

Ehud Greenspan

Encyclopedia of **Nuclear Energy**



Volume One



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Fundamentals and History of Development of Nuclear Energy: *Eric Norman*
Safety, Licensing and Decommissioning of Power Reactors: *Russell Stachowski*
Non-Electric Applications of Terrestrial Nuclear Reactors: *Alexander Stanculescu*

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ENCYCLOPEDIA OF NUCLEAR ENERGY

EDITOR-IN-CHIEF

Ehud Greenspan

*Retired Professor, Nuclear Engineering Department,
University of California at Berkeley*

VOLUME 1

**Fundamentals and History of Development of Nuclear Energy.
Commercial Nuclear Power Reactors and Their Design.
Advanced Nuclear Reactor Concepts Under R&D.**

SECTION EDITORS

Eric B. Norman

Nuclear Engineering Department, University of California at Berkeley

Russell E. Stachowski

Global Nuclear Fuel

Alexander Stanculescu

Retired; International Atomic Energy Agency, Idaho National Laboratory



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COVER PAGE

Bottom: View of a nuclear power plant by the sea in Vandellos, Spain. It is a 1043 MWe Pressurized Water Reactor.
Top, from left to right (sample applications of nuclear energy):

- a submarine powered by nuclear energy
- an exploration vehicle/lab on MARS powered by a Plutonium-238 nuclear heat source (insert)
- a nuclear medical diagnostic machine for humans with a sample of PET-CT scan (insert)
- an artist vision of a future Tokamak-type fusion energy demonstration power reactor

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Retired Professor and Head, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

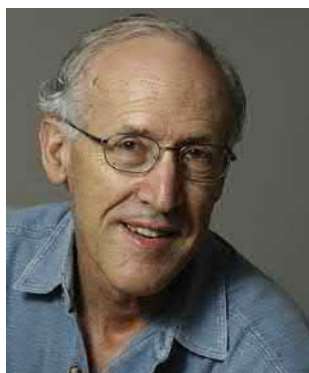
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EDITOR-IN-CHIEF



Ehud Greenspan

University of California at Berkeley
gehud@nuc.berkeley.edu

Professor Greenspan (Ph.D., 1966, in Nuclear Science and Engineering from Cornell University, Ithaca, N.Y., USA) retired from the Department of Nuclear Engineering of the University of California at Berkeley (UCB) in 2014 but served as a Professor in the UCB Graduate School until 2019. He has nearly 60 years of nuclear energy related research experience starting in 1961 as an M.Sc. student at the Department of Nuclear Engineering of the Israel Institute of Technology. His research spans a wide range of nuclear engineering related disciplines including: theoretical and experimental reactor physics; computational methods development including development of new optimization methods for determining the minimum critical mass and for finding optimal design of radiation shields, fusion reactor blankets and medical installations; conception and analysis of advanced fusion fuel cycles and of fusion-fission hybrid reactors; as well as advanced fission reactors and nuclear fuel-cycle conception, design and analysis. The latter activity was the focus of his research during the last twenty-five years. The thrust of this

research is improving sustainability of nuclear energy along with improving economics, safety, and proliferation resistance. He has been the Principal Investigator on studies of dozens of advanced nuclear systems, including those corresponding to Generation-IV reactors and to Small Modular Reactors (SMRs). He was one of the early advocates of SMRs and led a multi-institutional international team that developed a preliminary conceptual design of the Encapsulated Nuclear Heat Source (ENHS)—an innovative nuclear battery-type SMR that is free of pumps, pipes and valves in the primary reactor system. The advanced reactor technologies addressed in his research include: heavy water reactors, pressurized and boiling water reactors, prismatic and pebble-bed liquid-salt cooled reactors, molten-salt reactors, sodium and lead-alloy cooled fast reactors, breed-and-burn reactors, heat-pipe reactors as well as fusion-fission hybrid reactors and accelerator driven nuclear reactors. His research is documented in over 400 refereed publications, over 200 non-refereed publications, 5 book chapters, and a number of patents.

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Introduction to the Encyclopedia of Nuclear Energy

Ehud Greenspan, Department of Nuclear Engineering, University of California, Berkeley, Berkeley, CA, United States

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Introduction

From the beginning of the Universe through to the present day, nuclear energy, nuclear reactions, and nuclear radiations have played major roles in the evolution and shaping of our universe and planet Earth. All the chemical elements we find on Earth were produced by nuclear processes in the Universe. Planet Earth is continuously exposed to nuclear radiation from primordial radioactive isotopes produced in cosmological processes that followed the Big Bang. A number of primordial radioactive isotopes are still among the Earth's constituents, including thorium ^{232}Th , the uranium isotopes ^{235}U and ^{238}U as well as potassium ^{40}K . The spontaneous radioactive decay of the nuclides of these isotopes releases nuclear energy that provides a major contribution to the heat balance of planet Earth. The internal heat source is responsible for keeping the outer core of the Earth in a molten magma state and played a major role in the geologic evolution of the Earth's continents and oceans. The primordial radiation and Earth's heat source may have also contributed to the creation as well as evolution of life on Earth. Since its birth, the Earth has been also constantly bombarded by galactic cosmic nuclear radiation, which primarily consists of protons (hydrogen nuclei) and alpha particles (helium nuclei). Hence, mankind as well as any organism on Earth, were and are being continuously exposed to background nuclear radiation throughout their evolution. The background radiation level as well as the radioactive heat source due to the decay of the primordial radioactive isotopes were significantly higher at the time of Earth formation and are slowly decreasing with time.

It is only within the last hundred and twenty years or so that mankind discovered the structure of the atom and its nucleus and learned how to harness the enormous energy potential contained within the nucleus as well as the different types of radiation nuclei and atoms can emit. Nuclear radiation was discovered in 1896, when Henry Becquerel and Marie Curie discovered that salts of uranium emitted radiation that could be detected with photographic plates. In the 125 years since this discovery, mankind deciphered the structure of the atom and its nucleus, learned how to produce artificial radioactive isotopes (Irène Joliot-Curie and Frédéric Joliot in 1934), discovered how to fission certain heavy isotopes (Otto Hahn, Fritz Strassmann, Lise Meitner and Robert Frisch, 1938/39) and learned how to establish a fission chain reaction (Leo Szilard, Enrico Fermi et al., 1942) and developed numerous applications of nuclear radiation and nuclear energy. The feasibility of releasing nuclear energy by fusing nuclei of light isotopes was postulated already in 1920 (Arthur Eddington) who speculated that such nuclear reactions provide the enormous energy released in the sun and stars. Although research of harnessing nuclear fusion for controlled generation of nuclear energy is being pursued since the late 1940s, mankind has not yet succeeded in producing sufficiently more energy from fusion reactions than the energy invested in inducing these reactions. Additional information on the history and fundamentals of harnessing nuclear energy can be found in (Norman, 2021).

Encyclopedia scope

The Encyclopedia of Nuclear Energy (The Encyclopedia) provides a comprehensive authoritative coverage of a wide range of topics related to the generation, utilization and safe handling of nuclear energy, nuclear radiation and associated technologies, for the benefit of mankind and planet Earth. It covers past developments, the present status, and research and development of advanced innovative technologies that may enable improvements in the generation and utilization of nuclear energy (including nuclear radiation). The Encyclopedia consists of 272 thematic articles written by experts and arranged in 13 Sections; each Section covers the subject area identified in Table 1.

The fundamental concept of this encyclopedia is to provide all this vast and very diversified information in convenient to read cross-referenced articles within a single publication. Each article provides an effective solid foundation of its topic and references for readers interested in deeper exploration of the subject matter.

Table 1 Encyclopedia sections, number of articles per section and the section editors.

Sec. #	Section title (# of articles)	Section editor	Section editor affiliation
1	Fundamentals and history of development of nuclear energy (17)	Eric B. Norman	Professor in the Graduate School, Nuclear Engineering Department, University of California at Berkeley
2	Commercial nuclear power reactors and their design (22)	Russell E. Stachowski	Chief Consulting Engineer, Global Nuclear Fuel
3	Advanced nuclear reactor concepts under R&D (28)	Alexander Stanculescu	Retired Generation Four International Forum Technical Director; Retired Idaho National Laboratory Division Director
4	Safety, regulation, and decommissioning of commercial nuclear power reactors (24)	Joy L. Rempe	Rempe and Associates, LLC
5	Fuel cycle front-end: from mining through utilization (17)	Douglas C. Crawford	Director, Transient Reactor Test Facility, Idaho National Laboratory.
		Colby B. Jensen	National Technical Lead for Fuel Safety Research, Idaho National Laboratory
6	Fuel Cycle back-end: from recycling to deep geological disposal (19)	Christophe Poinssot	Deputy General Director (CEO) and Scientific Director of the French Geological Survey (BRGM) ^a
7	Radiation protection (21)	Laurence Miller	Professor Emeritus, Department of Nuclear Engineering, the University of Tennessee
8	Non-electric applications of nuclear energy (15)	Shannon M. Bragg-Sittou	Lead for Integrated Energy Systems, Idaho National Laboratory ^b
9	Nuclear power for space (16)	David I. Poston	Chief Reactor Designer, Los Alamos National Laboratory
10	Research reactors (23)	Sean O'Kelly	Associate Laboratory Director, Idaho National Laboratory. Chair of IAEA Technical Working Group on Research Reactors
11	Medical, industrial and agricultural applications of nuclear technology (26)	Alan E. Waltar	Retired Professor and Head, Department of Nuclear Engineering, Texas A&M University, College Station
12	Nuclear fusion R&D (26)	David J. Campbell	Munich, Germany. Former Director for Science and Operations, ITER Organization
13	Social issues of nuclear energy (18)	Andrew C. Kadak	Kadak Associates, Inc. Former Professor of the Practice Massachusetts Institute of Technology

^aAlso: Prof. of Nuclear Chemistry at the National Institute of Nuclear Science and technology (INSTN).

^balso National Technical Director for the DOE-NE Integrated Energy Systems program.

Intended readership

The articles are written at a level that users, with a bachelor degree in engineering, scientific and medical disciplines, especially in nuclear engineering, will find understandable and helpful. The Encyclopedia is intended to be an invaluable resource for students and will make an important contribution to the preservation of the vast amount of knowhow developed since the dawn of the nuclear era in the nuclear energy related fields and to its transfer to the new generation.

The Encyclopedia will also be very useful to industry and academia researchers and practitioners in the many scientific and engineering fields related to the generation and use of nuclear energy, including nuclear radiation. It will enable researchers and practitioners to refresh their memory in their field of specialization or expand their knowhow on less familiar subject areas.

The Encyclopedia will also be useful to members of the public with education in engineering or sciences who are interested in learning about specific nuclear energy or nuclear radiation related topics without having to struggle with textbooks or journal articles.

On the need for this Encyclopedia

The publication of this Encyclopedia is very timely for a number of reasons, including the following:

- Nuclear energy can play a very important role in drastically reducing the amount of fossil fuels burned for generation of electricity and, hence, the amount of greenhouse gas (GHG; primarily CO₂) emissions while, at the same time, making more electricity available for developing countries and for supporting the expected growth in the world population. Moreover, nuclear energy can make an important contribution to drastically reducing the amount of GHG emission from the transportation, industrial, residential and agricultural sectors of the economy. Hence, nuclear energy can make a vital contribution to meet the goal set by the 2015 Paris Agreement on climate change and ratified by many nations—to limit global warming to well below 2, preferably to 1.5 degrees Celsius, compared to pre-industrial levels. This point is elaborated upon below in the section: “On the Importance of Nuclear Energy in Avoiding Climate Change Catastrophe”. The state of the art in nuclear energy generation is covered in (Stachowski, 2021; Crawford, 2021).

- The Generation IV (GEN-IV) International Forum (GIF) was created in 2001 as a co-operative international endeavor of research, development, demonstration and deployment of advanced fourth generation innovative nuclear reactors that, hopefully, will be available for commercial deployment by 2030. The objectives of the innovative GEN-IV nuclear energy systems include improved sustainability, economics, safety and reliability, as well as proliferation resistance and physical protection. This encyclopedia provides an update on the state of the GEN-IV R&D efforts and explains the benefits the innovative GEN-IV reactors may provide. The Encyclopedia also describes (Stanculescu, 2021) the advanced nuclear fuel cycles that offer highly improved resource utilization, greatly reduced nuclear waste, and reduced proliferation risks.
- Over the past decade or so, there has been a surge in the number of private companies, mostly supported by venture capital, that are developing advanced innovative Small Modular Reactors (SMRs); reactors designed to generate between 10 to 300 MW of electricity per unit (Stanculescu, 2021). The SMRs offer a new paradigm in the generation and delivery of nuclear power; they promise to improve all aspects of design, engineering, construction and operation of commercial nuclear power plants, including enhanced safety performance, better economic affordability, and reduced financial risk. The components (modules) of SMRs are to be of a standard design, shop fabricated in assembly lines subjected to high quality control, and then transported as complete modules to the power plant site for rapid installation. The SMRs also offer flexible power generation for a wider range of users and applications—central power stations can consist of multiple units installed at a rate that best matches the growth in power demand; alternatively, a single SMR unit can be installed in the vicinity of the power customer using a so-called distributed (localized) power grid. It is even being proposed that the hundreds of shuttered coal power plant sites could be repurposed for SMRs. This encyclopedia describes (Stanculescu, 2021) a sample of the SMRs under development and explains the incentives for their development and the benefits they offer. The Encyclopedia also describes the motivation for, and examples of, so called “microreactors”—reactors producing an electrical power of less than 10 MW. Such reactors are primarily intended for deployment in remote areas which are not connected to the grid and to which the transport of fossil fuel is very expensive, as well as to provide reliable power to sensitive installations even in case of failure of the central power grid.
- The very ambitious space explorations being planned by leading industrial countries cannot succeed without nuclear energy. Space Fission Power (SFP) systems offer the potential to enable truly ambitious exploration of outer space, and ultimately a permanent human presence in space. Most of the benefits of SFP is tied to the very high amount of energy that can be extracted per unit weight of fissile material, which can allow high powers and/or long lifetime at a relatively low mass. SFP systems also offer great flexibility in terms of being able to operate in diverse environments, including environments with no or limited solar radiation, and the ability to provide abundant electrical power and heat from kilowatts to megawatts. Applications of SFP include propulsion and/or power for spaceships, electricity and heat for exploration rovers for landing on Mars (other planets and moons) or for human colonies on the Moon or Mars. This encyclopedia describes (Poston, 2021; Bennett, 2021) the many benefits nuclear energy offers to space exploration and the many challenges that need to be overcome.
- Controlled thermonuclear fusion is the focus of major international research programs that aim to demonstrate the scientific and technological feasibility of large-scale generation of nuclear energy by the fusion of light atomic nuclei. The widely distributed fusion fuel resources together with the favorable anticipated safety and environmental characteristics of future fusion power plants provide a significant motivation for the continued development of the challenging science and technology towards the exploitation of fusion energy for the benefit of mankind. In addition to the description of the fundamental elements of the fusion process and of the principal approaches to the controlled and sustained generation of fusion energy being pursued, this Encyclopedia presents the state-of-the-art for fusion research and development and the progress towards commercial fusion energy generation (Campbell, 2021). The advantages and challenges of nuclear fusion as an energy source are summarized.
- This encyclopedia also describes the many ways nuclear radiation can be harnessed to provide enormous benefits in the fields of medicine, agriculture, modern industry, environmental protection, and public safety (Waltar, 2021). Described in the Encyclopedia are more than a dozen unique properties of radiation technology that allow for these benefits to be achieved. Even though enormous and widespread, the benefits to society from these technologies are largely unknown and unappreciated. The global markets for these technologies exceed USA \$500 billion annually and are growing rapidly.
- Also provided in this Encyclopedia is information on dozens of research reactors (O’Kelly, 2021) that are operating or under construction around the world and provide unique research capability for scientists pursuing advanced research in physics, chemistry, biology, material and other sciences. Another invaluable utilization of research reactors is the production of special radioactive isotopes for numerous applications in medical, industrial and agricultural applications, and scientific research. Some research reactors are also used for demonstrating the performance of evolutionary as well as advanced nuclear fuels and of advanced nuclear power reactor components.
- Public perception of nuclear energy varies from country to country. It tends to be most negative in wealthy democracies like Germany, Australia, and Japan. Contributing to the negative perception is a fear of reactor accidents, a fear of nuclear radiation, a fear of long-lived radioactive waste, and a fear of the nuclear technology misuse for weapons proliferation. The encyclopedia addresses these concerns and provides authoritative objective information that, hopefully, will contribute to dispelling the public misconceptions (Kadak, 2021).
- The Encyclopedia describes measures implemented to ensure the health and safety of the public from nuclear power (Rempe, 2021). These measures evolved as new lessons were learned from operating experience and unplanned events, in particular events with core damage and radionuclide release (including in Three Mile Island, Chernobyl, and Fukushima). Clearly, the

experience gained from the design, evaluation, operation, decommissioning, and regulation of nuclear power plants has led to significant safety improvements. Future advanced reactor designs are expected to incorporate innovative inherent safety features (Stanculescu, 2021).

- The encyclopedia provides a detailed description of the biological effects of nuclear radiation, of how we reliably measure radiation levels and ensure that they do not exceed the permissible levels (Miller, 2021). The Encyclopedia describes the theory and practice of radiation protection that provide the basis for regulatory requirements that are essential for the safe and beneficial use of the many advanced technologies that involve exposure to nuclear radiation. The Encyclopedia explains that everyone is exposed to natural background radiation on a daily basis; that radiation is routinely in use for diagnostics of health conditions, such as for dental and medical X-rays, and that this is the largest source of radiation exposure for almost all human population in developed countries and that the benefits from these practices far exceed their risk. The regulatory permissible level of nuclear radiation is harmless much like a moderate exposure to sunshine is harmless (and may even be beneficial, whereas over-exposure can cause sunburns). In other words, the regulatory limits of doses induced by exposure to nuclear radiation are already substantively protective, and there is no evidence of health improvements by reducing the doses below these limits. Insisting to reduce radiation exposure to as low as possible below the regulatory limits may result in harm.
- Also provided in this Encyclopedia is detailed information about the nuclear waste and how it can be disposed of safely (Poinssot, 2021; Kadak, 2021). Using advanced fuel cycles (Stanculescu, 2021; Poinssot, 2021) the amount and activity of the long-lived nuclear waste can be reduced to few percent of the amount that needs to be disposed of per unit of energy generated using the contemporary “once-through” fuel cycle.

On the importance of nuclear energy in avoiding climate change catastrophe

The world’s population is expected to grow from current ~6.7 billion people to over 9 billion by 2050. As recognized in the United Nations Resolution 70/1 “Transforming our world: the 2030 Agenda for Sustainable Development” (Stanculescu, 2021; United Nations, 2015), providing “universal access to affordable, reliable and sustainable energy” and in particular to reliable electrical energy, is central to human development, social stability and even world peace. Currently, about 2.8 billion people are still lacking modern energy services (International Energy Agency, 2017), and more than one billion do not have access to electricity. Not only the growth of worldwide population, but also the economic growth in developing countries will lead to a continuing increase in the demand for energy.

The tremendous increase in energy generation required for the benefit of the global population has to be accomplished while reducing the rate of GHG (primarily CO₂) emission. Already, “human activities are estimated to have caused approximately 1.0 °C of global warming above pre-industrial levels” according to the Intergovernmental Panel on Climate Change (IPCC, 2018). In order to limit the global temperature increase to around 1.5 °C, total CO₂ emissions need to be reduced to net zero by around 2050.

Germany, for example, had decided to achieve this goal by installing renewable energy generators made of solar photovoltaic panels and wind turbines. Germany also decided to close down its nuclear power plants. As a result, according to Goldstein et al. (2019), “Germany, which went all-in for renewables, has seen little reduction in carbon emissions, and, according to our calculations, at Germany’s rate of adding clean energy relative to gross domestic product, it would take the world more than a century to decarbonize, even if the country wasn’t also retiring nuclear plants early. A few lucky countries with abundant hydroelectricity, like Norway and New Zealand, have decarbonized their electric grids, but their success cannot be scaled up elsewhere; the world’s best hydro sites are already dammed.”

Goldstein et al. (2019) are also saying “we actually have proven models for rapid decarbonization with economic and energy growth: France and Sweden. They decarbonized their grids decades ago and now emit less than a tenth of the world average of carbon dioxide per kilowatt-hour. They remain among the world’s most pleasant places to live and enjoy much cheaper electricity than Germany to boot. They did this with nuclear power. And they did it fast, taking advantage of nuclear power’s intense concentration of energy per unit weight of fuel. France replaced almost all of its fossil-fueled electricity with nuclear power nationwide in just 15 years; Sweden, in about 20 years. In fact, most of the fastest additions of clean electricity historically are countries rolling out nuclear power.”

In the words of World Nuclear Association Director General Agneta Rising in 2019 (Bisconti, 2021): “The 445 nuclear reactors in 30 countries are the low-carbon backbone of electricity systems, operating in the background, day in day out, often out of sight and out of mind, capable of generating an immense amount of clean power. They are the silent giants upon which we rely today.” Table 2 lists the fraction of electricity generated during 2018 in selected countries and globally from energy sources that do not emit greenhouse gases (Pioro et al., 2021). According to Poinssot and Bourg (2021), nuclear energy is within the top three of the most environmental-friendly energy sources; in the same range as hydropower and wind-power. Measured by CO₂ emissions, nuclear energy is even the most environmentally friendly energy source.

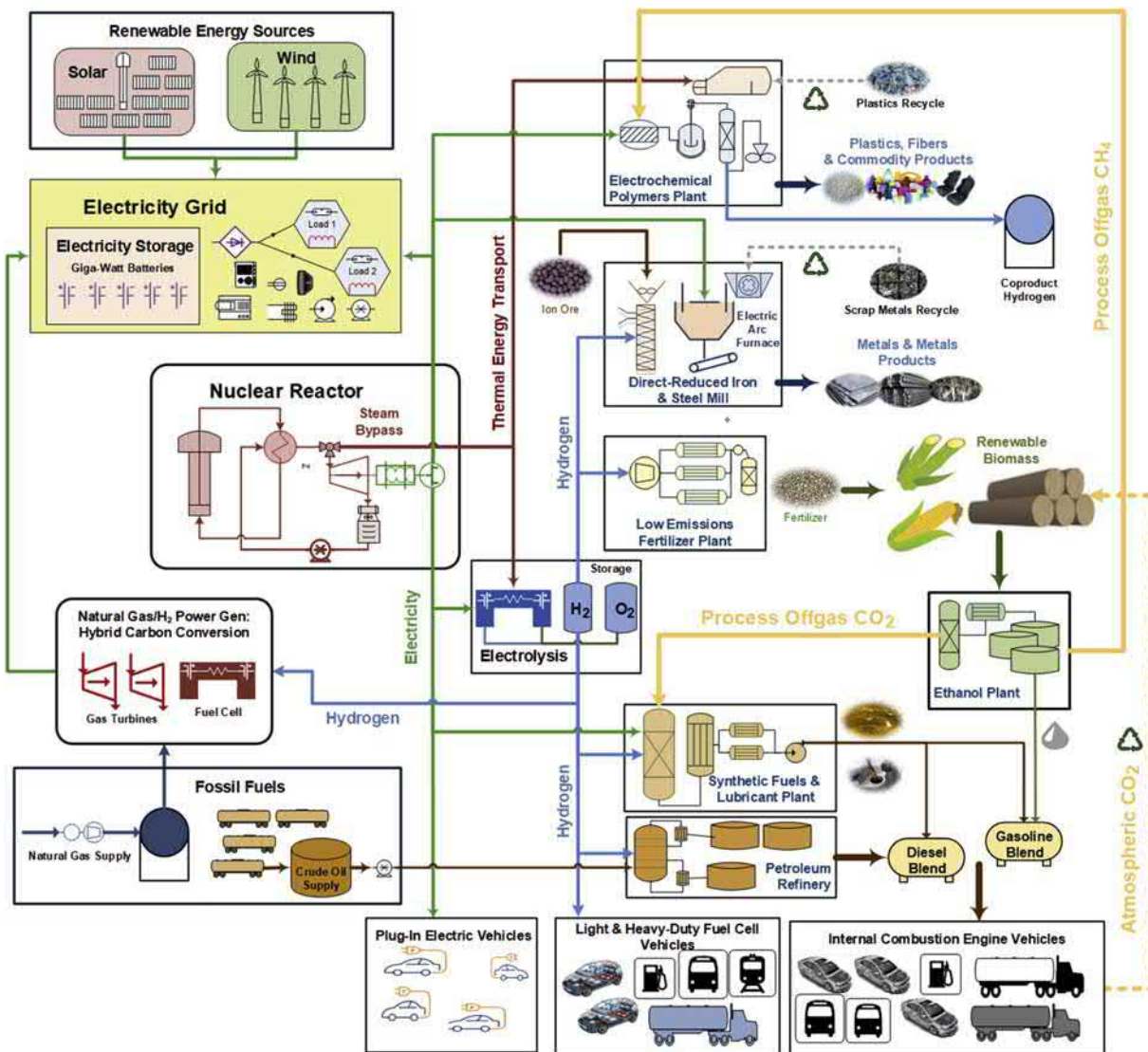
It is being suggested (Conca, 2021) that recent research has revealed the potential for extracting practically unlimited supplies of uranium from the oceans at a cost that may come close to that of land mining. If successful, this may make fission nuclear energy practically as renewable as wind and solar (Conca, 2021).

It is not sufficient to decarbonize the electricity generation sector. In the U.S., for example, 33% of CO₂ emissions are attributed to energy generated to support the electricity sector but an additional 36% of emissions result from transportation and 19% result

Table 2 Fraction of electricity generated by clean energy technologies in selected countries.

Country	Nuclear	Renewable	HYDRO
France	72	3	11
Ukraine	53.5	0.7	4.2
Sweden	41	10.2	39
UK	25	25	-
Korea	23.4	0	1.3
Russia	19.0	0	17
USA	19.7	10.9	6.6
Germany	11.6	22.2	-
World	10.2	6.6	16.3

from industrial processes (Bragg-Sitton and Boardman, 2021). There will be no global environmental benefits from switching transportation to use all electric vehicles if the electricity used for charging the batteries is generated in fossil fuel power plants (without sequestration of the CO₂ these plants emit). The realization of the above-mentioned facts is recently leading to a paradigm shift in the approach to the use of nuclear energy that is likely to bring about a restructuring in the entire energy market.

**Fig. 1** Illustrative examples of the near-term integration of existing LWRs with various industry applications (Bragg-Sitton and Boardman, 2021).

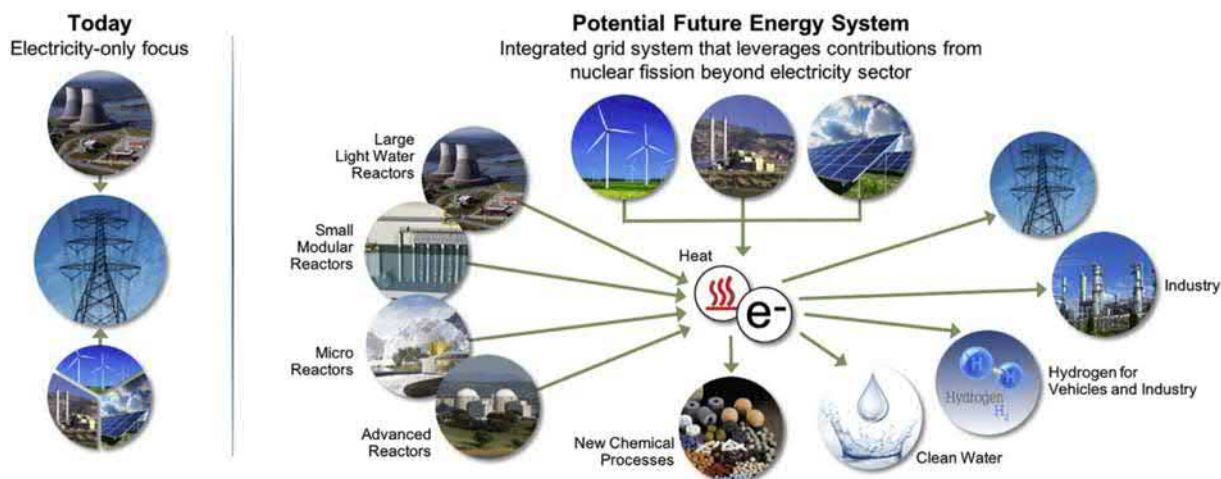


Fig. 2 Conceptualized coordination of energy generation sources and demand to maximize flexibility and economic performance while ensuring grid reliability and resilience.

Most energy experts around the world are advocating the development of a portfolio of low-carbon multi-energy technologies including, in addition to hydro-power plants (with very limited potential for expansion in capacity) and the currently favored “renewable” electricity generators, nuclear power plants. Because the power output of the renewable generators greatly fluctuates over the day and over the year, these generators need to be supplemented by effective energy storage devices that significantly add to the cost of energy.

Nuclear power plants, on the other hand, can generate power at a constant rate throughout the day and the year (excluding a shutdown period of ~1 month for refueling and maintenance once every 18–24 months), practically independent of weather conditions, while requiring a smaller land area per unit energy generated relative to the area required by renewable energy generators.

Moreover, nuclear energy can be generated in the form of high temperature heat (Stanculescu, 2021; Bragg-Sitton and Boardman, 2021) that is the most economical form of energy to store and to use as a backup for the intermittent nature of the renewable energy generators. Therefore, a suitable mix of nuclear and renewables is likely to be the optimal strategy for carbon-free energy generation.

Fig. 1 (Bragg-Sitton and Boardman, 2021) is a simplified illustration of the evolving perception of integrated energy systems using contemporary nuclear power reactors (primarily Light Water Reactors, LWR) in combination with solar and wind renewable energy generators to provide electricity and heat energy for transportation and many industries including non-grid energy users.

Fig. 2 (Bragg-Sitton et al., 2021) is a simplified conceptual perception of the proposed transition away from the use of contemporary large central nuclear power plants from their traditional electricity-only production mode to their exploitation for the provision of a variety of outputs. Future energy systems that seek to maximize energy utilization, generator profitability, and grid reliability and resilience must necessarily consider all available carbon-free energy sources, using each available resource in a way that maximizes its benefits while minimizing less-desirable attributes. The future energy systems should include advanced nuclear reactors including Gen-IV reactors, SMRs and even microreactors and should use distributed in addition to centralized electricity grids.

In summary, the need to meet the objective of zero net CO₂ emission by 2050 combined with the need to greatly increase the total global energy generation rate is an enormous challenge that can be achieved only by making use of all available clean energy generation technologies including nuclear in addition to solar and wind electricity generators. At the 2021 Earth Day climate summit, U.S. president Biden called (WSJ, 2021) for slashing U.S. greenhouse gas emissions by 2030 by 50%–52% compared with the baseline year of 2005, as he seeks to put America at the forefront of world-wide efforts to combat climate change. According to WSJ (2021), meeting this objective would require dramatically reshaping key sectors of the economy, from energy to transportation to agriculture.

Concluding remarks

This Encyclopedia of Nuclear Energy provides a very comprehensive, up-to-date and authoritative information related to nuclear energy, including nuclear radiation—from the discovery of the structure of the atom and its nucleus; through the discovery of three different ways to extract energy from nuclei (fission of heavy nuclei, fusion of light nuclei and radioactive decay); through the state-of-art of the nuclear industry for energy generation and for exploitation of several nuclear technologies for medical, industrial, agricultural and other applications; to the research and development of innovative nuclear technologies that are expected to further

increase mankind benefits from nuclear energy. The reader can find in the Encyclopedia a wide range of topics ranging from the contribution of nuclear energy to the evolution of the continents and oceans of planet Earth to the enabling of very ambitious space explorations including the provision of electricity and heat to human colonies on the Moon, Mars and beyond. The Encyclopedia explains why it is essential to greatly increase the nuclear energy generation rate in order to prevent a global warming catastrophe and, at the same time, provide sufficient energy to the growing world population and to the billions of people in underdeveloped countries.

It is hoped that this Encyclopedia will make an important contribution to transferring of the huge amount of technical knowledge accumulated in the field to the young generation, will inspire students to join research and development of innovative nuclear technologies and will contribute to better understanding of the nuclear technologies, their merits and safety by the general public and policy makers. The appreciation of the enormous benefits in the fields of steady reliable clean energy generation, medicine, agriculture, modern industry, environmental protection, and public safety that mankind enjoys every day and which is described in this Encyclopedia will, hopefully, contribute to much greater public acceptance and support of enhanced utilization of nuclear energy in the mix of clean energy technologies.

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CONTRIBUTORS TO VOLUME 1

Eric Abonneau

Commissariat à l'Energie Atomique et aux Energies Alternatives, Paris, France

Tim Abram

Manchester University, Manchester, United Kingdom

Alessandro Alemberti

Ansaldo Nucleare SpA, Genova, Italy

M Arcaro

Principal Engineer, GE-Hitachi Nuclear Energy-Systems Engineering, Wilmington, NC, United States

Benjamin R Betzler

Oak Ridge National Laboratory, Oak Ridge, TN, United States

Jeffery A Brown

Westinghouse Electric Company, Pittsburgh, PA, United States

Nicholas R Brown

Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

Mike W Davies

Jacobs Clean Energy Limited, Knutsford, Cheshire, United Kingdom

Daniel E DiGregorio

School of Science and Technology, National University of San Martín, Buenos Aires, Argentina; and Constituent Atomic Center, National Atomic Energy Commission, Buenos Aires, Argentina

RB Duffey

Idaho Falls, ID, United States

Lyndon Edwards

Australian Nuclear Science and Technology Organisation, Lucas Heights, Australia

Carlo Fiorina

Laboratory for Reactor Physics and System Behaviour, EPFL, Lausanne, Switzerland

Massimiliano Fratoni

Nuclear Engineering Department, University of California, Berkeley, CA, United States

Michael A Fütterer

European Commission, Joint Research Centre, Petten, The Netherlands

Alejandro Garcia

Department of Physics, University of Washington, Seattle, WA, United States

J Gehin

Idaho National Laboratory, Idaho Falls, ID, United States

E Greenspan

Department of Nuclear Engineering, University of California, Berkeley, CA, United States

W Halsey

Lawrence Livermore National Laboratory, Livermore, CA, United States

David A Hamon

GE-Hitachi Nuclear Energy, San Jose, CA, United States

Ke Han

School of Physics and Astronomy, Shanghai Jiao Tong University, Shanghai, China

Branislav Hatala

VUJE, A.s., Trnava, Slovak Republic

F Heidet

Argonne National Laboratory, Lemont, IL, United States

Pavel Hejzlar

TerraPower, LLC, Bellevue, WA, United States

Yanping Huang

Nuclear Power Institute of China, Chengdu, China

Victor V Ignatiev

NRC-Kurchatov Institute, Moscow, Russian Federation

Minwhan Kim
Korea Atomic Energy Research Institute, Daejeon, South Korea

PL Kirillov
State Scientific Centre of the Russian Federation, Al
Leypunsky Institute of Physics and Power Engineering
(IPPE), Obninsk, Russia

Karen Vierow Kirkland
Department of Nuclear Engineering, Texas A&M
University, College Station, TX, United States

Jennifer L Klay
California Polytechnic State University, San Luis
Obispo, CA, United States

Kevin J Kramer
TerraPower, LLC, Bellevue, WA, United States

Kenneth S Krane
Department of Physics, Oregon State University,
Corvallis, OR, United States

Jiri Krepel
Paul Scherrer Institut, Villigen, Switzerland; and Paul
Scherrer Institute, Villigen, Switzerland

Laurence Kim-Hung Leung
Canadian Nuclear Laboratories, Chalk River, ON,
Canada

Fu Li
INET, Tsinghua University, Beijing, China

Evzen Losa
Department of Nuclear Reactors, Czech Technical
University, Prague, Czechia

G Ivan Maldonado
Department of Nuclear Engineering, University of
Tennessee, Knoxville, TN, United States

RP Martin
Technical Consultant, BWX Technologies, Inc.,
Lynchburg, VA, United States

Reina Maruyama
Department of Physics, Yale University, New Haven,
CT, United States

Edward C Morse
University of California, Berkeley, CA, United States

Ondrej Muransky
Australian Nuclear Science and Technology
Organisation, Lucas Heights, Australia

Jim Napolitano
Temple University, Philadelphia, PA,
United States

Eleodor Nichita
Faculty of Energy Systems and Nuclear Science, Ontario
Tech University, Oshawa, ON, Canada

Eric B Norman
Department of Nuclear Engineering, University of
California, Berkeley, CA, United States

I Pioro
University of Ontario Institute of Technology, Oshawa,
ON, Canada

R Pioro
Faculty of Physics, Lomonosov Moscow State University,
Moscow, Russia

Manuel A Pouchon
Paul Scherrer Institut, Villigen, Switzerland

E Previtali
Department of Physics, University of Milano-Bicocca,
Milan, Italy

Staffan Qvist
Qvist Consulting Limited, London, United Kingdom

B Cameron Reed
Department of Physics (Emeritus), Alma College, Alma,
MI, United States

Frederik Reitsma
Ultra Safe Nuclear Corporation (USNC-Europe),
Olette, France

Geoffrey S Rothwell
Department of Economics (Retired), Stanford
University, Stanford, CA, United States; and Jean
Michard Pélissier, Antibes, France

Ramesh Sathankar
Canadian Nuclear Laboratories, Chalk River, ON,
Canada

Hiroyuki Sato
Japan Atomic Energy Agency, Ibaraki, Japan

Hiroshi Sekimoto
Tokyo Institute of Technology, Tokyo, Japan

Seok Bin Seo
Department of Nuclear Engineering, University of
Tennessee, Knoxville, TN, United States

Phil Sharpe
Studsvik Scandpower, Inc., Wilmington, NC, United
States

Jennifer Shusterman
Department of Chemistry, Hunter College of the City
University of New York, New York, NY, United States;
and Graduate Center of the City University of New
York, New York, NY, United States

Eugene Shwageraus
*Department of Engineering, University of Cambridge,
Cambridge, United Kingdom*

Craig F Smith
*Department of Physics, Naval Postgraduate School,
Monterey, CA, United States*

Russell E Stachowski
Global Nuclear Fuel, Wilmington, NC, United States

Alexander Stanculescu
*Idaho National Laboratory, Nuclear Systems Design and
Analysis Division, Idaho Falls, ID, United States; and
Generation IV International Forum (GIF)*

Gerhard Strydom
*Idaho National Laboratory, Idaho Falls, ID, United
States*

Andrew E Stuchbery
*Department of Nuclear Physics, The Australian National
University, Canberra, ACT, Australia*

M Hadid Subki
*Nuclear Power Technology Professional, Vienna,
Austria; and International Atomic Energy Agency,
Vienna, Austria*

T Taiwo
Argonne National Laboratory, Lemont, IL, United States

M Todosow
*Brookhaven National Laboratory, Upton, NY, United
States*

Pavel V Tsvetkov
*Department of Nuclear Engineering, Texas A&M
University, College Station, TX, United States*

Belle R Upadhyaya
*Department of Nuclear Engineering, University of
Tennessee, Knoxville, TN, United States*

Janne Wallenius
*Division of Nuclear Engineering, KTH, Stockholm,
Sweden*

William Walters
*Ken and Mary Alice Lindquist Department of Nuclear
Engineering, Pennsylvania State University, State
College, PA, United States*

David K Wehe
*Department of Nuclear Engineering and Radiological
Sciences, University of Michigan, Ann Arbor, MI,
United States*

Fred E Wietfeldt
*Department of Physics and Engineering Physics, Tulane
University, New Orleans, LA, United States*

R Wigeland
*Idaho National Laboratory, Idaho Falls, ID, United
States*

Richard T Wood
*Department of Nuclear Engineering, University of
Tennessee, Knoxville, TN, United States*

Metin Yetisir
*Canadian Nuclear Laboratories, Chalk River, ON,
Canada*

Michael Y Young
*Westinghouse Electric Company, Pittsburgh, PA, United
States*

Jinguang Zang
Nuclear Power Institute of China, Chengdu, China

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Introduction to Fundamentals and History of Development of Nuclear Energy

Eric B. Norman, Department of Nuclear Engineering, University of California, Berkeley, CA, United States

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From the beginning in the Big Bang through to the present day, nuclear energy, nuclear reactions, and radioactive decays have played major roles in the history and evolution of our universe. However, it is only within the last 100 years or so that we have discovered the secrets of the atom and how to harness the enormous energy potential contained within the nucleus. In this section of the *Encyclopedia of Nuclear Energy*, the fundamental concepts and history of the development of nuclear science are presented by a group of internationally recognized experts in the field. The following chapters are included in this section.

Chapter 1: *What is the Universe Made Of?*, by Prof. Alejandro Garcia, University of Washington

Throughout the ages, people have wondered what our universe is made of. The early Greek notion that all things are made of four *elements*, earth, air, fire, and water, stood for 2000 years. By the 1800s, it was recognized that there are many more chemical elements which could be organized in Mendeleev's famous *periodic table*. In the early 1900s, the reality of *atoms* was experimentally confirmed. Further experiments revealed the existence of the atomic *nucleus* and its constituent *neutrons* and *protons*. Additional studies showed that neutrons and protons are actually composite objects made of different combinations of three *quarks*. This chapter describes the evolution of our understanding of what our universe is made of, and the experimental evidence supporting each step in that development is presented.

Chapter 2: *Discovery of Radioactivity*, by Prof. Daniel DiGregorio, University Nacional de San Martin

Radioactive materials have been present on Earth since its formation 4.6 billion years ago. However, it is only within the last 100 years or so that this process was identified, characterized, and eventually understood theoretically. Initial studies were conducted using naturally-occurring radioactive materials such as uranium and thorium ores. Discoveries by Becquerel, Marie and Pierre Curie, and Rutherford established that through radioactive decays, one element is transformed into another via the emission of radiation (later named alpha and beta particles). Further experimentation showed that artificial radioactivity can be produced through nuclear reactions. The history of these discoveries is described in this chapter.

Chapter 3: *Discovery of the Quantum Structure of Atoms and Their Nuclei*, by Prof. Kenneth Krane, Oregon State University

In classical physics, variables such as energy, angular momentum, and electric charge are treated as continuous quantities that can take on any value. However, in the late 1800s and early 1900s, studies of atomic structure led to the discovery that only certain discrete values of these variables actually occur. In this chapter, the experimental evidence for and the theoretical understanding of the role of *quantization* in the structure of atoms and nuclei are described.

Chapter 4: *Origins of Nuclear Energy*, by Prof. Jim Napolitano, Temple University

The mass of an atomic nucleus composed of Z protons and N neutrons is observed to be a bit less than the sum of the masses of these constituents. This difference is due to the strong nuclear force which holds the nucleus together. Through the use of Einstein's famous equation, $E = mc^2$, we can quantify the so-called nuclear binding energy which is approximately a million times greater than the binding energy of electrons in atoms. Because of the variation of the nuclear binding energy with Z and N , it is possible to release large amounts of energy through nuclear reactions such as fission and fusion and via radioactive decays. The details of how these processes can be used to produce energy are described in this chapter.

Chapter 5: *Radioactive Decay*, by Prof. Ezio Previtali, University of Milan – Bicocca

Radioactive decay is the mechanism by which unstable nuclei release energy through the emission of radiation. Over 100 years ago, through his studies of the decays of naturally occurring radioactive materials, Ernest Rutherford discovered that there are three main types of radioactive decay. He named them alpha, beta, and gamma decays after the first three letters of the Greek alphabet. Through subsequent experiments he discovered that alpha particles are the nuclei of ^4He atoms; beta particles are electrons, and gamma rays are quanta of electromagnetic radiation known as photons. In this chapter, the details of these three types of radioactive decay are explained and examples are provided of some of the uses and applications of radioactivity in society.

Chapter 6: *Fission I*, by Prof. Jennifer Klay, California Polytechnic State University San Luis Obispo

In the 1930s, Enrico Fermi, Lise Meitner, Otto Hahn, and Fritz Strassmann irradiated uranium with neutrons in hopes of producing new "transuranic" elements. Instead, they discovered the process of nuclear fission. Meitner and her nephew Otto Frisch explained the results of these experiments in terms of the process by which a heavy nucleus splits apart in to two or more fragments, releasing large amounts of energy and neutrons that can thus sustain a chain reaction. This chapter describes the history of these discoveries and the basics physics involved in the fission process.

Chapter 7: *Fission II*, by Prof. Andrew Stuchbery, Australian National University

After the discovery of the fission process, it did not take long for scientists and engineers to develop the first man-made sustained nuclear chain reaction in the so-called Chicago Pile 1. This led to the development of nuclear reactors as major sources of electrical power. In order to accomplish this, much had to be learned about neutron-induced fission cross sections, the energetics of fission, fission product properties, neutron interactions with materials, and conditions necessary to operate a nuclear chain reaction in a controlled and stable manner. Somewhat surprisingly, in the 1970s, it was discovered that approximately two billion years ago in what is now the nation of Gabon in Africa, conditions were such as to allow natural nuclear reactors to operate for lengthy periods of time in a self-regulating pulsed mode. The details of these topics are described in this chapter.

Chapter 8: *Manhattan Project*, by Prof. B. Cameron Reed, Alma College

During World War II, the United States undertook the development of fission-based nuclear weapons under what was called the *Manhattan Project*. Using fast neutron-induced fission cross sections that had only recently been measured, it was determined that weapons based on the fissions of ^{235}U and ^{239}Pu were both possible. Weapons of both types were produced and were dropped on the Japanese cities of Hiroshima and Nagasaki. This chapter describes the details of the *Manhattan Project* and of the competing German program.

Chapter 9: *Phenomenology of Neutron Interactions*, by Prof. Fred Wietfeldt, Tulane University

The neutron was discovered in 1932 by James Chadwick. Within a decade, Enrico Fermi and his colleagues demonstrated that a self-sustaining nuclear chain reaction could be produced via the neutron-induced fission of ^{235}U . Depending upon their energies, neutrons can interact with matter in a number of different ways. Knowledge of these interactions was key to the development of nuclear reactors, radiation safety, materials studies, and a variety of other applications. The basic principles of neutron interactions are described in this chapter.

Chapter 10: *Fissile and Fertile Fuels*, by Prof. Jennifer Shusterman, Hunter College

Neutron-induced fission is the process by which nuclear reactors produce energy. Materials that undergo fission after capturing a thermal neutron are called *fissile*. Examples of fissile isotopes include ^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu . *Fertile* materials are those that can be converted into fissile materials via neutron capture and beta decay. ^{232}Th and ^{238}U are examples of fertile materials. These isotopes are also referred to as fissionable since they can fission by high-energy neutrons. In this chapter, the chemical and nuclear properties of both fissile and fertile isotopes are described. The roles of each of these types of materials in the operations of nuclear reactors are also presented.

Chapter 11: *Fusion*, by Prof. Edward Morse, University of California at Berkeley

Fusion is the process by which two nuclei combine to produce one or more different nuclei. Due to the increase in nuclear binding energy per nucleon as one moves up in mass from hydrogen to iron, nuclear fusion reactions can produce enormous amounts of energy. The fusion of hydrogen into helium powers our Sun and most other stars. Subsequent fusion reactions produce all of the elements up to iron. Because of its potential as a commercial energy source, much theoretical and experimental effort has been devoted to developing a means to produce a controlled fusion reactor. The details of stellar fusion and of our attempts to develop large scale fusion facilities are discussed in this chapter.

Chapter 12: *Phenomenology of Gamma Ray and Charged Particles Interactions*, by Prof. Reina Maruyama, Yale University

An understanding of the ways in which gamma rays and charged particles interact with matter was required for the development of nuclear reactors, nuclear medicine, radiation safety, radiation detectors, and many other applications of nuclear science and engineering. Gamma rays interact with matter via three mechanisms: the photoelectric effect, Compton scattering, and pair production. Charged particles interact with matter primarily through collisions with atomic electrons producing ionization. The details of these processes are described in this chapter.

Chapter 13: *Radiation Sources*, by Prof. David Wehe, University of Michigan

Radiation sources come in many forms, from naturally occurring radioactive materials, to man-made sources, to accelerator-based sources. A large number of radioactive isotopes are used in applications ranging from nuclear medicine to oil-well logging. Similarly, accelerator-based sources of neutrons are utilized in materials studies and basic nuclear science. This chapter describes how and where many radioactive isotopes are produced and what they are used for.

Chapter 14: *Experimental Tools – Radiation Detectors*, by Prof. Ke Han, Shanghai Jiao Tong University

Our senses are not keen enough to register the interaction of a single neutron, proton, electron, or gamma ray. Thus, in order to study nuclear properties and to utilize ionizing radiation in applications such as nuclear reactor operations, radiation safety, nuclear medicine, and oil exploration, we rely on numerous types of radiation detectors. Some detectors measure the direct ionization produced by radiation; others measure the light produced when radiation interacts with certain materials; still others are sensitive to the heat produced by radiation interactions. In this chapter, the physics principles behind such detectors are described and examples are provided of a number of types of modern radiation detectors.

Chapter 15: *Experimental Tools- Accelerators*, by Prof. Alejandro Garcia, University of Washington

The range of the strong nuclear force is extremely short. Thus, in order for nuclear reactions to occur, the nuclei involved must essentially “touch.” To allow this to happen, one must overcome the Coulomb repulsion between nuclei. Today this is accomplished through the use of particle accelerators that can produce beams of any stable nuclear species (and a number of radioactive ones as well) at energies from electron-volts (eV's) up to trillions of eV's (TeV's). This chapter describes the physical principles underlying a variety of types of accelerators and provides examples of a number of currently operating facilities around the world.

Chapter 16: *Atomic and Hydrogen Bombs*, by Prof. Jim Napolitano, Temple University

The Manhattan Project resulted in the production of fission-based nuclear weapons commonly identified as *atomic bombs*. The fusion of light nuclei can produce more energy per unit mass than the fission of heavy nuclei. This idea is the basis for fusion-based weapons known as *hydrogen bombs* that can be much more powerful than fission weapons. In this chapter the basic physics as well as the unclassified details of how such weapons work is described.

The material contained in these chapters will provide the reader with a basic understanding of the sources of nuclear energy, radioactive decays, fission, and fusion. The experimental tools used to study and utilize nuclei are discussed. These chapters also describe how we have learned about the atomic nucleus and the ways in which it can be employed in the many applications described in later sections of this Encyclopedia.

What Is the Universe Made of?

Alejandro Garcia, Department of Physics, University of Washington, Seattle, WA, United States

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From the Greek philosophers, the alchemists, to Mendeleev

The quest for understanding what the Universe is made of seems to be intrinsic to our species.

This curiosity, combined with the ability for organizing thoughts and discerning facts from fiction, has allowed humans to evolve from a hunter-gatherer society to its modern form, with all the associated advantages. Without understanding what things are made of, it is difficult to control them. So, it is not strange that one finds attempts to answer the question of the *composition of things* in the early history of human thought. Some of the Greek philosophers, including Aristotle, thought that all things on our planet were made of four elements: fire, water, air, and earth, adding quintessence for things in the heavens. However, this school thought of each of these as infinitely divisible. Leucippus, and his disciple, Democritus (c. 400 BCE), considered that indivisible units, *atomos*, were the basic components of all matter. However, without the rigor of modern experimental science, these ideas remained abstract for thousands of years. Interestingly, up to and during the middle ages, the alchemists were thought to have magical recipes that allowed *transmutation* of the more common elements into gold. This idea was also to remain dormant until the discoveries of the early 20th century.

Chemistry matured into a rigorous discipline through the renaissance. Toward the end of the 18th century, Lavoisier would give solid grounds to the atomic idea, identifying “simple substances” that could not be broken into parts by chemical reactions. In the 1800s, Dalton arrived at ways of determining the atomic weights of a couple dozen elements and an organizing chart in which they were classified by their atomic weight. It was Mendeleev, in 1869, who managed to organize 63 elements by increasing atomic weights and in columns, with similar chemical properties, showing periodicity. Mendeleev was able to predict elements that had not been discovered at the time or to correct the assigned atomic weight of some elements based on the periodic table. It is a wonderful example of how a correct organization can help discovery. Toward the end of the 19th century, J.J. Thomson would discover that the glow observed when high voltage was applied to metals under vacuum was caused by electrically-charged particles with a mass approximately 2000 times smaller than the mass of hydrogen. The electron would later become the principal character in explaining Mendeleev’s periodic table and remains considered an elementary particle to the present. J.J. Thomson conceived a *plum pudding model* for atoms, where electrons, with negative electric charge, would be confined by electric attraction to a positively charged substance that was thought to be uniformly distributed over the volume of the atom.

Rutherford and Chadwick, protons, neutrons, transmutations, basic nuclear properties, fission and fusion

In 1896, Becquerel discovered that salts of uranium emitted radiation that could be detected with photographic plates. Becquerel identified three types of radiation, now called alpha, beta, and gamma, according to the behavior of the radiation in magnetic fields. The alpha radiation was eventually identified as ions of helium, the beta radiation as electrons, and the gamma radiation as electromagnetic waves. In the early 1900s Rutherford and co-workers started studying the scattering of alpha particles from gold foils. With the plum pudding model for the gold atoms in his mind, Rutherford was amazed by the finding that alpha particles would backscatter: “It was almost as incredible as if you fired a 15-inch shell at a piece of tissue paper and it came back and hit you,” he later wrote. Rutherford realized that the positively charged part of atoms had to be made of a nucleus, of a size much smaller than the atom itself, but dominating its mass, and electrons orbiting around, determining the atomic size. More details are presented in (Krane, 2021). The chemical properties of the elements are mainly determined by the electrons. Rutherford also considered the nucleus as an agglomeration of protons and neutral particles of similar mass, explaining the existence of isotopes: elements with the same chemical properties but different masses. In 1933 Chadwick discovered the neutron, confirming these ideas. The neutron and proton have about the same mass and, apart from the electric charge, very similar properties within the nucleus. We refer to both neutrons and protons as *nucleons*.

Meanwhile, in 1905, considering how light is observed in different frames of reference, Einstein realized that matter and energy are connected by the now-famous formula: $E = mc^2$. This was essential in understanding subatomic phenomena.

Reflecting on the series of milestones summarized above, we can see that it is not just ideas on the constitution of things that are necessary to reach understanding: the edifice of concepts that are related to how things work is as essential, and sometimes indistinguishable, from the question on what things are made of.

Several puzzles came up as more knowledge was gained with respect to the constituents of atoms. An important one was that it was known from the study of electromagnetism that charged particles in orbits radiate energy. Thus, the model of atoms as electrons in orbits around a nucleus implied that they would lose energy by radiation, eventually collapsing into the nucleus. How could one then explain the stability of atoms? A few years before the discovery of the nucleus, Plank and Einstein had separately explained other observations by assuming that light is quantized into lumps, called photons. Each photon carries energy $E = h \nu$, with ν the frequency of the light, and h a constant, called Plank's constant. Bohr proposed a model in which the angular momentum of the electrons was restricted to having integer multiples of $\hbar = h/2\pi$ in orbits characterizing *quantum states*. Although Bohr's model did not explain why the angular momentum was *quantized* in this manner, it allowed explaining the stability of hydrogen, and matter in general. While excited states of atoms can decay by emission of photons, once the ground state is reached, there is no more radiation: the apparent contradiction with the expectation that particles in orbits radiate energy, was resolved by describing the electron by a *wave function* that determines a probability distribution. Rather than an orbit, we associate a static cloud to electrons in given quantum states. In the ground state there is not any real orbiting. The *electromagnetic interaction* induces transitions between excited states and lower-excitation states. In these *decays* the electrons switch from an initial to a final quantum state, each with a different associated cloud. Bohr's model also allowed to quantitatively explain the spectrum of light emitted by hydrogen (and other elements) when under electrical discharges. Sharp lines of well-defined frequency are predicted by the Bohr model, corresponding to electronic transitions between the allowed states, and they agreed well with observations at the time (Krane, 2021).

What ensued, the development of *quantum mechanics*, was to change our understanding of the behavior and constituents of nature at its most fundamental level. The hypotheses of the Bohr model were derived from a more general construct, developed among others, by Schrödinger and Heisenberg, that allowed understanding many other problems as well. The Coulomb potential from the nucleus generates a potential well and the equations of quantum mechanics show that there are *shells* with well-defined energies. Frequently, each shell can have a few quantum states, with different angular momenta. Remarkably, it was found that particles have an additional degree of freedom, called *spin*, which effectively works as an intrinsic amount of angular momentum, with a finite set of possible orientations. The electron, for example, is a spin- $\frac{1}{2}$ particle, and, in the absence of any orbital motion, will have two possible orientations, up and down, with angular momentum $\pm \hbar/2$ along a chosen axis. Another remarkable property, called the *Pauli exclusion principle*, is that particles with fractional spin (like the electron) cannot share quantum states. Thus, it is impossible to force two electrons with spin up to share all other quantum numbers. In the case of atoms, filled shells tend to be inert, yielding the noble elements. The alkali elements have a single electron in the last shell and tend to combine with elements that are missing one electron to fill a shell. Thus, the periodicity observed by Mendeleev and others is a result of the quantum behavior at atomic scales. More surprises came from quantum mechanics. Studying the union of relativity and quantum mechanics, Dirac showed the equations implied that for every existing particle there should exist a corresponding *anti-particle* with the same mass but opposite electric charge. The positron, the anti-particle of the electron was later found in experiments.

From the times of the discovery of the nucleus, about a century ago, our ideas of nuclear structure have evolved significantly. The nucleus has a size of a few *femtometers* (10^{-15} m), and, in agreement with what Rutherford thought, it is a factor of about 10^5 smaller than the atomic size. While the energies that bound electrons to atoms are on the order of a few *electron-volts* ($1 \text{ eV} = 1.6 \times 10^{-19}$ Joules), the corresponding quantity for nucleons is on the order of a few *Mega-eV* ($1 \text{ MeV} = 10^6 \text{ eV}$). The nucleus is held together by the nuclear force, much stronger than the electromagnetic force. This explains why the positively-charged protons do not fly away from the nucleus, yielding stable nuclei. The chart of nuclei, in Fig. 1, shows squares corresponding to nuclei, with the vertical axis as the number of protons, and the horizontal axis as the number of neutrons. Stable nuclei are shown in black.

Protons and neutrons in the nucleus satisfy the Pauli exclusion principle, just like electrons in the atom. Also, nuclei show excited quantum states, just like atoms show excited electronic states. Although, at first sight, one may imagine that the strong nuclear force, combined with the high density of nuclear matter, could lead to continual scattering and extremely short-lived states, it turns out that the scattering is blocked by the Pauli exclusion principle. The latter forbids scattering into occupied states, and the quantum states have long half-lives, exceeding, by orders of magnitude the time it takes for a nucleon to move through the nuclear volume. Although there is no external trapping potential (as in the atom, where the nucleus provides a Coulomb attraction) the nucleons can be considered as moving in an effective-potential well generated by the neighbors. For this reason, and accounting for the Pauli principle, light stable nuclei show an approximate neutron-to-proton ratio of 1:1, like ^{16}O with 8 protons and 8 neutrons. Fig. 2 shows a sketch of the nuclear potential with quantum states at different excitation energies. As the lower levels are filled, nucleons need to be placed into higher excitation energies, increasing the total energy. As a result, the most stable light-nuclei configurations correspond to nuclei with equal number of protons and neutrons. For the heavier nuclei, the electrostatic repulsion becomes more important, which explains why heavier stable nuclei tend to have a higher fraction of neutrons than light stable nuclei. The nucleus ^{208}Pb , for example, has 82 protons and 126 neutrons. This trend can be noticed following the stable nuclei (black squares) in Fig. 1 as a slope of approximately 1 at the light-nuclei side of the chart, and a smaller slope as nuclei get heavier.

Free neutrons turn out to be unstable, converting into a proton, an electron, and an anti-neutrino, a particle about which we will say more below. What is the origin of this lack of stability? We have discussed how the electromagnetic interaction mediates decays from excited states of atoms toward lower-energy states. In a similar way the weak interaction mediates radioactive decays. These

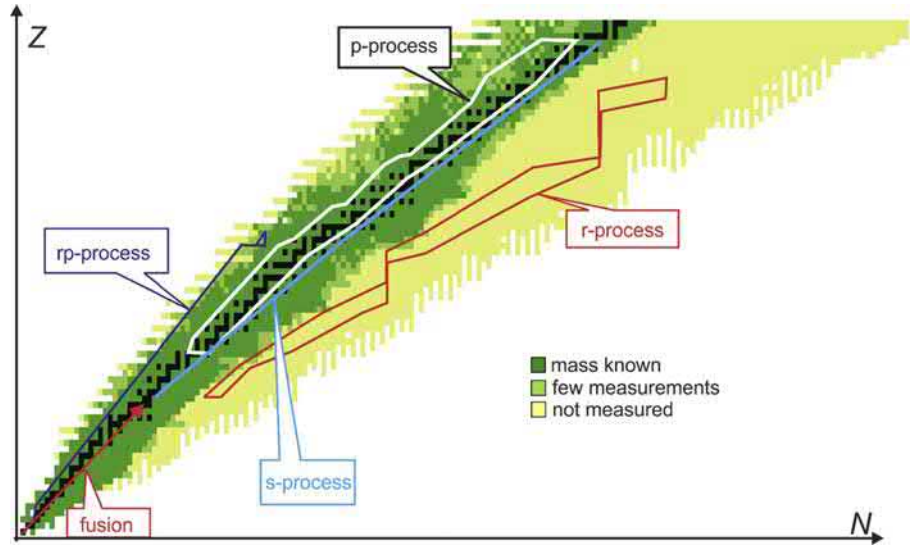


Fig. 1 Chart of nuclei and stellar processes that produce them. The vertical axis is the number of protons, Z , while the horizontal axis is the number of neutrons, N . Each square corresponds to a given nucleus. The chart starts at the bottom left with hydrogen. Stable nuclei are shown in black. Green is used for nuclei for which something has been measured. In yellow are shown nuclei for which we only have calculations, but no measurements. The uncolored nuclei, located beyond the *drip lines*, are not held together by the strong force. The labels indicate stellar processes that involve the nuclei in the different regions. Courtesy: Hendrik Schatz.

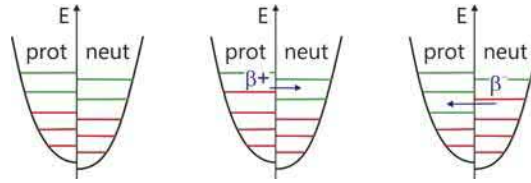


Fig. 2 Sketches of the effect of the Pauli exclusion principle in light nuclei. Red (green) indicates filled (empty) quantum states. The difference between the proton and neutron potentials is due to the electrostatic repulsion. Left: for a given total number of nucleons, identical number of protons and neutrons minimizes the total energy. Middle: a proton-rich nucleus can be unstable and decay by β^+ emission ($p \rightarrow n + e^+ + \nu$). Right: a neutron-rich nucleus can be unstable and decay by β^- emission ($n \rightarrow p + e^- + \bar{\nu}$).

radioactive decays are responsible for the beta radiation observed in the early days. Before the discovery of the neutron, some thought that electrons composing the beta radiation existed in the nucleus before ejection, and that the emission resembled that of alpha-particle emission. In this scenario for radioactive decays, an atom of ^{14}N , for example, would be composed of 14 protons and 7 electrons in the nucleus, and an additional 7 electrons in atomic orbits. This idea was eventually rejected: one consequence of quantum mechanics is that there is a relation between the momentum of particles and their spatial localization, known as the uncertainty principle. For the example at hand, given the small size of the nucleus, it would imply that beta-decay electrons should have much higher energies than were observed. The necessary conclusion was that the beta particles were produced at the time of decays. With the discovery of the neutron, it became clear that ^{14}N , in our example, is composed of 7 protons and 7 neutrons in the nucleus and 7 atomic electrons. While free neutrons decay, they are forbidden to do so by energy conservation in the case of ^{14}N , which is a stable nucleus. The transformation within the nucleus would require placing a proton in a quantum state that is already occupied by other protons, violating the Pauli principle, as shown in Fig. 2. Thus, stable nuclei work as neutron storage places, where neutrons can remain indefinitely, protected from decay. Free protons, on the other hand, are stable. The mass of the neutron is higher than the mass of the proton, so, protons are protected from decaying into a neutron, a positron, and a neutrino, by energy conservation. On the other hand, in proton-rich nuclei, the Pauli exclusion principle implies that protons can get transformed into neutrons without violating energy conservation. In this case the weak interaction mediates the transformation of a proton into a neutron, a positron (the anti-particle of the electron), and a neutrino. This is called β^+ decay, while the decay of the neutron is called β^- decay. Thus, proton-rich nuclei (above the line of stability) decay by β^+ emission and/or electron capture, while neutron-rich nuclei (below the line of stability), decay by β^- emission. (See Fig. 2.) More details on radioactive decays are given in (Previtali, 2021).

The nucleus is a *saturated medium*, very hard to compress, like a liquid. In fact, an early successful model described the nucleus as a liquid drop. The masses of nuclei are significantly affected by the interactions between the nucleons. The difference between the sum of the masses of the constituent protons and neutrons and the mass of the nucleus is called the *binding energy*:

$$\text{B.E.} = Z m_p c^2 + N m_n c^2 - m(Z, N) c^2$$

where m_p , m_n , $m(Z, N)$ are the masses of, respectively, a free proton, a free neutron, and a given nucleus with Z protons and N neutrons. The binding energy is described in more detail in Klay (2021). Here, it will suffice to indicate that the *binding energy per nucleon* is approximately constant at about 8 MeV/nucleon for a large range of nuclei. However, very light nuclei and very heavy nuclei have smaller binding energies than the average. Thus, nuclei can undergo fission or fusion, gain average binding energy, and liberate energy. This is the reason why fission can be used for nuclear reactors and weapons, as described in more detail in Klay (2021) and Wark (2021). Alternatively, the large increase of the binding energy per nucleon between hydrogen and helium can be used to liberate energy via fusion of light elements, as addressed in Morse (2021). The Sun shines from fusion reactions and it is hoped that in a not-too-distant future artificial fusion reactors will exploit this phenomenon in a controlled way.

Nuclear science became more sophisticated and accurate at describing phenomena through the 20th century. However, the problem of calculating nuclear structure is of phenomenal complexity. Nuclear science shares with atomic and solid-state physics the difficulties imposed by the *many-body problem*, but, in addition, the interaction between the nucleons is not given by a simple formula, like the Coulomb potential, for the atomic case. Many ingenious ideas have been used to overcome these difficulties, leading to *phenomenological* approximations that were tuned to reproduce data around a particular set of phenomena and section of the nuclear chart. Up until recently, *ab initio calculations*, that start from the very basic interactions and can explain all phenomena within the same framework, have not been possible. These calculations are now possible for light nuclei, and there are new ideas to extend this revolution in understanding for heavier nuclei. Accelerator facilities for exploring nuclei far from stability are being built to carry out experiments that will eventually allow doubling the number of nuclei accessible for experimentation.

Quarks and other elementary particles

The tale we have told so far seems to indicate that the answer to the question of what the Universe is made of is “protons, neutrons, and electrons.” With the advent of accelerators of higher energies, during the second half of the 20th century, subatomic physicists discovered a myriad of new particles and phenomena. Today we understand that all these particles are made only of the elementary components shown in Fig. 3. These are the modern equivalent of the four elements of Aristotelian days, but are also indivisible, as Democritus wanted, as far as we know. The six quarks and their anti-quarks get combined to make *baryons* with three quarks, qqq , and *mesons* with a quark and an anti-quark, $q\bar{q}$. The proton and neutron turned out to be baryons, composed of two up and one down quark for the proton, and one up and two down quarks for the neutron. There are many other baryons and mesons, but none of them are stable. The leptons, an example of which is the electron, are lighter (within each column) and insensitive to the strong force. A set of the leptons called neutrinos that were mentioned above in the context of beta decay, are electrically neutral and only sensitive to the weak interaction. Either a neutrino or an anti-neutrino is always emitted in nuclear beta decay.

All quarks and leptons are spin- $\frac{1}{2}$ particles. In addition, the *gauge bosons*, are responsible for the interactions. Their spin is an integer number and they are not affected by the Pauli exclusion principle. Finally, there is the Higgs boson, responsible for the masses of the other particles. It had been thought of in the 1960s, when it was predicted by a theory that seemed abstract at the time. The Higgs boson was discovered experimentally in 2012. This finding represented the culmination of many experiments that have confirmed the workings of the theory, which has now become the standard lore and is called the *Standard Model*.

← spin-1/2 particles →				← bosons →		
Quarks	mass charge spin					
	2.4 MeV/c ² +2/3 1/2 up	1.3 GeV/c ² +2/3 1/2 charm	173 GeV/c ² +2/3 1/2 top	0 0 1 gluon	125 GeV/c ² 0 0 Higgs	
	2.4 MeV/c ² -1/3 1/2 down	0.1 GeV/c ² -1/3 1/2 strange	4.2 GeV/c ² -1/3 1/2 bottom	0 0 1 photon		
	Leptons	0.5 MeV/c ² -1 1/2 electron	0.1 GeV/c ² -1 1/2 muon	1.8 GeV/c ² -1 1/2 tau	90 GeV/c ² 0 1 Z	
		≲1.1 eV/c ² 0 1/2 neutrino 1	≲1.1 eV/c ² 0 1/2 neutrino 2	≲1.1 eV/c ² 0 1/2 neutrino 3	80 GeV/c ² +1/-1 1 W^{+/-}	

Fig. 3 The elementary particles. Under each square, the top left has a list of the mass, charge, and spin of each particle. The three left-most columns are the elementary fermions, all spin-1/2 particles. Protons and neutrons are made of up and down quarks: $p \equiv uud$, $n \equiv ddu$. The fourth column lists the gauge bosons, responsible for the interactions. The last column contains only the Higgs particle, responsible for the masses of all the other particles.

Where were the elements produced?

We live in an expanding Universe. First evidence for this came in the late 1920s when Hubble noticed that all stars were moving away from us at a speed proportional to their distance from the Solar system. The rate of expansion can be calculated from the proportionality. It then followed that, at an earlier epoch, the Universe must have been smaller, denser, and hotter. The primordial explosion is called the Big Bang. We now believe that it occurred approximately 13.7 billion years ago. For reference, the oldest rocks on Earth are about 4 billion years old. Initially the Universe resembled a hot soup, with all elementary particles in equilibrium at very high temperatures. As the Universe expanded and cooled down, quarks coalesced into protons and neutrons. As soon as formed, neutrons started decaying, but, before they all decayed, some combined with protons to form ${}^4\text{He}$. The primordial abundance of ${}^4\text{He}$ was about 25% (by mass) and the rest was mostly hydrogen, with traces (less than 0.1%) of deuterium (${}^2\text{H}$), ${}^3\text{He}$ and ${}^7\text{Li}$. At this point electrons were moving too fast to get trapped into atoms, so the Universe still consisted of a plasma. Photons could not move freely through this plasma. For reference, photons take about 10,000 years to get from the center of the Sun to its surface. When the Universe was about 380,000 years old, we believe its temperature was approximately 3000 K, which was cold enough for atoms to form, allowing photons to decouple from the plasma. These photons permeate the whole Universe and can still be seen today as the *cosmic microwave background* (CMB) radiation. Due to the expansion of the Universe, the characteristic temperature of the CMB is now approximately 2.7 K, a factor of approximately 1000 colder than when emitted.

If the primordial elements were mainly hydrogen and helium, where were the rest of the elements made? Many elements are produced by the nuclear reactions that power stars. Typically, the initial gas that makes the star will heat up as it gets denser under gravity. Once the temperature is such that the kinetic energy of the particles allows for nuclear reactions, hydrogen gets transformed into helium via processes that involve several steps. An example of the latter is the, so-called, pp-chain, shown in Fig. 4, which fuels the Sun. Once the hydrogen fuel is consumed to a point that allows for the star to compress further under gravity, helium can be burned, and heavier elements produced. The fate of a star depends on its mass. For the Sun, the helium burning will end with an expansion that will make it puff into a *Red Giant*, extending out to engulfing the orbits of the closer planets. Eventually, some of the mass will collapse again under gravity, leaving behind a *White Dwarf*, with a radius similar to the Earth's radius. Heavier stars continue beyond the helium burning, developing a layered structure, as shown in Fig. 4. Many elements are made in explosive scenarios, like the collapse of an old star under gravity. Fig. 1 indicates sections of the chart of nuclides that are produced in different scenarios. The p-process is the slow capture of protons (extending near stability) and the rp-process, is the *rapid* proton capture, occurring when capture reaction rates are faster than the times for the radioactive decays of the isotopes produced in single-proton captures on stable isotopes. The p-process occurs most likely in core collapse or type Ia supernovae as a photodisintegration process, while the rp process is thought to take place in X-Ray Bursts (XRBs), X-ray pulsars and novae. The s-process is the slow capture of neutrons (like the p-process, near stability, but to the neutron-rich side.) It occurs mainly late in the life of stars after they go through the Red Giant phase. The r-process is the rapid capture of neutrons: this process occurs in sites with very large densities of neutrons, and these nuclei are much further into the neutron-rich area of the chart. While calculations indicated that the collapse of stars producing supernovae were good candidates for the r-process, they seemed to indicate that the neutron densities were not high enough. In 2017, the LIGO collaboration, set up to detect gravitational waves that come from the cosmos, identified the collapse of two neutron stars. LIGO alerted other observatories to orient their instruments in the direction of origin of the gravity wave, and soon images on the different parts of the electromagnetic spectrum (x-rays, infrared, UV, etc.) were recorded by approximately 70 instruments. The results identified the possible production of many r-process nuclei in the collapse of a binary system of two neutron stars. It was a spectacular event! This type of collaborative effort is now called *multi-messenger astronomy* and has become a very fertile field of research.

The pp-chain, mentioned above, is particularly important for us because it fuels the Sun via the conversion of four protons into ${}^4\text{He}$. In this process two of the protons are converted to neutrons. As we indicated above, this does not happen for free protons, but

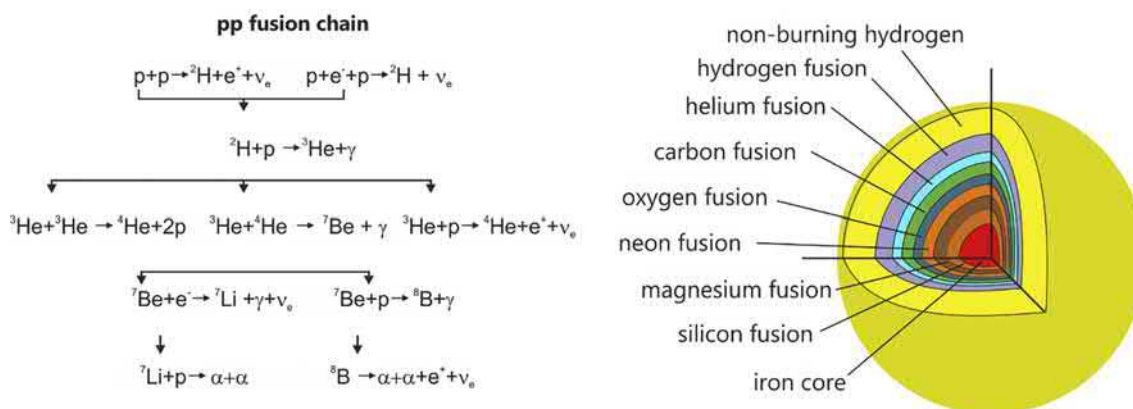


Fig. 4 Left: the pp-chain that dominates the energy generation in the Sun. Right: the layered structure developed by aging massive stars.

can happen through intermediary nuclear reactions in the Sun and other stars. The reactions involve the weak interaction, which makes the process slow and allows the Sun to yield energy at a rate slow enough to last for several billion years.

More knowledge, more questions: Dark matter and dark energy

Most of the mass in our planetary system is in the Sun, so it was expected that stars would account for most of the mass of the Universe. When measurements of the rotational velocities of stars through the Andromeda galaxy were compared to expectations, assuming the approximately known masses of the stars, it was discovered that there is, in addition to the stars, a much larger source of gravity, approximately equivalent to six times the mass of the stars. This is called *Dark Matter*. Using our understanding of primordial nucleosynthesis, we have concluded that it *cannot be baryonic* matter, like protons and neutrons, or electrons. The name thus reflects our lack of understanding of what the substance really is. It has been considered that this could be an artifact due, perhaps, to inaccuracies in our understanding of gravity itself. However, so far, the proposed models for modified gravity have been rejected by experiments. There are many on-going searches to try to identify the nature of Dark Matter.

We mentioned above that in the late 1920s, the Universe was found to be expanding. A projectile that is given an initial kinetic energy in the gravitational field of the Earth will lose kinetic energy as it moves out. Thus, it was expected that we would eventually find out that the Universe's expansion was slowing down. However, in the 1990s it was found that the expansion is actually accelerating. It is thought that this is due to a substance called *Dark Energy*. In terms of the Einstein energy-mass relation, the implication is that approximately 68% of the mass-energy of the Universe is in the form of Dark Energy.

Both Dark Matter and Dark Energy are considered at the top of the list of important unresolved problems in fundamental physics. As we can see, over the years, as we have improved our understanding of what the Universe is made of, we have also gotten to realize that 68% is Dark Energy, 27% is Dark Matter, and the stuff we know rather well, protons, neutrons, and electrons, makes only approximately 5% of the mass-energy balance of the Universe.

This section was supposed to describe what the Universe is made of. We know quite well what the Universe around us is made of: ultimately the particles of Fig. 3, in different combinations. We also know a fair amount about the forces between these particles and how they combine to yield the diversity of nuclei and elements we find around us. However, the adventure toward fully understanding the answer to the title of this section continues and promises more surprises as we unravel these mysteries.

Further reading

Beta decay: (DiGregorio, 2021; Krane, 2021; Previtali, 2021).
 Standard Model: (Griffiths, 2004; Henley and Garcia, 2007).
 Nuclear structure: (Krane, 2021; Napolitano, 2021; Krane, 2008).
 Nuclear astrophysics: (Rolfs and Rodney, 1988; Iliadis, 2007).
 Expansion of the Universe: (Perlmutter, 2012; Schmidt, 2012; Riess, 2012).

Acknowledgments

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See Also: Environmental Protection: The Oceans—Formation and Global Climate Change.

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Discovery of Radioactivity

Daniel E. DiGregorio, School of Science and Technology, National University of San Martín, Buenos Aires, Argentina; and Constituent Atomic Center, National Atomic Energy Commission, Buenos Aires, Argentina

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Glossary

Absorber Any material that stops or diminishes the intensity of the ionizing radiation. A thin sheet of paper or metal will stop or attenuate alpha particles and all except the most energetic beta particles. Lead, steel, copper, aluminum, and concrete attenuate gamma-rays.

Cathode rays A stream of electrons emitted by the cathode, or negative electrode, of a gas-discharge tube or by a hot filament in a vacuum tube, such as an old television tube.

Fluorescence Emission of light by a substance that has absorbed light or other electromagnetic radiation (X-rays, ultraviolet light). Fluorescent substances cease to glow nearly immediately when the incident radiation source stops.

Half-life The time it takes for a quantity of the atoms of a radioactive isotope to be reduced by half. Also called semi-disintegration period.

Ionization Removal of one or more electrons from an atom, leaving it positively charged.

Ionizing radiation Any radiation that is capable of producing ions either directly or indirectly. X-rays, neutrons, alpha, beta, and gamma radiation produce ionization.

Phosphorescence Ability of some materials to absorb light and re-emit light sometime after the exciting light source has been removed. Phosphorescent materials continue to emit light for a noticeable time after the incident radiation ceases.

Transmutation Transformation of one element into a different element by an induced nuclear reaction or a spontaneous nuclear decay.

Introduction

Radioactivity is a phenomenon produced in certain unstable atoms in which nuclei transform by emitting particles and/or electromagnetic radiation. Most of the atomic nuclei found in Nature are stable. However, some nuclei are unstable, turning into another kind while emitting radiation spontaneously; these radioactive nuclei decay over time. Their emission occurs at very variable times, from a fraction of a second up to billions of years. A radioactive atom has exactly the same physical, chemical or biological properties than an ordinary atom, except at the time it decays.

The energetic radiation emitted has been classified using the first three letters of the Greek alphabet: alpha (α), beta (β) and gamma (γ). Alpha radiation is composed of helium nuclei, beta radiation of positively- or negatively charged electrons and gamma radiation of high-energy photons.

Natural radioactivity was discovered at the end of the 19th century, however, since its formation approximately 4600 million years ago, the Earth has been radioactive. Consequently, from its origins, life on our planet was always exposed to a certain level of natural radiation, coming from the decay of radioactive nuclei or produced by cosmic rays.

Natural radioactive isotopes are found in air, water and soil in the environment, including in our bodies and in food. They can be classified as: (i) “*primordial*” radioisotopes, released in the birth and death of stars and created before the formation of the Earth, or (ii) “*cosmogenic*” radioisotopes, produced as the result of reactions between the nuclei of the gases that constitute the atmosphere and the galactic and solar cosmic rays that fall on Earth.

Uranium 238, uranium 235, thorium 232 and potassium 40 are primordial radioisotopes and can still be found in our planet because their half-lives are measured in billions of years. The rest of short half-lives radioactive isotopes produced at the time of the creation of the Universe have gradually disappeared.

Uranium and thorium are the heaviest elements that exist naturally in the Earth crust. The three radioisotopes (238U, 235U, 232Th) disintegrate spontaneously by means of different radioactive processes giving origin to other descendent radioisotopes. All of them, including radon, which is an inert gas, are the main sources of natural radioactivity. They generate what is called telluric radiation: gamma radiation emitted by these radioisotopes found in rocks and in the building materials. Potassium 40 (40K) forms part of all human bodies and it is present in food: milk, cheese, dairy products, vegetables and fruits.

Carbon 14 (14C) and tritium (3H), an isotope of hydrogen, Beryllium 7 (7Be), Beryllium 10 (10Be), are examples of radioisotopes of cosmogenic origin. The existence of these nuclei is permanent on Earth since they are constantly regenerated due to the bombardment of cosmic rays. The half-life of 14C is 5730 years. Plants absorb atoms of 14C along with ordinary carbon meaning that all living matter contains traces of this radioisotope. Due to these properties, Carbon-14 dating is a method that allows to precisely estimate the age of certain archeological objects of a biological origin up to about 50,000 years old.

Natural radioactivity, either from primordial or cosmogenic origins, is present in our everyday lives; it is an integral part of our environment. The air we breathe, the food we eat and our bodies contain radioactive isotopes.

Since the discovery of artificial radioactivity in 1934, new sources of radioactivity have been added to the list of natural sources.

The advent of the nuclear age, in the middle of the 20th century, brought about the production of new radioisotopes, now created by man, among others: cesium 137 (137Cs), strontium 90 (90Sr) and iodine 131 (131I), which are incidental waste products of the military and industrial nuclear development. These radioisotopes, called anthropogenic by their origin, globally added small quantities to the inventory of natural radioactivity present in the environment today.

The anthropological releases of radioactive materials to the air that occurred in the past, were of two kinds: (i) deliberate, due to the many atmospheric nuclear-weapon tests which began in 1950 and reached their peak in 1962, and (ii) involuntary, mainly as the result of the nuclear plant accidents in Chernobyl (1986) and in Fukushima (2011).

Fortunately, on the other hand, the use of artificially-created radioisotopes have allowed important advances in different fields. The most significant results are in medicine, where radioisotopes like: iodine 131 (131I), molybdenum/technetium 99 (99Mo-Tc), cobalt 60 (60Co), fluorine 18 (18F), are used in a variety of beneficial applications of radiation for health diagnoses and therapies.

It is important to emphasize that the exposure to radiation in medical applications, for example, in X-ray radiographs, exceeds the exposure caused by natural radioactivity sources. The effects of all these radiations can be harmful to living cells but, when used in the right way, they have a wide range of beneficial applications.

In this article, we will explore the relevant historical facts from 1895 until the 1930s: hypotheses, experimental researches, questions, discussions and controversies that led the discovery of natural and artificial radioactivity.

Natural radioactivity

Roentgen—December 28, 1895

X-rays were discovered in 1895 by Wilhelm Roentgen when he was working with electrical current flow in a cathode-ray tube. During one of his experiments, he covered the tube with a black paper. The experimental room was completely dark. However, he observed that a screen painted with fluorescent salt, which was on a near table for an independent experiment, gave off light every time an electrical discharge occurred. This phenomenon happened in spite of the fact that neither visible light nor cathode rays could directly reach the screen.

The “unknown radiation” originated from the location where the cathode rays (electrons) hit the glass tube or the anode plate inside the tube. It could pass through materials opaque to light, travel across the room, and activate a fluorescent screen. Roentgen was surprised about his finding which happened just by chance. In view of its uncertain nature, he thus gave the rays somewhat mysterious name of “*X-rays*,” though it also became known as Roentgen rays. Now, we know that X-rays are emitted when electrons lose their velocity due to its interactions with matter (Bremsstrahlung).

Roentgen began to study extensively the new discovered rays in order to determine some of its important properties. His experiments revealed that many materials, were transparent to the X-rays to some degree. This transparency decreased with increasing density of the material. Roentgen was unable to detect appreciable reflection or refraction of the rays, nor was he able to deflect them with a magnetic field. He found that X-rays made photographic plates black.

Roentgen recorded his first observation on November 8, 1895, and he announced the discovery of X-rays on December 28, 1895 in a preliminary paper sent to the secretary of the Physical-Medical Society of Würzburg (Roentgen, 1895). Then, he wrote only two more papers on X-rays (Roentgen, 1896, 1897). The scan of his wife’s hand was published in newspapers all around the world and caused quite a commotion. As of mid-January 1896, doctors and physicist in all Western countries were using the X-ray apparatus described by Roentgen to perform scans of their own and, at the end of December 1896, the bones of a broken arm were joined using X-rays. He was awarded the first Nobel Prize in Physics in 1901, for the discovery of X-rays.

Antoine Henri Becquerel—end of February–March 1, 1896

The discovery of X-rays by Roentgen was a matter of chance. The observation of radioactivity, however, was definitely the consequence of a mistake and a following accidental event.

Henri Becquerel decided to study the existence of a possible relation between X-rays and the phosphorescence phenomena in uranium salts. In those days, he thought, mistakenly, that minerals made phosphorescent by visible light, might emit X-rays. Becquerel had a wide collection of minerals, many of which showed phosphorescence. In order to test his hypotheses, he placed a phosphorescent uranium salt on top a photographic plate wrapped in black paper (to protect it from direct light) and exposed the whole set-up to strong sunshine for several hours. When he developed the plate, it showed the expected blackening with the image of the uranium salt. Initially, [Becquerel \(1896a\)](#) concluded that this result confirmed his hypotheses and published his results.

He wanted to continue the experiments but the sun did not shine in Paris for a few days at the end of February 1896. Since he was unable to expose another uranium salt to the sun, he put away the mineral placed on top of a fresh sheet of photographic plate together in a dark drawer of a cabinet. A few days later, he developed the photographic plate expecting a weak image and he was surprised to find out that it was totally black as intense as in the original sunlight experiment ([Becquerel, 1896b](#)). He discovered that penetrative rays emitted by the salt left an impression on the plate. With this new evidence, he had to change his previous conclusions: the exposure from the uranium salt came even in the dark and it was not related to sunlight ([Becquerel, 1896c](#)).

Becquerel systematically performed further experiments by using a variety of uranium minerals. He determined that several uranium salts, whether phosphorescent or not, as well as metallic uranium had the same property ([Becquerel, 1896c,d](#)). Therefore, he established that the radiation was an intrinsic characteristic of the uranium and it ionized air, as well as X-rays do. [Becquerel \(1896e\)](#) referred to the discovered penetrative rays as “uranic rays.”

Marie Skłodowska Curie and Pierre Curie—1897

Marie Skłodowska was born in Warsaw and she came to graduate school in Paris, because women in her country were not admitted to the University. She had married Pierre Curie in 1895, who had a well-known reputation working on piezoelectricity and magnetism. In November 1897, Marie Curie wanted to make her doctoral thesis in physics. Henri Becquerel recommended her to continue with his research on uranium minerals as the subject for her doctoral thesis.

She decided to conduct a systematic research on the “uranic rays” found in a number of elements, compounds and minerals. She measured quantitatively the ionization in the air created by the radiation of the uranium samples. The experimental setup consisted in an electrometer with a piezoelectric quartz developed by Pierre and his brother Jacques Curie, capable of measuring very low values of electrical current.

The experiments confirmed that the intensity of the radiation emitted by the uranium minerals was proportional to the amount of uranium present in it. In her systematic measurements, she also found that thorium minerals also behaved similar to uranium minerals ([Curie, 1898](#)). The emission of the “uranic rays” was a more general property of matter. Marie Curie proposed the name “radioactivity” to describe this phenomenon. She also noted that radioactivity appeared to be an atomic property.

Pierre Curie understood the importance of the results of these experiments, so he left his research on crystals and decided to join his wife in the search for the unknown radioactive element.

After examining several uranium minerals, such as pitchblende (uranium oxide) and chalcocite (copper and uranium phosphate), Marie Curie and her husband found that these minerals were much more radioactive than one would expected from the uranium content. The conclusion was that it contained other unknown radioactive components.

Starting from several tons of uranium mineral pitchblende, Marie and Pierre Curie succeeded in isolating, and thus discovered, two previously unknown radioactive elements, present in trace amounts in pitchblende. They announced the existence of a new element, named “polonium” (Po) in honor of Marie’s native country Poland, in a paper published on July 1898 ([Curie and Curie, 1898](#)). In this publication, the Curies used for the first time the word “radioactivity” even though H. Becquerel discovered it.

Later in the same year, the Curies, with the help of G. Bemont, announced the existence of a second new element, which they called “radium” (Ra) for its high intense radioactivity ([Curie et al., 1898](#)). Polonium was 60 times more radioactive than uranium and radium was 400 times more. Radium half-life is 1600 years very much shorter than 4.5×10^9 years of the uranium (^{238}U) half-life, therefore, for a same weight, its radiation is much more intense. There is one gram of radium in about three tons of natural uranium.

Pierre and Marie Curie continued their investigations trying to find new radioactive elements in pitchblende together with members of their research group. André Debierne announced the discovery a new radioactive element in a publication ([Debierne, 1899](#)). Later, this element became known as actinium.

Henri Becquerel, Marie and Pierre Curie won the 1903 Nobel Prize in Physics. Becquerel was awarded this prize for the discovery of spontaneous radioactivity and the Curies for their joint researches on these radiation phenomena. In 1911, Marie Curie won a second Nobel Prize, this time in Chemistry, for the discovery of the elements radium and polonium.

Types and properties of nuclear radiation

Ernest Rutherford— α , β , γ rays—1899

After the discovery of radioactivity, many physicists all over the world devoted their lives to study and to understand the properties of the different types of radiation emitted by radioactive nuclei.

In October 1895, Ernest Rutherford, a 24 year-old young New Zealander, arrived in England to work at the Cavendish Laboratory in Cambridge. A month after his arrival, the announcement of the discovery of X-rays by Roentgen shook the scientific world and public opinion. In 1897, J.J. Thomson guided his young assistant on the Becquerel rays that had just startled the scientific world.

Rutherford began his studies on the radiation emitted by uranium in 1898. His experiments consisted of studying the ionization of gases produced by this radiation. In 1899, the results led him to conclude that radioactivity is not homogeneous and must have at least two components: (a) a component which ionizes very heavily, but it is absorbed in a single sheet of paper, and (b) a component capable of penetrating several mm of Cu, Al, etc., which causes much less ionization. In summary, depending on their ability to penetrate matter and ionize air, Rutherford (1899) called them: alpha-radiation and beta-radiation, respectively.

At the end of 1899, several experiments performed in Germany (Giesel, 1899), Austria (Meyer and Schweidler, 1899a,b) and France (Becquerel 1899), demonstrated that a part of the rays emitted by radium were easily deflected in the presence of a magnetic field. From the direction of deflection, it was concluded that the “deviating” rays, named β -radiation by Rutherford, behaved like negatively charged particles. Curie and Curie (1900) demonstrated directly that the β -rays from radium had a negative charge. Becquerel (1900a), in an independent experiment, showed that the same rays were deflected in an electric field in the same direction as cathode rays.

Knowing that beta-rays were deflected by both magnetic and electric fields, Becquerel proved that they were electrons. He measured the charge-mass ratio (e/m) for these rays by the method that J.J. Thomson had used to study cathode rays and identify the electron. He found that the ratio was the same as Thomson’s electrons; therefore, Becquerel (1900b,c) suggested that the beta-radiation was, in fact, made out of electrons.

In 1900, Villard found that radium salts emitted an extremely penetrating radiation capable of being detected through a thin layer of lead or about 20 cm of iron (Villard, 1900). In 1903, Rutherford proposed to call gamma-radiation to Villard’s discovery because it was much more penetrating than alpha-radiation and beta-radiation that he had already distinguished and named in 1899. Shortly the gamma-radiation was proven to be electromagnetic radiation just as light, but with much shorter wavelengths and higher energies.

Fig. 1 displays a sketch similar to the one presented by Marie Curie in her Ph.-D thesis published in June 1903. The three kinds of rays (α , β and γ) emitted by a radium source, have different trajectories in the presence of a magnetic field. Alpha-rays are helium nuclei positively charged, thus they deflect to the left; beta-rays are electrons much lighter and negatively charged, thus they deflect

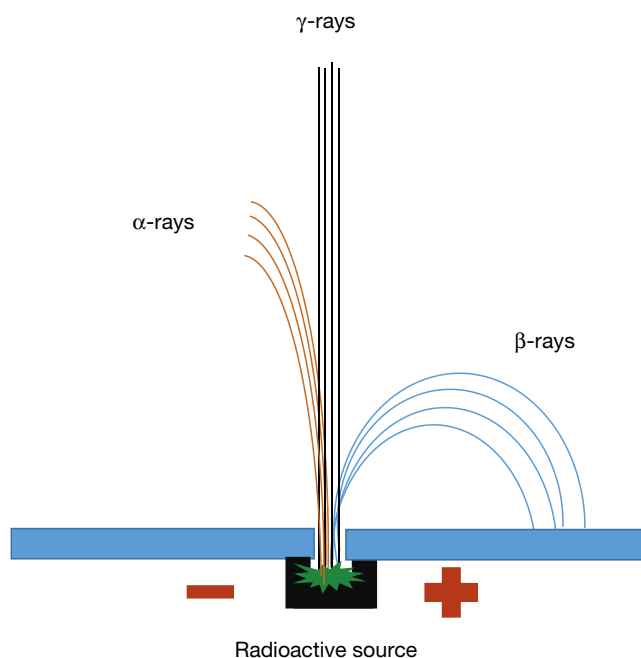


Fig. 1 A scheme similar to that presented by Marie Skłodowska-Curie in her Ph.D. thesis to describe some properties of the three kinds of radiation (α , β and γ) emitted by a radioactive source. They have different trajectories in a magnetic field. Alpha-rays, which are helium nuclei positively charged, deflect to left; beta-rays are electrons much lighter and negatively charged, deflect to right, and gamma-rays, which are very high-energy electromagnetic waves, do not deflect in the magnetic field since they are neutral.

to the right; and gamma-rays, which are very high-energy electromagnetic waves do not deflect in the magnetic field since they are neutral.

The definitive identification of alpha-radiation took a much longer time. Ramsay and Soddy (1903a,b) performed experiments in London and showed that helium is produced from radium emission. It was in 1909 when Rutherford and Royds (1909) performed an experiment to demonstrate that alpha particles were identical to double ionized helium. They put a large quantity of purified radon gas, an alpha-emitter, enclosed in a glass tube with very thin walls (0.01 mm thick) enough to allow the alpha particles go through to another evacuated tube surrounded the thin-walled tube containing radon. After a week, the whole spectrum of helium was visible. The alpha particles emitted by the radon gas gave rise to the characteristic helium spectrum.

Ernest Rutherford and Frederick Soddy—uranium decay series—1902

In the autumn of 1898, Ernest Rutherford was appointed professor of physics at McGill University in Montreal despite being only 27 years old. A few years later in 1901, Frederick Soddy, a young talented English chemist, joined Rutherford to work on radioactivity.

In 1902, while doing experiments on thorium compounds, Rutherford and Soddy discovered that radioactivity was associated with changes inside the atom that transformed thorium into a different element. They found that thorium continually generated a chemically different substance that was intensely radioactive. The radioactivity eventually made the new element disappear. They identified the radioactive emanation of radium as radon and demonstrated experimentally that radioactivity is the spontaneous transformation of one element into another through the emission of radiation (Rutherford and Soddy, 1902a,b, 1903).

The polonium and radium extracted from pitchblende and discovered by Pierre and Marie Curie were also direct descendants of the uranium it contained. When uranium and thorium emitted alpha and beta particles, they decayed into different elements. This sequence of transformations later was called the “*natural decay series*.” By studying the chemistry of the natural decay series of uranium (U) and thorium (Th), 30 new radioactive isotopes were discovered, of which radium (Ra) was the best known. During these times, 1903, the existence of a nucleus inside the atom was unknown. This was proposed by Rutherford in 1911 to explain the results of the series of experiments performed together with H. Geiger and E. Marsden.

The existence of isotopes was proposed by Soddy in 1913. He discovered that all radioactive preparations were not unique elements, but rather that some of them were variants of known elements. In other words, certain atoms could have the same chemical properties, but have different properties when it came to radioactivity.

Soddy (1913) coined the term “*isotopes*” in a letter to the editor published that year. Isotopes were atoms of certain elements that might exist in forms that differed on atomic weight while being indistinguishable and inseparable chemically, and therefore, they should occupy the same position in the Periodic Table.

We must remember that neutrons were not discovered until 1930. Nowadays, isotopes are atoms of the same nuclear charge but different mass; namely, they contain the same number of protons (are the same element) but different numbers of neutrons. From the detailed chemistry of the decay series Soddy, and others, noticed that the chemistry of the daughter element was connected with the emitted radiation.

Therefore, the researchers noticed that alpha particle emission moved the element two places down the Periodic Table, while beta emission moved the corresponding element one place up the Table.

Rutherford and Soddy obtained Nobel prizes in Chemistry. Ernest Rutherford won the Nobel Prize in 1908, thanks to these discoveries in the area of the disintegration of the elements and in the chemistry of the radioactive elements. Frederick Soddy won the Nobel Prize in Chemistry in 1921 for his investigations in the chemistry of radioactive substances and the origin and nature of isotopes.

The natural decay series provided valuable evidence for the idea that radioactivity accompanies atomic transmutation, one of the fundamental principles of subatomic physics. The disintegration of atoms of radioactive elements was a key discovery at that time, as atoms were believed to be a class of indestructible matter.

Exponential decay law—1903

The discovery of natural radioactivity by Becquerel in 1896 marked the beginning of the study of the atomic nucleus. Becquerel's observations of radioactivity were only qualitative. Pierre and Marie Curie (1902) mistakenly believed that each atom of radioactive material worked as a constant energy source, drawing it from the environment. They claimed that the radioactivity of uranium, thorium and radium did not change in time based on experiments for several years. It is known now that the half-lives of uranium and thorium are of the order of billions of years, and it is 1600 years for radium. Therefore, these half-lives are too long to obtain a measurable variation of their radioactivity even in months or years.

On the contrary, between 1901 and 1903, Rutherford and Soddy (1902a,b, 1903) derived a quantitative law from their experiments performed on thorium. Watching the process, they discovered that the decay of a radioactive element was not constant but decreased exponentially with time following the law $N(t) = N_0 e^{-\lambda t}$, where $N(t)$ is the number of radioactive nuclei at time t , N_0 is the number at $t = 0$ and λ is the probability for any particular nucleus to decay per unit time. The period or “half-life” for a specific radioactive nucleus is defined as $T = \ln 2 / \lambda = 0.693 / \lambda$, and represents the period of time in which half of the nuclei of a particular radioactive element disintegrate.

Fig. 2 shows a plot of the radioactive exponential decay law. It can be observed that the number of nuclei in a radioactive sample at a given time t , $N(t)$, reduces to one-half of the initial value N_0 after a time T (half-life). After two half-lives $2T$, N_0 reduces to one-fourth; after three half-lives $3T$, it reduces to one-eighth, and so on.

Rutherford and Soddy proved that the half-life was a characteristic property of each radioactive element having its own intrinsic half-life. If the radioactive atoms resulting from such a decay are also radioactive, they will in turn decay with their own “half-life.” This is the case for the disintegration of uranium and the thorium nuclei where we have a whole series of successive decays.

Half-lives can range from a fraction of a second to billions of years. The exponential time dependence is a characteristic of all radioactivity and indicates that it is a statistical random phenomenon. The exponential behavior observed by Rutherford and Soddy demonstrated that radioactivity is an irreversible transmutation process of matter.

Development of artificial radioactivity

James Chadwick—the neutron—February 1932

Prior to the discovery of the neutron, it was thought that atomic nuclei were made of protons, α particles and electrons. In 1920 Rutherford, had suggested that there might be a neutral particle, possibly a proton and an electron tightly bound together, which he called a neutron.

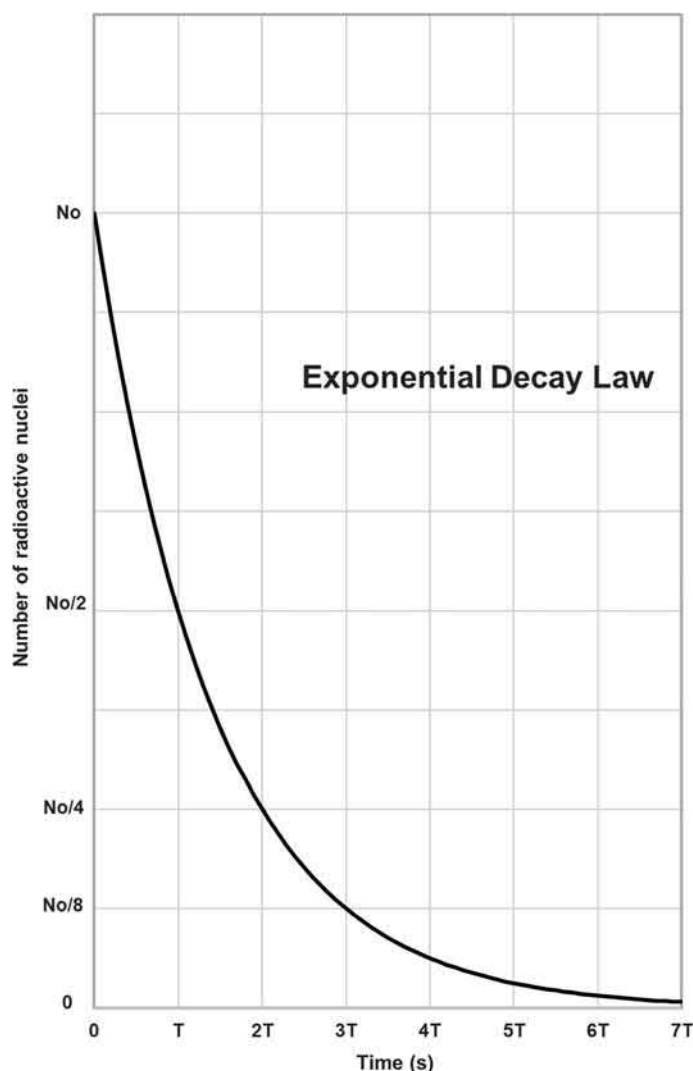


Fig. 2 Plot of the exponential decay law. The number of radioactive nuclei at given time t is, $N(t) = N_0 e^{-\ln 2 t/T}$, where N_0 is the number of nuclei at $t = 0$ and T is the “half-life” for a specific radioactive nuclei and represents the period of time in which half of the nuclei of a particular radioactive element disintegrate. It can be observed that the initial number of nuclei N_0 reduces to $N_0/2$ after a time T ; after two half-lives $2T$, reduces to $N_0/4$; after three half-lives $3T$, it reduces to $N_0/8$, and so on.

The 1930s were considered the golden decade of physics and chemistry. Research groups all over the world were making new discoveries about fundamental properties of matter. Their findings were published almost month by month, sometimes by a difference of weeks or even days.

In 1930, in Berlin, W. Bothe and H. Becker performed several experiments bombarding light nuclei (beryllium, boron, lithium) with α -particles (${}^4\text{He}_2$) emitted from a radioactive polonium source (Bothe and Becker, 1930). They observed the production of a very penetrating electrically-neutral radiation that they thought to be very high energy γ -rays (photons). In the case of beryllium, the proposed nuclear reaction was ${}^9\text{Be}_4 + {}^4\text{He}_2 = {}^{13}\text{C}_6 + \gamma$.

Soon afterwards, in Paris, Frederick Joliot and his wife, Irene Curie, repeated these experiments using a much stronger polonium α source. They measured the absorption of the unknown γ -rays using sheets of lead, aluminum, carbon, etc., with different thicknesses and found that the radiation had more penetration power than estimated initially by Bothe. They found that 4.7 cm of lead were needed to reduce the intensity of the radiation by one-half. Calculating the absorption expected by Compton scattering process, they found the energy of the γ -rays from the bombardment of beryllium with α particles to be about 15 MeV. This was much higher than the energy of γ -rays emitted by naturally radioactive substances. However, the Joliot-Curies reached a similar conclusion to that obtained by the German team: the radiation produced was very high-energy γ -rays (Curie, 1931; Joliot, 1931).

Early in 1932, they performed new experiments bombarding beryllium with α particles. This time they found that placing a sheet of material such as paraffin, which contained hydrogen, between the “assumed” γ -rays and their ionization-chamber detector produced a secondary radiation that greatly increased the ionization in their detector. They correctly assumed that this secondary radiation had the ability to eject protons. By measuring the range of the protons on air, they found the energy of the protons to be 5.7 MeV. In order to eject a proton from a hydrogenated substance with such energy, the “assumed” γ -rays must be at least 50 MeV. This result was quite unexpected and clearly showed a contradiction on the γ -ray hypothesis for the emitted primary radiation. They promptly argued the possible existence of a new mode of interaction of radiation with matter (Curie and Joliot, 1932a,b).

Just after the publication of Curie and Joliot (1932a) on their new experimental results, English physicist James Chadwick reported them to Rutherford. Both were not convinced about Curie-Joliot’s conclusions. Immediately Chadwick decided to repeat the experiment at the Cavendish Laboratory in Cambridge.

Chadwick thought that it was unlikely that protons were ejected by Compton scattering, as proposed by Curie and Joliot. He put forward two arguments: it required γ -rays of very high-energy, and the number of protons observed was several times greater than predicted for this scattering process. Therefore, he decided to study the properties of the radiation produced when beryllium was bombarded by polonium α particles. He exposed many materials to this radiation, including gases of hydrogen, helium, oxygen, nitrogen, and argon.

Chadwick’s experiments showed that the beryllium radiation ejected particles of about the same number from all the gases. Considering the ionization power and range, the particles ejected from hydrogen behaved like protons with an energy of 5.7 MeV; this recoil energy of the hydrogen atoms required incident gamma-rays of about 50 MeV. In the case of nitrogen, 90-MeV gamma-rays were needed to produce the recoil of nitrogen atoms observed with energy of 1.4 MeV. It is important to note that gamma-rays from radioactive decay are in the energy range from a few keV to approximately 8 MeV.

After 10 days of intense work, in February 1932 Chadwick, in a letter published in Nature (Chadwick, 1932a), concluded that his experiments and the previous Joliot-Curies’ data proved the possible existence of the neutron, which Rutherford had proposed 12 years earlier. Chadwick pointed out in his communication: “These results were very difficult to explain on the assumption that the beryllium radiation was a quantum radiation, if energy and momentum are to be conserved in the collisions. The difficulties disappear, however, if it be assumed that the radiation consists of particles of mass 1 and charge 0, or neutrons.” Therefore, the reaction he proposed was ${}^9\text{Be}_4 + {}^4\text{He}_2 = {}^{12}\text{C}_6 + {}^1\text{n}_0$. A more detailed account of Chadwick’s experimental results and conclusions appeared in the Proceedings of the Royal Society on May 1932 (Chadwick, 1932b).

In 1935, James Chadwick received the Nobel Prize in physics for this work. After Chadwick’s neutron discovery, the idea that electrons were permanent constituents of nuclei was abandoned. Instead, the nucleus was assumed to contain N neutrons and Z protons, a total of $A = N + Z$ particles.

Carl Anderson—The positive electron or “positron”—August 1932

In 1930, there were only three known particles: the electron, the proton and the photon. The discovery of the neutron in 1932 was a bonus to physics and allowed scientists to advance in the knowledge of the atomic nucleus. That same year, Carl Anderson, an American physicist at Caltech, announced the discovery of a positively charged particle with the mass of the electron.

While Anderson was studying cosmic rays by taking photographs produced in a Wilson cloud chamber, he observed unexpectedly particle tracks. He visualized them by the steam condensed into fine droplets along the track of electrically charged particles crossing the chamber. A magnetic field produced by a coil bent the tracks, which allows determining the charge sign, positive or negative, of the particle. He found a number of tracks whose orientation suggested that they were caused by positively charged particles; however, these particles were too small in mass to be protons (Anderson, 1932a,b, 1933).

In August 1932, after many experiments and observations, Anderson determined that the tracks were “positrons,” particles with the same mass as the electron, but with opposite electrical charge.

Early in 1932, I. Curie and F. Joliot failed to discover the positron during certain experiments they performed a few months before Anderson’s discovery. By analyzing the cloud chamber photographs of the neutrons and γ -radiation produced in the reaction ${}^9\text{Be}_4 + {}^4\text{He}_2$, they observed “several Compton electrons which seemed indeed to behave strangely with respect to the direction of

the magnetic field." They reported that while some tracks could be ascribed to Compton electrons passing through the volume of the chamber, "several tracks, having the same appearance as the electron tracks, showed a curvature opposite to that of others" (Curie and Joliot, 1932c).

The positron had been conjectured in 1928 by Paul Dirac, an English theoretical physicist, who proposed that electrons might have a positive charge and negative energy. Although a new particle was not explicitly predicted, the solutions of the Dirac equation did allow electrons to have either positive or negative energy. However, the positrons were observed and understood by Anderson without taking into account Dirac's conjecture. Soon after the positron was considered as the "antiparticle" of the electron.

Carl D. Anderson received the Nobel Prize in Physics 1936 for the discovery of the positron. He shared the prize with Victor F. Hess who discovered cosmic radiation.

Irene Curie and Frederick Joliot—artificial radioactivity—1934

At the beginning of 1934, Irene Curie and Frederic Joliot published several articles in which they announced that they had artificially produced new radio-elements (Curie and Joliot, 1934a,b; Joliot and Curie, 1934a,b).

The Joliot-Curies showed that by bombarding various non-radioactive light elements (boron, magnesium, aluminum) with α -particles emitted by a polonium source, they could create new radioactive isotopes. They observed that the irradiated material remained radioactive during a reasonable (long) period after the end of the bombardment. They also found that the new created radioisotopes emitted positrons in the case of aluminum and boron, while positrons and electrons were emitted in the case of magnesium.

In their experiments, the Joliot-Curies came across simultaneously two phenomena: one, it is possible to produce radioactive nuclei artificially, and two, the product of some nuclear-reaction decays emit a positron (e^+ , an antielectron), or an electron (e^-), now called β^+ or β^- decays.

They suggested that the phenomenon takes place in two stages. First, the capture of the α -particle and the instantaneous expulsion of a neutron, with the formation of an unstable radioactive nucleus. Second, the unstable nucleus decays transmuting into a stable nucleus.

In the case of the bombardment of a sheet of aluminum by α particles, a new radioactive nucleus, which they called "radio-phosphorus," was created; instantaneously, a neutron was ejected, and the remaining unstable nucleus decayed exponentially with a half-life of about 3 min emitting a positron and a positron.

The transmutation reaction was $^{27}\text{Al}_{13} + ^4\text{He}_2 \rightarrow ^{30}\text{P}_{15} + ^1\text{n}_0$. The new radioactive nucleus created was $^{30}\text{P}_{15}$, which was disintegrated by emitting positrons (e^+) and a neutrino (ν), giving a stable nucleus $^{30}\text{Si}_{14}$. In a formula, it can be written as, $^{30}\text{P}_{15} \rightarrow ^{30}\text{Si}_{14} + e^+ + \nu$.

The existence of neutrinos was proposed by Wolfgang Pauli in 1930, in a letter to the participants of a physics conference in Tuebingen, Germany. He suggested that a third particle might be emitted in β decay in order to explain the apparent non-conservation of energy after the emission of an electron (or positron) from an atomic nucleus.

Similarly, the Joliot-Curies interpreted the production of radioactive elements in the bombardment of B and Mg. In case of boron, the nuclear reaction was $^{10}\text{B}_5 + ^4\text{He}_2 \rightarrow ^{13}\text{N}_7 + ^1\text{n}_0$. The new radioactive nucleus was $^{13}\text{N}_7$, which disintegrated with emission of positrons and a neutrino with a half-life of 14 min, giving rise to a stable nucleus $^{13}\text{C}_6$. Therefore, the corresponding formula can be written as, $^{13}\text{N}_7 \rightarrow ^{13}\text{C}_6 + e^+ + \nu$.

The energy spectrum of the emitted positrons measured in the Joliot-Curies' experiment had a continuous form, similar to the β -ray spectrum, which perhaps, it might be attributed to the simultaneous emission of a neutrino in order to satisfy the principle of the conservation of energy and of angular momentum in the transmutation. The maximum energy of the measured positron spectrum was of the order of 1.5×10^6 eV for boron and 3×10^6 eV for aluminum. The radioactive elements formed during the irradiation were chemically separated from boron and aluminum, which as expected, had the chemical properties of nitrogen and phosphorus, respectively.

The Joliot-Curies were not able to observe similar effects when they used α -particles to bombard hydrogen, lithium, beryllium, carbon, nitrogen, oxygen, fluorine, sodium, silicon, or phosphorus. They ascribed this result, at least for some cases, to the too short half-lives of the products which made them impossible to be observed.

In their publications, the Joliot-Curies also suggested that radioactive elements might be produced "in different nuclear reactions with other bombarding particles: protons, neutrons, deuterons, neutrons."

Irene Curie and Frederic Joliot won the Nobel Prize of Chemistry in 1935, in recognition of their synthesis of new radioactive element.

The discovery of artificial radioactivity was fundamental for nuclear science and an important fact for the entire world. It brought the attention and inspired scientists all over the world to the possible use of different approaches to produce unknown radio-elements.

In Rome, Enrico Fermi had the idea to produce new radioactive elements by using neutrons, a particle electrically neutral, discovered 2 years earlier. He thought that neutrons might be more effective to make artificial radioactivity since they would not be repelled by the positive charge nuclei of the bombarded target atoms. However, the process could be more complicated because the neutrons were not produced spontaneously.

In June 1934, Fermi and his group from the University of Rome were able to discover more than 20 new radioactive isotopes using this method of neutron-induced nuclear reactions (Fermi, 1934; Fermi et al., 1934a,b).

They also made another important discovery. They proved that by decreasing the neutron energy (slowing down the neutrons) before they hit the target atoms, the production of new radioactive isotopes was more effective than using energetic neutrons, so called fast neutrons.

The cross-section of neutron absorption in nuclei largely increases when neutron velocity decreases. Slow neutrons turned out to be important in another phenomenon: induced nuclear fission, the discovery of which was a major event in the middle of the twentieth century.

Fission is a decay process that occurs when natural uranium nuclei (^{235}U , ^{238}U) are bombarded with slow neutrons and they break apart into two smaller, lighter nuclei. In 1938, the first induced fission was discovered by the German physicists Otto Hahn, Fritz Strassmann, and Lise Meitner (Hahn et al., 1937; Meitner et al., 1937). Spontaneous fission in ^{238}U was discovered in 1941 by Russian physicists Georgy Flyorov and Konstantin Petrzhak. It was difficult to distinguish between induced and spontaneous fission events because cosmic rays produce some neutrons,

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Discovery of the Quantum Structure of Atoms and Their Nuclei

Kenneth S. Krane, Department of Physics, Oregon State University, Corvallis, OR, United States

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Quantization

In physics, the term *quantization* refers to a process of transition from the domain in which classical laws apply, to the domain (often atomic or subatomic) in which it is necessary to apply the principles of quantum mechanics. More broadly, it refers to the necessity to take into account the discontinuity or graininess of certain variables. For example, a satellite can be placed into Earth orbit in any desired circular or elliptical path, but for an electron in an atom only certain orbits are allowed. In quantization of the electromagnetic field, the continuity of a wave-based description is replaced by the discontinuity of discrete particle-like quanta called *photons*.

Quantization of electric charge

In the 1800s, experiments with electric phenomena involved the transfer of large quantities of charge, such as in electric currents in circuits, so the question of the existence of a fundamental unit of charge did not arise. The first indication of a discrete unit of charge was revealed in the experiments of J. J. Thomson (1856–1940) at Cambridge University, in England.

Thomson's research was mainly focused on the properties of cathode rays, mysterious rays in gas discharge tubes (evacuated glass tubes containing two electrodes, one positive and the other negative). The rays appeared to leave the negative terminal (the cathode) and travel toward the positive terminal (the anode). In 1897, Thomson was able to deflect the rays in electric and magnetic fields, thus showing that they were particles, and he measured the ratio of their charge and mass. Thomson maintained that the particles, which he called *corpuscles* but are now known as *electrons*, were constituents of atoms and were released from the atoms of the small amount of residual gas in his discharge tubes.

More definitive evidence for the quantization of electric charge came in 1909 with the experiments of Robert Millikan (1868–1953) at the University of Chicago. By suspending oil droplets having just a few electric charges in an electric field, Millikan was able to measure the amount of charge on the drops. He observed that the charges were always small integer multiples of a quantity that he took to be the basic unit of charge. That basic unit of charge is now set at exactly the value $e = 1.602176634 \times 10^{-19}$ coulomb. All observed atomic, nuclear, and subnuclear particles have electric charges q that are small integer multiples of this basic unit of charge: $q = \pm ne$. Some theories have postulated the existence of fractional electric charges, and experiments with subatomic particles have been consistent with the predictions of these theories, but no isolated particles with fractional charges have so far been observed.

Quantization of energy

Although the 19th century had seen many successes in understanding the physical universe, particularly the development of the laws of electromagnetism and thermodynamics, a few unsolved problems remained. One in particular was the spectrum of *thermal radiation*, electromagnetic radiation (often in the infrared and visible regions) emitted by all objects because of their temperature T . This failure was especially troubling, because it suggested the inadequacy of the two great triumphs of 19th century physics, electromagnetism and thermodynamics.

The intensity of thermal radiation falls to zero at both large and small wavelengths, and it reaches a peak in between (Fig. 1). The result obtained from classical theory, called the *Rayleigh-Jeans formula*, approaches the data points at large wavelengths but fails completely at small wavelengths, where it predicts infinite radiation intensity:

$$I(\lambda) = \frac{2\pi c}{\lambda^4} kT \quad (1)$$

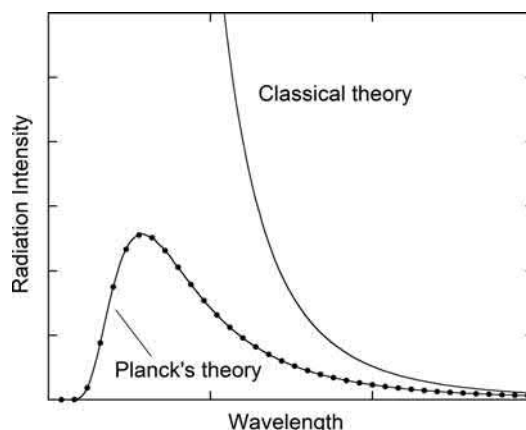


Fig. 1 The data points represent measurements of the spectrum of thermal radiation, which the classical theory cannot explain but which are fit perfectly by Planck's quantum theory.

where λ is the wavelength of the radiation, c is the speed of light, and k is the Boltzmann constant. The radiation intensity is defined so that $I(\lambda)d\lambda$ is the radiant power per unit area (measured, for example, in watts/m²) in the small wavelength interval $d\lambda$, such as might be absorbed by a suitable detector in the path of the radiation that is sensitive to wavelengths between λ and $\lambda + d\lambda$.

The solution to this dilemma was found in 1900 by Max Planck (1858–1947), working at the University of Berlin, in Germany. Planck analyzed theoretically the thermal radiation in a hollow metal box with the walls at temperature T . The radiation filled the inside of the box and could escape through a small hole in one of the walls. Planck considered the process by which the oscillating atoms in the walls of the box absorb and re-emit the reflected radiation. Clearly Planck had to find a way to suppress the intensity of short-wavelength (high-frequency) radiation that caused the failure of the classical theory.

He accomplished this by postulating that the atoms could absorb and re-emit radiation in discrete bundles that were integer multiples n of a basic quantity of energy ϵ . This basic quantity (or *quantum*) of energy was in turn determined by the frequency of the oscillating atoms (which was assumed to be the same as the frequency f of the radiation), so that $\epsilon = hf$, where h is the constant of proportionality between the energy and frequency, now known as *Planck's constant*. Thus Planck's quantization assumption is $E = n\epsilon = nhf$. An oscillating atom in the walls of the cavity can have 2 types of energy (kinetic and potential), and from statistical mechanics we learn that the average value of each form of energy of an atom at temperature T is $kT/2$. So on the average, the thermal energy kT of the atoms is not sufficient to provide an energy of hf at high frequency, and thus the short-wavelength (high-frequency) radiation is suppressed. With this assumption, Planck was able to derive an expression for the radiation intensity:

$$I(\lambda) = \frac{2\pi hc^2}{\lambda^5} \frac{1}{e^{hc/\lambda kT} - 1} \quad (2)$$

Planck's function fits the data perfectly, as shown in Fig. 1.

Five years later, in 1905, Albert Einstein (1879–1955) extended Planck's quantization ideas to the radiation itself, asserting that the radiation could be considered as composed of *light quanta*, now known as photons. On that basis, Einstein derived equations for the photoelectric effect, and in 1915 Millikan's experiments confirmed Einstein's theory.

Rutherford scattering and the nuclear atom

In 1895, Ernest Rutherford (1871–1937) traveled from New Zealand, where he earned his undergraduate physics degree, to the Cavendish Laboratory at Cambridge University in England to continue his physics education. There he worked under the supervision of J. J. Thomson. At first, Rutherford's research involved the transmission and detection of radio waves, a continuation of the investigations he had begun as an undergraduate in New Zealand. At one point Rutherford had the record for the largest distance over which radio waves could be transmitted and received.

The discovery of X rays in 1895 by Wilhelm Röntgen (1845–1923), in Germany, provided a new dimension to the research of Thomson and Rutherford. They set out to measure the conduction of electricity by various gases exposed to X rays (then known as Röntgen rays). Their hypothesized explanation for the effect, that the X rays removed electrons from neutral atoms, thereby changing them into charged ions, turned out to be correct. Rutherford published three papers describing these experiments, one jointly with Thomson in 1896 and two in 1897 under Rutherford's sole authorship.

Yet another dimension to the research arose in 1896 with the discovery of radiation emanating from uranium and its compounds by the Frenchman Henri Becquerel (1852–1908). In his second paper of 1897, in which he measured the speeds of the gas ions, Rutherford concluded:

Further experiments, which are not yet completed, have been made to find the velocity of the ions of a gas conducting under the influence of the radiation given out by uranium and its salts. It can be shown that the velocity of the ions in the conducting gas is the same as when the gas is acted on by Röntgen radiation, so that the carrier in the two cases is identical. Further results, however, must be reserved for a future paper. (Rutherford, 1897).

As promised, 2 years later Rutherford would produce the more detailed work on the effect of uranium radiation on electrical conduction by gases. Although he published several more papers on this subject, it is obvious from these papers that his interest was shifting from the effects of this radiation on gases to the origin and properties of the radiation itself.

In 1898, Rutherford left Cambridge to become Professor of Physics at McGill University in Montreal, Canada. He would remain there for 9 years, during which he published 70 papers on the subject of uranium radiation, or *radioactivity* as it soon came to be known. The research described in these publications included the sequence of elements produced in the radioactive decays of uranium and thorium and the chemical extraction and identification of the various decay products (research for which he would be awarded the 1908 Nobel Prize in Chemistry), as well as the nature of the alpha, beta, and gamma radiations emitted in the decays. Most of his publications in his last year in Canada (1906–07) concerned the nature of the alpha particle. In one experiment in particular (Rutherford, 1906), he obtained a photographic image of the pattern of alpha particles emitted by a source on a wire as they passed through a narrow slit. By covering half of the slit with a thin layer (0.003 cm) of mica, he could observe the amount of scattering of the beam due to the mica. His result is shown in Fig. 2, in which the bottom half of the image shows the more diffuse result of passing through the mica. Rutherford estimated that the average deviation of the beam was about 2° , and he concluded:

...it can easily be calculated that the change in direction of motion of some of the α particles in passing through the thickness of mica (0.003 cm) would require over that distance an average transverse electric field of about 100 million volts per cm. Such a result brings out clearly that the atoms of matter must be the seat of very intense electrical forces—a deduction in harmony with the electronic theory of matter.

In 1907, Rutherford returned to England as Professor and Chair of Physics at the University of Manchester, the personally hand-picked successor to the retiring Physics Chair, Arthur Schuster. Arriving in Manchester somewhat unexpectedly during a school holiday, he found the physics laboratory deserted except for a young German postdoctoral researcher, Hans Geiger (1882–1945), who had arrived the year before to work with Schuster after completing his doctoral degree in 1906 in Erlangen, Germany. Geiger's year at Manchester had not been particularly successful (only one minor paper published), and he was set to return to Germany to continue working in the area of his doctoral research, the passage of high currents through gases in discharge tubes. Geiger showed Rutherford around the laboratory, and Rutherford was so impressed with the young German that he asked him to stay on as his assistant. Despite having no background or experience in research with radioactivity, Geiger accepted and remained at Manchester for a total of 5 years, eventually becoming recognized as one of the world's leading experts on radioactivity.

Rutherford first assigned Geiger to obtain a more quantitative result for the diffuse scattering of α particles. Geiger designed an apparatus that allowed the α particles to travel through an evacuated tube (thus eliminating scattering by air), pass through a narrow slit (that could be covered by a foil), and then strike a fluorescent screen, on which Geiger could observe the flashes of light (scintillations) caused by the impact of individual particles and count them with a microscope. Fig. 3 shows Geiger's results for 0, 1, and 2 gold foils covering the slit. Geiger concluded that the average scattering angle for one gold foil was about 0.2° (Geiger, 1908).

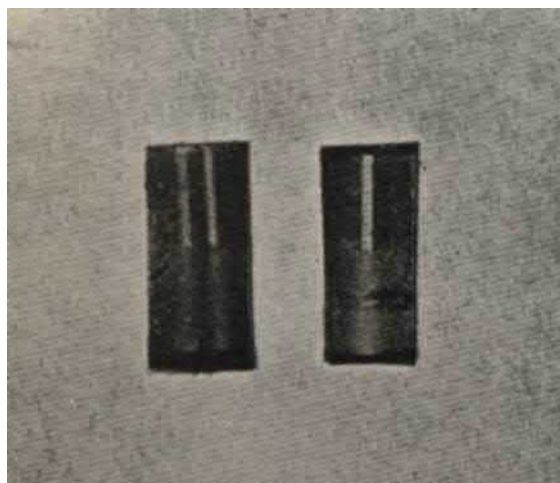


Fig. 2 Rutherford's alpha-particle scattering results, in which the particles pass through a narrow slit whose bottom half is covered by a thin layer of mica (Rutherford, 1906).

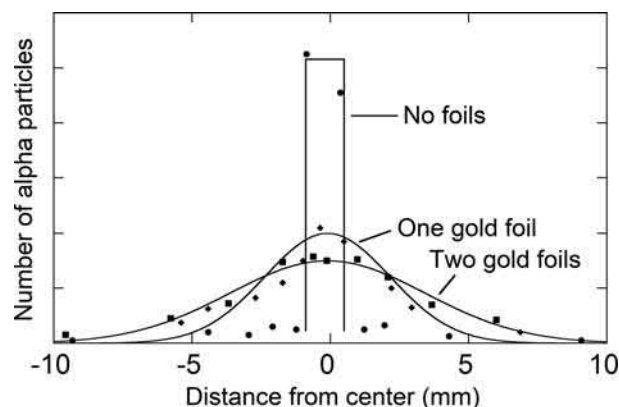


Fig. 3 Geiger's alpha-particle scattering results. (Data from Geiger, 1908).

Now working with undergraduate student Ernest Marsden, Geiger next used a similar arrangement with a scintillator and microscope to try to detect α particles reflected from foils, that is, scattered at angles greater than 90° . They concluded that about 1 particle in 8000 is reflected from the foil (Geiger and Marsden, 1909). Later Rutherford would recall upon hearing of their result:

It was quite the most incredible event that ever happened to me in my life. It was almost as incredible as if you fired a 15-inch shell at a piece of tissue paper and it came back and hit you (Rutherford, 1938).

Why was Rutherford so surprised at this result? Based on the prevailing model of the structure of the atom as a diffuse ball of positive charge in which the negative electrons were embedded (proposed by J. J. Thomson and known as the *Thomson model* of the atom), large scattering angles were expected only from multiple scatterings. If each scattering resulted in a deflection of about 0.2° (the average scattering angle) as Geiger had previously found, then a scattering angle of 90° would require at least 450 consecutive scatterings, *all in the same direction*. In its passage through the foil, the α particle is expected to strike atoms randomly and thus scatter with equal probability to the left or to the right, so such a result would be somewhat like tossing a coin 450 times and having it come up heads every time, which would have a probability of about 1 in 10^{135} . This enormous difference between the expected probability for backscattering (1 in 10^{135}) and the observed probability (1 in 8000) is what shocked Rutherford.

Later Geiger would recall:

One day (in 1911) Rutherford, obviously in high spirits, came to my room and told me he now knew what the atom looked like and how to explain the large deflections of the alpha particles. On the very same day I began an experiment to test the relation expected by Rutherford between the number of scattered particles and the angle of scattering (Eve, 1939).

Rutherford proposed that the mass and positive charge of the atom were not diffused throughout its volume, as in the model proposed by J. J. Thomson, but instead are concentrated in a tiny region at its center – the atomic nucleus. The large-angle deflections in this interpretation were then simply the result of an alpha particle encountering an object more massive than itself. Each scattering would then be the result of a single encounter, rather than of a succession of encounters. Based on this concept, Rutherford derived an expression for the number of scattered particles that would appear at a scattering angle ϕ :

$$N = \frac{(kze^2)^2 ntI}{4r^2 m^2 V^4 \sin^4 \phi / 2} \quad (3)$$

where k is the electronic constant (sometimes written as $1/4\pi\epsilon_0$), ze is the electric charge of the α particle, Ze is the electric charge of the nucleus, n is the density of atoms in the scattering foil, t is the thickness of the scattering foil, I is the intensity of the incident beam of α particles, r is the distance of the detectors from the scattering foil, and m and V are the mass and speed of the α particle. (Rutherford, 1911)

If Rutherford's hypothesis were correct, then the scattering should depend on (1) the square of the atomic number Z of the scattering material, (2) the thickness of the foil, (3) the speed of the α particles, and (4) the scattering angle. Geiger and Marsden set out to test all four of these dependences in a detailed series of experiments. (Because the concept of atomic number had not yet been developed, Geiger and Marsden did the test in terms of atomic weight instead.) Two sets of their results are shown in Fig. 4 (Geiger and Marsden, 1913). In Fig. 4A, the dependence of the scattering rate on the foil thickness is shown. This dependence is a critical test of the nuclear model, because according to Thomson's model the scattering should depend on the foil thickness as $t^{1/2}$, but Fig. 4 clearly shows the linear dependence on t as predicted by Rutherford's formula. This dependence serves clearly to differentiate

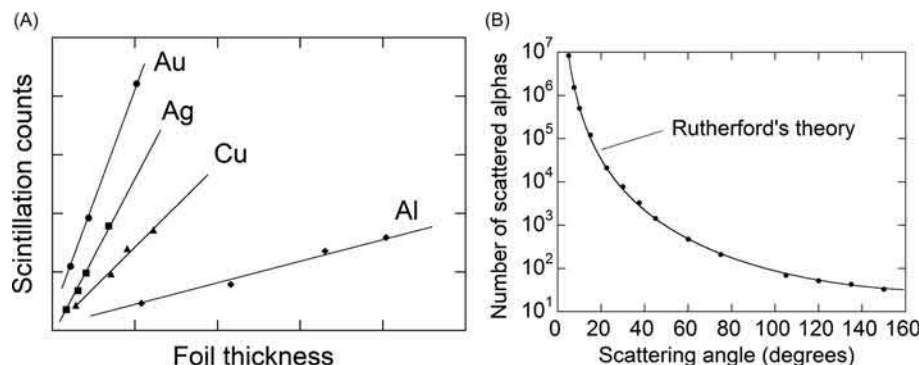


Fig. 4 Geiger and Marsden results for alpha scattering. (A) Thin foils of gold, silver, copper, and aluminum all show a linear dependence on the thickness. (B) The scattering probability as a function of scattering angle is perfectly fit by Rutherford's theory. (Data from Geiger and Marsden, 1913).

scattering by a single event (t dependence, as in Rutherford's model) from scattering from multiple events ($t^{1/2}$ dependence, as in Thomson's model). Fig. 4B shows the dependence on the scattering angle. Rutherford's formula, with the dependence of $\sin^{-4}(\phi/2)$, fits the data perfectly and constitutes a great triumph for Rutherford's idea and calculation, as well as for the careful and precise experiments of Geiger and Marsden.

Atomic energy levels and the Bohr model

The idea that matter was composed of atoms is an ancient abstraction, but it was given substance in the 1800s by English chemist John Dalton (1766–1844), who suggested that atoms of various elements could combine in definite proportions to form chemical compounds. The impetus for a physical model of atomic structure was engendered by the previously mentioned discoveries of the 1890s: the electron by Thomson, in 1897, and radioactive decay by Becquerel, in 1896. Electrons, including those observed as cathode rays in discharge tubes as well as those released from metallic surfaces illuminated by visible and ultraviolet radiation (the photoelectric effect), presumably did not arise spontaneously but were instead constituents of atoms ripped loose by electric fields, thus putting an end to the 19th-century idea of atoms as *indivisible*. Rutherford's research showed that in radioactive decay one type of atom would transform into another, ending the idea of the atom as *immutable*. It then became possible to imagine the existence of a physical model for the internal structure of atoms.

Although it was believed that electrons were constituents of atoms, the number of electrons in the atom was far from clear. Being electrically neutral, atoms must contain equal quantities of positive and negative charge, but the nature of the positive charge was unknown. Moreover, it was well known from classic electromagnetic theory that an assembly of positive and negative charges at rest could not achieve mechanical equilibrium. As a result, early models put the electrons in motion around or through the positive charge, but this led to another problem: as with all accelerated charges, the orbiting electrons would radiate electromagnetic energy, resulting in the collapse of the structure.

In the late 19th and early 20th century, atomic spectroscopists studied the light and other radiations emitted by atoms, for example when a vapor of an element is heated or when an electric current passes through the vapor. In the process they identified many hundreds and occasionally thousands of spectral lines from individual elements in the visible and near visible (ultraviolet, infrared) regions of the spectrum. Using equipment such as precisely ruled diffraction gratings, the wavelengths of these spectral lines had been accurately measured, often to a precision of less than 1 part in 10^4 . The early atomic model builders faced a challenge in accounting for the many different wavelengths or frequencies of the emitted radiations. It was initially assumed that each individual frequency was associated with an electron oscillating about its equilibrium position, suggesting that to account for the multitude of spectral lines there were many hundreds or even thousands of electrons in each atom.

An early model due to Thomson proposed that the electrons moved without resistance in an atom-sized sphere of massless positive charge. All of the mass of the atoms resided in the electrons, which must then number around $2000A$ for an atom of mass number A (an integer closest to the mass of the atom in atomic mass units, now known to be the total number of protons and neutrons in the nucleus). Soon scattering experiments showed that the number of electrons was instead of order A , which indicated that most of the mass of the atom must reside in the sphere of positive charge of unknown character. Rutherford's 1911 theory and the 1913 experiments of Geiger and Marsden showed that the positive charge was not diffuse but instead was concentrated at the center of the atom in a nucleus of diameter 10^{-5} times that of the atom.

Somewhat surprisingly, Rutherford's revelation did not lead to an immediate flurry of new atomic models based on the nuclear atom. Instead, the task fell to a young Danish visitor to Rutherford's Manchester lab in 1912, Niels Bohr (1885–1962). Bohr had finished his doctorate degree in physics at the University of Copenhagen in 1911; his dissertation (written in Danish) dealt with the theoretical analysis of the behavior of electrons in metals. With funding for postdoctoral study, Bohr chose to work at Cambridge

University with Thomson, who Bohr felt would help him advance his theoretical research into the properties of electrons. Bohr had produced an English translation of his dissertation, which he hoped Thomson would read. However, Thomson seemed indifferent to Bohr's research and the two never forged an effective working relationship. In December 1911, Bohr attended a talk in Cambridge given by Rutherford, and Bohr was so impressed with Rutherford's research on radioactivity that he arranged to leave Cambridge in March 1912 to spend the rest of his postdoctoral year in Rutherford's lab in Manchester. Bohr's goal at that time was to acquire some experimental proficiency in the field of radioactive decay. To that end, he participated in a laboratory practicum on radioactivity that Hans Geiger had developed as training for new students who wanted to undertake advanced research with Rutherford.

Bohr completed the radioactivity practicum in May 1912 and while waiting for radioactive material for the project he was assigned by Rutherford, he continued to work on his electron theory of metals. But within about 2 months he turned his attention to a topic that was more in line with the main thrust of the research at Manchester: the internal structure of the atom. His first project was a theoretical analysis of the passage of charged particles (alpha and beta particles from radioactive decays) through matter, in particular the effect of the bound electrons in atoms. He completed the analysis in August of 1912. Among his conclusions were that the H_2 molecule contained two electrons (thus one electron in the H atom) and the He atom contained two electrons. The number of electrons in an atom was at that time not well established, so this was the first step into Bohr's insight into the structure of the atom (Bohr, 1913a). He also tried to determine the number of electrons in more massive atoms, but the results had large uncertainties due to the then-current accepted values of the basic physical constants.

Bohr returned to Copenhagen in September 1912 and continued to work on the theoretical analysis of the structure of atoms. He published his results in a series of three papers in the *Philosophical Magazine* (Bohr, 1913b, c, d). The first of those papers discussed his model for the structure of atomic hydrogen. He began with a planetary structure, with a single electron (charge $-e$) in a circular orbit of radius a about a nucleus (according to Rutherford's model) of charge $+e$. (Other planetary models had been tried previously without success; they were based on a multitude of electrons circulating in a series of coplanar rings, like the planet Saturn.) Setting the electric force¹ e^2/a^2 equal to the centripetal force $ma\omega^2 = mv^2/a$ yields an expression for the orbital kinetic energy, $K = \frac{1}{2}mv^2 = e^2/2a$. (Here m is the electron's mass, ω its angular speed, and v its tangential speed around the circular orbit.) Combining with the potential energy $U = -e^2/a$ gives the total mechanical energy of the electron $E = K + U = -e^2/2a$. The negative value of E suggests a bound system, with binding energy (the energy necessary to remove the electron from the atom) $W = -E$. Manipulating these expressions, Bohr obtained two equations for the orbital parameters – the frequency f and the diameter $2a$:

$$f = \frac{\omega}{2\pi} = \frac{W^{3/2}\sqrt{2}}{\pi e^2\sqrt{m}} \quad (4)$$

$$2a = \frac{e^2}{W} \quad (5)$$

This orbital motion has mechanical stability, but the circulating electron would still radiate electromagnetic energy. Bohr avoided this problem by postulating that the electron could exist in special *stationary states* without radiating.

These two equations contain three unknowns: the orbital frequency f , radius a , and binding energy W , so it is not possible to solve for the values of all three parameters. Bohr needed another relationship among the parameters. For the missing ingredient, he applied the quantization condition $E = nhf$ invented by Max Planck in 1900 in his analysis of thermal radiation. Following Planck's lead, Bohr proposed that the binding energy could be expressed as $W = cnhf$, where c is a numerical factor of order unity. Bohr guessed at the value $c = 0.5$ by taking the average between the fully bound electron (binding energy = W) and the unbound electron (binding energy = 0). Subsequently he developed a more rigorous derivation that remarkably agreed with his guess.

Bohr's great insight was to marry the classical physics that gave Eqs. (4), (5) with the Planck-Einstein quantum theory, thereby producing the first quantum theory of atomic structure. Combining his quantum equation with the classical equations he obtained one equation for each of the three parameters:

$$W_n = \frac{2\pi^2 me^4}{n^2 h^2} \quad (6)$$

$$f_n = \frac{4\pi^2 me^4}{n^3 h^3} \quad (7)$$

$$2a_n = \frac{n^2 h^2}{2\pi^2 me^2} \quad (8)$$

The subscript n on each value indicates that the parameter takes different values for each value of the index n . Bohr assumed that the largest value of W would give the most stable state of the atom, so he put $n = 1$ and used the then-current values of e , m , and h to calculate $W_1 = 13$ electron-volts, $f_1 = 6.2 \times 10^{15}$ Hz, and $2a_1 = 1.1 \times 10^{-8}$ cm; he concluded that these were of the correct order of magnitude as ionization potentials, optical frequencies, and atomic sizes.

¹The notation adopted here follows as closely as possible that of Bohr's paper, in particular using cgs units for electromagnetic quantities. At the conclusion the switch to SI units can be made by replacing e^2 with $e^2/4\pi\epsilon_0$.

The problem remained to calculate the frequencies or wavelengths of the radiations emitted by atomic hydrogen. The frequencies of the orbital motion did not agree with those of the observed radiations. Bohr puzzled over this problem, but as soon as one of his colleagues made him aware of the Balmer series for the emitted radiations from hydrogen, it suddenly became clear to him. Johann Balmer (1825–98) was a Swiss mathematician who had (by trial and error) in 1885 discovered a numerical relationship for a set of frequencies in the visible and ultraviolet radiations emitted by hydrogen:

$$f_R = f_0 \left(\frac{1}{4} - \frac{1}{n^2} \right) \quad (9)$$

where $f_0 = 3.29 \times 10^{15}$ Hz and $n = 3, 4, 5, \dots$. Here f_R indicates the frequencies of the emitted radiation, as distinguished from the orbital frequencies f_n .

Balmer's formula led Bohr to two insights: (1) It was no accident that the binding energies W_n and the radiant frequencies f_R both depended on the index n as n^{-2} . The binding energies and emitted frequencies must be related to one another. (2) The minus sign in Eq. (9) suggests that the frequencies depend on the *difference* of two binding energies. Bohr proposed that the energy difference $W_{n_2} - W_{n_1}$ in passing from a state with $n = n_1$ to a state with $n = n_2$ appears as the energy hf_R of the emitted radiation, so that

$$f_R = \frac{W_{n_2} - W_{n_1}}{h} = \frac{2\pi^2 me^4}{h^3} \left(\frac{1}{n_2^2} - \frac{1}{n_1^2} \right) \quad (10)$$

Clearly putting $n_2 = 2$ gives the frequencies of the Balmer series, and Bohr calculated the value of the quantity outside the parentheses in Eq. (10) to be 3.1×10^{15} Hz, which agrees with Balmer's value within the then-current uncertainties of Planck's constant and the electron mass and charge. Putting $n_2 = 3$ reproduces the values of a series of radiations in the infrared region, called the Paschen series, and Bohr predicted the frequencies of the then-undiscovered series of radiations with $n_2 = 1, 4$, and 5, which turned out to agree with his formula once measured values were available. Moreover, a similar set of spectral lines observed in stars was soon shown to follow Bohr's theory for ionized helium (He^+), which also contains a single electron. In the case of He^+ , the electron is circulating about a nucleus with a charge of $2e$, and the change in the electric force equation introduces a factor of 4 into Eq. (10). The perfect agreement with the frequencies of the H and He spectral lines was a great triumph for Bohr's theory (Bohr, 1913e).

In summary, Bohr's theory for hydrogen represents the atom by a series of discrete levels (stationary states), in which the electron can circulate without radiating. Radiation is emitted only when the electron makes a transition from one state to another. The energy of the radiation is equal to the difference in energy of the initial and final states. Fig. 5 shows some of the electron states in atomic hydrogen and some of the emitted radiations. In more modern terminology, the energy levels are said to be *quantized*, and the energy difference appears as the energy of an emitted photon or quantum of radiation.

Additional confirmation of the idea of discrete states in atoms came within a year of Bohr's publication. In Germany, James Franck and Gustav Hertz carried out an experiment in which they passed a beam of electrons through mercury (Hg) vapor (Franck and Hertz, 1914). When the electron beam was accelerated through 4.9 V, the electron current suddenly dropped. Franck and Hertz interpreted this as evidence that the electrons in the beam were losing energy by colliding with Hg atoms and in the process driving the atoms from their ground state to an excited state known to exist at 4.9 eV. This energy difference corresponds exactly to an ultraviolet radiation emission in Hg of wavelength 254 nm. Although Bohr's theory was not able to calculate the excited states in a many-electron atom like Hg, the notion of radiation emission between discrete energy states was consistent with Bohr's ideas.

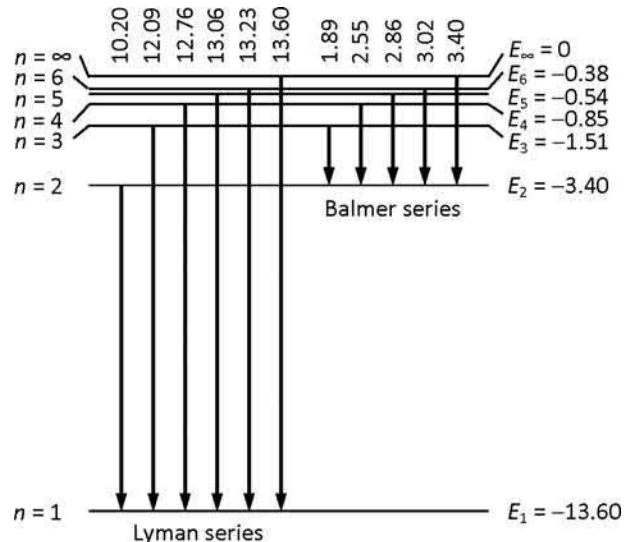


Fig. 5 Some energy levels in atomic hydrogen and their emitted radiations. All energies are given in units of eV.

For his discoveries, Bohr was awarded the 1922 Nobel Prize in physics. He was honored in particular for “his services in the investigation of the structure of atoms and of the radiation emanating from them.”

Bohr also was able to obtain a more rigorous justification for the factor $c = 0.5$ that he used in adapting Planck’s formula to the structure of hydrogen. Bohr’s guess amounted to taking the mean or average value between the energy of the unbound electron (zero) and the fully bound one. Instead, Bohr compared (for arbitrary value of c) the orbital frequency of the electron, as in Eq. (7):

$$f_n = \frac{\pi^2 m e^4}{2c^3 n^3 h^3} \quad (11)$$

with the frequency of the emitted radiation for a transition from a state n to state $n-1$:

$$f_R = \frac{\pi^2 m e^4}{2c^2 h^3} \left(\frac{1}{(n-1)^2} - \frac{1}{n^2} \right) \approx \frac{\pi^2 m e^4}{c^2 n^3 h^3} \quad (12)$$

where the last result is an approximation for large values of n . Bohr asserted that in the classical limit of the quantum atom, the atom will radiate at the orbital frequency, so the limit of the quantum frequency must match the orbital frequency. Comparing Eqs. (11) and (12) immediately gives $c = 0.5$. This agreement of classical and quantum physics in the intermediate range is known as the *correspondence principle*.

In developing his model, Bohr introduced not only the frequency condition (Eq. 7) but also the fundamental quantum ideas of the correspondence principle and nonradiating stationary states. It would be another decade before the formal mathematical basis of quantum mechanics was established (Schrödinger, 1926), incorporating these ideas. The full calculation reproduces Bohr’s energy levels but eliminates such classical concepts as orbits in fixed planes, and answers an objection raised by Rutherford in response to Bohr’s theory: how does the electron know beforehand which state it is going to jump to? The quantum calculation gives a set of probabilities to reach any final state from any initial state.

Although Bohr also tried to calculate the orbits for electrons in atoms with more than one electron, in his 1913 papers he did not have available the methods of quantum mechanics that a dozen years later would provide the tools necessary to analyze more complex atoms. Nevertheless, the rules he provided, and especially the idea that the emitted radiations corresponded to electron transitions between stationary states, would provide the keys to understanding the structure of atoms.

Quantization of angular momentum

The classical angular momentum of a particle of mass m moving in a circular orbit of radius r with tangential speed v is a vector $\mathbf{L}$ perpendicular to the plane of the orbit and of length $|\mathbf{L}| = mvr$. Taking the velocity from the kinetic energy (Eq. 6) and the radius from Eq. (8) gives

$$|\mathbf{L}| = mvr = m \left(\frac{2\pi e^2}{nh} \right) \left(\frac{n^2 h^2}{4\pi^2 m e^2} \right) = n \frac{h}{2\pi} \quad (13)$$

As Bohr states in his paper (Bohr, 1913b), “...the angular momentum of every electron round the center of its orbit will in the permanent state of the system be equal to $h/2\pi$...”. By “permanent state,” Bohr means lowest energy state, corresponding to $n = 1$ in Eq. (13).

Eq. (13) indicates that the orbital angular momentum cannot take any arbitrary value, but is restricted to the set of values $nh/2\pi$. In analogy with the quantization of energy and electric charge, this is known as the *quantization of angular momentum*. The full quantum mechanical calculation preserves this principle but gives a very different set of values – the length of the angular momentum vector is

$$|\mathbf{L}| = \sqrt{l(l+1)} \frac{h}{2\pi} \quad (14)$$

where $l = 0, 1, 2, \dots$. Note that in the lowest energy state, the quantum result is $|\mathbf{L}| = 0$, while in Bohr’s calculation, $|\mathbf{L}| = h/2\pi$. Moreover, in the quantum calculation the components of the vector $\mathbf{L}$ along any particular axis are limited to the values $mh/2\pi$, where m is an integer ranging from $-l$ to $+l$. That is, the vector $\mathbf{L}$ can take only certain directions in space corresponding to the allowed components on the chosen axis. This property is known as *spatial quantization*.

Quantized energy levels in nuclei

The acceptance of Bohr’s theory of atomic structure was enhanced by the excellent agreement between the energy levels he calculated from his planetary model and the extremely precise values of the energies of the emitted radiation quanta (photons) determined from the wavelengths of the emitted radiations measured by atomic spectroscopy. Unfortunately, attempts to carry over the methods to the structure of nuclei in the 1920s were hampered by the lack of a theory to calculate the expected energy levels of nuclei and the rather poor ability to measure the energies of the emitted photons (nuclear γ rays).

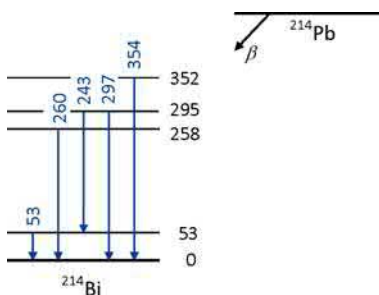


Fig. 6 Transition energies determined from internal conversion (Rutherford et al., 1930), shown in blue, compared with the presently accepted levels of ^{214}Bi . All energies are given in units of keV.

The situation was rescued somewhat by the existence of a competing process to γ -ray emission in nuclei, namely *internal conversion*. In γ -ray emission, the nucleus makes a transition from an initial energy level E_i to a final energy level E_f and emits a γ -ray photon whose energy hf satisfies the Bohr frequency condition $hf = E_i - E_f$. In internal conversion, the available transition energy $E_i - E_f$ instead ejects one of the atomic electrons. The kinetic energy K_e of the emitted electron is equal to the available transition energy less the energy cost to free a bound electron from the atom, the electron binding energy B_e , so that $K_e = (E_i - E_f) - B_e$. Values of the electron binding energies were already known in the 1920s, so by doing precise measurements of the kinetic energies of the emitted internal conversion electrons (for example, using a magnetic spectrometer) it was possible to deduce the nuclear energy levels.

Although using present-day equipment the energies of γ -ray photons can be measured to a precision often below 1 part in 10^4 , thus enabling equivalently precise conclusions about the energy levels in nuclei, in the 1920s the precision of electron energy measurements far exceeded that of γ -ray photons, and thus internal conversion often provided the key to understanding nuclear energy states.

For example, consider the energy levels of the nucleus ^{214}Bi populated in the radioactive β decay of ^{214}Pb (known as radium B).² Measurements of the internal conversion electrons revealed the following energy differences of the nuclear states (energies in keV): 53, 243, 260, 297, 354, with uncertainties of typically a few parts in 10^3 . (Rutherford et al., 1930) Knowing only the differences in energy of the nuclear states does not permit the ordering of the states to be deduced, but modern measurements led to the currently accepted energy levels of ^{214}Bi shown in Fig. 6. Note how well the early measurements of the transition energies match the accepted energy levels.

An alternative approach to studying quantum states in nuclei is to observe the α decays of radioactive species. High-resolution observations of the α particles from particular decays showed several different energies from the same decay process (called the *fine structure* of α decay). These were interpreted by Gamow (1930, 1937) in the case of the decay of ^{212}Bi (known as ThC) to ^{208}Tl as decays from the parent nucleus to different energy levels in the daughter, which could be verified by also studying the same energy levels by internal conversion. If the hypothesis were correct, the energy level differences deduced from the α particles should match those from internal conversion.

Fig. 7 shows the α decay and internal conversion transition energies reported by Ellis (1932). Assuming that the highest-energy α decay of ^{212}Bi leads to the ground state of ^{208}Tl , the remaining α decays lead to the energy levels shown in the figure. The internal conversion lines then fit nicely into the same scheme of energy levels. For example, the energy difference of α_0 and α_1 is 41 keV,

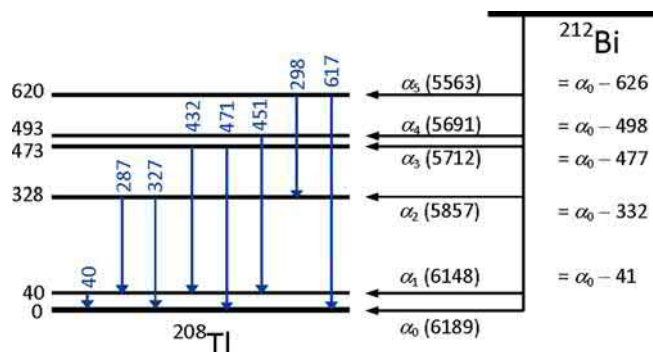
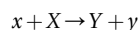


Fig. 7 Differences in alpha decay energies (shown at right) agree with energy level intervals (shown in blue) from internal conversion electrons (Ellis, 1932) and with the presently accepted values of the energy levels shown at left. All energies are given in units of keV.

²In the early nuclear physics literature, all electrons were referred to as β particles, whether they came from β decay or internal conversion.

which agrees (within experimental uncertainty) with the 40 keV internal conversion transition energy. Similarly, the energy difference of α_0 and α_2 (332 keV) and that of α_1 and α_2 (291 keV) match the observed 327 and 287 keV internal conversion transitions. And so on through the remainder of the levels. The good agreement of the energy level differences reinforces the validity of the concept of discrete energy states in nuclei.

Yet another way to observe discrete energy levels of nuclei is through a nuclear reaction such as



in which a projectile x strikes a nucleus X , yielding a residual nucleus Y and an ejected particle y . The amount of energy available in the reaction is determined by the masses of the four particles. If the residual nucleus Y is formed in its lowest energy state, then the maximum amount of energy can be carried by the projectile y . If the residual nucleus Y is left in a higher energy state, then there is proportionately less energy given to the ejectile y . Thus by measuring the various energies of the emitted particles y , it is possible to determine the energy levels of Y (assuming no internal excitations of y can occur). This method for studying nuclear energy levels, which is common and routine today, was developed in the 1930s when accelerators for producing beams of projectiles x were newly available but limited in energy, and spectrometers for measuring the energies of y were less common. A summary of the state of research in this area is given by [Livingston and Bethe \(1937\)](#).

One of the earliest experiments of this type was reported by [Duncanson and Miller \(1934\)](#). They studied the reaction $\alpha + {}^{27}\text{Al} \rightarrow {}^{30}\text{Si} + \text{p}$, in which the α particles came not from an accelerator but from the radioactive decay of radium, and the outgoing protons were detected with a Geiger counter. They deduced energy levels in ${}^{30}\text{Si}$ of 2.23, 3.60, and 4.74 MeV and suggested the possible existence of γ rays of those energies. Similar attempts to deduce values of discrete energy levels were made in the earliest cyclotron experiments. For example, [McMillan and Lawrence \(1935\)](#) studied the reaction ${}^2\text{H} + {}^{27}\text{Al} \rightarrow {}^{28}\text{Al} + \text{p}$ using ${}^2\text{H}$ projectiles accelerated to 2.2 MeV and obtained values for several energy levels in ${}^{28}\text{Al}$.

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Origins of Nuclear Energy

Jim Napolitano, Temple University, Philadelphia, PA, United States

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Glossary

Beta decay The process by which one nucleus transforms into a nucleus with the same mass number but different atomic number.

Binding energy The difference in mass between a composite object and the sum total of its constituent objects, expressed in energy units.

Fission The breakup of a large nucleus into (generally) two smaller fragments.

Fusion The assembly of two lighter nuclei into one heavier nucleus.

Gamma emission The release of a photon in a transition between states in a nucleus.

Nucleon Collective name for a proton or neutron.

Introduction

A straightforward implication of quantum mechanics is that if matter is confined to a volume of a certain size, then the constituents of that matter must have internal momenta that are inversely proportional to the size scale. The electrons bound in an atom have energies on the order of several electron volts (eV). This is, therefore, roughly the size of the energy that can be released, per atom, in a chemical reaction. Examples would include the energy one gets by burning gasoline into carbon dioxide and water in an automobile engine, or released in a chemical explosion.

Each atom contains an atomic nucleus, at a distance scale roughly 100,000 times smaller than the size of the atom. The constituent particles of the nucleus are protons and neutrons, which we collectively refer to as nucleons. The much smaller distance scale implies much larger energies are released in nuclear reactions relative to chemical reactions, per “atom” of fuel.

This is the essential element of nuclear energy, the extremely high potential for energy release for a given amount of “fuel” as compared to conventional chemical reactions. Indeed immediately after the discovery of nuclear fission in 1939, its potential as an extremely powerful weapon was immediately recognized.

This chapter outlines the basic nuclear physics needed to understand the origin of nuclear energy. Numerical examples are given and essential feature are highlighted.

Equivalence of mass and energy

Energy is a profound concept in physics. One fundamental principle of Nature is that energy can neither be created nor destroyed. That is, energy is *conserved*. Another fundamental principle is that a system will always seek the state with the lowest possible energy. These two principles lead to the concept of *binding energy*, which provides a useful basis for discussing the release of energy in atomic and nuclear reactions.

If two particles attract each other, then we describe the attraction in terms of a *potential energy well*. (Indeed, the force itself is written as the negative of the gradient of the potential energy function.) Therefore, when two particles combine to form a more

complex object, the system finds itself in a lower potential energy state than for the two particles by themselves. If energy is to be conserved, however, then the energy difference between the initial and final states has to be somehow “released” in the combination reaction. This energy difference is called the *binding energy* of the system.

Of course, things can happen in the other direction. That is, a complex system made up of more elementary objects can be broken apart. In this case, an amount of energy equal to the binding energy needs to be introduced to the complex system.

This is a formal description of what happens when two particles interact with each other, but it begs the question regarding the physical origin of the binding energy. Students learn in most elementary classes that energy has different forms, for example, light and heat, and that these different forms can be converted from one to the other. But what is the form of energy that leads to binding of elementary particles?

The answer to this question lies in Einstein’s famous formula $E = mc^2$, which summarizes the equivalence between energy E and mass m , in terms of the speed of light c . A particle of mass m_1 binds to a particle of mass m_2 and forms an object with mass M where $M < m_1 + m_2$. The binding energy E_B is then $E_B = m_1c^2 + m_2c^2 - Mc^2$.

The binding energy for an electron and a singly ionized atom, also known as the *ionization potential*, is typically on the order of 10 electron volts (eV). The binding energy for nucleons in a nucleus is discussed below, but typically several millions of electron volts (MeV) for each proton or neutron in the nucleus. *This is the reason that nuclear energy is so much more powerful than the energy released in chemical reactions.* That is, the binding energy is roughly a million times larger in a nucleus than it is for electrons in atoms or molecules.

Protons and neutrons

Protons (p) and neutrons (n) are the particles that make up the nucleus. These two particles have very similar properties, and are often referred to collectively as *nucleons*. Their masses and interactions are fundamentally important to the binding energy of the nucleus and the source of nuclear energy.

One important similarity between protons and neutrons is that they have very nearly the same mass. Since we are discussing energies, we use the equivalence of mass and energy to express the masses of subatomic particles in energy units. For example, the mass of the proton is $938.27 \text{ MeV}/c^2$, and the mass of the neutron is $939.57 \text{ MeV}/c^2$. (The need for so many significant figures will become apparent shortly.) That is, the proton and neutron masses differ by only 0.13%. The electron mass, $0.51 \text{ MeV}/c^2$, is approximately 2000 times smaller than the mass of a nucleon.

Another important similarity between protons and neutrons is their intrinsic angular momentum, called *spin*. In both cases, the angular momentum has the nominal value $\hbar/2$, also referred to as *spin-1/2*. The origin of spin is purely quantum mechanical (Krane, 2021), and when particles are combined, the spins add as vector (i.e. signed) quantities. Thus, for example, two spin-1/2 particles can have a total spin of zero or one, and three spin-1/2 particles can have total spin 1/2 or 3/2.

An important difference between protons and neutrons is their electric charge. The proton has a positive charge $+e$, where e is the fundamental unit of charge. (That is, the charge of an electron is $-e$, so a neutral atom is made from a nucleus that has the same number of protons as the atom has electrons.) The neutron, on the other hand, has zero charge.

The hydrogen atom consists of an electron bound to a single proton. That is, the proton itself is the nucleus of the hydrogen atom. The binding energy of this atom is 13.6 eV, and is just due to the electrostatic attraction between the negatively charged electron and the positively charged proton.

The simplest complex nucleus, called the *deuteron*, consists of one proton bound to one neutron. (The atom composed of one electron bound to a deuteron is called *deuterium*.) The mass of the deuteron is $1875.61 \text{ MeV}/c^2$. The binding energy is then the difference between this and the sum of neutron and proton masses, that is

$$938.27 \text{ MeV}/c^2 + 939.57 \text{ MeV}/c^2 - 1875.61 \text{ MeV}/c^2 = 2.23 \text{ MeV}/c^2$$

This result is well known to nuclear physicists. If a low energy neutron encounters a free proton, it will be captured to form a deuteron and release a 2.2 MeV gamma ray.

The distributions of charge and matter in protons and neutrons are spread over some distance. The result is a nucleon that has “size,” described by a spherical distribution with a distance scale of roughly 0.5 fm. (One fm, called a “femtometer” or “fermi,” equals 10^{-15} m .) These distributions do not have a hard cutoff at any radial distance, however, so it would be misleading to refer to nucleons as “hard spheres.” Nevertheless, this distance scale is useful for understanding the properties of nuclei.

Descriptions and properties of the nucleus

Complex nuclei are essentially combinations of different numbers of protons and neutrons. The number of protons is denoted by Z , known as the “atomic number,” terminology borrowed from chemistry. Indeed, Z must also equal the number of electrons in the neutral atom. That is, the charge of a nucleus is simply $+Ze$. The number of neutrons is written as N , and the total number of neutrons plus protons is the “mass number” $A = Z + N$.

An *isotope* refers to a member of a nuclear species with the same atomic number but different mass number. That is, the same number of protons but a different number of neutrons.

Any nucleus is completely specified by the values of Z and A , with the value of N determined by arithmetic. Since Z also specifies the name of the element corresponding to the neutral atom, we use that name to identify the nucleus while using a left hand superscript to specify the mass number. So, for example,

$$^1\text{H}, ^2\text{H}, ^3\text{H}, ^3\text{He}, ^4\text{He}, ^{12}\text{C}, ^{14}\text{C}, ^{208}\text{Pb}, ^{235}\text{U}, ^{238}\text{U}$$

identify, respectively, the proton, deuteron, and triton (three isotopes of hydrogen), a rare isotope of helium, normal helium, the most abundant isotope of carbon, a radioactive isotope of carbon used to date human artifacts and other organic materials, the dominant isotope of lead, and two isotopes of uranium that are particularly relevant for nuclear energy.

The total angular momentum, or “spin,” of the nucleus depends on the particular orientation of the spins of the protons and neutrons as well as their motions within the nucleus. Nuclei have some number of *excited states* with energies higher than the ground state, and these excited states typically have different spins than the ground state.

Protons and neutrons have a finite size, and quantum mechanics forbids them to occupy the same space. Therefore, nuclei have a finite size that increases with mass number A . Quantifying the nuclear size depends on several details, including which distributions are being considered and the nuclear “shape” which may be other than spherical but a useful and simple formula is that the matter radius of a nucleus is given by

$$R = R_0 A^{1/3}$$

where $R_0 = 1.2$ fm. The factor $A^{1/3}$ is easy to understand, as the volume of a sphere packed with nucleons should be proportional to A .

Nuclear binding energy

The mass of a nucleus depends directly on its binding energy. When calculating the binding energy from the mass, it is important to recognize that tabulated atomic masses include the mass (and atomic binding energy) of the electrons that make up the neutral atom. In fact, atomic masses are generally reported in terms of the *atomic mass unit* $u = 931.494$ MeV/ c^2 , which is defined as exactly one-twelfth of the mass of the ^{12}C atom.

As an example, let us calculate the binding energy of the ^{12}C nucleus. To account for the mass of the electrons, it is convenient to use the mass of the hydrogen atom, 1.00784 u , instead of the proton. We have

$$\text{BE} = 6 \times (1.00784 \times 931.494) + 6 \times 939.57 - 12 \times 931.494 = 92.274 \text{ MeV}/c^2$$

(Note that this calculation ignores the binding energy of the electrons to the nucleus.) It is instructive to express this result as $\text{BE}/A = 7.69$ MeV per nucleon. That is, the binding energy of this nucleus is nearly as large as 1% of the nucleon mass.

This calculation can of course be carried out on any nucleus for which the mass is known. **Fig. 1** below plots the result, as binding energy per nucleon versus the mass number, for a large number of relatively abundant nuclei. (From https://commons.wikimedia.org/wiki/File:Binding_energy_curve_-_common_isotopes.svg.)

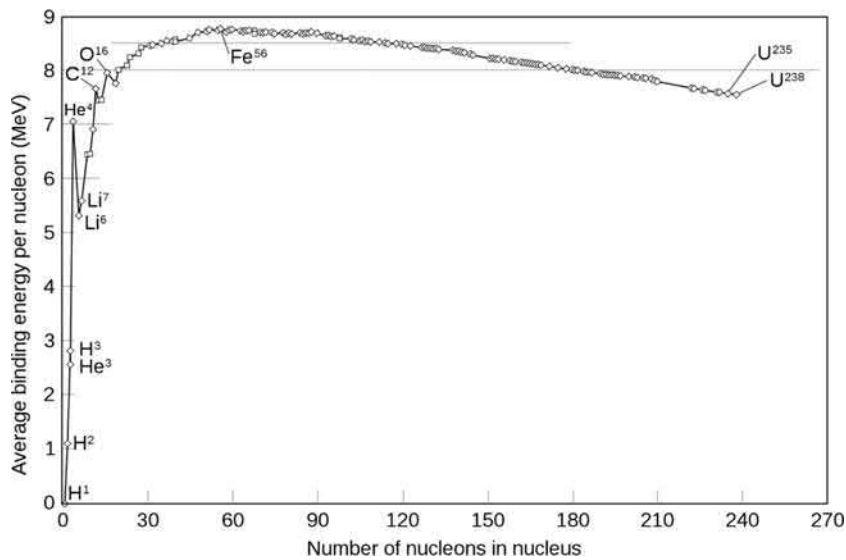


Fig. 1 Plot of the average binding energy per nucleon as a function of atomic mass A , averaged over the stable nuclei with that mass number. This generally smooth shape is well predicted by the Weizsäcker mass formula. The most stable nuclei are in the mass numbers near iron ($Z = 26$). There is a significant departure from the smooth curve for the ^4He nucleus, which is especially tightly bound.

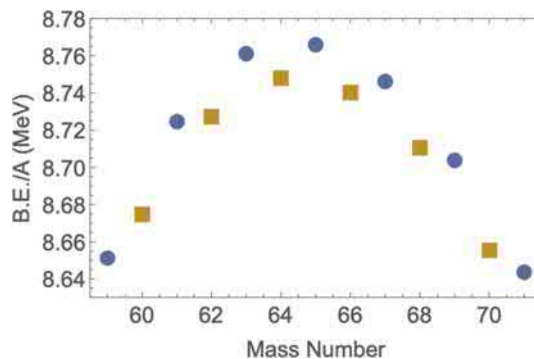


Fig. 2 The binding energy per nucleon for isotopes of copper ($Z = 29$), demonstrating the difference in pairing energy for odd or even numbers of neutrons.

A number of conclusions can be immediately drawn from this figure. Firstly, for the large majority of nuclei, the binding energy per nucleon is nearly constant, roughly equal to 8.5 MeV for nuclei heavier than oxygen. Secondly, the nuclei fall more or less on a smooth curve, rising rapidly for the light nuclei, reaching a maximum near ^{56}Fe , and then steadily decreasing through to uranium. This smooth shape is interrupted by a few particularly tightly bound nuclei, namely ^4He , ^{12}C , and ^{16}O .

These observations are key to understanding the fundamental sources of nuclear energy. Energy will be released if a large nucleus such as uranium breaks into smaller pieces that are larger than iron. This process is known as “nuclear fission.” Similarly, energy will be released if the light nuclei with $A = 1, 2$, and 3 combine to form ^4He , a process known as “nuclear fusion.” These processes are discussed below in more detail.

The smooth curve of the binding energy per nucleon versus mass number, is well described by the semi-empirical Weizsäcker mass formula. This formula expresses the binding energy E_B as a function of A , and is given by

$$E_B = a_V A - a_S A^{2/3} - a_C Z(Z-1)/A^{1/3} - a_A (A - 2Z)^2/A \pm \delta(Z, A)$$

These terms take into account, respectively, the overall binding of the A nucleons, a repulsive effect from “surface tension” of the nucleus, the electrostatic Coulomb repulsion of the protons, an “asymmetry” contribution from the “dilution” of coulomb repulsion by neutrons in heavy nuclei, and a quantum mechanical “pairing” energy that prefers neutrons and protons to come in pairs. The specific values of the coefficients a_i depend on the data set to which this formula is fit, but are given by (in MeV) to within a few percent as $a_V = 15.5$, $a_S = 16.8$, $a_C = 0.72$, and $a_A = 23$. The pair energy is reasonably approximated as $\delta(Z, A) = +a_P A^{-3/4}$ if Z and N are both even, $-a_P A^{-3/4}$ if Z and N are both odd, where $a_P = 34$ MeV, and zero if A is odd.

The “asymmetry” and “pairing” energies are nicely demonstrated in Fig. 2, which plots the binding energy per nucleon for isotopes of copper ($Z = 29$). The even isotopes (blue circles) and the odd isotopes (orange squares) are clearly separated from each other because of the pairing energy, but the overall trend for each is a parabolic dependence on the mass number, as expected from the Weizsäcker mass formula.

Nuclear interactions

The source of the binding energy of nucleons in a nucleus, called the *Strong Force*, acts only over very short distances, on the order of one femtometer (fm). It is therefore unfamiliar at the atomic scale, but nevertheless is responsible for overcoming the Coulomb repulsion of protons in the nucleus.

The exceptionally short range of the strong force is understood in the context of quantum field theory. In this framework, two particles interact with each other by virtue of an “exchange” particle. If the exchange particle is massive, then it is straightforward to show that the range of the interaction is inversely proportional to the mass. A range of 1 fm corresponds to an exchange particle mass of roughly $200 \text{ MeV}/c^2$, and this led to the discovery of the π -meson, or “pion,” with mass $135 \text{ MeV}/c^2$, as the exchange particle of the strong interaction.

Other components of the strong force are repulsive. These correspond to exchange particles other than the pion, which in fact are more massive and therefore of shorter range. The result is that nucleons are “packed” into a nuclear volume that is proportional to $A^{1/3}$. This is also responsible for the binding energy being proportional to the volume of the nucleus, to first approximation, as indicated in the Weizsäcker mass formula. The Coulomb force, however, is long range, so additional neutrons are needed to “dilute” the electrostatic repulsion and hold the nucleus together, as the atomic number increases.

During the process of nuclear fission, a large nucleus splits apart into two fragments. (This is discussed in more detail below.) The fragments are separated from each other by nuclear scale distances, but this is far enough so that the strong force becomes very weak, and Coulomb repulsion takes over. It is in this way that energy is released in nuclear fission.

A second interaction which only manifests itself at nuclear (and subnuclear) distances is the *Weak Interaction*. Mediated by exchange particles nearly one thousand times as massive as the pion, this interaction contributes only a minute amount to the

nuclear binding energy. However, in this case, the exchange particles allow the transformation of neutrons into protons, or vice versa, with the creation of other particles, including electrons, positrons (antimatter forms of electrons), and neutrinos.

Extracting nuclear energy: Fission, fusion, and radioactive decay

Now we have enough information to understand the dominant forms of energy release in nuclear reactions. These are “fission,” when a heavy nucleus splits into two fragments; “fusion,” when very light nuclei combine to make a heavier nucleus; and “beta decay,” when the weak interaction releases energy in the form of much lighter particles during transformations between neutrons and protons and “gamma decay” when a nucleus makes a transition from an excited state to a lower energy state. Indeed, radioactive decay itself provides a mechanism for extracting nuclear energy for practical applications.

Fission

Isotopes of iron ($Z = 26$) are particularly stable nuclei, with especially high binding energy per nucleon. Therefore, any nucleus with $Z > 52$ or so is less stable than the sum of the two nuclear fragments into which it would break up. In most cases, this does not happen spontaneously because the surface tension energy required to deform the large nucleus presents a barrier. However, in a number of special cases, the breakup of a large nucleus into two fragments, called “nuclear fission,” can be generated in controlled ways so that the resulting released energy is usefully harnessed.

Probably the most familiar example is the isotope ^{235}U , which provides the energy behind commercial nuclear power generators and many forms of nuclear weapons. A very low energy neutron is readily absorbed the ^{235}U which subsequently breaks into two large fragments. (In fact, typically a few neutrons are also released during fission process, and these can be used to propagate a chain reaction.)

A key aspect of ^{235}U is that it has an odd number of neutrons. The addition of even a very low energy neutron causes the nucleus to undergo fission, because of the binding attributed to the pairing energy. The nucleus ^{238}U also undergoes fission in the presence of neutrons, but only if the neutrons have high enough energy to overcome the fission barrier presented by the surface tension. In particular, ^{235}U will undergo fission by “thermal” neutrons, that is, neutrons in equilibrium with the thermal environment, namely with energies much less than 1 eV. Neutron induced fission of ^{238}U , on the other hand, requires neutrons with energies of more than 1 MeV.

Most natural uranium consists of the isotope ^{238}U . The natural isotopic abundance of ^{235}U is very small, less than 1%. Therefore, in many applications where fission is to be generated by thermal neutrons, for example nuclear reactors, natural uranium is “enriched” in ^{235}U to some level.

Let us investigate the energetics of ^{236}U (that is $^{235}\text{U} + n$) breakup and use this as an example of energy release in nuclear fission. Fission is notoriously asymmetric, so for the sake of illustration we assume ^{236}U splits into ^{133}Sn ($Z = 50$) and ^{103}Mo ($Z = 42$). The Weizsäcker mass formula tells us that the binding energies per nucleon for these three isotopes are 7.532 MeV, 8.224 MeV, and 8.542 MeV. (Compare these numbers to those plotted in the figure of binding energy per nucleon as a function of A .) The total energy released is

$$133 \times 8.224 + 103 \times 8.565 - 236 \times 7.538 = 196.5 \text{ MeV}$$

We have suggested that this energy appears in the Coulomb repulsion of the two fragments, so let’s test this idea. Using $R = R_0 A^{1/3}$ the radii of the two fragments ^{132}Sn and ^{103}Mo are 6.1 and 5.6 fm, respectively. Take these fragments to be two charged spheres that are just touching, so the centers are separated by $d = 11.7$ fm. The Coulomb energy just after fission is

$$(1/4\pi\epsilon_0) Z_1 Z_2 e^2/d = 258 \text{ MeV}$$

This is a crude calculation, involving several rough approximations, but the two calculations give results that are surprisingly close. In other words, the result of a fission of a heavy nucleus is that the fragments fly away from each other by virtue of their electrostatic repulsion. In other words, nearly all the energy released in fission appears in the kinetic energy of the fragments. The fragments slow down in the surrounding matter, imparting heat. This is the essence of how a nuclear power reactor generates steam to run turbines, for example.

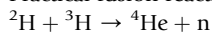
Fusion

Quantum mechanical effects in very light nuclei lead to peculiar results for nuclear binding energy, as shown in the figure of binding energy per nucleon. In particular, combinations of isotopes of hydrogen (^1H , ^2H , and ^3H) and ^3He into ^4He leads to a very large increase in the binding energy per nucleon, because ^4He is exceptionally tightly bound. Such processes are referred to as “nuclear fusion” and are in principle very copious sources of nuclear energy.

It is not obvious how to identify practical fusion reactions for generating nuclear power. Of the four lightest isotopes, the most abundant by far is simple hydrogen, that is ^1H . However, converting four ^1H into one ^4He requires turning two protons into neutrons. This can only happen through the weak interaction, so reaction rates will be very small. Indeed, this is how the Sun and other stars produce energy, but this is only possible because the temperatures and pressures in their cores are extraordinarily

large. Direct fusion of two ^2H into one ^4He is possible, through the reaction $^2\text{H} + ^2\text{H} \rightarrow ^4\text{He} + \gamma$. However in this case another interaction, the electromagnetic interaction, is required, which again is much weaker than the strong interaction.

Practical fusion reactions instead make use of “*dt* fusion,” that is, the fusion of a deuteron with a triton, in the reaction.



Although deuterium (^2H) is very rare (approximately 0.01% of the hydrogen on Earth), and tritium (^3H) is radioactive (with a half-life of only 12.3 years and therefore exists on Earth at impractical to use low concentrations), the energetics of this reaction are so favorable that the results are well worth the hurdles that need to be overcome.

The energy release in *dt* fusion is $M(^2\text{H}) + M(^3\text{H}) - M(^4\text{He}) - M(\text{n})$. Using the atomic (and neutron) masses, this is

$$(2.01410 + 3.01605 - 4.00260) \times 931.4941 - 939.565 = 17.6 \text{ MeV}$$

In other words, the energy release per unit of atomic mass is more than 3 MeV for fusion, whereas for fission it is closer to 1 MeV. That is, fusion is much more efficient energy source. Furthermore, it has the added benefit that the reaction products are considerably more benign than the radioactive daughter isotopes produced in fission.

The energy released in *dt* fusion is shared by the helium nucleus and neutron, with 4/5 of the energy (14 MeV) being carried off by the neutron. Harnessing this kinetic energy is one of the objectives of fusion reactor designers (Morse, 2021; Boccaccini, 2021). Another objective is converting the neutron to a triton via capture in ^6Li and (*n*,2*n*) reaction with ^7Li .

Radioactive decay

The radioactive decay of unstable nuclei is by itself a significant source of nuclear energy. Details of radioactivity are covered in Previtali (2021). One point made in particular, is that the heat of the Earth’s core is driven by the decay of radioactive isotopes. This has in fact been confirmed experimentally by detecting the neutrinos from the relevant beta decay processes.

^{238}U is one of the isotopes responsible for the heating at the core of the Earth. The radioactive decay chain of ^{238}U proceeds through eight alpha decays and six beta decays, ending up at ^{206}Pb . (Shusterman, 2021). From the mass difference of ^{238}U and ^{206}Pb and including the mass of the alpha particles, we find 48 MeV of energy released for each decaying ^{238}U nucleus. (In this calculation, we take only half the energy released in the beta decays, assuming that the other half is carried away by the neutrinos.)

Beta decay of the radioactive daughter nuclei in fission is also a significant source of energy in nuclear reactors. Heavier nuclei require relatively more neutrons for stability against Coulomb repulsion than do lighter nuclei, so the daughter nuclei are “neutron rich.” Therefore, they beta decay by turning neutrons into protons, releasing electrons in the process. Each daughter nucleus leads, typically, to three beta decays before turning into a stable nucleus, with each electron from the decay carrying roughly 1 MeV of energy. Thus the beta electron energy is roughly 6 MeV per fission, or about 2% of the energy released in the fission process itself. Beta decays also often produce nuclei in excited states that subsequently decay by emitting gamma rays.

Radioactive Thermal Generators (RTGs) make a practical use of nuclear energy released in radioactive decay, and one that is particularly useful when long term reliability is required in situations where maintenance is difficult or impossible, or where sunshine for solar power is lacking. In fact RTGs find many uses in space exploration (Previtali, 2021; Bennett, 2021). To be useful, an isotope must have a long enough half-life so that it spans the anticipated useful life of the application, but short enough so that the decay rate provides a reasonable power output per gram of material. The decay products are preferably dominated by alpha and beta emission, as neutrons and gamma rays are more difficult to stop, so would require more material surrounding the source to collect the energy.

The isotope ^{238}Pu is a generally attractive option for RTGs. It has a half-life of 87.7 years, and decays virtually 100% by alpha emission. Nearly 71% of the decays go to the ground state of ^{234}U , releasing a 5.5 MeV alpha particle, with nearly all remaining decays to the 43.4 keV excited state, which itself decays by gamma emission. We can confirm the energetics of ^{238}Pu decay by considering the difference of mass between this nucleus (238.0496u) and the daughter ^{234}U (234.0410u) plus an alpha particle. The result is 5.6 MeV, where 0.1 MeV is carried off by ^{234}U .

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Radioactive Decay

E. Previtali, Department of Physics, University of Milano-Bicocca, Milan, Italy

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Radioactive decays

Radioactivity is generated by spontaneous transitions of nuclei from their original states to other states to reach a more stable nuclear configuration. During this process, the excess energy is released by the emission of particles. Based on the emitted particle, we can classify the type of decay that drives the nucleus from the initial to the final state. From this point we can identify three main radioactive decay modes:

- Alpha decay
- Beta decay
- Gamma decay

Alpha decay

Alpha decay is directly related with the electromagnetic forces produced inside the nucleus by protons that create the condition for the emission of a ${}^4\text{He}$ nucleus: the so-called alpha particle. To represent such nuclear decay, we can write:

$$(AZ) \rightarrow (A - 4, Z - 2) + \alpha$$

In this transition the parent nucleus (A, Z) will release the α particle leaving the daughter nucleus with $N-2$ neutrons and $Z-2$ protons. This process will release energy equal to:

$$Q = [m(A, Z) - m(A - 4, Z - 2) - m_\alpha] c^2$$

where $m(A, Z)$ is the mass of the parent nucleus; $m(A-4, Z-2)$ is the mass of the daughter nucleus. m_α is the mass of the alpha particle, in other words the mass of the ${}^4\text{He}$ nucleus.

Applying this calculation to possible parents and daughter nucleus it is evident that alpha decays are possible only if $Q > 0$. Considering the average binding energy per nucleon, this relation clearly explains why alpha decays nuclei are concentrated in the high mass region. In practice only if the parent nucleus exhibits a binding energy per nucleon similar to that of the ${}^4\text{He}$ nucleus, could it be possible to have an alpha decay.

The Q value represents the total energy that will be released in the decay. For the alpha decay process in which there are two nuclei in the final state, the Q value will be distributed between the kinetic energies of the two products:

$$Q = T_\alpha + T_{(A-4, Z-2)}$$

Considering the parent nucleus to be at rest and using momentum and energy conservation, it is possible to evaluate the kinetic energy acquired by the alpha particle in the decay:

$$T = Q / \left[1 + \left(m_\alpha / m_{(A-4, Z-2)} \right) \right]$$

Considering the mass of the daughter nucleus, the alpha particle will acquire almost 98% of the Q value of the decay.

It is possible to evaluate the alpha decay probability starting from a simple model in which the alpha particle is trapped inside the Coulomb barrier generated by the daughter nucleus. The alpha particle can escape from the nucleus only because of the

quantum mechanical tunnelling effect. Using the Schrodinger equation for the penetration of a barrier, it is possible to evaluate the half life of an alpha decaying nucleus:

$$\frac{1}{T_{1/2}} \approx \frac{h}{m_\alpha R_n^2} \exp \left[-\frac{2}{h} \int_{R_n}^{R_n+d} \sqrt{2 m_\alpha (U(r) - E)} dr \right]$$

Where R_n is the radius of the nucleus, $U(r)$ is the barrier potential and E is the energy of the decay. It is important to observe that there is an exponential relationship between the half life and the energy of the alpha decay (Krane, 1987). This implies that a small variation in the Q value translates into a large change in the half life. In natural radioactive alpha decays, the Q values ranges from 2 MeV to 9 MeV but half life values range from fractions of microsecond to more than 10^{10} years.

Fig. 1 represents the alpha decay scheme of the ^{238}U nucleus. It is interesting to see how the transition probability indicated by the previous formula, strongly suppress the alpha decays to excited states of the daughter nucleus due to the reduction of the alpha decay energy E .

Beta decay

In the beta decay transition, there is no change in the atomic mass number of the nucleus. Depending on whether a neutron is transformed into a proton or vice versa, three different processes can take place:

$$\beta^- \quad (AZ) \rightarrow (AZ + 1) + e^- + \nu_e$$

$$\beta^+ \quad (AZ) \rightarrow (AZ - 1) + e^+ + \nu_e$$

$$EC \quad (AZ) + e^- \rightarrow (AZ - 1) + \nu_e$$

where the various symbols indicate: e^- the electron and e^+ the positron, $\nu_e(\bar{\nu}_e)$ the (anti)neutrino, and EC indicates atomic electron capture decay.

The neutrino was postulated as a new particle at the beginning of the 1930s by W. Pauli to justify the apparent non-conservation of energy produced by the emitted electron in beta decay (CERN, 2013). The name "neutrino" was introduced by Fermi (1934a,b) to distinguish such neutral particles from the other one discovered in the same period that was called the neutron. Neutrino interactions with matter have extremely low probability and the first observation of such interaction was made by Cowan Jr. et al. (1956).

Looking at the three indicated decay channels, the first one (β^-) transforms the parent to the daughter nucleus increasing the atomic number by one unit. On the contrary, the other two channels (β^+ , EC) both decrease the atomic number by one unit. To understand why we need two channels in which a proton can be transformed into a neutron inside the nucleus, we have to examine the energy conditions necessary for each of the three decay modes:

$$\beta^- \quad m(A, Z) > m(A, Z + 1) + m_e$$

where m is the nuclear mass and m_e is the mass of the electron (under the reasonable hypothesis that $m_\nu = 0$). Adding $(Z * m_e)$ on both sides of the equation, we obtain the equivalent condition using atomic masses (M):

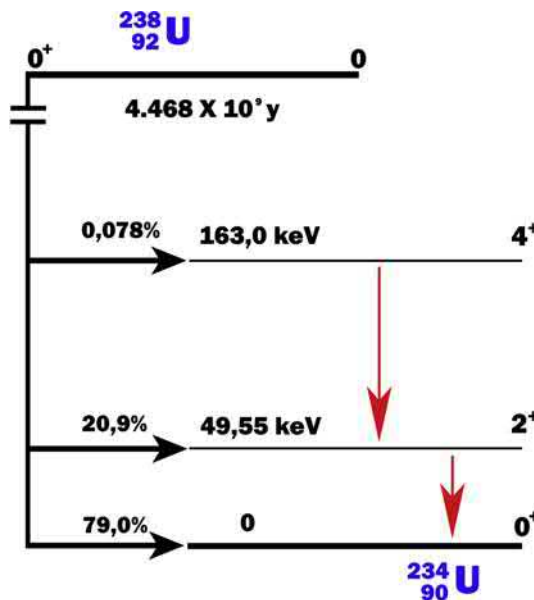


Fig. 1 Alpha decay scheme of the ^{238}U nucleus. The production of excited states in the daughter nucleus is strongly disfavored.

$$\beta^- \quad M(A, Z) > M(A, Z+1)$$

The result indicates that if the mass of the parent atom is larger than the mass of the daughter then β^- decay is energetically possible.

We can proceed in the same way for the other two decay channels.

$$\beta^+ \quad m(A, Z) > m(A, Z-1) + m_e$$

Adding $(Z * m_e)$ on both sides of the equation:

$$\beta^+ \quad M(A, Z) > M(A, Z-1) + 2*m_e$$

In the case of β^+ , the decay is energetically possible only if the mass of the parent atom is larger than the daughter one plus 2 electron masses. With this condition the β^+ decay will take place only if the energy mass difference between the two involved atoms is above a threshold of $2 * m_e = 1.022 \text{ MeV}$.

Now we can apply the same procedure to the process in which the nucleus captures an electron from the atomic shells:

$$EC \quad m(A, Z) + m_e > m(A, Z-1)$$

In this case we add $(Z-1) * m_e$ on the two sides of the equation:

$$EC \quad M(A, Z) > M(A, Z-1)$$

No threshold is present in this case. In the EC decay process, an atomic electron is captured by the nucleus. The EC decay probability depends on the overlap of the wave functions of the nucleus and the electron. Increasing the Z of the atom causes this overlap to increase which produces larger EC transition probabilities for heavy nuclei. This implies that mainly the electrons close to the nucleus will be captured. As a result, the binding energy of the atomic shell has to be considered in the total decay energy balance.

In β^- and β^+ decays there are three particles in the final state: the daughter nucleus, the electron (positron) and the (anti) neutrino. The energy released by the decay is shared mainly by the two emitted particles. Nuclear recoil can be considered as less important, but all the particles participate in sharing of the linear momentum. The energy spectrum of the electron is continuous starting from zero up to the Q value of the beta transition and (neglecting the nuclear recoil energy) obviously we have that:

$$Q = T_e + T_\nu$$

The shape of the energy spectrum is strongly related to the decay transition type, and depending on the characteristics of the initial and the final states we can produce a detailed classification of the beta decays (Konopinski, 1943).

Considering only the allowed beta decay transitions, we can write the electron energy spectrum in a relatively simple way as:

$$\frac{dN}{dE} = N_0 \frac{16\pi^2}{c^4} D \sqrt{E(E + 2m_e c^2)} (E - m_e c^2) (E_m - E)^2 F(Z, E)$$

where D is a proportionality factor, E is the electron energy expressed in electron mass units, E_m is the maximum energy acquired by the electron, and $F(Z, E)$ is the so-called Fermi factor that takes into account the electromagnetic correction to the decay spectrum.

The main difference between the β^- and β^+ spectra can be observed in the low energy part of the electron (positron) energy spectrum where the different electric charges of the electron and the positron play an important role.

We can now evaluate the half-life starting from the energy distribution of the electrons and integrating the previous formula:

$$\frac{1}{T_{1/2}} = \frac{1}{\ln 2} \frac{16\pi^2}{c^4} D \int_0^{E_m} \sqrt{E(E + 2m_e c^2)} (E - m_e c^2) (E_m - E)^2 F(Z, E) dE$$

Without entering into the details of the integral solution that requires a precise description of the Fermi factor, we consider only the case in which $E_m \gg m_e c^2$ obtaining an energy dependence for the half life:

$$\frac{1}{T_{1/2}} \propto E_m^5$$

We immediately observe that the decay probability for beta decays depends less strongly on the transition energy as compared with alpha decay described in the previous paragraph. With this dependence, possible transitions to excited states of the daughter nucleus are less suppressed than those for alpha decay. Considering the emitted particles in the final state, it is clear that the electron and neutrino spins contribute to a maximum change in the nuclear angular momenta of one unit. Such decays are identified as allowed transitions. All the other changes in nuclear angular momenta imply different angular momentum for the electron or neutrino in the final states and are normally labeled as forbidden transitions (Konopinski, 1943).

In Fig. 2, the beta transition scheme of ^{60}Co is illustrated. Due to the large variation of the angular momentum between the ground state of the parent nucleus and the ground state of the daughter one, beta transition to excited states are strongly favored with subsequent emission of gamma ray photons.

We now have to raise an important question: for β^- and β^+ decays we can observe the emitted charged particles as indicated in the transition schemes. What is the observable for the EC decay channel? It is important to remember that in this transition an

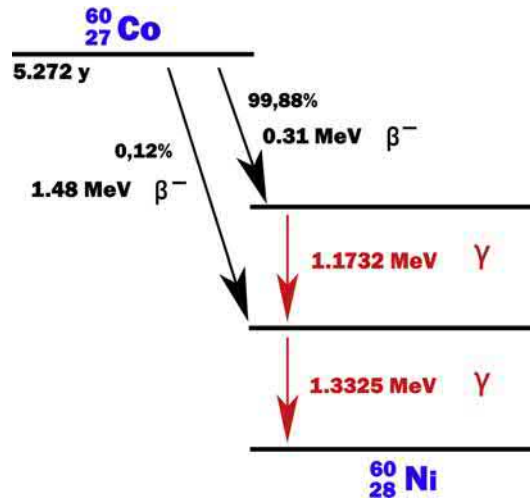


Fig. 2 Beta decay scheme of the ^{60}Co nucleus. The large difference in angular momentum between the ground state of the parent nucleus and that of the ground state of the daughter strongly favors the population of excited states and subsequent emission of gamma ray photons.

atomic electron is captured by the nucleus. As a result, there will be a hole in one of the shells of the daughter atom that will be filled by means of electron rearrangements. As a consequence, characteristic X-rays will be emitted from the daughter atom.

Gamma decay

A nucleus in an excited state will undergo gamma decay to reach a more stable nuclear configuration. Normally a photon with an energy equal to the energy difference between the initial and final state is emitted: this photon is usually identified as gamma ray.

$$(A, Z)^* \rightarrow (A, Z) + \gamma$$

The * indicates a nucleus in an excited state. In this case the energy balance is relatively simple (again neglecting nuclear recoil):

$$E_\gamma = E_n^i - E_n^f$$

where i and f indicate the initial and final states of the nucleus.

In what way can a nucleus reach an excited state? Normally it is the result of a previous alpha or beta decay or it is due to a nuclear reaction. Due to the decay probability, alpha decay normally generates the daughter nucleus in the ground state and if an excited state is produced it is usually of relatively low energy. In the case of beta decay, the conservation rules for angular momentum frequently generates a daughter nucleus in excited states that normally produces gamma transitions to reach the final ground state.

The typical energies of gamma ray photons are on the order of an MeV. For lower energy transitions another important process occurs: the conversion electron transition. In such case the energy of the excited nucleus is transferred directly to an atomic electron that will be emitted with an energy equal to the difference between the nuclear excited state and the atomic electron binding energy.

$$E_e = E_n^i - E_n^f - B(Z)_e$$

where B indicates the atomic electron binding energy. The conversion electron process is strictly connected with the overlap of the nuclear and atomic wave functions.

The timescale for gamma decays is normally very short. This is an electromagnetic transition, and the process will take place on times typically of the order of few psec (10^{-12} s). This is the reason why normally we associate the emitted gamma decay with the name of the beta or alpha parent nucleus, even though the gamma emission is produced by the daughter.

There are some cases in which the daughter nucleus remains in an excited state for a long time. In such cases the states are called metastable, and the nuclei that have such states are indicated as metastable nuclei. In many cases these nuclei will remain in excited states for times longer than months or years. In such cases it is obviously not possible to directly correlate the emitted gamma ray with the previous nuclear transition.

Source of Earth's internal heating

Our Solar System was born approximately 5 billion years ago from material expelled by previous generations of stars that produced all the elements that we now find on Earth. Among these elements, we find many radioactive isotopes still present in the Earth. These isotopes generate the so-called fossil radioactivity that is the most important part of the natural radioactivity on Earth. In order

to be considered as a fossil radioactivity element, the half lifetimes of such unstable nuclei have to be on the order of the age of the Solar System. We can also classify such radioactive elements in two main groups: radioactive isotopes that undergo a transition to stable nuclei or that are parents of a relatively long radioactive chain.

Radioactive chains

The radioactive chains are mainly produced by heavy nuclei. We normally identify four chains due to the dominant alpha decays that change the nuclear masses by four units. More specifically the four chains are:

$$\begin{aligned} A = 4n \quad {}^{232}_{90}\text{Th} \quad T_{1/2} &= 1.4 \cdot 10^{10} \text{ years} \\ A = 4n + 1 \quad {}^{237}_{93}\text{Np} \quad T_{1/2} &= 2.2 \cdot 10^6 \text{ years} \\ A = 4n + 2 \quad {}^{238}_{92}\text{U} \quad T_{1/2} &= 4.6 \cdot 10^9 \text{ years} \\ A = 4n + 3 \quad {}^{235}_{92}\text{U} \quad T_{1/2} &= 7.8 \cdot 10^8 \text{ years} \end{aligned}$$

Looking at the half lives of the four parent nuclei, it is clear that after 5 Gy, ${}^{237}\text{Np}$ has completely disappeared; ${}^{235}\text{U}$ is strongly reduced with respect to its original abundance; ${}^{238}\text{U}$ is now one half its initial value, and ${}^{232}\text{Th}$ is almost unchanged (Nucleardata, 1999). So at present, there are two main chains that are important for natural radioactivity.

In general, the parent nuclei will undergo a transition to a daughter that is also radioactive, and the daughter of the daughter (granddaughter) is also radioactive. In this way we can identify around 10 nuclei that directly derive from the progenitors: this generates the radioactive chains. The full chains of ${}^{232}\text{Th}$, ${}^{238}\text{U}$ and ${}^{235}\text{U}$ are represented in Figs. 3–5 respectively. Normally the last element of the chain is a stable lead isotope (Nucleardata, 1999).

The radioactive chains are dominated by the alpha decay with some beta minus transitions that are necessary in order to counterbalance the increase of the neutron/proton ratio generated by the alpha decays. The chain will reach secular equilibrium (all the nuclei involved in the chain show the same activity) after about five half lives of the daughter nucleus with the longest half life. This means that in order to reach secular equilibrium, the radioactive chains must remain isolated for a time related with the half lives of the daughter nuclei. If we consider the ${}^{232}\text{Th}$ chain, we see that the daughter nucleus with the longest half life is ${}^{228}\text{Ra}$, with $T_{1/2} \sim 5$ years (Nucleardata, 1999). Secular equilibrium will be reached in the relatively short time of around 25 years. On the contrary, the longest half life daughter of the ${}^{238}\text{U}$ chain is ${}^{234}\text{U}$ with $T_{1/2} \sim 2.5 \cdot 10^5$ years (Nucleardata, 1999). This means that at least a million years is needed to put the chain in the secular equilibrium condition.

Within the radioactive chains there are some elements that easily break the chains. Typically, Radium is an alkaline metal and under the action of a polar solvent (e.g. water) it can be released from rock and in such a way will break the chain into two parts. The same situation happens for Radon, which is a noble gas and does not establish chemical bounds with the materials in which it is produced. Consequently, Radon can be released by rocks and diffuse into the atmosphere or water generating a break in the radioactive chains. In all these cases, the mechanisms for radioactive chains breaks are strongly connected with the characteristics of the rocks and also with the half lives of the radioactive isotopes of Radium and Radon.

Long living radioactive isotopes

Apart from the radioactive chains described in the previous paragraph, there are also some other fossil radioactive isotopes present on Earth. These isotopes have a half life of the same order of the age of the Earth.

The most important isotope of this type is ${}^{40}\text{K}$ that has a half life of $T_{1/2} \sim 1.3 \cdot 10^9$ years. Its isotopic abundance is relatively low, 0.0118% (Nucleardata, 1999), but the concentration of Potassium on Earth is relatively high and that implies that the contribution to the natural radioactivity from this isotope is very important. Just to have an idea, inside each of our bodies there are approximately 4000 decays per second of ${}^{40}\text{K}$. This isotope decays by beta minus, $BR \sim 90\%$, and by electron capture $BR \sim 10\%$.

Another fossil isotope is ${}^{87}\text{Rb}$ that has a half life of $T_{1/2} \sim 4.8 \cdot 10^{10}$ years and an isotopic abundance of 27.8% (Nucleardata, 1999). ${}^{87}\text{Rb}$ also decays via a beta transition but, considering that the concentration of Rubidium on Earth is not so large, the overall contribution of this isotope to the Earth's radioactivity is relatively low.

An interesting case relates to ${}^{209}\text{Bi}$: $T_{1/2} \sim 2 \cdot 10^{19}$ years and isotopic abundance 100% (Nucleardata, 1999). This is the last remaining radioactive isotope of the ${}^{237}\text{Np}$ chain that, due to its relatively short half life has practically disappeared on Earth. The very long half lifetime of ${}^{209}\text{Bi}$ implies that its global contribution to the Earth radioactivity is extremely low and it could be considered, under many points of view, as a stable nucleus.

The radioactivity heat of the Earth

Looking to the natural radioactivity, it is reasonable to imagine that the energy released by the radioactive decays could contribute to the geothermal heat of the Earth. The question is how important is the contribution that comes from radioactivity with respect to what we observe from the thermal point of view.

In order to answer this question, there have been measurements that attempt to quantify the total number of radioactive decays per second that occur in the Earth's crust and mantle. Considering the main radioactive isotopes that could contribute to this energy

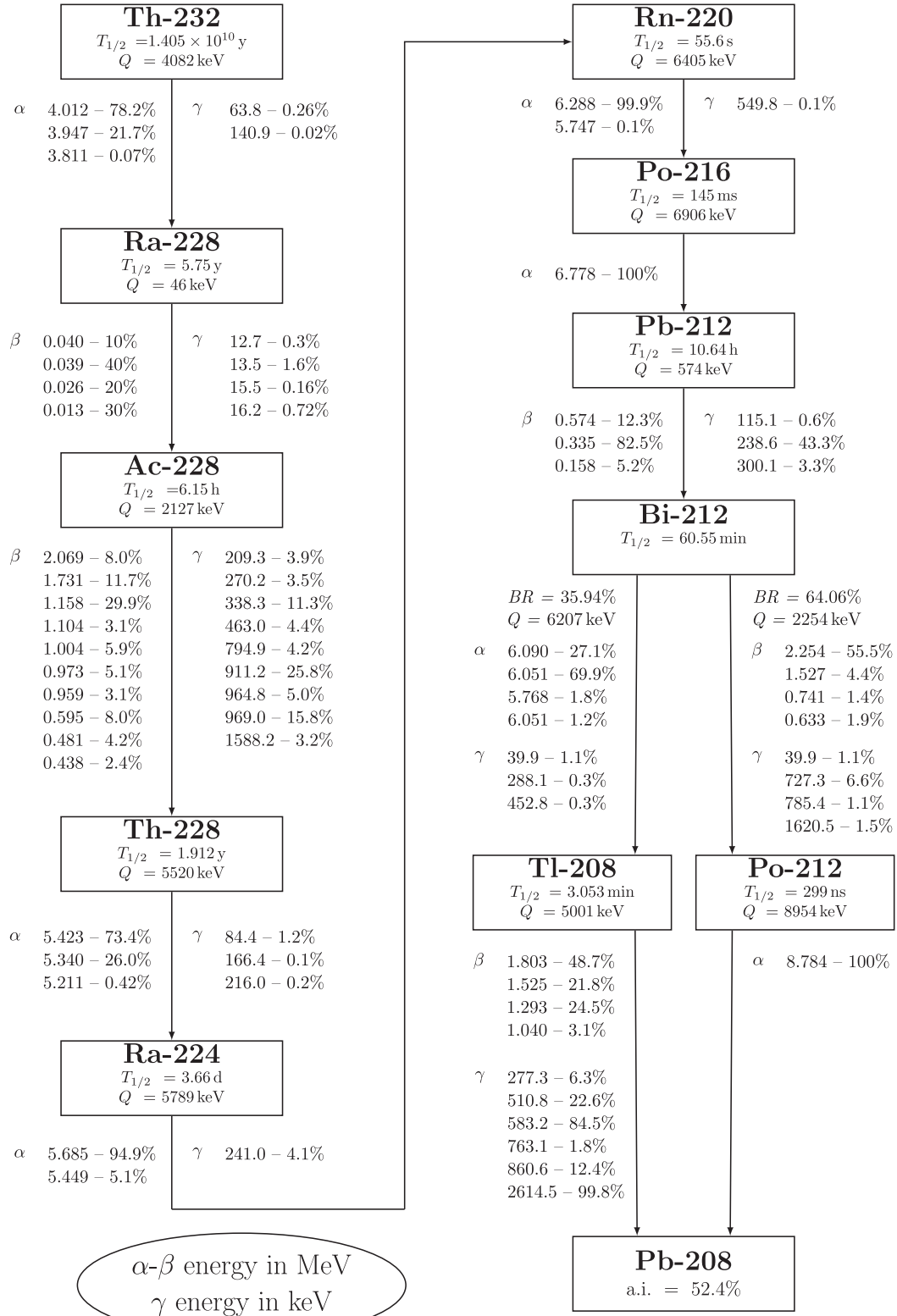


Fig. 3 ^{232}Th radioactive chain with the most important emitted alpha, beta and gamma radiations. Reported data from <http://nucleardata.nuclear.lu.se/toi/>

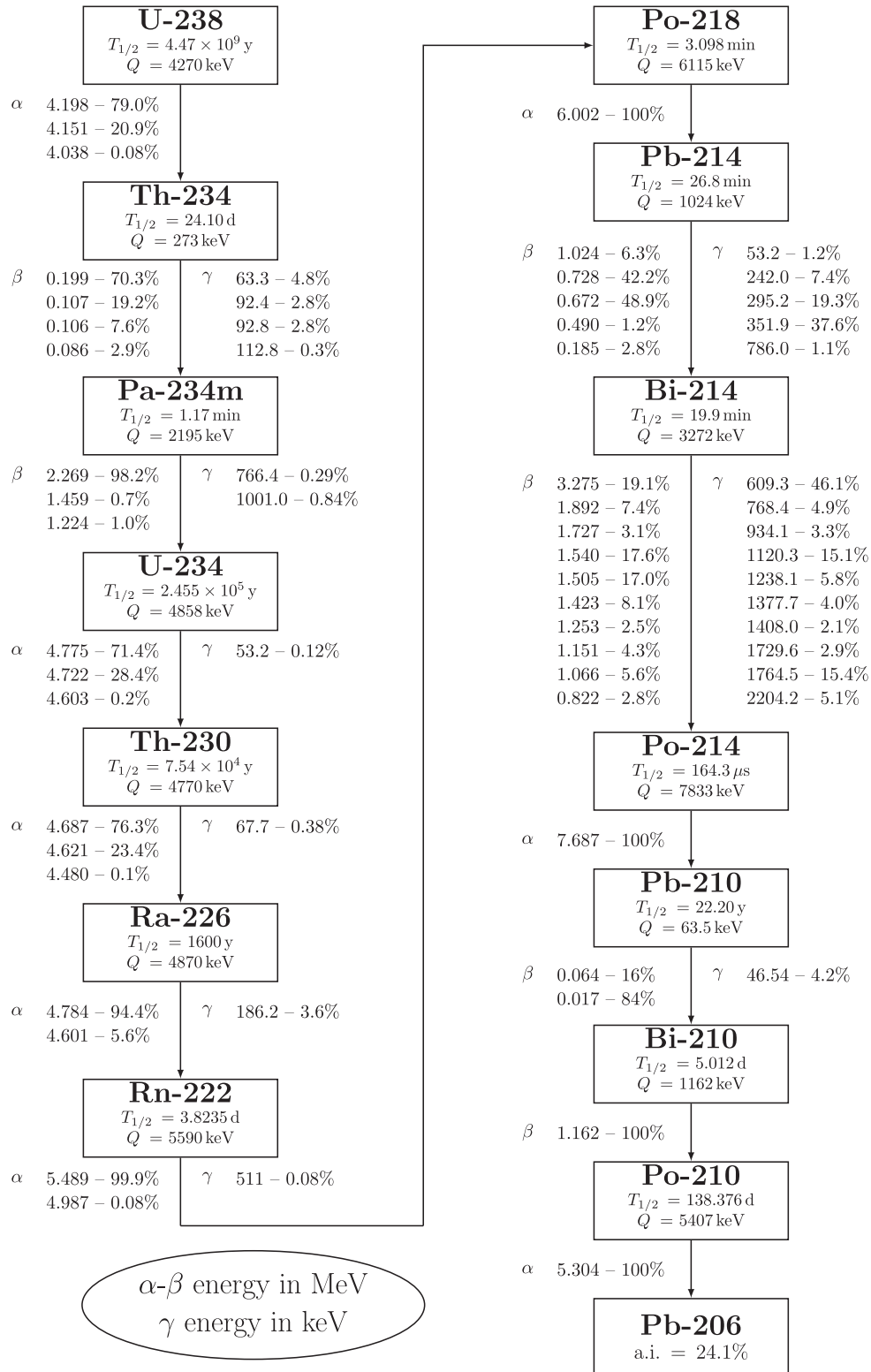


Fig. 4 ^{238}U radioactive chain with the most important emitted alpha, beta and gamma radiations. Reported data from <http://nucleardata.nuclear.lu.se/toi/>

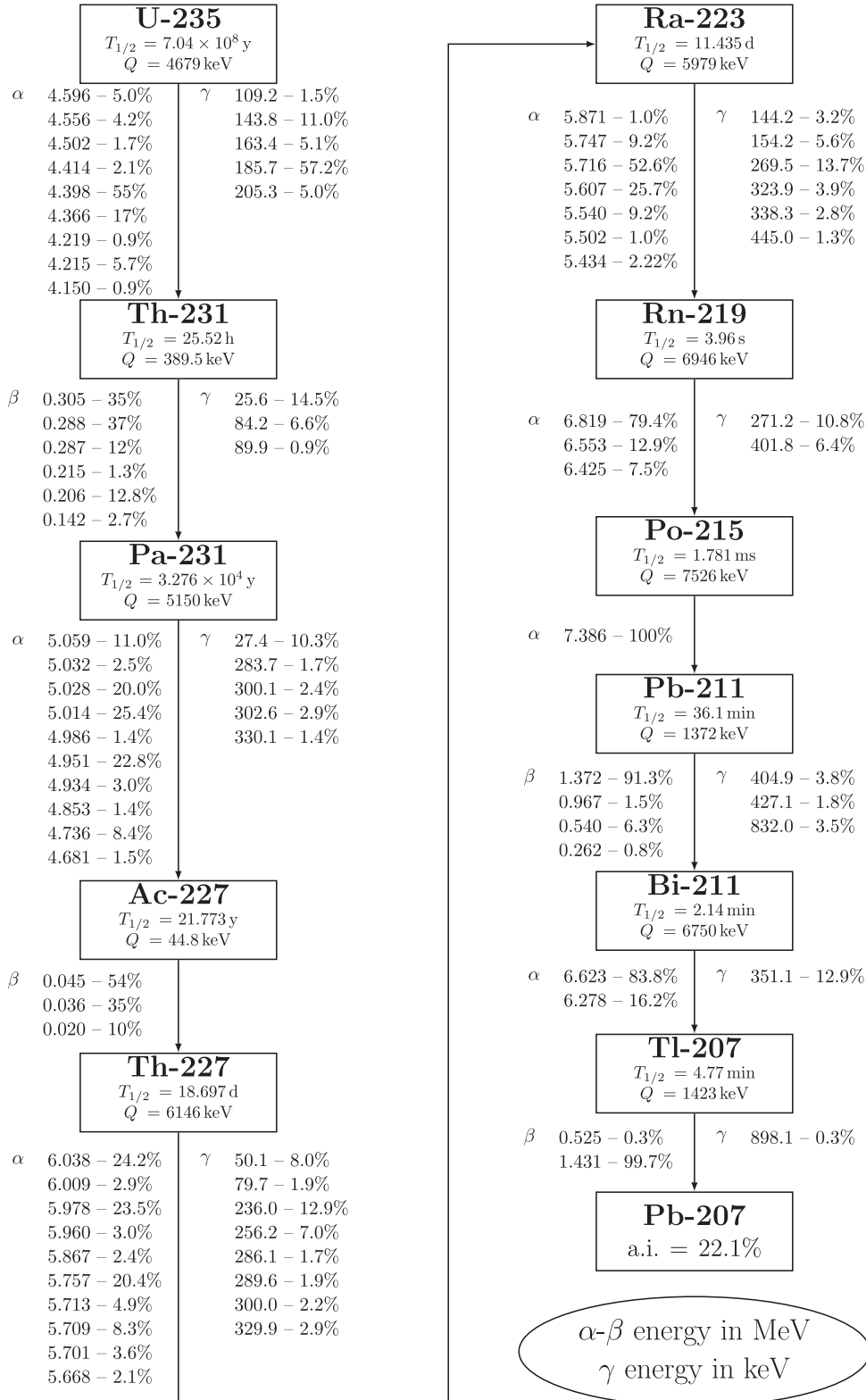
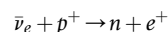


Fig. 5 ^{235}U radioactive chain with the most important emitted alpha, beta and gamma radiations. Reported data from <http://nucleardata.nuclear.lu.se/toi/>

release, we can identify ^{232}Th , ^{238}U and ^{40}K . All these isotopes are part of the Earth's ores and release their decay energy within the Earth's volume. To quantify the Earth's radioactivity, dedicated measurements of anti-neutrino fluxes from the Earth (geo-neutrinos) were done. Anti-neutrinos are produced by beta decays of the previously mentioned radioactive sources and their observation provides a direct quantification of the global radioactivity of the Earth. Geo-antineutrinos travel over a long distance inside the Earth's mantle and crust thanks to their very small interaction cross sections. To detect them, experiments typically use the inverse beta decay reaction:



Only geo-antineutrinos with an energy above 1.8 MeV can induce this reaction, so these experiments can only detect decays from the ^{238}U and ^{232}Th chains and not ^{40}K . These measurements are sensitive to mantle radioactivity that is not accessible with direct sampling or with other radioactivity measurements. Important results were obtained by Borexino and Kamland experiments that clearly identified the anti-neutrino signal that comes from the internal part of the Earth and determine the radiogenic heat from ^{238}U and ^{232}Th chains in the mantle of the order of 24 TW of power (Agostini et al., 2020; Gando et al., 2011).

In parallel with the anti-neutrino measurements, evaluations of the radioactivity of rocks and sediments in the Earth's crust were performed in order to identify the mean concentration of radioactive sources. These measurements determine the radioactivity from all the radioactive chains and for ^{40}K . Combining these results with those from the geo-antineutrino experiments, the total radiogenic heat for the Earth was determined to be equal to 38 TW of power, with an error of the order of 30% (Agostini et al., 2020). The results obtained with these measurements are compatible with a relatively high concentration of radioactive elements inside the Earth. If we compare these experimental results with the estimated total heat of the Earth to be around 47 TW (Davies and Davies, 2010) of power, we conclude that radioactive decay is the main source of geothermal heat.

Radioactive energy source for remote applications

Radioactive decay is an exothermic process and decay transitions release energy to the final decay products. The result is a net flux of energy that can generate thermal power. Selecting a suitable decay source, a radioactive energy generator can be used in many different applications.

The big advantage of this approach is connected to the large amount of energy produced by radioactive decays as compared to chemical reactions. By selecting long-lived isotopes, energy can be produced for long periods of time. Another important feature is that radioactive decay is not affected by environmental conditions. High vacuum conditions in space, high temperatures or high pressures don't affect the radioactive decay. Moreover, the radioactive source will remain in operation without any maintenance.

On the other hand, radioactive generators have low power density and low electrical conversion efficiency. Also, the application of such power generators is normally considered only for applications in which there are difficulties to actively manage the power sources. Examples include space stations and remote environmental instruments, or in situations in which possible recharging/refueling of the power source is difficult, or for implanted pacemakers for heartbeat stabilization.

The first aspect that has to be considered is the total radioactive power needed to guaranty a reasonable energy production. Just as an example, to generate 1 mW of power starting from a radioactive source that releases 1 MeV of energy per decay, a total activity of around 10^{10} Bq (0.3 Ci) is needed and, taking also into account a reasonable energy conversion efficiency, an increase of around a factor 10 in the total source activity is needed. This implies that few grams of ^{226}Ra , for example, are necessary (Nucleardata, 1999). This relatively large amount of radioactive material has to be taken into account because of potential safety issues for users and because it also has an impact on the costs of radioactive energy generators.

Alpha decay transitions will guaranty a large energy release per decay and safe operation of radioactive batteries (Knoll, 2010). On the contrary, beta decays are often connected with gamma emissions and, even in case of pure beta transitions, photons from bremsstrahlung of electrons can be emitted by the radioactive sources (Knoll, 2010). To increase the power density generated by the radioactive sources, nuclei with not too long half lives have to be selected. The most widely radioactive source used for these applications is ^{238}Pu . It has an alpha decay with a total energy of 5.5 MeV, a half life of 87 years and low probability for gamma ray and neutron emissions (Nucleardata, 1999). The main issue of such source is related with its relatively high production costs. However, with nuclear waste reprocessing it will be possible to generate strong radioactive ^{90}Sr sources at relatively low cost. This transition is a pure beta decay with practically no gamma ray emissions (Nucleardata, 1999), but it needs substantial radiation shields to stop the high energy electrons and their generated bremsstrahlung photons.

Radioisotope thermal generators (RTG)

In the most straightforward approach, to convert the radioactive decay energy into electric power the emitted particles interact with the source materials. These interactions produce a net increase of the material temperature that can be converted in electricity via various thermodynamic approaches: thermocouples, Peltier cells, or Stirling Engines.

Depending on the selected conversion technique, thermodynamic efficiency can vary substantially. A standard RTG made with thermocouples or Peltier cells has an energy conversion efficiency below 10%. However, applying the radioactive source to a Stirling motor generator a thermodynamic efficiency well above 20% was demonstrated

One of the main uses of RTGs is connected with space applications (NTRS, 2018). NASA has implemented RTGs since the Apollo missions in the 1960s. Long lasting space missions are directly supported by the long living radioactive isotopes for onsite electrical power production. The installation of an RTG on the Curiosity Rover on Mars will guaranty a long running time operation of the installed instrumentation.

Direct and indirect conversion batteries

A different strategy is related to the conversion of the emitted radiation directly into electric power. This approach derives from the possibility to generate a current in a semiconductor due to the characteristics of the p-n junction. There are two possible methods to have current production from radioactive sources: the direct way, in which emitted particles generate the current in the junction, and the indirect one, in which emitted particles produce secondary effects that generate the current.

A typical example of the direct approach is the “betavoltaic effect” (Olsen, 1973). In this case a beta particle emitted by the radioactive source interacts with the depleted region of a p-n semiconductor junction. The ionized charges, electron-hole pairs, produced in such regions will be responsible for the generated current across the junction. In this way the two-step process will be avoided but at the same time an optimization of the beta source vs. semiconductor junction configuration is needed. In fact, only the energy released in the depleted region of the p-n junction will be converted into a current. The energy released in an electric contact or in the semiconductor inert materials will not produce charges that will take part in the electric generation. Moreover the ionized electron-hole pairs can undergo recombination that will strongly reduce the conversion efficiency. This effect could be around 30% of the global interaction efficiency and are normally unavoidable (Knoll, 2010). So, at the end the conversion efficiency can be of the order of 10% but normally only values around few percent are really achievable.

The indirect method relies on the fact that the energy released by particles emitted in the decays will produce photons inside luminescent materials. The number of generated photons is directly connected with the source activity and to the photon conversion efficiency of the luminescent material. If we look at typical luminescent plastic materials, normally 10,000 photons per MeV are produced as the consequence of the interaction of a beta particle (Knoll, 2010). Produced photons are typically in the ultraviolet/visible range of the electromagnetic spectrum and so they can be converted in a current thanks to photovoltaic cells. The conversion efficiency is not very large due to the two steps process: conversion of radioactive emissions to light and then conversion of generated light to current. It needs a careful optimization of the luminescent material in order to match the spectral response of the photovoltaic cell. Using this approach, an overall efficiency of few percent has been demonstrated, and further optimizations are possible.

Some interesting developments have recently been achieved using the charge produced by emitted particles, alphas or betas, to directly produce current. The emitted charged particles generate a voltage difference between the two “plates” of a capacitor, each one isolated from the other and from the environment, and a current will be produced when the two plates are shunted (Li et al., 2002). This very simple approach is interesting if we consider the use of a modern MEMS (Micro Electro Mechanical System) in which the charges produced by the radioactive sources will generates an oscillation of the MEMS structures and the generation of electrical power.

See Also: Natural and Man-Made Sources of Radiation.

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Nuclear Fission I

Jennifer L. Klay, California Polytechnic State University, San Luis Obispo, CA, United States

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Glossary

Actinide The 15 chemical elements between actinium ($Z = 89$) and lawrencium ($Z = 103$) that include the naturally occurring radioactively unstable elements thorium ($Z = 90$) and uranium ($Z = 92$) as well as the artificially produced transuranics, such as plutonium ($Z = 94$) and californium ($Z = 98$), several of which can decay through spontaneous nuclear fission.

Fertile material A non-fissile material that can be converted to fissile material by neutron irradiation.

Fissile material Material that is fissionable by thermal neutrons.

Fissionable element Any element that can undergo fission, either spontaneously or by reaction with neutrons, photons, or charged particles.

MeV Million electron volts; 1 eV (eV) is a unit of energy that is equivalent to 1.6022×10^{-19} J.

Nuclide Species of atomic nucleus, characterized by the atomic number, Z and neutron number N .

Thermal neutrons Thermal neutrons have energies in the 0–0.5 eV range, corresponding to the equivalent Maxwell-Boltzmann kinetic energy for an ideal gas at the ambient (moderator) temperature.

Transuranic element Elements of the periodic table with atomic number larger than uranium. All are unstable.

History and introduction

The startling discovery of nuclear fission that ushered in the modern atomic age was the culmination of more than a decade's worth of focused investigation by a large community of chemists and physicists eager to understand the secrets of the atom. After the discovery of the neutron in 1932 and the development of the idea that nuclei are composed of collections of protons and neutrons held together against the long-range electrostatic repulsion of the positively charged protons by a strong, short-range nuclear force, many experiments and theoretical descriptions were developed to study and explain the structure of nuclei.

Enrico Fermi in Rome, Lise Meitner and her collaborators, Otto Hahn and Fritz Strassmann, in Berlin, among others, sought to produce new "transuranic" elements (heavier than uranium) through neutron bombardment. Meitner, Hahn, and Strassmann's results, using radiochemical techniques for particle identification, suggested they had found new elements in their experiments that were chemically similar to other known elements but the identities of these products were difficult to pinpoint (Meitner et al., 1937). When Meitner fled Germany to escape persecution by the Nazi regime, Hahn and Strassmann continued their investigations and stumbled upon the startling fact that one of the elements produced in their experiments was the relatively light element barium (Hahn and Strassmann, 1939). They shared their conclusions with Meitner before publication and she had the occasion to discuss them with her nephew, Otto Frisch, a physicist working on similar experiments at Niels Bohr's institute for theoretical physics in Copenhagen. Meitner proposed the idea that the liquid drop model of the nucleus could help explain their results and she and Frisch described the basic parameters of the idea shortly thereafter (Meitner and Frisch, 1939).

Fission comes from the Latin *fissio* "to split" and was first used to characterize biological cell division in the nineteenth century. Meitner and Frisch selected it as the perfect descriptor for the process by which a nucleus breaks up into two fragments of more or less equal mass. They worked out estimates for the energy available to these products from the difference in mass of the incoming and outgoing nuclei and the electrostatic repulsion of the products and discovered that the energy release was significant. The implications of this discovery were immediately apparent. Upon hearing the core idea of their manuscript from Frisch, Bohr is said to have exclaimed "Oh, what idiots we have been that we haven't seen that before. Of course this is exactly as it must be." Bohr was on his way to the United States to meet with collaborators and attend conferences and was able to work out the initial theoretical details of the fission process that he and collaborator John Wheeler refined into their seminal paper, The Mechanism of Nuclear Fission (Bohr and Wheeler, 1939).

Once it was known that fission could not only produce significant quantities of energy but also emitted additional neutrons that could themselves initiate fission and, possibly, establish a fission chain reaction, the race to harness this energy source was on. The start of the second world war both inhibited research and refocused it on applications that would support the war effort. Most notably, the application of nuclear fission for use as a weapon became the focus of the Manhattan Project in the United States (Cameron, 2021) that eventually led to the first deployment of nuclear weapons in Hiroshima and Nagasaki, Japan, in late 1945. The application of nuclear fission to energy production is discussed elsewhere in this volume. In the next sections, details of the fission process and its products are described.

The process of nuclear fission: Excitation and breakup

Heavy nuclei have a large positive electric charge due to their large atomic number, Z . The heaviest naturally occurring element, uranium, has an atomic number of $Z = 92$. This produces a large, long-range electric field that inhibits the close interaction of slow-moving nuclei and other positively charged particles. The neutron, on the other hand, having no net electric charge, is not repelled and has a high probability to interact with nuclei if they are brought together. Reactions between neutrons and nuclei can be set up experimentally, with energetic consequences. The left panel of Fig. 1 shows a schematic of the capture and excitation of the isotope ^{235}U by a bombarding low-energy neutron. The compound nucleus ^{236}U formed from this capture is in an excited state that allows it to decay by fissioning into two smaller mass fragments and one or more neutrons, accompanied by a significant release of energy. The product nuclei are themselves unstable and will decay through emissions of beta particles (electrons), gamma rays, and anti-neutrinos that carry away some of the energy, but the bulk of the energy released during this process goes into the kinetic energy of the fission fragments that are accelerated away from one another by their mutual electrostatic repulsion.

The nuclear breakup process can be understood as a consequence of the charged liquid-drop model of nuclear structure. It was Meitner who first intuited that this model, applied to an excited nucleus with severe spatial deformation, could cause the nucleus to break into two smaller charged drops because of their strong electrostatic repulsion (Meitner and Frisch, 1939). The right panel of Fig. 1 shows a schematic of the stages of fission from nearly spherical ground state to deformed excited states of various degrees of deformation, leading to the narrow neck between the two lobes that produces a strong electrostatic repulsion which drives the division of the nucleus into two smaller fragments.

The energy imparted to the fragments comes from the conversion of electrostatic potential energy to kinetic energy as the fragments separate. Fig. 2 shows a schematic of the total energy of the fragments at different stages of nuclear deformation adapted from Hodgson et al. (1997). In the ground state, the fragments are bound inside the nucleus with an energy corresponding to the sum of the masses of their constituents minus the binding energy that holds them together. The overall height of the barrier, E_b , represents the electric potential energy from the mutual repulsion of their constituent protons and is a measure of the energy available for release in the fission reaction. If the nucleus becomes excited, for example by absorbing a neutron and deforming as a result, the fragments may obtain enough energy, E_{ex} , to “climb over” the barrier and separate completely, releasing the barrier energy in the fission process.

For the example reaction illustrated in Fig. 1, the difference in mass between the bombarding neutron and its target ^{235}U and the compound nucleus of ^{236}U formed when the neutron is absorbed corresponds to an energy of 6.5 MeV. This slightly exceeds the excitation energy needed to overcome the barrier, allowing the fragments to separate through the fission process.

The masses of nuclei can be calculated in energy units using the semi-empirical mass formula (Napolitano, 2020), and the binding energy per nucleon is obtained by solving for the non-mass terms. As in Fig. 1 of Napolitano (2020), the binding energy per nucleon increases with mass number up to the elements near iron, where it has a value close to 8.5 MeV, then decreases with mass number. The decrease in binding energy for elements larger than iron means that energy is released when very heavy nuclei break apart. For fission reactions such as the example illustrated in Fig. 1, the compound nucleus of ^{236}U breaks up into two lighter

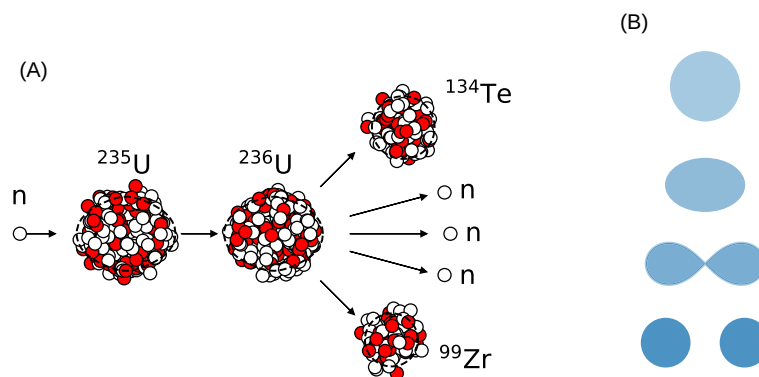


Fig. 1 (Left) A neutron bombards a nucleus of ^{235}U , which absorbs it and becomes excited. The excited nucleus of ^{236}U is unstable and splits into two lower mass nuclear fragments plus a few neutrons and gamma rays. (Right) Deformation of an excited nucleus during the stages of fission.

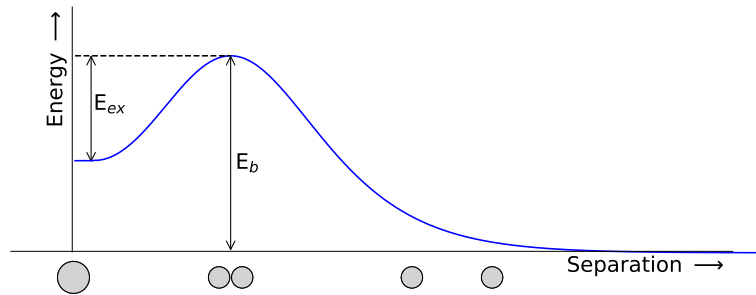


Fig. 2 Schematic of the potential barrier for a heavy nucleus as a function of the separation between two constituent fragment nuclei. Based on Figure 18.18 (page 401) from Hodgson PE, Gadioli E and Gadioli Erba E (1997) *Introductory Nuclear Physics*. Oxford University Press: New York.

mass fragments and the difference in mass between the initial and final states can be calculated directly or estimated from the binding energy curve by comparing the binding energy per nucleon for ^{235}U with that of the average fragment mass ($A = 118$). The result is approximately 200 MeV of energy that is liberated by this process, an extraordinary amount.

The products of nuclear fission

When a nucleus fissions, it does not typically fragment into equal mass products, although that is a possible outcome. More commonly the fragments are asymmetric, as seen in Fig. 3. The curves represent evaluated average yields of all the known nuclear fission products of ^{235}U , organized by mass number and tabulated for three different bombarding neutron energies: 0.025 eV (thermal), 500 keV (slow or epithermal) and 14 MeV (fast) by Chadwick et al. (2011). Thermal neutrons are so-called because those energies correspond to the equivalent Maxwell-Boltzmann kinetic energy for an ideal gas at the ambient temperature. The dashed lines indicate the masses of the fragments from the example illustrated in Fig. 1. These fission product nuclei are radioactively unstable and will further evolve through beta decay and gamma emission or delayed neutron evaporation to reach their most stable configurations.

In addition to the charged fragments of large mass that are emitted in the fission process, a small number of neutrons are also liberated through evaporation near the surface of the fissioning nucleus (prompt neutrons) and from the unstable fragments after they are released (delayed neutrons). These prompt and delayed neutrons carry away only a small fraction of the total energy but play a significant role in making nuclear fission a viable source of energy because they have the capacity to initiate further fissions within a sample of material. Fig. 4 shows the average number of neutrons emitted per fission, $\bar{\nu}$, for ^{233}U , ^{235}U , and ^{239}Pu as a function of neutron energy (Chadwick et al., 2011). Although the neutron yield grows with neutron energy above 1 MeV, the averages shown are typical for thermal and slow neutron-induced fission reactions.

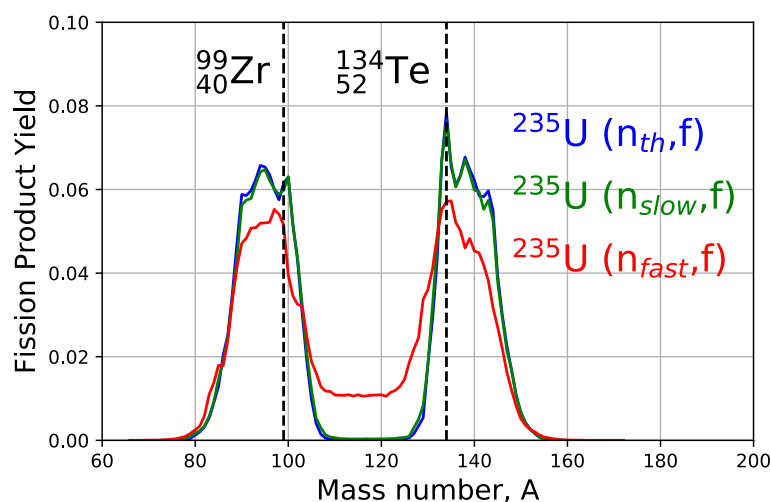


Fig. 3 Mass distribution of fragments emitted from neutron-induced fission of ^{235}U as a function of mass number for three different incident neutron energies: thermal (0.025 eV), slow (500 keV) and fast (14 MeV). The curves represent evaluated average yields tabulated by Chadwick et al. (2011). Data from Chadwick MB, Herman M, Obložinský P et al. (2011) *ENDF/B-VII.1: Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data*. Nuclear Data Sheets 112, 2887. National Nuclear Data Center, Brookhaven National Laboratory. Retrieved from: <https://www.nndc.bnl.gov/>.

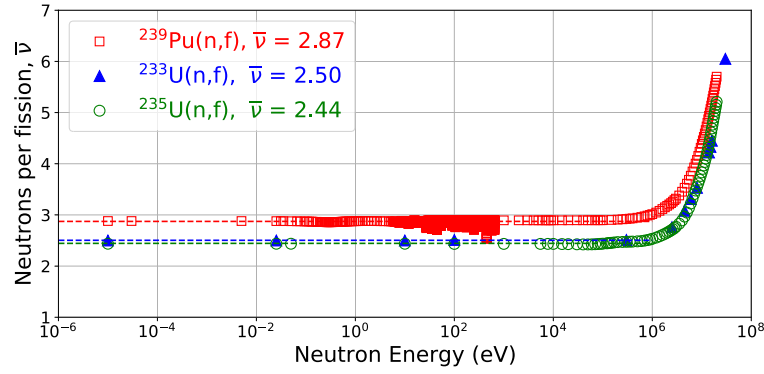


Fig. 4 Average number of neutrons per fission as a function of neutron energy for all available measured data tabulated by Chadwick et al. (2011). Data from Chadwick MB, Herman M, Obložinský P et al. (2011) *ENDF/B-VII.1: Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data*. Nuclear Data Sheets 112, 2887. National Nuclear Data Center, Brookhaven National Laboratory. Retrieved from: <https://www.nndc.bnl.gov/>.

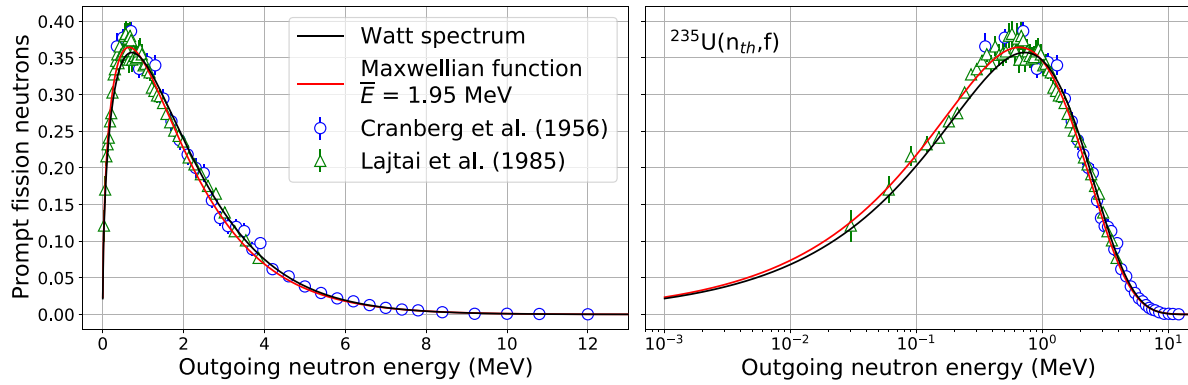


Fig. 5 Energy spectrum of outgoing neutrons from fission of ^{235}U induced by thermal neutrons on both linear (left) and logarithmic (right) energy scales. The data from Lajtai A, Kecskemeti J, Safar J, Dyachenko PP and Piksaikin VM (1985) Energy spectrum measurements of neutrons for energies 30 keV–4 MeV from thermal fission of main fuel elements. In: *Conference on Nuclear Data for Basic and Applied Science, Santa Fe, USA*, vol.1, p. 613.; Cranberg L, Frye G, Nereson N and Rosen L (1956) Fission neutron spectrum of ^{235}U . *Physical Review* 103: 662 [Retrieved from National Nuclear Data Center, Brookhaven National Laboratory, <https://www.nndc.bnl.gov/>] are compared to a Maxwellian function with a mean value of 1.95 MeV and the empirical Watt spectrum, Eq. (1) Curve from Watt BE (1952) Energy Spectrum of Neutrons from Thermal Fission of U235. *Physical Review* 87(6): 1037–1041.

The energy spectrum of the prompt neutrons from thermal neutron-induced fission of ^{235}U is shown in Fig. 5. The left panel shows the data on a linear energy scale while the right panel is logarithmic in energy. The data from Lajtai et al. (1985) and Cranberg et al. (1956) are compared to a Maxwellian function with mean energy of 1.95 MeV and the empirical Watt spectrum (Watt, 1952):

$$P(E) = 0.4865 \sinh(\sqrt{2E}) e^{-E} \quad (1)$$

In addition to the prompt neutrons emitted within $\sim 10^{-14}$ s of the fission process, the fission products themselves are highly unstable, neutron-rich nuclei that undergo further radioactive decays. This chain of reactions produces additional β -delayed neutrons that are capable of inducing further fission reactions. The time-scales for emission of these neutrons are driven by the β -decay chain half-lives and are typically grouped accordingly. Delayed neutrons play an important role in reactor design, discussed elsewhere in this volume. Table 1 summarizes the relative and total yield of delayed neutrons from neutron induced fission of ^{235}U , ^{238}U , and ^{239}Pu grouped by half-life (Koning et al., 2006).

While neutron absorption is the catalyst for fission, neutrons carry away only a small fraction of the total energy released during fission and constitute a small component of the overall momentum balance because of their small mass relative to the fragments.

The total energy released during fission, the so-called “Q-value” of the reaction, can be calculated from the difference in mass between the incoming and outgoing neutrons and nuclei, converted to energy units.

$$Q = \Delta mc^2 = [(m_U + m_n) - (m_{Te} + m_{Zr} + 3m_n)]c^2 \quad (2)$$

The right-hand side of Eq. (2) corresponds to the example illustrated in Fig. 1 and has a resulting value near 200 MeV that is shared among all of the products of the fission reaction, including those that arise from the radioactive decay of the unstable fragments.

Table 1 Delayed neutron yield by group half-life for neutron induced fission of ^{235}U , ^{238}U , and ^{239}Pu .

Group #	Group half-life (s)	Fractional delayed neutron group yield (ν_d/ν_t)		
		^{235}U (thermal)	^{238}U (fast)	^{239}Pu (thermal)
1	55.6	2.18×10^{-4}	1.39×10^{-4}	7.2×10^{-5}
2	24.5	1.023×10^{-3}	1.716×10^{-3}	5.33×10^{-4}
3	16.3	6.05×10^{-4}	6.19×10^{-4}	1.859×10^{-4}
4	5.21	1.31×10^{-3}	2.26×10^{-3}	4.1×10^{-4}
5	2.37	2.2×10^{-3}	4.85×10^{-3}	6.62×10^{-4}
6	1.04	6.0×10^{-4}	3.27×10^{-3}	1.836×10^{-4}
7	0.424	5.40×10^{-4}	2.11×10^{-3}	1.62×10^{-4}
8	0.195	1.52×10^{-4}	1.536×10^{-3}	4.16×10^{-5}
Total fractional yield (ν_d/ν_t)		6.65×10^{-3}	1.65×10^{-2}	2.25×10^{-3}
Total ν_d (per fission)		0.0162	0.0465	0.0065

Delayed neutron yields by group half-life for neutron induced fission of ^{235}U , ^{238}U , and ^{239}Pu .

Data from Koning A, Forrest R, Kellett M, et al. (2006) *The JEFF-3.1 Nuclear Data Library*, JEFF Report 21, OECD/NEA, Paris, France, 2006, ISBN 92-64-02314-3. Retrieved from the International Atomic Energy Agency Handbook of Nuclear Data for Safeguards 2008, Vienna, Austria. <https://www-nds.iaea.org/sgnuccat/>.

Table 2 Distribution of energy among fission products.

Quantity	^{233}U (MeV)	^{235}U (MeV)	^{239}Pu (MeV)
Kinetic energy of fragments	168.22 ± 0.50	169.13 ± 0.49	175.55 ± 0.40
Kinetic energy of prompt neutrons	4.90 ± 0.10	4.84 ± 0.07	6.13 ± 0.10
Kinetic energy of delayed neutrons	0.0310 ± 0.0047	0.0074 ± 0.0011	0.0030 ± 0.0004
Kinetic energy of prompt gammas	7.72 ± 0.52	6.60 ± 0.50	6.74 ± 0.47
Kinetic energy of delayed gammas	5.01 ± 0.06	6.33 ± 0.05	5.17 ± 0.06
Energy of delayed betas	5.16 ± 0.06	6.50 ± 0.05	5.31 ± 0.06
Energy of neutrinos	6.93 ± 0.09	8.75 ± 0.07	7.14 ± 0.09
Total energy released per fission, Q	197.97 ± 0.21	202.16 ± 0.13	206.04 ± 1.18
Total energy less neutrino energy	191.04 ± 0.23	193.41 ± 0.15	198.90 ± 1.09

Components of energy released in fission for the actinides ^{233}U , ^{235}U , and ^{239}Pu .

Data from Chadwick MB, Herman M, Obložinský P et al. (2011) *ENDF/B-VII.1: Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data*. Nuclear Data Sheets 112, 2887. National Nuclear Data Center, Brookhaven National Laboratory. [Retrieved from: <https://www.nndc.bnl.gov/>].

Table 2 shows the distribution of energy among the various fission products for ^{233}U , ^{235}U , and ^{239}Pu tabulated by [Chadwick et al. \(2011\)](#), with approximately 85% going into the kinetic energy of the fragments themselves, while nearly 3% is carried by the neutrons, 6% by gammas, and the remaining 12% is distributed among the electron and neutrino products of the fragment beta decays. Note that the energy imparted to the neutrinos is not available because of their extremely low interaction probabilities. That waste energy exits a reactor never to be recovered.

When the fission chain reaction in a nuclear reactor is halted, the rate of fission drops but there remains significant residual thermal power (heat) produced from delayed radioactive decays of the fission products that continues for a significant amount of time. **Table 2** illustrates that as much as 13% of the energy from each fission contributes to this delayed heat output. The precise power curve after shutdown depends on the mixture of isotopes present and their respective half-lives. **Fig. 6** shows a typical decay heat power curve from the fission of ^{235}U (thermal), ^{238}U (fast), and ^{239}Pu (thermal) over time after reactor shutdown tabulated by [Shibata et al. \(2011\)](#). The integrated heat output from the residual decays is significant and is an important consideration in reactor design.

Fission probabilities

As shown in **Fig. 5**, the prompt neutrons emitted during fission have average energies in the fast region of neutron energy classification (> 1 MeV), where the probability of capture by and fission of another nucleus of ^{235}U is highly suppressed relative to that for thermal neutrons. The probability of neutron-induced fission is typically reported in terms of a “cross-section,” measured in barns (10^{-24} cm²), that characterizes the effective areal overlap of a neutron projectile impinging on a nuclear target. **Fig. 7** shows the evaluated average cross-sections for neutron-induced fission of ^{235}U and ^{238}U as a function of neutron energy for all available measured data tabulated by [Chadwick et al. \(2011\)](#). The thermal and fast neutron energy regions are identified by the dashed vertical lines. The region between these two boundaries exhibits significant structure, corresponding to resonance absorption of

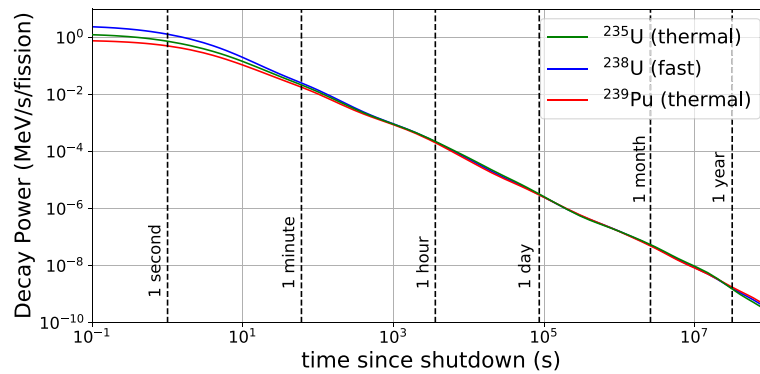


Fig. 6 Decay heat power from neutron induced fission of ^{235}U (thermal), ^{238}U (fast) and ^{239}Pu (thermal). Data from Shibata K, Iwamoto O, Nakagawa T et al. (2011) JENDL-4.0: A new library for nuclear science and engineering. *Journal of Nuclear Science and Technology* 48(1): –30. [Retrieved from the Japan Atomic Energy Agency, Nuclear Data Center 2020, Tokai-mura, Japan. <https://www.ndc.jaea.go.jp/>]

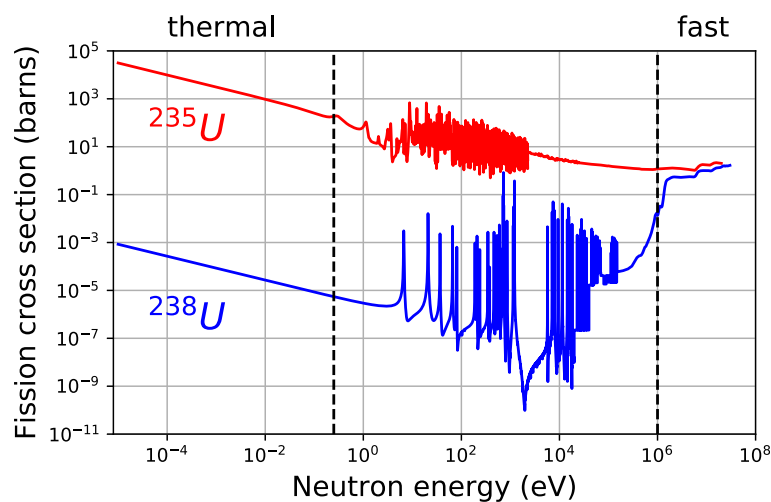


Fig. 7 Cross sections for neutron-induced fission of ^{235}U and ^{238}U as a function of neutron energy for all available measured data tabulated by Chadwick et al. (2011). Data from Chadwick MB, Herman M, Obložinský P et al. (2011) *ENDF/B-VII.1: Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data*. Nuclear Data Sheets 112, 2887. National Nuclear Data Center, Brookhaven National Laboratory. [Retrieved from: <https://www.nndc.bnl.gov/>].

neutrons into specific excited quantum states of the nucleus that are present at higher density for these energies. A comparison of Figs. 5 and 7 illustrates that the prompt neutrons from fission are not in the region where the fission cross-section is largest.

Both of the isotopes shown in Fig. 7 are present in naturally occurring uranium but at significantly different abundances, with ^{238}U at 99.2745% and ^{235}U at 0.72%. Notice that the cross section for thermal neutron induced fission of ^{238}U is seven orders of magnitude below that of ^{235}U and only becomes comparable in the fast region. This makes it a non-fissile material and may partly explain why fission was such a challenging discovery to make. Since the thermal neutrons and natural uranium used in the early experiments had such a low probability for fission and the idea that a nucleus could break up in this way was far outside the accepted understanding of nuclear physics at the time, it took imagination and insight to realize the obvious (Pearson, 2015). As noted by Nobel Laureate Emilio Segrè, “This is a tricky setup. Above all, it seems to me that the human mind sees only what it expects” (Segrè, 1989).

Spontaneous fission

In addition to fission induced by bombarding particles such as neutrons, photons, protons, and other charged particles, there are many actinides (thorium and beyond) that can undergo spontaneous fission. This process is a consequence of nuclear shell structure and the inherent deformation of neutron-rich heavy nuclei in their ground state. The result is a double-humped structure in the fission barrier of Fig. 2 that increases the chances that the fragments can quantum mechanically tunnel through the barrier and fission. One can predict the mean lifetime for spontaneous fission from the pairing term in the semi-empirical mass formula (Napolitano, 2020). The condition for spontaneous fission to be favorable occurs when the ratio of $Z^2/A > 47$, and the larger the value of this ratio,

the shorter the lifetime. Most nuclides that have a high branching ratio for spontaneous fission fall below this value ($Z^2/A = 38.1$ for ^{252}Cf). The discrepancy is due to the fact that nuclear shell effects, which are significant for the highest mass nuclei, are not well-described by the liquid drop model so the dependence on Z^2/A is only approximate (Krane, 1987).

Fig. 8 shows the high Z , N region of the chart of the nuclides, with the nuclides color-coded according to their largest nuclear decay branching ratio. Green boxes indicate those nuclides whose dominant decay mode is spontaneous fission. The vertical dashed lines correspond to the elements uranium and californium and the horizontal lines at constant neutron number show the known nuclides for these elements. While the dominant mode for ^{252}Cf (at the intersection of $Z = 98$ and $N = 154$) is via alpha decay, it has an appreciable branching ratio for spontaneous fission ($\sim 3\%$), making it a useful source of neutrons. Table 3 lists spontaneous fission characteristics of selected nuclides, including their neutron yields from Chadwick et al. (2011), Holden and Hoffman (2000), and Plompen et al. (2020).

The nuclides ^{233}U , ^{235}U , and ^{239}Pu are considered most useful for nuclear energy production because they are all fissile, have small spontaneous fission branching ratios, are relatively long lived, and have high thermal neutron cross sections. Although ^{233}U is not naturally present in uranium ore and ^{239}Pu is not naturally occurring, both can be created via neutron capture of the fertile materials ^{232}Th and ^{238}U , respectively, and thorium is three times more abundant than uranium.

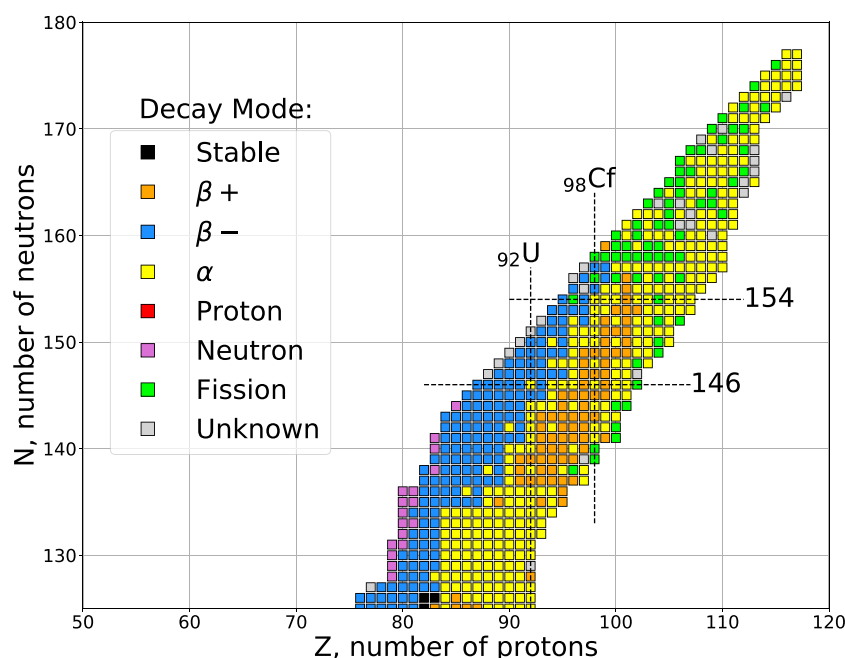


Fig. 8 Chart of the neutron-rich heavy nuclides, color-coded according to their dominant decay mode. Those nuclides whose dominant decay mode is spontaneous fission are shown in green. Data from ENSDF database as of April 4, 2016. Version available at <http://www.nndc.bnl.gov/ensarchivals/>.

Table 3 Spontaneous fission characteristics for selected nuclides.

Nuclide	Half-life (years)	SF branching ratio (%)	SF half-life ^a (years)	Total neutron yield (per SF)	Calculated total neutron yield ($\text{g}^{-1} \text{s}^{-1}$)
^{232}Th	1.40×10^{10}	1.1×10^{-9}	1.2×10^{21}	1.50	7.07×10^{-8}
^{235}U	7.04×10^8	7.0×10^{-9}	1.0×10^{19}	1.87	1.04×10^{-5}
^{238}U	4.468×10^9	5.4×10^{-5}	8.2×10^{15}	2.00	1.35×10^{-2}
^{239}Pu	24,110	3×10^{-10}	8.0×10^{15}	2.32	1.59×10^{-2}
^{242}Cm	0.446	6.2×10^{-6}	7.0×10^6	2.528	1.96×10^7
^{244}Cm	18.1	1.4×10^{-4}	1.32×10^7	2.6875	1.10×10^7
^{250}Cm	8305	74	1.13×10^4	3.30 ^b	1.53×10^{10}
^{252}Cf	2.645	3.09	86.0	3.7676	2.28×10^{12}

Half-life, branching ratio, and neutron yields from spontaneous fission for a selection of nuclides.

Data from Chadwick et al. (2011) except for "a" from Holden and Hoffman (2000) and "b" from Plompen et al. (2020).

Data from Holden NE and Hoffman DC (2000) Spontaneous fission half-lives for ground-state nuclides (Technical Report). *Pure and Applied Chemistry* 72(8): 1525–1562; Chadwick MB, Herman M, Obložinský P et al. (2011) *ENDF/B-VII.1: Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data*. Nuclear Data Sheets 112, 2887; Plompen AJM, Cabellos O, De Saint Jean C et al. (2020) The joint evaluated fission and fusion nuclear data library, JEFF-3.3. *European Physical Journal A* 56: 181 retrieved from the National Nuclear Data Center, Brookhaven National Laboratory, USA. [<https://www.nndc.bnl.gov/>].

Conclusion

Nuclear fission is a spectacular marvel of nature that allows us to harness the stored rest-mass energy of the heavy elements. The applications of this phenomenon to both weapons and energy production have been significant developments and drivers of human innovation, both destructive and beneficial, over the last century. Nuclear power, which currently provides 10% of the world's total electricity and 40% of the low-carbon electricity for advanced economies (Varro et al., 2019), has the potential to help us transform our energy systems away from fossil fuels so that we can mitigate the devastating effects of climate change while powering our modern world.

See Also: Physics of the Fission Chain Reaction in Thermal and Fast Reactors.

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Nuclear Fission II

Andrew E. Stuchbery, Department of Nuclear Physics, The Australian National University, Canberra, ACT, Australia

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Glossary

Atomic number, Z The number of protons in a nucleus. The number of protons determines the number of electrons that attach to the nucleus, and hence the chemistry; thus each atomic number corresponds to a chemical element. Examples are $Z = 1$ for hydrogen, $Z = 6$ for carbon, $Z = 90$ for thorium and $Z = 92$ for uranium.

Compound nucleus The nucleus formed by combining a projectile and target nucleus. The compound nucleus may subsequently break up in different ways. For example, the compound nucleus formed by addition of a neutron to ^{235}U is ^{236}U , and subsequently it may fission or decay by gamma-ray emission.

Coolant Liquid or gas used to remove heat from the reactor core.

Coulomb repulsion Like charges to repel. In atomic nuclei Coulomb repulsion refers to the electrostatic repulsion between the positively charged protons, either within a given nucleus or between two nuclei in close proximity.

Cross section A measure of the likelihood that a nuclear reaction takes place when a projectile encounters a target nucleus, expressed in terms of the effective cross-sectional area presented by the target nucleus to the projectile. The units are therefore those of area, usually cm^2 or barns (b), where $1 \text{ b} = 10^{-24} \text{ cm}^2$. A larger cross section means a higher probability that a reaction takes place. The interaction of a given particle with a given nucleus can have several possible outcomes, each with a certain probability or cross section. The cross section is usually represented by the Greek letter sigma, often with a subscript to denote the particular outcome of the reaction. The total cross section is a sum of cross sections for each possible outcome. For example, the absorption of a neutron on ^{235}U can result in fission, or to the formation of ^{236}U , which can be designated as neutron capture. Thus $\sigma_a = \sigma_f + \sigma_c$, where σ_a is the absorption cross section, and σ_f and σ_c are the fission and capture cross sections, respectively.

Elastic scattering A collision between two particles in which their directions of motion and velocities can change but the total kinetic energy is conserved. In nuclear physics elastic scattering means that neither the projectile nor the target nucleus gains internal excitation energy as a result of the collision.

Electron volt, eV, keV, MeV A unit of energy. One electron volt (eV) is the energy gained by an electron accelerated by a potential difference of 1 V. $1 \text{ eV} = 1.602 \times 10^{-19} \text{ joule}$. The abbreviation eV can be prefaced by standard metric prefixes, so

meV indicates 10^{-3} electron volts (or millielectron volts), keV indicates 10^3 electron volts, and MeV indicates 10^6 electron volts (or megaelectron volts).

Fast neutrons Neutrons with an energy of the order of 1 MeV that are emitted during fission.

Fission fragments The two nuclei that result from the fission of a heavier nucleus.

Fuel Source of nuclear energy usually uranium or plutonium; it can contain fissile and fertile material.

Inelastic scattering A collision between two particles in which total kinetic energy is not conserved. In nuclear physics inelastic scattering means that either the projectile or the target nucleus, or both, gain internal excitation energy as a result of the collision.

Isobars Nuclei that have the same number of nucleons and hence the same mass number, A . The parent and daughter nuclei in beta decay are isobars.

Isotones Nuclei that have the same number of neutrons but differing numbers of protons (cf. isotopes).

Isotopes Nuclei that have the same number of protons but differing numbers of neutrons (cf. isotones). Isotopes are variants of the same chemical element.

Mass number, A The number of protons plus neutrons (i.e. the total number of nucleons) in an atomic nucleus. Thus the mass number is the sum of the proton or atomic number (Z) and the neutron number (N), $A = Z + N$.

Moderator Material in a thermal reactor that slows down fast neutrons, usually water, heavy water, or carbon in the form of graphite.

Multiplication factor k The number of fission neutrons in one generation of a chain-reaction divided by the number of fission neutrons in the preceding generation.

Neutron fluence The neutron flux integrated over a period of time. It can be described as the total number of neutrons passing through a unit area over a period of time, or as the total length travelled by all free neutrons per unit volume over a period of time. The fluence is usually expressed in the units of neutrons per square centimeter.

Neutron flux (ϕ) A measure of the intensity of the free neutrons in a reactor and their rate of flow. It is evaluated as a product of the number of neutrons per unit volume n and the neutron velocity v , $\phi = nv$ and is usually expressed in the units of neutrons per square centimeter per second. It can also be described as the number of neutrons passing through a unit area per unit time, or as the total length travelled by all free neutrons per unit volume per unit time.

Neutron number, N The number of neutrons in a nucleus. Each chemical element with a specified number of protons can exist as many isotopes that have different numbers of neutrons. The isotopes can be stable or radioactive. For example most of the hydrogen that exists in nature has a nucleus that consists of a single proton. However a small fraction of natural hydrogen is deuterium, which has one proton and one neutron in the nucleus. Tritium, with one proton and two neutrons, is a radioactive isotope of hydrogen. The element tin (chemical symbol Sn), with $Z = 50$, has the most stable isotopes.

Nucleon A proton or a neutron. Nucleon is a generic name for these constituents of the atomic nucleus.

Reactor core The part of the reactor containing the fuel, where fission takes place and heat is generated.

Reflector Material to help reflect neutrons back into the reactor core (often simply more moderator).

Scission The point during fission when the neck between the two nascent fission fragments breaks and the two fission fragments first separate.

Thermal neutrons Neutrons with an energy of the order of 0.025 eV (electron volt), which corresponds to room temperature (20°C). Fission-emitted neutrons can slow-down by successive scattering collisions with the moderator (and also some other core constituents) to nearly thermal equilibrium with the moderator nuclei. For comparison, 300°C corresponds to 0.049 eV.

Review of neutron-induced fission

Fission was introduced in the previous chapter (Klay, 2021). Here we give a descriptive review of the properties of fission with an emphasis on those relevant to sustaining a chain reaction. The following discussion will consider neutron-induced fission of ^{235}U , however most of the discussion is equally relevant for other fissile nuclei like ^{239}Pu and ^{233}U .

When ^{235}U captures a neutron it forms the compound nucleus ^{236}U in a highly excited state. The excitation energy comes from the fact that nucleons bind strongly as pairs; it is for this reason that the familiar fissile isotopes all have an odd number of neutrons. Even in its ground state, the ^{236}U nucleus is already deformed, having a shape like a rugby ball with its long axis about 30% longer than the diameter of its circular profile at the center. With the added excitation energy, the nucleus becomes free to explore new shapes, and the strong Coulomb repulsion of the protons favors even more extreme nuclear deformations, eventually splitting the nucleus into two fission fragments. The point at which the connection between the fission fragments breaks is called scission. The separated nuclei so formed are then pushed apart by their mutual Coulomb repulsion. About 200 MeV of energy is released, with typically about 85% of this energy appearing as kinetic energy of the two fission fragments.

The fission fragments have unequal masses. The heavier one has an atomic number near $A = 140$, and the other near $A = 95$. Until recently, the origin of the mass difference was thought to be directly related to the extra stability of the heavy fragment

near the doubly magic nuclide ^{132}Sn ($Z = 50$ and $N = 82$), however [Scamps and Simenel \(2018\)](#) have recently shown, on the basis of sophisticated theoretical calculations, that it is octupole correlations that determine the mass split. In more pictorial language, it is clear from representations of fission like those in [Fig. 1](#) that the necking process forces the precursors of the fission fragments to take a pear (octupole) shape. The nuclear shell structure dictates that nuclei near $A = 140$ with 52–56 protons and 84–88 neutrons can acquire a pear-shaped deformation at little or no cost in energy, and it is this process that drives the asymmetry of the fission fragments. Obviously, if one fragment has about 140 nucleons, the other must have the remainder.

The fission fragments are neutron rich—that is they have more neutrons than the stable isotopes of the element. For example, say ^{236}U splits into $^{136}\text{Te} + ^{100}\text{Zr}$, the neutron/proton ratios for these three nuclei are $144/92 = 1.57$, $84/52 = 1.62$, and $60/40 = 1.50$, respectively. In contrast, the heaviest stable isotope of tellurium is ^{130}Te with $78/52 = 1.50$ and the heaviest stable isotope of zirconium is ^{96}Zr with $56/40 = 1.40$. The fission fragments are not only neutron-rich, but they have internal excitation energy, some of which is released by emitting neutrons. These neutrons from fission are emitted almost immediately after scission by the fission fragments and take about 5 MeV of the 200 MeV released. For thermal-neutron-induced fission of ^{235}U there are on average just under 2.5 neutrons per fission. The number of emitted neutrons is well described by a Gaussian distribution with a standard deviation close to unity. Thus the probability of emission is about 30% each for 2- and 3-neutron emission; for 1- and 4-neutron emission, the probability is about 15%. Emission of 1–4 neutrons therefore accounts for about 90% of the fission events. The fact that there are on average several neutrons emitted per fission enables the chain reaction.

About 10% of the energy from fission is held as binding energy in the neutron-rich fragments, and this energy is gradually released through beta decay. Neutrons are converted to protons, releasing beta particles (and antineutrinos) until a stable isotope with the same mass number is reached. In a few quite rare cases, the beta decay parent populates a highly-excited state in the daughter that then emits a neutron. This process gives rise to a beta-delayed component in the neutrons resulting from fission. The fraction of neutrons per fission with delayed emission is usually denoted as β . Although β is small ($< 1\%$), these beta-delayed neutrons help control the chain reaction in a reactor.

Fission chain reaction

The concept of the nuclear chain reaction is illustrated in [Fig. 2](#). An initial neutron induces the fission on a fissile nucleus (^{235}U in this example), which results in the emission of two (or more) neutrons. If two of the emitted neutrons in turn induce a fission producing another two neutrons, the number of fissions will double with every “generation.” The number of fissions and the energy release grows exponentially and rapidly if the process remains unchecked, resulting in the explosive power of an atomic bomb. To achieve this explosive potential requires highly pure fissile material (^{235}U or ^{239}Pu).

The fission rate R is determined by the product of the number of ^{235}U nuclei present N_{235} , the fission cross section σ_f , and the neutron flux, ϕ : $R = N_{235}\sigma_f\phi$. Note that the neutron flux determines the fission rate at a particular point in time, but the fission rate then determines how the neutron flux will develop with time.

For a nuclear reactor, a steady state is required. The rate of production of neutrons must balance the rate at which neutrons are being lost and subsequent fissions are being induced. On average only one of the neutrons that results from fission can be allowed to induce another fission event. Such a self-sustaining chain reaction can be achieved with low-enriched or natural uranium when it is properly mixed with a neutron moderator, or with uranium enriched in ^{235}U to higher than about 12% without the use of a moderator.

The condition in which the rate of fission is held steady in a chain reaction is called criticality: the number of neutrons in each generation is the same. In reactor theory, the multiplication factor is defined as $k = (\text{number of neutrons in current generation})/(\text{number of neutrons in the previous generation})$. Thus, the equilibrium condition is $k = 1$. If $k < 1$, the reactor is subcritical and cannot sustain the chain reaction. If $k > 1$, the reactor is supercritical and, if not checked, a dangerous runaway condition can result.

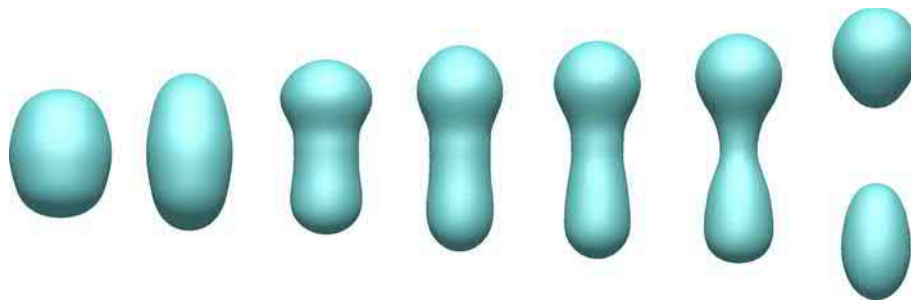


Fig. 1 Representation of the fission process in which a heavy nucleus divides into two. Elongation, neck formation and scission are indicated. For spontaneous and thermal-neutron induced fission there is an asymmetry in the fragment masses; one is near $A = 95$ and one is near $A = 140$. These images are artistic representations of fission based on microscopic calculations. Picture credit: Cedric Simenel.

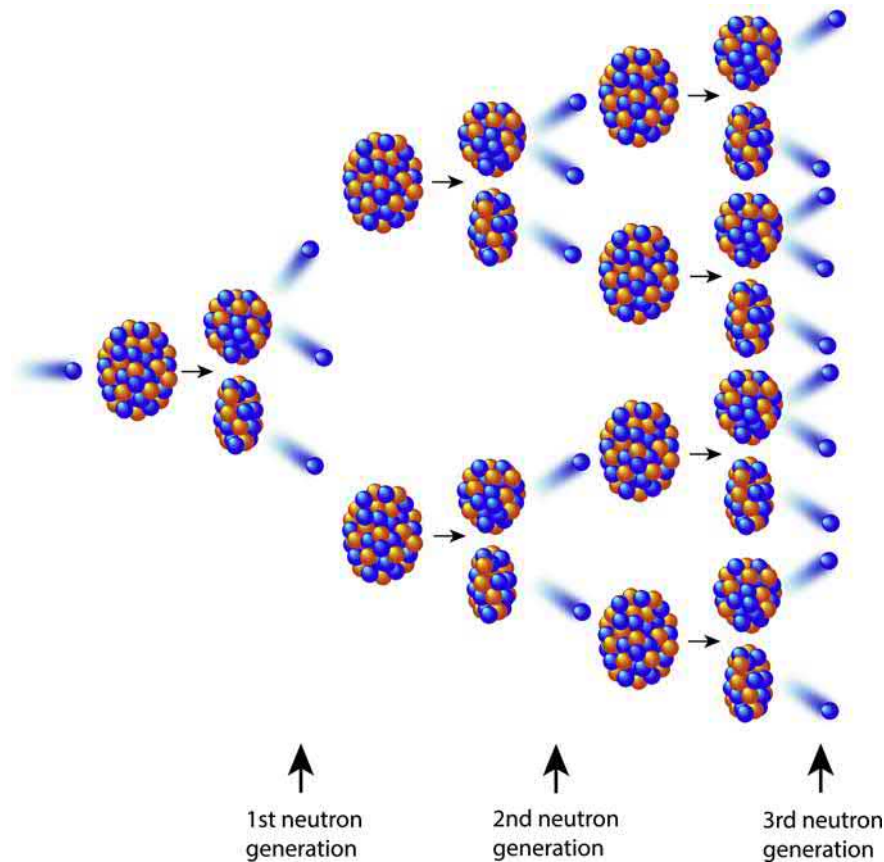


Fig. 2 Chain reaction by neutron-induced fission of ^{235}U .

A further distinction can be made between a reactor being prompt supercritical or delayed supercritical. The small beta-delayed component in the neutrons resulting from fission, which delays the release of neutrons by seconds to minutes, was noted above. A reactor is prompt supercritical if $k > 1$ without any contribution from the delayed neutrons, i.e. if $k > 1 + \beta$. It is delayed supercritical if $k > 1$ solely due to the delayed neutrons, i.e. if $1 < k < 1 + \beta$. Reactors are operated in the delayed supercritical state so that they can be safely controlled by adjustment of the amount of neutron-absorbing material in the core, such as by mechanical movement of control rods.

The chain reaction and the first human-made nuclear reactor

Leo Szilard in September 1933 is generally credited with being the first person to seriously consider a nuclear chain reaction as a means of releasing energy. He was reacting to a newspaper report of Lord Rutherford's statement that anyone who looked for a source of power in the transformation of atoms was talking moonshine. Szilard's insight came after the discovery of the neutron but before the discovery of fission. It occurred to him that neutrons would be more likely able to sustain a chain reaction because they are not charged and can readily pass through a solid material until they encounter a nucleus with which they can interact and induce a nuclear reaction. He realized that if one neutron could initiate some kind of nuclear reaction that led to the release of two neutrons, there was a good chance that one of these neutrons could initiate another reaction, in turn releasing another two neutrons. Such a process would be exothermic (i.e. release energy) and industrial power and/or nuclear weapons could be possible.

The discovery of fission is often marked by the publication of Lise Meitner and Otto Frisch in *Nature* on February 11, 1939 entitled "Disintegration of Uranium by Neutrons: a New Type of Nuclear Reaction" (Meitner and Frisch, 1939). In fact, Meitner and Frisch were re-interpreting experiments by Hahn and Stassmann (1939) in Berlin who did not know how to explain the appearance of the element barium in a sample of uranium they bombarded with neutrons. Some years previously Fermi had performed similar experiments in Rome but had mistakenly interpreted the products of the bombardment of uranium by neutrons as heavier rather than lighter elements (Fermi et al., 1934). Meitner and Frisch realized that if the uranium nucleus was excited and considered as a charged liquid drop, it would be energetically favorable for it to split into two roughly equal pieces. They estimated that the energy release would be about 200 MeV. In the same volume of the journal *Nature*, Bohr (1939) commented on the mechanism of fission, mentioning the factors that affect the stability of a heavy nucleus considered as a charged liquid drop. Frederic Joliot soon afterward

proved the fission hypothesis and realized that neutrons must be emitted in the process (Von Halban et al., 1939). Three patents were taken out in early May 1939, describing a device for energy production, a method for stabilizing a device for energy production, and a method for perfecting explosive charges (an atomic bomb).

The first human-made self-sustaining nuclear reaction was demonstrated by Enrico Fermi and co-workers at the University of Chicago on December 2, 1942. Designated CP-1 (Chicago Pile 1), this reactor consisted of about 40 tons of natural uranium, including lumps of metal and oxide, in a matrix of about 385 tons of graphite. The graphite blocks, which served as moderator and reflector, were stacked in a neat cubic structure. With about 2 kW of power, ambient air cooling was sufficient. There was no shielding or containment. This experimental reactor was part of the Manhattan project. Already plans were in place for the full scale Hanford reactors to make plutonium. The purpose of CP-1 was to measure the so-called multiplication factor or k -value of the reactor, and hence inform the final design of the Hanford reactors. (A more detailed discussion of the multiplication factor is given below.)

Detailed accounts of events on December 2, 1942 have been given elsewhere (e.g. Rhodes, 1986, Hewlett and Anderssen, 1962). In short, the reactor was kept from going critical during assembly by including rods of cadmium, a known strong neutron absorber, among the graphite blocks. To bring the pile to criticality—the self-sustaining reaction conditions—the cadmium rods were gradually removed while the neutron count rate (proportional to the flux) was monitored. This process of removing control rods and monitoring the neutron count rate continued through the morning, and included some safety tests. There was an automatic safety system to drop a control rod into the core if the neutron count rate (flux) rose too high. This system tripped because the threshold was set too low and the reactor was shut down while Fermi went for lunch. Testing and pursuit of the sustained chain reaction resumed at 2 pm. The reactor finally went critical at 3:35 pm and operated for 28 min before being shut down. It was determined that the multiplication factor was $k = 1.0006$.

CP-1 was experimented with for a few months before being dismantled and then revamped as Chicago Pile-2 (CP-2) on a new site and with new features such as radiation shielding, a small laboratory on top of the pile for experiments using neutrons from the reactor core, and a face with ports that allowed materials to be inserted into the core for irradiation. CP-2 had a thermal power level of 10 kW and ran for over a decade.

The achievements and outcomes from CP-1 were manifold. It demonstrated a controlled fission chain reaction and the production of a neutron flux for experiments and plutonium production. From the culmination of extensive experimental and theoretical studies, it established the foundations of nuclear reactor theory and measured the multiplication factor, which was difficult to calculate precisely at the time. As such, it provided essential input data for the design of larger-scale reactor projects already under construction in the Manhattan project, namely the X-10 experimental reactor at Oak Ridge National Laboratory and the Hanford reactors for plutonium production. Much later (in May 1955) a patent for an inhomogeneous nuclear reactor was issued to Fermi and Szilard.

CP-1 and Fermi reactor theory

CP-1 validated what is now known as the *Fermi reactor theory* (Fermi, 1947; Esposito and Pisanti, 2010). In this section CP-1 is discussed in terms of the concepts of that theory as relates to sustaining the chain reaction.

Achieving criticality—Requirements

At the time of CP-1, only natural uranium was available. Natural uranium contains only 0.72% fissile ^{235}U with the remaining 99.28% being fertile ^{238}U (Shusterman, 2021). To achieve criticality and sustain the chain reaction in a reactor using natural uranium as fuel was a challenging technical problem. The requirements were that:

- (i) There must be a high probability that fission results when a neutron is absorbed in the fuel (i.e. the natural uranium). Neutron absorptions in the fuel that do not result in fission must also be minimized.
- (ii) Neutron absorption by materials other than the fuel in the reactor must be minimized.
- (iii) The escape of neutrons from the reactor core must be minimized. In other words, the reactor must be of sufficient physical size.

The neutron cycle and reactor theory—Fermi's four-factor formula

These requirements for establishing a stable chain reaction in a reactor can be discussed in terms of a neutron cycle. It can be considered to begin with a specified number of thermal neutrons being absorbed in the fuel and producing a new generation of fission or fast neutrons. The neutron cycle follows the fate of these fast neutrons as they interact with the fuel, slow in the moderator, and eventually interact with the fuel as thermal neutrons to initiate a new cycle (or generation).

As described above, the multiplication factor is defined as $k = (\text{number of neutrons in current generation})/(\text{number of neutrons in the previous generation})$. The Fermi four-factor formula for k_{∞} , the multiplication factor for a reactor with an idealized infinite sized core that does not lose neutrons out of the core, is

$$k_{\infty} = \eta \epsilon p f,$$

where η is the number of neutrons produced by fission from the absorption of one neutron in the fuel, ϵ is the fast fission factor, p is the resonance escape probability, and f is the thermal utilization factor. The following discussion describes these quantities in relation to CP-1.

Increased fission probability through moderation

The first requirement for a sustained chain reaction is that a sufficient number of neutrons must result from the absorption of a neutron in the fuel. In order to achieve a chain reaction, the probability of fission following the absorption of a neutron must be high. The probability that fission results when a fast neutron interacts with natural uranium fuel is too low to establish a fission chain reaction. This probability can be increased by enriching the fuel in ^{235}U above the natural level of 0.72%; however, this option was not available for the first reactor in 1942.

Nature, however, gives an alternative way to increase the probability of inducing fission in natural uranium. The cross section (probability) for neutron-induced fission of ^{235}U increases by about a factor of 600 if the fast neutrons from fission (their energy is about 1 MeV or 10^6 eV) are slowed to “thermal” energies (0.025 eV or 2.5×10^{-8} MeV). This enhancement in the fission cross section compensates for the low abundance of ^{235}U in natural uranium. A *moderator* having low probability for neutron absorption must be introduced to slow the fast neutrons to thermal energies. The moderator in CP-1 was carbon in the form of high purity graphite.

While the above discussion distills the key points, the viability of a fast versus a thermal reactor is more subtle. It depends on the relative probabilities of the alternative outcomes of a neutron encounter with a uranium nucleus in the fuel (i.e. for both ^{235}U and ^{238}U). The important processes are fission, capture leading to ^{236}U or ^{239}U , and scattering (especially inelastic scattering at energies near 1 MeV). The total cross section is essentially the sum of the fission, capture and scattering cross sections.

The cross sections for neutron-induced reactions on ^{235}U and ^{238}U are shown in Figs. 3 and 4, respectively. Also indicated is the distribution of fast neutrons from the fission of ^{235}U , which peaks below 1 MeV or 10^6 eV.

For fast neutrons encountering a uranium nucleus, the most likely outcome—by an order of magnitude or more for ^{238}U —is that the neutron will scatter and lose energy. It will be noted below that neutrons lose little energy in elastic collisions with uranium nuclei. However for neutron energies near 1 MeV the inelastic scattering cross section dominates and scattering with loss of energy is dominant.

Once the neutron energy drops below 1 MeV, the fission cross section for ^{238}U becomes essentially zero while the capture cross section grows. It is this combination of features in the scattering, capture and fission cross sections of ^{238}U that make a fast reactor impossible with natural uranium. In contrast, the cross section for fast-neutron fission of ^{235}U remains of the order of 1 b, but the ^{235}U content of natural uranium is too small to compensate for the inadequate neutron production and strong absorption of neutrons by ^{238}U .

The 600-fold enhancement in the fission cross section for ^{235}U at thermal energies leads to an effective cross section of about 4.2 b for thermal-neutron induced fission in natural uranium. The (parasitic) neutron-capture cross sections leading to ^{236}U and ^{239}U shown in Figs. 3 and 4 are also increased, but the effective cross section of 3.4 b at 0.025 eV, mainly due to capture on the ^{238}U , is sufficiently small. In contrast to the adverse role of scattering at higher energies, it is not important at thermal energies,

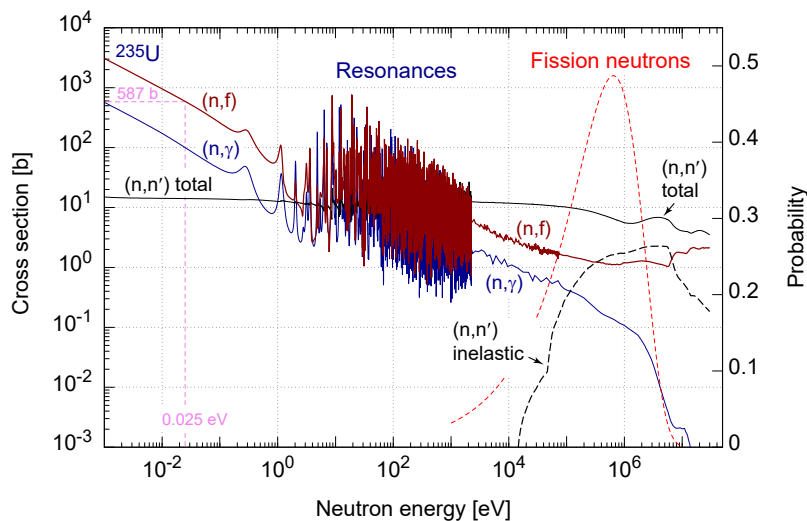


Fig. 3 Cross sections for neutron-induced reactions on ^{235}U as a function of energy. Fission is designated (n,f), capture (n, γ), and scattering (n,n'). The inelastic scattering contribution to the total scattering cross section is also indicated. The cross-section scale is on the left. The probability distribution of neutron energies that results from fission of ^{235}U is also indicated with its scale on the right.

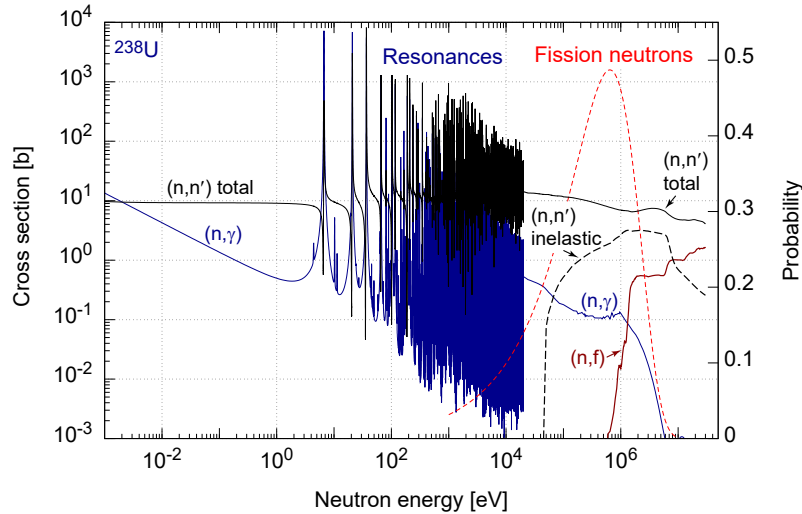


Fig. 4 Similar to Fig. 3 but for neutron-induced reactions on ^{238}U .

because thermal neutrons are (by definition) in thermal equilibrium with their surroundings. For each thermal neutron absorbed in natural uranium about 1.35 are produced on average. Criticality can thus be achieved with natural uranium at thermal energies.

The choice of carbon as moderator for CP-1

The moderation process slows fast neutrons with energies of the order of 1 MeV to thermal energies, usually specified as 0.025 eV, which is the energy equivalent to room temperature. The moderation process is based on the elastic scattering of neutrons from the nuclei of the moderator. In other words, the neutrons lose kinetic energy by imparting kinetic energy to the nuclei of the moderator. A good moderating material is one for which a neutron:

- (i) loses a large fraction of its energy in a single collision, which “speeds up” the moderation process,
- (ii) has a high probability of scattering off the moderator nuclei, i.e. the elastic scattering cross section should be high,
- (iii) has a low probability of being absorbed by the moderator, i.e. the absorption cross section should be low.

From the point of view of neutron energy-loss per collision, the lower the atomic mass of the moderator, the better. In terms of elastic collisions, hydrogen is best whereas uranium is very poor. The hydrogen in ordinary water is suitable as a moderator in this respect, but Fermi found in early experiments that it absorbs too many neutrons to be used in a reactor with natural uranium as fuel. Fermi did not have enriched uranium available, so light water was not a viable option for CP-1, or for the early reactors fuelled by natural uranium that followed. Szilard suggested to Fermi in July 1939 that carbon or possibly heavy water might be suitable alternatives.

Heavy water, D_2O , which replaces the hydrogen in ordinary water with deuterium (the isotope of hydrogen with one proton and one neutron) is an excellent moderator because deuterium has a low atomic number and it does not absorb neutrons. However, deuterium is only 0.015% of naturally occurring hydrogen. It was not a viable choice for the first reactor because it was both difficult to separate from ordinary hydrogen and needed in large quantities.

Carbon in the form of graphite then became the best choice for the moderator. It was much more readily available than D_2O , and although not as efficient at slowing neutrons as hydrogen or deuterium per elastic collision, the probability of absorbing neutrons was found by Fermi and his group to be sufficiently small as to allow the successful construction of a reactor using natural uranium as fuel and carbon as moderator.

Beryllium was also considered as a moderator. A.H. Compton argued that it not only has the required low atomic weight, but also would add (via $(n,2n)$ reactions) rather than remove neutrons (Hewlett and Anderson, 1962). Experiments proved it to be a good moderator but graphite was more readily available.

As will be discussed below, one engineering consequence of the use of graphite is that a heterogeneous reactor core is required. The resonance escape probability for natural uranium homogeneously distributed through a matrix of graphite is too small to achieve criticality.

The number of neutrons from fission

In the four-factor formula, η is the number of (fast) neutrons produced by the absorption of one thermal neutron in the fuel, including both the fissile and fertile isotopes present. In Fermi’s CP-1 reactor the fuel was natural uranium, which is 0.72% fissile ^{235}U and 99.28% fertile ^{238}U . Only the fissile ^{235}U can undergo thermal-neutron induced fission, releasing energy and neutrons.

Some of the thermal neutrons can be absorbed by the fuel and not result in fission. An important process is neutron-capture on ^{238}U (the majority of the fuel) to form ^{239}U , which then undergoes two beta decays to form fissile ^{239}Pu . This process of converting fertile material to fissile material is discussed below.

The average number of neutrons per thermal-neutron induced fission of ^{235}U is $\nu = 2.44$, however $\eta < \nu$ even for pure ^{235}U because neutrons can be absorbed without causing fission. Denoting the thermal fission cross section as σ_f and the total absorption cross section as $\sigma_a = \sigma_f + \sigma_c$, where σ_c is the capture cross section, gives $\eta = \nu \frac{\sigma_f}{\sigma_a} = \nu \left(\frac{\sigma_f}{\sigma_f + \sigma_c} \right)$. Evaluating the cross sections for pure ^{235}U at 0.025 eV from Fig. 3 gives $\eta = 2.08$ fast neutrons per fission. Essentially the same value is obtained for an average over the Maxwell-Boltzmann distribution, whether at room temperature or 300 °C.

For natural uranium $\eta = 1.36$. Because η is so close to unity, achieving criticality with natural uranium requires attention to the reactor design to minimize the non-fission absorption of neutrons in the core, and the loss of neutrons from the core.

In addition to the neutrons produced by thermal fission, some neutrons can be produced by fission of both ^{235}U and ^{238}U induced by fast neutrons (see Figs. 3 and 4). The fast fission factor ϵ accounts for these fast fission neutrons. The physical size, shape and chemical form of the fuel determines the probability that a fast neutron can induce a fast fission before slowing-down to below the fast fission threshold. It is a relatively small effect. Typically, $\epsilon \approx 1.03$ for a reactor fuelled with natural uranium; the contribution from ^{238}U is dominant.

Minimizing neutron absorption—Resonance escape probability

Neutron absorption by materials other than the fuel in the reactor must be minimized. As a first approximation, the reactor can be considered to consist only of the fuel and the moderator. In addition, cadmium rods, which strongly absorb neutrons, were included in the core of CP-1 to control the chain reaction and prevent the reactor from going critical in an uncontrolled fashion. Criticality was approached steadily by gradually withdrawing the cadmium rods. Carbon (in the form of graphite) was the chosen moderator because it can slow neutrons with little absorption.

During the moderation process there is a critical range of neutron energies where the probability of the neutron being absorbed in the ^{238}U (which is the dominant component of the fuel) becomes very high. This is the region between 10 and 1000 eV where there are many sharp peaks called resonances in the neutron capture cross section (see Figs. 3 and 4). Resonances occur when the neutron energy is matched to form an individual excited quantum state in the compound nucleus, which makes the capture particularly favorable. It is the resonant absorption of neutrons that make it impossible to achieve criticality with a homogeneous mixture of graphite and natural uranium for which $k \approx 0.8$.

For the design of a thermal reactor operating on natural or low-enrichment uranium, it is essential to minimize the resonant absorption of neutrons on ^{238}U , the dominant component of the fuel. In other words, the resonant escape probability, p , must be maximized.

The solution is to build an inhomogeneous (also referred to as a heterogenous) reactor, as was CP-1. In an inhomogeneous reactor the fuel is not uniformly distributed through the moderator but is contained in rods, lumps, or sheets. The dimensions and geometry of the fuel and moderator are chosen so that the fast neutrons from fission in the fuel have a high probability of leaving the fuel at high energy (above the energy regime where resonances occur in the ^{238}U neutron-capture cross section) and getting into the moderator where they can be slowed to thermal energies before diffusing back into the fuel where they have a high probability of initiating another fission. Thus the physical geometry of the reactor is designed to reduce the probability that a neutron is in the fuel when its energy is in the resonance region and thereby the *resonance escape probability* is increased.

It is worth noting that the moderation process takes about a microsecond, whereas the moderated neutrons at thermal energies typically diffuse through the core for a thousand times longer (about a millisecond) before being absorbed (Krane, 1988).

Due to the complexity of the cross sections as a function of energy in the resonance region, coupled with non-uniform (inhomogeneous) geometry of fuel and moderator material, which also affects the spatial and energy dependence of the neutron density, a manual calculation of the resonance escape probability is extremely difficult. Today it can be done with the benefit of detailed nuclear data and sophisticated computer codes, but such were not available in 1942. An experimental measurement of the resonance escape probability was an important outcome of CP-1.

Thermal utilization—The fate of thermal neutrons

The final stage of the neutron cycle concerns the fate of the thermal neutrons in the core. The thermal utilization factor, f , is the fraction of the thermal neutrons absorbed in the fuel relative to the total neutron absorption in the core. In order to achieve criticality, absorption in non-fuel components must be minimized. Comparing homogeneous and inhomogeneous reactors, the thermal utilization factor is reduced in an inhomogeneous reactor, but the gain in the resonance escape probability is greater. CP-1 demonstrated that a chain reaction can be established and maintained in an inhomogeneous reactor with natural uranium as fuel and graphite as the moderator.

Leakage—Minimizing loss of neutrons from the core

The four-factor formula for k_∞ ignores the loss of neutrons from the core of the reactor. Two strategies were available to reduce the loss of neutron from the core of CP-1. First, the escape of neutrons from the reactor core becomes less important if the core size is

increased. The critical size of the core can be determined by use of the reactor equation (Tsvetkov, 2021; Duderstadt and Hamilton, 1976; Lamarsh and Baratta, 2001; Lilley, 2001), also developed as part of the Fermi reactor theory. Second, the escape of neutrons from the reactor core can be minimized by use of a reflector, which physically is often simply extra moderator material.

It is worth noting that a reflector in a reactor does not reflect neutrons in the way that a mirror reflects light, where one pictures a ray of light bouncing off the surface of the mirror like a ball off a wall. A more accurate picture of the reflector in a reactor is that it provides a medium in which the neutrons can be thermalized (if they are not already at thermal energies), and once thermalized can bounce around, colliding at random with the reflector atoms, with a chance of eventually diffusing back into the reactor core. This process is not as efficient as the reflection of light by a mirror, but without the reflector, any neutron that leaves the reactor core will continue in a straight line path out of the reactor core, whereas with a reflector, some neutrons will diffuse back into the core. A small fraction of the fast neutrons that leak to the reflector may scatter back to the core even before they thermalize. Often the reflector is simply additional moderator material surrounding the core. Thus in CP-1 the graphite matrix served as both moderator and reflector.

The natural nuclear fission reactors of Oklo, Gabon

In 1956 Paul Kuroda used the Fermi reactor theory (the four-factor formula) to show that in the early history of the Earth a uranium deposit of appropriate dimensions with a suitable water content could easily sustain a chain reaction. Because the half-lives of ^{235}U ($T_{1/2} = 7.04 \times 10^8$ years) and ^{238}U ($T_{1/2} = 4.47 \times 10^9$ years) differ, the ^{235}U content in natural uranium is increased on geological time scales into the past. Today natural uranium contains 0.72% ^{235}U , whereas 1 billion years ago (10^9 years) the ^{235}U content was about 1.6% and 2 billion years ago it was near 4%, comparable to the enrichment used in a modern light-water moderated (and cooled) power reactor.

Kuroda's work was largely forgotten until 1972 when it was discovered that the ^{235}U content of some uranium samples from mines in the Oklo region in Gabon, Africa, was significantly below the usual 0.72%. This evidence of missing fissile ^{235}U came as a surprise and was perplexing at first, until the predictions of Kuroda (Kuroda, 1956, 1960) and the speculations of others (Wetherill and Inghram, 1953) were recalled. It then became evident that the ^{235}U content of the deposit was depleted because it had been consumed in an ancient natural nuclear reactor, very much along the lines of those predicted by Kuroda (Meshik, 2005).

Description of the Oklo reactors

In total 16 reactor zones were identified in the Oklo region and a seventeenth in nearby Bangombé (de Laeter and Hidaka, 2007). Unfortunately all but the Bangombé reactor have been mined out. However extensive studies of isotope ratios and other properties of the reactor zones have been performed that allow us to build up a rather complete picture of their operation.

The reactor zones range in depth from near surface (less than 100 m) to more than 400 m underground. Those nearer the surface have clearly been subject to weathering, while the deeper ones are considered to have better preserved the fission products and other consequences of the reactor operation. The reactors were typically within lens-shaped conglomerates with length and width of order 10 m and up to about 0.8 m thick. These dimensions are close to Kuroda's estimates, including that a thickness greater than "a few feet" is needed to achieve criticality. The reactors themselves were in uranium-rich zones of the deposit, consisting of uraninite (up to 60% uranium).

The requirements for a natural reactor

There are several requirements for a natural reactor to achieve criticality:

1. The uranium ore body must have high uranium content and appropriate dimensions and geometry; for example, it should be at least around 0.7 m thick. These requirements for the deposit ensure, first, that the infinite multiplication factor will exceed unity, and second, that the leakage of neutrons from the reactor zone is limited. Note that whereas the fissile material is ^{235}U , it is ^{238}U that provides the neutrons from spontaneous fission to initiate the reaction. In particular, for ^{238}U the spontaneous fission partial half-life is $T_{1/2}(\text{SF}) = 8.2 \times 10^{15}$ years, much shorter than for ^{235}U for which $T_{1/2}(\text{SF}) = 1.0 \times 10^{19}$ years. The dominant decay mode for both uranium isotopes is alpha decay, for which ^{235}U has the shorter partial half-life. Additional neutrons can be produced by alpha-particle induced reactions on light elements that may be present.
2. The uranium must contain a significant amount of fissionable ^{235}U . The ^{235}U level of 0.72% today in natural uranium is not sufficient to allow a natural reactor to operate in a uranium deposit. Natural reactors can therefore only have existed on Earth in the distant past.
3. A moderator is required to slow the fission-produced neutrons. This is most likely to be water in a natural reactor.
4. There must be limited quantities of neutron-absorbing elements in the deposit that would inhibit the chain reaction.

None of these requirements is exceptional for an ancient uranium deposit, nor are the geological features of the Oklo deposits unique when considered individually. Soon after the Oklo reactors were discovered, there was therefore some expectation that other ancient reactors would be discovered in other uranium deposits around the world (Maurette 1976; Cowan, 1976). However none have been found. For example, a search for fossil reactors in several mine sites in the Alligator River Uranium Field in Australia

found no evidence of a sustained chain reaction. The inferred low neutron fluence at Narbarlek, a site considered a good candidate for a natural reactor, was consistent with neutron production from spontaneous fission of ^{238}U , (α, n) reactions, and some slow-neutron induced fission of ^{235}U (Maas and McCulloch, 1990).

Although it is possible that some ancient reactors have been mined out without being noticed, it remains a puzzle as to why no other fossil reactors have been found.

Nuclear forensics of the ancient reactors and their operation: Isotopic analysis of fission fragments

Much of what we now know about the Oklo reactors has been determined by careful studies of the isotope ratios of elements created as fission fragments during the operation of the reactor (de Laeter and Hidaka 2007). The measurement method is known as mass spectrometry. Mass spectrometry covers a range of techniques which measure the relative abundances of the different isotopes of a given element in a sample. For example, we know that the consumption of ^{235}U in the Oklo deposits was indeed due to nuclear fission because the isotopic ratios of several rare earth elements collected from the reactor zone are very different from those normally found in nature but reflect those that result from fission.

The example of the neodymium isotopes is clear and instructive: In natural neodymium, the most abundant isotope is ^{142}Nd (27.13%), but this isotope cannot be produced in fission because the decay of the neutron-rich mass 142 isobars from fission terminates on stable ^{142}Ce , which blocks the production of ^{142}Nd . In contrast, the most abundant neodymium isotope from fission is ^{143}Nd . Samples of Nd from Oklo ore show a very low fraction of ^{142}Nd ($\sim 1\%$) and an isotopic distribution that is closer to that of fission-produced Nd than natural Nd. The ^{142}Nd in the ore can be assumed to represent the natural fraction, not originating from fission. When it is used to correct the isotope ratios of the ore for a small background of Nd from natural sources, what remains closely matches the distribution produced by the fission of ^{235}U .

The age of the reactors has also been determined by several radioactive dating procedures based on mass and isotope ratios, with general agreement that the age of the Oklo deposit is approximately 2×10^9 (or 2 billion) years. Thus, on the geological time scale, the chain reactions commenced shortly after the formation of the deposits. At the time of the formation of the Earth, the ^{235}U abundance was about 17%. It is natural therefore to ask why no evidence has been found of even older natural reactors. The answer is in the geology and biological history of Earth: Uranium ore deposits did not form until around 2.2 billion years ago, linked to the sudden increase of oxygen in the atmosphere and water due to the activity of cyanobacteria, which in turn affected the solubility of uranium oxides and enabled the deposition of high-grade uranium ores.

Other nuclear properties of the reactors investigated include the average neutron flux, the average power, the pulsed nature of the reactor operation, the total time the reactors operated and even the average reactor operating temperature. These studies make use of the fact that the distribution of neutron-rich fission products is known, as are the lifetimes of the fission fragments as they decay toward the valley of stability. Moreover, some fission products, as well as some of the isotopes in the surrounding clay and rock, are strong neutron absorbers, whereas others are not. This detailed knowledge of nuclear properties, together with extensive measurements on the isotopic residues in and near the reactor zones, has made it possible to build up a rather complete description of the reactors.

The reactors operated for some hundreds of thousands of years in a self-regulating pulsed mode. The uranium deposits in the absence of water were subcritical. However when water permeated the reactor zone it acted as a moderator and the reactors became supercritical. The heat generated then heated and vaporized the water, driving it off and returning the reactor zone to a subcritical state. By careful analysis of xenon isotope ratios, and understanding the properties of their precursors in the mass-decay chains, it was possible to deduce that the critical or heating period was about 30 min whereas the cooling or subcritical time was about 2.5 h, the whole cycle being about 3 h. Eventually, through the consumption of ^{235}U by fission as well as its normal radioactive decay, the ^{235}U enrichment level in the deposit dropped below the level at which criticality could be achieved, and the reactors ceased to operate.

What we learn from the Oklo fossil reactors

The Oklo reactors have provided extensive opportunities to study the fascinating phenomenon of natural nuclear reactors. Although all but the Bangombé reactor have been mined out, there is ongoing research on the geology and operation of these reactors.

For those interested in nuclear power and the management of nuclear waste, the Oklo reactors have provided an incredibly long-term nuclear-waste storage experiment. Early measurements of the migration of rare earth fission fragments found the surprising result that they had not moved far from the reactor zone into the surrounding rock over a period of the order of 1 billion years. The long-term geochemical behavior of fission products and actinides has since been studied extensively.

In drawing conclusions about geological storage and/or disposal of nuclear waste, the Oklo data are useful in two ways. Firstly, direct and detailed studies can be made on what actually happened to the fission products and actinides in these natural reactors over the last billion years or so. Secondly, the same data can be used to test and help develop sophisticated computer codes that are to be applied to new engineered waste disposal repositories and other related applications. The second step is important because conclusions drawn directly from the Oklo example may not be valid for the chemical waste form and possibly different rock types around a modern disposal site. The general conclusion, however, is that the Oklo data give confidence that radioactive waste can be safely stored in geological repositories, particularly with appropriately engineered barriers.

The Oklo reactor data have also found applications in fundamental physics. Many fundamental theories beyond the standard model of particle physics predict that the fundamental constants in the laws of nature may vary (slowly) with time. Certain isotopic production processes in a nuclear reactor depend on the value of a particular fundamental constant. Thus certain isotope ratios in the residue of the Oklo reactors can give a measure of the value of these constants a billion years ago for comparison with modern measurements.

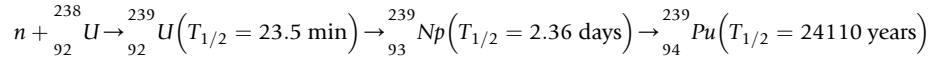
The case most often considered is the low energy $E_r = 97.3$ meV (milli-electron volt) resonance in the neutron capture on ^{149}Sm . For this resonance the capture cross section scales as E_r^{-2} and thus it is very sensitive to any shift in the resonance energy. Measurements of $^{149}\text{Sm}/^{150}\text{Sm}$ abundance ratios in the Oklo reactors have been used to set limits on the time and spatial variations in the fine structure constant, as well as limits on variations in the quark masses (Damour and Dyson, 1996; Flambaum and Wiringa, 2009; Berengut and Flambaum, 2012; Davis et al., 2014). Such endeavors require a firm theoretical understanding of the relationship between the fundamental constant and the energy shift as well as precise isotope-ratio data.

Feasibility of fuel self-sustaining fission (“breeding”)

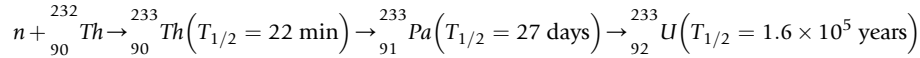
A typical power reactor with a ^{235}U enrichment of 3–5% clearly has more fertile ^{238}U in the fuel than fissile ^{235}U . The capture of neutrons on ^{238}U invariably leads to the production of ^{239}Pu , which is fissile. All power reactors therefore *convert* some fertile ^{238}U to fissile ^{239}Pu , and fission of this ^{239}Pu produced in the reactor core contributes to the power output of the reactor. There is the potential to increase the energy recovered from finite uranium and thorium resources by converting the fertile isotopes to fissile isotopes (Shusterman, 2021). It is even possible to *breed* more fissile material from the fertile isotope than is consumed in the fission process.

Neutron capture and decay paths for conversion and breeding

The dominant isotope in natural uranium, ^{238}U , can be converted to ^{239}Pu by neutron capture followed by two beta (β^-) decays:



In thorium-uranium conversion ^{232}Th is converted to ^{233}U by the following neutron capture and β^- decay path:



Conversion ratio and breeding ratio

The conversion ratio is defined as

$$C = \frac{\text{Fissile nuclei produced}}{\text{Fissile nuclei consumed}},$$

where the production of fissile nuclei follows the capture and decay paths indicated above, and the consumption includes both fission and capture. When the conversion ratio is larger than one, it is called the breeding ratio, sometimes designated B .

Table 1 shows the neutron production data for thermal and fast fission of the most relevant fissile isotopes for power generation. Along with the average number of neutrons produced per fission, ν , this table shows $\eta - 1$, which represents the number of neutrons available for breeding and losses following the absorption of one neutron on the pure isotope after allowing that one neutron is needed to sustain the chain reaction. We can write

$$\eta = 1 + C + \text{Capture} + \text{Loss}.$$

Table 1 Neutron production in fission. Values of ν are from ENDF/B-VII.1 (Chadwick et al., 2011). The cross sections required to evaluate the corresponding value of η are from ENDF/B-VIII.0 (Brown et al., 2018).

		^{235}U	^{239}Pu	^{233}U
Thermal neutrons (0.0253 eV)	ν	2.44	2.88	2.50
	$\eta - 1$	1.08	1.11	1.32
Fast neutrons (1 MeV)	ν	2.53	3.015	2.59
	$\eta - 1$	1.33	1.94	1.49

Note that *Capture* here is the parasitic neutron capture that does not result in the production of fissile material; capture by fertile isotopes producing fissile isotopes is included in C . Thus $\eta - 1$ is also the upper limit to the breeding ratio as it corresponds to the ideal case of no parasitic capture or loss. At thermal neutron energies neither ^{235}U nor ^{239}Pu with $\eta - 1 \approx 1.1$ are suitable fuels to drive a breeding cycle. Losses reduce the conversion ratio below one, which means that while conversion does take place, it is impossible to replace all of the fissile nuclei consumed.

Despite producing $\nu = 2.88$ neutrons per fission at thermal energies, ^{239}Pu has a low value of η because the neutron capture cross section at thermal energies is high. Breeding of ^{233}U from ^{232}Th is considered feasible at thermal energies. However, it is clear from **Table 1** that breeding is more favorable for fast-neutron induced fission in all cases of practical value. A breeding cycle based on ^{239}Pu as the fissile material and ^{238}U as the fertile material is particularly favorable.

Conversion in a thermal reactor

Returning to the case of a typical thermal power reactor, the conversion ratio can be estimated. The production of ^{239}U (and hence ^{239}Pu) can be attributed to two primary processes: (i) resonance capture of epithermal neutrons on ^{238}U , and (ii) capture of thermal neutrons on the ^{238}U . Thus, the conversion ratio can be written in terms of one minus the resonance escape probability $(1 - p)$ and in terms of η and η_{235} (the average number of neutrons produced by absorption of one neutron by pure ^{235}U). Assuming that the fast fission factor is unity, the conversion ratio in a thermal reactor can be written as

$$C = (1 - p)\eta_{235} + \left(\frac{\eta_{235}}{\eta} - 1\right),$$

where the first term is from epithermal resonant capture and the second term is from thermal capture (Bodansky, 2004). Note that the second (thermal) term is determined solely by the level of enrichment in ^{235}U , and that it increases as the enrichment decreases. The conversion ratio will therefore change with time, increasing as ^{235}U is consumed and η decreases. Graphite and heavy water moderated reactors tend to have higher conversion ratios ($C \approx 0.9$) than light water reactors ($C \approx 0.6$) because these reactors usually operate with natural or lower-enrichment uranium than light water reactors. For example, the thermal capture contribution to the conversion ratio $\left(\frac{\eta_{235}}{\eta} - 1\right)$ evaluated at 0.025 eV has the value of 0.54 for natural uranium, but is only 0.07 for 5% enriched uranium.

Because the conversion ratio depends on the uranium enrichment, it also depends on the burnup. The value of $C \approx 0.6$ corresponds to a burnup of about 60 gigawatt days per metric tonne of uranium.

Fast breeding, breeding gain and doubling time

It has been noted that breeding requires a conversion ratio in excess of 1, and that this can be achieved with fast reactors. In a fast reactor there is no moderator and cooling is usually achieved by use of a liquid metal circulating through the core. Several liquid metal fast breeder reactors have operated throughout the world. One example is the Russian-built BN-350 sodium cooled fast reactor, which was classified as a prototype with a target breeding ratio of about 1.4 (Orlov and Kochetkov, 1974). It has been reported to have achieved $C > 1.2$.

The degree of breeding achieved can be described by the *breeding gain* $G = C - 1$. It is also useful to define the doubling time, which is the length of time required to double the amount of fissile material in a breeder reactor (Murray, 2009). In other words, the doubling time is the time required to produce an excess of fissile material equal to the initial load. It can be estimated by considering a reactor with N_f fissile atoms per unit volume. The rate of fuel consumption is $N_f \sigma_a \phi$ where σ_a is the absorption cross section and ϕ is the neutron flux. From the definition of G , the rate of production of fissile atoms is

$$\frac{dN_f}{dt} = GN_f \sigma_a \phi.$$

The doubling time is then given by $T_d = \ln 2 / G \sigma_a \phi$. For $G = 0.2$, $\sigma_a \sim 1\text{b}$ and $\phi \sim 3 \times 10^{15} \text{cm}^{-2} \text{s}^{-1}$, the doubling time is over three decades. With typical doubling times of the order of decades for fast breeder reactors, it is clear that a nuclear power program based on breeding requires forward planning.

Generation IV fast reactors

Of the six Generation IV reactor types under development (Stanculescu, 2021), three are fast reactors and two are being developed with thermal and fast alternatives. In most cases the emphasis in Gen IV fast reactors is on burning actinides rather than on breeding. At present there is little economic advantage in breeding, which coupled with concerns about safeguards, has reduced the urgency for breeding. However the demonstrated possibility of breeding gives an option for almost unlimited nuclear energy in future, should it be needed.

See Also: Physics of the Fission Chain Reaction in Thermal and Fast Reactors.

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Fissile and Fertile Fuels

Jennifer Shusterman, Department of Chemistry, Hunter College of the City University of New York, New York, NY, United States; and Graduate Center of the City University of New York, New York, NY, United States

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Introduction

Nuclear reactors generate electrical power from the energy produced in nuclear fission. Fission occurs in the fuel of the reactor, which is made up of heavy elements, such as uranium, thorium, or plutonium. Upon interaction with a neutron, certain isotopes of these elements will either fission or form heavier isotopes that will subsequently fission. All elements that are capable of being used as fissionable material in fission-based reactors are part of a series of elements known as the actinides. A characteristic of each of these elements is that they are all radioactive and have no stable isotopes. Despite thorium and uranium having no stable isotopes, there are isotopes of each of them that are naturally occurring and undergo radioactive decay. In this chapter, fundamentals of the actinide elements, discovery and properties of the natural decay series for uranium and thorium, the use of these elements in nuclear fuels, and the presence of higher Z actinides in irradiated fuel will be discussed.

Actinides

The actinides are the group of elements that make up the second row of the f-block of the periodic table (Fig. 1) and span $Z = 89$ to $Z = 103$. These elements decay by alpha or beta decay or spontaneous fission. While spontaneous fission does occur and is unique to the heavier elements, alpha and beta decay are still the dominant processes within this series. Beyond their ability to fission, all isotopes of actinide elements are radioactive. Thus, fuel for a fission reactor is inherently radioactive prior to irradiation. For the purposes of this encyclopedia, actinides up through fermium will be discussed, however only those up through curium are typically present in notable quantities in nuclear reactor fuels.

Basic properties

Chemical

While actinide elements, especially uranium and plutonium, are well known because of their nuclear properties, their chemical properties are also unique. Unlike the top row of the f-block elements, the lanthanides, the actinides exhibit distinctly different behavior based on whether they are early or late period elements (Fig. 1). The break in the series is typically taken to be at or near curium. The early actinides exhibit many oxidation states in aqueous solution, similar to transition metals. The later actinides, however, often are only stable in one or two oxidation states in aqueous conditions, typically found in their trivalent form. Similarly, the lanthanides, are most often found in their trivalent state in aqueous solution with only a few of them even having accessible divalent or tetravalent forms. As a result of the transition-metal like behavior of the early actinides, the elements up through plutonium were originally assigned spots on the periodic table in the d-block. The d-block, named for its valence d-orbital electrons, is the region of the main block of the periodic table that ranges from Group 3 through Group 12. It was Glenn T. Seaborg's Actinide Hypothesis which related the similar behavior of americium and curium to the lanthanides and suggestion that the elements

1 IA																		18 VIIIA																																																																											
1	1.0079	H	Hydrogen	2 IIA														5	10.811	B	Boron	6	12.011	C	Carbon	7	14.007	N	Nitrogen	8	15.999	O	Oxygen	9	18.998	F	Fluorine	10	20.180	Ne	Neon																																																				
3	6.941	Li	Lithium	4	9.0122	Be	Beryllium															13	26.982	Al	Aluminium	14	28.086	Si	Silicon	15	30.974	P	Phosphorus	16	32.065	S	Sulphur	17	35.453	Cl	Chlorine	18	39.948	Ar	Argon																																																
11	22.990	Na	Sodium	12	24.305	Mg	Magnesium															19	39.098	K	Potassium	20	40.078	Ca	Calcium	21	44.956	Sc	Scandium	22	47.867	Ti	Titanium	23	50.942	V	Vanadium	24	51.996	Cr	Chromium	25	54.938	Mn	Manganese	26	55.845	Fe	Iron	27	58.933	Co	Cobalt	28	58.693	Ni	Nickel	29	63.546	Cu	Copper	30	65.39	Zn	Zinc	31	69.723	Ga	Gallium	32	72.64	Ge	Germanium	33	74.922	As	Arsenic	34	78.96	Se	Selenium	35	79.904	Br	Bromine	36	83.8	Kr	Krypton
37	85.468	Rb	Rubidium	38	87.62	Sr	Strontium	39	88.906	Y	Yttrium	40	91.224	Zr	Zirconium	41	92.906	Nb	Niobium	42	95.94	Mo	Molybdenum	43	96	Tc	Technetium	44	101.07	Ru	Ruthenium	45	102.91	Rh	Rhodium	46	106.42	Pd	Palladium	47	107.87	Ag	Silver	48	112.41	Cd	Cadmium	49	114.82	In	Indium	50	118.71	Sn	Tin	51	121.76	Sb	Antimony	52	127.6	Te	Tellurium	53	126.9	I	Iodine	54	131.29	Xe	Xenon																						
55	132.91	Cs	Caesium	56	137.33	Ba	Barium	57	138.91	La	Lanthanum	72	178.49	Hf	Hafnium	73	180.95	Ta	Tantalum	74	183.84	W	Tungsten	75	186.21	Re	Rhenium	76	190.23	Os	Osmium	77	192.22	Ir	Iridium	78	195.08	Pt	Platinum	79	196.97	Au	Gold	80	200.59	Hg	Mercury	81	204.38	Tl	Thallium	82	207.2	Pb	Lead	83	208.98	Bi	Bismuth	84	209	Po	Polonium	85	210	At	Astatine	86	222	Rn	Radon																						
87	223	Fr	Francium	88	226	Ra	Radium	89	227	Ac	Actinium	104	(261)	Rf	Rutherfordium	105	(262)	Db	Dubnium	106	(266)	Sg	Seaborgium	107	(270)	Bh	Bohrium	108	(270)	Hs	Hassium	109	(278)	Mt	Moscovium	110	(281)	Ds	Darmstadtium	111	(281)	Rg	Roentgenium	112	(285)	Cn	Copernicium	113	(286)	Nh	Nihonium	114	(289)	Fl	Flerovium	115	(289)	Mc	Moscovium	116	(293)	Lv	Livermorium	117	(294)	Ts	Tennessine	118	(294)	Og	Oganesson																						
																		58	140.12	Ce	Cerium	59	140.91	Pr	Praseodymium	60	144.24	Nd	Neodymium	61	145	Pm	Promethium	62	150.36	Sm	Samarium	63	151.96	Eu	Europium	64	157.25	Gd	Gadolinium	65	158.93	Tb	Terbium	66	162.50	Dy	Dysprosium	67	164.93	Ho	Holmium	68	167.26	Er	Erbium	69	168.93	Tm	Thulium	70	173.04	Yb	Ytterbium	71	174.97	Lu	Lutetium																				
																		90	232.04	Th	Thorium	91	231.04	Pa	Protactinium	92	238.03	U	Uranium	93	237.05	Np	Neptunium	94	(244)	Pu	Plutonium	95	(243)	Am	Americium	96	(247)	Cm	Curium	97	(247)	Bk	Berkelium	98	(251)	Cf	Californium	99	(252)	Es	Einsteinium	100	(257)	Fm	Fermium	101	(258)	Md	Mendelevium	102	(259)	No	Nobelium	103	(262)	Lr	Lawrencium																				

Alkali Metal

Alkaline Earth Metal

Metal

Metalloid

Non-metal

Halogen

Noble Gas

Lanthanide

Actinide

Superheavy Element

Z

mass

Symbol

Name

Synthetic

Fig. 1 Periodic table of the elements. Adapted from Griffin I (2009) *Periodic Table of Chemical Elements*, <https://texample.net/tikz/examples/periodic-table-of-chemical-elements/>.

starting at thorium be moved to the f-block. It is because the oxidation states of the early actinides can be manipulated readily, however, that they can be purified sufficiently to be used as reactor fuel. Lanthanides and the later actinides are challenging to separate from their neighboring elements because of their chemical similarities, which make them difficult to purify at industrial scales.

Nuclear

Alpha emission (Previtali, 2021) becomes a dominant mode of decay for elements with $Z > 83$. For actinide elements, the expected decay modes are alpha, beta, and spontaneous fission. Spontaneous fission only occurs for elements with $Z > 90$, and is a relatively minor contributor to the overall decay processes for the longer lived actinide isotopes. However, it is important to note that spontaneous fission, though minor, contributes to the overall neutron inventory of the reactor and also results in fission product formation particularly for isotopes of curium and californium with larger spontaneous fission branches. Unlike particle-induced fission processes, spontaneous fission, like alpha decay, occurs because of quantum mechanical tunneling (Krane, 2021). Even for the actinide isotopes with spontaneous fission branches that are more substantial, their spontaneous fission half-life is relatively long compared to alpha or beta decay. For ^{252}Cf , which has a 3.1% spontaneous fission branch, the spontaneous fission half life is 85.6 years compared to the alpha decay half life of 2.73 years.

Production of the actinides

Thorium and uranium both have naturally occurring isotopes which were produced prior to the formation of the earth and exist at concentrations of 12 and 2 ppm, respectively, averaged throughout the earth's crust. Naturally occurring thorium is 100% ^{232}Th with a half-life of 1.40×10^{10} years. Naturally occurring uranium, however, is a mixture of three isotopes ^{234}U , ^{235}U , and ^{238}U with abundances of 0.0054%, 0.7204%, and 99.2742%, respectively. There have been measurements of minute quantities of primordial ^{244}Pu , as well (Hoffman et al., 1971; Wallner et al., 2015). The primordial actinide isotopes are formed through the rapid neutron capture r-process which is thought to occur in the core collapse of supernovae or in neutron star-neutron star mergers. In the r-process, lower Z nuclides, undergo a series of neutron capture reactions and beta decays to produce about half of all isotopes heavier than iron (Garcia, 2021).

The naturally occurring actinide elements, thorium and uranium, were discovered decades and over a century, respectively, prior to those that are artificially produced. Uranium was discovered as a new element when Martin Heinrich Klaproth extracted it in the form of UO_2 from the uranium oxide mineral Pitchblende in 1789. It was not until 1841 when Eugène-Melchior Péligot successfully reduced UO_2 to uranium metal that it was found that Klaproth's material was in fact the oxide not the metal (Morss, 2006). Uranium was first given an atomic weight of 240 by Mendeleev in 1872, which was then measured by Zimmerman in 1882, though both of these values were overestimates based on the mass known today of 238.02891 (Meija et al., 2016). Thorium was discovered by Jöns Jacob Berzelius in 1828 when studying a thorium silicate mineral sample from Norway.

As with any other element found in geological deposits, uranium and thorium concentrations are increased in certain mineral ore bodies. Both elements tend to associate with oxygen, forming oxide and oxy-hydroxides species. There are many uranium and thorium bearing minerals, but two of the most well-known are uraninite (pitchblende) and monazite. Monazite, a rare earth phosphate mineral, is one of the most thorium-rich naturally occurring forms with Th exceeding 10^5 ppm in some samples. Thorium can replace the cerium monazite, as they can both exist as tetravalent cations and have similar ionic radii. Uraninite is composed primarily of UO_2 but has varying concentration of U_3O_8 depending on the environment in which it was formed. As a result of their decay, daughters of thorium and uranium (even pockets of gaseous radon) all the way down through lead can be found in their respective mineral deposits.

Unlike these select naturally occurring isotopes, the majority of actinides are artificially produced via nuclear reactions. The transuranic actinide elements were all discovered in a span of 21 years starting in 1940. The elements neptunium, plutonium, americium, curium, berkelium, and californium were all synthesized by the scientists at Berkeley, California, using the 60-in. cyclotron invented by Ernest Lawrence, a Berkeley physicist, or neutron irradiation at the Metallurgical Laboratory between 1940 and 1950 (Cameron, 2021). In 1940, McMillan and Abelson identified the 2.3 day activity formed in the neutron bombardment of ^{238}U as the new element 93, now known as Neptunium (Np) (McMillan and Abelson, 1940). A ^{238}U target was irradiated with neutrons at the 60-in. cyclotron and McMillan and Abelson observed that (1) the 2.3 day activity grew in from the decay of a 23 m activity, and (2) the 2.3 day activity exhibited chemistry different from known elements. At the time, element 93 was thought to be homologous (in the same periodic group) as Re, but they established that the chemistry was quite different with higher oxidation states behaving more like uranium and lower oxidation states like thorium. Seaborg, Kennedy, Mcmillan, and Wahl soon after, in 1941, reported the discovery of element 94, Plutonium (Pu). They performed several irradiations of ^{238}U with deuterons at the 60-in. cyclotron in the $^{238}\text{U}(\text{d},\text{n})^{239}\text{Pu}$ reaction, and then using the same previously established separations, isolated element 93. After awaiting the decay of the 2.3 day ^{239}Pu , alpha activity was detected. Through chemical exploration of this residual alpha emitting product, they established that it was chemically different from known alpha emitters (Seaborg et al., 1946a,b) and was presumed to be the ^{239}Pu decay daughter ^{239}Am . In the same year, Kennedy, Seaborg, Emilio Segre, and Wahl established the formation of ^{239}Pu via the reaction and decay: $^{238}\text{U}(\text{n},\gamma)^{239}\text{U} \rightarrow ^{239}\text{Pu} \rightarrow ^{239}\text{Pu}$ and also reported that ^{239}Pu undergoes neutron-induced fission (Kennedy et al., 1946). Interestingly, element 96 (Curium, Cm) was discovered in 1944, prior to element 95 (Americium, Am) in late 1944. Using the $^{239}\text{Pu}(\alpha,\text{n})^{242}\text{Cm}$ reaction, Seaborg, James, and Ghiorso, produced element 96 at the 60-in. cyclotron

(Seaborg et al., 1948a). The chemical processing and isolation, however, was performed at the Met Lab. Not long after, Seaborg, James, and Morgan produced, isolated, and characterized $^{241}\text{95}$ via reactor irradiation and subsequent separation at the Met Lab using the following sequence of reactions and decays (Seaborg et al., 1948b): $^{239}\text{Pu}(n,\gamma)^{240}\text{Pu}(n,\gamma)^{241}\text{Pu} \rightarrow ^{241}\text{95}$.

Soon after their discoveries, Am and Cm were used as target materials to explore production of heavier actinides. Due to challenges associated with previously unknown alpha decay systematics, low Am and Cm mass availability, radioactive doses from the targets, and complex trivalent actinide separations, it was not until 1950 that the discovery of element 97, Berkelium (Bk) was reported by Thompson, Ghiorso, and Seaborg. Using the Crocker Laboratory 60-in. cyclotron, Bk was produced via the reaction $^{241}\text{Am}(\alpha, 2n)^{243}\text{Bk}$ and subsequently isolated from the target material and byproduct Cm and trivalent lanthanide fission products via careful radiochemical separations (Thompson et al., 2008). With many of the production and separation challenges addressed in the Bk discovery, the isolation of element 98, californium (Cf), was reported soon after Bk in 1950 by Thompson et al. (1950). Similar to the Bk production, Cf was made in bombardment of Cm with α particles at the 60-in. cyclotron via the reaction $^{242}\text{Cm}(\alpha, xn)^{246-x}\text{Cf}$. The isotope of Cf, with a 45 min half-life was proposed to be ^{244}Cf in the original discovery paper but was later identified as ^{245}Cf .

Elements 99 and 100, einsteinium (Es) and fermium (Fm), respectively, were discovered in 1952 in the analysis of debris from Ivy Mike thermonuclear test (Ghiorso et al., 1955b). Elements 101, 102, and 103, Mendelevium (Md), nobelium (No), and lawrencium (Lr), respectively, were the first elements produced atom-at-a-time between 1955 and 1961 in charged particle reactions on targets of einsteinium, curium, and californium, respectively. For element 101, a target of ^{253}Es was bombarded with α particles at the 60-in. cyclotron at Crocker Laboratory, producing a new isotope that had chemical properties indicating that it was heavier than element 100 and decay properties expected to align with $^{256}\text{101}$ (Ghiorso et al., 1955a). There were efforts in the United States, Russia, and Sweden to produce nobelium. At the Berkeley Heavy Ion Linear Accelerator (HILAC), No was produced by bombarding Cm targets with C beams in the reaction $^{246}\text{Cm}(^{12}\text{C}, 4n)^{254}\text{102}$ (Ghiorso et al., 1958). At the 150-cm Cyclotron at the Institute of Atomic Energy in Moscow, $^{239,241}\text{Pu}$ targets were bombarded with ^{16}O to produce isotopes of No (Flerov et al., 1992). Lawrencium was produced by Ghiorso et al. also at the HILAC by irradiating a mixed isotopic Cf source with ^{10}B and ^{11}B , producing an isotope that emitted an 8.5 MeV α 's that were later attributed to ^{258}Lr (Ghiorso et al., 1961). Since that time, 15 more elements have been discovered at several laboratories around the world using a variety of reactions; these are the transactinide elements ranging from $Z = 104$ to $Z = 118$.

Fundamental processes in nuclear reactor fuels

Uranium is the most commonly used actinide element for nuclear fuel because it is the only element that has a naturally occurring fissile isotope. The nuclear reactor relies upon a careful neutron balance to sustain the nuclear chain reaction. This balance requires not only a knowledge of the number of overall neutrons, but the relative population with different energies. Fission neutrons, that is, neutrons that are emitted from a fission reaction, have a mean energy of approximately 2 MeV and an energy-dependent distribution that follows the Watt spectrum (Klay, 2021). These neutrons are considered fast because their velocity is very high. Fast neutrons have a significantly lower probability to induce fission in fissile isotopes such as ^{235}U , than thermal neutrons—neutrons that have slowed-down to thermal energies—in the vicinity of 0.025 eV at room temperature. At thermal energies, there are two dominant interaction processes that compete: neutron-induced fission and radiative capture. Both of these processes also compete with radioactive decay, which is independent of the neutron field.

Decay

Despite the early discoveries of thorium and uranium, their radioactive properties were not established for several more decades. In 1896, Henri Becquerel found that uranium-bearing materials caused blackening of photographic plates that was similar to that observed by Roentgen with X-rays. He established that this was a form of radiation. Following his methodology, in 1898, Gerhard Carl Schmidt reported that thorium-bearing materials resulted in the same effect on photographic plates and electrometers observed by Becquerel with uranium. In the same year, independent of Schmidt, Marie Skłodowska Curie measured the current generated by various compounds, including those that contained uranium and thorium, in a parallel plate condenser. Similar to Becquerel and Schmidt, she also employed photographic plates to study the activity of these materials. The two particularly notable findings from this work were that (1) thorium emitted radiation spontaneously, like uranium, and (2) uranium minerals, pitchblende and chalcocite, were more active than pure uranium indicating that a more active species than uranium was present (Curie and Lippmann, 1898). Curie's latter observation prompted her and other chemists to perform elemental separations on these minerals, from which she isolated polonium and radium.

Further investigations by Ernest Rutherford, F. Soddy, and F.E. Dorn led to the conclusion that when elements decayed via emission of radioactivity that they were converted into different elements (Rutherford and Soddy, 1902). With both chemical and radioactive properties, 40 species were isolated and then grouped into three series based on their decays. Two of these series started with uranium and one with thorium, while all terminated in stable lead. Across these series, however, there were species that had the same chemical properties but different radioactive properties. While it was not yet known what caused these differences, Soddy named these species that exhibit the same chemical properties of a given element but have different radioactive characteristics

isotopes. Once grouped by elements and isotopes, it became clear that the 40 species were actually just isotopes of the 11 elements present in the decay of ^{238}U , that make up these three decay series.

It was not until the atomic mass of uranium was measured more accurately to be closer to 238 amu that it became clear that it was the isotope ^{238}U that started the uranium series. With this information and the rule established by Fajans and Soddy that described an α -decay as $\Delta Z = -2$ and $\Delta A = -4$ while a β decay was a transition with $\Delta Z = +1$ and $\Delta A = 0$ (Soddy, 1923; Previtali, 2021) the decay series could be filled out (Fig. 2).

There are three naturally occurring decay chains: (1) uranium, (2) thorium, and (3) actinium series, each named after the naturally occurring parent element. The fourth series is the neptunium series, however, any naturally occurring ^{237}Np would have long since decayed. The mass number of all members of each of these four series can be describes as $4n + x$, where $x = 0, 1, 2$, or 3 , for the thorium, neptunium, uranium, and actinium series respectively. This classification was used to divide groups of nuclei as all mass numbers are either even divisible by 4 or have a remainder of 1, 2, or 3. The thorium series ($A = 4n$) is named for its parent ^{232}Th , the only naturally occurring isotope of thorium. This chain terminates in the formation of stable

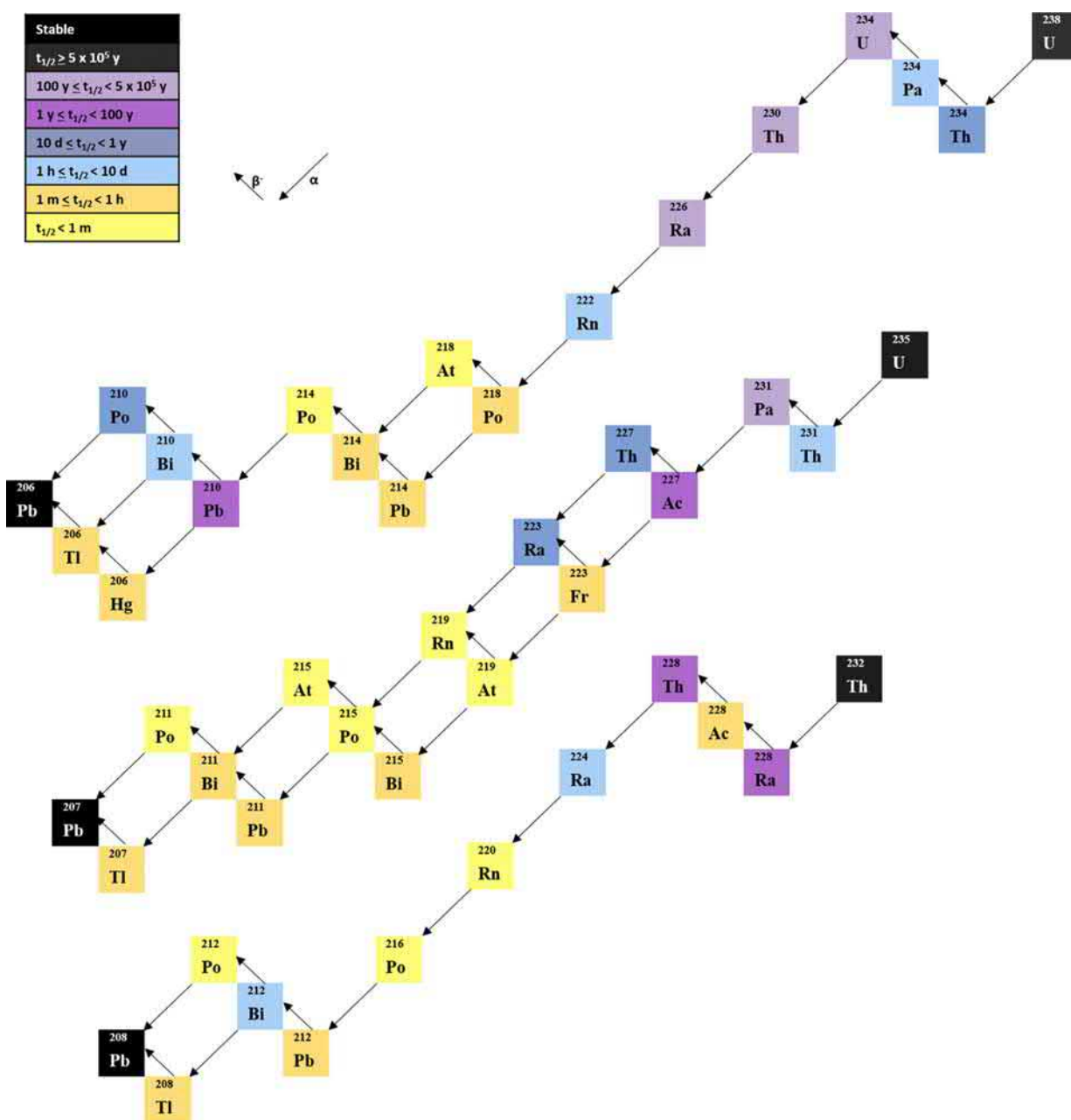


Fig. 2 The decay chains for the uranium, actinium, and thorium series from top to bottom, respectively. Each isotope is color-coded based on their half-life ($t_{1/2}$). Each chain begins with a long-lived, naturally occurring actinide isotopes and terminates in a stable isotope of lead.

^{208}Pb . The neptunium series ($A = 4n + 1$), while not naturally occurring, is important to consider particularly for nuclear energy. The ^{237}Np parent is one of the long lived minor actinides present in used nuclear fuel, and its granddaughter, ^{233}U the fissile isotope of uranium formed in ^{232}Th breeder reactors. The original terminus in the neptunium series was ^{209}Bi , which while generally treated as stable, actually is radioactive with a half life of 1.9×10^{19} years, and so the revised end member is ^{205}Tl . The uranium series ($A = 4n + 2$) is named for its parent and the most abundant uranium isotope, ^{238}U . Between the ^{238}U and the stable ^{206}Pb endpoint, 8 α decays and 6 β decays occur. One of these intermediary isotopes is ^{226}Ra , which is why this decay chain is sometimes referred to as the radium series. Unlike the other series, the actinium series ($A = 4n + 3$) is not named after the first member of its decay chain, but rather its great-granddaughter ^{227}Ac . The parent of this series is the fissile uranium isotope, ^{235}U , and its terminus is ^{207}Pb .

Fission

In particle-induced fission processes, a heavy nucleus absorbs a particle such as a neutron or proton, and this caused the nucleus to deform into a peanut-like shape. The two lobes of the deformed nucleus maintain the positive charge of the nucleus and thus repel each other causing greater deformation. Ultimately, the repulsion leads to the splitting (fission) of the deformed nucleus to form two lighter nuclei (fission products). For fission to be energetically favorable, the binding energy of the fission products must be greater than that of the originally heavy nucleus. In this process, depending on the fissile material and incident neutron energy, 2.5 neutrons are emitted on average and the total energy release is approximately 200 MeV. This energy is split between the kinetic energy of the fission products and the neutrons, as well as prompt γ -rays emitted.

Fission is most favorable for nuclides with an even number of protons and neutrons. Spontaneous fission occurs in even-even nuclei at rates in excess of 10^3 times those of even-odd and odd-even, and 10^5 times those for odd-odd nuclei (Hyde, 1964). Similarly, neutron-induced fission with thermal neutrons is observed in nuclei with an odd number of neutrons which reach a fissionable configuration following neutron capture.

In addition to fission, another source of neutron generation in reactors is through (α , xn) reactions.¹⁴ These result from alpha particles emitted through alpha decay of the actinides interacting primarily with oxygen or fluorine nuclei in oxide or fluoride based fuels, respectively. The neutron production rate can be as high as about 4×10^6 n/s/g of actinide decaying via alpha emission. Alpha particles with higher energies result in higher neutron production rates as they are able to overcome more reaction coulomb barriers. Based on the Geiger Nutall law, alpha particle energy is inversely proportional to alpha decay half-life. Thus, as shorter-lived, higher Z actinides emit higher energy alpha particles, fuels that generate larger quantities of minor actinides can have more substantial neutron generation via this mechanism.

Capture

For heavy nuclei, the nuclear process which competes with fission and decay is neutron capture. In this process, a nucleus with mass A absorbs most often a thermal neutron and emits γ -rays (radiative capture), resulting in the formation of the isotope of the same element with a mass of $A + 1$. The radiative capture products can then go on to decay, capture neutrons, or fission. It is by this process that elements heavier than the original fuel are formed. After radiative capture, the product can β -decay, converting a neutron into a proton, and in turn forming the next element on the periodic table. The longer the fuel is in a reactor, the greater abundance of elements heavier than the fuel as well as the higher in Z that can be observed. Independent of whether the initial fuel is Th, U, or Pu, the transuranium actinides that can be produced are Np through Fm (Fig. 3). Of these actinides, Np, Pu, Am, and Cm are produced in the greatest abundance and considered during the nuclear fuel cycle. The heaviest isotope that can be produced through successful neutron capture and beta decays is ^{257}Fm (Silva, 2010). The heavier isotopes of Fm, $^{258-260}\text{Fm}$ decay nearly 100% of the time by spontaneous fission and have short half lives, with the longest lived being ^{259}Fm with a half-life of 1.5 s. Therefore, isotopes heavier than ^{257}Fm are not believed to form in a nuclear reactor.

For a given actinide isotope, the relative probability of transmutation via neutron capture or fission can be quantified via cross sections for each of these mechanisms. Once the neutron flux is considered, the competition of these modes of transmutation with radioactive decay can be determined. Both neutron capture and neutron-induced fission cross sections are dependent on the energy of the incoming neutron. The probability of a neutron interacting with a given nucleus is greater for slower neutrons, so neutron capture cross sections are highest for slow neutrons, typically of thermal energy (0.025 eV). For fissile nuclides, neutron-induced fission is the dominant form of transmutation in the presence of thermal neutrons. As evident in Table 1 their (n,f) exceed their (n, γ) cross sections by a factor of 2.5 to almost 30. The fission-to-capture ratio is generally higher for fission neutrons than for thermal neutrons. This, together with the fact that the number of neutrons emitted is larger for fast than for thermal fission, enables fast spectrum reactors to breed more fissile fuel than they consume. Each of these fissile nuclides used are long lived with the shortest half-life being that of ^{241}Pu at 14.329 years. These long half-lives ultimately mean that the decay process is not a major contributor to fissile nuclide transmutation in a reactor.

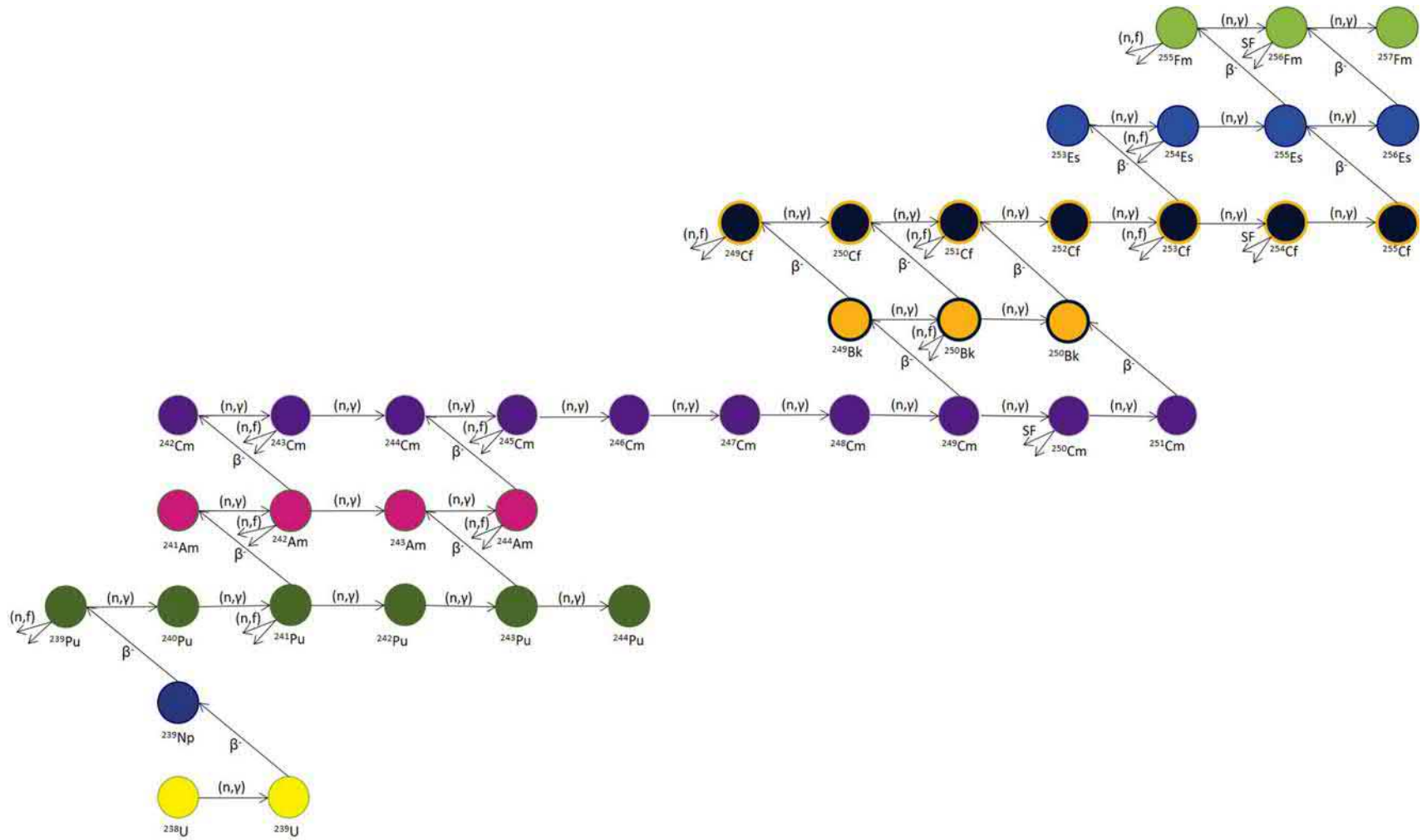


Fig. 3 Formation of transplutonium isotopes from uranium or plutonium fuels with thermal neutrons.

Table 1 Decay, neutron induced reaction cross sections, and nu-bar values for a selection of uranium and plutonium isotopes.

Nuclide	Half life (years) ^a	Decay % ^a	σ_f , b (thermal neutrons) ^b	ν_f (thermal, prompt)	σ_f , b (E_n = avg. fission spectrum) ^c	ν_f (E_n = avg. fission spectrum)	σ_γ , b (E_n = thermal) ^b	σ_γ , b (E_n = avg. fission spectrum) ^c
²³¹ U	1.15×10^{-2}	ϵ 100.00 α 4.0×10^{-3}	250.1 ^c		2.194		20.01 ^c	0.1035
²³² U	68.9	α 100.00 ²⁴ Ne: 9×10^{-10} SF 3×10^{-12}	76.76		2.063		75.19	0.1213
²³³ U	1.592×10^5	α 100.00 SF $< 6 \times 10^{-11}$	531.3	2.4968 ^d	1.908		45.26	0.05896
²³⁴ U	2.455×10^5	α 100.00 SF 1.6×10^{-9} Mg: 1×10^{-11} Ne: 9×10^{-12}	6.709×10^{-2}		1.222		100.9	0.1082
²³⁵ U	7.04×10^8	α 100.00 SF $< 7.0 \times 10^{-9}$	585	2.4208 ^e	1.218		98.69	0.08741
²³⁶ U	2.342×10^7	α 100.00 SF $< 9.4 \times 10^{-8}$	4.710×10^{-2}		0.5792		5.133	0.1075
²³⁷ U	1.85×10^{-2}	β^- 100	0.4165		0.5945		475.4	0.07336
²³⁸ U	4.468×10^9	α 100.00 SF $< 5.4 \times 10^{-5}$	1.679×10^{-5}	2.3468 ^e	0.306	2.819 ^f	2.683	0.07016
²³⁹ U	4.46×10^{-5}	β^- 100	14.11				20.57	
²⁴⁰ U	1.61×10^{-3}	β^- 100	1.079×10^{-3}				19.16	
²³⁶ Pu	2.858	α 100.00 SF $< 1.9 \times 10^{-7}$	164.8	2.9023 ^e	2.359		31.23	0.1043
²³⁷ Pu	0.125	ϵ 100.00 α 4.2×10^{-3}	2102	2.8876 ^e	2.432		540.7	0.06934
²³⁸ Pu	87.7	α 100.00 SF $< 1.9 \times 10^{-7}$	17.00	2.8699 ^e	1.968	3.00 ^g	560.8	0.1138
²³⁹ Pu	2.411×10^4	α 100.00 SF $< 3 \times 10^{-10}$	747.9	2.8836 ^d	1.802	3.17 ^{h, i}	270.7	0.05261
²⁴⁰ Pu	6561	α 100.00 SF $< 5.7 \times 10^{-6}$	6.404×10^{-2}	2.8863 ^e	1.328	3.086 ^g	287.5	0.08154
²⁴¹ Pu	14.329	β 100.00 α 2.5×10^{-3} SF $< 2 \times 10^{-14}$	1012	2.9198 ^e	1.626	3.17 ^{i, k}	363.0	0.07197
²⁴² Pu	3.75×10^5	α 100.00 SF 5.5×10^{-4}	1.042×10^{-3}	2.7901	1.151	3.189 ^g	19.16	0.09818
²⁴³ Pu	5.66×10^{-4}	β^- 100	181.4				88.13	
²⁴⁴ Pu	8.00×10^7	α 99.880 SF 0.12	1.715×10^{-3} ^c	2.79 ^e	1.038		1.83	0.05533

^aNNDC (2020).^bChadwick et al. (2011).^cShibata et al. (2011).^dBadikov et al. (2007).^eWright et al. (2015).^fChadwick et al. (2006).^gKoning et al. (2006).^h E_n 1.25–175 MeV.ⁱWang et al. (2019).^j E_n = 1.5 MeV.^kAkindele (2018) and Akindele et al. (2019).

Fissionable fuels

Nuclear reactors powered by nuclear fission all rely upon fissionable fuels. A fissionable material is one that fissions upon absorption of either a slow or fast neutron. Within this set of materials, there are two subsets to consider: fissile and fertile materials. A fissile element is one that fissions only upon absorption of thermal neutron. Alternatively, fertile materials do not fission after thermal neutron absorption, but can be transmuted to fissile elements through neutron capture and subsequent beta decay.

Fissile vs. fertile fuels

^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu are all fissile isotopes, fissioning as a result of thermal neutron absorption. It is worth noting that all of these nuclei have an odd number of neutrons; those nuclei that have an even number of neutrons require higher neutron energies to fission. Of these isotopes, only ^{235}U is naturally occurring, while the other are produced from neutron capture and beta decay of fertile isotopes. The energy dependent cross sections for neutron induced fission and radiative capture are shown in Fig. 5. The higher cross sections for radiative capture compared to neutron induced fission at most energies are evident for the nuclei with an even number of neutrons, ^{232}Th and ^{238}U (Fig. 5A and D, respectively). This compares to the fissile nuclei which all have an odd number of neutrons, ^{233}U , ^{235}U , and ^{239}Pu , which have higher neutron induced fission cross sections at all energies compared to radiative capture (Figs. 5B, C, and E, respectively).

To sustain a nuclear fission chain reaction, required for power generation, the neutrons produced during fission must have at least the energy needed to induce a subsequent (n,f) reaction. This is the reason why ^{238}U alone cannot maintain a chain reaction. The average 2.5 neutrons produced in fission, prior to moderation, have a median energy of 0.7 MeV, however the threshold energy for ^{238}U fission is 1 MeV. The number of neutrons release per fission is isotope and incident neutron energy dependent (Table 1). Fissionable and fertile fuel must be enriched or designed to build up an inventory of fissile material to maintain the neutron population for a chain reaction.

All existing fuel compositions contain both fissile and fertile material. As the fissile nuclei split and release neutrons, portions of the fertile material capture those moderated fission neutrons and beta decay to produce new fissile isotopes. In uranium based fuel, for example, a fraction of the higher energy neutrons produced in ^{235}U fission can go on to fission ^{238}U . Alternatively, the ^{238}U captures the thermal neutrons to form ^{239}U which undergoes two sequential beta decays to form the fissile nuclide ^{239}Pu (Fig. 4). Now that ^{239}Pu can go on to either fission or capture a neutron to produce ^{240}Pu which has a half-life of 6561 years and will capture another neutron to form ^{241}Pu , also fissile. Ultimately, after a period of burn up, uranium based fuels can contain three of these four fissile isotopes. As ^{233}U is not naturally occurring, it is not used in the fresh fuel matrices. Rather, it is produced in a ^{232}Th -based breeder reactor. The ^{232}Th captures thermal neutrons to form ^{233}Th which forms ^{233}U through two sequential beta decays (Fig. 4), with a ^{233}Pa intermediate.

Plutonium and MA

After an irradiation cycle of between 44 and 60 GWd/ton, used nuclear fuel (UNF) is still mostly uranium ($\sim 95\%$), specifically ^{238}U (Glatz, 2021). Even after 10 years of decay after removal of the fuel from the reactor, fission products are the next most abundance species at about 4% by mass, while plutonium makes up approximately 1% and the minor actinides about 0.2% (Salvatores and Palmiotti, 2011). With a combined inventory of less than 1.5 kg/MT (MT: metric ton) fuel 10 years post end of irradiation, the minor actinides make up a relatively small mass but have notable nuclear contributions that can make them problematic. Neptunium, for instance, exists primarily in fuel at ^{237}Np which has a half life of 2.144×10^6 years, making it a long term hazard in terms of waste storage. Not as long lived are the isotopes of Am, $^{241,242,242m,243}\text{Am}$, that each have intense gamma ray emissions and contribute to long-term radiotoxicity. The major isotopes of curium that are present, namely ^{242}Cm and ^{244}Cm are both relatively short lived compared to americium and neptunium isotopes of concern. Heavier isotopes of Cm that are present in smaller quantities can impact the neutron inventory as a result of spontaneous fission.

Despite their low concentration compared to uranium and plutonium isotopes, americium and curium can have technical handling challenges as a result of their high gamma ray intensities and neutron emissions requiring shielding. In addition to producing fissile isotopes, such as ^{239}Pu and ^{241}Pu , isotopes with larger neutron capture cross sections can build up and act as a burnable poison in the fuel (Table 2). Such isotopes include ^{241}Am , ^{244}Am , and ^{238}Pu which all have cross sections on

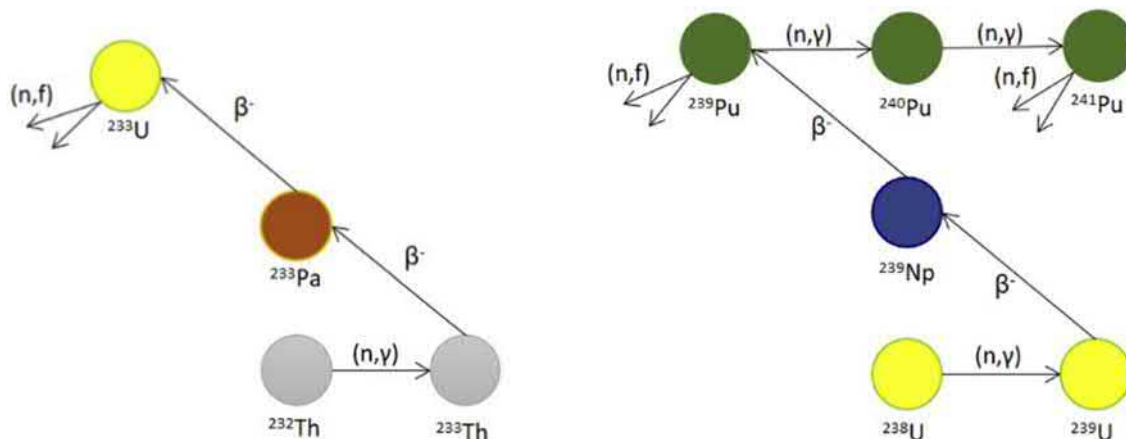


Fig. 4 Transmutation chains for the formation of ^{233}U from ^{232}Th (left) and ^{239}Pu from ^{238}U (right).

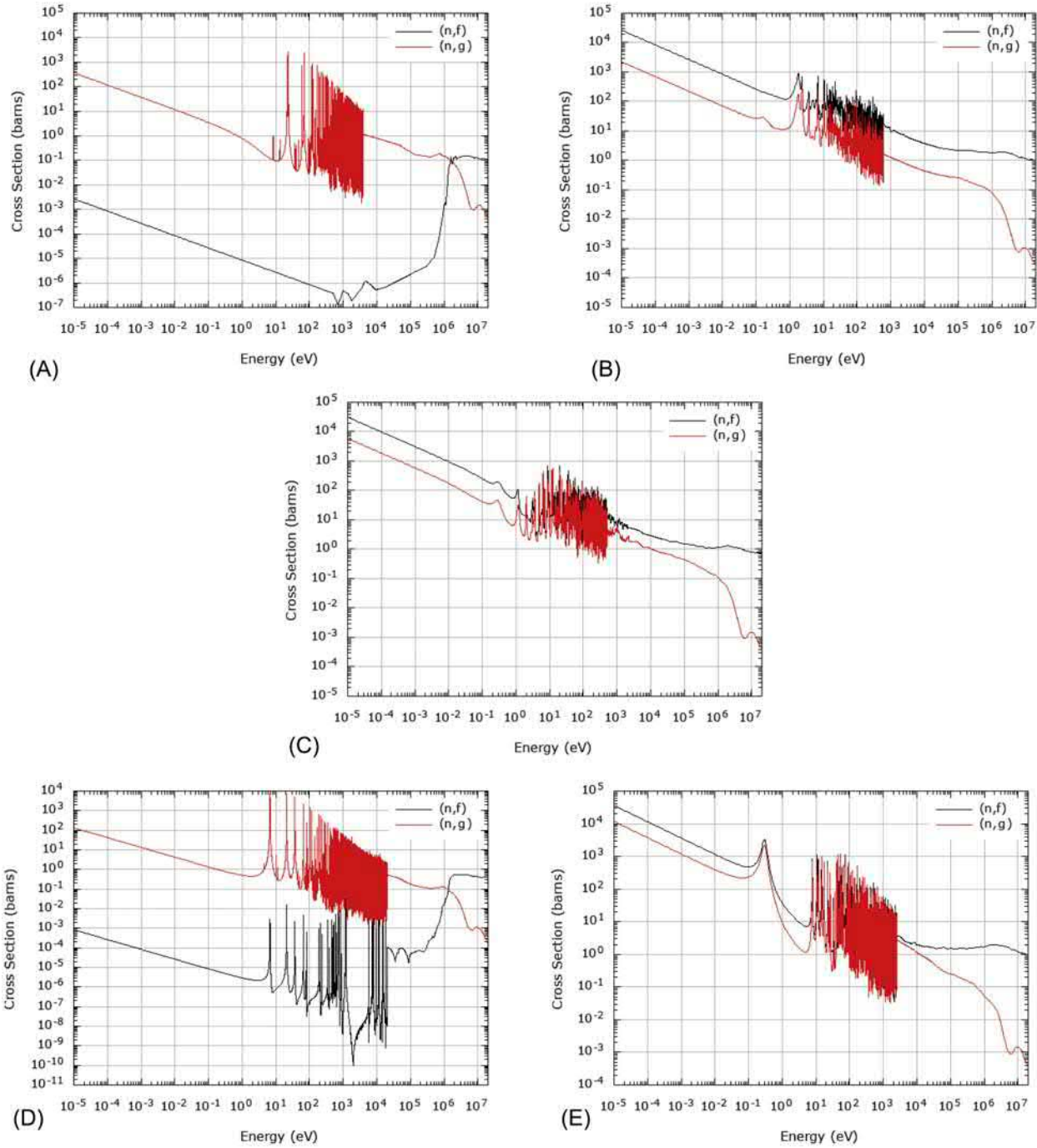


Fig. 5 Neutron induced fission and radiative capture cross sections for (A) ^{232}Th , (B) ^{233}U , (C) ^{235}U , (D) ^{238}U , and (E) ^{239}Pu at 300 K from JENDL 4.0 (Shibata et al., 2011).

the order of 600 b. High burnup fuel can go on to form isotopes of Bk and Cf, and both ^{250}Cf and ^{251}Cf have neutron capture cross sections of greater than 1500 b. The proportion of minor actinides in UNF is a function of initial fuel matrix, neutron flux, and irradiation time. In general, thorium based UNF has lower minor actinide content than uranium or mixed oxide ($\text{UO}_2\text{-PuO}_2$ blend) fuels because more neutron captures and beta decays would be required for thorium to transmute to these elements. The exception to this is for thorium breeder reactors using minor actinides as the driver material; prior to reaching equilibrium after about 150 years, there is a greater proportion of minor actinides than would form in the fuel (Heuer et al., 2014).

Table 2 Decay and neutron induced reaction cross sections for a selection of americium and curium isotopes.

Nuclide	Half life (years) (<i>NNDC</i> , 2020)	Decay (%) (<i>NNDC</i> , 2020)	σ_f , b (thermal neutrons) (<i>Chadwick</i> <i>et al.</i> , 2011)	ν_f (thermal, prompt) (<i>Chadwick</i> <i>et al.</i> , 2011)	σ_f , b (E_n = average fission spectrum) (<i>Shibata et al.</i> , 2011)	ν_f (E_n = 1 MeV) (<i>Wright</i> <i>et al.</i> , 2015)	σ_γ , b (E_n = thermal) (<i>Chadwick</i> <i>et al.</i> , 2011) ^a	σ_γ , b (E_n = average fission spectrum) (<i>Shibata et al.</i> , 2011)
²⁴¹ Am	432.6	α 100.00 SF: 4×10^{-10}	3.139	3.055	1.395		618.8	0.2386
²⁴² Am	1.83×10^{-3}	β^- 82.70 ϵ 17.30	2095		1.833		219.0	0.03607
²⁴³ Am	7364	α 100.00 SF: 3.7×10^{-9}	0.07393	3.2649	1.081		75.11	0.1859
²⁴² Cm	0.446	α 100.00 SF: 6.2×10^{-6} ³⁴ Si: 1×10^{-14}	3.02	3.4519	1.775	3.4366	16.87	0.172
²⁴³ Cm	29.1	α 99.71 ϵ 0.29 SF: 5.3×10^{-9}	613.5	3.4290	2.432		130.5	0.06773
²⁴⁴ Cm	18.1	α 100.00 SF: 1.4×10^{-4}	1.037	3.0000	1.733	3.5879	15.10	0.1159
²⁴⁵ Cm	8423	α 100.00 SF: 6.1×10^{-7}	2141	3.5900	1.754		358.9	0.06787

Summary

Within fissionable fuels in nuclear reactors, the three primary mechanisms that transmute the actinide elements are radioactive decay, radiative capture, and fission. The fuels are blends of fissile and fertile elements, in which the proportions rely upon the type of reactor design and use. Isotopes of neptunium and transplutonium isotopes that make up the minor actinides formed in nuclear fuel can not only impact the performance of the reactor but also post-irradiation handling of the fuel and its long-term storage.

See Also: Physics of the Fission Chain Reaction in Thermal and Fast Reactors.

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Manhattan Project

B. Cameron Reed, Department of Physics (Emeritus), Alma College, Alma, MI, United States

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Glossary

Cross-section Measure of the effective cross-sectional area a nucleus presents to an incoming particle in order that a particular reaction is induced. Cross-section depends on the type of struck nucleus and bombarding particle, as well as the energy of the latter. Usually measured in “barns” (bn; $1 \text{ bn} = 10^{-28} \text{ m}^2$).

Critical mass Mass of a fissile material that must be gathered in one place in order to sustain a neutron-mediated chain reaction. A subcritical mass is less than this amount, and a supercritical mass is greater.

Neutron flux Measure of the number of neutrons passing through a given area per unit time within a reactor.

Pile Historical term for a nuclear reactor.

Introduction

The energy liberated by nuclear fission, about 200 Million electron-Volts (MeV) per event, is equivalent, per kilogram, to the energy released by the detonation of some 17,000 metric tons of conventional explosive. In early 1939, the verification that each fission of uranium was accompanied by the prompt release of 2–3 fast neutrons pointed to the possibility of a powerful neutron-mediated chain-reaction explosion. Within weeks of the discovery of fission, attention turned to investigating the conditions necessary for such an explosion, and understanding the physics of the fission process.

The critical mass, and the possibility of nuclear weapons

In a sample of fissile material, secondary neutrons need not cause subsequent fissions: some might be captured in non-fission reactions, and some will inevitably escape, particularly if the sample is small. As the size of the sample increases, so does the probability that a neutron will cause a fission before leaking out. A minimal critical size, and hence a corresponding critical mass of material, has to be assembled in one place in order to achieve a self-sustaining chain-reaction which will continue until the material blows itself apart.

The travel and interactions of neutrons within a reactor or bomb core can be analyzed with a simplified time-dependent diffusion theory with a single-energy group of neutrons. The first analysis of this was published by Francis Perrin in early 1939, who considered natural uranium dioxide, that is, with uranium content 0.71% of the fissile isotope ^{235}U . With very rough estimates for some of the relevant reaction cross-sections, he arrived at a critical mass of about 40,000 kg, but remarked that this could be reduced by surrounding the material with a tamper (nowadays commonly referred to as a “reflector”) to reflect escaping neutrons back into the fissile material. Perrin’s result had no relevance for a bomb, which requires essentially pure ^{235}U , but he did establish the relevant physics and introduced the notion of tampering (reflecting). Rudolf Peierls subsequently refined Perrin’s analysis to develop explicit formulae for estimating the critical mass, but did not evaluate any numbers since cross-sections were still very uncertain.

Two results of diffusion theory are key to understanding criticality.

First, for a sphere of fissile material of radius r and density ρ , criticality will be obtained so long as the product $r\rho$ exceeds a definite value, which depends on the number of neutrons emitted per fission and the fission, capture and scattering cross-sections of the material. For material of normal density, the corresponding radius and mass are usually referred to as the bare critical radius and mass; for ^{235}U and ^{239}Pu , these masses are ~ 46 and 17 kg. For a sphere of mass M , this condition is equivalent to demanding that M/r^2 exceed a definite value. This shows that a bomb containing only a single critical mass of fissile material will yield a very inefficient explosion: As soon as the chain reaction begins, the core will heat up and expand, and M/r^2 will drop below the value needed to sustain criticality. To avoid this, it is necessary to provide more than a bare-critical mass, a so-called supercritical mass. But there is a loophole: If a subcritical lump of material can be crushed to a radius less than that corresponding to its normal density before the fission chain reaction is triggered, then M/r^2 will exceed the criticality condition. This is the premise of an implosion bomb, which makes more efficient use of available fissile material.

Second, in a nuclear explosion, the fission rate in a supercritical chain reaction grows exponentially with time with an e-folding time essentially equal to the prompt neutron generation time (Klay, 2021). The prompt neutron generation time in a fast spectrum core is on the order of 10^{-8} s whereas in a thermal spectrum core, like those used for most commercial nuclear power reactors, it is several orders of magnitudes longer. For an appreciable prompt energy yield, a bomb must utilize fast neutrons; there would be no point in making a slow-neutron bomb. To utilize a fast-neutron reaction, it is necessary to isolate ^{235}U from ^{238}U ; this is discussed in the following section.

The first calculation of the energy that might be yielded by a nuclear weapon was prepared by Otto Frisch and Rudolf Peierls at the University of Birmingham, England, in March 1940, in what became known as the Frisch-Peierls Memorandum. Frisch was aware of a prediction by Niels Bohr that slow-neutron fission was likely caused by the 235 isotope of uranium, and considered whether it would be possible to make an explosive chain reaction with that isotope by using fast neutrons. With a rough guess for the fission cross-section and Peierls' critical-mass formula, Frisch estimated a mass on the order of a pound; this was low due to overestimating the fission cross-section. Taking his calculation to Peierls, they estimated that the reaction might liberate energy equivalent to thousands of tons of ordinary explosive.

Frisch and Peierls summarized their conclusions in a secret memorandum, which they showed to their department chair, Marcus Oliphant. While this was a briefing intended for government officials, it described the concept of critical mass, how such a weapon might operate, and some of the associated strategic implications with great clarity. Oliphant forwarded the document to Henry Tizard, Chairman of the British Government's Committee on the Scientific Survey of Air Warfare. Tizard asked George P. Thomson of Imperial College, London, to convene a committee to investigate the matter. Thomson served as the chair; other members included Oliphant and James Chadwick. The committee first met in April 1940, and by that summer, research was underway at various British universities on cross-sections, uranium chemistry, and isotope enrichment.

Uranium isotope effects, and the discovery of plutonium

In parallel with criticality analyses, theoretical understanding of fission was explored by Niels Bohr and John Wheeler in the first half of 1939.

The experimental evidence seemed conflicting. Both uranium and thorium could capture slow neutrons and then beta-decay to heavier elements, but tended to fission under fast-neutron bombardment. However, uranium also fissioned under slow-neutron bombardment, whereas thorium did not. Pondering how both capture and fission could proceed concurrently in uranium, Bohr and Wheeler developed an analysis which indicated that it is the rare isotope ^{235}U which is responsible for all slow-neutron fission and much of the fast-neutron fission in that element, while explaining why thorium exhibited weak fast-neutron fission but no slow-neutron fission.

Their analysis focused on the number of neutrons in nuclei. Uranium comprises two isotopes, one with an odd number of neutrons (^{235}U ; 0.7% of naturally-occurring uranium), and one with an even number (^{238}U ; 99.3%), whereas thorium has only one stable isotope, ^{232}Th , which has an even number. Since uranium exhibited slow-neutron fission while thorium did not, it appeared that the odd-neutron isotope ^{235}U might be responsible for slow-neutron fission in that element, because thorium did not possess an odd-neutron isotope. The physics behind this surmise was that, due to inter-nucleon pairing forces, even-neutron heavy-element nuclei exist in more stable mass-energy configurations than odd-neutron ones. When a nucleus captures an incoming neutron, it becomes a heavier isotope of its element with its own characteristic mass, which is always less than the sum of the masses of the original nucleus and the neutron; the lost mass appears as excitation energy of the compound nucleus in accordance with $E = mc^2$. When an odd-neutron nucleus captures an incoming neutron, the result is a higher excitation energy in the compound nucleus than when capture is by an even-neutron-number one. This indicated that when an odd-neutron nucleus such as ^{235}U captures a neutron, it could find itself in a more excited energy state and hence more prone to fission than would an even-neutron one such as ^{238}U .

Bohr and Wheeler determined that any nucleus can be induced to fission under neutron bombardment, provided that the sum of the kinetic energy of the incoming neutron and any mass-energy liberated by neutron capture exceeds a necessary activation energy, the fission barrier. For middle-weight elements, fission barriers are ~ 50 MeV, but fall to 5–6 MeV for heavy elements such as uranium, with precise values depending on the neutron number involved. Independently of this, if a nucleus undergoes an even-to-odd neutron number transition, ~ 5 MeV of binding energy are released, whereas ~ 6.5 MeV are released if the transition is odd to even.

This explained the slow/fast behavior of uranium and thorium. For ^{235}U and ^{238}U , the binding energy releases are ~ 6.5 and 4.8 MeV, respectively. However, the fission barriers for these isotopes are ~ 5.0 and 6.2 MeV. For ^{235}U , the binding energy release by itself exceeds the fission barrier; any bombarding neutron, no matter what kinetic energy it has, can induce fission in ^{235}U . But for ^{238}U , the binding-energy release falls some 1.4 MeV short of the fission barrier, which means that to fission ^{238}U requires supplying neutrons of at least this amount of energy. ^{232}Th behaves similarly to ^{238}U .

As for fast neutrons, about half the neutrons liberated in fissions have kinetic energies in excess of 1.4 MeV, so it seemed that ^{238}U could fast-neutron chain-react, but a complication intervenes. Energetic neutrons striking ^{238}U nuclei are much more likely to be inelastically scattered and lose most of their kinetic energy than they are to cause a fission; this slows them to well below the 1.4 MeV threshold. To compound this, ^{238}U has a large neutron-capture cross-section for intermediate-energy neutrons, which explained the observed capture-decay behavior of natural-abundance uranium. ^{238}U is useless as a bomb material due to this combination of scattering and capture; in a fast-neutron environment, ^{238}U is a neutron-capturing poison. By late 1939, it was becoming clear that to make a fast-neutron explosion would require isolating kilograms of the rare ^{235}U isotope from its much more abundant sister isotope.

These predictions required experimental proof, which meant isolating separate samples of ^{235}U and ^{238}U and subjecting both to neutron bombardment. The only practical method of isotope separation at the time was mass spectroscopy, and microgram-scale samples were prepared by Alfred Nier in February 1940. The ^{235}U samples showed clear evidence for slow-neutron fission, while the ^{238}U samples showed none at all. The ^{238}U samples also fissioned under fast neutrons (the threshold effect), but the ^{235}U sample was too small to test for fast-neutron fission at the time.

Despite its lack of fissility, ^{238}U did play a key role in the Manhattan Project. The ^{239}U nucleus formed in neutron capture by ^{238}U sheds energy in a series of two beta-decays, transmuting to ^{239}Pu :



^{239}Pu is an odd-neutron isotope and behaves similarly to ^{235}U in its fission properties. The great advantage of synthesizing plutonium in this way is that it obviates having to develop isotope separation technologies in order to acquire fissile material. The feasibility of actually performing this synthesis rests on the enormous fission cross-section of ^{235}U for slow neutrons. If fission-generated neutrons can be slowed with a non-neutron-capturing moderator before they are captured by ^{238}U , their probability of subsequently fissioning ^{235}U nuclei dominates over their probability of being captured by ^{238}U despite the small abundance of ^{235}U . This is what makes possible slow-neutron controlled chain-reactions utilizing natural or slightly enriched uranium, while simultaneously synthesizing ^{239}Pu .

Reaction (1) was formally proposed by Louis Turner of Princeton University in May 1940, but withheld from publication until after the war. At the time, however, the discovery of plutonium began unfolding at the University of California at Berkeley based on a speculation by Glenn Seaborg that exactly this process might occur. The first step was the isolation of ^{239}Np , which was accomplished by Edwin McMillan and Philip Abelson, also in May 1940. About a year later, a team under the direction of Glenn Seaborg isolated a fraction of a microgram of ^{239}Pu , and verified that it was fissile.

The Manhattan Engineer District: Two pathways to atomic bombs

In mid-1939, Leo Szilard and Eugene Wigner, Hungarian-born Jewish émigré physicists living in America, felt that government officials needed to be alerted to the possibility of fission as a weapon. Reasoning that Albert Einstein, a friend of Szilard, was famous enough to be recognized by anybody, they prevailed upon him to sign a letter which they intended to submit to the State Department. Another acquaintance of Szilard and an advisor to President Roosevelt, Alexander Sachs, offered instead to take the letter directly to the White House.

Sachs briefed Roosevelt on October 11, and the President ordered the Director of the National Bureau of Standards, Lyman Briggs, to assemble an advisory committee comprising himself and ordnance experts from the Army and Navy. The committee held its first meeting on October 21, hearing briefings from Szilard, Wigner, and Enrico Fermi. The War and Navy Departments contributed \$6000 to Fermi, then at Columbia University after fleeing his native Italy, so that he could purchase supplies for neutron-capture experiments being carried out with a view to establishing a chain reaction.

In June 1940, Briggs's committee was absorbed by the National Defense Research Committee (NDRC), an agency established to coordinate research which might have military applications. The NDRC was directed by Vannevar Bush, a Massachusetts Institute of Technology electrical engineer, and James Conant, a chemist and President of Harvard University. In June 1941, the NDRC would be re-named the Office of Scientific Research and Development (OSRD); by the time of Pearl Harbor, the NDRC/OSRD had let contracts totaling some \$300,000 for fission and isotope-separation research.

For independent advice on the uranium issue, Bush asked Arthur Compton to gather a committee to conduct a comprehensive review. In three reports prepared between May and November 1941, Compton's group considered reactors and bombs, concluding that fission bombs were possible. These conclusions were independently supported by George Thomson's committee in Britain, and Bush received the go-ahead from Roosevelt in late November. In March 1942, it was decided to transfer the project to the War Department in view of the anticipated costs (estimated at \$500 million) and need for secrecy. Two programs would be pursued: enriching ^{235}U , and breeding plutonium.



Fig. 1 Left: Leslie R. Groves (1896–1970). Right: Robert Oppenheimer (1904–1967), ca. 1944. Source: <http://commons.wikimedia.org/wiki/File:JROppenheimer-LosAlamos.jpg>

The Army Corps of Engineers established the Manhattan Engineer District in August 1942; its first headquarters were located in Manhattan. In September, command was assigned to Brigadier General Leslie Groves (**Fig. 1**). At the time of his appointment, Groves was responsible for overseeing all Army construction within the United States, including the Pentagon; he had intimate knowledge of contractors who could undertake complex projects.

Uranium enrichment and plutonium production reactors

The Manhattan District established two factory complexes for producing fissile materials, one at Oak Ridge, Tennessee, and one at Hanford, Washington. In addition, a highly-secret bomb-design laboratory was established at Los Alamos, New Mexico, directed by Dr. J. Robert Oppenheimer of the University of California (**Fig. 1**).

Oak Ridge was the site of three uranium enrichment facilities and a pilot-scale reactor; the location was chosen for its access to electrical power from the Tennessee Valley Authority. The three uranium-enrichment facilities, code named Y-12, K-25, and S-50, each utilized a different process. The reactor, code-named X-10, was part of the plutonium-production program, and is discussed later in this section. Oak Ridge would consume \$1.2 billion in construction and operating costs.

Y-12: Electromagnetic separation

The operating principle of the Y-12 facility was identical to that of a cyclotron, wherein an ionized atom or molecule is directed into a magnetic field in such a way that it moves in a circular trajectory whose radius depends on its mass. At Oak Ridge, these devices were called calutrons, a contraction of California University Cyclotron.

See **Fig. 2**. The large rectangle represents a vacuum tank, which contains equipment to ionize and accelerate molecules of uranium tetrachloride. The molecules are shot vertically upwards into a magnetic field, which is created by a surrounding coil and is arranged to emerge perpendicularly from the page. If the ion stream comprises molecules of two different masses, the result is two streams that travel in orbits of different radii and which can be separated. While conceptually simple, this was difficult to realize in practice: the beams had diameters of just over 3 m, and a separation of only about a centimeter.

Calutron operation is constrained by the fact that the beams repel each other and can become disrupted from their paths. At Oak Ridge, this set a limit of isolating about 0.1 g of ^{235}U per day per vacuum tank. To collect enough material for a critical mass in 1 year would require over 1000 tanks in simultaneous operation, and Y-12 eventually boasted 1152. An efficient way to arrange many tanks is to place them between magnet coils, and copy such arrangements back-to-back in a closed track connected to a source of current. The track shown in **Fig. 3** contained 96 pairs of vacuum tanks placed within 48 magnet coils, which appear as rib-like structures. The tanks could be withdrawn to extract separated material; operators monitored and adjusted the beams in each tank. The first calutrons entered operation in November 1943, and by August 1945, some 60 kg of bomb-grade uranium had been isolated.

K-25: Gaseous diffusion

To 800-m long K-25 factory at Oak Ridge housed the gaseous diffusion plant of the Project, and consumed over \$500 million in construction and operating costs (**Fig. 4**).

Gaseous diffusion is based on a result of kinetic theory. An atom in an environment at a given temperature will move in random motion with a speed that is inversely proportional to the square root of its mass. Lighter atoms move faster on average than heavier ones, and, if a gas of atoms of mixed isotopic composition is pumped against a barrier containing millions of microscopic holes, those of lower mass will pass through slightly more frequently than those of higher mass, resulting in a minute level of enrichment

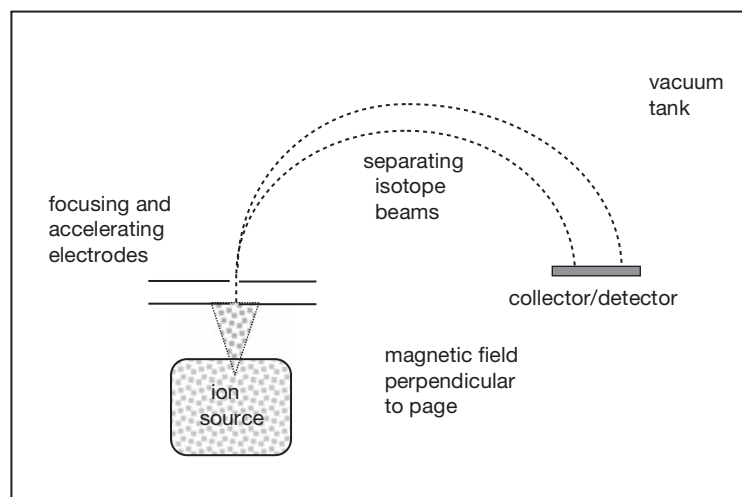


Fig. 2 Schematic illustration of a calutron tank. Sketch by author from Reed BC (2019) *The History and Science of the Manhattan Project*, 2nd edn. Berlin: Springer.



Fig. 3 A Y-12 track; spare vacuum tanks appear in the lower-left corner. Source: https://commons.wikimedia.org/wiki/File:Y-12_Calutron_Alpha_racetrack.jpg

of the lighter-isotope gas on the other side of the barrier. The resulting enrichment is dictated by the ratio of the masses, so to achieve bomb-grade enrichment requires repeating the process thousands of times.

In practice, large tanks containing diffusion membranes were linked together in a cascade (Fig. 5). In the figure, feed material enters the second tank from the bottom. Gas enriched in the lighter isotope is advanced to the next stage of the cascade, while that depleted in the lighter isotope—but which still contains atoms of that isotope—is recycled for additional processing. Lighter-isotope material accumulates toward the top of the diagram, while heavier-isotope material concentrates toward the bottom. K-25 incorporated 2892 such stages, and was built between 1943 and 1945; operations began in early 1945.

S-50: Liquid diffusion

The premise of liquid diffusion is that if a gas or fluid comprising two isotopes of an element is subjected to a thermal gradient, the lighter isotope will preferentially collect toward the hotter region, while the heavier one will collect toward the cooler region. Fluid



Fig. 4 The K-25 diffusion plant. Source: https://upload.wikimedia.org/wikipedia/commons/1/1b/K25_Aerial.jpg

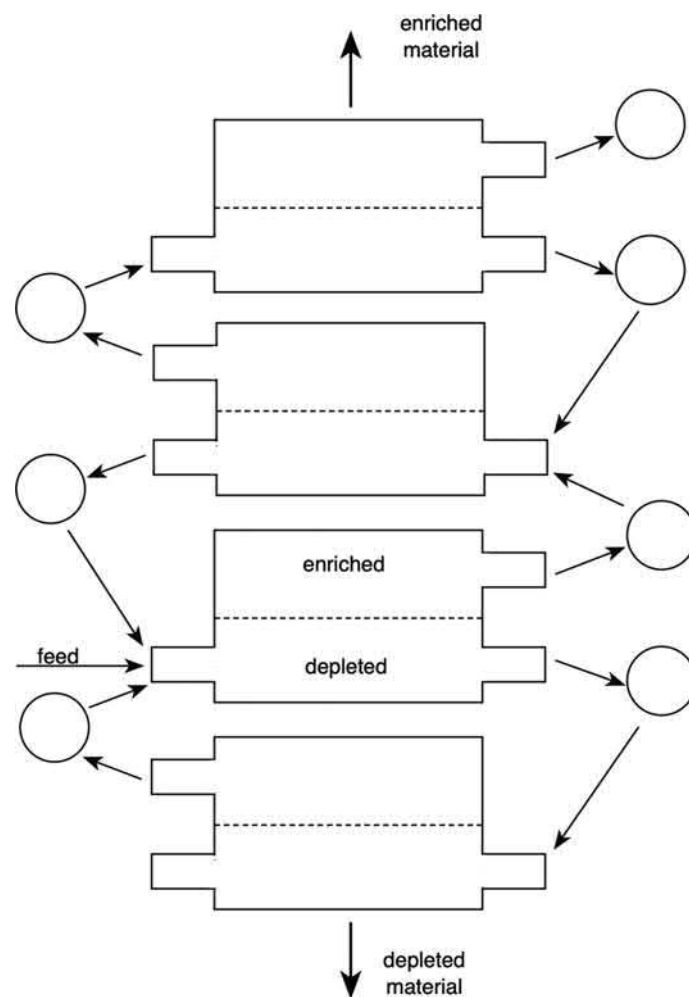


Fig. 5 Schematic illustration of a diffusion cascade. The circles represent pumps. In the K-25 plant, all tanks were at ground level. Sketch by author from Reed BC (2019) *The History and Science of the Manhattan Project*, 2nd edn. Berlin: Springer.

containing the lighter isotope will then naturally rise vertically by convection while that containing the heavier isotope falls, and the fluid slightly enriched in the lighter isotope can be drawn off for further processing.

Operationally, this employed columns of three nested pipes (Fig. 6). The inner one is heated by steam pumped through its center. The intermediate pipe surrounds the inner one, with only a quarter-millimeter separation between them, and the outer surface of the intermediate pipe is chilled by cold water pumped through the outer pipe. Liquefied uranium hexafluoride was then fed into the annulus between the two inner pipes, and material slightly enriched in ^{235}U rose to the top of the assembly, from where it was collected. The S-50 plant utilized 2142 such columns, each about 15 m high, all operating in parallel. A single column could enrich the natural-abundance feed material, 0.72% ^{235}U , to 0.86% ^{235}U . Construction began in June 1944, and all columns were in operation by March 1945.

In early 1945, Groves decided to harness the three enrichment methods in series as opposed having them compete in parallel. Ultimately, processing began by first feeding natural-abundance uranium hexafluoride to the S-50 plant. This product was then fed to the K-25 plant to be taken to 20% ^{235}U , and then to Y-12 to be enriched to 90%.

Plutonium production reactors

Enrico Fermi and his collaborators began building experimental slow-neutron piles soon after he arrived in America to take up a position at Columbia University in January 1939. Their first effort involved a cylindrical tank which held ~ 570 kg of water and into which a neutron source could be inserted. A glass bulb filled with uranium oxide was placed in the tank, and an increase in the neutron population above that when no oxide was present was detected, although criticality was nowhere near approached. Beginning about mid-1940, the group moved to constructing piles proper, columns of graphite bricks up to 8 ft. square in footprint by 11 ft. high into which could be inserted layers, cubes, or spheroids of uranium oxide or pure uranium metal. These devices were used to empirically determine optimal arrangements for the disposition of uranium within a - pile; ultimately, lumps of metal distributed in a cubical lattice pattern were deemed to be best. Column-pile work began at Columbia and was continued at the University of Chicago, to where the group relocated in early 1942 when Manhattan Project administrators decided to centralize pile work there. A total of some 30 such piles were constructed between the spring and fall of 1942; in addition, some 20 smaller piles were constructed for checking the neutron-capture properties of various batches of graphite.

By the fall of 1942, Fermi was ready to attempt a self-sustaining chain-reacting pile, CP-1. Some sources give this abbreviation as "Critical Pile 1" and others as "Chicago Pile 1." Construction began on November 16, with physicists and hired laborers assembling

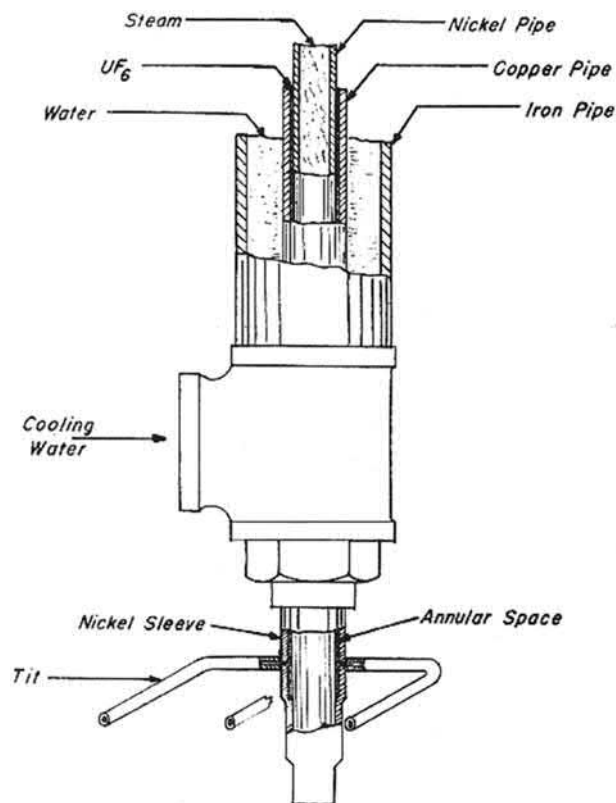


Fig. 6 Sectional view of a thermal diffusion process column. From Manhattan Engineer District documents.

nearly 400 tons of graphite bricks embedded with over 19,000 spheroidal slugs of uranium metal into an ellipsoid of equatorial radius nearly four meters and polar radius just over three meters. Layers of solid graphite bricks alternated with ones containing slugs configured to form a lattice as the pile was built up. Control was effected by a system of manually and motor-driven neutron-absorbing cadmium-sheathed wooden rods that could be withdrawn from or inserted into the pile as necessary. Self-sustaining criticality was achieved on the afternoon of December 2; Fermi estimated that the pile was operating at a power of about one-half of a Watt.

With proof that a chain reaction could be created and controlled, attention turned to design of both a pilot-scale pile and full-scale plutonium production piles. The X-10 pilot pile at Oak Ridge was built to produce a few hundred grams of plutonium for research and to train operators for the full-scale piles. This was a one-megawatt, air-cooled, graphite-moderated cubically-shaped pile about 7 m on a side. A reactor fueled with natural uranium produces about three-quarters of a gram of plutonium per day per megawatt of power output; if X-10 ran for a full year at 1 MW, it would produce about 275 g of plutonium. Ultimately, it exceeded expectations.

As the pile was assembled, beveled-edged graphite bricks were laid side-by-side to form channels into which slugs of uranium could be fed through concrete shielding (Fig. 7). Irradiated slugs fell from the back of the pile into a pool of water as new ones were pushed in, and their radioactivity allowed to decay before being processed.

X-10 went critical on November 4, 1943. Expansion of the cooling system allowed operation at 4 MW, and by mid-1944, tens of grams of plutonium were being produced per month. X-10 plutonium would lead to the discovery of the spontaneous-fission crisis at Los Alamos in the summer of 1944; this is discussed in the following section.

The site chosen for plutonium production piles was a remote area in south-central Washington state near the town of Hanford on the Columbia river, from which water could be drawn to cool the piles. The site occupied over 1700 km², large enough to accommodate three well-separated 250-MW piles. They were labeled B, D, and F, corresponding to survey locations. Equally monumental would be three chemical separation plants in which irradiated fuel slugs were dissolved and their plutonium extracted.

Each pile comprised a graphite core about 12 m square by 8.5 m deep, built of some 75,000 graphite bricks. Slugs of uranium were loaded into 2004 tubes which passed through the piles, with discharged slugs falling into a pool as with X-10 (Fig. 8). Operators could remotely monitor the flow and temperature of cooling water and adjust control rods.

B-reactor achieved criticality on September 26, 1944, with about 1600 tubes loaded, and within a day was operating at 9 MW. But about an hour after reaching 9 MW, operators noticed that reactivity was declining. By the evening of the 27th, the pile had shut itself down. Surprisingly, a few hours later, it spontaneously began operating again, only to shut itself down again. From the pattern of the reactivity, it was determined that the problem was likely a fission product with a high neutron-capture cross-section. The culprit was xenon-135, which has the largest-known slow-neutron capture cross-section of any nuclide. Fortunately, it has a half-life of only about 9 h. The solution to overcoming the problem was to increase the amount of fuel in the reactor, which meant shutting down while connecting unused tubes to the cooling system. B-pile achieved its full design power of 250 MW on February 4,

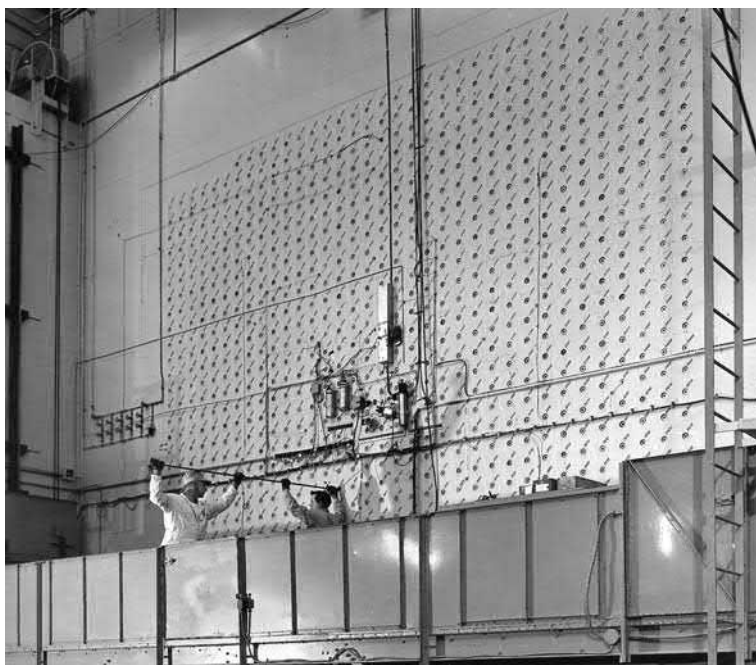


Fig. 7 Front face of the X-10 pile. Source: http://commons.wikimedia.org/wiki/File:X10_Reactor_Face.jpg

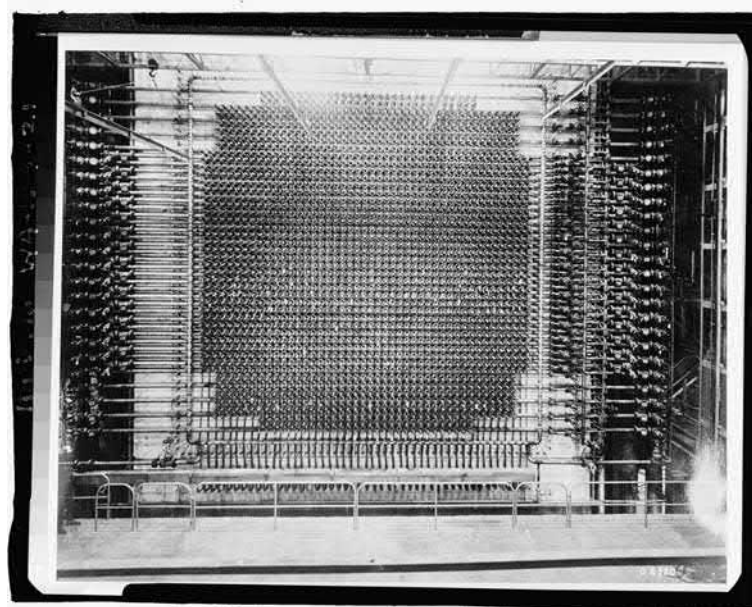


Fig. 8 Front face of F-pile, February 1945. Source: https://commons.wikimedia.org/wiki/File:View_from_the_work_area_of_the_front_face_of_the_pile_in_the_105_building,_in_this_case_at_the_F_Reactor_in_February_1945_The_2,004_pigtails_and_process_tube_nozzles_are_neatly_aligned_HAER_WA-164-21.tif

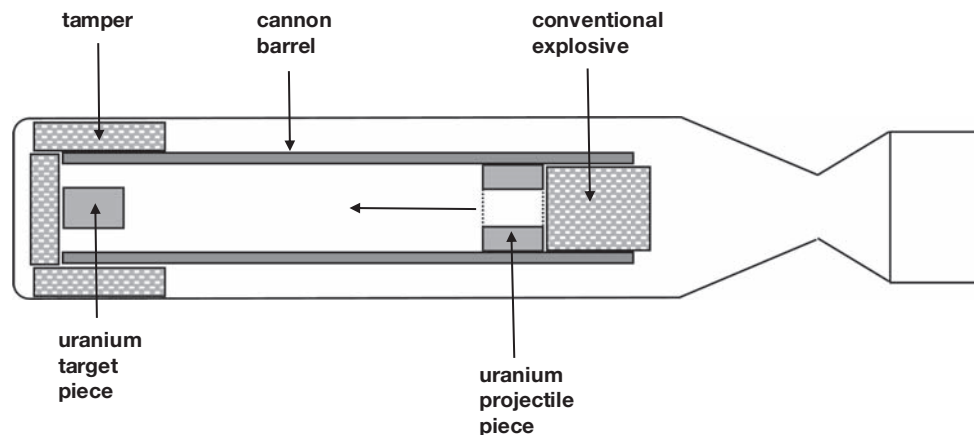


Fig. 9 Schematic illustration of a gun-type weapon. Sketch by author from Reed BC (2019) *The History and Science of the Manhattan Project*, 2nd edn. Berlin: Springer.

1945, and D and F piles started up with full fuel loads in December 1944, and February 1945. By the spring of 1945, kilogram quantities of plutonium were making their way to Los Alamos.

Los Alamos and trinity

The mission of the Los Alamos Laboratory was to design and build combat-ready bombs incorporating the products of Oak Ridge and Hanford.

Assembling a supercritical mass of ^{235}U proved remarkably easy with what was termed the gun method (Napolitano, 2021). As sketched in Fig. 9, this utilized two sub-critical pieces of fissile material placed within the barrel of an artillery cannon. A cylindrical target piece is held fixed at the muzzle end of the barrel, while a projectile piece is fired toward the target piece from the tail end; when fully mated, the two comprise more than one critical mass. With the cannons available in World War II, some 100 microseconds elapse between when the leading edge of the projectile piece first encounters the target piece and when full assembly is achieved. This number is very important.



Fig. 10 Little Boy. Source: http://commons.wikimedia.org/wiki/File:Atombombe_Little_Boy_2.jpg

The final Little Boy uranium bomb was about 3 m long and weighed about 4400 kg, much of which was the steel tamper; the fissile material totaled to about 65 kg (**Fig. 10**). Little Boy was ready by May 1945, and enough ^{235}U for one bomb was expected to be available around August 1. Little Boy would be deployed without a full-scale test.

In the summer of 1944, Los Alamos was plunged into a crisis. Calculations indicated that once the chain reaction was initiated, only about one microsecond would be required to fission the entire bomb core. This is much less than the projectile/target assembly time, and was the origin of the problem. At some point during assembly, a critical mass will come to be present in the partially-seated core. If a stray neutron initiates the explosion before assembly is complete (that is, before the bomb reaches its highest possible level of supercriticality), the result could be a very low-efficiency predetonation. The problem emerged in April 1944, with the arrival of plutonium from Oak Ridge: the material exhibited a high rate of neutron-liberating spontaneous fissions, an unavoidable consequence of how it was produced. When an already-synthesized ^{239}Pu nucleus is struck by a slow neutron in a reactor, it has some chance of capturing the neutron and becoming ^{240}Pu as opposed to fissioning. Reactor-produced plutonium inevitably contains some ^{240}Pu , and this isotope features a spontaneous fission rate of almost half a million per second per kilogram. It became clear that during the 100-microsecond assembly time, a predetonation would be virtually guaranteed. The only option was to speed up the assembly process to on the order of a few microseconds, which would be impossible with the gun mechanism. ^{235}U does spontaneously fission, but at a rate low enough not to be a problem for the gun assembly.

The solution was implosion, where an assembly of three-dimensional explosive blocks is configured to reverse an outwardly-progressing shock wave so that it focuses inward and crushes a sub-critical core to critical density. **Fig. 11** shows a sketch of a single block in side-view cross-section; in three dimensions, imagine a five or six-sided structure about 30 cm across. Each block comprises two castings of different explosives that fit together very precisely, and which interlocks with neighboring blocks to form a spherical shell. The outer casting of each block was a fast-burning explosive, Comp B, while the inner segment was a slower-burning material, Baratol. A detonator triggers an outward-expanding detonation wave in the Comp B, which progresses to the left in the figure. When the detonation wave hits the Baratol, it too begins exploding. If the interface is of just the correct curvature, the two waves will combine to create an inwardly-directed converging wave in the Baratol.

In the Trinity and Fat Man bombs, 32 such assemblies interlocked to create a spherical shell; see **Fig. 12**. The shell surrounded an inner assembly of 32 blocks of Comp B, which was detonated by the imploding Baratol to crush the core. For combat use, the entire assembly had to be enclosed within a ballistic casing (**Fig. 13**).

Implosion was considered so novel that it was decided to use some of the Hanford plutonium in a test of the method. This was the Trinity test of July 16, 1945, conducted in southern New Mexico. The test was a spectacular success, yielding an estimated 21 kt TNT equivalent (**Fig. 14**). On August 6, the Little Boy uranium bomb was dropped at Hiroshima (~ 13 kt), and on August 9 the Fat

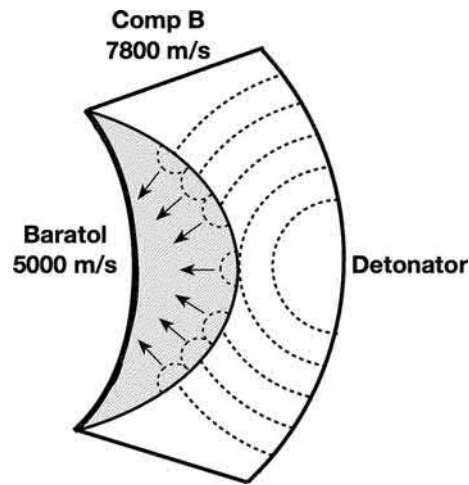


Fig. 11 Schematic illustration of a binary-explosive implosion block. Not to scale. Sketch by author from Reed BC (2019) *The History and Science of the Manhattan Project*, 2nd edn. Berlin: Springer.

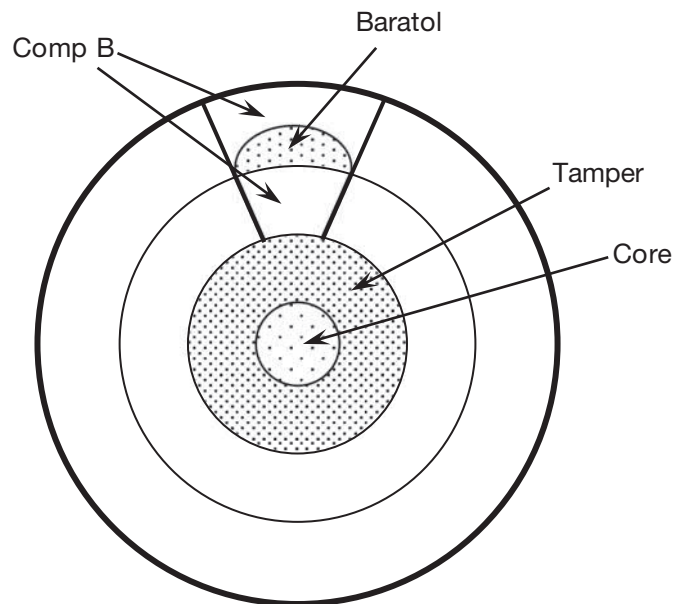


Fig. 12 Schematic illustration of an implosion bomb; not to scale. Only one of 32 implosion blocks is shown. Sketch by author from Reed BC (2019) *The History and Science of the Manhattan Project*, 2nd edn. Berlin: Springer.

Man combat version of the Trinity device was dropped at Nagasaki (~21 kt). Faced with these devastating new weapons and declaration of war by Russia, the Japanese surrendered a few days later.

The German weapons program

The German nuclear program began in April 1939, when Georg Joos of the University of Göttingen wrote to the Reich Ministry of Education to alert officials to the possibilities of fission. His letter reached Abraham Esau, head of the Ministry's Reich Research Council, who arranged for a ban to be placed on exports of uranium compounds.

Esau had a research program in mind, but a rival initiative was underway. On April 24, University of Hamburg physical chemist Paul Harteck wrote to the German War Office to alert officials to the fact that fission could lead to powerful explosives. This letter reached Erich Schumann of the Ordnance Department, who passed it on to Kurt Diebner, an Army expert on nuclear physics and explosives. Esau found himself sidelined by the more powerful War Office when, in early September, he met with General Karl Becker, head of Army Ordnance, to request provision of uranium compounds. To his surprise, he was ordered to cease uranium research and told that a cache of uranium oxide he had accumulated would be confiscated.

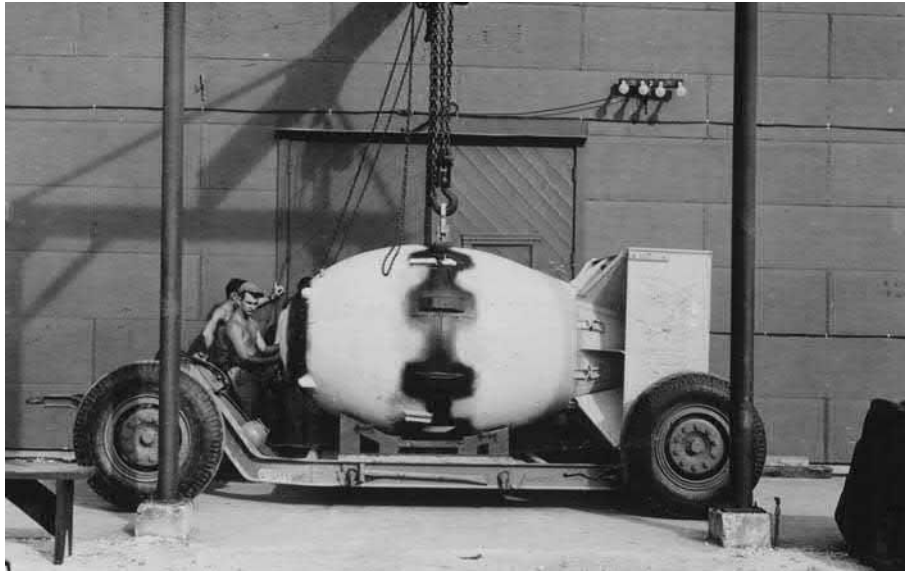


Fig. 13 Fat Man. Source: https://upload.wikimedia.org/wikipedia/commons/5/5f/FM_%28Fat_Man%29_unit_being_placed_on_trailer_cradle_in_front_of_Assembly_Building_%5E2_-_NARA_-_519397.tif

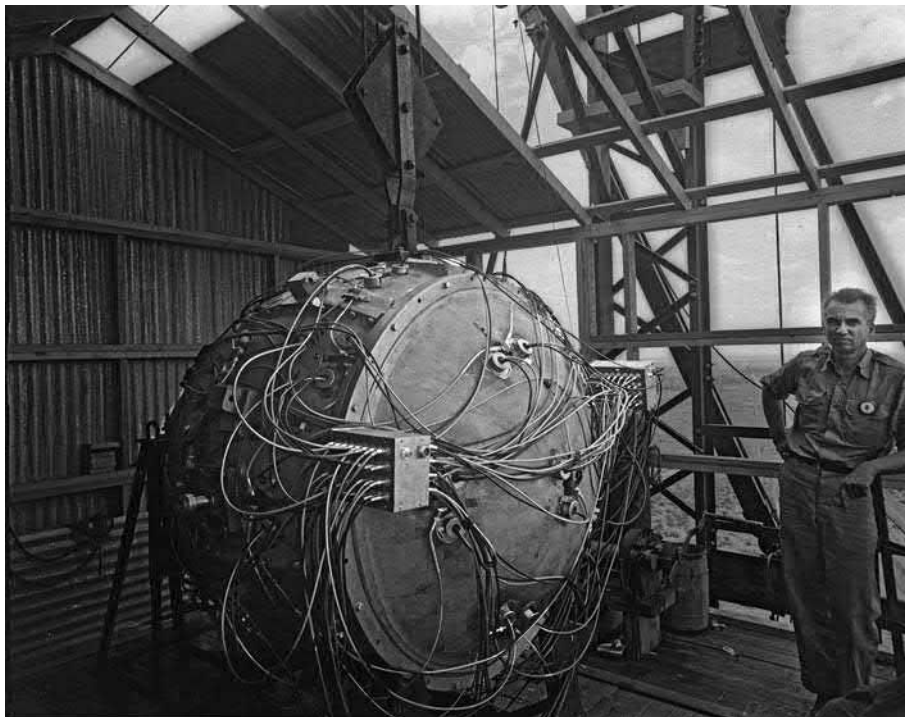


Fig. 14 The Trinity device atop its test tower on July 15, 1945, with Norris Bradbury. The cables feeding from the boxes go to implosion-block detonators. Source: http://commons.wikimedia.org/wiki/File:Trinity_Gadget_002.jpg

The same week as Esau's initiative was derailed, Erich Bagge of the Leipzig Institute for Theoretical Physics (and a student of Werner Heisenberg), was ordered to report to Berlin. There he met with Diebner and Schumann, who wanted him to arrange a conference to explore the feasibility of using uranium as a source of power or explosives. Otto Hahn, who participated in the meeting, was skeptical of achieving a chain reaction, but felt that the possibility could not be entirely dismissed. A research group under Diebner was set up in Gottow (a suburb of Berlin), and Heisenberg was brought in to work out the theory of a chain reaction. At another conference 10 days later, Heisenberg reported that two routes to utilizing fission might be possible: in reactors if a suitable moderator could be found, and/or as an explosive if ^{235}U could be isolated. Harteck had already begun to conceive of a reactor wherein uranium and heavy-water moderator would be arranged in layers, and was researching isotope separation methods.

Heisenberg was to develop the theory of chain reactions, Bagge to investigate the neutron-collision properties of heavy-water, and Harteck to consider isotope separation.

At about this time, Schumann had the War Office take over the Kaiser-Wilhelm Institute of Physics (KWIP) in Berlin as a location at which to centralize the work. The director of the KWIP, Peter Debye, left for America, and Schumann replaced him with Diebner over the objections of staff and administrators, who wanted Heisenberg. Diebner's efforts to coordinate the project were constantly frustrated by scientists preferring to stay at their home institutions; Heisenberg agreed to commute from Leipzig once a week as an advisor. Personality clashes were a motif of the German program.

In a report to the War Office in February 1940, Heisenberg initiated what has become one of the enduring mysteries of the German nuclear program: whether or not he knew how to calculate the critical mass. His result, some 500 metric tons of uranium, resulted from an erroneous expression for the critical radius and apparent confusion between the roles of reactors and bombs. That he would err so dramatically is mystifying given that papers on criticality had already appeared in the open literature. In contrast, an unsigned early-1942 Army Ordnance report on a series of lectures at which Heisenberg had been one of the participants included an unattributed estimate of from 10 to 100 kg, and at a meeting with Minister of Munitions Albert Speer a few months later he allegedly claimed that a bomb about the size of a pineapple could destroy a city. But when he was interned in England with other German scientists at the end of the war and their conversations secretly recorded, Heisenberg claimed, in private conversation with Otto Hahn after learning of the bombing of Hiroshima, to have never worked out the critical mass because he never believed that it would be possible to isolate enough pure ^{235}U . One interpretation of this story is that Heisenberg did in fact know how to calculate the critical mass, but deliberately overestimated it and slowed the program as part of an effort to deny Hitler the bomb. We will never know the truth; Heisenberg himself never addressed these conflicting versions after the war, perhaps preferring to put that time behind him.

The German invasion of Belgium in May 1940, was a boon for the program. The Union Minière du Haut-Katanga mining company was headquartered in Brussels, and the firm mined rich uranium ores in the Belgian Congo, tons of which were soon on their way to Berlin. In July, planning began to house a subcritical pile on the grounds of the Kaiser-Wilhelm Institute of Biology and Virus Research, which was located next to the Physics Institute in Berlin. To deter the curious, the laboratory was designated The Virus House. The site was ready by October, and Heisenberg and others began assembling their first pile, which involved layers of uranium oxide separated by paraffin as a moderator, all held within a container suspended in water. No chain reaction was observed, and Heisenberg concluded that an ordinary-water or paraffin-moderated pile would not achieve criticality; heavy water or graphite would be necessary.

The program began to stall in early 1941. A devastating setback occurred in January, when Walther Bothe determined that graphite captured too many neutrons to serve as a moderator; his material was probably contaminated with neutron-capturing boron. This meant that the program would have to rely on heavy water as a moderator, but the only large-scale source of that commodity was an electrolysis plant in Norway. The plant produced on the order of 100 kg per year as a by-product to its main operations, but a pile would require tons. Germany began pressing for increased production, particularly after occupying Norway. Allied intelligence became aware of German interest in the plant, which was subsequently targeted with commando and bombing raids. These disrupted production to the extent that the German program never had enough moderator to achieve a chain reaction, albeit at a cost of several dozen lives. By late 1941, German forces bogged down in the battle of Moscow, and Army Ordnance decided to reduced its funding of uranium work and relinquish control back to Abraham Esau; the Army eventually withdrew from the program altogether in 1943, but Diebner's group continued its research under Esau's oversight. In mid-1942, just as the Manhattan Engineer District was being established, the program was further curtailed when Albert Speer decreed that support would be limited to construction projects, particularly a bunker to house a larger pile at the KWIP. Heisenberg garnered a personal victory in October when he was made director of the KWIP, and Diebner retreated to Gottow.

At Gottow, Diebner conceived the idea of distributing uranium as chunks within a moderator as opposed to Heisenberg's preferred schemes of layers or shells. A test pile showed no increase in neutron flux, but this idea did initiate a new direction for the German program, one similar to Fermi's approach. Eventually, the concept evolved into suspending cubes of uranium on wires directly into heavy water.

In Berlin, Heisenberg carried on with the bunker pile, although work was hampered by relentless bombing raids. The reactor vessel was a magnesium-alloy cylinder just over a meter wide, a much smaller structure than CP-1; the German program never came close to having the materials of the Allied program. Despite Diebner's superior lattice concept, time was lost in trying different plate separations to see which one gave the best neutron enhancement. Further delays occurred when it was decided in early 1945 to relocate all pile research to Haigerloch, in southern Germany, where a wine cellar carved into a cliff was enlarged to accommodate a laboratory.

The last German pile, Berlin-VIII, was located in the Haigerloch wine cellar and used Diebner's system of suspended cubes (Fig. 15). Some increase in neutron number was detected, but the pile did not achieve criticality. By April, Allied forces were closing in, and the Haigerloch group dispersed. British and American intelligence agents located the cave, seized graphite and uranium, destroyed the vessel, and blew up the cave. Many of the leaders of the German effort, including Heisenberg, were rounded up and interned in England for 6 months after the end of the war. On the evening of August 6, they were shocked to learn just how far behind their Allied counterparts they had fallen.



Fig. 15 A replica of the B-VIII pile at the Atomkeller Museum in Haigerloch. Source: https://de.wikipedia.org/wiki/Forschungsreaktor_Haigerloch#/media/File:Haigerloch_Atomkeller-Museum_Versuchsreaktor_2013-08-18.jpg. This image is freely available for commercial use according as the terms of a Creative Commons license available at <https://creativecommons.org/licenses/by-sa/3.0/>

Summary

The Manhattan Project succeeded in developing nuclear weapons in time to affect the outcome of World War II, but thrust the world into an uncertain new era. The legacy of Manhattan lives on in the arsenals of the nuclear powers of the world, and continuing concerns with nuclear proliferation and the possibility of nuclear terrorism.

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Nuclear Fusion

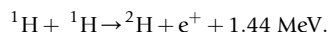
Edward C. Morse, University of California, Berkeley, CA, United States

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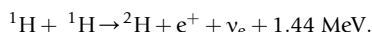
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Introduction

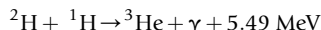
The earliest notion regarding nuclear fusion reactions was proposed by Eddington in 1920 (Eddington, 1920). The general idea is that the light elements have an average mass per nucleon M/A which decreases with the nucleon number A until A reaches about 60. This is called the curve of binding energy, discussed earlier. One might think of the most fundamental of all possible fusion reactions to thus be the transformation of ordinary hydrogen, with a proton as its nucleus, into something with an atomic number of two. However, there is no such thing as helium-2, and it took the discovery of deuterium in 1931 by Urey (Urey et al., 1932) and the positron by Anderson in 1932 (Anderson, 1933) to work out the details of proton-proton fusion. Thus a tentative form of the reaction might be given by (Bethe, 1939):



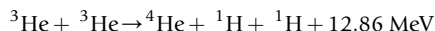
A more subtle problem remains, however. Although this reaction conserves charge, it does not conserve nuclear spin (i.e. angular momentum). Protons and positrons both have a spin of one-half, and the deuteron has a spin of one, so that there is a discrepancy of at least one-half unit of spin. (Units here are in the reduced Planck constant $\hbar = 1.05 \times 10^{-34}$ J-s.) The missing particle in the reaction given above is a neutrino, proposed by Pauli in 1930 to explain beta decay, the decay of the neutron into a proton and electron. Adding a neutrino with a spin of $\frac{1}{2}$ allows a reaction that conserves both charge and spin:



This is the start of the nuclear reaction sequence taking place in the sun and other stars with similar mass or less mass. (More massive stars have burned out this primary fusion fuel a long time ago.) Subsequent fusion reactions have much greater reaction probabilities and proceed quickly. These reactions include:



and



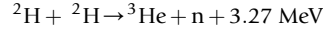
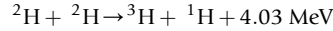
Thus, this sequence of reactions converts hydrogen into helium, with the overall result that six protons will eventually be involved to form a ${}^4\text{He}$ nucleus and returning two of the protons. The total yield of this net four-proton process, including the energy created in the neutrinos and the annihilation of the positrons with the “extra” electrons generated, is 26.7 MeV.

The neutrino production is key to understanding this process. This signals that the first reaction in this sequence is governed by the “weak force” as in beta decay. The cross section for this reaction is thus remarkably low and results in a reactivity $\langle \sigma v \rangle = 1.36 \times 10^{-47} \text{ m}^3 \text{ s}^{-1}$ at the temperature of the sun’s core. Such a low reactivity explains why the sun will be around for at least another 5 billion years, and why such a massive object is required to provide the earth with enough sunlight to sustain life. Confirmation of this process comes from direct measurements of the neutrinos produced, although reconciling the total neutrinos released from the sun took an understanding of the propensity for neutrinos to oscillate between three different “flavors.” (A clear experimental picture did not emerge until the years between 1998 and 2001, and resulted in Nobel prizes for two physicists (Kajita and McDonald) in 2015.)

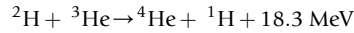
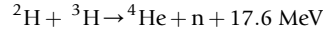
Subsequent nuclear reactions with alpha particles (${}^4\text{He}$ ions), ${}^3\text{He}$ ions, protons, and deuterons produce elements up to iron ($Z = 26$), representing the minimum in the curve of binding energy. To produce heavier elements, reactions with neutrons are

required. These can produce elements up to uranium ($Z = 92$). While these have not been produced to any extent in the sun, they are present in the cosmos as a result of supernovae and red giant stars.

This brings us to a discussion of fusion as an energy source on our small planet. There are a number of reactions at only slightly higher nucleon numbers which are not governed by the weak interaction but instead by the strong force. One of the more straightforward of these is the deuteron-deuteron reaction. It has two branches that occur with roughly equal probability. They are



These have reactivities $\langle \sigma v \rangle = 1 \rightarrow 100 \times 10^{-24} \text{ m}^3 \text{ s}^{-1}$ in the range of temperatures required for significant fusion reactions to take place, that is $kT = 10 \rightarrow 1000 \text{ keV}$. However, these are not the reactions that first-generation fusion reactors will employ. Instead, two other reactions are of interest:



The first reaction given, the D-T reaction, has a cross section approximately $100 \times$ that of either branch of the D-D reaction at a rather low energy (by nuclear fusion standards!) of around $80 \rightarrow 100 \text{ keV}$, and there is a similar behavior for D- ^3He . The reason for this disparate behavior between the D-D reactions and these latter reactions lies in the structure of two compound nuclei, ^5He for the D-T case and ^5Li for D- ^3He . In both cases, an excited state exists with energy just slightly above the combined rest mass of the D + T and D + ^3He nuclei, creating a resonance in the cross sections. Fig. 1 shows the situation for ^5He and D-T. A very small fraction of the D-T reactions will release a 16.8 MeV gamma, and this has been used as a diagnostic in past experiments. Fig. 2 shows the cross sections for these reactions. The resonant structure of D-T and D- ^3He is very apparent.

It is also important to consider not just the raw cross sections for these reactions, but also the reaction rate for these reactions taking place in a hot thermonuclear plasma with a distribution of plasma velocities. Thermonuclear reaction rates are given for two species 1 and 2 as

$$\text{R.R.} = \frac{n_1 n_2 \langle \sigma v \rangle}{1 + \delta_{12}},$$

with the Kronecker delta symbol being 1 if species 1 and 2 are the same and 0 if not. (This prevents counting each reaction twice when the species are the same.) The average $\langle \sigma \rangle$ is taken over the distribution of relative velocities v between two species. Often the plasma is close to Maxwellian conditions, and the distribution of relative velocities, normalized to one, is given by

$$f(v) = \left(\frac{\mu}{2\pi kT} \right)^{3/2} \exp(-\mu v^2 / (2kT))$$

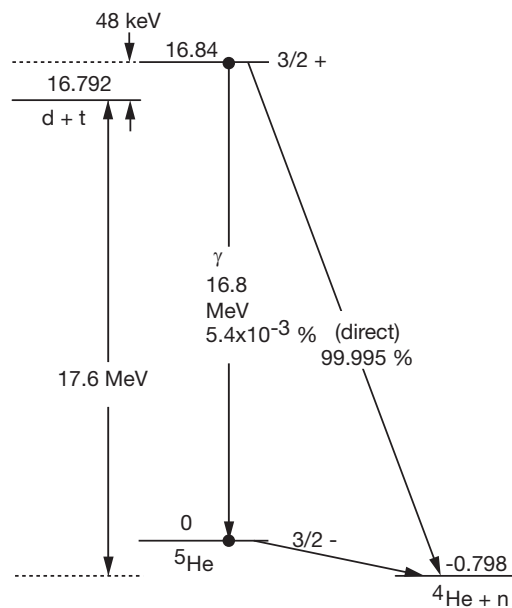


Fig. 1 Energy level diagram for ^5He , showing excited state with spin $3/2^+$.

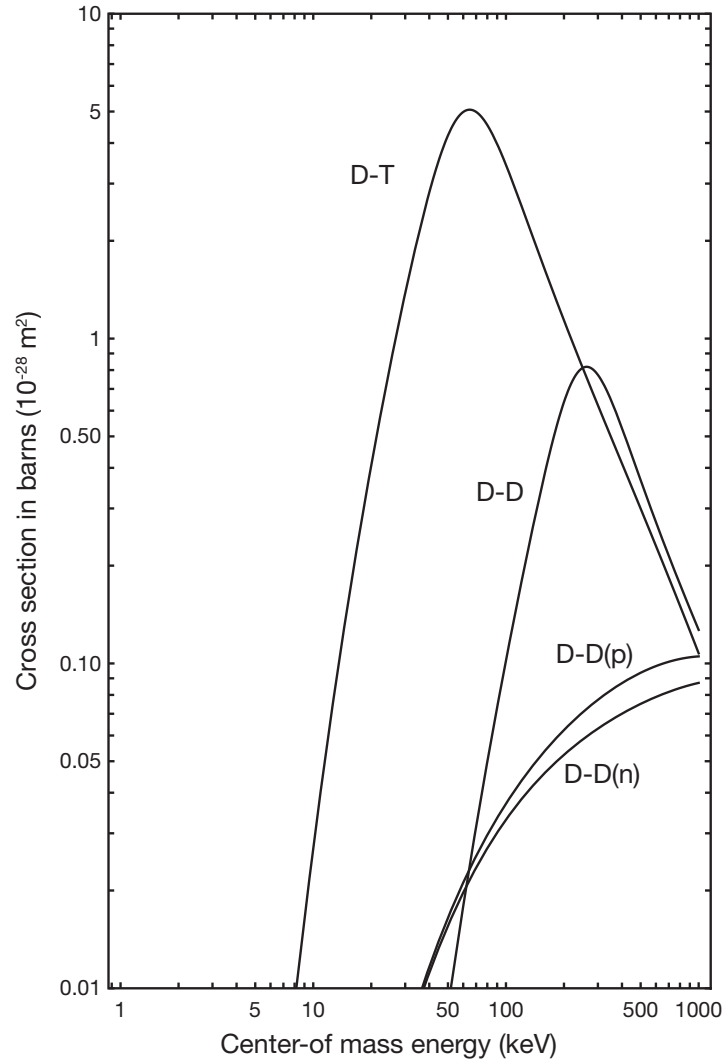


Fig. 2 Cross sections for D-T, D-³He, and D-D (both branches).

Here μ is the reduced mass $m_1 m_2 / (m_1 + m_2)$. For an isotropic distribution this can be written as a distribution in center-of-mass energies given by

$$f(E) = 2(\pi kT)^{-1/2} E^{1/2} \exp(-E/kT),$$

with E being the center-of-mass energy. Thus, for D-T fusion, most of the reactions involve particles well above the average energy $3/2 kT$. For all binary reactions, the Maxwellian-averaged reactivity $\langle \sigma v \rangle$ can be written as.

$$\langle \sigma v \rangle = \left(\frac{8}{\pi \mu kT^3} \right)^{1/2} \int_0^\infty \sigma(E) E \exp(-E/kT) dE.$$

Reactivity data is available in tabular form and as polynomial fits. See, e.g. (Bosch and Hale, 1992) for fit data for D-D, D-T, and D-³He reactions.

One might think that the ideal temperature for a fusion reactor would be the temperature for maximum reactivity, which turns out to be around $kT = 60$ keV for D-T. However, the real goal is to produce the maximum amount of fusion power per unit volume. In the case of magnetically confined plasma (explained later), the limiting factor is the amount of pressure that can be confined. This is given by

$$p = n_e T_e + n_i T_i,$$

where n_i is the total ion density and n_e is the electron density. If we assume that the electrons and ions have fully equilibrated, the electron and ion temperatures T_e and T_i are the same. Fusion plasmas are close to electrical neutrality, so that $n_e = \langle Z \rangle n_i$,

where $\langle Z \rangle$ is the average atomic number (e.g., $\langle Z \rangle = 1.5$ for a D- ^3He plasma with no impurities). Then the fusion power density (for a 50–50 mixture of two distinct ion components) is:

$$P_f = \frac{1}{4} \left(\frac{p}{1 + \langle Z \rangle} \right)^2 \frac{\langle \sigma v \rangle}{(kT)^2} E_f$$

where E_f is the reaction energy. The temperature dependence in the pressure-limited case is the term $\langle \sigma v \rangle / T^2$. The optimization of this expression generally happens at a lower temperature than the temperature for peak reactivity. For D-T, the optimum occurs at around $kT = 15$ keV; for D- ^3He the optimum is around $kT = 60$ keV and the optimum $\langle \sigma v \rangle / T^2$ for D- ^3He is about 50 times lower than the optimum in D-T. This explains why most fusion efforts are concentrated on D-T as the primary fuel for a reactor.

One should also note that inertial confinement schemes such as laser-driven or heavy ion-driven systems have no analogous pressure limit, and in general there is some time dependence in the fusion burn. The burn history has been modeled in computer codes which couple the hydrodynamics, fusion burn, and radiation loss over a complete cycle. It generally appears that a lower temperature (around 10 keV for D-T) is found to be optimum for an inertial fusion target.

Energy balance in fusion devices

The energy balance in a fusion reactor can be viewed in two different ways. One of these concerns the immediate balance of energy in the plasma, and the other is concerned with the power balance in the reactor as a complete system. Thus the plasma energy balance as well as the plant energy balance must both be considered.

First we will look at the plasma energy balance. We will consider a D-T plasma with $\langle Z \rangle = 1$ for simplicity. The conservation of energy equation then takes the form.

$$3n \frac{dT}{dt} = P_{f\alpha} + P_{inj} - P_{rad} - P_{hc},$$

with $P_{f\alpha}$ representing the portion of the fusion energy released in charged particles that is absorbed in the plasma (note that neutrons have negligible loss before exiting the plasma), P_{inj} represents the injected power from an external source such as neutral beam injection or radiofrequency heating (ohmic heating of the plasma by self-currents can also be included here), P_{rad} is the power loss from the plasma by radiation, and P_{hc} is the loss from the plasma by heat conduction. For the plasma to be in steady state, the time derivative dT/dt must be zero.

First we look at the alpha heating term $P_{f\alpha}$. The two particles ejected from a D-T fusion event are an alpha and a neutron. If we consider the D-T system to have zero momentum, Newton's law requires that the two reaction products have equal and opposite momenta. The energy released is small compared to the particle's rest masses, and therefore $E = p^2/(2m)$ for each particle. Since the alpha has a mass of (almost exactly) four times that of a neutron, the alpha carries away one-quarter of the neutrons energy, i.e. $E_\alpha = 0.2 E_f (= 3.52 \text{ MeV})$ and $E_n = 0.8 E_f (= 14.08 \text{ MeV})$. We then write $E_\alpha = E_f f_\alpha$, with $0 \leq f_\alpha \leq 0.2$, since all of the alpha energy might not be absorbed. We then can write.

$$P_{f\alpha} = \frac{n^2}{4} \langle \sigma v \rangle E_f f_\alpha.$$

Next we look at the injected power P_{inj} . We define a new quantity Q representing the ratio of the fusion power produced to the injected power: $Q = P_f/P_{inj}$. We then have the two energy gain terms as

$$P_{inj} + P_{f\alpha} = \frac{n^2}{4} \langle \sigma v \rangle E_f (f_\alpha + 1/Q)$$

Now we look at the loss terms. We will save a detailed look of the radiation loss until later and simply invent another ratio relating the radiation loss to the fusion power, $\chi_R = P_{rad}/P_f$. Plasma heat conduction is a subject of overwhelming interest in magnetic confinement fusion, but here we simply define a characteristic time τ_E for heat to leave the plasma. Since the energy content per unit volume is $3/2(n_e T_e + n_i T_i) = 3 nT$ for a $\langle Z \rangle = 1$ thermally equilibrated plasma, we have our overall energy balance.

$$P_{inj} + P_{f\alpha} - P_{rad} - P_{hc} = \frac{n^2}{4} \langle \sigma v \rangle E_f (f_\alpha + 1/Q - \chi_R) - \frac{3 n kT}{\tau_E} = 0.$$

The equation can be re-written by moving the last term to the right-hand side, dividing by n^2 and inverting, obtaining:

$$n\tau_E = \frac{3 kT}{(1/4) \langle \sigma v \rangle E_f (f_\alpha + 1/Q - \chi_R)}$$

Notice that only the product of density and energy confinement time appears in this equation and that all of the terms on the right are functions only of temperature.

We now turn our attention to the external energy balance for a fusion power plant. An energy flow diagram is shown as [Fig. 3](#). The analysis of this system was first done by J. D. Lawson in 1955 ([Lawson, 1955](#)). We envision a system where there is some

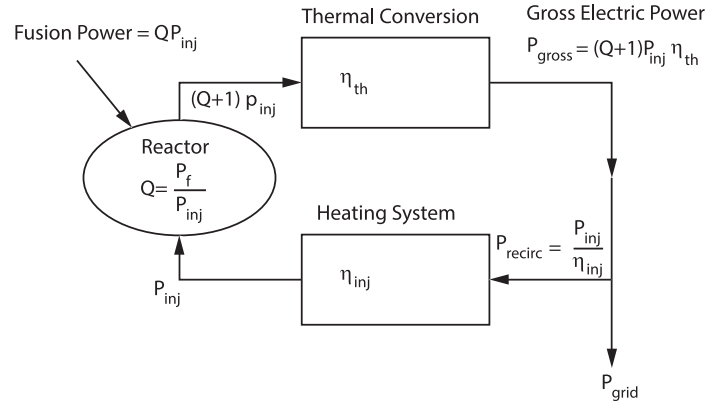


Fig. 3 Power flow diagram for a fusion power plant.

injection of energy into the fusion plasma, i.e. Q is finite. ($Q = \infty$ is referred to as “ignition”.) The energy multiplication is the primary figure of merit for the system. Lawson assumed that the thermal conversion step would consist of a conventional steam turbine and electrical generator. The efficiency of such units is governed by thermodynamics and must be less than the Carnot cycle efficiency. The maximum efficiency for a steam-based thermal converter is about $1/3$. Using this number, and assuming that the efficiency of the heating system is 1.0 , we can calculate the minimum value of Q to achieve breakeven condition, where no power is exchanged in or out of the grid. It is important to note that all of the injection power comes out as heat, along with the fusion power produced, and thus the output power is $(Q + 1)P_{inj}$. Setting the round-trip gain to one then yields

$$(Q + 1)\eta_{th} = 1,$$

which, for $\eta_{th} = 1/3$, yields $Q = 2$. (Note that $Q = 1$ often appears in the popular media as a breakeven condition, sometimes referred to as “scientific feasibility”, but $Q = 2$ is a better breakeven condition, and since the goal is to produce net power for the grid, $Q = 10$ or greater might be called “engineering feasibility”.)

We can then translate this into a required value of $n\tau_E$ for the plasma. This requires that we evaluate the radiation loss from the plasma. We look at the most optimistic case first, where only loss by bremsstrahlung is allowed. This type of radiation involves x-ray emission from the plasma caused by collisions between the electrons and ions. The x-ray photons produced have typical energies on the order of $kT_e \sim 10$ keV and are not reabsorbed by the plasma itself and not reflected by the walls of the reactor. The basic formula for bremsstrahlung power loss is

$$P_{brem} = 5.35 \times 10^{-37} n_e \sum_i n_i Z_i^2 (kT_e)^{1/2} \text{W m}^{-3},$$

with MKS units for all quantities except the electron temperature kT_e , which is given in keV. Note the sum over all ion species weighted by the square of their atomic numbers. Bremsstrahlung is increased by the presence of even small quantities of high- Z material, and furthermore, radiation from partially ionized high- Z ions (line and recombination radiation) will be present as well. But we can quantify the total radiation loss by using a simple model where all of the ions have a fictitious charge Z_{eff} . Then $n_i = n_e/Z_{eff}$ by charge neutrality and the above formula reduces to

$$P_{rad} = 5.35 \times 10^{-37} n_e^2 Z_{eff} (kT_e)^{1/2} \text{W m}^{-3},$$

and we see that Z_{eff} is really just the ratio $P(\text{dirty})/P(\text{clean})$ for the radiation found in the real, “dirty” plasma compared to the radiation expected from a pure D-T plasma with $Z = 1$ and the same electron density. Experimentally the best experience in large magnetic fusion devices finds Z_{eff} of around 2–3.

Now we can calculate the radiation parameter χ_R . A value for the reactivity is required. For D-T at $kT = 15$ keV, we have $\langle \sigma v \rangle = 2.71 \times 10^{-22} \text{m}^3 \text{s}^{-1}$. This gives a volumetric fusion power of $(n^2/4) \langle \sigma v \rangle E_f = 1.9 n_{20}^2 \text{MW m}^{-3}$. (Here n_{20} is the DT ion density in units of 10^{20}m^{-3} , a convenient quantity because a typical ion density in a magnetic confinement device is around 10^{20}m^{-3} .) Then χ_R at $kT_e = 15$ keV is $0.011 Z_{eff}$, which is to say that the radiation loss with $n_{20} = 1$ and $Z_{eff} = 2$ is around 41.4kW m^{-3} whereas the alpha heating power $= 0.2 P_f = 380 \text{kW m}^{-3}$, or about $10 \times$ larger. The energy confinement parameter is, for $Q = 2$:

$$n\tau_E = 5.85 \times 10^{19} \text{m}^{-3} \text{s}.$$

Lawson actually calculated this at a temperature $kT = 10$ keV, with a smaller reactivity, and got an easy-to-remember value for what is what we call the “Lawson criterion” for a reactor beyond breakeven:

$$n\tau_E \geq 1.0 \times 10^{20} \text{m}^{-3} \text{s}.$$

For a typical magnetic fusion device, this means a confinement time of one second.

We can carry this result over to the inertial confinement concept. “Inertial confinement” simply means that the explosive disassembly time of a compressed capsule of DT fuel is long enough to allow the Lawson criterion to be met. This time is given approximately by

$$\tau_d = \frac{R}{4c_s}.$$

Here R is the radius of the capsule and c_s is the sound speed, given by

$$c_s = \left(\frac{\gamma_e k T_e + \gamma_i k T_i}{m_i} \right)^{1/2}.$$

It is usually assumed that the adiabatic constant γ is 1.0 for electrons (isothermal) and 5/3 for the ions (adiabatic). We can write the density $n_i = \rho N_A / M_{DT}$. It is customary to use cgs units in inertial confinement applications, so Avogadro’s number $N_A = 6.02 \times 10^{23}$, and ρ is in g cm^{-3} . Putting these calculations together, we have.

$$n\tau_E \geq 5.2 \times 10^{14} \rho R \text{ cm}^{-3} \text{ s}.$$

This estimate does not include the time-dependent burn history due to compression and heating (see, e.g., [Fraleigh, 1974](#)), and a more realistic estimate drops this by a factor of roughly five, and then the equivalent Lawson criterion is

$$\rho R \geq 1 \text{ g cm}^{-2}.$$

Confinement approaches

As described above, the two basic types of confinement schemes are magnetic confinement and inertial confinement. Both of these approaches have evolved over more than 70 years of experimental and theoretical research. A historical view of the fusion program is given in ([Morse, 2018](#)).

Magnetic confinement

An early attempt at a magnetic confinement fusion device was done at the aeronautical laboratory run by NACA, the predecessor to NASA, at Langley Field near Washington, D. C. in 1938. It used a four-foot diameter aluminum torus which was covered with a copper-wire coil, which created a toroidal (in the $\hat{\phi}$ direction in cylindrical (R, ϕ, Z) geometry) magnetic field. No attempt at inducing a plasma current was done and confinement was poor. Further experimental attempts started after World War II, and most of these used some form of the “pinch effect”, where a current was passed through the plasma either along a cylindrical axis (a Z-pinch) or induced by a transformer effect in the $\hat{\phi}$ direction in a glass torus. These experiments all showed the development of instabilities causing distortion of the plasma column in the direction θ perpendicular to R and Z in the Z-pinch and perpendicular to the minor radius r and $\hat{\phi}$ in toroidal geometry, and described by a perturbation of form $\vec{\xi} \sim \exp.(im\theta)$, with $m = 1$ called a “kink” instability and $m = 0$ a “sausage” instability. These instabilities are fairly well explained by a model of the plasma assuming that it is a highly conductive fluid. This is called the ideal magnetohydrodynamic (MHD) model. The root cause of these instabilities was identified as the propensity of the plasma to seek a state of lower free energy than the symmetrical equilibrium by growing these perturbations. It was also found that adding an extra magnetic field along the $\hat{Z}$ -direction in the Z pinch or in the $\hat{\phi}$ direction stabilized these modes. [Fig. 4](#) shows the evolution of these magnetic fusion concepts.

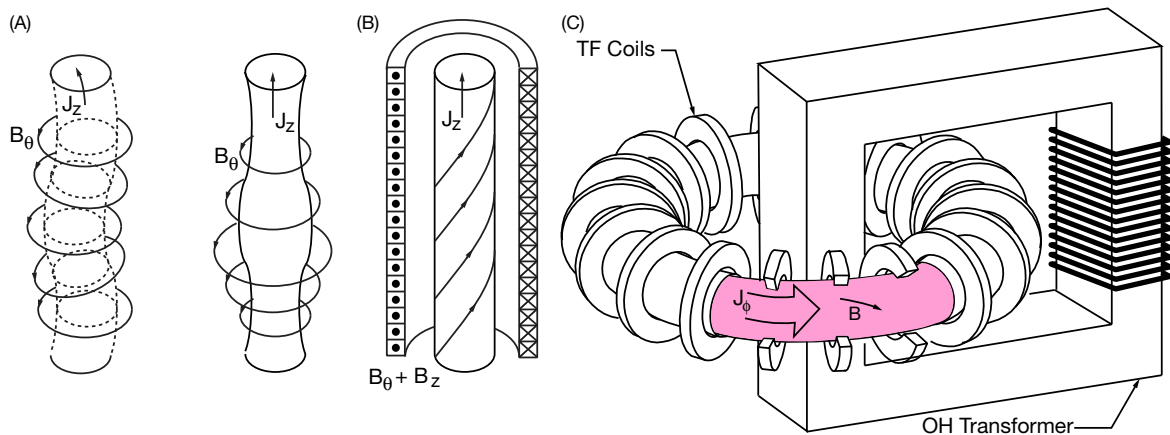


Fig. 4 Evolution of the pinch concept. (A): Kink and sausage instabilities in z-pinch. (B) Stabilized z-pinch. (C) Tokamak schematic.

The Russian tokamak concept is illustrated in Fig. 4C. While the first tokamak was built in the early 1950s, the concept was not adopted by others until a definitive measurement of the electron temperature in the Russian tokamak T-3 was done by a British team using laser Thomson scattering in 1969, and they reported that the electron temperature kT_e was greater than 1 keV in some shots. Tokamaks have since been built in almost every country in the world engaged in fusion research, and an international project called ITER (International Thermonuclear Experimental Reactor) is underway. This € 20 billion device is expected to generate about 500 MWth of fusion power with a gain factor Q of around 10.

While the MHD stability issues are generally mitigated in the tokamak approach, some problems remain to be completely understood and solved. At places in the plasma where the magnetic field closes on itself, (called the “rational surfaces”), the ideal MHD model breaks down and finite resistivity must be considered. The plasma has a propensity to grow islands of private flux at these places, and these islands can grow and eventually overlap, causing stochastic magnetic fields and anomalously high electron heat conduction. There are some trapped particles in the machine caused by the variation of the magnetic field strength along the magnetic lines, and these trapped particles cause island growth at lower plasma pressure than otherwise. These “neoclassical tearing modes” (NTMs) can limit plasma performance. Magnetic and electrostatic turbulence can also have a negative effect on confinement. Physics at the plasma edge is also much richer and potentially deleterious than originally thought, with “edge localized modes” (ELMs) being troublesome in some instances. In general, electron heat conduction is found experimentally to be a factor of $100 \times$ to $1000 \times$ that predicted from neoclassical theory. There are, however, empirical models from fits to experimental data that suggest, using a type of “wind tunnel scaling”, that ITER is large enough to deliver its promised performance.

Inertial confinement

An alternate path to producing fusion energy was inspired by the development of thermonuclear weapons. The first successful fusion-based nuclear weapon, called Ivy Mike, was detonated in 1952. This design was based on the D-D fusion reactions. The cryogenic deuterium fuel was heated and compressed by a fission device. The so-called “H-bomb” was refined and made light enough to mount on a missile over the next decade. Weapon physicists were interested in developing a “microfusion” experiment, to test physics concepts on a laboratory scale. Recently developed laser technology was an obvious choice for the driver. (In addition, inertial fusion concepts using heavy ions as a driver have been explored.) There were also tests of small D-T capsules in the proximity of an exploding fission primary, and these experiments were named Centurion/Halite and eventually were disclosed publicly. As a result, supporters of inertial confinement approaches claim that these efforts are an interpolation in the space of fusion yield vs. driver energy, whereas magnetic fusion is an extrapolation into the unknown.

Within the inertial confinement concept, there are two main variants. Fig. 5 shows the main approaches. There are schemes where the capsule is illuminated with the beams directly, and an ablation front is generated in the capsule material. By momentum conservation, remaining capsule material and fuel is accelerated radially inward, causing compression and shock heating. Alternatively, the driver energy can be directed to the interior wall of a high-Z container with some entrance holes: a “hohlraum”. The function of the hohlraum is to convert the laser/heavy ion energy into x-rays, which then bathe the capsule containing the fuel with high-energy photons. This “indirect drive” approach has the advantage that although laser photons cannot penetrate into the capsule plasma when that plasma is dense enough to reflect them, which happens when the critical density $n_{\text{crit}} \geq m_e \epsilon_0 \omega_L^2 / e^2$ (m_e = electron mass, ω_L = laser frequency, e = electron charge, ϵ_0 = permittivity of free space), x-ray photons are at a much higher frequency and penetrate the capsule easily. A disadvantage, however, is that other pathways for energy loss can be present because of the extra conversion step.

Just as magnetic fusion has discovered technical hurdles on the way to achieving breakeven conditions, inertial fusion has encountered challenges as well. The high compression factors required (to $100 \times$ to $1000 \times$ liquid density) can cause hydrodynamic instabilities such as Rayleigh-Taylor, Kelvin-Helmholtz, and Richtmyer-Meshkov instabilities to grow and cause mixing of the fuel and ablator (Morse, 2018). The high amplitudes of the laser electromagnetic waves can cause parametric instabilities to grow,

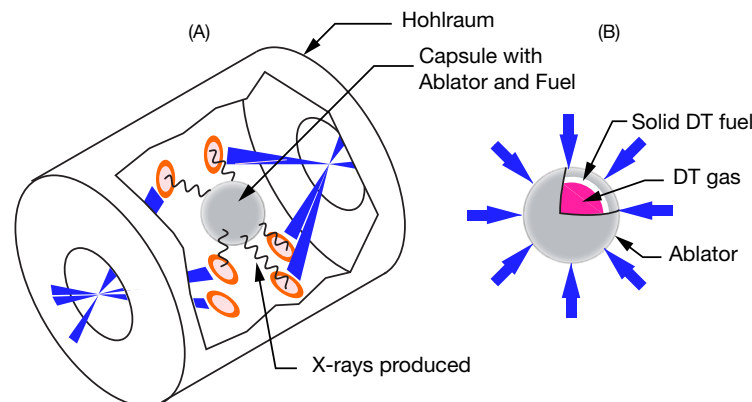


Fig. 5 Schematics of two inertial fusion concepts. (A) Indirect drive. (B) Direct drive.

leading to conversion of the waves into backscattered energy or energy absorption in unsuitable places. These modes are stimulated Raman scattering (SRS), stimulated Brillouin scattering (SBS), and intra-beam transfer (IBT). The choice of indirect drive moves these instabilities to the hohlraum wall from the capsule interface, but does not eliminate the problem.

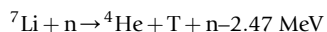
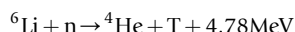
The largest inertial confinement fusion experiment is the National Ignition Facility (NIF) at Livermore, Ca USA. It is currently operational with 2.1 MJ of laser energy in 192 beams into an indirect-drive target. A key feature of this experiment is the conversion of the laser light from the neodymium-doped glass at a wavelength of 1.06 μm to one-third of this wavelength at 0.35 μm . The shorter wavelength mitigates the parametric instabilities to some extent. Presently the record fusion yield has only been about 3% of the laser drive energy, and much effort has been expended on capsule design, hohlraum design, laser timing and frequency spread in order to optimize performance.

Fusion technology

Fusion reactors are envisioned as advanced nuclear reactor technology and as such will share common elements with fission technology including a reactor containment structure, heat exchangers, gas or steam turbines, and electric generators. Some elements are very different from fission technology, however, and those differences are examined here.

Breeding blankets and first wall

The prime candidates for fusion fuel are D-T and D- ^3He as discussed previously. Both of these share a basic problem in that neither T nor ^3He are available in nature. But a symbiotic relationship exists between the fusion reactions and neutron-induced reactions in lithium. Lithium is present in nature in two isotopes, ^6Li and ^7Li . The neutron reactions are



The natural abundance of ^6Li is 6.7 at.%, with the balance being ^7Li . The exothermic ^6Li reaction can happen down to zero neutron energy, and the cross section is proportional to $1/v$, the neutron velocity, at lower energies. The ^7Li reaction has a threshold energy of around 2.8 MeV ($= -8/7\text{ Q}$) but has the advantage of returning a (slower) neutron. Since the D-T neutron energy is 14.08 MeV, this allows for the possibility that a mixture of the two lithium isotopes will have a tritium breeding ratio (TBR, tritium atoms made in the blanket divided by fusion neutrons produced) greater than one. It turns out that natural lithium is almost an optimum mixture, so that there is no need for enrichment or depletion of the ^6Li fraction. (Note that tritium breeding also solves the ^3He availability issue as well, because T decays to ^3He with a 12.3 y half-life.)

Different lithium formats have been proposed. Natural lithium, a liquid at blanket temperatures of around 330 $^{\circ}\text{C}$, has a high TBR but also a high electrical conductivity, which means that pumping it through a metallic structure into a magnetic field creates a situation where electromagnetic forces will require a large pumping power. Furthermore, it is reactive with air, water, and concrete. Other choices include Li_2O (a solid, perhaps gas-cooled), and FLiBe, a molten salt. These have lower TBR, but are less hazardous. Lithium-lead eutectics such as $\text{Li}_{17}\text{Pb}_{83}$ and $\text{Li}_{62}\text{Pb}_{38}$ also show high TBR as well as good energy multiplication, but have a tendency to form an oxide layer which can change their performance. Guebaly (1997) gives some details in the consideration of candidate blanket materials.

Equally important to the choice of the tritium breeding medium is the choice of first wall and structural materials. The first wall is directly exposed to fluxes of neutrons, heat, and particle exhaust from the plasma. The bulk of the particle exhaust in the form of charged particles is taken away by the magnetic field outside the burning plasma into a divertor structure, typically envisioned as an actively cooled tungsten or carbon fiber composite. The remaining hot neutral atoms and neutrons interact directly with the first wall. Since 80% of a DT fusion reaction comes out in the neutron channel, this represents the largest power component in the reactor. The 14 MeV fusion neutrons pose a threat to radiation damage on the first wall which is rather different from a fission reactor because at this higher neutron energy, more complex nuclear reactions such as (n,α) and (n,p) can take place, and these result in helium and hydrogen gas, respectively, building up in the metal matrix of the first wall. The propensity for radiation damage in materials is usually stated in "displacements per atom," or dpa; for fusion reactors, it is usually the dpa-to-helium ratio that is most important. Candidate wall and structural materials now include composites such as SiC/SiC, composite tungsten and tungsten/rhenium alloys, and reduced activation ferritic/martensitic (RAFM) and oxide dispersion strengthened (ODS) steels.

Heating systems for magnetic fusion

While creating ignited ($Q = \infty$) fusion plasma is an ultimate goal, near-term devices (including ITER) rely heavily on supplementary heating systems. While currents running through the plasma provide some ohmic (I^2R) heating, the plasma resistivity scales as $T_e^{-3/2}$ and thus becomes negligible at thermonuclear temperatures. Without some supplementary heating, there is a gap at around 2–5 keV where neither alpha heating nor ohmic heating can balance the heat loss. The two proven candidates for providing enough energy deposition to close this gap are neutral beam injection and radiofrequency (RF) heating.

Neutral beam injection (NBI) starts by making a beam of charged particles in an accelerator and then passing these ions through a gas cell which turns these energetic ions into neutrals by charge exchange. These neutrals can cross through the magnetic fields in the plasma and deposit their energy in the plasma's interior. The energy required depends on the size and particle density of the plasma: for the earlier TFTR experiment, 120 keV neutral deuterons were sufficient; for JET, 160 keV deuterons are available. But for the ITER device, beams with energies to 1.0 MeV are planned. While the former NBI systems used a relatively straightforward positive ion system, the latter will require the generation of negative H^- and D^- ions, which is done using reactive cesium metal. Issues still in question are the robustness of these high-voltage systems to electrical breakdown and thermal loads on accelerator grid structures.

RF heating systems have been used in several frequency ranges, starting with ion cyclotron resonance heating (ICRH) systems from 30 to 100 MHz, lower hybrid heating at 1–5 GHz, and electron cyclotron heating (ECH) at 90–170 GHz. ICRH systems require large antenna structures in the plasma, and the introduction of impurities into the plasma can be an issue. ECH technology is limited by the availability of high-power, high-frequency sources, and by the high power fluxes at vacuum windows into the plasma chamber.

Inertial fusion technology

A key element in taking inertial fusion to the reactor level is the development of high-efficiency (better than 10%), high repetition rate (10 Hz or higher) drivers. This can be contrasted with the large glass-based laser systems such as at the NIF facility, which have an efficiency of about one-half of 1% and can only be fired a few times a day. The current large laser systems (which produce light at 1.06 μm wavelength) also need to convert the light output to shorter wavelengths using frequency-conversion crystals, and alternative laser technologies such as KrF lasers may offer output at even shorter wavelength (0.25 μm) without needing this step. Heavy ion drivers offer some promise for high efficiency, but the experiments to date have been at a relatively small scale, and unknown beam optics and plasma-beam interactions at the target still remain as open questions.

See Also: Nuclear Fusion – Introduction and Overview.

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Atomic and Hydrogen Bombs

Jim Napolitano, Temple University, Philadelphia, PA, United States

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Glossary

Chain reaction Here referring to a fission chain reaction, can be established in a system that contains sufficient amount of fissile isotopes if at least one out of 2-to-3 neutrons emitted in a fission reaction is absorbed in another fissile nucleus and causes it to fission. A super-critical chain-reaction is established if more than one neutron from a fission is causing additional fissions. In a supercritical chain reaction the fission rate grows exponentially. In a nuclear reactor the chain-reaction is controlled to maintain a constant fission rate (fixed power).

Critical mass The minimum mass of a fissile element needed to generate a sustainable chain reaction.

Enhanced radiation weapon A thermonuclear device with a low fission yield and relatively high neutron radiation yield.

Fission weapon An explosive device whose yield derives from nuclear fission. Also known as an "Atomic Bomb."

Thermonuclear weapon A device that ignites a fusion reaction with a fission reaction, followed, typically, by a fast fission reaction. Also known as a "Hydrogen Bomb."

Introduction

The energy released in an exothermic chemical reaction is measured in electron volts (eV), whereas the energy released in a nuclear reaction is measured in millions of electron volts (MeV) (Napolitano, 2021). In other words, the energy released per kilogram of "fuel" is roughly a million times larger for a nuclear reaction than for a chemical reaction.

This simple fact was obvious immediately after a revolutionary discovery in Germany in the late 1930s. In December 1938 Otto Hahn and Fritz Strassmann chemically observed intermediate mass elements created when uranium were bombarded with neutrons. Within a month, Robert Frisch and Lise Meitner correctly interpreted this observation as the yet unknown nuclear fission reaction. Europe was in the midst of large scale conflict at the time, and this would soon grow into World War II. The intersection of these circumstances led to large scale Research and Development (R&D) efforts in both Germany and the United States to produce large amounts of fissile material, with the ultimate goal of developing enormously powerful weapons based on this discovery.

The combined R&D and manufacturing effort in the US was called the Manhattan Project (Reed, 2021). At the beginning of the Manhattan Project it was already known that the isotope ^{235}U underwent fission when bombarded by low energy neutrons, and that so does the newly discovered isotope ^{239}Pu . Therefore the Manhattan Project started with two parallel efforts to accumulate and isolate large enough amounts of each of the fissile isotope.

By the end of the war, the US had successfully built and deployed "atomic bombs" based on fission. On July 16, 1945, in the Alamogordo Bombing and Gunnery Range in the New Mexico desert, the first atomic bomb was exploded. It was based on the fission of ^{239}Pu and had an estimated yield equivalent to 20,000 tons (20 kT) of TNT. On August 6 and 9, bombs based on ^{235}U and ^{239}Pu , respectively, destroyed the cities of Hiroshima and Nagasaki in Japan. More than 100,000 people were killed, and a roughly equal number were severely injured.

Soon after the War, in 1949, the Soviet Union successfully detonated a fission weapon. Both the United States and Soviet Union developed much more powerful "hydrogen bombs" based on nuclear fusion in the 1950s, and the "nuclear arms race" had begun. Both countries began to amass arsenals of "strategic nuclear weapons" that were direct threats to each others' populations.

Other countries followed with their own weapons development programs. So called "Enhanced Radiation Weapons," also known as "neutron bombs," were developed in the 1960s, capable of destroying large populations with minimal damage to buildings and structures. These were deployed as "tactical nuclear weapons" for use on the battlefield, mainly in Europe.

Nuclear weapons have today faded into the background as more modern forms of warfare have been developed and deployed. Nevertheless, sizable nuclear arsenals still exist, and their existence is cataloged and documented, and publicly available. Research continues into the development of nuclear weapons, as well as into safeguards that are meant to deter their use during hostilities between governments.

Principles of atomic bombs: Fission weapons

The explosive release of nuclear energy in a fission weapon requires a rapidly escalating series of fission events. It is natural for a nucleus undergoing fission to release several neutrons in addition to the daughter nuclear fragments, and these neutrons can in turn generate fission in the surrounding fissile nuclei. This process is known as a *chain reaction*. In order for the chain reaction to proceed rapidly enough so that an explosion results, there need to be sufficient nuclei in proximity. This sufficient amount of nuclear material is called the *critical mass*, although the density is also important. That is, it is not enough for enough fissile nuclei to be present. They must be close enough to each other so that the chain reaction is a runaway event (supercritical).

It is not difficult to design an initial neutron source which would start the chain reaction. A typical choice is a mixture of ^{210}Po and ^9Be . The isotope ^{210}Po decays 100% through α -emission to the ground state of ^{206}Pb , so no other radiations are emitted in this decay. These α particles create neutrons through the reaction $\alpha + ^9\text{Be} \rightarrow ^{12}\text{C} + \text{n}$. Interestingly, this is the same combination of isotopes with which Chadwick made the measurements which demonstrated the existence of the neutron (Chadwick, 1932).

The next step in designing a fission weapon is to identify heavy isotopes that might make a suitable nuclear fuel. In order to have a significant fission probability when bombarded by low energy neutrons, the isotope should have even Z and odd N so that the pairing energy (Napolitano, 2021) leaves the compound nucleus in a highly excited state. The isotope must have a long enough half life so that it can be accumulated by extracting it either from ore or after synthesizing it in nuclear reactions.

It was known early on that the naturally occurring isotope ^{235}U ($Z = 92$, $N = 143$) was a good candidate for the fuel of a fission weapon. It has a half life of 700 million years so a sizable primordial abundance remains today as 0.7% of natural uranium, with ^{238}U making up over 99% of the rest. Enrichment techniques were developed in the Manhattan Project which accumulated many kilograms of uranium highly enriched in ^{235}U .

Research carried out in the US at the University of California at Berkeley, in 1940 and 1941, discovered two new elements just past uranium on the periodic table. These were named neptunium (Np, $Z = 93$) and plutonium (Pu, $Z = 94$). As expected, the researchers confirmed that the isotope ^{239}Pu underwent fission when bombarded by very low energy neutrons. See Norman et al. (2015) and references therein. The half life of ^{239}Pu is only 24 thousand years, so it does not occur naturally and must be created in nuclear reactors, where neutron capture on ^{238}U creates the short lived isotope ^{239}U which beta decays to ^{239}Np which in turn beta decays to ^{239}Pu . This technology was also developed during the Manhattan Project.

When ^{239}Pu captures a neutron, it is converted into ^{240}Pu that is not fissile but has a sufficiently high probability to undergo fission spontaneously—it emits 920 spontaneous fission neutrons per second per gram. This rate of spontaneous fission is large enough to lead to an intrinsic source of neutrons in the plutonium used for nuclear weapons and complicates this weapon design.

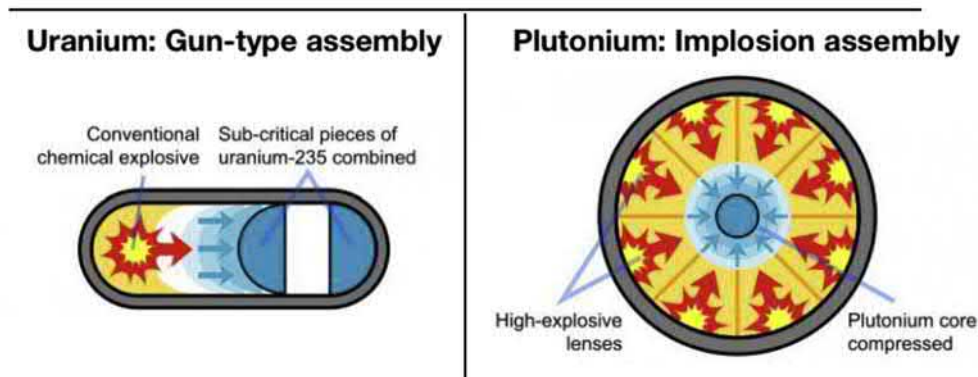
Given the difficulty in accumulating enough fissile material, it is of course critical to know how much of a particular isotope is necessary to achieve a critical mass. However, this is not a simple calculation (Chyba and Milne, 2014). Indeed, this determination was one of the major stumbling blocks during the Manhattan Project and other international efforts. For a given geometry, one needs to determine the size scale at which neutron production through (neutron-induced) fission is counterbalanced by the loss of neutrons through the surface. It is clear that a sphere is the right shape, since this maximizes the volume-to-surface ratio, but the “critical radius” of this sphere depends on details of the neutron interaction cross sections and its propagation through the volume. The purity of the fissile material is also important, as impurities not only dilute the fuel but may also absorb neutrons. The density of the material is also important, as a higher density decreases the mean free path (mfp; the mean distance the neutron travels until it collides with the fuel nucleus) of neutrons in the fuel.

Simple estimates can be made, and sequentially improved, but modern computational techniques are needed to make accurate determinations. The critical radius for Highly Enriched Uranium (HEU that is nearly pure ^{235}U) bare sphere at nominal uranium density is 8.4 cm. For ^{239}Pu , the bare sphere critical radius is 6.3 cm (Chyba and Milne, 2014).

After the fissile material production task is accomplished, a different challenge presents itself. How does one assemble a super-critical mass from sub-critical segments so that a runaway chain reaction follows? In the Manhattan Project, different solutions were used for the uranium and plutonium devices, as shown in Fig. 1, below. Both of these approaches lead to compressing the fissile core thus increasing the fissile material density while decreasing the surface area through which neutrons can escape. The compression of the fissile material makes an initially sub-critical core to a super-critical core. The compression reduces the mfp that is inversely proportional to the fuel atomic density. In a spherical configuration, for example, a factor of 2 reduction in radius (hypothetical; just for illustrating the principle) will result in a factor of 8 increase in the fuel density implying a factor of 8 decrease in the mfp. The mfp/radius ratio will increase by a factor of 4 and the neutron leakage probability will substantially decrease enabling a super-critical chain reaction to be established. As an example: the mean path-length an average fission neutron needs to travel in ^{235}U of nominal density ($N = 4.8 \times 10^{22} \text{ cm}^{-3}$) until it induces a fission is ($\sigma_f^{-1} N^{-1} =$) 17.1 cm; that is approximately twice the radius of a bare critical sphere (the cross section for an average fission neutron to induce fission in ^{235}U is $\sigma_f = 1.22 \times 10^{-24} \text{ cm}^2$). If the sphere is uniformly compressed to a radius of 4.2 cm, the mean path-length for fission becomes 2.3 cm that is significantly smaller than the sphere radius. Of course, neutrons change direction after each collision, may lose energy (inelastic collisions) and may even be captured so that the above numerical illustration is not a realistic simulation of the neutron evolution in the system but gives a numerical illustration of the effect of compression of the fissile material on the fission probability.

For a uranium device, two sub-critical masses of ^{235}U are assembled using a “gun-type” mechanism. Conventional chemical explosives are used to drive one sub-critical mass linearly where it collides and rapidly fuses with the other sub-critical mass. This is a relatively straightforward engineering problem, and was the solution of choice.

Assembly of an Explosive Critical Mass



<https://www.atomicheritage.org/sites/default/files/styles/large/public/Fission%20Bomb%20Assembly%20Wikimedia%20Commons.jpg>

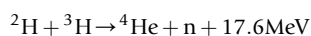
Fig. 1 The two ways to assemble for detonation a critical mass of fissile material in a fission weapon. The “Gun-type” mechanism, used for ^{235}U , fires one subcritical mass into another. The “Implosion” approach, used for ^{239}Pu , radially compresses the critical mass volume to achieve a super-critical density.

Unfortunately, this approach could not be used for a plutonium weapon. The plutonium extracted from the plutonium production reactors contains a small amount of the isotope ^{240}Pu which undergoes fission spontaneously. This required that the plutonium critical mass be assembled much more rapidly than can be achieved using a “gun-type” mechanism. A solution was devised that made use of a spherical lens of high explosives which radially compressed the plutonium core of the device. This was not as straightforward to engineer as the gun-type solution, requiring different types of chemical explosives which needed to focus their energy and to detonate synchronously around the spherical volume, but the result was nevertheless successful for both the Alamogordo test and for the Nagasaki bomb.

Principles of hydrogen bombs: Fusion weapons

It is a misnomer to refer to fission weapons as “atomic bombs,” as the explosion is due to a nuclear reaction, rather than an atomic (i.e. chemical) reaction. The term “hydrogen bomb” is more accurate, although it also has a serious misnomer with it, as much of the explosive energy from the largest of these weapons is due to fission of ^{238}U . A more accurate terminology for hydrogen bombs is “thermonuclear” devices.

The principle explosive reaction in a hydrogen bomb is the fusion of deuterium and tritium nuclei, generally called “D-T fusion” (Napolitano, 2021; Morse, 2021). In nuclear nomenclature, we write



This reaction releases about three times as much energy per unit atomic mass than does fission of ^{235}U or ^{239}Pu . There are also no limitations enforced due to critical mass, so any practical amount of fusion material can be assembled. This is one important reason that hydrogen bombs can be constructed with considerably more explosive power than fission-based weapons, typically 10 or more times the yield in the Hiroshima and Nagasaki bombs.

There is another important mechanism used to increase the yield of a hydrogen bomb. The isotope ^{238}U does not undergo fission with low energy neutrons because the neutrons and protons are all paired. However, higher energy neutrons have significant probability (here referred to as “cross-section”) for fissioning ^{238}U . The total cross section as a function of neutron energy is shown in Fig. 2, below, along with the cross sections for fission and for the release of two or three neutrons.

The ^{238}U fission cross section is effectively zero for thermal neutron energies, but rises sharply above 2 MeV, and is a large fraction of the total cross section at higher energies. Furthermore, the $(n,2n)$ and $(n,3n)$ cross sections are also large at high energies while the neutron capture cross-section at the MeV energy range is a small fraction of the fission cross section. As a result, high energy neutron interactions with ^{238}U generate more high energy neutrons, leading to a fission of one or more ^{238}U nuclei.

The additional energy needed for ^{238}U fission is found in the 14 MeV neutron released in D-T fusion. By packing in large amounts of natural uranium near the fusion fuel, again with no limitation due to critical mass constraints, the ^{238}U will undergo fission and add significantly to the explosive yield. In fact, weapons have been constructed this way which have demonstrated thousands of times the yield of the Hiroshima and Nagasaki bombs.

The successful development of thermonuclear weapons stems largely from the work of Edward Teller and Stanislaw Ulam, although the relative contribution of each is still a matter of debate. In late 1941, Enrico Fermi suggested to Teller that a fission device could be used to trigger fusion reactions. During the time he worked on the Manhattan Project, Teller devoted significant efforts toward the development of what he dubbed the “Super”—a thermonuclear weapon that could have thousands of times

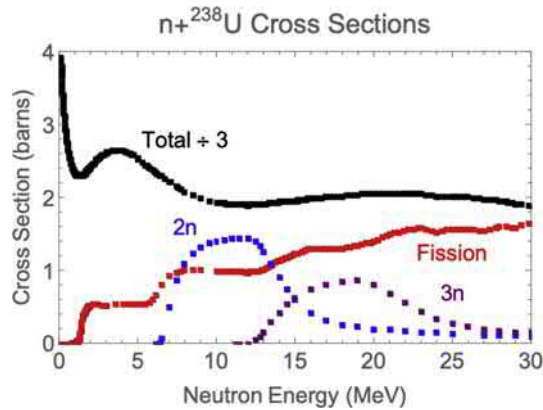


Fig. 2 Fission, $(n,2n)$, $(n,3n)$ and total cross sections in barns ($1 \text{ b} = 10^{-24} \text{ cm}^2$) of ^{238}U . The reaction probability is proportional to the cross-section. Source: Japanese Evaluated Nuclear Data Library.

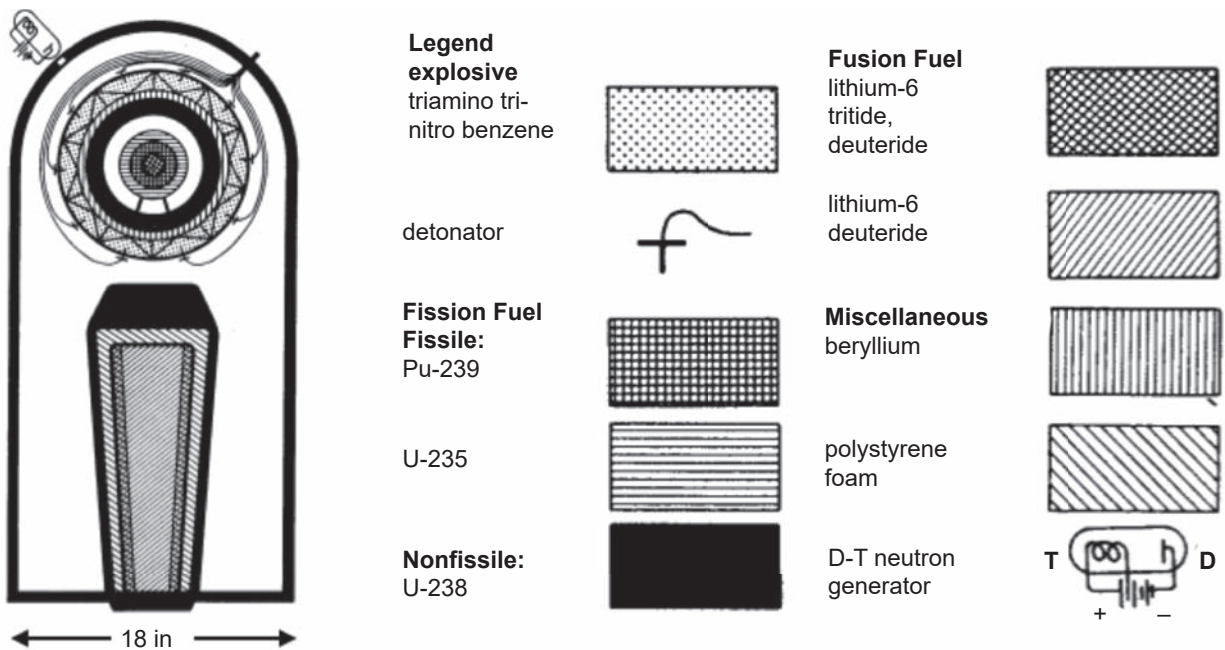
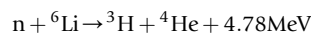


Fig. 3 A schematic of the fission-fusion-fission cascade style of thermonuclear weapon. A chemical explosion is detonated which implodes a fission device to achieve critical mass. Radiation from this explosion compresses the fuel column of $^6\text{Li}^2\text{H}$ and $^6\text{Li}^3\text{H}$. Fast neutrons from D-T fusion then impact the ^{238}U shell which then undergoes fission, adding enormously to the overall explosive yield. From Morland H (1979) *The H-bomb Secret*. Progressive Magazine.

the explosive power of a fission weapon. Many details of the construction of thermonuclear weapons remain highly classified. However, one of the key ideas, which may have been provided by Ulam, is apparently the physical separation of the fission trigger from the fusion fuel.

It is easiest to describe the series of events leading to the final explosion by following a specific design. Fig. 3, below, from Morland (1979), shows a schematic of a 300 kT thermonuclear weapon.

Focus first on the fusion fuel, lithium hydride that makes use of the isotope ^6Li , which is about 7% of natural lithium, and the hydrogen isotopes deuterium and tritium. The lithium binds the hydrogen isotopes into a solid form, which is not so unwieldy as gaseous hydrogen. Secondly, in the presence of neutrons, the reaction



proceeds copiously and therefore “breeds” tritium. This is particularly attractive because tritium is radioactive with a half-life of 12 years, so producing large amounts of tritium during the explosion itself makes it easier to increase the yield.

The fusion fuel is ignited by a fission explosion, triggered with an implosion device similar to that used in the plutonium bombs from the Manhattan Project. The fission explosion fills the cylindrical ^{238}U cavity with extremely intense electromagnetic radiation. This radiation comes to blackbody equilibrium on a rapid time scale, much shorter than the time scale of materials flying out from the fission explosion, owing to the fact that the speed of light is much larger than the mechanical speed of the explosion debris. The resulting radiation pressure compresses the lithium hydride fuel, raising its temperature to the point where fusion is ignited.

Fast 14 MeV neutrons from D-T fusion initiate fission in the ^{238}U shell, adding to the net explosive yield. The fission fragments are all unstable and add to the radioactive fallout from these sorts of weapons.

The first hydrogen bomb, with scale large enough so that the explosive yield was primarily due to hydrogen fusion, was detonated by the United States in November 1952. Code named "Ivy Mike", this device used cryogenic hydrogen instead of lithium hydride, and was meant to prove the principle of the staged fission-fusion-fission design. The 10 mega-ton explosion occurred on the Enewetak Atoll in the Pacific Ocean.

Enhanced radiation warheads

Soon following their development, hydrogen bombs, with their very high explosive yield, became the standard weapons in the nuclear arsenals of the United States, the Soviet Union, and other countries cultivating their nuclear capabilities. These were strategic nuclear weapons, that would be delivered by long range missiles, bombers, or submarines, to one country's soil from another.

The 1960s saw research and development of a different, so-called "tactical" nuclear weapons. These would be used on the battlefield, launched by armies attacking one another in an otherwise conventional theater of conflict. These were naturally weapons with much smaller yield, given the potential delivery range of a battlefield means of transportation. Other considerations, including the ability to maximize damage to troops, while minimizing the damage to buildings and noncombatants, figured into the designs of these weapons.

A particularly promising direction for this research was the "Enhanced Radiation Warhead," commonly called the "Neutron Bomb." This was essentially a hydrogen bomb where the first fission stage was designed with a minimal yield, and a large charge of lithium hydride was included for a large fusion yield. The ^{238}U components were removed so that the third stage that is responsible for the large yield of H-Bombs would not play a role. This meant that nearly the entire yield of neutrons from the D-T reaction would leave the explosion site.

It is this large neutron flux that is the source of the "enhanced radiation" of these weapons. With the fission blast at 10 kT or less, the blast radius could be confined to within 0.5 km, while the neutrons would deliver a dose that would cause "immediate permanent incapacitation" to troops within a radius of about 2 km. Delivery systems would be able to reach targets between 20 and 1000 km, depending on the design and deployment details. See [Kaplan \(1978\)](#).

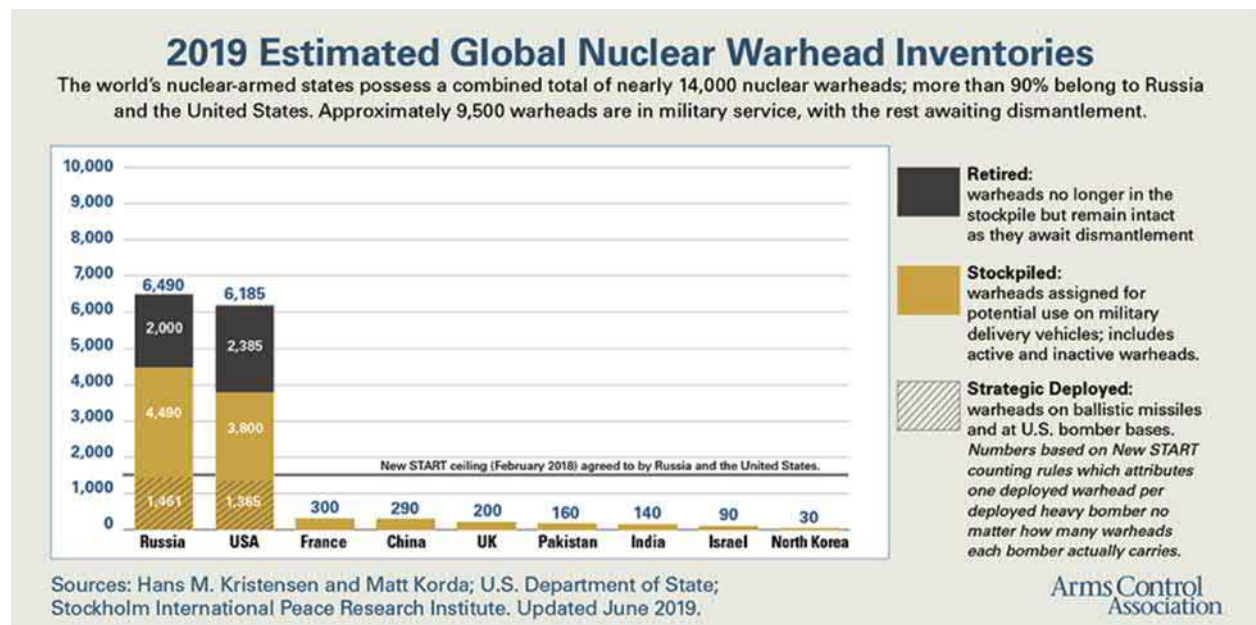


Fig. 4 Global nuclear warhead inventories in 2019, as compiled by the Arms Control Association (ACA), an international organization that monitors worldwide nuclear stockpiles.

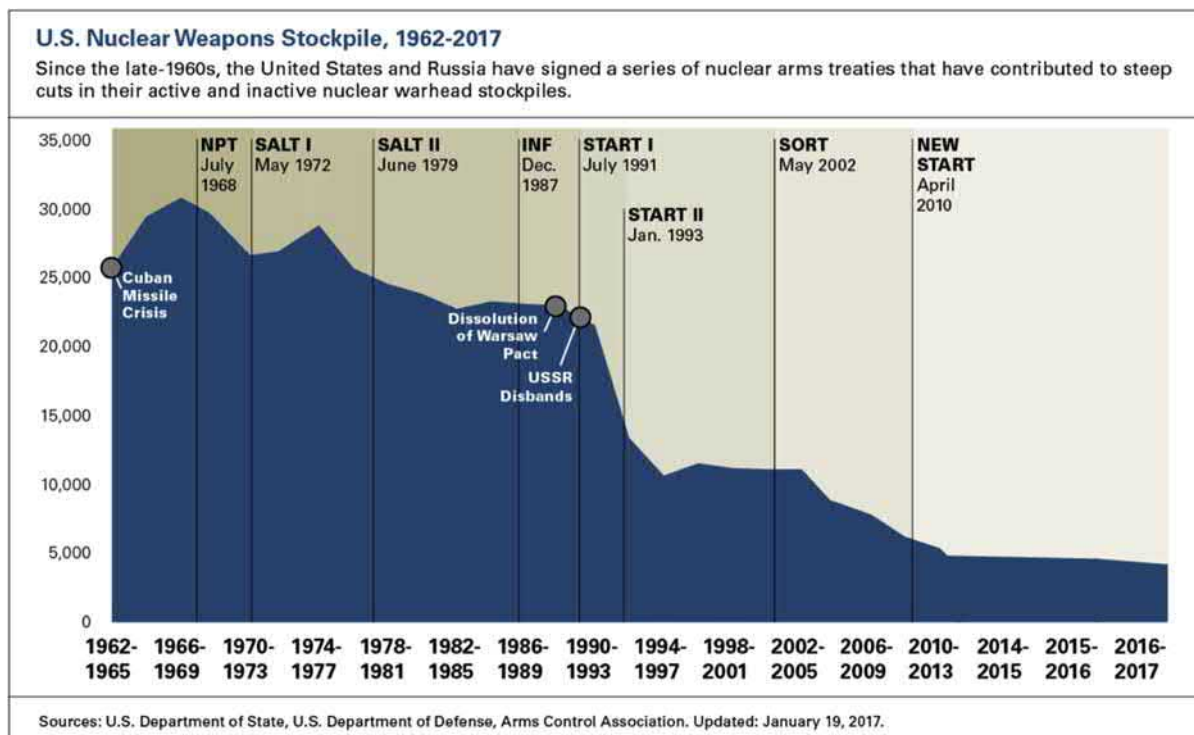


Fig. 5 The size of the US nuclear weapons stockpile over nearly 60 years, starting at the height of the Cold War with the Soviet Union. This is a dramatic demonstration in the power of negotiation to reduce the global nuclear threat.

The published arsenal of nuclear weapons

Nuclear weapons stockpiles by country are researched and recorded by the Arms Control Association (ACA) at <https://www.armscontrol.org/>. Fig. 4 shows a graphic from the ACA that summarizes the strategic nuclear weapon warheads in possession by all countries with nuclear arms capabilities.

Over 90% of the weapons are in the hands of the United States and Russia, and roughly one-third of these are retired and slated for dismantling. An even smaller fraction is actively deployed on ICBMs and long-distance bombers.

Thanks to the Nuclear Nonproliferation Treaty and other international efforts, the number of nuclear weapons in the world has decreased considerably since its peak in the early 1960s. Fig. 5, also from the ACA, shows the history of the US nuclear warheads stockpile over this period.

This is testimony to the success of negotiating for the common good of all parties, but it remains to remove the last of the nuclear weapons stockpiles.

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Phenomenology of Neutron Interactions

Fred E. Wietfeldt, Department of Physics and Engineering Physics, Tulane University, New Orleans, LA, United States

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Neutron energy regimes

A neutron can be freed from the atomic nucleus by various means including nuclear fission, fusion, proton spallation, and induced reactions such as ${}^9\text{Be} + \alpha \rightarrow {}^{12}\text{C} + n$ or ${}^9\text{Be} + \gamma \rightarrow {}^8\text{Be} + n$. The neutron is emitted with kinetic energy comparable to the neutron binding energy from the strong force in the nucleus, about 1–10 MeV (MeV = 10^6 electron Volts). In the case of spallation the neutron energy may be higher due to the incident proton's energy. Neutrons in this energy range are called fast neutrons. For many applications it is desirable to reduce the neutron energy; this is typically done by moderation in a low mass solid or liquid material. Neutrons that reach thermal equilibrium with a moderator are called thermal neutrons and have typical energy of 20–30 meV (meV = 10^{-3} electron Volts). Neutrons that are partially moderated are in the intermediate energy range, and epithermal if nearly thermalized.

The phenomenology of neutron interactions with matter depends very much on neutron energy. This is mainly a consequence of quantum mechanics; slow particles exhibit wave-like behavior while fast particles do not. The determining factor is the neutron's de Broglie wavelength λ , which is equal to Planck's constant divided by its momentum. (Planck's constant $h = 6.626 \times 10^{-34}$ J s = 4.136×10^{-15} eV s.) For example a fast neutron produced from a fission reaction (about 2 MeV), has $\lambda = 20$ fm (fm = 10^{-15} m), too small to be of consequence. A thermal (0.025 eV) neutron has $\lambda = 0.18$ nm (nm = 10^{-9} m), comparable to the distance between atoms in a liquid or solid, so its wave properties emerge to play an important role in its interactions. For some scientific purposes, most notably neutron scattering research, it is useful to reduce neutron energy to below the thermal regime. These are called cold neutrons (Fig. 1).

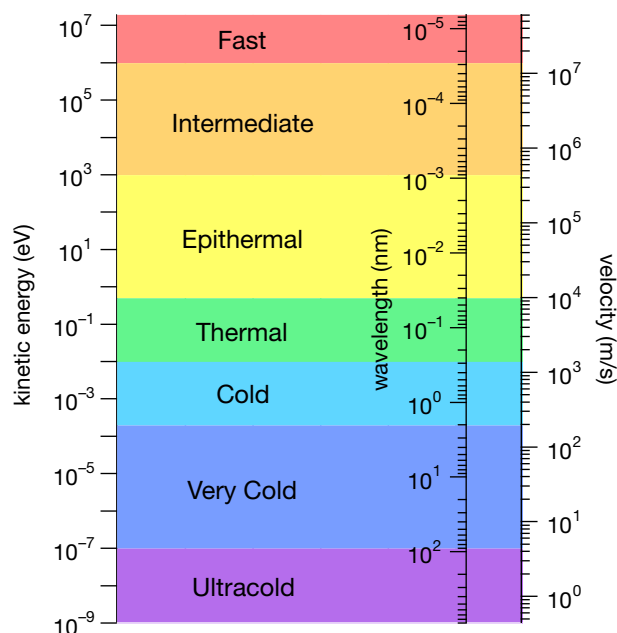


Fig. 1 Approximate neutron energy regimes showing the associated velocities and de Broglie wavelengths.

Ultracold neutrons comprise the lowest energy regime. The key feature of ultracold neutrons is that they can be trapped and stored in the laboratory. The neutron effective potential energy of a material surface, essentially a volumetric average of the strong force potential energy of its atomic nuclei, is typically about 100 neV (neV = 10^{-9} electron Volts), so a neutron with kinetic energy below that cannot penetrate the surface and must be reflected. The neutron's magnetic dipole moment is 60 neV/Tesla so neutrons that are roughly 100 neV or below can be energetically confined in a laboratory magnetic field. The mgh potential energy of a neutron in the gravitational field at the Earth's surface is 103 neV/m. Remarkably, while the origins and natural scales of the strong, electromagnetic, and gravitational forces are very different, they converge at a neutron energy of about 100 neV to enable practical trapping of neutrons using all three forces, either singly or in combination. Ultracold neutrons are produced by specialized techniques and are used to make precise scientific measurements of certain neutron properties such as its decay lifetime and hypothetical electric dipole moment.

Fast and intermediate neutron interactions

Neutrons with energy in these regimes interact as particles with atomic nuclei in matter through a variety of nuclear reactions. The probability of a reaction such as neutron capture or scattering by an atom is usually expressed as a *cross section* σ , which has units of area, usually barns where $1 \text{ b} = 10^{-24} \text{ cm}^2$. The probability P for the reaction to occur when a neutron travels a small distance δx in a material with isotopic number density N (atoms of that isotope per unit volume) is given by $P = \sigma N \delta x$.

In general, nuclear reactions are classified as *direct* when the time of interaction between the projectile, a neutron in this context, and the nucleus is comparable to the time it would take to traverse it, i.e. the nuclear diameter (a few fm) divided by the neutron velocity. Direct reactions are characterized by a weak dependence of the cross section on neutron energy and an angular distribution of final state particles that is peaked in the incident direction. There is a strong correlation between the incoming and outgoing particle momenta. Direct reactions are observed for incident neutron energies $> 10 \text{ MeV}$. Nuclear reactions are classified as *compound* when the time of interaction is much longer than the traversal time, implying a two step process. The projectile is first absorbed to form an excited compound nucleus that later decays with a characteristic lifetime and emits the final state particle(s), which are weakly correlated with the direction of the incident particle.

Cross sections for compound reactions depend strongly on energy and feature resonances (peaks) where the incident energy closely matches that of a particular state in the nuclear structure of the compound nucleus. Fig. 2 shows an example of resonance peak structure for neutron elastic scattering on ^{115}In . Nuclear resonances take the basic form of a Lorentzian distribution

$$\sigma(E) = C \frac{\Gamma^2}{(E - E_0)^2 + \Gamma^2/4}$$

where E_0 is the resonance energy and Γ is the resonance width, related to the reaction time τ by the Heisenberg uncertainty principle: $\Gamma \approx \hbar/\tau$. A short-lived compound nucleus produces a broad resonance, and a long-lived compound nucleus a narrow one. Resonance reactions are superimposed on the approximately constant scattering probability associated with the overall nuclear potential. These two phenomena may interfere quantum mechanically to produce characteristic dips in the cross section at the low energy side of each resonance. Such dips can be seen in Fig. 2.

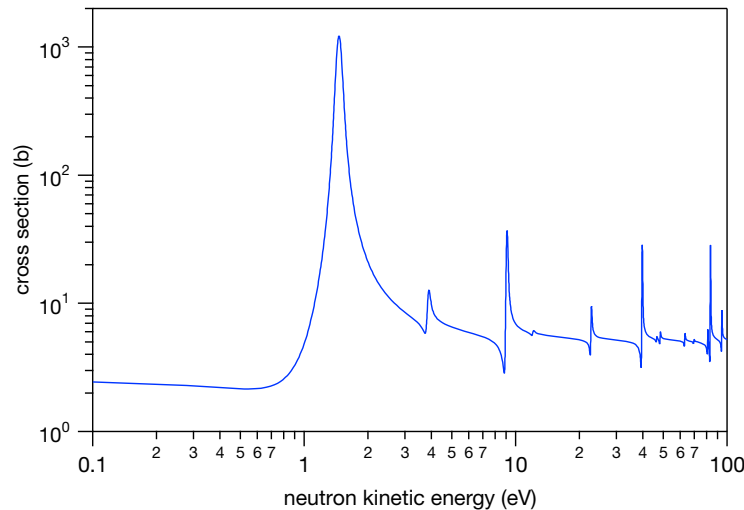


Fig. 2 The low-energy elastic neutron scattering cross section of ^{115}In , as a function of incident energy, showing resonance peak structure. This isotope has a strong resonance at 1.44 eV (ENDF/B-VIII).

In elastic scattering a single neutron is emitted but in a different direction than the incident neutron, leaving the structure of the nucleus unchanged, and total kinetic energy is conserved. If this not the case the reaction is called inelastic. Examples of inelastic reactions are (n,n') , (n,γ) , (n,p) , and $(n,2n)$, written in standard notation where the incident particle appears before the comma and the final particle(s) after. Inelastic (n,n') reactions leave the target nucleus in an excited state. At low energy the electrostatic energy (Coulomb barrier) in medium and heavy nuclei inhibits emission of a proton into the final state so (n,p) reactions become important above about 1 MeV. The (n,p) reaction cross section for a medium nucleus, ^{56}Fe , can be seen in Fig. 3. When the excitation energy of the compound nucleus is high enough it is possible to emit multiple neutrons into the final state. This can be regarded as a statistical process. As the “temperature” of the compound nucleus increases it becomes more probable for a higher number of neutrons to boil from its surface. The excitation energy needed for this is high in medium nuclei and lower in neutron-rich heavy nuclei such as ^{238}U where neutrons are more weakly bound. These effects can be seen in Figs. 3 and 4.

In medium and heavy nuclei (atomic number $Z \geq 18$), and neutron energy < 0.5 MeV, only elastic scattering, (n,n') , and radiative capture, (n,γ) , reactions are energetically allowed. In an (n,γ) reaction, a possible reaction for most nuclei, the neutron is absorbed by the nucleus to form an isotope of the same element with an increase of one unit of atomic mass. Due to the binding energy of the strong force, a small amount of mass is lost in the process. This is converted into gamma ray energy via the Einstein relation $E = mc^2$ and one or more gamma rays totaling about 8 MeV are emitted. An (n,γ) reaction may cause the atom to become radioactive.

In light nuclei ($Z < 18$) the Coulomb barrier is small enough to allow charged particles to escape the compound nucleus with minimal excitation and therefore (n,p) or (n,α) reactions are possible at low incident energy. These reactions are very useful in

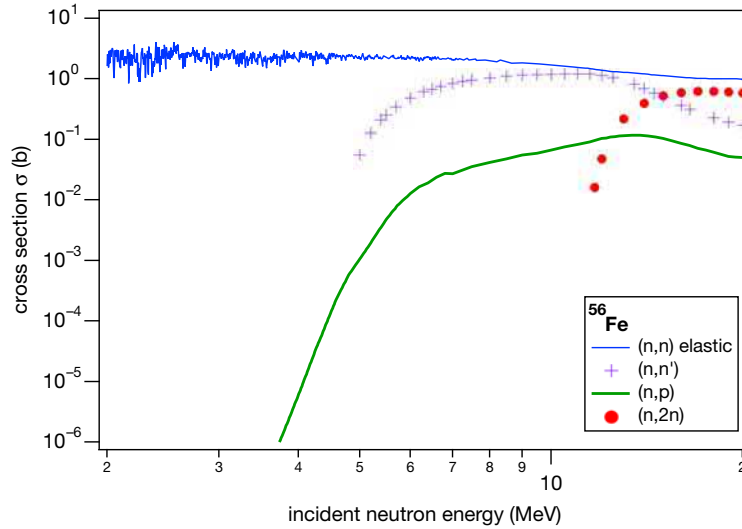


Fig. 3 Elastic (n,n) and inelastic (n,n') , (n,p) , $(n,2n)$ neutron cross sections of ^{56}Fe as a function of incident energy (ENDF/B-VIII).

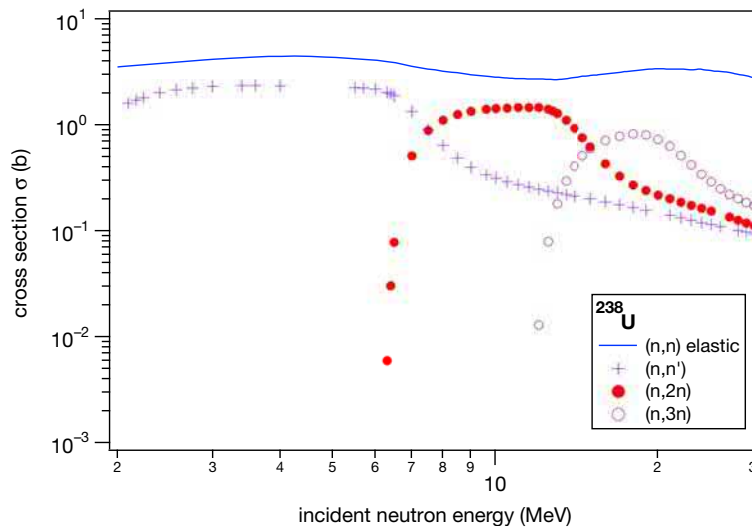


Fig. 4 Elastic (n,n) and inelastic (n,n') , $(n,2n)$, $(n,3n)$ neutron cross sections of ^{238}U as a function of incident energy (ENDF/B-VIII).

Table 1 A partial list of low energy (n,p) and (n, α) reactions. Q is the energy released.

Isotope	Natural abundance (%)	Reaction	Q (MeV)	Thermal neutron absorption cross section σ_0 (barns)
He-3	0.00013	(n,p)	0.765	5333
Li-6	7.5	(n, α)	4.78	940
B-10	19.9	(n, α)	2.79	3837
N-14	99.6	(n,p)	0.624	1.86
S-33	3.1	(n, α)	3.49	115
Cl-35	75.4	(n,p)	0.615	0.489

neutron detectors, and low background neutron shielding due to the absence of penetrating secondary gamma radiation. A partial list of these is given in [Table 1](#).

Slow neutron interactions

At thermal energy and below the neutron's de Broglie wavelength is much larger than the size of a nucleus and the range of the strong nuclear force, so it is necessary to treat the incident neutron in a reaction as a quantum mechanical matter wave, not as a small particle. In this picture the nucleus is point like and the orbital angular momentum of the system is zero, greatly simplifying the physical description. A single parameter, the *neutron scattering length*, is able to account for all phenomena of the neutron-nucleus interaction. In quantum scattering theory, the scattering length is the scattering amplitude $f(k, \theta)$ in the s-wave limit and corresponds to the scattering phase shift $\delta_0 \approx -2\pi a/\lambda$, where a is the scattering length and λ is the neutron de Broglie wavelength. The scattering length has units of length, usually expressed in fm. It is a property of the neutron-nucleus system so it depends on isotope and can have a very different values for isotopes of the same element. The underlying nuclear structure physics that determine its value are complicated, and while that physics is fundamentally understood there are no reliable methods presently available to calculate neutron scattering lengths theoretically; they must be measured by experiment.

Neutron scattering lengths are complex numbers, the imaginary part accounts for neutron absorption, e.g. (n, γ), (n,p), or (n, α) reactions. They are also spin dependent. The neutron is a spin-1/2 particle ([Garcia, 2021](#)) and so has two quantum mechanical spin states: spin-up and spin-down. If the nucleus has non-zero spin, as most do, there are two possible values for the combined spin state of the neutron-nucleus system, with two corresponding neutron scattering lengths a_+ and a_- . For free atoms and gases the scattering length a includes the kinematic effect of nuclear recoil when a neutron scatters from it. In solid materials and liquids the nucleus is constrained from recoiling due to interatomic forces and the value of the scattering length is somewhat larger. In this case the *bound scattering length* b , with spin-dependent values b_+ and b_- , is used. Because they differ only by a simple kinematic recoil effect, the free and bound neutron scattering lengths a and b are not independent. They are related by the simple formula $b = a(A + 1)/A$, where A is the atom/neutron mass ratio. At thermal energy and below, resonances in the neutron-nucleus interaction are rare, notable examples include Sm-149 (2.3 meV), Cd-113 (2.8 meV), and Hf-177 (5.0 meV). Otherwise, for the vast majority of isotopes where such low energy resonances are absent, the neutron scattering length is a constant, independent of neutron energy.

Neutron scattering cross sections can be computed directly from the scattering lengths. Due to wave interference of the neutron matter wave as it scatters coherently from different nuclei in a material, two cross sections result. The *coherent* scattering cross section is given by

$$\sigma_{\text{coh}} = 4\pi \langle b \rangle^2$$

and the *incoherent* scattering cross section by

$$\sigma_{\text{inc}} = 4\pi (\langle b^2 \rangle - \langle b \rangle^2)$$

where $\langle b \rangle^2$ and $\langle b^2 \rangle$ are averages taken over the different isotopes and the spin-dependent values b_+ and b_- . (In gases the bound scattering length b is replaced by the free scattering length a in these and following equations.) The physical interpretation is that coherent scattering describes the effect of scattering centers (nuclei) in the material acting together while incoherent scattering describes the effect caused by differences in the scattering lengths. Note that $\sigma_{\text{inc}} = 0$ in the case of an isotopically pure material where the nuclear spin is zero (hence $b_+ = b_-$), for example pure C-12 or He-4. In hydrogen and hydrogenous materials such as water and polyethylene, the incoherent scattering cross section is quite large due to the strong spin dependence of the neutron-proton scattering length. Slow neutron scattering from an isolated atom is isotropic, as is neutron scattering in a gas, liquid, or amorphous solid. In materials with structural order such as crystals, neutron wave diffraction can lead to highly directional scattering, see the discussion in Neutron Optics below.

The imaginary part of the scattering length gives the probability for a slow neutron to be absorbed by the nucleus, with emission of particles from the compound nucleus such as gammas, protons, or alphas. From quantum mechanics we obtain the following result for the slow neutron absorption cross section

$$\sigma_{\text{abs}} = 2\lambda b'' - 8\pi b''^2$$

where b'' is the imaginary part of the bound neutron scattering length and λ is the neutron de Broglie wavelength. In most isotopes $b'' \ll \lambda$ for slow neutrons, the exceptions being very strong absorbers or isotopes with low energy resonances, so the second term on the right side is usually negligible. The neutron de Broglie wavelength is $\lambda = h/mv$ where mv is the neutron momentum (mass times velocity) so we find that, to excellent approximation in most materials, the neutron absorption cross section is inversely proportional to neutron velocity. This is known as the " $1/v$ " law of neutron absorption, discovered and first explained by Fermi in 1935. The physical interpretation of the $1/v$ law is important. The range of the strong nuclear force very short ($\sim \text{fm}$), and the nucleus is point like in this regime, both much smaller than the neutron de Broglie wavelength, so the probability that the neutron is absorbed is proportional to the time that its matter wave spends in contact with the nucleus, hence proportional to $1/v$. The scattering length itself is independent of neutron velocity. Only the relative velocity of the neutron and the target material is considered here; there is no contribution from the thermal motion of atoms because each neutron overlaps many of them at once and their motion averages to zero. The neutron absorption cross section has no direct dependence on target temperature. However in the case where the target and neutron moderator are in thermal equilibrium, there is an indirect temperature dependence due to the effect of moderator temperature on the neutron velocity spectrum.

The neutron, being electrically neutral, has no long range electrostatic interaction with a charged particle such as an electron, but the neutron is composed of electrically charged quarks so there is a contact interaction when its matter wave overlaps electrons in a material. This leads to the neutron-electron scattering length b_{ne} that contributes slightly to neutron scattering in a material. Its value has been measured in neutron transmission experiments to be -0.00134 fm .

Extensive tables of neutron scattering lengths and cross sections are available (Mughabghab, 2005; NCNR, 2013). Fig. 5 shows the real part of the bound scattering length $\langle b \rangle$, averaged over spin states, for stable isotopes. Tabulated thermal neutron absorption cross sections are generally given at the reference thermal neutron velocity of exactly 2200 ms^{-1} (25.3 meV). From these, the absorption cross section for other velocities or neutron velocity spectra can be computed using the $1/v$ law.

Neutron moderation and thermalization

Fast and intermediate energy neutron interactions in materials are dominated by elastic scattering collisions with atomic nuclei. In the laboratory reference frame the nucleus gains a small amount of kinetic energy due to recoil, therefore the neutron loses that same energy. Each collision results in the transfer of a fraction of the neutron's kinetic energy to the nucleus and the neutron is gradually slowed, or *moderated*, in the material. In a single collision total kinetic energy and momentum are conserved and the initial kinetic energy of the nucleus is negligible so, in close analogy to elementary classical collision theory, the fraction of neutron energy lost depends on only two quantities: the mass of the nucleus, and the neutron's deflection angle θ , i.e. the change in direction of the neutron's velocity. The minimum possible fractional energy loss is zero, when $\theta = 0$, and the maximum is $4A/(A+1)^2$, where A is the atom/neutron mass ratio, when $\theta = \pi$. Elastic scattering is isotropic and θ is effectively a random variable between 0 and π , so we have a uniform probability distribution for energy loss between zero and the maximum. It is convenient to describe the neutron energy E in terms of a logarithmic quantity called the neutron lethargy u , defined by

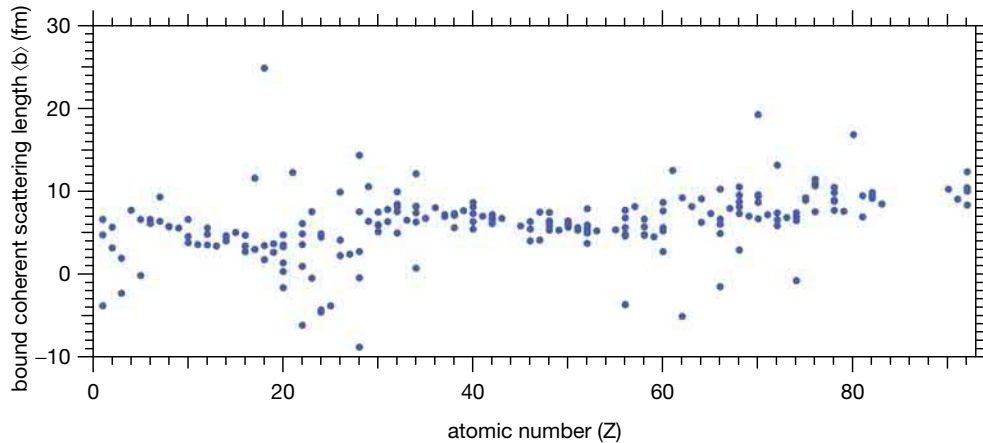


Fig. 5 A summary of measured real bound coherent scattering lengths $\langle b \rangle$ of stable isotopes. From Mughabghab SF (2005) *Atlas of Neutron Resonances*. Amsterdam: Elsevier.

$$u = \ln\left(\frac{10 \text{ MeV}}{E}\right),$$

and the *lethargy constant* ξ , the average logarithm of the ratio of initial to final neutron energy for a single collision which depends only on A

$$\xi = \left\langle \ln \frac{E_{\text{initial}}}{E_{\text{final}}} \right\rangle = 1 + \frac{(A-1)^2}{2A} \ln\left(\frac{A-1}{A+1}\right)$$

Fig. 6 shows a plot of the lethargy constant and consequently the average number of elastic collisions needed to moderate a neutron from initial energy 1 MeV to the bottom of the epithermal range 0.5 eV, as a function of A . In the thermal energy regime two effects emerge that greatly complicate the physics of neutron moderation: (1) the thermal kinetic energy of the nucleus may be comparable to the neutron energy so it cannot be ignored, in fact the neutron may gain energy in a collision; and (2) chemical and molecular binding energies in the material are not negligible so material dynamics may affect the collisions. In the thermal regime computer simulations are needed for quantitative studies of neutron moderation.

The distance that a neutron travels while being moderated depends on the total cross section for elastic scattering $\sigma_{\text{elas}} = \sigma_{\text{coh}} + \sigma_{\text{inc}}$ and the density of nuclei N . As may be expected from dimensional analysis, the mean free path λ between elastic collisions is given by $\lambda = 1/N \sigma_{\text{elas}}$. The path of the neutron can be regarded as a random walk between collisions, so mathematically it follows that the mean square distance $\langle R^2 \rangle$ equals the square of the step size times the number of steps. Therefore the average of the distance squared traveled by a neutron as it is moderated from initial energy E_0 to energy E is

$$\langle R^2 \rangle = n\lambda^2 = \frac{\ln(E_0/E)}{\xi N^2 \sigma_{\text{elas}}^2}$$

In neutron diffusion theory this is often quantified using the *Fermi age* parameter τ . In spite of its name, the Fermi age has units of length squared and is proportional to the mean square distance traveled in a moderator as the neutron lethargy goes from u_0 to u :

$$\tau = \frac{\lambda^2}{3\xi(1-2/3A)}(u-u_0) = \frac{1}{6}\langle R^2 \rangle$$

A population of free neutrons contained in a moderator will continue to lose energy until they reach thermal equilibrium with it, in other words until the average neutron kinetic energy closely approaches that of the atoms in the moderator so that in subsequent collisions the neutrons are as likely to gain energy as lose it. At that point the moderation process is effectively complete and the neutrons are said to be *thermalized*. Following statistical physics, the kinetic energy spectrum of thermalized neutrons is Maxwellian:

$$N(E) = N_0 \sqrt{E} e^{-\left(\frac{E}{kT}\right)}$$

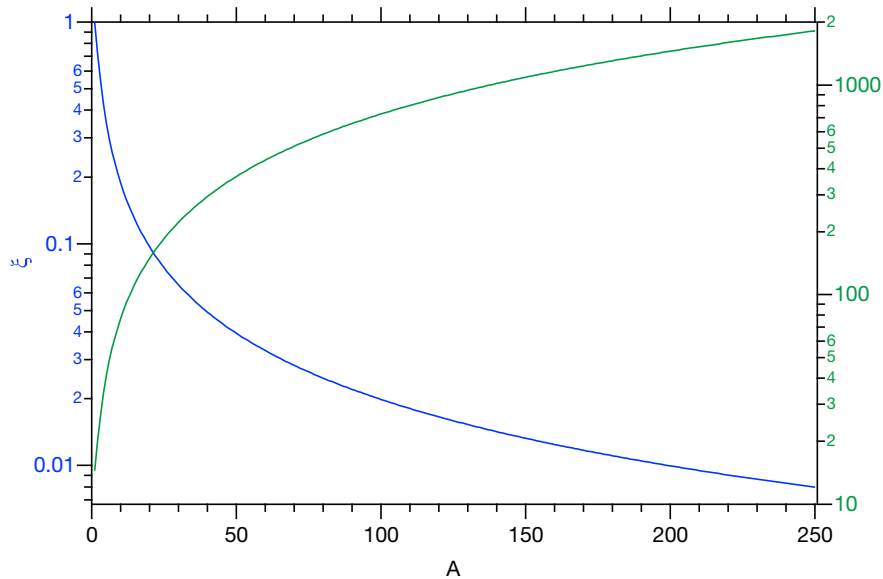


Fig. 6 Lethargy constant ξ (left axis), and the average number of collisions n needed to reduce neutron kinetic energy from 1 MeV to the bottom of the epithermal range 0.5 eV (right axis), as a function of A , the ratio of atomic mass to neutron mass. Low mass materials make much more efficient neutron moderators.

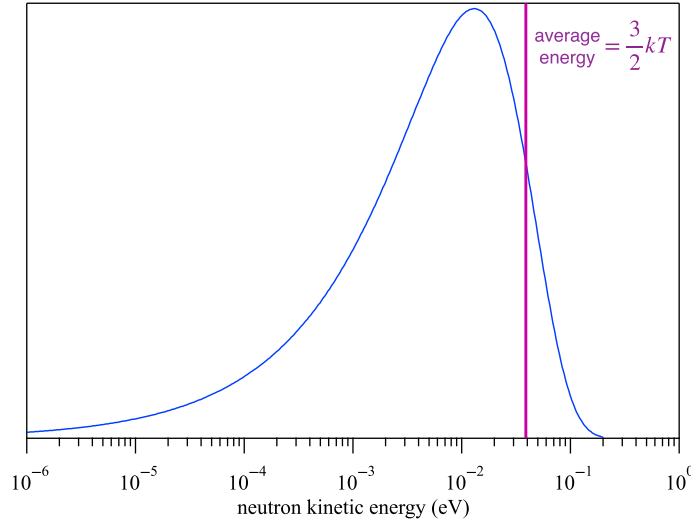


Fig. 7 The Maxwellian kinetic energy spectrum of a population of neutrons thermalized in a moderator at 28 °C ($kT = 0.026$).

where $k = 1.38 \times 10^{-23} \text{ J K}^{-1}$ is the Boltzmann constant and T is the temperature of the moderator in Kelvins, see Fig. 7. The most probable energy in a population of thermalized neutrons is $\frac{1}{2}kT$ and the average energy is $\frac{3}{2}kT$. Neutron flux $\phi(E)$ is defined as the number of neutrons passing through a plane surface, per unit area per unit time, so flux equals neutron density times velocity: $\phi(E) = v\rho(E) \propto \sqrt{E}\rho(E)$. Therefore in terms of flux the Maxwellian spectrum is:

$$\phi(E) = \phi_0 E e^{-\left(\frac{E}{kT}\right)}.$$

Due to the $1/v$ law, the probability of neutron absorption in a material increases as neutrons are moderated. This is important in the design of fission reactors: fast fission neutrons are moderated by a light material such as water or graphite to enable efficient absorption by the fissile isotope in the fuel; and also for fast neutron shields such as borated water: hydrogen in the water moderates neutrons to be absorbed via the (n, α) reaction in ^{10}B . Neutron absorption in the moderator itself may be undesirable. For example research reactors are generally designed to maximize neutron flux in the core while efficient power generation is not important. Therefore research reactors often use a heavy water (D_2O) moderator to take advantage of the much smaller neutron absorption cross section of deuterium compared to hydrogen, even though it is a less efficient moderator than light water (H_2O) due to its higher mass. Cold neutrons are produced at neutron research facilities using cryogenic moderators such as liquid hydrogen or liquid deuterium.

Thermal neutron absorption cross sections and resonance integrals

Thermal neutron absorption cross sections σ_0 are tabulated at the reference thermal velocity of $v_0 = 2200 \text{ m/s}$ ($E_0 = 25.3 \text{ meV}$). To measure these precisely, a neutron source that is very well thermalized in a moderator at known temperature is used, so the neutron kinetic energy spectrum is very close to Maxwellian, and a simultaneous comparison to a well known standard cross section σ_{0s} such as $^{197}\text{Au}(n, \gamma)^{198}\text{Au}$ is made. After irradiation, the ratio of the measured sample activity A to the standard activity A_s is

$$R = \frac{A}{A_s} = \frac{\int \sigma(E)\phi(E)dE}{\int \sigma_s(E)\phi(E)dE}.$$

In the case of a purely $1/v$ absorber, $\sigma(E) = \sigma_0 \sqrt{E_0/E}$ and $\sigma_s(E) = \sigma_{0s} \sqrt{E_0/E}$ giving simply $\sigma_0 = R\sigma_{0s}$. It is usually important to correct for small departures from the $1/v$ law and departures from a pure Maxwellian neutron spectrum, in which case

$$\sigma_0 = R\sigma_s \left(\frac{g}{g_s} \right)$$

where g and g_s are the generalized g -factors calculated from the known resonance spectra and neutron transport simulations of the irradiation facility. Also of interest is the Maxwellian averaged thermal cross section σ_{macs} :

$$\sigma_{macs} = \frac{2}{\sqrt{\pi}} \frac{\int \sigma(E)\phi(E)dE}{\int \phi(E)dE} = \frac{2}{\sqrt{\pi}} \frac{\int \sigma(E)E e^{-\left(\frac{E}{kT}\right)} dE}{\int E e^{-\left(\frac{E}{kT}\right)} dE}.$$

For purely $1/v$ absorber we have $\sigma_{macs} = \sigma_0 \sqrt{E_0/kT}$.

In a nuclear reactor, the high-energy tail of the neutron energy spectrum extends well into the epithermal region where neutron capture cross sections often have complicated energy dependence due to the presence of low energy resonances. It is useful to express the epithermal neutron capture probability of a material by a *resonance integral* RI , defined as

$$RI = \int_{E_0}^{\infty} \frac{\sigma(E)}{E} dE$$

Epithermal neutrons in a reactor usually have a $1/E$ spectrum so this applied as a weight factor inside the integral. By convention a low energy cutoff $E_0 = 0.5$ eV is used, applied in practice by using a cadmium filter. Important corrections include a high-energy weighting function to account for the typically rapid decrease in flux above 1 MeV, and the self-shielding factor that accounts for dips in the neutron flux spectrum caused by strong resonances in the sample. Resonance integrals are measured experimentally and are tabulated for many isotopes and materials.

Free neutron decay

The free neutron is not a stable particle, it decays into a proton, electron, and antineutrino with a mean decay lifetime of 880 s. This weak nuclear decay is possible because the neutron mass exceeds the total mass of the final state particles, so energy is released via the Einstein relation $E = \Delta mc^2$. Neutron decay is the most basic example of nuclear beta decay. It has important consequences in astrophysics, cosmology, and particle physics. The loss of neutrons due to free decay in a moderator or reactor is insignificant because the decay lifetime is much longer than the typical neutron capture lifetime in a material.

Neutron optics and scattering science

When a neutron's de Broglie wavelength exceeds the interatomic spacing in a material, as is the case for thermal energy and below, it may interact coherently with many nuclei at once. Each point-like nucleus acts as a scattering site for the neutron wave and produces a secondary spherical wave according to Huygen's principle. These scattered waves, all associated with a single neutron according to quantum mechanics, interfere with the incident wave and each other to produce optical phenomena such as diffraction and refraction that are closely analogous to those of light waves in a transparent medium.

In a crystal, the neutron wave scatters simultaneously from many lattice points to produce constructive interference (diffraction) when Bragg's Law, $n\lambda = 2d \sin \theta$, is satisfied for some set of lattice planes in the crystal. Here d is the spacing of lattice planes, θ is the neutron incident angle, λ is the neutron wavelength, and n is the diffraction order. Bragg's law is a geometric consequence of wave scattering in a crystal so it is essentially no different for neutrons and x rays. Neutron diffraction has a number of advantages over x rays in solid state physics and materials science research: (1) neutrons are much more penetrating and can illuminate a much larger sample volume; (2) in the relevant wavelength range, neutron kinetic energy is comparable to chemical and molecular binding energies in a material, so material dynamics can be effectively investigated; and (3) the neutron's magnetic dipole moment contributes to the spin-dependence of the scattering length, so magnetic phenomena such as ferromagnetism and superconductivity can be studied. The relative disadvantage of neutron diffraction is that the beam intensities at the best neutron sources are far lower than at a synchrotron x ray source so measurements are much slower. Fig. 8 shows an example of a crystal neutron diffraction pattern.

When a neutron enters a material it feels an effective potential energy, a volumetric average of the strong force potentials of the individual nuclei, that is proportional to the average scattering length $\langle b \rangle$ and density of nuclei N :

$$V_{\text{eff}} = \frac{h^2}{2\pi m} N \langle b \rangle.$$

In most materials V_{eff} is positive and on the order of 100 neV. When a neutron enters the material, it slows and its wavelength increases, causing refraction. This can be expressed as an index of refraction n , similar to that of light:

$$n = 1 - \frac{N \langle b \rangle \lambda^2}{2\pi}.$$

The neutron index of refraction in vacuum is 1, and it is slightly less than 1 in a medium, rather than larger as with light. This is because the neutron wave is nonrelativistic, its velocity is inversely proportional to wavelength, not proportional as for a light wave. The dependence of n on λ^2 shows that neutron refraction is dispersive. Refraction of neutron waves has been exploited in devices that manipulate slow neutron beams, such as neutron guides (analogous to fiber optics), neutron lenses, and neutron prisms.

In the last few decades neutron imaging has developed into a very useful tool for a broad range of scientific researchers. In a traditional x ray radiograph, contrast is obtained from differences in electron density within an object. High density materials such as calcium in bones absorb x rays much more strongly than low density materials such as skin and muscle. In a neutron radiograph, neutrons are removed from a beam by scattering from materials with high scattering length density. Neutrons scatter from nuclei, not electrons, and neutron scattering lengths have no significant dependence on atomic number, as seen in Fig. 5. In particular

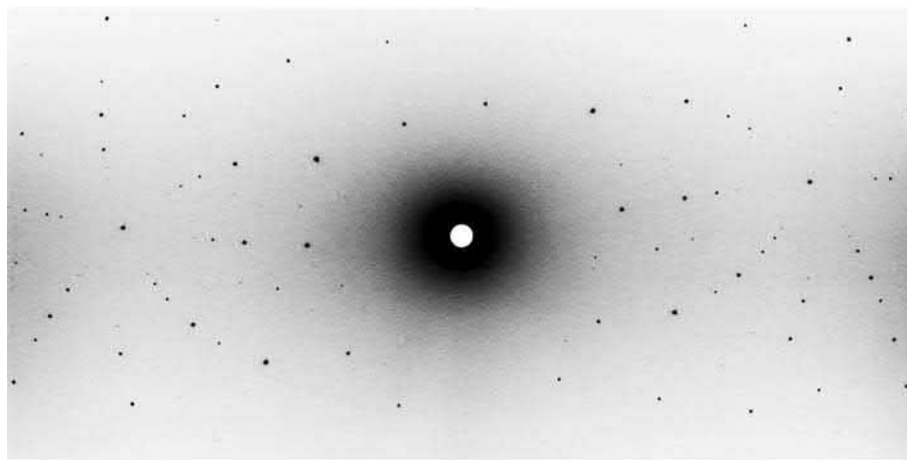


Fig. 8 A Laue photograph of neutron Bragg diffraction from a $\text{Cu}_2\text{ZnSnS}_4$ crystal (Iles and Schoor, 2016). Dark spots are caused by different Bragg reflections.

hydrogen has a large incoherent scattering length $b_{\text{inc}} = \sqrt{\langle b^2 \rangle - \langle b \rangle^2}$ while the scattering length densities in metals such as nickel and lead are relatively low, so neutron radiographs excel at something x ray radiographs cannot do: imaging hydrogenous materials imbedded within metal or stone. A beautiful example can be seen in Fig. 9. Because neutrons are much more penetrating than x rays, neutron radiography has been used to nondestructively image small features and defects within large dense objects such turbine blades, geological samples, and archeological objects. Neutron tomography, where multiple images are taken as a sample is rotated, can be used to construct virtual 3D neutron images of the interior of an object, similar to an x ray CAT scan.

Important centers for neutron scattering and optics research include the Spallation Neutron Source (Oak Ridge, Tennessee, USA), the Institute Laue-Langevin (Grenoble, France), the NIST Center for Neutron Research (Gaithersburg, Maryland, USA), and the European Spallation Source (under construction in Lund, Sweden).

Neutron cross sections and related data online

A number of online databases provide tabulated lists and plotting tools for neutron reaction data such as cross sections, resonance integrals, and scattering lengths. They are searchable by target element, isotope, reaction, and energy. These include the international Evaluated Nuclear Data File (ENDF), the Japanese Evaluated Nuclear Data Library (JENDL), and NuDat2.

<https://www.nndc.bnl.gov>

<http://www.nds.ciae.ac.cn>

<https://www.oecd-nea.org/dbdata/>

<https://www.nndc.jaea.go.jp>

<https://www.ncnr.nist.gov/resources/n-lengths/>

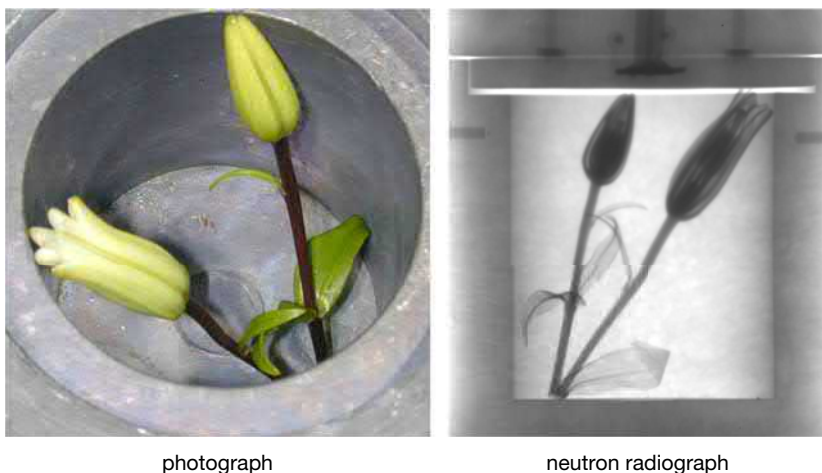


Fig. 9 Left: photo of the set up, two flowers inside a lead cask; Right: a neutron radiograph from the side, through the walls of the cask. Courtesy of Daniel Hussey, NIST.

See Also: Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection.

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Phenomenology of Gamma Ray and Charged Particles Interactions

Reina Maruyama, Department of Physics, Yale University, New Haven, CT, United States

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Introduction

Photons and charged particles are produced in naturally-occurring decays of radioactive isotopes and in nuclear reactions. They are produced in the cosmos and on Earth, via both naturally-occurring and man-made processes. In the cosmos, they fuel stars as protons fuse to form helium and go on to produce small amounts of heavier elements such as lithium, beryllium, boron, carbon, nitrogen, and oxygen in smaller stars, and in heavier stars, all the way up to iron. Elements heavier than iron are produced through two distinct neutron capture mechanisms known as the s-(slow) and r-(rapid) processes. Here on Earth, naturally occurring radioisotopes such as ^{238}U and ^{232}Th produce photons in the form of gamma rays and X-rays, neutrons, and charged particles such as alpha particles as they decay. We also produce photons and charged particles in particle accelerators and in nuclear reactors.

Knowing how photons and charged particles interact with matter is critical as we maintain and improve the safety and efficiency of current technology for nuclear energy generation, radiation detection, medical imaging, and radiation therapy.

Photons and charged particles will keep moving unaltered with the same energy and momentum in vacuum. When they encounter matter, photons and charged particles interact with the electrons and nuclei of the atoms of the matter and may be deflected, lose energy, or stopped. The amount of deflection and energy loss depend on the type of the incoming particle and what materials they encounter. Charged particles emitted at energies of MeV's from radioactive decay or nuclear reactions would typically travel less than 100 μm . Electrons such as those emitted in β decays may travel 0.1 to several mm in solids, while for MeV γ rays the range can be 5 cm or longer. The amount of energy lost per each encounter as well as the density of the material will determine how far the incoming photons and charged particles will travel.

Electrons and heavier charged particles such as protons and alpha particles interact through the coulomb force with the electrons in the materials they pass through. Uncharged radiations such as neutrons, X-rays and gamma-rays are not subject to coulomb force. They can either pass through matter without any interactions if thin enough or undergo a more catastrophic interaction that radically alters the properties of the incident radiation in a single encounter. These interactions result in the full or partial transfer of energy of the incident neutrons or photons to the electrons or nuclei of the atoms of the materials through which they pass.

Photons interact primarily with the orbital electrons and depending on their energy, can undergo pair production, Compton scattering, or photoelectric effect. In contrast, charged particles such as electrons, muons, alpha particles, and other ionized atoms interact with and impart energy to the orbital electrons and the nucleus. Through these individual interactions, charged particles may change their directions, excite or liberate electrons from their atomic orbits, excite nuclei, or induce nuclear interactions. High energy photons can also induce nuclear reactions, such as photofission.

In this article we discuss how photons interact with matter and how charged particles such as electrons, protons, and other ions interact with matter. We limit our discussion to radiations that we are likely to encounter in nuclear decay and nuclear energy production: photons in the X-ray and γ -ray regions, electrons and positrons up to MeV energies, and heavy charged particles (protons, α 's, and ions). Neutron interactions are considered separately in Chapter 1.9, *Phenomenology of Neutron Interactions*.

Interaction of photons with matter

Of the many interaction mechanisms possible, there are three primary ways X-ray and γ -ray photons interact with matter at relevant nuclear energies: photoelectric effect, Compton scattering, and pair production.

In the photoelectric effect, the photon is absorbed by an atomic electron and the electron is ejected from the atom. In Compton scattering, a photon scatters from an atomic electron. In pair production, a photon is converted to an electron-positron pair. When a photon is traveling through a medium, one can describe the interactions of photons with matter as a sum of all three processes. The energy of the photon determines which interaction dominates.

Photoelectric effect

The photoelectric effect occurs when all of the energy of the incident photon is transferred to an atomic electron and the electron is ejected from the atom. The photoelectric effect is important for photons below about 200–300 keV interacting with all but the lightest elements.

In the photoelectric effect, a photon interacts with an atom in the material and is completely absorbed. As a result, one of the orbital electrons is excited and ejected from the atom. The ejected electrons are called photoelectrons. The photoelectric effect occurs with the atom as a whole and cannot take place with free electrons. The kinetic energy of the ejected photoelectron (T_{e^-}) is the difference between the energy of the incident photon (E_γ) and the binding energy of the electron (E_b):

$$T_{e^-} = E_\gamma - E_b \quad (1)$$

After the photoelectrons are ejected, the atom is left behind with a vacancy in one of its bound shells. The vacancy is quickly filled by a free electron being captured from the surrounding medium and/or by electrons being rearranging from other shells of the atom. This process of vacancy filling generates one or more X-rays with characteristic energies. In some fraction of the cases, the excess energy is transferred to another electron bound to the atom. The excited electron is in turn ejected from the atom, resulting in an Auger electron.

The photoelectric process is the primary mode of interactions for photons at low energies (~ 100 keV). The interaction probability, τ , is difficult to calculate but rather shown empirically to increase rapidly with decreasing photon energy and increasing atomic number, Z , of the absorber material:

$$\tau \propto \frac{Z^n}{E_\gamma^{3.5}} \quad (2)$$

where n varies between 4 and 5 depending on E_γ . The strong Z dependence of photoelectric effect is the reason for the use of high- Z materials such as lead in gamma-ray shields.

Compton scattering

At energies between keV's to tens of MeV's, photons interacting with electrons will undergo a Compton scattering, an elastic process where a fraction of the energy of the incident photon is transferred to the electron, resulting in the scattered photon having lower energy than it had before the interaction. As in all such collisions, the kinematics of the scattering process conserve energy and linear momentum. In the limit that the photon energy is large compared with the orbital energies of the loosely bound outer atomic electrons (usually a good approximation), the electron can be regarded as free and at rest.

The basic principle of Compton scattering is shown in Fig. 1. The incoming photon with energy $E_\gamma = h\nu$ scatters off of an electron and is deflected with energy $E_\gamma' = h\nu'$ by an angle θ with respect to its original direction ($h = 6.626 \times 10^{-34}$ J $\cdot$ s is the Planck

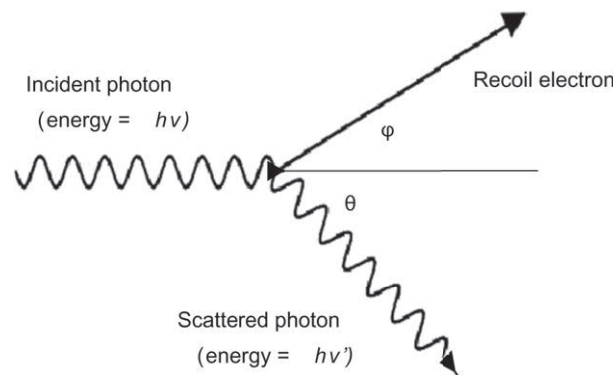


Fig. 1 Schematic of Compton scattering. By Reina Maruyama.

constant and ν is the frequency of the photon). The electron now has some momentum and emerges as a recoil electron. Using conservation of linear momentum and total energy, we arrive at the Compton scattering formula

$$E'_\gamma = \frac{E_\gamma}{1 + (E_\gamma/m_e c^2)(1 - \cos \theta)} \quad (3)$$

where $m_e c^2 = 0.511$ MeV is the rest mass energy of the electron. For small scattering angles θ , very little energy is transferred. At the other extreme of $\theta = 180^\circ$, the energy of the scattered photons is at a minimum and is roughly $m_e c^2/2 \approx 0.25$ MeV for incident photons with energies much larger than 0.511 MeV.

The probability of Compton scattering at an angle θ is described by the Klein-Nishina formula for the differential cross section per electron:

$$\frac{d\sigma_c}{d\Omega} = r_0^2 \left(\frac{1}{1 + \alpha(1 - \cos \theta)} \right)^2 \left(\frac{1 + \cos^2 \theta}{2} \right) \left(1 + \frac{\alpha^2(1 - \cos \theta)^2}{(1 + \cos^2 \theta)[1 + \alpha(1 - \cos \theta)]} \right) \quad (4)$$

where $d\Omega$ is the solid angle around θ , $\alpha = E_\gamma/m_e c^2$ and r_0 is the so-called classical electron radius defined as $r_0 = e^2/4\pi\epsilon_0 m_e c^2 = 2.818 \times 10^{-13}$ cm. The parameter r_0 is merely a convention and has nothing to do with the size of the electrons. The Klein-Nishina cross section is peaked forward strongly for scattering at higher gamma energies. The experimental data agree perfectly with Eq. (4) and is among the most convincing successes of quantum theory.

To calculate the total cross section of photons, we integrate Eq. (4) over all $d\Omega$ to obtain

$$\sigma_c = \frac{\pi r_0^2}{\alpha} \left(\left[1 - \frac{2[\alpha + 1]}{\alpha^2} \right] \ln(2\alpha + 1) + \frac{1}{2} + \frac{4}{\alpha} - \frac{1}{2(2\alpha + 1)^2} \right) \quad (5)$$

As the energy of the photon increases, the total cross section decreases and the probability of scattering becomes more forward peaked.

Pair production

If the energy of a photon is greater than twice the rest mass of an electron, i.e., $2m_e c^2 = 1.02$ MeV, the photon can interact with the electromagnetic field of a nucleus and undergo pair production to create an electron-positron pair. The photon disappears completely, and its energy is transformed into the rest masses and kinetic energies of the electron-positron pair. The total kinetic energy of the electron-positron pair is

$$T_{e-} + T_{e+} = E_\gamma - 2m_e c^2 \quad (6)$$

where the E_γ is the energy of the incoming photon and T_{e-} and T_{e+} are the kinetic energies of the electron and positron, respectively. Once formed, the electron and positron will propagate in the medium and lose energy by Coulomb interactions with the atomic electrons bound in the medium. The positron will slow down until it reaches thermal energies and eventually annihilate with an electron. Two photons that travel in opposite directions are produced as secondary products of this interaction, each with an energy of $m_e c^2$. The probability of pair production increases sharply as gamma-ray energy approaches several MeV but is especially important for energies above 4–6 MeV and increases approximately as the square of the absorber atomic number. Pair production can also occur in the electromagnetic field of electrons, however the minimum energy required for such a process is four times the rest mass of an electron.

Total cross sections and attenuation coefficients for photons

The interaction of photons with matter will proceed by all possible mechanisms including photoelectric effect, Compton scattering, and pair production as described above, as well as less prominent mechanisms such as Rayleigh scattering and photonuclear interactions. To quantify the interactions probability of photons with matter, we use total cross section (σ_{total}), linear attenuation coefficient (μ), and mass attenuation coefficient (μ/ρ). Units that are usually used to quantify these parameters are barns [$1\text{b} = 10^{-24} \text{ cm}^2$] for cross sections (σ), [cm^{-1}] for linear attenuation coefficient (μ), and [$\text{cm}^2 \text{ g}^{-1}$] for mass attenuation coefficient (μ/ρ).

The total cross section per atom, σ_{total} , is defined as the sum of the cross sections of the individual possible interaction mechanisms. Fig. 2 shows the total cross section and each of its components for photons interacting with lead and carbon as a function of energy. The figure shows how the cross section is dominated by the photoelectric effect at low energies (e.g., below about 10 keV for carbon and 100 keV for lead), Compton scattering at around 1 MeV, and pair production above 10 MeV. More detail can be found in Tanabashi et al. (2018). Data for photon cross sections can be found in Berger et al. (2010).

The total linear attenuation coefficient, μ , is defined as the total probability per unit length that an incoming photon is removed from the incoming beam. The total linear attenuation coefficient μ is the sum of the respective linear absorption coefficients for photoelectric absorption ($\mu_{\text{photoelectric}}$), Compton scattering (μ_{Compton}), and pair production ($\mu_{\text{pair production}}$):

$$\mu = \mu_{\text{photoelectric}} + \mu_{\text{Compton}} + \mu_{\text{pair production}} \quad (7)$$

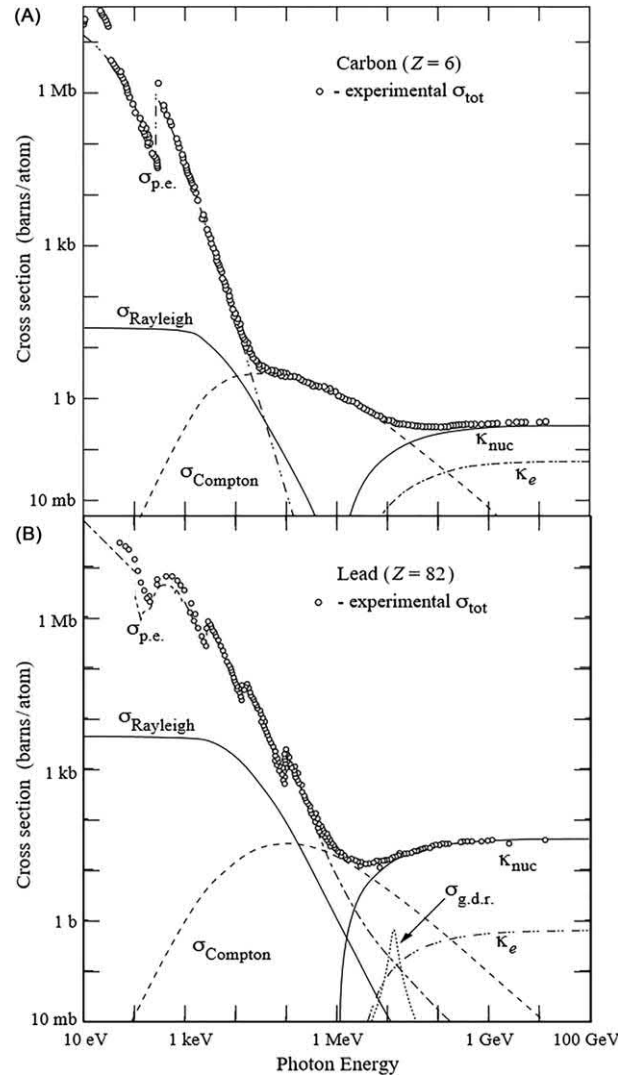


Fig. 2 Photon total cross sections as a function of energy in carbon and lead showing the contributions from photoelectric effect ($\sigma_{p.e.}$), Compton scattering (σ_{Compton}), pair production in nuclear field (κ_{nuc}), and pair production in electron field (κ_e), as well as less prominent contributions such as Rayleigh scattering (σ_{Rayleigh}) and photonuclear interactions, most notably the Giant Dipole Resonance ($\sigma_{g.d.r.}$). Figure from Tanabashi M, et al. (Particle Data Group) (2018). *Physical Review D* 98: 030001 with data from Berger MJ, et al. (2010) *XCOM: Photon Cross Sections Database, NIST Standard Reference Database 8 (XGAM)*, <https://doi.org/10.18434/T48G6X>.

The linear absorption coefficient takes into account the number of nucleons and electrons available per unit length while cross section is the probability for interaction per atom. For example, Compton scattering, μ_{Compton} is proportional to σ_c described in Eq. (5) by

$$\mu_{\text{Compton}} = \sigma_c N Z \quad (8)$$

where Z and N are the atomic number and number of atoms per unit volume of the material respectively.

The photon beam intensity I in ($\text{cm}^{-2} \text{s}^{-1}$) of transmitted photons through a medium of thickness x , given initial beam intensity I_0 is

$$I = I_0 e^{-\mu x} \quad (9)$$

The average distance traveled in a medium before an interaction takes place is characterized by the mean free path, $\lambda = 1/\mu$. Typical values of λ range from a few mm to tens of cm in solids for typical gamma-ray energies.

Although linear attenuation coefficients are convenient for engineering applications, they are proportional to the absorber density, ρ , which depends to some extent on the physical state of the material. Therefore, attenuation coefficients are often given as the mass attenuation coefficient (μ/ρ) which is the ratio of the linear attenuation coefficient to the density of a material, ρ :

$$\text{mass attenuation coefficient} = \frac{\mu}{\rho} \quad (10)$$

Mass attenuation coefficients do not depend on the physical state of a given absorber and has the units of $[\text{cm}^2 \text{g}^{-1}]$. The mass attenuation coefficient is proportional to the total photon interaction cross section per atom as:

$$\frac{\mu}{\rho} \left[\frac{\text{cm}^2}{\text{g}} \right] = \sigma_{\text{tot}} \left[\frac{\text{cm}^2}{\text{atom}} \right] \frac{N_A \left[\frac{\text{atoms}}{\text{g} \cdot \text{atom}} \right]}{M \frac{\text{g}}{\text{atom}}} = \sigma_{\text{tot}} \left[\frac{\text{b}}{\text{atom}} \right] \cdot N_A / M \cdot 10^{-24} \quad (11)$$

The mass attenuation coefficient of a mixture of elements can then be calculated from

$$\left(\frac{\mu}{\rho} \right)_c = \sum_i w_i \left(\frac{\mu}{\rho} \right)_i \quad (12)$$

where w_i factors are the weight fraction of element i in the compound or mixture.

Interaction of charged particle with matter

In this section, we will discuss the interactions of charged particles such as electrons, positrons, protons, and ions with matter. Interaction of charged particles with matter occur mainly through the Coulomb interactions with the orbital electrons in the medium. Charged particles also interact with atomic nuclei, however the Coulomb field of the nucleus is well shielded by the orbital electrons therefore such interactions are small and are negligible for most practical applications. The exception would be when those trajectories would bring the incident particle close to the nucleus.

We first discuss the interaction of heavy charged particles including protons, alpha particles, and other ions that have masses much greater than that of an electron. Their trajectories are mostly linear and simple. We will then discuss the interaction of light charged particles, in other words electrons and positrons, with matter.

Interactions of heavy charged particles in matter

As heavy charged particles such as protons, alpha particles, and ions travel through matter, they interact—via the Coulomb force—primarily with the negatively charged orbital electrons bound to the atoms in the medium. In these interactions, the electrons are excited into higher orbital states (a process called “excitation”) or may even be ejected from the atom (a process called “ionization”).

The Coulomb force has a $1/r^2$ dependence and can reach far. Therefore, a charged particle traveling through matter interacts with many electrons simultaneously. It loses energy gradually and continuously as it travels through the medium, until it has lost all of its energy. The total distance that it travels is called the range of the particle. The range is determined by the type and energy of the particle and the type of material it encounters. The mean range is defined as the range at which one-half of the particles have longer ranges and the other half shorter.

Because the mass of a heavy charged particles is much larger than the mass of an electron, each collision results in loss of only a small fraction of the particle’s kinetic energy. The change of momentum of the heavy charged particle is thus negligible. A large number of collisions is needed for a heavy charged particle to come to rest in a medium.

Stopping power

The linear stopping power, sometimes called specific energy loss, for charged particles in a given material is defined as the differential energy loss of the particle in the material ($-dE$) divided by the differential path length (dx). The energy loss of charged particles in matter was first calculated by Bohr using classical mechanics in 1913 and later by Bethe (1930), then by Bloch (1933) using quantum mechanics (Bohr, 1913; Bethe, 1930; Bloch, 1933). The small amount of energy lost ($-dE$) by a particle in the medium per unit length (dx) is, to a good approximation, by what we now call the Bethe equation:

$$-\frac{dE}{dx} = \frac{4\pi e^4 z^2}{m_0 v^2} N Z \left[\ln \frac{2m_0 v^2}{I} - \ln \left(1 - \frac{v^2}{c^2} \right) - \frac{v^2}{c^2} \right] \quad (13)$$

where v and ze are the velocity and charge of the incoming charged particle respectively, N and Z are the number density and atomic number of the atoms in the absorber medium, m_0 the electron mass, e is the electric charge, and I is the mean excitation and ionization potential of the atoms of the absorber the particle is going through. The quantity, $-dE/dx$ is called the stopping power.

In applications related to nuclear energy production, the charged particles are typically nonrelativistic ($v \ll c$). Therefore, v^2/c^2 is small and the expression inside the square brackets in Eq. (13) varies slowly with the particle’s energy. The stopping power varies as $1/v^2$ to a good approximation: the slower the incoming particle, the more energy it loses per unit length of travel inside the medium. dE/dx is proportional to z^2 , therefore incoming particles with greater charge z will have greater stopping power (for example, alpha particles will lose more energy per unit length than protons do). Absorber materials with larger NZ will exert larger stopping power. NZ represents the electron density of the absorber, and materials with high atomic number and high densities have larger stopping power.

In practice, it is useful to describe the energy loss in media in terms of ρx , the product of density of the medium ρ and distance traveled x . ρx is typically given in $[\text{g cm}^{-2}]$. It can be measured experimentally by taking the ratio of the mass and the area of the medium. The stopping power typically found in the literature is:

$$\frac{dE}{d(\rho x)} = \frac{1}{\rho} \frac{dE}{dx} \quad (14)$$

As noted in Eq. (13), the stopping power for particles much heavier than electrons depends on its charge and speed and does not depend on the mass of the particle.

By integrating Eq. (13) we can calculate the range R , the total distance that a particle of given initial kinetic energy E_0 travels in a given medium. To put it another way, to a good approximation, R is the average path length traveled by a charged particle as it slows down to rest. The range is given by

$$R = \int_{E_0}^0 \left(-\frac{dE}{dx} \right)^{-1} dE \quad (15)$$

where E is the kinetic energy of the particle along its path in the medium. Because the particles lose energy through scattering by atomic electrons, the range depends inversely on the density, and the product ρR in units of g cm^{-2} is often a more useful quantity than R (note: both R and ρR are referred to as “range” in the literature—check the units to determine which one is being used). Fig. 3 shows the range-energy relationship for protons and α particles in air, water, and graphite in terms of ρR . For database on range and energy for other particles and materials, refer to Berger et al. (2017).

Energy loss characteristics: The Bragg curve and energy straggling

As a heavy charged particle travels through matter, it loses its energy to the surrounding medium. The stopping power ($-dE/dx$) increases roughly as $1/E$ for most of its track as shown in Eq. (13). Fig. 4 shows the stopping power along the track of an alpha particle with an initial energy of 5.49 MeV traveling through air. As the energy of the particle decreases, the $1/E$ dependence causes a pronounced peak in the stopping power where the energy loss of the particle becomes significant. For the case of an alpha particle in air as shown in Fig. 4, this occurs at around 3.5 cm. The peak is called a Bragg peak after William Henry Bragg who discovered it in 1903. Beyond the Bragg peak and near the end of its track, the charged particle picks up electrons from the surrounding material and the $1/E$ dependence is no longer valid. As a result, the stopping power, i.e., $-dE/dx$ falls off rapidly.

The energy loss of a specific charged particle as it goes through a medium is a stochastic process and the details of the microscopic interactions of a specific charged particle vary somewhat randomly. After a beam of monoenergetic charged particles has passed

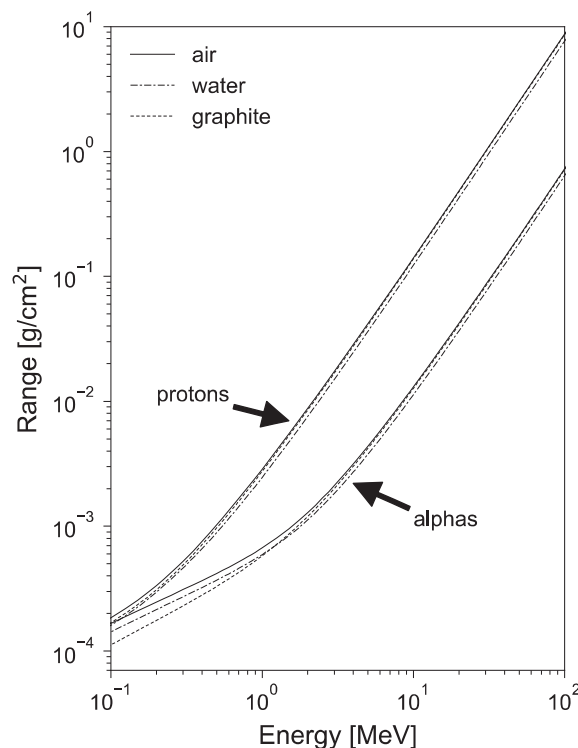


Fig. 3 Range-energy curves for protons and alpha particles in air (dry, near sea level), water (liquid), and graphite (density = 1.7 g/cm^3). Data from Berger MJ, et al. (2017) *Stopping-Power & Range Tables for Electrons, Protons, and Helium Ions*. <https://doi.org/10.18434/T4NC7P>.

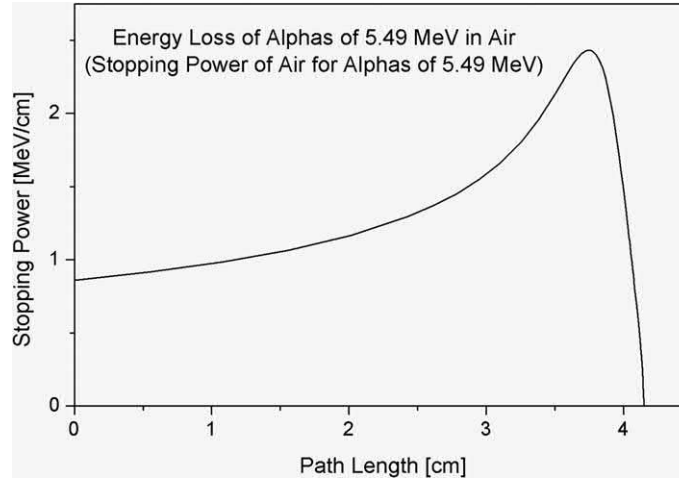


Fig. 4 Bragg curve: stopping power, i.e., specific energy loss, vs. path length for a 5.49 MeV alpha track in air. Figure by Helmut Paul-public domain, <https://commons.wikimedia.org/w/index.php?curid=770220>.

through a given thickness of the absorber material, there will be a spread in the particles' energies. The spread, or the width of the energy distribution, is a measure of "energy straggling." Energy straggling varies with the distance along the particle track. The distribution narrows again near the end of the track after the mean particle energy has greatly been reduced.

These phenomena are exploited in particle therapy of cancer. The presence of Bragg peak and energy straggling can concentrate the effect of light ion beams on the tumor being treated while minimizing the effect on the surrounding healthy tissue. A monoenergetic proton beam with the sharp peak is widened by increasing the spread of energies, so that a larger tumor volume can be treated. This can be achieved by using variable thickness attenuators like spinning wedges.

Interactions of electrons with matter

As in the case of heavy charged particles, electrons and positrons also interact with atomic electrons via the Coulomb force. However, there are several significant differences owing to the electron's smaller mass. As described in the previous section, heavy charged particles travel in near straight lines since they are much heavier than the electrons they interact with. However, we now have electrons interacting with electrons, in other words, their masses are the same so the incoming electrons will: (1) travel at relativistic speeds, particularly those that were emitted in β decays; (2) experience large deflections; (3) transfer a large fraction of the initial energy to the other electron; and (4) emit electromagnetic radiation (called bremsstrahlung radiation) as they experience large deceleration.

There are two primary energy loss mechanisms for electrons in matter. At energies below a critical energy, $E_c \approx 600 \text{ MeV}/Z$, energy loss via excitation and ionization of the bound electrons dominate (here, Z is the charge number of the atoms in the medium). Above E_c , energy loss is primarily via radiation. We discuss the interactions in the two energy regions below.

Ionization region ($E < E_c$)

The energy loss mechanism in the ionization region ($E < E_c$) is similar to what was described for heavy particles in section "Interactions of heavy charged particles in matter", and Eq. (13) can be used with small modifications. There is one significant difference: when heavy charged particles interact with electrons in the medium, each collision produces minimal change to their momentum. When electrons interact with other electrons, their momentum can undergo significant change since their masses are the same.

Radiation region ($E > E_c$)

There is a significant difference between the energy loss rate of electrons and heavier charged particles in the radiation region, $E > E_c$. When an electron passes by an atomic nucleus, its trajectory can be deflected as a result of Coulomb scattering with the nucleus. A charged particle that is accelerating or decelerating in an electric field will emit energy as electromagnetic radiation known as bremsstrahlung radiation (literally translated from German to English as "braking radiation"). The probability of emission of bremsstrahlung radiation is inversely proportional to the square of the mass of the particle. As a result, bremsstrahlung radiation is much more of a significant source of loss of energy for electrons than it is for protons or other heavy charged particles.

The energy of bremsstrahlung radiation is equal to the difference between the kinetic energy of the electron before and after the deflection, and is proportional to $(E_0/m_e c^2)^4$ where E_0 and m_e are the energy and mass of the incoming particle (electron in this case) respectively. Its energy ranges from zero to the maximum kinetic energy of the bombarding electrons depending on how much the

Table 1 Critical energy E_c and radiation lengths X_0 in (g cm^{-2}) and (cm) for selected materials.

Material	Z	Density (g cm^{-3})	Critical energy (MeV)	Radiation length	
				(g cm^{-2})	(cm)
H ₂ (liquid)	1	0.071	340	62.8	887
He (liquid)	2	0.125	220	93.1	745
C	6	2.2	81.74	42.7	19.32
Al	13	2.70	42.7	24.01	8.897
Fe	26	7.87	21.68	13.84	1.757
Pb	82	11.35	7.43	6.37	0.56
Air		0.0012	88	36.62	30,390
Water		1	78.3	36.08	36.08
Soft tissue		1	81.7	37.6	37.6

Data from Tanabashi M, et al., (Particle Data Group) (2018) *Physical Review D: Particles and Fields* 98: 030001.

electrons are influenced by the electric field of the nucleus, therefore when taken as an ensemble, can produce the full spectrum of energies between 0 and E_{max} .

The radiative energy loss for large electron energies is

$$-\left(\frac{dE}{dx}\right)_{rad} \approx \frac{E}{X_0} \text{ or } E = E_0 e^{-x/X_0} \quad (16)$$

where X_0 is the radiation length given in either g cm^{-2} or cm. The radiation length (also called attenuation length) is the distance over which the bombarding electrons lose energy by a factor of $1/e$. A few values of X_0 and critical energy E_c are provided in Tanabashi et al. (2018) and given in **Table 1**.

When the photons generated by bremsstrahlung radiation have energies higher than 1 MeV, they can successively produce electron-positron pairs via pair production. The average distance that a photon travels before it produced electron-positron pair (X_p) is related to the radiation length (X_0) by $X_p = (9/7)X_0$. The secondary electron will successively produce more electron-positron pairs creating a shower. The number of electrons rapidly increase in the beginning as they pass through the target medium. As the shower process proceeds, the average energy per electron becomes smaller, and ceases when pair production is no longer energetically possible.

See Also: Modern Industry: Gamma Methods for Materials Testing and Inspection.

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Radiation Sources

David K. Wehe, Department of Nuclear Engineering and Radiological Sciences, University of Michigan, Ann Arbor, MI, United States

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Radioactive decay background

A nuclear reaction typically involves an incident particle that penetrates the nucleus and induces a change that results in a different nucleus and exit particle. However, it is possible that a nucleus spontaneously changes itself into a different state or type of nucleus along with some emitted particle. Nuclei that can undergo this spontaneous change are termed “unstable” although the time scale may run from pico-seconds to many millennia.

Because the emitted particle has its birthplace in the nucleus, its energy is normally sufficiently high to be able to cause ionizations in the surrounding materials it encounters. The most common particles are the beta particle (an electron), an alpha particle (a helium nucleus), a gamma ray (a high energy photon, or a neutron. It is the effect of these emitted radiations in the surrounding material that are often of most concern to us since they can result in beneficial or deleterious physical changes.

This unstimulated change is a manifestation of the probabilistic nature of quantum mechanics. We associate this decay process for each unstable nucleus as having a constant probability per time λ for a decay, which naturally leads to the radioactive decay law for the population $N(t)$ of radioactive nuclei (Knoll, 2010):

$$\frac{dN}{dt} = -\lambda N(t)$$
$$N(t) = N(t=0)e^{-\lambda t}$$

The term λN represents the rate of decay of the material having $N(t)$ radioactive nuclei, and is referred to as the *activity* of the material. Activity is measured in units of disintegrations per second (i.e., 1 Bq = 1 dps), or in a more practical unit, in units of *Curies* (e.g., 1 Ci = 3.7×10^{10} dps).

While the activity is a measure of the rate of radioactive decay of a material, it tells us nothing about the impact of those decays on the surrounding materials. Other units must be defined that reflect the physical impact of these emitted radiations. For physical damage to most materials, we care about the energy deposited in a unit mass of the material, and the associated measurement units are *rads* (1 rad = 100 ergs/g) or *Grays* (1 Gy = 1 J/kg). For biological impact, we must also account for the nature of the cellular damage, and a unit that accounts for this is defined as either the *rem* (a rad with a biological impact factor) or a *Sievert* (a Gray with the same biological impact factor). Just to provide a context for the scale of the units, one might be concerned with electrical deterioration of a microchip if it receives ~ 1 Mega-rad of dose, or a fatal dose of ~ 500 rem delivered to a person. But most exposures encountered in daily life are $\sim 10^{-3}$ of a rad or rem, making exposure to natural radioactivity generally considered rather innocuous compared to commonly encountered risks. However, such may not be the case around intense, man-made sources of radiation, such as within nuclear power plants, as discussed below.

Natural radioactivity

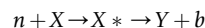
Radiation sources are usually classified as either natural or man-made. Natural ionizing radiation is ubiquitous because our Earth contains naturally radioactive isotopes that have survived the ~ 4.5 billion years since the earth was formed. Only a few lighter radioactive isotopes have survived, namely beta emitters of potassium-40 and rubidium-87. However, the heavier elements, i.e., the actinides, synthesized from repeated neutron capture and beta decay, have isotopes with half-lives on the scale of planetary evolution. Since these isotopes are prevalent in the Earth's crust, and all physical objects are derived from there, one cannot avoid some exposure to radioactive sources. Hence the familiar phrase “I’m radioactive and you are too.”

Whereas the midweight nuclei tend to decay by beta emission, these heavier nuclei prefer to decay through a succession of alpha emissions (with the occasional beta emission included) until they reach a stable endpoint. These long decay chains, which originate from either uranium-238, uranium-235, or thorium-232, give rise to a number of naturally occurring radioactive isotopes, such as the radon-222 noble gas of environmental concern.

Another source of natural radiation comes from the natural background of cosmic rays that interact in the atmosphere and form radioactive isotopes that enter our ecosystem. Most notable among these are carbon-14 produced by the interaction of neutrons and atmospheric nitrogen-14. In general, these cosmic sources of radiation are too small to be of biological interest under our protective atmosphere.

Man-made radioactive sources from neutron capture

Neutrons provide an easy means to create more interesting and substantial radioactive source. Without any electrical charge, they can enter the positively charged nucleus without hindrance. And with the addition of a neutron, the strong nuclear force within adds substantial energy to the enhanced nucleus that can permit a subsequent nuclear reaction. The resulting daughter nuclei are often-times unstable, and may lead to a further chain of radioactive decays. Such a process might be written as:



where n signifies a neutron, X is the target nucleus, X^* is the excited nucleus after absorbing the neutron, and Y and b represent the daughter nucleus Y and emitted particle b that emerge from the nuclear reaction. A more compact way to write this is $X(n, b)Y$, where we would classify the reaction type as (n, b) (Table 3 shows more general nuclear reactions of types (a, b)).

As a common example, exposing naturally occurring cobalt-59 to neutrons produces the radioactive cobalt-60 sources used for a variety of applications, including food sterilization or radiation therapy. Commercially viable radioactive materials can thus be made by exposing stable, naturally occurring isotopes to neutrons that are produced in either a research reactor or a dedicated accelerator source.

Radioactive sources and applications

Radioactive sources find common uses in industrial processes (e.g., thickness or moisture gauges (also known as “Troxler” gauges), sterilization, oil well exploration, radiography) as well as in space (e.g., thermoelectric generators) and medical applications (e.g., blood irradiators, brachytherapy, radiation therapy). Because a decaying radioactive source cannot be turned off, there has been a trend to transition to accelerator-based sources if the local infrastructure can support this more advanced technology. For example, modern radiation therapy centers have replaced cobalt-60 sources with electron (or even heavier ion) accelerators to deliver therapeutic doses of radiation. Furthermore, the accelerator-based generators of radiation do not carry the same level of security risks as the more portable radioactive sources.

Because of the potential health and environmental risks, most countries maintain regulatory controls over portable radioactive sources (and, to a lesser extent, accelerator-based sources of radiation). The International Atomic Energy Agency (IAEA) maintains a program (cf. IAEA, 2005) to help countries categorize and manage their inventory of radioactive sources. These include a surprisingly large number: Sr-90, Pu-238, Co-60, Cs-137, Se-75, Ir-192, Yb-169, Tm-170, Am-241, Pu-239/Be, Cf-252, Ra-226, I-125, Au-198, Kr-85, Pm-147, Cm-244, Ra-226, Cd-109, Gd-153, Po-210, Mo-99, I-131, Fe-55, Cd-109, Ni-63, H-3, Ru/Rh-106, Pd-103, Ge-68, and P-32. Table 1, drawn from these IAEA publications, lists some of these isotopes and their various applications (Ferguson et al., 2003).

Many people are familiar with smoke detectors although they may not realize the ionization-type of detector contains a very small amount (~ 1 microCurie) of radioactive Am-241. An alternate type, the photoelectric type, avoids this internal radioactive source, but is less sensitive to fast-burning fires. Combination units are available, but are more costly and thus less widely used. Because of what some consider an overabundance of concern, some communities now forbid installing ionization-type smoke detectors despite their safety advantages (U.S. Nuclear Regulatory Commission, 1979).

In addition to the IAEA, radioactive sources and their applications are also tabulated by the World Nuclear Association (n.d.-a). Within the United States, the Nuclear Regulatory Commission regulates and maintains information on current and former vendors of sealed sources and devices (The U.S. Nuclear Regulatory Commission, n.d.). Interestingly, a registry of the current and inactive sources located within the country is available.

The United States Department of Energy maintains an isotope development program (The U.S. Department of Energy Isotope Program, n.d.) whose goal is to produce *radioactive and enriched stable isotopes that are deemed critical or are in short supply*. A sample of the output from this national production network is shown in Fig 1. The DOE program matches radioactive source customers to source suppliers (nuclear reactors and accelerators) to satisfy needs not met by existing commercial suppliers.

As described above, much of the flow of commercial radioactive sources originates from dedicated low power nuclear reactors, and then is passed to commercial entities that convert the radioactive material into a viable product. A 2003 flowchart shows the global activity involvement in Fig. 2.

Radioactive materials as a byproduct of fission

One type of neutron induced reaction relevant to this encyclopedia is nuclear fission. For this reaction, the reaction products Y and b are termed fission products that are highly radioactive. In addition to the fission fragments, more neutrons are released during the reaction that perpetuate the reaction chain, leading to a buildup of highly radioactive nuclei. While each fission reaction immediately yields a large amount of energy (200 MeV) that can be used for electric power generation, a side effect is the initial highly radioactive waste products (e.g., neutron-activated structural materials along with fission products and their progeny) that remain in the core and persist after the reactions are terminated. While these fission products contain some valuable isotopes that could be extracted, with the exception of molybdenum-99, this has not proven to be cost-effective. Instead, the remnants from the generation of electric power from nuclear fission are a highly radioactive mixture of isotopes that must be managed.

Table 1 Applications using radioactive sources.

<i>Practice or application</i>	<i>Radioisotope</i>	<i>Typical radioactivity level (Curies)</i>
<i>Category 1</i>		
Radioisotope thermoelectric generators	Sr-90	30,000–300,000
Teletherapy	Co-60	1350–27,000
	Cs-137	13,500
Blood irradiation	Cs-137	50–2700
Industrial radiography	Ir-192	3–250
	Co-60	3–250
Sterilization and food preservation (irradiators)	Co-60	2700–11,000,000
	Cs-137	2700–11,000,000
Other irradiators	Co-60	27–27,000
	(Cs-137 rare)	27–27,000
<i>Category 2</i>		
High dose rate remote after loading brachytherapy	Co-60	0.27
	Cs-137	0.8×10^{-6} – 2.7×10^{-4}
	Ir-192	11
Low dose rate brachytherapy (manual or remote)	Cs-137	0.0014–0.014
	Ra-226	0.0008–0.008
	Co-60	0.0014–0.014
	Sr-90	0.0014–0.04
	Pd-103	0.0014–0.04
	Cs-137	0.027–2.7
Well logging	Am-241/Be	0.027–22
	(Cf-252 rare)	1.4
Level gauge thickness gauge conveyor gauge	Cs-137	0.27–27
	Co-60	0.027–0.27
Moisture/density detector (portable, mobile units)	Am-241	0.27–1.1
	Am-241/Be	0.0027–0.054
	Cs-137	Up to 0.0011
	Ra-226/Be	0.04
	(Cf-252 rare)	0.08
<i>Category 3</i>		
Level gauge density gauge	Cs-137	0.0027–0.54
	Co-60	0.0027–0.027
Thickness gauge	Kr-85	0.0027–0.08
	Am-241	0.027–0.27
	Sr-90	0.0027–1.1
	Tl-204	11
	Am-241	0.1×10^{-6}

Source: Adapted from IAEA. *Categorization of Radiation Sources*. Reproduced from Ferguson CD, Kazi T, and Perera J (2003) *Monterey Institute Occasional Paper 11. Commercial Radioactive Sources: Surveying the Security Risks*.

Surprisingly, most of these radioactive isotopes are relatively short-lived. In fact only seven have half-lives longer than 25 years, and, of those, only strontium-90 and cesium-137 have relevant half-lives (approximately 30 years) to be of health concern. Rather, the troublesome part of nuclear spent fuel is the low radioactivity of very long-lived transuranic isotopes produced from neutron absorption. This can be seen in the figure below which plots the activity per unit mass as a function of the time after shutdown. Determining an acceptable level of residual radiation, and thus defining the waiting time in isolation, remains an open question for society.

Before we close our discussion of radioactive sources produced by nuclear energy, we note an alternative measure of generating nuclear energy is through fusion. One of the main advantages of this mode of generating nuclear energy is avoiding the production of the very long-lived transuranic isotopes shown in Fig. 3. Thus, while fusion machines may actually produce larger volumes of radioactive materials (less so if neutrons are not released in the fusion reaction) the radioactive byproducts tend to be of shorter timescales and less objectionable to the public.

Accelerator based

With a limited number of nuclear reactors available to produce customized radioactive sources, accelerators are now widely used to produce many of the desired isotopes. Both cyclotrons and linear accelerators are employed to accelerate charged particles (protons,

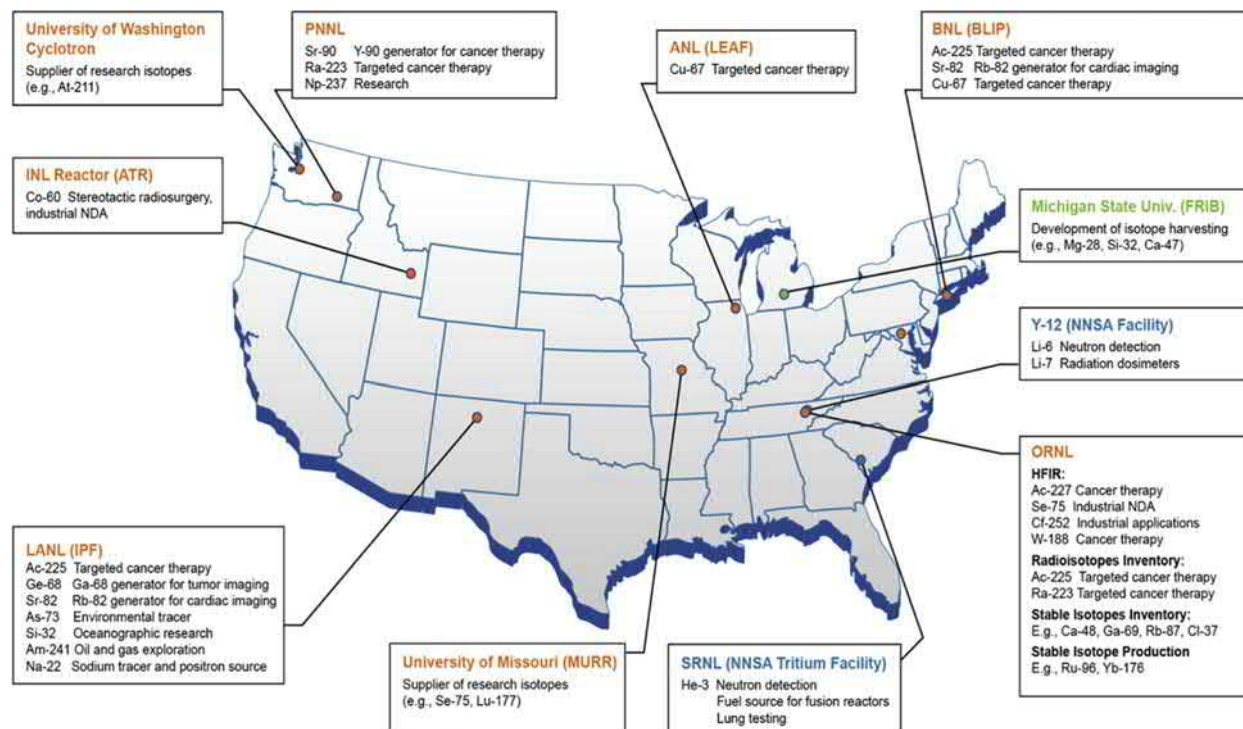


Fig. 1 Non-commercial production of radioactive sources within the United States.

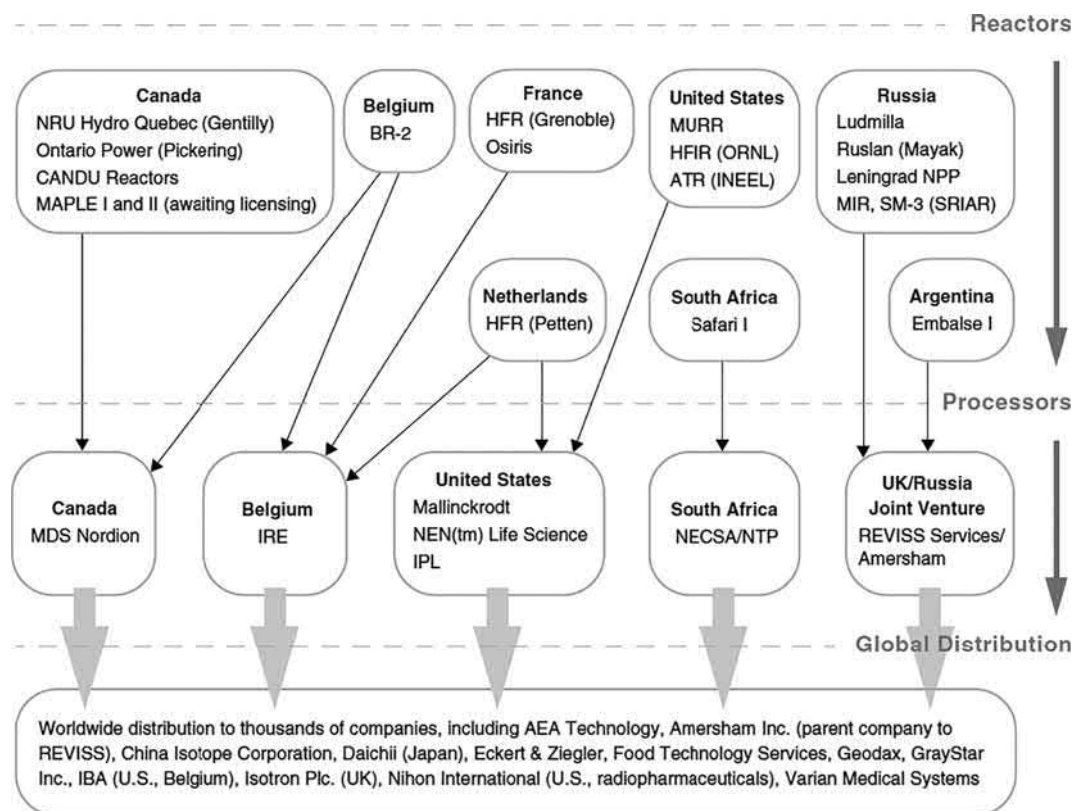
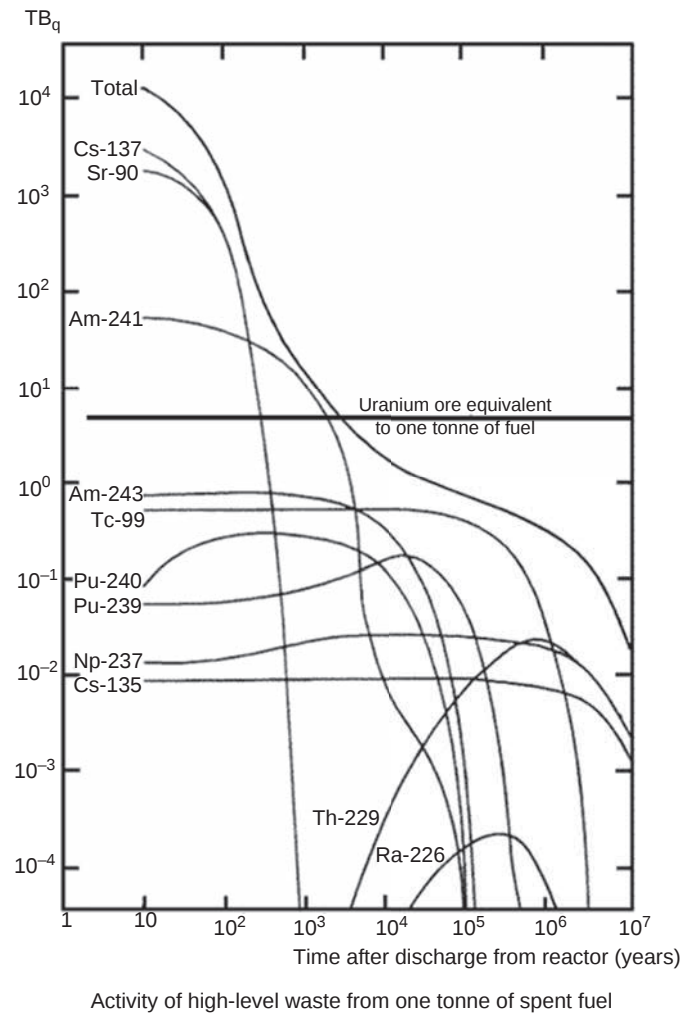


Fig. 2 Major radioisotope production and distribution pathways flowchart. Reproduced from Ferguson CD, Kazi T, and Perera J (2003) *Monterey Institute Occasional Paper 11. Commercial Radioactive Sources: Surveying the Security Risks*.



Source: IAEA. 1992 - radioactive waste management

Fig. 3 Most significant radioactive sources resulting from fission energy generation (World Nuclear Association, n.d.-b). International Atomic Energy Agency, 1992. Nuclear Power, Nuclear Fuel Cycle and Waste Management: Status and Trends 1992 (Part C of the IAEA Year Book 1992). IAEA, Vienna—radioactive waste management.

deuterons, alpha particles, or heavy ions) onto a target that is chosen to induce a particular nuclear reaction. The critical production parameters involve the incident particle energy (with lower energy requirements (<30 MeV) allowing the more compact cyclotron geometry) and beam current (defining the production rate). Commercial, university, and governmental entities operate accelerator facilities to produce specific radioisotopes.

Commercial entities (e.g., Nordion or IBA Radiopharma Solutions) usually operate lower power accelerators and have found a viable market as a reliable source of radiopharmaceuticals for clinic practice. A typical commercial accelerator might deliver a 1 mA beam of 30 MeV particles. In contrast, university hospitals typically operate even lower energy cyclotrons to produce quite short-lived positron emitters (C, F, N, O isotopes) used onsite in their nuclear medicine clinics for positron emission tomography (PET) imaging. Larger university accelerators are often associated with nuclear physics investigations, but these can also be adapted to produce radioisotopes. For example, the production of At-211 requires an alpha particle beam incident on a bismuth target—a configuration more akin to a physics experiment.

Government accelerators are also operated to provide a source of radioactive materials for societal use. They often can operate at higher energies and can produce either relatively rare isotopes being explored for new applications, isotopes unable to be produced through more conventional machines, or rarely needed isotopes that are essential in high priority situations. Table 2 provides a subset of international accelerators used for producing isotopes.

A major commercial and societal driving force for isotope production involves medicine, and Table 3 shows a list of commonly used isotopes and their production mechanisms. Accelerators would be used for the reactions in which the incident particle is charged, otherwise nuclear reactors would be used. The radiation emitted from these sources can be selected to penetrate deep into the body (for diagnostic testing) or to selectively destroy diseased cells (for therapy). The emergence of theranostics, i.e.,

Table 2 Parameters of government-operated higher-energy accelerator facility beams.

Facility	Energy MeV	Beam current (μ A)	Availability (months/year)	Main isotopes supplied
IPF USA	100	250	6–8	^{82}Sr , ^{68}Ge , ^{22}Na , ^{67}Cu , ^{32}Si , ^{73}As , ^{109}Cd , ^{72}Se , ^{88}Y
BNL USA	200	105	4–6	^{82}Sr , ^{68}Ge , ^7Be , ^{67}Cu , ^{86}Y , ^{65}Zn , ^{52}Fe , ^{83}Rb
INR Russia	160	100	3–6	^{82}Sr , ^{68}Ge , ^{22}Na , ^{103}Pd , ^{109}Cd
iThemba South Africa	66	120	6–8	^{82}Sr , ^{22}Na , ^{68}Ge , ^{73}As
TRIUMF Canada	500	150	6–8	^{32}Si , ^{82}Sr
PSI Switzerland	100	100	as required	^{82}Sr , ^{68}Ge , ^{67}Cu
Arronax France	72	10–70	as required	^{82}Sr , ^{68}Ge , ^{67}Cu
	70		as required	^{82}Sr , ^{64}Cu , ^{211}At

Reproduced from NSAC Isotopes Subcommittee (2015) *Meeting Isotope Needs and Capturing Opportunities for the Future: The 2015 Long Rang Plan for the DOE-NP Isotope Program*.

Table 3 Radioisotopes of interest in medicine and their major production routes.

Isotope	T1/2	Reaction	Comment/availability
^{44}Sc	3.93 days	$^{44}\text{Ca}(\text{p},\text{n})$	Low energy protons
^{47}Sc	3.35 days	$^{50}\text{Ti}(\text{p},\alpha)$	Enriched target needed
^{64}Cu	12.7 h	$^{64}\text{Ni}(\text{p},\text{n})$	University network
^{67}Cu	3.79 days	$^{70}\text{Zn}(\text{p},\alpha)$	Enriched target needed/spall. Mo
^{67}Ga	3.62 days	$^{70}\text{Zn}(\text{p},2\text{n})$	Commercial
^{68}Ga	1.13 h	$^{68}\text{Zn}(\text{p},\text{n})$	Low energy protons/generator
^{68}Ge	271 days	$^{68}\text{Ga}(\text{p},2\text{n})$	Used to make ^{68}Ga via generator
^{72}As	26.0 h	$^{72}\text{Ge}(\text{p},\text{n})$	Low energy protons
^{77}As	38.8 h	$^{80}\text{Se}(\text{p},\alpha)$	Low energy protons
^{76}Br	16.2 h	$^{76}\text{Se}(\text{p},\text{n})$	Enriched target needed
^{77}Br	57.0 h	$^{\text{nat}}\text{Mo}(\text{p},\text{spall})$	High energy protons
^{86}Y	14.7 h	$^{86}\text{Sr}(\text{p},\text{n})$	University network
^{90}Y	64.1	^{90}Sr generator	Commercial
^{89}Zr	3.27 days	$^{89}\text{Y}(\text{p},\text{n})$	University network
$^{99\text{m}}\text{Tc}$	6.0 h	^{99}Mo generator	Commercial
^{111}In	67.2 h	$^{112}\text{Cd}(\text{p},2\text{n})$	Commercial
$^{117\text{m}}\text{Sn}$	14.0	$^{116}\text{Cd}(\alpha,3\text{n})$	High energy alphas
^{123}I	13.2	$^{124}\text{Xe}(\text{p},2\text{n})$	Commercial
^{124}I	4.18 days	$^{124}\text{Te}(\text{p},\text{n})$	University network
^{131}I	8.0 days	Fission product	Commercial
^{149}Tb	4.12 h	$^{197}\text{Au}(\text{p},\text{spall})$	Online isotope separator
^{152}Tb	17.5 h	$^{197}\text{Au}(\text{p},\text{spall})$	Online isotope separator
^{161}Tb	6.91 days	$^{197}\text{Au}(\text{p},\text{spall})$	Isotope separator
^{177}Lu	6.65	$^{176}\text{Lu} + \text{n}$	Commercial via n capture
		$^{180}\text{Hf}(\text{p},\alpha)$	proton reaction for high specific activity
$^{195\text{m}}\text{Pt}$	4.0	n capture	
^{198}Au	2.69	n capture	High specific activity from $^{198}\text{Pt}(\text{p},\text{n})$
^{211}At	7.2 h	$^{209}\text{Bi}(\alpha,2\text{n})$	University network
^{211}Rn	14.6 h	$\text{U/Th}(\text{p},\text{spall})$	High energy/ ^{211}At source
^{223}Ra	11.4 days	Decay chain	Commercial
^{225}Ac	10.0 days	Multiple routes	Demand outstrips supply

Reproduced from NSAC Isotopes Subcommittee (2015) *Meeting Isotope Needs and Capturing Opportunities for the Future: The 2015 Long Rang Plan for the DOE-NP Isotope Program*.

the simultaneous execution of both therapy and diagnostics, offers the promise of improved outcomes based on patient-specific responses. For this modality, pairs of radioactive isotopes are chosen to accomplish the two tasks of locating and then treating. Since the chemistry for isotopes of the same element is the same (e.g., copper-64 is a positron emitter for imaging and copper-67 is an

electron emitter for therapy), one delivery pharmaceutical can be used to perform both tasks simultaneously. The exploration and development of these medically important isotopes has created a vibrant research area.

Neutron sources

In section *Man-made radioactive sources from neutron capture*, we mentioned that neutrons provide a convenient way to perturb the nucleus and induce a reaction. While nuclear reactors provide a copious supply of neutrons, more convenient and accessible sources have also been developed. These sources are used in density and moisture gauges, oil well exploration, neutron radiography, health physics instrument calibration, and physics research, among others. With the contemporary interest in homeland security and nonproliferation, these alternate sources have received renewed attention.

To produce free neutrons (other than from fission in a nuclear reactor), one can either (1) use nuclei that naturally fission as their radioactive decay mode, or (2) induce a nuclear reaction in which one product is a neutron (NRC, n.d.).

The first method mentioned, known as spontaneous fission, occurs in some particularly heavy nuclei with even numbers of neutrons and protons. Some examples include plutonium-238, curium-242, and curium-244, but by far the most commonly used neutron source is californium-252.

Californium-252 is quite convenient as a neutron source because its high specific radioactivity means that small amounts of the material are sufficient to produce a significant rate of neutron emission ($\sim 2 \times 10^{12}$ n/s/g). Unfortunately, Cf-252 has a relatively short half life of 2.6 years, so these sources weaken over time, and must ultimately be replaced. The radioactive material is usually doubly encapsulated in stainless steel, but still will occupy only a few cm³ or less. This makes them quite convenient and portable. Because the neutrons are produced by fission, they will have a characteristic average energy (~ 2 MeV) that proves to be so useful in the applications mentioned above. When a softer distribution of energies is needed for an application, the source can be buried inside a designed set of materials to tailor the fission neutron energy distribution.

The second method of producing neutrons involves inducing a nuclear reaction. As a target material, the beryllium nucleus is particularly attractive because beryllium-9 has one very loosely attached neutron. By striking that nucleus with an alpha particle produced by an alpha decay, a nuclear reaction can take place that produces a carbon-12 nucleus and a free neutron. By intimately mixing alpha emitting nuclei with beryllium, one produces a source of neutrons. These are known as alpha neutron sources and are the most common neutron source.

The neutron production rate and energies are thus governed by the mixture composition, and the decay rate and energy release of the alpha emitter. To ensure intimate contact, powdered beryllium metal is mixed with an oxide of the alpha emitter, compressed, and doubly encapsulated in a Type 304 stainless steel cylinder. Typical alpha activities ranging from 10 mCi to 40 Ci have been deployed for neutron sources. The alpha emitters most often chosen are either americium-241, plutonium-238, plutonium-239, polonium-210, or radium-226, and the resulting neutron sources are familiarly known as “AmBe”, “PuBe”, or “RaBe” sources. Unlike californium-252, the half-lives of these alpha matters are quite long and so the source retains its strength over time.

When lower energy neutrons are needed, americium can be mixed with fluorine or lithium, instead of beryllium, to produce average neutron energies of 1.5 or 0.5 MeV respectively.

Rather than using an alpha particle to produce an (alpha,n) reaction, high energy photons (~ 2 MeV gamma rays) can also be used. These types of neutron emitters are known as photoneutron sources, and typically use either deuterium (hydrogen-2) or beryllium (beryllium-9) as the target nucleus for the gamma ray. The incident gamma rays come from the radioactive decay of a high activity source, such as antimony-124 (half-life of 60 days). Because the gamma ray has a distinct energy, the neutrons are also produced at a fixed energy. By choosing different gamma ray emitters, different energy neutrons can be produced. Unlike the alpha neutron sources, the gamma rays are highly penetrating and can be kept physically separated from a surrounding neutron emitter. These neutron sources are generally larger in physical size ($\sim$ several cm³) and must be highly shielded due to the intense gamma rays. Photoneutron sources are useful when lower, fixed energy neutrons are needed, but because of the short half lives and high energy gamma rays involved, are only found in research laboratories.

Both alpha- and photo-neutron sources rely upon the energy provided by a radioactive source, and thus require no external power. On the other hand, they cannot be turned off. By using accelerators to provide the energy for the nuclear reaction, additional reactions become available (cf. Table 4), and the supply of neutrons can be turned off.

The most convenient and portable source of fast neutrons using an accelerator involves firing deuterium ions at either deuterium or tritium atoms. The resulting fusion reactions (known as (d,d) or (d,t) reactions) produce monoenergetic neutrons at ~ 2.5 or 14 MeV. Because of the larger reaction likelihood, the tritium targets typically will produce ~ 10 – 100 times more neutrons, and are the most favored. The target is usually composed of a copper or molybdenum substrate, covered by a metal halide film that incorporates either the deuterium or tritium atoms. The accelerators, which can be run in either pulsed or constant current modes, can be made small and highly portable, and the entire unit is commonly known as a neutron generator.

The available intensity of fast neutrons using commercial sealed-tube neutron generators is typically limited to 10^{10-11} n/s. More intense fast neutron sources that use the DT reaction are being developed with significantly higher intensity, but are much larger ($\sim$ a few m³) and expensive (Transforming Nuclear Technology, Phoenix Inc, n.d.). For R&D requiring even more intense sources of neutrons, spallation neutron sources have been deployed over the past few decades. These building-sized facilities use intense accelerators to drive protons (typically $\sim$ GeV) into heavy targets like lead, mercury, or tungsten. The accelerator type may be linear, or a circular variety, such as a synchrotron or cyclotron. When a high energy proton interacts with neutron-rich heavy nuclei, ~ 20 – 30

Table 4 Typical values for alpha- and photo-neutron sources.

Source	Neutron production rate ($\times 10^6$ n/s-Ci)	Average neutron energy (in MeV)	Neutron dose rate at 1 m/Ci (in mrem/h)	Gamma dose rate at 1 m/Ci (in mrem/h)
AmBe	2.2	4.2	2.5	2.5
PuBe	1.7	4.6	2	0.1
RaBe	15	3.6	—	High
Sb-Be	0.2	0.024	0.2	1000

neutrons are produced. These high energy neutrons are typically moderated to lower energies and used as the source for sophisticated neutron measurement stations. Because of the size, sophistication, and cost of these sources, these are typically international user facilities supported by national governments. A current list of 10 well-known global spallation sources is maintained by IAEA (IAEA, n.d.).

While the spallation neutron sources serve as our brightest source of neutrons, the equivalent international accelerator facilities for producing ionizing photons are known as synchrotron light sources or free-electron lasers. The photons are generated when high energy ($\sim$ GeV) electrons are bent by magnetic fields, and the emitted photon energies typically lie in the X-ray energy range of the electromagnetic spectrum. These also tend to be large scale user facilities particularly useful in materials science research. A list of global intense X-ray sources is maintained at [Light sources of the World \(n.d.\)](#).

See Also: Global Market for Radiation Applications; Harnessing Nuclear Radiation; ILW conditioning and performance; Medicine: Radionuclides Used in Nuclear Medicine; Modern Industry: Application of Neutrons for Materials Testing and Inspection; Natural and Man-Made Sources of Radiation; Public Safety: High Radiation Sources and Alternative Technologies.

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Experimental Tools I: Radiation Detectors

Ke Han, School of Physics and Astronomy, Shanghai Jiao Tong University, Shanghai, China

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Introduction

Radiation is the flux of microscopic particles or electromagnetic waves. In nuclear science, microscopic particles mainly refer to charged particles such as alpha particles, electrons, and heavy ions as well as neutral particles such as neutrons and neutrinos. X-rays and gamma rays are electromagnetic waves with energy in the range of keV to MeV and are treated as particles. Most radiations are the products of radioactive decay (Previtali, 2021). To characterize radiation, we study its energy deposition intensity, frequency, timing profile, etc., by measurement devices designed and constructed for this purpose, which we call radiation detectors. Particles interacting with the active medium in radiation detectors lose energy via complicated physical processes, including ionization and excitation.

Radiation detectors, or detectors for short, can be classified by the absorbing medium. Commonly used detector mediums include scintillators, gases, semiconductors, etc. Detectors can also be grouped by the type of radiation detected. For example, an alpha detector measures the number and energy of incident alpha particles—the nuclei of helium atoms. In this chapter, we focus on the detecting principles and characteristics of the three types of most widely used detectors. We will also briefly discuss detectors by the radiation types and auxiliary systems for detectors.

Physics of radiation detection

Radiation is detected via interactions with the detector medium. The interacting mechanism of charged particles and neutral particles in a detector varies significantly. Charged particles ionize and excite the atoms and/or molecules of the detector medium. Depending on the detector type, ionization and excitation signals may be measured either directly or indirectly. For example, excited molecules may return to the ground state by emitting visible light, which can be measured. Neutral particles can be detected only through the secondary charged particles generated when the neutral particles interact with the detector medium.

The physical process of radiation detection is also energy dependent. Possible interactions of gamma rays with the medium are the photoelectric effect, Compton scattering, and pair production. Depending on the incident gamma ray energy, one of the interactions may dominate the energy loss process. We discuss the energy dependence in the following subsections in more detail. A more extensive discussion can be found in Knoll (2010) and Leo (1994).

Energy loss in a medium: Charged particles

Common charged particles include alpha particles, beta particles (i.e., electrons), protons, heavy ions, etc. When a charged particle travels in a detector medium, it loses energy by ionizing and exciting the medium via electromagnetic interactions. In the ionization process, the incident particle knocks out one or more electrons from a target atom. Under the possible influence of an external electric field, electrons and ions may reach readout electrodes and send signals to detectors. In excitation, the incident particle pumps atoms or molecules to excited states, and photons are released in the subsequent de-excitation. Photons can then be collected by photosensors as a measure of the energy deposition of the incident particles.

The energy loss by ionization and excitation per unit travel distance dE/dx is proportional to the atomic number Z and inversely proportional to the atomic mass of the medium. dE/dx is also a function of the velocity of the incident particle. In nuclear physics, the unit commonly used for dE/dx is $\text{MeV}/(\text{g}/\text{cm}^2)$, where MeV is the unit of energy and g/cm^2 denotes the surface density of the medium. With this unit, dE/dx is largely independent of the physical properties of the medium. The energy loss does not vary monotonically with the incident energy. If we take the electron in silicon as an example (shown in Fig. 1), the energy loss dE/dx decreases as the energy increases from keV to approximately 1 MeV. Above 1 MeV, the energy loss has an additional contribution from radiation and increases with incident energy. For the energy around 1 MeV, the energy loss is relatively flat at approximately $0.2 \text{ MeV}/(\text{g}/\text{cm}^2)$.

Energy loss in a medium: X-rays and gamma rays

X-rays and gamma rays lose energy in a medium in three different ways: the photoelectric effect, Compton scattering, and electron-positron pair production. These three effects, respectively, dominate the energy loss when incident photons are in the low energy range (less than 100 keV), medium energy range (approximately 1 MeV), and high energy range (much higher than 1 MeV). Fig. 2 shows the total cross section of photon in germanium, which refers to the probability of photon interacting with the medium as a function of incident energy.

Photoelectric effect: the energy of a photon is fully absorbed by an electron in an atom, and the electron escapes from the atom or even the medium. This photon absorption and electron emission process was termed as the “photoelectric effect.” When an electron in the inner shell of an atom gets knocked out by a photon, the vacancy in the inner shell is filled by an electron from an outer shell. In this process, additional energy is carried away by an X-ray or another electron in the atom (called an Auger electron). The cross section of the photoelectric effect is proportional to the atomic number Z to the fifth power (Z^5).

Compton scattering: the incident photon scatters off one of the electrons in the target atom and transfers part of its energy to the electron. The energy of the outgoing photon is a function of the scattering angle θ , electron mass E_e and incident photon energy E_γ : $E_{\gamma'} = \frac{E_\gamma}{1 + E_\gamma/E_e(1 - \cos\theta)}$. The energy loss $\Delta E = E_\gamma - E_{\gamma'}$ increases with the scattering angle and reaches a maximum value when $\theta = \pi$,

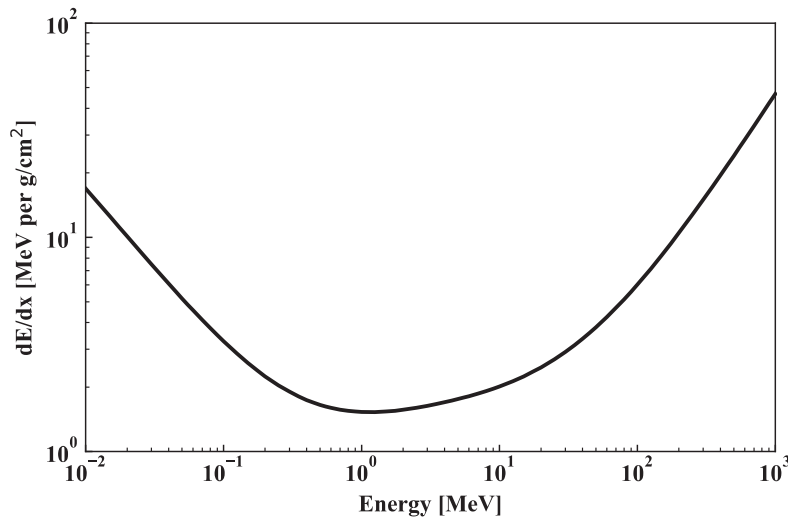


Fig. 1 Energy loss per unit travel distance in silicon as a function of incident electron energy. Data from NIST ESTAR database: <https://physics.nist.gov/PhysRefData/Star/Text/ESTAR.html>.

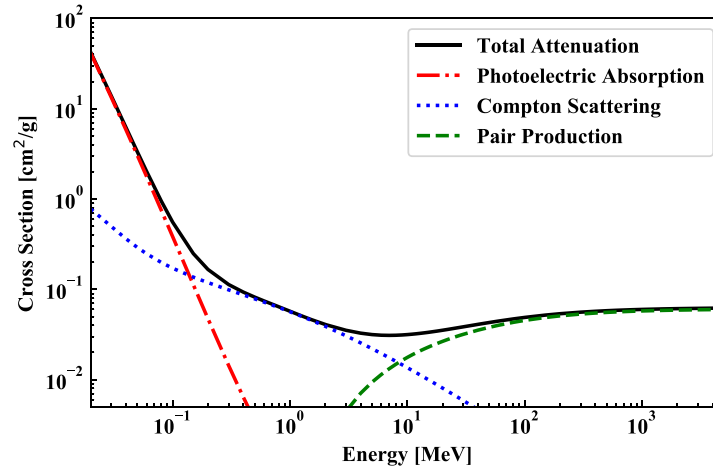


Fig. 2 Cross section of gamma interacting with germanium. The total cross section is dominated by three interaction mechanism at different energy ranges. The mean free path of a 1 MeV gamma ray is approximately 3 cm in germanium. Cross section data from NIST XCOM database: <https://physics.nist.gov/PhysRefData/Xcom/html/xcom1.html>.

that is, the photon is scattered backward by 180 degrees. In the medium, the scattering cross section is proportional to the atomic number Z and roughly inversely proportional to E_γ .

Pair production: above the energy threshold of approximately 1 MeV, an incident photon is converted to an electron-positron pair under the influence of the Coulomb field of the nucleus. The cross section for pair production increases with the photon energy and starts to dominate at energies above 10 MeV.

The total photon cross section is the sum of all three interactions, each dominating at different energy range. Photons are usually scattered by a large angle in Compton scattering and are absorbed when the photoelectric effect or pair production occurs. Accordingly, one cannot define a stopping range for incident photons. Instead, a signature attenuation length is often defined, after which distance the number of photons is reduced to $1/e$, or approximately 37% of the initial number.

Energy loss in a medium: Neutrons

Neutrons lose energy in a detector medium by nuclear interaction or elastic scattering off a nucleus. We usually classify neutrons into slow neutrons (energy lower than 0.5 eV) and fast neutrons (> 0.5 eV). Slow neutrons are absorbed by nuclear interactions, for example, $^{10}\text{B} + n \rightarrow ^7\text{Li} + \alpha$. The resulting charged particles, an alpha and a ^7Li nucleus in this example, are then detected. A fast neutron transfers energy to the medium more likely by elastic and inelastic scattering off the nuclei. As its energy decreases, the cross section for neutron absorption increases, increasing the probability that the neutron will be absorbed, as shown in Fig. 3.

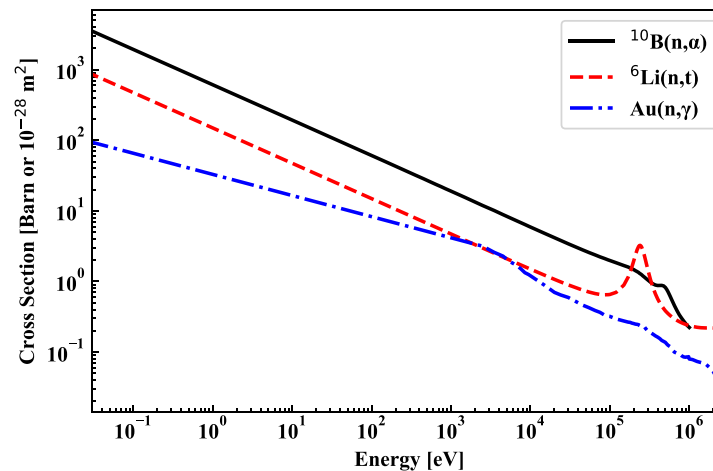


Fig. 3 Cross section of neutron capture in different isotopes. The commonly used detector material ^{10}B and ^6Li have larger cross section compared with Au. Data from IAEA Neutron cross section standards database: <https://www-nds.iaea.org/standards/>.

Counting statistics and energy resolution

Energy loss in a medium is a stochastic process. The number of quanta, a term sometimes used to refer to the unit energy carrier, including scintillation photons, ionization electrons, and vibrational excitations (phonons), created in a detection process, is subject to statistical fluctuation. The average energy required to create one quantum is denoted as w and differs significantly for different types of detectors. For example, the w value for a silicon semiconductor detector at room temperature is approximately 3.6 eV. Therefore, the average number of electron-hole pairs $\bar{n}$ created in a silicon detector by a 1 keV X-ray is approximately 280. However, the measured number n does not exactly equal $\bar{n}$ but follows a Poisson distribution with a mean value of $\bar{n}$.

The statistical fluctuation imposes an intrinsic limitation of the performance that can be achieved by any type of detector. For a monoenergetic energy input E , the full width at half maximum (FWHM) in the measured spectrum is used to characterize the detector performance and is called the energy resolution. The relative energy resolution, FWHM/E , is widely used as well and is often quoted as a percentage. Broadening of the energy peak due to the statistical fluctuation of n is intrinsic and determines the ultimate resolution achievable by certain types of detectors. For a large $\bar{n}$, the Poisson distribution approaches a Gaussian distribution, and the FWHM is proportional to 2.35σ , where $\sigma = \sqrt{\bar{n}}$ is the standard deviation of the Gaussian function. The relative energy resolution is therefore proportional to $1/\sqrt{\bar{n}}$. To achieve a better resolution, a larger $\bar{n}$ or equivalently a smaller w is desirable.

In reality, the statistical fluctuation is even more complicated. The creation of quanta in detectors is not fully independent; therefore, the observed variance differs from the predicted variance from the Poisson distribution. The ratio between the two is called the Fano factor F (Fano, 1947), and the relative energy resolution is proportional to $\sqrt{F/\bar{n}}$. Clearly, a smaller Fano factor is preferred to have a better energy resolution.

Scintillation detectors

Scintillation detectors are widely used for radiation detection in nuclear science, particle physics, medicinal imaging, homeland security, etc. They are versatile, fast, robust, and relatively low cost. The detector consists of two main parts: a scintillator and a photosensor. Scintillators are the target medium and emit light when radiation interacts with it (shown in Fig. 4). The target can be solid, liquid, or gas. The output light, mostly ranging from ultraviolet to infrared light, is then converted to electrons in the photosensors. The so-called photoelectrons are often multiplied and then converted to electric signals to be processed by electronics and computers.

Principles

The scintillator absorbs radiation and emits light. Incident particles excite scintillator molecules, and the subsequent energy release is in the form of photons in the de-excitation process. To work with photosensors, scintillating materials that emit photons in the visible light range are most widely used. With the development of novel photosensors, scintillation light from ultraviolet to infrared have become detectable with high efficiency, triggering new applications of scintillators.

To maximize the performance of scintillation detectors, the following aspects of the detection process contribute to the signal-to-noise ratio.

1. Light yield defined as the number of scintillation photons per unit energy deposition. The light yield depends on the properties of the scintillator, the incident radiation type, and the incident energy.

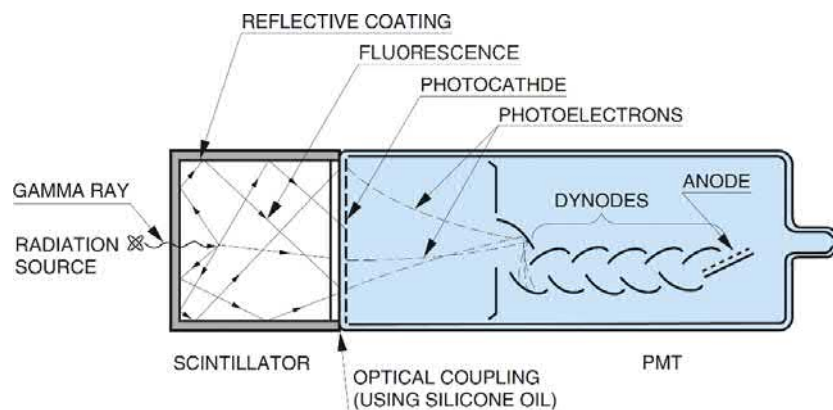


Fig. 4 A typical scintillation detector setup with a photomultiplier (PMT). Image courtesy: Hamamatsu (PHOTOMULTIPLIER TUBES Basics and Applications FOURTH EDITION).

2. Transparency of the emitted light in the scintillator. An ideal scintillator would have little overlap between the emission spectrum and absorption spectrum. Accordingly, the majority of emitted photons would reach the boundary of the scintillator, where the photosensors would collect them.
3. Collection efficiency of photosensors. This factor depends on the overall coverage and the quantum efficiency (QE) of the photosensors. Often, photosensors cannot cover the full surface area of the scintillator due to design and/or cost concerns. For example, a cylindrical NaI crystal is nominally coupled to a photosensor on one end, while all the other surfaces are covered with light reflectors. Light reaching the reflectors would be bounced back and inevitably lose its intensity before eventually reaching the photosensor. The photosensor's quantum efficiency, defined as the probability of converting one photon to an electron, is also directly coupled to the overall collection efficiency.
4. Photosensor multiplication gain. To achieve a sizable electric signal, the photosensors multiply the number of photoelectrons by factors from thousands to millions. The multiplication is achieved by either secondary emission in photo-dynode materials of a traditional photomultiplier tube or avalanches in semiconductors, as described later.

Other characteristics, such as the timing profile of light emission, should also be considered when selecting a scintillation detector.

Scintillators

Scintillators, in the form of solids, liquids, or gases, are normally categorized as either inorganic or organic. Inorganic scintillators, traditionally referred to as solid scintillators, usually have higher density and a larger atomic number Z , which means that they are effective in detecting both charged and neutral particles with relatively high energy resolution. However, the decay time of light emission is usually long (at the microsecond level), and the cost is relatively high. Note that noble gases, such as argon and xenon, are excellent scintillators with light emission spectrum in the UV region and light yields comparable to common benchmark scintillators. Noble gas scintillators are gaining more attention in fundamental research and applications (Aprile et al., 2006).

The most widely used inorganic scintillator type is alkali halides, including NaI and CsI. These compounds are often doped with trace amounts of activator to reduce the average energy required to emit a photon and thus improve the light yield. For example, thallium-activated NaI has a high light yield of approximately 40,000 photons per MeV. The emission spectrum peaks at 410 nm and can be easily measured by photosensors. Therefore, NaI(Tl) is often cited as the benchmark scintillator material for comparison with others.

Lanthanide halides are a relatively new family of inorganic scintillators that have very bright output and excellent energy resolutions. Among them, lanthanum bromide (LaBr_3) doped with cerium emits approximately 60 thousand photons per MeV and has good light yield proportionality. The best energy resolution can be as small as 2.8% FWHM at the 662 keV gamma line and is better than that of any other scintillator. For comparison, the typical resolution achieved with NaI(Tl) is approximately 5%. However, the crystals are still quite costly and limited in size when compared with NaI. Other types of inorganic scintillators include alkali-Earth halides, transition metal scintillators, post-transition metal scintillators, rare-Earth oxyorthosilicates, ceramic and glass scintillators, etc. (McGregor, 2018).

Organic scintillators are sold by commercial companies as a cocktail of two or three components in plastic or liquid form (see Fig. 5, for example). The scintillating component is a mixed with a solvent, mainly used as a stabilizer. For example, the KamLAND neutrino detector is equipped with a kiloton of commercial liquid scintillator with two main components: pseudocumene and normal dodecane (Suekane et al., 2004). The former is the scintillator, and the latter is a stabilizing agent used to improve the light



Fig. 5 A wide range of organic scintillators offered by Eljen Technology. Image courtesy: Eljen Technology; <https://eljentechnology.com/products/plastic-scintillators>.

transparency and stability. A third component might be added as a wavelength shifter to shift the emission spectrum of the scintillating component to a range more easily detected by photosensors.

Doping organic scintillators with trace amounts of impurities without significantly impacting the light-emitting performance is also relatively easy to accomplish. A sample to be measured can be mixed in the scintillator (liquid scintillator for practical reasons) for a high-efficiency measurement. Additionally, a special target material, for example, ^6Li with a large neutron absorption cross section, can be mixed in a scintillator to measure a specific radiation.

Photomultipliers and modern photosensors

Photomultipliers (PMTs) are workhorse photosensors that measure the amount of light output from scintillators. A PMT consists of a photocathode, multiple stages of dynodes, and an anode, all of which are packaged in a vacuum tube (see Fig. 6 for illustration). Photons enter the PMT via a front window, usually made of glass or quartz for different light ranges. Electrons are ejected through the photoelectric effect in the photocathodes coated inside the window. The quantum efficiency depends on the material of the photocathode and the wavelength of the incident photons. Under the influence of an electric field, an electron is accelerated to hit the first dynode and knock out multiple secondary electrons. The same process is repeated in later dynode stages, and the final number of electrons reaching the anodes can be 1 million or more. This large number of electrons will trigger a substantial electric signal to be read out by the electronics. The essential characteristics of PMTs are the sensitivity area, response spectrum, QE, and electron gain.

Silicon photomultipliers (SiPMs) are gaining popularity as a new generation of photosensors. The active areas of SiPMs are fabricated via standard lithographic techniques with silicon wafers. A SiPM has an area typically on the mm^2 to cm^2 scale and consists of millions of photodiodes, each of which has sides of a few tens to a few hundreds of microns. The photodiodes are biased in the avalanche mode and give a signal when a single photon triggers the avalanche. In the nominal working mode, not more than one single photon hits one photodiode, and thus, the number of photons can be counted by the number of triggered diodes. Similar to the QE concept for PMTs, a photon detection efficiency (PDE) exists for SiPMs, indicating the possibility of an incoming photon triggering the avalanche. Typical PDE can be comparable to or larger than the QE of PMTs. SiPMs can be sensitive to light from the UV to near-infrared range. The SiPM has a few clear advantages over the PMT. It is more compact in size, not susceptible to the influences of an external magnetic field, and operates with biases smaller than 100 V. Currently the cost per unit area is still higher than that of PMT, but is expected to decrease with wider usage and larger demand.

Gaseous detectors

Working modes

As in the general case, radiation loses energy in a gaseous detector medium by ionizing the gas molecules. A bias voltage is usually applied to the medium and is able to pull the electrons and ions in opposite directions. Although the charges from ions can be read out, a gaseous detector measures the electrons much more often. When drifting along the electric field lines, electrons may recombine with ions, become attached or absorbed by other molecules, and diffuse along transverse directions. Gaseous detectors are designed to carefully compensate for these negative effects and measure the energy, momentum, and trajectory of incident radiation or simply count the number of incident particles.

Gas mixtures of one type of noble gas and a quencher gas are most widely used in gaseous detectors. One example is argon gas mixed with 5% isobutane (C_4H_{10}). The W values, average energy required to create an electron-ion pair, are 26 and 23 eV, respectively, for the two gases. Therefore, the quencher gas decreases the W value of the mixture and reduces the Fano factor. More importantly, the quencher gas suppresses violent avalanches and improves the stability of operation, hence its name (Knoll, 2010).



Fig. 6 PMTs from Hamamatsu Photonics. Image courtesy: Hamamatsu Photonics; https://www.hamamatsu.com/us/en/product/optical-sensors/pmt/pmt_tube-alone/index.html.

Different effects dominate the charge collection process for different bias voltage ranges, which are often called operation regions, as illustrated in Fig. 7. At very small or no bias voltage, the recombination of electrons and ions plays a major role. As the bias increases, electrons more effectively drift away from the ionization site and are collected by the anode. This range is usually called the ionization region. With a higher drift voltage, electrons may be accelerated enough to trigger secondary ionization, where more electron-ion pairs are created and in turn may trigger more ionization. This cascade ionization is called an avalanche. Under a certain bias range, the total number of collected electrons, although perhaps orders of magnitude more than the number of primary electrons, is proportional to the initial energy deposition, and thus, the region is named the proportion region. When operated in ionization or proportional regions, the measured number of electrons is proportional to the energy deposition and thus useful for energy measurement. The proportionality can be maintained up to a certain bias voltage, above which the avalanche saturates and the total number of collected electrons no longer reflects energy deposition. This region is called the Geiger-Mueller region, named after the two scientists who discovered it.

The following three types of gaseous detectors are operated in the latter three regions and are the foundations of more complicated detectors.

Ionization chambers

Ionization chambers are operated in the ionization region. The geometry of the electric field in the chambers is often rectangular or cylindrical. A rectangular electric field is sandwiched between a pair of plate electrodes, and a cylindrical electric field has electrodes on the side and axis. Since no avalanche and thus no multiplication of ionized electrons occurs, the ionization chamber is more suitable for counting the number rather than measuring the energies of individual incident particles. It is also widely used for measuring heavy particles with high energy (MeV or higher). One application is to measure fission fragments from uranium or thorium. On the other hand, because the number of electron-ion pairs is not sensitive to the bias voltage, ionization chambers are robust and low-maintenance. A household smoke detector is an ionization chamber with open air as the gas medium and two plate electrodes biased by batteries. Alpha particles from an ^{241}Am source ionize the air continuously, and a stable signal exists in the smoke detector. In the case of excessive smoke in the ambient air and thus in between the electrodes, ionized electrons are absorbed by smoke particulates, and the electric signal is interrupted, triggering an alarm.

Proportional chambers

Proportional chambers can count the number and measure the energies of incident particles. The electric field is nominally 10 kV/cm or higher to reach the proportional region. Cylindrical chambers are often used to reach this high electric field around the electrode along the axis. With their high gain, proportional chambers are often used to measure neutrons, electrons, and X-rays. Proportional tubes with BF_3 or ^3He are widely used for neutron measurement.

Geiger-Mueller counters

Handheld Geiger-Mueller counters are ubiquitous equipment to measure the radiation level in an ambient environment (Fig. 8). Cylindrical tubes with electrodes biased at the Geiger-Mueller region are sensitive to very low levels of alpha, beta and gamma radiation. Due to its high multiplication factor, the large pulsed signal can be read out by straightforward, inexpensive, and robust

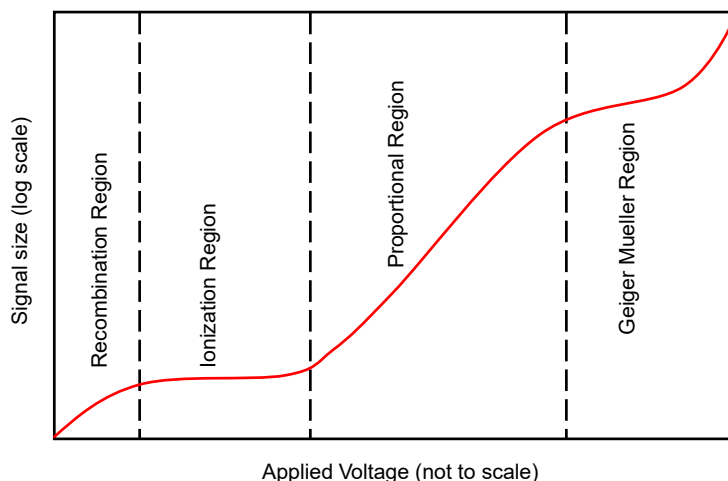


Fig. 7 Illustration of operation regions of gaseous detectors.



Fig. 8 A handheld Geiger-Mueller counter commonly seen in research labs. Public domain image. https://en.wikipedia.org/wiki/Geiger_counter#/media/File:Geiger_counter_2.jpg.

electric circuits. Furthermore, the gas tube itself is also of low cost and robust compared to other radiation counters (semiconductors, for example). However, the Geiger-Mueller counter cannot measure energy or identify radiation types.

Position-sensitive detectors with wire readouts

More complicated gaseous detectors have been developed to measure the energies and trajectories of incident particles. Instead of a single readout channel as in the basic types, these new types of gaseous detectors with high spatial granularity may have hundreds, thousands, or even more anode channels between two parallel cathode plates. Multiwire proportional chambers (MWPCs), as the name suggests, have multiple anode wires in the setup, each one serving as an independent proportional chamber (Charpak et al., 1968). The wires are usually equally spaced, and each wire is sensitive to the nearby space. Once an incident particle travels inside the chamber and deposits energy along the way, the corresponding wires register the signals. With two groups of anode wires in orthogonal directions, the trajectory's projection in the detector plane can be determined.

The time projection chamber (TPC) further develops the concept of the MWPC to achieve a three-dimensional spatial resolution (Nygren, 1974). The TPC enlarges the gap between the anode wires and cathode to obtain a dedicated drift region. Ionized electrons in the region drift to the anode wires, and the travel time multiplied by the electron drift velocity gives the position along the drift direction, i.e., normal to the anode plane. TPC detectors are widely used in nuclear and particle physics experiments. Fig. 9 shows a large TPC from the STAR experiment at Brookhaven National Laboratory during the construction phase.

With the rapid development of lithography technology, microstructured readout planes are becoming a viable substitute for the traditional anode wire planes. The micromesh-gaseous structure (MICROMEGAS) (Giomataris et al., 1996) and gas electron multiplier (GEM) (Sauli, 1997) are two successful electron multiplication and readout schemes.



Fig. 9 TPC of the STAR detector at Brookhaven National Laboratory, USA. Image courtesy: Roy Kaltschmidt, Berkeley Lab, <https://www.symmetrymagazine.org/article/october-2012/time-projection-chambers-a-milestone-in-particle-detector-technology>.

Semiconductor detectors

Semiconductor detectors have revolutionized radiation detection technology since they were first introduced approximately 60 years ago. Similar to gaseous detectors, semiconductor detectors measure the electron-ion pairs created by incident radiation. Commonly used semiconductors include silicon, germanium, and compounds such as CdTe and GaAs. The W value for Si is 3.6 eV at room temperature and slightly higher at lower temperatures. Germanium semiconductors, usually operating in a cryostat, have a W value of 3.0 eV at liquid nitrogen temperature. The W value in semiconductors is approximately an order of magnitude smaller than that in gas; thus, the number of electron-ion pairs created per unit energy deposition is much higher. The resulting energy resolution of semiconductor detectors is far superior to almost all other technologies.

In semiconductors, electric charge can be transported by either electrons or holes. When electrons are the dominant charge carriers, the semiconductor is called N-type. Otherwise, it is called P-type. Intrinsic P-type (N-type) semiconductors can be doped with implants to provide additional free electrons (holes) and reverse their type. With implantation, P-type and N-type materials can be joined together to form P-N junctions. When the N-type end is biased positively and the P-type end negatively, a large depletion region is formed in the P-N junction, where no free moving charge carriers exist. The sensitive volume of a semiconductor detector is the depletion region. With reversed biasing, the insensitive volume is thin, and the depletion region can occupy the vast majority of the detector volume. When an incident particle ionizes the detector and creates electron-hole pairs, it travels to the electrodes under biasing and creates a pulsed signal. Similar to gaseous detectors, charge carriers may also become trapped while drifting. To maintain good proportionality between energy deposition and pulse amplitude, the so-called lifetime of charge carriers should be much higher than the drift time. The lifetime depends on the crystal quality, purity, and fabrication of the semiconductor is key to good energy resolution.

Silicon detectors

Monolithic silicon detectors have been widely used to measure the energy of proton, alpha, heavy ion, and X-ray radiation. Aided by the ever-evolving silicon-based integrated circuit industry, strip or pixel silicon detectors are maturing and can measure the tracks of incident particles with high granularity.

The monolithic P-N-junction-type silicon detectors have good energy resolution and fast response to the energy loss of alpha and other charged particles. For 5.486-MeV alphas from ^{241}Am , the energy resolution (FWHM) is less than 10 keV. The electrodes can be fabricated via coating, thermal diffusion, and ion implantation. The surface areas of these alpha detectors can be as large as 3000 mm², but the thickness is limited to approximately 2 mm. Therefore, silicon detectors are suitable to measure the energy loss of alpha particles or charged particles but not high-energy gamma rays. When measuring penetrating particles, the thickness uniformity of the detector is also an important specification. Silicon detectors with drifted lithium, usually called Si(Li) detectors, create an intrinsic layer inside the P-N junctions. Si(Li) can be larger in volume and are used for X-ray detection.

Highly segmented silicon detectors such as silicon microstrip detectors and charge-coupled devices (CCDs) may have thousands or even millions of strip and/or pixelized channels. Each channel is a miniature P-N junction to measure the timing and energy information of passing radiation. Together, the detectors measure one- or two-dimensional spatial information with high granularity. Development of those tracking detectors has benefited from the integrated circuit industry. Currently, the possibility of integrating detection and front-end electronics on a single silicon wafer is being actively pursued.

Germanium detectors

Germanium detectors must be operated at low temperature, usually being cooled by liquid nitrogen (77 K) dewars. Compared with silicon semiconductors, the band gap of germanium semiconductors is so small that the leakage current due to thermal fluctuation at room temperature is too large to ignore. Germanium can be diffused with lithium to form Ge(Li) detectors, but the most common usage is high-purity germanium (HPGe) detectors (Fig. 10).

HPGe detectors are made of large, single germanium crystals with P-N electrodes. The impurity level must be less than the 10¹⁰ atom/cm³ level. The crystals can be larger than a kilogram. By maximizing the detector size while keeping the junction capacity low, HPGe detectors have the best energy resolution of all commonly used radiation detectors. For gamma radiation of a couple of MeV, the energy resolution (FWHM) can be as good as 0.1%. Fig. 11 demonstrates the sharp contrast between the performance of a sample HPGe detector and a NaI scintillation detector. Considering the relatively large atomic number ($Z = 32$) and thus the effective absorption of high-energy gamma rays, HPGe detectors are the ideal detector for gamma spectrometry from 0.01 MeV to a few MeV.

An array of germanium detectors can be closely packed together to measure the extremely low intensity of X-rays or gamma rays from rare nuclear interactions or decays. Arrays such as GAMMASPHERE (Lee, 1990), AGATA (Akkoyun et al., 2012), and GREINA (Fallon et al., 2016) are packed into spherical configurations to maximize the detection efficiency of events originating from the spherical center. One single HPGe crystal can also be segmented to make multiple detectors, with the objective of better position resolution.



Fig. 10 Selection of HPGe detectors with cooling dewars offered by Mirion Technologies, Inc. Image courtesy: <https://www.mirion.com/products/germanium-detectors>.

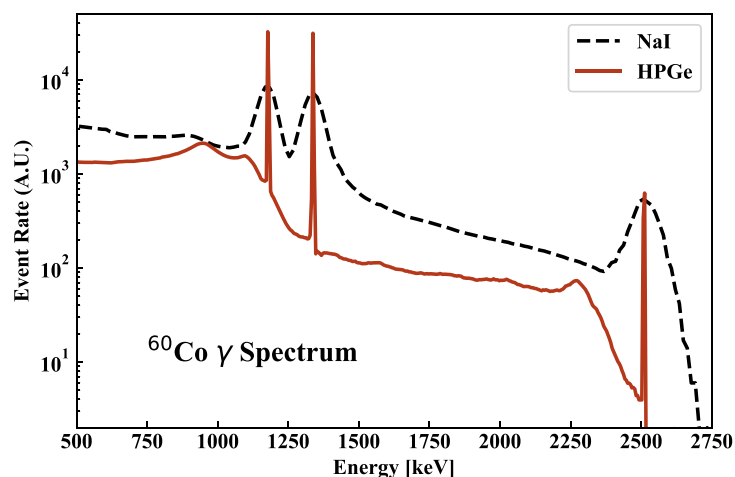


Fig. 11 Comparison of energy resolution from a sample HPGe semiconductor detector and a NaI scintillation detector.

Compound semiconductor detectors

Compound semiconductors such as CdTe, CdZnTe, and GaAs can be fabricated into room-temperature detectors to detect X-rays, gamma rays, and neutrons. The energy resolution is not as good as that of germanium detectors, but the operation at room temperature is a huge benefit. For this reason, compound semiconductor detectors are gaining popularity in the X-ray screening and medical imaging industries.

Specialty detectors

Specialty detectors, such as bolometers, bubble chambers, and emulsion-tracking detectors utilize special features of a detector medium. For example, bolometers measure the incident energy via the temperature change of the detecting medium itself. To have a measurable temperature rise with input energy levels in nuclear decays (MeV range), the detector has to be cooled to sub-Kelvin temperature to significantly reduce the heat capacity. The cryogenic requirement limits its application to research labs.

It is worth noting that the ubiquitous radiation monitoring badges use emulsion-etching detection techniques. Emulsion detectors consist of ionization-sensitive silver halide grains embedded in gelatin films. Incoming radiation sensitizes the grains and causes silver atoms cluster on the surface of grains. Once developed, the sensitized grains appear visibly darken and are good indicators of the flux of radiation. To enhance the dynamic range of radiation detection, a radiation monitoring badge may have emulsion films of different sensitivity, or with different types filters in front of the films. An emulsion-tracking detector differs from previously mentioned detectors in one significant way: it is a passive detector with no real-time detection. However, it offers a cheap and maintenance-free method to monitor the accumulated flux of radiation exposed to the badge-wearer.

Applications by radiation type

X-ray and gamma detectors

X-ray energy varies from sub-keV to approximately 100 keV. For the detection of low-energy (soft) X-rays, CCD and CMOS (Complementary Metal Oxide Semiconductor) image sensors are often used. In research labs, bolometers can be used for their good energy resolution and low threshold. For high-energy (hard) X-rays, the detection is not much different from that for gamma rays and is often done by scintillation and semiconductor detectors. Scintillators coupled with silicon PIN (P-Intrinsic-N type semiconductors) diodes, SiPMs, or PMTs are widely used in security and medical imaging applications. Silicon and germanium detectors are more often used for precise measurements of energies and intensities. For example, large (kg-scale) HPGe is essential in measuring ultralow levels of gamma radioactivity with high efficiency and unprecedented energy resolution.

Charged particle detectors

For alpha particles and ions in the sub-MeV and MeV energy range, surface barrier silicon detectors are often used for energy measurement. A silicon detector telescope, sometimes called a ΔE -E detector, measures the partial energy deposition (ΔE) in the first thin detector and the rest of the energy in the second thick detector. Since the energy distribution depends on the charge (Z) and mass (A) of an incident particle, one can identify the charged particle with the detector telescope.

Neutron detectors

Neutrons are usually detected by absorption or elastic scattering. For example, helium-3 can absorb a low-energy neutron with a large cross section and emit a proton and a tritium. The energy release of 764 keV is carried away by the two daughters. A ^3He proportional counter uses helium gas as the working medium and measures the energy of the daughters (Fig. 12). Similarly, ^6Li and ^{10}B are used for neutron absorption measurements. The isotopes can be loaded in scintillators (solid or liquid) or form gaseous compounds (e.g., $^{10}\text{BF}_3$) as the detector medium. To distinguish neutrons from other types of radiation in scintillators, special scintillators are developed to have different pulse shape responses by the radiation type. Fast neutrons can be slowed down with a moderator and then absorbed in a detector. A Bonner sphere, a lithium iodide scintillator wrapped inside a polyethylene shell moderator, can be sensitive to neutrons of different energy by varying its shell thickness. Fast neutrons can also be detected by elastic scattering, especially on light nuclei, such as hydrogen in organic scintillators.



Fig. 12 A selection of commercial helium-3 neutron detectors from VacuTec. Image courtesy: <https://www.vacutec-gmbh.de/en/news/details/news/neutron-detectors.html>.

Neutrino detectors

Neutrinos are neutral particles with tiny (but nonzero) masses. They interact extremely weakly with the medium; hence, detectors have to be specifically designed and fabricated to measure neutrinos from the Sun, the Earth, power reactors, neutrino beams, etc. The small interaction probability must be mitigated with an extremely large detection medium volume. While neutrinos can be detected by semiconductor and gas detectors, the cost of a gigantic detector is prohibitive. Among the three types of detectors we described earlier, scintillation detectors are the most widely used technology for neutrino detection. For example, the KamLAND detector in Japan, illustrated in Fig. 13, (Suekane et al., 2004) has 1 kt of liquid scintillator with approximately 2000 PMTs. More recently, liquid noble gas detectors, especially liquid argon TPCs, such as the proposed DUNE detector in the US (Abi et al., 2020), provide excellent tracking capability and energy measurement for 3-dimensional neutrino event reconstruction. All of the aforementioned detectors operate deep underground to minimize cosmic muons' impact.

Neutrino detection is also a promising technology for nuclear safeguards, which include monitoring of known and undisclosed nuclear reactors, spent fuel reprocessing, nuclear explosions, etc. Medium and large-scale detectors underground or near surface are often required for potential applications. For a comprehensive review, please refer to Bernstein et al. (2020).

Auxiliary systems for radiation detection

Electronics are an integrated part of a radiation detection system. The electric signal output from the detection medium, either continuous or pulsed, will be fed into electronics, where it may be amplified, shaped, and digitized. Electronics are designed to extract as much information from the detector output as possible without degrading it. One or more stages of amplification are often necessary when the output electric signal from the detector medium is small, such as in semiconductors. On the other hand, PMT and gaseous detectors may have amplified signals sent directly from the detector; hence, additional amplification in electronics is not necessary. The shape of the output pulses may also be reshaped for timing and amplitude measurements. Many modern detectors are connected to computers and have measured results displayed and stored there. Analog-to-digital conversion is therefore necessary. Traditionally, this conversion is performed on key parameters such as the arrival time and amplitude of pulses. With the rapid development of electronics and computers, whole pulses can be digitized and stored for further data analysis in computers. The conversion can be as fast as billions of sampling points per second.

Vacuum systems, high-voltage systems, and cryogenics are other auxiliary parts commonly seen in radiation detectors. For example, to measure alpha emissions from a sample, the sample is placed facing a silicon surface barrier detector. The setup has to be pumped down to good vacuum to avoid alpha particles being absorbed or slowed down before arriving at the detector. Many detectors, including PMTs, gaseous detectors, and HPGe detectors, require DC high voltages, sometimes as high as a few thousand volts, to provide the electric field potential essential to the operation of detectors. HPGe detectors are a good example for cryogenics, which are often cooled to slightly above liquid nitrogen temperature with a dewar.

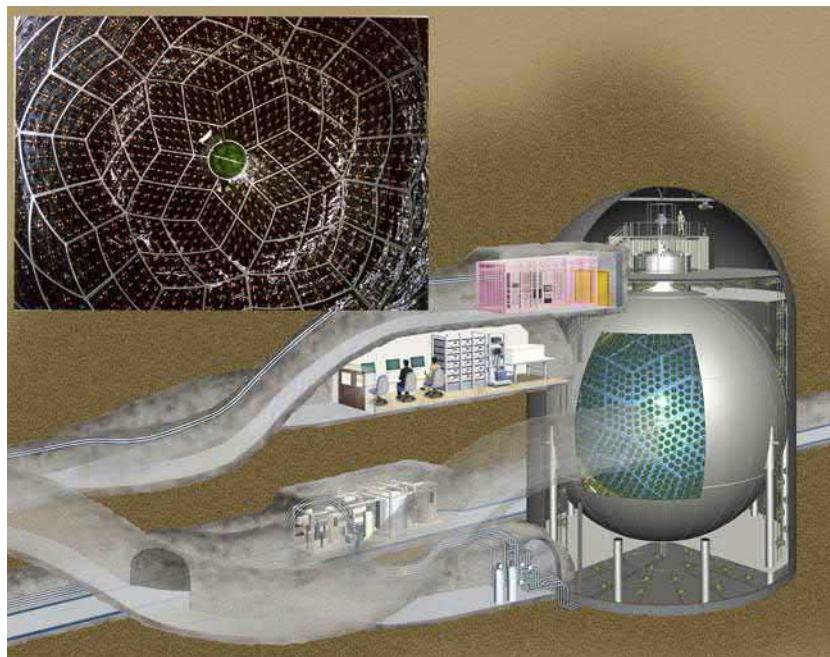


Fig. 13 The KamLAND anti-neutrino detector is a vessel filled with scintillating mineral oil and lined with photomultiplier tubes (inset), buried deep underground near Toyama, Japan. Image courtesy: KamLAND Collaboration.

See Also: External Dosimetry; Instrumentation and Practices for Radiological Assessments; Modern Industry: Gamma Methods for Materials Testing and Inspection; Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection.

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Experimental Tools II: Particle Accelerators

Alejandro Garcia, Department of Physics, University of Washington, Seattle, WA, United States

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Need for accelerators: Applications

Electron and ion accelerators can deliver beams of 10^{13} – 10^{16} particles per second with a very well-defined size ($\leq$ mm), direction ($\approx$ milli-radians), and energy. Although initially conceived to explore subatomic physics, they have become tools used for a wide variety of applications, such as: analysis of radiation damage, important for the aviation industry and neutron reactors, ion implantation for high-density electronics, food and medical instrument sterilization, radioisotope production for medical diagnostics and treatments, accelerator mass spectrometry for highly sensitive radiocarbon dating and geological applications. Accelerators producing proton and carbon beams have been installed at many thousands of hospitals around the world for cancer treatment: taking advantage of the fact that most of the ionization occurs near the end of the trajectory, beams are used to effectively destroy tumors with minimal damage of healthy tissue. Microbeams with sizes of microns or smaller can be used in combination with X-ray detection to produce high-spatial-resolution images of concentration of selected elements in biological, medical, or geological applications. Accelerators are now used as high-flux X-ray sources by undulating positrons and electrons which are used for exploration of condensed matter devices. More examples on applications are given in Bjørnstad (2021).

Ion and electron sources

The first stage in the acceleration process requires extracting the charged particles to produce a beam. For electrons, this is typically done by shining UV light on metals or heating up filaments. For ions of gaseous elements, radio-frequency (RF) discharges are used for ionization. A magnet is used in combination with a filament to generate a plasma, from which the ions are extracted. Fig. 1 shows a typical arrangement. Alternatively, to produce ions from solids, an alkali, like Li, or Cs, is ionized by heating and accelerated toward a cathode made of the wanted element. On collision, the alkali sputters charged ions from the solid. Another very successful type of source, developed in the 1980s, is the Electron Cyclotron Resonance (ECR) source (Lyneis et al., 2010). The basic idea is similar to the gaseous source mentioned above, in that a plasma is made, that allows for ionization. However, the RF works at the electron cyclotron resonance, allowing heating of electrons in the plasma at will. The radius of the ECR source is much larger to allow containing the higher-temperature plasma. The ECR ion source is efficient at producing ions with high charge states, which can be used to increase the acceleration in accelerators. Solids can be sputtered into the source gas allowing production of ions of a wide variety.

The beam properties are characterized by a *beam emittance*, expressed in units of area times solid angle, e.g. mm² steradian. As the beam is shaped with electromagnetic devices it can be focused into waists at certain points at the cost of a larger angular divergence, but conservation laws prevent improving the emittance without losing intensity. In an efficient machine, the emittance is defined at the source, so its design is crucial for efficient delivery.

The challenge in most ion sources is to generate the highest currents in stable fashion while minimizing emittance and maintenance expenses. Electron sources yielding up to several milli-Amperes of current, including the possibility of extracting polarized electrons, have been achieved with GaAs photocathodes. Similar currents can be reached with some ion sources.

Typical power supplies for beam extraction go up to a few tens of kilovolts. The ion or electron sources are placed on an insulated platform that can be brought up to a 100–300 kV potential. For higher energies, several ingenious machines have been invented, of which we will describe a few in what follows.

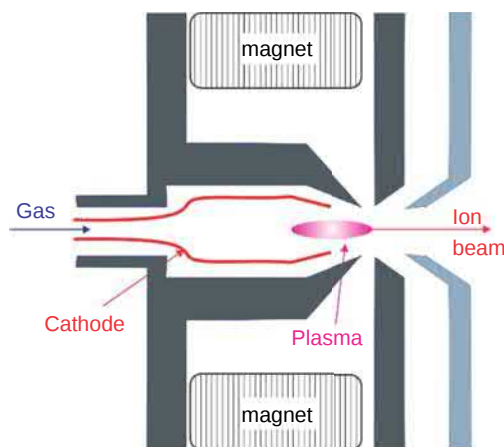


Fig. 1 Sketch of elements of a typical ion source. The combination of gas at appropriate pressures, a magnet, and a heated filament that eject electrons is used to create a plasma that ionizes the gas. Appropriately shaped electrodes help create an efficient emission pattern in a particular direction creating a beam.

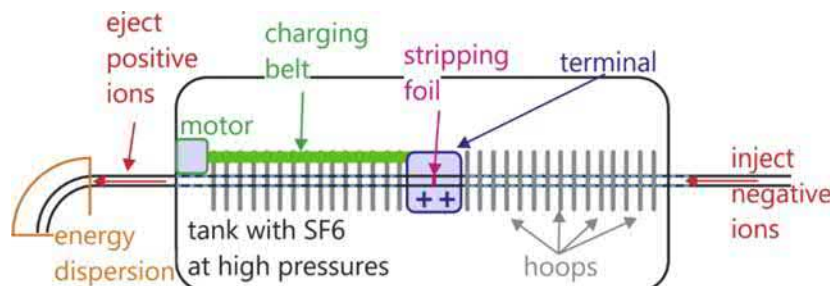


Fig. 2 Sketch of elements of a tandem Van de Graaff accelerator. To prevent sparking, a tank is used as shown to hold the accelerating tube embedded within high-pressure (typically 10 atm) Nitrogen + CO₂, or SF₆. For a 9-MV tandem, for example, the tank diameter and length are about 4 and 30 m, respectively.

Electrostatic accelerators

The first successful accelerator was produced by Cockroft and Walton, who made use of two chains of stacked capacitors. Using switches to move the connecting points between the chains, they were able to generate voltages of approximately 1 MV. Up until 2012, the Fermilab accelerator system made use of Cockroft-Walton accelerators for its initial stages. They were replaced by Radio Frequency Quadrupole cavities, which will be described below.

Van de Graaff accelerators, sketched in Fig. 2, use a belt to move charge from the ground to a high-potential terminal. The basic idea is commonly used in demonstrating units shown in science museums around the world. Van de Graaff accelerators contain a vacuum tube that allows for ions to propagate. This tube consists of a series of insulating and conducting rings, connected by resistors that guarantee a linear variation of the potential along the length of the acceleration tube. Hoops of toroidal shape surrounding the conducting rings, are used to decrease the chance of sparking. The potential can reach up to more than 20 MV. A *tandem* version with a positive high-voltage terminal at the center combined with negative ions is typically used to maximize the energy range of extraction energies. On their way through the vacuum tube, negative ions reach the terminal where they traverse a stripper foil or gas that changes the sign of the ion charge and can multiply it for ions with higher atomic number than hydrogen, thus allowing for a gain in energy at a given terminal voltage. Tandem Van de Graaff accelerators can deliver a wide variety of beams, from hydrogen to gold or heavier, and are very stable and easy to manipulate. In modern versions the charging belt is a special chain called a *pelletron* that improves stability.

Cyclotrons and synchrotrons

The cyclotron, invented by Lawrence, is based on another ingenious idea. Consider a uniform magnetic field, of magnitude B , used to bend ions' orbits into semicircles, as shown in Fig. 3. The ions move inside either of two flat half-cylindrical metallic cavities, called D's because of their shape.

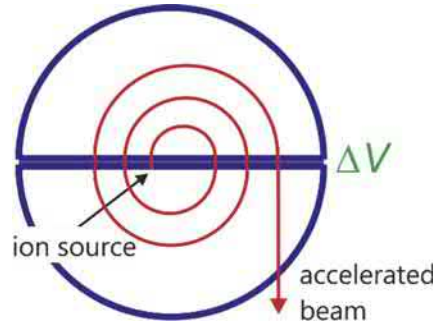


Fig. 3 Sketch of elements of a cyclotron accelerator. A magnetic field (oriented out of the page for negative ions) bends the orbits into semicircles. The potential difference between the two D's is flipped every time the particle is about to cross from one D to the other. Diameters can range from less than a meter to almost 20 m.

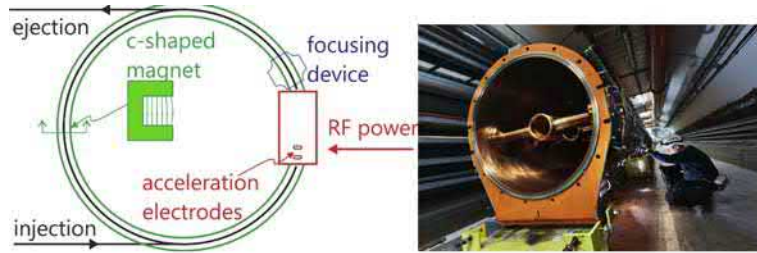


Fig. 4 Left: Sketch of elements of a synchrotron accelerator. Particles are ejected after several turns. What is shown as a single C-shaped magnet, is in practice done with several dipole and other magnets. Right: photo showing some of the RF accelerating cavities for the SPS at CERN. From <https://cds.cern.ch/images/CERN-PHOTO-201902-032-1>

A potential difference between the two D's accelerates the ions every time they move from one D to the other. The frequency of these events is twice the cyclotron frequency, f_c , which can be calculated by equating the magnetic force to the centripetal force:

$$f_c = \frac{qB}{2\pi E/c^2} \quad (1)$$

where q is the ion charge, and $E = K + mc^2$ is the ion's total relativistic energy, with K its kinetic energy and m its mass. In the *non-relativistic limit*, in which $\frac{K}{mc^2} \ll 1$, the cyclotron frequency is independent of the ions' kinetic energy. Thus, by flipping the voltages at a constant frequency, the ions are accelerated while going into larger and larger orbits, as shown in Fig. 3. An additional magnet allows ejection of the ions when they reached the outer radius.

Cyclotrons can have diameters from a fraction of a meter up to close to 20 m. Typically, they can deliver higher intensities and energies than tandem Van de Graaf accelerators. For this reason, they have been very successful at producing radionuclides for pharmaceutical applications, as well as for the study of unstable nuclei. There are more than 25 facilities worldwide dedicated to nuclear physics studies using cyclotrons.

In order to accelerate ions beyond the point where $K \ll mc^2$, the synchrotron was invented. It uses a principle similar to that of the cyclotron, synchronizing the times for the accelerating pulses to correctly account for the relativistic effect indicated in Eq. (1). In practice, the idea is implemented in a fashion sketched in Fig. 4. The ions move in a ring magnet with C-shaped cross section, allowing higher radii than would be possible with a magnet covering the whole circular area. The radius of the circle is fixed, but the magnetic field is ramped appropriately with time, to keep the ions in circular motion.

Synchrotrons are used, for example, in the CERN collider to bring protons to up to energies of 450 GeV, as part of their acceleration complex. The diameter of the 450-GeV Super Proton Synchrotron (SPS) is approximately 7 km. The "C-shaped" magnet, which appears as a single unit in the sketch at left in Fig. 4 is actually composed of 1317 magnets, of which 744 are dipole magnets.

Optical elements

In order to adjust the beam optics for efficient transport through the accelerator, several electromagnetic devices are used. Steering and focusing of the beam is typically achieved with dipole and quadrupole magnetic fields, respectively. Electrostatic elements can be used for energies below 1 MeV, but the necessary fields become too large for higher energies. Fig. 5 shows photographs and sketches the forces on a quadrupole. With the configuration shown in the sketch at right for the quadrupole, a beam of positive

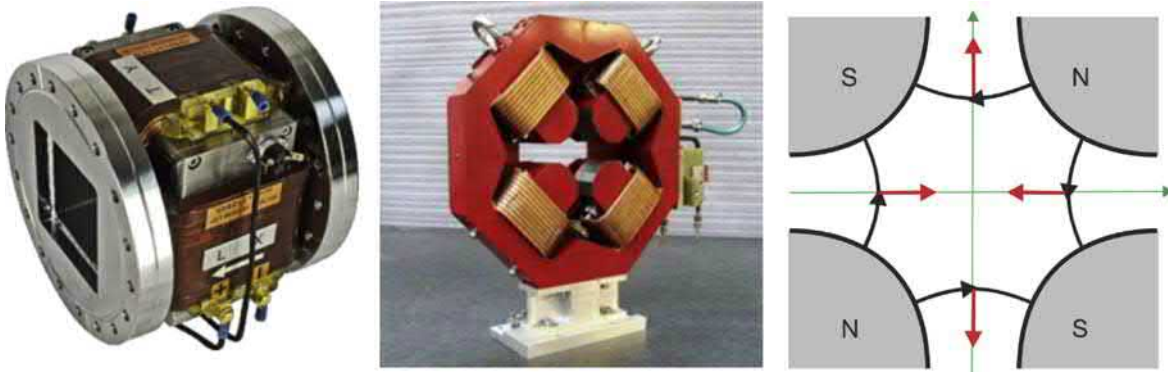


Fig. 5 Left: photo of typical magnetic dipole, used for steering, which produces relatively uniform fields in the vertical (horizontal) direction, that steer the beam in the horizontal (vertical) directions. Center: photo of quadrupole, used for focusing. Right: sketch showing the poles of the magnets and magnetic field lines (in black), and the resulting magnetic forces on positive ions moving into the page (in red).

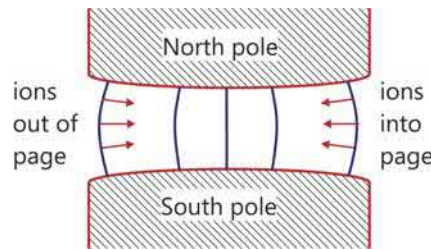


Fig. 6 Sketch of typical magnetic field lines in a cyclotron. The arrows indicate the magnetic force on positive ions. The poles are shaped to produce a vertical restoring force.

ions moving into the page, for example, would get stretched in the vertical direction, and compressed in the horizontal direction. A combination of two or three of these elements is used in sequence with a net result of focusing.

Betatron oscillations and stable orbits

Injected beams have a finite emittance, with the beam momentum having some spread in magnitude and direction. Considering the cyclotron, for example, a completely uniform magnetic field would result in a very low efficiency to accelerate ions. In what follows, one should bear in mind that the field is mainly in the axial (z) direction, with small components in the other directions.

The poles of cyclotrons are actually shaped to yield fields that decrease with increasing radius, as shown in Fig. 6. A quantity called the *field index* is used to characterize this variation:

$$n = - (dB_z/B_z)/(dr/r) \quad (2)$$

where B_z is the field in the axial direction. At a given radius, the field varies roughly as

$$B_z = K/r^n \quad (3)$$

with K a constant. Fields that decrease with r , as those shown in Fig. 6, have positive field index, $n > 0$. Fig. 6 indicates with arrows the direction of the magnetic forces. As can be seen, the arrangement naturally leads to restoring forces in the axial direction. The larger the field index, the larger the restoring force in the axial direction.

A sketch of the magnetic force versus radial position that results from the axial field magnitude variation of Eq. (3) is compared to the force needed to keep ions in a circle (the centripetal force) in Fig. 7. The desired radius for the ion tracks is called r_0 . Consider fields with indexes $n > 1$: an ion that finds its way into an orbit with $r < r_0$ will experience a magnetic force larger than the needed centripetal force and will be dragged toward the center; an ion that finds its way into an orbit with $r > r_0$ will find a magnetic force weaker than needed and will thus move further away. Thus, indexes $n > 1$ lead to unstable orbits. On the other hand, indexes $n < 1$ behave oppositely, leading to forces that tend to restore the radial position. The condition for orbits stable both vertically and horizontally is then $0 < n < 1$. Under these conditions, the ions move oscillating around a closed orbit. These are called *betatron oscillations*. The scheme leading to stable orbits, described above for the cyclotron, is also employed for synchrotrons.

This type of arrangement is called *weak focusing*. The fields in this case are mostly invariant in the main direction of motion, so ions experience small variations of field as they move around. For the higher energy synchrotrons, however, this weak focusing is not enough to prevent large unwanted oscillations of the beam. Fortunately, another mechanism, called *strong focusing*, or *alternating*

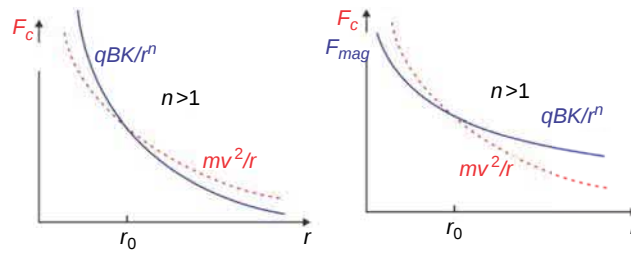


Fig. 7 Sketch of possible variations of the magnetic force (continuous blue) in comparison with the needed centripetal force (dashed red) versus radial position. The radial position marked r_0 is the wanted radius for the ion tracks. Left: field indexes $n > 1$ lead to unstable orbits. Right: field indexes $n < 1$ lead to stable orbits. After Livingood JJ (1961) *Principles of Cyclic Particle Accelerators*. Princeton, NJ: D. Van Nostrand Company, Inc.

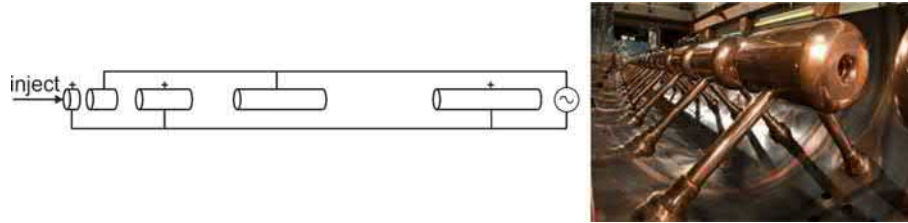


Fig. 8 Left: Sketch of a drift-tube linear accelerator. Charged particles move in bunches inside of the conducting tubes at constant speed, experiencing electric fields when moving in-between them. The length of the tubes varies to guarantee that particles are always accelerated in the same direction. Right: photo of inside cavity of a linac. From <https://www.isis.stfc.ac.uk/Pages/Inside-the-ISIS-linear-particle-accelerator.aspx>

gradients, was discovered that allowed controlling the large oscillations. It turns out that shaping the fields such as to yield alternation between large positive and negative field index values at suitable azimuthal intervals, yields net focusing forces an order of magnitude stronger than with weak focusing.

Practical cyclotrons and synchrotrons make use of these focusing effects, without which practical devices would be very inefficient. A practical cyclotron has several magnets, breaking the azimuthal symmetry suggested in Fig. 6. Likewise, for synchrotrons. The thousands of magnets indicated for the SPS above are partially for this purpose.

Linear accelerators

Linear accelerators (linacs) are based on a very simple idea, sketched in Fig. 8: oscillating electric fields accelerate bunches of ions during part of the oscillation. Conducting tubes prevent the ions from decelerating during the part of the cycle when the fields point in the unwanted direction. The length of the electrodes is designed according to the velocity of the particles.

Although ideas about linear accelerators existed in the 1930s, the technology for sufficiently intense oscillating fields at the needed frequencies (tens of MHz and higher) became available only after the 1940s, benefiting from the radar developments during World War II. A linac for GeV electrons in continuous-wave operation, as example, needs several tens of klystrons (RF high-voltage amplifiers), each delivering several kilowatts of power at ≈ 1 GHz. In practice, rather than the arrangement suggested on the left of Fig. 8, the tubes are placed within a copper cavity and RF of the proper mode is injected to generate the fields in-between the electrodes, as can be seen on the photo on the right of Fig. 8.

Because it is important to keep the particles grouped in bunches, careful considerations to avoid spreading are used. The shielding tubes can be set up such that particles accelerate only during a fraction of the cycle, when the acceleration is positive and increasing. In this way, particles that are late within the bunch are accelerated by larger fields than early ones, contributing to *phase stability*.

Linacs are the most common type of accelerators today. Applications range from electron linacs used for producing high-intensity radiation for industrial and medical applications, and for basic research in subatomic physics, to heavy ion accelerators used to explore properties of the strong force in relativistic heavy ion collisions. Fig. 9 shows a schematic diagram of the Facility for Rare Isotope Beams, planned to start operations in 2022 at Michigan State University. The facility will produce radioactive nuclei via a 400-kW superconducting linac to produce the primary reactions and then extract secondary beams to study nuclei far from stability.

For particles travelling at speeds near the speed of light, linacs can use travelling waves, synchronizing waves of electric field to accelerate the particle bunch, in a way similar as used by surfers at the beach. Interestingly, the phase velocity (e.g., the velocity at which a wave crest travels) for waves in RF guides is larger than the speed of light, so irises are used to slow down the electromagnetic wave phase speed to accompany the particle bunch.

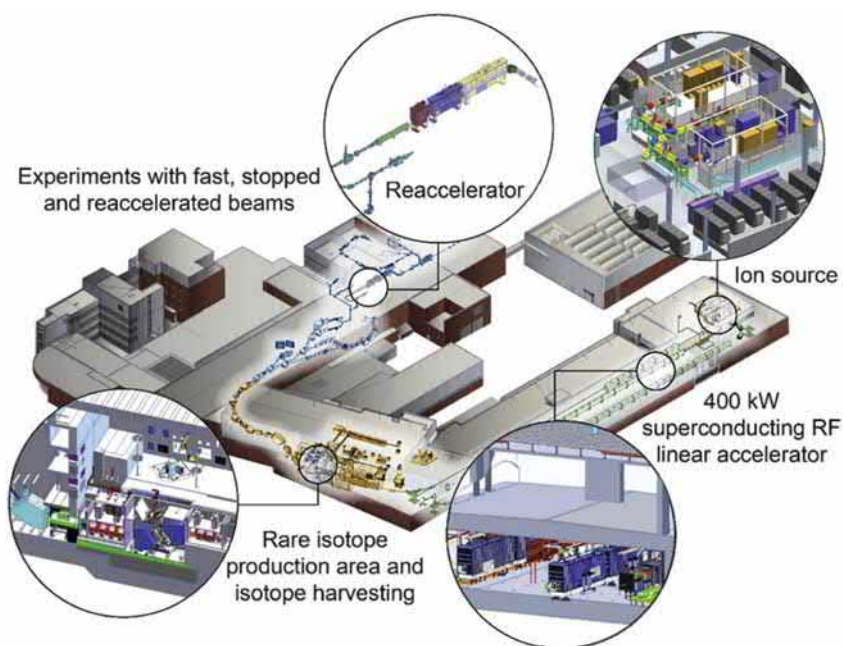


Fig. 9 Example of a linear accelerator used for exploring nuclei far from stability. This is a schematic drawing of the Facility for Rare Isotope Beams (FRIB) at Michigan State University, which will enable scientists to make discoveries about the properties of rare isotopes, nuclear astrophysics, fundamental interactions, and applications for society, including in medicine, homeland security, and industry. Credit: Michigan State University.

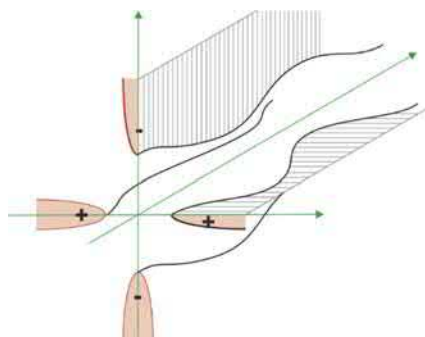


Fig. 10 Sketch of electrodes profile in a RFQ accelerator.

In order to achieve high fields minimizing dissipation, superconducting cavities are used. Many of the modern linacs are superconducting, but this increases the cost significantly. Remarkably, an accelerator under consideration for the future, called CLIC, is being designed to reach electron and positron beams up to 3 TeV, without the use of superconducting cavities.

Linear accelerators of the type described above are not well suited to accelerate ions directly from an ion source. Ions are typically pre-accelerated by *injectors* that bring ions to within $\approx 10\%$ of the speed of light. An alternative type of linear accelerator that has become popular for low-energy applications, or for injectors to the linacs described above, is the Radio Frequency Quadrupole (RFQ) accelerator. At a given time, the electric fields in the RFQ look similar to the magnetic fields shown in Fig. 5. In the RFQ, however, the fields are made to oscillate in time, and the poles profiles vary versus the axial position, as shown in Fig. 10. This modulation in the profile generates net longitudinal accelerations. As indicated above, the initial stages of proton acceleration at Fermilab, for example, have been replaced by RFQs. RFQs are also used for reaccelerating the secondary beams at the FRIB facility shown in Fig. 9.

Colliding beams, the energy and intensity frontiers

The quest for reaching higher beam energies and intensities, driven by the desire to study the frontiers of physics, has been the driving force for the development of accelerators.

As example of the high-energy frontier, the LEP (Large Electron-Positron) collider at CERN was very successful as a tool for physics discovery. What matters for discovery is the collision energy in the center of mass (COM), so two beams are made to collide

at special locations, surrounded by efficient detection systems. Compared to a fixed-target experiment, the energy can be doubled using the same accelerator system, but now the two beams need to be exquisitely controlled such as to yield the appropriate luminosity. The beams are focused as much as possible at the collision points. An important development that allowed the highest luminosities, was the development of *stochastic cooling* of the beams, using information on the transverse velocities to make corrections and improve emittances. The LEP collider achieved energies of up to 209 GeV and luminosity of $10^{32} \text{ cm}^{-2} \text{ s}^{-1}$ before it was shut down in the year 2000. It allowed for high-quality data about the Z and W bosons building great confidence in our theory for how nature works at its fundamental level through many studies. The LHC (Large Hadron Collider) is a machine built for proton-proton collisions at energies of 7–14 TeV in the COM. It has already made an important discovery in finding the Higgs boson while running at 7 TeV, which was announced in 2012. In 2016 the LHC started running at 13 TeV. The LHC collider is part of the accelerator complex at CERN, located underground in a tunnel of approximately 27 km in diameter.

A future electron-positron collider that could reach beyond 250 GeV (energies well below those of the LHC) could allow for searches for new physics with much higher sensitivity than achieved with the LHC. This is so due to the higher backgrounds generated in the proton-proton collisions. Two concepts are being developed, the International Linear Collider (ILC) and the Compact Linear Collider (CLIC). The main challenges for both concepts are yielding reliable and power-efficient acceleration (demanding RF technology developments) and achieving 100 times larger luminosity than LEP (demanding beam optics developments). The Future Circular Collider (FCC) would be similar to the LHC but bigger and reaching approximately 10 times its energy. These machines are expensive and require large international collaborations. However, the resolution of existing mysteries and the possibility of new findings at the physics frontier provide strong motivation.

As example of the intensity frontier, the electron-ion collider, a system of linacs to allow understanding of QCD, our theory of the strong force will be built at Brookhaven National Laboratory (Accardi et al., 2016).

Further reading

- Applications of accelerators: Chapter 11 in Myers and Schopper (2020);
- General references about charged particle beams (Myers and Schopper, 2020; Reiser, 2008; Shiltsev, 2020);
- Betatron oscillations (Livingood, 1961);
- Linacs: Chapter 7 in Myers and Schopper (2020) and (Leeman et al., 2001).

See Also: Agriculture: Electron Beam Irradiation Technology Applications in the Food Industry; Fuel Rod, Fuel Assembly, and Reactivity Control Assembly Design; ILW conditioning and performance; Nuclear Fusion – Introduction and Overview.

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Commercial Nuclear Power Reactors and Their Design—Introduction

Russell E. Stachowski, Global Nuclear Fuel, Wilmington, NC, United States

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The predominant use of nuclear energy today is in the generation of electricity for consumption by the world's electricity markets. While there have been some other uses that could be considered commercial, such as transportation or district heating, these instances have, to date, been extremely limited. This article provides the reader with a perspective on the world's 441 presently capable of operating (as of 2020) nuclear power plants, have the ability to generate some 393 Giga-Watts ($\text{GW} = 10^9 \text{ W}$) of electricity. That's approximately 10% of the world's electricity generation.

A brief historical perspective on the development of the technology that became the commercial nuclear power industry is given in the first chapter, "History of the Development of Nuclear Reactors" (Maldonado and Brown, 2021). This article covers the rapid advancement of the technology that was essentially originally geared towards military propulsion and then harvested to provide terrestrially based electricity generation, starting with the first light bulb lit in Arco, Idaho, and progressing to today's carbon-free gigawatt class power plants.

In the article "Physics of the Fission Chain Reaction in Thermal and Fast Reactors" (Tsvetkov and Stachowski, 2021), an introduction to the underlying physics of the nuclear chain reaction, and its control, is provided. Entire tomes are written that discuss various facets of these principles, so the material presented is structure to be accessible to those who do not have a nuclear engineering background, but are nonetheless familiar with basic physics and mathematical techniques of equation solving. A discussion of nuclear reactions, those resulting in nuclear fission, absorption and scattering, as well as nuclear radiation is provided, in order for coverage of reactor physics principles of moderation and reactivity to be understood. This article further introduces the concepts of neutron flux and its behavior in finite systems, in both thermal and fast neutron spectrum type reactors.

These governing concepts are expanded upon to introduce the basic design principles that underpin the engineering foundation of a nuclear reactor in the article "Principles of Design and Operation of Nuclear Power Reactors" (Hamon, 2021a). While there are numerous styles of reactor, which are based on different thermal-to-electrical energy conversion cycles as well as mechanisms of moderation, steam generation and reactivity (chain-reaction) control, the safety and design requirements are, at a high level, the same among reactors. Over the years, a relatively consistent set of standards, for both design and for licensing have been put into place. These standards cover everything from environmental assessments and grid interface to redundancy and diversity in control and safety systems.

The next six chapters each cover reactor types of different technological bases. The first of these, "Pressurized Water Reactors" (Brown, 2021), cover the reactor type (PWR) that comprises the largest portion of the commercial nuclear fleet. While there are several different styles of these reactors, these all share the principle of being moderated and cooled with non-voided (non-boiling) light water (i.e., H_2O) in a primary loop, with a secondary loop responsible for steam generation to feed the turbine-generator set. The chapter covers key equipment, such as the reactor pressure vessel, steam generators, fuel assemblies, emergency core cooling systems and the control systems, which use both dissolved boron in the coolant as well as a discrete control rod insertion system for shut down capability. The function and major design considerations of each component is discussed and illustrated with schematic type figures.

The next chapter deals with the type of reactor whose steam generation occurs directly in the core. In "Boiling Water Reactors" (Hamon, 2021b), the key systems of the BWR are discussed, including the channeled fuel assemblies, control rod system, steam separators, steam dryers, coolant recirculation system and emergency core cooling systems. The function and major design considerations of key components are discussed and illustrated with schematic type figures.

"Heavy Water Reactors" Nichita (2021) introduces a pressurized reactor, PHWR, which takes advantage of the moderating characteristic of heavy water (i.e., D_2O) that allows these reactors to be fueled with natural (unenriched) or low enriched uranium. These reactors have design that incorporate either pressure-tube or pressure-vessel concepts. The chapter surveys those designs, with emphasis on the CANDU system, which are the most numerous version of PHWRs worldwide. The function and major design considerations of key components, such as the primary and secondary coolant circuits, the calandria (moderator vessel), the pressurized fuel channels, fuel bundles, control systems, on-line refueling and are discussed and illustrated with schematic type figures. Operational aspects for this type of reactor are considered.

A final type of water-cooled reactor, the "Water-cooled Graphite Moderated Reactor" known as RBMK (*Reaktor Bolshoy Moshchnosti Kanalnyy* or "high-power channel-type reactor") was intended to be covered in this section, but the chapter was unavailable for publication. Instead, the reader is directed to the description of the RBMK provided in Sich (2021) and Pioro et al. (2021). Here, the primary moderation for the nuclear chain reaction is performed by a static structure of graphite, while the fuel is cooled by water. While a number of these former Soviet Union reactors were shut down after the Chernobyl event, several are still providing electricity to the grid.

Two chapters on non-water-cooled reactors are provided in Section 3 of this encyclopedia, which is on the topic of Advanced Nuclear Reactor Concepts under R&D. However, the principles involved in sodium fast reactors ([Heidet, 2021](#)) and high temperature gas-cooled reactors ([Fütterer, 2021](#)) are notably similar to those types of reactors in operation today. Each of these has a primary and a secondary circuit with an intermediate heat exchanger serving as the interface between the nuclear heat transport path and a steam supply side. Gas-cooled reactors, with a low density working fluid for coolant, must operate at different temperatures and pressures than their water-cooled counterparts. Key components are also substantially different due to the working fluid, but since the specific power density is also low, safety systems are also different. Power reactors in operation today that use gas (CO₂) as a coolant also leverage a thermal neutron spectrum, and thus use graphite as a moderator.

Liquid metal cooled reactors are a subset of fast spectrum reactors, so have no need for a moderator in order to sustain the nuclear chain reaction. These reactors have relatively high power densities, but due to the high thermal conductivity of the metal coolant, have a high degree of passive safety capability inherent to the system. Due to the use of a metal coolant, however, unique systems are required to work with the coolant, and also to maintain and refuel the reactor, since the coolant is not optically transparent. While very few of these reactors are considered commercial power reactors, the technology is envisioned also for future applications, with a well-known capability for being able to operate on various fuel cycles (uranium, plutonium, actinide, etc.) and be fuel-self-sustaining (breeders).

An article on “Commercial Nuclear Power by Country” ([Hamon, 2021c](#)) is provided to give the reader an overview of the current usage of nuclear energy across the globe. While this information has the potential to change quite frequently, there are references provided that do continuously provide updates to this information.

“Nuclear Fuel and Control Element Design” ([Young, 2021a](#)) provides the reader with the nuclear and mechanical bases for the design of both nuclear fuel elements used in light water reactors and also the control rods, which provide the primary means for shutdown of PWR and BWR reactors. Mechanical and thermal-mechanical performance is analyzed by the complex interaction of the fuel material (fuel pellets) and the cladding, as a function of load condition and irradiation condition. Specified Acceptable Fuel Design Limits are derived from the performance limits on the assemblies in order to stay within their design criteria.

The article on the design of nuclear reactor cores was unable to be published in this chapter. This topic involves how the Specified Acceptable Fuel Design Limits (SAFDLs) are employed, along with economic objectives, to design the core arrangement of fuel and, when applicable, the control assemblies. This core arrangement, called a loading pattern, is the dominant method of designing a reactor core to achieve energy production targets, subjected to both reactivity limits for controllability of the core, as well as “thermal limits” derived from the applicable SAFDLs. The reader is directed to [International Atomic Energy Agency \(2019\)](#) for a general description of the design, analysis and safety principles of today’s commercial reactor cores.

Each reactor type employs a strategy for its economical and safe operation, consistent with the underlying technology. In “Reactor Operations” ([Young, 2021b](#)), the operation of PWRs and BWRs is discussed. The chapter describes operations from cold shutdown after refueling, to full power operation. This includes the approach to full power operation, the Technical Specification Requirements, surveillance activities during normal operation, operational strategies, off-normal response and reactor shutdown.

Given the multiple physics disciplines interlaced in a nuclear reactor system of engineering principles involved in the function of nuclear reactors, rather unique approaches are required for the solution of the system of equations governing reactor operation. The topic is discussed in the article “Operational Analysis Methods” ([Walters, 2021](#)). This article on steady-state reactor analysis covers the neutronics calculations at the lattice level to generate effective cross-sections (probability for specific neutron induced nuclear reactions) and how these cross sections and other nuclear parameters are transposed to 3-dimensional core simulation and coupled to integral thermal hydraulic analyses for application to core design and safety analyses.

The next chapter, “Transient Analysis Methods” ([Martin, 2021](#)), describes how thermal hydraulic time dependency and neutron kinetics is added to the reactor physics, in order to analyze the dynamics of safety performance during abnormal as well as design basis accident events. The industry standards and regulatory requirements that govern necessary attributes of the software, as well as the rigors required in establishing an evaluation model and appropriate uncertainties based on qualification demonstrations of the model are discussed. Both deterministic and stochastic techniques are introduced.

A separate chapter on the reactor containment building is provided, given the importance of this structure in assuring that radioactive release to the environment, in the event of an accident, is minimized. “Containment” ([Sharpe, 2021](#)) discusses the design of the structure, the technologies employed for effluent containment and the mitigating strategies used for reducing pressure and dynamic forces that can be created during severe accidents.

Because of the intimate coupling between the means of power production and transport and the safety principles and strategies of ensuring safety, substantial attention is paid to heat transfer systems in Nuclear Power Plants (NPPs). Adequate heat removal from the nuclear fuel by the heat transfer systems, both for full power operation and post-shutdown residual heat removal, prevents fuel failure and subsequent release of radioactivity. The article “Heat transfer systems” ([Kirkland, 2021](#)) explores many of the main heat transfer systems found in NPPs and provides a fundamental understanding of the physics and technology used in their design and operation.

The instrumentation and control (I&C) systems in a NPP, along with their enabling features, serve as the “central nervous system” of a NPP. In “Control and Instrumentation Systems” ([Upadhyaya and Wood, 2021](#)) the reader is introduced to the I&C architecture, comprising plant control, protection, and monitoring systems that provide information to various actuators and devices to ensure the safe and efficient operation of the plant, and to the operation and maintenance personnel for making timely decisions. This article focuses on the I&C systems used currently in light water reactor plants (PWRs and BWRs).

Systems, components, and structures that comprise a complete NPP and are not included in the nuclear steam supply systems as defined under 10 CFR 170.3 Definitions fall into the category of balance of plant systems. These systems include Fuel Storage and Handling Systems, Fuel and Auxiliary Pools Cooling System, Reactor Component Cooling Water System, Plant Service Water System, Drywell Cooling System, Control room habitability, Heating, Ventilation and Air Conditioning systems. A description of the design and function of these and some auxiliary systems is provided in “Design of the Balance of Plant” (Arcaro, 2021).

“Economics of Nuclear Energy” Rothwell (2021a) is a article intended to provide the reader with an understanding of the mathematics to correctly assess the costs involved in the construction, operation and decommissioning of NPPs, and how to assess the levelized cost of electricity based on capital intensive projects. The method of leveling the cost of electricity is key in making determination of competitiveness for a given technology, deployment scenario and fuel cycle.

A subsequent chapter, “Economic Challenges of Nuclear Power” (Rothwell, 2021b), explores the challenges of operating a nuclear power plant in markets with other electricity generation. These include regulated and deregulated electricity markets, as well as varying degrees of renewable energy generation priority dispatching, which can lead to grid pricing behaviors not conducive to capital intensive power generation technologies. Additional factors affecting the economic competitiveness of nuclear include political climate and the manner in which greenhouse gas emissions is being addressed in the various jurisdictions.

While the design requirements of nuclear reactors are almost synonymous with the safety criteria, a separate chapter on the evolution of these safety principles is provided in “Safety of Nuclear Reactors” (Brown and Seo, 2021). Reactor safety objectives are provided for anticipated operational occurrences, design basis accidents and severe accidents (beyond design basis), which forms the historical basis for licensing, particularly in the US. Current trends in regulatory framework, towards a more risk-informed approach, are also discussed. While Section 4 of this encyclopedia, “Safety, Licensing and “Decommissioning of Power Reactors” is entirely devoted to these topics, this article provides an overview focused solely on the relevance of safety regulation to commercial nuclear power.

“Pros and Cons of Commercial Power Reactor Designs” Parts 1 and 2 Piro and Duffey (2021a,b) compares the generation of electricity from nuclear power against other forms of energy generation used commercially. Since all forms of power generation must exist within the context of an electrical grid of some kind, the relative merits of the technology for energy production tends to be a function of the country in which it is deployed. These comparisons are described in these chapters through the examination of thermal efficiency and capacity factors, both of which are major contributors to the relative competitiveness between power plants measured by levelized cost of electricity, as discussed in the above chapters on economics. In addition, each nuclear reactor type has a certain set of advantages and disadvantages that influence the degree and trends in technology adoption, with local preferences having influenced the choices between reactor types of relatively similar capabilities of the current and past generation technologies. Future trends in the development of next generation reactors, with their higher thermal efficiencies owing to the higher working fluid temperatures, substantially close the gap with fossil fuel plants, while increasing safety compared to the prior generation nuclear technologies. The cost competitiveness of combined-cycle natural gas turbine power plants, driven by low commodity fuel costs and high thermal efficiency, is off-set by the key advantages of nuclear energy, which include cheap, reliable, high capacity factor base-load power, low risk to population per unit energy generated (in spite of perception from high profile accidents), low carbon emissions, and minor environmental impact.

The final chapter in this section provides the reader with an overview of the science, economics and trade-offs of the back end of the fuel cycle. “Management of Nuclear Waste – via Disposition or Transmutation” (Tsvetkov, 2021) describes the basics of repository characteristics and performance, introduces the concepts of safeguards and non-proliferation, and compares the options of disposition versus transmutation in order to reduce the hazards potential for the commercial nuclear waste stream.

It is hoped that the reader, having been given a brief overview of most of the major aspects of commercial nuclear power, will be equipped with enough knowledge to delve deeper into the relevant topical areas.

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History of PWR and BWR Development

Jeffery A. Brown^a and G. Ivan Maldonado^{b, a} Westinghouse Electric Company, Pittsburgh, PA, United States; and ^b Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

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Glossary

AEC Atomic Energy Commission—The US agency in charge of nuclear energy.

AP1000 Advanced Plant 1000 MWe—a Westinghouse passive type Generation 3 PWR plant.

APR1400 Advanced Pressurized Reactor 1400 MWe—a South Korean (KHNP) PWR system.

BORAX The first series of boiling water reactor experiments at INL performed from 1953 through 1958.

Breeder A reactor that produces more nuclear fuel than it consumes. A fertile material, such as uranium-238, when bombarded by neutrons, is transformed into a fissile material, such as plutonium-239, which can be used as fuel.

Burnup A term for the specific thermal energy generated in the fuel material per unit mass of initial heavy metal (e.g., U).

BWR Boiling water reactor—A single coolant loop reactor type in which steam is generated directly inside the reactor vessel.

CP-1 Chicago pile 1—The first nuclear reactor. A graphite moderated natural uranium pile reactor.

CP-2 Chicago pile 2—The second nuclear reactor.

Criticality The normal operating condition of a reactor, in which a fission chain reaction is sustained at a given average fission rate. A reactor achieves criticality (and is said to be critical) when each fission event releases a sufficient number of neutrons to induce, on average, another fission event.

CVCS Chemical volume control system—A system that controls the primary.

DNB Departure from nucleate boiling—A PWR fuel clad design limit which represents the onset of film boiling resulting in a significant loss of heat transfer and increase in fuel clad temperature.

EFPD Effective full power days. A measure of the amount of energy generated if the reactor had run at full power for this number of days.

EFWS Emergency feed water system—An auxiliary system used to supply secondary feedwater to the steam generators when the normal feedwater supply is not available.

EPR Evolutionary pressurized reactor—A Framatome PWR system.

ESBWR The general electric economic simplified BWR reactor system.

Flux Flux is the neutron current in terms of neutrons/cm²-second.

LOCA Loss of coolant accident—An accident involving loss of primary cooling water to the reactor core.

Mark I The first PWR, a prototype of the naval reactor installed in the Nautilus submarine.

MTR Materials test reactor—A high flux water moderated reactor using high enriched uranium fuel located at the Idaho test lab.

Nautilus The first nuclear power submarine powered by a PWR.

Once-through SG A type of steam generator for PWRs in which the primary coolant enters at the top of the generator and is passed downward straight tubes to transfer heat to the secondary coolant.

Pressurizer The component of the RCS that maintains primary system pressure during normal operation.

RCP Reactor coolant pump—Use in PWRs to maintain flow of the primary coolant through the reactor vessel.

RCS Reactor coolant system—The primary coolant circuit of a PWR that provides pressurized coolant to the reactor core.

Reactivity A term expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

U-bend SG A type of steam generator for PWRs in which the primary coolant is passed through horizontal U-bend tubes to transfer heat to the secondary coolant.

U-tube SG A type of steam generator for PWRs in which the primary coolant is passed through vertical U-bend tubes to transfer heat to the secondary coolant.

VVER A Russian type Generation 2+ PWR using hexagonal fuel assemblies.

X-10 The Large Graphite Test reactor at Oak Ridge.

Abbreviations

ABWR Advanced boiling water reactor

B&W Babcock & Wilcox

CE Combustion engineering

EBWR Experimental boiling water reactor

ECCS Emergency core cooling system

GE General Electric Co

PWR Pressurized water reactor

USNRC United States Nuclear Regulatory Commission

VBWR Vallecitos boiling water reactor

Introduction

Although the number of operational plants varies over time, according to the 2020 data from the [IAEA \(2020\)](#), 302 of the 442 operational nuclear power reactors in the world are based on the Pressurized Water Reactor (PWR) design ([Brown, 2021](#)). Moreover, 59 of the currently planned 75 power reactors are PWRs, with roughly 43 PWRs already under construction. The reason for this has been the proven reliability and relative simplicity of design and operation of this reactor design. And also due to the efforts of one man, Admiral Hyman G. Rickover. Although there are fewer BWRs ([Hamon, 2021](#)) operational in 2020, about 63, they constitute the second most popular design, thus worthy of discussion. There are currently four new BWRs under construction.

Early reactor development

The first reactor to achieve a self-sustained criticality was the CP-1 located at the University of Chicago which occurred on December 2, 1942. It was conceived and designed by Enrico Fermi and Leo Szilard. The core consisted of a $7.3 \times 7.3 \times 5.8$ m ($24 \times 24 \times 19$ ft) high cube containing 42.2 metric t (46.5 t) of Uranium in the form of alternating natural enriched Uranium metal and graphite blocks within a wooden frame. It was limited to a maximum power 200 W since there was no shielding or cooling. The subsequent experiments with this reactor allowed Fermi and his team to confirm the basic parameters of graphite reactor physics and kinetics and most importantly demonstrated that a nuclear reactor could be controlled and operated safely provided that reactivity was added in small increments.

CP-1 was operated for 3 months and then dismantled and moved to a site outside of Chicago in what is now known as Argonne National Lab. The reconstructed reactor, now known as CP-2 was slightly larger ($9.8 \text{ m} \times 9.8 \text{ m} \times 6.4 \text{ m}$ / $32 \text{ ft} \times 32 \text{ ft} \times 21 \text{ ft}$) with 47.2 metric t (52 t) of natural Uranium metal and included shielding that allowed the reactor to be operated at a maximum power of 10 kW. This reactor also had provisions for inserting materials for nuclear reactor testing. Initial criticality was achieved on March 20, 1943.

The experiments and tests performed on CP-2 led to the construction of the X-10 Large Graphite Test reactor at the Clinton Laboratory at Oak Ridge, Tennessee. The reactor consisted of a 7.3 m (24 ft) graphite cube surrounded by a 2.1 m (7 ft) thick shield of high-density concrete. The reactor core also contained several channels through which air was forced as the coolant. The X-10 achieved initial criticality on November 4, 1943 and was operated at a power of 1 MW, which was eventually increased to 4 MW.

The X-10 also had a large test stand on top of the reactor and was used for testing of materials for radiation shields and for measuring nuclear properties including the multiplication factor of fuel arrays. This assembly was used to investigate the multiplication factor of various arrays of natural Uranium rods in light water. Prior experiments with fuel arrays in light water separated by the relative wide spacing typical of graphite or heavy water critical assemblies had indicated a very low neutron multiplication factor. The X-10 experiments which contained unenriched Uranium rods in light water arranged in tight lattices revealed a multiplication factor close to critical (> 0.95), suggesting that it would be possible to achieve criticality in a similar light water lattice using slightly enriched Uranium. This led Dr. Alvin Weinberg, then director of Clinton, to suggest in 1944 that a power producing reactor could be achieved with Uranium rods in a pressurized water system in which the water acted as both moderator and coolant. High pressure would assure that the water would not boil and the core would remain fully moderated at all times. Weinberg stressed

the inherent safety of the design (as loss of coolant would cause loss of criticality and subsequent reactor shutdown) and that such a system could be based on existing technology. Weinberg and his team subsequently received a patent for the PWR reactor concept.

The X-10 light water experiments led to the design of the high flux Materials Test Reactor (MTR) in Idaho which was designed to operate at a power of 40 MWth and test the effects of high neutron and gamma flux on reactor materials. The reactor core of the MTR consisted of a $0.41 \times 0.71 \times 0.61$ m ($16 \times 28 \times 24$ in.) lattice of tightly spaced high enriched Uranium fuel plates in light water surrounded by a beryllium reflector. This configuration was originally suggested by Eugene Wigner in order to maximize the flux in the core. The fuel elements were 93% enriched uranium alloyed with aluminum, clad in aluminum, and formed into curved plates approximately 61 cm (24 in) long, 7.6 cm (3 in) wide and 0.16 cm (1/16 in) thick similar to the Shippingport seed assembly shown on Fig. 7A. The fuel plates were brazed into aluminum side plates to form a $7.6 \text{ cm} \times 7.6 \text{ cm}$ (3 in $\times$ 3 in) boxlike assembly. Control rods were inserted in spaces between the small assemblies to control and shutdown the reactor. Although the fuel plate configuration was less neutron efficient than arrays using fuel rods, the significantly improved heat transfer allowed for a smaller core and thus higher power density and neutron flux. The core was cooled by subcooled water at a flow rate $566 \text{ m}^3\text{-min}^{-1}$ (20,000 gpm). The MTR achieved criticality in 1952.

Investigation of various power reactor concepts continued at Oak Ridge, Argonne, and elsewhere. Based on these studies, the AEC in 1953 selected five out of some 80 different reactor concepts for further development based on their readiness for construction and on their potential economic competitiveness for commercial power production. The results in order of perceived long term commercial competitiveness were:

1. Homogeneous reactor
2. Fast Breeder reactor
3. Boiling Water reactor
4. Sodium cooled reactor
5. Pressurized Water reactor

Although the PWR was considered to require the least development time, it was rated lower based on estimates of long-term fuel cost since Uranium was considered an extremely rare resource (We now know that Uranium is far more plentiful and easily obtained than previously thought.).

Navy reactor development

The U.S. Navy recognized early on the benefits of using nuclear power as the propulsion plant for submarines and surface ships. The Navy requirements for a nuclear power plant were different than those for a power station that Oak Ridge and Argonne had been considering. Specifically, the Navy required that the reactor power plant be small enough to fit into a submarine and have low enough radiation release during normal operation such that the exposure to the crew during a prolonged voyage would be acceptable. Economics and efficient use of Uranium were not concerns.

In 1946, the Navy sent a small team of personnel with technical backgrounds to Oak Ridge for several months to learn about reactor development and to ultimately recommend the steps forward for development of nuclear power for Navy application. The leader of this team was Captain Hyman G. Rickover.

Rickover had a strong technical background (an Annapolis graduate in 1922, master's degree in electrical engineering at Columbia University in 1929) and was qualified to command submarines. Rickover had distinguished himself by his performance as head of the Navy's electrical section during WWII. Rickover had insisted on having full engineering design capabilities under his direct command so that he could make informed decisions on major technical issues. Rickover personally selected the contractors and worked directly with the responsible official in each company. He and his staff worked directly with the contractors in designing the equipment, and once the plans and specifications were established, he insisted that the manufacturers follow them to the letter. In the process, Rickover established close working relationships with major electrical equipment contractors such as General Electric and Westinghouse and earned the reputation of being a tough-minded, exacting, but reliable customer. Rickover's intuitive practical engineering sense, his tenacity, and his refusal to compromise on technical excellence, resulted in a huge improvement in ship electrical system reliability during World War II. These characteristics made him the right man to lead the Navy nuclear program.

During Rickover's time at Oak Ridge, Weinberg had discussed the PWR reactor concept with him. Weinberg stressed the advantages of the PWR concept and that a working prototype small enough to fit into a submarine could be designed and built relatively quickly using existing technology. Rickover encouraged Weinberg to continue engineering studies of the PWR concept for submarine application. Rickover believed that the problems required to design and build a nuclear power plant suitable for navy application were engineering problems not scientific ones that could best be solved by properly managed industry experts.

In April 1948 the Atomic Energy Commission (AEC), in response to mounting concerns over Soviet submarine advances, decided to initiate a formal high-priority project to design and build a nuclear power plant capable of powering a submarine. Rickover was assigned to oversee and manage the program. Argonne, to which the AEC had earlier transferred responsibility for reactor development was to lead the technical design.

Two concepts perceived as being capable of meeting the compact size requirement in the near term were selected for development as a submarine power plant: the liquid sodium cooled reactor and the PWR. The Navy had already commissioned Westinghouse to perform studies related to heat transfer systems with pressurized water and General Electric to perform similar studies with liquid

sodium, so these vendors were the natural choice to be the industry leads. Although it was recognized that substantial technology development was needed for the sodium cooled reactor, both Argonne and General Electric had already been performing studies to develop a large nuclear power plant using a fast sodium cooled breeder reactor. For the PWR little fundamental technology development was necessary, and as a consequence, the 1948 Navy assigned Westinghouse to perform the full engineering design of a prototype submarine PWR plant (Mark 1). Westinghouse would be required to “do all detailed engineering, produce the working drawings, procure the necessary materials, and construct the Mark I plant,” which would be a land-based prototype. Argonne would provide design and engineering data and, as the Commission’s agent, would approve the final working drawings prepared by Westinghouse. Rickover would be responsible to assure success of the overall project in meeting the Navy’s requirements.

The AEC procured the Bettis site near Pittsburgh and set up lab and engineering offices for the sole purpose of performing the research and engineering necessary to produce the prototype Mark 1 PWR. Per Rickover’s demand, Westinghouse assigned many of its top engineers to the site and hired top experts in metallurgy, nuclear physics, and other disciplines. Westinghouse subcontracted major components fabrication work (reactor vessel and internals, steam generators) to other firms (such as B&W).

Although most of the technology required for the Mark 1 was not new, there were several major design changes necessary to accommodate a nuclear power system within the confines of a submarine. Most marine boilers at that time consisted of assemblies of tubes carrying water. Surrounding the tubes were extremely hot gases derived from the combustion of fuel. Heat from the gases flowed across the walls of the boiler tubes and flashed the water to steam. Nuclear propulsion required radical changes in boiler design because the heat source was subcooled water rather than hot combustion gas. Flowing through the boiler tubes, the water would transfer its heat, through the tube, where it would convert the secondary water into steam.

Also, although steam propulsion systems in naval surface ships had been commonplace for almost a century, the Navy had always found steam impracticable for a submarine. One of the complicated problems Rickover and his team faced was how to arrange the steam generators, piping, turbines, and condenser within the limited confines of a submarine hull. Even if this could be accomplished, there was the important requirement of providing enough radiation shielding and air conditioning to keep radiation exposure and temperatures in the steam machinery areas down to habitable levels. For this reason, Rickover insisted on having the Mark 1 system be installed in an actual submarine hull section complete with turbine driven propeller shaft submersed in pool at the Idaho test site (See Fig. 1). Fig. 2 shows a schematic layout of the PWR submarine nuclear propulsion system.

Although many details of the Mark 1 core are still classified, it can reasonably be inferred from known details of the MTR and Shippingport designs that the reactor power rating was ~60–70 MWth delivering about 11.2 MW (15,000 hp) to the propeller shaft and that the core was a squat cylinder consisting of fuel elements essentially the same as those used in the MTR core (i.e., thin plates of high enriched Uranium alloy sandwiched within metal clad). Rickover decided to use Zirconium instead of aluminum for the clad and alloy material as he believed that it had better long term corrosion properties even though an economical process for obtaining pure Zirconium metal free from the neutron absorbing Hafnium had not yet been developed.

In order to meet the space constraints, the compact Mark 1 PWR system included a single B&W recirculating steam generator comprised of a straight tube and shell heat exchanger with riser and downcomer pipes connected to a separate steam drum similar to that used in Shippingport (see Fig. 3). The reactor coolant pumps employed a “canned” design in which the rotor and stators were completely sealed inside the pressure boundary and the induction motor outside. These canned pumps were chosen in order to eliminate the problem of high pressure primary coolant leakage through the pump shaft seals. These pumps had the added advantage of being much quieter than a normal rotary pump, a distinct advantage for a navy submarine.

The Mark 1 PWR went critical on March 30, 1953. For the next 2 months testing went on day and night. Rickover had insisted on a 100 h run at full power that would simulate a voyage from North America to Ireland and thoroughly test all components of the power train. Even though some problems occurred with the non-nuclear equipment during the simulated run, Rickover refused to terminate the test, but instead used this as an opportunity to see if such problems could be fixed or mitigated by the submarine crew.

Construction of the *USS Nautilus* began in 1952 at the Electric Boat shipyard in Groton, Connecticut. A nearly identical copy of the Mark I PWR (Mark II) was built and installed into the submarine. The Mark II PWR achieved criticality on December 30, 1954. In January 1955, the *Nautilus* began her first sea trials during which the nuclear propulsion system functioned flawlessly. The success of the Navy PWR nuclear propulsion system was firmly established in dramatic fashion during the summer of 1958 when the *Nautilus* made the first under ice transpolar voyage on a cruise from Pearl Harbor to Portland, England.

A sodium cooled reactor plant, developed by Argonne and GE was installed in the submarine *Seawolf* that was launched in June 1956. However, persistent problems with sodium coolant leaks led the Navy to eventually replace the sodium plant with a *Nautilus* type PWR plant. Thereafter the PWR plant became the standard propulsion plant for US submarines.

BWR early history

The earliest concept of using nuclear as a power source involved simply using an atomic pile to directly produce steam to drive an external turbine. However, there was great concern whether the rapid local changes in moderator and coolant density associated with bulk boiling might result in fuel rod failure or even worse in a rapid catastrophic power excursion. Unlike in the relatively simple case of the PWR, there were no tools able to predict the much more complex local nuclear thermal hydraulic coupling.

However, in 1950 Samuel Untermyer II of Argonne performed calculations that predicted that allowing the moderator to boil in the reactor core would actually make the reactor more stable to power excursions and easier to control. Untermyer’s calculations showed that the reactor could more easily follow load demand without the need for complicated control mechanisms. However,

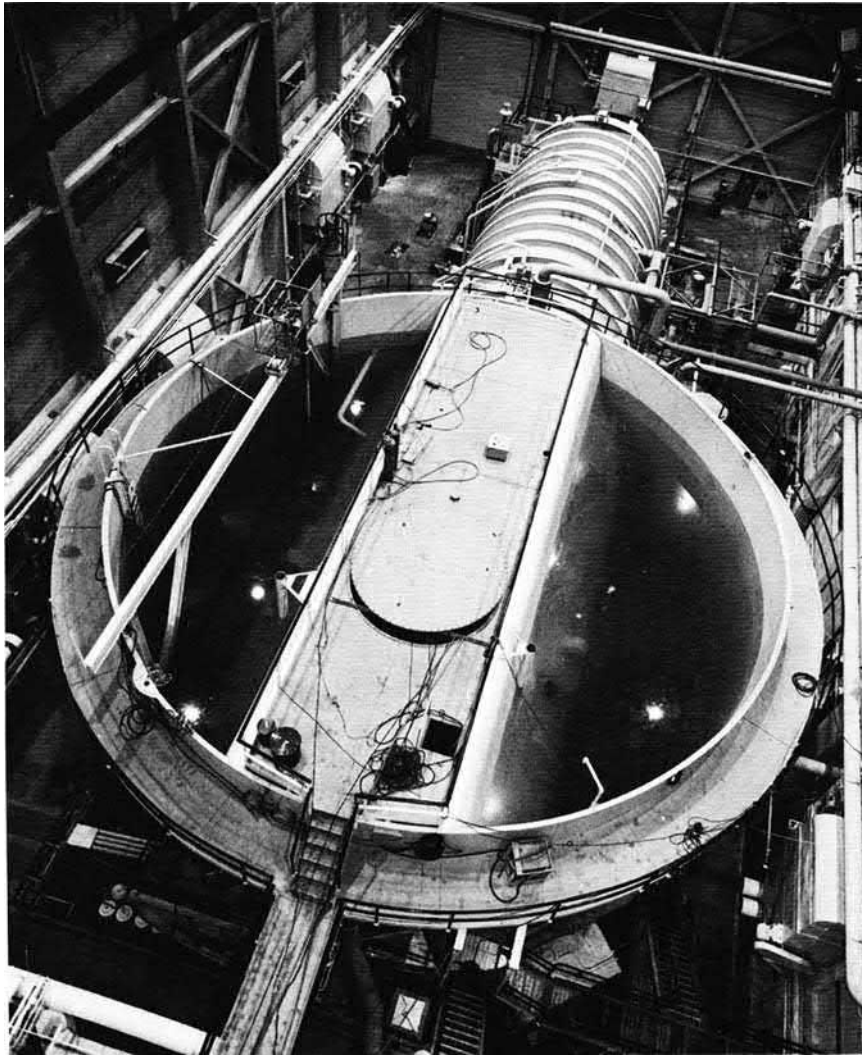


Fig. 1 Mark 1 nautilus prototype PWR. Roddis LH and Simpson JW (1954) The nuclear propulsion plant of the USS nautilus SSN-571. In: Paper presented at the Annual Meeting of the Society of Naval Architects and Marine Engineers, New York, NY, November 10–13, 1954.

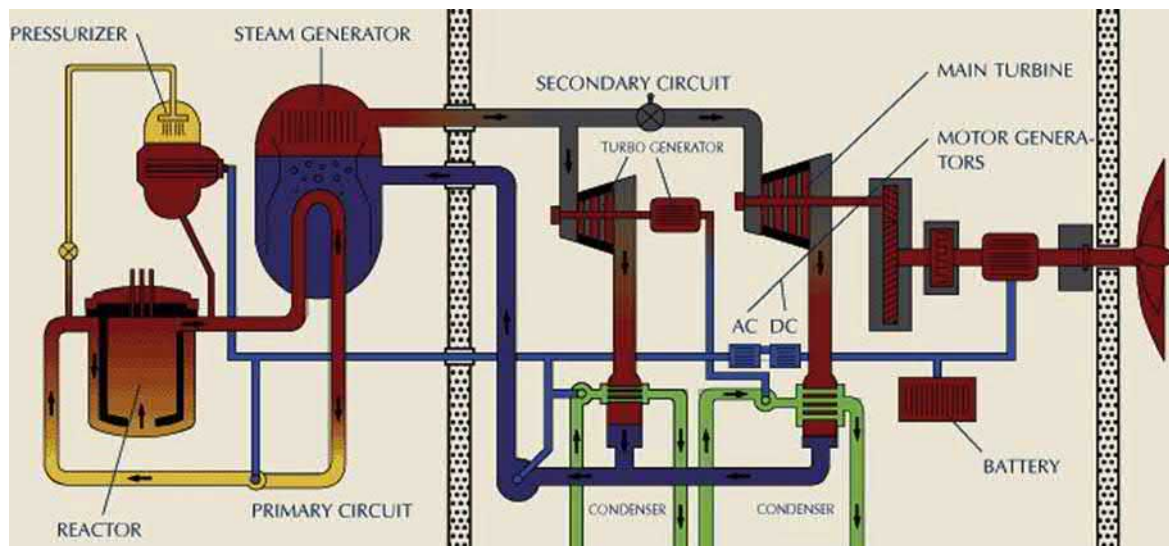


Fig. 2 Nuclear submarine propulsion system.

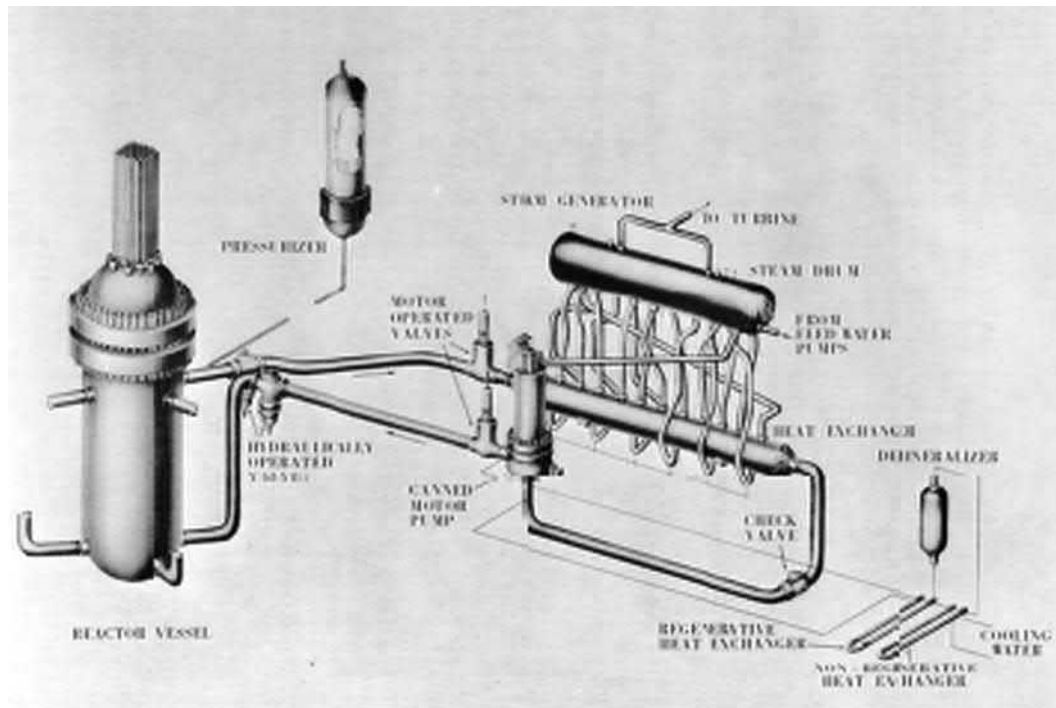


Fig. 3 Shippingport reactor system description. Bettis (1957) *Description of the Shippingport Atomic Power Station*, WAPD-PWR-970.

given the state of the analytical tools, tests were needed to confirm Untermeyer's predictions. Untermeyer convinced his boss, Walter Zinn, of the advantages of the boiling-water-reactor concept, and they were successful in getting the AEC to authorize small-scale tests at the Idaho National Reactor Test Site. The test reactor (BORAX-I), assembled from components scavenged from other test was installed on the ground, without any cover with the control room in a small trailer parked half a mile away.

From May 1953 for the next 14 months, Untermeyer and his team would inflict this collection of scrounged parts to every transient imaginable. Over 200 transients and power excursions were investigated with the last one being a prompt critical event that melted the reactor core but did not destroy the reactor vessel. These tests proved that the Boiling Reactor would be controllable and even in the event of a worst-case transient could be designed to limit release of radioactivity to the environment.

Over the next several years further BWR test reactors (BORAX-II, BORAX-III, BORAX-IV, and finally BORAX-V, a full power reactor prototype), were built and tested that confirmed Untermeyer's prediction of an inherently safe, self-controlling boiling water reactor (In fact, the BORAX-III reactor has the distinction of being the first nuclear power plant to fully support a public electric grid in that it supplied electricity for the entire town of Arco, ID for 1 h on July 17, 1955.).

These efforts were followed by two key developmental plants, the Experimental BWR and the Vallecitos BWR, which provided important and timely operational information for the ultimately development of the Dresden I BWR (Fennern, 2018), described in more detail in the subsection below labeled "Commercial Power Plants."

Shippingport

Until 1953, all nuclear reactor research had been classified, as the ultimate purpose had been mostly defense related. However, in 1953 President Eisenhower announced that the US would declassify and release the technology necessary for civilian and international peaceful application of nuclear power in his famous Atomic Power for Peace speech to the United Nations. With that, he opened the door for commercial application of nuclear power.

In March 1953, the U.S. National Security Council canceled plans for a large land-based prototype PWR reactor for an aircraft carrier. The AEC asked the President and Congress to redirect the funds previously approved for the project to be used for construction of a prototype civilian nuclear power plant. The AEC would build and own the nuclear reactor and would provide trained reactor operators to a private utility company which would own and operate the turbine-generator and balance of plant. Congress approved and the civilian power project based on the aircraft carrier reactor was assigned to Admiral Rickover. Rickover chose to work with Westinghouse and Bettis for the final design and fabrication.

In July 1953, the AEC established the general specifications for the prototype nuclear power plant. It should generate at least 60 MWe using slightly enriched uranium fuel and the coolant demineralized ordinary water. The life of the fuel elements was to be as long as possible and refueling was to take place with a minimum shutdown period. The AEC believed that target power rating

of 60 MWe was a reasonable extrapolation of existing technology. Going higher would mean larger components, higher costs, and would provide little additional data than provided by a 60 MWe plant.

Admiral Rickover and his team recognized that the project had two purposes: to show that a central station nuclear power plant could function on a utility network, and to provide technical information to advance civilian power reactor technology. He was well aware that he was setting precedents for a new industry and thus had to apply a conservative design philosophy that would guarantee the highest standard of safety. It was a responsibility he took seriously. No part of the station's design, development, construction, operation, or personnel escaped his oversight. Rickover insisted that a Naval Reactors representative who would have the authority to order reactor shutdown if safety was compromised, be present at all times during initial startup and early operation of the plant.

Although many utilities expressed interest in the project, the Duquesne Light Company of Pittsburgh offered at no cost, a site it owned at Shippingport, a small town on the Ohio River about 25 miles northwest of Pittsburgh. Duquesne Light proposed to build the turbine generator part of the station, and operate and maintain the entire facility and purchase the steam at the relatively good price of 8 miles/kW-h. The AEC would supply the reactor system and the nuclear fuel. The combination of Duquesne Light, the Shippingport site, Westinghouse, and Bettis all in the Pittsburgh area provided strong incentive to accept the Duquesne offer and locate the project there.

Shippingport would not be the first nuclear plant to put electricity to a public power grid. The EBR1 in Idaho had produced token amounts of power in 1951, the Russians built a small power plant at Obninsk in 1954, the BW4 provided electricity to a small Idaho town in July 1955, and the Calder Hall station supplied power to the United Kingdom national distribution system in October 1956. But in all these cases the reactor was primarily designed and operated for military purposes. Shippingport was the first full-scale plant designed and operated solely for the purpose of producing electricity for civilian use and advancing civilian reactor technology.

The Shippingport PWR system (Fig. 3) was essentially an enlarged four-loop version of the Mark 1/*Nautilus* system. The reactor vessel (Fig. 4) which was the largest that could be fabricated at that time (by Combustion Engineering) stood 18.3 m (60 ft) high. The system employed two different types of steam generators, two Foster-Wheeler straight horizontal steam drums similar to those used in large fossil power plants, and two B&WU-bend horizontal generators (Fig. 5) similar to those used in the *Nautilus*. The intent of including both types was that operating experience obtained from Shippingport would be used to select the steam generator design for future PWR plants. Although not normally used for reactivity control, boric acid could be injected into RCS as a means to achieve cold shutdown in the event of a stuck control rod. Emergency core cooling was achieved by redirecting the steam generator feedwater pumps to inject water directly into the reactor hot legs instead of the steam generator. The reactor containment building consisted of a mostly below ground spherical steel shell surrounded by concrete for extra radiation shielding.

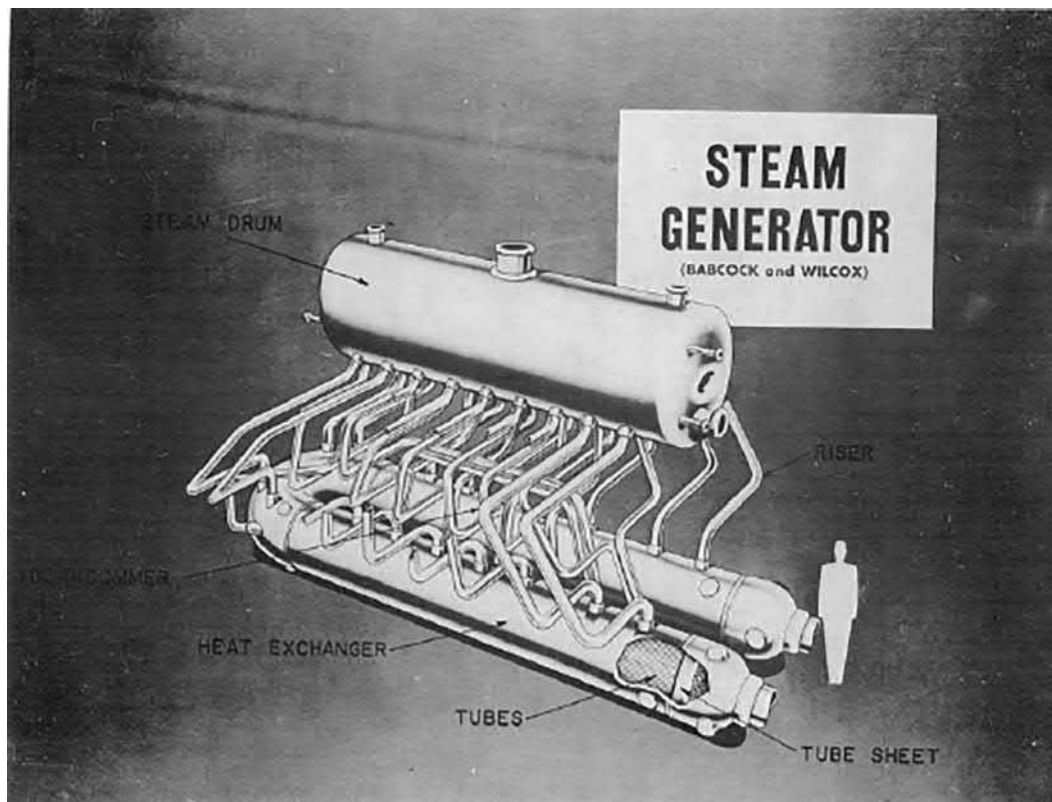


Fig. 4 Shippingport reactor vessel. Bettis (1957) *Description of the Shippingport Atomic Power Station*, WAPD-PWR-970.

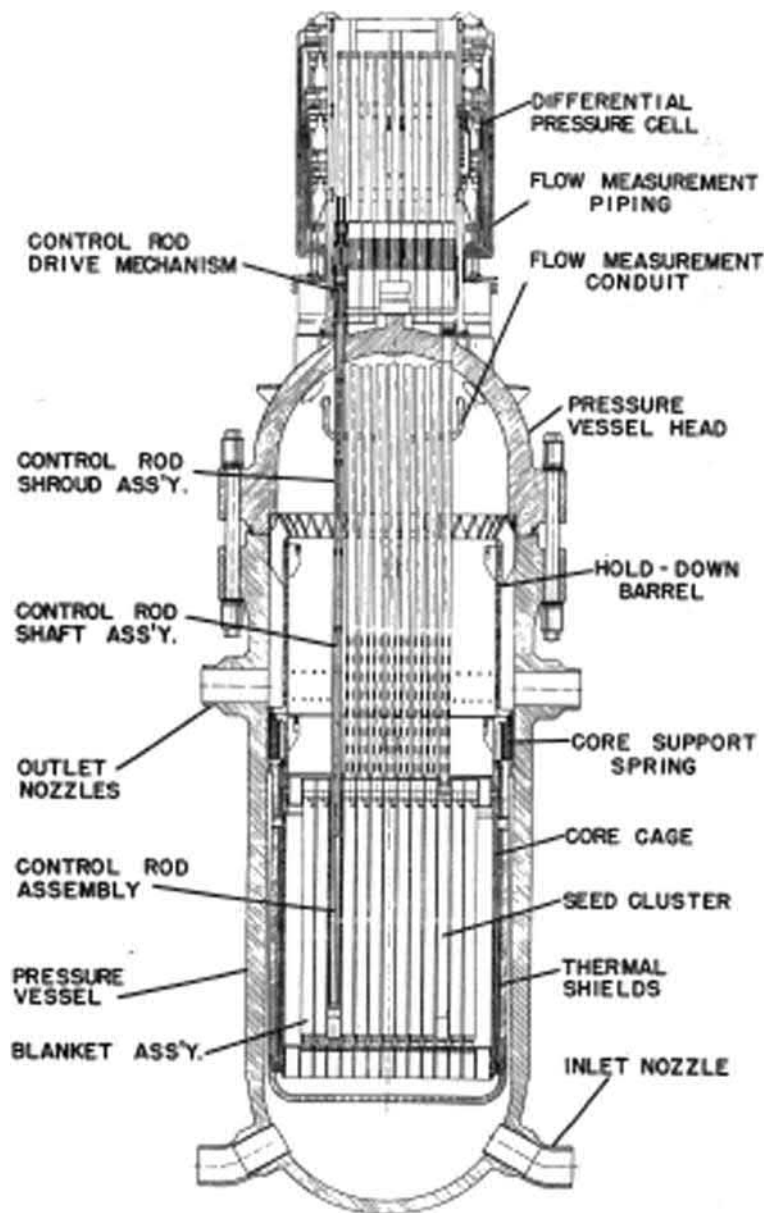


Fig. 5 Shippingport B&WU-bend steam generator. Bettis (1957) *Description of the Shippingport Atomic Power Station*, WAPD-PWR-970.

For the reactor core design, the primary goal was to achieve a long life core using low enriched Uranium. The obvious choice was to use the Uranium metal plate configuration previously used in the Mark 1 and Nautilus, especially since the plate design had good heat transfer characteristics and had performed well in there. But Rickover was concerned about the potential for corrosion of the Uranium metal fuel during long period of high power operation. The Rickover team decided to use a new fuel material consisting of Uranium dioxide (UO_2) pellets contained within thin Zircaloy 2 rods. Since little was known of the properties of UO_2 under irradiation and UO_2 had never been produced using enriched Uranium, it was decided that the power in the UO_2 assemblies should initially be limited, but then allowed to increase over time if there were no cladding integrity problems.

In order to meet these requirements, Shippingport employed a novel core "seed-blanket" design devised by Alvin Radkowsky of NRR in which 32 high enriched "seed" assemblies would provide the excess neutrons needed to drive 113 naturally enriched "blanket" assemblies (See Fig. 6). The seed assemblies for the first core would consist of a six-foot stack of fuel plate elements similar to those previously used in the Mark 1 and Nautilus (See Fig. 7A). The blanket assembly would consist of a 1.8 m (6 ft) stack of 26 cm (10.25 in) tall sub-assemblies of 12×12 arrays of Zircaloy 2 rods containing UO_2 pellets (Fig. 7B). The initial power generated in the blanket assemblies would be low, but would rise as plutonium was generated in the blanket assemblies by excess neutrons produced in the seed assemblies. Another advantage of this core design concept was that reactivity and power distribution control could be achieved by use of hafnium cruciform shaped control rods only in the 32 seed assemblies. Even without control

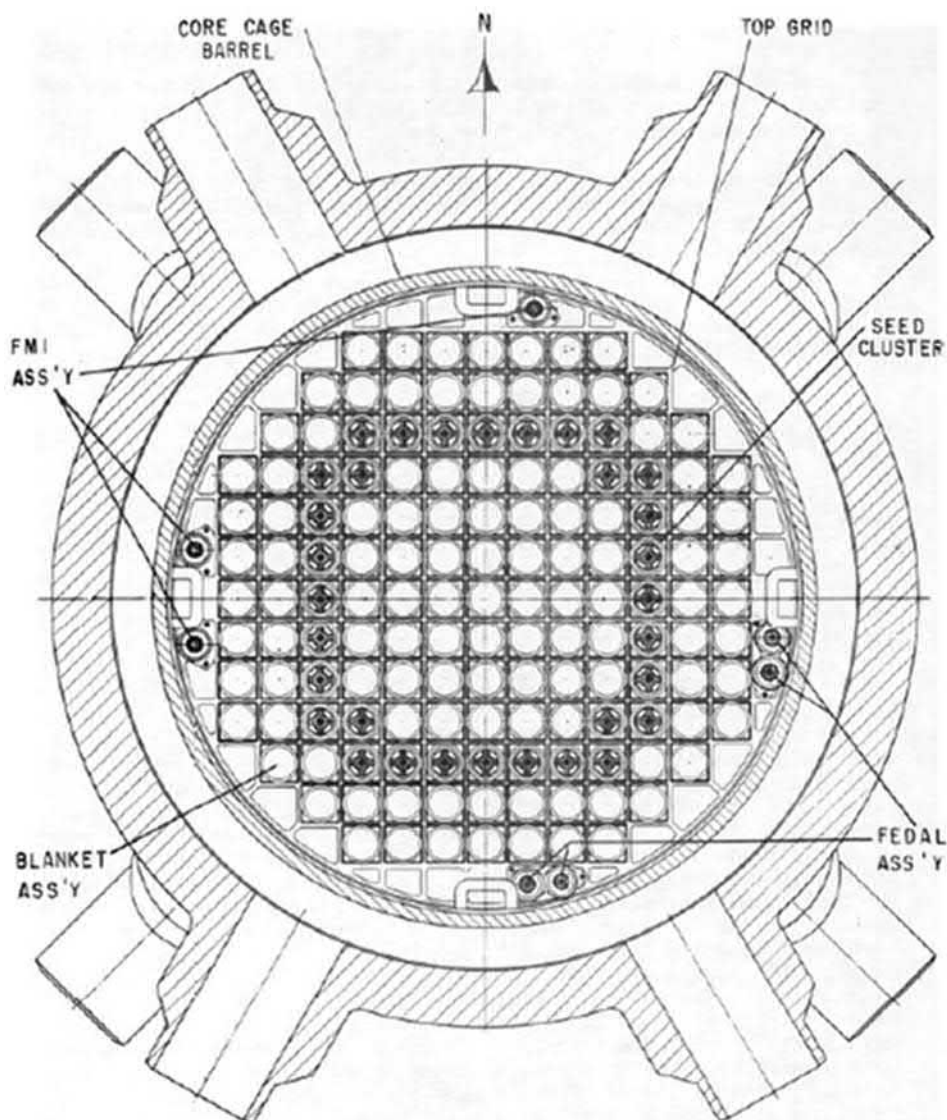


Fig. 6 Shippingport seed-blanket core 1 loading pattern. *FEDL*, failed element detection and location conduits; *FMI*, flow measurement instrumentation. Bettis (1957) *Description of the Shippingport Atomic Power Station*, WAPD-PWR-970.

rods the seed assemblies could achieve criticality only in the presence of some blanket assemblies to reflect and multiply some of the neutrons generated in the seed assemblies. The seed assemblies would be periodically replaced in order to obtain high burnup in the UO_2 fuel. The seed assemblies could be replaced without removing the head and reactor internals through refueling ports in the upper head (Fig. 4).

Shippingport achieved criticality on December 2, 1957 and operated flawlessly throughout its life. Core 1 was refueled with three additional seed regions and operated for a total of 1158 effective full power days (EFPD) and generated a total electrical output of 1.8×10^9 kW-h, half of which was ultimately generated by the UO_2 blanket assemblies. The final average burnup of the UO_2 assemblies was 8500 MWD/MTUO₂ with a peak UO_2 pellet burnup of 37,000 MWD/MTUO₂.

In 1965, Shippingport implemented the first power reactor uprate to 150 MWe (505 MWth). To support this uprate, the second Shippingport core used fuel elements consisting of UO_2 sandwiched between Zircaloy-4 plates for both the seed and blanket assemblies. This configuration provided better heat transfer to support the increased power density. The seed contained a mixture of highly enriched UO_2 and ZrO_2 , while the blanket contained UO_2 of natural enrichment. Core 2 also incorporated the first use of burnable absorbers in the form of borated stainless steel plates in the corners of each fuel assemblies. The Shippingport Core 2 operated with one seed refueling from April 1965 until February 1974 for a total of 992 EFPD.

In 1977, Shippingport began operation with its third and final seed-blanket core designed as a thorium-232 to uranium-233 "Light Water Breeder" in which excess neutrons from U-233 fission would convert the non-fissionable Th-232 into more U-233. The data and experience gained from operation of this core was crucial to the development of Thorium-based fuel cycle technology.

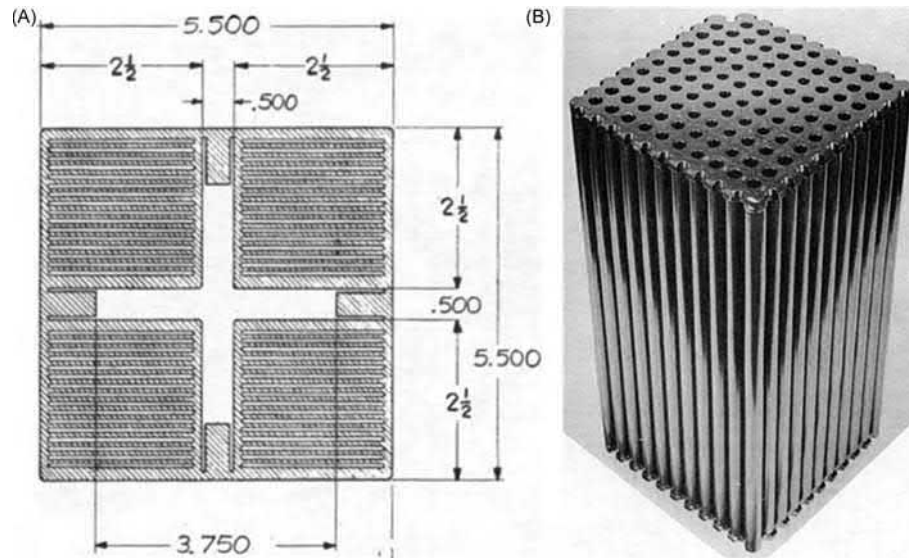


Fig. 7 Shippingport seed and blanket fuel elements. (A) Shippingport high enriched U-Zr metal seed fuel assembly. (B) Shippingport natural UO_2 blanket fuel assembly. Bettis (1957) *Description of the Shippingport Atomic Power Station*, WAPD-PWR-970.

Shippingport ceased operation in October 1982. Duquesne subsequently built two additional PWRs (Beaver Valley 1 & 2) on this site.

Overall, the Shippingport project contributed immensely to the development of nuclear power plant components, fuel element design, and analysis methods, and established the specifications, procedures, and standards for industry to follow in constructing and operating large nuclear power plants.

Commercial power plants

By the 1960s the various companies (Westinghouse, GE, CE, B&W) who had participated in the Naval and Shippingport reactor programs were well-positioned to develop and sell commercial nuclear systems of their own. And the Shippingport experience had demonstrated that utilities with properly trained personnel could safely operate a nuclear power plant. Westinghouse, B&W, and Combustion Engineering chose to develop and market the PWR concept, while GE chose the BWR. The national labs thereafter mostly focused their efforts on developing advanced reactor concepts, such as the fast breeder reactor.

Commercial PWR Development.

The 150 MWe, 485 MWth, Yankee Rowe plant, located in Rowe Massachusetts, containing a Westinghouse PWR was the first fully commercial nuclear power plant, owned and operated by a private company, to achieve criticality on January 16, 1961. Although nominally based on the Shippingport design, there were some important differences:

- An above-ground spherical steel containment building.
- Reactor coolant inlet and outlet nozzles above the core to facilitate the introduction of emergency core cooling water (See Fig. 8).
- A dedicated Emergency Core Cooling System.
- Use of boric acid in the primary coolant as a means of normal reactivity control.
- Ability to inject boron into the coolant to aid in normal shutdowns and was also used in the safety injection system (later known as the emergency core cooling system).
- Four Vertical U-tube Steam Generators.
- Uniform slightly-enriched UO_2 contained in bundles of 305 full length (90 in.) fuel rods instead of the stacked seed and blanket configuration.
- Stainless steel fuel rod cladding instead of Zircaloy (implemented in the early cycles as a cost reduction measure, but was later replaced by Zircaloy).
- Silver-indium-cadmium cruciform control rods (instead of hafnium) that were inserted in the gaps between the fuel assemblies. These rods also had a long Zircaloy-2 follower tip that would remain in the reactor core even when the poison region of the rods were fully withdrawn in order to minimize power peaking that would otherwise occur if the gaps were empty.
- Elimination of the reactor head refueling channels.
- Annual refueling cycles.
- Movable in-core cobalt wire flux detectors.

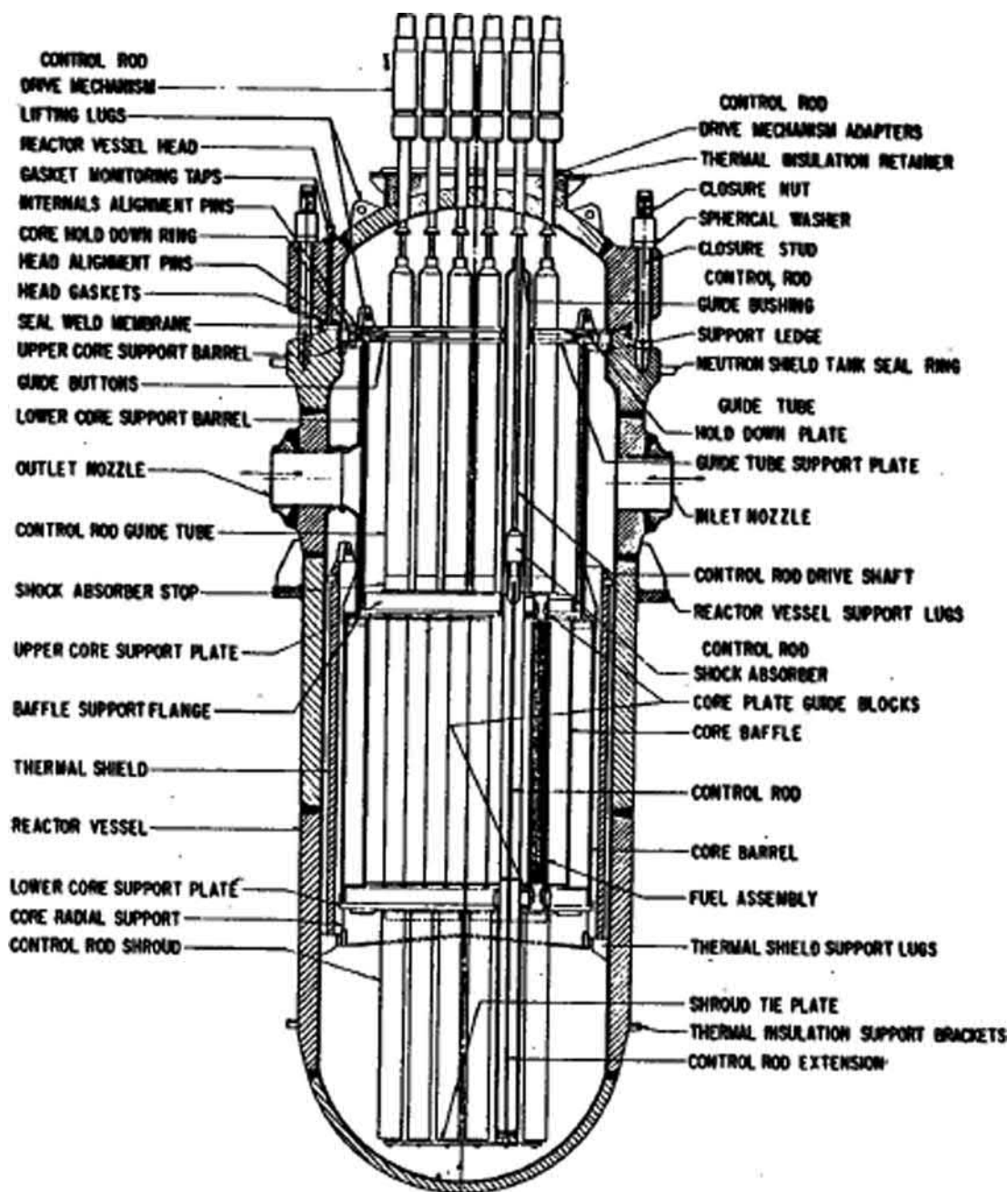


Fig. 8 Yankee Rowe reactor vessel. Westinghouse Electric Company (1967) *Engineering and Metallurgical Evaluation of the Yankee Core 1 Spent Fuel*, WCAP-6084.

By 1963, the stage was set for the first nuclear power stations that could function as large base load units in multi-station grids on a nearly equal availability factor and operational cost footing with coal- or oil-powered units. The first "full scale" stations were the three-loop 436 MWe San Onofre 1 PWR (Westinghouse), the 275 MWe Indian Point 1 (1962, B&W), and the four-loop 616 MWe Connecticut Yankee PWR (Westinghouse) Nuclear Generating Station.

Since these early days nearly 300 PWR power plants have been built throughout the world, all based on the design established for the Mark 1/*Nautilus* and Shippingport concepts. Current PWR designs are typically 1000–1400 MWe (3500–4000 MWth) with significant improvements in accident mitigation features. Recently there has been a drive to standardize nuclear power plant design in order to facilitate construction and licensing approval. Current standard PWR designs include the Westinghouse AP1000, the Framatome EPR, the Korean APR1400, and the Russian VVER1000. These standardized designs are being used as the template for the construction of a fleet of PWRs in several countries.

Commercial BWR development

The evolution of the BWR power plant occurred in parallel to that of the PWR since there were obvious significant commercial advantages of the BWR design including:

- Lower capital cost than the PWR due to single coolant loop design and smaller containment.
- Large negative moderator void coefficient of reactivity and reactivity control by the recirculation pumps makes the plant easier to load-follow than a PWR.
- Significantly reduced operating pressure (from about 2250 psia down to 1000 psia) that simplified the reactor coolant system and allowed for a smaller primary containment.
- Elimination of boric acid from the reactor coolant system significantly reduced corrosion of all the RCS components and assures that the moderator temperature feedback is negative at operating conditions.

The first commercial-size BWR plant was the Dresden Unit 1, which began operation in 1960. This was, in large part, pursued by GE and the Commonwealth Edison Company of Chicago following the success of the BORAX experiments, but preceded by two smaller scale projects; the experimental BWR (EBWR) which started construction in 1954 at ANL in Illinois, and went operational in 1956, as well as a developmental plant constructed by GE at its Vallecitos Atomic Laboratory near Pleasanton, California, the Vallecitos BWR (VBWR) became the first privately built nuclear reactor that provided electricity to a public grid in 1957. Cutaway illustrations are provided in Fig. 9A and B for the EBWR and VBWR, respectively (Fennern, 2018). The VBWR, in fact, was the first power reactor to obtain a commercial license (Power Reactor License No. 1), which helped promote the commercial expansion of the BWR in the US with the Dresden and Humboldt Bay plants, among others, and also internationally with the Dodewaard reactor, the first GE BWR generating ~58 MW in the Netherlands from 1969 through 1997, and the first large nuclear reactor in Germany, Gundremmingen (KRB) Unit A, producing ~237 MW from 1966 through 1977.

The table below (GE Hitachi Nuclear Energy, 2007) illustrates the history of commercial GE BWRs as they followed the Dresden Unit 1's success (Table 1).

To highlight the technical progression toward the modern versions of the GE BWR, it should be noted that the Dresden 1 BWR/1 system did not include the traditional direct steam cycle, but employed a steam drum and heat exchanger to produce turbine steam in a secondary system, basically a steam generator.

It wasn't until the mid-1960s and the BWR/2 at Oyster Creek that the steam generator concept was eliminated. The distinctive characteristics of the Generation II BWRs (BWR/2 through BWR/6) include direct-cycle, forced circulation, internal steam separation and a higher power output. Fig. 10 illustrates these distinctive BWR characteristics for the BWR/6 and the progression to the ABWR (Fennern, 2018). It should be noted that these evolving features were made feasible in large part due to improvements in nuclear fuel bundle and core design, leading to a higher power density (BWR/4, Vermont Yankee). Likewise, key thermal hydraulic advancements that facilitated this evolution included the transition from external loop plants to jet pumps (BWR/3, Dresden 2), and subsequently to reactor internal pumps (ABWR). GE/Hitachi's future technology strongly favors the use of natural circulation in currently marketed and future reactor concepts such as the Economic Simplified BWR (ESBWR) and the BWRX-300, a ~300 MWe small modular reactor (SMR).

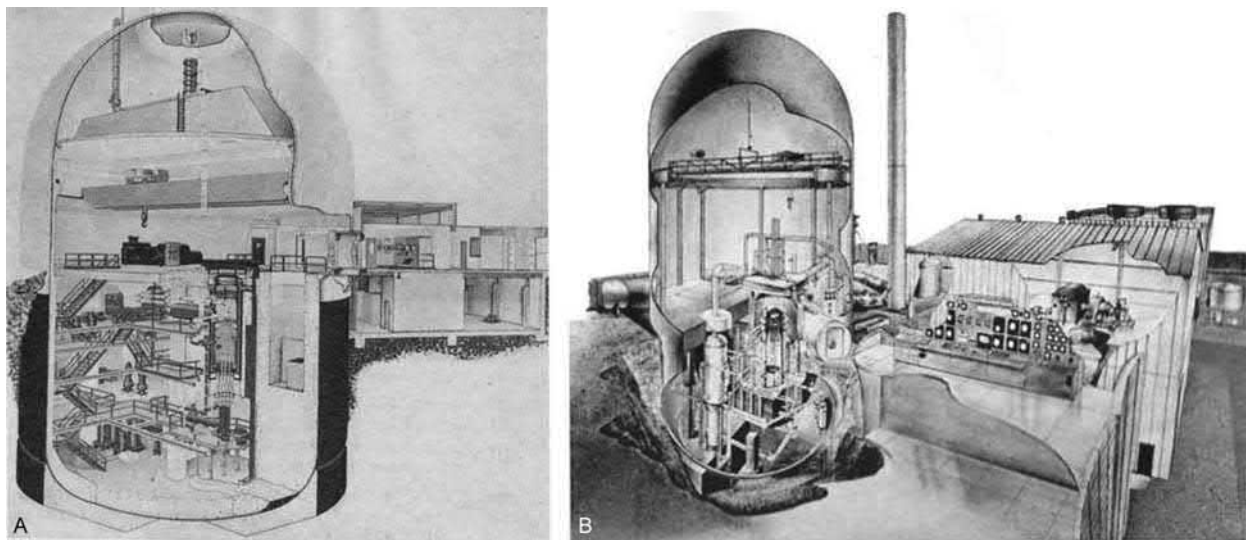


Fig. 9 Early developmental BWR prototypes (A) EBWR plant cutaway (Fennern, 2018). (B) VBWR plant cutaway (Fennern, 2018). From Kramer, 1958.

Table 1 Evolution of the GE BWR (GE Hitachi Nuclear Energy, 2007)

<i>Product line</i>	<i>First commercial operation date</i>	<i>Representative plant Characteristics</i>
BWR/1	1960	Dresden 1 Initial commercial-size BWR
BWR/2	1969	Oyster creek Plants purchased solely on economics Large direct cycle
BWR/3	1971	Dresden 2 First jet pump application Improved ECCS: spray and flood capability
BWR/4	1972	Vermont Yankee Increased power density (20%)
BWR/5	1977	Tokai 2 Improved ECCS Valve flow control
BWR/6	1978	Cofrentes Compact control room Solid-state nuclear system protection system
ABWR	1996	Kashiwazaki-Kariwa 6 Reactor internal pumps Fine-motion control rod drives Advanced control room, digital solid-state microprocessors Fiber optic data transmission/multiplexing Increased number of fuel bundles Titanium condenser Improved ECCS: high/low pressure flooders

In addition to the above noted advantageous features of the BWR, there has also been a substantial evolution of the BWR fuel assembly design, in comparison to the PWR. While the original BWR fuel resembled that of a PWR with the fuel rods assembled with fixed spacing throughout the entire height, this resulted in a relatively low fuel utilization (burnup) compared to PWRs due to the reduced moderation caused by the steam in the upper core region. This led to significant evolutionary changes in BWR fuel designs, such as variable pitch, part-length fuel rods, and internal water holes, that were developed to compensate for the two-phase flow impact upon moderation. Likewise, the complexity of fuel and burnable absorber radial/axial loading arrangements has grown exponentially so that today's BWR fuel assemblies are essentially optimized to specific plants and operating cycles. This has ultimately resulted into greatly improved fuel utilization for modern BWRs more comparable to that of PWRs.

Dimension wise, however, the BWR reactor vessel and active core are much larger than for a PWR of equivalent total power. This is due to the lower core power density necessary to maintain adequate margin to the fuel design limits in a BWR, and is an attribute that made the BWR unsuitable for submarine propulsion, where size and space considerations are paramount.

The containment for the BWR, given that is a more active structure than that for PWRs, has also evolved considerably over time. The first BWR containment systems were spherical steel structures and dry. These were superseded by the evolution of the Mark I, II, and III containments which included a pressure suppression pool that offers important advantages; primarily as a heat sink for steam from various sources such as LOCA (Loss of Coolant Accident) blowdown, safety/relief valves, and reactor discharge steam. The in-containment suppression pool also serves as a secondary source of water to support the Emergency Core Cooling System (ECCS), isolation make up, and cooling pumps. **Figs. 11A and B (Fennern, 2018)** provide the contrast between the early Mark I "inverted light bulb" drywell with the toroidal suppression pool, and the Mark III containment vessel used for the BWR/6 which exploited a simpler right circular geometry that combines a steel containment or liner surrounded by concrete shield or vessel, respectively.

As BWRs evolved from their first generation, concerns grew about the ability to provide sufficient core cooling during a LOCA, and the early BWRs did not have the ECCS, although the BWR/2 had limited capacity and spray, and some BWR/1 units were retrofitted to have some forms of ECCS. It wasn't until 1973 when US 10CFR50.46 defined the ECCS performance criteria, which limited peak clad temperature to 1200 °C, that prompted multi-layered approach to the ECCS, that for the BWR/5 and BWR/6 includes 2 ECCS pumps, a Residual Heat Removal (RHR) pump for Low Pressure Coolant Injection (LPCI), a Low Pressure Core Spray (LPCS), and a High Pressure Core Spray (HPCS) on a dedicated diesel generator.

Although the BWR had its roots with GE in the United States, BWRs were also successfully built and operated around the world, including Mexico, Netherlands, Italy, Spain, Switzerland, Germany, Japan, India, and Taiwan. In fact, in several instances, the original GE BWR technology was advanced by other organizations, such as by Kraftwerk Union AG, in Germany, originally founded by Allgemeine Elektrizitäts-Gesellschaft (AEG) and Siemens, which contributed to key modernized features of the Series 72 Gundremmingen

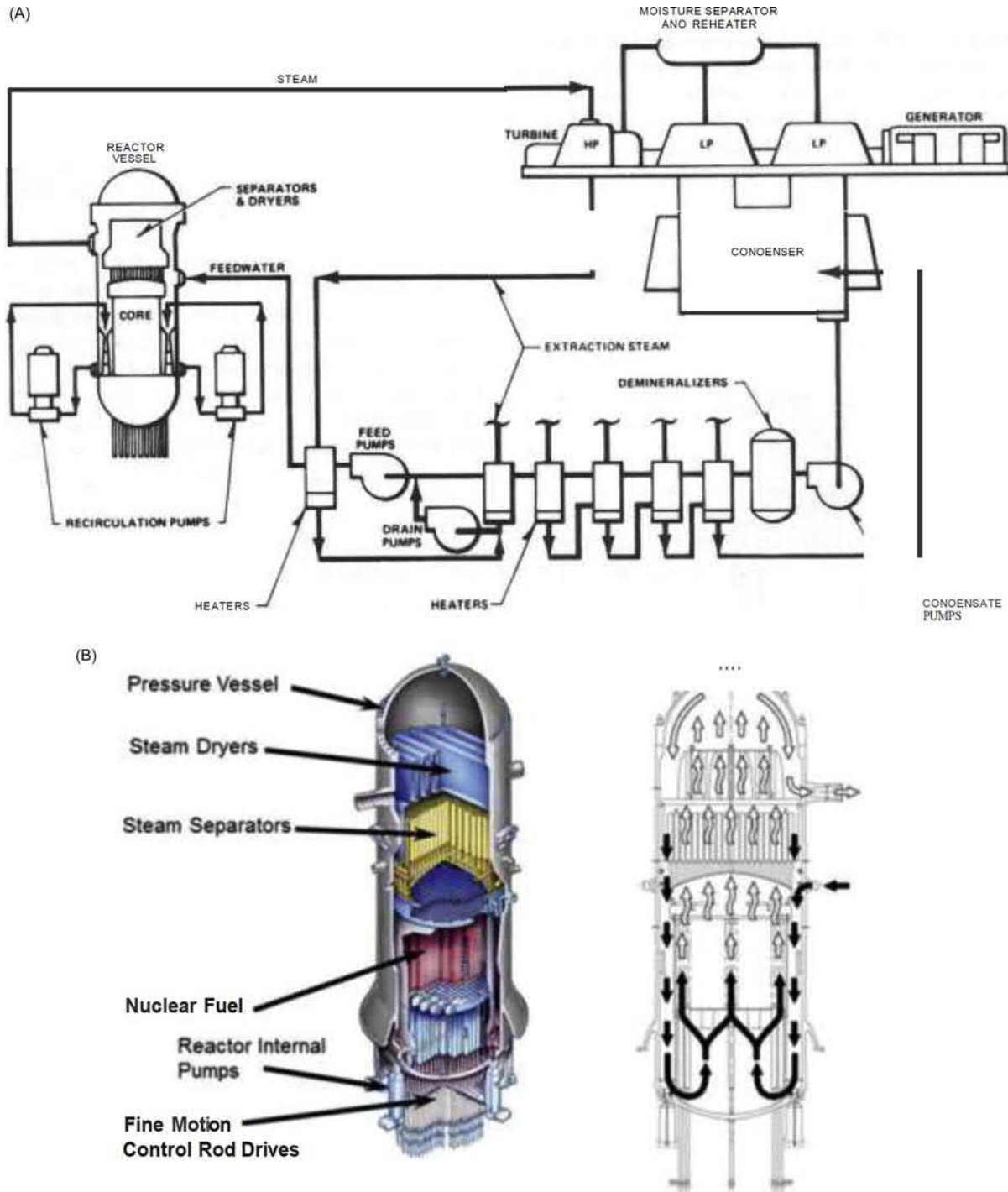


Fig. 10 Evolution to the BWR/6 and ABWR. (A) Direct Cycle BWR/6 (typical of BWR/6) (Fennern, 2018 from BWR/6, 1980). (B) ABWR reactor assembly and coolant flow path (Fennern, 2018). (B): From GE Hitachi Nuclear Energy (2007) *The ABWR Plant General Description*. https://nuclear. gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ABWR%20General%20Description%20Book.pdf.

Units B and C that began operation in 1984. In Sweden, the first western commercial LWR was designed, built, and operated without licensing from US vendors, where the Oskarshamnverket Kraftgrupp AB (OKG) consortium majority owned by Sydkraft ordered the construction of the Oskarshamn Unit 1 BWR from ASEA-Atom, which went operational in 1972 and was followed by Units 2 and 3 connected to the grid in 1974 and 1985, respectively. In Asia, two BWR/1 reactors were constructed at the Tarapur Atomic Power Station and brought online in 1969, while in Japan, several BWRs ranging from BWR/2 through BWR/5 were constructed, beginning with the Tsuruga BWR which went operational in 1970. One of the highlights of international cooperation constitutes the programs

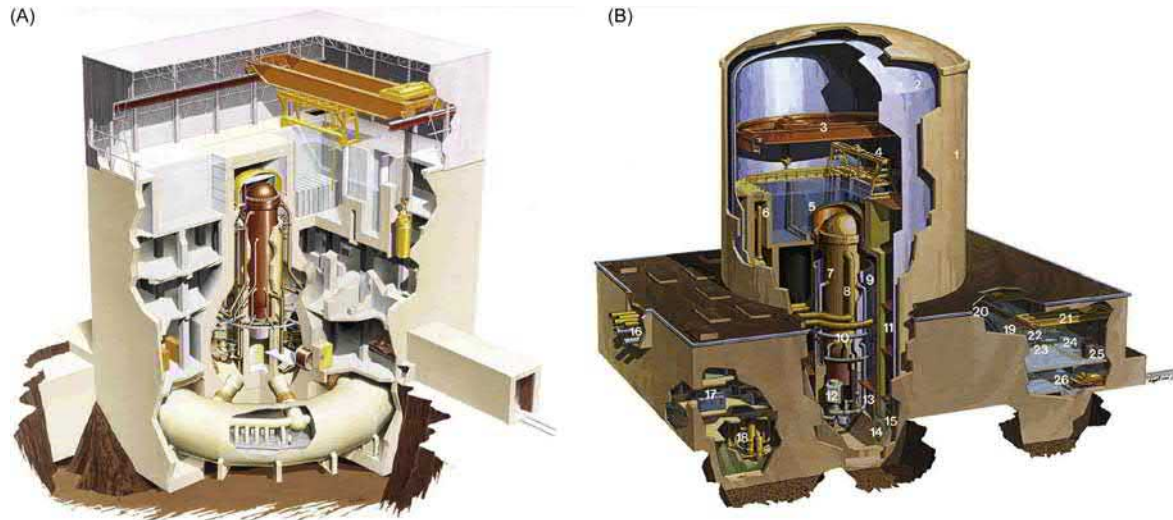


Fig. 11 Transition from the Mark I–III containment. (A) Mark I containment (Fennern, 2018). (B): Mark III containment (Fennern, 2018). Courtesy GE-Hitachi Nuclear Energy.

among GE, Hitachi, and Toshiba, which in addition to providing improvements for the BWR ultimately led to the design and construction of the Kashiwazaki-Kariwa ABWR in 1996, and three additional ABWRs now operational plus two more under construction in Japan, and two in Taiwan.

Summary

Modern PWR and BWR power plants have proven highly successful as a reliable, cost effective, and safe source of electrical power for the public grid. These modern designs were the result of many years of design improvements based on experience from early tests and prototype designs. And in both cases both of these concepts owe their success to the vision and relentless drive of a few individuals who were key in convincing others to commit to the long term development necessary to bring to fruition.

See Also: Advanced Light Water-Cooled Reactor Concepts.

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Physics of the Fission Chain Reaction in Thermal and Fast Reactors

Pavel V. Tsvetkov^a and Russell E. Stachowski^b, ^a Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States; and ^b Global Nuclear Fuel, Wilmington, NC, United States

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Glossary

Actinide The series of elements from atomic number 89 (actinium) through 103 (lawrencium), thus including uranium, neptunium and plutonium.

Binary nuclear reaction Change in the identity or characteristics of an atomic nucleus (target) induced by bombarding it with a nuclear particle (projectile).

Chain reaction A reaction yielding products that initiate reactions of the same kind.

Criticality (critical state) An equilibrium (time-independent) state of a neutron population.

Cross section Microscopic cross section is a measure of the probability of interaction of a neutron with a single nucleus. Macroscopic cross section is a measure of the probability of interaction of a neutron within a specific concentration of nuclei (within a volume of material).

Delayed neutrons Neutrons emitted by excited fission products in decay chains formed following fission reactions.

Electron volt (eV) A unit of energy $1 \text{ eV} = 1.6022 \times 10^{-19} \text{ J}$. $1 \text{ keV} = 10^3 \text{ eV}$; $1 \text{ MeV} = 10^6 \text{ eV}$.

Evaluated nuclear data files Digitized and analyzed collections of nuclear data sets gathered from all available experiments and nuclear models that are assessed (evaluated) to be uniquely describing relevant nuclear reactions.

Fast reactors Nuclear reactors characterized by dominance of fission reactions induced by high energy (fast) neutrons.

Fission fragments Fully ionized parts of a heavy atomic nucleus formed directly in a fission reaction either induced by interacting neutron or occurred spontaneously.

Fission neutrons Neutrons emanating in fission reactions.

Fission products Fully neutralized fission fragments formed in fission reactions and going through decay chains.

Fission reaction (fission) Subdivision of a heavy atomic nucleus, such as that of uranium or plutonium, into two or more fragments. The reaction results in a yield of a large amount of energy. It may occur spontaneously (spontaneous fission) or may be induced by the excitation of the nucleus with a variety of particles (induced fission).

Fission spectrum Energy distribution of neutrons (fission neutrons) emanating in fission reactions.

Multiplication factor A ratio of the number of neutrons in a generation to the number of neutrons in the preceding generation.

Neutron economy A relationship between the neutrons created (neutron production) in a configuration and the neutrons lost (neutron loss) through various nuclear processes (absorption, leakage out, etc.).

Nuclear fission chain reaction Series of nuclear fission reactions where each reaction is initiated (induced) by a neutron released in a previous reaction.

Nuclear physics The field of science that is focused on atomic nuclei, their structures and interactions.

Prompt neutrons Neutrons emitted by the fission event almost instantly (within 10^{-13} s).

Thermal reactors Nuclear reactors characterized by dominance of fission reactions induced by low energy (thermal) neutrons.

Abbreviations

AGR Advanced Gas Reactor
 BWR Boiling Water Reactor
 CANDU CANadian Deuterium Uranium reactor
 HTGR High Temperature Gas-cooled Reactor
 LWR Light Water Reactor
 MR Moderating Ratio
 MSDP Macroscopic Slowing Down Power
 PWR Pressurized Water Reactor
 SFR Sodium-cooled Fast Reactor

Fission reactions

In the fission process, heavy nuclei of actinides are split into several (usually two) light nuclei of fission fragments while emitting a number (typically 2 or 3) prompt neutrons (Klay, 2021). Combined masses of these fragments and emitted neutrons are less than masses of fissioning actinides. This results in the significant energy release. Today's nuclear energy technologies are the culmination of decades of development efforts to mature the means to induce fission reactions and harvest their energy release (Adler and Von Halban, 1939; National Research Council, 2013).

The challenge is to induce fissions of heavy nuclei. Some actinides heavier than plutonium have significant probability to undergo radioactive decay via spontaneous fission. However, for practical applications, it is necessary to hit heavy nuclei with sub-atomic particles, such as neutrons, to produce compound nuclei in extremely highly excited states. The unstable compound nucleus may release its excitation energy through a variety of decay reactions including the fission process (IAEA, 2000; Klay, 2021).

Any very heavy nucleus can be made to fission accompanied by a net release of energy if hit sufficiently hard by some incident particle. However, very few nuclides can fission by the absorption of a neutron with negligible incident kinetic energy. Such neutrons typically are thermal neutrons, neutrons which have lost almost all of their kinetic energy and are moving with speeds comparable to the speed of the thermal motion of the ambient atoms. At room temperature, thermal neutrons have kinetic energies lower than 1 eV. Neutrons with higher energies are called epithermal neutrons. Neutrons still carrying energies closer to their initial fission neutron energies are called fast neutrons.

The binding energy of the thermal neutron in the compound nucleus provides sufficient excitation energy that the compound nucleus can split apart (Klay, 2021). Nuclides, such as ^{235}U , ^{233}U , and ^{239}Pu , that fission upon the absorption of a slow moving (thermal) neutron, are called fissile nuclei. They play the key role in nuclear reactors because probabilities of fissions to occur are much higher at lower energies. On the other hand, nuclides that fission only when they are hit by neutrons with energies closer to fission neutron energies (fast neutrons), such as ^{238}U and ^{240}Pu , are called fissionable, they undergo fast fissions. Typically, these fissionable nuclides are characterized by high probability to capture thermal and epithermal neutrons, converting them into fissile nuclides through transformation chains (Kawano et al., 2008; Shusterman, 2021).

In addition to prompt fission neutrons emitted directly in the fission process, small fractions of fission neutrons, called delayed neutrons, are emitted by nuclides in decay chains of fission products (IAEA, 2008; Tudora and Hamsch, 2010; Klay, 2021). Delayed neutrons released per fission events depend on the fissioning nuclide and the energy of the neutron inducing the fission (Oblozinsky, 2009; Rudstam et al., 2002; Klay, 2021).

All fissionable nuclides produce distributions of prompt fission-neutron energies which go to zero at low and high energies and attain maxima around 0.7 MeV (Staples et al., 1995; Klay, 2021). The energies of the prompt fission neutrons are described by a fission spectrum distribution function, similar to that shown in Fig. 1 for prompt fissions in U^{235} . The average energy of prompt fission neutrons is about 2 MeV (Capote et al., 2016; Haight et al., 2009; Kapoor et al., 1963). Delayed neutrons, emitted through decay chains of fission products, carry considerably lower kinetic energies of a few hundreds of keVs (Wilson and England, 2000).

Cross sections

As a beam of neutrons travels through matter, the intensity of the neutron beam will decrease as neutrons are removed from the beam by absorption and scattering processes. The primary form of neutron absorption in a thermal system is the capture reaction, in the form of an (n,γ) interaction. This implies that a nucleus in the system will absorb one of the beam's neutrons and radiate

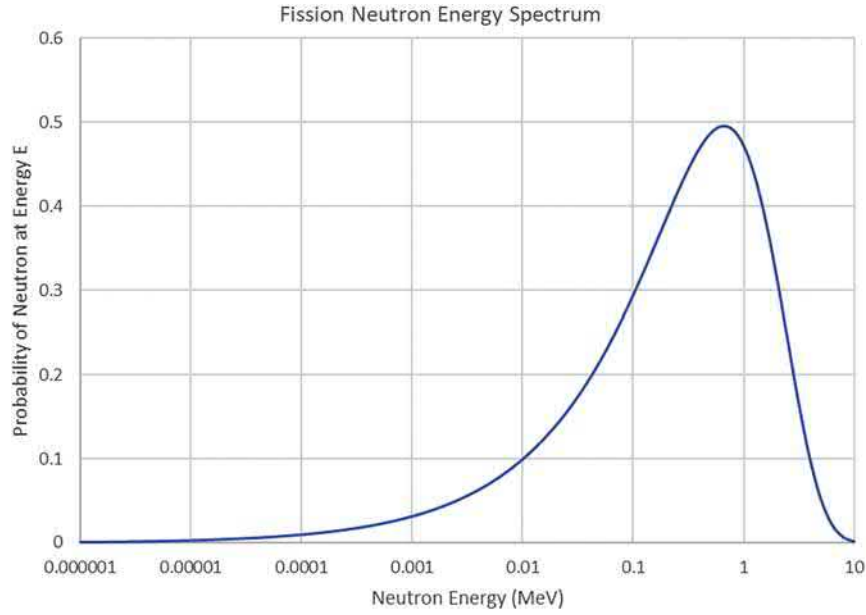


Fig. 1 Prompt neutron energy spectrum.

a gamma particle. The probability, or measure of, this neutron removal reaction occurring depends on the material the neutron is traveling in, that material's temperature and density, and the energy of the neutron. The "probability" is typically referred to as that material's cross section, for the given reaction type ((n, γ) in this case).

The cross section can vary greatly as a function of neutron energy, and is extremely significant to the understanding of the nuclear aspects of reactor simulation and operations.

The probability of a collision event is dependent not only upon the nucleus type, but on the relative velocity of the neutron and nucleus as well. The interaction rate in a bulk sample will also be proportional to the density of nuclei, N (number $\bullet$ cm $^{-3}$), and neutrons, n (number $\bullet$ cm $^{-3}$). Hence, the reaction rate in such a sample can be written as:

$$R(E_n) = \sigma_j(E_n)Nn(E_n)v \quad (1)$$

where R = reaction rate (reactions $\bullet$ cm $^{-3}$ $\bullet$ sec $^{-1}$), E_n = neutron energy (electron-volts), v = neutron speed (cm $\bullet$ sec $^{-1}$), σ_j = microscopic cross section for reaction type j (cm 2).

The proportionality constant σ is called the microscopic cross section and is a measure of the probability of interaction of a neutron with a single nucleus. The dependence of σ on neutron energy is explicitly shown in Eq. (1), but will henceforth be omitted for simplicity. Since collision reaction probabilities depend upon the relative velocity of neutrons and nuclei, the thermal motion of nuclei must be accounted for as low neutron energies (i.e., when the relative velocity of the neutron and nucleus may be significantly different from the neutron velocity), and at energies where the cross section varies rapidly. The latter situation is an important consideration for resonance absorption and, as will be discussed later, is usually accounted for by defining an "effective" cross section.

Each characteristic neutron interaction has an associated neutron cross section and is denoted accordingly:

- Elastic scattering σ_s .
- Inelastic Scattering σ_i .
- Radiative capture σ_γ .
- Charged-particle capture σ_α , σ_p .
- Fission σ_f .

The total cross section:

$$\sigma_t = \sigma_s + \sigma_i + \sigma_\gamma + \sigma_\alpha + \sigma_p + \sigma_f + \dots \quad (2)$$

measures the probability that an interaction of any type will occur when a neutron collides with a particular target nucleus.

The sum of the cross sections of all capture reactions is known as the absorption cross section, denoted by σ_a :

$$\sigma_a = \sigma_\gamma + \sigma_\alpha + \sigma_p + \sigma_f + \dots \quad (3)$$

Note that the fission cross section has been included in Eq. (3). While fission is probably the most significant single reaction type in a nuclear fuel system, it is nonetheless a constituent component of both the absorption and total cross sections.

The product $\sigma \bullet N$ arises frequently and is denoted as:

$$\Sigma \equiv N\sigma \text{ (cm}^{-1}\text{)} \quad (4)$$

While the microscopic cross section, σ , characterizes the probability of interaction with a single nucleus, Σ characterizes the probability of interaction with a specific concentration of nuclei and is therefore called the macroscopic cross section.

The product $n \bullet v$ is called the neutron flux and is assigned the symbol:

$$\phi \equiv nv \text{ (neutrons} \bullet \text{cm}^{-2} \bullet \text{sec}^{-1}\text{)} \quad (5)$$

Although in other physical systems the term flux is ordinarily used to denote a vector quantity, here flux is a scalar quantity and can be thought of as the total neutron path length per cm^3 per sec (i.e., neutron $\text{cm}/\text{cm}^3 \text{ sec}$). It logically follows that Σ is just the probability of interaction per unit path length, and Eq. (1) can be rewritten as:

$$R = \Sigma\phi \quad (6)$$

The term “thermal” refers to the neutron energy at which the cross section is evaluated, namely 0.025 eV (2200 m/s), the most probable energy for neutrons in thermal equilibrium with the system constituents at 20 °C. Materials that exhibit large absorption cross sections, such as B^{10} (used to control the reaction) or Xe^{135} (a product of the fission reaction) are frequently referred to as neutron poisons.

Fig. 2 shows the energy dependence of the total cross section for three important isotopes. The majority of the cross section is σ_a for B^{10} and U^{238} since the reactions are dominated by absorption. The cross section behavior of B^{10} is commonly called a “ $1/v$ ” absorber because the cross section is inversely proportional to the neutron velocity over a wide range of energy. As is typical of heavy nuclei, U^{238} exhibits a number of sharp absorption peaks. These peaks are referred to as resonances, since they occur at those neutron energies for which the combined energy of the neutron and nucleus match one of the naturally occurring energy levels of the compound nucleus (in this case U^{239}). The well-defined resonances at upper thermal and intermediate energies are called resolved resonances. Notably, there are thousands of unresolved resonances at higher energies which cannot be characterized by well-defined parameters and have to be accounted for together using reactor rate conservation together with stochastic approaches.

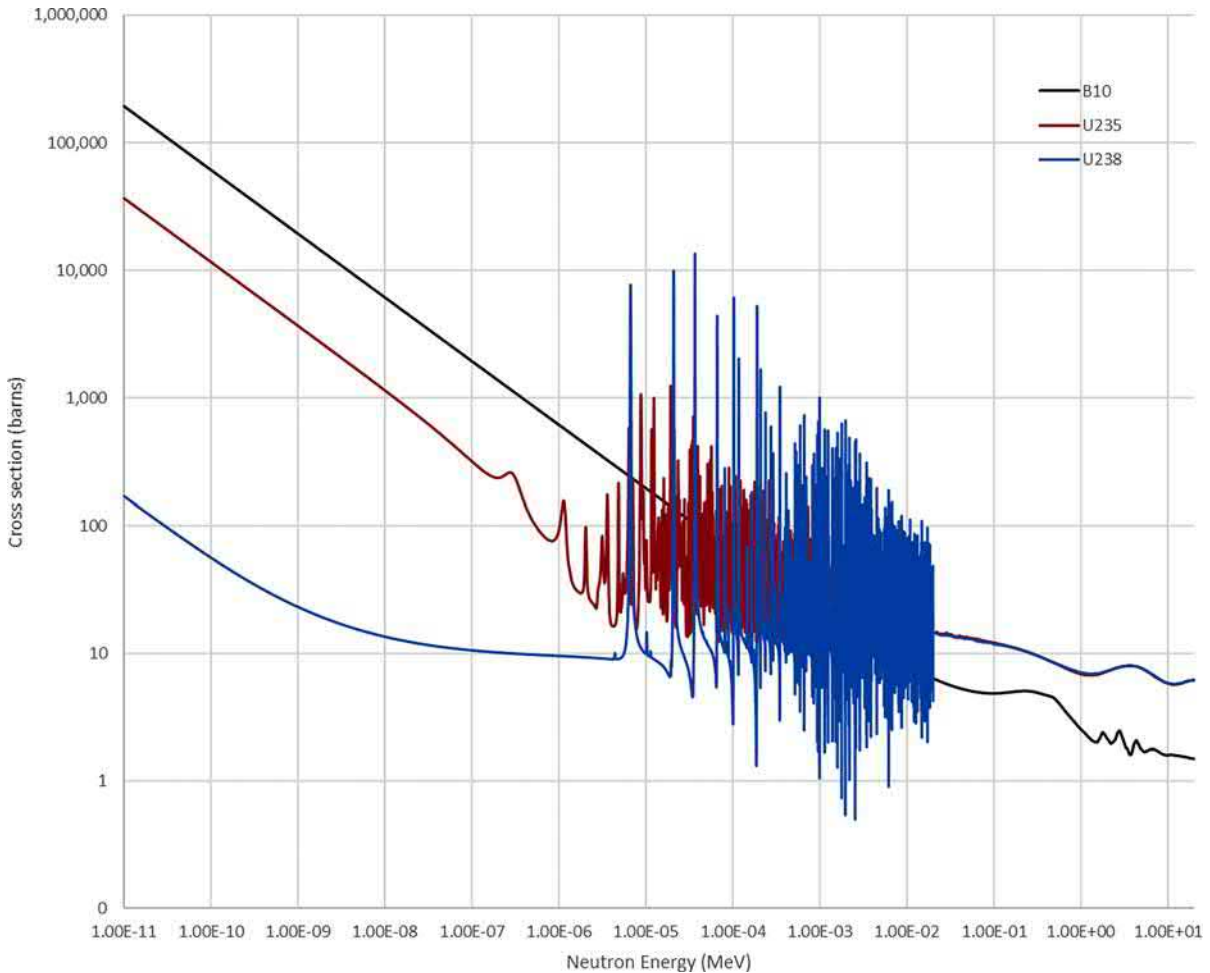


Fig. 2 Total cross sections for B^{10} , U^{235} and U^{238} as a function of incident neutron energy.

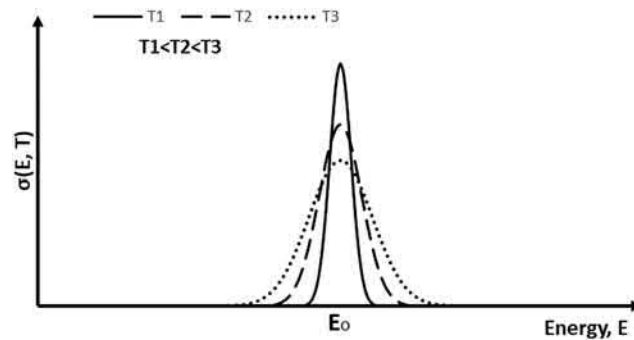


Fig. 3 Concept of Doppler resonance broadening.

The cross section of U^{235} is dominated by the fission cross section at low energies, and this is two orders of magnitude higher than the total cross section for the other constituent isotope of uranium (U^{238}).

The cross section in the high energy end of the plot is fairly small. The epithermal region energy range (between thermal and fast energies) for the U^{238} capture cross section is characterized by many large resonances, or spikes, while the thermal cross section is fairly small. The maximum U^{238} epithermal cross section is roughly four orders of magnitude larger than the corresponding fast and thermal cross sections. This implies a much greater probability (10^4) that an epithermal neutron will be interact with U^{238} , rather than a fast or thermal neutron.

Not all nuclei cross sections exhibit the large energy dependence of the U^{238} cross section, although all neutron cross sections are incident neutron energy dependent. Note the use of the term incident neutron; it refers to a free neutron with potential for interaction(s) with target nuclei. The term distinguishes those neutrons that effectively are input to interactions from those which are effectively output from those interactions.

The presence of a strong resonance tends to depress the neutron flux in the energy band of the resonance. The associated dip in the reaction rate is termed “energy self-shielding.” Doppler broadening of the resonance reduces this energy self-shielding effect. Recalling from Eq. (6) that reaction rate is proportional to cross section and flux, the reduced flux depression results in an increased absorption rate when integrated over the entire resonance.

Cross section resonances have a finite energy width, as opposed to occurring at discrete energies, due to the fact that σ is expressed as a function of neutron energy (or velocity v_n) but is actually dependent upon the relative velocity of the neutron and nucleus $|v_n - v_N|$. Due to the distribution of nuclei velocities from thermal motion, there exists a range of neutron velocities at which the relative velocity $|v_n - v_N|$ matches the resonance. Furthermore, an increase in the temperature of the absorber nuclei will result in a broader distribution of nuclei velocities, v_N , and therefore a broader range of neutron velocities for which $|v_n - v_N|$ matches the resonance. This effectively broadens the resonance peak, and the phenomenon is known as “Doppler broadening” of the resonance. Fig. 3 shows how, for increasing temperatures, the peak of the cross section centered about a particular neutron energy decreases, but the shape broadens, thereby increasing the probability interaction with the spectrum of neutrons. Hence, the net effect of a temperature increase is to increase resonance absorption. This effect has important implications with respect to reactor operation and safety, and will be discussed in a later section.

Moderation and neutron economy in thermal reactors

Since early nuclear engineering efforts, many fission reactor types have been conceived and developed conceptually. The thermal reactor family includes heavy water and light water moderated (and -cooled) reactors and graphite-moderated reactors.

Some of these developments resulted in commercially viable design lines, of which light water reactors (LWRs), both pressurized water reactors (PWRs) and boiling water reactors (BWRs) are the most successful. Because water is being used as coolant and moderator in LWRs, high energy fission neutrons lose their energy (slow down or, in other words, moderate) very quickly. Hence, the great majority of the fissions in the LWR cores is induced by neutrons with energies around 1 eV or lower. Customarily, such low energy neutrons are called thermal neutrons. Subsequently, all LWRs are thermal reactors.

The process whereby a neutron loses energy in a collision with a nucleus is called moderation. When a free neutron loses energy, we say that neutron “slows down” or is moderated. A fission neutron, born with an energy of 2 MeV slows down to below 1 eV while traveling in water, and has thus been moderated by water. The water in an LWR is a “moderator,” in addition to its being the coolant used for heat removal for power conversion to electricity and for safety cooling systems.

For a thermal reactor to be able to be critical, it is not enough to provide sufficient neutron moderation to slow fission neutrons down to thermal energies. It is paramount to complement neutron moderation with sufficiently small parasitic neutron capture probabilities. A neutron capture is considered parasitic if it does not result in a fission event. Therefore, the third necessity is to also enhance neutron-induced fission in the system.

The fast fission neutrons lose their kinetic energy primarily through elastic scattering from moderator nuclei with small mass numbers (except at MeV energies in which inelastic scattering by heavier constituents contributes to the neutron slowing down). Scattering nuclei with small mass numbers cause greater average energy loss per collision and thus thermalize fast neutrons with fewer scatters than do nuclei with large mass numbers. In addition to having a small mass number, a good moderator should be able to provide a high scattering probability and a low absorption probability (to avoid loss of neutrons before they can cause fissions).

The lowest mass moderator is hydrogen; the mass of its nucleus is comparable to that of a neutron. A neutron loses on average half of its kinetic energy upon scattering by a hydrogen nucleus, but hydrogen has a significant absorption cross section, as well. The deuterium atom, at twice the mass of hydrogen, is the next best slowing down agent. Since it already has an extra neutron in its nucleus, it has a very low probability of absorbing a neutron. Deuterium is the best moderator due to it having the lowest neutron absorption cross section thus, heavy water is a much better moderator than light (ordinary) water. Using heavy water, it is possible to sustain the chain reaction using natural uranium. If light water is used as the moderator, as in LWRs, the fuel must then be enriched in ^{235}U .

The carbon atom is also relatively “light,” graphite is relatively inexpensive, and carbon is a relatively weak absorber of neutrons. Accounting for low absorption as one of the key performance metrics for a feasible neutron absorber material, graphite is the 2nd best neutron moderator material outperforming light water. However, since the carbon atom is larger than hydrogen, neutrons have to undergo many more scattering reactions than in water in order to reach thermal energies and this requires a much larger moderator-to-fuel volume ratio than in an LWR. A fission chain reaction can be sustained using natural uranium with graphite moderator, but the graphite needs to be of very high purity. This special type of graphite is called a nuclear-grade graphite. As a result, the core of graphite moderated reactors tend to be large. Heavy water reactor cores also feature a much larger moderator-to-fuel volume ratio than LWR cores. Beryllium is also a good moderator for thermal reactors, but its biological toxicity complicates its use.

A convenient measure of energy in neutron scattering events is logarithmic energy decrement, ξ . It's defined as the average change in the logarithm of neutron energy per collision.

$$\xi = \overline{\ln E_0 - \ln E} = -\ln \overline{E/E_0} \quad (7)$$

By integrating over the angles of scattering to get the average change in energy (meaning $E - E_0$), one gets the following integral equation:

$$\xi = \ln \left(\frac{E_0}{E} \right) = \left(\frac{1}{n} \right) \int_0^\pi -\ln \left(\frac{A^2 + 2A \cos \vartheta + 1}{(1+A)^2} \right) \frac{n}{2} \sin \vartheta d\vartheta \quad (8)$$

Here A is the atomic mass and ϑ is the scattering angle. An additional derivation concludes that for large A only, the solution may be approximated as:

$$\xi = 1 + \frac{(A-1)^2}{2A} \ln \left(\frac{A-1}{A+1} \right) \quad (9)$$

The average loss in the log of the energy per collision is a function of the mass of the target nucleus and is independent of incident neutron energy. Note that for fast neutrons, energy loss will occur on subsequent collisions until thermal energies are achieved. The larger the value of ξ , the fewer the number of collisions required for a fast neutron to achieve thermal equilibrium. This log energy decrement does not completely describe the performance of a material as a moderator. As neutrons interact with moderator nuclei, there also exists a probability for the moderator isotopes to absorb those incident neutrons. Note that the interaction is with the nuclei. So even if the atom that contains the nucleus of interest (with atomic mass A) is bound in some other form, such as U in UO_2 , the A to use in the equation is that of the target nucleus.

When A is small, or is bound in a molecule that can affect the way the nucleus recoils, (such as H in water!!!), then these approximations don't in fact work. However, these data can be both calculated and measured. Thus, these are tabulated and supplied in the [Table 1](#) below.

Table 1 Comparison of moderators.

Material	ξ	#Collisions to thermalize	MSDP	MR
H (in H_2O)	0.930	19	1.425	62
D (in D_2O)	0.510	35	0.177	4800
He	0.427	42	8.0E-6	51
Be	0.207	86	0.154	126
B	0.171	105	0.092	8.0E-4
C (as graphite)	0.158	113	0.061	221
U	0.0083	2192	—	—

The ratio of the macroscopic slowing down power (MSDP), $\xi\Sigma_s$, and the macroscopic absorption cross section, Σ_a , is referred to as the moderating ratio, MR.

Good moderators have large moderating ratios and MSDPs, shown in [Table 1](#).

Due to the fact that it takes more collisions to thermalize in graphite than in other moderators as illustrated in [Table 1](#), graphite-moderated cores, such as helium-cooled high temperature reactors, offer options for nuclear waste management taking advantage of the spectral shift to higher energies. These cores provide favorable environments for high actinide incineration due to increased fractions of higher energy neutrons ([Tsvetkov et al., 2008](#)).

Neutron economy in fast reactors

Alternatively, unlike thermal reactors, other fission reactors provide environments allowing high energy (fast) neutrons to induce fission reactions before they lose their kinetic energy through interactions with various materials on their path. Continuing focusing on the energy range of the neutrons that induce most of the fissions, such fission reactors are called fast reactors. Typically quoted energies of fast neutrons are from about 100 keV to the upper boundary of the fission spectrum that is about 15 MeV. Inherently, fast reactor configurations use material combinations that do not slow neutrons down significantly before they have a chance to be absorbed in the fuel and induce fissions. Liquid metals and gasses as coolants and steels as structural materials are the examples of materials for fast reactors ([Department of Energy, 2021](#)).

[Table 2](#) summarizes generic design characteristics of the contemporary thermal and fast reactors. Physics fundamentals and relevant parameters of fission reactions and their realization in thermal and fast reactors are presented in the subsequent subsections ([Hewitt and Collier, 2000](#); [National Research Council, 2008](#); [Nichols et al., 2008](#)).

In fast reactors, it is desired to utilize fast neutrons before they lose much of their kinetic energies. Hence, in the design of fast reactor systems, not only must parasitic neutron absorption be minimized, but also neutron moderation must be minimized while maximizing neutron-induced fission. The relatively low probability of interaction in the fast neutron energy range is partly compensated by increased fissile inventory in the reactor, partly compensated by the higher density of the fuel material itself, and partly compensated by the increased fissions in fertile materials, such as U^{238} , which is the majority of the actinide content in reactor fuel. Because more of the isotopic content of the fuel either undergoes fission directly, or transmutes to create additional fissionable material, a nuclear fuel cycle economy that uses fast reactors is the most efficient system for the effective utilization of uranium resources, due to the possibility of using the uranium stored in used nuclear fuel to be recycled while producing energy. It is also possible to adopt a strategy of burning (meaning depleting by fission) plutonium along with the uranium stored as tailings from enrichment plants. [Krepel and Losa \(2021\)](#) has additional information of breeding and burning in both thermal and fast reactors.

With these unique features, the energy potential of uranium increases significantly (by a factor of approximately 60) compared to light water reactors (LWRs). In addition, the radioactive wastes containing long-lived actinides created through transmutation become practically insignificant ([OECD-NEA, 2007](#)). For fuel discharged from fast reactors, it is estimated that these long-lived radioactive isotopes, which are responsible for the long term heat load that needs to be managed when the fuel is disposed, contribute less than 0.1% of all the fission products. Hence, a fast spectrum reactor can minimize the mining effort and also reduce nuclear waste storage space and the time required in a repository (by a factor of ~ 1000) to reduce the waste radiotoxicity to the level of the uranium that was originally mined from the ground.

There are several constraints restricting neutron economy characteristics. The configurations must include provisions for controlling fission chain reactions. They must be able to produce utilizable energy outputs that can be harvested and converted into expected energy products. They must be designed to meet financial constraints ([Hamon, 2021](#)).

Notably, lumping fuel into fuel elements increases chances for neutrons that are slowing down to escape parasitic absorption by capture resonances of U^{238} and other actinides. As a result, they thermalize and induce fissions, enhancing the capabilities of heterogeneous configurations versus homogeneous environments to induce and sustain the fission chain reaction. Using heterogeneous configurations is required for establishing a self-sustaining fission process when using natural uranium and moderators with increased parasitic neutron absorption, such as light water. If graphite is used, it is possible to use configurations that mix graphite and fuel together. This is true for all fuel configurations with relatively low fissile content (i.e., low enrichment). However, if fissile

Table 2 Contemporary thermal and fast reactor configurations.

Type	Reactor	Fuel	Moderator	Coolant
Thermal	PWR	UO_2 , 3–4.5% U^{235}	H_2O	H_2O
	BWR	UO_2 , 3–4.5% U^{235}	H_2O	H_2O
	CANDU	UO_2 , 0.7% U^{235}	D_2O	D_2O or H_2O
	AGR	UO_2 , 3–4% U^{235}	Graphite	CO_2
	HTGR	UO_2 , 6–10% U^{235}	Graphite	He
Fast	SFR	$(U,Pu)O_2$, U, UN, 10–20% fissile content	None	Sodium

content is relatively high, with increased fractions of U^{235} in the fresh fuel composition, for example, the design approach is the opposite. It becomes more favorable for enhanced neutron economy to mix fuel and moderator together into homogeneous configurations. This increases the chance for neutrons to interact with the fissile nuclei.

Neutron-induced fission chain reactions and neutron multiplication

The multiplication factor is defined as the ratio of the number of fissions in any one generation to the number of fissions in the immediately preceding generation, as in (10). When this factor is exactly equal to unity the number of fissions in each succeeding generation is, on average, constant, and a chain reaction initiated in the system will continue at a constant rate. A system in such a state is called “critical.” In a critical system, the neutron generation rate and neutron loss rate are equal. Neutrons can be lost via absorption in the system constituents or leakage out of the system. If the neutron production rate is larger (smaller) than the neutron loss rate, the system is “super (sub)-critical” and its fission rate increases (decreases) with time (Lynn, 2004).

$$k \equiv \frac{\text{number of neutrons in one generation}}{\text{number of neutrons in preceding generation}} \quad (10)$$

This so called “life cycle” definition leads to a conceptually simple evaluation of k , illustrated in Fig. 3. We can represent the number of neutrons in one generation as a function of the number in the preceding generation by defining the probabilities associated with the various processes involved for a fission neutron to eventually cause subsequent fission neutrons to be born. For thermal reactors, this relationship can be expressed as the product of the following terms:

1. The fast non-leakage probability, $P_{FNL} \equiv$ the probability that a fast neutron will not leak out of the core before slowing down.
2. The resonance escape probability, $p \equiv$ the probability that a neutron will not be absorbed during its slowing down process.
3. The thermal non-leakage probability, $P_{TNL} \equiv$ the probability that a thermal neutron will not leak out of the core.
4. The thermal utilization factor, $f \equiv$ the probability that if a thermal neutron is absorbed, it is absorbed in fuel:

$$f = \frac{\Sigma_a^{\text{Fuel}}}{\Sigma_a} \quad (11)$$

5. The reproduction factor, $\eta \equiv$ the number of neutrons produced by thermal neutron induced fission per neutron absorbed in the fuel.
6. The fast fission factor, ϵ a factor which characterizes the additional neutrons produced by fast neutron induced fissions:

$$\epsilon \equiv \frac{\text{total number of fission neutrons (fast and thermal)}}{\text{number of neutrons from thermal fission}} \quad (12)$$

Hence,

$$k = \eta f p \epsilon P_{FNL} P_{TNL} \quad (13)$$

which is known as the six factor formula. By this definition, k is sometimes referred to as the “effective” multiplication constant, k_{eff} , since it characterizes a finite system. In nuclear theory, k_{eff} is the parameter that models how far away from criticality a reactor is, given a set of fuel properties. It may be thought of as a variable used to artificially modify fuel properties to achieve a critical reactor. When k_{eff} is equal to 1.0000, the reactor would be critical with a given flux distribution and fixed fission rate. The portion of this multiplication constant that is not associated with the finite core is the “infinite” multiplication factor, k_{∞} , wherein:

$$k_{\text{eff}} = k_{\infty} P_{FNL} P_{TNL} \quad (14)$$

Fig. 4 schematically demonstrates the fission “life cycle” for 1000 fast fission neutrons within a thermal critical core (i.e., $k_{\text{eff}} = 1.0$), including the slowing down of neutrons, the processes whereby neutrons can be lost to leakage and the probability of inducing subsequent fission.

In order to maintain a self-sustained chain reaction in a reactor, a balance must be established between the rate at which neutrons are produced in the core and the rate at which they disappear. Neutrons disappear in two ways; they either leak from the surface of the core or are absorbed within its interior. The rates at which neutron leakage and absorption occur are governed by the size and composition of the core. Operational control of the process is through increasing or decreasing the neutron capture rates and sometimes also leakage rates. Control materials are moved into, or out of, the core as required to achieve this goal.

If we assume an infinite nuclear fuel system, and therefore assume the probabilities of fast non-leakage, P_{FNL} , and thermal non-leakage, P_{TNL} , are 1.0 (i.e., no fast or thermal leakage), Eq. (14) reduces to what is essentially a property of the material component of the thermal reactor:

$$k_{\infty} = \eta f p \epsilon \quad (15)$$

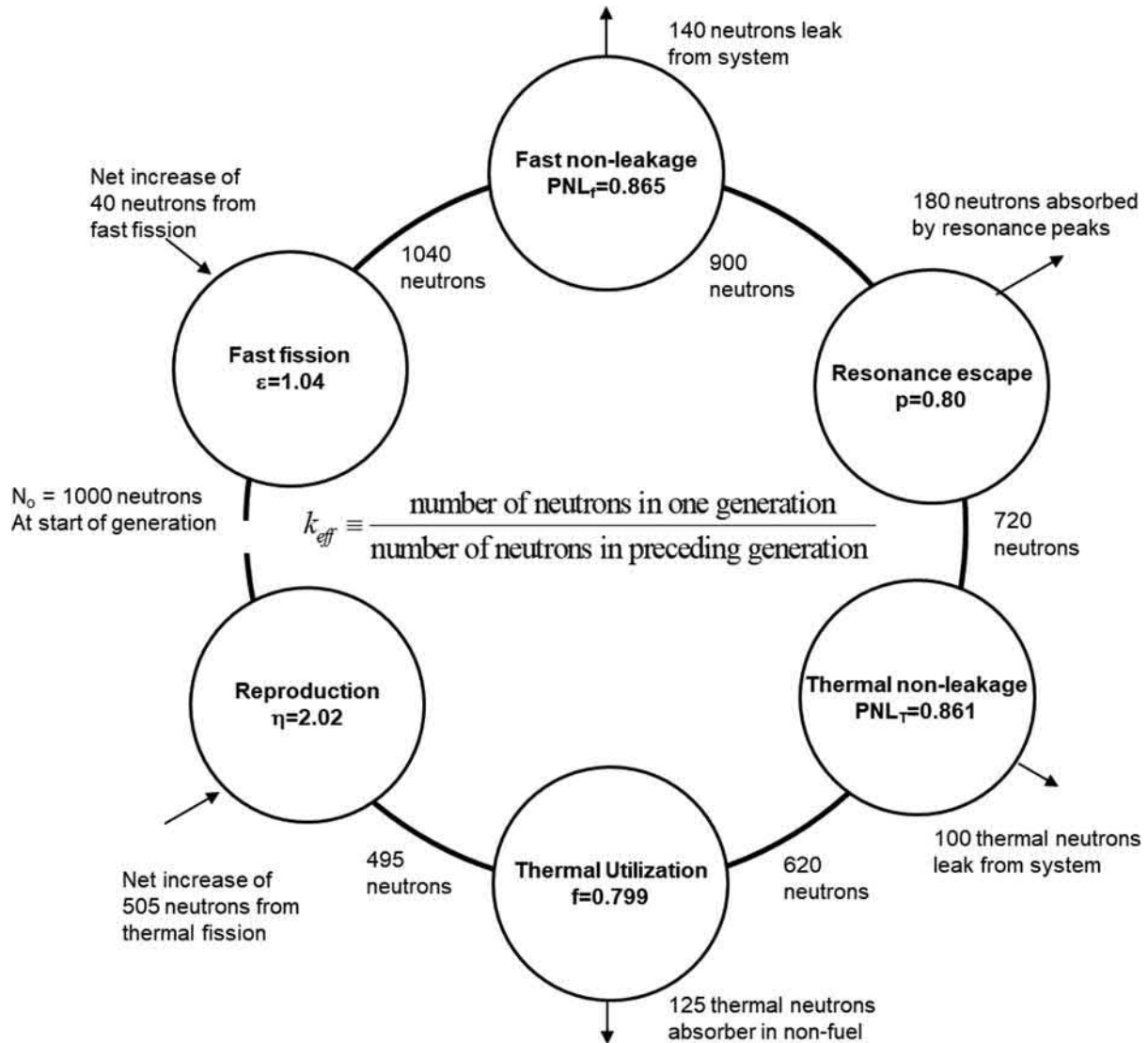


Fig. 4 Example of life cycle of 1000 neutrons.

Reactivity effects

“Reactivity” is a measure of the deviation of the multiplication factor from 1.0000, and is defined as:

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \quad (16)$$

The symbol ρ is commonly used to express reactivity, but this is not to be confused with mass density, which, unfortunately, uses the same symbol in nuclear engineering texts. It is also common practice to express $(k_{eff} - 1)$ as Δk and reactivity as $\Delta k/k$. In the following discussion, major reactivity effects in a BWR will be considered and the nature of the impact of these effects on k_{eff} will be discussed in terms of the 6-factor formula. Since actual changes in reactor conditions involve numerous, and often subtle, mechanisms and interactions, the validity of this sort of qualitative discussion is limited to first order effects.

U^{235} depletion

Depletion of U^{235} through burnup (that is, through fission), results in a decrease in k_{eff} due to a decreased thermal utilization factor, f :

$$f = \frac{\Sigma_a^{Fuel}}{\Sigma_a} \quad (17)$$

As the number of fissile atoms of U^{235} decrease (due to fission and capture), the macroscopic cross section, Σ_a^{Fuel} must likewise decrease, thereby reducing the thermal utilization factor.

Control density

The dominant effect of control insertion is the addition of B^{10} , which has a very large thermal neutron cross-section for the absorption reaction:



The addition of a strong absorber (B^{10}) increases the problem total macroscopic absorption cross section, thereby decreasing the thermal utilization factor, f , and therefore k_{eff} . Other neutron poison materials, such as AgInCd or Hf, are also used for reactivity control purposes. Different reactor types add a control material in different ways (Young, 2021). The most common method of inserting at least some neutron poisons into a reactor core is through the use of control rods (in the case of PWRs) or control blades (in the case of BWRs). For PWRs, though, the main reactivity control through the cycle is the use of soluble boron directly in the coolant (the concentration of the boron is adjusted throughout the cycle). Control rods are used there only for shutdown and for some power shaping. For BWRs, the main control mechanism is the use of control rods. Soluble boron injection is available for emergency shutdown purposes (Brown, 2021).

Moderator density or displacement

A secondary consideration in the insertion of control rods or blades in a core is that these devices occupy some space. That space was previously occupied by moderator. In general, light water reactors operate in an undermoderated state, meaning that removing moderator reduces the reactivity. So the insertion of control rods “inserts negative reactivity” both by introducing a thermal neutron absorber and by removing moderator from the core.

The same effect of lowering moderator density is also achieved by temperature increases in a PWR (at the same system pressure), or additional voiding of the water in a BWR. Such moderator reactivity feedback is an inherent aspect of the reactor’s response to abnormal events to return the reactor to a lower power state.

Burnable absorbers

Neutron absorbers that deplete (or “burn”) relatively quickly are commonly used in reactors to compensate for excess reactivity early in the cycle, and are referred to as “burnable poisons.” Gadolinium is used in BWR fuel and some PWR fuel for this purpose in the form of an oxide (Gd_2O_3) which is mixed in with the UO_2 in selected fuel rods at selected axial locations. Thin coatings of ZrB_2 within the fuel rods may also serve this same purpose. Burnable absorbers decrease k_{eff} in a manner similar to B^{10} , by decreasing the thermal utilization factor, f . As the fissile fuel concentration depletes with burnup, so does the concentration of the burnable absorbers so there is a reduced need to use control rods for long-term reactivity control.

Fuel temperature/Doppler effect

The resonance escape probability, p , may be expressed as the inverse of an exponential of the ratio of the resonance integral for the fuel to the scattering cross sections of the fuel, structure and moderators.

The resonance integral is essentially a method of integrating the absorption cross section of the fuel material in the with neutron energy in the epithermal region (the energy range with hash in Fig. 2). This is dependent on the mass density of the fuel material and the size (diameter) of the fuel rods. The fuel’s macroscopic absorption cross section in the resonance region is dominated for most reactors by the U^{238} , since it is the vast majority of the nuclei in the fuel material. Importantly, resonance cross sections, and therefore the resonance integral, are affected by temperature.

Because of increased resonance absorption arising from the Doppler broadening effect described earlier, the effective resonance integral will increase with increasing fuel temperature. Experimental measurements are represented by.

$$I(T) = I(300K) \left[1 + \beta \left(\sqrt{T} - \sqrt{300} \right) \right] \quad (19)$$

where $I(T)$ is the effective resonance integral at temperature T (Kelvin) and $I(300 K)$ is the integral at 300 K. β is a constant that depends on the nature of the fuel and on the radius of the fuel rod in a heterogeneous system. Therefore, as the temperature of the fuel increases, the resonance integral increases, which decreases the resonance escape probability. Through this mathematical model, a fuel temperature increase results in a lower neutron multiplication since the absorptions in U^{238} reduces the fraction of fission neutrons that reach thermal energies to cause fission, which is a negative reactivity. It is this Doppler effect that, in the

event of a power increase, and subsequent fuel temperature increase, results in an inherent prompt negative reactivity feedback to return the reactor to a normal state.

Plutonium buildup

When U^{238} absorbs a neutron, the resulting atom undergoes the following sequence of decays:



Pu^{239} is also a fissile isotope and therefore, it may then fission or undergo successive neutron captures to produce Pu^{241} , which is also fissile. In a reactor, as fuel burnup increases, parasitic neutron capture by the large concentration of U^{238} in the fuel results in plutonium quickly becoming a significant, and eventually dominant, contributor to the chain reaction. This is due to increased thermal utilization resulting from an increase in the macroscopic fission cross-section.

Production of plutonium is due primarily to resonance absorption, while the process that removes fissile material (i.e., fission) occurs primarily at thermal energies. As a consequence of this, the ratio of fissile Pu buildup to fissile U and Pu depletion, also called "conversion ratio," is strongly dependent upon the neutron energy spectrum. Any factor that tends to "harden the spectrum" (increase the fast flux/thermal flux ratio) will result in more buildup of Pu^{239} and Pu^{241} and less burnout of all fissile isotopes.

Reactor physics

Reactor physics is a mathematical means to determine the relationship between the characteristics of the reactor material (through expressions for k , as above in the four-factor formula of Eq. 15) and the finite geometry and arrangement, which essentially determines the fast and thermal non-leakage probabilities in the six factor formula of Eq. (14). These will be dependent on the total neutron flux, Φ , which is defined as the number of neutrons crossing a unit area per second. Reaction rates, such as the fission rate, and therefore reactor power, are proportional to the neutron flux.

Consider an arbitrary volume V within a medium containing neutrons. As time goes on, the number of neutrons in V may change if there is a net flow of neutrons in or out of V , if some neutrons are absorbed within V and if some neutrons are created within V . The equation of continuity is the mathematical statement of the obvious fact that all neutrons can be accounted for. This is summarized by the following:

$$\left[\begin{array}{c} \text{rate of change in} \\ \text{number of neutrons} \end{array} \right] = \left[\begin{array}{c} \text{rate of production} \\ \text{of neutrons} \end{array} \right] - \left[\begin{array}{c} \text{rate of absorption} \\ \text{of neutrons} \end{array} \right] - \left[\begin{array}{c} \text{rate of leakage} \\ \text{of neutrons out of } V \end{array} \right] \quad (21)$$

By invoking Fick's law to relate the rate of leakage from the volume to a proportionality constant, called the diffusion coefficient, D , and the gradient of the flux, the change in the neutron population (which is the flux divided by the velocity of the neutrons) is given by:

$$\frac{1}{v} \frac{d\phi}{dt} = k_{\infty} \Sigma_a \phi - \Sigma_a \phi + D \nabla^2 \phi \quad (22)$$

For the purposes of this explanation, we are only interested in the steady-state behavior of the neutron flux, so that the left hand side is set to zero. Upon rearrangement, Eq. (22) then becomes the homogenous, one energy group, reactor equation:

$$\nabla^2 \phi + \frac{(k_{\infty} - 1) \Sigma_a}{D} \phi = 0 \quad (23)$$

Solutions to Eq. (23) are well known eigenfunction solutions and the form of that solution is dependent on the nature of the geometry being analyzed (for instance, cylinder, sphere, parallelepiped, shown in Fig. 5) and the boundary conditions imposed.

Consider the five geometries in Fig. 5. These distinct geometries have distinct forms for their solutions. As an example, take the infinite slab geometry. This will have boundary conditions such that the flux is zero at the edges. Thus,

$$\phi\left(\frac{a}{2}\right) = \phi\left(-\frac{a}{2}\right) = 0 \quad (24)$$

It may also be noted that because of the symmetry of the problem, there can be no net flow of neutrons at the center of the slab. Since the current density is proportional to the spatial derivative of ϕ , we have:

$$\frac{d\phi}{dx} = 0, \text{ at } x = 0. \quad (25)$$

The general solution to (23) in one Cartesian dimension, with the above boundary condition is:

$$\phi(x) = C_1 \cos Bx \quad (26)$$

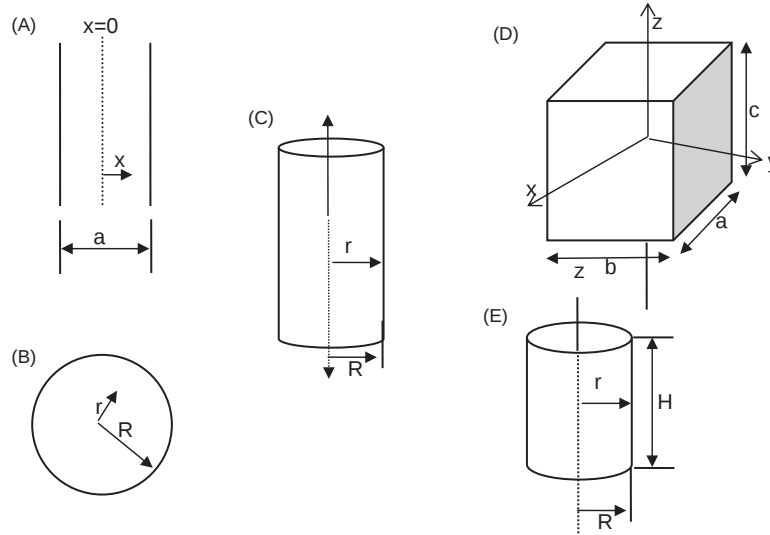


Fig. 5 Reactor core geometries. Reactor geometries: (a) the slab reactor; (b) the spherical reactor; (c) the infinite cylindrical reactor; (d) the parallelepiped reactor; (e) the finite cylindrical reactor.

Next, introducing the boundary condition given by Eq. (24) gives.

$$\phi\left(\frac{a}{2}\right) = C_1 \cos\left(\frac{Ba}{2}\right) = 0 \quad (27)$$

The non-trivial solution then demands that:

$$B_n = \frac{n\pi}{a}, n = 1, 3, 5 \dots \quad (28)$$

The various constants B_n are known as the eigenvalues and the corresponding functions, $\cos(B_n x)$, are called eigenfunctions. It can be shown that if the reactor were not critical, the flux would be the sum of all eigenfunctions, each multiplied by a function that depends on time. However, if the reactor is critical, all of these functions except the first die out quickly and the flux assumes the steady-state shape of the first eigenfunctions or fundamental, namely,

$$\phi(x) = C_1 \cos(B_1 x) = C_1 \cos\left(\frac{\pi x}{a}\right) \quad (29)$$

Note that the value of “ a ” has not yet been established. The reason is that, while the form has been established, no conditions on the reactor power have been imposed. To determine the magnitude of the flux, the power level needs to be considered as related to the flux by the relationship:

$$P = E_f \Sigma_f \int_V \phi(\mathbf{r}) d\mathbf{r} = E_f \Sigma_f \int_{-a/2}^{a/2} \phi(x) dx \quad (30)$$

where E_f = the recoverable energy per fission (about 193 MeV, or 3.1×10^{-11} J). Performing the integration for the slab geometry, we have the final form of the one-group bare reactor equation:

$$\phi(x) = \frac{\pi P}{2aE_f \Sigma_f} \cos\left(\frac{\pi x}{a}\right) \quad (31)$$

Other geometries are handled, in principle, in a likewise manner. See Table 3 for the parameters associated with the various reactor geometries. Note that this relationship holds for a “bare” reactor, meaning that no reflector around the outside (designed to reflect neutrons back into the reactor) has been included. For the reflected reactor, the equations would be modified through imposition of different boundary conditions.

Present day reactor physics design and analysis is using a much more sophisticated physics and mathematics models and computational tools than described above. For more details see Walters (2021) and Fiorina (2021).

Physics-driven design considerations for thermal and fast reactors

The nuclear design of a reactor involves defining the nuclear environment that will exist inside the reactor core. This phase of design work has a great bearing on thermal-hydraulic and mechanical analyses; hence, close coordination with these activities must be

Table 3 Bucklings and fluxes for critical bare reactors.

Geometry	Dimensions	Buckling	Flux	C_1	ϕ_{max}/ϕ_{avg}
Infinite slab	a	$\left(\frac{\pi}{a}\right)^2$	$C_1 \cos\left(\frac{\pi x}{a}\right)$	$\frac{1.57P}{aE_R \Sigma_f}$	1.57
Rectangular parallelepiped	$a \times b \times c$	$\left(\frac{\pi}{a}\right)^2 + \left(\frac{\pi}{b}\right)^2 + \left(\frac{\pi}{c}\right)^2$	$C_1 \cos\left(\frac{\pi x}{a}\right) \cos\left(\frac{\pi y}{b}\right) \cos\left(\frac{\pi z}{c}\right)$	$\frac{3.87P}{VE_R \Sigma_f}$	3.88
Infinite cylinder	Radius R	$\left(\frac{2.405}{R}\right)^2$	$C_1 J_0\left(\frac{2.405r}{R}\right)$	$\frac{0.738P}{R^2 E_R \Sigma_f}$	2.32
Finite cylinder	Radius R Height H	$\left(\frac{2.405}{R}\right)^2 + \left(\frac{\pi}{H}\right)^2$	$C_1 J_0\left(\frac{2.405r}{R}\right) \cos\left(\frac{\pi z}{H}\right)$	$\frac{3.63P}{VE_R \Sigma_f}$	3.64
Sphere	Radius R	$\left(\frac{\pi}{R}\right)^2$	$C_1 \frac{1}{r} \sin\left(\frac{\pi r}{R}\right)$	$\frac{P}{4R^2 E_R \Sigma_f}$	3.29

maintained (Fiorina, 2021). Safety and control requirements are closely tied to the nuclear design effort and must be considered for all phases of the fuel cycle. Space-dependent power distributions are needed in order to determine peak powers and temperatures for thermal-hydraulic and structural analyses (Walters, 2021). Reactor physics (also referred to as “neutronics”) calculations are needed to obtain, among other design parameters, the optimal moderator-to-fuel volume ratio, fissile fraction (enrichment), fixed and burnable absorber inventories required by the core design, optimal management of the fuel within the core (also referred to as “in-core fuel-management”), required reactor control systems (for both reactivity control and power shaping), attainable fuel burnup (amount of energy extracted per unit fuel weight), composition and radioactivity of the discharged fuel, reactivity coefficient due to change in components temperature and density, as well as the transient response of the reactor core during upset conditions (Martin, 2021).

The core of a fast reactor is smaller than that of a thermal reactor of comparable power. A thermal reactor is optimized at some particular fuel-to-moderator ratio, and any core smaller than optimum represents a less economic configuration. At optimum moderation, either the increase of moderator or the removal of moderator (for instance, due to steam formation in boiling), results in a decrease in the k_∞ of the core.

For a fast reactor, no moderator is needed; in fact, a moderator is normally excluded by design in order to maintain a fast spectrum. Since these reactors do not utilize thermal neutrons, the balance is optimized through the neutron leakage of the core. Whereas for thermal reactors one of the objectives of the reactor physics design is to find the optimal moderator-to-fuel volume ratio, for fast reactors the corresponding design objectives is to maximize the fuel volume fraction and minimize the coolant volume fraction. The higher is the amount of uranium and plutonium per unit core volume, the harder will be the neutron spectrum (higher will be the average energy of the neutrons) and the better will be the neutron economy more excess neutrons available for breeding or for incinerating nuclear waste (Tsvetkov, 2021). Because of the incentive to minimize the specific fissile inventory in the fast reactor, the spacing between fuel rods is different than in a light water reactor so that the pitch between rods is minimized in the lattice array of rods that makes up a fuel assembly (Young, 2021). A triangular lattice arrangement of the fuel rods intrinsically allows a higher fuel volume fraction than the square lattice typically used in LWRs. A triangular lattice arrangement intrinsically allows a higher fuel volume fraction than a square lattice. The higher fuel volume fraction minimizes fissile loading mainly by reducing reactor leakage. Hence, the triangular lattice or hexagonal structure has been universally selected for fast reactor design.

Nuclear data

The common approach of representing energy dependence of neutron-nucleus interactions consists of discretizing the energy dependence in a number of energy groups. During the fission reactions neutrons are emitted with an average energy of approximately 2 MeV. Neutron numbers are negligibly small above 10–15 MeV, making the maximum energy of interest in most nuclear fission reactors to be of about 15 MeV. Neutrons are then slowed down to much lower energies by collisions with the reactor components. The energy-dependent neutron spectrum in a reactor takes into account probabilities of absorption events, slowing down by elastic and inelastic collisions, and of leakage out of the system.

It is important to determine the rate at which various types of neutron-nuclear reactions occur within the reactor (Lucius and Marable, 1979). Because of the wide range of neutron energies and the very strong energy dependence of some neutron-nucleus cross-sections, this number of energy groups should be sufficiently high to capture necessary spectral details. Since the energy dependence of the cross-sections within these groups cannot be neglected, group average cross-sections have to be produced. The actual numbers of energy groups to be used in multigroup calculations vary according to the complexity of the calculations being performed and the accuracy desired.

All nuclear reactor calculations rely on the knowledge of the various relevant neutron-nuclear cross sections. The complicated dependence of such cross sections on neutron energy and angle of incidence combined with the large number of nuclides involved in nuclear reactor analysis implies that large databases are needed for storing nuclear data compilations. Such compilations have been accumulated over the past few decades from both experimental programs and theoretical calculations.

Nuclear cross section professionals have decided to consolidate and standardize all of the cross section information into the Evaluated Nuclear Data Files (ENDF) (Chadwick et al., 2006). The ENDF system contains both neutron and photon cross-section data. These data are stored in four Internet-accessible nuclear reaction databases:

- EXFOR, alias CSISRS (Cross Section Information Storage and Retrieval System), is a database containing experimental data. This data set contains unevaluated raw data from experimental measurements. The data presently included in the EXFOR database include compilations of experimental neutron-induced reaction data, charged-particle induced reaction data, and photon-induced reaction data. It covers more than 18,600 experiments and spans nearly all of neutron induced reactions experimental works worldwide.
- ENDF (Evaluated Nuclear (reaction) Data File) databases:
- ENDF/A data set contains both complete and incomplete sets of nuclear data as soon as they become available. For each isotope there may be more than one data set for a particular reaction, or there may be none at all for certain reactions of interest.
- ENDF/B is the main nuclear reaction database containing evaluated (recommended) data from the ENDF/B library (also JEFF, JENDL, BROND, ROSFOND and CENDL). It provides data in the ENDF format, covering all nuclides of practical relevance (393 in total) for neutrons up to 20 MeV and partly up to 150 MeV. It serves as principal input for neutronics calculations, including nuclear reactor design, national security, accelerators, criticality safety, shielding, radiation protection and detector simulations. The ENDF/B data set contains only complete, evaluated sets of nuclear data presented in a form that can be efficiently accessed and processed.
- RIPL (Reference Input Parameter Library) database contains input parameters for theoretical calculations of nuclear reactions.
- SIGMA (ENDF retrieval and plotting) database contains evaluated (recommended) nuclear reaction and decay data from ENDF/B, JEFF, JENDL, CENDL, ROSFOND libraries.

In addition to the nuclear data retrieval systems, there are two Internet-accessible bibliographical databases:

- CINDA (Computer Index of Nuclear (reaction) Data) database contains bibliographical neutron induced reaction information, including experimental, theoretical and evaluation works. It contains references to 275,000 reactions from 55,000 works worldwide.
- NSR (Nuclear Science References) database contains bibliographical nuclear physics information including more than 200,000 nuclear science articles, indexed according to content. It spans for 100 years of research, and currently covers 80 journals with about 3800 new articles added each year worldwide.

The ENDF data representation format and the ENDF/B data set are the accepted data presentation standard and the reference data set that are used worldwide for nuclear reactor data representation and for reactor analysis applications. Control over the ENDF formats has been retained by the US Cross Section Evaluation Working Group (CSEWG). The format specifications are published through the National Nuclear Data Center at the Brookhaven National Laboratory.

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See Also: Fissile and Fertile Fuels; Fission II; Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection; Nuclear Fission; Phenomenology of Neutron Interactions; Self-Sustaining Breeding in Advanced Reactors: Characterization of Natural Resources; Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors; Self-Sustaining Breeding in Advanced Reactors: Comparison of Fuel Cycle Performance.

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Relevant websites

- <https://www.gen-4.org/gif/>—Generation IV International Forum.
- <https://www-nds.iaea.org/>—International Atomic Energy Agency, Nuclear Data Services.
- <https://www.nndc.bnl.gov/>—National Nuclear Data Center.
- https://www.oecd-neo.org/jcms/mi_6530/jportal-data-bank-workspace—Nuclear Energy Agency, Data Bank.
- <https://www.world-nuclear.org/information-library.aspx>—World Nuclear Association, Information Library.

Principles of Design and Operation of Nuclear Power Reactors

David A. Hamon*, GE-Hitachi Nuclear Energy, San Jose, CA, United States

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Glossary

Anticipated operational occurrences Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power [10 CFR 50 Appendix A].

Loss-of-coolant accidents Loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system [10 CFR 50 Appendix A].

Reactivity A term expressing the departure of a reactor system from criticality; that is, of Keff value from 1 (1-Keff-1). A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Safety-related structures, systems and components Means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure: (1) The integrity of the reactor coolant pressure boundary; (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures [10 CFR 50.2].

Single failure A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions [10 CFR 50 Appendix A].

Abbreviations

ASME American Society of Mechanical Engineers

BPV Boiler and pressure vessel code

*Retired.

BTP Branch technical position
 CCF Common cause failure
 CFR US Code of Federal Regulations
 ECCS Emergency core cooling system
 EPRI Electric Power Research Institute
 EUR European utility requirements
 GDC General design criterion
 GDCs General design criteria
 I&C Instrumentation and control
 IAEA International Atomic Energy Agency
 IBC International Building Code
 LOCA Loss-of-coolant accident
 LWR Light water reactor
 NEI Nuclear Energy Institute
 NRC Nuclear Regulatory Commission
 NUREG US NRC technical report designation
 PRA Probabilistic risk assessment
 PSA Probabilistic safety assessment
 RG Regulatory Guide
 RISC Risk-informed safety class
 SAR Safety analysis report
 SRP Standard review plan (NNUREG-0800)
 SSCs Structures, systems and components
 URD Utility requirements document

Introduction

When designing a nuclear power plant, the plant designer has significant flexibility in choosing how the plant is configured and operated. General design criteria (GDCs) provide high level regulatory safety requirements for light water reactors that must be satisfied. The reactor designer establishes the specific plant/system configuration that results in compliance with the GDCs. In addition to regulatory requirements, utility groups in the United States and Europe have developed their own detailed design and operational requirements for new plant designs that expand on local regulatory requirements. Both safety and operational design requirements are discussed.

Utility design requirements

In the United States, the Electric Power Research Institute (EPRI) has developed the utility requirements document (URD) (EPRI, 2014) in cooperation with utilities and reactor vendors. The URD presents a clear, complete statement of Utility expectations for the design and operation of advanced light water reactors that is divided into three tiers. Tier 0 presents an executive summary of the URD. Tier 1 presents policy and top-tier design requirements, which also includes bounding environmental and seismic design requirements that will conservatively apply to most potential sites for new US nuclear power plants. Tier 2 contains thousands of specific requirements covering all major aspects of the plant design. Table 1 presents the list of chapter titles that comprise URD Tier 2.

A group of European utilities has developed a comparable document known as the European Utility Requirements (EUR) Document (EUR Organization, 2016). The EUR Document consists of a comprehensive nuclear power plant specification written by a group of the potential investors in electricity generation in Europe. The EUR document is structured into four volumes. Volume 1, Main policies and top-tier requirements, defines the major design objectives and presents the main policies that are implemented in the EUR document. Volume 2, Generic nuclear island requirements, contains all the generic requirements and preferences of the EUR utilities for the nuclear island. Volume 3, Specific nuclear island requirements, is prepared by a reactor vendor to describe how their plant design addresses the generic requirements in Volume 2. Volume 4, Power generation plant requirements, contains the generic requirements related to the power generation plant (i.e., turbine-generator systems). Additional background information on the development of the EUR document can be found in Roche (2001).

Some of the operational considerations found in the URD and the EUR include desired load following capabilities and operating design margin to Technical Specification limits for operation of the reactor core throughout a fuel cycle. In addition, while most

Table 1 Utility requirements document tier 2 chapter titles.

Chapter 1: Overall requirements
Chapter 2: Power generation systems
Chapter 3: Reactor coolant system and reactor non-safety auxiliary systems
Chapter 4: Reactor systems
Chapter 5: Engineered safety systems
Chapter 6: Building design and arrangement
Chapter 7: Fueling and refueling systems
Chapter 8: Plant cooling water systems
Chapter 9: Site support systems
Chapter 10: Man-machine interface systems
Chapter 11: Electric power systems
Chapter 12: Radioactive waste processing systems
Chapter 13: Main turbine-generator systems
Chapter 14: Tier 2 references

operating plants were originally designed for an operating life of 40 years, utility customers now expect new plants to be designed for an operating life of at least 60–80 years.

Regulatory requirements

The US nuclear regulatory process includes mandatory regulations (Bley, 2021; Bley and Malsch, 2021) which primarily appear in 10 CFR 50 and its various subparts, as well as guidance documents that include regulatory guides and the Standard Review Plan. The classification of structures, systems and components is an important part of regulatory guidance, as are requirements related to the defense in depth and diversity of the overall plant design and its control systems. An overview of these topics is provided below.

General design criteria

Appendix A of 10 CFR 50 defines a set of top-level requirements for light water reactors that are known as the General Design Criteria (GDCs). These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The GDCs are divided into six categories: Overall Requirements (GDCs 1–5), Protection by Multiple Fission Product Barriers (GDCs 10–19), Protection and Reactivity Control Systems (GDCs 20–29), Fluid Systems (GDCs 30–46), Reactor Containment (GDCs 50–57) and Fuel and Reactivity Control (GDCs 60–64). See Table 2 for a complete list of GDC requirements.

General Design Criterion 1 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. It also requires a quality assurance program to be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions.

GDC 2 addresses the design basis for protection against natural phenomena, such as earthquakes, tornadoes, hurricanes and floods. When selecting a proposed site for a nuclear power plant, the applicant for a permit must provide detailed assessments of the most severe external events and environmental conditions that might be experienced during the plant lifetime. Chapter 2 of the standard review plan (NUREG-0800) provides detailed guidance to the applicant on siting requirements.

GDCs 12 and 13 address instrumentation and controls for suppressing reactor power oscillations and for maintaining the reactor within its design limits during normal operation, anticipated operational occurrences and accident conditions to ensure the integrity of the reactor core, the reactor coolant pressure boundary, the containment and its associated systems.

Other GDCs address performance requirements for plant safety systems that provide primary safety functions during loss-of-coolant accident (LOCA) and other design basis accident events. The primary safety functions referred to in the GDCs include core cooling, containment heat removal and maintaining containment integrity to prevent releases of radioactive materials.

Regulatory guides and standard review plan

The NRC has issued over 200 regulatory guides that address what the nuclear regulatory commission (NRC) considers to be appropriate guidance for addressing various design issues in nuclear power plants. The application of these regulatory guides to a specific plant design is defined by NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.” NUREG-0800 (US NRC, n.d.) consists of several hundred individual sections, each under separate revision control, which correspond to the table of contents for a plant safety analysis report (SAR).

Table 2 General design criteria.

	<i>Criteria</i>	<i>Number</i>
I. Overall requirements:	Quality standards and records	1
	Design bases for protection against natural phenomena	2
	Fire protection	3
	Environmental and dynamic effects design bases	4
	Sharing of structures, systems, and components	5
II. Protection by multiple fission product barriers:	Reactor design	10
	Reactor inherent protection	11
	Suppression of reactor power oscillations	12
	Instrumentation and control	13
	Reactor coolant pressure boundary	14
	Reactor coolant system design	15
	Containment design	16
	Electric power systems	17
	Inspection and testing of electric power systems	18
	Control room	19
	Protection system functions	20
III. Protection and reactivity control systems:	Protection system reliability and testability	21
	Protection system independence	22
	Protection system failure modes	23
	Separation of protection and control systems	24
	Protection system requirements for reactivity control malfunctions	25
	Reactivity control system redundancy and capability	26
	Combined reactivity control systems capability	27
	Reactivity limits	28
	Protection against anticipated operational occurrences	29
	Quality of reactor coolant pressure boundary	30
	Fracture prevention of reactor coolant pressure boundary	31
IV. Fluid systems:	Inspection of reactor coolant pressure boundary	32
	Reactor coolant makeup	33
	Residual heat removal	34
	Emergency core cooling	35
	Inspection of emergency core cooling system	36
	Testing of emergency core cooling system	37
	Containment heat removal	38
	Inspection of containment heat removal system	39
	Testing of containment heat removal system	40
	Containment atmosphere cleanup	41
	Inspection of containment atmosphere cleanup systems	42
	Testing of containment atmosphere cleanup systems	43
	Cooling water	44
	Inspection of cooling water system	45
	Testing of cooling water system	46
	Containment design basis	50
	Fracture prevention of containment pressure boundary	51
V. Reactor containment:	Capability for containment leakage rate testing	52
	Provisions for containment testing and inspection	53
	Systems penetrating containment	54
	Reactor coolant pressure boundary penetrating containment	55
	Primary containment isolation	56
VI. Fuel and radioactivity control:	Closed systems isolation valves	57
	Control of releases of radioactive materials to the environment	60
	Fuel storage and handling and radioactivity control	61
	Prevention of criticality in fuel storage and handling	62
	Monitoring fuel and waste storage	63
	Monitoring radioactivity releases	64

Each standard review plan (SRP) section provides background information on the topic being addressed in the corresponding SAR section, defines applicable regulations and regulatory guides to be considered, and provides direction to the NRC reviewer regarding what to look for in this SAR section before approving the section. Some SRP sections make it mandatory to apply specific regulatory guides, while other SRP sections make recommendations to apply specific regulatory guides. When recommendations are

provided, the reactor designer may propose an alternative approach to addressing this topic provided it can be demonstrated that the alternative approach provides a comparable level of protection. Each SRP section also identifies important industry codes and standards that need to be considered during the design process.

The revisions of SRP sections and Regulatory Guides that apply to a plant are generally the most recent ones available 6 months prior to the licensing submittal that references them. Subsequent revisions of the SRP sections and Regulatory Guides do not automatically apply. If a plant owner chooses to apply newer revisions then additional NRC review will likely be required.

Classification of structures, systems and components

Several different types of classifications are applied to the structures, systems and components (SSCs) of a nuclear power plant.

Seismic classification

The SSCs of a nuclear power plant that are designated as Seismic Category I must be designed to withstand the effects of the SSE and remain functional. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 shall apply to all activities affecting the safety-related functions of seismic Category I SSCs. Section C of RG 1.29 ([US NRC, 2016b](#)) provides a detailed list of the types of SSCs that are required to be designated as Seismic Category I.

Structures, systems and components that perform no safety-related function, but whose structural failure or interaction could degrade the functioning of a Seismic Category I item to an unacceptable level of safety or could result in incapacitating injury to occupants of the main control room, are designated Seismic Category II. These items are structurally designed to withstand the effects of an SSE.

SSCs in radioactive waste management systems are required to be designed in accordance with requirements from RG 1.143 ([US NRC, 2001](#)). Most other SSCs that aren't designated as either Seismic Category I or II are only required to comply with seismic requirements from the International Building Code (IBC) or corresponding local building codes.

Quality group

Quality Group A corresponds to the category of components that must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) boiler and pressure vessel (BPV) code. This category is limited to components that are part of the reactor coolant pressure boundary, except for the portions excluded by 10 CFR 50.55a(c)(2) that are included in Quality Group B below. This exclusion applies to components whose failure would not prevent the reactor from being shut down and cooled down in an orderly fashion with normal makeup and components that are or can be isolated from the reactor coolant system by two valves in series (with automatic closure of open valves).

The Quality Group B standards given in Table 1 of RG 1.26 ([US NRC, 2017](#)) should be applied to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves that are either (1) part of the reactor coolant pressure boundary but excluded from the requirements of 10 CFR 50.55a(c)(1) for reactor coolant pressure boundary components pursuant to paragraph (c)(2) of that section (as mentioned above in the section on Quality Group A), or (2) not part of the reactor coolant pressure boundary but part of the following:

- (a) systems or portions of systems important to safety that are designed for (i) emergency core cooling, (ii) post-accident containment heat removal, or (iii) post-accident fission product removal;
- (b) systems or portions of systems important to safety that are designed for (i) reactor shutdown or (ii) residual heat removal;
- (c) those portions of the steam systems of boiling-water reactors extending from the outermost containment isolation valve up to but not including the turbine stop and bypass valves, and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation
- (d) those portions of the steam and feedwater systems of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation; and
- (e) systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.

The Quality Group C standards given in Table 1 of RG 1.26 should be applied to water-, steam-, and radioactive-waste-containing pressure vessels; heat exchangers (other than turbines and condensers); storage tanks; piping; pumps; and valves that are not part of the reactor coolant pressure boundary or included in Quality Group B but part of the following:

- (a) cooling water and auxiliary feedwater systems or portions of those systems important to safety that are designed for (i) emergency core cooling, (ii) post-accident containment heat removal, (iii) post-accident containment atmosphere cleanup, or (iv) residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems), although Quality Group B includes portions of those systems that are required for their safety functions and that (i) do not operate during any mode of normal reactor operation and (ii) cannot be tested adequately;

- (b) cooling water and seal water systems or portions of those systems important to safety that are designed for the functioning of components and systems important to safety, such as reactor coolant pumps, diesels, and the control room;
- (c) systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure; and
- (d) systems, other than radioactive waste management systems, not covered by items (a) through (c) above that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body.

The Quality Group D standards given in Table 1 of RG 1.26 should be applied to water- and steam-containing components that are not part of the reactor coolant pressure boundary or included in Quality Groups B or C, but are part of systems or portions of systems that contain or may contain radioactive material.

Safety classification

The simplest safety classification is to divide SSCs into two groups, safety-related (see glossary definition) and nonsafety-related. The safety-related SSCs are then often divided into subclasses that correspond to quality groups A, B and C.

Note: The term “safety-related” being used in this chapter generally corresponds to the term “important to safety,” which is more common internationally.

Safety-related structures, systems and components are required in the US to conform to the quality assurance requirements of 10 CFR 50 Appendix B. Nonsafety-related structures, systems and components have quality assurance requirements applied commensurate with the importance of the item’s function. ASME Standard NQA-1 (ASME, 2009) is used as a basis for implementing a quality assurance program that satisfies the requirements of 10 CFR 50 Appendix B.

Regulators in some countries have developed their own unique safety classification systems (Williams, 2021), which share much in common with US safety classification systems but have their own unique features. For example, the plant function to provide spent fuel pool cooling following a design basis accident is not considered to be safety-related in the US due to it being a longer-term need, but it is treated as safety-related in many European countries.

In recent years, risk-based safety classification systems have been developed that can potentially be used to replace the traditional safety classification system. Regulatory Guide 1.201 (US NRC, 2006), “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” describes a method that the NRC staff considers acceptable for use in complying with the Commission’s requirements in 10 CFR 50.69 with respect to the categorization of SSCs that are considered in risk-informing special treatment requirements. This categorization method uses the process that the Nuclear Energy Institute (NEI) described in Revision 0 of its guidance document NEI 00-04 (NEI, 2005), “10 CFR 50.69 SSC Categorization Guideline,” dated July 2005. Specifically, this process determines the safety significance of SSCs and categorizes them into one of four risk-informed safety class (RISC) categories.

IAEA Safety Series Guide SSG-30 (IAEA, 2014), “Safety Classification of Structures, Systems and Components in Nuclear Power Plants,” also provides guidance for a risk-based safety classification system.

Defense in depth and diversity

When most operating plants were originally designed, safety systems were generally designed with multiple redundant trains in order to meet reliability goals for the system being able to perform its design function on demand. Failure data for individual system components was used to calculate the probability of successful system operation on demand. In addition, safety systems operating on different principles (e.g., steam-driven versus motor-driven pumps) were included in many plant designs. Design basis accident analyses at this time only required the plant designer to postulate a single active component failure when evaluating the acceptability of emergency core cooling system (ECCS) performance during loss-of-coolant (LOCA) events.

More recently, probabilistic risk assessment (PRA), also known as probabilistic safety assessment (PSA), techniques were developed to provide a more in-depth evaluation of overall plant safety. These assessments allowed the plant designer to determine the relative safety significance of individual plant components, functions and initiating event types. PRA techniques also made it possible to determine plant vulnerabilities to common cause failures of like equipment that might exist in multiple plant systems. If the PRA analysis shows specific initiating events as large contributors to overall plant risk of core damage, the plant designer can then revise the plant design to lessen the significance of the identified vulnerabilities.

With the development of digital instrumentation and control (I&C) systems, the vulnerability to common cause failure (CCF) has increased because digital I&C systems are potentially vulnerable to CCF caused by software errors or software developed logic, which could defeat the redundancy achieved by hardware architecture (Upadhyaya and Wood, 2021). Based on reviews of the Advanced Light Water Reactor Design Certification applications for designs that use digital safety systems, the US NRC has established the following four-point position on defense in depth and diversity in Branch Technical Position (BTP) 7-19 of NUREG-0800 (US NRC, 2016a) for new reactor designs and for digital system modifications to operating plants.

Point 1 “The applicant shall assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common-mode failures have adequately been addressed.”

Point 2 “In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) *using best-estimate methods*. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.”

Point 3 “If a postulated common-mode failure could disable a safety function, then a diverse means with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function. *The diverse or different function may be performed by a nonsafety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.*”

Point 4 “A set of displays and controls located in the main control room shall be provided for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in Items 1 and 3 above.”

The term “best-estimate methods” in Point 2 is more accurately referred to as “realistic assumptions,” which are defined as normal plant conditions corresponding to the event.

The above four-point position is based on the US NRC concern that software-based or software logic based digital system development errors are a credible source of CCF. In this guidance, common software includes software, firmware and logic developed from software-based development systems. Generally, digital systems cannot be proven to be error-free and, therefore, are considered susceptible to CCF because identical copies of the software-based logic and architecture are present in redundant divisions of safety-related systems. Also, some errors labeled as “software errors” (for example) can result from errors in the higher-level requirements specifications used to direct the system development that fail in some way to represent the actual process. Such errors place further emphasis on the need for diversity to avoid or mitigate CCF.

The industry response to these BTP 7-19 requirements has been to allow plant safety systems to be initiated by two independent and diverse I&C systems to the extent practicable. When this isn’t possible, the diverse I&C system activates alternative diverse systems that were not credited in the standard accident analysis.

IAEA safety guides

Two key top-level International Atomic Energy Agency (IAEA) safety documents related to design and operation of nuclear power plants are listed below. They are supported by a number of individual safety guides on various topics.

SF-1 fundamental safety principles

Safety Fundamentals Guide SF-1 (IAEA, 2006) establishes the fundamental safety objective, safety principles and concepts that provide the bases for the IAEA’s safety standards and its safety related program. Related requirements are established in the Safety Requirements publications. Guidance on meeting these requirements is provided in the related Safety Guides. The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation. The 10 principles discussed in this guide are as follows:

- Principle 1: Responsibility for safety
- Principle 2: Role of government
- Principle 3: Leadership and management for safety
- Principle 4: Justification of facilities and activities
- Principle 5: Optimization of protection
- Principle 6: Limitation of risks to individuals
- Principle 7: Protection of present and future generations
- Principle 8: Prevention of accidents
- Principle 9: Emergency preparedness and response
- Principle 10: Protective actions to reduce existing or unregulated radiation risks

SSR-2/1 (Rev. 1) safety of nuclear power plants: Design

Specific safety requirements publication SSR-2/1 (IAEA, 2016) establishes design requirements for the structures, systems and components of a nuclear power plant, as well as for procedures and organizational processes important to safety that are required to be met for safe operation and for preventing events that could compromise safety, or for mitigating the consequences of such events, were they to occur.

It is intended for use by organizations involved in design, manufacture, construction, modification, maintenance, operation and decommissioning for nuclear power plants, in analysis, verification and review, and in the provision of technical support, as well as by regulatory bodies.

Operational design requirements

In addition to the many safety design requirements, there are also operational design requirements that must be considered. Plant control systems and the systems they control need to be designed to minimize the probability of inadvertent plant trips that could

adversely affect plant availability and might also challenge plant safety systems. Plant designers typically design these systems so that inadvertent trips of pumps can be accommodated without a reactor scram. Plants that have installed at least 100% turbine bypass capacity have the capability to avoid a reactor scram for inadvertent turbine trips or generator load rejection events. A typical goal is to design the operational systems with sufficient redundancy so that the reactor scram frequency is less than once per year over the life of the plant, and with an overall plant availability of greater than 90%.

NRC recommendations for plant organizational and staffing requirements are provided in Chapter 13 of the Standard Review Plan, NUREG-0800. This SRP chapter also provides requirements for training of operating personnel, emergency planning, physical security, cyber security and fitness for duty.

Technical specifications for plant operation

Each nuclear power plant is required to establish Technical Specifications that are used to ensure the plant is operated in accordance with its established design basis.

Technical Specifications are required by 10 CFR 50.36 and include specific requirements in the following categories:

Safety limits are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity (Lutz and Williamson, 2021). If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission.

Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions.

Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence.

Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described above.

Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

Design for maintenance

Most advanced reactors are designed with sufficient redundancy to accommodate continued operation with one division of safety equipment out of service and still be able to meet all analytical design requirements with one additional single failure. This capability allows significant maintenance to be performed on-line in areas of the plant that are accessible during normal operation. By performing significant maintenance during normal operation, refueling outage lengths can be shortened, leading to increased plant availability.

To facilitate ease of maintenance, building and component layouts should be arranged to provide adequate access for maintenance personnel and easy access to lifting devices where needed. For equipment that is expected to require replacement during plant operation, a pathway that allows for removal and replacement should be provided.

See Also: Physics of the Fission Chain Reaction in Thermal and Fast Reactors.

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Pressurized Water Reactors

Jeffery A. Brown, Westinghouse Electric Company, Pittsburgh, PA, United States

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Glossary

AP1000 Advanced Plant 1000 MWe – a Westinghouse passive type Generation 3 PWR plant.

APR1400 Advanced Pressurized Reactor 1400 MWe - a South Korean (KHNP) PWR system.

ABWR The General Electric designed Advanced BWR reactor system.

BWR Boiling Water Reactor – A single coolant loop reactor type in which steam is generated directly inside the reactor vessel.

B&W Babcock & Wilcox.

CE Combustion Engineering.

CPR1000 A Chinese Generation 2 PWR plant.

CVCS Chemical Volume Control system – a system that controls the water inventory, boric acid concentration, and chemistry of the primary Reactor Coolant System.

DNB Departure from Nucleate Boiling – a PWR fuel clad design limit which represents the onset of film boiling resulting in a significant loss of heat transfer and increase in fuel clad temperature.

EPR Evolutionary Pressurized Reactor – a Framatome PWR system.

ECCS Emergency Core Cooling System.

EFWS Emergency Feed Water system – An auxiliary system used to supply secondary feedwater to the steam generators when the normal feedwater supply is not available.

ESBWR The General Electric Economic Simplified BWR reactor system.

HPSI Pump High Pressure Safety Injection – Pump used to supply cooling water immediately following a LOCA or rapid RCS cooldown event when the RCS is pressurized.

Letdown The process by which primary reactor coolant is extracted from the RCS in small amounts during operation and full pressure conditions.

LOCA Loss of Coolant Accident – An accident involving loss of primary cooling water to the reactor core.

LPSI Pump Low Pressure Safety Injection – Pump used to supply cooling water during shutdown or following a LOCA when the RCS is depressurized.

Once-through SG A type of steam generator for PWRs in which the primary coolant enters at the top of the generator and is passed downward straight tubes to transfer heat to the secondary coolant.

Pellet Power peaking factor The ratio of maximum fuel pellet power to the average fuel pellet power.

Pressurizer The component of the RCS that maintains primary system pressure during normal operation.

PWR Pressurized Water Reactor.

Radial Power peaking factor The ratio of maximum fuel rod power to the average fuel rod power.

RCP Reactor Coolant Pump – Use in PWRs to maintain flow of the primary coolant through the reactor vessel.

RCS Reactor Coolant System – The primary coolant circuit of a PWR that provides pressurized coolant to the reactor core.

Reactivity The extent of the deviation in the core effective multiplication factor (usually denoted as k_{eff}) from its value at a just critical state (at which state $k_{\text{eff}} = 1.0$ and the fission rate is constant). A positive (negative) reactivity implies that the ratio of

the rate of neutron production to the rate of neutron losses (k_{eff}) is larger (smaller) than 1.0 due to a change in a performance characteristic such as temperature or density.

Reactor trip A shutdown signal actuating the reactivity control system of the reactor to quickly release all control rods into the core, shutting down the reactor.

RHR Residual Heat Removal – An auxiliary secondary cooling system that does not utilize the steam generators to cool the primary coolant during shutdown or post-LOCA condition.

RT_{NDT} Nil ductility transition temperature; the temperature below which steel transitions from a ductile to a brittle fracture mechanism.

RWT Refueling Water Tank – a storage tank filled with borated water that is used to auxiliary cooling of the reactor during shutdown or follow a LOCA.

Scram A sudden shutdown of the self-sustaining fission reaction by the sudden drop of several control rods into the reactor core (i.e. Safety Control Rods Activation Mechanism). Also known as a reactor trip.

SIT Safety Injection Tank (also called Accumulators in some designs) – A tank containing pressurized borated water used for reflooding the reactor core immediately after a large break LOCA.

USNRC United States Nuclear Regulatory Commission.

U-tube SG A type of steam generator for PWRs in which the primary coolant is passed through U-bend tubes to transfer heat to the secondary coolant.

VVER A Russian type Generation 2+ PWR using hexagonal fuel assemblies.

Introduction

The Pressurized Water Reactor Design was first developed as the power propulsion plant for US submarines (Anand and Herring, 2021), but was later applied to civilian electric power production. See Maldonado and Brown (2021) for a more detailed history of the PWR. Today 297 of the 448 power reactors in the world are based on the Pressurized Water Reactor (PWR) design. Moreover, 78 of the currently planned¹ 94 power reactors will be PWRs. The reason for this has been the large accumulated operating experience, proven reliability, and relative simplicity of operation. PWR plants are currently operating with electrical power ratings up to 1400 MWe at a reactor power of 4000 MWth (Hamon, 2021a).

PWR overview

The Pressurized Water Reactor (PWR) nuclear steam supply system (Fig. 1) is characterized by a two coolant stage design (i.e., a primary and secondary coolant circuit) in which the primary water coolant is maintained above saturation pressure such that no significant amount of boiling occurs in the reactor core during normal operation. The steam for the turbine-generator is produced in a separate steam generator where the heat from the primary coolant is transferred to a secondary water coolant stage at lower pressure. The coolants of primary and secondary coolant stages are separated and never in direct contact. The steam produced in the steam generator is at or near saturation conditions. This concept has the major advantage of maintaining fully moderated conditions in the reactor core at all times while providing an additional barrier to the potential release of radionuclides during severe accident situations.

Fig. 1 shows the basic PWR system configuration. The reactor vessel containing the fuel may be connected to two, three, or four steam generators, each having a supporting reactor coolant pump on the return cold leg. The pressurizer is required to maintain and control primary pressure at an operating pressure of approximately 2200 psi (15.2 MPa). Table 1 shows typical PWR system operating parameters.

Reactor core and fuel assemblies

In a PWR the reactor core water coolant/moderator is maintained at nearly equal density inside the reactor vessel thus enabling nearly optimal moderated conditions everywhere in the core during normal operation. Maintaining a fully moderated core results in a core power distribution that is relatively uniform² with lower power peaking factors. Since the core power is limited by margin of the hottest rod to fuel design limits, such as Departure from Nucleate Boiling (DNB) and fuel centerline melt temperature, lower power peaking factors mean that the core can be operated at a higher power density than a BWR without exceeding the fuel design

¹As of end of 2019.

²Relative to a Boiling Water Reactor (Hamon, 2021a, b) core in which the water density becomes lower with elevation making the upper part of the core undermoderated.

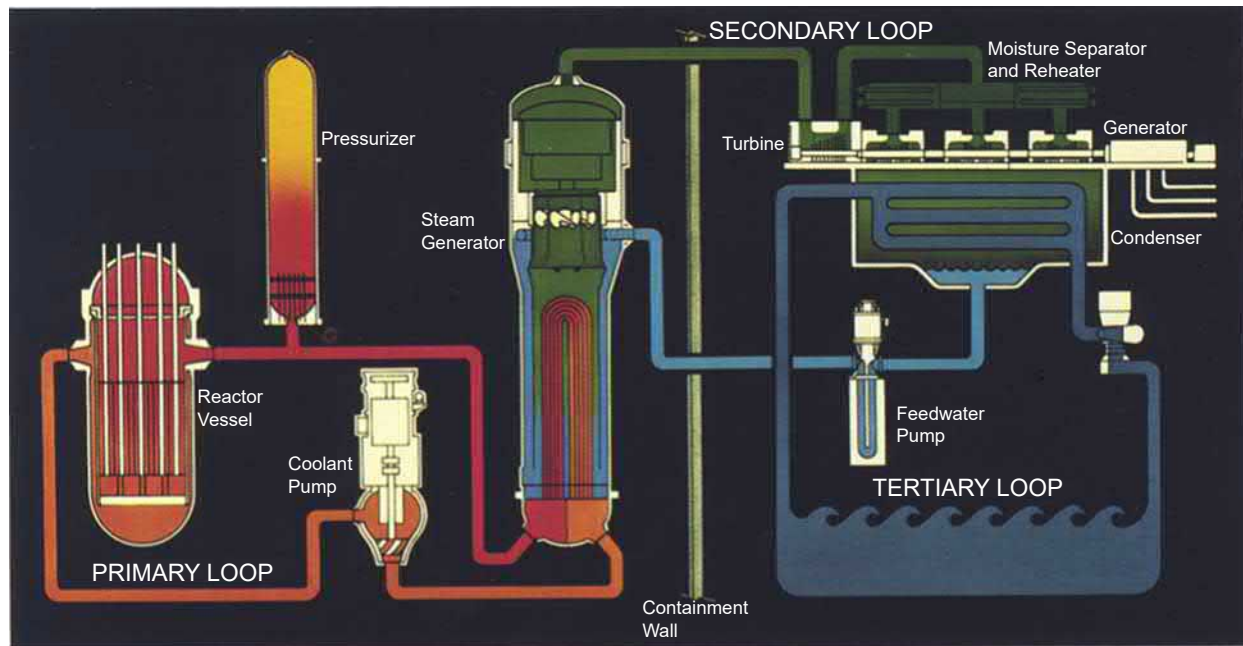


Fig. 1 PWR System Configuration. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

Table 1 Typical Westinghouse 4-loop PWR system parameters.

Total heat output, MWt	3411
Heat generated in fuel, %	97.4
Nominal system pressure, psia (bars)	2250 (155)
Total RCS Volume, ft ³ (liters)	12,064 (341,616)
Total coolant flow rate, lb·hr ⁻¹ (kg sec ⁻¹)	~138.4 × 10 ⁶ (17,438)
Gpm (liters sec ⁻¹)	359,200 (22,662)
Reactor Coolant temperature	
Nominal inlet, °F (°C)	557.5 (291.9)
Average rise in vessel, °F (°C)	61.0 (33.9)
Outlet from vessel, °F (°C)	618.5 (325.8)
Steam Pressure, psia (bars)	874 (60.3)
Steam Temperature °F (°C)	531 (277)
Steam Flow, lb/h (kg sec ⁻¹)	14.9 × 10 ⁶ (1877)
Plant MWe/MWt efficiency (%)	33.7
Equivalent core diameter, ft. (cm)	11.06 (338)
Core length, between fuel ends, ft. (cm)	12.0 (365.8)
#Fuel rods/assembly ^a	264
Fuel weight, initial uranium in core, kg	81,639
Number of fuel assemblies	193
Total # fuel rods in core	50,952
Typical number of feed assemblies ^b	77
Average power density (w/cc)	102.9
Average fuel power density (kw/kgU)	38.8
Typical feed Uranium enrichment, wt% U235 ^b	4.70
Average discharge burnup, GWD/MTU ^b	45–50
Typical maximum fuel rod radial power peaking factor	1.50
Typical maximum fuel pellet power peaking factor	1.80

^aSee (Young, 2021b) for details of fuel rod design.

^bAssuming 18 month refueling cycle.

Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

limits, thus allowing a smaller core size for a given total power. The fuel is also limited by peak rod burnup,³ which is meant to protect against fuel rod failure due to fuel clad corrosion during normal operation and anticipated operational occurrences (Young, 2021a). Operation with lower power peaking factors means that the fuel can be discharged at a higher average burnup, thus greatly improving the fuel utilization and fuel cycle cost relative.

The PWR core (Fig. 2) is comprised of fuel assemblies configured in a semi-cylindrical shape that approximates an optimal geometry for nuclear criticality.⁴ The fuel assembly consists of the nuclear fuel pellets contained within fuel rods that are bundled into fuel assemblies. The UO₂ fuel pellets, typically enriched to 4–5 wt% U235, are typically 0.74–1.0 cm in diameter and approximately 1.3–2.5 cm long. A 3.66 m stack of these pellets are loaded into a sealed zirconium alloy fuel rod tube that is typically 20–30 cm longer than the stack of fuel pellets in order to provide space for the accumulation of gaseous fission products released during operation. Some of the fuel rods may contain a burnable absorber (Young, 2021b) such as boron, gadolinia,⁵ or erbia in the form of an admix to the UO₂ or in a coating applied to the pellet that provides extra neutron absorption for a portion of the cycle to assist in offsetting the core excess reactivity during the early portions of the reactor operating cycle.

The fuel assemblies (Fig. 3) combine up to 264 fuel rods in a 4 m tall steel-Inconel-zircaloy “skeleton” to provide structural integrity of the active core and also to serve as the container for loading and unloading fuel from the reactor core. The fuel assembly skeleton consists of the bottom nozzle, fuel assembly grids, top nozzle as well as various thimble tubes to provide structural support and to guide the control rods and in-core instrumentation into the core.

The steel bottom nozzle contains support legs that fit into the alignment features in the bottom core support plate. It also contains a flow plate to evenly channel the coolant flow distribution to the coolant channels between the fuel rods. The bottom nozzle may also contain a debris filter to catch and hold metallic debris (usually material that was inadvertently introduced during the refueling outage) that could damage the fuel clad during operation.

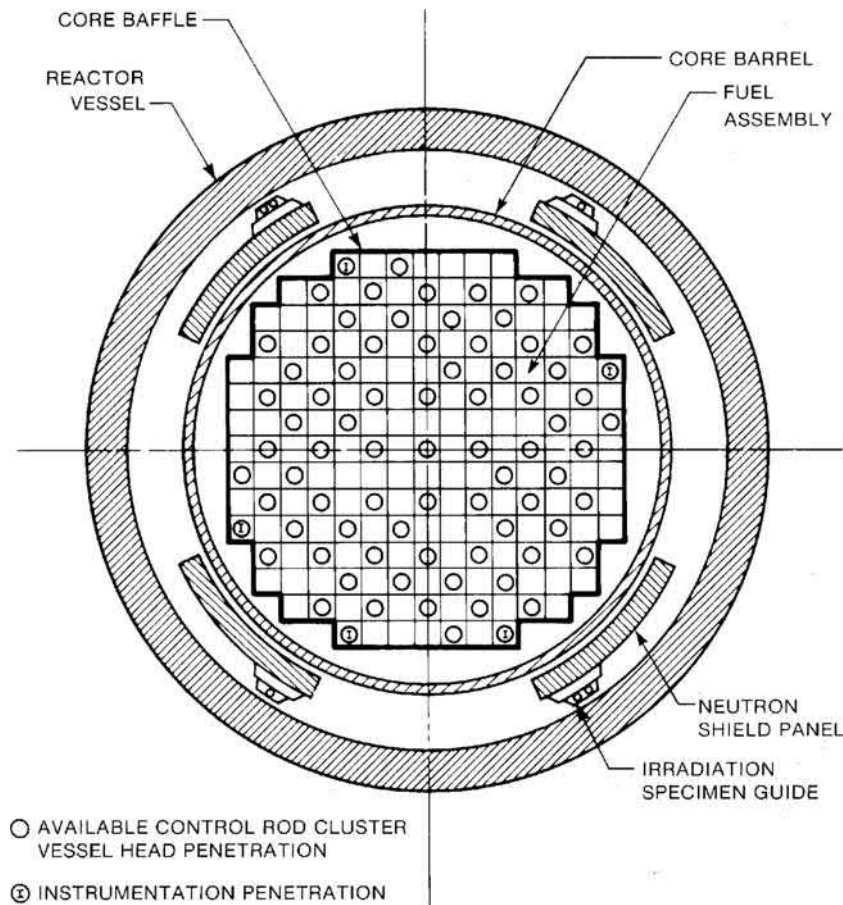


Fig. 2 PWR reactor core cross section. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

³Burnup is a measure of the amount of energy generated via fissions per unit weight of uranium loaded into the core.

⁴For establishing and sustaining a fission chain reaction.

⁵Gadolinium oxide (Gadolinia) or Erbium oxide (Erbia).

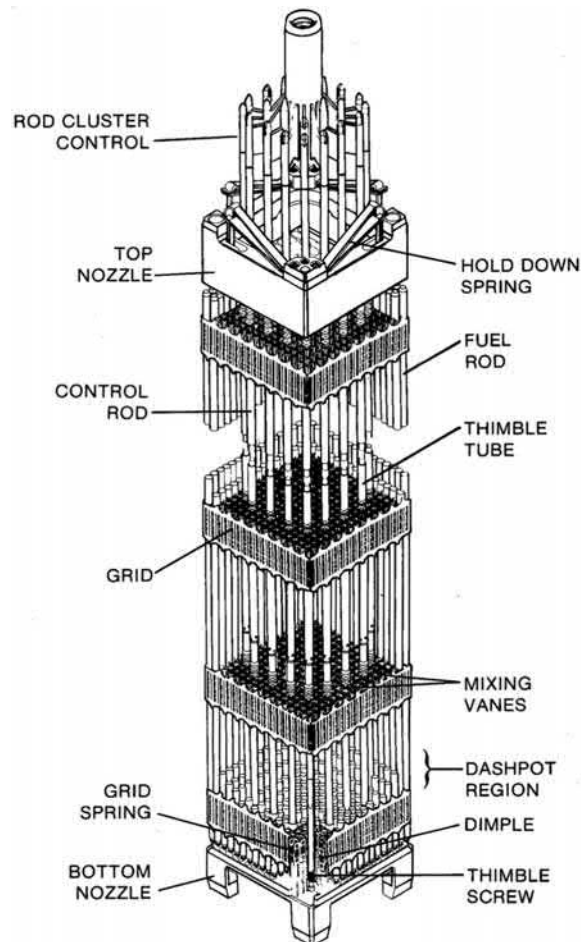


Fig. 3 PWR Fuel Assembly (Westinghouse). Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

The fuel assembly grids are the structures that hold the fuel rods into the tightly spaced array required for optimal nuclear performance and adequate cooling of the fuel rods. Each grid consists of strips that contain spring-like structures that allow insertion (and sometimes removal) of the fuel rod, but still have sufficient contact force to secure the fuel rod. The grids are attached to the guide thimbles at various elevations within the skeleton. The grids in the active core region are usually made of zirconium alloy in order to minimize parasitic neutron absorption. Some of these grids may have mixing vanes to thermally mix the coolant flowing upwards in the channels between fuel rods to improve heat transfer, reduce hot spots on the fuel rod clad, and increase DNB margin. In addition, some fuel assembly designs incorporate “intermediate flow mixing” grids that contain mixing vanes, but do not provide structural rod support.

The steel top nozzle contains features to interface with the top fuel alignment plate and maintain the fuel assembly spacing. The top nozzle also contains hold down springs that allow for fuel assembly thermal expansion during operation, while maintaining sufficient hold down force to prevent “liftoff” of the fuel assembly from the high velocity reactor coolant flow.

Approximately one third to one half of the fuel assemblies are replaced at each refueling shutdown which may occur at 12, 18, or 24 month intervals. The fresh assemblies are judiciously placed at locations in the core to assure good power sharing and maintain sufficient margin to peak rod power and temperature limits. The carryover fuel is also reshuffled as necessary during the refueling outage in order to meet fuel design limits and to maximize the energy extraction from the fuel prior to final discharge. **Fig. 4A** shows a typical core loading pattern. Some of the fresh assemblies may contain burnable absorber either within the fuel pellet or in discrete burnable absorber rods inserted into the assembly thimble guide tubes. The burnable absorber rods are judiciously placed within the fuel assembly (See **Fig. 4B**) to minimize the power peaking and to assure maximum fuel burnup by the end of cycle.

The reactor core also contains other components such as control rods and in-core instrumentation and, in some cases, discrete burnable absorber rods, neutron startup sources, and special isotope (e.g. Co-60, H3) generation capsules that are inserted in the guide thimbles for a single cycle of operation.

Control rods containing B4C, Ag-In-Cd, or hafnium are inserted into some or all of the guide thimbles in approximately half of the fuel assemblies to control reactor power. The control rod absorber in most PWRs is contained in tubes which are inserted into up to 24 of the guide thimbles in the assembly. The control rod fingers for Combustion Engineering PWRs are larger and are inserted

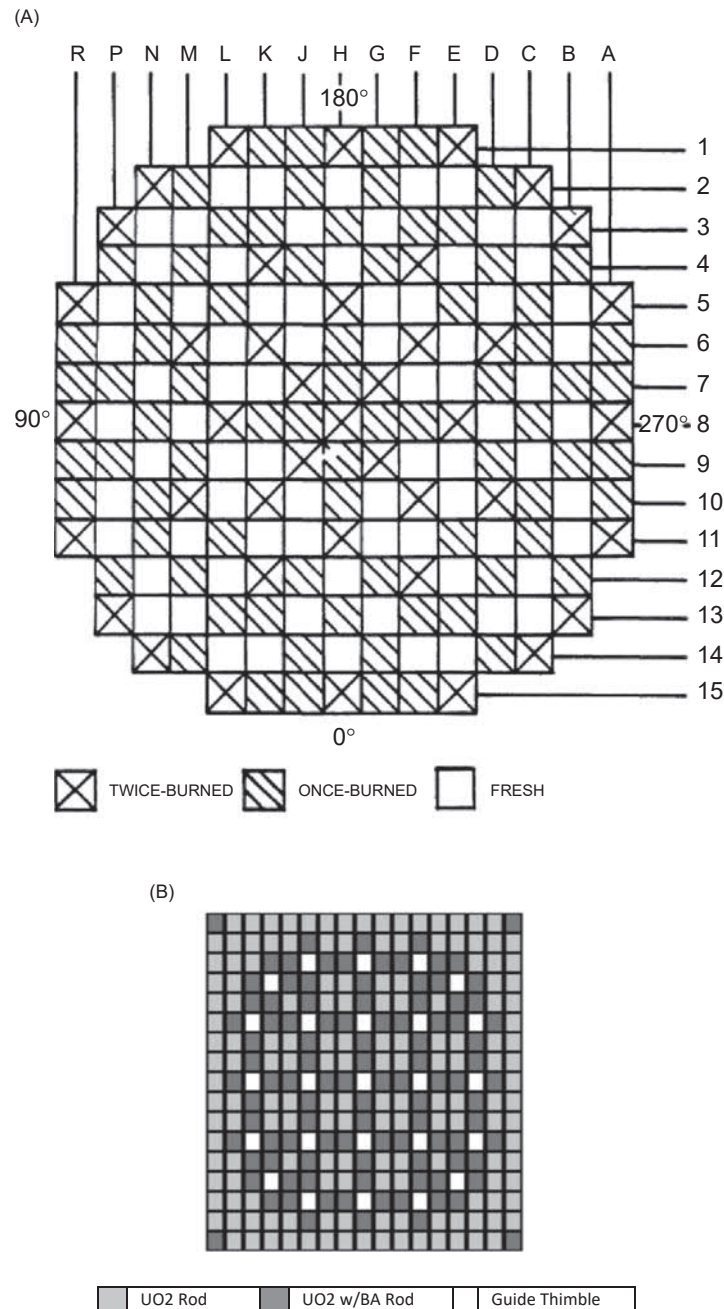


Fig. 4 Typical PWR Core and Assembly Fuel Loading Pattern. (A) Core Loading Pattern. (B) Assembly Rod Loading. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

into four or five large guide tubes that occupy a 2×2 rod cells in the assembly. A few PWR plants (e.g. Palisades Nuclear Generating Station in Michigan) still use the cruciform control blades similar to BWRs that are inserted into the gaps between fuel assemblies.

The control rods are assigned to rod groups (Fig. 5A) that are electrically controlled together based on their function during normal reactor operation. Some groups are designated as regulating groups that are inserted in a pre-defined sequence to control reactor power or to change core power or to offset core excess reactivity. Some groups are designated as shutdown groups that are only inserted during a reactor scram or during shutdown. In some cases, a control rod group may be designated as a power shaping rod group to control the reactor axial power distribution. Power shaping control rods may be composed of weaker neutron absorbers such as tungsten or Inconel. The control rods in a PWR are normally held above the active core by magnetic grips which release and allow the rods to drop into the core following a reactor scram or loss of electrical power. This is one of the primary engineered safety features of this reactor type (Lutz and Williamson, 2021).

In order to assure that the fuel rod integrity is maintained during normal operation and during anticipated operational occurrences, such as loss of load or loss of offsite power, two key power distribution parameters are monitored to assure they stay within

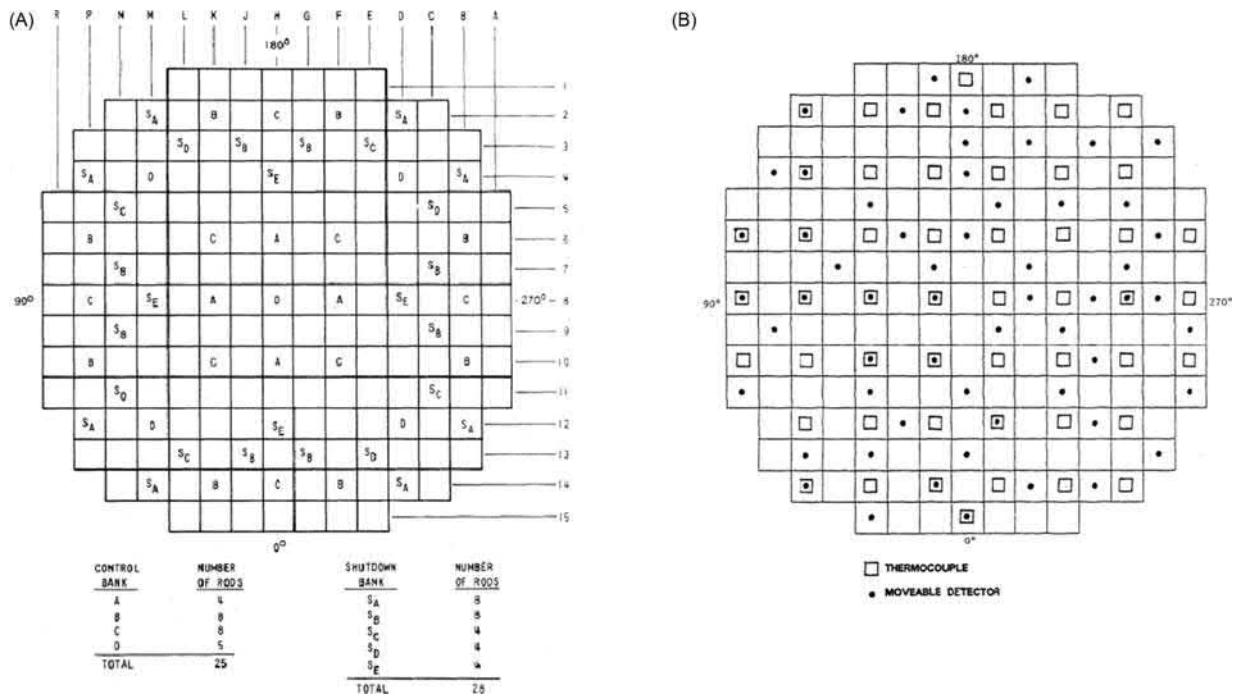


Fig. 5 Locations of Control Rods and Incore Instruments. (A) Control Rod Locations. (B) Incore Instrument Locations. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

predefined limits. These parameters are the peak rod power, which is related to the margin to DNB of the fuel rod clad and the peak pellet power, which is related to the margin to fuel centerline melt and to the peak clad temperature during a Large Break Loss of coolant accident (Rempe, 2021).

The monitoring of these limits is performed using in-core neutron flux detectors and thermocouples which are located in the center thimble tube of approximately a fourth of the fuel assemblies (Fig. 5B). The in-core flux detectors may either be fixed and always inserted during operation, or movable that are normally withdrawn but periodically (usually monthly) inserted to obtain a core flux map. Fixed in-core detectors depend on the beta decay of activated rhodium or vanadium or by gamma interaction to obtain a signal proportional to the local fission power level. Movable in-core instruments are usually small fission chambers that are sequentially driven into each of the “instrumented” assemblies on a monthly basis.

The data obtained from the in-core instruments are used to determine a detailed power distribution map for all locations in the core in order to confirm compliance with power peaking limits. Since most in-core detectors have a delayed response, or in the case of movable detectors are not continuously used, they are currently only used for monitoring core limits during normal operation and are not used by the reactor protection system to initiate a reactor trip. Reactor protection systems currently use the prompt responding ex-core neutron detectors, which are located immediately outside the reactor vessel, to determine instantaneous reactor power and core axial power distribution to initiate a reactor trip if maximum core power, power peaking, or power rate increase limits are exceeded.

Reactor vessel and internals

The PWR reactor vessel (Fig. 6) is the pressure vessel used to contain and support the reactor core. The reactor vessel is designed and manufactured to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. The vessel is fabricated from low carbon steel with a diameter up to 6.7 m (22 ft) and thickness of approximately 25 cm (10 in.). The inside surface in contact with coolant is clad with a minimum of 0.32 cm austenitic stainless steel to minimize corrosion from the boric acid⁶ containing coolant. The removable upper head of the reactor vessel contains a bolting flange employing studs and nuts. Hydraulic tensioning of the studs permits uniform nut loading. Two hollow, metallic O-rings form a pressure-tight seal in concentric grooves in the head flange. The reactor vessel is supported by steel pads integral with the coolant nozzles. The pads rest on steel base plates atop a support structure attached to the concrete foundation. An elongation gauge is employed to facilitate uniform loading.

Primary coolant inlet and outlet nozzles are located near the top of the vessel at an elevation above the top of the active core such that the core will remain covered with coolant in the event of a break in one of the coolant pipe legs. The removable top head

⁶Soluble boron in the form of boric acid is added to the coolant via the CVCS (See discussion later in this chapter) in order to provide core reactivity control.

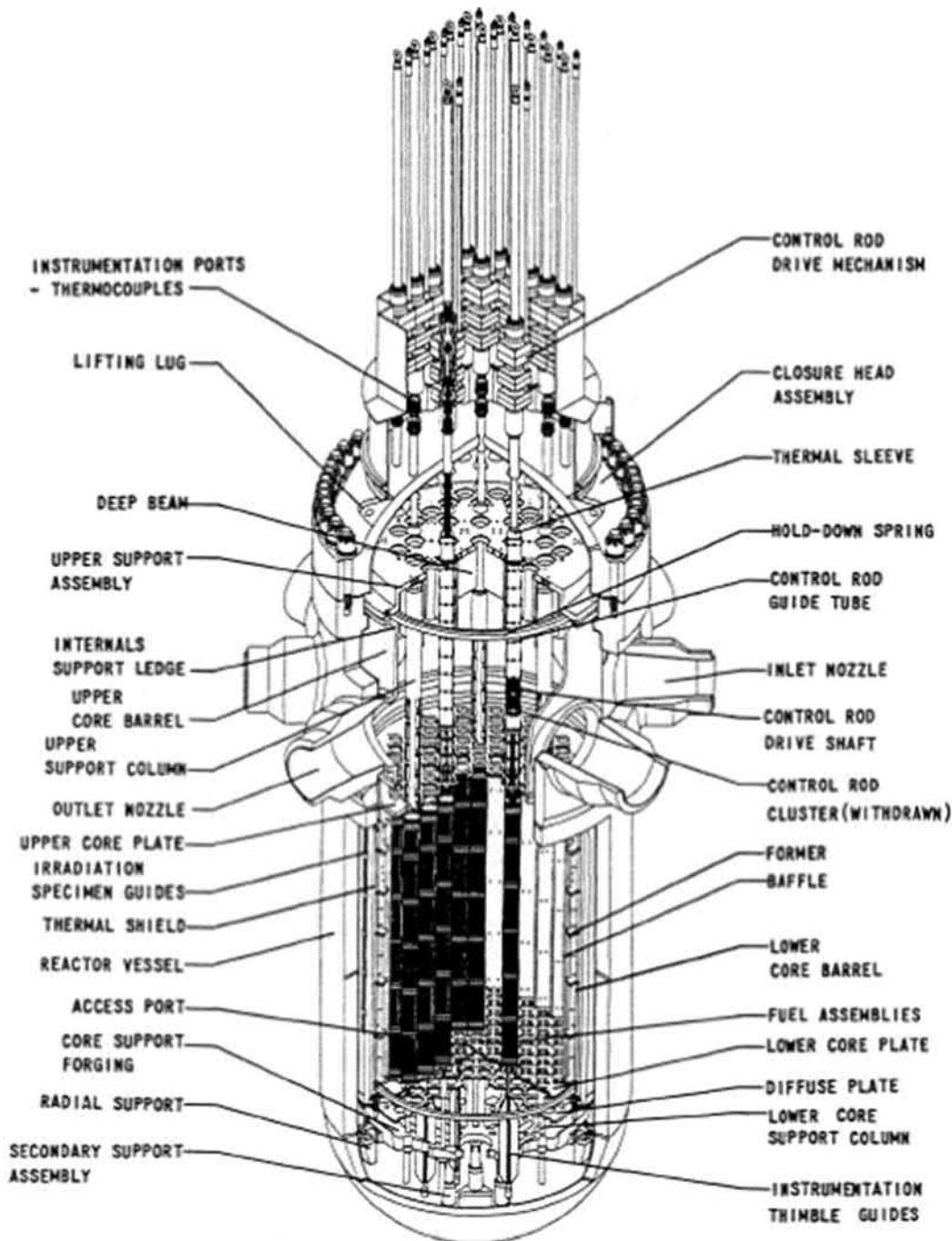


Fig. 6 PWR Reactor Vessel. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

contains penetrations for the control rods and in-core instruments. (In some PWR designs the in-core instrument enters through the bottom of the reactor vessels in order to have a straight path for insertion into the bottom of the fuel assembly.)

Reactor pressure vessel design and operation requires allowance for changes in steel properties due to fast neutron irradiation of the vessel wall surrounding the core. These changes in properties include increased ultimate and yield strengths, and decreased notch ductility, the latter being characterized as an increase in the nil ductility (embrittlement) transition temperature (RT_{NDT}) of the steel. The shift in RT_{NDT} from an initial temperature to a higher temperature is a function of the fast neutron fluence (nvt) and amount of shift sets limits on the maximum allowable pressure during heatup and cooldown for the vessel and may ultimately limit the operating life of the plant. (The allowable increase in RT_{NDT} is limited by a concern that over pressurization of the reactor vessel or the reflood of the reactor vessel following a Loss of Coolant Accident with the relatively cold water from the Emergency Core Cooling System could lead to vessel fracture if the RT_{NDT} temperature were high enough.)

The fast neutron fluence is predicted by neutron transport calculations that are benchmarked to “surveillance” capsules located between the neutron shield pads and the vessel wall directly opposite the center of the core. The surveillance capsules contain reactor vessel steel specimens obtained during vessel fabrication as well as other dosimeter materials, which are material samples that are used to obtain the nvt at these locations. Surveillance capsules are periodically extracted from the vessel and tested to determine the nvt (from the dosimeter materials) and the RT_{NDT} (from the vessel specimens) to assure that the predicted RT_{NDT} shift for the reactor vessel at End of plant Life will be within acceptable limits.

Reactor vessels in the early PWR plants were fabricated by welding three sections of curved plates into rings which were then welded into the vessel cylinders since there were no forges large enough to forge a cylinder of that diameter. However, this approach placed vertical welds at the location of the reactor core which raised a concern of weld failure due to prolonged exposure to high energy neutrons especially for plant lifetime extension beyond 40 years. Today, there are facilities capable of forging the large rings required to fabricate large PWR reactor vessels without beltline welds at the active core elevation that can achieve operating lifetimes of 80 years or longer.

The reactor internals are removable structures inside the reactor vessel that are required to support the core and to establish the flow path for coolant flow from reactor vessel inlet nozzle, through the fuel region, and out the reactor vessel outlet nozzle. The reactor internals are divided into two major structures: the upper internals and the lower internals. The upper internals include the top support plate, the upper core plate, support columns, and control rod guide structures the function of which is to align and locate the upper ends of the fuel assemblies, to protect withdrawn control rod clusters from lateral forces due to coolant cross-flow, and to guide the control rod cluster and associated drive shaft. The principal components of the lower internals are the lower core support plate and the core barrel whose functions are to support the weight of the core, to align and position the lower ends of the fuel assemblies and to reduce the radiation dose to the reactor vessel wall. Some plants have lower plenum instrument tubes to guide the in-core instrumentation into their proper core locations.

In addition to the upper and lower internals, there is the core baffle (or shroud) that encloses the active core, for which the main purpose is to channel the coolant flow into the active core region. This component experiences the highest exposure to neutron flux and gamma heating of any of the non-fuel components and thus it requires special attention when plant lifetime extension is being considered.

The reactor vessel is located in the reactor refueling cavity in the containment building floor which is flooded with water during refueling to provide radiation shielding to personnel in containment during movement of the spent fuel assemblies. During operation the reactor cavity is covered with a missile shield in order to protect the control rod drives during a severe accident.

Steam generator

Modern PWR plants have two to four steam generators to generate steam to drive the turbine and to cool the primary coolant and thus reactor core. In a PWR steam generator, the pressurized primary coolant is channeled through a large number of tubes that are immersed in a reservoir of low pressure secondary coolant to which the primary heat is transferred in sufficient amount to generate steam. The steam generators for most modern PWRs power plants come in three basic configurations: vertical U-tube in which the primary coolant tubes enter and exit at the bottom of the steam generator, vertical once-through in which the primary coolant enters on top and exits at the bottom, and horizontal steam drum generator in which the steam is generated in a generator tank and then collected into a secondary header tank similar to the configuration used in large fossil plants. All steam generator configurations, except for the vertical once-through, produce saturated steam. The vertical once through steam generator (used in B&W PWRs) is capable of producing slightly superheated steam which results in a small increase in plant thermodynamic efficiency. Most modern PWRs employ the U-tube design that allows for a relatively small compact steam generator and thus smaller containment building.

The U-tube steam generator (Fig. 7) consists of three sections: a lower plenum region, an evaporator section and a steam drum section. The lower plenum region is divided into two halves, one connected to the primary coolant hot leg and one connected to the primary coolant cold leg. Hot primary coolant enters the hot side and is diverted into U-tubes that sit in the evaporator section. Cool primary coolant return from the U-tube on the cold plenum side and is channeled into the primary cold leg. Cold secondary feed-water coolant is injected into an annulus region surrounding the tube bundle near the top of the generator and is preheated while it flows to the bottom of the evaporator section. The evaporator section is filled with secondary coolant that covers the U-tubes. Heat transfer from the primary coolant in the U-tubes generates steam in the secondary coolant. This steam then enters the steam drum section in the upper part of the steam generator that contains moisture-separating equipment, which delivers the high quality saturated steam required to minimize turbine blade damage.

The B&W PWR plants use two once-through steam generators capable of producing steam with a few degrees superheat. In the B&W design (Fig. 8), hot primary coolant enters the top of the steam generator and exits the bottom, thus allowing heat transfer from the hottest primary coolant to the saturated steam generated in the lower elevations of the steam generator thus resulting in a few degrees of superheat.

The tubes in most of the original PWR steam generators are fabricated from mill-stressed Inconel 600, which proved to be susceptible to stress corrosion cracking in the upper portions of the steam generator where rapidly fluctuating water levels induce vibration and rapid changes in temperature. As tubes failed, they were plugged to prevent leakage of primary coolant to the secondary system. Many plants have replaced their original steam generators with generators which have tubes made of heat treated Inconel 690, which has proved to be significantly less susceptible to cracking.

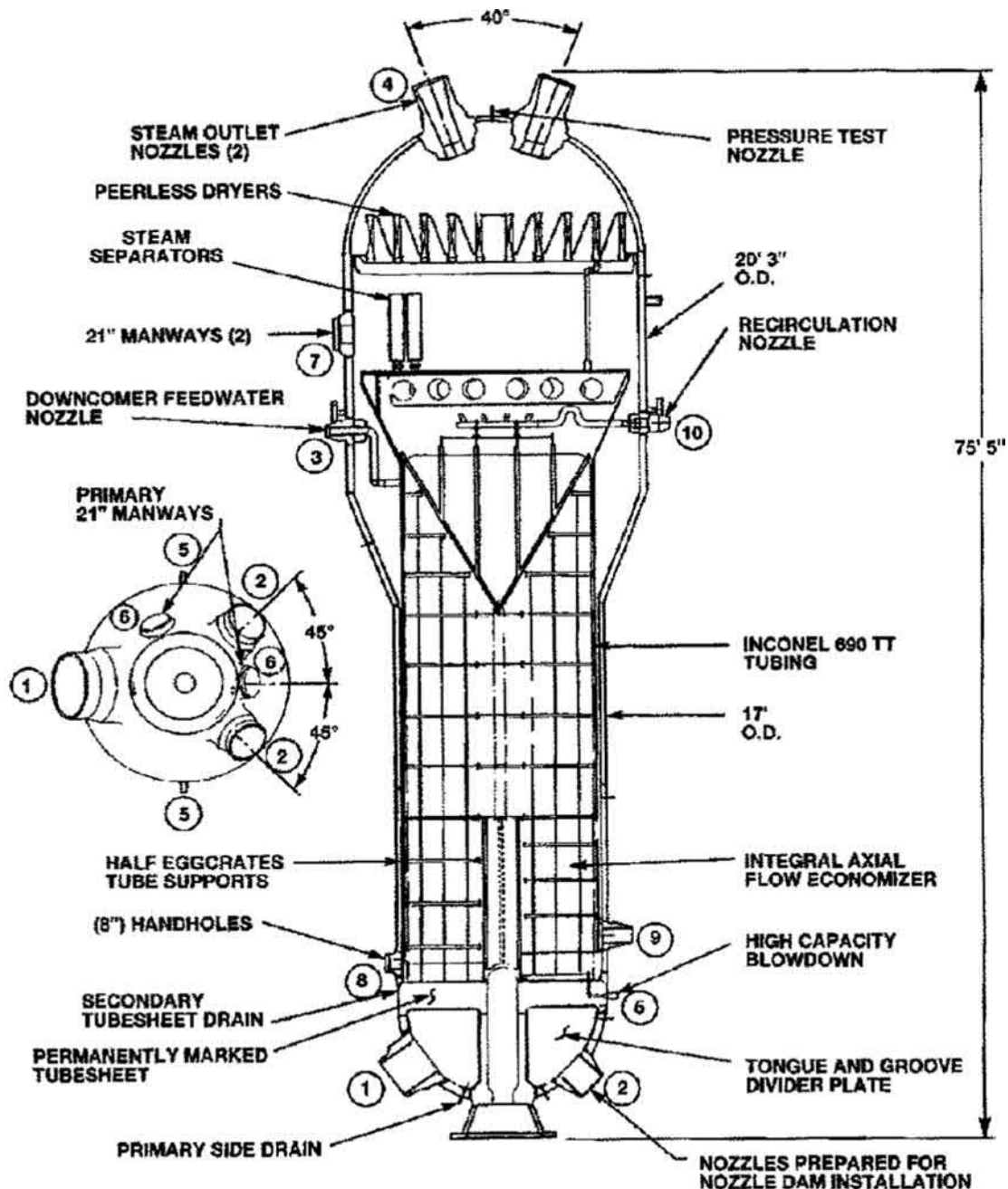


Fig. 7 PWR U-Tube Steam Generator (CE Type). System 80+ Design Control Document (1997) Available in the NRC Public Documents Archive.

Reactor coolant pump

A Reactor Coolant Pump (RCP) is attached to each primary cold leg to maintain the high primary flow rate needed to provide sufficient cooling of the PWR reactor core. In the typical PWR RCP (**Fig. 9**), reactor coolant is pumped by an impeller attached to the bottom of the rotor shaft. Each RCP on a typical 4-loop Westinghouse plant has an operating speed of 1189 rpm and delivers about 97,600 gpm (6158 l s^{-1}) of coolant to the reactor vessel. The coolant is drawn up through the bottom of the casing, up through the impeller, and discharged through the diffuser and an exit nozzle in the side of the casing. A diffuser and turning vane assembly converts velocity head from the impeller to pressure head, and directs the flow to the single discharge nozzle in the circular casing. A thermal barrier incorporated as an integral part of the diffuser and turning vane assembly limits the heat transfer from the casing into the seal and bearing regions. The impeller is driven by an AC induction motor mounted on top of the pump assembly.

A unique design problem for the PWR RCP is the design of the seals where the motor shaft passes through the pump casing. The seal must seal against a pressure head of $> 15.2 \text{ MPa}$ (2200 psi) while still allowing the pump shaft to rotate at a high speed. The

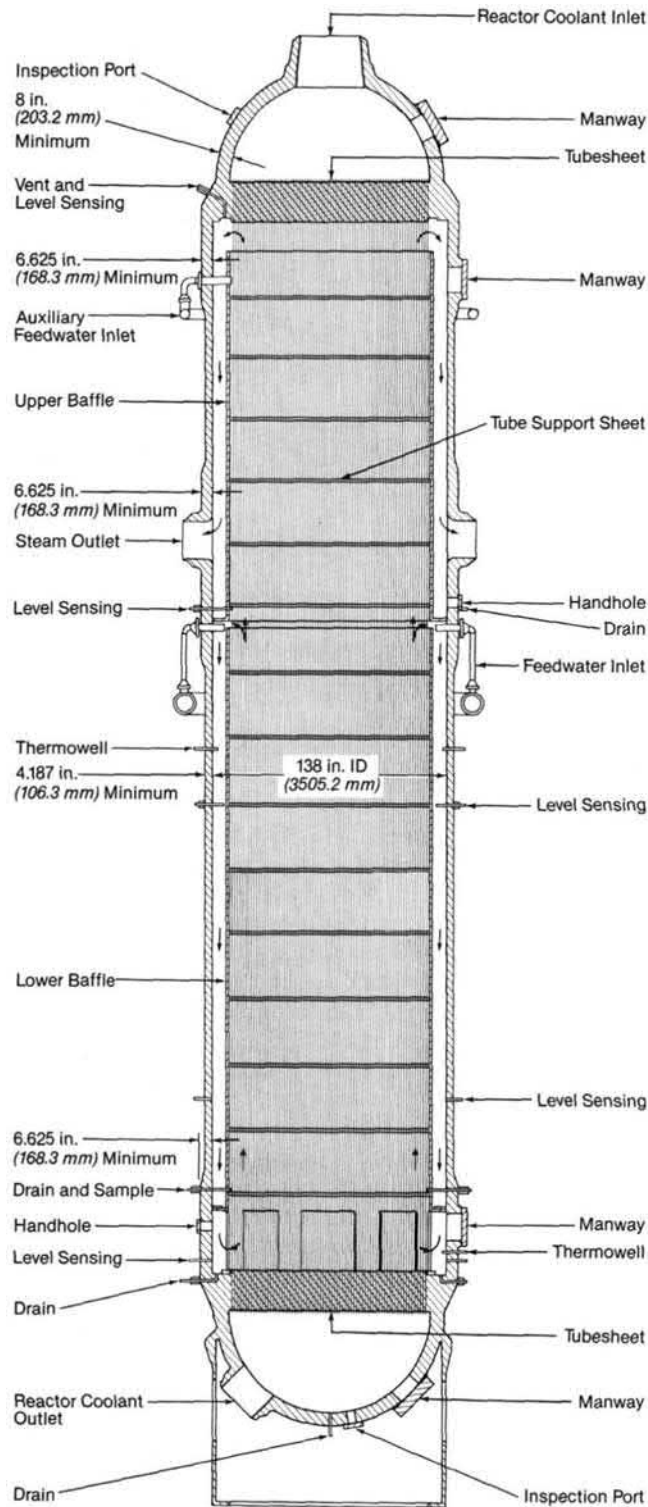


Fig. 8 B&W Straight thru Steam Generator. Babcock and Wilcox (1992) *Steam – Its Generation and Use*, 40th edn.

solution has been to allow a small amount of primary coolant leakage, about $\sim 0.23\text{--}0.68\text{ m}^3\text{ s}^{-1}$ ($\sim 1\text{--}3\text{ gal/min}$), through the seal to form a barrier but still allow for pump rotation. The Generation 3 Westinghouse AP1000^{®7} nuclear power plant employs canned

⁷AP1000 is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited.

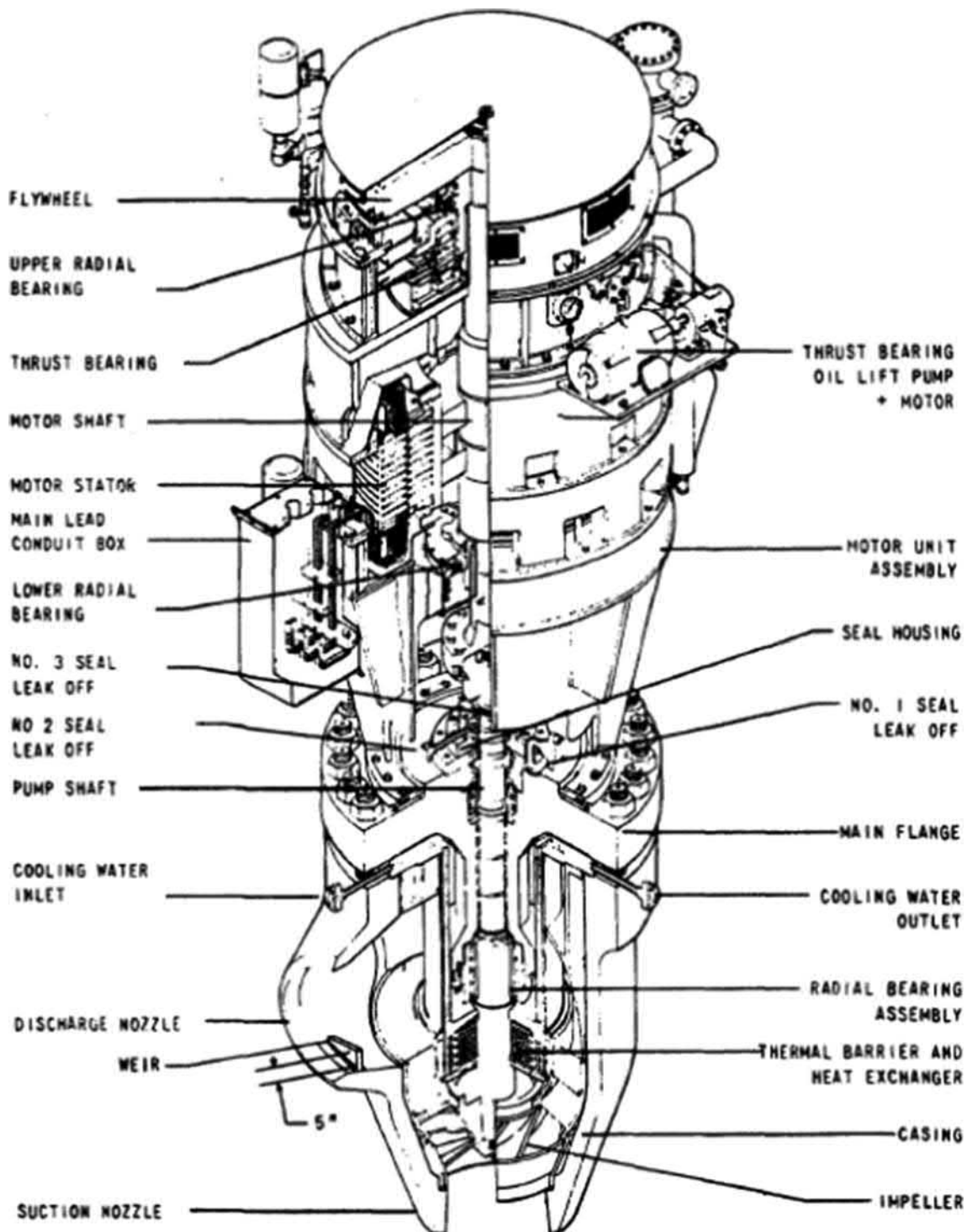


Fig. 9 Reactor Coolant Pump. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

RCP pumps similar to, but much larger than, those used in nuclear submarines. In the canned reactor coolant pump design there is no pump shaft as the rotor, and stators are completely sealed inside the pressure boundary while the induction motor coils are wrapped outside the pump housing (See Fig. 10). This design completely eliminates the problem of pump seal leakage.

Another consideration in PWR RCP design is that the impeller must have sufficient inertia such that in the event of loss of AC power, the decrease in Reactor Coolant System (RCS) flow will be slow enough to allow for sufficient reactor power decrease

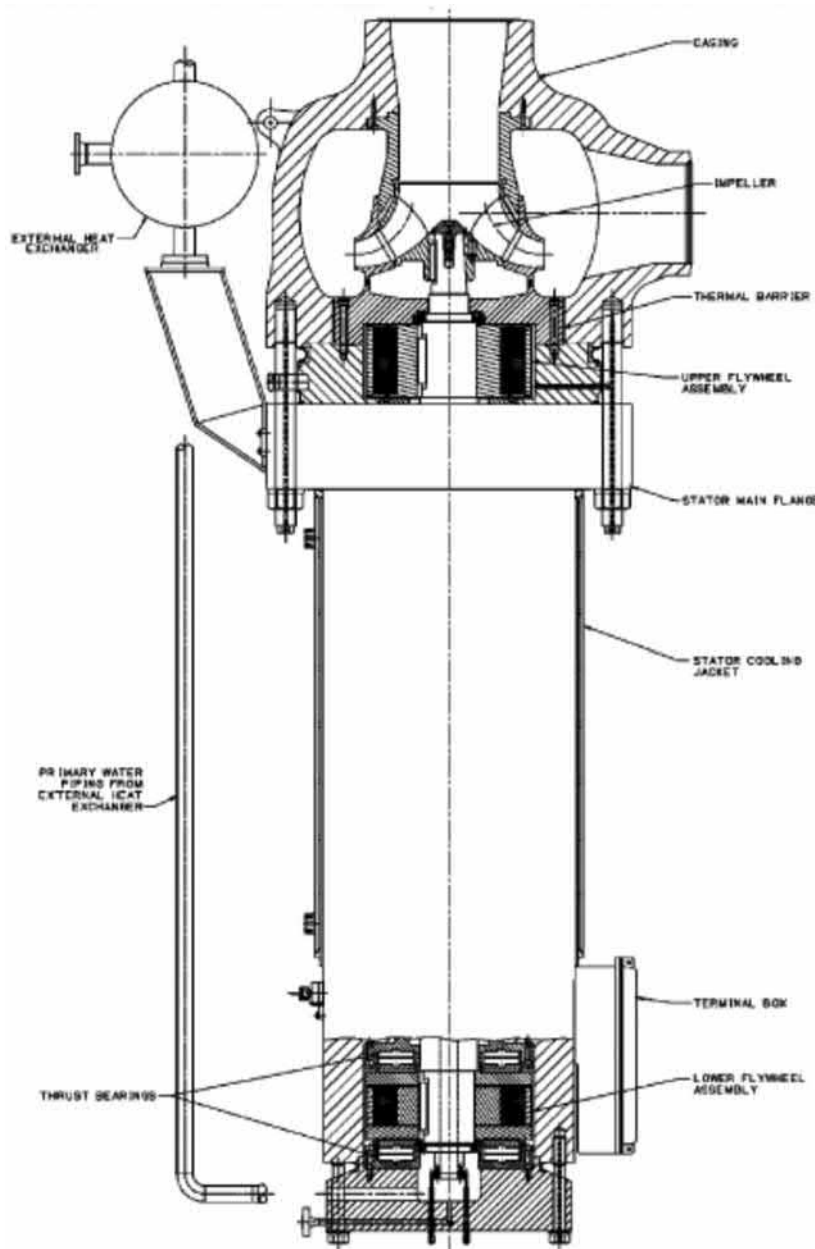


Fig. 10 AP1000® plant “Canned” Reactor Coolant Pump. Westinghouse Electric Company (2007) *Westinghouse AP1000 Design Control Document - Tier 2, Rev. 16*, (available in NRC Adams system Number: ML071580759 to ML071580937).

following a reactor trip to avoid DNB fuel failure. For the canned RCP pump design this often requires the addition of an extra flywheel since the pump impellers lack the inertia of the motor shaft.

At full capacity the Reactor Coolant Pumps delivers a net thermal power to the primary coolant of about 12 MWth. The RCP heat, in combination with the pressurizer heaters, is used to heat up the RCS to operating temperature ($\sim 288^\circ\text{C}/550^\circ\text{F}$) prior to bringing the core to critical.⁸

Pressurizer

The function of the pressurizer (**Fig. 11**) is to maintain the RCS pressure during steady-state operation and to limit pressure changes during system transients. This function is accomplished by maintaining and managing a steam bubble in the top $\sim 40\%$ of the

⁸A critical core is a core that sustains a fission chain reaction.

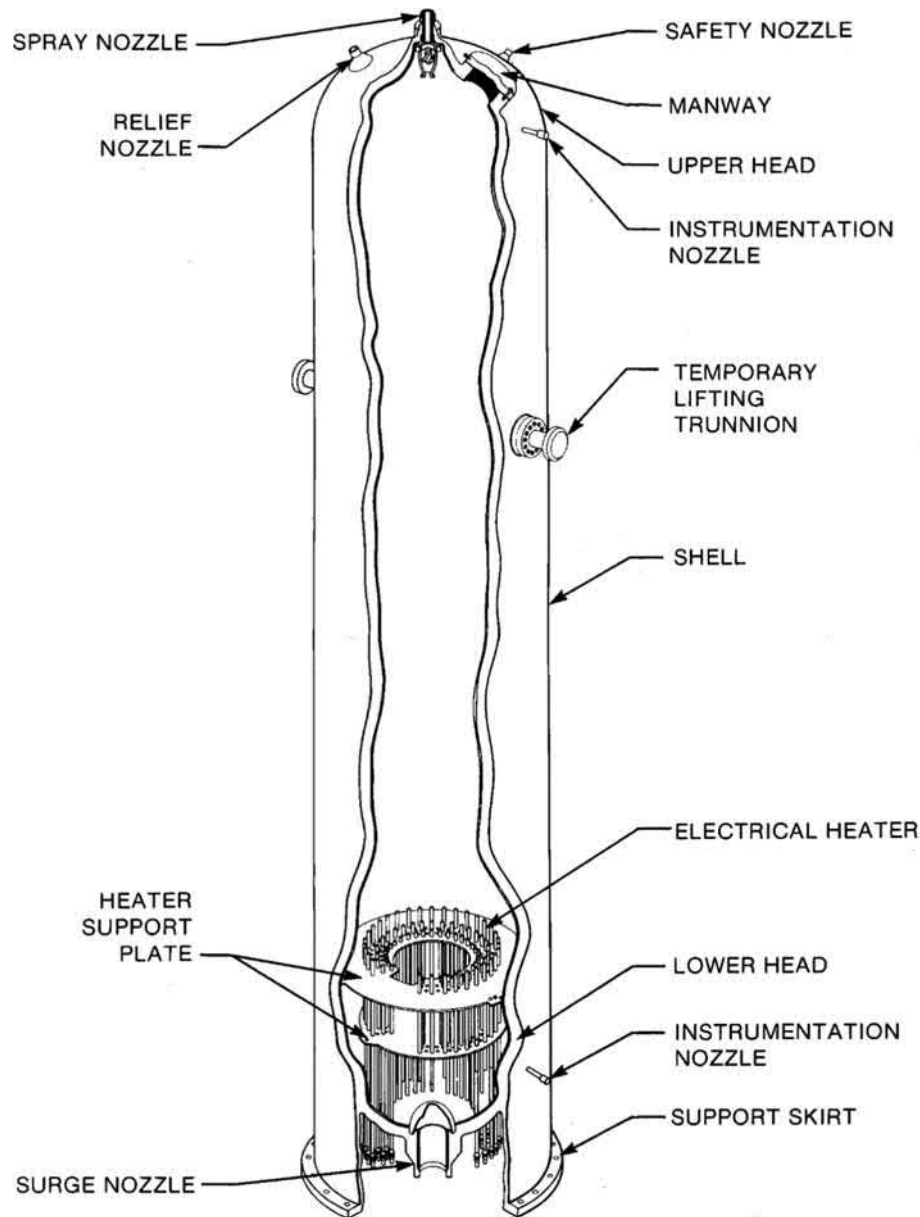


Fig. 11 Pressurizer. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

pressurizer. Electric immersion heaters, located in the bottom of the pressurizer vessel, keep the water at saturation temperature and maintain a constant system operating pressure of approximately 15.2 MPa (2200 psi). Activation of the heaters results in an increase in system pressure. System pressure is decreased by injecting reactor coolant from the cold leg of a coolant loop through the spray nozzle located at the top of the pressurizer to condense the steam bubble. The pressurizer steam bubble also compensates for potential system pressure changes resulting from changes in the primary coolant temperature due to changes in turbine load. A reduction in primary coolant outlet temperature will increase the water density, and thus the level, decreasing the pressure above the water level, which results in some of the water in the pressurizer flashing to steam, thus increasing and restoring the primary system pressure. Likewise, an increase in primary coolant temperature will increase the pressure inside the pressurizer resulting in condensation of the pressurizer steam bubble thus decreasing and restoring the primary system pressure. This results in a natural tendency for the PWR RCS to follow moderate changes in the turbine demand. Flexible operations, which are intentional changes in reactor power are discussed in [Young \(2021b\)](#). The pressurizer heaters, in combination with heat imparted to the coolant by the Reactor Coolant Pumps, are also used to heat up the RCS to operating temperature ($\sim 550^\circ\text{F}$) prior to bringing the core to critical.

Emergency core cooling system

All nuclear power plants are required to have an emergency core cooling system (ECCS) that can rapidly inject coolant into the reactor core during accident situations (Rempe, 2021). The emergency systems are required to have redundant components to assure that the safety function is maintained even in the event of a failure of a key component. The ECCS for a PWR requires unique design considerations due to the large pressure of the primary coolant. For this reason, a PWR ECCS consists of two coolant delivery systems, one for high pressure injection and one for low pressure injection (See Fig. 12). The ECCS system is designed to provide not only emergency core cooling, but also continued cooling during the long-term phase following the accident.

The initial source of the ECCS water is the highly borated water contained in the Safety Injection (or Accumulator) Tanks, which are located inside the containment building. The accumulators deliver water by passive gravity feed to the cold leg and are designed to rapidly and completely reflood the core following a large break Loss of Coolant Accident (LOCA). The High Pressure and Low Pressure Safety Injection (LPSI) pumps, which are located outside but close to containment, draw cooling water from the Refueling Water Tank (RWT). When the RWT water is nearly depleted, the system automatically switches the source of the Injection pump to the containment sump. The LPSI pumps continue to inject water from the containment sump to the cold leg until the system is manually realigned to inject to the hot leg. The purpose of this re-alignment is to terminate boiling in the core and to prevent the buildup of precipitated boron from interfering with core cooling following a large cold-leg break.

The Emergency Feedwater System (EFWS) is used to supply feedwater to the steam generators following accident or transient conditions when the main feedwater system is not available. The EFWS thereby maintains the capability of the steam generators to remove plant stored heat and core decay heat by converting the emergency feedwater to steam which is then discharged to the condenser or to the atmosphere. The EFWS is also capable of supplying feedwater to the steam generator during normal plant operations of startup, shutdown, and hot standby.

The power for the Emergency Core cooling systems is normally supplied from the offsite electrical grid (which also must have redundant connections). In the event of loss of offsite power, there are two onsite emergency diesel generators which are automatically started. As a final backup, there are two sets of battery packs that are designed to maintain the emergency core cooling function from 2 to 4 h after loss of all other power.

Reactor containment

The nuclear steam system for all PWR power plants are enclosed in a large steel lined ~40 in. thick concrete containment building (Fig. 13). The primary function of the containment building is to contain and control any release of radioactivity to the environment under normal, upset, or emergency conditions including severe earthquakes (Sharpe, 2021). A PWR containment building is much larger than that for a BWR, as there must be sufficient volume to accommodate full and sudden release and subsequent flash of the high pressure RCS coolant to steam without exceeding the containment building stress limits.

The containment contains various systems that are designed to maintain containment integrity even during accident conditions:

- Containment spray system to reduce the containment temperature and pressure
- Containment Cooling System to cool the containment atmosphere during normal and accident conditions
- The Containment Isolation System provides the means of isolating the various fluid systems passing through the containment walls in order to prevent the release of radioactivity to the outside environment in the case of an accident

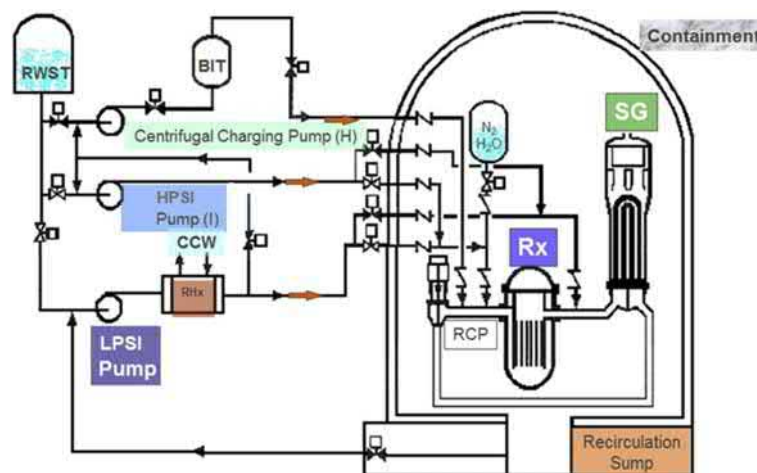


Fig. 12 Emergency Core Cooling System. Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*.—Courtesy Westinghouse Electric Company.

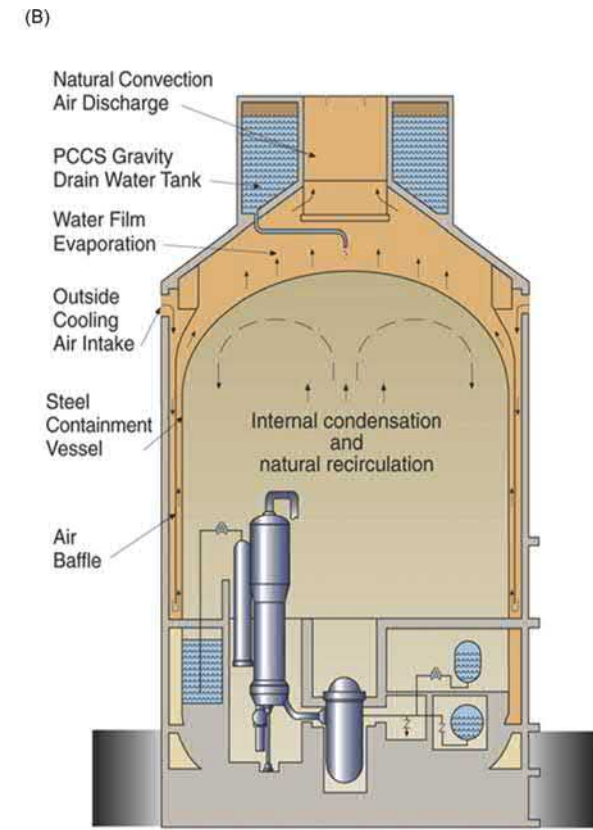
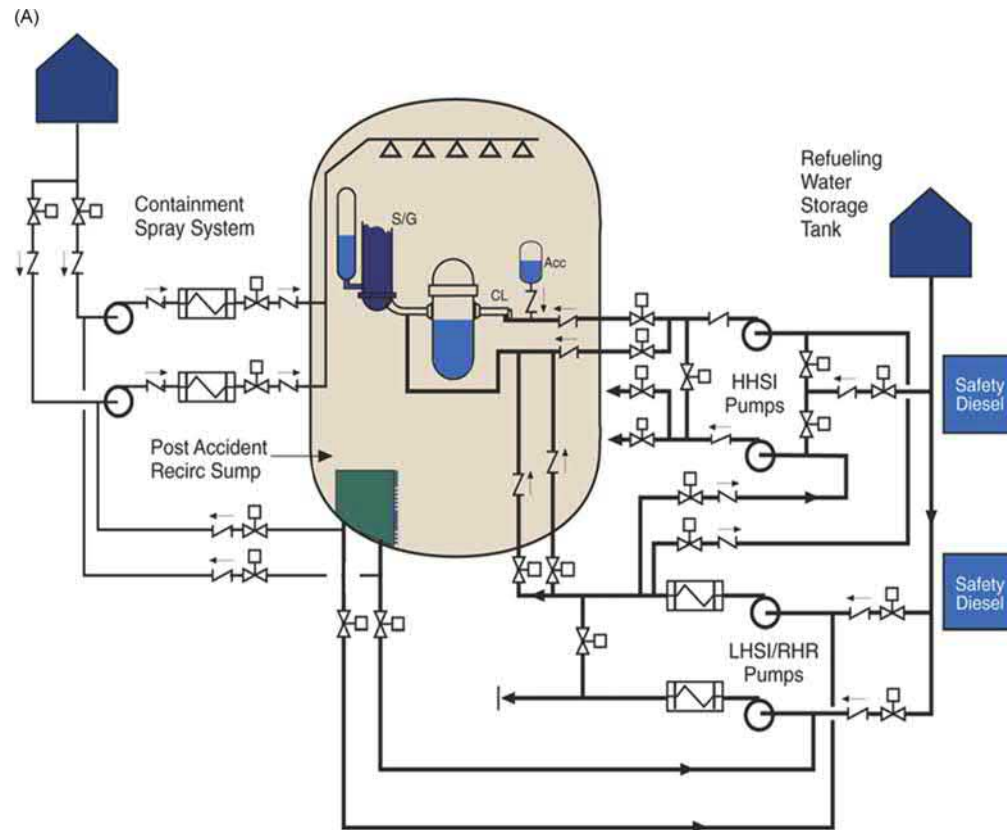


Fig. 13 PWR Reactor Containment Building. (A) Standard PWR Containment. (B) AP1000® Plant Passive Containment. (A) Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*. (B) *Westinghouse AP1000 Nuclear Power Plant: Coping with Station Blackout*, April 2011.

- Hydrogen suppression system that can detect and recombine free hydrogen that may have been generated by water radiolysis or from Zircaloy-water reactions under severe accident conditions

The Westinghouse AP1000[®] plant uses passive containment cooling (See Fig. 13B) which can function for several days even in the event of a full station blackout event similar to the Fukushima event.

Ancillary PWR components

PWR power plants also contain components required to maintain operation of the plant, and to mitigate consequences during accident situations.

The Chemical Volume Control System (CVCS) which is mainly located in the auxiliary building outside the containment building is used to control water inventory, boric acid concentration, and chemistry of the primary RCS. These functions are satisfied by maintaining a continuous feed-and-bleed stream to and from the RCS during plant operation. Water from the Boric Acid Tank, which contains a highly concentrated solution of boric acid, or the Reactor Water Makeup tank, which contains demineralized pure water, is injected into the RCS by charging pumps, depending on whether the RCS boron concentration is to be increased or decreased. The feed rate to the RCS is automatically controlled by pressurizer water level while the bleed rate can be set by selecting the proper combination of letdown orifices to meet plant operational requirements. Letdown water leaves the RCS and flows through the shell side of the regenerative heat exchanger where it gives up its heat to makeup water before being returned to the RCS. By using this heat exchanger to heat up the makeup water, heat loss from RCS during RCS boration and dilution is minimized.

The primary function of the Residual Heat Removal System (RHRS) is to transfer heat energy from the core and primary Reactor Coolant System to heat exchangers, cooled by the plant's ultimate heat sink (either via a cooling tower, or body of water, such as a river) located outside containment during plant cooldown and refueling operations. The system is designed to reduce the RCS temperature to 140 °F (60 °C) within 20 h following reactor shutdown. The RHRS extracts coolant from a depressurized RCS hot leg and routes it through a heat exchanger located outside containment. During normal plant operation, the RHR pumps and heat exchangers are normally isolated from the RCS in order to preserve the RCS pressure boundary and containment isolation requirements. Cold RHR coolant is returned to the cold leg of the RCS. Provisions are made for continued flow of the reactor coolant to the CVCS during shutdown. The RHRS may also be used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of refueling operations. The low pressure RHR pumps can also draw from the containment sump and Refueling Water Tank to provide the Low Pressure Safety Injection cooling to the depressurized RCS system following a LOCA.

Balance of plant components

The components of the secondary and other Balance of Plant Systems are functionally the same as those used in fossil steam plants. The major differences in the secondary system of PWR plants are:

- a) Due to the continuous production of decay heat in the reactor core even while shutdown, the secondary system must always maintain coolant to the reactor. This requirement manifests itself in emergency feedwater pumps and extra heat exchangers. No cooling is needed for a fossil plant during shutdown.
- b) The PWR delivers a high flow of low temperature saturated steam to the turbine whereas a fossil plant delivers a smaller flow of superheated steam. This requires a "wet turbine" design, which has a different turbine blade design than those used in a fossil plant's turbine, which accepts superheated steam for higher thermal efficiency energy conversion.

Operation of PWR

Today most PWRs are operated as large (800–1300 MWe) base load units which are maintained at or near full power conditions during their entire operating cycle. Most PWRs operate on an 18 month refueling schedule with a few plants on 12 or 24 month refueling schedules. PWRs are typically operated with all the control rods fully withdrawn and the core excess reactivity is controlled by a combination of burnable absorber located in the fuel assemblies and boric acid dissolved in the RCS. The RCS soluble boron concentration at the beginning of cycle is usually set at 1000–1300 ppm which is diluted (by the CVCS) to ~10 ppm at the end of the cycle.

Future of the PWR

To date, there are four standardized US PWR designs with thermal power ratings up to 4000 MWth that have been approved through the design certification by the USNRC for future US application: Westinghouse AP600, System80+ and AP1000[®] plants,

and the Korean APR1400 plant (NRC, 2019). Two standard BWR designs, ABWR and ESBWR (Hamon, 2021b) have also been approved. PWRs (EPR) are currently under construction will have power ratings up to 1600 MWe, 4500 MWth. In addition, a significant number of other standard PWR designs are being deployed elsewhere (APR1400-Korea, VVER1000 -Russia, CPR1000- China). There are also several Small Modular Reactor designs, most notably the NuScale SMR (NRC, 2020), based on the PWR design with power ratings less than 100 MWe.

Advancements in material science and PWR core design have allowed for plant lifetime extension beyond the original 40 year design life to 60 years even 80 years in some cases. PWR fuel designs are being tested that will increase the fuel discharge burnup to 80 GWD/MTU or beyond with fuel enrichments up to 10 wt% U235. Given these trends it is clear that the PWR reactor design will likely continue to be a major contributor to the world energy supply for many years to come.

See Also: Advanced Light Water-Cooled Reactor Concepts; Commercial Nuclear Power Reactors and Their Design—Introduction; Floating Nuclear Power Plants With Small Modular Reactors; Light Water Reactors (LWR) Fuel Cycle Options; Pros and Cons of Commercial Reactor Designs: Part 2—Current Status and Future Trends in the World Nuclear-Power Industry and Technical Considerations of Nuclear-Power Reactors; Water Cooled Small Modular Reactors (Integral PWR and BWR).

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Boiling Water Reactors

David A. Hamon*, GE-Hitachi Nuclear Energy, San Jose, CA, United States

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Glossary

BWRX-300 GE 300 MWe natural circulation BWR design undergoing design certification review.

KRB Gundremmingen Unit A (BWR/1 in Germany).

LOCA Loss-of-Coolant Accident. Loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system [10 CFR 50, Appendix A].

SCRAM Reactor trip, involving rapid full insertion of all control rods.

Abbreviations

ABWR Advanced boiling water reactor

ADS Automatic depressurization system

AET Advanced Engineering Team

AOOs Anticipated operational occurrences

BWR Boiling water reactor

CRD Control rod drive

CST Condensate storage tank

ECCS Emergency core cooling system

ESBWR Economic simplified boiling water reactor

FMCRD Fine motion control rod drive

FWCS Feedwater control system

GE General Electric Company

HPCF High pressure core flooders

HPCI High pressure coolant injection

HPCS High pressure core spray

LPCF Low pressure core flooders

LPCI Low pressure coolant injection (mode of RHR system)

*Retired

LPCRD Locking piston control rod drive
 LPCS Low pressure core spray
 MWe Megawatts electric
 MWth Megawatts thermal
 NSSS Nuclear steam supply system
 PCS Pressure control system
 PG&E Pacific Gas and Electric Company
 PWR Pressurized water reactor
 RCCV Reinforced concrete containment vessel
 RCIC Reactor core isolation cooling
 RCS Reactivity control system
 RHR Residual heat removal
 RIPs Reactor internal pumps
 RPV Reactor pressure vessel
 SLC Standby liquid control
 SMR Small modular reactor
 USE Upper shelf energy

Introduction

The Boiling Water Reactor (BWR) is based on two fundamental principles that distinguish it from other nuclear power plant reactors: (1) bulk boiling of water occurs in the reactor core; and (2) steam produced from boiling in the reactor core is sent directly to the turbine that is used to turn a generator to produce electricity.

The General Electric Company (GE) signed a contract with the Commonwealth Edison Company of Chicago, Illinois, in 1955 to deliver the 180 MWe Dresden Nuclear Station (Unit 1). To assure the success of the Dresden-1 reactor, GE embarked on construction of a smaller scale development plant at its Vallecitos Atomic Laboratory near Pleasanton, California, to develop operational data for the Dresden-1 design. Pacific Gas & Electric Company (PG&E) also participated in the developmental reactor project by supplying a turbine-generator. In 1957 the Vallecitos Boiling Water Reactor (VBWR) became the first privately built atomic power reactor to supply electricity to a public utility grid ([Fig. 1](#)). Additional information about early BWR development can be found in [Kramer \(1958\)](#) and in [Lahey and Moody \(1977\)](#).

Today approximately one third of the operating nuclear power reactors in the United States and one half of the operational nuclear power reactors in Japan are of the BWR design. The largest BWR plants are currently operating with electrical power ratings of slightly over 1400 MWe with a reactor power of approximately 4000 MWth. All US BWRs are GE designs.

BWR overview

GE pursued different design concepts for the earliest Generation I BWRs, that is, natural circulation boilers (Humboldt Bay, Dodewaard), forced circulation boilers (Dresden-1, Big Rock Point, KRB, Oyster Point), and considered nuclear superheat (steam cooled) boilers in development programs to achieve higher levels of electrical power output.

Dresden-1, the first commercial BWR, was a dual-cycle plant that incorporated an external steam drum as well as steam generators ([Fig. 2](#)). This configuration provided a convenient capability to follow load changes. Given an increased load demand, a signal would be sent to the steam generator's control valve to admit additional steam to the lower stage of the turbine, allowing the primary pressure and steaming rate from the reactor to remain unchanged, initially. As steaming rate in the steam generator increased, reactor inlet coolant became more subcooled causing a reduction of in-core steam void formation and higher moderation of neutrons, which in turn caused an increase in reactor power to meet the new load requirement. While this provided a load following advantage, it came at additional plant capital cost and maintenance expense. Therefore, other means of load following (i.e., varying core recirculation flow rate) as well as improved means for steam separation (i.e., in-vessel steam separators and dryers) were developed.

During the 1950s and into the 1960s, when the BWR was in its infancy, it grew by exploring different options and design configurations and feeding back operating experience gained. As a result, standardization did not exist to the extent that it would for future product lines of the Generation II reactors. While the various Generation I BWR configurations successfully operated for many years, operating experience along with improvements in manufacturing capabilities eventually drove toward forced circulation, internal vessel steam separation, and direct-cycle BWRs as a standard configuration.

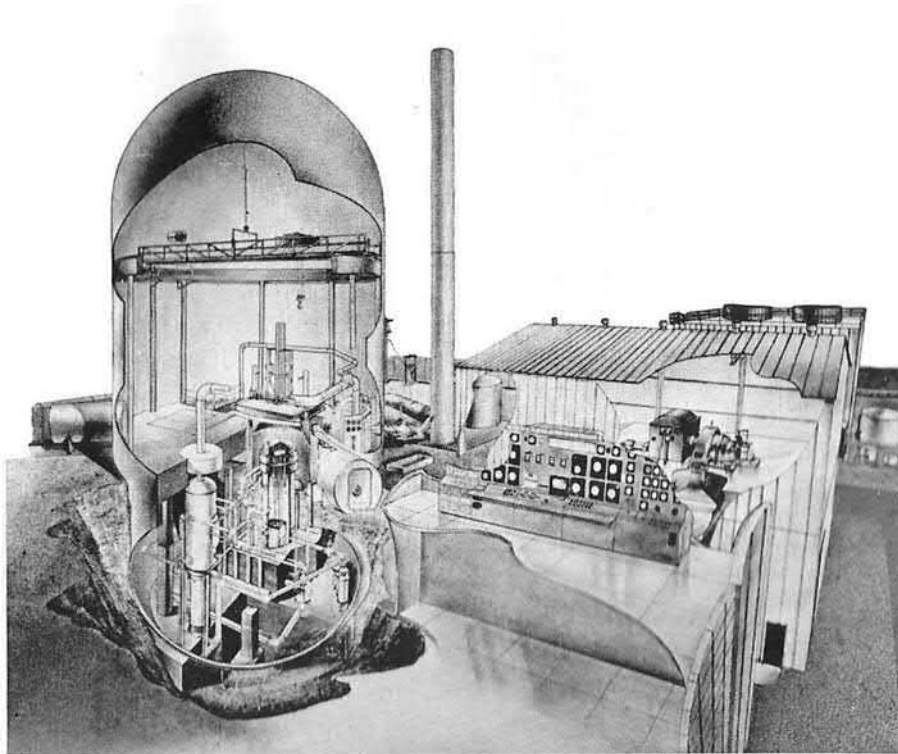


Fig. 1 Vallecitos BWR plant cutaway. Kramer AW (1958) *Boiling Water Reactors*. Addison-Wesley Publishing Company Inc.

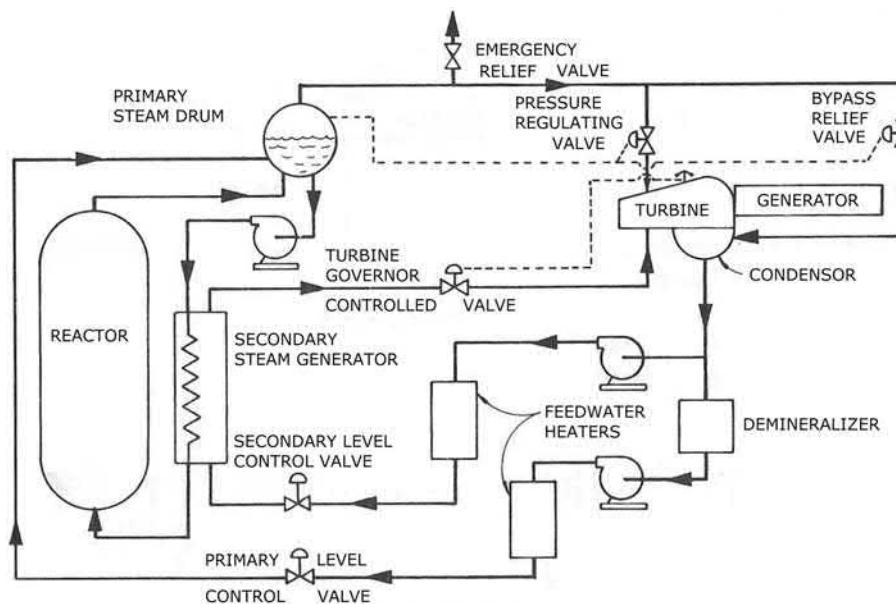


Fig. 2 Dresden-1 schematic diagram. Lahey Jr. RT and Moody FJ (1977) *The Thermal-Hydraulics of a Boiling Water Nuclear Reactor*. Prepared under the American Nuclear Society for United States Energy Research and Development Administration.

With the growth of nuclear power electrical generation worldwide during the 1960s and the following decades, improved BWR product lines—designated as BWR/2, BWR/3, BWR/4, BWR/5 and BWR/6—evolved to meet demand. The Oyster Creek BWR/2 was the first modern plant characterizing the distinctive characteristics of the Generation II BWRs: direct-cycle, forced circulation, internal steam separation and higher power output. **Fig. 3** illustrates these distinctive BWR characteristics for the BWR/6. Additional information about the BWR/6, beyond what is found in this encyclopedia chapter, can be found in the BWR/6 General Description of a Boiling Water Reactor ([General Electric \(GE\) Nuclear Energy Group, 1980](#)).

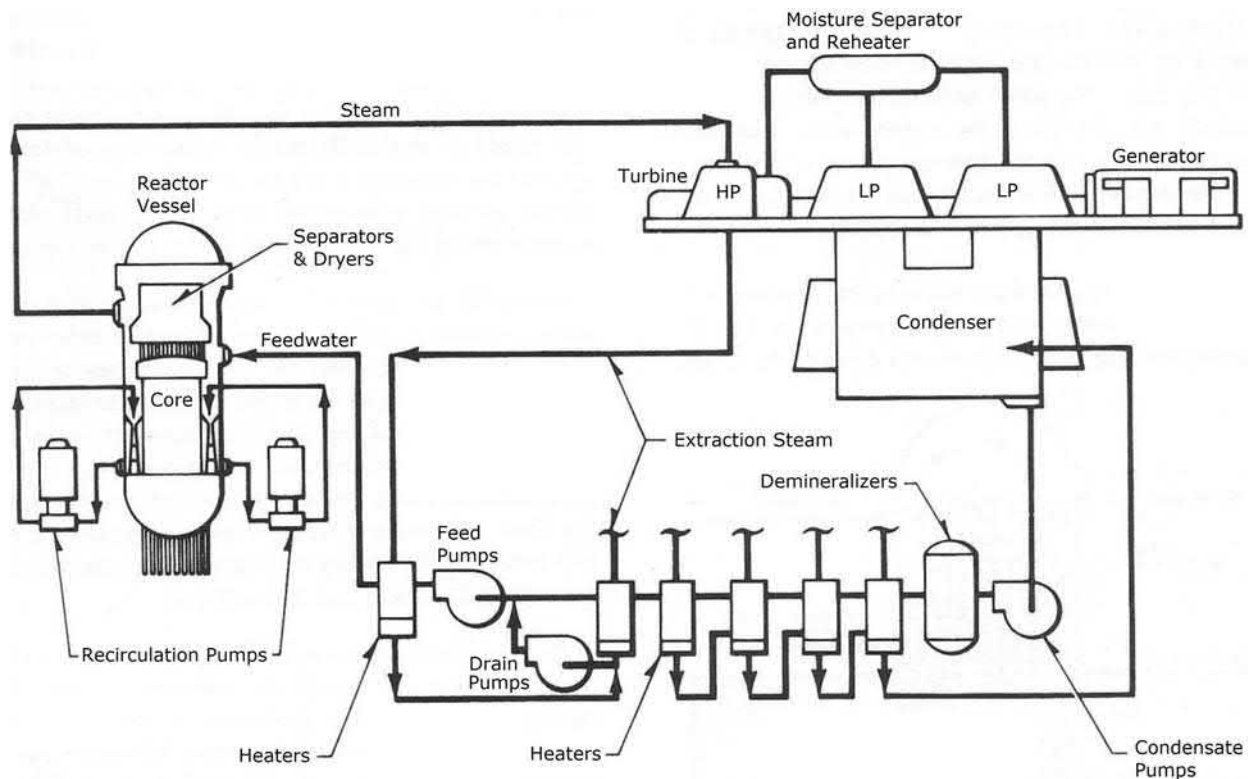


Fig. 3 Direct cycle BWR (typical of BWR/6). General Electric (GE) Nuclear Energy Group (1980) BWR/6 general description of a boiling water reactor. Revised September 1980.

Key motivators for evolution of the Generation II product lines included emerging safety regulations and improved performance and economic expectations. In the descriptions to follow, key features that distinguish the Generation II BWRs from these perspectives are described.

Generation III designs incorporate evolutionary improvements in areas of safety, performance, efficiency, technology and standardization relative to design and experience of the typical Generation II operating fleet. The ABWR is an example of a BWR Generation III reactor that is being actively pursued worldwide. International BWR development is discussed in a later section.

The primary design objectives during the ABWR development were: improve operation and maintenance; improve safety with diversity; employ advanced technology; reduce construction time; and reduce power generation costs. Key design features that achieved these objectives included fine motion control rod drives, reactor internal pumps, optimized Emergency Core Cooling Systems (ECCS), digital instrumentation and controls, and a reinforced cylindrical concrete containment vessel with a steel liner. Additionally, automated maintenance tools and radiation reduction features contributed to operation and maintenance improvements. The ABWR electrical output is approximately 1350 MWe. Additional information about the ABWR, beyond that which is found in this encyclopedia chapter, can be found in the ABWR Plant General Description (GE-Hitachi Nuclear Energy, 2007). A more complete description of the ABWR can be found in Chapter 1 of the ABWR Design Control Document (GE-Hitachi Nuclear Energy, 2016).

Table 1 shows typical BWR system operating parameters for the ABWR, one BWR/6 and one BWR/5.

Emergency core cooling systems (ECCS)

Early BWRs were designed without separate safety systems to provide core cooling in an emergency, although they did make use of highly reliable feedwater systems for that purpose. However, as concerns were raised with respect to providing adequate core cooling during a Loss-of-Coolant-Accident (LOCA) event that is severe enough to threaten containment integrity, it was required that all nuclear plants must have ECCS. Separate ECCS was included in the BWR/2, and some forms of ECCS were also retrofitted for BWR/1s (Mehta and Pappone, 2009). For BWRs, an ECCS network typically consists of both high and low pressure coolant makeup systems, the use of selected safety-relief valves as part of an Automatic Depressurization System (ADS) to reduce reactor pressure rapidly when necessary to allow injection of low pressure coolant makeup, and supporting systems (e.g., water source, electrical, instrumentation) that are required for proper functioning of the ECCS. High pressure ECCS coolant makeup systems are designed to inject coolant flow into the Reactor Pressure Vessel (RPV) quickly, and can often prevent the need for a rapid reactor

Table 1 Typical GE BWR system parameters.

<i>Design</i>	<i>ABWR 278–872</i>	<i>NMP-2 BWR/5 251–764</i>	<i>Grand Gulf BWR/6 251–800</i>
Rated power (MWth)	3926	3323	3833
Nominal steam dome pressure (psia)	1040	1020	1040
Steam flow rate (Mlbm/h) at 420 °F feedwater temperature	16.843	14.263	16.491
Feedwater flow rate (Mlbm/h)	16.807	14.564	16.455
Core coolant flow rate (Mlbm/h)	115.1	108.5	112.5
Feedwater temperature (°F)	420	420	420
Core average exit quality (% steam)	14.5	13.1	14.6
Number of fuel assemblies	872	764	800
Core weight as UO ₂ (lbm)	379,221	265,551	365,693
Core diameter (equivalent) (in.)	203.3	160.2	191.5
Core height (in.)	146	146	150

Modified from GE-Hitachi Nuclear Energy (2016) ABWR Design Control Document—Tier 2, Chapter 1, Table 1.3-1, Rev 6, 25A5675AC, February 2016.

depressurization for less severe events. Low pressure ECCS coolant makeup systems must wait for reactor pressure to be substantially reduced, either by the break or the operation of the ADS, in order to begin injecting coolant flow into the RPV.

For the BWR/3 and BWR/4, two electrical divisions of ECCS and an ADS were carried forward from the BWR/2. The two divisions contained three ECCS pumps each, consisting of two Residual Heat Removal (RHR) pumps that were also used for Low Pressure Coolant Injection (LPCI) and one Low Pressure Core Spray (LPCS) pump. A steam turbine driven High Pressure Core Injection (HPCI) systems was also included in the ECCS network. Some later BWR/4 projects (e.g., Limerick Generating Station, Hope Creek Nuclear Generating Station) added a third and fourth electrical division for increased redundancy and redistributed the loading of the ECCS pumps among the increased number of divisions.

At the time the BWR/2, BWR/3, and BWR/4 ECCS were designed, there were no regulations defining the ECCS performance acceptance criteria. In 1973, US 10 CFR 50.46 was issued, which defined the ECCS performance acceptance criteria, for example, the peak cladding temperature was limited to approximately 1200 °C.

Improvement of ECCS continued in the BWR/5 and BWR/6 product lines, with three electrical divisions. Two divisions contained two ECCS pumps each, an RHR pump that was also used for LPCI, and either an LPCS or a second LPCI pump dedicated to reactor vessel coolant injection. The third electrical division consisted of a High Pressure Core Spray (HPCS) system powered by a dedicated diesel generator. The HPCS system replaced a turbine-driven HPCI used in the earlier product lines.

A three divisional ECCS was adopted for the ABWR as shown in Fig. 4. Each division provides both low pressure and high pressure coolant makeup capability and incorporates its own emergency diesel generator power supply. One division includes a Reactor Core Isolation Cooling (RCIC) high pressure makeup system and a Residual Heat Removal/Low Pressure Flooder (RHR/LPFL) low pressure system, while the other two divisions include High Pressure Core Flooder (HPCF) and RHR/LPFL trains. RCIC employs

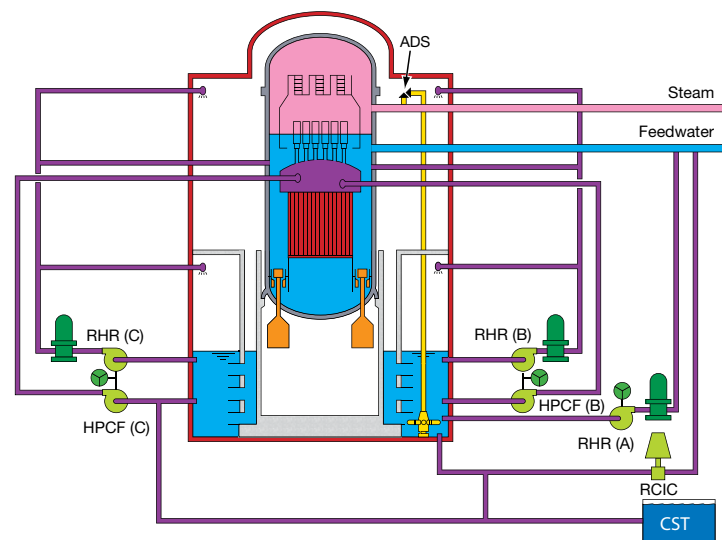


Fig. 4 ABWR ECCS network. GE-Hitachi Nuclear Energy (2007) The ABWR plant general description. Revised July 1, 2007. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ABWR%20General%20Description%20Book.pdf.

a turbine-driven pump feed by reactor steam to provide high pressure makeup when the reactor is isolated from the normal power cycle, drawing water from either the containment Suppression Pool or the Condensate Storage Tank (CST). As with previous BWR product lines, the ABWR also incorporates an Automatic Depressurization System (ADS) to provide effective makeup from the low pressure pumps over the full range of potential LOCA break sizes.

The internal recirculation pumps in the ABWR eliminated the large external recirculation loop piping. By eliminating this piping and the associated large vessel nozzles below the top of the active fuel, it became possible to design the ECCS with less capacity and to keep the core covered during a design basis LOCA.

Containment

Reactor containment systems (Sharpe, 2021) are primarily designed to contain reactor coolant losses from a LOCA event within the containment, which in turn minimizes any potential release of radioactive material to the environment. There are two primary types of reactor containments, dry and wet. A dry reactor containment is sized to remain below design pressure after absorbing all the energy release from a LOCA in its air volume. A wet containment is designed to direct most of the steam released from the reactor during a LOCA through a pool of water (i.e., suppression chamber or suppression pool) that condenses the steam, thereby reducing the containment volume that is necessary to absorb the energy from the coolant release.

The initial BWR containment systems were constructed from steel in a spherical shape, and were dry. However, this was soon superseded by the Mark I, Mark II and Mark III pressure suppression containments, each incorporating a suppression chamber containing a large pool of water.

There are several advantages of an in-containment pressure suppression pool for the BWR:

- Provision of a heat sink for LOCA blowdown steam, safety/relief valve discharge steam, and reactor steam discharged from turbine driven pumped water make up systems;
- Provision of a secure source of coolant water for ECCS and reactor isolation make up and cooling pumps;
- Provision to filter and retain fission products that may be released during an accident or transient event.

Because the BWR may discharge steam when the reactor is isolated during Anticipated Operational Occurrences (AOOs) that pressurize the direct cycle, it was a natural fit for the BWR to adopt the pressure suppression system. This also allowed a smaller containment vessel than a dry containment by taking advantage of the pressure reduction from the suppression pool.

Early BWR product lines utilized the Mark I containment vessel, which consisted of an inverted “light bulb” shaped drywell surrounding the Reactor Pressure Vessel (RPV), and a suppression chamber consisting of a steel pressure vessel with a toroidal shape (“torus”) and a large body of water inside with large surface area for venting steam (Fig. 5).

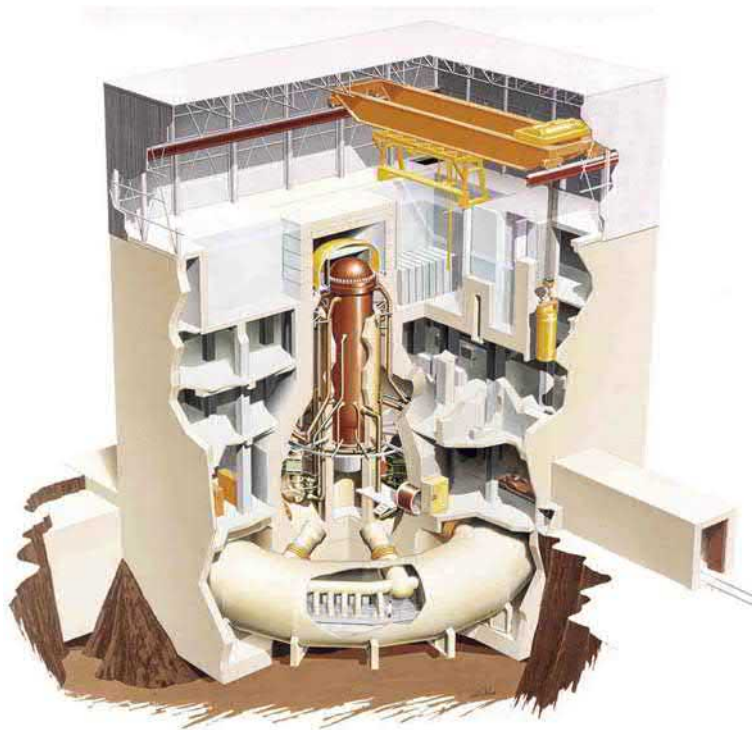


Fig. 5 Mark I containment configuration. Courtesy GE-Hitachi Nuclear Energy.

The Mark II containment vessel was introduced for some BWR/4s and for BWR/5s, and consisted of a steel dome head and either a post tensioned concrete wall or reinforced concrete wall standing on a base mat of reinforced concrete. The inner surface of the containment was lined with a steel plate to provide a leak tight membrane. The Mark II design provided better access to piping and equipment in the drywell, a simpler vent configuration using straight pipes, the potential to use different construction materials, and a smaller reactor building (Fig. 6).

The Mark III containment vessel was used for the BWR/6 product line and adopted the simpler geometry of a right circular cylinder, which was easier to construct and provided better access to equipment for maintenance. It could be built as either a free-standing steel containment surrounded by a concrete shield building or as a concrete pressure vessel with a liner (Fig. 7).

The ABWR cutaway showing the reactor building, control building and turbine building, consistent with the US certified design, is illustrated in Fig. 8. The Reinforced Concrete Containment Vessel (RCCV) is cylindrical and consists of a top slab, a shell and a foundation. The RCCV is divided into a drywell and a suppression chamber by the diaphragm floor and the RPV pedestal. The compact structure and reactor building integration improves seismic stability and resistance and provides construction and cost effectiveness.

Reactor pressure vessel and internal assembly

Elimination of steam generators in Generation II BWRs resulted in considerable simplification of the BWR Nuclear Steam Supply System (NSSS). Early prototypes and commercial BWRs demonstrated that radiation levels in the Turbine Island (TI) for a direct-cycle BWR were manageable. More shielding is required for the BWR TI, and the Reactor Pressure Vessel (RPV) is larger than that for a Pressurized Water Reactor (PWR), but these increased costs are offset by elimination of steam generators and application of smaller volume pressure suppression containments.

The BWR RPV is larger than the PWR RPV for two reasons: (1) the BWR core operates at a lower power density than the PWR core; and (2) the BWR incorporated steam separators and dryers within the RPV, eliminating the external steam generators and steam drum and greatly reducing in-containment piping to these components characteristic of the earlier dual-cycle BWRs.

Fig. 9 shows the BWR/6 RPV assembly, with details of the steam separator and dryer assemblies illustrated in Fig. 10. Steam rises into the separators whereby it impinges on fixed vanes that spin the steam-water mixture to establish a vortex wherein centrifugal forces separate water from the steam in each of three stages. The steam dryer assembly, mounted above the steam separators, receives steam exiting from the separators, impinging on dryer vanes where residual moisture is collected and drained through a series of troughs to the recirculation downcomer annulus between the core shroud and RPV wall.

The ABWR reactor assembly, illustrating the coolant flow path through the reactor internal pumps, is shown in Fig. 11.

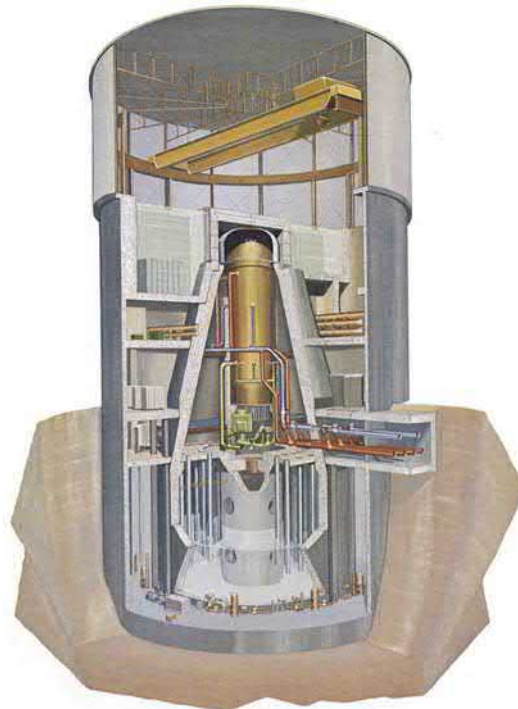


Fig. 6 Mark II containment configuration. Courtesy GE-Hitachi Nuclear Energy.

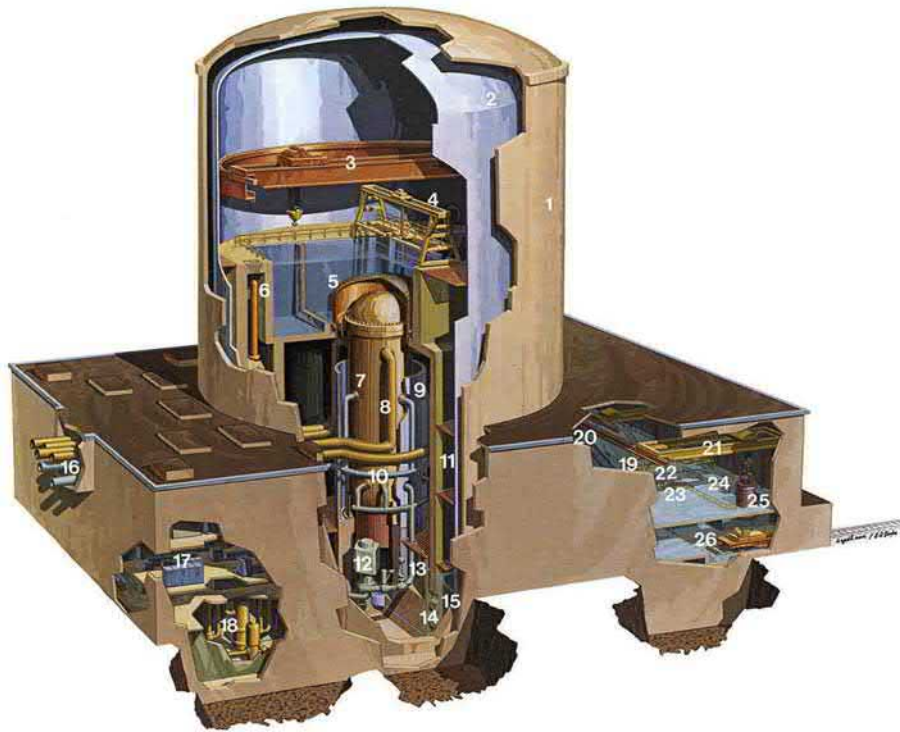


Fig. 7 Mark III containment configuration. Courtesy GE-Hitachi Nuclear Energy.

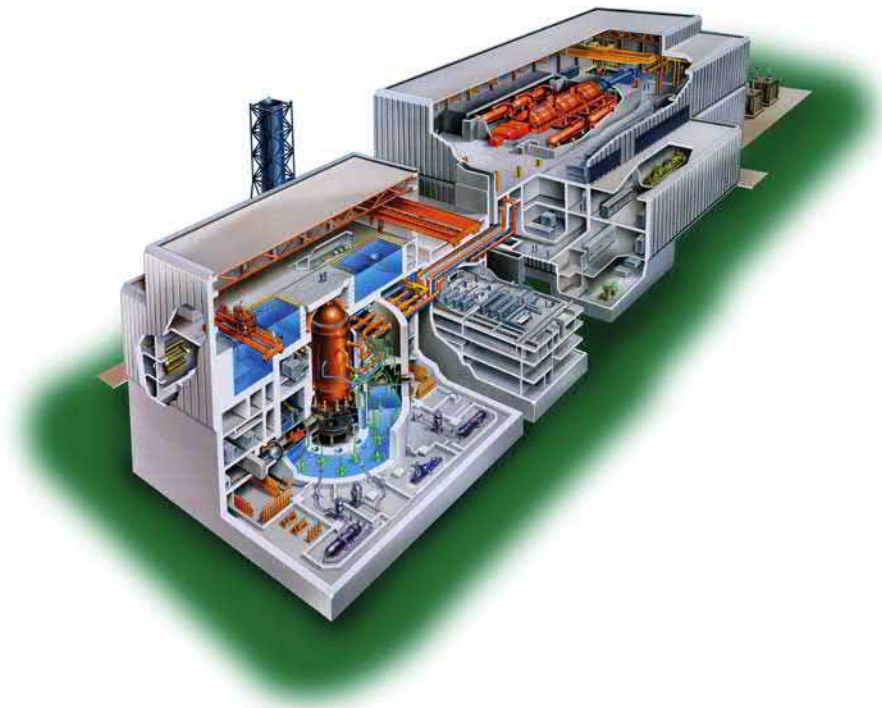


Fig. 8 ABWR cutaway of reactor building, control building and turbine building. GE-Hitachi Nuclear Energy (2007) The ABWR plant general description. Revised July 1, 2007. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ABWR%20General%20Description%20Book.pdf.

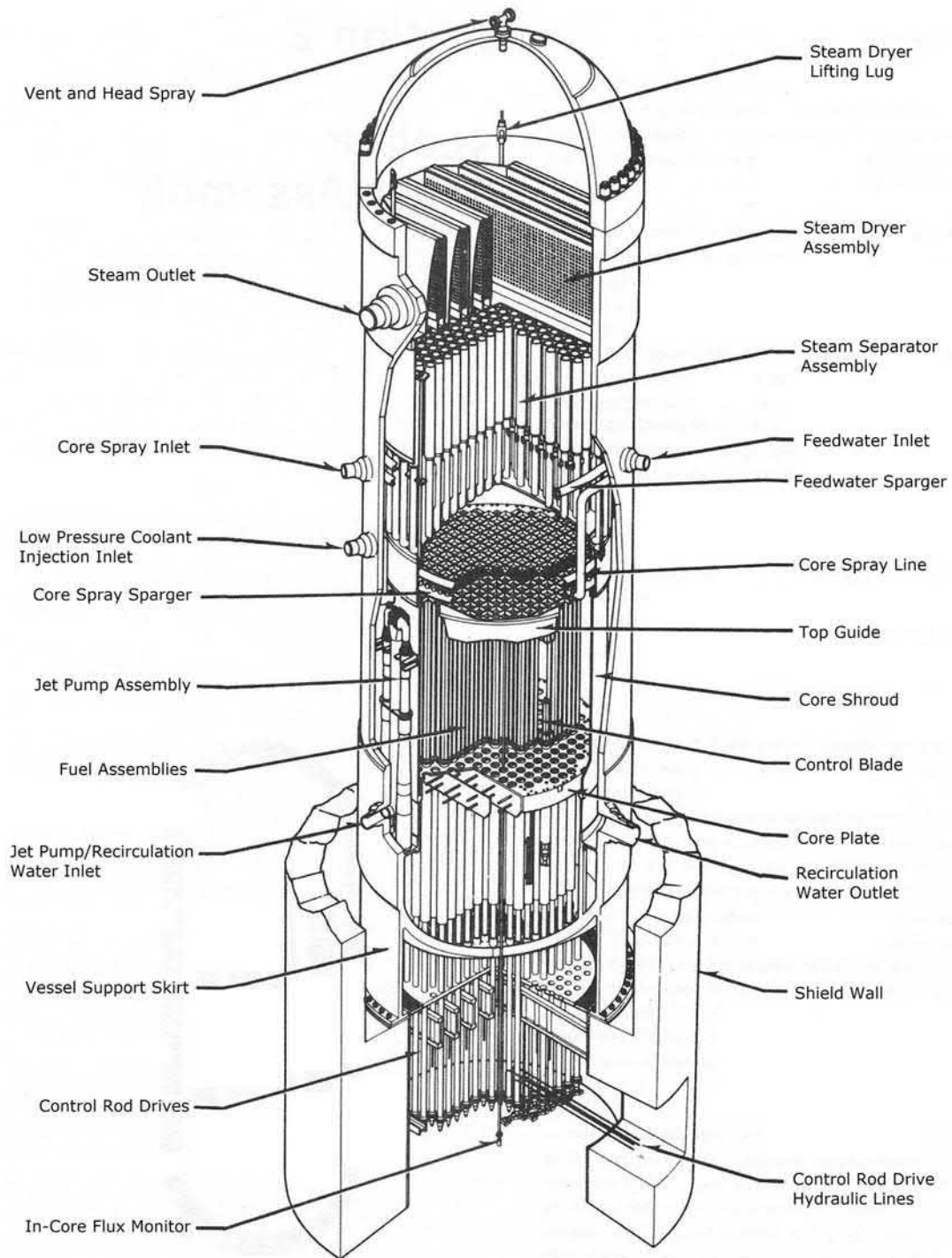


Fig. 9 BWR/6 reactor assembly. General Electric (GE) Nuclear Energy Group (1980) BWR/6 general description of a boiling water reactor. Revised September 1980.

The ABWR has adopted improvements in design and fabrication of the Reactor Pressure Vessel (RPV) relative to earlier BWR product lines. Earlier RPVs were constructed from welded low alloy carbon steel rolled plate. The ABWR uses low alloy carbon steel forged shell rings at and below the core elevation, thereby avoiding welds in high fluence locations in the core beltline. Shell rings above the core beltline region may be made of low alloy carbon steel forged rings or welded plate.

The bottom head consists of a spherical bottom cap, made from a single forging, extending to encompass the Control Rod Drive (CRD) penetrations and a conical transition section to a toroidal knuckle between the bottom head and vessel shell, also made from

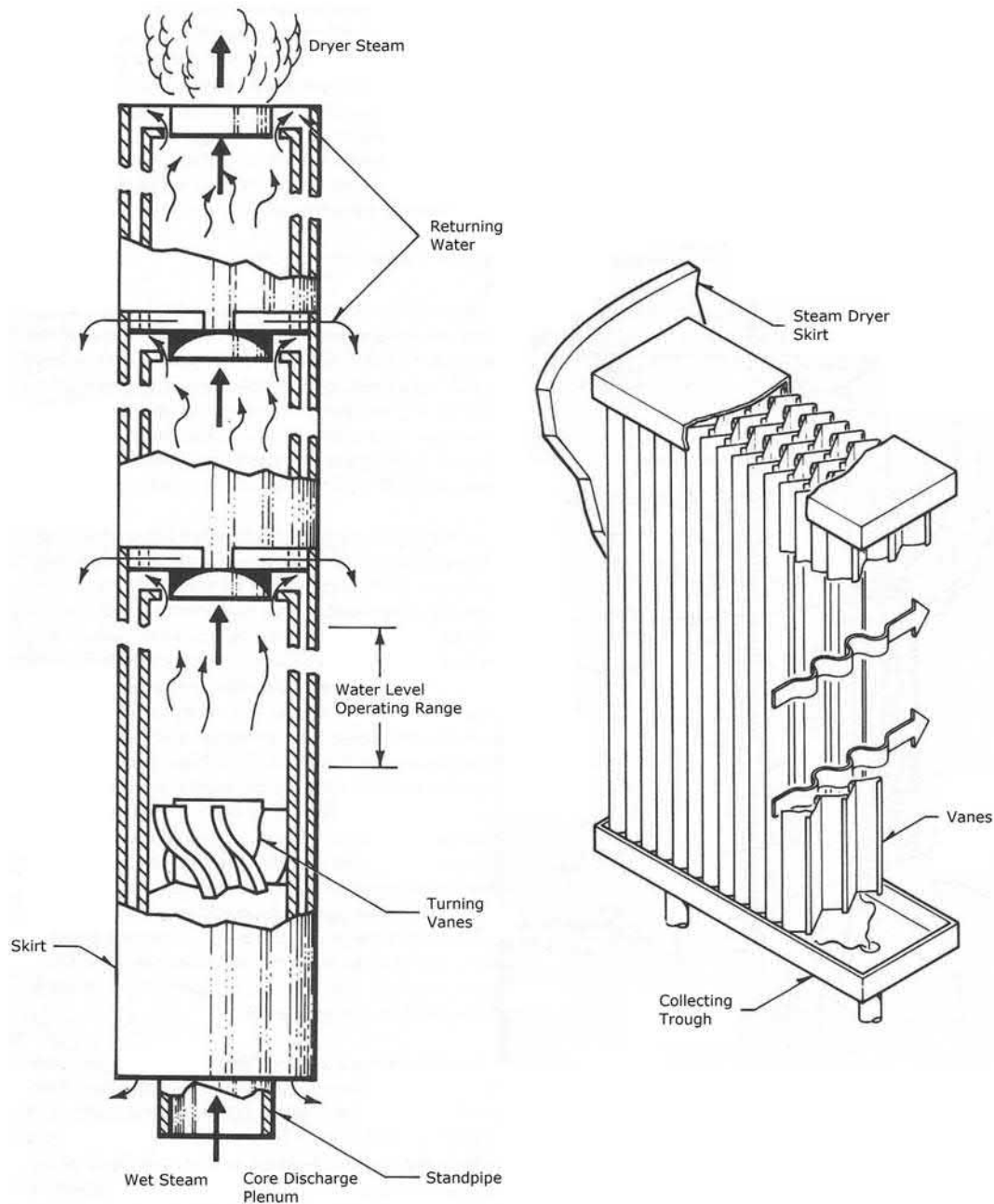


Fig. 10 BWR/6 steam separator and dryer. General Electric (GE) Nuclear Energy Group (1980) BWR/6 general description of a boiling water reactor. Revised September 1980.

a single forging. This eliminates bottom head welds within the CRD pattern. Penetrations are included for the Reactor Internal Pumps (RIPs), and the RIP motor casings are welded to the vessel bottom head.

The vessel support skirt has a conical geometry and is attached to the lower vessel cylindrical shell course. The support skirt attachment is an integral part of the vessel shell ring. Steel anchor bolts extend from the RPV pedestal through the flange of the skirt to secure the support skirt with the pedestal.

To minimize the number of main closure bolts, the ABWR RPV has an inside type flange. This flange allows a hemispherical head closure with a radius less than the vessel radius, which helps minimize the weight of the head closure. The vessel closure seal consists of two concentric "O" rings.

The steam outlet nozzles incorporate a flow restricting venturi. This provides for steamline flow and steamline break detection, and serves as a flow-choking device to limit blowdown and associated loads on the RPV and internals in the event of a postulated main steamline break.

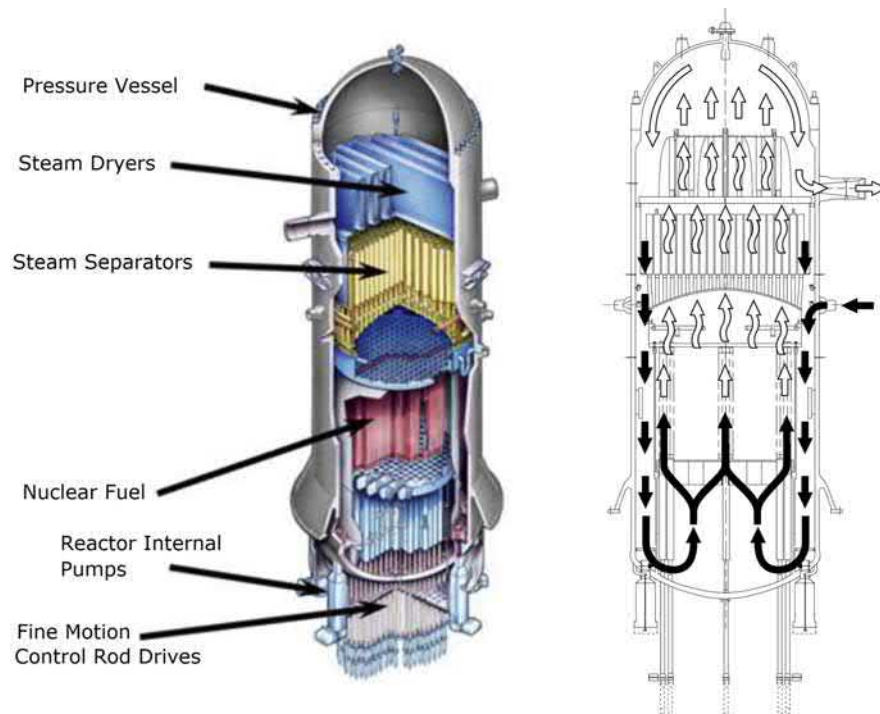


Fig. 11 ABWR reactor assembly and coolant flow path. GE-Hitachi Nuclear Energy (2007) The ABWR plant general description. Revised July 1, 2007. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ABWR%20General%20Description%20Book.pdf.

The feedwater nozzles utilize double thermal sleeves welded to the nozzles. The double thermal sleeves protect material of the vessel nozzle inner bend radius from the effects of high frequency thermal cycling that could result from 216 °C (420 °F) feedwater being introduced into the higher temperature saturated fluid inside the RPV. If not mitigated by the double thermal sleeves, rapid cycling of surface temperature at the RPV nozzles could eventually lead to localized material fatigue damage that would shorten the life of the RPV.

In the United States, reactor pressure vessels have been designated as Safety Class 1, in accordance with ASME Code Section III Design Basis. Specifically, Part 50 of the US Code of Federal Regulations, Title 10 (10 CFR 50) regulates the construction of nuclear power plants. Section 10 CFR 50.55(a) defines the reactor vessel to be part of the reactor coolant boundary and requires that the vessel meet the requirements contained in the ASME Boiler and Pressure Vessel Code Section III for Class 1 vessels.

This code deals generally with specified loading conditions in the equipment specification. The specification identifies the design conditions and operating conditions (normal conditions, upset conditions, emergency conditions, faulted conditions and testing conditions). The code further addresses stresses and stress limits to be considered in the analysis of the component. The methods of analysis and stress limits depend upon the category of loading conditions, that is, the requirement for normal conditions are considerably more stringent than those for faulted conditions. Finally, special requirements are given that have to be met by Class 1 vessels, including provisional thickness requirements for shells, reinforcement requirements for nozzles and recommendations for welding nozzles, for example.

The German reactor vessel designs follow the German KTA standards for LWRs. The KTA requirements are very similar to those in the ASME code regarding the definition of stress intensities and allowable stresses, although differences exist in the design requirements for Upper Shelf Energy (USE) and mid-thickness tensile and Charpy values, as well as for in-service inspections.

The RPV is the most critical component susceptible to brittle fracture failure due to irradiation damage from the neutron fluence. In order to avoid conditions wherein brittle fracture is avoided with ample safety margin, a limit is set on the set of Pressure-Temperature that the RPV is allowed to experience. The ASME Boiler & Pressure Vessel Code Sec. XI and 10 CFR 50 Appendix G provide the requirements for deriving the Pressure-Temperature limit curve. When the Pressure-Temperature limit curve is derived in this way, the area of interest (and postulated crack initiation location for the fracture mechanics evaluations) is the beltline of RPV, which directly surrounds the effective height of active core. This is because almost all RPVs have a semi-spherical bottom head shape, making the distance between core and beltline shorter than the distance between core and bottom head. Thus, the beltline is the high neutron fluence area, and therefore, the most susceptible region of the RPV to brittle fracture failure. Monitoring of the material condition through the use of material sample coupons are used to assess fluence effects for this beltline area. In

addition, the neutron fluence characteristics for the RPV and other reactor internal components are evaluated and maintained throughout the life of the reactor.

Reactor core and fuel assemblies

Although different fuel designs and materials were considered during early BWR development, including flat plates and stainless steel clad fuel elements, Dresden-1 set a trend for the BWR by deploying low enriched uranium dioxide pellets encased in Zircaloy-2 tubes arranged in a 6×6 lattice. The fuel bundle lattice was enclosed within a Zircaloy-2 channel to complete the fuel assembly for loading into the reactor. The interior core layout consisted of repeating cells of four fuel assemblies surrounding a boron steel cruciform control rod (Fig. 12). In order to accommodate the control blade, the gap between fuel channel exterior walls adjacent to wings of the cruciform control rod was wider than the gap for channel walls that were not adjacent to a control rod.

As the BWR product line evolved, the core and fuel design also evolved. The original Dresden-1 core layout ("D-lattice") with its non-uniform gap width evolved to a centered lattice ("C-lattice") and BWR six lattice ("S-lattice") with uniform gap and 6 in. [15.2 cm] fuel pitch (distance from fuel assembly centerline to the adjacent fuel assembly centerline) (Fig. 13). The fuel lattice evolved to 7×7 , 8×8 , 9×9 and 10×10 fuel rod arrays, with progressively smaller diameter fuel rods. Initially, the BWR fuel assembly consisted of a uniform enrichment without burnable poisons, but enrichment variation, rod to rod and axially within a rod, addition of burnable poison (gadolinia) in select fuel pellets, and replacement of some fuel rods with water rods to improve neutron moderation were introduced to improve performance and fuel cycle economics.

The introduction of axially varying enrichment and gadolinia concentrations in the BWR core was effective to flatten power distributions and excess reactivity during the fuel operating cycle, reducing the amount of control rod movement needed to compensate for fuel burnup. Generally, only a small fraction (25% or less) of control rods are required to control excess reactivity during operation. A Control Cell Core loading strategy was introduced that loaded lower reactivity fuel around control rods used for power adjustments, reducing a need for periodic control rod exchanges.

Additionally, the use of adjustable flow recirculation systems to change core flow and power further simplified load adjustments in the BWR. Increasing core flow sweeps steam voids from the core to provide added reactivity as a result of the negative designed void reactivity coefficient. Reducing core flow results in higher void fractions in the core and subsequent reduced core power. Using core flow for load following adjustments provides smoother, global changes in core power output, and provides a means of spectral shift (i.e., lower flow, harder neutron spectrum during the cycle, higher flow, softer spectrum toward end of cycle) to improve fuel cycle economics.

As noted above, BWR power density is lower than that for the PWR. Lower power density coupled with lower operating pressure provides operating thermal margin for the BWR, and optimization of power density, core flow and operating pressure provides nuclear-thermal-hydraulic stability during operation of the BWR. A benefit of lower power density is less buildup of Xenon-135 that absorbs neutrons and reduces core reactivity, facilitating lower enrichment requirements and improved fuel utilization than would be the case if power density were higher. Low power density coupled with void reactivity feedback in the BWR also provides strong damping of Xenon transients. However, lower power density of the BWR does result in a larger diameter core and fuel loading inventory than would be the case if power density were higher.

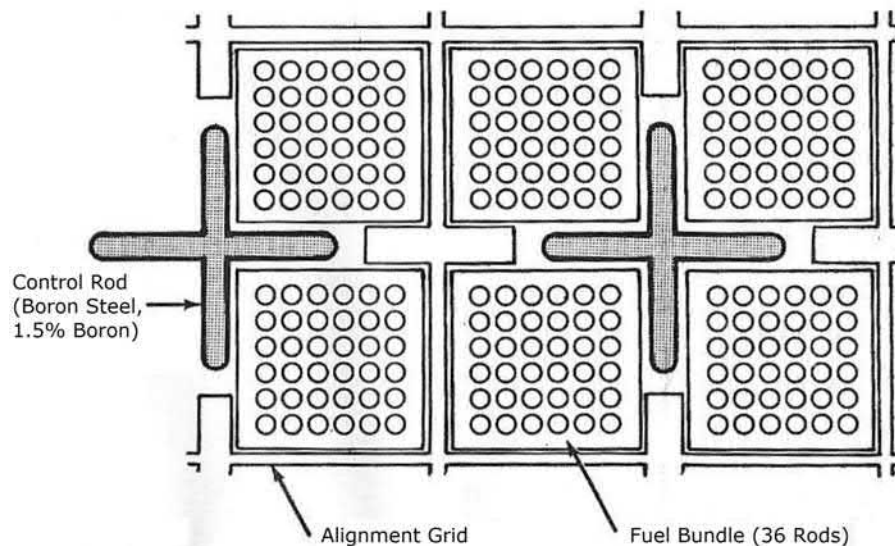


Fig. 12 Dresden-1 interior core arrangement. Kramer AW (1958) *Boiling Water Reactors*. Addison-Wesley Publishing Company Inc.

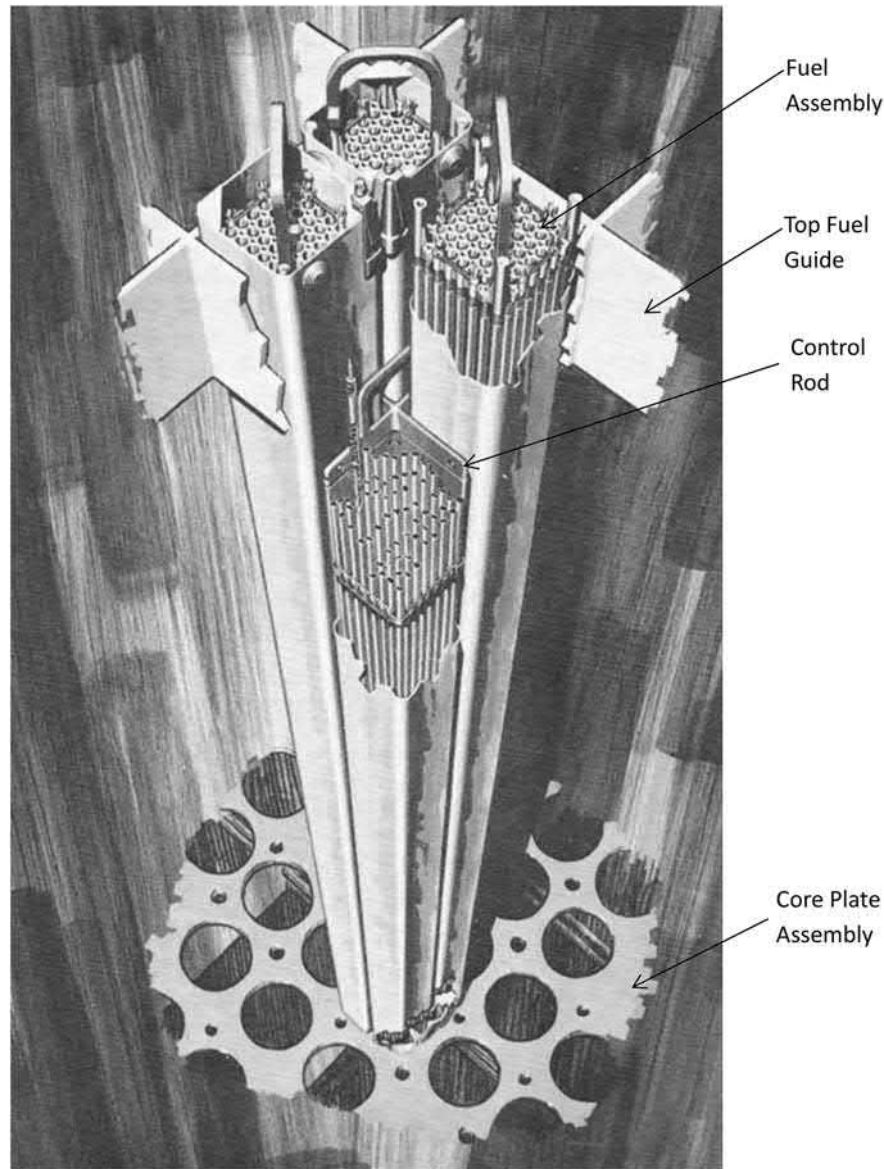


Fig. 13 BWR/6 fuel assemblies and control rod cell. Courtesy GE-Hitachi Nuclear Energy.

Control rod drive system

Dresden-1 employed bottom mounted, hydraulic Locking Piston Control Rod Drives (LPCRD) that set a trend for the Generation II BWR product lines. The Control Rod Drive (CRD) system provides controlled, notched rod movement and positioning for power shaping and burnup compensation, as well as rapid full insertion of all control rods for reactor trip (SCRAM).

Typically, a notch was discrete movement of 6 in. [15.2 cm] for the Generation II BWRs. The control rod was moved by applying pressure greater than reactor pressure to either the top or bottom of a main drive piston. The control rod was held at a given position by a collet engaging one of several notches in an index tube (Figs. 14 and 15). An accumulator provides hydraulic pressure greater than reactor pressure. Loss of electrical power to the system or reduction of accumulator pressure results in opening of scram pilot valves, thereby effecting rapid shutdown of the reactor.

The ABWR diversified the CRD to incorporate fine motion of control rods with an electric motor for normal operations, while retaining and simplifying the hydraulic scram (Fig. 16). Because the Fine Motion Control Rod Drive (FMCRD) has the additional electrical motor, it drives the control blades into the core even if the primary hydraulic system fails to do so, providing additional capability to hydraulic insertion of the control rods. The drive also includes safety enhancements against rod run out (electromechanical brake) and rod ejection (internal restraint) (see Fig. 16).

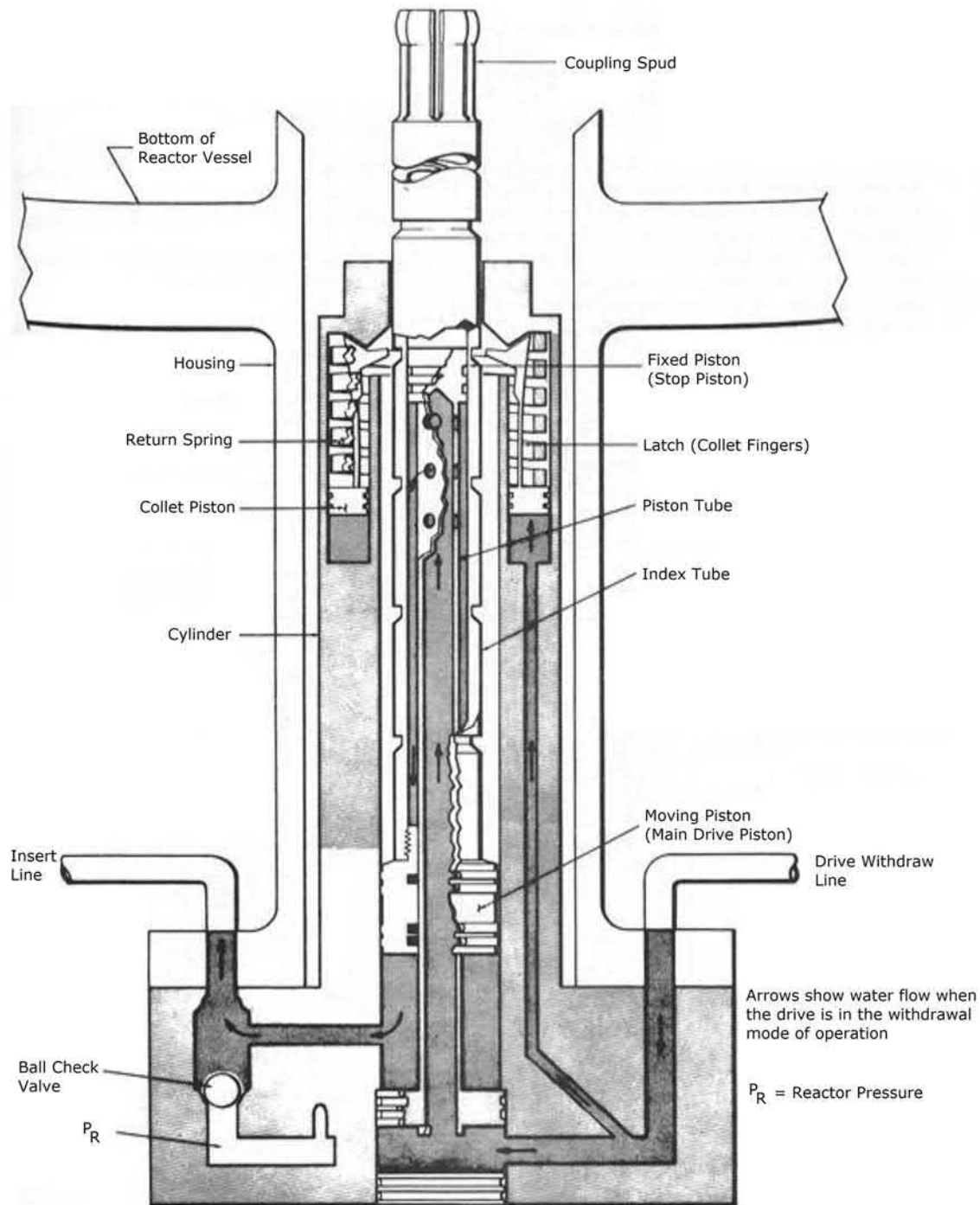


Fig. 14 BWR/6 locking piston hydraulic drive. General Electric (GE) Nuclear Energy Group (1980) BWR/6 general description of a boiling water reactor. Revised September 1980.

Recirculation system

The initial Generation II BWRs used five external recirculation loops with pumps to provide forced recirculation coolant flow through the core. Subsequently, an internal jet pump design was developed that reduced the number of external loops to two. The recirculation pumps take suction from downward flow in an annulus between a core shroud and the vessel through two recirculation nozzles. The water is pumped to higher pressure and distributed through a manifold to which a number of riser pipes are connected, that return flow through vessel inlet nozzles to be discharged to a jet pump throat. Through a momentum exchange

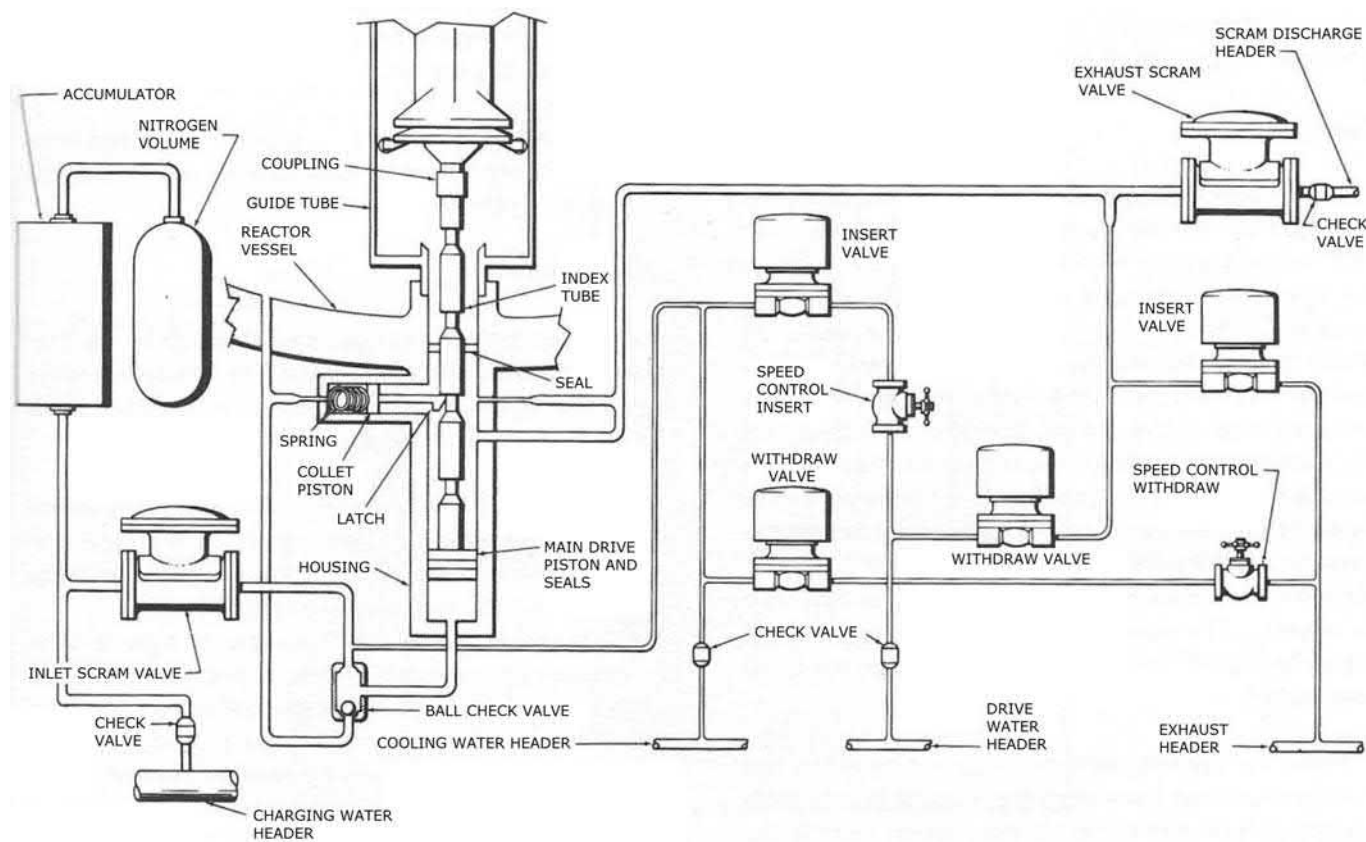


Fig. 15 BWR/6 locking piston hydraulic drive system. General Electric (GE) Nuclear Energy Group (1980) BWR/6 general description of a boiling water reactor. Revised September 1980.

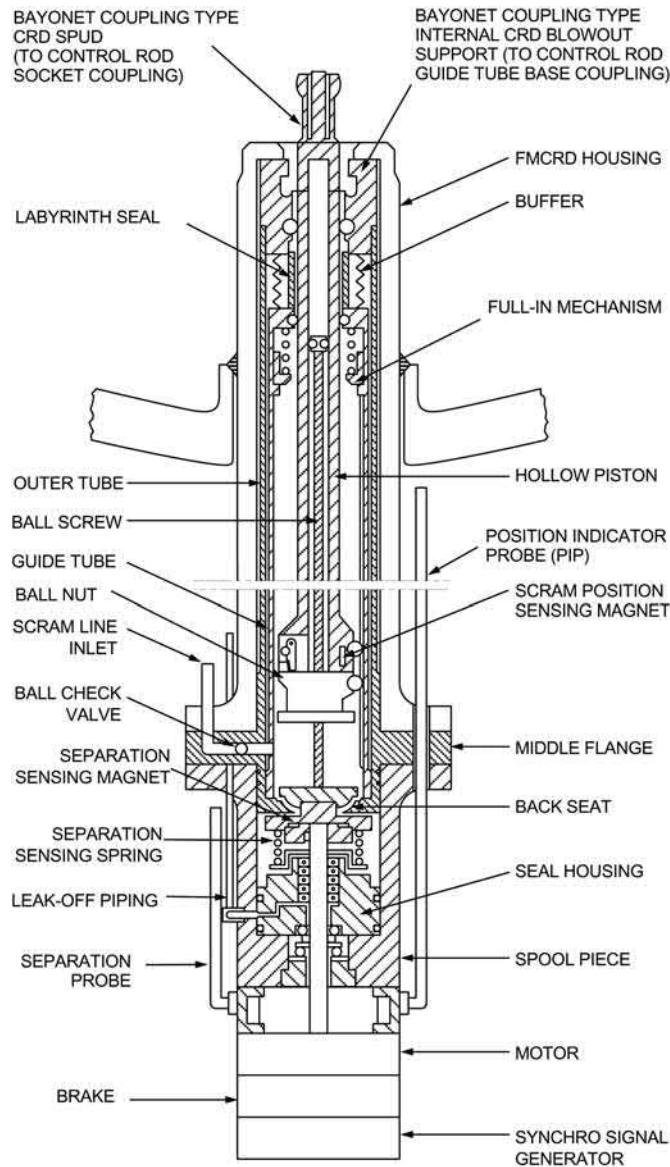


Fig. 16 ABWR fine motion control rod drive. GE-Hitachi Nuclear Energy (2007) The ABWR plant general description. Revised July 1, 2007. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ABWR%20General%20Description%20Book.pdf.

process, this flow induces surrounding water in the downcomer region to be drawn into the throat where the flows mix, diffuse into a diffuser, and finally are discharged into a lower core plenum. This recirculation system is illustrated in Fig. 17.

With the jet pump recirculation system, it became possible to reflood the core region of the vessel up to the top of the jet pump. This also provided an ECCS safety improvement, in that no recirculation line break could prevent reflooding of the core up to this level.

The ABWR design includes 10 Reactor Internal Pumps (RIPs) whose diffusers and impellers are arranged circumferentially between the shroud and the RPV near the RPV bottom head and whose motors are installed in the motor casing at the bottom of the RPV (Fig. 18). The RIPs function collectively to force the reactor coolant through the lower plenum of the reactor and upward through openings in the fuel support, upward through the fuel assemblies and steam separators, and down into the annulus to be mixed with feedwater and recirculated through the core. The RIPs are variable speed to adjust core flow. A change in RIP speed conditions will vary the core flow in the reactor, which, in turn, will change the reactor power during normal power range operation.

Ancillary BWR components

BWR power plants also contain components that are required to maintain normal operation of the plant, and to mitigate consequences during accidents.

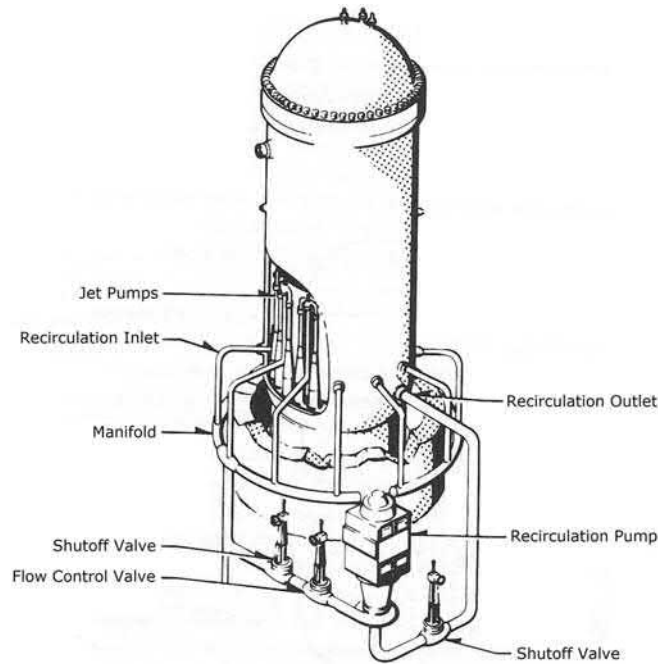


Fig. 17 BWR/6 jet pump recirculation system. General Electric (GE) Nuclear Energy Group (1980) BWR/6 general description of a boiling water reactor. Revised September 1980.

The BWR design includes uses a Feedwater Control System (FWCS) to maintain water level control within the RPV. The FWCS returns condensed steam back to the RPV and provides makeup water as needed in case of any inventory losses.

The BWR design includes a Reactivity Control System (RCS) to control movement of control rods during normal operation and in response to anticipated operational occurrences and accidents. The RCS initiates a reactor scram on a variety of signals (e.g., high neutron flux, closure of all Main Steam Isolation Valves, low reactor water level).

Reactor pressure is maintained during plant operation using a Pressure Control System (PCS). The PCS controls the positions of the turbine control valves to maintain a constant turbine inlet pressure for GE BWRs. The ASEA Atom BWRs typically use the PCS to maintain a constant steam dome pressure in the RPV.

In the event of a failure of the control rods to insert when needed, the BWR uses a Standby Liquid Control (SLC) system to inject a sodium pentaborate solution into the RPV. The boron in this solution is a neutron absorber, and the SLC system is sized to be able to bring the plant to cold shutdown without control rod insertion.

Radioactive Waste Management Systems are also included in the BWR design to process solid, liquid and gaseous sources of radiation such that offsite doses to the public remain below acceptable limits.

Balance of plant components

Balance of Plant Systems (Arcaro, 2021) for a BWR are functionally the same as those used in fossil steam plants. Major differences between Balance of Plant Systems for a BWR and a fossil steam plant are:

- a. Due to the continuous production of decay heat in the reactor core even while in shutdown, the shutdown cooling mode of the Residual Heat Removal system must remain in operation. No cooling is needed for a fossil plant during shutdown.
- b. The BWR delivers a high flow of low temperature saturated steam whereas a fossil plant delivers a smaller flow of superheated steam. This requires a "wet turbine" design, which has a different turbine blade design than those used in a fossil turbine that accepts superheated steam and results in lower energy conversion efficiency because of the lower turbine inlet temperature.

Operation of BWR

Today most BWRs are operated as large (800–1300 MWe) base load units which are maintained at or near full power conditions during their entire operating cycle (Young, 2021). Most US BWRs operate on a 24-month refueling schedule with a few plants on 18-month refueling schedules. Non-US BWRs generally operate on a 12-month refueling schedule, although there are exceptions.

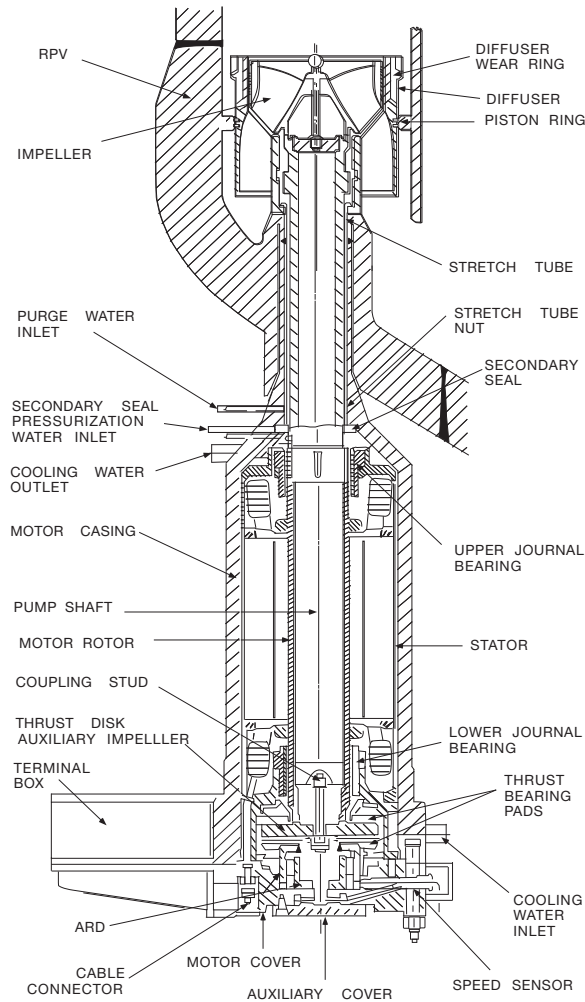


Fig. 18 ABWR reactor internal pump. GE-Hitachi Nuclear Energy (2007) The ABWR plant general description. Revised July 1, 2007. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ABWR%20General%20Description%20Book.pdf.

BWRs start an operating cycle with a mixture of partially and fully withdrawn control rods, then periodically withdraw the partially inserted control rods as needed to remain at rated power conditions. Core excess reactivity is also controlled by a burnable absorber (gadolinium) that is located in several fuel rods in each fuel assembly.

International BWR development

While the BWR prospered in the United States, BWRs were constructed in Mexico, Europe and Asia as well. In Europe, BWRs were constructed from the product lines described above in the Netherlands, Italy, Spain, Switzerland, and Germany. In Germany, Kraftwerk Union AG (founded by AEG and Siemens, and subsequently integrated into Areva) further advanced the design of the earlier imported BWRs of GE design, culminating with the construction of the Gundremmingen Nuclear Power Plant Units B and C. Features of these units included fine motion control rod drives, internal recirculation pumps, a three train full-range residual heat removal system and a pre-stressed cylindrical concrete containment with steel liner.

In 1947, Sweden established a domestic atomic energy research program that led to the country's first experimental reactor commissioned in 1954. In 1966, a Sydkraft-led consortium (OKG AB) ordered the Oskarshamn 1 BWR from ASEA Atom. This was the first western light water reactor designed and built without requiring a license from US vendors (World Nuclear Association, 2017). Several additional BWRs of increasing output were supplied by ASEA Atom (subsequently Westinghouse Electric Sweden AB, and currently owned by Toshiba Corporation) in Sweden and Finland. The first five ASEA Atom BWRs were similar in design to the GE BWR/2 product line. The last six ASEA Atom BWRs introduced fine motion control rod drives and internal recirculation pumps in their reactor design.

In Asia, BWRs were constructed from the product lines described above in Japan, India and Taiwan. India constructed two BWR/1 units at the Tarapur Atomic Power Station.

Several commercial BWRs were constructed of the BWR/2, BWR/3, BWR/4 and BWR/5 types with Mark I and Mark II containments in Japan, the first being Tsuruga Unit 1. Cooperative programs among the designers, GE, Hitachi and Toshiba introduced further improvements for the BWR in Japan, culminating with the construction of the Advanced Boiling Water Reactor (ABWR) at Tokyo Electric Power Company's Kashiwazaki-Kariwa Nuclear Power Station.

To develop the ABWR design, an Advanced Engineering Team (AET) was formed in the late 1970s to review design, construction and operating experiences in Japan, the United States and Europe. The AET consisted of a 25-member-team with representatives from a world-wide range of technology suppliers such as General Electric (United States), Hitachi and Toshiba (Japan), ASEA-Atom (Sweden) and Ansaldo (Italy). Architect engineering support was provided by NUCON of the Netherlands and Bechtel and Ebasco from the United States. The AET effort was completed in 1979.

Future of the BWR

To date, two standardized US BWR designs (ABWR and ESBWR) with thermal power ratings up to 4500 MWth have been approved through the USNRC's 10 CFR 52 design certification process for US applications (Ray, 2021). The ABWR design certification is currently undergoing a renewal review by the USNRC. A third standardized US BWR design (BWRX-300) is currently undergoing design certification review by the USNRC.

The ABWR design is currently offered by three companies: GE-Hitachi Nuclear Energy, Hitachi-GE Nuclear Energy, and Toshiba. While there are some differences in the three companies' designs, and some differences that are location specific, as mentioned above for the UK ABWR, the descriptions above have focused on key common similarities between the designs that provide a general description of ABWR.

Supplementary characteristics suggested for Generation III+ plants are simplicity and improved economics. GE-Hitachi's Economic Simplified Boiling Water Reactor (ESBWR) design, which incorporates several passive safety systems with a reduction of active components exhibits these characteristics, as well as the characteristic of improved safety. Although not yet constructed, the ESBWR is under active consideration for construction in the United States, and has also been proposed for construction in other countries. Two US utilities have received NRC approval for a Combined License that would allow them to construct the ESBWR on their approved sites. Additional details about the ESBWR are provided in the following section.

The BWRX-300 is a ~300 MWe water-cooled, natural circulation Small Modular Reactor (SMR) with passive safety systems, which represents the simplest, yet most innovative BWR design since GE began developing nuclear reactors in 1955. The BWRX-300 is based on the US NRC-licensed, 1520 MWe ESBWR and is designed to provide clean, flexible baseload electricity generation that is competitively priced and estimated to have the lifecycle costs of typical natural gas combined-cycle plants. Additional information about the BWRX-300 can be found on the [GE-Hitachi Nuclear Energy, n.d.](#) website referenced in the bibliography.

International cooperative efforts have led to today's Generation III/III+ BWR designs with improvements in safety, operation and maintenance, construction and economics. The Boiling Water Reactor is well positioned to serve in conjunction with the current operational fleet as well as emerging Generation IV reactor designs of the future.

ESBWR

The ESBWR is designed as a large-size reactor, approximately 1500 MWe, with natural circulation adopted for core coolant recirculation. The core height is reduced relative to ABWR, which reduces core pressure drop, and a "chimney" is placed in the two-phase flow region above the core to enhance driving head for natural circulation flow. Elimination of recirculation pumps also reduces pressure drop in the natural circulation flow circuit.

The ESBWR RPV is taller than the ABWR RPV with the addition of the "chimney" above the core, but the large vessel volume provides a mild response to transients. The larger steam volume during normal operation leads to a slower and lower pressure rise for pressurization transients. In order to provide enhanced margin to keep the core covered with water using the passive GDCS for LOCA events, the ESBWR core is located lower in the RPV than the ABWR core.

The number of fuel assemblies for ESBWR is increased while remaining within the RPV diameter limit established by the ABWR. Modifications were also made to the steam separator assembly based on additional test data. The US certified ESBWR design introduced 10 × 10 fuel with part length fuel rods and improvements in analytical methods, for example, the Transient Reactor Analysis Code-GE (TRACG) to assess reactor natural circulation, transient and accident conditions. Load following capability was enhanced by providing feedwater temperature control. The resulting reactor assembly is illustrated in [Fig. 19](#).

The ESBWR Passive Safety Systems are illustrated in [Fig. 20](#). The ESBWR incorporates four redundant mechanical trains of passive Gravity-Driven Cooling System (GDCS), an Automatic Depressurization System (ADS), and a Passive Containment Cooling System (PCCS). Heat removal and inventory addition are also contributed by the Isolation Condenser (IC) System and the Standby Liquid Control (SLC) System, with the latter providing reactor core shutdown as well. All mechanical trains of the passive safety systems are designed to remain operable even if one electrical division fails and another division is out of service for maintenance.

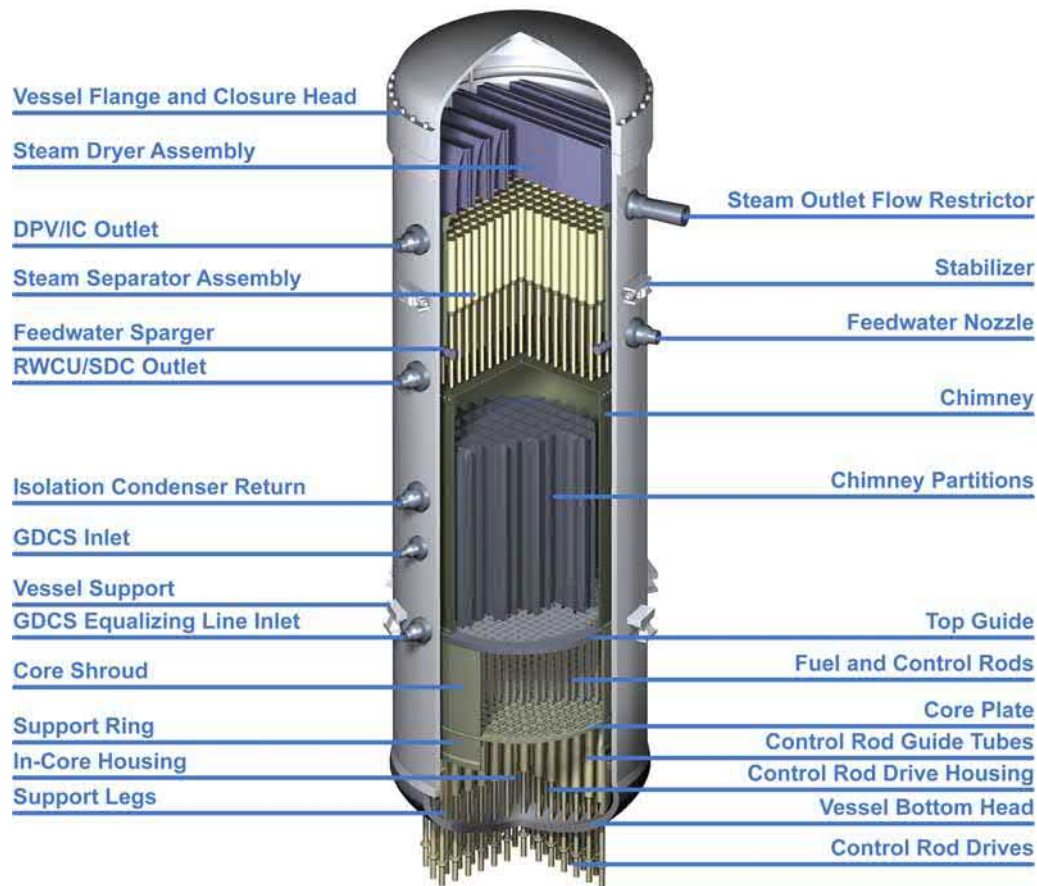


Fig. 19 ESBWR reactor assembly. GE-Hitachi Nuclear Energy (2011) The ESBWR plant general description. Revised June 1, 2011. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ESBWR_General%20Description%20Book.pdf.

High pressure inventory control and heat removal is accomplished with the use of the IC System if the reactor becomes isolated from the normal heat sink. The IC System initiates without dependence on AC power, either off-site or on-site. Elevation differences between the IC heat exchanger and RPV provide gravity driven natural circulation flow. The IC pool is sized to provide 72 h of heat removal before refilling, and beyond 72 h through simple operator action to refill the IC pool using on-site resources, even without the availability of off-site or on-site AC power. The IC System includes provisions to assure that excess hydrogen or air does not accumulate in the IC steam supply line and heat exchanger during either standby or normal IC System operation.

Each train of the GDCS consists of three independent subsystems: a short-term core makeup and cooling (injection) system from the GDCS pool, a long-term cooling (equalizing) connection from the suppression pool, and a deluge line that can provide makeup to the Basemat-internal Melt Arrest Coolability (BiMAC) core catcher, a severe accident feature. The short-term and long-term subsystems provide gravity driven cooling water to replace RPV water inventory lost during a postulated LOCA and subsequent decay heat boil off.

GDCS actuation valves are squib valves, gas propellant shear valves that are normally closed but open when a pyrotechnic booster charge is ignited using DC power. Once the squib valve is actuated, it remains open to provide a flow path to the vessel. Check valves are incorporated in the system design to mitigate consequences against spurious valve initiation. The GDCS check valves remain fully open when zero differential pressure exists across the valve, in order to minimize the potential for sticking in the closed position during long periods of nonuse.

The ADS is automatically initiated if an RPV low water level signal or if high drywell pressure is sustained for specified periods. Redundant trip channels are arranged in divisionally separated logic, such that each Depressurization Valve (DPV) or Safety-Relief Valve (SRV) receives separate initiation signals from three safety-related divisions and also a Diverse Protection System (DPS). Each SRV is initiated by a solenoid valve controlling a pneumatic actuator. Each DPV is initiated by a squib located on the valve. The ADS valve openings are staggered in time to control the blowdown rate and prevent excessive suppression pool swell and reduce dynamic loads.

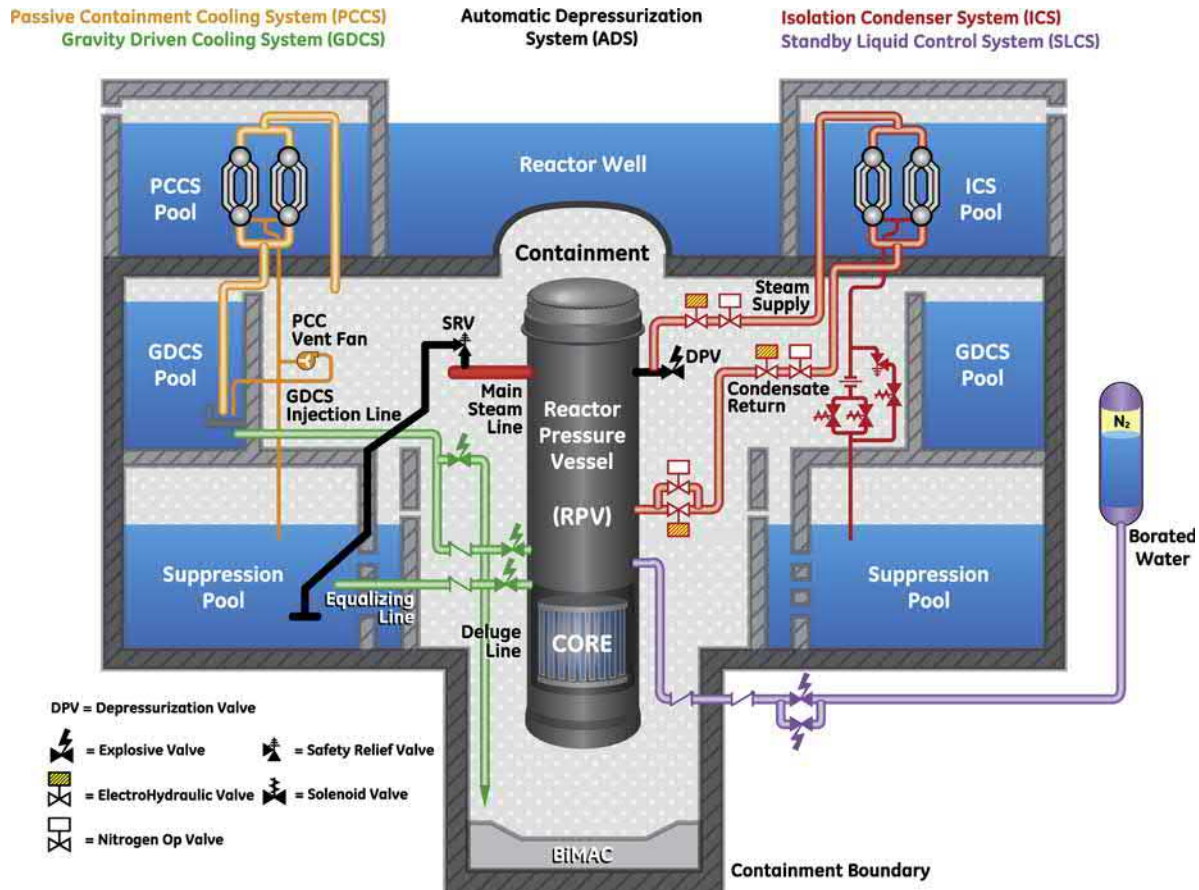


Fig. 20 ESBWR passive safety systems. GE-Hitachi Nuclear Energy (2011) The ESBWR plant general description. Revised June 1, 2011. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ESBWR_General%20Description%20Book.pdf.

The PCCS maintains the containment within its pressure limits for Design Basis Accidents (DBAs). The PCCS consists of six independent trains, each containing a steam condenser. The PCCS heat exchangers receive steam-gas mixture supply directly from the containment drywell, driven by pressure difference created between the drywell and the suppression pool during LOCA. Steam condensed in the PCCS heat exchanger tubes is returned to the GDSC pool by gravity, and subsequently to the RPV. Gases are returned by a vent line to the suppression pool airspace. Together with the pressure suppression containment, the PCCS condensers limit containment pressure to less than its design pressure for 72 h after a LOCA with makeup to the IC/PCC pool, and beyond 72 h with pool makeup and PCCS vent fan operation.

The PCCS vent fans aid in long-term removal of non-condensable gas from the PCCS condensers for continued condenser efficiency. The discharge of each PCCS vent fan is submerged in the GDSC pool drain pan to prevent backflow that could otherwise interfere with the normal venting of the PCCS. The fans are manually initiated by operator action and are powered by an ancillary diesel generator.

The SLC System provides a backup method to bring the nuclear reactor to subcriticality and to maintain subcriticality as the reactor cools. It is automatically initiated in case of signals indicative of LOCA or Anticipated Transient Without SCRAM (ATWS). The SLC System is a passive system incorporating two identical and separate trains using pressurized accumulators to inject borated water rapidly and directly in the bypass volume of the core.

The ESBWR cutaway showing the Reactor Building, Fuel Building, Control Building and Turbine Building, consistent with the US certified design, is illustrated in Fig. 21. The Reinforced Concrete Containment Vessel (RCCV) is cylindrical and consists of a top slab, a shell incorporating a raised suppression pool, lower drywell and supporting foundation, and is integrated with the Reactor Building to improve seismic stability and resistance. The Reactor Building is designed to provide a secondary containment. The IC/PCC pools, consisting of inner and outer replenishing pools, are located above the RPV in the Reactor Building. To provide for IC/PCC volume in the reactor building, the spent fuel pool is located in a Fuel Building adjacent to the Reactor Building and shares a common basemat. Fuel is transported to the spent fuel pool using an Inclined Fuel Transfer System.

Additional information about the ESBWR can be found in the ESBWR Plant General Description (GE-Hitachi Nuclear Energy, 2011).

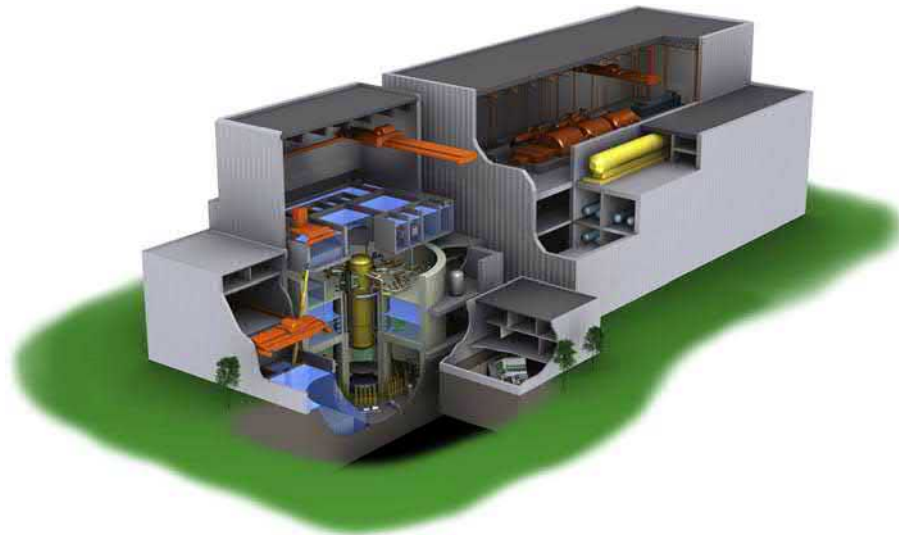


Fig. 21 ESBWR cutaway of reactor building, fuel building, control building and turbine building. GE-Hitachi Nuclear Energy (2011) The ESBWR plant general description. Revised June 1, 2011. https://nuclear.gepower.com/content/dam/gepower-nuclear/global/en_US/documents/ESBWR_General%20Description%20Book.pdf.

Acknowledgment

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See Also: Advanced Light Water-Cooled Reactor Concepts; Commercial Nuclear Power Reactors and Their Design—Introduction; Floating Nuclear Power Plants With Small Modular Reactors; Light Water Reactors (LWR) Fuel Cycle Options; Pros and Cons of Commercial Reactor Designs: Part 2—Current Status and Future Trends in the World Nuclear-Power Industry and Technical Considerations of Nuclear-Power Reactors; Water Cooled Small Modular Reactors (Integral PWR and BWR).

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Heavy-Water Reactors

Eleodor Nichita, Faculty of Energy Systems and Nuclear Science, Ontario Tech University, Oshawa, ON, Canada

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Glossary

Criticality The constant-power operating condition of a reactor, in which nuclear fuel sustains a steady-state fission chain reaction. A reactor achieves criticality (and is said to be *critical*) when neutrons released in a fission event induce exactly one other fission.

Reactivity A measure of a reactor's departure from criticality. A reactor with positive reactivity is said to be *supercritical* and its power increases. A reactor with negative reactivity is said to be *subcritical* and its power decreases.

Abbreviations

ACR Advanced CANDU Reactor
AR Adjuster Rod
CANDU Canada Deuterium Uranium
DCC Digital Control Computer
ECCS Emergency Core Cooling System
HWR Heavy Water Reactor
LIN Liquid Injection Nozzle
LWR Light Water Reactor
MCA Mechanical Control Absorber
PHTS Primary Heat Transport System
RIH Reactor Inlet Header
ROH Reactor Outlet Header
RRS Reactor Regulating System
SDS1 Shutdown System 1
SDS2 Shutdown System 2

SG Steam Generator
 SHTS Secondary Heat Transport System
 SOR Shutoff Rod
 ZCU Zone Control Unit

Introduction

Heavy-water reactors (HWRs) are nuclear reactors that are moderated¹ (and, possibly, also cooled) by heavy water. Heavy water, chemical formula D₂O, differs from regular, or light, water, by having two deuterium atoms instead of the usual two hydrogen ones. Deuterium is an isotope of hydrogen whose nucleus contains both a proton and a neutron. Because heavy water has better moderating properties than light water, heavy-water moderated lattices can attain criticality for lower uranium enrichments than light-water ones, including for natural uranium, which has only $\sim 0.7\%$ ²³⁵U.

A substance's moderating properties are expressed by the *moderating ratio*, which captures three of the desirable properties of a moderator: large decrease in neutron energy (increase in neutron lethargy) per collision, large scattering microscopic cross section, and small absorption cross section. The moderating ratio is calculated by taking the product between the average lethargy increment with the scattering cross section and dividing by the absorption cross section. Heavy water has a moderating ratio of ~ 5500 , compared to ~ 70 for light water (see Wietfeldt, 2021).

The high moderating ratio of heavy water is due primarily to heavy water's very low absorption cross section. A consequence of that is that the infinite multiplication constant of a UO₂-heavy-water lattice keeps increasing with the moderator-to-fuel volume ratio up to values of ~ 20 , much higher than the value of ~ 2 in a light-water reactor (LWR). Consequently, HWRs have higher moderator-to-fuel ratios than LWRs and, therefore, generally larger cores for a given power level, implying significantly lower average power densities than LWRs (~ 10 MW/m³ vs. ~ 50 MW/m³). The large amount of heavy water impacts the dynamic properties of HWRs as well. In particular, HWRs have a longer neutron generation time² than LWRs (~ 0.9 ms vs. ~ 0.05 ms) and a small number of additional delayed neutrons³ (~ 0.0003 per fission), in the form of *photoneutrons* resulting from the reaction of gamma rays with deuterium.

In systems where the coolant and moderator are physically separated and where coolant represents only a small fraction of the heavy-water volume in the core, loss of coolant leads only to a marginal decrease in moderation. In such systems, because of the reduction in absorption and because of subtle neutron-spectrum effects, the reactivity of the core increases following loss of coolant unaccompanied by loss of moderator. This effect is mitigated by the long generation time, resulting in slow transients, which can be safely terminated by engineered shutdown systems. As an additional measure of safety, some designs employ two separate shutdown systems.

Because the heavy water is exposed to a high neutron flux, deuterium can absorb a neutron and transmute into tritium, which undergoes beta decay (maximum energy 18.6 keV) with a half-life of ~ 12 years and can be of radiological concern if ingested or inhaled. The tritium activity in the coolant can reach ~ 1.4 Ci/kg (Son et al., 2009). Its release is usually prevented by the use of advanced seal technology for the coolant and moderator systems. Some plants also employ a Tritium Removal Facility (TRF) whereby tritium is separated from tritiated heavy water by catalytic exchange followed by cryogenic distillation, and is subsequently stored as titanium hydride (Son et al., 2009) for subsequent industrial use (e.g., luminescent signs).

High heavy-water purity (less than 1% light water content in moderator) is essential for maintaining good neutron economy. A distillation facility is used to upgrade the isotopic content, as necessary.

Over the years, a number of heavy-water power reactor designs were developed in Canada, the UK, India, Germany and Argentina. The most common is the Canadian CANDU⁴ (CANada Deuterium Uranium) design. There are several variants of the CANDU design, of which the CANDU-6 is the closest to a reference design and also the one with the widest worldwide deployment. This overview of heavy-water power reactors therefore consists of a fairly detailed presentation of the CANDU-6 reactor system and a summary of the other designs and how they differ from the CANDU-6.

The CANDU-6 reactor system

The CANDU-6 reactor is a pressure-tube HWR which uses heavy water as both moderator and coolant. However, the two are physically separated and belong to different circuits. CANDU-6 fuel consists of fuel bundles approximately 0.5 m long. Each fuel bundle

¹Moderation is the process of neutron slowing down, usually by collision with light nuclei.

²The neutron generation time can be interpreted as the average "age" (time since "birth" through fission) of neutrons in a reactor. See also Tsvetkov (2021) and Hamon (2021).

³Delayed neutrons are neutrons emitted after the fission event, as a consequence of fission-product decay. See also Tsvetkov (2021) and Hamon (2021).

⁴CANDU is a trademark of AECL and SNC Lavalin.

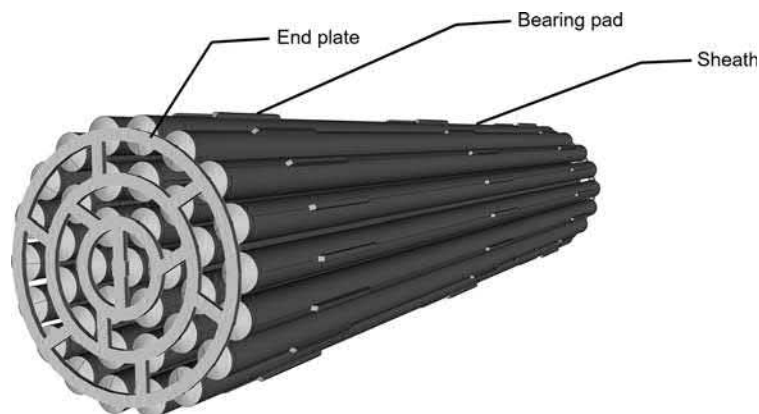


Fig. 1 CANDU-6 fuel bundle.

has 37 fuel elements arranged in three concentric rings (6, 12 and 18 elements each) plus a central element (**Fig. 1**). A fuel element consists of UO_2 fuel pellets approximately 1 cm in diameter, encased in a collapsible Zircaloy sheath. The elements forming a bundle are held together by end plates and are prevented from rubbing against each other by spacers. The diameter of a fuel bundle is approximately 10 cm. Fuel-bundle parameters are shown in **Table 1** (IAEA, 2002).

Fuel bundles reside in horizontal pressure tubes made of a Zr-Nb alloy, each holding 12 fuel bundles. It follows that pressure tubes are approximately 6 m long. Heavy water coolant runs through the pressure tubes and cools the fuel during reactor operation. Pressure tubes are arranged in a rectangular array with a pitch of 28.575 cm (a trade-off between excellent moderation and the cost of heavy water) and are immersed in the heavy-water moderator contained in a large, horizontal, cylindrical non-pressurized vessel, called the *calandria*. Unlike the coolant, the moderator is kept at near-atmospheric pressure and $\sim 70^\circ\text{C}$. To avoid heat loss from the hot coolant to the cooler moderator, each pressure tube is surrounded by a slightly-larger-diameter tube (also made of Zr-Nb alloy), called the calandria tube. There is a gas-filled gap between the two tubes which acts as a thermal insulator to prevent heat loss from the coolant to the moderator. The gas can also be monitored for heavy-water vapor which can indicate a crack/leak in the pressure tube, thus allowing preventative measures before a break occurs. The pressure tube and the calandria tube form the fuel channel. Because the pressure tube holds the weight of the fuel, while the calandria tube doesn't, the pressure tube would normally tend to sag and possibly touch the calandria tube. This is prevented by spacers (garter springs) placed at several axial locations between the pressure and calandria tubes. A CANDU-6 reactor has 380 fuel channels and 4560 fuel bundles. A simplified diagram of the CANDU-6 reactor is shown in **Fig. 2**.

CANDU reactors are refueled on-line. New fuel is pushed into the channel at one end and burnt fuel is removed from the channel at the other end. On refueling, fuel is pushed in the same direction as the coolant flow (fueling with flow). While refueling against flow was used at one point in some CANDU units, currently all CANDU units are fueled with flow. Because flow and refueling direction alternate from one channel to the next, CANDU reactors are said to have bi-directional refueling.

Fueling is performed using fueling machines. A refueling operation involves two fueling machines, one at each end of the reactor. The first machine initially contains the fresh fuel, while the second machine is empty. After refueling, the first machine is empty, while the second machine contains the burnt fuel. Each fueling machine consists of a machine head that attaches to the fuel channel, a revolving fuel magazine, and a ram that pushes the fuel into the fuel channel. The refueling operation is automated and computer-controlled.

Table 1 CANDU-6 fuel bundle characteristics.

Fuel material	Natural UO_2
Bundle Weight	24 kg
Uranium Weight per Bundle	19.2 kg
Sheath Outside Diameter (cold)	13.1 mm
Sheath Thickness	0.4 mm
Sheath Material	Zircaloy 4
Elements per Bundle	37
Bundle Length	495.3 mm
Bundle Outside Diameter	102.4 mm

From IAEA (2002) Technical Reports Series No. 407, 2002. *Heavy Water Reactors: Status and Projected Development*. International Atomic Energy Agency. Retrieved from: https://www-pub.iaea.org/MTCD/publications/PDF/TRS407_scr/D407_scr1.pdf.

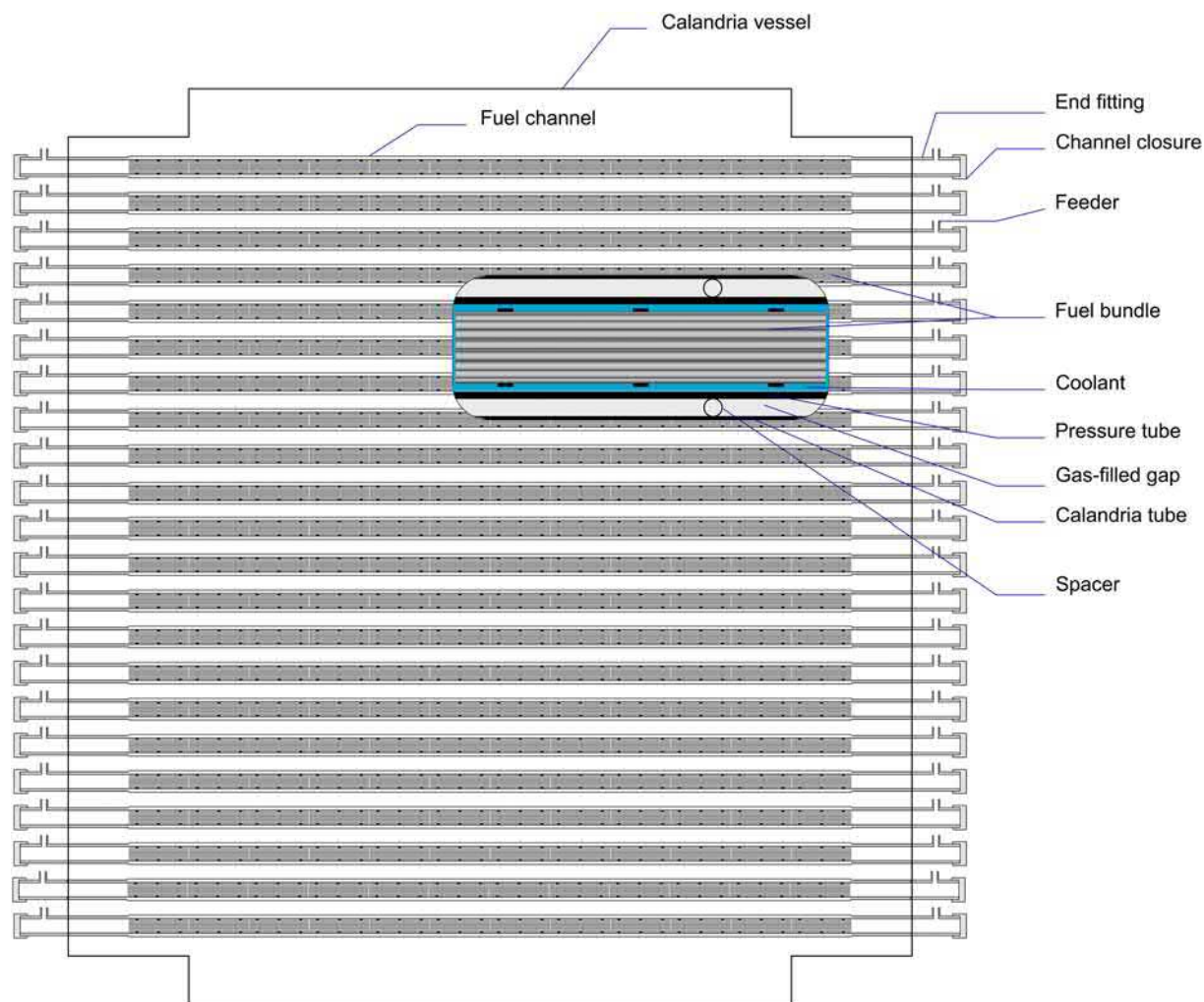


Fig. 2 CANDU-6 simplified reactor diagram.

Primary and secondary circuits

In a CANDU reactor, heavy-water coolant runs through the pressure tubes at high temperature ($\sim 280^{\circ}\text{C}$) and high pressure ($\sim 12\text{ Mpa}$). Coolant in adjacent channels flows in opposite directions, in a checkerboard pattern. Coolant reaches each pressure tube through an inlet *feeder* pipe and leaves it through an outlet *feeder* pipe. Inlet feeders draw their coolant from inlet *headers* and outlet feeders send their coolant to outlet *headers*. A CANDU-6 unit has two separate cooling loops, each comprising 190 channels. Each loop has two passes comprising channels in which coolant flows in the same direction. Each pass also includes a pump on the inlet (cold) side and a steam generator on the outlet (hot) side. The pressure tubes, piping, pumps and steam generators make up the primary circuit, also called the Primary Heat Transport System (PHTS). A simplified diagram of the PHTS is shown in Fig. 3.

A pressurizer controls the pressure in the primary circuit when the reactor is at power. A feed-and-bleed circuit can adjust the coolant inventory when that is made necessary by the change in coolant specific volume during reactor heat-up or cool-down. The pressurizer is isolated from the PHTS at low reactor power and during shutdown. The main parameters of the PHTS are shown in Table 2, adapted from (IAEA, 2002).

The moderator in the calandria vessel is part of a sealed moderator circuit which also comprises a heat exchanger and moderator pump. The heat exchanger is used to remove the moderator heat that comes primarily from gamma radiation emitted during operation. The pressure in the calandria vessel is near atmospheric pressure and four relief pipes are provided, each terminated with a rupture disk designed to break if the moderator pressure increases beyond a prescribed limit.

The secondary circuit, also called the Secondary Heat Transport System (SHTS), is similar to that of a LWR. It consists of the secondary side of the 4 steam generators, steam lines, turbines, condenser and feedwater pump and lines. A simplified diagram of the SHTS is shown in Fig. 4 and its main parameters are shown in Table 3 (IAEA, 2002).

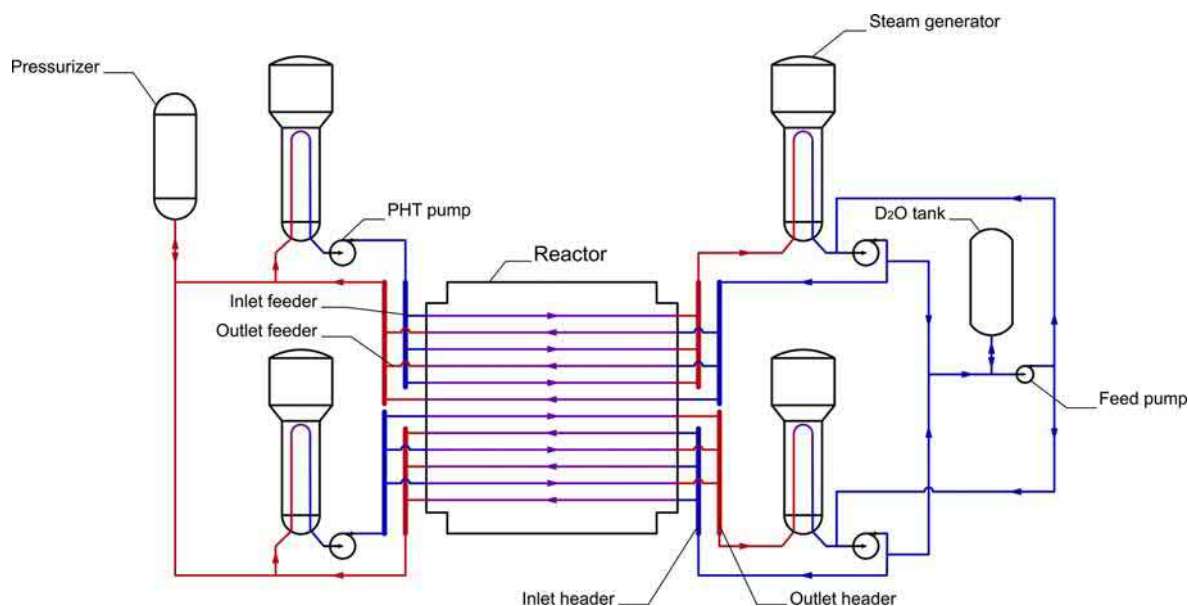


Fig. 3 CANDU-6 simplified PHTS diagram.

Table 2 CANDU-6 PHTS parameters.

Thermal output	2064 MW (th)
Coolant Flow Rate	7700 kg/s
Design Temperature (RIH)	279 °C
Design Pressure (RIH)	12.7 MPa (abs)
Design Temperature (ROH)	318 °C
Design Pressure (ROH)	11.0 MPa (abs)
Number of PHT Pumps	4
Pump Rated Capacity	2228 l/s
Pump Rated Head	215 m
Pressure Tube Inside Diameter	103.4 mm
Core Length	5.94 m
Number of Pressure Tubes	380
Pressure Tube Coolant Flow (nominal)	24 kg/s
Pressure Drop (12 bundles)	758 kPa

Adapted from IAEA (2002) Technical Reports Series No. 407, 2002.
Heavy Water Reactors: Status and Projected Development. International Atomic Energy Agency. Retrieved from: https://www-pub.iaea.org/MTCD/publications/PDF/TRS407_scr/D407_scr1.pdf.

Operational considerations

Online refueling

The main method of maintaining core criticality in a CANDU reactor is refueling. CANDU reactors are refueled online. In a CANDU-6 unit, each channel is refueled on average every 6.75 months, which corresponds to almost two refueling operations taking place each day. The standard refueling scheme is a push-trough scheme which replaces 8 of the 12 bundles. However, if a defective bundle needs to be removed, all 12 bundles can be replaced in a single refueling operation.

Between refuelings, core criticality is maintained by 14 Zone-Control Units (ZCUs) which are vertical tubes filled with light water. By slowly decreasing the water level in ZCUs, the core reactivity⁵ is increased to compensate the decrease due to fuel burnup. When a channel is refueled, the water level in ZCUs is increased almost in step fashion. The total reactivity worth of the 14 ZCUs is ~700 pcm.

⁵Reactivity is the relative imbalance between the neutron production rate and the neutron loss rate in the core. It is measured in pcm. One positive pcm corresponds to the situation where the neutron production rate is 0.001% larger than the neutron loss rate. See also Tsvetkov (2021) and Hamon (2021).

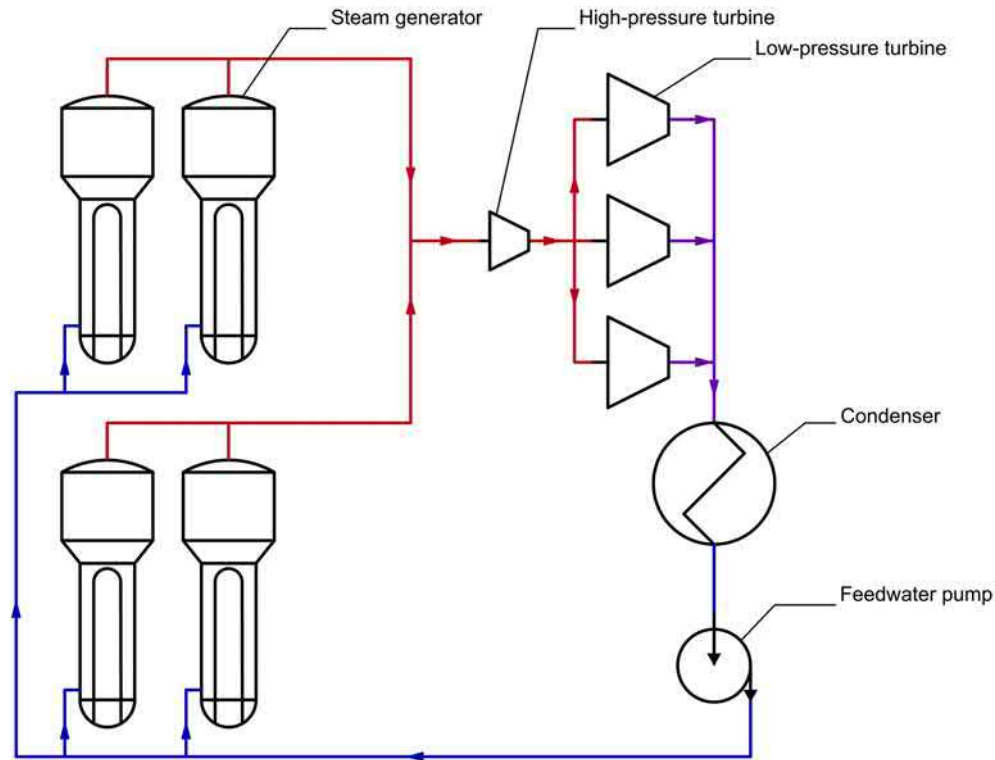


Fig. 4 CANDU-6 Simplified SHTS Diagram.

Table 3 SHTS parameters (IAEA, 2002).

Number and type of steam generators	4, vertical, inverted U tubes
Total Steam Flow (4 SG)	1033 kg/s
Steam Pressure (full power)	4.7 MPa (abs)
Steam Temperature (full power)	260 °C
Maximum Moisture	0.25%
Feedwater Temperature	187 °C
Number of High-Pressure Turbines	1
Number of Low-Pressure Turbines	3
Energy Conversion Efficiency (gross/net)	35%/33%

From IAEA (2002) Technical Reports Series No. 407, 2002. *Heavy Water Reactors: Status and Projected Development*. International Atomic Energy Agency. Retrieved from: https://www-pub.iaea.org/MTCD/publications/PDF/TRS407_scr/D407_scr1.pdf.

Equilibrium core

When a reactor is initially commissioned, it usually starts with a core full of fresh fuel, which is slightly supercritical. The excess reactivity is compensated by the addition of poison (boron) in the moderator. Poison is slowly removed from the moderator as fuel burns. Fine reactivity control is achieved by the ZCUs. Once all the poison has been removed from the moderator, reactor fueling begins (after ~100 Full-Power Days). As refueling continues, the core properties at each bundle position vary continuously due, on one hand, to fuel burnup and, on the other hand, to continuous refueling. Over a few months, the core reaches a state whereby properties and power at each bundle position vary only little with respect to some average value. Such a core state is known as the *equilibrium core*. The average burnup, average *dwelt time* (inverse of refueling frequency) for each channel and average power distribution in the equilibrium core are found using a *time-average* calculation. By contrast, the burnup and power distribution at any given time are determined from an *instantaneous* calculation based on *core-follow* which accounts for the actual refueling times of each channel and specific instantaneous burnup of each fuel bundle. The core-average discharge burnup at equilibrium is ~7.5 MWd/kg(U). The core is usually divided in two burnup regions, an inner one and an outer one. The inner region has a higher discharge burnup than the outer region in order to achieve a level of power flattening in the core and improve the hot-channel

factor.⁶ The individual average dwell time of each channel depends on its prescribed average discharge burnup and on the average power level in that channel. The average dwell time increases with the former and decreases with the latter. Although the discharge burnup of CANDU reactors is only a fraction of that of LWRs, their fuel utilization is slightly better and, consequently, they consume slightly less natural uranium per unit of generated energy. That is because the ratio of the LWR-to-CANDU discharge burnup is slightly lower than the ratio of the LWR-to-CANDU enrichment. The low CANDU discharge burnup compared to LWR results in a correspondingly larger (~ 6 times) amount of used fuel produced per unit of generated energy.

Daily refueling

While the average refueling frequency for each channel is found using the time-average core model, the practical decisions on which specific channels to refuel in any given week have to account for additional considerations, such as:

1. Time elapsed since a channel's last refueling.
Channels for which this exceeds the time-average dwell time for that channel are prioritized for refueling.
2. Average burnup of last 8 bundles (as determined from an *instantaneous* calculation) in the channels (the ones to be discharged in a refueling operation).
Channels for which this burnup exceeds the time-average discharge-burnup value for that core region are prioritized for refueling.
3. Channel power.
Channels with power lower than the time-average one are given preference.
4. Zone power.
A CANDU-6 core is formally divided into 14 *power zones*. Channels located in a zone whose current power (as determined from an *instantaneous* calculation) is lower than its time-average power are given preference.
5. Overall core power symmetry and closeness to the time-average power distribution.
Preference is given to channels whose refueling ensures overall power symmetry and brings the instantaneous power shape close to the time-average power shape.
6. Resulting change in the ZCU fills.
Following a refueling operation, ZCU fills will adjust automatically both to ensure core criticality and to ensure symmetry of the power distribution. Individual ZCUs will fill to different levels depending on which channels are refueled and how that changes the core reactivity and power distribution. Channels are selected for refueling with an aim to avoid putting excessive demands on the ZCU system as it tries to maintain core criticality and compensate for flux distortions. The goal is to maintain the individual ZCU fills between 20% and 70% and the average ZCU fill between 40% and 60%.

If refueling cannot be performed due to fueling-machine unavailability or any other reason, the core becomes subcritical in less than 1 day. To avoid shutting down the reactor in such circumstances, core reactivity can be increased by removing banks of *adjuster rods* from the core. By sequentially withdrawing banks of adjuster rods, the reactor can be maintained critical for a month without refueling. The reactivity worth of any bank of adjuster rods is slightly smaller than the total reactivity worth of the ZCUs, in order to maintain the latter's ability to keep the core critical. Adjuster rods are *gray* rods made of stainless steel or cobalt that reside in the core during normal operation and also serve as a second means (besides differential discharge burnup) of flattening the power distribution and hence decrease the hot-channel factor. Adjuster rods can also be withdrawn to provide Xe reactivity override,⁷ to allow for reactor restart up to 30 min after a spurious shutdown (subject to regulatory constraints).

Control and safety systems

CANDU-6 reactors have completely separate control and shutdown systems. The control system is known as the Reactor Regulating System (RRS). There are two separate shutdown systems known as SDS1 and SDS2. The RRS consists of a set of routines running on a digital control computer, while shutdown systems employ hard-wired circuitry. The reactor regulating system and the shutdown systems each employ an array of detectors and each operate a series of reactivity devices.

Reactivity devices

CANDU-6 reactors employ six types of reactivity devices, as follows.

- Zone Control Units
Zone Control Units (ZCUs) consist of a vertical tube which can be filled with light water, which acts as a mild absorber.⁸ There are 14 ZCUs, one per power zone. ZCUs are part of the Reactor Regulating System. They serve to maintain core criticality between refuelings, adjust the power level and shape, and effect slow power maneuvers.

⁶The hot-channel factor is defined as the ratio of the maximum heat flux in the core to the average heat flux in the core.

⁷¹³⁵Xe is a fission product and a strong neutron absorber. Its concentration increases temporarily after the reactor is shut down. In order to be able to restart the reactor a short time after it has been shut down, ¹³⁵Xe absorption needs to be *overridden* by removing some other absorbers from the core.

⁸Because of the large amount of heavy water in the lattice, neutrons are already very well thermalized and the addition of light water simply increases absorption, without significantly altering the spectrum.

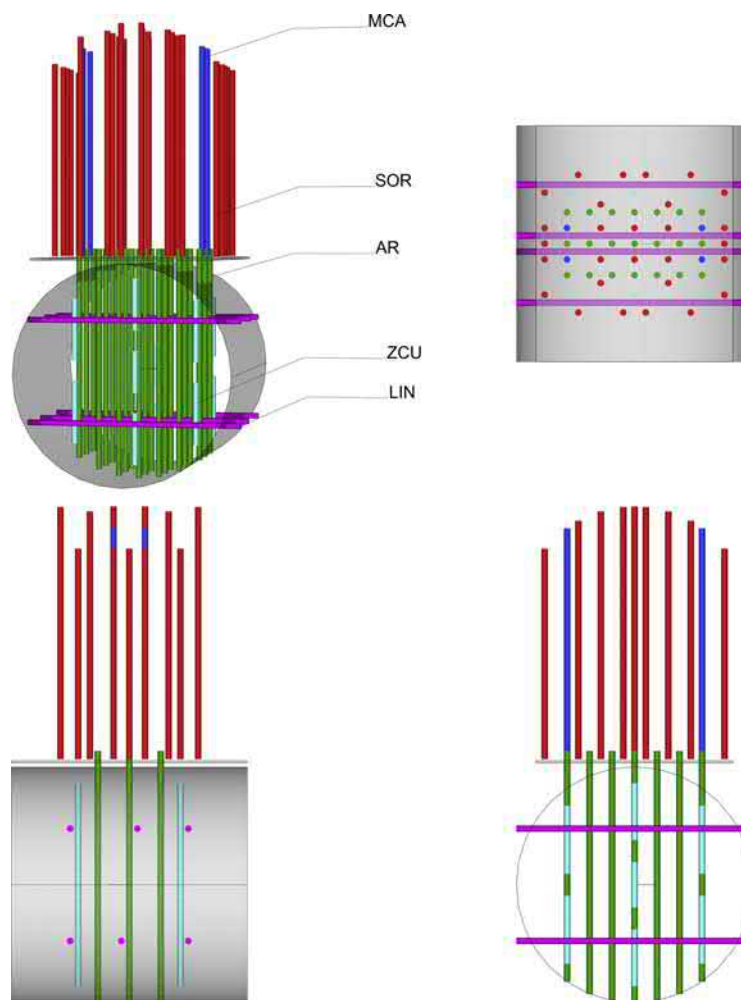


Fig. 5 Location of CANDU-6 reactivity devices during normal operation. MCAs are shown in blue, ZCUs in light blue, SORs in red, ARs in green, and LINS in fuchsia. Guide tubes are not shown.

- Adjuster Rods

The 21 vertical Adjuster Rods (ARs) are stainless-steel “gray” rods⁹ which are positioned in the core during normal operation, inside guide tubes located between fuel channels in the unpressurized calandria vessel, as shown in Fig. 5. They can be driven in and out of the core using electrical motors. Adjuster rods are part of the Reactor Regulating System.

- Shutoff Rods

The CANDU-6 reactor has 28 vertical Shutoff Rods (SORs), which are “black” rods consisting of cadmium cylinders sandwiched between concentric stainless-steel ones. They are positioned above the core during normal operation, are spring-loaded and can be dropped into the core (with spring assist) inside guide tubes located between fuel channels, in the unpressurized calandria vessel. Shutoff rods are part of SDS1.

- Mechanical Control Absorbers

The CANDU-6 reactor has 4 vertical Mechanical Control Absorbers (MCAs), which are physically identical to the shutoff rods and reside outside the core during normal operation. They can be driven into the core for fast power decrease and can also be dropped into the core. MCAs are part of the Reactor Regulating System.

- Poison Injection Nozzles

A CANDU-6 core has six horizontal Liquid-poison (gadolinium nitrate solution) Injection Nozzles (LINS) located in the calandria vessel. Poison injection is part of SDS2. When SDS2 is triggered, a fast valve releases pressurized helium from a tank into the poison tanks, pushing poison into the unpressurized moderator.

⁹Rods made of materials characterized by modest neutron absorption are called “gray” rods. Rods made of materials characterized by strong neutron absorption are called “black” rods.

- Moderator Poison

Moderator poison (boron) is usually used to ensure guaranteed shutdown and is slowly removed during reactor startup to bring the reactor to criticality. It is also used when a reactor is first started up with a fresh core and until refueling equilibrium is reached. It is not used during normal operation after refueling equilibrium is reached. Moderator poison is part of the Reactor Regulating System.

A synopsis of reactivity devices is given in [Table 4](#) ([Rouben, 2002](#)) and a diagram of their location is shown in [Fig. 5](#).

It is important to note that all rods are perpendicular to the main axis of the cylindrical reactor core and that they reside or are inserted in guide tubes located between fuel channels in the unpressurized calandria vessel ([Fig. 6](#)) and thus cannot be accidentally ejected from the core.

Detectors

CANDU-6 reactors use a vast array of detectors for control and shutdown purposes, including proportional counters, ion chambers, and self-powered flux detectors (see [Han, 2021](#)). There are 174 active in-core detectors and 9 out-of-core detectors. Detectors are grouped into two separate groups depending on whether they are used for control or for shutdown. No detector belongs to both groups. Additionally, detectors for SDS1 and SDS2 are also separate. A list of detectors and their functions is given in [Table 5](#), adapted from ([Rouben, 2002](#)).

Reactor regulating system

CANDU-6 reactors have a fully digital control system. The Reactor Regulating System is a set of routines that run on a Digital Control Computer (DCC). There are two such DCCs, the first of which is active while the second one is in hot standby, able to take over seamlessly if the first one becomes unavailable. For power control purposes, the CANDU-6 core is divided into 14 power zones. Each zone has one ZCU and one detector assembly containing two (one active and one spare) fast self-powered platinum in-core detectors as shown in [Fig. 7](#). The power level in each zone is read by the respective detectors and is adjusted by lowering (for zone power increase) or raising (for zone power decrease) the water level in the respective ZCU.

Table 4 CANDU-6 reactivity devices ([Rouben, 2002](#)).

Function	Device	Total reactivity worth (pcm)	Maximum reactivity rate (pcm/s)
Control	14 Liquid Zone Controllers	700	± 14
Control	21 Adjuster Rods	1500	± 10
Control	4 Mechanical Control Absorbers	1000	± 7.5 (driving) -350 (dropping) - 1 (extracting)
Control	Moderator Poison		- 5000
Shutdown (Safety)	28 Shutoff Rods	8000	- 5000
Shutdown (Safety)	6 Liquid-Poison Injection Nozzles	> 30,000	- 5000

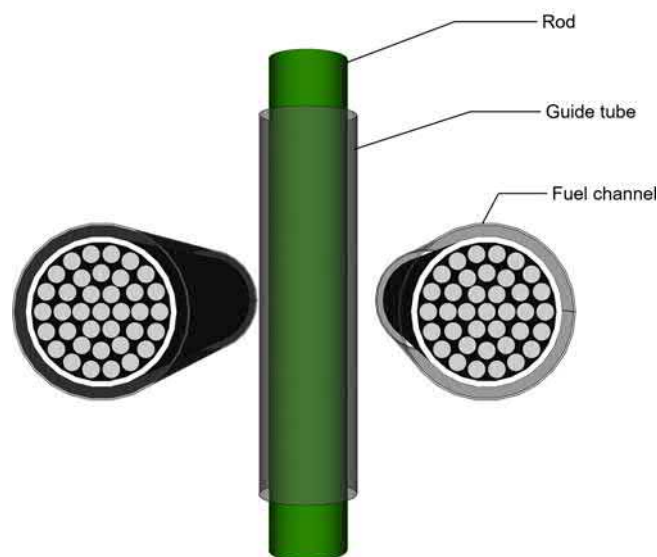


Fig. 6 Location of a rod and guide tube between fuel channels.

Table 5 CANDU-6 neutron detectors.

Purpose	Type	Location	Number
RRS	boron-lined ion chamber	out of core	3
SDS1	boron-lined ion chamber	out of core	3
SDS2	boron-lined ion chamber	out of core	3
RRS (zone powers)	platinum self-powered	in core, vertical	14
RRS (flux mapping)	vanadium self-powered	in core, vertical	102
SDS1	platinum self-powered	in core, vertical	34
SDS2	platinum self-powered	in core, horizontal	24

Adapted from Rouben B (2002) *Reactor-Physics Analysis Basis for Current CANDU*. Presentation to US Nuclear Regulatory Commission. Retrieved from: <https://www.nrc.gov/docs/ML0300/ML030020086.pdf>.

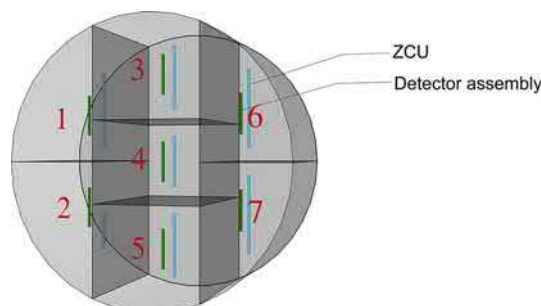


Fig. 7 Power zones and locations of ZCUs and zone detector assemblies in a CANDU-6 reactor (Front half of reactor, zones 1–7, shown. Back half of reactor consists of zones 8–14.)

The RRS routines serve the following functions:

- Maintaining core criticality between refuelings
 - o By slowly lowering the water level in all ZCUs, the core is maintained critical as fuel burns. When a channel is refueled the water level goes up in all ZCUs.
 - o If a fueling machine becomes unavailable, criticality is maintained by removing adjuster rods, one bank at a time.
- Bulk (total) power control
 - o Total power (captured by the sum of the 14 zone detector readings) is checked every 0.5 s and the water levels in all ZCUs are adjusted in the same direction if power deviates from its set value.
- Spatial power control
 - o Individual zone powers (captured by individual zone detector readings) are checked every 2 s and if a zone power deviates from its time-average target value, the water level in its ZCU is raised (to decrease zone power) or lowered (to increase zone power) as necessary.
- Dampening of Xenon oscillations¹⁰ using the 14 ZCUs
- Flux mapping
 - o Flux mapping is performed every 2 min using readings from 102 self-powered vanadium detectors.
- Startup maneuvers
- Rapid power reduction, if necessary, using the 4 MCAs

Shutdown systems

CANDU-6 reactors have two, completely separate, independent and physically different (using different physical principles) shutdown systems, labeled SDS1 and SDS2. Each of them is capable of independently shutting down the reactor. SDS1 consists of 28 “black” spring-loaded cadmium shutoff rods which drop into the core when SDS1 is triggered. Shutoff rods fall into the core through guide tubes located between fuel channels. Because the moderator is not pressurized, there is no ejection mechanism for these shutoff rods. SDS1 uses 34 in-core self-powered platinum detectors for high-power neutronic trips and has a total negative reactivity of 8000 pcm which can be inserted at a rate of 5000 pcm/s.

¹⁰¹³⁵Xe is a fission product and a strong neutron absorber. Its concentration increases in regions with high neutron flux and decreases in regions with low neutron flux. Because it is a strong neutron absorber, its accumulation leads to a flux decrease in the regions with originally high flux and its depletion leads to a flux increase in the regions with originally low flux. This phenomenon can lead to unwanted flux and power oscillations which need to be dampened.

SDS2 consists of six liquid-poison (gadolinium nitrate solution) horizontal injection nozzles which can inject up to 30,000 pcm negative reactivity at a rate of 5000 pcm/s. When SDS2 is triggered, liquid poison is pushed into the core by pressurized helium. Liquid injection will work even in a deformed-geometry core that may occur in a severe accident. SDS2 uses 24 in-core self-powered platinum detectors for high-power neutronic trips.

Emergency core cooling system

The Emergency Core Cooling System (ECCS) floods the core in case of a major break in the primary circuit. It injects (light) water into all headers so water reaches the core regardless of the location of the break. ECCS has three stages: a high-pressure stage which uses gas pressure to inject water into the core from a water tank outside the reactor building, a medium pressure stage which uses water from a dousing tank located at the top of the containment building, and a low-pressure stage which pumps water from the reactor building sump back into the reactor core via the emergency core cooling heat exchanger.

Containment system

For a single CANDU-6 reactor, the containment system consists of a plastic-lined concrete structure, a dousing system, air coolers, a filtered discharge system, and access air locks.

For very small breaks in the primary circuit, the air coolers are used to condense the steam released in the containment. For larger breaks, when pressure in the containment rises, the dousing system may be started either manually or automatically (on a pressure setpoint). The dousing system sprays light water from the dousing tank located at the top of the containment into the steam environment, thus lowering the temperature, which causes the steam to condense and, consequently, the pressure to drop. Dousing is automatically shut off when the pressure reaches a lower limit but is restarted if pressure increases again. This cycle can be repeated multiple times.

The containment system is designed to withstand the additional pressure caused by a large break in the primary circuit and its leak rate is designed to be smaller than 0.5% of the total volume per day. Any radioactivity release is expected to be reduced by a factor of 10,000 by the containment.

Variants of the CANDU system

The CANDU-6 is the most widely deployed CANDU system. CANDU-6 plants have been built and operated successfully in Canada (Point Lepreau and Gentilly-2), Argentina (Embalse), China (Qinshan phase II), Korea (Wolsong) and Romania (Cernavoda).

The CANDU-6 system is, however, only one example of a wider family of CANDU designs. In the early days of the CANDU program, each new build was seen as an opportunity to try different technical solutions and to improve the design. As a consequence, reactor and steam-supply system characteristics differ from site to site and from the CANDU-6 design. As an illustration, [Table 6](#), adapted from ([IAEA, 2002](#)), compares the characteristics of the CANDU-6 system with those of CANDU nuclear-power plants built at several locations in the Canadian province of Ontario, as well as at sites in India that also use CANDU-type technology.

Pressure-vessel heavy-water reactors

As the pressure-tube CANDU system was being developed and plants were built in Canada and internationally, a pressure-vessel system was being developed in Germany with a prototype being built in Germany and commercial plants being built in Argentina at the Atucha site.

The Atucha pressure-vessel HWRs use heavy water as both coolant and moderator. Although the two belong to different circuits, they are not physically separated, as the two circuits communicate. The fuel consists of 37-rod vertical fuel assemblies, each approximately 5-m tall. Fuel rods are arranged in a concentric-ring arrangement similar to the CANDU-6 one (1, 6, 12, 18). Fuel assemblies reside in vertical fuel-channel tubes approximately 11.5 cm in diameter. Fuel-channel tubes are arranged in a triangular pattern with a lattice pitch of 27.2 cm. The moderator is found outside the fuel-channel tubes and there is no calandria tube and no gas gap to provide thermal insulation between the fuel-channel tube and the moderator. The moderator is contained in a vertical cylinder-shaped moderator tank which is traversed axially by the fuel-channel tubes. The moderator tank resides inside the pressure vessel, with a lower plenum below it and an upper plenum above it. The top of the moderator tank communicates with the upper plenum through holes serving to nearly equalize the coolant and moderator pressures. The PHTS has two loops. The coolant enters the pressure vessel through the coolant inlets located near the top of the pressure vessel and flows downwards, outside the moderator tank, towards the lower plenum through a downcomer similar to the one in LWRs. The coolant then flows upwards through the fuel-channel tubes and through slits at the top of the fuel-channel tubes into the upper plenum and then out of the pressure vessel through the coolant outlets. The moderator enters the pressure vessel through moderator inlets and then through a ring-shaped inlet header located at the bottom of the tank and into the moderator tank. It leaves through an outlet ring-shaped header located at the top of the moderator tank and out of the pressure vessel through moderator outlet nozzles. As a special feature distinguishing the pressure-vessel HWRs from the LWRs, the upper and lower plena do not occupy the entire space between the cylindrical moderator tank and the approximately hemispherical top and bottom of the pressure vessel, but rather a small portion of it. In order to save on

Table 6 Main parameters of CANDU pressure-tube HWR systems.

	<i>CANDU-6</i>	<i>Pickering A</i>	<i>Pickering B</i>	<i>Bruce A</i>	<i>Bruce B</i>	<i>Darlington</i>	<i>Rajasthan 1/2 Kalpakkam 1/2</i>	<i>Narora 1/2 Kakrapar 1/2</i>	<i>Tarapur 1/2</i>
Thermal Power (MW)	2064	1742	1744	2551	?	2949	693.5	862	1835
Gross Electrical Power (MW)	728	540	540	791	807	936	100–200	220	~ 500
Number of loops	2	2	2	1	1	2	1	1	2
Core Orientation	horizontal	horizontal	horizontal	horizontal	horizontal	horizontal	horizontal	horizontal	horizontal
Number of channels	380	390	380	480	480	480	306	306	392
Number of bundles per channel	12	12	12	13	13	13	10	10	13
Number of fuel elements per bundle	37	28	28	37	37	37	19	19	37
Fuel type	Natural UO ₂	Natural UO ₂	Natural UO ₂	Natural UO ₂	Natural UO ₂	Natural UO ₂	Natural UO ₂	Natural UO ₂	Natural UO ₂
Lattice pitch (cm)	28.58	28.58	28.58	28.58	28.58	28.58	22.86	22.86	28.58
Shutoff Rods	28 Cd rods	11 Cd rods	28 Cd rods	30 Cd rods	32 Cd rods	32 Cd rods	14	14	28
Liquid poison injection (shutdown)	6 Gd nitrate injection nozzles	n/a	6 Gd nitrate injection nozzles	Gd nitrate injection nozzles	8 Gd nitrate injection nozzles	Gd nitrate injection nozzles	Liquid poison injection—12 nozzles	Liquid poison injection	Liquid poison injection—6 nozzles
Moderator dump (shutdown)	n/a	available	n/a	n/a	n/a	n/a	available	available	n/a
Adjuster rods (control)	21	18	21	n/a	24	16	n/a	n/a	17
Booster rods (control)	n/a	n/a	n/a	Initially available, later removed	n/a	n/a	available	available	n/a
H ₂ O Liquid Zone Control Units	14	14	14	14	14	14	14	14	14
moderator poison (control)	n/a	Moderator B	Moderator B	Moderator B	Moderator B	Moderator B	Moderator B	Moderator B	n/a
Mechanical Control Absorbers (control)	4	n/a	4	4	4	4	available	available	4
Containment	Prestressed concrete, cylindrical	Reinforced steel cylinder	Reinforced-steel cylinder w/ elliptical top	Rectangular building, reinforced concrete	Rectangular building, reinforced concrete	Rectangular building, reinforced concrete	Reinforced concrete cylindrical, pre-stressed hemispherical dome, single wall Rajasthan, partial double-wall shell, single dome, Kalpakkam	Full double-wall shell, single dome	Cylindrical full double containment

Adapted from IAEA (2002) Technical Reports Series No. 407, 2002. *Heavy Water Reactors: Status and Projected Development*. International Atomic Energy Agency. Retrieved from: https://www-pub.iaea.org/MTCD/publications/PDF/TRS407_scr/D407_scr1.pdf.

Table 8 Main characteristic of prototype heavy-water power reactors.

Reactor	Country	Core orientation	Pressure boundary	Moderator	Coolant	Fuel	Fuel assembly	Lattice type	Pitch (cm)	Refueling	Thermal power (MW)	Gross electrical power (MW)	Operating years	Notes
NPD	Canada	horizontal	Pressure tubes	Low-pressure D2O	D2O	Natural UO ₂	Annular cluster, 7 elements, 9 bundles per channel	square	26.4	On-power	81	22	1962–1987	
Douglas Point	Canada	horizontal	Pressure tube	Low-pressure D2O	High-pressure D2O	Natural UO ₂	Annular cluster, 19 elements, 12 bundles per channel	square	22.86	On-power	693	220	1967–1984	
KANUPP	Pakistan	horizontal	Pressure tube	Low-pressure D2O	High-pressure D2O	Natural UO ₂	Annular cluster, 19 elements, 11 bundles per channel	square	23.5	On-power	432	137	1972-present	
CVTR	USA	vertical	U-shaped pressure tubes	Low-pressure D2O	pressurized D2O	1.5% and 2% enriched UO ₂	Annular cluster, 19 elements, 1 assembly per section of channel	rectangular	16.51 cm × 20.32 cm	Off-power	65	19	1963–1967	Included oil-fired superheater
SGHWR	UK	vertical	Pressure tubes	Low-pressure D2O	Boiling H2O	2.28% enriched UO ₂	Annular cluster, 36 elements, 1 assembly channel	square	26.03	On-power	308.2	102.4	1968–1980	Direct cycle
Gentilly 1	Canada	vertical	Pressure tubes	Low-pressure D2O	Boiling H2O	Natural UO ₂	Annular cluster, 18 elements, 10 bundles per channel	square	27.94	On-power	832.9	260	1971–1977	Direct cycle
Fugen	Japan	vertical	Pressure tubes	Low-pressure D2O	Boiling H2O	2% enriched UO ₂ + PuO ₂ (MOX)	Annular cluster, 28 elements, 1 assembly per channel	square	24.0	On-power	557	165	1979–2003	Direct cycle
Cirene	Italy	vertical	Pressure tubes	Low-pressure D2O	Boiling H2O	natural and 1.15% enriched UO ₂	Annular cluster, 18 elements, 8 bundle per channel	square	27.0	On-power	130	40	Not commissioned	Direct cycle
MZFR	Germany	vertical	Pressure vessel	Pressurized D2O	Pressurized D2O	Natural UO ₂	Annular cluster, 37 elements, 2 assemblies per channel	Triangular/hexagonal	27.2	On-power	200	57	1966–1984	Served as prototype for the Atucha plants
Ågesta	Sweden	vertical	Pressure vessel	Pressurized D2O	Pressurized D2O	Natural UO ₂	Annular cluster, 19 elements, 4 bundles per channel	square	27	off-power	65	10	1964–1974	
Marviken	Sweden	vertical	Pressure vessel	Pressurized D2O	Boiling D2O	1.35% enriched UO ₂ and 1.75% enriched UO ₂ for superheated channels	Annular cluster, 36 elements for boiling channels, 45 elements for superheated channels	square	25.0	On-power	463	138	Construction stopped in 1969, never commissioned	Direct cycle, included superheated channels
EL 4	France	horizontal	Pressure tubes	Low-pressure D2O	Pressurized CO2	1.37% and 1.65% enriched UO ₂	Annular cluster, 19, 9 bundles per channel	square	23.5	On-power	250	73	1967–1985	Gas cooled, fuel pins have a square cross section
Niederaichbach	Germany	vertical	Pressure tubes	Low-pressure D2O	Pressurized CO2	1.15% enriched UO ₂	Annular cluster, 19 elements, 4 bundles per channel	square	24.5	On power	316	106.4	1973–1974	Gas cooled
Lucens	Switzerland	vertical	Pressure tubes	Low-pressure D2O	Pressurized CO2	0.95% enriched U metal alloyed with 0.1% Cr	Annular cluster, 7 elements encased in graphite matrix	square	24 inner, 29 outer	On-power	30	10.4	1968–1969	Gas cooled, Located in underground rock cavern
Bohunice	Slovakia	vertical	Pressure vessel	Pressurized D2O	Pressurized CO2	Natural U metal	150–200 rods/assembly	square		Off-power	590	150	1972–1979	Gas cooled

Adapted from IAEA (2002) Technical Reports Series No. 407, 2002. *Heavy Water Reactors: Status and Projected Development*. International Atomic Energy Agency. Retrieved from: https://www-pub.iaea.org/MTCD/publications/PDF/TRS407_scr/D407_scr1.pdf.

Boric acid can be inserted into the system and then slowly removed to compensate for fuel burnup after the initial load. Finally, moderator temperature can be varied in order to achieve fine reactivity control in certain operating modes. The reactor can be shut down either by banks of black shutdown rods (SDS1) or by rapid injection of boric acid into the moderator (SDS2).

A generic diagram of the core is shown in Fig. 8, adapted from (IAEA, 2002). The Atucha 1 and 2 reactor systems differ in power and other technical specifications. Table 7, adapted from (IAEA, 2002), shows the main parameters of the two systems.

Prototypes, advanced designs, and advanced-fuel-cycles

Prototypes

In addition to the large heavy-water power reactors in current commercial operation, a number of prototype heavy-water power reactors have been designed, built and operated over time. These were usually lower-power reactors (<200 MWe), oftentimes operated only for a few years, primarily as proofs of concept, to demonstrate the feasibility of specific technical solutions. Table 8, adapted from (IAEA, 2002), summarizes the characteristics of early prototype reactors. Interested readers can find additional details in (IAEA, 2002).

As Table 8 illustrates, prototype reactors have utilized a wider array of technical options (e.g., gas and boiling-light-water coolant) than the heavy-water systems in current commercial operation.

Advanced designs

In recent years, advanced CANDU designs have also been developed, most notably the Advanced CANDU Reactor (ACR). The ACR retains the horizontal pressure-tube design but has a tighter lattice pitch, of only 24 cm, and is cooled by pressurized light, rather than heavy, water. The fuel bundles have 43 fuel elements, instead of the 37 fuel elements used by the CANDU-6 design and fuel pellets consist of slightly enriched (<2.5%) UO_2 . The ACR retains both the shutoff-rod and liquid-poison-injection shutdown systems but the light-water zone control units are replaced by control blades. Finally, a supercritical-water-cooled pressure-tube heavy-water-moderated reactor is part of the Generation-IV International Forum whereby a preliminary design is still being developed.

Advanced fuel cycles

Because of their excellent neutron economy, fuel-channel design and on-line refueling, CANDU reactors can accommodate other fuel cycles besides natural uranium, such as low-enriched (usually <1.2%) uranium, recycled uranium (from LWR) and thorium (Foster and Critoph, 1975). Because of their good neutron economy, CANDU reactors can allow the direct use of used PWR fuel (including fission products), a technology known as Direct Use of PWR Fuel in CANDU (DUPIC, Park et al., 2008).

CANDU cores can also use thorium-based fuel cycles. ^{232}Th is a fertile nuclide which, under neutron irradiation, is converted to the fissile nuclide ^{233}U . To initiate fission, some fissile material (driver) needs to be initially present in the core. Driver fuel can come in the form of natural uranium, recycled uranium or DUPIC fuel (Jeong et al., 2008). Used fuel can either be disposed of, in the case of a once-through fuel cycle, or processed to extract ^{233}U to be used, alongside ^{232}Th , to fuel the reactor, in the case of a closed fuel cycle. Theoretically, over time, all fissile material fueling the reactor (alongside ^{232}Th) could be in the form of ^{233}U recovered from used fuel. In practice, the amount of ^{233}U that can be extracted from used fuel is less than the amount of fissile material needed to refuel the reactor to maintain criticality. Consequently, even in the case of a closed Th fuel cycle, some fissile material besides ^{233}U needs to be used in the core (either mixed with recycled ^{233}U or separate). The additional fissile material can be in the form of natural uranium, recycled uranium, or DUPIC fuel (Jeong et al., 2008; St-Aubin and Marleau, 2015).

See Also: Commercial Nuclear Power Reactors and Their Design—Introduction; Pros and Cons of Commercial Reactor Designs: Part 2—Current Status and Future Trends in the World Nuclear-Power Industry and Technical Considerations of Nuclear-Power Reactors.

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Commercial Nuclear Power Plant Statistics

David A. Hamon*, GE-Hitachi Nuclear Energy, San Jose, CA, United States

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Glossary

ANS American Nuclear Society

BWR boiling water reactor

FNR fast neutron reactor

GCR gas-cooled reactor

*Retired.

GWe gigawatts electric
 IAEA International Atomic Energy Agency
 LGR light water-cooled graphite reactor
 LMFBR liquid metal fast breeder reactor
 LWGR light water cooled graphite reactor (used in Fig. 4)
 MW megawatts
 PHWR pressurized heavy water reactor
 PRIS power reactor information system
 PWR pressurized water reactor
 TWh terawatt hours
 WNA World Nuclear Association

The number of operating commercial nuclear power reactors in each country is something that changes on a regular basis as new units come into service and others are permanently shut down. As a result, the information presented in this section is a snapshot for each country as of January 2021.

More up-to-date information can be obtained by accessing the IAEA Power Reactor Information System (PRIS) database or the World Nuclear Association (WNA) country profile pages using the links provided in the Relevant Webpages section. The American Nuclear Society (ANS) also publishes a list of operable, under construction, or on order nuclear reactors once per year in its Nuclear News magazine. ANS also publish a booklet with this information and maps that show the locations of the reactors in each country.

Global statistics

Fig. 1 presents total world-wide nuclear electricity production from 1970 to 2019, with color coding to show how much was produced in each of six regions. Total production for 2019 was 2657 TWh.

Fig. 2 shows the nuclear generation capacity in GWe for reactors that were classified as operable from 1970 through 2019. The top of the blue bar for each year is the actual amount of nuclear generation that occurred from operable reactors.

Fig. 3 presents the global average capacity factor for reactors that generated electricity during the period from 1970 to 2019. The global average capacity factor was 82.5% in 2019.

Fig. 4 presents the capacity factor by reactor type for the years 2014 through 2018, as well as for 2019. Five of the six reactor types had better performance in 2019 than their average performance for 2014–18.

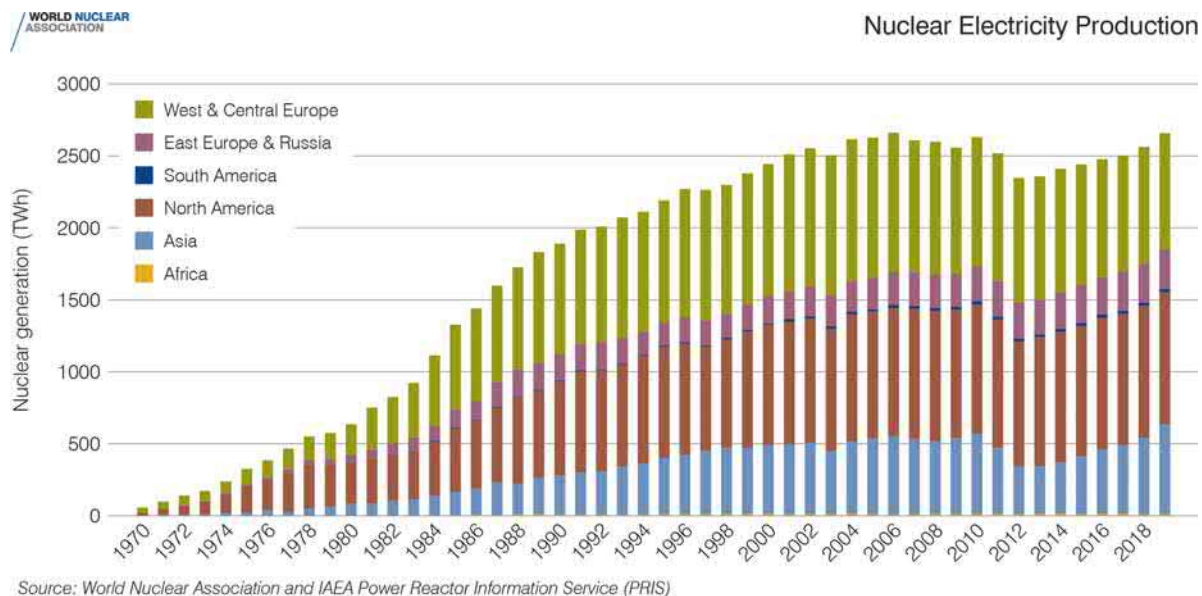


Fig. 1 Nuclear electricity production by region. World Nuclear Association, World Nuclear Performance Report 2020, Report 2020/008, August 2020, Figure 1.

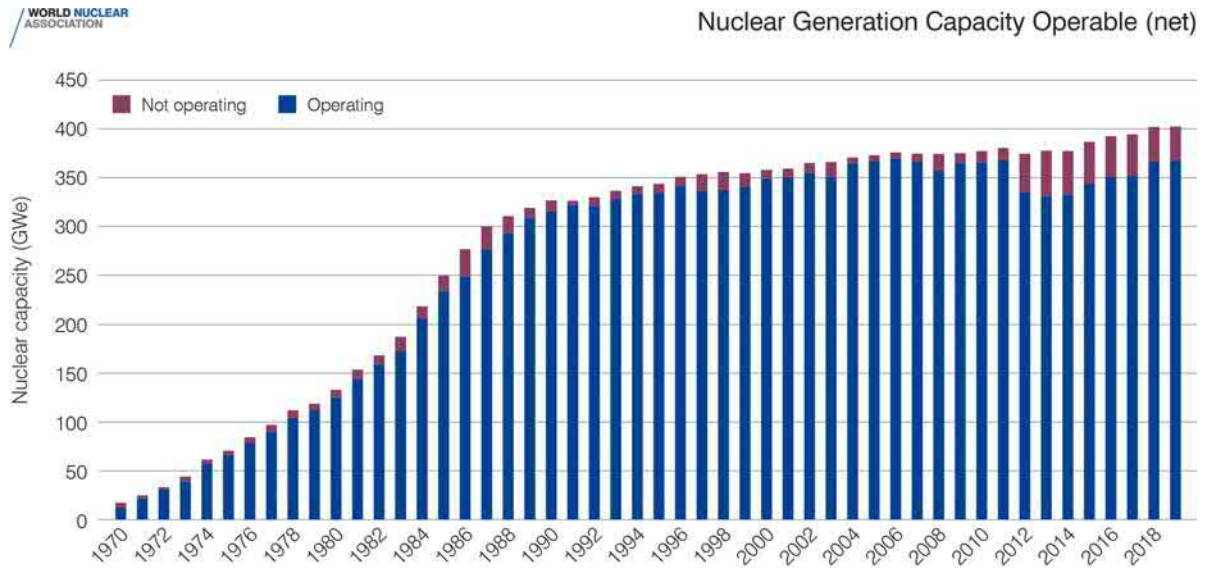


Fig. 2 Nuclear generation capacity for operable reactors. World Nuclear Association, World Nuclear Performance Report 2020, Report 2020/008, August 2020, Figure 2.

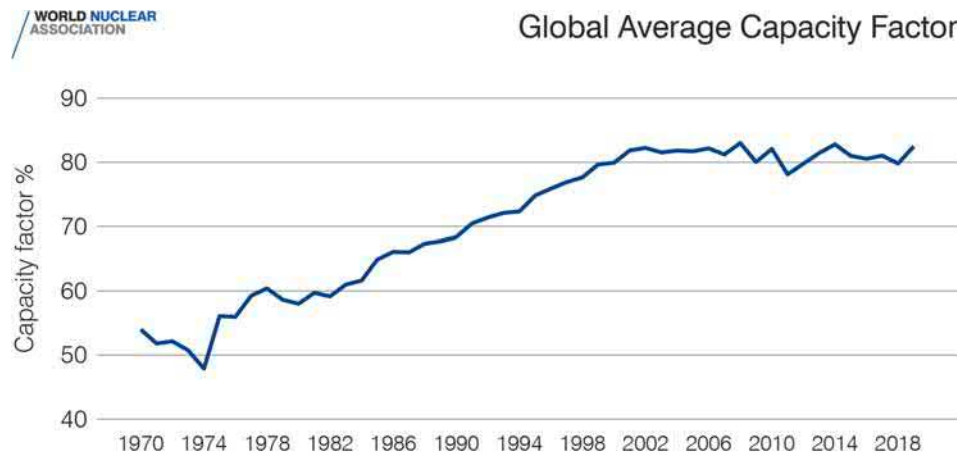


Fig. 3 Global average capacity factor. World Nuclear Association, World Nuclear Performance Report 2020, Report 2020/008, August 2020, Figure 4.

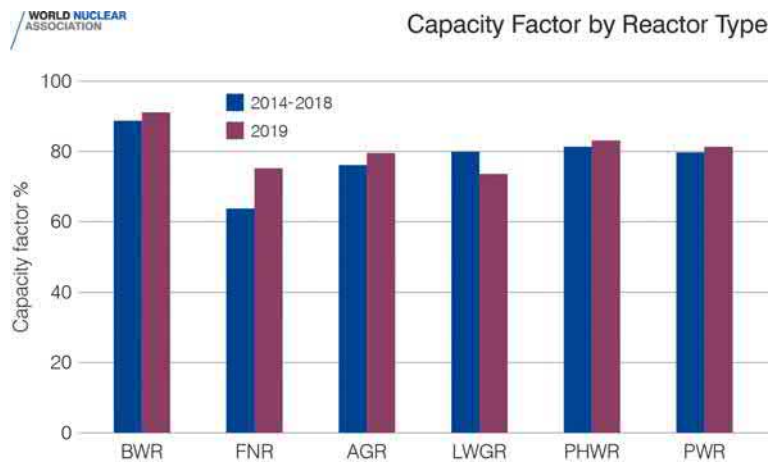


Fig. 4 Capacity factor by reactor type. World Nuclear Association, World Nuclear Performance Report 2020, Report 2020/008, August 2020, Figure 5.

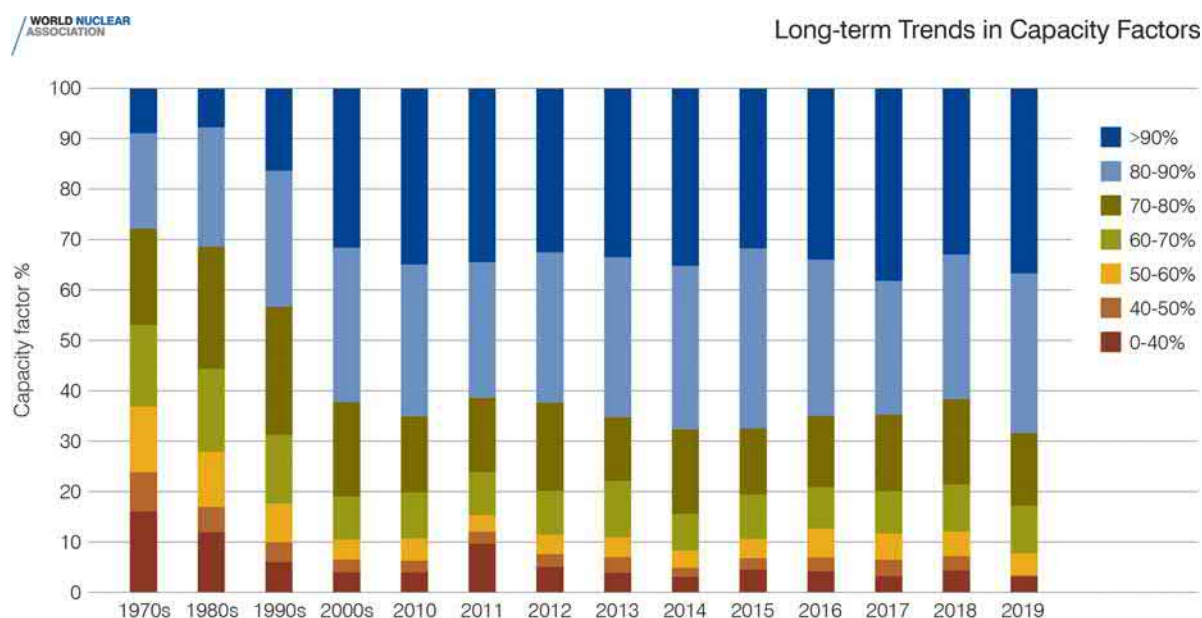


Fig. 5 Long-term trends in capacity factor. World Nuclear Association, World Nuclear Performance Report 2020, Report 2020/008, August 2020, Figure 9.

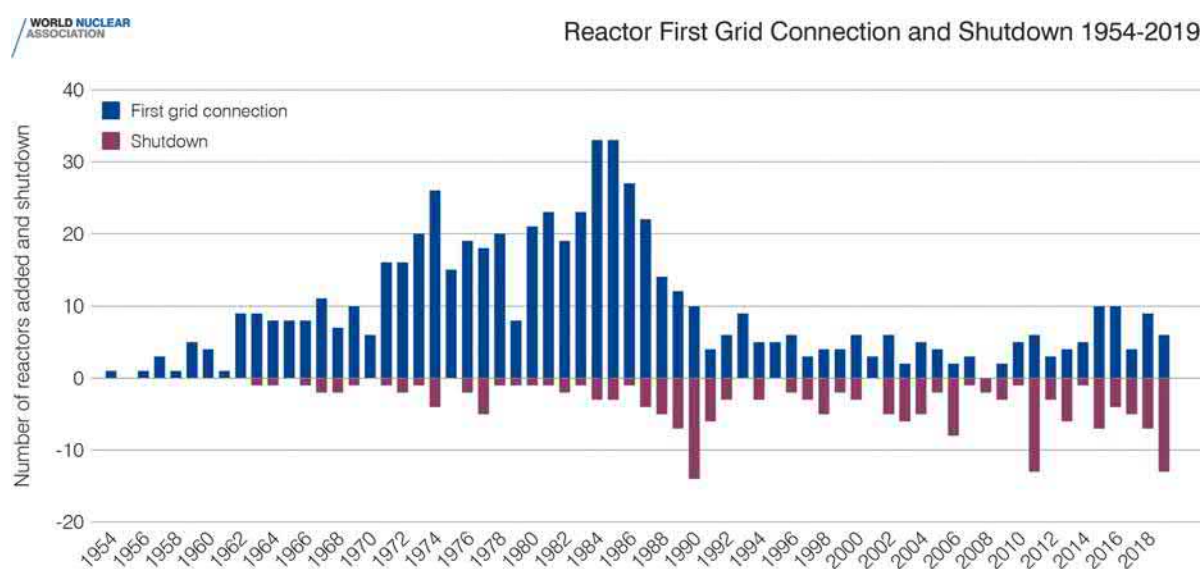


Fig. 6 Reactors first grid connection and shutdown, 1954–2019. World Nuclear Association, World Nuclear Performance Report 2020, Report 2020/008, August 2020, Figure 13.

Fig. 5 shows the percentage of operating reactors that obtained specific ranges of capacity factors. Capacity factors improved substantially from the 1970s through to the 2000s. Since then, this high performance has been maintained.

Fig. 6 shows the number of reactors that first connected to the grid and the number that were permanently shut down from 1954 through 2019. The majority of the reactors that permanently shut down in 2019 actually ceased generation between 2011 and 2017.

The above figures and additional information about them can be found in the World Nuclear Performance Report 2020, which is published by the World Nuclear Association. New versions of this report are issued on a yearly basis.

Country statistics

Table 1 summarizes the number of reactors in operation and under construction and their associated net electrical capacity in each country. Approximate capacity factor values for 2018 are also presented in Table 1 for each country with operating reactors.

The subsections which follow for each country provide details on the type of reactors that are found in each country.

Table 1 Reactors in operation, under construction and shut down by country.

Country	In operation		Under construction		Shut down	
	Number	Electrical net output (MW)	Number	Electrical net output (MW)	Number	2018 Production (GWh) Approx. capacity factor (%)
Argentina	3	1641	1	25		6453 44.9
Armenia	1	375			1	1898 57.8
Bangladesh			2	2160		
Belarus	1	1110	1	1110		
Belgium	7	5930			1	26,996 52.0
Brazil	2	1884	1	1340		15,674 95.0
Bulgaria	2	2006			4	16,125 91.8
Canada	19	13,554			6	95,037 80.0
China	50	47,518	11	10,806		286,501 68.8
Czech Republic	6	3932				28,255 82.0
Finland	4	2794	1	1600		21,889 89.4
France	56	61,370	1	1630	14	393,200 73.1
Germany	6	8113			30	71,866 101.1
Hungary	4	1902				14,857 89.2
India	23	6885	6	4194		35,389 58.7
Iran	1	915	1	974		6300 78.6
Italy					4	
Japan	33	31,679	2	2653	27	49,199 64.5
Kazakhstan					1	
Lithuania					2	
Mexico	2	1552				13,200 97.1
Netherlands	1	482			1	3341 79.1
Pakistan	5	1318	2	2028		9290 80.5
Romania	2	1300				10,422 91.5
Russia	38	28,578	3	3459	9	191,331 76.4
Slovakia	4	1814	2	880	3	13,789 86.8
Slovenia	1	688				5490 91.1
South Africa	2	1860				10,564 64.8
South Korea	24	23,172	4	5360	2	127,075 62.6
Spain	7	7121			3	53,295 85.4
Sweden	7	7740			6	63,849 94.2
Switzerland	4	2960			2	24,496 94.5
Taiwan	4	3844	2	2600	2	26,656 79.2
Turkey			2	2228		
Ukraine	15	13,107	2	2070	4	84,398 73.5
United Arab Emirates	1	1345	3	4035		
United Kingdom	15	8923	2	3260	30	59,098 75.6
United States	94	96,553	2	2234	39	807,078 95.4
Total	444	393,965	51	54,646	191	2,573,013

Approximate capacity factor has been calculated based on total nuclear production for 2018 divided by the theoretical maximum generation if all operating plants in the country were to run at full rated electrical net output for the entire year. For Japan, it was calculated based on the capacity of the nine operating units instead of the entire fleet of 37 units that remain operational.

Source: IAEA PRIS database summary, January 2021.

Argentina

Argentina's first nuclear reactor was designed by Siemens and began operation in 1974. In 2018 with one of its reactors down for a lifetime extension program, nuclear power accounted for 5% of Argentina's electricity production.

Argentina has three operating pressurized heavy water reactor (PHWR) units with a combined net electrical capacity of 1641 MW. One pressurized water reactor (PWR) unit with a planned net electrical capacity of 25 MW is under construction.

Armenia

Armenia's nuclear reactors were an early Soviet VVER design, the first of which began operating in 1976. In 2018, nuclear power accounted for 27% of Armenia's electricity production.

Armenia has one operating PWR unit with a net electrical capacity of 375 MW. Armenia also has one reactor that has been permanently shut down.

Bangladesh

Bangladesh started construction of its first nuclear power reactor in November 2017. The unit is scheduled to be commissioned in 2023.

Bangladesh has two PWR units under construction with a planned combined net electrical capacity of 2160 MW. Atomstroyexport is building the two VVER-1200 reactors.

Belarus

Belarus connected its first nuclear power reactor to the grid in November 2020.

Belarus has one operating PWR unit with a net electrical capacity of 1110 MW, and one PWR unit under construction with a planned combined net electrical capacity of 1110 MW. Atomstroyexport is building the two VVER-1200 reactors.

Belgium

Belgium's first commercial nuclear reactor began operating in 1974. In 2018, nuclear power accounted for 38% of Belgian electricity production.

Belgium has seven operating PWR units with a combined net electrical capacity of 5942 MW. Belgium also has one reactor that has been permanently shut down. Westinghouse and Framatome have both designed PWRs in Belgium.

Brazil

Brazil's first commercial nuclear reactor was designed by Westinghouse and began operating in 1982. In 2018, nuclear power accounted for 3% of Brazilian electricity production.

Brazil has two operating PWR units with a combined net electrical capacity of 1884 MW. One PWR unit is under construction with a planned net electrical capacity of 1340 MW. Kraftwerk Union (KWU) designed the second and third Brazilian PWR units.

Bulgaria

Bulgaria's commercial nuclear reactors were designed by the Soviets and the first unit began operating in 1974. In 2018, nuclear power accounted for 34% of Bulgarian electricity production.

Bulgaria has two operating PWR units with a combined net electrical capacity of 2006 MW. Bulgaria also has four reactors that were permanently shut down as a result of Bulgaria's agreement to join the European Union in January 2007.

Canada

In 2018, nuclear power accounted for 15% of Canadian electricity production.

Canada has 19 operating PHWR units designed by Atomic Energy of Canada, Ltd. (AECL) with a combined net electrical capacity of 13,554 MW. Canada also has six reactors that have been permanently shut down.

China

The first nuclear power plant in China was a PWR provided by Mitsubishi Heavy Industries of Japan. Westinghouse, Areva/Framatome, Atomstroyexport (Russia) and Atomic Energy of Canada, Ltd. have also supplied PWRs and PHWRs to China, but most Chinese plants are now of Chinese design. In 2018, nuclear power accounted for 4% of Chinese electricity production.

China has 47 operating PWR units, two operating PHWR units and one liquid metal fast breeder reactor (LMFBR) unit with a combined net electrical capacity of 47,518 MW. Eleven PWR units with a combined net electrical capacity of 8200 MW are under construction. One gas cooled reactor (GCR) unit, one liquid metal fast breeder reactor (LMFBR) units and about 20 PWR units are planned but not yet under construction.

Czech Republic

The Czech Republic's commercial nuclear power reactors were designed by the Soviets and the first unit began operating in 1985. In 2018, nuclear power accounted for 34% of electricity production in the Czech Republic.

The Czech Republic has six operating PWR units with a combined net electrical capacity of 3932 MW.

Finland

In 2018, nuclear power accounted for 32% of Finnish electricity production.

Finland has two operating boiling water reactor (BWR) units and two operating PWR units with a combined net electrical capacity of 2784 MW. One PWR unit with a planned net electrical capacity of 1600 MW is under construction. One additional PWR with a planned net electrical capacity of 1200 MW is planned but has not started construction. The operating BWRs were designed by ASEA Atom and the operating PWRs were designed by the Soviets. The PWR that is under construction was designed by Areva and is currently expected to begin operating in 2021.

France

In 2017, nuclear power accounted for 71% of French electricity production.

France has 56 operating PWR units with a combined net electrical capacity of 61,360 MW. One PWR unit with a planned net electrical capacity of 1600 MW is under construction. France has 14 reactors that have been permanently shut down. All French reactors were designed by Framatome or its successor, Areva.

Germany

In 2018, nuclear power accounted for 12% of German electricity production. General Electric designed the first BWR in Germany, which is now shut down. All operating German reactors were designed by Siemens-KWU. Several Soviet-designed reactors existed in East Germany at the time of German reunification, but all of them were shut down due to safety concerns.

Germany has one operating BWR unit and five operating PWR units with a combined net electrical capacity of 8113 MW. Germany has 30 reactors that have been permanently shut down.

Hungary

Hungary's commercial nuclear power reactors were designed by the Soviets and the first unit began operating in 1982. In 2018, nuclear power accounted for 49% of Hungarian electricity production.

Hungary has four operating PWR units with a combined net electrical capacity of 1902 MW.

India

General Electric designed India's two BWRs, which began operation in 1969. AECL designed the first two PHWRs in India. Subsequent PHWRs were designed locally in India. More recently, Atomstroyexport started providing PWRs to India. In 2018, nuclear power accounted for 2.4% of Indian electricity production.

India has two operating BWR units, 19 operating PHWR units and two operating PWR units with a combined net electrical capacity of 6885 MW. One LMFBR unit, three PHWR units and two PWR units with planned combined net electrical capacities of 4194 MW are under construction. Additional PHWR units and PWR units are planned but not yet under construction.

Iran

In 2018, nuclear power accounted for 2% of Iranian electricity production.

Iran has one operating PWR unit with a net electrical capacity of 915 MW. One additional PWR unit with a net electrical capacity of 974 MW is under construction. The operating PWR started construction in 1975, but did not connect to the grid until 2011. The original design was from Siemens-KWU, but construction was abandoned after the Iranian revolution. The plant was eventually completed with help from the Russians.

Italy

Italy has had four operating nuclear power reactors but the last two were permanently shut down following the Chernobyl accident.

Japan

In 2017, nuclear power accounted for 3% of Japanese electricity production.

The first Japanese BWRs of BWR/2 through BWR/5 product lines were designed by General Electric. Subsequent BWRs of these product lines were built by Hitachi or Toshiba under license from General Electric. The Kashiwazaki Kariwa ABWR units were designed jointly by General Electric, Hitachi and Toshiba.

The first Japanese PWR was designed by Westinghouse. Subsequent Japanese PWRs were designed by Mitsubishi Heavy Industries under license from Westinghouse.

Prior to the Great East Japan Earthquake of March 11, 2011, and the subsequent disaster at the Fukushima Daiichi Nuclear Power Plant, there were 54 operating nuclear reactors in Japan.

As of the writing of this chapter, Japan has 17 BWR units with a combined net electrical capacity of 17,559 MW that remain operational, but none of them are currently in operation. Japan also has 16 PWR units with a combined net electrical capacity

of 14,120 MW that remain operational. Nine of these PWRs with a combined net electrical capacity of 8706 MW are currently in operation. Two BWR units with a combined net electrical capacity of 2653 MW are under construction. Japan has 27 reactors that are permanently shut down.

Kazakhstan

Kazakhstan operated a single Russian nuclear power LMFBR from 1972 to 1999, generating electricity and desalinating water. It is now permanently shut down.

Lithuania

Lithuania has had two operating Russian light water cooled graphite moderated reactors (LGRs). However, due to strong European Union (EU) concerns about the LGR design, by the time Lithuania applied to join the EU it was required to close them both down.

Mexico

Mexico's commercial nuclear reactors were designed by General Electric, and the first one began operating in 1989. In 2018, nuclear power accounted for 4% of Mexican electricity production. Mexico has two operating BWR units with a combined net electrical capacity of 1552 MW.

Netherlands

In 2018, nuclear power accounted for 3% of electricity production in The Netherlands.

The Netherlands has one operating PWR unit designed by Siemens-KWU with a net electrical capacity of 482 MW. The Netherlands also had a smaller General Electric natural circulation BWR unit that operated from 1968 until it was permanently shut down in 1997.

Pakistan

In 2018, nuclear power accounted for 7% of electricity production in Pakistan.

Pakistan has four operating PWR units and one operating PHWR unit with a combined net electrical capacity of 1318 MW. Two PWR units with a combined net electrical capacity of 2028 MW are under construction. One additional PWR unit with a net electrical capacity of 1000 MW is planned but not yet under construction. Pakistan's PWR units were designed by the Chinese. AECL designed the PHWR unit.

Romania

In 2018, nuclear power accounted for 18% of Romanian electricity production.

Romania has two operating PHWR units with a combined net electrical capacity of 1300 MW. Two PHWR units with a combined net electrical capacity of 1440 MW are planned but are not yet under construction. AECL designed the Romanian PHWRs units.

Russia

In 2018, nuclear power accounted for 18% of Russian electricity production.

Russian companies have designed all of Russia's reactors and are a major exporter of nuclear reactor technology. Russia has 12 operating light water-cooled graphite reactor (LGR) units, two operating LMFBR units and 24 operating PWR units with a combined net electrical capacity of 28,578 MW. Three PWR units with a combined net electrical capacity of 3459 MW are under construction. Russia has nine reactors that are permanently shut down.

Slovakia

In 2018, nuclear power accounted for 53% of Slovakian electricity production.

Slovakia has four operating PWR units with a combined net electrical capacity of 1814 MW. Two additional PWR units with a combined net electrical capacity of 880 MW are under construction. Slovakia also has three reactors that are permanently shut down. Two of them were shut down as a precondition for Slovakia's entry into the European Union in 2004.

Slovenia

Slovenia has shared ownership of its one reactor with Croatia since 1981. In 2018, nuclear power accounted for 39% of electricity production in Slovenia.

Slovenia has one operating Westinghouse PWR unit with a net electrical capacity of 688 MW.

South Africa

South Africa's commercial nuclear power reactors were designed by Framatome (now Areva) and the first one began operating in 1984. In 2018, nuclear power accounted for 5% of South African electricity production.

South Africa has two operating PWR units with a net electrical capacity of 1860 MW.

South Korea

In 2018, nuclear power accounted for 23% of South Korean electricity production.

The early PWRs in South Korea were designed by Westinghouse, Combustion Engineering and Framatome. Subsequent PWRs based on the Combustion Engineering design have been built by South Korean suppliers. South Korea has three operating PHWR units and 21 operating PWR units with a combined net electrical capacity of 23,172 MW. Four additional PWR units with a combined net electrical capacity of 5360 MW are under construction. South Korea has two reactors that are permanently shut down.

Spain

In 2018, nuclear power accounted for 20% of Spanish electricity production.

BWRs in Spain were designed by General Electric. Kraftwerk Union AG (KWU) designed one PWR with the remaining PWRs being designed by Westinghouse. Spain has one operating BWR unit and six operating PWR units with a combined net electrical capacity of 7121 MW. Spain has three reactors that are permanently shut down.

Sweden

In 2018, nuclear power accounted for 42% of Swedish electricity production. BWRs in Sweden were designed by ASEA Atom, which is today part of Westinghouse. Swedish PWRs were designed by Westinghouse. Sweden has four operating BWR units and two operating PWR units with a combined net electrical capacity of 6859 MW. Sweden has seven reactors that are permanently shut down.

Switzerland

In 2017, nuclear power accounted for 32% of Swiss electricity production. General Electric Technical Services Company (the international division of General Electric) designed two Swiss BWR plants. Westinghouse and KWU designed Swiss PWR units. Switzerland has one operating BWR unit and three operating PWR units with a combined net electrical capacity of 2960 MW. Switzerland has two reactors that are permanently shut down.

Taiwan

In 2017, nuclear power accounted for 8% of electricity production in Taiwan. General Electric designed all BWRs in Taiwan, while Westinghouse designed all PWRs. Taiwan has two operating BWR units and two operating PWR units with a combined net electrical capacity of 3844 MW. Two additional BWRs are under construction at the Lungmen site with a combined net electrical capacity of 2600 MW, but construction has been suspended. Taiwan has two reactors that are permanently closed.

Turkey

Turkey is constructing its first nuclear power plants, which are designed by Atomstroyexport of Russia. Turkey has two PWR units under construction with a planned net electrical capacity of 2228 MW. Two additional PWR units with a combined net electrical capacity of 2228 MW are planned but have not started construction.

Ukraine

In 2017, nuclear power accounted for 55% of electricity production in Ukraine. Ukraine's nuclear plants were all designed by a Russian supplier. Ukraine has 15 operating PWR units with a combined net electrical capacity of 13,107 MW. Two additional PWR units with a combined net electrical capacity of 2070 MW are under construction. Ukraine has four reactors that are permanently closed.

United Arab Emirates

The United Arab Emirates is constructing its first nuclear power plants, which are designed by South Korea. The first of four units was connected to the grid in August 2020. The United Arab Emirates has one operating PWR with a net electrical capacity of 1345 MW. Three additional PWR units with a combined net electrical capability of 4035 MW are under construction.

Table 2 Annual capacity factors for nuclear generation in the United States.

Period	Annual capacity factors for nuclear generation
2013	89.9%
2014	91.7%
2015	92.3%
2016	92.3%
2017	92.2%
2018	92.6%

U.S. Energy Information Administration (2019) *Electric Power Monthly, Data for July 2019, Table 6.7.B. Capacity Factors or Utility Scale Generators Not Primarily Using Fossil Fuels, January–July 2019*. Available from https://www.eia.gov/electricity/monthly/epm_table_grapher.php?t=epmt_6_07_b

United Kingdom

In 2018, nuclear power accounted for 20% of electricity production in the United Kingdom. Operating plants were designed by companies in the United Kingdom. Areva designed the PWRs that are under construction.

The United Kingdom has 14 gas-cooled reactor (GCR) units and one PWR unit with a combined net electrical capacity of 8923 MW. Two additional PWRs with a combined net electrical capacity of 3260 MW are under construction. The United Kingdom has 30 reactors that are permanently shut down.

United States

In 2018, nuclear power accounted for 19% of electricity production in the United States. The United States has 31 operating BWR units and 63 operating PWR units with a combined net electrical capacity of 96,553 MW. Two PWR units with a combined net electrical capacity of 2234 MW are under construction. The United States has 39 reactors that are permanently shut down.

The annual capacity factors for nuclear power generation in the United States for 2013 through 2018 are provided in **Table 2**.

See Also: Boiling Water Reactors; Heavy-Water Reactors; Pressurized Water Reactors; Sodium-Cooled Fast Reactors; The High Temperature Gas-Cooled Reactor.

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Pros and Cons of Commercial Reactor Designs Part 1: Current Status of Electricity Generation in the World and Selected Countries

I. Pioro^a, R.B. Duffey^{b,*}, P.L. Kirillov^c, and R. Pioro^d, ^a University of Ontario Institute of Technology, Oshawa, ON, Canada; ^b Idaho Falls, ID, United States; ^c State Scientific Centre of the Russian Federation, AI Leipunsky Institute of Physics and Power Engineering (IPPE), Obninsk, Russia; and ^d Faculty of Physics, Lomonosov Moscow State University, Moscow, Russia

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Nomenclature

P, p Pressure, MPa
Q Heat transfer rate, W
s Specific entropy, J/kg K
T Temperature, °C
W Work, J

Greek symbols

η Efficiency, %

Subscripts

c condenser
cr critical
el electrical
in inlet
max maximum
out outlet
sat saturated or saturation

Acronyms/Abbreviations

ABWR Advanced Boiling Water Reactor
AECL Atomic Energy of Canada Limited
AGR Advanced Gas-cooled Reactor
BN Fast Sodium (reactor) (Быстрый Натриевый (in Russian abbreviations)) (Russia)
BREST-OD Fast Reactor with Inherent Safety Lead Coolant—Experimental Demonstration (БРЕСТ-ОД—Быстрый Реактор Естественной безоПасности со Свинцовым Теплоносителем—ОПытно-Демонстрационный от Быстрый Реактор ЕСТественной безоПасности—ОПытно-Демонстрационный (in Russian abbreviations)) (Russia))

*Retired.

BWR Boiling Water Reactor
CANDU[®] CANada Deuterium Uranium (reactor) (Registered Trademark of AECL, used under license by Candu Energy Inc., Member of SNC-Lavalin Nuclear Group)
EC Enhanced CANDU[®] (reactor)
EEC Electrical-Energy Consumption
FW Feed Water
GFR Gas-cooled Fast Reactor
GIF Generation-IV International Forum
HDI Human Development Index
HP High Pressure
HPH High Pressure Water Heater
HPT High Pressure Turbine
IPT Intermediate Pressure Turbine
IHX Intermediate Heat Exchanger
KLT Container-Carrier Cargo-Lighter Transport (reactor) (Контейнеровоз Личтеровоз ТрансПортный (реактор) (in Russian abbreviations)) (Russia)
LFR Lead-cooled Fast Reactor
LGR Light-Water-cooled Graphite-Moderated Reactor
LMFBR Liquid-Metal Fast-Breeder Reactor
LNG Liquefied Natural Gas
LP Low Pressure
LPH Low Pressure Water Heater
LPT Low Pressure Turbine
LWR Light Water Reactor
M Million
MS Moisture Separator
MSR Moisture Separator and Reheater
MHI Mitsubishi Heavy Industries
NPP Nuclear Power Plant
NRC Nuclear Regulatory Commission (USA)
PHWR Pressurized Heavy-Water Reactor
PWR Pressurized Water Reactor
RBMK Reactor of Large Capacity Channel Type (Реактор Большой Мощности Канальный (in Russian abbreviations) (Russia))
RITM-200M Reactor Integral Type Modular 200 MW_e Modernized (Реактор ИнтеГральнОго ТиПа Модульный мощностью 200 МВт Модернизационный (in Russian abbreviations) (Russia))
S, South
SC SuperCritical
SCF SuperCritical Fluid
S-CO₂ Supercritical Carbon-Dioxide
SCP SuperCritical Pressure
SCWR SuperCritical Water-cooled Reactor
SFR Sodium Fast Reactor
SG Steam Generator
SMR Small Modular Reactor, also, Small and Medium-size Reactor
UK United Kingdom
USA United States of America
VHTR Very High Temperature Reactor
VVER Water Water Power Reactor (Водо-Водяной ЛнерГетический Реактор (in Russian abbreviations) (Russia))
X Steam Extraction

Introduction: Electricity generation in the world and selected countries

It is well-known that electricity generation and its consumption are the key factors for advances in industry, agriculture, technology, and the level of living. **Table 1** shows clearly a strong dependence of Human Development Index (HDI) (Rows No. 4 and 5) from Electrical-Energy Consumption (EEC) per capita (Row 3); current population is shown in Row 1 (for details, see [Pioro et al. \(2019, 2020\)](#); [Pioro \(2016\)](#); [Pioro and Duffey \(2015\)](#)). Also, robust power industry with diverse energy sources is very important for a country's economic and political independence.

In general, electricity (see **Table 1**) can be generated from: (1) traditional fuels (i.e., non-renewable sources) such as coal, natural gas, oil, and nuclear utilizing mined resources; and (2) so-called, "renewables" such as hydro, biomass, wind, geothermal, solar, and marine (tidal and wave) power utilizing geotechnical resources.

Today, the main sources for global electrical-energy generation (see **Table 1**) are: (1) Thermal power—primarily using coal (38.3%) and secondarily using natural gas (22.9%); (2) "Large" hydro-electric power plants (16.3%); and (3) Nuclear power (10.2%). The rest 12.3% of the electrical energy is generated using renewables (wind, biomass, geothermal, solar, and marine energy) (6.6%), oil (3.3%), and other sources (2.4%).

Table 1 also lists 13 countries, which have the largest number of nuclear-power reactors/their installed capacities. These countries are listed according to the decreasing population.

Analysis of the data in **Table 1** shows that the world (~38%) and, especially, countries with the largest population, i.e., China (~66%) and India (~75%), as well as Germany (~37%) and South Korea (~42%) rely heavily on coal for the electricity generation. The United States (~39%), Russia (~59%), Japan (~37%), and United Kingdom (~44%) use mainly natural gas or

Table 1 Electricity generation in the world and selected countries by source (data on the world and 13 countries with largest installed capacities of nuclear-power reactors are located by decreasing population).

No	Country	World	China	India	USA	Russia	Japan	Germany	UK	France	S. Korea	Spain	Ukraine	Canada	Sweden
		0	1	2	3	4	5	6	7	8	9	10	11	12	13
Population, EEC, and HDI per country															
1	Population, M	7838	1442	1387	332	146	126	84	68	65	51	47	44	38	10
2	EEC ^a TW h/year	22,347	5935	1177	4034	890	946	539	307	455	512	242	124	509	133
3	W/Capita	327	461	97	1391	919	857	732	515	799	1146	588	354	1529	1518
4	HDI ^b Total	0.731	0.758	0.647	0.920	0.824	0.915	0.939	0.920	0.891	0.906	0.893	0.750	0.922	0.937
5	Rank	93	85	129	15	49	19	4	15	26	22	25	88	13	8
Electricity-generating sources															
Non-renewable (% of total)															
1	Coal	38.3	65.7	74.5	23.5	—	28.3	37	3.6	2	41.9	17	34	10.8	—
2	Gas	22.9	3.1	3.6	38.8	59	37.4	13.1	43.5	12	26.8	23	6.2	6.8	10
3	Nuclear	10.2	4.8	2.7	19.7	19	4.7	11.6	25	72	23.4	21	53.5	16.8	41
4	Oil	3.3	—	—	0.5	—	3.7	0.9	—	—	1	6	—	—	—
Renewable (% of total)															
1	Hydro	16.3	17.8	10.4	6.6	17	7.8	3	—	11	1.3	8	4.2	60.6	39
2	Wind	—	5.5	4.5	7.3	—	0.7	16.1	—	1	—	18	0.7	—	10
3	Solar	—	3.1	2.9	1.8	—	6.5	6.1	—	2	—	5	—	—	0.24
4	Geothermal	—	—	—	0.4	—	—	—	—	—	—	—	—	—	—
5	Biomass	—	—	—	1.4	—	2.4	—	—	—	—	—	—	—	—
6	Renewables	6.6	—	—	—	—	—	—	25	—	—	—	—	5	—
Other (% of total)															
1	Other	2.4	—	—	—	5	—	4.3	2.9	—	5.6	2	1.4	—	—
2	Other thermal	—	—	—	—	—	8.5	—	—	—	—	—	—	—	—
3	Biomass/other	—	—	1.4	—	—	—	7.9	—	—	—	—	—	—	—

Population is taken from <https://www.worldometers.info/world-population/population-by-country/> (data for January 2021); EEC from https://en.wikipedia.org/wiki/List_of_countries_by_Human_Development_Index (data are from 2017; for exact details see the reference); and HDI from https://en.wikipedia.org/wiki/List_of_countries_by_Human_Development_Index and http://hdr.undp.org/sites/default/files/2018_human_development_statistical_update.pdf (data from 2018—Report of 2019); and data on electricity-generating sources from 2018 (<https://www.statista.com/>). (More diagrams for other countries and/or for previous years are shown in [Pioro et al. \(2019, 2020\)](#); [Pioro \(2016\)](#); [Pioro and Duffey \(2015\)](#)).

^a $EEC/\frac{W}{Capita} = \left(\frac{EEC, \frac{GW \cdot h}{year}}{(Population, Millions) \times 10^6} \right) \times \frac{10^9}{365 \text{ days} \times 24 \text{ h}}$; Electrical Energy Consumption (EEC) compares the total electricity generated annually plus imports and minus exports, expressed in GigaWatt-hours (GW · h).

^bHDI: Human Development Index by United Nations (UN); HDI is a comparative measure of life expectancy, literacy, education, and standards of living for countries worldwide. HDI is calculated with the following formula: $HDI = \sqrt[3]{LEI \times EI \times II}$, where LEI, life expectancy index; EI, education index; II, income index. It is used to distinguish whether the country is a developed, a developing or an underdeveloped, and also to measure the impact of economic policies on quality of life.



Fig. 1 Aerial view of the largest operating NPP in the world—6288-MW_{el} Bruce NPP (courtesy of Bruce NPP: www.brucepowers.com). (The Douglas Point NPP (see left upper corner) was Canada's first full-scale NPP and the second CANDU reactor. Its success was a major milestone for Canada to enter the global nuclear-power scene. Construction began on Feb. 1, 1960 and decommission date: May 4, 1984).

Liquefied Natural Gas (LNG) for electricity generation, which is better than to use coal, but still, for now, we cannot avoid emissions of carbon dioxide and NO_x due to a combustion process. On opposite, France (~72%) and Ukraine (~54%) rely heavily on nuclear power, which is, in general, the lowest emitter of carbon dioxide compared to all other electricity-generating sources including renewables (for details, see [Pioro et al., 2019](#)). Canada and Sweden are the only countries from the 13 ones, which rely heavily on the hydro electricity generation ~61% and ~39%, respectively.

Analysis of the data on installed capacities of various power plants in the world (https://en.wikipedia.org/wiki/List_of_largest_power_stations) shows that the 22 largest power plants have installed capacities ranging within 5700–22,500 MW_{el}. Within these power plants the largest group is hydro plants (in total 12) with installed capacities ranging from 5850 and up to 22,500 MW_{el}, six Nuclear Power Plants (NPPs) (installed capacities 5700–7965 MW_{el}) (for now, the largest in the world NPP—Kashiwazaki-Kariwa NPP is not in service after the Fukushima-Daiichi NPP severe accident in March of 2011), 2 coal- and 1 gas-fired thermal power plants (installed capacities 6040–8695 MW_{el}), and 1 wind power plant (installed capacity 7965 MW_{el}). **Fig. 1** shows aerial view of the largest operating NPP in the world—6288-MW_{el} Bruce NPP (Province of Ontario, Canada).

Basic general parameters of power plants

Thermal efficiency

In general, efficiency (thermal efficiency) is one of the most important parameters of any power plant ([Pioro et al., 2019, 2020; Pioro, 2016](#)):

Gross (overall) efficiency (for details, see [Table 2](#)) during a given period of time is the ratio of the gross electrical energy generated by a unit to the energy consumed during the same period by the same unit. The difference between gross and net efficiencies is the internal usage of a power plant, which might be not so small (5% or even more). Also, the thermal efficiency is the driving force for advances in thermal power plants and, nowadays, in NPPs.

Analysis of the data listed in [Table 2](#) shows that the top power plants by the thermal efficiency are: (1) Combined-cycle power plants (combination of Brayton gas-turbine cycle and subcritical-pressure Rankine steam-turbine cycle) (for details, see Appendix A1 in [Pioro, 2016](#)) with the highest thermal efficiency in the power industry of up to 62%, which means that roughly 2/3 of thermal energy introduced inside the combined power cycle is used for electricity generation, i.e., useful energy (in general, “useful” energy can include electricity; process heat; hot water/steam for heating of buildings, hot-water supply, etc.; and desalination) and only 1/3 of the total energy is lost into environment; and (2) Supercritical-pressure Rankine steam-turbine cycle of coal-fired power plants with up to 55% thermal efficiency (for details, see Appendix A1 in [Pioro \(2016\)](#) and [Pioro and Duffey \(2007\)](#)). Unfortunately, NPPs

Table 2 Typical ranges of thermal efficiencies (gross) of modern thermal and nuclear power plants (Pioro et al., 2019, 2020; Pioro, 2016).

No	Power plant	Gross thermal efficiency
1	Combined-cycle power plant combination of Brayton gas-turbine cycle (fuel—natural gas or LNG; combustion-products parameters at gas-turbine inlet: $P_{in} \approx 2.5$ MPa, $T_{in} \approx 1650$ °C) and Rankine steam-turbine cycle (steam parameters at turbine inlet: $P_{in} \approx 12.5$ MPa ($T_{sat} = 327.8$ °C), $T_{in} \approx 620$ °C ($T_{cr} = 374$ °C)). (For details, see Fig. 2)	Up to 62%
2	Supercritical-pressure coal-fired power plant (Rankine-cycle steam inlet turbine parameters: $P_{in} \approx 23.5$ –38 MPa ($P_{cr} = 22.064$ MPa), $T_{in} \approx 540$ –625 °C ($T_{cr} = 374$ °C); and $P_{reheat} \approx 4$ –6 MPa ($T_{sat} = 250.4$ –275.6 °C), $T_{reheat} \approx 540$ –625 °C). (For details, see Fig. 3)	Up to 55%
3	Internal-combustion-engine generators (Diesel cycle and Otto cycle with natural gas as fuel)	Up to 50%
4	Subcritical-pressure coal-fired power plant (older plants; Rankine-cycle steam: $P_{in} = 17$ MPa ($T_{sat} = 352.3$ °C), $T_{in} = 540$ °C ($T_{cr} = 374$ °C); and $P_{reheat} \approx 3$ –5 MPa ($T_{sat} = 233.9$ –263.9 °C), $T_{reheat} = 540$ °C). (For T -s diagram, see Fig. 5C)	Up to 43%
5	Carbon-dioxide-cooled reactor (Advanced Gas-cooled Reactor (AGR)) NPP (Generation-III) (reactor coolant: $P = 4$ MPa, $T = 290$ –650 °C; and steam: $P_{in} = 17$ MPa ($T_{sat} = 352.3$ °C) and $T_{in} = 560$ °C ($T_{cr} = 374$ °C); and $P_{reheat} \approx 4$ MPa ($T_{sat} = 250.4$ °C), $T_{reheat} = 560$ °C). (For schematic and layout, see Fig. 4A and B, and for T -s diagram—Fig. 5C)	Up to 42%
6	Sodium-cooled Fast Reactor (SFR) (BN-600 & BN-800) NPP (steam: $P_{in} = 14.2$ MPa ($T_{sat} = 337.8$ °C), $T_{in} = 505$ °C ($T_{cr} = 374$ °C); and $P_{reheat} \approx 2.5$ MPa ($T_{sat} = 224$ °C), $T_{reheat} = 505$ °C). (For details, see Fig. 5A–C)	Up to 40%
7	Pressurized-Water-Reactor (PWR) NPP (Generation-III ⁺) (reactor coolant: $P = 15.5$ MPa ($T_{sat} = 344.8$ °C), $T_{out} = 327$ °C; steam: $P_{in} = 7.8$ MPa, $T_{in} = T_{sat} = 293.3$ °C; and $P_{reheat} \approx 2$ MPa ($T_{sat} = 212.4$ °C), $T_{reheat} \approx 265$ °C). (For details, see Fig. 6A–C)	Up to 36–38%
8	Pressurized-Water-Reactor (PWR) NPP (Generation-III, current fleet) (reactor coolant: $P = 15.5$ MPa ($T_{sat} = 344.8$ °C), $T = 292$ –329 °C; steam: $P_{in} = 6.9$ MPa, $T_{in} = T_{sat} = 284.9$ °C; and $P_{reheat} \approx 1.5$ MPa ($T_{sat} = 198.3$ °C), $T_{reheat} \approx 255$ °C). (For details, see Fig. 6, and for T -s diagram—Fig. 6C)	Up to 34–36%
9	Boiling-Water-Reactor (BWR) NPP (Generation-III, current fleet) reactor coolant: $P = 7.2$ MPa, $T_{out} = T_{sat} = 287.7$ °C; steam: $P = 7.2$ MPa, $T_{in} = T_{sat} = 287.7$ °C and $P_{reheat} \approx 1.7$ MPa ($T_{sat} = 204.3$ °C), $T_{reheat} \approx 258$ °C. (For schematic and layout, see Fig. 7A and B, and for T -s diagram—Fig. 6C)	Up to 34%
10	Light-water-cooled Graphite-moderated Reactor (LGR) (Russian RBMK-1000) NPP (Generation-III, current fleet) (reactor coolant: $P = 6.4$ MPa, $T_{out} = T_{sat} = 279.8$ °C; steam: $P = 6.4$ MPa, $T_{in} = T_{sat} = 279.8$ °C and $P_{reheat} \approx 0.3$ MPa ($T_{sat} = 133.5$ °C), $T_{reheat} \approx 263$ °C). (For schematic and layout, see Fig. 8A and B and for T -s diagram—Fig. 6C)	Up to 34%
11	Pressurized Heavy Water Reactor (PHWR) NPP (Generation-III, CANDU [®] -6, current fleet) reactor coolant: $P_{in} = 11$ MPa/ $P_{out} = 9.9$ MPa ($T_{sat} = 310.3$ °C) and $T = 260$ –310 °C; steam: $P_{in} = 4.7$ MPa, $T_{in} = T_{sat} = 260.1$ °C; and $P_{reheat} \approx 1.2$ MPa ($T_{sat} = 188$ °C), $T_{reheat} \approx 240$ °C. (For schematic and layout, see Figs. 9A and B and for T -s diagram—Fig. 6C)	Up to 32% (34%)
12	PWR SMR NPP (RITM-200M, Russia) (Generation-III ⁺) (not yet in operation as SMR NPP ^a) reactor coolant: $P = 15.7$ MPa ($T_{sat} = 345.8$ °C), $T = 277$ –313 °C; steam: $P_{in} = 3.82$ MPa, $T_{in} = 295$ °C ($T_{sat} = 247.6$ °C). (For layout, see Fig. 10B and for T -s diagram—Fig. 10D)	Up to ~31%
13	PWR SMR NPP (KLT-40S, Russia) (Generation-III, current fleet) reactor coolant: $P = 12.7$ MPa ($T_{sat} = 329$ °C), $T = 280$ –316 °C; steam: $P_{in} = 3.72$ MPa, $T_{in} = 290$ °C ($T_{sat} = 246.1$ °C). (For details, see Fig. 10A, C and D)	Up to ~26%

^aIn 2021, Rosatom will begin work on a site of the future SMR NPP, which is planned to be built in Yakutia. The SMR will be a RITM-200 ship-based reactor (<https://www.neimagazine.com/news/newsrosatom-to-being-work-on-land-based-smr-8436408>).

with subcritical-pressure Rankine steam-turbine cycle have significantly lower thermal efficiencies up to 1.5–2.4 times compared to those of the top thermal power plants.

It is reasonable to compare NPPs with various types of nuclear-power reactors based on thermal efficiencies (for details, see Table 2 and Figs. 3C, 6C and 8C), where AGR NPPs have the highest thermal efficiency in the nuclear-power industry of up to 42% (Figs. 2 and 3C), and SFR NPPs—up to 40% (Fig. 3). Such relatively high thermal efficiencies are due to higher temperatures inside reactors (in AGRs the outlet temperature of the carbon-dioxide reactor coolant is up to 650 °C, and in SFRs with the liquid-sodium reactor coolant—up to 550 °C), adopting subcritical-pressure Rankine cycles with primary and secondary superheated steam and relatively high primary-steam pressures—17 MPa and 14.2 MPa, respectively.

However, the last AGR (only 14 left in United Kingdom) was connected to a grid in 1989, i.e., with no new AGRs have been built for last 32 years. Possible reasons for stopping AGRs in the United Kingdom were the cost, build difficulties (problems with intermediate heat exchangers and thermal insulation), and issues with attaining on-line refueling at full-power plus lower costs alternatives. Therefore, unfortunately, these reactors/NPPs will be never built again.

In terms of SFRs, only two are currently in operation in Russia, and the relatively low boiling point of sodium (882.8 °C) means that these unique fast reactors/NPPs will not possibly have higher thermal efficiencies in the future. Currently, China is finalizing

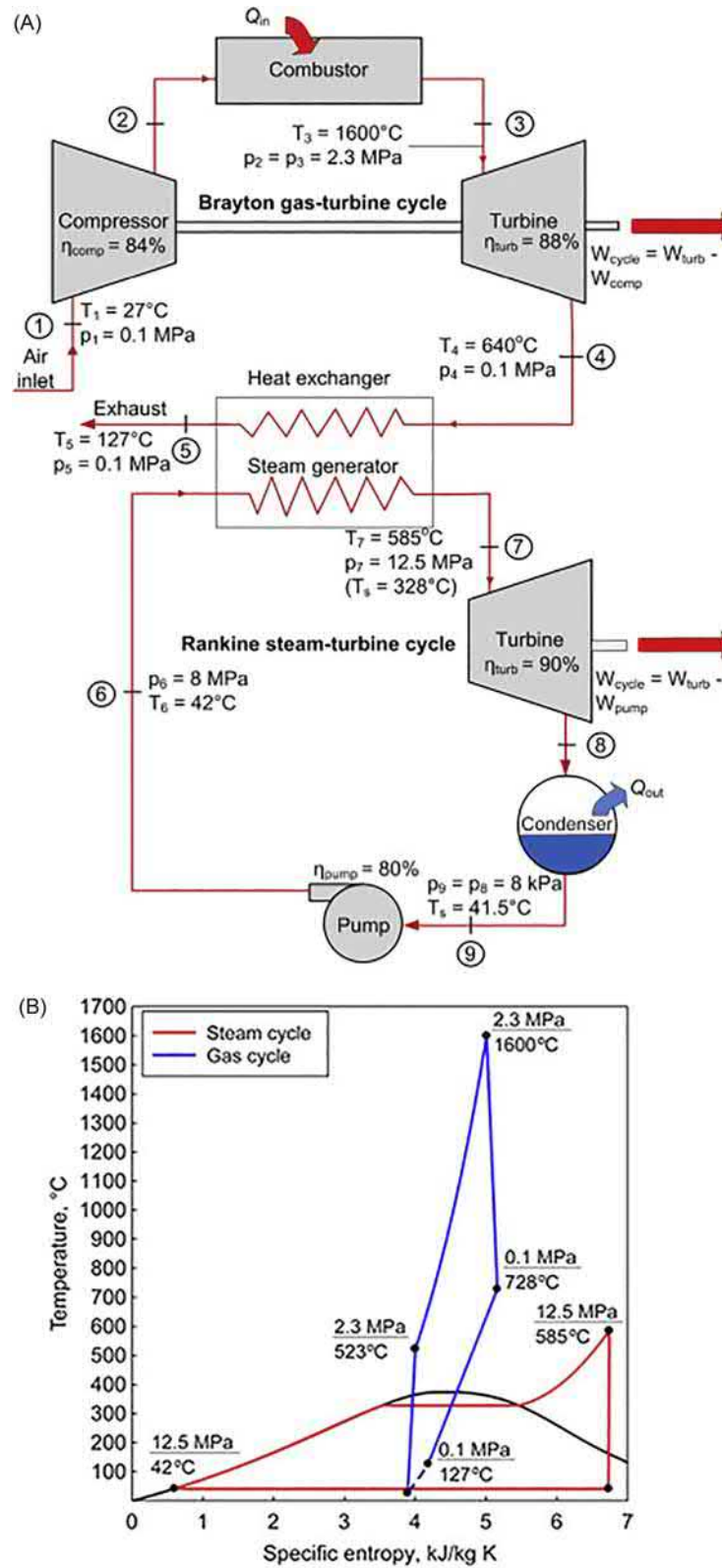


Fig. 2 Simplified layout (A) and T - s diagram (saturation line is only related to Rankine cycle) (B) of modern combined-cycle power plant (based on data from MHI and Siemens) (Pioro, 2016).

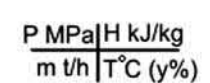


Fig. 3 (A) Single-reheat regenerative cycle 600-MW_{el} Tom'-Usinsk thermal power plant (Russia) layout (Kruglikov et al., 2009): Cyl—Cylinder; H—Heat exchanger (feedwater heater); CP—Circulation Pump; TDr—Turbine Drive; Cond P—Condensate Pump; GCHP—Gas Cooler of High Pressure; and GCLP—Gas Cooler of Low Pressure. (B) Simplified *T-s* diagram of supercritical-pressure Rankine steam-turbine cycle. Shown Rankine cycle is typical one for next generation or Generation-IV SuperCritical Water-cooled Reactors (SCWRs) (for exact parameters of SCWRs, see Table 3) and can be used in Lead-cooled Fast Reactor (LFR) NPP (at least, for Russian BREST-300 reactor).

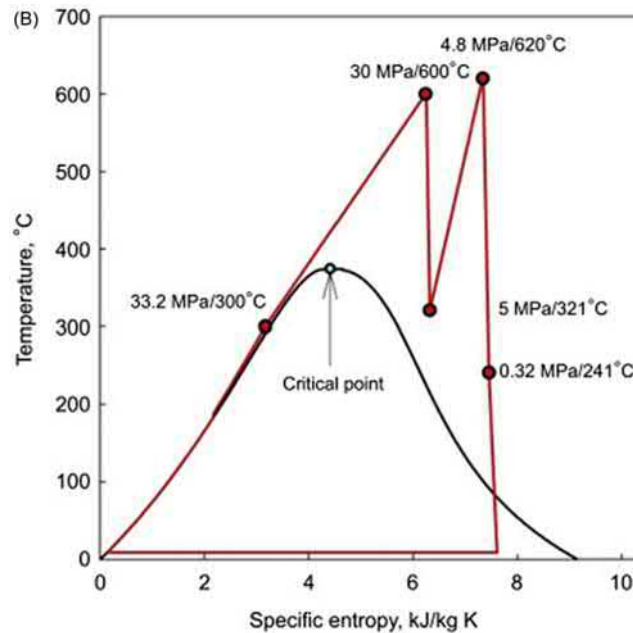


Fig. 3 (continued).

construction of the first China Fast Reactor (CFR-600) and has started construction of another one (<https://www.neimagazine.com/news/newschina-begins-construction-of-second-cfr-600-fast-reactor-8435608>).

The vast majority of NPPs are equipped with light- and heavy-water-cooled reactors (377 and 49, respectively), which is currently the largest group of nuclear-power reactors in the world, i.e., 426 from 442 or $\sim 96\%$. 424 of these reactors use a subcritical-pressure Rankine cycle with saturated primary steam (at pressures up to 7.8 MPa) and slightly overheated secondary steam (see Fig. 6C), so these NPPs have significantly lower thermal efficiencies (32–38%) compared to those of advanced modern thermal power plants (see Table 2).

The only exceptions are power cycles of Russian SMRs: KLT-40S (two of which are currently in operation) and RITM-200M (Pioro et al., 2019, 2020). These SMRs are equipped with subcritical-pressure Rankine cycles with overheated primary steam, but without a reheat option (see Table 2 and Fig. 10). Due to that, thermal efficiencies of these SMR NPPs are the lowest ones (i.e., 26% and 31%, respectively) compared to those of other light- and heavy-water-cooled reactors NPPs.

Analysis of T-s diagrams and the corresponding NPPs layouts (see Appendix A1 in Pioro (2016)) shows that NPPs with water-cooled reactors have possibly reached their maximum achievable thermal efficiencies (close to 36–38%), because further increase in reactor-coolant pressures (i.e., beyond 15–16 MPa) will lead to a small rise in the coolant saturation temperature (see Fig. 1 in Dragunov et al. (2015)). Moreover, it will lead to a significant decrease in Critical Heat Flux (CHF) at flow boiling (see Fig. 2 in Dragunov et al. (2015)).

Thermal efficiency of an NPP and any thermal power plant is a very important general parameter for comparison of various power plants. Higher thermal efficiency is better, because nuclear or thermal power plant will have less thermal/various emission pollutions per kg of fuel, less fuel per kW h of electricity generated, less cost of kW h, and less cooling water required. Currently, cooling shortfall is a serious concern due to the climate change and more often and more severe dryouts/higher water temperatures in rivers, lakes, and even seas/oceans around the world, etc.

However, it should be stated that in spite of the thermal efficiency being the driving force for advances in thermal and nuclear power plants, quite often other parameters, policies, or conditions influence decisions, which type of reactors/NPPs to build in a country or in a region. As such, Canada develops PHWRs, in particular, CANDU^{®1} reactors, from 1950s, and due to that, up till now, all 19 Canadian nuclear-power reactors are of PHWR type. India, originally a partner with Canada, has quite similar, but now an independent approach, therefore, the vast majority of their reactors are also of PHWR type.

United States and Japan have a number of large nuclear vendors, which, after being in a cooperative license, developed PWRs and BWRs/ABWRs. Therefore, they have only these types of nuclear-power reactors (United States—63 PWRs and 31 BWRs, and Japan—16 PWRs and 17 BWRs). On the other hand, France has only one nuclear vendor, and after terminating an United States cooperation license and stopping gas-reactor builds, now have all their NPPs equipped with only PWRs. Also, it should be mentioned that thermal efficiencies of water-cooled SMRs NPPs are lower than those of modern NPPs equipped with LWRs.

¹CANDU[®]—Registered Trademark of AECL, used under license by Candu Energy Inc., Member of SNC-Lavalin Nuclear Group.

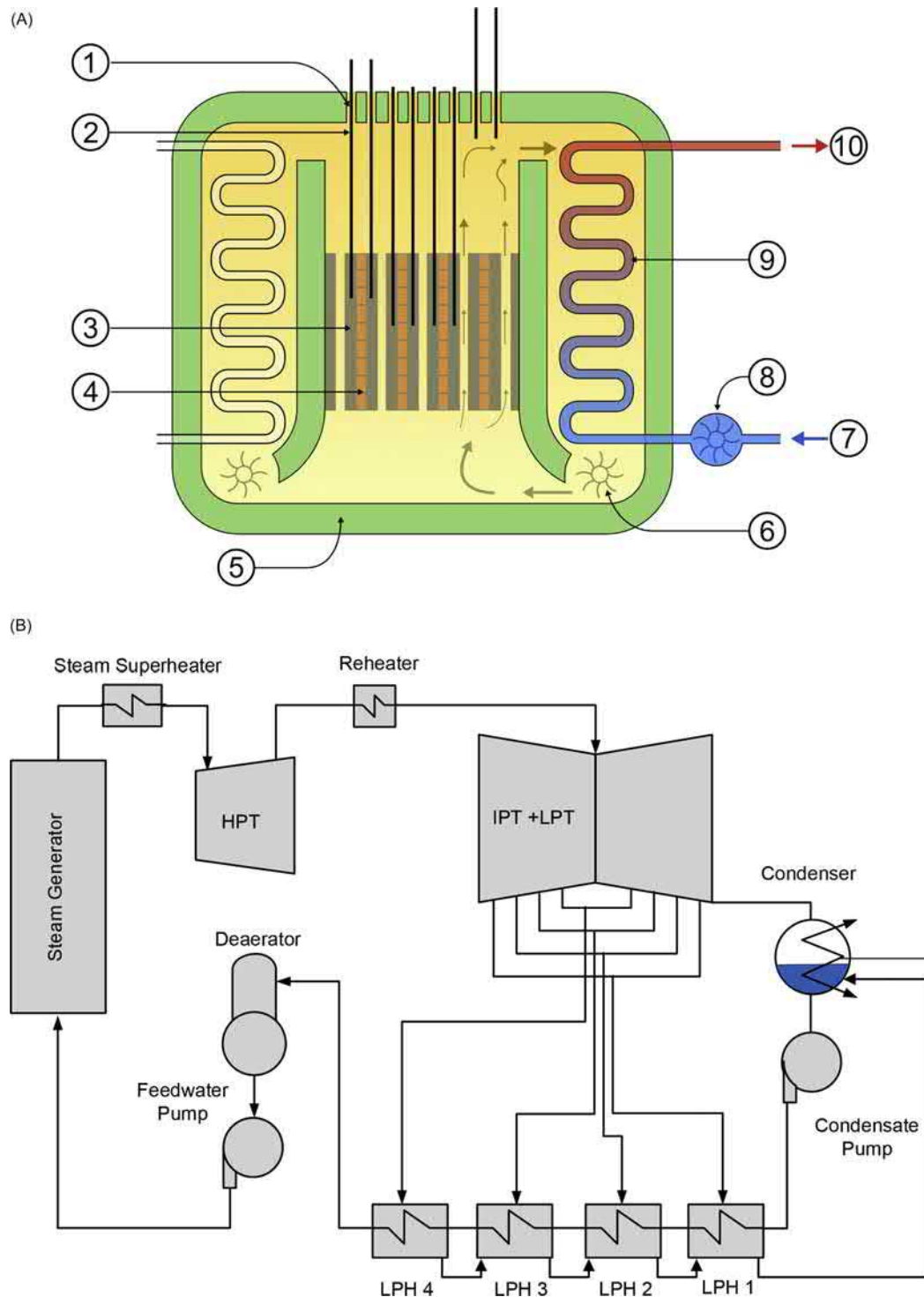


Fig. 4 (A) Simplified schematic of Advanced Gas-cooled Reactor (AGR) (Author: Messer Woland, https://commons.wikimedia.org/wiki/File:AGR_reactor_schematic.svg) (for details, see Pioro (2016)); (1) Charge tubes; (2) Control rods; (3) Graphite moderator; (4) Fuel assemblies; (5) Concrete pressure vessel and radiation shielding; (6) Gas circulation pump; (7) Water; (8) Water circulation pump; (9) Steam generator; and (10) Superheated steam. Steam generator is contained within steel-reinforced-concrete combined pressure vessel and radiation shield. (B) Simplified layout of AGR Torness NPP. Based on data from <http://large.stanford.edu/courses/2017/ph241/barry2/docs/nks-rak2-96-tr-c2.pdf>.

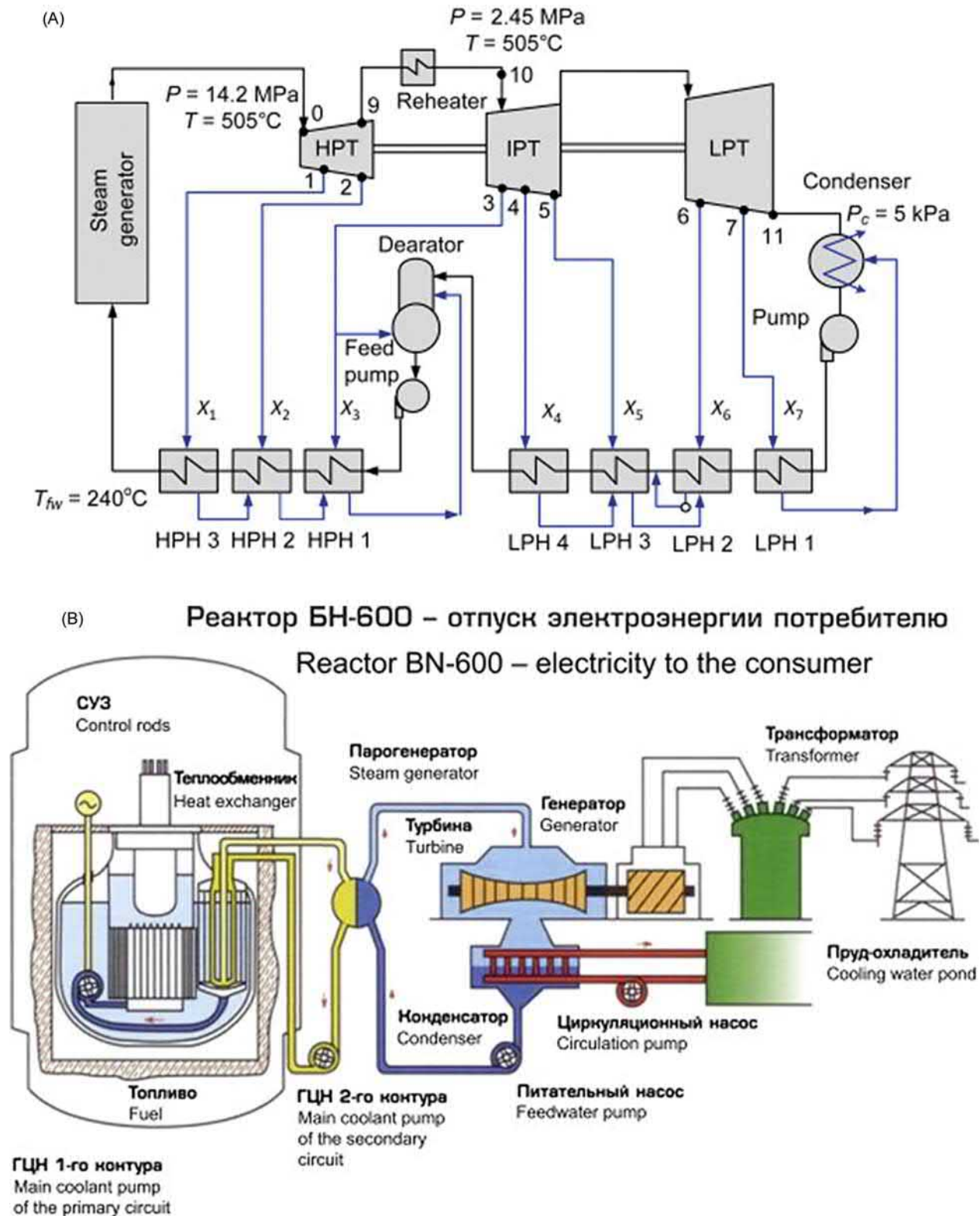


Fig. 5 (A) Simplified schematic of Liquid-Metal Fast-Breeder Reactor (LMFBR)—Sodium-cooled Fast Reactor (SFR) (Russian BN-600) NPP (courtesy of Rosenergoatom) (ROSENERGOATOM, 2004) (Pioro, 2016). (B) Detailed layout of SFR (BN-600) NPP (based on data from Grigor'ev and Zorin (1988) and Margulova (1995)) (for details, see Pioro (2016)). Secondary steam is reheated with liquid sodium from intermediate loop in preheater. (C) Simplified T - s diagram of subcritical-pressure superheated-steam Rankine cycle with reheat option of SFR (BN-600) NPP. Shown Rankine cycle is typical one for older, i.e., subcritical-pressure thermal power plants, AGR NPPs, and next generation or Generation-IV LFR NPPs (for exact parameters, see Tables 2 and 3, and Pioro (2016)).

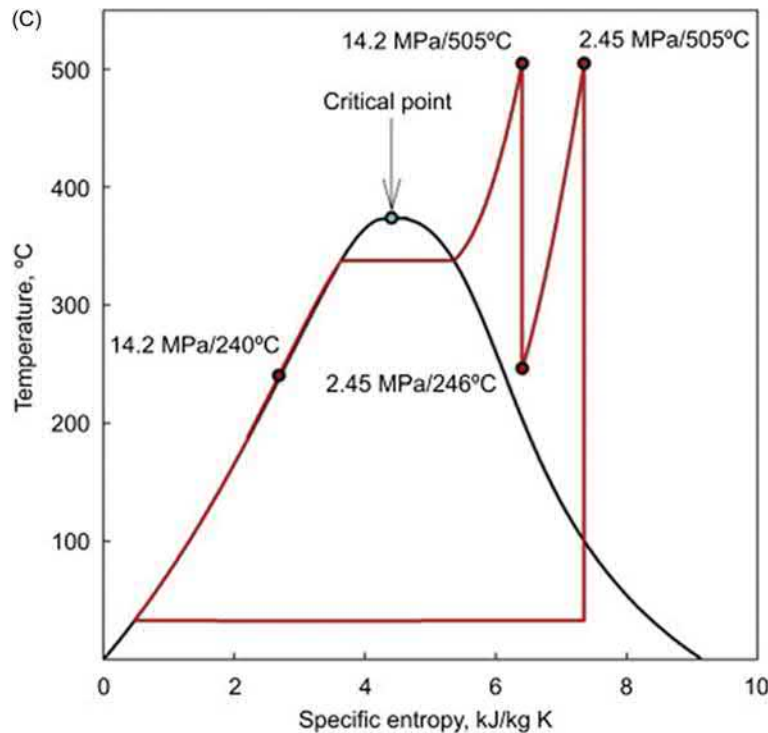


Fig. 5 (continued).

However, SMRs will undoubtedly have their unique “niche” applications of being implemented in remote areas, small electrical grids, military facilities, and as floating NPPs (Pioro et al., 2019, 2020; Pioro, 2016).

Current NPPs generate mainly electricity. Just few of them are used for district heating and hot-water supply and for water desalination. However, there are needs for process heat, hydrogen co-generation, etc., which cannot be achieved at modern NPPs due to relatively low temperatures inside reactors, especially, cooled with light and heavy water. Therefore, to increase thermal efficiencies to the level of those at modern advanced thermal power plants and to cover these new needs, the world has to move to high-temperature reactors, such as next generation or Generation-IV reactors (see Pioro, 2016).

Thermal efficiencies of next generation or Generation-IV NPPs

It should be mentioned that thermal-power industry is ahead of nuclear-power one in terms of using advanced power cycles (Pioro, 2016). Due to this next generation or Generation-IV nuclear-power reactors will be connected to combined power cycles, Brayton cycle, or to supercritical-pressure Rankine cycle (for details, see Table 3; Chapter 00301 by A. Stanculescu in this Encyclopedia; GIF (<https://www.gen-4.org/gif/>); and Pioro (2016)). However, in Brayton cycles combustion products will be replaced with helium, nitrogen, mixture of helium and nitrogen, or supercritical-pressure carbon dioxide (for details, see Pioro (2016) and Mahdi et al. (2018)) (Figs. 11–14).

Capacity factor

Another important general parameter of any power plant is a capacity factor.

Net capacity factor of a plant is the ratio of the actual output of a power plant over a period of time (usually, during a year) to its potential output, if it had operated at its full “nameplate” capacity the entire time. Capacity factors vary significantly depending on the type of a plant (for details, see Table 4).

It should be mentioned that, in general, capacity factors can be induced by Mother Nature, i.e., mainly, all renewable-sources power plants, or artificially induced, e.g., for thermal power plants to compensate variations in consumption of electricity and/or generation of electricity with renewable-sources power plants (for details, see Fig. 14); unreliable operation of plant equipment, long-terms for reactor re-fueling, etc.

In terms of capacity factors, NPPs have proven records of high capacity factors up to 100% for new plants, especially, compared to those of renewable-sources power plants (see Table 4). In general, there are two groups of nuclear-power reactors with: (1) Batch refueling (PWRs, BWRs, and SFRs) and (2) Refueling on-line (PHWRs, LGRs, and, partially, AGRs). In previous years, refueling on-

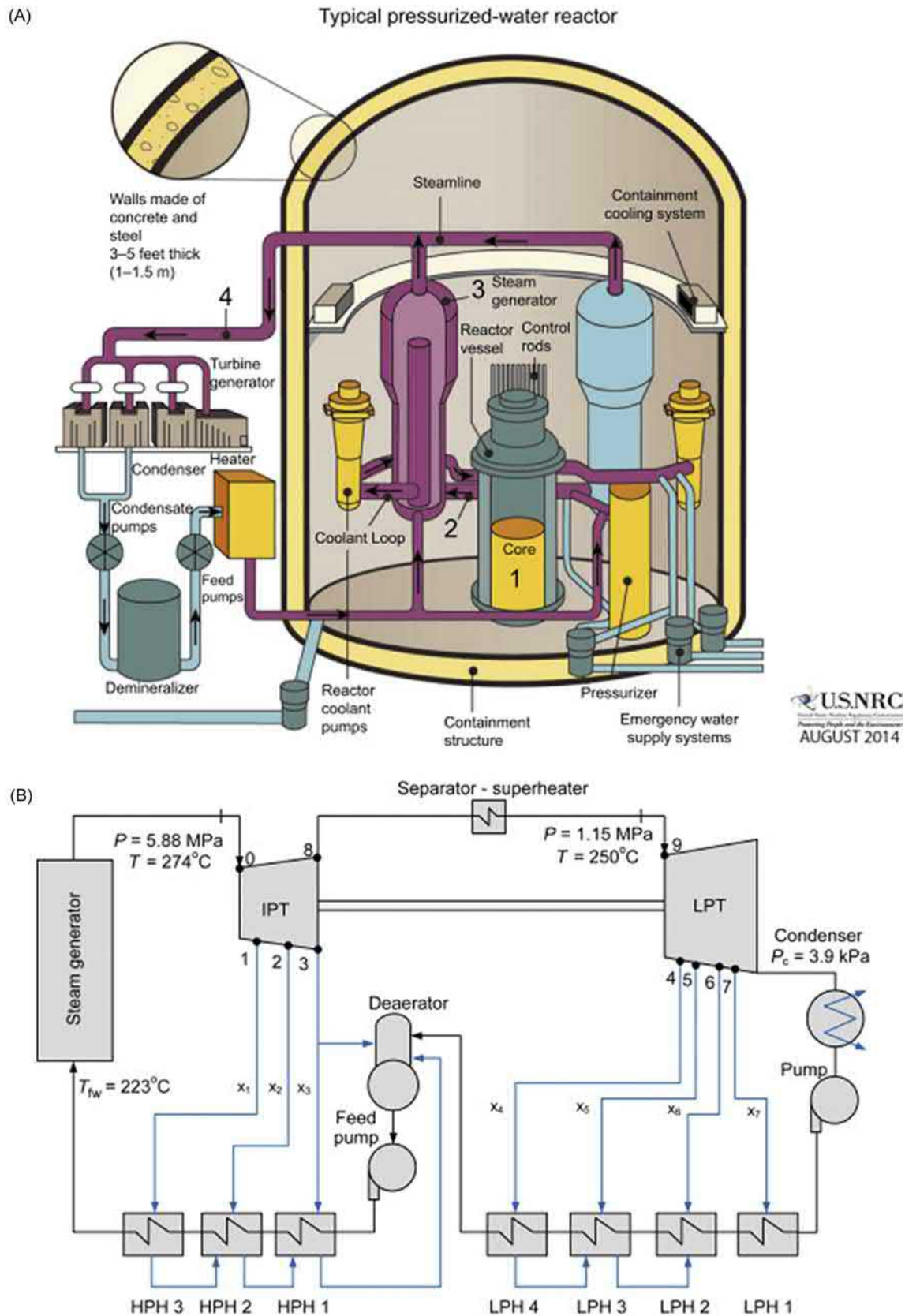


Fig. 6 (A) Simplified layout of typical US Pressurized Water Reactor (PWR) NPP (courtesy of USA NRC) (for details, see [Pioro \(2016\)](#)). (B) Detailed layout of typical PWR (VVER-1000 NPP (winter operation)) (based on data from [Grigor'ev and Zorin \(1988\)](#) and [Margulova \(1995\)](#)) (for details, see [Pioro, 2016](#)). (C) Simplified T - s diagram of subcritical-pressure saturated-steam Rankine cycle with reheat option (secondary steam is reheated with primary steam in preheater) of PWR (VVER-1000) NPP (summer operation): HP, high pressure; LP, low-pressure; MS, moisture separator; SG, steam generator. Shown Rankine cycle is typical one for light- and heavy-water-cooled reactors: PWRs, BWRs, LGRs, and PHWRs (for exact parameters, see [Table 2](#) and [Pioro, 2016](#)).

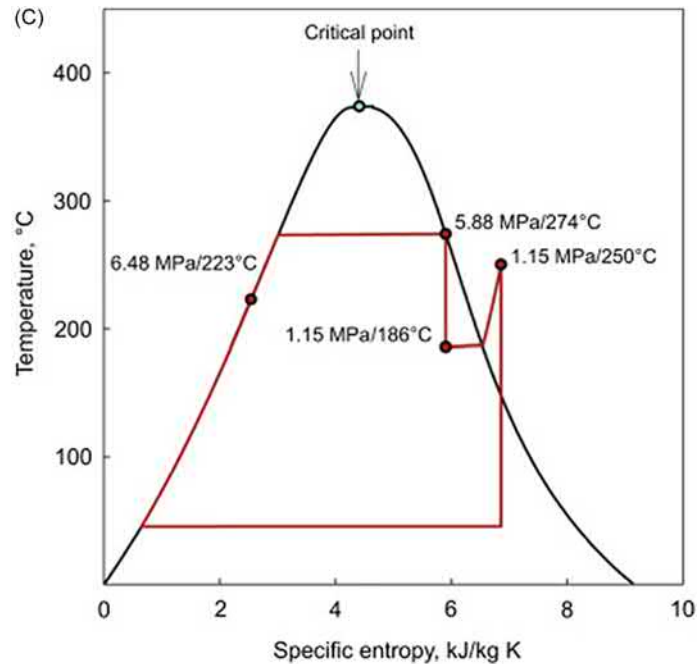


Fig. 6 (continued).

line allows reactors to run continuously for longer time, and, at that time, it was some advantage of such reactors/NPPs compared to batch-refueling ones. However, nowadays, batch-refueling reactors started to use slightly higher fuel enrichments of up to 5% and higher fuel burnups, which allowed to prolong their continuous operation, and effective NPPs outage management decreased a number of days required for refueling. Therefore, now the advantages of the on-line refueling are not so obvious compared to that of the batch refueling.

How various energy sources generate electricity in a grid can be illustrated based on the Province of Ontario (Canada) system. It should be noted that all, i.e., 19, nuclear-power reactors in Canada are of CANDU®-reactor type, and 18 of these reactors are located in the Province of Ontario (for details, see Pioro et al. (2019) and Pioro (2016)). Currently, the Province of Ontario has completely eliminated coal-fired power plants from its electrical grid. Some of them were closed, others—converted to natural gas. Ontario major installed capacities in 2015 were nuclear (38%), natural gas (29%), hydro (25%), and renewables (mainly wind) (8%). However, electricity was mainly generated by nuclear (60%), hydro (24%), natural gas (8.7%), and renewables (mainly wind) (6.9%).

Fig. 14A shows power generated, and Fig. 14B—capacity factors of various energy sources in Ontario electrical grid in summer (June 17, 2015). Analysis of the data in Fig. 14 shows that a full mix of nuclear, hydro, gas, wind, biofuel, and solar were the major sources for electricity generation (for operation of the Ontario grid with coal-fired power plants, see Pioro (2016) and Pioro and Duffey (2015)).

Electricity that day from midnight till 3.00 a.m. was mainly generated with nuclear, hydro, gas, wind, and biofuel (see Fig. 14A). After 3.00 a.m., biofuel power plants have increased slightly electricity generation followed by hydro and gas-fired power plants due to increased consumption of electricity in the province. Also, at the same time, wind power plants have also slightly increased electricity generation. However, after 7.00 a.m. wind power started to fluctuate and, eventually, decreased significantly. After 6.00 a.m., solar-power plants started to generate electricity.

During a day, hydro, gas-fired and biofuel power plants had variable electricity generation to compensate changes in consumption of electrical energy and variations in generating electricity from wind and solar power plants. After 9 p.m. in the evening, energy consumption started to drop in the province, and at the same time, wind power increased. Therefore, gas-fired, hydro, and biofuel power plants have decreased their energy generation accordingly due to the preferred feed-in-tariffs allocated to wind.

It should be noted that NPPs operated at about 100% of installed capacity providing reliable basic power to the grid. This example shows clearly that any grid that includes NPPs and/or renewable-energy sources must also include “fast-response” power plants such as gas-fired, coal-fired and/or large hydro-power plants to compensate changes in consumption of electrical energy per day and variations in electricity supply by wind and/or solar power plants.

Also, it should be mentioned that in the current chapter only two basic general parameters of power plants were considered. However, there are other important parameters of power plants, which include capital and operational costs, eventually, generated electricity costs, fuel costs, environmental impact, operational term, area covered by a plant, etc.

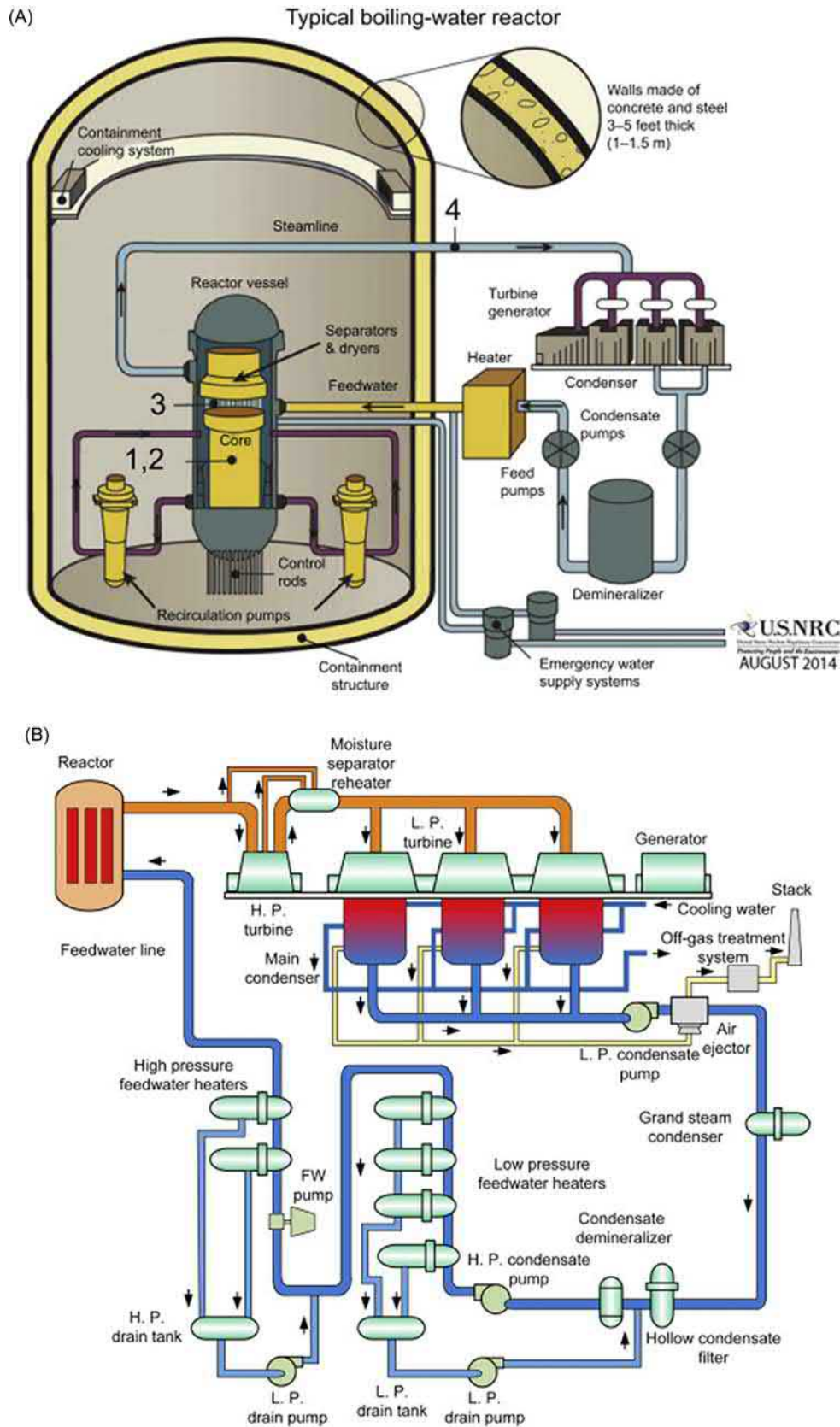


Fig. 7 (A) Simplified layout of typical Boiling Water Reactor (BWR) NPP (courtesy of USA NRC) (for details, see [Pioro \(2016\)](#)). Steam quality at outlet of reactor core is $\sim 10\text{--}15\%$. (B) Detailed layout of ABWR NPP (adapted and simplified from Advanced Boiling Water Reactor (ABWR), Booklet, Toshiba Corporation, 2011).

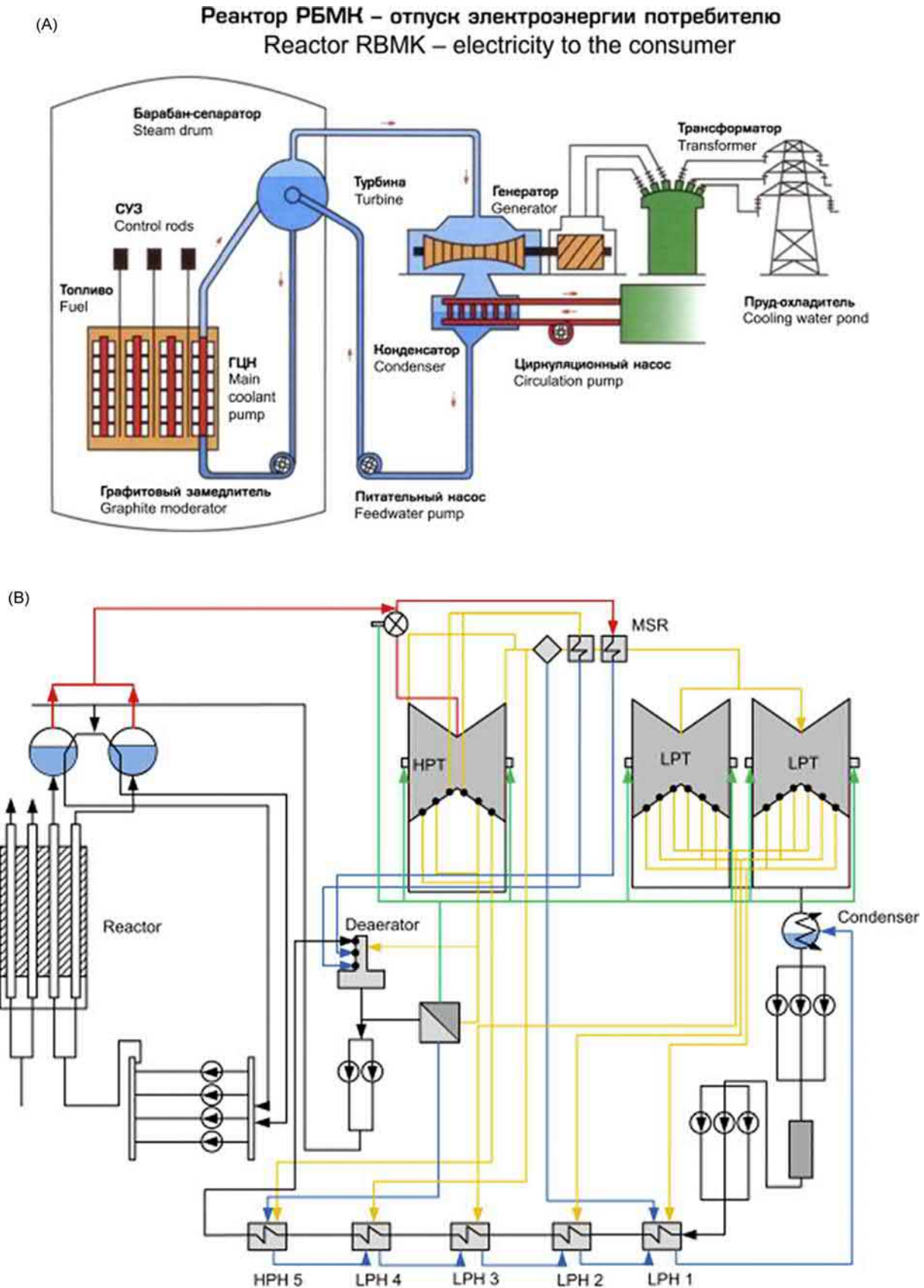


Fig. 8 (A) Simplified layout of 1000-MW_{el} Light-water-cooled Graphite-moderated Reactor (LGR) (Russian RBMK-1000) NPP (courtesy of Rosenergoatom) (ROSENERGOATOM, 2004) (for details, see Piro, 2016). Steam quality at outlet of reactor core is ~15%. (B) Detailed layout of RBMK-1000 NPP (based on data from RBMK (2006) and Piro (2016)).

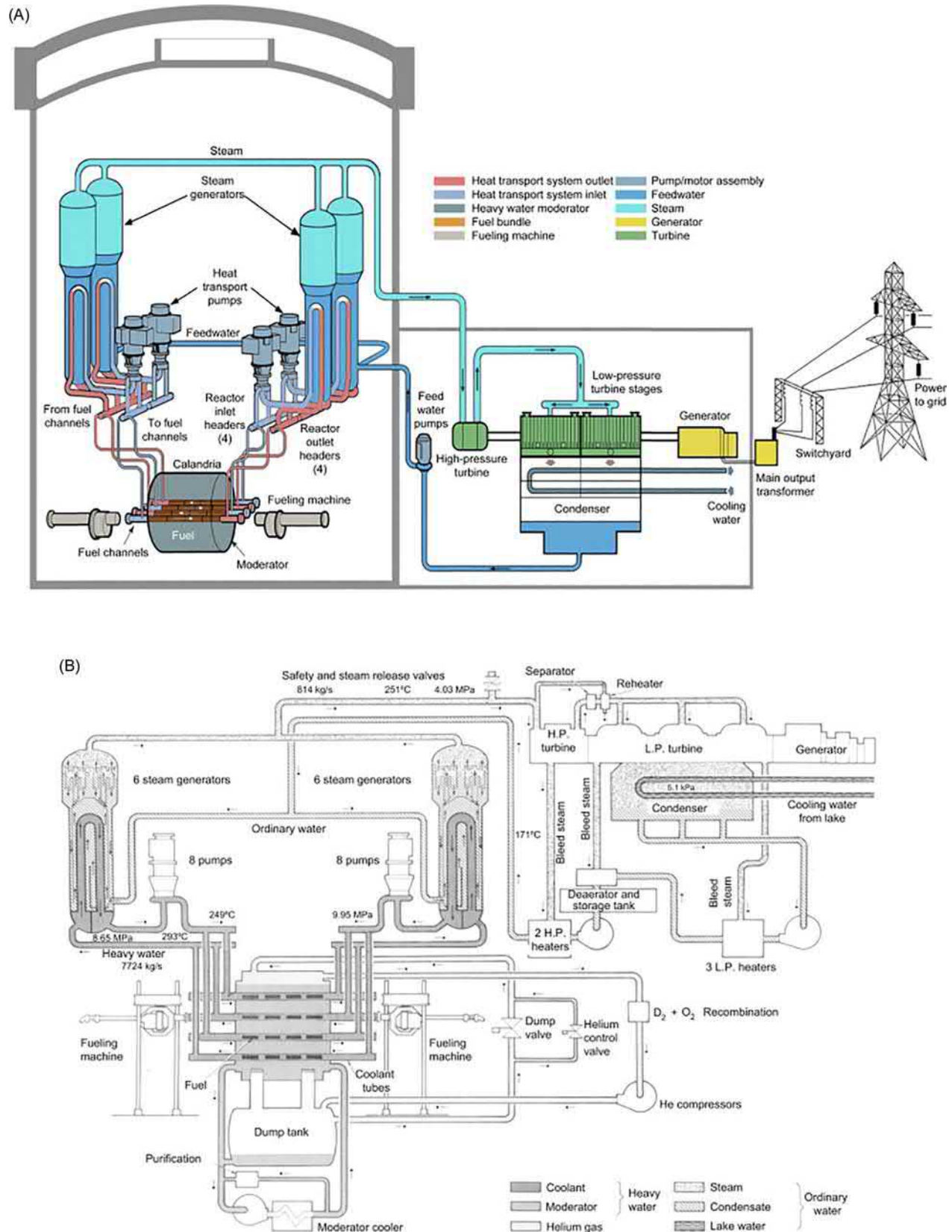


Fig. 9 (A) Simplified layout of CANDU®-reactor NPP (used with the courtesy and permission of SNC-Lavalin nuclear group): Moisture Separator and Reheater (MSR) between high- and low-pressure turbine cylinders (stages) are not shown. (B) Detailed layout of 515-MW_e CANDU-reactor NPP (Pickering NPP, Ontario, Canada) (AECL Report, 1969): This 515-MW_e CANDU reactor is the smallest and the oldest one in Canada (was connected to grid in 1971) (for details on CANDU® reactors, see Duffey et al. (2021) and Piro (2016)).

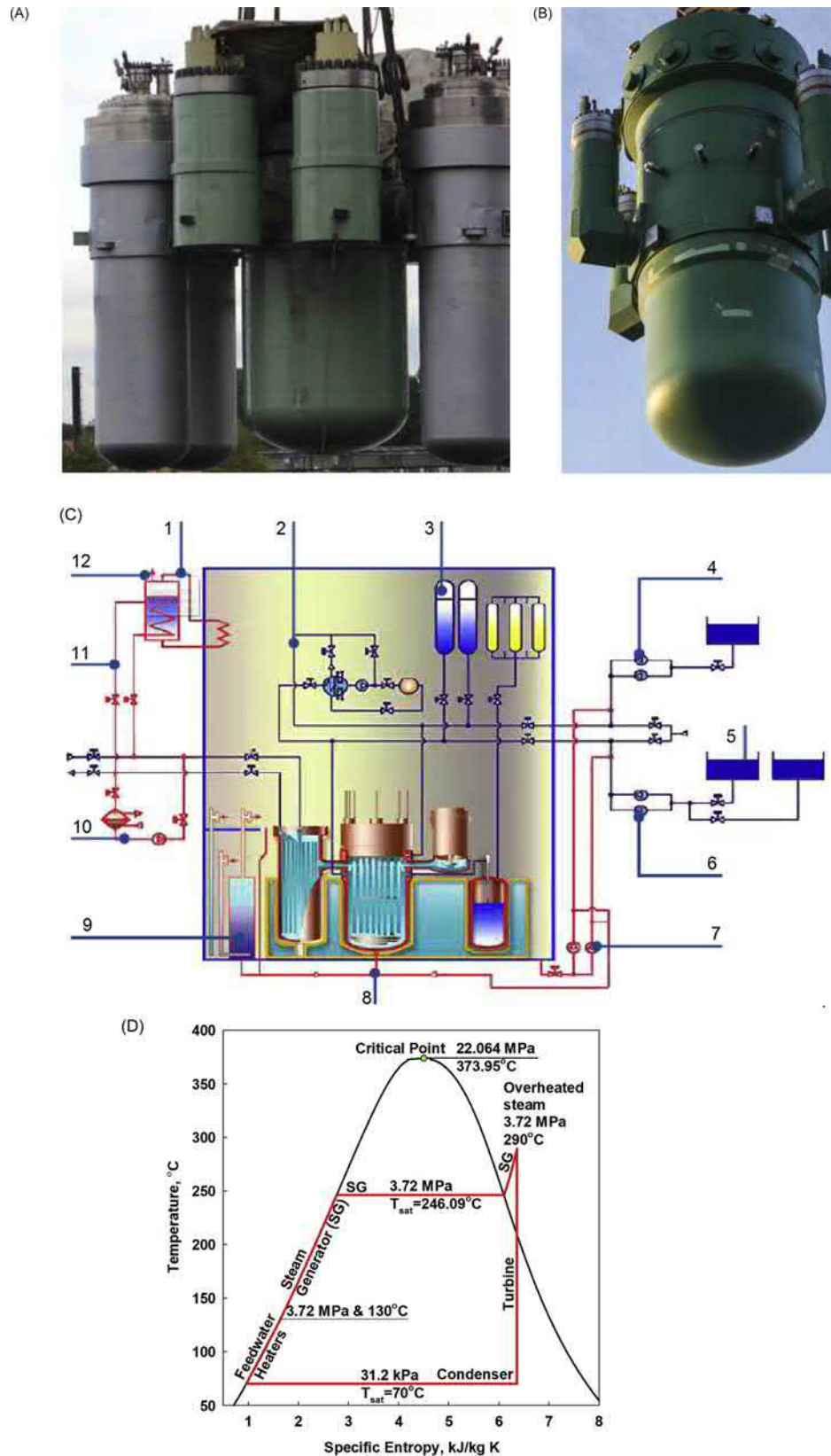


Fig. 10 (A and B) Russian PWRs-SMRs (courtesy of Rosatom) (Pioro et al., 2020; <http://www.okbm.nnov.ru/media-center/booklets/>): (A) KLT-40S (in center) with four steam generators (larger cylinders) and four reactor-coolant circulation pumps (smaller cylinders) located outside (<https://www.flickr.com/photos/rosatom/42425233062/in/album-72157692330711570/>); and (B) RITM-200M with steam generators integrated into reactor pressure vessel and four reactor-coolant circulation pumps (smaller cylinders) located outside. (C) Schematic of KLT-40S reactor and its systems (Based on original figures from AO OKBM by the name of I.I. Afrikanov, Brochures on KLT-40S (in red—newly introduced safety systems): (1)

Conclusions

One of the major parameters of any power plant is efficiency or thermal efficiency. The thermal efficiency is the major driving force for all advances in thermal and nuclear power plants. Ranges of gross thermal efficiencies of modern power plants are as the following: (1) Combined-cycle thermal power plants—up to 62%; (2) Supercritical-pressure coal-fired thermal power plants—up to 55%; (3) Subcritical-pressure coal-fired thermal power plants—up to 43%; (4) Carbon-dioxide-cooled reactor NPPs—up to 42%; (5) Sodium-cooled Fast Reactor (SFR) NPPs—up to 40%; (6) Modern water-cooled-reactor NPPs—30–36% (up to 38%); and (7) Current PWR-SMRs NPPs—26–31%.

However, all current Generation-II and III and oncoming Generation-III⁺ NPPs, especially, those equipped with light- and heavy-water-cooled reactors, are not competitive with modern thermal power plants in terms of thermal efficiency (30–36% (up to 38%) for current NPPs with light- and heavy-water-cooled reactors and up to 55%/62% for supercritical-pressure coal-fired and combined-cycle power plants, respectively).

Comparison of various types of reactors/NPPs showed that there is no reactor design, which will be the best for all the possible parameters. In general, thermal efficiency of an NPP can be one of the important factors to be considered. In this case, higher thermal efficiency is better, because NPP will have less thermal/various emissions pollutions per kg of fuel, less fuel per kW h of electricity generated, less cost of kW h, less cooling water required (which, currently, is a serious concern due to the climate change and more often and more severe dryouts/higher water temperatures in rivers, lakes, and even seas/oceans around the world), etc. Also, higher thermal efficiencies of NPPs make nuclear power more competitive with thermal and other power plants.

Current NPPs are mainly generate electricity. Just few on them are used for district heating and hot-water supply, and for water desalination. However, there are needs for process heat, hydrogen co-generation, etc., which cannot be achieved at modern NPPs due to relatively low temperatures inside reactors, especially, cooled with light and heavy water. Therefore, to increase thermal efficiencies to the level of those at modern advanced thermal power plants and to cover these new needs, the world has to move to high-temperature reactors, such as next generation or Generation-IV reactors.

Another important parameter of any power plant is a capacity factor. It should be mentioned that, in general, capacity factors can be induced by Mother Nature, i.e., mainly, all renewable-sources power plants, or artificially induced, e.g., for thermal power plants to compensate variations in consumption of electricity and/or generation of electricity with renewable-sources power plants (for details, see Fig. 14); unreliable operation of plant equipment, long-terms for reactor re-fueling, etc.

Also, it should be mentioned that in the current chapter only two basic general parameters of power plants were considered, i.e., thermal efficiency and capacity factor. However, there are other important parameters of power plants, which include capital and operational costs, eventually, generated electricity costs, fuel costs, environmental impact, operational term, area covered by a plant, etc.

The major advantages of nuclear power are well-known, including cheap reliable base-load power, high capacity factor, low carbon-dioxide emissions, and minor environmental impact. However, these factors are offset today by a competitive economic disadvantage with cheap natural gas combined-cycle power plants. Hopefully, Generation-IV reactors/NPPs under development will close or reduce the efficiency gap.

Introduction: Electricity generation in the world and selected countries

It is well-known that electricity generation and its consumption are the key factors for advances in industry, agriculture, technology, and the level of living. Table 1 shows clearly a strong dependence of Human Development Index (HDI) (Rows No. 4 and 5) from Electrical-Energy Consumption (EEC) per capita (Row 3); current population is shown in Row 1 (for details, see Pioro et al. (2019, 2020); Pioro (2016); Pioro and Duffey (2015)). Also, robust power industry with diverse energy sources is very important for a country's economic and political independence.

In general, electricity (see Table 1) can be generated from: (1) traditional fuels (i.e., non-renewable sources) such as coal, natural gas, oil, and nuclear utilizing mined resources; and (2) so-called, "renewables" such as hydro, biomass, wind, geothermal, solar, and marine (tidal and wave) power utilizing geotechnical resources.

Today, the main sources for global electrical-energy generation (see Table 1) are: (1) Thermal power—primarily using coal (38.3%) and secondarily using natural gas (22.9%); (2) "Large" hydro-electric power plants (16.3%); and (3) Nuclear power (10.2%). The rest 12.3% of the electrical energy is generated using renewables (wind, biomass, geothermal, solar, and marine energy) (6.6%), oil (3.3%), and other sources (2.4%).

Passive system of containment emergency pressure decrease (condensing system); (2) Active emergency cooling system through heat exchangers of loops I–III; (3) Passive emergency core cooling system (hydraulic accumulators); (4) Active emergency core cooling system from feedwater pumps; (5) Active system for injecting liquid absorber; (6) Active emergency core cooling system from feedwater pumps; (7) Active emergency core cooling system through recirculation pumps; (8) System of reactor caisson filling with water; (9) Containment passive emergency pressure decrease system (bubbling); (10) Active emergency shutdown cooling system (through process condensers); (11) Passive emergency shutdown cooling system; and (12) To atmosphere. (D) Simplified T-s diagram of subcritical-pressure overheated-steam PWR-SMR (KLT-40S) NPP (Akademik Lomonosov floating NPP, Chukotka, Russia). Shown Rankine cycle is typical one for RITM-200M NPP (for exact parameters, see Table 2 and Pioro et al. (2020)). Diagram is prepared based on data available in open literature.

Table 3 Estimated ranges of thermal efficiencies (gross) of Generation-IV NPP concepts (Generation-IV concepts are listed according to thermal-efficiency decrease) (Pioro, 2016).

No	Nuclear power plant	Gross eff. %
1	Very High Temperature Reactor (VHTR) NPP reactor coolant—helium (SCF): $P = 7$ MPa and $T_{in}/T_{out} = 640/1000$ °C; primary power cycle—direct SCP Brayton helium-gas-turbine cycle (however, originally, VHTR concept was intended for hydrogen generation); possible back-up—indirect Brayton or combined cycles (see Fig. 11)	≥ 55
2	Gas-cooled Fast Reactor (GFR) or High Temperature Reactor (HTR) NPP (reactor coolant—helium (SCF): $P = 9$ MPa and $T_{in}/T_{out} = 490/850$ °C; primary power cycle—direct SCP Brayton helium-gas-turbine cycle (see Fig. 12); possible back-up—indirect SCP Brayton or combined cycles (see Fig. 13))	≥ 50
3	SuperCritical Water-cooled Reactor (SCWR) NPP (one of Canadian concepts; reactor coolant—SC light water: $P = 25$ MPa and $T_{in}/T_{out} = 350/625$ °C ($T_{cr} = 374$ °C); direct cycle; SCP Rankine cycle with high-temperature secondary-steam superheat: $T_{out} = 625$ °C; possible back-up—indirect SCP Rankine “steam”-turbine cycle with high-temperature secondary-steam superheat). (For details, see Fig. 3)	45–50
4	Molten Salt Reactor (MSR) NPP (reactor coolant—sodium-fluoride salt with dissolved uranium fuel: $T_{in}/T_{out} = 700/800$ °C; primary power cycle—indirect SCP carbon-dioxide Brayton gas-turbine cycle; possible back-up—indirect Rankine steam-turbine cycle or combined cycles)	~ 50
5	Lead-cooled Fast Reactor (LFR) NPP Russian design BREST-OD-300 ^a : reactor coolant—liquid lead: $P \approx 0.1$ MPa and $T_{in}/T_{out} = 420/540$ °C; primary power cycle—indirect subcritical-pressure Rankine steam cycle: $P_{in} \approx 17$ MPa ($P_{cr} = 22.064$ MPa) and $T_{in}/T_{out} = 340/505$ °C ($T_{cr} = 374$ °C); high-temperature secondary-steam superheat; (for typical T - s diagram, see Fig. 5C); in one of the previous designs of BREST-300 NPP primary power cycle was indirect SCP Rankine “steam” cycle: $P_{in} \approx 24.5$ MPa ($P_{cr} = 22.064$ MPa) and $T_{in}/T_{out} = 340/520$ °C ($T_{cr} = 374$ °C) (For details, see Fig. 3); also, note that power-conversion cycle in a different LFR designs from other countries is based on SCP carbon-dioxide Brayton gas-turbine cycle	~ 41 –43
6	Sodium-cooled Fast Reactor (SFR) NPP Russian design BN-600: reactor coolant—liquid sodium (primary circuit): $P \approx 0.1$ MPa and $T_{in}/T_{out} = 380/550$ °C; liquid sodium (secondary circuit): $T_{in}/T_{out} = 320/520$ °C; primary power cycle—indirect Rankine steam-turbine cycle: $P_{in} \approx 14.2$ MPa ($T_{sat} \approx 337$ °C) and $T_{in\ max} = 505$ °C ($T_{cr} = 374$ °C); secondary-steam superheat: $P \approx 2.45$ MPa and $T_{in}/T_{out} = 246/505$ °C; (For details, see Fig. 5); possible back-up in some other countries—indirect SCP carbon-dioxide Brayton gas-turbine cycle	~ 40

^aBREST-OD-300 is Fast Reactor with “NATural safety”-Test-Demonstration in Russian abbreviations (БРЕСТ-ОД-300, Быстрый Реактор с Естественной безоПасностью—ОПытно—Демонстрационный).

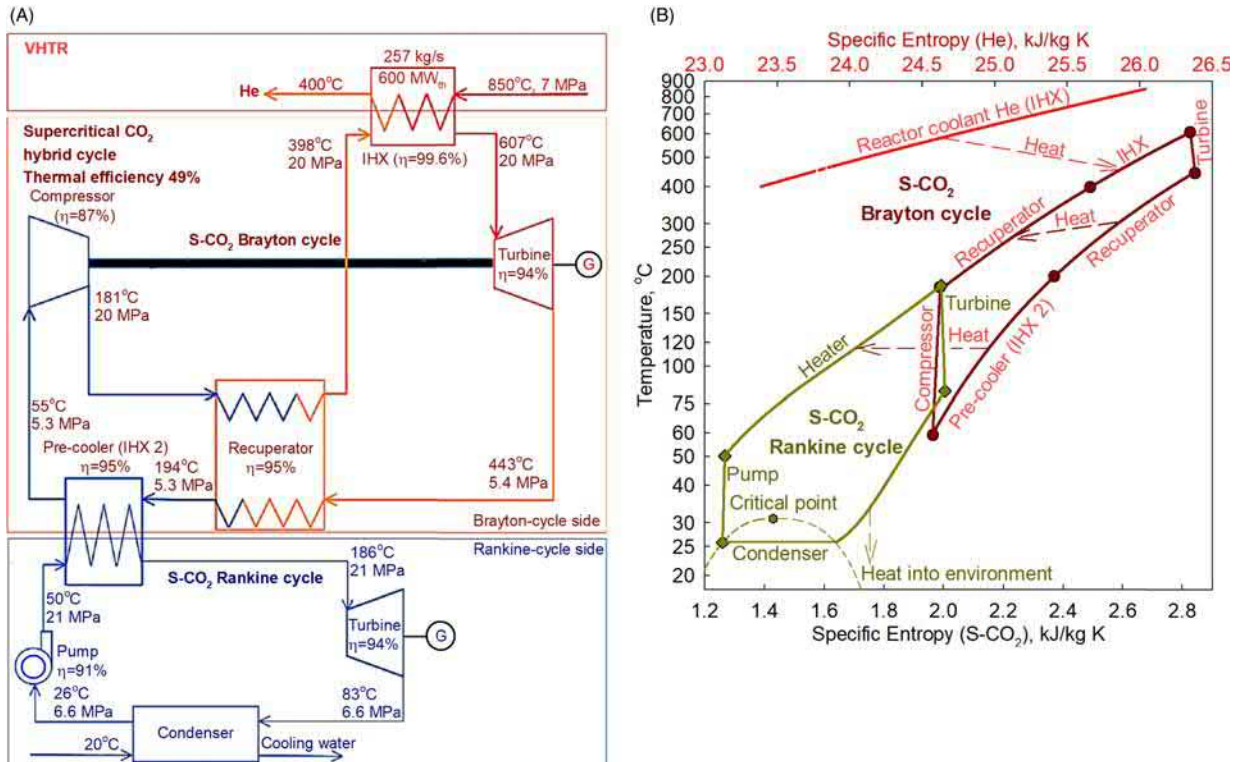


Fig. 11 (A) Simplified layout of 600-MW_{th} VHTR NPP with combined power cycle: Supercritical carbon-dioxide (S-CO₂) Brayton and Rankine cycles (based on figure from Bae et al. (2014)) and (B) Simplified T - s diagrams (Mahdi et al., 2018).

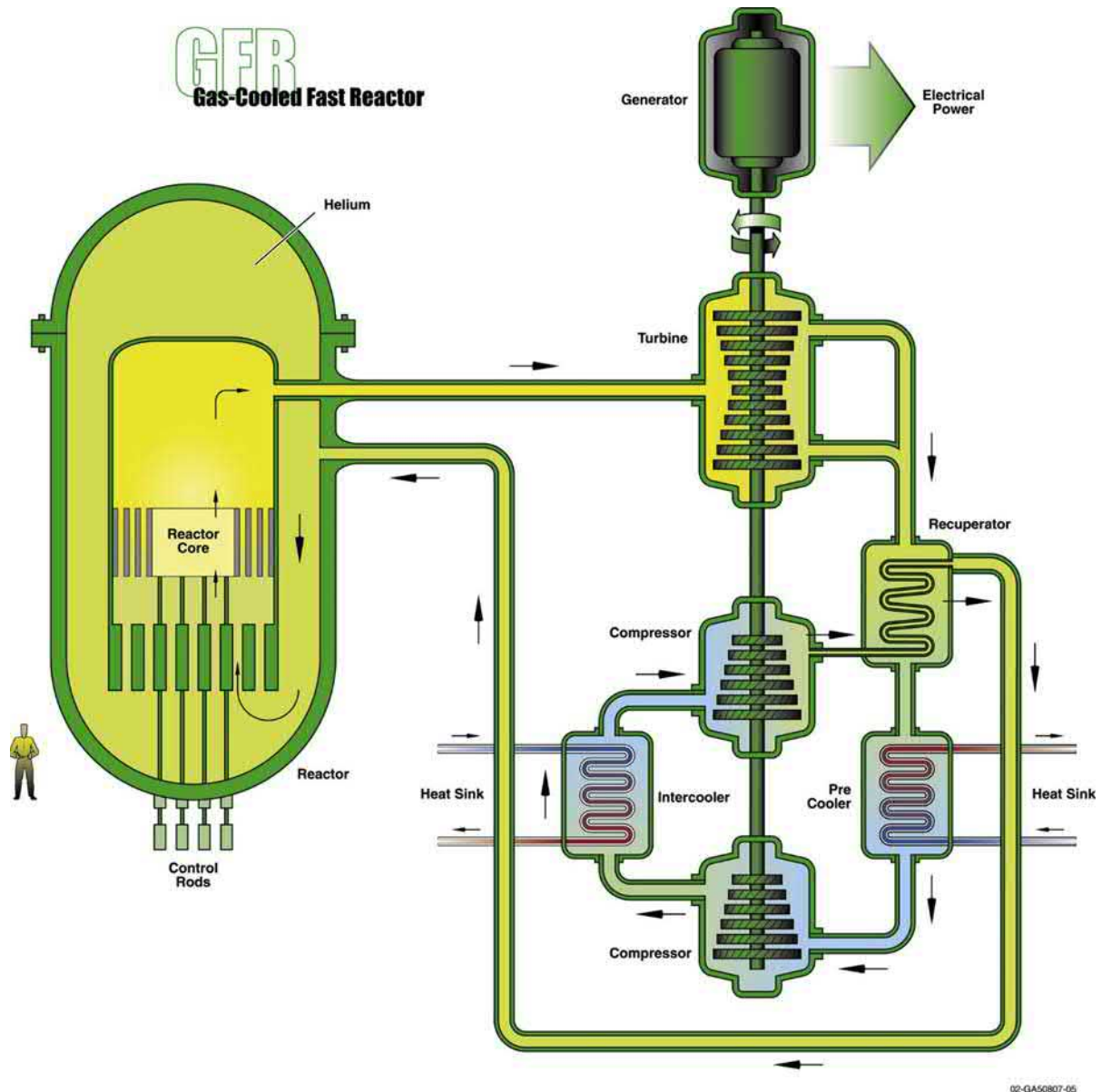


Fig. 12 Simplified layout of 600-MW_{th} GFR concept considered initially by GIF with direct Brayton helium power cycle (Courtesy of GIF) (see also, Pioro, 2016).

Table 1 also lists 13 countries, which have the largest number of nuclear-power reactors/their installed capacities. These countries are listed according to the decreasing population.

Analysis of the data in **Table 1** shows that the world (~38%) and, especially, countries with the largest population, i.e., China (~66%) and India (~75%), as well as Germany (~37%) and South Korea (~42%) rely heavily on coal for the electricity generation. The United States (~39%), Russia (~59%), Japan (~37%), and United Kingdom (~44%) use mainly natural gas or Liquefied Natural Gas (LNG) for electricity generation, which is better than to use coal, but still, for now, we cannot avoid emissions of carbon dioxide and NO_x due to a combustion process. On opposite, France (~72%) and Ukraine (~54%) rely heavily on nuclear power, which is, in general, the lowest emitter of carbon dioxide compared to all other electricity-generating sources including renewables (for details, see Pioro et al., 2019). Canada and Sweden are the only countries from the 13 ones, which rely heavily on the hydro electricity generation ~61% and ~39%, respectively.

Analysis of the data on installed capacities of various power plants in the world (https://en.wikipedia.org/wiki/List_of_largest_power_stations) shows that the 22 largest power plants have installed capacities ranging within 5700–22,500 MW_{el}. Within these power plants the largest group is hydro plants (in total 12) with installed capacities ranging from 5850 and up to 22,500 MW_{el}, six Nuclear Power Plants (NPPs) (installed capacities 5700–7965 MW_{el}) (for now, the largest in the world NPP—Kashiwazaki-Kariwa NPP is not in service after the Fukushima-Daiichi NPP severe accident in March of 2011), 2 coal- and 1 gas-fired thermal power

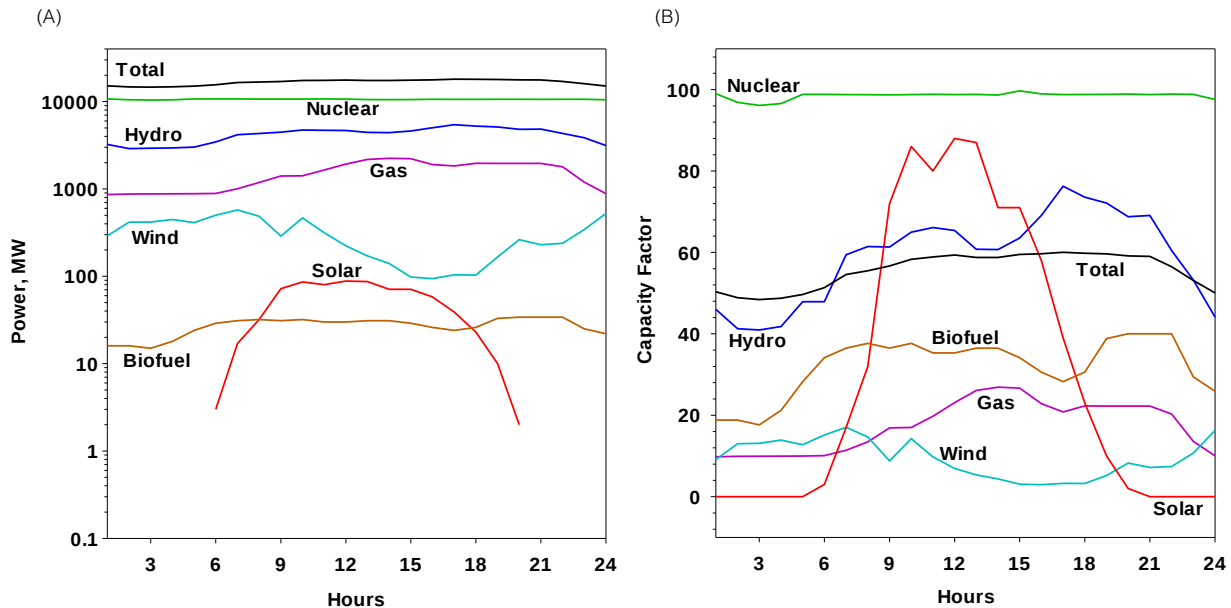


Fig. 14 Power generated (A) and capacity factors (B) of various energy sources in Ontario (Canada) in hot summer working day—June 17 (Wednesday), 2015 (based on data from <http://ieso.ca/imoweb/marketdata/genEnergy.asp>) (shown here for reference purposes) (Pioro et al., 2019 and Pioro, 2016).

Table 4 Average (typical) capacity factors of various power plants (Pioro et al., 2019; Pioro, 2016; https://en.wikipedia.org/wiki/Capacity_factor).

No.	Power plant type	Locations	Years	Capacity factor %
1	Nuclear	United States	2017	92
		Canada (CANDU®-reactors)	Life-time (average)	87
		Russia	2014	81
		United Kingdom	2015	75
		World	2017	81
2	Geothermal	United States	2017	76
3	Bioenergy	United States	2017	51–71
4	Combined cycle	United States	2017	55
5	Coal fired	United States	2017	54
6	Hydroelectric	United States	2017	45
		World (average)	2011–13	~45
		United States	2017	37
7	Wind	World	2011–13	20–40
		United States	2017	22
		Spain (molten salt with storage)	2014	63
9	PhotoVoltaic (PV) solar	United States	2017	27
		United Kingdom	2015	12
10	Wave	United Kingdom	2015	3

the internal usage of a power plant, which might be not so small (5% or even more). Also, the thermal efficiency is the driving force for advances in thermal power plants and, nowadays, in NPPs.

Analysis of the data listed in Table 2 shows that the top power plants by the thermal efficiency are: (1) Combined-cycle power plants (combination of Brayton gas-turbine cycle and subcritical-pressure Rankine steam-turbine cycle) (for details, see Appendix A1 in Pioro, 2016) with the highest thermal efficiency in the power industry of up to 62%, which means that roughly 2/3 of thermal energy introduced inside the combined power cycle is used for electricity generation, i.e., useful energy (in general, “useful” energy can include electricity; process heat; hot water/steam for heating of buildings, hot-water supply, etc.; and desalination) and only 1/3 of the total energy is lost into environment; and (2) Supercritical-pressure Rankine steam-turbine cycle of coal-fired power plants with up to 55% thermal efficiency (for details, see Appendix A1 in Pioro (2016) and Pioro and Duffey (2007)). Unfortunately, NPPs with subcritical-pressure Rankine steam-turbine cycle have significantly lower thermal efficiencies up to 1.5–2.4 times compared to those of the top thermal power plants.

It is reasonable to compare NPPs with various types of nuclear-power reactors based on thermal efficiencies (for details, see [Table 2](#) and [Figs. 3C, 6C and 8C](#)), where AGR NPPs have the highest thermal efficiency in the nuclear-power industry of up to 42% ([Figs. 2 and 3C](#)), and SFR NPPs—up to 40% ([Fig. 3](#)). Such relatively high thermal efficiencies are due to higher temperatures inside reactors (in AGRs the outlet temperature of the carbon-dioxide reactor coolant is up to 650 °C, and in SFRs with the liquid-sodium reactor coolant—up to 550 °C), adopting subcritical-pressure Rankine cycles with primary and secondary superheated steam and relatively high primary-steam pressures—17 MPa and 14.2 MPa, respectively.

However, the last AGR (only 14 left in United Kingdom) was connected to a grid in 1989, i.e., with no new AGRs have been built for last 32 years. Possible reasons for stopping AGRs in the United Kingdom were the cost, build difficulties (problems with intermediate heat exchangers and thermal insulation), and issues with attaining on-line refueling at full-power plus lower costs alternatives. Therefore, unfortunately, these reactors/NPPs will be never built again.

In terms of SFRs, only two are currently in operation in Russia, and the relatively low boiling point of sodium (882.8 °C) means that these unique fast reactors/NPPs will not possibly have higher thermal efficiencies in the future. Currently, China is finalizing construction of the first China Fast Reactor (CFR-600) and has started construction of another one (<https://www.neimagazine.com/news/newschina-begins-construction-of-second-cfr-600-fast-reactor-8435608>).

The vast majority of NPPs are equipped with light- and heavy-water-cooled reactors (377 and 49, respectively), which is currently the largest group of nuclear-power reactors in the world, i.e., 426 from 442 or ~96%. 424 of these reactors use a subcritical-pressure Rankine cycle with saturated primary steam (at pressures up to 7.8 MPa) and slightly overheated secondary steam (see [Fig. 6C](#)), so these NPPs have significantly lower thermal efficiencies (32–38%) compared to those of advanced modern thermal power plants (see [Table 2](#)).

The only exceptions are power cycles of Russian SMRs: KLT-40S (two of which are currently in operation) and RITM-200M ([Pioro et al., 2019, 2020](#)). These SMRs are equipped with subcritical-pressure Rankine cycles with overheated primary steam, but without a reheat option (see [Table 2](#) and [Fig. 10](#)). Due to that, thermal efficiencies of these SMR NPPs are the lowest ones (i.e., 26% and 31%, respectively) compared to those of other light- and heavy-water-cooled reactors NPPs.

Analysis of *T-s* diagrams and the corresponding NPPs layouts (see Appendix A1 in [Pioro \(2016\)](#)) shows that NPPs with water-cooled reactors have possibly reached their maximum achievable thermal efficiencies (close to 36–38%), because further increase in reactor-coolant pressures (i.e., beyond 15–16 MPa) will lead to a small rise in the coolant saturation temperature (see [Fig. 1](#) in [Dragunov et al. \(2015\)](#)). Moreover, it will lead to a significant decrease in Critical Heat Flux (CHF) at flow boiling (see [Fig. 2](#) in [Dragunov et al. \(2015\)](#)).

Thermal efficiency of an NPP and any thermal power plant is a very important general parameter for comparison of various power plants. Higher thermal efficiency is better, because nuclear or thermal power plant will have less thermal/various emission pollutions per kg of fuel, less fuel per kW h of electricity generated, less cost of kW h, and less cooling water required. Currently, cooling shortfall is a serious concern due to the climate change and more often and more severe dryouts/higher water temperatures in rivers, lakes, and even seas/oceans around the world, etc.

However, it should be stated that in spite of the thermal efficiency being the driving force for advances in thermal and nuclear power plants, quite often other parameters, policies, or conditions influence decisions, which type of reactors/NPPs to build in a country or in a region. As such, Canada develops PHWRs, in particular, CANDU^{®1} reactors, from 1950s, and due to that, up till now, all 19 Canadian nuclear-power reactors are of PHWR type. India, originally a partner with Canada, has quite similar, but now an independent approach, therefore, the vast majority of their reactors are also of PHWR type.

United States and Japan have a number of large nuclear vendors, which, after being in a cooperative license, developed PWRs and BWRs/ABWRs. Therefore, they have only these types of nuclear-power reactors (United States—63 PWRs and 31 BWRs, and Japan—16 PWRs and 17 BWRs). On the other hand, France has only one nuclear vendor, and after terminating an United States cooperation license and stopping gas-reactor builds, now have all their NPPs equipped with only PWRs. Also, it should be mentioned that thermal efficiencies of water-cooled SMRs NPPs are lower than those of modern NPPs equipped with LWRs. However, SMRs will undoubtedly have their unique “niche” applications of being implemented in remote areas, small electrical grids, military facilities, and as floating NPPs ([Pioro et al., 2019, 2020; Pioro, 2016](#)).

Current NPPs generate mainly electricity. Just few of them are used for district heating and hot-water supply and for water desalination. However, there are needs for process heat, hydrogen co-generation, etc., which cannot be achieved at modern NPPs due to relatively low temperatures inside reactors, especially, cooled with light and heavy water. Therefore, to increase thermal efficiencies to the level of those at modern advanced thermal power plants and to cover these new needs, the world has to move to high-temperature reactors, such as next generation or Generation-IV reactors (see [Pioro, 2016](#)).

Thermal efficiencies of next generation or Generation-IV NPPs

It should be mentioned that thermal-power industry is ahead of nuclear-power one in terms of using advanced power cycles ([Pioro, 2016](#)). Due to this next generation or Generation-IV nuclear-power reactors will be connected to combined power cycles, Brayton cycle, or to supercritical-pressure Rankine cycle (for details, see [Table 3](#); Chapter 00301 by A. Stanculescu in this Encyclopedia; GIF

¹CANDU[®]—Registered Trademark of AECL, used under license by Candu Energy Inc., Member of SNC-Lavalin Nuclear Group.

(<https://www.gen-4.org/gif/>); and Pioro (2016)). However, in Brayton cycles combustion products will be replaced with helium, nitrogen, mixture of helium and nitrogen, or supercritical-pressure carbon dioxide (for details, see Pioro (2016) and Mahdi et al. (2018)) (Figs. 11–14).

Capacity factor

Another important general parameter of any power plant is a capacity factor.

Net capacity factor of a plant is the ratio of the actual output of a power plant over a period of time (usually, during a year) to its potential output, if it had operated at its full “nameplate” capacity the entire time. Capacity factors vary significantly depending on the type of a plant (for details, see Table 4).

It should be mentioned that, in general, capacity factors can be induced by Mother Nature, i.e., mainly, all renewable-sources power plants, or artificially induced, e.g., for thermal power plants to compensate variations in consumption of electricity and/or generation of electricity with renewable-sources power plants (for details, see Fig. 14); unreliable operation of plant equipment, long-terms for reactor re-fueling, etc.

In terms of capacity factors, NPPs have proven records of high capacity factors up to 100% for new plants, especially, compared to those of renewable-sources power plants (see Table 4). In general, there are two groups of nuclear-power reactors with: (1) Batch refueling (PWRs, BWRs, and SFRs) and (2) Refueling on-line (PHWRs, LGRs, and, partially, AGRs). In previous years, refueling on-line allows reactors to run continuously for longer time, and, at that time, it was some advantage of such reactors/NPPs compared to batch-refueling ones. However, nowadays, batch-refueling reactors started to use slightly higher fuel enrichments of up to 5% and higher fuel burnups, which allowed to prolong their continuous operation, and effective NPPs outage management decreased a number of days required for refueling. Therefore, now the advantages of the on-line refueling are not so obvious compared to that of the batch refueling.

How various energy sources generate electricity in a grid can be illustrated based on the Province of Ontario (Canada) system. It should be noted that all, i.e., 19, nuclear-power reactors in Canada are of CANDU[®]-reactor type, and 18 of these reactors are located in the Province of Ontario (for details, see Pioro et al. (2019) and Pioro (2016)). Currently, the Province of Ontario has completely eliminated coal-fired power plants from its electrical grid. Some of them were closed, others—converted to natural gas. Ontario major installed capacities in 2015 were nuclear (38%), natural gas (29%), hydro (25%), and renewables (mainly wind) (8%). However, electricity was mainly generated by nuclear (60%), hydro (24%), natural gas (8.7%), and renewables (mainly wind) (6.9%).

Fig. 14A shows power generated, and Fig. 14B—capacity factors of various energy sources in Ontario electrical grid in summer (June 17, 2015). Analysis of the data in Fig. 14 shows that a full mix of nuclear, hydro, gas, wind, biofuel, and solar were the major sources for electricity generation (for operation of the Ontario grid with coal-fired power plants, see Pioro (2016) and Pioro and Duffey (2015)).

Electricity that day from midnight till 3.00 a.m. was mainly generated with nuclear, hydro, gas, wind, and biofuel (see Fig. 14A). After 3.00 a.m., biofuel power plants have increased slightly electricity generation followed by hydro and gas-fired power plants due to increased consumption of electricity in the province. Also, at the same time, wind power plants have also slightly increased electricity generation. However, after 7.00 a.m. wind power started to fluctuate and, eventually, decreased significantly. After 6.00 a.m., solar-power plants started to generate electricity.

During a day, hydro, gas-fired and biofuel power plants had variable electricity generation to compensate changes in consumption of electrical energy and variations in generating electricity from wind and solar power plants. After 9 p.m. in the evening, energy consumption started to drop in the province, and at the same time, wind power increased. Therefore, gas-fired, hydro, and biofuel power plants have decreased their energy generation accordingly due to the preferred feed-in-tariffs allocated to wind.

It should be noted that NPPs operated at about 100% of installed capacity providing reliable basic power to the grid. This example shows clearly that any grid that includes NPPs and/or renewable-energy sources must also include “fast-response” power plants such as gas-fired, coal-fired and/or large hydro-power plants to compensate changes in consumption of electrical energy per day and variations in electricity supply by wind and/or solar power plants.

Also, it should be mentioned that in the current chapter only two basic general parameters of power plants were considered. However, there are other important parameters of power plants, which include capital and operational costs, eventually, generated electricity costs, fuel costs, environmental impact, operational term, area covered by a plant, etc.

Conclusions

One of the major parameters of any power plant is efficiency or thermal efficiency. The thermal efficiency is the major driving force for all advances in thermal and nuclear power plants. Ranges of gross thermal efficiencies of modern power plants are as the following: (1) Combined-cycle thermal power plants—up to 62%; (2) Supercritical-pressure coal-fired thermal power plants—up to 55%; (3) Subcritical-pressure coal-fired thermal power plants—up to 43%; (4) Carbon-dioxide-cooled reactor NPPs—up to 42%; (5) Sodium-cooled Fast Reactor (SFR) NPPs—up to 40%; (6) Modern water-cooled-reactor NPPs—30–36% (up to 38%); and (7) Current PWR-SMRs NPPs—26–31%.

However, all current Generation-II and III and oncoming Generation-III⁺ NPPs, especially, those equipped with light- and heavy-water-cooled reactors, are not competitive with modern thermal power plants in terms of thermal efficiency (30–36% (up to 38%) for current NPPs with light- and heavy-water-cooled reactors and up to 55%/62% for supercritical-pressure coal-fired and combined-cycle power plants, respectively).

Comparison of various types of reactors/NPPs showed that there is no reactor design, which will be the best for all the possible parameters. In general, thermal efficiency of an NPP can be one of the important factors to be considered. In this case, higher thermal efficiency is better, because NPP will have less thermal/chemical emissions/pollutions per kg of fuel, less fuel per kW h of electricity generated, less cost of kW h, less cooling water required (which, currently, is a serious concern due to the climate change and more often and more severe dryouts/higher water temperatures in rivers, lakes, and even seas/oceans around the world), etc. Also, higher thermal efficiencies of NPPs make nuclear power more competitive with thermal and other power plants.

Current NPPs are mainly generate electricity. Just few on them are used for district heating and hot-water supply, and for water desalination. However, there are needs for process heat, hydrogen co-generation, etc., which cannot be achieved at modern NPPs due to relatively low temperatures inside reactors, especially, cooled with light and heavy water. Therefore, to increase thermal efficiencies to the level of those at modern advanced thermal power plants and to cover these new needs, the world has to move to high-temperature reactors, such as next generation or Generation-IV reactors.

Another important parameter of any power plant is a capacity factor. It should be mentioned that, in general, capacity factors can be induced by Mother Nature, i.e., mainly, all renewable-sources power plants, or artificially induced, e.g., for thermal power plants to compensate variations in consumption of electricity and/or generation of electricity with renewable-sources power plants (for details, see Fig. 14); unreliable operation of plant equipment, long-terms for reactor re-fueling, etc.

Also, it should be mentioned that in the current chapter only two basic general parameters of power plants were considered, i.e., thermal efficiency and capacity factor. However, there are other important parameters of power plants, which include capital and operational costs, eventually, generated electricity costs, fuel costs, environmental impact, operational term, area covered by a plant, etc.

The major advantages of nuclear power are well-known, including cheap reliable base-load power, high capacity factor, low carbon-dioxide emissions, and minor environmental impact. However, these factors are offset today by a competitive economic disadvantage with cheap natural gas combined-cycle power plants. Hopefully, Generation-IV reactors/NPPs under development will close or reduce the efficiency gap.

See Also: Commercial Nuclear Power Plant Statistics.

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Pros and Cons of Commercial Reactor Designs: Part 2—Current Status and Future Trends in the World Nuclear-Power Industry and Technical Considerations of Nuclear-Power Reactors

I. Pioro^a, R.B. Duffey^{b,*}, P.L. Kirillov^c, and R. Pioro^d, ^a University of Ontario Institute of Technology, Oshawa, ON, Canada; ^b Idaho Falls, ID, United States; ^c State Scientific Centre of the Russian Federation - A. I. Leypunsky Institute of Physics and Power Engineering (IPPE), Obninsk, Russia; and ^d Faculty of Physics, Lomonosov Moscow State University, Moscow, Russia

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Nomenclature

P Pressure, MPa
T Temperature, °C

Subscripts

el Electrical
in Inlet
max Maximum
out Outlet
sat Saturated or saturation

Acronyms/Abbreviations

ABWR Advanced Boiling Water Reactor
ACPR Advanced Chinese Pressurized-water Reactor
AECL Atomic Energy of Canada Limited
AEP AtomEnergoproekt (Russia)
AGR Advanced Gas-cooled Reactor
AHWR Advanced Heavy Water Reactor
AP Advanced Plant (USA)
APR Advanced Pressurized-water Reactor (S. Korea)
ASE AtomSroyExport (Russia)
BN Fast Sodium (reactor) (Быстрый Натриевый (in Russian abbreviations)) (Russia)
BWR Boiling Water Reactor
CANDU[®] CANada Deuterium Uranium (reactor) (Registered Trademark of AECL, used under license by Candu Energy Inc., Member of SNC-Lavalin Nuclear Group)
CGNPC China General Nuclear Power Group
CNNC China National Nuclear Corporation
EGP Power Heterogeneous Loop (reactor) (ЛнерГетический ГетероГенный Цетлевой (реактор с 6-ю Цетлями циркуляции теПлоносителя) (in Russian abbreviations))
el. element (fuel rod)

*Retired.

EPR European Pressurized-water Reactor (original acronym, later changed to Evolutionary Power Reactor) (France)
 GCR Gas-Cooled Reactor
 GE General Electric (USA)
 Hor. Horizontal
 HPR Hualong one Pressurised-water Reactor (China)
 HTC Heat Transfer Coefficient
 ID Inside Diameter
 IEA International Energy Agency
 Inc. Incorporated
 Jan. January
 KEPCO Korea Electric Power COporation
 KLT Container-carrier cargo-Lighter Transport (reactor) (Контейнеровоз Личтеровоз ТрансПортный (реактор) (in Russian abbreviations)) (Russia)
 LGR Light-water-cooled Graphite-moderated Reactor
 LMFBFR Liquid-Metal Fast-Breeder Reactor
 LWR Light Water Reactor
 Mar. March
 MOX Mixed OXide (fuel)
 MTM Ministry of Heavy Machine Building (in Russian abbreviations) (Russia)
 N/A Not/Applicable
 NEA Nuclear Energy Agency
 NNC National Nuclear Corporation (UK)
 NPP Nuclear Power Plant
 OECD Organisation for Economic Co-operation and Development
 OKBM Experimental Design Bureau of Mechanical-engineering (ОПытно-Конструкторское Бюро Машиностроения (in Russian abbreviations))
 PCh Pressure Channel (reactor)
 PHWR Pressurized Heavy-Water Reactor
 PV Pressure Vessel
 PWR Pressurized Water Reactor
 RBMK Reactor of Large Capacity Channel type (Реактор Большой Мощности Канальный (in Russian abbreviations) (Russia)
 R&D Research and Development
 Rep. Republic
 RITM-200M Reactor Integral Type Modular 200 MW_{el} Modernized (Реактор ИнтеГральноГо ТиПа Модульный мощностью 200 МВт Модернизационный (in Russian abbreviations) (Russia)
 RPV Reactor Pressure Vessel
 S. South
 SFR Sodium Fast Reactor
 SMR Small Modular Reactor
 Th. Thermal
 UAE United Arab Emirates
 UK United Kingdom
 USA United States of America
 USSR Union of Soviet Socialist Republics
 V Vessel
 Vert. Vertical
 VVER Water-Water Power Reactor (Водо-Водяной ЛнерГетический Реактор (in Russian abbreviations) (Russia)

Current status and future trends in the world nuclear-power industry

Nuclear power without used-fuel reprocessing and recycling is often considered to be a non-renewable energy source like the mined fossil fuels, such as coal, oil, and natural gas. Moreover, the resources of naturally occurring fissile fuels (uranium and thorium) are

limited in extent, and geographically concentrated in regions of relatively lower population. However, nuclear resources can be used for significantly longer time than some fossil fuels, and in some cases almost indefinitely, if recycling of unused or low burn-up uranium fuel, thorium-fuel resources, and fast-neutron-spectrum reactors that literally can “breed” plutonium are used.

Current statistics on all world nuclear-power reactors connected to electrical grids are listed in **Tables 1–6** and shown in **Figs. 1–4**. Statistical data from previous years are shown in [Pioro et al. \(2019, 2020\)](#), [Pioro and Duffey \(2015\)](#), and [Handbook of Generation IV Nuclear Reactors \(2016\)](#). **Table 6** lists current activities worldwide on new nuclear-power-reactors build, and **Tables 7a and 7b** - basic parameters of various nuclear-power reactors and thermal efficiencies of the corresponding Nuclear Power Plants (NPPs), respectively.

The largest group of nuclear-power reactors by type is PWRs (302 from 442 reactors or 68% of the total number), and quite a significant number of Pressurized Water Reactors (PWRs) are planned to be built (about 39 (+35?)) (for details, see **Table 1**). **Fig. 1** shows the data from **Table 1** in the form of sector diagrams for a number of reactors by type and by installed capacities. The second largest group of reactors is Boiling Water Reactors (BWRs)/Advanced BWRs (ABWRs) (63 reactors or 14% of the total number). The third group is Pressurized Heavy-Water Reactors (PHWRs) (49 reactors or 11% of the total number). Considering the number of forthcoming reactors, the number of BWRs/ABWRs and PHWRs will decrease globally within next 20–25 years. Furthermore, within next 10–15 (maximum 20) years or so, all Light-water-cooled Graphite-moderated Reactors (LGRs) and Advanced Gas-cooled Reactors (AGRs) (carbon-dioxide-cooled) will be shut down forever.

Table 2 lists average, maximum, and minimum installed capacities of nuclear-power reactors of the world of various types (values were rounded to nearest 0 or 5). Analysis of the data in **Table 2** shows that the largest reactor by installed capacity is 1660-MW_{el} PWR (EPR, Generation-III⁺, Areva design, France) (two EPRs are in operation in China and more are under construction in several countries), and the smallest one is a 10-MW_{el} LGR (EGP, former USSR design) (three EGPs are in operation in Russia, but all of them will be shut down in several years).

Table 3 lists a number of nuclear-power reactors connected to grid by selected nations (13 nations with the largest number of nuclear-power reactors/their installed capacities ranked by installed capacities). Analysis of the data in **Table 3** shows that real

Table 1 Number of nuclear-power reactors connected to electrical grid and forthcoming units as per March 2021 (based on data from [Nuclear News \(2021\)](#); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>) and prior to Japan earthquake and tsunami disaster (based on data from [Nuclear News \(2011\)](#)).

Rank by No	Reactor type (% of total reactors/average installed capacity)	No. of units		Installed capacity, GW _{el}		Forthcoming units	
		As of Mar. 2021	Before Mar. 2011	As of Mar. 2021	Before Mar. 2011	No. of units	GW _{el}
1	Pressurized Water Reactors (PWRs) (68%/955 MW _{el})	302 ↑	268	288 ↑	248	39+35?	45+37? ^a
2	Boiling Water Reactors (BWRs) or Advanced BWRs (14%/1030 MW _{el})	63 ↓	92	65 ↓	84	4?	5.3?
3	Pressurized Heavy Water Reactors (PHWRs) (11%/500 MW _{el})	49 ↓	50	24 ↓	25	2 + 5?	1.3 + 3.3?
4	Advanced Gas-cooled Reactors (AGRs) (CO ₂ -cooled) (3%/550 MW _{el})	14 ↓	18	8 ↓	9	1 ^b	0.2 ^b
5	Light-water-cooled, Graphite-moderated Reactors (LGRs) (3%/700 MW _{el})	12 ↓	15	8 ↓	10	0	0
6	Liquid-Metal Fast-Breeder Reactors (LMFBRs – SFRs) (0.5%/690 MW _{el})	2 ↑	1	1.4 ↑	0.6	2+1?	1.1 + 0.6?
In total		442 ↓	444	395 ↑	377	44 + 45?	48 + 46?

Technical parameters of various reactors types are listed in **Tables 7a and 7b**, and in details are shown in [Handbook of Generation IV Nuclear Reactors \(2016\)](#).

Arrows mean decrease or increase in a number of reactors and installed capacities.

Data in **Table 1** include 33 reactors in Japan, 24 of which were not in service as of October 2020.

Explanations to **Table 1**.

^a? – Means “Commercial start date – indefinite” ([Nuclear News, 2021](#)).

^bForthcoming reactor is a helium-cooled reactor – High Temperature Reactor Pebble-bed Modular (HTR-PM) (China).

Table 2 Average, maximum and minimum installed capacities of nuclear-power reactors of the world of various types (values rounded to nearest 0 or 5) (based on data from [Nuclear News \(2021\)](#); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>).

Type of reactor	PWRs	BWRs	PHWRs	AGRs (CO ₂ -cooled)	LGRs	LMFBRs (SFRs)
Parameter	Installed capacities, MW _{el}					
Average	955	1030	500	550	700	690
Maximum	1660	1435	880	620	925	820
Minimum	30	150	90	480	10	560

Table 3 Number of nuclear-power reactors connected to grid by nation (13 nations ranked by nuclear-reactor installed capacities) as per March 2021 (based on data from Nuclear News (2021); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>) and before the Japan earthquake and tsunami disaster (based on data from Nuclear News (2011)).

No	Nation	No. of units (PWRs/BWRs/Other types)		Installed capacity, GW _{el}		Changes in No. of reactors from March 2011	% of electricity generated by nuclear
		As of March 2021	Before March 2011	As of March 2021	Before March 2011		
1	USA	94 (63/31)	104	98	103	↓ 10	19.7
2	France	56 (56/—)	58	61	63	↓ 2	72.0
3	China	49 (47/—/2 ^a)	13	47	10	↑ 36	4.8
4	Russia	38 (24/—/12 ^b /2 ^c)	32	29	23	↑ 6	19.0
5	Japan ^d	33 (16/17)	54	32	47	↓ 21	4.7
6	S. Korea	24 (21/—/3 ^a)	20	23	18	↑ 4	23.4
7	Canada	19 (—/—/19 ^a)	22	14	15	↓ 3	16.8
8	Ukraine	15 (15/—)	15	13	13	N/A	53.5
9	UK	15 (1/—/14 ^e)	19	9	10	↓ 4	25.0
10	Germany	6 (5/1)	17	8	20	↓ 11	11.6
11	Spain	7 (6/1)	8	7	8	↓ 1	20.4
12	India	23 (2/2/19 ^a)	19	7	4	↑ 4	2.7
13	Sweden	6 (2/4)	10	7	9	↓ 4	36.0
In total		385 (258/56/12^b/2^c/43^a/14^e)	391	355	343	↓ 6, but installed capacity ↑ by 12 GW_{el}	

Data for all countries with nuclear-power reactors are listed in Table 4.

Explanations to Table 3:

Arrows mean decrease or increase in a number of reactors.

^aPWRs.

^bNo of LGRs.

^cLMFBRs.

^dAs of October 2020, only nine PWRs were in operation.

^eAGRs.

nuclear “renaissance” is in China (36 reactors built and put into operation within past 9 years), in Russia (addition of 6 reactors), in South Korea (addition of 4 reactors), and in India (addition of 4 reactors). Meanwhile, the most significant drop in a number of reactors is in Japan (21 reactors were shut down) (only about 9 reactors out of 33 are currently in operation), in Germany (11 reactors), in USA (9 reactors), in UK (4 reactors), in Canada (3 reactors), and in Sweden (3 reactors). In addition, Canada, Germany, and Sweden have no plans to build new reactors. It should be noted that, in spite of, outstanding achievements in nuclear-power industry, especially, in China, and, partially, in India, electricity share by nuclear power is very small, i.e., in China - only 4.8% and in India - 2.7%.

Table 4 lists a number of nuclear-power reactors connected to grid or to be connected to grid in a reasonable time by all countries in the world. Analysis of the data in Table 4 shows that, currently, 33 countries in the world have operating nuclear-power reactors (within these countries: 20 plan to build new reactors, and 13 don't plan to build new reactors) and 3 countries without nuclear-power reactors (Bangladesh, Egypt, and Turkey) are working towards introducing nuclear energy on their soils.

Also, in addition to data in Tables 1–4, Fig. 2 shows a number of nuclear-power reactors in the world connected to grid (as per July, 2020) vs. years of their commercial operation (Fig. 2A) and their installed capacities (Fig. 2B). And Fig. 3 shows a number of nuclear-power reactors in the USA connected to grid (as per July, 2020) vs. years of their commercial operation.

Analysis of the data in Fig. 2 shows that the Chernobyl NPP accident (April 26, 1986), which had happened with the RBMK-1000 – LGR (former USSR design), has forced Ukraine to shut down this NPP, and Russia – to cancel any further R&D and construction of new LGRs. In the same way, a small number of BWRs/ABWRs planned to be built is due to the Fukushima-Daiichi NPP severe accident, which had happened with older design BWRs in March of 2011. However, it should be mentioned that all nuclear vendors, of course, including BWRs and ABWRs, have updated their designs with additional features/systems to enhance safety based on lessons learned from all nuclear accidents.

In the case of the USA (see Fig. 3), their nuclear industry was hit quite hard by the Three Mile Island NPP accident and right after that by the Chernobyl NPP accident, which, eventually, decreased the number of reactors built and connected to grid to only six per last 25 years.

Therefore, the history of nuclear-power industry shows that nuclear accidents, especially, severe ones, can override any advantages of various reactors' types, and significantly affect future builds of certain types of reactors, e.g., LGRs and BWRs, or even all types of reactors. We have to remember that one more severe accident somewhere in the world can and does adversely affect the whole nuclear-power industry and broader societal acceptance. Due to this, we must continuously to enhance safety of all nuclear-power reactors.

Fig. 4 shows a number of nuclear-power reactors in the world by installed capacity as per August, 2020. Analysis of the data in this Figure shows that the largest group of reactors is within the range of installed capacities from 900 MW_{el} and up to 999 MW_{el}.

Table 4 Number of nuclear-power reactors connected to electrical grid and forthcoming units as per March 2021 (based on data from [Nuclear News](https://www.nuclear-news.org/) (2021); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>).

No	Nation	# Units (type)	Net MW _{el}	# Units	Net MW _{el}	Type
		(connected to grid)			(forthcoming)	
1	Argentina	3 (PHWRs)	1641	1?^a	25?	PWR
2	Armenia	1 (PWR)	375	0	0	—
3	Bangladesh	—	—	2	2160	PWR
4	Belarus	1 (PWR)	1110	1	1110	PWR
5	Belgium	7 (PWRs)	5942	0	0	—
6	Brazil	2 (PWRs)	1889	1?	1245?	PWR
7	Bulgaria	2 (PWRs)	2006	0	0	—
8	Canada	19 (PHWRs)	13,554	0	0	—
9	China	49 (47 PWRs; 2 PHWRs)	47,458	9 + 15?	9522 + 16,918?	PWR
				1	200	GCR^b
				1 + 1?	600 + 600?	LMFBR
10	Czech Rep.	6 (PWRs)	3932	0	0	—
11	Egypt	—	—	1 + 3?	1194 + 3582?	PWR
12	Finland	4 (2 PWRs; 2 BWRs)	2794	2	2800	PWR
13	France	56 (PWRs)	61,370	1?	1600?	PWR
14	Germany	6 (5 PWRs; 1 BWR)	8113	0	0	—
15	Hungary	4 (PWRs)	1902	2?	2400?	PWR
16	India	23 (19 PHWRs; 2 BWRs; 2 PWRs)	6885	2 + 3?	1260 + 1890?	PHWR
				2 + 2?	1834 + 1834?	PWR
				1	470	LFMBR
17	Iran	1 (PWR)	915	2	1889	PWR
18	Japan^c	33 (16 PWRs; 13 BWRs; 4 ABWRs)	31,679	2?	2653?	ABWR
19	Mexico	2 (BWRs)	1552	0	0	—
20	Netherlands	1 (PWR)	482	0	0	—
21	Pakistan	5 (4 PWRs; 1 PHWR)	1318	3?	3028?	PWR
22	Romania	2 (PHWRs)	1300	2?	1440?	PHWR
23	Russia	38 (24 PWRs; 12 LGRs; 2 LMFBRs)	28,578	5 + 3?	5515 + 3333?	PWR
24	Slovakia	4 (PWRs)	1848	2?	880?	PWR
25	Slovenia	1 (PWR)	696	0	0	—
26	S. Africa	2 (PWRs)	1860	0	0	—
27	S. Korea	24 (21 PWRs; 3 PHWRs)	23,150	4	5360	PWR
28	Spain	7 (6 PWRs; 1 BWR)	7121	0	0	—
29	Sweden	6 (2 PWRs; 4 BWRs)	6873	0	0	—
30	Switzerland	4 (3 PWRs; 1 BWR)	2960	0	0	—
31	Taiwan	4 (2 PWRs; 2 BWRs)	3844	2?	2600?	ABWR
32	Turkey	—	—	4	4456	PWR
33	Ukraine	15 (PWRs)	13,107	2?	2070?	PWR
34	UAE	1 (PWR)	1345	3	4035	PWR
35	UK	15 (1 PWR; 14 AGRs)	8923	2	3260	PWR
36	USA	94 (63 PWRs; 31 BWRs)	98,411	2	2200	PWR
In total		442 (302 PWRs; 63 BWRs; 49 PHWRs; 14 AGRs; 12 LGRs; 2 LMFBRs)	394,939	44 + 45?	47,865 + 46,193?	—

Countries planning to build new reactors are in bold.

Summary: 33 countries have operating nuclear-power reactors; 20 countries from these 33 and 3 countries without reactors plan to build new nuclear-power reactors (in bold color). In addition, 30 countries are considering, planning, or starting nuclear-power programs, and about 20 countries have expressed their interest in nuclear power.

Explanations to Table 4.

^a? – Means “Commercial start date – indefinite” ([Nuclear News](https://www.nuclear-news.org/), 2021).

^bGCR is a helium-cooled reactor – High Temperature Reactor Pebble-bed Modular (HTR-PM) (China).

^cAs of October 2020, only nine reactors were in operation.

However, oncoming reactors of Generation-III⁺, usually, have higher installed capacities compared to those of Generation-III reactors, i.e., within the range of 1100 MW_{el} and up to 1660 MW_{el} (the exception is Small Modular Reactors (SMRs)).

It should be mentioned that globally current successes in nuclear-power industry are far away from previous “glorious” days, when ~80 and ~120 reactors have been built and connected to grid within 1980–1984 and 1985–1989, respectively (see Fig. 2A). Just within last 5 years, i.e., from 2015–2019, about 40,000 MW_{el} of new installed capacities have been added, which match the same amount added within 1975–1979 (see Fig. 2B).

Table 5 Latest years when various types of reactors have been built and connected to grid (based on data from [Nuclear News \(2021\)](#)).

No.	Type of reactor	Model	Reactor supplier	Country	Installed capacity, MW _{el}	Year	Reactor age ^a , years
1	PWR	VVER-1200	ASE	Belorus'	1109	2020	1
		HPR	CNNC	China	1000	2020	1
		ACPR-1000			1000	2020	1
		EPR	Areva		1660	2019	2
		VVER-1200	AEP	Russia	1085	2021	0.5
		KLT-40S (2 SMRs)	OKBM		32	2019	2
		APR-1400	KEPCO	UAE	1345	2020	1
2	BWR	ABWR	Hitachi	Japan	1108	2006	15
3	PHWR	PHWR-700	Owner	India	630	2021	0.5
4	AGR	AGR	NNC	UK	605	1989	32
5	LGR	RBMK-1000	MTM	Russia	925	1990	31
6	LMFBR	BN-800	OKBM	Russia	820	2016	5

Reactors built before 1995, i.e., 25 and more years old are in red and *Italic*.

^aReactors age shown for year 2021.

Table 6 Current activities worldwide on new nuclear-power-reactors build (based on data from [Nuclear News \(2021\)](#)).

No	Country/Nuclear supplier	Countries, which are looking forward for new builds (No. of possible units)
1	Russia/Rosatom (outside Russia - ASE (AtomStroyExport) (Nuclear-power activities are supported by Russian government)	Russia (5 + 3?), Bangladesh (2), Belarus (1), China (2 + 2?), Egypt (1 + 3?), Finland (1), Hungary (2?), India (2 + 2?), Iran (2), and Turkey (4): In total: 22 + 10? = 32
2	China/Various vendors (Nuclear-power activities are supported by the Chinese government)	China (9 + 14? ^a), Pakistan (3?), Romania ^b (2? CANDU [®] reactors): In total: 9 + 19? = 28
3	S. Korea/Doosan and KEPCO	S. Korea (4) and UAE (3): In total: 7
4	India/Various vendors	India (4+3?): In total: 7
5	France/Framatome	Finland (1), France (1?), and UK (2): In total: 3 + 1? = 4
6	USA/GE and Westinghouse	USA (2) and Taiwan (2?): In total: 2 + 2? = 4
7	Czech Rep./Skoda	Slovakia (2), Ukraine (2?): In total: 2 + 2? = 4
8	Japan/Toshiba + Hitachi	Japan (2?): In total: 2?
9	Canada/AECL (Candu Energy, Inc.) together with CGNPC (China)	Romania ^b (2? CANDU [®] reactors): In total: 2?
10	Germany/KWU (KraftWerk Union AG)	Brazil (1?): In total: 1?
11	Argentina/CNEA (Comisión Nacional de Energía Atómica)	Argentina (1?): In total: 1?

^a? - Means "Commercial start date – indefinite" ([Nuclear News, 2021](#)).

^bTwo CANDU[®] reactors in Romania for the Cernavoda NPP are a joint venture proposal from China and Canada.

Table 5 lists latest years, when various types of reactors have been built and connected to grid. The latest AGR (carbon-dioxide-cooled) in UK was connected to grid in 1989, i.e., no new AGRs have been built for last 31 years. And, unfortunately, as it was mentioned before, these reactors/NPPs will never be built again.

Table 6 lists current activities in various countries worldwide on new nuclear-power-reactors build. Analysis of the data in **Table 6** clearly shows that Russia and China are the front runners in new nuclear builds in their countries and abroad, largely because both governments provide significant political and long-term support with various funds for nuclear-power R&D and for their government-controlled nuclear vendors, as, also, do South Korea and France, especially, to build NPPs abroad plus credits and other incentives for foreign buyers to introduce nuclear power.

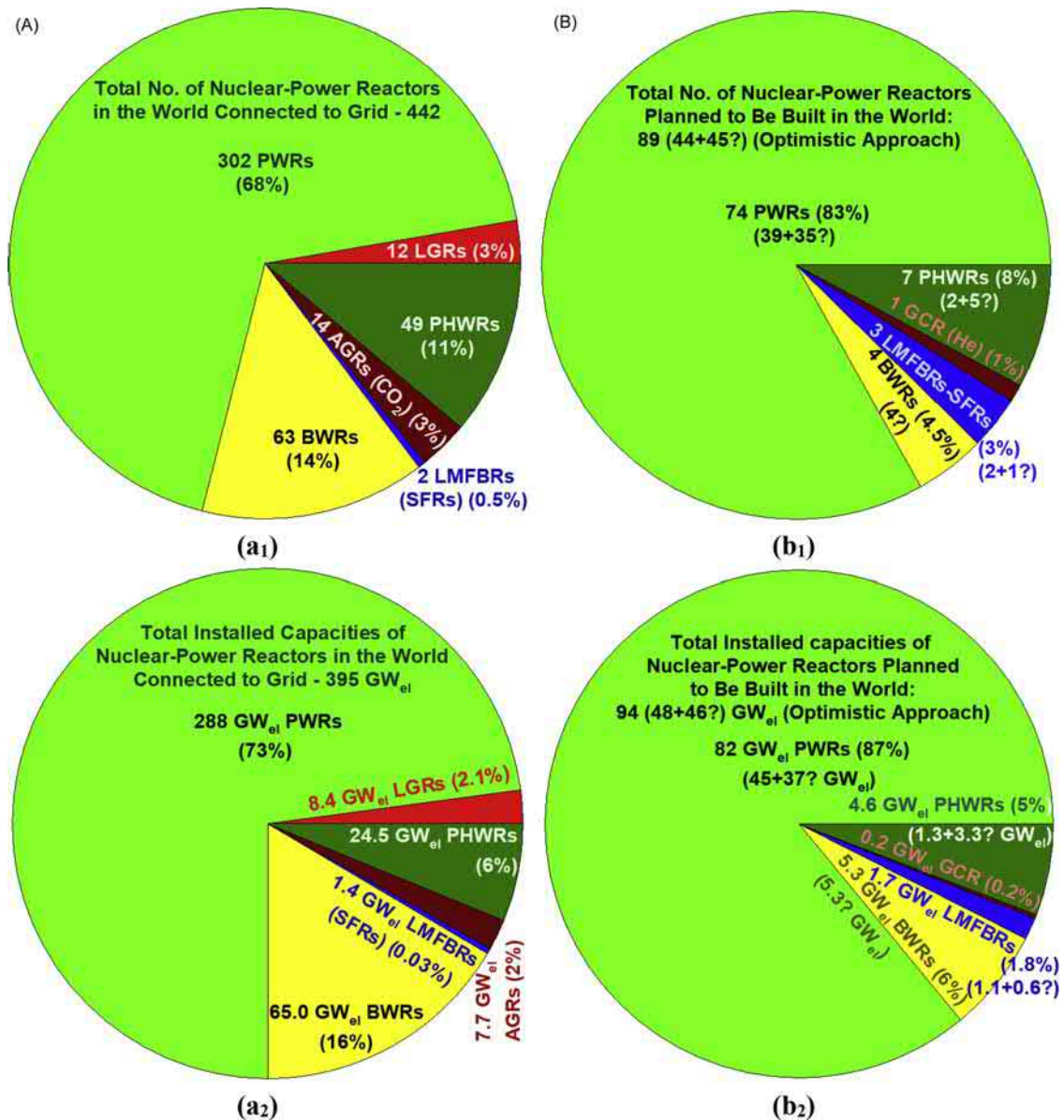


Fig. 1 Nuclear-power reactors of the world (based on data from [Table 1](#) and [Nuclear News \(2021\)](#); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>): (A) connected to grid – (a₁) by type and (a₂) by installed capacities; and (B) planned to be built (optimistic approach, i.e., including those under?) – (b₁) by type and (b₂) by installed capacities.

The last several years were very important for the nuclear-power industry of the world. Russia put into operation a number of Generation-III⁺ VVERs (PWR – VVER-1200) and the SFR – BN-800 reactor in 2016 and continues to lead the SFR technologies in the world ([Pioro et al., 2019](#); [Handbook of Generation IV Nuclear Reactors, 2016](#)).

China put into operation many reactors/NPPs including the largest in the world Generation-III⁺ PWR – EPR (Areva design) with amazing installed capacity of 1660 MW_{el}. In addition, several AP1000 reactors (Westinghouse design) ([Friedman, 2019](#)), also, a Generation-III⁺ design, were put into operation in China first time in the world ([Nuclear News, 2021](#)). In general, Generation-III⁺ water-cooled reactors/NPPs have enhanced safety and can reach slightly higher thermal efficiencies up to 36–37% (38%) compared to those of Generation-III water-cooled reactors/NPPs (the exception is SMRs). Also, China put into operation the Hua-long One reactor or HPR-1000 – a domestically developed Generation-III reactor design, and plan to put into operation first in the world Gas-Cooled Reactor (GCR) – a helium-cooled reactor: High Temperature Reactor Pebble-bed Modular (HTR-PM) in 2021 plus two SFRs – China Fast Reactors (CFR-600) within next several years.

South Korea put into operation several their Generation-III⁺ APR-1400 (Doosan design) on their soil, another APR-1400 (KEPCO) in UAE, and plans to put six more these reactors into operation soon: 3 inside country and 3 in UAE ([Nuclear News, 2021](#)).

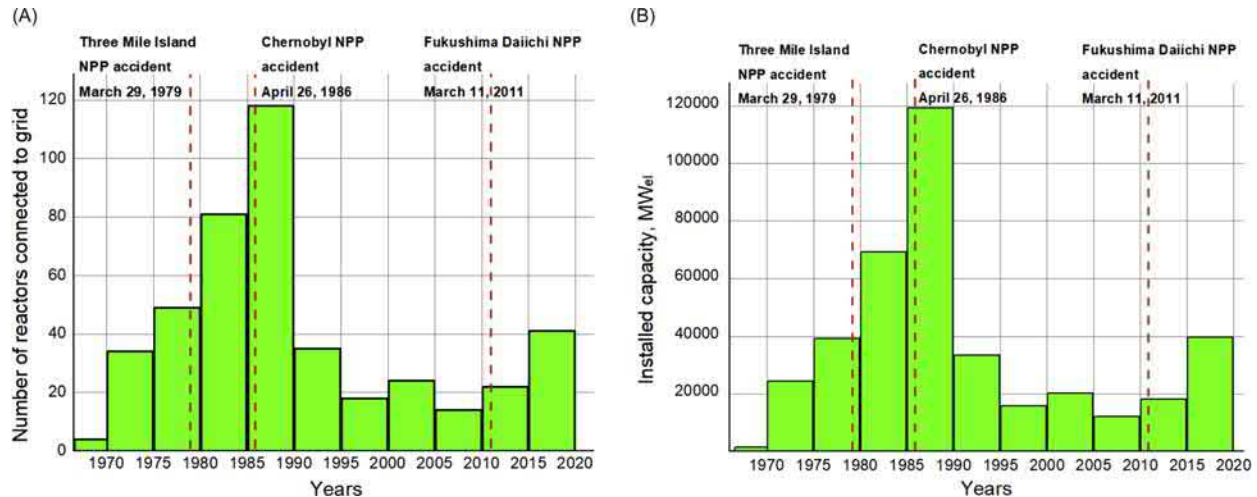


Fig. 2 Number of nuclear-power reactors in the world connected to grid (as per July, 2020) vs. years of their commercial operation (A) and their installed capacities (B) (based on data from Nuclear News (2021); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>) (for previous data, see Pioro et al. (2019) and Handbook of Generation IV Nuclear Reactors (2016)).

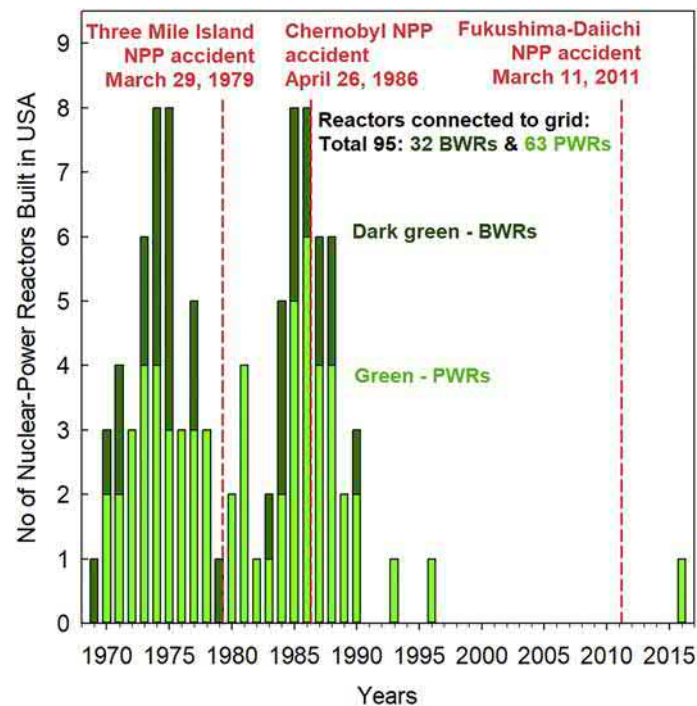


Fig. 3 Number of nuclear-power reactors in the USA connected to grid (as per July, 2020) vs. years of their commercial operation (based on data from Nuclear News (2021); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>).

India put into operation their latest domestically developed design of PHWR – PHWR-700 in January of 2021.

Current trend in new builds is to build Generation-III⁺ reactors, mainly PWRs. Another trend, which just has appeared, is to build SMRs with installed capacities up to 300 MW_{el} for applications in remote areas, small electrical grids, military facilities, and as floating NPPs; and first two SMRs are 35-MW_{el} PWRs installed on a floating barge, so-called, Floating Nuclear Thermal-Power Plant (FNThPP) by the name of Academician Lomonosov (Russia) (for details, see Tables 7a and 7b, and Pioro et al. (2019, 2020)). Also, in 2021, Rosatom will begin work on a site of the future SMR NPP, which is planned to be built in Yakutia. The SMR will be a RITM-200 ship-based reactor (<https://www.neimagazine.com/news/newsrosatom-to-being-work-on-land-based-smr-8436408>).

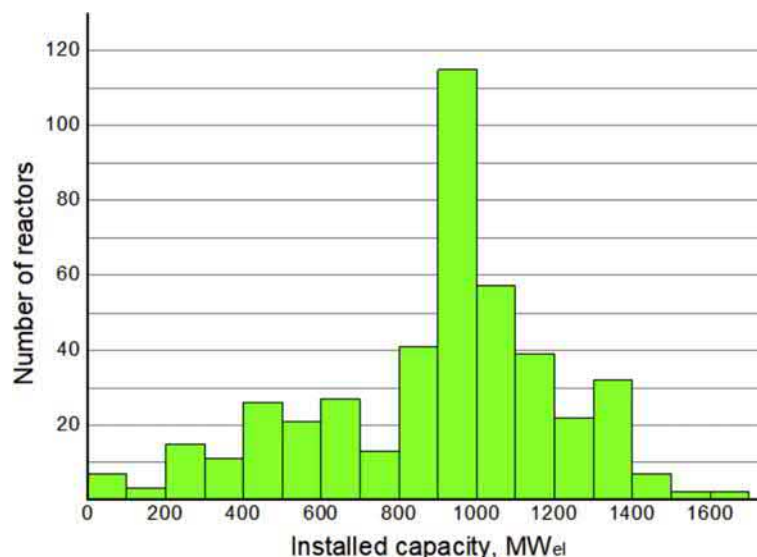


Fig. 4 Number of nuclear-power reactors in the world by installed capacity as per July, 2020 (based on data from [Nuclear News \(2021\)](http://www.world-nuclear.org/); <http://www.world-nuclear.org/>; and <https://pris.iaea.org/pris/>). For better understanding of this Figure, the largest number of reactors have installed capacities within the range of 900–999 MW_{el}. Generation-III⁺ reactors, usually, have installed capacities from 1,100 and up to 1,660 MW_{el} (exception is SMRs).

Technical considerations of various types of nuclear-power reactors

Basic parameters of all current reactors' types are listed in [Table 7a](#), and their power cycles – in [Table 7b](#) (for other details, see also Appendix A1 in [Handbook of Generation IV Nuclear Reactors \(2016\)](#)). It should be stated here that there is no such term as “the best” reactor design by all features or parameters. Each reactor design has some pros and cons.

Analysis of the basic parameters of current reactors (see [Tables 7a and 7b](#)) shows that various types of reactors have the following differences:

By neutron spectrum

- 1) Thermal spectrum - vast majority of reactors, i.e., 440 from 442 or 99.5%, which includes Light Water Reactors (LWRs) (PWRs and BWRs/ABWRs), PHWRs, LGRs; and AGRs; and
- 2) Fast spectrum – LMFBRs, i.e., SFRs - two Russian BN-600 and BN-800 reactors. In general, fast reactors are considered to be the future of nuclear-power industry, because fast-neutron-spectrum reactors can “breed” plutonium and can reprocess radioactive wastes. However, fast reactors require higher enrichment of nuclear fuel compared to that of thermal-spectrum reactors (see [Table 7a](#)), and, for now, light water is not considered as a reactor coolant in such reactors. Due to that, there are some limitations, which reactor coolant can be used in fast reactors (currently, sodium, lead, lead-bismuth, helium, and molten salt are considered). A number of nuclear-power advanced countries have tried to develop fast-reactors technologies, but, as of today, only Russia has two operating SFRs.

By reactor coolant and moderator

- 1) Reactor coolants: (a) the most widely used – light water (LWRs and LGRs); (b) heavy water (PHWRs); (c) carbon dioxide (AGRs); and (d) liquid sodium (SFRs).
- 2) Moderator: (a) light water used in LWRs, and there is no separation between reactor coolant and moderator; (b) heavy water in PHWRs, and here reactor coolant and moderator are separated with fuel channels, i.e., reactor coolant is inside a fuel channel and moderator - outside; (c) graphite in LGRs, and here reactor coolant and moderator are separated with fuel channels; (d) graphite in AGRs and here graphite is in contact with reactor coolant; and (e) there is no moderator in fast reactors, i.e., in SFRs.

Properties of various reactor coolants and comparison of heat transfer coefficients in various reactors are shown in [Handbook of Generation IV Nuclear Reactors \(2016\)](#). Comparison of light- and heavy-water thermophysical properties is shown in [Petriw et al. \(2019\)](#) and [Sabir et al. \(2019\)](#).

Table 7a Basic parameters of all current reactors' types (based on data from [Handbook of Generation IV Nuclear Reactors \(2016\)](#) and <http://www.world-nuclear.org/>).

No	Reactor type	Fuel bundle orientation	Sheath (cladding) material ^a	Neutron spectrum	Reactor		Reactor coolant		Refueling	Fuel ^b	Fuel enrichment (%)	HTC ^c (kW/m ² K)
					Coolant	Moderator	P, MPa	T (°C)				
1	PWR	RPV	Vert.	Zr	Th.	H ₂ O	15–16.2	295 → 330	Batch	UO ₂	3–5	~30
	SMR KLT-40S	RPV	Vert.	Zr	Th.	H ₂ O	12.7	280 → 316	Batch	UO ₂	18.6	–
2	BWR	RPV	Vert.	Zr	Th.	H ₂ O	7.2	287.7	Batch	UO ₂	~2	~60
3	PHWR (CANDU®)	PCh	Hor.	Zr	Th.	D ₂ O	11 → 10	260 → 310	On-line	UO ₂	0.7	~50
4	AGR	PV ^e	Vert.	SS	Th.	CO ₂	~4	290 → 650	Batch ^f	UO ₂	2.5–3.5	~2–5
5	LGR (RBMK)	PCh	Vert.	Zr	Th.	H ₂ O	6.9	284.9	On-line	UO ₂	2–2.4	~60
6	LMFBR (BN-800)	V	Vert.	SS	Fast	Na	~0.1	354 → 547	Batch	MOX	17/20/24	55–85

^aZr – Zirconium alloys; SS – Stainless Steel.

^bCommonly used fuel.

^cHeat Transfer Coefficients (HTCs) are approximate values, shown just for reference purposes.

^dCANDU®-reactor moderator has $P = \sim 0.1$ MPa at the top of calandria vessel and $T = \sim 70$ °C.

^eConcrete RPV.

^fAGRs were designed to be refueled on-line. However, it was found that during refueling at full power fuel assemblies can vibrate, due to that the on-line refueling was suspended from 1988 till the mid-1990s. Nowadays, only refueling at a part load or in shut-down state is now undertaken in AGRs.

Table 7b Basic parameters of all current reactors' types power cycles (based on data from [Handbook of Generation IV Nuclear Reactors \(2016\)](#)).

No	Reactor type	Cycle ^a	No of loops	Rankine-cycle parameters						Thermal efficiencies (gross), %
				Primary steam			Secondary-steam reheat			
				P _{in} , MPa	T _{in} , °C	Steam	P _{in} , MPa	T _{in} , °C	Steam	
1	PWR	Indirect	2	7.72	292.5	Saturated	2	265	Overheated	Up to 38
	SMR KLT-40S	Indirect	2	3.72	290	Overheated	N/A			Up to 26
2	BWR	Direct	1	7.2	287.7	Saturated	1.7	258	Overheated	Up to 34
3	PHWR (CANDU®)	Indirect	2	4.7	260.1	Saturated	~1.2	240	Overheated	Up to 34
4	AGR	Indirect	2	17	560	Superheated	4	560	Superheated	Up to 42
5	LGR (RBMK)	Direct	1	6.9	284.9	Saturated	~0.3	~263	Overheated	Up to 33
6	LMFBR (BN-800)	Indirect	3 ^b	14.2	505	Superheated	2.5	505	Superheated	Up to 40

For more details, see [Handbook of Generation IV Nuclear Reactors \(2016\)](#).

^aAll current reactors connected to Rankine steam cycle (light-water working fluid).

^bBN-800 has 3 loops: (1) liquid sodium circulating inside reactor; (2) intermediate loop with liquid sodium; and (3) water-steam in Rankine cycle.

By pressure-boundary design

- 1) Reactor pressure vessel (RPV) - vast majority of reactors, i.e., 365 LWRs (RPVs made of steel) and 14 AGRs (RPVs made of concrete) or in total 86% of all reactors. In general, RPV of LWRs have two levels of pressures: (1) PWRs – 15 – 16 MPa (older reactors, e.g., VVER-440, and modern PWR SMR KLT-40S have pressures within 12–13 MPa); and (2) BWRs – around 7 MPa. AGRs carbon-dioxide cooled have pressure around 4 MPa.
- 2) Vessel reactors – two Russian SFRs, because pressure above the pool of liquid sodium is close to the atmospheric one; and
- 3) Pressure-channel reactors – 49 PHWRs (pressure up to 11 MPa (e.g., in CANDU[®] reactors) and 12 LGRs (pressure about 6–7 MPa) or in total 14% of all reactors.

Which pros and cons we have for these different pressure-boundary designs:

Pressure vessel (PV) contains reactor core, i.e., bundle strings or fuel-elements (rods) assemblies, which oriented vertically. Pressure vessel has an upper cover and internal insert to guide reactor-coolant flow downwards from the inlet through the gap between the reactor vessel and after that upward through the reactor core to the outlet. Pressure-vessel wall thickness depends mainly on inside pressure and internal diameter of a pressure vessel, and can be for: (1) VVER-1000: Pressure around 16 MPa and ID = 4.14 m - wall thickness ~19 cm; (2) US PWR: Pressure around 16 MPa and ID = 3.96 m - wall thickness - ~22 cm; and (3) US BWR: Pressure around 7 MPa and ID = 6.4 m - wall thickness - 15 cm. Therefore, a pressure vessel is a very important part in a reactor. Due to the three severe nuclear accidents a number of pressure-vessel manufacturing plants in the world were closed or transferred to another production affecting a maximum number of pressure vessels, which can be manufactured in the world. Also, some of these RPV manufacturing plants are oriented only to a certain nuclear vendor, e.g., Russian Atommasht (“АТОММАШ”) for VVER RPVs.

Traditionally, PHWRs are pressure-channel or pressure-tube reactors. In particularly, CANDU[®] reactors have horizontal orientation of fuel channels. Due to small OD of fuel channels, many plants worldwide can manufacture them.

By power cycle

Currently, all nuclear-power reactors are connected to subcritical-pressure Rankine steam-turbine cycles. The vast majority of NPPs equipped with AGRs, LMFBFRs, PWRs, and PHWRs (367 reactors from 442 or 83%) have, so-called, an indirect Rankine cycle, which includes a double loop (AGR, PWRs, and PHWRs): (1) reactor-coolant circulation loop, which includes PV with reactor core, circulation pumps, and steam generators (all of these equipment are located inside a containment building), and Rankine-cycle loop; and (2) triple loop (LMFBFRs – SFRs), which includes a primary loop - reactor vessel with fueled core, circulation pumps, and heat exchangers; intermediate loop with liquid sodium as the working fluid, circulation pump(s), and steam generator(s); and Rankine-cycle loop.

In general, the indirect cycle in which reactor coolant circulates only inside a containment building enhances reactor safety. However, this cycle requires steam generators or heat exchangers through which ~30–40 °C of a temperature difference is usually lost (as such, for PWRs (double loop) - APR-1400: Reactor outlet $T = 324$ °C, steam-generator saturated steam $T_{\text{out}} = 285$ °C, and $\Delta T = \sim 40$ °C; for EPR - $\Delta T = \sim 35$ °C; in terms of SFR – BN-800 (triple loop): Inside reactor $T_{\text{max Na}} = 547$ °C; inside intermediate loop $T_{\text{max Na}} = 505$ °C; and superheated steam $T_{\text{max}} = 490$ °C, and $\Delta T = \sim 57$ °C).

Therefore, in general, the direct cycle is considered to be more efficient from the thermodynamics point of view. Actually, Generation-III⁺ ABWRs have reached levels of pressures up to 7.2 MPa and saturated steam temperature of 287.7 °C (NPP thermal efficiency of about 34%). However, Generation-III⁺ PWRs, in particular, the largest in the world 1660-MW_{el} EPRs, have reached saturation-steam pressures close to 7.8 MPa and corresponding to that $T_{\text{sat}} = 293.3$ °C in the Rankine cycle. Therefore, as of today, NPPs with advanced PWRs have reached thermal efficiency of 36%–37% (38%), which is the highest thermal efficiency among all NPPs equipped with light- and heavy-water reactors.

Based on the above mentioned and accounting that AGRs and LGRs will never be built again, these two reactors types will not be considered in the following comparison. Also, SFRs will not be compared to PWRs/BWRs/PHWRs, because only two of them are in operation in Russia, and just a couple of them are planned to be built in other countries in the future.

Therefore, first comparison is PWRs vs. BWRs (see Table 8).

The second comparison is PWRs vs. PHWRs (CANDU-type reactors) (see Table 9), and Table 10 lists technical and deployment advantages and disadvantages of PHWR-type reactors.

One of the unique features of the CANDU[®]-reactor design is its ability to operate with alternative fuels such as a Recovered Uranium (RU) from the reprocessing of LWR spent fuel, MOX (80% UO₂ and 20% PuO₂), and thorium-based fuels, in addition to the conventional natural-uranium fuel.

In summary, Table 11 provides an overview of a possible “top ten” listing of the Pros (Advantages) and Cons (Disadvantages) for just the five principle general types of classes of reactors still actively involved or being considered in and for the power and energy markets. This Table 11 simply contrasts and compares the possible claims and issues for the generic PWR, BWR, PHWR, SFR, and SMR types, but without any distinction or differentiation between the multitude of design details and engineering variations, which have been illustrated in this chapter.

And, possibly, the final feature, which influences a decision for implementation of a type of reactor, is reactors/NPPs generation changing due to technological advances, accumulated operational knowledge, and design evolutions. Currently, we have nuclear-power reactors mainly of Generation-III. However, from 1990s, reactors of Generation-III⁺ have been built. The first ones were

Table 8 Major similarities and differences between PWRs vs. BWRs (for details, see Tables 2, 5, and 7a and 7b).

No.	Major features	
	PWRs	BWRs
1	Reactor design – reactor pressure vessel	
	Pressures up to 15–16 MPa	Pressures up to 7.2 MPa
	BWR RPV diameter and height can be larger than those in PWRs, but wall thickness is smaller	
	Bundle strings (assemblies) quite similar with vertical orientation and a large number of fuel elements (rods)	
	Control and shut-down rods installed from the top of RPV;	Control and shut-down rods installed from bottom of RPV;
	Control rods are moved with electrical motors; Shut-down rods are kept in the upper position with magnetic lock, dropped down by gravity force (passive safety system)	Control and shut-down rods are moved with electrical motors or/and hydraulic mechanism
	Batch refueling	
2	Fuel enrichment of 3–5% (can be up to 20% for SMRs)	Fuel enrichment of 1.9–2%
	Reactor coolant and moderator – light water, no separation between them	
	Non-boiling reactor coolant	Boiling reactor coolant
	Small changes of water density from inlet to outlet of RPV	Significant changes of water density from inlet to outlet of RPV
	At the RPV outlet – subcooled water with maximum temperatures up to $\sim 330^\circ\text{C}$ or $\sim 15\text{--}25^\circ\text{C}$ below saturation temperature	At the RPV outlet – saturated water-steam mixture with maximum temperatures up to 287.7°C
3	Wide range of installed capacities from 30 MW _{el} (SMRs) and up to 1660 MW _{el}	Smaller range of installed capacities from 150 MW _{el} and up to 1435 MW _{el}
4	Power cycle – subcritical-pressure Rankine steam-turbine cycle	
	Indirect (double loop) with steam generators	Direct
5	Thermal efficiency up to 37% (38%)	Thermal efficiency up to 34%

Table 9 Major differences and similarities between PWRs vs. PHWRs (CANDU-type reactors) (for details, see Tables 2, 5, and 7a and 7b).

No.	Major features	
	PWRs	CANDU®-type reactors
1	Pressure vessel	Pressure channel or pressure tube
2	Vertical bundle strings	Horizontal fuel channels
3	Relatively long bundle strings (~ 4 m)	Short bundles (495.3 mm each, 12 bundles per channel)
4	Bundle-string cross section - square or hexagonal	Bundle cross section - circular
5	Large number of fuel elements (rods) in bundle string, e.g., 17×17	Relatively small number of fuel elements (rods) in bundle - usually, 37 el., proposed 43 el.
6	Reactor coolant/moderator – H ₂ O (light water), no separation between them	Reactor coolant – D ₂ O (heavy water), Moderator - D ₂ O (heavy water), reactor coolant - inside fuel channel, moderator - outside
7	Higher $P_{\text{out}}/T_{\text{out}}$ inside reactor: 15.5 MPa/330 °C	Lower $P_{\text{out}}/T_{\text{out}}$ inside reactor: 9.9 MPa/310 °C
8	Higher $P_{\text{in}}/T_{\text{in}}$ to turbine inlet: 7.72 MPa/292 °C	Lower $P_{\text{in}}/T_{\text{in}}$ to turbine inlet: 4.6 MPa/260 °C
9	Higher thermal efficiency of NPP: up to 37% (38%)	Lower thermal efficiency of NPP: up to 32% (33%)
10	Batch refueling	Refueling on-line
11	Fuel enrichment 3–5%	Fuel – natural UO ₂ (enrichment $\sim 0.7\%$)
12	Wide range of installed capacities from 30 MW _{el} (SMRs) and up to 1660 MW _{el}	Small- and medium-size reactors: 90–880 MW _{el}

ABWRs built and put into operation in Japan. However, nowadays, advanced PWRs of Generation-III⁺ are being built around the world. Major differences between Generation-III⁺ and III are as the following:

- 1) In general, NPPs with Generation-III⁺ reactors have slightly higher thermal efficiencies compared to those of the same type-reactor NPP of Generation-III.
- 2) Generation-III⁺ reactors are equipped with passive-safety systems, which enhance safety.
- 3) Generation-III⁺ reactors are designed for operation of 60 years compared to 30–45 years of Generation-III reactors.

In summary, we would like to stress the major advantages of nuclear power:

- 1) Concentrated and reliable source of almost infinite energy, which is independent of weather conditions (however, it should be noted that within last summers, which were very hot on a record due to fast climate changes, some reactors/NPPs were forced to decrease power load or even were shut down for some time, because of lower levels of water in rivers, etc. and/or of relatively high water temperatures including not only in-land water resources, but, also, sea/ocean waters;

Table 10 Technical and deployment advantages and disadvantages of PHWR-type reactors.

No	
<i>Advantages</i>	
1	Ease of build as does not require large items that can be manufactured by very few heavy industrial plants in the world
2	Mass-flow rates and power shapes can be controlled in individual fuel channels
3	Reactor-coolant flow in pressure channels is interlaced (i.e., in neighboring channels flow has opposite directions), which allowing more uniform axial-density distribution/neutron flux
4	Flexible fuel cycles use natural-uranium plus spent fuel from PWRs and BWRs, and using thorium fuel allow optimum resource use
5	Refueling on-line provides higher capacity factors compared to batch-refueling reactors
6	Extra passive-safety system, as liquid-moderator heat sink, separated from reactor coolant
7	Channel and steam generator replacement/refurbishment allow long plant operation
8	Proven performance in fuel and major components have been demonstrated
9	Inherently modular-core design and SMR design
10	Extensive operating and build experience in multiple countries
<i>Disadvantages</i>	
1	Heavy-water reactor coolant and moderator more expensive than light water and require production facilities
2	Heavy water and natural uranium use gives slightly positive void and power coefficients, necessitating auto-control and two independent and diverse shut-down systems
3	Pressure tubes subjected to creep and sagging, which require their replacement in 20–30 years
4	Non-proliferation issues, because of on-line refueling and potential plutonium diversion, with potential for mixing civilian with overt or covert military uses
5	Limited additional development potential demonstrated to date and no new prototypes
6	No known or existing long-term high-level waste repository despite many years of study
7	Significant new build commitment in India, none in Canada
8	Focus on life extension with limited R&D investment in PHWRs
9	Move towards “fashionable” SMR developments has seemingly bypassed AHWRs

Table 11 Technical and deployment advantages and disadvantages of the various types or classes of reactors.

No	PWR	BWR	PHWR	SFR	SMR (PWR)
<i>Advantages</i>					
1	✓	✓	✓		
2	✓	✓	✓	✓	✓
3	✓	✓	✓	✓	✓
4			✓	✓	✓
5	✓	✓	✓		
6	✓	✓	✓		
7	✓	✓	✓	✓	
8	✓	✓	✓		
9			✓		✓
10	✓	✓	✓		
<i>Disadvantages</i>					
1	✓	✓		✓	
2	✓	✓	✓		
3	✓	✓	✓	✓	✓
4			✓	✓	✓
5			✓		
6	✓	✓	✓	✓	✓
7	✓	✓	✓	✓	✓
8	✓	✓	✓		
9	✓	✓	✓	✓	✓
10	✓	✓		✓	

Note: A check mark (✓) indicates the possession of the numerically listed advantage or disadvantage by that reactor type.

- 2) High capacity factors are achievable, often in excess of 90% with long operating cycles, making units suitable for continuous base-load operation;
- 3) Essentially negligible operating emissions of carbon dioxide (see [Figs. 5 and 6](#)) and relatively small amount of wastes generated (see [Table 12](#)) compared to alternate fossil-fuel thermal power plants plus the nuclear has the lowest death rate per TWh compared to that for other energy sources ([Fig. 7](#));

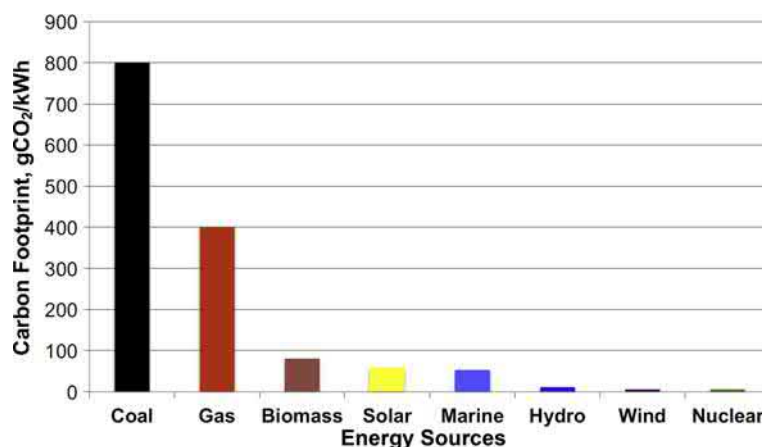


Fig. 5 Carbon footprint for various energy sources. If Carbon Capture and Storage (CCS) are used then the carbon footprint can be decreased for coal and gas by about 6 fold. (For details on carbon footprint of NPPs – see Fig. 6). Courtesy of Dr. J. Roberts, University of Manchester, UK; based on data from <http://researchbriefings.files.parliament.uk/documents/POST-PN-268/POST-PN-268.pdf>.

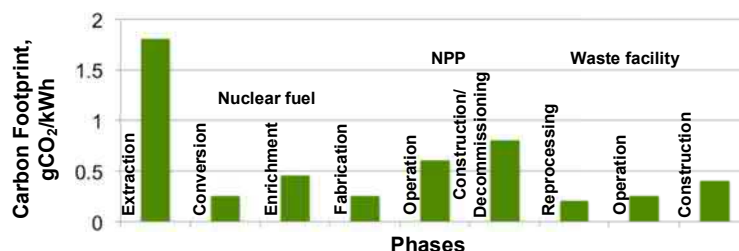


Fig. 6 Carbon footprint of nuclear power plant various phases. Courtesy of Dr. J. Roberts, University of Manchester, UK; based on data from British Energy for Torness AGR NPP.

Table 12 Per cent of various wastes in total amount.

No	Wastes	% of total amount
1	Mining and Quarrying	27.30
2	Agriculture	20.13
3	Demolition and Construction	18.51
4	Industrial	12.73
5	Dredged spoils	7.64
6	Household	6.94
7	Commercial	6.48
8	Sewage sludge	0.23
9	Radioactive	0.04

Courtesy of Dr. J. Roberts, University of Manchester, UK; partially based on data from <https://publications.parliament.uk/pa/cm200405/cmselect/cmenvfru/130/130we13.htm>.

- 4) Relatively small amount of fuel required compared to that of fossil-fuel thermal power plants (Table 13); and
- 5) NPPs can supply relatively cheap electricity (see Fig. 8) for re-charging of electrical vehicles during night hours as they usually operate on full load (capacity) 24/7.
- 6) As a result, nuclear power is considered as the most viable source for electricity generation within next 50–100 years. However, nuclear power must operate and compete in energy markets based on relative costs and strategic advantages of the available fuels and energy types.
- 7) In spite of all current advances in nuclear power, NPPs have the following deficiencies: (a) generate radioactive wastes; (b) have relatively low thermal efficiencies, especially, NPPs equipped with light- and heavy-water-cooled reactors (up to 1.6 times lower than that for modern advanced thermal power plants; (c) risk of radiation release during severe accidents; and (d) production of nuclear fuel is not an environment-friendly process. Therefore, all these deficiencies should be addressed in next generation – Generation IV reactors and NPPs.

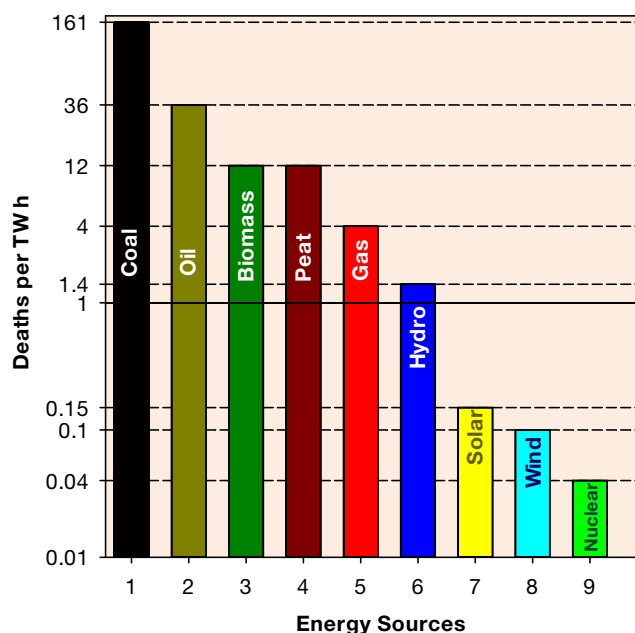


Fig. 7 Deaths per TW h for various energy sources (based on data from <https://www.nextbigfuture.com/2011/03/deaths-per-twh-by-energy-source.html>).

Table 13 Approximate tonnage of wastes per 1000-MW_{el} power per year for nuclear and coal-fired power plants.

Nuclear Power Plant	Coal-fired Power Plant
Fuel	
25 t of UO ₂	2.6 million t of coal (5 × 1400-t trains a day)
Wastes	
35 t High Level Wastes (HLW)	6,500,000 t of CO ₂
310 t Intermediate Level Wastes (ILW)	900 t of SO ₂
460 t Low Level Wastes (LLW)	4500 t of NO _x
	320,000 t of ash
	400 t of toxic heavy metals

Courtesy of Dr. J. Roberts, University of Manchester, UK.

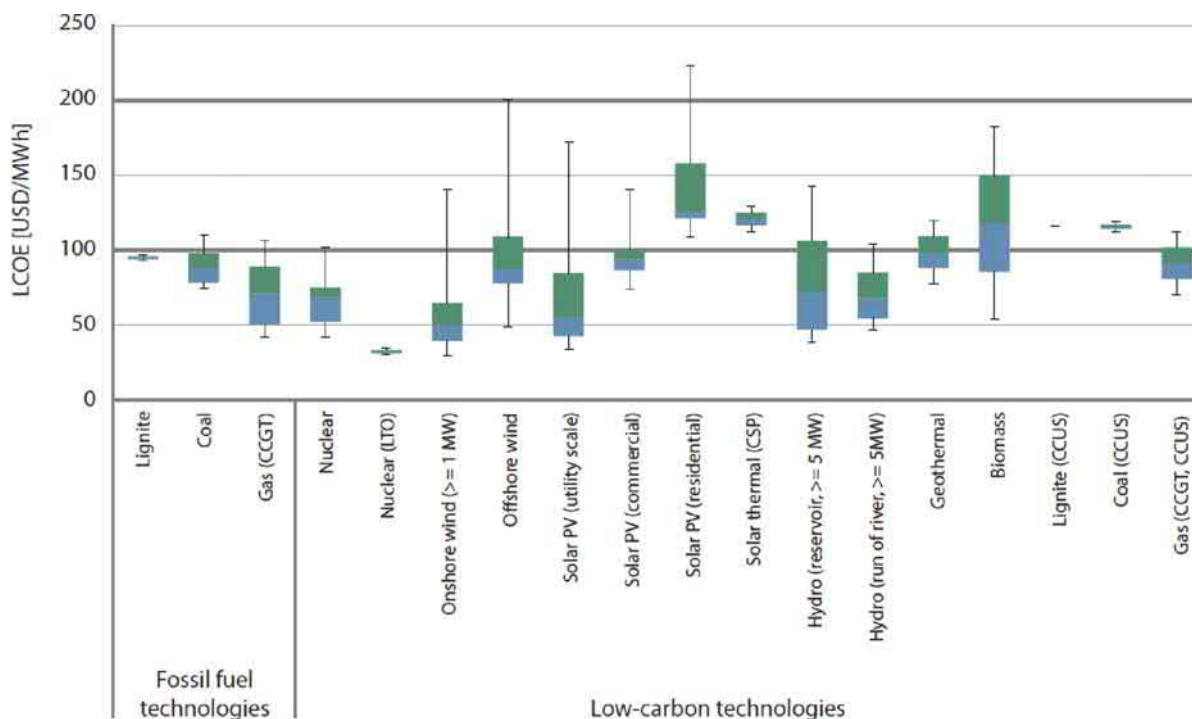


Fig. 8 Levelized Cost Of Energy (LCOE) by technology. Note: Values at 7% discount rate. Box plots indicate maximum, median, and minimum values. Boxes indicate central 50% of values, i.e., second and third quartile. USD – US Dollar; CCGT – Combined-Cycle Gas Turbine; LTO – Long-Term Operation; PV – PhotoVoltaic; CSP – Concentrated Solar Power; CCUS – Carbon Capture, Utilization and Storage. Courtesy and copyright by IEA, NEA, and OECD; <https://www.iea.org/reports/projected-costs-of-generating-electricity-2020>.

Conclusions

1. Currently, 33 countries have operating nuclear-power reactors, and 3 countries plan to build nuclear-power reactors. In addition, 30 countries are considering, planning or starting nuclear-power programs, and about 20 countries have expressed their interest in nuclear power. However, 13 countries with NPPs don't plan to build new nuclear-power reactors.
2. In January of 2021, 442 nuclear-power reactors operated around the world, which is less by 2 reactors compared to that before the Fukushima NPP severe accident in March of 2011 (however, the total installed capacity increased by 18 GW_e). This number includes 302 PWRs, 63 BWRs, 49 PHWRs, 14 AGRs, 12 LGRs, and 2 LMFBRs (SFRs). Considering the number of forthcoming reactors, the number of BWRs/ABWRs and PHWRs will decrease within next 20–25 years. Furthermore, within next 10–15 years or so, all AGRs (carbon-dioxide-cooled) and LGRs will be shut down forever.
3. Also, it should be stated that the history of the nuclear-power industry shows that nuclear accidents, especially, severe ones, can override any advantages of various reactors' types, and significantly affect future builds of certain types of reactors, e.g., LGRs or even all types of reactors.
4. Today, based on a summary of various parameters including all pros and cons, PWRs are considered as the most "popular" reactors, which are being built around the world. However, tomorrow and in the future the next generation or Generation-IV reactors/NPPs offering improved performance are planned to be eventually built.
5. In general, the major advantages of nuclear power are: (a) Concentrated and reliable source of almost infinite energy, which is independent of weather conditions; (b) High capacity factors are achievable, often in excess of 90% with long operating cycles, making units suitable for continuous base-load operation; (c) Essentially negligible operating emissions of carbon dioxide and relatively small amount of wastes generated compared to alternate fossil-fuel thermal power plants plus the nuclear has the lowest death rate per TWh compared to that for other energy sources; (d) Relatively small amount of fuel required compared to that of fossil-fuel thermal power plants. As a result, nuclear power is considered as the most viable source for electricity generation within next 50–100 years. However, nuclear power must operate and compete in energy markets based on relative costs and strategic advantages of the available fuels and energy types.
6. In spite of all current advances in nuclear power, NPPs have the following deficiencies: (1) Generate radioactive wastes; (2) Have relatively low thermal efficiencies, especially, NPPs equipped with water-cooled reactors (up to 1.6 times lower than that for modern advanced thermal power plants); (3) Risk of radiation release during severe accidents; and (4) Production of nuclear fuel is not an environment-friendly process. Therefore, all these deficiencies should be addressed in next generation – Generation IV reactors and NPPs.

See Also: Commercial Nuclear Power Plant Statistics.

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Economics of Nuclear Energy

Geoffrey S. Rothwell, Jean Michard Pélissier, Antibes, France; and Department of Economics (Retired), Stanford University, Stanford, CA, United States

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Glossary

CRF Capital Recovery Factor is the (annual) rate at which capital expenditures financed with debt and equity capital are paid to banks and investors.

D&D Decontamination and Demolition (Decommissioning) is the process by which an operating facility site is returned to its original state before operations. This involves the decontamination of contaminated structures and equipment, the removal of equipment, and the demolition of structures. Under different regulatory regimes this leads to the end (“decommissioning”) of the operating license.

Debt Two forms of debt are (1) a loan from a financial institution and (2) sales of bonds to investors. Both involve repayment of principle and *interest*. Both forms are repaid before payments to equity providers in the case of bankruptcy (which involves the restructuring of assets and liabilities).

Equity At least two forms of equity are (1) funds provided by the owner/operator out of earnings and (2) funds provided by investors hoping for a *rate of return* from profits.

GIF Generation IV International Forum is a group of countries working together to develop advanced reactor technologies, known as “fourth generation” technologies.

IDC Interest During Construction is the amount of financing charges (interest and returns to equity) that accumulate during the construction of a facility.

Interest rate It is the rate of return required by banks or on bonds, see debt.

LC Levelized cost is the value that equates (1) the discounted costs of a facility over its lifetime, including expenditures during construction and decommissioning with (2) the discounted revenues of a facility’s lifetime.

LCOE Levelized cost of electricity is the lifetime levelized cost of a generated megawatt-hour.

Nominal It is the cost or price of a good or service in the year of currency when expended.

PPI Producer Price Index is a measure of inflation relevant to producers, similar to the CPI, the Consumer Price Index, which is a measure of inflation facing consumers.

Real is the nominal cost or price of a good or service inflated or deflated to a currency in a particular year, e.g., 2018.

WACC Weighted Average Cost of Capital is an average of the rate of *interest* on debt investments (e.g., by banks) and the *rate of return* required on equity investments (e.g., by partners) weighted by the percentage of debt and equity to total capitalization (debt + equity). Some authors use WACC as “the cost of money.”

Introduction

Two types of costs, total cost (TC) and levelized cost of electricity (LCOE), provide much of the economic information necessary to determine the competitiveness of nuclear power. Total cost is important because nuclear power plants (NPPs) are capital intensive and required large upfront outlays. Levelized cost is important because it determines whether the NPP can compete in the long run against other generating technologies. The LCOE is a measure of discounted lifetime costs divided by discounted electricity revenues. The LCOE (or simply *LC*; here, the italic font denotes a variable and the roman font denotes a parameter), is that cost that equates the left and right sides of Eq. (1):

$$\sum_t [(I_t + O\&M_t + F_t) \cdot (1 + r)^{-t}] = \sum_t [LC \cdot E_t \cdot (1 + r)^{-t}] \quad (t = -LT, \dots, T) \quad (1)$$

where I_t are investment expenditures (including decommissioning costs) in year t ; $O\&M_t$ are Operations and Maintenance expenditures (including refurbishments) in year t ; F_t are fuel expenditures in year t ; r is the discount rate and $(1 + r)$ is the discount factor, e.g., $(1 + 5\%)$; LC is the levelized electricity (or energy product) generation cost over the plant's lifetime; E_t are the MWh of electricity generated in year t ; LT is the construction lead time; and $LC \cdot E_t$ are electricity revenues in year t , which are discounted to $t = 0$.

Because levelized cost is by definition constant for all years, LC can be taken out of the summation and both sides can be divided by the remainder on the right-hand side. Hence, LC can be defined as in NEA/IEA (2015, p. 28):

$$LC = \sum_t [(I_t + O\&M_t + F_t) \cdot (1 + r)^{-t}] / \sum_t [E_t \cdot (1 + r)^{-t}] \quad (t = -LT, \dots, T). \quad (2)$$

The summations in Eq. (2) run from (1) the start, as defined in IAEA (2019, p. 8), of large construction expenditures when the foundation is poured ($t = -LT$, the construction lead time) to the start of commercial operation ($t = 0$) and (2) from the start of commercial operation to the end of commercial operation, T , the life of the plant. (There is a third period from the end of the commercial life to the end of decommissioning; the cost of which is treated as a capital cost or as a contribution to a sinking fund accounted for in $O\&M$.) LC can be calculated in two ways: (a) using a “top-down” approach by making appropriate simplifying assumptions to Eq. (2) such that each cost is the same in each year of operation, or (b) using a “cash flow” approach by estimating the value of each input in each year on the right-hand side of Eq. (2) and discounting to $t = 0$ (for an example of a cash flow model see Table 3, below). Because of its simplicity and transparency, the first approach is used in this article.

Before discussing each of the inputs in Eq. (2), a distinction should be made between nominal and real values. Nominal costs are those in the currency of the year of the expenditure. Real costs are those in the currency of a specific year (in this article all dollars have been converted to 2018 unless otherwise stated). It is often difficult to tell in the literature whether costs and prices are in nominal currency or in real currency if not explicitly stated. Because taxes are assessed on, and budget outlays are stated in, nominal values, most cost estimates associated with governmental or regulatory oversight are in nominal currency.

Confusion enters when these nominal values are discounted (inflated or deflated) to a particular year. Some estimates use escalation rates. However, escalation rates are associated with particular costs. The escalation rate for fuel would not be the same as the escalation rate for construction materials or labor. Therefore, currency should be discounted using a price index, e.g., a producer price index, and real escalation rates (not including currency inflation) should be applied to those particular costs. Picking the proper escalation rates can be difficult because published versions of these rates generally include inflation and because an assumption is made that a particular input is the same over time, i.e., change orders during construction are not well represented by escalation rates. Further, escalation rates, e.g., for NPP construction, are usually calculated for specific regions. Finally, escalation rates are often calculated by observing changes in final costs; so, some NPP construction cost escalation rates confound real changes in the price of inputs, such as copper or welding labor, with increases in the productivity, availability, reliability, or safety of NPPs.

This article avoids using real escalation rates and discounts US dollars (USD) in each year to USD 2018 using the Producer Price Index for All Commodities (PPIACO) from Federal Reserve Bank (2019). The rate of inflation for producers, such as electric utilities (that use a different set of inputs than consumers for whom food is an important component in calculating the Consumer Price Index, CPI) increased an average of 5.7% per year from 2000 to 2008, then experienced an average of 0% from 2008 to 2018 (the average from 2000 to 2018 was 2.6%). This means that cost estimates for NPP construction, fuel, and $O\&M$ made before July 2008 should be discounted by the PPI, but cost estimates after July 2008 are *similar* to what they would be in 2018, 2019, or 2020 (ignoring real escalation and real changes in the costs of construction, $O\&M$, and fuel). Compare with inflation rates in Rothwell (2021a).

Examples of NPP electricity levelized costs can be found in NEA/IEA (2015). Table 1 updates these calculations. However, the ABWR cost estimate in NEA/IEA (2015) has been replaced with the estimated costs of a System 80+ from ORNL (2011). In each case the investment cost (construction cost plus financing cost) is the largest component of NPP LCOE. Investment costs are lowest in East Asia. The $O\&M$ cost (which is dominated by the cost of labor, see NEA/IEA, 2018a,b) is less than the fuel cost for Chinese

Table 1 LCOE and cost components for nuclear power plants at a 5% discount rate (2018\$).

Country		China	Korea	China	Belgium	US	Hungary	Japan	UK
Type		CPR	APR	AP	Gen III	PWR	VVER	ALWR	EPR
		1000	1400	1000		Gen III	1200		
Size	MWe	1,080	1,343	1,250	1,000	1,144	1,180	1,152	1,650
Overnight cost	\$/kWe	1,825	2,041	2,641	5,132	4,920	6,277	3,922	7,092
Investment cost	\$/kWe	2,175	2,310	3,147	6,117	5,858	7,208	4,673	8,322
Investment + D&D	\$/MWh	\$15.13	\$16.08	\$21.90	\$42.50	\$43.70	\$50.59	\$32.54	\$57.87
$O\&M$ cost	\$/MWh	\$6.57	\$9.75	\$7.39	\$13.69	\$19.06	\$10.50	\$27.70	\$23.68
Fuel + waste cost	\$/MWh	\$9.42	\$8.67	\$9.42	\$10.56	\$7.40	\$9.70	\$14.29	\$11.42
Variable cost	\$/MWh	\$15.99	\$18.41	\$16.73	\$24.25	\$26.46	\$20.20	\$42.00	\$35.11
Levelized cost	\$/MWh	\$31.00	\$34.00	\$39.00	\$67.00	\$70.00	\$71.00	\$75.00	\$93.00

Source: Based on data from NEA (Nuclear Energy Agency) (2018) *Sustainable Development and the Application of Discounting to the Calculation of the Levelized Costs of Electricity*. Nuclear Technology Development and Economics, NEA/NDG/R(2018). Available from [http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDG/R\(2018\)1&docLanguage=En](http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDG/R(2018)1&docLanguage=En); updated to 2018\$.

NPPs and greater than fuel cost for all other NPPs. LCOE is lowest in China and Korea, about the same in the United States, Continental Europe, and Japan, and highest in the UK. (Russia did not participate in writing NEA/IEA (2015), but Hungary provided data on an EU-approved VVER 1200.)

Nuclear Power Plant Construction Costs section discusses the total capital investment costs using the cost estimates for the System 80+, which was based on the System 80 design used at Palo Verde Units 1, 2 and 3, the largest NPP in the United States. This design was eventually purchased by the Korea Electric Power Corporation (KEPCO) and updated to the APR1400, licensed for construction in the United States by the US NRC in August 2019. *Operating and Maintenance Costs* section discusses O&M costs. *Fuel Cycle Costs* section discusses the nuclear fuel cycle and nuclear fuel costs. Finally, *Concluding Remarks* section discusses some caveats regarding Table 1.

Nuclear power plant construction costs

Fig. 1 shows the inputs, systems, and outputs of an NPP (on the history of the development of NPPs, see Maldonado and Brown, 2021). The “systems” correspond to “accounts,” to be discussed below. Here, the two most important metrics are (1) total construction cost and (2) levelized construction cost (GIF/EMWG, 2007, p. 9). This section calculates levelized capital cost, *LKC*, as a function of total capital investment costs, *KC*:

$$LKC = KC\{OC(MW) \cdot [1 + IDC(r, LT)]\} \cdot CRF(r, T)/E \quad (3)$$

where *KC* is total capital investment costs, a function of *OC* and *IDC*; *OC* is overnight cost (including contingency) a function of size, *MW*; *MW* is the size of the plant in net megawatts-electric; *IDC* is the rate of Interest During Construction to finance *KC*, a function of *r* and *LT*; *r* is the appropriate real cost of capital; *LT* is the construction lead time; *T* is the economic life of the facility; *CRF* is the capital recovery factor, a function of *r* and *T*, see Eq. (5); and *E* is the annual electricity output expressed in megawatt-hours (MWh).

Construction cost, *KC*, is the total amount spent on construction and commissioning before any electricity or revenues can be generated (as defined in GIF/EMWG, 2007, p. 79). *KC* is equal to the overnight construction cost with contingency plus financing costs (as measured by *IDC*). To measure these consistently, a set of standard definitions of construction accounts, structures, equipment, and personnel is required. The Code of Accounts (COA) used here is derived from the *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (GIF/EMWG, 2007), developed by the Economic Modelling Working Group (EMWG) of the Generation IV International Forum (GIF).

The GIF/EMWG (2007) COA is based on (1) the Energy Economic Database (EEDB) that contains a detailed description of how to calculate the LCOE for nuclear power and fossil-fired electric plants (US DOE, 1988) and on (2) the International Atomic Energy Agency (IAEA) updating the EEDB (IAEA, 2000), giving an international consensus COA that differs from the original EEDB (which is still in common use in the United States).

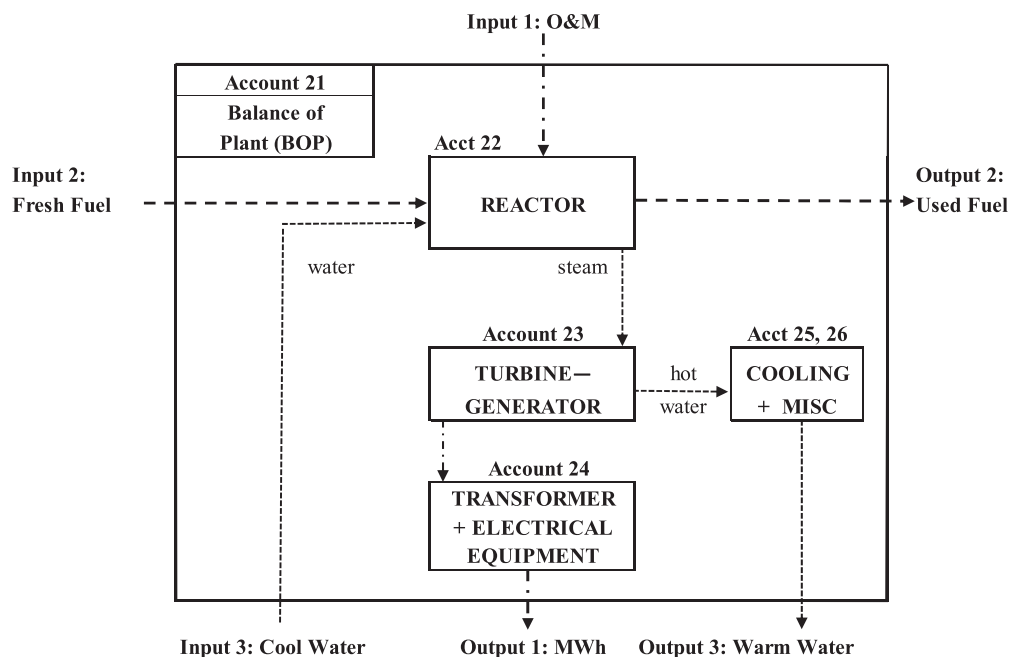


Fig. 1 Inputs, systems, and outputs of a nuclear power plant. Based on Rothwell GS (2016) *Economics of Nuclear Power*. Routledge Publishers, Francis and Taylor Group. Available from https://www.academia.edu/37151189/The_Economics_of_Future_Nuclear_Power.

One of the primary differences between the EEDB and the IAEA accounting systems is that the latter does not break out labor in the cost accounts. One of the secondary differences is that the IAEA's "Series" are not numbered the same way as is EEDB's "Accounts," particularly for indirect costs. The IAEA numbering system is used in the GIF/EMWG COA. In 2003, the GIF/EMWG began developing a hybrid of the original EEDB and IAEA cost accounting systems; primarily adding back labor, so that labor productivity and wages could be adjusted across GIF member countries. This became the 2007 *Guidelines* (GIF/EMWG, 2007). This article uses the latter to guide the cost estimation of new NPPs.

To calculate total capital investment cost, *KC*, **Table 2** presents the costs for a System 80 + 4-loop (four steam generators) PWR with a net size of 1144 MWe. *KC* includes:

- (1) Account 20, Direct construction costs plus Account 10, which includes pre-construction costs, i.e., those costs before the first concrete is poured for the foundation; in **ORNL (2011)** these costs were included in Account 21.
- (2) Account 30, Indirect costs include engineering and administrative costs that cannot be associated with a specific cost category in Account 20.
- (3) Account 40, Owners' costs include expenses incurred by the owner(s) associated with the plant and plant site, but not including off-site transmission system costs (although some cost estimates include transmission costs in owners' costs, increasing construction costs tremendously); the owners' costs associated with the development of the site, e.g., US NRC and US Environmental Protection Agency (EPA) licensing fees, are set to \$200 million plus 5% of base construction costs plus a contingency of 20%.
- (4) Account 50, Supplemental costs are primarily first fuel core costs (and if first fuel core costs are levelized in the cost of fuel, as is the case here, Account 50 can be set to zero). For heavy water reactors this account includes the cost of heavy water. This account can include D&D costs if these are accounted for as capital costs and not as annual contributions to a sinking fund.
- (5) In **Table 2**, a contingency of 20% is added to direct, indirect, owners', plus any supplemental costs; the appropriate contingency percentage for a "preliminary estimate" is discussed below.
- (6) Account 60, Interest During Construction, *IDC*, expressed as a percentage mark-up on overnight costs, is known as the "IDC factor."

The sum of Accounts 10–50 is the base overnight construction cost, *BASE*. The term "overnight" is used to describe what the construction cost would be if money had no time value. *BASE* plus contingency, *C_{on}*, equals total overnight cost, *OC*. Some references define *OC* without contingency and some references define overnight cost with contingency, as does the US Energy

Table 2 Construction cost estimates for a single unit of a twin PWR in the United States.

<i>System 80+</i>			
<i>COA</i>	<i>General description</i>	<i>Cost in 2018\$ (in millions)</i>	<i>Line</i>
10	Capitalized pre-construction costs (or in Series 21)	\$0	1
21	Structures and improvements	\$639	2
22	Reactor plant equipment	\$856	3
23	Turbine plant equipment	\$632	4
24	Electric plant equipment	\$230	5
25	Heat rejection system	\$138	6
26	Miscellaneous plant equipment	\$132	7
20	Capitalized direct costs	\$2,626	8
30	Capitalized indirect costs	\$1,556	9
40	Capitalized owners cost	\$486	10
50	Capitalized supplemental costs (e.g., first fuel core)	\$0	11
	Base construction costs (<i>BASE</i>)	\$4,668	12
	Contingency (20% of <i>BASE</i>)	\$904	13
	<i>Overnight construction cost</i>	\$5,573	14
	<i>Overnight construction cost/kWe</i>	\$4,871	15
60	Capitalized financing costs (<i>IDC</i>)	\$1,062	16
	<i>Total capital investment cost (KC)</i>	\$6,635	17
	<i>Total capital investment cost/kWe</i>	\$5,800	18
	<i>Annual payment to capital (5% for 60 years)</i>	\$351	19
	<i>Annual generation with 85% capacity factor (GWh)</i>	8,524	20
	<i>Levelized capital cost (LKC)</i>	\$41.12	21

Note: Cost have been increased from **ORNL (2011, pp. 32–33)** with a 2% real annual escalation rate for NPPs not yet under construction. Based on data from ORNL (2011) *Advanced High Temperature Reactor Systems and Economic Analysis*, ORNL/TM-2011/364, prepared by Oak Ridge National Laboratory under the direction of UT-Battelle, Oak Ridge (Table 6). Available from <http://info.ornl.gov/sites/publications/files/Pub32466.pdf>

Information Administration (US EIA, 2020). To make this distinction, *BASE excludes* contingency and *OC includes* contingency. *OC* plus Account 60, *IDC*, equals total capital investment cost, *KC*. To understand the relationship between lead time and compounding *IDC*, see Rothwell (2021a).

Although generally not a part of *KC*, “decontamination and dismantling” (D&D) costs should be set aside before or during operation for plant D&D after operation. In the latter case, D&D funding is treated as a sinking fund (and is assigned to O&M costs, see next section). In some IAEA software programs, the cost of decommissioning is added to the cost of construction, and no return is allowed on these set-aside funds under the implicit assumption that the D&D escalation rate is equal to the rate of return on the decommissioning funds. This greatly increases the total amount to finance by the time of commercial operation and leads to differences in levelized cost when compared to non-IAEA cost estimates.

The total capital investment cost, *KC*, can also be calculated with a cash flow model. *KC* is calculated in Table 3 based on the overnight cost from Table 2. The financing costs are assumed to be paid once in the middle of the year. The *KC* in Table 3 is slightly higher (\$6,644,760) than in Table 2 (\$6,635,000) due to differences in the calculation of *IDC*.

The remainder of this section discusses assumptions made above regarding (1) the appropriate contingency rate, see line 13 in Table 2; (2) the appropriate discount rate, see line 19; and (3) the appropriate capacity factor, see line 20. First, the appropriate contingency rate depends on the level of project definition or estimate “class.” Traditionally, cost contingency estimation relied heavily on expert judgement based on various cost-engineering standards, whereby contingency was added to the cost estimate to provide a reserve for cost risks and uncertainties during construction. OPEC (2018) discusses the typical estimating method for each estimate class: a Class (5) estimate is for screening among alternatives using parametric models, analogy, and/or judgement; a Class (4) estimate is for feasibility using parametric models with specified equipment; a Class (3) estimate is for budget allocation using semi-detailed unit costs for activity level line items; a Class (2) estimate is for establishing a project baseline using detailed unit costs with a general material take-off list (MTOL) with quantities and types; and a Class (1) estimate is for bidding or evaluating a bid using detailed unit costs with a detailed MTOL. Table 4 gives contingencies recommended by the Electric Power Research Institute (EPRI, 1993) by class. A 20% contingency, as in Table 2, is appropriate for a preliminary estimate. (See D’haeseleer, 2013, pp. 73–77, for a discussion of these issues.) Rothwell (2005) shows how to translate standard deviations of the cost estimate (e.g., from a Monte Carlo simulation) into a contingency rate and vice versa. This method can be applied to construction lead time, fuel prices, etc.; see Rothwell (2016).

Table 3 Cash flow calculation of total capital investment cost.

Year	Year to completion	%	KC	KC	KC	KC
		Construction	$r = 10\%$	$r = 7\%$	$r = 5\%$	$r = 3\%$
		Expenditure	M USD	M USD	M USD	M USD
2018	–7	0.00%	\$0.00	\$0.00	\$0.00	\$0.00
2019	–6	14.000%	\$1,449.63	\$1,211.15	\$1,071.36	\$945.47
2020	–5	14.000%	\$1,317.84	\$1,131.92	\$1,020.34	\$917.93
2021	–4	15.000%	\$1,283.61	\$1,133.43	\$1,041.16	\$954.85
2022	–3	15.000%	\$1,166.92	\$1,059.28	\$991.58	\$927.04
2023	–2	14.000%	\$990.12	\$923.98	\$881.41	\$840.03
2024	–1	14.000%	\$900.10	\$863.53	\$839.44	\$815.57
2025	0	14.000%	\$818.28	\$807.04	\$799.46	\$791.81
Totals		100.00%	\$7,926.51	\$7,130.34	\$6,644.76	\$6,192.70

Note: *KC* is the total capital investment cost.

Based on data from Based on data from ORNL (2011) *Advanced High Temperature Reactor Systems and Economic Analysis*, ORNL/TM-2011/364, prepared by Oak Ridge National Laboratory under the direction of UT-Battelle, Oak Ridge (Table 6).

Available from <http://info.ornl.gov/sites/publications/files/Pub32466.pdf>

Table 4 Suggested contingency rates as a function of estimate class.

Estimate	EPRI estimate	EPRI
Class	Designation	Contingency
5	Conceptual	Not available
4	Simplified	30–50%
3	Preliminary	15–30%
2	Detailed	10–20%
1	Finalized	5–10%

Based on EPRI (Electric Power Research Institute) (1993) *Technology Assessment Guide*. Palo Alto, CA, TR-102276-V1R7.

Second, there is the question of the appropriate discount rate. Traditionally, economists have argued that the discount rate should be the real corporate, tax-adjusted weighted average cost of capital (WACC). However, the tax rate varies significantly from one electric utility to another, so is ignored here.

$$r = \text{WACC} = [\mathbf{d} \cdot \text{debt} / (\text{debt} + \text{equity})] + [\mathbf{e} \cdot \text{equity} / (\text{debt} + \text{equity})], \quad (4)$$

where $\mathbf{d}$ is the real rate of return on corporate debt (bond) financing, debt is the total amount of corporate debt (borrowing), equity is the total “stock” financing, and $\mathbf{e}$ is the real rate of return on corporate equity financing. The cost of debt and equity to the nuclear power sector is discussed in Rothwell (2021a).

Focusing on the definition of appropriate long-term cost-benefit evaluations in the US DOE Federal Energy Management Program (US DOE/FEMP, 2017) recommends a discount rate of 2.8% for US government energy projects of 30 years or more, following the Office of Management and Budget guidance. This is approximately equal to economists’ consensus that the “social discount rate” is about 3%. Karoly (2018) presents a literature review of this rate. Hence, fixing the discount rate at 3% real for all expenditures and revenues *after* the electricity generating facility starts selling electricity into the transmission grid is reasonable. This is equivalent to refinancing a construction loan at a lower rate (3%) *after* construction. However, applying a risk premium during construction is also reasonable. For compatibility with NEA/IEA (2010, 2015), the cost of financing during construction are set equal to 3%, 5%, 7%, and 10% for purposes of comparing the costs of electricity generating alternatives; see discussion in NEA (2018). On the other hand, following NEA (2016), the rate of return on decommissioning trust funds can be assumed to be zero, given that cost escalation of decontamination and demolition (D&D) and radioactive waste management are likely to be equal to or greater than the return on these funds (primarily because regulators restrict NPP owner/operators from investing nuclear decommissioning trust funds in low-risk assets).

Also, the NPP owner/operator borrows funds from banks (debt) and uses internal funds and funds from investors (equity) to finance the construction of the plant, where the debt to total liabilities (debt plus equity) ratio can be as high as 80%. Under rate-of-return regulation, amounts of these funds were transparent because of regulatory requirements. In deregulated markets some of these amounts are consider strategic information. To simplify capital accounting, the capital recovery factor (CRF) is the rate at which the NPP owner/operator pays off its debts and pays a return to equity investors. The CRF is a function of r (the WACC in Eq. 4) and the payback period, T (CRF increases with r and decreases with T):

$$\text{CRF} = \left\{ \mathbf{r} \cdot (1 + \mathbf{r})^T / \left[(1 + \mathbf{r})^T - 1 \right] \right\} \quad (5)$$

Third, one must estimate how much electricity is sold in each year. The US NRC (2001) defines several capacity factors, each with a different measure of capacity. In Eqs. (2) and (3), let E be the US NRC’s “net electrical energy generated”; MW is the “net maximum dependable capacity” (dependable capacity is the capacity upon which the transmission grid operator can rely for dispatch); and TT is the total time in a year, i.e., 8766 h (averaging over leap years): $CF \equiv E / (MW \cdot TT)$. Average CFs were less than 78.2% before 1999 (with a minimum average monthly capacity factor of 47% in 1974) and greater than 78.2% after 1998 (with a maximum average monthly capacity factor of 94% in 2018); US EIA (2020).

Operating and maintenance costs

This section discusses non-fuel operations and maintenance (O&M) costs. The GIF/EMWG Guidelines (2007, pp. 97–101) defined the following cost accounts, as well as contingency on estimated O&M costs (these account definitions have been updated to more closely correspond to ORNL, 2011):

- (1) Account 71 On-site staffing expenses includes all management and O&M personnel assigned to the plant site.
- (2) Account 72 Salary related costs: Costs of pensions and benefits, including worker’s compensation insurance, provided for the on-site staff. (These costs can be included in Account 71.)
- (3) Account 73 Consumables includes items such as resins, chemicals, make-up fluids, materials, and other non-depreciable items, such as small equipment and tools required for maintenance. This account also includes the costs of supplies for the spent fuel pool.
- (4) Account 74 Repairs and spare parts: Funds for unanticipated repairs and expenditures on spare parts.
- (5) Account 75 Subcontracted services: Expenditures for personnel not assigned full time to the plant site, e.g., safety reviews, off-site training, environmental monitoring, meteorological surveys, fuel studies, refueling services, other owner home office activities directly supporting the plant, and the costs of external disposal of operational radioactive wastes.
- (6) Account 76 Regulatory fees: In the United States this account includes US NRC annual fees and review costs, as well as other routine safety, environmental, and health physics inspections. In other countries NPP owner/operator annual costs for this category depend on their regulatory environment.
- (7) Account 77 Plant refurbishments and upgrades: Cost of large capital items that must be purchased during operation (e.g., steam generator replacement), averaged over the lifetime of the plant. These expenditures can be estimated as a percentage of construction direct costs.
- (8) Account 78 Taxes and insurance: Property taxes, sales and value-added taxes, and any other non-income-based taxes, as well as premiums for commercial and government liability insurance, property damage insurance, and replacement power insurance.

Table 5 O&M cost estimates for a single unit of a two-unit PWR in the United States.

<i>PWR 4-loop of MWe = 1144</i>			
<i>COA</i>	<i>General description</i>	<i>Cost 2018\$ (millions)</i>	<i>% of annual O&M</i>
71	On-site staffing expenses	\$36.27	22.2%
72	Salary related costs (pensions and benefits)	\$9.68	5.9%
73	Consumables	\$28.72	17.5%
74	Repairs and spare parts	\$7.02	4.3%
75	Subcontracted services	\$9.82	6.0%
76	Regulatory fees	\$6.28	3.8%
77	Capital replacements as 0.5% of direct capital	\$13.13	9.4%
78	Taxes and insurance	\$10.85	6.6%
79	Other general and administrative costs	\$12.27	7.5%
	Contingency (as a percent of total O&M costs)	\$26.81	16.7%
70	<i>Annual O&M Costs</i>	\$160.85	100.0%
	<i>Annual Capital Costs per MWh</i>	\$41.12	
	<i>Annual D&D Costs per MWh</i>	\$2.15	
	<i>Annual O&M Costs per MWh</i>	\$18.87	

Based on ORNL (2011) *Advanced High Temperature Reactor Systems and Economic Analysis*, ORNL/TM-2011/364, prepared by Oak Ridge National Laboratory under the direction of UT-Battelle, Oak Ridge. Available from <http://info.ornl.gov/sites/publications/files/Pub32466.pdf>, updated to 2018\$, with a 2% real annual escalation rate for NPPs not yet under construction.

- (9) Account 79 Other general and administrative costs: Other O&M costs that cannot be accounted for in Accounts 71 through 78, such as office supplies, balance of plant (non-nuclear) O&M, visitors' center, etc.

Table 5 shows O&M cost estimates from [ORNL \(2011, p. 81\)](#). In this estimate annual O&M costs are approximately \$19/MWh. Other discussions of the relationship between staff size and O&M can be found in [Rothwell \(2016, Section 4.3\)](#) and [NEA \(2018, Chapter 2\)](#).

Here refurbishments are estimated at 0.5% of the direct cost of the plant. This provides approximately \$13 million per year, which can be spent every year or accumulated like the D&D fund. There are three categories of refurbishments: uprates, upgrades, and life extensions. Uprates involve increasing the maximum power level at which a commercial NPP is allowed to operate. Upgrades involve the replacement of equipment with (generally) larger or more reliable equipment to allow the production of more electricity or to increase the safety of the NPP. Life extension (also known as "Long Term Operation") can involve uprates and upgrades to operate the plant longer than originally anticipated. See [EPRI \(2013\)](#) and [IAEA \(2018\)](#). In many countries this requires replacing older equipment to bring the plant up to more recent safety standards.

Uprates and upgrades can be found by comparing NPP characteristics from one year to the next in the IAEA's *Nuclear Power Reactors in the World*, e.g., comparing [IAEA \(2019\)](#) to the previous addition. For example, in Sweden after the closure of Barsebäck Units 1 and 2 in 1999, uprates and upgrades at the remaining nine reactors increased their capacity by 1050 MW by the end of 2008 and another 569 MW by mid-2014 ([WNA, 2019](#)). Unfortunately, it is difficult to determine the costs of these uprates and upgrades because announcements regarding future projects are usually in nominal currency and are not broken down in terms of specific equipment or systems.

To this must be added contributions to an Nuclear Decommissioning Trust (NDT, see [NEA 2016](#)) fund such that the value of the fund is equal to the cost of the D&D of the plant, the on-site storage of irradiated nuclear fuel, and the D&D of the nuclear fuel storage facility; see [GIF/EWMG \(2007\)](#) and [McCullum \(2021\)](#). Contributions to an NDT fund, managed by a fiducially distinct entity, are accumulated during operation. The consensus value of D&D is 20% of the Overnight Construction Cost. (This is higher than the 15% assumed in [NEA/IEA, 2015](#)). There is no consensus on the appropriate rate of return on the NDT. Recent experience has shown that real escalation in D&D costs is approximately equal to the rate of return on long-term low-risk investments. See, for example, [CPUC \(2014, pp. 110–129\)](#) and [Southern California Edison \(2016\)](#). Further, without large investments in radioactive waste storage facilities, the cost of waste management could also rise faster than the general rate of inflation (e.g., PPI), eating into the return on D&D funds. Therefore, the annual contribution is assumed to be equal to the estimated decommissioning cost divided by the number of contributing years. Considering the Overnight Cost in **Table 2**, 20% is approximately \$1.1B, divided for 60 years (as assumed in [NEA/IEA, 2015](#)), implies an annual contribution of approximately \$18M or \$2.15/MWh. Finally, the D&D estimates can be very low if it is assumed that the return on NDT funds is equal to the rate investors require on construction expenditures.

Fuel cycle costs

The generic nuclear fuel cycle (**Fig. 2**) consists of (1) uranium mining and milling to create uranium oxide; (2a) conversion from uranium oxide to UF₆ (uranium hexafluoride) and enrichment to increase the percentage of the fissionable U²³⁵ from 0.7% up to

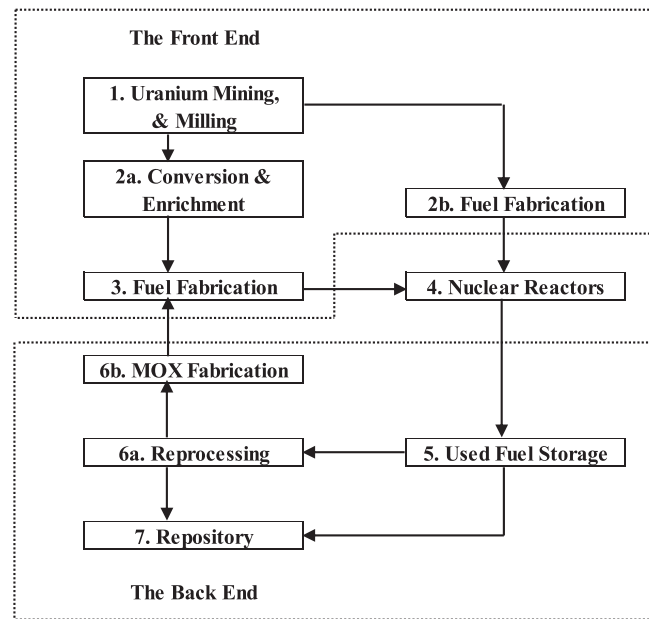


Fig. 2 Generic nuclear fuel cycle (once and twice through). Based on Rothwell GS (2016) *Economics of Nuclear Power*. Routledge Publishers, Francis and Taylor Group. Available from https://www.academia.edu/37151189/The_Economics_of_Future_Nuclear_Power.

5% yielding enriched product and depleted tails, or (2b) fabrication of fuel using natural uranium, such as in Pressurized Heavy Water Reactors; (3) conversion from UF_6 to UO_2 and fabrication into fuel rods and assemblies; (4) irradiation in a nuclear reactor; (5) irradiated fuel storage; (6) reprocessing to separate out (6a) U^{235} and plutonium and the fabrication (6b) of mixed oxide (MOX) fuel by combining recovered U^{235} and plutonium with natural uranium or depleted tails to be reused in a nuclear reactor; and (7) disposal of irradiated fuel and reprocessing residuals into a deep geologic repository (DGR). (On nuclear fuel cycles.) A “once-through” nuclear fuel cycle does not include reprocessing. A “twice-through” nuclear fuel cycle includes one MOX step. A “closed” fuel cycle includes many recycles to burn long-lived isotopes, generally with fast-spectrum nuclear reactors, not discussed in this article; see Heidet (2021).

Most commercial NPPs are Light Water Reactors (LWRs), Pressurized Heavy Water Reactors (PHWRs, including CANDUs (Canada Deuterium, Heavy Water, Uranium reactors), Gas-Cooled Reactors (GCRs), or Light Water Graphite Reactors (LWGRs). Table 6 presents reactor types, NPP types, and a simple description of the fuel they use. Low enriched uranium (LEU) fuel accounting is complex if done precisely reflecting the periods during which each fuel bundle is used in the reactor. Generally, it is assumed that fuel is paid for in a uniform stream over the life of the plant, without regard to the changing isotopic composition of a reactor’s fuel. The isotopic composition of LEU (Low Enriched Uranium), MOX (Mixed Oxide), and CANDU (Canadian Deuterium) natural uranium (NU) fuels is given in Table 7.

One characteristic that distinguishes LEU fuels is their potential burnup, primarily associated with their enrichment (see Fig. 3), defined as the number of megawatt-days, MWd, produced per kgU of fuel (MWd are equal to the number of thermal megawatts, MWth, produced per day times the thermal efficiency, ϵ , of the power plant, yielding MWd.) Consider levelized fuel costs over the life of the NPP and fuel expenditures in each year. Following Rothwell (2016, Section 4.2), the cost per MWh of LEU fuel, F , is

Table 6 Types of nuclear fuel in commercial use (not including fast reactors).

Type	NPP Type	Physical specification
LWR	PWR, VVER BWR, ABWR	Cubic/hexagonal cross-section, 4–5 m long 200–500 kgU per assembly
PHWR	CANDU	10 cm dia. $\times$ 50 cm long, 20 kgU bundle
GCR	Magnox AGR	3 cm dia. $\times$ 1.1 m long slug 24 cm dia. $\times$ 1 m long assembly
LWGR	RBMK	8 cm dia. $\times$ 10 m long assembly

Source: Based on IAEA (2007). *Operation and Maintenance of Spent Fuel Storage and Transportation Casks/Containers*. Vienna, Austria: International Atomic Energy Agency, IAEA Nuclear Energy Series, No. NF-T-3.5; Table 1, p. 5. Available from https://www-pub.iaea.org/MTCD/Publications/PDF/te_1532_web.pdf

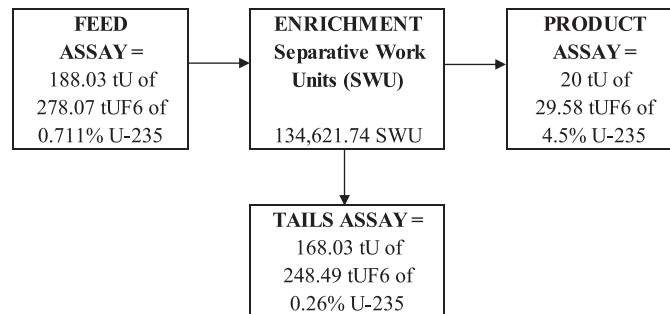
Table 7 Isotopic composition of LEU, MOX, and NU-CANDU.

	LEU		MOX		CANDU	
Burnup	Fresh	Spent	Fresh	Spent	Fresh	Spent
(GWd/MTHM)	50	50	50	50	7	7
U-234	0.04	0.02	0.002	0.018	0.036	0.010
U-235	4.50	0.96	0.228	0.119	0.720	0.230
U-236	0.00	0.58	0.000	0.026	0.000	0.070
U-238	95.46	92.01	91.040	87.999	99.244	98.570
Pu-238	0.00	0.03	0.232	0.225	0.000	0.000
Pu-239	0.00	0.63	4.843	2.514	0.000	0.250
Pu-240	0.00	0.26	1.991	1.791	0.000	0.100
Pu-241	0.00	0.14	0.934	0.774	0.000	0.020
Pu-242	0.00	0.08	0.587	0.695	0.000	0.010
Am-241	0.00	0.04	0.143	0.143	0.000	0.000
Other TRU	0.00	0.10	0.000	0.266	0.000	0.004
Total actinides	100.00	94.85	100.000	94.570	100.000	99.264
Total U	100.00	93.57	91.270	88.162	100.000	98.880
Total Pu	0.00	1.15	8.587	5.999	0.000	0.380
Total TRU	0.00	1.19	8.730	6.408	0.000	0.384
Fission products	0.00	5.15	0.000	5.430	0.000	0.736
Total	100.0	100.0	100.000	100.000	100.000	100.000

Note: TRU are Transuranic elements with the atomic weights greater than uranium (92), including plutonium with an atomic weight of 94.

Fission products are the elements created with the fission of uranium and TRU isotopes.

Based on data in MIT (Massachusetts Institute of Technology) (2011) *The Future of the Nuclear Fuel Cycle: An Interdisciplinary Study*. Available from <https://energy.mit.edu/wp-content/uploads/2011/04/MITEL-The-Future-of-the-Nuclear-Fuel-Cycle.pdf> and Glodeanu F, Miller W, and Patrascoiu S (2015) Recycle or direct disposal of spent fuel? In: Presented at the *International Symposium on Nuclear Energy*, Bucharest, Romania. Available from https://www.researchgate.net/publication/283299237_Recycle_or_direct_disposal_of_spent_fuel

**Fig. 3** Low enriched uranium at 4.5% U-235 product assay. Author calculations based on equations in the text.

$$F = [FC/(\varepsilon \cdot \text{BURN} \cdot 24)] + \text{SNF} \quad (6)$$

where FC is the cost of nuclear fuel in USD per kilogram of uranium (USD/kgU); ε is the thermal efficiency of converting MW-thermal into MW-electric; BURN is the burnup rate measured in thermal megawatt-days per kgU; 24 is the number of thermal MWh in a thermal megawatt-day; and SNF is the interim storage cost of Spent Nuclear Fuel per MWh (assumed to be \$1/MWh real) plus geologic disposal costs (assumed to be \$1/MWh real).

For example, if the burnup rate is 50 megawatt-days per kgU and the efficiency is 33.3%, then the cost of fuel per MWh is the cost of irradiated fuel management and disposal, often assumed to be equal to \$1.33/MWh for management and \$1/MWh for disposal plus the cost of fuel per kgU divided by $[(24 \cdot 50)/3] = 400$. If the cost per kgU is \$2000, then F equals \$5 plus \$2.33, or about \$7.33/MWh. (Fuel prices in Table 1 were estimated in 2014 when both the prices of natural uranium and SWU were higher; see Fig. 4.)

FC in Eq. (6) can be calculated with Eq. (7), in which all prices are given in 2018 USD/kg of product (and ignoring financial charges on expenditures less than one year before using the fuel in an NPP and fractional losses during processing of less than 1%, see Bunn et al., 2003, p. 98):

$$FC = (\text{URAN} \cdot P_{\text{UF6}}) + (\text{SWUS} \cdot P_{\text{SWU}}) + P_{\text{FAB}}, \text{ where} \quad (7)$$

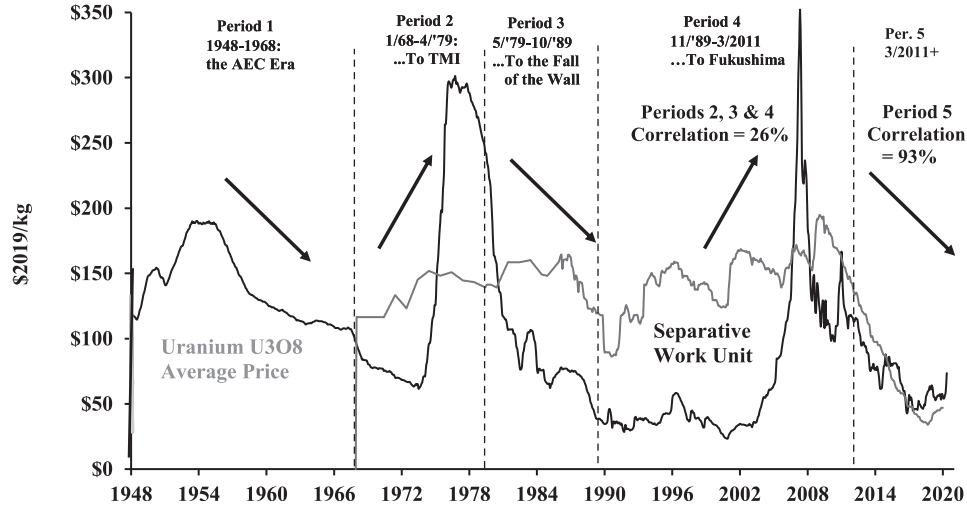


Fig. 4 Natural uranium and SWU spot prices (2019\$/kg). Based on Rothwell GS (2016) *Economics of Nuclear Power*. Routledge Publishers, Francis and Taylor Group. Available from https://www.academia.edu/37151189/The_Economics_of_Future_Nuclear_Power, updated with prices collected semi-annually since 2006 from the UxC LLC website and converted to 2019\$ prices: <https://www.cameco.com/invest/markets/uranium-price>.

$URAN$ is the amount of natural uranium input to produce enriched uranium output; P_{UF_6} is the price of natural uranium plus the conversion of U_3O_8 to UF_6 (e.g., \$10/kgU); $SWUS$ is the number of Separative Work Units required in enrichment, measured in kg; P_{SWU} is the price of enriching uranium hexafluoride, UF_6 ; and P_{FAB} is the price of converting and fabricating UO_2 fuel from enriched UF_6 .

Separative Work Units (SWUs) are a measure of work needed to separate U^{235} from natural uranium; see Rothwell (2009).

In Eq. (8), $URAN$ is the natural uranium input for one kgU of enriched uranium product. As shown in Fig. 3, 20 tU, the approximate amount of 4.5% enriched uranium to generate one gigawatt-year of electricity, requires 188 tonnes of natural uranium and 134,622 SWU, and yields 168 tonnes of depleted uranium with 0.26% U^{235} .

$$URAN = (X_{PE} - X_{TE}) / (X_{FE} - X_{TE}) \quad (8)$$

where X_{FE} is the percentage of U^{235} (assay) in initial natural uranium feed about 0.711%; X_{PE} is the percentage of U^{235} in the enriched uranium product, for example 4.5%; and X_{TE} is the optimal percentage of U^{235} in the tails, e.g., 0.26%.

The optimal tails assay can be approximated by an exponential function of the ratio of P_{SWU} and P_{UF_6} for enrichments with feed assays of 0.711%. Eq. (9) was calculated with ordinary least squares (OLS) with data from numerical optimization with a fixed product assay (here, 4.5%) and is not applicable to other product assays (a similar equation must be estimated for each product assay; X_{TE} is an endogenous variable and without Eq. (9) it must be calculated with an optimization routine). In Eq. (7), $SWUS$ is the number of separative work units required in enrichment, as given in Eq. (10):

$$X_{TE} = \exp \{ -6.085 + 0.468 \cdot [\ln (P_{SWU} / P_{UF_6}) - 0.074 \cdot [\ln (P_{SWU} / P_{UF_6})]^2] \} \quad (9)$$

$$SWUS = [V(X_{PE}) - V(X_{TE})] - URAN \cdot [V(X_{FE}) - V(X_{TE})] \quad (10)$$

$$V(X_i) = (2X_i - 1) \cdot \ln [X_i / (1 - X_i)], \text{ where} \quad (11)$$

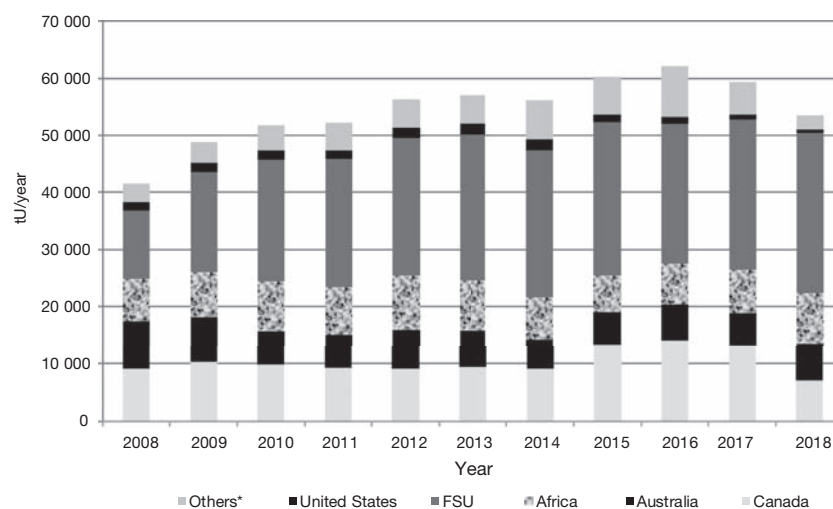
where i is either the feed assay, FE , the product assay, PE , or the tail assay, TE , and $V(X_i)$ is known as the "value function," with $0 < X < 1$, i.e., $V(X_{FE})$, $V(X_{PE})$ and $V(X_{TE})$.

Fig. 4 presents spot market uranium and average SWU prices. (Spikes in spot uranium prices were not as pronounced for long-term contract prices; see data at UxC LLC). The period from 1948 to 2020 is divided into (1) Period 1 from 1948 to 1968, before US electric utilities were allowed to own uranium; Period (2) from 1968 to April 1979 when the Three Mile Island accident occurred; Period (3) from the TMI accident to the collapse of the Soviet Union (when the Soviet weapons-grade highly-enriched uranium was dumped on the uranium market); Period (4) from the collapse of the Soviet Union to the Fukushima Dai-ichi accident; and Period (5) from the closure of the Japanese NPPs after the Fukushima accident to early-2020. The peaks and troughs of these series are discussed in Rothwell (2016, pp. 10–12). Using equations from Rothwell (2016, Section 4.3) and assuming the cost of re-conversion and fuel fabrication is \$300/kgU (Rothwell, 2010), LEU fuel prices ranged between about \$1760/kgU in 2007 and 2012 to about \$780/kgU in 2018.

One of the reasons the price of enrichment has been declining is the increasing enrichment capacity in the world market; Table 8. Between 2015 and 2020 the WNA projected that Russian capacity would increase by approximately 2,000,000 SWU and China's capacity would increase by 5,000,000 SWU. Russia and China are attempting to provide increasing capacity for their growing home and international fleets. In the case of China this means that the Chinese NPPs will be using less SWU from the international market. The Chinese will likely continue to increase their indigenous enrichment capacity until they can supply all of their

Table 8 Uranium enrichment capacity (thousands of SWU).

Country	Facility	2010	2015	2020 ^a
France	Orano, Georges Besse I & II	5,500	7,000	7,500
Europe	Urenco: Germany, Netherlands, UK	14,200	14,400	14,900
Japan	JNFL, Rokkasho	75	75	75
USA	Urenco, New Mexico	3,500	4,700	4,700
Russia	Tenex: Angarsk, Novouralsk, Zelenogorsk, Seversk	26,000	26,578	28,660
China	CNNC: Hanzhun & Lanzhou	2,200	5,760	10,700
Other	Various: Argentina, Brazil, India, Pakistan, Iran	75	100	170
	<i>Total SWU/year approximate</i>	51,550	58,600	66,700
	<i>Requirements (WNA reference scenario)</i>	49,154	47,285	57,456
	<i>Difference (surplus)</i>	2,396	11,315	9,244

^aProjection.Based on data in WNA (World Nuclear Association) (2020) *Uranium Enrichment*. Available from <https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/conversion-enrichment-and-fabrication/uranium-enrichment.aspx>.**Fig. 5** Recent world uranium production, 2008–2018. Note: FSU = Former Soviet Union. <https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/mining-of-uranium/world-uranium-mining-production.aspx>

indigenous needs. This increase to supply their own needs is putting downward pressure on the world SWU price. This will put downward price pressure on Orano's plants in France and on Urenco's plants in Germany, the Netherlands, the United Kingdom, and the United States. Because of the inverse tradeoff between enrichment and natural uranium feed, falling prices for SWU should increase the optimal tails assay for uranium product. Thus, the demand for uranium could increase if uranium supply were constant. However, with fewer NPPs operating in Japan, NPPs retiring in Germany, Europe, and the United States, uranium demand is also declining. This could continue until the NPPs under construction come into operation at a rate greater than those retiring. The most recent "Red Book" (NEA/IAEA, 2018b) shows a slight decline in uranium production, see Fig. 5, particularly in the United States where uranium production fell to 735 tons in 2018 (enough to supply about 4 LWRs, see Fig. 3), one-third less than in 2017. As an alternative to enriched uranium fuel, mixed uranium oxide and plutonium oxide fuel (MOX) is available if irradiated nuclear fuel can be reprocessed, see Table 7 and Fig. 6.

Approximate fuel costs, assuming conversion from natural uranium to uranium hexafluoride costs \$10 per pound of uranium feed, uranium hexafluoride and SWU are \$100 per kg, and fuel fabrication is \$300 per kg, for typical burnups and enrichment levels for various NPPs are given in Table 9 with the amounts of natural uranium required for a kgU of LEU and the number of SWU required assuming a tails assay of 0.26% and a product assay of 4.5%. As an alternative to LEU, consider MOX. MOX costs for incumbent MOX producers (those that can produce MOX without new investment) are shown in Fig. 6. (The cost of reprocessing and MOX fuel fabrication is unknown at this time because so little of this type of investment has been done since the delays associated with Rokkasho in Japan and the cancellation of the MOX fuel fabrication facility at the Savannah River Site in South Carolina.) The cost of MOX is higher than any fuel in Table 9. This is primarily because the labor requirements to produce MOX are much higher per tonne of output (measured in Metric Tonnes of Heavy Metal, MTHM) due to radiation safety measures compared to a LEU fuel fabrication facility (Rothwell, 2016, Section A4). ("Minimum Efficient Scale" is that level of output with a cost that is within 10% of the cost of the largest possible facility.)

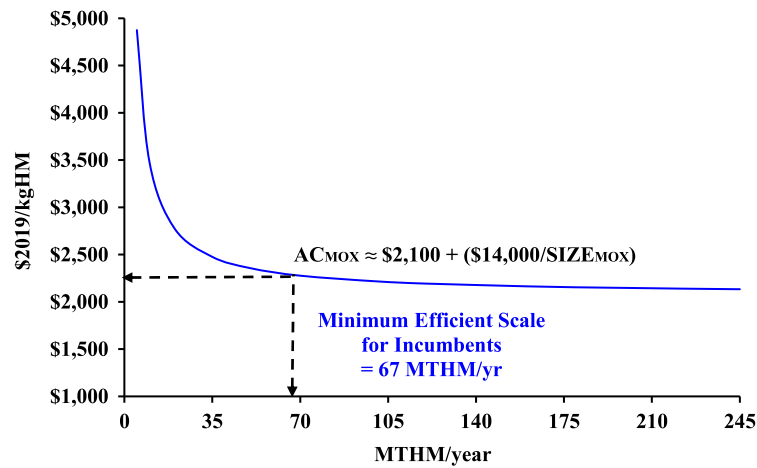


Fig. 6 The cost of MOX as a function of facility size; MTHM, Metric Tonnes of Heavy Metal. Based on Rothwell GS (2016) *Economics of Nuclear Power*. Routledge Publishers, Francis and Taylor Group. Available from https://www.academia.edu/37151189/The_Economics_of_Future_Nuclear_Power.

Table 9 Approximate fuel costs for nuclear power plants in Table 6.

	Burnup	Xp, Prod	UF6	SWU	Xt, Tails	Fuel	Cost/
	GWd/tU	Assay	kg	kg	Assay	Cost/kg	GWd
GCR (UK)	5	0.71%	1.00	0.00	0.71%	\$390	78
PHWR	7	0.71%	1.00	0.00	0.71%	\$390	56
BWR, t < 1973	30	2.70%	5.41	3.15	0.26%	\$1,112	37
AGR (UK)	30	3.30%	6.74	4.31	0.26%	\$1,348	45
VVER-440	30	3.60%	7.41	4.91	0.26%	\$1,511	50
PWR, t < 1973	35	3.20%	6.52	4.12	0.26%	\$1,366	39
RBMK	40	3.00%	6.08	3.73	0.26%	\$1,294	32
BWR, t > 1972	40	3.40%	6.96	4.51	0.26%	\$1,443	36
PWR, t > 1972	45	3.90%	8.07	5.51	0.26%	\$1,638	36
VVER-1000	45	4.40%	9.18	6.53	0.26%	\$1,848	41
EPR	50	4.20%	8.74	6.12	0.26%	\$1,779	36
AP1400	50	4.20%	8.74	6.12	0.26%	\$1,790	36
VVER-1200	50	4.50%	9.40	6.73	0.26%	\$1,907	38
ABWR	56	4.30%	8.96	6.32	0.26%	\$1,832	33

Calculations based on Eqs. (6)–(11).

Concluding remarks

Finally, there are some caveats regarding Table 1. China is not a member of the NEA, is an associate member of the IEA, and did not participate in the preparation of NEA/IEA (2015). Cost data was gathered by a consultant under contract to the NEA. Therefore, if China participates in the 2020 survey, these values could change. The Finnish expert submitted cost estimates that mixed values for the EPR under construction and the VVER under consideration. The original table in NEA/IEA (2015, pp. 48–49) included a French “EPR.” However, the experts that submitted these projected cost estimates did so for 2030 and hence were not comparable to projected cost estimates for 2020. The Slovak Republic experts submitted estimates for a “Generation II” LWR (NPP designs that were licensed before 1996), whereas all others submitted estimates for “Generation III” LWRs (NPP designs that were licensed after 1996). For these reasons the cost projections from Finland, France, and the Slovak Republic were not reproduced in Table 1. The Chinese data was included because it was similar to the Korean data.

The cost estimates in this article generally follow the GIF/EMWG (2007): *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (where “Generation IV” nuclear power plants are for non-light water reactors under research) and the implementation of these *Guidelines* in ORNL (2011). Where the *Guidelines* did not provide adequate detail, such as in the calculation of fuel costs, calculations following Rothwell (2016) were used. Table 1 summarizes levelized costs for a PWR 4-loop dual unit nuclear power plant (based on the costs of a System 80+) and compares these with the levelized costs of other nuclear power plants for China and OECD member countries. Rothwell (2021b) compares these with other electricity generating technologies.

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Economic Challenges of Nuclear Energy

Geoffrey S. Rothwell, Department of Economics (Retired), Stanford University, Stanford, CA, United States

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Glossary

AP1000 Advanced Pressurized reactor (1000 MWe, net) designed by Westinghouse; under construction in the United States, in operation in China.

BTU British Thermal Unit (= 1055 J), MMBTU is one million BTUs.

CRF Capital Recovery Factor is the (annual) rate at which capital expenditures financed with debt and equity capital are paid to banks and investors.

D&D Decontamination and Demolition (Decommissioning) is the process by which an operating facility site is returned to its original state before operations. This involves the decontamination of contaminated structures and equipment, the removal of equipment, and the demolition of structures. Under different regulatory regimes this leads to the end (“decommissioning”) of the operating license.

Debt Two forms of debt are (1) a loan from a financial institution and (2) sales of bonds to investors. Both involve repayment of principle and *interest*. Both forms are repaid before payments to equity providers in the case of bankruptcy (which involves the restructuring of assets and liabilities).

EPR European Pressurized Water Reactor (1650 MWe) designed by Areva/Orano; under construction in Finland, France, and the United Kingdom, in operation in China.

Equity At least two forms of equity are (1) funds provided by the owner/operator out of earnings and (2) funds provided by investors hoping for a *rate of return* from profits.

FOAK First-of-a-Kind technology refers to the set of early (“first”) units over which design and licensing costs can be recovered, justifying the construction of dedicated manufacturing facilities.

GIF Generation IV International Forum is a group of countries working together to develop advanced reactor technologies, known as “fourth generation” technologies.

IDC Interest During Construction is the amount of financing charges (interest and returns to equity) that accumulate during the construction of a facility.

Interest rate The rate of return required by banks or on bonds, see debt.

LC Levelized cost is the value that equates (1) the discounted costs of a facility over its lifetime, including expenditures during construction and decommissioning with (2) the discounted revenues of a facility’s lifetime.

LCOE Levelized cost of electricity is the lifetime levelized cost of a generated megawatt-hour.

Nominal Is the cost or price of a good or service in the year of currency when expended.

Real Is the nominal cost or price of a good or service inflated or deflated to a currency in a particular year.

WACC Weighted Average Cost of Capital is an average of the rate of *interest* on debt investments (e.g., by banks) and the *rate of return* required on equity investments (e.g., by partners) weighted by the percentage of debt and equity to total capitalization (debt + equity). Some authors use WACC as “the cost of money.”

Introduction

The calculation of the Levelized Cost of Electricity (LCOE) is a convenient context in which to examine challenges to the nuclear power industry. See Rothwell (2021a) for more details regarding the economics of nuclear energy. The LCOE is a measure of discounted lifetime costs averaged over discounted lifetime electricity revenues. LCOE can be defined as in NEA/IEA (2015, p. 28):

$$LCOE = \frac{\sum_t [(I_t + O\&M_t + F_t) \cdot (1 + r)^{-t}]}{\sum_t [E_t \cdot (1 + r)^{-t}]} \quad (t = -LT, \dots, T) \quad (1)$$

where I_t are capital investment expenditures (including decommissioning costs) in year t ; $O\&M_t$ are Operations and Maintenance expenditures (including refurbishments) in year t ; F_t is the product of fuel expenditures, F_{MWh} , and E_t in year t ; r is the cost of capital and $(1 + r)$ is the discount factor, e.g., $(1 + 5\%)$; E_t are the MWh of electricity generated in year t ; and LT is the construction lead time.

While there are “non-economic” challenges to nuclear power, such as public acceptance of the siting of nuclear power plants (NPPs) and related facilities, and the misunderstanding of the probability and consequences of NPP accidents, the economic challenges discussed in this chapter are (1) unrealistically high costs of capital imposed by risk-averse private financial markets and (2) the imposition of deregulated electricity markets that do not necessarily compensate the fixed costs of any electricity generation technology. With the introduction of electricity market deregulation and the absence of government support for commercial nuclear power in the European Union (EU), Japan, and the United States, the nuclear power industry in the “West” is giving way to the emergence of nuclear power industries in the “East”: Russia, China, and to some extent, the Republic of Korea.

This can be seen in the levelized cost estimates found in NEA/IEA (2015). Table 1 presents levelized costs for NPPs, where “variable” cost is equal to O&M plus Fuel and Waste costs. (Variable costs must be covered by the price.) The investment cost (construction cost plus financing cost) is the largest component of NPP LCOE. Investment costs are lowest in China and Korea. The O&M cost (which is dominated by the cost of labor, see NEA/IAEA, 2018) is less than the fuel cost for Chinese and Korean NPPs and greater than fuel cost for European and US NPPs. LCOE is lowest in China and Korea, about the same in the United States, Continental Europe, and Japan, and highest in the United Kingdom. (Russia did not participate in NEA/IEA, 2015; however, the NPP in Hungary is an EU-qualified Russian plant).

“The cost of capital to the nuclear sector” section of this chapter discusses problems associated with high costs of capital facing builders of NPPs. “Competition on the transmission grid: Regulated tariffs versus market pricing” section discusses the impact of electricity market liberalization on NPP owner/operators. “Summary of challenges” section discusses the contraction of the nuclear power industry in North America and Western Europe, and the intense competition mounted by the “state capitalist” countries of Russia and China.

The cost of capital to the nuclear sector

Levelized capital cost, LKC, is a function of total capital investment cost and the cost of capital:

$$LKC = KC \{ OC(MW) \cdot [1 + IDC(r, LT)] \} \cdot CRF(r, T) / E \quad (2)$$

where KC is the total capital investment cost, a function of OC (Rothwell, 2021a); OC is overnight cost, including contingency, a function of size, MW ; MW is the size of the plant in net megawatts-electric; IDC is the rate of Interest During Construction to finance KC , a function of r and LT ; r is the appropriate real cost of capital; LT is the construction lead time; T is the economic life of the facility; CRF is the capital recovery factor, a function of r and T , see Eq. (3); and E is the annual electricity output expressed in megawatt-hours (MWh).

The capital recovery factor (CRF) is defined as:

$$CRF = \left\{ r \cdot (1 + r)^T / [(1 + r)^T - 1] \right\} \quad (3)$$

Table 1 LCOE and cost components for nuclear power plants at a 5% discount rate (2018 USD).

Country		China	Korea	China	Belgium	United States	Hungary	Japan	United Kingdom
Type		CPR	APR	AP	Gen III	PWR	VER	ALWR	EPR
		1000	1400	1000		Gen III	1200		
Size	MWe	1080	1343	1250	1000	1144	1180	1152	1650
Overnight Cost	\$/kWe	1825	2041	2641	5132	4920	6277	3922	7092
Investment cost	\$/kWe	2175	2310	3147	6117	5858	7208	4673	8322
Investment + D&D	\$/MWh	\$15.13	\$16.08	\$21.90	\$42.50	\$43.70	\$50.59	\$32.54	\$57.87
O&M Cost	\$/MWh	\$6.57	\$9.75	\$7.39	\$13.69	\$19.06	\$10.50	\$27.70	\$23.68
Fuel + Waste cost	\$/MWh	\$9.42	\$8.67	\$9.42	\$10.56	\$7.40	\$9.70	\$14.29	\$11.42
Variable Cost	\$/MWh	\$15.99	\$18.41	\$16.73	\$24.25	\$26.46	\$20.20	\$42.00	\$35.11
Levelised Cost	\$/MWh	\$31.00	\$34.00	\$39.00	\$67.00	\$70.00	\$71.00	\$75.00	\$93.00

Based on data from NEA (Nuclear Energy Agency) (2018a) “Sustainable Development and the Application of Discounting to the Calculation of the Levelised Costs of Electricity,” Nuclear Technology Development and Economics, NEA/NDC/R(2018)1 (June). [http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDC/R\(2018\)1&docLanguage=En](http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDC/R(2018)1&docLanguage=En), NEA (Nuclear Energy Agency) (2018b) MDEP Annual Report 2018. Paris: OECD. <https://www.oecd-neo.org/mdep/annual-reports/mdep-annual-report-2017.pdf>, updated to 2018.

where T is the payback period on debt, for example, 30 or 40 years, and r is equal to the “debt and equity weighted average discount rate.” The higher the cost of capital and the lower the pay-back period, the higher the CRF. Traditionally, economists (particularly in the United States) have argued that the discount rate, r , should be the real corporate, tax-adjusted weighted average cost of capital (WACC): “The discount rate of 7.86% was used to calculate interest during construction (IDC)...” (MIT, 2018, p. 239). However, the tax rate varies significantly from one electric utility to another, so is ignored here:

$$r = \text{WACC} = [d \cdot \text{debt} / (\text{debt} + \text{equity})] + [e \cdot \text{equity} / (\text{debt} + \text{equity})] \quad (4)$$

where d is the real rate of return on corporate debt (bond) financing, debt is the total amount of corporate debt (borrowing), equity is the total “stock” financing, and e is the real rate of return on corporate equity financing.

In this discussion, a distinction must be made between nominal and real values. Nominal interest rates (and nominal rates of return on equity) are those in the currency of the year of the expenditure. Real interest rates are those in the currency of a specific year. (Interest rates are determined in debt markets; rates of return on equities are determined in “stock” markets.) This chapter uses inflation rates implied by the difference debt markets required to compensate for the risk of inflation.

Further, to understand the relationship between lead time and compounding IDC, consider capital construction expenditures, discounted to the beginning of commercial operation, i.e., when revenues from the project begin to flow:

$$\text{IDC} = \sum_t (\alpha_t \cdot \text{OC}) \cdot [(1 + m)^t - 1], (t = \text{LT}, \dots, 0) \quad (5)$$

where α_t are construction expenditures as a percent of OC in month t ; and m is the monthly cost of capital during construction, $(1 + m) = (1 + r)^{1/12}$.

In addition, the IDC factor, idc , is the percentage add-on to account for financing charges during construction. Because IDC depends on the construction expenditure rate (how much is spent in each month), Eq. (5) can be complicated, because the expenditure rate is not the same over the construction period, with smaller amounts being spent early to excavate the site, larger amounts being spent on equipment in the middle of the project, and smaller amounts being spent at the end on instrumentation, training, and fuel loading. Eq. (5) can be simplified by assuming that the construction expenditures have a uniform distribution, such that $\alpha_t = (1/\text{LT})$, i.e., overnight cost divided by construction lead time, LT, in months. Rothwell (2016, pp. 86-87) shows that idc can be approximated by the following equation:

$$\text{idc} \cong [(m/2) \cdot \text{LT}] + [(m^2/6) \cdot \text{LT}^2] \quad (6)$$

For example, if the construction lead time is 7 years and the annual discount rate is 5%, $\text{idc} = 19\%$. The longer the lead time and the higher the cost of capital, the greater the financing cost, if the lead time is 12 years and the discount rate is 7.86%, $\text{idc} = 59\%$, i.e., over one-half the overnight cost. The construction lead time and the real cost of capital have an enormous influence on the apparent competitiveness of nuclear power in the marketplace of competing electricity generating technologies. The Diablo Canyon NPP required 19 years to complete, the real IDC portion of the KC (known in regulatory accounting as “Allowance for Funds Used During Construction”) was 35%, implying that the real cost of capital was approximately 3.13% (Cox and Gilbert, 1991).

What is the appropriate real cost of capital for evaluating nuclear new build? First, what is the appropriate cost of debt? Many interest rates are tied to the rates on government bonds. The Federal Reserve Bank (the “Fed”) serves as the central bank in the United States. It sells government bonds with different maturities in an open (and international) market to finance government borrowing. The “90-day bond rate” serves as the foundation for other bonds sold by the Fed. In the finance literature this rate is known as the “risk-free” rate under the assumption that the US government will always pay its short-term debts. (During US federal government shutdowns, the risk of non-payment is greater than zero, hence the US government credit rating is not the highest possible.)

The “30-year bond rate” serves as the foundation for rates on 30-year mortgages, as well as infrastructure investments. The Fed sells two types of 30-year bonds: (1) one that yields an annual nominal interest payment and (2) one that yields an annual payment covering changes in the value of the interest payment due to inflation: Treasury Inflation Protected Securities (TIPS). The difference between these rates that buyers are willing to buy bonds is the expected (general) rate of inflation. In Fig. 1 bonds yielding a nominal interest payment varied from the end of 2014 to the beginning of 2020 between 2.1% and 3.3%. Bonds yielding an inflation-adjusted interest payment varied between 0.5% and 1.3%. This implied that the expected (general) annual inflation rate varied between 1.6% and 2.1%, i.e., less than that experienced by US producers from 2000 to 2018, but greater than from 2008 to 2018.

Second, the next issue is determining the appropriate risk premium to add to the “risk-free” rate. Generally, this is a function of an entity’s credit rating. There are many credit-rating services, for example, Standard & Poor’s (SP Global, 2019) and Moody’s Investor Services. Usually, the lower the credit rating, the higher the required yield that must be offered to sell bonds to investors. For example, the highest rated country, Switzerland, has a credit rating of AAA and can sell its bonds at -0.95% , i.e., investors must pay the Swiss government almost 1% to be insured that their principle will be paid back to them. On the other hand, Brazil has the lowest credit rating in Table 2 and must pay investors a return of approximately 7.26%.

Third, to give context to the relationship between Wall Street and the nuclear power industry, in the 1970s, with double-digit inflation and associated increases in the cost of capital of over 20% a year, capital intensive nuclear plants were cancelled. After the accident at Three Mile Island on 28 March 1979, the time taken to license NPPs doubled, leading to even more nuclear plant cancellations. In 1983, the Washington Public Power Supply System defaulted on \$2.25 billion in bonds (\$4.5 billion in 2018 US dollars)—something that Wall Street has neither forgotten nor forgiven. Therefore, Wall Street bond rating agencies have been very risk averse regarding the financing of NPP construction. Regarding the required rate of return on electric utility equity, this is more

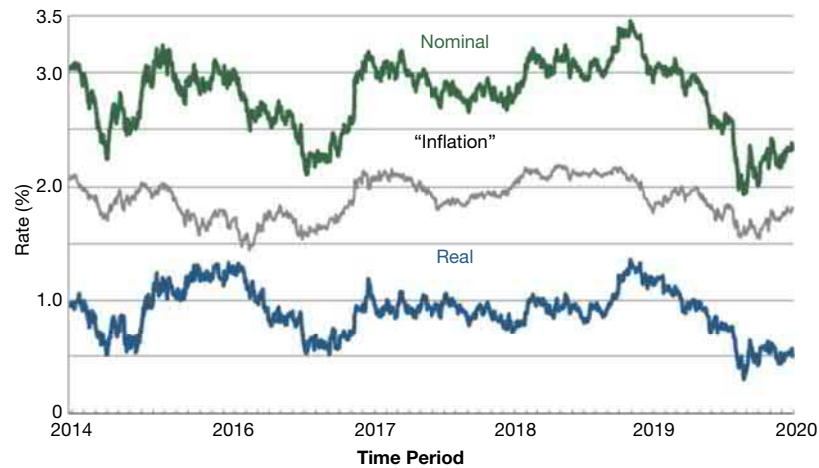


Fig. 1 Nominal and real rates on (US) debt: 23 October 2014 to 1 January 2020. Based on data from US Treasury (2018), “Resource Center: Historic yield curve chart,” <https://www.treasury.gov/resource-center/data-chart-center/interest-rates/Pages/Historic-LongTerm-Rate-Data-Visualization.aspx> and <https://www.treasury.gov/resource-center/data-chart-center/interest-rates/Pages/TextView.aspx?data=yieldYear&year=2019>.

Table 2 Bond ratings, bond interest rates, and inflation (mid-2019).

Country	S&P Rating (see 1)	Moody's Rating (see 1)	Nominal yield (26 AUG 2019) (2)	GDP Deflator (3,4)	Real yield
Switzerland	AAA	Aaa	−0.95%	0.2%	−1.15%
United States	AA+	Aaa	0.57%	1.6%	−1.01%
France	AA	Aa2	−0.37%	1.2%	−1.55%
Korea	AA	Aa2	1.19%	0.4%	0.79%
United Kingdom	AA	Aa2	1.87%	2.1%	−0.23%
China	A+	A1	3.08%	1.4%	1.66%
Japan	A+	A1	−0.27%	0.2%	−0.47%
Russia	BBB-	Baa3	7.11%	8.5%	−1.31%
Brazil	BB-	Ba2	7.26%	3.7%	3.43%

Based on NEA (Nuclear Energy Agency) and International Energy Agency (IEA) (2015) *Projected Costs of Generating Electricity, 2015 Update*. Paris: OECD. <https://www.oecd-neo.org/ndd/pubs/2015/7057-proj-costs-electricity-2015.pdf>, updated with data from (1) https://en.wikipedia.org/wiki/List_of_countries_by_credit_rating, (2) <https://tradingeconomics.com/bonds>, (3) <http://www.oecd.org/economy/economic-outlook/>, and (4) <https://www.ceicdata.com/en/indicator/russia/gdp-deflator-growth>.

difficult to determine given the strategic (stock) market implications of such information. A second-hand source (Utility Dive, 2018) announced,

A new report from credit ratings agency Moody's Investor Services paints a generally stable picture of utility holding company finances, though that is heavily dependent on the extent to which revenues are generated by regulated operations... While the “vast majority” of North American investor-owned utility holding companies are rated Baa, the agency concluded Exelon and Public Service Enterprise Group have the highest levels of operating risk due to their generation assets in competitive markets... The firm said Exelon's credit profile is constrained by its higher-risk merchant power subsidiary, Exelon Generation Co., which owns unregulated nuclear generation facilities and a retail energy trading and marketing business.

Therefore, US electric utility debt and equity are considered about as risky as Russian government bonds, which had a *nominal* yield of 7.11% (in August 2019). This is unlikely to be the appropriate *nominal* weighted cost of capital for NPP owner/operators. Fig. 2 shows the relationship between the WACC and the nuclear's LCOE, primarily a function of the levelized capital investment cost in Eq. (2). For example, assuming a WACC of 7.86% leads to a nuclear LCOE of \$107/MWh, much higher than any in Table 1.

The consequences of using such high WACCs are several. First, while WACCs higher than 5% might reflect the risk of building an NPP, they do not reflect the riskiness of producing electricity revenues, unless deregulation has been imposed *after* the NPP has been built, see next section. Second, there is an implicit assumption that sinking funds, for example, for D&D costs and waste management, are earning rates of return equal to the WACC charged in the market for construction. If these costs are increasing at real rates higher than the general inflation rate (due to the increases in regulation and lack of storage sites), there will be less put aside during operation

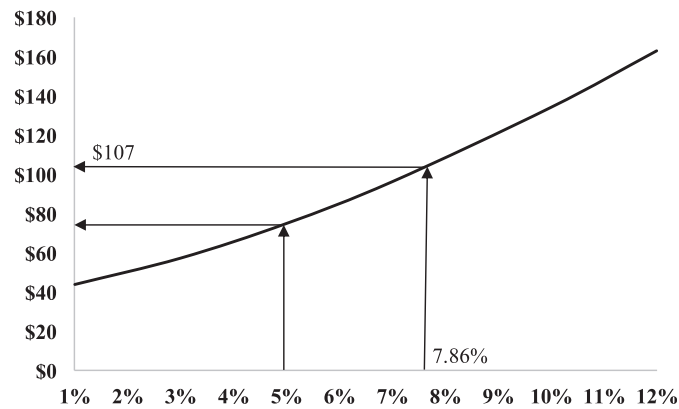


Fig. 2 Nuclear energy LCOE as a function of the discount rate. Author calculations based on Eq. (2).

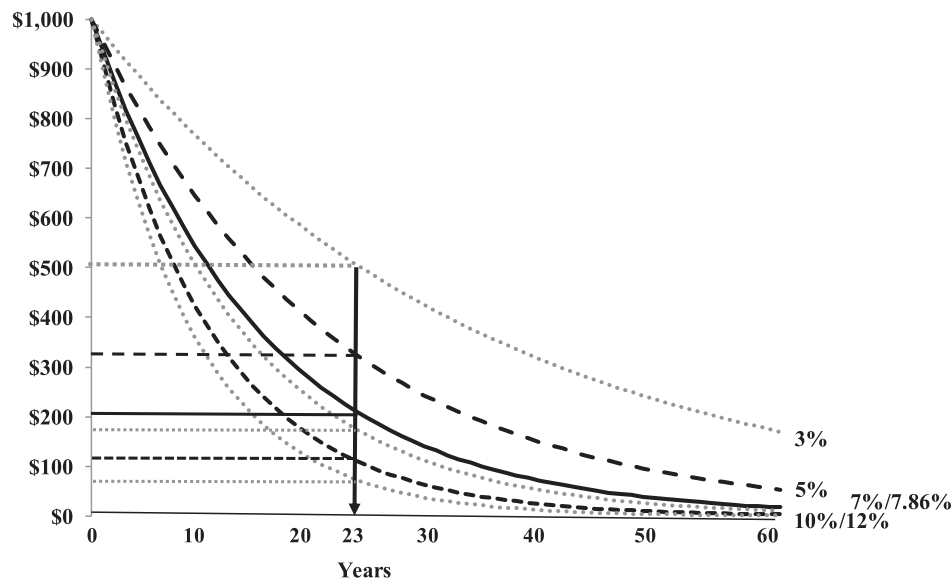


Fig. 3 The present value of benefits as a function of the discount rate. Based on NEA (Nuclear Energy Agency) (2018a). “Sustainable Development and the Application of Discounting to the Calculation of the Levelised Costs of Electricity,” Nuclear Technology Development and Economics, NEA/NDC/R(2018)1 (June). [http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDC/R\(2018\)1&docLanguage=En](http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDC/R(2018)1&docLanguage=En), NEA (Nuclear Energy Agency) (2018b) *MDEP Annual Report 2018*. Paris: OECD. <https://www.oecd-neia.org/mdep/annual-reports/mdep-annual-report-2017.pdf>, Fig. 2.

than will be required. Fig. 3 shows after revenues received 23 years (the approximate length of a human “generation”) at a discount rate of 3%, are only worth half as much as revenues in the present (NEA, 2018a). At a discount rate of 7.86% the present value of revenues after 23 years is only worth 17.5% of revenues generated in the present. For long-life assets, such as NPPs, at discount rates above 7% the value of the assets is next to nothing for the children and grandchildren of the builders of power plants in the present. This contradicts the goal of sustainability. For these reasons, the levelized cost methodology requires different discount rates during different periods in the life cycle of a long-term asset. For example, a more sustainable-oriented approach would use the corporate cost of capital to discount expenses during construction and use the social discount rate to discount revenues (Karoly, 2018) during operation.

Thus, determining the appropriate rate for discounting costs and benefits over multiple lifetimes, such as one finds when evaluating investments in the electricity generation and distribution system, is objectively difficult. Therefore, one of the most difficult economic challenges facing the nuclear power industries in countries with liberal capitalist economies is to find financing for new NPP construction due to its perceived riskiness, particularly the risks associated with building on time and on budget. Many electric utilities in these countries have turned to either (1) their governments for aid, such as loan guarantees (Rothwell, 2016, Chapter 2) or (2) foreign financing.

Competition on the transmission grid: Regulated tariffs versus market pricing

Both “old” and “new” nuclear power compete with “old” and “new” fossil-fired electricity generators, as well as “new” renewable generators (Rothwell, 2021b). Because of greenhouse gas concerns, many countries have adopted policies that encourage the

phasing out of old generators in favor of new generators. For example, old coal plants are being replaced with new CCGTs (Combined Cycle Gas Turbines) that produce half the CO₂ of coal-fired generators and little of the other pollutants that coal burning produces. At the same time, renewables with high capital costs and low capacity factors are being given priority in electricity markets, as well as subsidies, such as tax breaks. This section compares the LCOEs of various electricity generating alternatives. Following the assumptions in [US EIA \(2019\)](#) and in [NEA/IEA \(2015\)](#), [Table 3](#) compares levelized cost of (“Advanced”) CCGT, (“Advanced”) Nuclear, (“On-shore”) Wind, and (“Fixed-tilt”) Solar PV as defined in [US EIA \(2019\)](#), supplemented with assumptions from [US EIA \(2016\)](#) and [NEA/IEA \(2015\)](#), assuming a weighted cost of capital of 5%.

For the United States (only), assumptions in [NEA/IEA \(2015\)](#) are updated using the assumptions in [US EIA \(2019\)](#). The CCGT alternative includes a carbon fee based on \$30.00/tCO₂ and the nuclear alternative includes an irradiated fuel fee of \$2.33/MWh (for storage and disposal). At \$5.50 per MMBTU (\$5.21 per Gigajoule), CCGT is the cheapest alternative. The levelized cost of nuclear and onshore wind are approximately the same. Solar PV is significantly higher due to its low capacity factor. These levelized costs are based on published assumptions and can be updated with updates of the relevant publications. These results depend on the price of natural gas ([Federal Reserve Bank of St. Louis, 2020](#)).

Comparing these results with the calculations in [Tables 1 and 3](#), new nuclear was competitive with natural gas CCGT between 2000 and 2008. (Note: the projected costs of new nuclear in [Table 3](#) are slightly different than in [Table 1](#) due to different assumptions.) However, new nuclear cannot compete with natural gas prices below \$4/MMBTU (\$3.79/Gigajoule) in deregulated markets in the United States: it is unlikely that revenues would cover the fixed costs of NPP construction. See [Table 3](#). Even if there were carbon taxes of \$30/tCO₂, new nuclear would not be competitive outside of China and Korea, see [Table 1](#).

Although most CCGT generators have long-term contracts, the opportunity cost of using their contracted natural gas for electricity generation is the spot price. For this reason, this price is used to convert the price of natural gas into the price of electricity as in [Fig. 4](#). This is used to calculate the “spark spread” ([US EIA \(Energy Information Administration\), 2013](#)): “The spark spread is the difference between the price received by a generator for electricity produced and the cost of the natural gas needed to produce that electricity.” To understand the influence of natural gas prices, [Fig. 5](#) presents “CCGT electricity prices” as a function of the monthly Henry-Hub spot-market prices in Louisiana from 1994 to 2020 in 2018 USD. If existing nuclear variable costs are similar to new nuclear variable costs (i.e., construction costs have been paid off for plants older than 30 years), existing NPPs could be competitive with CCGT with natural gas prices above \$4/MMBTU (\$3.79/Gigajoule). However, with natural gas bouncing between \$2 and \$4/MMBTU (\$1.90 and \$3.79/Gigajoule), it would be difficult to justify investment in upgrades, uprates, or life extensions without the certainty that these investments could be recovered. On the other hand, natural gas prices have generally been twice as high in Europe and three times as high in Japan, see [Knoema \(2019\)](#). Therefore, it is more likely that incremental investments and new nuclear power could be justified in Europe assuming rising natural gas prices and carbon taxes.

Table 3 Levelised cost for four electricity generation technologies, 2018 US\$, $r = 5\%$.

Input		CCGT	Nuclear	Wind	Solar PV
Levelized Capital Cost + D&D	\$/MWh	\$7.45	\$49.57	\$50.48	\$88.74
Levelized O&M Cost	\$/MWh	\$3.44	\$16.24	\$19.73	\$18.30
Levelized Fuel	\$/MWh	\$34.10	\$9.33	\$0.00	\$0.00
Levelized Fuel CO ₂ Fee	\$/MWh	\$9.89	\$0.00	\$0.00	\$0.00
Levelized Cost without waste fee	\$/MWh	\$45.00	\$72.80	\$70.20	\$107.04
Levelized Cost with waste fee	\$/MWh	\$54.89	\$75.13	\$70.20	\$107.04

Based on Rothwell GS (2021b) “Economics of Nuclear Power Versus Other Energy Sources,” Chapter 01305, this Encyclopedia.

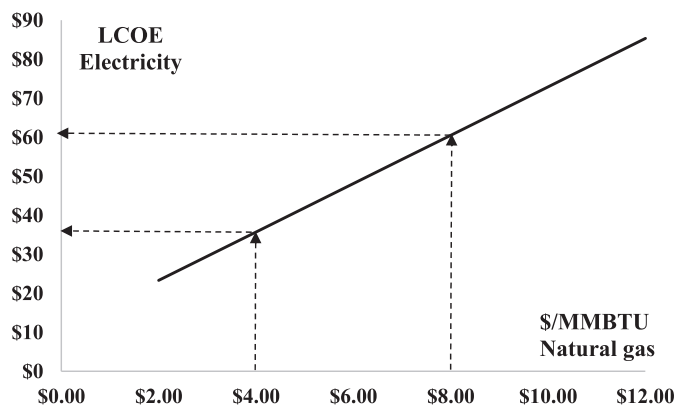


Fig. 4 Levelised cost for new natural gas generation, 2018 US\$. Based on data from Rothwell GS (2021b) “Economics of Nuclear Power Versus Other Energy Sources,” Chapter 01305, this Encyclopedia, Table 9.

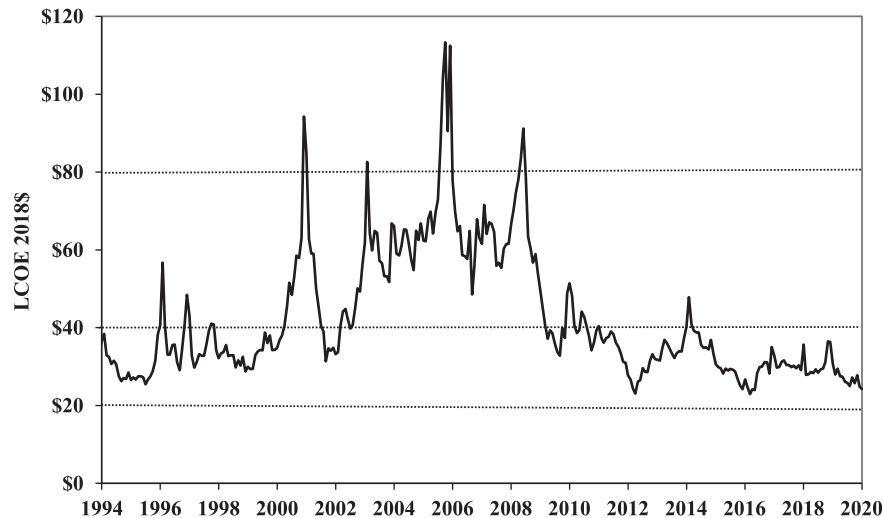


Fig. 5 Cost of CCGT electricity as a function of the price of natural gas in year, 1994–2020. Calculations based on Fig. 4 with data from Federal Reserve Bank of St. Louis (2020) “Henry Hub Natural Gas Spot Price”. <https://fred.stlouisfed.org/series/DHHNGSP>.

With US states’ experience in deregulating tariffs in many industries and with increasing construction of natural gas CCGTs in the “dash for gas” in the 1990s state legislatures directed public utility commissions to create markets to set prices in the electric power industry. This involved breaking the electric industry into (1) generation, (2) transmission, and (3) distribution sectors. Because strong network economies, transmission was determined to be a “natural monopoly,” and remained under control of the state-regulated utility (Rothwell and Gomez, 2003, p. 75).

Under rate-of-return regulation generation, dispatch was a function of average cost, similar to levelized cost. Under a regulated monopoly, compensating electric utilities based on the load factor of their plant (in particular those not owned by the utility) is equivalent to paying each generator its technology specific (fixed) capacity and (variable) energy costs. This can be observed on the left-hand side of Fig. 6, where three different generation technologies, GN, are dispatched to cover a load duration curve (Oren, 2000). The generator with the highest fixed costs and the lowest variable cost, GN₃ (a base-load generator), receives its fixed, F_3 , and variable costs, C_3 , for producing energy: $[F_3 + C_3 \cdot (T_1 + T_2 + T_3)]$, where $(T_1 + T_2 + T_3)$ is its total generating time. Further, the generator with the lowest fixed cost and the highest variable cost, GN₁ (a peaking generator), receives its fixed and variable costs during the period it is producing energy, $F_1 + (C_1 \cdot T_1)$, where T_1 is its total generating time. In this way generation revenues cover the fixed and variable costs of each technology.

However, in a competitive generation market, in each period electricity generators receive the highest bid in each period, assumed to be equal to the variable cost of the most expensive (highest cost) generator dispatched in that period. On the right-hand side of Fig. 6, hours with higher load factors, the marginal energy cost (market price) is equal to the lowest variable generation cost, C_3 . In hours with lower load factors, the marginal energy cost is equal to the highest variable generation cost, C_1 . Therefore, generators with the highest load factors will receive $(C_1 \cdot T_1 + C_2 \cdot T_2 + C_3 \cdot T_3)$ and generators with the lowest load factors will receive $(C_1 \cdot T_1)$. The sum of the marginal cost times the duration during which it applies $(\sum C_j \cdot T_j)$ is equal to *only* the variable costs of producing electricity, albeit at the bid (variable cost) of the last generator dispatched. Thus, if generators are paid a uniform average variable cost for their energy, they can end up with a shortfall in covering the fixed cost of their technology. This is the reason that some systems have introduced an explicit *capacity payment* (here, equal to F_1) to generators to pay generators’ fixed cost, e.g., in MISO (see below).

In states with deregulated generation, markets were established in which generators would, for example, place day-ahead bids for each period (e.g., each half hour) and the market operator would dispatch the lowest bidder first, the second lowest bidder next, etc., until demand in each period was met. The price in each period was determined by the highest bid. Bids were assumed to be the marginal (average variable) cost of electricity provision (e.g., fuel costs), with capital costs covered under the highest bid. Further, renewables, in most cases, have been given “must-run” status, e.g., whenever the sun is shining, solar generators are dispatched, displacing other bidders. Also, but independently, consumers were given the right to choose their distributor, who, in turn, purchased rights to future electricity for their customers. Given that generation and retail markets were established on a state-by-state basis, the electricity industry in the United States became a complicated patchwork of overlapping jurisdictions, Kulkarni (2016).

The US Federal Energy Regulatory Commission (US FERC, 2020), describes 10 electricity markets in the United States where many south-eastern and western states of the United States are still rate-of-return regulated within the state and compete in interstate markets:

CAISO California Independent System Operator “operates a competitive wholesale electricity market in California and manages the reliability of its transmission grid.”

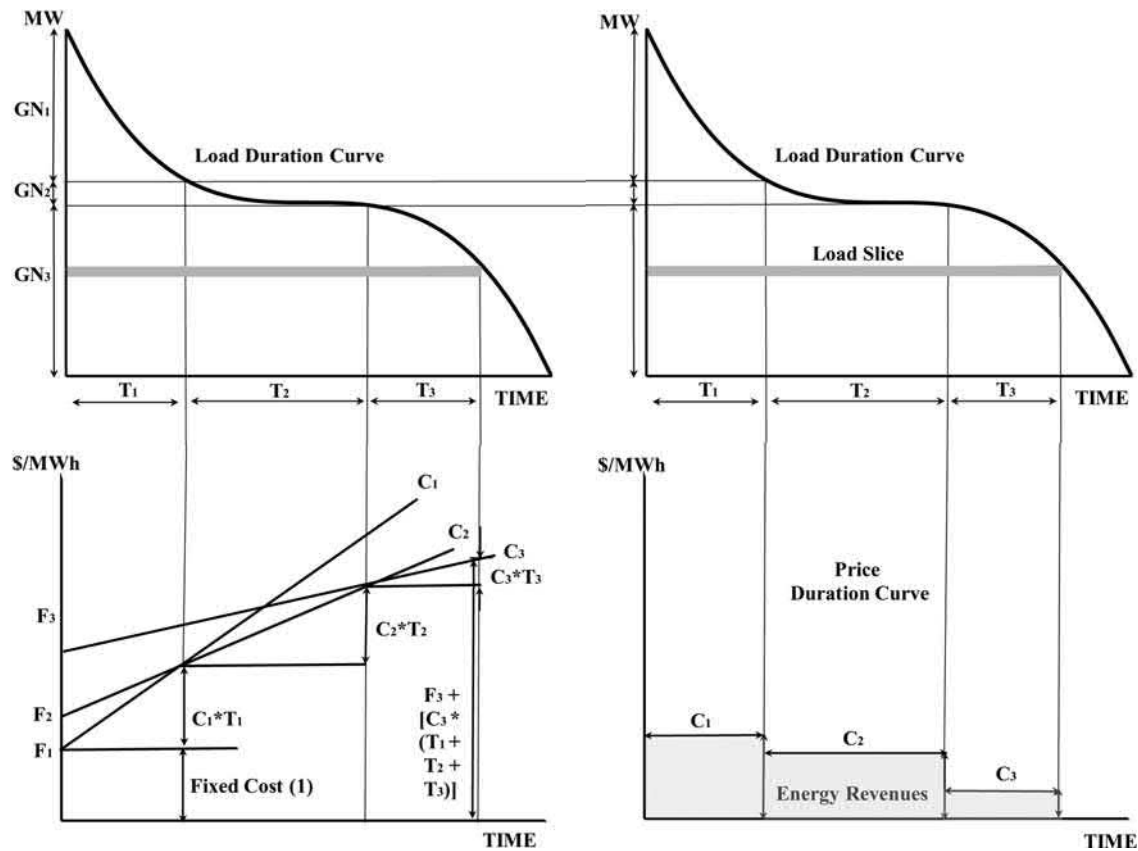


Fig. 6 Load slice and marginal cost pricing. Based on Oren S (2000) "Capacity Payments and Supply Adequacy in Competitive Electricity Markets," presented at the VII Symposium of Specialists in Electric Operational and Expansion Planning (SEPOPE), May 21–26, Curitiba, Brazil.

ERCOT Electricity Reliability Council of Texas coordinates approximately 90% of Texas' electrical load.

MISO Midcontinent Independent System Operator "operates the transmission system and a centrally dispatched market in portions of 15 states in the Midwest and the South, extending from Michigan and Indiana to Montana and from the Canadian border to" Louisiana and Mississippi.

ISO-NE Independent System Operator-New England "serves six New England states: Connecticut, Maine, Massachusetts, New Hampshire, Rhode Island, and Vermont."

NYISO New York Independent System Operator covers all of the state of New York.

Northwest Is "composed of all or major portions of the states of Washington, Oregon, Idaho, Wyoming, Montana, Nevada, Utah, a small portion of Northern California" and the Canadian provinces of British Columbia and Alberta.

PJM Pennsylvania, New Jersey and Maryland "centrally dispatches generation and coordinates wholesale electricity in all or part of 13 states."

Southeast "Includes all or parts of Florida, Georgia, Alabama, Mississippi, North Carolina, Missouri, and Tennessee."

Southwest "Encompasses the Arizona, New Mexico, southern Nevada, and the Rocky Mountain Power Area."

SPP Southwest Power Pool manages transmission in 14 states.

To further complicate the choice between generation technologies, the cost of natural gas has appeared to be "too cheap to avoid" in North America. Most analysts assume that the price of natural gas will continue to be low (below \$4/MMBTU, \$3.79/Gigajoule) for the foreseeable future. With the market price of electricity driven by natural gas prices, consumers must absorb more of the risk of electricity price spikes because the electric utilities have invested so heavily in CCGT capacity. For example, if natural gas pipelines become congested, the price of electricity can climb above \$1000/MWh. As GTM (2019) stated, "Last week as Texas' ERCOT grid reached its price cap of \$9,000 per megawatt-hour and the price map on ERCOT's website became a solid and deep red, many energy market wonks highlighted that this is a feature, not a failure, of the market."

Because deregulation schemes as implemented in the United Kingdom, the United States, and, to some extent in the EU, favored low capital cost (with high fuel costs) electricity generators, as well as schemes to favor renewables (to some extent to discourage high fossil fuel usage) with must-run status, NPPs are losing market share. As long as there are no payments to capacity and subsidies to renewables, nuclear power faces an almost insurmountable economic challenge in the face of market liberalization while providing the highest share of non-greenhouse gas electricity in many hydro-starved countries with few energy storage

resources. (Future energy storage technologies for nuclear heat could increase the complementarity of nuclear power with non-dispatchable renewables, see [Forsberg, 2021](#)).

Summary of challenges

To conclude there are two primary problems facing the NPP construction and operation industries: (1) extraordinarily high costs of capital on capital intensive NPP projects, and (2) the deregulation of electricity markets that favors fuel intensive generators and “must-run” renewables. Two related issues are (1) overcoming problems associated with maintaining a viable industrial supply chain, allowing the construction of new nuclear on time and on budget, and (2) public perceptions of perceived safety issues after the Fukushima Daiichi accident and the little understood management of radioactive wastes and irradiated fuel. Unless these synergistic issues are resolved the NPP manufacturing industry could end up becoming more concentrated in China and Russia during the remainder of the 21st century where governments are actively supporting NPP construction and the maintenance of a viable nuclear supply chain and actively discouraging dissent over nuclear power policy.

Because of the economies of scale in constructing NPPs, without new orders the nuclear supply chain is disintegrating in the “West.” The cost of constructing new NPPs has increased dramatically since the financial crisis of 2007–08 and the accident at Fukushima Daiichi in 2011, which saw the cancellation of many new orders and the announced closure of NPPs in Germany. With these cancellations and closures came a slowdown in construction of NPPs in the United States (e.g., AP1000) and Europe (e.g., EPR) and the increase in construction costs, leading to more cancellations. While Russia is exporting a dozen VVERs, the cost of construction outside Russia is just below the bids of American and European suppliers and includes up to 80% financing. China hopes to export its Hualong One (HPR1000) NPP, now being constructed in China. Further, Korea has been building the APR1400 in the United Arab Emirates, but is having a difficult time finding Korean government support for building more APR1400s in Korea.

All NPP manufacturers are attempting to market standardized plants, but the regulations in different countries require local amendments to standard designs. The MDEP (Multinational Design Evaluation Programme) is attempting (1) to enhance multi-lateral cooperation within existing regulatory frameworks; (2) to encourage multinational convergence of codes, standards, and safety goals; and (3) to implement the MDEP products to facilitate the licensing of new reactors; [NEA \(2018b\)](#). However, its working groups are focused on the EPR, AP1000, APR1400, VVER, and HPR1000, i.e., no BWR or PHWR NPPs. Should MDEP or a similar body standardize the regulations for LWRs, the supply chains could intertwine, allowing for increased reliance on more international suppliers.

While NPPs are becoming more efficient with increases in capacity factors, the staff size at NPPs is not declining. Because of the high number of staff required to respond to regulatory (e.g., safety) requirements, there are high economies of scale: fewer staff are required per kW with larger plants than with smaller plants ([Rothwell, 2016](#), p. 167, where the minimum staff size of any NPP is estimated to be about 250; also see [NEA/IAEA, 2018](#), p. 31). Compare this to an average staff size at a CCGT of about 3 dozen. [IEEE \(2017\)](#):

On August 1, Michigan-based DTE Energy revealed plans to spend almost \$1 billion to build a 1,100-megawatt gas-fired power plant. When the station enters service in 2022, it will replace three existing coal-fired units that currently employ more than 500 people. Job openings at the new gas-fired plant? Thirty-five full-time employees, says a DTE spokesman.

If NPPs employ 10 times as many people as CCGTs, NPPs might never be competitive with CCGTs as CCGTs become even more automated. Perhaps with integrated PWRs staff sizes will be smaller.

Finally, there appear to be no supply problems with the front-end of the nuclear fuel cycle. Although little uranium is being mined in the United States and there is no endogenous enrichment technology, there are well-functioning markets for both uranium and uranium enrichment services. On the other hand, the public perceives a back-end “waste” problem: the typical 1000 MWe NPP produces about 20 t (see [Rothwell, 2021a](#)) of irradiated nuclear fuel that can be stored in dry casks ([Kessler, 2021](#)). However, the typical CCGT produces 330 kg/MWh, or millions of tonnes of CO₂ per year (as a function of the “carbon intensity factor,” see [US EIA, 2016](#)) with no competitively priced “carbon capture and storage” technology in sight. Therefore, there appears to be a public perception/relations problem that must be overcome for the public to accept nuclear new build. (On other aspects of public opinion, see [Bisconti, 2021](#).)

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See Also: Agriculture: Improving Crop Production; New Reactor Concepts Deployment Challenges and Opportunities; Water Cooled Small Modular Reactors (Integral PWR and BWR).

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Fuel Rod, Fuel Assembly, and Reactivity Control Assembly Design

Michael Y. Young*, Westinghouse Electric Company, Pittsburgh, PA, United States

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Glossary

AOO anticipated operational occurrence Events and transients that are expected to occur in a reactor during its operating lifetime (e.g., reactor shutdown, power change, refueling)

Burnup A measure of the amount of fission energy released from a given mass of uranium (heavy metal). Unit is Megawatt-days per kilogram (MWd kg⁻¹), or Gigawatt-days per ton (GWd ton⁻¹). Also referred to as exposure.

BWR Boiling water reactor, a type of light water reactor in which the water moderator in the core is boiling (Hamon, 2021).

CANDU A heavy water power reactor design (Canadian Deuterium-Uranium).

CHF, DO Critical heat flux, dryout. Used to describe the boiling crisis in a BWR fuel assembly.

Cladding The structural material that separates the nuclear fuel from the coolant. In fast reactors it is usually made of steel. In light water reactors it is usually made of an alloy of Zirconium.

Control blade The component used for reactivity control and reactor shutdown in a BWR.

Crud A material that deposits on surfaces exposed to LWR water, becomes radioactive and forms an adherent layer on fuel cladding.

DNB Departure from nucleate boiling. Used to describe the boiling crisis in a PWR fuel assembly.

Fuel assembly A structure containing a group of fuel rods arranged in a regular pattern and supported by other components for operation in the core.

Gen II reactors Refers to reactor designs developed and constructed during the 1970s and 1990s. Many are in their final years of operation.

Gen III reactors Refers to reactor designs developed in the 1990s. A small number have been completed and are operating today.

GDC General design criteria.

GTRF Grid-to-rod fretting. Wear caused by relative motion between the fuel assembly grid and the fuel rod.

HWR Heavy water reactor. This type of reactor uses heavy water as the moderator (Nichita, 2021).

*Retired.

LOCA Loss of coolant accident. This is one of the types of hypothetical accidents that must be considered in the design of a reactor.

LWR Light water reactor. Refers to the class of reactors that use water as the moderator and coolant. PWRs and BWRs belong to this class.

Moderator Substance used to slow neutrons generated by fission for efficient energy transfer.

Neutron flux Number of neutrons crossing a unit area per second. Reaction rates, such as the fission rate, are proportional to the neutron flux.

PCI Pellet-cladding interaction. Refers to several types of interaction between the fuel pellet and the cladding. Interactions include contact, sliding, chemical reaction, etc.

PCMI Pellet-cladding mechanical interaction. Refers specifically to mechanical types of interactions between the fuel pellet and the cladding (e.g., friction, contact, etc.).

PWR Pressurized water reactor. A type of light water reactor that uses pressurized water as the moderator and coolant. The water is maintained below its boiling point. (Brown, 2021).

RCA Reactivity control assembly, generic term for components used to control reactivity.

RCCA Rod cluster control assembly, used for reactivity control and reactor shutdown in a PWR.

Reactivity The extent of the deviation in the core effective multiplication factor (usually denoted as k_{eff}) from its value at a just critical state (at which state $k_{eff} = 1.0$ and the fission rate is constant). A positive (negative) reactivity implies that the ratio of the rate of neutron production to the rate of neutron losses (k_{eff}) is larger (smaller) than 1.0 due to a change in the condition of the core.

SAFDL Specified acceptable design limit. Specified limits have been established by regulation for various calculated conditions in the reactor (see “Fuel assembly and RCA design considerations” in this article).

SCC Stress-corrosion cracking. A material degradation process that involves chemical corrosion processes in conjunction with applied tensile stresses that open small cracks in the material.

USNRC United States Nuclear Regulatory Commission. This is the regulatory body that controls all aspects of nuclear safety in the United States.

Zirconium A transition metal with low neutron absorption, alloys of which have good strength, thermal and corrosion properties in reactor water chemistries.

Introduction

The evolution of modern water reactor nuclear fuel for commercial power production began with the successful design and operation of compact nuclear reactors for submarines, in 1953. Details of the subsequent history of nuclear reactor development are provided in other articles in this encyclopedia (Brown, 2021; Hamon, 2021; Maldonado, 2021). In the United States today, pressurized water reactors (PWR) and boiling water reactors (BWR), both with light water as the moderator, are the primary reactor types. They also operate in several countries worldwide. These are also grouped under the broader category of light water reactor (LWR). Heavy water power reactors (HWR), specifically the CANDU design, are operating in Canada as well as a few other countries. The fuel designs for these three types will be described below.

The theoretical and experimental work underpinning the reactor and fuel designs described here is contained in a large body of published work, of which only a small fraction can be cited here. For general physics, design and operation of power reactors and their fuel, the books by Glasstone and Sesonske (1980) and Olander (2001) are suggested.

Fuel rod, fuel assembly and reactivity control assembly configuration

PWR and BWR

Fuel rod and assembly configurations continued to develop through the 1960s, as a balance was sought between efficient and safe neutron production and utilization (at the simplest level, the ratio of moderator volume to uranium volume) and efficient and safe transfer of nuclear heat to steam production (heat transfer surface area, etc.). For additional details on the nuclear design of a reactor core. Typically, the addition of more, thinner fuel rods in each fuel assembly resulted in more efficient heat transfer and lower temperatures in the fuel. This evolution resulted in a fairly standard configuration appearing in most of Gen II reactor designs. This standard configuration also appears, with some modifications, in the Gen III reactors that have been developed. The detailed descriptions below apply to standard Gen II Westinghouse PWRs, General Electric BWRs, and the CANDU.

A typical PWR or BWR fuel rod is shown in Fig. 1. The fuel rod consists of a zirconium alloy tube (normally referred to as “cladding”) approximately 3.65 m (12 ft) long, with a diameter of about 1 cm, filled with a pressurized inert gas to approximately 1 MPa (150 psia), sealed at each end with end caps or plugs, and containing a stack of pellets, each of which are approximately 1 cm long.

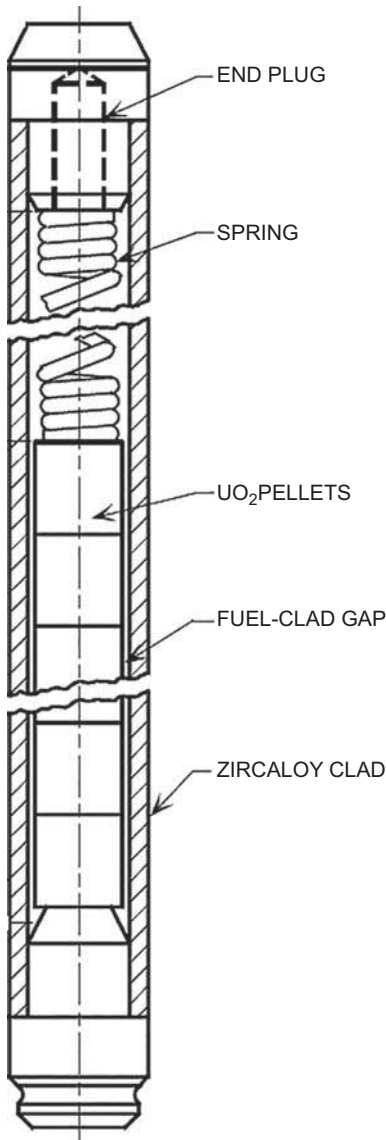


Fig. 1 Typical PWR fuel rod.

The pellets are comprised of slightly enriched (in U^{235}) uranium dioxide, with, in some cases, additional compounds for reactivity control (see “**Neutron absorber materials**” in this article). To allow for some movement of the pellets, their diameter is slightly smaller (a few dozen microns) than the tube inner diameter. The stack is held down by a spring to prevent movement during transportation. It should be noted that the uniform stack of pellets depicted in the figure is an idealization; the pellets will actually be randomly offset, with most pellets touching the cladding at some point. PWR and BWR fuel rods are very similar, except that in a BWR some of the rods may be shorter in length to improve nuclear characteristics. As will be seen, CANDU fuel rods are also similar in design, except that they use natural uranium and are significantly shorter.

Fig. 2 shows the key components of a PWR fuel assembly, with a reactivity control assembly inserted (in PWRs these assemblies are normally called rod cluster control assemblies, or RCCAs). **Fig. 3** shows a top down view of four assemblies. The number of assemblies in a typical PWR core ranges from 121 to 241, depending on the size of the reactor.

The fuel assembly consists of a “skeleton” or “cage” designed to hold the fuel rods in a fixed, typically square lattice (some Russian reactor types employ a hexagonal lattice), as shown in **Fig. 3**. There is no barrier or channel wall between the assemblies; coolant resides in the space between the assemblies. The assemblies shown contain 264 fuel rods in a 17 by 17 square lattice, with the remaining lattice locations taken up by control rod guide tubes or instrument tubes. Other designs exist with different numbers of fuel rods and control rod guide tubes. The assembly structure consists of a number of tubes (guide thimbles) to which grids made of interlocking stamped sheet metal straps are attached (the grid may be welded to the tubes or attached by bulge joints as shown in **Fig. 2**). These grids will hold the fuel rods in place. There are typically six to seven such grids, for a span length of about 55 cm

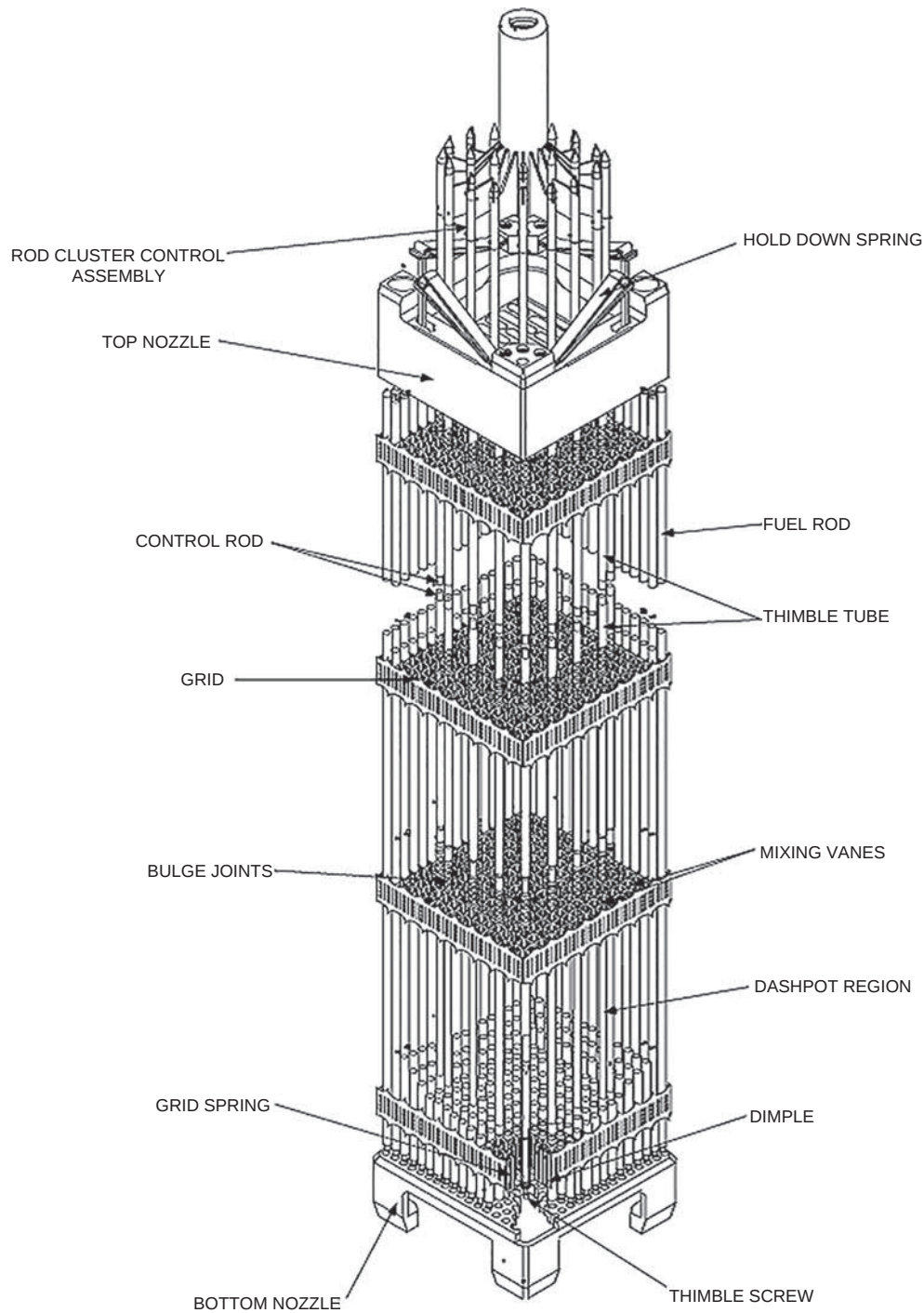


Fig. 2 Westinghouse PWR fuel assembly.

(22 in.) between grids. In some designs, additional “mixing” grids may be added in the upper spans, to promote additional coolant mixing and heat transfer by inducing lateral momentum to the coolant flow. To accommodate axial growth of the fuel rod due to thermal expansion and radiation effects, as well as ease of manufacture, the support of the fuel rod is usually accomplished by the use of flexible springs attached (or formed by stamping) to the grids.

The tubes serve as guides for the reactivity control assemblies (RCA), as described below. In a Westinghouse design, the center tube serves as a channel for nuclear instrumentation. In other PWR designs, a smaller number of larger tubes and control rods are used.

The tubes are attached to end fittings (or “nozzles”). These provide a means for fixing the assembly in place between the upper and lower core plates of the core, and allowing for lifting and movement during refueling. The top fitting in a PWR includes a set of

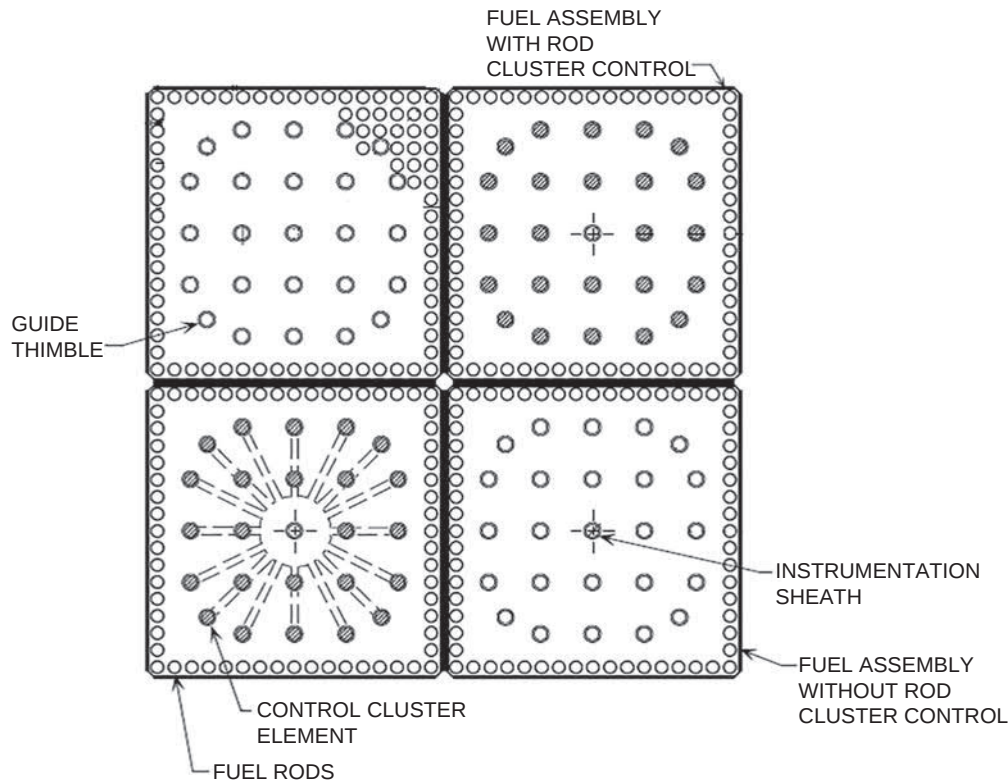


Fig. 3 Typical PWR fuel assembly; cross-section view of four assemblies.

leaf springs (shown in the figure; in some PWRs these are coil springs), that are compressed when the top core plate is replaced, to hold the assembly down.

The RCA (also referred to as a rod cluster control assembly, or RCCA in a PWR) for a typical Westinghouse PWR is shown in **Fig. 4**. A number (in a Westinghouse design there are 24) of “rodlets” with length equal to the fuel rods and containing neutron absorbing material (see section “**Neutron absorber materials**”) are attached to a central “spider” which is then attached to a control rod mechanism that keeps the control rods suspended above the fuel assembly during normal operation. The rodlets are lowered into the control rod tubes of the fuel assembly as needed for reactor reactivity control or shutdown, in addition to the chemical shim (borated water for neutron absorption) used in the reactor coolant. The arrangement of control rods varies among PWR designs. Some designs have fewer control rods with larger diameter.

Although there are several differences in geometry and nomenclature between PWRs and BWRs, the basic configuration of the fuel rod and assembly is about the same. **Fig. 5** shows a typical BWR fuel assembly, inserted into its fuel channel.

In a PWR, a single assembly contains typically 200 to 300 fuel rods and tubes that serve as channels through which control rods move. In a BWR, the equivalent component, often called a “bundle,” is a smaller array of typically 100 fuel rods, with grids and end fittings that serve a similar purpose as in a PWR. In addition, there are a smaller number of tubes, and they are designed to allow water flow for neutron moderation, not for control rods. The location and number of these water channels plays an important role in efficient utilization of the fuel; there are a number of configurations being used today. In addition, each fuel assembly is contained within a solid wall fuel channel. A group of four bundles in their respective fuel channels is shown in **Fig. 6**. In a BWR core the number of such fuel assemblies ranges from 240 to 872.

The reactivity control assembly (usually referred to as the control blade assembly) in a BWR consists of a cruciform blade with four wings that moves in the gaps between the fuel channels as shown in **Fig. 6**. The blades, containing absorber material (see section “**Neutron absorber materials**”) are attached to a hub as shown in **Fig. 7**. The hub, or rod, is attached to a drive mechanism at the bottom of the reactor vessel that pushes the blades into the core from below.

HWR (CANDU)

The heavy water reactor known as the CANDU (Canadian Deuterium-Uranium) reactor is described here. Its design is significantly different than an LWR, as outlined below. More details can be found in [Nichita \(2021\)](#).

1. Natural uranium rather than enriched uranium is used as the fuel.
2. The moderator and coolant is heavy water rather than light water.

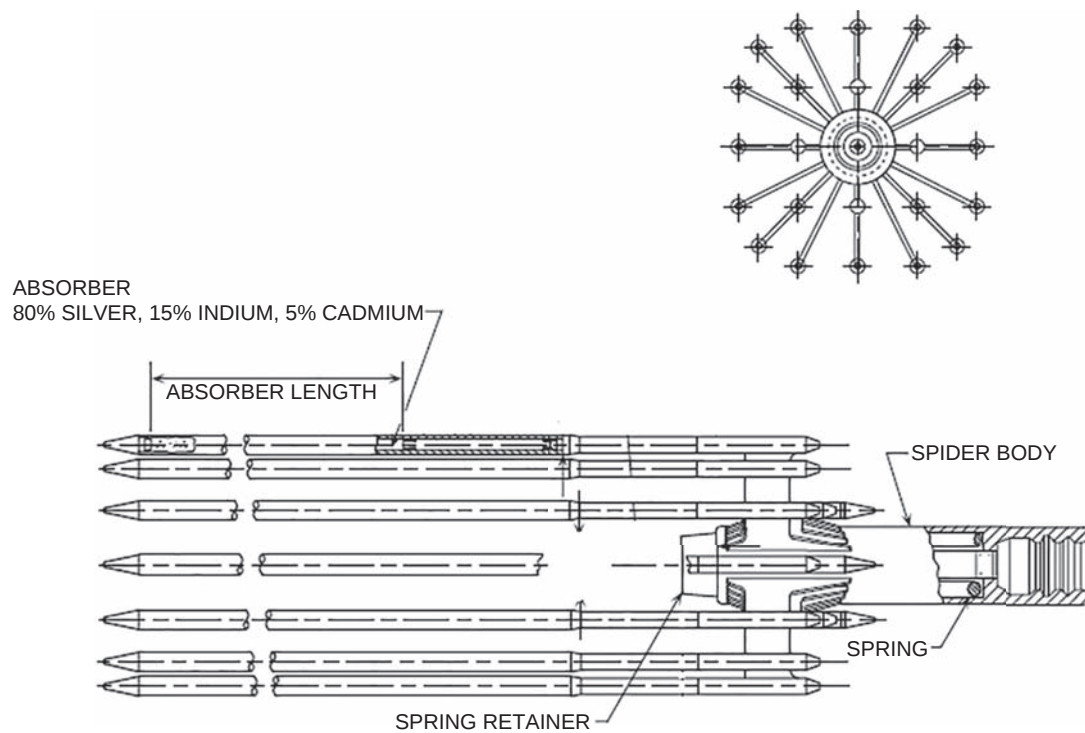


Fig. 4 Typical Westinghouse PWR control rod cluster (RCCA).

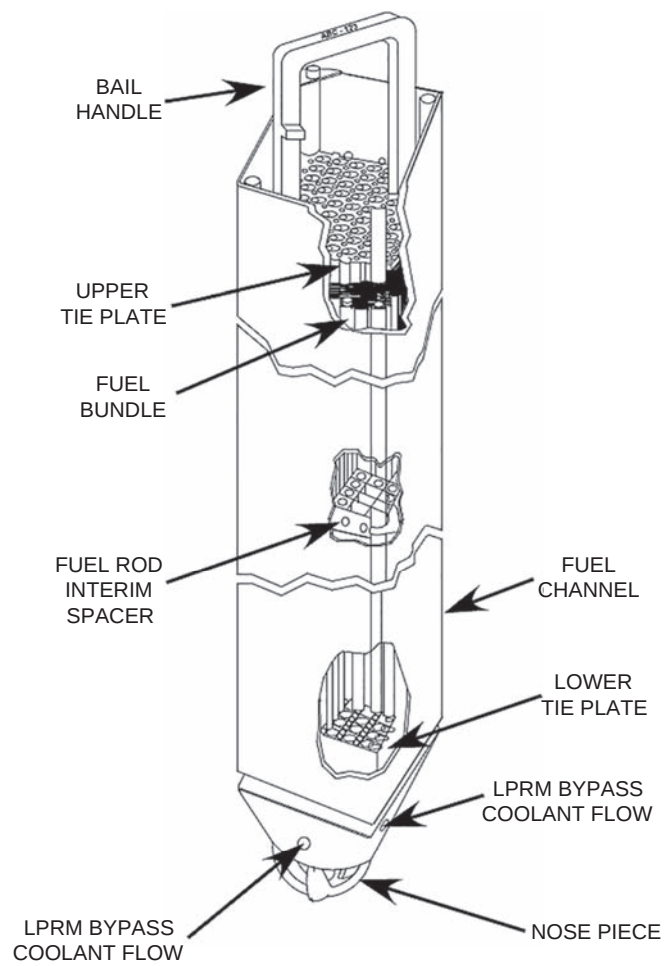


Fig. 5 GE BWR fuel assembly.

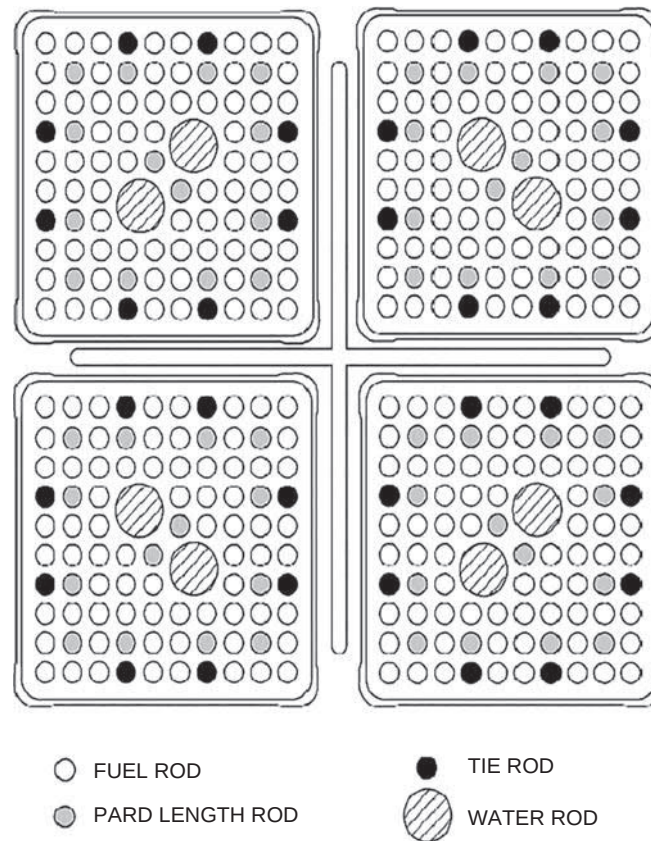


Fig. 6 Cross-section of four BWR assemblies with control blade inserted between the channels.

- Whereas in the LWR the same water is used as both moderator and coolant, in a CANDU there are two separate water circuits; one for heavy water at low pressure and temperature to serve primarily as moderator, and one at high pressure used as coolant to transfer the heat generated in the fuel rods to a steam generator.
- The fuel rods and assemblies are arranged horizontally rather than vertically, in individual pressure tubes (instead of pressure vessels in PWR and BWR).
- The pressure tubes can be isolated and accessed individually, so that refueling is performed continuously, while the reactor is at power.

A CANDU fuel bundle is shown in Fig. 8.

From 28 to 37 fuel rods are arranged as shown, attached via the end plugs to an end support plate. The fuel bundle is about 0.5 m long, and is inserted into a horizontal pressure tube of about 10 cm diameter. The pressure tube itself is about 6 m long; a dozen fuel bundles fit inside. Inside the pressure tube, heavy water circulates as coolant and is directed to steam generators. Several hundred of these pressure tubes reside inside a large tank, called a “calandria vessel,” containing cool, low pressure heavy water, as shown in Fig. 9. The pressure tubes are isolated from the moderator by another set of tubes, called “calandria” tubes. Carbon dioxide gas is circulated between the pressure tube and the calandria tube to provide thermal insulation.

Each pressure tube is connected via fittings to individual external piping to circulate the high pressure, high temperature coolant.

The reactivity control assemblies for a CANDU are substantially different than in an LWR. They reside outside of the higher pressure zone, within the calandria vessel, and run vertically among the pressure tubes. They consist of:

- Pipes containing light water, which provide for fine tuning the reactivity and power distributions.
- Stainless steel “adjuster” rods, normally residing in core, and withdrawn when extra reactivity is needed.
- Mechanical control absorbers, tubes of cadmium sandwiched in stainless steel, normally withdrawn, normally used for rapid power control.
- Shutoff rods, with same construction as item 3, used for reactor trip—rapid shutdown.

Similar to the PWR, CANDUs also can use chemical shim to control reactivity (via boron or gadolinium added to the moderator water).

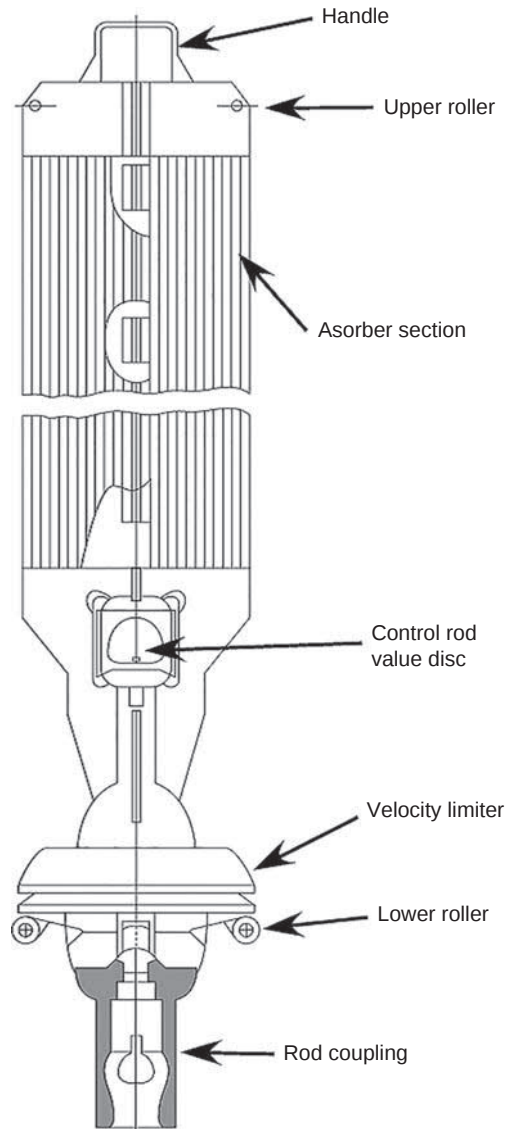


Fig. 7 GE BWR control blade assembly.

Materials of construction

Fuel assembly materials

Modern fuel rod, fuel assembly, and reactivity control structural materials consist primarily of zirconium and nickel alloys, and stainless steel. Because zirconium has a low neutron cross-section in the neutron energy range of interest, nearly all the components within the active core are now constructed of zirconium alloys to minimize neutron loss.

Nickel based alloys, such as Inconel™, are corrosion resistant in the pressure and temperature of reactor environments. This alloy in thin sheet form, is used for example for grids; it retains strength over a wide temperature range. Typically, the top and bottom supporting grids at the ends of the fuel assembly are made of this material, where neutron absorption is not a concern.

Assembly end fittings, control rod clusters and absorber material claddings are typically made of stainless steel.

The materials of construction of fuel and control rod assemblies in a PWR normally consist of:

- (a) Zirconium alloy fuel rod cladding
- (b) Zirconium alloy mid-grids and control rod tubes.
- (c) Nickel based alloy top and bottom grids
- (d) Stainless steel end fittings
- (e) Stainless steel control rod hubs and control rod cladding.

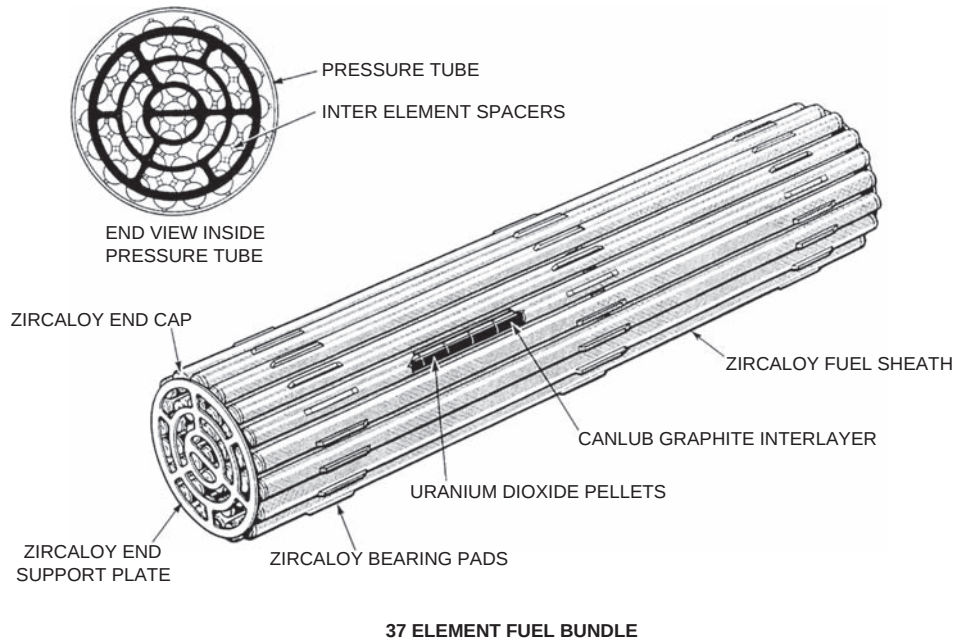


Fig. 8 CANDU fuel bundle.

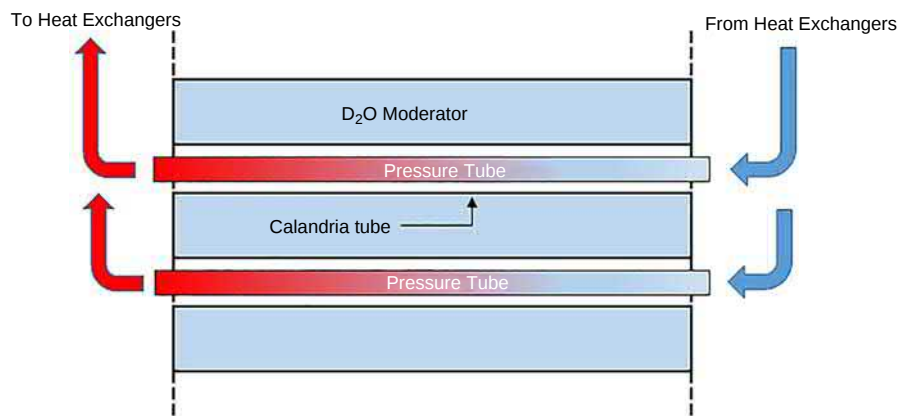


Fig. 9 Schematic of pressure tube/calandria arrangement.

In a BWR, the materials of construction are typically:

- (a) Zirconium alloy fuel rod cladding
- (b) Zirconium alloy channels and water tubes
- (c) Nickel based alloy grids
- (d) Stainless steel end fittings
- (e) Stainless steel control rod hubs and control rod cladding.

In a CANDU, the pressure tube, fuel cladding, and fuel spacers are all made of zirconium alloy.

Fuel materials

The fuel rod pellet is made of sintered uranium dioxide, UO₂. In LWRs, approximately 96% of the uranium consists of U-238, with the remainder consisting of fissile U-235. The typical pellet is about 1.5 cm long, and about 0.9 cm in diameter.

CANDU fuel consists of 100% natural uranium.

Neutron absorber materials

Neutron absorber materials are used in two ways: for rapid shutdown of the reactor, and for more modest control of the core reactivity to achieve more efficient use of the fuel.

Absorber materials in the control assemblies are typically silver-indium-cadmium (AgInCd) in PWRs, and boron carbide (B_4C) in BWRs (some PWRs also use this material). These compounds are strong neutron absorbers. They are contained as pellets within stainless steel tubes or sandwiched between plates. CANDU reactors typically employ cadmium sandwiched in stainless steel in the form of tubes. When fully inserted into the core, these materials cause a rapid termination of the fission chain reaction.

To improve nuclear performance, PWRs and BWRs also employ absorbers added to the fuel pellets. Gadolinium (in the form of oxide, Gd_2O_3) or Erbium (Er_2O_3) are mixed into some of the fuel pellets, while boron (in the form of ZrB_2) is applied as a coating on the pellets. These materials are weaker neutron absorbers. They are used to reduce the reactivity of the feed assemblies, early in the fuel cycle. As the fuel cycle proceeds, their concentration and, hence, reactivity effect is reduced. They are therefore referred to as “burnable poisons.” In PWRs, these burnable poisons are used to avoid high concentrations of chemical shim in the water early in the cycle. In BWRs, they are used to minimize the amount of control rod maneuvering required for reactivity control through the cycle.

Reactor environment

In order to better understand the key elements of fuel rod and control rod design, it is important first to become familiar with the environment in which these components must operate. The operating environment can be divided into two main areas.

Normal operation

A reactor will spend the vast majority of its operating life in this mode, operating at 100% power, with intermittent shutdowns, primarily for refueling. Many design considerations focus on the ability of the fuel to operate without difficulty, for long periods of time, in this mode.

Tables 1–3 provide a summary of the typical operating conditions for modern PWR, BWR, and CANDU cores.

It can be seen from the tables that the fuel rod components of all three reactor designs are exposed to approximately the same set of environmental conditions, with the exception perhaps of chemistry, which is different among the reactor designs. The materials used and operating environments summarized above lead to physical processes that modify material properties, weaken materials, limit heat transfer, and cause vibration and wear. Any design must be evaluated against all these factors to be successful in-reactor. The discussion will focus primarily on fuel rods and fuel assemblies, where the environment severity is greatest. At the end of each subsection, a discussion of how the environment affects the control rod assembly will be presented.

Table 1 Typical PWR operating conditions.

Parameter	Units	Operation range	Comment
Core pressure	MPa	15.5	Some reactors operate at slightly lower pressure
Heat flux: core average-maximum	(W/cm ²)	55–150	
Thermal neutron flux	Neutrons/cm ² -s	$0-1 \times 10^{14}$	
Core coolant temperature	°C	290–330	Maximum core exit temperature typically 20 °C below the saturation temperature (=346 °C at 15.5 MPa)
Coolant velocity in-core	m/s	5–6	Lithium added to offset boron. Current plants operate at constant pH. Coolant is maintained at high purity, but some small debris, primarily from refueling maintenance activities, is unavoidable.
Coolant chemistry	pH	pH > 7.0	

Table 2 Typical BWR operating conditions.

Parameter	Units	Operation range	Comment
Core pressure	MPa	7.2	Core exit coolant is at saturation temperature with steam quality approximately 15%
Heat flux: core average-maximum	(W/cm ²)	50–110	
Thermal neutron flux	Neutrons/cm ² -s	$0-1 \times 10^{14}$	
Core coolant temperature	°C	215–290	
Coolant velocity in-core	m/s	2–8	Primarily low density steam at core exit
Coolant chemistry	pH	Neutral pH	Contains dissolved oxygen.

Table 3 CANDU operating conditions.

Parameter	Units	Operation range	Comment
Core pressure	MPa	10–11	Larger pressure drop through core than LWRs
Heat flux: core average-maximum	(W/cm ²)	70–130	
Thermal neutron flux	Neutrons/cm ² -s	0–1 × 10 ¹⁴	
Core coolant temperature	°C	265–310	
Coolant velocity in-core	m/s	10–12	
Coolant chemistry	pH	Neutral pH	

Transients and accidents

In the United States, the requirements for the design of LWRs, including their fuel design, are spelled out in the General Design Criteria (GDCs) established by federal regulations (10 CFR 50, Appendix A; “General Design Criteria for Nuclear Power Plants”). There are four specific requirements relating to fuel.

1. The fuel system will not be damaged during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).
2. Fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when required.
3. The number of fuel rod failures is not underestimated for postulated accidents.
4. Coolability is always maintained.

There are various transients that need to be considered in the design of a nuclear fuel rod, its assembly, and the control rods. These range from transients that may occur on a regular basis, such as reactor shutdown for refueling, to those that occur several times during the reactor’s lifetime, to those that are postulated to occur, but are very unlikely. For moderate frequency transients it is generally required that the fuel survives the transient without damage, so the reactor can be restarted. For postulated accidents it is generally assumed that some fuel damage will occur. As long as long term cooling can be initiated and maintained, this fuel damage is limited and poses little risk to the public. These anticipated transients and more severe events are discussed in detail in (Martin, 2021).

The more frequent transients that may occur, and against which the fuel must be protected, involve (for example) reductions in flow caused by tripping of a main coolant pump, increases in heat local flux caused by inadvertent withdrawal of control rods, and changes in water temperature and/or pressure caused by sudden changes in steam demand. Numerous automatic systems are available and become activated to shut down the reactor and mitigate the changes that are caused by these events. The challenge in safety analysis (see Martin, 2021) is to accurately calculate these conditions, and confirm with a high degree of confidence that fuel damage is avoided.

The impact of these transients on fuel assembly design is summarized in the section “Fuel assembly and RCA design considerations.”

The intense radiation within the core, coupled with high operating temperatures, introduces significant changes in the properties of the materials of construction. These changes must be considered in the design of the components for other aspects such as heat transfer. For this reason, this environmental factor is discussed first. Under normal operating conditions, the environment in the core includes the following:

Reactivity (radiation)

Within the fuel pellet, the fission and radioactive decay processes generate fission gases, causing the pellet to swell and crack. The fission gases may potentially attack the cladding in a process called pellet-cladding interaction (see section “Pellet-cladding interactions”). If the fuel pellet has a boron coating for use as a burnable absorber, helium is also generated, increasing the internal pressure of the fuel rod.

The high neutron flux leads to changes in the basic properties of materials, altering their density and ductility to name a few. In particular, a reduction in density due to the creation of voids in the material means that the mechanical design must accommodate significant growth in some of the components. At the same time, this growth may affect the ability of the material to withstand corrosion.

The effects of radiation are primarily important during normal operation, since their effects require long term exposures. However, large changes in reactivity are possible and must be considered.

Temperature (pressure)

Materials of construction used in reactors (see section “Materials of construction”) are well suited to the high operating temperatures and pressures encountered in a reactor. Where materials are under some stress, however, higher temperature will cause these materials to creep, to relieve the stress. The primary example of this process is the fuel rod cladding, where the high pressure on the outside of the rod will cause the cladding diameter to decrease, until it contacts the fuel pellet. Grids made of zirconium alloy, and to a lesser extent nickel based alloy, will also relax their grip on the fuel rods due to material creep.

It will be seen in “**Conditions within the fuel rod**” that there is a rather large temperature gradient across the fuel pellet. Because the uranium dioxide material is brittle, cracks form shortly after operation begins, changing the pellet diameter.

An additional effect of temperature is to increase the tendency of zirconium alloy materials to pick up hydrogen from the coolant. This tends to reduce the ductility of the metal.

In a PWR, the control rod assemblies are positioned above selected assemblies in the reactor vessel upper plenum, where the average coolant temperature may be moderately higher than the average. In a BWR, they are located below the reactor vessel, in slightly subcooled water.

During several postulated transients, large and rapid changes in heat flux, coolant temperature and pressure may occur. The stresses discussed above may increase significantly in the short term, and fuel rod design must consider these changes.

Water chemistry

Water plays a fundamental role in LWRs and HWRs as a moderator and heat transfer medium. As it transports heat from the reactor core, irradiated materials are also carried to other areas of the reactor. All reactor materials exposed to water react with it to form corrosion products. An early treatise on the issues related to water chemistry in reactors was provided in [Cohen, 1969](#), with significant advances in understanding since then.

Fuel thermal-hydraulics

Fuel thermal-hydraulics normally refer to the environment adjacent to the fuel cladding-water interface. The following paragraphs describe the heat transfer conditions within a fuel assembly during normal and transient operation in summary form to highlight the key elements.

The fluid conditions within the assembly surrounding the fuel rods is adequately described by the simplified one dimensional picture shown in [Fig. 10](#).

In the BWR, the heat generated in the fuel rod is transferred to the water, which then heats up as it flows through the channel. Many additional factors exist to complicate this picture in the actual fuel channel, but the essential features are here. Near the bottom of the channel, heat is transferred through forced convection heat transfer, and consideration is given to sub-cooled boiling. As the water approaches the boiling point, bubbles begin to form on the tube surface, and both the water and tube temperatures become constant, slightly above the saturation temperature of the water. This is the result of a much more efficient form of heat transfer imparted by the vaporization process. This situation persists for some distance, while the steam content in the water becomes greater.

The water-steam mixture goes through several “flow regimes,” as shown the figure. The high pressures and temperatures in PWR and BWR fuel prevent direct observation of these flow regimes. These regimes have been observed in air-water experiments at low pressure, as well as experiments using Freon, which has a low boiling point and scales reasonably well to water. Supporting evidence at reactor conditions has been obtained by measuring the steam fraction through the use of gamma densitometers in the test facility.

At some point the mixture is so dominated by steam that the tube wall becomes dry. This condition is termed “dryout” (DO). At this point, since the heat that must be transferred remains constant in time, the tube temperature suddenly increases. This is the boiling crisis point, and is gauged through the critical power ratio, which is the ratio of power at which this boiling crisis occurs to the operating power of the fuel assembly in core. Limits are placed on the critical power ratio to prevent boiling crisis from occurring during an AOO.

The situation for a PWR in normal operation is similar, except that the heat transfer regimes are limited to those near the bottom of the channel shown in the figure. The incoming water is much colder relative to the temperature at which the water begins to boil, so most of the fuel length is in forced convection. Near the top of the hottest channels, however, subcooled nucleate boiling begins. Bubbles are created at the surface, grow for some period of time, then detach and condense quickly after entering the main stream. If this boiling and bubble generation are intense enough, the bubbles generated cannot escape or condense rapidly enough and a vapor film is formed on the surface. This is the boiling crisis for the PWR, typically called “departure from nucleate boiling” (DNB).

Both DNB and dryout are also referred to as “critical heat flux” (CHF) in the literature.

The detailed heat transfer processes occurring in light and heavy water reactors (including the steam generators) has been well documented in many papers and texts, such as [Collier and Thome \(1994\)](#). The fundamentals of two-phase flow have also been well documented, as in [Wallis \(1969\)](#) and [Lahey and Moody \(1996\)](#).

Conditions within the fuel rod

In a nuclear fuel rod the power generation rate is independent of conditions at the surface of the cladding. Temperatures within the fuel rod and in the cladding will almost instantly respond to changes in the surface heat transfer coefficient, even when nuclear fission is rapidly being shut down. Temperatures in the interior of the fuel rod respond as needed to maintain transfer of the power generated. [Fig. 11](#) shows a typical temperature distribution during normal operation. In the figure, the center of the fuel is at radial position 0, the pellet-inner cladding gap is at approximately 0.9, and the cladding surface is at 1. Because of the relatively low thermal conductivity of the fuel pellet, the centerline temperature is quite high. The temperature drops sharply across the gap filled gap.

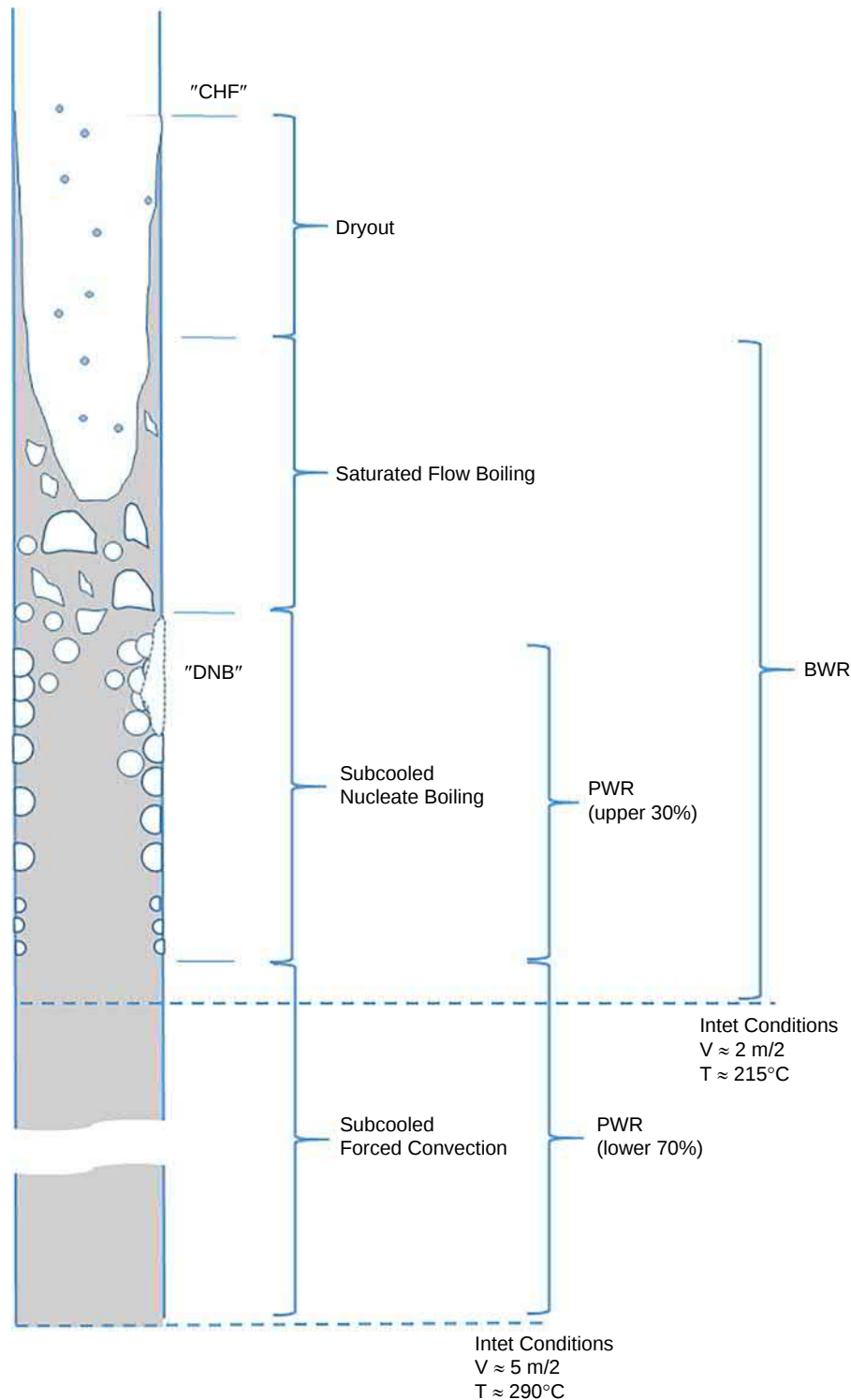


Fig. 10 Fluid conditions within a fuel assembly.

As the fuel cycle proceeds, the pellet tends to expand and crack as it responds to the high temperatures and radiation, as previously noted. The high temperature and high coolant pressure cause the cladding to shrink, or "creep down" until it contacts the pellet. The typical condition of the fuel rod after several months of operation is also shown in Fig. 11.

In a transient, rapid changes in local fluid conditions occur within the fuel channel. For example, a reduction in flow due to failure of a pump will lower the axial location at which the coolant becomes saturated and significant additional steam formation begins. In a PWR, the boundary between subcooled forced convection and subcooled nucleate boiling rapidly moves downward in

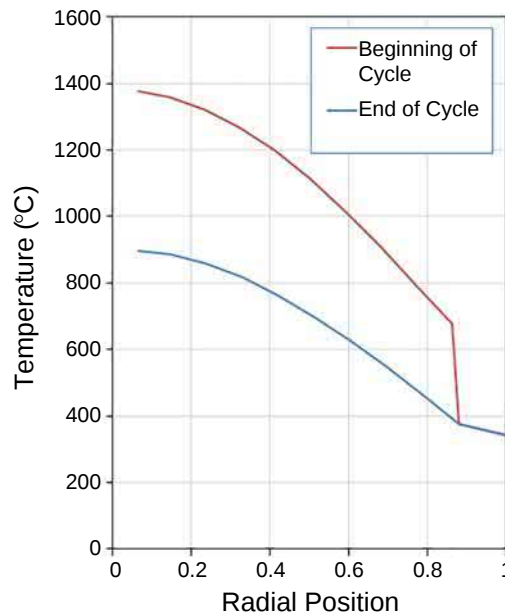


Fig. 11 Typical fuel and cladding temperature under PWR operating conditions, beginning and end of cycle.

the core (see Fig. 10). The potential for DNB or dryout rapidly increases. At the same time, the heat flux into the channel is rapidly diminishing as control rods are inserted and power is reduced. The challenge in safety analysis is to accurately predict these competing conditions, and confirm with a high degree of confidence that fuel damage, DNB or dryout are avoided.

Conditions on the fuel rod surface

As shown in Fig. 11, the cladding surface temperature (at relative radial position 1) under normal operating conditions is a few degrees above the saturation temperature of water. For a PWR operating at 15.5 MPa (2250 psi), this temperature is approximately 350 °C (662 °F) in the upper region of the core where nucleate boiling is occurring, and for a BWR operating at 7.3 MPa (1050 psi), it is approximately 290 °C (554 °F). These temperatures are well within the normal operating capability of the zirconium alloys in use today. However, the conditions at the surface may result in a more severe chemical environment, as explained below.

In the subcooled nucleate boiling region, shown in Fig. 10, there is a tendency for dissolved or particulate materials carried in the coolant to concentrate at the cladding surface. This process, commonly called “fouling,” occurs on virtually all industrial heat transfer surfaces. The concentrated materials collect as “crud.” This process is illustrated schematically in Fig. 12. With thick crud deposits, solid boron compounds may form, affecting core reactivity. Such events are avoided by careful control of plant chemistry and minimization of corrosion. Crud deposits removed from PWR fuel rods have been examined in detail to understand their structure and chemistry (Byers and Deshon, 2004).

These crud deposits have presented a significant environmental challenge recently, as power densities have increased. In PWRs, boron (dissolved in the water for long-term reactivity control) concentrating within the deposits has resulted in unanticipated changes in core reactivity (PWR Axial Offset Anomaly (AOA), 2004). In both PWRs and BWRs locally thick crud deposits have led to increased cladding temperatures and isolated fuel leaks.

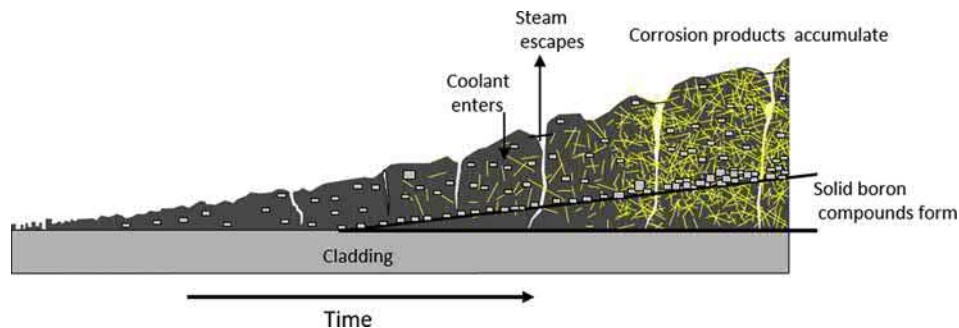


Fig. 12 Schematic of the corrosion product accumulation process on a surface with subcooled nucleate boiling.

Flow induced vibrations

The interaction of the fuel assembly with the coolant and with contacting or nearby structures is a third major environmental challenge. In the relatively high velocity fluid environment several vibration and wear mechanisms come into play. In addition, operations such as shipping and handling of a fuel assembly, and transients the reactor may experience, impose stresses that must be considered in the design. These types of interactions are summarized in Fig. 13. Additional details may be found in Au-Yang (2001) and Pettigrew et al. (1998).

Fluid flow induced vibration occurs in both PWR and BWR fuel assemblies. The vibration in PWRs is more problematic for two reasons:

- PWR fuel assemblies typically use grids made of zirconium alloys, which tend to relax and lose their grip on the rods, as discussed below. BWR fuel assemblies use nickel based alloy grids that do not relax as much, and prevent gaps from forming between the rod and grid support.
- Crossflow across the fuel assembly, which exists in a PWR core and increases the potential for vibration, is prevented in BWRs by the fuel bundle channels.

In a PWR, the control assembly rodlets, when in their normal retracted positions, are located within control rod guide tubes in the reactor vessel upper plenum. The guide tubes contain support structures at several intervals, to guide the rodlets. At the lower end of the guide tube, the axial core flow must turn as it enters the upper plenum and hot leg piping. The rodlets are therefore subject to crossflow and potential vibration effects.

BWR control assemblies move in between the fuel channels as shown in Fig. 7, are not exposed to high crossflows, and are not subject to significant flow induced vibrations.

Pellet-cladding interactions

In the first cycle, the pressure inside the fuel rod is significantly less than the RCS pressure. As shown in Fig. 1, a gap exists initially between the pellet and the cladding, so the fuel rod cladding is in a compressive state during most of the first cycle.

During initial power ascension, the fuel pellet develops small cracks due to its brittle nature. Over two or three cycles of operation (a typical cycle lasts between 12 and 24 months, depending on the reactor type and the core configuration), fission gases formed within the pellet will escape through these cracks into the volume within the cladding, increasing the internal pressure of the fuel rod. At the same time, the cladding strains, or creeps inward in response to the high pressure and temperature of the coolant. Typically, near the end of the first cycle or shortly after the start of the second cycle, hard contact occurs between the pellet and cladding. After hard contact, continued pellet swelling causes the cladding to strain outward, and cladding stresses go from compression to tension.

The changes in the fuel and cladding that occur over the operating life of a fuel assembly are a complex interaction involving nuclear, chemical, heat transfer and material physics that can only be handled accurately with advanced computer models. Fuel suppliers typically use their own proprietary computer codes, focused on their specific fuel design features. These codes require technical review and approval by the United States Nuclear Regulatory Commission (USNRC), as well as other international regulatory agencies. The NRC employs a computer code called FRAPCON (Geelhood et al., 2015) to perform check calculations.

Much design effort has been expended to improve margins to PCI failure. One mitigating feature is to add an inner liner of pure zirconium to the cladding. This softer material is better able to accommodate local stresses. Additives to the fuel pellet achieve the same purpose. Emphasis is also placed on assuring that pellet flaws (typically missing pellet sections cause by chipping during the manufacturing process) are eliminated.

The fuel rod must be capable of withstanding stresses from a number of anticipated transients. These transients may result in sudden increases or decreases in RCS pressure, or increases in local power due to unexpected reactivity insertions. As a result, the pellet may expand or the cladding may experience increased stresses. Verification of the ability of the fuel rod to withstand significant power transients has been performed in test reactors (typically research facilities). In these tests, fuel rod segments are inserted

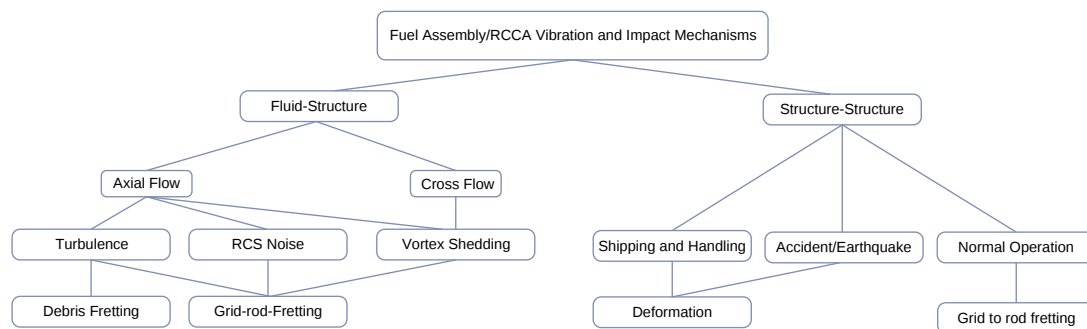


Fig. 13 Potential interaction sources within the core.

and subjected to sudden power transients to assess their ability to withstand changes in temperature without failure, and to determine failure limits. The design criteria that must be considered are discussed in the next section.

Fuel rod-debris interactions

Small pieces of debris are always present in small quantities in the coolant, typically left behind by maintenance operations during refueling or plant modifications. These may be caught on the grids, and may vibrate against the fuel rod cladding, causing wear and potential cladding failure. This is an important source of potential damage for both PWR and BWR designs, and is typically mitigated by debris filters attached to the fuel assemblies.

Fuel assembly and RCA design considerations

The previous sections described the different environments, both steady state and transient, that could be experienced by fuel rods and control rods, and that pose a challenge to the successful operation of these components.

The geometrical and material configurations described above have remained basically unchanged since the 1970s, having been designed to accommodate the competing mechanical, thermal-hydraulic and, of course, nuclear design constraints and optimization goals. What has driven design since then has been the desire to make greater use of the available energy in the nuclear fuel, primarily by increased power density and increased burnup, while maintaining safety. The total amount of fissile uranium (enrichment) in the fuel assembly drives the ability to support the length of the operating fuel cycle at the reload batch size. The reactivity of the assembly is governed by both the amount of fissile material and the amount of moderator (coolant) within the assembly. This ratio obviously changes throughout the life of the assembly as the fissile material is consumed. The amount of assembly enrichment (and in BWRs, the axial distribution of the enrichment) is chosen to meet both the energy requirements, but also to meet the requirements for the ability of the reactor to meet shutdown reactivity requirements at all points during the cycle.

While each specific fuel design has a fuel rod pitch-to-diameter ratio that gives the optimum moderation point (Tsvetkov, 2021) for reactivity (k -infinity), this ratio is dependent on fuel fissile content and thus exposure as the fuel material depletes in U^{235} and increases Pu^{239} . K -infinity of the lattice is also dependent on the geometry. Increasing the diameter for a given pitch-to-diameter ratio, increases the reactivity, but this also changes the amount of moderation from the interstitial water between fuel rods. These factors are all balanced in the fuel assembly, which is designed to be operated within the core constraints, requiring that, at operating conditions, the temperature or void reactivity be negative. This is affected by both fuel design characteristics and also the core design.

Beyond this, the arrangement of nuclear material within the fuel assembly is driven primarily by limitations on the fuel rod thermal limits, linear heat generation rate and the dryout performance. Water incorporated in non-fuel components of the assembly, such as in control rod and instrumentation thimbles in PWRs and the intra-assembly gaps in BWRs (and so-called water rods) generally increase the power produced in nearby fuel rods. These water containing components also act as neutron traps in the cold, shutdown condition. So both thermal limits and shutdown margin must account for the effect of these features.

Pin-wise power distribution across the fuel assembly is also affected by the presence of burnable absorbers mentioned earlier. While these are primarily used to control reactivity, they have an effect on neighboring fuel rods. As with any feature, this effect can either be an obstacle or a design feature, as it is in BWRs which tailor the lattice power distribution using the burnable absorber.

After several decades of design evolution, the increased use of intermittent power sources such as solar have pushed reactors to move away from constant baseload power, to frequent changes in power during a fuel cycle (see Young, 2021), as well as challenges for large steps in fuel cycle cost performance. This has resulted in additional changes in fuel design focused on:

- (a) Reductions in fuel rod diameter and increases in number of fuel rods to improve nuclear performance. In BWR's there may also be variations in fuel rod length in some portions of the bundle to improve the balance between thermal-hydraulic and fuel rod performance.
- (b) Modifications to the grids to promote heat transfer, in order to accommodate higher power density.
- (c) Improved fuel pellets to accommodate higher power density, more rapid power changes, and higher burnup.
- (d) Improved alloys to reduce corrosion and allow for higher burnup.
- (e) Various improvements to reduce manufacturing costs and increase reliability.

Additionally, the criteria established for fuel performance under normal and accident conditions are being examined for more precise, and potentially beneficial, handling of the fuel response under those conditions. A detailed description of the accidents and transients that must be considered in the design of LWRs and HWRs in the United States is provided in Martin (2021). From the regulatory requirements and guidance provided, Specified Acceptable Design Limits (SAFDLs) for all known damage mechanisms (see "Reactor environment") have been developed over time that form the basis for acceptance of a fuel assembly design. Typical SAFDLs for the fuel rod, fuel assembly, and reactivity control assembly are shown in Table 4.

Table 4 Selected design limits for PWR and BWR fuel assemblies and reactivity control assemblies.

<i>Component</i>	<i>Condition to be avoided</i>	<i>Design limits (SAFDLs)</i>	<i>Comments</i>	<i>Failure(s) prevented</i>
Fuel pellet	Fuel melting	Fuel centerline temperature < melting point		Fuel melting
Cladding	Excessive oxidation	Maximum total oxidation at end-of-life < specified limit		Cladding failure
Cladding	Excessive stress and strain	Rod internal pressure < system pressure	Due to fission gas release	Cladding failure
Cladding	Excessive stress	Clad stress < yield strength	Transients in which the fuel pellet expands against the cladding	Cladding failure
Cladding	Excessive fatigue	The cumulative strain fatigue cycles are less than the design strain fatigue life.	Consider anticipated number of cycles during component life, plus safety factors	Cladding failure
Fuel pellet	Excessive burnup	Discharge burnup < limit established for the fuel		Operation in unverified operational space
Fuel rod	Excessive local power	Peak linear heat generation rate < allowed by tech specs at each axial position	Must consider effect on local power of manufacturing variations, measurement uncertainties, fuel densification, rod to rod gap reduction, etc.	DNB, fuel centerline melt, fuel damage
Fuel rod	DNB/DO	For all transients, ratio of DNB, DO heat flux to maximum local heat flux > 1.0 with 95% probability, 95% confidence		Fuel melting, cladding failure
Fuel rod	Excessive growth	Axial growth < available clearances between top and bottom nozzle		Fuel rod bow, assembly distortion (PWR only)
Fuel assembly grids (PWR)	Excessive oxidation or hydrogen pickup	Grid hydrogen pickup < specified limit		Fuel coolability, incomplete RCCA insertion
Fuel assembly grids	Excessive impact loads during shipping, handling, accidents	Impact load during earthquake < grid damage load established by testing	Accidents up to and including LBLOCA + design basis earthquake	Fuel coolability, incomplete control rod insertion
Fuel assembly end fittings	Excessive impact loads during shipping, handling, accidents	Maximum stress < maximum allowable with safety factor	Accidents up to and including LBLOCA + design basis earthquake	Incomplete control rod insertion
Control rods	Excessive stresses	Stresses in control rod components < yield during shipping and reactor trip	ASME Section III limits and procedures generally used	Incomplete RCCA insertion
Control rods	Excessive resistance to insertion	Control rod insertion time < maximum allowable value	Absorber swelling < limit	Incomplete control rod insertion
BWR channel box	Excessive distortion	Differential irradiation growth, differential hydrogen pickup	Assembly bow < limit	Incomplete control blade insertion
Core	Excessive reactivity insertion	Maximum reactivity change rate < specified limit	Channel distortion < limit accounting for seismic/LOCA loads	DNB, fuel damage
			Interaction of boron dilution or flow change and control rod withdrawal	

See Also: Fresh Fuel Properties: Ceramic Compounds; Fuel Design and Fabrication: Pellet-Type Fuel; Introduction to Fuel Cycle Front End: From Mining Through Utilization.

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Heat Transfer Systems

Karen Vierow Kirkland, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

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Glossary

Criticality The normal operating condition of a reactor, in which nuclear fuel sustains a fission chain reaction. A reactor achieves criticality (and is said to be critical) when each fission event releases a sufficient number of neutrons to sustain an ongoing series of reactions.

Fuel burnup The amount of energy released per mass of fuel

Heat flux Heat flow rate (energy per time) per unit area of heated surface

Heat rate Heat flow rate (energy per time)

Power density Heat rate per unit volume (energy per time per volume)

Quality Ratio of the mass flow rate of the gas to the mass flow rate of the gas-liquid mixture.

Reactivity A term expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Abbreviations

ADS	Automatic depressurization system
BWR	Boiling water reactor
ECCS	Emergency core cooling system
ESBWR	Economic simplified boiling water reactor
ESF	Engineered safety feature
GDCS	Gravity-driven cooling system
HPCI	High-pressure coolant injection
HPCS	High-pressure core spray
HPSI	High pressure safety injection
IRWST	In-containment refueling water storage tank
LOCA	Loss of coolant accident
LPCS	Low-pressure core spray
LPSI	Low pressure safety injection
NSSS	Nuclear steam supply system
PCCS	Passive containment cooling system (for the ESBWR)
PCS	Passive containment cooling system (for the AP1000)
PWR	Pressurized water reactor
RCIC	Reactor core isolation cooling (system)
RCS	Reactor coolant system
RHR	Residual heat removal (system)
RPV	Reactor pressure vessel
RWST	Refueling water storage tank
SIT	Safety injection tank
US NRC	United States Nuclear Regulatory Commission

Introduction

The purposes of heat transfer systems in nuclear reactors are two-fold:

- To produce useful energy resulting from the nuclear fission reaction and
- To maintain the reactor systems, structures and components in a safe state, thereby preventing release of radioactivity and protecting the public's health and safety.

To produce useful energy from the fission of nuclear fuel, heat undergoes a number of conversions to other forms of energy, along with being transported from the fuel to the location of final conversion. The final use can be as electricity, as high-temperature heat to drive a chemical reaction such as hydrogen production, or as lower-temperature process heat for industrial applications. The heat transfer systems are designed to enable the energy conversion and transport.

The latter function, maintaining nuclear safety, is ensured by adherence to the nuclear safety goals published by the US Nuclear Regulatory Commission ([US NRC, 1986](#)). Qualitative and quantitative goals were established to limit the risk to individual members of the public and to society from nuclear power plant operation. Adequate heat removal from the nuclear fuel by the heat transfer systems prevents fuel failure and subsequent release of radioactive fission products, and thereby, fulfills the nuclear safety goals. Adequate heat removal is achieved by providing core cooling at a rate sufficient to remove heat generated by nuclear fission, decay heat released by radioactive decay heat of unstable nuclei and heat release associated with system cooldown. Following termination of the fission reaction by control mechanisms, decay heat is the largest heat source that needs to be accommodated.

Energy transport and conversion within reactor heat transfer systems

The energy conversion processes and heat transport in power reactors are summarized in this section. The energy flow and conversion processes for water-cooled power reactors are identified in [Fig. 1](#). The primary source of energy is a nuclear fission reaction in the fuel. A fission reaction produces radioactive fission products that deposit all of their energy as frictional heat in the nuclear fuel elements (in addition to most of their radioactive decay energy). This heat conducts radially outward through the fuel, through the

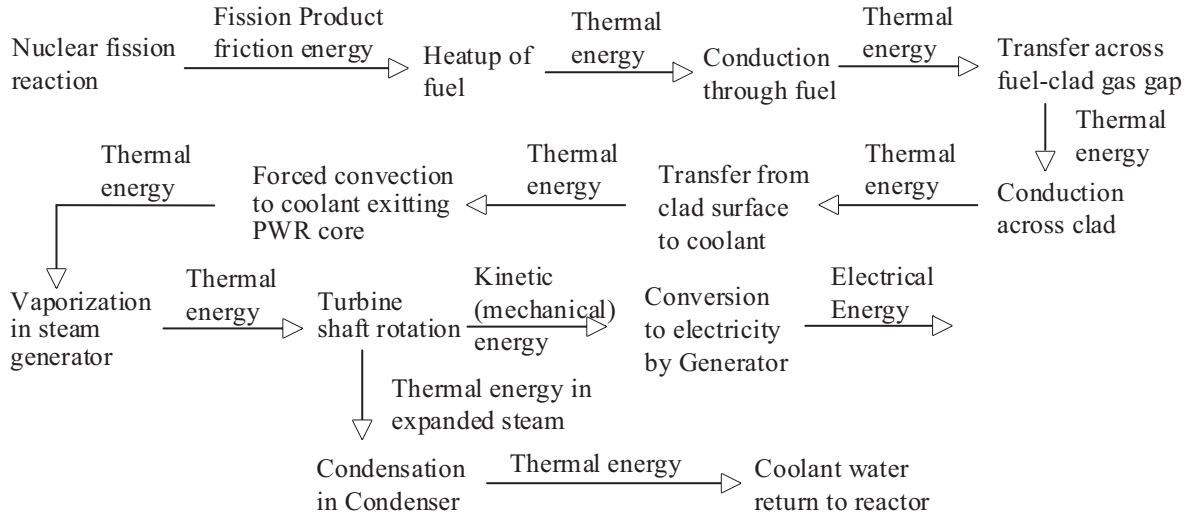


Fig. 1 Energy flow in water-cooled power systems.

gas gap or contact interface between the fuel pellet and the clad inner surface, and finally through the clad to the surrounding coolant (Young, 2021).

For a direct-cycle water-cooled reactor (e.g. BWR), the coolant water removes heat from the clad outer surface by forced convection and boiling. Coolant water evaporates within the core region, producing steam which carries its thermal energy to the steam turbines. For an indirect cycle water-cooled reactor (e.g. PWR), the coolant water removes heat from the clad outer surface by forced convection and carries its thermal energy to an external steam generator. Secondary-side feedwater to the steam generator removes heat from the outer surface of the steam generator tubes by forced convection and boiling. The secondary-side coolant evaporates within the steam generator, producing steam which carries its thermal energy to the steam turbines.

The thermal energy imparted by the steam on the turbine blades provides the motive force for turbine shaft rotation. The thermal energy of the steam is, in this way, converted to kinetic (i.e. mechanical) energy of the turbine. The turbine powers a generator which produces the end product, energy in the form of electricity.

The steam exhausted from the turbine carries energy which is not used in the Rankine cycle to produce work output. This lower-enthalpy steam is condensed in the main condensers and the resulting condensate is returned to the secondary side of the steam generators in a Pressurized Water Reactor (PWR) or to the Boiling Water Reactor (BWR) feedwater inlet.

The radial temperature distribution in PWR fuel is illustrated in Fig. 2. The classical heat conduction equation is solved to obtain the temperature distribution as a function of space and time within the fuel pellet.

$$\rho c_p \frac{\partial T}{\partial t} = \nabla k \nabla T + q''' [\text{kW m}^{-3}] \quad (1)$$

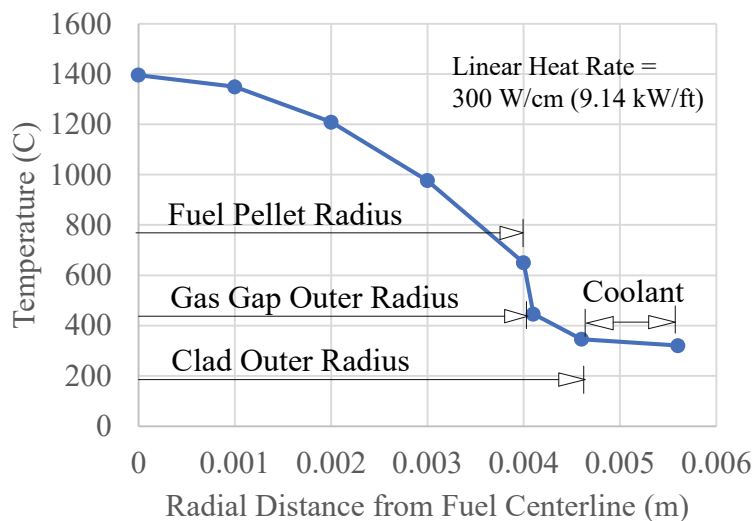


Fig. 2 Radial temperature distribution for a PWR fuel rod.

where

ρ = fuel pellet density [kg m^{-3}]

c_p = specific heat [$\text{kJ kg}^{-1} \text{K}^{-1}$]

T = temperature [K]

t = time [s]

k = thermal conductivity [$\text{kW m}^{-1} \text{K}^{-1}$] and

q'' = volumetric heat generation rate within the fuel [kW m^{-3}].

The pellet thermal conductivity is a function of several variables including temperature, burnup, fuel porosity and composition. The thermal conductivity is highly dependent upon temperature (Fig. 3). This dependency must be considered in analysis, as the temperature across the radius of the fuel pellet varies by several hundred degrees (Fig. 2).

For a steady-state situation with circumferential symmetry, the heat conduction equation simplifies to:

$$\frac{1}{r} \frac{d}{dr} \left[k(T) r \frac{dT}{dr} \right] = -q''(r) [\text{kW m}^{-3}] \quad (2)$$

where r [m] is the radial location within a fuel pin.

For the special case of constant thermal conductivity and a uniform volumetric heat generation rate, the heat conduction equation for the radial temperature distribution in the fuel pellet further simplifies to:

$$\int_{T(r)}^{T_{\max}} k dT(r) = q'' \frac{R_{\text{fuel}}^2}{4} [\text{kW m}^{-1}] \quad (3)$$

The outer surface temperature of the fuel pellet serves as a boundary condition for analysis of heat transfer through the gas gap. The gap is very thin, such that heat transfer by convection of the gas molecules within the gap is negligible. Thermal conduction and radiative heat transfer are the primary heat transport mechanisms. The thermal resistance at the fuel-gap interface is largely a function of fuel surface and clad surface protrusions, which determine the amount of fuel-clad contact (Fig. 4), and also a function of contact pressure.

The heat transfer rate between the fuel and clad can be expressed as a function of the gap conductance, h_g .

$$q_g'' = h_g (T_{\text{fuel, outer}} - T_{\text{clad, inner}}) [\text{kW m}^{-2}] \quad (4)$$

$$h_g = \frac{k_{\text{gas}}}{\delta_{\text{eff}}} + \frac{\sigma}{\frac{1}{\epsilon_{\text{fuel}}} + \frac{1}{\epsilon_{\text{clad}}} - 1} \frac{T_{\text{fuel, outer}}^4 - T_{\text{clad, inner}}^4}{T_{\text{fuel, outer}} - T_{\text{clad, inner}}} [\text{kW m}^{-2} \text{K}^{-1}] \quad (5)$$

where

k_{gas} = thermal conductivity of the gas in the gap [$\text{kW m}^{-1} \text{K}^{-1}$].

δ_{eff} = effective gap width [m].

σ = Stefan-Boltzmann constant for radiative heat transfer [$\text{kW m}^{-2} \text{K}^{-4}$].

$\epsilon_{\text{fuel}}, \epsilon_{\text{clad}}$ = fuel pellet and clad emissivities [–].

$T_{\text{fuel, outer}}, T_{\text{clad, inner}}$ = fuel pellet outer and clad inner surface temperatures [K].

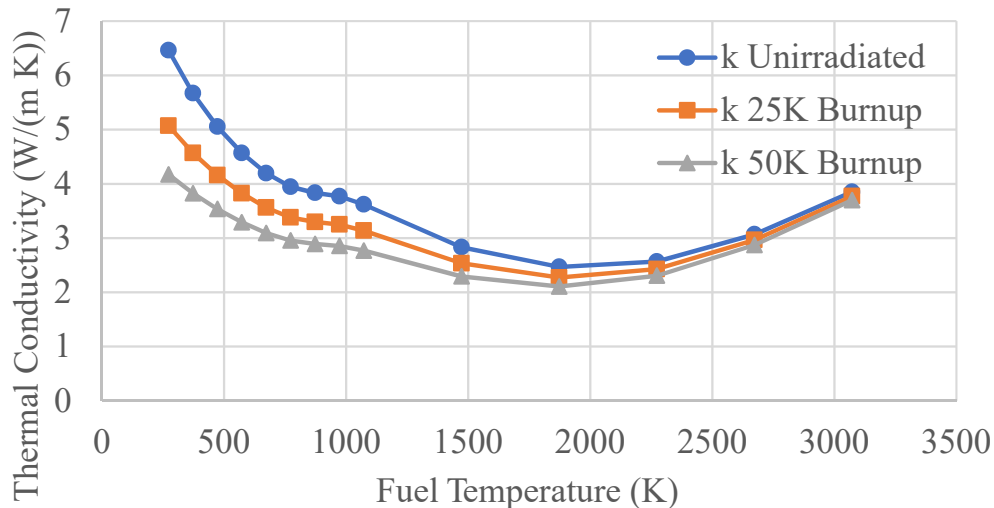


Fig. 3 Thermal conductivity of UO_2 (calculated with AREVA model in Todreas and Kazimi (2012), p. 370).

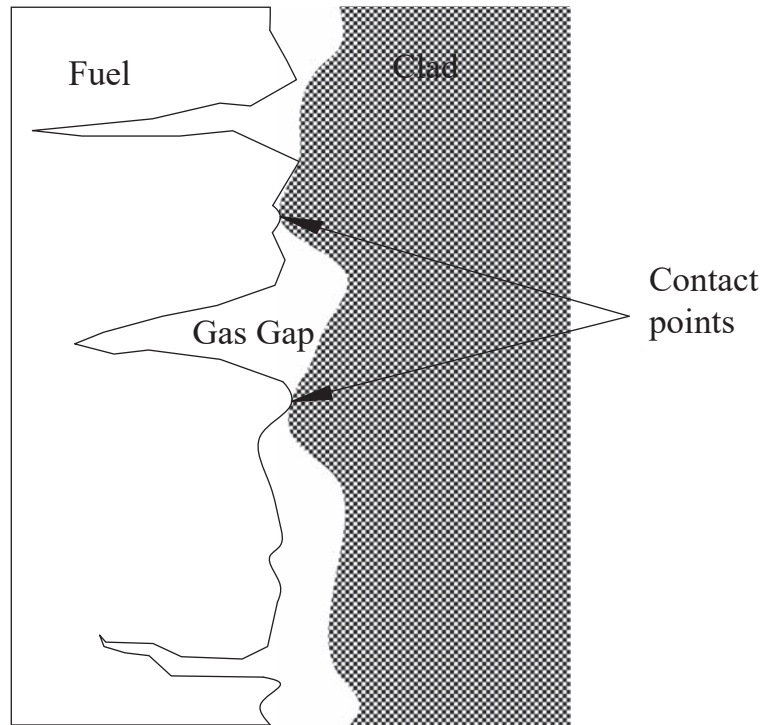


Fig. 4 Physical contact between fuel pellet and clad inner surface.

The first term on the right-hand side of the equation accounts for conduction while the second term represents the radiative heat transfer.

The conduction heat transfer rate from the clad inner surface to the clad outer surface is solved similarly as for the fuel. The volumetric heat generation rate, q''' [kW m^{-3}], within the clad is much smaller than that due to fission in the fuel, however it is often accounted for in fuel vendor calculations. Heat generation within the clad is mainly due to gamma-ray absorption and elastic and inelastic scattering of neutrons.

Convective heat transfer from the clad outer surface to the coolant occurs under turbulent flow conditions. The heat flux, q'' [kW m^{-2}], is expressed as the product of the heat transfer coefficient, h [$\text{kW m}^{-2} \text{K}^{-1}$], and the coolant temperature change from the heated surface to the bulk coolant temperature at the given elevation, z [m].

$$q''(z) = h(z) \Delta T(z) [\text{kW m}^{-2}] \quad (6)$$

For single-phase heat transfer, the forced convection heat transfer coefficient is appropriately selected to predict the change in sensible heat. Heat transfer by convection dominates over heat transfer by conduction for many coolants. For these coolants, the Dittus-Boelter correlation with a viscosity correction [Todreas and Kazimi \(2012\)](#) can be applied. Liquid metal coolants are a notable class of coolants for which a heat transfer coefficient must be chosen that accounts for both conduction and convective heat transfer, because conduction heat transfer is important even with significant convective heat transfer due to bulk motion of the coolant.

For two-phase heat transfer, a heat transfer formulation is applied that accounts for heat transfer both by bulk motion of the coolant (sensible heat) and by convective boiling (latent heat).

Reactors cooled by fluids other than water may have other steps in the energy conversion process, and additional components in their heat transfer systems, to transfer energy from the primary coolant to the coolant which delivers energy to power a turbine. In the energy flow of a gas-cooled power reactor, the steam generator is not needed if it is operating on the Brayton (gas) cycle.

Additional energy conversion processes take place in the vapor generation equipment and turbomachinery (turbines and pumps) within the heat transport systems. Power reactors operating on the Rankine (vapor) cycle have vaporization of liquid and condensation of the vapor. Most Rankine cycles use water as the working fluid. The steam generator vaporizes water into steam. Liquid metals including sodium or lead, or another fluid that does not moderate neutrons well, may be used in fast reactors. More exotic fluids such as mercury and potassium are used as the working fluid in compact space reactors.

Heat transfer system components in the basic electric power cycle

The basic electric power cycle is shown in [Fig. 5](#). The steam source for a power reactor is the Reactor Pressure Vessel (RPV) for a direct cycle reactor and the steam generator for an indirect cycle reactor. Direct cycle reactors include the BWR and the gas-cooled reactor,

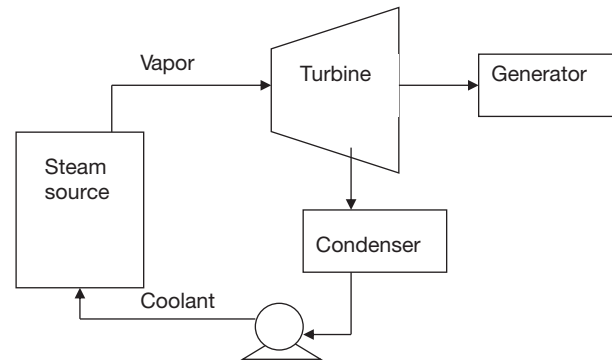


Fig. 5 The basic electric power cycle for the rankine cycle.

which operates on the Brayton cycle. For the latter, hot gas from the core region flows directly to a gas turbine and the condenser and pump in **Fig. 5** are replaced by a compressor. Indirect cycle reactors include the PWR and a gas-cooled reactor operating on the Rankine cycle. For the latter, hot gas from the core region flows through a steam generator which boils water on the secondary side. The steam from this steam source then travels to a steam turbine. [Todreas and Kazimi \(2012\)](#) provide a convenient thermodynamic overview of the power cycles for various reactor types.

Heat transfer systems relevant to nuclear power generation

The systems discussed in this article are the Nuclear Steam Supply System (NSSS), the Reactor Coolant System (RCS) and the Emergency Core Cooling System (ECCS), with a focus on Light Water Reactor systems.

The NSSS is defined by the US NRC as “The reactor and the reactor coolant pumps (and steam generators for a pressurized water reactor) and associated piping in a nuclear power plant used to generate the steam needed to drive the turbine generator unit” ([US NRC, 2020a](#)). The RCS is defined as “The system used to remove energy from the reactor core and transfer that energy either directly or indirectly to the steam turbine” ([US NRC, 2019a](#)). Thirdly, the ECCS is stated to be “Reactor system components (pumps, valves, heat exchangers, tanks, and piping) that are specifically designed to remove residual heat from the reactor fuel rods in the event of a failure of the normal core cooling system (reactor coolant system)” ([US NRC, 2020b](#)).

The NSSS and the RCS share many of their major components. “RCS” appears far more often than “NSSS” in formal technical documents. In particular, NUREG-0800 ([US NRC, 2007](#)), which dictates the Standard Review Plan for US BWRs and PWRs and therefore provides guidance for the contents of Safety Analysis Reports and other documents submitted to the US NRC, refers exclusively to the RCS. For consistency, this article will describe the RCS and the ECCS and mention separately any significant aspects of the NSSS for heat transfer systems that are not discussed within the context of the RCS.

Reactor coolant system (RCS)

Description and functions

As described in the Introduction, the heat transfer systems have the functions of producing power and ensuring nuclear safety. With respect to the safety functions, the core and RCS are designed to enable operation such that no fuel damage occurs as a result of normal operation or a moderate frequency transient. By designing against fuel damage, the release of radioactivity is minimized and the nuclear safety goal of protecting the public’s health and safety is achieved. With respect to the power production, the core and RCS are designed to enable operation at rated core power and possibly load following or load maneuvering without core damage.

In a PWR, the RCS provides steam heat from the nuclear fuel to the steam generators. The RCS consists of the RPV, the steam generators, the reactor coolant pumps, the pressurizer and the connecting piping. Also termed the “primary loop” or the “primary system”, the RCS is illustrated in **Fig. 6**.

Depending on the particular plant, the RCS may have two or three or four primary loops. The most common configuration in the US is a Westinghouse four-loop plant. Each loop includes one steam generator and one reactor coolant pump. One loop is equipped with a pressurizer, which regulates the primary system pressure. Three-loop Westinghouse plants consist of three steam generators, three reactor coolant pumps and one pressurizer. Two-loop Westinghouse plants have two steam generators, two reactor coolant pumps and one pressurizer. Reactors designed by Combustion Engineering or Babcock and Wilcox have the one pressurizer and two steam generators. Different from the Westinghouse designs, two cold legs return from each steam generator to the RPV, for a total of four reactor coolant pumps in the primary system. See also [Brown \(2021\)](#).

In current BWRs, the steam is generated within the RPV. Termed “direct cycle”, the BWR does not need an external steam generator as the PWR does. The NSSS consists of the RPV, the nuclear core, the turbines and associated piping. The RCS consists of the

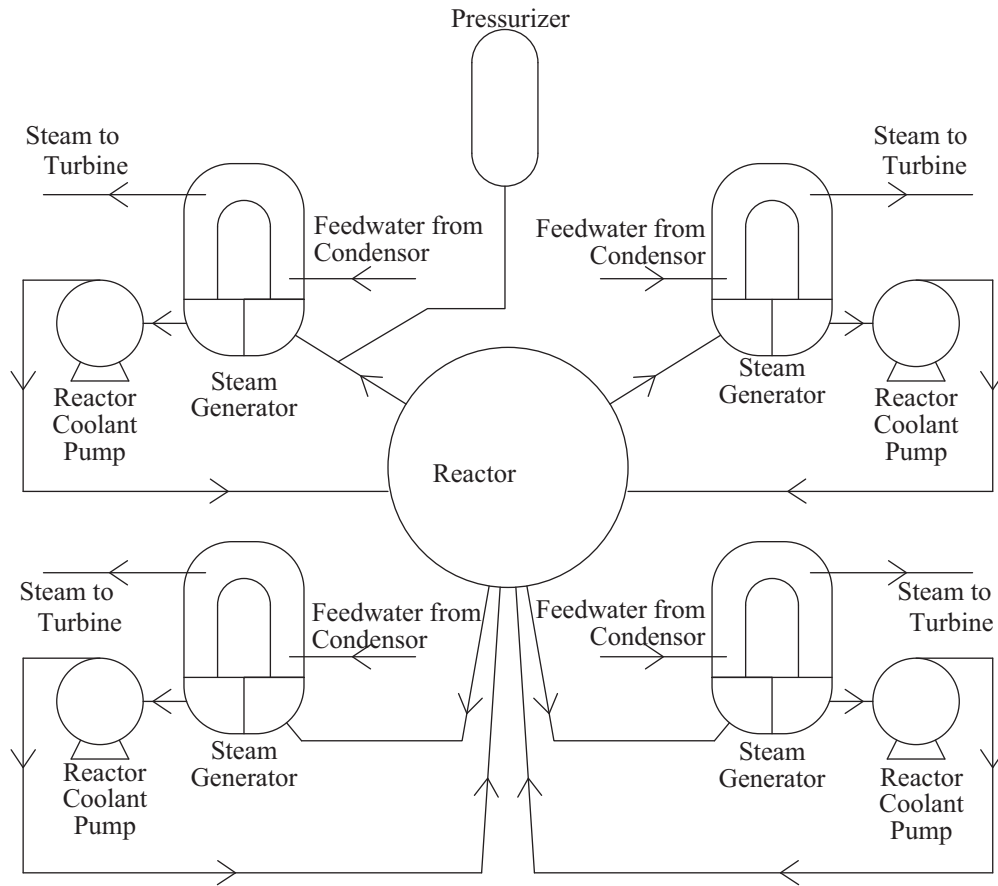


Fig. 6 Simplified schematic of a PWR RCS.

NSSS plus the condenser downstream of the turbines, the return feedwater pumps and associated piping. A wet steam mixture exits the top of the core and is directed into moisture separators and dryers. The dry steam has a quality of greater than 99.5%, suitable for entering the steam turbines, which power the generators to produce electricity. The expanded steam leaving the turbine is condensed and then returned to the RPV through feedwater pumps. [Fig. 7](#) shows the NSSS and the RCS for a BWR.

Water droplets in the wet steam leaving the top of the core are separated from the steam by the steam separators and dryer, located above the core and within the RPV. This water represents a significant fraction of the coolant within the RPV. A recirculation flow path internal to the RPV is established such that this water mixes with incoming feedwater flow and is returned to the core inlet. The separated water droplets do not have enough momentum to overcome pressure losses and recirculate. Therefore, upon separation from the steam flow, the water travels through the downcomer and into jet pumps, which add momentum. The returning water is the lower pressure “suction flow” in [Fig. 8](#).

Recirculation pumps take water from the downcomer, the annular region between the core shroud and the inner surface of the RPV wall, and direct this flow to the inlet of the jet pumps as the driving flow for coolant recirculation. [Figs. 8 and 9](#) show the components that enable internal recirculation. See also [Hamon \(2021\)](#).

The jet pump principle and the concept of recirculation within the RPV are unique to the BWR designs. Key features are that the jet pump contains no moving parts and can provide flow at the full range of flow rates to enable load following. The jet pumps provide enough momentum to supply about two-thirds of the recirculation flow within RPV ([General Electric, 1980](#), pp. 1–4). The principle is momentum exchange between water flows. The recirculation pumps provide the higher pressure “driving” flow into the jet pump driving nozzle ([Fig. 8](#)). The driving flow is accelerated by the area reduction at the nozzle outlet. The “suction” flow is the water returning from the separators and dryers. The suction flow enters at a lower pressure, which is further lowered as it is accelerated in the converging section outside of the driving fluid nozzle. The suction flow and the driving flow mix in the “mixing section”. Momentum transfer due to fluid mixing, along with development of a new velocity profile, cause a pressure recovery. The diffuser downstream of the mixing section slows the relatively high-velocity flow stream, converting part of the dynamic component of total pressure to static head, and thereby providing the flow mixture with appropriate velocity and head to flow into the lower head and return to the reactor core.

A key advantage of the recirculation concept is that regulation of the core power can be achieved by changing the speed of the recirculation pumps. A change in pump speed causes a change in the coolant flow rate through the core. With a higher or lower flow

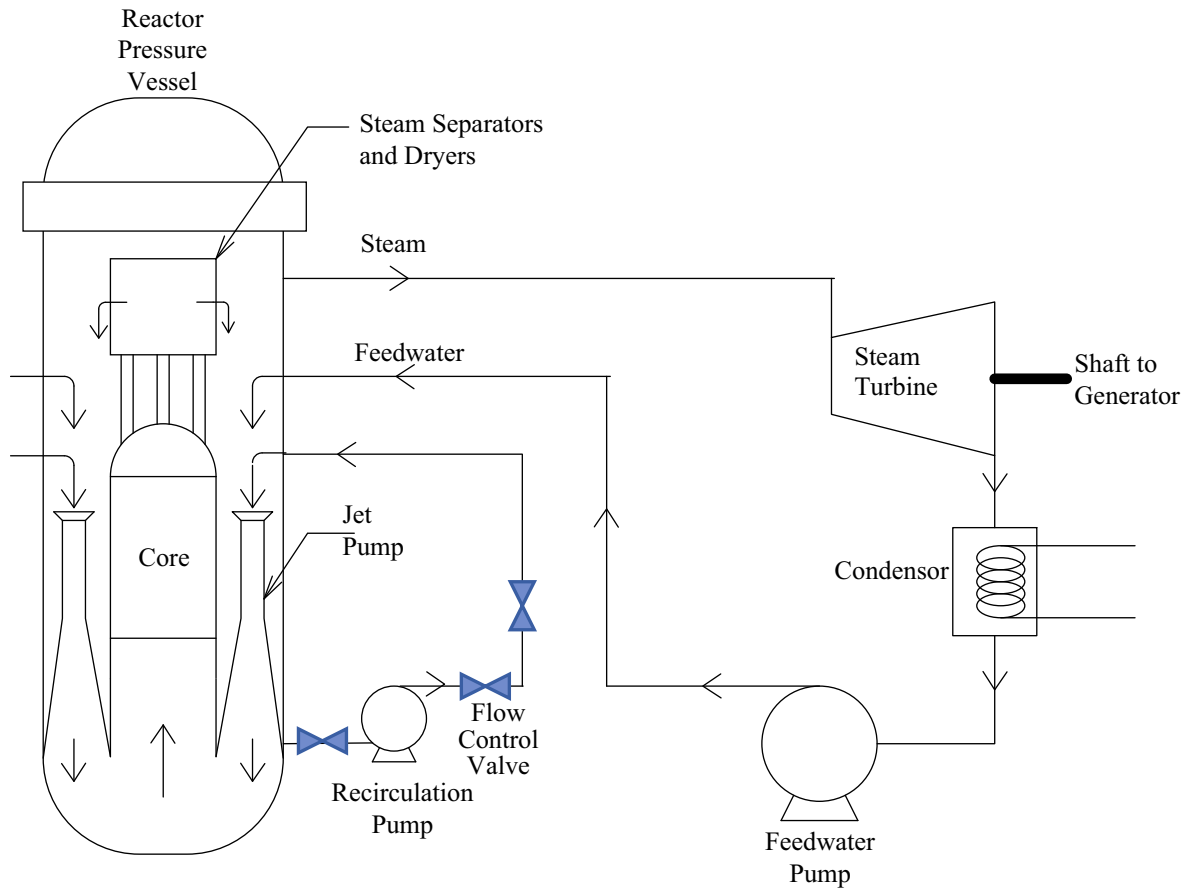


Fig. 7 Simplified schematic of a BWR reactor coolant system.

rate into the core, the amount of liquid coolant in the core region is changed. This, in turn, affects the amount of neutron moderation, thereby changing the core's reactivity and the core power level. This ability to change the core power without moving control rods or adding/deleting chemical shim as in PWRs (Brown, 2021) allows for more efficient usage of the nuclear fuel.

As described by Lahey and Moody (1993), the turbine is "slaved" to the reactor, meaning that the turbine demand follows the reactor steam output. This allows for the RPV pressure to remain approximately constant. To accommodate load changes (changes in steam demand by the turbine), the BWR can vary the reactor power level by flow control of the reactor coolant through the core, as described in the previous paragraph. This flow control concept is illustrated in Fig. 10.

The above discussions have been for active plants that require powered pumps for forced circulation of the coolant and heat removal by forced convection heat transfer. Reactor design techniques to promote natural circulation flow and natural convection heat transfer has matured to the extent that a reactor cooled by natural circulation of the coolant (no pumps!) has been Design Certified by the US NRC. This design is discussed further in the section on passive reactors.

Design requirements

The main objective of the thermal and hydraulic design of the core is to provide sufficient heat removal for the core power distribution such that fuel integrity is not sacrificed. As described in the Introduction, the heat transfer systems have functions to both promote power production and maintain nuclear safety. The design requirements for the RCS, therefore, have both safety design bases and power generation design bases. The following are typical safety and power generation criteria that must be met:

- Fuel integrity can be maintained for normal operation, Anticipated Operational Occurrences (AOOs) and transients considered within the NUREG-0800 Chapter 15 basis (US NRC, 2019b).
- The reactor can be maintained in a safe state for events within the Design Basis Event range.
- For BWRs: The reactor shows no signs of oscillations that could threaten the integrity of the fuel or the primary pressure boundary.

The fuel integrity can be maintained as long as the heat rate from a fuel rod does not exceed some limit. Tong and Weisman (1996) describe the major factors in maintaining fuel integrity:

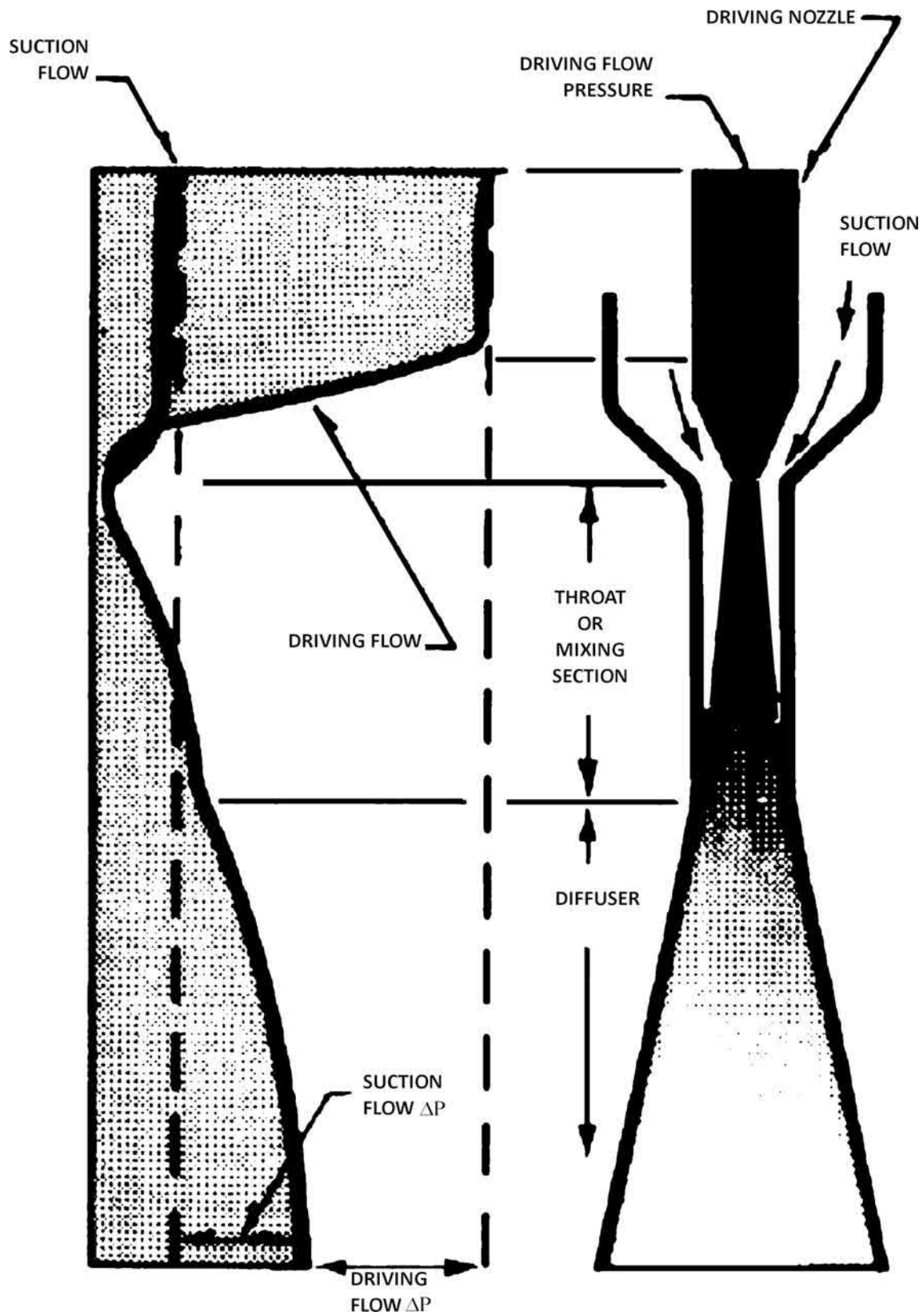


Fig. 8 Jet pump principle (General Electric Company, 1980). Courtesy of GE Hitachi Nuclear Energy Americas LLC.

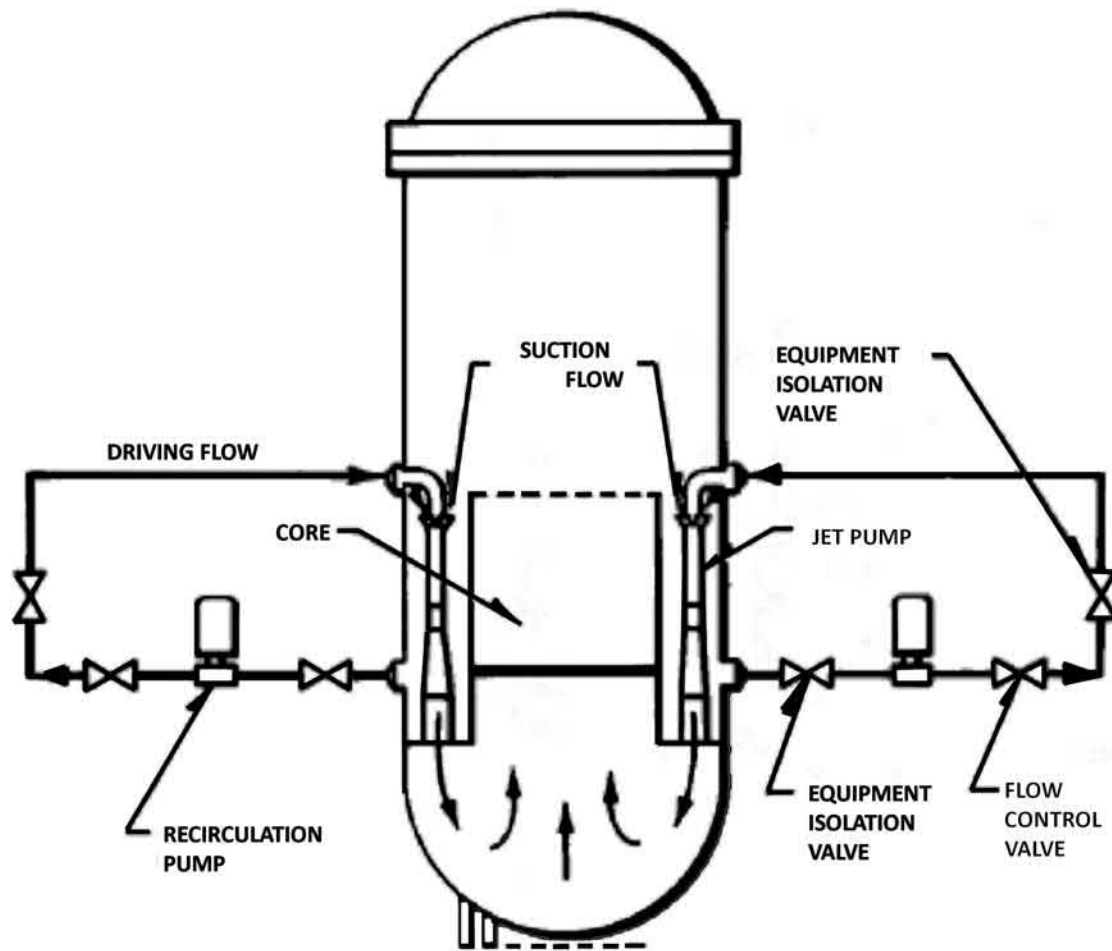


Fig. 9 BWR recirculation system with jet pumps (General Electric, 1980). Courtesy of GE Hitachi Nuclear Energy Americas LLC.

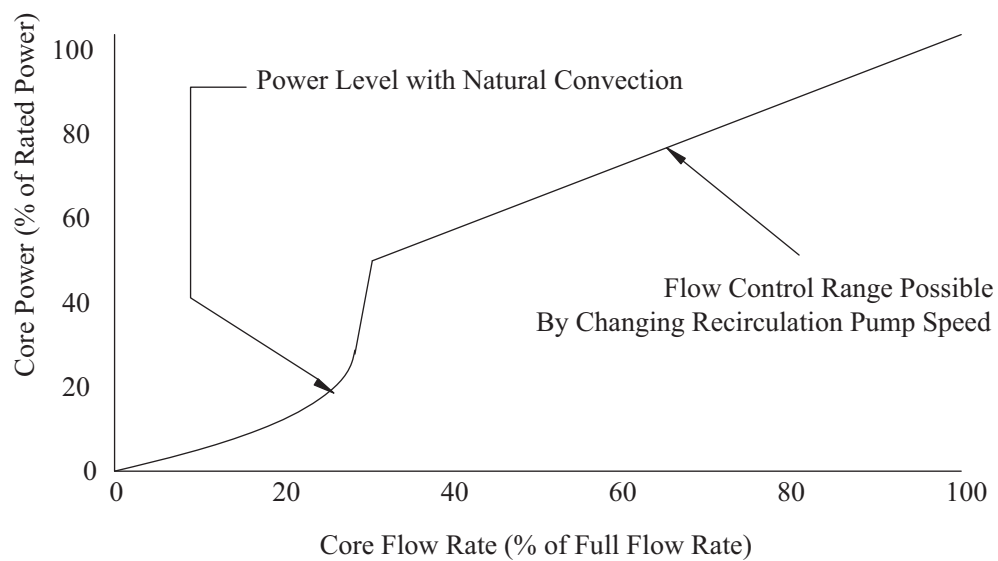


Fig. 10 Flow control concept of a direct-cycle BWR.

- Fuel centerline temperature well below the fuel melting temperature;
- Heat flux on the clad outer surface below a limiting value determined by the coolant conditions;
- Fuel burnup and fission gas release limited to avoid excessive internal pressure in the clad and clad creep or embrittlement;
- Heat flux low enough to avoid unacceptable behavior of the clad during postulated accident conditions;
- Appropriate power density; and
- Limitation on the rate of power change to avoid development of local stresses that are too large to be accommodated.

The limiting core power level can be determined by knowing where the “hot spot” is in the core, that is the location of highest temperature, and evaluating the heat removal rate needed to adequately cool this hot spot. Thus, knowledge of the radial and axial temperature distributions of the core are required. It should be noted that a reactor core typically experiences a very small fraction of fuel rods that “fail” during operation, or release radioactivity to the RCS. The US nuclear industry has led a successful campaign to reduce the fraction of “leakers” to on the order of less than 10 in 50,000 or 60,000 rods in the core.

Major heat transfer components of the RCS

The heat transfer phenomena and operation of the major RCS components in PWRs and BWRS are described below.

PWR heat transfer components of the RCS

PWR steam generator

The high-temperature coolant leaving the RPV passes through the PWR steam generator depositing the heat used to create steam for the main turbines. The steam generator, therefore, is a tube-and-shell heat exchanger designed to use heat from the higher temperature primary fluid on the tube-side to vaporize liquid on the secondary side into steam. The steam generator is a vertically-mounted pressure vessel with a tube bundle, through which the primary coolant travels, and steam separators and dryers for some designs. To be noted, some Russian-designed PWRs (VVERs) employ horizontal tube-and-shell heat exchangers as their steam generators. In a closed-loop design, the primary coolant makes two passes through the Westinghouse-type inverted U-tube steam generator and a single pass through the Once-Through Steam Generator design before being pumped back to the RPV. **Fig. 11** shows the evaporator section, housing the tube bundle, and the steam drum section, housing the moisture separation equipment for

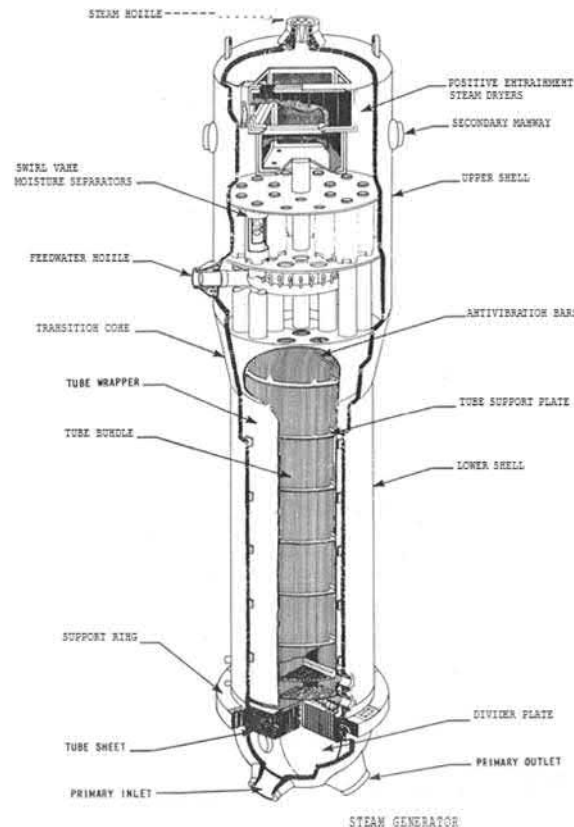


Fig. 11 Westinghouse steam generator (Cummins and Matzie, 2018).

a Westinghouse design. Fig. 12 illustrates the Babcock and Wilcock Once-Through Steam Generator design. Combustion Engineering employs a third design for its CE System 80 steam generator, with unique features being an axial flow feedwater economizer and steam outlet nozzles with venturi flow restrictors (Cummins and Matzie, 2018).

The higher-temperature primary coolant travels through a bundle of several thousand tubes. Heat is transferred from the primary coolant by thermal conduction through the tube walls to the secondary coolant. To minimize thermal resistance, the tubes are thin-walled and made of a high thermal conductivity material such as Inconel, a nickel alloy.

Most steam generators are natural circulation units on the secondary side. The secondary-side feedwater enters the steam generator subcooled. Heat from the primary side goes toward heating the secondary side coolant to the saturation temperature. As the secondary side water travels further along the tube bundle, the water vaporizes. The decrease of coolant density along the flow path promotes the natural circulation flow. In the inverted U-tube design, the water passes through steam separators and dryers before exiting to the main turbines. In the B&W design, the steam is superheated near the top of the tube bundle and exits with as much as 33°C (60°F) superheat. The thermal efficiency of the B&W design is comparable to that of other LWRs.

The primary and secondary flows are in countercurrent flow in both steam generator designs. Care must be taken in the design stage to have a primary-to-secondary side temperature difference for efficient heat transfer along the entire tube length. Fig. 13 illustrates the temperature difference in a once-through steam generator. The secondary-side water enters subcooled but holds at saturation temperature at the secondary-side pressure for most of the tube bundle length. (This figure does not include the superheater at the top of the tube bundle.) The primary-side water is subcooled for the entire tube length because of the higher primary-side pressure. The pinch point, where the two temperatures are closest, is a key factor in determining an appropriate tube length and heat transfer area.

For an inverted U-tube, the same type of diagram reveals that the flow is co-current along the riser side of the steam generator tubes and counter current along the second half of the tubes (Fig. 14). The x-axis is shown as tube axial location instead of enthalpy. The primary-side coolant is once again subcooled for the entire length. The pinch point is at P1 for the original design. If a preheater is added, the secondary fluid in the preheater section follows line B.

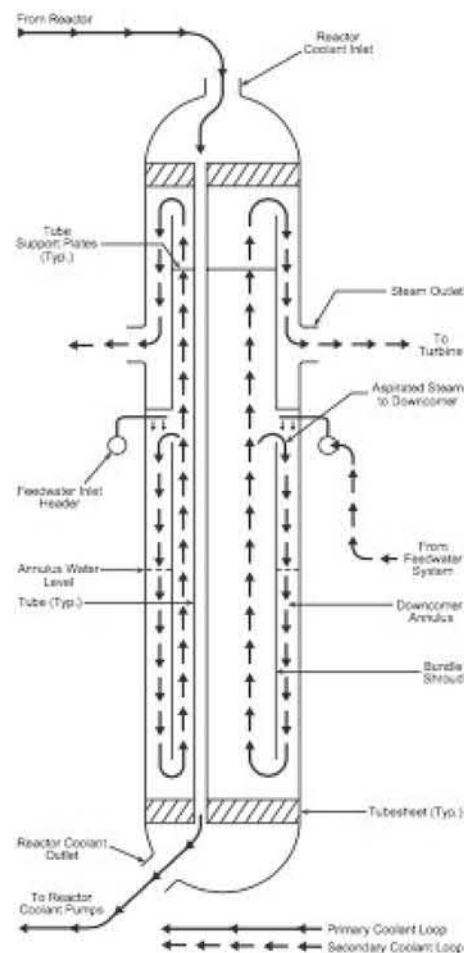


Fig. 12 B&W once-through steam generator (Cummins and Matzie, 2018).

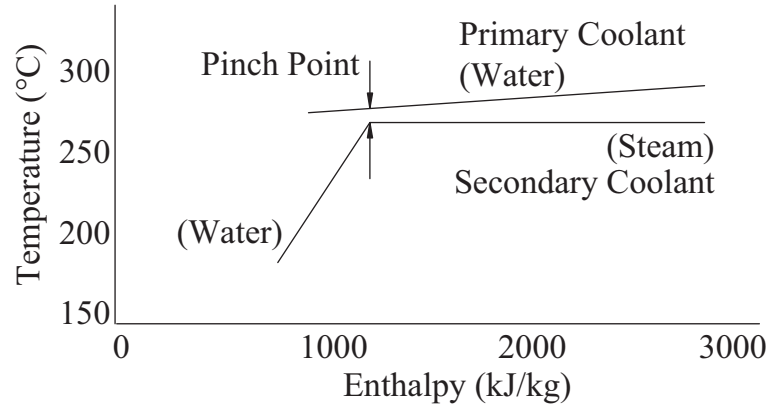


Fig. 13 Temperature variation in a once-through steam generator.

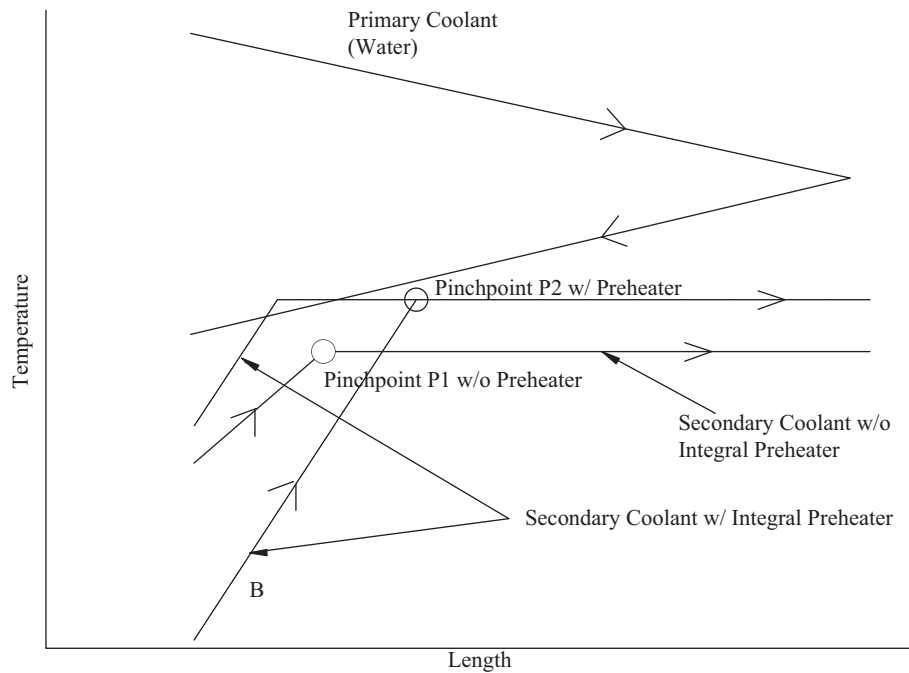


Fig. 14 Temperature variation in a U-tube steam generator.

The heat transfer resistance from primary to secondary side is the sum of the thermal resistances of the primary-side forced convection heat transfer, the tube wall conduction, fouling on the tube outer surface, and the secondary-side heat transfer.

$$\frac{1}{U} = \frac{A_{SGtube,outer}}{h_{inner}A_{SGtube,inner}} + \frac{r_{SGtube,outer} - r_{SGtube,inner}}{k_{SGtube}} \frac{A_{SGtube,outer}}{A_{SGtube,mean}} + f + \frac{1}{h_{outer}} \quad (7)$$

where

A = heat transfer area

h_{inner} = the primary-side forced convection heat transfer coefficient

r = tube radius

f = fouling factor

h_{outer} = the secondary-side heat transfer coefficient

The length on the secondary side which is not in boiling must be known. For this section, a forced convection heat transfer coefficient is used. Along the boiling length of the secondary side, boiling coefficients are appropriate.

Fouling on the outer side of the tubes must be considered. Primary-side fouling is generally considered negligible because of the extremely high level of water chemistry control. However, purity on the secondary side is not as high and some fouling can be present. To more accurately assess heat transfer rates, a thermal resistance term for fouling must be included, as shown above.

Operational issues for the U-tube steam generators include localized corrosion in low velocity zones on the secondary side where sludge can deposit. Improvements to the secondary-side water treatment have largely eliminated this problem. The addition of baffles to increase the velocity in these areas have also reduced the problem. In the 1970s, Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600 tubes was observed. Tube cracking was found in areas of high stress, such as in the 180° bend of the first row, the tightest row, of tubes and where tubes transition in size at the tube sheet entrance. Changes in the manufacturing processes and the use of Inconel 690, which is more resistance to PWSCC, have reduced this problem.

The once-through steam generators have shown problems with thermal stresses, since the straight tubes have less freedom to expand axially than the U-tube bundles. These long tubes may expand several inches when heated. The higher section of the tubes is at about the primary temperature while the lower portion is at the saturation temperature. Greater care is now given to minimize the difference in amount of thermal expansion of the tube bundle and the shell, especially during full-load operation (Tong and Weisman, 1996).

PWR pressurizer

The pressurizer has the function of regulating the primary-side pressure. The pressure is maintained at a set value during steady-state operation and pressure transients are mitigated during transients and some accidents. Only one pressurizer is needed in the primary side since the RCS should be full of water everywhere except in the vapor space of the pressurizer. Therefore, a change in pressure in the pressurizer will propagate throughout the entire RCS (nearly) immediately because water is essentially incompressible.

Fig. 15 shows a schematic of a Westinghouse-design pressurizer. The vessel is a tall, narrow vessel to enable water level change, and thereby, a change in elevation head (ρgh). Designs by other vendors are very similar. To maintain a constant operating pressure, the pressurizer water is to be kept at saturation temperature by electric heaters installed in the bottom of the pressurizer. To mitigate a pressure decrease, the heaters can heat the water and increase vapor pressure.

The pressurizer water level can increase or decrease in response to a change in primary water temperature change. This water level change affects the vapor space pressure. To mitigate a pressure rise, sprays at the vessel top release water droplets to condense steam and decrease the vapor pressure. The spray water source is one of the cold legs, which supplies water at less than saturation temperature. As a last resort, if the sprays are unable to keep the pressure below a setpoint value, then relief valves can be opened at the top of the pressurizer to vent steam into the containment.

The relevant heat transfer processes in the pressurizer are:

- vaporization from the water surface due to depressurization (and water level decrease)
- condensation in the vapor space due to pressurization (and water level rise)
- steam condensation onto water droplets due to operation of the sprays
- condensation on the pressurizer vessel wall inner surface.

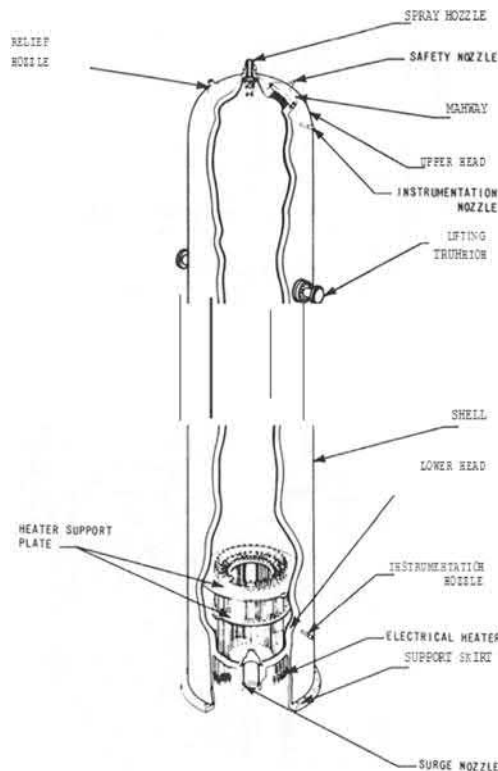


Fig. 15 Westinghouse pressurizer (Cummins and Matzie, 2018).

The control volume approach with the assumption of thermodynamic equilibrium between the vapor and liquid can be assumed for mild transients (Todreas and Kazimi, 2012, p. 340). For more rapid transients, the two phases cannot be assumed to be in equilibrium and an analysis by a method such as the two-fluid model gives more accurate results.

Condenser

The description of the condenser herein is common to PWR and BWR cycles.

The condenser is a device that transfers heat from the steam exhausted from the main turbines to a secondary-side fluid within the condenser. The steam expands in the turbine from a very dry state to a wet steam mixture. This wet steam mixture no longer carries enough energy to efficiently transfer further thermal energy to the turbine. Rather, the wet steam is condensed in the condenser to single-phase liquid water, pumped back to the secondary-side pressure, and returned to the steam generator.

The condenser is an essential heat transfer system because it brings the wet steam to a state that it can be pumped up to the secondary-side pressure and returned to the steam generator as slightly subcooled feedwater. The relevant heat transfer mode is phase change (wet steam to single-phase liquid water). To maximize the amount of work done by the turbine and increase the thermal efficiency of the Rankine cycle, the condenser operates at a partial vacuum (0.067 MPa, or 20 in. of mercury, or less). For a condenser operating at 0.02 MPa (5.9 in. of mercury), the corresponding saturation temperature is about 60 °C. The condenser is usually a single-pass heat exchanger with thousands of straight-through tubes.

Cooling towers and ultimate heat sinks

An Ultimate Heat Sink is the destination for rejection of unused thermal heat from the power plant. A cooling tower is a heat exchanger that transfers the condenser rejection heat into the air instead of into a body of water. The description herein of these components is common to both PWRs and BWRs.

The heat removed from the steam by the condenser is no longer used in the Rankine cycle and is exhausted to the environment. The Ultimate Heat Sink is either a cooling tower or a large body of water such as a lake, river or ocean. Rejection of the heat from steam condensation may be done in one of three ways (World Nuclear Association, 2020):

- The simplest method is “direct” or “once-through” cooling. If the power plant is near a large body of water, the heat may be removed by running water through the condensers and discharging the heated water back to the body of water.
- Another method is by dry cooling. This method, by which air removes the heat by forced convection, is less commonly employed. The air flow may be in a closed loop. This method would be used only by plants not located near a large body of water.
- The third method is “wet recirculating” or “indirect cooling” (US Department of Energy, 2011). In wet recirculating systems, the water in the condenser which removes heat from the steam is pumped to a cooling tower. The heat from the warm water is removed by air flowing upward through the cooling tower. Evaporation is the main heat transfer mode, with a smaller rate of heat removal by forced convection to the air flow. A small fraction of the warm water, 3–5% per the World Nuclear Association (2020), evaporates from the cooling tower and appears above the cooling tower as the white vapor plume which characterizes the cooling tower appearance. The draft induced by reduced air density and buoyancy of the lighter air provides natural pumping action for these flows. Completing the cycle, the cooled water is returned back to the condenser. Due to the evaporation of some of the cooling water, buildup can occur of minerals and sediment in the water that could degrade performance. A fraction of the cooling water is continuously removed and replaced to prevent buildup and performance degradation.

To be noted, this natural draft reduces the amount of pumping power required and involves only nonradioactive water/steam. This attribute of energy conservation may seem ironic because of the “negative” image of nuclear power that the hyperbolic cooling towers can symbolize.

- The US Department of Energy (2011) also lists a fourth method, referred to as a “cooling pond”. A cooling pond is essentially the same as a wet recirculating system, however, natural conduction and natural convection heat transfer from the water to the atmosphere is used together with evaporation to cool the recirculating water.

According to the US Department of Energy (2011), as of 2011, 38.1% of US nuclear power plants used once-through systems, 0% used dry cooling systems, 43.6% of US nuclear power plants used wet recirculating systems and 18.3% used cooling ponds.

It should be noted that all of these systems are designed such that the heat rejected to the environment, be it a body of water or the air, is not harmful to the surroundings.

PWR main coolant pumps

The PWR main coolant pumps are all centrifugal pumps that operate according to a pump curve as that shown in Fig. 16. They are single-stage, vertically mounted pumps designed to move large flow rates of primary cooling water (as much as 6.2 m³/s, or 100,000 gpm!) through the primary loop under high pressure and temperature conditions. The optimal operating point for the pump is at the intersection of the pump curve and the system characteristic curve.

The pumps must have sufficient Net Positive Suction Head (NPSH) at the pump suction (pump inlet). The pressure at the entrance to the impeller of a centrifugal pump is lower than that at the pump suction. Cavitation, or bubble formation, at the impeller is possible if the local pressure falls below the vapor pressure of the fluid corresponding to its local temperature. Cavitation is detrimental to the performance and integrity of the pump and therefore the pumps must be operated with sufficient NPSH to avoid this situation. Locating the pumps on the cold legs is one means to increase the NPSH on the pumps and avoid cavitation.

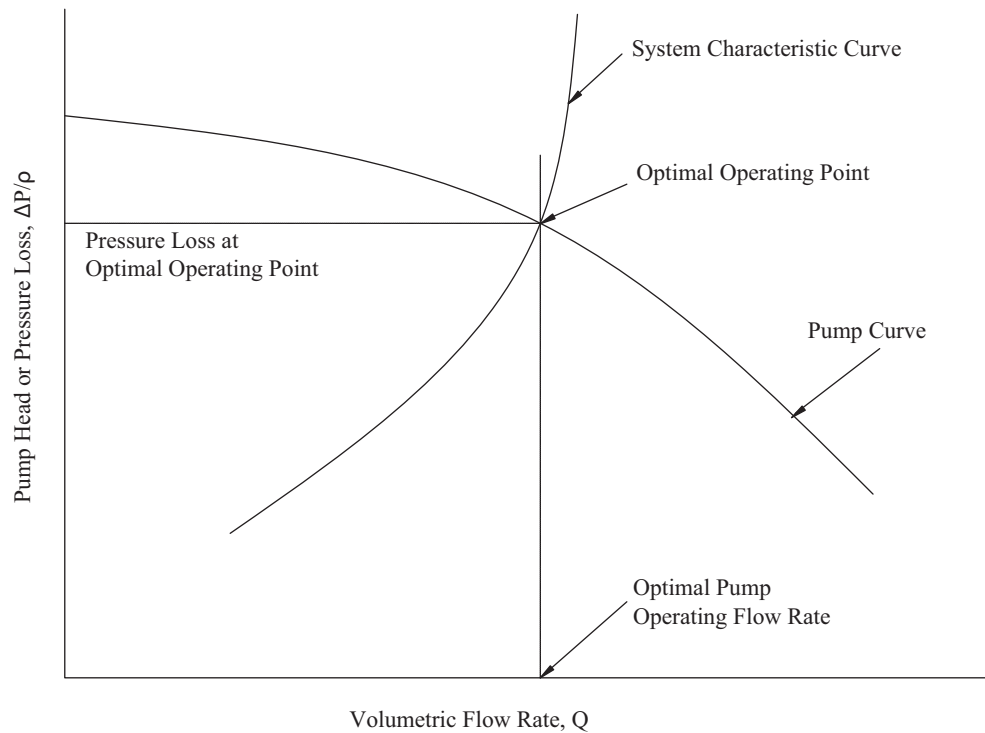


Fig. 16 Pump curve for PWR main coolant pumps.

PWR piping and valves

The piping and valves on the primary loop must withstand the high temperature and pressure environment for the life of the reactor. To meet this requirement, they are made of carbon steel with a stainless steel clad or are made of stainless steel. The pressurizer surge line from the main steam line and other pressurizer connections and lines are of austenitic stainless steel. Most joints and connections are welded as permanent connections instead of flanged.

Being made of metal, all of these components and the RPV are good thermal conductors. As such, their outer surfaces will be within a few degrees of the temperature of the primary flow. Heat loss to the surrounding containment gas space would be by thermal radiation and natural convection and would be significant. These heat losses must be reduced to enable more efficient use of fission heat and to maintain the containment conditions within acceptable ranges.

Piping and valves that operate at high temperature are insulated to reduce heat losses. To prevent the development of high thermal stress spots due to rapid changes in fluid temperature during a transient, thermal sleeves are installed. The RPV is often insulated with a reflective metal insulation system (US NRC, n.d.).

BWR heat transfer components of the RCS

BWR steam separators and dryers

In contrast to the indirect cycle of the PWR, the coolant in a direct-cycle BWR is vaporized within the active core region by heat transfer from the clad outer surface. The BWR does not employ steam generators external to the RPV.

To ensure adequate cooling of the fuel, the core and RCS are designed such that the fuel rods are always covered with water. The upper portion of the active core is within a two-phase steam-water mixture: the fuel rods are “wetted”, or covered by a liquid film, while the bulk of the fluid is a two-phase steam-water mixture. Fig. 17 (Young, 2021) illustrates the two-phase configuration in a BWR core region. Because the heat transfer coefficient associated with water vaporization is very high, the fuel rods are sufficiently cooled when they are wetted.

The basic physics promotes this favorable distribution of steam and water within the two-phase region. Following the path of least flow resistance, the higher velocity steam flows through the center of the flow channels and away from the fuel rods which impose a frictional wall resistance. The lower velocity liquid is pushed away from the flow channel center and toward the fuel rod outer surfaces. Some water is also entrained as droplets by the steam to form a two-phase wet steam mixture. The requirement that the fuel be wetted places a limitation on the core height. The core height is specified such that the water inventory does not boil off and result in dryout of the clad outer surface. With this two-phase distribution, the two-phase portion of the active fuel region is efficiently cooled by forced convection boiling even at the core exit elevation.

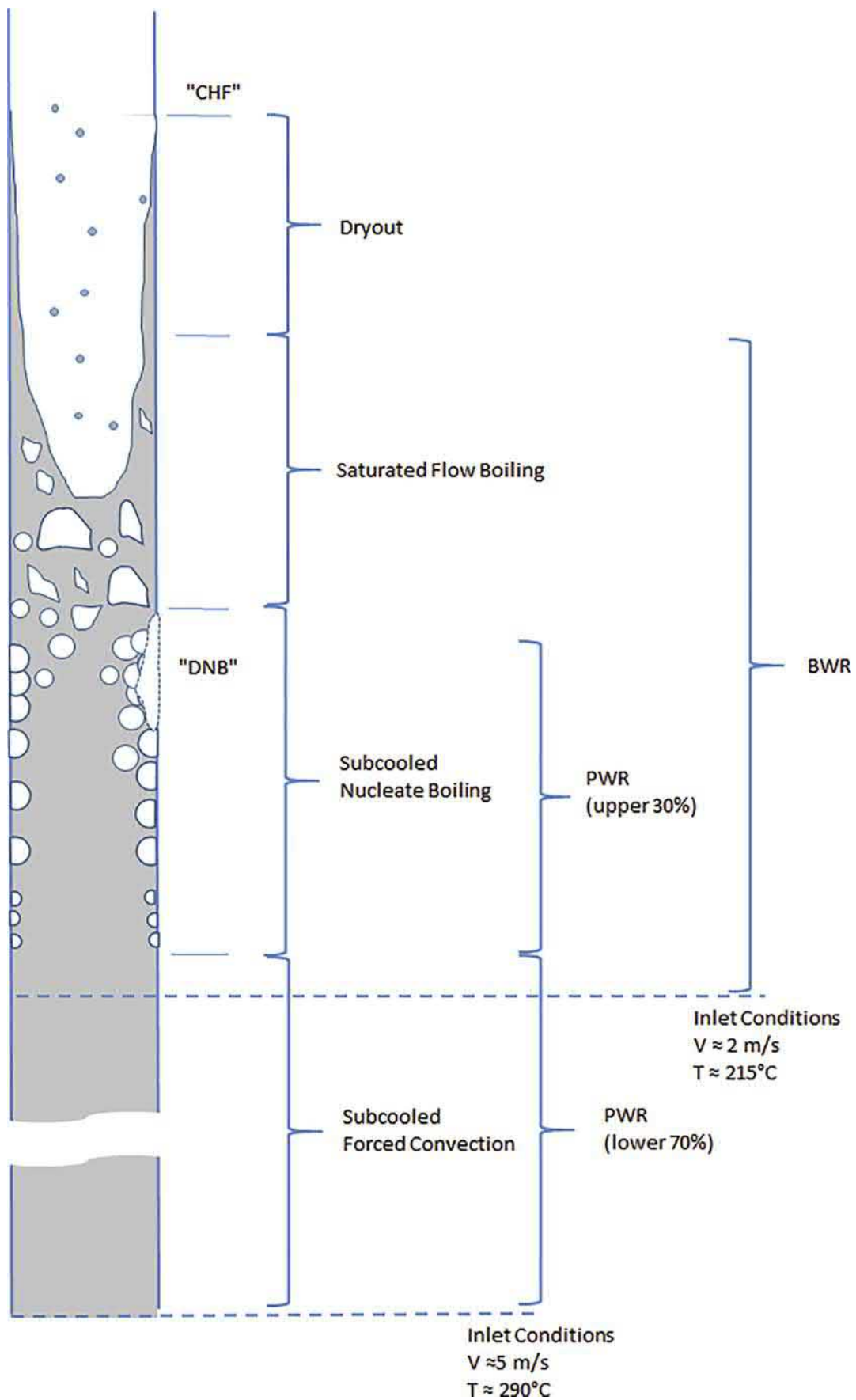


Fig. 17 Two-phase configuration in the BWR core region (Young, 2021).

Wet steam should not be ingested by the turbines due to the high potential for turbine blade damage by water droplets. The quality of the steam at the core exit is low, ranging between about 10–15% depending on the reactor. To dry the steam, separators and dryers are installed in the BWR RPV above the core.

BWR condenser

The description of the condenser is common to PWR and BWR cycles. The reader is referred to the earlier discussion of the condenser for the PWR RCS components.

Cooling towers and ultimate heat sink

The description of the cooling towers and Ultimate Heat Sink is common to PWR and BWR cycles. The reader is referred to the earlier discussion of these components for the PWR.

BWR pumps

The BWR uses Feedwater Pumps to return primary coolant (“feedwater”) from the condenser back to the RPV, Recirculation Pumps to provide driving flow to the recirculating coolant within the RPV downcomer, and jet pumps to provide most of the motive force for recirculating flow. The Feedwater Pumps are steam-driven pumps with a capacity less than that of the pumps for a comparable PWR due to the recirculating flow within the RPV. They operate at about half the pressure of the PWR Main Coolant Pumps. The Recirculation Pumps and jet pumps have been described in detail earlier.

BWR piping and valves

The piping and valves on the primary loop must withstand the high temperature and pressure environment for the life of the reactor. To meet this requirement, they are made of carbon steel with a stainless steel clad or are made of stainless steel. Most joints and connections are welded as permanent connections instead of flanged.

Being made of metal, all of these components and the RPV are good thermal conductors. As such, their outer surfaces will be within a few degrees of the temperature of the primary flow. Heat loss to the surrounding containment gas space would be by thermal radiation and natural convection and would be significant. These heat losses must be reduced to enable more efficient use of fission heat and to maintain the containment conditions within acceptable ranges. Piping and valves that operate at high temperature are insulated to reduce heat losses to the containment.

Heat removal systems during normal shutdown and following primary system breach

As mentioned earlier, the goal of nuclear safety is to protect the public’s health in the unlikely event of a release of radioactivity. Radioactive fission products will not be released if the fuel integrity is maintained. To prevent fuel failure and maintain a safe state in a nuclear system, system components must be kept within acceptable temperature limits. Adequate heat removal from the nuclear fuel by heat transfer systems prevents fuel failure and subsequent release of radioactivity, and thereby, promotes fulfillment of the nuclear safety goal.

Adequate heat removal is achieved by maintaining the RPV water level above the top of the active fuel at all times, including after termination of the nuclear fission reaction by control mechanisms. The heat removal mechanisms during normal operation have been described earlier. Additional heat removal systems are in place for normal shutdown cooling at high and low pressure and the ECCA. Each of these systems is described below.

From a heat transfer perspective, these systems are reconfigurations of a Rankine system components (piping, heat exchangers, turbines, pumps) in order to transport the core heat under a variety of thermodynamic conditions (ranging from high pressure to low pressure) and flow conditions (ranging from steam flow rate at full core power to steam flow rate corresponding to decay heat rate). Some configurations involve only single-phase heat transport (forced convection or natural convection heat transfer) while others involve two-phase heat transport (forced convection with boiling and/or condensation). The PWR steam generators were noted earlier to involve natural circulation. The technology to enable natural circulation flow and natural convection heat transfer in reactors has matured to the extent that a fully passive reactor has been Design Certified by the US NRC. These so-called “passive” systems belong in a class of their own and are described in a later section.

Normal shutdown cooling at high pressure for PWRs

Under normal shutdown cooling, the nuclear fission reaction has been essentially terminated and the steam generation rate in the steam generators is greatly reduced. The secondary coolant heat removal system is rerouted such that the turbine is bypassed and steam is directed from the steam generators to the main condenser. The condensate and feedwater systems return coolant to the steam generator secondary side as in normal operation. All systems are operating at much lower flow rates than for operation at reactor power because the largest contributor to heat generation is now heat released from decay of radioactive fission products.

Normal shutdown cooling at low pressure for PWRs

After an initial cooldown to 177 °C (350 °F) and depressurization to about 2.8 MPa (400 psig) (US NRC, n.d.), the Residual Heat Removal (RHR) system takes over to provide shutdown cooling. The steam generators and secondary loop used in power operation are no longer employed. Instead, primary reactor coolant from a hot leg is directed to RHR heat exchangers and returned through cold legs to the RPV. The secondary-side heat removal is provided by a closed-loop, single-phase system, named the Component Cooling Water System (CCWS), which rejects heat to a tertiary loop, and finally to the Ultimate Heat Sink. All heat transfer is single-phase, forced convection heat transfer except as dictated by the equipment for heat transfer to the Ultimate Heat Sink.

Normal shutdown cooling at high pressure for BWRs

For normal shutdown cooling at high pressure, earlier vintage BWRs and some newer passive BWR designs are equipped with Isolation Condensers, while later non-passive BWRs use the Reactor Core Isolation Cooling (RCIC) System.

In the Isolation Condenser system, an elevated, vertical tube bundle takes steam from the RPV into the top of the tube bundle and returns condensate from the bottom of the tube bundle back to the RPV. For this natural circulation loop, a small pressure difference drives flow to the condenser and gravity returns condensate to the vessel. As its name implies, the system was designed for use when the RPV is “isolated” from the main steam turbines. The tube bundle sits in one of several water compartments outside of containment. Heat from the condensed steam boils water on the secondary side and the nonradioactive steam is vented to the Ultimate Heat Sink. A key feature of the Isolation Condenser System is that heat is that decay heat is transferred outside of containment while no coolant mass leaves the primary system.

The RCIC System takes decay heat steam from the RPV to provide motive energy to a Terry steam turbine. The turbine is on the same shaft as the RCIC pump and therefore powers the pump to provide cooling water to the RPV. The RCIC pump takes suction from the Condensate Storage Tank. If the RCIC System runs long enough to deplete the Condensate Storage Tank, the pump suction is realigned to the bottom of the Suppression Pool water volume in containment. Steam flowing into the Terry turbine is exhausted as wet steam to the Suppression Pool. The RCIC System gained recent attention due to performance far in excess of its expected capabilities during the accidents at Fukushima Dai-ichi Units 2 and 3 (Solom and Vierow Kirkland, 2015). Since RCIC turbine steam is vented to the Suppression Pool and RCIC pump suction is from the Suppression Pool, a cooling system is in place to transfer heat from this pool to the Ultimate Heat Sink and maintain acceptable containment pressure.

Normal shutdown cooling at low pressure for BWRs

The RHR System provides cooling once the primary system has been depressurized to about 0.95 MPa (125 psig) and undergone an initial cooldown. Configured with the recirculation loops shown in Fig. 9, the RHR System removes decay heat by passing primary coolant water through RHR heat exchangers and returning the cooler water back to the RPV. The RHR shutdown cooling suction line is aligned with the suction side of one of the recirculation pumps to extract primary coolant. After passing through the RHR heat exchangers, the coolant is returned to the RPV either through the recirculation line on the discharge side of a recirculation pump or via a main feedwater line.

Emergency core cooling systems (ECCS)

Following a hypothetical failure of the primary pressure boundary, water is lost from the RCS and core cooling must be restored. After the Reactor Protection System has served to terminate the nuclear fission reaction, Engineered Safety Features (ESF) are called upon in LWRs to provide continued heat removal. The ESFs are safety systems that provide a safety function to mitigate consequences of accidents that may cause major fuel damage and cannot be accommodated by the Reactor Protection System alone.

Adequate heat removal is achieved by:

- Terminating the heat generation. Heat generation through the nuclear fission reaction can be stopped by control mechanisms. However, heat generation by radionuclide decay of fission products in the fuel cannot be stopped. Therefore, an additional function is needed:
- Providing heat removal

The ESF which provides this heat removal is the ECCS.

ECCS design bases

According to 10 CFR 50, Appendix K (US NRC, 2017a), the ECCS must be able to achieve its function with certain equipment damage. The single-failure criterion states that, “Single Failure Criterion. An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.”

The acceptance criteria for LWR ECCS as codified in 10 CFR 50.46 (US NRC, 2017b) as:

- Peak fuel clad temperature is not to exceed 1200 °C (2200 °F).
- Total cladding oxidation is not to exceed 0.17 times the total clad thickness.
- Total amount of hydrogen generation from cladding chemical reaction is not to exceed 0.01 times the total hydrogen generation if all of cladding were to react.
- Any core geometry changes must leave core in a coolable state.
- Appropriate long-term cooling must be provided to keep the core at acceptably low temperatures

The first two criteria assure that the cladding remains sufficiently intact to retain the fuel pellets and radioactive fission products. The third criterion assures that hydrogen gas will not be generated in amounts that could combust.

ECCS for PWRs

In a PWR, the ECCS both provides core cooling and supplies additional neutron poisons. The latter function ensures that the core is in a shutdown mode by adding large amounts of borated water to the solid water system. This second function is accomplished by the injection of large amounts of cool, borated water into the RCS to help ensure that the reactor remains shutdown. This borated water source is called the Refueling Water Storage Tank (RWST).

Multiple systems are available to provide core cooling water and ensure adequate cooling, as determined by heat and mass balances. All PWR ECCS include pressurized Safety Injection Tanks (SITs) and high-pressure and low-pressure safety injection (HPSI and LPSI) pumps. The cooling water is injected via a hot leg, a cold leg or directly into the RPV, depending on the reactor design and accident stage.

A Loss of Coolant Accident (LOCA) scenario can be divided into the ECCS injection phase and the ECCS recirculation phase. The ECCS injection phase involves ensuring adequate water supply and heat removal (heat and mass balance). During a large-break LOCA, the primary system rapidly depressurizes and water is supplied by the safety injection accumulators once the primary pressure falls below the accumulator pressure. Both HPSI and LPSI pumps are aligned to deliver water from the RWST to the primary side. During a small-break LOCA, because of the slower loss of coolant inventory from the primary side, an energy balance is achieved between the heat generation rate in the core and the total of heat loss rates through the break, steam generators and other systems. Makeup coolant can be provided in some PWR designs by the HPSI pumps.

After the RWST supply has been depleted, the recirculation mode of ECCS operation is commenced. The containment sump has collected a large volume of water discharged during the LOCA and from operation of containment sprays. The LPSI pumps are reconfigured to take suction from the containment sump instead of from the RWST. For a large LOCA, the recirculation will be in a low-pressure mode, while for a small LOCA, the system may be operating at either high pressure or low pressure. Heat exchangers, if available for operation, may be used in the low-pressure safety injection system to transfer heat to the Ultimate Heat Sink via intermediate heat transfer systems.

ECCS for BWRs

In a BWR, the ECCS includes a system that injects coolant via piping or spray nozzles into the RPV at high pressure; one or more systems which can inject or spray coolant into the RPV at low pressure; and an Automatic Depressurization System (ADS). The High-Pressure Coolant Injection (HPCI) system and the High-Pressure Core Spray (HPCS) system have pumps that can overcome the large pressure difference to provide coolant at up to the normal operating pressure of the RPV. The Low-Pressure Coolant Injection (LPCI) system and the Low-Pressure Core Spray (LPCS) are low pressure systems intended to respond to large LOCAs by supplying makeup coolant at larger flow rates than the high-pressure systems can. The ADS depressurizes the primary side, enabling the low-pressure ECCS systems to actuate and provide core makeup water when the high-pressure systems are unable to adequately maintain the cooling function.

Mass and energy balances dictate which ECCS systems are needed when. While the low-pressure systems can add coolant more rapidly to the primary system, the trade-off of rapidly releasing primary coolant to the containment that are associated with ADS operation must be considered when determining when to transition from high-pressure ECCS systems to low-pressure ECCS systems. Brief descriptions of the physics of each type of ECCS subsystem follow. Fig. 18 shows a schematic of the ECCS for a BWR/6, a more recent BWR design.

High-pressure ECCS cooling systems in BWRs

The HPCS is employed at high pressure in the more recent BWR/5 and BWR/6 designs. The intention of the HPCS is to provide makeup water and depressurize the primary system. The HPCS prevents fuel failure if the coolant level were to fall below the Top of Active Fuel by spraying water droplets down onto the core, thereby maintaining adequate cooling of the fuel clad. The spray is injected from a ring of nozzles above the reactor core. The water droplets that fall on the fuel assemblies remove heat by forced convection and vaporization. The HPCS reduces the primary pressure by steam condensation on the water droplets. Heat and mass are transferred from the primary system to the Suppression Pool in the containment by opening of the primary Safety/Relief Valves. While most of the normal shutdown systems discussed earlier were closed systems, the HPCS initially operates as an open loop, drawing water from the Condensate Storage Tank. If the Condensate Storage Tank water supply runs low, water is taken from the Suppression Pool and the system becomes closed. Suppression Pool cooling becomes necessary to transport heat to the Ultimate Heat Sink.

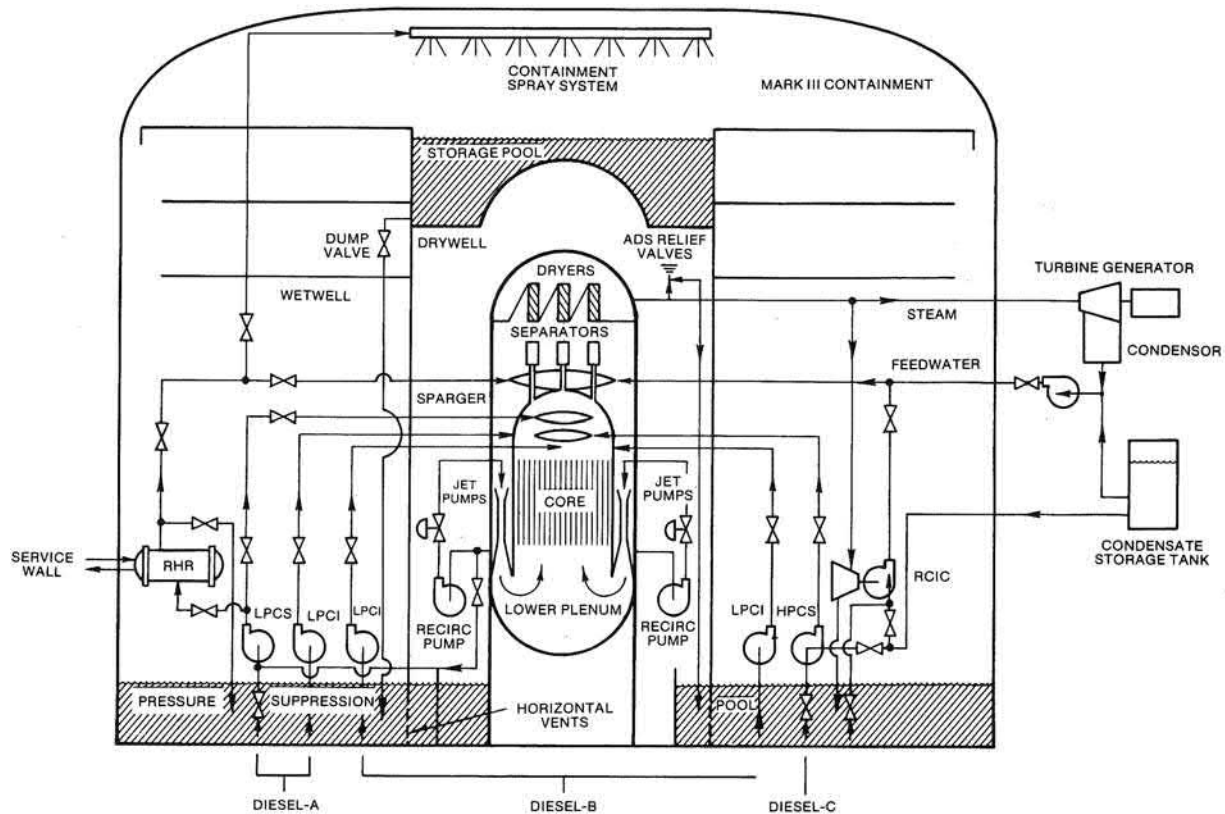


Fig. 18 BWR/6 ECCS (Lahey and Moody, 1993). Copyright 1993 by the American Nuclear Society, La Grange Park, Illinois.

The HPCI System is employed at high pressure in the earlier BWR/3 and BWR/4 designs. The HPCI System cools the core by forced convection heat transfer and does not have the same depressurization function at the HPCS. Similar to the HPCS, the HPCI System initially runs on an open cycle. The HPCI System takes water from the Condensate Storage Tank and injects into the RPV through one of the main feedwater lines. The HPCI System pump is powered by a steam turbine that ingests steam from one of the main steam lines and exhausts to the Suppression Pool. Heat and mass are transferred from the primary system to the Suppression Pool in the containment by opening of the primary Safety/Relief Valves. If the Condensate Storage Tank water supply runs low, water is taken from the Suppression Pool and the system becomes closed. Suppression Pool cooling becomes necessary to transport heat to the Ultimate Heat Sink.

Low-pressure ECCS cooling systems in BWRs

All BWRs have low-pressure systems to provide coolant makeup in response to a large LOCA. Except for Dresden I, all plants can depressurize via Safety/Relief Valve discharge to the Suppression Pool.

All BWRs have a LPCS system that delivers water from the Suppression Pool to one or more spray spargers located above the reactor core. Similar to the HPCS, the function of the LPCS System is to prevent fuel failure if the coolant level were to fall below the Top of Active Fuel by directing jets of water into the fuel assemblies, thereby maintaining adequate cooling of the fuel clad.

The LPCI System in BWR/3 and later designs provides the ability to "flood" the core, that is, supply water to raise the coolant level above the top of the core. The probability of fuel failure is much lower when the core is flooded because fuel is in contact with coolant at all locations and should be adequately cooled by forced convection heat transfer or subcooled boiling. Water is supplied by the Suppression Pool to the RPV. In some BWR/3 designs, the LPCI System is a dedicated system that injects water via one of the recirculation loops. In later BWR/3 designs and all BWR/4 to BWR/6 plants, the LPCI function is one of the functions of the "multi-mode" RHR System. That is, the RHR System equipment is used for LPCI operation. Coolant is injected via one of the recirculation loops or directly into the RPV, as shown in Fig. 18.

Heat removal systems for passive light water reactors

Passive systems are those that rely on natural forces such as gravity and small pressure differences to drive natural circulation flow and heat transfer and do not depend on operator actions. The use of natural forces promotes increased system reliability. Passive

systems are usually simpler in design, have fewer parts for easier/lower cost maintenance and do not depend on active power sources for heat removal. Differing from active systems which often have forced convection as the dominant heat transfer mode, the heat transfer modes in many passive systems are natural convection heat transfer or condensation of steam on cool containment surfaces or in submerged tube bundles.

A key feature of reactor designs with passive safety systems is large elevated pools of water that can drain into the RPV to supply core coolant. They may also have large heat transfer surfaces, such as the inner surfaces of containment structures or tube bundles, on which steam released from the primary or secondary loop can condense and transfer heat outside of containment.

Passive safety features of heat transfer systems in leading PWR and BWR designs are summarized below.

AP1000 passive heat transfer systems

The AP1000 was designed by Westinghouse Electric Company and has had four units operational in China since 2018, with two more units nearing completion in the US (Vogtle site). It is a Gen III+ reactor with a rated power output of 1100 MWe (Cummins and Matzie, 2018). Figs. 19 and 20 illustrate the AP1000 passive safety systems for core cooling and containment cooling respectively.

The AP1000 achieves long-term passive heat removal with a Passive Residual Heat Exchanger (PRHR). The PRHR employs a large heat exchanger in an elevated water pool in containment, the In-Containment Refueling Water Storage Tank (IRWST). The PRHR heat exchanger is a large C-shaped tube bundle that is connected at the top to a primary system hot leg by an inlet line that is always open during operation and to cold legs from the bottom. The PRHR operates at primary-side operating pressure to remove decay heat. At least one valve on the return line opens to enable natural circulation flow of single-phase water. Denser water cooled in the PRHR heat exchanger drains to the cold legs, promoting flow of hotter, less dense water from the reactor core to the PRHR inlet. The heat is transferred to the IRWST. Secondary-side heat transfer is initially natural convection until the IRWST water reaches saturation temperature. Steam produced by secondary-side boiling is vented to the containment gas space.

The Passive Containment Cooling System (PCS) removes heat from containment to avoid exceeding containment design pressure and temperature limits. Steam from the IRWST and/or from a LOCA condenses on the cooler surfaces of the containment and heats the containment shell. Cooling by the PCS enables continued steam condensation on the shell inner surface and transfer of heat from inside to outside of containment. The condensate flows back to the IRWST.

The PCS provides cooling by a combination of air and water. Water flows on the external top surface and vertical side of the containment steel pressure vessel dome (Fig. 20). The cooling water on the containment shell outer side evaporates and leaves

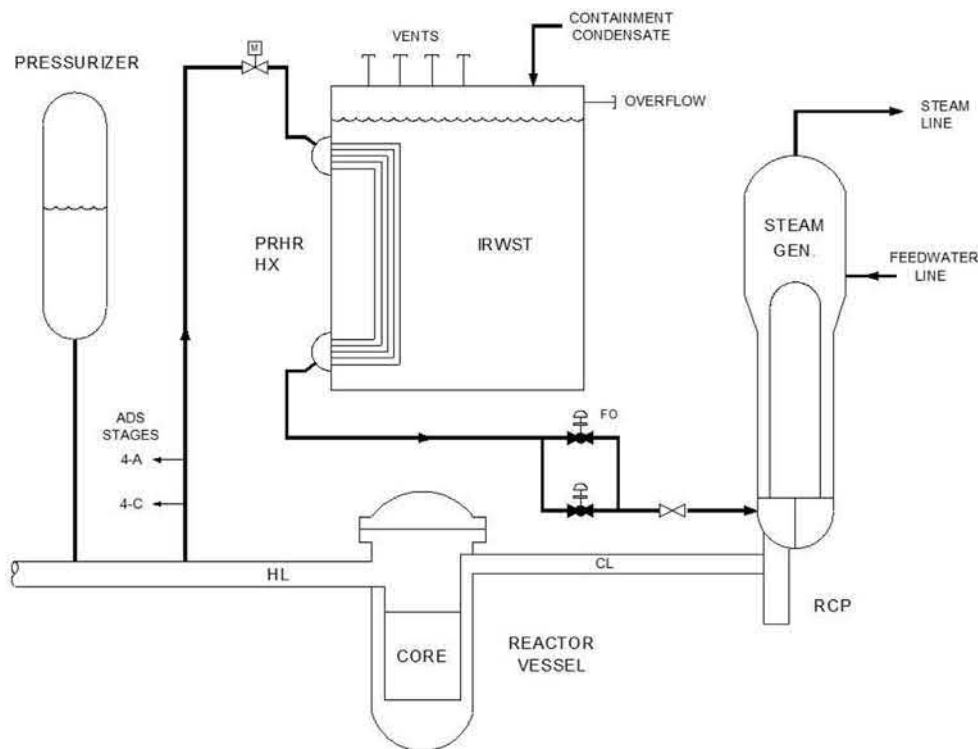


Fig. 19 AP1000 passive core cooling system (Cummins and Matzie, 2018).

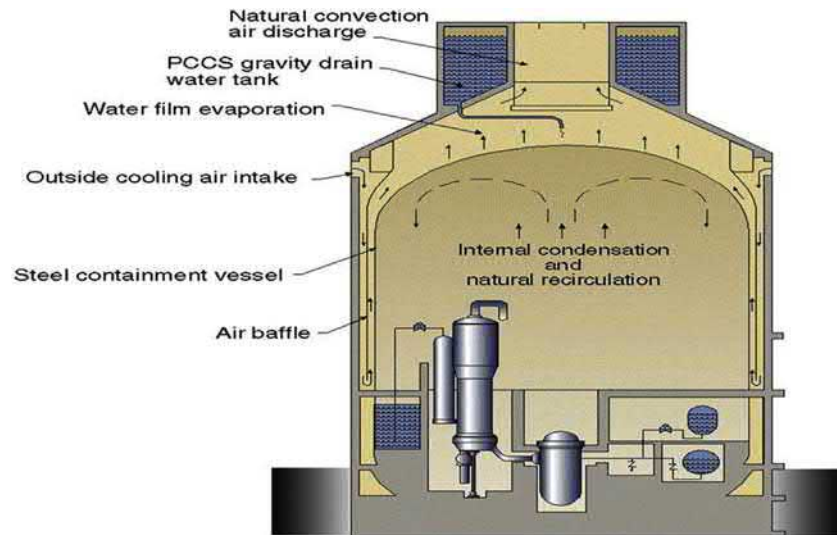


Fig. 20 AP1000 passive containment cooling systems (Cummins and Matzie, 2018).

the shield building with a natural circulation air flow that is induced by the upward steam flow. Air enters the shield building near the top of the cylindrical shield building and flows down an air baffle. The cooling air makes a 180° turn and then flows up between the containment shell and the air baffle. As it flows upward, the air is heated by natural convection heat transfer. The lower density results in an increase in velocity which enables the air to carry the evaporating PCS water upward and out of the shield building roof through a large opening in the center of the roof.

ESBWR passive heat transfer systems

The Economic Simplified Boiling Water Reactor (ESBWR), designed by GE Hitachi Nuclear Energy (2011) and Design Certified by the US NRC, is a Gen III+ reactor with a rated power output in excess of 1500 MWe (Fennern, 2018). Not only does the ESBWR have passive safety systems, but it is also a natural circulation reactor. That is, the ESBWR achieves primary coolant flow without any active coolant circulation pumps. The primary flow is driven by buoyancy differences in the “chimney” region above the core, small pressure differences and jet pumps in the reactor recirculation loops. Fig. 21 illustrates the ESBWR passive safety systems. Most of the passive components in the ESBWR design are employed in other designs that have been built and operated. The ESBWR design is presented herein to illustrate how the systems function.

Normal shutdown cooling at high pressure when the reactor has been isolated from the main steam turbines is done by the Isolation Condenser System. The Isolation Condensers are vertical tube bundles which sit in water compartments outside of containment and at a higher elevation than the RPV. The units are connected to the RPV by a steam inlet line that is always open during operation and a return line that allows for water flow back to the RPV. Upon passive opening of a valve on the return line, the water-filled tubes drain to the RPV and steam is ingested into the top of the tubes. The steam condenses in the tubes and drains by gravity, inducing a natural circulation flow path between the reactor and the condensers.

Low-pressure makeup of reactor coolant inventory is accomplished with the Gravity-Driven Cooling System (GDCS). Three elevated GDCS water pools are located in the containment Drywell. Following a LOCA and depressurization of the primary system, the GDCS pools can drain by gravity into the RPV, providing short-term makeup water. Long-term makeup is enabled by an Equalization Line, a line between the Suppression Pool and the RPV. The Equalization Line allows flow into the primary side from the Suppression Pool due to the hydrodynamic head ($\rho g \Delta H$ term) between the Suppression Pool and the RPV. The much larger Suppression Pool water volume is sufficient to flood the RPV to an elevation well above the core. With adequate coolant inventory, the decay heat is removed by natural convection heat transfer and/or vaporization.

With the core well cooled at low pressure by the GDCS and/or Equalization Line water, containment heat removal is next considered. Under Design Basis Accident conditions, heat released from the primary system to the containment must be removed in order to remain within containment pressure and temperature limits. The Passive Containment Cooling System (PCCS) provides this function by ingesting a steam-noncondensable gas mixture from the containment Drywell and condensing the steam in heat exchangers located outside of containment. These six PCCS heat exchangers are vertical tube bundles located in elevated water compartments outside of containment, similar in design to the IC units. The steam condensed in the PCCS units drains to the GDCS Pools by gravity, and thereafter to the RPV as core makeup. The PCCS units vent noncondensable gases to the Suppression Pool. The pressure difference between the Drywell and the Suppression Pool drives the natural circulation flow through the PCCS.

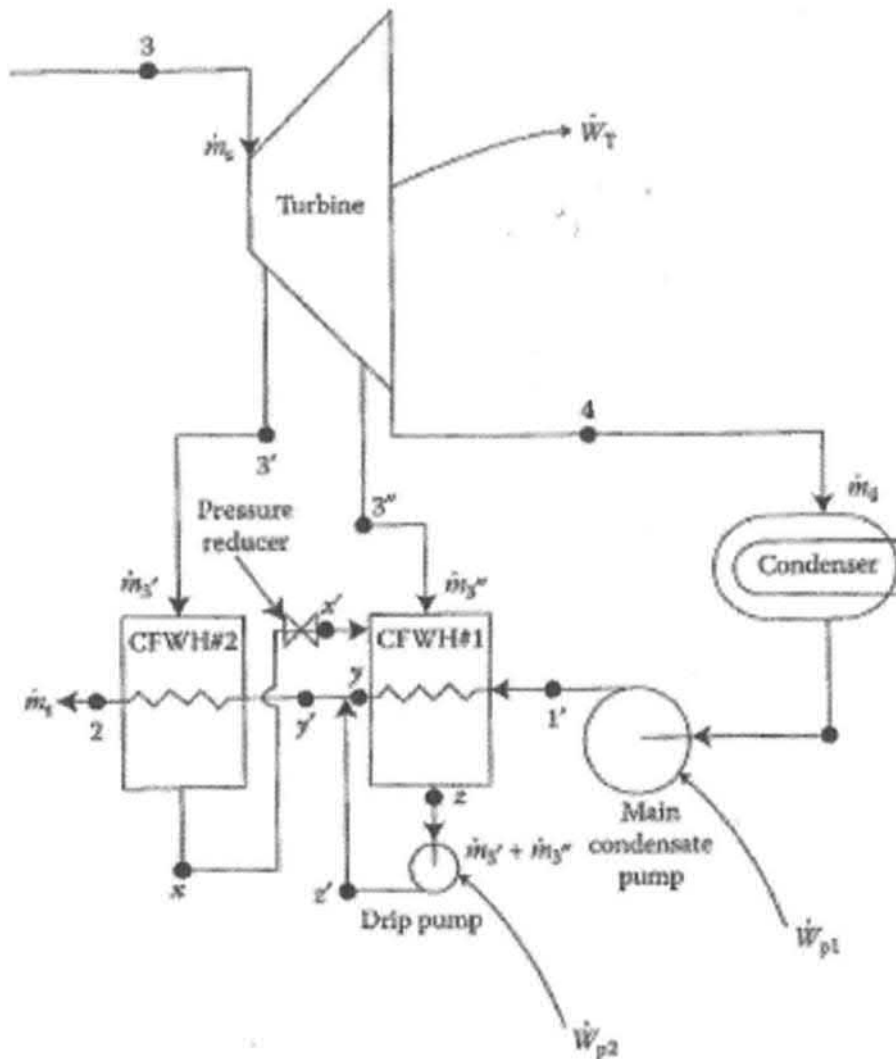


Fig. 22 BWR/6 electric power cycle (General Electric Company, 1980). Courtesy of GE Hitachi Nuclear Energy Americas LLC.

returns from the condenser to the steam generator (PWR) or RPV (BWR). The processes are further explained in the Temperature-Entropy (T-s) diagram of Fig. 24.

In superheating, a heater is added to heat steam into the superheated region, to the right of the vapor dome on a T-s diagram (Fig. 25). The sensible heat of the steam entering the turbine is increased. The results are drier steam into the turbine and increased work from the turbine, to increase the cycle thermal efficiency. If the system is properly designed, the cost of the superheater is offset by the increased turbine work and thermal efficiency.

With reheating, the fluid is heated again after partial expansion in the turbine (Fig. 26). The steam flowing into the low-pressure turbine is therefore drier and can undergo more expansion in the low-pressure turbine than without reheating. The results are drier steam into the turbine and increased work from the turbine, to increase the cycle thermal efficiency. Similar to superheating, if the system is properly designed, the cost of the superheater is offset by the increased turbine work and thermal efficiency.

As the final Rankine cycle improvement, moisture separation is used to remove moisture from the partially expanded steam before it flows into the low-pressure turbine. The benefit of this process is increased life of the turbine by protecting the turbine blades from impinging water droplets. The thermal efficiency is not increased. Fig. 27 shows the components involved and the T-s diagram (Fig. 27).

An actual nuclear reactor plant uses a combined moisture separator-reheater between the high-pressure and low-pressure turbines to remove moisture from the wet steam exiting the high-pressure turbine and to add superheat to the steam before entering the low-pressure turbine. This configuration is shown in Fig. 22.

The range of thermal efficiency for a commercial nuclear power plant is between 30–38% with these improvements included.

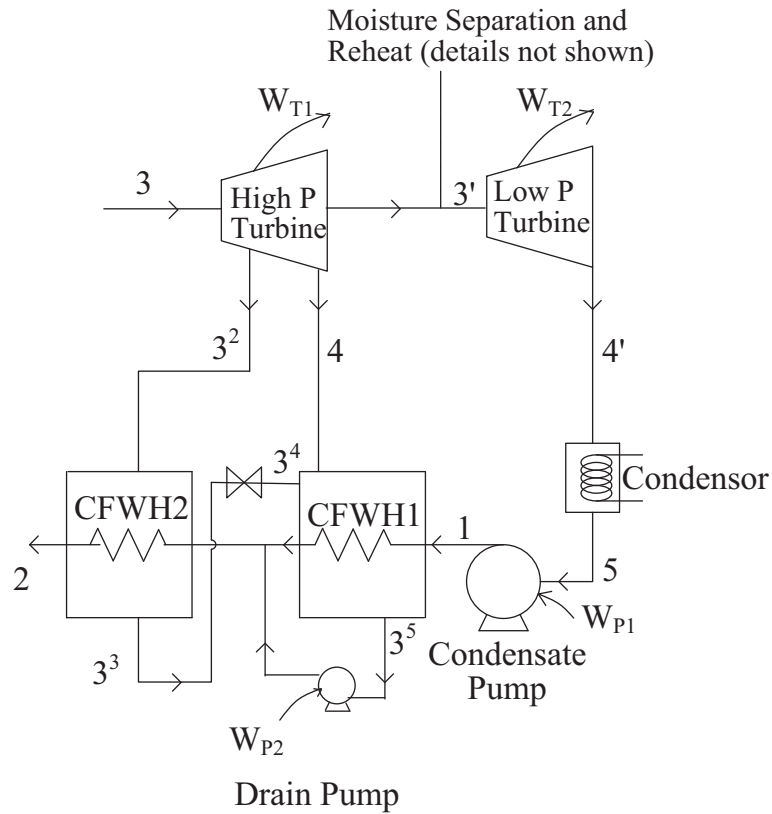


Fig. 23 Regeneration with two closed feedwater heaters.

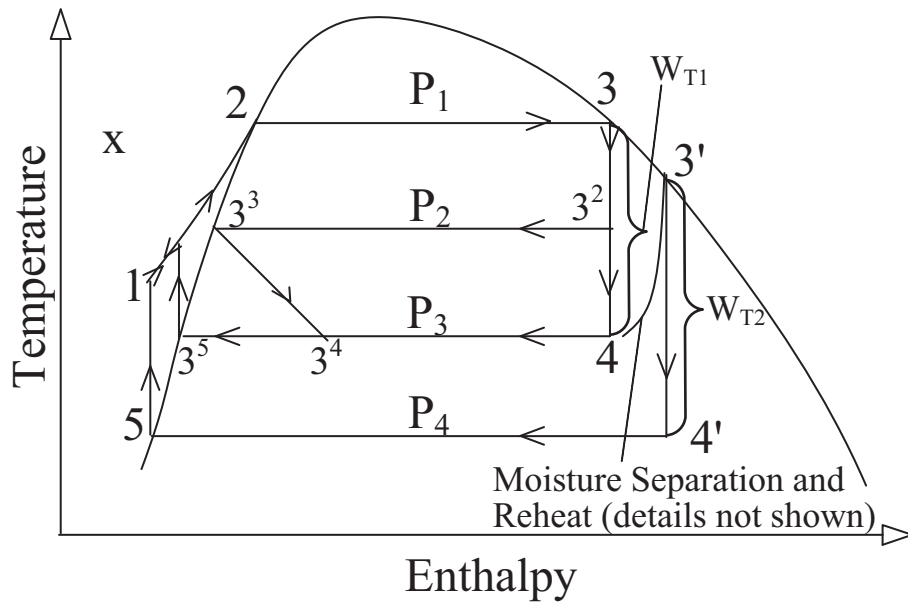
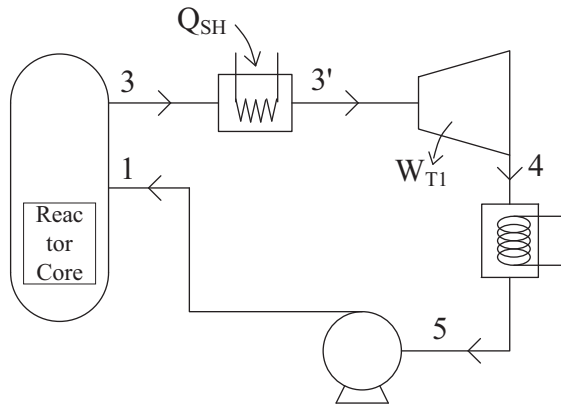
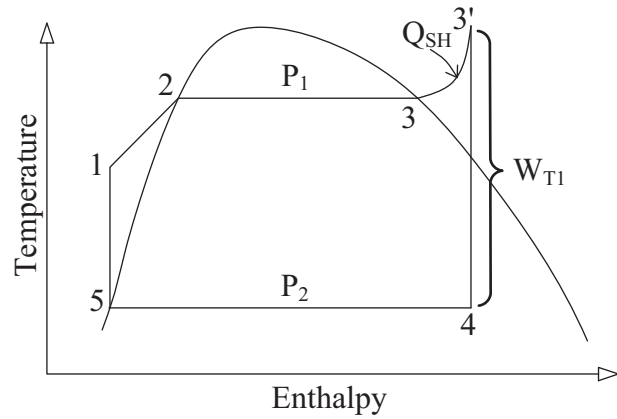


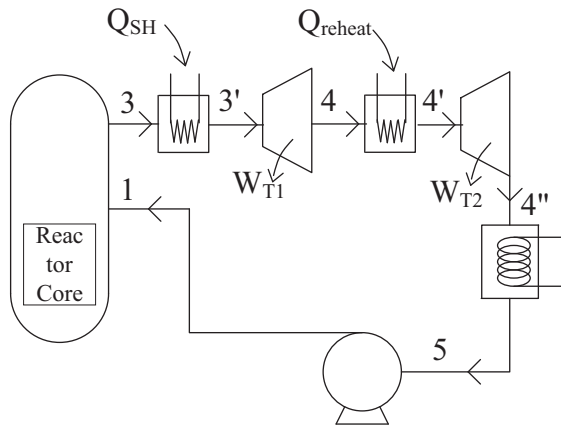
Fig. 24 T-s diagram for regeneration with two closed feedwater heaters.



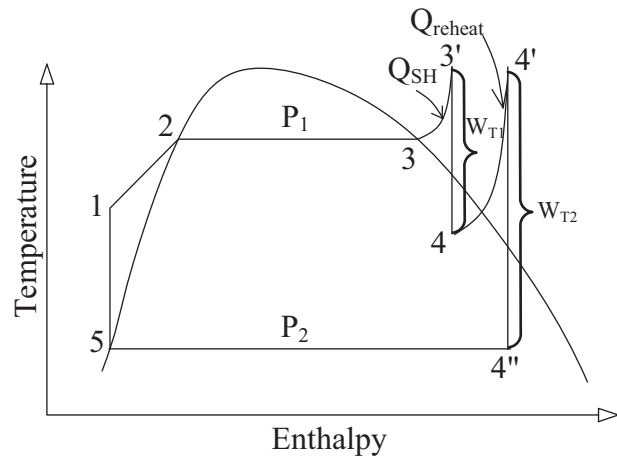
Components with Superheating

Fig. 25 Component diagram and T-s diagram for superheating.

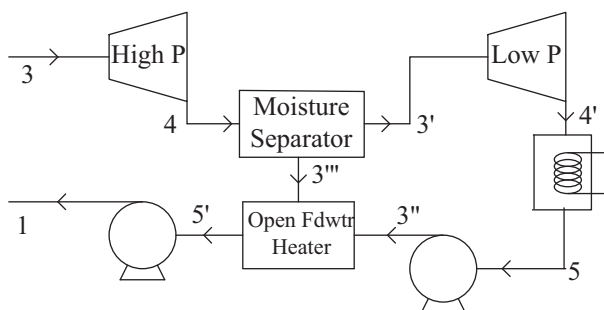
T-s Diagram for Superheating



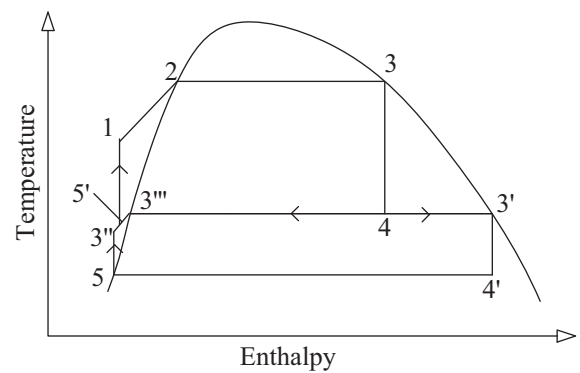
Components with Superheating and Reheating

Fig. 26 Component diagram and T-s diagram for reheating.

T-s Diagram for Superheating and Reheating



Components for Moisture Separation

Fig. 27 Component diagram and T-s diagram for moisture separation.

T-s Diagram for Moisture Separation

Summary

The basic phenomena occurring in nuclear reactor heat transfer systems were described in this article. Included are heat transfer systems for normal operation, normal shutdown and breach of the primary system, starting with the nuclear fission reactor and extending out to and including the Ultimate Heat Sink. LWR systems were given primary focus, including active and passive reactors and passive safety systems. The intention is to summarize the basic concepts so that the reader is prepared to investigate specific reactor designs of interest in greater detail.

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Instrumentation and Control Systems of Nuclear Power Plants

Belle R. Upadhyaya and Richard T. Wood, Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

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Abbreviations

ADC	Analog-to-digital conversion
APRM	Average power range monitor
ATWS	Anticipated transient without scram
BWR	Boiling water reactor
B&W	Babcock and Wilcox
CCF	Common-cause failure
COTS	Commercial-off-the-shelf
CVCS	Chemical and volume control system
DAS	Diverse actuation system
DiD	Defense-in-depth
GDC	General design criteria
ESFAS	Engineered safety features actuation system
FCV	Feed control valve
GWR	Guided wave radar
HART	Highway Addressable Remote Transducer
HIS	Human system interface
IAEA	International Atomic Energy Agency
I&C	Instrumentation and control
LPRM	Local power range monitor
MOV	Motor-operated valve
NPP	Nuclear power plant
NSSS	Nuclear steam supply system
O&M	Operation and maintenance
OTSG	Once-through steam generator
PHWR	Pressurized heavy water reactor
PI	Proportional-integral
PID	Proportional-integral-differential
PIE	Platform for information exchange
PWR	Pressurized water reactor
RTS	Reactor trip (rapid shutdown) system
RTD	Resistance temperature detector
SMR	Small Modular Reactor
SPND	Self-powered neutron detector
TCV	Turbine control valve
TIP	Traveling in-core probe
UTSG	U-tube steam generator

Introduction**Importance of instrumentation and control systems in nuclear power plants**

The instrumentation and control (I&C) systems in a nuclear power plant (NPP), along with their enabling features, serve as the “central nervous system” of a NPP. These systems ensure the safe, reliable, and economic operation of the power plant. The I&C architecture includes plant control, protection, and monitoring systems that provide information to various actuators and devices to ensure the safe and efficient operation of the plant, and to the operation and maintenance (O&M) personnel for making timely decisions. The information about plant parameters and plant status are communicated continuously to various modules in real time so that appropriate actions may be taken by the plant sub-systems.

This chapter focuses on a technical description of the instrumentation and control systems used currently in light water reactor (pressurized water reactor (PWR) and boiling water reactor (BWR)) plants.

Outline of the chapter

The chapter covers the following topics related to nuclear plant I&C.

- Overview of I&C systems
- Instrumentation in light water reactors and heavy water reactors. This addresses field device level technology that is cross cutting elements of I&C, with a limited discussion of actuators
- Control strategies in PWRs and BWRs
- Protection systems with focus on principles such as separation, redundancy, defense-in-depth and diversity
- Digital I&C modernization
- I&C concepts for small modular reactors and advanced reactors
- Summary

Overview of I&C systems

Overall objective of I&C systems

The functions performed by I&C systems in a NPP are threefold.

- The plant measurements are communicated to control, monitoring, and protection systems. The operator or a plant monitoring system uses the sensory information to track the operational behavior of the plant and look for deviations from normal conditions.
- The measurements are used by control systems to regulate the overall plant output and outputs of various plant systems. The goal is to minimize the deviations of important variables from their set points.
- The third function is the actuation of safety systems in case of abnormal deviation from normal plant behavior, some of which may be caused by human operators. Under these situations, the safety systems provide “automatic action to protect the plant and the environment.”

Fig. 1 is a block diagram showing the functional relationship among these three major systems.

Characterization of I&C systems using a functional approach

Plant I&C systems are represented in terms of sensors, data acquisition, display, communication, monitoring, control, and safety systems that are integrated with plant dynamics (reactor power, thermal-hydraulics, energy conversion) and operator interfaces, referred to as human system interface (HSI). Fig. 2 is a schematic of the functional representation of I&C systems in a nuclear plant (IAEA Nuclear Energy Series, 2011). In digital systems, analog outputs from instrument channels are sampled at desired rates to provide the required information that flows to monitoring, control, and safety systems. Actuators are part of the control and safety loops.

The functionality in the I&C system architecture includes the following (IAEA Nuclear Energy Series, 2011): “Data acquisition, actuator activation, signal and command validation, arbitration, control, limitation, checking, monitoring, command, prediction, communication, fault/alarm management, and configuration management.” In the U.S., the safety system design principle follows the IEEE 1E classification. Important elements of the design approach include “functional isolation; redundancy; physical, electrical

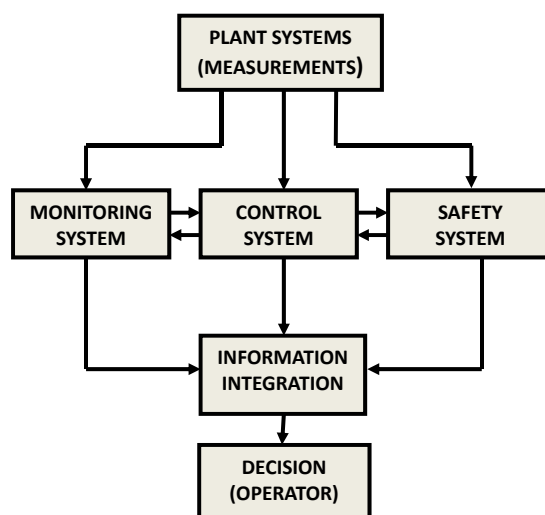


Fig. 1 Integration of I&C systems.

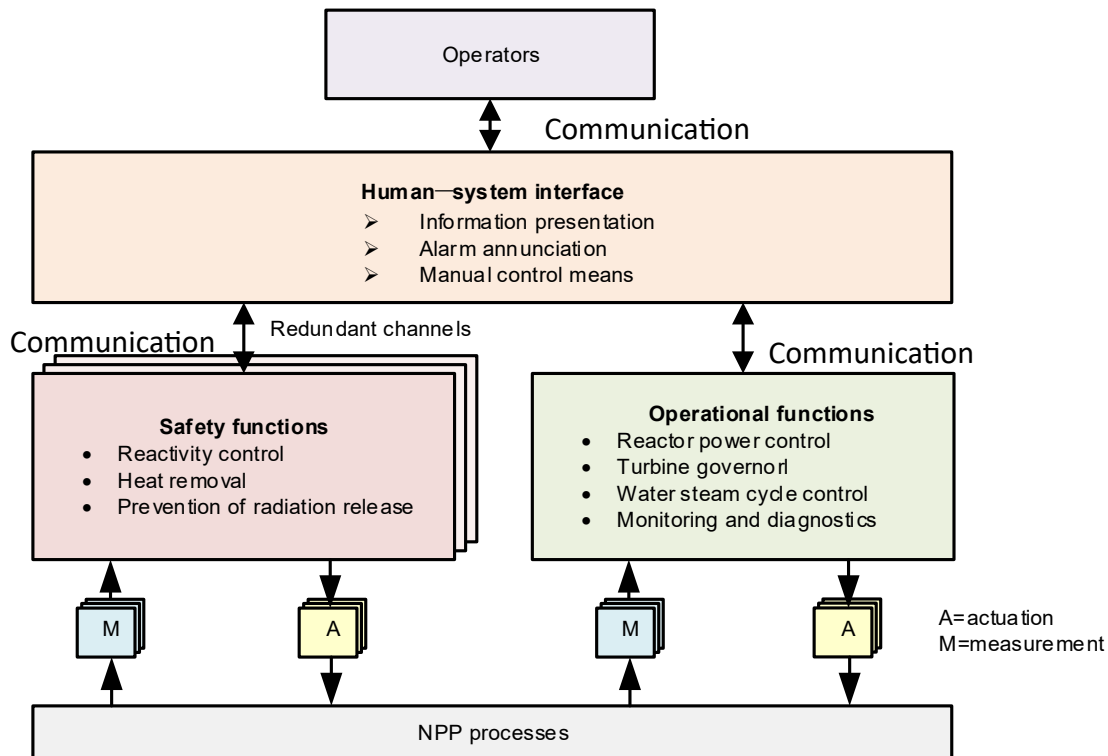


Fig. 2 Schematic showing functional architecture of I&C in a nuclear power plant. Adapted from IAEA Nuclear Energy Series (2011) Core Knowledge on Instrumentation and Control Systems in Nuclear Power Plants. *Technical report NP-T-3.12*.

and communications isolation; diversity and defense-in-depth; electrical noise emissions and immunity; installation, maintenance and design change control" (IAEA Nuclear Energy Series, 2011).

Characterization of I&C systems using a physical approach

Another description of I&C systems is to present their physical layout, along with sensors and actuators used, and the communication of information among the different sub-systems. This provides a clearer visualization of the functions of the NPP I&C systems compared to a functional description (IAEA Nuclear Energy Series, 2011).

Process interfaces include primarily sensors that measure various process variables, neutron flux, and other parameters. Examples of sensors include temperature, pressure, liquid level, flow rate, neutron flux detectors, radiation monitoring, chemical parameters, position, number of rotations, and shaft speed. Other related devices include signal processing electronics, smart transmitters, trip logic, process sensing lines (such as pressure instrument tubes), and equipment monitoring and diagnostics measurements (such as vibration sensors, ultrasonic transducers, infrared thermography).

There is a multitude of major and minor control loops distributed in the plant. A large number of actuators is used for providing control actions. Examples include motor-operated valves (MOVs), solenoid-operated valves, air-operated valves, switchgears, motor controls, control rod drive mechanisms, heater banks, and water spray systems (pressurizer, core, containment). Communication from the sensors to other parts of the system is achieved by cables carrying both analog and digital signals. HART (Highway Addressable Remote Transducer) protocol and wired digital fieldbus interfaces are common. Digital wireless communication is being considered in many applications.

A digital data acquisition system consists of signal conditioning (filters and amplifiers), analog-to-digital conversion (ADC) device, and data storage. Control systems are either centralized or distributed. In a centralized control system, the processing capabilities and the plant data are at one central location. In a distributed control, these features are distributed or localized throughout the process. In this case, the control actions and the database are available individually throughout the system, with necessary communication among the distributed controllers.

Plant I&C operator support systems include "safety parameter display systems, computerized procedures, I&C system documentation, and process performance monitoring." Different forms of plant simulators are available to operators. These include training simulators, engineering simulators, research simulators, and multifunction simulators. Some of these simulators mimic plant operational transients and accident scenarios.

Instrumentation in nuclear power plants

Introduction

This section presents a summary of instrumentation used in light water reactors (PWRs and BWRs). Kerlin and Upadhyaya (2019) contains a summary of instrumentation systems in pressurized heavy water reactors (PHWRs) and certain advanced reactors. Additional information and details on instrumentation are given in DOE Fundamentals Handbook (1992a,b) and Harrer and Beckerley (1973, 1974).

Instrumentation systems in a large NPP communicate sensor measurements to control, safety, and plant monitoring systems. They consist of process sensors, neutron detectors, radiation monitors, logic devices and other devices, totaling about 10,000 in a large NPP. A factor contributing to this large number is the redundancy in safety system sensors.

Process instrumentation

Process instrumentation in PWRs and BWRs consist of temperature, pressure, flow, and level sensors. The sensors are manufactured and supplied by vendors who specialize in nuclear grade instrumentation.

Temperature sensors

Commonly used temperature instrumentation consists of resistance temperature detectors (RTDs) and thermocouples. RTDs are more accurate for temperature measurement than thermocouples and are used to measure hot leg and cold leg temperatures in a PWR. Thermocouples measure PWR core-exit temperature and other fluid temperatures in a power plant.

Temperature measurement using RTDs is based on the principle that the electrical resistance of a wire changes with temperature. Platinum wire is the common sensing material used in RTDs; its resistance increases almost linearly with temperature. Temperature vs. resistance calibration provides an excellent relationship for accurate temperature measurement. RTDs are installed either by using a thermo-well that makes contact with a fluid in a pipe or in a bypass loop attached to the pipe. The RTD wire is generally wound around a mandrel and is encased in a stainless-steel sheath with an insulating material such as magnesium oxide. The response time of RTDs depends on the RTD sheath thickness and the installation (Kerlin and Shepard, 1982).

Thermocouples are used extensively for fluid temperature measurement in NPPs. They are less expensive and easier to manufacture than RTDs. A thermocouple has two dissimilar wires joined at a junction which is the measuring point. The temperature difference between the junction and the open end of the wire creates a voltage drop. Thermocouple wires are encased in a stainless-steel sheath with an insulating material between the wires and the sheath inner wall (Kerlin and Shepard, 1982). The common thermocouple is of Type-K that uses chromel-alumel wires. For higher temperature measurements Type-N (nicrosil-nisil) thermocouples are used.

Pressure sensors

Pressure sensors used in reactor applications must measure high pressures in the range 50–153 bars (≈ 750 –2250 lbf/in²). The devices must be able to withstand high temperatures and radioactive environment. The most common pressure sensor has a metallic diaphragm that moves (expansion and contraction) when a pressure is applied (DOE Fundamentals Handbook, 1992a). This movement can be measured electronically using changes in capacitance between two plates. Pressure sensors measure both absolute pressure and differential pressure.

New piezo-electric pressure sensors are being developed. The applicability of these sensors is limited by the temperature of the medium. The piezo-electric property is effective up to a temperature of about 300 °C. Artificial piezo materials are common in these applications. An example is the material lead-zirconate-titanate (PZT) which has a Curie temperature higher than natural crystals.

Flow sensors

The common flow measurement technique in light water reactors is based on measuring pressure drop across an orifice meter or a venturi meter. The pressure drop, Δp , is measured by a differential pressure transmitter. The flow velocity is proportional to $\sqrt{\Delta p}$. The volumetric flow rate is calculated knowing the cross-sectional areas at the pipe and the constriction, and the pressure drop. Since a bend in a pipe would also result in a pressure drop, flow velocity can also be estimated by measuring this pressure drop and the knowledge of the pipe diameter.

Ultrasonic flow meters are used extensively for monitoring liquid flow rate in industrial processes. An advantage of the ultrasonic flow meter is that it is not necessary to expose the device directly in the flow regime. The principle of reflection transit time ultrasonic flow meter (Lish et al., 2017) can be adapted to measurement in a reactor plenum, such as in the downcomer region. Two ultrasonic transducers are installed outside the pipe or plenum on the same side at a known distance apart. Each device acts as a transducer and a receiver. An ultrasonic wave is pulsed from the transducer in the direction of flow (downstream) to strike an opposite surface. The reflected wave is then received by the downstream transducer. The total time of transit is recorded (t_d). In a similar fashion, a wave is pulsed in the direction opposite to the flow (upstream). The reflected wave is received at the upstream transducer. This total time of transit is recorded (t_u). Because of the manner in which the two pulses are transmitted, $t_u > t_d$. Knowing these two transit times and the angle of incident of the ultrasonic pulses on the opposite surface, and the distance between the detectors, we can calculate the average flow velocity (Lish et al., 2017). This approach is well-suited for integral small reactors where the space available for sensor placement is limited. Light-water small modular reactors have no primary piping,

and direct measurement of primary coolant flow rate is challenging. A similar challenge is presented by BWRs with internal pumps, or natural circulation.

For flow measurement in liquid metal reactors (such as the sodium fast reactor), magnetic and eddy current flow meters are good candidates (Vaidyanathan et al., 2012).

The above flow measurement techniques require physical placement of sensors. An alternate approach for flow velocity measurement uses cross correlation between two signals from sensors that are installed to measure either temperatures or in-core neutron flux. This is the case with in-core neutron detector strings in BWRs or in-core SPND (self-powered neutron detector) strings used in PWRs. As the water (single or two-phase) flows by these detectors, the neutron detection gets affected and generates fluctuations that can be detected at two adjacent detectors. A cross-correlation function between the detector signals may be used to estimate the transit delay time, which is the correlation lag where the cross correlation has a maximum value (Sweeney et al., 1985).

Nitrogen-16 (N-16) detectors are used for monitoring primary coolant flow rate in PWRs. Cross correlation between adjacent N-16 detectors is used to estimate the flow transit time, and the average coolant velocity. See section “**Instrumentation in pressurized water reactors**” for details. The use of diverse measurement methods would complement each other and provide reliable measurements of flow rates.

Level sensors

The differential pressure between two points, above and below the water column in a vessel, is proportional to the height of the water column, as shown in Eq. (1).

$$p = \rho gh \quad (1)$$

p = differential pressure; ρ = density of water; g = acceleration due to gravity; h = height of the water column.

This relationship is used to make level measurements in light water reactors. This requires the knowledge of liquid density at a given temperature. Examples of such measurements are: downcomer level in a U-tube steam generator (water level above the U-tubes), pressurizer water level, and water level in a BWR vessel.

The section on “**Control of PWRs and BWRs**” describes the regulation of vessel liquid levels at desired set point values.

Another approach for level measurement is the use of time of flight instruments. The time of flight measurement may be made either by using ultrasonic pulses or high frequency guided radio waves. Ultrasonic waves are transmitted from a piezo-electric transducer towards the liquid. The ultrasonic pulse strikes the liquid level and reflects back (because of a change in the density of the two media), and the reflected wave is received by the pulse-echo device. Knowing the time of flight and the velocity of sound in the medium above the liquid level, the distance between the transducer and the liquid level is calculated.

Guided wave radar (GWR) is another technology for level measurement (Endress and Hauser, 2020). In this approach a high frequency radio wave is passed through a waveguide which extends into the liquid. When the wave hits the liquid surface, it gets reflected through the waveguide back to the receiver. By measuring the time of flight, an estimate of the distance between the probe and the liquid level is made. This technique needs further development and demonstration for reactor applications.

Neutron detectors

The measurement of nuclear power is made using neutron detectors. Neutron flux, is defined as the number of neutrons/cm²-sec. A typical value of thermal neutron flux in a PWR at full power is in the range 10¹³–10¹⁵ neutrons/cm²-sec. Neutron flux is the product of neutron density and the average velocity of neutrons at a certain energy level. The reactor power is proportional to core average neutron flux. The estimation of reactor thermal power is outlined in sections “**Major control systems in a PWR**” and “**Control of boiling water reactors**”.

The flux measurement is made during reactor start-up, intermediate power range, and power operation. Different detectors are being used depending on the reactor operation regime.

Fission chambers

Fission chambers are used during reactor start-up to measure low flux and can cover an extended range of reactor operating conditions. The inside of a fission chamber is coated with a thin layer of highly enriched U-235 and filled with oxygen-free argon gas. The fission caused by the interaction of neutrons with the coating results in high energy fragments (fission products) that ionize the argon gas and produce a current that is measured, either in the pulse mode or in the current mode (Campbell mode). The charge pulses generated by neutron capture are much greater than the pulses generated by gamma capture. Consequently, the measurement content arising from the fission flux is much greater than the gamma-ray flux, even at start-up and low-power conditions. Thus, fission chamber measurement overlaps both the source range and power range modes. Fission chambers are used both in PWRs and in BWRs.

Ionization chambers

Ionization chambers are neutron detectors with a neutron absorbing material that is filled into the chamber as a gas with the central element acting as anode and the chamber wall as cathode. The chamber is filled with BF₃ gas. Boron-10 has a high thermal neutron absorption cross section. A neutron absorption in B-10 produces two charged particles—Li-7 and α -particle (see Eq. 2), and the

resulting current is measured. The sensitivity to gamma energy can be reduced by proper design of the chamber, often referred to as compensated ion chamber.



Compensated ionization chambers are suitable for measuring flux in the range 10^4 – 10^{12} neutrons/cm²-sec, and are used in the intermediate power range mode.

Self-powered neutron detectors (SPNDs)

These detectors are used for measurement of localized neutron flux. These are generally configured as in-core neutron detector strings with four detectors placed along each instrument tube. A thermocouple is installed at the tip of the instrument tube for core-exit temperature monitoring in PWRs. SPNDs consist of a neutron absorbing material such as Vanadium, Rhodium, Cadmium or Cobalt inside a stainless-steel sheath. SPNDs consist of an emitter, insulation, and a collector (sheath). The neutron activation of the emitter material produces beta decay, and the resulting electrons migrate towards the collector and a current is measured. The current level is very low and requires amplification for proper data acquisition.

SPNDs are slow responding detectors because of the large half-life of the beta emitter material. These detectors remain calibrated longer than fission and ionization chambers. They are well-suited for in-core monitoring and flux mapping.

Instrumentation in pressurized water reactors

Fig. 3 is a schematic of a PWR plant showing the major sensors and their placement on the primary and secondary sides of the plant (Kerlin and Upadhyaya, 2019).

Following are the process and neutron flux instrumentation in a PWR. See Fig. 3.

- Hot leg and cold leg RTDs. These are used in control and safety system functions.
- Pressurizer pressure (primary pressure) and pressurizer water level.
- Primary coolant flow rate.
- Steam generator feedwater flow rate, steam flow rate, downcomer level (in U-tube steam generators).
- Core-exit thermocouples (Type-K) to measure the coolant temperature at ≈ 45 locations. The thermocouples are introduced through conduits at the vessel head ports.
- Digital rod position indicator signal.
- Traversing in-core probes (TIP) are used to perform three-dimensional flux mapping. The TIP detectors are small fission chambers that are introduced by a drive system through instrument thimble guides that extend through the entire length of the core. Fig. 4 shows the arrangement of in-core instrumentation in a Westinghouse PWR (Westinghouse Electric Company, 1984).
- Ex-vessel neutron detectors are used for power monitoring from start-up to full power operation. Two source range detectors (fission chambers) and two intermediate range neutron detectors (compensated ionization chambers) are placed in the biological shield surrounding the reactor vessel. Two sets of four ionization chambers (BF_3 -filled) measure the neutron flux in the upper half and in the lower half of the core. These long detector chambers are placed symmetrically between each pair of hot leg

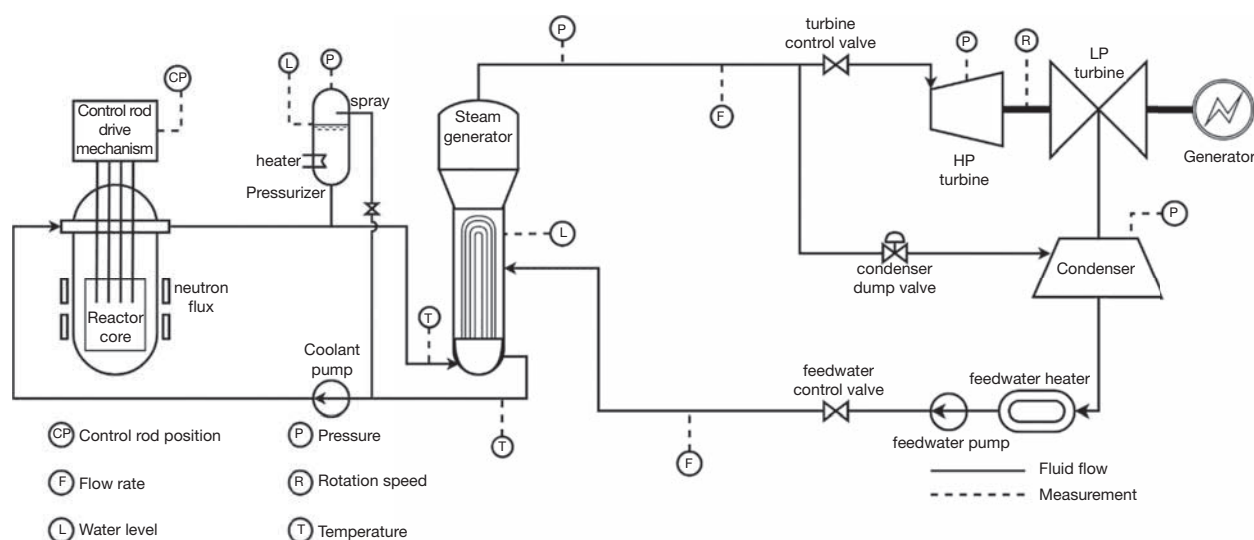


Fig. 3 Important plant instrumentation of a PWR with U-tube steam generators. Kerlin TW and Upadhyaya BR (2019) *Dynamics and Control of Nuclear Reactors*. Cambridge, MA: Elsevier—Academic Press.

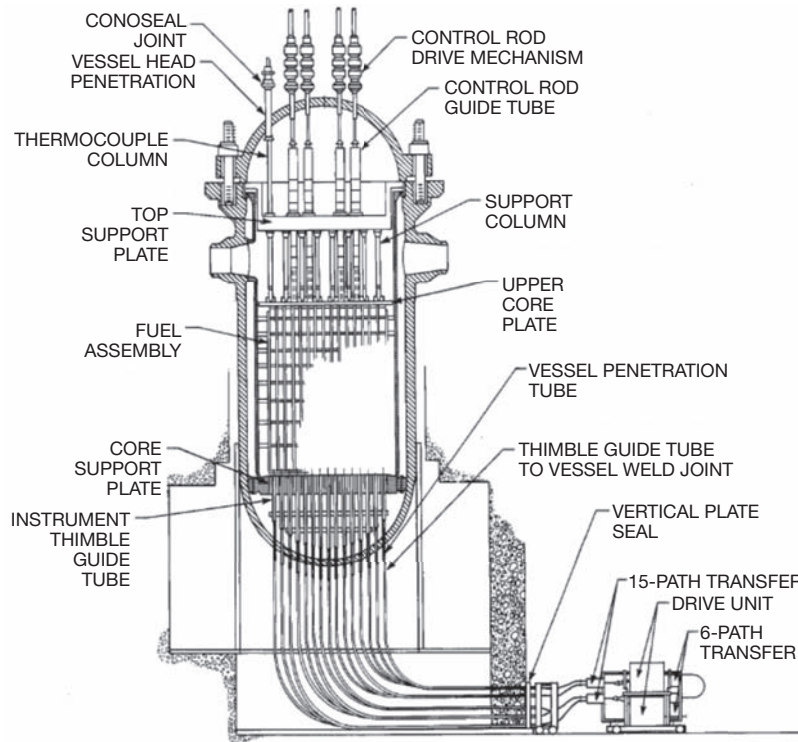


Fig. 4 In-core instrumentation system in a PWR. Reprinted with permission from Westinghouse Electric Company (1984) *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*. Pittsburgh: Westinghouse Electric Company, Water Reactor Division.

and cold leg pipe nozzles as shown in Fig. 5. Four current signals are calculated by adding the outputs of a pair of corresponding upper and lower detector currents and provide information about any power offset. The power range detectors measure up to 120% of full power.

- Nitrogen-16 (N-16) detector monitors the thermal power of the NSSS by measuring the level of N-16 present in the coolant due to a neutron reaction with the O-16. The decay of N-16 isotope produces high energy gamma rays that are measured by ionization chambers placed on hot leg piping close to the reactor vessel (Westinghouse Electric Company, 1984). The N-16 fluctuations may be measured by gamma detectors at two locations in the pipe to estimate the transit time. Cross correlation between the two detector signals is used to estimate the transit time. Knowing the distance between the detectors, the transit time of N-16 activity, and the pipe inside area, the volumetric flow rate of the primary coolant is estimated. This parameter complements flow rate measurement using standard flow meters.

Instrumentation in boiling water reactors

Instrumentation requirement in a BWR is somewhat different compared to that for a PWR. This is primarily due to the differences in the NSSS and the reactor control strategy. The following are the typical measurements in a BWR (Kerlin and Upadhyaya, 2019).

- Reactor coolant flow rate. This is measured by venturi meters at the recirculation pump discharge locations. There are two recirculation pumps in a BWR-6 plant that supply water to a set of jet pumps.
- Reactor vessel water level.
- Steam flow rate.
- Reactor pressure (steam pressure).
- Feed water flow rate.
- Feed water temperature.
- Local power range monitors (LPRMs). An LPRM detector tube has four fission chambers, placed 36-in. apart, with the bottom and top detectors 18-in. from the top and bottom of core.
- Average power range monitor (APRM) signal is an average of about 20 LPRM detector signals from across the core. The APRM signal provides the global core power behavior.

A large BWR core (1220 MWe) has approximately 45 LPRM detector tube locations. Each detector tube is placed among four fuel channels, with the central tubular space available for a TIP detector. The TIP detector is used for calibrating the LPRM fission

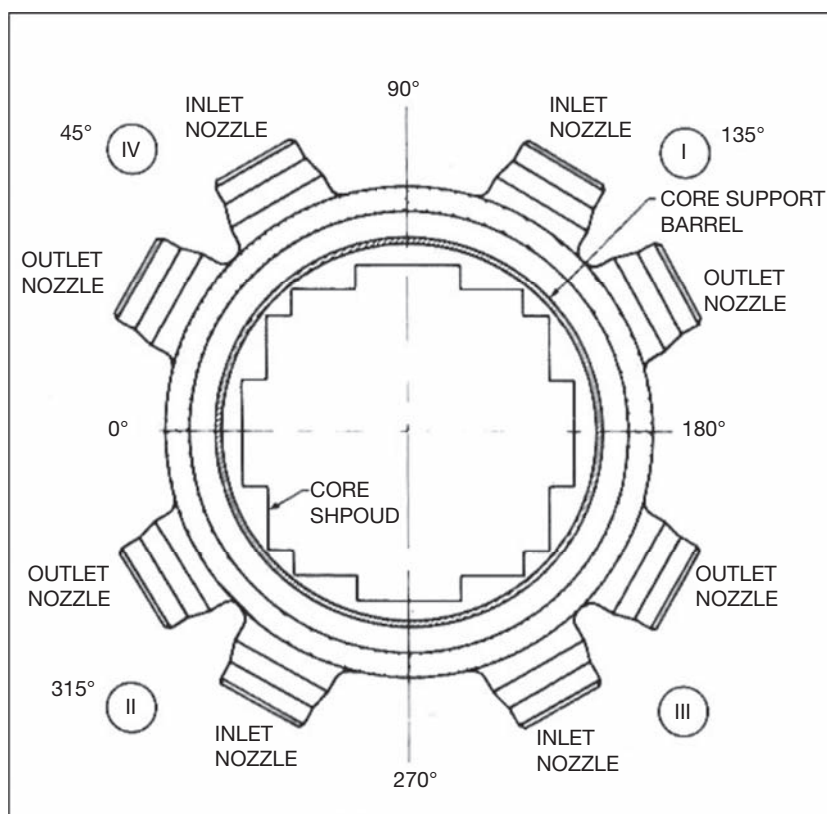


Fig. 5 Locations of four ex-vessel ionization chambers in a PWR, numbered I, II, III and IV. Detectors I & II and III & IV are placed diametrically opposite to each other. Figure shows the reactor core, core support barrel, pressure vessel, hot leg and cold leg nozzles in a four-loop plant. Fry DN, March-Leuba J and Sweeney FJ (1984) Use of Neutron Noise for Diagnosis of In-Vessel Anomalies in Light Water Reactors. *ORNL/TM-8774*.

chambers. **Fig. 6** shows a BWR core map. The detector locations are shown as dark circles and the crosses (+) indicate control assembly locations (Fry et al., 1984).

Actuators

A nuclear power plant has a large number of control loops performing major control actions centrally and minor control actions in a distributed architecture. Effective process regulation requires reliable actuators. Examples of some of the actuators are MOVs, solenoid-operated valves, air-operated valves, switchgears, motor controls, control rod drive mechanisms, heater banks, and spray systems. These actuations result in changes in flow rate, liquid level, temperature, pressure, nuclear reactions and others.

In all these situations it is not only necessary to provide the right command signal but also to monitor the amount of actuation performed by the devices. Examples are valve position, control rod position, switch positions, power on-off condition, etc. Actuators are used in control and safety systems of a NPP.

Remarks

See Garland (2019) for details of instrumentation in the CANDU reactor. Vaidyanathan et al. (2012) describe instrumentation in sodium fast reactors. See Kerlin and Upadhyaya (2019) for a summary of instrumentation in high temperature gas-cooled reactors and molten salt reactors.

Control of PWRs and BWRs

Introduction

Control systems have two objectives: (1) to move the process from one set point to another set point as quickly as possible without compromising safety and exceeding process parameter limits; (2) to regulate the process at a desired operating condition by overcoming the undesirable effects of external and internal disturbances.

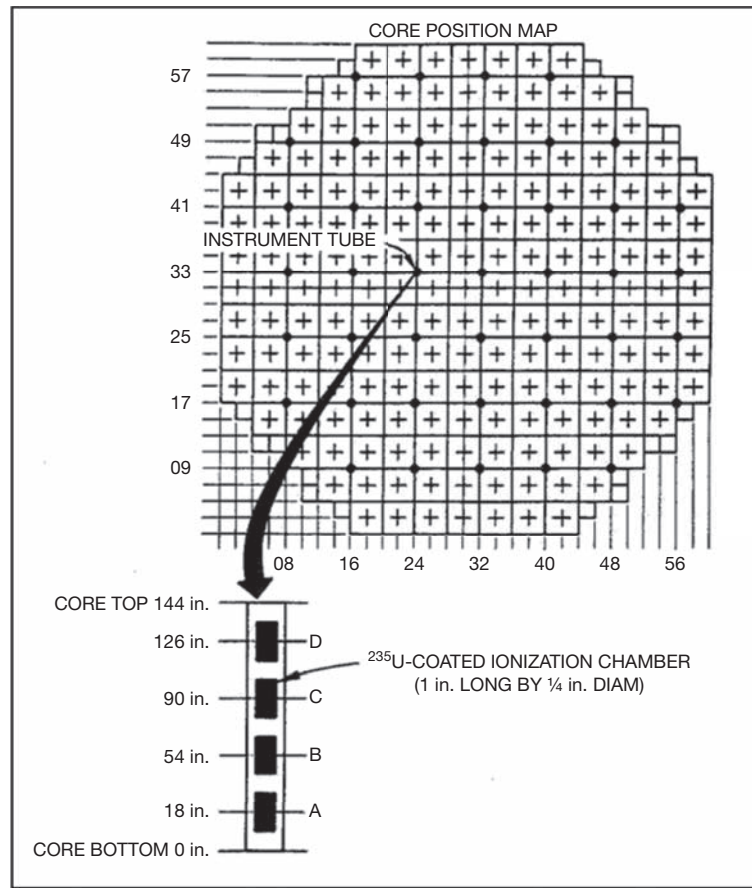


Fig. 6 BWR core map showing LPRM detector string locations. Detector locations are indicated by dark circles and control assemblies by crosses (+). Fry DN, March-Leuba J and Sweeney FJ (1984) Use of Neutron Noise for Diagnosis of In-Vessel Anomalies in Light Water Reactors. ORNL/TM-8774.

The primary control objective in a nuclear power plant is to regulate the plant output at a desired power level. In some situations, it may be necessary to maneuver the plant to follow a variable load demand. The control actions result in changes in process parameters. The control strategy should maintain the plant in a safe mode and ensure that all parameters are within safe limits. Because of coupling among plant systems, control maneuver in one system would result in the activation of other control loops. Most control strategies are of the closed-loop type. The following topics are presented in this section:

- Basic control actions
- Control strategy in PWRs with U-tube and once-through steam generators
- Control strategy in BWRs and impact of BWR evolution.

Control actions

Simple controllers

Most industrial control systems are of the closed-loop type. In an open-loop controller, the system input or perturbation is not affected directly by its output. An implicit calibration between the input and the output exists, such as in opening and closing a water tap to achieve a desired flow rate. In a closed-loop control system, the output is compared with its set point value, and the control action (actuation) is applied to minimize the error between the set point and the measured output. The error signal, $e(t)$, is given by.

Error signal, $e = \text{Set point, } X_{set} - \text{Measurement, } X_m$

$$e(t) = X_{set}(t) - X_m(t) \quad (3)$$

Simple control actions are on-off and proportional controllers. These control actions are enhanced by using a dead band, such that the control action does not change (or remains the same) when the error falls in a pre-defined error band.

Proportional-integral-differential controllers

The error signal using a proportional controller remains non-zero at steady-state conditions. This can be overcome by adding a control action that depends on continuous integration of the error. Such controllers are called integral controllers. A proportional-integral (PI) controller has the form

$$f(t) = K_p e(t) + K_i \int_0^t e(r) dr \quad (4)$$

$f(t)$ = control action; K_p = proportional constant; K_i = integral constant.

In some dynamics, where there is a delay between the control action and the system response, the system dynamics may become unstable because of the time mismatch between error signal and the control actuation. This issue may be resolved by using differential control action in the form

$$f_D(t) = K_d \left(\frac{de}{dt} \right) \quad (5)$$

This control action has an anticipatory effect on the behavior of the error signal. In general, a combination of the three control actions may be used as shown in Eq. (6).

$$f(t) = K_p e(t) + K_i \int_0^t e(r) dr + K_d \left(\frac{de}{dt} \right) \quad (6)$$

This control design is referred to as a proportional-integral-differential (PID) controller. Some experimentation is necessary for choosing the PID parameters K_p , K_i , and K_d . Differentiating an error signal may result in jumps (due to noise) in the parameter (de/dt) , and thus cause unnecessary variations in the control signal $f(t)$. This is avoided by filtering the error signal $e(t)$ and then performing the differentiation. PID controllers are the most used control strategies presently in the process and power industries.

Major control systems in a PWR

The following are the important control loops in a PWR:

- Reactor power control through reactivity insertion by control rod actuation.
- Primary loop pressure control using spray systems and heater banks in the pressurizer.
- Pressurizer water level control using CVCS flow perturbation either to increase or to decrease the level to the set point.
- Steam generator level control in a U-tube steam generator using feedwater flow perturbation.
- Steam generator pressure control in a once-through steam generator by varying feedwater flow demand that changes as the steam flow demand to the turbine changes.
- Turbine power by actuating turbine control valve either to match power set point or to regulate steam pressure set point.

These control strategies are presented for PWRs with U-tube steam generators (UTSGs) and once-through steam generators (OTSGs).

Steady-state temperature program in a PWR with UTSGs

The purpose of the reactor power control system in a PWR is to respond to the changes in the load demand and to regulate the power at a given steady-state condition. Sudden changes in the turbine demand can be handled without scrambling the reactor, by bypassing the steam. Automatic control is employed for power levels above approximately 20%. The set points of the control system are based on the desired power level. Such a procedure is called a steady-state program.

Consider steady-state conditions. The heat balance on the primary side is given by

$$P_R = WC_p (T_{HL} - T_{CL}) \quad (7)$$

P_R = reactor power; W = total core coolant mass flow rate; C_p = specific heat capacity of coolant; T_{HL} = average hot leg temperature; T_{CL} = average cold leg temperature.

The power delivered to the secondary side, P_s , is given by

$$P_s = (h_{eff} A)_{SG} (T_{avg} - T_s) \quad (8)$$

h_{eff} = average effective primary-to-secondary heat transfer coefficient for the steam generator; A = total heat transfer area in the steam generator; $T_{avg} = \frac{T_{HL} + T_{CL}}{2}$, average primary coolant temperature; T_s = average temperature of the secondary fluid, which is at saturated condition in most of the shell-side volume.

Eq. (7) indicates that, for a fixed coolant flow rate, $(T_{HL} - T_{CL})$ increases with reactor power. From Eq. (8), the power delivered to the secondary side can be affected by the three parameters $(h_{eff} A)_{SG}$, T_{avg} and T_s . It is desired to maintain almost constant steam temperature T_s in order to minimize the duty on the turbine. This requires that either $(h_{eff} A)_{SG}$ or T_{avg} , or both change with the power level. Also note that Eqs. (7) and (8) do not require that the cold leg temperature T_{CL} be constant, increasing, or decreasing. The variations in T_{CL} and T_s depend on the type of steam generator, and in general these temperatures adjust to the desired steady-

state conditions. That is, T_{CL} and T_s are not directly controlled. The steam generator pressure remains at approximately constant value due to the manipulation of the turbine control valve as the turbine demand changes and the steam production changes accordingly. In a UTSG, $(h_{eff}A)_{SG}$ remains approximately constant. But, T_{HL} increases with power, its rate of change being set by the power level. Thus, the quantity that changes in order to deliver necessary secondary power is the average primary coolant temperature, T_{avg} . The programmed change in T_{avg} as a function of turbine load (power) is called the T -average program, or the steady-state temperature program.

The following is the sequence of actions during a load-following maneuver: the case of load change from 80% to 100%. Control in the range 0–20% power is primarily manual.

- The action by the operator creates an error signal between the desired turbine load and the actual turbine output. The turbine header pressure (inlet pressure to the turbine) is directly correlated to turbine output.
- The error signal goes to the turbine control valve (TCV) which is manipulated to minimize the error. For an increase in the turbine load demand, the error increases causing the TCV to open.
- The mismatch between the turbine power and the actual measured reactor power initiates a signal to move the control rods, causing an increase in the reactor power. This may be considered as a feed-forward control action. The control action uses the rate of power mismatch. If the power mismatch rate is zero, then this error signal is not utilized by the controller. Also, the gain (or the rate constant) may be changed, a larger value for a larger rate of change of power mismatch, thus providing a quicker action during power maneuvers. This variable gain is sometime called a non-linear gain action.
- The $\{T_{avg}(\text{ref}) - T_{avg}(\text{actual})\}$ error is generated based on the actual turbine output. The turbine output is used for calculating the percent power in the T_{avg} – Power program.
- As the error between the reference T_{avg} and measured T_{avg} increases, the control rods move to increase the reactor power. The total error signal to the control rod drives is a combination of power mismatch signal and the $\{T_{avg}(\text{ref}) - T_{avg}(\text{actual})\}$ error signal.
- The three-element controller generates a signal to feed control valve (FCV) to supply desired feed water rate, and in turn regulate the water level in the steam generator (above the tube bundle). The feed pumps are variable speed, and the pump control tries to maintain constant pressure drop across the FCV.
- There is a pressurizer level versus reactor power program that defines the pressurizer level. Depending on the level mismatch, the charging or the let-down flow changes.
- As the reactor power increases, the measured T_{avg} increases. The control rod motion stops once $\{T_{avg}(\text{ref}) - T_{avg}(\text{actual})\}$ is within a specified dead band.
- The TCV actuation ceases when the desired turbine load matches the calculated turbine output.

Fig. 7 shows the programmed T_{avg} used in a PWR with UTSG, similar to that in a 1100 MWe Westinghouse plant (Kerlin and Upadhyaya, 2019). A PWR with UTSGs operates in the reactor-follow-turbine mode.

Full-length control rods are used for power control. These are divided into four banks A, B, C, D and are moved according to a pre-specified program during start-up. During transient operation, both power mismatch (converted to a temperature scale) and T_{avg} mismatch are used as error signals to determine the rod speed. During steady-state regulation only T_{avg} signal is used for control.

The full-length control rod clusters are operated by a magnetic jack mechanism. The part length rods are manually operated and are used for power shaping purpose. The control rod program is shown in Fig. 8. The rod motion has a dead band of ± 1.5 °F. The rod motion goes to zero for errors in the range ± 1.0 °F. The dead band includes a 0.5 °F lock-up. This avoids a continuous chatter in the control rod motion due to small errors when near a 1 °F temperature mismatch signal, thus ensuring mechanical integrity of the actuator.

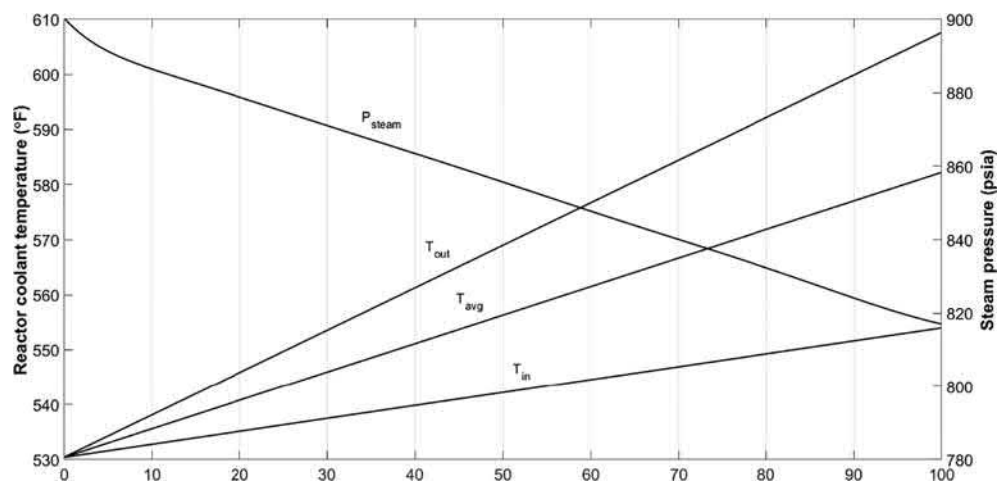


Fig. 7 Steady-state temperature program (sliding average temperature program) as a function of reactor power (%) for a typical 1100 MWe PWR with UTSGs. Kerlin TW and Upadhyaya BR (2019) *Dynamics and Control of Nuclear Reactors*. Cambridge, MA: Elsevier—Academic Press.

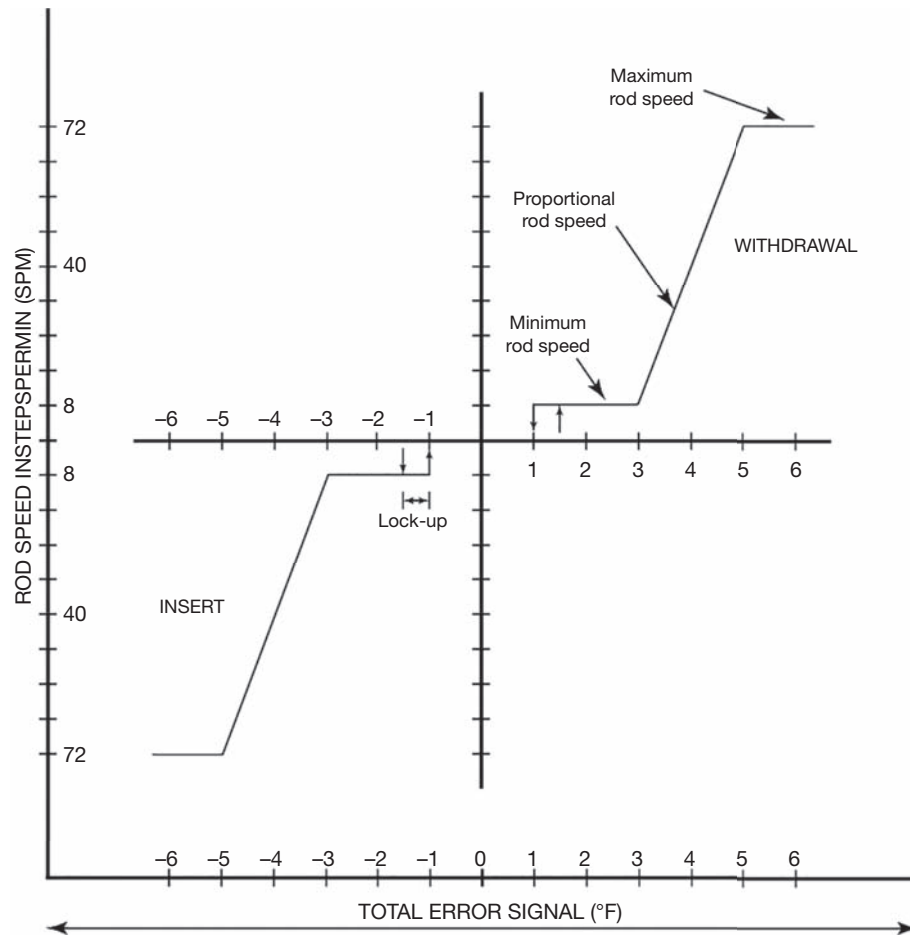


Fig. 8 Reactivity controller and control rod speed program used in a typical Westinghouse PWR. SPM: steps per minute. Adapted from the U.S. Nuclear Regulatory Commission. Westinghouse Electric Company (1984). *The Westinghouse Pressurized Water Reactor Nuclear Power Plant*. Pittsburgh: Westinghouse Electric Company, Water Reactor Division.

Feedwater control in a PWR with UTSGs

The purpose of feedwater regulation in a UTSG or a recirculation type steam generator is to control the flow of water to the steam generator in order to maintain the level of water in the downcomer (the level of water above the top of the tube assemblies).

A mixture of saturated water and steam is generated in a UTSG and flows to the top of the steam generator where the water is separated from the mixture in the steam separators and passes through steam driers before entering the turbine. The separated water flows back to the downcomer region.

Consider the situation when the steam pressure is initially decreased because of increased flow rate to the turbine due to valve opening. This decrease in the pressure causes flashing of the water and as the saturated mixture flows to the steam separators, more water gets recirculated, and flows into the downcomer region. This causes a temporary increase in the water level. This effect is called the *swell phenomenon*. As a result the controller tries to bring the level back to the set point by decreasing the feed flow rate, even though more steam is taken out of the generator. There should be an increase in the feed flow rate. When there is a reduction in the steam flow rate, the opposite happens and there is a shrink in the water level, or the occurrence of a shrink phenomenon. This behavior in a UTSG is called the *shrink-and-swell* phenomenon, resulting in a change in the water level (initially) opposite to the real situation, thus prompting the controller to apply a wrong control action.

For this reason, the control system uses two error signals—mismatch in the steam and feed flow rates and mismatch in the level set point and the measured level. The sum of the water level error signal and the steam flow-feed water flow rate error signal is used to increase or decrease the feed water flow rate to the steam generator. Fig. 9 shows the block diagram of the three-element controller (Kerlin and Upadhyaya, 2019). It is called a three-element controller because three signals are used by the controller to determine the feed flow rate. These are water level, steam flow rate and feed water flow rate.

The error term that is used in the three-element controller to control the feed flow, is a combination of level signal error and the mismatch between steam flow rate and feed flow rate.

Level error, $\Delta L = \text{SG level set point} - \text{SG measured level signal}$

Flow error, $\Delta W = \text{Steam flow rate} - \text{Feed water flow rate}$

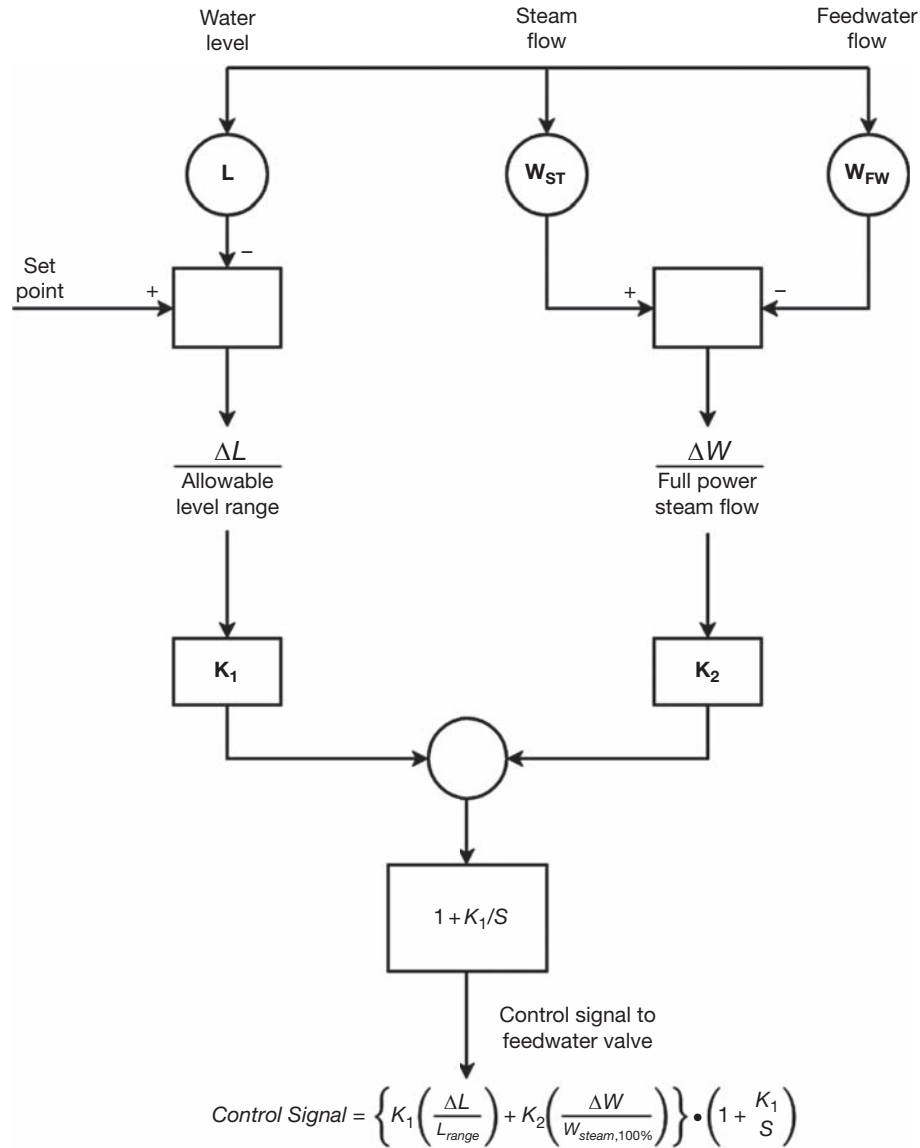


Fig. 9 Three-element controller for a U-tube steam generator level control. Kerlin TW and Upadhyaya BR (2019) *Dynamics and Control of Nuclear Reactors*. Cambridge, MA: Elsevier—Academic Press.

Since the flow error and the level error have different physical units, the errors are normalized by the corresponding nominal ranges of these variables. The total error signal feeds into a PI controller that generates a perturbation signal to actuate the feed flow valve. Note that there is no controller for regulating the steam pressure in the UTSG. The change in the steam flow rate (through the TCV) and the resulting change in the feed flow rate determines the steam pressure.

Steady-state temperature program in a PWR with OTSGs

Unlike in UTSGs, the overall heat transfer coefficient from the primary to the secondary fluid in an OTSG changes with power level. The boiling length increases as the power increases to generate more steam, and the portion of the tube surface available for heat transfer in the superheated region decreases. The steam temperature initially follows the hot leg temperature, and then starts to decrease with increasing power.

Referring to Eq. (8), repeated below, since the overall primary-to-secondary heat transfer coefficient increases (with reduced superheat) and the steam temperature decreases with power, a PWR with OTSGs can be operated with a constant T_{avg} program in most of the power range (15–100%). It is necessary to maintain proper water level in the steam generator at low power levels.

$$P_s = (h_{eff}A)_{SG}(T_{avg} - T_s) \quad (8)$$

Because of the increasing boiling with increasing power level, the cold leg temperature decreases with power. Fig. 10 shows the steady-state temperature program for a PWR with OTSGs, such as in Babcock & Wilcox (B&W) PWRs.

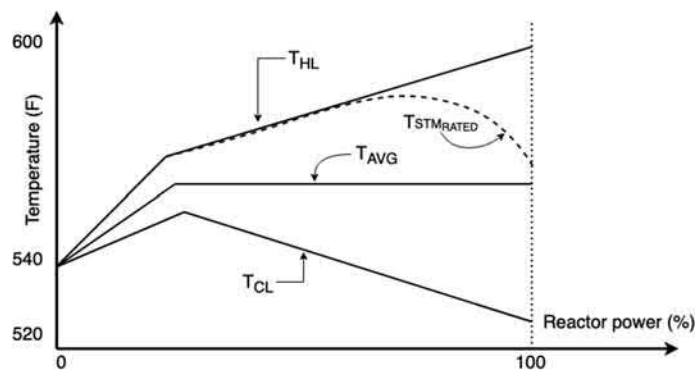


Fig. 10 Steady-state temperature program for a PWR with OTSGs. Kerlin TW and Upadhyaya BR (2019) *Dynamics and Control of Nuclear Reactors*. Cambridge, MA: Elsevier—Academic Press.

Feedwater control in a PWR with OTSGs

The control strategy of PWRs with OTSGs is different from PWRs with UTSGs. The OTSG generates superheated steam; the downcomer level is not constant because of changing boiling length. The feedwater flow rate must match the steam flow rate. The feed flow rate increases with the power level (turbine demand) and changes according to a feed flow rate—power program. The steam pressure is regulated at a set point by actuating the steam valve. For example, as the steam production rate increases the steam pressure increases, and more steam is let out of the steam generator, thus bringing the pressure back to its set point. PWRs with OTSGs operate in a turbine-follow-reactor mode.

Turbine control

In PWRs with UTSGs, the TCV responds to a turbine demand and actuates the valve to regulate the steam flow until the mismatch between the turbine demand and actual turbine power is minimized. In PWRs with OTSGs, a change in the steam generation results in a change in the steam pressure. The TCV actuates to change the flow rate so that the steam pressure settles at its set point.

Primary pressure control in a PWR

The primary pressure in a PWR is regulated by a pressurizer which is connected to one of the hot legs using a surge line. The pressurizer contains water and steam, with 60% of the volume filled with water at 100% power. The pressurizer is the only component in the primary system that contains vapor and a free liquid surface. The overall height of the pressurizer in a four-loop plant is ≈ 16.1 m with a diameter of 2.3 m, and a total volume of 31 m^3 . The pressurizer pressure is controlled by electric heaters and a spray device. The bank of heaters has a total power of 1800 kw. The spray device receives water supply from one of the cold legs and has a maximum flow rate of 57 L/s, with a minimum continuous water injection at 63 mL/s.

The nominal primary coolant pressure is 2250 lbf/in^2 (≈ 153 bars). The primary pressure (pressurizer pressure) is regulated using the heater banks and the spray device, respectively to increase or decrease the pressure. There is a combination of on-off and proportional heaters that are actuated as a function of pressure set point error. The spray flow rate is proportional as a function of the pressure set point error. The water level changes due to the change in the hot leg temperature. The water inventory in the pressurizer vessel is controlled by charging and let-down flows provided by the chemical and volume control system (CVCS).

Control of boiling water reactors

The dynamics of boiling water reactors is different from those of PWRs because of the two-phase fluid in the reactor vessel. This affects the core reactivity due to changes in the two-phase fluid density changes (also called void reactivity). This dynamics is unique to BWRs and must be taken into account during reactor start-up, load-following, and steam pressure regulation.

A BWR has the following three major control systems:

- Reactor power control
- Reactor vessel water level control
- Steam pressure control.

BWR power control

The power generation and control in a BWR is achieved by a combination of control rod reactivity and recirculation flow rate. During start-up, criticality is achieved by control rod withdrawal up to about 30% power, with natural circulation providing core flow. Beyond this and up to about 65% power, a combination of control rod withdrawal and increased recirculation flow rate (to achieve a core flow rate of $\approx 40\%$). The control rod actuation ensures that the power reaches the power trajectory line

referred to as the “100% line.” Beyond this point in the power-flow map, reactor power is increased to 100% by increasing the recirculation flow to achieve a core flow rate 100%. Recirculation flow is also used for load-following in the 75–100% power range. It is important to avoid instability in BWR operation with low flow-to-power ratio up to about 30% power. BWRs operate in a turbine-follow-reactor mode, ensuring that enough core flow is available as the reactor power is steadily increased.

The steady-state thermal power generated in a BWR is calculated using the following equation.

$$P_R = W_{ST}(h_{ST} - h_{FW}) \quad (9)$$

P_R = reactor power; W_{ST} = flow rate of steam exiting the reactor vessel; h_{ST} = enthalpy of steam at the exit quality; h_{FW} = enthalpy of feedwater at the water inlet temperature.

Any vessel heat loss to environment should be added to Eq. (9) in estimating the total reactor power. Accurate measurements of reactor pressure, steam flow rate, and feedwater temperature are important for minimizing the error in calculating the total reactor power generated.

Reactor vessel water level control

The water level in the reactor vessel is maintained at a set point value above the active core. This is similar to the downcomer level control in a UTSG. A three-element controller, using measurements of vessel water level, feedwater flow rate, and steam flow rate, is implemented to ensure enough flow is supplied to the vessel. This controller responds to the “shrink-and-swell” phenomenon, similar to that in a UTSG dynamics. As the water level is restored to its set point, the mismatch between the steam flow and feedwater flow is minimized.

Steam pressure control

As the reactor power increases to respond to a change in the power demand, steam generation increases, causing an increase in steam pressure. This sends a signal to the TCV actuation to increase the valve opening, thus increasing the steam flow rate and reducing the pressure back to the set point. The TCV actuation ceases when the steam pressure reaches the set point.

Consider a load-following operation to increase the power from 80% to 100%. The recirculation flow increases to a predetermined value, thus increasing the core flow. The increased coolant flow reduces coolant void fraction which increases core reactivity causing an increase in reactor power, and in turn increased steam production. This results in an increase in the steam pressure that sends a signal to the TCV to increase the valve opening. This increases the steam flow to the turbine and decreases the steam pressure. The valve actuation ceases when the steam pressure settles to its set point value. At the same time, the feedwater valve actuation increases the feedwater flow rate to regulate the vessel water level. This sequence of control actions in a BWR is similar to the turbine-follow-reactor operation. Control action is also initiated if there is a mismatch between generator frequency and grid frequency (GE Hitachi Nuclear Energy, 1972).

Safety systems

Introduction

Safety systems in a nuclear power plant are designed to sense abnormal occurrences and/or accidents in the reactor and to initiate automatic actions to place the plant in a safe condition. The primary purpose of safety systems is to protect the physical barriers that contain radioactive fission products in the reactor core and, thereby, prevent exposure of the general public. Safety systems have two objectives: (1) terminate the nuclear chain reaction and (2) remove residual thermal energy from the core. The first objective is accomplished through a reactor protection or trip system, which rapidly inserts safety assemblies of neutron absorber material into the core to “starve” the fission chain reactions and end the sustained nuclear generation of thermal power. The second objective is accomplished through actuation of engineered safety features to remove residual heat from the core arising from fission product decay. These protective I&C systems, reactor trip system (RTS) and engineered safety features actuation system (ESFAS), act automatically when design basis event conditions are detected to ensure avoidance of excessive energy buildup (heat and pressure) that could compromise the protective barriers.

Safety system architectures

Safety systems are typically implemented in redundant configurations consisting of (1) sense and command divisions for detecting events and determining when a safety action is required and (2) execute trains for coincidence logic voting on the safety commands issued by the sense and command divisions and then actuating a safety response when required. These safety I&C architectures generally provide instrumentation channels for each protective function within the redundant divisions, feeding voting structures for actuation signaling within a one-out-of-two equipment train scheme. The approach to implementing safety functions is for the sense and command divisions to compare critical plant parameters to trip set points associated with postulated initiating events and for the execute trains to initiate protective action based on coincident occurrence of trip set points being exceeded. The sense and command divisions consist of independent sensors and electronics while the execute trains consist of independent voting arrays (e.g., relays, breakers) and actuators (e.g., control rod drive mechanisms, valves, pumps). The voting scheme for PWRs and advanced BWRs is typically two-out-of-four (or 2oo4) general coincidence among the trip/no-trip outputs from the sense and command

redundancies. The traditional voting scheme for BWRs is one-out-of-two taken twice (i.e., at least one of two redundancies in each train indicating trip conditions).

Reactor trip system

The RTS initiates a reactor trip or scram to prevent the reactor from operating in a condition that could cause damage to the core. The tripped state is achieved by the rapid insertion of safety assemblies (e.g., rods or blades). For PWRs, this is accomplished by de-energizing the electromagnetic coupling of the safety rods to the control rod drive mechanism, which causes the rods to drop into the core from above. For BWRs, this is accomplished by de-energizing relays that release pressurized gas from accumulators to hydraulically push the safety blades into the core from below. In each case, the trip action stops the nuclear chain reactor and thereby limits peak fuel centerline and cladding temperatures along with coolant system temperatures and pressures. The RTS along with safety relief valves and the ESFAS constitute the reactor protection system.

Engineered safety features actuation system

Upon detection of accident conditions, the ESFAS actuates or realigns safety-related systems and equipment assigned to engineered safety feature functions to ensure that accident consequences are kept within acceptable limits. These features include emergency core cooling systems, residual heat removal systems, and isolation valves. Additionally, BWRs provide the Standby Liquid Control System to inject negative reactivity through liquid chemical control as an alternate to reactor trip by control element insertion. This feature serves to address anticipated transient without scram (ATWS) scenarios. PWRs also provide ATWS functions to enable diverse automatic means of initiating rod drop.

Digital I&C modernization

Introduction

The benefits of digital technology include the following ([IAEA Nuclear Energy Series, 2011](#)): improved accuracy of instrument and controller outputs; absence of drifts in electronic component; ease of implementing complex mathematical functions; capability for high-capacity data capture, information extraction, and integrated O&M actions and activities; on-line monitoring, diagnostics, and fault correction; self-diagnostics of instrumentation and protection systems; and improved human system interaction. The deployment of digital I&C technology can enhance plant safety and economic benefits. Additionally, analog equipment is becoming more difficult to acquire in a marketplace dominated by the process industry, which has embraced the benefits and flexibility of digital technology. The increasing obsolescence of analog I&C technology for power plants hampers the nuclear power industry, but regulatory uncertainty constrains the transition to digital. Modernization of I&C systems at NPPs requires special treatment of the characteristics of digital technology and the focused enforcement of key safety principles ([Upadhyaya et al., 2013](#); [IAEA Nuclear Energy Series, 2015](#)).

Characteristics of a digital I&C system

The fundamental difference between analog and digital I&C systems relates to the way plant signals/data are acquired and processed functionally. In a digital I&C system, the signals from various sensors and devices are sampled at a specified rate (for example, samples/s) and the digitized data are transmitted to processing modules to provide desired outputs for use by other plant systems and personnel. The nature of digital systems tends towards aggregated functions that are executed sequentially in software code using data “snapshots.” Conversely, an “analog I&C system responds to instantaneous values of continuously varying signals, and the information is propagated and processed in parallel as part of separate functions embodied in discrete, dedicated components” ([IAEA Nuclear Energy Series, 2011](#)). Despite the identified benefits of the technology, the capabilities and characteristics it provides also pose challenges to developing a safety justification for implementing protection functions in digital systems. For example, the integration of function in fewer devices, time-sequence dependence of data and computation, and the abstraction of functionality in software versus analog physical processes can result in complex systems that are difficult to analyze, too complex to model, and subject to an untestable number of potential states. Consequently, the nuclear power industry employs rigorous quality assurance and qualification processes throughout a digital system’s lifecycle.

Safety principles for digital I&C systems

I&C systems at NPPs adhere to safety principles that originated and evolved from the outset of reactor development. The most fundamental principle is defense-in-depth. The general design criteria (GDC) provided in Appendix A of Title 10, Part 50 of the Code of Federal Regulations (10 CFR 50) establish the minimum design requirements for light-water reactors. These design criteria embody principles such as high quality, integrity, reliability, independence, and qualification. Independence and redundancy, as well as physical barriers and electrical isolation, are generally applied as design measures to address potential vulnerabilities related

to a single failure of equipment and the propagation of failure effects. Diversity is the prevalent means of addressing common-cause failure (CCF).

The U.S. Nuclear Regulatory Commission (NRC) and digital I&C vendors have identified several issues associated with the application of digital technology that must be addressed in the *design, licensing, installation, and operation of a digital safety system* (Upadhyaya et al., 2013; IAEA Nuclear Energy Series, 2015; National Research Council, 1997). The primary safety principles that involve unique considerations for digital I&C systems are.

- Diversity and defense-in-depth (D3)
- Quality assurance
- Cyber security

Diversity and defense-in-depth (D3)

Defense-in-depth (DiD) is a fundamental safety principle for the design, construction, and operation of nuclear reactors with a substantial historical basis. It can be described in terms of a concentric arrangement of protective barriers to ensure public health and safety. Before any harmful radiological release can occur to adversely affect the public or the environment, all of the barriers (i.e., fuel rod cladding, reactor coolant system pressure boundary, containment, and emergency response) must be breached. I&C systems have an important role in maintaining the integrity of these barriers. The DiD principle, as applied to I&C systems, was advanced mainly in response to the anticipation that degradation may not be wholly predictable, and that safety can be best assured by multiple independent lines (or echelons) of defense. Thus, NPP I&C architectures provide progressively compensating systems for “active” control and protection, even should one or more of the systems fail.

DiD within I&C architectures provides independent means to maintain desired operational conditions, prevent accidents, and ensure adequate protection during adverse events (e.g., failures). The U.S. NRC has identified four echelons of defense. These overlapping and redundant systems are defined as the RTS, the ESFAS, the plant control system, and the monitoring and indicator system(s) in the main control room. The means of accomplishing a safety objective for a specific echelon of defense can involve either avoidance of adverse conditions or mitigation of their effects.

A companion safety principle in the application of DiD within an I&C architecture is independence, which has the objective of preventing the failure of one system from inducing the failure of another system. This objective is achieved by the enforcement of independence among the echelons of defense. Independence is ensured through physical separation, electrical isolation, and functional segregation. The I&C systems must not be coupled physically, electrically, or functionally to ensure that failure of one system does not compromise the function of other systems. The capability of digital technology to aggregate functions and interconnect systems through networking poses a special challenge to ensuring independence among digital systems. Thus, I&C systems designs consider communications independence so that an echelon does not functionally depend on output (e.g., data, commands) from other echelons.

Another safety principle arises from the single failure criterion expressed in the GDC. A single failure is a random occurrence that results in the loss of capability of a system to perform its safety functions. The common approach to address single failures is the application of redundancy as a design principle for safety systems. Consequently, protection systems, such as RTS and ESFAS, consist of independent redundant divisions supporting a voting scheme for actuation, as described in the discussion of safety system architectures above. The independence of the divisions (or channels) is maintained from the sensor through the system voting. In effect, each division consists of dedicated sensors and electronics that are redundant to and independent of the equipment of the other divisions. Thus, the failure of one division does not propagate to affect the function of the other divisions. Additionally, the voting of the division commands for actuation mitigates the failure of any one division. Digital technology offers the capability for self-testing and error-checking and the avoidance of single random failures because such conditions can be more readily detected and mitigated than for analog systems.

The safety principle of diversity and DiD, or D3, serves to address the potential for concurrent failure of multiple systems or divisions due to activation of a systematic fault common to each system. This vulnerability is of special concern for safety systems because they operate in an on-demand fashion (i.e., safety systems only take action when a PIE is detected). Consequently, safety systems subject to CCF can “fail silent,” resulting in their undetected unavailability should a design basis event occur. Common systematic faults can result from various sources (e.g., requirements deficiency, implementation error, or manufacturing defect) and may not be detected during design, implementation, or acceptance testing. By itself, the redundancy of identical divisions in safety systems offers no protection against CCF. Thus, even for well-established analog-based I&C system designs, the potential for CCF of multiple systems (or redundancies within a system) constitutes the principal credible threat to defeating the DiD provisions within NPP I&C system architectures. For analog safety systems, the primary approach to address CCF is to establish suitability through equipment qualification and thereby minimize the potential for a common environmental stressor to cause concurrent failure of redundant equipment. It is generally accepted that the unique characteristics and inherent complexity of digital I&C systems can exacerbate vulnerability to CCF.

Diversity, in the context of DiD, is the primary means employed to address CCF vulnerability. The principle of D3 was employed in the first reactor, Chicago Pile #1 (CP-1). For CP-1, three reactor trip capabilities were provided: (1) automatic control rods, (2) a manually initiated, gravity-driven shutdown rod, and (3) manual liquid control (i.e., reactivity control by flooding the pile with a cadmium-salt solution). Thus, the original application of DiD for safety systems was enhanced by equipment and functional diversity, which relate to different actuation means and different underlying methods, respectively.

The approach to ensuring safety through diversity provides different means to accomplish the same or equivalent function to compensate for a CCF that disables one or more echelons of defense. Thus, diversity is complementary to the principle of DiD and increases the likelihood that a line of defense will be actuated when needed. For current NPPs that have primarily analog-based safety systems, diverse anticipated transient without scram (ATWS) systems are employed to provide additional safety assurance through alternate means of responding to design basis events. For more recent NPPs and those undergoing extensive modernization, the common approach is to implement a non-safety diverse actuation system (DAS) to act should the safety systems fail due to CCF.

The application of diversity to I&C systems involves approaches such as sensing different parameters, using different technologies, using different logic or algorithms, or using different actuation means to provide several ways of detecting and responding to a significant event. As noted in the CP-1 example, the foundational use of diversity involved functional diversity. In common practice, functional diversity is generally coupled with signal diversity. Basically, different underlying methods to detect design basis events is achieved by using different sensors (e.g., pressure vs. temperature) to serve as diverse indications of PIEs.

For digital I&C systems, the potential for and consequences of CCF are evaluated through a D3 analysis (U.S. Nuclear Regulatory Commission, 1994). Based on analysis results, diversity may be introduced into the I&C system architecture to mitigate identified CCF vulnerabilities. The value of diversity for digital I&C architectures arises because dissimilarities in technology, function, implementation, and so forth can mitigate the potential for undetected common faults among redundant divisions or independent echelons of defense. Diversity for digital I&C systems is characterized in terms of design, functional, signal, equipment, software, and human (system lifecycle) attributes.

Quality assurance

The regulations in 10 CFR 50 specify that consensus standards and quality assurance practices must be applied in the design, development, implementation, testing, installation, and operation of safety systems. Specific criteria for quality assurance are provided in Appendix B of 10 CFR 50. These criteria are supplemented by regulatory guidance on software quality assurance practices that apply to the entire system lifecycle, from requirement specification to retirement. In the design and development of software, extreme care is assured by requiring a detailed formal program with well-developed and independently verified plans. These plans and associated procedures address aspects of the digital system lifecycle such as requirements and design specifications, design, development, configuration management, installation, testing, maintenance, software management, verification and validation program, operations, lifecycle maintenance. These programs and plans are based on consensus standards and provide the means for developing high integrity software for reliable operation of a safety system.

The nuclear industry typically obtains its components from vendors who adhere to nuclear-grade quality assurance practices. However, because of the small market for nuclear I&C equipment, the availability of nuclear qualified equipment suppliers is limited. Consequently, it is becoming more common for the nuclear industry to employ commercial off-the-shelf (COTS) digital equipment. While this approach brings with it the advantage of the lower cost, a large user base, and an extensive history of performance, it is necessary for the NPP to demonstrate that the equipment meets the same or equivalent quality requirements as nuclear-grade equipment. As a result, it has become a common practice for utilities and other companies to purchase COTS or commercial-grade items and then to qualify them for use in safety systems by performing a special dedication process.

Cyber security

This issue requires the development and implementation of a security program to protect critical digital assets from cyber attack. There are both software and hardware aspects to cyber security. The hardware can be protected by isolating the safety system from external networks and establishing an internal network which allows only one-way communication from the safety system to any other I&C or data systems in the plant. The software, however, can be more problematic because of the need to provide security throughout the supply chain. If the software-based system or its digital support services are altered or infected, either during development or later, then the vulnerabilities to malicious attack or intrusion can be introduced that may result in anomalies in reactor operation or compromise of control system and/or safety system functions. The U.S. NRC has several recommendations for establishing cyber security for digital assets to preserve and protect software integrity.

Integral and small modular reactors

Several designs of small modular reactors (SMRs) are being developed by reactor vendors. The SMRs range in capacity from 25 to 300 MWe. The capability to implement nuclear power plants based on smaller reactors can help match the energy needs for regions of the world that lack highly stable, densely interconnected electrical grids. These reactors can present lower capital costs than large reactors, allow for incremental additions to power generation capacity, and support multiple energy applications (i.e., power generation and process heat supply). The design and configuration of these reactors provide enhanced safeguard capability. Additionally, SMRs can be introduced through phased construction of modules at a plant site to incrementally achieve a large-scale power park (Ingersoll, 2021).

The SMR designs under development include innovative light-water-cooled reactor concepts as well as non-water-cooled reactor concepts. The light-water-cooled reactor concepts tend to be integral primary system designs, in which all of the major NSSS components are in a large reactor vessel. These include reactor core, control rod drive mechanisms, steam generators, coolant pumps, and

in some cases residual heat removal system. SMRs are designed for possible remote deployment with remote monitoring and operation. Ingersoll (2021), and Carelli and Ingersoll (2014) provide excellent descriptions of both light water and other types of SMRs.

As an example of a near-term SMR concept, the U.S.-developed SMR design by NuScale Power is an integral PWR rated at 50 MWe per unit, with natural circulation of the primary coolant. The reactor vessel contains all the NSSS components and is housed in a steel containment which is submerged in a large pool of water. The power generation station is designed for 12 units with a single control room.

Realizing the benefits of SMR deployment requires innovation and advanced capabilities development of I&C technologies to address management of multi-unit, multi-product-stream generating stations and to offset the reduced economy-of-scale savings. Additionally, the unique characteristics of SMR designs compared to traditional reactor designs pose some technical challenges that remain to be resolved. As an example, the unusual geometry and internal flow paths for integral reactors presents a challenge in the measurement of process parameters, as well as neutron flux, that emphasizes the value of in-vessel sensors. IAEA Nuclear Energy Series (2017) describes the characteristics of SMRs that affect the design and implementation of I&C systems and discusses the distinctive features and issues of I&C technologies relevant to SMR plant management and operation.

Summary

Instrumentation and control systems serve as the ‘central nervous system’ of a nuclear power plant. This chapter provides brief descriptions of process instrumentation and neutron detectors used in light water reactors. Reactor control strategies implemented in PWRs and BWRs are presented. The chapter provides a summary of digital I&C. Some of the operating reactors are being modernized with digital systems. New designs of light water reactors utilize digital I&C.

The licensing aspects of digital I&C, including probabilistic risk assessment, are not presented in this chapter. They are covered in Section 4 of this encyclopedia.

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Reactor Operations

Michael Y. Young^a, Westinghouse Electric Company, Pittsburgh, PA, United States

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Glossary

AFD Axial flux difference is the difference between the core power generated in the upper half of the core and the lower half, normalized to the total core power. The operational goal is to maintain AFD within a relatively narrow band around zero.

Burnup A measure of the amount of fission energy released from a given mass of uranium (heavy metal). Unit is Megawatt-days per kilogram (MWd/kg), or Gigawatt-days per ton (GWd/ton). Also referred to as exposure.

BWR Boiling water reactor. See [Hamon \(2021\)](#) for detailed description.

CRD Control rod drive system is the mechanism used to control insertion or extraction of the control rods.

CVCS Chemical and volume control system in a PWR. See section “PWR operation” for detailed description.

Drywell A steel or concrete structure surrounding the reactor vessel in a BWR. It is designed to contain steam released in a loss of coolant accident, and channel it to a water pool for condensation.

IAEA The International Atomic Energy Agency (IAEA) is an international organization that seeks to promote the peaceful use of nuclear energy, and to inhibit its use for any military purpose, including nuclear weapons. The IAEA was established as an autonomous organization on 29 July 1957.

K_{eff} The ratio of rate of production of fission neutrons to the rate of neutron losses (via absorption or leakage). In critical reactor K_{eff} value is maintained at 1.0.

Mode The state of the reactor during startup, normal operation, and shutdown.

Moderator Substance used to slow neutrons generated by fission and with features an efficient neutron kinetic energy loss per collision.

MTC Moderator temperature coefficient (MTC): The effect of a change in the moderator temperature on the core reactivity ($\Delta K_{eff}/K_{eff}/^{\circ}\text{C}$).

^aRetired.

NA Not applicable. Refers to certain table entries in this article that do not contain any information.

Neutron flux Number of neutrons crossing a unit area per second. Reaction rates, such as the fission rate, are proportional to the neutron flux.

NOP/NOT A term used in reactor startup operations to indicate: Normal operating pressure/normal operating temperature

PCI Pellet-cladding interaction. See [Young \(2021\)](#) for details.

PWR Pressurized water reactor. See [Brown \(2021\)](#) for details.

RCA Reactivity control assembly. See [Young \(2021\)](#) for details.

RCS Reactor coolant system. See section “PWR operation.”

Reactivity A term expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

RFP Reactor feedwater pump. A pump or pumps designed to provide feedwater to the reactor vessel (BWR).

RHR Residual heat removal system. See sections “PWR operation” and “BWR operation.”

RWCU Reactor water cleanup system. See section “BWR operation.”

SJAE Steam jet air ejectors are devices used to remove non-condensable gases from a steam system.

Technical Specifications A set of conditions or limits that must be met before a reactor can remain at power or change a reactor condition. Establishes requirements for items such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

USNRC The United States Nuclear Regulatory Commission is the principal government body charged with establishing regulations to ensure the safety of nuclear power plants in the United States.

Xenon A decay product (Xenon-135) created during the fission process. It has a large neutron cross section, which acts to suppress local reactor power. Significant changes in local reactor power affect xenon concentrations, which in turn affect local core reactivity.

Introduction

Reactor operations comprise the activities involved in maintaining the reactor in a stable configuration while operating, and, at intervals of 1 to 2 years, bringing the reactor to zero power and atmospheric pressure so that fuel can be replaced and maintenance performed. These activities are described in this article.

The startup, normal operation and shutdown operations are described separately for the PWR and BWR. The PWR is assumed to be a four loop Westinghouse design, and the BWR is assumed to be a General Electric BWR/4. Other designs will differ in some details, and these differences will be discussed as appropriate. Some of the materials for this article were derived from materials available in the USNRC public document room ([USNRC, 2002a](#)) and ([USNRC, 2002b](#)).

PWR operation

Several components of the plant play critical roles in reactor startup and normal operation and are listed here in summary form. Details on these components are provided in ([Brown, 2021](#)).

Reactor coolant system (RCS): this system consists of the reactor vessel containing the core, pipes connecting the reactor vessel to the steam generators, and reactor coolant pumps to circulate the pressurized water.

Residual heat removal system (RHR): this system’s main function is to provide water to the core during a loss of coolant accident. In normal operation it is secured, but ready for use in case of emergency. During refueling operations, it also serves to remove residual heat and maintain sufficient flow through the core.

Chemical and volume control system (CVCS): this system controls the chemistry of the coolant during normal operation, in particular the concentration of the boron in the water moderator. It also serves to remove corrosion products from the coolant. During refueling and startup it is used in conjunction with the RHR to control pressure and temperature.

Control rods: Control rod movement is important to control both power level and power distributions during normal operation, and to shut down the reactor, as needed. Control rods in a PWR are grouped into “control banks” and “shutdown banks.” These banks are “stepped” into or out of the core by latching mechanisms in the control rod drives, attached to the reactor vessel head. Shutdown banks are used to help shut down the reactor. At a reactor trip signal, all banks are released into the core.

Nuclear instrumentation: The type of instrumentation used to measure neutron flux varies by reactor design. In current Westinghouse reactors, nuclear instrumentation located just outside the reactor vessel (ex-core) monitors the power level by detecting neutron leakage from the reactor core. The neutron flux (neutrons/cm²/s) generated in the core, from shutdown conditions to 100% power, ranges from a few neutrons/cm²/s to over 10¹⁵ neutrons/cm²/s. This large range requires three types of detectors with overlapping ranges. The three different types of neutron detectors are designated source, intermediate, and power. Each

detector is specifically designed to accurately measure neutron flux in its designated range. The source range detector operates when the reactor is critical but is producing essentially zero power. It is typically a proportional counter (Han, 2021) that detects ionization events (counts per minute) directly. The intermediate range detector, active up to 120% of rated power, is typically a compensated ionization chamber, where the neutron flux induces a current in the detector circuitry (hence, the power state of the reactor in this range is described in terms of “amps,” where 10^{-8} amps corresponds to approximately 0.001% thermal power). The power range detectors, applicable from 1% to 120% power, are also ionization chambers, calibrated to convert the neutron flux measured to percent rated thermal power.

Reactor startup

Reactor startup proceeds in the form of a countdown, through a series of “Modes.” These are specifically defined in the plant Technical Specifications (USNRC, 2012a, 2012b) as shown in Table 1 below. The values given below for average coolant temperature are from the Standard Technical Specifications; there may be some variation among plants. These specifications tightly control the tests that shall be performed and the conditions that must be met prior to proceeding to the next Mode.

Heatup and power ascension (Modes 4 to 2)

The startup evolution begins from the point when the core has been refueled, the reactor vessel has been resealed, and the RCS has been filled with water. The reactor coolant pumps are off, so coolant circulation is maintained by the RHR pumps. The reactor core is generating a small amount of residual heat at this time. This heat is removed from the core by the RHR system.

The RCS is completely full of water, and the pressure in the RCS is controlled between 320 and 400 psig (2–3 MPa) by maintaining a flow balance between the coolant removed from the RCS, referred to as letdown flow, and the coolant being returned, referred to as charging flow, using both the RHR and the CVCS. The steam generators are in “wet layup,” filled to the 100% level with water, and all secondary systems are secured with the exception of one circulating water pump. The main turbine and both feedwater pump turbines are on their respective turning gears (slowly being rotated to prevent distortion of the heavy turbine rotors). All pre-startup checklists have been completed.

Heatup

A steam bubble is first established in the pressurizer. This ensures that a compressible volume is available in the pressurizer for water expansion caused by the heatup of the coolant. Steam bubble formation in the pressurizer is accomplished by heating the water volume of the pressurizer with the pressurizer heaters. With a steam bubble established in the pressurizer, RCS pressure is controlled by pressurizer heater (to generate steam in the pressurizer) and spray valve operation (to condense steam).

Once a steam bubble is established in the pressurizer, the reactor coolant pumps are started. After all reactor coolant pumps are operating, the RHR pumps are stopped, since they are no longer needed to remove decay heat or to provide forced flow through the core. With all RCPs operating and the RHR system secured, the reactor coolant begins to heat up at a rate of approximately 50 °F (10 °C) per hour, due to the heating energy supplied by the RCPs.

Steam formation begins in the steam generator secondary side. The reactor is now in Mode 4 (Hot Shutdown).

The RCS continues to heat up, until it reaches normal operating pressure and temperature (referred to as NOP/NOT), typically 2250 psia (15.5 MPa), 570 °F (300 °C). The RHR system is now isolated from the RCS and aligned for its normal emergency cooling function. The reactor is still subcritical.

The primary plant heatup is limited by automatic actuation of the steam dumps when the pressure inside the steam header reaches its normal operating pressure, approximately 1000 psig (6.9 MPa). The RCS temperature remains constant, with the steam dumps removing any excess energy that would tend to drive the RCS temperature higher. The reactor is now in Mode 3 (Hot Standby).

Reactor coolant leak rate tests and control rod drop tests are normally conducted at this point.

Power ascension

Boron concentration in the coolant is first adjusted so that the reactor will be near critical at a given desired control rod position. Control rods are then withdrawn in steps to establish criticality. Criticality is reached when there is a steadily increasing source range

Table 1 Typical PWR modes for startup.

Mode	Title	Core reactivity condition (K_{eff})	% rated thermal power	Average reactor coolant temperature
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350 °F (177 °C)
4	Hot Shutdown	< 0.99	NA	200 °F (90 °C) to 350 °F (177 °C)
5	Cold Shutdown	< 0.99	NA	≤ 200 °F (90 °C)

count rate with no control rod withdrawal. After criticality is achieved, the control rods are positioned to increase nuclear power until the intermediate range instruments read 10^{-8} amps (equivalent to approximately 0.001% thermal power). Power is stabilized at this level to record actual critical conditions. Various physics tests are performed to confirm the nuclear characteristics of the core. These data include:

Critical boron concentration (all rods out)
 Moderator temperature coefficient (MTC)
 Control rod reactivity worth
 Core power symmetry

After recording the critical data, the control rods are positioned to increase power to the point of adding heat, approximately 1% power as indicated on the power range instruments, and then to stabilize power at about 2%. As nuclear power increases above the point of adding heat, the fuel and moderator temperatures increase. These temperature increases add negative reactivity because of the negative fuel and moderator temperature coefficients.

As the moderator temperature increases, the coolant expands into the pressurizer, causing the level and pressure to increase. Also, the steam generator temperatures and pressures begin to increase. The steam dump valves modulate open, maintaining the steam generator header pressure at approximately 1000 psia (6.9 MPa).

The power level is maintained at 2% while a main feedwater pump is started and aligned to supply the steam generators through the main feed regulating bypass valves. With a main feedwater pump feeding the steam generators and the auxiliary feedwater system realigned to its normal at-power standby lineup, turbine startup may proceed. When the reactor power reaches and exceeds 5% of rated power, the plant is in Mode 1 (Power Operations). The Technical Specifications for power operation then apply.

With nuclear power at 15%, the main turbine is rolled off the turning gear and accelerated to a specified rpm. With the turbine running, the turbine auxiliary systems are aligned for normal operation, and the turbine protection system is tested to verify proper operation. Generator synchronization is performed after satisfactory turbine protection system testing.

When the main generator voltage, frequency, and phases are matched with those of the grid, the generator output breakers are closed, and the generator picks up a minimum load (typically, approximately 60 MWe for a large nuclear unit). The operators can now increase the generator's electrical output to the grid. To prevent axial xenon oscillations and to maintain peaking factors within Technical Specification limits, the operator must insure that the indicated axial flux difference (AFD) is maintained within predefined limits during power ascension and when full power is achieved (see section "Normal Operation" for description of AFD measurement). This is accomplished by applying strict limits to the rate of removal of control rods. At approximately 50% load, a calorimetric calculation (heat balance) is performed on the secondary system, and an adjustment to the power range nuclear instruments is performed, if necessary. Further calorimetric calculations are performed at 90% and 100% power. These additional calculations are performed to ensure that the power range nuclear instruments are properly calibrated.

Full power operations

During normal operation, all control rods are typically fully withdrawn, and reactivity is controlled by boron in the coolant. The boron concentration typically increases early in the cycle to offset increasing core reactivity as burnable poisons (Young, 2021) within the fuel are burned out, then decreases as fuel is depleted.

Core power is generally controlled by control rod insertion. At all times when the reactor is critical, the control banks must be maintained above their respective insertion limits. Maintaining the rod banks above their respective insertion limits ensures that sufficient negative reactivity is available to achieve and maintain the required shutdown margin in the event of a reactor trip.

For a PWR, the response of the RCS (the primary side) to a change in turbine load occurs at the heat transfer interface with the steam generators (the secondary side). At the simplest level, the heat transfer from the primary side to the secondary side is proportional to the difference between the average reactor coolant temperature, T_{avg} (essentially the average of the core inlet and the core exit temperature), and the average steam-water mixture temperature on the secondary side, T_{stm} . Since the secondary side is a steam-water mixture, the secondary side temperature varies directly with pressure. For example, as turbine load increases, secondary side temperature and pressure fall as more energy is removed than supplied. Increased heat transfer from the primary then reduces T_{avg} . For a PWR, in response to this change, three options are possible:

1. Maintain primary side T_{avg} constant, and let the reduced secondary side temperature increase the heat transfer from the primary side as needed. This requires a minimal change in reactivity, but has the disadvantage (among several others) of significantly reducing steam pressure, with a corresponding deteriorating steam condition (increased moisture carryover).
2. Increase reactor power to increase T_{avg} , so that the secondary pressure remains at the optimal value. The disadvantage here is the potential for significantly higher core exit temperature, which has an adverse effect on system materials, and may cause some bulk boiling in the higher powered assemblies.
3. The optimal approach, taken by all PWRs today to varying degrees, is a combination of options 1 and 2: allow T_{avg} to increase slightly (along a programmed trajectory), with an increase in turbine load. Steam pressure and temperature will fall slightly, but within acceptable limits.

Operating strategies

Baseload operation

This has been the traditional operational strategy for PWRs (and BWRs, see below). The reactor operates at 100% power, 24 h a day, 365 days a year, for up to 2 years, until the fission chain reaction can no longer be sustained and the reactor must be refueled. Normally in a PWR, all control rods are out, and core power is maintained by slowly reducing the amount of boron in the coolant or other poisons in the core. Short term responses to turbine load are accomplished by control rods.

Recently, a “coast down” period has been added to the last few months of the cycle, to increase fuel utilization. In this operational mode, boron has been diluted to the maximum extent practically possible. Core power slowly declines to approximately 80–90% until the plant is shut down for refueling.

Load follow operation

PWR's (and BWR's, see below) have been designed to allow the reactor power to be reduced and increased in a programmed fashion depending on the electricity demand from the grid. In a PWR, reactor power is reduced by adding boron to the coolant, and partially inserting selected control rods. The power change must be accomplished such that changes in axial offset (difference between power in the top half and bottom half of the core) are maintained within limits established in the Technical Specifications. In the United States until recently, it has been more economical to manage load demand by temporarily shutting down or throttling down other generation forms (which comprise over 80% of the total generation). In contrast, in France, where a much larger fraction of the total electrical power is generated by nuclear, other options for responding to load demand do not exist. Consequently, the PWRs in France have operated in load follow for years.

In a load follow operation, the core power is reduced from 100% over several hours to a predetermined level, typically not less than 70%, and then maintained at that level for several days. The power reduction is primarily achieved by control rods, assisted by boron concentration increase. As originally envisioned, load follow would occur at regular, preplanned intervals.

Flexible operation

Flexible operation is essentially load follow, except that required response times are generally much shorter, and the times during which the reactor is at lower power can vary from a few hours to weeks. This is the result of intermittent forms of electricity production, such as wind turbines or residential solar panels that have been connected to the grid. Changes in electricity production from these alternative forms cannot be easily planned in advance.

Economics of load follow operation in PWRs

Reactor operating costs generally fall into three categories:

1. Capital costs
Most power reactors such as those described here have been designed to allow safe operation while responding to changing grid demands, so additional capital investment to allow a plant to load follow is generally small.
2. Operation and maintenance costs
Operation and maintenance (O&M) costs cover all the costs of operating the plant, maintenance and repair, and replacement of supplies and equipment.
3. Fuel cycle costs
Fuel cycle costs generally encompass the cost of uranium mining, enrichment, and the manufacturing of fuel assemblies.

The preferred operational mode for nuclear reactors is, for fairly obvious reasons, base load operation. This is reinforced by economic analysis (Rothwell, 2021) demonstrating that minimizing operating costs and fuel costs while maximizing annual power output are key to reducing the cost of nuclear electricity production for a specific reactor. Minimizing changes in reactor conditions also minimizes operation complexity, wear and tear, and fuel costs. As explained, however, in a recent IAEA publication (IAEA, 2018), consideration of the overall makeup of the electric grid and its various generating sources often takes precedence over increased reactor operating and fuel costs.

Surveillance

Normal operation

During normal operation, the ex-core and in-core instruments described at the beginning of section “PWR operation” are used to monitor for any changes in the overall core power distribution. In particular, the axial flux difference (AFD), the difference in power between the top and bottom of the core is determined from the ex-core power range measurements. This measurement must be kept within specific limits established by the core design and safety analysis at all times.

In addition, detailed assembly power distributions are determined. This is accomplished through use of in-core instrumentation. In most Westinghouse PWRs, the instrumentation, consisting of miniature fission chambers (Han, 2021) and contained within a long sheath, is inserted through the bottom of the reactor vessel into the central guide tube of selected assemblies. The instrument is moved axially within the assembly to measure the local neutron flux. In other PWRs, the instruments remain in place in the core, and measure the local power distribution continuously (while the in-core instrumentation provides real-time data, it is less

detailed). Data from these more detailed measurements are used to confirm maximum local fuel rod linear powers (expressed as kW per meter or per foot of fuel rod, and discussed further in Young, 2021) are also within Technical Specification limits.

At startup and during normal operation, calorimetric calculations (essentially a heat balance, establishing the steam flow and change in enthalpy between the supplied feedwater and the generated steam) are performed to confirm agreement with core power calculated from the nuclear instruments.

In addition, coolant samples are taken at regular intervals, and tested for the presence of various elements and radioisotopes. Generally these measured quantities fall into three categories and indications:

- (a) Boron, lithium, hydrogen: Boron is crucial in controlling core reactivity. Lithium controls the reactor coolant pH, which in modern PWRs is usually set at a constant target value of 7.0–7.2. Hydrogen is added to the coolant via the pressurizer vapor space to ensure that any excess oxygen, which affects corrosion rates, is quickly removed.
- (b) Coolant radio isotopes that are created from stable nickel (Co-58), iron (Mn-54), and chromium (Co-60) isotopes that comprise the major structural materials in a PWR are monitored to confirm no departure from historical values. Departures may indicate unusual corrosion rates or other changes.
- (c) Fission products such as iodine (I-131) and xenon (Xe-133) are monitored. Large changes indicate the presence of leaking fuel rods.

Additional reactor monitoring during normal operation include:

- (d) Control rod drive testing: control rods are moved down and up a few steps, sufficient to confirm the control rod drive mechanisms are operational, without affecting the core power distribution significantly.
- (e) Calorimetric tests to confirm reactor power

The description above does not include the many surveillance functions performed automatically by the reactor protection system. This system continuously monitors system pressure, flow, and many other operating parameters to determine if a serious malfunction has occurred and to automatically activate the safety systems to shut down the reactor.

Reactor trips and restarts

Reactor trips and reductions in power tend to be rare in today's reactors, occurring about once every few fuel cycles. For a shutdown, the reactor is usually maintained in Mode 2 while the problem is addressed. Normally restart occurs within a few hours or days. A typical restart proceeds by removing control rods to achieve 50% power, maintaining this power for a specified hold period, then continuing the power ascension in steps to 100%, again with specified hold periods in between. These hold periods are put in place to allow the fuel rods to "re-condition" to the increased power. The conditioning process that PWR fuel rods require following a power change is similar to that for a BWR and is described in additional detail in the section BWR: Flexible Operation.

Normally, returns to 100% power from reduced power conditions such as occur during load follow can be implemented without any required hold periods, provided the change is of relatively short duration.

Reactor shutdown for refueling

The procedure for reactor shutdown is essentially the reverse of the startup procedure with some key differences. A key objective is to minimize the potential for uncontrolled release of activated corrosion products, initially residing on the core, to the rest of the RCS. To accomplish this, the RCS is first exposed to an acid reducing environment by removing lithium from the coolant, reducing the pH. The purpose is to reduce any accumulated nickel oxide to nickel, so it will be susceptible to dissolution when the chemistry becomes oxidizing. In addition, there will be some limited dissolution of other corrosion products such as nickel ferrite and activated corrosion products. The coolant is then converted to an oxidizing environment. This dissolves most of the Ni and Co-58 metals, and smaller amounts of Fe, Mn-54, and Co-60, which can then be removed by ion exchange before the RCS is opened to the atmosphere. The large amounts of nickel and Co-58, relative to other metals, are due to the large surface area of nickel alloy steam generator tubes that exist in a US PWR. Co-58 generally poses the greatest challenge to maintaining low dose rates during refueling.

A typical shutdown proceeds as described below, although there may be significant differences in the timing and use of RCPs among plants.

Prior to shutdown, one of the mixed bed demineralizers will have the resin replaced with a dual purpose charge that will also remove lithium and will have sufficient capacity to clean up the RCS (including potential elevated Co-58 or Co-60) throughout the shutdown period.

The outage starts when main generator load and reactor power are reduced to 15%. Main turbine overspeed trip testing is typically performed next. Immediately following completion of the overspeed test, reactor power is reduced to less than 5%, and with the reactor in Mode 2 the control rods are inserted to enter Mode 3. Emergency boration is initiated as soon as Mode 3 is entered, and with continued lithium removal, acid reducing (low pH) conditions are achieved in the coolant.

A few hours after reactor shutdown, cooldown from normal operating temperature is initiated, to reach Mode 4 (177°C) within a few hours. During the time period that acid reducing conditions exist, letdown purification flow is maintained as high as possible to remove the corrosion products that are being dissolved. Adjusting RCS inventory is accomplished, as in startup, by adjusting charging flow while keeping letdown flow constant and maximized.

All except one of the reactor coolant pumps are typically secured after reactor shutdown and prior to Mode 5 to minimize heat input and pressure drop loading of the reactor internals.

With RCS pressure less than 1000 psi (6.8 MPa), accumulators (large tanks with water that are used in the event of a loss of coolant accident) are isolated and the RCS pressure is reduced to less than 360 psi (2.5 MPa). Then RHR shutdown cooling is established. Cooldown from approximately 175°C to 90°C on RHR generally requires a few hours. When the RCS temperature falls below 90°C the RCS is taken water solid (pressurizer full).

After approximately 12 h of acid reducing conditions, the dissolved Co-58 activity level is expected to have increased by a factor of about 10. Increases in iron, Mn-54, and Co-60 also occur, although these levels are much smaller.

The next phase in the shutdown procedure is to achieve acid oxidizing conditions, by adding hydrogen peroxide into the RCS. The dissolved Co-58 activity level is expected to increase by another factor of 10 during this time.

During the peroxide addition activities, the RCS is cooled down to between 60°C and 35°C with one RCP operating. The RCS is further depressurized and then partially drained for reactor disassembly.

When the cleanup endpoint is reached, the flood up of the reactor cavity is initiated. With the cavity flooded it is desirable to maintain RHR letdown purification in operation, to the maximum extent possible, to continue to remove corrosion products that are being dissolved at a reduced rate.

Reactor disassembly and refueling operations

The disassembly of the reactor vessel begins with the removal of the missile shield. All cables, air ducts, and insulation are then removed from the vessel head. The thermocouple leads are disconnected from the reactor vessel head, and protective covers are installed over the top of the thermocouple support columns. The in-core instrumentation thimbles are disconnected and pulled back through the bottom of the vessel. The reactor vessel stud tensioners are used to de-tension the vessel studs, and after removal of the studs, stud hole plugs are installed to protect the threads (three stud holes are left unplugged and are fitted with threaded guide studs which help to position the various lifting rigs).

The refueling cavity is prepared for flooding by sealing off the reactor cavity and removing the blind flange from the fuel transfer tube. With the refueling cavity prepared for flooding, the vessel head is unseated and raised approximately 1 ft above the vessel flange. The vessel head and the water level in the refueling cavity are raised simultaneously, keeping the water level just below the head, so that the head can provide temporary protection from radiation. When the water reaches a safe shielding depth, the vessel head is taken to its storage pedestal. The control rod drive shafts are then disconnected and, with the upper internals, are removed from the vessel and placed in their temporary storage location. The fuel assemblies and rod cluster control assemblies are now free from obstructions and the core is ready for refueling.

Fuel handling can start after the reactor has been subcritical for 100 h (Technical Specification requirement). This allows time for the short-lived fission products to decay. The refueling sequence is started with the manipulator crane. As determined in the refueling guide, which is prepared before each refueling, used fuel assemblies are removed from the core. The positions of partially used assemblies that will be operated for a second cycle are changed and new assemblies are added to the core. This process is called a fuel shuffle. The fuel assembly loading and reloading procedure proceeds in a pattern that minimizes the risk of interaction between assemblies. In some cases, the entire core is offloaded, then replaced (with fresh and partially used assemblies).

The fuel handling sequence proceeds as follows:

The manipulator crane is positioned over a fuel assembly in the core. The fuel assembly is lifted by the manipulator crane to a predetermined height sufficient to clear the reactor vessel and still leave sufficient water covering to eliminate any radiation hazard to the operating personnel.

If the removed fuel assembly contains a rod cluster control assembly (RCCA), the fuel assembly is placed in a special fixture where the rod cluster control assembly is removed from the spent fuel assembly and placed into a new or partially used fuel assembly also located in the change fixture.

The fuel transfer (conveyor) car is moved into position in the transfer canal in the containment near the upender, which is used to switch the fuel assembly between horizontal and vertical orientations. The fuel assembly container section of the transfer car is pivoted to the vertical position by the upender, and the manipulator crane is moved to line up the fuel assembly with the fuel transfer system. The manipulator crane loads a fuel assembly into the container section of the conveyor car. The container is then pivoted to the horizontal position by the upender, and the fuel assembly is moved through the fuel transfer tube to the spent fuel pool. There, the fuel assembly container is raised to the vertical position by the upender, the fuel assembly is unloaded by the spent fuel pool bridge crane, and a new fuel assembly is brought from dry storage, and loaded into the fuel assembly container of the transfer car. The fuel assembly container is pivoted to the horizontal position and the conveyor car is moved back into the containment, where the assembly is relocated in the core.

Plant activities during a refueling outage

A refueling outage represents a critical time period for the operating success of a nuclear reactor. In addition to replacing the fuel, it is the only time when large portions of the reactor and its supporting systems, including the secondary side, are accessible for equipment inspection, maintenance, and replacement. Examples of typical activities are:

Pump and valve inspections and repair.

Metallurgical inspections of various components of the pressure boundary such as the reactor vessel and the steam generator tubes.

Containment inspections.

On the other hand, every day not producing power is a significant cost impact. Consequently, the refueling outage is generally scheduled for times when electricity demand is lower (in the United States, this is in spring or fall). The outage is packed with hundreds of activities performed around the clock, all planned months in advance with tightly controlled schedules. The workforce during a refueling outage typically increases from a few hundred to potentially several thousand workers. The resident inspector from the US Nuclear Regulatory Commission is also at the reactor site observing the activities. Typical well-planned and executed refueling outages are accomplished within 1 to 2 months.

Reactor re-assembly and startup

Reactor assembly, following the refueling outage, is essentially achieved by reversing the operations described in the beginning of this section. Startup then proceeds as described at the beginning of this section.

BWR operation

Several components of the plant play a role in reactor startup and normal operation and are listed here in summary form. Details on these components are provided in [Hamon, 2021](#).

Reactor coolant system (RCS): this system consists of the reactor vessel containing the core and steam separators, pipes connecting the reactor vessel to the steam turbines, and the recirculation loop and associated pumps.

Condensate and feedwater system: The condensate and feedwater system collects condensed steam, purifies, preheats, and pumps water from the main condenser to the reactor vessel.

Residual heat removal system (RHR): This system's main function is to provide water to the core during a loss of coolant accident. In normal operation it is secured, but ready for use in case of emergency. During refueling operations, it also serves to remove residual heat and maintain sufficient flow through the core.

Reactor water cleanup system (RWCU): The reactor water cleanup system maintains reactor water quality by filtration and ion exchange and provides a path for removal of reactor coolant when required.

Control rod drive system (CRD): positions control rods within the reactor core to change reactor power or to rapidly shutdown the reactor. These positions are referred to as notches, since there are mechanical stops that lock the CRDs into position. Typically, a notch is a discrete movement of 6 in. (15.2 cm) for the Generation II BWRs. The control rod is held at a given position by a collet engaging one of several notches in an index tube. The control rod is moved by applying pressure (via an accumulator) greater than reactor pressure to either the top or bottom of a main drive piston. Control rods are designated in groups. Only a selected number of groups are chosen to control reactor power during normal operation. These groups are banked into or out of the core as needed. The remaining groups of control rods are used to help shut down the reactor. At a reactor trip signal, all groups are driven into the core. Loss of electrical power to the system or reduction of accumulator pressure results in opening of scram pilot valves, thereby causing the rapid shutdown of the reactor.

Nuclear instrumentation: BWRs have various in-core nuclear instrumentation to measure neutron flux. The neutron flux (neutrons/cm²/s) generated in the core, from shutdown conditions to 100% power, ranges from a few neutrons/cm²/s to over 10¹⁵ neutrons/cm²/s. This large range generally requires three types of detectors with overlapping ranges. The three different types of ranges are designated source, intermediate, and power. Each detector is specifically designed to accurately measure neutron flux in its designated range. The source range detectors operate when the reactor is critical but is producing essentially zero power. It is typically a proportional counter that detects ionization events (counts per minute) directly and as such, is withdrawn when power is starting to be produced. The intermediate range detectors, active up to ~25% power, are typically ionization chambers, where the neutron flux induces a current in the detector circuitry. These are also withdrawn before full power operations begin.

The local power range monitor (LPRM) is a set of fixed in-core proportional detectors to measure local neutron flux at various radial and four axial in-core locations. The instrumentation tubes, in which the LPRMs are housed, are long sheaths, permanently fixed from the bottom of the reactor vessel into the region between selected fuel assemblies. The LPRMs are fixed in these instrumentation tubes along with a pathway for a movable detector. The LPRMs are periodically calibrated using the traversing in-core probe system (TIP), which is sensitive either to the neutron flux or gamma flux. This system is capable of finely spaced axial flux measurements and the shape of the flux is used to calibrate the LPRMs within an axial instrument string, and also across the radially spaced instrument strings.

During normal operation, the LPRMs are used to monitor reactor thermal power and the overall core power distribution. Signals are used by the average power range monitoring (APRM) system and rod block monitors. In addition, detailed assembly power distributions are determined, both by TIPs and by LPRMs.

Reactor startup

A BWR follows basically the same "countdown" procedure as a PWR, with some specific differences outlined in [Table 2](#). In particular, reactor heatup and pressurization are achieved primarily by nuclear heating, as opposed to heating only by reactor coolant pumps. Therefore, during much of the startup evolution, the reactor is not in "shutdown" mode in the sense that it is subcritical.

Table 2 Typical BWR modes for startup.

Mode	Title	Reactor mode switch position	Average reactor coolant temperature (F)
1	Power Operation	RUN	NA
2	Startup	Refuel or startup/hot standby	NA
3	Hot Shutdown	Shutdown	> 200 °F (93°C)
4	Cold Shutdown	Shutdown	≤ 200 °F (93°C)
5	Refueling	Shutdown or refuel	NA

The startup evolution begins from the point when the core has been refueled, and the reactor vessel has been resealed. The vessel has been partially filled with water (typically to just below the steam line). The recirculation pumps are off, and the RHR pumps are aligned to remove core decay heat.

Heatup and power ascension

The reactor mode switch is placed in STARTUP, source range (SRM) and intermediate range (IRM) monitors are inserted, and control rods are pulled in a predefined sequence. As rods are pulled, source range counts and reactor period are monitored. As criticality is approached, it takes longer and longer to return to steady state after notching a rod. Eventually the reactor period will stay on scale between 50 and 300 s with no rod motion and counts will be continuously increasing. This is the point of criticality.

After the reactor is critical, a positive reactor period is established to raise power to the heating range. The power increase continues until heating power is reached. At this point control rod withdrawal is governed by the heatup rate, usually about 90 °F (50 °C) per hour. This heatup rate is a procedural limit. The Technical Specification limit is 100 °F (55.5 °C) per hour. Control rod withdrawal is constrained to an approved sequence such that control rod positions remain symmetrical and the reactivity worth of a given rod is minimized.

Shortly after, the point of adding heat is reached. This is when the nuclear heat from the fuel is greater than ambient losses from the system. The fuel warms up, the moderator temperature increases, and reactor period and power stabilize.

Once the point of adding heat is reached, rods will continue to be withdrawn to achieve heating power. Once boiling starts, the reactor head vent is closed, the main steam header equalizing valve and MSIVs are opened, the steam plant begins warming. Around 230 °F (110 °C) the steam drains are opened to de-aerate the reactor. Power ascension continues to produce sufficient steam to warm other components including the turbine.

As reactor water temperature increases, the coolant expands, causing the water level in the reactor vessel to increase. In order to maintain a constant level during heatup, water must be removed from the reactor vessel. The reactor water cleanup system (RWCU) is aligned to reject water to the main condenser or to the Liquid Radwaste system. It is usually more desirable to reject the excess water to the condenser in order to minimize the water processing load on the Liquid Radwaste system. This evolution continues until rated temperature and pressure are reached.

As temperature increases above 212 °F (100 °C) the reactor vessel begins to pressurize. At about 100 psig (0.68 MPa), the reactor core isolation cooling (RCIC) system and the high pressure coolant injection (HPCI) system, low steam pressure isolation is reset, and the steam lines to auxiliary steam loads such as reactor feedwater pump (RFP) turbines and steam jet air ejectors (SJAEs) are warmed. As pressure approaches 150 psig (1.14 MPa), the system pressure regulator begins to open bypass valves. To avoid bypassing steam to the main condenser and to allow plant pressurization, the pressure setpoint is increased to 920 psig (6.45 MPa). At about 350 psig (2.41 MPa), an RFP is started and placed in service. At about 450 psig (3.21 MPa), the SJAEs are placed in service to maintain a vacuum in the main condenser. Also at this time, the main turbine is reset and prepared for starting. As pressure reaches 920 psig (6.45 MPa), the bypass valves begin to open to control reactor pressure at the EHC system pressure setpoint. Control rods are withdrawn to increase power, which results in further opening of the bypass valves.

When power reaches 5–12%, the reactor mode switch is shifted to the “RUN” position (Mode 1), and the IRMs are fully withdrawn. Control rods continue to be withdrawn until 2–3 bypass valves are open, representing sufficient steam flow to roll the main turbine and provide minimum loading following synchronization to the grid. At about this time, the feedwater control system is placed in automatic.

Control rod withdrawal is used to increase power to approximately 60%. When approximately 60% power is reached, the control rods have been withdrawn to what is called a 100% or “target” rod pattern. This rod pattern, in turn, yields 100% reactor power when recirculation flow is increased to 100%. The power increase from 60% to 100% is accomplished by increasing recirculation flow, which increases moderation, subsequently increasing the reactor power. The recirculation flow control system is aligned with the individual loop controllers in master manual control. Recirculation flow is then increased to near 100% with the master flow controller.

After the generator is synchronized to the transmission grid and producing a substantial output, reactor power output is changed to meet the grid system requirements by adjustment of control rod position, manual adjustment of reactor recirculation flow, or a combination of these two methods. Withdrawing a control rod reduces neutron absorption and adds positive reactivity. Reactor power then increases until the increased steam formation just balances the change in reactivity caused by the rod withdrawal. The increase in boiling rate tends to raise reactor pressure, causing the EHC system pressure regulator to open the turbine control valves sufficiently to maintain a programmed throttle pressure. When a control rod is inserted, the reverse effect occurs.

Full power operations

Operating strategies

Baseload operation

As with the PWR, constant, full power operation has been the desired operational mode for BWRs. Normal power operation, including adjustment to rapid load changes, is controlled by manual or automatic adjustment of reactor recirculation flow. Also, manual adjustments of control rods are used to shape core power distribution and can be used to change reactor power levels due to daily load requirements.

Load follow

Reactor power output can be varied over a power range of approximately 25% of rated power by adjustment of reactor recirculation flow, which maintains a near uniform power distribution. Reactor power change is accomplished by using the negative void coefficient of reactivity. In practice, power adjustments of 15% or so are managed by core flow alone. An increase in recirculation flow temporarily reduces the volume of steam in the core by reducing the steam void fraction. This addition of reactivity to the core causes reactor power level to increase. The increased steam generation rate then returns the steam volume in the core to approximately its original value, and a new constant power level is established. When recirculation flow is reduced, reactor power is reduced in a similar manner. During initial power operation, the operating curve or power/flow map is established relating reactor power to recirculation flow. The first point of the curve is full flow and rated power. When a rod pattern is established for this point, recirculation flow is reduced in steps, at that rod pattern, and the relationship of flow to power is plotted for steady state conditions. Other curves are established at lower power ratings and other rod patterns as desired.

Flexible operation

Like for PWRs, flexible operation is essentially load follow, except that required response times are generally much shorter, and the times during which the reactor is at lower power can vary from a few hours to weeks. Flexible operation refers to operation responding to grid conditions that cannot be easily planned in advance. Such operation places additional stress on the fuel rods, unless operational constraints are observed, as discussed below.

To reliably operate the plant in this non-baseload condition requires a combination of fuel management strategies and power maneuvering guidelines, which have proven effective at significantly reducing the risk of pellet-cladding interaction (PCI) that may induce cladding failure during flexible operation (see also [Young, 2021](#)). Steps that are taken to prepare for such operation include managing the power levels on the fuel rods such that the stresses from the pellet on the cladding are given adequate time to relax. Guidelines for these preconditioning power levels and rates of power increase vary with the fuel type and fuel vendor. Following periods of extended low power operation, these guidelines assure that proper reconditioning to relax the stress in the fuel rod is obtained during the ascension to power. These guidelines are informed by both analytical models and by data from test reactors ([Young, 2021](#)). These methods also show that design features, such as barrier liner of the cladding or small amounts of material doping within the fuel pellet, can improve resistance to PCI, and allow for slightly more aggressive ramp rate limits ([Young, 2021](#)).

Economics of load follow operation in BWRs

The economics of load following in BWRs are similar to those in a PWR. Minimizing operating costs and fuel costs while maximizing annual power output are key to reducing the cost of nuclear electricity production for a specific reactor.

Surveillance

Normal operation

During normal operation, constant surveillance of the reactor power and its distribution is maintained (for additional details see Section 2.1). In addition, coolant samples are taken at regular intervals, and tested for the presence of various elements and radioisotopes. Generally these measured quantities fall into four categories and indications:

- (a) The purity of the circulating coolant is confirmed by monitoring coolant conductivity and pH.
- (b) Chloride levels are monitored to assure the mitigation of stress corrosion cracking of structural materials.
- (c) Oxygen, iron particulates, and silica are also monitored.
- (d) Fission products such as Iodine and Xenon are monitored. Large changes indicate the presence of leaking fuel rods.

Additional reactor monitoring during normal operation include:

- (e) Control rod drive testing: control rods are moved down and up a few steps, sufficient to confirm the control rod drive mechanisms are operational, without affecting the core power distribution significantly.
- (f) Calorimetric tests to confirm reactor power.

Reactor trips and restarts

The plant is designed to accommodate abnormal shutdowns such as loss of generator load or turbine trip. Depending on the operating condition of the plant and size of the load rejection, a portion of the steam may be bypassed to the condenser. When the

bypass system is not capable of accommodating the excess steam generation, the reactor will automatically scram and the reactor isolation cooling and HPCS systems will be actuated, if required to maintain the vessel water level. Any of the foregoing shutdown modes of operation may be established after the emergency shutdown.

The normal plant operating features that delay achieving power following a scram are:

- Control rod withdrawal rates
- Cold plant heatup rates
- Turbine loading rates
- Xenon considerations

The exact recovery time, dictated by rod removal rates, is a function of elapsed fuel cycle time. At the beginning of the cycle, a shorter time could be realized, but toward the end of the fuel life, more control rods must be removed to reach power and a longer time is required.

Constraints are imposed on the rate of power increase to ensure that prior to a fuel failure threshold being reached (a PCI limit), the induced stress is relieved, or that the release of corrosive fission products from the fuel is sufficiently slow to prevent reactive chemical species traveling from the release site in the fuel to the cladding, and chemically attacking the inner-surface, causing embrittlement. The impact of these constraints on load follow operation is discussed further in (IAEA, 2018). The physical process in the fuel rod during power changes is described further in (Young, 2021).

The fuel rod local power density has threshold values, generally as a function of burnup and residence time in the reactor, below which rates of power change are not constrained. These are implemented generally as either absolute power levels (and rates of change of power), or in terms of a calculated stress level. Lower values apply to fuel that has experienced sustained control rod insertion or sustained operation at low power; this is termed the conditioned power level of the fuel. Above these thresholds, power ramp rates (and step changes in power) are constrained.

If the reactor systems are to be cooled down, a heating rate may be the limiting factor based on temperature differences existing in particular regions of the pressure vessel. The turbine loading rates are a function of the turbine temperature.

Reactor shutdown for refueling

A plant shutdown is accomplished by essentially reversing the steps described for startup and power operation.

Preparation for plant shutdown involves testing of systems required to maintain cold shutdown and other equipment checks such as the turbine oil systems, flushing of the RHR system (shutdown cooling piping).

Reactor power is reduced from 100% to approximately 60% by using the recirculation flow control system. This requires adjusting the master flow controller to reduce recirculation pump speed to minimum. Control rods are now inserted in reverse of the withdrawal sequence to obtain a reactor power of approximately 10%. At this time, the IRM detectors are inserted, and the reactor mode switch is transferred to the startup position. Generator load is reduced to a minimum and the turbine generator is separated from the transmission system. Turbine steam flow is picked up automatically by the bypass valves. Control rod insertion continues until all rods reach the full in position. Reactor cooldown is commenced by withdrawing steam from the reactor via the bypass valves and maintaining vessel inventory using a reactor feed pump. A cooldown rate of 90 °F (32 °C)/hr. is established. When reactor pressure has been reduced to approximately 50 psig (0.35 MPa), the shutdown cooling mode of the RHR system is initiated and is used to cool the nuclear system down to a temperature of 150 °F (66 °C). The reactor shutdown is essentially complete. Some plant shutdowns may require flooding of the reactor vessel. This is normally done using a condensate pump and would be necessary if the plant was going into refueling operations.

Reactor disassembly and refueling operations

In order to refuel, the reactor must be disassembled. After reactor shutdown and during cooldown, the reactor pressure vessel is vented and filled to above the flange level to promote cooling. The drywell head cavity is drained and drywell head bolts removed during this time in preparation for head removal.

The drywell head typically utilizes swing bolts, which are loosened to clear the head and swing outward to lean on the reactor well wall. The unbolted drywell head is lifted by the polar building crane to its storage place on the refuel floor. The drywell seal surface protector is installed before any other activity proceeds in the reactor well area.

With the drywell head removed, an array of piping is exposed that must be serviced. Various vent piping penetrations through the reactor well must be removed and the penetrations made water tight. Vessel head piping is stored on the head insulation and a common lift is used to transport them to storage on the refueling floor.

Water level in the vessel is now lowered to the flange level in preparation for reactor vessel head removal. Using a strongback and the reactor building crane, the reactor vessel head is removed to the head holding pedestals on the refueling floor, keeping the vessel head elevated in order to facilitate inspection and O-ring replacement. If necessary, the vessel studs in line with the transfer canal are removed to provide a path for fuel movement.

The dryer is lifted via strongback and reactor building crane to its storage location. Prior to removal of the steam separators, the steam line plugs are installed in the main steam line nozzles from inside the vessel. With these plugs in place, servicing of the safety/relief valves or main steam isolation valves can be accomplished without interference or holdup due to refueling activities.

The separator is unbolted and unlatched from the shroud. With strongback and the reactor building crane, the separator is lifted and transferred underwater to its storage location in the pool.

The gate isolating the fuel transfer area is removed, connecting the reactor well and the fuel transfer area. The refueling platform is the principal device to handle and transfer the fuel assemblies in and out of the reactor. To move fuel, the grapple is aligned over the fuel assembly, lowered and attached to the fuel bundle bail handle. The fuel bundle is raised out of the core and moved through the refueling slot, positioned over the storage rack and lowered into the rack. Fuel is shuffled to respective new locations for the coming fuel cycle, and then new fuel is moved from the storage racks to the reactor vessel in the same manner.

Plant activities during the refueling outage

As in the PWR, a refueling outage at a BWR represents a critical time period for performing equipment inspection, maintenance, and replacement. A similar level of activity over a short period of time takes place at a BWR.

Reactor re-assembly and startup

Reactor assembly, following the refueling outage, is essentially achieved by reversing the operations described in the beginning of this section. Startup then proceeds as described at the beginning of this section.

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Operational Analysis Methods

William Walters, Ken and Mary Alice Lindquist Department of Nuclear Engineering, Pennsylvania State University, State College, PA, United States

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Glossary

Burnup The amount of energy generated per unit mass of fuel. Usually expressed as megawatt-days per metric ton of uranium. The amount of fission products in the fuel is directly related to the burnup of the fuel.

Control element A component of a reactor that is controlled by the operator to affect the reactivity of the core. A common control element is a control rod that contains a neutron absorbing material such as boron that can be inserted or retracted from the core.

Cross section A material property that measures the probability of an interaction, such as neutron-induced fission. Reaction rates, such as the fission rate, are proportional to the neutron flux and the fission cross section of the material.

Crud A colloquial term for corrosion and wear products (rust particles, etc.) that become radioactive (i.e., activated) when exposed to radiation ([Nuclear Regulatory Commission, 2020](#)).

Deterministic method A method to non-randomly calculate the neutron flux and reaction rates by numerically solving the neutron transport equation. Contrast with the Monte Carlo method.

Fissile A nuclide is considered fissile if a neutron with zero kinetic energy can induce fission in the nucleus.

Fission product A nuclide produced when another nucleus breaks apart during nuclear fission. Some examples are Strontium-90, Cesium-137, Xenon-135, and Iodine-131.

Fissionable A nuclide is considered fissile if fission can be induced by a neutron with high kinetic energy (a fast neutron).

Fluence Total number of neutrons that have crossed a unit area; equal to the integral of the flux over time. Usually used in the context of material damage. Units of neutron flux are neutrons per square centimeter.

Flux spectrum The energy dependence of the neutron flux.

Flux The number of neutrons crossing a unit area per second. Reaction rates, such as the fission rate, are proportional to the neutron flux. Units of neutron flux are neutrons per square centimeter per second. The angular flux includes the dependence on neutron direction, and the units include solid angle in the denominator.

Moderator A material used to slow down the fast neutrons that are born from fission to a lower energy where there is a higher probability of causing fission.

Monte Carlo method A stochastic method to calculate neutron flux and reaction rates based on random sampling of known probability distributions of physical processes.

Neutronics The study of neutron transport within a reactor.

Nuclear resonance A large peak in a nuclear cross section at a particular incident neutron energy.

Nuclide A species of nucleus with a particular number of protons and neutrons (e.g., a Uranium-235 nucleus has 92 protons and 143 neutrons).

Reactivity worth The change in reactivity due to a specified change in the core (e.g., reactivity worth of a control element).

Reactivity Typically denoted by the symbol ρ , the reactivity represents the extent of the deviation in the core effective multiplication factor (usually denoted as k_{eff}) from its value at a just critical state (at which state $k_{\text{eff}} = 1.0$ and the fission rate is constant). A positive (negative) reactivity feedback implies that the ratio of the rate of neutron production to the rate of neutron losses (k_{eff}) is larger (smaller) than 1.0 due to a change in a performance characteristic such as temperature or density.

Shutdown margin The amount of reactivity by which the reactor is subcritical when all control elements have been inserted. The shutdown margin usually assumes that the control element with the highest worth remains fully withdrawn.

Subchannel The region of coolant in between fuel elements. This is usually a repeatable unit and forms the basis of most thermal hydraulic analysis.

Thermal feedback The impact of temperature change on reactivity, through mechanisms such as Doppler broadening of nuclear resonances in the fuel or changes in moderator density.

Thermal hydraulics The study of heat transfer, fluid mechanics, and thermodynamics needed to determine the temperature inside a nuclear reactor.

Validation The process of ensuring that a computer simulation or analysis method produces a result that is consistent with physical experiments.

Introduction

Other chapters (Hamon, 2021; Young, 2021a) discussed some of the general design and operations considerations for safe and reliable operations of nuclear reactors. This section will discuss some of the mathematical and computational methods that are used for the quantitative analysis of nuclear reactor core design and operations.

From a basic perspective, reactor design has two objectives. Number one is to always ensure the safety of the reactor under any condition, and number two, always behind safety considerations, is to produce power in an economical fashion.

To ensure safety, there are a few main considerations:

- The fuel and cladding temperatures need to stay within the design limits such that they do not melt or fail in any way.
- The reactor should be stable and have inherent passive negative thermal feedback (moderator temperature coefficient, fuel temperature coefficient, core expansion, etc.).
- Active control elements should be able to shut down the reactor under any conditions (i.e., have sufficient shutdown margin), and failure or loss of a control mechanism should not be harmful (e.g., rod ejection reactivity worth).

From an economical point of view, the main factors are:

- The reactor should operate at the maximum possible power for the longest amount of time (power $\times$ time = money).
- The reactor should be as neutronically efficient as possible to use least amount of input fuel as possible to save money on refueling costs.

Very often, these objectives may conflict with each other. For example, the most neutronically efficient core is one with most of the power in the center, and thus a low leakage of neutrons from the core. This, however, will have a very high local power in the center (peaking factor), which would raise the fuel temperature there to unacceptable levels.

Multi-physics computational methods: General considerations for multi-physics coupling

The core design and analysis is fundamentally a multi-physics problem, generally consisting of:

- Neutronics methods to calculate power distributions and reactivity within the core, and neutron fluence that may dictate long-term material damage in the pressure vessel, for example.
- Thermal hydraulics methods to calculate heat transfer and temperature distributions within the fuel and coolant.
- Materials, thermo-mechanical, and structural analysis methods to calculate the effect of temperature distributions, coolant induced vibrations and erosion etc. on the physical materials in the system.

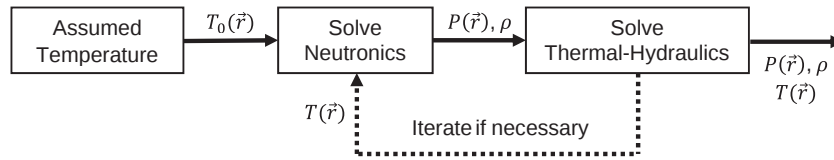


Fig. 1 The process for the standard method to solve the coupled neutronics/thermal-hydraulics problem.

While the thermo-mechanical aspect can sometimes be ignored, or at least not included in all analyses, there is a fundamental and strong feedback between neutronics and thermal-hydraulics that must always be considered. The temperature of the system (fuel, coolant/moderator, etc.) affects the reactivity, which in turn affects the power level, which then affects the temperature again.

In theory, it is possible to create a set of equations governing the neutronics and thermal hydraulics and solve them as a single coupled system, such as in the MOOSE code system (Gaston et al., 2009). In practice, for large systems, this is a challenging problem, and not usually necessary. The standard approach is to solve each problem independently, and couple to each other. This process is outlined in Fig. 1. A temperature distribution, $T_0(\vec{r})$, is assumed. The neutronics problem is solved to obtain a power distribution, $P(\vec{r})$, and reactivity, ρ . The power distribution is input to a thermal-hydraulics solver, which will solve for the temperature distribution. If necessary, the temperature can be fed back into the neutronics solver until converged.

Neutronics computational methods

The full-scale direct solution of the stand-alone, steady-state neutron transport problem (Tsvetkov, 2021) is extremely challenging, even disregarding thermal feedback. In order to balance the accuracy needed for safe and economical operation with the speed required for everyday operational and design use, the industry has generally settled on the “two-step” computational approach for core neutronics:

1. Detailed (in space, energy, and angle) neutron transport solution for a set of simplified 2D problems (e.g., pin cell or single fuel-assembly), and generation of homogenized group constants.
2. Use of the homogenized group constants in a coarse spatial mesh 3D core model, using relatively few energy groups and simplified angular dependence (usually the diffusion approximation).

The proper generation of group constants using the full neutron transport solution generally ensures good accuracy for the quantities of interest (eigenvalue, reaction rates), while the coarse 3D solution can be obtained fairly quickly (usually less than 1 s).

Nuclear Data

The fundamental nuclear data required for reactor calculations include:

- Point-wise (continuous energy) neutron cross-sections for all reactions of interest.
- Resolved and unresolved nuclear resonance parameters for important nuclides.
- Fission product yield data for all fissile and fissionable nuclides.
- Decay constants (half-lives, decay products, branching ratios) of fission and transmutation products.

There are several sources of this data available, which include:

- US-developed Evaluated Nuclear Data File Version B (ENDF/B) (Brown et al., 2018).
- European Joint Evaluated Fission and Fusion Library (JEFF) (Koning et al., 2011).
- Japanese Evaluated Nuclear Data File (JENDL) (Shibata et al., 2011).
- Russian Library of Evaluated Neutron Reaction Data (BROND) (Blokhin et al., 2016).
- Chinese Evaluated Nuclear Data Library (CENDL) (Ge et al., 2017).
- TALYS-based evaluated nuclear data library (TENDL) (Koning et al., 2019).

Additional information on the selection and use of nuclear data sets can be found in other documentation such as ANSI/ANS 19.1-2019 (American Nuclear Society, 2019).

Multi-group (application-dependent) cross sections

The most important nuclear data is the nuclear macroscopic cross section $\Sigma(E)$, which represents the probability per unit length of particle travel that a nuclear reaction will occur. The macroscopic cross section is calculated as the microscopic cross section $\sigma(E)$

times the number density of a given nuclide. The neutron flux and neutron cross section can be used to calculate the reaction rate density:

$$R \left[\frac{\text{reactions}}{\text{cm}^3 \text{ s}} \right] = \int_0^\infty dE \Sigma(E) \phi(E)$$

In a nuclear reactor, neutron energies can vary from around 0.01 eV to 10 MeV. Cross-sections have a strong dependence on energy, and can vary by many orders of magnitude at local nuclear resonances. For a more thorough discussion of neutron flux, cross-sections, and reaction rates, the reader is directed to [Tsvetkov \(2021\)](#).

Most neutron transport solvers, with the exception of Monte Carlo-based solvers, cannot use continuous energy cross-sections directly, and require the use of multi-group cross-sections. In general, a multi-group cross section for energy group g , Σ_g , can be written as a function of the continuous energy cross section $\Sigma(E)$, and a weighting flux spectrum $\phi(E)$:

$$\Sigma_g = \frac{\int_{E_g}^{E_{g+1}} dE \phi(E) \Sigma(E)}{\int_{E_g}^{E_{g+1}} dE \phi(E)}$$

Since the weighting flux spectrum should logically vary between reactor types, all multi-group cross section libraries are inherently application-dependent, and care must be taken in their use. In general, the more energy groups in the multi-group library, the less impact there is on the accuracy of the weighting flux. There are various methods for determining the weighting flux spectrum, which may assume homogenous or 1D geometry. There are many computer codes in common use to perform this multi-group cross-section generation. These include BONAMI ([Greene, 1984](#)) and CENTRM ([Williams et al., 2007](#)) for LWR applications and MC2-3 ([Lee and Yang, 2017](#)) for fast reactor applications.

The number of energy groups used in the multi-group cross section libraries varies significantly with the application. In general, fast spectrum reactor problems require many more than thermal reactor problems. Light water reactor libraries are generally in the 10s to 100s of energy groups, while liquid metal fast reactors may use 1000s to 100,000s of energy groups. This initial simplified-geometry cross-section generation process is usually followed by lattice physics calculations on a more representative geometry to further reduce the number of energy groups.

Lattice physics calculations

Given an application-dependent set of multi-group cross-sections, the goal of lattice physics calculations are to homogenize (i.e., remove the spatial dependence), and perform energy condensation (reduce the number of energy groups) to allow for accurate and fast full-core calculations. This is usually done by solving the transport equation for the multi-group flux on a single fuel assembly (or lattice) with reflective boundary conditions in 2D geometry. When the multi-group flux $\phi_g(\vec{r})$, has been solved for in the lattice, homogenization and group condensation is performed on the basis of preserving reaction rates in the homogenized cell (such as a fuel assembly as shown in [Fig. 2](#)).

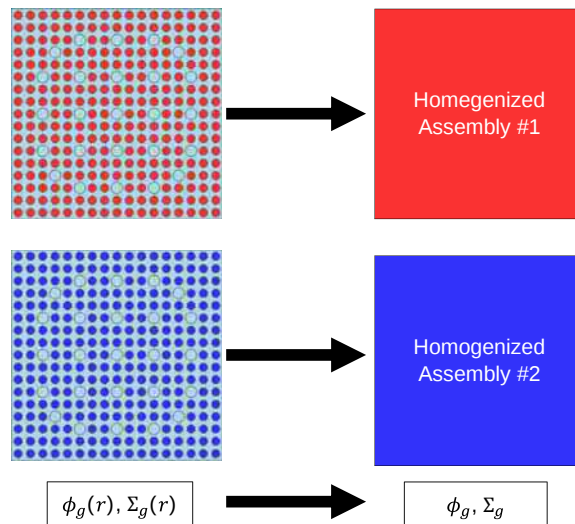


Fig. 2 Assembly lattice homogenization for two fuel assembly types.

The homogenization process to obtain the homogenized cross-sections $\bar{\Sigma}_g$, based on the preservation of reaction rates over the volume of the cell V_{cell} is given by:

$$V_{cell} \bar{\phi}_g \bar{\Sigma}_g = \int_V dV \phi_g(\vec{r}) \Sigma_g(\vec{r})$$

The cell averaged group g flux over volume V_{cell} is defined as:

$$\bar{\phi}_g = \frac{1}{V_{cell}} \int_V dV \phi_g(\vec{r})$$

Starting with G energy groups, it is also possible to reduce this to smaller set of groups H , called a broad-group or few-group cross-sections $\bar{\Sigma}_h$. This is process, called energy collapsing or condensation, is based on preservation of reaction rates as given by:

$$\bar{\phi}_h \bar{\Sigma}_h = \sum_{g \in h} \bar{\phi}_g \bar{\Sigma}_g$$

One challenge with the spatial homogenization process is that local properties such as the pin-wise power distribution are lost. It is possible, however, to recover this information in a process known as pin-power reconstruction (Koebeke and Wagner, 1977; Rempe et al., 1989). This is generally done by multiplying the flux calculated in the full-core homogenized medium by a pin-wise form function obtained during the lattice calculation, as shown below (Rempe et al., 1989).

$$\phi_g(xy) = \phi_g(xy)_{homogenous} \times \phi_g(xy)_{form\ function}$$

Pin power reconstruction is important because it is needed to determine the local power needed for thermal and safety analysis. An example of a pin power distribution in a typical pressurized water reactor (PWR) assembly is shown in Fig. 3, which was calculated based on the geometry in Fig. 2. The power is higher near the guide tubes where there is additional moderator present.

While the homogenization process aims to preserve the transport solution as much as possible, there are errors introduced in this approximation when applied to the full-core solution. For one, there must be an assumption made for the boundary conditions during the lattice calculation, which is generally considered reflective. This is, of course, not true in the real core, and the assembly may be greatly affected by nearby dissimilar assemblies. The reaction-rate preserving homogenization process also does not necessarily preserve the current at the assembly boundary. There are two general approaches to increase the accuracy of the homogenized result: Generalized Equivalence Theory (Smith, 1986) and the Super Homogenization (SPH) method (Hébert and Benoist, 1991). Homogenization is an important but challenging process, with many subtle details that may affect the result. Research is ongoing in the area to improve the accuracy (Smith, 2017).

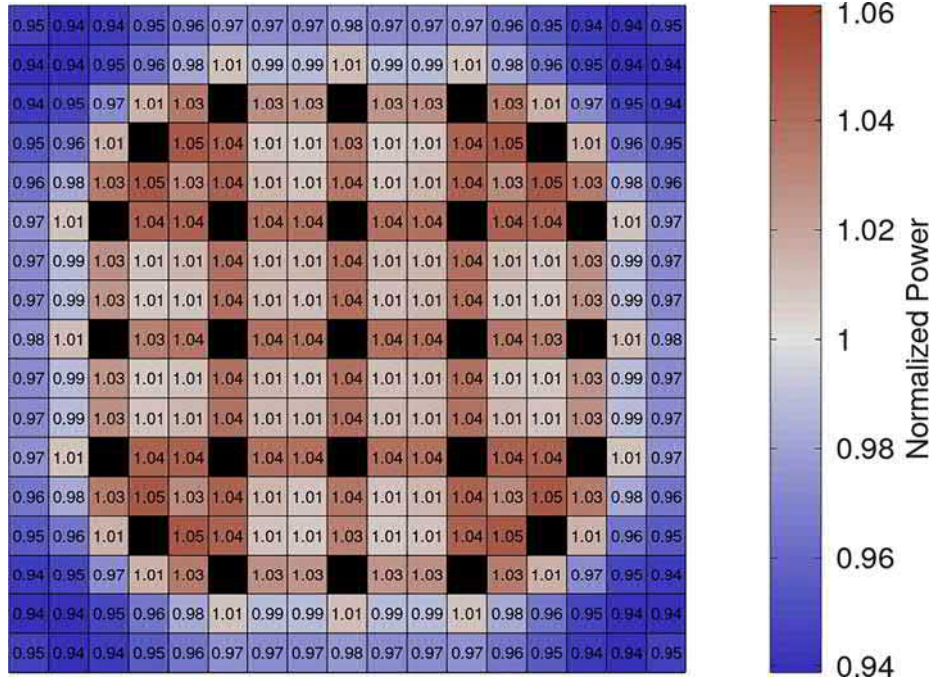


Fig. 3 Normalized pin-power distribution calculated for a sample pressurized water reactor lattice.

Lattice physics methods

The calculation of the flux in the fuel lattice can be accomplished using a variety of lattice physics methods. The neutron flux is generally a function of space, angle (or direction of travel), and energy (or velocity). Lattice physics methods must have high fidelity in all these variables in order to accurately calculate the neutron flux needed for homogenization.

The discrete ordinates (or Sn) method is a deterministic method based on the differential form of the neutron transport equation, and solves for the angular flux along a set of discrete directions (ordinates), on a spatial mesh. This method is not as popular for lattice physics problems due to the extremely small spatial mesh size that is usually required for accurate solutions. The NEWT code (DeHart and Bowman, 2011) in the SCALE package is an example of a discrete ordinates lattice physics code.

The Method of Characteristics (MOC), and Collision Probability (CP) deterministic methods are based on the integral form of the neutron transport equation. They attempt to solve the equation along a set of characteristic rays. These have the advantage that they can better handle the curved geometry of fuel rods using fewer variables. Examples of MOC codes include HELIOS (Wemple et al., 2008), POLARIS (Jessee et al., 2014), CASMO (Rhodes et al., 2006), and APOLLO (Sanchez et al., 2010). Some CP codes include PARAGON (Matsumoto et al., 2006) and DRAGON (Hébert, 2016).

Monte Carlo methods solve the transport problem by following randomly sampled neutrons statistically through a probabilistic representation of each component of the transport equation until the neutron's termination by absorption or leakage from the system. These techniques have gained popularity for few-group cross section generation in recent years, due to increased computational power, and the fact that they can use continuous energy cross sections directly, without the need for problem-dependent multi-group cross-sections. Monte Carlo methods also do not suffer from spatial and angular discretization errors, which can be a challenge with deterministic codes. The main disadvantage of Monte Carlo is that it is a stochastic method with inherent statistical uncertainty and can require longer computation time to reach an acceptable level of uncertainty. Monte Carlo codes currently in use for few-group cross section generation include Serpent (Leppänen et al., 2015), KENO (DeHart and Bowman, 2011), and OpenMC (Romano et al., 2015).

Depletion calculations

Along with generation of group constants and few-group cross-sections, the other task of lattice physics codes is to perform depletion calculations. This process requires the tracking of changes in number density for all nuclides, as well as the resulting changes in cross-sections. Many different reactions must be tracked that account for loss and production mechanisms for each nuclide.

Loss mechanisms for a given nuclide include:

- Any non-scattering (i.e., absorption) reaction: (n, γ) , (n, f) , (n, xn) , (n, p) , (n, α) , etc.
- Radioactive decay.

Production mechanisms include:

- Direct production from fission.
- Transmutation from another nuclide (usually n, γ).
- Radioactive decay from parent nuclide.

Putting these together, one generally arrives at a rate-of-change equation for every nuclide in the system resembling the one below:

$$\frac{dN_x}{dt} = \gamma_x N_f \sigma_f \phi + N_y \sigma_{\gamma} \phi + \lambda_p N_p - \sigma_{a,x} \phi N_x - \lambda_x N_x$$

Where,

- $\gamma_x N_f \sigma_f \phi$ is production of nuclide x from fission;
- $N_y \sigma_{\gamma} \phi$ is the transmutation of nuclide y into nuclide x;
- $\lambda_p N_p$ is the radioactive decay of nuclide p into nuclide x;
- $\sigma_{a,x} \phi N_x$ is the transmutation of nuclide x from neutron absorption;
- $\lambda_x N_x$ is the loss rate of nuclide x from radioactive decay.

There are a few challenges with this approach. First is the cross sections σ_f , σ_{γ} , and $\sigma_{a,x}$ which are one-group cross sections. Since they are only one energy group, they will vary significantly as the neutron spectrum changes throughout the operating cycle. This necessitates recalculating these cross sections fairly often with fuel burnup. Second is the sheer number of nuclides that must be simulated. Depending on the application, this is usually in the hundreds. The compositions vary spatially, in which case hundreds of nuclides must be tracked throughout hundreds (or millions) of different zones. Lattice physics codes automate this depletion process, and a variety of mathematical methods have been used to solve the basic depletion equations.

Few-group cross section parameterization

The depletion calculations are normally performed at nominal conditions (e.g., like fuel temperature, moderator temperature, boron concentration). At several points in the depletion calculations, branch calculations are usually performed at off-nominal

conditions in order to determine the dependency of the group constants on reactor conditions. Branch calculations are a set of calculations done at high and low values of certain parameters (such as fuel temperature, moderator density, boron concentration). It is then possible to estimate the cross-sections for any combination of these conditions at a given burnup. For example, the cross section representation in PARCS (Downar et al., 2002) for a PWR system is considered as follows:

$$\Sigma(\alpha, T_f, T_m, \rho_m, N_b) = \Sigma^r + \alpha \Delta \Sigma^{cr} + \frac{\partial \Sigma}{\partial \sqrt{T_f}} \Delta \sqrt{T_f} + \frac{\partial \Sigma}{\partial T_m} \Delta T_m + \frac{\partial \Sigma}{\partial \rho_m} \Delta \rho_m + \frac{\partial \Sigma}{\partial N_b} \Delta N_b + \frac{\partial^2 \Sigma}{\partial \rho_m^2} (\Delta \rho_m)^2$$

Where,

- Σ^r is the cross section at nominal conditions
- α and $\Delta \Sigma^{cr}$ are factors accounting for the control rods being present or not in the region
- ΔT_f is the difference in fuel temperature from nominal conditions
- ΔT_m is the difference in moderator temperature from nominal conditions
- $\Delta \rho_m$ is the difference in moderator density from nominal conditions
- ΔN_b is the difference in boron concentration from nominal conditions
- The derivatives such as $\frac{\partial \Sigma}{\partial T_m}$ are determined from the branch calculations (at high/low conditions away from nominal to calculate the derivative).

Additionally, other conditions may be considered for cross section parameterization, such as xenon concentration; or void history and control rod history, which both affect the spectral conditions and thus also the composition of the fuel at a given burnup level (Baturin and Vygovskii, 2001).

Core simulation

Given the few-group cross-sections, parameterized by all the factors required for the given reactor type, the most common core analysis method are deterministic nodal diffusion methods. Speed of calculation is extremely important for core analysis, because many cases must be run. Since the composition of the fuel is constantly changing over a cycle, the core analysis code has to be run over many time-steps to account for this. Additional runs are also required for the criticality search to determine the level of reactivity control required to keep the reactor critical (e.g., critical rod position, critical boron concentration, etc.).

Nodal codes use a coarse representation of the angular variable, which is generally based on diffusion theory or simplified P3 (SP3). Diffusion theory considers that the angular flux only has a linear dependence on angle, and can break down in areas of high flux gradients. The P3 considers a third order Legendre polynomial expansion of the angular flux in the transport equation, while the SP3 method ignores many of the terms for a much simpler and faster solution (Brantley and Larsen, 2000).

Spatially, the core is divided into segments called nodes; typical node size is a quarter-assembly per node in the radial direction, and 24 nodes axially, although smaller nodes are possible. There is also some effort to perform pin-wise nodal calculation to increase the pin-wise accuracy. Some examples of nodal core calculation codes include: ANC (Liu et al., 1985), SIMULATE (Rempe et al., 1989), ARES (Leppänen et al., 2014), NESTLE (Turinsky et al., 1994), PANACEA (Chiang et al., 1985), PANTHER (Hutt et al., 1991), PARCS (Downar et al., 2002), DONJON (Hébert, 2016), and REBUS/DIF3D (Collins et al., 2016). Most nodal codes can perform a single core calculation on a single CPU in fractions of a second.

Although nodal diffusion has been the workhorse of the nuclear industry for decades, there has also been some effort to develop higher-fidelity whole-core analysis codes based on transport theory. These codes generally use hundreds of energy groups and much higher spatial and angular resolution than nodal codes. Although this can increase the accuracy, especially for pin-resolved results, it comes at a very high computational cost. Deterministic whole-core transport solvers are typically run on thousands of CPUs, and may take minutes-hours-days of computation time. This makes it difficult to use for routine analysis and design, but can be used in special cases. Some examples of high-fidelity transport codes for full-core analysis are MPACT (Collins et al., 2016), nTRACER (Jung et al., 2013), and DeCART (Joo et al., 2004).

The Monte Carlo method is the most accurate method for reactor physics, since it does not require discretization in space, energy, or angle, unlike deterministic methods. However, it suffers from statistical uncertainty, especially if local results (like pin-power) are required. For whole-core analysis, this requires the simulation of billions or more particles. Although computational speeds have increased over the years, this is still a very challenging prospect, especially for routine calculations. Generally Monte Carlo methods are mostly used occasionally as “reference” solutions to make sure that the everyday deterministic methods are working well. Some of the most prominent Monte Carlo codes that have been used for core analysis include MCNP (Goorley et al., 2012), Serpent (Leppänen et al., 2015), KENO (DeHart and Bowman, 2011), Shift (Pandya et al., 2016).

Loading pattern development

Typically, most power reactors around the world operate on a 12, 18, or 24 month cycle. Some reactors, such as CANDU reactors (Nichita, 2021), the fuel is shuffled almost continuously. In molten-salt reactors, the fuel is homogenous, and there can be a near-continuous removal of spent fuel and addition of fresh material. Generally, however, at the end of every cycle, the reactor shuts down, some fraction of the fuel assemblies are removed (typically around 1/3), and a corresponding number of fresh assemblies

are added. Where in the core to place the new and old assemblies for the next cycle, as well as the properties of the incoming fresh fuel, is called a loading pattern. This pattern is of critical importance, as it will determine the safety and efficiency of the system. As with most things, the goals are to maximize fuel efficiency (at the economic concerns) while staying within important safety constraints. Specifically, one usually wants to:

- *Minimize neutron leakage* from the core to maximize neutron efficiency. This can be done by putting low-enriched or highly-burned fuel towards the radial edge of the core. Increased neutron efficiency will allow for either a longer cycle length before refueling, or require fewer fresh assemblies to be loaded. Reduced leakage also has the benefit of reducing the neutron fluence to the pressure vessel, which is the lifetime-limiting factor for many reactor designs.
- *Minimize power peaking* in the core to keep local power within safety limits. Since a reactor is usually operated to run at a given average power, any power peaking means a higher local power. Reactor safety is limited by the ability to keep the hottest part of the reactor cool, so a high peaking factor would require a lowering of the average power to keep the peak power below the limit. Limiting power peaking can be done by putting fresh and burned assemblies in a checkerboard pattern, and by putting some higher-reactivity assemblies near the core periphery.
- *Stay within all other safety constraints*, which may include reactor control shutdown margin, the worth of the highest worth control rod in the case of a rod-ejection accident, reactivity coefficients, etc.

Unfortunately, these objectives are usually conflicting in nature, and optimization is challenging. The most efficient type of core neutronics is low-leakage with high-reactivity in the middle and low-reactivity on the periphery. However, that is also the worst from a power peaking point of view, as all the power will be concentrated in the center of the core.

Several example core loading options are shown in Fig. 4 (Tsoulfanidis, 2013). The Out-In pattern is very simple: fresh fuel is loaded on the outside of the core, and subsequent batches are moved to towards the middle, and then removed. This results in power peaking towards the edge of the core, but still has relatively less peaking than a uniform core or one loaded with the fresh fuel in the middle. One modification to this is the checkerboard loading, yields very low power peaking. Both of these loading patterns, however, have a high amount of leakage which is bad for neutron economy and pressure vessel damage. One slight modification can be obtained with a low leakage core loading, in which low-reactivity fuel is placed on the periphery, fresh fuel just inside of that, and a mixture of fuel in the middle. The ring of fresh fuel helps to draw power to the outside to help with power peaking, while the low reactivity fuel at the periphery reduces the leakage. Radial power distributions for uniform, checkerboard, and low-leakage core loading patterns are shown in Fig. 5. The low leakage pattern increases power peaking slightly while significantly reducing neutron leakage.

The traditional method of core loading optimization is manual, relying on trial-and-error and engineering judgment. This can certainly be successful, but requires years of core design experience and can be difficult to teach. It is also prone to personal preference and biases, and the overall design space that is explored will likely be very small.

There are several automated algorithms that are available, which due to the automated nature can explore a much larger design space, and do not rely nearly as much on personal judgment and years of expertise. This area has been well studied due to the cost savings that are possible (Levine, 1981; Yamamoto, 1997). The two most popular methods currently are genetic algorithms and simulated annealing.

Genetic algorithms are based on concepts from biological evolution. A semi-random set of possible initial layouts (the population) is evaluated, and the “best” candidates are taken forward to the next generation. A new population is created by combining aspects of the chosen candidates (crossover). Random mutations are also added to ensure the optimization does not get stuck in local minima. These algorithms have seen some success in core design, as demonstrated by Haibach and Feltus (1997) and Yamamoto (1997).

Simulated annealing is another optimization method widely used across many disciplines for problems with large solution spaces (Kirkpatrick et al., 1983). The fundamentals of simulated annealing are based on the concept of annealing in metallurgy. The algorithm starts with an initial random loading pattern. A perturbation is made to the layout, and the new layout is randomly accepted based on a combination of the fitness of the layout, and a “temperature” parameter. A high temperature parameter increases the change to move to a “worse” layout. The temperature starts out high, allowing many “worse” permutations, but avoiding getting stuck in local minima. As the algorithm progresses, the temperature is lowered (as in metallurgical annealing), and the

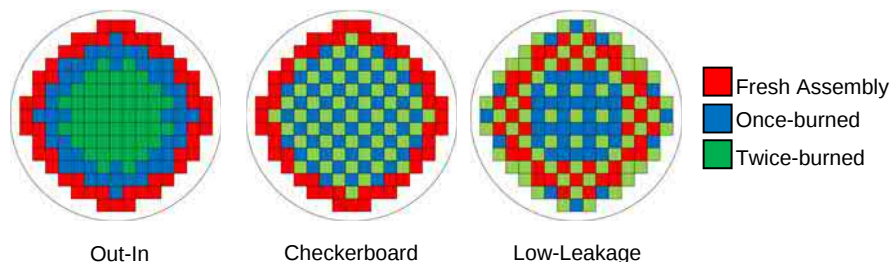


Fig. 4 Example loading pattern options for a 3-batch PWR. Adapted from Tsoulfanidis N (2013) *The Nuclear Fuel Cycle*. American Nuclear Society.

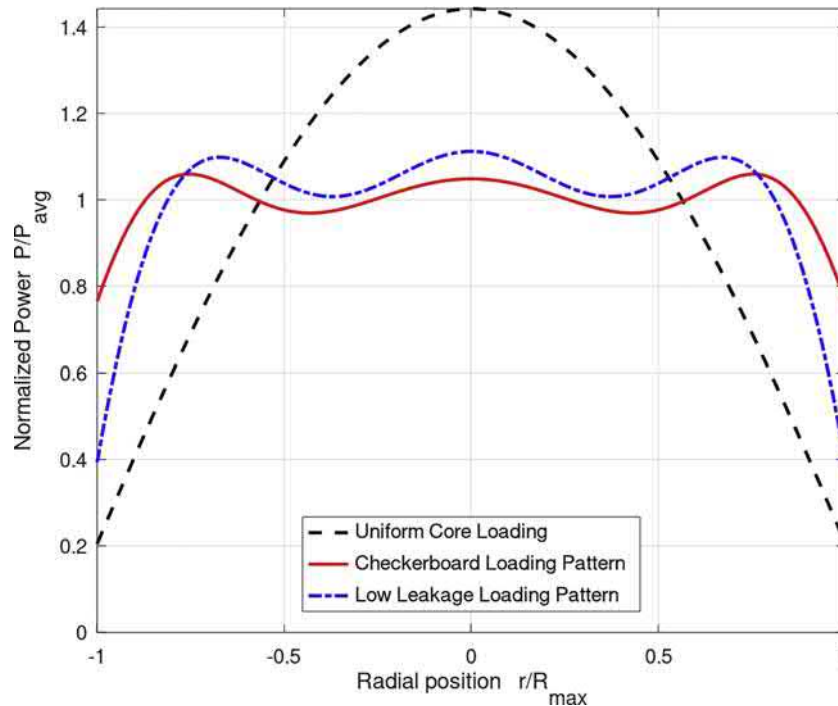


Fig. 5 Radial power distributions for uniform, checkerboard, and low-leakage core loading patterns.

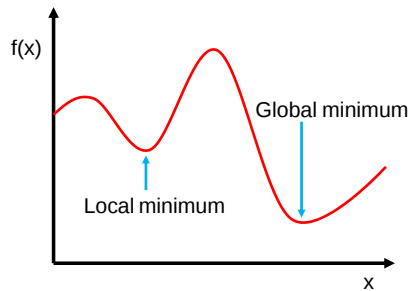


Fig. 6 An illustration of the challenge of global optimization of a function in 1D.

algorithm hopefully settles onto a good solution. This method has been successfully applied by Westinghouse with their Shuffleworks code, the Studsvik code SIMAN (Stevens et al., 1995), and others (Kropaczek and Turinsky, 1991).

A key aspect of both of these methods, and in optimization of large complex problems, is avoiding solutions that appear to be locally optimal. That is, any *small* change will result in a worse system. However, the system may be better if some large changes are made. A simple example for a 1D optimization problem, finding the minimum of a function $f(x)$, is shown in Fig. 6. The local minimum on the left can look like a good solution to a simple optimization method (e.g., gradient descent), while missing the global minimum on the right. Simulated annealing avoids this by allowing large, unfavorable changes at the beginning. Genetic algorithms avoid this with both the mutation and crossover features.

Thermo-hydraulic computational methods

As discussed earlier, there is a strong connection feedback between neutronics and temperature distributions within the reactor. The above methods only solved the neutronics side, and generally thermal-hydraulics calculations are done to solve for the temperature distribution in the system for a given power distribution.

These methods must solve for:

- The mostly 1D radial heat transfer from the fuel, cladding, and fuel-cladding gap;
- Heat transfer into the coolant with turbulent flow and possibly various boiling regimes;
- Coolant flow properties (3D or approximated as 1D), and pressure drop through the core and secondary systems.

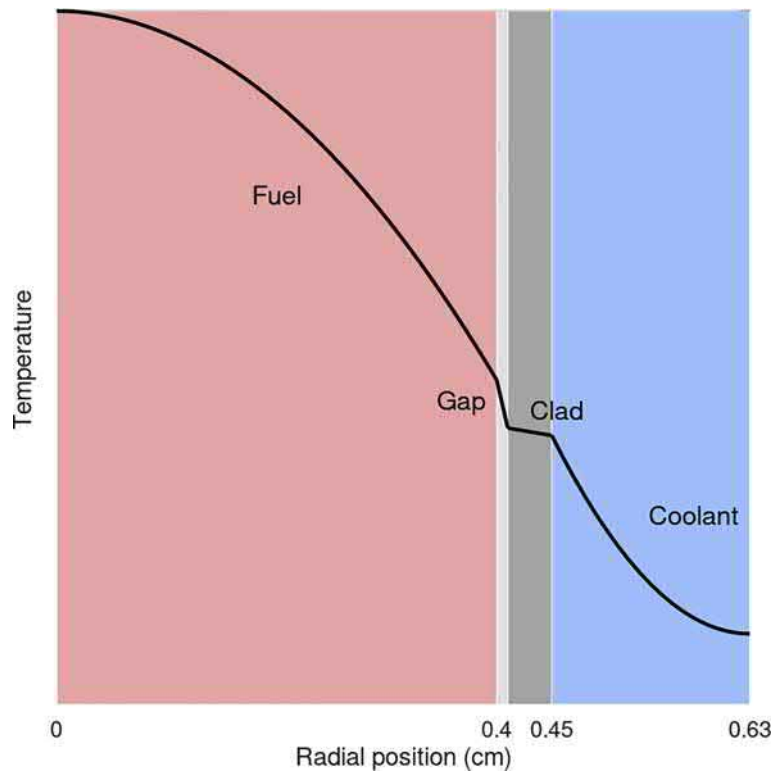


Fig. 7 Radial temperature profile in a nuclear fuel rod.

A typical temperature profile for 1D radial heat transfer in a fuel rod is shown in Fig. 7. This is characterized by a quadratic profile in the fuel where heat is being generated, a rapid drop in the fuel-clad gap due to the low conductivity, and a relatively small drop within the cladding. The profile within the coolant is generally quite complex.

An example of the temperature feedback on the power distribution in a PWR is shown in Fig. 8. A uniform temperature distribution results in a cosine-shaped axial power profile. Since most systems are designed to have negative moderator temperature

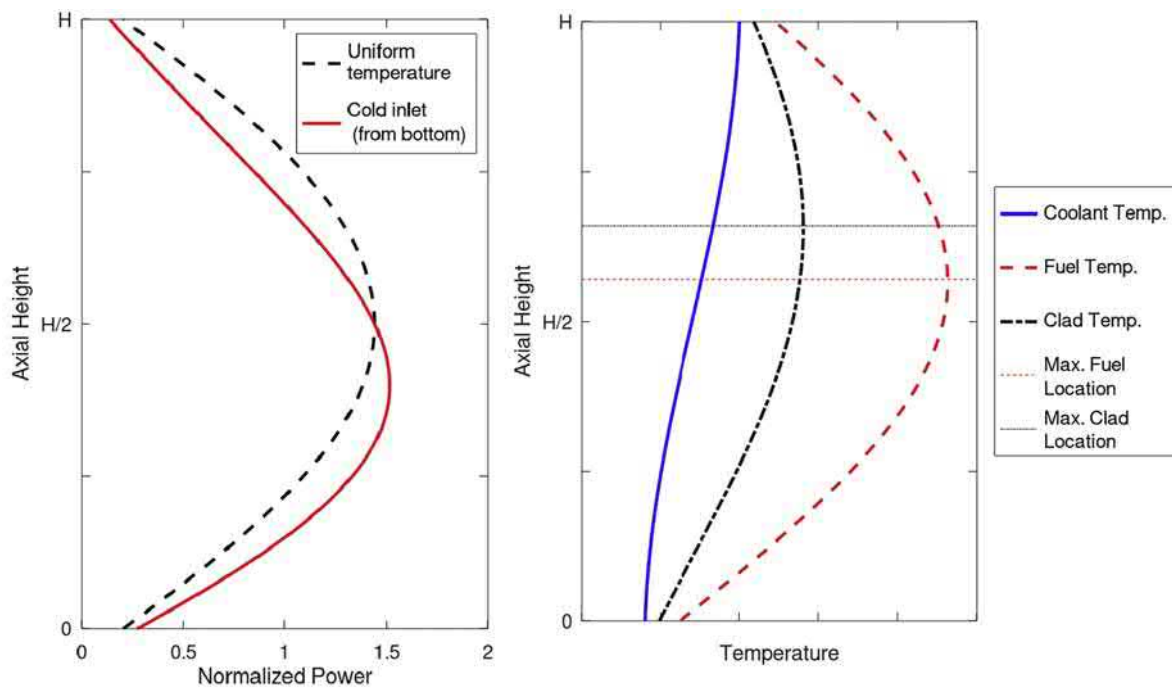


Fig. 8 Axial power distribution for a uniform temperature and with a downward shift due to cold inlet temperature (left); and the resulting coolant, fuel, and cladding temperatures (right).

feedback, cold coolant entering at the bottom will tend to push the power towards the bottom of the reactor. The resulting coolant, fuel, and cladding temperatures are also shown in Fig. 8.

Subchannel methods

Most reactor thermal analysis is done on the basis of the *subchannel*, which is defined as the area in between fuel rods, as shown in Fig. 9. The fluid state (e.g., temperature, density, velocity, pressure, vapor fraction) is not resolved within the subchannel, but instead taken as an average over the area. The 1D heat conduction equation can be solved radially, and then empirical correlations must be used to determine the turbulent heat transfer from rods to the fuel. Generally, changes in the axial flow direction will be resolved by considering multiple nodes in the axial direction. Conservation equations for mass, momentum, and energy are used for the fluid flow. In the case of two-phase flow, which is common in water-cooled reactors, these equations can be solved separately for the liquid, vapor, and entrained droplets.

In some cases, one may do an analysis for an “average” subchannel or “hottest” subchannel, in which case there is assumed to be no flow between adjacent subchannels (similar to the reflective boundary condition used in fuel assembly neutronics calculations), and only a single subchannel will be analyzed. If more fidelity is required, then all subchannels within an assembly, or the entire core, could be modeled. In this case, there will be cross-flow between adjacent subchannels, and a 3D result can be obtained.

Non-dimensional numbers and empirical correlations

Since the details of the fluid flow within the subchannels are not resolved, and only the bulk coolant properties are known, we must rely on empirical correlations to determine the heat transfer from the fuel rods to the coolant. These correlations are usually made on the non-dimensional numbers that characterize the flow conditions (primarily Reynolds, Prandtl, and Nusselt numbers). The use of non-dimensional numbers allows the correlations to be used in a wider variety of conditions than being limited strictly to the experimental conditions. Before discussing these non-dimensional numbers, it is useful to know the main dimensional parameters associated with these problems, as shown in Table 1.

The Reynolds number is the ratio of inertial forces to viscous forces, and effectively represents the effect of turbulence in the system. Viscous forces tend to damp out turbulence, so a high Reynolds number indicates a higher effect from turbulence.

$$Re = \frac{\text{Inertial forces}}{\text{Viscous forces}} = \frac{\rho u D}{\mu}$$

The Prandtl number represents the ratio of momentum diffusivity to thermal diffusivity, and is effectively a property of the fluid as it does not depend on any parameters of the system (e.g., diameter, velocity, etc.). The Prandtl number is very low (< 0.1) for

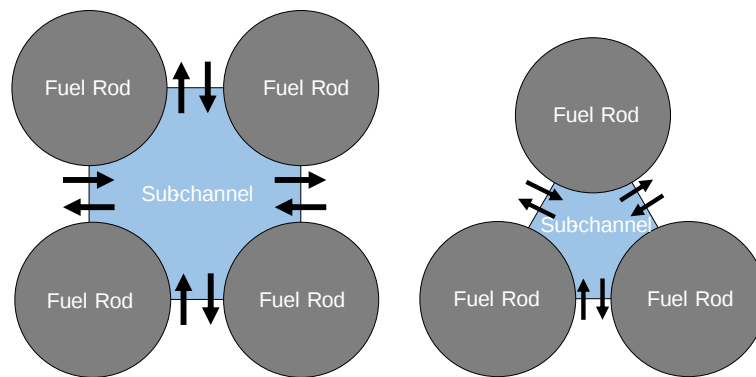


Fig. 9 Illustration of the subchannels for square pitch (left) and triangular pitch (right) fuel rods that are typically used for subchannel analysis.

Table 1 Some dimensional parameters associated with thermal fluid heat transfer problems.

Parameter	Symbol	Unit (SI)
Density	ρ	kg/m ³
Flow speed	u	m/s
Hydraulic diameter	D	m
Dynamic viscosity	μ	kg/(m · s)
Specific heat	c_p	J/(kg · K)
Thermal conductivity	k	W/(m · K)
Convective heat transfer coefficient	h	W/(m ² · K)

liquid metal coolants, since they have very high thermal conductivity and low viscosity. For water and gas coolants, the Prandtl number is around 1.

$$\text{Pr} = \frac{\text{Momentum diffusivity}}{\text{Thermal diffusivity}} = \frac{c_p \mu}{k}$$

The Nusselt number corresponds to the relative importance of convective and conductive heat transfer mechanisms.

$$\text{Nu} = \frac{\text{Convective heat transfer}}{\text{Conductive heat transfer}} = \frac{hD}{k}$$

Empirical correlations such as the Dittus-Boelter equation (Dittus and Boelter, 1930), which applies to forced convection with no boiling, can be used to estimate the Nusselt number (and thus heat transfer from the fuel rods to the coolant).

$$\text{Nu} = 0.0243 \text{Re}^{0.8} \text{Pr}^{0.4}$$

Numerous such correlations exist for a variety of parameters of interest (e.g., pressure drop) and possible flow conditions. Empirical correlations can and have worked very well in nuclear reactor thermal-hydraulic analysis. However, there are significant limitations. First, they are only valid for the experimental conditions in which they were obtained.

In two-phase systems, there is the addition of the void fraction, which represents the relative volume of the vapor phase to the total volume.

$$\chi = \frac{V_{\text{vapour}}}{V_{\text{vapor}} + V_{\text{liquid}}}$$

Two-phase flow

The liquid-vapor flow regimes that are prevalent in boiling water reactors present additional challenges. Depending on flow rates, void fractions, and other conditions, a variety of two-phase flow regimes are possible, which have very different properties and interactions between the phases (Rouhani and Sohal, 1983). Precise delineation between regimes is challenging, but some common regimes include bubble flow, plug flow, slug flow, churn-turbulent flow, and annular flow. These involve bubbles or droplets of various shapes and sizes, films, etc., and have very complex relationships. A discussion of these regimes can be found in Young (2021b). As with single-phase flow, conservation of mass, momentum and energy must be considered, but for both liquid and vapor phases. This should also include some accounting for the interactions at the interface (e.g., interfacial shear, heat transfer) between the phases. Several approaches are possible, with varying levels of complexity (Faya et al., 1979).

The simplest approach is the homogeneous equilibrium model: the fluid and vapor are modeled as a single mixture, with equations for continuity, momentum, and energy. This assumes that both phases flow at the same velocity, and are at saturated conditions.

Higher in complexity is the drift-flux model (Kataoka and Ishii, 1987), which still combines the two fluids into a single mixture, but supplements this with equations for the relative velocity between the two phases (i.e., the drift-flux). This allows for better accounting for interfacial phenomenon than the homogenous model, but at significantly less complexity than a full two-fluid model. Correlations and closure relationships can be made based on the flow conditions, which vary significantly throughout the core, and can be difficult to predict.

Liquid and vapor phases can also be considered separately, with their own conservation equations, as in two-fluid models. Some models, such as in CTF (Salko et al., 2015), consider three fields: vapor, continuous liquid, and droplets. For the two or three-field approaches, there must be terms for the interactions at the interfaces of the three phases. One such approach is the interfacial area transport method (Ishii et al., 2002), which derives a transport equation for the liquid-vapor interface. This can allow for mechanistic treatment of two-phase phenomenon such as bubble break-up and coalescence, as well as different types of bubbles such as cap and spherical.

Subchannel codes

Subchannel methods have been the workhorse of the nuclear industry, and many codes have been used over the years (Moorthi et al., 2018). These include:

- COBRA-TF, which has been adopted by the Consortium for Advanced Simulation of Light Water Reactors (CASL) (Kelly et al., 2017; Salko et al., 2015);
- TRACE/RELAP, which is used by the US Nuclear Regulatory Commission (USNRC) (Fletcher and Schultz, 1992; Odar et al., 2003);
- VIPRE, sponsored by the Electric Power Research Institute (EPRI), approved by the USNRC, and widely used by US utilities (Srikantiah, 1992);
- COBRA-FLX, developed by AREVA (Bhatt et al., 2015).

Computational fluid dynamics methods

As discussed above, subchannel methods with empirical correlations have limitations, especially when it comes to predictive capabilities for new systems, and calculation of complex, 3D flows.

As an alternative, computational fluid dynamics (CFD) methods try to solve the Navier-Stokes equations that govern fluid flow to obtain the locally resolved fluid properties. Although there are several levels of detail that can be resolved in CFD, all methods require many orders of magnitude more unknowns than traditional subchannel or lumped-parameter methods. As such, they are generally too computationally expensive for most design problems at the industry level on desktop systems. Most examples of CFD for nuclear applications are for research or benchmarking, and require large-scale supercomputers.

There are many ways to solve the Navier-Stokes equations, with varying levels of complexity. The major distinguishing factor between methods is how turbulence is modeled, which has a significant impact on computation cost and accuracy. In decreasing order of computational cost, some of the major methods are summarized below:

- Direct Numerical Simulation (DNS) (Merzari et al., 2020; Tiselj et al., 2020). The Navier-Stokes equations are solved directly without additional models for turbulence. This requires an extremely small spatial mesh, down to the smallest eddy size, along with a corresponding short time-step. This method is impractical except for extremely small problems.
- Large-Eddy Simulation (LES) (Merzari et al., 2020; Tiselj et al., 2020). This method has a larger mesh size than DNS, and only spatially resolves the “large” eddies in the problem. Since the smaller, unmodeled turbulence still causes energy dissipation, a sub-grid model is needed to account for them.
- Reynolds-averaged Navier-Stokes (RANS). This method involves taking an ensemble averaging approach to the equation, which allows for larger mesh sizes than LES, but requires more assumptions about the underlying turbulence (Baglietto and Ninokata, 2005).
- Porous media approach. Here, the Navier-Stokes equations are averaged over space. This can result in very large meshes compared to the above methods, and may be considered to be effectively equivalent to subchannel methods but with more flexibility (Novak et al., 2018). Correlations for heat transfer and wall friction factors are required for accurate treatment.

Some codes that have been used for nuclear analysis include: OpenFOAM (Jasak et al., 2007), Nek5000 (NEK5000, 2019), ANSYS CFM, ANSYS FLUENT, STAR-CCM+, and Pronghorn (Novak et al., 2018).

Mechanical, thermo-mechanical, structural, and chemical analysis computational methods

For most analysis, the neutronics/thermal hydraulics is sufficient. However, it is also sometimes necessary to assess the fuel performance over cycle. The large temperature gradients present in nuclear fuel, as well as radiation damage and creation of fission products, greatly affects the microstructure of the fuel, which in turn affects the macroscopic thermal and mechanical properties.

Due to the large range of scales (microstructure vs. macroscopic structural), several modules are usually used. A mesoscale code such as MARMOT (Tonks et al., 2015) can predict microstructure evolution and determine the impact of fuel properties at a local scale. Next, an engineering-scale code such as BISON (Williamson et al., 2012) can predict the fuel stress and performance based on these properties.

Large-scale structural considerations can also be important for operations. Bowing and thermal expansion of fuel elements is especially important for fast reactor systems. These could be solved using a structural finite-element code such as ANSYS Mechanical (Bhashyam, 2002).

Finally, there has been an interest in modeling the chemistry of the fluid, especially as it relates to the formation of crud on the fuel rods. This importantly leads to crud-induced power shift (CIPS), and crud-induced localized corrosion. As a part of the CASL program, the MAMBA code has been developed to model this phenomenon (Short et al., 2013).

Core simulator frameworks

With all of the methods above, and the necessity to consistently couple them and repeatedly define complex problems, there is a need for core simulators or packages to bring these modules together under one framework. Many of these give a platform to use both low and high-fidelity methods where appropriate and to help verify accuracy. Although there are many ad hoc coupling programs that have been developed, some of the more advanced frameworks include:

- Simulation-based High-fidelity Advanced Reactor Prototyping (SHARP), a tool-kit for fast reactor analysis developed by Argonne National Laboratory (Mahadevan et al., 2014).
- Virtual Environment for Reactor Applications (VERA) and VERA-Core Simulator (VERA-CS), developed by the CASL program for light water reactor applications (Godfrey et al., 2017; Turner et al., 2016).
- Reactor Physics Workbench, developed in the United Kingdom for LWRs and Gas-cooled reactors (Hutt et al., 1991).
- The European NURESIM framework, designed to be a reference platform for reactor applications with legacy and modern tools (Chauliac et al., 2011).

Computational validation

At all stages of the computational simulation process to get a practical solution, there are approximations introduced. In order for users and regulators to have confidence in the nuclear simulations, verification and validation of these code results is required. Ideally, this comparison would be with a physical experiment, but this is not always possible or practical. In many cases, a code-to-code validation can be performed to get confidence that the approximations made in the low-order models are valid (Oberkampf and Roy, 2010).

Code-to-code validation involves the comparison of a low-order model that one wants to use for routine calculations to a high-order model that incorporates more realistic physics. A common example of this is comparison of nodal diffusion to Monte Carlo for neutronics (Leppänen et al., 2014). This would validate the spatial, angular, and energy discretization involved with the diffusion approximation. It is important to validate not only integral results, such as reactivity, but local quantities such as 2D and 3D pin (that is, fuel rod)-power.

The main limitation of code-to-code validation is that the codes may not adequately describe reality in several ways:

- Fundamental data they are given (such as cross-sections, which have significant uncertainty).
- Geometry and material information given, which may have uncertainties or be missing details or impurities, for example.
- Certain physics are not incorporated (e.g., thermal expansion or fuel bowing affecting neutronics).

However, code-to-code validation has a few advantages, such as being massively less costly than experiments, with the ability to test many configurations easily. It also allows the comparison of many local variables that may be challenging or impossible to measure, such as detailed, 3D power distributions.

Experimental validation

The gold-standard would be in validation against a full-scale reactor experiment. However, that is rarely possible, especially for new designs. The most important validation is for the thermal-hydraulics models, since they rely so heavily on experimental correlations. These can generally be broken down into separate effects tests, which try to focus on a single phenomenon, and integral effects tests, which incorporate many phenomena. Examples of such facilities include:

- Penn State University Rod Bundle Heat Transfer (RBHT) Test Facility (Hochreiter et al., 2010).
- Oregon State University Advanced Plant Experiment (APEX) (Reyes and Hochreiter, 1998), Multi-Application Small Light Water Reactor (MASLWR) (Reyes et al., 2007).
- Purdue University Multi-dimensional integral test Assembly (PUMA).
- University of California, Berkeley compact integral effects test (CIET) facility (Zweibaum et al., 2016).
- Nuclear Power Engineering Corporation PWR Sub-channel and Bundle Tests (PSBT) (Rubin et al., 2012).

For neutronics evaluations, there have been a number of high quality benchmark experiments done. Most of these have been incorporated into the International Reactor Physics Benchmark Experiment Evaluation Project (IRPhEP) and International Criticality Safety Benchmark Evaluation Project (ICSBEP) databases (Briggs et al., 2014).

The ultimate test of simulated predictions is real-world operational data from an operating reactor. This can be difficult to obtain for academic research, because plants typically do not share this proprietary information. Nuclear reactor vendors, on the other hand, do have access to this data in order to validate their proprietary thermal hydraulic, nuclear, and thermal-mechanical methods. Licensing and validation of these codes is required by regulatory agencies, including validation with plant operational data. However, since the reactor vendor goal is primarily to generate electricity safely and the operational environment is challenging, instrumentation may be more limited and operational conditions not as well-defined as in a designed experiment. One recent example of such operational data is from the Watts Bar Units 1 and 2, which are the most recently constructed reactors in the United States, and provided significant amount of data as part of the CASL program (Godfrey et al., 2016, 2017).

Core monitoring methods

Predicting reactor conditions must also be combined with measurements during operation. This provides another source of validation for the modeling tools and ensures that the reactor is always operating within the approved limiting conditions that were used for the safety analysis. The General Design Criteria in US CFR 50 Appendix A, mandates the monitoring of system variables and automated responses (GDC-13, GDC-20) to ensure that limits are not exceeded (U.S. Nuclear Regulatory Commission, 2007). Furthermore, higher accuracy models and on-line core monitoring can allow for smaller margins for operating reactors, leading to more advantageous power distributions in the core, which can result in better profitability due to the increased allowable power output.

Inlet and assembly outlet temperatures are generally measured with thermocouples, while the measurement of power distribution is more complex (Upadhyaya and Wood, 2021). During operation, the power is usually measured by a set of fixed self-powered neutron flux detectors throughout the core. Since the detector response may change throughout the core burnup (e.g., due to

changes in the number of neutrons released per fission with more plutonium), these detectors are periodically calibrated using movable neutron or gamma detectors. The overall heat balance of the plant can be used for an absolute calibration to the reactor power. Ex-core detectors (external to the core) can also be used, especially when operating at very low power levels, such as during startup, when the in-core power range detectors may not provide enough sensitivity. Ex-core detectors operate in a much more forgiving physical environment and with fewer spatial constraints, but also provide much less spatial resolution than in-core instrumentation.

Data collected from the detectors is used for automated protection systems, and is also evaluated with core simulator results for monitoring the fuel thermal limits. These comparisons can ensure that technical specifications limits were followed, such as departure from nucleate boiling, and Specified Acceptable Fuel Design Limits, such as local power peaking limits, are met.

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Transient Analysis Methods

R.P. Martin, Technical Consultant, BWX Technologies, Inc., Lynchburg, VA, United States

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Glossary

Actinides Any of the series of 15 metallic elements from actinium (atomic number 89) to lawrencium (atomic number 103) in the periodic table. They are all radioactive, the heavier members being extremely unstable and not of natural occurrence.

Analytical model A mathematical description of physical process or structure, system or component.

Anticipated operational occurrence Those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include, but are not limited to, a loss of all off-site power, an inadvertent control rod withdrawal and tripping (shutting off) of the turbine generator set.

Anticipated transient without scram Events initiated as anticipated operational occurrences compounded by a failure in the normal reactor scram (i.e., immediate insertion of all safety rods) system.

Critical safety function Those safety functions that are essential to prevent a direct and immediate threat to the health and safety of the public. For conventional light water reactors, these are the accomplishing or maintaining of:(1)Reactivity control;(2)Reactor core cooling;(3)Reactor coolant system integrity (as a fission product barrier);(4)Primary reactor containment integrity; and(5)Radioactive effluent control.

Critical heat flux The condition in boiling heat transfer where the steam produced forms an insulating layer over the heated surface, leading to a reduction cooling at the heater surface. Surface temperature subsequently increase until a stable vapor film forms. This term is commonly associated with departure from nucleate boiling (DNB), which identifies the onset of the boiling regime transition from nucleate to film boiling.

Critical power The assembly power at which critical heat flux occurs. It is an empirical quantify used in Boiling Water Reactor safety analysis that must also consider the fluid quality of the coolant.

Design-basis event A design-basis event is a condition of normal operation, including anticipated operational occurrences, accidents (or transients), external events or natural phenomena for which the plant must be designed to ensure the three basic safety-related functions are achieved. Refer to the definition for “safety-related.”

Effective dose The tissue-weighted sum of the equivalent doses in all specified tissues and organs of the human body and represents the stochastic health risk to the whole body, which is the probability of cancer induction and genetic effects, of ionizing radiation. It takes into account the type of radiation and the nature of each organ or tissue being irradiated, and enables summation of organ doses due to varying levels and types of radiation, both internal and external, to produce an overall calculated effective dose. $E = \sum_R W_T \cdot H_T$, where E is the effective dose to the entire organism, H_T is the equivalent dose in

Sievert (Sv) absorbed by tissue ($1 \text{ Sv} = 1 \text{ joule/kg}$ —the equivalent biological effect of the deposit of a joule of radiation energy in a kilogram of human tissue), W_T is the tissue weighting factor defined by regulation.

Emergency core cooling system The emergency core cooling system is a system (or group of systems) required for LWRs that is designed to provide reactor core cooling following postulated accidents up to and including the design-basis loss-of-coolant accident. Its performance is analytically demonstrated and must conform to the acceptance criteria set forth in 10CFR50.46¹ regarding peak fuel cladding temperature, fuel cladding oxidation, hydrogen production, the maintenance of a coolable core geometry and successful core cooling in the long-term.

Engineered safety features Engineered safety features (ESF) are structures, systems, or components (SSCs) included in the plant design for the specific purpose of transient and accident prevention and mitigation. These SSCs may be dual purpose; that is, they may also serve non-accident mitigation functions. An example of a dual-purpose system is the decay heat removal system, which also typically provides low-pressure safety injection flow for a large loss-of-coolant accident. Typically, these SSCs are classified as safety-related, but certain ESFs (e.g., the BWR standby liquid control system) are not classified as safety-related because they exceed the requirements of the single-failure criterion and instead provide diversity and defense-in-depth for accident mitigation.

Figures-of-Merit Figures-of-merit are those quantitative standards of acceptance that are used to define acceptable answers for a safety analysis.

Fission products The nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclide formed by the fission fragments' radioactive decay.

Limiting conditions for operation The lowest functional capability or performance levels of equipment required for continued operation of the facility without undue risk to the health and safety of the public.

Loss-of-coolant accident Those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

Margin (i.e., design or safety) A quantitative relationship between a design evaluation result for a given event and a limit associated with a functional requirement.

Neutron activation the process in which neutron radiation induces radioactivity in materials, and occurs when atomic nuclei capture free neutrons, becoming heavier and entering excited states.

Postulated accidents Those design-basis occurrences not expected to occur during the life of the nuclear power plant, such as a loss-of-coolant accident.

Reactivity A term expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Reactivity initiated (insertion) events (accidents) Events (or accidents) that involve positive reactivity insertion in the reactor core, thereby causing a power excursion and corresponding temperature increase, locally or globally.

Safety analysis An analysis that simulates the integrated response of the plant to a design-basis event (DBE) in order to confirm that functions of structures, systems and components are accomplished within their design basis and the nuclear safety criteria and safety analyses acceptance requirements are satisfied for the DBE.

Simulation A mathematical imitation of the response of a physical process or structure, system or component, represented by a model, to a state perturbation.

Introduction

National nuclear safety authorities, such as the Nuclear Regulatory Commission in the United States (USNRC) and the Autorité de Sûreté Nucléaire (ASN) in France, require that nuclear power plant designs be evaluated through physical testing and analysis. Because of the practical limits of physical testing, analysis is used as a surrogate to examine a broader range of equipment performance and possible failures or malfunctions. Of particular interest to nuclear regulators are failure sequences presenting the greatest physical challenge to the protective barriers separating radiological hazards, i.e., chemical elements derived from the splitting of the atom or fission products, from nuclear power plant staff and the public. The transient behavior resulting from such equipment failures or malfunctions are referred to as design-basis events. Using testing and analysis, nuclear power plant designers demonstrate that the resulting transient nature of the nuclear power plant's physical states, such as temperature, pressure, inventory, chemistry, or neutron balance, do not lead to a failure of the protective barriers. The use of transient analyses informs the nuclear power plant licensing process by validating its design envelope as defined by the limiting conditions of operation, limiting safety system settings, and technical (design) specifications.

¹The AEC was the predecessor of the USNRC, serving as both the promoter and regulator of nuclear energy project from 1946 until 1975.

Using analysis to confirm the design and safety adequacy of a nuclear power plant relies on strict adherence to an auditable calculational process, referred to as the evaluation model. An “evaluation model” is the collection of evidences related to a particular analysis. It will include one or more computer programs and all other information necessary to characterize a particular transient or accident scenario. This will include an analytical model provided by a mathematical description of the laws of (fluid) motion, physical phenomena and processes, plant design details and transient characteristics. In addition, the uncertainties, assumptions and limitations of the computer code(s) are also described and used to inform procedures for treating the program input and output information. Computer codes are to be qualified against the expected system behavior as observed from applicable experimental programs. Similarly, the way the computer codes are applied in the conduct of the evaluation model analysis is also subject to verification and validation.

This Transient Analysis Methods chapter summarizes the preparation of evaluation models from high-level requirements to the performance of nuclear-quality simulation and the interpretation of results for design and safety applications.

Regulatory requirements and guidance for nuclear safety analysis

Regulatory agencies like the USNRC and ASN are established and tasked with protecting public health and safety related to nuclear energy. Before a nuclear power plant can be built and operated, regulatory agencies require that the engineering organization advancing the project to submit a safety analysis report (SAR). The SAR describes the scope of the nuclear project in broad detail with an emphasis on analyses of transients and accidents that demonstrate design robustness. It identifies the anticipated hazards and the requirements for containing and managing these hazards. Results from physical testing and transient analysis appear in the SAR as evidence demonstrating that the role of physical barriers and other safeguards to protect and/or mitigate the release of fission products is maintained for all credible scenarios involving equipment failures or malfunctions.

The government regulations and guidance related to the content of a SAR begins with policies and standards for protections against ionizing radiation. These statements set regulatory objectives that build upon the principle that nuclear power plants provide adequate margins of safety and utilize effective means to accomplish their safety and security functions. Woven through the regulatory framework is the Defense-in-Depth requirement that nuclear power plants include the following: (a) design features for safe and reliable plant performance under normal operating conditions, (b) a plant protection system to protect the reactor against the consequences of any off-normal event, and (c) engineered safety features to provide additional safety margins for extremely unlikely and unforeseen events. Corresponding to this Defense-in-Depth philosophy, the full scope of physical testing and safety analysis that appears in a SAR examines the following:

- Prevention of abnormal operation and failures;
- Control of abnormal operation and detection of failures;
- Control of accidents within the design basis;
- Control of severe plant conditions;
- Individual effective dose from radiation to workers and public;
- Mitigation of radiological consequences;
- Integrity of physical barriers to the release of radioactive materials (i.e., fuel, fuel cladding, reactor coolant system and containment);
- Sustained capability of operators and safety systems; and
- Practical exclusion of large early releases of radioactive material.

For the breadth of physical testing and analysis required in the preparation of a SAR, the principal safety criteria relate to the effective dose from radiation. Dose estimates factor into how nuclear power plant staff performs periodic maintenance, how they respond during abnormal events, and where a nuclear power plant can be sited. Regarding the latter, plant siting explicitly considers the reactor design, the population density near the site and the site’s physical characteristics. It includes the acceptance criteria, in terms of the radiation dose, that are ultimately quantified from interpretations of transient and accident analysis. [Bley \(2021\)](#) details the safety and licensing requirements and acceptance criteria as they relate to radiological measures.

As [Bajorek \(2019\)](#) explains, transient analysis methods address how calculations are prepared, which, of course, incorporates regulations, but also includes the procedures and technical bases that provide evidence that the analyses are meaningful. Surrogate measures are emphasized that relate to the performance of one of the Defense-in-Depth measures, such as the integrity of the physical fission product barriers (e.g., fuel, fuel cladding, and reactor coolant pressure boundary) or the performance of an engineered safety feature. These measures, such as fuel bundle critical power in boiling water reactors (BWRs), critical heat flux in pressurized water reactors (PWRs), and peak clad temperature in all light water reactors (LWRs), are then compared against acceptance criteria. Events assumed to have radiological consequences are emphasized in safety assessments.

The particular set of measures and applicable acceptance criteria evaluated in transient and accident analysis begins with the assumed initial plant state and the nature of the equipment failures or malfunctions. Extensive failures are not considered credible for purposes of assessing the plant’s resilience to design-basis events. The regulatory intent for the set of events analyzed is that they represent a sufficiently broad number of situations to demonstrate a thorough assessment of credible risk-significant events. The basis of credibility for a particular event is determined from an assessment of risks, drawn from knowledge of the expected frequency of the associated set of failures (or malfunctions) and consequences from these failures.

Two general event classifications are recognized by national nuclear safety authorities, distinguished based on frequency:

- (1) Anticipated operational occurrences (AOOs)—Events expected to occur on a frequency of one or more times during the lifetime of the plant;
- (2) Postulated accidents (PAs)—Events not expected to occur that are evaluated to demonstrate design robustness.

Categorization of initiating events by either AOOs or PAs provides a foundational characteristic between events that serves to distinguish the more credible events from severe accidents. In 1973 the American Nuclear Society (ANS) published a standard (ANS 18.2 1973) entitled, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." For safety analysts, this Standard organized events by their "condition." This "condition" primarily refers to event frequency and the classifications defined in that Standard are:

- Condition I: Events expected to occur frequently in the course of power operation, refueling, maintenance or plant maneuvering.
- Condition II: Events expected to occur on a frequency of once per year during plant operation.
- Condition III: Events expected to occur once in the lifetime of the plant.
- Condition IV: Events not expected to occur that are evaluated to demonstrate the adequacy of the design.
- Condition V: Hypothetical core damage events of very low frequency but with high radiological consequence, exceeding siting limits.

Tolerance for consequences from failure vary inversely with Condition. For Condition I and II events, for instance, there is zero tolerance for radiological releases. For Condition V events, radiological consequences are possible but, through design measures, the likelihood of these events is so low as to be practically eliminated. Because events in the Condition III and IV categories cannot be practically eliminated and have radiological consequences, they have received considerable regulatory attention. Events in these categories include anticipated transients without scram (ATWS) and loss-of-coolant accidents (LOCA).

An ATWS is an AOO followed by a failure of the reactor control and protection system to shut down the reactor via an automatic scram or reactor trip. Failure of the reactor protection system to provide a scram when required could cause an otherwise minor event to turn into a high consequence accident. Because of the potential consequence of ATWS, regulations have evolved to require diverse, independent, reactor trip functionality. Rules pertaining to ATWS are generally met by either demonstrating that the reactor protection system meets the diversity requirements or by providing a separate hardware backup scram system. The protection system must include a completely independent reactor scram function, including diversity in everything from sensors to final actuation of the scram.

A LOCA is an event characterized by a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system to mitigate. This is expected from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system. The larger sized LOCAs can be severe, and for this reason, it is frequently referred to as the Maximum Hypothetical Accident. Given this potential, the complexity of the phenomena associated with the event and the performance of a plant's safety features designed to restore cooling the reactor core, analysis of the LOCA has received special attention in nuclear power plant regulations. Notably, these regulations are specific with regard to reactor and fuel design, LWRs with zirconium-clad fuel elements.

Since the 1970s, LOCA-related regulations worldwide include a specific procedure for simulating the performance of a nuclear power plant's emergency core cooling system (ECCS) deterministically. In the United States, this procedure is referred to as "Appendix K," a reference to its location in the U.S. Code of Federal Regulations on nuclear energy (i.e., 10 CFR 50 Appendix K). The features of this model that incorporate penalizing assumptions to ensure estimates for peak fuel cladding temperature and other acceptance criteria by this method are conservative.

Beginning in 1988 in the United States, followed by several other countries with nuclear power plants, rules allowing the use of best-estimate computer codes with some relaxation on the conservatism described above emerged. The degree to which best-estimate solutions to the LOCA analysis are permitted vary considerably among these countries. Evaluation methodologies described as best-estimate plus uncertainty (BEPU) methods have emerged as the preferred approach and most aligned with principles of best-estimate analysis that rely on the complementary application of science- and data-based analytics.

Design-basis event characterization

Given the predominance of LWR technology in the commercial nuclear industry, the characterization of design-basis events has been largely vetted through decades of deliberation between regulatory agencies and the industry. Consequently, regulations for the commercial nuclear power industry in most countries, with Canada and the UK notably exceptions, incorporate LWR-specific prescription for the particular set of design-basis events (DBE) that must be analyzed. To support the emergence of LWR and non-LWR advanced reactors, the responsibility for DBE characterization is with the design and engineering organization. To do so, engineers apply hazards analyses.

A successful hazards analysis requires the analyst to include all risk-significant failure modes for each contributing element in the plant. A DBE derives its detail beginning at the plant scale then through an assessment of progressively shrinking scales: facility, system, subsystem, component, and part. Methods vary with the degree to which probabilistic characteristics are considered. Not surprisingly, the original assessments in this regard, prepared by the Atomic Energy Commission² (AEC) in the United States

²Presented in generalized units of T = Time, L = Length, Θ = Temperature, M = Mass, E = Energy.

in 1950s and 1960s, had relatively little data from which to draw probabilistic insights. Rather, the AEC incorporated experience from their sponsorship of test programs and high-level consideration of risk. This experience provided the basis for identifying system-level failures corresponding to one or more critical safety functions (Martin, 2019).

System-level failure modes are failures reflecting operability deviations challenging system-level structures, systems or components that are relied upon to remain functional during and following DBEs. Risk insights are considered to the extent to which DBEs can be categorized into the Conditions previously defined. The severity of the event is then compounded by the requirement that the challenged critical safety function, as served by one or more engineered safety features, is degraded by a “worst single failure.”

The importance of the single-failure criterion cannot be overstated. This deterministic criterion specifies an approach to obtaining optimal redundancy of a system or of a group of components ensuring a high degree of reliability. It requires a systematic search of potential single failures followed by an evaluation of the effect of each single failure on the ability of the plant’s engineered safety features to satisfy the specified principal design criteria, each of which relate to the protection of one or more critical safety functions or fission product barriers.

A single-failure analysis methodology incorporates both qualitative and quantitative evaluation measures, yielding candidate limiting equipment failures and malfunctions for further characterization and, ultimately, to serve as the basis for a plant’s design-basis envelope. The assessment does not require consideration of any conceivable failure that could occur; rather, “only those systems or components judged to have a credible chance of failure are assumed to fail in applying the single-failure criterion” (Institute of Electrical and Electronics Engineers, 2001). The common activities performed in the assessment of the worst single-failure include:

- Identifying structures, systems and components important to safety;
- Listing the credible vulnerabilities with respect to each critical safety function;
- Defining figure(s)-of-merit and criteria that assure the critical safety function;
- Evaluating the impact of failure for each vulnerability in terms of its influence on the figure(s)-of-merit and design and acceptance criteria.

Other DBE attributes are evaluated deterministically, incorporating a nuclear power plant’s limiting conditions of operation and technical specifications targets. In the characterization of DBEs, these targets are commonly implemented as initial conditions in a transient or accident analysis. The deterministic approach to characterizing the initial condition is to identify the plant state (i.e., in terms of model parameters in an analysis) that will lead to the smallest calculated margins in the figures-of-merit relative to acceptance criteria. Many of these will be set consistent with the plant’s Technical Specifications. For example, core power is set at nominal full power plus instrument uncertainty, core decay heat is set assuming infinite operating time, core outlet temperature is set at the maximum allowable, and core flow rate is set at a minimum.

Design-basis characterization methods that were originally applied to the commercial nuclear industry were technology neutral. Given that the industry favored LWR technology (with government encouragement), the efforts by national regulators to identify and characterize DBEs has been documented only in terms of how they have been applied for LWRs. In the United States, the most complete description of LWR DBEs is documented in NUREG-0800, Standard Review Plan (U.S. Nuclear Regulatory Commission, 2007). The many identified events can be generalized for a broader spectrum of nuclear power plant designs as:

- Increase in heat removal by a cooling system
- Decrease in heat removal by a cooling system
- Flow rate transients (such as hydrodynamic flow instability events)
- Decrease in coolant system inventory
- Increase in coolant system inventory
- Reactivity transients

where the reference to “coolant systems” can refer to systems dedicated to the core, primary coolant, containment or one of several auxiliary coolant systems.

Since the AEC’s original determination of LWR DBEs, reliability analysis has matured into analytically-consistent methodologies, such as functional hazard assessment, failure modes and effects analysis, and event and fault tree analysis (Lee and McCormick, 2011). More recent methods, such as Aldemir (2013) and Idaho National Laboratory (2017), extend the reliance on risk insights to a broader selection of event characteristics. Regardless of the preferred method, the objective of the hazards analysis is to systematically assess postulated failures and determine the plant’s response.

Evaluation model, phenomenological importance and scaling

The development of the knowledge bases supporting transient and accident analysis relies upon both “top-down” and “bottom-up” elements. The “top-down” element relates to the applicability of governing equations, constitutive relationships or models (i.e., models describing phenomenological behavior) and their organization as an analytical solution. The “bottom-up” element relates to the completeness, consistency and overall performance of the evaluation model.

Given the complexity of nuclear power plant transient analysis, the analytical solution is provided by a computer code, often consisting of thousands of lines of coding. As the centerpiece of the evaluation model, every facet of the computer code is

characterized and described. Since the computer code is expected to emulate a real system, the analytical solution must be shown to apply to the transient scenarios that the analyst expects to resolve through simulation. As such, the more important mass, momentum, and energy transport mechanisms, i.e., the phenomena and processes, are explicitly represented in the analytical model. Once the governing equations and constitutive models describing these transport mechanisms are organized into a computer code, the code's performance is reviewed and assessed in terms of numerical convergence, stability, and property conservation and its fidelity against separate- or integral-effects test data. This review involves three distinct activities of verification, validation, and uncertainty quantification, which are described in later sections; but fundamental to these activities is the knowledge base of physical phenomena expected to be present during the DBEs of a real system. This knowledge base establishes the set of requirements for the evaluation model and the computer code that facilitates transient analysis.

Managing knowledge comprises the strategy used to translate the insights and experiences relating nuclear power system performance during DBEs to a complete and self-consistent evaluation methodology. In raw form, this knowledge exists in individuals and appears in documents that preserve our understanding of nuclear power plant DBEs. Given the breadth of this knowledge base, the development of an evaluation model begins by identifying a reduced set of more important phenomena and processes influencing the physical state of a nuclear power plant during a DBE. The Phenomena Identification and Ranking Table (PIRT) methodology is broadly applied in the nuclear industry for this purpose (Wilson and Boyack, 1998).

Fundamentally, computer codes for transient analysis of nuclear power systems must model the principal phenomena associated with the particular plant and scenario of interest. This specificity provides the initial prioritization of the knowledge base. The PIRT process provides a framework for the structured collection and documentation of subject matter expert judgment with respect to phenomena and their importance in an evaluation model. The particular transient event or accident is then broken into multiple phases characterized by the dominant phenomena and processes present and their effect on measurable loads (i.e., figures-of-merit) on the real system. The PIRT organizes the identified phenomena by each system and component involved in the transient and by ranking the importance of the phenomena relative to the figures-of-merit in each phase. In addition, the quality or state of the phenomena-specific knowledge base is also assessed and an overall risk assessment is made based on the combination of importance ranking and the state of knowledge. As such, it provides part of the basis for the following evaluation model development objectives:

- Determining the scope of the effort required to verify the computer code (i.e., does the computer code properly model the important phenomena).
- Identifying phenomenological parameters that can be quantified for evaluating and propagating uncertainties.
- Identifying or developing test data that contain the appropriate phenomena during each accident phase and support the quantification of evaluation model uncertainties.

The majority of the effort in developing an evaluation model is that associated with the last objective. With the important phenomena identified, test programs are identified or designed that provide data appropriate for measuring the performance of the computer code and adequacy of the evaluation model, generally.

Since events in nuclear power plants involve complex interactions between fluid and structure exchanging mass, momentum, and energy, the hierarchical nature of the phenomena and processes as modeled in a computer code must be aligned with that appearing in a real system. To address this requirement, a scaling analysis is performed. Scaling analysis is broadly practiced in all disciplines of engineering. For transient analysis of nuclear power plants, the Hierarchical Two-Tiered Scaling (H2TS) methodology is commonly applied (Zuber, 1991), which builds on and extends the Ishii and Jones (1976) method. The Dynamic System Scaling methodology (Reyes and Frepoli, 2019) presents a contemporary unifying approach to scaling by incorporating a geometric analogy.

Similar to the evaluation model itself, the two tiers of the H2TS methodology are the top-down or system approach and the bottom-up or process approach. The top-down approach yields a set of nondimensional scaling or similarity parameters that are expressed as characteristic time ratios. The second tier or bottom-up approach provides more detailed, localized analyses of the important processes. The two tiers provide an overall analysis methodology that is objective, tractable and technically justifiable.

In contrast to the PIRT process that relies on the observational subjectivity of experts, the H2TS emphasizes a physically-based hierarchy (i.e., of the form facility, system, subsystem, etc.). Since phenomena or processes can appear at any scale, a pattern emerges between the chronology of knowledge development and phenomena or process scale. Notably, statics and dynamics are fundamental to processes at all scales and appear associated with the system and subsystems, respectively. Modules are commonly equated with components; however, generically, they represent contributions of work or energy from one or more physical domains (e.g., hydraulics, heat transfer, fission, etc.). The actual fluid(s) or, more generically, the "medium of motion" are the constituents. Fluid state in terms of its appearance as a solid, liquid or gas relates to its phase. The configuration of phases is described by both the geometry encompassing the fluid(s) and the topological arrangement within that geometry. Fig. 1 illustrates a conceptual hierarchical knowledge base decomposition of a PWR DBE. At the process level appear a set of phenomena that would appear originally in a PIRT.

Expressions for the nondimensional similarity parameters are derived from an analytical model that relates the phenomenon or process at the scale of interest to figures-of-merit. Using test program design data and results from physical or computer-simulated testing, these parameters can be quantified and compared for alignment. Where test program data and computer simulations compare well, the quality of the two is considered adequate. In contrast, discrepancies suggest shortcomings in either the test or the computer code or, possibly, both, requiring further investigation or other analytical treatment to resolve.

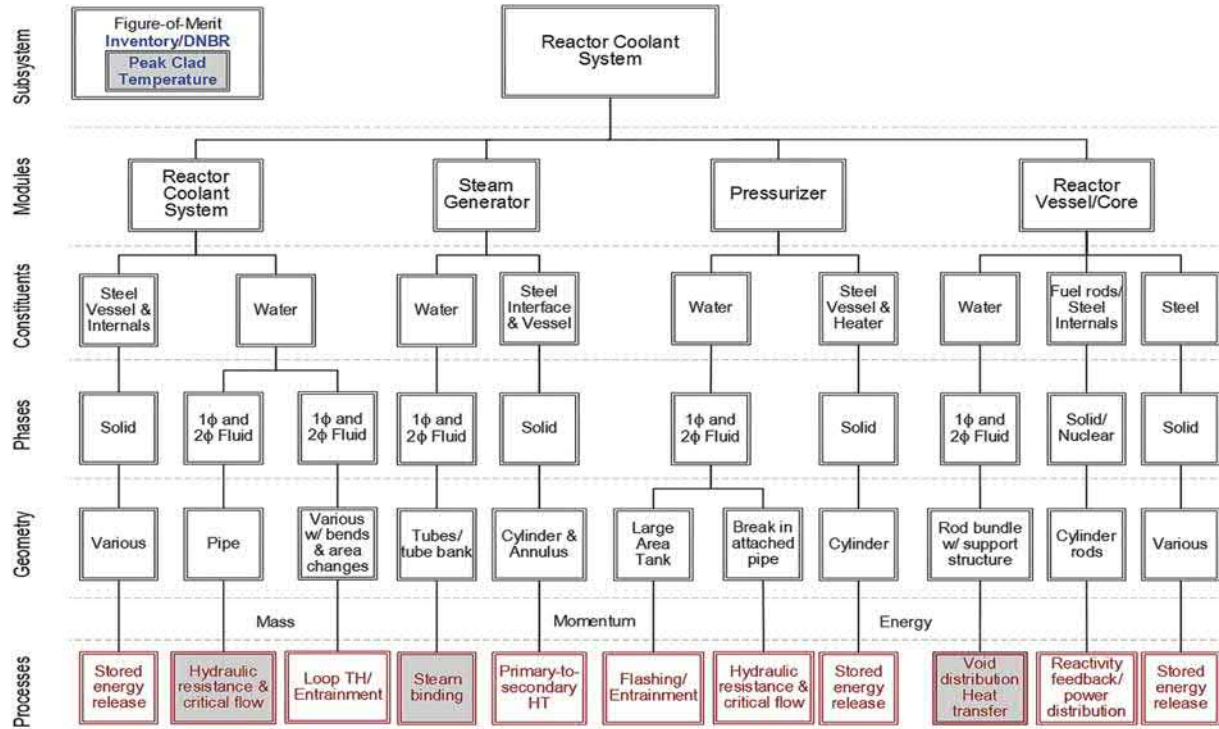


Fig. 1 Hierarchical knowledge base decomposition example for a PWR DBE.

Analytical requirements and software

The objective of the evaluation model for transient analysis of a nuclear power plant is to establish or verify a measure of design adequacy under normal operation and DBEs. Given the complexity of these applications, analysis relies on a hybrid of physics- and evidence-based computer codes for simulation of nuclear power plant processes. The virtual environment of this software combines the knowledge of the plant geometry, boundary and initial conditions with knowledge of the important physical phenomena. Ensuring quality transient analysis begins with a strong foundation in thermal-hydraulic phenomenology, which is to be translated into requirements, model theory, software and applications.

For the commercial nuclear power industry based on LWR technology, the greatest challenge to this effort has been the modeling of two-phase fluid (i.e., liquid water and steam) behavior. An inaccurate solution could lead to an under prediction of important measures of safety like fuel and clad temperatures or the timing of when coolant starts to boil. As such, the development evolution of these computer codes has been driven by emphasizing such figures-of-merit. This has required coincident experimental program for data to characterize two-phase flow. Over the decades of related research, these pursuits have revealed that the related phenomena impacting nuclear safety measures are many and varied.

The method of modeling in these computer codes involves a set of governing or field equations used to model the two-phase fluid flow. “Governing equations” refer to equations for the conservation of mass, energy and momentum of the working fluid and, optionally, a non-condensable gas and/or a liquid solute. These equations are solved to provide predictions of the spatial distribution of liquid, steam and non-condensable mass as well as the velocity and energy state for each phase. These governing equations contain parametric dependencies related to geometry, phenomena and processes modeled by these equations. Solution is only possible with a complete and consistent set of models and correlations providing the necessary mathematical closure.

Since the mid-1970s, the six-equation, Two-Fluid Model is the predominant description for two-phase flow in thermal-hydraulic systems codes for nuclear safety analysis (Ishii, 1975). The distinguishing characteristic of the Two-Fluid model is that it considers mass, momentum, and energy transport between the liquid and vapor phases. In the more common solutions to the Two-Fluid model, appearing below, the dependent variables are vapor void fraction (α_v), pressure (P), liquid-phase total energy or enthalpy (h_l), and vapor-phase enthalpy (h_v), liquid velocity (v_l) and vapor velocity (v_v). From Martin et al. (2019a), the phasic mass continuity equations are

$$\overbrace{\frac{\partial}{\partial t}(\alpha_k \rho_k)}^{\text{time rate of change}} + \overbrace{\frac{\partial}{\partial z}(\alpha_k \rho_k v_{kz})}^{\text{advection}} = \overbrace{\Gamma_k}^{\text{source/sink}}, \quad (1)$$

where ρ_k are liquid and vapor density (i.e., ρ_l and ρ_v), respectively, α_k is liquid and vapor void fraction (i.e., α_l and α_v), respectively and v_{kz} is the phasic velocity in the z -direction (v_{lz} and v_{vz}), respectively. For the mass continuity equation, the only source or sink for the liquid and vapor phases of water is interfacial mass transfer (Γ); namely, boiling, flashing, evaporation (in the presence of non-condensables) and condensation. In order to conserve mass, the relationship between the phasic terms at the interface (also known as the jump conditions) can be written as

$$-\Gamma_l = \Gamma_v = \Gamma, \quad (2)$$

which states that a given mass of liquid gained is balanced by the loss of an equivalent amount of vapor.

The phasic momentum equations are

$$\begin{aligned} & \overbrace{V(\alpha\rho)_k \frac{\partial}{\partial t}(v_{kz})}^{\text{time rate of change}} + \overbrace{V(\alpha\rho v_z)_k \frac{\partial}{\partial z}(v_{kz})}^{\text{advection}} = \\ & \overbrace{-\Delta z \frac{\partial}{\partial z}(p\alpha_k A_z)}^{\text{Pressure Gradient}} - \overbrace{\alpha_k \rho_k V g \sin\theta}^{\text{gravity body force}} - \overbrace{\tau_{kw} P_{kw} \Delta z}^{\text{wall shear}} \\ & - \overbrace{\tau_{ki} A_i}^{\text{interfacial shear}} + \overbrace{(v_{ki} - v_{kz}) \Gamma_k V}^{\text{momentum transfer by mass transfer}} \end{aligned} \quad (3)$$

where z , P_w , A_z , A_i , and V are, respectively, the control volume length, perimeter, cross-sectional area, phasic interfacial area, and volume; g is the gravitational constant; τ_{kw} and τ_{ki} are the wall and interfacial shear stress terms; and other variables and subscripts as defined with Eqs. (1) and (2). In this expression, the final two equation terms must sum to zero when the phasic equations are added together to form the mixture equation. In practice, the terms are typically treated individually instead of as a sum. Namely,

$$\tau_{li} A_i = -\tau_{vi} A_i \text{ and } v_{li} \Gamma_l V = -v_{vi} \Gamma_v V \quad (4)$$

The interfacial shear term (τ_{li}) is typically written as

$$\tau_{li} = f_i \frac{1}{2} \rho (v_v^2 - v_l^2), \quad (5)$$

where the interfacial friction factor (f_i) and the phasic density (ρ) to be used in the expression depend on the flow topology, describing the configuration of vapor and liquid together, such as a predominant liquid field with bubble or slugs of vapor, or a predominant vapor field with dispersed droplets. These are examples of flow regimes.

The energy equation is most often written in terms of internal energy or enthalpy. Both of these state conditions are directly related to temperature via the specific heat at constant volume and constant pressure, respectively. A common form of the energy equation is given as

$$\begin{aligned} & \overbrace{V \frac{\partial}{\partial t} [(\alpha\rho h^0)_k]}^{\text{time rate of change}} + \overbrace{\frac{\partial}{\partial z} [(\alpha\rho v_z h^0)_k V]}^{\text{advection}} = \\ & - \overbrace{(\alpha\rho v_z)_k V g \sin\theta}^{\text{gravity}} - \overbrace{pV \frac{\partial \alpha_k}{\partial t}}^{\text{pressure work}} + \overbrace{q_k''' V}^{\text{generation}} + \overbrace{q_{wk}'' A_w}^{\text{wall heat transfer}}, \\ & + \overbrace{q_{ik}'' A_i}^{\text{interfacial heat transfer}} + \overbrace{\Gamma_k h_{ki}^0 V}^{\text{energy with mass transfer}} \end{aligned} \quad (6)$$

In this equation, h_k^0 is the phasic enthalpy and h_{ki}^0 is the phasic enthalpy associated with mass transfer. q_k''' , q_{wk}'' , and q_{ik}'' are the internal (volumetric) heat generation, wall heat flux and interfacial heat flux, respectively. For a system in thermal equilibrium, this would represent a saturated phasic enthalpy. However, in the case of thermal non-equilibrium the use of purely saturated enthalpies for the interfacial terms can lead to non-physical results. To avoid this problem, the calculation of the effective enthalpy is chosen based on the direction of mass transfer. Namely, the disappearing phase uses the local phasic enthalpy and the appearing phase uses the saturated value. In other words, the disappearing phase takes all of its energy with it, while the appearing phase is born at the saturated value.

Significant thermal non-equilibrium is common during a LOCA. As such, thermal-hydraulic system codes must be able to predict thermal non-equilibrium between the fluid phases. Specifically, the codes must be able to predict stable ($T_l < T_{sat}$ and $T_v > T_{sat}$) and meta-stable conditions ($T_l > T_{sat}$ and $T_v < T_{sat}$). Interfacial heat transfer (denoted as the product $q_{ik}'' A_i$) is included in the phasic energy equations. This energy is commonly described as the transfer of energy from the fluid at its phasic temperature to a massless interface that exists at a saturation temperature. The interfacial heat flux (q_{ik}'') is defined using an analog of Newton's Law of Cooling.

$$q''_{ik} = h_{ik}(T_{\text{sat}} - T_k), \quad (7)$$

In order to conserve energy, the sum of the last two terms in the energy equation must be equal to 0.0 when summed over the liquid and vapor phases.

$$\Gamma_l h_{li}^0 V + q''_{il} A_i = \Gamma_v h_{vi}^0 V + q''_{iv} A_i, \quad (8)$$

Rearranging the equation and remembering the relationship between the liquid and vapor side interfacial mass transfer terms allows us to develop an equation for the interfacial mass transfer, as a function of the interfacial heat transfer rates and the effective enthalpies.

$$\Gamma = -\frac{(q''_{iv} + q''_{il})A_i}{(h_{vi}^0 - h_{li}^0)V}, \quad (9)$$

The previously described set of governing equations actually contains more variables than equations. Model closure is the task of resolving those variables that are outside of the equation set. Among these variables in the described two-fluid model are contributions from large-scale processes like work on the fluid from a pump or energy from a nuclear reactor. It also includes smaller-scale phenomena like wall friction and fluid properties. The principal complexity of a two-phase hydrodynamic model closure resides in the derivation and solution of the constitutive relationships addressing interfacial phenomena. These are commonly resolved by applying (empirically derived) flow-regime-dependent models for interphase friction (drag) and mass and heat transfer. Additional constitutive relations include solutions for the coefficient of virtual mass, wall friction, wall heat transfer and direct (sensible) heat transfer.

Model closure for simulating nuclear reactor transients must include the power response from changes in core reactivity. For most transient analyses, the point kinetics model is considered conservative and thus adequate for safety assessments. The point kinetics model is the simplest model that can be used to compute power behavior in a nuclear reactor. Power is computed using the space-independent or point kinetic approximation that assumes that power can be separated into space and time functions. Eqs. (1)–(5) describe the point kinetics model (Tsvetkov, 2021) used in most thermal-hydraulic system codes, including RELAP5-3D.

$$\frac{dn(t)}{dt} = \frac{\beta_{\text{eff}}}{\Lambda} [\rho_s(t) - 1]n(t) + \sum_{i=1}^N \lambda_i C_i(t) + S \quad (10)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t) \quad i = 1, 2, 3, \dots, N \quad (11)$$

$$\psi(t) = V \Sigma_f n(t) v \quad (12)$$

$$P_f(t) = Q_f \psi(t) \quad (13)$$

$$\rho_s(t) = \sum_i \rho_i(T, x) \quad (14)$$

where t , Time³ (T); n , neutron density (#/L³); C_i , concentration of delayed neutron precursors of group i (#/L³); β_i , delayed neutron fraction of group i ; β_{eff} , effective delayed neutron fraction; Λ , prompt neutron generation time (T); ρ_s , ρ_i , reactivity ((dk/k)/ β_{eff}); λ_i , decay constant of group i (1/T); S , neutron source (#/(L³·T)); v , neutron velocity (L/T); ψ , fission rate (#/T); Σ_f , fission cross section (1/L); P_f , immediate fission power (E/T); Q_f , immediate fission energy per fission (E); V , fuel volume (L³); N , number of delayed neutron precursor groups; T , temperature (Θ); x , spatial position (L³)

The fuel heat generation includes contributions from (1) fission power, (2) power from fission product decay, (3) power from actinide product decay, and (4) power from decay due to neutron capture in core constituents (i.e., neutron activation). Fission power is important only until a short period following reactor trip, vanishing as the reactor becomes subcritical. The remaining decay power terms are commonly calculated based on an ANSI/ANS decay heat standard. As such, total volumetric power ($Q_t'''(t)$) at time t is the summation of the fission power and delayed contributors ($P_i(t)$) divided by the core volume:

$$Q_t'''(t) = \frac{1}{V} \left(Q_f \psi(t) + \sum_{i=1}^N P_i(t) \right) \quad i = 1, 2, 3, \dots, N \quad (15)$$

The simple reactivity model given in Eqs. (10)–(15) is broadly applicable to the analysis of PWR AOOs. In BWR AOOs and accidents, the presence of coolant voids requires at least one dimensional spatial resolution and preferably three-dimensional.

Reactivity initiated accidents (RIA) are a class of design-basis events that require higher spatial fidelity for both PWRs and BWR. Neutron kinetics in RIAs is modeled using either a multi-dimensional diffusion or transport model (Avramova and Ivanov, 2019). At a minimum, the time-dependent neutron flux in these models depends on space and energy. To benefit from this high fidelity model, there must be a proper mapping between thermal-hydraulic control volumes and spatial kinetics nodes. Given the increased

³Title 10 of the U.S. Code of Federal Regulations (CFR), Part 50, Section 46.

modeling requirements, these multi-physics simulations can be computationally challenging. As such, they historically have only been utilized in those scenarios where pronounced 3-D asymmetric characteristics are expected; however, with advances in computing, 3-D multi-physics modeling is becoming more commonplace.

Verification and validation

Nuclear regulatory frameworks require that the sufficiency of transient analysis computer codes be demonstrated through either analysis, test programs and experience, or combination thereof. In addition, this sufficiency must be explained specific to individual phenomena and processes and for the collective or integral effects of these phenomena and processes. Of particular interest are the interdependent effects involving engineered safety features. In all settings, assessments involve experimental data spanning a sufficient range of normal conditions and transient conditions, in addition to accident sequences.

The utility of computational methods is invaluable to nuclear system designers and safety analysts. Given the strong societal interest in the reliability and robustness of these complex systems, the credibility of software prediction used in these applications must be extensively scrutinized. Verification and validation (V&V) are the primary methods for building and quantifying confidence in analytical capability.

Verification is the process of determining that the numerical algorithms are correctly implemented in the computer code and of identifying errors in the software. It involves the confirmation that documented statements accurately reflect the coding and evaluation methodology procedural mechanics. It typically takes the form of a line-by-line review of coding and supporting evaluation documentation, failure-modes-and-effects analysis, and confirmation of compliance with an accepted quality assurance plan. Standard problems, benchmarks with other codes, or analytical exercises with known solutions (e.g., *Method of Manufactured Solutions* (Salari and Knupp, 2000)) are also useful complements to explicit code-to-data comparisons.

Validation is the process of determining the degree to which a model is an accurate representation of the real world from the perspective of the intended uses of the model. Computational tools are validated using an appropriate developmental assessment matrix consisting of both separate and integral effects test program data that address the more important phenomena influencing the ultimate figure(s)-of-merit. The assessment matrix supports the evaluation model development in defining nodalization, quantifying code accuracy, and demonstrating any code or model scaling effects. The principal objectives are to demonstrate sufficient accuracy in modeling dominant physical processes, appropriate nodalization, independence of scale effects, and the relative insensitivity of compensating error.

V&V activities are derived from the same source of requirements used to develop the computer code: the PIRT and scaling analysis. Application of these methodologies provide the modeling and simulation requirements related to the more important physical phenomena and processes. As is the nature of complex systems, a thorough PIRT and scaling analysis contain multiple levels of organization defined by the phenomena and processes that appear and interact at a common scale. A V&V hierarchy is prepared to match equivalent tiers of model theory expressed in computer code predictions and experimental outcomes, such as that shown in Fig. 2 (Oberkampf and Roy, 2010).

This approach divides the complex engineering system of interest, i.e., the nuclear power plant, into progressively more fundamental tiers. The strategy in the tiered approach is to assess how accurately the computational model (i.e., governing equations and closure relations) and application results compare with independently sourced information such as theory manuals, journal articles, and data from test facilities. Individual V&V tests then proceed down and across the hierarchical model. Verification is demonstrated by the alignment of theory and its translation into lines of computer code. Validation is demonstrated by the alignment of simulation outcomes with experiment results. Ohkawa and Ratnayake (2019) provide an extensive summary of datasets compiled for the nuclear industry.

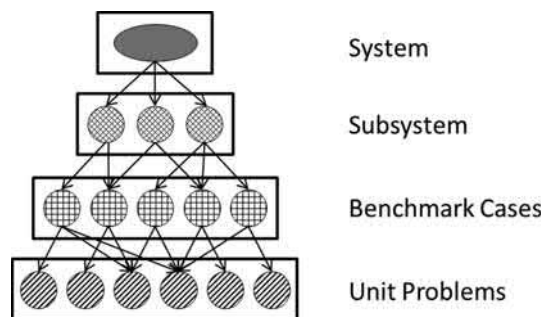


Fig. 2 Validation hierarchy tiers. From Figure 10.7 in Oberkampf WL and Roy CJ (2010) *Verification and Validation in Scientific Computing*. Cambridge, UK: Cambridge University Press.

Deterministic and best-estimate analysis methods

Per Martin et al. (2019b), transient and accident analyses are prepared to determine a nuclear power plant's physical state following equipment failure or malfunction. This involves a logical progression from requirements to inquiry to experimentation to modeling, simulation, and the interpretation of results. Ultimately, this exercise concludes with procedures for quantifying the effective dose from radiation from a hypothetical release of radioactive material. In between resides a mix of deterministic and probabilistic insights into plant design and its nuclear and thermal-hydraulic response to transients and accidents.

Objectivity is essential in an evaluation model. In early years of the nuclear industry, this objectivity was mandated through regulatory requirements explicitly specifying analytical procedures. This was necessary to address large gaps in the nuclear thermal-hydraulic knowledgebase at the time. As this knowledgebase was expanded through experimentation and theory, regulatory emphasis shifted toward uncertainty management. With perfect knowledge, precise evaluation of actual margins relating to the limits and set points would be possible. In practice, even with detailed knowledge of physical phenomena and processes, uncertainties remained. The benefit of this knowledge is that uncertainties are quantifiable and, thus, lend themselves to explicit treatment in evaluation models.

Analysis uncertainty in modeling and simulation has many sources. These include those associated with approximated models that describe the underlying physics; those associated with the settings of parameters used in physical models and correlations; those associated with performing a simulation at a given spatial resolution; and those associated with approximations in the numerical algorithms. Uncertainty quantification is the process of characterizing, estimating, propagating, and analyzing all kinds of uncertainty to support decision-making.

Fig. 3 graphically illustrates the difference in the convolution of uncertainty within deterministic and statistically (or probabilistically) based (i.e., BEPU) evaluation models. Within deterministic evaluation models, key parameters are conservatively bounded, effectively “stacked” upon other conservatisms in a single calculation. “Appendix K” evaluation models, as recognized in the United States, are deterministic. With statistically based evaluation models, key parameters are sampled over an uncertainty range, requiring several calculations.

Since the 1988 U.S. regulatory rule change allowing for statistically-based evaluation models for LOCAs (U.S. Nuclear Regulatory Commission, 2017), there has been notable industry investment in uncertainty quantification methods for thermal-hydraulic models in computer codes for nuclear power plant applications. These methods center upon the evaluation of parametric and system response uncertainties, where system response uncertainties are derived from the convolution of one or more parametric uncertainty models. In this context, uncertainty quantification begins with a model or correlation, data, and a clear qualitative statement that captures the assumptions associated with measured values. The latter element is key in that observations of phenomena will often inherit a bias associated with data acquisition methods. For example, core heat flux datum is a function of the movement of coolant, the fluid properties, the geometry of the core, fuel materials and configuration, the possible stochastic nature of the underlying phenomena, the location and density of instrumentation, calibration of the instrumentation, and post-test data reduction. As with the type of data, such assumptions are not equally important. What is important is that the application of uncertainty models is consistent with the original assumptions.

The outcome from PIRT and scaling analyses serve to identify a manageable set of uncertainty parameters. The quantification process requires that data from separate-effects tests be segregated into control and validation sets. The control set is used to derive the uncertainty and, as the name implies, the validation set is used to validate the integrity of the uncertainty model. A general

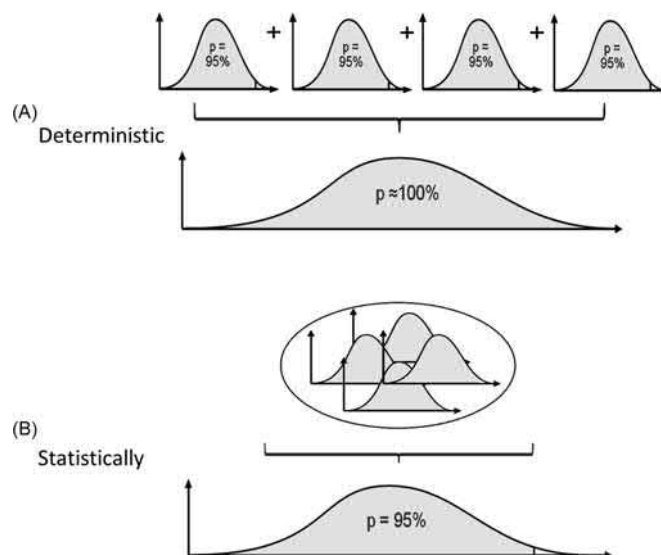


Fig. 3 Contrast between deterministic and statistical convolution of uncertainty.

uncertainty model is characterized by a bias and a probability density function; however, the uncertainty model does not necessarily reflect a task unique to performing a statistically-based analysis. A deterministic analysis can be viewed as one built from uncertainty models consisting of parameter biases. While the bias may simply reflect a static error in a model parameter, it can also be used to define a conservative or bounding treatment based on a limited set of test data.

With the principal elements of the evaluation model completed, meaningful analysis of nuclear power plant transients and accident is possible. Regardless of whether analysis is performed deterministically or using statistically-based methods, the preparation of analysis is similar:

- Gather pre-requisite code input data (e.g., design, fuel/core analysis or plant data)
- Identify code parameters important to analysis measures-of-interest
- Quantify uncertainty treatment for parameter set (e.g., biases and/or probability density)
- Develop base input for the computer codes (i.e., design information)
- For statistically-based (BEPU) evaluation models, build calculation suite from sampling the uncertainty domain
- Perform calculations
- Evaluate parametric sensitivity of figures-of-merit
- Formalize analysis guidance
- Perform analyses of record

Anticipated operational occurrences and postulated accidents

The scope of analyses performed for nuclear power plants is broad. These include (Ergün et al., 2019):

- Reactor coolant system—Characterizes the RCS pressure boundary, a physical fission product barrier and defines the design envelope for thermal-hydraulic conditions during normal operation and design-basis events.
- Fuel design and performance—Characterizes the configuration of fuel and the material, and structural requirements for the fuel rod cladding, a physical fission product barrier.
- Core design and performance—Characterizes the design envelope for fuel materials and reactor physics during normal operation and design-basis events.
- Instrumentation and controls (I&C)—Characterizes the protection and safeguards systems, defined by several monitored plant process inputs, translated as binary signals that result in the actuation of systems serving the critical safety functions of reactivity control, RCS pressure and inventory control, RCS heat removal, containment environment and isolation, containment integrity, and radiation and radioactive effluent control.
- Containment performance—Characterizes the design envelope for the containment, a physical fission product barrier.
- Structural and loss-of-coolant accident (LOCA) loads—Characterizes the design envelope of structural forces on the containment and critical load-bearing and safety equipment.
- Radiological, severe accidents and probabilistic risk assessment (PRA)—Characterizes the design envelope of the radiological source term and the risk and consequences related to the release and transport of radionuclides from the fuel into the environment.
- Auxiliary systems—Characterizes new and spent fuel handling and storage, water systems, process auxiliaries, air handling systems, fire protection and other systems that support the overall mission of plant operation.
- Technical specifications—Captures the design envelope requirements for SSCs supporting critical safety functions from other engineering disciplines to support administrative coordination and controls, including procurement.
- Accident management—Characterizes programs, plans, training, exercises and resources necessary to prepare personnel to operate, maintain and effectively react to off-normal situations, including those arising from terrorism or natural events. It includes emergency preparedness, whose objective is to ensure that nuclear power plant operators can implement measures to protect public health and safety in the event of a radiological emergency.

For LWRs, the DBEs are identified and characterized per methods described in Section “**Design-Basis Event Characterization.**” As noted in that section, analysis categories fall into the following general types: undercooling, overcooling, flow rate transients, an increase in coolant system inventory, a decrease in coolant system inventory, reactivity transients. In addition, consideration is given to what might occur if the reactor protection system fails to respond to these events as intended, generally by providing an automatic reactor scram. Such scenarios, referred to as anticipated transients without scram (ATWS), are analytically conservative and contribute to assuring a robust design.

In the following, several PWR- and BWR-specific DBEs are described under the categories of Anticipated Operational Occurrences and Postulated Accidents, drawn from Buschman and Meholic (2019), Todd (2019) and Avramova and Ivanov (2019).

Anticipated operational occurrences

Anticipated operating occurrences (AOOs) are Condition II events, as such, the regulatory design criteria require that these transients result in no plant damage. A strategy of both prevention and mitigation is incorporated into the design to assure that this objective is met. These include engineered safety features, core monitoring, and the reactor protection system.

Among the six LWR event categories listed above, the most significant AOOs to the thermal designer (limiting transients) are those in the undercooling and power distribution anomaly category. Notably, decreases in coolant system inventory are typically LOCAs, which are postulated accidents. Since core damage during an AOO must be precluded by design, system response to these events is relatively mild and uncomplicated.

With core damage precluded, the emphasis of AOO transient analysis is on the margins to specified acceptable fuel design limits (SAFDLs). The set of SAFDL figures-of-merit depend on the plant design. A list of SAFDLs include the departure from nucleate boiling ratio (DNBR) in PWRs, minimum critical power ratio (MCPR) in BWRs, fuel centerline temperature and core flow (including hydrodynamic flow instability), coolant enthalpy distribution and fuel rod cladding strain. By definition, satisfying the SAFDL (i.e., acceptance criteria) means that no core damage occurs.

Undercooling transients

Undercooling transients are those in which there is a decrease in the rate of heat removal from a failure on the secondary system in a PWR or in the power conversion or coolant feedwater system in a BWR. Typical transients of this nature are loss of load (turbine trip), increase in feedwater temperature, and loss of feedwater flow. In each of these cases, the degradation of secondary-side heat removal capability leads to a rise in the reactor coolant system pressure and temperature.

The loss-of-load transient is the most common of these events. Such events include load rejection, turbine trip, loss of condenser vacuum and inadvertent closing of a main steam isolation valve. The main characteristic of this transient is the sudden reduction in steam flow. This causes a rapid reduction in heat removal from the core, leading to a pressurization of the reactor coolant system. In both PWRs and BWRs, increased coolant temperature will decrease core reactivity and, hence, power. While the reduction in power can be to a level below that of the rate of heat removal, plant protection systems are designed to recognize the conditions of an under cooling event and initiate a reactor trip. The SAFDLs emphasized in analysis are the core DNBR, primary pressure, and steam generator pressure in PWRs.

Overcooling transients

Overcooling transients include those caused by an increase in feedwater flow, decrease in feedwater temperature, excessive load, and an inadvertent opening of a relief, safety, or dump valve. Since these events cause a decrease in the reactor coolant system temperature, both core reactivity and power increase. In addition to an increase in power level, the transient also gives rise to an initial decrease in pressure. A reactor scram may be initiated by high flux levels, high thermal power, or low reactor coolant system pressure. A conservative analysis of this transient is usually accomplished using the maximum moderator reactivity feedback and the minimum fuel temperature feedbacks. The SAFDL emphasized in this analysis is core DNBR in PWRs and MCPR in BWRs.

Core flow anomaly

The principal core flow anomaly DBEs are the loss of flow and hydrodynamic flow instability. In PWRs, the loss of flow DBE is generally the most limiting of all AOOs. Primary coolant flow may be reduced by the loss of a single or multiple reactor coolant pump. In these events, the core flow rate decreases. The reduced primary system flow rate degrades core heat transfer, resulting in increased reactor coolant system temperatures. The PWR SAFDL emphasized in this DBE is DNBR.

In BWRs, operation can become unstable under certain circumstances for neutronic or thermal-hydraulic reasons when operating with a mismatch of low core flow and high core power (March-Leuba et al., 1986). This BWR instability may be realized as a global power oscillation, a power oscillation that occurs regionally within the core (hence total core power varies very little), or as a density wave oscillation within individual fuel assemblies that result in large fluctuation of the assembly flow rate. Power oscillations occur because of reactor feedback from changes in coolant void fraction or density and/or moderator and fuel temperature. Density wave oscillations are primarily driven by fuel assembly pressure drop considerations.

Since in a BWR the coolant channels are isolated, a large void fraction in the core leads to a substantial void-power feedback. During normal operating conditions, the density reactivity and the fuel temperature (Doppler) reactivity are negative, which ensures that the reactor power remains stable. In nuclear power systems, instabilities are possible even with these negative feedback mechanisms, if the feedback gain or the phase lag is increased beyond some critical value. Experiments have shown that BWRs are susceptible to reactivity instabilities when operated at low flow and relatively high power levels (e.g., <30% flow and >50% power). Transient analysis of BWR instability events emphasizes the characterization of operation limits with respect to core flow.

Increase in reactor coolant inventory

The AOOs that lead to an increase in reactor coolant inventory include the inadvertent operation of the ECCS, malfunction of the chemical and volume control system (i.e., for coolant makeup and letdown control) in PWRs or a feedwater malfunction in BWRs.

The more serious concern for both PWRs and BWR is reactor coolant pressure. As coolant inventory increases, so does the reactor coolant system pressure. Like overcooling transient, if the injected coolant is significant cooler than the core inlet temperature, reactor power may also increase. The increase in power can lead to higher reactor coolant pressure. The transient is usually examined to show that the operator has time to end the accident before the pressurize increases above a relief or safety valve setpoint.

Reactivity and power distribution anomalies

The AOO reactivity transient involves the positive addition of reactivity in the reactor core. Consequently, core power and temperature increase either locally or core wide. Transients in this category result from control system failures, control element ejections or uncontrolled withdrawal, and variations in coolant/moderator temperature, void fraction or density. Other specific examples include control assembly misalignment (fail to insert), startup of inactive loop (i.e., overcooling), or a chemical volume control system malfunction, such as the inadvertent dilution of boron concentration in a PWR. The AOO reactivity transient generally involves the slow addition of core reactivity relative to those DBEs considered reactivity accidents. The SAFDL emphasized in the analysis of reactivity transients is enthalpy (energy) deposition in the fuel. Failure to preserve the SAFDL means that the rate of energy increase in the fuel could result in its failure.

Postulated accidents

Postulated accidents are Condition IV events that are not expected to occur during the life of a nuclear power plant. The occurrence of a postulated accident is expected to result in the release of fission products from the fuel or coolant system. The fission product release rate and total release inventory are equated to the potential radiological dose received by an occupational worker or an individual from the general public. The more common postulated accidents considered are drawn from the same general categories identified for AOOs; however, these are dominated by events described as decreases in system inventory, certain loss-of-flow events, and reactivity initiated accidents. In the licensing of nuclear power plants, postulated accidents also extend to fuel handling accidents.

Loss-of-coolant accident (LOCA)

A LOCA is any decrease in system inventory due to inadvertent valve opening or breach of the reactor coolant system pressure boundary. For very minor leaks, the makeup system may be able to make up the loss. If so, there may be no reactor scram. For an inadvertent valve opening, the charging system probably cannot keep pace. The letdown system would be isolated and the event probably terminated by low pressurizer level (in a PWR) or pressure. The class of LOCA events also includes scenarios involving open safety or relief valve that fail to close on reduced pressure.

The most serious accident generally considered in the evaluation of a nuclear power plants design-basis, often referred to as the Maximum Hypothetical Accident, is the double-ended rupture of a main coolant pipe. If such an unlikely accident were to occur, the course of expected events would be (Tong and Weisman, 1996):

- (1) Subcooled blowdown: The system is rapidly depressurized until the vapor pressure of the coolant is reached. The reactor protection system will signal a reactor trip, most likely on low pressure. Voiding in the core region will also contribute to a cessation of fission.
- (2) Saturated blowdown: after system pressure falls to the level of coolant vapor pressure, steam voids are formed. Sonic velocity is drastically reduced, pressure waves are damped out, and choked flow occurs at the break. The length of this period depends on the location and size of the break. During this period, the fuel rods are cooled by fluid in the reactor core. Initially, high forced convection or nucleate boiling heat transfer coefficients prevail. As fluid enthalpy increases, the boiling crisis can be reached, leading to a marked reduction in core cooling. Cladding temperature rise is reduced as the time between blowdown initiation and boiling crisis is increased.
To extend the period during which the core is well cooled, an ECCS is provided. When the primary system pressure falls to a predetermined level, a large quantity of borated coolant water is injected from high-pressure accumulators. In most PWR designs, bottom flooding is used and ECCS water enters the reactor vessel through the downcomer.
- (3) Dry period: after the accumulator inventory is exhausted, continued blowdown leads to a condition where the liquid, or rather the liquid-steam froth, falls below the top of the core. When the froth level is above the bottom of the core, the dry portion is cooled by steam generated in the lower region. When the froth level drops below the bottom of the core, the only core cooling is by radiation to the surrounding structure and by convection and radiation to the small amount of steam generated by froth contact with the vessel wall. During this period, fuel rod cladding temperatures are expected to rise to a level at which the zirconium-based cladding reacts with steam appreciably. The heat produced by this exothermic reaction must be considered in assessing the maximum temperature rise.
- (4) Vessel reflood: after the accumulator injection has stopped and system pressure has been further reduced, high flow pumps of the ECCS system add water to the reactor vessel to recover the core. Since most designs call for bottom-up reflood of the core

region, substantially improved cooling does not result until the coolant level reaches the bottom of the core. The steam produced from quenching the lower portion of the fuel rods flows upward and cools the upper portion of the core. This steam cooling generally limits the maximum temperature reached by the cladding. The water from ECCS pumps gradually refills the core; the transient is terminated when the vessel is refilled. Provision is made for the subsequent addition of water at a low rate to replace that lost through boiloff.

Steam line rupture

Steam line ruptures (SLR) are postulated accidents involving the rupture of a secondary system steam line in a PWR or the main steam line in a BWR. In addition to the breach of the reactor coolant system pressure boundary, they are also considered overcooling transients and the increase in reactivity from overcooling is modeled. The most serious of these accidents is a main steam line break based on the assumption of an instantaneous guillotine rupture.

When an SLR occurs, there is a sudden increase in steam flow. In a PWR, the rapid cooling associated with the SLR occurs via the steam generator. BWRs will cool down from the rapid depressurization of the reactor coolant system. If unmitigated, the profile of the SLR in a BWR aligns with that described previously for a LOCA. In PWRs, the rapid cooling of the secondary system causes the primary system to cool and depressurize until the secondary system completely depressurizes. In either LWR design, the reactor protection system initiates a reactor trip on either low reactor coolant system pressure or liquid level (steam generator in PWRs or the core in BWRs). Closure of the main steam isolation valve close mitigates the blowdown in many scenarios. In PWRs, the SLR is often difficult to distinguish from a LOCA.

Containment response to a LOCA or SLR

LOCAs and certain SLRs directly expel coolant inventory into the containment. The depressurization of the reactor coolant system (or the secondary cooling system in PWRs) in these events can challenge the integrity of the steel and/or concrete containment that encloses the reactor coolant system. As such, the design criteria on containments is that they withstand the loads imposed by LOCAs and SLRs and contain any fission products that escape the reactor coolant system pressure boundary.

The mass and energy leaving the reactor coolant system in both LOCAs and SLR has an immediate effect on the containment pressure and temperature. The impact on the containment is compounded as the air that was resident in the containment warms and expands due to the addition of the hot steam and water. The short-term effect on containment pressure and temperature is mitigated passively by condensation of the steam on the containment walls. As such, peak pressure and temperature occurs very soon after the initiation of the LOCA or SLR.

After the temperature of the structures within the containment equilibrate with the atmosphere, the pressure and temperature stabilize. In the SLR, this is the nominal conclusion of the containment response. In contrast, the recovery from a LOCA can last several days depending on the mitigation strategy. Most long-term cooling scenarios for LOCA mitigation utilize water first from storage water tanks and then switch to water residing in the containment sump. Following the switchover to the sump, water is recycled back to the reactor coolant system. If the sump water is not cooled or if other mechanisms are not used remove decay heat, the continued boil-off of water cooling the core can lead to a repressurization of the containment (Lane and Franz, 2019; Sharpe, 2021).

Reactivity initiated accidents

Reactivity initiated accidents (RIAs) resemble reactivity transients but involve larger and faster reactivity changes. Control rod ejection accidents (REAs) in PWRs and Russian VVERs and control rod drop accidents (CRDAs) in BWRs occur by mechanical failure of the control rod driving mechanism or its housing. Since the reactivity rates and the resulting power transients are much faster than the limits imposed for reactivity transients, these DBEs are classified as accidents. The immediate consequence of a RIA is a fast rise of power and fuel temperatures. The power excursion may lead to failure of fuel rods and the release of radioactive material into the reactor coolant system. The damage mechanisms are all related to high temperature and the degree of fuel damage correlates well with the peak value of fuel pellet specific enthalpy.

As described in section “Design-Basis Event Characterization,” ATWS are unique reactivity initiated events that involve an AOO compounded by a failure in the normal reactor protection system to initiate a timely reactor trip (i.e., scram). This postulated complete failure of the required single-failure proof reactor protection system takes the scenario beyond the definition of DBE. Nonetheless, it is considered a special category of events that must be evaluated with analysis method procedures and acceptance criteria unique to both AOOs and postulated accidents.

Radiological evaluations

Radiological evaluations for DBEs and DBAs are conducted whenever such an event or accident is determined to release radioactivity. The release of radioactivity occurs anytime there is a release of primary and/or secondary coolant (the latter only for

PWRs). The release of coolant involves, as a minimum, the radioactivity dissolved in the coolant. It can also involve activity found in the pellet/cladding gap within the fuel rod (the so-called “gap activity”), activity released from localized fuel melting (due to a localized reactivity insertion), or activity released from generalized fuel overheating and melting due to a sustained loss of core cooling.

The evaluation of radiological consequences involves calculations for post-accident radiological dose to individuals from exposure to sources of radioactivity (Metcalf and Chang, 2019). These evaluations consider shielding, leakage rates, and the transport of fission products offsite. As with SAFDLs or LOCA acceptance criteria, the calculation of the effective dose from radiation is compared to regulatory acceptance criteria to assess the adequacy of dose mitigation features incorporated in a nuclear power plant. Bley (2021) details the safety and licensing requirements and acceptance criteria as they relate to radiological measures.

Licensing considerations related to transient analysis

Transient and accident analysis as described in this chapter demonstrates part of a nuclear power plant’s overall safety mission against the breadth of applicable regulation and regulatory guidance (McIntyre, 2019). The set of events selected to satisfy these requirements define a nuclear power plants design-basis envelope. The design-basis envelope is the collection of the requirements and conditions that serve as the reference point for a design’s safety case throughout the lifecycle of a plant, beginning with the design conceptualization and development, deployment and operation, and, lastly, decommissioning. Organizations involved in the engineering and operations of nuclear power plants establish and implement processes to control the design and design changes of items that are subject to the quality assurance requirements set by national nuclear safety authorities.

Transient and accident analyses are initiated to establish the facility’s hazards and operability characteristics. Credible transients and accidents are identified per methods outlined in Section “Design-Basis Event Characterization” that emphasize a plant’s critical safety functions, distinguishing which structures, systems and components in the plant are to be considered safety-related (and which are not). Among these will be one or more events considered the maximum hypothetical accident (MHA). Safety analyses of the set of transients and accidents and the MHA, in particular, provide the analytical basis supporting the plant’s technical specifications—the statement of relevant limits of operation associated with the design-basis envelope. The principal measure derived from these analyses is the effective dose from radiation, but, as noted in Section “Deterministic and Best-Estimate Analysis Methods,” other thermal-hydraulic measures are typically often used as surrogates to effective dose.

Analysis, along with physical testing, are essential parts of controlling the design-basis envelope. As the lifecycle of a nuclear power evolves, so must the analyses performed, verifying the safety of a nuclear power plant. This means that this set of safety analyses is performed many times both during and following the design phases to account for the refinement and changes to the equipment’s performance parameters and system configurations.

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Containment

Phil Sharpe, Studsvik Scandpower, Inc., Wilmington, NC, United States

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Glossary

Containment building The air-tight structure, which houses a nuclear reactor core vessel, pressurizer, reactor coolant pumps, steam generators, and other equipment or piping that might otherwise release fission products to the environment in the event of an accident. Such enclosures are usually dome-shaped and made of steel-reinforced concrete. The term applies for both PWR's and BWR's.

Decay heat The inherent heat produced by the decay of radioactive fission products after a reactor has been shut down.

Drywell The portion of a BWR containment in the immediate vicinity around the reactor system which accepts reactor coolant discharged during a postulated LOCA and routes the discharge to the suppression pool.

Emergency core cooling system (ECCS) An Engineered Safety Feature system responsible for makeup of cooling water into the reactor core following postulated LOCAs or other occurrences.

Engineered safety Nuclear plant systems, structures, and components designed to mitigate.

Features The consequences of postulated accidents.

Loss of coolant accident(s) (LOCA) Postulated accidents resulting from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

Source term Types and amounts of radioactive or hazardous material released to the environment following an accident.

Suppression pool A large pool of water that suppresses pressure and temperature increase in the free volume of the BWR containment by virtue of water's large heat capacity. The pool also serves as a source of water for long-term core cooling.

Ultimate heat sink The structures, components, and environmental conditions serving to absorb and dissipate plant heat loads after normal or emergency reactor shutdown.

Wetwell The portion of a BWR containment enclosing and above the free surface of the suppression pool that accepts non-condensable gas displaced from the drywell during a postulated LOCA.

Role of containment

Nuclear power reactors are designed and operated with Engineered Safety Features (Lutz and Williamson, 2021) having active or passive attributes aiming to protect against adverse consequences to people and the environment arising from highly unlikely postulated accidents. Radioactive material is continually produced during power generation as fission reactions consume fuel in the core, producing so-called radioactive fission products comprising the source term material which must be retained by physical barriers and Engineered Safety Features of the plant. For standard power reactors presently in service, the solid fuel form serves as a first barrier retaining the radioactive source term, while further material barriers are provided by structural components of the fuel and core- namely the cladding tubes containing solid fuel, and the vessel and piping boundaries of the normally pressurized reactor coolant system. As a standard reactor core operates at elevated temperature and pressure for optimal thermal efficiency and associated plant economic performance, a significant amount of stored energy is naturally held in the fuel, core structure, and coolant. Additional energy from decay heat is produced during power generation and notably persists at a decreasing rate even after fission power generation ceases (reactor shutdown). Management and removal of this stored energy and decay heat during postulated accident sequences determines the design requirements and specifications placed upon several Engineered Safety Features, such as the

Emergency Core Cooling System (ECCS) for controlled removal of core heat when the normal coolant pathway becomes compromised in a postulated Loss of Coolant Accident (LOCA).

The final physical barrier serving also an Engineered Safety Feature required by general design criteria implemented by most nuclear regulatory authorities (Hamon 2021a)—encloses the entire nuclear power reactor system and prevents release of radioactive source term material in quantities having unacceptable effects on people and the environment. This containment barrier normally includes a substantially large and robust structure (the Containment Building) with associated systems functioning to preserve structural integrity throughout postulated accident scenarios. Possible rapid and violent release of stored energy from the core and coolant during a LOCA is a key internal challenge to containment structural integrity, as is longer term (several minutes up to many days) redistribution and removal of decay heat, radioactive source term material, and other reaction products resulting from interacting materials (e.g., hydrogen generation from zirconium metal-water reactions). The containment must handle these energy and material challenges while maintaining an essentially leak-tight boundary to prevent any material release. The Containment Building must also be suitably robust to survive intact many possible hazards from external sources, such as an earthquake, hurricane, tornado, or airplane strike.

Containment structure and system design requirements for nuclear power reactors are devised from and demonstrated by comprehensive engineering analysis and simulation. These steps provide reasonable assurance that any release of source term material remains within acceptable limits established by the regulatory authority. Accident scenarios selected for analysis are established by the reactor system design basis and chosen from postulated yet credible scenarios with severe consequences. For example, energy discharge into containment by postulated design basis LOCA events would inevitably cause temperature and pressure to increase within the enclosure, so that the structure analysis must demonstrate resistance to static and dynamic pressure, temperature, and fluid flow loadings. Eventual cooling of the core and containment by active coolant injection or passive heat removal systems may cause the containment pressure to fall below atmospheric pressure. Cooling water collects in the lower reaches of the containment structure, serving as a sump for possible recirculation into the core or containment via sprays, though perhaps limited somewhat by the available inlet pressure for the associated circulating pumps. Possible relocation of radioactive source term material following core heatup requires measures to clean the containment atmosphere at these varying pressure and temperature conditions. Scrubbing of radioactive material from the containment atmosphere would be performed to reduce amounts potentially leaked to the environment. Removal of excessive hydrogen also requires a specific system designed to alleviate the risk of combustion. All equipment required to perform these safety-related functions must be shown to operate and survive conditions for the event duration, thereby proving so-called equipment environment qualification. Practical plant operation utilizes several penetrations into the containment structure, requiring systems to isolate these openings during a postulated accident, perhaps even incorporating a secondary containment structure. Clearly, the many design requirements for containment and core cooling systems are rather complex and deeply intertwined for mitigating full accident sequences.

Most nuclear reactors deployed worldwide use water for the heat transfer that enables power generation, with the nuclear system either preventing boiling in the core (the Pressurized Water Reactor—PWR) (Brown, 2021) or allowing steam production by boiling in the core (the Boiling Water Reactor—BWR) (Hamon, 2021b). Containment systems for these water reactors have similar functional needs yet have different characteristics to address these needs. A brief overview of these characteristics for PWR's and BWR's are provided herein. Other less common existing or proposed reactor types may use a heat transfer medium other than water, however this article will only consider containments for water-cooled reactor systems.

PWR containment characteristics

The highly energetic discharge of hot ($\sim 288^\circ\text{C}/550^\circ\text{F}$) and pressurized ($\sim 15.5\text{ MPa-g}/2250\text{ psig}$) reactor coolant into containment during postulated LOCA's must be accommodated by the very large free volume surrounding reactor components (vessel, steam generators, pressurizer, piping, etc.). A correspondingly large structure composed of reinforced concrete, several feet thick in places and often sealed with a steel liner, encloses the large volume and includes interconnected rooms. This very large structure is a dominant characteristic of PWR containments that are designed and built to be self-supporting, but also to handle loads by internal and external hazards. Fig. 1A shows a representation of a PWR containment with the large free space around the primary reactor system. In addition to the expansive volume, large internal surface areas are available for condensing the discharged coolant, and generally a spray system is used to cool the containment atmosphere by evaporation of small water droplets. Performance of these heat transfer mechanisms are evaluated with conservative assumptions to ensure containment design temperature and pressure are not exceeded. Clearly the size and robustness of the containment systems and structure are important factors in the total capital cost of a nuclear power plant. Various PWR containment configurations have evolved in an attempt to balance plant costs while meeting specified general design criteria. Most PWR plants have one of four design variants implemented: dry ambient, dry sub-atmospheric, ice condenser, and passively cooled.

Dry containments rely on coolant expansion into the large volume filled with ambient air ($\sim 0.103\text{ MPa}/15\text{ psia}$), or sub-atmospheric held at partial vacuum ($\sim 0.069\text{ MPa}/10\text{ psia}$), to limit the peak atmospheric pressure during all LOCA scenarios, including the challenging scenarios of a break in a reactor recirculation pump suction line or a main steam line. Consider as an example the PWR plant containment as displayed in Fig. 1B. With a rated thermal power of 3645 MWth and dry ambient containment free airspace volume at $7.87\text{E}04\text{ m}^3$ ($2.78\text{E}6\text{ ft}^3$), the characteristic discharge capacity is around $21.6\text{ m}^3/\text{MWth}$ ($760\text{ ft}^3/\text{MWth}$) and peak design pressure is 0.41 MPa-g (60 psig). A representative PWR plant operating at 2940 MWth with

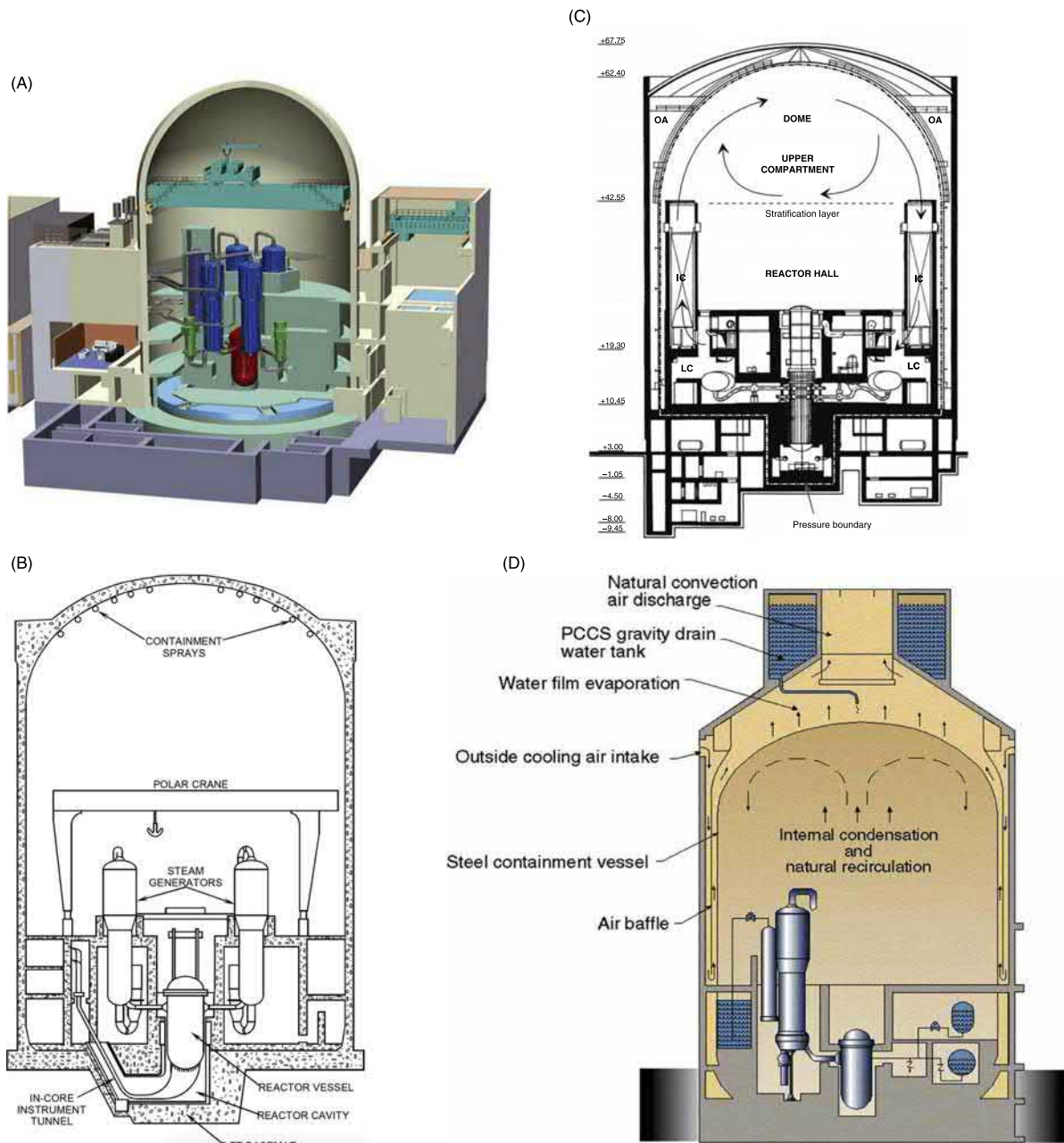


Fig. 1 (A) General PWR containment building layout and relative scale for (B) dry ambient/sub-atmospheric, (C) ice condenser, and (D) passively cooled containments. (A: <https://www.nuclear-power.net/nuclear-power-plant/containment-building/>; B: <https://www.nrc.gov/reading-rm/basic-ref/students/for-educators/04.pdf>; C: <https://www.nuclear-power.net/nuclear-power-plant/containment-building/>; D: <https://www.westinghousenuclear.com/Portals/5/Other%20PDFs/UKP-GW-GL-793NP-sm.pdf>).

a dry sub-atmospheric containment free airspace volume of $5.17\text{E}04\text{ m}^3$ ($1.825\text{E}6\text{ ft}^3$) has a roughly similar yet smaller characteristic discharge capacity of $17.5\text{ m}^3/\text{MWth}$ ($620\text{ ft}^3/\text{MWth}$) and a lower peak design pressure of 0.31 MPa-g (45 psig). A somewhat smaller relative structure size is therefore possible with the dry sub-atmospheric containment compared to dry ambient, however the cost of additional equipment to maintain the partially evacuated atmosphere may offset a portion of the potential capital savings. [Note: the representative values for containment volume, reactor power, and design pressure (expressed in absolute, gauge (g), or differential (d)) presented within this article were obtained from public documents and provided, for example, in the Final Safety Analysis Report of operating plants. See Hamon 2021A for additional information.]

An alternative PWR containment design implemented to further reduce size and cost incorporates beds of ice over which the containment atmosphere flows and cools during a LOCA scenario. Fig. 1C indicates the relative size and annular location of Ice Condenser (IC) equipment in this containment type. Large sections of ice doped with boron (for eventual melting and injection

into the core) are continually refrigerated during normal operation and provide rapid condensation of discharge from postulated LOCA scenarios, thereby reducing possible peak containment atmosphere pressures and temperatures. For comparison, a representative PWR plant rated at 3411 MWth with an ice condenser containment free airspace volume of $3.31\text{E}04\text{ m}^3$ ($1.17\text{E}6\text{ ft}^3$) has a smaller still characteristic discharge capacity of $9.34\text{ m}^3/\text{MWth}$ ($330\text{ ft}^3/\text{MWth}$) but a much-reduced peak design pressure of 0.1 MPa-g (15 psig). Cost savings may be found with reduced structural requirements, but the tradeoff comes by way of a relatively complex refrigeration system. Furthermore, operational challenges may occur during refueling with a relatively smaller space available for maneuvering fuel and equipment.

A more recent PWR containment design approach targets reduced reliance on active equipment by using natural forces for cooling. The passively cooled containment system (PCCS) relies upon gravity to supply a large inventory of water for heat removal rather than pumps requiring electric power. Natural circulation of air on the outside of the containment draws heat out of the structure without use of an active system. Fig. 1D shows the layout of passive components and systems for containment cooling, with the large water inventory in the PCCS water tank situated above the containment structure. The passively cooled containment configuration requires a characteristic discharge capacity of $\sim 17.3\text{ m}^3/\text{MWth}$ ($610\text{ ft}^3/\text{MWth}$) and maintains a design pressure of 0.41 MPa-g (59 psig), similar to the dry ambient containments, but without the associated costs of safety-related active containment cooling system components.

BWR containment characteristics

Nuclear reactors that boil water and generate steam in the core are designed to operate at lower pressure ($\sim 6.89\text{ MPa-g}/1000\text{ psig}$) and directly transport reactor coolant to the steam turbine for power generation. Though somewhat less than a PWR, a similarly rated BWR also retains a tremendous amount of stored energy and decay heat that must be accommodated within containment for postulated LOCA scenarios. Components such as steam generators and a pressurizer are not used in the BWR, thus a comparatively smaller space is needed for reactor components within containment. An alternative approach to the dry containment is implemented in part to further reduce containment size and hence possible savings in plant capital costs. Containment size reduction is achieved by using a large inventory of water connected by a series of flow conduits, or vents, to the free volume surrounding the reactor system—a space generally referred to as the drywell. Heat transfer and steam condensation within the water pool suppresses the maximum pressures and temperatures of the containment atmosphere following a LOCA discharge. Non-condensable gas present in the drywell is pushed into the water suppression pool then passes into the enclosed free space above the pool, referred to as the wetwell. Active heat transfer systems supply pool water to cool the core (ECCS), drywell, and wetwell for longer term removal of decay heat. All modern BWR's utilize some variant of this pressure suppression containment approach, with change to vent configuration and structure being a key difference. Three principal containment variations rapidly evolved throughout the history of BWR plant design, each being developed, tested, and certified by the General Electric Company.

The first variant, referred to as the Mark I BWR containment, incorporates the suppression pool in a large steel tank of toroidal geometry connected by several sizeable pipes to the drywell. Fig. 2A indicates the cross-section of the torus at the lower section of the reactor building, along with the layout of vent pipes connecting to the drywell and the vent ring header within the wetwell. Projecting vertically downward from the ring header and submerged into the suppression pool are numerous downcomer pipes that complete the vent system. The upper drywell region provides a relatively tight fit to the reactor vessel, yet enlargement in the lower region gives space for expansion and routing of LOCA discharge flow into the containment vents. This configuration is colloquially termed the “inverted lightbulb” drywell (perhaps in deference to Thomas Edison, founder of General Electric and generally credited inventor of the incandescent lightbulb!). Fig. 2B shows an isometric perspective for scale and layout of the reactor system, Mark I containment drywell, wetwell, and vent systems, along with the surrounding reactor building serving as a secondary containment boundary. For comparison to PWR containments and BWR Mark II and III containments, a representative BWR power reactor rated at 2923 MWth with a Mark I containment has a drywell free volume of 4644 m^3 ($164,000\text{ ft}^3$), a corresponding wetwell free space volume of 3483 m^3 ($123,000\text{ ft}^3$), and a suppression pool volume of 2492 m^3 ($88,000\text{ ft}^3$ or $\sim 660,000\text{ gal}$). Assuming robustness of structure holding the pool of water is similar to the structure comprising the drywell structure, the total representative Mark I containment volume is $10,620\text{ m}^3$ ($375,000\text{ ft}^3$), giving a characteristic discharge capacity of $3.62\text{ m}^3/\text{MWth}$ ($128\text{ ft}^3/\text{MWth}$), and reflecting the general effectiveness of the water suppression compared to dry containment. Although the structure is overall smaller compared to the PWR containments, the design pressure of the Mark I BWR containment remains quite high at $\sim 0.43\text{ MPa-g}$ (62 psig) for both the drywell and wetwell. The relatively compact size of the drywell and wetwell volumes facilitates use of a nitrogen-inerted atmosphere to reduce the likelihood of hydrogen combustion from metal-water reaction postulated for severe LOCA's. Some additional complexity remains with the structural requirements of the vent system, since LOCA discharge forces result in rather large blowdown, clearing, and quenching/chugging loads from pool water at the exit of downcomer pipes—an area for improvement in BWR containment design evolution that in part led to the Mark II containment.

With increased BWR reactor size and power output, the Mark II containment allows more water volume available for pressure suppression. Fig. 2C shows the Mark II configuration including the tapered drywell overtop a large water pool separated by a concrete deck. The reactor vessel sits high in the drywell on a concrete pedestal that continues through the wetwell and onto the containment building basemat. Rather than a toroidal tank, the suppression pool is held within a cylindrical concrete chamber and fills into the lower interior section of the reactor pedestal. Drywell and wetwell interior surfaces are lined with steel, and the outer boundaries housed within the reactor building serving as secondary containment. The simplified vent system uses many

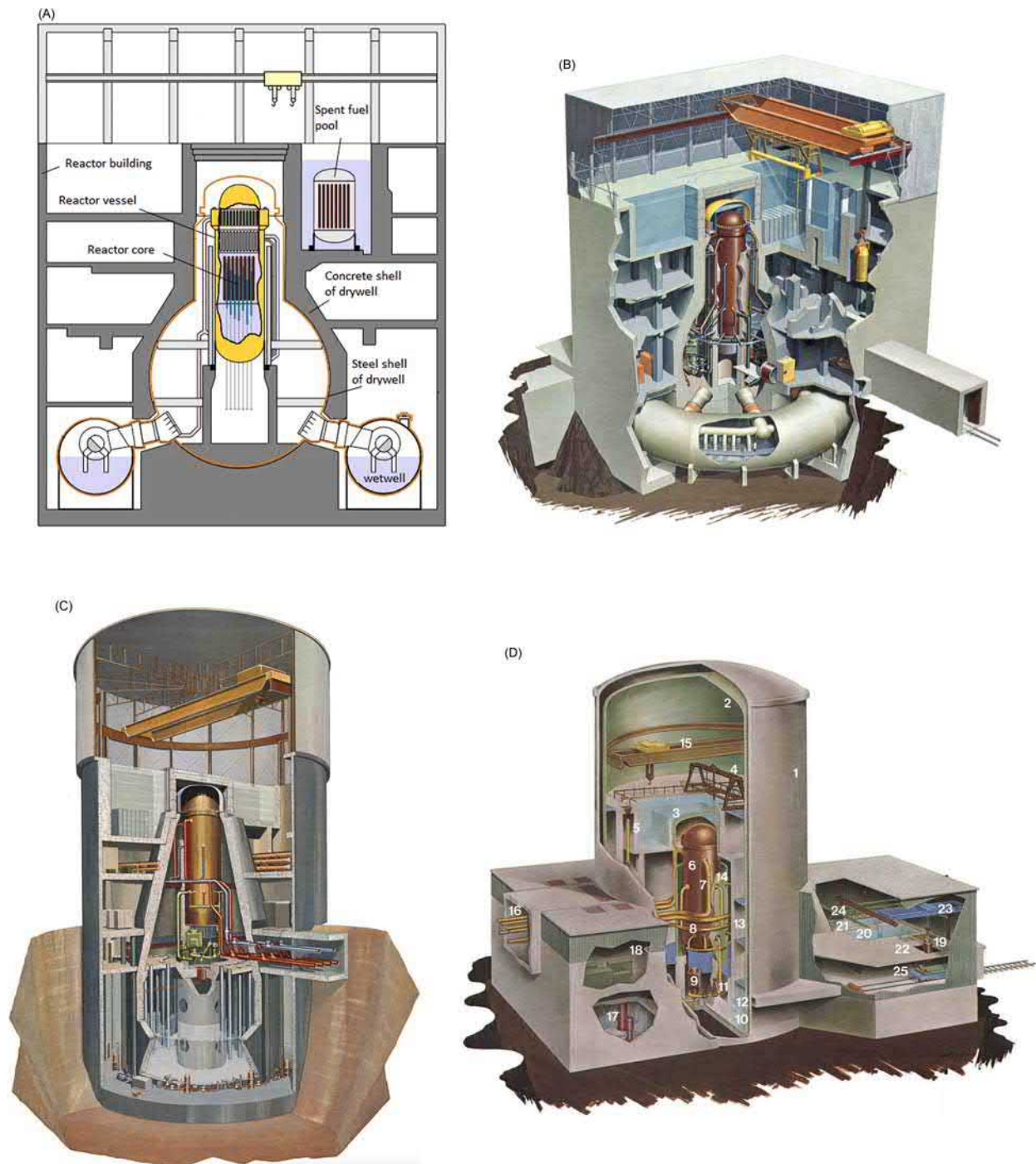


Fig. 2 Pressure suppression containment structure layout and evolution applied to the General Electric line of BWR's. (A) Mark I containment outlining the "inverted lightbulb" shape of the steel-lined drywell and toroidal structure of the wetwell. Reactor and containment component layout for (B) Mark I, (C) Mark II, (D) Mark III BWR containments. (A: <https://www.nuclear-power.net/nuclear-power-plant/containment-building/>; B: <https://www.nrc.gov/reading-rm/basic-ref/students/for-educators/03.pdf>; C: <https://www.nrc.gov/reading-rm/basic-ref/students/for-educators/03.pdf>; D: <https://www.nrc.gov/reading-rm/basic-ref/students/for-educators/03.pdf>).

downcomer pipes penetrating the drywell deck and directly connects the drywell atmosphere to the suppression pool. Thrust and lateral loads are eased somewhat through axisymmetric reactions, although the downcomer length requires additional buttressed support. Like the Mark I containment, the Mark II containment is also inerted with nitrogen gas. A representative BWR plant rated at 3546 MWth has a drywell free volume of 6513 m³ (230,000 ft³), a wetwell free volume of 4700 m³ (166,000 ft³), and suppression chamber water volume of 3680 m³ (130,000 ft³ or ~975,000 gal). The total representative Mark II containment volume is thus

14,895 m³ (526,000 ft³), giving a characteristic discharge capacity of 4.19 m³/MWth (148 ft³/MWth), slightly higher than the Mark I containment. A comparatively lower design pressure of 0.31 MPa (45 psig) is allowed, however, with the caveat that the pressure differential between the drywell to wetwell is no more than 0.17 MPa-d (25 psid), and 0.34 MPa-d (5 psid) from wetwell to drywell, to ensure structural integrity of the drywell deck. Though the simplified downcomers were an improvement, discharge loads at the downcomer pipe exit could see further reduction with a completely different venting concept.

A third, or Mark III, BWR containment design variant evolved in part to relieve the need for downcomer pipes and associated structures in handling LOCA discharge and quenching loads. The Mark III containment also leverages features of the dry ambient containment concept with a very large available free volume but quite low design pressure—attempting to marry the best of both dry and pressure suppression containment approaches. Fig. 2D gives a view Mark III containment with reactor component placement, pool location, and the large containment structure and free airspace. Rather than incorporating header pipes and downcomer vents, the Mark III configuration has the drywell volume interface the suppression pool by way of an annular weir wall extending upward from the basemat and placed just inside the drywell wall. Water from the suppression pool fills the annular gap between the weir wall and drywell wall by three vertical rows of horizontal vent holes bored into and distributed around the azimuth of the drywell wall. The outer perimeter of the annular suppression pool is bound by the containment vessel inner wall and the containment airspace volume above. LOCA discharge mass and energy into the drywell will displace the water in the narrow annulus and push into the horizontal vents, with each row of vents uncovering at different times during the discharge event. With horizontal vents configured in this fashion, discrete structural loads are more smoothly distributed compared to those for a downcomer vent pipe. The containment also houses a large volume of water in an upper pool, along with a large pool for refueling, that may serve as a water source for ECCS or containment spray, but not directly available for pressure suppression. A BWR plant rated at 4408 MWth with a Mark III containment includes a drywell free volume of 6682 m³ (236,000 ft³), a containment free airspace volume of 3.375E4 m³ (1.192E6 ft³), and suppression pool water volume of 3936 m³ (139,000 ft³) (~1000,000 gal), resulting in a total available volume for containment of 4.45E4 m³ (1.57E6 ft³). The resulting characteristic discharge capacity is 10 m³/MWth (355 ft³/MWth)—the largest characteristic size for the BWR containments—and comparable to the PWR ice condenser containment. A distinctive reduction is achieved, however, for the Mark III containment design pressure of 0.10 MPa-g (15 psig), comparatively reducing design requirements for the large containment structure. The drywell design pressure is similar but slightly reduced at 0.28 MPa (40 psig) relative to the Mark II drywell design pressure (0.31 MPa-g/45 psig). Similar to the PWR dry ambient containment, the large free volume of the Mark III containment makes impractical any inerting with nitrogen gas.

Further evolution of the BWR containment has occurred for the Advanced Boiling Water Reactor (ABWR) and the Economic Simplified Boiling Water Reactor (ESBWR), but each implement features similar to those of Mark II and Mark III. Very few ABWR's and no ESBWR's have been deployed to date, thus further consideration is not provided herein for these designs.

Common engineered safety features

Nuclear power reactors require systems that interface or utilize containment features to perform a given safety function, such as cooling the core or maintaining integrity of the containment structure. Similar functions are required for both PWR and BWR containments, and they may be considered generally applicable to each variant. General design criteria for containments are satisfied by Engineered Safety Features implemented in various systems affecting the containment atmosphere, cooling water inventory, and control of radioactive source term material. A few of these systems are briefly described below, with focus on the active systems (requiring electric power and operator action) that are in use at the predominant nuclear power plants.

Containment heat removal systems

Energy released from the reactor core during postulated LOCA scenarios acts over the long term (beyond the blowdown and pressure reduction/suppression period) to heat the containment atmosphere, structure, and components. Removal of this heat requires cooling systems to ensure design specifications are not exceeded. Fan coolers are available to circulate the containment gaseous atmosphere through heat exchangers cooled, for example, by plant service water into the site ultimate heat sink. These fan coolers may also be used during normal operations to sustain containment temperature within specifications. A containment spray system is implemented to actively cool the containment atmosphere with small water droplets that are very effective for evaporative cooling. A source of water for the containment spray system is the suppression pool inventory in BWR's and the containment sump for PWR's (supplied, for example, by condensed reactor coolant, injected makeup water for core cooling, and return of unevaporated containment spray). Residual heat (the combination of stored energy and decay heat) removal systems also provide containment cooling with water sourced from the ECCS or suppression pool.

Source term material control systems

Radioactive source term material transported during postulated accident scenarios that breach the first two barriers (cladding failure and reactor pressure boundary failure) shall be retained with minimal release from containment. Several safety-rated systems work in concert to prevent such release. The containment isolation system acts to prevent leakage from fluid systems by use of isolation valves, interlocks, or other barriers for all pipes which penetrate or pass-through containment except those needed for accident

response. Components of the isolation system are triggered by instrumentation monitoring reactor conditions as well as containment atmosphere conditions and operate during anticipated abnormal conditions as well as postulated accidents. Prevention of long-term containment pressurization is often accomplished with a venting system that may incorporate filtration, allowing greater flexibility for operators responding to accident conditions. Many nuclear plants incorporate a secondary containment structure (often as the reactor building) that may collect and clean source term material that may leak from the primary containment structure. Requirements are established to periodically test leakage from both primary and secondary containments to assure performance of the isolation and venting components. Relocation and removal of radioactive source term material within containment improves performance of these components in retaining the material. This role is performed by containment atmosphere cleanup systems that generally include in-containment recirculation system (fan coolers), stand-by gas treatment systems, and air-cleaning systems for connected rooms. Containment spray, for example, provides an important feature to capture radioactive gases (primarily iodine) and aerosols in the containment atmosphere or wash debris from surfaces within containment upon which radioactive debris has settled.

Combustion control systems

Combustible gases may be generated during postulated accident scenarios where metal-water reactions (primarily from zirconium) generate and release hydrogen gas when core components overheat. In the highly unlikely, yet more severe, postulated accidents (extending beyond design basis accidents), molten core material may interact with concrete in the containment structure to produce combustible hydrogen, carbon monoxide, and oxygen gases. Mixing of combustible gases with oxygen in the containment airspace (for non-inerted containments) may undergo exothermic reactions sufficiently rapid or explosive thereby resulting in containment breach, or more likely, a substantial increase in leakage beyond the capacity of the containment isolation system. Mitigation of possible hydrogen explosions is achieved by deploying controlled hydrogen reaction systems, generally in the form of passive autocatalytic recombiners (PAR) and hydrogen igniters. While these components are effective in locally controlling hydrogen concentration, complications with predictions of combustible gas generation rate, gas mixing, dead-space volumes, and hot surfaces requires careful attention to determine their distribution within containment.

Conclusion

The high degree of importance given to the final barrier protecting people and the environment from postulated accidents at nuclear power plants is reflected in the size, complexity, and associated cost of the containment systems, structures, and components. With Engineered Safety Features, containment is categorized as a safety-related system with correspondingly high attention provided by regulatory authorities in certifying plant construction and operation. A number of key characteristics have been described and compared here, giving a brief introduction to some of the concepts considered, evaluated, and demonstrated in order to achieve regulatory approval. This discussion begins to only scratch the surface of containment performance, with many excellent resources available for further pursuit by the interested reader. Only a few resources are listed below, recognizing there are many, many more excellent treatises available.

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Design of the Balance of Plant

M. Arcaro, Principal Engineer, GE-Hitachi Nuclear Energy-Systems Engineering, Wilmington, NC, United States

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Glossary

Advanced Light Water Reactor Advanced LWR (ALWR) is a reactor, BWR or PWR, which meets the requirements of the ALWR program, U.S. and international utility sponsored effort, managed by the Electric Power Research Institute, to establish the technical foundation for the design of the next generation of nuclear power plants.

Air Handling Units A component in a HVAC system consisting of an inlet area, filters (as specified by the system), heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust).

As Low as Reasonably Achievable As defined in Title 10, Section 20.1003, of the Code of Federal Regulations (10 CFR 20.1003), The acronym, As Low As Reasonably Achievable (ALARA) means making every reasonable effort to maintain exposures to ionizing radiation as far below the dose limits as practical, consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

Auxiliary Boiler Steam System The Auxiliary Boiler System (ABS) consists of package boilers. During plant startup and shutdown and at any other time when insufficient main steam or extraction steam is available, the ABS can provide the necessary steam at sufficient pressure to the various equipment items such as; the feedwater system, to provide hot water during plant startup when decay heat is not present or is insufficient on its own to startup the plant in a timely manner, the Steam Jet Air Ejectors, to maintain the motive power required to perform a continuous evacuation of the non-condensable gasses from the Main Condenser and to the Offgas System in a BWR, the Turbine Gland Sealing System, to allow the Main Condenser to

reach vacuum; Preoperational testing of Offgas System equipment; and evaporation of liquid nitrogen for inserting of the Containment.

Chilled Water System (CWS) Consists of redundant independent subsystems that do not perform safety class functions. The CWS provides chilled water to the cooling coils of air conditioning units and other coolers in the nuclear island and the primary containment. Each subsystem would consist of redundant 100% capacity, independent loops with crossties between the chilled water piping.

Condensate and Feedwater System (C&FS) Receives condensate from the condenser hotwell(s), supplies condensate to the condensate purification system, and delivers high purity feedwater (FW) to the reactor, at the required flow rate, pressure and temperature. The C&FS does not perform, ensure or support any safety-related function, and thus, has no safety design basis.

Condensate Storage and Demineralized Water Transfer System Consists of redundant 100% capacity pumps and lines taking suction from one storage tank that is the normal source of water for makeup to selected plant systems. These systems are used for storage of makeup water and excess condensate rejected from the condensate & feedwater systems and the condenser hot well. These systems do not provide any safety class functions.

Design Basis Accident (DBA) Design basis accidents are postulated accidents used in the design to establish the acceptable performance requirements of the structures, systems, and components.

Emergency Core Cooling System (ECCS) The ECCS shall provide abundant core cooling. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate so that fuel and clad damage will not interfere with continued effective core cooling.

Engineered Safety Features The design and functional requirements of engineered safety features (ESF) of the plant are provided to mitigate the consequences of postulated accidents. The ESF consist of containment systems, core cooling systems, habitability systems, and fission product removal and control systems.

Equipment Drain/Floor Drain Systems (EFDS) Collects waste liquids from their point of origin and transfers them to a suitable processing or disposal system. The Reactor Coolant Pressure Boundary leakage detection systems utilize features of the EFDS to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage.

Evolutionary plant The evolutionary plant is envisioned as a large unit, employing substantial simplifications and improvements in plant safety systems and instrumentation and control systems. It is a direct descendent of today's ALWRs and its safety systems and regulatory base largely follow conventional approaches.

Instrument and Service Air Systems Are compressed gas systems which also include the Containment Inserting System (CIS) and the High Pressure Nitrogen Supply System (HPNSS). The CIS and the HPNSS provide nitrogen gas for instruments and valve operators within the inserted containment. Compressed air operated components having safety-related functions, either have safety-related accumulators or are fail-safe and do not rely on any of the compressed air systems to perform these functions.

Light Water Reactor LWRs are power reactors that are cooled and moderated with ordinary water. There are two basic types: the PWR and the BWR.

Loss of Preferred Power (LOPP) The offsite power system is referred to in industry standards and regulatory guides (RGs) as the "preferred power system." The preferred power supply interfaces with an alternate ac (AAC) power source for safe shutdown in the event of a station blackout (SBO) (non-DBA).

Quality Assurance Quality assurance (QA) comprises all planned and systematic actions that are necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.

Reactivity A term expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Safety function This term applies to systems, structures, components, procedures, and controls (of a facility or process) that are relied upon to remain functional during and following design basis events. Their functionality ensures that key regulatory criteria, such as levels of radioactivity released, are met. Examples of safety-related functions include shutting down a nuclear reactor and maintaining it in a safe shutdown condition.

Seismic Category I Structures, systems, and components that are designed and built to withstand the maximum potential earthquake stresses for the particular region where a nuclear plant is sited.

Seismic Category II Nonsafety-related piping and components that support accident recovery functions.

Standby Liquid Control System (SLC) Provides an alternate method of reactor shutdown (i.e., without control rods) from full power to cold subcritical by the injection of a neutron absorbing solution into the Reactor Pressure Vessel for a BWR. A SLC system would have redundant and independent 50% capacity trains, which include piping, valves, accumulator and instrumentation that can inject a neutron absorber solution into the reactor. The system is designed to operate over the range of reactor pressure to inject sufficient neutron absorber solution to reach hot subcritical conditions after system initiation.

Turbine Component Cooling Water System (TCCWS) Provides cooling water to all turbine island auxiliary equipment to provide a continuous supply of cooling water to the turbine island auxiliary equipment. The TCCWS is designed to permit the maintenance of any single active component without interruption of the cooling function. Heat is removed from the TCCWS and transferred to the Nonsafety- Related PSWS.

Introduction

Balance of plant systems consist of the remaining systems, components, and structures that comprise a complete nuclear power plant and are not included in the nuclear steam supply systems as defined under 10 CFR 170.3 Definitions for nuclear plants licensed in the US or equal for international design. LWR NSSS are described for PWR in (Brown, 2021) and BWR (Hamon, 2021).

These systems utilized in nuclear power plants include Fuel Storage and Handling Systems, FAPCS, RCCWS, PSWS, DCS, Control Room Habitability HVAC systems and Radiation Monitoring systems.

There are numerous nuclear island and non-nuclear island auxiliary systems, such as instrument and service air, condensate/feedwater and demineralized water transfer, chilled water, TCCWSs, HVAC, equipment drain, floor drain, fuel handling and cranes, makeup water systems, potable water and sanitary waste systems, sampling systems, SLC, auxiliary boiler steam supply, water chemistry systems, communications, lighting, standby electrical power systems, fire protection and other systems which are similar between LWR technology BWR and PWR plants and are not covered here, since the designs are essentially the same as non-nuclear counterparts, and for space reasons, are not further described in this article.

Balance of plant systems are separate from power generation systems of nuclear plants that include main/extraction steam, feedwater and condensate systems, chemical addition system, condensate makeup purification system, auxiliary steam system and passive plant PWR backup feedwater system.

Balance of plant system designs utilized at nuclear plants have similarities between current BWR and PWR designs, ALWR plants, Small Modular LWR (smLWR) and evolutionary plants. The similarities lead to designs which incorporate numerous common requirements.

Balance of Plant systems may support safety-related or nonsafety-related functions based on the design of the plant. Safety-related applies to systems, structures, components, procedures, and controls (of a facility or process) that are relied upon to remain functional during and following design basis events. Their functionality ensures that key criteria, such as levels of radioactivity released, are met. Examples of safety-related functions include shutting down a nuclear reactor and maintaining it in a safe shutdown condition. Advanced and evolutionary nuclear plants are being designed to minimize safety-related functions fulfilled by balance of plant systems.

Fuel storage and handling

Nuclear reactor plants include facilities for the storage of new and spent fuel assemblies. The safety functions of the new fuel storage vault, new fuel storage racks, spent fuel pool, and spent fuel storage racks are to maintain the fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the spent fuel assemblies into shipping or storage casks.

Nuclear reactor plants include facilities for storage of new and spent fuel. The new fuel storage facility includes the fuel assembly storage racks, the concrete storage vault that contains the storage racks, and the auxiliary components. The spent fuel storage facility includes the spent fuel storage racks, the spent fuel storage pool that contains the storage racks, and the associated equipment storage pits.

For BWRs, upon receipt of the new fuel bundles at the reactor site, each fuel bundle container is normally uncrated from its shipping crate and the fuel bundle container moved to the refueling floor. The fuel bundles are removed from the container and moved to a new fuel inspection stand where the fuel bundles are inspected and the fuel channels are installed to create fuel assemblies. For PWRs, the fuel assemblies are generally removed from the shipping container, into the fuel inspection stand, and then directly to designated new fuel storage or directly to the spent fuel pool.

The fuel assemblies would then be placed in the spent fuel pool until time to move them into the reactor. The fuel assemblies are normally stored in underwater storage racks. The racks are accessed using a refueling machine.

Spent fuel immediately removed from the reactor vessel must be stored underwater for decay heat removal. Spent fuel racks are used for long-term storage of spent fuel. These spent fuel storage racks are located at the bottom of the storage pools at a depth sufficient to provide adequate radiological shielding.

Another aspect of spent fuel storage is dry fuel storage. Nuclear plants typically reserve space within the site security fence for dry fuel storage. The area reserved shall be sufficiently large to accommodate a storage facility with a capacity for storage of spent fuel elements for up to a 60-year period of plant operation. Because of the potential for extension to an 80-year period of plant operation, consideration is given for reserving an area for a dry fuel storage facility with a capacity for storage of spent fuel elements for an 80-year period of plant operation.

The spent fuel storage pool water is processed by a FAPCS, which provides cooling to the spent fuel and maintains the spent fuel storage pool water quality.

New fuel storage

New fuel storage racks contain storage space for new fuel assemblies.

Nuclear design basis

The new fuel storage racks are designed to assure that the fully loaded array is subcritical by a predetermined margin.

The biases between the calculated results, experimental results, and the uncertainty in the calculation, are taken into account as part of the calculation procedure to assure that the specific limit is met.

Material considerations

Material used in the fabrication of the new fuel storage racks is normally limited to the use of stainless steel in accordance with the latest issue of the applicable American Society for Testing and Materials (ASTM) specifications at the time of equipment order.

Safety evaluation

Criticality control

The design of the new fuel storage racks provides for **reactivity** limits for storage conditions. To ensure that design criteria are met, the following normal and abnormal new fuel storage conditions are normally analyzed:

- Racks are loaded with fuel of the maximum fuel assembly **reactivity**;
- Normal positioning in the new fuel array; and
- Eccentric positioning in the new fuel array.

New fuel storage racks criticality control meets the requirements of 10 CFR Part 50.68(b) for nuclear plants licensed in the US or equal for international design.

Structural design

The new fuel storage racks are designed to meet **Seismic Category I** requirements. Stresses in a fully loaded rack do not exceed stresses specified by the applicable American Society of Mechanical Engineers (ASME) Codes and Standards when subjected to seismic loads.

The storage rack structure is designed to withstand the impact resulting from falling fuel assemblies and the fuel grapple, which is the portion of the fuel handling machine that interfaces with the fuel assembly handle.

Procedural fuel handling requirements and equipment design limit the number of assemblies that can be handled over the storage racks at any one time, typically only one. The structural arrangement is such that no lateral displacement of the fuel occurs; therefore, subcritical spacing is maintained. An irradiated fuel assembly is not to be placed in a rack specifically designated as new fuel storage.

The racks are fabricated from material specified to ASTM or equivalent international standards.

Protection features of the new fuel storage facilities

The new fuel storage racks are housed in a **Seismic Category I** structure designed for natural phenomena such as tornadoes, tornado missiles, floods and high winds.

A refueling machine is used to move fuel. It contains interlocks to prevent overloads from being applied to the bail of the fuel assembly. Thus, the transfer devices used for new fuel handling to the new fuel rack cannot impose excessive uplift loads on the rack.

Should it become necessary to move major loads along or over the pools, administrative controls require that the load be moved over the empty portion of the pool and avoid the area of the new fuel racks. Procedures are written for fuel handling.

Spent fuel storage

Additional details associated with nuclear spent fuel storage are provided in for more details see ([Glatz, 2021](#)) and ([Kessler, 2021](#)).

Design bases

On-site storage is provided in the spent fuel pool for unconsolidated spent fuel assemblies resulting from a minimum of 5 years of plant operation plus the total number of assemblies in one core (typical loading) (NUREG-0800 Standard Review Plan-9.1.2; US Nuclear Regulatory Commission, Revision 4—March [2007](#)). In addition to this fuel storage, space shall also be provided in the pool for storage of control elements, fuel channels, equipment for inspecting fuel and channels and leak testing equipment ([Young, 2021](#)).

Nuclear design basis

New and spent fuel storage racks are capable of maintaining fuel subcritical. A full array in the loaded spent fuel rack is designed to be subcritical by a predetermined margin. Neutron absorbing material (ex. borated stainless steel in accordance with ASTM A-887 ([ASTM, 2020](#))), as an integral part of the design, is normally employed to assure that the calculated **reactivity** coefficient including biases and uncertainties, does not exceed the limit under all normal and abnormal conditions.

The biases between the calculated results and experimental results, as well as the uncertainty involved in the calculations, are taken into account as part of the calculative procedure to assure that the specific **reactivity** coefficient limit is met.

Criticality control

The spent fuel storage racks are designed to assure that the fully loaded array is subcritical by a predetermined margin.

Because of the complexity of the configuration, Monte Carlo neutron transport techniques are employed in the calculations performed to assure that the **reactivity** of the worst case configuration does not exceed the predetermined margin of subcriticality under all normal and abnormal conditions, including consideration for the exposure-dependent **reactivity** of the fuel assemblies.

The biases between the calculated results, experimental results, and the uncertainty in the calculation, are taken into account as part of the calculative procedure to assure that the specific k_{eff} limit is met. Spent fuel storage racks criticality control meets the requirements of 10 CFR Part 50.68(b) for nuclear plants licensed in the US or equal for international design.

Structural design and material compatibility requirements

The spent fuel pool is normally a reinforced concrete structures with a stainless steel liner. Fuel storage racks and pool liners are designed to meet **Seismic Category I** requirements. Pool liner and anchorage are designed to the same loads and load combinations as the pool concrete structure except that load factors for all cases are equal to 1.0, and the acceptance criteria follow ASME Section III, Division 2, CC-3700 (ASME, 2019). Pool liners are evaluated to ensure structural integrity under fuel handling accidents.

The fuel racks support axially the weight of the fuel assembly or bundle. Individual racks are spaced less than one fuel assembly apart so that a fuel assembly cannot be inserted between racks. In the event that a fuel assembly is lowered adjacent to an exterior rack, this configuration is analyzed.

The support structure allows sufficient pool water flow for natural convection cooling of the stored fuel and allows the rack material temperatures to stay within limits.

Following the Fukushima-Daiichi accident, advanced and evolutionary nuclear plants are designing spent fuel pools with sufficient pool volume adequate to accommodate a loss of normal pool decay heat removal for a period of time (typically 72 h) without a need for additional makeup while satisfying the shielding requirements for the fuel. The spent fuel pool cooling system will be analyzed to ensure that for all postulated accidents, the site boundary dose will be below regulatory limits.

The racks are fabricated from materials specified in accordance with the latest issue of applicable ASTM specifications, or international equivalent, at the time of equipment order.

The racks are designed to withstand the impact force generated by the vertical free-fall drop of a fuel assembly and its handling tool from the maximum height expected during normal fuel handling.

The rack is designed to withstand a pull-up force in the event a fuel assembly is stuck.

The fuel storage pools are designed to have adequate water shielding for protection against the neutron and photon radioactivity emitted from the stored spent fuel. The depth is such that even when fuel assemblies are raised above the storage racks containing other fuel, there is adequate depth of water to provide this shielding.

The structures which contain the fuel storage equipment, including the storage racks and pool, are designed to protect the fuel from damage caused by:

- Natural events such as earthquake, high winds and flooding; and
- Mechanical damage caused by dropping of fuel assemblies, bundles, or other objects onto stored fuel.

Summary of radiological considerations

By adequate design and careful operational procedures, the design bases of the spent fuel storage arrangement are satisfied. Thus, the exposure of plant personnel to radiation is maintained well below regulatory limits and in accordance with ALARA principles.

Fuel and auxiliary pools cooling system (FAPCS)

Nuclear reactor plants include a spent fuel pool for the storage of spent fuel assemblies. The methods used to provide cooling for the removal of decay heat from the stored assemblies vary from plant to plant, depending upon the individual design. The safety function to be performed by the system in all cases remains the same; that is, the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions. Other functions performed by the system but not related to nuclear safety include water cleanup for the spent fuel pool, refueling canal, refueling water storage tank, and other equipment storage pools; means for filling and draining the refueling canal and other storage pools; and surface skimming to provide clear water in the storage pool.

A second function of the FAPCS is to maintain visual clarity and purity of the spent fuel pool cooling, safety pool (passive plants), and refueling cavity fill water. The typical system consists of independent subsystems, each with pumps, piping, a heat exchanger, filter and demineralizer capacity and associated equipment and controls for regulating the system flow. Typically, only the cooling function is safety-related for evolutionary plants. Both the cooling and maintenance of clarity functions are non-safety-related for passive plants.

Legacy spent fuel pool cooling systems utilized at PWR and BWR nuclear plants utilize active cooling systems requiring (Alternating Current) (AC) power. Power is supplied from either on-site or offsite sources. The cooling system consists of appropriately redundant equipment so that a single active component failure will not result in a loss of normal cooling. Passive FAPCS have been specified for advanced and evolutionary nuclear designs that do not include the provision for safety-related AC power. In addition,

the spent fuel storage pool will be designed with a pool volume adequate to accommodate a loss of normal fuel pool decay heat removal for a period of time (usually 72 h) without the need for AC power while satisfying the shielding requirements of the fuel.

Power generation design basis

FAPCS provides continuous cooling and cleaning of the spent fuel storage pool during normal plant operation. It also provides occasional cooling and cleaning of various pools located inside the containment during normal plant operation and refueling outages.

System description summary

The FAPCS consists of physically separated cooling and cleanup trains. Redundancy and physical separation are provided for active components in lines dedicated for supporting pool cooling functions. Each train typically contains a pump, a heat exchanger and a water treatment unit for cooling and cleanup of various cooling and storage pools. The FAPCS includes high point vents and component vents necessary to avoid gas accumulation.

The primary design function of FAPCS is to cool and clean pools located in the containment and nuclear island structures during normal plant operation. FAPCS provides flow paths for filling and makeup of these pools during normal plant operation and during post-accident conditions, as necessary.

Instruments are made for indication of operating conditions to aid the operator during the initiation and control of system operation. Provisions are provided to prevent inadvertent draining of the pools during FAPCS operation by including anti-siphon holes on FAPCS piping that is normally submerged.

The FAPCS is designed to provide for the collection, monitoring, and drainage of pool liner leaks from the spent fuel pools and auxiliary pools to the Liquid Waste Management System.

The FAPCS piping and components that are required to support safety-related or accident recovery function have Seismic I or II classification. A Seismic I classification is required for all safety-related functions. The remaining nonsafety-related piping and components that support accident recovery functions are **Seismic Category II**.

System operation

FAPCS cooling and cleanup trains operate continuously to cool and clean the water in the spent fuel pool during normal plant operation and refueling outages. Operation of only one FAPCS cooling and cleanup train of redundant train designs is sufficient to handle the cooling requirements under the normal heat load condition in the spent fuel pool. Operation with up to two FAPCS cooling and cleanup trains is sufficient to handle the cooling requirement under the maximum heat load condition. At least one FAPCS cooling and cleanup train is available for cooling the spent fuel pool, except for a short period as long as the water temperature in the pool remains below the maximum temperature limit for normal operation.

During a refueling outage, FAPCS can be operated in the Fuel and Auxiliary Pool Cooling and Cleanup mode with both cooling and cleanup trains under the maximum heat load condition in the spent fuel pool.

Plant service water system (PSWS)

The PSWS provides essential cooling to safety-related equipment and may also cool nonsafety-related auxiliary components used for normal plant operation. The ultimate heat sink is the intake source of water to the PSW for long-term cooling of station features required for plant shutdown and also of any special equipment required to prevent or mitigate the consequences of postulated accidents and as such is an PSW interface system. The ultimate heat sink and plant cooling water system designs are required to operate under extreme environmental conditions such as extremes in ambient temperatures and drought conditions. Systems utilizing water conservation measures are applied to advanced and evolutionary nuclear plant designs.

BWR and PWR plants have many similarities in their cooling water systems. The similarities lead to designs which incorporate numerous common requirements, i.e., the basic systems utilize the same types of pumps, heat exchangers and piping assembled into similar system configurations.

Power generation design bases

The PSWS provides the required cooling to the RCCWS and TCCWS heat exchangers.

The PSWS is typically designed so that neither a single active nor single passive component failure results in a complete loss of nuclear island cooling and/or plant dependence on any safety-related system. This is achieved by redundant components, automatic valves and piping cross-connects for increased reliability.

The PSWS is designed for ease of restoration of its function after a component failure without plant operating mode or power level change.

The PSWS is designed to operate during a Loss of Preferred Power (LOPP). Typically, independent on-site power supplies are specified to provide a high availability of electric power to the PSWS.

The PSWS is designed for remote operation from the main control room.

Summary description

The PSWS rejects heat from RCCWS and TCCWS heat exchangers to the environment. The source of cooling water to the PSWS is site specific with the heat rejection facilities dependent on actual site conditions.

Operation

The PSWS operates during startup, normal power operation, hot standby, cooldown, shutdown/refueling, and LOPP.

Operation of individual redundant PSWS pumps is sufficient for the design heat load removal in any normal operating mode. During normal and LOPP cooldown mode sufficient redundancy is provided in system design to provide for operational convenience to bring the plant to cold shutdown condition in limited time (normally 24 h).

During a LOPP, the running PSWS pumps restart automatically using power supplied by the nonsafety-related standby diesel-generators.

Reactor component cooling water system (RCCWS)

Service water shall not be used directly for equipment cooling due to the issues of fouling and corrosion caused by raw service water. Therefore, water from the closed loop component cooling water system shall be utilized to provide a barrier between contaminated and raw water systems and reactor component heat loads.

The RCCWS provides a closed loop of cooling water for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the ECCS required for safe shutdown during normal operations, anticipated operational occurrences, and accident conditions and for prevention or mitigation of the consequences of accidents.

Service water also supplies the Turbine Building Closed Cooling Water System that typically cools chillers, turbine lube oil coolers, feedwater pumps, condensate pump motors, booster pump motors, main generator cooling and air compressor cooling loads.

Power generation design bases

The RCCWS provides cooling water to plant auxiliary equipment during startup, normal power operation, hot standby, and plant cooldown. The RCCWS is also designed to operate during a LOPP.

The RCCWS is designed so that neither a single active nor credible single passive component failure results in a complete loss of active nuclear island cooling and/or plant dependence on any safety-related system. Typically, this requirement is addressed through redundant trains of RCCWS.

The RCCWS is designed for ease of restoration after a single component failure without plant operating mode or power level change. Typically, this requirement is addressed through redundant trains of RCCWS or multiple components arranged in parallel. The RCCWS is designed to limit leakage of radioactive or chemical contamination to the environment. Typically, this is accomplished by heat exchanger design that is not susceptible to cross contamination (ex. Plate heat exchanger) and design of leak detection methods to preclude unmonitored release of radioactive liquids to the environment.

Summary description

The RCCWS provides cooling water to components in the nuclear island and provides a barrier against radioactive contamination of the PSWS.

The RCCWS operates during startup, normal power operation, hot standby, normal cooldown, shutdown/refueling, and LOPP.

RCCWS heat exchanger operation is coordinated with PSWS flow. RCCWS cooling water flow through a RCCWS heat exchanger is only allowed if there is a corresponding PSWS water flow to absorb the heat load.

Control room area ventilation system (CRAVS)

HVAC systems utilized for ventilation and cooling functions at nuclear plants provide a suitable environment ensuring the safety and comfort of plant personnel and operability of plant equipment during normal operating conditions (passive and evolutionary plants) and postulated DBA conditions (evolutionary plants).

The CRAVS provides a controlled environment for the comfort and safety of control room personnel and assures the operability of control room components during normal operating, anticipated operational transient, and DBA conditions. Portions of the CRAVS also may be relied upon to ensure coping with and recovering from a station blackout event.

For accident conditions where the CRAVS is available, the system shall be designed to transfer from its normal operating mode either to its isolation/recirculation mode upon detection of conditions that could result in induction of airborne hazardous chemicals or smoke in the control room, or to its pressurization mode (through the supplementary filter unit) to mitigate accidental exposure of control room personnel to a high level of airborne radioactivity.

The control room ventilation system and control building layout and structures ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gasses and to assure conformance with the requirements of specific design criteria and regulations.

The CRAVS maintains space design temperatures, quality of air and pressurization. It provides a controlled environment for personnel safety and comfort, and for the proper operation and integrity of equipment.

- The CRAVS will meet the acceptance criteria of General Design Criteria 5, Sharing of Structures, Systems, and Components (Appendix A to Part 50—General Design Criteria for Nuclear Power Plants, 2021) for sharing of structures, systems, and components for sites with multiple reactors:
 - Each unit at a multi-unit site has a separate control room;
 - An accident in one unit does not impair the ability to perform safety functions of the remaining units; and
 - An orderly shutdown and cooldown of remaining unit(s) would not be impaired.
- The CRAVS meets the acceptance criteria of General Design Criteria 19, Control Room (Appendix A to Part 50—General Design Criteria for Nuclear Power Plants, 2021) for the control room. The control room ventilation is isolated on loss of normal AC power or if high radioactivity is detected with a safety-related system automatically initiated to pressurize and provide radiologically filtered air to the control room area. Radiation detectors in the intake air supply provide warning and initiate actions to protect control room personnel under accident conditions. This safety related ventilation system design maintains a habitable control room under accident conditions by providing adequate radiation protection and breathing air.
- The CRAVS meets the acceptance criteria of General Design Criteria 60 (Appendix A to Part 50—General Design Criteria for Nuclear Power Plants, 2021) for control of releases of radioactive materials to the environment. The CRAVS meet the guidance of RG 1.52 as related to design, testing and maintenance criteria for atmosphere cleanup system and normal ventilation system air filtration and adsorption units.

Advanced and evolutionary nuclear plants have applied passive means to meet control room habitability requirements including pressure integrity and temperature control. For accident conditions if the main control room HVAC system is not operational (e.g., if AC power is not available), a safety grade main control room pressurization system shall provide a positive air pressure in the control room so that the main control room can be maintained as the primary location from which technical personnel can safely operate. The main control room can also be designed with the capability to provide suitable temperature conditions for room heat loads generated in the control room during loss of AC power events during the duration of the AC loss of power event (normally assumed for 72 h).

Safety design bases

The CRAVS provides the following safety-related design basis functions:

- Monitors the control room area air supply for radioactive particulate and/or iodine concentrations; and
- Isolates the normal CRAVS air supply, starts an ESF fan, and aligns the air supply through the safety-related filter unit, upon a high radiation detection signal in the control room area normal air supply, or upon an extended loss of AC power.

The portions of the CRAVS which penetrate the control room area habitability envelope are safety-related and designed as **Seismic Category I** to provide isolation of the space from the outside and surrounding areas in the event of a DBA. The ESF portion of the CRAVS subsystem is safety-related and designed and supported as **Seismic Category I** including the air intakes, ductwork, dampers, fans, instrumentation and controls. The remaining CRAVS functions are nonsafety-related. The penetrations contain safety-related isolation dampers or valves that fail closed upon a loss of control signal, power, or instrument air. An ESF unit is automatically actuated upon radiological isolation of the control room area habitability envelope or an extended loss of AC power. If the initial ESF unit fails to start or is otherwise unavailable, the second standby ESF unit automatically actuates.

CRAVS equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed as **Seismic Category II**. The remaining portion of the system is nonsafety-related and nonseismic.

Power generation design bases

The CRAVS:

- Provides a controlled environment for personnel comfort and safety. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality as defined under ASHRAE 62.1, Ventilation for Acceptable Indoor Air Quality (62.1, Ventilation For Acceptable Indoor Air Quality, 2019);
- Provides a controlled environment for the proper operation and integrity of equipment in the control room area during normal, startup and shutdown operations;

- Provides redundant active components to increase reliability, availability and maintainability of the ventilation system;
- Provides shutoff dampers on the inlet and outlet of fans and **Air Handling Units (AHUs)** if necessary to allow for maintenance;
- Provides shutoff valves at the inlet and outlet of cooling coils if necessary to allow for maintenance;
- Provides access for AHU fans, filter sections and duct mounted dampers to allow for maintenance;
- Provides the capability for manual control of system fans to facilitate maintenance and testing;
- Minimizes exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of high radiation;
- Maintains higher than atmospheric (positive) pressure to minimize the infiltration of outside air. Construction materials and processes that ensure the control building structure maintains low leakage or leak tight conditions above and below grade. The CRAVS envelope penetrations are sealed and access doors are designed with self-closing devices that close and latch the doors following use. Normally, there are double door airlocks in the control room area habitability envelope for access and egress during emergencies when the CRAVS is isolated and an ESF unit is operating;
- Reduces the potential spread of airborne contamination by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination. The CRAVS is maintained at a higher pressure than surrounding areas except during the isolation and smoke exhaust modes;
- Detects and limits the introduction of airborne hazardous materials (radioactivity or smoke) into the control room habitability area;
- Provides the capability to exhaust smoke, heat and gaseous combustion products from inside the control room area to the outside atmosphere in the event of a fire. Construction materials and processes ensure that materials of construction are non-combustible and heat and flame resistant wherever possible. Materials that produce toxic or noxious vapors when subjected to a fire are avoided;
- Smoke control and removal functions are in accordance with National Fire Protection Association guidelines;
- Is designed such that failure of nonsafety-related equipment does not compromise or otherwise damage safety-related equipment; and
- Is provided with bullet-resistant exterior HVAC openings that penetrate the control room vital envelope in accordance with 10 CFR 73 (10 CFR 73, Physical Protection of Plants and Materials, 2021b).

Drywell and containment cooling system

The drywell (BWR) or containment cooling (PWR) systems provide the functions of an auxiliary and radwaste area ventilation system to maintain ventilation, permit personnel access, and control the concentration of airborne radioactive material in the auxiliary and radwaste equipment areas of a nuclear plant during normal operation and anticipated operational occurrences, and after postulated accidents.

The containment design for a BWR or PWR includes a drywell cooling or containment cooling system to maintain temperatures during normal operation within acceptable limits for equipment operation. Additional details associated with BWR and PWR containments is contained in [Sharpe \(2021\)](#).

Safety design bases

The drywell or containment cooling system has no safety-related function, and thus, has no safety design basis. The drywell or containment cooling system shall be designed as a nonsafety, nonseismic system powered from the permanent nonsafety distribution system. A separate safety-related system to remove heat from the reactor containment under DBA conditions is also provided. Advanced and evolutionary reactor designs employ passive heat removal systems to fulfill this function.

Power generation design bases

The drywell or containment cooling system performs the following functions during stable and transient operating conditions through the entire operating range, from startup to full load condition to refueling:

- Maintain temperature in the drywell/containment spaces within specified limits during normal operation;
- Accelerate drywell/containment cooldown following a LOPP during the period from hot shutdown to cold shutdown;
- Aid in complete purging of nitrogen from the drywell (BWR) during shutdown;
- Maintain a habitable environment for plant personnel during plant shutdowns for refueling and maintenance.
- Limit drywell/containment temperature during LOPP.

Summary description

The drywell or containment cooling system maintains the thermal environment within the drywell/containment to specified conditions during normal reactor operation, hot standby and refueling using Fan Cooling Units (FCUs) typically.

The drywell or containment cooling system is a closed loop recirculating air/nitrogen cooling system with no outside air/nitrogen introduced into the system except during refueling. The system uses direct drive-type FCUs to deliver cooled air/nitrogen to various areas of the upper and the lower drywell/containment. Ducts distribute the cooled, recirculated air/nitrogen through diffusers and nozzles. The drywell heat loads are transferred to the cooling medium circulating through the cooling coils of the FCUs.

The cooling medium of the FCUs is chilled water or reactor component cooling water. There are typically separate FCUs for the upper and the lower drywell/containment regions to optimize space cooling.

PWR and BWR containment design differences lead to differences in drywell cooling and containment cooling system design:

- BWR drywell volumes are typically smaller than PWR containment structures requiring an inserted nitrogen atmosphere for combustion gas control under an accident. This requires the DCS to operate in a nitrogen environment compared to an air atmosphere for the PWR,
- PWR containment design pressures and allowable leakage rates are typically greater than those allowed in BWR designs.

Radiation monitoring system

Criteria exist in 10 CFR 50 that require the control, monitoring, and radiological evaluations of all releases, documentation and reporting of all radiological effluents discharged into the environment. The NRC requires power reactor licensees apply ALARA principles to radioactive effluents for the protection of the public. The ALARA criteria contained in Appendix I of 10 CFR Part 50 are:

“The licensee shall establish an appropriate surveillance and monitoring program to:

1. Provide data on quantities of radioactive material released in liquid and gaseous effluents...;
2. Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation doses to individuals from principal pathways of exposure; and
3. Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure.”

To meet these requirements, nuclear plants employ radiation monitoring systems of two basic types: process radiation monitoring and area radiation monitoring. With the former, certain plant processes are monitored in order to detect radioactivity in excess of acceptable limits. These vary somewhat, depending on the type of reactor in which they are employed, because the processes vary between those reactor types. For instance, in BWRs, the main steam line is monitored in the steam tunnel in order that gross fuel failures may be detected. High radiation levels may initiate a reactor trip. The detectors are generally ionization chambers, sensitive to gamma radiation, and are linked to the reactor protection system. Additional monitoring is performed on the offgas treatment system, the offgas vent pipe, the liquid radioactive waste treatment system and the containment ventilation discharge. Limits are set so as to protect against releases to the environment exceeding those determined to be acceptable during plant licensing.

An example of a processing radiation monitoring application for a PWR is the steam generator blowdown sample monitor which typically monitors the liquid phase of the steam generators for radioactivity indicative of a primary-to-secondary system leak. The detectors are gamma scintillation detectors, mounted in liquid samplers surrounded by lead shielding.

The second form of radiation monitoring is area monitoring. For this, gamma-sensitive detectors, with several sensitivity ranges to match expected radiation levels, are distributed throughout the plant. The data are fed back to recorders in the control room. Levels are also indicated locally, and will have audible indication of exceeding allowable limits, for areas that are frequently occupied. These area monitors are intended to protect plant personnel.

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Safety of Nuclear Reactors

Nicholas R. Brown and Seok Bin Seo, Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

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Glossary

AOO Anticipated operational occurrences
ASME American Society of Mechanical Engineers
ATWS Anticipated transients without scram
AVR The German reactor: *Arbeitsgemeinschaft Versuchsreaktor*
BDBA Beyond design basis accident
BWR Boiling water reactor
CCF Common cause failures
CCFP Conditional containment failure probability
CDF Core damage frequency
DBA Design basis accident
DVI Direct vessel injection
ECCS Emergency core cooling system
ESF Engineered safety feature
ESFAS Engineered safety features actuation system
FFTF Fast flux test facility
GEM Gas expansion modules
HPSIS High pressure safety injection system
HTTR High-temperature engineering test reactor
LCO Limiting conditions for operation
LOCA Loss of coolant accident
LPSIS Low pressure safety injection system
LSSS Limiting safety system settings
LWR Light water reactors
MSR Molten salt reactor
MSRE Molten salt reactor experiment
NPP Nuclear power plant
NRC Nuclear regulatory commission
PPS Plant protection system
PWR Pressurized water reactor
Reactivity The extent of the deviation in the core effective multiplication factor (usually denoted as k_{eff}) from its value at a just critical state (at which state $k_{\text{eff}} = 1.0$ and the fission rate is constant). A positive (negative) reactivity implies that the ratio of the rate of neutron production to the rate of neutron losses (k_{eff}) is larger (smaller) than 1.0 due to a change in a performance characteristic such as temperature or density.
RHRS Residual heat removal system
RIA Reactivity insertion accident
RPS Reactor protection system
RTS Reactor trip (immediate shutdown) system
RWST Refueling water storage tank
SAFDL Specified acceptable fuel design limits

SARRDL Specified acceptable radionuclide release design limits
 SHRT Shutdown heat removal tests
 SIT Safety injection tank
 SMR Small modular reactor
 TRISO Tristructural isotropic fuel

Reactor safety principles

Although nuclear energy is very safe compared to other forms of electricity generation, with an excellent operational record approximately 1/1000 of casualty rates per terawatt-hour generated from fossil fuel (Ritchie, 2020), several reactor accidents have progressed beyond the design basis, becoming high-profile severe accidents. These events have had significant effects both from actual consequences and also from impact on the public perception of nuclear energy. In particular, Three Mile Island in Pennsylvania, Chernobyl in the former Soviet Union, and Fukushima Daiichi in Japan, have had a negative impact on the public perception of nuclear energy. See Skillman and Rempe (2021), Sich (2021), and Mizokami and Rempe (2021), respectively, for more complete discussions of these accidents. These major accidents were preceded by other historic events, like the partial meltdown at Fermi-1 in Michigan. Fermi-1 was widely chronicled in the 1975 Readers Digest book “We Almost Lost Detroit.” Later, the rebuttal, “We Did Not Almost Lose Detroit” noted the utility stressed that the incident at Fermi-1 was well within the plant’s safety envelope, even though it was built at a time when the regulator in the United States was not independent. Additionally, the accident was due to the last-minute addition of a component meant to enhance safety. This was an early example of how accidents, even without significant consequences to the public, can have a significant impact on public perception.

Safe reactor operation, as well as the culture of safety within all aspects of the nuclear fuel cycle from mining to final disposition, is key to the successful application of fission energy. The definition of reactor safety has evolved over the years. Historically, the first “critical piles” were operated in an ad hoc manner, with relatively few safety procedures. For example, the safety protocols at the US Army-operated Stationary Low Power Reactor 1 (SL-1) were only a few pages long. This reactor was the site of the deadliest nuclear accident resulting in the first immediate fatalities from nuclear energy in the United States. A maintenance crew of the reactor operator, the shift supervisor, and a trainee passed away due to the sudden steam explosion caused by a reactivity accident. The fuel ultimately melted down, as well, though this aspect of the accident did not cause additional casualties. This was all initiated by an error made by operational personnel. Similar errors contributed to other events, like the partial fuel meltdown at the 105 KW reactor (Sturges et al., 1955) in Washington state or the fire at Windscale in the United Kingdom, as detailed in Rempe (2021b).

These early life-and-death incidents helped transform the safety culture of the overall nuclear power enterprise to become more rigorous and proactive. For example, Admiral Hyman Rickover had developed a more rigorous safety culture within Navy program in contrast with the Army-operated SL-1. Similarly, the Three Mile Island accident revealed the concomitant risk of small equipment failures and human factors that can result in severe accidents, and thus emphasized the importance of operator training, emergency planning, plant operating histories, and other matters. More recently, the Fukushima Daiichi accidents have further emphasized the role of human factors in the emergency, even for the reactor designs that were safe, and emphasized the importance of *passive, sometimes also called inherent, safety* in reactor and reactor system design. The value proposition of inherent safety was demonstrated by the success of the sodium-cooled Experimental Breeder Reactor II (EBR-II) shutdown heat removal tests (SHRT) (McFarlane, 2021). These rigorous series of tests were instrumental in showcasing the potential for a particular reactor concept to be inherently safe in the postulated scenarios that were tested.

In general, testing and validation regimes are a vital part of reactor safety. A validation hierarchy exists where separate effects tests are used to understand phenomena and validate individual models, integral effects tests support multi-physics understanding, and engineering demonstration tests where these principles are demonstrated in an actual reactor. Analysis methods validated with actual data, along with an independent regulator and associated regulatory oversight, are key tenets of the modern reactor safety philosophy.

A key objective of reactor safety is to ensure there is no unacceptable release of fission products or other radioactive elements to plant personnel and especially to the public during all anticipated operational occurrences (AOOs) or design basis (postulated) accidents (DBAs) (IAEA, 2006), as discussed in Rempe (2021a). The key in AOOs is to ensure that there is no fuel damage and survives the event to generate energy in the future. The reactor safety objective in a DBA is to maintain a coolable geometry and avoid the release of any radioactive elements. If an AOO, DBA, or other event progresses outside of the design basis, to a beyond design basis accident (BDBA), the objective of reactor safety is to contain and limit any release of radioactivity to personnel within the *site boundary* and especially to the public beyond the site boundary.

Most AOOs or DBAs are mainly a problem of mitigating power-cooling mismatches due to overpower, under-cooling, or both. There are seven categories of events typically considered:

1. Increase in heat removal by the secondary system
2. Decrease in heat removal by the secondary system

3. Decrease in reactor coolant system flow rate
4. Reactivity and power distribution anomalies
5. Increase in reactor coolant inventory
6. Decrease in reactor coolant inventory
7. Radioactive release from a subsystem or component

Licensing for Nuclear Power Plants (NPP) requires a safety analysis report under 10 CFR 50 (U.S. NRC, 2011), that analyzes the design and performance of structures, systems, and components that have a safety function, and their adequacy for the prevention and mitigation of the consequences of such events. For details, see Rempe and Williams (2021). NUREG-0800 (U.S. NRC, 1987) and other Regulatory Guides discuss a subset of the transient and accident events that are within the scope of DBAs of a NPP, and are considered acceptable for use in developing and assessing evaluation models (Gauntt, 2021) for conservative safety analysis of DBAs.

For a more reliable framework towards risk-informed regulation, the NRC has defined two surrogate variables, core damage frequency (CDF) and conditional containment failure probability (CCFP) that are quantitative measures of reactor safety in Regulatory Guides for the regulation and licensing of NPPs. These variables are used to benchmark reactor safety performance against two quantitative measures of short term and long-term risk, respectively:

1. The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 % (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
2. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 % (0.1%) of the sum of cancer fatality risks resulting from other causes.

Performance issues are defined as risk significant if they lead to an increase in CDF that is greater than 10^{-4} reactors per year. Risk significant issues involving core damage which lead to large, rapid, and unmitigated releases of radioactivity have the potential to cause deaths within the general populace before they can be evacuated (U.S. NRC, 2002). This risk is relevant to the first quantitative measure. The second quantitative measure is related to the longer-term risk of cancer from radioactive releases from the reactor.

The key to attaining these objectives is the principle of *defense-in-depth* where several layers of defense in the system serve as a credited barrier to fission product release (IAEA, 2000). Typically for many of the existing nuclear power reactors, these layers consist of the fuel matrix itself, the cladding, the coolant pressure boundary (in a pressurized system) or the primary coolant boundary (in a low pressure system), and the containment (primary or secondary). This concept is shown as an example in Fig. 1. Various reactor concepts under development may use different layers of defense-in-depth or so-called functional containment. One example of a functional defense-in-depth layer is the integral coolant system layouts of small modular reactors (SMRs) which aim to strengthen the first and subsequent layers by incorporating inherent and passive safety features. Regulatory bodies have identified potential opportunities and challenges in the features and the application of defense-in-depth for SMRs

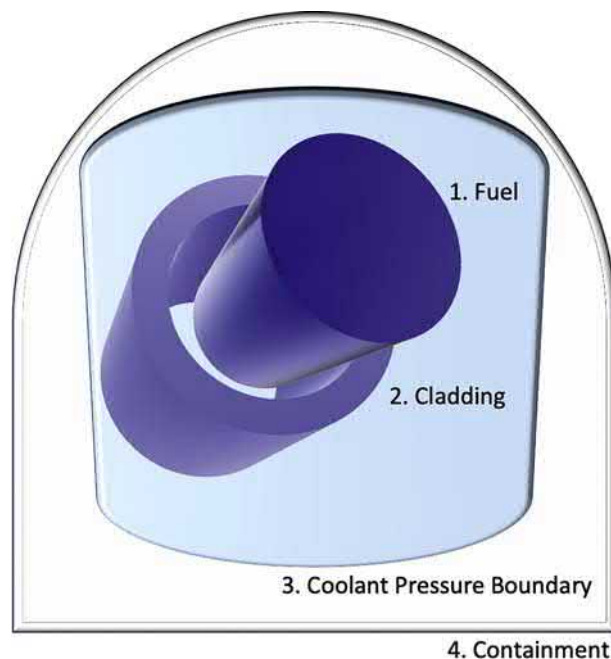


Fig. 1 A conventional defense-in-depth approach where the fuel matrix, cladding, coolant pressure boundary (or primary coolant boundary), and containment provide barriers to the release of radioactive fission products, actinides, and other transmutation products.

(IAEA, 2018). Another example of a defense-in-depth layer in an advanced reactor concept is the use of the SiC-layer in tristructural isotropic (TRISO) fuel within a modular high-temperature gas-cooled reactor (mHTGR), which acts as a very small pressure vessel to contain fission products (Helmreich, 2021; Gerczak, 2021).

Layers of defense-in-depth can be uncredited, in which case they help to prevent the release of radioactivity but are not considered in the safety case, or credited, in which case they are an essential component of the reactor licensing case. Typically, in power reactor licensing the Standard Review Plan (SRP), NUREG-0800 may be applied prescriptively and deterministically to evaluate the safety of a particular light water reactor with uranium fuel and zirconium-based cladding. In this example, NUREG-0800 is used to evaluate against safety criteria regulations in the U.S. code of federal regulations, 10 CFR 50, relating to power reactor safety. While the methods in NUREG-0800 are prescriptive, use of NUREG-0800 is not, in fact, mandatory. Multiple examples of compliance with 10 CFR 50 exist that do not, in fact, conform to the methods provided in NUREG-0800.

The non-power reactor SRP, NUREG-1537 (U.S. NRC, 1996), provides an alternative *performance-based* approach to defense-in-depth, where demonstration of safety performance is acceptable. A performance-based approach is useful when a particular reactor technology cannot rely on the historical precedent of many decades of operational experience, and instead needs to demonstrate its performance via operation. For example, mHTGR TRISO fuel performance requires statistically significant measurements of demonstrated reactor safety performance over the range of operational conditions, AOOs, and DBAs. More recently, the Regulatory Guide for non-LWRs (U.S. NRC, 2019) addressed *risk-informed and performance-based (RIPB)* approaches that appropriately account for risk significance in establishing technical and regulatory requirements while increasing effectiveness and efficiency.

Many risk significant events are by definition caused by a breach of the credited layers of defense-in-depth. For example, fuel cladding, the coolant pressure boundary, and the containment boundary are all barriers to fuel and fission product release in a typical light water reactor.

The design principles of reactor safety are important technical requirements in the design of safety-related systems. These design principles (IAEA, 1999) include proven technology, redundancy, diversity, physical separation, independency, fail safe, and interlock principles, as defined below.

- (i) Principle of proven technology: The functional performance and the safety of all systems and components should already be proven by commercial usage in other plant types or industrial fields. One example is the application of the ASME Boiler and Pressure Vessel Code (ASME Code) to nuclear power plant construction, providing requirements for manufacturer certification and quality assurance for the design, materials, manufacturing, inspection, testing, and operation of boilers and pressure vessels. If not, the usage of the system component should be tested following the guidelines from the regulation agency.
- (ii) Principle of redundancy: The design of engineered safety systems requires the extra provision of more than the minimum number of components for a specific safety function. Such redundant provisions allow a safety function to be satisfied when one or more items are unavailable, due to a variety of unspecified potential failure mechanisms or maintenance.
- (iii) Principle of diversity: The diversity of system requires more than two system components operating for the same objective but in different means. The diversity is achieved by the use of different physical means or different equipment made by different manufacturers. The purpose of diversity is to prevent the unavailability of systems caused by the Common Cause Failures (CCFs).
- (iv) Principle of physical separation: The system components operating for the same functions should be physically separated by locating them far from each other or having physical protections between them, to minimize the CCFs or other unavailability caused by external sources.
- (v) Principle of independence: Each safety-related component should provide a single specific function and avoid being used for more than one system, to reduce the uncertainties in the system and potential for CCFs.
- (vi) Principle of fail safe: The design of main components should inherently lead to the reactor system safety, even if the components fail to operate as intended.
- (vii) Principle of interlock: Some safety features should be locked to operate or stop only if certain specific conditions are satisfied.

Inherent safety features

Inherent safety features are the first level feature to ensure nuclear reactor safety during the abnormal operation of a NPP. The key concept of inherent safety features is to recover the nuclear reactor system from abnormal operation by inherent feedback of physical phenomena.

Feedback can be either prompt and inherent or delayed and inherent, depending on the event in question. Inherent safety doesn't require a specific system component, but is achieved by unique physical features of the system. Inherent feedback is based on an unavoidable and totally reliable physical phenomena (e.g. thermal expansion in fast reactors). Inherent feedback is prompt if it immediately follows the prompt fission energy deposition: in particular, if it is directly related to the temperature or physical dimensions of the nuclear fuel where fission fragments are generated. Inherent feedback that is delayed requires an additional physical phenomenon, such as heat transfer to the coolant.

The inherent safety features can be different among the types of nuclear reactors, while the representatives are Doppler broadening, moderator temperature coefficient of reactivity, and void coefficient of reactivity, in Light Water Reactors (LWRs). All three of

these are inherent feedback, with Doppler broadening being prompt feedback and moderator temperature or void coefficient being delayed feedback. All three of these coefficients contribute to the overall power coefficient of reactivity.

- (i) Doppler broadening (i.e. fuel temperature): As the increase of reactor power results in the increase of fuel temperature, it broadens the resonances in the neutron absorption cross section of fertile material (e.g. ^{238}U) due to the thermal motion of target nuclei in the nuclear fuel. The broadened resonances increase the parasitic neutron absorption probability and consequently results in prompt negative reactivity feedback.
- (ii) Moderator temperature coefficient of reactivity: As the temperature of moderator increases, its density decreases resulting in the reduction of neutron slowdown. In thermal reactors, the decrease of neutron slowdown increases the probability of parasitic neutron absorption in the resonances on ^{238}U and reduces the probability of fission in thermal neutron region, giving negative reactivity to the system. There is also a spectral impact of the change in water temperature on cross sections in the thermal energy range.
- (iii) Void coefficient of reactivity: As the increase of the moderator (water for LWRs) temperature leads to boiling, the voids (steam for LWRs) are produced inside the moderator. The formation of voids reduces the neutron slowdown by the moderator even more than due to moderator temperature increase resulting in even more negative reactivity feedback.

Among these example feedbacks, the fuel temperature coefficient is the main prompt feedback mechanism. Doppler broadening provides the shortest possible feedback loop. Doppler broadening occurs simultaneously with the deposition of kinetic energy from fission products. The Doppler coefficient is typically calculated as a temperature coefficient ($\partial\rho/\partial T$ in which ρ denotes reactivity). Moderator temperature and void coefficients of reactivity are examples of delayed feedback mechanisms, which are initiated by the heat transferred from the fuel to the coolant. Inherent safety features of some advanced reactor concepts are discussed in the last section.

Operational modes of NPPs

In general, normal operation of a NPP refers to the operation within the power operating range. However, there are also other operational modes in the category of normal operation. Each operational mode is defined as each combination of core reactivity condition, power level, and average reactor coolant temperature. For example, the operational modes of the Westinghouse Advanced Passive 1000 (AP 1000) plant include power operation, startup, hot standby, hot shutdown, cold shutdown, and refueling. However, the operational modes can differ for various NPP designs. Here, we briefly describe the representative operational modes of the existing PWRs as follows. More information on the reactor operation is discussed in [Young \(2020\)](#).

- (i) Power operation: A critical operation with a power over 5% of its full-power capacity is defined as power operation mode by NRC. During the power operation mode, a power cycle is established from a nuclear fuel where heat energy generated from nuclear fission reaction is transferred to a primary coolant, then to a heat sink where the energy is transferred to a secondary system for generation of electricity (e.g. a steam generator for a Pressurized Water Reactor [PWR]). The primary concern during the power operation is to keep the core critical. Typically, the reactivity control is achieved by using control rods and soluble poisons. The insertion or withdrawal of control rods is for the prompt reactivity control, while the long-term reactivity control is achieved by the soluble poisons (e.g. boron) in the primary coolant and the burnable poison rods. The entire systems including main control systems (e.g. reactor control system, feedwater control system, etc.) and auxiliary control systems (e.g. chemical and volume control system, etc.) are automatically controlled during the power operation mode in most of the operating NPPs.
- (ii) Startup: The startup of a NPP begins from a subcritical state and includes hot standby, hot shutdown, or cold shutdown states. When the nuclear reactor is shut down due to minor component or instruments malfunctions, and re-operates within a few hours, the startup of reactor is initiated from a hot standby state whose system conditions are close to those of power operation mode. Therefore, it takes a relatively short time, about an hour, to reach the full-power level after startup from hot standby. However, it takes longer, over 12 h, to reach the full-power level after the startup from cold shutdown in the case of refueling.
- (iii) Cooldown: Cooldown is a process leading the nuclear reactor from zero power state to cold shutdown state. The first step of cooldown is the heat removal from the primary system by the steam generator. The circulation of primary coolant is achieved by forced flow by pump or natural circulation inside the system. If the pressure and temperature of primary coolant fall below the limit, the heat removal of primary system is achieved only by the Residual Heat Removal System (RHRS). Usually, it takes a minimum of about a day for the temperature of primary coolant to reach that of the refueling condition. During the cooldown of reactor, the boron concentration inside the primary system is raised to compensate the positive reactivity feedback caused by the decrease of temperature.

Example reactor safety systems

The Plant Protection System (PPS) provides prevention and mitigation of reactor from exceeding safety limits during AOO or postulated accidents in the reactor. The PPS consists of Reactor Protection system (RPS) and Engineered Safety Features Actuation system (ESFAS).

- (i) Reactor protection: The RPS, also known as the reactor trip system (RTS), monitors system variables related to the reactor safety and automatically shuts down the reactor if one or more of those variables reach the certain point, referred to as the trip setpoint. The trip setpoints of RPS are set not to exceed the safety limits of the reactor system for AOOs. Example safety limits for a PWR typically include a local heat flux limit on the fuel surface (e.g. Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio), a limit on linear heat generation rate (i.e. Maximum Linear Heat Generation Rate, MLHGR), a limit on system pressure less than 110% of design value, and so on. The heat flux limit and linear heat generation rate limit are deterministically Specified Acceptable Fuel Design Limits (SAFDL) that should not be exceeded during any condition of normal operation including the effects of AOOs (U.S. NRC, 2011). For an advanced reactor with a performance-based approach the equivalent concept is the specified acceptable radionuclide release design limits (SARRDLs). For example, the safety case for TRISO fuel forms is typically performance-based and is focused on the definition of appropriate SARRDLs, as opposed to deterministic SAFDLs. Within the criteria of safety limits, Limiting Safety System Settings (LSSS) is set as a conservative limit to initiate the RPS during the design basis transients of reactor. Finally, for the commercial operation of NPP, one should follow the Limiting Conditions for Operation (LCO) defined by technical specification.
- (ii) Engineered safety feature actuation: When the reactor system variables approach the relevant safety setpoints, the ESFAS generates an actuation signal for an Engineered Safety Feature (ESF) (Lutz and Williamson, 2021) to mitigate the risk from the accidents below the acceptable criteria. Typical actuation signals generated by ESFAS for a PWR include a safety injection, containment isolation, containment spray, recirculation, and emergency actuation signals, and main steam isolation signal.

ESF refers to the safety systems necessary to prevent and mitigate the consequences of postulated accidents in the nuclear reactor. Typical ESFs for a PWR include core cooling system, containment system, residual heat removal system, emergency feedwater system, fission product removal and control system, habitability system, and so on. A detailed description of main ESFs for PWRs are described as follows:

- (i) Emergency core cooling: The Emergency Core Cooling System (ECCS) provides fuel cooling by supplying emergency coolant into the core during the Loss of Coolant Accident (LOCA). The main purpose of an ECCS is to keep the integrity of fuels by removing stored energy and decay heat in the core during LOCA, which mitigates progression to a severe accident. In a PWR, the ECCS consists of a High Pressure Safety Injection System (HPSIS), a Safety Injection Tank (SIT), and a Low Pressure Safety Injection System (LPSIS), which operate independently. The HPSIS provides emergency coolant into the core from a Refueling Water Storage Tank (RWST) using pumps with low flow and high head, while the LPSIS uses pumps with high flow and low head for enhanced operation in low pressure conditions. The SIT injects borated coolant through the cold leg if the system pressure decreases below the pressure of nitrogen gas filling the upper volume of accumulator.
- (ii) Containment system: The containment system consists of the containment building, containment isolation system, containment heat removal system, and combustible gas control system. The containment building provides physical barrier to prevent the release of radioactive products during the normal operation and postulated accidents. The containment isolation system isolates the containment by closing penetrated pipes during the accidents. The containment heat removal system aims to keep the integrity of containment by reducing inner pressure and temperature. The representative system is a containment spray system located at the upper part inside the containment, which sprays coolant pumped from RWST or containment sump. The combustible gas control system, or hydrogen control system, prevent the containment building from the potential damage by hydrogen explosion.
- (iii) Residual heat removal: The RHRS operates to remove the decay heat from the core and reduce the temperature of the reactor coolant to the cold shutdown temperature. The RHRS is typically a multiple use system with modes of operation for low pressure injection, shutdown cooling, suppression pool or containment sump cooling, and/or containment spray.
- (iv) Emergency feedwater: The decay heat removal from the primary system requires a heat sink, which is usually a steam generator cooled by feedwater. When the main feedwater system doesn't operate for some reason (e.g. loss of external power), the emergency feedwater system provides auxiliary feedwater to the secondary side of steam generator from a condensate storage tank or demineralized water storage tank.

Safety features of advanced reactors

The improvement of safety for the advanced reactors is introduced in a large variety of reactor designs including Generation III+ LWRs, mHTGRs, liquid metal reactors, and molten salt reactors, among other examples. These advanced reactor concepts have an enhanced focus on inherent safety characteristics relative to older Generation I, II, and III plants. The objective of inherent safety is to minimize the utilization of safety-related active systems and components (IAEA, 2007). In this section, the basic principles of safety and some examples of improved safety systems of the advanced reactors are briefly introduced. More detailed information is discussed in Stanculescu (2021).

The design of an advanced reactor should first satisfy the fundamental principles of safety discussed for the current-generation reactors (IAEA, 1999). Two principles may be considered as the basic requirements for the design of an advanced reactor. First, the defense-in-depth can only be complemented or operated by specific safety design features, but cannot be replaced by them. Second, the basic concepts for advanced reactor safety should include the control of the nuclear fission power and decay power, cooling of

the reactor in all scenarios, and containment of radioactive material, during the design, construction, and operation. In addition, the extended concepts of safety features are proposed to improve the safety of advanced reactors (IAEA, 1992). They include extension of plant design processes to include maintenance, simple and intuitive user-friendly design, reduced dependence on operator intervention, improvement of system reliability even in severe accident conditions, removal of high-risk event sequences, effective defense strategies from intentional damage or attack, reduction of uncertainty in safety and operational margins, and employment of passive safety systems. Based on these highly desirable extended safety features, various designs of advanced reactors have been developed. A passively safe advanced reactor is one whose design employs passive systems to function as the main safety systems in the reactor.

The main feature of advanced safety systems is the inherent operation by unavoidable natural physical phenomena. The definition of an inherent, or passive safety system is a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation. The employment of passive safety systems significantly reduces the risk of nuclear reactor. Some specific examples of passive safety system for Generation III+ LWRs are described below (IAEA, 2009).

- (i) Safety injection tank: The safety injection tank is a coolant tank pressurized by nitrogen gas inside the tank, to supply auxiliary coolant to the core during a postulated LOCA. The operational principle is based on the difference between nitrogen gas pressure and primary system pressure.
- (ii) Core make-up tank: The core make-up tank, which is located above the core, injects coolant to the core by gravitational force. The upper valve of the core make-up tank provides the pressure balance between the tank pressure and primary system pressure, which allows enough head to drive the flow, regardless of the primary system pressure.
- (iii) Natural circulation inside the primary loop: If the primary loop is still closed during the accident, the natural circulation is achieved inside the primary system as long as the loop inventory is filled. The heat sink can be either the steam generator or an in-containment refuel water storage tank. The isolation condenser of the advanced BWRs adopts the same concept for the residual heat removal system, which is composed of closed circuit connected to the reactor vessel and the additional heat exchanger submerged in the cooling tank.
- (iv) Passive containment spray system: The external cooling of the containment building is achieved by natural circulation inside the passive containment spray system. The sprayed water is boiled by heat transferred from the surface of containment, and the steam is removed from the system along with the air. Inside the containment building, the steam generated from the core (or the entire primary system) condenses via external water and then flows down through the inner surface. This is followed by a natural circulation operational mode.
- (v) Containment cooling system: Locating external water pool above the containment building leads to natural circulation inside a closed loop between a heat exchanger in the containment building and the water pool. The natural circulation can continuously remove the heat from the containment building better than slow conduction across the containment building.
- (vi) Direct Vessel Injection (DVI) system is an advanced safety system to improve the challenges in injecting emergency coolant into cold leg by conventional safety injection system. DVI injects emergency coolant directly to the core instead of injecting it through the cold leg. After the AP-1000 adopted DVI for the improved safety system, US Advanced Pressurized Water Reactor (US-APWR) and Korea's Advanced Power Reactor 1400 (APR1400) followed by selecting the DVI system for delivering borated ECCS water from safety injection pumps into the core.

One prominent example of a demonstration of inherent safety in an advanced reactor is the EBR-II SHRT mentioned in this Chapter (Feldman et al., 1987; McFarlane, 2021). EBR-II was an experimental sodium-cooled fast reactor which operated for 30 years in the state of Idaho. The EBR-II SHRT examined the inherent capability to passively shutdown the reactor by steam load and primary flow control followed by decay heat removal by natural circulation of the primary coolant. The program demonstrated (Planchon et al., 1986) the inherent safety of EBR-II for a wide range of postulated accidents (Planchon et al., 1987) including loss-of-flow without SCRAM (LOFWS), loss-of-heat sink without SCRAM, and overpower accidents.

Similarly, the Fast Flux Test Facility (FFTF) was another reactor which demonstrated the inherent safety potential of liquid metal cooled reactor designs (Wootan and Casella, 2016; Zhang et al., 2017). FFTF conducted series of passive safety tests including steady state reactivity feedback tests, delayed pony motor trip, steady state natural circulation, controlled flow transient, LOFWOS with gas expansion modules (GEMs), inadvertent pump startup with GEMs, and core demonstration experiment safety tests. The results of the FFTF demonstrated the inherent safety features of sodium-cooled reactor designs including thermal expansion coefficients, Doppler coefficient, sodium density changes, and control rod driveline expansions that act as the passive negative reactivity feedbacks to counter a rise in sodium temperature.

Similar tests of inherent safety were also conducted in HTGRs. In the AVR reactor (German: *Arbeitsgemeinschaft Versuchsreaktor*), which was an experimental HTGR, the inherent safety features of HTGRs were examined under a pressurized loss of forced cooling accident (Ding et al., 2009). The experiments and simulations demonstrated the inherent control of reactivity by temperature change of the pebble bed core (Krüger et al., 1991). Similarly, the High-Temperature Engineering Test Reactor (HTTR) in Japan conducted tests for a control rod withdrawal in HTGRs (Nakagawa et al., 2004). The tests demonstrated another inherent safety of HTGRs that the fuel would not fail even if a large reactivity is inserted in a very short time. Additional fuel safety tests in the Nuclear Safety Research Reactor established useful limits for energy deposition in reactivity initiated accidents for HTTR fuel (Umeda et al., 2010). The inherent safety features of HTGRs were also examined in HTR-10 (Gou et al., 2018) under two postulated anticipated transients without scram (ATWS) accidents (Hu et al., 2006), including loss of forced cooling and Reactivity Insertion Accident

(RIA) tests (Boer et al., 2010). These examples of separate effects fuel safety tests are additional examples of the kind of testing needed to support reactor safety.

Molten salt reactor (MSR) designs achieve their inherent safety by the liquid state of fuel salt (the fuel cannot melt), strong inherent reactivity feedback characteristics, and low excess reactivity. This inherent capability was experimentally examined in Molten Salt Reactor Experiment (MSRE) (Guymon et al., 1966) which was an 8 MW_{th} fluid-fueled test reactor (Křepel et al., 2007). The fuel in some MSR designs can be dumped into a storage tank located below the core that can assure sub-criticality and decay heat removal.

Most of the inherently safe advanced reactor designs focus on the benefits of a small modular or integrated design. These design approaches enhance the inherent safety of the reactor and reduce uncertainties in the entire reactor system.

Conclusions

The first priority in the design of an NPP is always its safety in design, construction, operation, and eventual decommissioning. Lessons learned from past reactor accidents have led to the development of rigorous safety standards for the licensing and regulation of NPPs and to an enhanced focus on the passive safety characteristics of reactor concepts and systems. Various passive safety tests have been conducted on engineering facilities and experimental reactors, both in the United States and internationally. These tests build on the foundation of separate and integral effects tests required to generate the data necessary for model validation.

Although the current fleet is very safe, consistent with the Institute for Nuclear Power Operation's mission to promote the highest levels of safety and reliability—to promote excellence—in the operation of commercial nuclear power plants, ongoing studies leading towards a new level of safety in both existing and future nuclear reactors are under way.

See Also: Safety, Regulation, and Decommissioning of Commercial Nuclear Power Reactors—Introduction.

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Management of the Nuclear Waste – Via Disposition or Transmutation

Pavel V. Tsvetkov, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

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Glossary

Actinides Any of a series of 15 consecutive chemical elements in the periodic table from actinium to lawrencium (atomic numbers 89–103). As a group they are significant largely because of their radioactivity. Although several members of the group, including uranium (the most familiar), occur naturally, most are man-made.

Conditioning Chemical improvement of nuclear waste prior to storage.

Deep Geological Repository An underground site, excavated from the surrounding rock formation, where nuclear waste is to be isolated from the biosphere for long periods of time.

Double Strata Fuel Cycle Proposed by JAERI (Japan Atomic Energy Research Institute), consists of the power reactor fuel cycle (Stratum 1) and the partitioning-transmutation cycle (Stratum 2). It consists of a reprocessing plant, a partitioning plant and a transmutation plant on a single site. In the power reactor stratum, only U and Pu are recycled. In the P-T cycle, the high level waste is partitioned (into four groups: TRUs, Sr-Cs, Tc, others), fabricated into fuel elements, and then transmuted in a dedicated accelerator driven system. In this scenario the MAs are separated from the power cycle due to the fact that they can be completely recovered and transmuted. Since the MAs are strong neutron emitters, this avoids potential problems of radiation shielding in the conventional reprocessing. The final HLW should contain mostly short-lived fission products, but also low concentrations of Pu and MA that escaped with process waste streams.

High Level Waste (HLW) Refers to the liquid waste from reprocessing which contains fission products and the minor actinides and, as a result, requires massive shielding and provision for cooling.

Minor Actinides (MA) Refers mainly to the elements neptunium, americium and curium. These minor actinides (MA) are produced as radioactive by-products in nuclear reactors. The term “minor” refers to the fact that they are produced in smaller quantities in comparison to the “major” actinide plutonium.

Mixed Oxide (MOX) Mixed oxide fuel containing plutonium instead of 235 U-enriched uranium as the main fissile component.

Partitioning Refers to aqueous or pyroprocesses which can be used to separate (partition) radioactive by-products or waste produced in nuclear reactors from the spent fuel.

Reprocessing General term for applying chemical and physical processes to used (spent) reactor fuel.

Used Fuel Nuclear fuel that has been used in a reactor.

Transmutation The conversion of a nuclide into a shorter lived or stable nuclide in a reactor or with an accelerator.

TRU Transuranic elements.

Abbreviations

ADS Accelerator Driven System
 DOE Department of Energy
 FR Fast Reactor
 GFR Gas Fast Reactor
 HLW High Level Waste
 HTR High Temperature Reactor
 ILW Intermediate Level Waste
 LFR Lead-cooled Fast Reactor
 LLFPs Long-Lived Fission Products
 LLW Low-Level Waste
 LWR Light Water Reactor
 MA Minor Actinides
 MOX Mixed Oxide
 P&T Partitioning and Transmutation technologies
 PWR Pressurized Water Reactor
 SCWR Super Critical Water Reactor
 SFR Sodium-cooled Fast Reactor
 TRU TRans-Uranic nuclides
 UOX Uranium Oxide
 VHTR Very High Temperature Reactor

Nuclear waste

Contemporary nuclear energy systems are the result of extensive development efforts during which the technology has reached industrial maturity. Most of these developments have been concentrated on light water reactors (LWR) and their fuel cycles (OECD-NEA, 2003; Cochran and Tsoulfanidis, 1999).

Note that those parts of the fuel cycle that precede the use of the fuel in a reactor core are collectively called the front end of the nuclear fuel cycle (Crawford, 2021; Cochran and Tsoulfanidis, 1999). That portion related to the fuel after it has been removed from a reactor is called the back end of the nuclear fuel cycle. According to the front-end options, there are three well-developed fuel cycle technologies:

- Once-through (open) fuel cycle based on enriched uranium,
- Once-through (open) fuel cycle based on natural uranium,
- Fuel cycle with plutonium recycling.

The once-through (open) fuel cycle is the reference fuel cycle. It is the main alternative for Canada, Spain, Sweden, USA and some other countries. This scenario gives, with the present low uranium prices, the cheapest nuclear energy production. However, it implies that the residual fissile material content (1% Pu and 0.8% ^{235}U ; weight % of the metallic elements in the discharged fuel), as well as the remaining fertile material (^{238}U), of the spent fuel will not be recovered and becomes a waste material (Cochran and Tsoulfanidis, 1999).

Radioactive wastes comprise a variety of materials requiring diverse types of management to protect people and the environment (Baetsle, 2001; Cochran and Tsoulfanidis, 1999). They are normally classified as low-level (LLW), intermediate-level (ILW) or high-level wastes (HLW), according to the amount and types of radioactivity in them:

- Low-level waste (LLW) comprises the bulk of waste from the nuclear fuel cycle. It comprises paper, rags, tools, clothing, filters etc. which contain small amounts of mostly short-lived radioactivity. It does not require shielding during handling and transport and is suitable for shallow land (near surface) burial. To reduce its volume, these wastes are often compacted or incinerated before disposal. Disposal sites for low-level waste are in operation in many countries. Worldwide they make up 90% of the volume but have only 1% of the total radioactivity of all radioactive wastes.
- Intermediate-level waste (ILW) contains higher amounts of radioactivity and normally requires shielding. Shielding can be barriers of lead, concrete or water to give protection from penetrating radiation such as gamma rays. Intermediate-level wastes typically comprise resins, chemical sludges and metal fuel cladding, as well as contaminated materials from reactor decommissioning. It may be solidified in concrete or bitumen for disposal. Generally short-lived waste (mainly from reactors) is buried.
- Operating a nuclear reactor implies fissioning ("burning") the fuel, producing fission products, such as various isotopes of barium, strontium, cesium, iodine, krypton and xenon (Ba, Sr, Cs, I, Kr, and Xe). Many of these light nuclides formed as fission

products within the fuel are highly radioactive, and hence short-lived. In addition to nuclides formed through fissions, neutron capture reactions transform fertile actinides into various transuranic (TRU) isotopes. These include ^{239}Pu , ^{240}Pu and ^{241}Pu , as well as others arising from ^{238}U by neutron capture and subsequent beta decays. Looking at the backend of the fuel cycle, all these nuclides are radioactive and remain within the used fuel composition when it is removed from the reactor. The transuranic nuclides constitute the most of the long-lived portion of HLW. The long-term hazard of used fuel and HLW is associated with actinides, particularly the TRUs, while the short and long term risks are due to the mobility of fission products in the geosphere and the possibility of their entering the biosphere. During the first 150 years after fuel discharge, the fission products ^{137}Cs and ^{90}Sr determine the radiotoxicity and the heat emission. From 150 years till 250,000 years the plutonium isotopes are the main contributors to the radiotoxicity. Beyond 250,000 years ^{237}Np emerges together with the daughter products of uranium. One may expect developments aiming towards solutions improving the back-end of the nuclear fuel cycle, i.e. management of transuranics in the used fuel.

Partitioning and transmutation (P&T) of transuranic nuclides (TRU), in particular minor actinides (MA), in nuclear reactors offer benefits of significant reductions or complete conversion of long-lived HLW TRU inventories into short-lived fission products. Thus, these technologies carry a potential of reducing expectations for future nuclear waste repositories.

Disposition of nuclear waste

Most countries with nuclear facilities planned or operational have active programs seeking geological disposal sites. The objective is to locate areas where multiple reliable barriers can be established between the wastes and the environment (Cochran and Tsoulfanidis, 1999; Hewitt and Collier, 2000; Mekki, 2021; Ewing and Park, 2021). Some of the barriers, both natural and engineered, being considered are:

- Encapsulate in an insoluble form of waste (glass, Synroc or UO_2) that is immobilized either as ceramic oxide (e.g. the spent fuel itself) or through vitrification (separated HLW from reprocessing);
- Seal in corrosion-resistant containers (stainless steel or copper);
- Immobilize in wet rock packing with bentonite clay to inhibit groundwater movement and insulate from minor earth movement;
- Locate deep underground (e.g. 500 m deep) in stable rock formations.

Synroc is an advanced ceramic substance principally comprising three natural titanate minerals which are geochemically stable and which together have the capacity to incorporate into their crystal structures nearly all of the elements present in high-level radioactive waste, thereby immobilizing them. Three types of geological structures are being widely studied for this purpose: hard crystalline rocks, clays and rock salt beds (Cochran and Tsoulfanidis, 1999; Lamarsh and Baratta, 2001).

Suitable locations have been defined in several countries and the sites are now undergoing detailed evaluation. Most approaches plan to utilize conventional mining techniques involving shaft-sinking and development of extensive drives and rooms. These will provide sufficient area for the canisters to be placed in suitably-spaced holes in the floor on each level, or some other arrangement (IAEA, 2005).

One purpose-built deep geological repository is now operating in New Mexico, USA, but this is only for long-lived military wastes: The WIPP – Waste Isolation Pilot Plant in Lea County, located in the remote Chihuahuan Desert in a salt formation, began operations on March 26, 1999.

Moving forward with an integrated waste management strategy, the U.S. Department of Energy (DOE) established the policy moving to a long term solution. The plan is to setup interim storage facilities first, and then move to a long-term repository. The strategy is sufficiently flexible to incorporate advanced fuel cycles. As a result, to date there has been no practical need for final HLW repositories, as surface storage for 30–50 years is first required so that heat and radioactivity can dissipate to levels which make handling and storage easier (Cochran and Tsoulfanidis, 1999; Department of Energy, 2021). Extensive spent fuel evaluations have been conducted to fully characterize compositions, storage conditions, dry cask loading sequences of interim storage options, and expectations for permanent disposal. These efforts results in the spent fuel database and associated tools for waste management evaluations (Yancey and Tsvetkov, 2014). The analysis identified and quantified variations in the inventory data showcasing both fidelity of older historical data as well as the need for regular characterization of accumulating inventories (Yancey and Tsvetkov, 2015).

Reprocessing

Reprocessing involves dissolving the spent fuel to enable the re-usable plutonium and uranium content to be separated from the residual waste fission products and actinides. The fuel assemblies are chopped up and placed in nitric acid. This enables the fuel content, which dissolves in the acid, to be separated from the insoluble zirconium alloy or stainless steel cladding (Cochran and Tsoulfanidis, 1999; Marin, 2021).

The solution of uranium, plutonium, other actinides and fission products is then chemically treated in a series of stages which are designed to produce solutions of plutonium nitrate and uranyl nitrate of high chemical purity. The waste products (other actinides, fission products and unwanted impurities) are stored as a highly radioactive solution in water-cooled, double-walled high-integrity stainless steel tanks before further conditioning. The separate solutions of uranyl nitrate and plutonium nitrate are further processed. The uranium can be converted to uranium dioxide for storage or for the production of new fuel, by blending with fissile material or conversion to uranium hexafluoride for return to the enrichment plant. The plutonium nitrate is converted to plutonium dioxide for storage or for incorporation into Mixed Oxide fuels for thermal or fast reactors (Baetsle, 2001).

Conditioning of the wastes produced by reprocessing is a well established operation that has been rigorously examined and approved by regulatory authorities in several countries. The removal of the plutonium and the uranium via reprocessing reduces the volume of high level waste, but leads to the production of low and intermediate level wastes.

Following conditioning and, in most cases, interim storage for a number of decades to allow further reduction of radioactivity and heat generation, vitrified HLW, suitably encased, can be transported to and placed in a deep geological repository. Here, it can be held under supervision and, when considered appropriate, sealed off permanently. The glass matrix in which the highly radioactive wastes are incorporated, the method of encapsulation and the geological formation chosen to isolate the radioactivity from the biosphere, are carefully selected to ensure long term safety (Cochran and Tsoulfanidis, 1999; Hewitt and Collier, 2000; Bruno, 2021).

Partitioning and transmutation

The role of nuclear energy in a sustainable energy future has multiple facets, a significant number of which relate to the nuclear fuel cycle. Many sustainability issues are associated with the fuel cycle, e. g. use of natural resources, economics, wastes, public acceptance, proliferation resistance, and etc. (Brogli and Krakowski, 2002; OECD-NEA, 2001; Wydler and Baetsle, 2000).

Partitioning and Transmutation (P&T) is a complex technology which implies the availability of advanced reprocessing plants, facilities for fuel fabrication of transuranic elements (TRUs) and irradiation facilities beyond the present facilities (OECD-NEA, 2003; Salvatores, 2005; Salvatores et al., 2004, 1994; Boullis and Charbert, 2021).

Partitioning is to a certain extent a broadening to other nuclides of the current reprocessing technique which has been operating at industrial level for several decades and for which the main facilities exist or can easily be extrapolated from present day nuclear plants (Jackson, 2003).

Partitioning is the technology which can be considered as a form of “super - reprocessing” by which, in addition to U, Pu and Iodine (^{129}I) also the Minor Actinides (MA) and the Long lived Fission products (LLFP: ^{99}Tc , ^{93}Zr , ^{135}Cs , ^{107}Pd and ^{79}Se) would be extracted from the liquid high level waste (Baetsle, 2001; Jackson, 2003).

The closed fuel cycle with P&T incorporated is without any doubt the most comprehensive approach. It requires very important and dedicated fuel cycle and reactor facilities which go far beyond the presently existing nuclear technology. Especially, the transmutation approach calls for the development of FR-burners and accelerator-driven systems (ADS) which may take 20 to 30 years before these are industrially available (Wydler and Baetsle, 2000; Boullis and Charbert, 2021; Krepel, 2021; Abderrahim and De Bruyn, 2021).

Fig. 1 illustrates typical nuclide transformations in a legacy PWR, assuming the fuel was irradiated by thermal neutrons. The use of higher energy neutrons would lead to enhanced fissions of TRUs ultimately reducing their inventories (Tsvetkov et al., 2008).

Moving towards sustainable nuclear energy, the transmutation strategy is to close the fuel cycle of conventional, i.e. LWR-dominated, reactor parks also for the MA by recycling all actinides homogeneously or heterogeneously in existing and innovative reactor types. If the fuel reprocessing losses are sufficiently small, complete closure of the fuel cycle would result in a considerable reduction of the actinide content, and hence the long-term radiotoxicity, of the HLW. Innovative reactor types and new recycle infrastructures, including the pyrochemical or “dry” reprocessing technique, would be necessary, especially for burning highly concentrated TRU and minor actinides (Brogli and Krakowski, 2002; National Research Council, 2008).

If some of the long-lived fission products could also be transmuted or separately conditioned, the radiotoxicity of the remaining HLW would decay within a few hundred years, meaning that repository designs for such HLW may meet licensing requirements more easily. Thus, since the radiotoxic nuclides in the HLW can only partially be eliminated, transmutation does not make geologic disposal concepts superfluous, but must be considered as a complementary waste management method which may ease the design and licensing requirements for geologic repositories (IAEA, 2004; Salvatores, 2005).

However, with regard to the repository requirements for vitrified HLW, it should be noted that the total mass and the initial heat production of the vitrified waste are dominated by the fission products and, hence, mainly depend on the total thermal energy produced by nuclear reactors. Moreover, due to the non-linear relationship between the actinide content of the waste matrix and the repository release rates, actinide mass reductions achieved by transmutation do not necessarily translate into proportional actinide risk reductions (Baetsle, 2001).

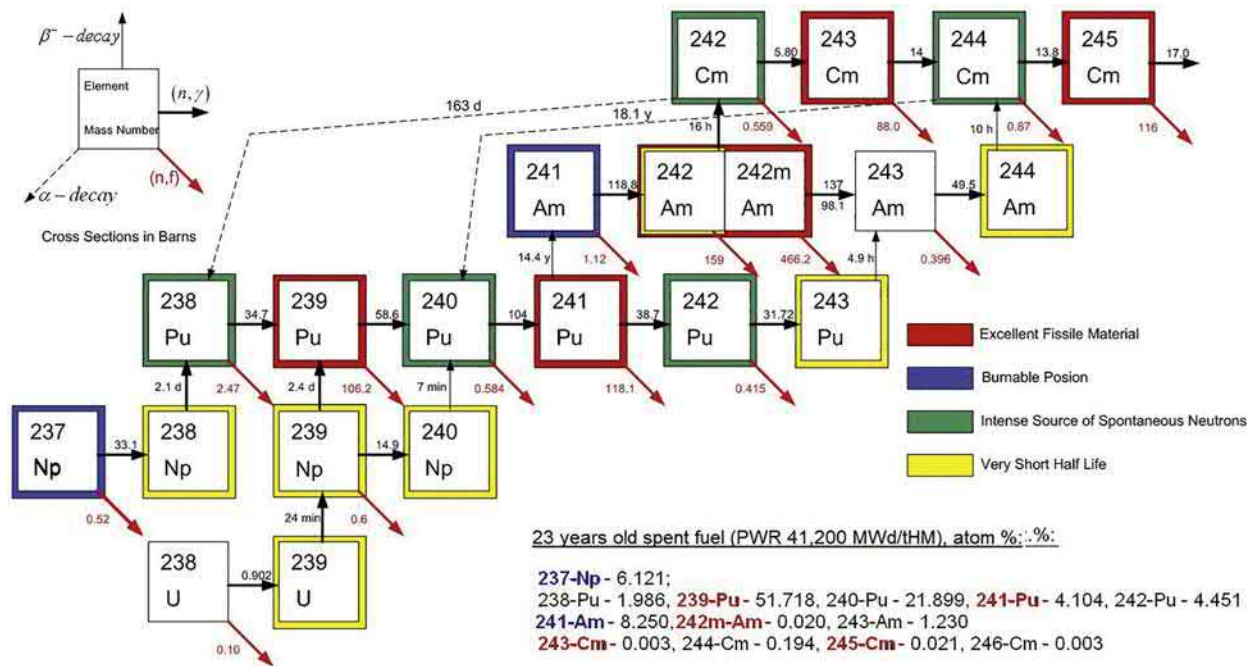


Fig. 1 Thermal neutron induced transformations in the reference legacy-PWR fuel.

Considering various reactor options and related fuel cycle scenarios, there are several feasible fuel cycle development directions:

- Once-through LWR-to-HLW;
- TRU-burning LWR-to-FR-to-HLW;
- TRU-burning LWR-to-ADS-to-HLW;
- FR-to-HLW.

In the conventional LWR fuel cycle, the radiological hazard of the depleted uranium arising from the enrichment process and the recovered irradiated uranium (irradiated enriched uranium from LWR-UOX fuel and irradiated depleted uranium from LWR-MOX fuel) is only of secondary concern. However, in P&T scenarios with fully closed fuel cycles, the long-term radiotoxicity of this residual uranium becomes comparable with the HLW radiotoxicity (Brogli and Krakowski, 2002; Cochran and Tsoulfanidis, 1999; Kolb and Martinsen, 2003).

Whereas all transmutation strategies with fully closed fuel cycles perform similarly with respect to the resource efficiency and environmental friendliness parameters, different requirements and consequences arise for the establishment of the fuel cycle. In this context, important parameters are the TRU inventory of the fuel cycle and the decay heat and neutron source strength of the fuel.

The analyses show that, among the different transmutation strategies, TRU burning in ADS is associated with the lowest TRU inventory and hence the lowest activity inventory. This means that this strategy can respond flexibly to unexpected changes in the nuclear energy scenario and has safety advantages (OECD-NEA, 2002; IAEA, 2004; Nakagawa, 2003).

On the other hand, the decay heat of the ADS fuels is well beyond the limit for which the radiation stability of the organic extractant in the aqueous process can be guaranteed. For these and all other systems with fully closed fuel cycles, the less developed pyrochemical reprocessing ("dry" reprocessing) is the appropriate reprocessing method because it circumvents unnecessary separation processes (only fission products are extracted) and can handle highly active product streams without major radiation degradation (OECD-NEA, 2002).

Thus, P&T techniques could contribute to reducing the radioactive inventory and its associated radiotoxicity by a factor of 100 or more and to reducing the time needed to approach the radioactivity levels of the raw materials: uranium and its equilibrium decay products originally used to produce energy. Great scientific progress has been made in metallic barrier building and in mining technology. Nobody can fully guarantee total confinement of radiotoxic materials in human-made structures beyond 10,000 years.

The reduction of the solubility of long lived radio-nuclides in underground aquifers may be a first step in reducing their migration. Partitioning from nuclear waste streams and conditioning of long lived radio-nuclides into stable matrices may be a second step in this direction before disposal. The economic implications of a P&T policy are not negligible, however, and the responsibility associated with this societal choice has to be taken by the present generation for an unknown number of future generations. The decision to reduce the radiotoxic inventory of nuclear material by transmutation is an important one that has to be taken by the responsible authorities.

The main goals of P&T are:

- A reduction of the hazard associated with spent fuel over the medium and long term (> 300 years) by a significant reduction of the inventory of plutonium and MAs.
- A reduction of the time interval required to reach a reference level of radiotoxicity inventory by recycling transuranic elements (TRUs).
- A decrease of the spent fuel volume by separation of uranium to enable more efficient storage or disposal. This should result in an increase in the effective capacity of a final repository. However, this approach might require special handling of strontium and cesium after partitioning.

Evaluations confirm the following general observations:

- Radiotoxicity: P&T must first be concerned with the actinides, MAs – Am and Cm. The toxicity of the fission products is two orders of magnitude smaller after 10^3 years.
- Repository risk: The mobile fission products are more important than the actinides. The fission products, ^{129}I , ^{135}Cs , ^{79}Se , ^{99}Tc , ^{126}Sn , are dominant dose contributors.
- The fission product risk peaks between 10^4 and 10^6 years after the closure of a repository, whereas an actinide risk arises “only” after one million years.

The major potential benefits of application of P&T technologies are:

- Resource conservation by making maximum use of the energy available from the primary fuel material, uranium. This has been pursued through the separation of re-useable uranium and plutonium from spent nuclear fuel by means of reprocessing. The current price of uranium relative to fossil fuels makes this uneconomic and there is perceived to be limited value in extending reprocessing operations to this end alone;
- Reduction of the heat generation of wastes. Repository designs can incorporate temperature limits designed to guard against thermal degradation of the surrounding rock and prevention of boiling that impedes natural groundwater movements. The benefit derives from enlarging the capacity of a repository and reducing the number of sites required. This aim underpins R&D in the USA in both partitioning and transmutation;
- Reduction of radiotoxicity of waste. By transmuting elements, such as plutonium and americium, it is theoretically possible to transform the spent fuel so that it will decay to the activity of the original uranium in a few hundred (rather than tens of thousands) years, reducing the period over which it is a hazard.

All these potential benefits have to be judged against other forms of management. If direct disposal, for example, is a reliable means of isolating the wastes from the environment (thus preventing the escape of the more mobile radionuclides most of which are not suitable for transmutation) then significant additional cost of transmutation technology would have little practical benefit in reducing hazard (Wydler and Baetsle, 2000).

Nuclear fuel cycle problems and challenges

Aqueous reprocessing

Considering various P&T options and technologies, there are many problems and challenges that should be mentioned:

- Current proposals for P&T technology rely on the aqueous (PUREX-type) reprocessing of spent fuel as a preliminary step preceding minor actinide partitioning. The drastic increase of the plutonium content from 15% up to 43% and the fuel burn-up from 80 GWd/tHM up to 210 GWd/tHM for fast reactor concepts make aqueous reprocessing more difficult because of the low solubility of plutonium and the radiation damage to the organic extractant (tributyl phosphate).
- The present state-of-the-art in aqueous reprocessing based on the PUREX process is almost satisfactory with regard to uranium and plutonium separation. Recovery yields of nearly 99.7% have been achieved and 99.9% is potentially achievable in a near future. However, reprocessing of industrial quantities of LWR-MOX and FR-MOX (FR – Fast Reactor) within short cooling times will require gradual adaptations of the PUREX flow-sheet and involve the construction of additional extraction rigs.
- The decay heat of the separated plutonium fraction enters the organic phase and remains in contact with the actinide fraction (U, Pu, Np) until separation and purification of the major actinides occurs. The ^{238}Pu concentration in the plutonium stream is the main source of radiation damage to the tributyl phosphate extractant, a strong source of neutrons in the separation plant, and an additional radiolysis agent during the oxalate precipitation and conversion to PuO_2 . On the other hand, the high ^{238}Pu concentration in the separated PuO_2 for the same reasons (heat, spontaneous neutrons) improves the nonproliferation resistance of the product during storage.
- The MOX fuel fabrication will have to adapt its handling technology to reduce the radiation dose to the workers in the plant and during the transport of MOX fuel assemblies.
- The separation of the MA is currently under intense investigation. The separation of the bulk of ^{237}Np from the aqueous product solution during the reprocessing operations is technically feasible but requires an adaptation of the first cycle extraction flow-sheet. The residual ^{237}Np which goes directly to the HLW solution could, in principle, be recovered by recycling the HLW

solution through a secondary recovery cycle after a valence adjustment. With liquid extraction processes, the separation yield can be increased up to the desired value of 99.9% by increasing the number of extraction stages.

- The separation of americium and curium is much more difficult, but considerable progress has been made during the past decade. Bulk separation of MA together with lanthanides has been demonstrated under hot-laboratory conditions. The separation of MA from lanthanides, however, remains an obstacle for the industrial up-scaling of Am-Cm separation from HLW. Several promising laboratory methods have been tested. Yields of 99% and 97.6% have been achieved for americium and curium, respectively. Further improvements are necessary in order to include these methods in a cycle of multiple reprocessing.

Pyrochemical reprocessing

The proliferation risk potentially associated with the clean plutonium produced by the aqueous reprocessing has drawn the attention on the pyrochemical reprocessing which makes it difficult to separate individual TRU elements. Whereas the pioneering work was performed by ANL in the USA, most of the recent advances have been made in Russia and Japan. In the Russian Institute of Atomic Research (RIAR), a pyrochemical process has been demonstrated with highly irradiated spent oxide fuel with burn-ups of 210–240 GWd/tHM (OECD-NEA, 2003; Cochran and Tsoulfanidis, 1999; Jackson, 2003).

Good results were obtained during the demonstration. The recovered PuO_2 are mixed with UO_2 and processed by vibropacking into fresh FR-MOX fuel. Pyrochemical reprocessing of metallic fuel was developed in the USA in the framework of the integral fast reactor (IFR) program with capability for actinide burning. Recovery yields of 95% for uranium and 99% for mixtures of uranium and minor actinides have been obtained on laboratory and pilot-plant scales (Jackson, 2003).

The advantages and disadvantages of the pyrochemical reprocessing can be summarized as follows:

- Highly concentrated TRU product streams can be handled without major radiation degradation of the reagents, allowing the facilities to be more compact than aqueous reprocessing facilities for the same TRU throughput. Because of the absence of water, much smaller criticality risks during purification and metallic fuel re-fabrication arise when processing industrial quantities of TRUs.
- Since all TRU elements remain together throughout the process, the proliferation risk is much reduced. The separation of the TRUs from the lanthanides, however, is difficult, requiring the development of multistage countercurrent extraction using highly corrosive reagents at high temperature. The most challenging issue is the selection and industrial manufacture of corrosion resistant equipment which must be designed for remote operation and maintenance.
- For spent LWR-UOX fuel, a genuine pyrochemical process has to cope with the problem of the elimination of the uranium, which is the major constituent of the LWR fuel. Therefore, a mixed approach using a (P)UREX-type process for uranium-neptunium elimination has been selected as the first step. Fluoride volatility has also been suggested as an alternative, but the mixed volatility of U-Pu and zirconium leads to complex waste streams which can be difficult to control.
- Starting from metal or nitride TRU fuel, a complete pyrometallurgical process involving a series of electro-refining steps in a wide range of molten salt baths has recently attracted much interest throughout the nuclear research communities in the USA, Russia, Japan and France. Most of the respective flow-sheets remain in the pre-conceptual phase and will require several decades of R&D to become a mature technology.
- Whereas the aqueous reprocessing consists of an independent industrial process which supports a large reactor park and can operate independently on continental or even world scales, the pyrochemical process will predominantly be applied in small facilities installed in the immediate vicinity of the reactors.
- However, in the long-term, pyrochemical reprocessing will become indispensable, if very hot fuel has to be multi-recycled in fast reactors or accelerator-driven systems on an industrial scale with a limited out-of-pile inventory.

Actinide transmutation

Depending on regional boundary conditions and political factors, countries with P&T programs have developed different transmutation strategies:

- The evolutionary approach, adopted mainly in Europe and Japan, aims at closing the fuel cycle in successive steps, starting with the recycling of plutonium in LWRs and later in fast reactors using conventional reprocessing and MOX fuel technology, and finally complementing the system with a dedicated P&T cycle which features MA burners with fast neutron spectra.

This approach has the advantage that it can respond flexibly to changes in the priorities for plutonium and minor actinide management, and that new technologies have to be developed only for a comparatively small number of MA burners which support a large system of conventional LWRs and fast reactors.

An option has been explored to perform the transmutation of MA in the form of targets to be loaded in critical cores of a “standard” type. The mode of recycling has been called “heterogeneous”, the potential advantage being to concentrate in a specific fuel cycle the handling of a reduced inventory of MA (separated from Plutonium).

- The innovative approach, first suggested in the USA, aims at co-processing plutonium and MA to avoid the use of technologies with a potentially high proliferation risk. After the initial separation of the uranium from the LWR spent fuel, the unseparated

transuranic actinides are recycled in a transuranic burner with a closed fuel cycle based on pyrochemical reprocessing. Compared with the double strata strategy, the number of burners in the equivalent system is four to six times larger, but the burners are not subjected to a (fast) neutron-spectrum condition. Nevertheless, most of the currently evaluated critical and sub-critical transuranic burners feature a fast neutron spectrum.

The homogeneous recycling has been explored in U.S. The concept is mainly designed to produce energy, making an optimized use of resources and using a robust reactor and fuel cycle layout. The fuel cycle does not imply the separation of Pu and MA.

The concept can accommodate in principle several options in terms of reactor size, reactor coolant, waste-forms, etc. In general, the homogeneous recycling has equivalent performances for whatever the type of fuel in the fast reactor. However, the consequences on the fuel cycle have to be taken into account and their impact evaluated.

The transmutation of an actinide is completed, when the transformation chain, which involves “generations” of neutron reactions and radioactive decays, terminates with a fission. In order to get a net MA destruction yield, the elimination of the nuclide must exceed the internal production. Conventional LWRs cannot be used to decrease significantly the Np and Am radiotoxic inventories resulting from LWR- UO_2 reprocessing as neutron capture reactions dominate the nuclear transmutation processes. The presence of highly active α -emitters (especially ^{244}Cm) in irradiated targets make multi-recycling nearly impossible (Baetsle, 2001; IAEA, 2004; Salvatores, 2005).

The low MA transmutation capability of LWRs has been demonstrated by different R&D groups. It has been shown that by following the once-through irradiation, only with very long irradiations (20 to 30 years) and with higher ^{235}U enrichment, significant results could be obtained.

The stabilization of the TRU inventory with LWRs is only possible by generalizing the use of LWR-MOX fuel with an increased ^{235}U enrichment in the entire reactor park. It would profoundly influence the fuel fabrication needs for LWR-MOX and require additional ^{235}U enrichment capacity (Kolb and Martinsen, 2003; OECD-NEA, 2001).

Complete elimination of the actinide inventory by neutron irradiation can only be accomplished in a fast neutron spectrum, of either “critical” or “subcritical,” reactor. The real net reduction results from fission and is consequently proportional to the burnup of the fuel or target. High burnups are the prerequisite condition for a reasonably fast reduction pace but involve on the other hand the industrial implementation of the most difficult reprocessing operations on very “hot” nuclear materials (OECD-NEA, 2002).

The maximum burnup obtainable in a fast neutron spectrum device (FR or ADS) is in principle not limited by the fissile material content, since all TRU's are to a certain extent fissionable, but by the resistance to fast neutron radiation damage of the cladding material expressed in “displacements per atom.” If the fast neutron dose exceeds 200 dpa the stability of the cladding material is at stake and the highly irradiated fuel or target (150–250 GWd/tHM) will have to be withdrawn from the reactor and recycled or disposed of (OECD-NEA, 2002).

From neutron economy considerations, it follows that the complete closure of the fuel cycle of a fission-based nuclear energy system is eased by integrating fast spectrum devices into the system. From the viewpoint of both resource utilization and actinide waste production, the ideal system is a single component system based only on fast reactors with a fully closed fuel cycle. It is well known that such a system has the potential of fissioning natural uranium and thorium resources with a close to 100% efficiency while producing only a very small amount of actinide waste.

Fission product transmutation

The neutron capture process is currently the only promising nuclear reaction for transmuting fission products. Other processes which have been proposed in the past rely on technologies which are still at a very early stage of development (e.g. fusion neutron sources) and generally suffer from a poor energy balance. The capture process consumes neutrons but, theoretically, fast reactors could deliver enough surplus neutrons to allow the potentially troublesome long-lived fission products to be completely transmuted to shorter-lived or stable species.

The transmutation of a fission product makes sense only if the reaction rate (microscopic cross section times neutron flux) is higher than the natural decay rate of the nuclide (Cochran and Tsoulfanidis, 1999). With the practically achievable neutron fluxes, this condition cannot be met for the most abundant fission products, ^{137}Cs and ^{90}Sr with half-lives of only about 30 years, i.e. these fission products are “non-transmutable.” However, since their radioactive life is limited to less than 300 years, they can be safely enclosed using engineered barriers only. On the other hand, the fission products determine the size of the vitrified waste repository which, consequently, cannot be much reduced by P&T operations.

The fission products which influence the long-term risk of a repository are, in order of radiologic importance, ^{129}I , ^{99}Tc , ^{135}Cs , ^{93}Zr , and ^{126}Sn . Activation products (^{14}C and ^{36}Cl) can also contribute to the dose. Obviously, the relative contribution of these nuclides to the integral risk varies according to the type of repository host rock. In the following, the problems associated with the transmutation of these five fission products are discussed separately. It should be noted that the determination of the separation yield and the decontamination factor (DF) for HLW depends to a great extent on policy decisions and that, depending on the nuclide, special conditioning and confinement is an alternative to transmutation.

The primary risk of geologic repositories is related to the release of long-lived fission products. For the long-lived fission products, the goal is to transform them into shorter-lived or stable species by means of neutron capture reactions. With the exception of ^{99}Tc , however, the transmutation of long-lived fission products appears to be difficult because of low neutron reaction cross-sections

and the necessity of isotopic separations. This means that, for most fission products, special conditioning and confinement is the only practical method to reduce the radiological impact.

It should be noted that the transmutable fission products represent only a small fraction of all fission products and that the vitrified waste mass, which is primarily determined by the fission products, therefore, cannot be much reduced by P&T operations. A combined transmutation-conditioning strategy for the long-lived fission products, however, would allow easing the licensing requirements for vitrified waste repositories because the geosphere barrier would no longer have an important safety function.

Generation IV transmutation impacts

The Generation IV systems considered are the thermal-spectrum VHTR and SCWR, and the fast-spectrum GFR, LFR, and SFR. The base fuel cycle for the thermal reactor concepts (VHTR and SCWR) is a once-through fuel cycle using low-enriched uranium fuels. The higher burnup and thermal efficiency of the VHTR gives an advantage in terms of heavy-metal waste mass and volume, with lower decay heat and radiotoxicity of the spent fuel per electrical energy produced, compared to a PWR. Fuel utilization might, however, be worse compared to the PWR, because of the higher fuel enrichment essential to meeting the VHTR system design requirements. The SCWR concept also features improved thermal efficiency; however, benefits are reduced by the lower fuel discharge burnup (National Research Council, 2008; OECD-NEA, 2001; Stanculescu, 2021).

The base fuel cycle for the fast reactor concepts (SFR, GFR, and LFR) is a closed fuel cycle using recycled TRU and depleted uranium fuels. Waste management gains from complete recycle are substantial, with the final disposition heat load determined by processing losses. The base Generation-IV concepts allow consumption of ^{238}U significantly extending uranium resources (up to 100 times).

For both thermal and fast concepts, recent design studies have pursued the development of dedicated burner designs. Fig. 2 illustrates typical neutron spectra in Generation IV helium-cooled HTRs and SFRs comparing their conditions to a generic Generation II PWR neutron spectrum.

Preliminary results suggest that a burnup of 50–60% is possible in a HTR burner design using non-uranium (transuranics) fuel. However, practical limits related to higher actinide buildup and safety impact may limit the extent of TRU burning in thermal reactors. Fast burner designs have been developed for both conventional and high TRU content fuel forms. In general, the conversion ratio can be varied within a system by changing the uranium loading. Recent studies indicate a low conversion ratio (0.25) SFR retains the favorable passive characteristics of conventional designs, and the cost is similar.

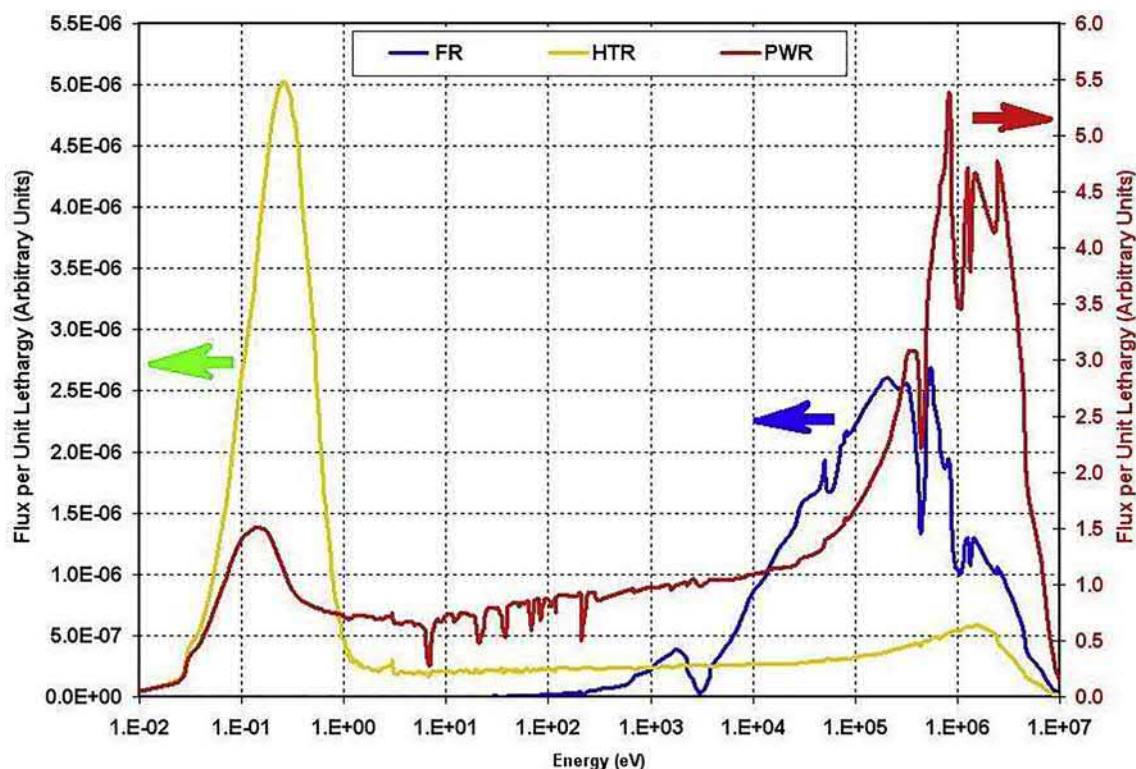


Fig. 2 Spectral conditions in reactor environments with light-water (PWR), graphite (HTR) and sodium (FR).

Transmutation systems and safety

At a basic level, there are six safety functions to be fulfilled when operating fissioning systems (Wydler and Baetsle, 2000).

- The nuclear fuel must remain contained within a controlled space because of its radiotoxicity; this is traditionally accomplished by use of multiple containment barriers.
- Shielding must be kept in place between fissioning and fissioned fuel and humans to avoid suffering radiation damage.
- A heat-transport path must be in place to carry energy away from the fissioning medium to a heat sink; usually an energy conversion plant.
- The rate of release of fission energy in the chain reacting medium must be regulated to remain in balance with the rate of energy delivery to the heat sink, so as not to overheat the containment barriers around the fuel and challenge their integrity; a capacity to store heat in the reacting medium and heat transport channel will buffer mismatches of short duration or small amplitude.
- Since some 5% of the energy liberated from each fission event is initially retained in nuclear bonds of unstable fission products, and since these fission products subsequently decay according to their natural radioactive-decay time constants, a means must be provided for transporting heat from the fission products and transuranics in the fuel for all times subsequent to the fission event. Failure to satisfy the latter two safety functions could lead to overheating of the fuel with the potential to defeat the integrity of the containment and shielding.
- Operation of the fissioning device in a quasi steady state mode requires a balance of neutron production and destruction rates from one fission chain generation to the next – even as the composition of the chain reacting medium changes due to transmutation and as the absorption, leakage, and neutron production properties of the fissioning medium change with changes in composition and temperatures.

Strategies to fulfill the six basic safety functions have been developed and refined over many years for conventional (critical) reactors. The strategy employs defense in depth such that any single failure will not defeat the strategies for meeting safety functions and thereby result in unacceptable release of radiotoxicity; multiple barriers (fuel cladding, primary coolant boundary, and reactor containment building) are used to prevent release of radiation even under accident conditions. Highly reliable (diverse and redundant) systems for controlling and terminating the chain reaction are used to match heat production to removal rate. Highly reliable, redundant/diverse systems for decay heat removal are provided. High quality construction and verification norms minimize manufacturing flaws, and rigorous maintenance, formal procedures, and exhaustive training and certification of operators and maintenance workers are used to minimize the occurrence of human error which could subvert the achievement of the safety functions. Once safety is “designed into” the system, its efficacy is judged by an independent safety regulative authority on a plant-by-plant basis prior to deployment and during its operation.

Economic considerations

It seems at present the only grounds for considering reprocessing would be to improve the overall situation for long-term storage of nuclear fuel. Calculating the costs and benefits would require an in depth analysis of factors such as the cost of reprocessing versus the benefits of a smaller depository. The likely high costs of transmutation, if and when it becomes commercially feasible, would have to be weighed against the presumably lower cost of burying the fuel for the long-term in an appropriate geological depository (OECD-NEA, 1994, 2001).

Engineering cost estimates would be a factor in decisions on which option to pursue but social, political and economic considerations would be likely be decisive.

Given the uncertainties that surround the technology, it is not surprising to find that there are no detailed cost estimates available. At present, reliance has to be placed on very broad assessments.

If the technology is to be linked to a future power reactor program, there are several items that would have the effect of raising the cost of production of the end product, namely, electricity. These include:

- Additional fuel manufacture costs because of the need to produce targets;
- Reduced power output from a reactor because the targets will carry a reactivity penalty;
- Additional reprocessing costs because of the additional plant and processing required.

If P&T was to be applied in the absence of a power reactor program, ADSs would need to be built. As no fully developed ADS conceptual design yet exists, it is not possible to offer even a crude estimate, but clearly implementation could prove to be very expensive indeed.

Environment, transportation, reprocessing safety

Environment

The environmental issues of reprocessing are closely related to the economic issues. It would seem that minimizing the environmental impact over the entire nuclear fuel cycle would be a more plausible optimization criterion instead of minimizing its total

cost. The complicated fuel cycles aimed at consuming most of the radioactive isotopes arising from spent fuel, of the type being studied in particularly Europe, involve more than one reactor and multiple reprocessing steps. They need to be examined very carefully, particularly in terms of the resulting reprocessing wastes. Comparisons need to be made with the environmental impacts of long-term geological storage to ensure that the radiotoxicity and net radioactive burden to the biosphere were really decreased by reprocessing (OECD-NEA, 2001; Wydler and Baetsle, 2000).

During the early stages of the Cold War in the 1950s and 1960s, an intense nuclear arms race gave rise to very large scale processing of nuclear fuel to extract plutonium. This fuel was irradiated in purpose-built reactors. Virtually no attention was paid to the environment in what many on both sides perceived to be a contest for survival. The outcome was very large quantities of poorly stored and badly characterized reprocessing wastes needing costly remediation in locations like Hanford in the US and Chelyabinsk in Russia. Therefore, the issue is whether reprocessing is fundamentally an environmentally risky business or whether it was just done badly in the Cold War period but now is sustainable in a civilian context.

La Hague (France) pursues a rather open approach and regularly publishes the occupational exposure of its staff and analyses of seafood, local produce and other biosphere radiation measurements. La Hague's record of minimal environmental impact, at least in the short term, indicates that civilian nuclear fuel reprocessing can be done in an environmentally responsible manner.

Transportation

Reprocessing requires the transportation of nuclear fuel from the reactor sites to the reprocessing plants and the subsequent transportation of the products (plutonium, uranium and fission products) in the case of commercial plants, back to their countries of origin. This is now done routinely throughout the world using massive containers to shield persons from the high levels of radiation from their contents. These containers are specially engineered to withstand the worst rail and road accidents that can be hypothesized. Severe tests such as long exposure to high temperature fires and dropping from heights onto an unyielding surface, and onto hardened steel pins, must be passed before the containers are approved for use. Spectacular tests such as train crashes have also been carried out (Wydler and Baetsle, 2000).

Security in the transportation of materials such as plutonium is also stringent. Certain government agencies specialize in securing such shipments and Japan, for example, has built armed vessels especially for the transport of materials that might be used in weapons.

The record on transportation has been excellent in terms of both radiation safety and security. Evidently transportation to and from reprocessing facilities can be done safely and the primary issue is likely to be the cost of specialized transportation facilities and personnel rather than technical feasibility.

Reprocessing safety

The early days of reprocessing saw many accidents within the plants. The organic solvents used must be carefully handled and fires can occur even in operations such as encasing low level radioactive wastes in bitumen as happened in 1995 at TRP (Japan).

It was found that as the technology matured that the key to preventing these incidents is a continuous emphasis on safety culture among the employees combined with a high degree of quality control. The consequences of such accidents are mitigated by the design of modern reprocessing plants, which retain and contain any radioactive materials released within thick-walled vault structures (IAEA, 2004; Salvatores et al., 2004; Wydler and Baetsle, 2000).

A particularly damaging reprocessing accident occurred in 1957 at Kyshtym, some 80 km from Chelyabinsk. Kyshtym was a waste area associated with the military plutonium production operation at Chelyabinsk (Russia). A powerful chemical explosion occurred in a concrete waste storage tank containing 80 tones of fission products and MAs. Failure of the tank's cooling system caused a mostly solidified mixture of sodium acetate and sodium nitrate to ignite. About 20 million curies of radioactivity were released contaminating an area of 15,000 km². For comparison, the Chernobyl reactor accident released about 50 millions curies. This Soviet era accident was not officially reported to the IAEA for some 30 years.

There is another type of accident unique to nuclear facilities known as a criticality accident. This occurs when a mixture of nuclear materials and usually water is put together in a confined space accidentally creating the conditions for a chain reaction. There have been more than 30 such accidents since the beginning of nuclear processing and a particularly severe one took place at Tokai, Japan in 1999. A company known as JCO was preparing enriched uranium fuel and instead of using the carefully designed system of pumps and vessels put in place to avoid criticality, the workers routinely added uranium solutions to a vat with buckets. On the occasion in question this practice resulted in a chain reaction, which went on for some 20 h. Two workers died from the radiation sickness and another one was badly injured. Since this particular building did not have a containment system, residents of the surrounding area were exposed to levels of radiation that were later assessed not to be harmful. The causes of this easily avoided accident were incompetence and stupidity.

The Tokai experience illustrates the reality that even in an advanced industrial country, no amount of safety culture or regulatory supervision can completely eliminate the possibility of accidents. Hence, nuclear facilities such as reprocessing plants must be built to contain and limit any accidental releases of radioactivity within the plant itself. The technology to do this is now widely available and, if deployed, safety although it must always be an important concern would probably not form a technical obstacle to building additional reprocessing plants.

Impact of transmutation on waste management and disposal

In general, the sources of secondary wastes from the MA separation methods are the following:

- Aqueous liquid waste from solvent cleanup,
- α -contaminated solvent,
- Condensate from evaporation and denitration steps,
- Scrub liquors from valency adjustment (in case of Np),
- Different liquid and solid wastes from Np, Am (Cm) product preparation,
- Solid contaminated wastes from Np, Am-Cm target preparation.

Elimination of ^{237}Np during the advanced reprocessing operations will slightly increase the secondary waste as the stripping of Np from the Np-Pu stream takes place in the second Pu-purification cycle. However, most of these waste streams can be recycled to the HLW acid recovery step and mixed with the concentrated HLW. Very little is presently known about the secondary waste generation resulting from the introduction of Am-Cm separation methods.

Once separation-conditioning and target fabrication processes have been developed and demonstrated on a pilot scale, transmutation by irradiation in dedicated facilities can theoretically be envisaged. Since this implies also the simultaneous development of dedicated irradiation facilities (FRs or ADSs), it will take several decades before this option could become available. However, the impact of the transmutation option starts as soon as the targets and/or fuels are ready for irradiation, which may be much earlier than the implementation of dedicated irradiation facilities. The safety and security of this option requires much attention (OECD-NEA, 2002, 2001; IAEA, 2004; Wydler and Baetsle, 2000).

Transmutation of separated MA and, more generally, TRUs makes their handling and storage operations in engineered storage facilities more difficult. It increases the heat load of the waste and generates secondary waste. These drawbacks have to be weighed against the very long term benefit of decreasing the dose to humans in the very distant future (millions of years). However, it gives the present generation an ethical guarantee that the present nuclear legacy will not unduly burden future civilizations. By transmuting TRUs the very long term hazard will be decreased, but it still has to be compared with the natural actinide decay chains of uranium (depleted and reprocessed).

Proliferation risk and proliferation resistance of nuclear waste management options

Proliferation risk and proliferation resistance of nuclear waste management options have to do with protection measures of fuel cycle facilities and with presence of nuclides of concerns in reprocessing streams. Inherently avoiding fuel cycle scenarios with separated plutonium streams and keeping TRU together mitigates the associated risks. Furthermore, high radioactivity levels of reprocessed used fuel streams poses tremendous accessibility challenges. Research efforts are in progress to enhance waste stream characterization capabilities (National Research Council, 2011).

Acknowledgments

This chapter uses materials derived from the openly available peer reviewed publications, textbooks, government reports, and online resources. The utilized major information sources are listed in the references section. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

See Also: Advanced Fuel Cycles Identification and Evaluation; Assessment of the Relative Environmental Footprint of Nuclear Energy and Its Fuel Cycle; Fuel Rod, Fuel Assembly, and Reactivity Control Assembly Design; International Nuclear Waste Disposal Centers; Light Water Reactors (LWR) Fuel Cycle Options; Main Repository Projects Worldwide; Nuclear Waste Can Be Disposed of Safely; Nuclear Waste Can Be Reduced by Recycling and Transmutation; Routes for Pu Multi-recycling; Self-Sustaining Breeding in Advanced Reactors: Characterization of Natural Resources; Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors; Self-Sustaining Breeding in Advanced Reactors: Comparison of Fuel Cycle Performance; Social Acceptability of Geologic Disposal; The Concept of Geological Disposal of Highly Radioactive Nuclear Waste.

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Relevant websites

- <https://www.energy.gov/ne/fuel-cycle-technologies/fuel-cycle-research-development>—DOE Fuel Cycle Research & Development.
- <https://www.energy.gov/ne/initiatives/spent-fuel-and-waste-disposition>—DOE Spent Fuel and Waste Disposition.
- <https://www.gen-4.org/gif/>—Generation IV International Forum.
- <https://www.iaea.org/resources/databases/advanced-reactors-information-system-aris>—IAEA Advanced Reactors Information System (ARIS).
- <https://www.iaea.org/resources/databases/application-of-safeguards-to-geological-repositories>—IAEA Application of Safeguards to Geological Repositories (ASTOR).
- <https://www.iaea.org/topics/nuclear-fuel-cycle>—IAEA Nuclear Fuel Cycle.
- <https://www.iaea.org/topics/radioactive-waste-and-spent-fuel-management>—IAEA Radioactive Waste and Spent Fuel Management.
- <https://www.world-nuclear.org/Information-Library.aspx>—World Nuclear Association, Information Library.

Worldwide Status of Advanced Reactors (GEN IV) Research and Technology Development

Alexander Stanculescu^{*}, Idaho National Laboratory, Nuclear Systems Design and Analysis Division, Idaho Falls, ID, United States; and Generation IV International Forum (GIF)

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Glossary

B&B Breed-and-Burn reactor
BREST-OD-300 Fast reactor with lead (Pb) coolant (Russian abbreviation)
Doppler feedback Reactivity feedback effect due to temperature changes
ELFR European lead (Pb) fast reactor
FHR Fluoride salt-cooled high-temperature reactor
GEN IV Generation Four
GFR Gas-cooled Fast Reactor
GHG Green House Gas
GIF MOU GIF memorandum of understanding
GIF PA GIF project arrangement
GIF SA GIF system arrangement
GIF Generation IV International Forum
Gt 10⁹ metric ton
GW 10⁹ watt
IEA International Energy Agency
LFR Lead-cooled fast reactor
LWR Light-water reactor
MCFR Molten chloride fast reactor
MOSART Molten salt actinide recycler and transmuter
MPa 10⁶ Pa
MSFR Molten salt fast reactor
MSR Molten Salt Reactor
MWe 10⁶ watt electrical
MWth 10⁶ watt thermal
NEA Nuclear Energy Agency
OECD Organization for Economic Co-operation and Development
Power Density Average fission power per unit core volume
R&D Research and development
R&D&D&D Research, development, demonstration and deployment
SCWR Supercritical Water-cooled Reactor

^{*}Retired.

SFR Sodium-cooled Fast Reactor
SiCf/SiC Silicon carbide fiber-reinforced/silicon carbide
SMR Small and medium-sized reactor (also Small Modular Reactor)
Specific Power Average power per unit fuel weight
SSTAR Small secure transportable autonomous reactor
TRISO Tristructural isotropic
UCO Uranium oxi-carbide
UO₂ Uranium-dioxide
VHTR Very-high Temperature Reactor
Void coefficient Describes the reactivity change due to coolant voidening

Background

From currently 6.7 billion people, the world's population is expected to grow to over 9 billion by 2050. As recognized in the United Nations Resolution 70/1 "Transforming our world: the 2030 Agenda for Sustainable Development" (United Nations, 2015), providing "universal access to affordable, reliable and sustainable energy" and in particular to reliable electric energy, is central to human development, social stability and even world peace.

When looking ahead at the next 50 years or so, it should be reminded that currently, about 2.8 billion people are still lacking modern energy services (International Energy Agency, 2017), and more than one billion do not have access to electricity. Not only the growth of worldwide population, but also the economic growth in developing countries will lead to further increase of the demand for energy. There will most likely be diverse energy sources co-existing side by side. These various sources will have to be integrated into "smart" (i.e. efficient and flexible) energy systems catering to various applications and needs. The share of fossil energy sources (mostly coal and natural gas) will decrease, albeit slowly, while its impact on the Green House Gas (GHG) emissions will be mitigated by the introduction of carbon capture technologies.

The seriousness of climate change and the related importance of having abundant carbon-free energy sources available to replace the world's current dependency on GHG emitting energy sources are understood. Therefore, it is recognized (United Nations, 2000) that a paramount objective of sustainable energy policy is the need to reduce GHG emissions, in particular carbon-dioxide.

Since 1971, the release of approximately 60 Gt of carbon-dioxide has been avoided by nuclear energy generation (International Energy Agency, 2014). Currently, nuclear energy is avoiding the release of about 1.7 Gt carbon-dioxide per year. While producing no carbon emissions during electricity generation, nuclear energy has a very small life-cycle carbon footprint originating from fossil fuels used during mining, manufacturing, and transportation of materials and components.

Moreover, nuclear energy can help smooth the integration of renewable, albeit intermittent capacity into electric grid systems. Consequently, going forward, the IEA projects that nuclear will play a cornerstone role in climate-friendly scenarios. For the strategy that limits global temperature increase to 2 °C, nuclear capacity would more than double by 2050, representing 17% of worldwide electricity production. In this scenario, nuclear would be the technology that is maximizing cumulative carbon-dioxide emissions avoidance (International Energy Agency, 2015).

In spite of its potential, the current situation of the nuclear energy sector is difficult. The nuclear energy share of global primary energy production (currently just over 4%) is steadily shrinking, as is also the worldwide nuclear energy share of commercial electricity production (International Energy Agency, 2019a). New build projects are almost entirely limited to a handful of countries in Asia, and to Russia. New build projects require high investments, and their financing is introducing considerable financial risks. Therefore, fewer investments are canalized into the nuclear sector. This is also the case (with few exceptions) for public investment related to nuclear energy Research and Development (R&D).

There are massive global investments in alternative (renewable) energy concepts and the development of the related infrastructure. For example, a recent IEA publication (International Energy Agency, 2019b) is forecasting a 50% increase of the global renewable power capacity (corresponding to 1200 GW) over the 2019–2024 five-year period. This increase will be realized mostly thanks to the implementation of distributed photo-voltaic technology installed on private, commercial and industrial structures. Contrasting this trend, nuclear energy is considered by many as "not renewable" and is losing its ground as the sustainable, practically zero-carbon footprint option it really is. The reasons for this overall negative assessment are a combination of socio-economic and technological perceptions. Nuclear energy is being perceived by many as a complex and expensive technology that is no longer necessary, given the advent of "renewable" solar and wind energy; as unacceptably dangerous (since the Fukushima Dai-ichi accident); and also as polluting (radioactive waste). Moreover, in many administrative, political, financial and academic circles, proponents of nuclear energy are met with distrust, seen as lacking social responsibility, accountability, and transparency.

More specifically, there are various barriers faced by innovative nuclear research, development, demonstration and deployment (R&D&D&D) efforts, e.g. (International Framework for Nuclear Energy Cooperation/Nuclear Energy Agency, 2016): negative social and political perception of nuclear energy safety characteristics; uncertain and changing political support coupled with explicit governmental support for renewables in some countries; insufficient carbon pricing to promote nuclear investments along with

electricity market structures (unstable electricity prices in liberalized markets) that do not provide investment signals for low-carbon technologies; the long-term nature of capital investments in the nuclear sector, along with very high costs, technical risks, as well as past and current nuclear built budget and schedule overruns.

In order to reverse these perceptions and the worldwide continuous decline in funding for long-term research and technology development that these perceptions provoke, it is imperative that the nuclear energy option regain public trust and acceptance by continuously demonstrating social responsibility and uncompromisingly adopting the wider sustainability criteria – environmental soundness, safety, security of supply, costs, and non-proliferation (Bisconti, 2021; Golay, 2021; Parsons, 2021; Rothwell, 2021; Ingersoll, 2021; Shellenberger, 2021; Kadak, 2021; Cheng, 2021; Hore-Lacy, 2021).

Based on the recognition that, on the one hand side, nuclear energy is losing political support and, more fundamentally, the understanding of and legitimacy with a wide-ranging fraction of society, and firmly convinced, on the other side, that nuclear energy, as a practically zero-carbon footprint option, has a crucial role to play as part of a worldwide long-term sustainable energy mix, it must be concluded that radically novel nuclear energy systems must be developed, offering options that are both acceptable to society and attractive to the investor in the energy market. Since the industrial revolution, it was arguably proven that, in all technological areas, such a goal can only be achieved through innovation, which, in turn, is achieved through R&D to fill knowledge gaps, adapt the technology to new requirements, and foster inventions. On the other hand, it must be acknowledged that the required investments to resolve the technological barriers and R&D risks are large and beyond the means of most individual organizations and many countries. This recognition provides the basic rationale for the R&D&D&D efforts undertaken within the framework of the Generation IV International Forum (GIF) to develop “4th generation” (GEN IV) nuclear energy systems.

Section 3 of the “New Encyclopedia of Nuclear Energy” is presenting the worldwide status of advanced and innovative nuclear energy systems research and technology development. It is structured around four topical blocks and one cross-cutting block. The four topical blocks comprise the

- i. Six GEN IV systems
- ii. Advanced LWRs
- iii. Breed-and-Burn (B&B) reactors
- iv. Small and Medium-sized Reactors (SMRs).

The cross-cutting block comprises

- i. Advanced modeling and simulation
- ii. Deployment challenges and opportunities.

The fluoride salt cooled high temperature reactor concept and the micro reactor concept can also be considered, for practical reasons, to fall into the SMR category.

Status of GEN IV nuclear energy systems development

Introduction

The GIF is providing a multinational collaborative framework for R&D efforts aiming at establishing the feasibility and performance capabilities of innovative GEN IV nuclear energy systems that meet the objectives of improved sustainability, economics, safety and reliability, as well as proliferation resistance and physical protection (GIF, 2014).

The 12 GIF partners (Australia, Canada, European Union, France, Japan, Republic of Korea, People’s Republic of China, Russian Federation, South Africa, Switzerland, United Kingdom, and United States) actively participating in the GIF are pursuing technological innovation and advanced nuclear reactor design efforts toward meeting the aforementioned objectives.

The R&D focus of the GIF partners is documented by their respective commitment to the various GIF System Arrangements (SA), Memoranda of Understanding (MOU), and/or Project Arrangements (PA), specifically for the

- Sodium-cooled Fast Reactor (SFR): European Union, France, Japan, Republic of Korea, People’s Republic of China, Russian Federation, and United States
- Very-high Temperature Reactor (VHTR): Australia, Canada, European Union, France, Japan, Republic of Korea, People’s Republic of China, Switzerland, and United States
- Gas-cooled Fast Reactor (GFR): European Union, France, and Japan
- Molten Salt Reactor (MSR): Australia, European Union, France, Russian Federation, Switzerland, and United States
- Lead-cooled fast reactor (LFR): European Union, Japan, Republic of Korea, Russian Federation, and United States)
- Supercritical Water-cooled Reactor (SCWR): Canada, European Union, Japan, People’s Republic of China, and Russian Federation.

The GIF partners are investigating nuclear energy cost reduction potentials by studying simplified designs that are addressing new market opportunities and delivering (in addition to electricity) heat, hybrid and dispatchable energy systems, as well as by increasing thermal efficiencies. The latter is achieved by adopting high-temperature coolants like liquid metals, helium, liquid salts, or super critical water. In addition to increasing thermal efficiencies, such coolants are also widening the application range of nuclear energy to industrial heat applications, thus displacing industrial fossil fuel burning. With regard to enhanced sustainability, GIF’s

GEN IV nuclear energy systems, when combined with innovative fuel recycling options, are improving uranium utilization by a factor of 70–100, and also reducing both volume, radiotoxicity and heat load of the waste to be disposed of in the final deep geological repository.

Recognizing the strategic importance of hybrid energy systems with an ever-increasing share of intermittent energy sources and demanding energy storage and supply flexibility, the GIF partners are addressing dispatchable energy concepts in both electricity and heat applications.

With regard to safety, GIF's GEN IV nuclear energy systems are maintaining and improving the current stringent safety standards. Moreover, and as an example for GIF's institutional innovation efforts, the GIF partners are pursuing activities aiming at the harmonization of regulatory requirements (Nuclear Energy Agency, 2011). By the same token, a GIF Task Force on defining safety design criteria and safety guidelines for GEN IV sodium cooled fast reactors designs is striving to engage regulators and share results of the safety research conducted in the field of such advanced GEN IV reactor concepts.

Another example for the institutional and organizational innovation pursued within the GIF framework is the effort to investigate new ideas and business models for engaging private capital in the development of small modular reactors and GEN IV systems. With this, the partners in the GIF collaboration framework are acknowledging the reality of strong support coming from venture capital (proof of investment leaders' confidence in the potential of the nuclear energy option) to numerous start-up companies that are developing innovative SMR designs.

GEN IV nuclear energy systems

The original GIF Roadmap (GIF, 2002), which was updated in 2014 (GIF, 2014), identified six nuclear energy systems (viz. the Sodium-cooled Fast Reactor (SFR), the Very-high Temperature Reactor (VHTR), the Gas-cooled Fast Reactor (GFR), the Molten Salt Reactor (MSR), the Lead-cooled fast reactor (LFR), and the Supercritical Water-cooled Reactor (SCWR) that were the most promising to meet the stated objectives (improved sustainability, economics, safety and reliability, as well as proliferation resistance and physical protection) for a nuclear system to qualify as "4th generation."

Fig. 1 (GIF, 2018) provides an overview of the generations of nuclear power: from the early prototype reactors to the revolutionary designs of the "fourth generation." The time ranges indicated in this figure correspond to the design and the first deployments of the respective reactor generation. For a more detailed historical overview, the reader is referenced to (Schwageraus, 2021).

Table 1 (Pioro, 2016) provides a summary of the main design characteristics of the various reactor concepts that are being investigated by the GIF partners.

Each of the GEN IV nuclear energy systems includes the nuclear reactor, its energy conversion systems, and the respective fuel cycle technologies.



Fig. 1 The Nuclear Power Reactor Generations.

Table 1 GEN IV nuclear energy systems: main design characteristics.

	Neutron System spectrum	Fuel type	Coolant	Outlet temperature [°C]	Electrical power rating [MW]	Energy conversion cycle	Thermal efficiency [%]	Fuel cycle
SFR	Fast	Mixed U-Pu-minor actinide oxide or metal; U-Pu-minor actinide-Zr Alloy	Sodium	500–550	50–150 300–1500 600–1500	Indirect steam Rankine; 40 indirect supercritical pressure carbon- dioxide Brayton gas turbine		Closed
VHTR	Thermal	TRISO-coated particles (pebbles or compacts)	Helium	750–950 (ultimately 1000)	250–300/thermal power 10–625 MW in co-generation mode	Indirect Rankine steam; > 55/>80 direct helium Brayton; (electrical indirect helium- nitrogen cycle; hydrogen co- generation mode)		Open (closed fuel cycle potential with U-Th fuel)
GFR	Fast	Mixed U-Pu oxide /carbide/nitride	Helium	850	1200	Indirect combined (helium-nitrogen and steam circuit); direct Brayton helium turbine; indirect Rankine steam	> 50	Closed
MSR	Thermal Fast	U, Th-U fluorides U-Pu, Th-U, Pu fluorides	Same as fuel (fluoride salts)	700–800	1000	Brayton gas turbine (carbon-dioxide); indirect steam Rankine	50	Closed
LFR	Fast	Mixed U-Pu oxide; Mixed U-Pu-nitride (both options with/ without minor actinides)	Lead	480–570	20–180 300–1200 600–1000	Indirect Rankine steam / super-heated steam; gas turbine Brayton (carbon-dioxide)	41–43	Closed
SCWR	Thermal Fast	U-oxide; Th-Pu oxide Mixed U-Pu oxide	Water	510–625	300–700 1000–1500	Direct super-heated once-through steam	45–50	Open Closed

For illustration purposes, [Figs. 2–7 \(Pioro, 2016\)](#) are showing overview sketches of some GEN IV nuclear energy systems.

While the original high-level four GEN IV objectives remain valid, the updated 2014 GIF Roadmap acknowledged that the GEN IV nuclear energy systems (i) have the primary mission of generating power, (ii) have the capability to deliver non-electric products like process heat and hydrogen, (iii) aim at minimization of waste, and (iv) offer the flexibility needed for cost-effective integration into global low-carbon energy systems.

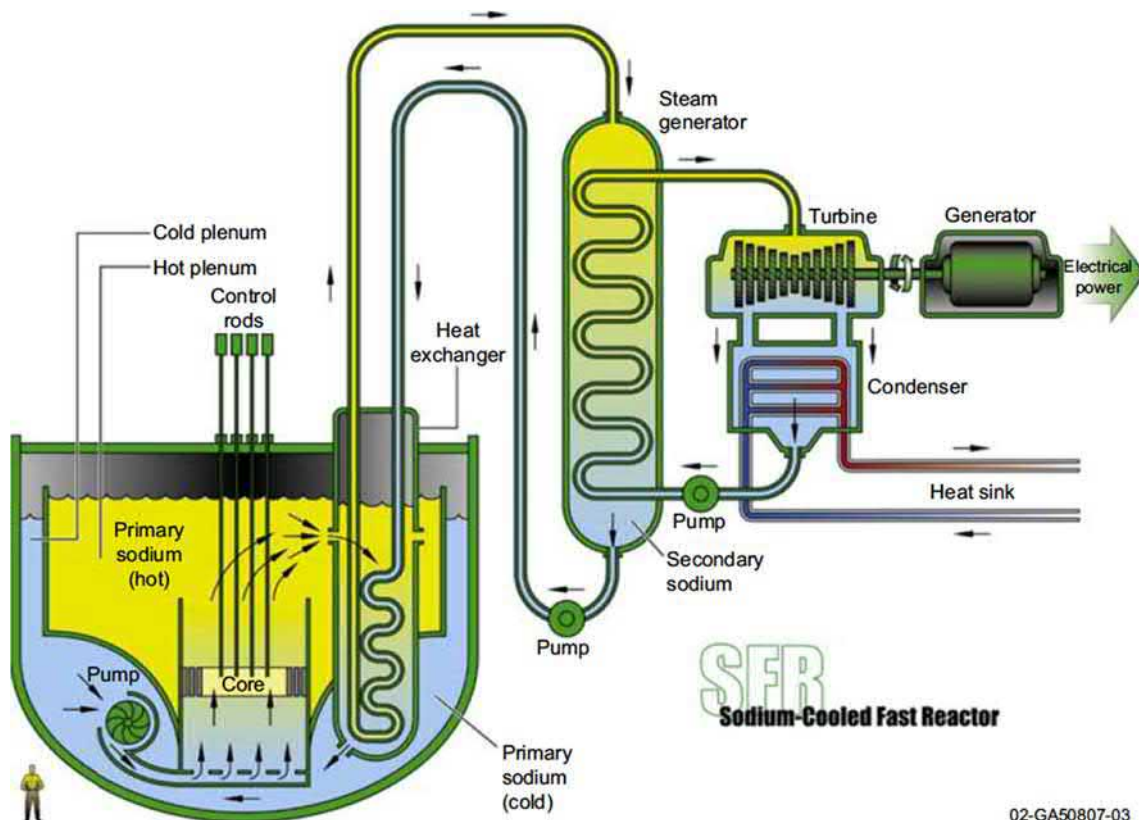
All six systems deliver power. VHTR, GFR, LFR and MSR are higher temperature systems and offer process heat and hydrogen production capabilities. All fast neutron spectrum systems (SFR, GFR, LFR, and the fast SCWR concept) and the MSR offer the capability to transmute actinides and thus are pushing the waste management objective.

Given their role in a hybrid energy world with an ever-increasing share of intermittent energy sources, the GEN IV systems will have to respond to broad flexibility requirements in terms of operational flexibility (compatibility with hybrid systems, island mode operation, diversified fuel use), deployment (scalability, siting, constructability), and energy products (power, process heat). Last but not least, in an effort to enhance market attractiveness, a metrics to evaluate the “flexibility” and design maturity of GEN IV nuclear energy systems is needed that will also take into consideration the impact on the deployment of GEN IV nuclear energy systems of factors other than cost (e.g. electricity market characteristics, national policies, products/services requirements, etc.) ([Electric Power Research Institute, 2018](#)).

The following sub-sections are providing brief descriptions of the design characteristics and major operational and safety characteristics of the six GEN IV systems. Detailed information, in particular in terms of the major design objectives, as well as the challenges and opportunities on the path to commercialization can be found in [Heidet \(2021\)](#), [Fütterer et al. \(2021\)](#), [Hatala \(2021\)](#), [Ignatiev \(2021\)](#), [Alemberti \(2021\)](#), and [Zang and Huang \(2021\)](#) for the SFR, VHTR, GFR, MSR, LFR, and SCWR, respectively, as well as in ([Sadhankar, 2021](#)). Moreover, the reader wanting in-depth technical immersion into the GEN IV systems is referred to ([Pioro, 2016](#)).

Sodium-cooled fast reactor (SFR)

The salient characteristic of the GEN IV sodium-cooled fast reactor (SFR) is its capability to utilize the transuranic actinides for energy production. This permits vastly extending the primary energy resource uranium, and, at the same time, reducing the



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Fig. 2 The pool-type GEN IV SFR design with an indirect steam turbine Rankine power cycle.

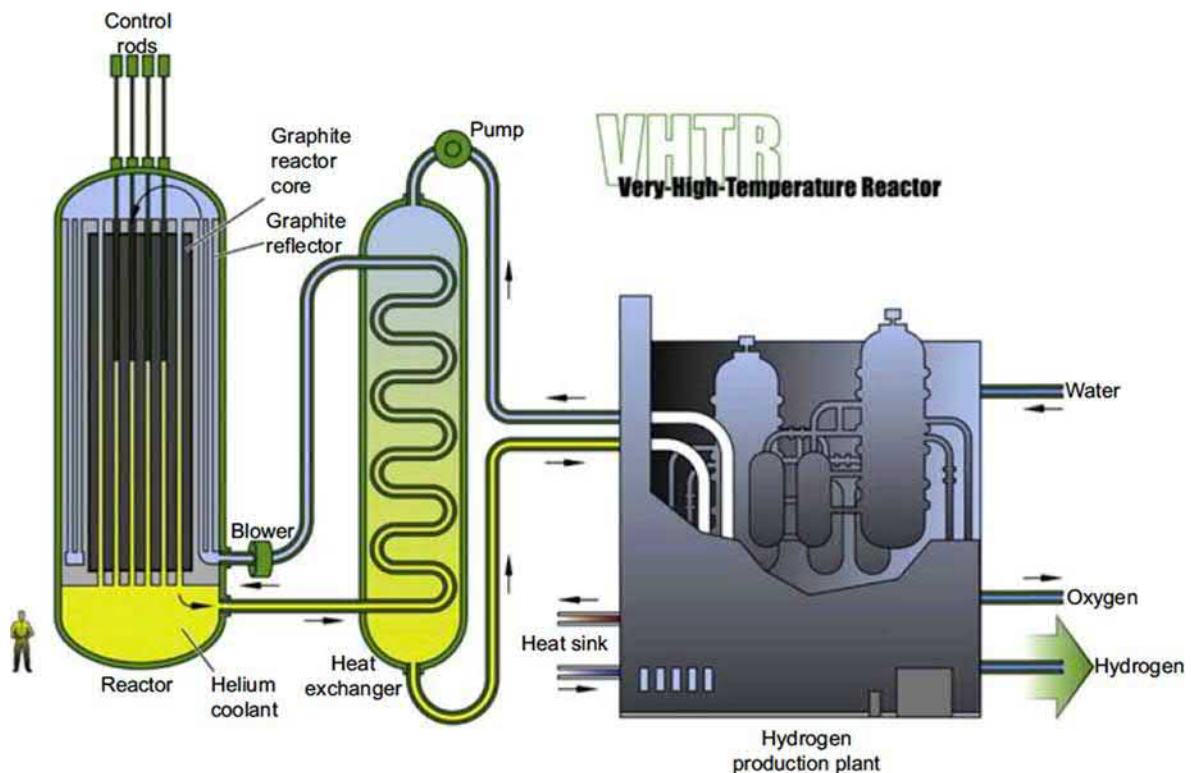


Fig. 3 The prismatic fuel block GEN IV VHTR reactor concept with hydrogen co-generation.

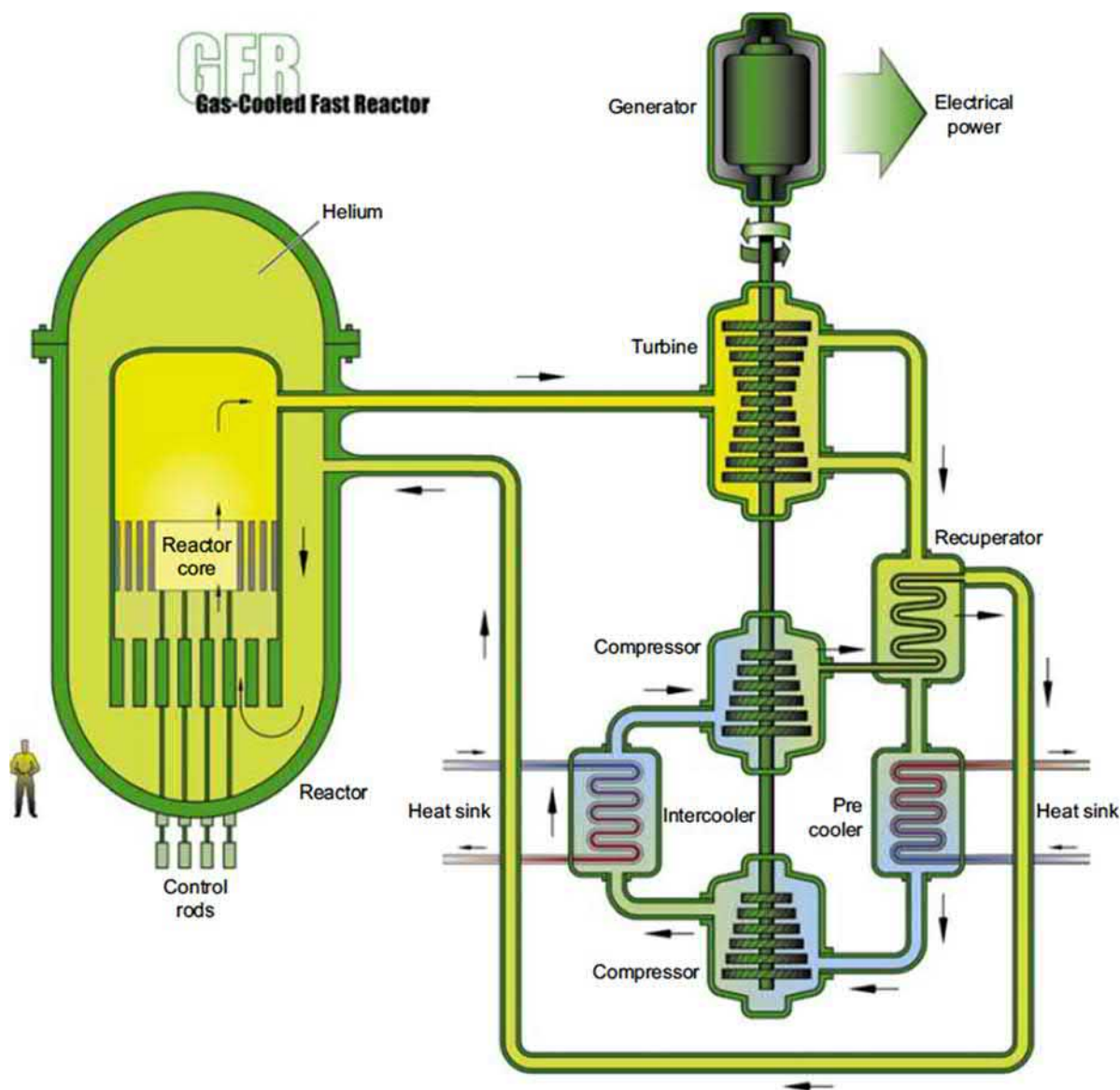


Fig. 4 The GFR GEN IV concept with direct gas turbine Brayton power cycle.

volume, radiotoxicity of, and heat load in the waste to be disposed of in the final geologic repository. To meet these fuel cycle management objectives of regenerating fissile material and enhancing the utilization of minor actinides, the SFR must operate in a closed fuel cycle (Grenèche, 2021). Accordingly, the development and qualification of safe and cost-effective fuel recycle technologies is a major challenge addressed by the GIF partners on the road toward industrial deployment of SFRs.

The unique characteristics of the SFR are due to the fact that it operates with fast neutrons. It is cooled by liquid sodium and lacks moderating materials. Given the cooling properties of sodium, the coolant volume fraction in the core of the SFR core is low compared with a light water-cooled thermal reactor, while the core power density as well as specific power are much higher. Because of the chemical reaction of sodium with oxygen, the SFR requires a sealed coolant system which guarantees an oxygen-free environment that is preventing corrosion.

For the power conversion system, water/steam, nitrogen, and supercritical carbon-dioxide are considered as working fluids. SFR power rating options range from 50 to 300 MWe small modular reactors to larger plants of up to 1500 MWe.

There are two fuel recycle technology options being pursued, viz. the advanced aqueous, and the pyrometallurgical reprocessing option (Taylor, 2021; Bourg, 2021; Miguiditchian, 2021). Advanced mixed (uranium-plutonium-minor actinides) oxide, and mixed metal alloy are the primary fuel types for the advanced aqueous reprocessing option, and the pyrometallurgical reprocessing option, respectively. Mixed nitride fuel that can be recycled by both advanced aqueous and pyrometallurgical reprocessing methods is also being considered.

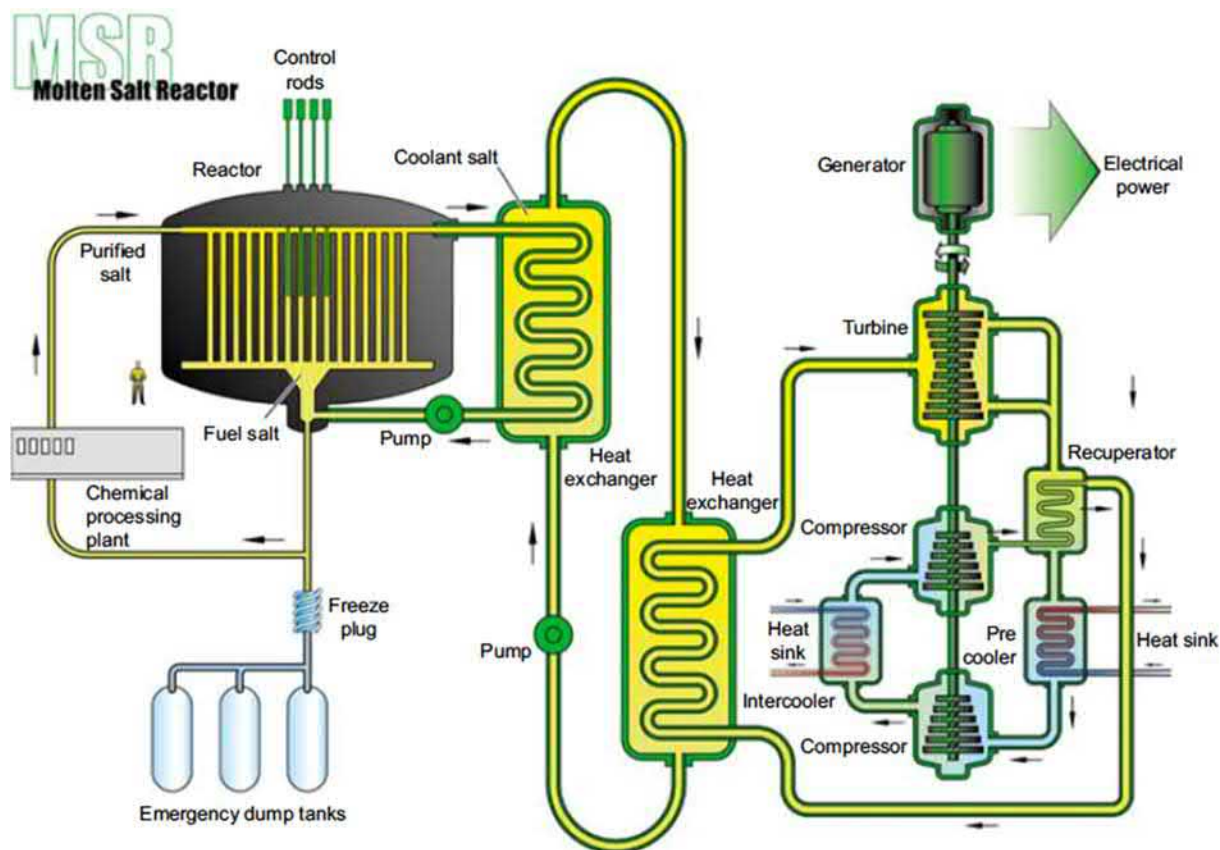


Fig. 5 MSR concept with indirect Brayton power cycle.

The major safety features of the GEN IV SFR allowing to achieve a high safety level through inherent and passive means and to accommodate transients and bounding events with significant safety margins, are long thermal response time, reasonable margins to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system.

Very-high temperature reactor (VHTR)

The salient characteristics of the GEN IV very-high temperature reactor (VHTR) are the fully ceramic coated TRISO particle fuel, the graphite moderator, and the helium coolant.

The TRISO fuel is composed of small kernels of ceramic fissile material (UO_2 or UCO) that is coated with pyrocarbon, silicon carbide, and pyrocarbon layers. Many experimental and operational performance tests have proven that the three-layer coating design provides an excellent barrier against fission product release under both normal operational and accident conditions. The kernels are surrounded by a porous carbon buffer and assembled either into pebbles or prismatic blocks. The shape of the VHTR fuel defines the two design concepts being pursued, viz. the pebble-bed core and the prismatic-block core.

Various engineering-scale tests and the operation of prototype high temperature gas-cooled reactors have demonstrated the VHTR concept's inherent safety characteristics. All VHTR cores are designed to ensure passive decay heat removal.

The long-term objective of the GEN IV VHTR development efforts is a reactor delivering heat at about 850°C (that would require a reactor outlet temperature around 1000°C) to be used by a hydrogen production facility (based on, e.g., the sulfur-iodine thermochemical process). R&D efforts are also pursued to develop innovative heat transfer technologies allowing to reduce the reactor outlet temperature requirements (also in view of ongoing efforts to reduce the hydrogen production temperature requirements).

In parallel, the recognition that process heat in form of steam at less than 600°C is currently representing a considerable market (globally in the order of several hundred GWth, almost entirely supplied by fossil fuelled power plants), short-term efforts are pursued to develop less ambitious high temperature gas cooled reactors (having core outlet temperature around 750°C) for industrial applications, as a zero-carbon alternative to the fossil-fuelled power plants.

Accordingly, in the short-term, GEN IV VHTR concepts could be designed having core outlet temperatures in the range 700 to 950°C (for which there are available high-temperature alloys employed in heat exchangers and other reactor metallic components) to deliver both power and process heat. In the long-term, however, higher core outlet temperatures would require the development of innovative materials (e.g. ceramics, super-alloys, etc.).

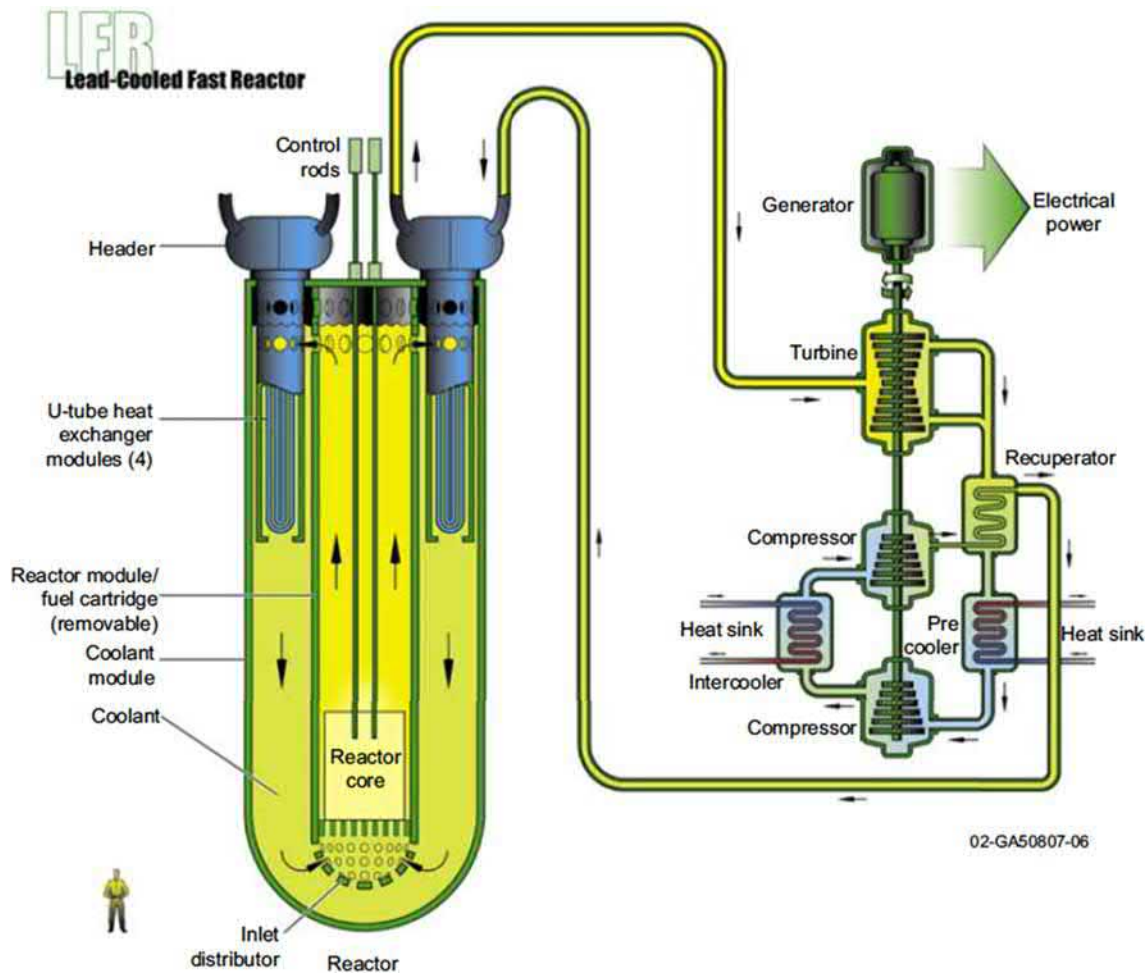


Fig. 6 LFR concept with indirect Brayton power cycle.

Gas-cooled fast reactor (GFR)

The salient characteristic of the GEN IV gas cooled fast reactor (GFR), which is put forward as a longer-term alternative to the sodium cooled fast reactor, is its single-phase inert gas coolant. GFR's coolant is chemically inert, does not dissociate or become activated in the neutron field, and is transparent. This latter characteristic, in addition to simplifying coolant handling, is facilitating inspection and repair operation.

The power density of the GFR core is relatively high. Its core outlet temperature is also high, in the range of 800–850 °C, opening up non-power applications in addition to the sustainability improvements linked to the closed fuel cycle.

The GFR coolant void coefficient, while still positive, is small, and the core's safety characteristics are dominated by the negative Doppler feedback.

The reference concept for GFR is a 2400 MWth nuclear power plant featuring a break-even core with 850 °C outlet temperature, which allows adopting an indirect combined gas-steam cycle that is achieving approximately 48% efficiency. The primary circuit with three loops is contained within a secondary pressure boundary that constitutes a guard containment against depressurization accidents. The heat is converted into electricity in the indirect combined gas-steam cycle with three gas turbines and one steam turbine. A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle containing a helium-nitrogen mixture that drives a closed cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator that is then used to drive a steam turbine.

The major challenges for the GFR are linked to the development of the fuel, cladding and structural materials meeting the requirements of a high-power density, high outlet temperature fast neutron core (i.e. 1000 °C normal operation clad temperature, no fission product release at 1600 °C clad temperature during a few hours, ensuring the core cooling capability up to 2000 °C clad temperature). The reference GFR core design consists of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hex-tube. Currently, the reference material for both the pin clad and the hex-tubes is silicon carbide fiber-reinforced silicon carbide (SiCf/SiC).

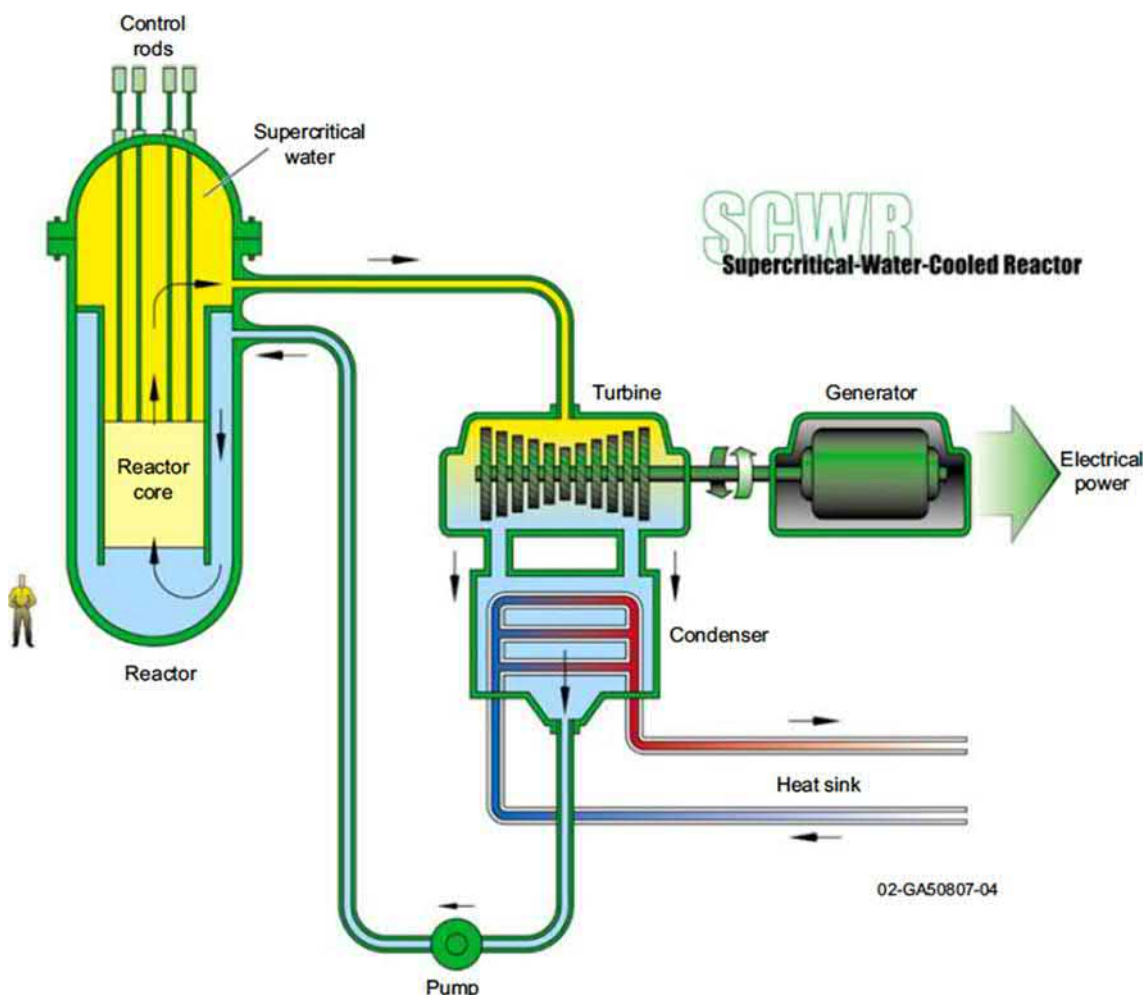


Fig. 7 The SCWR GEN IV concept with direct steam turbine Rankine power cycle.

The GFR developers are considering that an experimental reactor rated at less than 100 MWth is needed to demonstrate the GFR viability and to qualify specific GFR technologies such as the refractory fuel elements and the specific safety systems, in particular the decay heat removal function. Current activities within the GIF framework are focused on this objective.

Molten salt reactor (MSR)

Molten salt reactors (MSRs) are broadly grouped in two categories: reactors in which a fuel salt containing fissile material that is fissioning in the core is circulated through the primary system serving both purposes – fissile material and coolant; and reactors fuelled with solid fissile material in which the coolant is a molten salt. In both cases, the neutron spectrum can be thermal, epithermal or fast. Thermal neutron spectrum MSRs usually employ fluoride salts, and their design relies on solid moderator materials within the core. Fast neutron spectrum MSRs do not need moderator materials in the core and can be designed employing fluoride or chloride salts.

The partners collaborating on MSR within the GIF framework are pursuing R&D for both MSR types. The salt-fuelled MSR R&D efforts are concentrated on fast neutron spectrum concepts, with the objective of combining the advantages of fluoride molten salts (viz. low pressure, high temperatures, optical transparency) with the advantages of fast neutron systems (viz. improved fissile utilization and waste management).

R&D on fast neutron spectrum MSRs is focused on three concepts: the 1000 MWe MOSART design with the main objective of recycling actinides; the 1400 MWe thorium-molten-salt fueled MSFR; and the MCFR design that has a very hard neutron spectrum and the objective to require fissile seed only for the start-up core and run without the need to separate the fissile material from the circulating fuel salt.

In theory, the fast neutron spectrum MSRs have a number of advantageous characteristics, when compared to solid fuel fast reactors: large negative reactivity coefficients, lower fissile inventories, a homogeneous isotopic composition of the fuel, no radiation damage constraints on attainable fuel burnup, no used nuclear fuel, no requirement to fabricate and handle solid fuel.

In terms of a solid fuel reactor cooled by molten salt, R&D efforts are focused on the fluoride salt cooled high temperature reactor (FHR). The FHR has a near-thermal neutron spectrum and is adopting a once-through fuel cycle based on low enriched uranium. It relies on ceramic fuel and on low pressure fluoride salt cooling. The FHR is designed to have a high temperature power cycle and fully passive decay heat removal characteristics (Fratoni, 2021).

There remain considerable challenges in terms of demonstrating the viability and performance of MSR concepts, requiring long-term research and technology development efforts addressing, e.g.: salt physical, chemical and thermodynamic properties; system design and safety studies; development of advanced materials, studies of their compatibility with molten salts, and their performance in high neutron and temperature fields; corrosion and tritium release prevention technologies; development of efficient techniques for gaseous fission products extraction from the fuel salt; chemical processing flowsheets; development of safety, safeguards, security and proliferation resistance approaches applicable to liquid-fuelled reactors.

Lead-cooled fast reactor (LFR)

The salient characteristics of the GEN IV lead cooled fast reactor LFR are its fast neutron spectrum (comparable to the SFR) and closed fuel cycle aiming at improved fissile utilization and waste management. Given the properties of the lead coolant (very high boiling temperature, chemically inert, operating at near atmospheric pressure), the LFR has the potential to achieve considerable design simplification compared to the SFR and very good safety characteristics.

The R&D efforts within the GIF framework are focused on three LFR options: a large 600 MWe reactor (ELFR) meeting central station power generation objectives; an intermediate size 300 MWe reactor (BREST-OD-300), and a small transportable reactor rated in the 10–100 MWe range Small Secure Transportable Autonomous Reactor (SSTAR) featuring very long core lifetime and offering the possibility to provide power and heat to remote locations. For all the LFR options, the objective is to achieve at least 42% energy conversion efficiency.

The ELFR and the BREST-OD-300 designs feature secondary water loops with superheated once-through steam generators and a steam Rankine cycle achieving energy conversion efficiency well above 40%. The SSTAR design features a supercritical carbon-dioxide power conversion system in a direct Brayton cycle and achieves a similar level of efficiency.

Design simplifications like the elimination of the intermediate cooling system and the adoption of reduced height core components allow the reduction of both capital cost and construction time, thus improving the economics of the LFR.

The BREST-OD-300 reactor uses nitride fuel and adopts a closed fuel cycle with the nitride fuel manufacturing and reprocessing facilities co-located at the reactor site. The BREST-OD-300 design adopts an integrated layout with a multilayer reinforced metal-concrete vessel. The coolant and the main primary circuit components are located in this vessel. There are no coolant valves in the primary circuit. Emergency cooling is ensured by convective circulation, removing the heat directly from the primary circuit to air as the ultimate heat sink. Building on the properties of the lead coolant and of the high-density, heat-conducting uranium-plutonium nitride fuel, the BREST-OD-300 design ensures efficient fuel utilization. In tandem with a small fuel temperature power reactivity effect, this allows reactor operation with a small reactivity margin, which constitutes an important prevention measure against unprotected transient overpower accident initiators. Loss-of-coolant type accidents are prevented by the aforementioned integrated reactor layout.

The major design and operational features of the reference 20 MWe SSTAR design are molten lead coolant, nitride fuel containing transuranic elements, fast neutron spectrum, natural circulation, and transportable reactor vessel. These design characteristics constitute an unique approach to proliferation resistance by enabling a long core lifetime, autonomous load following, simplicity of operation, reliability, transportability, and a high degree of passive safety.

Supercritical water-cooled reactor (SCWR)

The salient characteristics of the GEN IV super-critical water-cooled Reactor (SCWR) is the high temperature, high pressure water coolant above the thermodynamic critical point of water (374 °C, 22.1 MPa).

Most SCWR designs are developed for power generation and are rated above 1000 MWe. The SCWR operates at pressures of about 25 MPa and reactor outlet temperatures up to 625 °C. The SCWRs could generate electricity with thermal efficiencies ranging from 43 to 48%, as compared to 32% for the current fleet of nuclear reactor systems. Moreover, the relatively high SCWR core outlet temperature offers the opportunity for co-generation, such as hydrogen production. The SCWR coolant does not change phase, allowing the design to adopt the direct power cycle without moisture separator, reheaters and recirculation pumps as in the boiling-water reactors, or steam generators as in the pressurized light-water and heavy-water cooled reactors. These design simplifications are vastly reducing the size of the SCWR nuclear power plant, allowing a balance-of-plant configuration that is the same as that of a fossil-fired power plant.

Because of the potential advantages of the SCWR (viz. improved economics due of the high thermodynamic efficiency, and the potential for plant simplification), the GIF partners are pursuing R&D for two types of SCWR concepts: the pressure-vessel and the pressure-tube concept. Other than specifics of the core design, these concepts have many similar features (viz. the outlet pressures and temperatures, thermal neutron spectra, steam cycle options, materials, etc.). Improvements in the areas of safety, sustainability, proliferation resistance and physical protection are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

The SCWR has a long-term vision for light water reactors that requires significant development in technical areas (such as materials, chemistry, thermal hydraulics, neutronic, instrumentation, control, etc.). At the same time, the SCWR benefits from the interest in LWRs as well as an established technology for supercritical water power cycle equipment in the fossil-fired power industry. The

path forward for the SCWR efforts within the GIF framework consists in demonstrating the SCWR viability and designing and operating a prototypical fuelled test loop, thus preparing the grounds for designing and deploying a prototype reactor. The main R&D activities include system Integration and assessments, materials and chemistry, as well as thermal hydraulics and safety analyses.

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Sodium-Cooled Fast Reactors

F. Heidet, Argonne National Laboratory, Lemont, IL, United States

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Glossary

BG Breeding Gain is defined as $BR-1$; that is the measure of the extra fissile material in the used nuclear fuel relative to the fissile inventory in the initial fuel loading.

Breeding ratio (BR) Defined like the CR and usually applied to situations in which the CR is equal to or larger than unity.

Conversion ratio (CR) The ratio of fissile material inventory in the used nuclear fuel to the fissile material inventory loaded into the core. Usually used when this ratio is smaller or equal to unity.

Electron-volt (eV) A measure of the neutron j = kinetic energy; $MeV = 10^6$ eV.

Gigawatt-day per initial ton of heavy metal (GWd/t) A measure of the amount of fission energy generated per unit fuel weight.

Reactivity and Reactivity Coefficients “Reactivity” is a term used to denote the extent of the deviation in the core effective multiplication factor (usually denoted as k_{eff}) from its value at a just critical state (at which state $k_{eff} = 1.0$ and the fission-rate is constant). A positive reactivity Coefficient implies that the ratio of the rate of neutron production to the rate of neutron losses (k_{eff}) is increasing due to a unit change in a performance characteristic such as temperature or density. One of the necessary characteristics for a reactor to be inherently safe is that the combined effects of all possible changes in reactor conditions due to a power increase will result in a negative reactivity feedback.

Abbreviations

ADTR Advanced demonstration and test reactor

ALMR Advanced liquid metal reactor

B&B Breed and burn
BDBA Beyond design basis accident
DOE Department of Energy
DOE-NE Department of Energy, Office of Nuclear Energy
dpa displacement per atom
DRACS Direct reactor auxiliary cooling system
EBR-I Experimental breeder reactor I
EBR-II Experimental breeder reactor II
FIMA Fission per initial metal atom
GEN-IV Generation IV
GIF Generation IV International Forum
HLW High-level waste
HTGR High-temperature gas-cooled reactor
IAEA International Atomic Energy Agency
IFR Integral fast reactor
IHX Intermediate heat exchanger
IRACS Intermediate reactor auxiliary cooling system
LEU Low-enriched uranium
LWR Light-water reactor
MeV Mega electron-volt
MHA Maximum hypothetical accidents
MWe Electrical megawatt
MWth Thermal megawatt
PRACS Primary reactor auxiliary cooling system
PRISM Power reactor innovative small module
PWR Pressurized water-cooled reactor
RVACS Reactor vessel auxiliary cooling system
SFR Sodium-cooled fast reactor
SGACS Steam generator auxiliary cooling system
SMR Small modular reactor
SWU Separative work unit
TRISO TRi-structural ISOtropic particle fuel
VTR Versatile test reactor

Introduction

Sodium-cooled fast reactors (SFR) are one of the reactor technologies part of the Generation IV (GEN IV) reactors (GIF, 2018), in Fig. 1. They are considered advanced reactors as opposed to the water-cooled reactors currently used commercially across the world based on their enhanced safety characteristics and additional capabilities. Compared to other advanced reactors, SFRs are one of the most mature technology as they have benefited from many years of technology development and reactor operation. Their benefits have been successfully demonstrated around the world through construction and operation of a number of SFRs. However, early nuclear energy history in the United States led to advancing water-based nuclear reactor systems more than sodium-based nuclear reactor systems.

A bit of history

As with most of the early nuclear energy history, this is taking us to the United States. Liquid metal cooled reactors history starts only a few years after the successful demonstration of the feasibility of a sustainable fission chain reaction in Chicago Pile 1 (DOE, 1982), in 1946, with the Clementine reactor. This reactor was cooled with mercury as it is in its liquid phase at room temperature and atmospheric condition. The Clementine reactor (IAEA, 2013) paved the way to sodium-based systems and led to the construction of the Experimental Breeder reactor (EBR-I) in 1951 (Lichtenberger et al., 1953). EBR-I designed and operated by Argonne National Laboratory, and sited in the state of Idaho, was cooled with a mixture of sodium and potassium. It was the very first nuclear reactor to produce electricity as captured in the photo shown in Fig. 2.

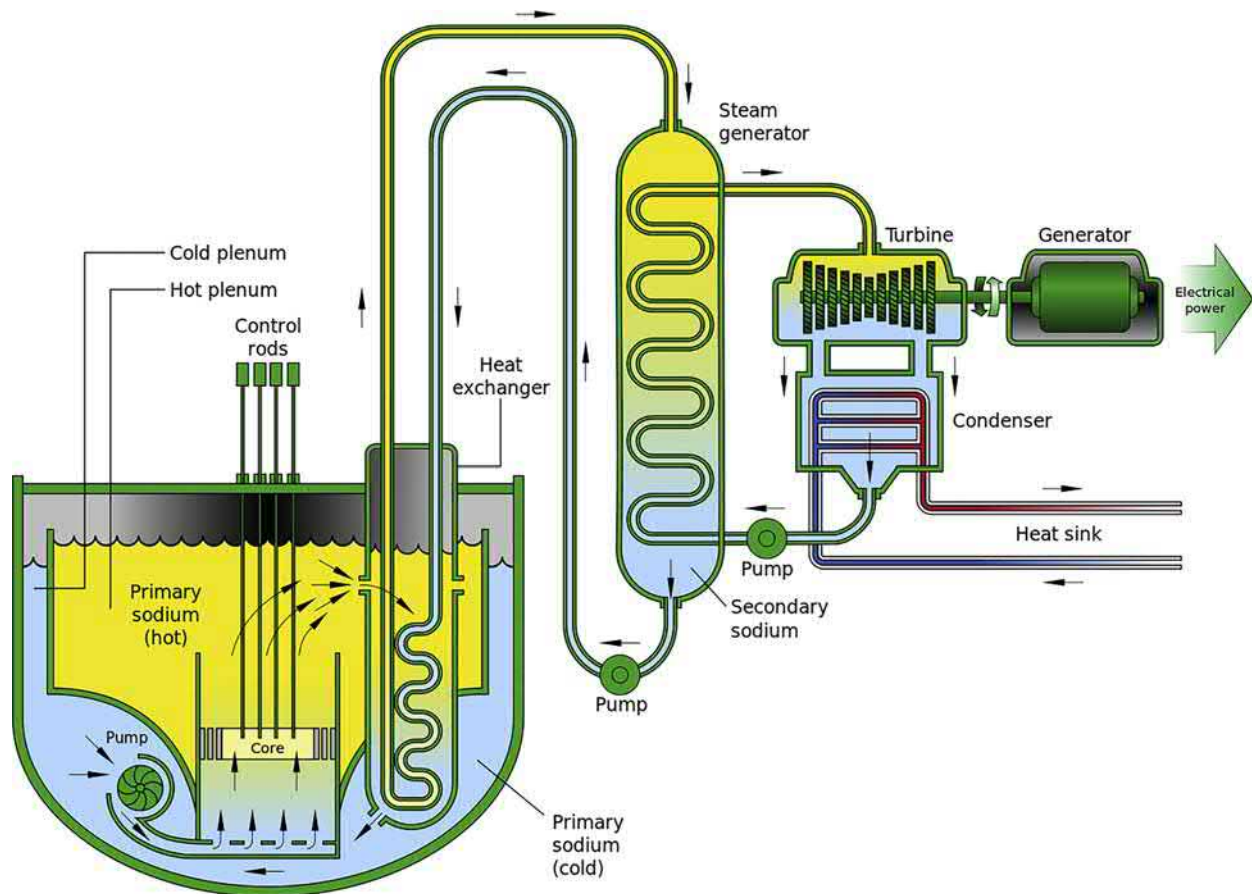


Fig. 1 Sodium-cooled fast reactor general layout (GIF, 2002). https://commons.wikimedia.org/wiki/File:Sodium-Cooled_Fast_Reactor_Schemata.svg

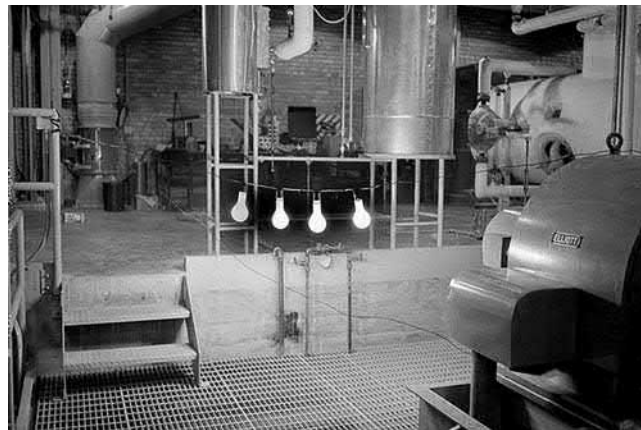


Fig. 2 First light bulbs powered by nuclear energy from EBR-I. https://commons.wikimedia.org/wiki/File:First_four_nuclear_lit_bulbs.jpeg

Shortly after EBR-I, the U.S. Navy which had gained a significant interest in nuclear power and spearheaded a lot of the early technology development, ordered in 1952 the development of a nuclear-powered submarine relying on a SFR. The U.S.S. Seawolf (SSN-575) resulted from this effort and was the second nuclear-propelled submarine in the world, successfully operating for its full planned 30-year life (Friedman, 1994). However, initial challenges with having a sodium system submerged in a water environment led to focusing development efforts on a water-based nuclear reactor for naval applications. With the sodium-cooled systems being less mature than water-cooled systems, commercial application of nuclear power logically leveraged the water-cooled reactor technology due to its higher maturity.

Yet, a subsequent number of experimental, research or demonstration SFR were designed, built and operated around the world. Countries like the United States, the United Kingdom, France, Germany, Russia and Japan established programs to support development of this reactor technology. With many SFRs built and operated (IAEA, 2007), some of these landmark efforts included the

Experimental Breeder Reactor II (EBR-II) in the United States (IAEA, 2013), the Phenix reactor in France (Sauvage, 2006) and the BOR-60 reactor in Russia (Trojanov and Rinejsjij, 1992).

EBR-II, shown in Fig. 3, was built in the United States in 1958 and underwent its final shutdown in 1998. It was a 62.5 MWth/20 MWe pool-type reactor. It was the successor to the very first SFR, EBR-I, and aimed at demonstrating several of the capabilities

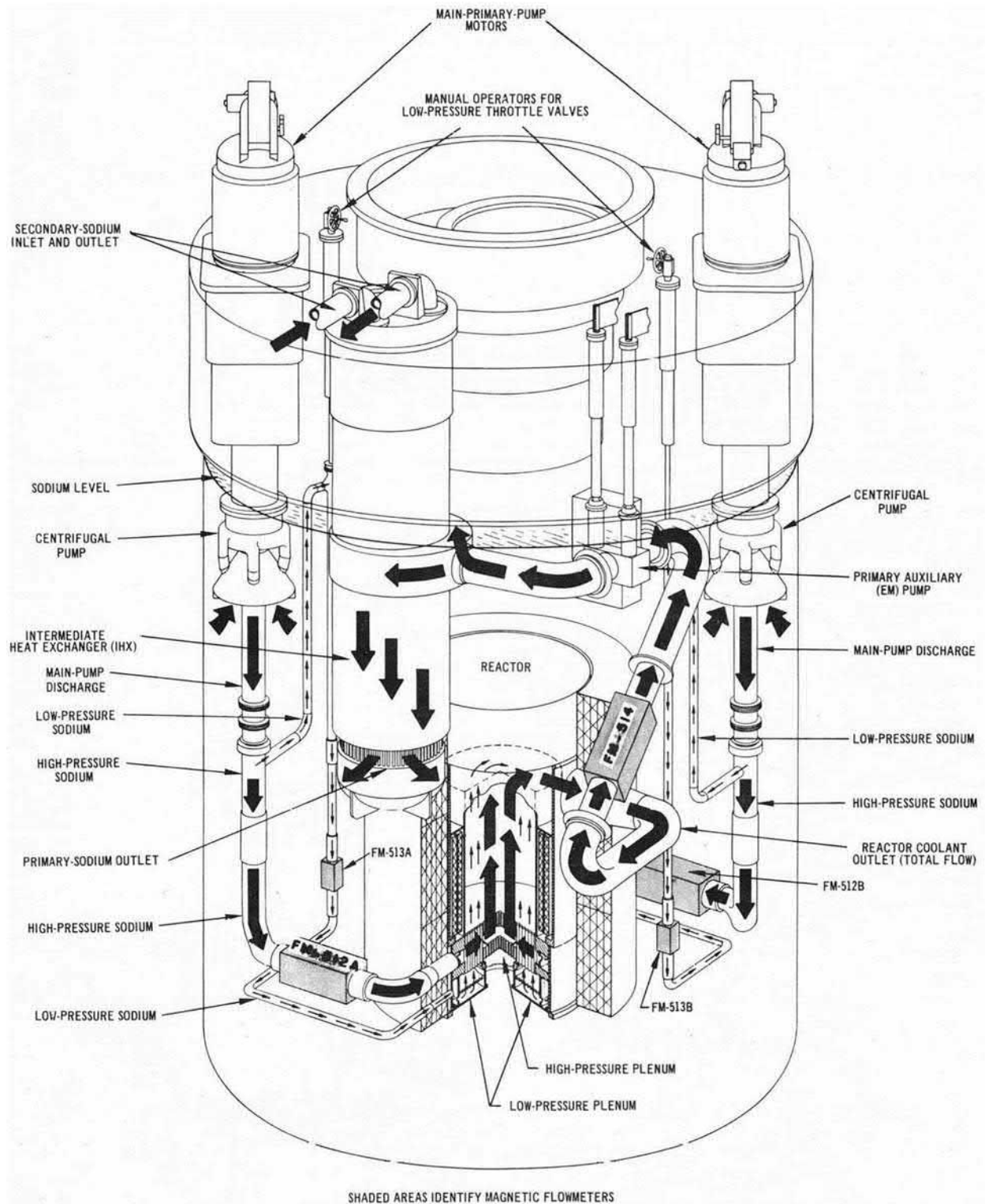


Fig. 3 EBR-II primary vessel and its internals (Keeton, 1984).

enabled by that reactor technology. While initially operated as an experimental reactor, it served as a demonstration reactor for a large fraction of its life. In particular, it supported demonstrated fuel reprocessing, inherent safe behavior of SFRs—including unprotected transients, and possible use of various fuel forms (Imel, 2021). EBR-II was the stepping stone to the Advanced Liquid Metal Reactor (ALMR) (Hardy et al., 1992), part of the Integral Fast Reactor (IFR) project (Till and Chang, 2011). With the ALMR having been developed by Argonne National Laboratory in collaboration with General Electric, when the IFR project got canceled in 1994 due to political reasons, the ALMR concept survived through the Power Reactor Innovative Small Module (PRISM) designed by General Electric Hitachi (Triplett et al., 2012).

The Phenix reactor was built in France in 1968 and underwent its final shutdown in 2009. It was preceded by the Rapsodie reactor (Vautrey and Zaleski, 1962), the first French SFR. Similarly to other countries, Rapsodie was an experimental reactor serving as a stepping stone for the Phenix reactor, which operated as a demonstration reactor. Phenix was a 563 MWth/255 MWe pool-type reactor, although during the latter part of its operational life its power level was limited to 350 MWth/145 MWe. It first served to demonstrate electricity generation at a large scale, and subsequently also served to demonstrate transmutation of nuclear wastes with a SFR. The successor to Phenix was Superphenix (Schneider, 2009), built in 1976. While it operated for many years, Superphenix suffered from poor public opinion and political support, which precipitated its shutdown and closure in 1998, way ahead of its intended lifetime.

The BOR-60 reactor, shown in Fig. 4, was built in Russia in 1964 and is still operating in 2021. It was preceded by the BR-5 and BR-10 reactors (Trojanov and Rinejsij, 1992) which served as stepping stone to gradually increase the power level of the Russian SFRs. BOR-60 is a 60 MWth/12 MWe loop-type experimental reactor the main mission of which is to provide irradiation services to enable nuclear fuel and material development and qualification, benefiting advanced reactor technologies. It also served studying

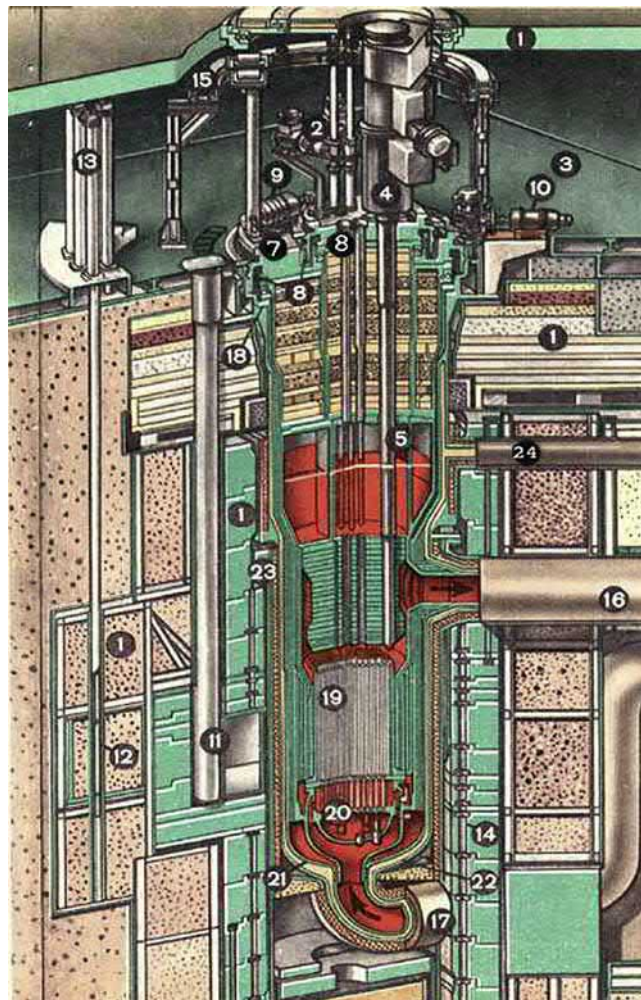


Fig. 4 BOR-60 general layout (IAEA, 2007). Legend: 1-biological shield; 2-driving and control rod gear; 3-reactor pit; 4-junction cylinder; 5-reloading channel; 6-small rotating plug; 7-large rotating plug; 8-plug sealing; 9-small rotating plug gear; 10-large rotating plug gear; 11-external irradiation channel; 12-ionization chamber; 13-ionization chamber gear; 14-shield cooling channels; 15-power supply; 16-exit socket; 17-head socket; 18-basket flange; 19-core and steel screen subassemblies; 20-head collector; 21-frame; 22-casing; 23-reactor support; 24-gas socket.

various steam generator technologies for electricity production. BOR-60 enabled development of subsequent BN-600 demonstration SFR (Trojanov and Rinejsijj, 1992) and the BN-800 commercial SFR (Krivitski, 2001).

Overall, many SFRs succeeded EBR-I and have been developed and operated by various countries. They served to demonstrate the potential applications and benefits of SFRs. These include, but are not limited to, their ability to be passively safe, to significantly increase resource utilization, to take advantage of used fuel, to consume actinides, and to accelerate irradiation testing and qualification for nuclear fuel and materials. Related technologies, such as fuel reprocessing and isotope separation, were also demonstrated in the process.

SFRs remain the most mature and advanced technology among fast-spectrum reactor technologies. At the time of this writing, there are five SFRs operating in the world and one more currently on stand-by as listed in Table 1. Half of these are located in Russia, which include two relatively large size reactors producing sizable amounts of electricity.

SFR basics

Sodium-cooled fast reactors are relying on fast (i.e., unmoderated) neutrons to sustain the chain reaction in the core. In contrast, in light-water cooled reactors and other non-fast reactors neutrons are first moderated before they contribute to sustaining the chain reaction. This means that SFR and other fast reactors should not contain significant amounts of moderating materials as that would reduce the neutrons energy. In particular, any material containing hydrogen, graphite, or beryllium would act as a moderator and is not typically found in the core of a SFR. Water as a coolant is therefore not a viable option but liquid metals (IAEA, 2012) are a potent way to cool fast reactors as they have a high heat capacity and excellent thermal conductivity, without significantly slowing-down the fission neutrons. Furthermore, liquid metals do not require operating the primary coolant system at high pressure to achieve attractive heat transfer and extract the fission heat at high temperature thus enabling high efficiency for conversion into electricity. While many liquid metals have been considered for fast reactors over time, sodium has been the preferred option due to its characteristics and extensive availability. Sodium is in its liquid phase from 97.8 °C to 882.8 °C at atmospheric pressure and is exclusively composed of ^{23}Na in its natural form. It is non-corrosive with materials typically encountered in nuclear reactors, and remains chemically stable even in the event of fuel constituents being mixed into it. The main drawback of sodium is that when exposed to water, and to a lesser extent to oxygen, it undergoes an exothermic chemical reaction. The reaction with water occurs regardless of the temperature, while with oxygen it occurs only at elevated temperatures. This makes controlling humidity and oxygen contents in SFRs of the utmost importance, in a similar way that controlling impurities in light-water reactors is necessary to avoid unwanted corrosion. This is the primary reason driving SFRs to contain an intermediary coolant loop between the primary loop and the steam plant, when applicable, to avoid potential reactions between activated primary sodium and steam. Sodium can be activated by neutron capture ($^{23}\text{Na} + n \rightarrow ^{24}\text{Na}$ that β^- decays into ^{24}Mg with a half-life of ~ 15 h).

Because they rely on fast neutrons to sustain the chain reaction, SFRs are able to use a larger variety of structural materials than light-water reactors (LWR), since the reactivity penalty (i.e., impact on the reactor neutronic performance, or “neutron economy” (Krepel and Losa, 2021)) is orders of magnitude less than in LWRs. Iron-based alloys such as stainless steels are commonly used in SFRs as structural materials as they are very affordable and well understood materials. For the same reasons, discussed in more details in the following paragraphs, isotopes typically considered as neutron poisons (DOE, 1993a,b) in thermal reactors, are mostly inconsequential in SFRs and other fast reactors. Of particular relevance is the ^{135}Xe isotope. This is a well-known fission product isotope, produced as part of the fission reactions, which has an excessively large cross-section in the thermal energy range, Fig. 5, and therefore leads to significant operational constraints for thermal reactors (e.g., limiting power ramp rates, restart availability) (DOE, 1993a,b). For fast reactors, this isotope does not have any impact besides being a fission gas, possibly contributing to fuel swelling, enabling flexible load-following capabilities in fast reactors.

Fast neutrons lead to additional differences with thermal spectrum reactors, two of which are sometimes considered as potential drawbacks. The first one is that a larger fraction of fissile material, that is uranium enrichment in case of low-enriched uranium (LEU), is required to make the core critical. This is not as much of a drawback at it may initially appear because SFRs can operate at a much higher specific power than thermal spectrum reactors, meaning that the power generated per unit mass of fuel is higher than in most of other reactors (Waltar et al. 2012). Also, the fuel in SFRs can be driven up to about twice the burnup levels than in

Table 1 List of currently active SFRs (IAEA, 2007).

Name	Type, coolant	Power, MW	Configuration	Fuel type	Country	Start date
		Thermal/electric				
BOR-60	Experimental	55/10	Loop	Oxide	Russia	1969
BN-600	Demonstration	1470/600	Pool	Oxide	Russia	1980
BN-800	Commercial	2100/864	Pool	Oxide	Russia	2014
FBTR	Experimental	40/13	Pool	Carbide	India	1985
CEFR	Experimental	65/20	Pool	Oxide	China	2010
Joyo	Experimental	140/—	Loop	Oxide	Japan	1978 (stand-by since 2007)

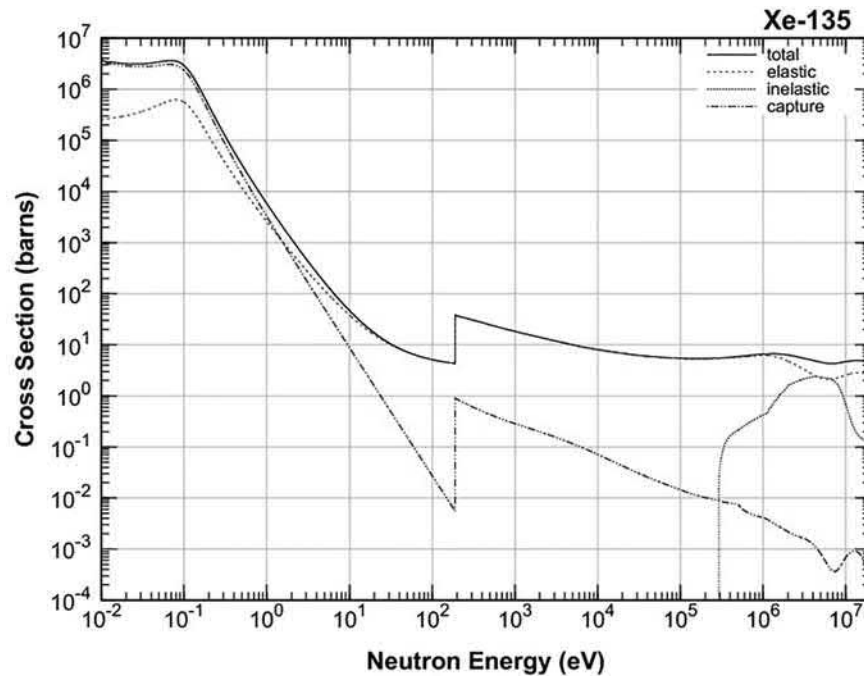


Fig. 5 Xe-135 microscopic cross-sections. Cahalan J, Fanning T, Farmer M, Grandy C, Jin E, Kim T, Kellogg R, Krajtl L, Lomperski S, Moiseyev A, Momozaki Y, Park Y, Reed C, Salev F, Seidensticker R, Sienicki J, Tang Y, Tzanos C, Wei T, Yang W and Chikazawa Y (2007) *Advanced Burner Reactor 1000Mw Reference Concept*. Argonne National Laboratory, ANL-AFCI-202. On page 22, Figure II.1-1.

LWRs and can be recycled, so that the resulting natural resource utilization can be nearly two orders of magnitude higher than in contemporary LWR. The second potential drawback is that shielding of fast neutrons can require larger shield volumes and thicknesses than when shielding for thermal neutrons (Waltar et al. 2012). This is primarily a consideration for the smallest units considered (micro-reactors and small-modular reactors (SMR)) where compactness of the overall system can be of the highest importance. For SFR systems typically considered for base-load electricity generation, shielding, while an important aspect, is not typically a major limitation.

Reactor physics aspects of fast versus thermal neutrons

Some basic reactor physics discussion is necessary to appreciate the differences between thermal and fast neutron systems (DOE, 1993a,b; Waltar et al., 2012). Neutron having an energy above 0.1 MeV (100,000 eV) are generally considered fast neutrons, and those with an energy below 0.625 eV are considered thermal neutrons. In a simplified way, neutrons arising from a fission event have an energy distribution peaking around 0.5–1.0 MeV (Klay, 2021). When a moderating material is present, the energy of those neutrons is quickly knocked down from the fast energy range to the thermal energy range, which results in spectra observed in thermal reactors with the usual two-hump shape as shown in Fig. 6. When no moderator is present, neutron scattering is still occurring but neutrons lose significant less energy and are typically absorbed or leak out of the core before reaching thermal energies.

Granted that every isotope has different cross-sections, some general trends can be observed when looking at cross-section distribution for commonly used isotopes: cross-section values at fast energies tend to be significantly smaller than at thermal energies but the fission-to-capture probability is higher and the number of neutrons emitted per fission is significantly larger at high energies (Klay, 2021). From those basic observation, we can globally understand some of the key differences between thermal and fast reactor systems, as summarized in Fig. 7. This helps understand why fast reactors require a higher minimum fuel enrichment level, compared to LWRs, to be critical, why they are able to serve multiple roles such as being capable of incinerating (fissioning) minor actinides or breed actinides, why a larger variety of structural materials can be used, and why extremely high fuel utilization can be achieved.

A potential miss-conception is that SFRs require more initial fissile material than LWRs. However, because of their superior neutron economy, SFRs can readily be designed to be breeders; that is to make at least as much fissile fuel as they consume, so no enriched uranium is required after the initial fuel loading; depleted uranium or natural uranium can be used for the SFR makeup fuel. Additionally, SFRs can be designed to have a fissile material requirement, per unit energy, lower than contemporary LWRs.

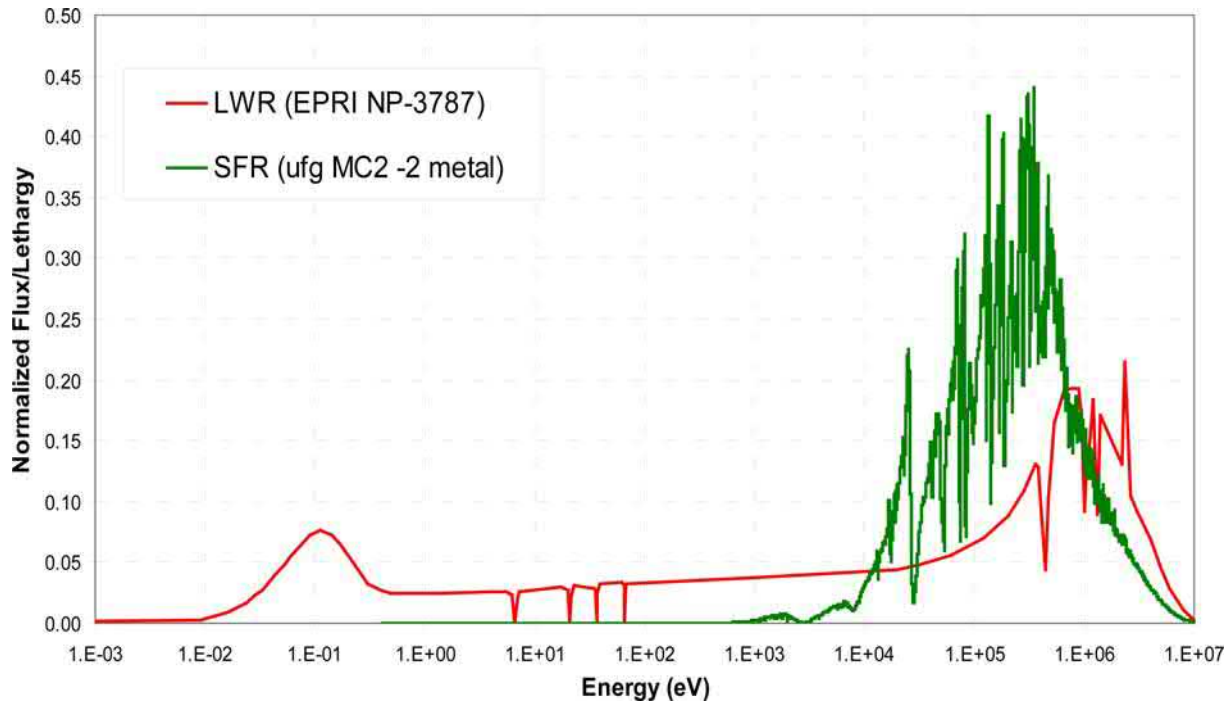


Fig. 6 Representative neutron spectra in a thermal (red) and fast (green) reactor system.

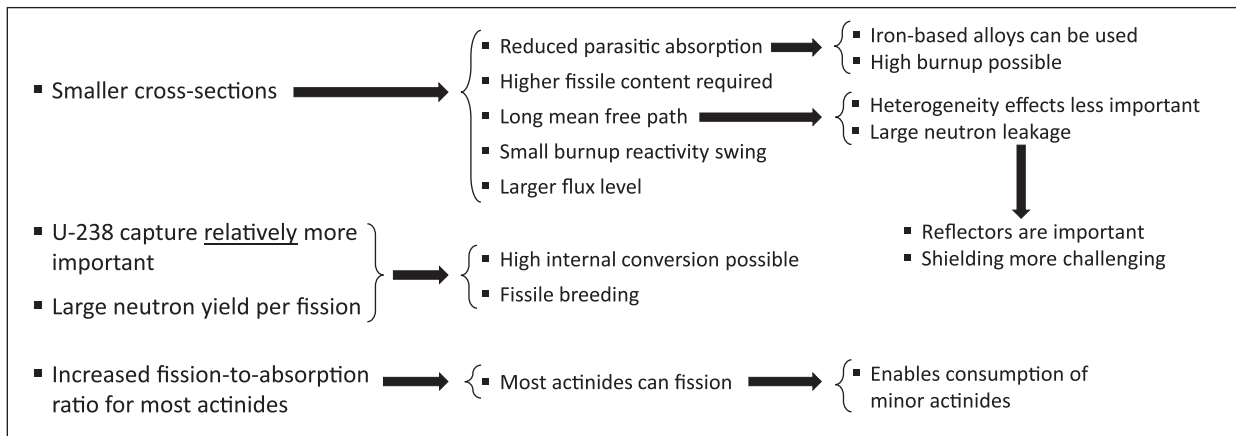


Fig. 7 Causality-consequence relationships for thermal versus fast neutron basic observations.

Major technology description

While at a high level, the key difference between SFRs and LWRs is the absence of a moderator and water being substituted with sodium as coolant, there are a lot more differences when diving into the technology details. Some insight is provided here on the technology details pertaining to the nuclear island, some of the reactor systems, and the safety impacts. It must be noted that only dedicated books (or other supports) will be able to provide a comprehensive description and discussion of those details. Like for all reactor technologies, there are many systems present in SFRs that cannot be highlighted in an encyclopedia chapter. These systems are serving a multitude of purposes, from purification systems to sodium cleanup systems. They aim at increasing the safety, reliability, and maintainability of the power plant.

Core features/options

SFR cores can be designed in many ways due to the flexibility enabled by fast neutrons. A large number of materials, typically ruled out from use in thermal reactors because of their negative impact on the reactor performance, are possible choices for SFRs (Walters et al., 2011). With sodium as the coolant, SFRs are typically expected to operate with coolant temperatures in the range of 300–600 °C

(IAEA, 2007). At those temperatures, special consideration needs to be given to structural material performance and compatibility. Those considerations are unique for each design and as such no additional discussion is provided on this topic in this article.

Almost any fuel type can be used in a SFR (Walters et al., 2011). This goes from the well-known oxide fuels to the less common inert matrix fuels, encompassing metal, carbide, nitride and even coated particle fuels. SFR operating experience primarily pertains to oxide fuels (UO_2 and UPuO_2) as this fuel form has been used the most around the world for nuclear reactors of all types, but other fuel forms have also been used. In particular, metal fuels (Hofman et al., 1997) have been extensively irradiated in SFRs such as the experimental EBR-II because of their enhanced safety performance benefits (FRWG, 2018; Sofu, 2015). While any fuel form, beside those bearing moderating materials (e.g., TRISO), can be used in SFRs, not all those fuel forms are qualified or even mature enough to undergo a licensing review. Characteristics of importance for SFR fuels are fuel forms allowing to achieve high heavy metal loading, good thermal performance and good irradiation stability.

Materials used for structural integrity within the core, including fuel cladding, have typically been steel-based alloys. These materials are preferred due to their compatibility with sodium, relatively low impact on the neutron economy of a SFR, and general availability. Furthermore, they are easy to work with, are relatively inexpensive, and benefit from a tremendous knowledge basis.

Another material of importance in a SFR core is the absorber material, typically used as part of the control elements and potentially in shield components. Noteworthy is that EBR-II did not use an absorber in its control elements, but rather relied on the opposite approach, consisting in withdrawing specific fuel elements when needing to reduce the core reactivity. Although the absorption cross-section of boron is significantly smaller in fast-spectrum systems than in thermal-spectrum systems, it still remains one of the most effective absorber for fast neutrons, especially the ^{10}B isotope. While there are other materials with larger absorption cross-sections in the fast energy range, the commercial availability of B_4C and experience as a control material in thermal reactors makes it the preferred absorber material in SFRs (IAEA, 1996a).

Some typical SFR core layouts are shown in Figs. 8 and 9. It is typical for SFRs to use an hexagonal lattice of fuel assemblies, rather than the traditional square lattice used in LWRs, as it leads to more compact cores featuring high fuel volume fraction. This is enabled by the thermal hydraulics properties of sodium, making it possible to achieve very high-power densities: the footprint of a SFR core is typically smaller than that of LWRs for the same power level. Fuel assembly designs are different across the various SFRs, but are typically made of a duct providing both structural support and a coolant flow path to each assembly. Fuel assemblies, schematically shown in Fig. 10 would contain a number of fuel pins assembled in a triangular array, with space between pins maintained either by spacers or by a wire wrapped around the pin. The fuel pins are made of a cladding holding together the fuel, which can be in the form of pellets or slugs depending on the fuel used.

Other assemblies in a SFR core include control assemblies, reflector assemblies and potentially shield assemblies. Reflector and shield assemblies can be made of reflector/shield rods or other geometries such as block-type components when desirable. The control assemblies, shown in Fig. 11, are typically double-duct systems. The outer duct, being the same as that of other assemblies, fixed in position within the core during normal operations. The inner duct can be moved up and down as needed by the reactor

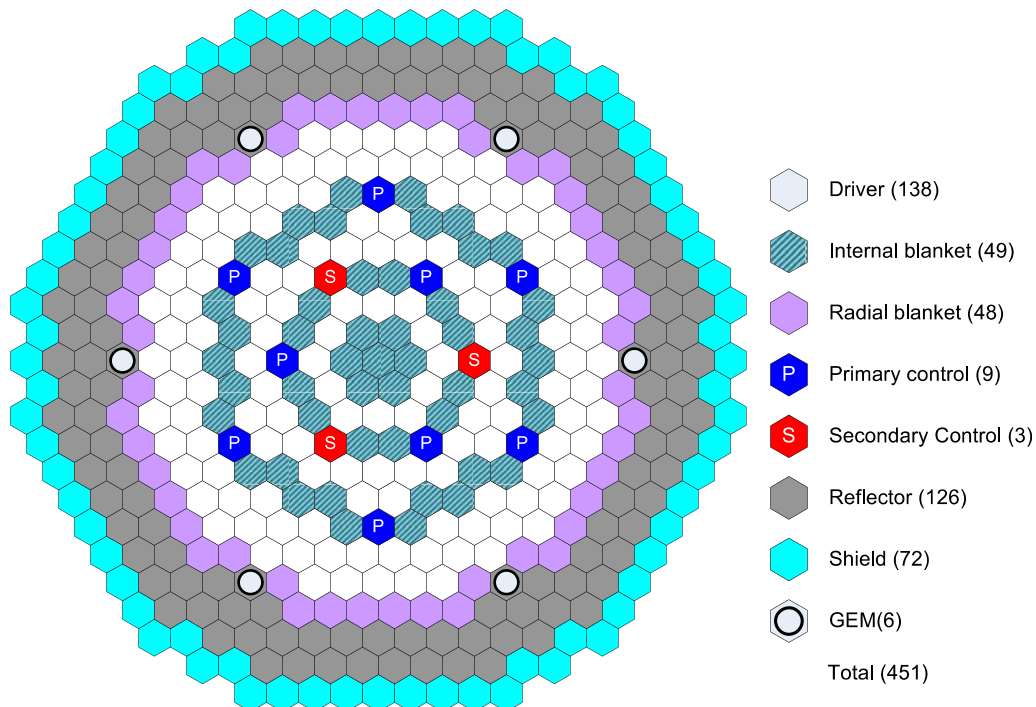


Fig. 8 General Electric S-PRISM core layout, showing a breeding configuration.

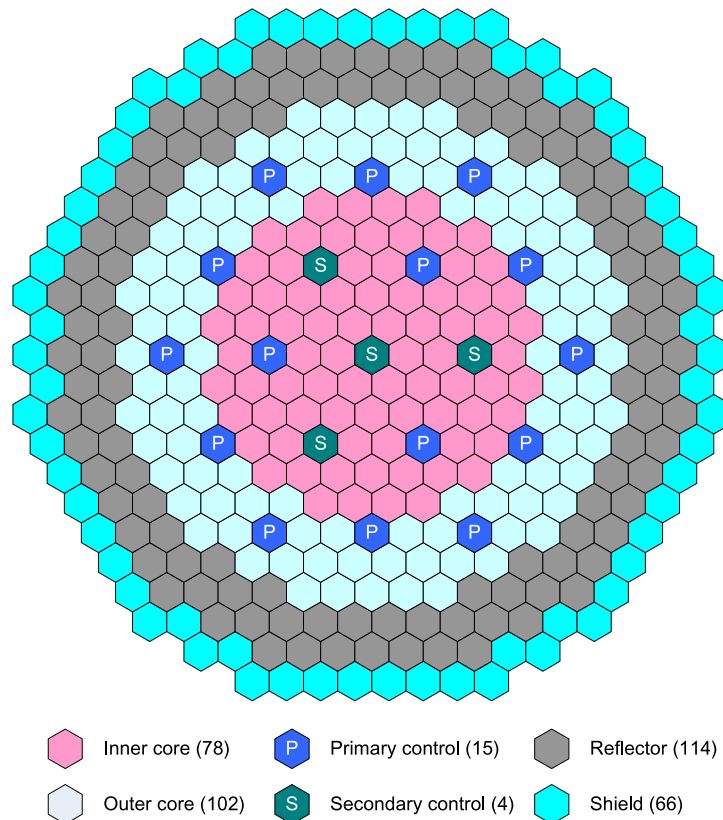


Fig. 9 Argonne National Laboratory Advanced Burner Reactor concept core layout (Cahalan et al., 2007), showing a burner configuration.

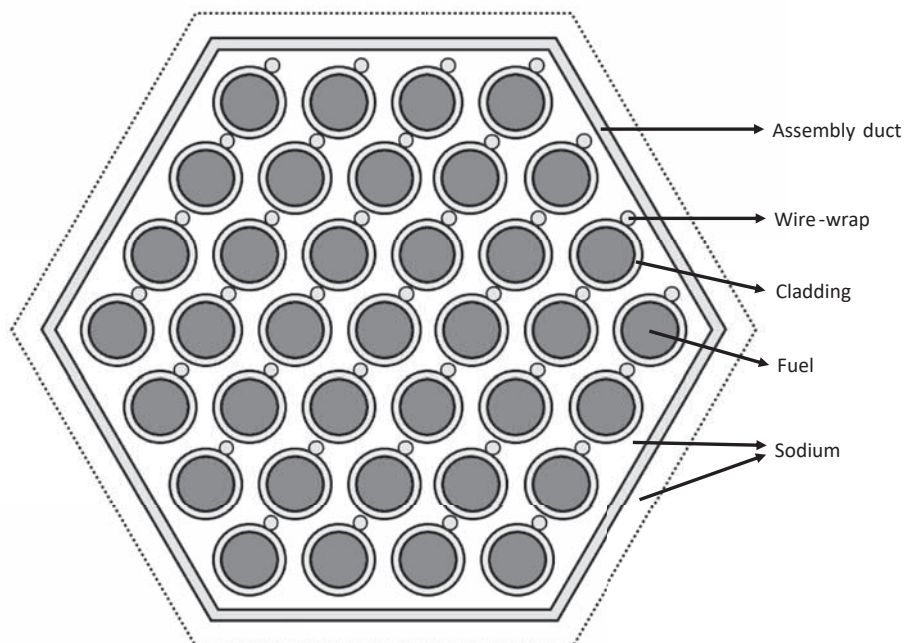


Fig. 10 Cross-cut view of a representative SFR fuel assembly. https://commons.wikimedia.org/wiki/File:LMFBR_schematics2.svg

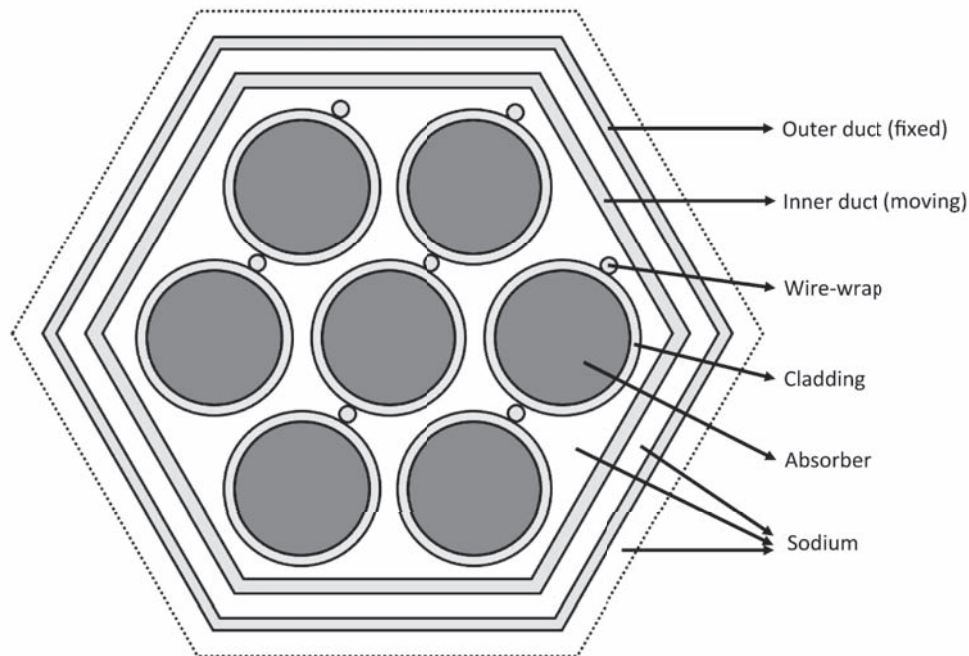


Fig. 11 Cross-cut view of a representative SFR reactivity control assembly. Till CE, Chang YI, Kittel JH, Fauske HK, Lineberry MJ, Stevenson MG, Amundson PI and Dance KD (1980) *Fast Breeder Reactor Studies*. Argonne National Laboratory, ANL-80-40.

control system and control rod driveline mechanism. This inner duct contains an array of absorber rods organized in a triangular lattice. Those rods are made of a cladding containing the absorber material, typically B_4C .

The general assembly arrangement depends on the particular mission tackled. SFRs can be configured as burner, breeders, or break-even reactors (Hoffman et al., 2008). Burner reactors are reactors which are net consumers of fissile material, are able to transmute actinides or consume minor actinides. Breeder reactors are reactors which allow generating more fissile material than they are consuming, or turning a given fissile isotope into a different fissile one (e.g., fueling with fuel bearing ^{235}U to generate ^{233}U in thorium fuel). Break-even reactors, are reactors which are fissile-fuel self-sustaining, without generating more fissile material than they are consuming. These various SFR missions are discussed in more details further down in this article. The characteristics of the assemblies, their arrangement, the number of control assemblies, and the dimensions of the core can vary greatly depending on the targeted SFR mission.

With fast neutrons having a large mean free path, shielding is very important. Two of the main concerns addressed by shielding, in addition to biological shielding, are activation of the primary or secondary sodium and radiation damage of the load-bearing in-vessel components. Sodium activation can lead to high dose rate arising from the sodium circulating outside of the reactor vessel, which would lead to needing excessive shielding around the sodium circuit. Radiation damage, often calculated in displacement per atom (dpa), can lead to deterioration of the mechanical properties of structural materials in the vessel, such as embrittlement. While radiation damage occurs with any reactor technology, this is of a higher concern in fast reactors as neutron flux levels tend to be higher and neutrons are more energetic, leading to higher dpa rate (NEA, 2015). This can typically be addressed through proper shielding design and maintenance planning.

There are two main approaches used for neutron shielding in SFRs. One consists of using an absorber material, such as B_4C , in large enough amounts to achieve the desired effect. The other one consists of using a moderating material, such as graphite, paired with a material absorbing thermal neutrons, such as steel. The tradeoffs of both approaches are complex and depend on the specific reactor design. The first approach could lead to a more compact shielding solution, while the second relies on somewhat simpler materials which may have a cost advantage.

The sodium flows from the bottom to the top of the core, which allows taking full advantage of the natural circulation capabilities of sodium during transient scenarios. The flow rate in each assembly is determined by the hydraulic resistance of the assembly, as typically there is no active mechanism to allocate flow rate to the various assemblies, and all assemblies are hydraulically connected together at their extremities. The hydraulic resistance in different radial parts of the core can be adjusted to approximately match the radial power distribution so that the sodium temperature rise across different fuel assemblies will be as close as practical to the core average. As the coolant exits at the top of the assemblies, it is mixed together and continues through the rest of the primary system towards the heat exchanger.

Nuclear Island configurations

Looking at a SFR from the outside, there are no obvious differences with any other nuclear power plant. Once heat is extracted from the sodium coolant, the technologies used in the rest of the energy transfer system are not unique to SFRs, even though their specifications are tailored to the specific conditions expected. The operating temperatures of SFR are such that thermal efficiencies of around 40% are common when heat is converted to electricity. The heat from the coolant (or that transferred to additional loops) could enable some process-heat applications, but cannot achieve temperatures as high as those achievable with reactor technologies such as high-temperature gas-cooled reactors (HTGR) and molten-salt reactors (MSR) (Stanculescu, 2021). When a SFR is used as a research, test, experimental or demonstration reactor, it is possible that neither an energy conversion system, nor a process heat plant is associated with the reactor, and that the heat is simply rejected into the atmosphere.

Inside the containment building (IAEA, 1996b; Grandy et al., 2016), there are two different plant configurations commonly used for SFRs (Waltar et al., 2012): the loop-type and pool-type configurations illustrated in Fig. 12.

The loop-type configuration is conceptually similar to the general configuration of LWRs. The core components are tightly contained in a fitted vessel and the primary coolant system exits the vessel to reach the intermediate heat exchanger (IHX). This layout still requires a secondary coolant loop as sodium in the primary system is highly activated and in the very unlikely event of a leak from the primary to the secondary coolant system, reaction between radioactive sodium and water will be prevented.

The pool-type configuration consists of a large vessel containing the entire primary coolant system, including the pump(s), IHX(s) and other components. It is labeled as a “pool” as all the components are submerged in sodium. In that configuration, the secondary coolant loop passes through the vessel boundary.

The major benefits and drawbacks of both configurations are summarized in Table 2. Maintenance of the systems is made easier in the loop configuration as fewer components are submerged in the primary sodium, making accessing them and replacing them a somewhat simpler endeavor. The sodium inventory in the loop configuration is significantly less than that of the pool configuration which results in a smaller overall mass of the primary system. Shielding of the secondary sodium, that in the secondary coolant loop, is made easier in a loop configuration as the IHX can be located as far as needed from the reactor core and shielding to that effect can be placed outside of the vessel, in-between the vessel and the secondary coolant system.

The pool configurations are less compact but also offer more space within the vessel. This provides more flexibility to incorporate the various components needed in the system. It can also simplify fuel handling operations, and even include used fuel storage locations within the vessel. The main benefit of the pool configurations has to do with the safety aspects. Pool-type configurations are virtually immune to loss of coolant from the primary system, offer a significant heat sink which leads to highly favorable transients and make passive heat removal systems simpler to integrate.

There is no clear preference across the SFR community regarding the pool-type versus loop-type configurations (IAEA, 2007). While certain countries seem to favor one over the other, this is possibly due to the experience accumulated with the configuration selected for their first SFR, making it preferable to retain that configuration for subsequent SFRs. Understanding the application intended for a SFR design can drive the preference towards one configuration or the other. For instance, a SFR to be operated as a test reactor might favor a pool configuration due to the enhanced inherently safety behavior and the extra space available to handle experiments.

Reactivity control and shutdown systems

While some of the early SFRs relied on a single reactivity control system, the current standard is to include two independent reactivity control systems (U.S. NRC, 2018). The first one, commonly referred to as the reactivity control system, is used to control the core reactivity level and therefore its power level over the entire range of operational conditions. It can also be used to single-handedly bring the reactor from any anticipated operational condition into a safe sub-critical state. The second reactivity control system, commonly referred to as the safety or shutdown system, is not used to control the power of the reactor and its sole role is to provide an independent means to quickly bring the reactor from any anticipated operational condition into a safe sub-critical state.

Both systems are usually made of several assemblies containing an array of absorber pins, which are able to slide into fixed assembly duct locations. The absorber material of choice is B_4C due to its availability and extensive knowledge basis. However, other absorber materials are sometimes considered such as Hf_xH as they can provide benefits such as a lower rate of reduction of the reactivity worth with absorber burnup.

Due to the low burnup reactivity swing occurring in SFRs over the course of a cycle, burnable absorbers typically used in LWRs are unnecessary, leading to a simplification of the reactivity control strategy. The initial, uncontrolled, excess reactivity in a SFR is 5–10 times lower than the typical value in LWRs.

Reactor and guard vessels

The reactor internals, which at a minimum include the core components, control systems and instrumentation, are contained within the reactor vessel. The reactor vessel is a solid shell, typically made of steel, which role is to contain the primary sodium, the various internals, and constitutes one of the protection barriers against the release of radioactive products. The thickness of the reactor vessel depends on the size of the reactor, volume of sodium, weight of the internals it supports, and radiation shielding needs. Typical thicknesses are less than 5 cm. It can be hung from the top, or directly supported at the bottom. These two approaches lead to slightly different seismic response behavior. The reactor cover head is a thick plate of steel closing the reactor vessel and allowing

Liquid Metal cooled Fast Breeder Reactors (LMFBR)

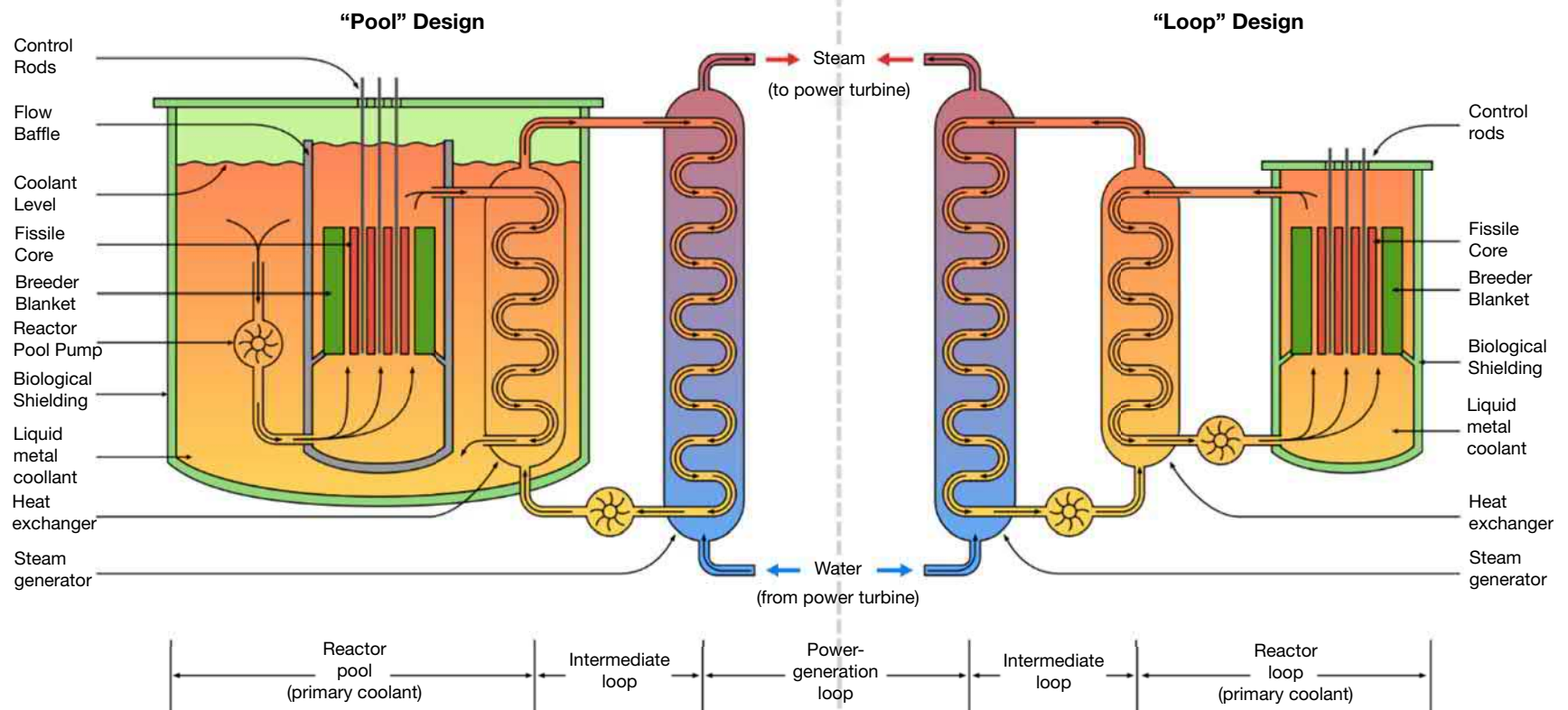


Fig. 12 Schematic comparison of pool-type versus loop-type layouts for SFRs. Source: Wikipedia.

Table 2 Major differences between loop-type and pool-type SFR configurations.

Characteristics	Loop-type	Pool-type
Primary coolant	Exit vessel	Within vessel
Primary vessel	Compact, less in-vessel space	Large
Primary system components	Outside vessel	Within vessel
Safety aspects	Reduced sodium inventory	Minimized coolant pipe break impact, large heat sink
Leak prevention	Guard vessel; double-walled coolant pipes	Guard vessel
Maintenance	Ease of access and to isolate the primary loop	Mostly under sodium

necessary penetrations for the control systems, instrumentation, and such. Specific reactor vessel layout and arrangement depends on the SFR design and its overall configuration.

The reactor vessel is usually surrounded by a guard vessel which plays the role of a failsafe vessel as it envelopes the reactor vessel. The gap between the two vessels does not contain sodium and typically is filled with an inert gas or with air. This gap is large enough to allow inspection through remote means, but small enough to ensure that in case of failure of the reactor vessel, the core will remain submerged under sodium and decay heat removal systems remain operational.

Fuel handling and storage/refueling system

The primary sodium activity builds up during normal operations due to activation of ^{23}Na into ^{24}Na through neutron capture. With ^{24}Na half-life being about 15 h, it would take several days or weeks for the activity levels to reach a low enough value to allow direct access for refueling. In addition, with the sodium refueling temperature typically being in the 200 °C range, refueling operations in SFRs are routinely performed with the cover head in place. There are several ways in which refueling can occur in a SFR, but all have some similarity. For instance, assemblies being removed from the core are handled under sodium or inert gas atmosphere. With sodium being opaque, careful planning of assembly handling is necessary and specific features are built into the designs to prevent miss-loading of assemblies. Additionally, under-sodium viewing techniques have been developed but do not replace careful planning (Griffin et al., 2009).

As part of the process of removing used assemblies from a SFR core, discharged assemblies are moved from the sodium environment to a sodium washing station. There, residual sodium still present in the assemblies is being removed under inert atmosphere before used assemblies can be stored. Before being discharged from the sodium environment, fuel assemblies are kept within the reactor vessel until their decay heat is compatible with the fuel transfer technology and protocol in place for that particular reactor. It can take several months for the decay heat to drop to the desired levels, interim storage or in-vessel storage space is typically available away from the active core region.

Heat transport systems

The primary coolant system removes heat from the nuclear core, and transfers it to the secondary coolant system through the IHX. As sodium is a light liquid metal at typical operating conditions, it can be circulated through the primary system using either mechanical pumps or electromagnetic pumps. The tradeoffs between these two pump technologies do not result in either technology being significantly more attractive than the other. While natural circulation can be relied upon to remove heat from the core, especially in some of the postulated unprotected transient scenarios, it is insufficient to establish flow rates consistent with the high power density achieved under normal operating conditions in most SFRs.

Intermediary heat exchangers come in various shapes and forms, but in all cases they transfer heat from the primary coolant system to the secondary coolant system. Sodium is used as the cooling material in the secondary system in order to avoid any risk of the activated primary sodium to come in contact with air or water. This would lead to burning, pressurization of the primary coolant system and potential dispersion of radioactive isotopes outside of the primary coolant system. Similar to the primary coolant system, the sodium contained in the secondary system is circulated by mechanical or electromagnetic pumps.

Heat is removed from the secondary coolant system through a steam generator, for electricity generation, or directly dumped in the environment, typically into the air, in case of research or test reactors. This part of the plant does not differ in any major way from that encountered with the other reactor technologies. For instance, either conventional steam cycles or supercritical CO_2 Brayton cycles can be used for the energy conversion system. As operating temperatures are higher in SFRs than in LWRs, the thermal efficiency of SFRs is typically around 40% versus about 32% of LWRs.

Decay heat removal systems

After a reactor is shutdown, decay heat is still generated over time and initially amounts to several percent of the nominal reactor power. Proper removal of the decay heat is key in order to ensure fuel integrity and has been the source of most of the adverse consequences of the various nuclear accidents around the world. In SFRs, during normal shutdown the decay heat is simply removed

through the balance-of-plant, dumping the heat in a heat-sink. During hypothetical accident scenarios where the balance-of-plant is unavailable, the decay heat is removed by safety-grade heat removal systems (passive or not) which become enabled as the reactor temperature rises. More details about those systems are provided in the safety section of this article.

Containment

The containment building is the last barrier of defense-in-depth (Grabaskas et al., 2015) against the uncontrolled release of radioactivity. During hypothetical accident scenarios a SFR containment would experience a set of conditions different from that expected from LWRs or other types of reactors. Being a low pressure system, pressurization does not play a major role for an SFR containment building, and the same goes for highly energetic Maximum Hypothetical Accidents (MHA). Experiments, analyses and experience have shown those situations to be excessively unlikely for SFRs. Rather, the design basis for the containment building shall consider large sodium fires as the bounding set of conditions. Protection against large sodium fire is typically achieved through appropriate design of the guard vessel for pool-type SFRs, and through double-walled piping for loop-type SFRs.

Instrumentation and Controls

Various instrumentation is used to monitor a SFR, similarly to other reactor technologies. Characteristics of interest include the neutron flux, various temperatures, coolant flows and pressure. These are needed to inform the reactor control system and the plant protection system, as well as the operators in order to ensure the safe operation of the plant. The conditions experienced in a SFR and the properties of sodium being unique to that reactor technology, this leads to a different selection of the proper monitoring instruments.

For instance, flux monitoring can be achieved with neutron detectors such as fission chambers located outside of the reactor vessel in order to protect them from high temperatures and neutron fluxes, which could reduce their lifetime or reliability. Positioning of these detectors several meters away from the core and still obtaining accurate flux monitoring is made possible by the high flux and large neutron mean free path which are unique features of fast reactors.

Coolant temperature and flow are measured at different points in the primary and secondary coolant loops and are used to determine the core thermal power. Propagation of the coolant conditions through the coolant loop and response time of the instrumentation selected can insert some small delay in obtaining the accurate core condition, which needs to be consistent with the desired response time of the reactor control system.

Safety considerations

Passive safety

One of the main value proposition of the advanced reactor technologies is to offer improved safety performance and behavior (Ruggieri et al., 2017), compared to the previous generations of reactors. While existing reactors have been shown to be extremely safe, advanced reactors and including SFRs, offer the opportunity to achieve passive safety—sometimes referred to as inherent safety (IAEA, 2020; ANL, 2002; Till and Chang, 2011). The main benefit is that the reliance on safety-grade safety systems needing to be activated to ensure the safety and preservation of the plant is significantly reduced. This can help reduce the complexity of the instrumentation and control systems, and therefore of the associated costs and risks. While passive safety can theoretically be achieved with any of the advanced reactor technologies, the design space enabling it is quite limited for some of those technologies. For SFRs, while passive safety is not guaranteed for every design configuration, it is a reactor technology for which it is readily achievable.

Before jumping into the passive safety features of SFRs and how it is achieved, a general sense for the meaning behind “passive safety” is required. A reactor can be considered safe if during normal operation and any possible anticipated operational occurrence it is able to remain in a safe state, preventing radioactivity release, exposure, or failure of critical components (e.g., vessel rupture, core melting, etc.). A safe reactor is not immune to unexpected events happening or to random equipment failure; but it guarantees that in such scenarios the public, workers and capital asset will not be jeopardized. A passively safe reactor does not guarantee more than that, but it is able to achieve those safety objectives through relying uniquely on passive systems or behaviors. It is also important to note that because a reactor is passively safe, it does not mean it does not contain any active safety systems, or does not rely on active systems to maintain control of the reactor. Passive safety can be seen as the guarantee that even when everything goes unexpectedly wrong, the reactor will remain in a safe state. Autonomous reactor operation, while potentially synergistic with passive safety, is a different matter, not discussed here.

Excess reactivity and self-compensation

In a simplified way, two key requirements of reactor safety are the ability to prevent excessive increase of the core reactivity and to prevent excessive temperatures to be reached. Controlling excess reactivity is typically achieved through relying on activation of the control (or safety) rods/blades when the reactor control system detects abnormal conditions. A passively safe reactor is able to mitigate reactivity excursions even without control rods being activated. For SFRs this is achieved through favorable reactivity feedback coefficients, which are combining together to maintain the reactor in a controllable state. Although there are multiple postulated

accident scenarios to consider and analyze when performing a safety assessment, all with different assumed successions of events, we can provide a general description of the general trends. When reactivity of the core unwillingly increases (e.g., accidental removal of a control element, core compaction, etc.) the power generated by the core will increase. In turn, this inevitably leads to an increase of the fuel, cladding and coolant temperatures. The increase in coolant temperature will also lead to an increase of the core support structure temperature and therefore to its thermal expansion. All of these temperature changes lead to a number of feedback mechanisms: Doppler feedback from the fuel, fuel density feedback, cladding density feedback, coolant density feedback, and core axial and radial expansion feedbacks. Not all feedbacks build up at the same moment, and the particular reactor configuration quantitatively affects the actual trends. However, it is well understood that one could not expect significant increase of the coolant temperature without an increase of the fuel and cladding temperature, since the heat is mostly arising from the fuel itself. In SFRs, the combination of these feedback mechanisms commonly leads to a compensation of the accidental reactivity insertion. That reactivity compensation is achieved at temperatures above those observed at nominal conditions. Depending on the particular accident scenario assumed, when relying only on passive mechanisms (i.e., assuming control/safety rods are not responding), passively safe SFRs can either remain critical at higher operating temperatures or shutdown on their own. Those trends, described here in a simplified fashion, have been demonstrated in most past and present SFRs and the ability of SFRs to passively handle all kind of transients is unequivocal. This includes even the most severe beyond design basis accident (BDBA) scenarios postulated for SFRs.

Decay heat management

The second aspect mentioned, pertaining to a safe reactor, is its ability to prevent temperatures to exceed design limits, which could lead to fuel melting, coolant boiling, cladding failure or other highly undesirable and potentially impactful events. While preventing excessive increase of the core reactivity is essential, it does not guarantee that temperatures during transient scenarios would remain acceptable. Peak temperatures reached during postulated transients are highly dependent on the SFR design characteristics. Characteristics that affects those temperatures are the heat transfer characteristics of the system and the power to flow ratio during transients. A key challenge for passive safety with any reactor technology is the management of decay (residual) heat for transient scenarios postulating a loss of forced cooling or of the ultimate heat sink. In those situations decay heat cannot be removed through the intended pathways and if no appropriate design features exist it will lead to ever increasing temperatures. There exist several systems allowing passively removing the residual heat (Ashurko, 1996; ORNL, 1985) during the most stringent transient scenarios, such as Direct Reactor Auxiliary Cooling System (DRACS), Primary Reactor Auxiliary Cooling System (PRACS), Intermediate Reactor Auxiliary Cooling System (IRACS), Reactor Vessel Auxiliary Cooling System (RVACS), or Steam Generator Auxiliary Cooling System (SGACS). All these rely on the ability of SFRs to establish natural circulation through the core, as the sodium temperature differential allows overcoming the primary system hydraulic resistance. For the extremely low flow rates required to remove the residual heat, most of the hydraulic resistance is potential energy. Usually only one of the passive heat removal systems mentioned is implemented in a given design, but all consist in creating an alternate pathway for small amounts of heat to be passively removed from the core. They are sized based on the anticipated decay heat to be handle, and to only become effective when the system temperature rises, as to avoid heat losses during normal operations. Since these systems rely on the primary system to not be drained from all its coolant, SFRs have design features preventing large loss of coolant situations (guard vessel and double-wall pipes).

Many additional safety features exist in the various SFRs, some shared among designs, some design-specific. Overall, the physics of SFRs and the thermal-hydraulics properties of sodium lead to extremely favorable safety behaviors. In particular, SFRs can easily survive BDBA without adverse consequences to the public, workers or even capital asset. While higher-than-normal temperatures can be reached during some BDBA, these retain margins from temperatures that would lead to irreversible plant damage.

SFR missions

With the flexibility exhibited by SFRs (due, to a large extent, to their intrinsic reactor physics characteristics) with respect to the range of viable design options (fuel, materials, layouts, etc.), SFRs are well suited to address a number of potential missions. SFR can be designed as power reactors all the way to radio-isotope producers, including all kinds of material transmutation, and irradiation testing capabilities (Waltar et al., 2012).

Power reactor

When intended as power reactors, SFRs can allow to significantly increase resource utilization as well as significantly reduce the amount of waste produced. Because they tend to require higher enrichment level than thermal reactors, the absolute Separative Work Unit (SWU) per mass of fuel is higher. However, as a significantly higher burnup and a larger fraction of the mined materials can be consumed in a SFRs, the importance metrics is the amount of energy produced per mass of fuel, per SWU, or per mass of mined resource.

In thermal spectrum reactors, accumulation of fission products leads to a significant reactivity penalty which is one of the main limitation to the burnup the fuel can achieve. In fast spectrum reactors accumulation of fission products does not bear such a significant reactivity penalty as the cross-sections of all fission products are orders magnitude lower than in a thermal spectrum reactors. As such, the fuel burnup of SFRs is rarely limited by fission product accumulation, allowing to reach very high burnups. Instead, the

burnup achievable in SFRs tends to be limited by the radiation damage limits imposed on the fuel or cladding materials. Burnup levels of about 20% FIMA (Fission per Initial Metal Atom) have been achieved in SFRs without needing fuel reprocessing or any fission product removal.

While consumption of the fissile material can theoretically limit the fuel burnup, SFRs can be configured as burner, break-even or breeder reactors. For break-even and breeder configurations, depletion of the fissile material is not typically a limitation for the fuel burnup. Instead, the burnup in SFRs is usually limited by either degradation of the fuel performance or of its cladding. With increasing burnup, the fuel composition is changing and its properties as well. Some of the fuel performance characteristics that may become limiting are the fuel swelling, thermal conductivity, mechanical integrity, fission gas release, and constituents relocation (Pahl et al., 1992; Hofman et al., 1997; Jensen and Woolstenhulme, 2021). From the cladding perspective, cladding wastage and degradation of its mechanical properties due to neutron induced radiation damage leads to a limitation of the burnup that the fuel can reach. In those cases where the burnup is not limited by reactivity considerations, the fuel could theoretically be reprocessed or reconditioned and directly reused in the reactor.

The resource utilization in a SFR relative to a LWR can be roughly estimated using approximate values. Assuming LWRs using 4.2% LEU and achieving a fuel burnup of 50 GWd/t, their resource utilization is approximately 0.6%, meaning that not even 1% of the energy contained in the mined uranium is extracted. While assuming that the fuel burnup of SFRs is limited by the fuel or cladding performance under irradiation, burnup levels of 200 GWd/t have been regularly achieved. With SFRs requiring LEU containing around 15% ^{235}U , the amount of energy per unit mass of ^{235}U is only about 13% more than in a LWR. When reprocessing the fuel, SFRs can recycle the fuel indefinitely, using depleted uranium as the makeup fuel (no uranium enrichment is needed beyond the initial fuel loading). This allows SFRs to make use of the large amount of the depleted uranium accumulated during the fuel enrichment process. While in theory this could lead to fuel utilization of nearly 100%, losses occurring during reprocessing and fuel fabrication reduce the achievable fuel utilization to $\sim 90\%$ (Krepel and Losa, 2021). This estimate is highly dependent on the reprocessing and fabrication losses, the ability to recover those losses, and the particular processes used. Furthermore, losses for a given process are typically getting reduce as the process keeps getting refined.

While this thought exercise is focused on LEU for the initial fuel loading, SFRs can be fueled with fissile materials other than ^{235}U , such as excess plutonium from the military program or plutonium extracted from LWR used fuel.

Core configurations

The configuration of a SFR core can be set in different ways depending on the objective. Typically, the possible core configurations can be sorted into three categories: burner, breeder and break-even. Those are not three widely different approaches to core configurations, but rather should be seen as a continuous configuration morphing from burner to breeders, with the conversion ratio being the differentiator.

Burners

In a burner configuration (Hoffman et al., 2008) the core consumes more fissile material than it generates through neutron capture in fertile isotopes such as ^{238}U or ^{232}Th . The conversion ratio (CR) of burners is less than unity. This is the preferred configuration when one of the reactor missions is to transmute materials such as trans-uranium (TRU) actinides from used fuel. Burner SFRs are purposely designed to require large fissile and low fertile fuel loadings such as to minimize new fissile material generation and maximize TRU incineration rate. This is achieved through high parasitic neutron capture and leakage probability. The capture process is typically part of achieving the transmutation goals. Usually, burner configurations, illustrated in Fig. 10, feature a large reactivity swing during an operating cycle and therefore tend to require a larger number of control rods than other SFR configurations. By burning TRU from LWR used fuel the LWR resource utilization is effectively increasing.

Breeders

In a breeder configuration, illustrated in Fig. 9, the core generates more fissile material (usually primarily ^{239}Pu) through neutron capture in fertile isotopes (usually ^{238}U) than it consumes. It has a breeding ratio (BR) larger than unity (Till et al., 1980; Krepel and Losa, 2021) which depends on the fuel considered, as shown in Fig. 13. Alternately, a given fissile material can be turned by a breeder reactor into a different type of fissile material; for instance using LEU to breed ^{233}U in thorium fuel.

Breeder SFRs benefit from making the most of the neutron economy, as every neutron captured in the fertile material contributes to the end goal. Parasitic neutron capture and large neutron leakage probabilities are undesirable. However, when neutron leakage probability is excessively reduced it may adversely impact the favorable passive safety features of SFRs. When the sodium density is reduced (e.g., temperature increase), the spectrum tends to harden and reactivity increases. This is balanced by a fractional increase of the neutron leakage probability that reduces the core reactivity. As such, if the initial leakage probability is very small, the fractional increase in leakage probability during transients may not be sufficient to compensate the reactivity effect of spectrum hardening.

Breeder SFRs take advantage of fuel loading with low fissile fraction and large fertile fraction and are designed with high volumetric fuel fraction, leading to relatively compact cores with hard neutron spectrum. It is common for these to include radial and/or axial blankets at the periphery of the core to make the most of neutrons leaking out of the active core region, maximizing achievable breeding. However, breeding in blankets elements can raise some proliferation concerns that must be addressed carefully. As the

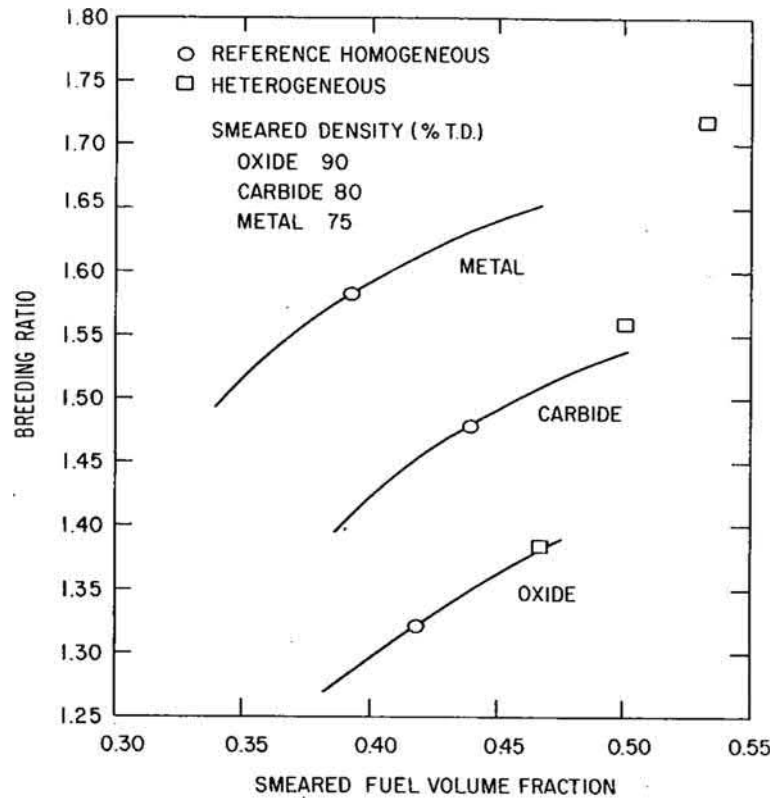


Fig. 13 Breeding ratio for various fuel types as a function of the fuel volume fraction in the SFR core (Till et al., 1980).

burnup reactivity swing in the breeder SFRs tends to be about zero or slightly positive, they usually require fewer control rods to achieve the desired shutdown and control performance.

Breeder SFRs are primarily intended for situations where there is a need to increase the fuel fissile inventory in order to support a growing fleet of nuclear reactors by using the extra fissile material bred for the initial fissile loading of new reactors (without the need for uranium enrichment). The high breeding gain that can be achieved with SFRs is attractive for all nuclear energy growth scenario envisioned by the different governmental agencies around the world, making SFRs a strategic technology in growing nuclear capacities without taking a toll on our natural resources.

Break-even

In a break-even configuration the core generates as much fissile material as is being consumed. It has a breeding ratio equal or slightly larger than unity. Their makeup fuel is a fertile material such as depleted or natural uranium. That is, break-even SFRs are self-sustainable from a fissile standpoint. In practice, the core might generate a little bit more fissile material than it consumes in order to make up for reprocessing/reconditioning and fabrication losses. Break-even SFRs are at the mid-point between burner and breeder SFRs, somewhat mitigating proliferations challenges of having a breeder SFR and the resource utilization limitations of the burner SFRs.

Break-even SFRs would be advantageous in a situation where the system electricity production capacity through SFRs has reached a stable point, with no need for increasing the fissile fuel inventory to start new reactors, and a desire to conserve natural resources.

Transmutation mission

SFRs have regularly been considered as a highly attractive option for High Level Waste (HLW) transmutation (IAEA, 1993). They can use minor actinides from spent fuel, some of which have half-lives of hundreds of thousands of years, and progressively fission those (Krepel and Losa, 2021; Boullis and Chabert, 2021). This results in a tremendously reduced amount of minor actinides, or even a near complete consumption if kept long enough in the reactor (Wigeland et al., 2015, 2021). Beyond transmuting minor actinides, SFRs can also serve to transmute used fuel from other reactors and in particular consume or denature plutonium found in those fuels. Some physics considerations are discussed here to provide a basic understanding of these transmutation capabilities.

From a transmutation perspective the key reactor physics characteristic of interest is the ratio between capture and fission for each actinide. Some representative values for these ratio are shown in Fig. 14 for actinides created by successive neutron capture in the uranium and transuranium isotopes. In essence, when a fission occurs in an actinide nucleus, it is consumed. However, when

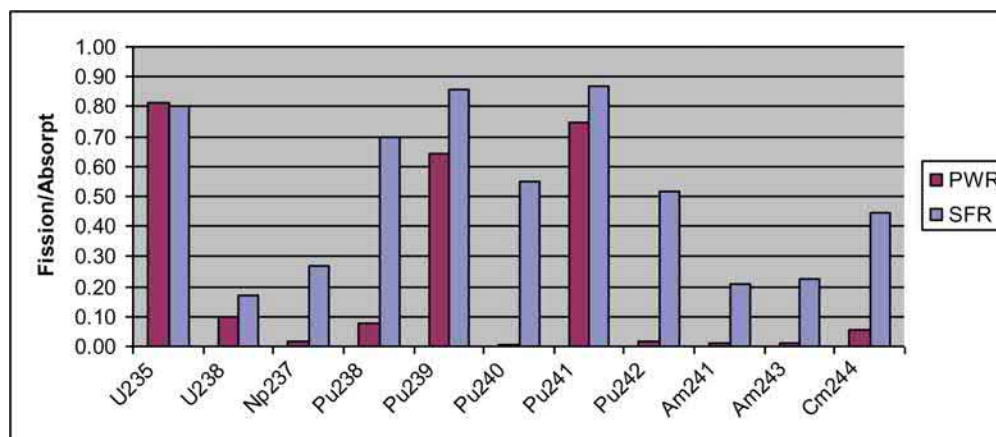


Fig. 14 Representative fission to absorption ratio for selected actinides in SFRs and LWRs (PWR). (absorption = fission + capture).

a capture occurs that actinide nucleus is merely turned into another actinide. While this can be beneficial if the newly formed actinide has favorable properties (e.g., shorter half-life than its parent), this leads to accumulation of heavier actinides (Koch, 1986).

Based on the information of Fig. 14, we can understand what happens to actinides generated from capture in uranium in thermal-spectrum and fast-spectrum systems, and how the heavier actinides accumulate. For instance, when a neutron is absorbed in ^{238}U , it will result in a fission event about 10% of the time in a thermal reactor and about 18% of the time in a fast reactor. This means that, respectively, 90%/82% of the time the ^{238}U atom will not be destroyed, but rather transmuted into another actinide. While for the fissile isotopes ^{235}U , ^{239}Pu and ^{241}Pu as well as for the fertile isotope ^{238}U the difference between thermal and fast reactors is not game-changing, it is for other isotopes. Using ^{242}Pu for illustration, when a neutron gets absorbed in ^{242}Pu in a thermal reactor, it will get transmuted into another actinide about 98% of the time, not resulting into any noticeable reduction of the actinide inventory. In contrast, when a neutron gets absorbed in ^{242}Pu in a fast reactor, it will get transmuted into another actinide only about 48% of the time. The remaining 52% of the time ^{242}Pu will fission and contribute to reducing the actinide inventory.

The successive actinides build up is hence schematically shown in Fig. 15, starting from ^{238}U . In most reactors ^{238}U would not be present in isolation and other actinides would be initially present. The way to read this figure is “for one neutron absorbed in ^{238}U , we would get 0.90/0.82 atoms of ^{239}Pu in a thermal/fast spectrum system,” and so on. This does not account for the different neutron economy of thermal and fast reactors or the different initial actinide concentrations. Overall, LWRs are expected to accumulate over 20 times more minor actinides than SFRs. This can also be stated as SFRs being able to consume minor actinides at over 20 times the rates of what could be achieved in LWRs. This is caused by the fact that in thermal spectrum no isotope beside ^{233}U , ^{235}U , ^{239}Pu and ^{241}Pu has any significant fission cross-section.

Test mission

Neutron irradiation capabilities are a cornerstone to advancing nuclear R&D in particular with respect to new material development, fuel qualification, and life extension of the current fleets (NEAC, 2017). Most irradiation testing needs for existing reactors are met through a limited number of thermal-spectrum reactors. Given that those test reactors are operating with neutron flux of the same order of magnitude as that of current reactors, studying new materials and fuel behaviors at high irradiation levels takes a very long time. This is of particular importance for development of new reactor technologies, in particular fast spectrum systems, as this would require decades of irradiation in existing thermal-spectrum test reactors to achieve the desired irradiation level.

SFRs are uniquely suited to provide accelerated material and fuel testing capabilities as they are the most mature fast-spectrum reactor technology and can offer extremely high flux levels, large irradiation volumes, and significant experimental flexibility (Grandy et al., 2016). SFRs can easily deliver 30-60 dpa/year over large irradiation volumes, effectively speeding up material and fuel testing by a factor 10 or more as compared with LWRs.

While SFRs can deliver on several missions concurrently, the test mission stand somewhat apart from the others. Material and fuel testing can be hindered by power production operational constraints, and the reactor design itself is unlikely to be optimized for power production and testing capabilities at the same time. For instance, testing may require frequent outages, and a core configuration and operating temperatures different than those optimal for power production. Irradiation testing provision embedded in the design may further diverge from the power production mission.

Two SFRs being currently developed to tackle the testing mission are the MBIR developed by AEM Technology in Russia (Dragunov et al., 2012), and the Versatile Test Reactor (VTR) developed by the DOE-NE in the United States of America (Heidet, 2019). On paper, both reactors plan to offer similar flux levels, with peak fast fluxes in excess of $4 \times 10^{15} \text{ n/cm}^2\text{-s}$, but the MBIR design is more compact than the VTR one due to its reliance on fuel containing a very large fraction of fissile materials while the VTR will rely on low fissile contents and also offer larger available testing volume.

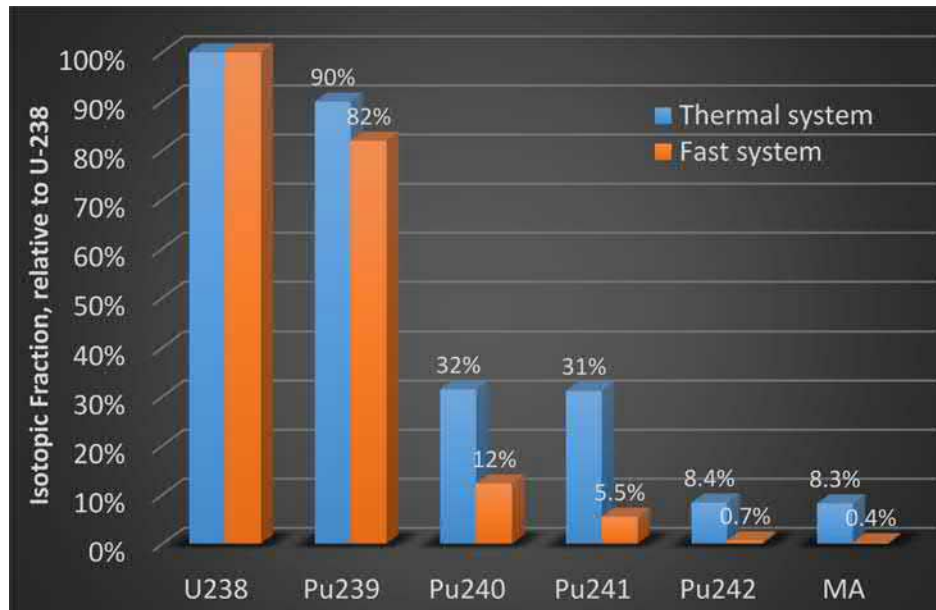


Fig. 15 Representative plutonium and minor actinides accumulation in thermal and fast spectrum systems.

Radio-isotope mission

SFR can also be used to produce radio-isotopes. This is effectively leveraging the same mechanisms as at play with transmutation, making use of excess neutrons to convert target materials into the isotope of interest. Looking at the world of experimental, research and test reactors, production of radio-isotopes has allowed several of these reactors to sustain operations through operational cost recovery. Sales of radio-isotopes has been a major contributor to offset the operational costs without significantly impacting the primary mission of those facilities (NEA, 2012).

SFRs having different flux and spectrum conditions than thermal-spectrum reactors, they are able to produce different radio-isotopes. SFRs typically have fluxes order of magnitude higher than thermal-spectrum systems, have the ability to tailor the neutron energy to a particular production need, have their core performance mostly unaffected by irradiation targets, and can have large volumes available for target irradiation. These characteristics make them uniquely suited for radio-isotope production (Schenter and Myjak, 1987). However, with only few SFRs around the world, this potential is mostly untapped at this time. Furthermore, unique radio-isotopes that can be produced in a SFR (e.g., alpha-emitters) do not yet have extensive commercial applications. This is for the most part due to the inability to produce them in reactors currently used for radio-isotope generation.

While it has not been determined whether or not a SFR dedicated solely to radio-isotope production could be commercially viable, many fast reactor concepts currently under development are considering incorporating capabilities to support radio-isotope production.

Future SFR outlooks

With the growing interest around the world in building advanced reactor systems, SFRs are at the forefront of it. The versatility of the missions that can be efficiently tackled with this reactor technology, its high maturity, and the safety benefits have led several countries around the world to plan design and constructions of SFRs. For some, as part of bridging experimental gaps, for others as a mean to keep up with the growth of their nuclear capacity and associated fuel resources utilization, and yet for others as a mean to advance their reactor technology to the next stage.

While SFRs are not yet commercially deployed world-wide, several commercial companies have been working on various types of SFRs, offering reactors ranging from large-scale units for electricity production all the way down to micro-reactor size for remote applications. Among the various designs of SFR there exists a specific category, called Breed and Burn reactors (B&B), which offer an innovative way of managing fuel loading in order to completely circumvent the need for fuel enrichment, after the very first fuel loading as well as avoiding used fuel reprocessing (Hejzlar, 2020; Qvist, 2021). These B&B reactors have gained significant traction over the last decade as they would enable achieve maximal resource utilization without the need for fuel separation, which still remains a highly political matter.

Although SFRs have demonstrated significant benefits over the current generation of nuclear reactor in use, there exist several challenges slowing down their widespread deployment. One of those major challenges is the unknown and uncertainties surrounding the capital and operating cost of SFRs (Mooz and Siegel, 1979; Kim et al., 2013). While most cost comparison studies

conclude that the capital cost of SFRs would be more than that of LWRs (Till and Chang, 2011), valuation of the benefit offered by SFRs is impossible without corresponding policy changes. For instance, this includes how we value resource utilization, amount of waste generated, sustainability, safety behavior, and other such consideration that are significantly enhanced by SFRs.

Another challenge faced by SFR is the political and public acceptance. Some of the technologies and capabilities enabled by SFRs, such as fuel reprocessing and breeding, are suffering from a stigma in the view of non-experts. In the past, this led to the early cancellation of the IFR project (Till and Chang, 2011) and early closure of the Superphenix reactor. With communication efforts over the last several decades, those stigma are slowly shifting away, but some still remain as they require more in-depth understanding to dispel. For instance, some decision makers are challenging the safety behavior of a reactor by requiring that each of its reactivity coefficient will be negative, while in practice one needs to look at the compounded effect, as discussed above in this article. Such miss-interpretation leads to forcing design choices onto the experts, potentially resulting in a more complex and expensive reactor.

At a higher level, SFRs are facing global challenges common to all advanced reactor technologies. There is a lack of market opportunities in many countries which already have a fleet of nuclear reactors and the energy demand of most of these countries is flat and no new power plant construction is warranted. Additionally, most countries do not have a fully established licensing framework for advanced reactors, which complicates moving advanced reactor projects forward and results in an increased risk for investors. Those challenges are being tackled and progresses made (GIF, 2018; U.S. NRC, 2018).

From the technical perspective, a pending item needing attention for commercial deployment of SFRs is fuel qualification. Even when uranium oxide is used, the condition experienced in SFRs are much different than in LWRs and therefore the fuel needs to be carefully qualified for these conditions. As it can include high burnups, the shortage of fast spectrum irradiation facilities across the world creates a bottleneck to fast reactor new fuel development and qualification. In addition to the relatively large fuel irradiation databases (Yacout et al., 2017) accumulated through operation of SFRs, efforts to develop very high flux irradiation facilities are underway in Russia through the MBIR project (Dragunov et al., 2012), and in the United States through the VTR project (Heidet, 2019).

Lastly, it is important to note that a number of studies focusing on non-reactor aspects of nuclear energy have concluded that fast reactors and in particular SFRs are the most essential reactor technologies with respect to various strategic objectives. For instance, an exhaustive fuel cycle study determined in 2014 that the most promising fuel cycles of the future will be based on fast reactor to enable resources sustainability and limited environmental impact (Wigeland et al., 2014, 2021). In 2017 the Advanced Demonstration and Test Reactor (ADTR) options study, carried in collaboratively by many institutions in the United States determined that SFRs were the most viable technology to offer accelerated irradiation testing capabilities (Petti et al., 2017; Grandy et al., 2016). Numerous additional studies across the world similarly point to SFRs being a keystone to advancing nuclear energy.

See Also: Advanced Fuel Cycles Identification and Evaluation; Historical Survey of Fast Test Reactors; Historical Survey of Test Reactor Programs at INL Over 70 Years; Introduction to Breed and Burn Reactors; Irradiation Performance: Fast Reactor Fuels; Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors; Traveling Wave Reactor (TWR®).

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The High Temperature Gas-Cooled Reactor

Michael A. Fütterer^a, Gerhard Strydom^b, Hiroyuki Sato^c, Fu Li^d, Eric Abonneau^e, Tim Abram^f, Mike W. Davies^g, Minwhan Kim^h, Lyndon Edwardsⁱ, Ondrej Muransky^j, Manuel A. Pouchon^j, and Metin Yetisir^k, ^aEuropean Commission, Joint Research Centre, Petten, The Netherlands; ^bIdaho National Laboratory, Idaho Falls, ID, United States; ^cJapan Atomic Energy Agency, Ibaraki, Japan; ^dINET, Tsinghua University, Beijing, China; ^eCommissariat à l'Energie Atomique et aux Energies Alternatives, Paris, France; ^fManchester University, Manchester, United Kingdom; ^gJacobs Clean Energy Limited, Knutsford, Cheshire, United Kingdom; ^hKorea Atomic Energy Research Institute, Daejeon, South Korea; ⁱAustralian Nuclear Science and Technology Organisation, Lucas Heights, Australia; ^jPaul Scherrer Institut, Villigen, Switzerland; and ^kCanadian Nuclear Laboratories, Chalk River, ON, Canada

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Glossary

AGR Advanced Gas-cooled Reactor
 AVR Arbeitsgemeinschaft Versuchsreaktor
 BISO Bi-Structural Isotropic Fuel
 GCR Gas-Cooled Reactor
 GIF Generation IV International Forum
 GT-MHR Gas Turbine Modular Helium-cooled Reactor
 HTGR High Temperature Gas-Cooled Reactor
 HTR High Temperature Gas-Cooled Reactor
 HTR-10 10 MW High Temperature Reactor
 HTR-PM High Temperature Reactor – Pebble-bed Module
 HTTR High Temperature engineering Test Reactor
 HWR Heavy Water Reactor
 INET Institute of Nuclear and New Energy Technology (Tsinghua University, Beijing)
 JAEA Japan Atomic Energy Agency
 LANL Los Alamos National Laboratory
 LWR Light Water Reactor
 MWe Megawatt electric
 MWth Megawatt thermal
 NGNP Next Generation Nuclear Plant
 ORNL Oak Ridge National Laboratory
 PBMR Pebble Bed Modular Reactor
 R&D Research and Development
 S-I Sulfur-Iodine thermochemical process for hydrogen production
 SFBR Sodium-cooled Fast Breeder Reactor
 THTR Thorium High Temperature Reactor
 TRISO Tri-Structural Isotropic Fuel
 VHTR Very High Temperature Reactor

What is a high temperature gas-cooled reactor?

High Temperature Gas-cooled Reactors (HTR or HTGR) are helium-cooled graphite-moderated nuclear fission reactors utilizing fully ceramic fuel. They are characterized by inherent safety features, excellent fission product retention in the fuel, and high temperature operation suitable for the delivery of industrial process heat, in particular hydrogen production. Typical coolant outlet temperatures range between 750 °C and 850 °C, thus enabling power conversion efficiencies up to 48%.

The Very High Temperature Reactor (VHTR) is a longer term evolution of the HTR targeting even higher efficiency and more versatile use by further increasing the helium outlet temperature to 950 °C or even higher (Gougar, 2011).

From groundbreaking technology...

The HTR has evolved from the early gas-cooled reactors (GCRs) that gained widespread popularity for their simplicity and high power conversion efficiencies (Beech and May, 1999). The first commercial nuclear power plant was a CO₂-cooled graphite-moderated Magnox reactor (Calder Hall in 1956). In total, 26 Magnox reactors were built (270–1760 MWth), with the last one (Wylfa-1, 1971–2015) shut down at the end of 2015. As a second generation, 14 Advanced Gas-Cooled Reactors (AGRs) were deployed in seven nuclear power plants at six sites in the UK with a total capacity of approx. 8 GWe. All these AGRs are expected to remain in operation until 2023–30, although their life extension required clearance of graphite cracking issues and two power plants have to run at lower power because of the observation of cracks in boilers. This multi-decade effort in the development and operation of gas-cooled reactors allowed for collection of a considerable technical background and operational experience, which then served as the basis for the development of current HTRs. GCRs have an extremely clean primary cooling circuit (few radiological and chemical contaminants) and use a conventional steam cycle (~ 540 °C, same as for coal fired power plants) resulting in high thermal efficiencies (> 40%). However, GCRs had to observe a temperature limitation due to the dissociation of CO₂ and the resulting carburization of structural materials and oxidation of graphite at elevated temperatures. Modern HTR are characterized by increased operating temperature and thermal efficiency, which could be achieved by two major changes: the designs adopted helium as the cooling gas along with fully ceramic fuel, which is discussed in more detail in Section “**TRISO fuel: Key to performance and safety.**”

The first HTR was proposed in a 1945 design study in the US, but was never realized. It featured a primary circuit (helium at 1.55 MPa, 438–732 °C) coupled to a secondary Brayton power conversion cycle (air at 2.9 MPa, 677–22 °C) leading to an expected power rating of 5 MWe.

In 1962–63, a 3.3 MWth Mobile Low-Power Reactor (ML-1) with 140 (330 nominal) kWe was built in the US with a closed-cycle nitrogen turbine. The project was not pursued because it could not fulfill the power output expectations.

In 1964, the Experimental Gas-Cooled Reactor (EGCR) was built at ORNL in the US, but not completed. This was basically a helium-cooled AGR-type reactor using stainless steel fuel rod clusters. EGCR was expected to produce 85 MWth/25 MWe with helium at 566 °C.

Ensuing developments led to conceptual changes in the existing gas-cooled reactors involving, as mentioned, in particular the use of helium instead of CO₂, and the substitution of metallic fuel clads by fully ceramic fuel, both in view of a further increase of reactor outlet temperature and improved safety performance.

The first tangible step in this direction was made in the UK with the DRAGON reactor (see also Section “**Which HTR versions were developed?**”). With a power of 21.5 MWth, it was an OECD project and operated from 1964 to 1975 primarily as a test bed for HTR fuel development. It used already early versions of fully ceramic coated particles as its own fuel.

In the US, the Ultra-High-Temperature Reactor Experiment (UHTREX) operated at LANL from 1966 to 1970. Its rated power was 3 MWth using helium at 3.4 MPa (870–1300 °C). It used extruded fuel with TRISO coated particles in an annular rotatable core for on-line refueling.

More details on the development of HTR technology can be found in a recent authoritative summary (Kugeler and Zhang, 2019).

... to modern characteristics

The following developments led to basic technical characteristics and design features shared by all modern HTR.

- can be built with passive safety features up to 625 MWth/core (prismatic block type core) and 250 MWth/core (pebble bed core); this is the power range of Small Modular Reactors (SMR);
- long grace time after an accident (large heat capacity, low power density);
- self-stabilization of power transients (negative temperature coefficient);
- low source terms (outstanding fission product retention in robust TRISO coated fuel particles and structures);
- fully ceramic core (fuel and moderator/reflector);
- high-purity graphite as moderator/reflector, high thermal inertia;
- chemically and neutronically inert helium as primary coolant;
- high operating temperatures for high efficiency, capability for nuclear cogeneration of heat and power, including for bulk hydrogen production;

- high burn-up capability;
- high fuel utilization (good neutron economy and possible use of thorium).

There are two competing HTR designs based on the TRISO coated fuel particle (**Fig. 1**): the prismatic block core and the pebble bed core (**Fig. 2**).

- Invented by Peter Fortescue and his team at General Dynamics in the US, the prismatic block core is built from hexagonal graphite blocks containing vertical holes. Some of these holes are used for helium cooling, while others receive the fuel in the form of “compacts”, which are little cylinders (typically $\varnothing 12.3 \times 25$ mm) pressed from graphite and coated fuel particles (**Fortescue, 1975**).
- The pebble bed HTR was conceived in 1942 by Farrington Daniels in the US (**Daniels, 1944**). This early vision was later developed to a power plant design by Rudolf Schulten in Germany, which employed $\varnothing 60$ mm fuel spheres made of graphite and coated fuel particles (**Schulten et al., 1959**). These pebbles are filled into the reactor pressure vessel, which is internally lined with graphite blocks. The resulting pebble bed constitutes the reactor core. The pebble bed can flow and allows discharge and (re-)injection of pebbles during operation, enabling online refueling.

TRISO fuel: Key to performance and safety

One of the major challenges and key to achieving a fully ceramic reactor core was fuel development (**IAEA, 2010**). The initially used UO_2 or UC fuel was placed in ceramic clads which showed poor fission product retention. Coated particle fuel was invented between 1957 and 1961 by the United Kingdom Atomic Energy Authority (UKAEA) and Battelle, but no patent was granted at that time. The UO_2 fuel kernels were made by external gelation of uranyl nitrate in ammonia and, after a heat treatment, coatings were deposited on top of these kernels via pyrolysis of hydrocarbons in a fluidized bed. The next development step was the early BISO (bi-structural isotropic) particle fuel comprising a buffer layer directly deposited on the kernels and an additional pyrolytic carbon (PyC) layer on top. Finally, modern TRISO (tri-structural isotropic) particles were given an additional SiC diffusion barrier leading to confirmed fission product retention up to 1600°C or even higher (**Gougar et al., 2020**). These TRISO coated particles, typically in the order of 1 mm in diameter, are the basis for all modern HTR fuel designs (**Gerczak, 2021; Helmreich, 2021**). As shown in **Fig. 1**, they feature (from inside out) the kernel, a porous PyC buffer to accommodate fuel swelling and fission gases, a dense PyC buffer and a dense SiC layer as diffusion barriers against fission product escape, and a final PyC layer (missing in **Fig. 1**) for better bonding with the matrix graphite into which they will be integrated.

Baked into matrix graphite, the TRISO coated particles can now be given a macroscopic shape (**Fig. 2**), usually in the form of thumb-thick cylinders (“compacts”) either solid or annular, or in the form of spherical fuel elements (“pebbles”). The compacts are inserted into hexagonal blocks made of graphite, which are then assembled to constitute the reactor core contained in a pressure vessel.

Typical pebble and compact design characteristics are given in **Table 1**:

Which HTR versions were developed?

Based on these characteristics, in the 1960s two different types of reactors were designed and built, primarily to produce electricity. Experimental HTRs with a prismatic block core and TRISO coated particle fuel were developed in the UK (DRAGON reactor,

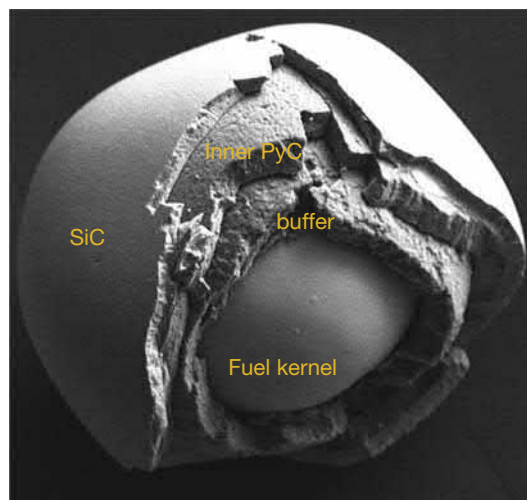


Fig. 1 SEM picture of a modern TRISO coated particle broken up to visualize the coatings; the top outer PyC layer is still missing on this particle.

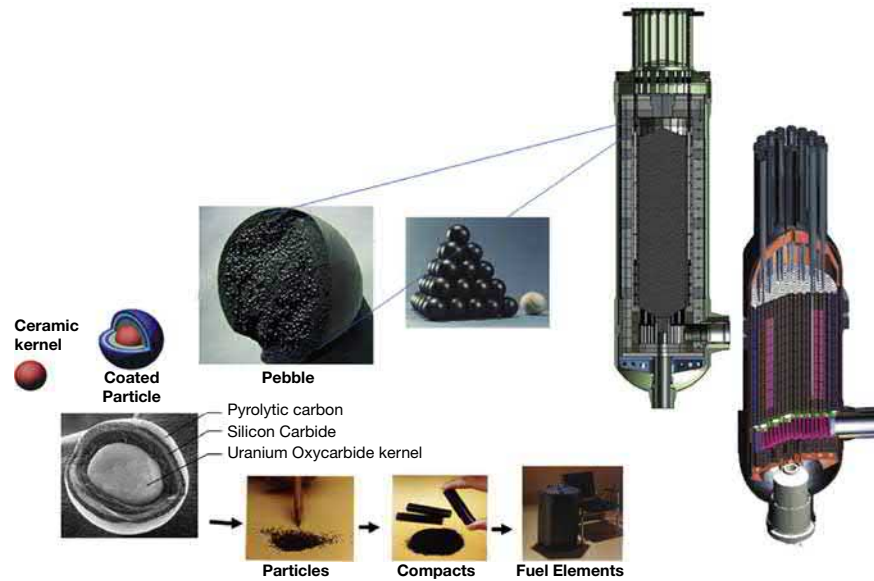


Fig. 2 TRISO coated particle fuel as the basis for hexagonal block and pebble bed core designs (Gougar et al., 2020).

Table 1 Typical examples for nominal characteristic data of German AVR GLE-4 particles and pebbles and US NGNP particles and compacts.

Coated particle	AVR pebble	NGNP compact
Kernel composition	UO ₂	UCO
Kernel diameter [μm]	502	425
Enrichment [U-235 wt%]	16.76	14
Thickness of coatings [μm]:	92	100
buffer	40	40
inner PyC	35	35
SiC	40	40
outer PyC		
Particle diameter [μm]	916	855
Fuel element (FE)	Pebble	Compact
Dimensions [mm]	Ø60 (spherical)	Ø12.3 × 25 (cylindrical)
Heavy metal loading [g/FE]	6.0	1.27
U-235 content [g/FE]	1.00	0.18
Number of coated particles per FE	9560	3175
Volume packing fraction [%]	6.2	35
Fraction of factory defective SiC coatings	7.8×10^{-6}	$< 1.2 \times 10^{-5}$
Matrix density [kg/m ³]	1750	1600
Temperature at final heat treatment [°C]	1900	1850

operated 1964–1975, 21.5 MWth, an OECD project (Price, 2012) and in the US (Peach Bottom, operated 1966–1974, 115 MWth/40 MWe Beck and Pincock, 2011). They were followed by the prototype of the Fort St. Vrain Generating Station (operated 1976–1989, 842 MWth/330 MWe, Beck and Pincock, 2011). This reactor established the technical feasibility of HTRs although it experienced problems (Rempe, 2021) of power fluctuations, jamming of a control rod and leakage of moisture into the core, which finally caused its decommissioning for economic reasons.

Over the same period, Germany developed and built an experimental pebble bed reactor (AVR, 46 MWth/15 MWe, Pohl, 2008) at the Jülich Research Centre that successfully operated from 1967 to 1988 and produced valuable feedback on different types of pebble fuels and overall reactor operation. In particular, it was used for several demonstrations of passive safety performance. After a water ingress accident provoked by a steam generator leak it could be repaired, dried and returned to service. Following this experience, a 300 MWe prototype power reactor that aimed at using thorium fuel was built and operated: the Thorium High Temperature Reactor (THTR-300, 750 MWth/300 MWe, Baumer and Kalinowski, 1991; Dietrich et al., 2019). This prototype, however, met a number of technical difficulties. Examples of design issues are the direct insertion of the control rods in the pebble bed (causing

pebble damage) and the pebble discharge system, which allowed for jamming. The THTR was closed in 1989 in the aftermath of the Chernobyl accident after only 3 years of operation.

In the same period, the Power Nuclear Project (PNP-500, 500 MWth, Neef and Weisbrodt, 1979) started in Germany aiming at using nuclear heat to produce hydrogen by steam methane reforming. This project led to development and testing of large modules of heat exchangers and a steam reformer. It was brought to a halt in 1989 after the Chernobyl accident, which caused a temporary stop of HTR development worldwide.

In the 1980s, Interatom/Siemens in Germany developed the 200 MWth HTR-Modul as the first modular pebble bed design consisting of a metallic reactor pressure vessel connected to an adjacent steam generator through a hot gas duct (Siemens, 1988). The concept features a simplified design with a size and power rating chosen to enable passive decay-heat removal after a loss-of-coolant-accident solely by conduction and radiation. No natural or forced convection is necessary (Reutler and Lohnert, 1984; Kugeler et al., 2017). Although it was never built, the HTR-Modul has served as the basis for the PBMR in South Africa and for the HTR-10 and HTR-PM reactors in China.

The Gas Turbine Modular Helium Reactor (GT-MHR, LaBar, 2002) is a 600 MWth design developed by a group of Russian and US enterprises, Framatome in France and Fuji Electric in Japan. It was based on the earlier MHTGR-350 design by General Atomics. It employs an annular **prismatic core** and utilizes a direct helium **Brayton cycle** for electricity generation with an efficiency of up to 48% based on a reactor outlet temperature of 850 °C. Extensive analysis has shown that this reactor, and more generally most HTR designs, are particularly suitable for the incineration of excess plutonium which became an issue in the US and in the former USSR for the implementation of the START I disarmament treaty in 1991. Hydrogen production with the Sulfur-Iodine (S-I) process was also envisaged. The Preliminary Design of the reactor plant and GT-MHR prototype power plant was completed in 2001. The GT-MHR regulatory process started in 2002 but was not completed. More recently, the GT-MHR design was proposed by General Atomics as one of the options for the US NGNP project until the NGNP Alliance expressed in 2012 a preference for the ANTARES concept (625 MWth) developed by AREVA (Lommers et al., 2012), based on the GT-MHR but with an indirect steam cycle. A smaller version (SC-HTGR, 350 MWth) equally with indirect steam cycle was proposed by AREVA/Framatome as well (AREVA, 2014). The GT-MHR was also the basis for the Japanese GT-HTR300 designed by JAEA (Kunitomi et al., 2004).

A review summary on the 7 built reactors (Dragon, Peach Bottom, Fort St. Vrain, AVR, THTR, HTTR and HTR-10) can be found in (Beck and Pincock, 2011). The experience of past experimental and prototype HTRs demonstrated their technical viability, however, they were not given the time to prove their economic competitiveness with LWR for electricity production. No further developments were to occur until the late 1990s when the interest in HTRs revived owing to the needs of low carbon high temperature heat supply for a variety of industrial processes.

One of these new projects was the Pebble Bed Modular Reactor (PBMR, Matzner, 2004) in the Republic of South Africa. PBMR Pty. Ltd. is a public-private partnership established in 1999 in response to threats of nation-wide power outages in South Africa and to initiate the development of a modular pebble-bed reactor with a rated capacity of 165 MWe. This design featured a thermal power of 400 MWth and a direct power conversion with a gas turbine operating with a helium outlet temperature of 900 °C. In June 2003 the South African government approved a prototype of 110 MWe for the utility Eskom on the site of Koeberg. This prototype was intended to be put in service in 2014 and expected to precede a fleet of 24 PBMRs so as to make up 4000 MWe out of the 12,000 MWe additional nuclear capacity planned by 2030. Large facilities dedicated to PBMR specific technologies testing were built in 2007: a "Heat Transfer Test Facility", a "Helium Test Facility", a "Pebble Bed Micro Model" and an "Electro-magnetic blower." A fuel laboratory developed manufacturing processes and quality assurance testing techniques in collaboration with NECSA and successfully manufactured coated fuel particles with enriched uranium in December 2008.

In 2009 the PBMR project, like other projects of nuclear equipment in South Africa, faced funding difficulties and had its business plan re-oriented towards the supply of industrial process heat, a difficult endeavor in a country with large coal reserves and no CO₂ emission limits. The new focus of the PBMR was on onsite power, cogeneration, seawater desalination and direct process heat delivery. Target process heat applications included coal-to-liquid or gaseous fuels, petrochemicals, ammonia/fertilizer, refineries, steam for oil sand recovery, bulk hydrogen for future transportation and water desalination. Thus, PBMR Ltd. started developing options for commercial fleets with Sasol (the South African coal liquefaction company), with the utility Eskom for electricity, as well as with US and Canadian cogeneration end users including oil sand producers. The PBMR project was accordingly revisited to develop one standard design that meets all requirements for these applications, thus leading to a cogeneration steam plant with a power of 200 MWth, a helium temperature of 750 °C at the core outlet and a steam generator directly placed in the primary loop. A conventional subcritical steam turbine was selected for first generation plants whereas super-critical cycles were envisaged for next generation plants.

Due to funding issues and problems in the interaction between PBMR and the South African regulator the project was stopped in 2010. This development was analyzed critically in (Thomas, 2011). Another investigation with negative conclusions from operational performance of HTR in the past with a pessimistic outlook is summarized in (Ramana, 2016).

Since then, the aforementioned technological problems encountered with test reactors (e.g. moisture leakages into the core) have been solved to a large extent, so that most recent HTR designs could be deliberately geared towards short-term realization with minimum R&D efforts and development risks. In addition, with a much longer-term view, a number of research organizations cooperate internationally on the Very High Temperature Reactor, which is usually understood to produce heat above 950 °C to maximize power conversion efficiency and to enable ambitious process heat applications such as thermochemical hydrogen production with the S-I cycle. The VHTR is thus a long-term concept requiring new materials and design codes along with fuel qualification for the higher temperatures. The very significant progress of this cooperation is summarized in (Fütterer et al., 2014).

Recent results from test reactors

In the 1990s, the Japan Atomic Energy Agency (JAEA) built a research reactor in Oarai, the High Temperature Test Reactor (HTTR, (Kunitomi, 2013), Fig. 3). It is a prismatic block type reactor with annular compacts. It was put in service in 1998 and reached its full design power of 30 MWth in 2001 with a helium outlet temperature of 850 °C. Subsequent tests until 2010 have demonstrated the safe behavior of the reactor. This included reactivity insertion as well as partial and complete loss of forced cooling, but not yet at full power. The HTTR was successfully operated at the design temperature of 950 °C first in 2004, then for 50 continuous days in 2010. In parallel with tests on the HTTR, JAEA is developing the S-I thermo-chemical process to produce hydrogen (Fig. 4). A first demonstration of this process was achieved in 2003 when a continuous production of 30 l/h of hydrogen was maintained for several days. During the March 2011 earthquake, which triggered the Fukushima accident, the HTTR was only slightly damaged. After extensive inspection, some repair and after the review by the regulator, a restart is planned for 2021, pending a positive outcome of the public hearing. JAEA intends to conduct further safety tests in the frame of an OECD-NEA Loss of Forced Cooling Project.



Fig. 3 External view of the HTTR building in Japan.

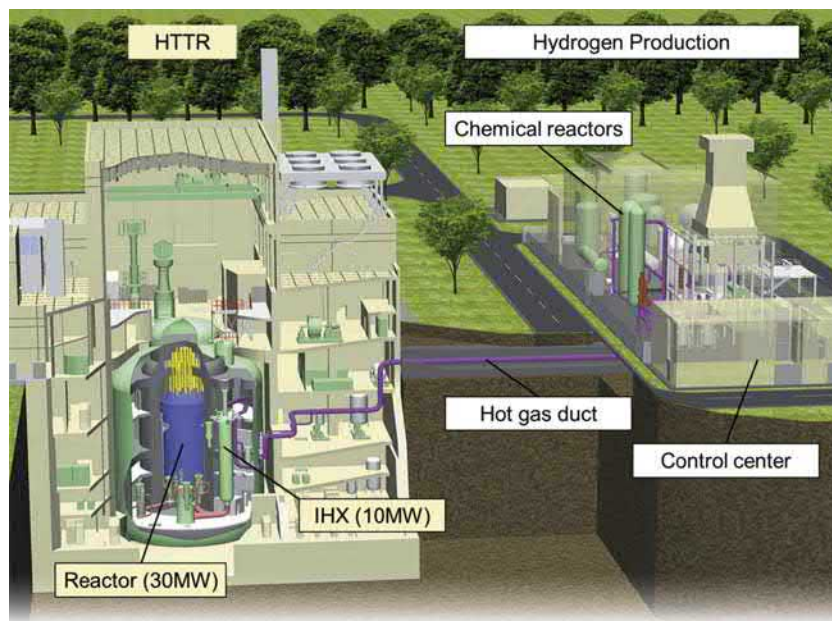


Fig. 4 Schematic of HTTR and future heat use facilities.



Fig. 5 External view of HTR-10 building in China and Control Room.

The Institute of Nuclear and New Energy Technology (INET) of the Tsinghua University in China has built the experimental reactor HTR-10 (10 MWth) (Dong, 2012; Wu et al., 2002; Fig. 5) that was put into service in 2000. The successful operation of this reactor demonstrated an updated pebble bed core HTR technology. In particular, it served as a test bed for fuel, components and for code validation. The HTR-10 was also employed for district heating of the INET campus in the vicinity of the reactor. With several successful demonstrations of its benign safety performance for the public and the licensing authority it paved the way for scaling up this technology to the High Temperature Reactor – Pebble bed Module (HTR-PM, 210 MWe) project (Zhang et al., 2016).

Together with their predecessors, HTTR and HTR-10 have significantly contributed to the establishment of the rather high technology readiness level both for block type and pebble bed HTR designs.

The case for new next generation HTRs

Why have past HTRs not been successful economically and why do we think that this is changing?

GCRs were developed worldwide, but only the AGRs in the UK remain in commercial operation. After reasonable experiences with the first HTR plants in the UK (Dragon), the US (Peach Bottom Unit 1) and Germany (AVR), national HTR programs ended with no commercial deployments for various reasons. In the UK, the Thatcher government decided to build PWRs essentially because of absence of confirmed economic data for other designs, higher perceived financial risk of HTR designs compared to the mainstream PWR, and because of the then unsolved difficulty to integrate the HTR into a long-term sustainable closed fuel cycle that included Fast Breeder Reactors and reprocessing. In Germany, AVR was shut down in 1988 due to public opposition to nuclear energy, shortly after the Chernobyl accident. In the US, poor capacity factors of the Fort St. Vrain demonstration plant led to its premature shutdown in 1989. This has coincided with the time period of three decades without new nuclear orders in the US starting with the Three Mile Island accident in 1979.

In general, most HTR operational issues were associated, as already mentioned, with leakages, e.g. moisture ingress that resulted in corrosion of components, core temperature oscillations caused by coolant flow bypass and in-core behavior of graphite (cracking, dimensional changes, movement of blocks and distortions, dust formation) (Beck and Pincock, 2011). Most of these were first-of-a-kind operational issues and took a long time to resolve without the benefit of the broader industry experience that is dominated by water-cooled reactors. As a result, it led to poor performance in some HTR reactors, most notably the Fort St. Vrain reactor in the US.

On the positive side, however, the operational experiences with HTRs showed excellent fuel performance and demonstrated the concept's inherent safety features. Many lessons learned through past HTR experiences led to improvements in modern HTR concepts, such as the use of magnetic bearings in the helium circulator, or the use of a steel pressure vessel for improved reliability instead of a pre-stressed concrete vessel. Passive cooling systems, requiring no pumps or monitoring systems to initiate them, have been adopted. The excellent performance of TRISO fuel is further improved by recent extensive research programs (Electric Power Research Institute, 2019), which benefitted both fuel types, compact and pebble. These developments eliminate major known issues experienced by early HTRs and further corroborate HTR safety characteristics.

Ongoing HTR development

The last decade has seen significantly growing interest worldwide in Small Modular Reactors, which the IAEA defines as units producing less than 300 MWe. A snapshot of the very dynamic SMR landscape is given in (IAEA, 2018). These reactors are being designed by several classical vendor companies and start-ups for flexibility, affordability, for a wide range of users and applications,

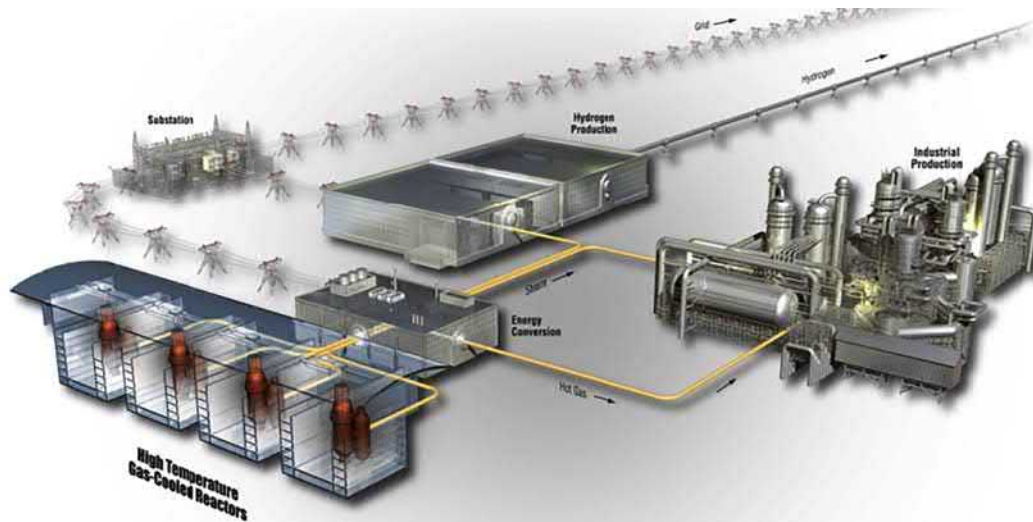


Fig. 6 Artist's view of a 4-pack modular HTR for process heat, hydrogen production and electricity generation (INL).

and to replace fossil generation plants including in off-grid areas. These advanced reactors are deployable either as single or multi-module nuclear power plants, and are designed to be built in factory workshops and shipped to utilities for installation as demand evolves. **Fig. 6** shows how a multi-module pack could be configured to polygenerate heat, hydrogen and electricity.

Several designs ensure enhanced safety performance through inherent and passive safety features as well as suitability for cogeneration and non-electric applications thus opening opportunities for hybrid energy system architectures combining nuclear, fossil and renewable energy carriers. They have reached different stages of development and target near-term deployment with several vendor companies participating in feasibility and licensing studies.

About 16 of these SMR designs are HTRs with one currently under construction and commissioning in China (HTR-PM). Several of these reactors are derivatives or evolutions of earlier parent concepts, e.g. the HTR Modul for the pebble bed designs and the MHTGR-350 (designed by General Atomics) for several prismatic block designs. For the HTR concepts in **Table 2**, publicly available design information can be found in (IAEA, 2018).

R&D efforts as well as cooperation between all stakeholders (vendors, suppliers, regulators, utilities/end-users, investors, politicians, public etc.) are ongoing and organized at different national and international levels including GIF, IAEA, OECD-NEA, and are including economic analyses, as well as novel investment options and licensing approaches, e.g. (Gougar et al., 2020; Kalilainen et al., 2019). While the nuclear accident in Fukushima in 2011 has dealt a blow to nuclear energy development for several years, the ongoing debate about climate change mitigation has created new interest in low-carbon technologies in several countries and specifically awareness of the need to address the massive energy requirements of the process heat market in industrialized countries. As shown in **Table 2**, interest in the inherently safe, highly efficient and versatile HTR technology is steadily growing, and new

Table 2 Summary of HTR-type small modular reactor concepts.

Concept	Developer
<i>Pebble Bed</i>	
HTR-PM	Tsinghua University, China
Xe-100	X-energy, USA
HTMR-100	Steenkampskraal Thorium Ltd., South Africa
PBMR-400	Pebble Bed Modular Reactor SOC Ltd., South Africa
AHTR-100	Eskom Holdings SOC Ltd., South Africa
<i>Hexagonal Block</i>	
GTHTR300	Japan Atomic Energy Agency, Japan
MHTGR-350	General Atomics, USA
GT-MHR	OKBM Afrikantov, Russian Federation
MHR-T Reactor/Hydrogen Production Complex	OKBM Afrikantov, Russian Federation
MHR-100	OKBM Afrikantov, Russian Federation
SC-HTGR	Framatome Inc., USA
MMR-5, MMR-10	UltraSafe Nuclear Corporation, USA
StarCore HTGR	StarCore Nuclear, Canada
U Battery	U Battery, UK

demonstration projects, in particular for the coupling of the nuclear reactor with a process heat end-user installation, are being implemented to help de-risk (and possibly shorten the time to) industrial deployment.

Beyond electricity: Emission-free process heat and cogeneration

Because HTRs are particularly fit for process heat applications and cogeneration of heat and power, this section is dedicated to non-power utilization aspects of nuclear energy, which has very significant potential impacts since it reduces fossil fuel consumption in areas beyond the electric power market, and thus enhances energy security, further increases the reduction of noxious emissions, and helps mitigating climate change. Already with earlier reactor types, nuclear cogeneration was performed in many countries and with several types of reactors including Light Water Reactors (LWR), Heavy Water Reactors (HWR), and Sodium Cooled Fast Breeder Reactors (SFBR). District heating (80–150 °C) is probably the most widely found application of nuclear heat: 46 reactors in 12 countries, including for instance Slovakia, Switzerland, Russia and China were and are used for this purpose.

Examples for low temperature applications of nuclear heat include seawater desalination (Japan, Kazakhstan), paper and cardboard industry (Norway, Switzerland), heavy water distillation (Canada), or salt refining (Germany).

The technology options for nuclear process heat utilization with HTRs were already documented quite early (Schulten, 1976). A survey of two decades of activities in Germany is given in (Verfondern, 2007a), and further potential is outlined in (Verfondern, 2007b).

The HTR produces heat at a much higher temperature level (exergy) than the LWR. This opens the possibility to replace a large number of existing industrial cogeneration plants delivering process steam in the 500–600 °C temperature range. Very significant amounts of such process steam are consumed in the chemical and petrochemical sector as well as in the fertilizer industry, where today this steam is mostly produced by gas or coal firing.

For several stakeholders, in particular in those countries where natural gas is expensive, the prospect of hydrogen production continues to be the main driver for development and potential deployment of the HTR and VHTR. Process heat from an HTR can be used for several more or less advanced methods of hydrogen production. The most near-term option is steam methane reforming of natural gas with steam at 700 °C, 5.5 MPa. Owing to the external heat supply, more than a third of natural gas is saved. In the 1980s, the necessary components, e.g. heat exchangers or reformers, were developed and tested under nuclear conditions in Germany and in Japan (Harth et al., 1990).

Processes and components for allothermal and steam coal gasification processes were also tested in Germany. They require typically steam in the range of 750–900 °C at 0.1–4 MPa. Although external heat supply makes coal upgrading more efficient, these processes release large amounts of unwanted CO₂.

These activities were brought to a temporary halt in an anti-nuclear climate after the Chernobyl accident, with inexpensive oil and gas and in absence of CO₂ emission restrictions.

As steam methane reforming to produce hydrogen consumes natural gas and generates CO₂ emissions in the process, direct water splitting methods are under investigation in several countries as a clean alternative. HTRs can provide steam for a rather low temperature process, the copper-chlorine (Cu-Cl) cycle, requiring steam at just over 500 °C (Rosen et al., 2012). Other prominent hydrogen production methods are (i) High Temperature Steam Electrolysis (750–950 °C) where a part of the required water dissociation energy is delivered in the form of heat, and (ii) thermo-chemical cycles such as the Sulfur-Iodine Cycle where one of the three process steps (SO₃ decomposition) requires heat input at 850 °C (Yan and Hino, 2011). This process is particularly suitable for VHTR operating at 900–1000°. The market for bulk hydrogen is currently very large and growing fast, with distribution networks already in place in several countries. To justify large-scale production of hydrogen, the development of a specific “hydrogen economy” is not required. Hydrogen uses include upgrading of increasingly heavy oils to lighter fractions, hydrogenation processes, hydro coal gasification, metal refining, ammonia production for fertilizers, the synthesis of methanol or synfuel, or the use of hydrogen in combination with fuel cells as a transport fuel. For some Asian countries, the replacement of coke by hydrogen for direct iron ore reduction is of particular interest to cut back emissions from steel making. Finally, hydrogen can also play a role in carbon capture and utilization processes, which would use CO₂ together with hydrogen as a feedstock for the fabrication of a wide array of possible products ranging from plastics or synfuel for aviation to construction materials. A summary of such processes and products is provided in (Styring et al., 2011).

In the context of energy system integration efforts with growing fractions of variable renewable electricity in many countries, it is of particular interest that the cogeneration capability of HTRs would allow it to contribute to grid stabilization (“peak shaving”), e.g. by modulating the production of (storable) hydrogen depending on the electricity demand in the grid, similar to what is currently envisaged for wind energy (“power to gas”).

To further corroborate the incentive for process heat and hydrogen production with nuclear energy, several market research, economic analyses, trade studies, and business plans were recently prepared in several countries, some of which are publicly available (e.g. Angulo et al., 2012; Bredimas, 2012; INL, 2012; Konefal and Rackiewicz, 2008; Shropshire, 2013).

Outlook

The unique capability of the HTR to produce process heat above 600 °C makes it an efficient reactor type to displace fossil fuels in various applications such as producing electricity, non-conventional hydrocarbon fuels from coal or biomass, and process heat for



Fig. 7 Installation of RPV into HTR-PM reactor building in 2016.

energy-intensive industries (oil refining, petro-chemistry, oil sand recovery, chemistry, steelmaking, etc.). Several market studies confirmed the potential for the HTR system to be used in such applications while the economic boundary conditions (e.g. price of natural gas, CO₂ tax) for market deployment have become clearer. The inherent safety characteristics of the HTR are a precious asset in contributing convincing answers to today's concerns in terms of nuclear safety, energy security, and climate change.

Current research performed within frameworks supported by GIF, IAEA and OECD-NEA, as well as specific national programs address primarily issues related to R&D, licensing, demonstration, and deployment. In particular, the multinational cooperation within GIF (GIF, 2018) allows sharing efforts to advance the technologies and to accelerate development in view of licensing and deployment. Currently, cooperation on the VHTR within GIF focuses on development and qualification of (i) fuel, (ii) structural and functional materials, (iii) hydrogen production processes and (iv) computer tools. GIF has also produced guidance for (V) HTR designers, e.g. in the areas of sustainability, economy, reactor safety, non-proliferation questions or energy system integration. The cooperation is clearly geared towards producing licensing-relevant information across the signatory countries and has recently opened to closer interaction with competing designer and vendor companies. Furthermore, the experimental reactors in Japan (HTTR) and in China (HTR-10) offer unique opportunities to qualify technologies and design codes. The next hurdle towards deployment is being taken by China with the ongoing commissioning of the HTR-PM demonstrator (Fig. 7). Japan will perform further safety demonstrations on the HTTR.

Since 2002, the bi-annual International Topical Meeting on High Temperature Reactor Technology is the sole international conference with focus on HTR and process heat applications (<https://htr2020.org/>).

Although very substantial results were produced, in particular by the signatories of the GIF VHTR System Arrangement, funding opportunities for a demonstrator coupled with an end-user process will have to be found soon to capitalize on previous investments. Several such international initiatives are on the way. Their success will depend on how much and where nuclear will be allowed to contribute to climate change mitigation, be it for political and public acceptance reasons or for economic boundary conditions (cheap natural gas, CO₂ tax, financial risk).

See Also: Fuel Design and Fabrication: TRISO Particle Fuel; Pebble Bed Gas Cooled Reactors; Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors.

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Lead Cooled Fast Reactors

Alessandro Alemberti, Ansaldo Nucleare SpA, Genova, Italy

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Introduction

The Generation IV international Forum started its activity with the choice of six reactors reference systems, including the LFR. These systems were identified as meeting the objectives of the “fourth” generation nuclear systems, viz. sustainability, safety and reliability, economic competitiveness as well as proliferation resistance and physical protection (Stanculescu, 2021; GIF, 2014a; GIF, 2018b).

At that time the design of modern reactors based on lead coolant was practically at its inception although some previous experience was already available. In fact, the use of lead or lead-bismuth coolants was proposed and investigated in the United States starting from 1942 and then abandoned around 1950 due to technological problems related to corrosion and coolant purification. In Russia however the investigation continued bringing to the realization of lead-bismuth cooled reactors for submarine propulsion in the 1960s (Zrodnikov, 2000). In the United States the interest for LFR systems was restarted thanks to the activities on Small Secure Transportable Autonomous Reactor (SSTAR) around 2000. In Europe proposals came out essentially in relation to Accelerator Driven Systems proposed for nuclear waste transmutation during the 1990s. During the Russian early and pioneering applications, the main aspects of the technology were investigated, and solutions identified to face technological problems related to the use of heavy metal coolants. Although some issues remain to be faced, the technology of LFR is becoming recently very attractive, and subject both to an increase of investigations and actors, thanks to some specific characteristics of the coolant and expected system safety performance. Several initiatives are taking place around the world and this article will briefly review them, providing references for readers interested in additional details.

Molten lead technology is today considered valuable for a number of nuclear applications: Lead cooled Fast Reactors are the most investigated application. Lead or lead-bismuth eutectic is also considered for transuranic elements transmutation, i.e. transmutation of plutonium and heavier elements accumulated in the nuclear fuel of light-water and other types of nuclear power reactors. This is typically achieved by neutron absorption in the transuranic elements in subcritical cores that are driven by protons

accelerators that generate neutrons by protons-induced spallation reactions (when very high energy protons or ions impinge on heavy-element targets like tungsten or lead, the spallation reaction will “knock-out” a number of neutrons); these systems are also called Accelerator Driven Systems. Lithium-lead is also proposed as a coolant and as a breeder of tritium (Coen, 1985) for fusion applications. After a description of the coolant characteristics, advantages and issues, a synthetic review of the main initiatives around the world is presented.

LFR and the goals of Generation IV

The objectives of the Generation IV International Forum are (Stanculescu, 2021): Sustainability, Safety, Economics, Proliferation Resistance and Physical Protection. LFR is expected to excel when challenged against such goals. The basic arguments are briefly summarized below.

Sustainability

The LFR, as a fast reactor system with its associated closed fuel cycle, is effective in fully utilizing the energy value of uranium. It is well known that fast reactor with closed fuel cycles can provide factors of up to 100 times greater natural uranium utilization than existing thermal reactor fuel cycles. This capability thus fully supports the first-Generation IV goal of increased sustainability. The neutronic properties of lead and lead-bismuth coolants (low neutron absorption and moderation) readily enable sustaining a hard neutron energy spectrum, providing good neutron economy and performance flexibility.

In order to achieve nearly 100% natural uranium utilization, it is necessary to reprocess the fuel after it reached its maximum permissible burnup (burnup is, effectively, the amount of energy that can be generated per unit fuel weight before the fuel has to be removed from the core) and recycle all the uranium and transuranic elements back to the core. The initial core must be loaded with fuel enriched with plutonium. However, after a few recycles the core reaches an equilibrium composition. In such “equilibrium mode”, the fuel composition at the beginning (end) of a cycle is the same as the composition at the end (beginning) of the previous cycle and either natural or depleted uranium is the only needed feed material as schematically illustrated in Fig. 1 (Artioli et al., 2010). The waste is made of only fission products as well as reprocessing and fabrication losses. Such a fuel cycle can be implemented as well in other types of fast reactors such as sodium-cooled (Heidet, 2021) and gas-cooled (Hatala, 2021) fast reactors.

Economics

Economic performance is generally difficult to quantify for new, advanced systems, yet there are several characteristics of LFR systems that suggest a potential in terms of economic performance respect to present Gen III or Gen III+ systems. Key among these characteristics are:

- the chemical nature of lead (or lead-bismuth) coolant, should it come in contact with air or water, guarantees the absence of strong interactions supporting the development of simplified and efficient designs that do not require complex, expensive intermediate coolant systems to isolate the primary coolant from the energy-conversion system (usually water) coolant.
- lead coolant retention properties of fission products are very important to reduce the source term and expected to enable a reduction (or possible elimination) of the emergency planning zone;
- the very high boiling temperature of the lead coolant (1749 °C) virtually eliminates coolant boiling and related safety challenges enabling further simplification of design and safety demonstration;
- the low partial vapor pressure of lead (2.9×10^{-5} Pa at 400 °C) allows primary system operation close to atmospheric pressure (as is also the case for sodium cooled fast reactors and molten salt reactors);
- the high operating temperature of LFRs enables high power conversion efficiencies, above 40%;
- the use of oxide and nitride fuels that can reach high burnup thus reducing fuel cycle costs;
- load following operations are strongly favored by the small difference between “cold state temperature” and operational temperature and the system can be coupled to an additional energy stored capability;

In summary, a good economic performance of LFRs can be expected due to the favorable inherent characteristics of the lead coolant.

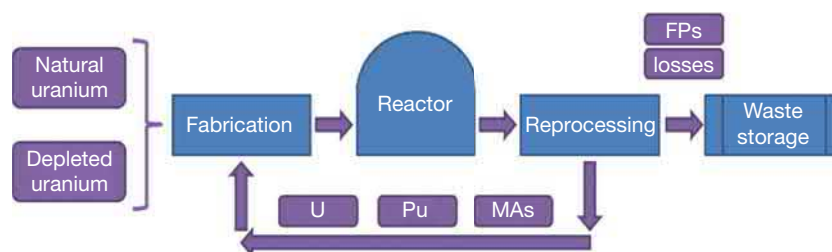


Fig. 1 Equilibrium fuel cycle.

Safety and reliability

Increased safety and reliability of the next generation nuclear systems is becoming a major condition for public acceptance of nuclear power. Again, the basic and intrinsic characteristics of lead coolant plays an important role in terms of safety as well as reliability of the LFR systems. Following is a highlight of the most important aspects and consequences; a more detailed discussion can be found in (GIF, 2014b) and (GIF, 2020).

- high primary coolant boiling temperature protects the core against adverse neutronic feedbacks from voiding effects and protects the system against one of the possible initiators of a severe accidents. Only transient voids generated by cladding rupture and consequent fission gas release needs to be considered, but their effect will be naturally limited in time and value;
- thanks to the scattering properties of the coolant and its high atomic mass an LFR core can be designed with a large pitch-to-diameter ratio, providing enhanced safety against flow blockage and improved heat removal through higher natural circulation;
- LFR systems are not dependent on off-site power or emergency on-site back-up power thanks to their high thermal inertia and natural circulation capability enabling effective passive decay heat removal;
- the lead coolant provides a capacity for retention of volatile fission products, such as cesium and iodine, so that significant accidental release outside the primary coolant system is unlikely (Dubenkov et al., 2018);
- The potential energy stored in an LFR primary system is very low thanks to the relatively inert chemical nature of the lead coolant and the absence of hydrogen accumulation (Toshinsky et al., 2011), enhancing LFR's resilience in accident situations and facilitating the success of mitigation actions;

With respect to reliability, high levels can be achieved through design and extensive use of passive safety systems.

Proliferation resistance and physical protection

The last goal area for Generation IV systems aims at increased proliferation resistance and physical protection (PRPP). These characteristics are enhanced for LFR systems since:

- A common feature of fast reactor concepts with closed fuel cycle is the already cited sustainability, not requiring the generation/supply of additional enriched uranium, plutonium or other fissile material after reaching the equilibrium cycle.
- Some LFR concepts integrate/co-locate fuel reprocessing and fabrication functions into the reactor site so that, once equilibrium cycles are reached, only natural uranium is transported to the site as fuel feed, and only the relatively short-lived fission products leave the site for disposal;
- The possibility of long or very-long core life (and the corresponding increase of the time between refueling operations) reduces the proliferation risk associated with fuel handling operations.

For further insights on this topic the interested reader may consult the (GIF, 2021).

In summary, due to the potential for substantial improvements arising from the intrinsic characteristics of the coolant as well as from design and safety choices implemented in the considered LFR designs, in light of the technology advances and studies completed in recent years, it is appropriate to claim that the LFR operational and safety characteristics are meeting the GIF goals for the development of the next generation nuclear energy systems.

LFR design

LFRs designers have been facing from the beginning the novelty to work in a completely new environment (very different from lighter fluids) with the possibility to exploit as much as possible the specific characteristics of the fluid. Some early work, starting from the so called "loop" configuration, suggested immediately, due the weight of the coolant and associated seismic and supporting problems, to switch to the "pool" configuration with a primary side circulation organized inside a main vessel, and, in most cases, contained inside a safety vessel to prevent accidents caused by coolant leakages. However, such common characteristics have been declined in very different ways, thanks to engineer's innovation attitude and imagination. It is important therefore, before presenting general system arrangements and design solution, to highlight briefly the main coolant characteristics that can be used extensively in the design development.

Lead and in general heavy liquid metals are characterized by a very high density, 10 times higher than the water density. As a result, most of the pool immersed components are "floating" in the pool, so that they are normally "supported," i.e. pushed down, by the roof and not suspended. This fact generates some possible difficulties when trying to push some long elements inside the pool; the higher density is also limiting the maximum speed of lead coolant if we want to minimize erosion and maintain a dynamic pressure close to that of the water. Lead is also characterized by a high melting point (327 °C) and a very high boiling point (1749 °C). For lead-bismuth eutectic (LBE) the melting point is reduced to 123 °C with a boiling point still very high, 1670 °C. The high melting and boiling points introduce the need to consider the possibility of coolant freezing as well as practically exclude the existence of two-phase flow in the primary side, while the system pressure can be atmospheric, so no need of pressurization. The low vapor pressure provides some advantages since practically no evaporation takes place at the normal operating temperatures presently envisaged (2.9×10^{-5} Pa at 400 °C). Another very important characteristic is the low chemical reactivity

of lead with coolants like water or air. Since the oxidation reaction is slow, secondary side water coolant can be allocated directly inside tubes immersed in the primary pool, without the need of an intermediate loop. From a “nuclear and neutron economy” point of view lead has some very important advantages. As you can imagine such big nuclei are going to scatter the very small neutron particles in the core without absorbing energy during the collision process and, thanks to the low absorption cross section, the core can easily be designed as a breeder. This fact has important consequences in terms of the possibility to implement a closed fuel cycle. While its “neutronic” behavior is an advantage, it has also an important impact on the thermal hydraulic characteristics of the primary system. In fact, due to the scattering properties of such heavy nuclei, we are not forced to use packed fuel pins in our fuel assemblies. The pitch between the pins can be adequately spaced and designed to obtain very low-pressure losses in the primary system, in favor of natural circulation and of the overall safety characteristics. This fact induced designers to extensively use passive safety systems, as a natural way to promote an increased level of safety for such advanced nuclear reactors. The high boiling point of lead also has important effects on possible severe accident initiators related to coolant boiling reactivity void effects: they are considered highly unlikely due to the very high coolant boiling temperature, even in the case of loss of off-site and on-site backup power. Last, but not least, the lead coolant provides a capacity for retention of volatile fission products, such as cesium, iodine, and polonium, so that significant accidental release outside the primary coolant system is strongly limited.

LFRs areas of application

The possible areas of application of LFRs are very wide; as demonstrated by the growing worldwide interest, the possible sizes of LFRs range from the micro reactors to the whole range of small and medium reactors while future, longer term, options will cover also the large size reactors. Areas of applications are also strongly dependent from the maximum achievable primary side temperature and many efforts are dedicated by the international community to increase the operating temperature looking for maximum flexibility and efficiency of the thermal cycle. Apart from the obvious use of LFRs for electricity generation, with its primary cycle temperatures (depending on design) between 400 and 520–620 °C the LFR is highly compatible with solar thermodynamic plants (normally designed at 520–550 °C) and permit to explore possible synergies between nuclear and renewables.

In terms of flexibility the LFR presents important advantages depending essentially from the high “cold” temperature imposed by the lead melting point which limits in case of power modulation the temperature excursion. Practically speaking, if the primary flow is maintained constant, the primary temperature difference is roughly proportional to the power produced and for an LFR with a constant minimum cold temperature of 400 °C, only the “hot” structures are subjected to temperature excursion. These facts can greatly enhance the reactor flexibility allowing some degree of load following with important benefits in terms of electricity cost and revenues, although the preferred option in terms of flexibility is the introduction of energy storage, already introduced by some recent designs. With the suitable future increase of the hot temperature the LFR may be used also for hydrogen or industrial heat production, extending further the range of applicability of the technology. Thanks to its high safety performance and low likelihood of environmental radioactive release LFRs can be sited relatively close to small towns and there are projects considering this technology to provide district heating grids. For additional information on economics aspects, flexibility of Generation IV systems (GIF, 2019).

Research challenges

In the case of the LFR, the challenges include those related to the high melting point of lead; its opacity; coolant mass as a result of its high density; and the potential for corrosion when the coolant is in contact with structural steels. A specific challenge of the LFR is related to the high melting temperature of lead (327 °C) requiring that the primary coolant system be maintained at temperatures to prevent the solidification of the lead coolant. The pool type configuration and appropriate primary system design can provide effective solutions of the issue. The opacity of lead, in combination with its high melting temperature, presents challenges related to inspection and monitoring of reactor in-core components as well as fuel handling. Such issues can be addressed by appropriate and specific design features, for example, by innovative core configurations with the upper (non-active part) of fuel elements extended above the lead-free level (a choice pursued by some European designs). The high density and corresponding high mass of lead as a coolant are normally faced by structural design to prevent seismic impacts on the reactor system. Innovative primary systems configurations with short reactor vessels coupled with the introduction of seismic isolators are possible options to address such issues (EU 7FWP, SILER, n.d.). Perhaps the most significant challenges result from the tendency of lead at high temperatures to be corrosive, through a dissolution mechanism, when in contact with structural steels. This tendency, which is accelerated at higher temperatures, requires careful material selection and component and system monitoring during plant operations. Pending the development of new materials resistant to lead corrosion at high temperatures (e.g., for fuel cladding), approaches including anti-corrosion surface treatment of already qualified steels with careful coolant chemical (especially oxygen) control have been proposed to protect materials immersed in lead from corrosion (IAEA, 2019). In the presently proposed design configurations, a relatively low operating temperature strongly helps to reduce and practically eliminate the impact of this issue. The coolant purification needs to be carefully controlled and actively pursued in order to minimize deposits and the chances of flow blockages. The polonium issue (^{210}Po is a highly toxic alpha-emitter formed, in a reactor core, by neutron capture in the natural isotope ^{209}Bi) should be carefully addressed not only for LBE cooled system where the ^{210}Po production is very high, but also for pure lead systems due not only to bismuth impurities in the lead but also to double transmutation processes. Each of these areas of technology challenge is the topic of active ongoing research or is object of a specific provision at design level.

Finally, it is recognized that LFR fuel development can benefit from the vast data base and ongoing research being conducted for other reactor types, and especially from the Sodium Fast Reactor (SFR) fuel development activities. However, it is also recognized that where specific features of the LFR linked to irradiation conditions (i.e., linear power, temperature, neutron energies, fluence, and fuel-cladding interactions) differ from other reactor types, these differences need to be considered and addressed.

A reference LFR design: ALFRED

Following is a description of one reference LFR design, used as an example, to go through its main features in more detail.

Alternative design solutions will be described using cursive characters through the following ALFRED design description.

The Advanced LFR European Demonstrator (ALFRED) has been conceived in the frame of a European Commission funded project called LEADER from 2010 to 2013 (EU 7th Framework Program) (EU 7FWP, LEADER, n.d.). Sixteen partners were involved in the design activities and the reason for using it as a reference here is that most of the design details and data are public and so can be used by researchers without restrictions. ALFRED had been sized at 300 MWth, corresponding to 125 MWe and is cooled by pure lead. The configuration originally developed in the frame of LEADER project is used as a reference, further details on the actual design evolution will be presented as part of the following overview of LFR systems. A GIF webinar is available on internet on the ALFRED design solutions (GIF, 2018a, ALFRED webinar) and can be used to complement the information provided below. A previous webinar on the reference systems considered for activities of GIF System Steering Committee is also available (GIF, Lead Fast Reactor, 2017) and can be consulted to explore other possible solutions in more detail.

The configuration of the primary system is pool-type. This concept permits containment of all the primary coolant within the Reactor Vessel (RV), thus eliminating problems related to out of vessel circulation of the primary coolant (Fig. 2).

The Reactor assembly presents a simple flow path, with a Riser and a Downcomer. The heat source (the Core), located below the Riser, and the heat sink, the Steam Generators (SG), at the top of the Downcomer, allow for efficient natural circulation of the coolant. The primary coolant moves upward through the pump impeller to the vertical shaft, then enters the SG through the inlet holes, flows downwards on the shell and exits the SG to reach again the core.

The free level of the hot pools inside the Steam Generator and Primary Pump units is higher than the free level inside the cylindrical structure surrounding and supporting the core, the so called Inner Vessel (IV), Figs. 2 and 3, the different heads depending on the pressure losses across component parts of the primary circuit. The plenum between the primary coolant free levels and the reactor roof is filled by cover gas.

One of the basic characteristics of ALFRED primary system configuration is the primary pump being located in the hot region.

Other configurations exists as well, some of them are similar to SFR configuration with the pump located in the cold region with pressurization of the lower part of the core while in others the pump is used to elevate the cold fluid, before entering the SG, to generate the elevation head needed to balance the primary system pressure losses. Each system configuration presents advantages and disadvantages and the location of the pump in the primary flow path completely characterize and impact other design choices.

In the case of ALFRED, the RV is cylindrical with a toro-spherical bottom head. It is anchored to the reactor cavity from the top, by means of a "Y" junction (Fig. 2). A steel layer covering the reactor pit constitutes the Safety Vessel (SV).

Other types of vessel exist and will be highlighted in the following sections, while solutions using a main vessel surrounded by a safety vessel are also under investigation.

The dimensions of the gap between the SV and the RV are normally strongly conditioned by the size of the In-service Inspection tools. The SV mitigates the consequences of through-wall cracks in the RV with leakage of lead: any RV leakage is discharged into the SV. The RV and the SV are arranged in such a manner that, in case of a RV leak, the resulting primary coolant always covers the SG inlet and the lead circulation flow path is indefinitely maintained. The IV has two main functions: Fuel Assemblies (FAs) support

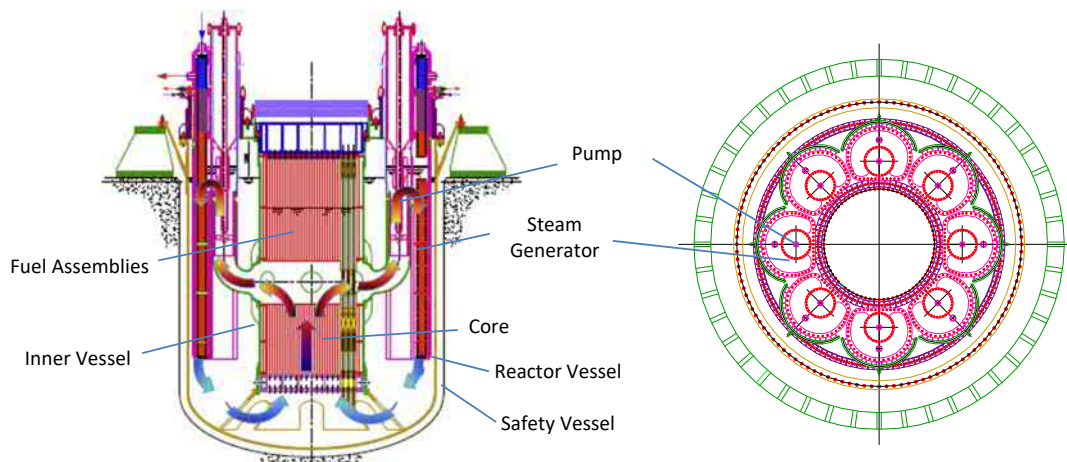


Fig. 2 ALFRED primary system: vertical (with primary flow path) and horizontal sections.

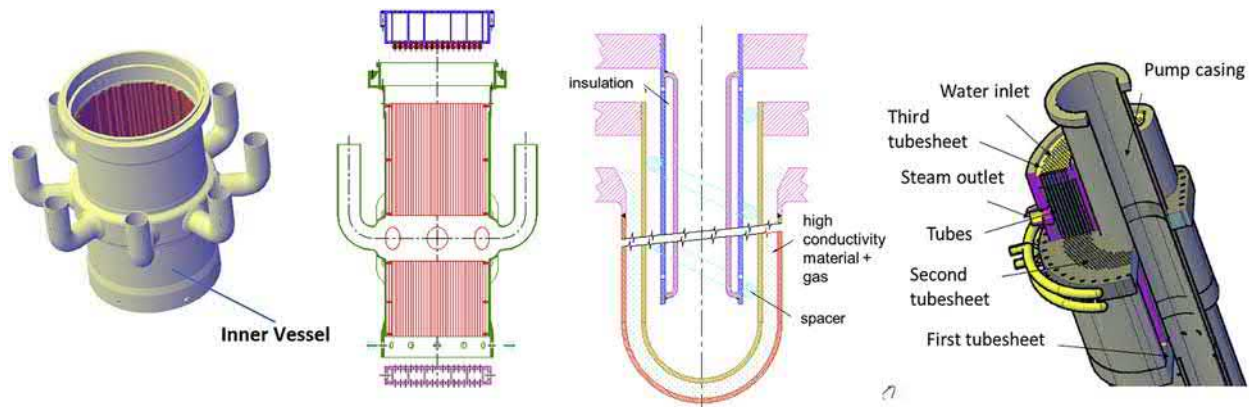


Fig. 3 Inner Vessel 3-D sketch, axial section and bayonet SG concept and arrangement.

and separation between hot plenum and cold plenum. It is fixed to the cover by bolts and is radially restrained at the bottom. Lead flow is guided from the FAs outlet towards the primary pump (PP) inlet pipes by a toroidal half-ring. Moreover, the pipes that connect the hot zone with the inlet of the PP are integrated in the IV. The cylindrical IV (Fig. 3) has a double wall shell: the outer thick wall has a structural function, while the inner thin wall follows the core section profile. The Core Lower grid is a box structure with two horizontal perforated plates connected by vertical plates. The plate holes are the housing of the FAs feet and the plate distances must be sufficient to assure the verticality of FAs. The lower grid is mechanically connected to the IV with pins. The Core Upper grid is a box structure like the lower grid but stiffer. It has the function to push down the FAs during the reactor operation. A series of pre-loaded disk springs press each FA to its lower housing. A hole is present for each disk to allow the passage of instrumentation (i.e., thermocouples, neutron detectors, etc.).

The steam generator and primary pump are integrated into a single vertical unit. Eight SG/PP units are located in the annular space between the cylindrical IV and the RV wall. The primary pump is placed on the hot side of the steam generator, having its mechanical suction in the hot pool inside the IV. The primary coolant moves upward through the pump impeller to the vertical shaft, then enters the SG through the lead inlet holes, flows downwards on the shell and exits the steam generator. The pump motor is located above the reactor roof. Each SG consists of a bundle of 542 bayonet tubes immersed in the RV pool for 6 m of their length. The bayonet tube is a vertical tube with external safety tube and internal insulating layer, composed of four concentric tubes (Fig. 3): a slave tube, an inner tube, an outer tube and an outermost tube. The internal insulating layer (delimited by the slave tube) has been introduced to ensure the production of superheated dry steam: in fact, without an insulating system, the high ΔT ($\approx 115^\circ\text{C}$) between the rising steam and the descending feedwater would promote steam condensation in the upper part of the steam generator. The gap between the outermost and the outer bayonet tube is filled with pressurized helium and high thermal conductivity particles (such as synthetic diamonds) to enhance the heat exchange capability. In case of an external tube break this arrangement guarantees that the primary lead does not interact with the secondary water. Moreover, a tube break can be easily detected by monitoring the Helium gap pressure. The Primary Pump is surrounded by the SG tube bundle, and its housing is the SG casing. The Pump is fixed to the top of SG casing by a bolted joint for easy removal of the component.

For what the SG units are concerned, several different possibilities have been explored and used by designers: helically coiled steam generators (already used for some advanced LWRs as well as SFRs systems), single bayonet tubes, plate heat exchangers, etc. Again, in the following overview of systems under development the different designers choices will be highlighted.

The core configuration (Fig. 4) is constituted by wrapped Hexagonal Fuel Assemblies. It utilizes MOX as fuel and uses hollow pellets and a low active height in order to improve natural circulation. The total power is 300 MWth. The core is made of 171 FAs, 12 CR (Control Rods) and four SR (Safety Rods), surrounded by 108 Dummy Elements ($\text{ZrO}_2\text{-Y}_2\text{O}_3$) shielding the IV. Each Fuel Assembly is about 8 m long and consists of 127 fuel pins, fixed to the bottom of the wrapper and restrained sideways by grids. Tungsten deadweight (Ballast) prevents buoyancy forces in lead. Upper elastic elements (cup springs) prevent lifting induced by hydrodynamic loads and accommodate axial thermal expansions. The FAs upper end extends beyond the lead free surface in the cover gas for easy inspection and handling. In this way it is possible to achieve refueling without the need of in-vessel refueling machines.

This solution is strongly characterizing the ALFRED design, other solutions have shorter fuel assemblies with blocking devices in the FAs lower part (foot) or an above core structure is provided to keep the elements in position. The choice is strongly connected to other design characteristics and the chosen design option strongly condition other design aspects.

The Decay Heat Removal system (DHR) consists of two passive, redundant and independent systems, DHR1 and DHR2, both composed of four Isolation Condenser (ICs), immersed in water pools, connected to four SGs on the secondary side (i.e., one IC for each SG). The system design considers the single failure criterion, since three out of four ICs are sufficient to remove the decay heat power.

The DHRs are dedicated safety systems, not used for normal operation. The separation is achieved by placing the two DHRs pools in physically different locations. A physical structural barrier or another means of protection will be placed between adjacent

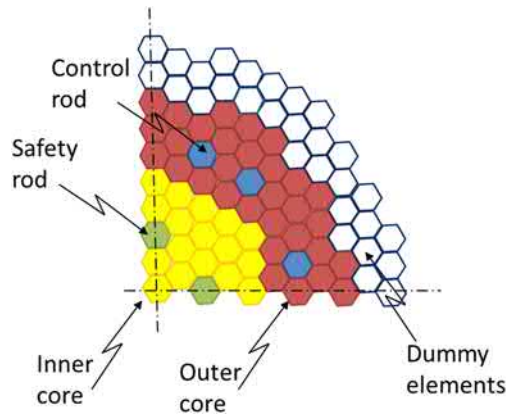


Fig. 4 ALFRED core.

ICs to ensure that failure of one of them could not harm another one. The diversity requirement has been relaxed (in any case, two DHR systems will be fabricated by diverse manufacturers) due to the high redundancy and considering that the SG tube bayonet concept allows a continuous monitoring of the SG status.

Other systems and ALFRED itself in its new system configuration include a second DHR system using independent deep coolers.

Both systems are completely passive, with an active actuation through valves equipped with redundant and diverse energy sources (batteries or locally stored energy). Each DHR system must be ready to operate after the reactor trip in order to remove the decay heat power, in case of unavailability of the normal path (i.e., the by-pass to the Condenser). The actuation logic guarantees the actuation of DHR1 first, whereas the DHR2 actuates only in the case of failure of the first system. The total number of isolation condensers called to operate can never exceed the four units, in order to avoid an excessive cooling of the primary coolant leading to fluid solidification.

Each of the four independent IC sub-systems (Fig. 5) consists of:

- One heat exchanger (isolation condenser), constituted by a vertical tube bundle with an upper and a lower horizontal header.
- One water pool, where the isolation condenser is immersed; the amount of pool water is sufficient to guarantee 3 days of decay heat removal operation.
- One condensate isolation valve (to meet the single failure criterion, this function must be performed at least by two parallel valves).

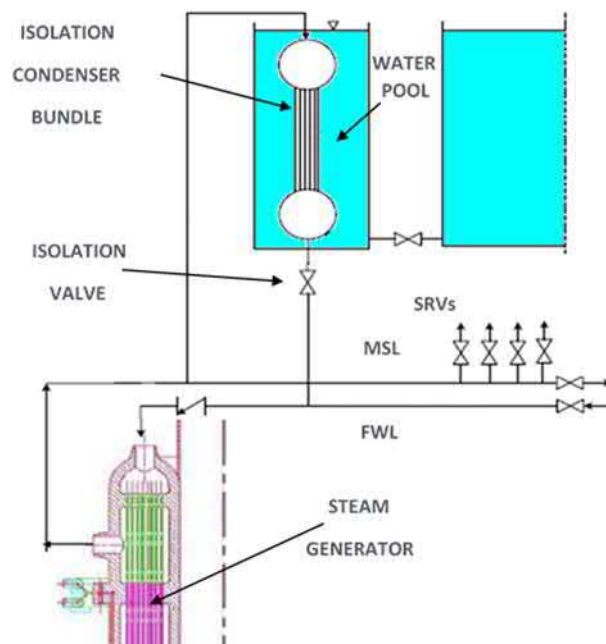


Fig. 5 ALFRED DHR scheme.

Each IC is connected to a SG: the upper header of the IC is connected to the main steam line and the lower header of the IC is connected to the main feed water line.

In normal operation, the isolation valve below the condenser is closed, the condenser is full of water and no heat exchange takes place. As the IC subsystem is called to operate, the feed water line and steam line are isolated, and the condensate isolation valve opens. The subcooled water stored in the IC tube bundle drains into the SG, due to the hydrostatic head existing between the reactor and the IC. The IC tubes and headers become empty and their internal cold surface starts the steam condensation and hence to transfer heat to the cold pool water. The steam condensation causes a pressure reduction which calls for other steam from the SG. The water injected into the secondary system from the IC during the draining phase, vaporizes into the SG tube bundle contributing to the secondary system pressure rise: the safety relief valves continue to operate to reject to the atmosphere the excess of steam and to guarantee a secondary side pressure of 195 bar(a). When the IC reaches its steady state condition and starts to remove the thermal power from the primary coolant, the secondary side pressure rise decreases and finally stops leading to the closure of the safety relief valves on the main steam line.

For LFR designs the preference has been from the beginning to introduce passive decay heat removal systems thanks to the favorable characteristics of the lead coolant. The envisaged solutions possible are many and depend basically on the size of the reactor. For small sized reactors a so called Reactor Vessel Air Cooling System has often been used to remove the decay heat directly from the vessel wall through radiation, other systems (in the very small range) may use directly conduction through Vessel walls to provide cooling (with the disadvantage of having high heat losses in operation and the advantage of having no need of actuation). Of course, active systems can be used but the preferred way of introduction is their use as non-safety grade systems so that additional cost is eliminated (if passive systems are already available in case of loss of AC power). In case of pure lead systems and use of water as secondary fluid the use of boiling condition is practically mandatory due to the high temperature of the primary system. In case of LBE and lower temperature condition also the DHR pressure can be reduced and, in some cases, single phase conditions can be used. Other fluids can be considered such as diathermic oil or molten salt as well. They provide advantages in term of single-phase conditions, no or lower pressure but introduce problems related to coolant interaction with air (possibility of fires) and/or coolant dissociation or properties change over the range of allowed temperatures. In any case the design of DHR systems must consider the "freezing" temperature of the coolant in order not to impair the natural circulation of the primary side.

The Secondary System proposed for ALFRED is constituted by a steam-water cycle characterized by a turbine inlet pressure of 18 MPa, feed water entering the SGs at 335 °C and superheated steam at 450 °C. This is exactly the same cycle that was used for the sodium fast reactor Superphénix (SPX) and has an overall thermal efficiency greater than 40%. *Studies have shown that using supercritical water on secondary side does not provide a big advantage in terms of efficiency while supercritical CO₂ cycle are considered but must be the subject of further development before industrialization and should take into account the fact that they present a real advantage only at higher temperature conditions (over 700 °C).*

LFR development status and initiatives around the world

This part of the paper is far from being exhaustive but will try to give a brief overview of the present initiatives in different countries in both public and private sectors. We will try to cover both the pure lead and LBE technologies as well as the critical reactors and the ADS initiatives, starting obviously from the reference systems assumed as a basis for the Generation IV activities: BREST-OD-300 for Russian Federation, SSTAR for United States of America and ELFR for the European Community.

BREST-OD-300

The BREST-OD-300 reactor is a pilot demonstration reactor (300 MWe) considered as a prototype of future commercial reactors of the BREST family for large-scale nuclear plants characterized by the idea of "natural safety." It is the most advanced LFR design and the start of the construction is expected to take place in 2020. Fig. 6 provides an overview sketch of the BREST-OD-300 system. BREST-OD-300 (Figs. 6 and 7) is of pool-type design, which incorporates within the pool the reactor core with reflectors and control rods; the lead coolant circulation circuit with steam generators and pumps; equipment for fuel reloading and management; and safety and auxiliary systems. The reactor equipment is arranged in a steel-lined, thermally insulated concrete vault. BREST has a widely spaced fuel lattice with a large coolant flow area, resulting in low pressure losses, enabling the establishment of primary coolant natural circulation for decay heat removal.

It shares with other designs the absence of uranium blankets, replaced by lead reflector with the proper albedo improving power distribution, providing negative void and density coefficients, and ruling out the production of weapons-grade plutonium. The BREST decay heat removal systems are characterized by passive and time-unlimited residual heat removal directly from the lead circuit by natural circulation of air through air-cooled heat exchangers, with the heated air vented to the atmosphere. The fuel type considered for the first core of the BREST fast reactor is nitride of depleted uranium mixed with plutonium, whose composition corresponds to that of irradiated (used) fuel from VVER's following cooling and reprocessing. This mode of operation is characterized by full reproduction of fissile nuclides in the core (Core Breeding Ratio (CBR) ~ 1) with irradiated fuel reprocessing in the closed fuel cycle. Reprocessing is limited to the removal of fission products without separating Pu and minor actinides (MA) from the mix (U-Pu-MA). One of the notable characteristics of the BREST-OD-300 is that a reprocessing plant is co-located with the reactor, eliminating in principle any accident or problem due to fuel transportation. The main characteristics of this unique

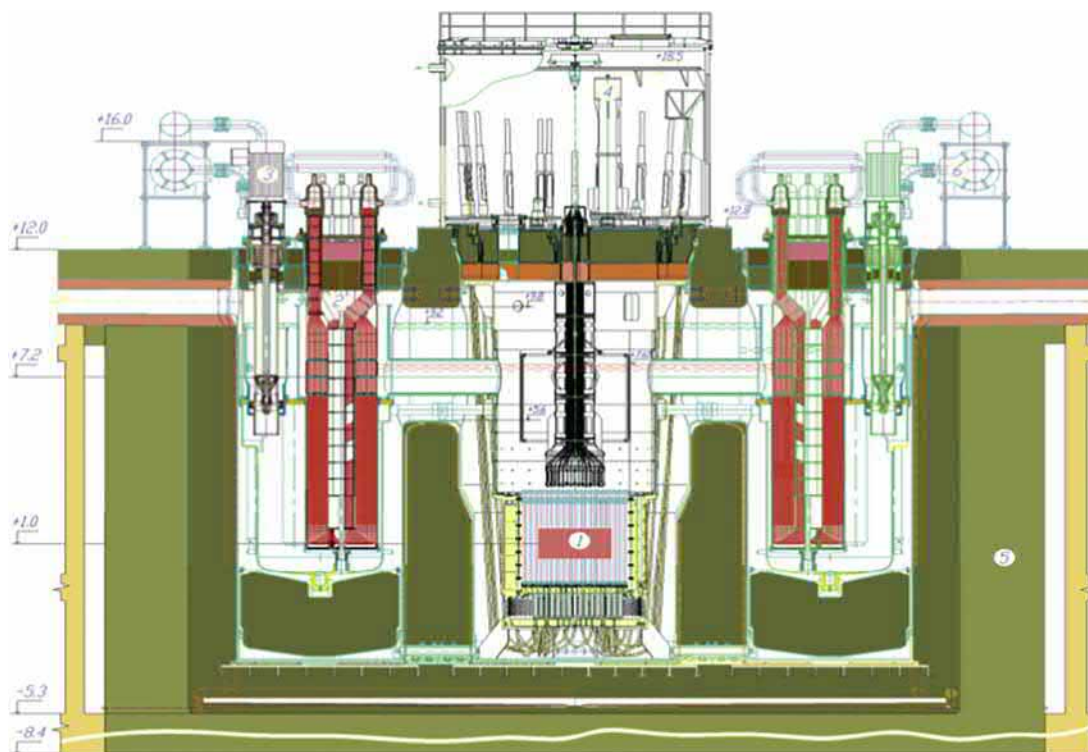


Fig. 6 BREST-OD-300.

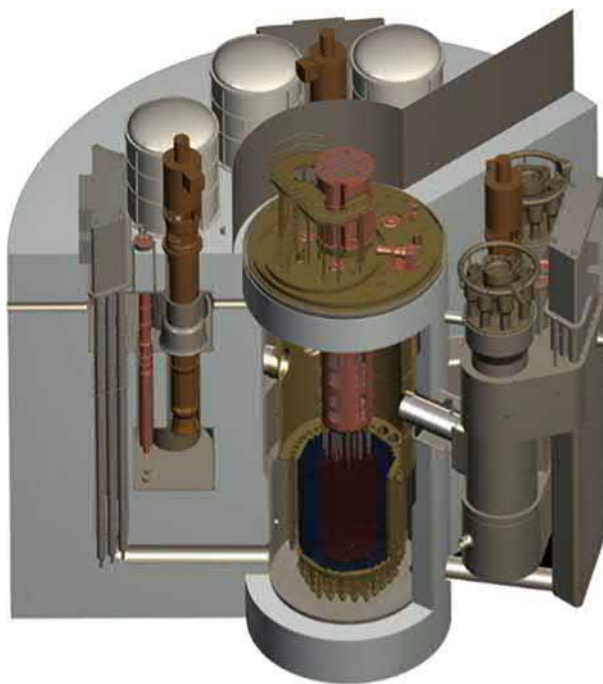


Fig. 7 BREST-OD_300 3D view.

demonstrator are a primary cycle between 420 and 535 °C, a power of 700 MWth, a superheated steam-water cycle reaching 505 °C at 170 bars, allowing for a thermal cycle efficiency of around 42%.

SSTAR

SSTAR is a small modular reactor (SMR) that can supply 20 MWe/45 MWth with a reactor system that can be transported in a shipping cask. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life without refueling; and an innovative supercritical CO₂ (S-CO₂) Brayton cycle power conversion system. Fig. 8 provides an overview sketch of the SSTAR system. The pre-conceptual design of SSTAR represents a small shippable reactor (12 × 3.2 m vessel), with a 15–30-year life open-lattice cassette core and large-diameter (2.5 cm) fuel pins held by spacer grids welded to control rod guide tubes. The main mission of the 20 MWe (45 MWth) SSTAR is to provide incremental energy generation to match the needs of developing nations and remote communities without electrical grid connections, such as those that exist in Alaska or Hawaii, island nations of the Pacific Basin, and elsewhere. Design features of the reference SSTAR, in addition to the lead coolant, 15–30-year cassette core and natural circulation cooling, include autonomous load following without control rod motion, and use of a supercritical CO₂ (S-CO₂) Brayton cycle energy conversion system. The incorporation of inherent thermo-structural feedbacks imparts a high degree of passive safety, while the cartridge core long life imparts strong proliferation resistance. It is presently a legacy system, not under continuing development, but is considered as representative concept, within GIF, of the “small-size” category of LFR developments.

The ELFR

The ELFR system is a modification of the earlier ELSY reactor concept. Fig. 9 provides an overview sketch of the ELFR reactor vessel and system arrangement. The ELFR primary system has a pool-type configuration, with the main vessels hanged by a Y-support holding the main vessel in the upper part. The reactor vessel has been kept as compact as possible, in order to reduce the coolant inventory and the corresponding seismic loads, while being of sufficient size to accommodate the required number of components (i.e., 8 Steam Generators (SGs), 8 Primary Pumps (PPs), and 8 Decay Heat Dip Coolers (DCs)). The hot pool of the ELFR vessel is enclosed by an inner vessel, connected to the PPs through suction pipes. Each PP is installed at the center of its corresponding SG, which transfers the heat from primary lead coolant to water-steam in a superheated cycle. The design is based on a core pressure loss of 0.9 bar and a total primary pressure loss of 1.4 bar. The core inlet and outlet temperatures are 400 and 480 °C. The internal reactor component arrangement and design presents a simple flow path for the primary coolant. The locations of the heat source (within the core) and of the heat sinks (SGs) allow for efficient natural circulation of the coolant under emergency shutdown conditions. Two safety systems for decay heat removal have been considered as an integral part of the design from the beginning of the activities. They are characterized by passive operation, diversity and redundancy while, in addition, being completely independent from one another. The design of the core has been driven by the implementation of the so-called “adiabatic” reactor concept, a closed fuel cycle approach (as above presented) maintaining constant the quantity of plutonium for no proliferation but supporting nuclear fuel sustainability.

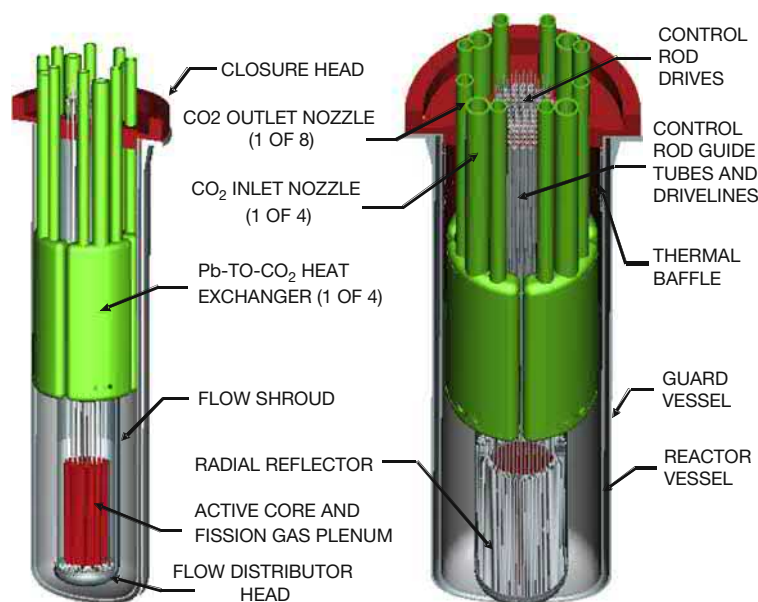


Fig. 8 SSTAR conceptual sketch.

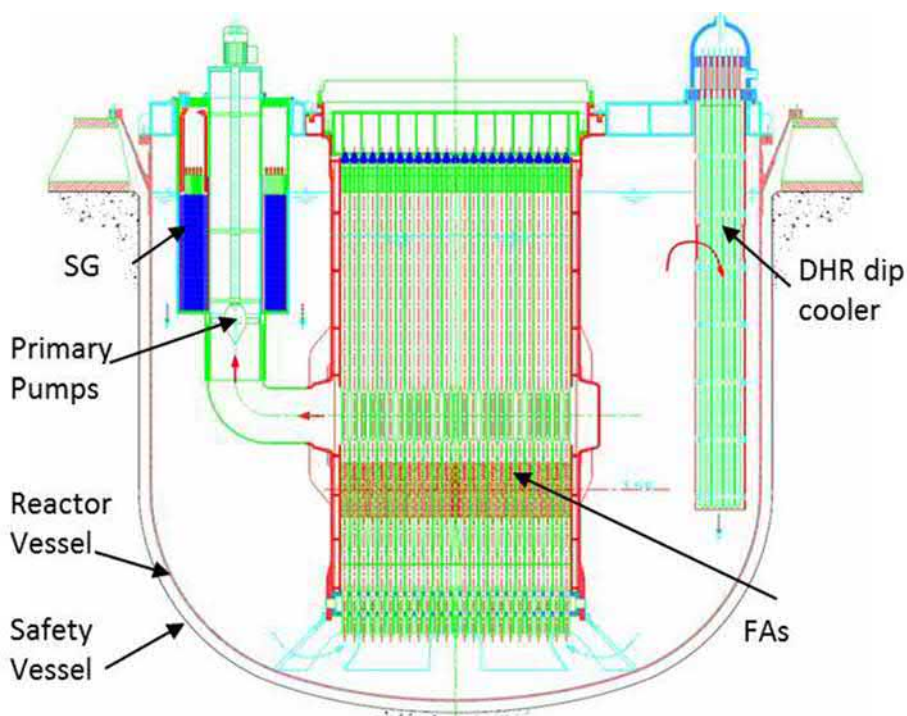


Fig. 9 ELFR concept.

ALFRED

In the description of ALFRED presented above the LEADER project design was used as a reference to explain solutions and differences with other systems. However, the ALFRED design is evolving thanks to the efforts of the so-called FALCON consortium trying to support the construction of the reactor in a relatively short time frame. The new system arrangement is a result of a design review which tried to identify the limitations and issues of the previous configuration. The main modifications can be summarized as follows: elimination of the vessel thermal stratification through a new primary system flow path defined by the use of a “hot” and a “cold” pool, use of single wall bayonet tubes in the SGs thanks to the new flow path design, three SGs, three primary pumps, three additional dip coolers connected to isolation condensers for a completely independent and diversified DHR-2 system, introduction of an anti-freezing system in the DHR concept. The new design has been presented recently in several conferences and journal papers the interested reader is referred to. Figs. 10 and 11 presents the reactor cover top view and the new primary system flow path. Falcon consortium also addressed the path to licensing through an innovative strategy in terms of R&D development, the so-called staged approach that foresees after the construction a progressive increase of operating temperature in order to qualify the structural materials and/or corrosion protection strategy in the lead coolant environment as well as prototypical irradiation conditions.

SVBR-100

The SVBR-100 is a multipurpose small modular fast reactor lead–bismuth eutectic (LBE) cooled with an equivalent electric power of 100 MW. In the Russian Federation, LBE cooled reactor technology has been used in several nuclear submarines (NSs). The experience (more than 80 reactor-years) gained was used in the development of the SVBR-100 reactor. The SVBR technology, according to its basic parameters and outstanding technical characteristics, is claimed as a Generation IV nuclear reactor. The entire primary equipment circuit of SVBR-100 including the reactor core, steam generators, circulation pumps and in-vessel radiation shielding is contained within a robust vessel of the reactor mono-block (see Fig. 12).

A protective casing surrounds the reactor mono-block (RMB). The reactor has negative void reactivity effect and negative temperature reactivity coefficient. Natural circulation of coolant in the reactor heat removal circuits is able to passively cool down the reactor for at least 72 hours preventing core temperature increase. Enhanced inherent self-protection and passive safety level, provided by the very high boiling point and chemical inertness of LBE, significantly simplify the reactor design as well as the entire NPP and improve the economy.

SVBR is a multi-purpose reactor and can be used for regional scale grids, industrial heat and desalination systems with the appropriate power levels (100–600 MW) depending on the specific needs. Reactor can operate with different type of fuel in different fuel

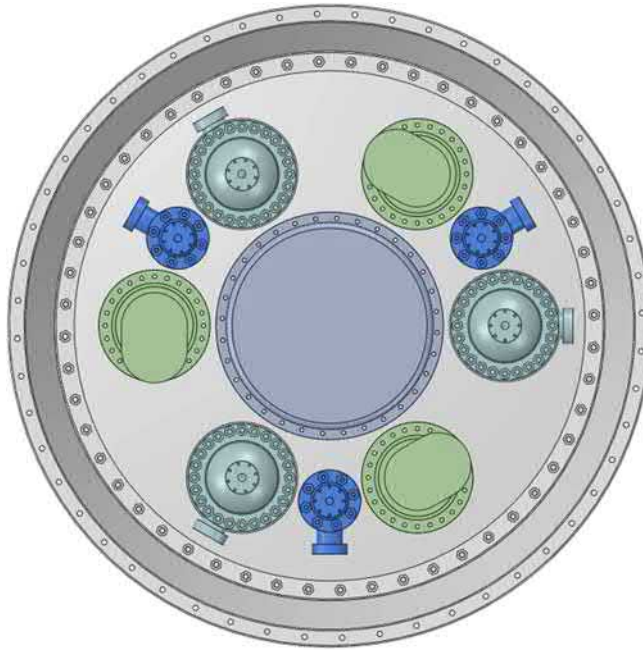


Fig. 10 Reactor cover top-view.

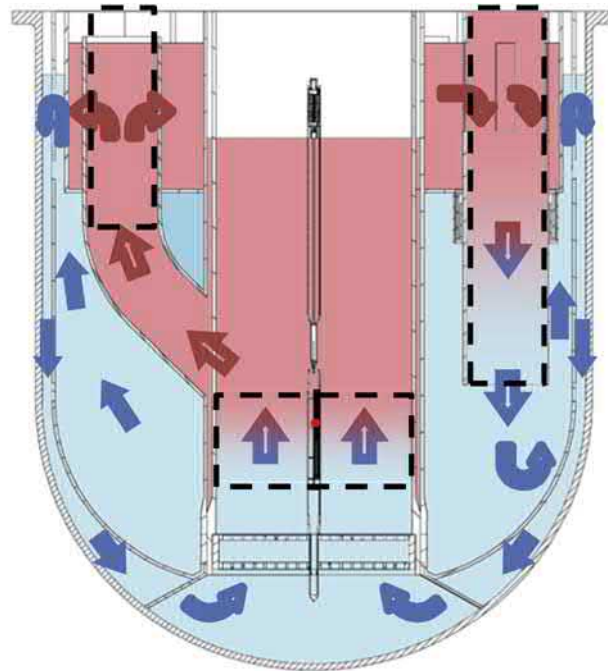


Fig. 11 Alfred primary system flow path.

cycles (period of operation without refueling not less than 7–8 years). At the first stage it will use typical uranium oxide fuel providing a core breeding ratio (CBR) of 0.83. MOX fuel can also be used, leading to a CBR just about 1.

Using UO_2 as the starting fuel, the closed fuel cycle can be realized in 15 years. Compact design and maximum factory readiness of the RMB are provided by its railway transportability and short construction time. The coolant technology system includes heat-exchangers, gas mixture ejectors, sensors of oxygen activity in LBE; its function is to maintain the LBE chemistry, inhibiting structural materials corrosion. The SVBR-100 reactor pursues resistance to nuclear fissile material proliferation by using uranium with enrichment below 20% while using uranium oxide fuel in the core.

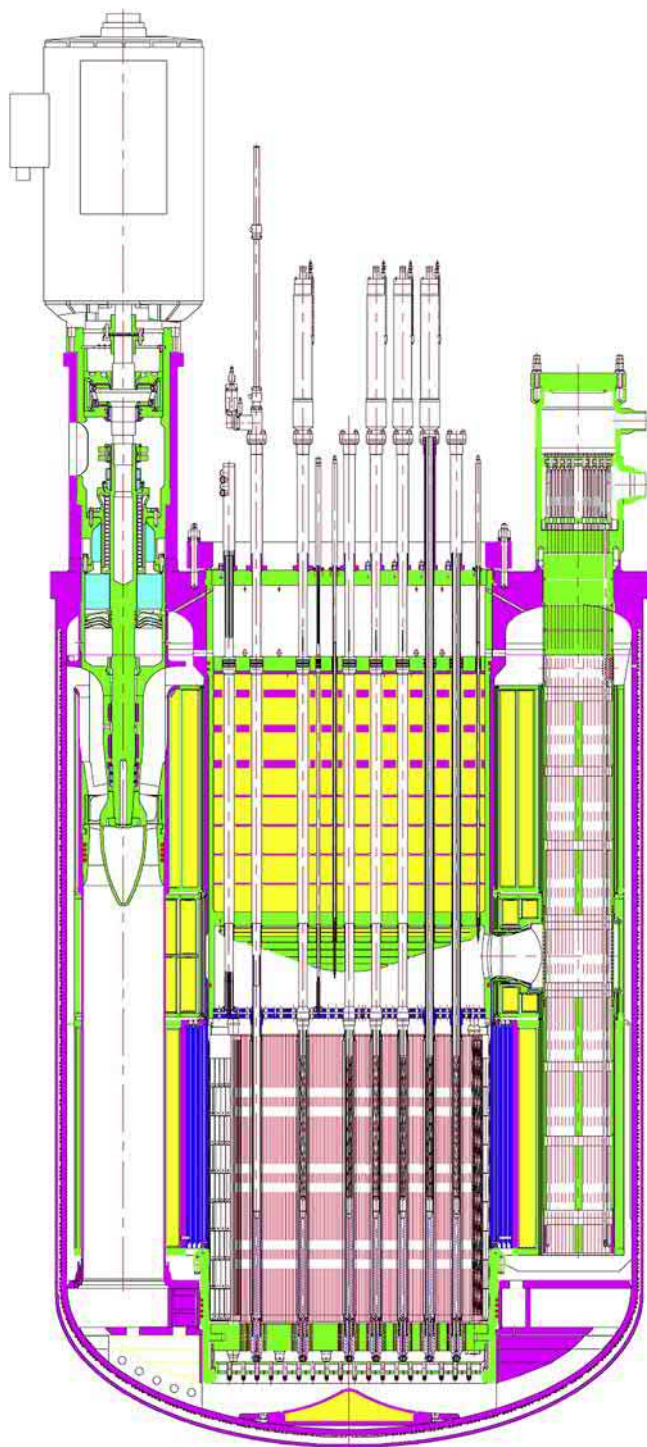


Fig. 12 SVBR-100 reactor mono-block.

LFR-AS-200

The LFR-AS-200 (**Fig. 13**) is an innovative small reactor cooled by molten lead under development by Hydromine Nuclear Energy S.à.r.l., Luxembourg; AS stands for Amphora-Shaped, referring to the shape of the Inner Vessel (ASIV), and 200 represents the electrical power in MWe. The LFR-AS-200 is an integral, pool-type fast reactor with all the primary components installed inside the reactor vessel (RV). The main components of the primary system are six innovative Spiral-Tube Steam Generators (STSG), six mechanical pumps (MP), reversed-flag-type control rods, three + three Dip Coolers belonging to two diversified, passive,

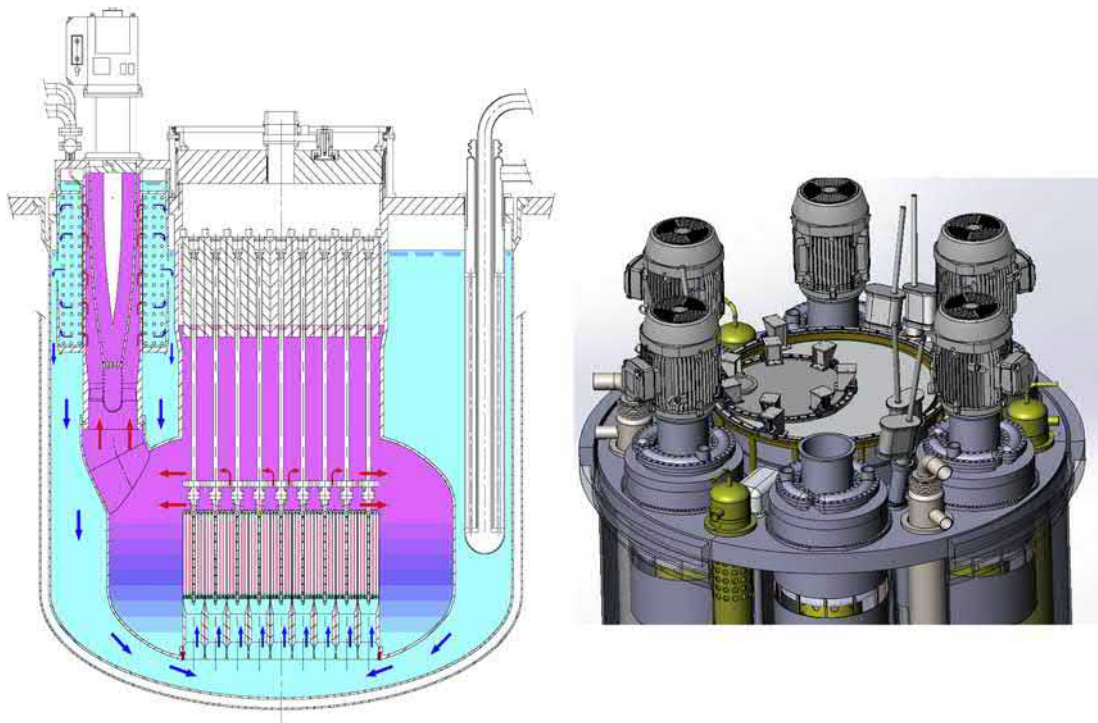


Fig. 13 The LFR-AS-200.

redundant decay heat removal (DHR) systems and the ASIV, that is large in the bottom part in order to dispense of shielding assemblies. Thanks to the properties of lead, intermediate loops are not required. Because the STSGs are located within the RV, several special precautions deterministically eliminate any risk of important primary system pressurization in the case of tube rupture: among them water and steam collectors, located outside the RV, and short STSGs, partially raised above the lead-free level of the cold collector. The Fuel Assemblies (FA) are characterized by stems extending above the lead-free level with heads interacting to form a self-sustaining core. Refueling is performed under visual control and sealed conditions by means of two rotating plugs and an ex-vessel refueling machine. The absence of intermediate loops, the primary system specific volume of less than $1 \text{ m}^3/\text{MWe}$ and the compact reactor building, are key factors for a competitive kWh cost. The elimination of critical components like the in-vessel refueling machine, core strongback, and above-core structures immersed in lead, and hence submitted to thermal transients and fast neutron flux, reduces the need of In-Service Inspection (ISI) and increases the reliability of the plant. The design of the core aims at a system nearly self-sufficient in plutonium. The conversion ratio of about 0.9 is obtained without blankets.

CLFR

The CLFR series reactors (Fig. 14) are developed by the China Nuclear Power Technology Research Institute Co., Ltd. (CNNPTRI) as the next generation of commercial reactors of the China General Nuclear Power Corporation (CGN). The CLFR series include CLFR-100 and CLFR-300, developed to demonstrate the scalability of heavy liquid metal (HLM) cooled reactors. The CLFR-100 is a LBE cooled fast reactor with 100 MWe electric power, designed to demonstrate the technical feasibility of the small modular type HLM reactor with a target for operation by 2030. The CLFR-300 is a lead cooled fast reactor with 300 MWe electric power, designed to demonstrate the economic competitiveness of HLM cooled reactor relative to LWR technology. The CLFR series introduce advanced design ideas like the integral arrangement, modular design, whole core refueling, and smart operation and maintenance. The reactor primary system is an integral pool-type concept with all primary components arranged in the Main Vessel (MV), including the reactor core, steam generators (SGs), reactor coolant pumps (RCPs) and the internal structure. The integral pool-type design can provide safety advantages eliminating main vessel penetrations, reducing the likelihood of loss of coolant accidents (LOCA) and economic benefits due to simplification of the primary system. All primary components are designed to be extractable from the primary system for easy maintenance and replacement, with increased system reliability. A new safety concept named Natural Driven Safety (NDS) has been developed and applied to the design of CLFR reactors. The NDS of CLFR imply that the activation and operation of reactor safety systems are entirely driven by nature laws without the use of any active system (including batteries or electronic devices). Several specific NDS technologies are under investigation in CNNPTRI for CLFR reactors, including the Natural Driven Shutdown System (NDSS), the Natural Driven Decay Heat Removal System (NDDHRS), etc. The NDSS can virtually eliminate risks of unprotected accident while the NDDHRS can virtually eliminate risks of core damage and large release of radioactivity.

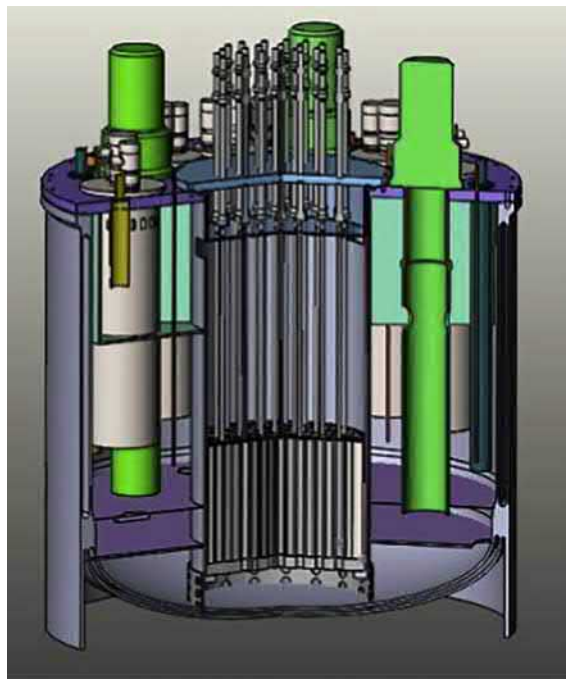


Fig. 14 CLFR sketch.

These excellent safety features can help CLFR reactors to improve both economic performance and nuclear power safety, ruling out the requirement of evacuation of the local population in case of accidents.

BLESS

A R&D project of a Lead-Bismuth Eutectic (LBE) cooled Fast Reactor named BLESS (Breeding Lead-based Economical and Safe System) has been implemented by State Power Investment Corporation Research Institute (SPICRI), the research institute of State Power Investment Corporation (SPIC) in China. In this project, the first reactor design is called BLESS-D (Demonstration) for the purpose of demonstrating the most mature and achievable technology of LFR in the near future. BLESS-D is cooled by LBE with thermal power of 300 MW while the electric power is set at about 100 MW. Fig. 15 provides an overview sketch of reactor primary system configuration of the BLESS-D candidate conceptual design. BLESS-D is a pool-type reactor in which the steam generators (SGs), pumps and reactor core are placed in the reactor vessel filled with LBE. The reactor core is centered and installed in an inner vessel. Two core designs of BLESS-D with UO_2 and MOX ($\text{UO}_2 + \text{PuO}_2$) fuel are proposed. UO_2 fuel is chosen in order to take the advantages of mature fuel-fabrication industry, while the MOX ($\text{UO}_2 + \text{PuO}_2$) fuel is chosen in order to obtain a longer life cycle and to reduce the long-lived high-level radioactive wastes. The core inlet and outlet temperatures are 340 and 490 °C. Eight once-through helical SGs and four pumps are arranged symmetrically between the space of inner vessel and reactor vessel in order to achieve forced circulation. Both water-steam Rankine cycle and supercritical CO_2 (S- CO_2) Brayton cycle energy conversion system are considered and designed for BLESS-D based on the technology capabilities of SPIC. Two types of passive decay heat removal systems with diversity and redundancy have been considered and designed from the R&D experience of the passive safety systems for large passive PWR in SPIC. The main goal of the 100 MWe BLESS-D is to meet the public demands of a safer, more flexible, more economical and more environmental-friendly nuclear energy system, especially for the energy demand from remote residential area and industrial area without national grid connection in China.

Westinghouse LFR

The Westinghouse LFR is a medium-output (~ 460 MWe), passively-safe, modular plant harnessing a lead-cooled, fast spectrum core operating in pool configuration and coupled with an air-cooled Supercritical CO_2 (s CO_2) Balance of Plant (BoP) system. The plant design results from business choices and technology solutions driven by the need to achieve competitive economics, versatility in future and diverse markets and high levels of safety at the same time, with baseload electricity production and load levelling as the primary plant's missions. The plant's output, selected based on economic considerations, is sufficiently small to integrate into lower-capacity grids while also being substantial enough to be used in standard baseload plant applications. Load levelling is achieved through an integrated thermal energy storage system using low-cost materials and coupled to existing BoP equipment, which allows load follow to be performed without requiring the core power to be varied. Moreover, being



Fig. 15 Conceptual design of BLESS-D.

air-cooled, the plant does not require vicinity to large water bodies, thus allowing for a greatly expanded variety of plant siting options. **Figs. 16 and 17** show a schematic of the primary system and of the plant layout, respectively. Reactor vessel compactness is promoted by the use of extremely robust, highly-compact hybrid microchannel-type primary heat exchangers and by the use of a water-assisted reactor vessel cooling system for emergency decay heat removal. This latter system allows removal of in-vessel deep coolers since, in case of accidents, it permits liquid lead natural circulation to be simply driven by passive heat transfer through the reactor vessel and guard vessel to an outer annular water pool. Transition to natural convection air cooling, circulating outside of the guard vessel, occurs when water is depleted. Fuel cycle flexibility is embedded in the plant design, with multiple fueling options and strategies being considered to satisfy market demand and customer preference. Starting with operation at lead temperatures up to $\sim 530^\circ\text{C}$, the plant is designed to transition to higher hot leg operating regimes through implementation of innovative materials

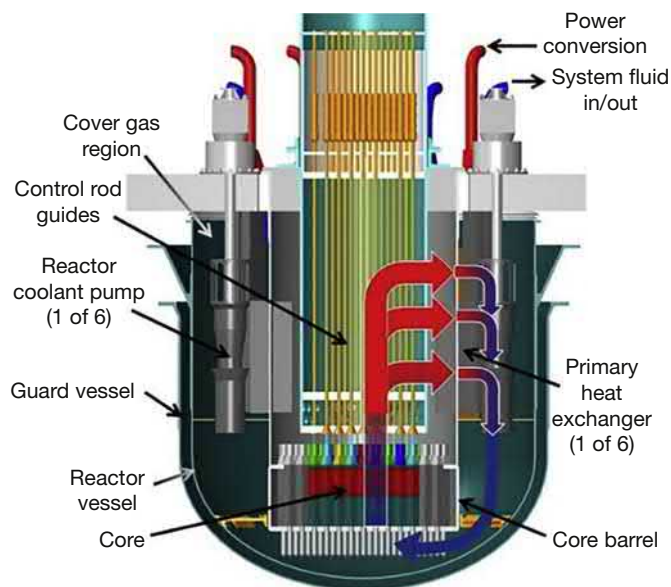


Fig. 16 Westinghouse primary system scheme.



Fig. 17 Westinghouse LFR plant lay-out.

currently being tested, for selected components that can be replaced while keeping the vessel and the majority of BoP components the same thus allowing the transition to occur at the same unit.

CANDLE

CANDLE reactor is a new concept of a lead-cooled fast reactor in which the fission burning wave moves in the axial direction of the reactor core (Fig. 18). Once a burning equilibrium state is reached, it requires only natural uranium or depleted uranium fuel feed. Details are provided by Prof. Sekimoto as part of this Encyclopedia.

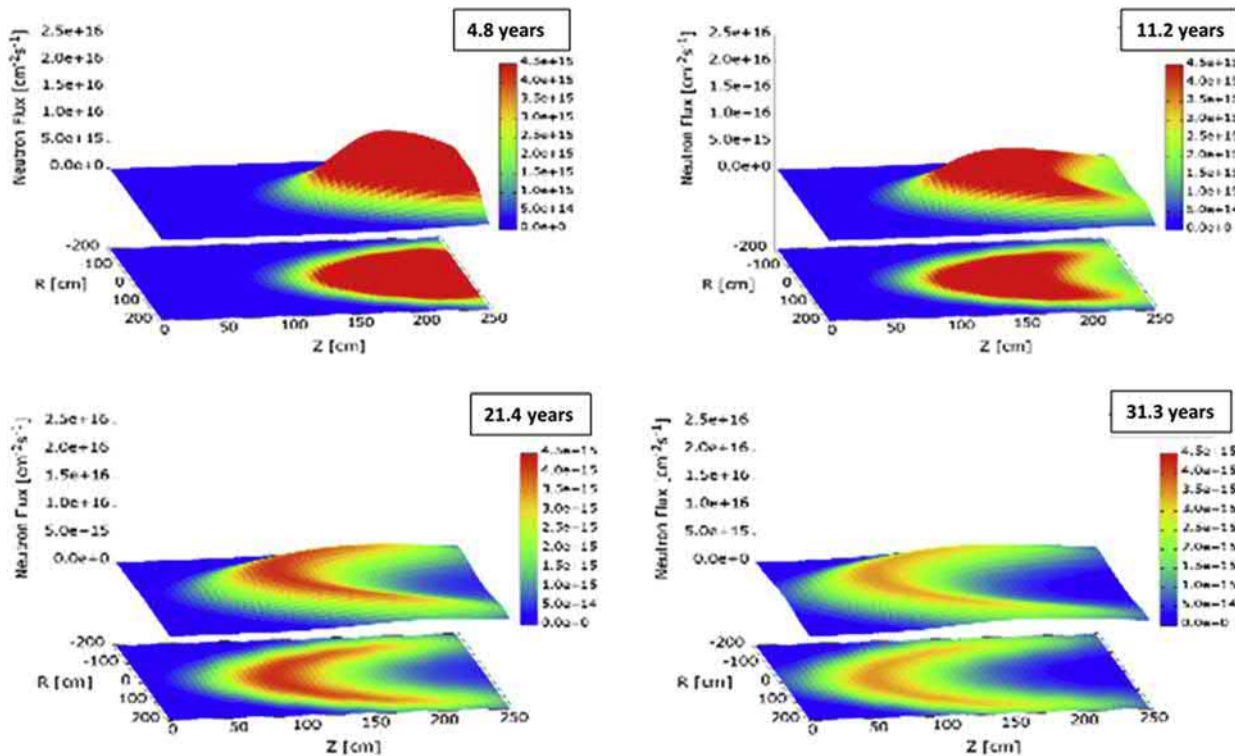


Fig. 18 Neutron flux wave at the start-up of CANDLE reactor.

Lead cooled accelerator driven systems

Accelerator Driven Systems (ADS) consist of a high-power proton accelerator coupled to a subcritical nuclear reactor through a spallation target, generating the neutrons needed to reach criticality. For such systems the “neutron economy” is of great importance and this, together with the fact that the spallation reaction, in most cases, is provided by the impingement of the proton beam on a liquid metal target, makes the Lead or Lead-Bismuth the preferable coolants for this application. The goal and interest around this system, apart from the research interests, resides in its use for waste incineration, given the possibility to add to the fuel an important quantity of Minor Actinides without compromising the safety characteristics of the system thanks to the sub-critical mode of operation, intrinsically providing a “scram” when the proton beam is shut down. For further detailed information please see (IAEA, 2015).

MYRRHA

MYRRHA is the Multi-purpose hYbrid Research Reactor for High-tech Applications developed at SCK•CEN (in Mol, Belgium). It will demonstrate the accelerator driven system (ADS) concept by coupling the accelerator, the spallation target and the subcritical reactor at power levels capable of providing experience feedback which is scalable to an industrial demonstrator. In addition, it will allow the study of efficient transmutation of high-level nuclear waste (Fig. 19). The MYRRHA-facility is conceived as a subcritical flexible irradiation. It will also be able to work in critical mode albeit with different performance characteristics. Both modes of operation have their specific energy and flux distributions which permit a wide range of applications: from fuel development for innovative reactor systems; material development for GEN IV systems and fusion reactors, to radioisotope production for medical and industrial applications. Since MYRRHA is based on heavy liquid metal technology with lead-bismuth eutectic (LBE) as coolant, it can also significantly contribute to the development of lead fast reactor technology and will therefore play the role of the European Technology Pilot Plant in the roadmap for lead fast reactors (LFR). The MYRRHA reactor is a pool-type concept integrating all the primary systems in the reactor vessel. These include a diaphragm separating the hot and cold plenum, four primary heat exchangers, two primary pumps, two in-vessel fuel handling machines, and a core unit. The latter contains, among others, fuel assemblies, control rods, safety rods, the spallation neutron target, instrumentation and experimental devices. The reactor is located in the reactor pit which features a liner able to serve as secondary containment in case of a reactor vessel leakage or break. The reactor cover closes the reactor vessel and supports all the components. Because of structures above the core, the fuel and reflector assemblies have to be loaded from underneath. Thanks to the buoyancy, the fuel and reflector assemblies do not need locking devices to hold them on the core support structure. The cooling system of the reactor can remove up to 110 MWth in active mode. It is also designed to remove the decay heat in a passive way with quadruple redundancy. In case of a common mode failure of the cooling system, the diverse Reactor Vessel Auxiliary Cooling System can, in a passive way, take over the cooling function. The primary system operates at full power, that is, 100 MWth in critical mode or 65 MWth in subcritical mode, at a temperature range from 270 up to 325 °C with a flow rate of 13,800 kg/s LBE. In September 2018 the Belgian Government decided to support the construction in a phased approach of MYRRHA at the site of Mol, Belgium. In the first phase, budget is allocated for the construction of the first part of the accelerator up to 100 MeV including an ISOL (isotope separator on-line) and a proton target facility. This first phase, called

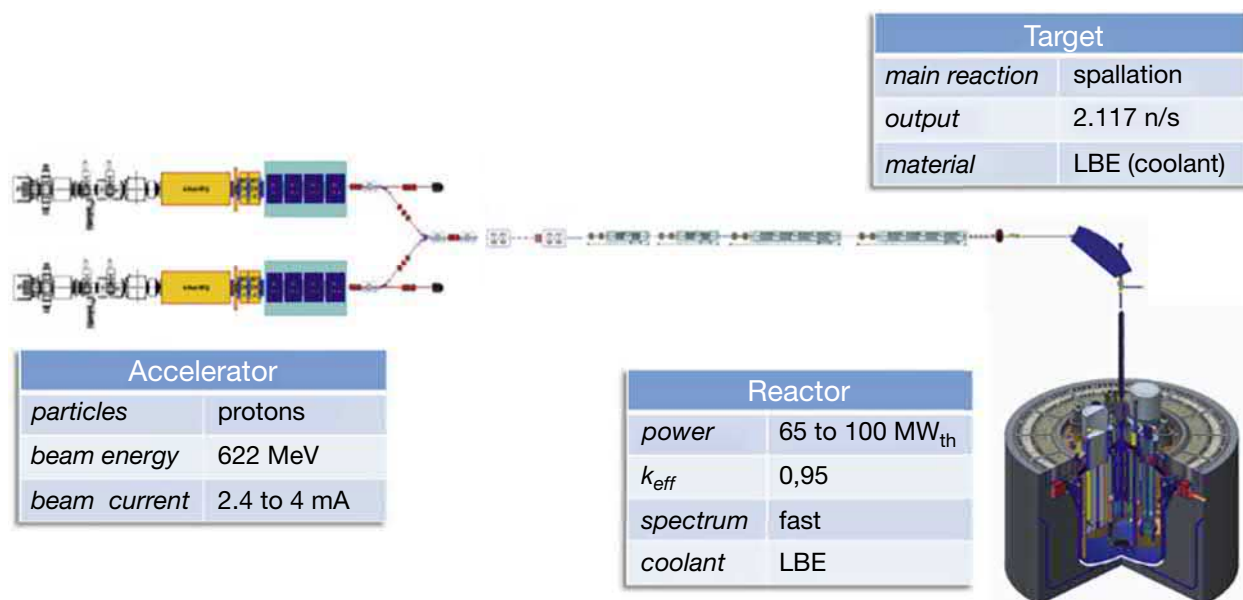


Fig. 19 MYRRHA—an accelerated driven system: coupling of accelerator, spallation target and reactor.

MINERVA, has to be finalized in 2026. In this first phase further design, R&D and licensing for the second part of the accelerator and the reactor is also being performed with the objective to obtain in 2026 the necessary permits for construction.

CLEAR-I

China Lead-based Research Reactor CLEAR-I is a 10 MWth reactor cooled by lead-bismuth eutectic (LBE). It was designed to be operated in a dual way: critical operation mode for technology test of lead cooled fast reactor and subcritical operation mode for ADS operation technology validation. The experimental objectives are expected to be achieved step by step, such as loading different types and amounts of fuel or adjusting the intensity of the proton beam, which will be carried out in the same experimental platform. CLEAR-I reactor is a compact pool type structure, which is composed of reactor vessel, reactor internals, vessel closure head, rotating plug system, reactor core and spallation neutron target, etc. The fuel assembly is composed of 61 fuel rods with triangular arrangement. The UO_2 with ^{235}U enrichment of 19.75% is chosen as the first loading fuel, and the cladding material is austenitic steel. A vacuum beam tube is placed in the center of the core unit which let a 250 MeV/10 mA proton beam enter from the top through the window to the flowing LBE target. The Core inlet and outlet temperature are 300 and 400 °C. The reactor vessel is designed as a double-walled pool-type vessel. The CRDM adopts the methods of counterweight and driving spring to overcome the descending resistance initiated from the high density of the lead-bismuth alloy. Double wall bayonet tubes for four primary heat exchangers are used aiming to reduce the risk of the heat exchanger tube rupture event. Two primary pumps are vertically installed in the cold pool. Reactor vessel air cooling system (RVACS) is designed to remove decay heat passively in accident conditions. Three integrated test facilities have been built at INEST (Institute of Nuclear Energy Safety Technology, HEFEI, China), the lead-based engineering validation reactor CLEAR-S, the lead-based zero power critical/subcritical reactor CLEAR-0 coupled with HINEG neutron generator and the virtual reactor CLEAR-V, to validate and test the key components and integrated operating technology of CLEAR-I. CLEAR-S is a large lead-bismuth pool-type facility with full scale height and 1/2.5 diameter with respect to CLEAR-I. The LBE inventory is larger than 200 t. The total heating power of the seven fuel assemblies simulators is 2.5 MWe. The prototype components test and integrated circulation thermal hydraulic test under steady and transient conditions are carried out to support CLEAR-I project (Fig. 20).

TORIA

TORIA, a ThO_2 -based subcritical burner naturally cooled by LBE has been designed to eliminate TRU produced in LWRs, leaving behind only low and intermediate level wastes. Thoria-based TRU-bearing fuels will be produced from advanced TRU extraction by PyroGreen of waste materials (Korea Atomic Energy Research Institute's pyro-processing). The spent fuels from TORIA can be continuously recycled by PyroGreen process while discharging low and intermediate level waste only. The Toria core consists of 162 thoria-based TRU-bearing fuel assemblies and 102 LBE reflector assemblies in hexagonal lattice. At the core center a spallation target channel about 15 cm in diameter is positioned for 590 MeV proton beam from a superconducting cyclotron with a separate LBE circulation system (Fig. 21). The overall height of the TORIA is 130 cm with a central active core of 70 cm height with keff of 0.99.

CiADS

The China initiative Accelerator Driven System (CiADS) is a transmutation research facility consisting of an accelerator, a spallation target and a sub-critical LBE cooled fast reactor. The initial purposes of this facility include: research on the strategy of the stable,



Fig. 20 CLEAR-I sketch and picture of CLEAR-S facility.

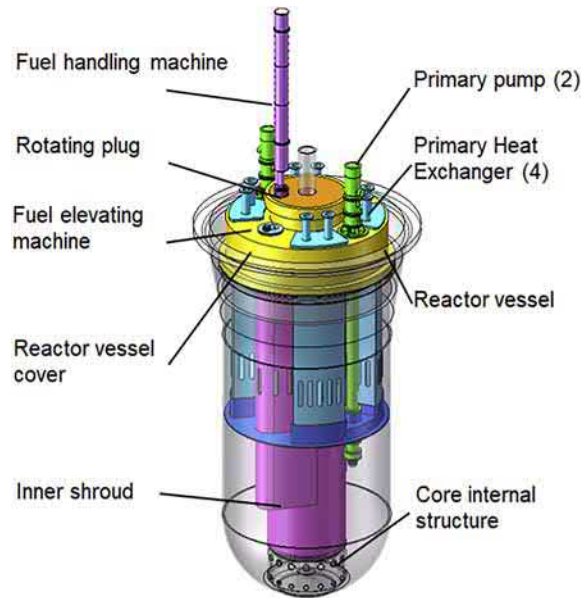


Fig. 21 TORIA conceptual design schematic.

reliable and long-term operation of the superconducting linear accelerator, high-power spallation target and sub-critical reactor; research on the interaction between systems during the coupling operation mode; breakthrough of the coupling technology under the full power operation mode. After that, the device will be operated as a comprehensive experimental platform to carry out the experiments on MAs transmutation and structure materials irradiation. Meanwhile, the special design software for ADS system will be developed in this project to support the development of industrial demonstration ADS facility in the next step after the end of the construction of CiADS. The total thermal power of the sub-critical reactor is 10 MWth (including the spallation target power). The reactor has been chosen as a pool type configuration, as shown in Fig. 22. From the figure one can see that there are two main coolant pumps and four main heat exchangers to remove the core power. The core inlet and outlet temperatures are 280 and 380 °C, respectively, which can effectively reduce the corrosion induced by the LBE. The secondary coolant has been chosen as molten salt so that it can be operated in the atmospheric pressure condition. During the core design phase, much more spaces have been left for the experimental devices installed in the future. The core decay heat can be removed promptly through natural circulation.

A forward look

The main goal of this LFR article was to highlight the benefits achievable through the development of the lead cooled fast reactor concept in terms of Generation IV goals. The interest in the technology is continuously growing worldwide covering a large number

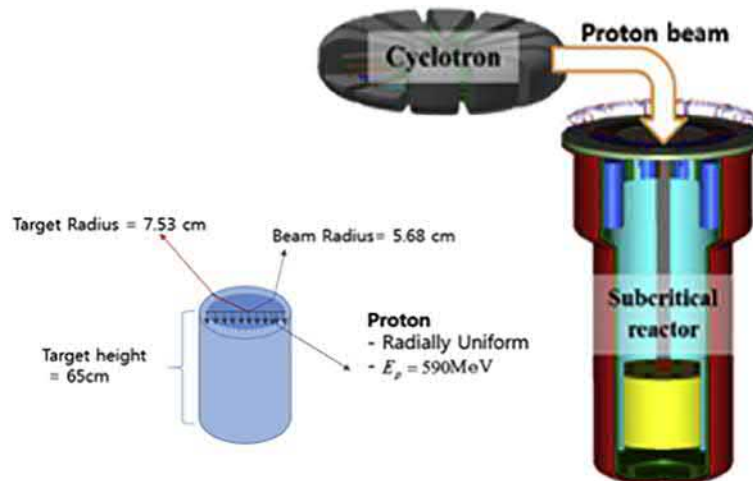


Fig. 22 CiADS sketch.

of potential applications. What is necessary for advancement of this technology is the realization of one or more demonstrators. While this is already a reality in the Russian Federation, with the start of BREST-OD-300 construction activities, we hope that other Countries or the international community will be able to make the same step, bringing the LFR to industrial reality, for the deployment of this so promising technology in the middle of this century.

Acknowledgments

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See Also: Heavy Liquid Metal (HLM) Cooled Fast Small Modular Reactors; Irradiation Performance: Fast Reactor Fuels; Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors.

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Gas Cooled Fast Reactor System (GFR)

Branislav Hatala, VUJE, A.s., Trnava, Slovak Republic

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Glossary

DHR Decay Heat Removal
ESNII European Sustainable Nuclear Industrial Initiative
ETGBR Existing Technology Gas Breeder Reactor
GBR Gas Breeder Reactor
GEN IV Generation Four (nuclear reactors)
GFR Gas-cooled Fast Reactor
GCFR Gas-cooled Fast Reactor
GIF Generation IV International Forum
gHM Gram of Heavy Metal
LMFBR Liquid Metal Cooled Fast Breeder Reactor
MOX Mixed Oxides (fuel type)
SiCf/SiC Silicon carbide fiber-reinforced/silicon carbide
SNETP Sustainable Nuclear Energy Technology Platform
SS Stainless Steel
UOX Uranium Oxide (fuel type)
UPuC Uranium Plutonium Carbide
ZrC Zirconium Carbide
MPa 10^6 Pa
MWe 10^6 watt electrical
MWth 10^6 watt thermal

Introduction

Gas cooled fast reactors (GFR, GCFR) were subject of intensive studies and design activities in the 1960s and 1970s induced by the effort to cope with expected enlargement of nuclear power plant fleet and limited uranium resources. Together with liquid metal cooled reactors, the GFR were the preferred option. Presently, the fast reactors are gaining increasing attention for several reasons - mainly for their potential contribution to sustainable development by using a much larger fraction of the uranium resources; for the high number of excess neutrons available in a fast reactor which allows their application as actinide transmutation reactors, to reduce the long-term radiotoxicity of nuclear waste; and high temperature of the coolant, providing high thermal cycle efficiency and enabling co-generation of hydrogen and other industrial uses.

For fast reactors, the choice of coolant is dictated by the desire to introduce the smallest amount of absorption and moderation, while still being able to reliably remove the heat from the reactor core. Commonly a liquid metal is chosen, but a gas coolant is also possible.

The GFR cooled by helium is proposed as a competitive longer-term alternative to liquid metal cooled fast reactor ([Heidet, 2021](#); [Alemberiti, 2021](#)). It provides several attractive features. Helium is a single phase, chemically inert and transparent coolant. It does not dissociate or become activated, and in spite that the coolant void coefficient is generally positive, it is a comparatively small effect compensated effectively by negative Doppler and other reactivity feedback mechanisms. The high core outlet temperature above 750 °C (typically 800–850 °C) is an added value of GFR technology.

The advantages of a relatively simple system layout in GFRs, a potential of high breeding gain, and higher thermal efficiency are offset by the need of the engineered safety precautions for depressurization. In practice the temperature limits on conventional cladding material (stainless steel) do not allow operation at much higher temperature than in a typical sodium-cooled fast reactor. No GFRs were ever constructed, and the “classic” GFR concept was abandoned in the late 1970s.

During the next decades several conditions changed and GFR (as well as some other reactor concepts) are subject of restored attention and conceptual studies, reflecting both changes in energy market and advances in technologies to deal with gases and to produce refractory construction materials. The GIF Technology Roadmap (GIF, 2014) identified the GFR as a technology that associates the advantages of fast spectrum systems with industry needs.

Within the Generation IV framework the GFR targets sustainability, that is, optimal use of resources while maintaining good safety and economic performance. The reference design of the Generation IV GFR features the following (van Rooijen, 2009):

- Fast reactor core without fertile blankets, that is, all new fissile fuel is bred in the core. The background is that in a fertile blanket, the fissile element (e.g., Pu-239) is isotopically almost pure, which poses a proliferation concern.
- Breeding enough fissile material to refuel the same reactor, recycling all heavy metal, adding only fertile material; only fission products and reprocessing losses are discharged to a repository.
- The fuel specific power is comparatively low, typically around 40 W/gHM. By allowing a low specific power, the volumetric power density in the core remains limited, typically between 50 and 100 MW/m³, which ensures manageable safety characteristics.
- Helium is chosen as the reference coolant.
- The design of the core and fuel elements aims at promoting passive decay heat removal and providing adequate margins to core melt, by using refractory (high melting point) materials, and by allowing a large coolant fraction in the core.

The reference GEN IV concept for GFR is a 2400 MW_{th} plant operating with a core outlet temperature of 850 °C enabling an indirect combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high power density necessary for good neutron economy in a fast reactor core. This represents the biggest challenge in the development of the GFR system. The second significant challenge for GFR is ensuring decay heat removal in all anticipated operational and fault conditions.

A necessary step in the development of a commercial GFR is development of an experimental demonstration reactor, to allow qualification of the refractory fuel elements and full-scale demonstration of the GFR-specific safety systems. The proposed demonstration reactor for the reference GIF GFR concept shall be the ALLEGRO a low power GFR facility (50–100 MW_{th}) with the ability to operate in different core configurations starting from a “conventional” core featuring steel-cladded MOX fueled pins, followed by all-ceramic fuel elements in the latter stages of operation.

History of GFR

In the following, there is a very brief overview of GFR projects worldwide. It is intended not to be exhaustive but only to illustrate those concepts which were subject of more intensive research and development in the past.

In 1962, General Atomics (USA) designed a prototype reactor with a rated power of 300 MW_e as well as a 1000 MW_e industrial-scale gas cooled fast reactor (GCFR) (IRSN, 2014). The detailed design and certification program for a 300 MW_e helium cooled high temperature prototype reactor began in 1965–66. This GCFR was designed to use fuel consisting of uranium and plutonium, with austenitic stainless-steel cladding. Helium entered the reactor core at a temperature of 385 °C and exited at 550 °C. The helium coolant pressure was 8.5 MPa, and the primary circuit was contained within a reinforced concrete shell. The core of the GCFR was very similar to those of sodium-cooled reactors, apart from the addition of equipment to distribute the helium supply to the base of the fuel assemblies and the adoption of ridged cladding to enhance the heat exchange with the helium. Development work for the GCFR continued until 1981, when development of fast reactors was abandoned in the United States.

In Europe (van Rooijen, 2009) a number of players in the nuclear field joined forces to establish the Gas Breeder Reactor Association for development of a gas-cooled fast reactor. This group proposed a first design (GBR-1) in 1970, a 1000 MW_e reactor featuring helium coolant, pin-type fuel, conventional outlet temperature, and a secondary steam cycle. This design was followed by GBR-2 and GBR-3 (1971), also 1000 MW_e reactors but using coated particle fuel, slightly elevated outlet temperature, and helium coolant for GBR-2; CO₂ coolant for GBR-3 (Chermaine (1972)). The design finally evolved into the GBR-4, a 1200 MW_e reactor with helium cooling and pin-type fuel.

The GBR-2 design is interesting because it resurfaces in modern design proposals for GCFRs, for instance, in Japan (Konomura et al., 2003). The objective of the coated particle fuels was to increase the core outlet temperature to improve the thermodynamic efficiency of the secondary steam cycle. For both GBR-2 and GBR-3 coated particles were only used for the driver fuel, the blankets employed traditional pin type fuel. This solution was chosen because, at the time of design, reprocessing of coated particle fuel was not proven. GBR-2 and GBR-3 required several ceramic parts, most notably the structures at the outlet side. The fabrication difficulties related to large ceramic parts led to the development of GBR-4, which is a much more conventional design. In GBR-4, the outlet temperatures are decreased, enabling the use of stainless steel components throughout the core. The plant efficiency is lower, which is offset by a larger total output of the reactor, elevated from 1000 to 1200 MW_e. A last reference to the GBR-4 design was found in Chermaine and Burgsmuller, 1980), where the safety case for large GCFR cores is discussed.

In the late 1970s, a UK program (van Rooijen, 2009) was initiated into an “Existing Technology Gas Breeder Reactor” (ETGBR). This design focused on joining the experience gained in the UK on sodium systems (Dounreay) and the thermal CO₂ - cooled advanced gas cooled reactors. The fuel assemblies used stainless steel cladding with surface roughening, while the entire system was to be housed in a concrete vessel as used for the advanced gas cooled reactors. ETGBR also used CO₂ coolant, and had a lower power density than Liquid Metal Cooled Fast Breeder Reactors (LMFBRs) (Kemish et al., 1982). The ETGBR was not very different from other designs of the same era for GCFRs. However, the ETGBR idea lingered on for a long time well into the late 1990s. At that late stage, the ETGBR was rebranded as the Enhanced Gas-Cooled Reactor which was proposed as an actinide burner, first within the European Fast Reactor program, and later in the CAPRA/CADRA study (Sunderland et al., 1999). By then, the reactor featured 3600 MW_{th}, CO₂ cooling, and nitride fuel in fuel pins.

In Japan (van Rooijen, 2009) a fast reactor program was initiated in the 1960s, including sodium and gas-cooled reactor concepts. Kawasaki Heavy Industries investigated GCFR concepts cooled with steam, CO₂ and helium (Mochizuki et al. (1972)). The helium concept was based on LMFBR technology, but opted for a very low core, to reduce the pumping power requirements. The flat core also increases breeding gain but requires a larger fissile fraction. Investigations into the GCFR concept seem to have continued without interruption in Japan, culminating in the late 1990s in a GCFR design proposal (Konomura et al., 2003).

GEN IV GFR design options

Besides the generic features of fast neutron spectrum reactors, which are common to all types of fast reactors, the GFRs provide some specifics resulting from light gas coolant. Helium (or some other gasses) complies with the requirement of the smallest amount of neutron absorption and moderation, while still being able to reliably remove the heat from the high power density configuration of the core. In comparison with sodium or other liquid metal coolants, gas coolants have the following advantages for fast reactor applications:

- Chemical compatibility with water and generally good chemical compatibility with structural materials
- No restriction related to operation temperature
- Negligible activation of the circulating medium in the core
- Optical transparency (which simplifies fuel reload operations and inspection)
- No pockets of coolant in cooled down fuel assemblies (compared to liquid metal reactors)
- No phase change in the core, reducing the potential of reactivity instabilities, particularly during transients and accidents
- Reduced (but still present) positive void feedback coefficient
- Potential for a hard neutron spectrum, which increases the breeding potential of the reactor
- Possibility to design core with increased neutron leakage, adding design flexibility in terms of improved breeding gain

Nevertheless, the gas coolant brings several challenges and potential disadvantages. They are mainly the following:

- More demanding means to keep coolant circulating in the system
- Need to maintain high pressure in the system, with associated challenges to prevent leakage of the coolant out of the system
- Low heat transfer coefficient from fuel to coolant, with impact on pumping power, gas velocities, requirements to peak fuel and rod clad temperatures, as well as higher peak-to-average fuel and clad axial temperature distributions (due to the reduced volumetric heat capacity of the coolant)
- High coolant flow velocity, which can lead to significant vibrations of the fuel pins
- Safe removal of decay heat from the high-power density core may be difficult

There are several additional specifics and limitations relevant to GFR, coming e.g. from restrictions placed on plutonium content, relatively very low thermal inertia of the core, capability of burning fission products and others. This is reflected more or less in present design options of individual components of the GFR.

Here below, the reference GEN IV GFR concept is described, based on a project developed by CEA until 2009 (RGN, 2014), with the design power 2400 MW_{th} (Fig. 1).

The fuel consists of pellets of mixed uranium and plutonium carbide ((U,Pu)C). The fuel element of the GFR concept is based on a refractory metal lined silicon carbide fiber reinforced (SiCf) clad pin. The pins contain a plenum in which fission gas can collect for the whole duration of fuel operation and other features to prevent pellet-clad interaction and reduction of the fuel cladding swelling. Fuel element design is based on refractory and high thermal conductivity materials, with the ability to ensure radioactive material confinement up to very high temperatures.

The core layout of the reference GFR concept has been chosen to be consistent with the maximum power derived from thermal-mechanical and thermal-hydraulic analyses, the requirements of the reactivity control system and the optimized power distribution. It shall allow fast neutron spectrum with a zero (self-breeding) or positive breeding gain, with no or very limited use of fertile blankets (due to proliferation concerns), to generate as much fissile material as it consumes, with an optimal use of uranium, to have a fuel cycle fed with only depleted or natural uranium and to achieve homogeneous recycling of all actinides, in order to allow operation without separation of plutonium from other actinides (again due to proliferation resistance considerations). Core plutonium inventory shall not exceed 10 tons/GW_e, in order to have a realistic reactor fleet deployment (in a few decades) and high fuel burn-up.

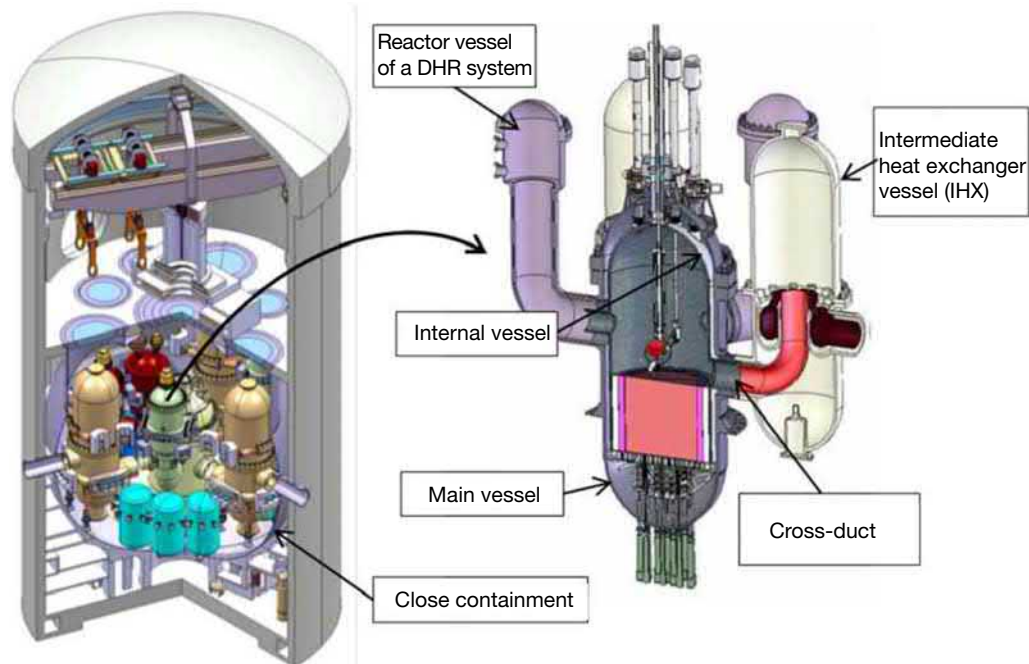


Fig. 1 Reactor building and vertical cross-section of the GFR reactor vessel (IRSN, 2014).

The primary system of the reference GFR concept is relatively simple. The cylindrical vessel with upper and lower spherical parts contains the core and internal cylindrical shell which separates the cold zone (volume in the upper part, downcomer and lower plenum) from the hot zone (upper plenum). Directly interconnected there are one to three Power Conversion Systems with main heat exchangers and blowers and several Decay Heat Removal (DHR) loops. Additionally, the reference design features several gas tanks, connected to the primary circuit. Specific loops for DHR are equipped with heat exchangers and potentially with forced convection devices.

The whole primary system is enclosed in an air tight envelope, a pressure guard containment, which shall limit the consequence of a second safety barriers rupture (of the primary system), but potentially also damage of the first barrier, the fuel clad. The guard containment is designed to provide and maintain a backup pressure in case of large gas leak from the primary system. It shall be a metallic structure, initially filled with e.g. nitrogen slightly above the atmospheric pressure to reduce air ingress capabilities.

Power conversion system utilizes advantages, provided by the high temperature gas in the primary circuit. The choice could be an indirect combined cycle with Helium or Helium-Nitrogen mixture for the intermediate gas circuit. The produced heat will be converted into electricity in the direct or indirect combined cycle, using gas turbines or steam turbine. The cycle efficiency can reach up to around 50%, or even higher if counting with cogeneration or utilization of the excess heat to appropriate technological purposes.

Concerning the objectives of ultimate waste minimization, proliferation resistance and natural resources optimization (zero or positive breeding gain), the major corresponding reactor design options are:

- No fertile blanket and multi-pass recycling of all actinides without separation
- Loading of 1.1% of minor actinides (corresponding to self-recycling)
- A high density fuel with maximization of actinide content
- High core power density of about 100 MW/m³
- Mean overall core Burn-Up: 5% FIMA (fission per initial metal atom) or more, depending on fuel cycle orientation and optimization.

These high level objectives imply various additional secondary specifications such as minimization of the reactivity swings per cycle or minimization of the core pressure drops.

In parallel, the safety architecture was thought to cover the potential defects fitted to this system, due to the following elements:

- Fuel elements that uses refractory materials and withstands very high temperatures
- Gas voiding reactivity effect in the core is not significant, but still positive hence its effect is not negligible
- Decay heat can be removed in any accidental situations (pressurized or not, even in case of large primary break, including an additional single failure or multiple failures), due to different systems of moderate power supply and to a gas tight guard containment
- Natural convection capabilities in DHR can be retained in most of the situations

Nevertheless, several technical fields are only partially covered today, or they are still in need of optimization. One of these critical fields is a full scope and reliable decay heat removal system.

ALLEGRO, as a demonstrator of the GFR technology, represents a necessary step on the path to the deployment of the GFR technology and realization of its high-potential in terms of industrial heat utilization, more efficient use of fuel, and highly competitive costs of both electricity and industrial heat.

Allegro demonstrator development status and R&D major challenges

The ALLEGRO demonstrator is an essential step to establish confidence in the innovative Gas Cooled Fast Reactor (GFR) technology. The development of ALLEGRO is supported by the European Sustainable Nuclear Energy Technology Platform (SNETP). The European Sustainable Nuclear Industrial Initiative (ESNII) as a one pillar of SNETP addresses the need for demonstration of Generation IV Fast Neutron Reactor technologies.

The proposed demonstrator would be the first ever gas cooled fast reactor to be constructed. The ALLEGRO was originally designed by the French Alternative Energies and Atomic Energy Commission (CEA) (Poette, 2009; Poette and Stainby, 2010; Poette and Morin, 2012), among others under EU financed GOFATR project and it is successor of ETDR 50 MW concept (Poette et al., 2007).

The objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as fuel, fuel elements, helium-related technologies and specific safety systems, in particular, the decay heat removal function, together with demonstration that these features can be integrated successfully into a representative system. The ALLEGRO reactor would function not only as a demonstration reactor hosting GFR technological experiments, but also as a test pad of using the high temperature coolant of the reactor in a heat exchanger for generating process heat for industrial applications and a research facility which, thanks to the fast neutron spectrum, makes it attractive for fuel and material development and testing of some special devices or other research works.

The original design of the ALLEGRO consists of two He primary loops, three decay heat removal (DHR) loops integrated in a pressurized cylindrical guard vessel (Figs. 2 and 3). The two secondary gas circuits are connected to gas-air heat exchangers.

The 75 MW_{th} reactor shall be operated with two different cores (Fig. 4). The starting core with UOX or MOX fuel in stainless steel cladding will serve as a driving core for six experimental fuel assemblies containing the advanced carbide (ceramic) fuel. The second core will consist solely of the ceramic fuel and will enable to operate ALLEGRO at the high target temperature. Reflecting the

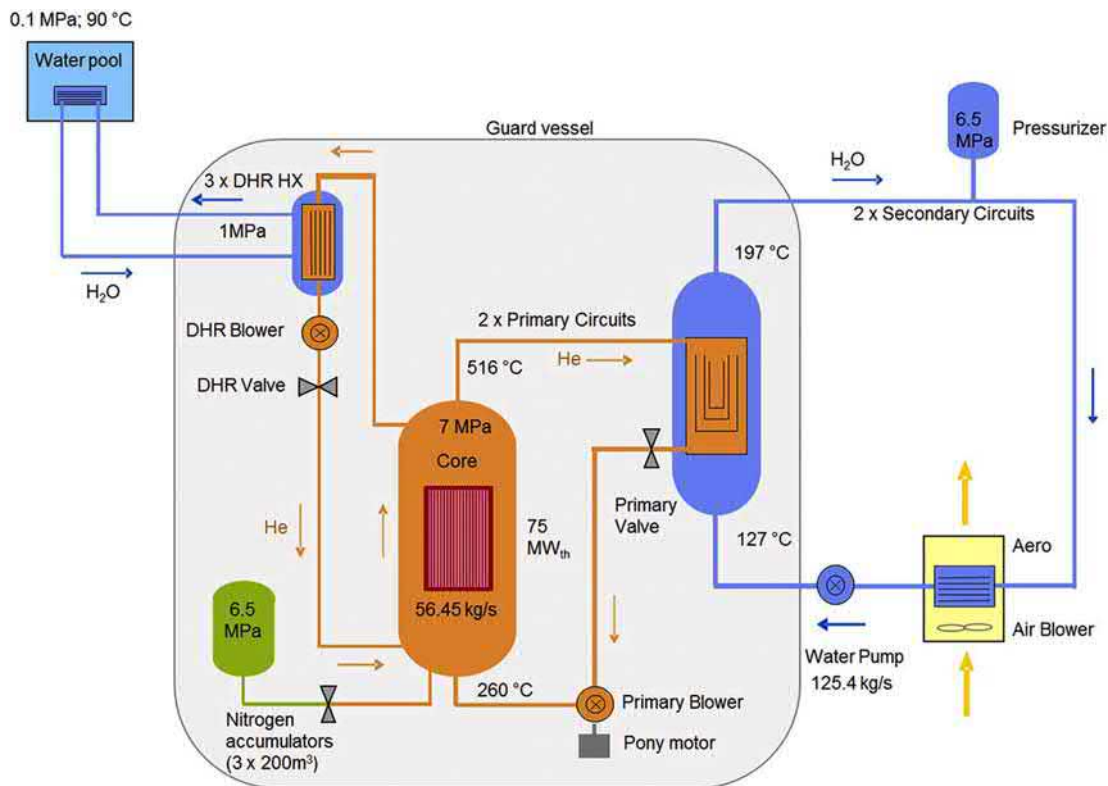


Fig. 2 ALLEGRO Systems.

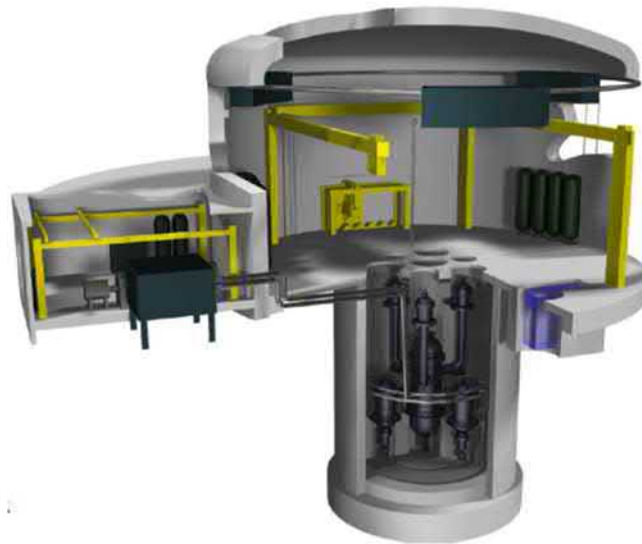


Fig. 3 Quasi-isometric cross view of the ALLEGRO nuclear plant.

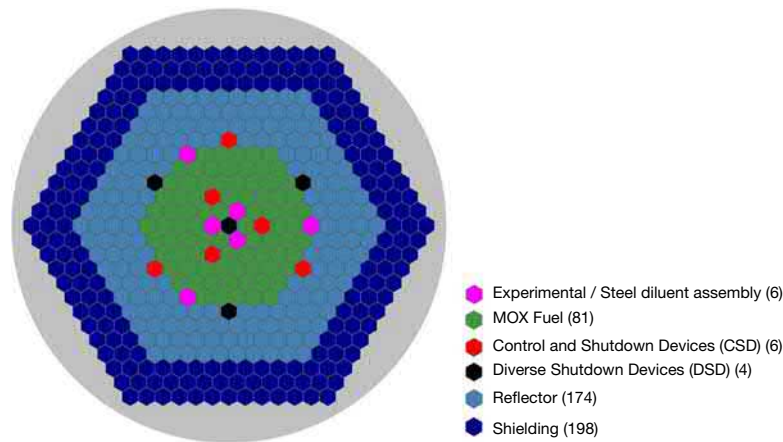


Fig. 4 ALLEGRO reactor core layout.

demonstrator character of the ALLEGRO, several experimental positions are foreseen. If experimental assemblies are not used, the positions are filled with “steel diluent assemblies” to balance core mixture of steel and helium (Table 1).

Fuel and core design issues

Fuel development to satisfy the needs of a GFR is one of the basic goals of the ALLEGRO project. As the target temperature of the coolant is very high, there is a need to utilize refractory fuel. The ALLEGRO cannot use this type of fuel from the very beginning since this fuel is not developed and cannot be qualified/licensed without irradiations in GFR conditions and without subsequent post irradiation examinations. Therefore, according to previous considerations the first cores will be built up from stainless steel clad fast reactor oxide fuel. Some core positions will be reserved for the development of the refractory fuels through the irradiation of fragments, rods and sub-assemblies. In these positions an elevated helium outlet temperature (800–850 °C) is created by reducing the coolant flow rate.

In relation to the MOX fuel utilization, a serious problem may emerge if ALLEGRO (or any other Generation IV reactor) is built e.g. in eastern Europe EU countries. This is because qualification and use of MOX fuel involves several legal and proliferation issues. In order to overcome these potential difficulties, now it is investigated whether using low enriched UOX pellets can be feasible.

In the existing design of the ALLEGRO core, safety and control rods of identical type are grouped into two independent groups of absorbers. In order to increase the safety of the core design a completely diverse types of absorbers have to be developed which would be activated purely by physical principles in a completely passive manner. It can be for instance system of flow levitated absorbers initiated by changes in coolant flow rate or of curie point latches, initiated by temperature changes.

Table 1 ALLEGRO main design characteristics.

<i>ALLEGRO main design characteristics</i>		
Nominal Power - Thermal	75 MW _{th}	Reduced power is being considered in the range 30–75 MW
Nominal Power - Electrical	0 MW _e	
Fuel / Cladding	UPuC/SiC-SiCf	Reduced power density is being considered in the range 50–75 MW/m ³ Start-up core. Feasibility of low enriched UOX for the start-up core is investigated Long term core
Power density	100 MW/m ³	
Fuel element	MOX/SS cladding	
	UPuC/SiCSiCf cladding	
Number of primary loops	2	
Primary coolant	Helium	
Primary pressure	7 MPa	
Number of secondary loops	2	
Secondary coolant	Water	
Secondary pressure	6.5 MPa	

Decay heat removal issues

Decay heat removal in accident conditions is one of the crucial challenges in design of a GFR reactor because of low thermal inertia of the reactor system and relatively high volumetric power density, one order of magnitude higher in comparison with Very-High-Temperature Reactor. The situation in ALLEGRO is further complicated by the wide use of stainless steel in the driver (start-up) core, because e.g. the 15-15Ti stainless steel claddings start to melt already at ~ 1320 °C. The acceptance criterion (temperature) for fuel rod failure is even lower.

Specific loops for decay heat removal in case of emergency are directly connected to the primary circuit using a cross duct piping, in extension of the pressure vessel, and are equipped with heat exchangers and forced convection devices. In addition, thanks to the low pressure drop of the core design, a passive gas natural circulation can be used in most of the situations.

The original concept of decay heat removal in the 75 MW_{th} ALLEGRO CEA 2009 (~ 100 MWt/m³ power density) was mainly based on active elements in the decay heat removal system, and its malfunction in some cases can easily result in core melting. As the increased use of passive features in safety systems is a pre-requisite of licensing a Generation IV reactor after the Fukushima accident (IRSN, 2014), a new concept of combined active-passive DHR system has to be developed. This is why there are several potential elements of the new DHR system under development.

Allegro research and technology development needs

Coordinated effort is ongoing in EU to restore the ALLEGRO concept and to advance its design. Starting from a reference design studied at CEA prior to 2009, the project explored a new target of nominal power (in the range of 30–75 MW_{th}) and power density (in the range 50–100 MW/m³) compatible with the safety limits and the design requirements. At the same time, the feasibility of a low enriched UOX start-up core as alternative to a standard MOX core is being considered. This start-up core, to be used in the first period of the reactor operation, will include experimental positions dedicated to the refractory fuel development. The development of an acceptable fuel system that meets the target criteria (1000 °C normal operation cladding temperature, no fission product release at 1600 °C cladding temperature during a few hours, and maintaining the core-cooling capability up to 2000 °C clad temperature) is a key viability issue for the GFR system. It is necessary to develop an initial cladding material that meets the core specifications, fabrication capacities and material characterization under full scope of conditions.

The GFR also requires a specific dense fuel element that can withstand very high temperature transients, due to the lack of thermal inertia of the system. UPuC pins fabrication methods, as well as their behavior under irradiation must be studied. Ceramic or refractory metal clad should be selected, developed and qualified. Such a program requires material properties measurements, selection of different materials, their arrangement and their interaction, out-of pile and in-pile tests up to qualification, as well as demonstration tests.

The specific operating conditions of the ALLEGRO oxide fuel pins (i.e. maximum fuel temperature close to 1000 °C, linear pin power below 100 W/cm) are not covered by fuel pin behavior codes. The results of such codes appeared to be very sensitive to fission gas release predictions. In this context, a program of post-irradiation examinations on selected pins of the in PHENIX Sodium Fast Reactor irradiated CPed6106 standard fuel subassembly is suitable for obtaining experimental data on fission gas release for operating conditions similar to those foreseen for the ALLEGRO oxide core fuel pins. Alternatively, a specific irradiation program in other potentially available fast reactors could be envisaged.

In addition to the qualification of the oxide fuel, the reference GFR fuel (i.e. carbide fuel with composite SiC and fiber reinforced SiC clad) development efforts must continue. In particular, it is necessary to develop a ceramic clad that meets the specifications in terms of length, diameter, surface roughness, apparent ductility, level of leak tightness (including the potential need of a metallic

liner on the clad), compatibility with polluted helium, and with the irradiation conditions. The needs include fabrication capacities and material characterization under normal and accidental conditions for fresh and irradiated fuel.

In the area of neutronics, existing calculation tools and nuclear data libraries have to be validated for gas-cooled fast reactor designs. The wide range of validation studies on sodium-cooled fast reactors must be complemented by specific experiments that incorporate the unique aspects of gas-cooled designs: slightly different spectral conditions, innovative materials and various ceramic materials (UC, PuC, SiC, ZrC, Zr₃Si₂). In addition some unique abnormal conditions (e.g. depressurization, steam ingress if relevant, etc.) must be considered.

Given the high temperature environment of the ALLEGRO ceramic core, the design margins considered in terms of material characteristics, as well as in the applied thermal hydraulics correlations must be as low as possible. Therefore, air, followed by helium tests on subassembly mock-ups under representative temperature and pressure conditions are necessary to assess the heat transfer and pressure drop uncertainties for the specific GFR design.

Moreover, large-scale air and helium tests to demonstrate the passive decay heat removal function will be required for the licensing process of ALLEGRO.

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Molten Salt Reactors

Victor V. Ignatiev, NRC-Kurchatov Institute, Moscow, Russian Federation

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Glossary

ARE 2.5 MWt Aircraft Reactor Experiment, ORNL, USA.
DMSR 2250 MWt Denaturated Molten Salt Reactor, ORNL, USA.
IMSR 400 MWt Integral Molten Salt Reactor, Terrestrial Energy, Canada and USA.
MCFR 30 MWt Molten Chloride Fast Reactor, Terra Power, USA.
MOSART 2400 MWt Molten Salt Actinide Recycler Transforming System, NRC-Kurchatov Institute, the Russian Federation.
MSBR 2250 MWt Molten Salt Breeder Reactor, ORNL, USA.
MSCR 2250 MWt Molten Salt Converter Reactor, ORNL, USA.
MSFR 3000 MWt Molten Salt Fast Reactor, France/Euratom.
MSRE 8 MWt Molten Salt Reactor Experiment, ORNL, USA.
ThorCon 557 MWt Modular Reactor Plant, ThorCon International, Singapore.
TMSR-LF 395 MWt Thorium Molten Salt Reactor–Liquid Fuel, SINAP, China.

Abbreviations

CR Conversion ratio
EFPD Equivalent full power day
GIF Generation IV International Forum
HLW High level waste
IGC Intergranular cracking
LWR Light water reactor
MA Minor actinides
MOX Mixed oxide fuel
MSR Molten salt reactor
ORNL Oak Ridge National Laboratory, USA
PR&PP Proliferation resistance and physical protection
REE Rare earth elements
RW Radioactive waste

SNF Spent nuclear fuel
 TRU Transuranic elements
 UOX Uranium oxide fuel

Introduction

Historically, the molten salt reactor (MSR) concept, where fissile and fertile materials are dissolved in liquid fluorides/chlorides salt, and fission occurs within the fuel salt, which then flows into an intermediary heat exchanger where the heat is transferred to a secondary liquid-salt coolant, is based on a philosophy that presupposes the use of fuel in a form that permits the continuous management of nuclear-physical, chemical and heat transfer processes in the fuel, as well as the regulation of its nuclides content. This degree of freedom can be used for the optimization of the nuclear energy system and gaining maximal profits from the physical potential of nuclear phenomena, but can also advantageously or disadvantageously affect both the reactor safety and the fuel material attractiveness for nuclear weapon proliferation misuse.

The potential advantages of a fluid-fueled reactor design have been recognized for a long time. Accordingly, the MSR has usual benefits attributed to fluid-fuel systems:

- A high coefficient of thermal expansion, which ensures a large negative temperature reactivity feedback. Because the fuel is liquid, it expands when heated, thus slowing the rate of nuclear reactions and facilitating reactor self-regulation and safety.
- The combination of self-regulation and the ability to dump fuel to a previously prepared storage system, with a geometry designed to render criticality impossible and which also has built-in independent heat removal, was considered the better arrangement from a safety point of view;
- The liquid fuel makes it possible in principle to eliminate a need for reactivity excess required for reactor startup, which can be accomplished by preheating the fuel salt in the subcritical drain tanks up to the working temperature and then filling the primary circuit containing the core;
- The liquid nature of fuel eliminates problems associated with loss of coolant;
- The possibility of continuous fission product removal using physical (helium sparging for noble gases removal) and/or pyrochemical processes. Fuel salt can be processed in an on-line mode or in batches in order to retrieve soluble fission products (lanthanides) and actinides. The actinides are then reintroduced into the fuel circuit.
- No xenon poisoning in thermal spectrum cores.
- Simplifies plutonium and minor actinides handling. Transuranic elements including the actinides could in principle remain in the fluid fuel of the core and be utilized in the neutron flux, either by direct fissioning or transmutation to a fissile element until eventually they all undergo fission. On-line fuel processing offers a safer, more efficient and sustainable nuclear power option.
- The avoidance of the transport and fabrication expenses for new fuel elements.
- Thermal stresses and accumulation of defects in the fuel accompanying a change in power are eliminated;
- The chemical interaction of the fuel salt with the intermediate coolant as well as with the air and water is not accompanied by a release of heat and formation of hydrogen as result of exothermic reactions;
- Radiation damage to the fuel is eliminated, since the radiation resistance of fluoride melts is 10^5 times higher than that of water;
- Heat transfer from the fuel salt to intermediate coolant is outside the neutron field, and the fuel-coolant interface does not affect the neutronic processes in the core;
- Low concentration of the actinide's fluorides in the fuel salt and its large margin relative to the saturation temperature ensure that the thermal reservoir of the primary circuit has a significant capacity and that transient processes proceed without sharp changes of temperature;

As for main challenges it has to be remembered that the fuel salt in MSR fills the whole volume of the primary circuit. It becomes highly active owing to fission products. Sometimes the critics of the MSR say that the use of liquid fuel means the loss of safety barrier (fuel cladding). The number of barriers is not a magic number. What is of real importance is a reliability of barriers not only for liquid, but also gas phase. And if one wants to keep a number of barriers, why shouldn't we to put an additional guard vessel for fluid fuel reactor.

From the 1940s up to now, many molten salt reactor concepts have been proposed all over the world using different fuel (U, Pu, Th, etc.) and salt compositions (fluorides, chlorides, etc.). These projects have tried to find the optimum solution for the fuel salt composition meeting, at the same time, requirements due to: neutronic properties (neutron moderation, breeding ratio, fissile inventory, etc.), reactor core operating temperature (salt melting temperature, radiation stability, transport properties), actinide and fission products solubility in the molten salt to guarantee a homogeneous core composition, materials compatibility and salt chemistry control, and, last but not least, processing and low waste requirements.

This article deals with molten salt reactor concepts where the fuel materials are dissolved in the high temperature molten salt fluoride systems. It is mainly focused on high power rating reactor designs with fast/intermediate neutron spectrum and full

processing of fuel salt. For reactor concepts with a solid fuel and a molten salt fluorides mixture as a primary coolant, which are not addressed here, the reader is referenced to [Fratoni \(2021\)](#).

Historical experience

The fluid-fuel reactor which enjoyed the one of the most extended runs of investigatory life was the molten salt fluoride-based system. This began with an aircraft reactor experiment (ARE) ([Cottrell et al., 1955](#)), which operated successfully in 1954 in a “proof-of-principle” short-term test at a thermal power level of 2.5 MW and at temperatures up to 860 °C. The fuel was a solution of UF₄ in other fluorides, with Inconel-clad beryllium as the moderator. Those who took part in this program realized from the early stages that, potentially, the molten salt system was a candidate for a power reactor, and a team at the US Oak Ridge national Laboratory (ORNL) began work on such a concept in 1957 ([Rosenthal et al., 1970](#)).

Through the 1960s, ORNL investigated large power graphite moderated molten salt reactor designs, both converter reactors and breeder reactors based on the Th-²³³U cycle ([Briggs, 1966](#)). Several designs were proposed using single or two fluid graphite moderated cores. In the two-fluid design, a fuel LiF-BeF₂-UF₄ salt carries the fissile ²³³U, and a separate blanket LiF-BeF₂-ThF₄ salt is used for the fertile thorium. The thorium blanket effectively captures the neutrons that leak from the core region leading to highly efficient neutron use (neutron economy), and a high breeding ratio. ²³³U produced in the blanket, is transferred to the fuel salt by a fluorination process. Fluorination is accomplished by bubbling fluorine gas through the blanket LiF-BeF₂-ThF₄ salt, converting produced UF₄ to volatile UF₆. Gaseous UF₆ is converted back to solid UF₄ before being added as LiF-UF₄ eutectic pellets into the fuel salt. This chemical process is still considered nowadays as a powerful chemical step in the liquid fueled molten salt reactors processing flow sheet. The main advantages of a two-fluid system are a reduced initial fissile inventory, and the simplification of fuel salt clean up from both protactinium (protactinium removal is not necessary as it is diluted in the blanket salt) and soluble rare earth fission products from the fuel salt in absence of thorium. Because thorium and rare earth fission products have similar chemical behavior, the removal of rare earth fission products is greatly simplified without thorium in the core fluid. Rare earth fission products absorb neutrons and reduce the production of new fissile fuel, their concentration in the salt needs thus to be kept low, as they have a large cross section for neutron capture in a thermal neutron spectrum. The two-fluid design offers absolute and incontrovertible advantages, but rises however the problem of the barrier wall (either nickel-based alloy or graphite) between the core and the blanket region, that implies a more complex vessel exposed to fuel redox chemistry and radiation damages.

Vessel complexity and limited graphite lifetime (4 years) in contact with the molten fuel salt steered ORNL toward the development of a simpler single fluid 1000 MWe Molten Salt Breeder Reactor (MSBR) reactor plant design. The proposed MSBR graphite moderated core featured as fuel a LiF-BeF₂-ThF₄-UF₄ (72–16–12–0.4 mol%) salt mixture in the primary circuit, and a NaF-NaBF₄ (8–92 mol%) salt mixture as a coolant in the secondary circuit. An improved but complex processing flowsheet was intensively developed to achieve the difficult separation of thorium from rare earth fission products required for a single fluid design ([Mc Neese et al., 1974](#)).

By the end of 1969 as further development of the concept, a molten salt reactor experiment (MSRE) was built and operated successfully at a thermal power level of 7.3 MW and temperatures up to 700 °C with Li, Be, Zr, U, Pu/F fuel salt ([Haubenreich and Engel, 1970](#)). In MSRE primary Li, Be, Zr, U, Pu/F and secondary Li, Be/F circuits were made of the Hastelloy N alloy (base Ni-16Mo-7Cr-5Fe). The MSRE was a very successful experiment, in that it answered many questions and posed a few new ones. Arguably, the most important result was the conclusion that such a concept had the potential to become a practical reactor:

- Keeping salt in the molten state was not difficult.
- Moving salt among tanks was routine.
- No salt leaks during operation.
- Adding enriching salt during operation was uneventful (²³⁵U and ²³³U added as LiF-UF₄ eutectic; plutonium added as PuF₃).
- Static nuclear properties were accurately predicted.
- Excellent dynamic stability with both ²³⁵U and ²³³U as predicted.
- Reactivity change with time as expected (no evidence of uranium loss or deposition).
- Heat transfer and hydraulic performance as predicted.
- No pump maintenance required.
- Stripping of noble gas fission products was effective.
- Good gaseous UF₆ recovery.
- Effective oxide removal from fuel salt
- Substantial corrosion of reactor vessel, as expected.

MSRE ran for long periods of time. It is of interest that it operated initially on ²³⁵U, then on ²³³U, and finally on plutonium. There has always been some controversy as to how a ²³³U-Th system should be started initially, and the plutonium experiment was intended to demonstrate the feasibility of operating with plutonium as the initial fissile material. It is of interest that substantial dismantling of MSRE was achieved very satisfactorily, results being obtained on fission-product deposition and take-up. Although this was only on small-scale plant, the experience from an active-handling point of view was encouraging. Following MSRE

shutdown, the Oak Ridge program concentrated on a number of important items requiring development if the system was to be of commercial interest. Materials problems had been encountered (Haubenreich and Engel, 1970):

1. Hastelloy N was susceptible to damage from helium formation in a neutron radiation field ("radiation hardening"). Later it was found that modified alloys that had fine carbide precipitates within grains would hold the helium and restrain its migration to the grain boundaries.
2. The circuit material specially developed for the system by ORNL (Hastelloy N) was deteriorating in its properties as a result of intergranular attack caused, it was believed, by the fission product tellurium. Later work showed that this tellurium attack could be controlled by keeping the fuel on the reducing side. This is done by adjustment of the chemistry so that about 2% of uranium is in the form of UF_3 as opposed to UF_4 .
3. Tritium formed from the primary salt by neutron reactions mainly with lithium was presenting difficulties as it diffused (as does hydrogen) through the primary and secondary heat exchanger tube walls and led to radioactivity in the steam circuit. After considerable development work it was found that the intermediate salt coolant NaF-NaBF_4 , would capture the tritium that could be then removed and isolated in the gas purge system.
4. In the graphite, xenon and iodine were being taken up to the detriment of the nuclear performance—a non-wetting graphite was needed.
5. In addition, further work in the processing field was clearly identified.

Despite the success of the MSRE reactor and the large research and development efforts including natural and forced convection molten salt loop tests demonstrating many aspects of the reactor technology, the molten salt reactor program at ORNL was closed down in the early 1970s in favor of the liquid metal cooled fast-breeder reactor (Rosenthal et al., 1970). Oak Ridge did continue a modest program until the early 1980s and proposed another molten salt reactor design, the denatured molten salt reactor (DMSR) developed in the 1970s to address proliferation concerns. In the DMSR concept, full fuel salt processing was not performed (other than krypton and xenon gas removal), enriched uranium (19.75%) was used at start-up and as feed material to maintain criticality, and ^{238}U was added as needed to maintain denatured state (Engel et al., 1980). The simplified DMSR converter design with a low power density core allowing a full 30-year graphite lifetime was called the "30 Year Once Through Design". During this period, more countries, including China, France, Japan and the Russian Federation, also placed some efforts in the molten salt reactor concept developments.

Developments within the Generation IV International Forum (GIF) collaboration framework

The new interest in molten salt reactors is a consequence of changing goals and changing technologies. The today family of MSRs (Serp et al., 2014), can be split into several categories depending on the choice of the fissile and fertile materials, the neutronic spectrum, the salt type, the power level, the fuel salt processing mode (limited or full), etc. Although the design of all these reactors will be different, some key safety features will be quite similar, for example the approach for defining the barriers, the definition of the severe accidents, etc. However, some features can differ in particular between graphite moderated and non-graphite moderated cores.

There is a renewed interest in breeder reactors and reactors for the utilization of transuranic elements (TRU) from used fuel. For the latter option, molten salt reactors have the unique advantage that there is no fuel fabrication step—an extremely complex task for fuel elements with higher actinides. Several national or international molten salt reactor concepts were conceived and new research programs were recently launched worldwide (see Table 1).

The MSR developers are in some cases directly supported by national government funded programs. In many cases, however, the developments are being primarily supported by private capital. The different development organizations have differing development strategies. Some are constructing non-nuclear demonstration facilities that are intended to support test reactor development. Others are preparing for performing component and system tests to support first-of-a-kind power reactor deployment. The private companies view each other as competitors, often have patents (or have filed patent applications) on key design features, and are protective of their technology and design information. This section discusses relevant molten salt reactor features being investigated within the GIF collaboration framework.

Within the GIF collaboration framework, a memorandum of understanding between Australia, Canada, France, Euratom, the Russian Federation, Switzerland and United States (with China, Japan and the Republic of Korea as observers) is providing a flexible structure for R&D cooperation on molten salt reactor concepts (Serp et al., 2014).

The common objective of molten salt reactor projects in different GIF countries is to propose a conceptual design with the best system configuration—resulting from physical, chemical and material studies—for the reactor core, the fuel salt processing unit and wastes conditioning. The mastering of molten salt reactor technically challenging technology is requiring concerted, long-term international R&D efforts, specifically:

- Additional studies of the salt physical, chemical and thermodynamic properties,
- System design and safety analyses, including development of advanced coupled neutronic and thermal hydraulic simulation models,

Table 1 List of the different MSR systems (Dolan, 2017).

Name	Developer	Power (MWt)	Fuel/carrier/moderator
Thermal Spectrum MSRs			
Thorium Molten Salt Reactor–Liquid Fuel (TMSR-LF)	SINAP, China	395	ThF ₄ - ²³³ UF ₄ /LiF-BeF ₂ /graphite
Integral Molten Salt Reactor (IMSR)	Terrestrial Energy, Canada and USA	400	UF ₄ /fluorides/graphite
ThorCon Reactor	ThorCon International, Singapore	557 × 2	ThF ₄ - ²³³⁻²³⁵ UF ₄ /NaF-BeF ₂ /graphite
Liquid-Fluoride Thorium Reactor (LFTR)	Flibe Energy, USA	600	ThF ₄ - ²³³ UF ₄ /LiF-BeF ₂ /graphite
FUJI-U3	Japan	450	ThF ₄ - ²³³ UF ₄ /LiF-BeF ₂ /graphite
Advanced Molten Salt Break-Even Inherently-Safe Dual-Mission Experimental and Test Reactor (AMBIDEXTER)	Ajou University, Republic of Korea	250	²³³ UF ₄ -ThF ₄ /LiF-BeF ₂
Transatomic Power MSR (TAP)	Transatomic Power, USA	1250	UF ₄ /FLiNaK/SiC clad ZrH _{1.6}
Compact Used Fuel Burner (CUBE)	Seaborg Technologies, Denmark	250	SNF/fluorides/graphite
Process Heat Reactor	Thorenco, USA	50	UF ₄ /NaF-BeF ₂ /Be rods
Stable Salt Thermal Reactor (SSR-U)	Moltex Energy, UK	300–2500	UF ₄ /fluorides/graphite
Fast/Epithermal Spectrum MSRs			
Molten Salt Fast Reactor (MSFR)	SAMOFAR, France–EU–Switzerland	3000	ThF ₄ -UF ₄ /LiF-
Molten Salt Actinide Recycler and Transformer (MOSART)	Kurchatov Institute, Russia	2400	TRUF ₃ or ThF ₄ -UF ₄ /LiF-BeF ₂ or NaF- ⁷ LiF-BeF ₂
U-Pu Fast Molten Salt Reactor (U-Pu FMSR)	VNIINM, Russia	3200	UF ₄ -PuF ₃ /LiF-NaF-KF
Indian Molten Salt Breeder Reactor (IMSB)	BARC, India	1900	ThF ₄ -UF ₄ /LiF-
Stable Salt Fast Reactor (SSR-W)	Moltex Energy, UK	750–2500	PuF ₃ /fluorides
TerraPower Molten Chloride Fast Reactor (MCFR)	TerraPower (USA)	30	U-Pu/chlorides
Molten Chloride Salt Fast Reactor (MCSFR)	Elysium Industries (USA and Canada)	100–5000	U-Pu/chlorides
Dual Fluid Reactor (DFR)	Dual Fluid Reactor, Germany	3000	U-Pu/chlorides

- Development of advanced materials, including studies of their compatibility with molten salts and behavior under high neutron fluxes at high temperature conditions,
- Mastering of corrosion and tritium release prevention technologies, based on proper molten salt Redox control,
- Development of efficient techniques of gaseous fission products extraction from the fuel salt by helium bubbling,
- Fuel salt processing flowsheet, including reductive extraction tests (actinide-lanthanide separation),
- Development of a safety, safeguards, security and proliferation resistance approaches dedicated to liquid fuelled reactors.

Thermal spectrum cores

As can be seen from Table 1, Canada, Japan, Singapore and South Korea are paying attention to the development of the small and medium power liquid fuel units with thermal-neutron spectrum graphite moderated cores and limited fuel salt processing. In China, the Thorium Molten Salt Reactor (TMSR) Program was initiated by the Chinese Academy of Sciences (CAS) at Shanghai Institute of Applied Physics (SINAP) in 2011, which involves a closed U-Th fuel cycle for MSR (Serp et al., 2014). The new candidate site for the 2 MWt TMSR-LF1 test reactor was selected in 2017. It will be located in Wuwei, Gansu Province, about 2000 Km from Shanghai.

The thermal spectrum MSR approach involves moderator material, with the most commonly proposed being graphite, as it has been shown to be directly compatible with fluoride fuel salts. Several other moderators have been suggested such as metal hydrides (or deuterides), Be or BeO, and even water, heavy water and molten sodium hydroxide. Each non-graphite alternative has specific challenges and all of them do require cladding materials to be validated for use in contact with fuel salts in a high neutron flux environment. All thermal spectrum MSR variants offer benefits in terms of the safety case development including large thermal inertia, longer neutron lifetimes and practically absence of out of core criticality issues due to the low fissile enrichment levels in the fuel salts.

Thermal spectrum MSR variants can suit almost all fuel cycles. Traditional thermal-neutron spectrum graphite moderated MSRs require full and relatively rapid processing of the molten salt if they are to be breeder reactors (Dolan, 2017). For the thorium-²³³U breeder cycle there is the advantage of very low starting fissile inventory to compensate for lower breeding ratios compared to fast spectrum variants. The thermal spectrum opens the possibility of simplified converter options, removing the need for fuel processing by employing low enriched uranium (LEU), on its own or in combination with thorium. In fact, the use of commercially available standard assay uranium enrichment (<5%) is possible to further simplify development time-lines. Converter designs also allow consideration of carrier salts that are free from the need for enriched lithium 7 and/or beryllium (and their tritium production) as neutron economy is of lower importance for converters. The thermal spectrum approach has a larger historic development foundation and ability to significantly lower neutron flux reaching reactor vessel material compared to fast spectrum MSR.

ThorCon

ThorCon is a straightforward scale-up of the successful US MSRE at Oak Ridge National Laboratory. ThorCon uses only commercially available components and materials, such as SS316 steel, but avoiding unobtainable lithium-7 for molten salt. Fuel salt with dissolved thorium and uranium (enriched to 19.7%) fluorides is similarly removed and replaced, on an 8-year cycle (Dolan, 2017). ThorCon is divided into 250 MWe power modules or PMODs. Each module contains two replaceable reactors in sealed Cans. The Cans sit in silos. At any one time, just one of the Cans of each module is producing power. The other Can is in cooldown mode.

The Can contains the reactor, called the Pot, a primary loop heat exchanger (PHX), and a primary loop pump (PLP). The pump takes fuel salt from the Pot at 704 °C, and pushes the fuel salt over to the PHX at a rate of just under 3000 kg/second. Flowing downward through the PHX, the fuel salt transfers heat to a secondary salt, and is cooled to 565 °C in the process. The fuel salt then flows over to the bottom of the Pot, and rises through the reactor core, which is mostly filled with graphite blocks.

The Pot pressure is 3 bar gage. The outlet temperature of 704 °C results in an overall plant efficiency of 46%. The Can is a cylinder 11.6 m high and 7.3 m in diameter. It weighs about 400 tons. The Can has only one major moving part, the primary loop pump.

Every 4 years the Can that has been cooling is removed and replaced with a new Can. The fuel salt is transferred to the new Can, and the Can that has been operating goes into cool down mode. Every 4 years the entire primary loop is changed out, returned to a centralized recycling facility, decontaminated, disassembled, inspected, and refurbished. Incipient problems are caught before they can turn into casualties. Upgrades can be introduced without significantly disrupting power generation.

Each ThorCon plant is based on one or more hulls, each containing two 250 MWe power modules, a 500 MW super-critical turbogenerator, gas insulated switchgear (GIS), a decay heat pond, and auxiliaries. The fission island is at the forward end of the hull. Aft of the fission island is the steam generating cell (SGC). Aft of the SGC is the turbine hall, which contains the turbogenerator, exciter, condensers, feed heaters, pumps, and condensate treatment.

The complete ThorCon is manufactured in 150–500-t blocks in a shipyard, assembled, then towed to the site. This produces order of magnitude improvements in productivity, quality control, and build time. A single large reactor yard can turn out 20 GW of ThorCon power plants per year. ThorCon is a system for building power plants. Shipyards can rapidly fabricate 500 MW power plants on hulls to be towed to near-shore sites. Claim is made that prototype could begin testing in 4 years.

ThorCon has been working with the Indonesian government to add reliable electric power to the grid. In 2019 the Ministry of Energy successfully completed a study of the safety, economics, and grid impact of the 500 MWe prototype ThorConIsle. Phase 1 is to build and test it with step by step commissioning, ending in a type license for future power plants. Phase 2 is shipyard production of ThorCon plants to provide an additional 3 GW of cheap, reliable electric power.

IMSR

The integral molten salt reactor is so called because it integrates into a compact, sealed and replaceable nuclear reactor unit, called the IMSR Core-unit (LeBlanc, 2010; Dolan, 2017). The Core-unit comes in a single size designed to deliver 400 MW of thermal heat which can be used for multiple applications. If used to generate electricity then the notional capacity is 190 MW electrical. The unit include all the primary components of the nuclear reactor that operate on the liquid molten fluoride salt fuel: moderator, primary heat exchangers, pumps and shutdown rods. The Core-unit forms the heart of the IMSR system. In the Core-unit, the fuel salt is circulated between the graphite core and heat exchangers. The Core-unit itself is placed inside a surrounding vessel called the guard vessel. The entire Core-unit module can be lifted out for replacement. The guard vessel that surrounds the Core-unit acts as a [containment vessel](#). In turn, a shielded silo surrounds the guard vessel.

The IMSR is designed to have all the safety features associated with the MSR family including low pressure operation (the reactor and primary coolant is operated near normal atmospheric pressure), the inability to lose primary coolant (the fuel is the coolant), the inability to suffer a meltdown accident (the fuel operates in an already molten state) and the robust chemical binding of the fission products within the primary coolant salt (reduced pathway for accidental release of fission products).

The design uses standard assay [low-enriched uranium](#) fuel, with less than 5% U²³⁵ with a simple converter [fuel cycle](#) objective (as do most operating power reactors today). The IMSR is a [thermal-neutron reactor](#) moderated by vertical [graphite](#) tubular elements. The molten salt fuel-coolant mixture flows upward through these tubular elements where it goes critical. After heating up in this moderated core the liquid fuel flows upward through a central common chimney and is then pulled downward by pumps through heat exchangers positioned inside the reactor vessel. The liquid fuel then flows down the outer edge of the reactor core to repeat the cycle. All the primary components, heat exchangers, pumps etc. are positioned inside the reactor vessel. The reactor's integrated architecture avoids the use of external piping for the fuel that could leak or break.

The piping external to the reactor vessel contain two additional salt loops in series: a secondary, nonradioactive coolant salt, followed by another (third) coolant salt. These salt loops act as additional barriers to any radionuclides and improve the system's heat capacity. It also allows easier integration with the heat sink end of the plant; either process heat or power applications using standard industrial grade [steam turbine](#) plants are envisioned by terrestrial energy.

The IMSR Core-unit is designed to be completely replaced after a 7-year period of operation. This ensures that a sufficient operational lifetime of materials used in the IMSR reactor core can be achieved. During operation, small fresh fuel/salt batches are periodically added to the reactor system. This online refueling process does not require the mechanical refueling machinery required for solid fuel reactor systems.

Fast spectrum and intermediate cores

R&D within the GIF collaboration framework is mainly focused on MSR options with homogeneous cores without graphite moderator. Fast MSRs have strongly negative reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors (Allibert et al., 2016). Compared with solid-fuelled reactors, those systems have lower fissile inventories, no radiation damage constraints on attainable fuel burnup, reduced radiation damages to materials since no solid materials are located in the center of the core, no used nuclear fuel, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic composition of fuel in the reactor.

A molten chloride fast spectrum reactor (MCFR) concept based on U/Pu fuel cycle (Krepel et al., 2021) has been under development by TerraPower Inc. for the past 5 years. The MCFR is intended to have a very hard neutron spectrum to avoid requiring any fissile material input after its initial core load or separation of fissile materials from the remainder of the fuel salt.

The fast neutron spectrum MSRs loaded with fluoride fuel salt open promising possibilities to exploit the ^{232}Th - ^{233}U cycle and can also contribute, in the transmuter mode, to significantly diminishing the radiotoxic inventory from current-reactor used fuel in particular by lowering the mass of TRUs. Developments in Russia (Ignatiev et al., 2014) on the 2400 MWt molten salt actinide recycling and transmuter (MOSART) and in France (Heuer et al., 2014) on the 3000 MWt thorium molten salt reactor (MSFR) address the concept of large power units. The first concept aims to be used as an efficient burner of TRU waste from used LWR fuel without uranium and thorium support. The fast-spectrum MSRs have a higher breeding ratio that enables the reactor to be a breeder reactor with longer cycle times to remove soluble fission products. When comparing with the reference 2250 MWt MSBR design, for which a 4 m³/day reprocessing rate was mandatory to insure a sufficient breeding ratio, the processing rate can be reduced to 0.04 m³/day in the MSFR concept (Doligez et al., 2014). Pyrochemical salt processing in MCFR, MOSART and MSFR designs is mandatory to avoid the accumulation in the fuel salt of large quantities of lanthanides and zirconium that could be detrimental to several physical-chemical properties, such as actinide trifluorides solubility. The availability of a high neutron flux and the lack of structural materials in the liquid homogeneous core, lead to the optimization of the neutron balance, as well as to increased flexibility allowing fuel salt composition changes without core modification and/or reactor shutdown. The plutonium and minor actinides (MA) burning rate is directly proportional to the core power density.

MOSART

A promising single fluid configuration for the 2400 MWt MOSART is the homogeneous cylindrical core 3.6 m high and 3.4 m in diameter (See Fig. 1, right side) filled with 100% of 73LiF-27BeF₂ (mol%) salt mixture (Ignatiev et al., 2014). It is feasible to design a critical homogeneous core fuelled only with TRU trifluorides from UOX or MOX LWR used fuel and ensuring that the TRU equilibrium concentration is kept below the solubility limit (~ 3 mol%) in trifluorides at 650 °C, that is, the minimum fuel salt temperature in the primary circuit. The effective neutron flux in the fuel is close to $1 \times 10^{15} \text{ n cm}^{-2} \text{ second}^{-1}$ (Ignatiev et al., 2014). The 2400 MWt MOSART core that was initially loaded with TRU fluorides derived from the used fuel of a LWR having 10–15% MA/(plutonium + MA) ratio is able to accommodate TRU fluorides make-up with up to 33% MA/(plutonium + MA) ratio, without any core modification. The equilibrium TRU loading in primary circuit was calculated to be in the range of 3.6–20.0 t, depending on the MA/(plutonium + MA) ratio in the core make-up. The studies showed that the 2400 MWt MOSART can transmute up to 250 kg of minor actinides and about 500 kg of plutonium per year (in the case of make-up loadings having 33% MA/(plutonium + MA) ratio).

The structural material selected for the MOSART fluoride-salt container is the special Ni-Mo HN80MTY alloy type with a low concentration of Cr alloyed by 1% of Al. The composition of the alloy is optimized for corrosion resistance (both in a low oxygen gas atmosphere and in molten salt fluorides), irradiation resistance, and high temperature mechanical properties (Ignatiev and Surenkov, 2017).

Interior Ni-base reflectors/shielding are employed to reduce the radiation damage to the reactor vessel (see Fig. 1, right). Fuel salt chemistry control is employed to substantially limit oxidizing the container alloy constituents. For those systems that include graphite, the salt is maintained in a redox window oxidizing enough to prevent damage to the graphite while reducing enough to prevent alloy oxidative corrosion. The systems without graphite can maintain the salt in a more reducing condition.

The radiation resistance of reflectors/shielding material determines the upper limit for the fuel salt power density in the MOSART design. While the damage caused by fast neutrons is critical for graphite, for high-nickel alloys the reduction in plasticity at a temperature above 500 °C, associated with the formation of helium, due to neutron absorption by Ni, along the grain boundaries, is the most important process, caused by both fast and thermal neutrons. To obtain an acceptable service life for reflectors (> 5 years), the core power density should not exceed 130–150 W/cm³. Otherwise, frequent stops to replace the reflector will result in a reduction in the reactor load factor and an unjustified increase in operating costs. In addition, with this limitation for the core power density, there is no problem of heat removal from the fuel circuit. In this case the service life for the reactor vessel, made of the HN80MTY alloy, will be about 50 years. For the 2400 MWt MOSART, taking into account the adopted limits, the primary circuit will contain 50 m³ of fuel salt, of which only half is in the core.

For MOSART, the proposed fuel salt processing to remove fission products (see Table 2), and the required fuel maintenance operations include (1) continuous removal (by the sparging and stripping section of the reactor) of the fission products krypton and xenon, (2) addition of TRUs to replace those lost by burnup, (3) recycling of all actinides, (4) removal of soluble fission

products (principally rare earths), (5) removal of inadvertent oxide contaminants from the fuel, if any, and (6) removal of a portion of the insoluble noble and semi-noble fission products.

For multiple TRUs recycling, since plutonium and minor actinides must be removed from the fuel solvent before yttrium and rare earth fission products, the MOSART must include a system that provides removal of TRUs from the fuel salt and their reintroduction to the purified solvent. This plutonium reintroduction circuit has the advantage of also returning americium and curium to the reactor fuel (low separation coefficients within TRU group). Since the higher actinides would always accompany the plutonium, this operation would never produce a “clean” fissile material that would be attractive for diversion. Adoption of the molten Li, Be/F salt mixture, due to the high separation coefficients between actinides and lanthanides, thus providing effective removal of soluble fission products based on the reductive extraction. This, in turn, allows (in comparison with solid fuel reactors) to substantially reduce the time, during which actinides reside in the external fuel cycle, as well as the losses in waste streams in the case of multiple recycling.

The flexibility of the single fluid MOSART concept fuel cycle is allowing its operation in self-sustainable mode using different loadings and make-up. For example, the single fluid 2400 MWt Li, Be/F MOSART core containing an initial loading of 2 mol% of ThF_4 and 1.2 mol% of TRUF_3 , adopting a LnF_3 removal cycle of 300 EFPD after 12 years of a slow increase of its thorium content in the solvent, can operate as a self-sustainable system without any TRUF_3 make-up, relying only on thorium support. The maximum concentration of TRUF_3 during this transition does not exceed 1.7 mol%. At equilibrium the molar fraction of ThF_4 fertile material in the fuel salt is nearly 6%, and it is enough to support the system with a conversion rate equal to one for a period of 50 years reactor lifetime.

MSFR

The reference MSFR supported by the European Union is a 3000 MWt reactor with a total fuel salt volume of 18 m^3 , half of the salt (9 m^3) located in the core and half in the external circuits: salt collectors, salt-bubble separators, fuel heat exchangers, pumps, salt injectors and pipes (Heuer et al., 2014). The fuel salt runs through the total cycle in 3–4 seconds. The fuel salt is composed of lithium fluoride and thorium fluoride, and the proportion of heavy nuclei is fixed at 22.5 mol%. The preliminary design of the primary circuit of the MSFR is a single compact cylinder 2.25 m high $\times$ 2.25 m diameter (see Fig. 1, left), where the nuclear reactions occur within the liquid fluoride fuel salt acting also as the coolant. The operating core temperatures for fuel salt chosen are 650°C (inlet temperature) and 750°C (outlet temperature). The lower limit is set by the salt's melting point (565°C), while the upper limit is imposed by the Ni base structural materials performance (limit around 800°C). The fuel salt flows in the central part of the core freely from the bottom to the top without any solid moderator. The return path of the salt is divided into 16 sets of pumps and heat exchangers located around the core. Bubbles are injected in the fuel salt circulation after the exchangers and separated from the liquid at the core outlet. As in MOSART the lower neutron reflector is connected to a drain system enabling the reactor core to be drained for planned shut downs or in case of incidents that lead to a temperature increase in the core. Thus, the entire fuel inventory

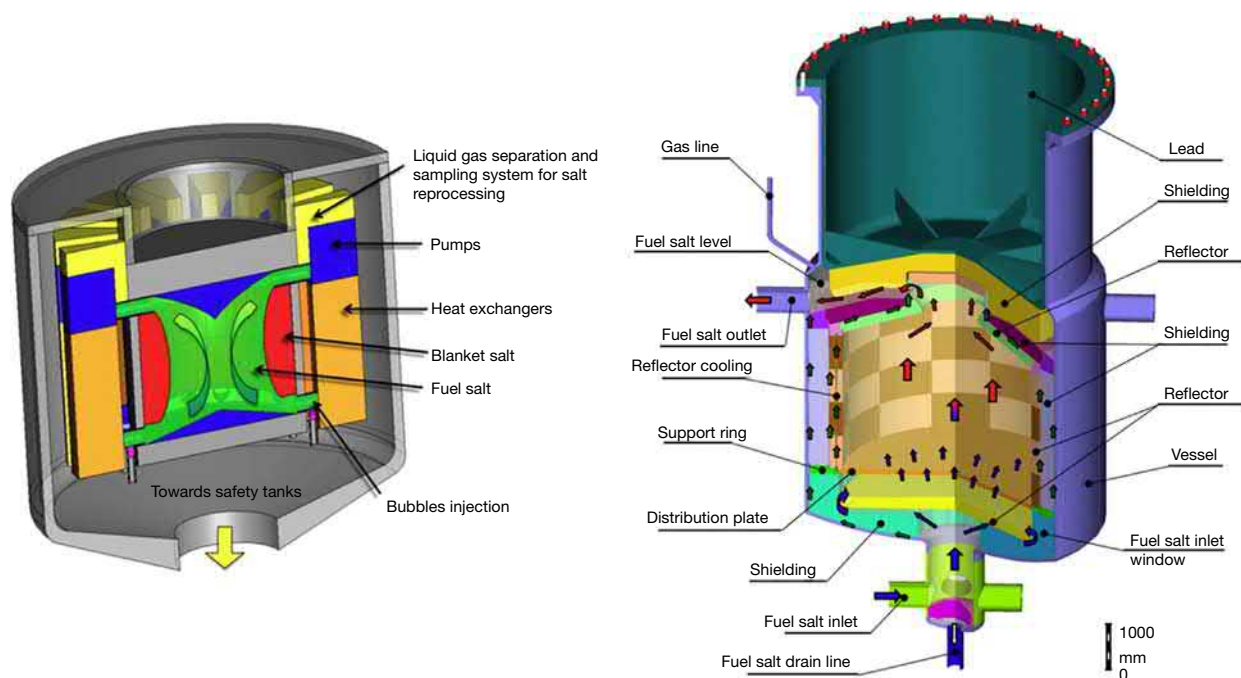


Fig. 1 MSFR (left) and MOSART (right) concepts.

Table 2 Summary time-lines for fission product removal and actinides recycling in MOSART (Ignatiev et al., 2014).

Element	Time
Kr, Xe	50 seconds
Zn, Ga, Ge, As, Se, Nb, Mo, Ru, Rh, Pd, Ag, Tc, Cd, In, Sn, Sb, Te	2–4 hours
Zr	1–3 years
Ni, Fe, Cr	1–3 years
Pu, Am, Cm, Np, U	1–3 years
Y, La, Ce, Pr, Nd, Pm, Gd, Tb, Dy, Ho, Er	1–3 years
Sm, Eu	1–3 years
Sr, Ba, Rb, Cs	5–10 years
Li, Be, Th	50 years

can be passively drained by gravity into subcritical, passively cooled tanks. The MSFR can be started either directly with ^{233}U as fissile matter or with a mix of enriched uranium (13%), plutonium and minor actinides from PWRs spent fuel. MSFR configurations corresponding to various starting modes of the reactor are all characterized by excellent safety reactivity coefficients and have the same very good deployment capabilities. In the MSFR, the liquid fuel processing is an integral part of the reactor where a small fraction of the molten salt (40 L/day) is set aside to be processed for fission product removal and then returned to the reactor. Actinides and soluble fission products that are strongly bound to the salt, can be separated by pyro-chemical techniques. The most effective technique to achieve uranium removal seems to be fluorination as UF_6 . The fast neutron spectrum relaxes the requirements for soluble fission product removal times by reductive extraction process up to 1 year.

The common objective of the MOSART and MSFR projects is to develop a conceptual design for intermediate/fast-spectrum MSRs with an effective system configuration—resulting from physical, chemical and material studies—for the reactor core, the reprocessing unit and waste conditioning. The conceptual design activities are intended to increase the confidence that fast-spectrum MSR systems can satisfy the goals of GIF reactors in terms of sustainability (Th breeder), non-proliferation (integrated fuel cycle, multi-recycling of actinides), resource savings (closed Th/U fuel cycle, no uranium enrichment), safety (no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burner). No show stoppers have been identified but almost all the technology remains to be tested and a safety approach dedicated to this type of reactors has to be developed to assess the potential advantages and to prove the viability of such innovative of fast-spectrum molten salt systems.

Open technical issues and future R&D priorities

Specific scientific and technological challenges remain for the development of liquid fueled MSRs and the safety approach has yet to be established (Serp et al., 2014). Liquid fueled MSR viability assessment should address the essential R&D issues discussed below in this section.

Salt and container material

Experimental investigations of physical-chemical properties of fuel salts are carried out to elucidate the influence of the salt composition on thermal and physical properties (melting temperature, solubility of actinides and vapor pressure). Experimental results, received in the Russian Federation and Euratom laboratories on key physical and chemical properties of MOSART and MSFR fuel salts, which determine main core parameters (e.g., inlet/outlet temperatures, critical loadings and temperature reactivity coefficient of the fuel salt, etc.) are described in detail elsewhere (Ignatiev et al., 2016). These data are now used in multi-physics numerical codes (neutronics and thermo-hydraulics) for the simulations of the fuel salt heat and flow distribution in the reactor cavity (reactor reactivity, fuel irradiation, reactivity feedback coefficients, reactor core wall temperature, etc.).

A large body of literature exists on the corrosion of metallic alloys by molten salt fluorides. Much of this work was done in the past at ORNL (Mc Neese et al., 1974) and later at Kurchatov Institute (Novikov et al., 1984). It involved both thermal and forced convection loops. Uniform corrosion is in the form of bare alloy dissolution driven by the oxidants in the salt. The thermal gradient driven corrosion is inevitable in the MSR circuits. Thus, the corrosion rate at hot sections must be minimized to avoid the clogging problems at cold sections for heat exchangers. The major impurities that must be removed to prevent severe corrosion of the container metal are moisture/oxide contaminants. During the US MSR program, considerable effort was devoted to salt purification by HF/H_2 sparging of the molten salt, which is described in numerous ORNL reports (Mc Neese et al., 1974). A great deal of effort has still to be devoted to develop analytical and purification methods able to:

- Identify the oxygen-containing species (oxide type, hydroxyl, etc.),
- Purify the molten salt mixture,
- Determine accurately the oxygen content in salt melts.

The structural materials retained for the MSR container are special Ni-Mo alloys with a low concentration of Cr down to 6–7 mass%. The composition of the alloy was optimized by ORNL researchers for corrosion resistance (both in a low oxygen gas atmosphere and in molten fluorides), irradiation resistance and high temperature mechanical properties. Less severe corrosion attack ($< 10 \mu\text{m}/\text{year}$) was seen for the Hastelloy N in contact with the MSRE fuel $\text{LiF-BeF}_2\text{-ZrF}_4\text{-UF}_4$ salt at temperatures up to 700°C for 3 years (26,000 hours). The most surprising observation was the almost complete absence of corrosion for Hastelloy N during the 3-year period to the MSRE coolant LiF-BeF_2 salt.

Hastelloy N, is not currently approved by ASME code as a material for high-temperature nuclear power plants. Thus, a materials qualification effort would be required for use of Hastelloy N in any part of the containment boundary or to perform any other safety function. Developing a safety case for limited term Hastelloy N use at temperatures less than 700°C may be possible based upon existing data from the earlier MSR program and/or limited term supplemental qualification testing. The maximum allowable stress for Hastelloy N decreases rapidly above 600°C , becoming too low for practical use above 700°C . Compared to the leading candidate materials that are usually considered for high-temperature nuclear reactor construction, Hastelloy N has significantly lower high-temperature strength. However, developing and qualifying improved performance alloys for longer term high stress applications will require substantial investment over a number of years. Kurchatov Institute in the Russian Federation and US ORNL has continued the development of successor alloys to Hastelloy N (Ignatiev and Surenkov, 2017). Samples of the new alloys (e.g., Russian HN80MTY alloy) show promise for application in high stress applications at higher temperatures along with good fluoride salt corrosion resistance. As a first step, early phase, long-term advanced salt-compatible alloy property measurements will need to be performed. Thus, improved alloys can be developed and made available for commercial MSR deployments while a limited-term safety case for use of Hastelloy N at test reactors can be developed in parallel.

Material integrity is directly linked to the instrumentation and control of liquid salt because the corrosion of the structural materials by the molten salt strongly depends on the redox potential of the salt. It has been demonstrated that the salt redox potential is a key parameter in the corrosion phenomena of structural materials of MSRs. The chemical corrosion can be controlled by a redox buffer, which controls the potential of the fuel salt. The redox buffer considered for MSR is the redox system U(IV)/U(III) . This potential has to be measured on line in the reactor core because the potential increases with operation time due to the fission reactions. Addition of a reducing agent leads to a decrease of the fuel salt potential.

From 1965 to 1969, the successful operation of the MSRE proved that the fission of $^{235}\text{UF}_4$ as well as of $^{233}\text{UF}_4$ made the fuel salt moderately oxidizing and proved the absence of metallic uranium deposition or uranium carbide formation incidence due to successive fissioning. The U(IV)/U(III) ratio could be maintained within the projected range by periodic dissolution of beryllium metal bar suspended in the pump bowl. A 10 g Be metal bar equivalent to 300 g of U(III) and could compensate fissioning effect for 10,000 full power days operation of the 7.5 MW MSRE. During the post MSRE work, it was found that the significant inter-granular cracks in Hastelloy N due to presence of fission product tellurium could be suppressed by adjusting the U(IV)/U(III) ratio no higher than 70.

In 1968, before the MSRE was switched to ^{233}U operation, ORNL claimed that no significant differences are believed to exist in the yields or chemistry of the principal species of fission products which would result from incorporation of PuF_3 in MSR fuels. Indeed, there were fission yield data for thermal fission of ^{235}U in a nuclear engineering handbook at that period, but no data were available for ^{239}Pu fission.

The feasibility of using PuF_3 in startup operation of an MSR did not therefore appear to require separate research program relative to fission products from plutonium. ORNL described further that with UF_4 as the fissile solute, a slight excess of fluoride ions developed. Use of a trifluoride solute, however, should result in a cation excess, and should cause the fuel solution to generate a mild reducing potential. This last sentence was undisputable; however, there might be some prejudice even if he had described “trifluoride solute” but not “plutonium trifluoride.”

The recommendation by ORNL at that time was; redox potential control of the salt loop was achieved by a combination of redox adjustment and redox buffering. A redox buffer, such as the U(III)/U(IV) couple used during MSRE, is a sink for absorbing the chemical effects of fission in safe manner. For MSR systems that did not contain uranium it was proposed that the Ce(III)/Ce(IV) couple be used as a buffer. The level of buffer need not be any greater than that of the uranium used during the last phase of MSRE operation (0.13 mol%), and can probably be much less. If in-line, side stream, or semi-batch adjustment of redox state of the salt was accomplished (either electrochemically or by hydrofluorination) then it might not be necessary to add a redox buffering agent at all. Hydrofluorination of the salt to set the redox state of the salt has been well established and the only barrier to its implementation was the natural desire to exclude HF from the primary off-gas systems. Because of this bias, it was probably more practical to consider a side stream or semi-batch hydrofluorination operation.

The adoption of fast spectrum MSR designs and the possible application of MSR as industrial heat sources have introduced new challenges such as higher salt operating temperatures (up to 750°C), metallic barriers and reflectors able to function in a strong neutron flux. The compatibility of salts with structural materials for fuel and coolant circuits, as well as fuel processing material development needs thus to be addressed.

Special attention will have to be paid to the measure and control of the U(IV)/U(III) ratio in order to reach the desired corrosion resistance of Ni based alloys (Engel et al., 1980). The instrumentation and control of liquid salt must be developed with in-situ measurements at a lab-scale in order to:

- Measure and control the redox potential of the salt and to limit the corrosion.
- Measure the composition of the salt containing fissile and fertile elements.

In MSR Te fission product diffuses to container material preferentially along the grain boundaries and causes IGC by the formation of brittle metallic tellurides on grain boundaries. The formation of these telluride compounds is probably by the reactions of Te with Ni and Cr (Ignatiev and Surenkov, 2017). There exist two options to reduce the tellurium induced IGC. The first option is to minimize the preferential formation sites (e.g., intergranular carbides) for tellurides by alloy modifications with Nb or Al. The other option is to reduce the Te to its negative oxidation states in the salt by controlling the redox condition.

To improve the resistance of Ni-based alloys against irradiation embrittlement, additions of alloying elements to form finely dispersed carbides within the grains have been found to be effective (Ignatiev and Surenkov, 2017).

Experiments have shown that the radiation damage accelerates the corrosion of Hastelloy N exposed to molten fluoride salts. The resistance of Hastelloy N to damages caused by simultaneous thermal neutron radiation and molten salt corrosion seems adequate based on MSRE experience (Haubenreich and Engel, 1970). However, effect of fast neutron radiation damage and salt corrosion attack must be investigated for fast MSRs.

Advanced Ni-Mo-Cr alloys are developed and tested in the Russian Federation (Ignatiev and Surenkov, 2017). In the molten Li, Be, U/F and Li, Be, Th, U/F salts with $[U(IV)]/[U(III)]$ ratio ≤ 60 up to temperatures 750 °C the HN80MTY alloy with 1% added aluminum is the most resistant to tellurium intergranular cracking of Ni-base alloys under study (including US Hastelloy N, Chinese GH3535, Czech MONICR and French EM-721 alloys). The metallurgy and in-service properties of these alloys need to be investigated in further details regarding irradiation resistance and industrialization.

The next steps needed to develop these alloys must involve:

- Irradiation, corrosion, tellurium exposure, mechanical property and fabrication tests to finalize the composition for scale up,
- Procurement of large commercial heats of the reference alloy,
- Mechanical property and corrosion tests of at least 10,000 hours duration,
- Development of design methods and rules needed to design a reactor (breeder or burner) to be built of the modified alloy.

Tritium control is an important issue for MSRs because it is the only radionuclide that under normal operating conditions, without failed fuel, has the potential for significant release (Mc Neese et al., 1974). Tritium is generated primarily by the interaction of neutrons with lithium and beryllium in the primary coolant. MSRs will produce significantly more tritium compared to LWRs. Tritium capture at MSR designs is complicated because at high temperatures tritium permeates available structural alloys. The large contact surface area and thin walls of heat exchanger tubes means that heat exchangers will be the primary release pathway. Tritium trapping in the secondary NaF-NaBF₄ coolant was proposed and experimentally proved in molten salt loop by ORNL for the MSBR design.

Fuel salt clean up

The different steps of the fuel salt processing process (except fluorination for which a large experimental feedback from US research work is found in the literature (Mc Neese et al., 1974)) are currently tested in the Russian Federation, France and Euratom laboratories (Ignatiev et al., 2016). The work includes the measurement of actinide and fission product distribution coefficients to evaluate the efficiency of the reductive extraction process using two different liquid pools of bismuth containing lithium as developed by ORNL (Mc Neese et al., 1974). The other steps of the fuel salt cleanup are also tested in Russian and French laboratories such as the back extraction of lanthanides in a chloride or/and fluoride molten salt and the precipitation of lanthanide oxides using bubbling of Ar-H₂O gaseous mixture (Ignatiev et al., 2014). The extraction efficiencies for lanthanides and actinides were estimated using several literature sources leading to the construction of a database available for molten LiF-BeF₂, LiF-BeF₂-ThF₄ and LiF-ThF₄ salt mixtures (Serp et al., 2014).

The coupling of neutronic and salt processing simulation codes in an integrated numerical tool is also used to calculate the extraction efficiencies of fission products, their location in the whole system (reactor and processing unit). Studies are however limited by the uncertainties on the design and knowledge of the complete chemical processing scheme.

The thermodynamic properties of all phases considered in a multi-component system such as the fuel salt have to be assessed in order to collect new data which are necessary for developments of MSR designs, reprocessing scheme and simulation codes. Physical-chemical behavior of fuel salts and in particular coupling between neutronic, thermal-hydraulic and chemistry need to be thoroughly investigated.

Acquisition of fundamental data for the extraction processes is still needed especially for the actinide-rare earth fission products separation. The extraction of rare earth elements has to be done because of the low solubility of these trifluoride elements and their neutron captures that decrease the reactivity balance. This critical step of the pyrochemical process have to be examined both from a thermodynamic and kinetic point of view. The different steps involved in the processing of the fuel salt have not yet been studied from the technological and engineering points of view and will have to be implemented as the system moves from the lab-scale to the industrial scale. Non-proliferation issues related to the fuel salt processing scheme should also be more deeply addressed.

The efficiency of the non-soluble fission products (noble metal and gases) extraction by helium bubbling, which is needed for circuit components life-time and thus safety, will have to be demonstrated in forced convection molten salt loops.

Design and safety

MSRs are very different in the safety approach from solid fuel reactors. In the new MSR concepts developed worldwide, safety will be an essential point to be considered all along the R&D studies. The first step of the safety approach is a systematic description of the reactor components, limited to the main systems surrounding the core, in order to identify the typical accident initiators. The development and the qualification of appropriate simulation tools to study normal and incidental situations have been initiated within the Euratom and the complementary Rosatom projects (Ignatiev et al., 2016). In order to assess the behavior of the fuel salt after reactor shut down, a tool based on point kinetics able to calculate the decay heat have been developed and validated for fast spectrum MSR. Loss of heat sink transients can be calculated allowing the evaluation of their impact on the fuel salt temperature. These residual heat calculations will be the basis for the design of the safety draining system, as drainage of the salt is foreseen for reactor shut down, whether in normal or in incidental conditions.

Complete pre-conceptual designs of MOSART and MSFR do not yet exist. However, initial candidate technologies have been identified for all required MOSART and MSFR functions. Substantial uncertainty remains in key elements of plant design and technology implementation. Developing a complete conceptual design will provide a unifying framework to enable an integrated set of technology choices to be made. Concept development also identifies technologies that require longer term development (e.g., structural alloy creep testing) that, in turn, assist in determining the overall timeline for development and deployment of the MSR.

The licensing-related design and safety features of liquid fueled MSR combine those of a reactor and a chemical processing plant. The MSR safety approach will have to take into account a fuel in a liquid form within the coolant which depends upon keeping actinides and fission products in solution.

Several categories of active components are needed to be qualified for the operation of MSRs like heat exchangers, pumps, valves. These items face the same thermal and chemical compatibility challenges as the instrumentation equipment. Among these, the heat exchangers are likely to represent the most significant challenges. A heat exchanger is a cold spot in the system and supports a significant heat flux, both factors that can drive corrosion and corrosion product deposition. In addition, the heat exchangers may be required to provide a barrier for tritium diffusion to control tritium releases to the balance-of-plant and the environment. The designs must also allow inspection and be serviceable in the event of plugging of tubes or MS leaking. These challenges are compounded by a desire to maximize the thermal efficiency of the heat exchanger by minimizing its wall thickness and maximizing its surface area.

For all MSRs, the general principle of operation is the same (Serp et al., 2014):

- A fuel circuit where the liquid fuel is located during normal operation and circulates between a first zone where the criticality is reached, and a second zone where the heat of the salt is exchanged with a secondary salt through exchangers. In this second zone, there are also delayed neutrons and residual power on the contrary to classical solid-fuel reactors.
- An intermediate circuit with a cooling salt that will extract heat from the fuel circuit via exchangers and transmit it to a steam generation circuit to produce steam either for electrical use or for other applications. This last circuit is named classically “energy conversion circuit”.
- In the chemical processing unit, several operations can be made with the fuel salt, such as extraction of some soluble fission and corrosion products, control of the salt composition, and injection of fresh fissile and possibly fertile materials necessary to keep the salt liquid without solid accumulation over the operation temperature range and to maintain criticality for the continued operation.

MSRs provide unique challenges in terms of the safety evaluation:

- Compatibility of salts with reactor materials (at high temperatures and radiation damage).
- High melting point.
- Significant quantities of fuel outside the reactor core: heat exchanger, various tanks, pumps, possible associated fuel processing; possible continuous addition/removal of fuel.
- Distributed delayed neutrons (mobile fuel).
- Salt vapor deposition in cover gas lines.
- Potential for larger volumes of high activity components (filters and replaceable components).
- Fuel composition continuously changing.
- Fuel performs cooling function.
- Tritium production (lithium and beryllium containing fuel salts).
- Presence of bubbles (fission product gasses) passing through the core.
- Noble gas fission products evolve out of the salt into cover gas; noble metal fission products plate out onto surfaces; fuel salt retains most other fission products. The fuel salt processing units both for gaseous and soluble fission products including actinides recycling requests a specific safety analysis.
- The fuel salt composition is complex and some products could get their solubility limit leading to uncontrolled deposition in a cold spot or by selective extraction during salt cooling procedures. If this would happen with fissile elements that have low critical masses, it could create criticality accident.
- The notion of barrier remains to be specified, the aim being to avoid any significant radiological releases in any situation.
- The precise operating mode of these reactors remains to be investigated.

For the MOSART and MSFR, which have a fast neutron cores, the criticality in the critical zone of the fuel circuit is a function of the shape of this critical zone and temperature. According to the parameters, the criticality is linked to a precise value of the temperature of the fuel salt. At a higher temperature, the salt is too diluted, inducing lower density of fission matter, then the reactivity goes down inducing a lower production of heat and a decrease of the temperature of the fuel salt. At a lower temperature, the fuel salt is too dense, inducing higher density of fission matter in the critical zone, then the reactivity goes up inducing a higher production of heat and an increase of the temperature of the fuel salt. The higher the feedback effect coefficient, the shorter the response time of the system to return to its equilibrium state. The temperature of the fuel linked to the criticality is called “critical temperature”.

Some differences in design, or even in neutron behavior, will emerge between thermal spectrum and fast spectrum concepts. Thermal flux reactors have a moderator in their criticality zone, such as graphite elements, and are very sensitive to fission products poisoning. A prototype, MSRE, was operated in the United States at ORNL during the 1960s. This is so far the only MSR experience feedback. In addition, fast reactors are less sensitive to fission product poisoning and accept a longer removal times.

Control of reactivity function

Regarding the neutron characteristics, the negative reactivity feedback effect of the MOSART and MSFR, coming from the density effect and the Doppler effect, provides intrinsic reactor stability. Unlike solid-fuel reactors, the high negative density feedback coefficient acts instantaneously in the case where the fuel salt also acts as coolant.

Also in such a circulating-fuel system, the fraction of delayed neutrons is reduced because the fuel motion drifts a part of the delayed neutron precursors to low importance areas. Reactivity insertion up to 1000 pcm ($0.01 \Delta k/k$) has been calculated and results shows that the equilibrium state can be restored without involving a large temperature increase.

This negative and instantaneous temperature feedback effect has several consequences on the MOSART and MSFR designs:

- Control rods may not be necessary; the control of the reactor being carried out by simply controlling the power extraction via heat exchangers. If the cooling increases, the power will increase to maintain the critical temperature. By contrast, if the cooling stops, it will have the opposite effect and the reactor power drops to zero (excluding residual power) without large temperature increase
- The reactor can be started up by progressive filling until the criticality is reached.

A possibility of draining of the salt is necessary for maintaining a sub-critical state. On the other hand, the salt chemistry is complex. The salt contains products with low critical masses, such as plutonium, uranium or other minor actinides. In case of deposition of these products for example in the cold zones of heat exchangers, criticality could occur. It will therefore be necessary to take measures both to avoid this type of accident and to mitigate their consequences.

Decay heat removal function

In the event of a reactor shutdown, the residual power will be distributed throughout the fuel circuit. Good physical properties of the liquid fuel make it possible to organize efficient heat removal based on natural circulation. Various devices must then evacuate it. The fact that the salt does not have any exothermic reaction with air is favorable for cooling by air. Natural circulation of salt, covering gas and air has to be investigated. The gas cooling could be directly done with the fuel salt or through an intermediate salt circuit. Using the air as heat sink requires investigating how the confinement function is achieved at this level.

Containment function

In the MSR, the first barrier is the fuel system, which plays the role of the cladding of a solid fuel. However, certain elements would have to be taken into account in the analysis of by-pass or weak points of this first barrier. This includes:

- Plates or tubes of heat exchangers;
- The pump and the pump shaft;
- Sampling and reintroduction systems for combustible and fertile salts;
- The removable cover in the upper reflector;
- The gas sampling system of the salt treatment unit;
- Valves and lines connecting the fuel system to the storage tanks of combustible and fertile salts;
- The valve and the pipe connecting the cover gas of the fuel circuit;
- Salt drain valves.

The difficulty lies in the definition of the second barrier. This must include the first barrier, in particular the heat exchangers, the draining circuit and allow interventions in case of leakage of the liquid fuel. The draining of the core appears necessary, either for the maintenance and inspection (e.g., replacement of protective baffles of the tank in the critical zone), or in case of leakage to minimize the leak of fuel between the first and the second barrier, or for maintaining the salt in a sub-critical state. The problems

of solid species that could be formed during cooling and or remain during liquefaction of the fuel salt for reinjection will have to be analyzed and specified. In conclusion, this second barrier will be necessary more for operation and maintenance than for safety analysis. The last barrier will be the containment that allows avoiding any radioactive large release.

Proliferation challenges

The reference US ORNL designs of a breeder (MSBR) and denatured (DMSR) MSRs developed in the 1970s clearly illustrate disadvantage and advantages in terms of fuel material attractiveness for the case of U-Th fuel cycle. The low material attractiveness DMSR fuel cycle has been incorporated in some of the new MSR designs. Fuel salt additions in MSRs become part of a homogeneous mixture upon being added to operating fuel, so bred fissile isotopes cannot be readily chemically separated from non-fissile isotopes. Isotopes with higher fission cross sections tend to preferentially burn out lowering the fissile actinide fraction generating deep-burn fuel over time. Development of a lower fissile fraction plutonium isotopic composition tends to happen faster in thermal spectrum reactors because their fissions are dominantly generated from fissile materials whereas the fast spectrum systems consume both fissile and fissionable nuclei.

Today wide variety of MSRs, in which a flowing molten salt mixture based on fluorides or chlorides contains fissile/fertile material serving as fuel and coolant, have begun development around the world (see Table 1). The proposed new MSRs have widely varying fuel cycles to the point that identifying common elements is challenging. Proposed neutron spectra range from thermal to fast and also include time varying spectra. Almost every known form of fissile/fertile material or fuel cycle is under consideration as a fuel source.

Several of the prospective vendors (see Table 1) have indicated that they intend to employ a Th-²³³U breeding equilibrium fuel cycle. Breeding ²³³U from thorium, however, continues to involve creating a separated stream of ²³³Pa that is allowed to decay in a lower thermal flux region. Separated ²³³Pa has high material attractiveness.

Basic information is provided concerning fuel cycle of the MOSART power plant (Ignatiev et al., 2019) integrated with facilities of Experimental Demonstration Centre (EDC) (see Fig. 2). The EDC is under construction in Russia at the site of the Mining and Chemical Combine (MCC) and after 2020 will begin reprocessing of spent nuclear fuel (SNF) from VVER-1000 reactors on the basis of innovative technology, providing a recovered nuclear material (refined products) for recycling in thermal and fast solid fuel reactors. In accordance with the EDC flowsheet, the highly active raffinate, containing long-lived minor actinides, is sent for conditioning. Use of dedicated reactor unit as a TRU burner, remaining after the main part of uranium and plutonium recycling to solid fuel thermal and fast reactors, may reduce the volume and radiotoxicity of HLW. The principle attraction of the MSR technology in this application is the use of fuel material flexibility (easy of “short cooled” fuel preparation, processing and multiple recycling) for gaining additional profits as compared with solid materials.

It is proposed to use the technical and technological capabilities of the MCC site to place single fluid fertile free MOSART system in the immediate vicinity of SNF aqueous reprocessing plant, linking it to the EDC infrastructure. The main design objective of the single fluid 2400 MWt MOSART is to close nuclear fuel cycle for all actinides, including neptunium, plutonium, americium and curium. It is assumed that the fuel cycle of this complex will be organized as follows (see Fig. 1): the bulk of the removed uranium and plutonium return to thermal and fast solid fuel reactors, and the remaining transuranic elements are transferred for utilization in the MOSART system. The co-location of MOSART and SNF reprocessing plant, will provide the MCC site and the surrounding customers by electricity (7.92 TWe hours per year), facilitates the problems of nuclear materials transport and radwaste management. As with fluid fluoride-based fuel the entire fuel element fabrication process is excluded, providing for exceptional flexibility. The fuel can be blended into reactor straightly as needed at any time. Also, there is no need for long cool times and interim storage. This saves for MOSART significant part of the front-end effort (including radioactive doses) and cost. All fresh fuel fluorides containing significant quantities of fissile materials for initial loading and make-up, will be manufactured onsite at the EDC by hydro-fluorination process.

The main advantages of MOSART are the ability to vary widely the minor actinides content in fuel salt without losing the inherent safety and the absence of stages related to the fuel fabrication and re-fabrication in multiple actinides recycling. As result there are significant proliferation resistance and safeguards implications related to the fuel make-up and chemical processing in MOSART plant: (1) there will be continuous variation of isotopic concentrations in the fuel salt from both actinides transmutation and chemical processing; (2) refueling scheme include the ability to continuously feed the core with fresh fissile material; (3) plate-out of noble metals in the primary circuit could complicate inventory tracking.

Fuel salt represents a unique combination of high-temperature and high-radiation environments that will be challenging for diversion as well as measurement techniques and instrumentation: (a) temperature in the reactor or fuel processing plant will always be kept in liquid state within 650–750 °C; and (b) fuel salt will be highly radioactive even outside the primary circuit.

Note, that after each recycling fuel is getting less attractive for fissile material diversion. During 50 years of operation 2400 MWt MOSART can utilize more than 12 t of minor actinides. Last TRU loading will be transferred to the next MOSART unit to be constructed at the MCC site.

In order to avoid nuclear matter diversion, the MOSART reactor plant is integrated (1) at the front end with VVER SNF aqueous reprocessing plant and (2) at the back end with the high temperature fuel salt clean up facility all located at the MCC site. All fresh fuel fluorides containing significant quantities of fissile materials (plutonium plus minor actinides) for initial loading and make-up,

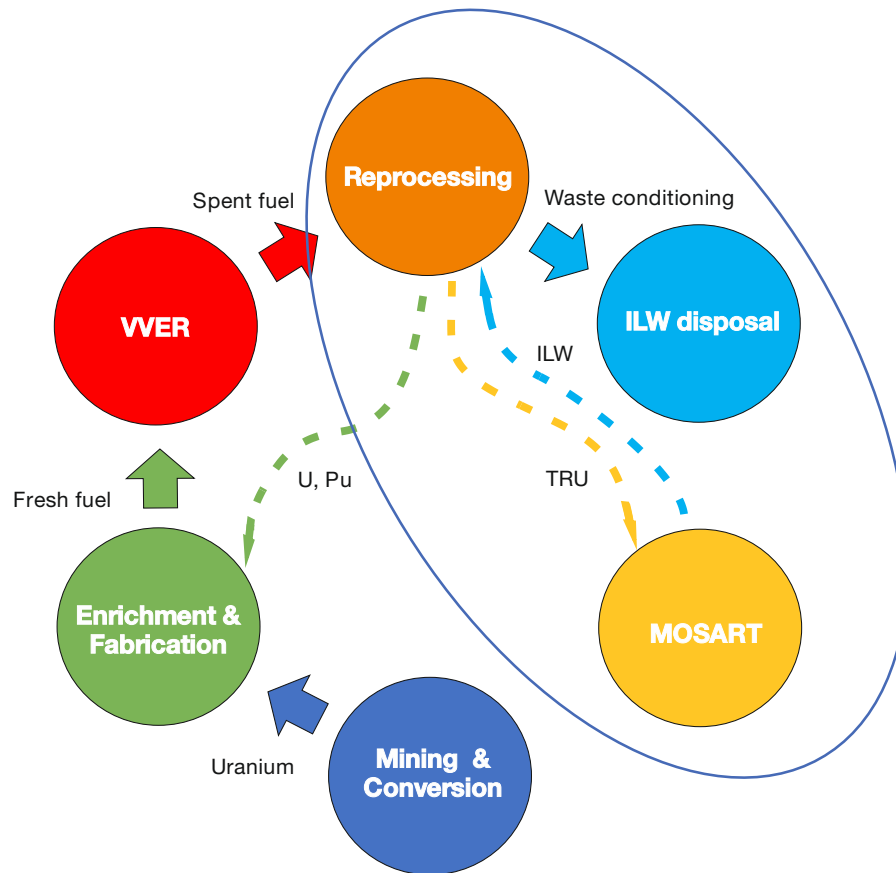


Fig. 2 Nuclear fuel cycle with MOSART at MCC site.

will be manufactured onsite by hydrofluorination process. In molten salt pyroprocessing facility the higher actinides would always accompany the plutonium, this operation would never produce a “clean” material that would be attractive for diversion.

Conclusion

Molten salt fluorides as coolants offer interesting features such as very good transport properties, strong irradiation resistance, high thermal stability and boiling points. They share some advantages with liquid metal coolants like reactor operation at low pressure. This constitutes a significant safety and cost advantage.

The MSR has many advantages compared to other reactor types, like the capability of burning actinides or breeding fissile fuel, continuous processing of spent fuel, no need for fuel manufacturing, safe operation, and a high level of sustainability because MSR can be run as a transmuter of long-lived actinides from used nuclear fuel or as a breeder reactor in the uranium fuel cycle or (much better) in the thorium fuel cycle. World-wide, all these operation modes of the MSR are investigated. The MSRs with liquid fuel can be designed with a neutron spectrum ranging from thermal to fast.

The thermal spectrum approach has a larger historic development foundation and ability to significantly lower neutron flux reaching reactor vessel material compared to fast spectrum MSR. The thermal spectrum designs, like IMSR and ThorCon, open the possibility of simplified converter options, removing the need for fuel processing by employing LEU, on its own or in combination with thorium. Converter MSR designs also allow consideration of SS316 as container material for reactor vessel and carrier salts that is free from the need for enriched lithium 7 and/or beryllium (and their tritium production) as neutron economy is of lower importance for converters.

Over the last years, R&D on MSRs has been mainly focused on fast spectrum concepts with or without thorium support (MSFR and MOSART), which have been recognized within GIF as long term alternative to solid-fueled fast neutron reactors with attractive features (strongly negative reactivity feedback coefficients, smaller fissile inventory, simplified fuel cycle). A negative coolant temperature reactivity coefficient is universally recognized as a desirable safety feature for power reactors. MSFR and MOSART can operate within technical limits using high nickel alloy as container material with different fuel loadings and make-up based on TRUs (from used LWR fuel with MA/TRU ratio up to 0.45) as special actinide transmuter, as self-sustainable system ($CR = 1$) or even as a breeder ($CR > 1$).

The risks of corrosion by the impurities (oxygen, water mainly) dissolved into a molten salt coolant or by the fission products present in fuel salt are a significant issue in MSRs that has been the subject of R&D work since the 1950s. The R&D on advanced

materials for MSR designs are under way now in different countries. Although more experimental and qualification efforts for special materials involved in the MSR design are still needed to demonstrate the viability of MSRs, past experience gained at the ORNL in the 1960s and 1970s remains a source of knowledge that demonstrated many positive aspects of the reactor technology.

The safety analysis methods in their current form cannot be applied to liquid fueled MSRs, in particular due to the fact that the fuel is molten during normal operation. Its outstanding safety level is ensured by the strongly negative reactivity feedback coefficients, even in a fast neutron spectrum, and the capability to drain the liquid fuel into dump tanks, which excludes overheating due to stagnant decay heat removal. A novel methodology for the design and safety evaluations of liquid fueled MSRs is needed, relying on current accepted safety principles such as the principle of the defense-in-depth, the use of multiple barriers and the three basic safety functions: reactivity control, decay heat removal and radioactive products confinement. New methodology applied to the reactor and reprocessing units shall be robust and comprehensive, integrate both deterministic and probabilistic approaches.

MSR designs under considerations, in which a flowing molten salt mixture based on fluorides contains fissile/fertile material serving as fuel and coolant, represent a reactor plant integrated with a pyro processing unit. Existing IAEA PR&PP approaches are mainly applicable to the U-Pu, but not to MA-based and Th-U fuel cycles, and to item counting for solid fuel reactors, as well as to bulk material accountancy for the front and back end of the nuclear fuel cycle. These techniques and associated instrumentation for bulk accountancy have been developed predominantly for enrichment, fuel fabrication, and aqueous reprocessing. However, none of these bulk accountancy measures can be directly applied to high temperature MSR designs, in general, and for MSFR and MOSART in particular, without evaluation and potential modification. Almost no work has been done to determine how will the fissile material content in the fuel salt mixture be determined, when it is in the primary circuit, in drain tanks, or in processing unit.

The various MSR projects have common themes in basic R&D areas, of which the most prominent are the liquid salt technology and materials behavior, the fuel and fuel cycle chemistry and modeling, and the numerical simulation and safety design aspects of the reactor. Research in these areas will not only support the development of this particular reactor and fuel cycle technology, but can find applications in other Generation IV systems as well, like the SFR (Heidet, 2021), LFR (Alemberti, 2021) and VHTR (Fütterer et al., 2021), for example, for the development of alternative fluids for intermediate heat transport, where liquid salts offer potential advantages regarding in particular the high volumetric heat capacity, reduced equipment size, absence of chemical exothermal reactions, intermediate loop and power cycle coolants.

The strong international cooperation between the countries involved, has shown to be very effective in tackling the most prominent remaining R&D issues. However, a continued effort is needed by all parties to converge to a basic reactor design excelling in safety and sustainability around 2025–30.

See Also: Fluoride-Salt-Cooled High-Temperature Reactors; Molten Salt Fuels: Properties, Purification, and Corrosion Control; North Carolina State University PULSTAR Reactor; Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors.

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Super-Critical Water-Cooled Reactor (SCWR)

Yanping Huang,^a Jinguang Zang^a, and Laurence Kim-Hung Leung^{b,*}, ^a Nuclear Power Institute of China, Chengdu, China; and
^b Canadian Nuclear Laboratories, Chalk River, ON, Canada

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Introduction

The Super-Critical Water-cooled Reactor (SCWR) is a high temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point (374 °C, 22.1 MPa). It is evolved from the current fleet of nuclear reactors (such as boiling-water reactors (BWRs), pressurized-water reactors (PWRs) and pressurized heavy-water reactors (PHWRs)), which has accumulated over 50 years of design and operating experience. The balance of plant configurations of SCWRs are based on those of the supercritical water fossil-fuel power plants. This has simplified significantly the development process, which focuses mainly on the reactor core, and could expedite the deployment.

Fig. 1 illustrates two types of SCWR core concepts that are being considered (GIF, 2014): pressure-vessel (PV) type and pressure-tube (PT) type. Other than the reactor core internals, the overall plant configurations are the same for both types.

The flexibility of SCWR designs accommodates traditional (i.e., uranium oxide (UO₂)) and advanced (i.e., thorium and plutonium-uranium mixed oxide (MOX)) fuel cycles with thermal, fast and mixed spectra. Selected parameters for various conceptual designs are summarized in Table 1. All SCWR concepts are designed to operate at reactor outlet temperatures between 500 and 625 °C matching closely the current and advanced supercritical-pressure turbines. This range of reactor outlet temperatures would improve the thermal efficiency ranging 43–48%, compared to 33–35% for the current fleet of nuclear reactors. The UO₂ fuel composition is similar to those used in Gen-III and Gen-III+ nuclear-reactor technologies. A mix of thorium and plutonium oxides, (Pu,Th)O₂, is adopted as the fuel composition for the pressure-tube type SCWR. Light water (H₂O) is used as coolant and moderator for the pressure-vessel type thermal-spectrum SCWR. However, light water is used as coolant while heavy water (D₂O) is adopted as moderator for the pressure-tube type SCWR.

The safety approach is based on that of the Gen III + nuclear-reactor technology but includes additional passive safety systems. This could reduce the core damage frequency for postulated design-basis accidents compared to the current fleet of nuclear reactors. The compact design of the SCWR core and the elimination of certain systems (e.g., steam generators as in PWRs and recirculation

* Retired

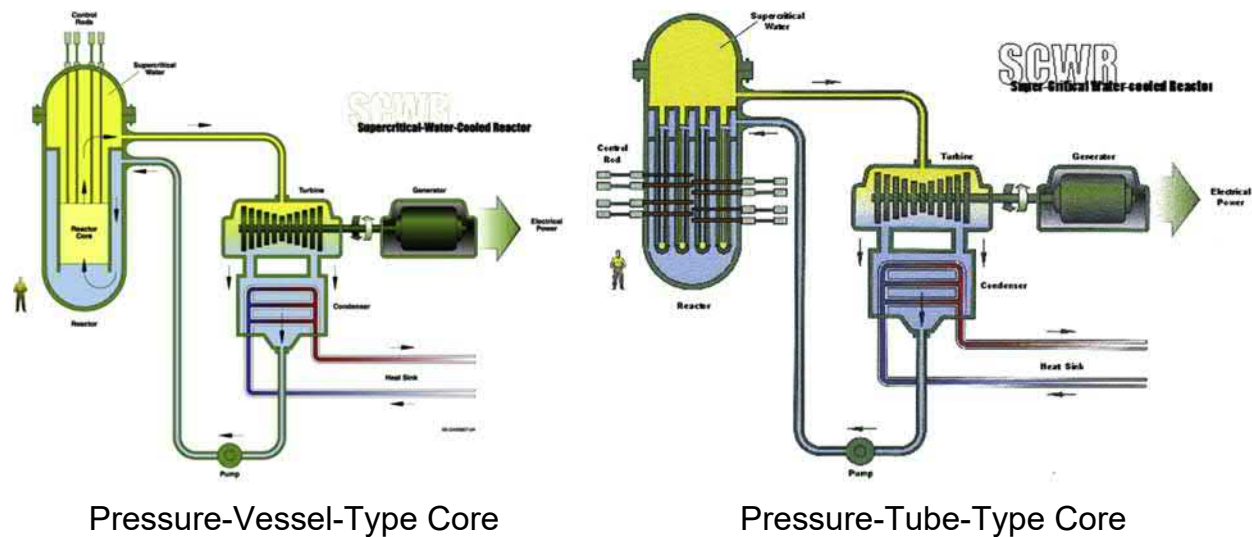


Fig. 1 Schematic diagram of SCWR concepts (GIF, 2014).

Table 1 Key parameters of SCWR concepts.

Type Spectrum	Canada	China		EU	Japan		Russian Federation
	PT Thermal	PV Thermal	PV Mixed	PV Thermal	PV Thermal	PV Fast	PV Fast
Pressure (MPa)	25	25	25	25	25	25	24.5
Inlet Temp. (°C)	350	280	280	280	290	280	290
Outlet Temp. (°C)	625	500	510	500	510	501	540
Thermal Power (MW)	2540	2300	3800	2300	4039	1602	3830
Efficiency (%)	48	43	44	43.5	43	44	45
Active Core Height (m)	5	4.2	4.5	4.2	4.2	2.4	4.07
Fuel	(Pu,Th)O ₂	UO ₂	UO ₂ /MOX	UO ₂	UO ₂	MOX	MOX
Burnup (GWd/t HM)	60	45	NA	60	45.3	53.8	60
Moderator	D ₂ O	H ₂ O	H ₂ O/none ^a	H ₂ O	H ₂ O	None/ZrH ^b	none
# of Flow Passes	1	2	2	3	1 and 2	2	1 and 2
Blanket	No	No	Yes	No	No	Yes	Yes

^aLight water in the thermal zone and no moderator is needed in the fast zone.

^bNo moderator in the seed fuel and Zirconium Hydride in the blanket fuel.

pumps and steam separators/dryers as in BWRs) have led to reductions in the footprint and containment size. This has positive implications for protection against external events in addition to the reduction in capital cost.

The main improvement in the SCWR is in the area of economics due to its high thermal efficiency and plant-design simplifications that result from the use of supercritical (SC) water as coolant (GIF, 2014). Furthermore, the safety characteristics of the SCWR have also been advanced through the introduction of additional passive safety systems. Depending on the concept configuration, the aspect of proliferation resistance and physical protection (PR&PP) has also been improved.

Historical developments

The SCWR development can be divided into two stages. The first stage covered earliest proposals of the SCWR concept in 1950s and 1960s (Oka et al., 2010; Schulenberg et al., 2014). These proposals included a light-water moderated, supercritical-steam-cooled reactor (which was designed at Westinghouse in 1957) with 538 °C outlet temperature to generate 70 MW thermal power, a heavy-water-moderated, light-water cooled reactor (which was designed at General Electric in 1959) to generate thermal power of 300 MW, a graphite moderated, light-water-cooled pressure-tube reactor (which was designed at Westinghouse in 1962) to generate electric power of 1000 MW, and a light water cooled and moderated pressurized water reactor (which was proposed at Westinghouse in 1966) at supercritical pressure to generate 800 MW electric power. In addition, a supercritical water cooled

in-pile fuel testing loop was constructed and installed in the Saxton Research Reactor for irradiating collapsed clad fuel elements in a reactor environment but the program was suspended in April 1965 after a shutdown of the loop (Cohen, 1968).

The second stage was initiated as a means to improve fuel utilization and reduce capital cost (Oka et al., 2010). Many countries participated in the development after the SCWR was selected as one of six advanced nuclear-reactor systems for collaborative R&D in the Generation-IV International Forum (GIF) (GIF, 2014). The first SCWR development proposal was introduced in 1986 at USSR for the design of a two-circuit reactor plant (VVER-SCP-I) to generate electric power of 500 MW (Ryzhov et al., 2011). A thermal neutron spectrum and later an alternative concept with a fast neutron spectrum, light water cooled, pressure vessel type SCWRs were proposed in the late 90s at the University of Tokyo (Oka et al., 2010). From 1999 to around 2006, a multi-year program was initiated to support the development of the SCWR in USA (Buongiorno and MacDonald, 2003). The European Commission supported a consortium of European research institutes and industrial partners to develop a pressure-vessel type SCWR, which is referred to as the High Performance Light Water Reactor (or HPLWR) in 2000 (Schulenberg and Starflinger, 2012). In 2006, Research and Development Institute of Power Engineering (RDIPE) in the Russian Federation developed a concept of supercritical water-cooled and graphite-moderated power reactors (VGERs) operating at the pressure of 25 MPa and outlet temperature of 550 °C (Ryzhov et al., 2011). Natural Resources Canada (NRCan) established a national program (Generation-IV National Program) in 2006 to support Generation-IV R&D and develop a SCWR concept based on the pressure-tube configuration (Leung et al., 2018a). In 2007, Shanghai Jiao Tong University along with other institutes performed the basic research of SCW and proposed a mixed-spectrum SCWR core concept (Cheng et al., 2007). In 2012, a thermal-spectrum SCWR conceptual design, referred to as the CSR1000, was established at Nuclear Power Institute of China (IAEA, 2015).

Thermodynamic cycle

The thermodynamic cycle of a SCWR is basically the same as that of a supercritical fossil plant (IAEA, 2014). Superheated steam at a pressure of 24–25 MPa and a temperature of 500 °C or more from the reactor outlet is supplied directly to the High Pressure turbine (HP), then to the Intermediate Pressure turbine (IP) and afterward to the Low Pressure turbine (LP). The internal energy of the superheated steam is converted to kinetic energy of turbine rotor and then to electric power through the Generator (G). A moisture separator and reheater is installed between the high pressure and the intermediate-pressure turbines or between the intermediate pressure and the low-pressure turbines, depending on the steam quality, to reduce the moisture content before entering the downstream turbine. Steam exhausted out of the low-pressure turbine at low pressure (~abs 5 kPa) and temperature (~33 °C) is further cooled in the condenser. The condensate pressure is increased to 25 MPa by the Condensate Extraction Pump (CEP) and the Feedwater Pump (FP). Several preheaters (PHs) with turbine steam extractions are introduced to heat up the water temperature to the designed value ranging from 280–350 °C. This heat recovery process increases the reactor inlet temperature and thus the reactor average temperature and thermal efficiency. A general sketch of the SCWR steam-cycle concept is shown in Fig. 2 to illustrate the once-through design principle.

Thermodynamic cycle designs for SCWR concepts are similar to each other. As mentioned above, the location of the moisture separator and reheaters could be different. Furthermore, seven (three low pressure and three high-pressure preheaters with the deaerator in between), eight (four low pressure and three high-pressure preheaters with the deaerator in between) or nine (four low pressure and four high-pressure preheaters with the deaerator in between) heat recovery stages could be implemented (IAEA, 2014).

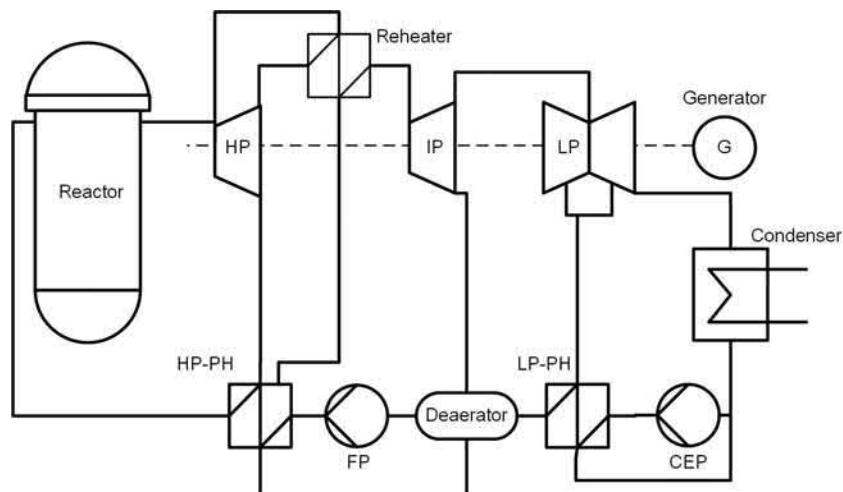


Fig. 2 Simplified energy conversion system for typical SCWR designs.

SCWR core concepts

The SCWR thermal-spectrum core concepts have been based mainly on the pressure-vessel configuration (Schulenberg and Leung, 2016). However, the one developed in Canada is based on the pressure-tube configuration. Coolant is circulated from the bottom to the top of fuel assemblies in a single path for both Japan's and Canada's concepts (Schulenberg et al., 2014). It passes through two zones of fuel assemblies (two-path system) in China's concept (IAEA, 2015) and three zones (three-path system) in EU's concept (Schulenberg and Starflinger, 2012). Fig. 3 illustrates the thermal-spectrum core configurations (Leung et al., 2018b).

Canada's concept consists of an inlet plenum, where the coolant enters into the core and distributes to the fuel channels, and an outlet header, which collects the high-temperature coolant from the fuel channels and directs to the high-pressure turbine (Schulenberg and Leung, 2016). The fuel channels are submerged within the low-pressure heavy-water moderator in the calandria vessel. Control and shutdown rods are inserted from the side of the calandria vessel into the low-pressure moderator.

Both China's and EU's core concepts have a similar configuration, except that China's core is divided into two zones (IAEA, 2015), while the EU's core into three zones (Schulenberg and Starflinger, 2012). Coolant enters the vessel from the inlet nozzles and is divided into two streams. In China's concept, the majority of coolant travels up the passage besides the vessel wall and down through the first zone of fuel assemblies (IAEA, 2015). It mixes with the remaining coolant in the lower plenum and travels up through the second zone of fuel assemblies. The remaining coolant entering the vessel travels down to the lower plenum and mixes with the bulk coolant. In EU's concept, most of the coolant travels down the passage beside the vessel wall and up through the first zone of fuel assemblies (Schulenberg and Starflinger, 2012). The remaining coolant flows up another passage to the top of the vessel, mixes with the coolant through the first zone, down through the second zone and up through the third zone of fuel assemblies. The high-temperature coolant collected at the outlet header is directed to the high-pressure turbine. Control and shutdown rods are inserted from the top of the vessel into the core.

Japan's core concept is similar to that of a boiling water reactor (Yamada et al., 2011). Coolant enters the vessel from the inlet nozzles and is divided into two streams. Most of the coolant travels down the passage beside the vessel wall. It mixes with the

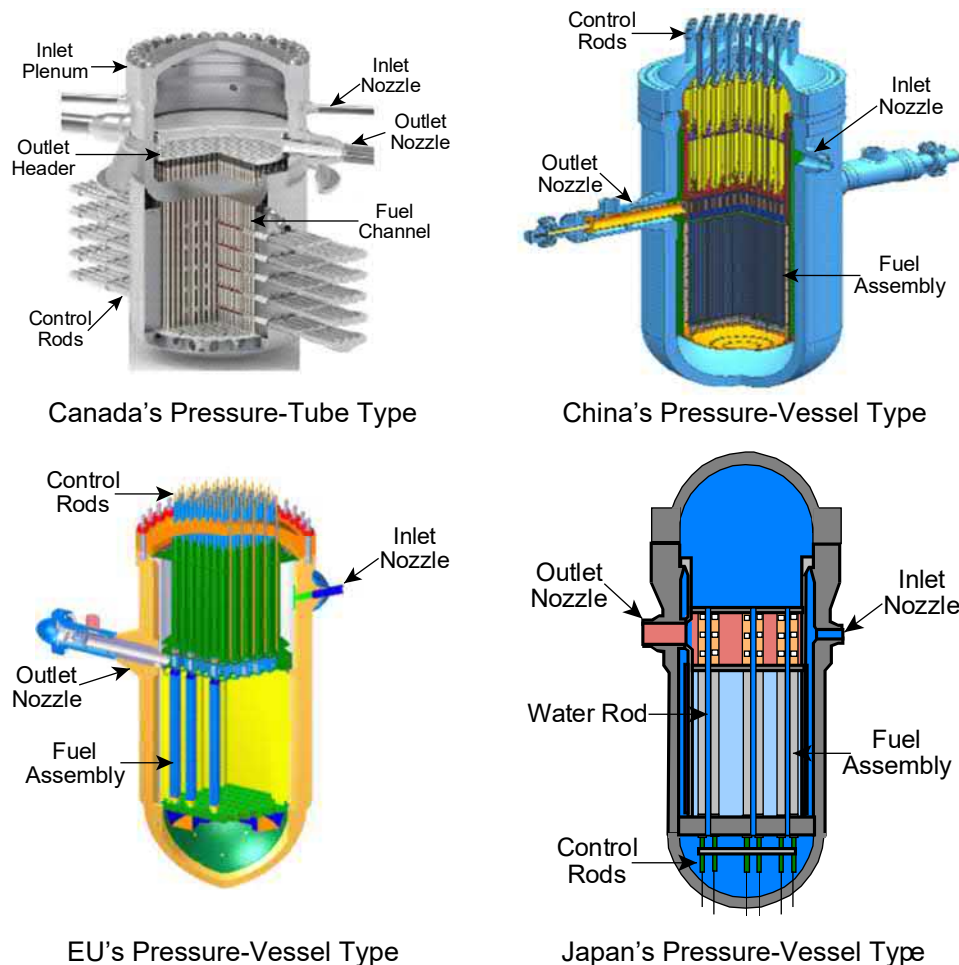


Fig. 3 Thermal-spectrum SCWR core configurations (Leung et al., 2018b).

remaining coolant at the lower plenum and travels up through the fuel assemblies. The remaining coolant entering the vessel flows up another passage to the top of the vessel and then down to the bottom of the vessel along the wall mixing with the bulk coolant. It cools the vessel wall and maintains the wall at low temperatures. The high-temperature coolant exiting the fuel assemblies is collected in the outlet header and is directed to the high-pressure turbine. Control and shutdown rods are inserted from the bottom of the vessel into the core. There is an option for a two-path core configuration, which is similar to China's SCWR concept.

The flexibility of SCWRs facilitates the development of mixed- and fast-spectrum core concepts, as illustrated in Fig. 4, other than thermal-spectrum core concepts (Schulenberg and Leung, 2016). A mixed-spectrum SCWR core concept is being developed at the Shanghai Jiao Tong University in China (Cheng et al., 2007). Its configuration is similar to that of China's thermal-spectrum core. The first zone of fuel assemblies is the thermal-spectrum region while the second zone is the fast-spectrum region. Control and shutdown rods are also inserted from the top of the vessel to the core.

The configuration of Japan's fast-spectrum core concept differs from that of the thermal-spectrum core concept (Oka et al., 2010). It is separated into two zones with a two-path flow pattern (similar to that of China's thermal-spectrum core configuration). The first zone consists of separated seed and blanket fuel assemblies with the downward coolant flow, while the second zone consists of seed fuel assemblies with the upward coolant flow. Control and shutdown rods are inserted through the top of the vessel into the core (unlike Japan's thermal-spectrum core where the rods are inserted through the bottom of the vessel).

Russian Federation's fast-spectrum core concept has the option of adopting the single-path or two-path flow pattern (Ryzhov et al., 2011; Glebov et al., 2015). For the single-path core concept, the coolant entering the core travels to the top of the vessel and then directs to the bottom of the vessel. It flows upwards through the fuel assemblies and discharges to the high-pressure turbine. For the two-path core concept, the coolant entering the core is separated into two streams. The majority of the coolant travels up to the top of the vessel and then downward through the outer zone of fuel assemblies. The remaining coolant flows down through the passage beside the vessel wall. It mixes with the bulk flow and travels upward through the inner zone of fuel assemblies.

SCWR fuel assembly concepts

Fuel concepts for thermal-spectrum cores of the pressure-vessel type of SCWRs resemble closely to those of light water reactors. Fig. 5 illustrates the fuel assemblies for thermal-spectrum cores of SCWR (Schulenberg and Leung, 2016). Uranium fuel with various levels of enrichment and neutron absorber has been included in the pellets. Differences are mainly in the spacing devices and the introduction of a water rod at the central region of the fuel assembly to enhance the moderation of the coolant at high pressure and high-temperature conditions.

The fuel concept for Canada's SCWR consists of a "flask"-like structure that contains the fuel assembly and is connected to the outlet header (Schulenberg and Leung, 2016). Several nozzles are introduced for the coolant entering into the structure. These nozzles also serve as orifices to control the flow rate matching the power generation in the channel (another set of openings are also installed at the top of the pressure tube). The fuel assembly resembles to fuel bundles of the heavy-water reactors with two rings of 32 fuel rods and an active length of 5 m. Spacing between fuel rods is maintained with a wire-wrapped spacer. A central flow tube is installed for the coolant to travel down from the nozzles to the bottom of the fuel channel. Furthermore, this improves the moderation for the inner-ring rods resulting in a balanced radial power profile. The coolant travels upward through the fuel

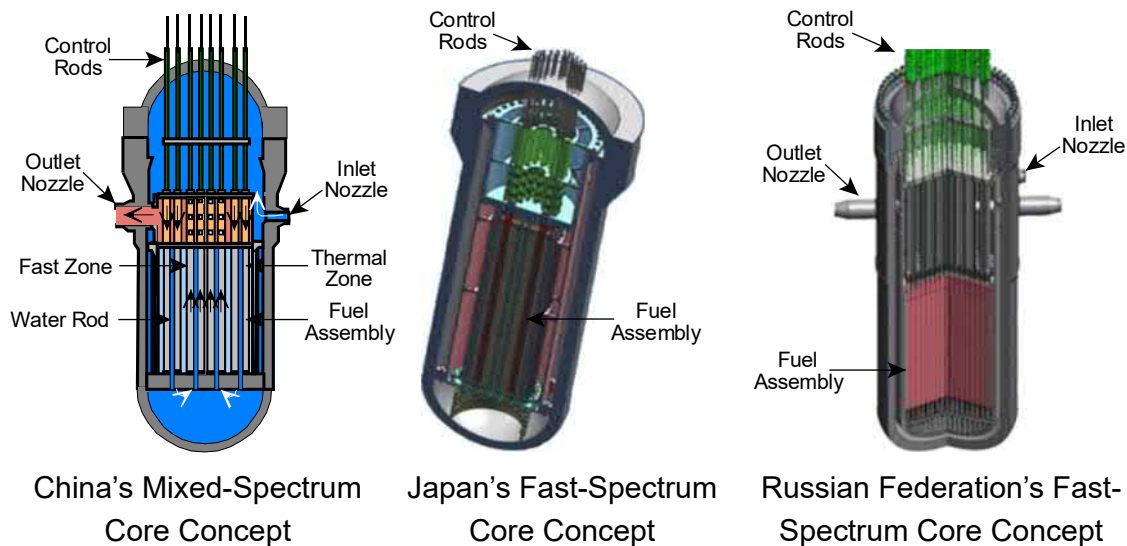


Fig. 4 Mixed- and fast-spectrum SCWR core configurations (Leung et al., 2018b).

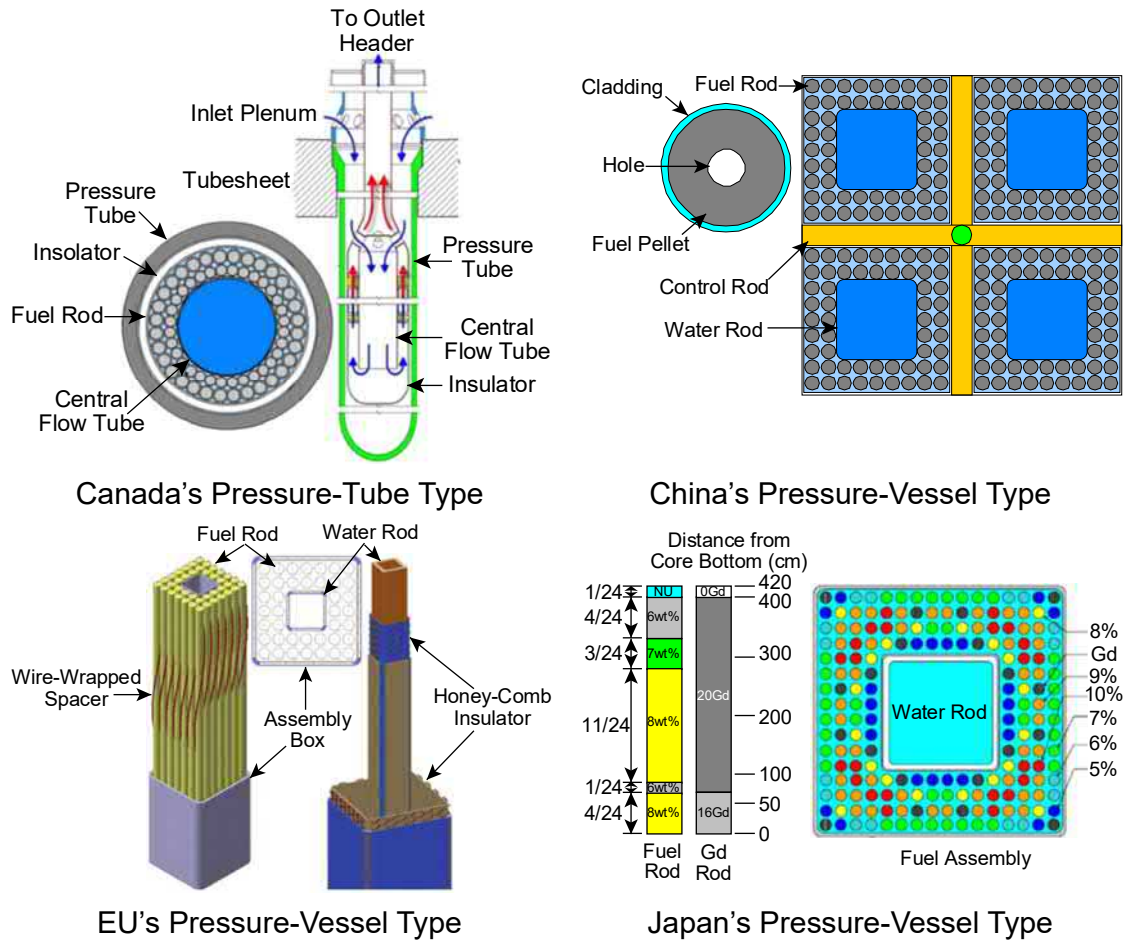


Fig. 5 Thermal-spectrum SCWR fuel concepts (Schulenberg and Leung, 2016).

assembly to the outlet header, and is discharged to the high-pressure turbine. Pellets inside the fuel element consist of a mixture of thorium and plutonium (15 wt% on average).

The fuel-assembly concept of China's thermal-spectrum SCWR is configured into a square array of 56 fuel rods (IAEA, 2015). Wire-wrapped spacers are introduced to maintain the gap size between fuel rods. A water rod is installed in the central region to increase the moderation. Pellets in the fuel rod consist of enriched uranium and are similar to those of light-water reactors. Each pellet has a central hole to reduce the fuel centerline temperature. Four assemblies are grouped together with a boiling-water-reactor type of control-rod configuration.

The fuel assembly concept of EU's thermal-spectrum SCWR has a square-array configuration with 40 fuel rods (Schulenberg and Starflinger, 2012). Wire-wrapped spacers are also used to maintain the gap size between fuel rods. A water rod is installed in the central region to increase the moderation. Pellets in the fuel rod consist of enriched uranium and are similar to those of light-water reactors. Nine assemblies are grouped together with common head and foot pieces.

The fuel assembly concept of Japan's thermal-spectrum SCWR is the largest among all concepts (Yamada et al., 2011). It consists of 192 fuel rods and resembles closely to that of the boiling water reactor. Grid spacers are used to maintain the gap size between fuel rods. A water rod is installed in the central region to increase the moderation. Pellets in the fuel rod consist of uranium fuel of various levels of enrichment. Several rods contain only neutron absorbers for initial reactivity suppression. Four assemblies are grouped together with a boiling-water-reactor type of control-rod configuration.

Fuel assembly concepts for mixed- and fast-spectrum SCWRs are illustrated in Fig. 6 (Schulenberg and Leung, 2016). Two separate fuel assemblies are proposed for China's mixed-spectrum SCWR; one for the thermal zone and the other for the fast zone (Cheng et al., 2007). The fuel assembly concept for the thermal zone consists of 180 fuel rods with nine water rods distributed within the assembly. Wire-wrapped spacers are used to maintain the gap size between fuel rods. Pellets in the fuel rod consist of enriched uranium. Three grades of enriched uranium pellets are inserted at different levels of the fuel rod (i.e., lower grades at the top where the coolant temperature is the highest and higher grades at the bottom). The fuel assembly concept for the fast zone consists of 324 fuel rods. Wire-wrapped spacers are used to maintain the gap size between fuel rods. Each fuel rod contains both the seed and blanket pellets wafered in different sections. Mixed oxide fuel is used for the seed pellet and depleted uranium fuel for the blanket pellet.

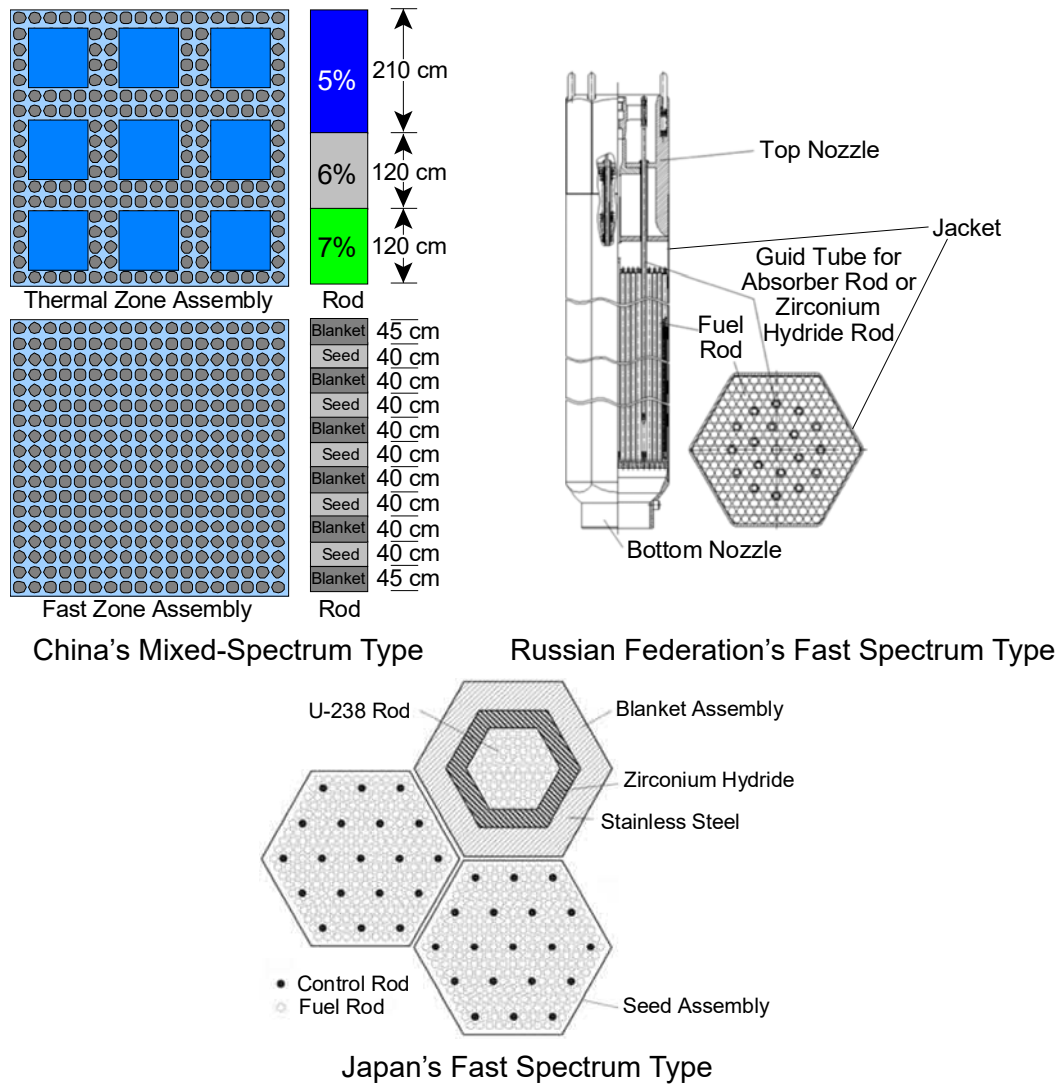


Fig. 6 Mixed- and fast-spectrum SCWR fuel concepts (Schulenberg and Leung, 2016).

Two separate fuel-assembly types are introduced for Japan's fast-spectrum SCWR: seed assemblies each with 252 fuel rods and blanket assemblies each with 127 fuel rods (Oka et al., 2010). These fuel assemblies are configured in hexagonal arrays. Mixed oxide fuel and stainless-steel cladding are used for seed fuel rods, and depleted uranium and stainless-steel cladding are used for blanket fuel rods. In the blanket assembly, the fuel rod region is surrounded by a solid moderator (Zirconia Hydride layer) so that fast neutrons coming from the seed fuel are slowed down and are absorbed by the blanket fuel without causing fast fissions. It enables the fast-spectrum SCWR to have a negative void reactivity without adopting a flat core shape or additional devices. Nineteen control rods are inserted into each seed fuel assembly.

The fuel assembly concept for Russian Federation's fast-spectrum SCWR is configured into a similar array to that of Japan's seed fuel assembly (i.e., 252 fuel rod in a hexagonal array) (Ryzhov et al., 2011). However, only one fuel assembly is adopted with seed and blanket fuel layers wafered in each fuel rod. Mixed-oxide fuel is used for the seed layers and depleted uranium with zirconium hydride is used for the blanket layers. Nineteen absorber or zirconium hydride rods are inserted into the fuel assembly.

SCWR safety system concepts

Safety-system concepts of SCWRs are evolved from the Gen-III and Gen-III+ nuclear reactor systems. Additional passive safety systems have been incorporated to enhance the safety characteristics. Deterministic and probabilistic safety analyses were performed for the Canadian SCWR, HPLWR and JSCWR. Schematic diagrams of SCWR containment and safety system concepts are shown in Fig. 7.

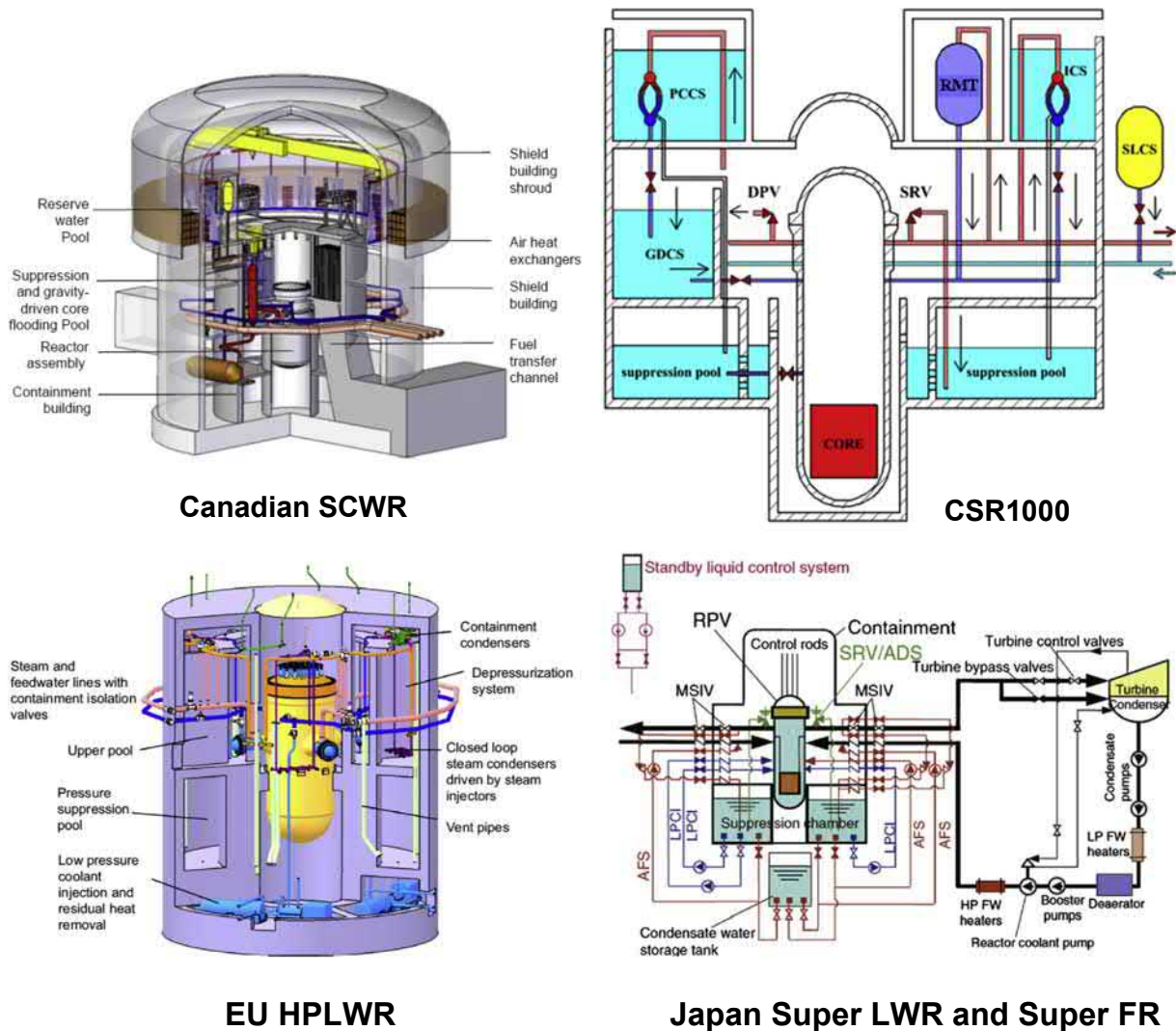


Fig. 7 SCWR containment and safety systems.

Main components of the Canadian SCWR safety system consist of the containment pool, reserve water pool, isolation condensers and passive cooling systems (Schulenberg and Leung, 2016). The containment pool is an annular-shape tank that is located in the containment building above the reactor. Steam condensers and passive autocatalytic recombiners are housed in a condenser gallery located above the pool liquid level. The condenser gallery floor has a series of drains equipped with suppression nozzles, discharging into the containment pool below the liquid level. The containment pool is also used as the pressure suppression pool and the water source for the gravity-driven core flooding system. The reserve water pool serves as a buffer between the passive safety systems and the ultimate heat sink. It is located in the upper section of the shield building and occupies an annular space against the building outer wall. The pool is divided into two sections, each section housing one train of isolation condensers (ICs) and the passive moderator-cooling system (PMCS). Each train of the IC system consists of a piping loop running from the reactor outlet, to heat exchangers and returning to the reactor inlet. The IC relies on the difference of densities between the IC hot leg and cold leg to initiate and maintain a gravity-driven circulation. The PMCS consists of a piping loop running from the reactor calandria to heat exchangers located in the reserve water pool and returning to the calandria. The PMCS uses a flashing-driven natural circulation loop to remove heat from the moderator transferred from the cladding through radiation. The atmospheric air heat exchangers are designed to reject heat from the reserve water pool to the atmosphere which serves to extend the period of time in which the water pool can function as a heat sink before intervention under a high core decay heat regime.

The safety system of CSR1000 adopts the idea of passive and active measures both in PWRs and BWRs (IAEA, 2015). It includes five main systems: High-Pressure Injection System (HPIS), Automatic Depressurization System (ADS), Passive Core Cooling System (PXS), Pressure Suppression Type Containment System (PCS) and Reactor Pit Injection System (PIS). The high-pressure injection system consists of two high pressure feedwater tanks (RMT), each of which is connected to the reactor coolant circuit loop. Each high-pressure feedwater tank (RMT) is filled with high-pressure water and is located above the reactor core. The Automatic

Depressurization System composes of a series of Safety Relief Valves (SRV) and Depressurization Valves (DPV). The SRV are used to protect the circuit from over-pressurization and DPVs are used to accelerate the depressurization process from supercritical pressure to ambient pressure. The Passive Core Cooling System (PXS) composes of the Passive Residual Heat Removal (PRHR) subsystem and passive reactor injection subsystem. The PRHR subsystem removes the residual decay heat from the reactor to the condensation pool through a passive heat exchanger (PRHR HX). The passive reactor injection subsystem connects the condensation pool directly to the reactor with a passive reactor injection line. When the reactor core system (RCS) depressurizes to the atmospheric pressure, it could provide enough water for cooling the reactor. The CSR1000 containment is of pressure suppression type composed of reactor pit, wet well and dry well. The reactor pit is used for containing the reactor vessel. The dry well is used for containing the RCS components and some safety system components. The wet well has a large condensation pool for suppressing the containment pressure when the steam is discharged from the RCS to the containment.

The HPLWR compact containment contains the reactor pressure vessel, an annular pressure suppression pool, four upper pools and a drywell (Schulenberg and Starflinger, 2012). Four feedwater lines with check valves and four steam lines with containment isolation valves, each inside and outside of the containment, connect the reactor with the steam cycle. Four automatic depressurization systems (ADSSs), each equipped with two safety relief valves and two depressurization valves, open to eight spargers in the upper pools. Four redundant and independent low-pressure coolant injection pumps supply coolant from the pressure suppression pool via a heat exchanger and a check valve to the feed-water line for the purpose of residual heat removal. Overflow pipes from the upper pools to the pressure suppression pool close the coolant loop inside the containment. Sixteen vent tubes connect the drywell with the pressure suppression pool for pressure suppression in case of loss-of-coolant accidents (LOCAs), as well as a discharge of hydrogen from the drywell to wetwell. The pressure suppression pool can also be vented to the stack through aerosol and iodine filters. Four emergency condensers are connected with the four steam lines and the four feed-water lines hanging from the top in the upper pools. In addition, four containment condensers are mounted at the ceiling of the drywell, with the secondary side permanently open connected to pools above the containment, to condense the steam in the containment as soon as possible in the unlikely case. A boron poisoning system on top of the containment with a tank of Boron-10 is connected with the feed-water lines which serve as the second, redundant shut-down systems.

The unique feature of the Super LWR and Super fast reactor (FR) of Japan is that depressurization is used to provide a flow to cool the core under accident conditions (Oka et al., 2010). Turbine-driven auxiliary feed water systems and motor-driven low-pressure core injection (LPCI) system are applied on the cold-leg side to provide the function of keeping the coolant supply under abnormal conditions. Safety relief valves (SRVs) are applied on the hot-leg side to provide functions of keeping the coolant outlet open and mitigating over-pressurization. The SRVs also have a function of automatic depressurization to induce effective coolant flow. For reactor shutdown, the control rods and standby liquid control system (SLCS) are employed as in boiling water reactors.

SCWR plant concepts

Up till now, Canada, Euratom and Japan have successfully completed the development of their SCWR plant concepts, which were reviewed by international peers (Leung et al., 2018b). Fig. 8 shows the proposed SCWR plant concepts. Both Canada's and Euratom's core concepts are developed for the thermal spectrum. On the other hand, Japan has developed a thermal spectrum and a fast-spectrum cores using the same plant concept. China and the Russian Federation are continuing their development of their pressure-vessel type of core concepts. China is focusing on the thermal-spectrum core concept while the Russian Federation is working on a fast-spectrum core concept.

Improvements in economics, sustainability, safety and proliferation resistance for SCWRs

The GIF has set six technology goals for the Gen-IV nuclear reactor systems: improved economics, enhanced safety and reliability, improved sustainability, and enhanced proliferation resistance and physical protection (GIF, 2014). SCWRs have the potential of meeting most, if not all, these technology goals.

Economics

SCWRs are designed to operate at high pressures and high temperatures, which have led to increase in thermal efficiency. Compared with the current fleet of water-cooled reactors where the average plant efficiency is around 33%, the plant efficiency of SCWRs could achieve 43–49%. The increase in thermal efficiency would improve the fuel utilization, reduce waste heat and minimize spent fuel storage (Duffey and Leung, 2010).

In addition, the capital cost of SCWRs is reduced from the plant simplification and size reduction. The high operating temperature for SCWRs eliminates the need of inner circulation pumps, steam separators and dryers as in BWRs. On the other hand, the adoption of a direct thermal cycle for SCWRs eliminates the need of steam generators, pressurizer and main circulation pumps as in PWRs. Furthermore, the significant increase in enthalpy through the core and low mass flow rate of the system have led to reduction in sizing of equipment, piping, containment and reactor building. Economic analyses have shown reductions in capital cost by up to 20% for SCWRs compared to the Advanced BWRs (Schulenberg and Starflinger, 2012).

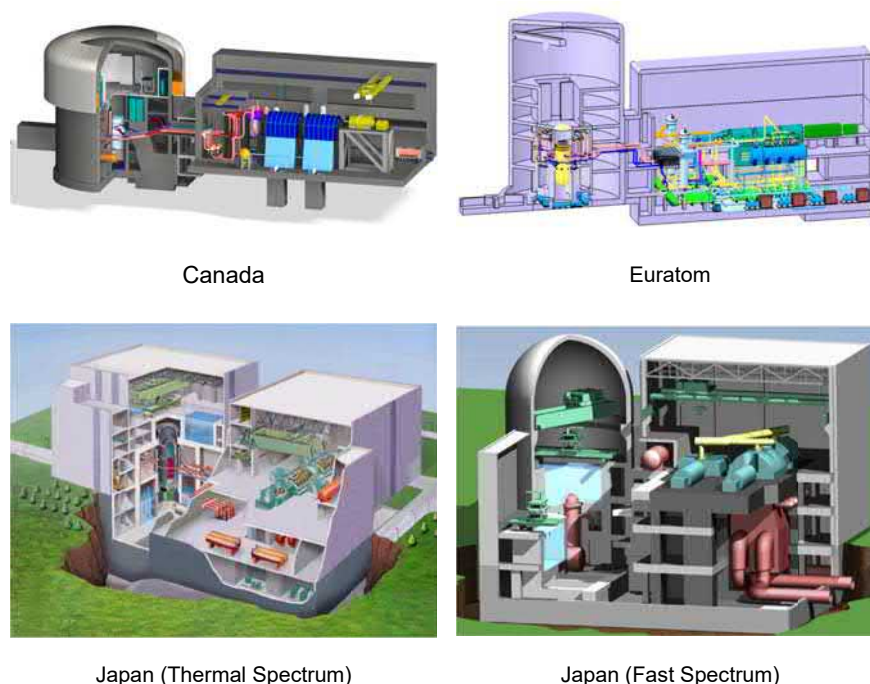


Fig. 8 SCWR plant concepts (Leung et al., 2018b).

Sustainability

Most thermal-spectrum SCWRs adopt the once-through uranium fuel cycle. As mentioned above, the improvement in thermal efficiency would improve the fuel utilization and enhance their sustainability. The introduction of the thorium fuel cycle for the Canadian SCWR would enhance further the sustainability as uranium fuel is no longer needed (Leung et al., 2012). Since thorium is fertile, the reactor-grade plutonium is used as the igniter and to breed U-233.

Fast-spectrum SCWRs (i.e., the Super FR of Japan and VVER-SCP of the Russian Federation) contain blanket regions in the fuel. The plutonium bred into these blanket regions (a breeding ratio of 1.021) contributes to the overall reactor power during its time in the core. Thermal-spectrum SCWRs do not rely on breeding fissile material in separate regions. However, plutonium is bred in the UO_2 fueled SCWRs and U-233 in the thorium fueled Canadian SCWR. The recycling of plutonium from the thermal and fast reactors is foreseen after prototype reactors have been demonstrated. The recycling of U-233 from the Canadian SCWR back into fresh fuel, eventually displacing plutonium as the starting fissile component, has been extensively studied in the context of other thorium-fueled heavy water reactors (Hyland et al., 2009).

Safety and reliability

The introduction of additional passive safety systems (such as the passive containment cooling system in the pressure-vessel type SCWR and the passive moderator-cooling system in the pressure-tube type SCWR) has improved the safety characteristics of SCWRs (Oka et al., 2010; Schulenberg and Starflinger, 2012; Leung et al., 2012). In addition, the avoidance of the critical heat flux phenomena due to non-existence of phase change of water at supercritical pressures has eliminated the sharp rise in cladding temperature in cases of the loss-of-reactivity-control or small-break loss-of-coolant accident scenarios. The stainless steel applied for fuel cladding reduces risks of hydrogen explosion. Moreover, simplification of systems and elimination of components, such as steam generators and primary heat-transport pumps in PWRs or recirculation pumps and steam separator and dryers in BWRs greatly reduce the possibilities of system and equipment failures causing abnormal conditions. That would enhance the reliability of the SCWR plants.

Deterministic safety analyses have demonstrated the maintenance of cladding and fuel integrities for postulated accident scenarios. In addition, preliminary probabilistic safety analyses have shown reductions in core damage frequency for postulated accident scenarios in SCWRs compared to current fleet of nuclear reactor systems (Oka et al., 2010; Leung et al., 2012).

Proliferation resistance and physical protection

The proliferation resistance aspect of SCWRs with UO_2 fuel are similar to that of existing LWRs or CANDU reactors. Enhanced safeguards (such as virtual monitoring) have been incorporated in the plant design phase. The use of thorium fuel in the pressure-tube

type SCWR would improve the proliferation resistance characteristics as thorium is fertile, lesser amount of plutonium is produced, separating U-233 from U-238 is difficult, and the production of U-232 (a high gamma-emitting decay chain) deters extraction (Leung et al., 2012).

The reduction in SCWR plant footprint and sizes of containment and reactor building due to the elimination of components required in PWRs or BWRs would reduce the areas to be monitored and surveyed enhancing the physical protection aspect.

Challenges

Technical challenges were encountered over the course of the SCWR development. These challenges arose mainly from the high core outlet temperature and the high enthalpy rise of the coolant in the core compared to current water-cooled reactors. Key challenges have been identified in six main technology areas.

System design and neutronic

- The SCWR core concepts are more complex than those of conventional LWRs to accommodate the high enthalpy rise. Multiple flow passes have been introduced to mitigate this challenge. This requires an internal separation of regions in the core further complicating the design and manufacturing processes.
- Water at high temperatures has lower moderation than that at low temperatures. Separate flow passes within the fuel assembly are needed to increase the moderation. This has increased the complexity of the fuel assembly design.

Materials and chemistry

- Materials for most core components have been identified. However, the selection of a cladding material to withstand elevated temperatures (above 600 °C) remains a significant challenge (Guzonas et al., 2018). Several potential material candidates have been identified and additional R&D are required to confirm their applicability.
- Operation under SCW conditions introduces unique water chemistry challenges related to water radiolysis and corrosion product transport (Guzonas et al., 2018). A chemistry control strategy is required to mitigate these challenges on cladding-material selection and inspection and maintenance of components.

Thermal-hydraulics and safety

- The large coolant density variation within the core could lead to instability and subsequently large neutronic variation and high fuel cladding temperature.
- Proven safety systems known from advanced boiling water reactors have been adopted in SCWR plant concepts. Due to high operating pressures and low water inventory, the pressure decreasing time scale in SCWR is shorter for LOCAs than that in conventional LWRs. This poses increased demand for the engineered safety systems.

Future plans

Development and experimental plans have been established to improve the current conceptual SCWR designs and address identified challenges. All designers recognize that a collaborative effort is required to fulfill these plans. The ultimate goal from this collaboration is to complete the design and operation of a demonstration plant. Specific areas of interest are summarized below.

System designs

Designers in Canada, European Union and Japan have completed their conceptual design of the thermal-spectrum SCWR plant (Leung et al., 2018b). Those in China are focusing on completing their reference design of the thermal-spectrum SCWR plant (i.e., CSR1000), and are not planning to pursue the mix-spectrum SCWR concept. Designers in Japan and the Russian Federation are continuing their work on the fast-spectrum SCWR concepts.

The modular design of SCWR can accommodate sizing for local deployment needs. Small modular SCWR concepts are being developed for electricity generation from 15 to 350 MW (Yetisir et al., 2012; Novikova and Bogoslovskaya, 2017).

Materials and chemistry

Material research aims to finalize the selection for fuel cladding. Testing of material candidates will be continued to improve the understanding of their corrosion and cracking behaviors. A round-robin testing of stress-corrosion cracking for selected material candidates has been scheduled. Research organizations from China, Canada and European Union will participate in the exercise.

Testing of a small-scale fuel assembly in a research reactor is considered a mandatory step before the construction of a prototype SCWR. A material test loop capable of operating with supercritical-pressure water has recently been constructed at Centrum Vyzkumu Rez (CVR) in Czech Republic. It has provided ample experience on design, construction and installation of a supercritical-water facility. The loop is undergoing the licensing process and will come online shortly.

Neutronics

Reactor physics analyses have been performed using analytical tools developed for light water and heavy-water reactors. Nuclear data are available for high-temperature conditions but not at relevant pressures. Previous analyses concluded the effect of pressure on reactor physics parameters is small. However, nuclear data would be required for the construction of the demonstration SCWR plant.

Safety system design

An integral test facility is required to demonstrate the effectiveness of the safety system design in SCWRs. Design and construction of a full-scale integral facility are time consuming and costly. A scaled-down system could be applicable to examine the phenomena and minimize the time and cost. Scaling analyses have been performed for major components in the safety system to confirm feasibility. An integral analysis is being carried out to quantify the appropriate scaling factor using a safety analytical tool.

Full scale fuel bundle test

Thermal-hydraulics experiments were performed using simple channels (such as tubes and annuli) and bundle subassemblies with three, four or seven rods. Licensing of the demonstration unit and a full-scale SCWR plant would require thermal-hydraulics data for full-scale fuel assemblies. Significant expansion of current facilities is required for licensing purposes.

A demonstration of SCWR plant

A demonstration SCWR plant is required before deployment. The design and construction of the demonstration plant require a strong collaborative effort among international community.

Summary

SCWRs are the natural evolution of the current water-cooled reactor technology (i.e., PWRs, BWRs or HWRs), which have accumulated over 50 years of design and operation experience. While improving economic is the main objective for the development, SCWRs have the potential of meeting all GIF technology goals of improving economics, safety and reliability, sustainability, and proliferation resistance and physical protection.

Seven SCWR core concepts have been developed covering thermal and fast spectra in pressure vessel and pressure-tube configurations. Fuel assembly concepts were established specifically for these SCWR cores. Most of these concepts were based on the traditional uranium fuel while the one for the pressure-tube type SCWR adopted the thorium fuel. Three groups of developers from Canada, European Union and Japan have also completed their plant concepts. Those from China and the Russian Federation are working on their concepts.

Despite of the success in completing the SCWR core and fuel concepts, technical challenges attributing to the high cladding temperature were encountered. Mitigation strategies of these challenges have been established with specific work planned.

See Also: Advanced Light Water-Cooled Reactor Concepts.

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Advanced Light Water-Cooled Reactor Concepts

Eugene Shwageraus, Department of Engineering, University of Cambridge, Cambridge, United Kingdom

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Glossary

ABWR Advanced boiling water reactor
BWR Boiling water reactor
CGN China general nuclear power group
CHF Critical heat flux
CNNC China national nuclear corporation
CPR Critical power ratio
CR Conversion ratio
EDG Emergency diesel generator
EPR Evolutionary (or European) pressurized water reactor
ESBWR Evolutionary boiling water reactor
FCM Fully ceramic microencapsulated fuel
FP Fission products
GE General electric
KAERI Korea Atomic Energy Research Institute
MA Minor actinides
MDNBR Minimum departure from nucleate boiling ratio
MOX Mixed (U–Pu) oxide fuel
NSSS Nuclear steam supply system
PWR Pressurized water reactor
RBWR Resource-renewable boiling water reactor
RPV Reactor pressure vessel
SOEC Solid oxide electrolyzer cell
TRISO Tri-structured Isotropic fuel particle

TRU Trans-uranic elements
 VVER Water-cooled water-moderated power reactor

Brief history and current status of LWRs

Light water-cooled reactors are by far the most mature nuclear technology, delivering reliable power since the first commercial pressurized water reactor (PWR) began its operation in Shippingport, PA in 1957. Commercialization of PWR technology was a direct consequence of the successes of the US Naval Reactors program, borrowing many technology features identified by the program, such as zirconium-based cladding, UO_2 fuel and pressurized water as a coolant. Generating steam directly in the reactor core was originally in question due to core stability concerns. However, the series of BORAX experiments in the 1950s and 1960s showed that boiling water reactors (BWRs) are also feasible, leading to the progressively improving line of BWR designs. BWRs offer the advantages of simpler plant layout with no need for steam generators as in PWRs (although earlier designs did have them), lower operating pressure, and a somewhat more flexible load following capability. The main drawback of BWRs is the presence of slightly activated water circulating through the turbine and the rest of the power plant. Overall, the economics of PWRs and BWRs have proven to be comparable, and so are the safety characteristics of modern designs.

LWRs have undergone significant evolution since the early days of nuclear power. Advanced cladding alloys, experience with UO_2 fuel and its manufacturing process, as well as improved understanding, monitoring and control of the coolant water chemistry, have led to substantial increases in fuel burnup (up to 60 MWd/kg), increases in operating cycle length between refueling outages (up to 24 months), improved capacity factors, and continual reductions in fuel failure and radiation doses to plant personnel in modern LWRs. Many existing LWRs were successfully re-licensed, allowing them to operate at higher than their originally designed power, or for longer than their original operating lifetime, or both. The successful power uprates range from a few percent to up to 20% of the original power capacity; these were enabled by improved instrumentation (thus, lower measurement uncertainties of the plant parameters), better understanding of critical components' degradation mechanisms, and some equipment refurbishment or replacement. In the United States, many LWRs built in the 1970s and 1980s have successfully gained an operating license extension from their original 40–60 years. At the time of writing, two PWR units at Turkey Point, Florida, and two BWR units at Peach Bottom, Pennsylvania, have been granted their operating license extension to 80 years.

Direct descendants of early prototypes, such as Shippingport and the BORAX experiments, resulted in what is known as the first generation of LWRs. These were relatively low power (< 1000 MW electric), relatively inefficient and somewhat complex designs. This was due to the limited experience and relatively large uncertainties, which therefore required large engineering safety margins. These reactors, however, shared most technological features, layouts, and materials choices with the modern-day LWRs. With better understanding of operational issues and component manufacturing, as well as construction techniques, the efficiency and size of reactors substantially increased, giving rise to the Generation-II LWRs, which also had improved safety features. The majority of LWRs operating today are of Generation-II. The most significant changes were associated with the growing emphasis on safety following the accidents at Three Mile Island in 1979 and Chernobyl in 1986. The main LWR design challenge with regards to safety is related to satisfying two conflicting objectives: effective cooling of the core in all expected accident conditions and containment of radioactive fission products (FP) should the core integrity be compromised. Meeting the latter objective requires designing ever more robust and tightly sealed containment structures, while the former requires efficient ways of removing the stored energy from within the containment into the environment (i.e., to the ultimate heat sink).

Originally, PWR designers opted for large containment volumes, sometimes with integrated intermediate heat sinks such as an “ice condenser” that could partially absorb the energy released in an accident. In addition, cooling water sprays would scrub the containment atmosphere of FPs and condense the steam, relieving the containment pressure and reducing the probability of leakage.

BWR designers introduced the concept of a suppression pool as an intermediate heat sink within the containment, temporarily absorbing some of the energy generated by the core in an accident, allowing the containment to be very compact. Blowing potentially radioactive steam from the reactor pressure vessel (RPV) through the suppression pool also allows partial removal of fission products and condensation of the steam—provided the water in the suppression pool remains below saturation temperature.

Another challenge associated with safety of LWRs is that the cooling water must be pressurized to achieve a reasonably high temperature of heat addition to the power cycle and, therefore, reasonable thermodynamic efficiency of power conversion. If the integrity of the cooling circuit were compromised, the loss of coolant would be very rapid and would need to be counteracted by makeup water injection. This is typically accomplished by motor-driven pumps, which require an alternative source of power such as Emergency Diesel Generators (EDGs). In BWRs, the task of reflooding the core with water was further complicated by the presence of coolant recirculation pumps, which control the coolant flow through the core and had to handle the entire recirculation flow rate. Recirculation pumps in early BWR designs required piping connections to the RPV below the core. Therefore, reflooding the core in case of a large break of one of these pipes would be impossible. Later BWR designs featured a larger suppression pool and the introduction of jet-pumps, which substantially reduced the flow rate external to the RPV and allowed the lowest level of large pipe penetrations to be at about two-thirds of the core height level.

Nearly all LWRs being built today are of Generation-III, which offer further improvement in efficiency and attempt to address the shortcomings of the previous Generation-II designs. In terms of economics, the prevailing trend until recently was to increase the power output of reactor units in pursuit of “economies of scale,” with the largest design currently on the market being the European (or Evolutionary) Pressurized water Reactor (EPR), with electric power output of nearly 1700 MW. This trend, however, led to an increase in complexity of the designs and challenges with the management of construction projects. The latest experience with construction delays has raised concerns over the viability of large reactors. This led the proposal of an alternative strategy of modular construction of smaller units in factories. Small Modular Reactors—SMRs, discussed in detail in [Reitsma \(2021\)](#) and [Ingersoll \(2021\)](#), can then be transported and assembled on-site with minimal construction effort.

In terms of safety, Generation-III LWR designs have many commonalities. For example, both PWR and BWR designers recognized the benefits of having a large water tank in close proximity of the core (the large suppression pool in BWRs and the In-containment Refueling Water Storage Tank, IRWST, in PWRs), which can serve as temporary heat sinks, allowing fast but controlled de-pressurization of the core and serving as a source of makeup cooling water. To improve reliability and reduce costs through simplification, modern LWRs increasingly rely on natural phenomena—such as gravity-driven flow or natural convection of coolant and/or air—to accomplish their safety functions, rather than requiring backup power from EDGs. Safety systems of some designs are entirely passive (e.g. AP-1000—a PWR designed by Westinghouse and ESBWR—Evolutionary Simplified BWR designed by GE-Hitachi) or include a combination of passive and active systems: active, when a fast response time is desirable, and passive, where its use leads to improved reliability of the design and requires long periods of operation (e.g. APR-1400, VVER-1200, ABWR). These latest designs are often referred to as Generation-III+ reactors. Jet-pumps have been replaced by recirculation pumps which are fully integrated into the RPV in the GE-Hitachi ABWR with no pipe connections at the core level—a feature successfully used in earlier ABB and Siemens BWR plants. GE-Hitachi has gone even further and introduced fully natural circulation-driven flow in their ESBWR design, which, as mentioned earlier, also offers fully passive safety systems to cope with all design basis accidents.

In summary, Generation-III/III+ LWRs have made unprecedented advances in the efficiency and reliability of operation, service life, and in safety, reducing the probability of core damage by up to two orders of magnitude compared to Generation-II LWRs. However, a number of frequently voiced reproaches remain to be addressed and have resulted in a vibrant research field, occupied by academics, national labs, reactor designers as well as nuclear plant operators. Have the LWRs reached their potential or can they do more or do better? The remainder of this article will present an overview of advanced LWR concepts aiming to address some of these challenges.

Challenges of the current fleet

In 2001, an international forum of experts was tasked with developing criteria for assessing the societal attractiveness of future nuclear energy systems, screening advanced reactor concepts against these criteria, and identifying those worthy of further development through international collaboration—also known as the Generation IV International Forum (see [Stanculescu, 2021](#)). The basic requirement was that Gen-IV reactors should offer substantial advantages over Gen-III LWRs in terms of economics, safety and security, sustainability of resources, and waste management. Therefore, existing LWR design challenges can be grouped in accordance with these themes.

Economics

The practically achievable thermodynamic efficiency of power conversion in LWRs is relatively low, on the order of 33%, which is substantially less than that of modern Combined Cycle Gas Turbines (> 60%) or Supercritical Coal power plants (> 50%). The efficiency of LWR power cycles is limited by the maximum coolant temperature (around 300 °C) and pressure (about 15 MPa in PWRs). These operating conditions are already pushing the boundaries of state-of-the-art manufacturing of large reactor components, such as steam generators and RPVs.

Furthermore, the cost of LWR-generated electricity is largely determined by the construction (capital) cost of the plant with only a minor contribution from operating costs. It is therefore desirable to reduce the capital costs as much as possible in order to achieve favorable economics. Capital costs strongly correlate with the reactor core power density: the more compact the core is for a given reactor power output, the smaller the expenditure is on many of the plant components. Modern PWR power densities are hard to challenge; it is not a coincidence that this reactor type is preferred in most naval propulsion applications, where compactness is the key feature. A substantial amount of R&D, as well as operational experience, has resulted in highly optimized LWR fuel and core designs, maximizing the core power output while meeting all safety criteria, seemingly leaving little room for further substantial improvements.

Sustainability of fuel resources

Current LWRs operate primarily in an “open” (or “once-through”) fuel cycle, in which freshly mined uranium is enriched, converted to UO₂, fabricated into fuel assemblies, and loaded into a core for generating nuclear heat. Following discharge from the core, the fuel is typically stored in cooling ponds and then dry cask storage for a number of years, pending ultimate disposal in geological formations suitable for immobilization of radioactive spent fuel constituents over time scales sufficient for their decay. This practice

allows only a small fraction ($< 1\%$) of the originally mined uranium's energy value to be utilized. For a typical LWR fuel with 4–5% initial enrichment, about 10 kg of natural uranium needs to be mined in order to produce 1 kg of enriched uranium. Furthermore, only about 5% of the enriched uranium loaded into the core will be fissioned, producing useful heat. This simple arithmetic suggests that 99.5% of the mined uranium's energy remains unused. The open cycle practice is presently the most economically preferred option because uranium is cheap and plentiful. The currently known reserves are sufficient to support even the most aggressive nuclear power expansion imaginable, at least until the end of the century. In addition, fuel costs constitute only a small fraction of the overall nuclear generation cost. Therefore, the cost of uranium is almost irrelevant to the economics of nuclear power. This approach, however, can be challenged on moral grounds: is it ethically justifiable for modern society to deprive future generations of a valuable energy source by following current practice?

Spent fuel contains non-negligible amount of fissile isotopes (mostly plutonium generated from neutron captures in U^{238} and remaining U^{235}) which can be recycled in existing LWRs as Mixed U–Pu Oxide (MOX) fuel—but only once—saving about 20% of natural uranium. This is a relatively well-established practice in many countries, most notably in France, but also in the United Kingdom, Germany, Belgium, Switzerland, Russia, and Japan. Further recycling is difficult because of the degradation of the plutonium isotopic vector, as it becomes less and less rich in fissile isotopes, as well as the increasing complexity of Pu recycling and MOX fuel fabrication. It has been argued, therefore, that in order to truly take advantage of the large energy content in spent LWR fuel (as well as in the depleted uranium tailings of the enrichment process), a completely different type of reactor with a fast neutron energy spectrum is needed, along with sophisticated spent fuel reprocessing and a recycling infrastructure to support them.

Safety

It has been noted that modern LWRs have come a long way in terms of improving their safety since the early days of the technology. The efforts to simplify the safety systems or eliminate certain fault scenarios continue and could result in substantial improvement in LWR economics. Two particularly high-impact issues are common to all LWRs, which led to complexity and high costs of safety systems: rapid loss of coolant in the case of a large-break in primary circuit piping, and rapid oxidation of zirconium cladding following the loss of coolant with an eventual loss of coolable geometry, preventing re-flooding of the core and leading to its further damage.

Weapons non-proliferation

Plutonium in spent nuclear fuel discharged from existing LWRs is not well-suited for use in nuclear weapons. It has a relatively small fraction (typically about 50%) of Pu^{239} , which is the most desirable isotope. Other plutonium isotopes significantly complicate weapon design because they have a high probability of spontaneous fission, which creates a high neutron background, leading to pre-initiation of an explosion and a substantial reduction in its explosive yield. Furthermore, Pu^{238} (constituting 2–3% of Pu in a typical spent LWR fuel) has a short half-life resulting in high heat generation, which would melt or cause spontaneous combustion of all generic high explosives used to compress the plutonium to achieve critical conditions. Although these complications may be viewed as a substantial deterrent against the use of Pu from spent LWR fuel for weapons proliferation, some argue that this possibility cannot be ruled out completely. Following discharge from the core, the Pu in spent fuel is self-protected from unauthorized diversion for several decades by the presence of highly radioactive fission products. However, after a few hundred years, once fission products have largely decayed, Pu could, in principle, be more easily diverted if access to a geological repository is gained by rogue states or terrorist groups. It is therefore desirable that the amount and attractiveness of weapons-usable material in commercial reactors spent fuel be minimized.

Spent fuel management

In spent nuclear fuel from LWRs, about 5% of the original uranium mass is converted into fission product isotopes. Many of these isotopes are highly radioactive. For this reason, they are also relatively quick to decay away, with the vast majority becoming stable isotopes in a few hundred years—a timeframe for which the environmental isolation of nuclear waste is a relatively simple engineering task. Other nuclides generated from multiple successive captures of neutrons in the original uranium isotopes—but without causing their fission—are known as trans-uranic elements (since they are positioned higher than uranium in the periodic table, e.g., neptunium, plutonium, americium, curium, etc.). Plutonium isotopes constitute about 1% of typical irradiated LWR fuel, while the rest of the trans-uranic (TRU) nuclides collectively constitute only about 0.1% and are therefore known as Minor Actinides (MA). The majority of plutonium and MA isotopes are less radioactive than FPs, i.e., have longer half-lives. Additionally, some MAs have relatively high mobility in the environment. Therefore, they require environmental isolation for hundreds of thousands of years. Demonstrating the integrity of engineered barriers for nuclear waste packages over such time scales is a challenging task. Thus, a nuclear waste repository safety case relies on geology and historical analogues (e.g., studies of the natural nuclear reactor in Oklo, Gabon). Although licensing of geological waste repositories has been successful in several countries, in others, the process became prohibitively expensive financially or politically due to local public opposition (Metlay, 2021; Kadak 2021). Furthermore, a similar argument on the moral responsibility for the sustainable use of energy resources can also be made with regards to the nuclear waste legacy left to future generations. Therefore, reprocessing and recycling of all TRU isotopes until their complete fission has been proposed, not just to improve utilization of natural resources, but also to keep the long-lived radioactive isotopes away

from a geological repository. Numerous studies have shown that the task of transmuting the long-lived TRUs into short-lived FPs can be accomplished most efficiently by fast spectrum Gen-IV reactors.

Emerging solutions

High operating temperature

One way of improving the economics of LWRs is to increase the temperature of heat delivered to the power cycle. BWRs use a direct cycle, thus saving a small, but notable, loss of temperature difference required for heat transfer across a heat exchanger. A natural evolution of BWR designs would be to raise their pressure and temperature as much as technologically affordable. This direction of reasoning resulted in the development of the super-critical water-cooled reactor (SCWR) concept discussed in detail in [Zhang et al. \(2021\)](#).

Increasing the operating pressure and coolant temperature may not be feasible due to manufacturing challenges and the availability of high-temperature corrosion-resistant materials for SCWR core components. An alternative solution could be to keep the operating pressure within the conventional BWR range but allow super-heating of the saturated steam produced in the core. This task can be accomplished by dedicating certain regions of the core to be cooled by steam rather than a steam-water mixture, or by an external super-heater with a non-nuclear heat source. In fact, many early LWRs included either a fossil fuel (oil or coal)-fired or a nuclear superheater, which would increase the temperature of the saturated steam produced in other parts of the core and therefore increase the thermal efficiency of power conversion. These early attempts were only marginally successful. The most typical problems were with geometry distortion of the fuel in the superheater region of the core due to differential thermal expansion of materials or stress corrosion cracking. Additionally, drastic differences in neutron moderation in water- and steam-cooled regions of the core made control of the power distribution particularly challenging. Eventually, the complexities of having a superheater were not economically justifiable. Instead, increasing the pressure (and thus the temperature) of saturated steam produced in later generations of LWRs was found to be a more attractive design direction, with modern LWRs achieving higher power cycle efficiency with saturated steam than earlier designs with a superheater.

The idea of superheated steam has not been abandoned completely, however. Recent technology developments have led to the reemergence of interest in superheated steam reactors. These are:

- BWRs have accumulated vast operational experience with many issues of early designs being successfully resolved;
- Development of new cladding and structural materials along with understanding of mechanisms for their degradation;
- Successful operational record of superheated steam fossil plants;
- Substantial development effort in unconventional, high-performance fuel concepts such as annular, internally and externally-cooled fuel geometry ([Hejzlar and Kazimi, 2007](#)) discussed in some detail later in this article.

A recent study by [Ko \(2010\)](#) has combined the developments mentioned above and proposed a BWR with a double pass of coolant flow through a reactor core composed of annular fuel elements—the Annular-fueled Superheat Boiling Water Reactor (ASBWR). In the first pass, the cooling water flows along the outside surface of annular fuel elements, generating saturated steam. The steam is then diverted back into the core, this time, flowing inside the fuel annulus, picking up additional heat, which raises its temperature up to 520 °C before directing it to the steam plant. The ASBWR concept was shown to achieve power conversion efficiency in excess of 40%, which is considerably higher than existing LWRs, while having a core power density of about 50 W/cm³, which is comparable to conventional BWRs. The study, however, does not go beyond the basic feasibility of the concept and would require substantial follow-up research. As future work, the study suggests focusing on fuel performance modeling in such unusual conditions, as well as safety analysis. Assessment of the economic case is also needed to determine whether incremental revenues from more efficient power conversion would offset the incremental investment in more complicated core structures and less neutronically efficient fuel due to higher neutron absorption in the double cladding, which may also have to be made out of stainless steel rather than Zircalloy.

As discussed earlier, external superheaters may have more advantages. In addition to being simpler (the core operation can be decoupled from the superheater operation), it can potentially allow variation in the degree of the superheating or in the fraction of steam flow which is superheated. Such a mode of operation can extend the load following capabilities of nuclear power plants. This could be an extremely valuable feature for future low-carbon energy systems with a growing fraction of intermittent renewable energy generators. There are many options for energy sources to power the superheaters. For example, this could be a separate, independently-controlled nuclear core, cooled directly by steam or indirectly by another coolant (e.g. helium gas or molten salt), which then carries the heat from the core to the superheater. Alternatively, the source could be the exhaust of a conventional natural gas-powered turbine, which can provide heat at close to 600 °C, or direct combustion of natural gas or other fuels (such as hydrogen or ammonia) which can be synthesized using the energy produced by the system during periods of low electricity demand. Although the use of fossil fuels to superheat the nuclear generated steam will produce carbon emissions, these can be relatively low because it will be used only to meet the peak electricity demand and to raise the temperature of already hot steam, implying that the power conversion efficiency of fossil heat to electricity will be relatively high (comparable to a combined cycle gas turbine, [Wibisono and Shwageraus \(2016\)](#)). Finally, the superheater can be coupled to a heat storage system based, for example, on molten salt, and powered by a renewable energy source, such as a concentrated solar farm.

The idea of coupling LWRs with external superheaters has been recently explored in a series of studies by Wibisono and Schwageraus (2016) and Wibisono et al. (2019). It has been shown that a small BWR with an external superheater system can accommodate a range of power generation loads between 65% and 100% of total system power, while always operating the nuclear reactor core at its full power. Such a combined system can achieve a thermal efficiency of power conversion of about 40%, which is considerably higher than conventional LWRs.

High power density

Another approach for improving the economics of LWRs is to increase their power output for the same volume of the core and other components. An LWR's core power density is typically limited by the ability of cooling water to remove the heat produced in the core during reactor operation and under several limiting accident scenarios, such as a Large Break Loss of Coolant Accident. Degradation of fuel can occur as a result of multiple mechanisms under accident conditions. Broadly speaking, the most important of those are fuel melting and cladding failure due to excessive oxidation or strain at high temperatures. Increasing safety margins to any of these limiting phenomena could lead to cost savings due to either simplification (or complete elimination) of some safety systems, or the ability to operate at higher power while maintaining the same margins, or both. Therefore, the design strategies for achieving higher core power density aim at reducing the fuel and cladding temperatures at nominal and accident conditions, as well as searching for materials which can maintain high strength and oxidation resistance at high temperatures.

Since heat removal from fuel elements occurs through the fuel surface, increasing the fuel surface area would increase the heat removal capabilities for a given margin to fuel failure. A number of approaches have been suggested to increase the fuel surface area by transitioning from the standard cylindrical fuel pin geometry to alternative fuel shapes.

Annular fuel

In PWRs, one option is the aforementioned annular, internally- and externally-cooled fuel (Hejzlar and Kazimi, 2007). Cross-sectional views of the conventional and annular fuel pin geometries are compared in Fig. 1. By nearly doubling the fuel surface area, the surface heat flux is reduced proportionally, which also leads to a corresponding increase in the DNBR margin. By reducing the heat conduction path from fuel to coolant, the temperature difference across the thickness of the annular fuel pellet can be reduced by over a 1000 °C (Fig. 2). Cooler fuel increases the margin to melting and the amount of stored energy to deal with in loss of cooling accidents. Lower fuel temperature also leads to more robust fuel performance over irradiation, with less swelling and fission gas release. A number of neutron economy effects largely offset each other. For example, although there is a smaller Doppler reactivity decrement¹ because of the lower fuel ΔT between cold and hot full power core state, this is mostly offset by the larger magnitude of the Doppler Coefficient due to an increase in resonance captures in U^{238} and parasitic absorption in additional cladding material (Xu et al., 2007).

The initial testing of the concept by a consortium of US universities, industry, and national labs was very promising and drew interest from South Korea, where a parallel project was launched by KAERI (Koo et al., 2013). The R&D activities showed that a substantial power up-rate of existing PWRs of up to 50% is possible, while maintaining or improving the safety margins. The activities covered a wide range of topics including core physics, thermal-hydraulics, safety analyses, mechanical testing, manufacturing (including actual pellet and fuel pin fabrication) and economic assessments. Safety analyses performed by KAERI for 120% of nominal power cases suggest substantially improved fuel characteristics (lower fuel and cladding temperatures) in a range of typical design basis accidents, such as reactivity insertion and loss of coolant. Small-scale irradiation tests have also been performed and showed favorable performance of the fuel. Nevertheless, at the moment, there are no large-scale follow-up projects being carried out aiming at commercialization of annular fuel.

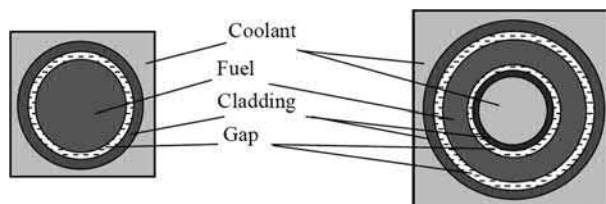


Fig. 1 Cross sectional view of cylindrical and annular fuel pin geometries. Hejzlar P, and Kazimi MS (2007). Annular fuel for high-power-density pressurized water reactors: motivation and overview. *Nuclear Technology* 160 (1).

¹Doppler Effect is an increase in neutron absorption in fuel nuclides with an increasing fuel temperature. It occurs due broadening of resonances in the fuel isotopes' absorption cross section as the fuel temperature increases. Therefore, hotter fuel will have lower reactivity.

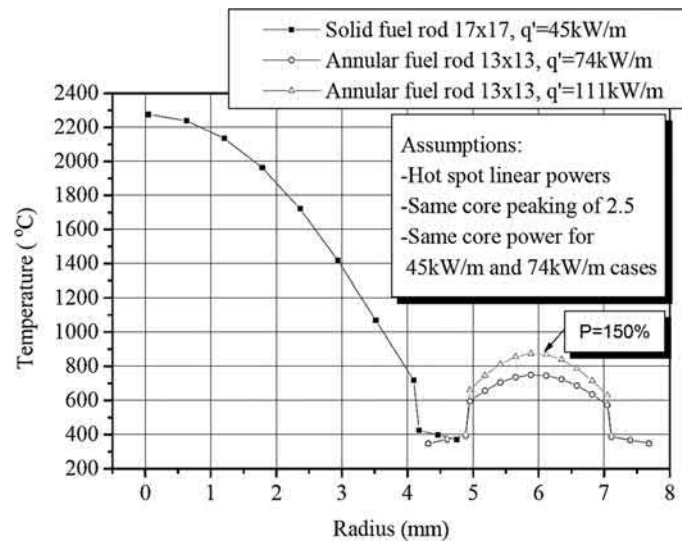


Fig. 2 Comparison of radial temperature profiles at the core location with highest power (hot spot) produced per unit of fuel pin length (also known as linear heat generation rate expressed in kW/m) for solid and annular fuel. Hejzlar P, and Kazimi MS (2007). Annular fuel for high-power-density pressurized water reactors: motivation and overview. *Nuclear Technology* 160 (1).

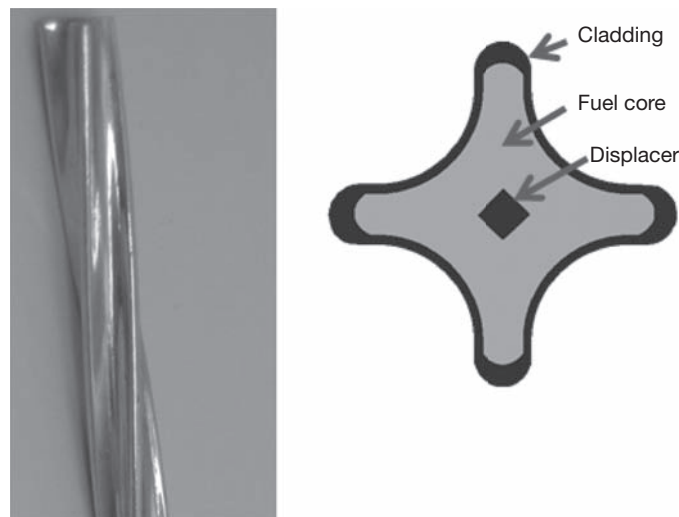


Fig. 3 Lightbridge fuel rod (left) and its cross section (right). Courtesy: Lightbridge Corporation 2020, used with permission.

Lightbridge fuel

Another fuel type with alternative geometry is being developed by Lightbridge Ltd. in cooperation with Framatome (Malone et al., 2012). In this case, the increase in fuel surface area is achieved by corrugating the edges of cylindrical cladding to form four fins, as shown in Fig. 3. This fuel shape is allegedly inspired by Russian marine propulsion fuel (Bays et al., 2012). Furthermore, the cruciform-shaped pin is twisted in order to provide coolant flow mixing, which enhances heat transfer and also removes the need for spacer grids, reducing the flow resistance. The Lightbridge fuel is in metallic form, which offers additional advantages due to its high thermal conductivity. Zirconium-niobium alloy cladding is metallurgically bonded with the fuel and deforms together with it to accommodate the irradiation swelling, which is claimed to be similar to conventional UO_2 fuel. A displacer in the center of the rod helps to lower the maximum fuel temperature. It is also made of zirconium alloy or may contain some form of burnable poison.

Similar to the annular fuel, a larger fuel surface area and low thermal resistance to heat transfer substantially reduce the Lightbridge fuel temperature and increase the MDNBR, allowing power up-rates while offering greater safety margins. The fuel being designed and tested by Enfission (a joint venture between Lightbridge and Framatome) is fully retrofittable into existing PWRs and would allow greater operational flexibility and a power uprate of up to 30%. A series of extensive irradiation testing has been performed in research reactors and the development team seems to be on track to complete the fuel qualification, although a detailed timeline for commercialization is not currently available.

A computational and thermal-hydraulic experimental study by Conboy et al. (2014) on power uprates of LWRs using similar helical cruciform fuel geometries have concluded that up to a 47% power uprate of PWRs is possible with such fuel, similar to what was found to be achievable in the annular fuel studies.

Larger number of pins

The simplest way of increasing the fuel surface area for the same fuel volume, however, is to reduce the fuel pin diameter and increase the number of pins in the core. Transitioning from conventional 17×17 pins PWR fuel assembly designs to 19×19 or more fuel pins per assembly will, of course, have some detrimental effects. These could include increased pressure drop and higher pumping power requirements, less mechanically rigid fuel—requiring a redesign or re-evaluation of fuel vibration limits on flow velocity—as well as an increased resonance capture rate due to increased surface area and parasitic absorption due to larger cladding volume. These effects can be potentially compensated by lower capital costs due to the higher core power density. This has been shown, for example, in an alternative tight-pitch core design of the IRIS reactor, proposed and jointly developed by an international consortium of Westinghouse with several universities (Saccheri et al., 2007).

Alternative fuel geometries, such as more numerous pins of smaller diameter, or annular or Lightbridge-type fuel, are somewhat less effective in allowing power uprates of BWRs than PWRs (Karahane et al., 2010; Ellis, 2006; Conboy et al., 2014). Increases in power density on the order of 20% appear to be possible for these fuel geometries. For annular fuel, the reduction in surface heat flux is offset by the requirement for greater coolant mass flux, which has a negative effect on the Critical Heat Flux² (CHF) in BWR conditions, as opposed to its beneficial effect in PWR conditions (Todreas and Kazimi, 2012). Higher coolant mass flux is necessary to compensate for the lack of mixing in the inner channels and the generally less uniform radial power distribution in BWR fuel assemblies.

A systematic re-evaluation of the BWR fuel design space has been performed by Shirvan and Kazimi (2013) with an objective of increasing the power density of conventional BWR cores while maintaining the same safety margins. The optimized design can achieve a 26% higher core power density as compared to the ABWR and has a 65% higher specific power.³ It uses 35% shorter fuel pins of smaller diameter arranged in 16×16 fuel assemblies. Higher power for the same CPR⁴ margin has been achieved through traditional strategies, namely, reducing the linear heat generation rate and coolant mass flux to compensate for slightly higher exit quality. The findings of this study are of significant importance because the proposed design uses conventional fuel geometry, materials, validated analysis tools as well as proven manufacturing techniques, effectively enabling a short path to licensing and commercialization. It appears that this study discovered a region in the BWR core design space which was previously overlooked by the designers because of the greater focus on fuel cycle efficiency rather than on capital costs. The latter, however, has evidently become substantially more important.

Hydride fuel

In LWRs, the coolant serves as the moderator. Therefore, the amount of water needs to be sufficient to satisfy both requirements, although some design compromises are often inevitable. In a recent study, for example, it has been proposed to use U–Zr hydride fuel in an attempt to reduce the coolant volume without compromising neutron moderation (Greenspan et al., 2009). U–Zr hydride has been used in research reactors for many decades operating at low, often nearly room, temperature. Its properties, however, in principle, do not necessarily preclude using it as a fuel form in LWRs, if the shortcomings are well-understood and properly addressed in fuel design (Olander et al., 2009). Metal hydrides typically have a thermal conductivity close to metals, which is much better than that of UO_2 and would limit the operating fuel temperatures to an acceptable level, even at typical LWR power densities. The study covered a wide design parameter space: both PWR and BWR conditions, square and triangular fuel pin lattices, conventional UO_2 and U–Zr hydride fuel, and, finally, “inverted” fuel pin geometry (a prismatic fuel block with coolant channels passing through it, which allows for larger fuel volume fractions than geometrically possible with conventional pins, even in triangular, tightly-packed lattice).

The study showed that a hydride fuel PWR core, with a tight triangular pitch and wire-wrap support, can achieve a sizable power density uprate of approximately 53% in comparison to conventional designs, provided about a factor of two higher coolant pressure drop across the core can be accommodated. In BWRs, the use of hydride fuel can eliminate the necessity for water rods, reducing heterogeneity of the fuel assembly and radial pin power peaking factors. The predicted power uprate possible through switching to hydride fuel in BWRs is insignificant if the core pressure drop were to be kept the same as in a conventional UO_2 core. Accommodating a 50% higher pressure drop would allow a power uprate of about 20% in comparison with modern 10×10 assembly designs. The most important technological uncertainty of hydride fuel use in LWRs remaining to be demonstrated is the compatibility of the fuel with Zr-based cladding materials and water coolant.

²Critical heat flux is the amount of heat transferred per unit area of cladding surface at which direct contact of boiling coolant with the cladding is no longer maintained. In PWR conditions, this typically occurs when vapor bubbles on the cladding surface either grow too fast or detach too slowly, resulting in their overcrowding and coalescence. In this case, they eventually form a continuous vapor film with poor heat transfer coefficient which leads to significant cladding temperature increase. In BWRs, direct contact of vapor with the cladding occurs when liquid film covering the fuel rods dries out up-stream the coolant flow in a channel with the same undesirable cladding temperature increase.

³Specific power is the amount of heat produced per unit mass of uranium in the fuel. It is typically measured in kW/kg.

⁴BWR cooling channel dry-out is a function of total power input into the channel rather than local heat flux as in PWRs. Therefore, BWR thermal margin is measured as a ratio of power input into the channel at which dry-out occurs to the actual operating channel power. It is known as the Critical Power Ratio (CPR), which must be sufficiently greater than unity to account for uncertainties and anticipated transients.

Improved thermal conductivity

Currently, the Critical Heat Flux (Departure from Nucleate Boiling Ratio in PWRs and Critical Power Ratio in BWRs) is more limiting to the achievable power density than fuel temperature in steady-state operation or in anticipated transients. Nevertheless, a greater thermal margin to fuel melting is generally welcomed because it can improve tolerance to accidents, such as loss of coolant. Similar to the use of metallic uranium alloys as in the Lightbridge design, fuel temperatures can also be reduced by using other fuel forms with substantially higher thermal conductivity than UO_2 , such as uranium silicide (U_3Si_2) or uranium nitride (UN). Enhanced thermal conductivity can also be obtained by admixing high-conductivity materials to UO_2 ceramic. Numerous additives have been proposed including beryllium oxide (BeO), silicon carbide (SiC), molybdenum, chromium oxide (Cr_2O_3), diamond, graphene, and others. The main idea here is to create a high-conductivity matrix in which fuel particles are dispersed. If the matrix network is not continuous, the enhancement of thermal conductivity is minimal or not observed at all. All additives, however, have their disadvantages. These include displacement of fuel, high neutron absorption, degradation of thermal properties with irradiation, chemical reactions with water, and complicated or costly manufacturing processes, to name a few. One of the leading candidates is BeO. It improves the thermal conductivity of UO_2 by approximately 10% for each percent of BeO added, with the conductivity of UO_2 essentially doubled at about 10% of BeO in the mixture (Garcia et al., 2017). The enhanced thermal conductivity alone (without changes in fuel pin geometry) cannot justify substantial power uprates to improve the economics of power plants. Instead, they are marketed as fuels with enhanced accident tolerance which can increase the safety margins in some accident scenarios and, therefore, lead to simplifications in the safety case and potential associated cost savings. Accident-tolerant fuels will be discussed further in a later section of this article.

Improved fuel utilization

Fuel utilization is typically measured in terms of the amount of energy generated per unit mass of fuel (fuel burnup) relative to the amount of natural uranium required to manufacture the fuel. In a typical LWR fuel, about two thirds of energy is produced from fissioning the original fissile atoms of U^{235} . The remainder is produced mostly from fissions in Pu^{239} generated as a result of neutron captures in fertile U^{238} . The amount of initial U^{235} represents the initial investment in the fuel (enrichment), while energy produced represents the return on that investment. It can be shown (Fei et al., 2014) that the levelized fuel cycle cost (Rothwell, 2021) is roughly proportional to the ratio of initial fuel enrichment to the fuel discharge burnup. The fuel cycle cost of LWR fuel is almost independent of enrichment because natural uranium, separative work, and attainable burnup all scale approximately linearly with enrichment. Therefore, fuel cycle costs and fuel utilization cannot be improved substantially by varying the enrichment but only by extending the fraction of energy produced from plutonium fissions, because their fractional contribution to total energy would increase with burnup. This can be accomplished by maximizing the rate of neutron captures in fertile ^{238}U (thus, maximizing the rate of fissile ^{239}Pu production) and the number of neutrons available for these fertile captures.

The first mechanism is related to the Conversion Ratio (CR), which is defined as the generation rate of new fissile material (essentially, the rate of neutron captures in ^{238}U) relative to the rate of fissile isotope disappearance (neutron absorption rate in ^{235}U and ^{239}Pu). If CR is greater than one, the system is said to be self-regenerating or a “breeder,” i.e., the system generates new fissile material faster than it consumes it. In a typical LWR, the average core CR lies within the range of 0.5–0.6.

The abundance of neutrons available for breeding is related to the η -factor, which, for a fissile isotope, represents the average number of neutrons released per neutron absorbed in that isotope. η -factors for the most relevant fissile isotopes as a function of the absorbed neutron energy are presented in Fig. 4.

As can be seen from the figure, the most abundant neutrons available for breeding are released as a result of fissions induced by high energy ($E > \sim 10^5$ eV) neutrons, making fast spectrum reactors particularly attractive with respect to achieving high fuel utilization. In thermal spectrum reactors ($E < 0.1$ eV), such as LWRs, the η -factors are only slightly greater than 2, which means that there are barely enough neutrons to sustain the chain reaction (at least one neutron is needed for criticality) and to compensate for the fissioned atom by producing a new one through fertile capture (at least one neutron is needed to sustain the population of fissile atoms). Since neutron losses to leakage and parasitic absorption in structural materials and coolant are inevitable in realistic systems, achieving self-sustainable breeding ($\text{CR} > 1$) in LWRs is extremely difficult in practice. Nevertheless, numerous concepts have been proposed to improve the conversion ratio, bringing it as close to unity as possible. A number of strategies are used in these concepts.

The excess reactivity of fresh LWR fuel is typically on the order of 0.3, i.e., it generates about 30% more neutrons than needed to sustain criticality (self-sustaining chain-reaction). This excess is offset by a combination of reactivity control means such as burnable poisons, soluble boron, or control rods. These excess neutrons are effectively lost and do not contribute to regeneration of fissile material through captures in ^{238}U because ^{238}U is not competitive enough for neutron absorption with much stronger absorbers, such as ^{235}U and reactivity control materials (e.g. boron or gadolinium). Therefore, at the beginning of fuel life, the excess of neutrons is high, but the CR is low. With irradiation, however, both the original ^{235}U and reactivity control poisons are depleted, diluted, or mechanically withdrawn, and the competitiveness of ^{238}U in terms of neutron absorption increases. Unfortunately, as the reactivity is lost, fewer excess neutrons are available for breeding, and a high CR leads to shortening of the fuel cycle and reductions in discharge fuel burnup. Therefore, the “holy grail” of high conversion LWR designs is to reverse this trend. At the beginning of life, it is desirable to have the CR as high as possible to maximize the capture of excess neutrons in ^{238}U , while it is desirable to reduce the CR towards the end of irradiation to increase the fuel reactivity, prolonging the burnup and allowing newly-bred fissile plutonium to contribute a larger fraction to total energy produced by the fuel.

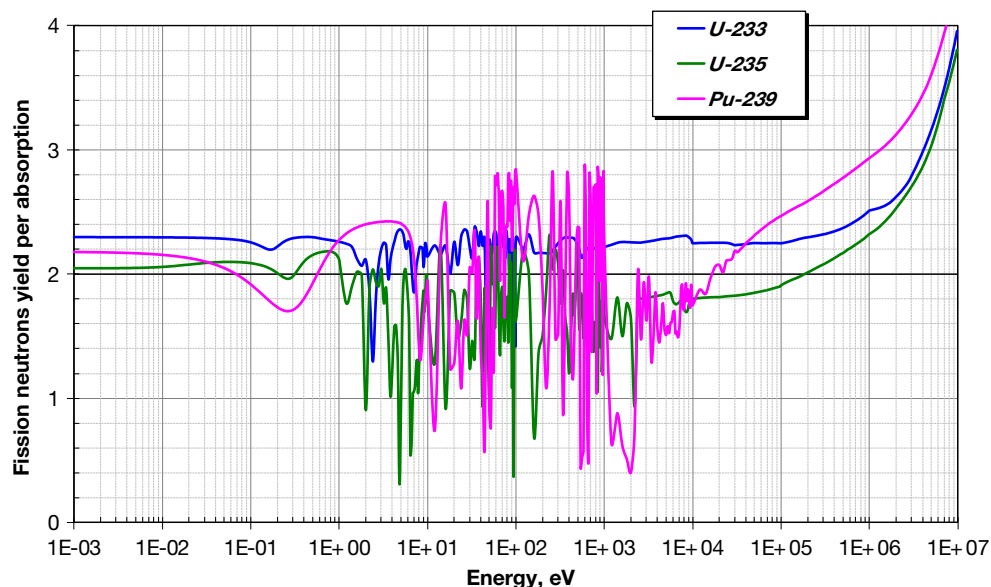


Fig. 4 Average number of neutrons released per absorption in ^{233}U , ^{235}U , and ^{239}Pu .

Spectral shift

One of the approaches to achieve the favorable behavior described above is to vary the amount of neutron moderation along the burnup cycle. Reducing the moderation initially would make ^{235}U and reactivity control materials—which are mainly thermal neutron absorbers—comparatively less competitive with the primarily resonant absorber ^{238}U . Reduced moderation effectively shifts the energy spectrum of the neutron population away from thermal and towards the epi-thermal (resonant) energies. As a result, the CR is increased and the need for reactivity control is reduced.

The fractional change in the amount of ^{238}U atoms throughout the burnup cycle is relatively small. Therefore, without changing the level of moderation, captures in ^{238}U will contribute a relatively constant negative reactivity, shortening the achievable burnup. The overall CR will be improved somewhat, but the energy share of plutonium fissions could be extended if moderation is gradually increased with irradiation time. Shifting the neutron spectrum back towards thermal energies will increase the fuel reactivity by reducing the neutron absorption in ^{238}U and increasing the relative competitiveness of fissions in newly bred ^{239}Pu .

A certain amount of spectral shift is already a routine practice for currently-operating BWRs. Long-term reactivity control of BWRs is accomplished by gradual withdrawal of control rods. Smaller reactivity adjustments, however, can also be made by varying the speed of the coolant recirculation pumps. A higher coolant flow rate reduces the core average void fraction (fraction of water that boils), providing better moderation. Conversely, when the flow rate is low, the void fraction is high, leading to less moderation. Therefore, many BWRs tend to adjust their control rod positions such that, at the beginning of cycle, the flow rate is low, but then gradually increases with irradiation.

This effect can be amplified by using the Spectral Shift water-Rod (SSR) concept being developed by Hitachi (Kondo et al., 2010). In contrast to traditional BWR fuel assembly water rods (Hamon, 2021)—which are always filled with water during reactor operation—SSR has a U-shape with a large diameter ascending flow section and a small diameter descending flow section, with openings below and above the fuel pin support plate, respectively, at the bottom of the core. Neutron and photon energy deposition in the SSR water leads to boiling, while the pressure drop across the support plate allows for adjusting the water level in the SSR by varying the coolant flow rate. At low flow rates, the SSR would be mostly voided, while, at high flow rates, it would be filled with water.

Reduced moderation can be achieved in PWRs by a variety of means, the neutronic design implications of which are extensively discussed in Ronen (1989). Reducing the fuel pin lattice pitch increases the CR but leaves limited practical opportunities to adjust moderation throughout the cycle. It must be noted, however, that increasing the CR through reduced moderation can be a viable strategy if the newly generated fissile material can be extracted from the spent fuel and subsequently reused in a closed fuel cycle. This strategy will be briefly addressed later in this article, as well as in Shwageraus (2021) in more detail.

Instead of reducing the fuel lattice pitch, light water (H_2O) can be replaced with the same amount of heavy water (D_2O). Since heavy water is a less efficient moderator, this approach is somewhat equivalent to reduction in lattice pitch. Variable moderation can be achieved by gradually diluting the heavy water coolant with light water. This will obviously be hard to justify economically as heavy water would be a consumable commodity with considerable associated costs.

In order to reduce the loss of available neutrons to parasitic absorption in reactivity control materials, a number of options are available—some have been attempted and some are currently in use. Reactivity can be controlled by adjusting the liquid moderator level in pressure tube-type reactors e.g. the Plutonium Recycle Test Reactor in the United States, the Nuclear Power Demonstration (NDP) reactor in Canada (Mitchell III, 1964) and the Steam-Generating Heavy Water reactor in the United Kingdom (Cartwright,

1968). More recently, a pressure tube-type reactor concept has been proposed with double wall horizontal calandria channels (Rachamin et al., 2013). The space between the walls can be flooded for each channel separately, such that the neutron spectrum can be gradually softened during irradiation allowing to control the core reactivity.

Some reactors use continuous on-line refueling, such as the CANDU (Nichita, 2021) and British AGRs. LWRs use multiple fuel batch refueling—a strategy which makes use of excess neutrons in fresh fuel assemblies to generate power in adjacent subcritical fuel batches and allows achieving higher fuel burnup for the same initial enrichment.

The moderator can be initially displaced by mechanical means. For example, a high-conversion PWR concept has been developed by Framatome (Hittner et al., 1988). In the proposed design, hexagonal, tight-pitch fuel assemblies include guide tubes that can be occupied by either control rods or water displacement rods loaded with natural uranium. The water displacement rods are fully inserted initially. The CR, in this case, is high because of the reduced moderation and large amount of ^{238}U . With irradiation, the displacement rods are gradually withdrawn, improving the moderation and reducing the amount of ^{238}U capture, which compensates for the reactivity loss. This design can theoretically achieve CR very close to one, although it has never been implemented in practice. A similar concept is currently being developed for the next generation of Russian VVER reactors (Teplov et al., 2014).

The main challenge with increasing the CR of fresh fuel in LWRs is the large amount of initial fissile material and reactivity control poisons which compete for neutrons with ^{238}U . An alternative approach to increase the competitiveness of ^{238}U is to physically separate the fissile and fertile components of the fuel. This idea is known as the “seed-blanket” concept. The fissile material is concentrated initially in a seed region, which produces a large number of excess neutrons. These neutrons leak out of relatively small (on the order of one neutron mean free path) seed regions into the blanket fuel regions loaded mostly with fertile isotopes, where they are absorbed with high probability. The concept was pioneered by Alvin Radkowsky and implemented originally in the first commercial PWR at Shippingport, PA. The Shippingport core configuration was modified several times and the last one is known as the Light Water Breeder Reactor (LWBR) program (Atherton, 1987). This core consisted of seed fuel assemblies inserted into annular-shaped blanket fuel assemblies, loaded with ^{233}U and thorium, respectively. The reactivity control was achieved by mechanically moving the supercritical seed units in or out of blanket units. As can be observed from Fig. 4, ^{233}U has the largest number of neutrons released on average per neutron absorbed in the thermal energy range. This, in combination with exceptionally good neutron economy thanks to variable geometry reactivity control, allowed the core to achieve net breeding (i.e., $\text{CR} > 1$), which was then experimentally verified through post irradiation examination of LWBR spent fuel.

The idea of seed-blanket configurations has been recently revisited with the aim of achieving high CR, as well as other objectives discussed later in this article, such as improved proliferation resistance of spent LWR fuel.

In relation to the thorium fuel cycle, it has been shown that mixing enriched uranium and thorium homogeneously within LWR fuel does not provide any benefits with respect to uranium savings or extended burnup (Galperin et al., 2002). Heterogeneous, seed-blanket designs, on the other hand, can be optimized to achieve high CR and notably improve fuel utilization in a once-through fuel cycle, or achieve near-breeding performance in cores intended for multi-recycling of fissile material. Spatial separation can be implemented on a micro-scale, e.g., fissile and fertile regions within a single fuel pellet (Shwageraus et al., 2004), or macro-scale, on the order of full fuel assembly size (Baldova et al., 2016; Kotlyar and Shwageraus, 2012). Heterogeneity can be in the radial (groups of pins or full assemblies) or axial (alternating axial fuel zones) directions, or both. One of the major challenges of seed-blanket designs is accommodating significant power peaks resulting from drastic differences in fissile fuel loading in different assembly regions. Coolant flow can, in principle, be distributed in proportion to the power expected to be generated in different regions. However, this is still difficult to achieve without negatively impacting the core average power density. The addition of even small amounts of fissile material to the blanket in order to flatten the power distribution reduces the CR significantly.

All the methods applied to an LWR once-through fuel cycle described so far in this section are capable of only relatively modest improvement in fuel utilization—on the order of 20–30%. An order of magnitude greater improvements can be achieved if a high CR or even a breeder core is combined with a closed fuel cycle infrastructure. Multiple irradiated fuel components can be recycled. These include the remaining uranium with depleted fissile content, plutonium with a high fraction of fissile isotopes, as well as minor actinides, some of which are fissile and, thus, have high energy value. Recycling of spent fuel is discussed in more detail in Shwageraus (2021).

At the moment, one of the most advanced LWR concepts, under active development for the past 20 years, is the Resource-renewable Boiling Water Reactor or RBWR (Takeda et al., 1995; Hino et al., 2015).

The concept uses a tight-pitch hexagonal fuel lattice as well as a higher-than-conventional BWR void fraction to achieve reduced moderation and increase the CR. Furthermore, spatial separation of fissile and fertile fuel zones in the axial direction is employed. In addition to improving the CR, heterogeneous arrangement serves the purpose of achieving negative void reactivity feedback.

In reduced moderation, plutonium-driven lattices, the coolant density coefficient of reactivity can be positive, especially for fuels with high plutonium content (Aniel et al., 1997). The positive void reactivity effect can be counteracted by a negative neutron leakage effect. With loss of moderation due to voiding, the probability of neutron loss due to leakage increases, which reduces the core reactivity. The leakage effect can be amplified by changing the core aspect ratio to increase the surface area through which the neutrons can leak. For large cores, however, this strategy has limited benefit because the fraction of leaking neutrons is relatively small and because, after all, neutrons are needed for sustaining the core criticality for as long as possible to achieve favorable economics and for efficient breeding of new fissile material. Neutrons lost to leakage would negatively affect both.

In the RBWR, the core is much shorter than, for example, in the ABWR (1.3 vs. 3.8 m), while having the same equivalent diameter, making it significantly more leaky. The fuel is arranged in an axially heterogeneous “parfait”-like core, where fissile (seed)

layers—loaded with MOX fuel—are sandwiched between fertile layers—loaded with natural or depleted uranium—as schematically shown in Fig. 5. Upon an increase in the core void fraction resulting in spectrum hardening, the neutron leakage from fissile into fertile zones will increase, reducing the core reactivity. The leaking neutrons in this case are not lost, but are absorbed in the fertile blankets.

The RBWR concept can be applied not only in the context of fuel multi-recycling to improve its utilization manyfold, but also for efficient burning of Pu and MA if this is chosen to be the fuel cycle objective. Furthermore, the RBWR can operate in a closed U^{233} –Th fuel cycle with some modifications to the core design (Greenspan et al., 2015).

The concept is particularly attractive because it essentially allows for achieving many of the Gen-IV goals while using well-established LWR reactor technology. This is, of course, provided that the closed fuel cycle infrastructure is available and can operate cost-effectively. RBWR core modifications (for the U–Pu or thorium fuel cycles, breeder or actinide burner) are designed to fit into an ABWR pressure vessel, and have the same operating characteristics—such as system pressure, feed water flow rate and total power—so that the conventional ABWR power plant can be used. The RBWR development and design studies have been mostly theoretical, supported by experiments in key areas, such as boiling thermal-hydraulics in tight lattices and axially heterogeneous heat fluxes. However, there are plans to establish an experimental program based on the findings of the studies performed so far.

Simplified safety case

The challenges of LWRs described earlier in this article have been known for many years and recognized by the industry—reactor designers, operators, and regulators. Numerous attempts have been made to address these issues so that the safety case can be simplified, reducing the costs of nuclear energy and improving public acceptance. The proposed new features that could achieve the simpler safety case objective can be broadly grouped into the following categories:

- Reliance on passive mechanisms to assure reactor shutdown and residual heat removal without the need for off-site power. This can be accomplished through natural convection, gravity-driven flows, and large quantities of cooling water stored in close proximity to the core, so that availability of a heat sink is assured for prolonged periods of time. Natural circulation of coolant could be employed only for removing the decay heat or used also in nominal operation at full power. Electric batteries for instrumentation and valve actuation, as well as compressed gas energy storage for fast, high-pressure, coolant injection may still be needed, but the requirement for Emergency Diesel Generators to move the decay heat to the ultimate heat sink will not be necessary.
- Integral design of a PWR Nuclear Steam Supply System (NSSS), in which steam generators are located inside the reactor pressure vessel. This allows eliminating a whole family of faults associated with breaking of the primary coolant circuit piping and a subsequent loss of coolant.
- New fuel forms and cladding materials which can survive the loss of core cooling for prolonged periods of time without losing their integrity and releasing fission products.

One of the earlier concepts worth mentioning is PIUS—Process Inherent Ultimate Safety (Hannerz, 1987). It was under development in the 1970s as part of a collaboration between Sweden and Finland to develop an inherently safe LWR called SECURE for district heating and electric power generation. Part of the PIUS primary loop containing the core is submerged in a large pool of cold borated water. The loop has openings into the pool below and above the core but, under normal operation, the primary water does not mix with the pool water. The mixing is prevented by so-called density locks (interfaces between cold and hot water), which are kept in place if the primary loop temperatures and flow rate are within the normal operation window. In case of power loss to the

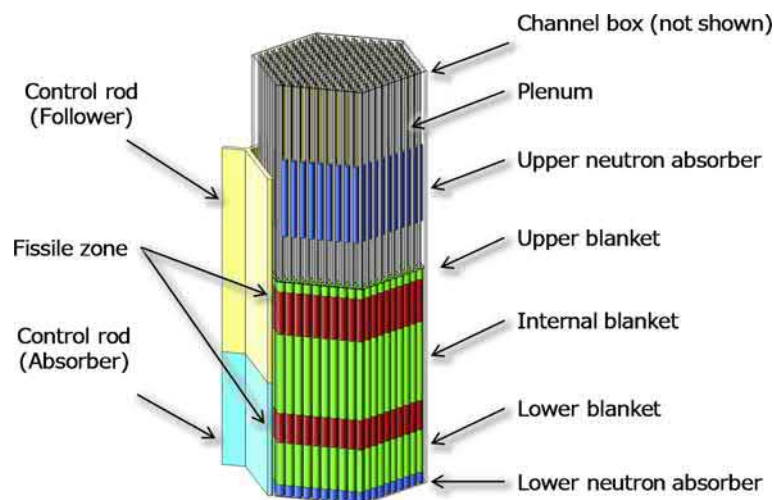


Fig. 5 Schematic view of an RBWR fuel assembly. Courtesy Hitachi 2020 used with permission.

primary coolant circulation pumps, a natural circulation is passively established between the core and the pool water. The concept can also be extended to BWRs (Forsberg, 1986).

Another concept is an early version of an integral PWR called SIR—Safe Integral Reactor (Matzie et al., 1992). It was developed as a collaboration between ABB Combustion Engineering, Rolls-Royce, Stone & Webster and AEA Technology. Apart from placing all the primary coolant loop components into a single pressure vessel to eliminate large pipe break loss of coolant accidents, the design also featured a compact pressure suppression containment—a feature previously common only to BWRs. Safety was achieved by primarily passive means through natural convection and without the need for EDGs.

Many of the features mentioned above have already been adopted in current Gen-III+ designs such as the AP-1000 and ESBWR. Both designs benefit from fast, automatic depressurization, passive decay heat removal by natural convection, and a large volume of cooling water near the core. The ESBWR also has no coolant pumps to force the coolant flow through the core in normal operation, relying instead on natural convection. Due to simplification and passive safety features, both designs claim significant reductions in the number of needed pumps, piping, and valves.

The first four AP-1000 power plants have recently been brought online in China, and two more are under construction in the United States at the time of writing. The reported financial, regulatory, manufacturing, and project management difficulties of the first units can be attributed to the construction of first-of-a-kind plants. Therefore, it would be somewhat premature to speculate whether the promise of cost reduction due to the simplification of safety systems has been realized. There are no firm ESBWR orders at the moment, but the design is mature, gaining the US NRC certification in 2014. Several sites in the US either applied or were already issued a combined Construction and Operating License (COL).

Simplification of the safety case is likely to be easier to achieve if the physical size of the reactor is small. It is easier to accommodate large quantities of cooling water, substantially increasing the volumes of primary and emergency cooling systems per unit area of fuel to be cooled. It is also possible to submerge the entire primary vessel in a large pool, serving as a heat sink, or integrate all the primary circuit components into a pressure vessel. Therefore, multiple Small Modular Reactor (SMR) designs taking advantage of the above features have been proposed and are currently under active development. Further chapters (Reitsma, 2021; Ingersoll, 2021) dedicated to SMRs describe their designs in more detail.

Integral designs of large PWRs do not have quite the same level of maturity as some of the SMRs. A conceptual design of a large integral PWR has recently been studied by a consortium of US universities, Westinghouse, and a number of international partners. The Integral Inherently Safe—I²S-LWR has 1GWe power output (Memmott et al., 2017). It is extremely challenging to fit all the primary components into the RPV. Therefore, integral, compact heat exchangers are used to transfer heat to the secondary pressurized water, which is then directed into “flashing drums” where the pressure is reduced, and a large fraction of water is flashed into steam. The remaining water is pressurized again by an external pump and directed back into the RPV. The decay heat removal from the I²S-LWR is fully passive and accomplished through dedicated natural circulation heat exchangers also located in the RPV. The core has 121 fuel assemblies with a larger than conventional number of fuel pins (19×19), providing a large cooling surface area and allowing the I²S to achieve about 20% higher power density than in conventional PWRs. The fuel is uranium silicide (U_3Si_2) which has a higher heavy metal density and thermal conductivity than UO_2 .

Advanced fuels with high thermal conductivity and large surface area have already been mentioned in relation to increasing the core power density and, thus, achieving better economics. In this section, it is worth mentioning advanced cladding materials also, which can maintain their strength and resist oxidation at high temperatures. Since the Fukushima accident in 2011, a search for more oxidation-resistant cladding materials became a particularly active area of research. Traditional Zr-alloy-based cladding poses a number of safety challenges: it loses its mechanical strength dramatically when heated beyond approximately 700 °C; it exothermically reacts with steam at temperatures above 1200 °C, generating heat which is comparable, or even up to an order of magnitude higher than decay heat a few hours after LOCA initiation; hydrogen gas is generated during oxidation which can accumulate and combust with an additional large and rapid energy release.

A number of options are currently being pursued as advanced cladding materials for Accident-Tolerant Fuel (ATF) with various degrees of technological maturity (Terrani, 2018). The simplest and the most near-term solution is oxidation resistant coating (e.g. chromium) of traditional Zr-based cladding. Fuel qualification work on the coated cladding is well on the way by many fuel vendors. The first commercial products are expected to be available within the next few years. This concept, however, does not resolve the problem of having large amounts of Zr in LWR cores which would still represent a significant source of stored energy and which can be released if high temperature water does get in contact with Zr.

A somewhat more advanced concept is Fe-Cr-Al alloys. They have a similarly low oxidation rate as Zr-coatings. However, in this case, the exothermal Zr oxidation energy source is eliminated entirely. The cladding is also stronger and can maintain its integrity in relevant accident scenarios for up to several hours longer. High neutron absorption compared to Zr is a disadvantage and would require a slight enrichment increase to compensate.

A possible “gamechanger” technology is based on SiC/SiC composite materials. Fibers of SiC can be weaved into a desired tubular shape and infiltrated with crystalline SiC, forming an air-tight matrix. It can then be combined with either additional monolithic SiC layers or metallic layers, so that the hybrid material would have an optimal combination of all the desirable properties. SiC/SiC composites have shown high strength and good oxidation resistance under variety of conditions. SiC also has low neutron absorption and it is a good moderator. If the behavior of such materials is well understood and manufacturing techniques mastered, the ceramic composite cladding can potentially survive without cooling for sufficiently long periods of time to convince regulators around the world that the reliability and redundancy requirements on LOCA protection equipment can be relaxed. Before these advantages can be realized, however, further development is still required in areas such as radiation-resistant bonding of cladding

components, understanding of micro-crack formation mechanisms and migration of fission products through them, as well as the development of fracture mechanics codes and standards necessary for qualification of ceramic cladding materials—these currently exist only for metals (Terrani, 2018).

SiC has also been successfully used as the coating of small (about 0.5 mm in diameter) spherical UO_2 kernels. These are known as TRISO fuel particles. The particles can be either embedded into graphite, 6 cm diameter spheres (pebbles), or pressed into cylindrical rodlets (compacts), for use in gas- and salt-cooled high temperature reactors. TRISO particles have been shown to be incredibly robust, maintaining integrity of the coatings and retention of fission products at up to 1600 °C (IAEA, 2010).

TRISO particles embedded into SiC matrix pellets can be loaded into a SiC composite cladding tube. An additional layer of protection (particle coating plus cladding tube) and elimination of many undesirable and difficult to predict fuel irradiation effects, such as cracking, swelling and fission gas release, can provide even further accident tolerance. This concept is known as Fully Ceramic Microencapsulated (FCM) fuel (Terrani et al., 2012). Although technologically feasible, manufacturing of such fuel is likely to be costly. An additional complication is much lower fuel heavy metal density because fuel is displaced by the TRISO coatings and the matrix. The enrichment required to compensate for such a loss of heavy metal loading can be as high as 19% (Shapiro and Frantoni, 2016), which would further escalate the costs of FCM fuel manufacturing. However, these costs may well be offset by the safety case simplification as alluded to previously.

Non-power applications of LWRs

Expanding a nuclear energy contribution to the decarbonization of the world economy is currently a matter of urgency. It is therefore highly desirable for nuclear reactors to offer other energy products in addition to the generation of electricity. Non-power applications also represent a potentially attractive commercial opportunity to improve the economics of nuclear power. A number of such non-power applications are possible. Many have been hypothesized—some of them became mature designs—others have actually been constructed and operated commercially.

LWRs operate at relatively low temperatures—around 300 °C. This feature somewhat limits the extent of non-power applications. Energy intensive industrial processes typically require much higher temperatures, achievable only in gas-cooled or other Gen-IV reactors (Stanculescu, 2021). This is also the case for high-efficiency hydrogen generation methods. At room temperature, electrolysis of water is possible, but is much less efficient and therefore expensive. That being said, such a hydrogen production plant could be completely decoupled from the nuclear power plant, which would only supply the electricity. A hydrogen plant can be thermally coupled to an LWR (Frick et al., 2019). In this case, the low-grade nuclear heat can be topped up with resistive heating by electricity or other means and then used in a much more efficient high-temperature solid oxide electrolyzer cell (SOEC). However, the fractional contribution of low-grade nuclear heat to the total energy consumed in hydrogen generation is modest (on the order of 10%). A number of studies have been conducted around the world on using existing LWRs for hydrogen production employing these strategies and several reactor operating companies expressed interest in pilot projects. The biggest technical challenge at the moment is degradation of the electrolyzer cell electrodes, which is currently a topic of particularly active research.

The most attractive commercial opportunities for using relatively low-grade LWR heat efficiently lie in sea water desalination and district heating. Both have been proven successful with several operating commercial prototypes. Many countries face a scarcity of water resources or envisage such scarcity in the future. However, currently only a small proportion of the world desalination capacity is met by nuclear power. A similar situation exists with district heating applications.

Multiple options for coupling nuclear and desalination plants are available. Reverse Osmosis (RO) methods require only electricity, although the efficiency of the process can be boosted by about 10–15% if feed water is preheated. This is a mature, widely-deployed technology.

Part of the nuclear-generated steam can be diverted at various stages of power turbines and used directly to drive thermal desalination processes, such as multi-stage flash (MSF) and multi-effect distillation (MED). The latter was shown to be somewhat more cost-effective (IAEA, 2007). Thermal processes also have an advantage of producing an order of magnitude lower salinity product than RO. Thermal processes can be combined with RO and offer greater flexibility for the plant to follow the variations in electricity demand, which can be particularly attractive in electricity grids with a high penetration of intermittent renewable energy generators.

Many northern countries have domestic heating as a large fraction of their energy consumption. Some have well-developed district heating networks, transporting hot water over long distances (over 10 km) with modest temperature losses (about 1 °C/km). Numerous LWR plants have been used successfully to provide domestic heating in addition to electricity.

Some concepts developed in the past were designed exclusively for the purpose of district heating. One of the main concerns with such reactors is the proximity to population centers. This can be addressed by low-temperature, low-pressure operating conditions, as well as passive safety features. In fact, the inherently safe SECURE concept mentioned earlier in this article was originally developed for district heating and with these safety concerns in mind (Leppänen, 2019). The most active R&D on dedicated district heating reactors is currently ongoing in China. Government companies, CNNC and CGN, are pursuing alternative designs in cooperation with Tsinghua university. CNNC is developing a 400 MW deep pool-type reactor (Jiafu et al., 1998), known as Yanlong, which does not require a pressure vessel and operates at low temperatures of less than 100 °C. CGN is pursuing a pressure vessel-type design, NHR200-II (Wenxiang and Dazhong, 1995), which has already obtained design certification from the Chinese regulatory authority, with plans in place for the construction of the first units.

LWRs have also been used for marine propulsion, although mostly for military applications. Commercial shipping currently accounts for a non-negligible fraction of global carbon emissions, therefore, it is an attractive market for advanced LWRs. The

requirements of compactness (small physical size) and long periods of operation without refueling lead to the choice of highly enriched uranium fuel in most military marine propulsion applications. For merchant ships, the enrichment will have to be lower than 20% to stay within the non-proliferation limit. A number of high-level conceptual design studies have been conducted recently, demonstrating basic design trade-offs. There are no firm follow-up plans for further development and commercialization of the technologies. Marine reactors fall into the SMR category and will be discussed in further detail in [Reitsma \(2021\)](#).

Proliferation-resistant LWR fuel

With regards to the fuel cycle front-end, LWRs can be considered more proliferation-resistant than many of the Gen-IV reactor concepts simply because they use lower enriched fuel. As for the back-end, most existing LWRs operate with an open fuel cycle, meaning that Pu and other fissile actinides remain in spent fuel which, following a period of cooling in reactor ponds and interim storage, is intended to be disposed in geological repositories. It has been argued that any isotopic composition of Pu originating from LWR spent fuel is potentially suitable for weapons use ([Mark et al., 2009](#)). It does, however, become more technologically challenging to use it if it has a less favorable isotopic vector and less of it is produced per unit energy generated. A number of strategies can be employed to reduce the quantity and attractiveness of Pu in spent LWR fuel.

Pu recycling would improve uranium resource utilization, but it can also be viewed as a strategy to mitigate the proliferation potential of spent LWR fuel. This is provided that the risks of Pu diversion during reprocessing and re-fabrication are deemed lower than those of direct disposal. Fuel recycling strategies are discussed in more detail in [Schwageraus \(2021\)](#).

The simplest approach to reduce the attractiveness of Pu in LWRs is to increase fuel burnup. As already mentioned, the burnup limits are primarily determined by Zr cladding performance (corrosion and hydrogen pick-up). However, with the development of new cladding materials discussed earlier, fuel burnup can be extended substantially. This will most definitely require higher initial fuel enrichment, which is currently limited to 5%. The fuel manufacturing and handling infrastructure can, in principle, be re-licensed, but at an additional cost, which can be prohibitive.

The total amount of Pu in LWR fuel grows monotonically with burnup, but this growth is not linear ([Fig. 6](#)). It is higher in the beginning and slows significantly towards the end. This is because the ^{239}Pu isotope concentration saturates with time as a result of its relatively constant generation rate being gradually matched by its destruction rate as fissions shift from ^{235}U to ^{239}Pu . Such non-linear behavior implies that, with higher burnup, less Pu will be generated per unit of energy produced by the fuel. Furthermore, all Pu isotopes that make it less attractive increase in quantity. The fraction of ^{238}Pu isotope, for example, which is mostly responsible for decay heat, more than doubles from about 3% at 50 MWd/kg to more than 7% at 100 MWd/kg ([Xu et al., 2005](#)). High decay heat generation of Pu with a high fraction of Pu^{238} is often cited as a particularly effective deterrent against proliferation. Studies have shown that isotopic mixes with ^{238}Pu comprising more than about 6% would be impractical to use in weapons, even for technologically advanced agents ([Kessler, 2007](#)). Even-numbered isotopes (^{240}Pu and ^{242}Pu) with high spontaneous fission source also increase with burnup, which would reduce the reliability of a weapon (reduce the probability of delivering a desired energy yield).

The concentration of ^{238}Pu in irradiated fuel can be increased by a variety of means. Many actinides would include ^{238}Pu in their transmutation chains under typical LWR conditions. Increasing the initial loading of these nuclides would result in larger quantity of ^{238}Pu after irradiation. Americium-241 is a product of relatively rapid β -decay of Pu^{241} and would inevitably be present in reactor-grade Pu. If separated ^{241}Am is available from countries practicing fuel reprocessing or as a by-product of warheads maintenance in nuclear weapons states, it can be added in small quantities to commercial LWR fuel and increase the end of life ^{238}Pu fraction ([Ronen et al., 2010](#)). Less than 0.2% of ^{241}Am would be sufficient to ensure over 6% of ^{238}Pu in the fuel end of life Pu isotopic vector. ^{241}Am has an additional advantage of having a high neutron capture rate initially, but, with irradiation, it depletes

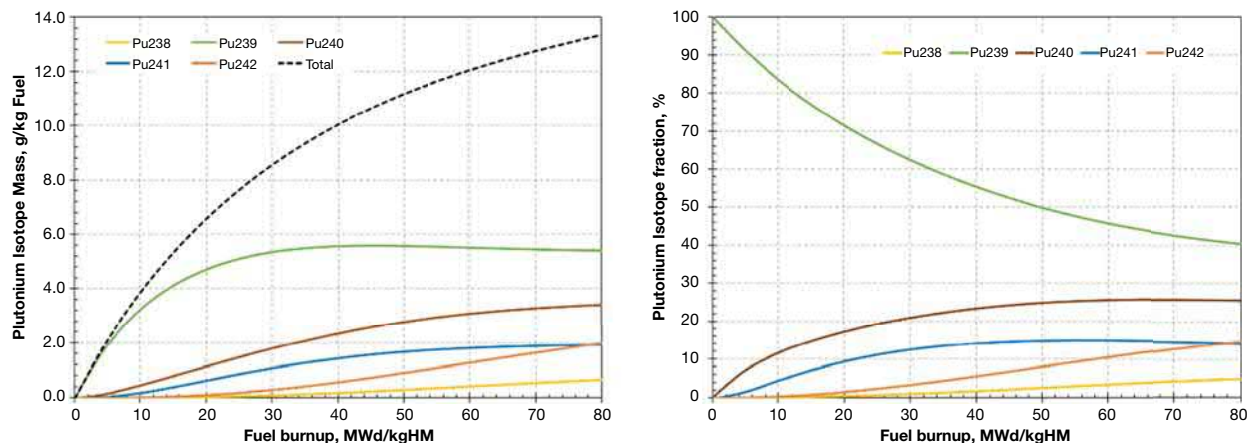


Fig. 6 Evolution of Pu isotopes with burnup in a typical PWR fuel.

while generating highly fissile ^{242m}Am , increasing the fuel reactivity towards the end of life. It essentially serves as a fertile burnable poison, helping to reduce the fuel reactivity swing and reactivity control requirements.

Other additives to fresh LWR fuel can perform a similar function. It was shown that recycled and re-enriched uranium—with or without an addition of either recycled reactor grade Pu, Pu + MA or thorium—in a slightly modified fuel lattice would result in discharge fuel Pu vector with Pu 238 comprising more than 6% (Kessler, 2017). Doping LWR fuel with the mentioned additives, however, has never been considered commercially and these studies remain, at the moment, in the realm of academic research.

Replacing some or all of the fertile ^{238}U in fresh LWR fuel with thorium can drastically reduce the total quantity of Pu generated in their operation because thorium is located lower than uranium in the Periodic Table. A typical modern LWR would generate between 200 and 250 kg of Pu per GW $_e$ -Year of energy it produces. In practice, however, ^{238}U cannot be eliminated entirely from the fresh fuel because natural thorium (Th 232) is not fissile. Therefore, ^{232}Th has to be primed with a fissile component. The two most practical choices are enriched uranium or recycled Pu from other reactors or dismantled weapons. In the former case, the enrichment cannot be higher than 20% for the uranium itself not to become a proliferation concern. This implies that the remainder of uranium will have to be ^{238}U . In the latter case, fissile ^{233}U will be generated from ^{232}Th , which is also a weapons usable material with an even smaller critical mass than ^{235}U . In order to avoid the proliferation risk of ^{233}U diversion, the fresh fuel must contain some ^{238}U , so that, at the end of life, the content of ^{233}U in the uranium isotopic mixture will be less than 12% (which is equivalent to 20% enrichment limit for ^{235}U – ^{238}U mixtures, (Forsberg et al., 1998)). In both cases, the presence of ^{238}U is required, leading to the generation of Pu in spent fuel.

Another complication of using thorium in LWR fuel for improving its proliferation resistance is economic. ^{233}U tends to be generated from Th in LWRs much slower than Pu from U 238 . Therefore, in order to extract comparable amounts of energy from ^{233}U fissions in Th-containing fuel as from Pu in conventional enriched uranium fuel, one needs to keep the fertile Th part in the core for as long as possible. The initial reactivity of Th-containing fuels is typically lower because of the higher thermal absorption cross-section of ^{232}Th compared to ^{238}U . This initial investment of neutrons can be “repaid” by later fissions in ^{233}U , but only if a sufficient amount of ^{233}U is generated by allowing longer Th irradiation.

As mentioned earlier in this article, a homogeneous mixture of Th and enriched uranium does not lead to uranium savings (Galperin et al., 2002). The Pu production rate for such fuel, however, will be approximately halved, and the proliferation attractiveness of the Pu isotopic vector in spent fuel will also be slightly reduced. Physical separation of the fertile Th blanket and fissile uranium seed parts of the fuel on the micro-scale of fuel pellets in axial and/or radial direction increases the contribution of ^{233}U fissions to the total energy generated, increasing the fuel burnup (Schwageraus et al., 2004). As a result, the Pu generation rate reduces further, reaching about one third that of conventional uranium LWR fuel.

Macro-scale separation of fissile seed and fertile blanket parts of fuel assemblies can be realized in the form of a Seed-Blanket Unit (SBU), proposed by Radkowsky as an extension of the Shippingport LWBR idea. An additional advantage of the approach is that it allows refueling the seed and blanket parts with different frequencies. This is to allow the thorium blanket to remain in the core longer, further increasing the energy return per unit mass of initial fissile material invested (Radkowsky and Galperin, 1998). The macro-heterogenous configuration can be also realized on a scale of full conventional 17×17 pin fuel assemblies, as shown in Fig. 7 (Todosow et al., 2005). Such seed-blanket arrangements can reduce the Pu production rate to as little as one quarter that of conventional LWR fuel.

As mentioned in the discussion of high-conversion seed-blanket concepts, the main technological challenge of such configurations is to manage a much larger power imbalance between the seed and blanket regions than in typical LWRs, especially at the beginning of fuel life, when fission power in the blankets is small. Addition of fissile material to the blankets drastically reduces

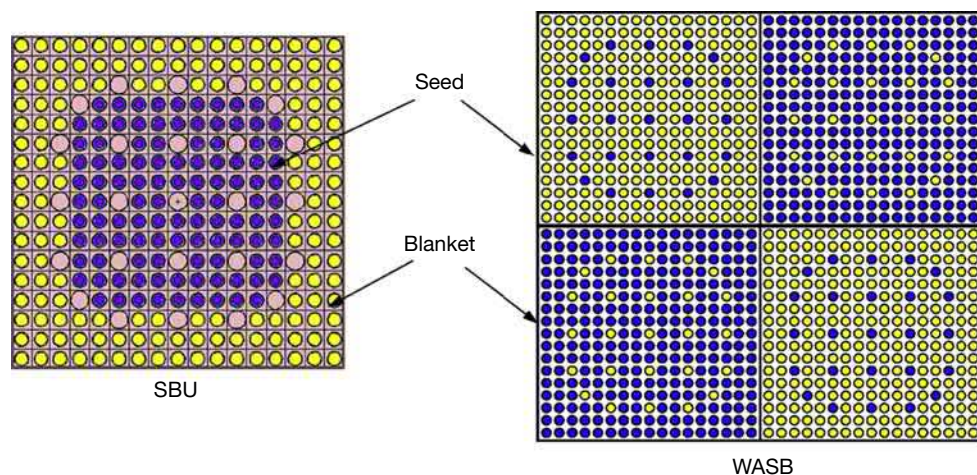


Fig. 7 Schematic view of seed-blanket unit and whole-assembly seed blanket (WASB) configurations.

the effectiveness of spatial separation physics and leads to significant deterioration of both high fuel utilization and proliferation resistance performance characteristics.

Nuclear waste management

The issue of high-level waste management has many economic, political, environmental and other aspects, the nature of which is discussed in more detail in Metlay (2021) and Kadak (2021). Here, we simply address how LWR technology can be modified to improve some of the waste management characteristics of the fuel cycle. In a once-through fuel cycle, spent fuel assemblies are first cooled in water ponds at reactor sites for a number of years. Then, they are either moved to a temporary dry-cask storage or directly to geological repositories—if and when they become available. The costs of these storage activities strongly correlate with volume, decay heat load, radioactivity and radiotoxicity of spent fuel. Therefore, there is a clear incentive to reduce all of them. Decay heat affects the density of waste packages for transportation and storage, radioactivity affects the shielding, while radiotoxicity is relevant to the dose rates received by the public in case the content of waste packages leaks at any stage of handling or storage. The radioactivity, decay heat, and radiotoxicity are obviously closely correlated, but not exactly proportional to each other as different isotopes have a different energy release per decay and different radiotoxicity dose conversion coefficients.

The simplest option to reduce these characteristics is to increase fuel burnup. Fewer fuel assemblies would need to be discharged per year of reactor operation. The mass of fission products, however, is directly proportional to the energy produced (burnup). Therefore, each fuel assembly would be hotter, having implications for the costs of storage, transportation and handling.

Similar to the build-up of Pu isotopes, some fission products saturate with irradiation faster than others, leading to their non-linear accumulation. As a result, less decay heat is generated per unit energy produced by the fuel, at least initially, as shown in Fig. 8 (Xu et al., 2005). This is the period relevant to spent fuel pond storage. Although high-burnup assemblies would generate more heat, there will be fewer of them, potentially allowing an increase in pond storage capacity. Even though the total amount of FPs per unit energy produced is smaller, the isotopic mix of FP nuclides is different for high- and low-burnup fuel. This leads to a flip in the trend a few years following discharge and, shortly after, the high-burnup fuel at 100 MWd/kg would have up to 25% higher decay heat than 33 MWd/kg fuel (Fig. 8). After the decay of FPs, the decay heat is dominated by Pu and MAs, with the decay heat behavior favoring the high burnup between a few hundred to 10^5 years—the period of relevance for geological repository design.

As mentioned in the earlier discussion on non-proliferation, similar considerations apply to what extent LWRs' fuel burnup can be increased. Advanced fuel and cladding materials can certainly enable a transition to higher burnups, provided that higher than 5% enriched fuel becomes commercially available.

Thorium-based fuels in a once-through fuel cycle are often viewed as a more favorable option with respect to high-level waste management. This is based on the observation that, in conventional uranium-based fuel, the radiotoxicity and decay heat in the long term is dominated by the decay of Pu and MAs. In thorium-based fuel, many more successive neutron captures would be required to produce Pu and MAs. Therefore, the long-term characteristics of spent fuel can be improved. This, unfortunately, turns out not to be the case for a number of reasons.

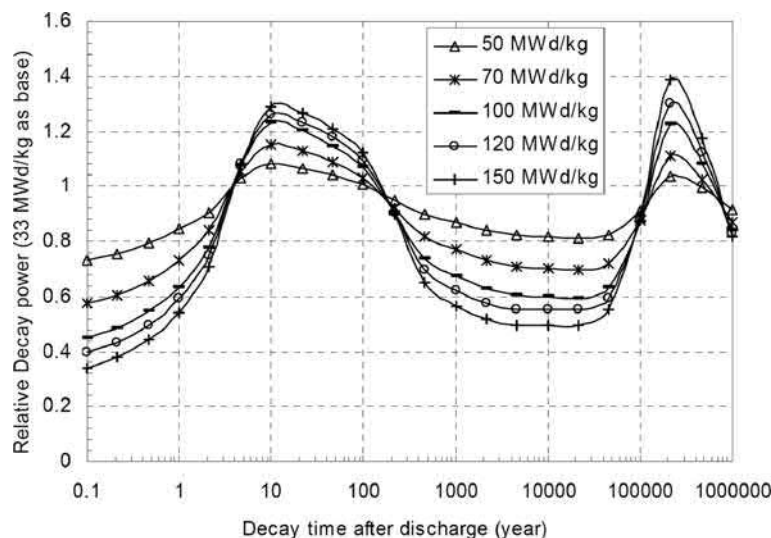


Fig. 8 Decay heat power of high-burnup PWR fuel per GWe-Year relative to low-burnup fuel. Xu Z, Kazimi MS, and Driscoll MJ (2005) Impact of high burnup on PWR spent fuel characteristics. *Nuclear Science and Engineering* 151 (3): 261–273.

As mentioned earlier, thorium fuel will inevitably contain some uranium to mitigate the power imbalance in heterogeneous designs and dilute the generated ^{233}U down to the non-proliferation limit. At least 80% of that uranium has to be ^{238}U (due to the 20% limit on enrichment), which will generate considerable amounts of Pu and MAs.

Just like in uranium fuel, the short-term decay heat in thorium-based fuels is dominated by fission products rather than actinides. Although the isotopic mix of FPs is slightly different, the overall behavior is very similar for the two fuels, exhibiting similar trends as noted above in the high-burnup fuels discussion.

Although the quantities of Pu and MA in the thorium transmutation chain are practically negligible, it does contain long lived ^{233}U and ^{231}Pa , which would contribute substantially to radiotoxicity in the long-term.

Overall, thorium-based fuels in homogeneous, or micro- or macro-heterogeneous geometries introduced earlier, would have a relatively modest effect on the short- or long-term spent fuel characteristics, and are unlikely to simplify or reduce the costs of spent fuel storage and disposal substantially. In the most promising macro-heterogeneous seed-blanket arrangements, the number of discharged fuel assemblies per $\text{GW}_e\text{-Year}$ can be reduced by up to a factor of two compared to typical LWR fuel. The decay heat per unit energy is also lower by up to 40% for the first 5 years after discharge, becoming essentially identical to that of UO_2 for the next 100 years—the period most relevant to interim storage and geological repository construction costs. Thorium fuel decay heat becomes notably lower than UO_2 again in the period from 100 to 20,000 years with the largest difference of up to a factor of two at about 2000 years. In this period, however, the decay heat is less relevant and the dose rate (or radiotoxicity) of waste becomes a concern. Beyond 20,000 years, thorium fuel has up to an order of magnitude higher radiotoxicity, peaking at about 10^5 years exclusively due to thorium transmutation chain nuclides. (Tudosow et al., 2005).

The once-through fuel cycle options discussed above can offer only marginal gains with regards to simplification and cost reduction of high-level waste management. In order to realize a substantial benefit, long-lived actinides and fission products would need to be kept within a closed fuel cycle, that is, away from a geological repository. A major fuel cycle study in the United States, for example, has concluded that removing 99.9% of Am and Pu from spent fuel would increase the design capacity of Yucca Mountain repository by a factor of 5.4, while removing, in addition, Cs and Sr fission products—which dominate the short-term decay heat—could increase repository capacity by a factor of 43 (Wigeland et al., 2006) as compared to the direct disposal of LWR fuel without reprocessing.

In order to realize these gains, two attributes are necessary. First, an available fuel reprocessing infrastructure capable of keeping the TRU nuclides within the fuel cycle with minimal losses, high separation efficiency, and tolerance to high radiation and decay heat load of TRU isotopic mixes after multiple recycle passes. Secondly, one would need a reactor system capable of indefinite recycling of TRU nuclides with transmutation efficiency sufficient to constrain their growth. It has been established through multiple national and international studies that Gen-IV fast spectrum reactors are the preferred choice for this mission. LWRs, although less efficient than fast reactors in handling TRUs, are still capable of actinide transmutation and recycling. One such example is the RBWR, already mentioned in the context of LWRs with a high conversion ratio. In fact, LWRs operating in a closed fuel cycle can simultaneously accomplish multiple Gen-IV objectives, i.e., sustainability of resources, non-proliferation and long-term waste management, all using existing and well-proven reactor technology. Various aspects of fuel recycling in LWRs are addressed in further detail in Schwageraus (2021).

Chapter summary

LWRs are a well-understood and mature technology, ready for a large-scale worldwide deployment. Substantial improvements have been made to their design, manufacturing, and operating practices, with remarkable gains in efficiency, reliability, and safety. Modern Gen-III+ LWRs have pushed the boundaries on all fronts of the design and operational performance. Design simplifications and the adoption of passive safety systems have reduced the probability of core damage and large radioactive releases by more than two orders of magnitude over the past few decades. In light of these observations, it has been speculated that LWRs have practically reached their theoretical limits. Therefore, fundamentally different, new Gen-IV technologies are needed to make substantial, rather than incremental further progress in tackling the societal issues faced by the nuclear industry.

This article summarized and discussed some of the recent and ongoing research into advanced LWR designs with potentially high impact, showing that LWRs still have much to offer in each of the areas identified by the Gen-IV International Forum as objectives for the development of new systems.

It has been shown that LWRs can significantly increase the temperatures of heat delivery to their power conversion system, improving their thermodynamic efficiency. Their core power density can be increased, leading to capital cost reduction. Fuel burnup can be extended through employment of advanced fuel and cladding materials, as well as new geometrical arrangements, offering significant gains in safety, fuel utilization, proliferation resistance, and waste minimization. LWRs with a high conversion ratio operating on U–Pu or U^{233} –Th fuel cycles can achieve orders of magnitude higher fuel utilization than the current practice if fuel reprocessing infrastructure is available. LWRs capable of indefinite recycling of all trans-uranic nuclides would meet not only the resource utilization objective but, by keeping the Pu and MA within the fuel cycle, would also address the non-proliferation and long-term waste radiotoxicity concerns associated with direct disposal of spent fuel. Accident Tolerant Fuels in combination with passively safe designs can potentially offer significant capital cost savings through elimination of some of the safety systems and a simpler safety case, shortening the time necessary for licensing and construction. Finally, LWRs can be used to generate a variety of products other than electricity, such as district heating, hydrogen, and desalinated water, opening a wide range of commercial opportunities and increasing the contribution of nuclear energy to decarbonization of the world economy.

Even if Gen-IV reactors become technologically mature and commercially competitive in the near future, transition from an LWR fleet to new technologies would take many decades because LWRs currently being constructed are likely to have an 80-year life or even longer. Since LWRs are bound to be part of our energy system for a while, it is sensible to continue exploring and exploiting the advantages of this technology, which show promise in achieving many of the Gen-IV goals without the need to wait for Gen-IV reactors' availability and wide deployment.

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See Also: Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors; Super-Critical Water-Cooled Reactor (SCWR).

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Fluoride-Salt-Cooled High-Temperature Reactors

Massimiliano Fratoni, Nuclear Engineering Department, University of California, Berkeley, CA, United States

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Glossary

ATWS Anticipated transient without scram, a theoretical event where a major malfunction occurs and the shutdown system does not intervene.

Excess reactivity Maximum reactivity that a reactor can achieve.

Reactivity feedback Change in reactor multiplication factor upon change of an operational parameter.

TRISO Tri-structural isotropic particles of less than 1 μm in diameter, consisting of a spherical fuel kernel surrounded by multiple carbon/SiC layers with the purpose to retain fission.

Tritium Hydrogen isotope whose nucleus contains one proton and two neutrons.

Introduction

The fluoride-salt-cooled high-temperature reactor (FHR) uses solid fuel and molten salt coolant. Originally known as Advanced High Temperature Reactor or AHTR, the FHR concept was first proposed in 2003 (Forsberg et al., 2003) as the result of combining two technologies: TRISO coated particle fuel dispersed in a graphite matrix, developed for high-temperature gas-cooled reactors (HTGRs) starting in 1960s (Price, 2012; Goodjohn, 1991), and molten fluoride salt technologies developed within the molten salt reactor program conducted at the Oak Ridge National Laboratory (ORNL) in 1960s and 1970s (Haubenreich and Engel, 1970). Fluoride salt coolants provide a unique combination of properties: high volumetric heat capacity, low chemical reactivity with air and water, very low volatility at high temperature, effective natural circulation heat transfer, and high retention of many fission products (Andreades et al., 2014). When combined with coated fuel particles, they create a reactor design with outstanding safety features, high efficiency and potential economic advantages.

The following sections illustrate the main characteristics of FHR reactors, review some of the proposed design concepts, and summarize challenges and related R&D efforts.

Distinguishing features of FHRs

FHRs can be generally classified as graphite-moderated, molten-salt-cooled reactors. Nevertheless, the contribution of the salt to neutron moderation cannot be neglected. Rather, it is one of the features used to identify suitable salt options. In fact, although multiple fluoride salt mixtures can in theory be considered as coolants, $2\text{LiF}\text{-BeF}_2$, commonly referred to as flibe, is the almost universal choice due to its low neutron absorption, as long as lithium is enriched to 99.995% in ^7Li , and its large scattering cross section (Williams et al., 2006). As for any coolant, the reactivity feedback due to a change of its density is the result of opposite effects: a positive effect caused by reduced neutron absorption in the coolant and a negative effect caused by reduced moderation. The latter must be dominant in order for the total feedback to be negative, therefore, salts made of elements more absorbing and less moderating are not viable choices (Fratoni and Greenspan, 2011). Mixtures of fluorides are preferred to single-component salts for their considerably lower melting temperatures. Chloride salts, instead, are not suitable for thermal reactors due to chlorine relatively large absorption cross section. Molten salts are characterized by excellent heat transfer properties, high volumetric heat capacity, and high boiling temperature that enable near atmospheric pressure operation and ample margin to boiling in accident scenarios. The other defining characteristic of these salts is the high melting point that constraints the core's coolant inlet temperature above 500°C and requires adequate heating throughout the primary coolant system to prevent the salt from freezing. At the other end, the outlet temperature is constrained by the limits of the metallic structure in the heat exchangers and is typically set at around

700 °C. Even with such limitations, FHRs can provide heat at an average temperature higher than most other coolants and are suitable for industrial applications beyond electricity generation (Table 1).

Due to the use of TRISO particles, fuel density in the core is limited and requires enrichment typically between 10% and 20%. Although the use of thorium for breeding purposes has been considered, limited studies exist on such fuel option. FHRs are envisioned to operate at a power density of about 20 MW.m³ that is 2–5 times that of gas-cooled reactors, but lower than light water reactors and liquid metal cooled reactors (Table 1). When comparing electric power generated per unit volume of vessel, instead, FHRs value is about three times larger than the sodium-cooled reactor and less than one-third that of the PWR.

FHRs have unique safety characteristics. As the coolant outlet temperature—thus, the fuel temperature—is considerably lower than in gas-cooled reactors, extraordinarily large thermal margins are available between peak accident fuel temperatures and fuel damage temperatures. TRISO particles are an extremely robust form of fuel (Gerczak, 2021), but even in case of failure, the coolant presents high solubility for key fission products, particularly cesium, and could be credited as an additional barrier to the release of radionuclides. Operation at near atmospheric pressure combined with the salt high boiling point eliminates the possibility for accident events with abrupt coolant boiling and rapid control rod ejection as it can occur when there is a large pressure difference between inside and outside the reactor vessel as in light water reactors. Although design basis and beyond design basis events remain to be thoroughly addressed, the characteristics illustrated above combined with long neutron generation time, indicate that transient events in FHRs are expected to be slow and manageable. Finally, decay heat removal under emergency conditions can rely on the natural circulation of salt as long as adequate design solutions are implemented to prevent salt freezing.

Reactor designs

Three main designs have emerged for FHRs based on fuel geometry: a plate fuel design, a pebble bed design, and a stringer fuel design. Early designs also considered prismatic block fuel as gas turbine modular helium reactors (Potter and Shenoy, 1996; Fütterer, 2021), nevertheless, such option was quickly discarded because the average density of the fuel blocks would be lower than the salt density raveling some reactor operations like refueling (attempts to make the fuel blocks heavier adding ballasts were unsuccessful due to the excessive neutron absorption by the added material) (Casino Jr, 2006). This section summarizes.

Pebble bed designs

The Mark-1 Pebble Bed FHR (Mk1 PB-FHR), developed at the University of California, Berkeley (USA) (Andreades et al., 2016), is a 100 MWe design that features an annular pebble core cooled by flibe. The pebbles are 3 cm in diameter and use an annular fuel layer (Fig. 1). The center of the pebbles is made of porous carbon whose density varies in order to control the average pebble density. Due to their smaller size as compared to the conventional 6-cm diameter pebbles used in gas-cooled reactors (Reitsma, 2021), Mk1 pebbles have double the surface area per unit volume and half the heat diffusion length, enabling a substantial increase in power density while maintaining relatively low peak fuel particle temperature. Low fuel temperature reduces the thermal transient caused by hypothetical anticipated transient without scram (ATWS) accidents. Pebbles are slightly lighter than flibe, therefore, buoyant in the salt. They are inserted at the bottom of the core and slowly move upward driven by the salt drag and buoyancy. At the top of the core, defueling machines collect pebbles that are then reinserted into the core or discarded according to the measured burnup level. The Mk1 PB-FHR is expected to operate in continuous refueling mode with limited excess reactivity.

The internal vessel layout is illustrated in Fig. 2. Fuel pebbles form an annular bed in between two graphite reflectors. The external reflector is designed to withstand the lifetime of the plant, therefore, it is protected from radiation damage using a thin layer of graphite pebbles that are also continuously recirculated and can be replaced as needed. The central reflector is designed,

Table 1 Comparison of design parameters of a FHR with those of a 4-loop Westinghouse PWR, a gas-cooled reactor and a sodium-cooled fast reactor.

Reactor	Mk1 PB-FHR	PWR	PBMR	S-PRISM
Coolant	Flibe	Water	Helium	Sodium
Thermal power (MW)	236	3411	400	1000
Thermal efficiency (%)	42.5	32.0	43.8	38.0
Core inlet temperature (°C)	600	292	500	355
Core average outlet temperature (°C)	700	326	900	510
Core power density, (MWt/m ³)	22.7	105.2	4.8	321.1
Fuel enrichment (%)	19.9	4.5	9.6	8.9
Discharge burnup (GWd/tHM)	180	48	92	106
Reactor vessel diameter (m)	3.5	6.0	6.2	9.2
Reactor vessel height (m)	12.0	13.6	24.0	19.6
Vessel power density (MWe/m ³)	0.866	2.839	0.242	0.292

Andreades C, et al. (2014) *Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant*. Technical Report UCBTH-14-002. Berkeley: University of California.

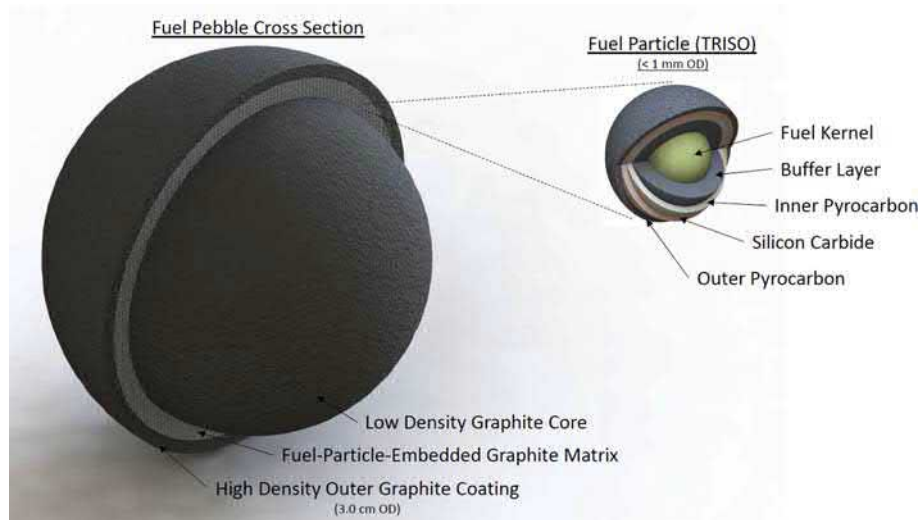


Fig. 1 Mk1 PB-FHR annular pebble design. Andreades C, et al. (2014) *Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant*. Technical Report UCBTH-14-002. Berkeley: University of California.

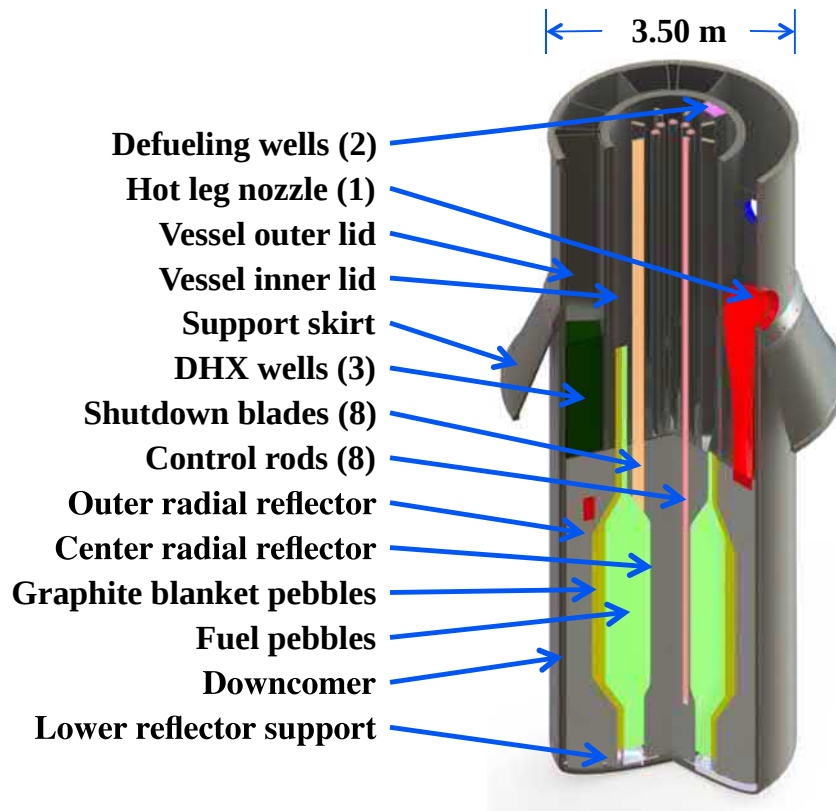


Fig. 2 Components and layout of the Mk1 PB-FHR vessel. Andreades C, et al. (2016) Design summary of the Mark-I pebble-bed, fluoride salt-cooled, high-temperature reactor commercial power plant. *Nuclear Technology* 195(3): 223–238. ISSN 00295450. <https://doi.org/10.13182/NT16-2>.

instead, as a modular unit for periodic replacement (roughly, every 10 years). The unique shape of its components (see Fig. 3) fulfills multiple functions: reduces stresses due to graphite shrinkage under irradiation, accommodates eight channel for insertion of buoyant control rods, houses 16 instrumentation guide tubes, and provides flow channels for radial coolant injection. The core barrel and other core internal structures are made of metallic material (316 stainless steel or in alternative 304 stainless steel or Alloy N) to enable near-term deployment.

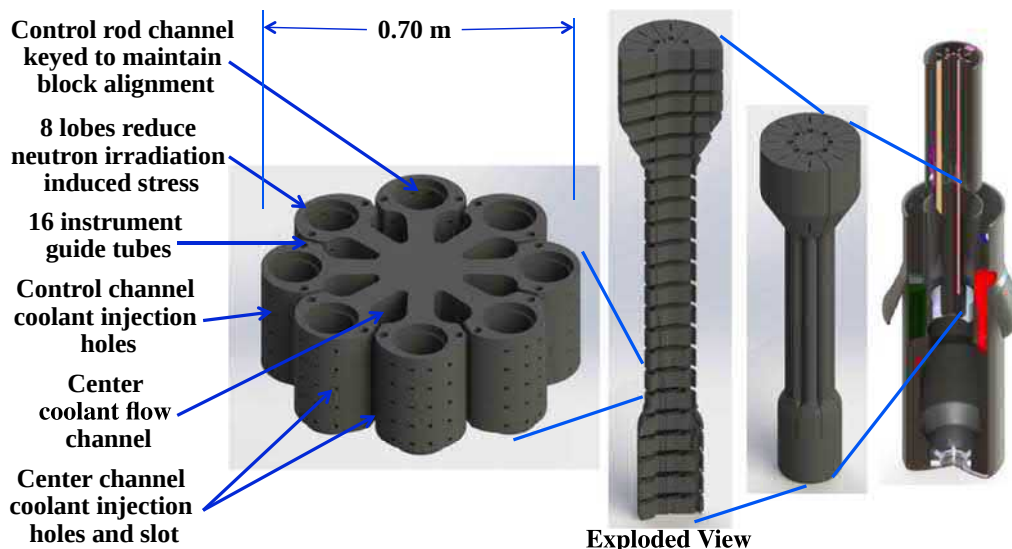


Fig. 3 Mk1 PB-FHR center graphite reactor. Andreades C, et al. (2014) *Technical Description of the “Mark 1” Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant*. Technical Report UCBTH-14-002. Berkeley: University of California.

The Mk1 PB-FHR does not use an intermediate coolant loop and instead directly heats the power conversion fluid. It uses a refractory reactor cavity liner system in place of the conventional reactor guard vessel used in sodium fast reactors. All components for the Mk1 are modular and rail-transportable.

The Mk1 is designed to have negative fuel, moderator, and coolant temperature reactivity feedbacks. The reactivity control system uses buoyant rods both for normal reactivity control and shutdown. This is a passive shutdown mechanism as the buoyant rods insert if the reactor coolant temperature in the control-rod channel exceeds 615 °C, i.e., the buoyant stability limit. A separate shutdown system uses control blades inserted directly into the bed.

Emergency heat removal is achieved by employing a passive check valve that activates natural-circulation-driven heat transport to a set of three Direct Reactor Auxiliary Cooling System (DRACS) loops and ultimately to Thermosiphon-cooled Heat Exchangers (TCHXs) upon loss of flow condition.

A preliminary analysis of two ATWS (reactivity initiated accident and over-cooling transient) reveals that ample margins are available to accommodate the peak temperature experienced by the fuel during the event without compromising the integrity of the coated particles. Potential challenges could rather arise from the temperature limit of the metallic structures used to manufacture heat exchangers (Wang, 2018).

The Mk1 PB-FHR employs a nuclear air-Brayton combined cycle (NACC) that produces 100 MWe of base-load electricity when relying only on nuclear heat and increases this power output to 242 MWe using gas co-firing for peak electricity generation. Such co-firing capability brings a value proposition as additional revenues can be collected by providing dispatchable peak power at higher efficiency (66.4%) than natural gas plants (60%) in addition to the traditional base-load electrical power generation.

The Thorium Molten Salt Reactor Solid Fuel (TMSR-SF) is another pebble bed FHR design developed in China by The Shanghai Institute of Applied Physics (SINAP) (Dai, 2017). The TMSR-SF features a cylindrical core, 3-cm diameter pebbles, and (despite the name) 19.75% enriched uranium fuel. The core size is optimized for railway and highway transportation and the system is conceived as multipurpose, e.g., electricity production, hydrogen or methyl alcohol production, steam supply, heat delivering or seawater desalination. The total thermal power is 395 MWt (168 MWe). The system uses an intermediate salt loop and a combined air Brayton and steam Rankine cycle.

Commercial deployment of a PB-FHR is currently being pursued by Kairos Power LLC (KP; Kairos Power, n.d.). Its proprietary technology named KP-FHR features a cylindrical core and a steam conversion cycle.

Plate fuel designs

Plate or plank fuel consists of long and relatively thin planks made of carbon within which TRISO particles are dispersed. Early phase reactor concepts with plate fuel were developed at ORNL (Greene et al., 2011; Varma et al., 2012). They range from small 125-MWt plants for high temperature process heat production to large 3400 MWt reactors. The optimized plate design consists of two TRISO-bearing layers separated by a graphite layer to form a 2.55 cm thick plank. A hexagonal fuel element is obtained combining three sets of six planks as shown in Fig. 4. The assembly is completed by a central Y-shaped support structure and a channel box, both made of a carbon-carbon composite. The central structure also provides room for insertion of a control blade. The hexagonal assemblies (252 units) form a pseudo-cylindrical core with a 5.5 m long active region. When using a 9% enriched

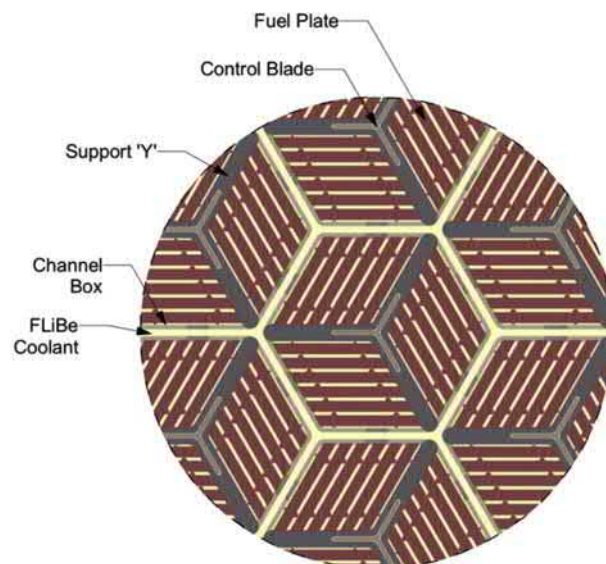


Fig. 4 Transverse cross section of a group of fuel assemblies with plate fuel. Varma V, et al. (2012) *AHTR Mechanical, Structural, and Neutronic Preconceptual Design*. Technical Report ORNL/TM-2012/320. Oak Ridge National Laboratory.

uranium fuel, half of the core is replaced every 6 months that is at least twice the frequency of light water reactors; therefore, a rapid refueling scheme was designed based on visually guided automated refueling manipulators in order to limit the downtime to 3 days or less. The large size plant uses a super-critical steam cycle and an intermedium salt (KF-ZrF_4) loop. This FHR concept also features fully passive safety. The primary shutdown system consists of control blades to be inserted into the support structure at the center of the fuel assemblies. A secondary shutdown system is based on the injection of rare earth fluoride salt into the core inlet plenum triggered by the salt temperature exceeding a set limit. Natural circulation decay heat rejection is obtained by DRACS if other heat rejection systems are unavailable. Finally, the nuclear facilities including the spent fuel pool are located below grade, providing shielding from aircraft impact, and wind or waterborne missiles.

Stringer fuel designs

The term stringer fuel is here used to indicate fuel designs based on long cylindrical elements. Stringer fuel was originally used in the ORNL concepts before transitioning to plate fuel. Currently, stringer fuel is employed in FHR designs inspired by the British Advanced Gas-cooled Reactor (AGR) (Forsberg et al., 2019). The AGR-FHR, indeed, employs so-called fuel in radial moderator, or FIRM, assemblies that consist of a graphite cylinder with channels for coolant flow and holes to host fuel compacts (short cylinders made of a graphite matrix with TRISO particle fuel) as illustrated in Fig. 5.

The FIRM assembly is placed in a permanent graphite structure. As the length of graphite elements is typically limited to roughly 2 m, multiple short fuel sub-assemblies would be held together by a stringer for longer cores. The design of the FIRM fuel assembly is optimized to increase fuel burnup by reducing parasitic neutron losses. Fast neutrons are generated in the fuel, migrate to the graphite, are thermalized away from the fuel, and return back toward the fuel to cause fission, resembling the neutronic strategy used in CANDU, AGR, and RBMK reactors to increase fuel burnup by increasing neutrons' resonance escape probability.

This fuel assembly design reduces secondary graphite moderator waste generation because only the part of graphite that contains the fuel and sees significant fast neutron radiation damage is replaced. It also borrows high-temperature re-fueling procedures from the AGR and through a double tank system avoids salt freezing and high temperatures beyond design limits. Finally, the passive decay heat removal system employs heat pipes that remove decay heat upon reactor shutdown to avoid excessive temperatures but at lower temperatures coolant vapor formation inside the heat pipes is very limited, therefore, there is effectively no heat transfer and freezing of the salt is avoided.

R&D needs

The major challenges and R&D needs for the commercial deployment of FHRs are extensively discussed by Holcomb et al. (2013). A brief summary is provided in this Section.

Tritium production is largely recognized as the main challenge for FHRs. Tritium is produced upon neutron absorption in ^6Li that is continuously replenished from neutron absorption in ^9Be (Fratoni and Greenspan, 2011). Tritium is the only radionuclide that under normal operating conditions, with no fuel failure, can be potentially released to the environment as tritium penetrates structural alloys and the heat exchangers provide a pathway to its release.

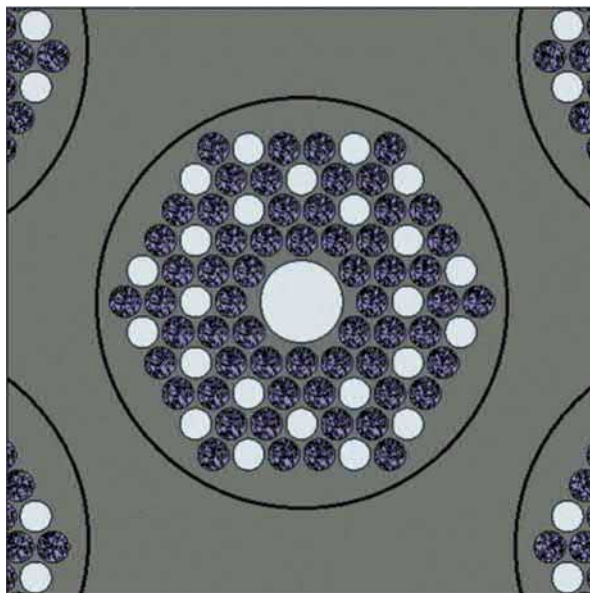


Fig. 5 Notional design of the AGR-FHR FIRM assembly. Forsberg C, et al. (2019) Fluoride-salt-cooled high-temperature reactor (FHR) using british advanced gas-cooled reactor (AGR) refueling technology and decay heat removal systems that prevent salt freezing. *Nuclear Technology* 205(9): 1127–1142, 2019. <https://doi.org/10.1080/00295450.2019.1586372>.

It is estimated that at equilibrium the Mk1 PB-FHR, for example, produces 0.023 mol of tritium per effective full power day (EFPD), or 0.069 g/EFPD, or 670 Ci/EFPD (Andreades et al., 2014). This is 3500 times more tritium per MWe than the 700 Ci/year of a typical 1000-MWe PWR. Maintaining the total tritium emission rates similar to that for PWR would require recovering 99.9% tritium. Multiple tritium management strategies are being evaluated and capture in carbon beds outside of the reactor core appears particularly promising (Forsberg et al., 2017).

Fuel development and qualification is an additional R&D challenge. Although the United States is testing TRISO fuel as part of the gas-cooled reactor program, the different operating envelope in FHR core is likely to require additional testing. Furthermore, high throughput production of TRISO particles and high assay enriched uranium are not readily available and will require the development of dedicated infrastructure. Similarly, FHRs require lithium enriched at 99.995% in ^7Li . Several alternative techniques have potential for producing industrial scale volumes but a preferred process remain to be selected and deployed.

High temperature, neutron and gamma radiation and salt make a unique environment and a potential challenge for instrumentation. Although no fundamental hurdles have been identified, measurement performance requirements for instrumentation during normal and accident conditions need to be determined and demonstrated. Optical measurement and diagnostic technologies for liquid salt, gamma-based flow-meter for primary flow-loop measurements and electrochemical salt redox-condition measurement instrumentation are among the technologies under development for FHR applications.

See Also: Molten Salt Reactors.

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Introduction to Breed and Burn Reactors

Staffan Qvist, Qvist Consulting Limited, London, United Kingdom

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Glossary

Breeding Ratio Ratio of the rate of conversion, by neutron capture, of fertile into fissile fuel to the rate of fissile fuel consumption.

Breed-and-Burn (B&B) Reactors Reactors that can breed fissile material and then fission (burn) a significant fraction of the bred fissile material without having to reprocess the fuel, when fed with make-up fuel the k_{∞} of which is smaller than 1.0.

Burnup A measure of the amount of fission energy released from a given quantity of fuel.

FIMA A unit of burnup defined as the fraction (usually given in %) of Initial Heavy Metal Atoms that has fissioned.

Neutron Flux Number of neutrons crossing a unit area per second.

Neutron Fluence Neutron flux integrated over time.

MOX Mixed Oxide Fuel, typically combinations of UO_2 and PuO_2 resulting from the reprocessing of used light water reactor fuel.

Infinite multiplication factor (k_{∞}) k_{∞} is the ratio of the rate of neutron production to the rate of neutron losses in a hypothetical infinitely large system without neutron losses to leakage.

Reactivity feedback A reactivity feedback implies that the ratio of the rate of neutron production to the rate of neutron losses (k_{eff}) changes due to a change in a performance characteristic such as temperature or density.

Smear density The fraction of the area inside a fuel rod cladding tube that separates the nuclear fuel from the coolant that is made up of fuel assumed to be at its nominal density.

Stationary Wave Reactors (SWRs) A subset of Breed-and-Burn reactors in which the fuel in the core is periodically relocated such that the breed-and-burn wave appears nearly stationary to an outside observer.

Traveling Wave Reactors (TWRs) A subset of Breed-and-Burn reactors in which a self-propagating breeding-and-burning wave travels through the static fuel material.

Traveling Wave Reactor (TWR[®]) Registered trademark of Traveling Wave Reactor under development by TerraPower, LLC.

Introduction and motivation

Present-day light water reactors (LWR) extract less than 1% of the potential energy of the uranium mined to make their fuel. About 90% of the unused uranium is in the form depleted uranium (DU, containing $\sim 0.25\%$ ^{235}U) left over from the enrichment process, and the remaining $>9\%$ is left over as used nuclear fuel (UNF). The world-wide stock of DU is about 1.6 million tons,¹ and every year more than 50,000 tons is added to this inventory (Lopez-Solis and Francois, 2017).

¹There is approximately 500,000 tons of DU stored in the form of UF_6 in the US alone (Schneider et al., 2007).

This material can, in principle, be utilized for energy production in nuclear reactors, primarily via conversion of ^{238}U by neutron capture to ^{239}Pu ² and subsequent fission. The conventionally proposed approach for attaining high uranium utilization is to use breeder reactors with fuel re-cycling. This process enables utilization (fission) of all the mined uranium minus inevitable losses in the reprocessing and fuel recycling operations. However, fuel reprocessing is a technically complicated, often costly and a politically difficult prospect in many countries. It has therefore been of great interest to explore fuel cycles that can attain high levels of uranium utilization without the need for reprocessing.

To enable a high utilization of uranium while using a once-through fuel cycle without reprocessing, a special class of nuclear reactors collectively known as “breed-and-burn” (B&B) reactors have been under consideration since the late 1950s (Feinberg and Kunegin, 1958). Any reactor containing fertile material (like ^{238}U or ^{232}Th) will inevitably breed and subsequently burn some fraction of the bred fissile material. *A breed-and-burn reactor, uniquely, relies solely on this process for criticality in its equilibrium state.* B&B systems may formally be defined as follows:

- A reactor whose operation includes the periodic or continuous loading and discharge of fuel may be defined as a B&B system if it is capable of sustaining an equilibrium state of critical operation fed only by fuel that, if the full core fuel inventory was made up of this feed fuel, would render the system subcritical³ (that is, the core infinite multiplication factor would be smaller than unity).
- A static fuel reactor (with no periodic or continuous loading and discharge of fuel) may be defined as B&B system if it is capable of initiating and self-propagating a breeding and burning wave through the fuel that, in its equilibrium critical state, is independent of the initial fissile starter fuel.

A large number of “battery” type fast reactor concepts have been developed that make very significant use of the breed and burn principle, typically with static (unshuffled) fuel maintaining criticality for extended periods of time and to significant levels of burnup (~ 10 – 15% FIMA). For some examples, see refs. (Greenspan, 2003; Wade, 2010; Grandy, 2011; ANL/CEA/JNC, 2005; Choi et al., 2011; Tsuboi et al., 2012; Brown, 1999). While these concepts may appear similar to B&B reactors (and sometimes are categorized as such Lopez-Solis and Francois, 2017), these systems do not fall in to the formal category of B&B reactors in this encyclopedia since they do not conform to the definitions as stated above. The breeding ratio of “battery” type cores is slightly above unity which is significantly smaller than the breeding ratio of B&B cores.

In order to establish initial criticality, the initial fuel loading of any B&B core has to contain an adequate amount of fissile material. While the definition of a B&B system formally allows for enriched or reprocessed feed-fuel even in the equilibrium state, the long-term aim of B&B reactor development is for fuel reloads to be made up of either depleted or natural uranium; a system like this will require no fuel enrichment and no fuel reprocessing. New B&B cores still require new fissile material for the first core loading to establish initial criticality. However, this material can be supplied by reconditioning⁴ of the fuel discharged from an existing B&B core (Petroski et al., 2011; Heidet and Greenspan, 2012a,b; Kim et al., 2017). The asymptotic requirement for uranium enrichment and reprocessing in a steady (not expanding) fleet of future B&B reactor systems is therefore zero.

Fast breeder reactors (FBRs) can also operate without fuel recycling using a once-through fuel cycle. Although the standard once-through FBR discharge burnup is two to three times higher than that of Light Water Reactors (LWRs), the uranium utilization of a once-through FBR is not significantly different from that of a once-through LWR because the level of uranium enrichment required to achieve criticality in the FBR is often three times that required to fuel the LWR. A conventional FBR operating without reprocessing is thus not able to use fuel resources more efficiently or make any use of the untapped energy potential of the available DU stockpiles.

In contrast, the uranium utilization of a B&B system will approach the fraction of the loaded uranium that has been fissioned. For realistic once-through solid-fuelled B&B systems with depleted or natural uranium reloads, this is around 20%, whereas for liquid-fuel designs or solid-fuel designs utilizing fuel reconditioning this value may be as high as 40%. B&B systems are therefore an extremely resource-efficient way of making use of the mined uranium without the use of fuel reprocessing technology.

If fully utilized for electricity production (in a system with 40% efficiency of converting thermal energy to electricity), each ton of mined uranium can in principle provide ~ 9 terawatt-hours (TWh) of electricity through fission. The electricity production possible from the currently available stockpiles of DU is therefore equivalent to current global electricity consumption for more than 500 years.⁵ Utilizing the B&B cycle without any reprocessing or fuel reconditioning, an amount of electricity corresponding to more than 100 years of current global consumption could in theory be unlocked just from this un-used depleted uranium waste

²There are existing reactor technologies that can operate using natural uranium as fuel, such as the heavy-water-moderated CANDU and the graphite-moderated Magnox type of reactors. Neither can run on a depleted uranium fuel loading. These reactors operate by fissioning fissile ^{235}U present in natural uranium, rather than by converting fertile ^{238}U , and thus achieve very low levels of uranium utilization in line with LWRs.

³A more ambitious definition would be that the equilibrium cycle of a B&B system should be fuelled only by fertile fuel with a ^{235}U content at or below that of natural uranium (0.711%). While this remains the longer-term objective of any B&B system, it does not formally define the reactor category. In a transitional phase while materials and fuels are being qualified for higher burnup and irradiation damage tolerance, prototype B&B systems will likely need to operate with equilibrium cycles fed by enriched fuel.

⁴Reconditioning typically means to simply reclad the fuel rods. This process inevitably involves the removal of gaseous fission products and refabrication of fuel rods, but reconditioning does not involve any kind of chemical fuel reprocessing operations.

⁵Current worldwide electricity consumption is $\sim 27,000$ TWh/year (BP., 2019), and the electricity potential from fully fissioning current DU stockpiles is 1.6×10^6 (tons) $\times 9$ TWh/ton = 14.4 million TWh. ($14.4 \times 10^6 / 27000 = 533$ years)

material without the need for additional uranium mining.⁶ The fuel supply implications of various nuclear fuel cycles are summarized in **Table 1**. By the time a significant fleet of B&B reactors will be deployed the inventory of DU waste will be significantly larger than the one accounted for in composing this table.

A brief history of breed & burn reactor development

The general principle of a breed-and-burn type of reactor was introduced by USSR scientists E. P. Kunegin and S. M. Feinberg in the late 1950s (Kunegin and Feinberg, 1958). The results of their initial research was briefly introduced to the world at the Second United Nations International Conference on the Peaceful Use of Atomic Energy in 1958 with the following comment (Feinberg and Kunegin, 1958):

I should like to draw attention to a possible modification of the conventional fast breeder reactor scheme, in which the reactor has only an active zone, i.e., there is no separate breeder zone and so there is no longer any need for chemical and metallurgical reprocessing of the fuel in the breeder cycle as a whole. A fast neutron reactor is continuously loaded with blocks of ^{238}U . The blocks are also continuously unloaded, after about 30% of the uranium has been consumed, when they contain about 7% of plutonium. Criticality is maintained because of the plutonium which accumulates in the reactor. Theoretical calculations for the breeder, which have been made by Mr. Kunegin, show that such a scheme is feasible. [...] An estimate has shown that the dimensions and the overall power of a profitable plant might have to be very large. Moreover, there are as yet no fuel elements capable of ensuring a burn-up of 30% of the uranium.

The issues identified by Feinberg & Kunegin in the late 1950s remain the main challenges to B&B reactor development to this day, namely a requirement for very low neutron leakage (large core dimensions) as well as fuel and structural materials that can survive very high levels of burnup and neutron irradiation damage.

In 2007, a small team at Intellectual Ventures funded by Bill Gates initiated work on physics and engineering concepts of B&B reactors and in 2008 TerraPower, LLC (TerraPower[®]) was officially launched (TerraPower, 2015). In the years following this announcement, a large number of groups have started research on the topic, making it difficult to give a full and exhaustive overview of all major B&B research developments after this point. An extensive, though certainly not complete, chronologic history of B&B research and development from the late 1950s up until 2007 is shown in **Fig. 1**. The first or most important publication from each identified B&B research group in **Fig. 1** is given in references: (Feinberg and Kunegin, 1958; Fuchs and Hessel, 1961; Kouts et al., 1979; Atefi, 1980; Slesarev et al., 1983; Feoktistov, 1988; Goldin and Anistratov, 1992; Seifritz, 1995; Teller et al., 1996; Toshinsky, 1997; Akhiezer et al., 1999; Sekimoto and Ryu, 2000; Van Dam, 1998; Van Dam, 2000; Khizhnyak, 2001; Pilipenko et al., 2003; Fomin et al., 2005; Chen et al., 2005; Yarsky, 2005; Driscoll et al., 2005; Heidet and Greenspan, 2010; Gaveau et al., 2005; Gilleland, 2008). Some of the more recently published research studies on the topic are presented in ref. (Pavlovich et al., 2007; Rusov, 2011; Hartanto and Kim, 2012; Osbourne et al., 2012; Choi et al., 2013; Si, 2013; Heidet and Greenspan, 2013a,b).

Types of breed & burn reactors

B&B reactors come in three basic flavors: Standing Wave Reactors (SWRs), Traveling Wave Reactors (TWRs), and molten-salt B&B (BBMSRs; Greenspan, 2012). They are presented below in the chronological order in which they were invented.

Table 1 Fuel utilization comparisons between LWRs and DU-fed B&Bs (Greenspan and Heidet, 2011; BP., 2019).

Concept	Uranium utilization	Uranium utilization relative to LWR	Years of current global electricity consumption from current DU stockpiles
Light Water Reactor (LWR)	~0.6%	1.0	N/A
Once-through FBR	Up to ~1%	1.5	N/A
LWR with recycling reprocessed mixed-oxide (MOX) fuel	~1%	1.5	N/A
Solid-fuel B&B reactor with DU fuel reloads, No fuel reconditioning	~20%	> 30	> 100
Solid-fuel B&B reactor with DU reloads and one fuel reconditioning or a fluid fuel B&B reactor	Up to 40%	> 65	> 200
Breeder reactor with fuel recycling	> 95%	> 150	> 500 ⁵

⁶However, in order to establish initial criticality, B&B cores will require enriched uranium (some additional uranium mining and enrichment) or another form of fissile material.

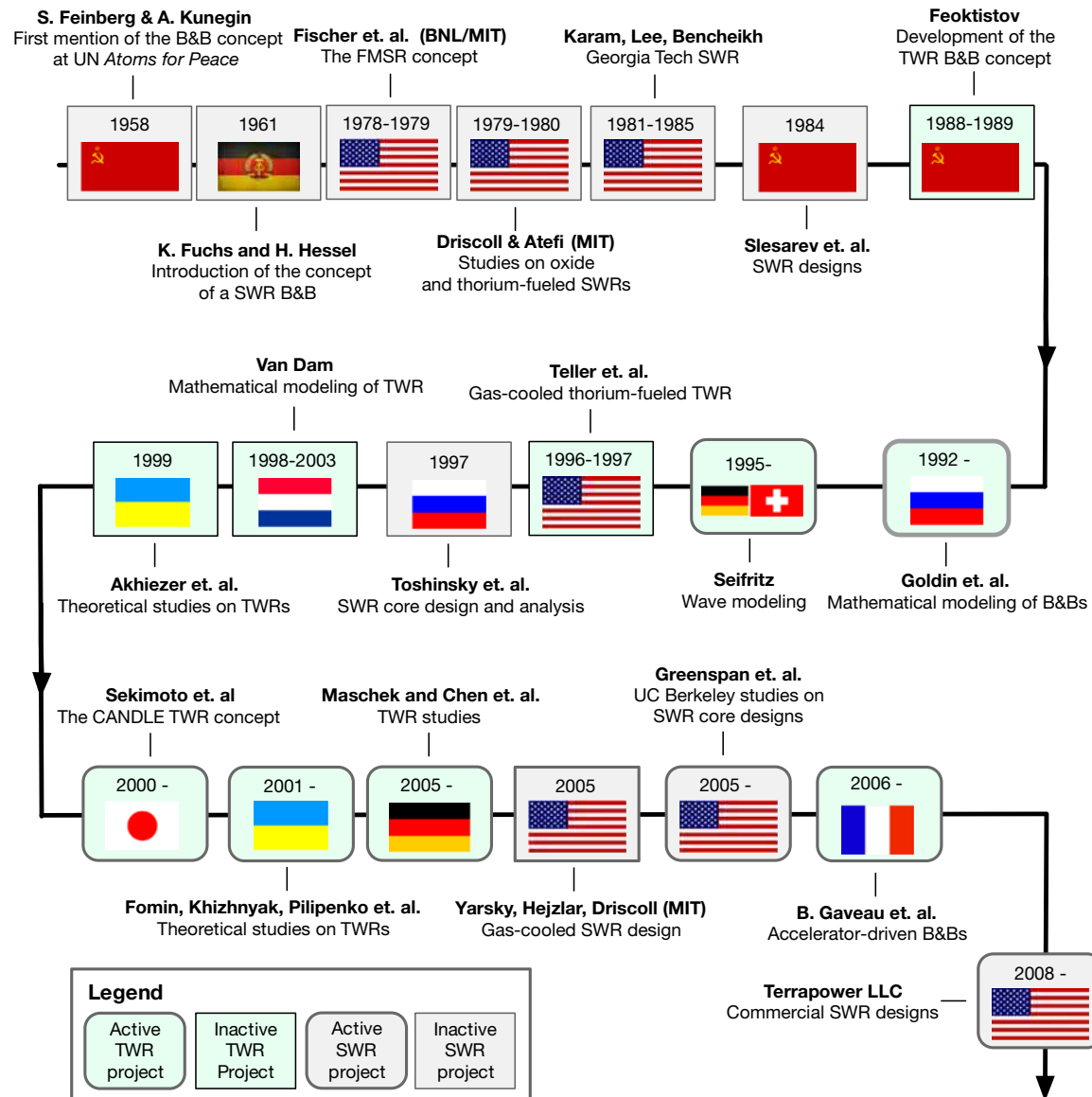


Fig. 1 The history of breed-and-burn reactor R&D until 2008.

Standing wave reactor (SWR)

A “standing wave reactor” (SWR), the type of B&B originally proposed by Feinberg and Kunegin, is essentially a type of fast breeder reactor with an exceptionally good neutron economy (very low neutron leakage, hard spectrum and high core actinide density) which enables it to operate on the unique B&B-type fuel cycle. In SWRs, fuel assemblies are periodically shuffled in the core, burned fuel batches are discharged and new fertile fuel assemblies are loaded. Fertile fuel batches, ideally containing depleted or natural uranium fuel, are typically loaded at the low-worth radial periphery of the core⁷ and are subsequently moved into inner regions of the core before being discharged when the target equilibrium cycle burnup is reached. The location of “breeding and burning” in an SWR therefore stays fairly stationary while the fuel “travels”, which is the feature which gives this type of B&B system its name (Standing, Stationary or Soliton Wave Reactor – SWR). Apart from the characteristics that typically define any nuclear reactor technology (the spectrum and type of fuel & coolant), SWRs can be broadly differentiated by the type of shuffling strategy employed. A large variety of optimized 2D B&B shuffling strategies have been developed (Toshinsky et al., 2000; Toshinsky et al., 1999; Heidet and Greenspan, 2013a,b; Qvist et al., 2015; Heidet and Greenspan, 2013a,b; Kuwagaki et al., 2018; Kuwagaki et al., 2019; Zheng, 2014; Su’ud et al., 2013), while more limited work has been done on three-dimensional shuffling for SWRs (Qvist and Greenspan, 2014a,b; Qvist et al., 2015). Some examples of SWR concepts,

⁷Thereby capturing excess fission neutrons generated in inner, high fission density regions of the core and minimizing neutron leakage out from the core.

distinguished by the overarching approach to fuel shuffling, are given in [Table 2](#). TerraPower, LLC's TWR concept is also a SWR but uses the TWR[®] trademark designation for reasons discussed in Chapter Traveling wave reactor (TWR).

Traveling wave reactor (TWRs)

TWRs, a category of B&Bs first explored by Lev Feoktistov in the late 1980s ([Feoktistov, 1988](#)),⁸ are cores traditionally defined to have static (unshuffled) fertile fuel with a small, enriched, “starter” region typically located at one axial end. The fissile starter region establishes criticality and initiates a self-propagating breeding-and-burning wave that travels through the static fuel material toward the other axial end. The stages of operation for this reactor class are shown in [Fig. 2](#), in which the third state refers to the established B&B equilibrium in which all the main parameters, such as flux, flux profile and power level of TWRs remain essentially constant as the breeding and burning “waves” self-propagate through the core.

In principle, the fissile starter zone of TWRs could be omitted by instead irradiating a portion of the fuel with an external neutron source or an internal accelerator driven spallation neutron source ([Cisneros et al., 2012](#)).⁹ The leading Japanese TWRs researcher Hiroshi Sekimoto has dubbed this type of principle “CANDLE”, as the burnup wave propagation is similar to a burning candle.¹⁰ While this wave propagation principle is very elegant and simple, it has so far proven very difficult to engineer a commercial design that makes full and direct use of it. To establish a self-propagating B&B wave through stationary fuel material with low fissile content, a very high level of burnup and associated peak fast neutron fluence is needed, which exposes core structures (like the fuel rod cladding and fuel assembly duct wall steel) to high levels of radiation damage ([Heidet, 2021](#)).

There are essentially five different possible options to configure TWRs, which are defined by where in the core that the initiating fissile fuel is placed, as shown in [Figs. 3 and 4](#). For axially propagating burnup waves, the fissile fuel portion can either be placed at one axial end

Table 2 Examples of SWR concepts.

SWR type	Concept	Design specs.
2D Radial Fuel Shuffling	TerraPower TWR [®]	For full details, see Chapter Traveling wave reactor (TWR) of this encyclopedia
	UCB SWR (Heidet and Greenspan, 2013a,b ; Qvist, 2013 ; Heidet and Greenspan, 2012a,b)	Sodium & lead coolant, Metallic fuel
	FMSR (Loh, 1980 ; Fischer et al., 1979 ; Kouts et al., 1979 ; Atefi, 1980)	Beryllium moderator, metallic feed fuel (DU or NU), gas and sodium-coolant options
	Georgia Tech SWR (Karam and Lee, 1981 ; Bencheikh and Karam, 1985 ; Lee and Karam, 1982)	Sodium coolant, Various fuel options
	Xi'an SWR (Zhang et al., 2014 ; Zheng, 2014)	Sodium coolant, Metallic fuel
	TokyoTech SWR (Obara et al., 2017 ; Ku wagaki et al., 2018)	LBE coolant Metallic fuel
	SFRP (Toshinsky, 1997 ; Toshinsky et al., 2000 ; Toshinsky et al., 1999)	LBE coolant Metallic fuel
3D Fuel Shuffling	UCB 3D-SWR (Qvist and Greenspan, 2014a,b ; Qvist et al., 2015 ; Hou et al., 2016 ; Hou et al., 2015)	Sodium coolant, metallic DU-Zr feed fuel
Pebble-bed SWR	UCB Pebble-bed (Greenspan, 2016)	Sodium coolant Metallic fuel pebbles
	TokyoTech Pebble-bed (Ryu and Sekimoto, 2000)	Helium coolant Nitride-fuel pebbles
Molten salt fuel, but separated from coolant	Variations on the Moltex SSR concept (Kasam, 2018)	UCL ₃ fuel, 50-30-20% NaCl-KCl-MgCl ₂ coolant

⁸Lev Petrovich Feoktistov (1928–2002) was a leading USSR scientist originally active in the development of thermonuclear weapons. In the 1970s he left this work behind to join the Kurchatov Institute and became a very active proponent of nuclear disarmament and a member of the Pugwash movement. In the USSR and present-day Russia, the “Traveling Wave” is also known as the “Feoktistov wave”, named after its discoverer.

⁹Studies have also examined fusion/fission hybrids where neutrons from the fusion reaction is the driving source for a breeding and burning fission blanket ([Brown et al., 2016](#))

¹⁰An alternative analog would be the burning of a matchstick, where the match head is the equivalent of the fissile starter region that initiates the burning wave, which then propagates axially and leaves behind burnt unproductive fuel-ash as it propagates down the match.

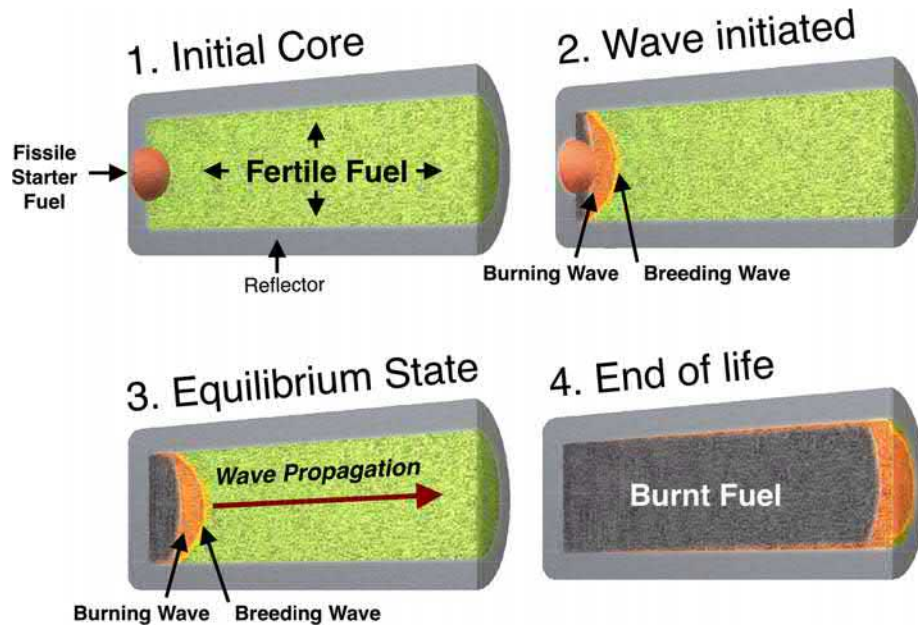


Fig. 2 Stages of B&B wave propagation in TWRs. This figure is meant only to illustrate the concept and does not represent a realistic core design. Adapted from Weaver KD, Gilleland J, Ahlfeld C, Whitmer C, and Zimmerman G (2009) A once-through fuel cycle for fast reactors. In: *Proceedings of the 17th International Conference on Nuclear Engineering, ICONE17* July 12–16, Brussels, Belgium.

Axial TWR options

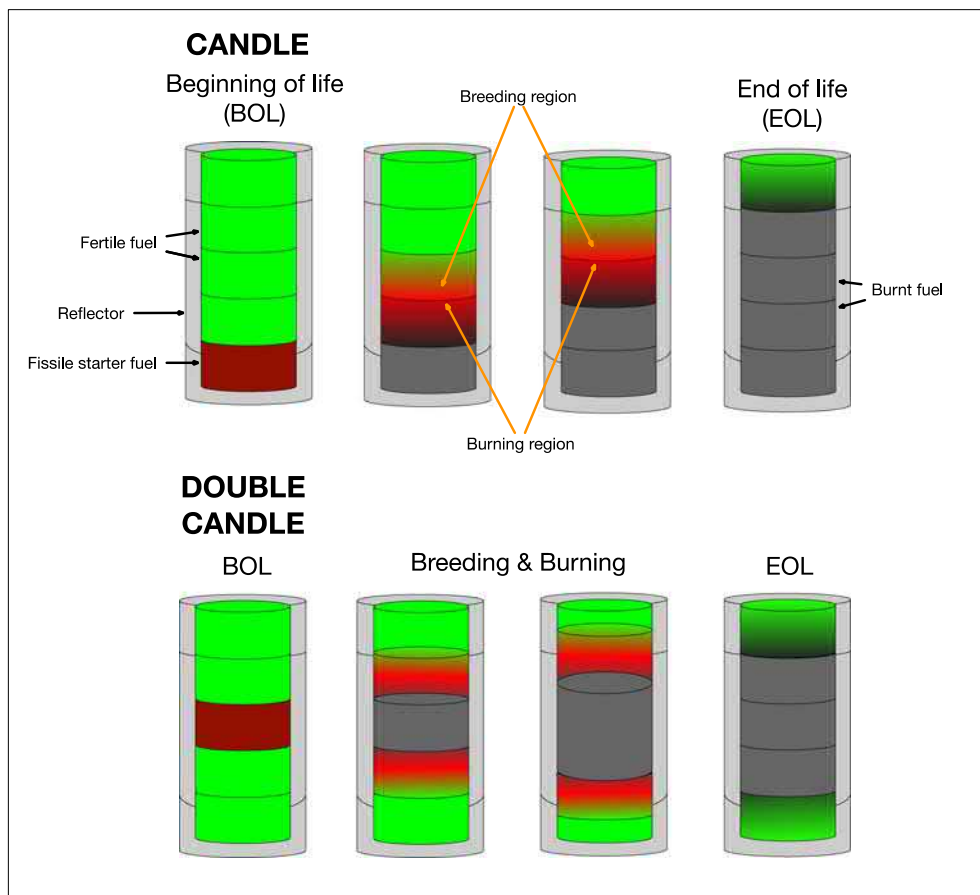


Fig. 3 "Axial" B&B wave principles and options for TWRs.

Radial TWR options

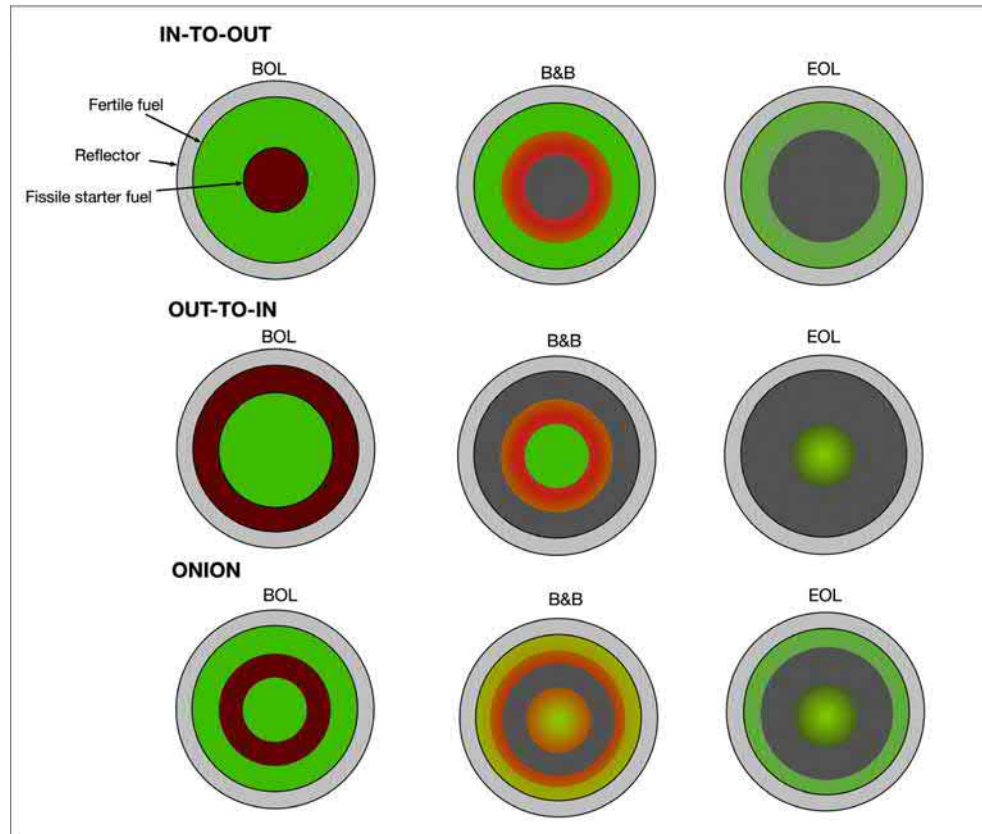


Fig. 4 “Radial” B&B wave principles and options for TWRs.

Table 3 Examples of TWR concepts.

TWR type	Concept	Design specs.
Axial “CANDLE”	CANDLE (Sekimoto and Ryu, 2000), See Chapter “CANDLE Reactor Concept” of the encyclopedia for full details UCFR (Tak et al., 2012; Jung et al., 2013)	Uranium-Nitride feed fuel, Lead or LBE ^a coolant and many other options
	KIT SWR (Chen et al., 2010; Zhang et al., 2011)	Metallic Uranium-Zr feed fuel, Sodium coolant Supercritical water coolant Enriched oxide fuel
Double “CANDLE”	Teller et al. TWR (Teller et al., 1996)	Metallic Thorium feed fuel, Helium-gas coolant
Radial “Onion”	ULFR (Kim and Taiwo, 2010)	Metallic Uranium-Zr feed fuel, Sodium coolant
Radial “In-to-out”	General Atomics EM ² (Choi et al., 2013)	Carbide uranium feed fuel, Helium-gas coolant

^aLBE: Lead-bismuth eutectic.

(top of Fig. 3) or in the central axial zone (bottom of Fig. 3).¹¹ For radially propagating waves, the fissile fuel can be placed either centrally (propagating outwards), at the periphery (propagating inwards) or in the radial central zone (propagating in both radial directions).

Some examples of TWR concepts available in literature are given in Table 3.

¹¹In principle one could also place fissile starter regions at *both* axial ends, with two waves propagating toward the axial center, however no concept aiming to make use of such a principle was found in the literature review.

Table 4 Examples of BBMSR concepts.

<i>MSRBB type</i>	<i>Concept</i>
Single fluid molten salt reactor	TerraPower MCFR (Cisneros et al., 2018)
	Swiss BBMSR (Hombourger, 2018; Hombourger et al., 2015, 2017)
	Aalto BBMSR (Koivisto, 2019)
	Elysium BBMSR (Pheil, 2018)
	Aalto BBMSR (Koivisto, 2019)
Dual fluid molten salt reactor	Aalto BBMSR (Koivisto, 2019)

Breed & Burn Molten Salt Reactor (BBMSR)

In addition to solid-fuel options, the B&B fuel cycle has more recently been shown to be possible in a fast-spectrum fluid fuel molten salt type of reactors (BBMSR) with both single fuel-salt and multiple fuel-salt designs. In a BBMSR system, the initial fuel load contains a significant fraction of fissile material to establish criticality. The system then operates by continuously feeding in fertile fuel salt and simultaneously discharging homogeneously mixed molten fuel with an equilibrium-state average burnup level of up to 10–40% FIMA (Martin et al., 2017; Hombourger, 2018; Hombourger et al., 2019; Kramer, 2018; Krepel and Kramer, 2021). Some examples of BBMSR concepts under development are given in Table 4.

Fundamentals of breed-and-burn reactor physics

Neutron spectrum and fuel cycle

The B&B fuel cycle puts extremely challenging requirements on the neutron economy, particularly when aiming to realize its full potential in systems that in equilibrium are fed only by fertile fuel such as a natural or depleted uranium. All major B&B design efforts rely on a hard (fast) neutron spectrum to maximize ν (the average number of neutrons per fission) and to minimize neutron parasitic absorption. The fertile-feed B&B fuel cycle is not possible in any type of reactor system with a thermal or intermediate neutron flux spectrum. Thorium-based B&Bs have been shown to be possible at least in theory (Hombourger et al., 2019), but the Uranium-Plutonium fuel cycle is the main development path for all major current B&B efforts due to its superior neutron economy in a fast spectrum.

The neutron balance concept

The neutron balance concept is a simple mathematical framework for analyzing the requirements and limits of discharge fuel burnup in B&B systems. Unique among nuclear fuel cycles, the B&B reactor requires a certain minimum level of discharge burnup to be attained in its equilibrium cycle.¹² Rather than trying to increase this level (as is typically the goal for designers of nuclear energy systems), the main aim of the B&B system designer is to minimize the required discharge burnup level in order to minimize problems related to irradiation damage of structural materials.

To explore the physical principles of the B&B fuel cycle, we follow a batch of fertile fuel in a SWR B&B core from its introduction until its discharge. We follow the derivation used by Heidet and Greenspan et al. (Heidet et al., 2009; Heidet, 2010; Heidet and Greenspan, 2012a,b; Petroski et al., 2010; Petroski et al., 2011), which in principle is similar to the method used earlier by Yu et al (Yu et al., 2002). This generalized methodology is valid regardless of the type of B&B system studied.¹³ A fertile B&B fuel batch will pass five distinct operating milestones as relating to the neutron balance as defined below (and shown in Fig. 5).

1. Fuel introduction to core

A fuel batch of fertile fuel (ideally depleted or natural uranium) loaded to the core initially produces far fewer neutrons than it absorbs, and its infinite multiplication factor - k_{∞} , is thus significantly lower than unity. The fuel is consuming excess neutrons that leak from fissions occurring in other, higher k_{∞} , fuel regions. As burnup progresses, neutron capture converts fertile material into fissile isotopes thus increasing the fissile fuel concentration and, along with it, the batch k_{∞} .

2. Net neutron producer

¹²Equilibrium for an SWR means that the system parameters between subsequent burn cycles are identical, for the TWR and MSRBB it means the state when system criticality and the B&B wave propagation no longer relies on the initial fissile fuel load.

¹³For a TWR, the batch simply corresponds to a region of stationary fertile fuel. For an MSRBB the analog of a “batch” is simply a unit volume of fertile fuel salt, starting from its introduction in an equilibrium core.

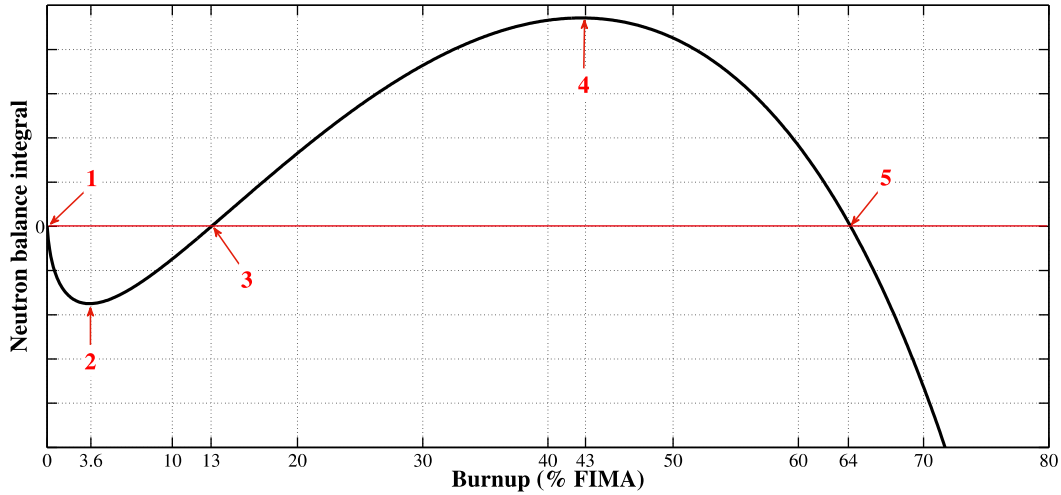


Fig. 5 Example of a B&B neutron balance integral with balance points marked (unit cell, sodium-coolant, metallic U–6Zr fuel, HT9-steel structure; $L = 0$).

Once the batch k_{∞} reaches unity, it turns from a net neutron consumer to a net neutron producer. From this point on, the batch starts to pay back its debt to the neutron economy.

3. B&B neutron balance point #1

At this point, the burnup-integrated neutron production and loss (absorption + leakage) of the batch are equal, and the overall neutron economy is balanced. This point corresponds to the lowest level of *average discharge burnup* that an equilibrium cycle of the B&B core can be operated at. This is the main design point of interest to the B&B core designer.

4. Net neutron absorber

Due primarily to the accumulation of fission products, the excess neutron production diminishes after having reached point 3. At point 4, the neutron absorption rate equals the production rate (k_{∞} is again equal to one), and after this point the batch again becomes a net neutron absorber. In the neutron economy, the net neutron “profit” accumulated between points 3 and 4 is now being spent to “drive” the high-burnup subcritical fuel on to point 5.

5. B&B neutron balance point #2

Once the net neutron gain accumulated between point 3 and 4 has been consumed, the net neutron excess is again 0, and the neutron economy is balanced for the second time. This corresponds to the maximum level of average discharge burnup that an equilibrium cycle of a B&B system can, in theory, be operated to.

The values (in terms of burnup) that correspond to points 1–5 can be quantified using a very simple neutron balance analysis. The point of primary interest for the B&B reactor designer is point 3, the minimum burnup required for the system to operate in a B&B mode. The total number of neutrons produced (P) by N_{HM} atoms of actinide fuel up to a level of burnup Bu in units of FIMA is given by Eq. (1) in which $\bar{\nu}$ is the average number of neutrons per fission:

$$P = N_{HM} \int_0^{Bu} \bar{\nu}(Bu) dBu \quad (1)$$

Since the infinite multiplication factor is the neutron production¹⁴ divided by neutron absorption, the total number of neutrons absorbed in the fuel up to a burnup Bu is:

$$A = N_{HM} \int_0^{Bu} \frac{\bar{\nu}(Bu)}{k_{\infty}(Bu)} dBu \quad (2)$$

Combining Eqs. (1) and (2), the points in burnup (Bu) where the integrated neutron production and absorption are equal are given by the solutions to:

$$\int_0^{Bu} \left(\bar{\nu} - \frac{\bar{\nu}}{k_{\infty}} \right) dBu = 0 \quad (3)$$

¹⁴Which includes neutrons from fission as well as from (n,2n) and (n,3n) reactions (Yang et al., 2016).

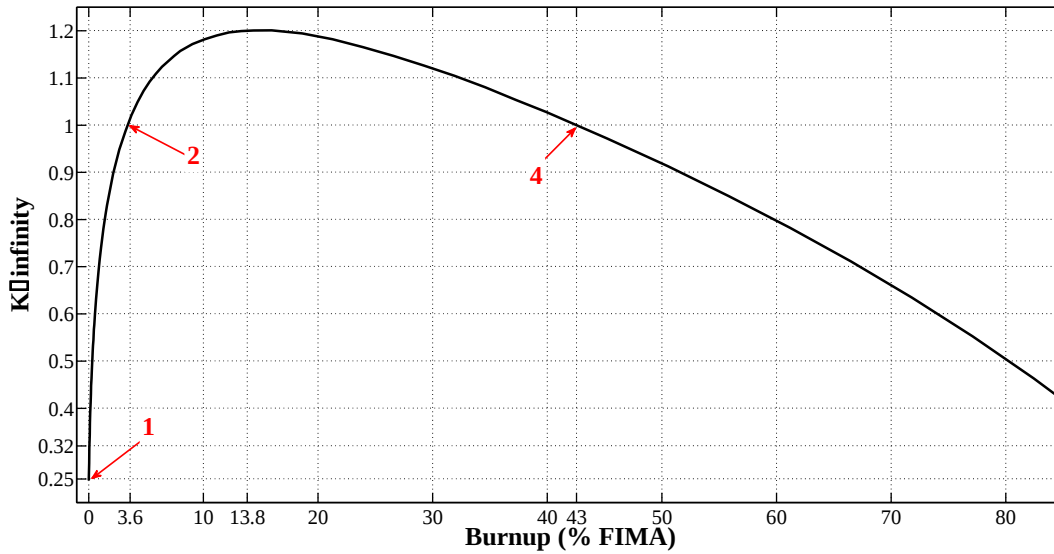


Fig. 6 Example of the k_{∞} -evolution of a batch of metallic depleted uranium (U–6Zr) fuel in a unit cell with sodium-coolant and HT9-steel structure.

In a real system, neutrons are also lost to leakage and to burnup reactivity control systems. Combining the loss terms for neutrons into a single term denoted as L , Eq. (3) becomes the *neutron balance equation*:

$$\int_0^{Bu} \bar{v} \left[1 - \frac{1}{k_{\infty}(1-L)} \right] dBu = 0 \quad (4)$$

To illustrate the points of neutron balance of a typical B&B system described in the equations above, neutron transport calculations were performed in a 0-dimensional geometry representing a typical sodium-cooled fast reactor with metallic alloy of depleted uranium (0.25% ^{235}U) with 6 wt% Zirconium (U–6Zr) feed-fuel using the Serpent neutron transport code (Leppänen, 2010) with ENDF/B-VII.0 cross-sections (Chadwick, 2006). The resulting neutron balance plot (the left side of Eq. 4) is given in Fig. 5 and the k_{∞} evolution plot in Fig. 6.

For the B&B core designer, the extended neutron balance (ENB) method, an expansion of the basic neutron balance presented above, can be used to further enable parametric and sensitivity analyses of candidate B&B core design options (Qvist and Greenspan, 2016; Qvist, 2013). In practice, the value of neutron loss probability (L) needs to be kept well below 10% for the fertile-feed B&B cycle to be possible at all, and below $\sim 6\%$ in order to limit peak levels of fuel burnup and irradiation damage to realistic levels ($\sim 20\%$ FIMA). In a metal-fueled core, increasing the neutron loss from $L = 0\%$ to $L = 3\%$ corresponds to an increase in the average minimum required burnup of about 3% FIMA (from 13% to 16.5%), while a further increase to $L = 6\%$ raises the minimum burnup to just over 20% FIMA (Qvist and Greenspan, 2016).

Engineering aspects of B&B systems

Minimum burnup requirement & irradiation damage

The main engineering challenge of any system relying on the B&B principle is the minimum required discharge burnup and the associated fast neutron fluence and neutron irradiation damage to the fuel and structural components of the core. The ambition to minimize the peak fuel burnup and irradiation damage drives many of the main design and material choices of B&B systems.

Current fast reactor structural steels were so far exposed to a peak fast neutron fluence of $\sim 4 \times 10^{23} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$), corresponding to a radiation damage of $\sim 200 \text{ DPA}$ (displacements per atom) (Toloczko et al., 1994). Solid-fuel B&B designs with a DU-feed-fuel suffer peak equilibrium cycle neutron irradiation damage to the cladding and fuel assembly duct steel structures starting from a theoretical low of around 350 DPA (for very large 3D-shuffled SWRs) to well over 1000 DPA (for TWRs). Structures in more near-term realistic designs utilizing conventional 2D-fuel shuffling B&Bs suffer a peak neutron irradiation damage in the range of 500–700 DPA. The difference in peak neutron damage between the 2D and 3D shuffling systems is due to the theoretical possibility of greatly reducing the axial peaking factor of burnup, fluence and DPA in the 3D-shuffled system. Molten salt reactor fuel is liquid and thus does not suffer lasting irradiation damage. BBSRs have no solid in-core structures in high-flux regions, and thus suffer much lower levels of peak structure DPA while operating on a B&B type fuel cycle. Representative burnup and irradiation damage levels of B&B systems are summarized in Table 5.

There are essentially five current approaches to tackle the challenge of high neutron irradiation damage levels in solid-fuel B&B systems:

Table 5 Approximate ave. burnup and irradiation damage in B&B systems fed by DU.

B&B Type	Required average minimum discharge burnup (% FIMA)	Approx. min. Peak structure DPA
TWRs (Tak et al., 2012; Sekimoto, 2007)	~30–40%	> 1000
2D-shuffled SWR (Qvist et al., 2015) and TerraPower TWR® (see Chapter Traveling wave reactor (TWR))	~15–20%	~500
3D-shuffled SWR (Qvist et al., 2015)	~17–25%	~350
BBMSR (Martin et al., 2017; Hombourger et al., 2019)	~10–40%	N/A due to no solid in-core structures. (the vessel can in theory be protected by sacrificial replaceable materials)

1. Increasing the irradiation damage tolerance limits for steels

Samples of HT9 steel previously irradiated to ~200 DPA in the FFTF reactor in the USA have now been further irradiated in the BOR-60 reactor in Russia to extend the knowledge base for HT9 steel to ~300 DPA (Maloy, 2015). TerraPower, LLC is developing optimized versions of HT9 steel with the eventual aim of tolerating damage well above 500 DPA without excessive swelling or material property degradation. In parallel, Oxide Dispersion Strengthened (ODS) alloys are under development and testing to further expand irradiation tolerance (Maloy, 2015). Ion irradiation results up to 500 DPA indicate that ODS steels such as MA957 and 14YWT, as well as TerraPower's optimized ferritic martensitic steels may outperform HT9.

2. Minimizing the peak irradiation damage by design

This involves, for SWRs, optimizing the fuel shuffling strategy to minimize peaking factors and in general to minimize the neutron leakage probability.

3. Use of enriched feed fuel at equilibrium state

While materials are being qualified for higher limits, it is highly likely that the first generation of B&B reactors will use enriched fuel rather than fertile fuel in order to enable a B&B equilibrium cycle at lower burnup and irradiation damage. This is the approach presently taken by TerraPower (Gilleland et al., 2016) as further described by Hejzlar (Hejzlar, 2021).

4. Use of source-driven subcritical B&B system

The idea is to reduce the minimum burnup and, hence, also irradiation damage at which the fuel need be discharged by supplementing the neutrons emitted in fissions in the B&B system by neutrons generated by an external neutron source. This will move Point 3 in Fig. 5 to the left; that is, will lower the minimum required burnup. The two potentially relevant external neutron sources are spallation reactions induced by high energy ions impinging on a heavy metal such as lead or tungsten, and fusion reactions to be generated in future fusion devices (Moses et al., 2009; Kramer et al., 2008; Abbott et al., 2008; Meier et al., 2014). Another approach that is described in detail by Greenspan in Chapter Seed driven breed-and-burn blanket reactor concepts is to use excess neutrons that are generated by a high k_{∞} critical fission region (the "seed") to drive a subcritical B&B blanket that radially surrounds the seed.

5. Periodic replacement of structural materials and reconditioning of fuel

One design solution investigated by Sekimoto and colleagues for TWRs is to re-clad the fuel rods with new steel every ~10 years of operation to limit peak damage levels (Nagata and Sekimoto, 2007; Nagata et al., 2009). At UC Berkeley, researchers have looked in to the possibility of B&B operation with periodic and limited fuel reconditioning (Heidet, 2010; Greenspan and Heidet, 2011; Greenspan, 1998), as well the option to double-clad the fuel (Di Sanzo et al., 2011).

To enable the required high levels of burnup in the fuel, solid-fuel B&Bs (typically employing metallic fuel) are designed with a low smear density (60–75%) which allows for sufficient fuel swelling without causing excessive stress in the cladding. To limit fission gas pressure in the cladding tubes, the fuel rods are typically designed to vent gaseous fission products to a fission-gas plenum above the active fuel zone. A number of fission gas vent-to-coolant devices have been developed for both sodium cooled fast reactors (SFR) and gas-cooled reactors (General Atomics, 1974; Schleicher et al., 2013; Farmer et al., 2006; Steward, 1978; O'Neill, 1965; McNelley, 1964; Martini and Gerosa, 1973; Dorner and Sommer, n.d.; Smidt and Sommer, 1965; Lipps, 1968). A couple of examples of fuel rod designs optimized for use in B&B systems, with annular metallic fuel and fission gas venting, are shown in Fig. 7.

Fuel and coolant choices

A large number of potential B&B fuel and coolant options have been assessed in scoping studies (for example Petroski et al., 2013; Qvist and Greenspan, 2016). An ideal B&B fuel candidate has to offer the following characteristics:

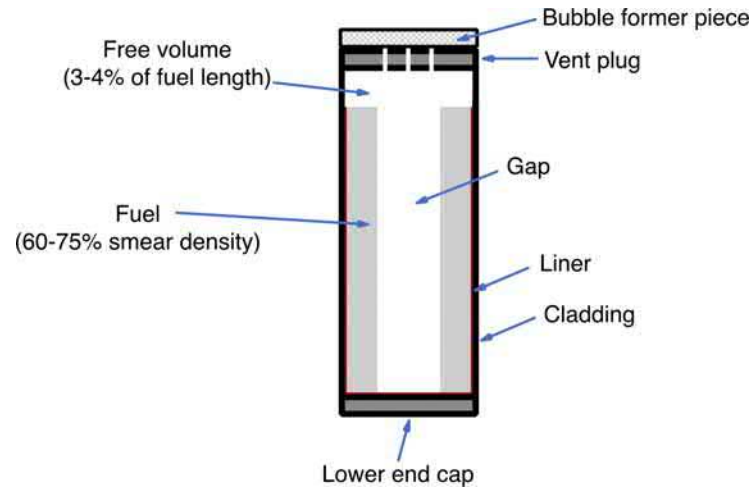


Fig. 7 Example of a vented B&B fuel rod design (not to scale) (Qvist et al., 2015).

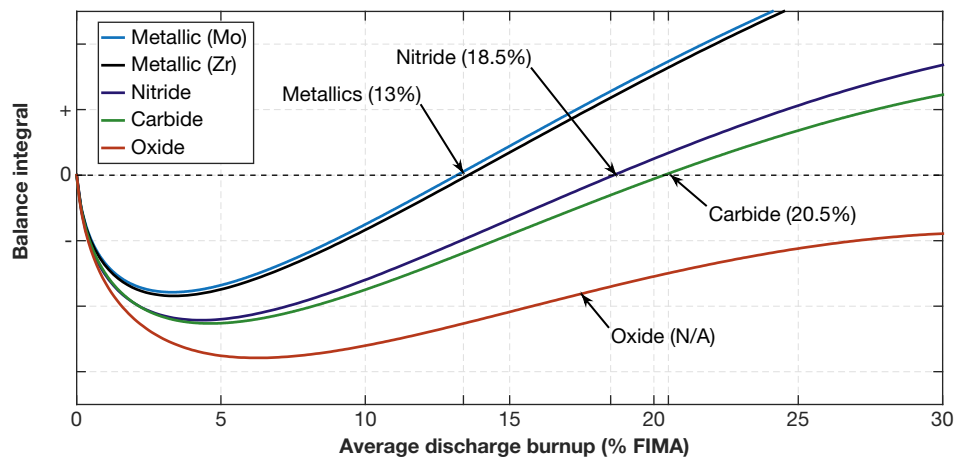


Fig. 8 Unit-cell neutron balance analysis results for different fuel types (zero neutron loss, sodium coolant, HT9 steel structures), nitride-fuel is 99% ^{15}N (Qvist and Greenspan, 2016).

1. Maximized actinide atom density in the core
2. Minimized moderation of neutrons by nonactinide components of the fuel
3. Minimized absorption of neutrons by nonactinide components of the fuel

The effective core-averaged actinide density depends on the fuel volume fraction, fuel chemical composition, fuel nominal density and smear density. Metallic fuels (typically uranium alloyed with molybdenum or zirconium) have consistently been found to be the most viable fuel options for solid-fuel B&B reactors. It is not possible to sustain *fertile-feed* B&B operation using the oxide fuel type (UO_2) in use in light and heavy water reactors. Non-metallic fuel options such as U_3Si_2 and U^{15}N are potentially viable for B&Bs but require a higher burnup cycle than metallic alternatives. Carbide fuels may only work in very low-leakage systems at very high levels of discharge burnup and, hence, neutron irradiation damage. An example of the neutron balance integral (left side of Eq. 4) for various solid-fuel options in an example sodium-cooled system with HT9-steel structure is shown in Fig. 8. The corresponding technical data for the fuel options is summarized in Table 6.

Essentially all major fast reactor coolant options have been studied and shown to be viable for solid-fuelled B&B reactor applications (for examples see Tables 2 and 3), including:

- Sodium
- Lead & Lead-bismuth eutectic
- He and CO_2 gas

For fluid-fuelled BBMSR systems, only chloride-based fuel (and coolant) salts, with chlorine enriched in ^{37}Cl , have so far been shown to be neutronically viable. The fuel options studied and found viable with fertile feed-fuel include various compositions

Table 6 Fuel parameters for B&B core in Fig. 8 (Fuel/coolant/structure vol. fraction = 43.7%/39.9%/16.4% Qvist and Greenspan, 2016).

Fuel type	UO ₂	UN	UC	U-10Zr	U-10Mo
Fuel mass fraction of actinide (%)	88.1	94.4	95.2	90	90
Smear density (%)	85	85	85	75	75
Fuel porosity (as manufactured, %)	6	6	6	0	0
Fuel density at operating temp. (g/cm ³)	10.26	13.59	12.58	15.80	16.70
Fuel density after swelling (g/cm ³) ^a	8.73	11.56	10.81	11.86	12.54
Fuel actinide density (g/cm ³)	9.05	12.84	11.98	14.22	15.03
Swollen fuel actinide density (g/cm ³)	7.69	10.91	10.30	10.67	11.29
Core fuel density (g/cm ³)	4.48	5.94	5.49	6.90	7.30
Core actinide density (g/cm ³)	3.95	5.61	5.24	6.21	6.57
Core fuel non-actinide density (g/cm ³)	0.53	0.33	0.25	0.69	0.73

^a“Swelling” here refers to the state at full radial swelling (out to the inner wall of the cladding) and no axial swelling. The metallic fuels will suffer axial swelling (the value of which can vary greatly with different compositions) which has not been accounted for in this table.

Table 7 Passive reactivity control concepts developed specifically for B&B reactors.

Reactivity control system concept	Actuation signal	Absorber
TWR Thermostat (Teller et al., 1996)	Coolant temperature	⁶ Li
Autonomous Reactivity Control (ARC) (Qvist and Greenspan, 2014a,b; Qvist et al., 2016)	Coolant temperature	⁶ Li
FAST (Hartanto et al., 2015; Hartanto, 2015; Hartanto et al., 2016a,b)	Coolant temperature and coolant flow rate	B ₄ C
SAFE (Hartanto, 2015; Hartanto et al., 2016a,b)	Coolant temperature	B ₄ C

of NaCl-UCl₃, NaCl-UCl₃-UCl₄, UCl₄-UCl₃ as well as pure UCl₄ and a mixed Uranium-Thorium feed fuel composition of NaCl-ThCl₄-UCl₄ (Hombourger et al., 2015, 2017, 2019; Martin et al., 2017; Koivisto, 2019).

Designing for inherent safety

An important design objective of fast reactor cores is inherent safety – that is, the combined effect of all the inherent changes in the core due to loss of coolant or due to increase in power will result in the slowing down of the chain reaction without active intervention of reactor operators. It is very difficult to design a very low-leakage fast reactor core, such as required for B&B systems, to have a combination of inherent reactivity feedbacks to keep core component temperatures sufficiently low in accidents. If the neutron leakage probability at nominal operating conditions is small, which is a requirement for a good neutron economy, the negative feedback associated with loss of coolant will have a small negative impact on the core reactivity. Moreover, the negative reactivity feedbacks due to core radial expansion and due to fuel axial expansion caused by core temperature increase in physically large low-leakage core are smaller as compared to conventional smaller, higher-leakage cores (Wade and Fujita, 1989; Qvist and Greenspan, 2012). The challenge of designing large low-leakage fast B&B reactor cores to be inherently safe has led to the development of passively actuated engineered reactivity control systems, as summarized in Table 7. A very extensive review of other applicable solutions (although not specifically developed for B&B systems) is available in ref. Qvist (2017).

Summary

After many decades of mainly theoretical studies, the physics of B&B systems is now well understood and established, and the technology is now being commercialized. The B&B fuel cycle offers the potential of extremely efficient use of existing uranium resources with minimal need for mining, enrichment and reprocessing. The challenges identified in the very first publication on B&B systems in the late 1950s, the requirement of a large low-leakage core and high burnup, still present a challenge to present-day B&B developers. However, a wide range of possible solutions are being developed to overcome these issues. This chapter serves as an introduction to the general concept of B&B and the basic physical principles involved. In the following four chapters, researchers and commercial reactor vendors present in greater detail specific B&B reactor systems and their current state of development. Chapter Molten Chloride Fast Reactors (MCFRs) and Traveling wave reactor (TWR) presents the efforts of Terrapower in developing solid and liquid fuel B&B reactors respectively. In Chapter CANDLE Reactor Concept, Professor Hiroshi Sekimoto presents the concept of CANDLE, perhaps the most well-known and most comprehensively researched of all traveling-wave B&B systems to date. In Chapter Seed driven breed-and-burn blanket reactor concepts, Professor Ehud Greenspan presents the novel “seed-and-blanket” concept that makes extensive use of the B&B principle in a solid-fuelled system while adhering to existing irradiation damage limitations and constraints.

B&B development and commercialization will likely follow a stepwise approach, with initial prototype reactors either making partial use of the B&B principle or first operating on a low-enriched feed-fuel rather than the more challenging fertile fuel such as natural or depleted uranium. The key enabling technology for conventional solid-fuel B&B technology is the development of fuels and structural steels that can tolerate very high levels of burnup and irradiation damage. In the last few years, a completely new category of B&B systems is being developed that may offer an effective way to circumvent the problems related to irradiation damage in solid core materials by using a liquid fuel and coolant mixture. Technology pathways are maturing that can lead the way toward fulfilling the *full* promise of breed & burn, that of unlocking the massive energy content of depleted uranium without reprocessing.

See Also: A Seed-Driven Breed-and-Burn Blanket Core Concept; CANDLE Reactor Concept; Molten Chloride Fast Reactors (MCFRs); Traveling Wave Reactor (TWR®).

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Molten Chloride Fast Reactors (MCFRs)

Jiri Krepel^a and Kevin J. Kramer^b, ^a Paul Scherrer Institute, Villigen, Switzerland; and ^b TerraPower, LLC, Bellevue, WA, United States

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Glossary

Actinides Eigen-composition Actinides composition which results from closed fuel cycle operation. It is specific and inherent for every reactor and it depends on its spectrum and feed type.

ALARA As Low As Reasonably Achievable.

ANL Argonne National Laboratory.

B&B Breed-and-Burn; a mode of operation that when at equilibrium, the core breeds fissile fuel in an un-enriched fuel feed and later fissions in-situ a fraction of the bred fuel without need for fuel reprocessing/recycling.

Breeding Converting a fertile isotope into a fissile one by capturing a neutron. Usually the fertile isotope ^{238}U to the fissile isotope ^{239}Pu . Also the fertile isotope ^{232}Th (natural thorium) to the fissile isotope ^{233}U .

Burnup A measure of the amount of fission energy released from a given mass of uranium (heavy metal). Unit is FIMA (Fissions per Initial Metal Atom; usually given in %) or Megawatt-days per kilogram (MWd/kg).

Cladding The structural material that separates the nuclear fuel from the coolant. In fast reactors it is usually made of steel.

DRACS Direct reactor auxiliary cooling system.

Excess reactivity When $K_{\text{eff}} > 1.0$, the core is said to have excess reactivity. Excess reactivity can provide a reserve at the beginning of the fuel cycle to compensate for fuel composition changes during the cycle. It has to be compensated at any given time to maintain a self-sustaining chain reaction.

FCS Functional containment system.

FPS Fission products.

HTGR High temperature gas cooled reactor.

IXC Ion exchange column.

LWR Light water reactor

MCFR Molten chloride fast reactor.

MSBR Molten salt breeder reactor.

MSFR Molten salt fast reactor.

MSR Molten salt reactor.

MSRE Molten salt reactor experiment.

Neutron economy Neutron economy term representing balance between neutron generation and neutron losses in a system. Good neutron economy has high neutron generation and small neutron losses by parasitic absorption in structural material and small leakage from the system.

PHX Primary heat exchanger.

Reactivity "Reactivity" denotes the extent of the deviation in the core effective multiplication factor (usually denoted as k_{eff}) from its value at a just critical state (at which state $k_{\text{eff}} = 1.0$ and the fission rate is constant). When ignoring the effect of

neutron leakage from the core, the neutron balance in the core is measured by the “infinite multiplication factor,” denoted as k_{∞} or k_{inf} .

RVACS Reactor vessel auxiliary cooling system.

SFR Sodium fast reactor.

Subcritical/critical core A core that cannot/can sustain a fission chain-reaction.

Temperature coefficient Relationship between fuel temperature and reactivity.

TWR[®] Traveling Wave Reactor; a solid fuel B&B reactor under development by Terra-Power.

XS Cross section; measures probability of neutron interaction.

Introduction

The Molten Chloride Fast Reactor (MCFR) is a type of Molten Salt Reactor (MSR) utilizing fuel in form of molten chloride salts. Accordingly, it shares many of the general MSR advantages and disadvantages listed in [Ignatiev \(2021\)](#). In theory, they can also be operated with a solid salt, which is occasionally re-melted and purified; the chloride salts can also act as a coolant in solid fuel reactor (e.g. [Taube and Schumacher, 1969](#)). These two options are not discussed here.

Chloride salt based concepts have been proposed already during the MSR pioneering time. At the end of the 1950s, the fluoride salt thorium thermal breeder was, nonetheless, considered as superior option with shorter doubling time. Accordingly, the past MCFR research was mainly based on supplementary theoretical concept studies. The historical experimental data for chlorides are also rather scarce compared to the fluorides as main research direction, which were studied in more detail ([Ignatiev, 2021](#)). In general, it seems that the melting temperatures for both salts are similar. For both salts additional studies are needed to address their corrosivity, compatibility with structural material, redox control, and many other safety and performance oriented parameters. Nonetheless, based on the Hard-Soft and Acids-Bases (HSAB) concept ([Pearson, 1968](#)), chlorine is generally a softer element than fluorine. This can cause issues with stability of chlorine compounds, redox control of the chlorides salt and with their corrosivity. The scarcity of experimental data is also the reason why this chapter focus on MCFR concepts overview and on major differences in neutronics and fuel cycle performance from fluoride based MSRs.

MCFR utilizes, similar to a fluoride based Molten Salt Fast Reactor (MSFR), a fast neutron spectrum to support in-situ fissile fuel breeding and/or utilization of the legacy used fuel. Whereas the MSFR spectrum features the softest fast spectra, the MCFR, due to the absence of moderator and moderating compounds like Be, Li, and F, ensures one of the hardest spectra of all fast reactors. It is common for fast reactors, including the MCFR and MSFR, for their neutron economy to be less sensitive to fission products buildup. This is caused primarily by the lower neutron capture cross-section (XS) of fission products in the fast spectrum, but also by the higher fraction of fissile isotopes in the fuel. In the actinides Eigen-composition ([Krepel and Losa, 2019](#)), which results from the repetitive fuel recycling, the share of fissile nuclides is typically above 10% of the heavy-metal (fuel) in fast reactors. This number does not differ strongly between U-Pu and Th-U cycle. For any thermal thorium breeder the fissile actinide fraction is below 2%. The importance of fission products in thermal spectrum is correspondingly higher. Since the thermal MSR is more sensitive to fission products buildup, aggressive online processing is often applied to counter the effects. Furthermore, because of the constrained neutron economy several thermal concepts rely on ²³³Pa (the precursor of the fissile ²³³U) separation to reduce its parasitic neutron absorption. For these reasons, thermal MSR breeders can only be operated with fluoride salts and low absorption moderators.

Several chloride salts envisioned for MCFRs can provide higher actinide fraction, at the respective melting temperature, as compared to fluoride salts of MSFR or a thermal MSR. The hard neutron spectrum and the related high neutron economy of the MCFR enables strong breeding in the Th-U and U-Pu closed cycles ([Hombourger, 2018](#)) or even operation in the open breed-and-burn U-Pu cycle, in which there is not a necessity to recycle the used fuel. This kind of self-sustaining cycle was first proposed by Feinberg and Kunegin ([Feinberg and Kunegin, 1958](#); [Kunegin and Feinberg, 2011](#)) and is reviewed by [Qvist \(2021\)](#). The possibility to operate breed-and-burn cycle with liquid fuel was pointed out in several publications, among the first being ([Hombourger et al., 2015, 2017, 2019](#); [Latkowski, 2015](#); [Kasam and Shwageraus, 2016, 2017](#); [Martin et al., 2017](#)). In this type of open fuel cycle, substantial fuel utilization can be achieved, as outlined in the dedicated breed-and-burn chapter ([Qvist, 2021](#)).

Recently, the MCFR concept based on chloride fuel salt, along with its breed-and-burn fuel cycle capability has garnered significant commercial interest and is currently under development by TerraPower, LLC (TerraPower[®]) ([Kramer, 2019](#); [Latkowski, 2019](#)). Their MCFR feasibility studies started before 2015, but were not published until 2015 and later ([Latkowski, 2015](#); [TerraPower, 2016](#)). This design is attractive, because the chloride salt offers a high uranium/transuranic solubility. It can be operated as a low-pressure compact system. Structural materials are absent in the core because it is unmoderated and the liquid molten salt fuel serves as the coolant. The breed-and-burn fuel cycle can provide high natural resources utilization without need of fuel enrichment and reprocessing or actinides separation, all of which are considered proliferation risks. As such, the MCFR concept distinguishes itself from other MSR concepts ([Ignatiev, 2021](#)).

As with other MSR concepts, the low-pressure nature of the MCFR and the high solubility of actinides and fission products reduces the driving force for reactor core source term release and makes passive decay heat removal possible, even with high power densities. During normal operation, part of the fission products (FPs), comprising an important part of the source term, can be

removed from the core via gas sparging and mechanical filtering. This will require safety attention and immobilization in engineered areas outside of the reactor core as mentioned in Ignatiev (2021). The fuel salt's large negative temperature coefficient and low excess reactivity, enabled by breed-and-burn operation or by continuous refueling, help ensure that overpower transients are not long lasting or run away. These characteristics coupled with the ability to passively remove decay heat prevent such transients from driving large temperature excursions.

Historical background

The history of the MCFR goes back to the beginning of the nuclear era. Some of the MCFR concepts belong to the family of homogeneous reactors. The aqueous homogeneous reactor from this family was the first reactor considered when the theoretical possibility of chain reaction was assessed in 1939 (Lane et al., 1958). Coincidentally, also in 1939, significant chlorine isotope separation was achieved (Beams and Skarstrom, 1939) by improving earlier established method (Beams and Haynes, 1936). The discovery of ^{239}Pu in 1940 as a ^{238}U transmutation product (Seaborg, 1994) opened new perspective and later resulted in a whole class of fast breeder reactors to which MCFR belongs. Around 1944, a Pressurized Water Reactor (PWR) concept was identified and patented (Wigner et al., 1945) and the term breeder was born (Weinberg, 1987). The design of first fast neutron breeder test reactor cooled by sodium was ongoing soon after (Zinn, 1946).

The MCFR concept was not considered in the period during and after the Manhattan project. Nonetheless, several properties of uranium chlorides were revisited and measured already within this project (Ginnings and Corruccini, 1947). Within the Aircraft Nuclear Propulsion (ANP) project (1947–61), at ORNL the attention moved towards higher temperatures and several additional reactor concepts were evaluated. In Ellis and Thompson (1950), melting temperatures of uranium chlorides are listed. However, other halides than fluorides were not considered for further research, because of their unfavorable neutronic properties. This could be related to the requirement of compact cores for the nuclear-powered jets. In later ANP reports, fast reactors were, in general, excluded because of the potential reactivity control problem caused by the shorter neutron lifetime (Fraas and Savolainen, 1954). When the ANP program was finished, ORNL decided to develop MSR for civilian applications. The graphite moderated thorium breeder Molten Salt Breeder Reactor (MSBR) was chosen for several reasons, the major ones being low fissile inventory, short doubling time, high thermal inertia and appealing Th-U fuel cycle (Briant and Weinberg, 1957). The chlorides were recognized from neutronic perspective as suitable for fast reactors. The need of ^{37}Cl enrichment was perceived as more difficult than the ^7Li enrichment for fluoride salt. Another considered weakness of the MCFR was the large fissile fuel inventory and the related longer doubling time of decades. Also MSFR was excluded because of the longer doubling time and so MSBR was chosen, at that time, as a major research direction with strong experimental support.

Around 1952–53 a group of students at the Oak Ridge School Of Reactor Technology (ORSORT) started designing fused salt fast breeder reactor based on fluoride salts and the Th-U cycle (Wehmeyer et al., 1953). It is a predecessor of the MSFR design. At the same time, a nuclear engineering project at M.I.T. was dedicated to non-aqueous liquid fuels for the U-Pu cycle (Goodman et al., 1952). The resulting MCFR concept was based on homogeneous core surrounded by reflector and blanket. Argonne national laboratory compiled a report (Glassner, 1953) with thermochemical properties of the chlorides up to 2500 K, which was later updated (Glassner, 1957). In 1954 it was followed by a UK report, which was declassified in 1957 (Bradley, 1957) and dedicated to chlorides of actinides and fission products. Interestingly, thorium chloride was included in all these reports. From 1957, there is also one Russian report with chlorides properties (Belyaev et al., 1957). In 1955 and 1956, another two groups of students at ORSORT considered chlorides as immiscible direct coolant for liquid fueled reactor (Burch et al., 1955) and as homogeneous two-fluid MCFR (Bulmer et al., 1956), respectively. In 1958, the *Fluid Fuel Reactors* book (Lane et al., 1958) was published, where one third was dedicated to MSRs. The MCFR concept is not mentioned in this book. In 1959, ORNL published a report with phase diagrams of several fluoride and chloride binary mixtures (Thoma, 1959).

During the 1960s, the majority of the above reports were publicly available and chloride salts were proposed also for pyro-reprocessing (Harder et al., 1969). In 1962, a cooperation agreement was signed between USAEC and EURATOM which covered also MSR (EURATOM, 1963). In 1963, there was a conference on breeding, economics, and safety in large fast power reactors at ANL (Okrent, 1963) with one MCFR contribution (Alexander, 1963). Around this time, the chloride salts as a potential fuel or fuel reprocessing medium were recognized also in other countries, e.g., Poland (Taube, 1961) and later Switzerland (Taube and Ligou, 1972), Germany (Naumann, 1962), and the United Kingdom (Moore and Fawcett, 1966). In 1969, there was a symposium on reprocessing of nuclear fuels at Iowa State University (Chiotti, 1969), where pyro-chemical methods based on chlorides were presented also by scientists from Japan and India. There was some research dedicated to chloride salt properties in Russia (Desyatnik et al., 1970). Nonetheless, the research in Russia, Japan, India, Germany (Vornhusen, 1965) and France (Hery and Lecocq, 1984) was oriented rather on fluorides salts.

Neutronics characteristics of chloride salts

Compared to fluorine, which has only one stable isotope ^{19}F , chlorine has two stable isotopes ^{35}Cl 76% and ^{37}Cl 24%. The ^{35}Cl capture XS is significantly higher than that of ^{37}Cl (see Fig. 1); even though, there is some difference between the XS libraries' values. Therefore, in order to reduce the parasitic neutron capture in chlorine and thus improve the neutron economy, a majority of MCFR

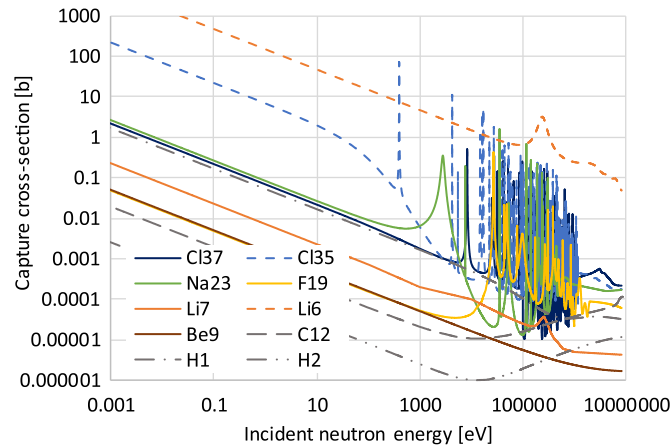


Fig. 1 Neutron capture XS of selected isotopes from ENDF/B-VII.0 XSs library.

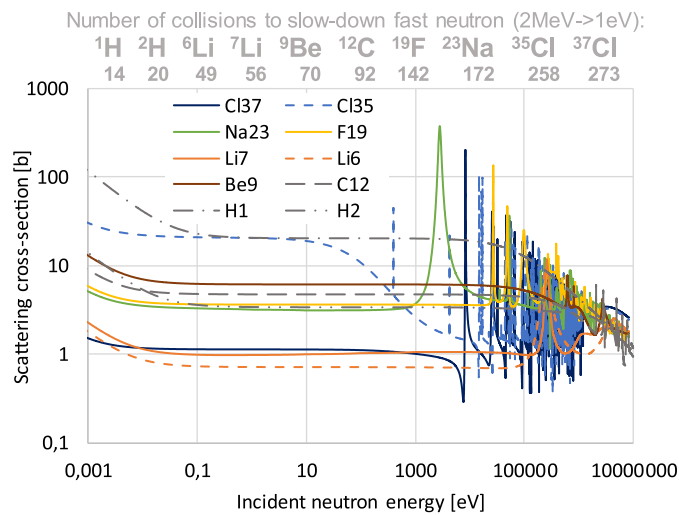


Fig. 2 Neutron scattering XS of selected isotopes from ENDF/B-VII.0 XSs library.

designs assume isotopic enrichment and application of nearly pure ^{37}Cl . This is very similar to the situation with ^6Li (92% of Li) and ^7Li (8%) isotopes where many MSFR and MSBR concepts rely on enriched ^7Li . In both cases the enrichment not only increases the neutron economy, but also reduces the production of radioactive daughter products. Capture XSs of ^{37}Cl and ^{23}Na have similar nature and amplitude. They are more than one order of magnitude higher in the thermal spectrum (below ~ 1 eV) than XSs of ^7Li , ^9Be and ^{19}F , but similar to ^1H capture. In general, the isotopes ^1H , ^{23}Na and ^{37}Cl can be applied for thermal reactors; however, their higher capture burden will prevent a self-sustaining breeding in the Th-U cycle. In the U-Pu cycle, the equilibrium actinides Eigen-composition alone prevents the self-sustaining breeding practically in all thermal reactors (Krepel and Losa, 2019). The data presented in Figs. 1 and 2 are adopted from ENDF/B-VII.0 XSs library.

The second most relevant neutronic parameter is the scattering XS. From this perspective, ^{37}Cl behaves rather like ^7Li than as ^{23}Na and has a quite low scattering XS. It is much lower than the scattering XSs of ^{19}F and ^{23}Na for neutron energies below 1 keV. Furthermore, slowing-down properties of an isotope depend also on the average logarithmic energy decrement imparted during a neutron scattering interaction. This decrement can be used to enumerate number of required collisions to reduce the neutron energy, for instance, from 2 MeV to 1 eV. Since ^{37}Cl is the heaviest isotope in this discussion, it needs the highest number of collisions ~ 273 to slow down a neutron to 1 eV (see Fig. 2). The same number for ^7Li is 56. Furthermore, ^7Li has quite broad resonance around 260 keV. This is also the reason, why homogeneous MSFR with $^7\text{LiF-BeF}_2$ carrier salt was considered as moderated reactor during the MSBR project at ORNL. Even if BeF_2 is omitted and solely ^7LiF carrier salt is used, the fast MSFR spectrum is still very soft. On the other hand, MCFR based on a Na^{37}Cl carrier salt provides one of the hardest neutron spectra possible. Harder spectra can be obtained only with actinide chlorides omitting ^{23}Na , or with molten metallic actinide slurries. The discussion of available chloride carrier salts with limited moderation was thoroughly done in Taube (1961) considering also other criteria. As most suitable compounds Na, K, Mg, Ca and eventually CuCl (CuCl_2) and ZnCl_2 were identified. In the TerraPower patent (TerraPower, 2016) also BaCl_2 , SrCl_2 , VCl_3 , CrCl_3 , TiCl_4 and ZrCl_4 are listed together with the mixture of chloride and fluoride salts.

The neutronic properties of ^{37}Cl provide very hard neutron spectrum and very low parasitic neutron absorption. The actinides' Eigen-composition of equilibrium Th-U and U-Pu cycles in such spectrum results in a very high neutron economy (Krepel and Losa, 2019) and thus enables new fissile fuel breeding and/or utilization of legacy used fuel. At the same time, the absence of strong scattering and capture reaction rates has a consequence in very large neutron mean free path (average distance a neutron travels until it collides). A typical homogeneous MCFR core is transparent for neutrons and thus leaky. With high associated neutron leakage, the need of reflector or blanket is more pronounced in MCFR than in MSFR. It is also the reason, why salts with high actinides fraction are considered for MCFR.

Historical and current MCFR concepts

MCFR designs can be generally divided into homogeneous and heterogeneous reactors. In the (Goodman et al., 1952 and Bulmer et al., 1956) reports the external and internal cooling option is discussed. It is practically a synonym for homogeneous (external cooling) or heterogeneous (internal cooling) core geometry. The advantage of external cooling is that the parasitic capture of structural materials and coolant in the core is minimal. At the same time, the fuel salt acts as a coolant and must be pumped to the external heat exchanger. This layout increases fissile fuel inventory and results in delayed neutrons loss in primary circuit. The necessary salt volume can be optionally reduced by integral core design, where volume of piping and size of the heat exchangers is minimized. The advantage of internal cooling, on the other hand, is that the salt does not need to be pumped out of the core for the cooling purpose. Presence of the dedicated coolant and separating structural materials in the heterogeneous core, however, deteriorates the neutron economy by softening of the neutron spectrum and increasing parasitic neutron absorption. Many of these concepts rely on metallic tubes with very thin walls, profiting from the low pressure difference, or on SiC tubes, because of its low parasitic capture.

The advantages of homogeneous and heterogeneous core geometry can be combined in the directly cooled MCFRs. In (Burch et al., 1955) direct cooling of the liquid fuel, based on uranium solution in molten bismuth, was considered by an immiscible coolant (KCl-LiCl). In this internal cooling case, the structural material can be avoided and liquid fuel circulates only within the core. The largest drawback of the directly cooled MCFRs is the materials exchange between the fuel and coolant. An extreme direct cooling option is represented by boiling coolant salt (Taube et al., 1967). Based on the terminology from several reports, e.g. (Smith et al., 1974 or Taube, 1977), three types of MCFR concepts can be identified depending on heat exchange properties. These types are labeled here as:

1. Homogeneous MCFR: externally indirectly cooled
2. Heterogeneous MCFR: internally indirectly cooled
3. Directly cooled MCFR: internally or externally directly cooled

Since the third option, direct cooled reactors, is rather exotic, the MCFR can be generally divided into heterogeneous and homogeneous concepts. The MCFR designs can be further classified as with other MSRs by number of actinides-containing salts including one-fluid, two-fluid or multi-fluid options. As with any other reactors, MCFRs can be also classified by purpose as burner, converter, or breeder operated in open or closed Th-U or U-Pu cycle. Very particular is the breed-and-burn operation mode.

Breed-and-burn cycle in MCFR

Breed-and-burn cycle is a special type of open fuel cycle, where substantial fuel utilization can be achieved, as outlined in the dedicated breed-and-burn chapter (Qvist, 2021). Both, homogeneous (Hombourger et al., 2015, 2017, 2019; Latkowski, 2015; Martin et al., 2017) and heterogeneous (Kasam and Shwageraus, 2016, 2017; Kasam et al., 2020; Cuoc et al., 2020). MCFR have a potential to be operated in breed-and-burn cycle. Nonetheless, the neutron economy is tighter in heterogeneous MCFR.

There is substantial difference between liquid-fuel and solid-fuel reactors operating in the breed-and-burn cycle. In both cases, the fresh fuel used for repetitive refueling of the reactor consists only of fertile material, which is either natural or depleted uranium. In the solid-fuel reactor, the highest-burnup fuel is discharged from the reactor and the average content of fissile material makes the core critical, even though the fresh fuel has nearly zero fraction of fissile material (see Fig. 3). On the other hand, in case of single-fluid MCFR, the fuel is homogenized in the reactor and the discarded fuel has the average burnup (see Fig. 4). As a consequence, the discharged fuel can be accumulated and later used to start daughter breed-and-burn MCFR. To achieve higher burnup or to reduce slightly the core size, multi-fluid MCFR layout can be applied (see Fig. 5). In this case, the molten-salt in the outermost annulus acts as a blanket (Raffuzzi and Krepel, 2020) and the discharged fuel, typically from the inner core zone, will have higher than average burnup and it may not be used to start another breed-and-burn MCFR.

The performance of a breed-and-burn reactor can be assessed by neutron balance concept, which is discussed in (Qvist, 2021). This concept is applicable also to MCFR. Nonetheless, there is one major difference caused by liquid fuel homogenization. The burnup and composition of a single solid fuel assembly is evolving according to the local flux, but independently from the other assemblies. At the end of irradiation, it is discharged from the core at maximal burnup. In case of liquid fuel, the fuel at different stages of burnup is mixed together. Hence, the discharged fuel has an average burnup and composition. The composition can be simulated by averaging over the residence time distribution (Hombourger et al., 2019). Fig. 6 illustrates the impact of the fuel mixing on the neutron balance evolution. The example was obtained for 68%Na ^{37}Cl -32%U $^{37}\text{Cl}_3$ salt, which was treated in two

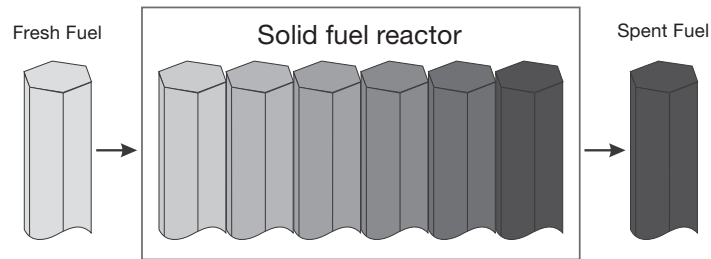


Fig. 3 Schematic illustration of burnup evolution in a solid fuel breed-and-burn reactor. From Hombourger B (2018) *Conceptual Design of a Sustainable Waste Burning Molten Salt Reactor*. Doctoral Thesis, EPFL Lausanne, Switzerland.

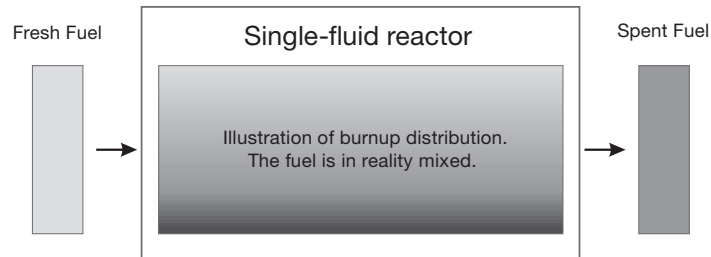


Fig. 4 Schematic illustration of burnup evolution in a single-fluid breed-and-burn reactor. From Hombourger B (2018) *Conceptual Design of a Sustainable Waste Burning Molten Salt Reactor*. Doctoral Thesis, EPFL Lausanne, Switzerland.

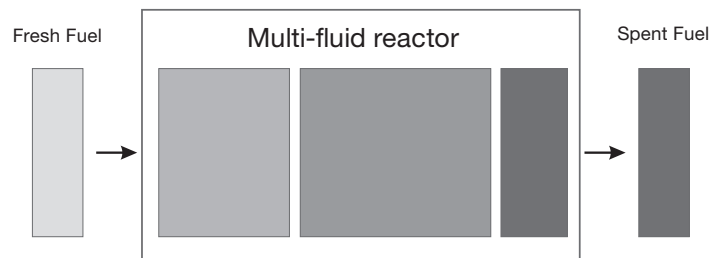


Fig. 5 Schematic illustration of burnup evolution in a three-fluid breed-and-burn reactor. From Hombourger B (2018) *Conceptual Design of a Sustainable Waste Burning Molten Salt Reactor*. Doctoral Thesis, EPFL Lausanne, Switzerland.

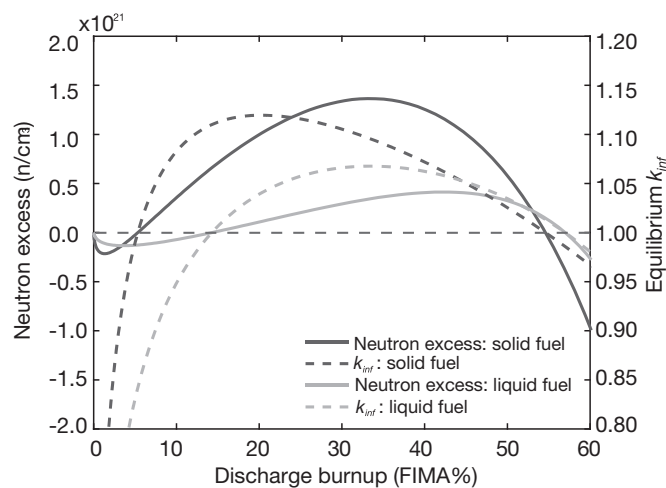


Fig. 6 Neutron excess and infinite neutron multiplication factor k_{inf} as a function of discharge burnup for solid fuel reactor and liquid fuel reactor with fuel mixing. From Hombourger B, Krepel J, and Pautz A (2019) Breed-and-burn fuel cycle in molten salt reactors, progress in the science and technology of nuclear reactors using molten salts. *EPJ Nuclear Sciences & Technologies* 5.

independent cell-level simulations as a liquid and as a solid fuel, respectively. In both cases, the specific power of the MSFR design $\sim 300 \text{ W/cm}^3$ was adopted. This value can be too high for large MCFR cores. Nonetheless, the dependence of the U-Pu cycle performance on the specific power is much milder than for the Th-U cycle (Krepel and Losa, 2019). Hence, the curves will not change significantly for lower specific power.

Fig. 6 shows that in liquid fuel reactor, the neutron excess curve becomes positive for much higher burnup than in solid fuel reactors. It has a lower maximum and flatter decline. For more detailed discussion about the important points of the neutron excess curve and their meaning refer to (Qvist, 2021). The maximal infinite neutron multiplication factor k_{inf} of the liquid fuel reactor is achieved at burnup around 30–40% FIMA. In solid fuel reactor, the curve is peaking earlier and at higher values. Since each refueling rate in liquid fuel reactor results in specific discharge burnup, the neutron excess and the related k_{inf} are, actually, a function of the refueling rate. The maximal k_{inf} is obtained for refueling rate corresponding roughly to 3% per year. Nonetheless, the position of k_{inf} maximum slightly differs between different salt compositions (see left Fig. 7). The k_{inf} curve for liquid fuel reactor in Fig. 6 and left Fig. 7 is rather symmetric with burnup. Similar curve as in (Hombourger et al., 2019) can be found also in (Martin et al., 2017) or (Raffuzzi and Krepel, 2020).

The minimal and maximal burnup, which allows for MCFR operation in the breed-and burn mode, depends on the k_{inf} curve and on the core size, which determines the neutron leakage and so the relationship between k_{inf} and k_{eff} . Hence, in the actual MCFR design, core size as one design parameter directly competes with the fuel burnup. From neutron balance perspective, the burnup, in the cases with very high uranium concentration in the salt, may be as high as 55% FIMA. The minimal MCFR burnup, necessary for breed-and burn mode of operation, is around 20–25% FIMA.

Based on the maximal k_{inf} values from left Fig. 7, the minimum dimensions of reflected single-fluid MCFR cores as estimated by Hombourger et al. (2019) are given in right Fig. 7. These core volumes are quite large: $\sim 50 \text{ m}^3$ for salts with very high actinides share and $\sim 100 \text{ m}^3$ for more realistic salt without UCl_4 . If the core is to deliver a thermal power of 5000 MW, as of the largest commercial light-water-reactor, the respective specific power needs to be 100 and 50 W/cm^3 . This power density is lower than the 300 W/cm^3 assumed in the calculations the results of which are reported in Figs. 6 and 7. However, Martin et al. (2017) are reporting that k_{inf} slightly increases as the assumed power density is lower implying that it might be possible to slightly reduce the Fig. 7 reported core volumes. The actinides inventory in the respective cores is slightly less than ~ 100 tons and ~ 120 tons. Accounting for the salt volume outside of the core (circulating through the heat exchangers) the overall inventory will be even higher. The specific actinides inventory of the example core power is, respectively, $> 20 \text{ kg/MW}_{th}$ and $> 24 \text{ kg/MW}_{th}$. These are to be compared with $> 17 \text{ kg/MW}_{th}$ in conventional solid-fuel liquid-metal cooled reactors and $\sim 24 \text{ kg/MW}_{th}$ in commercial pressurized water reactors.

The initial loading of a breed-and-burn MCFR core can be either enriched uranium or Trans-uranic elements recycled from existing used nuclear fuel. In both cases, the fissile fuel inventory in the initial fuel is around 10–12% (Hombourger et al., 2019). The TerraPower MCFR design, described in this chapter as a reference, will be started by enriched uranium and achieves burnup of 20% FIMA with an 1800 MW_{th} design over 60 years if employing a once through fuel cycle. After this initial cycle, the reactor could continue operations in breed-and-burn cycle beyond 60 years with irradiated fuel used to support daughter reactors.

The discharged burnup in a natural (or depleted) uranium fed breed-and-burn core is, after reaching equilibrium composition, the natural uranium utilization. Accordingly, breed-and-burn MCFR might achieve a very high uranium utilization of $\sim 35\%$. This is ~ 60 times the utilization of natural uranium in contemporary light water reactors; both operating in a once-through fuel cycle. In a solid fuel breed-and-burn reactors the discharge burnup is limited by radiation damage to the fuel cladding to less than $\sim 20\%$ FIMA (Qvist, 2021; Hejzlar, 2021). The molten salt, as an ionic liquid, is not constrained by the radiation damage. What is

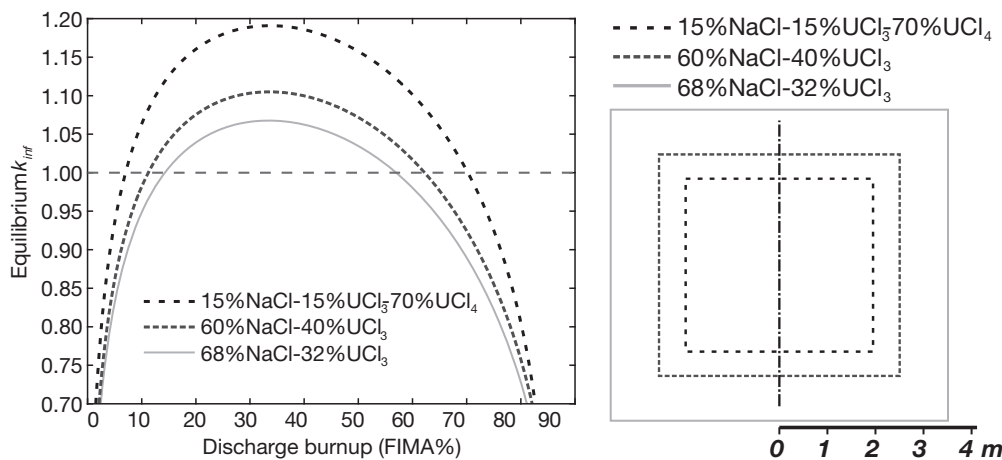


Fig. 7 Evolution of k_{inf} (left) and minimum core size (right) dependence on the salt composition; the dimensions pertain to the peak k_{inf} FIMA. From Hombourger B, Krepel J, and Pautz A (2019) Breed-and-burn fuel cycle in molten salt reactors, progress in the science and technology of nuclear reactors using molten salts. *EPJ Nuclear Sciences & Technologies* 5.

challenging the MCFR design are: (1) finding a fuel salt mixture which allows for high actinides and FPs concentration and, at the same time, has suitable chemical and physical properties, (2) fast neutron induced radiation damage to the core vessel and (3) the large volume required for the core.

Directly cooled MCFR

Molten salts are limited as coolants due to the poor heat transfer properties. Compared to liquid metals their thermal conductivity is very low (typically 0.1–0.6 W/mK) and rather comparable to water. Hence, large heat-exchange surface is usually needed. The idea of liquid fuel cooling by direct contact with salt coolant liquid was presented in Burch et al. (1955) where uranium solution in molten bismuth was cooled by KCl–LiCl. This solution may reduce the fissile fuel inventory. The heat exchangers can be designed for the cooling medium and not for the fuel. However, the coolant will be strongly contaminated by the direct contact with fuel. As mentioned by Taube (1977) this is extremely exotic reactor type. Since there are not many concepts in this category, few non-chloride designs with fluoride salts or metallic fuels are also included. The overview of the concepts is shown in Table 1 and Fig. 8.

Heterogeneous MCFR

The basic characteristics of heterogeneous MCFR concepts is that the primary heat exchange between the fuel salt and coolant take place in the active core. The material separating these fluids must be compatible with both of them and withstand strong neutron irradiation and high temperatures. Furthermore, it should not have high neutron capture XS, which would deteriorate the neutron economy. Since the pressure difference between the liquids may be low, thin metallic tubes are proposed to minimize the neutron absorption. Other proposed option with low neutron capture XS is SiC.

The heterogeneous MCFR acts to a certain extent as a reactor with solid fuel. Hence, it must rely on heat conduction by fluid coolant. Furthermore, the heat removal from the active core is proportional to the heat-exchange surface area. Accordingly, the power rating and core size are tightly linked together as for solid fuel reactors. This link does not exist for homogeneous MCFRs, where the heat can be conducted directly by the fuel and the core size is limited rather by neutronic constraints or by thermal properties of the surrounding solid materials. The heterogeneous MCFR concepts overview is presented in Table 2 and Fig. 9.

Homogeneous MCFR

One major advantage of a homogeneous MCFR is that the core zone is filled solely by the fuel salt, which acts simultaneously as a coolant. In opposite to heterogeneous MCFR, where the core power depends on heat exchange surface in the core, in homogeneous MCFRs the heat is released directly in the circulating salt. The power of the MCFR is thus limited rather by thermal-hydraulics and thermal-mechanics constraints. The typical power considered for MCFR is 3GW_{th} .

The homogeneous MCFR concepts tend to have superior neutronic economy. At the same time, the fuel salt, as a coolant, must be pumped to the primary heat exchangers. The salt volume outside of the core is thus substantial and the actinides load is usually high. The necessary salt volume can be optionally reduced by integral core design. The superior neutronic economy enables operation in breed-and-burn fuel cycle. Nonetheless, it is still needed to minimize neutron leakage, hence the breed-and-burn MCFRs are usually large. Accordingly, they would be capable to produce much higher power than 3GW_{th} . The homogeneous MCFR concepts overview is presented in Table 3 and Fig. 10.

Table 1 Overview of reactor concepts with liquid fuel cooling by direct contact with coolant liquid.

<i>Directly cooled MCFRs</i>						
<i>No. Reference</i>	<i>Breeder/cycle</i>	<i>Fuel type</i>	<i>Blanket type</i>	<i>Primary coolant</i>	<i>Core layout/moderator</i>	<i>Core shape</i>
1 Burch et al. (1955)	Yes/Th-U cycle	U solution in molten bismuth	Th dispersion in bismuth	KCl–LiCl	Heterogeneous/graphite	Cylindrical
2 Hammond et al. (1961)	Yes/U–Pu cycle	Molten Pu alloy	UO ₂ –Na paste	Sodium	Homogeneous/none	Spherical
3 Moore and Fawcett (1966)	Yes/U–Pu cycle	Chloride salt	Chloride salt	Lead	Homogeneous/none	Cylindrical
4 Taube et al. (1967)	Yes/U–Pu cycle	Chloride salt	Chloride salt	Boiling AlCl ₃	Homogeneous/none	Semi-spherical
5 Gat (1967)	Yes/Th-U cycle	Fluoride salt	Fluoride salt	Lead	Heterogeneous/graphite	Cylindrical
6 Smith et al. (1974)	Yes/U–Pu cycle	Chloride salt	Chloride salt	Lead	Homogeneous/none	Cylindrical
7 Smith et al. (1974)	Yes/U–Pu cycle	Chloride salt	Chloride salt	Lead	Homogeneous/none	Cylindrical

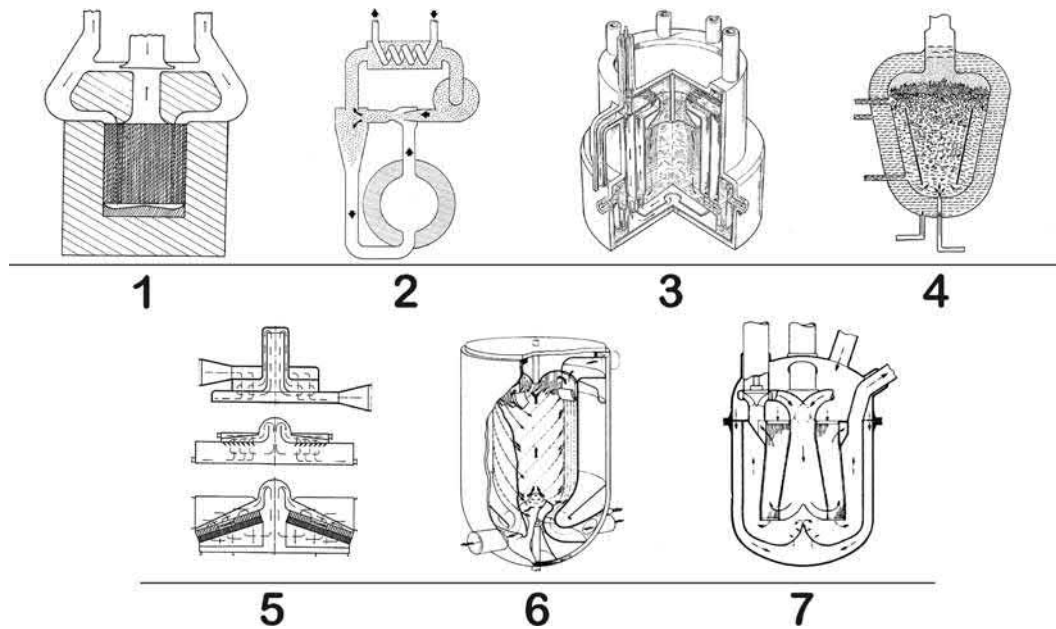


Fig. 8 Depiction of each reactor concept with direct contact of liquid fuel and coolant identified in **Table 1**. From (1) Burch WD, Cater CH, Hathway DL, Maxon BS, Williamsen NR, and Yarbrow OO (1955) *Immiscible-Liquid-Cooled Fluid-Fuel Reactor*. Oak Ridge, TN: Oak Ridge School of Reactor Technology, CF-55-8-188; (2) Hammond RP, Stanford REL, and Humphreys JR (1961) *Mobile fuel Plutonium Breeders*. Los Alamos Scientific Laboratory, LA-2644; (3) Moore RV and Fawcett S (1966) Present and future types of fast breeder reactors. In: *Proceedings of the London Conference on Fast Breeder Reactors Organized by the British Nuclear Energy Society*; (4) Taube M, Mielcarski M, Poturaj-Gutniak S, and Kowalew A (1967) New boiling salt fast breeder reactor concepts. *Nuclear Engineering and Design* 5 (2), 109–112; (5) Gat U (1967) Cooling concepts for a compact MOSEL (molten salt) reactor. *Nuclear Engineering and Design* 5: 113–122; (6) Smith J, et al. (1974) *An assessment of a 2500 MWe Molten Chloride salt Fast Reactor*. United Kingdom Atomic Energy Authority, AEEW-R956; (7) Smith J, et al. (1974) *An assessment of a 2500 MWe Molten Chloride salt Fast Reactor*. United Kingdom Atomic Energy Authority, AEEW-R956.

Table 2 Overview of the heterogeneous MCFR concepts.

Heterogeneous MCFRs						
No.	Reference	Breeder/cycle	Fuel type	Blanket type	Primary coolant	Core shape
1	Alexander (1963)	Yes/mixed cycle	Chloride salt with U-Pu	Chloride salt with Th-U	Gas	Semi-spherical
2	Taube and Ligou (1972, 1974); Taube (1974a)	Yes/U-Pu cycle	Chloride salt with Pu	Chloride salt with U	Blanket salt	Cylindrical
3	Taube (1974a)	No/waste burner	Chloride salt with Pu	Salt with FPs and Be	Salt with FPs and Be	Cylindrical
4	Taube (1974a)	Yes/waste burner	Chloride salt with Pu	Chloride salt with U	Salt with FPs and Be	Cylindrical
5	Huke et al. (2015)	Yes/U-Pu cycle	Chloride salt	Chloride salt	Lead	Cylindrical
6	Scott et al. (2015)	Yes/U-Pu cycle	Chloride salt	None	Fluoride salt	Cylindrical
7	Kasam and Shwageraus (2016, 2017); Kasam et al. (2020); Cuoc et al. (2020)	Breed-and-burn cycle	Chloride salt with U-Pu	None	Chloride salt	Cylindrical

Representative current MCFR development

One currently proposed MCFR concept under TerraPower, LLC commercial development utilizes a breed-and-burn approach based upon the uranium-plutonium fuel cycle (Kramer, 2019). This MCFR core is well-mixed because all the fuel salt is fully circulating about once per second. This design utilizes a potential breed-and-burn fuel cycle is illustrated in **Fig. 11**. Following initial start-up with enriched ($\sim 12\%$ ^{235}U) fuel, this MCFR design does not require ongoing fissile material makeup. Instead, an MCFR can be fed by depleted or natural uranium. During normal operations, the reactor reactivity tends to slowly increase as a result of a neutron conversion ratio exceeding 1. To counter this increase in reactivity, a small quantity of fully mixed fuel salt is removed and replaced with fertile feed salt. The addition of fertile material acts, in effect, as a burnable poison that reduces reactivity.

This MCFR does not require enriched fissile material following its initial charge.

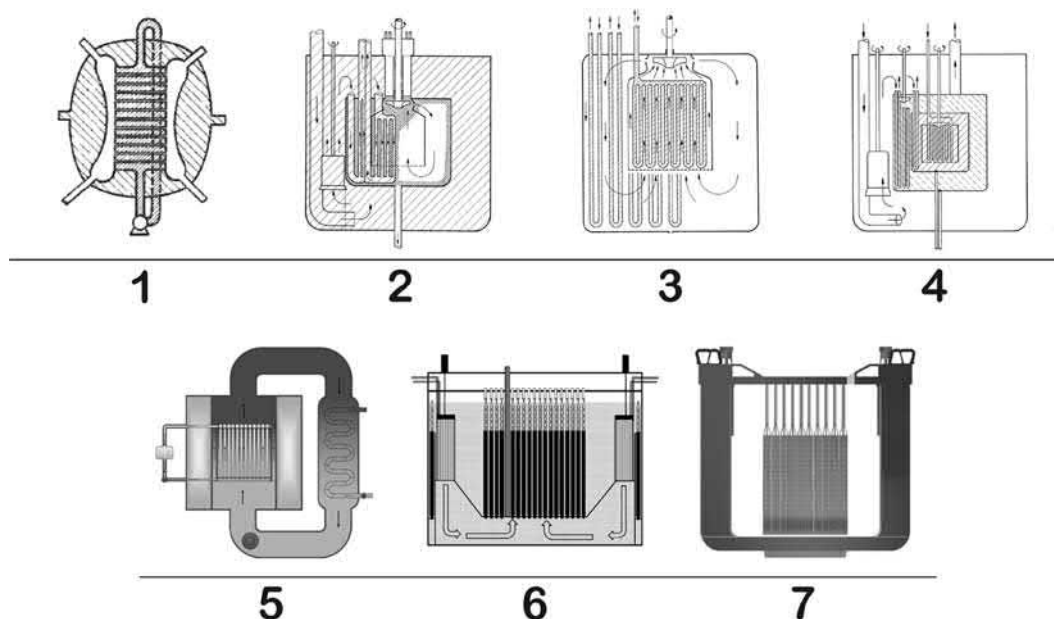


Fig. 9 Depiction of each heterogeneous MCFR concept identified in Table 2. From (1) Alexander LG (1963) Molten-salt fast reactors. In: *Proceedings of the Conference on Breeding, Economics and Safety in Large Fast Power Reactors*, October 7–10; (2) Taube M and Ligou J (1972) *Molten Chlorides Fast Breeder Reactor Problems and Possibilities*, Report EIR-215. Wurenlingen, Schweiz: Eidg. Institut für Reaktorforschung; (3) Taube M (1974a) *A Molten Salt Fast Thermal Reactor System With No Waste*, Report EIR-249. Wurenlingen, Schweiz: Eidg. Institut für Reaktorforschung; (4) Taube M (1974a) *A Molten Salt Fast Thermal Reactor System With No Waste*, Report EIR-249. Wurenlingen, Schweiz: Eidg. Institut für Reaktorforschung. (5) Huke A, et al. (2015) The dual fluid reactor—A novel concept for a fast nuclear reactor of high efficiency. *Annals of Nuclear Energy* 80: 225–235; (6) Scott I, et al. (2015) Stable salt reactor design concept. In: *Thorium Energy Conference 2015 (ThEC15)*, Mumbai, India, October 12–15; (7) Kasam A, et al. (2020) Thermal-hydraulic design methodology and trade-off studies for a dual-salt breed-and-burn molten salt reactor. *Nuclear Engineering and Design* 360: 110481.

Table 3 Overview of the homogeneous MCFR concepts.

Homogeneous MCFRs					
No. Reference	Breeder/cycle	Fuel type	Blanket type	Specific feature	Core shape
1 Goodman et al. (1952)	Yes/U-Pu cycle	Chloride salt	Chloride salt	Reflector between fuel and blanket	Spherical
2 Bulmer et al. (1956)	Yes/U-Pu cycle	Chloride salt	UO ₂ -Na paste		Spherical
3 Nelson et al. (1967)	Yes/U-Pu cycle	Chloride salt	Chloride salt		Spherical
4 Smith et al. (1974)	Yes/U-Pu cycle	Chloride salt	Chloride salt	Secondary coolant lead	Spherical
5 Smith et al. (1974)	Yes/U-Pu cycle	Chloride salt	Chloride salt	Secondary coolant helium	Spherical
6 Taube (1974b)	Yes/mixed cycle	Chloride salt with U-Pu	Chloride salt with Th-U		Spherical
7 Taube et al. (1975a, 1975b)	No/waste burner	Chloride salt in outer sphere	Central waste burning zone	Intermediate Be moderator loop	Spherical
8 Ottewitte (1978, 1982)	Yes/Th-U cycle	Chloride salt	Chloride salt		Cylindrical
9 Taube and Heer (1980)	Yes/U-Pu cycle	Chloride salt	None	Reflected by chloride salt	Spherical
10 Mourgov and Bokov (2004, 2006)	Yes/U-Pu cycle	Chloride salt	None		Cylindrical
11 Hombourger et al. (2015, 2017, 2019)	breed-and-burn cycle	Chloride salt	None		Cylindrical
12 Latkowski (2015); Kramer (2019); Cisneros et al. (2019)	Breed-and-burn cycle	Chloride salt	None		Cylindrical
13 Raffuzzi and Krepel (2020)	Breed-and-burn cycle	Chloride salt	Fuel salt	Preliminary study of 1 or 2 blankets	Cylindrical
14 De Oliveira and Hombourger (2020)	Breed-and-burn cycle	Chloride salt	None	Fuel flow shaping by baffles	Cylindrical

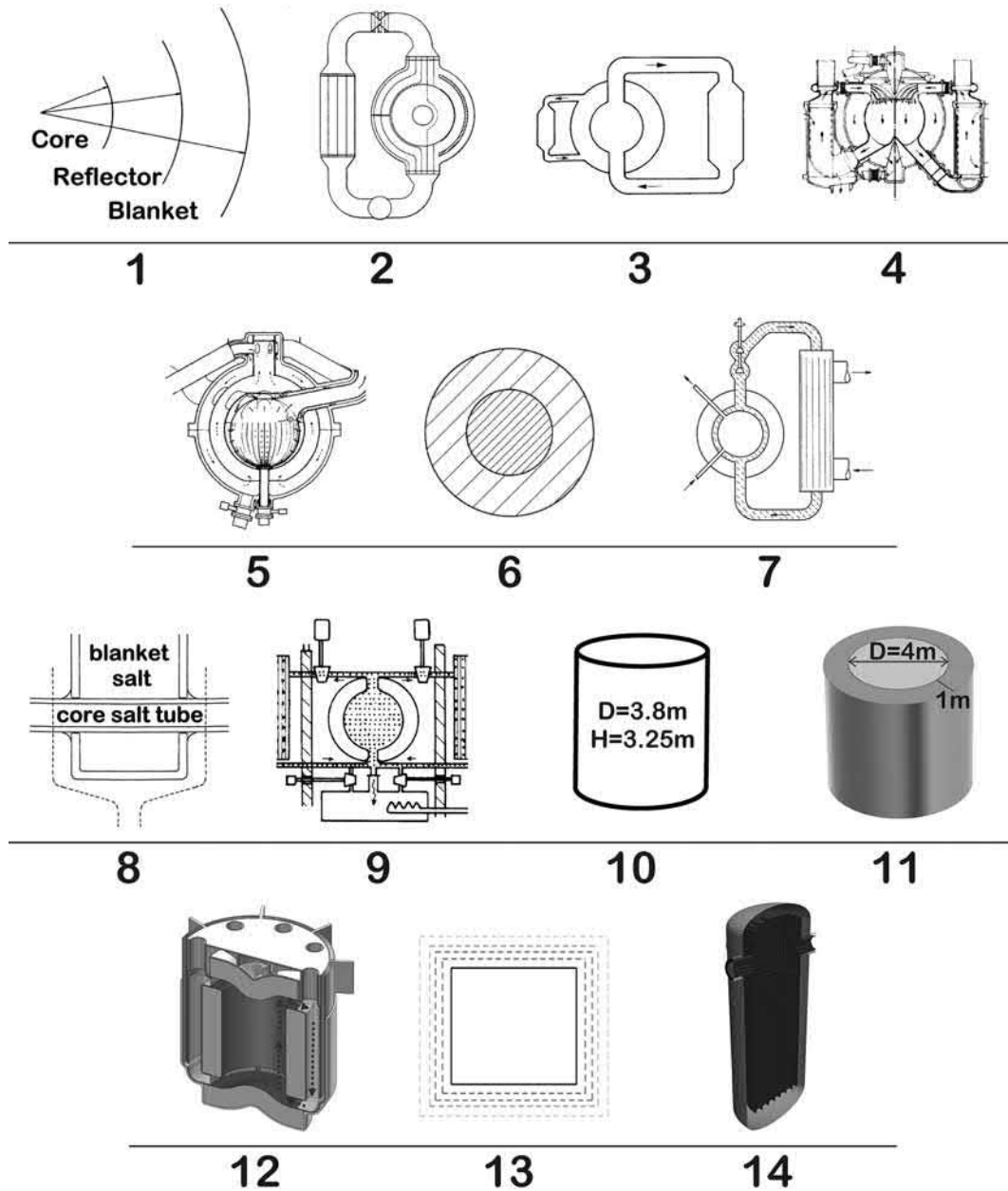


Fig. 10 Depiction of homogeneous MCFR concepts identified in **Table 3**. From (1) Goodman C, Greenstadt JL, Kiehn RM, Klein A, Mills MM, and Tralli N (1952) *Nuclear Problems of Non-aqueous Fluid Fuel Reactors*. Massachusetts Institute of Technology, MIT5000; (2) Bulmer JJ, Gift EH, Roll RJ, Jacobs AM, Jaye S, Kassman E, McVean RL, Oehl RG, and Rossi RA (1956) *Fused Salt Fast Breeder*. Oak Ridge, TN: Oak Ridge School of Reactor Technology, CF-56-8-204; (3) Nelson PA, et al. (1967) Fuel properties and nuclear performance of fast reactors fueled with molten chlorides. Argonne, IL: Argonne National Laboratory; (4) Smith J, et al. (1974) *An Assessment of a 2500 MWe Molten Chloride salt Fast Reactor*. United Kingdom Atomic Energy Authority, AEEW-R956; (5) Smith J, et al. (1974) *An Assessment of a 2500 MWe Molten Chloride salt Fast Reactor*. United Kingdom Atomic Energy Authority, AEEW-R956; (6) Taube M (1974b) *Thorium-Uranium Fast/Thermal Breeding System with Molten Salt Fuel*, Report EIR-253. Wurenlingen, Schweiz: Eidg. Institut für Reaktorforschung; (7) Taube M, Ligou J, and Bucher KH (1975a) *The Transmutation of Fission Products (Cs-137, Sr-90) in a Liquid Fuelled Fast Fission Reactor With Thermal column*, Report EIR-270. Wurenlingen, Schweiz: Eidg. Institut für Reaktorforschung; (8) Ottewitte E (1982) *Configuration of a Molten Chloride Fast Reactor on a Thorium Fuel Cycle to Current Nuclear Fuel Cycle Concerns*. Thesis, University of California, Los Angeles; (9) Taube M and Heer W (1980) *Reactor With Very Low Fission Product Inventory*, Report EIR-411. Wurenlingen, Schweiz: Eidg. Institut für Reaktorforschung; (10) Mourougov A and Bokov PM (2006) Potentialities of the fast spectrum molten salt reactor concept: REBUS-3700. *Energy Conversion and Management* 47: 2761–2771; (11) Raffuzzi V and Krepel J (2020) Simulation of breed and burn fuel cycle operation of Molten Salt Reactor in batch-wise refueling mode. In: *PHYSOR 2020*, Cambridge, United Kingdom, 29 March–2 April, 2020; (12) De Oliveira RG and Hombourger BA (2020) Fuel tap: A simplified breed and burn MSR. In: *PHYSOR 2020*, Cambridge, United Kingdom, 29 March–2 April 2020.

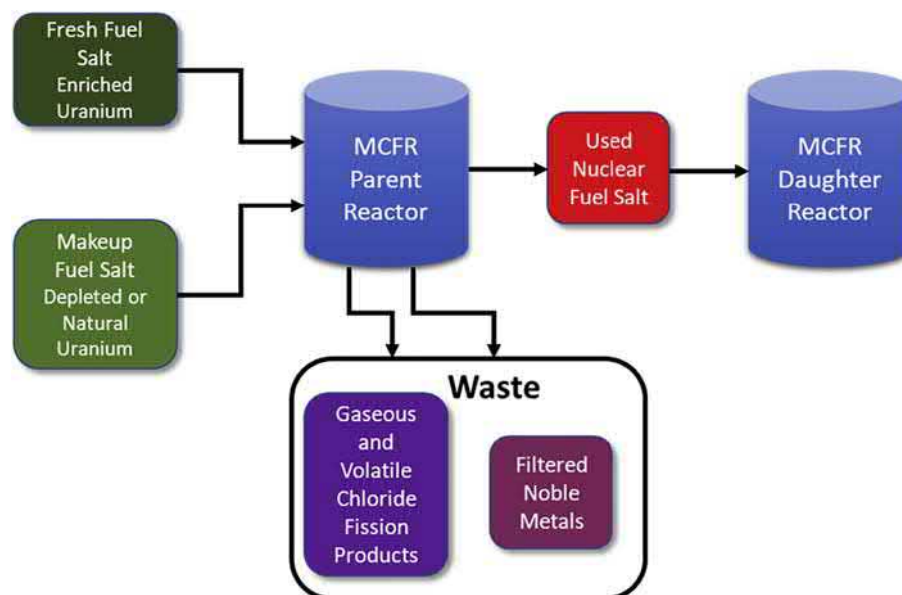


Fig. 11 Graphical depiction of the proposed breed-and-burn fuel cycle with irradiated fuel from an MCFR stored for future use in daughter plants (without reprocessing) or sent to waste for disposal.

Table 4 Summary of operational and fuel cycle characteristics of LWRs, MCFRs and other advanced reactors.

Characteristic	LWRs	SFRs	HTGRs	Other MSRs	MCFR
Online refueling	No	No	Possible with pebbles	Yes	Yes
Fuel shuffling avoided	No	No	No	Yes	Yes
Fuel processing avoided	Yes	No (except TWR [®])	Yes	Online	Yes
Breed-and-burn variant	No	No (except TWR [®])	No	No	Yes

Rather than going to disposal, used MCFR fuel can be collected and stored in a critically safe geometry until a sufficient amount is available to start a new reactor. As this fuel was accumulated from an operating reactor, it should contain sufficient fissile material to support initial critical operations of a second generation reactor. These daughter reactors utilizing this fuel salt will be able to start up without any new enriched feed fuel (see Fig. 11). By transferring used fuel, in total, to a daughter plant, this MCFR significantly reduces the use of actinides and defers the need to dispose of the vast majority of radioactive waste until the end of fleet build-out. For ultimate disposal of actinide-bearing fuel salt, prior work found that the salt itself could be packaged, without chemical conversion, into a suitable waste form (Toth et al., 1993). This and additional waste forms requires further research and development.

The open breed-and-burn fuel cycle supported by the breed-and-burn MCFR can result in lower volumes of waste than a conventional open fuel cycle due to higher potential burnup and lack of structural material being part of the waste.

During normal operations, it is likely that MCFR fuel salt will receive some cleanup or polishing including mechanical filtering of the noble metals and removal of noble gases to improve neutron economy. Polishing is defined as mechanical filtering and inert gas sparging to remove volatile fission products, while avoiding electrochemical separations. This fuel polishing is a feature of MSRs in general and can coexist with online refueling. The MCFR can similarly avoid fuel shuffling to support breed and burn as the fuel salt is continuously mixed. For a comparison of these features for different reactor types, refer to Table 4. A thermal spectrum MSR has substantial in-core surface area for gas accumulation. This potential for gas accumulation coupled with a large thermal neutron capture XS of xenon requires aggressive fission gas removal to maintain system criticality. In contrast, the lack of a moderator and the lack of in-core structure in an MCFR greatly limits surface area for gas accumulation. Similarly, the fast neutron spectrum limits the reactivity worth of these gases due to lower capture XS at higher neutron energies. These two combined factors allow for less aggressive gas removal. These polishing operations have the potential to produce a stream of nearly pure fission products. Since this stream will contain only trace quantities of actinides, it is possible to consider processing this material and isolating the long-lived fission products for simplified handling of the resulting waste. Fluoride fuel salt waste forms have been discussed in the literature (Toth et al., 1993); these may be applicable to chloride fuel salts and fission products as well.

Operationally and economically, the MCFR could perform well due to inherent simplicity. The low pressure and high boiling point of the fuel salt avoids the need for thick-walled, expensive forgings and reactor components. The high solubility of actinides in the salt and high volumetric heat capacity of the salt enables a high core power density, thereby potentially reducing capital

requirements. Molten salt fueled reactors can, in general, support online refueling due to the liquid fuel form. With online refueling, plant availability can be enhanced driving overall economics in a positive direction. The avoidance of online reprocessing also arguably increases proliferation resistance, reduces capital costs, and increases reliability (with a reduced requirement for complex processing trains) and availability (by reducing outage requirements). Additionally, the MCFR like other MSR designs requires relatively inexpensive fuel synthesis rather than complex high tolerance fuel fabrication of hundreds of kilometers of fuel pins, cladding, and coatings. MSR's can support high temperature operation because the fuel is already molten and a large margin exists at operating temperatures relative to the fuel salt boiling point. Existing MCFR designs typically employ a core outlet temperatures $> 700^{\circ}\text{C}$, but can be even higher with advanced high-temperature materials like refractory alloys. This high temperature operation can provide a high thermal efficiency ($> 42\%$) when coupled to a commercially-available steam cycle equipment. All of these factors combine to provide the potential for simplified operations and maintenance and improved economics.

Table 5 provides a summary of these and other safety features that together distinguish the MCFR from present LWR technology and other proposed non-LWR advanced reactors. In addition to providing the basis for defense in depth safety strategies, many of these features also contribute to operations and economic performance.

Technical description

The MCFR under development by TerraPower, LLC is of homogeneous type and uses a salt mixture composed of various sodium chloride and uranium chloride components. An example ternary phase diagram for this family of salts is shown in [Fig. 12 \(Cisneros et al., 2019\)](#). The blue shaded region shows the extent of a representative 500°C melting envelope. Multiple fuel salt compositions

Table 5 A summary of key safety characteristics of LWRs, the MCFR, and other advanced reactors.

Characteristic	LWRs	SFRs	HTGRs	Other MSRs	MCFR
Chemical reactivity	Steam reacts with the cladding to produce H_2 gas	Steam/water exothermically react with liquid sodium	Graphite can burn in air (MIT, 2018)	If using graphite, graphite can burn in air	Limited/slow reaction of salt with O_2 , soluble in water
Nuclear reactivity	Excess reactivity to provide 18–24 month cycle	Slight excess reactivity to provide economic cycle	Excess reactivity to provide economic cycle	Near-zero excess reactivity	Near-zero excess reactivity
Pressure	High pressure	Low pressure	High pressure	Low pressure	Low pressure
Temperature coefficients	Overall negative	Overall negative	Overall negative	Strongly negative, overall negative if using graphite	Strongly negative
Decay heat removal	Gen 2 active, very limited; Gen 3+ limited but passive	Passive	Passive, cause significant power density limitation	Passive	Passive

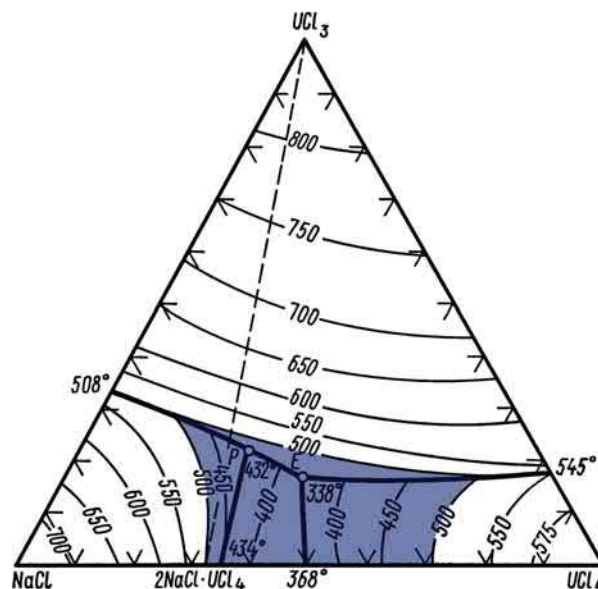


Fig. 12 The ternary phase diagram for UCl_3 - UCl_4 - NaCl in mole % ([Desyatnik et al., 1971](#)).

can be considered and should be capable of net breed-and-burn behavior with final composition being determined by a variety of factors including corrosion, solubility, viscosity, and reactor size.

Liquid fuels have an inherent advantage over solid fuels in that a solid fuel must first conduct heat to the outer surface of the fuel element, next conduct heat through the cladding (including past a physical gap or through a bond material), then convect the heat from the cladding surface to the primary coolant, and finally advect the heat out of the core. By comparison, a liquid fuel can directly transport the fuel salt/primary coolant out of the core and to the primary heat exchanger. Whereas there is up to 1000 °C difference between the maximum fuel temperature and coolant outlet temperature in the solid fuel reactor; in a homogeneous MCFR the highest fuel temperature is equal to the salt outlet temperature. Additionally, the liquid salts have volumetric heat capacities that are nearly twice that of liquid sodium at similar temperatures.

Another key advantage provided by a fuel salt is a strong and prompt negative temperature coefficient. Due to temperature dependent density changes, the MCFR core with hot fuel salt has a lower reactivity than core with cold salt. As a result, transients that result in increased heating are limited in severity by the expansion of the fuel salt. As the chloride salt is heated from 600 to 800 °C, its density drops significantly. This inherent feature provides a negative reactivity feedback that is much stronger than that provided by the Doppler effect.

Fig. 13 shows a view of a potential MCFR design. In addition to the vessel, the figure shows the flow path for the fuel salt (red dashed lines) and the neutron reflector (gray regions). In this concept, fuel salt flows up through the center of the core, flows past the upper axial reflector, and enters one of the pump assemblies. Upon exiting a pump assembly, the fuel salt immediately enters one of the primary heat exchangers (PHX).

Materials survival is an ever-present challenge to technology readiness for molten salt reactor concepts. In response, MCFR candidate materials are being tested early in non-nuclear environments such as flow loops at TerraPower, LLC and the MCFR can be designed for modular replacement of key components. Molten salt corrosion dictates the use of advanced materials, such as refractory alloys, for fuel salt-facing components, such as reflectors, primary heat exchangers (PHX) and the vessel. However, components only need to be clad or coated using these advanced materials, while the bulk of such component structural material can be constructed from more traditional materials. Radiation damage is also a factor for structural material survivability and component replacement intervals will be defined by those radiation damage limits like a solid fueled reactor. However, in a molten salt reactor, that radiation damage lifetime will likely be set by the first structural wall nearest to the high radiation flux center of the core as opposed to fuel or cladding limits in solid fueled reactors.

The MCFR reactor vessel and reactor head can utilize relatively thin plate due to the low normal and low design basis pressures.

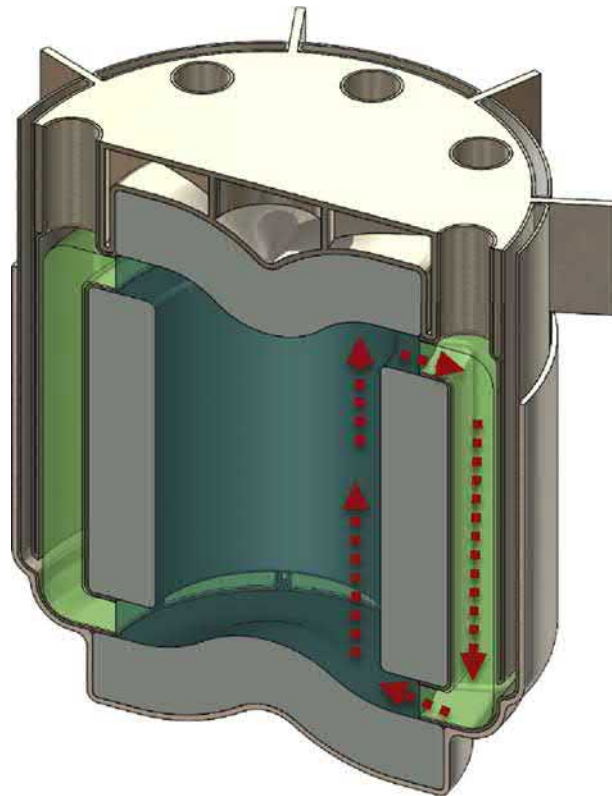


Fig. 13 The MCFR modular design for the reflectors and primary heat exchangers.

Operational, safety and public acceptance features

The MCFR design lends itself to simplified operations and maintenance. The fact that the fuel salt is non-chemically reactive and at low pressure leads to significantly simpler operations and maintenance. Similarly, the fact that the fuel salt is molten ensures “continuous shuffling” or mixing of the fuel and no mechanical fuel shuffling system is required. This has the potential to significantly improve plant availability and minimize downtime. Maintenance of some systems can be performed online, while others will require draining of the fuel to storage tanks. The drained reactor core will consist of modular components that can be directly accessed and removed via remote handling once the vessel head is removed. Internal core components, vessel and head, and support structures can be inspected as well.

Decay heat removal can be accomplished by natural convection through one or more primary heat exchangers, out through a naturally-circulating intermediate loop, and passively dumped to air heat exchangers. Typically, this is accomplished in concert with reactivity shutdown systems as the negative temperature reactivity coefficient of the fuel salt will lead to increased reactivity as the decay heat system eventually lowers the fuel salt temperature. Multiple options exist for reactivity control including fuel displacement, control rods, or control drums. This simplified approach reduces the number of additional systems for passive decay heat removal under anticipated operational occurrences and design basis accidents. This can all be accomplished entirely via natural convection and without any site power.

As part of the TerraPower approach to include multiple fission product barriers, MCFR's can readily employ a Functional Containment System (FCS) as proposed in INL/EXT-14-31179 (Idaho National Laboratory, 2014). Per this definition, a FCS will serve as “a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, anticipated operational occurrences, and accident conditions. Functional containment is relied upon to ensure that dose at the site boundary as a consequence of postulated accidents meets regulatory limits.”

In an MCFR, a developer could consider a FCS to be comprised of the Fuel Barrier, the Containment Structure, and the Reactor Building. The Fuel Barrier is the first barrier against uncontrolled release of radionuclides. It is comprised of corrosion resistant cladding on reactor vessel plate, piping, primary heat exchanger tubing and tube sheets and is designed to withstand design basis pressures.

The containment structure is likely a high temperature alloy lined reinforced concrete building structure and includes equipment and personnel hatches, process piping penetrations through the building with isolation valves, and expansion joints to allow differential movement between penetrating process piping and the building structure. During normal operation and maintenance, the containment structure would be kept at a slightly negative pressure, and during operation it is back filled with an inert gas. The containment structure will be designed and constructed to the applicable portions of ASME Boiler and Pressure Vessel Code to assure that it reliably performs its safety related functions.

If the fuel barrier were to leak, the containment structure is designed as the second barrier to uncontrolled release of radionuclides. The containment structure includes passive over pressure protection and when actuated, vents through filters to the reactor building, which has sufficient passive cooling.

The reactor building is likely the third barrier against uncontrolled release of radionuclides. The reactor building will have a ventilation system which circulates and monitors the atmosphere and is equipped with a filtration system to process normal effluents as well as limit the release of radionuclides during all anticipated operational occurrences and design basis accidents.

The combination of the multiple barriers would comprise the FCS and will be designed to limit radiation exposure to operational personnel and the public to within the applicable regulatory limits during operation as well as during and after postulated design basis accidents and to maintain any such exposures as low as reasonable achievable (ALARA), in accordance with national and international standards.

The FCS provides radiation shielding for the reactor core and associated equipment and provides protection from internal and external missiles to assure that the FCS can reliably perform its designed functions.

In addition, the containment and reactor building can be sized such that over pressurization cannot occur. Due to the fact that the molten salt fuel is non-reactive (even at high temperatures) and the reactor is normally operated at low pressures, thereby preventing a number of the driving forces by which a source term can be mobilized.

The MCFR employs multiple passive features that enhance safety. Modular components do have to be periodically replaced via remotely handled equipment. These remote handling systems can be continuously monitored to enhance safeguards. Similarly, the infrequent nature of reactor core component replacement yields infrequent and very controlled entering of containment. In the breed-and-burn MCFR design, fuel is moved to safe storage when any maintenance is performed, and these storage systems can be incorporated into material accountability systems.

Safeguards and non-proliferation

In contrast to other reactor designs, MCFR fuel design makes use of a high-actinide content chloride salt. Additionally, the MCFR performance improves significantly with chlorine that is enriched to at least 75% ^{37}Cl . The use of enriched chlorine reduces both neutron parasitic absorption and production of ^{36}Cl , which is a long-lived activation product. Chlorine enrichment is not available today from commercial sources, but either centrifugal or ion exchange column (IXC) methods of enrichment appear viable and extensible to the required quantities (Holcomb et al., 2011).

With its potential avoidance of reprocessing and ability to operate without fissile feed material after initial start-up, the MCFR offers an improved proliferation resistance among MSRs. Additional non-proliferation advantages are realized due to the near-zero excess reactivity. Removal of fuel salt beyond that required for online refueling would be a detectable event as the fuel salt contains multiple fission products producing significant dose. Similarly, removal of fuel salt beyond the designed volume would result in the reactor entering a subcritical configuration. The reactor could not start back up without addition of more enriched fuel. As a result, the MCFR does not require pyroprocessing, electrochemistry, or other separative online reprocessing technologies that raise potential proliferation concerns.

Used MCFR fuel salt will contain plutonium chloride and other transuranics, but these are fully mixed with uranium and chemically similar lanthanide chlorides (from fission products). Early material attractiveness analyses utilizing a Figure of Merit approach, which was originally developed by Bathke et al. (2012) for other fuel cycles, have been applied to the MCFR. These analyses indicate that MCFR used nuclear fuel could be less attractive than that from a fluoride molten salt from which lanthanides have been separated. A summary of some key proliferation characteristics are outlined in Table 6.

Economic competitiveness

The MCFR concept has the potential to simplify some operations and maintenance, potentially improving economic competitiveness. The fact that the fuel salt is non-reactive, transparent and at low pressure leads to significantly simpler operations and maintenance and lower operating costs. However, those operations and maintenance must be performed using remote handling, which must be further developed. Maintenance of many systems can be performed online, while others require draining of the fuel to storage tanks. Internal core components, vessel and head, and support structures can be inspected or replaced after fuel is drained to storage tanks. The steam generators can be located outside of primary containment to allow for maintenance without full shut-down of the core. With the exception of remote handling, these operational aspects of an MCFR have the potential to positively impact overall plant economics.

Typical MCFR waste streams are unlike conventional light water reactor waste streams. It has two components, one is the actual discharged fuel with all the actinides, the second are the separated fission products. The most prominent aspect is that these fission products contain only trace actinides. The MCFR fission product waste can be separated from unburned uranium or other actinides. This waste is proposed to be captured through mechanical filtering and gas sparging. Similarly, noble fission product gasses can be captured and allowed to decay in holding tanks. The filters containing the insoluble and longer lived fission products form a portion of the waste stream. This waste also has the benefit of minimizing criticality concerns as there are few fissile isotopes separated from the fuel salt.

Reactor core components and structures that are periodically replaced, assuming a modular design, will need to be cleaned of fuel salt and mechanically packaged for waste disposal. These components become activated from their presence in the reactor core. However, the chlorine (Cl) in the MCFR salt does produce a concern if it is placed in a geologic repository, activated ^{36}Cl is a radionuclide of concern in the performance assessment of such facilities (Croff and Krahn, 2015); this concern is mitigated, to a degree, by using Cl that is enriched in ^{37}Cl , but not eliminated.

Summary and path forward

MCFRs belong to the family of liquid-fuel molten salt reactors and utilize chloride salts containing actinides as a liquid fuel. MCFRs can be generally subdivided into homogeneous and heterogeneous concepts, based on the core structure. The neutronic properties of chlorine and especially of its isotope ^{37}Cl provide a very hard neutron spectrum and very low parasitic neutron absorption. The resulting very high neutron economy can be used for new fuel breeding or legacy waste burning. At the same time, the low scattering and capture XS has consequence in very high neutron leakage probability. Typical homogeneous MCFR must thus rely on efficient reflector and/or blanket and bulky core. It has also implication on possible doubling time for both open breed-and-burn and closed fuel cycles.

The homogeneous and heterogeneous categories are practically synonyms for internal and external cooled reactors. The advantage of external cooling is that the parasitic capture of structural materials and coolant in the core is minimal. At the same time, the

Table 6 A summary of proliferation characteristics of traditional MSRs and the MCFR.

Characteristic	Thermal molten salt reactors	Molten chloride fast reactor
Fissile material	^{233}U , ^{235}U and/or ^{239}Pu	^{235}U and/or ^{239}Pu
Operating mode	Neutron moderated, thermal spectrum	Unmoderated, fast spectrum
Fission product poisoning	Significant	Limited
Online processing	Yes, including fissile separations	Limited, no fissile separations
Ongoing fissile feed	Yes	No, system is fed depleted or natural uranium after start-up
Supports eventual elimination of enrichment	Not in U-Pu cycle	Yes
Material attractiveness	Varies	Reduced due to presence of lanthanides, which are chemically similar to Pu

fuel salt acts as a coolant and must be pumped to the external heat exchanger. This layout increases fissile fuel inventory and results in delayed neutron loss in the primary circuit. The advantage of internal cooling is that the salt does not need to be extensively pumped out of the core. The presence of coolant and separating structural materials in the core, however, deteriorate the neutron economy. Many of these concepts rely on metallic tubes with very thin walls, profiting from the low pressure difference, or on SiC tubes, because of its low parasitic capture.

The very hard neutron spectrum and the related excellent neutron economy of MCFR enables strong breeding in the U-Pu cycle or even the operation in breed-and-burn cycle. In this case, discharged fuel will be replaced exclusively by fertile feed, typically depleted or natural uranium, reducing or eliminating the need for continued enrichment. For concepts with a single fuel salt, or actually single fissile zone, the discharged fuel represents an average fuel composition. It can be used, once a sufficient amount is collected, to start another breed-and-burn reactor. This option can substantially postpone the final decision about recycling or disposal of the spent fuel. Breed-and-burn MCFR might achieve a very high uranium utilization of $\sim 35\%$. This is ~ 60 times the utilization of natural uranium in contemporary light water reactors; both operating in a once-through fuel cycle.

MCFR concepts have been developed since the 1950s and continue to draw interest due to key technology differences and advantages relative to solid fueled reactors. However, many of the key technologies are at low technical readiness and require further research and development. Specific areas of focused research and development include molten salt synthesis, chemistry and corrosion, accommodation of the irradiation damage to the core vessel, source term development, irradiation testing, reactor licensing, nuclear safeguards and nonproliferation, instrumentation and controls, and component vendor development. Applied research in the areas of molten salt chemistry and source term quantification are required. Additional structural material development and irradiation testing to support operational and economic requirements are also underway. Lastly, remote handling technologies in support of continuous operations remain an area for future development.

Active development in these areas along with further reactor design and modeling is underway by numerous international partners across industry, government, and academic research labs. Government agencies, the International Atomic Energy Agency, non-governmental organizations, and other research agencies maintain active forums and working groups with industry vendors to facilitate the advancement of MSRs. Although much work remains in specific areas, the promise of MCFR technology remains a strong driver to advance to the technology to support worldwide deployment.

See Also: Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors; Traveling Wave Reactor (TWR[®]).

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Traveling Wave Reactor (TWR[®])

Pavel Hejzlar, TerraPower, LLC, Bellevue, WA, United States

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Glossary

ATR advanced test reactor.

BDBA beyond design basis accident.

BOEC/EOEC beginning/end of equilibrium cycle.

BOL/EOL beginning/end of life.

Breeding Converting a fertile isotope into a fissile one by capturing a neutron. Usually the fertile isotope ^{238}U to the fissile isotope ^{239}Pu . Also the fertile isotope ^{232}Th (natural thorium) to the fissile isotope ^{233}U .

Burnup A measure of the amount of fission energy released from a given quantity of fuel. One unit used for such a measure is FIMA = Fraction (usually given in %) of Initial Heavy Metal Atoms that has fissioned. A more wide-spread unit is MWd kg^{-1} ($= \text{GWd ton}^{-1}$) that measures the amount of thermal energy released from unit weight of the initially loaded HM.

Cladding The structural material that separates the nuclear fuel from the coolant. In fast reactors it is usually made of steel. In light water reactors it is usually made of an alloy of zirconium.

DBA design basis accidents.

EPZ emergency planning zone.

FCCI fuel-cladding chemical interaction.

FFTF fast flux test facility.

IES integrated energy system.

INL Idaho National Laboratory.

IVS in-vessel storage.

LCOE levelized cost of electricity.

LWR light water reactor.

Neutron flux Number of neutrons crossing a unit area per second. Reaction rates, such as the fission rate, are proportional to the neutron flux.

PIE post-irradiation examination.

PPS plant protection system.

SFR sodium-cooled fast reactor (Heidet, 2021).

STC Sodium temperature coefficient; the effect on the effective multiplication factor (k_{eff} ; the ratio between neutron production rate and neutron loss rate) due to a change in the sodium temperature.

Subcritical/critical core A core that cannot/can sustain a fission chain-reaction.

TREAT transient reactor test facility.

TRL technology readiness levels.

TWR[®] traveling wave reactor.

UTOP unprotected transient over power.

TWR definition and key advantages of TWR technology

The traveling wave reactor is a fast spectrum reactor that operates in a once-through fuel cycle and uses conversion of fertile material into fissile material and subsequently burns bred-in fissile material in situ, thus enabling substantial reduction of the need for enrichment and higher utilization of uranium resources. It is a breed-and-burn reactor whose operation includes the periodic loading and discharge of fuel capable of sustaining an equilibrium state of critical operation with fuel reloads that without in situ breeding would render that specific fast spectrum core subcritical. Thus, the reload fuel, which if fully loaded into the core would render the core subcritical, is bred in the core through neutron captures on fertile fuel isotopes and converted into fissile isotopes that are then fissioned (burned) in situ, maintain criticality and produce power. More detailed discussion on breed-and-burn fast reactors and their background is discussed in [Qvist \(2021\)](#) and [Hejzlar et al. \(2013\)](#). The concept operates on the basis of so called breed-burn waves and their relative movements versus fuel, where either the fuel or the waves may move relative to stationary observer. The concept with stationary fuel and moving breed-burn waves is described in CANDLES concept by [Sekimoto \(2021\)](#). The version with the stationary wave that keeps nuclear reactions fixed in one place and moves the fuel into the wave was selected by TerraPower, LLC (TerraPower®) as the most practical embodiment for development and will be described in this Chapter.

The selection of TWR concept and origins of TerraPower, LLC have their beginnings in the 2006 deliberations between Bill Gates, Nathan Myhrvold and Lowell Wood on how to provide sustainable, scalable low carbon energy for the world. A large number of reactor types, both existing and new, were considered with respect to improvements in areas of safety, waste, efficiency, economics, weapons-proliferation resistance and terrorist-risk reduction. During these assessments, the TWR concept, advocated at that time by Lowell Wood, was identified as offering improvements in all these areas. Many possible fuel and coolant options were considered and explored in order to evaluate which option was the most promising and practical embodiment. Sodium cooled Fast Reactor (SFR) using metal uranium fuel was identified as the most promising TWR concept. This is because sodium coolant offers excellent cooling capabilities enabling high power densities and metal fuel has high uranium density and hard neutron spectrum that is important for effective breed-and-burn wave process. In addition, sodium coolant technology is the most mature among other fast reactor coolant candidates due to appreciable operating experience in experimental, prototype, demonstration, and commercial SFR units worldwide and is compatible with early deployment time scales.

TWR cores are distinguished from other fast reactors in that they can operate in equilibrium with subcritical reload fuels, i.e., fuels that if fully loading these cores would make the fast neutron spectrum cores subcritical. These fuels include medium assay low enriched uranium (enrichment less than 10%), low enriched uranium (less than 5%), natural uranium, depleted uranium or spent fuel from light water reactors (LWRs). The ultimate TWR concept with most benefits is realized when TWR core operates with fuel reloads from depleted uranium. This means that after reactor startup period, the only fuel that is required is depleted uranium and there is no need for enrichment and reprocessing facilities to sustain operation of the first reactor and its successors. As a result, fuel cycle is significantly simplified compared to the current LWR fuel cycle, as shown in [Fig. 1](#).

Such simplification of fuel cycle has a number of benefits:

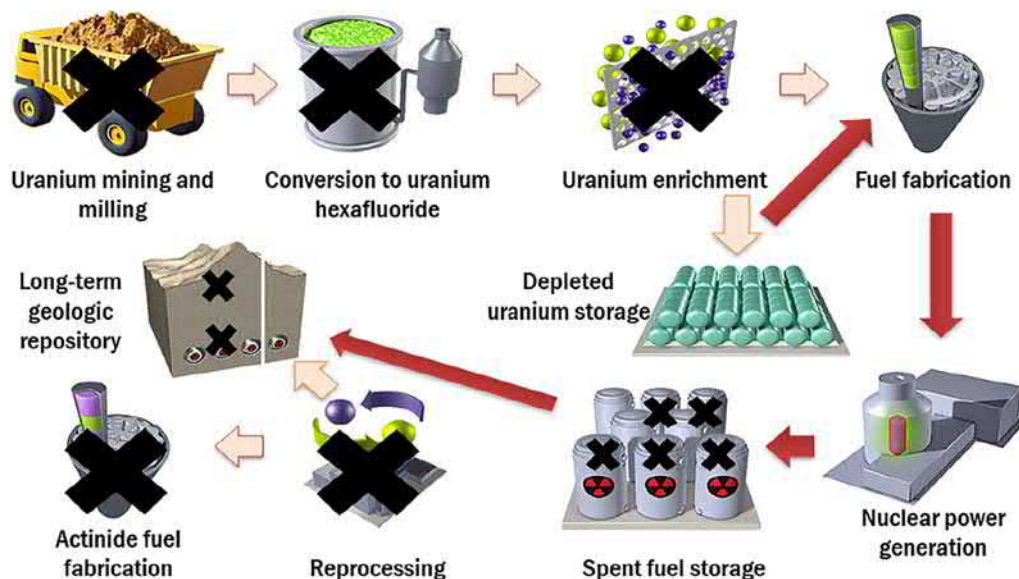


Fig. 1 Simplified TWR fuel cycle versus LWR fuel cycle. Black crosses indicate eliminated step of LWR cycle or reduction of waste, such as less spent fuel storage and less waste to repository. TWR fuel cycle minimizes the need for mined natural uranium, enrichment services, temporary fuel storage, fuel reprocessing and final waste disposal.

- Significantly reduced mining of natural uranium resources. TWR reactors operating on depleted or natural uranium use up to 30 times less natural uranium than LWRs. The reasons for such a large improvement include the elimination of enrichment for fuel reloads and much more energy extracted from TWR fuel assemblies than in LWRs.
- Substantially reduced number of spent fuel assemblies and waste volume; by about 80% less than for LWRs per unit energy generated. This is primarily due to significantly larger energy extraction per unit fuel mass due to high burnups of TWR fuel and also because of higher TWR fuel loading per unit volume and higher plant efficiency than for LWR. Consequently, the requirements for temporary spent fuel storage and for waste repository are correspondingly reduced. Furthermore, using depleted uranium rather than enriched uranium, TWR plants consume existing tailings from the enrichment process that would otherwise add to waste obligations.
- Large reduction of enrichment services. Only the first core requires enriched fuel, all reloads and successor plants¹ do not require enriched fuel reloads.
- No need for reprocessing. TWR plants avoid the cost of building and operating reprocessing plants as well as the related waste cleanup costs, while still supporting large increase of utilization of natural uranium compatible with worldwide scale deployment (Petroski and Wood, 2012).
- High proliferation resistance. By eliminating the need for reprocessing and reducing enrichment needs, TWR technology bypasses two main proliferation concerns. Very high burnup of TWR fuel renders plutonium isotopic composition of spent TWR fuel unattractive for proliferators. In addition, long TWR fuel residence time and infrequent refueling operations—about once in 10 years, restrict access ability to fuel.
- Lower fuel cycle cost. Simplified fuel cycle reduces the cost of fuel cycle due to reduced enrichment and reduced volume of spent fuel and associated repository and waste management costs.

In addition to the benefits from TWR fuel cycle, TWR plant also exhibits safety benefits of pool-type SFRs with metal fuel. These include:

- Elimination of loss of coolant accident due to low pressure sodium coolant, no large pipes with primary coolant outside the reactor vessel, and the use of a guard vessel surrounding the reactor vessel with no penetrations below the coolant surface level.
- Long times available before fuel and coolant temperatures reach their limits due to large thermal inertia of the sodium pool in the vessel.
- No need for electricity supply to drive emergency cooling systems due to passive decay heat removal systems that remove decay heat to air by natural circulation indefinitely.
- Inherent reactor transition to a safe state in beyond design basis accidents with failure to shut down. This is because in case that control rods and even standby shutdown rods fail to drop into the core to stop the fission reaction, reactivity coefficients of metal fueled TWR core result in such reactor response that significant damage to fuel is prevented.
- Compatibility of metal fuel with sodium coolant that eliminates fuel/coolant reaction and progression of fuel damage even in case of cladding breach. In addition, dispersal characteristics of metal fuel in severe accidents lead to reactivity reduction even in case of fuel melting.
- Iodine and cesium fission products that drive dose rates stay within metal fuel and the sodium pool, so their release in severe accidents is thus very limited (Grabaskas et al., 2016). This feature of the TWR reactor together with other safety benefits listed above result in a much lower source term and radiological risk and thus in a greatly reduced emergency planning zone (EPZ). TWR plants are expected to be able to achieve having a small EPZ, possibly within the plant site boundary. Eliminating the need for evacuations beyond plant site reduces cost of emergency planning and allows more flexible plant siting.

Overall, TWR technology can lead to a sustainable, scalable reactor infrastructure that in its ultimate mode of operation requires only depleted or natural uranium, tremendously enhances energy security, reduces fuel costs, and exhibits good proliferation resistance without need for reprocessing facilities. The very significant benefits offered by the TWR system provide strong incentives for the expeditious development and deployment of the TWR technology.

Key challenges for TWR technology development and mitigation strategies

The concept of breed-and-burn reactors is not new, first being proposed in the late 1950s by Feinberg (1958). The review of various studies of breed-and-burn reactors since that time is provided by Qvist (2021). However, these early studies focused primarily on neutronic analyses of the wave propagation without looking closely at other aspects of the reactor design, such as fuel performance, thermal hydraulic and mechanical design limitations. Considering all engineering and fuel design limits in a systematic manner to develop practical embodiment of a functional reactor brings challenges that need to be overcome before TWR plants can be successfully demonstrated and deployed. This section will discuss key challenges of the TWR core design and solutions including innovative approaches developed by TerraPower, LLC, to resolve these challenges.

Any reactor design has to meet many limits, but for the purpose of this discussion, we will focus only on the key limits impacting the feasibility of a breed-and-burn reactor. Fig. 2 shows these limits and main design levers that can be used to affect these limits.

¹Successor TWR plant is a new TWR unit that uses as its initial core the fuel discharged at end of life from its predecessor plant.

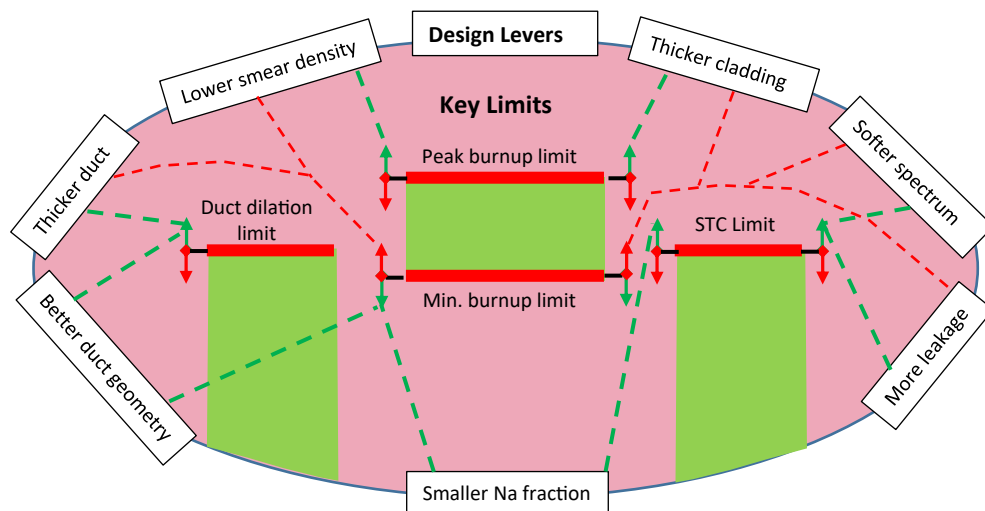


Fig. 2 Key TWR core design limits and design levers to impact the limits. Green fields indicate desirable regions for key design parameters that need to be kept within design limits designated by thick red lines. Impact of design levers on the limits is depicted by dashed lines where green color indicates benefit, i.e., increasing the desirable green field and red color indicates negative benefit. For example, softening the neutron spectrum reduces sodium temperature coefficient (STC), but increases minimum peak burnup limit.

High minimum peak fuel burnup

Fuel burnup, which is a key parameter for fuel performance, needs to lie in the green field above the minimum peak burnup and below the thermo-mechanical peak burnup limit, which is determined by cladding strain from irradiation or thermal creep. The former limit is unique to breed-and-burn reactors as the reload uranium assemblies with very low fissile content need to first breed enough fissile material to sustain the chain reaction, which requires certain minimum burnup. The lower the enrichment of reload fuel, the higher the minimum peak burnup limit. For the most attractive TWR embodiment running on depleted uranium, minimum peak burnup limit is about 30%.² The peak allowable burnup limit can be increased by increasing cladding thickness. However, larger cladding thickness results in larger neutron parasitic losses and leads to a higher required minimum burnup. Typically, the required minimum burnup is more sensitive than the peak burnup limit is to variations in cladding thickness, and hence a reduced design window may result from increasing thickness. A more effective design lever to increase the peak burnup limit is reduced smear density.³ Less fuel per pin provides more space for swelling caused by fission products accumulation reducing cladding stress. However, less fuel leads to higher minimum required burnup. Moreover, for metal fuel with sodium bond⁴ more sodium in the pin yields a more positive sodium temperature coefficient (STC).

To meet the requirement of high minimum burnup while still meeting fuel performance limits, TerraPower, LLC is developing fuel pins⁵ with smaller smear density than the ~75% used in traditional SFR metal pins. In addition, other design levers shown in Fig. 2 that tend to increase minimum fuel burnup are optimized to keep the peak minimum burnup as small as possible. The ultimate TWR plant that can operate with depleted or natural uranium fuel will require venting gaseous fission products from the fuel pins to reduce pressure buildup and thus reduce stresses and cladding strain. This is accomplished through a pin vent allowing venting of gases in a controlled manner into sodium coolant and with features minimizing escape of cesium. The impact on plant design is a small amount of Cs137 released into the sodium coolant, which is removed with reticulated vitreous carbon traps, and an increased amount of fission gas that requires higher capacity of the cover gas processing system and storage of Kr85 in shielded vessels. Due to this continuous online removal of Cs137, the end result is a primary coolant activity similar to that of current LWRs.

Another consequence of high burnups necessary to achieve breed-and-burn operating mode on depleted uranium is a high neutron dose on cladding material. The highest dose that has been reached up to date in metal fueled test pins with ferritic martensitic steel cladding is ~200 dpa⁶ (Toloczko et al., 1994), while the TWR cladding requires doses of up to ~500 dpa. The higher dose increases cladding strain from void swelling and irradiation creep. The above strategies for addressing high peak burnup (using low smear density and fuel pin venting) also mitigate the consequences of high dose on irradiation creep-driven cladding strain since they eliminate the stress from internal gas pressure and reduce the creep strain from fuel cladding mechanical interaction. Fig. 3 shows an example of how pin venting can significantly increase burnup capability under total cladding strain limit in comparison with sealed pin.

²That is, at least 30% of the fed uranium fuel needs to be fissioned.

³The smear density is the fraction of the maximum nominal fuel density.

⁴The sodium bond is sodium that is inserted into the as-fabricated fuel rod to fill the initial radial gap between the fuel and the cladding for better heat transfer.

⁵Also referred to as “fuel rods.”

⁶Displacements per atom; a measure of the radiation damage induced by, primarily, energetic neutrons bombardment.

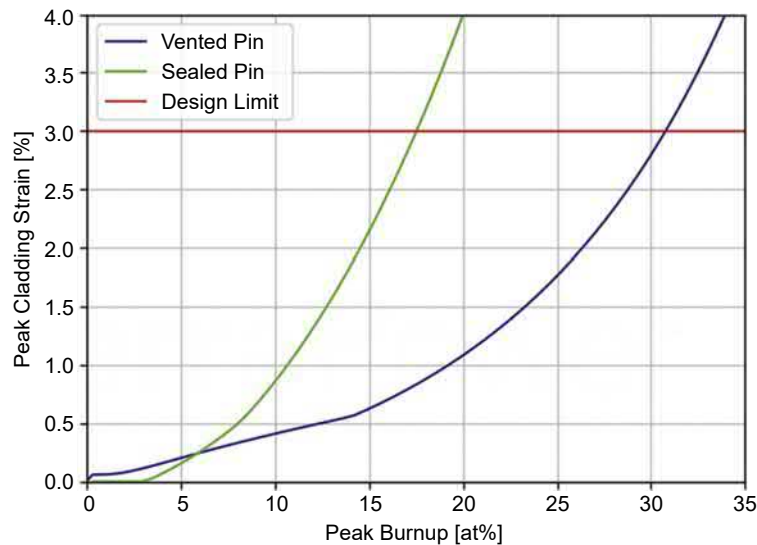


Fig. 3 Comparison of cladding strain for sealed and vented pins for irradiation history in TWR core. For the same cladding strain limit, achievable peak burnup can be significantly increased.

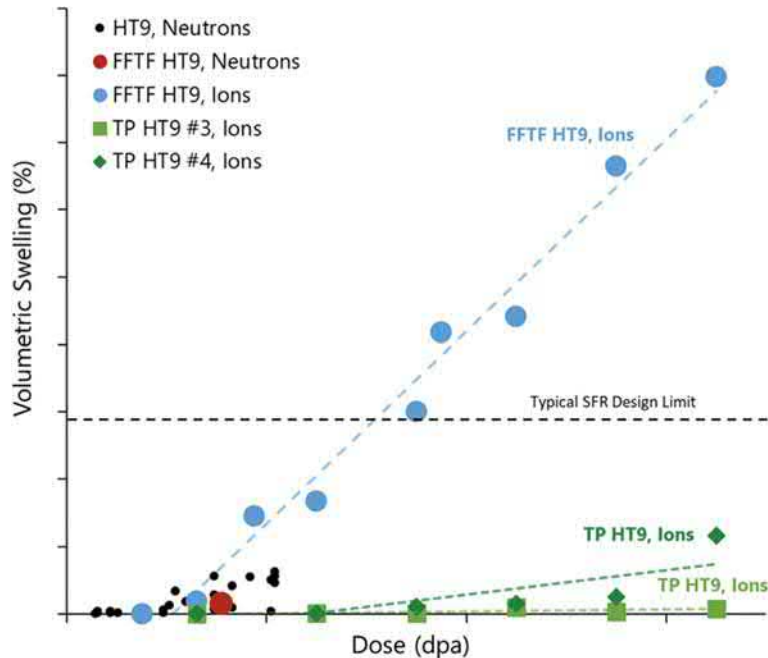


Fig. 4 Demonstration of irradiation performance improvement of TerraPower optimized HT9 (TP HT9) using ion irradiation with direct comparison against FFTF HT9 cladding. Black dots are data from neutron irradiation in FFTF reactor. Blue dots show ion irradiation of archived HT9 material from FFTF and squares and diamonds shows ion irradiation data for Heats 3 and 4 for optimized TP HT9.

To further reduce cladding strain and duct dilation,⁷ TerraPower, LLC is developing an advanced ferritic-martensitic steel HT9 with an optimized microstructure that is expected to exhibit very low void swelling rates and irradiation creep up to ~ 500 dpa. Fig. 4 shows that in direct comparison with archived HT9 used for FFTF (Fast Flux Test Facility) reactor, TerraPower optimized HT9 steel has substantially lower swelling.

⁷Duct dilation is bulging of hexagon duct walls in radial direction towards adjacent duct walls.

Duct dilation

Fuel pins are surrounded by ducts that direct the sodium flow through assemblies and are exposed to stresses from pressure differential⁸ and to irradiation creep causing the ducts to dilate with time. Similarly as for cladding, TWR ducts are exposed to high dose⁹ of high energy neutrons increasing void swelling and irradiation creep. As the duct dilates it closes the gap available between the ducts of adjacent assemblies. It is desirable that at the end of assembly life, the duct walls of adjacent ducts do not come into contact to avoid excessive withdrawal and insertion forces that would impair fuel shuffling. Duct dilation can be reduced by increasing the duct wall thickness. This design approach adds steel in the core and increases minimum fuel burnup, reducing the design window for fuel pins. Traditional fast reactors use thicker ducts and boost enrichment to overcome the penalty of parasitic losses, but this approach is not possible for the TWR cores where depleted uranium is feed.

To keep the duct dilation within prescribed limits innovative duct design that exhibits smaller dilation has been developed. The geometric features of the duct design and the optimized HT9 material with low swelling performance shown in Fig. 4 are key aspects of the design enabling smaller dilation.

Sodium temperature coefficient

The third important limit comes from the sodium temperature coefficient. This coefficient is typically positive for large SFR cores.¹⁰ However, its value needs to be kept small enough so that it can be balanced by other reactivity feedbacks¹¹ that are negative, such as Doppler coefficient, core radial expansion/bowing coefficient, fuel axial expansion coefficient and control rod driveline expansion coefficient, to yield overall net negative reactivity feedback. This is typically achieved in sodium cooled reactors by a large enough increase in the neutron leakage probability. This contradicts low leakage needed for TWR core to support breed-and-burn operation at reasonable minimum burnups. Another approach is the use of softer spectrum through introduction of moderators, but this leads to more parasitic absorptions and increased minimum burnup, reducing the burnup design space as indicated on Fig. 2. Therefore, minimizing the sodium fraction in the assembly is the main design lever remaining to keep the sodium temperature coefficient within acceptable bounds. This is accomplished through the following means:

1. the use of a tight pitch hexagonal lattice to keep the amount of coolant between the fuel pins as small as possible while keeping an acceptable pressure drop,
2. the minimization of the gap between the fuel assembly ducts while providing enough space for duct dilation and swelling to maintain acceptable assembly withdrawal forces, and
3. the design of a fuel pin without a sodium bond, replacing it with mechanical bond, as discussed below.

In addition, TerraPower, LLC is developing special passive reactivity devices that upon coolant temperature increase reduce reactivity by passive means to counterbalance positive reactivity insertion from sodium temperature feedback.

Feasibility of TWR core operating on depleted uranium

This section provides the physical description and layout of an example (not necessarily optimized) TWR core and its key performance parameters to assess if the operation on depleted or natural uranium in equilibrium is feasible using the design approaches outlined above. The example is shown for commercial units sized for 1200 MWe power rating.

Fig. 5 shows an example TWR core map with illustration of fuel shuffling scheme. The core initially contains a total of 864 fuel assemblies with 510 enriched fuel assemblies and 354 depleted uranium fuel assemblies. The core effective radius is 3.25 m and the active core height is 2 m. The TWR fuel assemblies have innovative features enabling achievement of high burn-up and dpa capabilities consistent with the requirements of TWR breed-and-burn operation on depleted uranium described in the above section. These assemblies have 217 wire wrapped fuel pins arranged in a tight triangular lattice surrounded by advanced ducts that can withstand neutron induced radiation damage up to ~500 dpa. The pins have a significantly lower effective smear density in the range of 50% to 60% versus 75% for typical SFR metallic fuel pins and are vented. The ducts, as well as pin cladding will be fabricated from optimized ferritic martensitic steel with high resistance to swelling to accommodate high neutron fluence, but other components of the assembly (e.g., end fittings and nozzles) will use stainless 316H. Key fuel assembly dimensions are listed in Table 1.

The key feature of the TWR core is its ability to sustain critical operation indefinitely while being refueled with depleted or natural uranium. To demonstrate this feature, the reactor must be able to operate in an equilibrium cycle using only depleted or natural uranium as feed fuel. In an indefinitely sustainable equilibrium cycle, the reactor is critical over the entire cycle, and it is possible to restore the Beginning of Equilibrium Cycle (BOEC) state from the End Of Equilibrium Cycle (EOEC) state by shuffling the fuel

⁸Between the coolant pressure inside the fuel assembly and the sodium filling the gap between the fuel assemblies.

⁹Also referred to as fluence = time integrated flux.

¹⁰A reduction in the sodium density results in a hardening of the core neutron spectrum that causes a positive reactivity feedback and an increase in the leakage probability of neutrons out from the core that has a negative reactivity feedback.

¹¹A reactivity feedback coefficient is the change in the core effective multiplication factor (See definition in Glossary) due to a one degree change in either the coolant (affecting core radial expansion and control rods driveline expansion) or the fuel (affecting Doppler effect and fuel axial expansion).

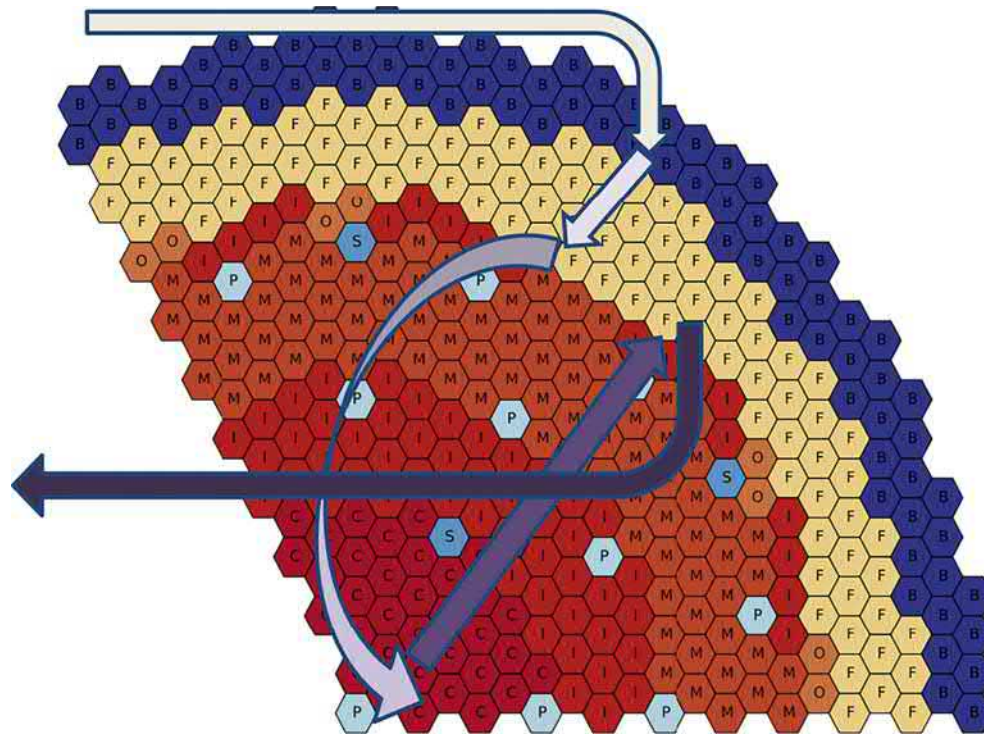


Fig. 5 Example of TWR core map with illustration of fuel management using convergent-divergent shuffling. Dark red colored C assemblies denote center driver fuel, assemblies designated by letter I designate inner driver fuel, M and O assemblies indicate middle and outer driver fuel, respectively, F assemblies are depleted uranium feeds, blue colored B assemblies designate reflectors, P are control rods and S are standby shutdown rods. The arrows indicate how the shuffling of assemblies is performed during shutdown.

Table 1 Key fuel assembly dimensions for example TWR core.

Parameter	Value (mm)
Clad outer diameter	10.2
Clad thickness	0.40
Assembly outer flat-to-flat	174.0
Duct thickness	4.00
Wire diameter	0.86

assemblies. Shuffling the fuel assemblies involves discharging the most highly burned fuel from the core, repositioning the assemblies left in the core, and introducing new fuel assemblies consisting of fresh fertile fuel (depleted or natural uranium).

Analyzing the equilibrium cycle state is important because it sets the minimum requirements on fuel burn-up and dpa needed for burning depleted uranium. Burn-up and dpa levels affect the distortion and swelling of both the fuel pins and assembly ducts and therefore need to be accommodated in the design. Because the equilibrium cycle consists of a large amount of bred plutonium and fission products, it is also the most challenging from the perspective of reactivity coefficients and transient response.

Equally important to the equilibrium cycle is the transition from a fresh enriched startup core to equilibrium state. This must be done in such a manner that the power produced in each assembly location remains nearly constant to avoid power-to-flow discrepancies at any point during the transition period. The primary challenge of designing the transition fuel management is minimizing the amount of enriched material required to reach equilibrium while maintaining a constant radial power distribution.

The initial fresh core layout is shown in Fig. 5. The uranium in the enriched assemblies vary between 5.5% and 13% U-235. The initial core layout, cycle length, and enrichment distributions were determined by iteratively matching the target equilibrium cycle power distribution, which in turn was determined as the optimal result of a systematic optimization sweep over a variety of fuel management options.

TWR Reactor's fuel management scheme using a convergent-divergent path through the core is shown on Fig. 5. The cycle length is 550 EFPDs. There is a feed stream of 30 natural uranium assemblies per cycle. Feed assemblies enter the core at the periphery, as indicated on the figure, and are gradually shuffled inward as fissile fuel is bred. Then they are moved to the center of the core and start moving away from the center until they reach the burn-up limit, when they are discharged from the core at around ring 15. The

discharged assemblies are stored in the In-Vessel Storage (IVS) for 5 cycles, hence the fresh feed fuel delivery into the reactor vessel and removal of burned assemblies from the vessel happen once every 10 years.

The primary parameters of interest are the peak burnup (for the fuel performance limits), k -effective (for neutronic constraints), the structural dose (for cladding and duct performance), and the power density profile (for cooling constraints).

The k -effective evolution of the TWR core over several selected equilibrium cycles is shown in Fig. 6. It remains between 1.006 and 1.021, with a typical burn-up reactivity swing¹² of 2% dk/k. The larger k_{eff} than 1.0 is used to provide margin to account for uncertainties. In an actual reactor, k_{eff} will be maintained at 1.0 all the time using control rods. During every cycle, the k_{eff} increases because the TWR core is breeding new plutonium. The saw-tooth drops of reactivity are associated with each shuffle operation, which are designed to bring k_{eff} down through addition of depleted uranium feed assemblies that behave as neutron absorbers.

Neutronic parameters and reactivity coefficients vary significantly between the Beginning of Life (BOL) and EOEC due to transition of primarily fissioning isotopes of ^{235}U at BOL to ^{239}Pu at EOEC with effective delayed neutron fraction changing from 0.0068 at BOL to 0.0035 at EOEC. Doppler coefficient is negative and relatively small, consistent with metal fuel cores, undergoing BOL to EOEC change from $-0.051 \text{ } ^\circ\text{K}^{-1}$ to $-0.092 \text{ } ^\circ\text{K}^{-1}$. Fuel axial expansion coefficient is also negative changing from $-0.04 \text{ } ^\circ\text{K}^{-1}$ to $-0.08 \text{ } ^\circ\text{K}^{-1}$ and negative radial expansion coefficient increases from $-0.062 \text{ } ^\circ\text{K}^{-1}$ to $-0.15 \text{ } ^\circ\text{K}^{-1}$ between BOL and EOEC. Sodium temperature coefficient is positive with small value of $0.032 \text{ } ^\circ\text{K}^{-1}$ at BOL increasing to $0.27 \text{ } ^\circ\text{K}^{-1}$ at EOEC, without considering the effect of engineered passive reactivity devices. Even with EOEC positive sodium temperature coefficient, the net temperature and power reactivity coefficients are negative.

Keeping the power shape relatively constant throughout each cycle and during the transition to equilibrium is required to match power to flow in each fuel assembly at all times. The BOL and EOEC radial power distributions are shown in Fig. 7. The low-burnup assemblies that jump into the center of the core are less reactive than the higher-burnup assemblies in the central rings, leading to an annular power distribution. There is very little power in the outer region of the core where the depleted uranium captures most of the neutrons leaking from the power-producing regions of the core while generating very little fissions.

The flow to each assembly is delivered through receptacles located in a high pressure plenum between a lower and upper grid plates. To maintain appropriate power to flow ratio at all times, each receptacle employs orifice plates designed to deliver appropriate flow rates. There are 20 orifice zones to distribute flow rates for matching power to flow in each region to maintain peak clad temperature below $625 \text{ } ^\circ\text{C}$ (a steady state operating temperature limit that accounts for uncertainties). Maximum bundle pressure drop is 0.5 MPa and the core provides $500 \text{ } ^\circ\text{C}$ core average outlet temperature.

Table 2 summarizes key TWR core parameters of interest. The peak burnup of 28% and peak cladding radiation damage of 520 dpa is achieved after residence time of 46 years. The ALCHEMY fuel performance code (Latta et al., 2012; Touran et al., 2017), developed in house, was used to evaluate TWR fuel pin compliance with the engineering design basis limits. Fuel performance results show that peak cladding strains could meet the 3% design basis limit. The thermal creep component which may cause creep rupture was well below the 1% limit. The models used to predict fuel performance need to be confirmed through an irradiation program.

The fuel duct assembly dilation is required to be less than the envelope created by the surrounding fuel assemblies. For the TWR reactor core assembly design, allowable dilation is equal to 3.5 mm at refueling conditions.

The unrestrained dilation of the TWR fuel assembly was calculated using the commercial finite element analysis tool ABAQUS 6.11.1 (ABAQUS, 2011) with custom creep and swelling materials subroutines incorporating irradiation and thermal creep and swelling correlations for optimized HT9 material. The important loadings on the assembly include the dpa rate, the differential pressure across the duct wall, and the duct material temperature histories over life of the duct. These loads are spatially dependent (i.e., vary as a function of assembly height from the bottom of the duct) and time dependent (i.e., vary as the assembly is shuffled in the

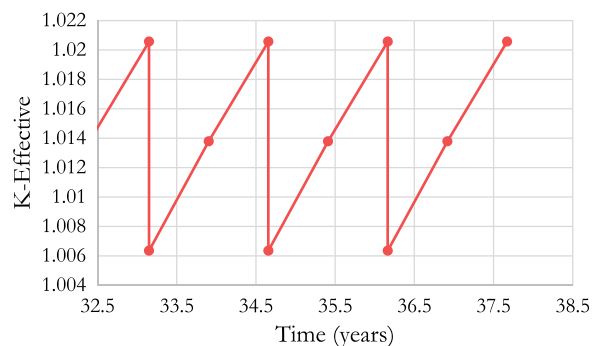


Fig. 6 k -Effective evolution over several equilibrium cycles.

¹²Change in reactivity over the cycle.

¹³¢ - cent is 1/100 of a dollar (\$), which is a unit of reactivity normalized to delayed neutron fraction (i.e. neutrons coming with delay from fission products versus prompt neutrons coming directly from fissions) useful in determining the response of reactor to insertion of reactivity.

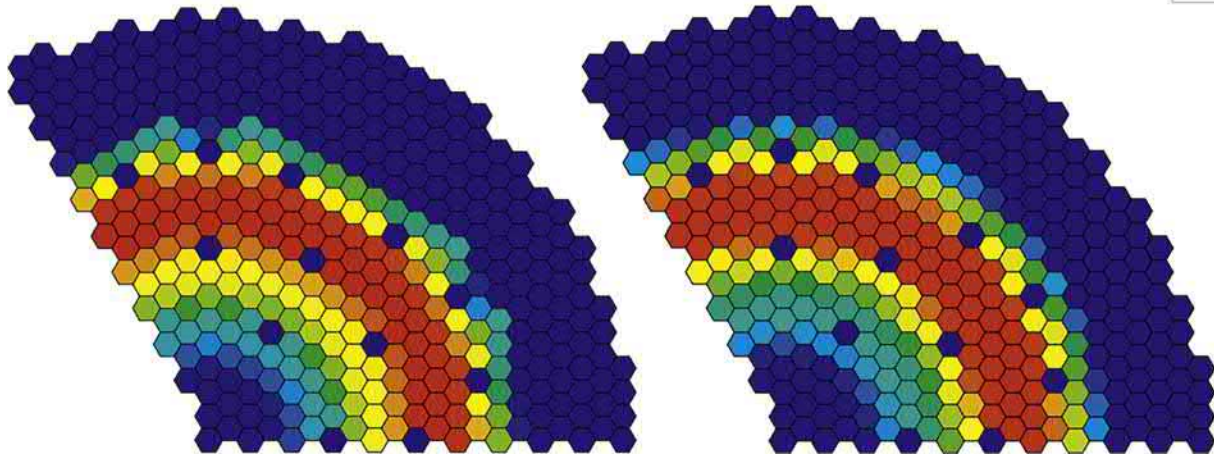


Fig. 7 Example of Power distribution at BOL (Left) and EOEC (right). Dark red indicates highest power, dark blue indicates lowest power.

Table 2 Key parameters of interest for TWR core analysis.

<i>Parameter</i>	<i>Value</i>
Peak burnup (%FIMA)	28
Peak cladding dose (dpa)	520
Core pressure drop (MPa)	0.45
Cycle length (EFPD ^a)	550
Peak linear heat rate (kW/m)	25
Peak cladding strain (%)	3
Peak duct dilation at refueling conditions (mm)	3.3

^aEffective full power days.

core). The duct performance analysis confirmed that for the load case and geometry resulting in the greatest amount of duct dilation, the duct would remain within the allowable envelope when lifted from or inserted into the core.

Safety analyses were performed using SAS4A-SASSYS1 code (Cahalan and Fanning, 2012). A number of Design Basis Accidents (DBA) have been analyzed and confirmed to meet all the limits with margin. Only one representative DBA, which is one of the most challenging DBAs, protected loss of flow, is presented as an example on Fig. 8. This transient is initiated by partial loss of primary coolant flow due to pump failure. With the loss of flow, the Plant Protection System (PPS) shuts down the reactor either due to higher temperature, low flow or high power to flow ratio. Before the shutdown occurs, peak cladding temperature is increasing due to power-to-flow imbalance. After the shutdown signal is transmitted to control rods, control rods drop inserting negative reactivity. Peak cladding temperatures never exceed the 650 °C design limit for DBAs.

The TWR technology's goal is to achieve higher safety protection for Beyond Design Basis Accidents (BDBA), such as in unprotected events, where the reactor fails to shut down. An example of such a scenario is an Unprotected Transient Over Power (UTOP) accident initiated by a severe positive reactivity addition. All PPS systems are assumed not to function. For this BDBA class of accidents it is desirable that the reactor response achieves stable conditions without any sodium boiling in the core. An example of TWR response to reactivity insertion of 0.14\$ from partial rod withdrawal (limited by rod stops) is shown in Fig. 9. The reactor power peaks at 2.3 times the nominal power and stabilizes back at original power level due to net negative coefficient of reactivity. The maximum peak cladding temperature reaches about 900 °C and coolant temperature remains below sodium boiling. The best-estimate results confirm that TWR core is nominally stable without any additional passive reactivity device. Passive reactivity devices will reduce peak temperatures and they are needed for unprotected loss of flow.

Areas of application

Traditional application of plants based on TWR technology is electricity generation. However, because this is a sodium-cooled reactor with outlet temperatures in the range 500 °C–540 °C, e.g., significantly higher than those of < 300 °C from LWRs, its applicability goes well beyond electricity generation. To address the need for innovative technology solutions for clean energy generation (Moniz et al., 2019), TerraPower, LLC is developing a technical architecture that combines nuclear power and thermal storage to reduce cost, broaden application and increase the value stream. This architecture can provide maximum flexibility for nuclear applications and sizing (e.g., energy capacity; electric, thermal, or both) as both energy generation and distribution transition to their

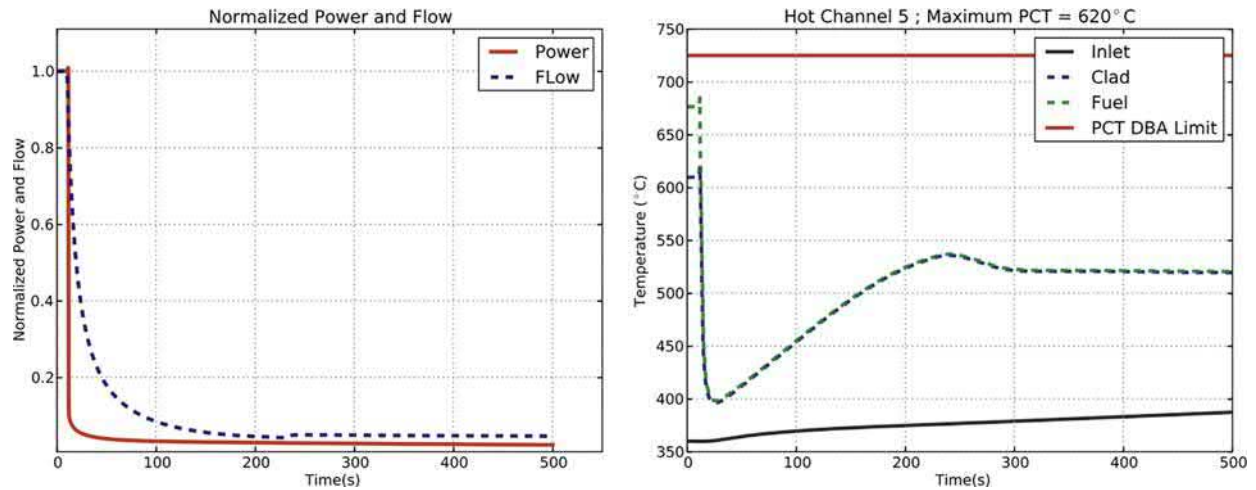


Fig. 8 Example of TWR plant response to protected loss of flow.

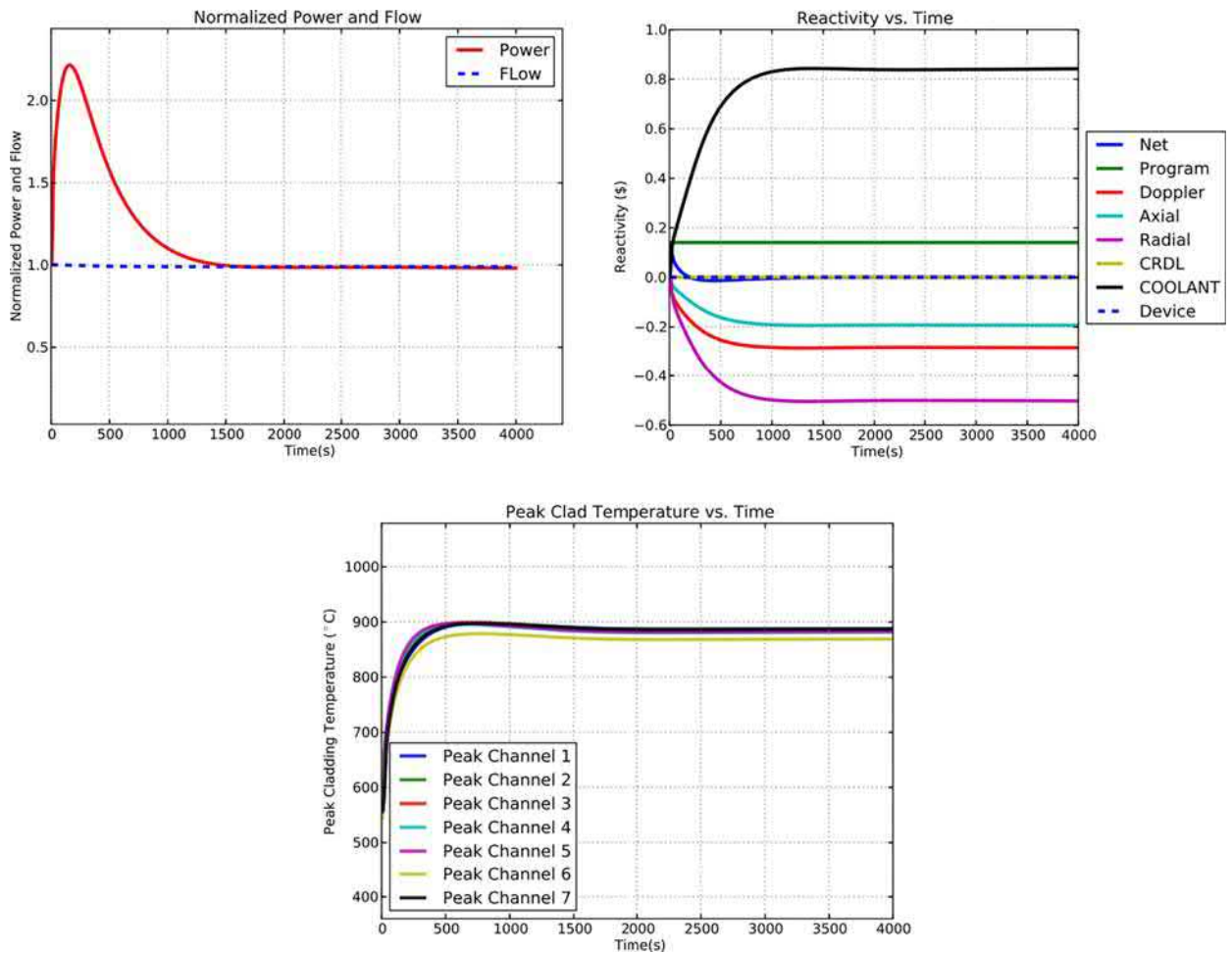


Fig. 9 Example of TWR plant response to protected loss of flow.

future form (which is naturally uncertain). An example illustration of this new architecture, designated TWR Integrated Energy System (IES) referred to as TWR-IES is shown in Fig. 10. In this architecture, the nuclear reactor is used as a source of heat that is sent to a separate thermal energy storage system which is located outside of the nuclear site boundary. This integrated energy system architecture potentially brings four main beneficial features:

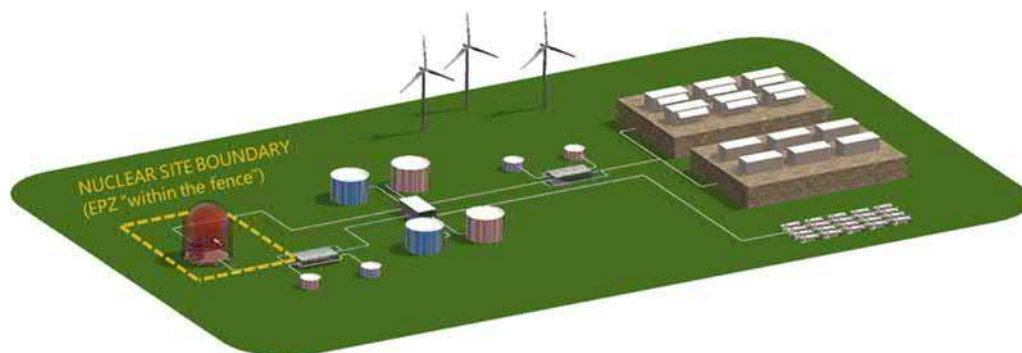


Fig. 10 Integrated energy system—synergy between thermal energy storage and TWR technology increases nuclear versatility. Storage (blue and brown large tanks) allows decoupling of reactor from power conversion applications (illustrated in the right bottom corner as modular supercritical CO₂ cycle) and enables rapid load following for electricity generation and use of industrial heat off-site (large buildings on the right illustrating heat consumers). Commercially available solar salt can be used as heat transport and storage fluid.

- (1) reduces total system cost,
- (2) enables flexible electric demand (load) following and “profit following” in grids with larger penetration of renewables,
- (3) provides high temperature process heat at competitive cost with natural gas, not possible with LWRs, and
- (4) supports hydrogen generation through high temperature electrolysis.

These capabilities make carbon reduction possible in industrial processes and the transportation sector that account for about 75% of the world’s greenhouse gas emissions.

The cost of nuclear power plants is the main challenge facing nuclear power in many Western markets. Recent studies have shown that the main cost driver is not the nuclear technology itself, but rather the cost of large scale construction projects regulated by strict nuclear standards (Dawson and Sabharwall, 2017) and that the greatest promise for capital cost reduction is not necessarily in the technology advances of a reactor itself but in the overall plant design (Buongiorno et al., 2018). By greatly simplifying and reducing the scope/complexity of the construction project within the nuclear site, TWR-IES directly addresses what experts have concluded to be the main cost drivers. For example, in the TWR-IES architecture the nuclear plant and the scope of the nuclear construction project is reduced to its most basic form. The simplified reactor becomes a producer of heat energy. The interface between the reactor and the rest of the system is a heat exchanger. The remaining system components are functionally and spatially separated from the reactor outside of the nuclear site beyond the heat exchanger. In this architecture, the thermal energy storage, and balance of plant with power conversion systems are now constructed and operated in a less regulated, less expensive, and fully commoditized environment. Molten salt thermal storage systems are relatively inexpensive, an order of magnitude less than battery storage, and have recently achieved commercial readiness on GWh scale, such as 4.2GWh Solana Generating Station in Arizona (Peltier, 2014), in support of the concentrated solar power industry. Also, very small EPZ due to excellent safety benefits (see section “TWR definition and key advantages of TWR technology”) enables closer siting of these reactors to the nuclear process heat consumers than is possible for current LWRs.

The proposed TWR-IES system also addresses another challenge facing nuclear power in current and future electricity markets with increasing variability of electricity production, as discussed in Forsberg (2018). As the fraction of power generated by intermittent renewable sources is increasing, there is large variation in electricity supply with overproduction during typically the 9 a.m.–4 p.m. time window with solar energy pushing electricity prices down to very low values or even negative territory. Current nuclear plants have typically limited flexibility of rapid load following and they also need to keep high capacity factor to achieve low Levelized Cost of Electricity (LCOE). So even if they could match daily varying power demand, their LCOE is increased as capacity factor is decreased, making it harder for them to compete. Salt thermal storage allows TWR plant operation at 100% capacity factor and store energy in salt tanks and sell electricity during periods when demand is high and price is also high. Initial studies, using actual market data and a hypothetical flexible generation nuclear plus storage system, have shown that appreciable increases in revenue can be obtained by accessing these higher value prices.

As discussed in MIT (Buongiorno et al., 2018) and Breakthrough Energy (Moniz et al., 2019) reports, expansion of decarbonization into other industrial processes is an important challenge to be addressed. Energy consumption primarily in the form of heat in this sector is enormous with petroleum and chemicals being major consumers. TWR-IES with its high outlet temperatures of ~510 °C–540 °C and molten solar salt compatible with these temperatures provides opportunity to supply heat to large number of consumers up to temperature of 500 °C, such as petroleum refineries, various chemical plants, soda ash production plants, pulp and paper production plants, food processing plants and others.

The transportation sector is responsible for the second largest share of global energy consumption after industrial manufacturing. Until recently, transportation has been solely driven by gasoline fuels with no participation of clean nuclear energy in this sector. This is changing with recent arrival of electric cars driven by batteries and fuel cells running on hydrogen. TWR-IES can

provide both electricity and hydrogen products carbon-free and have significant impact on decarbonization of the transportation sector.

The TWR-IES can generate hydrogen using high temperature electrolysis and heat. It uses the heat from the heat storage to generate steam from water and electricity to raise temperature in the electrolyzer to 750 °C–900 °C through ohmic heating and thermal recuperation from effluent streams. The heat exchangers in the electrolyzer recuperate the heat from hydrogen and oxygen streams so only a small amount of ohmic heating is needed to keep the electrolyzer at desired temperature. Efficiency of this process is high as documented for a similar application by [Yildiz and Kazimi \(2004\)](#). TWR-IES can generate both electricity to charge car batteries and hydrogen at the same time. When electricity is not needed, it can generate more hydrogen and store hydrogen for distribution throughout the country as is done with gasoline. Unlike GW-scale thermal storage, which is limited to hours-long duration and relatively short distances for transport, hydrogen can be stored for much longer times and can be transported over long distances. The main challenge today remains in reduction of cost of electrolysis equipment and cost of hydrogen transportation to be competitive with gasoline.

Because climate change is a worldwide problem, any nuclear reactor providing carbon free heat, electricity and clean energy carrier, such as hydrogen, needs to be deployable on a worldwide scale, exhibit good uranium utilization and be proliferation resistant. TWR Reactors operating on depleted or natural uranium in a once through cycle is the perfect candidate as it has all these features. For more detailed discussion on non-electric applications of nuclear power reactors, readers are referred to Section 8 of this encyclopedia, specifically chapters by [Forsberg \(2021\)](#), [Bienvenu \(2021\)](#), [O'brien \(2021\)](#), as well as other chapters.

Development status and overall TWR program

Confirmation of the feasibility of the core operating on depleted uranium fuel reloads, described in section “Feasibility of TWR core operating on depleted uranium” is only the first small step before the actual TWR plants can be deployed. The key areas of development include qualification of new mechanically bonded fuel with FCCI barrier,¹⁴ cladding and duct performance up to doses of 500 dpa, development and validation of analysis tools and many other aspects of the design.

The fuel and core components represent a significant leap in performance beyond existing experience. Satisfactory material and fuel performance at high burn-up and dpa level are essential to the TWR Program. TerraPower, LLC has a significant ongoing fuels and material development program to develop and qualify fuel and materials for TWR core components. The program addresses key technology challenges and involves the following key activities:

- Legacy data collection. Data from sodium bonded fuel pins from EBR-II and FFTF reactors are analyzed to expand the database for benchmarking the models against irradiated data and better understanding of key phenomena for modeling behavior of metallic fuel. Database of fuel pin as built characteristics, irradiation history and Post-Irradiation Examinations (PIE) has been developed. FFTF metal fuel pins are of special value since their size and histories are much closer to commercial fuel form than EBR-II pins.
- Development of commercial scale fuel material production. Extrusion was selected as a commercially scalable metal fuel fabrication for TWR fuel because it is a mature industrial process with high throughput and capable of delivering dimensionally precise products without major waste stream associated with historic casting process. First U-10Zr¹⁵ extruded fuel was made in December 2016 at lab-scale fuel fabrication facility at INL, and enriched U-10Zr fuel was made in October 2018 for irradiation at Advanced Test Reactor (ATR).
- Barrier development against Fuel-Cladding Chemical Interaction (FCCI). FCCI is a phenomenon where fission products (typically lanthanides) diffuse into the fuel cladding steel and embrittle the cladding. FCCI depends in part on time at temperature. A practical long lifetime TWR fuel pin requires a barrier between the fuel and the cladding. Various materials and manufacturing methods are being explored to develop effective barriers against FCCI, followed by diffusion tests in laboratory furnaces and irradiation testing in ATR to verify performance.
- Metal fuel irradiation at ATR. ATR is used to obtain early performance verification of innovations in the metal fuel design. Cadmium shrouds are used to eliminate 99% of thermal neutrons to simulate a fast reactor neutron environment. Irradiation of advanced fuel rodlets started in November 2013 and of sodium bonded fuel in July 2019. Two specimens of advanced fuel were discharged at 4% burnup and PIE was completed confirming excellent performance of FCCI barrier and as predicted filling of the inner hole with porous fuel—see [Fig. 11](#). Two additional specimens were discharged at 6% burnup and are undergoing PIE while a fifth rodlet is undergoing irradiation to 15% burnup. After these early tests, fuel assembly irradiation tests will ultimately be performed in a fast test reactor under prototypical flux and dose conditions. The purpose of the test of sodium bonded metal fuel is to confirm consistency of irradiation performance of newly manufactured/extruded fuel with historic injection cast fuel.
- Transient performance of ferritic martensitic steel clad fuel pins. Transient performance data for validation of fuel performance models are key for licensing of metal fuel, especially during overpower transients. The Transient Reactor Test Facility (TREAT) at Idaho National Laboratory (INL) was recently restarted, and TerraPower, LLC is performing design and testing of a new sodium loop in collaboration with DOE to support transient testing of irradiated fuel pins from FFTF and advanced fuel pins.

¹⁴Barrier protecting cladding material from chemical interaction with lanthanide fission products.

¹⁵An alloy of uranium with 10 wt% Zirconium.

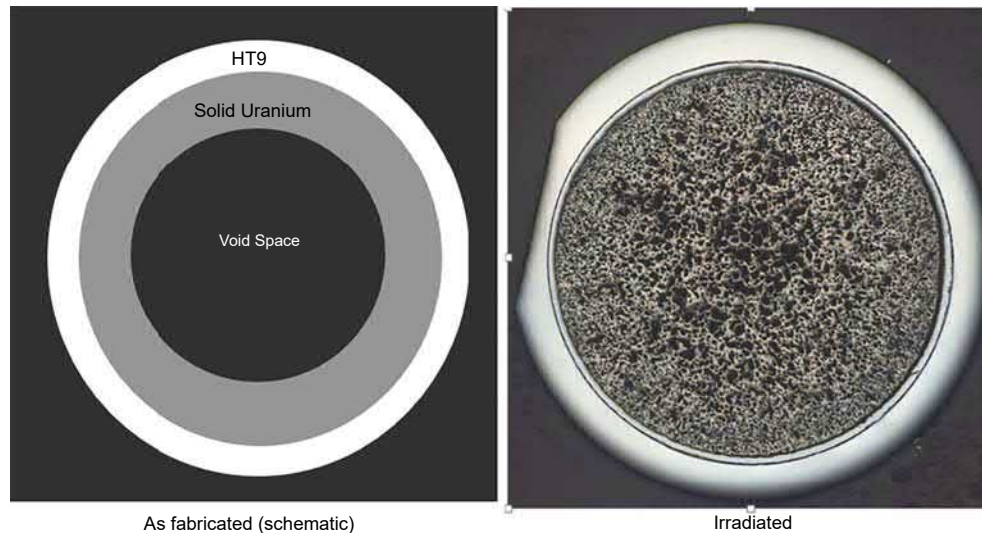


Fig. 11 Schematic of as fabricated (left) and irradiated (right) TerraPower advanced metallic fuel. PIE of fuel irradiated to 4% burnup confirms expected closure of central voided space with porous fuel and shows no signs of FCCI.

- Optimization and testing of optimized ferritic martensitic steel for cladding and ducts. Studies of material characteristics and former industry experience has resulted in the selection of ferritic martensitic steels, with HT9 as the primary candidate. To improve understanding of HT9 irradiation behavior, five different heats of archived HT9 samples from the FFTF irradiation programs were characterized. The data has enabled completion of an optimization program of the HT9 microstructure with the goal of improved irradiation performance, while remaining within the original HT9 specification; a critical distinction that allows historical irradiation data of HT9 to be used for qualification and licensing. Heavy ion irradiation of HT9 up to 600 dpa was performed to measure swelling behavior. While heavy ion irradiations are not capable of emulating all aspects of neutron induced irradiation damage, they have proven successful in replicating specific radiation effects such as swelling. [Fig. 4](#) shows substantial reduction of void swelling for TerraPower's optimized HT9 versus that of archived HT9 samples. Mechanical testing also confirmed that improved microstructure in optimized HT9 led to performance improvements of mechanical properties. Lessons learned from HT optimization were used to produce heats of other ferritic martensitic steels for both heavy ion and neutron irradiations. Over 1000 samples, including FFTF HT9 already irradiated to 155 dpa, as well as TerraPower optimized HT9, and advanced steels have been irradiated in the BOR-60 reactor ([Izhutov et al., 2015](#)). A supply chain for the optimized HT9 has also been developed. More than 8 MT of HT9 has been manufactured to support development and qualification activities, and a network of domestic and international suppliers has been created. Cladding tubes have been produced with multiple suppliers; successful production of ducts using stainless steel 410 has been demonstrated and duct manufacturing process development work using HT9 is currently underway.

In parallel with fuel and structural material development, work has been also performed on other primary and intermediate system components and equipment, such as core support structure, sodium pumps, intermediate heat exchangers, to evaluate their Technology Readiness Levels (TRLs) to provide guidance for development, testing, and the equipment qualification program. TerraPower, LLC has contracted with more than 50 industrial, national laboratories and academic institutions to address these needs including selected proof of manufacturing activities, such as FCCI barriers, fuel extrusion, HT9 ducts or manufacturing of full size fuel assembly. This is coupled with in-house testing to progressively reduce risk and advance TRLs. Selected examples are given in [Gilleland et al. \(2016\)](#). Significant development is also ongoing on computer codes for TWR performance and safety analyses and their qualifications. These are discussed in [Touran et al. \(2017\)](#).

Even with the robust fuel and material development program, the first TWR plants to be built will not be operating in equilibrium on depleted uranium. Rather, a stepwise approach to the deployment of TWR plants will be used where early plants will use medium assay low enriched uranium feed and non-vented fuel operated to lower peak burnups than 30%. As data and experience with operation and performance of TWR cores is accumulated, burnups will be increased while reload enrichment will be decreased in successive TWR units to ultimately reach the operation on depleted uranium reloads. Also, some aspects such as core radial expansion/bowing reactivity feedbacks, operating performance of passive reactivity devices need to be validated in a prototype or demonstration reactor. TerraPower, LLC first developed a conceptual design of 600MWe prototype reactor TWR-P to demonstrate the breed-and-burn process in a flux profile and shuffling scheme similar to a commercial TWR core, to validate the overall core performance and key characteristics and accommodate lead test assemblies to qualify fuel for commercial design. This was followed later at the request of an international customer with a conceptual design of a 300MWe version. Currently, TerraPower, LLC is focusing on the development of demonstration and commercial versions of TWR-IES plants coupled to salt storage technology

to accommodate rapid load following needs in grids with large penetration of renewables and cost reduction to be competitive with natural gas, while providing clean energy beyond electricity only.

See Also: Introduction to Breed and Burn Reactors.

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CANDLE Reactor Concept

Hiroshi Sekimoto, Tokyo Institute of Technology, Tokyo, Japan

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Glossary

Breed-and-burn reactor Reactor whose criticality is maintained only with bred fissile material.

Burnable poisons (BP) Materials that have a high neutron absorption cross section that are converted into materials of relatively low absorption cross section as the result of neutron absorption. To control large amounts of excess fuel reactivity without control rods, they are loaded into the core.

Burnup A measure of the amount of fission energy released from a given quantity of fuel. It is proportional to the neutron fluence with good accuracy.

Cladding The structural material that separates the nuclear fuel from the coolant.

Neutron economy The balanced account of the neutrons created and lost in a reactor. For more excess neutrons which can be utilized for breeding, it is said better neutron economy.

Neutron fluence The neutron flux integrated over a time period.

Neutron flux Number of neutrons crossing a unit area per second. Reaction rates, such as the fission rate, are proportional to the neutron flux.

Neutron spectrum The neutron flux energy distribution.

TRISO Tri-isotropic coated fuel particles which have been developed for HTGR.

Abbreviations

BOC/EOC Beginning/end of cycle

CDA Core disruptive accident

dpa Displacement per atom

FP Fission product

LBE Lead bismuth eutectic
 LFR Lead and/or LBE cooled fast reactor
 MOTTO cycle Multichannel once-through-then-out cycle
 SFR Sodium-cooled fast reactor

Introductions

A future ideal nuclear energy system

An advanced civilized society involves a large amount of energy consumption. Since the industrial revolution, mankind has met this demand for energy through the massive consumption of fossil fuels. We have been worried about the depletion of resources, but global warming and climate change due to the massive release of carbon dioxide gas are now more pressing issues, and there is a consensus to try to avoid using them. Renewable energies, especially solar energy and wind power, are expected. However, power they can generate is greatly influenced by the weather and fluctuates with time. Therefore, it is attempted to store energy with a battery or the like. However, storing energy increases the cost of energy considerably.

Nuclear power does not release carbon dioxide, and a breeder reactor can solve resource problems. However, as shown in Fig. 1, there is a problem of the technologies common to atomic bomb and a problem of the production of radioactive materials (Sekimoto, 2010b).

The technologies common to atomic bomb causes the problem of nuclear proliferation in terms of nations and the problem of physical protection in terms of terrorists. The production of radioactive materials also causes problems with accidents when the reactor is in operation, and wastes problems even after the operation is over. The risk of these problems cannot be reduced to zero, but can be reduced to an acceptable level by properly designing the nuclear system. These problems must be solved within an economic tolerance. There are many problems, and the researches on advanced nuclear reactors are trying to solve some of these, but no reactor has yet been proposed that can solve all of them simultaneously. Section “How CANDLE reactors satisfy the requirements for future ideal nuclear energy system” explains how the CANDLE reactor can solve these problems.

What is the CANDLE reactor concept?

Concept of the CANDLE burning strategy

CANDLE stands for Constant Axial shape of **Neutron flux**, nuclide densities and power profile During Life of Energy production (Sekimoto and Ryu, 2000a). This acronym also implies candle-like **burnup**. When this burnup strategy is employed, the fissioning region propagates along the core axis at a speed proportional to the power output without changing the spatial distributions of the nuclide densities, neutron flux, and power density. Significantly, unlike conventional reactor designs, it is not necessary to equip movable components for burnup control (e.g., control rods and control reflector), despite the fuel being fixed in the core. Sekimoto (2010a) has written a book on this topic for general readers, which has a Chinese translation (Zhang and Guo, 2014).

The infinite neutron multiplication factor, k_{∞} , of the CANDLE reactor fuel varies with increasing **neutron fluence** or fuel burnup in the manner shown in Fig. 2A and in (Qvist, 2021). It is less than unity for the fresh fuel, but increases with increasing burnup, eventually exceeding unity. After reaching a maximum, it decreases, becoming less than unity again. k_{∞} becomes unity at $B = B_1$ and B_3 , where B is the burnup. B_4 is the maximum attained burnup.

Fig. 2B plots the same data along the core axis. k_{∞} increases with burnup on the left side of the peak, and decreases on the right side. Consequently, the peak is shifted to the left (i.e., to the fresh fuel side) with burnup. In this Chapter, we will consider a vertical core and the peak is shifted to the bottom.

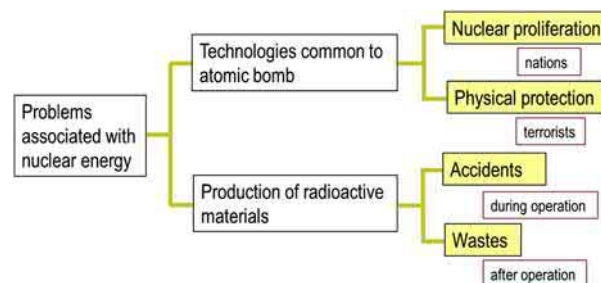


Fig. 1 Problems associated with nuclear energy.

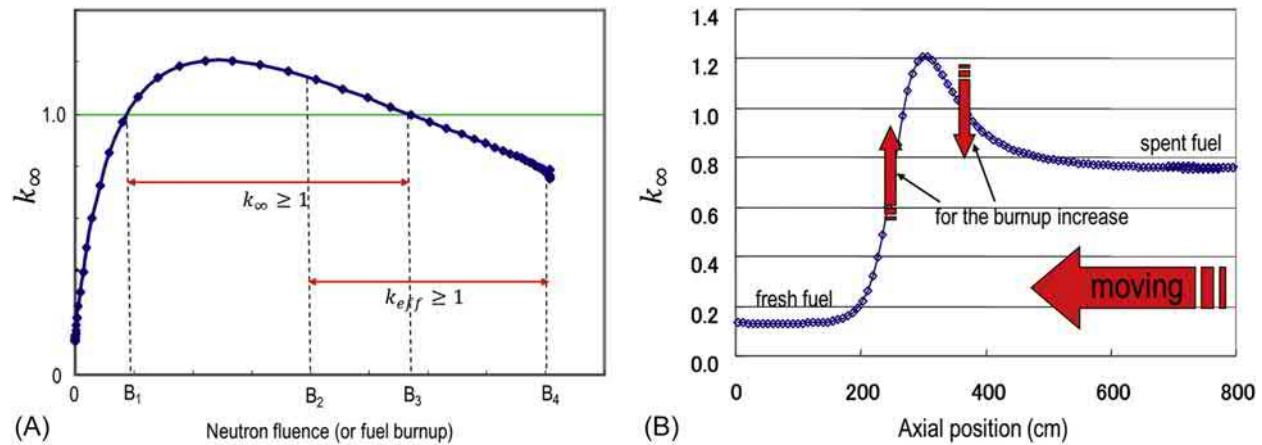


Fig. 2 Infinite-medium neutron multiplication factor variations in CANDLE fast reactor core. (A) Plotted against fuel burnup. The graph is for the infinite core. k_{eff} and B_2 are entered for the later explanation for the finite core. (B) Plotted along the central axis (z-axis). Arrows indicate the directions of shift with burnup.

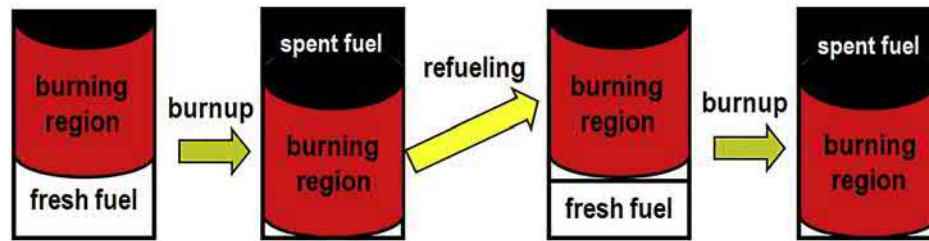


Fig. 3 Refueling in the CANDLE burnup strategy. From Sekimoto H (2010a) *Light a CANDLE*, 2nd edn., Tokyo, Japan: CRINES, Tokyo Tech. ISBN: 978-4-905205-00-5 C3053 E. Retrieved from https://www.researchgate.net/publication/305391375_Light_a_CANDLE_2nd_ed.

Although the power density level at a given axial location in the core varies, the shape of the power distribution remains constant. The burning region moving speed is proportional to the power level; the principle behind this is explained mathematically in section “**Mathematical explanation of CANDLE burning**”.

In reality, the core height is finite. When the burning region reaches the end of the core, the spent fuel should be removed from the top of the core, and adding fresh fuel at the bottom of the core as shown in Fig. 3. In this way, CANDLE burning can be sustained.

This refueling scheme is realized by cutting the fuel pins into several pieces. It is also possible by using a pebble-bed reactor (Ryu and Sekimoto, 2000).

Here we will consider k_{eff} expected for different discharge burnup. Since the relationship between fuel burnup and k_{∞} at different axial position for finite cores is almost the same for the infinite core, we will use Fig. 2A for our consideration. Now we consider the horizontal axis of this figure as the discharge burnup. The effective neutron multiplication factor, k_{eff} , of the core is less than unity for the fresh fuel ($B = 0$), but increases with increasing discharge burnup, and becomes unity at $B = B_2$, where $B_2 > B_1$. It continues to increase with increasing discharge burnup and takes its maximum, and then it decreases. It becomes unity again at $B = B_4$, where $B_4 > B_3$. If the neutron economy of the core becomes better, B_2 becomes smaller and B_4 becomes larger.

The discharge burnup B must be selected to satisfy $B_2 \leq B \leq B_4$. When the spent fuel is not discharged, the spent fuel burnup takes its maximum value B_4 . If we try to use uranium effectively, B_4 is preferred for the discharge burnup. However, the maximum permissible burnup for the fuel is usually much less than B_4 to satisfy material integrity, and we will consider B_2 later.

The following subsections describe two examples of CANDLE burning: one for a fast reactor and the other for a thermal reactor.

Fast reactors

When the average neutron energy of the neutron spectrum is high, it is called hard spectrum. Hard-spectrum fast reactors have excellent neutron economy. CANDLE burning was tried for these reactors using natural or depleted uranium as fresh fuel (Sekimoto and Ryu, 2000a,b; Sekimoto et al., 2001).

^{238}U in the fresh fuel absorbs neutrons and becomes ^{239}Pu . ^{239}Pu density of the fresh fuel is zero, but increases with burnup. Eventually it becomes saturated, where the production rate of ^{239}Pu is equal to its destruction rate, and the ratio of nuclide number density of ^{239}Pu to ^{238}U is constant. On the other hand, the total FP density increases with burnup. Therefore, at high burnups k_{∞} drops with burnup as seen in Fig. 2.

Since natural uranium is highly subcritical, many neutrons need to be absorbed by ^{238}U to produce a large amount of ^{239}Pu to bring the system into a critical state. Thus, it is important to use a nuclear reactor that has an excellent neutron economy. The fuel burnup is increased by supplying many neutrons to the fresh fuel. This results in a high burnup of spent fuel and considerably reduces the burning region moving speed. Edward Teller proposed a similar idea (Teller et al., 1996; Hyde et al., 2008) using thorium. However, CANDLE burning cannot be achieved in the truest sense when only thorium is used.

A possibility of the pebble-bed fast reactor for CANDLE burning was numerically demonstrated using natural uranium as a fresh fuel (Ryu and Sekimoto, 2000).

Thermal reactors with burnable poison

In the case of a fast CANDLE reactor, the average burnup of the spent fuel is as large as 40%FIMA (Fissions per Initial Heavy Metal Atoms). From now on, %FIMA is simply written as %. 40% is a great advantage in terms of effective use of natural uranium, but there is no fuel element that can withstand such a high burnup in the current fast reactor design. How to solve this problem is described in section "Solving high burnup problems". Apart from this, this subsection describes tri-isotropic (TRISO) coated fuel particles (Sawa et al., 1996), which have been developed as a fuel that can withstand such high burnup.

Since this fuel has a large parasitic neutron absorption and neutron slowing-down, it cannot be used in the fast CANDLE reactor, but used as a normal fuel in a High Temperature Gas cooled Reactor (HTGR). In fact, a CANDLE burning HTGR had been proposed (van Dam, 1998; Ohoka and Sekimoto, 2004). By using enriched uranium fuel with burnable poisons (BP), k_∞ can be varied as shown in Fig. 2.

By increasing the uranium enrichment and gadolinium BP concentration, the burnup of the spent fuel can be increased. Although it can be used as a long-life reactor for special purposes, it is not a **breed-and-burn reactor** and will not be discussed further in this chapter.

Mathematical explanation of CANDLE burning

Some principles and characteristics of the CANDLE burning are easy to understand when they are mathematically explained. In this section, a steady state CANDLE burning equation is derived from basic equations of neutron transport and fuel burnup, and derive some relations among important parameters by using approximate treatment.

Basic equations of neutron transport and fuel burnup (Sekimoto et al., 2001)

For the investigation of CANDLE burning, we have to treat both neutron flux, $\phi(\vec{r}, \vec{\Omega}, E, t)$, and nuclide number density distributions, $n_j(\vec{r}, t)$, as dependent variables. The variables t and $\vec{r}$ are time and position in three dimensional space, respectively, and E and $\vec{\Omega}$ are energy and direction of motion of neutron, respectively. The neutron flux satisfies the following neutron transport equation:

$$\begin{aligned} & \frac{1}{v} \frac{\partial}{\partial t} \phi(\vec{r}, \vec{\Omega}, E, t) + \vec{\Omega} \cdot \nabla \phi(\vec{r}, \vec{\Omega}, E, t) + \sum_j n_j(\vec{r}, t) \sigma_{T,j}(E) \phi(\vec{r}, \vec{\Omega}, E, t) \\ &= \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}, t) \sigma_{S,j}(\vec{\Omega}' \cdot \vec{\Omega}, E' \rightarrow E) \phi(\vec{r}, \vec{\Omega}', E', t) \\ &+ \frac{1}{4\pi} \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}, t) \chi(E) \nu_j (1 - \beta_j) \sigma_{F,j}(E') \phi(\vec{r}, \vec{\Omega}', E', t) \\ &+ \sum_i \frac{\chi_i(E)}{4\pi} \lambda_i c_i(\vec{r}, t) + s(\vec{r}, \vec{\Omega}, E, t) \end{aligned} \quad (1)$$

where

- v : velocity of the neutron with energy E ,
- $\sigma_{T,j}(E)$: microscopic total cross-section of nuclide j for the neutron with energy E ,
- $\sigma_{S,j}(\vec{\Omega}' \cdot \vec{\Omega}, E' \rightarrow E)$: microscopic scattering cross-section of the nuclide j for the neutron with energy E' and direction $\vec{\Omega}'$ transferred to E and $\vec{\Omega}$. This reaction includes any neutron emitting reactions except fission,
- $\sigma_{F,j}(E)$: microscopic fission cross-section of the nuclide j for the neutron with energy E ,
- ν_j : average number of neutrons emitted by the fission of nuclide j ,
- $\chi(E)$: prompt fission neutron energy spectrum,
- $\chi_i(E)$: delayed neutron energy spectrum for the precursor i ,
- β_j : delayed neutron precursor fraction produced by the fission of nuclide j ,
- λ_i : decay constant of the delayed neutron precursor i ,
- $c_i(\vec{r}, t)$: number density of the delayed neutron precursor i ,
- $s(\vec{r}, \vec{\Omega}, E, t)$: external neutron source.

Although $\chi(E)\nu_j(1 - \beta_j)$ is a function of the energy, E' , of impinging neutron which causes the fission, this dependence is omitted for simplicity.

The delayed neutron precursor density $c_i(\vec{r}, t)$ satisfies the following equation:

$$\frac{\partial}{\partial t} c_i(\vec{r}, t) + \lambda_i c_i(\vec{r}, t) = \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}, t) \beta_{ij} \nu_j \sigma_{F,j}(E') \phi(\vec{r}, \vec{\Omega}', E', t) \quad (2)$$

where

$\beta_{i,j}$: fraction of delayed neutron precursor i produced by the fission of nuclide j , which satisfies

$$\beta_j = \sum_i \beta_{i,j} \quad (3)$$

The nuclide density satisfies the following equation:

$$\frac{\partial}{\partial t} n_j(\vec{r}, t) = -n_j(\vec{r}, t) \left(\lambda_j + \int_0^\infty dE' \sigma_{A,j}(E') \int_{4\pi} d\vec{\Omega}' \phi(\vec{r}, \vec{\Omega}', E', t) \right) + \sum_k n_k(\vec{r}, t) \left(\lambda_{k \rightarrow j} + \int_0^\infty dE' \sigma_{k \rightarrow j}(E') \int_{4\pi} d\vec{\Omega}' \phi(\vec{r}, \vec{\Omega}', E', t) \right) \quad (4)$$

where

$\sigma_{A,j}(E)$: microscopic absorption (including transmutation) cross-section of nuclide j for the neutron with energy E ,

$\sigma_{k \rightarrow j}(E)$: microscopic neutron cross-section of nuclide k that is transmuted into nuclide j ,

λ_j : decay constant of nuclide j ,

$\lambda_{k \rightarrow j}$: decay constant of nuclide k that is transmuted to nuclide j .

Here the nuclide j covers both actinides and FPs produced from the fuel. The production of FP from fission can also be treated by Eq. (4) by choosing $\sigma_{k \rightarrow j}(E)$ properly.

In addition to these equations we consider the equations of temperature and motion of materials. Coolant flow distribution affects strongly temperature distributions, and the temperature distributions affect effective microscopic cross sections and material densities and shapes (Sekimoto and Udagawa, 2006).

Steady state CANDLE burning equation (Sekimoto et al., 2001)

In the steady state CANDLE burning case, the reactor is in a critical state and the external neutron source does not stay in the core. Only time dependent phenomena considered is the burnup. In this situation the time derivative of neutron flux in Eq. (1) and the time derivative of delayed neutron precursor density in Eq. (2) can be neglected. These two equations can be combined to the following equation:

$$\begin{aligned} \vec{\Omega} \cdot \nabla \phi(\vec{r}, \vec{\Omega}, E, t) + \sum_j n_j(\vec{r}, t) \sigma_{T,j}(E) \phi(\vec{r}, \vec{\Omega}, E, t) = \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}, t) \sigma_{S,j}(\vec{\Omega}' \cdot \vec{\Omega}, E' \rightarrow E) \phi(\vec{r}, \vec{\Omega}', E', t) \\ + \frac{1}{4\pi k_{eff}} \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}, t) \chi_j(E) \nu_j \sigma_{F,j}(E') \phi(\vec{r}, \vec{\Omega}', E', t) \end{aligned} \quad (5)$$

Here the reactor is usually kept critical by adjusting the control rods. However, it is complicated to implement this procedure in neutron transport equation, and the eigenvalue k_{eff} is introduced instead. The reactor is controlled to produce the designed total power output, $P(t)$:

$$\Gamma \int_{Core} d\vec{r} \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}, t) \sigma_{F,j}(E') \phi(\vec{r}, \vec{\Omega}', E', t) = P(t) \quad (6)$$

where $\int_{Core} \bullet d\vec{r}$ denotes the integration over the whole core, and Γ is the energy released per fission.

The basic equations for our study are Eqs. (4), (5), (6).

The ideal CANDLE burning can be realized for core geometries where the core length is infinite and the geometry perpendicular to the core axis is the same at different axial positions. We also assume a constant power operation and Eq. (6) becomes

$$\Gamma \int_{Core} d\vec{r} \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}, t) \sigma_{F,j}(E') \phi(\vec{r}, \vec{\Omega}', E', t) = P_0 \quad (7)$$

In this scenario the neutron flux shape and nuclide densities move with burnup, whose relative shapes are not changed and their positions move with a constant speed V along z -axis for CANDLE burning. For this burnup scheme, when the following Galilean transformation (Goldstein, 1950) given by

$$\vec{r}_G = \vec{r} + \vec{V}t \quad (x_G = x, \quad y_G = y, \quad z_G = z + Vt) \quad (8)$$

$$t_G = t \quad (9)$$

is applied to Eqs. (4), (5), (7), and by replacing the variable $\vec{r}_G$ and t_G by $\vec{r}$ and t , respectively, then these equations will be changed to

$$V \frac{\partial}{\partial z} n_j(\vec{r}) = -n_j(\vec{r}) \left(\lambda_j + \int_0^\infty \sigma_{A,j}(E') \phi(\vec{r}, E') dE' \right) + \sum_k n_k(\vec{r}) \left(\lambda_{k \rightarrow j} + \int_0^\infty \sigma_{k \rightarrow j}(E') \phi(\vec{r}, E') dE' \right) \quad (10)$$

$$\begin{aligned} \vec{\Omega} \cdot \nabla \phi(\vec{r}, \vec{\Omega}, E) + \sum_j n_j(\vec{r}) \sigma_{T,j}(E) \phi(\vec{r}, \vec{\Omega}, E) &= \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}) \sigma_{S,j}(\vec{\Omega}' \cdot \vec{\Omega}, E' \rightarrow E) \phi(\vec{r}, \vec{\Omega}', E') \\ &+ \frac{1}{4\pi k_{eff}} \int_0^\infty dE' \int_{4\pi} d\vec{\Omega}' \sum_j n_j(\vec{r}) \chi_j(E) \nu_j \sigma_{F,j}(E') \phi(\vec{r}, \vec{\Omega}', E') \end{aligned} \quad (11)$$

and

$$\Gamma \int_{Core} d\vec{r} \int_0^\infty \sum_j n_j(\vec{r}) \sigma_{F,j}(E) \phi(\vec{r}, E) dE = P_0 \quad (12)$$

where we have used the scalar flux, $\phi(\vec{r}, E)$, which is defined by

$$\phi(\vec{r}, E) \equiv \int_{4\pi} \phi(\vec{r}, \vec{\Omega}, E) d\vec{\Omega} \quad (13)$$

In many cases, we can ignore radioactive decay in Eq. (10). Usually the decay of ^{241}Pu and some FPs may contribute some, but their contribution to k_{eff} is almost negligible. In this case Eq. (10) becomes

$$V \frac{\partial}{\partial z} n_j(\vec{r}) = -n_j(\vec{r}) \int_0^\infty \sigma_{A,j}(E') \phi(\vec{r}, E') dE' + \sum_k n_k(\vec{r}) \int_0^\infty \sigma_{k \rightarrow j}(E') \phi(\vec{r}, E') dE' \quad (14)$$

If $\phi(\vec{r}, E)$ and $n_j(\vec{r})$ are the solution of Eqs. (11)–(14), $\alpha \phi(\vec{r}, E)$ and $n_j(\vec{r})$ are also the solution of these equations for αV instead of V , and for αP_0 instead of P_0 . It means that if the power level is changed, the relative power shape does not change but only power level and burning region speed change by that rate.

Approximated treatment

Here, we will use one-speed diffusion approximation (Bell and Glasstone, 1970). Eq. (14) suggests that each nuclide density can be expressed by using neutron fluence, θ , where

$$\theta(z) = \int_{-\infty}^z \phi(z') \frac{dz'}{V} \quad (15)$$

It is rewritten as

$$\phi(z) = V \frac{d\theta(z)}{dz} \quad (16)$$

Therefore, k_∞ can be also written as dependent on θ . By using the one-group diffusion approximation, Eq. (11) can be written as

$$-L^2 \frac{d^2}{dz^2} \phi(z) + \phi(z) = \frac{k_\infty(\theta(z))}{k_{eff}} \phi(z) \quad (17)$$

where L is the diffusion length.

In order to obtain a simple analytical solution, van Dam (2000) approximates $k_\infty(\theta)$ as

$$k_\infty(\theta) = -4(k_{max} - k_{min}) \left(\frac{\theta}{\theta_M} \right)^2 + 4(k_{max} - k_{min}) \left(\frac{\theta}{\theta_M} \right) + k_{min} \quad (18)$$

$k_\infty(\theta)$ has the value k_{min} at $\theta = 0$ and θ_M , and its maximum value k_{max} at $\theta = \theta_M/2$. Using Eqs. (16), (18), the solution of Eq. (17) is given by

$$\theta(z) = \theta_M \frac{1}{1 + e^{-\alpha z}} \quad (19)$$

where the axial position is chosen so that the flux peak position locates at $z = 0$. The flux is given by

$$\phi(z) = V \theta_M \alpha \frac{1}{(1 + e^{-\alpha z})(1 + e^{\alpha z})} \quad (20)$$

where α is related to the half width, W , of flux shape as

$$W = \frac{2 \ln(3 + 2\sqrt{2})}{\alpha} = \frac{3.53}{\alpha} \quad (21)$$

k_{max} and k_{min} are expressed by using α as

$$\frac{k_{max}}{k_{eff}} = 1 + \frac{1}{2} L^2 \alpha^2 \quad (22)$$

$$\frac{k_{min}}{k_{eff}} = 1 - \frac{1}{2} L^2 \alpha^2 \quad (23)$$

From Eqs. (21), (22)

$$W = \frac{2.49L}{\sqrt{\frac{k_{max}}{k_{eff}} - 1}} \quad (24)$$

From this equation W is decreased by decreasing L and/or increasing k_{max} .

The maximum value of flux, ϕ_{max} , is given from Eq. (20) as

$$\phi_{max} = \frac{1}{4} V \theta_M \alpha = 0.88 \frac{V \theta_M}{W} \quad (25)$$

Therefore, the velocity of burning region is determined as

$$V = 1.13 \frac{\phi_{max}}{\theta_M} W \quad (26)$$

This equation is easy to be understood from the physical meaning of θ_M/ϕ_{max} and W/V . As already mentioned, the burnup of spent fuel for CANDLE reactor is very high. It means θ_M is very large and therefore V is very small.

Chen et al. (2012) extend the approximated treatment including a few important actinide nuclides.

Typical examples of the CANDLE reactor (parametric survey)

Though both fast and thermal CANDLE reactors were considered in the previous sections, the following sections focus only on fast CANDLE reactors. Even when discussing fast CANDLE reactors, there are many different designs that can be considered; for example, several kinds of fuels and coolants can be used, and the core designs can be varied. This section shows how the performance of fast CANDLE reactors vary when fundamental parameters are varied (Sekimoto and Ryu, 2000b; Sekimoto, 2001). The performances are compared in the equilibrium state. Table 1 shows the reference reactor design parameters.

The core is cylindrical with a radius of 2 m and an infinite height. However, setting the core height to infinity is not possible in practical calculations, so instead it was set to 8 m. When the core height used in the calculation is sufficiently large that the neutron flux and leakage are negligibly small at the upper and lower boundary of the core, the neutron flux distribution in the burning region will be unaffected by a change in the core boundaries and all other distributions can be considered as those for an infinitely long core. Furthermore, it was confirmed by investigating the effects of changing the boundary conditions from vacuum to reflective conditions.

Natural uranium was used as the fresh fuel. Since CANDLE burning requires an excellent neutron economy, a lead-bismuth-eutectic (LBE) cooled metallic fueled reactor was used for the reference reactor. The percentage of fuel volume (excluding cladding) was set to 50%, which is larger than that in current reactors; this increases the neutron economy but reduces the cooling capacity.

Table 1 Design parameters of reference reactor design.

Reactor	Thermal output	3000 MWt
	Core radius	200 cm
	Radial reflector thickness	50 cm
Fuel pin	Fuel form	U-10w%Zr
	Fuel pellet diameter	0.8 cm
	Cladding tube material	HT-9
	Cladding tube thickness	0.035 cm
Coolant		Pb-Bi (44.5%, 55.5%)
Fuel volume fraction (excluding cladding)		50%

Table 2 Effective neutron multiplication factor, burning region moving speed, and spent fuel burnup for various core design parameters.

(A) For different fuels				
Fuel	Oxide	Nitride	Metal	
Effective neutron multiplication factor	0.926	0.990	1.015	
Moving speed of burning region	4.7 cm/year	3.5 cm/year	3.8 cm/year	
Average burnup of spent fuel	452 GWd/t	445 GWd/t	426 GWd/t	
(B) For different coolants				
Coolant material	Sodium	Lead	LBE	Helium
Effective neutron multiplication factor	1.006	1.012	1.015	1.035
Moving speed of burning region	3.8 cm/year	4.1 cm/year	3.8 cm/year	3.8 cm/year
Average burnup of spent fuel	415 GWd/t	427 GWd/t	426 GWd/t	413 GWd/t
(C) For different fuel volume fractions				
Fuel volume fraction (excluding cladding)	40%	50%	60%	
Effective neutron multiplication factor	0.989	1.015	1.035	
Moving speed of burning region	4.8 cm/year	3.8 cm/year	3.2 cm/year	
Average burnup of spent fuel	427 GWd/t	426 GWd/t	427 GWd/t	
(D) For different core radii				
Core radius	150 cm	200 cm	250 cm	
Effective neutron multiplication factor	0.999	1.015	1.023	
Moving speed of burning region	3.8 cm/year	3.8 cm/year	3.8 cm/year	
Average burnup of spent fuel	429 GWd/t	426 GWd/t	426 GWd/t	

(A) and (B) From Sekimoto H (2001) Applications of "CANDLE" burnup strategy to several reactors, ARWIF-2001. Chester, UK.; (C) and (D) From Sekimoto H and Ryu K (2000b). Feasibility study on the CANDLE new burnup strategy. *Transactions of the American Nuclear Society* 82: 207–208.

The fuel and coolant materials, fuel volume fraction, and core radius are varied, and the effective neutron multiplication factor, k_{eff} , the burning region moving speed, V , and the average burnup of spent fuel are calculated and shown in **Table 2**. The value of V is very low—about 4 cm/year in all cases. The average burnup of spent fuel was extremely high—about 400 GWd/t, which indicates that about 40% of the loaded fuel was burnt.

Table 2A shows that only the k_{eff} of the metallic fuel exceeds unity. It is clearly difficult to realize CANDLE burning using oxide fuel, whereas it may be possible with a little effort using nitride fuel. Both V and the average burnup of spent fuel differ for different fuels, but V varies more. This large variation in V is attributed to the different fissile material densities of the fuels.

Table 2B shows that k_{eff} also varies for different coolants. The differences are mostly attributed to the different neutron economies of the coolants. Helium has the highest k_{eff} and sodium has the lowest k_{eff} . However, the differences are not significant and CANDLE burning is possible for any coolant when a metallic fuel is used. The cooling performance of sodium is better than of the other coolants, and it will be often employed in the following parts of this chapter.

Coolants with low neutron slowing-down and absorbing powers are suitable for fast reactors. A large scattering cross section is desirable for small reactors since it results in effective neutron confinement. Since lead and LBE are better in terms of these parameters than sodium, they have been used as coolants for small long-life fast reactors (Sekimoto and Zaki, 1995). Previous lead- or LBE-cooled small long-life reactors employ natural lead for economic reasons, but natural lead contains several isotopes. On the other hand, ^{208}Pb , whose isotopic abundance is 52.4%, is a double magic nucleus and has a much smaller capture cross section and a higher threshold energy of inelastic scattering than the other isotopes of natural lead. An economical separation of this isotope has been proposed (Khorasanov, 2013). A ^{208}Pb -cooled metallic fuel fast reactor is expected to have a harder neutron spectrum than natural-lead-cooled (Okawa and Sekimoto, 2010). The void reactivity coefficient shifts in the negative direction when ^{208}Pb is used as a coolant, since the spectrum hardening effect by coolant voiding becomes small. Sekimoto and Nakayama (2014) changed the void coefficient in a medium-sized CANDLE reactor from \$ 7.94 to \$ 3.09 by changing the coolant from sodium to ^{238}Pb . When CANDLE reactors are improved by design change, the ^{208}Pb coolant can further improve their performance. They are explained later in section "Characteristics of CANDLE reactor".

The effect of fuel volume fraction is shown in **Table 2C**. When the fuel volume fraction is large, k_{eff} is large so that CANDLE burning can be realized more easily. The change in V is inversely proportional to the fuel volume fraction, but the average burnup of spent fuel does not change for different fuel volume fractions.

Table 2D shows that the core radius also exerts a considerable influence. When the core radius is large, k_{eff} is large so that CANDLE burning can be realized more easily. However, neither V nor the average burnup of spent fuel changes for different core radii.

The properties of CANDLE burning obtained here by numerical simulation are consistent with the properties derived mathematically in the previous section.

Startup of CANDLE burning

The above demonstrations of CANDLE burning only considered equilibrium steady states since they typically exhibit CANDLE characteristics. However, the equilibrium core contains many radioactive materials including higher actinides and FPs. It may thus seem to be difficult to construct the initial core, but there are several methods for overcoming these problems.

The initial core may be realized by supplying a sufficient number of neutrons from an external neutron source (Teller et al., 1996). However, this method is expensive and the power profile varies drastically in the early stages of the first cycle. By using enriched uranium (Sekimoto and Miyashita, 2006) and/or plutonium (Sekimoto et al., 2003) substituted for actinides in the equilibrium core, it may be possible to construct an initial core that has a similar power profile to the equilibrium profile and that can reach an equilibrium state without any drastic changes.

The isotopic fraction of plutonium (plutonium vector) varies for different spent fuels, which makes it difficult to construct the first core for CANDLE burning using plutonium from spent fuel. On the other hand, uranium enrichment can be precisely controlled and makes it easier to construct the first core (Sekimoto and Miyashita, 2006). Fig. 4A shows the axial nuclide number densities of ^{235}U and ^{238}U . The maximum enrichment is about 13% at an axial position of about 350–400 cm, which is well below the constraint of 20%. The power profile remains almost constant along operation time as shown in Fig. 4B, but it shifts at a constant speed in the negative axial direction, as expected.

Characteristics of CANDLE reactor

Advantageous characteristics (Sekimoto et al., 2001; Sekimoto, 2010a)

The advantages of using CANDLE burning in a fast reactor with an excellent neutron economy are summarized. The items discussed here will be further discussed in section “How CANDLE reactors satisfy the requirements for future ideal nuclear energy system”.

- No fissile material is required except for the initial core.
Accordingly, if the initial core can be constructed, no enrichment or reprocessing facilities are required. This feature is common to breed-and-burn fast reactors (Qvist, 2021).
- There is no need to control burnup excess reactivity.
Accordingly, it is not necessary to equip movable components for burnup control such as shim control rods.
- The average burnup of spent fuel is about 40%.

About 40% of natural (or depleted) uranium fissions to generate energy. For a simple once-through cycle, the energy extracted from natural uranium is about 50 times greater than in presently used LWRs, and the volume of spent fuel for geological disposal will be one-eighth of LWRs with 4% enriched uranium fuel (5% discharge burnup) per unit energy production. Miscellaneous waste, especially which associated with fuel cycle, will also be extremely small.

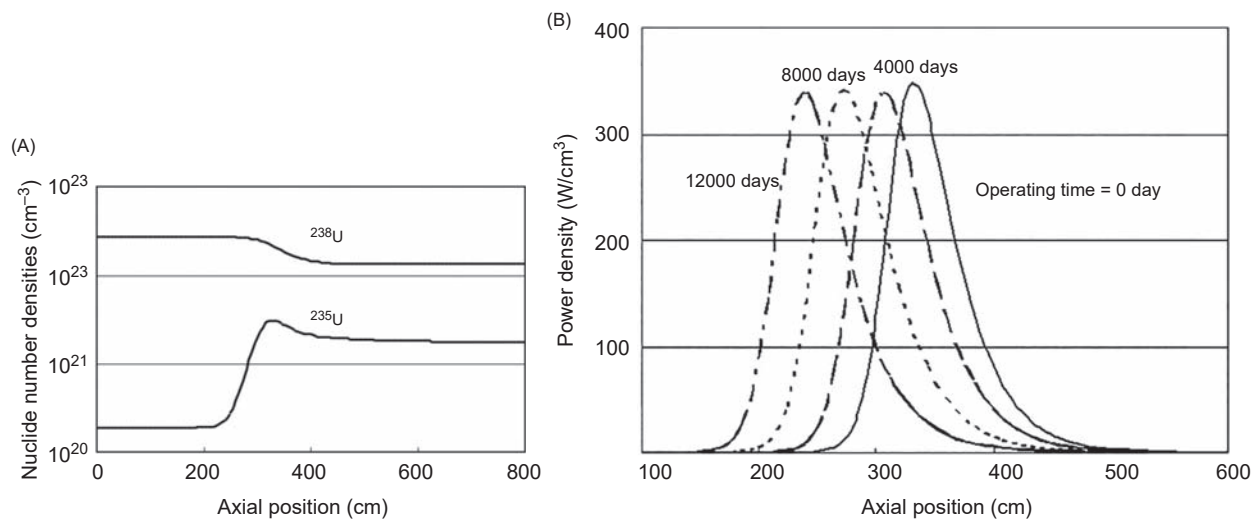


Fig. 4 Startup of CANDLE burning using enriched uranium. (A) Nuclide number densities of ^{235}U and ^{238}U as functions of position along the initial core axis. (B) Power density distributions for four different burnup durations. From Sekimoto H and Miyashita S (2006) Startup of “candle” burnup in fast reactor from enriched uranium core. *Energy Conversion and Management* 47(17): 2772–2780.

- The speed of burning region, V , is about 4 cm/year.
This extremely slow speed makes it easy to design a long-life reactor. For example, to increase the core life by 20 years, the core height just needs to be increased by only 0.8 m.
- Even in case of a core disruptive accident (CDA), it is less likely for a recriticality accident to happen.
There is no excess fissile material in the core to produce excess reactivity, and no need for neutron absorber to suppress it. And the amount of coolant, which suppresses k_{eff} , is smaller than the conventional fast reactors. Therefore, when the core is disrupted and fuel rearrangement occurs, it is less likely to lead to a recriticality accident. Even if it happens, the induced excess reactivity is expected to be much less than in conventional fast reactors.
- The number density distribution of each nuclide does not change due to fuel burnup.
The number density distribution of each nuclide does not change due to fuel burnup but only shifts along the core axis. Therefore, the reactor characteristics such as power peaking and power coefficient of reactivity do not change due to fuel burnup. Therefore, the estimation of core condition becomes very reliable. The reactor operation strategy remains un-changed for the entire core life.
- The radial power shape is very smooth.
The power density distribution is smooth, and does not change due to fuel burnup but only shifts along the core axis. In conventional reactors, even if the radial power profile is optimized at the beginning, it will change along burnup. This is not the case for CANDLE. Therefore, it is possible to highly optimize the radial power distribution. This is explained in more detail in section “**Power density flattening and core height shortening**”. This can increase the energy output from the core.

Problematic characteristics (Sekimoto et al., 2001; Sekimoto, 2010a)

As mentioned above, the CANDLE reactor has many wonderful properties, but at the same time has the following challenging problems: Some solutions for these problems are given in section “**How to solve the problematic characteristics**”.

- Difficult to establish criticality
As mentioned in section “**Fast reactors**”, in the fast CANDLE reactor, fissile materials to maintain criticality are mostly created by ^{238}U absorbing neutrons in the core. Conventional fast breeder reactors use the same principle, but the criticality can be easily maintained by removing FPs from the core by refueling. On the other hand, in the CANDLE reactor, the burning region moves, so the FPs slowly exits the burning region. This method leaves a significant amount of FPs in the core, which makes it much more difficult to maintain criticality than in conventional fast breeder reactors. This is a big problem. However, we have already introduced some CANDLE core design examples that can achieve criticality.
As burnup increases, FPs accumulate, more neutrons are absorbed into the FPs. In the original CANDLE reactor, where the discharged burnup is taken its maximum limit as shown in Fig. 2A, many neutrons will be absorbed by FPs. Extraction with a lower burnup reduces such parasitic neutron absorption (Toshinsky et al., 1999). A standing-wave breed-and-burn core (Hejzlar et al., 2013; Qvist, 2021) thus facilitates the criticality problem. In the present chapter, a similar approach is adopted for the modified CANDLE described in section “**Solving high burnup problems**”.
- High burnup problems
It is also important to consider ways of controlling the reactor, including power level regulation and shut down. It is important to design a fast reactor that excels in neutron economy and can be operated safely. These will be further explained in detail in section “**How to solve the problematic characteristics**”.
- High burnup problems
For realizing an ideal CANDLE reactor, it is necessary to use materials that can withstand over its maximum discharge burnup, which is usually around 47%. There is currently no fuel element material that can withstand such a high burnup. The volume of accumulated FPs becomes large with high burnup. In this state, the pressure in the fuel element will become excessively high, making it necessary to release gas from there. This necessitates a major design change. Even if the gaseous FPs can be managed, the solid FP volume will also become high, especially when metallic fuel is used; to counteract this, the fresh fuel smear density will need to be reduced. The most severe problem is that current **cladding** can only withstand a burnup much lower than 47%. This problem is discussed in section “**Solving high burnup problems**”. The power flattening, which is discussed in section “**Power density flattening and core height shortening**”, can somewhat alleviate the high burnup problem because the maximum discharge burnup approaches the average discharge burnup.
- Low power density
In the CANDLE reactor, it is necessary to reduce the coolant volume fraction in order to increase the neutron economy. For this reason, the cooling performance is deteriorated and the power density is lowered. The power density is directly related to the economy, and lower density increases the electricity price.

How to solve the problematic characteristics

In the previous section, problematic characteristics are introduced. This section describes how these problems can be mitigated.

Reactor control without control rods

The use of control rods, which consist of shim control rods, regulation control rods and shutdown (scram) rods significantly hurts the neutron economy by reducing the fuel volume fraction in the core and increasing the parasitic neutron absorption. If they are eliminated from the core, it is expected that the neutron economy can be greatly improved. The control rods also disturb power shape. They decrease average power density and deteriorate the economic efficiency. Since CANDLE reactors do not require the shim control rods, only regulation control rods and shutdown rods need to be considered.

- Power level control without control rods (Sekimoto and Nakayama, 2014)

Even without regulation control rods, the power level can be changed by adjusting the coolant flow rate and/or inlet coolant temperature. Three methods shown in Table 3 are considered.

These methods are attempted for two CANDLE reactors, sodium cooled fast reactor (SFR) and ^{208}Pb cooled fast reactor (LFR). The calculation results are shown in Fig. 5, where the outlet coolant temperature is the averaged value over all coolant channels. The results for SFR is shown in (A) and LFR in (B). For both reactors, it can be demonstrated successfully to change the power level from its full power to zero by Method 3 adjusting both coolant flow rate and inlet coolant temperature while the coolant outlet temperature is kept at its designed value. The change of coolant inlet temperature is smaller for LFR.

- Shut-down and start-up by moving some fuel assemblies

Since the shutdown rod is not in the core during power operation, it does not deteriorate the neutron economy due to neutron absorption, but it needs the space for its insertion in the core, thus deteriorating the neutron economy due to neutron leakage. When movable fuel assemblies are introduced and removed at shutdown, the shutdown rods are no longer needed.

A potentially dangerous issue with this system is unplanned insertion of removed fuel during cold shut down state. This problem can be mitigated by removing the fuel downward.

The second issue to consider is that k_{eff} should not increase while removing out the fuel. This is unlikely to happen because the neutron importance profile takes a single peak.

At present, however, the design of this core has not been studied enough.

Table 3 Adjusted and fixed parameters for investigated three methods for power level control without control rods.

	Adjusted parameters	Fixed parameters
Method 1	Coolant flow rate	Coolant inlet temperature
Method 2	Coolant inlet temperature	Coolant flow rate
Method 3	Coolant flow rate Coolant inlet temperature	Coolant outlet temperature

From Sekimoto H and Nakayama S (2014) Power level control of CANDLE reactor without control rods. *Annals of Nuclear Energy* 63: 427–431.

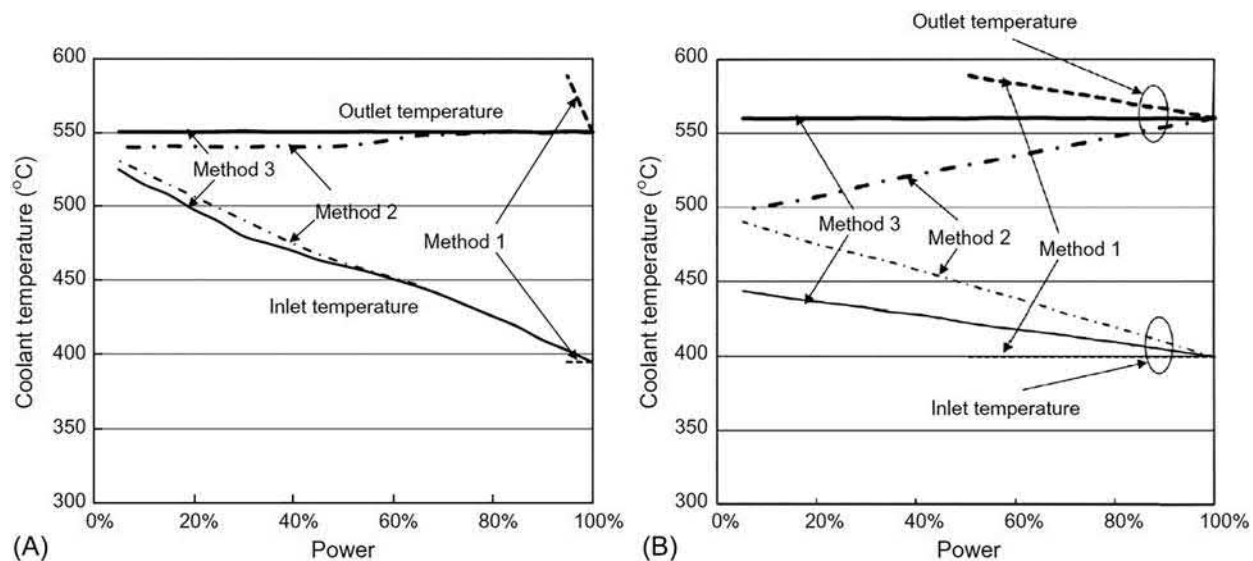


Fig. 5 Inlet and outlet coolant temperatures for different power level for power level control without control rods. (A) Sodium cooled fast CANDLE reactor (SFR). (B) ^{208}Pb cooled fast CANDLE reactor (LFR). From Sekimoto H and Nakayama S (2014) Power level control of CANDLE reactor without control rods. *Annals of Nuclear Energy* 63: 427–431.

Solving high burnup problems

The material integrity of both fuel pellets and cladding should be considered for high burnups. However, since we cannot expect the fuel pellets to remain intact, it is necessary to rely on the cladding integrity. Even in this case, we should be concerned about pellet swelling (especially for metallic fuels) since the swollen pellets may break the cladding. To mitigate this problem, the fuel smear density should be reduced. To ensure the integrity of the fuel pin by means of the cladding, both the displacement per atom (dpa) in the cladding material caused by fast neutrons and the FP gas pressure in the pin should be considered. The dpa is related to the fast neutron (> 0.1 MeV) fluence; its maximum permissible value for HT-9 is about $4.0 \times 10^{23}/\text{cm}^2$, which is much less than that expected for CANDLE burnup ($1.8 \times 10^{24}/\text{cm}^2$). The gas plenum volume necessary for FP gas is much larger than of conventional reactors. The FP gas may be better to be vented from the cladding can.

Replacement of cladding of fuel elements (recladding) and modified CANDLE are tried to solve this problem. These two methods will be described below. Both of these methods are expected to be available in the near future. However, a more ideal approach in the long term is to develop new materials that can withstand high burnup.

- Recladding

This high burnup problems could be solved by recladding after a certain burnup is attained (Hesson et al., 1963; Nagata and Sekimoto, 2007). Two recladding methods are possible. One is a recladding of a whole fuel pin (Nagata et al., 2009), and the other is a recladding of several axial elements of a fuel pin (Karim et al., 2016).

For the metallic fuel, melt and refining procedures can be applied for the axial element recladding, where the fresh fuel and discharged fuel may be added to the fuel in the recycled elements (Hoang et al., 2018).

- Modified CANDLE burnup

Since recladding removes the fuel meat from the cladding, the radiation protection problem becomes more difficult and the facility becomes more expensive and complex. This problem can be solved by the modified CANDLE burnup method (Zaki and Sekimoto, 2012), which can be performed with the fuel meat confined in the cladding.

The CANDLE reactor discharge burnup B must be selected to satisfy $B_2 \leq B \leq B_4$ in Fig. 2A. If we try to use uranium effectively, B_4 is preferred. However, the maximum permissible burnup for the material is very low. At present, in order to avoid the recladding, we must choose a low burnup, usually B_2 . This is the principle of the modified CANDLE burnup.

The fuel pin of modified CANDLE burnup is divided into many axially stacked fuel elements, and each element is covered with cladding material. At the outer boundary of the discharged fuel element ($B = B_2$), the neutron flux level is still high. Therefore, it is better to put the fresh fuel element adjacent to the discharged fuel element for increasing neutron economy. A typical modified CANDLE burnup scheme is shown in Fig. 6. Considering the axial temperature distribution, it may be better to turn it upside down. The obtained discharge burnup for a typical modified CANDLE reactor is about 26% (Zaki and Sekimoto, 2012).

At present, the achievable peak burnup value is not clear. At EBR-II, X425 lead test with U-Pu-Zr ternary fuel had achieved 19.3% burnup and the X435 Mk-III driver qualification test had achieved 19.9% burnup when the EBR-II was shut down permanently in 1994. There was no indication these tests needed to be terminated. Much higher burnup levels might well have been achieved if irradiation had been allowed to continue (Till and Chang, 2011).

Power density flattening and core height shortening

To enhance the economic performance, power density flattening is important. Especially coolant core-outlet temperature flattening is very important. It is also important to shorten the core height, since it increases the power density and reduces the coolant pressure drop in the core. The power density flattening usually increases the criticality, since core surface-to-volume ratio becomes smaller.

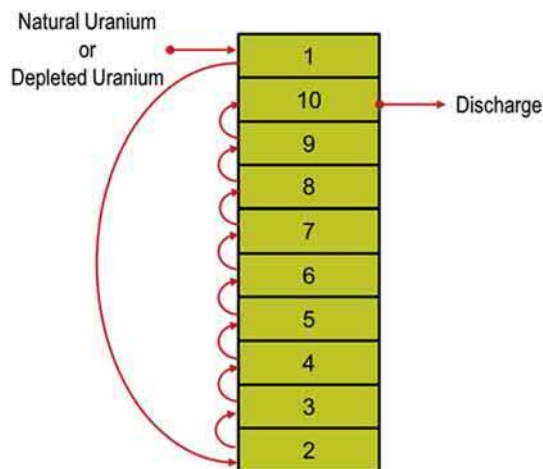


Fig. 6 Fuel element shuffling scheme for modified CANDLE.

- MOTTO cycle for core height shortening

The axial location of the burning region moves more rapidly for the higher power density. The power density at the radially central position is higher than at the peripheral position. If there is no neutron diffusion coupling between these positions, the burning region at the central position moves faster than at the peripheral position. However, the neutron diffusion coupling between these positions is considerable, and the burning regions at both positions move at the same speed but the one at the central position stays at a preceding location than at the peripheral position as shown in Fig. 7A for a typical CANDLE burning in a uniform core. This makes it necessary to increase the core height for covering whole burning region. In order to make the axial power peak position radially flat and to reduce the core height, we introduce the multichannel once-through-then-out (MOTTO) cycle (Sekimoto and Nagata, 2010; Sekimoto, 2010a). Detailed procedures for this method are given below:

For each radial position r , the axial center position of power density (centroid position) is defined as:

$$z_c(r) \equiv \frac{\int_{core-bottom}^{core-top} zP(r, z)dz}{\int_{core-bottom}^{core-top} P(r, z)dz} \quad (27)$$

where $P(r, z)$ is the power density at the position (r, z) . At BOC, the centroid surface made by connecting the centroid positions is made horizontal ($z_c(r) = \text{constant}$) as shown in Fig. 7B by adjusting the amount of spent fuel taken out from the top of each radial position. The amount of fresh fuel added to the bottom is the same as the amount of spent fuel removed at the same fuel channel. The burning region shifts downward with the progress of burning ($z_c(0) < z_c(r_1) < z_c(r_2)$ for $0 < r_1 < r_2$). Therefore, at EOC, the centroid surface is not level as shown in Fig. 7C: it is lower in the central region than the periphery. This surface is again made level as shown in Fig. 7B by adjusting the amount of discharged spent fuel and charged fresh fuel at each radial position.

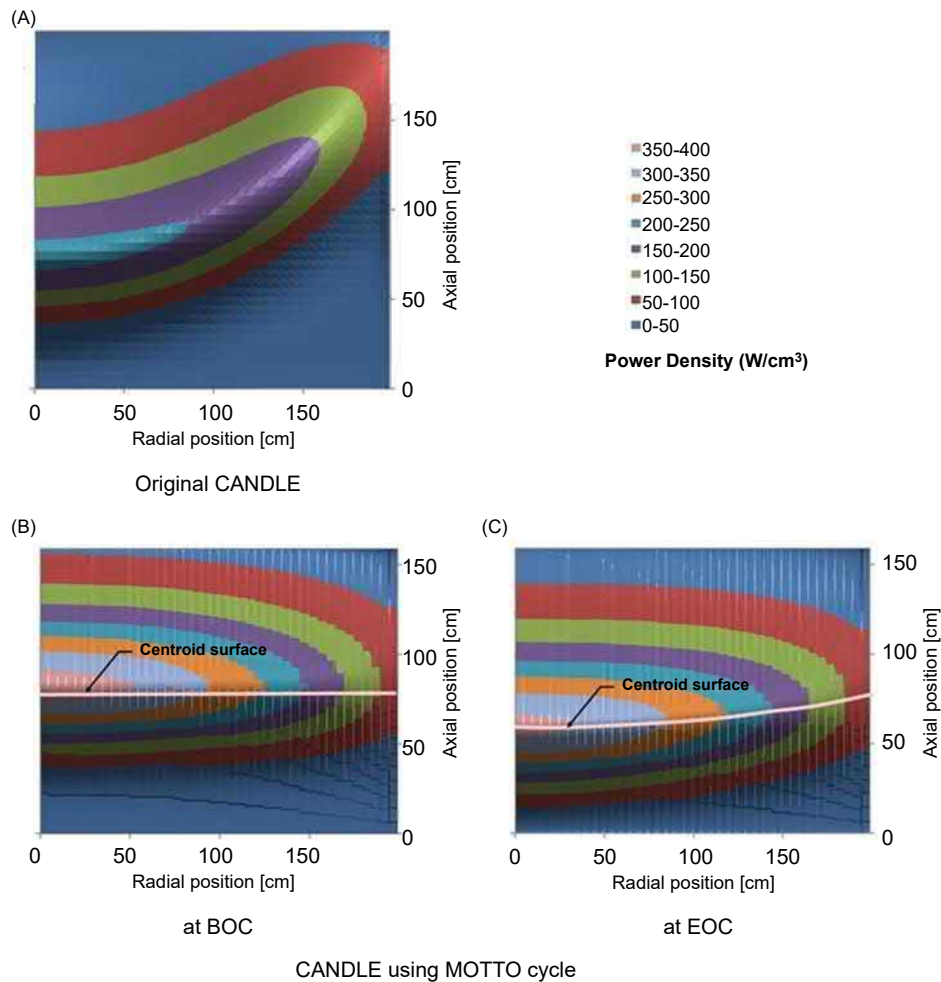


Fig. 7 Power density distributions for CANDLE core using MOTTO cycle. From Sekimoto H (2010a) *Light a CANDLE*, 2nd edn., Tokyo, Japan: CRINES, Tokyo Tech. ISBN: 978-4-905205-00-5 C3053 E. Retrieved from https://www.researchgate.net/publication/305391375_Light_a_CANDLE_2nd_ed.

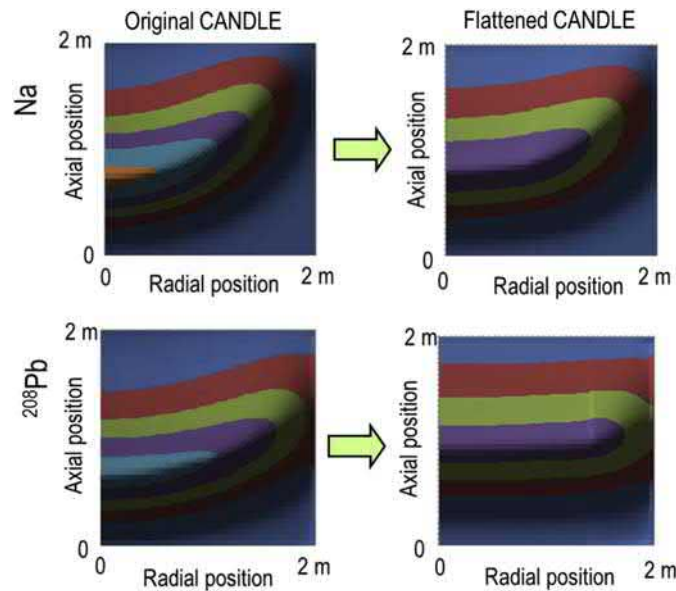


Fig. 8 r - z power density distributions for two reactors (Na cooled and ^{208}Pb cooled) before and after flattening. From Sekimoto H (2010a) *Light a CANDLE*, 2nd edn., Tokyo, Japan: CRINES, Tokyo Tech. ISBN: 978-4-905205-00-5 C3053 E. Retrieved from https://www.researchgate.net/publication/305391375_Light_a_CANDLE_2nd_ed.

- Thorium added in the inner fresh fuel core region for power density flattening

In order to flatten the radial power distribution, an amount of thorium is added to the fresh uranium fuel uniformly in the inner core region (Sekimoto, 2010a; Okawa et al., 2012). For only uranium fresh fuel (original CANDLE), as already mentioned, the neutron flux has a peak on the core axis and neutrons flow from the core axial region to the periphery. If thorium is added to the inner part of the fresh fuel region, the flux peak will decrease since the η value of ^{233}U produced from thorium is less than that of ^{239}Pu produced from ^{238}U . The thorium addition rates in the fresh fuel inner core region, where thorium-to-uranium ratio is constant over this region, and the boundary position between the inner and outer core regions are adjusted to flatten the radial power density distribution in the inner core. The uranium density in the outer core can be simultaneously adjusted to make the power density distribution continuous at the boundary. Core criticality should be maintained during these processes. In this way, the radial power density distribution in the inner core can be flattened.

Fig. 8 shows the obtained power density distributions for sodium-cooled and ^{208}Pb -cooled fast reactors with metallic fuel (Sekimoto, 2010a); this figure also shows those for the original cores for comparison. The number density ratio of thorium to uranium is 22:78 for sodium cooling core and 37:63 for ^{208}Pb cooling core. The thickness of the outer core is 120 cm for sodium cooling core and 60 cm for ^{208}Pb cooling core. From this, the radial power peaking factor for ^{208}Pb cooling core is 1.063 very close to unity.

How CANDLE reactors satisfy the requirements for future ideal nuclear energy system

Section “A future ideal nuclear energy system” describes the problems imposed on nuclear energy. This section describes how the CANDLE reactor solves these problems (Sekimoto, 2010b).

Resource

The average burnup of spent fuel is about 40% (400 MWd/tHM); in other words, 40% of natural uranium burns up without enrichment or reprocessing.

The present once-through fuel cycle of 4% enriched uranium in LWRs burns up about 5% of the inserted fuel. When tail (depleted uranium) enrichment is 0.1%, the amount of the product (4% enriched uranium) is 16% of the feed natural uranium. The burnup of 5% of this product means the utilization of 0.8% ($=0.16 \times 0.05$) of original natural uranium. Remaining 84% of the original natural uranium is the depleted uranium. If this depleted uranium is utilized as fuel for a CANDLE reactor, where the average discharge burnup is about 40%, about 34% ($=0.84 \times 0.4$) of the original natural uranium will be utilized. Therefore, if an LWR has already generated X Joules of energy, a CANDLE reactor can produce about $42.5(=34/0.8)X$ Joules from the depleted uranium stored at an enrichment facility for LWR fuel.

The following scenario is conceivable. If LWRs produce energy for 50 years and the nuclear energy production rate does not change in the future, we can produce energy for more than 2000 years by using CANDLE reactors with the depleted uranium. Thus, it is not necessary to mine any more uranium ore, and reprocessing facilities are not required.

Nuclear proliferation and physical protection

The enrichment and reprocessing are the most important technologies for the production of atomic bombs. CANDLE reactor can be operated without enrichment or reprocessing forever, once it starts, if only natural or depleted uranium is available. Therefore, CANDLE reactor shows excellent features on both non-proliferation and physical protection.

Safety

The best way for increasing safety features of nuclear reactor is considered to reduce both frequency of undesired events and consequence of severe accidents.

The frequency of undesired events for CANDLE reactor is reduced by many ways as follows. Firstly, the excess burnup reactivity becomes zero, and the shim control rods are not required for CANDLE burning. Therefore, the reactor becomes free from reactivity-induced accidents at operating condition. Secondly, the number density distribution of each nuclide does not change due to fuel burnup. Therefore, the reactor characteristics such as power peaking and power coefficient of reactivity do not change due to burnup. Therefore, the estimation of core condition becomes very reliable. The reactor operation strategy remains unchanged for different burnup stage. Thirdly, since the radial power profile does not change due to burnup, the required flow distribution rate for each coolant channel does not change. Therefore, the orifice control for each channel due to fuel burnup is not required. The operational mistakes are avoided.

The most severe accident of fast reactors is considered CDA. The consequences and possibility of re-criticality accidents at CDA is considerably reduced for CANDLE reactors, since the control rods are not inserted in the core under operating condition, and coolant amount in the core is small.

Reprocessing also has severe problems with regard to the safety of ordinary fast reactors. The CANDLE reactor does not require reprocessing, so there is no such problem.

Waste

The present LWR performs the average discharge burnup of about 5% of the inserted fuel of 4% enriched uranium. On the other hand the average burnup of the spent fuel for CANDLE reactor is about 40%. It is eight times as high as for LWR. Therefore, the spent fuel amount per produced energy is reduced to be one-eighth of once-through cycle LWR.

Reprocessing of discharged fuel and vitrification may reduce the volume of high level wastes, but the total volume of radioactive wastes increases. The once-through fuel cycle system of CANDLE reactor reduces the total volume of radioactive wastes.

The amount of actinides is decreased since they are stored in the core much longer than conventional reactors and fissioned in a considerable amount during this period.

We can use the depleted uranium. The wastes from uranium mining do not appear.

Economy

The cost of nuclear reactor consists of capital, fuel, and O&M (operation and maintenance).

We can expect low O&M cost, since CANDLE reactor is simple.

We can also expect low fuel cycle cost, since the fresh fuel is only depleted uranium or natural uranium, and the reprocessing of discharged fuel is not required.

However, since coolant channel space is smaller than conventional reactors, the core cooling performance is poor. It may result in lower average power density. The low average power density deteriorates strongly its economic performance. Nevertheless, this deterioration can be significantly mitigated by the blanket-less design, no space for control absorbers, short core height and radially flat power distribution (see section **"Power density flattening and core height shortening"**).

See Also: Introduction to Breed and Burn Reactors; Traveling Wave Reactor (TWR®).

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A Seed-Driven Breed-and-Burn Blanket Core Concept

E. Greenspan, Department of Nuclear Engineering, University of California, Berkeley, CA, United States

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Glossary

BOEC/EOEC Beginning/End Of Equilibrium Cycle.

Breeding Converting a fertile isotope into a fissile one by capturing a neutron. Usually the fertile isotope ^{238}U to the fissile isotope ^{239}Pu . Also the fertile isotope ^{232}Th (natural thorium) to the fissile isotope ^{233}U (Shusterman, 2021).

Burnup A measure of the amount of fission energy released from a given quantity of fuel. One unit used for such a measure is FIMA = Fraction (usually given in %) of Initial Heavy Metal Atoms that has fissioned. A more wide-spread unit is MWd kg^{-1} ($= \text{GWd ton}^{-1}$) that measures the amount of thermal energy released from unit weight of the initially loaded HM.

Cladding The structural material that separates the nuclear fuel from the coolant. In fast reactors it is usually made of steel. In light water reactors it is usually made of an alloy of Zirconium (Porter, 2021).

Conversion Ratio (CR) $[\text{amount of TRU in the discharged fuel} - (\text{amount of TRU loaded into core} - \text{amount of TRU fissioned})] / [\text{amount of TRU fissioned}]$. Quantities are for equilibrium cycle.

Neutron Flux Number of neutrons crossing a unit area per second. Reaction rates, such as the fission rate, are proportional to the neutron flux.

Neutron Flux Spectrum The neutron flux energy dependence; i.e., the neutron flux distribution as a function of the neutron energy. Also referred to as the Neutron Spectrum.

Neutron Leakage Probability Fraction of fission-born neutrons that leak-out from the core or from a specific region of the core (such as from the Seed).

Passive Safety (also referred to as Inherent Safety) (Rempe, 2021) A reactor designed in such a way that it mitigates the effects of any mal-function by its inherent design characteristics and can eliminate fuel damage without intervention. The inherent safety features are provided by the laws of physics.

Radiation damage A measure of degradation in the mechanical properties of a structural material (or fuel) induced primarily by high energy neutrons that can displace (knock out) atoms from their original lattice location.

Reactivity and Reactivity Feedback “Reactivity” is a term used to denote the extent of the deviation in the core effective multiplication factor (usually denoted as k_{eff}) from its value at a just critical state (at which state $k_{\text{eff}} = 1.0$). A positive reactivity feedback implies that the ratio of the rate of neutron production to the rate of neutron losses (k_{eff}) is increasing due to a change in a performance characteristic such as temperature or density. One of the necessary characteristics for a reactor to be inherently safe is that the combined effects of all possible changes in reactor conditions due to a power increase will result in a negative reactivity feedback (Tsvetkov, 2021).

SFR Sodium-cooled Fast Reactor (Heidet, 2021).

TRU/HM ratio Ratio of nuclei of all fuel isotopes pertaining to elements heavier than uranium (TRansUranium or TRU) to all fuel isotopes including uranium in addition to TRU.

Introduction

Breed-and-Burn (B&B) reactors have been proposed in the past (Qvist, 2020) as an alternative mode of operation of fast breeder reactors and TerraPower™ is presently pursuing the development of commercial B&B reactors (Gilleland et al., 2008; Hejzlar, 2021). Whereas conventional fast reactors are to be fed with relatively high fissile concentration fuel and their discharged fuel is to be reprocessed and recycled (Heidet, 2021), B&B cores are to be fed with depleted or natural uranium, breed plutonium, and then fission a significant fraction of the bred plutonium in situ. It is expected that B&B reactors will improve the economics of fast reactors relative to that of conventional fast reactors (Heidet, 2021) as they require no enriched fuel (except for the initial fuel loading that will establish the chain reaction) and no fuel reprocessing and recycling. However, in order to sustain the B&B mode of operation, it is necessary to fission at least ~20% of the depleted uranium feed (Petroski et al., 2010; Heidet and Greenspan, 2012; Qvist, 2020); that is, the fuel has to reach an average burnup of approximately 20% FIMA (Fissions per Initial Metal Atom). This corresponds to a peak discharge burnup of about 30% FIMA. The corresponding peak radiation damage of the fuel cladding material will be in the vicinity of 500 DPA (Displacements Per Atom). The maximum neutron induced radiation damage that structural materials had been exposed to so far is approximately 200 DPA (Qvist, 2021). Hence, an extensive R&D effort is required to develop and certify a cladding material that can retain the fuel integrity up to approximately 500 DPA. Such a program will have to include irradiation experiments in fast spectrum together with post-irradiation analysis and may take a long time and be expensive. If thorium is to be used as fertile feed rather than depleted uranium, it is practically impossible to design a critical core with sustainable B&B mode of operation (Petroski et al., 2010; Heidet and Greenspan, 2012). Several concepts of cores for B&B reactors are described in this encyclopedia by Hejzlar (2021), Krepel and Kramer (2021), Qvist (2021), and Sekimoto (2021). The fuel of the molten chloride B&B core described by Krepel and Kramer (2021) is in a liquid state and is not subjected to the same radiation damage constraint as do the solid fuel B&B cores.

The Seed-and-Blanket (S&B) core concept was recently proposed (Greenspan, 2012) to enable to start benefiting from the B&B mode of operation without having to wait for the development of a cladding material that can be licensed for operation at radiation damage levels exceeding 200 DPA. The objective of this Chapter is to describe the S&B core concept and to summarize its attainable performance characteristics as predicted by computer simulations; no such core has ever been built. Selected characteristics of a couple of S&B core designs are compared against those of a conventional Sodium-cooled Fast Reactor (SFR).

The Seed-and-Blanket core concept

Typical Sodium-cooled Fast Reactor (SFR) cores, such as cores for the Advanced Burner Reactor (ABR) and Advanced Recycling Reactor (ARR) (Dubberley et al., 2000; Hoffman et al., 2006; Kim et al., 2009; Heidet, 2021), are designed to have a pancake shape—their height-to-diameter ratio is significantly smaller than 1.0 (Fig. 1A) with the height being typically in the vicinity of 1 m. Such cores feature relatively large probability for neutrons to leak out—typically, on the order of 20% of the fission-born neutrons leak out, mostly in the axial direction. The large neutron leakage probability enables designing the SFR to feature passive (also referred to as

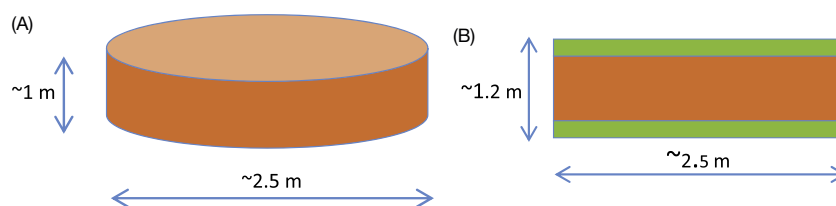


Fig. 1 (A) Highly simplified schematic layout of a conventional Advanced Burner Reactor (ABR) core. There is no blanket fuel in such cores. (B) Highly simplified schematic vertical view of an SFR core with axial depleted uranium blankets (in green). Such cores are typically designed to be breeders; that is, to make fissile fuel at a rate that is equal or larger than the rate of fissile fuel consumption.

inherent) safety by reducing the core positive reactivity feedback to coolant density reduction (causing an increase in the neutron average energy = “spectrum hardening”) and increasing the negative reactivity feedback due to the core radial expansion and fuel axial expansion resulting from increase in core coolant temperature. Passive safety requires that the net reactivity effect of all changes in the core that are induced by core power increase is negative. Besides the safety reason, there is no beneficial use of these leaking neutrons except in cores that are designed to be breeders that feature axial depleted uranium blankets as illustrated in Fig. 1B. In such cores a fraction of the neutrons that leak out from the core are captured in the blanket in the fertile isotope ^{238}U and convert it to plutonium (Pu) thus contributing to the core breeding. But in order to benefit from the blanket, the blanket fuel has to be reprocessed and the Trans-Uranium (TRU), or only Pu, need be recycled back to the core.

The S&B concept, on the other hand, features lower neutron leakage probability from the core—as low as few percent, and achieves passive safety by a unique synergism between the seed and the blanket to be elaborated below. Although many combinations of seed and blanket geometry and composition are possible, we’ll focus on one of the most promising combinations identified: The specific S&B cores addressed consist of a low Conversion Ratio (CR) seed that is fueled with TRU extracted from Light Water Reactor (LWR) Used Nuclear Fuel (UNF) and a thorium-fueled B&B blanket. The primary objective of the seed is to transmute, that is, incinerate the TRU elements that are of proliferation concern, health hazard and expensive to store. The excess neutrons that are generated in the seed from the fission of the TRU nuclei are used to “drive” the sub-critical blanket that radially surrounds the seed, also referred to as the “driver,” as schematically illustrated in Fig. 2A. Whereas the seed fuel is recycled as in a conventional ABR, the blanket operates on a once-through fuel cycle as in B&B reactors; there is no need to enrich the blanket fuel feed and to reprocess and recycle the used blanket fuel. As a result, the fuel cycle cost of the blanket is a very small fraction of that of the seed. A design objective of the S&B core is, therefore, to maximize the fraction of core power that is generated by the blanket. This objective is best obtained by designing the seed to be of an elongated annular shape that features large radial and small axial neutron leakage probabilities that is radially surrounded by thorium blankets, as schematically illustrated in Fig. 2B.

Upper-bound performance—An illustration

The leftmost column of Table 1 summarizes selected design and performance characteristics of a S&B core, denoted as S&B-1, designed to give the maximum power fraction from its thorium blanket, in comparison with those of a conventional ABR core (Hoffman et al., 2006). All the performance characteristics presented in the table pertain to the equilibrium fuel cycle; that is, to cores in which the fuel had been recycled a number of times so that the core composition at the beginning (end) of a given irradiation cycle equals to that of the following cycle. The upper-bound S&B core design referred to in Table 1 significantly deviates from design practices followed by the SFR community in several aspects: (1) The core height and pressure drop are more than twice the commonly used values. This height is typical, though, to that envisioned for B&B cores (Hejzlar et al., 2013). The sensitivity of the S&B core performance to these two design variables will be addressed in the following section. (2) The seed fuel conversion ratio (CR) is zero; its attainment requires use of TRU fuel without a fertile fuel (usually depleted U) component (TRU/HM = 0.995). Such fuels are referred to as fertile-free fuels or inert-matrix fuels. There is no experience in irradiating such a fuel in SFR to the desired

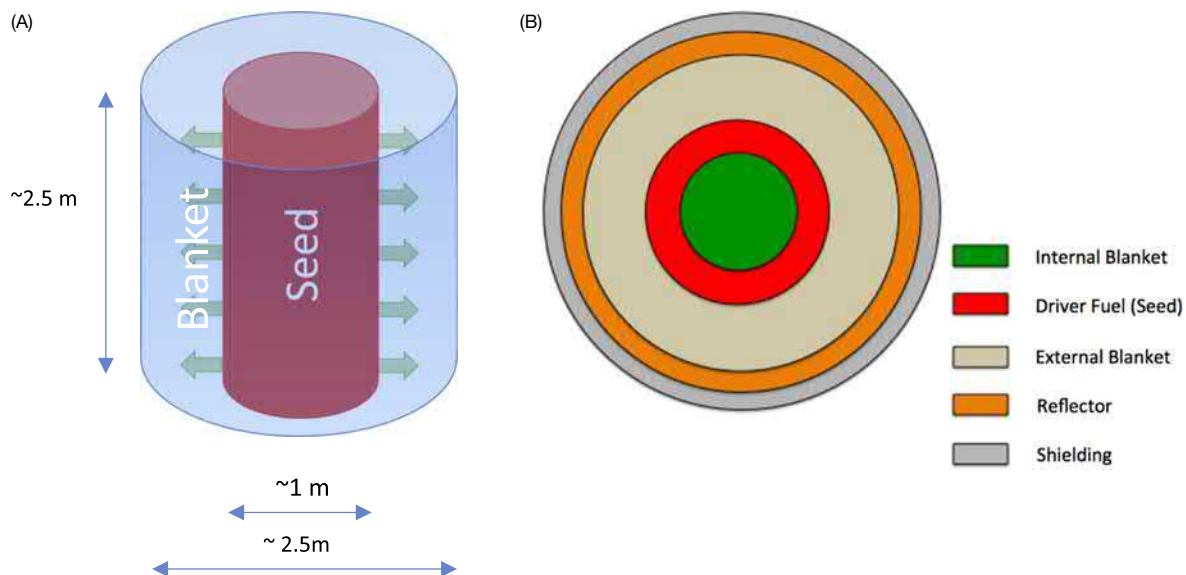


Fig. 2 (A) Schematic layout of a simplified Seed-and-Blanket core. (B) A horizontal cut through a highly simplified S&B core that features an annular seed. More than 40% of the Seed fission-born neutrons leak out in the radial direction. These leaking neutrons drive the subcritical blanket that operates in the B&B mode to generate up to ~ 50% of the core power.

Table 1 Comparison of selected design and performance characteristics of S&B and ABR (Hoffman et al., 2006) cores.

Property	S&B-1	S&B-2	ABR
	<i>Upper bound</i>	<i>Easier to implement</i>	
Core thermal power (MWth ^a)	3000	2000	1000
Fuel form (seed/blanket)	TRU-10Zr/Th	TRU-40Zr/Th	U-TRU-10Zr
TRU/HM at BOEC (wt%)	99.5	99.5	27.3/34.1/40.9 ^b
TRU CR of seed	0.0	0.0	0.5
Core height (cm)	250	120	101.6
Core effective outer diameter (cm)	270.3	270.3	218 ^c
Number of fuel assemblies			
Inner blanket	96	57	–
Seed (or “driver”)	30	71	144
Outer blanket	145	143	–
Number of batches			
Inner blanket	2	4	–
Seed (or “driver”)	2	5	6/6/7
Outer blanket	3	10	–
P/D ratio (seed/blanket)	1.406/1.104	1.190/1.117	1.224
Leakage probability from core			
Axial	3.5%	10.7%	Unspecified
Radial	3.9%	3.6%	
Cycle length (EFPD ^d)	1550	525	221
Total residence time (EFPD) (seed/blanket)	3100/7750	2625/7350	1326/1326/1547 ^b
Burnup reactivity swing (%Δk/k)	–3.60	–3.98	–2.90
Burnup reactivity swing rate (%Δk/k/EFPY ^d)	–0.85	–2.77	–4.79
Average blanket power fraction	57.7%	43.1%	–
Average discharge burnup (MWd ^a kg ^{–1})	312.4/70.2	396.6/84.5	131.9
Peak radiation damage (DPA)	185/207	189/197	– ^e
HM at BOEC (tons)	4.2/63.7	2.5/24.6	9.4
Specific power (MWth tHM ^{–1})	100.8/9.1	151.1/11.8	106.4
Coolant pressure drop in core (MPa)	0.9	0.3	0.3
TRU transmutation rate (kg EFPY ^{–1})	158.1/0.0	140.0/0.0	173.8
DU feed rate (kg EFPY ^{–1})	0.2/0.0	0.2/0.0	217.5
Thorium feed rate (kg EFPY ^{–1})	0.0/3024.2	0.0/1242.5	0.0
Trans-Th discharge rate (kg EFPY ^{–1})	0.0/223.3	0.0/102.3	0.0
Reprocessing capacity (kg (GWth ^a -EFPY) ^{–1})	494.5	348.9	2767.2
Specific reproc. capacity (kgHM per kgTRU transmuted)	3.1	2.5	14.2
Safety parameters at EOEC			
Effective delayed neutron fraction	0.0030 ± 0.0002	0.0030 ± 0.0002	0.003
Sodium void worth (Δk/k)			
Seed only	0.012 ± 0.001	0.014 ± 0.001	–
Blanket only	0.014 ± 0.001	0.005 ± 0.001	–
Full core	0.026 ± 0.001	0.018 ± 0.001	0.029
Doppler coefficient (¢ °C ^{–1})	–0.08 ± 0.02	–0.07 ± 0.02	–0.09
Axial expansion coefficient (¢ °C ^{–1})	–0.27 ± 0.03	–0.31 ± 0.03	–0.54
Radial expansion coefficient (¢ °C ^{–1})	–0.16 ± 0.02	–0.22 ± 0.02	–0.43

^aMWth = Thermal power in MW (Mega-watts); GWth = likewise in Giga-watts; MWd = energy generated per day per 1 MWth of power.^bInner/Middle/Outer core regions.^cABR core includes 19 control and safety assemblies; no such assemblies are included in the S&B cores.^dEFPD/EFPY = Effective Full Power Days/Years.^eThe peak radiation damage on the cladding of the ABR design is constrained by peak fast fluence of 4×10^{23} neutrons (of energy > 0.1 MeV) per cm².

burnup levels. For comparison, typical fuels for ABR cores feature a TRU-to-HM ratio that does not exceed ~40%; they are designed to have a CR of 0.5–0.75 (Kim et al., 2009).

The fuel of both reactors is in a metallic form and the HM is alloyed with 10 wt% Zirconium that improves the fuel performance under irradiation (Porter, 2021). The fuel rods of the S&B cores examined do not have a fission gas plenum above the fuel section, as do the fuel rods of the ABR and all conventionally designed SFR cores (Heidet, 2021). The function of the fission gas plenum is to accumulate the fission products that are gaseous in the fuel operating conditions without building too high pressure inside the fuel cladding. In typical SFR designs, the length of the fission gas plenum is comparable to the active fuel length so that it almost doubles the coolant pressure drop in the core. In the S&B cores it is assumed that the gaseous fission products are vented from the top of the fuel rod to the coolant from which they get into the inert gas volume that is located above the sodium at the top of the reactor vessel.

These gaseous fission products are then extracted by circulating the inert gas through filters. An illustration of a gaseous fission products venting system concept that allows the fission product gas to get out from the fuel rod and prevents the coolant to get into the rod can be found in [Qvist et al. \(2015\)](#).

The upper-bound S&B (S&B-1) core ([Zhang et al., 2017](#)) has only 30 fuel assemblies in the annular seed that surround 96 fuel assemblies in the inner blanket and are surrounded by 145 fuel assemblies of the outer blanket. Each fuel assembly is of a hexagonal shape of identical outer dimensions and contains 271 fuel rods arranged in a hexagonal lattice as common in SFR core designs ([Heidet, 2021](#)). The seed and blanket fuel rod diameter and spacing between the fuel rods are optimized to meet thermal-hydraulic constraints ([Qvist and Greenspan, 2014](#); [Zhang et al., 2017](#)) while offering the following design objectives: (1) maximum fraction of fission neutrons born in the seed leak out in the radial direction and (2) providing a critical core throughout the cycle. With 47% of the fission neutrons born in the seed leaking out to the thorium blankets, the blanket generates 57.7% of the core total power!

This extremely high leakage probability from the seed to the blanket is a result of using (1) the maximum possible TRU concentration in the seed fuel—enabling to maximize the fraction of fission-born neutrons to be excess neutrons; that is neutrons not required for maintaining the fission chain reaction; (2) the very small annular seed width of only ~ 16 cm, combined with the relatively large seed fuel lattice Pitch-to-Diameter (P/D) ratio¹ that enable most of the excess neutrons to leak from the seed to the inner and outer blankets. In practice, the seed is not a perfect annulus because it has to be made out of hexagonal fuel assemblies of the same dimensions as the blanket fuel assemblies. The width of the S&B core fuel assembly is 16.1 cm—on the order of 2 mean-free-paths (mfp) for fission neutrons collision. The mfp is the distance a neutron will travel, on average, from the location of its emission to its collision with one of the seed constituents. A fission-born neutron has to undergo, on average, many collisions before it is absorbed.

The following subsections discuss unique selected performance characteristics of the S&B-1 core and compares some of which with those of the reference ABR core. A comparison of an easier to implement S&B core, to be denoted as S&B-2, with the S&B-1 core and with the ABR core is provided in the following section.

Small TRU fuel inventory

The TRU containing Heavy Metal (HM) inventory of the S&B-1 core is 4.2 tons versus 9.4 tons of the ABR core. Normalized by the rated thermal power—3000 MWth for the S&B-1 versus 1000 MWth for the ABR cores, the respective values are 1.4 tons per GWth versus 9.4 tons per GWth. Moreover, whereas in the S&B-1 core the TRU is contained in 30 fuel assemblies, in the ABR core all the 144 fuel assemblies contain TRU. However, the ratio of the total length of fuel slugs² that need be fabricated is $(30 \times 2.5 / (144 \times 1.016)) = 0.51$. Normalized by the rated power this ratio becomes 0.17. As the fabrication and handling of TRU containing fuel is very expensive relative to the fabrication and handling of thorium or depleted uranium or natural uranium, the cost of fuel for the S&B-1 core is expected to be smaller than for the ABR.

Ultra long cycles

After reaching equilibrium composition, the S&B-1 core operates in cycles each of which lasts the equivalent of 1550 days at full power (Effective Full Power Days—EFPD). This is a factor of 7 longer than the 221 EFPD of the ABR. At the end of each cycle half of the seed fuel assemblies are replaced by new fuel assemblies; the new assemblies are made of the TRU extracted from the used seed assemblies that are being reprocessed supplemented by TRU extracted from LWR UNF. The innermost 48 blanket fuel assemblies (a blanket fuel “batch”) are removed from the core for cooling and storage; the following 48 fuel assemblies are shuffled to the innermost locations and their locations are being loaded with the innermost 48 fuel assemblies of the outer blanket. Likewise, the remaining fuel assemblies of the outer blanket are shuffled one batch inward and fresh thorium fuel assemblies are loaded into the outermost blanket locations.

In the ABR core, on the other hand, the core is divided into three radial composition zones: the inner zone has the lowest, whereas the outer zone has the highest TRU/HM ratio. This enrichment zoning flattens the radial power density distribution across the core thus enabling to maximize the core average power density and this way to minimize the core volume required for generating the rated power of 1000 MWth. What limits the ABR cycle length to 221 EFPD is, primarily, the so called “burnup reactivity swing”—the change, usually drop, in the core reactivity due to the change in the fuel composition—depletion of fissile fuel and buildup of fission products inventory. Safety considerations restrict this change to approximately 3.5% drop in the core effective multiplication factor ($\Delta k_{\text{eff}}/k_{\text{eff}}$). In the ABR core that features a CR of 0.5, k_{eff} drops with burnup pretty steeply dictating a short cycle duration and 6–7 refueling batches. At the end of each cycle, 1/6th of the fuel assemblies in the inner and central core zones and 1/7th of the fuel assemblies of the outer core zone are removed from the core (for cooling followed by reprocessing) and fresh fuel assemblies are loaded instead. It is apparent that the design of the seed of the S&B-1 core and its fuel management is much simpler than those of the ABR core; both offering a comparable TRU transmutation capability.

The remarkable difference in the cycle length of the two cores being examined is one outcome of the unique synergism between the seed and the blanket. Even though the reactivity ($\Delta k/k$) of the seed drops with burnup steeper than that of the ABR core (because

¹P/D = Pitch/Diameter. The Pitch is the distance between the center of neighboring fuel rods while the Diameter is the fuel rod cladding outer diameter.

²The fuel rod is made of few cm long cylindrical fuel slugs inserted into a steel tube (“the cladding”) and supported from the top by a spring ([Young, 2020](#); [Abram, 2021](#)).

its CR is smaller than that of the ABR core: 0 versus 0.5), the reactivity of the thorium blanket keeps increasing with burnup due to the conversion of thorium to the fissile isotope ^{233}U so that the total change in the S&B-1 core reactivity is much flatter with burnup than that of the ABR core. The longer cycles will enable to operate the S&B cores with a higher capacity factor that is expected to reduce the operation-and-maintenance cost component of the generated electricity.

It is possible to design ABR cores to have longer cycles; the minimum desirable cycle length is 1 year; but this requires to increase the ABR core CR from 0.5 to 0.6 or higher. A higher CR implies a lower transmutation rate per unit power generated implying a smaller specific transmutation capacity.

Very small reprocessing capacity

Even though the rate of TRU transmutation is nearly the same in the two cores compared, the required fuel reprocessing capacity per unit energy generated by the S&B-1 core is smaller than 1/5th that for the ABR core! This remarkable reduction in the required reprocessing capacity will significantly reduce the S&B reactor fuel cycle cost and reduce the investment in the infrastructure required for supporting a fleet of SFRs (Zhang et al., 2018a).

Another useful measure of the required reprocessing capacity is the specific reprocessing capacity—Kilograms of TRU containing fuel that need be reprocessed per Kilogram of TRU transmuted (a primary objective of the ABR and of the seed). The specific reprocessing capacity of the S&B-1 core is nearly 1/5th that of the reference ABR.

What makes the reprocessing capacity per energy generated by the S&B-1 core so much lower is a combination of higher seed discharge burnup—312 MWd kg^{-1} versus 132 MWd kg^{-1} for the ABR, and the large contribution of the blanket—57.7%, to the core power. The seed fuel can reach such a high burnup without exceeding the presently acceptable radiation damage constraint due, primarily, to its very high fissile (and TRU) content that implies a lower flux amplitude for a given specific power—fission power per given HM mass. The radiation damage rate is proportional to the neutron flux amplitude but also depends on the neutron spectrum—the softer the spectrum, the lower is the radiation-damage rate. The neutron spectrum in the seed is softer than in the ABR core due, primarily, to the larger coolant volume fraction (larger fuel rod pitch-to-diameter (P/D) ratio) and high leakage probability of fission neutrons from the seed.

Use of inert-matrix transmutation fuel

The S&B seed fuel is free of fertile fuel; it is an “inert-matrix fuel.” Such fuels are usually considered for so called “source-driven” subcritical reactors designed for transmutation of the LWR TRU (or Pu) waste (Wigeland et al., 2014). Two types of non-fission neutron sources are being considered for such reactors: (1) Spallation neutrons that are to be generated by impinging high energy ions onto a high atomic number target such as lead or tungsten. High energy particle accelerators are required for generating the needed current of high energy ions. Such reactors are also referred to as “Accelerator-Driven Systems” or ADS (Brown et al., 2016). (2) Fusion neutrons that are to be generated by fusion devices having low “energy gain.” Such reactors, also referred to as “Fusion-Fission Hybrid Reactors,” are being considered as possible early application of the fusion technology (Stacey, 2020). Inert matrix fuel is usually considered inadequate for use in critical reactors not only because of the steep drop in the core reactivity with burnup but also because of safety concerns—such reactors tend to have a positive Doppler coefficient—an increase in reactivity as a result of an increase in the fuel temperature—unacceptable from safety considerations. In the S&B core designs reported in Table 1 the Doppler coefficient is almost as negative as in the reference ABR core that does not use inert matrix fuel. This is attributed to the large amount of thorium (fertile fuel) in the blanket that is tightly neutronicly coupled with the seed fuel, combined with the relatively high power generated by the blanket batches adjacent to the seed. One of the unique features of the S&B core concept is that it enables to use fertile-free fuel without an external neutron source. The S&B approach to use of inert-matrix fuel is likely to be significantly simpler and less expensive than use of spallation or fusion neutrons driven sub-critical reactors.

Small positive sodium reactivity feedback enabling inherently safe design

As mentioned above, conventional SFR cores are usually designed to have a high neutron leakage probability in order to reduce their positive reactivity feedback to coolant voiding below a level that enables their design to be inherently safe. Although the S&B-1 core features much lower net leakage probability than the ABR core, its positive reactivity feedback to coolant voiding is comparable, or even slightly smaller, than that of the small volume high-leakage ABR core (Table 1). This is attributed to a combination of three features: (1) the large neutron leakage probability from the annular seed to the blanket (47.7%) and its increase with seed coolant voiding (to 47.9%); (2) the neutron that leaks from the TRU seed to the thorium blanket becomes less “important”—its probability to contribute to the chain reaction diminishes; (3) the unique physics characteristics of the thorium fuel: (a) the number of fission neutrons generated per neutron absorbed in ^{233}U (the so called “ η ” value) increases with the energy of the neutron that causes fission much slower than in ^{239}Pu ; (b) the threshold for fission of ^{232}Th is at higher neutron energy than that of ^{238}U and the ^{232}Th fission probability increase with neutron energy is significantly smaller. As a result of “a” and “b,” voiding a thorium blanket results in a significantly smaller positive reactivity insertion than voiding a blanket fueled with depleted uranium.

Thorium utilization

The S&B core concept enables significant thorium resource utilization in a critical fission reactor without a need for thorium reprocessing technology. In the metallic thorium fueled S&B-1 core, approximately 7% of the fed thorium is fissioned without recycling. This thorium utilization is about 12 times the natural uranium utilization in LWRs. This high thorium utilization is possible due to the beneficial use of the excess neutrons from the seed that drive the blanket in a subcritical breed-and-burn mode. Significantly higher thorium utilization can be achieved by softening of the neutron spectrum in the blanket and/or by the development of a cladding material capable of withstanding a higher radiation damage (Zhang et al., 2018b).

More conservative S&B core design

The performance characteristics of the S&B-1 core set an upper bound on the fraction of the core power that can be generated by the thorium B&B blanket and on the cycle length that could be provided by the S&B core concept. The present section quantifies implications on the S&B core performance resulting from more conservative design practices that result in the S&B-2 design reported in the 2nd column of Table 1. The specific design examined features a core height of 120 cm instead of 250 cm, a pressure drop of 0.3 MPa rather than 0.9 MPa. Its power level was down-rated to 2000 MWth to meet thermal-hydraulic design constraints. The S&B-2 seed fuel uses TRU alloyed with 40 instead of 10 wt% Zirconium. TRU-40Zr is assumed for the seed fuel because it is supported by irradiation experiments (Hayes et al., 2015; Zhang et al., 2018b). The outer dimensions of the S&B-2 core are comparable to those of the S-PRISM core that is being offered by GE (Dubberley et al., 2000) and could fit within the S-PRISM reactor vessel.

Relative to the S&B-1 design, the S&B-2 design requires more than twice fuel assemblies for the seed and correspondingly reduced number of inner blanket seed assemblies. The axial neutron leakage probability from this core is three times higher than from the S&B-1 core. As a result of these changes, the fraction of excess seed neutrons that is available for driving the blanket is lower and, correspondingly, the fraction of core power that is generated by the blanket is lower—only 43.1%. The S&B-2 core reactivity swing is significantly higher than that of the S&B-1 core and this forces to shorten its fuel cycle to only 525 EFPD—which is still more than twice that of the reference ABR core. However, the TRU inventory of the S&B-2 core is significantly lower than that of the S&B-1 core—2.5 versus 4.2 tons, and its burnup can be noticeably higher for the same radiation damage. As a result, the reprocessing capacity as well as the specific reprocessing capacity required for the S&B-2 core are lower—only 349 (kg GWth⁻¹ EFPY⁻¹) and 2.5 (kg reprocessed/kg transmuted) versus 495 and 3.1 for the S&B-1 core. The higher seed discharge burnup as well as higher thorium utilization are due to (1) the somewhat softer neutron spectrum of seeds the fuel of which contains 40 instead of 10 wt% Zirconium, and (2) The increase in the ratio of the axially averaged—to peak power density; that is, to the flatter axial power density distribution in the shorter core; the radiation-damage constraint limits the peak rather than the average power density. The safety related characteristics of the two S&B cores are comparable.

It is concluded that even using more practical design practices the fuel cycle benefits from S&B cores are expected to be significant.

Fuel cycle characteristics

Zhang et al. (2018a) compare the fuel cycle performance of the S&B-1 reactor against the reference ABR and a conventional Pressurized Water Reactor (PWR). The fuel cycle performance is evaluated assuming a 2-stage PWR-SFR energy systems for the fast reactors (SFR: ABR and S&B) and a once-through fuel cycle for the PWRs. In the 2-stage energy system it is assumed that the ratio of the installed capacity of the SFR and PWR is such that the transmutation rate of TRU from the PWR UNF equals the TRU production rate in the PWR. Other study assumptions, including the cost of various fuel cycle activities, can be found in Zhang et al. (2018a).

Fuel cycle cost

Fig. 3 compares the fuel cycle costs of the PWR-ABR, PWR-S&B, and PWR systems. As the fuel reprocessing capacity of the S&B core is only about one-fifth that of the equal power ABR, the fuel cycle cost of the PWR-S&B system is lower than that of the PWR-ABR system; it is even lower than that of present PWRs. There is significant uncertainty in the fuel cycle costs reported in Fig. 3 due to uncertainties in the unit cost assumptions used by Zhang et al. (2018a).

Due to the significantly longer fuel cycle, the capacity factor of the S&B-1 reactor is expected to be about 10% higher than that of the reference ABR. The higher capacity factor is expected to result in lower Operation and Maintenance (O&M) and capital cost components of the S&B reactor cost of electricity (Rothwell, 2021). The capacity factor benefits of the S&B-2 reactor will be significantly smaller due to its shorter cycle length.

Waste characteristics

Fig. 4 compares the specific radioactivity—in Curies (Ci) per unit of electricity generated (in GWe EFPY) of the three energy systems at short-term (10 years) and long-term (100,000 years) after the fuel is discharged from the core.

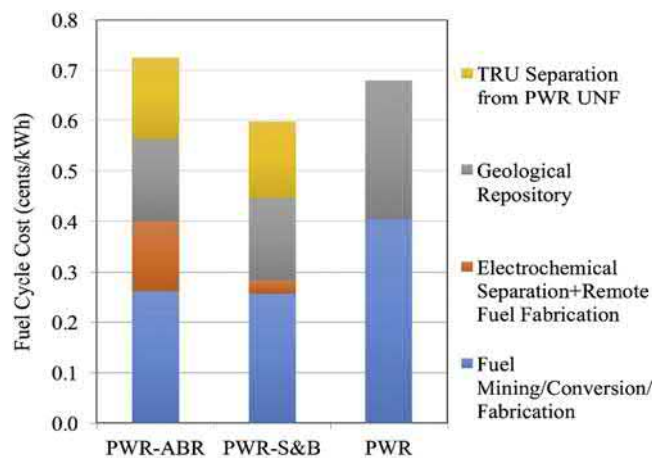


Fig. 3 Fuel cycle cost of PWR-ABR, PWR-S&B, and PWR.

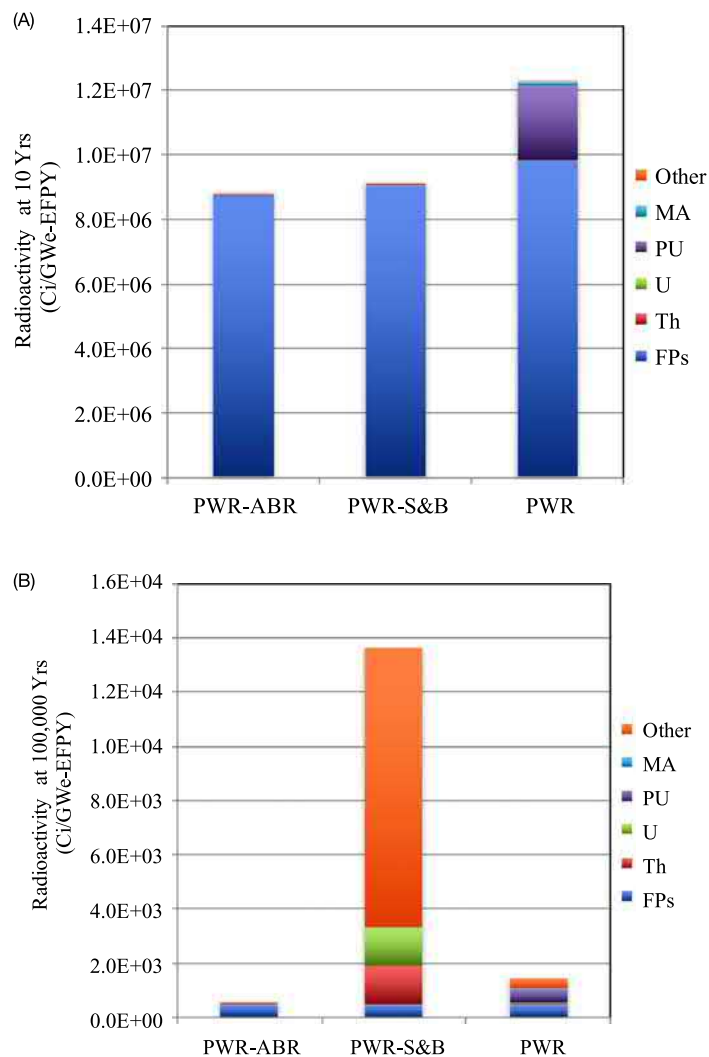


Fig. 4 Radioactivity of UNF + HLW from PWR-ABR, PWR-S&B, and PWR energy systems at 10 years (A) and 100,000 (B) years after fuel discharge. $1 \text{ Ci} = 3.7 \times 10^{10}$ decays per second.

The higher short-term radioactivity of the Fission Products (FPs) from the PWR (Fig. 4A) is mainly attributed to the lower thermal efficiency compared with that of the fast reactors ($\sim 30\%$ versus $\sim 33\%$). In addition, ^{235}U fissions yield FPs of slightly higher radioactivity than those from ^{239}Pu fissions, but this makes a small difference. The ^{233}U FPs are slightly more radioactive than the ^{235}U FPs making the short-term radioactivity of the S&B reactor waste slightly higher than that of the ABR. The ^{233}U discharged from the S&B thorium blanket has no significant impact on the short-term radioactivity because ^{233}U has a very long half-life of 159,200 years. In the fuel discharged from the PWR the plutonium, primarily ^{241}Pu , contributes notably to the short-term radioactivity.

In the long-term, the discharged ^{233}U is the predominant contributor to the higher radioactivity of S&B relative to the reference PWR and ABR (Fig. 4B). The long-life ^{233}U decays into highly radioactive isotopes including ^{209}Pb , ^{213}Bi , ^{217}At , ^{221}Fr , ^{225}Ra , ^{225}Ac , and ^{229}Th . The discharged plutonium and minor actinides from the PWR-ABR and the PWR systems as well as the S&B seed undergo substantial decay by 100,000 years, and their decay daughters are less radioactive.

Similar conclusions regarding the short-, and long-term ranking were found (Zhang et al., 2018a) for the radiotoxicity of the waste from the three energy systems. The radiotoxicity was evaluated using a couple of measures—the inhalation radiotoxicity and the ingestion radiotoxicity. For details see Zhang et al. (2018a).

Proliferation resistance

The proliferation resistance is evaluated based on the plutonium inventory, fissile plutonium fraction, decay heat and spontaneous fission neutrons emission rate per unit weight of the recovered plutonium, $^{238}\text{Pu}/\text{Pu}$ ratio, and $^{232}\text{U}/^{233}\text{U}$ ratio. ^{238}Pu , ^{240}Pu , and ^{242}Pu have high spontaneous fission rate which can significantly reduce a weapon's yield; ^{238}Pu also has a large decay heat per unit weight that further complicates the design of an explosive device (Pellaud, 2002; Xu et al., 2005).

Due to the higher burnup, the fuel discharged from the S&B-1 seed has the lowest fissile plutonium fraction and the highest $^{238}\text{Pu}/\text{Pu}$ ratio of 6.1%: much higher than that of the ABR (4.1%) and PWR (3.0%). Therefore, the plutonium recovered from the S&B core is the least attractive for making weapons.

The ^{233}U produced in the thorium blanket of the S&B reactor is, in principle, applicable for weapon use because the critical mass of ^{233}U is close to that of ^{239}Pu whereas its spontaneous fission rate is much lower (Kang and Hippel, 2001). Nevertheless, the decay chain of the co-product ^{232}U includes ^{208}Tl which generates highly penetrating 2.6 MeV gamma rays; therefore, a contamination of ^{232}U makes ^{233}U a less desirable weapon material as it is very difficult to separate ^{232}U from ^{233}U due to their close atomic mass. After the in-growth of ^{208}Tl , the dose rate from ^{233}U containing 1 ppm ^{232}U is about the same as that from reactor-grade plutonium (IAEA, 1999; Kang and Hippel, 2001). The thorium fuel discharged from the blanket has a $^{232}\text{U}/^{233}\text{U}$ ratio of 2233 ppm, well above the contamination level that requires remote operations to extract ^{233}U on a large scale without incurring large occupational dose (Forsberg and Hopper, 1998).

Nevertheless, the uranium in the fuel discharged from the S&B core blankets will need to be safeguarded to prevent proliferation, similar to the situation for the PWR reactor-grade plutonium. However, the breed-and-burn thorium blanket has an intrinsic proliferation resistance since the ^{233}U concentration in the discharged fuel is 8%. According to Bathke et al. (2014) dilution with $> 80\%$ ^{238}U or 70% ^{232}Th makes the fissile fuel unattractive to a sub-state actor. Moreover, the fissile uranium (mainly ^{233}U) in the blanket discharged fuel can be diluted with depleted uranium to make it even harder to extract. It is also possible to increase the $^{232}\text{U}/^{233}\text{U}$ ratio in the discharged fuel by softening the spectrum in the blanket (Zhang et al., 2018b). More information regarding proliferation resistance of the nuclear fuel and fuel cycle can be found in Cheng (2021).

Phased development of B&B reactors

The radiation damage constraint assumed for the cladding of all S&B cores examined so far is 200 DPA. Improved structural materials that are capable to withstand a higher radiation dose are under development (Allen et al., 2015). A sensitivity study was conducted to quantify the improvement of the S&B core performance in response to a successful development and certification of structural materials that will enable increasing the radiation damage limit from 200 to 300 or 400 DPA. For this sensitivity analysis, the DPA value of the seed fuel is kept at 200 DPA since the discharge burnup of its fuel (approaching 400 MWd kg^{-1}) is already higher than $\sim 200 \text{ MWd kg}^{-1}$ demonstrated so far (Hayes et al., 2015). The cores examined are 250 cm tall and feature 0.9 MPa pressure drop. The seed fuel is TRU alloyed with 40 wt% Zirconium.

It was found (Zhang et al., 2017) that when the blanket fuel is discharged at a higher DPA value and, hence, higher burnup, the reactivity of the blanket fuel increases. As a result: (1) the blanket can generate a larger fraction of the core power; (2) the cycle length be extended due to a smaller burnup reactivity swing. The positive feedback to coolant voiding also decreases due to a larger thorium blanket power. At 400 DPA, the B&B blanket can generate 64.2% of the core power; the thorium utilization increases to $\sim 17\%$ FIMA—close to 30 times the natural uranium utilization in contemporary LWRs; the reprocessing capacity required to support this S&B core drops to $373.4 \text{ kg GWth}^{-1} \text{ Yr}^{-1}$ which is only 13.5% the $2767.2 \text{ kg GWth}^{-1} \text{ Yr}^{-1}$ capacity required for the reference ABR core.

Alternative fuel cycle options

A couple of alternative options for advanced fuel cycle could be enabled by the S&B core concept; they involve utilization of the fissile fuel accumulated in the thorium blanket.

One option is to recycle the ^{233}U (or all Trans-Thorium elements) bred in the S&B core blanket into PWRs. It was found (Zhang et al., 2018b) possible to reduce the SFR-to-PWR installed capacity ratio required for a sustainable PWR-SFR-PWR energy system from 0.59 for the PWR-S&B system to 0.29 for the PWR-S&B-PWR system. In this 3-stage energy system, the first stage is made of conventional enriched uranium fueled PWRs (BWRs could be used as well), the second stage are S&B reactors featuring TRU-incinerating seed and thorium B&B blanket, while the third stage are PWRs (BWRs) that use the fissile fuel bred in the S&B core blankets.

The other option is to use the fissile fuel extracted from the thorium blanket to feed, and thus enable, new promising fuel-self-sustaining thermal and epithermal nuclear energy systems that operate on the Th- ^{233}U fuel cycle. Examples are heavy-water and molten-salt reactors (Ignatiev, 2021).

For all of the above ^{233}U -based fuel cycle options, the S&B core can be viewed as an enabling device that converts TRU waste into a comparable or even larger amount of ^{233}U resource that is not available in nature.

Technology gaps

The study of the feasibility of the S&B core concept results from which are reported above devoted much of its resources to search for S&B core designs that either minimize the required capacity for fuel recycling per unit of electricity generated or per amount of TRU transmuted, or maximize the blanket fuel discharge burnup without violating the presently acceptable radiation damage and other constraints. Being a concept-feasibility-assessment the study focused on the equilibrium core performance and the analysis was done using a simplified R-Z core model with the fuel batches represented as annuli. Moreover, neither transient/accident analysis nor detailed fuel performance and economic analyses were performed. In addition, the 200 DPA used for the radiation damage constraint is inconsistent with the fast-fluence constraint of $4 \times 10^{23} \text{ n/cm}^2$ in use by Argonne National Laboratory and other fast reactor designers. Hence, technological gaps exist between the present state of knowledge of S&B core design and defendable license-able practical designs of S&B cores. Following is a brief description of these technological gaps.

Realistic core designs

Rather than using R-Z geometry to represent the core, future S&B core analyses should represent the core design assembly-by-assembly and incorporate a right number of control and safety assemblies at discrete locations. An explicit shuffling pattern will have to be used for each fuel assembly so that peak assembly power, peak burnup and peak radiation damage could be accurately determined.

Detailed safety analysis

Detailed safety analysis will have to be performed for selected S&B core designs to assure that these designs are indeed inherently safe and license-able. Whereas the feasibility study performed so far only quantified and compared reactivity coefficients of the different designs, the detailed safety analyses will have to perform time-dependent transient and accident simulations using licensed safety codes.

Approach to equilibrium

A study need be undertaken to identify a promising strategy for transitioning from beginning-of-life (BOL) to equilibrium core composition in the most cost-effective way. Among the approaches to be explored: (1) Enriching the first blanket loading with enriched uranium (or Pu or TRU) trying to match the equilibrium core geometry and power density; (2) Starting with a larger volume lower TRU concentration seed that will enable to safely operate the SFR at full power from day one and gradually build up the ^{233}U concentration and power density in the thorium blanket assemblies.

A comprehensive system analysis

A systematic comprehensive comparison needs to be made between couple of sustainable PWR-SFR energy systems: one using S&B type and the other using conventional ABR type SFRs. The comparison needs to include an estimate of the investment required for the commercialization of these energy systems and an estimate of the relative cost of electricity from the two energy systems when commercial. Also need be compared is the environmental impact of the waste coming out from these energy systems, as well as these systems proliferation resistance.

Fertile-free seed fuel

Acceptable design of TRU-40Zr fuel rods to high burnups is required. In addition to numerical fuel performance analysis, irradiation experiments and post-irradiation analyses will have to be performed to determine the acceptable discharge burnup.

Thorium-base blanket fuel

There is no sufficient experience in the fabrication, irradiation and storage of metallic thorium fuel envisioned for the blanket of the S&B cores. Much more scarce is experience in the fabrication and irradiation of thorium hydride fuel as well as of fuel containing ThN fuel kernels the use of which was found (Zhang et al., 2018b) to offer significant increase in the thorium utilization. As in the case of the seed fuel, there is a need to develop performance prediction capability for thorium-based blanket fuel.

Thorium fuel waste disposal

Large inventories of used nuclear fuel will be accumulated from the breed-and-burn blankets of the S&B cores. There is no commercial experience in storage of thorium-based UNF as well as in the disposal of such fuel. Such technologies need be developed.

Thorium fuel recycling capability

If one wishes to implement one of the applications described in the section “Alternative fuel cycle options” it will be necessary to reprocess the thorium-based blanket fuel. For such applications it will be necessary to develop commercial-scale thorium fuel reprocessing capability as well as reprocessed thorium fuel re-fabrication capability.

Summary

The Seed-and-Blanket core concept presented in this chapter is found highly promising as it enables to benefit from the breed-and-burn mode of operation without exceeding the radiation damage limits structural materials had been successfully exposed to in fast spectrum reactors. The benefits identified include the following:

- Improvement in the economic viability of SFRs due to a significant reduction in the fuel cycle cost and, possibly, an increase in the capacity factor that may be enabled by the longer cycles.

Table 2 Summary of the unique synergism between low CR seed (“driver”) and thorium B&B blanket.

<i>Cause</i>	<i>Effect</i>
Lower CR seed	A. Higher TRU/HM and larger P/D ratio B. Larger fraction (>40%) of fission neutrons leaking from the seed to the blanket C. Larger fraction of core power generated by the blanket D. Lower fast neutron flux in the seed and higher discharge burnup of the seed fuel for radiation damage of 200 DPA E. Lower fuel reprocessing and remote fabrication capacity
Thorium breed-and-burn blanket	A. No thorium reprocessing technology required B. Large reactivity gain in the blanket with burnup which significantly compensates for the reactivity loss in the seed → longer cycles and higher capacity factor C. Small positive feedback to coolant voiding—comparable or smaller than that of high leakage ABR cores D. Negative Doppler reactivity coefficient even when non-fertile fuel is charged to the seed E. Application of non-fertile fuel for the seed without a need for a spallation or fusion neutron sources F. Significant thorium resource utilization without exceeding the proven radiation damage constraint G. In case blanket fuel is reprocessed → Very effective conversion of TRU waste into ^{233}U fissile resource thus enabling new fuel-self-sustaining thorium-based energy systems
Larger internal blanket	A. Larger leakage from seed and higher fraction of the core power generated by the blanket B. Smaller burnup reactivity swing per year C. Smaller void reactivity worth D. Longer cycles; higher capacity factor E. Smaller radial power peaking factor

- Significantly smaller investment in the construction of the fuel reprocessing and recycled fuel fabrication infrastructure required to support a given capacity of SFRs and more economical transmutation of light-water reactor (LWR) TRU.
- Significant utilization of thorium without having to develop thorium fuel reprocessing capability.
- A very efficient conversion of TRU (could be only Pu) to ^{233}U thus enabling deployment of new promising energy systems that use the Th- ^{233}U fuel cycle. These applications will require reprocessing of the thorium blanket fuel.

A more comprehensive comparison between the S&B and ABR core performance is required in order to accurately quantify the claimed benefits being offered by the S&B cores. A more thorough R&D is required before it will be possible to commercialize S&B type SFRs.

The benefits from the S&B core concept outlined in this chapter are due to the unique synergism that exist between a TRU (or Pu) incinerating seed (driver) and a breed-and-burn thorium blanket. This unique synergism is summarized in [Table 2](#).

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See Also: Introduction to Breed and Burn Reactors; Traveling Wave Reactor (TWR®).

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Worldwide Status of Small Reactors Research and Technology Development

Frederik Reitsma^a and M. Hadid Subki^b, ^a Ultra Safe Nuclear Corporation (USNC-Europe), Olette, France; and ^b International Atomic Energy Agency, Vienna, Austria

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Abbreviations

BWR	Boiling Water Reactor
GIF	Generation IV International Forum
HTGR	High Temperature Gas-Cooled Reactor
HTR	High Temperature Gas-Cooled Reactor
I&C	Instrumentation and Control
LOCA	Large Loss of Coolant Accident
LWR	Light Water Reactor
MWe	Megawatt electric
MSR	Molten Salt Reactors
MWth	Megawatt thermal
NPP	Nuclear Power Plant
PHWR	Pressurized Heavy Water reactors
PWR	Pressurized Water Reactor
R&D	Research and Development
TRISO	Tri-Structural Isotropic Fuel

Introduction

In order to combat climate change and make some strides towards the Paris agreement goals ([United Nations, 2015](#)), the world will need to harness all low carbon energy sources. The specific goal is to “Holding the increase in the global average temperature to well below 2 °C above pre-industrial levels and to pursue efforts to limit the temperature increase to 1.5 °C above pre-industrial levels, recognizing that this would significantly reduce the risks and impacts of climate change.” Nuclear is currently the second largest low carbon source of electricity (after hydro) and deserves due consideration to play its part in a future sustainable energy mix. To support the reduction of greenhouse gases and the further deployment of low carbon but intermittent wind and solar generation, nuclear power – as a dispatchable low carbon source – should surely play a key role. Together with hydropower, nuclear is the only low-carbon source of energy that can replace fossil fuels for 24/7 baseload power.

The potential role of nuclear power was also addressed in the Intergovernmental Panel on Climate Change (IPCC) Special Report on Global Warming of 1.5 °C ([IPCC, 2018](#)) where nuclear need to play a substantial role if the climate targets are to be reached: “Nuclear power increases its share in most 1.5°C pathways with no or limited overshoot by 2050.” In the more recent International Conference on Climate Change and the Role of Nuclear Power ([IAEA, 2020b](#)) the need to include the SMRs to meet climate and sustainable development goals were clearly recognized. “Some Member States may not be able to integrate large nuclear projects in the immediate future, due to existing grid size, lack of domestic demand, or affordability. For these countries, SMRs could be an effective option” ([IAEA, 2020b](#)).

Electricity production from nuclear power first become a reality in 1954 (the AM-1 Obninsk Nuclear Power Plant in the Soviet Union became the first reactor connected to an electrical grid and supplied 5 MW), followed by the first commercial power operation of Calder Hall Nuclear Power Station in the United Kingdom, consisting of four 60 MWe Magnox reactors ([ICTP, 2019](#)). Since

then, hundreds of commercial power plants of various designs and different materials were deployed in countries all around the world. These reactors (and also all other experimental or prototype reactors) are generally classified by their materials, purpose, moderator, coolant, or the neutron energy spectrum. More recently, also the time of construction or the generation of the design has been used to classify the reactors.

Reactors are thus classified by coolant such as Light Water Reactors (PWR or BWR) cooled by normal water, or heavy water moderated and cooled (PHWR). Another example is the molten salt reactor (MSR), where the fissile fuel is dissolved in a molten salt that is also the coolant or heat transfer medium. Final examples include High Temperature Gas-cooled Reactors (HTGR) or fast neutron spectrum reactors cooled by either sodium or a heavy liquid metal like lead or the lead-bismuth eutecticum. Coming to Small Modular Reactors (SMRs), it must be noted that these are newer generation reactors typically rated at 300 MW electric power and having distinct features, as to be defined in the following sections. The SMRs can include any of the reactor types defined above.

There is increasing interest in SMRs and their applications especially from countries with smaller grids and less access to capital to afford large capital projects. SMRs also target the rest of the energy value chain to produce heat or to be used to support industrial processes. This will also be explored in the sections below.

What is a small modular reactor?

SMRs are newer generation reactors designed to generate electric power typically up to 300 MW, whose components and systems can be shop fabricated and then transported as modules to the sites for installation as demand arises. SMRs include both evolutionary designs, which are based on improvements to existing designs by minimizing the technological risk, and innovative designs, that incorporate radical changes in the use of materials, fuel and system configurations. They can also respectively be seen as Generation III+ and Generation-IV designs (Reitsma, 2020).

In general, at the one end of the technology readiness level, the water cooled SMRs are built onto the well-established technology of the large fleet of operating nuclear power plants, while, e.g., the MSRs at the other, are truly innovative with a limited technology base. The fast neutron spectrum and high temperature SMRs are both considered as Generation-IV reactors, but are also built upon the experience obtained in the respective prototype and commercial plants.

SMRs are under development for all principal reactor lines: water cooled reactors (Subki, 2021a), high temperature gas cooled reactors (Fütterer, 2021; Reitsma, 2021) liquid-metal, sodium (Smith, 2021) and gas-cooled reactors with fast neutron spectrum (Hatala, 2021), molten salt reactors (Fratoni, 2021), and most recently microreactors (Wallenius, 2021). The IAEA booklet on Advances in Small Modular Reactor Technology Developments (IAEA, 2020a) provides an excellent and comprehensive summary of the leading SMR designs. Most of these designs are land-based but the deployment can also be a floating nuclear power plant (Subki, 2021b) constructed on a barge.

General design characteristics

As explained above SMRs include a loose family of modern advanced power reactors with lower power output that share certain characteristics. Enhanced safety is the most important feature shared by all SMR designs. This is followed by simplicity and reduced cost to make it more affordable. It represents a nuclear option to meet the need for flexible power generation for a wider range of users and applications.

One of the initial motivations to develop SMRs was to have a smaller nuclear module that could replace aging fossil-fired units while making use of the existing infrastructure. The unit size was therefore similar sized to the older fossil fired units. It also targets cogeneration (electricity and heat) needs in remote and off-grid areas.

As smaller advanced reactors SMRs have enhanced safety margins through inherent and/or passive safety features. These will be discussed in more details later.

SMRs designs introduce simplifications by modularization and system integration. By decreasing the thermal power many systems and components can be integrated or even discarded. For example the use of natural convection will exclude the need for primary pumps.

SMRs are to be deployed as multiple modules. Modules are typically self-sustained and independent but two or more modules could be located in the same location or even same reactor building. These could be controlled from a single control room with one operator overseeing multiple reactors. Although this could reduce staff it requires a new approach for I&C systems.

SMRs also include the extensive use of modularization in construction. Some of these concepts are already being used on large NPP construction but these are typically complex, very heavy and need additional infrastructure to be added on site. For SMRs it means factory manufactured, tested and quality assurance and then transported to site by heavy truck, rail, or barge shipping. This should lead to faster construction and allow the incremental increase of capacity addition as needed.

SMRs promise much smaller sites and that the Emergency Planning Zone can be reduced, possibly to the site boundary. This could mean that SMRs could be located close to population centers and therefore the end users of its electricity and possibly district heating. Similarly, SMRs should be located next to heat users in industrial complexes.

Rationale for introducing SMRs: Deployment and market

Economic considerations and especially increased affordability are an important promise made by SMR developers that make it attractive for many more countries and potential users. The capital needs for large nuclear power stations are beyond the means of most utilities and private companies, even many smaller states. So SMRs are envisioned for niche electricity or energy markets where large reactors would not be viable. SMRs designs are more flexible and attract a wider range of potential users, be it utilities serving a smaller electricity grid, an industry complex that try and decarbonize its process heat sources, or a remote village or mining operation that currently relies on diesel generators to provide electricity at high cost.

Advantages on larger more established electricity grids have also come to the fore. SMRs can contribute to load follow operations, switching between different products and enable larger renewable penetration through hybrid nuclear/renewables energy systems (Hussain et al., 2019). A distributed generation will stabilize the grid network, reduce losses due to long transmission lines and by replacing aging fossil plants existing infrastructure can be used and social impact around these sites minimized (compared to the total closure of fossil plants). Modules can also be added as and when needed preventing price instabilities due to oversupply.

The financing of large nuclear projects is challenging. Through modularization technology, SMRs target the economics of serial production with shorter construction times. For capital incentive projects a shorter project duration and earning revenue from the first units, while constructing later units, can make a huge difference. In factory manufacturing also contribute to lower risks since delays and the assurance of quality standards are much easier in the controlled environment of a factory, compared to a construction site.

Certain families of SMRs such as HTGRs are better suited for partial or dedicated use in non-electrical applications such as providing heat for industrial processes, hydrogen production or sea-water desalination. Process heat or cogeneration results in significantly improved thermal efficiencies leading to a better return on investment. Innovative power conversion cycles with thermal storage enhance the flexibility of SMRs further as short periods of over generation (> 100%) is made possible without impact on safety while benefiting from better utilization of capital.

The marine based deployment on floating platforms brings new markets to SMRs. It opens new business models for ownership or operators, may be more flexible to be sited (many well protected harbors) and has the advantage that it is floating on the ultimate heat sink – the sea. It also has the possibility to be deployed in emergencies for humanitarian assistance to provide electricity, heat and clean water to affected communities.

Safety characteristics of SMRs

All SMRs claim enhanced safety features mostly through the use of passive safety systems and simplifications.

For LWR based SMRs options include:

- the use of natural convection as the primary flow, excluding the primary pumps and related accident scenarios.
- SMRs designs introduce simplifications by modularization and system integration also to enhance safety. For example, an integrated PWR design could include all the primary system components within a single vessel. This eliminates loop piping and external components, thus enabling compact containment and plant size and reduced cost. It further eliminates accident scenarios like a large loss of coolant accident (LOCA) since no large pipes are connecting the primary pumps with the reactor vessel. See Ingersoll (2021) for an illustration of this concept.
- Improved containment options such as passively cooled containment, concrete containment with spray system and pressure suppression containment.
- Severe accidents mitigation features such as in-vessel corium retention and hydrogen passive autocatalytic recombiner.
- Some innovative LWR SMR designs could eliminate many of the safety systems required for large NPPs. The NuScale design is one example that has eliminated many safety systems reducing the number by more than a factor of two. Due to the guard vessels used in a large reactor pool, systems such as containment spray, emergency diesel generators, safety injection system or a residual heat removal system, is no longer necessary.

General safety features include:

- The use of internal (within the vessel) control rod drive mechanisms therefore excluding control rod ejection accidents.
- Residual heat removal system (passive).
- Underground construction for enhanced security and seismic stability. It can also provide protection against external impacts including large aircraft impacts.
- Ability to survive a station blackout.
- The designs have a long grace period where no operator interaction is required, typically between 24 and 72 h, or even longer. This give ample time to evaluate the situation carefully and take any actions that may be needed to bring the reactor to a safe state.

Some of the advanced designs make use of other safety characteristics. In the case of HTR SMRs the use of TRISO coated particle fuel is the main safety feature leading to significantly improved safety, the removal of decay heat by natural means only, i.e. no melt-down. In these designs the early or large release of radionuclides are not seen as a credible event since almost all radioactivity is contained within the coated particle fuel (Reitsma, 2021).

Current status of SMR designs

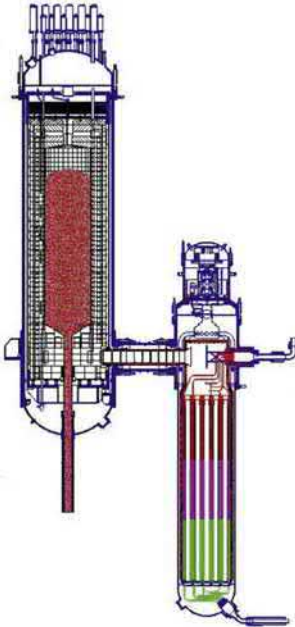
A major milestone has been reached in SMR technology deployment when the Akademik Lomonosov floating power unit has been connected to the grid and started commercial operation in May 2020. The plant consists of two modules KLT40S and deployed in the Russian Federation. KLT-40S is a compact-PWR to produce 35 MW(e) per module.

Two industrial demonstration SMRs are in an advanced stage of construction. In the People's Republic of China the HTR-PM, a high temperature gas cooled reactor with its two reactor modules is progressing well. Civil construction of HTR-PM in the Shidao Bay site is completed, the two reactor pressure vessels were installed in 2016, and the two steam-generators hoisted in-place in July 2019. It targets initial criticality in 2021 with commercial demonstration to follow.

In Argentina the CAREM reactor, an integral type PWR, is being constructed at the site of Atucha-II NPP. Despite recent project delays due to funding from the government the project is back on track. It is scheduled to start operations in 2023. CAREM-25 is an integral-PWR to produce 27 MW(e), adopting advanced features that include an integral primary cooling system using natural circulation, in-vessel hydraulic control rod drive mechanisms and fully passive safety systems.

A summary of these three forerunners can be seen in **Table 1** with some of its main characteristics highlighted. Many SMR designs make use of slightly higher enrichments predominantly to increase the cycle lengths and overcome higher neutron leakages than in larger reactors.

Table 1 SMRs under construction and their main characteristics.

	KLT-40S	HTR-PM	CAREM
			
Country	Russian Federation	China	Argentina
Reactor type	PWR	Modular pebble bed HTGR	Integral PWR/Natural Circulation
Power MW _{th}	Total 150	2 × 250	100
Electric output MW _e	2 × 35	210	30
Fuel type	UO ₂ pellet in silumin matrix	TRISO Spherical elements with coated particle fuel	UO ₂ pellet/hexagonal
Enrichment %	18.6%	8.5	3.1
Fuel cycle (months)	30–36	Online refueling	24
Distinguishing features	Floating power unit for cogeneration of heat and electricity; no onsite refueling; spent fuel take back	Inherent safety, no need for offsite emergency measures	Core heat removal by natural circulation, pressure suppression containment
Status	Akademik Lomonosov FNPP Connected to the grid in Pevek in December 2019. Entered full commercial operation in May 2020.	Finalizing construction, startup commissioning test of primary circuit in 2020. First operations expected in 2021	Under construction (as prototype). First criticality expected in 2023

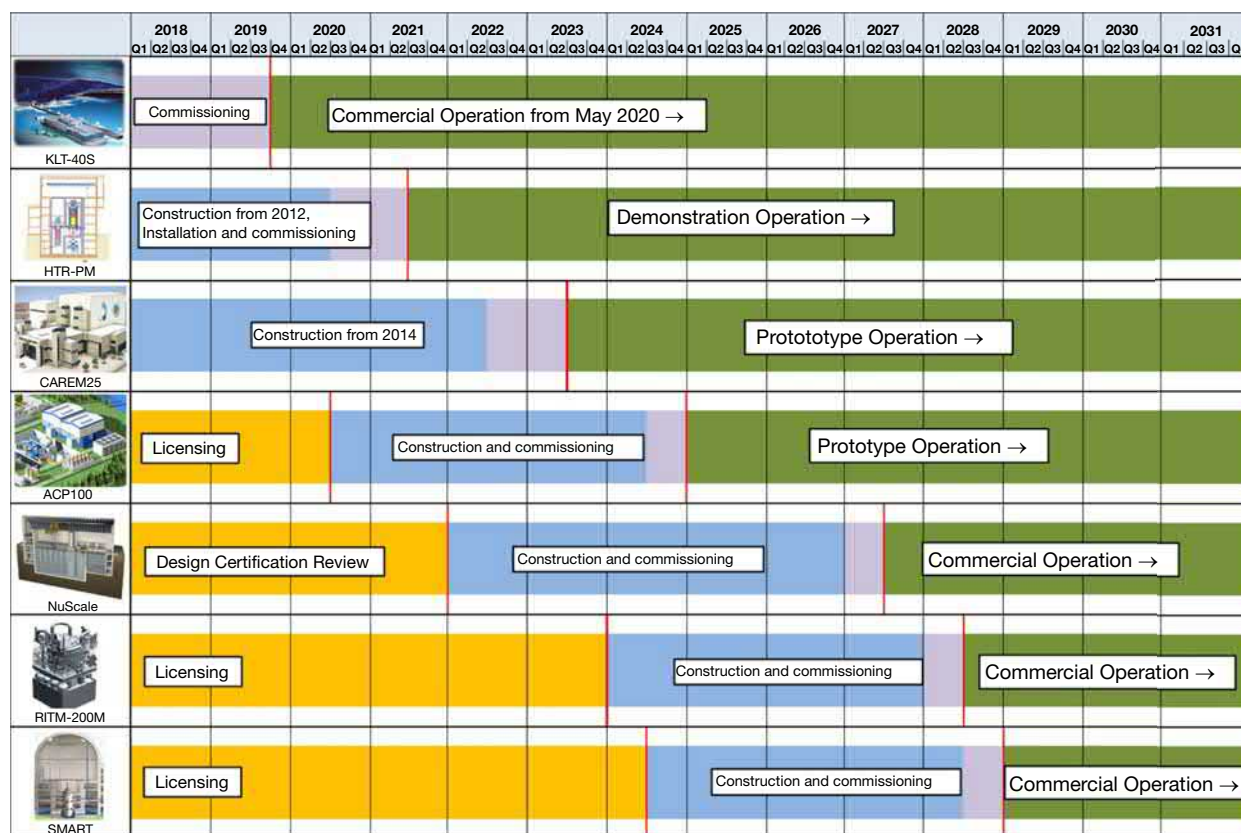


Fig. 1 Status of projected deployment timelines for some SMRs as expected in summer 2020. International Atomic Energy Agency (2020a) *Advances in Small Modular Reactor Technology Developments – 2020 Edition*, https://aris.iaea.org/Publications/SMR-Book_2020.pdf.

The deployment of several more SMRs are expected soon. In the Russian Federation it was announced that a construction license has been issued for the BREST-OD-300 reactor. It will be the world's first experimental demonstration power unit featuring a lead-cooled fast neutron reactor (WNN, 2021). The ACP100 small modular reactor at Changjiang in Hainan province, China, has also received a construction license since the preliminary safety analysis report (PSAR) was reviewed and approved at the National Nuclear Safety Administration. Construction of the demonstration unit - also referred to as the Linglong One design can commence (NBN Media, 2020). The NuScale SMR has received US design certification (WNN, 2020) in September 2020 and plan to build at a site at the Idaho National Laboratory. In Fig. 1 a scenario for possible SMR deployment is given taking the design maturity and current licensing status into account. Many of the SMR designs are targeting the 2020's to start construction of their first unit.

There are currently more than seventy (70) SMR designs under development for different application (IAEA, 2020a). As can be expected most of the designs are based on water cooled technology but the number of other designs has been increasing recently. A selection of these designs are summarized in Table 2 for water (and heavy water) reactors and in Table 3 for fast spectrum, HTGR and micro reactors.

Outlook and challenges to deployment

The first and perhaps biggest challenge are commercial demonstration. This requires the development, testing, licensing, construction and operation of a nuclear power plant. Many potential clients, especially from countries that plan to introduce nuclear power for the first time, rightly require a reference plant licensed and operating in the country of origin to limit their financial and technical risks.

The second challenge is to get to a situation where the economy of numbers come into effect. Whereas the power output of large LWR has increased over the years to benefit from economies of scale, the SMR vendor need to build many plants to bring the cost down. The vendor also needs to make a strategic decision when to establish the factory manufacturing facilities to get the benefit of reduced cost and increased quality.

The flexibility requirements on SMRs are expected to increase over time in a deregulated market and with an increased share of intermittent renewables. SMR designers aim to design a reactor module that could be flexible enough to serve many needs, ideally without any design changes. This concept is illustrated in Fig. 2.

Table 2 Water based SMRs under development with their main characteristics.

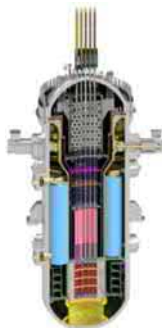
	<i>ACP100</i>	<i>RITM-200</i>	<i>NuScale</i>	<i>SMART</i>
				
Country	China	Russian Federation	United States of America	Republic of Korea and Saudi Arabia
Reactor type	Integral PWR/Forced Circulation	Integral PWR	Integral PWR/Natural Circulation	Integral PWR/Forced Circulation
Power MW_{th}	385	175	200	365
Electric output MW_e	125	50	60	107
Fuel type	UO ₂ pellet/17 × 17 square	UO ₂ pellet/hexagonal	UO ₂ pellet/17 × 17 square	UO ₂ pellet/17 × 17 square
Enrichment %	4.95	< 20	4.95	< 5
Fuel cycle (months)	24	Up to 120	24	30
Distinguishing features	Integrated reactor with tube-in-tube once through steam generator, nuclear island underground	Integral reactor, in-vessel corium retention, double containment	Unlimited coping time for core cooling without AC or DC power, water addition, or operator action	Coupling with desalination and process heat application, integrated primary system
Status	Detailed Design for construction	Six reactors built for icebreakers/Land-based NPP development	Design certification	Licensed/Standard design approval
	<i>BWRX-300</i>	<i>CANDU SMR</i>	<i>NUWARD</i>	<i>UK-SMR</i>
				
Country	United States of America and Japan	Canada	France	United Kingdom
Reactor type	Boiling Water Reactor/Natural Circulation	PHWR with/Calandria	Integral PWR/Forced Circulation	3-loop PWR/Forced Circulation
Power MW_{th}	870	960	2 × 540	1276
Electric output MW_e	280	300	2 × 170	443
Fuel type	UO ₂ pellet/10 × 10 array	37-element fuel bundle	UO ₂ pellet/17 × 17 square	UO ₂ pellet/17 × 17 square
Enrichment %	3.4–4.95	Natural U	< 5	4.95
Fuel cycle (months)	12–24	Online refueling	24	18–24
Distinguishing features	Natural circulation BWR utilizing RPV isolation valves and isolation condenser system: enable dry containment and eliminate safety relief valves	Natural Uranium fuel (no enrichment); high degree of localization	Highly compact NSSS and containment, boron-free design, load follow	Modular approach facilitating rapid and cost-effective build
Status	Pre-Licensing in UK, Canada, and the United States	Conceptual	Conceptual	Conceptual, Pre-Licensing in the UK

Table 3 Non-water based SMRs under development with their main characteristics.

	<i>BREST-OD-300</i>	<i>MMR</i>	<i>Xe-100</i>	<i>Aurora</i>
				
Country	Russian Federation	United States of America	United States of America	United States
Reactor type	Liquid metal cooled fast reactor	HTGR micro-reactor	Modular HTGR	Fission battery, fast spectrum
Power MW _{th}	700	15	200	4
Electric output MW _e	300	5	82.5	1.5
Fuel type	Mixed uranium plutonium nitride	FCM™/Hexagonal	UCO TRISO/pebbles	metal UZr
Enrichment %	up to 14.5	19.75	15.5	19.75
Fuel cycle (months)	900–1500 days	20 years	Online refueling	Up to 20 years
Distinguishing features	High level of inherent safety due to natural properties of the lead, fuel, core and cooling design	No core meltdown; modular adjacent non-nuclear power conversion plant with thermal storage	Online refueling, core cannot melt and fuel damage minimized by design	Microgrid scale applications and long core life
Status	Construction license and potential start-up in 2026	Basic/Preliminary Design; Site license application /Pre-licensing engagement	Design Certification Review	Accepted combined license application by the US NRC
	<i>Integral MSR</i>	<i>Stable Salt Reactor</i>	<i>ThorCon</i>	<i>KP-FHR</i>
				
Country	Canada	United Kingdom, Canada	Indonesia	United States of America
Reactor type	Molten salt reactor	Static fueled molten salt fast reactor	Molten salt reactor	Modular, pebble bed, HTR, salt-cooled
Power MW _{th}	440	750	557	320
Electric output MW _e	195	300	250	140
Fuel type	Fluoride fuel salt	Molten salt fuel in a conventional hexl array fuel assembly	UF ₄ , ThF ₄	TRISO particles in graphite pebble matrix/pebble bed
Enrichment %	< 5	4.95	5–19.7	19.75
Fuel cycle (months)	84	150	48	Online refueling
Distinguishing features	Core-unit is replaced completely as a single unit every 7 years	Molten salt fuel in conventional fuel assemblies; burning of nuclear waste	Full passive safety, short construction time	
Status	Basic engineering in progress	Conceptual	Basic design completed	Conceptual

International Atomic Energy Agency (2020a) *Advances in Small Modular Reactor Technology Developments – 2020 Edition*, https://aris.iaea.org/Publications/SMR-Book_2020.pdf.

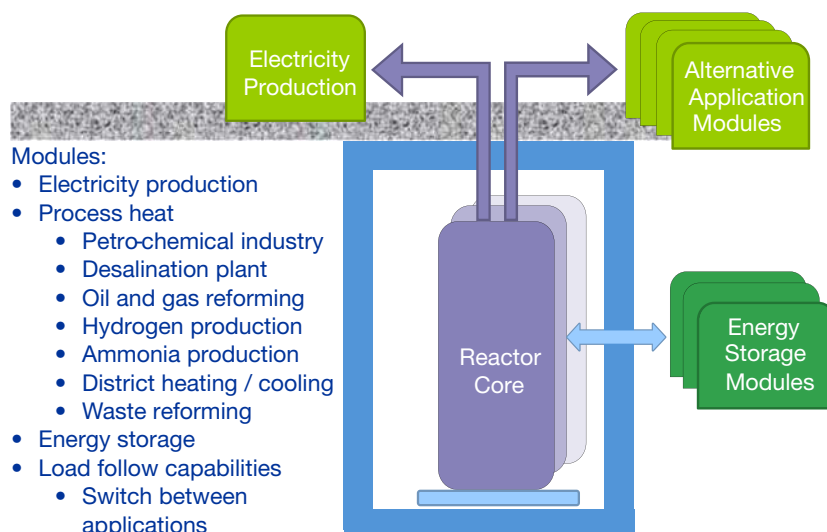


Fig. 2 Flexible utilization of SMR modules for different power and heat applications.

This takes the modular concept beyond just the reactor unit and extend it to different non-nuclear modules that can be coupled and configured to fulfil a specific customer needs, be it a heat storage module, a module providing process heat at specific conditions to one or more of multiple possible process heat application, cogeneration, or integration with renewables to perform load following. This can of course be applicable to a single or multiple reactor modules configured in this way. The huge challenge would be to come up with designs that can be licensed and then deployed in different configurations and sites in many different countries.

Excellent progress has been made in clarifying the licensing process of SMRs. Various regulators have established a specific focus on the future licensing of SMRs. The US Nuclear Regulatory Commission (NRC) has worked to developed better guidance for light water based SMRs (NRC, 2010) and more recently on reducing the licensing risks of Advance non-water cooled reactors (NRC, 2018). In Canada, the Pre-Licensing Vendor Design Review (VDR) process has attracted more than 10 SMR submissions (CNSC, 2021). In March 2019, Global First Power applied for a license to prepare a site for a small modular reactor on Atomic Energy of Canada Limited's property at the Chalk River Laboratories location. The plan is to deploy the MMR design of Ultra Safe Nuclear Corporation, a HTR micro-reactor. Regulators in many other countries are also prepared to receive submissions and license e SMRs.

See Also: Impact of Small Modular Reactors on the Acceptance of Nuclear Power by the Public, Investors, and Owners.

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Water Cooled Small Modular Reactors (Integral PWR and BWR)

M. Hadid Subki, Nuclear Power Technology Professional, Vienna, Austria

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Acronyms

ADS	Automatic Depressurization System
ASME BPV	American Society of Mechanical Engineers—Boiler and Pressure Vessel
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CEA	The French Alternative Energies and Atomic Energy Commission.
CES	Containment Enclosure Structure
CNEA	National Atomic Energy Commission of Argentina
CNNC	China National Nuclear Corporation
CNV	Containment Vessel
CRDM	Control Rod Drive Mechanism
CS	Containment Structure
DBA	Design Basis Accident
DBC	Design Basis Condition
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
EDF	Électricité de France
EDG	Emergency Diesel Generator
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Features
GHG	Green House Gas
GW	10 ⁹ Watt
I&C	Instrumentation and Control
IEA	International Energy Agency
iPWR	Integral or integrated PWR
IVR	In Vessel Retention
KAERI	Korea Atomic Energy Research Institute
LBLOCA	Large Break LOCA
LOCA	Loss of Coolant Accident
LWR	Light-water reactor

MCP Main Circulation Pump
 MPa 10^6 Pa
 MW(e) 10^6 Watt electrical
 MW(th) 10^6 Watt thermal
 NI Nuclear Island
 NSSS Nuclear Steam Supply System
 PCCS Passive Containment Cooling System
 PCV Primary Containment System
 PRHRS Passive Residual Heat Removal System
 PSFPCS Passive Spent Fuel Pool Cooling System
 PSIS Passive Safety Injection System
 PWR Pressurized Water Reactor
 R&D Research and development
 R&D&D&D Research, development, demonstration and deployment
 RCP Reactor Coolant Pump
 RCPB Reactor Coolant Pressure Boundary
 RCS Reactor Coolant System
 RIA Reactivity initiated accident
 RPV Reactor Pressure Vessel
 RVV Reactor Vent Valves
 SBO Station Black Out
 SG Steam Generator
 SLIS Small Leaks Injection System
 SMR Small and medium-sized reactor (also Small Modular Reactor)
 UO₂ Uranium-dioxide

Background

From the more than 70 SMR designs and concepts presently being pursued, at least 20 are of the water-cooled, water-moderated type. These designs are under various stages of development, demonstration, and deployment. In terms of the nuclear steam supply system (NSSS) configuration, they can be categorized into the followings four categories:

iPWRs

The term “integral” refers to integrating (entirely or partially) the primary components within the NSSS, i.e., the steam generators, pressurizer, control rod drive mechanism, and reactor coolant pumps (RCPs) are colocated in inside the reactor pressure vessel in the proximity of the core, hence eliminating the recirculation loop (Belles, 2015; Carelli, 2015). The primary design objectives of this integration are of fourfold:

Simplification: The integral configuration results in reduced nuclear island size and thus enhanced components transportability. Some designs implement horizontally or vertically mounted circulation pumps at the reactor vessel through nozzles, or internally mounted at the bottom of the reactor vessel. Other designs adopt natural circulation for the primary cooling system, thus eliminating the need of heavy recirculation pumps and reducing mechanical complexity. It should also be noted that the internal steam generator has a large heat transfer area in a compact geometry.

Enhanced safety: With only small piping connected to the reactor vessel, integral NSSS configuration enables practical elimination of loss of coolant accidents due to a large pipe break. For integral PWR designs with internal control rod drive mechanism, rod ejection accidents can be precluded. The potential of large and medium breaks, such as hot/cold leg, pressurizer surge line, and primary pump suction/discharge line breaks are eliminated by design.

Improved reliability: The simplified design and consequent enhanced robustness contribute to the improvement of the reliability of integral PWR systems.

Economic Competitiveness: A high degree of modularity of shop fabricated structures, systems and components, as well as the phased deployment of additional modules as demand increases, result in the “economics of multiple” (as oposed to the “economy of scale”) for countries and regions for which large reactors would not be viable, or which are experiencing a slower electricity growth rate.

The integral design, in which, as mentioned, all major components of the primary coolant system are collocated either entirely or partially in the single pressure vessel, enables elimination of piping of the primary cooling system. By contrast, the typical large-rated PWRs are loop systems with the primary system components, e.g., the steam generators, primary coolant pumps and pressurizer connected by piping to each other, and to the pressure vessel which houses the reactor core and the control elements.

One simplistic way to describe an iPWR relative to the current pressurized-water reactor (PWR) designs is “more water—less pipe.” Because of the integral nature of the various iPWR designs, the available water volumes in an iPWR relative to the thermal power rating of the reactors is substantially increased compared with current PWR designs. Furthermore, the size and length of the reactor coolant system (RCS) piping is significantly reduced compared with current PWR designs. This reduces the need for many of the active safety system components found in larger LWRs and generally increases the operator response time to plant upsets.

Differences from current PWR designs and commonalities with PWR designs are highlighted in this article. Except for the option of air cooling, the secondary plant in an iPWR is very similar to the current PWR reactor fleet. Therefore, secondary plant components are fairly common and will not be discussed in detail in this article.

Compact-loop PWRs

The term “compact” indicates the use of conventional loop configuration as applied in the operating medium and large-sized PWRs, but with external vessels for steam generators and pressurizer that are flanged directly to the reactor pressure vessel. This design eliminates the need of recirculation piping in the primary loop. Compact loop is a transition between traditional loop-type and the integral configuration. It is applied in marine-based nuclear power plants, e.g., floating NPP and nuclear icebreaker ships.

Traditional loop-type PWRs

Typical loop-type configuration as applied in the operating medium and large PWR nuclear power plants. A loop would consist of a steam generator, a recirculation pump, and the associated pipings (hot leg, cold leg, and cross-over leg) that connects the components with the reactor pressure vessel.

Boiling water reactors (BWRs)

Direct cycle BWRs with natural circulation, without external pumps or external pipings, adopts predesigned features from the large-sized Economic Simplified BWR, for its primary circuit, or nuclear boiler system. Passive safety features are fully adopted.

“Modular” refers to the unit assembly of the NSSS, which when coupled to a power conversion system or process heat supply system, delivers the desired energy product. The unit assembly can be assembled from one or several submodules. The desired power plant can then be developed from one or several modules as necessary to deliver the desired power rating. Importantly, the deployment of modules can also be sequenced over time both to match regional load growth and to levelize the timing of capital spending over a prescribed timeframe. Construction of the plant by assembly of factory-built elements or modules is the technique of modular construction. Although it is an integral part of the construction strategy envisioned for all SMRs, this technique is not uniquely applied to SMRs. Rather, it is now being employed for relevant construction elements of nuclear power plants of all power ratings, although the modules for large plants are considerably different in size, not typically amenable to ease of transportation and to the rapid assembly as is being proposed for SMRs.

Status of water-cooled, water-moderated SMR design and technology development

Many design organizations pursue integral PWR-type or BWR-type concepts that can form a reactor module or be combined to multi-module SMR power plants. Each module is independent of each other and operated from a common main control room. In this article, the following nine land-based water-cooled water-moderated types of SMRs are described:

- CAREM: a 30 MW(e) integral PWR type SMRs with natural circulation developed by CNEA, Argentina;
- ACP100: a 100 MW(e) integral PWR type SMRs with forced circulation developed by Nuclear Power Institute of China, China National Nuclear Corporation, China;
- NUWARD: a 170 MW(e)/module integral PWR type with forced convection for twin-module configuration SMR plant, jointly developed by EDF, CEA and the Technic Atome, France.
- SMART: a 125 MW(e) integral PWR type SMRs with forced circulation originally developed by the Korea Atomic Energy Research Institute, Republic of Korea, then the design was jointly developed with the King Abdullah City of Atomic and Renewable Energy, Saudi Arabia through shared an Intellectual Property arrangement.
- RITM-200: a 50 MW compact-loop PWR type SMRs with forced circulation already deployed for nuclear icebreakers, developed by OKBM Afrikantov, Russian Federation;
- UK-SMR: a 450 MW(e) three-loop PWR type SMRs with forced circulation, developed by Rolls-Royce, Plc., United Kingdom;
- NuScale Power Module: a 77 MW(e)/module integral PWR type SMRs with natural circulation and full passive safety features for twelve (12) module configuration NuScale power plant, developed by NuScale Power Inc., United States;

- SMR-160: a 160 MW(e) integral PWR type SMRs with natural circulation and full passive safety features, developed by Holtech International, Inc., United States.
- BWRX-300: a 300 MW(e) direct-cycle boiling water reactor (BWR) with natural circulation and full passive safety features, developed by GE Hitachi Nuclear Energy, United States.

Design description of PWR- and BWR-based SMRs

The following sub-sections are providing brief descriptions of the design characteristics and major operational and safety characteristics of the nine representative advanced designs of PWR- and BWR-based SMRs, which qualify as Generation III (advanced evolutionary) or Generation III+ (advanced passive). Detailed information, in particular in terms of the major design objectives, as well as the challenges and opportunities on the path to commercialization can be found in (IAEA, 2020a,b).

Table 1 provides a summary of the main design characteristics of the various reactor concepts that are being investigated.

CAREM

CAREM stands for the Central Argentina de Elementos Modulares developed by the National Nuclear Energy Agency of Argentina (CNEA). CAREM is an integral PWR type SMR designed as a prototype to generate 25 MW(e). CNEA started the development project for CAREM as a new generation SMR design in 1984. The project activities were intensified in 2006 as a national priority. CNEA completed the preliminary safety analyses report for CAREM in 2009 as a pre-requisite for construction. In 2011, site excavation began to prepare for civil engineering work. The first concrete pour for construction was conducted in February 2014. While the construction has experienced several delays, in the beginning 2021 the project is in an advanced stage of construction of about 70% completion on a site near the Atucha unit-2 nuclear power plant and aims for fuel loading and startup commissioning by mid-2023.

The objective of this prototype is to demonstrate the technology that will be used as the basis for commercial Nuclear Power Plant (NPP) versions with larger power ratings of up to 300 MW(e). A unique characteristic of CAREM is the fact that it will operate in natural circulation mode to drive its primary system coolant flow. This eliminates the need for reactor coolant pumps. Its integral nuclear steam supply system consists of 12 helical once-through steam generators that are installed inside the reactor vessel above the core. The hydraulic control rod drive mechanism is also located in the upper part of the reactor vessel. System pressure is controlled and maintained through self-pressurization, so no pressurizer component is required. This self-pressurization feature aims at developing thermal equilibrium of the steam and liquid phases in the reactor vessel dome region. This provides pressure stability in the reactor coolant system under normal operation, power ramps, and abnormal operating occurrences. For the power conversion system, CAREM has a similar approach to that of operating PWRs. During normal operation, pressurized feedwater is supplied by the electrically driven pumps to the 12 helical steam generators.

The **Fig. 1** shows the reactor configuration of CAREM.

CAREM's passive heat removal system is permitting it to achieve a high safety level. For core decay heat removal, CAREM adopts the pressure suppression pool, commonly used in the operating boiling water reactors for fission product scrubbing and absorption of steam blowdown energy. The CAREM suppression pool is located inside the cylindrical containment vessel that constitutes the final defence-in-depth barrier. The adopted key safety features allow mitigation of the consequences of design basis extended conditions, including severe accidents with core melt, during a 36-h grace period. The safety features ensures safe peak cladding temperature in the core in the case of station blackout (SBO) or loss of ultimate heat sink.

ACP100

ACP100 stands for Advanced Chinese PWR-type SMRs to generate 100 MW(e). It is designed as a Generation III advanced reactor by the China National Nuclear Corporation (CNNC) based on the experience in designing and operating the latest PWR technology in China. The ACP100 follows a deployment strategy of multi-purpose SMRs to supply electricity and heat to medium/small grids, islands, remote areas as well as mining industries. This SMR design is also intended to perform seawater desalination to coastal region with difficult access to potable freshwater. ACP100 is an integrated PWR design with forced convection to drive the primary coolant flow. To achieve a high level of safety, passive safety systems are adopted by including features, such as the Passive Decay Heat Removal System (PDHRS) to prevent core meltdown in the case of design basis accident (DBA), beyond DBA and mitigate postulated severe accidents. To enhance safety and security, the reactor building and spent fuel pool are located below ground level, which gives additional protection from external hazards and reduces any release of radioactive materials. To protect the reactor building against large earthquakes and flooding, segregated waterproof reactor compartments are provided.

Fig. 2 depicts the reactor configuration of ACP100.

The ACP100 reactor pressure vessel includes a vertically mounted reactor coolant pump, a magnetic force-type control rod drive mechanism, and a once-through steam generator. The pressurizer is located outside the reactor vessel. The layout of the reactor vessel and equipment is configured to promote natural circulation between the reactor core and steam generator. The 57 fuel assemblies have a conventional 17×17 square PWR fuel configuration. The reactor will be able to operate on a 24-month refueling cycle.

The safety system is composed of a number of passive safety subsystems covering core cooling, residual heat removal, containment spray cooling, and containment hydrogen control, as well as an automatic depressurization system. A large primary coolant

Table 1 PWR and BWR-based SMR designs: Main design characteristics (IAEA, 2020a,b).

	<i>CAREM</i>	<i>ACP100</i>	<i>NUWARD</i>	<i>SMART</i>	<i>RITM200-M</i>	<i>UK-SMR</i>	<i>NuScale</i>	<i>SMR-160</i>	<i>BWRX-300</i>
Country of origin	Argentina	China	France	South Korea	Russian Federation	United Kingdom	United States	United States	United States
Design organization(s)	CNEA	CNNC/NPIC	EDF, CEA, TA	KAERI and K.A.CARE (Saudi Arabia)	OKBM Afrikantov	Rolls-Royce, Plc.	NuScale Power Inc.	Holtec International	GE Hitachi Nuclear Energy, Inc.
Reactor type	Integral PWR	Integral PWR	Integral PWR	Integral PWR	Compact PWR	3-Loop PWR	Integral PWR	Integral PWR	BWR
Thermal output, MWt	100	385	540	365	175	1276	200	525	870
Electrical output, MW(e)	30	125	170	110	50	443	77	160	270–290
Primary cooling mode	Natural circulation	Forced circulation	Forced circulation	Forced circulation	Forced circulation	Forced circulation	Natural circulation	Natural circulation	Natural circulation
Prim. system pressure, MPa	12.25	15	15	15	15.7	15.5	12.8	15.5	7.2
Core inlet temp. °C	284	286.5	280	296	277	296	258	209	270
Core outlet temp. °C	326	319.5	307	323	313	327	314	321	287
Fuel type	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet
Assembly array	Hexagonal	17 × 17 Square pitch	17 × 17 Square pitch	17 × 17 Square pitch	Hexagonal	17 × 17 Square pitch	17 × 17 Square pitch	17 × 17 Square pitch	17 × 17 Square pitch
Enrichment, %	3.1	< 4.95	< 5	< 5	< 20	4.95	< 4.95	4.95 Max.	3.4 Avg.
Fuel cycle, months		24	24	36	60	18–24	24	18–24	12–24
Reactivity control	Control rods	Control rods, Gd ₂ O ₃	Control rods	Control rods, soluble boron	Control rods	Control rods, Gd ₂ O ₃	Control rods, soluble boron	Control rods	Control rods, soluble boron
Safety system	Passive	Passive	Passive	Passive	Active and passive	Active and passive	Passive	Passive	Passive
Design life, years	40	60	60	60	60	60	60	80	60
RPV height/diameter, m	11/3.2	10/3.35	13/4	18.5/6.5	8.5/3.3	11.3/4.5	17.8/3.0	15/3	26/4
Seismic design	0.25 g	0.3 g	0.25	> 0.18 g	0.3 g	> 0.3 g	0.5 g	0.3 g	0.3 g
Design status	Under construction	Detailed design	Basic design	Detailed design	Detailed design	Basic design	Certification review	Design review	Design review

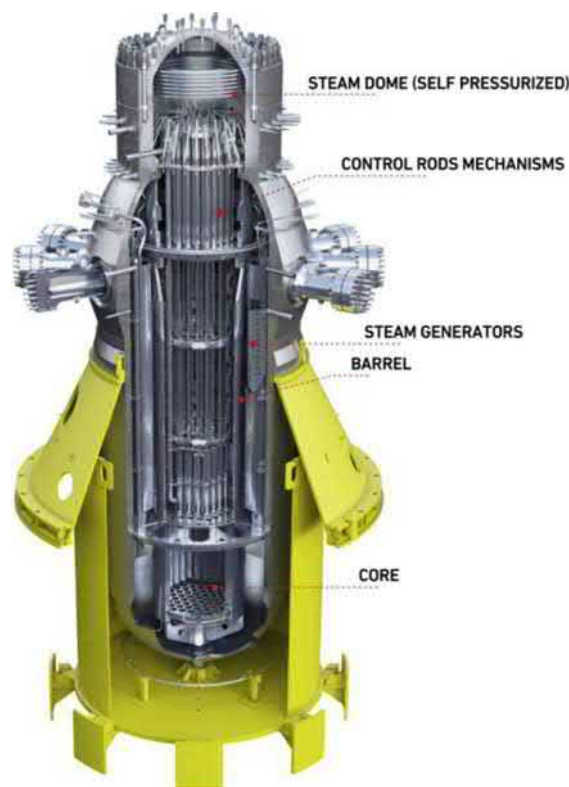


Fig. 1 Reactor configuration of CAREM developed by CNEA, Argentina.

inventory is also included as a safety feature. To protect against an SBO event, there is a 72-h DC battery system to provide power in support of all accident mitigation systems. Auxiliary power units are located inside the reactor building to recharge the batteries for up to 7 days. The ACP100's passive decay heat removal system provides cooling for 72 h without operator intervention or 14 days with water supply from the cooling pool, drained by gravity force.

The plant also has an enhanced spent fuel pool configuration to cope with a loss-of-heat sink event such as what occurred at the Fukushima Daiichi nuclear plant. Even with the pool filled with 10 years' worth of used fuel, the cooling system would be able to cope for 7 days in the case of a severe accident before it would boil dry and uncover fuel.

CNNC began an R&D program for ACP100 in June 2010 with a basic design that was approved in June 2010. At the end of 2013, a preliminary design for the ACP100 was completed that included enhanced safety features that have incorporated the lessons learned from the Fukushima Daiichi accident. In 2014, the PSAR was submitted to the IAEA Department of Nuclear Safety and Security to undertake a Generic Reactor Safety Review. The review lasted for 8 months and resulted in a set of recommendations to enhance the overall performance of the design. In June 2020, the National Nuclear Safety Administration of China (NNSA) approved the PSAR to enable Construction Start Date (CSD) in 2021 in the Fujian Province as a demonstration unit to be in operation in 2026.

NUWARD

Officially launched in 2019, NUWARD is a conceptual design of integral PWR to generate 340 MW(e) from two independent modules that enable flexible operation. This Generation III+ SMR is being developed by a French EDF-led consortium including CEA, TechnicAtome, and the Naval Group. As an integral PWR, the main components of the NSSS including compact steam generators, pressurizer and control rod drive mechanisms are installed within the Reactor Pressure Vessel (RPV) in proximity with the core. Design simplification leads to a shortened RPV that enables the NSSS to be installed in a steel containment vessel submerged in a large volume underground water-wall.

The reference core is based on a proven 17×17 fuel assemblies design used in currently operating PWRs with a shortened core height and UO_2 rods with enrichment $< 5 \text{ wt\% } ^{235}\text{U}$. Various levels of ^{235}U enrichments and burnable poisons are used in the core to achieve a boron-free design. Based on a 2-year refueling interval, fuel handling uses state-of-the-art techniques applied in the operating PWRs. The reactor coolant system of NUWARD uses a newly invented Compact Steam Generator (CSG), once-through steam generator technology derived from the plate heat exchanger concept that allows direct connection to the reactor vessel. The overall size of the reactor coolant system can, therefore, be significantly reduced.

Fig. 3 illustrates the reactor configuration of NUWARD™.

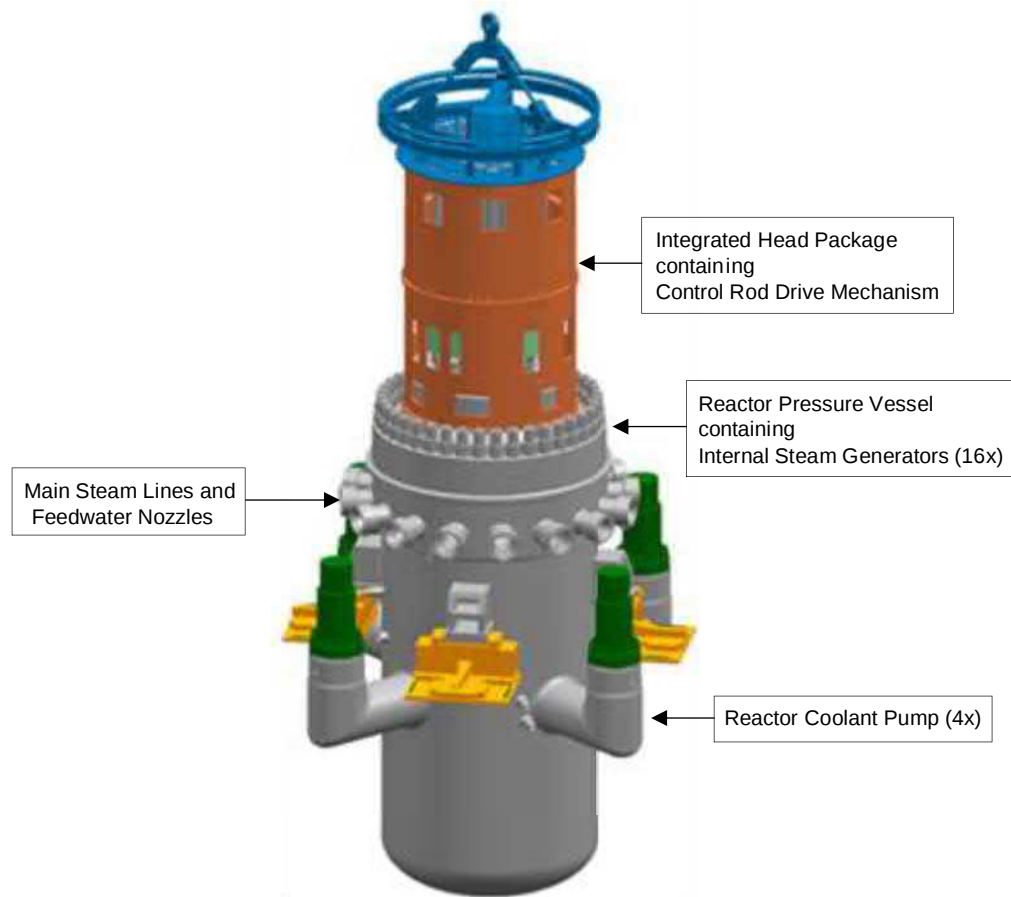


Fig. 2 Reactor configuration of ACP100 developed by NPIC/CNNC, China.

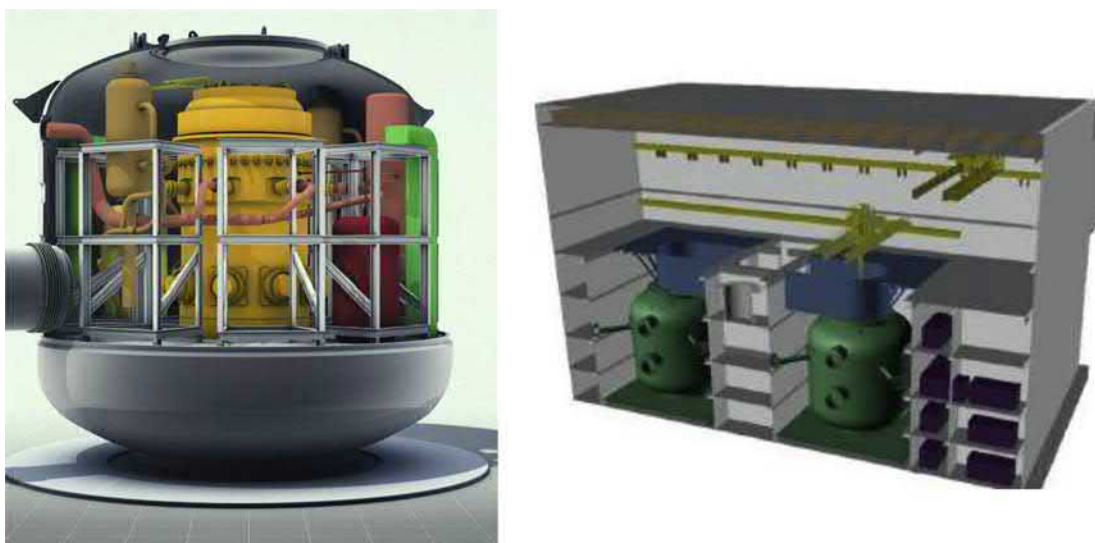


Fig. 3 Reactor configuration of NUWARD™ developed by a consortium of CEA, EDF and TechnicAtome, France.

To achieve high safety performance, NUWARD adopts passive safety systems to cope with Design Basis Conditions (DBC) without the need for any external power supply and is self-reliant on an internal ultimate heat sink for a grace period of more than 72 h. No safety classified 1E electrical power is required (except DC power to I&C). No operator action required for more than 3 days after any design basis accidents (DBA). To cope with severe accidents with core melt, in-vessel retention of the corium (IVR concept) is adopted.

The following inherent features benefit NUWARD in achieving its safety performance goals:

- Large reactor coolant inventory provides inertia in coping with power transients;
- The boron-free operation provides large and constant moderator counter-reaction and is preventing boron dilution;
- The integral NSSS architecture minimizes the pipe break size in LOCA, thus providing more time for mitigation. Internal control rod drive mechanism (CRDM) prevents reactivity accidents caused by rod-ejection;
- A submerged containment provides passive cooling for several days;
- A small core in a large RPV facilitates an in-vessel retention strategy for severe accidents with core melts.

The Nuclear island (NI) houses two independent modules and an associated fuel storage pool. The NI is designed to withstand design basis earthquakes. No heat sink outside the NI is required to ensure the safe-state for at least 72 h. The NI is self-reliant for at least this period due to the water-wall in which each containment vessel is immersed. To account for seasonal variations, the concept of “a set of reactors inside the same plant” provides the operator with a solution to adapt the maintenance schedule with priority given to the electrical grid supply needs. There is always at least one reactor of the plant in operation and supplying the grid, while another reactor may be in maintenance. As twin-module configuration, the reactors are operated from a common control-room with dedicated panels for normal operation.

SMART

The 365 MW(th) System-integrated Modular Advanced ReacTor (SMART), developed by the Korea Atomic Energy Research Institute (KAERI), is an integral PWR with a rated electrical power of 110 MW(e). SMART adopts advanced design features to enhance the safety, reliability and economics. The advanced design features and technologies implemented were verified and validated during the standard design approval review. To enhance safety and reliability, the design configuration incorporates inherent safety features and passive safety systems. Economics improvement is achieved through system simplification, component modularization, reduction of construction time and high plant availability. The advanced design features of SMART include integrated SGs, in-vessel pressurizer, and the horizontally mounted canned motor pumps. SMART is a multi-purpose application reactor for electricity production, seawater desalination, district heating, process heat for industries, and suitable for small or isolated grids. SMART has a unit output adequate to meet the demands of electricity and freshwater for a city of 100,000 inhabitants.

Fig. 4 provides the reactor configuration of SMART.

The SMART design adopts an integrated primary system, modularization and advanced passive safety system to improve the safety, reliability as well as the economics. Improvement in the economics is achieved through a system simplification, in-factory fabrication, reduction of construction time and high plant availability.

SMART has an integral reactor coolant system configuration for the NSSS that enables the elimination of large bore pipe connections resulting in the removal of the large break loss of coolant accident (LOCA) from the design basis events. The NSSS consists of the reactor core, helically coiled steam generators and a pressurizer integrated in the reactor pressure vessel (RPV). The primary cooling system is based on forced circulation during normal operation. The system has natural circulation capability for emergency condition.

The safety systems of SMART are designed as fully passive that function automatically on demand. These consist of a reactor shutdown system, a passive safety injection system (PSIS), PRHRS, a shutdown cooling system and a passive containment cooling system (PCCS). Additional safety systems include reactor overpressure protection system such as automatic depressurization system (ADS) and pressurizer safety valves, and severe accident mitigation system. As a result the reactor safety is enhanced substantially and the core damage frequency is also reduced.

As for the engineered safety system approach and configuration, SMART adopts passive safety features to cope with abnormal operating occurrences and postulated design basis accidents, and severe accidents with core melts. The passive safety system is being developed to maintain a safe shutdown condition following design basis accidents such as LOCA and non-LOCA transient events without AC power or operator actions. The passive safety system consists mainly of PSIS, PRHRS, PCCS, and the automatic depressurization system (ADS). All the active safety features are being substituted with passive versions, eliminating the necessity for emergency diesel generator (EDG) or operator actions for at least a 72 h period.

KAERI received the standard design approval from the Korean nuclear safety commission in July 2012. In September 2015, a pre-project engineering agreement was signed between the Republic of Korea and the Kingdom of Saudi Arabia for the deployment of SMART. A construction period of less than 3 years from the first concrete to fuel load is predicted. A safety enhancement program to adopt a passive safety system in SMART began in March 2012, and the testing and verification of the PRHRS and PSIS were completed at the end of 2015.

RITM-200

The RITM series reactor RITM-200 is the latest development in the Generation III + SMR line designed by the OKBM Afrikanov. It has incorporated all the proven features from its predecessors. The RITM-200 based NPP is a reference technology, currently at

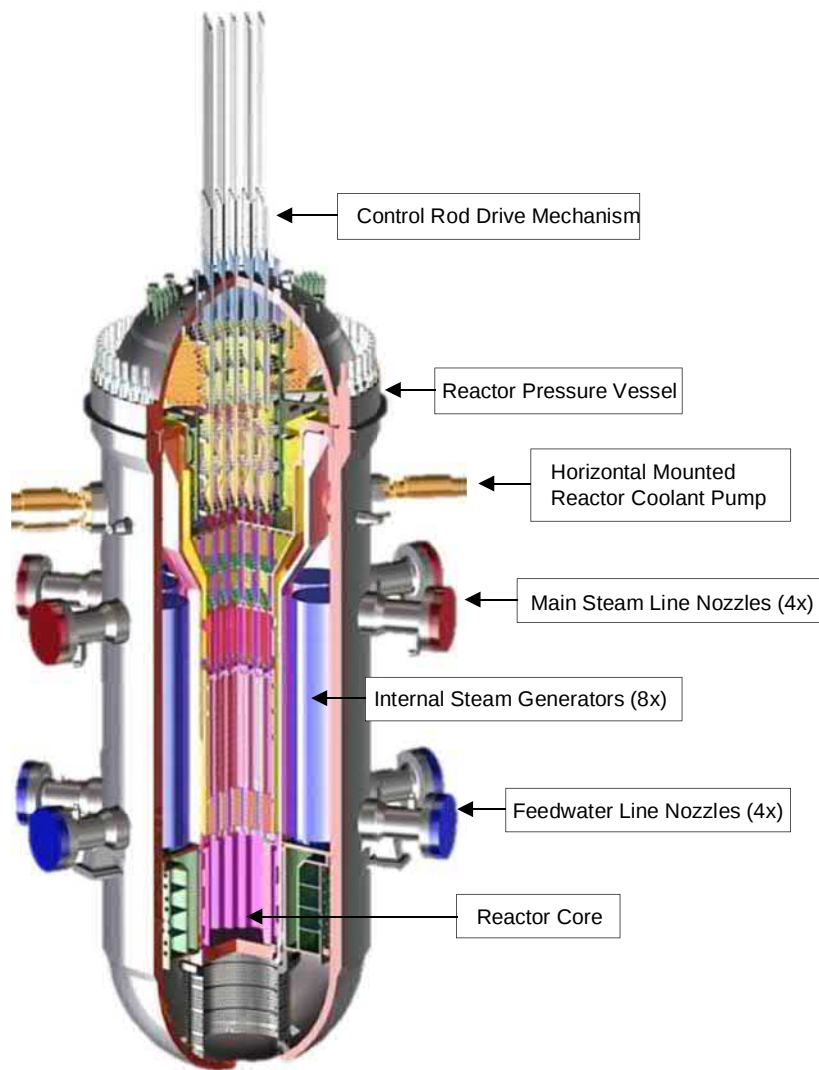


Fig. 4 Reactor configuration of SMART developed by KAERI, Republic of Korea.

a rather advanced development stage and will be available in short/medium term for commercial deployment. RITM-200 based NPP modularity enables to provide power both for commercial/local needs (e.g., in the 100 MW thermal rating range), as well as for regional needs (300 MW). Being initially designed for icebreakers, the RITM series reactors are already responding to the key market requirements, such as small size, long refueling cycle and flexible load following capabilities (100 to 30 to 100% nominal power). Six RITM-200 reactors have been already manufactured for nuclear icebreakers. Four reactors are already installed on the Sibir and the Arktika nuclear icebreakers. Based on these factors, the RITM series reactors are the most optimized solution in terms of size and other technical and economic parameters.

The RITM-200 series reactors are land-based multi-purpose application reactors for electrical power generation, water desalination, and district or industrial heating. RITM-200 based NPPs is suitable for small, isolated, or distributed grids.

The RITM series reactors are the evolutionary development of the reactors (OK-150, OK-900, KLT-40 series) for Russian nuclear icebreakers with a total operating experience of more than 50 years (more than 400 reactor-years). With incorporation of the steam generators into the reactor pressure vessel, the reactor system and containment is very compact compared to the KLT-40. RITM design makes it possible to increase electric output (40% more) and reduces the dimensions (45% less) and the mass (35% less) in comparison with KLT-40S. While integral reactor configuration almost eliminates the classic large loss-of-coolant accident (LOCA) and the other inherent features and active and passive safety systems apply concepts of diversity, redundancy, physical separation, and functional independence to achieve the necessary safety level and reliability.

The RITM nuclear steam supply system consists of the reactor core, four steam generators integrated in the reactor pressure vessel, four canned main circulation pumps (MCP), and pressurizer. The primary cooling system is based on forced circulation during normal operation and allows natural circulation in the case of emergency condition.

For the reactor core, RITM-200M adopts a low enriched cassette core similar to KLT-40S that ensures long time operation without refueling and meets international non-proliferation requirements. The core consists of 199 fuel assemblies with fuel enrichment up to 20%.

The safety concept of the RITM-200 is based on the defence-in-depth principle combined with the inherent safety features and use of passive systems. Properties of inherent safety features are intended for automatic control of power density and reactor self-shutdown, limitation of primary coolant pressure and temperature, heating rate, primary circuit depressurization scope and outflow rate, fuel damage scope, maintaining of reactor vessel integrity in severe accidents and form the image of a “passive reactor,” impervious to possible disturbances. RITM-200 optimally combines passive and active safety systems to cope with abnormal operating occurrences and design basis accidents. More specifically:

- Passive pressure reduction and cooling systems have been included (system reliability is confirmed by test bench);
- Pressure compensation system is divided into two independent groups to minimize size of potential coolant leak;
- Main circulation path of the primary circuit is located in a single vessel;
- Steam header of primary coolant circulation is added, which ensures safety of the plant during SG and MCP failures.

The exposure dose for the staff in normal operation and design basis accidents does not exceed 0.01% of the natural radiation limit. The public exposure dose in case of severe accident is below the value requiring protective measures.

The basic architectural design of the RITM-200M land-based NPP is similar to conventional large NPPs. However, it is housed within a relatively small footprint and includes nuclear island, spent fuel pool, turbine island, auxiliary buildings for water treatment, maintenance, and switchyard connections. The RITM-200 design was developed in conformity with Russian laws, norms and rules for nuclear power plants and safety principles developed by the world community, as well as based on IAEA recommendations.

Fig. 5 shows the RITM-200M reactor configuration.

UK-SMR

Rolls-Royce, Plc. is undertaking SMR design and technology development for a Gen III+ three-loop, close-coupled PWR that provides a power output of 443 MW(e) from 1276 MW(th) using industry standard UO_2 fuel. Coolant is circulated via three centrifugal RCPs to three corresponding vertical u-tube SGs. The design includes multiple active and passive safety systems, each with

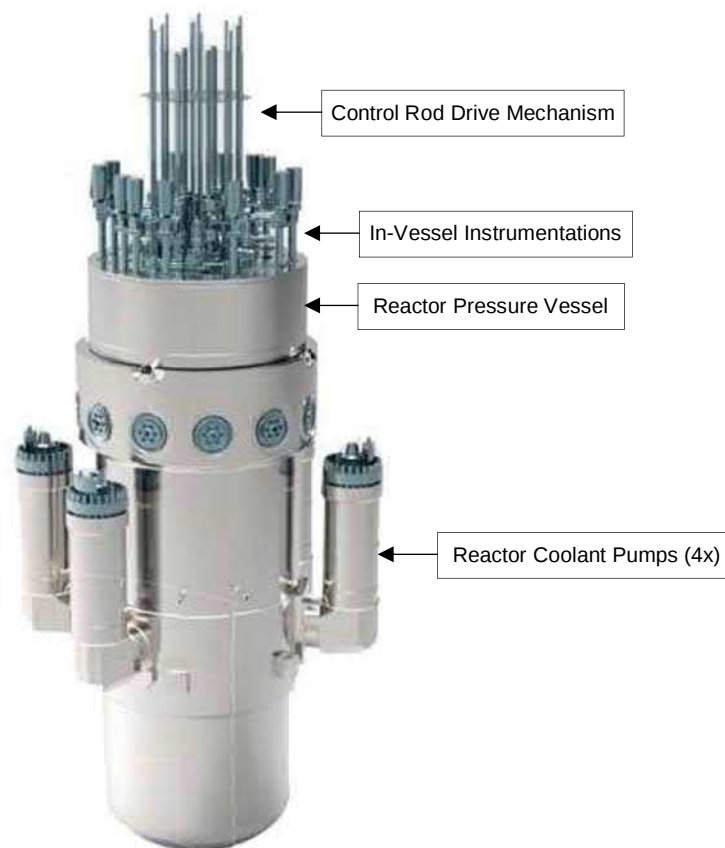


Fig. 5 Reactor configuration of RITM-200M designed by OKBM Afrikantov, Russian Federation.

substantial internal redundancy. Rapid, certain and repeatable build is enhanced through site layout optimization and maximizing modular build, standardization and commoditization. The power station is designed for installation on an extensive range of inland and coastal sites, across a wide range of soil conditions. This flexibility is enabled through design features such as seismic isolation for safety related areas. The three-loop PWR is located in the Nuclear Island, adjacent to the Turbine Island with the Cooling Water Pump House following. Support buildings and auxiliary services are situated within a berm that sweeps around the site and provides protection from external hazards such as tsunami or aircraft impact.

Fig. 6 reactor configuration of the UK-SMR.

The plant is capable of load following, operation on house load where required, even a black start in which the plant is capable of restoring an electric power station without relying on the external electric power transmission network. As a result of the passive nature of safety systems, the plant is not reliant on grid power for safety related functions. The UK SMR is primarily intended for electricity production; however, the design can be configured to support other heat-requiring or cogeneration applications.

The UK SMR primary circuit comprises a reactor core, containing the reactor fuel, mounted in an RPV that is closely coupled through large bore pipework to three SGs. Nuclear fission in the reactor core produces heat which is transferred to coolant in the primary loop through thermal conduction and convection. A pressurizer is connected to the primary circuit via a surge line to provide the necessary overpressure to prevent boiling in the reactor coolant circuit.

The reactor core has been sized to optimize power output within an RPV that is sized to maximize road transportability, producing 1276 MW(th). Coolant is transferred via three canned sealless centrifugal RCPs to three corresponding vertical u-tube steam generators, each rated to remove approximately 425 MW(th) during normal power operation. The RCPs are mounted, via close-coupled nozzles, to the bottom of the SG outlet head.

For all Design Basis Accidents (DBA), defence in depth is provided through the provision of robust safety measures, designed against conservative conditions, which meet the guidelines from the deterministic design basis analysis.

Multiple layers of fault prevention and protection are provided through diverse and independent active and passive systems, comprehensively ensuring UK SMR safety for design basis and design extension conditions, for all modes of operation and during all lifecycle stages.

In addition to heat removal via the closed loop SG steam and feed cycle, the Passive Decay Heat Removal (PDHR) system and the Emergency Core Cooling System (ECCS) are passive, redundant, diverse and segregated protective safety measures that provide multiple means of decay heat removal in response to faults.

All design basis Loss of Coolant Accidents (LOCA) are protectable by ECCS, with diverse protection additionally available from the Small Leak Injection System (SLIS) for smaller leaks. Control Rods (scram) and Emergency Boron Injection provide two diverse and highly reliable means of reactor shutdown. Three Safety Relief Valves are provided to protect against over pressure hazards, each fully capable of providing depressurization.

The UK SMR consortium includes among others the National Nuclear Laboratory (NNL), the Nuclear AMRC, and Rolls-Royce, Plc. The consortium aims to have the first UK SMR power station in operation around 2030 in the United Kingdom.



Fig. 6 Reactor configuration of UK-SMR developed by Rolls-Royce, Plc., United Kingdom.

NuScale power module

The NuScale Power Module (NPM) is a small, light water cooled pressurized-water reactor (PWR) developed by the NuScale Power Corporation, United States. The design has received standard design approval from the US Nuclear Regulatory Commission in September 2020. The NuScale design is a modular reactor for electricity production and non-electrical process heat applications. A NuScale nuclear power plant is scalable and can be built to accommodate a varying number of NPMs to meet a customer's energy demands. The 77 MW(e) NPM provides power in increments that can be scaled to 924 MW(e) gross in a single facility. A 12 module configuration is the current reference plant size for design and licensing activities. Each NPM is a self-contained module that operates independently of the other modules in a multi-module configuration, as shown in Fig. 7. All modules are managed from a single control room. The plant design contains the following significant features:

- reliable and passively safe systems that are simple in design and operation;
- safety features that assure a core damage frequency substantially lower than the current light water reactor fleet;

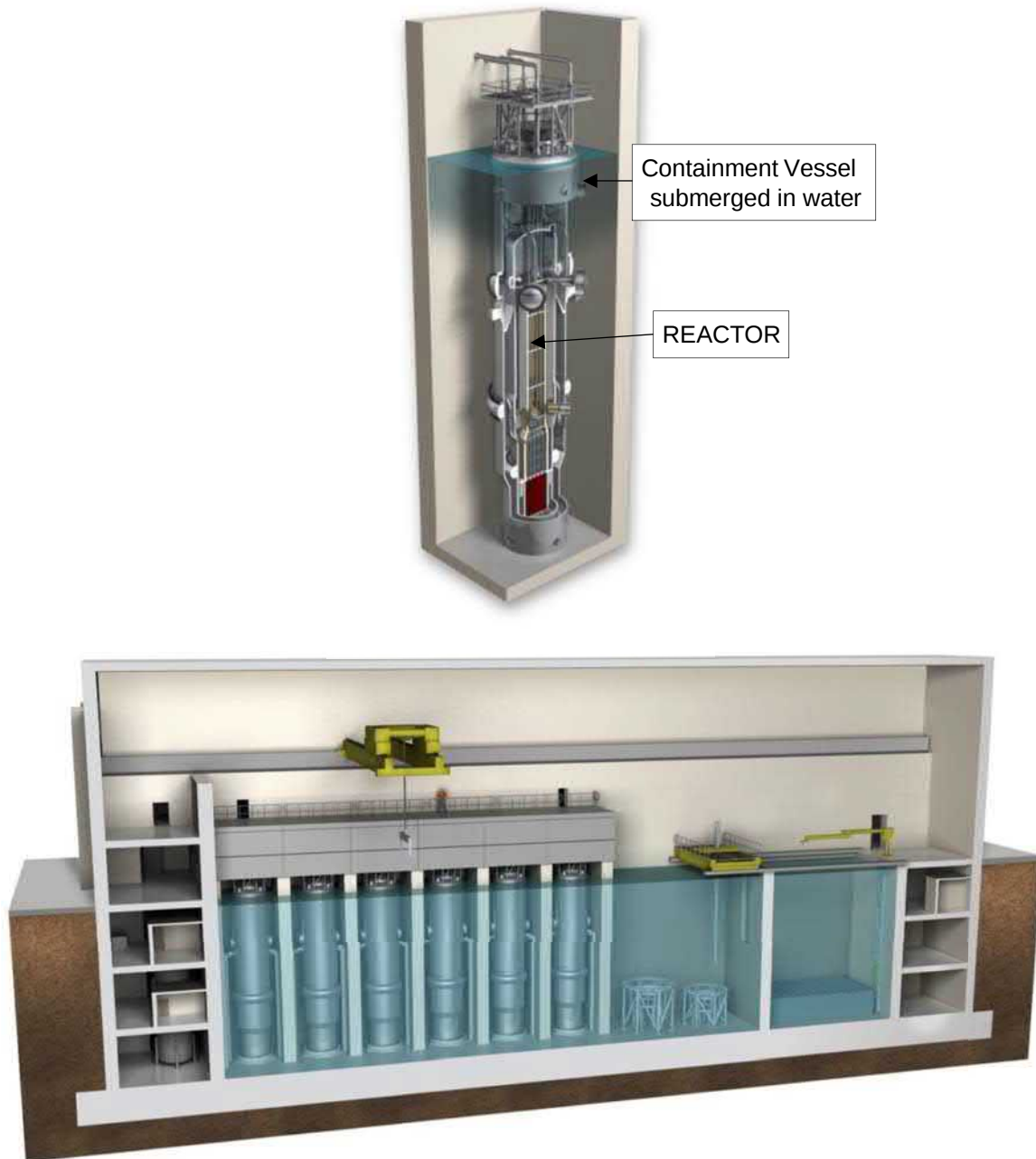


Fig. 7 Reactor configuration of NuScale Power Module developed by NuScale Power Inc., United States of America.

- reactor building designed to protect against aircraft impact while maintaining spent fuel pool integrity and preserving containment integrity;
- 60-year design plant life;
- reduced construction schedule compared to nuclear plants of a comparable output;
- modularization to enable in-shop fabrication of reactor and containment components; and
- use of half-height traditional 17×17 nuclear fuel assemblies.

The NuScale power plant design philosophy is based on the following elements: a design that uses proven light-water reactor technology, modular nuclear steam supply system, factory-fabricated power modules with shippable components, and passive safety systems that allow for unlimited coping time after a design basis accident without power, operator action, or makeup water. The NPM is designed to operate efficiently at full-power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps. The NSSS consists of a reactor core, helical coil steam generators, and a pressurizer within the RPV. The NSSS is enclosed in an approximately cylindrical containment vessel (CNV) that sits in the reactor pool structure. The reactor core is located below the helical coil steam generators inside the RPV.

Each NPM incorporates several simple, redundant, and independent safety features, specifically:

Decay Heat Removal System: The decay heat removal system (DHRS) provides secondary side reactor cooling for non-LOCA events when normal feedwater is not available. The system is a closed-loop, two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each steam generator loop. Each train is capable of removing 100% of the decay heat load and cooling the reactor coolant system. Each train has a passive condenser immersed in the reactor pool. During normal operations, the condensers are maintained with sufficient water inventory for stable and effective operation.

Emergency Core Cooling System: Emergency core cooling system (ECCS) consists of two independent reactor vent valves (RVVs) and two independent reactor recirculation valves (RRVs). For LOCAs inside containment, the ECCS returns coolant from the CNV to the reactor vessel. This ensures that the core remains covered and that decay heat is removed. The ECCS provides a defence-in-depth means of decay heat removal in the unlikely event of a loss of feedwater flow, combined with the loss of both trains of the DHRS. The ECCS removes heat and limits containment pressure by steam condensation on, and convective heat transfer to, the inside surface of the CNV. Heat is then transferred by conduction through the CNV walls and convection to the water in the reactor pool. Long-term cooling is established via recirculation of reactor coolant to the RPV through the ECCS recirculation valves.

Reactor Pool: The reactor pool is a large stainless steel lined pool located below the plant ground level. Water in the pool provides cooling for the NPM for a minimum of 72 h following any design basis accident. During normal plant operations, heat is removed from the pool through a cooling system and ultimately rejected into the atmosphere through a cooling tower or other external heat sink. In an accident where off-site power is lost, heat is removed from the NPMs by allowing the pool to heat up and boil. Water inventory in the reactor pool is large enough to cool the NPMs for an unlimited time without adding water.

In December 2016, NuScale Power submitted the Design Certification Application (DCA) to the US NRC. The design certification review went on track. The NRC issued a design certification approval (DSA) in September 2020, with final design approval (FDA) in 2021. The first plant owner, the Utah Associated Municipal Power System (UAMPS) has target commercial operation date of 2029 for the first module, to be built in Idaho.

SMR-160

The SMR-160 conceptual design has been developed by the US company Holtec International as an advanced PWR-type small modular reactor producing power of 525 MW(th) or 160 MW(e) and adopting passive safety features. Simplification in the design is achieved by using fewer valves, pumps, heat exchangers, instrumentation, and control loops than conventional plants, simplifying operator actions during all plant modes, including diagnosing and managing off-normal and accident conditions. The SMR-160 uses fuel similar to existing commercial LWR fuel, includes no reactor coolant pumps and utilizes a large vertical steam generator. A modular construction plan for SMR-160 involves pre-assembling the largest shippable components prior to arrival at a site. A 24-month construction period is envisaged for each unit.

Fig. 8 depicts the reactor configuration of the SMR-160.

The primary application of SMR-160 is electricity production with optional cogeneration equipment (i.e., hydrogen generation, district heating, and seawater desalination). Target applications include distributed electricity production, repowering coal facilities, uprate for existing nuclear facilities and providing electricity and low-temperature process heat for commercial and military installations. Design optimization includes air-cooled condensation for no wet cooling. The plant design includes a unique fuel management solution that incorporates an integrated state-of-the-art underground dry fuel storage facility with the base plant offering.

The SMR-160 is a pressurized water reactor with the RCS using natural circulation for all power and accident modes and states. The RCS is comprised of the RPV and a SG in an offset configuration with an integrated pressurizer flanged to the top of the steam generator. The RPV and the SG are connected by a single connection that houses both the hot leg and the cold leg. The hot leg is the inner pipe and the cold leg is the annular region of this single connection. Unique among integrated PWRs, the offset configuration allows easy access to the core without moving the RPV or SG during refueling. The SG has a superheating feature which eliminates

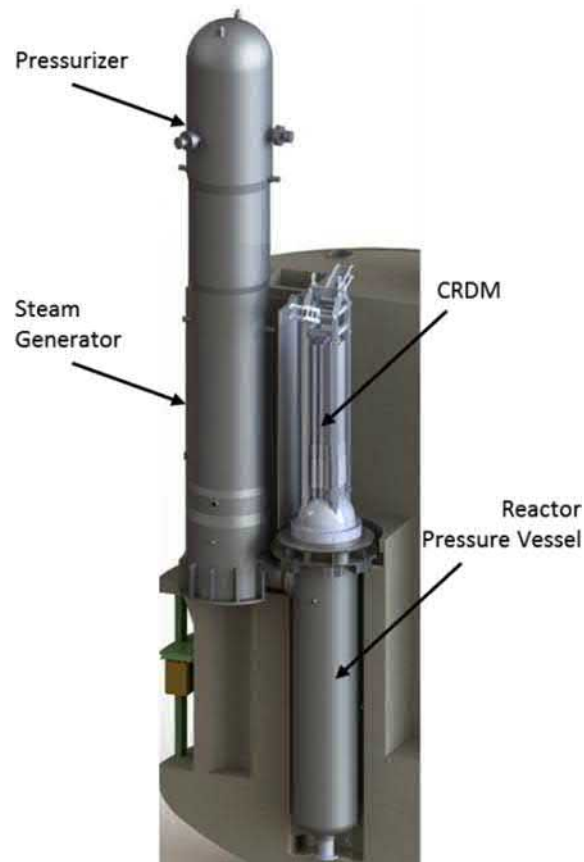


Fig. 8 Reactor configuration of SMR-160 developed by Holtec International Inc., United States of America.

the need for a moisture separator reheater and trains of feedwater heaters, while not compromising the thermodynamic efficiency of the plant. The secondary side has only two feedwater heaters simplifying plant operations and maintenance.

The SMR-160 employs an efficient core design that uses traditional reload shuffle at the end of each cycle. The core is designed for up to a 2-year cycle with flexibility for shorter cycles depending upon utility requirements. Because of the relatively low number of fuel assemblies and overall outage schedule, the shuffling of nuclear fuel is not a critical path activity during an outage. The core contains an array of 96 square-lattice assemblies having an area of approximately 35.5 cm^2 , and an 4.3 m active height. Fuel bundles are surrounded by 16 reflector assemblies that contain pins of solid stainless steel or other reflecting material contained within stainless steel tubes. The reflector assemblies allow for an axial variation in the radial reflector by varying the composition of the pins along their length. The reflector assemblies can also be changed on a cycle to cycle basis if necessary depending upon fuel utilization requirements. Surrounding all of the fuel and reflector assemblies is a solid stainless steel reflector structure. This provides reflection of fast neutrons in order to improve the neutron economy, while reducing the fast neutron fluence through the reactor pressure vessel thereby, increasing vessel life.

To achieve the desired safety performance, the SMR-160 safety basis incorporates defence-in-depth via multiple and diverse simple pathways for heat rejection from the core. All safety systems are within the containment structure (CS), making them secure and safe from external threats. A large inventory of water within a reservoir outside the containment structure provides long-term post-accident coping, and that structure facilitates air cooling for decay heat removal if the reservoir inventory boils off; for an unlimited coping period. SMR-160 engineered safety features include the passive core cooling system (PCCS), passive spent fuel pool cooling system (PSFPCS), and automatic depressurization system (ADS).

The SMR-160 ICS consists of a free standing steel containment vessel called the CS, supported within a reinforced concrete reactor building called the containment enclosure structure (CES), which also provides a missile protection. An annular coolant reservoir between the CS and the CES is filled with water and serves as the UHS for SMR-160. In addition to preventing the release of radioactive fission products to the environment, the ICS acts as a large passive heat exchanger to cool the core and the spent fuel pool. The PCCS and PSFPCS are connected to the integrated containment system (ICS) via the CS metal surface to reject heat to the environment, which is the plant ultimate heat sink (UHS). The reservoir provides sufficient inventory to remove the peak decay heat; as the water in this reservoir depletes (reservoir becomes dry > 60 days), the ICS transitions to air cooling to remove subsequent decay heat. The operator has the option of replenishing the reservoir.

BWRX-300

GE Hitachi Nuclear Energy's (GEH's) BWRX-300 is a 300 MW(e) water-cooled, natural circulation SMR utilizing natural phenomena driven safety systems. The BWRX-300 is an evolution of the U.S. NRC-licensed, 1520 MW(e) ESBWR. It is designed to provide clean, flexible energy generation that is cost competitive with natural gas-fired power plants. Target applications include baseload electricity generation, load following electrical generation within a range of 50–100% power and district heating.

Fig. 9 shows the BWRX-300 reactor configuration.

The key features of the BWRX-300 include:

- 10th generation BWR technology.
- 300 MW(e) SMR.
- Compliance with the international high safety standards.
- Designed to be cost competitive with gas.
- Up to 60% capital cost reduction per MW.
- Evolved from the licensed ESBWR.
- Designed to eliminate large-break LOCA.
- Reduced on-site staff and security.
- Proven components, fuel, and supply chain.
- Constructability integrated into design.

BWRX-300 is being designed with the proven supply chain of the ABWR, and predesigned features from the ESBWR, for its primary circuit, or nuclear boiler system. The primary functions of the nuclear boiler system are:

- To deliver steam from the RPV to the turbine main steam system;
- To deliver feedwater from the condensate and feedwater system to the RPV;
- To provide overpressure protection of the reactor coolant pressure boundary (RCPB);
- To provide the instrumentation necessary for monitoring RPV pressure, steam flow, core flow, water level, and metal temperature.

The RPV has a minimum inside diameter of 4 m, a wall thickness of about 136 mm with cladding, and a height of 27.4 m from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 5.2 m from elevation zero and the active core is 3.8 m high. The relatively tall vessel, due to the chimney, permits natural circulation driving forces to produce abundant core coolant flow.

The major reactor internal components include:

- core (fuel, channels, control rods and instrumentation),
- core support structures (shroud, shroud support, top guide, core plate, control rod guide tube and orifice fuel support),
- chimney
- chimney head and steam separator assembly,
- steam dryer assembly,
- feedwater spargers,
- in-core guide tubes.

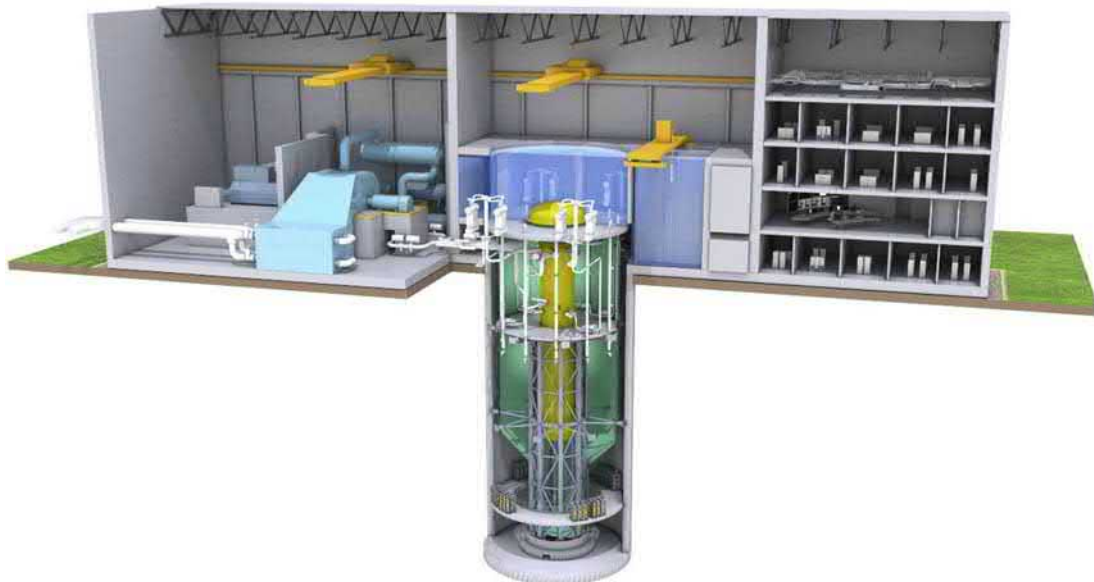


Fig. 9 Reactor configuration of BWRX-300 developed by GE Hitachi Nuclear Energy, United States of America.

Containment/confinement

The BWRX-300 Primary Containment Vessel (PCV) is a metal tank that encloses the RPV and is related systems and components. The PCV is dry and is located mostly below grade. The dry PCV is a leak tight gas space surrounding the RPV and the reactor coolant pressure boundary.

Isolation condenser system (ICS)

The isolation condenser system (ICS) removes decay heat after any reactor isolation during power operations also provides over-pressure protection in accordance with ASME BPV code, section III, class 1 equipment. Decay heat removal limits further increases in steam pressure and keeps the RPV pressure low. The ICS consists of four independent loops, each containing a heat exchanger, with a capacity of 33 MW(th), that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC pool, which is vented to the atmosphere.

The ICS is initiated automatically, and the ICS will also be initiated if a loss of DC power occurs. To start an Isolation Condenser (IC) into operation, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers. The ICS can also be initiated manually by the operator from the MCR by opening the IC condensate return valve. The IC pool has an installed capacity that provides 7 days of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC pool inventory. The ICS passively removes sensible and core decay heat from the reactor when the normal heat removal system is unavailable. Heat transfer from the IC tubes to the surrounding IC pool water is accomplished by natural convection, and no forced circulation equipment is required.

Table 2 Expectation for enhanced safety of iPWR type SMRs (Hidayatullah et al., 2015).

Features	Added values to design	Expected safety improvements
Low core power	- Reduces fission product source term - Low level of decay heat and, therefore, would require less cooling after reactor shutdown	- Enhances in-vessel corium retention - Reduces accident consequences - Simplifies emergency planning
The RCS integrated within the RPV	- No large external primary coolant piping - Longer RPV lifetime due to reduced fast neutron fluence	Lower or eliminate susceptibility to certain potential events, such as LBLOCA
Once-through helical coil integrated steam generator with feedwater and steam inside the tube	- Steam generator is designed to withstand the primary pressure without pressure in the secondary side - Steam system is designed to withstand primary pressure up to isolation valves - Steam generator tubes are in compression - Reduced tube-side water inventory	- Improved steam generator tube integrity with steam generator tube rupture frequency reduced - Excessive cooling during a steam line break accident: the addition of reactivity would be limited and the reactor over power may not compromise critical safety limits, due to small secondary water inventory hence heat removal
Natural circulation	Simplified design and reduced maintenance costs, due to the absence of main coolant pumps	- Eliminate loss of flow accident (LOFA) - Eliminates reactor coolant pump accidents (shaft breaks, seal leakage, pump seizure and pump leaks)
Passive safety systems	- The passive safety systems mitigate or potentially eliminate the need for external power under accident conditions - The auxiliary feedwater system may not be required - Spray systems are not required to reduce steam pressure or to remove radioiodine from containment	- Station-black-out (SBO) can be coped - Active safety systems are not required (low core damage frequency) - Removal of core heat without an auxiliary feedwater system may be a significant safety enhancement - No safety-related pumps for accident mitigation; therefore, no need for sumps and protection of their suction supply
Internal CRDMs	- Elimination of rod ejection event - Elimination/reduction of vessel head penetrations	Reactivity initiated accident (RIA) due to rod ejection is precluded
High pressure design, temperature and evacuated metallic containments	- Containment pressure and temperature for worst-case design basis accident remains below containment design values - All water lost from RPV stays within containment and returns to reactor vessel by passive means - More sub atmospheric pressure during normal operation - The engineered safety systems are simplified - Improve seismic capability	- No postulated small-break LOCA (SBLOCA) capable of uncovering nuclear fuel - Containment integrity assured (due to the metallic containment vessel, molten core concrete interaction is not applicable) - The deep vacuum enhances steam condensation rates for containment heat removal during a postulated SBLOCA - Prevent Hydrogen explosion during a severe accident as little Oxygen will be available
Soluble Boron free core	- No Boron dilution - Less corrosion - Reduces volume of liquid radwaste - Strong negative moderator temperature coefficient - Boron monitor and adjustment systems eliminated	- Reactivity-initiated events are precluded - Reduced occupational radiation dose - Improve reactor transient performance as well as operational safety

Passive containment cooling system (PCCS)

The PCCS is a passive system that removes the decay heat and maintains the containment within its pressure limits for design basis accidents such as a LOCA. It consists of several low-pressure, totally independent heat exchangers suspended in the PCV. The heat in the PCV is transferred to the Reactor pool which is located above the PCV upper head, and which is filled with water during normal operation. The Reactor pool is vented to the atmosphere. PCCS operation requires no sensing, control, logic or power actuated devices for operation. The PCCS condensers are a closed-loop integral part of the containment pressure boundary. Since there are no containment isolation valves between the PCCS condensers and the drywell, they are always in “ready standby” mode.

Expectation of safety enhancement in integral PWR type SMRs

The integral PWR type SMRs with modularization are designed for cost reduction and further safety enhancements. Safety improvements are achieved through design simplicity, incorporation of passive safety systems, small fuel inventory, and use of additional fission product barriers. The NSSS with integral module configuration eliminates external coolant loop piping, thus eliminating the large-break loss-of-coolant accident (LBLOCA) scenario. The passive engineered safety features (ESFs) eliminate the need for external power under accident conditions. The expectations for improved safety of iPWR type SMRs are listed in **Table 2** below. The expected core damage frequency (CDF) will be much less and may be of the order of 10^{-6} – 10^{-8} per year or even lower. The CDF of the order of 10^{-8} per reactor-year has been claimed by vendors so far (IAEA, 2016).

See Also: Impact of Small Modular Reactors on the Acceptance of Nuclear Power by the Public, Investors, and Owners.

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Floating Nuclear Power Plants With Small Modular Reactors

M. Hadid Subki, Nuclear Power Technology Professional, Vienna, Austria

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Glossary

CRDM	Control Rod Drive Mechanism
ECCS	Emergency Core Cooling System
FNPP	Floating Nuclear Power Plant
GHG	Green House Gas
IEA	International Energy Agency
LWR	Light-water reactor
NIKIET N.A.	Dollezhal Scientific Research and Design Institute of Energy Technologies, Russian Federation
NSSS	Nuclear Steam Supply System
PCV	Primary Containment System
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
R&D	Research and development
R&D&D&D	Research, development, demonstration and deployment
RPV	Reactor Pressure Vessel
SG	Steam Generator
SGA	Steam Generating Aggregate
SMR	Small and medium-sized reactor or Small Modular Reactor
TGP	Turbine Generator Package
TNPP	Transportable Nuclear Power Plant
UO ₂	Uranium-dioxide
GW	10 ⁹ watt
MPa	10 ⁶ pascal
MW(e)	10 ⁶ watt electrical
MW(th)	10 ⁶ watt thermal

Background

Small and medium-sized power reactors (SMRs) are not only being developed for inland on-the-grid and off-shore constructions, but also for floating nuclear power plant (FNPP) and off-grid applications, including oil and gas rigs far from the shore. FNPP with SMRs have the potential to open up new market segments in countries wishing to adopt nuclear energy for various off-grid applications. Small-sized reactors are intended for the remote or isolated regions where the hydrocarbon fuel delivery cost is very high, in which electricity networks are either absent or underdeveloped. By the same token, FNPPs with SMR are options for regions where large capacities are not required, and where consumers are establishing the demands for reliable electricity supply systems, as well as

for alternatives to hydrocarbon based energy production. FNPPs are also a potential power producer for mining, oil and gas industries in different region.

Traditionally, Russia has deployed and is operating marine-based units with SMRs for nuclear icebreakers. The success and lessons-learned from operating nuclear icebreakers are incorporated in the development of FNPP. The technology developer classifies FNPP based on SMR as follows (Fadeev, 2014a, b):

- By the power capacity of a single module (in line with the IAEA power definition): Small power units are designs rated up to typically 300 MW(e); and medium-sized power units are designs that produce between 300 and 700 MW(e);
- By the energy consumption sectors, which means:
 - Local Sector with 1–20 MW(e) SMRs: the sector is isolated from the Russian energy system and other energy sources; it consists of one NPP and one or several consumers;
 - Territorial Sector with 20–100 MW(e) SMRs: the sector is centralized; isolated from the Russian energy system; consists of several sources and consumers;
 - Regional Sector with 100–700 MW(e): this sector is centralized; it consists of balanced sources and consumers; connected to the Russian energy system.

The deployment of SMR NPPs in the Russia is in line with the development strategy for the Arctic region, and a promising market niche. Its foundation in terms of proven technologies and past developments can be outlined as follows:

- (i) Long experience in development and operation of marine-based nuclear power plants. OKBM Afrikantov has designed and supplied reactors for nuclear submarines up to the 4th generation, as well as for surface naval ships with VVER reactors, that have accumulated a total operating time of over 9500 reactor-years. The R&D activities are focused on engineering standardization for the entire power range; enhancing the reliability, safety, and manoeuvrability; as well as on enhancing maintainability by reducing maintenance scope and expanding the service operation between maintenance outages.
- (ii) Long experience in the development and operation of reactor plants for nuclear icebreakers, with a total number of 20 reactors (including 8 installed on current nuclear icebreakers). More than 50 years operation of three generations of nuclear icebreakers in the Arctic region, amounting to a total operating time of more than 365 reactor-years. Design of an innovation SMR for the multipurpose nuclear icebreaker of the 4th generation (RITM-200M).
- (iii) Experience with R&D, design and fabrication of reactor plants for the FNPP: Within 3 years two KLT-40S reactors were supplied for the FNPP “Academician Lomonosov” confirming the efficiency of combining the functions of chief designer and complete supplier.

This shows that proven reactor technologies and innovative solutions are available for countries and utilities with specific needs of floating NPPs with SMRs.

Status of SMR technology development for floating nuclear power plants

In addition to land-based integral-type PWR-type designs that could be deployed as a single reactor module or combined to multi-module SMR NPPs in which each module is independent and operated from a common main control room, there are efforts to develop SMR floating power concepts. In this chapter, six marine-based water-cooled, water-moderated SMR types for floating power plants are described (IAEA, 2020a, b).

- KLT-40S: a 35 MW(e) compact-PWR type SMR for the Akademik Lomonosov Floating NPP that is in operation since December 2019 using two reactor modules producing 70 MW(e);
- RITM-200M: a 50 MW(e) integral-PWR type SMR developed by OKBM Afrikantov as the next generation after KLT-40S for small, isolated, or distributed grids. Six RITM-200 units have been manufactured for nuclear icebreakers. Four reactors are already installed on the Sibir and the Arktika icebreakers;
- ABV-6M: a 6–9 MW(e) per module integral-PWR type with natural circulation primary cooling for shipyard fabricated, multipurpose, transportable NPP designed for safe operation over 10–12 years without refueling at the berthing platform or along the coast, developed by OKBM Afrikantov;
- VBER-300: a larger 325 MW(e) integral-PWR type developed by OKBM Afrikantov, to supply thermal and electric power to remote areas where centralized power is unavailable, and to substitute capacities of fossil fuels for cogeneration, deployed both as land-based or floating marine-based;
- ACPR-50S: a 50 MW compact-loop PWR type SMRs with forced circulation already deployed in nuclear icebreakers, developed by the China General Nuclear Power Group (CGNPC);
- SHELF: a 28.4 MW(th)/6.6 MW(e) integral PWR, as power source for remote and hard-to-reach locations with decentralized power supplies, developed by NIKIET.

Design description of SMR-based floating nuclear power plants

The following sub-sections are providing brief descriptions of the design characteristics and major operational and safety characteristics of the six representative advanced designs of marine-based PWR-type SMRs for floating nuclear power plants or units. Detailed

information, in particular in terms of the major design objectives, as well as the challenges and opportunities on the path to commercialization can be found in IAEA (2020a, b).

Table 1 provides a summary of the main design characteristics of the various reactor concepts that are being investigated.

KLT-40S

The KLT-40S is a PWR-type SMR developed for the “Akademik Lomonosov” FNPP to produce 70 MW(e) from two reactor modules. The FNPP has started its commercial operation in May 2020 in the Pevek Region, Russia, after it was connected to the grid in December 2019. The KLT-40S is based on the commercial KLT-40 marine propulsion plant and is an advanced variant of reactor plants that power nuclear icebreakers. The “Akademik Lomonosov” FNPP, featuring the two KLT-40S reactors, was manufactured in shipyards and then delivered to the customer fully assembled, tested and ready for operation. It can be moored in any coastal region.

Fig. 1 shows the reactor configuration of the KLT-40S.

The reactor is a modular design with the core, steam generators (SGs) and main circulation pumps connected with short nozzles. The reactor has a four-loop system with forced and natural circulation, a pressurized primary circuit with canned motor pumps and leak tight bellow type valves, a once-through coiled SG and passive safety systems. Fuel utilization efficiency is provided by the use of all improvements of nuclear fuel and fuel cycles of nuclear icebreaker reactors, spent fuel reprocessing and the increase of fuel burnup through the use of dispersion fuel elements.

One of the advantages is long-term autonomous operation in remote regions with a decentralized power supply. The design requires that after every 3–4 years of operation the reactor is refueled. The used nuclear fuel is then stored on board the FNPP, and no special maintenance or refueling ships are necessary.

The KLT-40S is designed with proven safety solutions such as a compact structure of the SG unit with short nozzles connecting the main equipment, without large diameter primary circuit pipelines, and with proven reactor emergency shutdown actuators based on different operation principles, emergency heat removal systems connected to the primary and secondary circuits, elimination of weak design points based on the experience of prototype operation and use of available experimental data, certified computer codes and calculation procedures.

The keel of the first FNPP carrying the KLT-40S, the “Akademik Lomonosov,” was laid in 2007. The “Akademik Lomonosov” was completed in 2017. The FNPP was moored in Pevek, Chukotka region of Russia for the connection to the grid in December 2019 and launched for commercial operation in May 2020 producing 70 MW(e).

RITM-200M

The RITM-200M is designed by OKBM Afrikantov as an integral reactor with forced circulation for use in nuclear icebreakers. For the next deployment, RITM-200M is designed for land-based and floating-based multi-purpose application reactors for electrical power generation of 50 MW(e) per module, water desalination, and district or industrial heating.

Fig. 2 depicts the reactor Configuration of the RITM-200M.

It employs a low enriched cassette type reactor core similar in design to the KLT-40S. The uranium oxide fuel enrichment is up to 20 wt% ^{235}U , and the reactor houses 199 fuel assemblies. The design also allows for a low fluence on the reactor vessel.

The reactor is designed as an integral vessel with the main circulation pumps located in separate external hydraulic chambers and with side horizontal sockets for steam generator (SG) cassette nozzles. The four section SGs have 12 rectangular cassettes, while the four main circulation pumps are located in the cold leg of the primary circulation path and separated into four independent loops. The reactor is also designed to use forced circulation of the primary coolant and an external gas pressurizer system. The SGs produce steam of 295 °C at 3.82 MPa flowing at 248 ton per hour.

The core is designed to operate for 7 years at a 65% capacity factor before the need for refueling. The assigned service life of the plant for replaceable equipment is 20 years with a continuous operation period of 26,000 h; for permanent equipment it is double that. The RITM-200 is designed to provide 30 MW of shaft power on a typical nuclear icebreaker and can be used on vessels of 150–300 t displacement. The reactor can be considered for floating heat and power plants, power and desalination complexes, and offshore drilling rigs. The designers also claim that the overall size of the steam generating unit allows transportation of the reactor by rail. The reactor plant in containment has a mass of 1100 t and is 6 m × 6 m × 15.5 m.

The Atomenergomash, as the primary manufacturer for nuclear reactor components has signed a contract for the supply of RITM-200M reactor units for two more nuclear icebreakers. The contract was signed between the Atomenergomash subsidiary OKBM Afrikantov and the Baltic Shipyard. The new vessels will join the LK-60 icebreakers Arktika, Sibir and Ural, which are dual-draught (8.55 or 10.5 m) wide-beam (34 m) ships of 25,450 dead weight or 33,540 dead weight with ballast, able to handle 3 m of ice. They each have two RITM-200 reactors of 175 MW(th) each, delivering 60 MW(e) at the propellers via twin turbine-generators and three motors. Arktika is expected to enter operation in 2020, Sibir in 2021 and Ural in 2022.

ABV-6M

The ABV-6M reactor installation is a nuclear steam generating plant with an integral pressurized LWR and natural circulation of the primary coolant. The ABV-6 M design was developed using the operating experience of the Russian water-cooled, water-moderated power reactors (VVERs) and has incorporated all the lessons-learned in the field of NPP safety. The main mission of the ABV-6M is to offer multipurpose power sources based on proven marine reactor technologies, providing easy transport to the site, rapid assembly, and safe operation.

Table 1 Key PWR-based SMRs for floating NPP: main design characteristics.

	<i>KLT-40S</i>	<i>RITM-200 M</i>	<i>ABV-6 M</i>	<i>VBER-300</i>	<i>ACPR-50S</i>	<i>SHELF</i>
Country of Origin	Russian Federation				China	Russian Federation
Design organization(s)	ROSATOM, JSC OKBM Afrikantov				CGNPC	NIKIET
Reactor type	Integral-PWR	Integral-PWR	Integral-PWR	Integral-PWR	Compact Loop-PWR	Integral-PWR
Thermal Output, MWt	150	175	38	917	200	28.4
Electrical output, MW(e)	35	50	6–9	325	50	6.6
Primary cooling mode	Forced circulation	Forced circulation	Natural circulation	Forced circulation	Forced circulation	Forced and Natural circulation
Prim. System pressure, MPa	12.7	15.7	16.2	16.3	15.5	14.7
Core inlet temp., °C	280	277	250	292	299.3	270
Core outlet temp., °C	316	313	325	328	321.8	310
Fuel type	UO ₂ pellet in silumin matrix	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet
Assembly array	Hexagonal	Hexagonal	Hexagonal	Hexagonal	17 × 17 square	Hexagonal
Enrichment, %	18.6	<20	<20	4.95	<5	19.7
Fuel cycle, months	30–36	60	120–144	72	30	6 years
Reactivity control	Control rods	Control rods	Control rods	Control rods, soluble boron	Control rods, solid burnable poison and boron solution	Control rods
Safety system	Active (partially passive)	Active and Passive	Passive	Active and Passive	Passive	Active and Passive
Design life, years	40	60	40	60	40	60
RPV height/diameter, m	4.8/2.0	8.5/3.3	6/2.4	Not available	7.2/2.2	14/8
Seismic design	9 (MSK scale)	0.3 g	7 Richter scale	0.25 g	Not available	64 (MSK scale)
Design status	In operation for the Akademik Lomonosov FNPP since 12/2012	Deployed in 6 nuclear icebreakers, floating and land-based version available	Detailed design completed	Detailed design, in licensing	Detailed design	Detailed design

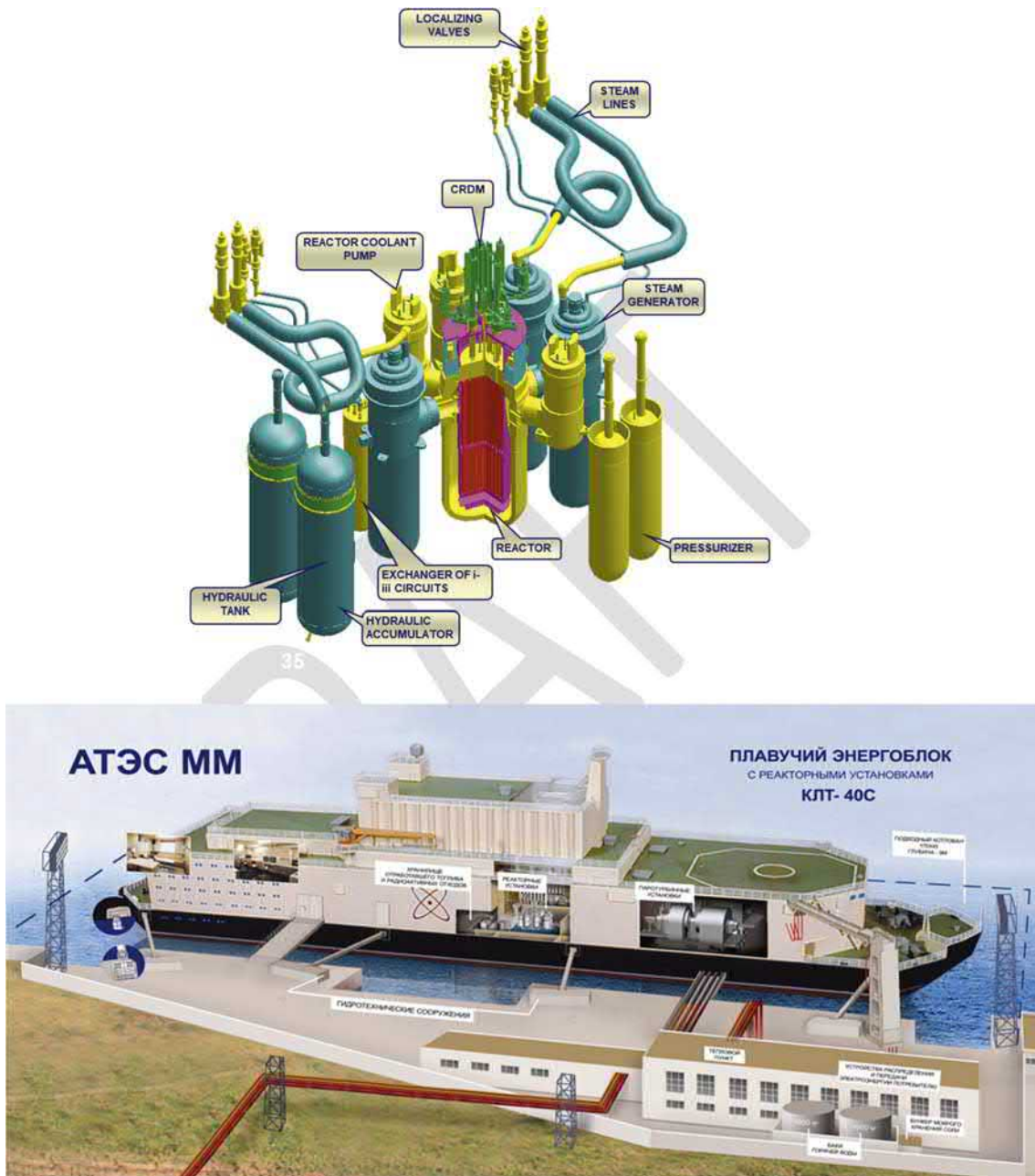


Fig. 1 Reactor Configuration of KLT-40S developed by OKBM Afrikantov.

Fig. 3 shows the reactor configuration for the ABV-6M.

The reactor is designed to produce 45 MWt and 8.6 MWe in condensation and 14 MWt and 6 MWe in cogeneration mode. The reactor operates under the normal PWR conditions of 15.7 MPa in the reactor pressure vessel. The steam generators (SGs) located inside the vessel generate 290 °C steam at 3.14 MPa flowing at 55 t per hour.

The reactor cover is set under biological shielding and the control rod drive mechanism is located above the shield outside the vessel. The main reactor pump equipment is arranged on the shield tank as a single steam generating aggregate (SGA) that is transportable by rail. The SGA has a mass of 200 t and is 5 m long by 3.6 m wide with a height of 4.5 m.

The core lifetime without reloading or shuffling of fuel is 10 years with 16,000 h of continuous operation.

Specifically, the ABV reactor installation is intended as a universal power source for FNPPs. The reactor is designed with the capability of driving a floating unit with a maximum length of 140 m, a beam of 21 m, a draft of 2.8 m and a displacement of 8700 t. Depending on the needs of the region, the FNPP can generate electric power, or provide heat and power cogeneration, or heat generation for seawater desalination, or can be used for other applications.

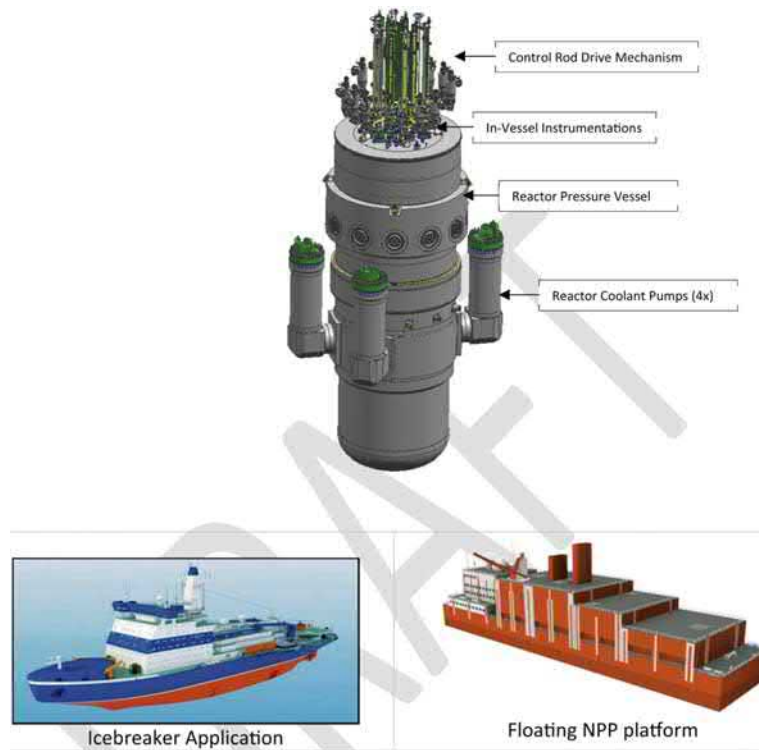


Fig. 2 Reactor Configuration of RITM-200 developed by OKBM Afrikantov.

The ABV-6M reactor is fabricated as a large ready-made unit that is transported by special truck or waterway to the shipyard. The total reactor module has a mass of 600 t and is 13 m in length and 8.5 m in diameter. This allows for a small footprint for the plant building, which would be approximately 67 m long and 47 m wide.

VBER-300

The VBER-300 reactor is a medium sized power source for land based NPPs and nuclear cogeneration plants, as well as for floating NPPs. The VBER-300 design is a result of the evolution of modular marine propulsion reactors. The design was developed using the operating experience of VVER type reactors and achievements in the field of NPP safety. Features of the VBER-300 include the use of proven nuclear ship building technologies, engineering solutions and operating experience of VVER reactors as well as the possibility to enlarge or reduce the source power using only unified VBER-300 equipment (reactors consisting of two, three or five loops).

Fig. 4 shows the reactor configuration of the VBER-300.

The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard VVER fuel assemblies (FAs). The fuel cycles are 3×2 years and 4×1.5 years. The number of FAs in the refueling batch is either 15 or 30; maximal fuel burnup does not exceed 60.0 MWd/kg U for the cycle with 30 fresh FAs in the reloading batch and maximum initial uranium enrichment. The main reactor coolant pump is an axial, single-stage canned motor pump with experience from operation of more than 1500 main circulation pumps on ships. The steam generators are once-through modules with the secondary coolant flowing within the tubes.

The safety assurance engineering solutions incorporated into the design are: prioritization of accident prevention measures, design simplification, inherent safety, and the defense in depth principle; passive safety systems and enhancement of safety against external impacts (including acts of terrorism); and limitation of severe accident consequences. The VBER-300 emergency shutdown system consists of the control rod drives, two trains of liquid absorber injection, and two trains of boron control from the make-up system. The ECCS consists of two stages of hydraulic accumulators and the RHR system consists of two passive emergency heat removal systems and a process condenser. The estimated reactor unit strength holds up against seismic impacts of maximum magnitude VIII on the MSK-64 scale.

The VBER-300 preliminary design was completed in 2002, and a technical and commercial proposal (a shorter version of technical and economic investigation) for construction of a land based or floating NPP with the VBER-300 was prepared. The preliminary design passed the branch review by Rosatom and was approved by the Scientific and Technical Council. Currently, there are

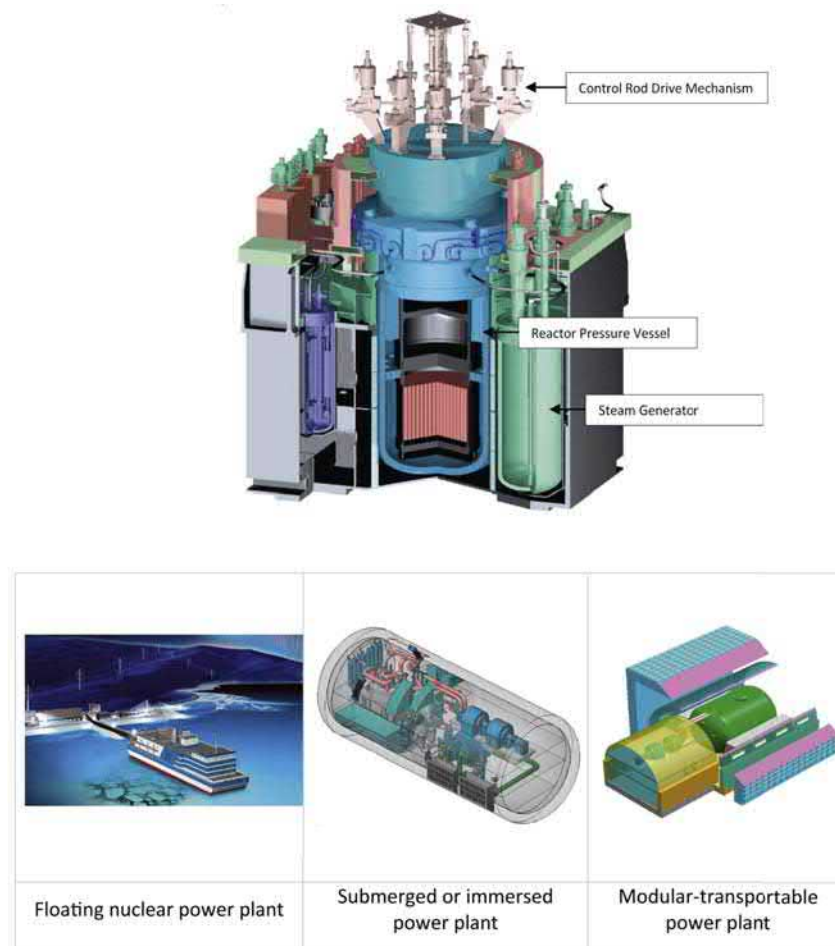


Fig. 3 Reactor Configuration of ABV-6 M developed by OKBM Afrikantov.

two directions of further project development: first, within the framework of scientific and technical cooperation between the Russia and Kazakhstan, and second, replacement of ageing NPP in Russia. Kazatomprom JSC of Kazakhstan completed a feasibility study for the VBER-300 in Aktau city in 2009 and planned to start construction.

ACPR-50S

Rated at 200 MW(th) and 50 MW(e), the ACPR-50S is a small modular offshore floating reactor developed by the China General Nuclear Power Corporation (CGNPC). It is intended as a flexible solution for combined supply of heat, electricity and fresh water for marine resource development activities, energy supply and emergency support on islands and along the coastal area, (IAEA, 2020a, b).

Fig. 5 displays the reactor configuration of the ACPR-50S.

As an offshore floating SMR, the ACPR-50S is designed as a multipurpose power reactor for the following applications: combined energy supply for offshore oil drilling platform; offshore combined energy supply; coastland and island combined energy supply; energy supply for offshore mining, nuclear power ship; and distributed clean energy for islands together with solar energy and wind power.

The ACPR-50S adopts design simplification to reduce cost and investment risks to be competitive with conventional offshore energy sources. Modular design is adopted through standardized and streamlined manufacturing, aiming for shorter construction period as well as lower cost.

The compact loop-type PWR NSSS design of the ACPR-50S consists of the reactor pressure vessel (RPV) that houses the core, two once-through steam generators (OTSG), two main reactor coolant pumps (RCP) and a pressurizer (PZR), all of which are interconnected by short reactor coolant system legs. The compact layout of the primary loop reduces the probability of LOCA significantly. The system has natural circulation capability for heat removal capacity when operating of up to 10% thermal power.

The low power density design with a low enriched UO_2 fueled core ensures a thermal margin of greater than 15% to the departure to nuclear boiling in the core which can accommodate any anticipated transient event. The 37 fuel assemblies (FAs) of ACPR-50S core, with an axial length of 2.2 m, have a square 17×17 configuration. The reactor will be able to operate 30 months per fuel cycle.

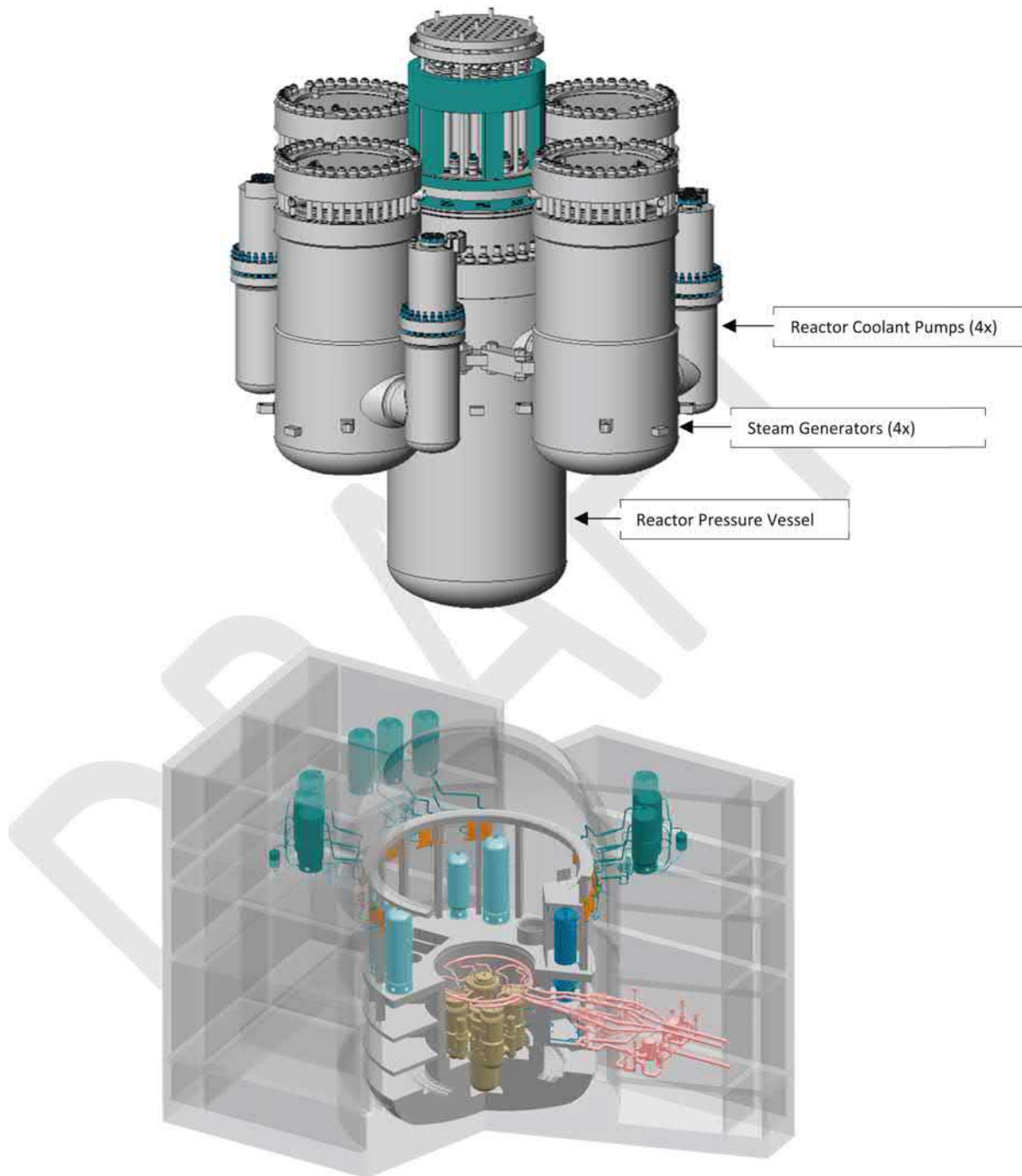


Fig. 4 Reactor Configuration of VBER-300 developed by OKBM Afrikantov.

CGNPC has completed the preliminary design work of the ACPR-50S and is now preparing for detail design. The National Nuclear Safety Administration (NNSA) is performing safety review for this SMR. The ACPR-50S planned to apply for the construction permit to the NNSA in 2020. An industrial demonstration plant of ACPR-50S is being planned to be constructed in China, with a target of startup commissioning in mid of 2020s.

SHELF

SHELF is designed by NIKIET as a local power source for users in remote and hard-to-reach locations. The power unit is a power capsule to generate 6.6 MW(e). The power capsule is developed in two options: containing only all reactor components, and the capsule of a bigger size that includes also the turbine generator package (TGP), the automated and remote-control system,

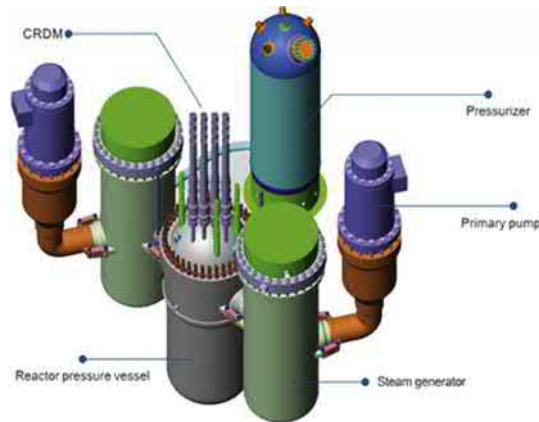


Fig. 5 Reactor Configuration of ACPR-50S designed by CGNPC.

monitoring and protection system and the electricity output regulation. The SHELF power capsule can be used as both, a floating and a submerged nuclear power plant.

Fig. 6 shows the configuration of the SHELF reactor.

Depending on the customer requirements, the SHELF plant can be equipped with a system for direct heat supply to resident and production premises with a capacity of 12 Gcal/h, or a desalination plant with a capacity of 500 m³/h of fresh water. In case of the land based plant, the heat is removed by external heat exchangers cooled by mechanical air pumping. The SHELF plant does not require local water sources.

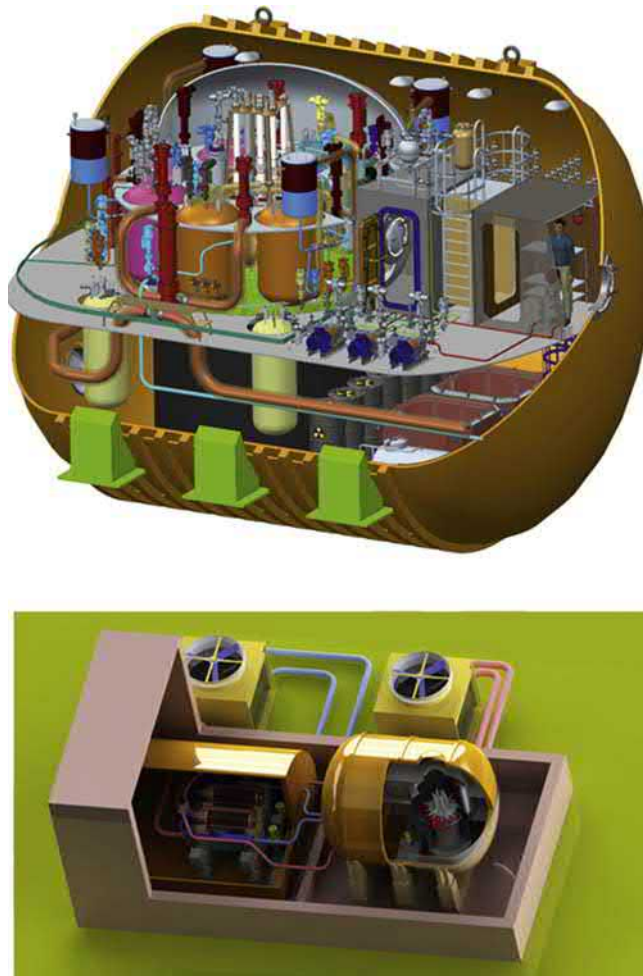


Fig. 6 Reactor Configuration of SHELF developed by NIKIET.

SHELF is a water-cooled reactor of integral layout with combined (forced and natural) coolant circulation modes. The fuel campaign is 6 years at a capacity factor of 80%. The components for SHELF are assembled inside a cylindrical power capsule with an inner diameter of 8 m and a length of 14 m. The unit's external systems include an automated instrumentation and control system (AICS) and uninterrupted power supply systems.

The core is of the heterogeneous cartridge type and consists of 163 hexahedral fuel assemblies (FA) of three different types. A number of FAs contains burnable poison and control absorber rods. Absorber rods for reactivity control are united in six identical shim groups.

High level safety is achieved through the following features:

- Use of an integral water-cooled water-moderated reactor with well-developed intrinsic self-protection properties and the following inherent features:
- A defense-in-depth system of barriers to the spreading of ionizing radiation and radioactive products of uranium fission into the environment.

The licensing of the nuclear plant design was scheduled for 2019–2020. During 2017 to 2019, the detailed design was performed for the SHELF power capsule with an electric power of 6.6 MW.

Major challenges ahead and R&D approaches to resolve them

With the deployment of the Akademik Lomonosov FNPP using 2 units of KLT-40S in Pevek, Russian Federation, referenced-design and –deployment projects are now available for countries interested to consider FNPP for their specific conditions. The rationale of the development of FNPP using SMRs is to offer a factory-fueled and tested, transportable nuclear facility for generating electricity and heat in remote regions and small islands where land-based facilities would not be viable. The major challenges ahead lie in addressing legal and institutional issues linked with the deployment of marine-based SMRs when categorized as transportable NPPs (IAEA, 2013). These include preparedness of legal and regulatory frameworks, nuclear safety, security, liability and safeguards and transboundary aspects of Transportable NPPs (TNPP). Implications of floating NPPs on the infrastructure of the recipient countries ought also to be examined. The R&D on both ships or marine platforms with nuclear reactors is quite established. By the same token, the R&D for water-cooled SMRs on floating NPPs is also mature. A recent trend has been to increase R&D aiming at investigating the viability of SMRs based on high temperature and of molten salt reactor technologies for floating NPP, modelling of heat and electricity cogeneration, and commercialization issues.

See Also: Floating Nuclear Plants to Improve Economics, Safety, Siting, and Proliferation Resistance; Marine Propulsion.

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Pebble Bed Gas Cooled Reactors

Frederik Reitsma*, Ultra Safe Nuclear Corporation (USNC-Europe), Olette, France

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Glossary

AVR Arbeitsgemeinschaft Versuchsreaktor; first experimental pebble-bed reactor that operated in Julich, Germany (1967–1988).

Burnup The amount of fission energy generated per unit weight of fuel.

CP Coated Particle.

GCR Gas-Cooled Reactor.

GIF Generation IV International Forum.

HEU Highly Enriched Fuel (>20% enriched; in the AVR up to 93%).

HPC High Pressure Compressor.

HTGR High Temperature Gas-Cooled Reactor.

HTR High Temperature Gas-Cooled Reactor.

HTR-10 10 MW High Temperature Reactor; a 10 MWt prototype pebble bed reactor at Tsinghua University in China (2000 -).

HTR-PM High Temperature Reactor – Pebble-bed Module; a demonstration power plant at the Shidao Bay site in Rongcheng, Shandong Province, China,

INET Institute of Nuclear and New Energy Technology (Tsinghua University, Beijing).

LEU Low Enriched Fuel (<20% enriched; in the AVR 7%, 10% and 16%).

LPC Low Pressure Compressor.

LWR Light Water Reactor.

MWe Megawatt electric.

MWth Megawatt thermal.

NGNP Next Generation Nuclear Plant.

PBMR Pebble Bed Modular Reactor.

PCR Pre-stressed Concrete Reactor pressure Vessel.

Reactivity coefficient Express the change in the effective multiplication factor, k_{eff} , per unit change in component temperature or density. k_{eff} is the ratio of neutron production rate to neutron loss rate. In a critical reactor $k_{\text{eff}} = 1.000$ and the power is constant. If $k_{\text{eff}} < 1.0$ (> 1.0) the core is sub (super)-critical and the power drops (increases). Excess reactivity is the extent of k_{eff} increase above 1.0.

RPV Reactor Pressure Vessel.

R&D Research and Development.

S-I Sulfur-Iodine thermochemical process for hydrogen production.

*The director of analysis at USNC that aim to deploy the first HTGR micro reactor in Canada. Till recently he was the team leader for SMR technology development and technical lead for gas cooled reactors at the International Atomic Energy Agency (IAEA). Prior to that he was a member of the PBMR project in South Africa for 13 years.

THTR Thorium High Temperature Reactor; was a prototype thorium high-temperature pebble bed reactor in Hamm-Uentrop, Germany (1983–1989).

TRISO Tri-Structural Isotropic Fuel.

VHTR Very High Temperature Reactor.

Introduction

Pebble bed gas cooled reactors are part of High Temperature Gas-cooled Reactors (HTR or HTGR (Fütterer et al., 2021)). They are helium-cooled graphite-moderated nuclear fission reactors utilizing fully ceramic fuel. All pebble bed reactor designs considered since the late 1980s applied the same principles established in the HTR-Module design (Reutler and Lohnert, 1984) that ensure no significant or early radioactivity (or radionuclide) releases under all accident conditions.

In general, the pebble bed gas cooled reactor technology is seen as mature. Two pebble bed reactors, the AVR and THTR-300 in Germany, have operated in the past, one test reactor (the HTR-10 in China) is current in operation and a commercial demonstration plant (the HTR-PM with two reactor units) is currently being commissioned in China. The Very High Temperature Reactor (VHTR) has been selected as one of the six Generation-IV International Forum reactor systems and investigates long term concepts aiming at increasing the helium outlet temperatures above 950 °C to reach even higher efficiency and to service the niche of very high temperature process heat markets.

More details on the development of HTR technology can be found in a recent authoritative summary (Kugeler and Zhang, 2019).

The reactor system

The nuclear steam supply system

The pebble bed SMR core is made up by loosely packed fuel spheres loaded into the core cavity formed by the graphite top, bottom and side reflectors. The graphite serves as a neutron reflector but also as the neutron moderator and the main core structural material. The pebble fuel is typically 60 mm in diameter and typically contains between 10,000 and 15,000 TRISO coated fuel particles (Helmreich, 2021) within the inner part of the fuel sphere surrounded by an outer ‘fuel free’ protective layer, all formed from graphitic material. This is illustrated in Fig. 1 (derived from Venter, 2010) with the fuel sphere’s components shown and as loaded into a cylindrical core.

The core geometry is generally cylindrical, tall and slender but can also be an annular shape (with a fixed (or dynamic) cylindrically-shaped central graphite reflector). The core volumes are relatively large and at the upper scale of designs (250–425 MW_{th}) the core may contain nearly half a million fuel spheres. The large core volume limits the power density to 3–6 MW/m³ and a tall and slender core increases the heat transfer from the fuel to the environment so that no active cooling is needed in case of loss of coolant accidents and decay heat can be transferred passively through a non-insulated steel reactor pressure vessel to coolers outside of the reactor (Reutler and Lohnert, 1984; Kugeler and Zhang, 2019; Said et al., 2006). The small core diameters also shortens the heat transfer path from the core radial center to the reactor vessel thus limiting maximum fuel temperatures in accident conditions.

In this severe loss of all active cooling as well as coolant condition only a very small volume of the core may reach temperatures above 1500 °C and this is still far below the temperatures where any failures are expected. Historically, pebble fuel has been tested for accident conditions up to 1600 °C for several days without any significant failures (IAEA, 2010). More recently these heat-up tests were also repeated for the newly manufactured HTR-PM pebble fuel elements (Freis et al., 2020) and for TRISO coated particles up to temperatures of 1800 °C and beyond, with good results (EPRI, 2020).

Fuel spheres are added to the top and extracted at the bottom of the core through a defuel chute (see Fig. 2 left). The bottom cone angle and defuel chute diameter can be chosen to prevent bridge formation and therefore any pebble flow blockages. The design may allow the fuel to be recycled many times through the core to flatten the radially-averaged axial burnup and power profile while also increasing the average discharge fuel burnup (due to less leakage). The flatter axial power profile also reduces the maximum accident temperatures (Geringer and Reitsma, 2010) since the decay heat axial production profile is also flatter allowing a more evenly conduction and radiation to the vessel and outside cooling panels. This loading and unloading of the fuel spheres are performed on-line at full power operation conditions facilitated by a fuel handling system (see Fig. 2 right). The main functions of the fuel handling system are listed in the figure.

The burnup of the fuel is measured after it has been extracted from the core, and based on the measured value, the fuel is either returned to the core to pass through once more or if a target burnup has been reached, it is discharged as spent fuel. The huge advantage of on-line refueling is the increase in the plant availability since no refueling outages is required. The fuel circulation rate and discharge burnup is a function of the average number of passes and fuel enrichment, and may need adjusted to ensure the core remains critical with the desired excess reactivity. With on-line refueling the excess reactivity throughout the lifetime of the reactor can be limited to just overcome operational transients due to power changes (xenon buildup).

Figures and Captions

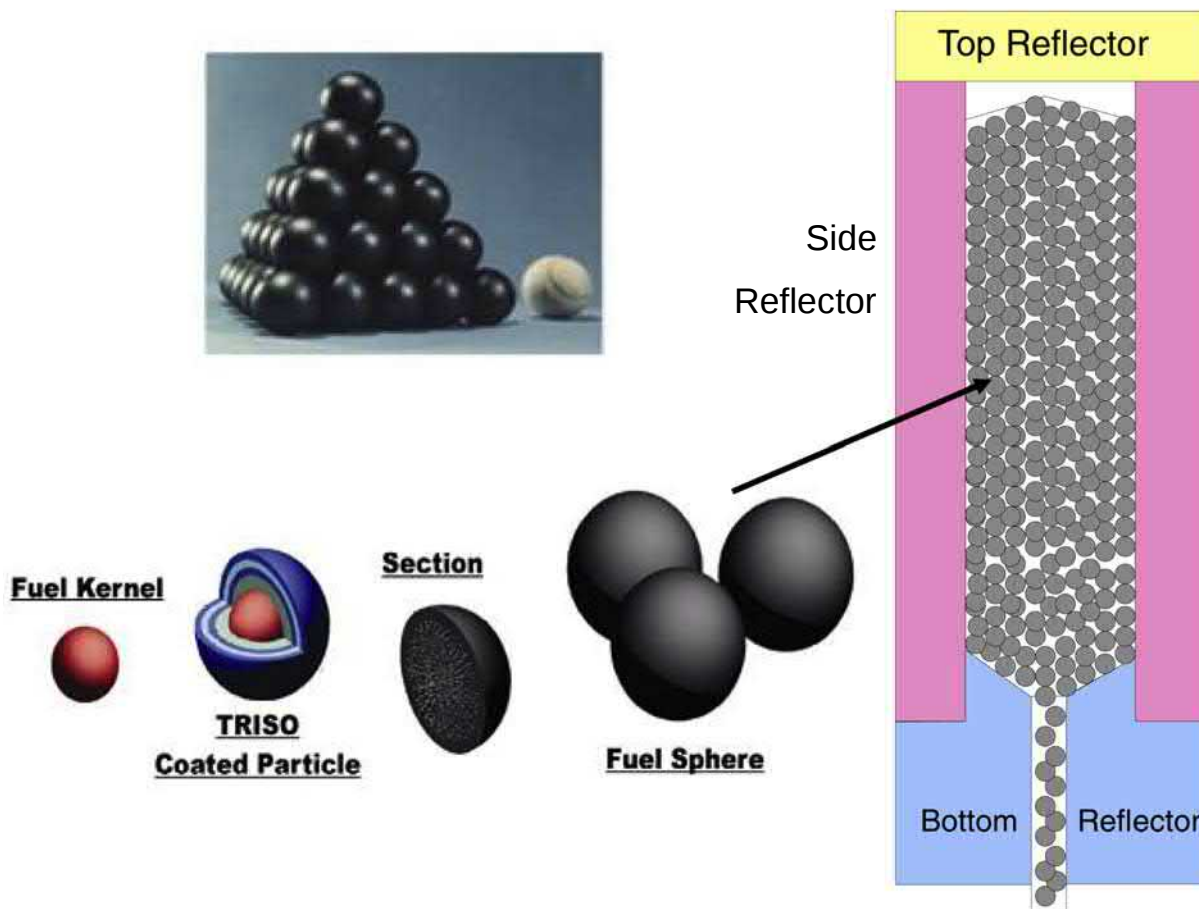


Fig. 1 View of a pebble bed gas cooled reactor with TRISO particle fuel contained in 60 mm diameter graphite spheres (Pebbles). Redrawn based on elements from *HTGR Technology Course for the Nuclear Regulatory Commission*; May 24–27, 2010, *Module 6a: Pebble Bed HTGR Core Design Description*. <https://art.inl.gov/NGNP/Training%20Modules%20%20HTGR%20Fundamentals/Forms/AllItems.aspx>.

For electricity generation a direct helium power conversion cycle (Brayton) or an indirect steam cycle (Rankine) has been studied. The Brayton cycle leads to increase efficiencies ($\sim 50\%$) if the outlet gas temperature is increased (up to 900°C typically) while for the Rankine cycle ($\sim 41\%$) gas outlet temperature (typically 750°C) is dictated by the ideal steam conditions ($\sim 540^\circ\text{C}$) based on conventional steam power turbines. Simplified system diagrams (Reitsma, 2010) are included in Fig. 3. Many alternatives and combinations of cycles are possible that may use intermediate heat exchangers (IHX) or heat storage to do co-generation (electricity and heat) or to deliver process heat at multiple temperatures.

In Table 1 the typical core parameters and maximum thermal power outputs for pebble bed gas cooled reactors are given for these two power conversion cycles. The study ensures compliance to the modular design approach and specifically the maximum fuel temperatures (1600°C was used) in accident conditions (Said et al., 2006).

The majority of pebble bed designs actively being pursued today has a cylindrical core shape and is making use of the indirect steam cycle. The typical layout and features of the reactor and nuclear steam supply system is shown in Fig. 4 (derived from Reitsma, 2010). The pebble bed core and graphite reflectors are shown. The control rods drive mechanism and secondary shutdown storage-and release systems are situated above the core and will insert control rods or release small absorber spheres into holes in the side reflector graphite blocks. This prevents damage to the fuel spheres (no insertion into the pebble bed) but also limit core diameter to ensure adequate shutdown. The fuel spheres are unloaded through a central defuel chute and can be reloaded to the top of the pebble bed. This is all contained within the reactor pressure vessel (RPV). The positioning of the control rod drives within the RPV exclude control rod ejection as a postulated accident in this example.

The reactor is coupled via a cross vessel duct to the steam generator and helium blower. The RPV, cross connecting vessel and steam generator vessel contain the helium coolant typically pressured between 4 and 9 MPa. The helium coolant is circulated in the outer volume of the cross duct, through the bottom reflector, up in the side reflector and then downwards through the pebble bed. The hot gas is collected below the core in the hot gas plenum and returned to the steam generator through the inner hot gas pipe where it flows downwards on the outside of the steam generator tubes returning on the inner boundary of the steam generator vessel

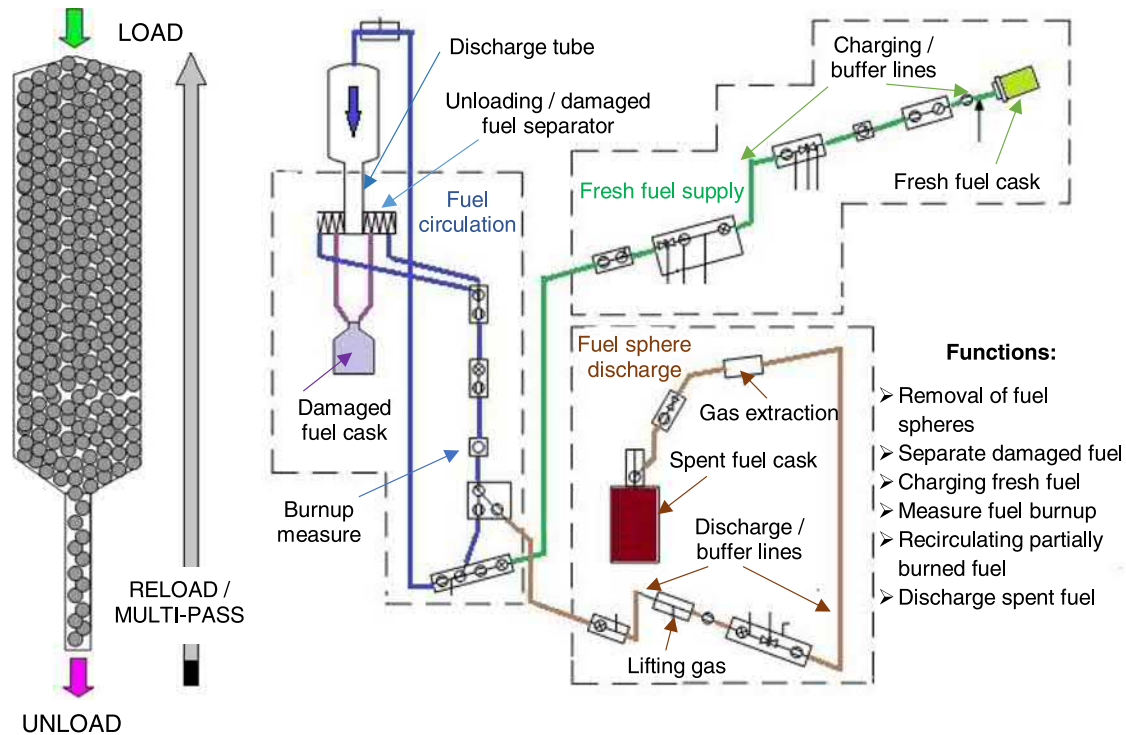


Fig. 2 Fuel sphere load and multi-pass concept (left) and fuel handling system (on the right).

to the circulator. The steam generator is positioned below the active core height to limit water and steam ingress in the case of a tube leak or rupture. The water enters the steam generator from below with the steam line exiting at the top, feeding the steam power turbine.

Operations

The pebble bed gas cooled SMR has several characteristics that enhances its safety but specifically also simplify operations, since margins are large and the response typically slow. The heat transfer fluid used is helium, an inert compressible gas with no phase change (no sudden changes in heat removal) and excellent heat transfer properties. The large helium temperatures difference across the core ($\Delta T = T_{\text{outlet}} - T_{\text{inlet}}$) requires a smaller coolant mass flow resulting in smaller pumping power needs. The large amount of graphite used as moderator, reflector and core structural material has a large thermal capacitance that, combined with the low power density, result in very slow temperature transients. This simplify the control system and allows for time before any operator actions are needed. Although these salient features suggest ease of operation with much simpler control and operator actions, it is better to rather look at actual experience.

The AVR 15MWe operated for 21 years till 1988. It achieved a high average availability factor of 66.4% over its operational life, even though it was used as a test reactor to perform bulk tests of different fuel types (Hittner, 2019), and used to do many safety demonstrations. It achieved a maximum availability factor of 92% in 1976 (Baumer et al., 1990). These are exceptionally high values for that time. The irradiation dose to personnel was very low (compared to the levels at other plants at that time), and the plant demonstrated it flexibility and robustness running on different fuel (HEU-Thorium, LEU, etc.), surviving oil and water ingress accidents, and performing many safety demonstrations. "Operation of the plant presents no problems and makes no high demands on the operating personnel. All the faults which occurred could essentially be eliminated with our own plant crew" (Baumer et al., 1990). Now under decommissioning, the plant also contributes towards the future knowledge needed in this area.

The THTR 300 operated for only 4 years (Hittner, 2019). It was built before the modular HTGR principles were generally accepted and had, for example, a relatively higher power density at 6 MW/m^3 (compared to the HTR-Modul value of 3 MW/m^3), made use of 42 absorber rods directly driven into the pebble bed core, and used a large pre-stressed concrete reactor pressure vessel (PCRV). It therefore required active cooling to remove the decay heat. It had good operational behavior and low activity in the primary circuit, but some failures and operational difficulties contributed to the lessons learned for modern pebble bed reactors. Many of the fuel spheres ($\sim 1.2\%$) were damaged due to core control rod insertions into the pebble-bed in the core. The on-line defueling could not operate at full power (due to back pressure design issues) and this required operation at reduced power and flow that negatively impacted on the capacity factor. The best availability factor of 61.7% was achieved in 1987.

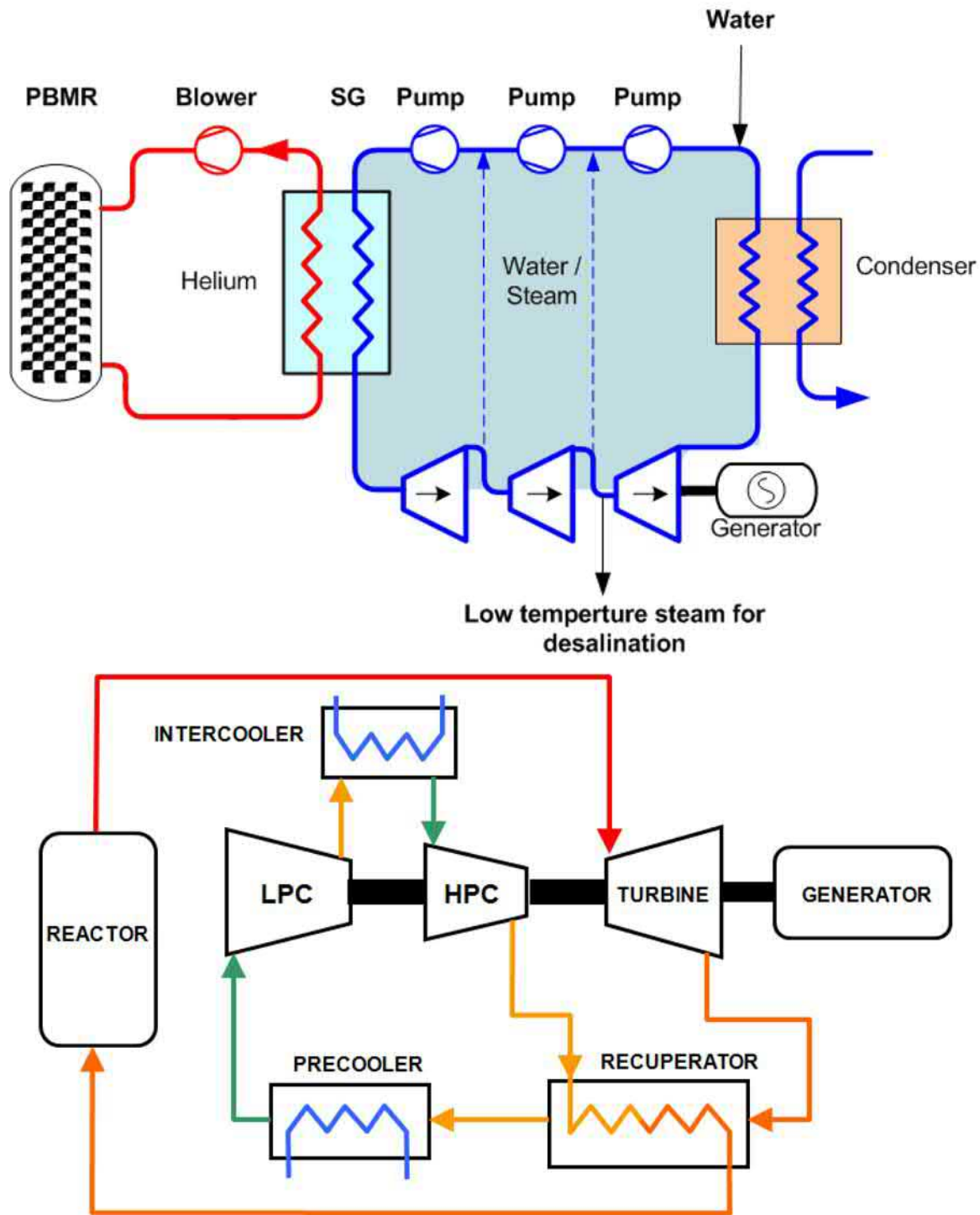


Fig. 3 Simplified power conversion system diagrams for indirect steam cycle (Rankine upper) and direct helium cycle (Brayton lower) with low and high pressure compressors (LPC and HPC). Reitsma F (2010) Presentation made at the PHYSOR 2010 Workshop on “Reactor Physics and Design of Next Build Reactors” - The Pebble Bed Modular Reactor: From V.S.O.P. (Very Superior Old Product) to Generation-IV candidate. 14 May 2010.

In general, the operating experience from the German prototypes has confirmed the good behavior of coated particle fuel under irradiation with very low release of fission products into the coolant gas (Gerczak, 2021). It has also confirmed “easy control, high thermal inertia and significant operating safety margins” (Kugeler and Zhang, 2019).

The HTR-10 pebble bed test reactor in China reached first criticality in December 2000, and by January 2003, after steady increase from initial 30% operations, reached the full 100% rated power and 700 °C outlet temperature, and then sustained

Table 1 Typical conditions and maximum power output for modular pebble bed gas cooled reactors.

Design	Cylindrical		Annular	
Core height [m]	11.0		11.0	
Inner core diameter [m]	–		2.0	
Outer core diameter [m]	3.0		3.7	
RPV outer diameter [m]	5.7		6.4	
Variant	Brayton	Rankine	Brayton	Rankine
Inlet temperature [°C]	500	250	500	250
Outlet temperature [°C]	900	750	900	750
Thermal power [MW _{th}]	220	250	425	464
Mean power density [MW/m ³]	2.83	3.21	5.07	5.54
Helium mass flow rate [kg/s]	105.9	96.24	204.5	178.6
Pressure-drop over the core [bar]	1.27	0.84	3.55	2.16

From Said NB, Lohnert G, Buck M, and Bernnat W (2006) The impact of design on the decay heat removal capabilities of a modular pebble bed HTR, *Nuclear Engineering and Design* 236: 648–656.

72 h of full power operation satisfying the technical specifications and design requirements. Many planned commissioning tests have been performed to test the fuel while several tests were conducted to demonstrate the passive safety characteristics. “For the future INET first plans to raise the outlet temperature to 750°C in the HTR-10 core with the same sphere fuels and to replace the steam generator with a gas turbine power conversion unit. The new designed reactor is named HTR-10GT” (Lang et al., 2017). More operational lessons learned has been summarized by (Beck and Pincock, 2011).

In Table 2 the main operating conditions of the three pebble bed reactors that was built and some of the leading pebble bed SMR designs are summarized in Table 2 (Baumer et al., 1990; IAEA, 2017; Hittner, 2019; IAEA, 2020). It should be pointed out operational experience are only for indirect steam cycles with no experience with helium gas turbine direct cycles.

The high-temperature gas-cooled reactor pebble-bed module (HTR-PM) demonstration power plant (Zhang et al., 2016) at the Shidao Bay site in Rongcheng, Shandong Province, China, has completed cold commissioning of both units in the last quarter of 2020 (World Nuclear News, 2020). First criticality and power operations are expected in 2021. Even though the same ‘easy and non-demanding’ behavior of the AVR is expected, a positive operational experience will be invaluable to the prospects of pebble bed gas cooled reactor deployment.

Safety characteristics

The safety philosophy and characteristics of modular HTGRs is well documented and apply to both pebble bed and prismatic designs. It is very well summarized by (Ball, 2014): “No fuel failure for loss of coolant flow and coolant accidents, or for all foreseeable reactivity events, even with no corrective actions.”

Within the framework of an IAEA coordinated research project on Modular HTGR safety design, Reitsma et al. (2016) made a detailed review of all the safety characteristics of HTGRs summarized here. Some of these characteristics were already mentioned such as the use of an inert single-phase helium gas and the slow thermal transients due to the large mass (heat capacity) of graphite. Other favorable inherent safety characteristics includes the use of high quality ceramic coated particle fuel and the strong negative reactivity coefficients.

Most of the other favorable safety features are obtained by the correct design choices mostly to ensure no core meltdown or significant radionuclide release are possible even in the event of depressurization of the helium coolant and loss of all active forced convection cooling systems. This can only be achieved by selecting a low power density and relying on natural means only to transport the decay heat away from the fuel, i.e. by radiation, conduction and the natural convection outside the RPV. To ensure a relatively large heat rejection surface area a long slender core and uninsulated RPV are typically used. The heat is removed to cooling panels outside the reactor and the building structures and surrounding earth can serve as the ultimate heat sink if the system fails. During loss of active cooling transients, the maximum fuel temperature must remain below levels (~1600 – 1800 °C or even higher, EPRI, 2020) where significant fuel damage may occur with some margin. It is also important to note that only a very small fraction of the fuel (typically < 5%) will be at these high temperatures, while the rest of the fuel is substantially cooler (a significant fraction may be lower than the normal peak operating temperatures).

The critical element of HTGR safety is the TRISO (TRi-structural ISOtropic) coated particle (CP) fuel. It is capable to contain almost all fission products for the expected operating and postulated accident temperatures (Gerczak, 2021). The core design must ensure sufficient margins in the operating and accident conditions to prevent damage to the fuel. The most important of these conditions is the maximum accident temperatures that, together with the other conditions, being the fuel operating temperatures, fast fluence, burnup and maximum power density, must be bounded by the fuel design and test envelope (EPRI, 2020). No large early release (~first 24 h) of fission products due to coated particle failure is therefore possible (not a credible scenario).

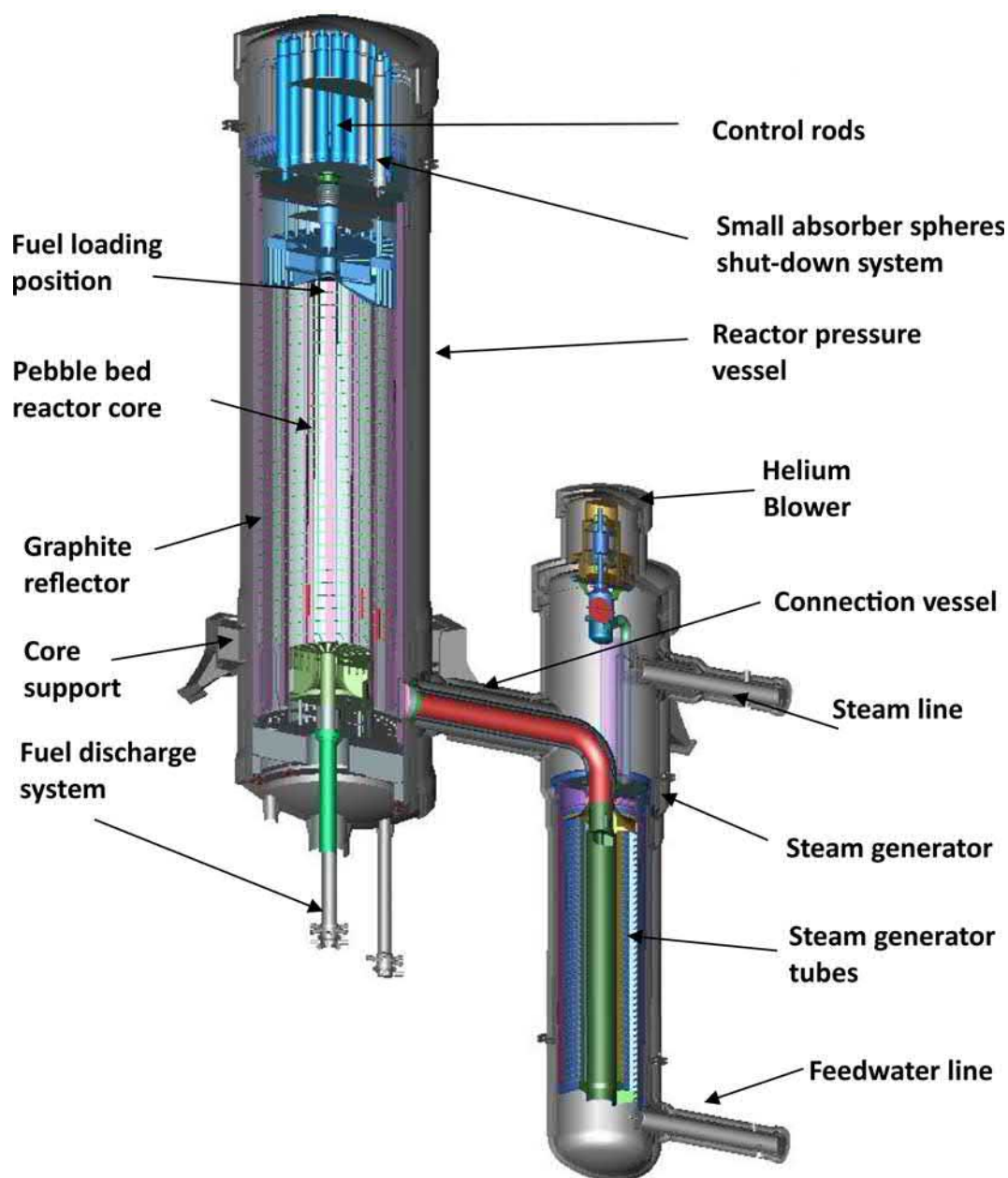


Fig. 4 Typical layout of the reactor and nuclear steam supply system (Rankine cycle) for a pebble bed gas cooled reactor. Redrawn based on the HTR-Module Reitsma F (2010) Presentation made at the PHYSOR 2010 Workshop on “Reactor Physics and Design of Next Build Reactors” - The Pebble Bed Modular Reactor: From V.S.O.P. (Very Superior Old Product) to Generation-IV candidate. 14 May 2010.

In the worst-case scenario with the loss of all the coolant (and therefore no active cooling) the core will heat up and reach a maximum typically between 1–3 days after the accident and then starts cooling down. The increased fuel temperatures will cause the release of some additional fission products through increased diffusion through the TRISO coatings even if no additional failures occur. This may contribute to an increased radionuclide to the environment, called the delayed release. The modular HTR design ensures that these values remain far below the regulatory limits.

A HTGR core contains billions of coated particles and the failure mechanisms of CP fuel are decoupled and totally independent. When CP failures occur, they have been shown to develop after a long time (hours and days) at very high accident temperatures. In the German HTR fuel irradiation program over several decades no additional coated particle failures were found due to accident temperatures of up to 1600 °C (IAEA, 2010). The expected failure rate can only be expressed as a statistical probability dependent on the burnup, temperature and time at temperature. The failure of one coated particle cannot lead to the failure of a neighboring CP. A CP failure also has no effect on the cooling of the fuel as a failure will not change the heat removal path. The amount of fission

Table 2 Main operating conditions of pebble bed gas cooled reactors and designs.

Design	Power [MW _{th}]	Power [MW _e]	Power density [MW/m ³]	Inlet/outlet temperature [°C]	Helium pressure [bar]	# Of fuel spheres	²³⁵ U enriched [%]	Cycle
AVR	46	15	2.6	275/950	10.8	100,000	7,10,16, 93	Steam
THTR-300	750	296	6	250/750	39	675,000	93/Th	Steam
HTR-10	10	~3	2	250/700	30	27,000	17	Steam
HTR-PM	2 × 250	210	3.2	250/750	70	420,00	8.5	Steam
HTR-Module	200	80	3	250/700	60	360,000	7.8%	Steam
Xe-100	200	82.5	4.9	260/750	60	220,000	15.5	Steam
HTMR-100	100	35	3.6	250/750	40	150,000	LEU/Th	Steam
AHTR-100	100	50	4.9	406/1200	90	110,250	LEU	Combined
PBMR-400	400	165	4.8	500/900	90	452,000	9.6	Direct

From Baumer R, et al. (1990) AVR: Experimental High-temperature Reactor; 21 Years of Successful Operation for a Future Energy Technology / Association of German Engineers (VDI), The Society for Energy Technologies (Publ.). Dusseldorf: VDI-Verlag GmbH; Hittner D (2019) *History of HTGR, Lecture 6 at the Joint ICTP-IAEA Workshop on Physics and Technology of Innovative High Temperature Nuclear Energy Systems I (smr 3281)*, 14 – 18 October 2019, Trieste, Italy; International Atomic Energy Agency (2017) *Industrial applications of nuclear energy*, IAEA Nuclear Energy Series No. NP-T-4.3, Vienna, (2017); IAEA (2020) *Advances in Small Modular Reactor Technology Developments – 2020 Edition*, https://aris.iaea.org/Publications/SMR-Book_2020.pdf.

products that can potentially be released when a failure occurs is very small, and many CP will need to fail to be of any consequence. The fuel failures exhibit no cliff edge effects.

The large negative temperature coefficient (and large fuel temperature margins) implies that the Doppler, or fuel temperature reactivity coefficient, acts promptly thus stabilizing the nuclear chain reaction. The on-line refueling in the pebble bed reactor also ensures very small excess reactivity at any given time.

The HTGR designs must guard against water ingress due to steam generator tube leaks or ruptures that can cause reactivity excursions and limited graphite corrosion and fission product release (washout of damaged CP). In such conditions, a standard scram and isolation of the reactor from the steam generator is performed. The prevention of massive air ingress and its corrosion effect on graphite and fuel above 400 °C has also been studied and found to be a very low probability event, and design and operational measures are included to mitigate it.

Areas of application

The pebble bed gas cooled SMRs as part of HTGRs were always envisaged for both electricity generation and process heat applications. The higher outlet temperatures of pebble bed reactors already lead to higher efficiencies (~41%) compared to water cooled reactors (~34%) and match the efficiencies of conventional fossil fueled power plants (for more details see [Pioro et al., 2021](#)). As explained above the use of direct cycle gas turbines can increase this substantially (~50%). The electricity market is only about 18% of the total energy market with the rest used for heat and transportation. HTGRs are ideally suited to make a carbon free contribution to all these energy markets ([Bragg-Sitton and Boardman, 2021](#)).

Due to the high outlet temperatures HTGRs can match the range of temperatures required for process heat applications in most industries or to produce synthetic fuels to replace gasoline, diesel and kerosene for transportation in a carbon dioxide neutral way. In the International Atomic Energy Agency report on Industrial applications of nuclear energy ([IAEA, 2017](#)) as well as in ([Bragg-Sitton and Boardman, 2021](#)) a comprehensive summary of all these potential markets is provided with HTGRs and pebble bed reactors specifically referenced.

The AVR reactor ([Baumer et al., 1990](#)) operated at outlet gas temperatures of 950 °C through most of its operating life demonstrated the potential for pebble bed gas cooled reactors to deliver high temperature heat for industrial processes and hydrogen production. In fact, the melt-wire gas temperature measurements above the core and helium core bypass studies have shown that the average gas temperatures above the pebble bed was probably as high as ~1136 °C and even higher maximum temperatures ([Viljoen et al., 2008](#)). This shows the potential of the fuel and core structures to tolerate even higher operating temperatures in the future, although many other aspects still need to be developed and tested. Additional information is provided in [Fütterer et al. \(2021\)](#).

Development status

A summary of the main pebble bed designs is given in [Table 3](#). More details on each of the designs can be found in the 2020 edition of *Advances in Small Modular Reactor Technology Developments* ([IAEA, 2020](#)).

Most of these designs closely follow the modular principles established with the HTR-Module design ([Reutler and Lohnert, 1984](#)). The PBMR400 design propose an annular core and direct cycle while the rest are all cylindrical configurations and use

Table 3 Summary of pebble bed gas cooled reactor designs.

<i>Design</i>	<i>Developer</i>	<i>Development status</i>
HTR-10 (10 MW _{th})	Tsinghua University, China	Test reactor in operation since 2000
HTR-PM (2 × 250 MW _{th})	Tsinghua University, China	Under commissioning. Commercial demonstration in 2021
Xe-100 (200 MW _{th})	X Energy, LLC, United States of America	Basic design development under way
HTMR-100 (100 MW _{th})	Steenkampskraal Thorium Ltd., South Africa	Concept design
AHTR-100 (100 MW _{th})	Eskom Holdings SOC Ltd., South Africa	Concept design completed; R&D work on hold since 2019
PBMR-400 (400 MW _{th})	Pebble Bed Modular Reactor SOC Ltd., South Africa	Preliminary design completed; test facilities demonstration; project stopped in 2010
RDE (10 MW _{th})	National Nuclear Energy Agency (BATAN), Indonesia	Detail design development is in progress
HTR-Module (200 MW _{th})	Interatom/Siemens in Germany	The reference / first modular pebble bed design developed in the 1980s and never constructed

indirect cycles (Matzner, 2004). Two designers propose a once-through fuel management system (AHTR-100 and HTMR100) while the others propose a multi pass fuel management scheme (average number of passes between 6 and 15). The HTR-PM is already under commissioning (World Nuclear News, 2020) and will be the first Generation-IV SMR to operate in the world. It is expected to increase confidence in the pebble bed SMR technology and encourage many countries to proceed with planned HTGR activities. Development of the Xe-100 is also progressing well and engagement with the US regulator and start of construction is foreseen in the next few years (IAEA, 2020).

Further insight into the different historical reasons why pebble bed gas cooled reactors (and more generally HTGRs) were not further developed and commercialized can be found in Fütterer et al. (2021).

Objectives of ongoing development work

In the near term the goal is full-scale demonstration and commercialization of pebble bed gas cooled SMRs based on the HTR-PM commissioning and operational feedback. Despite past experience and high technology readiness levels many of the suppliers that were available in the 1980s and 1990s are no longer available, and a new supply chain is needed. For the HTR-PM all major systems and components were manufactured in China. Many test facilities were also built to perform full-scale engineering verification experiments at hot conditions in a helium environment. This includes the main helium blower, steam generator module, the fuel-handling and spent fuel storage systems, control and shutdown systems and the helium purification system (Zhang et al., 2016). These experiments exposed and solved many engineering technical problems.

Almost all the designs presented in Table 3 are based on known technology and systems with a relatively high technology readiness level that had been demonstrated before or in the HTR-PM project. However, the PBMR400 makes use of a direct cycle for which a helium turbine is required. When the project was executed in the 2000s this represented a huge challenge, and substantial development costs were incurred. No such large (400 MW_{th}) helium turbine has been built before and the high temperature (900 °C) presented material challenges. The proposed blade cooling solutions represent a direct cold helium bypass and thus a substantial loss of efficiency. New materials are needed to be qualified if the direct cycle is to be a viable option.

The niche market for pebble bed gas cooled SMRs must surely be cogeneration and the process heat market. Many market studies have been performed to look at hybrid energy systems or integration with existing industrial complexes to provide steam and electricity. Hydrogen production is also increasing in importance and several demonstration plants are under development. The development of innovative intermediate heat exchangers (IHX), especially of compact printed circuit designs, has been studied in the past for hydrogen production, but is also important for cogeneration and thermal storage (with different coolant options such as helium-helium, or helium-molten salt). The developments using existing materials are being carried out while new materials are under development for the future.

The AHTR-100 design is really a VHTR design with outlet helium temperature of 1200 °C. It also introduces novel power conversion concepts that include a direct cycle, thermal storage and bottom indirect cycle, that would require further research and development. The use of a concrete reactor vessel also requires new innovative decay heat removal such as the use of a passive heat pipe system that needs additional R&D.

Major challenges and R&D needs

Licensing is often singled out as one of the main challenges to deployment new advanced reactors. However, pebble bed gas cooled reactors have been licensed and operated before, so what has changed? “The answer is two-fold: In the last 30 years the safety and licensing requirements have been strengthened substantially but in the process it also became more light water reactors (LWR) technology specific with the consequence that it is more difficult to interpret and apply to HTGRs. The second is the change in the safety philosophy used in all modern HTGR designs where many of the safety systems required in LWRs are made obsolete” (Reitsma,

2017). It is possible to license pebble bed reactors under LWR requirements and licensing approach. This was the case in Germany before and recently again with the HTR-PM in China. However, in both cases agreement has been reached before license applications were made on specific exceptions and additions to the requirements and approach.

In recent years excellent progress has been made with the development of advanced non-water cooled reactor licensing. For example, the US NRC published RG 1.232 (NRC, 2018) that not only attempt to make requirements technology neutral, but also includes a specific appendix on Modular High-Temperature Gas-Cooled Reactor Design Criteria. Similar efforts have been made by regulators around the world that should enable the appropriate focus in licensing HTGRs.

The licensing of pebble bed reactors to be used for cogeneration and directly coupled to industrial processes for process heat applications may present a specific challenge. Potential transients and external hazards must be included in the design considerations of both the nuclear and industrial plant, or innovative decoupling systems needs to be applied. Studies on safety, technology and economics are being performed.

The research and development need for pebble bed reactors (VHTRs) are well summarized in the R&D Outlook for Generation IV Nuclear Energy Systems (GIF, 2018). The TRISO fuel technology was successfully reestablished to produce excellent fuel quality both in China (Freis et al., 2020) and the USA (EPRI, 2020). The fuel test envelope has been expanded in burnup, power density, neutron fluence, irradiation temperatures and accident temperatures. R&D in the future may look at expanding this test envelope further by investigating alternative coatings and matrix materials.

Development for higher temperature applications require new materials and instrumentation. High temperature metals will allow new designs of vessels, pipes, valves, gas turbines and IHX that can withstand the higher operating and accident conditions. R&D on improved graphite grades is expected to result in better mechanical and thermal properties at higher temperatures and fluences while showing better resistance to air and water ingress. Ceramics and composite materials for control rod cladding or in-core structural materials are also needed.

Finally, R&D on improved software codes and simulation methods are continuing to reduce uncertainties and improve the verification and validation status of these codes through benchmarking and comparisons with experiments and plant data.

See Also: The High Temperature Gas-Cooled Reactor.

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Heavy Liquid Metal (HLM) Cooled Fast Small Modular Reactors

Craig F. Smith, Department of Physics, Naval Postgraduate School, Monterey, CA, United States

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Glossary

AKME Irkutskenergo - AKME-Engineering Joint Stock Company (JSC)
ALFRED Advanced Lead-cooled Fast Reactor European Demonstrator
ANL Argonne National Laboratory
BREST Bystry Reaktor so Svintsovym Teplonositelem or “Fast Reactor with Lead Coolant”
CLEAR China Lead-based Reactor
CP-1 Chicago Pile-1
CPS Control and Protection System
ELFR European Lead-cooled Fast Reactor
ENHS Encapsulated Nuclear Heat Source
eV Electron volt (a unit of energy equivalent to 1.6022×10^{-19} J)
FDS Fusion Design Study
GIF Generation IV International Forum
HLM Heavy Liquid Metal
IAEA International Atomic Energy Agency
INEST Institute of Nuclear Energy Safety Technology
keV Kilo electron volts (= 1000 eV)
kW Kilowatt
LBE Lead bismuth eutectic
LFR Lead-cooled fast reactor
LFR-AS-200 200 MWe LFR with Amphora-Shaped Inner Vessel
LFR-TL-X Transportable long-lived LFR with “X” MWe output
LANL Los Alamos National Laboratory
LLNL Lawrence Livermore National Laboratory
MA Minor Actinide
MeV Million electron volts
MOX Mixed Oxide
MWe Megawatts electric
MWth Megawatts thermal
NIKIET N.A. Dollezhal Research and Development Institute of Power Engineering
OECD-NEA Organization for Economic Co-operation and Development - Nuclear Energy Agency

Reactivity A measure of the deviation of the fission chain reaction from a critical state – a state in which the ratio of the rate of neutron generation to the rate of neutron loss (usually denoted as k_{eff}) equals 1.000. Reactivity is $(k_{\text{eff}} - 1)/k_{\text{eff}}$.

Reactivity Coefficient The change in reactivity per unit change in constituent temperature or density

SEALER Swedish Lead-cooled Reactor

SMR Small Modular Reactor

SSTAR Small Secure Transportable Autonomous Reactor

STAR Secure Transportable Autonomous Reactor

STSG Spiral Tube Steam Generator

SVBR Svintsovo-Vismutovyi Bystryi Reaktor or “Lead-Bismuth Fast Reactor”

UNIST Ulsan National Institute of Science and Technology

URANUS Universal, Robust, Accident-tolerant, Non-proliferating, Ultra-lasting, Sustainer

Introduction

Since the earliest days of nuclear reactor development, small reactors – generally considered reactors with power less than 300 MWe – have seen wide application. A principal reason for this is the fact that initial development and demonstration of new reactor technologies generally start with small scale demonstration reactors that can form the basis for larger implementations of the particular reactor type. Additionally, there are many special applications – aside from large scale grid electricity production – in which small systems are most appropriate. These include reactors designed for scientific research and testing, reactors for power supply in remote or isolated locations, reactors for training and radioisotope production, and reactors for ship propulsion. For purposes of electricity production, there has long been a trend toward larger plant sizes to achieve economy of scale since nuclear plants are very capital-intensive systems. However, in recent years there has been progressively increasing interest in the development of small modular systems to address a different set of objectives including modularity, transportability, factory fabrication, passive safety and decentralization/localization of power generation facilities – hence the growing interest in Small Modular Reactors, or SMRs (Smith, 2009).

One of the most important defining characteristics of a reactor technology is the mechanism chosen for heat removal and reactor cooling; a wide variety of coolants have been considered including gases such as helium or carbon dioxide; liquids such as water, heavy water, molten salt, sodium and potassium; and the heavy liquid metals mercury, tin, lead and the eutectic alloy of lead and bismuth known by its acronym LBE. Additionally, technologies such as heat pipes can be used for heat removal without relying on the circulation of bulk coolant fluids.

Another defining reactor technology characteristic is the neutron energy spectrum of the reactor. The two most prominent options are reactors operating at thermal neutron energies (centered around ~ 0.1 eV) and those operating with neutrons in the fast-energy range (i.e., from ~ 100 keV to greater than 1 MeV). Coolants and other reactor materials (e.g., moderators) containing high concentrations of light elements (e.g., hydrogen in water or carbon in graphite) are effective in slowing down fission neutrons to thermal energies, and therefore are favored for thermal reactors. On the other hand, coolants and other reactor components in which materials comprised of light elements are minimized in favor of those containing heavy elements are used for fast reactors. Liquid metals are a category of such materials, and heavy liquid metals (HLMs), in particular, can be highly effective as coolants for fast spectrum reactors. Advantages associated with fast spectrum reactors include the following:

- Greatly enhanced fuel utilization.
- Toxicity reduction of spent fuel and waste.
- Safety advantages since fast reactors can operate at low pressure with large thermal inertia and negative power and temperature coefficients of reactivity (IAEA, 2006).
- Potential dramatic extension of fuel lifetime and reduction of refueling requirements.

The last of these advantages, enabled by the ability to breed and burn fuel constituents, is of particular importance to many HLM-cooled fast SMR concepts which have fuel designs that in some cases result in core lifetimes of up to 30 years and can therefore operate for such a period without the need for refueling.

This chapter focuses on the topic of HLM-cooled fast SMRs.

History

The first operational nuclear reactor was the graphite-moderated thermal reactor known as Chicago Pile-1 (CP-1) which reached first criticality on December 2, 1942 as part of the wartime Manhattan Project. Following the end of World War II, there was a great deal of exploratory research on reactor technologies, and as part of that effort, the first fast spectrum reactor – known as Clementine – was developed (LANL, 1993; LANL, 2018). Clementine, first operated in 1946 at Los Alamos National Laboratory, was not only the first

fast reactor; it was also the first reactor fueled by Plutonium, and it was the first reactor cooled by a HLM (mercury). At 25 kW power, it can be considered the first HLM-cooled fast SMR.

Mercury had been chosen for Clementine in order to provide a coolant that would sustain the desired fast neutron spectrum, but also because it is a liquid at room temperature and would therefore avoid the engineering difficulties of other metal coolants that would need to be maintained above their melting temperature for operational and safety reasons. However, it was found that mercury as a coolant also had several disadvantages including its toxicity, relatively high vapor pressure, relatively low boiling temperature and, perhaps most important, its high neutron absorption cross sections. As a result, following the Clementine reactor experience, mercury was not seriously considered for future development.

On the other hand, lead and its alloy with bismuth LBE¹ were found to have very favorable characteristics for use as coolants in fast spectrum reactors. During the Cold War, the US and the Soviet Union each explored reactor technologies for submarine and surface ship propulsion. While the US declined to pursue HLM coolant technologies due to concerns over corrosion and material compatibility, the Soviet Union (and subsequently Russia) embarked on a substantial development program that ultimately resulted in the production of a fleet of high-performance “Alfa Class” submarines powered by reactors cooled by LBE. Fig. 1 is a photo of the Alfa Class submarine (National Archives, 1983). Said to be the world’s fastest and deepest diving submarine of its time (Military Today, n.d.), a total of 15 reactor cores were built and operated in support of 8 submarines and 2 onshore reactors. In all, ~80 reactor-years of operating experience were accumulated and led to continued Russian interest and technology leadership with regard to HLM-cooled reactors following the dissolution of the Soviet Union.

More recently, interest in HLM-cooled reactor technology has grown significantly with national and industrial initiatives around the world. Current activities, ranging from early pre-conceptual design and analysis to final design, site preparation and construction are taking place in many parts of the world with activities under the Generation IV International Forum (GIF) in Russia, the United States, the European Union, China, Japan and Korea (Alemberiti et al., 2017; Alemberiti, 2021).

In the late 1990s, an effort was initiated by Lawrence Livermore National Laboratory (LLNL) to organize a consortium of US national laboratories, universities and industry to explore SMR concepts that would exhibit characteristics most suitable for small, transportable and highly proliferation-resistant nuclear power systems capable of meeting the growing power demands of many regions of the developing world. The initiative originated the acronym STAR standing for Secure Transportable Autonomous Reactor (Brown et al., 1999). Multiple small reactor concepts were considered, including those cooled by a full range of different coolants - gas, liquid metal, light water and molten salt. Among the most promising concepts resulting from this initiative were HLM-cooled fast reactor systems. Two worth noting in particular are reactor concepts known as the Encapsulated Nuclear Heat Source (ENHS) (Greenspan et al., 2008; Greenspan, 2003; Wade et al., 2002) and the Small Secure Transportable Autonomous Reactor (SSTAR) (Smith et al., 2008).

The ENHS concept envisioned a reactor that achieved the STAR objectives of small size, transportability and high degree of proliferation resistance by implementing a sealed module in which there were no penetrations to provide paths for coolant flow out of or returning to the module. Instead, the sealed module contained all the active reactor components and primary LBE coolant which served to remove the heat from the core by natural circulation. Heat would be transferred to an external secondary-coolant pool by passive, through-the-wall conduction.

The SSTAR concept was similarly based on the achievement of the STAR objectives – proliferation security through the sealed nature of the reactor vessel, very long core life, transportability by virtue of its physical dimensions and capability of autonomous operations, and passive safety through simplicity of design. It should also be noted that a follow-on concept – known as SuperSTAR was subsequently proposed. Details on SuperSTAR can be found at (ANL, 2012 and Sienicki et al., 2011).

Both ENHS and SSTAR are discussed further in “Section Current/recent HLM fast SMR technology developments” of this chapter.



Fig. 1 Alfa class submarine. From National Archives (1983) A Port View of a Soviet Alfa Class Submarine. National Archives Identifier 6383596. Available at <https://catalog.archives.gov/id/6383596>.

¹Note that LBE is the eutectic alloy of bismuth and lead, namely 55.5 wt% bismuth and 44.5 wt% lead (OECD-NEA, 2015).

More recently, a number of HLM-cooled SMR concepts have been conceptualized and are in varying stages of development. These additional concepts are also described further in “Section [Current/recent HLM fast SMR technology developments](#)” of this Chapter.

Heavy liquid metal coolants

Heavy liquid metals have been explored as primary coolants for fast nuclear reactors since the early days of reactor technology development primarily because they can effectively remove heat from reactor cores without appreciably degrading the energy spectrum of the neutrons in the reactor. Candidates that have been considered at various times include tin, mercury, lead and LBE. While mercury was the coolant used in the first fast reactor Clementine, its properties, including toxicity, vapor pressure and high tendency for neutron absorption, resulted in disfavor for concepts developed after this first application. Another heavy liquid metal, tin, has also been considered ([Weeks, 1971](#); [Can, 2006](#)) but, due to its high corrosivity and other factors, has not to date been significantly developed or incorporated into design concepts for HLM-cooled reactor systems.

The two remaining potential coolants, lead and LBE, are the principal HLM coolants being actively incorporated into small fast reactor concepts. While the fundamental characteristics of these coolants are well described in technical detail elsewhere ([OECD-NEA, 2015](#); [Cinotti et al., 2010](#); [Smith and Cinotti, 2016](#)), an overview of some of the advantages and disadvantages of each is provided in the following paragraphs.

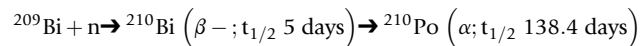
Lead and LBE coolants are highly suitable for fast reactor applications due to properties they share, namely:

- The high atomic masses of lead (~ 207 g/mol) and bismuth (~ 208 g/mol) ([OECD-NEA, 2015](#)) ensure that the fast neutron energy spectrum characteristic of fission neutrons will not be excessively degraded due to neutron scattering interactions with these coolant materials despite the non-negligible level of inelastic scattering in these coolants.
- The very high boiling points of lead (1737°C) and LBE (1630°C) ([OECD-NEA, 2015](#)) make coolant boiling extremely unlikely under any conceivable condition, and these very high temperature margins contribute to the safety and design simplification potential for HLM-cooled systems; in addition, the problem of void formation in the primary circuit from coolant vaporization is virtually eliminated.
- Lead and LBE coolants have relatively benign chemical properties such that they present no rapid chemical reactions with either water or air. This is an important factor in contrast to other liquid metal fast reactors (i.e., sodium cooled systems) which must introduce complex and expensive measures to ensure that the radioactive and chemically energetic primary coolant is reliably isolated from contact with water and air.
- The low saturation vapor pressures of both lead and LBE (about 1×10^{-2} Pa at 800 K or 527°C , ([OECD-NEA, 2015](#))) enable operation of the systems at or very close to atmospheric pressure.
- Lead and LBE exhibit the ability to retain radionuclides, including volatile species such as iodine and cesium, should a fuel failure occur
- As heavy elements, lead and LBE provide inherent shielding of gamma radiation from fission products
- They exhibit favorable neutronic properties, including ([OECD-NEA, 2015](#)):
 - small (fast) capture cross-sections (for small parasitic loss of neutrons);
 - high scattering cross-sections (for small leakage of neutrons from the core);
 - small energy loss per elastic collision (to limit spectrum softening (moderating) effects)
- The thermodynamic properties of lead and LBE are favorable in enabling passive cooling by natural circulation. HLM-cooled fast SMR concepts generally rely on this natural circulation process for shutdown heat removal, and some concepts are designed to operate without forced circulation during normal operations, relying instead on natural circulation for operational cooling.

Some additional characteristics shared by lead and LBE (and which must be overcome by system engineering, material selection and design) include:

- Both coolants have material compatibility issues ([OECD-NEA, 2015](#)), most importantly the potential for corrosive attack on conventional structural materials. While this corrosion potential is relatively low for temperatures below 450°C , it increases with temperature above this level. Design approaches to address this challenge include careful management of coolant chemistry, especially oxygen concentration; restriction of coolant temperatures to limit maximum temperatures below those that present heightened corrosion rates; and use of materials (alloys, coatings and coolant additives) that provide corrosion resistance.
- Both coolants have melting temperatures that present engineering challenges in that the primary coolant must be maintained above the coolant melting temperature throughout the primary system to avoid freezing and the potential for flow blockage or other undesirable impacts. The melting points for lead and LBE are 327°C and 125°C , respectively ([OECD-NEA, 2015](#)). In both cases, this presents engineering and operational constraints; however, this is clearly a greater challenge for lead as a coolant.
- The activation of these coolants in a neutron flux creates undesired radioactive activation products that must be addressed, and the formation of polonium-210 is undoubtedly the contaminant of greatest concern due to its very high radiotoxicity and volatility and its contribution to heat generation within the coolant itself in the case of LBE as a coolant; issues related to generation of polonium-210 represent much greater challenges for LBE as a reactor coolant.

These considerations are central to the choice of lead- versus LBE-cooled reactor designs. The elevated melting temperature of lead in comparison to LBE creates engineering design challenges and also constrains the operational temperature cycle of the primary circuit for lead-cooled systems. On the other hand, the production of polonium-210 as a coolant activation product is primarily a challenge for LBE-cooled systems due to the presence of bismuth in the alloy mixture and the reaction:



Although polonium-210 is also formed in lead, the rate of formation is extremely small whereas in an LBE-cooled reactor, the formation and equilibrium level is greater by nearly 4 orders of magnitude (Can, 2006). Note that polonium-210 decays with a half-life of 138.4 days into lead-206 by an α emission of 5.3 MeV. As a result, polonium-210 represents a substantial heat load in the coolant as well as being highly radiotoxic in the event of release. The polonium production in an LBE-cooled reactor is so high that in an 80 MWe, LBE-cooled accelerator-driven subcritical reactor developed in Europe, the equilibrium polonium inventory was calculated to be 2 kg and represented a decay heat in the coolant that, 5 days after a shutdown, would equal the decay heat power of the fuel itself (Cinotti et al., 2011). As a result, LBE-cooled reactor concepts must incorporate systems, design features and/or operational constraints to address both the toxicological and heat-generation issues related to accumulation of polonium-210.

In comparing lead and LBE as coolants for prospective fast SMR concepts, the balancing between advantages and disadvantages of each have led to the current situation in which both coolant types are well represented in concepts under current development. This is true for HLM-cooled systems of different sizes, including small and very small reactors.

Current/recent HLM fast SMR technology developments

While HLM-cooled small fast reactors have been considered for development since early in the era of nuclear technology (e.g., Clementine in the 1940s (LANL, 2018) and Soviet/Russian reactors for the Alfa class submarine beginning in the 1950s (Military Today, n.d.)), in recent years interest in this technology has rapidly expanded. Current developments now are represented by design and development activities throughout the world and especially in the Russian Federation, the European Union, China, the US, the Republic of Korea and Japan, each of which are members of the Generation IV International Forum (GIF) provisional System Steering Committee for the lead-cooled Fast Reactor (LFR) – see Stanculescu (2021) and Alemberti (2021). Some of these key recent and ongoing developments are summarized in this section.

Note that while the small reactor categorization is generally defined in terms of power (up to 300 MWe), three separate types may fall into this definition. The first type is the category of small, special purpose reactors intended for training, laboratory experimentation and/or radioisotope production. The second type, representing the majority of systems discussed in this section, includes reactors designed for operation in the SMR power range as low as a few MWe and up to 300 MWe. The third type is represented by demonstration reactors that are purposely designed and operated in this lower power range but as prototypes for larger follow-on designs that would not be considered to be in the SMR category. Each of these types is represented in the summaries that follow. Table 1 summarizes the systems discussed in this section.

SVBR-100

The logical follow-on to the experience gained in the development and operation of the Soviet/Russian Alfa-class submarine reactors is the SVBR-100 (Petrochenko, 2011; Zrodnikov et al., 2011). The project is being implemented by a joint venture between the Russian state corporation Rosatom and the private energy company Irkutskenergo - AKME-Engineering Joint Stock Company (AKME). The SVBR-100 is an LBE-cooled reactor with a power rating of 100 MWe. By the end of 2014, most of the capital-intensive and high risk stages of research and design work had been completed, a license for placement (siting) was obtained (2015) and the project moved to the stage of licensing and preparation for construction of a first-of-a-kind demonstration reactor

Table 1 Summary of recent and current HLM-cooled fast SMR concepts.

Country/sponsor	Reactor	HLM Coolant	Power, MWe	Type	Current Status
Russian Federation	SVBR-100	LBE	100	SMR by design	Licensed for placement (siting); continuing development
	BREST-OD-300	Lead	300	Demonstrator	Final Design, Licensed for construction
European Union	ALFRED	Lead	125	Demonstrator	Conceptual design
	SEALER	Lead	3–10	SMR by design	Conceptual design
China	CLEAR-M	Lead	10	Experimental	Pre-conceptual
United States	ENHS	LBE	47/68	SMR by design	Discontinued Development
	SSTAR	Lead	20	SMR by design	Discontinued Development
US-Europe	Hydromine LFR-AS-200	Lead	200	SMR by design	Conceptual design
	Hydromine LFR-TL-X	Lead	3–40	SMR by design	Conceptual design
Korea	Micro-URANUS	LBE	20	SMR by design	Conceptual design

in Dimitrovgrad of the Ulyanovsk region of Russia. (Petrochenko et al., 2017) In preparation for construction and operation of the demonstration plant, work is underway to analyze IAEA safety requirements, complete required regulatory documents of the Russian Federation, clarify the technical and economic parameters of the project, and optimize design solutions for a follow-on serialization of the SVBR-100. A description of SVBR-100 demonstrates that this concept is closely aligned with and benefits from the prior development and operational experience gained from the Soviet/Russian submarine propulsion HLM reactor program. **Table 2** provides an overview of selected technical parameters of the SVBR-100 reactor.

The SVBR-100 is designed for use in remote, isolated, or coastal locations, or for dedicated industrial applications. It also envisioned the ability to provide a variety of different outputs such as electricity, process heat, or desalination. (Smith and Cinotti, 2016) **Fig. 2** provides a sketch of the SVBR-100.

BREST-OD-300

The BREST-OD-300 is a 300 MWe demonstration reactor cooled by lead intended as a prototype for future larger-scale reactors for the production of electricity to the grid. This system is one of the reference designs adopted by the Generation IV International Forum provisional System Steering Committee for the Lead-cooled Fast Reactor. (Alemberti et al., 2017) **Table 3** provides selected technical parameters of the BREST-OD-300, and **Fig. 3** provides a sketch.

Notable characteristics of the BREST reactor are the use of a multi-layer metal-concrete vessel with integral layout in which the lead coolant and the primary circuit equipment are located inside the vessel – thus eliminating the potential for the loss of coolant accident. The primary circuit is organized using free coolant levels, and since there are no shutoff valves in the primary circuit, coolant circulation cannot be lost. The design provides for a passive emergency cooling system with natural air circulation and heat removal to the atmosphere. The design features a unique hybrid loop-pool concept for coolant circulation. The mixed nitride fuel with its high density and thermal conductivity, enables full replenishment of fuel in the core (core breeding ratio ~ 1.05) and compensation of reactivity for fuel burnup. See also Stanculescu (2021), Alemberti (2021), and Adamov et al. (2020) for additional discussions of this concept.

It is important to note that on February 10, 2021 a construction license was issued for the BREST-OD-300 making it among the first Generation IV reactors to achieve this important milestone (Rostechnadzor, 2021).

ALFRED

ALFRED, the Advanced Lead-cooled Fast Reactor European Demonstrator, is being developed as a demonstration reactor for the purpose of testing and qualifying the design, components and procedures for the larger-scale commercial European Lead-cooled Fast Reactor, or ELFR. ALFRED additionally represents a potential early realization of a LFR-SMR (Frignani et al., 2017). ALFRED features a compact design with an emphasis on removability and replaceability of major components, a simplified coolant circulation design and an inner vessel to enclose the hot coolant pool (Falcon Consortium, 2021). **Table 4** provides some high-level parameters of ALFRED, and **Fig. 4** provides a primary system sketch. See also Alemberti (2021) for expanded discussion of this reactor concept and Falcon Consortium (2021) as an online resource website for ALFRED.

SEALER

SEALER (Swedish Lead-cooled Reactor) is a small lead-cooled nuclear battery-type reactor under development by LeadCold Reactors for commercial power production in off-grid applications, especially in remote towns and mining locations in the Canadian Arctic. The reactor is cooled by lead, and is sized for 3 MWe output for remote community power generation and 10 MWe for mining operations. An additional variation, developed to meet the SMR needs identified for the United Kingdom, would be scaled for an output of 55 MWe. (LeadCold, 2021) **Table 5** provides some high-level parameters of SEALER and **Fig. 5** provides a sketch. See also Wallenius (2021) for expanded discussion of this reactor concept.

Table 2 Selected parameters of the SVBR-100 (Compiled from data in (Zrodnikov et al., 2009)).

<i>SVBR-100 parameters</i>	
Thermal Power, MWth	280
Electric power, MWe	106
Primary coolant	LBE
Primary coolant temperature: inlet/outlet, °C	345/495
Fuel type	UO ₂
Uranium loading, kg	~9016
Average U-235 enrichment, %	~16.5
Core lifetime, full power hours	50,000
Time interval between refueling, years	~8
Reactor module dimensions, m	4.53/7.55 (diameter/height)

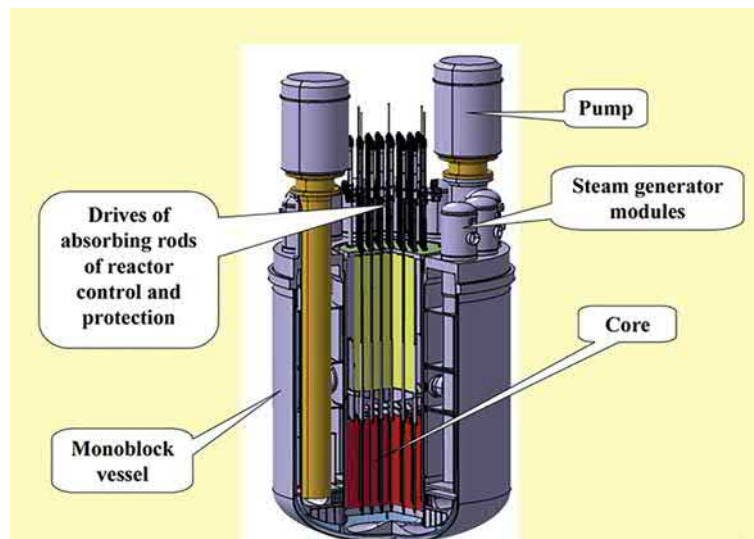


Fig. 2 Sketch of SVBR-100 (Zrodnikov et al., 2009). Zrodnikov A, Toshinsky G, Komlev O, Stepanov V, and Klimov N (2009) SVBR-100 module-type reactor of the IV generation for regional power Industry. In: International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities, IAEA-CN-176-FR09P1132, 7–11 December 2009, Kyoto, Japan. Available at https://inis.iaea.org/collection/NCLCollectionStore/_Public/41/070/41070041.pdf. Permission courtesy of AKME-engineering JSC.

Table 3 Selected technical parameters of BREST-OD-300.

<i>BREST-OD-300 parameters</i>	
Thermal Power, MWth	700
Electric Power, MWe	300
Primary Coolant	Lead
Core diameter, m	2.4
Core height, m	1.1
Core fuel	(U + Pu + MA)N
Average Fuel lifetime, y	5
Refueling interval, y	1
Lead inlet/outlet temperature, °C	420/540
Maximum fuel cladding temperature, °C	650
Net efficiency of power unit, %	42
Design service life, y	30

Adapted from (Smith and Cinotti, 2016).

CLEAR-M

The China Lead-based Reactor (CLEAR) is an advanced family of nuclear energy systems cooled by lead and LBE proposed and under development by the Institute of Nuclear Energy Safety Technology (INEST)-FDS team which engages in R&D of advanced nuclear systems and applications of nuclear technology. The team has operated since the 1980s, and has been supported by a series national programs and international cooperation projects. The CLEAR initiative has included a variety of new systems, most recently the CLEAR-M lead-cooled SMR system. As an SMR, CLEAR-M is designed to be easy to transport and install, inherently safe and economical due to its long refueling period, among other characteristics. Table 6 provides some high-level parameters of CLEAR-M, and Fig. 6 provides a sketch.

ENHS

The ENHS concept envisioned a reactor vessel that was a sealed module in which there were no penetrations to provide paths for coolant flow out of or returning to the module. Instead, the sealed module contained the reactor core, one central and one peripheral control/safety assembly and the primary LBE coolant which served to remove the heat from the core by natural circulation. This module, in turn, was to be immersed in a pool of HLM (i.e., a secondary LBE coolant pool). The transfer of heat from the primary coolant within the ENHS module to the secondary coolant pool outside the module was by conduction through the specially designed wall of the ENHS module. This characteristic was highly innovative and one of the most unique features of the ENHS

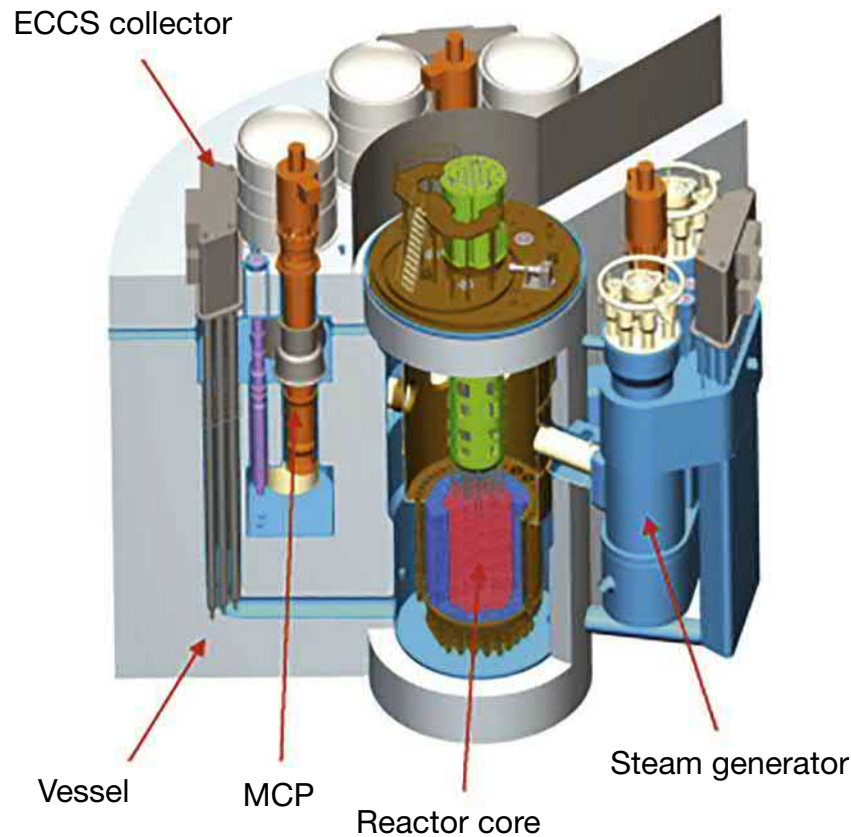


Fig. 3 Sketch of BREST-OD-300 (Lemekhov et al., 2017).

Table 4 Selected summary parameters of ALFRED.

<i>Parameters of ALFRED</i>	
Thermal Power, MWth	300
Electric Power, MWe	125
Primary Coolant	Lead
Net efficiency, %	42
Core inlet/outlet temperature, °C	400/480

Condensed and updated from (Smith and Cinotti, 2016).

reactor. There are no pipes, valves or pumps in the primary and secondary LBE coolant loops. Natural circulation results in the capability of passive load following and autonomous control while eliminating the potential for loss-of-flow accidents. Over the 20 year life of the core, the small burnup reactivity swing ensures that the reactor reactivity control system would require adjustment only once every few years.

ENHS was designed for passive load following and a very high level of passive safety. Its fuel burnup concept enabled a 20-year period of operation without refueling or fuel shuffling. The ENHS modules were to be fabricated, fueled and welded-sealed in the factory. They would be transported to the power plant site with the fuel embedded in solidified LBE; no on-site refueling or fueling hardware is required. At end of life, the ENHS module would serve as a spent fuel storage cask and, later, as a spent fuel shipping-cask for return to a central fuel cycle facility. The module concept enables an essentially sealed module that would be easy to install and replace, and this provides a unique approach to achieving a very high level of proliferation resistance for the reactor concept and its associated nuclear fuel cycle. One configuration of ENHS featured operational heat removal by natural circulation only (ENHS1), and in another, natural circulation provided for shutdown heat removal while operational cooling used also a lift pump achieved by bubbling cover gas into the LBE as it emerges from the top of the core (ENHS2). **Table 7** provides selected parameters of both configurations of ENHS, and **Figs. 7 and 8** provide descriptive sketches of ENHS1.

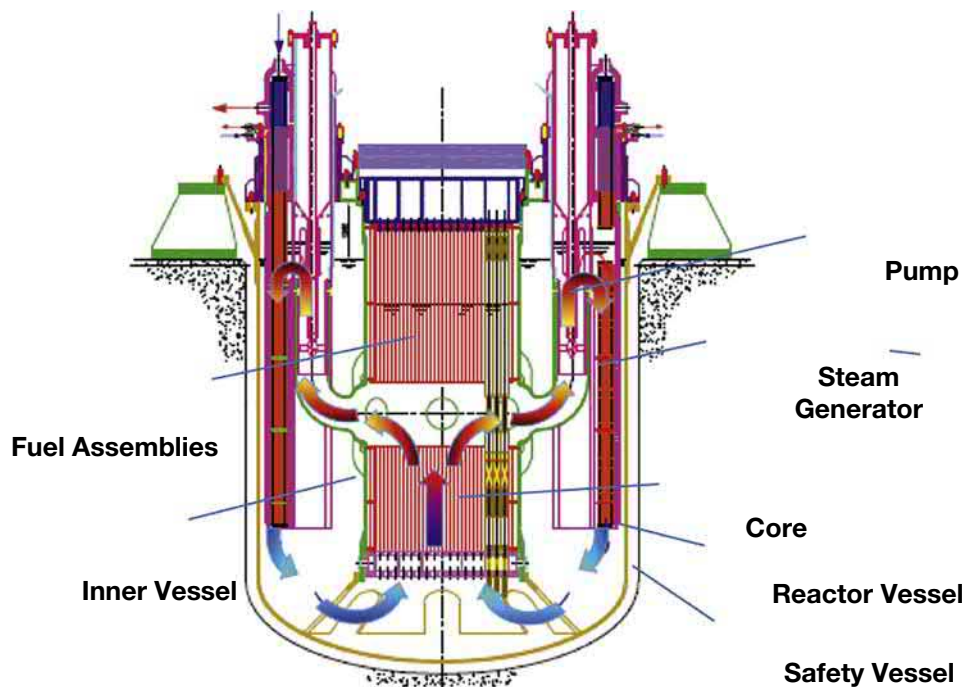


Fig. 4 Sketch of ALFRED (Alemberti, 2021).

Table 5 Selected summary parameters of SEALER.

<i>Parameters of SEALER</i>	
Electric Power, MWe	3, 10 or 55
Primary Coolant	Lead
Net efficiency, %	42
Coolant inlet/outlet temperature, °C	390/432 ^a
Core lifetime, y	10–30

^aFor the 3 MWe SEALER application in Arctic communities.

Compiled from (LeadCold, 2021) and (Wallenius, 2021).

SSTAR

The SSTAR (Small Secure Transportable Autonomous Reactor) concept was introduced as a conceptual design in the early 2000s by a consortium of contributors from US National Laboratories, universities and industry led by Lawrence Livermore National Laboratory with Argonne National Laboratory as the primary design organization. The driving objective behind this initiative was the development of a highly proliferation-resistant small reactor system that would be suitable for deployment to remote locations and developing countries throughout the world. SSTAR is a small natural circulation fast reactor of 20 MWe/45 MWth output. The compact active core would be removed by the supplier as a single cassette and replaced by a fresh core. Key characteristics include transportability by virtue of its physical dimensions, capability of autonomous operation, and passive safety through simplicity of design and the characteristics arising from the inherent properties of lead as a coolant. Some of the unique features of the SSTAR design include a very long core life – up to 30 years – to enable the vessel to be sealed and to keep its fuel inaccessible for its full lifetime of operation. Inherent safety was enhanced by design simplifications such as the reliance on natural circulation for operational as well as shutdown heat removal. Efficiency and compactness was supported by the use of a supercritical CO₂ power conversion system (Smith et al., 2008).

The SSTAR concept was developed to a conceptual design level, but further advancement of the system was limited due to lack of funding following the initial design effort. When the Generation IV International Forum provisional System Steering Committee for the Lead-cooled Fast Reactor was organized, the SSTAR concept was chosen as one of the LFR reference systems despite its legacy status. It remains one of the GIF reference systems. Table 8 provides some high-level parameters of SSTAR, and Fig. 9 provides a sketch. See also Smith et al. (2008), Stanculescu (2021), and Alemberti (2021) for further discussion of SSTAR.

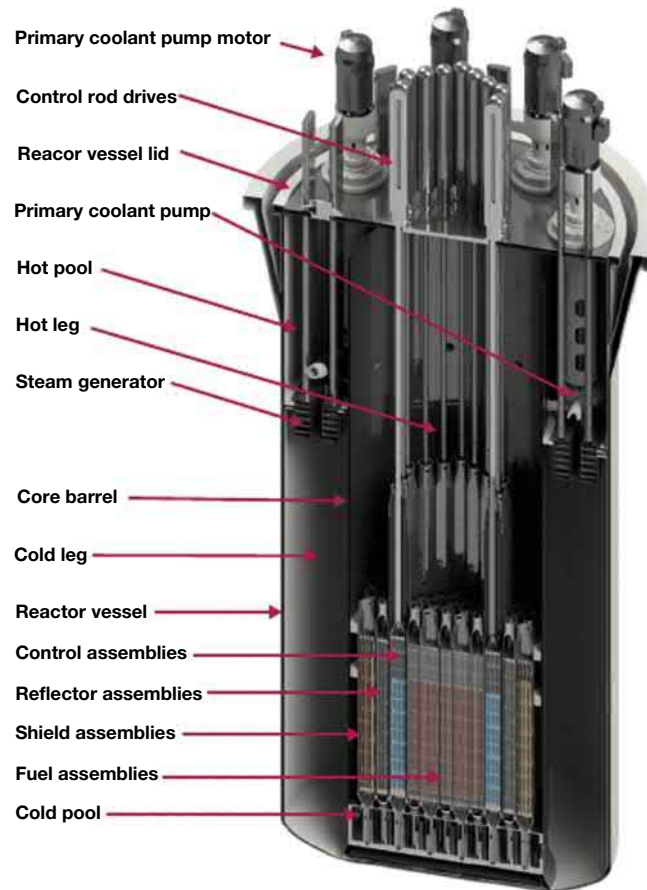


Fig. 5 SEALER primary system sketch (Wallenius, 2021).

Table 6 Selected summary parameters of CLEAR-M.

<i>Parameters of CLEAR-M</i>	
Thermal Power, MWth	35
Electric Power, MWe	10 + 17 MWth
Primary Coolant	Lead
Fuel (Average Uranium Enrichment, %)	UO ₂ (18.5)

Hydromine LFR-AS-200 and LFR-TL-X

The Hydromine LFR-AS-200 is a highly compact and innovative HLM-cooled SMR. The reactor is cooled by lead. Its compact primary system produces more than 1 MWe per cubic meter of primary system volume, a metric about 4 times greater than that of a large sodium-cooled fast reactor (i.e., SuperPhénix), and about 2–3 times greater than current designs of integral pressurized water SMRs.

In the LFR-AS-200, AS stands for Amphora-Shaped, because of the shape of the inner vessel, and 200 refers to the electrical power in MWe (Cinotti et al., 2017). The innovations incorporated into the design exploit the intrinsic properties of the lead coolant and emphasize plant simplification and compactness, while ensuring passive safety.

The LFR-AS-200 is a pool-type fast reactor with six innovative spiral-tube steam generators (STSG), six mechanical pumps, and flag-type control rods. There is no need of in-vessel refueling machinery nor of intermediate loops. Table 9 provides selected parameters of the LFR-AS-200.

A related family of very small reactors (micro reactors) known as the LFR-TL-X series is also under development. This family of reactors capitalizes on many of the same features as the LFR-AS-200, with the main difference being in the number of major components (pumps and steam generators) and their physical arrangement. In this family of microreactors, TL indicates the characteristics of transportability and long-lived core, while X is a variable representing potential electrical outputs of 5, 10 or 20 MWe. The objective of this conceptual design is to implement the simplifications embodied in the LFR-AS-200 in a family of very small reactors with

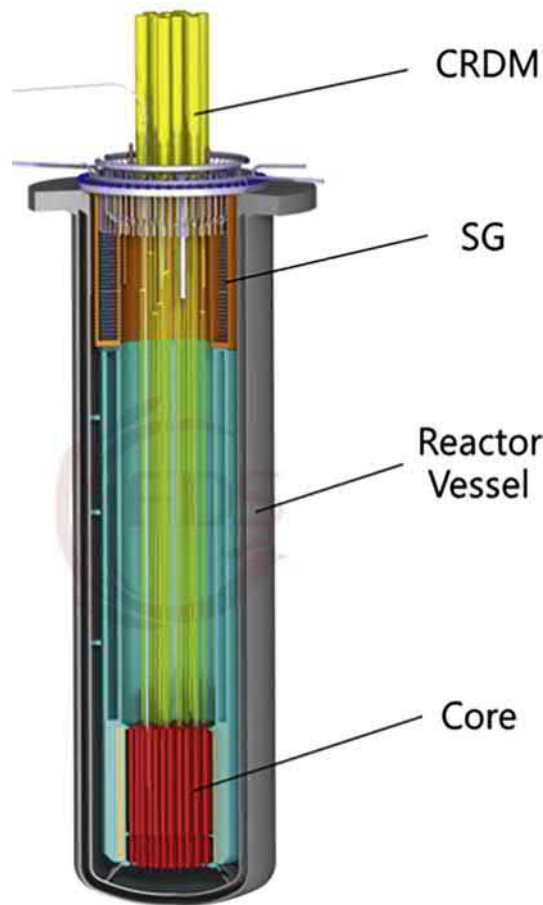


Fig. 6 Sketch of CLEAR-M. Courtesy of INEST-FDS.

Table 7 Selected characteristics of ENHS reference designs.

Design Parameter	ENHS1	ENHS2
Coolant operational circulation	Natural Circulation	Lift pump
Power, MWth/MWe	125/47	180/68
Power conversion Efficiency, %		38
Fuel ^a		12% Pu, 78% U _{depl} , Zr 10%
Annular core inner/outer radius, m		0.164/1.118
Effective core height, m		1.25
Overall module height, m	19.6	10.1
Outer module diameter, m	3.52	3.72
Coolant temperature, in/out, °C	400/504	400/471
Peak burnup, GWdays/tonHM		100
Peak fast neutron fluence, n cm ⁻²		3.8×10^{23}
Burnup reactivity swing, %dk/k		0.22

^aAs an alternative, the ENHS systems can be fueled with enriched uranium.

Adapted from (IAEA, 2007).

a similar level of compactness and simplification. Table 10 provides selected parameters of the LFR-TL-X, and Fig. 10 provides sketches of both LFR-AS-200 and LFR-TL-X.

MicroURANUS

Nuclear scientists and engineers at Seoul National University and elsewhere in the Republic of Korea have a long history of conceptualizing and developing HLM-cooled fast SMRs. Most recently, they have pursued a new land-transportable micro reactor concept,

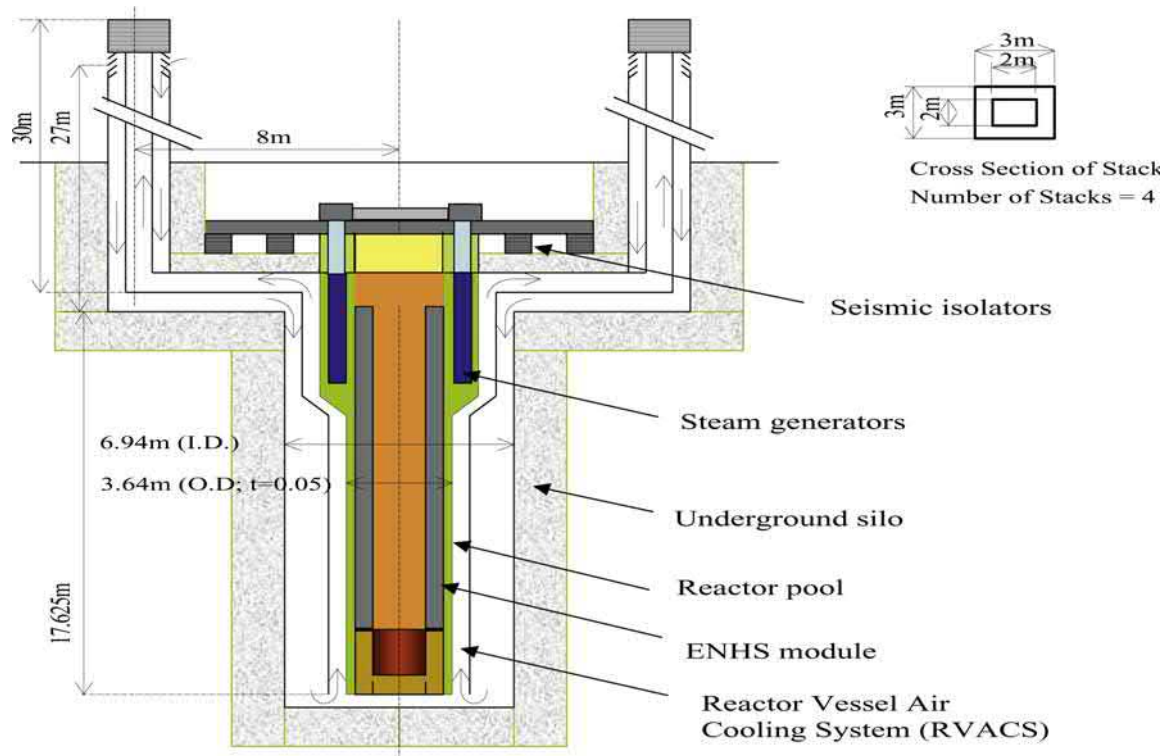


Fig. 7 Sketch of ENHS from IAEA (2007).

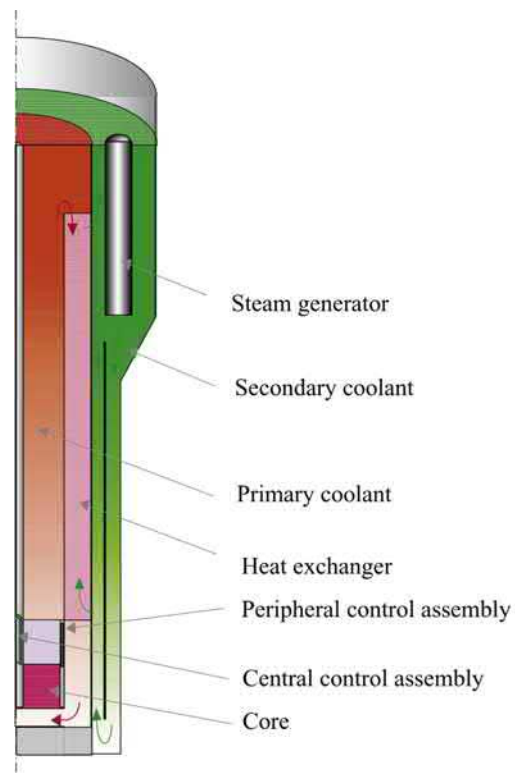
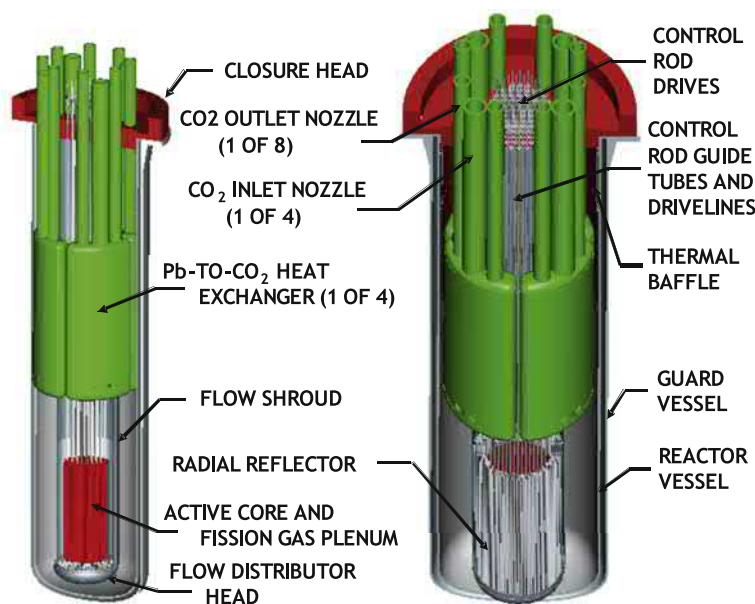


Fig. 8 Cross section Sketch of ENHS from IAEA (2007).

Table 8 Selected summary parameters of SSTAR.

<i>Parameters of SSTAR</i>	
Thermal Power, MWth	44
Electric Power, MWe	20
Primary Coolant	Lead
Fuel	Transuranic Nitride (enriched in N ¹⁵)
Net efficiency, %	42
Active Core Dimensions Height/Diameter, m	0.976/1.22
Core lifetime, y	30

Extracted from (Smith et al., 2008).

**Fig. 9** Sketch of SSTAR (Smith et al., 2008).

the MicroURANUS system which is a 20 MWe LBE-cooled system targeted for hydrogen production and maritime power. The term URANUS is an acronym derived from the following characteristics: Ubiquitous, Robust, Accident-forgiving, Non-proliferating and Ultra-lasting Sustainer.

The MicroURANUS development, led by researchers at the Ulsan National Institute of Science and Technology (UNIST), keys on the following features: The system would be long-lived and would be designed as a non-refueling modular LFR. After its full operational life, the MicroURANUS reactor would be removed with its fuel contained in the reactor vessel and transported by shipping cask to an international recycling center for secure processing. Recovered heavy actinide constituents from spent fuel processing as well as trace long-lived nuclides would be refabricated into fuels for future MicroURANUS units. Table 11 provides selected parameters of the MicroURANUS reactor, and Fig. 11 provides a sketch.

Special potential applications

Having summarized ongoing developments related to HLM-cooled fast SMRs, it is worthwhile to note some especially suitable applications beyond simple production of electricity and other energy-related products to meet local or regional requirements. Two such applications are (1) to meet the needs of isolated communities (e.g., islands or communities isolated by severe climate conditions such as the Arctic), and (2) commercial maritime applications.

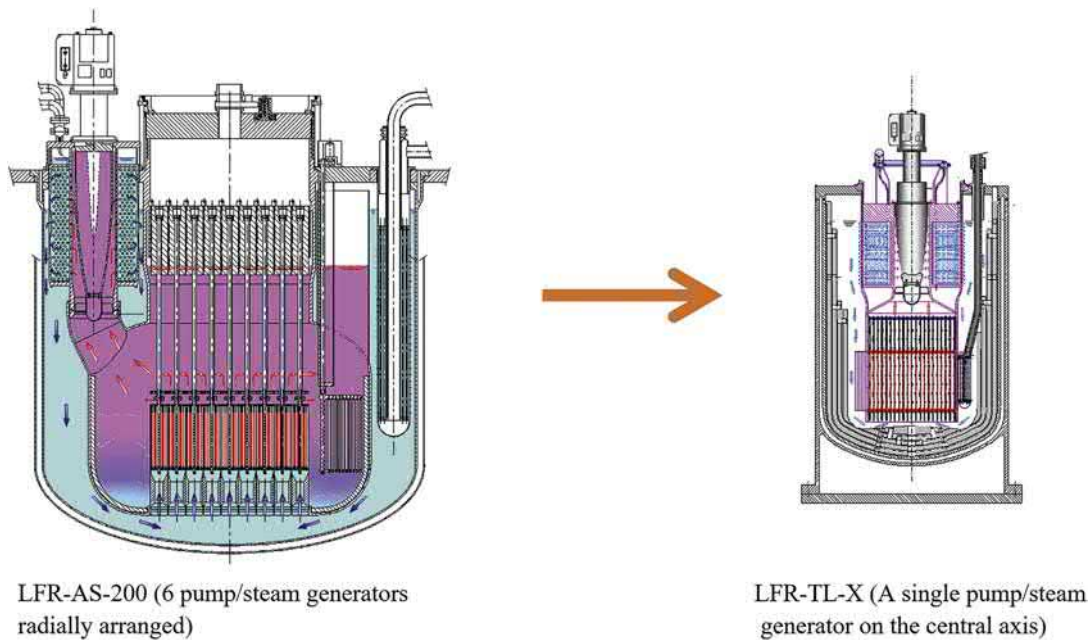
To meet the energy product needs of remote or isolated locations (including those in developing parts of the world for which power distribution infrastructures are poorly established), important considerations include transportability, compactness, passive safety, very long core life and autonomous operations. HLM-cooled fast SMRs envisioned to date excel in meeting these goals. Examples include SSTAR, SEALER and others – deliberately designed to achieve such objectives by capitalizing on the intrinsic characteristics of HLM coolants. Note that the Hydromine LFR-AS-200, for example, achieves a compactness milestone of >1 MWe per m³ of primary system volume.

Table 9 Selected summary parameters of the Hydromine LFR-AS-200.

<i>Parameters of LFR-AS-200</i>	
Thermal Power, MWth	480
Electric Power, MWe	200
Primary Coolant	Lead
Fuel	MOX
Lead Coolant temperature cycle, °C	420/530

Table 10 Selected summary parameters of the Hydromine LFR-TL-X.

<i>Parameters of LFR-TL-X</i>	
Thermal Power, MWth	15, 30, 60
Electric Power, MWe	5, 10, 20
Primary Coolant	Lead
Fuel	UO ₂ enrichment < 20%
Lead Coolant temperature cycle, °C	360/420

**Fig. 10** Sketches of LFR-AS-200 and LFR-TL-X – not to scale (Smith and Cinotti, 2016).**Table 11** Selected summary parameters of the MicroURANUS.

<i>Parameters of MicroURANUS</i>	
Thermal Power, MWth	60
Electric Power, MWe	20
Primary Coolant	LBE
Fuel	UO ₂ enrichment < 20%
LBE Coolant temperature cycle, °C	250/350

A second special application of note is that of commercial maritime propulsion, an application receiving increasing attention as it is recognized that current commercial shipping contributes significantly to the CO₂ release to the atmosphere. (Haas, 2014) Two HLM-cooled fast SMR concepts are actively targeting this area of application: MicroURANUS and the Hydromine LFR-TL-X. Table 12 summarizes the benefits from the maritime ship propulsion application.



Fig. 11 Sketch of MicroURANUS (IAEA, 2020).

Table 12 Maritime ship propulsion benefits.

<i>Desired feature</i>	<i>HLM-cooled fast SMR characteristics</i>	<i>Implications for ship propulsion</i>
Passive safety/ simplified design	- HLM-cooled small fast reactors feature a high level of intrinsic passive safety - relative chemical inertness of lead - high boiling temperature - ability to operate at atmospheric conditions - favorable thermo-hydraulic properties that support natural circulation heat removal	Simplified design and operation result in high reliability, low maintenance operations over very long time frames
Safety in the event of catastrophic occurrence	In the event of catastrophic occurrence, the core becomes encapsulated in the frozen lead coolant and prevents release of significant radioactivity	Eliminates the potential for environmental dispersion of contaminants in the event of otherwise catastrophic conditions
Compactness	HLM-cooled SMR designs are very compact	Reduced volume frees up cargo capacity. Elimination of the need for diesel fuel storage also increases cargo capacity
Very Long refueling interval/sealed system	These systems can operate for 15–30 years without refueling	Eliminates service outages/routing restrictions for refueling; eliminates safety issues with fuel transfer operations

Summary and conclusions

This chapter has reviewed HLM-cooled Fast SMRs. While HLM coolants have been considered since the earliest days of nuclear reactor technology, in recent years development activity has rapidly accelerated. The Russian systems SVBR-100 and BREST-OD-300 benefited from design and operational experience during the Cold-War application of HLM-cooled reactors to submarine propulsion, while other applications have reflected broad global HLM-cooled fast SMR interest (e.g., ALFRED and Sealer in the EU; SSTAR in the US; Hydromine LFRs in the US and Europe; CLEAR-M in China; and MicroURANUS in the Republic of Korea).

Different options for heavy liquid metal coolants having been explored, nearly all current activity is focused on either lead or LBE coolants. The engineering challenges associated with their significant differences in melting temperatures and production rates of polonium-210 are important in the choice between these coolants, and to date both options are represented in different ongoing reactor development activities.

The fundamental characteristics of these coolants are very favorable for fast reactor operations by virtue of their impacts on reactor neutronics as well as thermal hydraulics. The lack of rapid and energetic reactions with water and air eliminate the need for an intermediate cooling circuit to isolate activated primary coolant from steam, water or air, and this promotes system simplification, economic performance and inherent safety. Likewise, the very high boiling temperatures and low vapor pressures of both of these coolants enable operation near atmospheric pressure and virtually eliminate the possibility for coolant boiling or coolant vapor voids in the primary system.

Several current and recent HLM-cooled fast SMR concepts were summarized, including those that are designed as SMRs, and those that are demonstrator units intended to support scale up for future full-size units. In each case, the fundamental characteristics of the coolant have been taken into account to provide designs that are highly compact, passively safe, efficient in both electricity generation and fuel utilization. Some special applications, including units intended for remote and isolated locations and units appropriate for maritime propulsion, were also discussed.

See Also: Lead Cooled Fast Reactors.

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Micro-Reactors

Janne Wallenius, Division of Nuclear Engineering, KTH, Stockholm, Sweden

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Abbreviations

ABV	Atomic modular pressurised water reactor
CERMET	Ceramic-metallic
CLEAR	Chinese lead-based advanced reactor
CRDM	Control rod drive mechanism
dpa	Displacement per atom
GWd/t	Gigawatt-day-per-ton heavy metal
INEST	Institute of nuclear energy safety
MH	Military high (power)
ML	Military low (power)
MMR	Micro modular reactor
MW _{th(e)}	Megawatt [subscript th(e) for thermal(electric)]
OKBM	Опытное конструкторское бюро машиностроения
pcm	Percent mille [reactivity unit corresponding to $(k_{\text{eff}} - 1) \cdot 10^{-5}$]
PL	Portable low (power)
PM	Portable medium (power)
SEALER	Swedish advanced lead-cooled reactor
SG	Steam generator
SILUMIN	Silicon-aluminum
SL	Stationary low (power)
SM	Stationary medium (power)
SMR	Small modular reactors
TES	ТрансПортальная Лектростанция
TRISO	Tri-structural isotropic particle fuel
URENCO	Uranium enrichment company
USNC	Ultra-safe nuclear corporation

Historical background

In the early 1950s, the United States and USSR military forces developed micro-reactors for use as power supply on military bases disconnected from national grids. The typical electrical power requirement of such bases is a few MW. Four of the water-cooled US reactors listed in [Table 1](#) (PM-2A, PM-3A, SM-1A and MH-1A) were constructed, shipped to a remote destination and operated for a number of years before being returned to mainland US for decommissioning ([Suid, 1990](#); [DOE, 2001](#)). The USSR reactors were intended to be truck- or tank mounted, but were operationally tested on the site of the institute where they were developed ([Sinev et al., 1964](#); [Paliukhovich, 2002](#); [Baranaev et al., 1997](#)).

In particular, the PM-2A reactor deployed 220 km north-east of Thule base on Greenland deserves special attention ([Clark, 1965](#)). The reactor was transported to site on sleigh over the ice-cap, for installation in a 12 m wide trench buried under a surface of snow. A staff of 18 operated the plant. The cost of designing, building and installing the plant on Camp Century is stated by Clark as (1961) USD 5.7 million, equivalent to USD 49 million in 2019. Operating costs are quoted as 6.5 cents/kWh in 1961 currency, or 0.56 USD/kWh in today's value. Initial problems with the turbine generator lead to three extended shutdowns in 1961, whereas in 1962, the reactor operated with 74% availability, producing 5.4 GWh of electrical energy. In 1964, the reactor was disassembled, removed from Camp Century and returned to the United States.

Concepts under development

This article is addressing micro-reactor concepts and designs rated at maximum 10 MWe. It is complementary to [Cochran and Lovering \(2021\)](#), which is expanding the considered reactor power to the > 10 MWe and < 50 MWe range, thus addressing also reactors in the SMR category ([Table 2](#)).

In what follows, these concepts are described in terms of their particular design solutions, to the extent that this information is available in the open literature.

Toshiba's 4S-design

4S is short for "Super-Safe, Small and Simple." It is a sodium cooled reactor design with metallic fuel that was developed by Toshiba in order to reconcile demands for safety and economics ([Ueda et al., 2005](#)). It is intended for deployment in Arctic and other remote areas.

The final 10 MWe (30 MWth) version of 4S features a core with an active fuel height of 2.5 m and 18 hexagonal fuel assemblies with 169 fuel rods each ([IAEA, 2007](#)). The fuel rod has an outer diameter of 10.0 mm, and a cladding thickness of 0.50 mm. The U–Zr metallic alloy fuel has a ^{235}U enrichment of 17/19%, leading to a reactivity loss of 5500 pcm during the intended 30-year lifetime at an average linear power of 3.9 kW/m, during which the fuel reaches an average burn-up of 34 GWd/t. All reactivity coefficients of the core are negative, but due to fission product accumulation, the neutron spectrum softens, and the coolant temperature feedback becomes less negative and reaches a net-zero value at end-of-life. The reactivity loss of the core is compensated by a combination of a fixed absorber rod that is lifted out of the core at middle-of-life, and a cylindrical silicon-carbide reflector being lifted from beneath the core toward the top.

The primary sodium inlet/outlet temperatures are 355 °C and 510 °C respectively. Cooling during normal operation is achieved by forced convection using an EM-pump and an intermediate heat-exchanger located in a separate vessel. The height of the reactor vessel is 14 m, in order to permit passive removal of decay heat.

Toshiba proposed to demonstrate the 4S concept in Galena, an Alaskan community. Whereas the city council of Galena was positive to the project ([Alaska Journal of Commerce, 2008](#)), it did not proceed to licensing.

Table 1 Micro-reactors operated or tested by US and USSR military forces.

Name	Coolant	Power	Operator	Location	BoL	EoL
PM-1	H ₂ O	1 MWe	US army	Sundance, Wyoming	1962	1968
PM-2A	H ₂ O	1.6 MWe	US army	Camp Century, Greenland	1960	1963
PM-3A	H ₂ O	1.5 MWe	US army	McMurdo, Antarctica	1962	1972
SM-1	H ₂ O	2 MWe	US army	Fort Belvoir, Virginia	1957	1973
SM-1A	H ₂ O	2.0 MWe	US army	Fort Greely, Alaska	1962	1969
SL-1	H ₂ O	0.2 MWe	US army	Idaho	1958	1961
ML-1	N ₂	0.3 MWe	US army	Idaho	1962	1965
MH-1A	H ₂ O	10 MWe	US army	Panama canal	1968	1977
TES-3	H ₂ O	1.5 MWe	USSR army	IPPE, Obninsk	1961	1965
ARBUS	Organic	0.75 MWe	USSR army	RIAR, Dimitrovgrad	1961	1963
Pamir-630D	N ₂ O ₄	0.6 MWe	USSR army	JINP Sosny, Belarus	1985	1986

Table 2 Micro-reactors that have been developed after 1990, or that are currently under development.

Name	Developer	Coolant	Power	Fuel	Market
4S	Toshiba	Na	10 MWe	U-Zr	Arctic, remote, desalination
ABV-6 M	OKBM	H ₂ O	6–9 MWe	UO ₂ -silumin	Power, heat, desalination
Angstrom	Gidropress	PbBi eutectic	6 MWe	UO ₂	Arctic
CLEAR-M	INEST	Pb	10 MWe	UO ₂	Remote areas
e-Vinci	Westinghouse	Na	0.2–5 MWe	UO ₂ /U ₃ Si ₂ /U-Zr	Off-grid industrial and military
MMR	USNC	He	5 MWe	UO ₂ -TRISO	Arctic communities/mining
Aurora	OKLO	Na	1.5 MWe	U-Zr	Communities
SEALER	LeadCold	Pb	3–10 MWe	UO ₂	Arctic communities/mining
U-Battery	URENCO	He	4 MWe	UO ₂ -TRISO	Process steam
UNITHERM	NIKIET	H ₂ O	6 MWe	UO ₂ -Zr	Desalination, process steam

OKBM's ABV-6M design

The 6–9 MWe version of OKBM's ABV design is a factory produced PWR for use on floating nuclear plants, that would supply heat and desalination services, in addition to power (IAEA, 2012). An ²³⁵U enrichment of 19.7% gives the core an operational life of 10–12 years.

The total reactor module has a mass of 600 tons, a height of 13 m and a diameter of 8.5 m. The reactor is designed to be able to drive a floating nuclear power plant with a maximum length of 140 m, a beam of 21 m, a draft of 2.8 m and a displacement of 8700 t. Depending on the needs of the region to where it is shipped, the floating nuclear power plant can generate electric power, provide heat and power cogeneration, heat generation for sea water desalination, or can be used for other industrial applications.

The core lifetime is 10 years with 16,000 h of continuous operation, without reloading or shuffling of fuel.

The inlet/outlet temperature of the primary coolant is 247/330 °C, operating under a pressure of 15.7 MPa. Passive safety is implemented by natural convection of the primary coolant for removal of full power.

Gidropress' Angstrom design

Gidropress ("Hydropress") designs the Angstrom plant to achieve passive safety features in conjunction with a very short factory construction time of 12 months (Stepanov et al., 1998).

The 30 MWth (6 MWe) plant operates on lead-bismuth eutectic as primary coolant and may produce up to 14 MW of heat. Its core consists of about 2500 UO₂ fuel rods with an active fuel height of 0.7 m, an outer fuel rod diameter of 12.0 mm and a cladding thickness of 0.4 mm. The enrichment of the fuel is 26% and the peak fuel burn-up 8.6% (86 GWd/ton). At an average linear power of 17 kW/m, the effective full power life of the core is 8 years.

The primary coolant inlet/outlet temperatures are 290 °C and 465 °C, respectively, and the steam generator operates at a pressure of 3.5 MPa. The transportable steam supply system module of the plant, enclosing the reactor unit, steam generators, pump tanks, condensers and radiators, has a dimension of 5.1 × 3.2 × 24.4 m.

OKLO's Aurora design

Aurora is a 1.5 MWe (4 MWth) potassium heat-pipe reactor designed by OKLO. It utilizes uranium-zirconium metal alloy fuel and the reactor is to be located underground. A license application has been submitted to US NRC (OKLO, 2020). The intended market is "communities."

Heat exchange from the primary system is provided by super-critical CO₂.

Aurora has negative reactivity feedbacks due to thermal expansion of the fuel and structural materials, as well as doppler broadening.

Three control drums are used to control reactivity. Three shut-down rods are suspended above the core by electromagnets such that the rods insert via gravity drop when the electromagnets are de-energized.

Decay heat is passively transported from the fuel through the zirconium based reflector to thermal heat sinks within the reactor module.

INEST's CLEAR-M design

CLEAR-M10d is a 10 MWe lead-cooled reactor design by INEST (Wu et al., 2019). It is fueled with 18.0–19.75% enriched UO₂, having a core-life of 10–20 years, corresponding to a burn-up of 62 GWd/t. The core consists of 3500 fuel rods with an outer diameter of 16 mm, a cladding thickness of 1 mm, and a rod pitch to diameter ratio of 1.2. The active fuel height is 1.2 m.

The primary coolant inlet/outlet temperature is 375/495 °C. The average velocity of the coolant is 0.46 m/s, yielding a maximum cladding temperature of 541 °C.

A single spiral tube steam generator is used for heat transfer, having an outer diameter of 1.75 m and a bundle active height of 1.46 m.

Decay heat removal is achieved by natural convection in a primary vessel having a height of 8.5 m and a diameter of 2.5 m.

Westinghouse's eVinci design

The eVinci reactor is designed by Westinghouse to achieve passive heat extraction using liquid sodium heat pipes (Arafat and Van Wyk, 2019). The electrical power of the reactor is between 0.2 and 5 MW. It is designed to be manufactured and fueled in a factory for transport to site, where installation would be made within 30 days. The core is designed as a steel monolith with channels for heat pipes and fuel pellets, requiring refueling after 10 years. The secondary coolant temperature reaches 600 °C, enabling process heat applications. The uranium fuel may be either an oxide, silicide, or metal alloy.

The intended markets of e-Vinci are Arctic regions and military bases (Fig. 1).

USNC's MMR design

The US based Ultra-safe Nuclear Corporation is developing a 5 MWe helium cooled reactor with coated particle TRISO fuel in a prismatic block configuration. A core lifetime of 20 years without refueling is foreseen. The intermediate heat transfer medium is a molten salt. In 2019, USNC's Canadian partner Global First Power filed an application for license to build a demonstration unit on the Chalk River site of Canadian Nuclear Laboratories in Canada (Global First Power, 2019). The intended markets for the MMR are remote communities and mining operations in Arctic regions (Fig. 2).

LeadCold's SEALER design

LeadCold has designed a lead-cooled reactor for power production in Arctic communities and mining operations (Wallenius et al., 2018). The core consists of 19 fuel assemblies with 19.75% enriched UO_2 fuel, surrounded by 12 control rod assemblies and six shut-down rod assemblies. Each fuel assembly hosts 91 fuel rods with an outer diameter of 14.4 mm, a clad thickness of 0.5 mm and a pellet diameter of 13.3 mm. The fuel column height is 1.1 m, and the total rod length 1.59 m. The 15–15Ti cladding is protected from corrosion by a surface alloy consisting of Fe–10Cr–6Al–RE, where RE stands for an optimized composition of Ti, Zr, Nb and Y.

When deployed in an Arctic community, it would operate at 8 MW of thermal power (maximum 3 MWe), corresponding to an average linear power of 4.2 kW/m. The average burn-up is limited by reactivity losses and available control-rod compensation worth of 4500 pcm to 33 GWd/ton, which in this operational mode is achieved after 27 full power years. In this case, the coolant inlet/outlet temperature is limited to 390/432 °C, permitting the use of conventional SS316 steel as structural material in the reactor. The end-of-life peak damage dose to the fuel cladding is 66 dpa.

If deployed at a mining site, the reactor would operate at 25 MW of thermal power (maximum 10 MWe), corresponding to a linear power of 13 kW/m, and a coolant outlet temperature of 520 °C. The core-life in this case is nine full power years, which corresponds to the typical duration of a remote mining business (IAEA, 2007). In this case, all pressure boundary materials exposed to lead have to be protected by an alumina forming weld overlay, while structures under low stress may be manufactured by bulk Fe–10Cr–4Al–RE (Fig. 3).



Fig. 1 Westinghouse eVinci plant. Reprinted with permission of Westinghouse Electric Company.



Fig. 2 USNC's MMR plant configuration. Reprinted with permission from USNC.

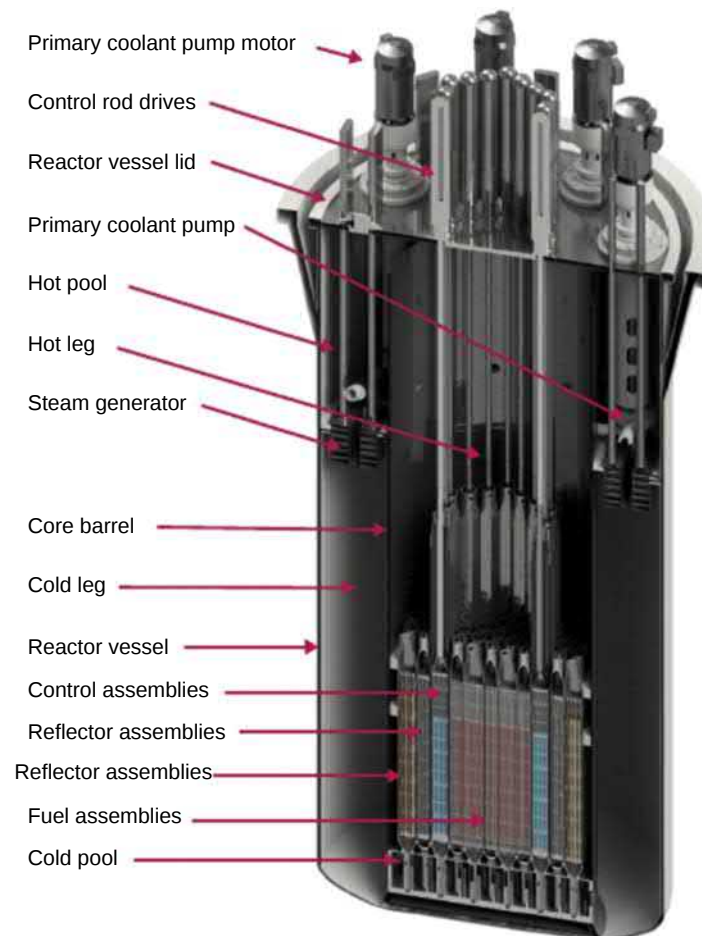


Fig. 3 Primary system configuration of SEALER (Wallenius et al., 2018). Reprinted with permission from LeadCold.

U-battery

U-Battery is a consortium including URENCO, Wood, Laing O'Rourke and Kinetrics. Its 9 MWe (21 MWth) helium cooled reactor was designed by Manchester University and the University of Delft to operate on coated particle TRISO fuel embedded in prismatic blocs (Ding et al., 2011). The core consists of 120 blocks stacked in four horizontal levels, including seven internal reflector blocks in each level. The inventory of 19.75% enriched UO_2 fuel is 1.8 tons and the packing fraction of microspheres in the fuel compacts is 30%. The core is refueled each 5 years, during which the fuel reaches 90 GWd/tburn-up.

The coolant inlet/outlet temperature is 420/750 °C, and power is produced by a nitrogen turbine.

The intended market for U-Battery is process heat for industrial applications where the first demo-unit would be powering the industrial site of URENCO in Capenhurst (Fig. 4).

NIKIET's UNITHERM design

The water-cooled UNITHERM design of NIKIET is intended for production of a mixture of electricity, district heating, seawater desalination, and process steam production, or to meet other specific demands (IAEA, 2007). The reactor would be produced in a factory shop.

The 30 MWth (2.5–6.0 MWe) power unit uses a CerMet (Ceramic-Metallic composite) UO_2 –Zr fuel with 19.75% enrichment of ^{235}U , in a zircalloy cladding tube. The cladding tube features four helical ribs, used as rod spacers. The inner/outer cladding diameter is 4.8/5.8 mm. The fuel rods are arranged in a hexagonal pattern having a pitch of 7.05 mm, in cylindrical fuel assemblies with an inner/outer shroud diameter of 58.4/60.0 mm.

An average/peak burn-up of the fuel equal to 110/180 GWd/t is foreseen. The core features 265 fuel assemblies, a diameter of 1.13 m and an active fuel height of 1.1 m. A particular feature of this design is a very low fuel temperature, estimated at 365 °C. The life-time of the fuel is 16.5 full power years. Reactivity loss during operation of the core is not provided by NIKIET, yet is supposed to be compensated by 9 control assemblies with 53 or 54 absorber rods each.

The primary coolant inlet/outlet temperatures are 250/330 °C, respectively, operating under a pressure of 16.5 MPa. The dimension of the primary vessel is 3.22×5.05 m for the case of producing 20 MWth of steam and 2.5 MW of electricity, and 3.33×6.15 m for a system producing 6 MW of electricity.

R&D required for commercial applications

Water-cooled reactors

The water-cooled ABV-6M reactor applies conventional oxide fuel, hence the design is of mature character, and requires little R&D to be qualified, with the possible exception of the particular solution for passive decay heat removal, which is natural convection of the primary coolant.



Fig. 4 U-Battery plant configuration. Reprinted with permission of U-Battery limited.

NIKIET's UNITHERM's concept relies on the use of $\text{UO}_2\text{-Zr}$ composite CERMET fuel, which has not been used in commercial power plants. The fuel manufacturing route has to be qualified by means of irradiation in test reactors of fuel produced by an industrial fuel supplier with the actual process foreseen by NIKIET.

Helium-cooled reactors

Whereas high temperature micro-reactor designs of U-Battery and USNC rely on proven technology in terms of the primary cooling circuit and the coated-particle fuel, these concepts have to qualify their respective industrial (i.e. non-prototype) fuel manufacturing route by irradiation in materials test reactors or a demonstration unit prior to commercial operation.

A remaining phenomenon of possible concern for radiological exposure to maintenance staff is the observed migration of silver fission products through intact TRISO encapsulation of fuel kernels (Geng, 2014; van Rooyen et al., 2014).

Sodium-cooled reactors

Licensing of Toshiba's 4S design will require the demonstration of its reactivity control system, which consists of an annular absorber ring surrounding the core. In particular, passive response to control element ejection must be evaluated. As for other concepts, the U-Zr fuel manufacturing route needs to be qualified by irradiation in a materials test or a demonstration reactor.

The eVinci concept of Westinghouse is considerably more innovative, e.g. in terms of its horizontal configuration. In particular the functionality and materials performance of sodium heat pipes during extended operation mandates investigation. It is to be noted that the lab-scale tests of the technology conducted to date were made for potassium heat pipes, an element having a substantially lower evaporation temperature. Use of a silicide or metal alloy fuel will require qualification of the respective manufacturing route to be used in an actual industrial fuel manufacturing facility.

Lead- and lead-bismuth-cooled reactors

The long-term application of LeadCold's SEALER-unit requires the qualification of alumina forming steels for corrosion protection of fuel cladding tubes, steam generator tubes and hex-cans. Lab observations indicating excellent corrosion tolerance at $T = 550\text{--}850\text{ }^\circ\text{C}$ need to be confirmed under prototypical flowing lead and varying oxygen conditions (Ejenstam et al., 2013; Dörmstedt et al., 2020).

In the case of the ANGSTREM design, a similar qualification program is required for the Fe-12Cr-2Si steel developed in Russia for heavy liquid metal cooled reactors. A particular difficulty with the latter appears to be low temperature irradiation embrittlement.

The structural material to be used for corrosion mitigation in CLEAR-M10d appears not to be defined. Once the approach is defined, it has to be verified through a program including exposure under flowing lead as well as under irradiation.

All lead-cooled micro-reactors here listed intend to use conventional UO_2 fuel, for which no further qualification appears to be necessary.

See Also: Microreactors for Decentralized Power Sources.

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Light Water Reactors (LWR) Fuel Cycle Options

Eugene Shwageraus, Department of Engineering, University of Cambridge, Cambridge, United Kingdom

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Glossary

CR Conversion ratio—Fissile material production rate relative to its loss rate in a system.

DC Doppler coefficient of reactivity—Change in core reactivity per unit change in fuel temperature due to Doppler broadening of resonances in fuel nuclides.

HFP Hot full power—Operating condition at which all reactor component temperatures are at their nominal values.

HZP Hot zero power—Operating condition at which reactor coolant and fuel temperatures are lower than their nominal values. Only the coolant temperature at the core inlet is nominal. However, since the core does not generate power, the coolant is not heated up as it flows along the fuel rods. Furthermore, there is also no temperature difference between the fuel pellet surface and its center.

MA Minor actinides; isotopes of neptunium, americium, curium and heavier elements.

MTC Moderator temperature coefficient—Change in core reactivity per unit change in moderator temperature. The reactivity effect is related to the balance between neutron absorption and neutron moderation effects, both of which change to a different extent due to thermal expansion of the moderator (light water coolant in case of LWR).

MWd/kg A measure of fuel burnup: the amount of energy in MW-days produced per kg of initial uranium (or any other Heavy Metal—HM) content in the fuel.

Neutron spectrum Relative energy distribution of neutron flux. It is measured in number of neutrons/cm²/s/eV. The spectrum is said to be hardening, if the relative distribution of neutrons shifts towards higher energies, while opposite is the in case for spectrum softening.

Radiotoxicity Effective radiation dose to human body in Sieverts (Sv) per given quantity of radioactive nuclide either ingested or inhaled. It accounts for the specific activity of the nuclide (i.e., its half-life), type of radiation it emits and relative damaging effects to biological tissues affected by the radiation.

Reactivity Fraction of neutrons in a reactor in excess of what is required for perfect balance between the neutron production and loss rates. Reactivity is equal to zero when such balance is achieved. In that case, the system is said to be critical and its neutron population remains stable.

Reactivity excess Positive reactivity of fresh nuclear fuel which needs to be compensated by various reactivity control means such as control rods, soluble boron and burnable poisons in order to keep the system critical during normal operation at full power.

Reactivity worth Change in reactivity as a result of introduction of unit of absorbing material into the system—e.g., insertion of a control rod or change in soluble boron concentration.

SDM Shutdown margin—Difference between core reactivity at Hot Full Power and Hot Zero Power (see above the definitions of HFP and HZP). Since both DC and MTC are typically negative, cooling down of the core will result in an increase in reactivity which must be compensated by insertion of control rods or other means to assure sub-criticality in case of an emergency shutdown. Several conservative assumptions are typically applied when calculating SDM, such as assuming that the most reactive control rod is stuck out (failed to insert) as well as various engineering and modeling uncertainties.

TRU Transuranic nuclides—Plutonium and minor actinides.

Abbreviations

BWR	Boiling Water Reactor
EOC	End of Operating Cycle
FFF	Fertile-Free Fuel
FP	Fission Products
HLW	High Level Waste
LLFP	Long-lived Fission Products
LLW	Low Level Waste
LWR	Light Water Reactor
MOX	Mixed Oxide (PuO ₂ -UO ₂) fuel
PWR	Pressurized Water Reactor
RIA	Reactivity Insertion Accident
US DOE	United States Department of Energy

Commercial LWR used fuel characteristics

The vast majority of currently operating commercial nuclear power reactors are of the Light Water-cooled Reactor (LWR) type. Most LWRs use an open fuel cycle. This mode of operation implies that the fuel used in the reactors is manufactured from freshly mined uranium, which is subsequently enriched and fabricated into fuel rod assemblies. After irradiation, used fuel is cooled for a number of years in water pools on reactor sites and, when its activity is reduced, is typically moved to interim air-cooled concrete storage casks awaiting ultimate disposal in a geological repository when it becomes available (Glatz, 2021). Several countries (e.g., Finland, Sweden and France) have been successful in identifying suitable geological formations for the final disposal of radioactive waste and are now at different stages of design, licensing and construction of these facilities (Kadak, 2021). In other countries, used nuclear fuel repository siting and approval has faced significant public opposition with no clear pathway for resolution (Metlay, 2021). The use of deep boreholes is a potentially attractive alternative to a centralized repository. This emerging technology benefits from a smaller footprint and a wider availability of suitable sites. Their modular nature may also result in a lower marginal cost of capacity expansion.

The need for a geological repository arises from the fact that used nuclear fuel contains radioactive nuclides which need to be isolated from the environment for long periods of time to prevent harm to humans and the biosphere in general. The half-life for radioactive decay of some of these nuclides, however, is substantially longer than the time over which the integrity of engineered isolation barriers can be assured. Therefore, the safety case for an ultimate repository must rely on local geological characteristics to ensure that radioactive nuclides do not enter the biosphere. Studies of natural analogues, most notably a natural nuclear reactor at Oklo, Gabon provide a robust scientific basis for the licensing of radioactive waste repositories (Gauthier-Lafaye, 2002).

Before presenting the reasons for recycling used nuclear fuel, it is worthwhile examining the constituents of the fuel following irradiation and discharge from a typical LWR (Glatz, 2021). The cumulative thermal energy produced by each kilogram of a typical LWR used fuel is on the order of 50 Mega-Watt-days (MW-days or 1.2 million kWh). This amount of energy corresponds to about 5% of the initial uranium atoms having been fissioned and converted into fission products (FP). It means that the FPs constitute about 5% of the used fuel mass, while the originally loaded uranium still accounts for the majority of the rest of the fuel, or about 94% (Fig. 1). The remaining uranium is only mildly radioactive and still contains about the same amount of fissile ²³⁵U as natural uranium. Recycled uranium can be re-enriched and re-used in LWRs, although a somewhat higher enrichment would be required because of the presence of ²³⁶U, which is a moderately strong neutron absorber.

The fission products include many isotopes, the majority of which are highly radioactive. Heavy nuclides such as uranium require many neutrons to provide stability of their nuclei. When split in fission into two (or more) fission products, the resulting lighter nuclei do not require as many neutrons for stability. Such excess of neutrons is energetically unfavorable and leads to spontaneous radioactive decay. High radioactivity implies short half-life. As a result, the vast majority of FPs decay to negligible activity levels in a few hundred years. Most of the FPs also have a relatively low mobility in the environment, so that they will stay within or close to the waste package even after it is degraded.

The high radioactivity of FPs also means that used fuel needs to be cooled for long periods of time after discharge from a reactor core to avoid reaching temperatures that would compromise its integrity or the integrity of the waste package. Therefore, the management of decay heat produced by spent used fuel remains an important consideration for the design of all facilities involved in the spent used fuel handling. Most of the decay heat in the time period relevant to interim storage and disposal of used fuel is produced by only two FP isotopes: ¹³⁷Cs and ⁹⁰Sr, both with half-lives of approximately 30 years. Both isotopes are produced in fission with a high probability and are therefore present in used fuel in high concentrations. From about 10 years following fuel discharge, Cs, Sr and their short-lived decay chain daughters dominate the heat generation, accounting for over 90% of the heat generated by FP decay for several decades.

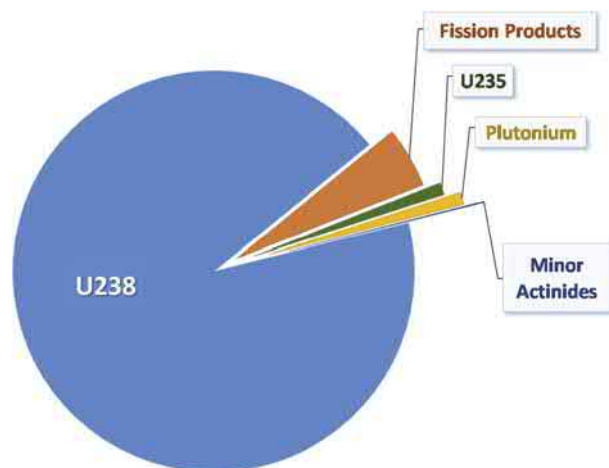


Fig. 1 Composition of a typical LWR used nuclear fuel.

Modern LWRs produce, through successive neutron captures starting in ^{238}U and radioactive decay (Shusterman, 2021), about 250 kg of Pu per Giga-Watt-years (GW-year) of electricity they produce. Plutonium comprises approximately 1% of used LWR fuel mass. A significant fraction of this is made up of the fissile isotopes ^{239}Pu and ^{241}Pu , with properties similar to ^{235}U . Plutonium isotopes are more radioactive than uranium and, if separated from the rest of the used fuel constituents, produce non-negligible amount of decay heat and require shielding for handling. The radioactive decay of some of plutonium isotopes, however, is not sufficiently fast to avoid the requirement for geological isolation.

The remainder of the used fuel mass is composed of actinides other than uranium and plutonium. Their cumulative fraction is typically small—on the order of 0.1%. Therefore, these are collectively known as Minor Actinides (MA), while, together with plutonium, these are referred to as Trans-Uranium elements (or TRU) because they are located beyond uranium in the Periodic Table. MAs are produced as a result of a series of neutron captures and subsequent decays, starting from uranium and plutonium isotopes. Despite their small quantity, some MA have long half-lives and high mobility in the environment. Therefore, they, like plutonium, require geological isolation. Moreover, many MA isotopes generate not just decay heat but also emit high-energy photons and/or neutrons, further complicating handling, transportation and storage of used fuel.

Motivations for Used Fuel Recycling

Several reasons have been identified to justify recycling some of the constituents of irradiated nuclear fuel from LWRs. The recycling of some subsets of these has been deemed sufficiently valuable to spur industrial scale technology development and commercial deployment. The recycling of other constituents has remained a mostly academic exercise with small-scale lab based experimental programs. The practice of recycling and re-use of any of the used fuel constituents is often referred to as a closed fuel cycle.

Historically, the primary reason for recycling used nuclear fuel from commercial power reactors has been the desire to recover the energy value remaining in the fuel after discharge. As briefly described in Chapter 00308, uranium from used fuel can be either re-enriched or mixed with plutonium, also recovered from used fuel, then refabricated into new fuel which would continue generating energy. The recycling of uranium and plutonium from used fuel can also be viewed as an assurance of energy security, in case the supply of energy resources (either nuclear or otherwise) is interrupted due to hard-to-predict events such as natural disasters or military conflicts.

Another reason for recycling is to enable an alternative (to the open fuel cycle) approach to radioactive waste management, perhaps increasing the capacity or simplifying the design of the geological repository. Separating out environmentally mobile nuclides results in less reliance needing to be placed on engineered barriers. Separating out the long-lived nuclides means that any barriers, either engineered or geological, need to perform their role for a shorter period. The long-lived nuclides can be re-introduced into reactor cores, where they can be fissioned directly or transmuted into other nuclides with a shorter half-life or with better fissile properties which can also eventually fission and, thus, be converted into short-lived fission products.

An extensive study conducted by the US DOE concluded that a used fuel repository capacity can be increased by up to a factor of 43 if 99.9% of Cs and Sr, Am and Pu are separated and not sent to geological disposal, as illustrated in Fig. 2 (Wigeland et al., 2006). Separated Cs and Sr would be stored in a supervised interim storage, while the Am and Pu would be recycled and transmuted within a closed fuel cycle.

In addition to realizing the energy potential of recycled uranium and plutonium or transmuting long-lived MA nuclides, separation of short-lived FPs from uranium, Pu and MA also provides an opportunity to recondition the components that are not recycled and destined for the repository into more environmentally-robust waste forms such as glasses or ceramics.

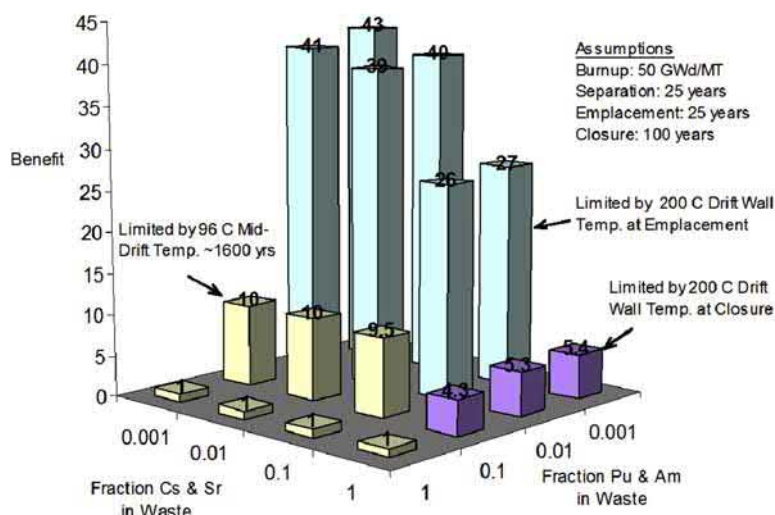


Fig. 2 Geological repository compaction factors for different efficiencies of MA and FP separation. From Wigeland RA, Bauer TH, Fanning TH, and Morris EE (2006) Separations and transmutation criteria to improve utilization of a geologic repository. *Nuclear Technology*, 154(1): 95–106.

Finally, the recycling of fissile actinides may also prevent diversion of these nuclides to weapons proliferation. Eliminating the potential for the diversion of fissile material from the used fuel repository, however, does not eliminate the risk of proliferation altogether; other opportunities of diversion from various stages of the closed fuel cycle will arise. Therefore, weighing the risks of proliferation between the open and closed fuel cycles is a subject of vibrant research and ongoing public debate.

The various options for used fuel recycling can be distinguished by which components of the fuel are chosen to be recycled: Pu, MA, and sometimes Long-lived Fission Products (LLFP) e.g., ^{129}I and ^{99}Tc , or any combination of the above. In addition, recycling these components could be done only once, multiple times, or indefinitely. The latter option implies that these nuclides are kept within the fuel cycle and never leak into the environment apart from losses arising as a result of imperfect reprocessing methods. Finally, different materials have been considered as host matrices for recycling, such as UO_2 , ThO_2 or neutronically inert matrix fuels (IMF), also known as Fertile-Free Fuels (FFF). It is legitimate to investigate at least the theoretical possibility of all these options. The environmental benefits of recycling increase with more components being recycled and with the number of recycling passes, but so do the technological complexity and costs of such fuel cycles. The remainder of this chapter will discuss some of the benefits and practical limitations of these options, focusing mostly on the implications for reactor operation and design.

Single-Pass Recycling of Pu and MA in LWRs

Recycling Pu in commercial LWRs has been carried out successfully for several decades in many countries. The motivations for this practice have been mainly the realization of additional energy value stored in used nuclear fuel and conditioning of FPs and MAs in carefully engineered glass (through a process called vitrification) for long-term storage or disposal.

In this scenario, used fuel is reprocessed to separate plutonium and uranium from the rest of the used fuel constituents using the PUREX process or its derivatives (Genetz, 2021). It involves mechanical shredding of the fuel with subsequent dissolution in nitric acid, from which U and Pu are selectively removed using the organic molecule tributyl phosphate (or TBP). This process has been demonstrated on an industrial scale with a total global capacity of several thousand tons of reprocessed fuel per year. The largest existing LWR reprocessing operation is in La Hague, France. Oxide fuel reprocessing in the United Kingdom at Sellafield THORP plant was terminated in 2018 due to a lack of commercial demand, while a large-scale reprocessing plant at Rokkasho, Japan, has been recently completed after a lengthy construction and licensing period but has not yet entered its full commercial operation at the time of writing. In the US, following a brief politically motivated ban on fuel reprocessing in the late 70s due to concerns over nuclear weapons proliferation, commercial fuel reprocessing has never been restarted, although scientific research is ongoing with state-of-the-art capabilities in many universities and national laboratories.

The preferred fuel of choice for recycling Pu from used nuclear fuel in LWRs is Mixed Oxide (MOX). In this fuel form, PuO_2 is mixed with UO_2 which can either be fresh natural uranium, depleted uranium from enrichment process tailings or recycled uranium from used fuel.

The evolution of Pu isotopes with burnup in a typical LWR is presented in Fig. 6 in Shwageraus (2021). At a typical burnup of about 50 MWd/kg, used LWR fuel would contain a mixture of Pu isotopes with only about 65% of them being fissile in a thermal spectrum. All Pu isotopes are strong absorbers of thermal neutrons. This is the main cause of MOX fuel's implications for LWR core design and operation. Each Pu isotope has its own peculiar features that are relevant to the practicality of using MOX in LWRs.

^{238}Pu appears in relatively small quantities of around 2–3%. It is not fissile in a thermal spectrum but can be converted to fissile ^{239}Pu through neutron capture. It has a relatively short α -decay half-life of about 87 years which, in addition to a high spontaneous fission probability, can be viewed as providing an important deterrent against nuclear proliferation (see also [Shwageraus, 2021](#) for additional discussion). However, high levels of decay heat also complicate MOX fuel manufacturing and handling, which can be overcome in principle with technological solutions such as remote operations and active cooling, but these will inevitably add costs to the fuel cycle.

^{239}Pu is the main fissile isotope with the highest concentration of around 50% in typical used LWR fuel. It has a much higher thermal fission cross-section than ^{235}U , which makes it a dominant consumer of thermal neutrons in the fuel, leading to a substantially harder neutron spectrum than in uranium fuel. It also has much higher resonance integral than ^{235}U with a strong low-energy resonance at about 0.3 eV, which makes MOX fuel more sensitive to spectral shifts due to changes in temperatures and densities of core materials. The probability of neutron capture without fission in ^{239}Pu is also substantially higher than in ^{235}U , leading to significant generation of ^{240}Pu .

^{240}Pu is not fissile. It comprises about a quarter of LWR-grade Pu. Its importance lies in the fact that it has a giant low-energy resonance (about 100,000 barn at its peak). As for ^{239}Pu , this makes MOX fuel particularly sensitive to spectral changes. Because of its large absorption cross section, it is an excellent source of fissile ^{241}Pu .

^{241}Pu has similar fissile properties to those of ^{239}Pu . At discharge, it constitutes a notable fraction (about 15%) of total Pu. However, it has relatively short half-life of 14 years, decaying to ^{241}Am . This means that the Pu fissile content is sensitive to both cooling time before reprocessing and the subsequent storage time of already separated Pu. The latter matters also because the presence of ^{241}Am further complicates MOX fuel manufacturing and handling due to decay heat and γ -radiation dose resulting from ^{241}Am decay. Additionally, ^{241}Am is a strong neutron absorber contributing further to Pu fuel reactivity loss in addition to that arising from the loss of fissile ^{241}Pu .

^{242}Pu is very similar to ^{238}Pu in terms of its relevant neutron reactions. It is, however, much longer-lived (nearly 400,000 years). Unlike ^{238}Pu , it does not directly produce a fissile isotope and, therefore, it is essentially just a parasitic neutron absorber. Furthermore, the decay of ^{241}Pu and neutron captures in ^{242}Pu are the two main routes to production of minor actinides such as americium and curium isotopes, recycling of which in LWRs adds an additional layer of complexity, as will be discussed later.

Implications of using Pu MOX for LWR operation and core design

As mentioned earlier, the strong thermal absorption of Pu isotopes leads to a hardening of the neutron spectrum when MOX fuel is introduced to an LWR core. As a result of that, the relative competitiveness of other core materials for neutron absorption diminishes. For example, the reactivity worth of soluble boron and control rods is significantly reduced. Therefore, LWRs loaded with MOX fuel will have a reduced control rod shutdown margin.

The operational flexibility of existing reactors is often limited, and it is not always possible to increase the soluble boron concentration to make up for this change because of the constraints imposed by needing to avoid a positive Moderator Temperature Coefficient (MTC) and coolant water chemistry. The loss of soluble boron reactivity worth is partially compensated by the fact that MOX fuel has flatter reactivity as a function of burnup than UO_2 fuel. Therefore, there may be less initial excess reactivity to compensate for.

These considerations currently restrict the fraction of MOX fuel assemblies in existing LWRs to about one third, while the remainder are conventional slightly enriched UO_2 fuel assemblies. Modern, Gen-III+ LWR designs have provisions such as a larger number of control rods to accommodate a full MOX fuel core ([Arslan et al., 2013](#)).

MOX fuel assembly designs are slightly different from conventional UO_2 LWR fuel assemblies. Non-uniformity in the amount of water around fuel pins across a fuel assembly (next to inter-assembly gaps or control rod guide tubes) leads to slight variations in neutron spectrum. This has a negligible effect on UO_2 fuel. Pu bearing fuel, however, is very sensitive to spectrum variation due to low-energy resonances and high thermal neutron absorption cross section. This can result in a greater variation of power across the fuel assembly. Therefore, fuel pins in MOX assemblies typically have varied concentration of Pu in order to compensate for this. Three different Pu loadings are common for PWR MOX, while in BWRs, up to six different Pu loadings are used because of the generally higher heterogeneity of BWR fuel assemblies.

In partially MOX fuel loaded LWR cores, MOX fuel assemblies tend to operate at higher power towards the end of life because of the flatter reactivity decline with burnup than that of UO_2 assemblies. This is undesirable because high power in high-burnup assemblies may lead to an excessive release of fission gas, limiting the achievable fuel burnup.

Both the Doppler effect and MTC in MOX fuel assemblies are more negative than in UO_2 fuel, again, due to the low-energy resonances in fertile Pu isotopes. This effect leads to a further deterioration of shutdown margin because the reactivity increase due to fuel and moderator cooling from Hot Full Power (HFP) to Hot Zero Power (HZP) conditions will be larger. Therefore, even larger number of control rods will be required, further restricting the fraction of MOX assemblies in conventional LWR cores.

An additional complication related to resonance structure of Pu isotopes is a potential reactivity insertion due to significant boiling or complete loss of coolant. Although the MTC and void reactivity coefficient at low void fractions are negative, the void coefficient at high void fraction can be positive for MOX fuel with high Pu content. Low-energy neutron capture resonances of ^{240}Pu and ^{242}Pu contribute negative reactivity to void coefficient at low void fractions (more neutrons are shifted towards these resonance energies). This is demonstrated in [Fig. 3](#), which shows the neutron spectrum in a typical LWR MOX fuel at different void fractions superimposed on the shape of neutron capture cross sections of ^{240}Pu and ^{242}Pu , as well as fission cross section

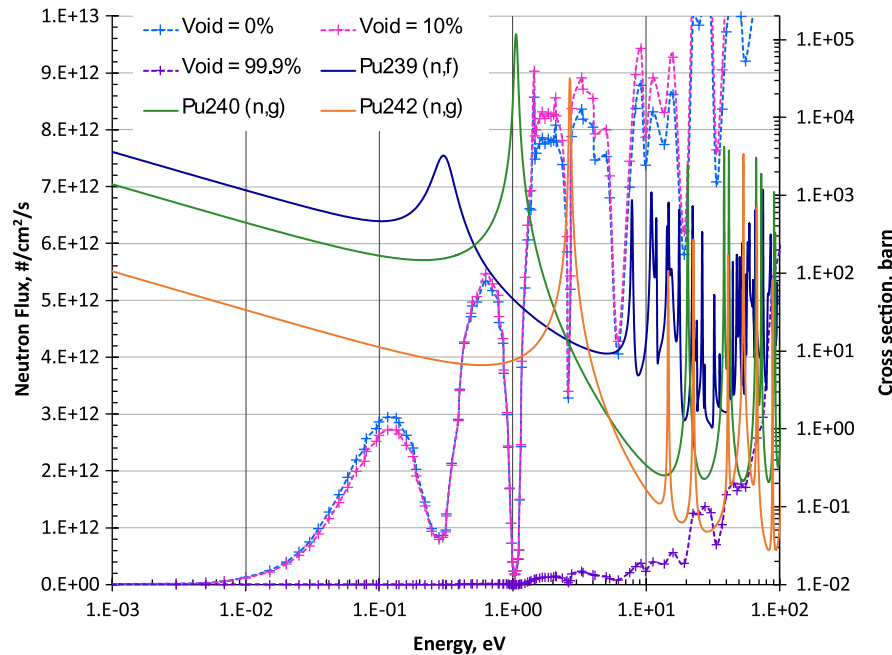


Fig. 3 Neutron flux spectrum at different void fractions (left axis) and cross sections of important Pu isotopes in a typical LWR MOX fuel.

of ^{239}Pu . At low void fractions, the magnitude of thermal flux spectrum is slightly reduced, while epithermal flux is increased. This leads to a loss of absorption in ^{239}Pu fission resonance, while increasing absorption in ^{240}Pu and ^{242}Pu capture resonances (**Fig. 3**), with an overall result being a loss of reactivity.

At high void fractions, the thermal neutron population diminishes, while neutron production in all U and Pu isotopes at high energies increases with spectrum hardening. If the reactivity of fully voided core is higher than the reactivity at nominal operating conditions, the safety case for such a core cannot be easily made without additional provisions for reactivity control in design basis accidents. Therefore, the Pu content in MOX fuel assemblies is typically restricted to below approximately 10%, which is still sufficient for achieving a cycle length comparable to UO_2 fuel with up to 5% enrichment. The limit on maximum Pu loading, however, depends on the fraction of ^{240}Pu and ^{242}Pu . Therefore, MOX fuel with Pu originating from high-burnup UO_2 fuel or Pu that has been recycled more than once (both of which typically have larger fraction of ^{240}Pu and ^{242}Pu) would have more a restrictive loading limit.

Pu isotopes have a slightly different distribution of fission product yields than uranium. The net effect of this is slightly higher parasitic absorption in MOX fuel fission products. This effect is partially compensated by the lower reactivity worth of ^{135}Xe because it is primarily a thermal neutron absorber and MOX fuel has significantly harder spectrum.

The final effect worth mentioning is the considerably lower effective delayed neutron fraction, β_{eff} , of ^{239}Pu compared to ^{235}U . It is lower by about a factor of two. This is a relatively minor issue. It should be considered in assessments of fast reactivity insertion accident scenarios (RIA) such as control rod ejection, main steam line break or boron dilution. Energy deposition in the fuel during a power excursion due to RIA is roughly proportional to the difference between the reactivity inserted and delayed neutron fraction. As the delayed neutron fraction is reduced, fuel energy deposition would increase and could potentially exceed the fuel damage limit. MOX fuel-loaded cores, however, are not substantially different from UO_2 fueled cores because, at the more limiting end of cycle (EOC) conditions, most of the power would be produced by fissions in Pu isotopes in both core types. Furthermore, the effect of a smaller β_{eff} in MOX fuel on the energy deposition in fast transients is more than offset by the stronger negative Doppler reactivity feedback, typically leading to more favorable behavior (even smaller energy deposition) in RIA than in UO_2 fuel cores. In general, power transients in a core with small β_{eff} would be more rapid and could impose some restrictions on the reactor control systems. This may become an issue if large quantities of MA with even smaller β_{eff} are recycled in addition to Pu.

Efficiency of recycling

As mentioned earlier, Pu and MA can be recycled in a variety of host matrices such as UO_2 (MOX fuel), ThO_2 or a range of fertile-free matrix materials. Therefore, the initial Pu and MA will not only be fissioned but may also be newly generated. The set of newly generated nuclides and the rate of their generation will obviously depend on the host matrix and other fuel and core design parameters.

Single-pass recycling of Pu and MA can have two competing objectives. On one hand, Pu can be considered a liability. Therefore, it is desirable to reduce its amount as fast as possible to avoid the risk of proliferation and/or cost of its storage, safeguarding and

final disposal. To achieve this objective (maximize Pu burning rate), recycled Pu fuel needs to be designed such that generation of new fissile plutonium, or the conversion ratio (CR), should be minimized.

On the other hand, Pu (or TRU) can be considered a valuable resource because of the cost and complexity of Pu (or TRU) extraction from used fuel and recycled fuel manufacturing. This will also increase the energy value of natural uranium from which the original fuel containing Pu to be recycled was manufactured. Therefore, there is a clear incentive for the energy recovered from the recycled Pu to be maximized. This can be achieved if the power produced by the secondary, in situ generated fissile isotopes, is as high as possible. This suggests that the CR should also be as high as possible in order to amplify the initial fissile material energy value. Efficient utilization of the recycled Pu (or TRU) would allow achieving high cumulative burnup per unit of initial fissile loading, which in LWRs, also minimizes the fraction of residual fissile material in the used fuel.

The most effective option for achieving the first objective is the use of a fertile-free fuel (FFF) because it has $CR = 0$, that is, every unit of energy produced by the fuel comes from fissioning the original Pu. The transmutation rate in FFF is determined solely by the core power and only slightly varies with initial TRU composition because of the small differences in energy release per fission and atomic mass of different TRU nuclides. For a typical value of about 200 MeV of energy released per fission, slightly over one metric ton of Pu or TRU can be burnt in one 1GW-electric PWR core loaded with FFF per one full power year of operation.

Numerous options for FFF have been proposed and studied (Pöml et al., 2012), but FFF remain a technology with relatively low maturity for it to be readily used in LWRs. Apart from lacking sufficient experience with manufacturing, irradiation performance data and a non-ideal combination of properties (such as thermal conductivity, melting point, radiation damage resistance etc.), FFFs pose a number of operational challenges to an LWR core design. The most important ones are a very large and typically non-linear reactivity loss with burnup resulting from low CR and greatly diminished Doppler reactivity feedback due to the absence of fertile nuclides. A large reactivity loss with burnup may lead to prohibitively high power peaking factors in a multi-batch core, while a reduced Doppler coefficient (DC) can result in excessive energy deposition in the fuel during fast reactivity transients which rely on Doppler Effect to limit the power excursion. Furthermore, for some combinations of TRU with their inert host matrices and burnable poisons, void coefficient of reactivity can become positive, which is another safety concern. These challenges, however, can be in principle mitigated by a careful choice of burnable absorbers and fuel assembly design (Fridman et al., 2007; Schwageraus et al., 2007).

In the case of Pu-UO₂ MOX, which is by far the most well-understood fuel form, the conversion ratio (generation of new Pu from ²³⁸U) is high but can be suppressed somewhat and Pu burning rate increased by varying the relative amount of fuel and moderator. Unfortunately, for a standard PWR fuel assembly geometry, the rate of Pu transmutation is approximately a factor 3 lower compared to FFF.

In order to address the issues related to harder neutron spectrum in Pu (or TRU) bearing fuels without changing the standard fuel geometry, using U-Pu-Zr hydride as a fuel form has been proposed (Ganda and Greenspan, 2009). Hydrogen atoms contained in the fuel provide additional moderation softening the neutron spectrum. As a result, reactivity of the fuel is increased allowing to achieve higher burnup and longer cycle. Alternatively, smaller Pu loading is required in hydride fuel than in MOX to achieve comparable fuel cycle length. In either case, larger fraction of initial Pu can be burnt. About half of the loaded Pu can be transmuted using hydride fuel form in a PWR fuel assembly with conventional design.

Thorium dioxide can serve as an attractive alternative fuel matrix for the transmutation of Pu and MA because it is a very stable fuel form with which we have reasonable experience and, more importantly, does not generate new Pu or MA. In terms of the Pu (or TRU) burning rate, Pu-Th MOX performs in between the Pu-U MOX and FFF, burning about 700 kg of Pu/GWe-Year in a typical PWR fuel assembly geometry (Table 1). It does, however, generate ²³³U and associated thorium transmutation chain nuclides which, similar to Pu and MA, are long-lived and represent a concern for long-term geological storage. If thorium transmutation chain nuclides are added to the total count of TRU mass, the net burning rate of actinides drops to values similar to those of traditional Pu-U MOX.

Th-Pu MOX has other advantages, however. For example, a higher loading of Pu (or TRU) is possible because of the less restrictive void reactivity coefficient (Kotlyar et al., 2017). Otherwise, Pu-Th MOX has very similar operational characteristics to those of traditional Pu-U MOX (Fridman and Kliem, 2011).

Uranium-233 generated from ²³²Th poses a proliferation concern as it is, in principle, a weapons-usable fissile material. It has been argued that because of contamination with trace amounts of ²³²U, which has very high energy photon emitters in its decay chain, ²³³U would be self-protected and can be considered proliferation resistant. However, this argument is not universally accepted in the non-proliferation community and often accompanied by an additional requirement for isotopic dilution (or denaturing) of newly bred ²³³U with ²³⁸U before fabrication of the fresh Pu-Th MOX fuel. The presence of ²³⁸U further deteriorates the Pu (or TRU) burning efficiency (Schwageraus et al., 2004).

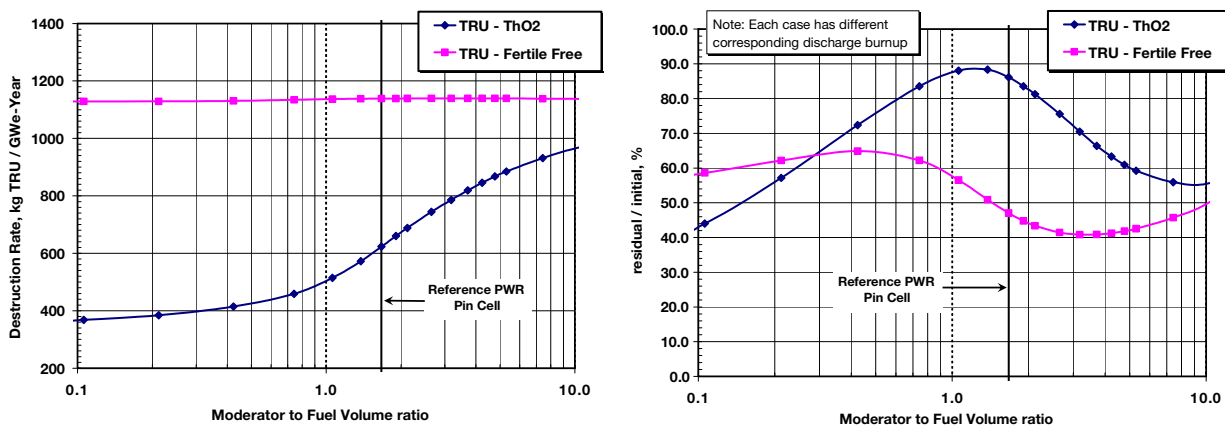
Table 1 compares the single-pass Pu or TRU burning performance of the different fuel types discussed above if used in a typical PWR with a standard fuel assembly geometry and operational characteristics, such as core power density, discharge burnup and fuel cycle length.

The left-hand side of Fig. 4 shows TRU burning rates that can be achieved in Pu-Th MOX and FFF fuel if the constraint on fuel geometry (moderator to fuel volume ratio) is relaxed. The TRU burning rate in FFF is obviously insensitive to the fuel geometry, but in Pu-Th MOX, it can be improved somewhat by increasing the amount of moderator relative to the fuel.

With regards to achieving the second objective of maximum fractional TRU burnup, FFF fuel again has an advantage, as can be seen in Table 1, while UO₂- and ThO₂-based MOX fuels have practically identical performance if ²³³U is counted as TRU.

Table 1 Performance comparison of single-pass Pu and TRU recycling options in a PWR.

	UO ₂	Pu MOX	TRU MOX	Pu ThO ₂	TRU ThO ₂	TRU FFF
Discharge Burnup, MWd/kg	51	46	52	54	55	541
TRU generation rate, kg/GWe-Year	+260	−310	−480	−686	−793	−1150
Initial TRU, kg/assembly	0.0	32.5	74.8	42.8	100.5	48.4
Discharged TRU, kg/assembly	5.5	25.3	63.6	26.8	82.0	21.4
Fractional burnup: TRU	—	0.22	0.15	0.37	0.18	0.56
Fractional burnup: TRU + ²³³ U	—	0.22	0.15	0.25	0.14	0.56
Discharged Pu, kg/GWe-Year	237	152	179	122	173	82

**Fig. 4** Comparison of TRU transmutation efficiency in single-pass Pu-Th MOX and Fertile-free fuel.

The plot on the right-hand side of Fig. 4 demonstrates that a typical PWR fuel design is nearly the worst possible for ThO₂-based MOX, if the TRU burnt fraction is to be maximized. Note that each point along the plot lines has the same initial TRU loading but very different corresponding discharge burnup because the neutron spectrum is significantly affected by the amount of moderator available and so is the proportion of fissile versus fertile absorptions. Both FFF and ThO₂-based MOX can benefit from a softer spectrum because it generally results in higher reactivity during irradiation and, thus, a longer fuel cycle. However, a substantial increase in moderator volume would be impractical without detrimental effects on other operational and core performance characteristics.

Uranium savings and effect on repository

In order to operate a single full PWR core on recycled Pu or TRU fuel, multiple conventional UO₂ cores would be required to generate the necessary fissile material. It can be seen from Table 1 that Pu from about five conventional cores would need to be extracted to feed a single Pu-U MOX core. This so called “support ratio” for other fuel types is even higher, with over 18 conventional cores required to support a single TRU-ThO₂ MOX core. Furthermore, only marginal gains in terms of reducing the support ratio through alterations of fuel dimensions can be practical without negatively affecting the core performance in terms of safety or economics. As a result, the fuel savings achieved through single-pass recycling are relatively modest—on the order of 20% of natural uranium can be saved if single-pass Pu recycling is employed instead of direct disposal of used UO₂ fuel.

Used fuel characteristics relevant to geological disposal such as radiotoxicity and decay heat can be improved if the amount of actinides is reduced. However, this benefit is also marginal. Fig. 5 compares waste disposal characteristics of the fuel types presented in Table 1. The plot on the left-hand side of the figure presents the decay heat generation for actinides (no FP) from the used fuel normalized to total energy produced by the fuels. The energy for normalization includes both the conventional UO₂ cores and a TRU-loaded core they support. As evident from the figure, all fuel types are relatively close to each other and to the reference UO₂ case. The radiotoxicity of actinides in used fuel from Pu/TRU cores relative to those from UO₂ cores with an equivalent amount of energy output is compared in the right-hand side of Fig. 5. The radiotoxicity of actinides is not particularly important in the first hundred years or so because during this period, the used fuel characteristics are dominated by the FP. In the longer term, different options do offer some benefit, reducing the radiotoxicity by up to factor of 2–3 depending on the fuel option and time period of interest.

This reduction, however, was found to be insufficient to make a substantial difference to simplification of the safety or the costs of construction and operation of a geological repository. Instead, at least two orders of magnitude (factor of 100) reduction in loss rate of TRU from a fuel cycle into the repository is needed. Exceeding a 99% burnup fraction of TRU in a single-pass recycling is impractical to achieve. It can only be achieved if all TRUs are recycled indefinitely within either the same reactor fleet (e.g.,

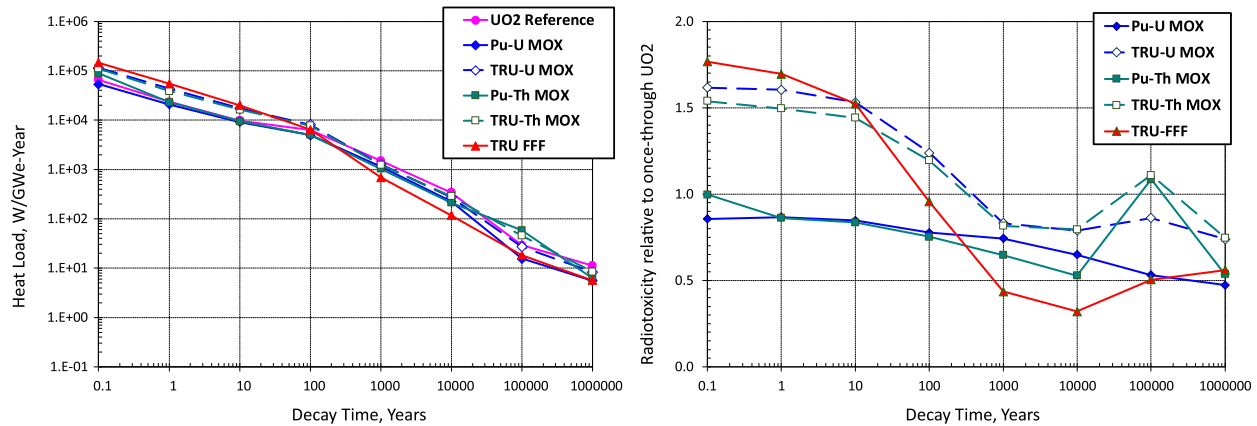


Fig. 5 Comparison of decay heat and ingestion radiotoxicity for single-pass Pu/TRU burning options.

LWRs) or a fleet of dedicated TRU transmutation systems (e.g., critical or external neutron source-driven fast spectrum reactors) with only 0.1% of TRU mass allowed to be leaked to the waste streams due to inefficiencies in the reprocessing and fabrication stages (Van Tuyle and Finck, 2001). The allowable TRU loss rate is determined by the requirement to reduce the time taken for the radiotoxicity of used fuel to reach that of the natural uranium used for its fabrication from several hundred thousand to less than a thousand years.

Multi-recycling of Pu and MA in LWRs

As mentioned earlier in this chapter, the meaningful environmental benefits of a closed fuel cycle can only be realized if the following conditions are met:

- MA are recycled in addition to Pu;
- Pu and MA are not just recycled once, but kept within the fuel cycle indefinitely;
- TRU losses to the repository as a result of inefficiencies in recycling processes are kept below 0.1%.

These conditions pose several challenges to fuel cycle facilities, core design and reactor operation. This chapter will mostly focus on the latter. The challenges are mostly linked to the increasing amount of MA relative to Pu and the reduced fraction of fissile isotopes with the number of consecutive recycling stages (passes through the system).

Practically all the LWR operational challenges mentioned in the discussion of single-pass Pu recycling as MOX fuel, such as degradation in reactivity worth of reactivity control materials and restrictive void reactivity coefficient, are also relevant to TRU multi-recycling. Many of the issues, however, become progressively more severe. In fact, recycling of Pu as MOX beyond a second pass becomes impractical without modification of the fuel design.

The reduction in the relative fractions of fissile ^{239}Pu and ^{241}Pu in irradiated MOX fuel implies that if this Pu is further recycled, a higher Pu loading would be needed in the next recycle of MOX in order to maintain the same fuel cycle length and discharge burnup. It also implies that subsequent recycles of MOX fuel will have even higher fractions of ^{240}Pu and ^{242}Pu isotopes. As stated earlier, these are the isotopes responsible for the limit on Pu loading associated with the positive void reactivity effect.

The most fundamental challenge to feasibility of multi-recycling, however, even before considering any of the practicality issues mentioned above, is the ability of an LWR fleet to achieve steady state operation in which the total amount of TRU in the system is stabilized (zero net production of TRU). In order to achieve such a steady state, a sufficient amount of neutrons should be available in the system which, in addition to being able to maintain the criticality of the system for a reasonable time, can also be used to transmute TRU nuclides at the rate required to assure zero net production.

Some TRU nuclides are fissile in a thermal LWR spectrum and therefore produce on average a surplus of neutrons. Other TRU nuclides, however, are non-fissile in thermal spectrum and therefore require at least one or more neutron captures before they are converted into fissile nuclides and, eventually, into fission products.

It has been argued and generally accepted (Wigeland et al., 2021) that fast reactors are more efficient for the recycling and transmutation of TRU than LWRs (or thermal reactors in general). This is because the ratio of fission to capture reaction cross sections in most TRU is more favorable in a fast spectrum (see for example Fig. 5 in Shusterman, 2021) and the average number of neutrons released per fission is greater. In other words, more neutrons are available in the fast spectrum while fewer are required on average to transmute a relevant mix of TRU into short-lived fission products (Salvatores, 2002). Furthermore, the core physics of fast reactors is less sensitive to TRU isotopic mix, because practically all TRU isotopes are fissile, offering greater tolerance of the wide range of composition variations that result from multi-recycling.

The greater number of neutrons produced by fission in a fast spectrum is significantly offset by the high neutron leakage in fast reactors which typically have compact (high surface to volume ratio) cores to feature high neutron leakage probability in order to limit the magnitude of positive reactivity effect due to coolant thermal expansion or voiding—a major safety concern in sodium-cooled fast reactors.

LWRs too have an excess of neutrons during their operating cycle. This excess is typically compensated by various absorber materials (e.g., control rods, soluble and/or burnable poisons). It can be argued that, in a carefully designed LWR core, these excess neutrons can be used instead for transmutation of TRU. If the available excess of neutrons is insufficient for achieving the transmutation rate needed for zero net production of TRU, one solution is to boost the feed of fissile nuclides by using a slightly higher uranium enrichment. In this case, the trade-off to be assessed is the environmental benefits of zero net TRU production versus the costs of a higher enriched uranium feed.

Two basic alternatives can be imagined for achieving a zero TRU balance in an LWR fleet of reactors. One method is to dedicate certain regions in every reactor core to fuel containing TRU generated by the same core in previous cycles, while the rest of the core is loaded with conventional UO_2 fuel. In this case, the core physics behavior could still be dominated by the conventional UO_2 fuel and minimal modifications to the reactor system would be required.

The alternative is to accept the significant design modifications required to handle multi-recycled TRU-loaded fuel and operate a small number of dedicated LWRs with such a modified core design as net TRU burners. In this case, other reactors in the fleet can operate with conventional fuel with no modifications.

It has been shown in multiple studies that both options are viable and can, in principle, operate with a variety of fuel forms for recycled TRU, including but not limited to uranium or thorium-based MOX or FFF.

Pioneering work on Pu recycling in PWRs has been done in France as their nuclear program from its earliest days considered various options for stabilizing the Pu inventory as an alternative to, or in combination with, sodium-cooled fast reactors. For enabling multi-recycling of Pu and some MA, an Advanced Plutonium Assembly (APA) concept was proposed (Puill and Bergeron, 1997). It is schematically shown in the right-hand side of Fig. 6. The assembly consists of a combination of low-enriched UO_2 fuel pins and large annular Pu/TRU-bearing inert matrix fuel pins. Large amount of water around TRU pins allows better moderation to mitigate the spectrum hardening effects of MOX fuel and provides better cooling to accommodate the higher power generation. Furthermore, the fertile-free fuel matrix facilitates high transmutation efficiency (high burnup fraction of TRU) and, thus, helps keep the TRU inventory in the assembly at a minimum. It has been shown that a fleet of PWRs, of which only about 30–40% of cores (depending on whether only Pu or also MA are recycled) are fully loaded with APA fuel, can achieve zero net TRU production (Golfier et al., 2001).

Another option, also discussed in (Shwageraus, 2021), which has already attracted notable development effort and has promising performance characteristics, is a derivative of the Resource-renewable BWR concept (RBWR-TB/TB2) by Hitachi. It is a tight-pitch, high-void fraction BWR designed to maximize TRU transmutation efficiency (Hino et al., 2015). A smaller amount of

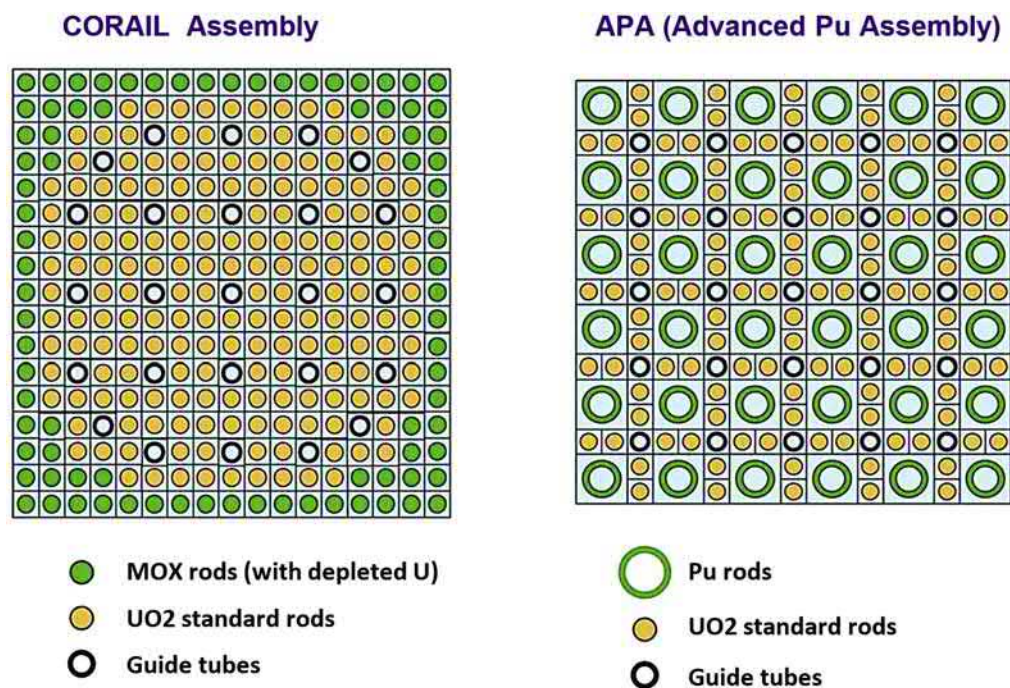


Fig. 6 PWR fuel assembly designs for Pu recycling. From Thomas JB (2002) Nuclear power plant types and the management of plutonium and minor actinides—In search of fuel cycle flexibility. *Comptes Rendus Physique* 3(7–8): 783–796.

moderator makes the neutron spectrum sufficiently hard so as to take advantage of fast neutron properties for transmutation such as a higher probability of fission relative to capture in all nuclides and a larger number of neutrons released per fission.

The RBWR-TB core has smaller aspect ratio (flatter) than a conventional ABWR core to increase the neutron leakage and mitigate the positive void reactivity effect typical of fast reactors. With increasing void fraction, the neutron spectrum becomes harder, while the core materials' cross-sections become smaller. This leads to higher neutron leakage and loss of reactivity, compensating for the positive void reactivity effect at high void fractions which is also a common feature of MOX fuel with high TRU content. The RBWR-TB fuel is axially heterogeneous and consists of a natural or depleted uranium oxide zone sandwiched between two fissile U-TRU MOX zones (Fig. 7). With increasing void fraction, neutron leakage from the fissile zones to the internal blanket (a neutron sink) and to the axial reflectors is enhanced, compensating for the positive reactivity effect of spectrum hardening, to provide a net negative void reactivity effect.

More exotic and less developed alternatives have also been proposed, for example, a BWR fuel in which the top third of the length, where the void fraction is high, is loaded with recycled TRU in non-fertile nitride form or as U-TRU MOX (Wallenius and Westlén, 2008). Zirconium cladding is replaced in the top third of the fuel length with hafnium which is a strong thermal neutron absorber. These design modifications also result in sufficiently hard neutron spectrum to operate such a reactor as a net TRU burner supporting a fleet of reactors with conventional fuel.

Operating a fleet of LWRs with considerably different characteristics has an advantage of limiting the complexities of handling TRU-containing fuel to only a fraction of the reactors in the fleet. If this fraction, however, is significant, it might be more convenient and cost-effective to operate a fleet of identical plants, all operating with recycled TRU fuel, provided that such fuel characteristics do not drastically deviate from those of conventional fuel and do not require major reactor design, operating procedures and safety case modifications.

The above requirements generated a range of fuel assembly concepts which can recycle TRU within the same core. The fuel assemblies would consist mostly of fuel pins with conventional, low-enriched UO_2 fuel and a small fraction of TRU containing pins. The TRU pins would contain Pu and MA from both UO_2 and TRU pins, generated in or leftover from the previous irradiation cycle respectively. In order to maintain the core characteristics as close to conventional UO_2 fuel as possible, while maintaining zero net generation of TRU, as a general rule, the transmutation efficiency (TRU fractional burnup per recycling pass) should be as high as possible. This will help minimize the TRU inventory in an assembly and mitigate undesirable effects from TRU on the core physics.

Examples of such heterogeneous fuel assembly concepts include CORAIL (COMbustible Recyclable A ILot—French for “recycled fuel island”) which uses U-TRU MOX pins for transmutation in an otherwise conventional PWR fuel assembly (Aniel et al., 2000). A schematic view of CORAIL PWR fuel assembly is shown in the left-hand side of Fig. 6. A similar design uses a different TRU host fuel matrix. The CONFU (Combined Non-Fertile and Uranium) fuel assembly uses an inert matrix fuel (Shwageraus et al., 2005a). Due to the superior transmutation efficiency of FFF, i.e., no new TRU generation in TRU pins, the inventory of Pu and MA in the CONFU assembly is notably smaller than in the CORAIL assembly.

Thorium oxide can also be used as a matrix for TRU, achieving stabilization of the TRU inventory after several recycling passes (Shwageraus et al., 2005b). The inventory of TRU per assembly is comparable to CORAIL, if ^{233}U is recycled into UO_2 rather than ThO_2 pins, but higher than in CONFU. Other conceptual studies showed that the TRU inventory can also be stabilized in other heterogeneous arrangements (Lindley et al., 2014) or in a homogeneously mixed tertiary U-Th-TRU MOX fuel. In the latter case, all pins within a fuel assembly would have a uniform composition, i.e., no dedicated TRU bearing pins (Fridman et al., 2006).

All of the above concepts share a number of features. They all operate in a typical PWR environment with a thermal neutron spectrum. Therefore, the fissile content of the TRU isotopic vector progressively deteriorates with the number of recycling passes.

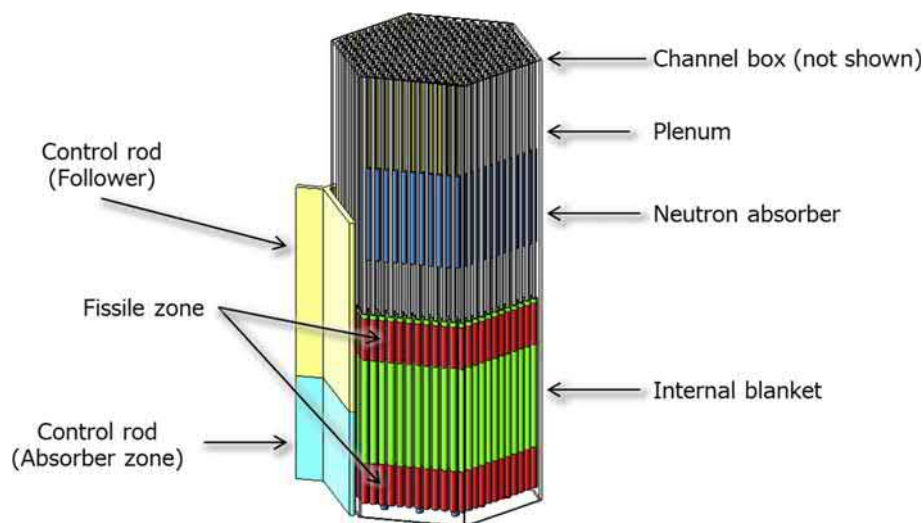


Fig. 7 Schematic view of RBWR-TB (Transuranics Burner) fuel assembly. Courtesy of Hitachi Ltd. 2020 with permission.

It takes multiple recycles for the most important isotopes to approach equilibrium. The concentration of some isotopes continues growing because of their small absorption cross section. The fuel cycle length is reduced with a deteriorating fissile quality of TRU, but this saturates after only about 3–4 recycling passes at a value about 10% lower than the reference conventional UO₂ fuel. It means that a similar fractional increase in feed uranium enrichment would be needed to compensate for that fuel cycle length reduction.

These studies have also found that these conceptual fuel assembly designs exhibit similar characteristics to those of traditional UO₂ fuel, meaning that no major changes to core design, reactor operation and safety case would be required.

The main challenges with multi-recycling of TRU are therefore associated with operation of fuel cycle facilities—not the reactor fleet. With the evolving TRU isotopic vector, the decay heat and radiation dose from the recycled TRU generally increase. This is due to the accumulation of relatively short-lived isotopes such as ²³⁸Pu, ²⁴¹Am, ²⁵²Cf and Cm isotopes with a high decay rate and associated emission of α -particles, high-energy photons or neutrons. These isotopes are very difficult to handle because of the cooling and radiation shielding requirements, which may become prohibitively expensive or even make the reprocessing and fabrication of TRU pins technically unfeasible.

Some of the isotopes, such as ²⁴²Cm ($T_{1/2} = 162.7$ days), ²⁴⁴Cm ($T_{1/2} = 18.1$ years) and ²⁵²Cf ($T_{1/2} = 2.64$ years), are sufficiently short-lived that longer cooling times can partially mitigate these challenges allowing a greater number of recycles in a given fuel cycle infrastructure. Fig. 8 compares the neutron source from spontaneous fission in TRU from CONFU fuel recycled using different cooling periods between fuel discharge and reprocessing. If the cooling is increased from 7 to 20 years, the spontaneous fission source is reduced to the levels comparable to conventional Pu MOX fuel routinely handled commercially.

An additional challenge is the technical feasibility, efficiency and cost of TRU chemical separation processes which must only allow < 1% TRU loss to the repository. A vast international R&D effort has already been invested in this area and many promising candidate processes have been identified for commercial application. Review of these techniques is presented in (Miguirditchian, 2021).

Summary

This chapter presented a brief overview of the conventional once-through (open) fuel cycle and discussed incentives for transitioning to a closed one while still using existing or evolutionary LWRs.

Partially closing the fuel cycle is possible by only recycling Pu or Pu and MA in LWR cores only once. The strategy of single-pass Pu recycling is already well-established in many countries with multiple available commercial scale facilities. All aspects of reactor design and operation as well as those of the associated fuel cycle infrastructure are well-understood. However, the benefits of such single-pass recycling for increase in uranium resource utilization, or for reduction in the radiotoxicity of fuel cycle waste products are marginal. This is because Pu from multiple conventional UO₂ fuel LWR cores is needed to feed a single MOX core while the net burnup of Pu in MOX fuel is small.

Studies investigating alternative fuel matrices for single-pass Pu or TRU recycling such as ThO₂ or Fertile-free Fuels arrived at similar conclusions for similar reasons.

Radiotoxicity of used LWR fuel reduces with time. It reaches the point at which it is equal to radiotoxicity of the originally mined uranium used to manufacture it after several hundred thousand years. It has been shown that in order to shorten this time to less than a thousand years, the fraction of TRU which is allowed to be lost to the repository in addition to fission products should be less than 0.1%. This is possible only if TRU are kept within the fuel cycle (recycled) indefinitely and chemical processes are available to deliver such a TRU separation efficiency.

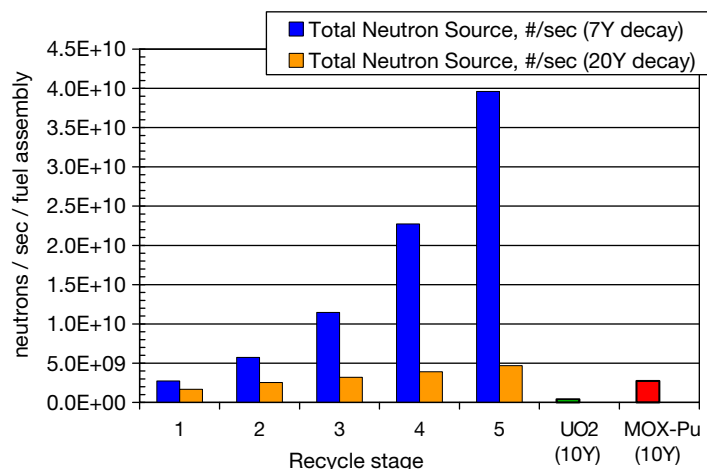


Fig. 8 Effect of cooling time on Spontaneous Fission Source of recycled TRU fuel. From Shwageraus E, Hejzlar P, and Kazimi MS (2005a) A combined non-fertile and UO₂ PWR fuel assembly for actinide waste minimization. *Nuclear Technology* 149(3): 281–303.

Indefinite TRU recycling is easily achievable in fast spectrum reactors due to abundance of neutrons available for transmutation and the fact that fewer neutrons are required for TRU conversion to short-lived fission products.

Indefinite recycling in LWRs, however, is more challenging. It is possible by either developing fuel designs with a sufficiently hard neutron spectrum or supplementing the neutrons required for transmutation in thermal spectrum by slightly increasing the enrichment of the uranium used as a driver fuel. Several relatively mature designs exist for both options such as Hitachi's RBWR-TB or the CORAIL PWR fuel concept and their derivatives. A significant body of knowledge has been generated from these studies, suggesting that TRU recycling in LWRs is feasible—that is, a closed fuel cycle with zero net generation of TRU using well-established and well-understood LWR technology can be achieved.

The limiting issue remains the development of fuel cycle infrastructure capable of handling multiple TRU recycling passes. Changes in the TRU isotopic vector lead to increasing radiation shielding and cooling requirements which complicate practically all fuel cycle operations associated with recycling, manufacturing and transportation of TRU fuel. Allowing a greater cooling period between used fuel discharge and reprocessing operations can somewhat mitigate these challenges.

It appears that technological feasibility of a closed LWR fuel cycle is not the reason for lack of progress in implementing it. The relatively slow growth of nuclear power over recent decades and advances in geology and extraction technologies of natural uranium diminish the economic incentive for used LWR fuel recycling. The non-economic incentive of preserving a valuable energy resource for future generations is also not seen as sufficiently compelling. Furthermore, the environmental benefits of closed fuel cycle are difficult to quantify in monetary terms because of the time scale of the radioactive waste isolation problem, but it is still important for each generation to internalize its industrial waste instead of leaving it to future generations. Therefore, it can be argued that pursuing the closed fuel cycle is the ethically responsible choice for modern society, even at the expense of somewhat higher economic costs.

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See Also: Impact of Small Modular Reactors on the Acceptance of Nuclear Power by the Public, Investors, and Owners.

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Advanced Fuel Cycles Identification and Evaluation

R. Wigeland^{a,*}, T. Taiwo^b, M. Todosow^c, W. Halsey^{d,*}, and J. Gehin^a, ^aIdaho National Laboratory, Idaho Falls, ID, United States; ^bArgonne National Laboratory, Lemont, IL, United States; ^cBrookhaven National Laboratory, Upton, NY, United States; and ^dLawrence Livermore National Laboratory, Livermore, CA, United States

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Introduction

In principle there are numerous options for nuclear fuel and fuel cycles based on the underlying nuclear physics. Research and development (R&D) of various nuclear fuel and fuel cycle options has been ongoing over the past 80 years, with many implementing technologies being explored. Today, fuel cycles and new technologies are being proposed and developed for a variety of purposes, including more efficient use of fuel resources, higher thermal efficiency, waste reduction, and lower capital and operating costs. The U.S. Department of Energy, Office of Nuclear Energy (DOE-NE) has been conducting R&D on a range of nuclear fuel and nuclear fuel cycles, guided by past studies and ongoing research. (US Department of Energy, 1980, 2002, 2008; International Atomic Energy Agency, 1980, 2009; Los Alamos National Laboratory, 2001; Massachusetts Institute of Technology, 2003, 2010; Wigeland et al., 2009, 2010).

All of these previous studies are constrained in some manner, either by the scope and consistency of the criteria used for evaluating different fuel cycles or by the range of fuel cycles considered, and as such are of limited use. To address the need for complete information to support R&D decisions, a study was conducted which reflected both the broad range of issues relevant to the present time and that examined a comprehensive range of fuel cycle options. This chapter briefly describes the scope of this comprehensive study, the methodology it used including the Evaluation Criteria, and selected findings regarding the benefits from and challenges in the implementation of the different fuel cycle options.

The nuclear fuel cycle evaluation and screening study

In late 2011, the U.S. Department of Energy, Office of Nuclear Energy (DOE-NE) chartered a new study on the identification and analysis of nuclear fuel cycle options, referred to as the Evaluation and Screening Study (E&S Study), to provide information about the potential benefits and challenges of nuclear fuel cycle options and, if possible, to identify a relatively small number of promising

* Retired.

fuel cycle options with the potential for achieving substantial improvements compared to the current nuclear fuel cycle in the United States (US Department of Energy, 2011b). The E&S Study defined the nuclear fuel cycle as shown in Fig. 1, which shows the components of a nuclear fuel cycle.

In the E&S Study Charter, criteria were specified to be used for evaluation of each fuel cycle option. The E&S Study was part of a larger effort to achieve Objective 3 “Develop sustainable nuclear fuel cycles” in the DOE Nuclear Energy Research and Development Roadmap (US Department of Energy, 2010), and followed the pilot demonstration of an evaluation and screening process (US Department of Energy, 2011a). The E&S Study addressed feedback from internal and external reviews of the pilot demonstration which identified the required areas of refinement and improvement of the process (US Government Accountability Office, 2011). A complete description of the E&S Study, including all results and analyses, is available in the final report for the study (Wigeland et al., 2014). A subsequent report discusses further details concerning minor actinide recycle (Wigeland, 2015). The final report on the E&S Study is available at: www/fuelcycleevaluation.inl.gov/SitePages/Home.aspx.

The evaluation and screening software (the Evaluation and Screening Tool, SET) and metric data are also available for download and use on the website, allowing exploration of any desired combination of metrics and criteria by any interested individuals.

Fuel cycle Evaluation Criteria

The E&S Study Charter (US Department of Energy, 2011b) specified nine Evaluation Criteria representing broadly-defined economic, environmental, safety, non-proliferation, security and sustainability goals to identify promising fuel cycle options and measure improvements as compared to the current nuclear fuel cycle in the United States. The first six criteria are related to the potential for benefit and the last three reflecting the challenges for developing and deploying a new fuel cycle.

Nine DOE-specified Evaluation Criteria

- Nuclear Waste Management (Study considered waste generation only)
- Proliferation risk
- Nuclear material security risk
- Safety
- Environmental impact
- Resource utilization
- Development and deployment risk
- Institutional issues
- Financial risk and economics

Nuclear Waste Management Criterion—Some nuclear fuel cycles may generate less waste than others, but all fuel cycles create similar wastes, both low-level and high-level radioactive wastes. Consequently, all nuclear fuel cycles require waste disposal capabilities, including the need for long-term isolation of some wastes. As a result, the E&S Study focused on the quantity and characteristics of the radioactive wastes generated by the different fuel cycles, including the current U.S. fuel cycle, not on the details of waste disposal such as geologic disposal environments.

Proliferation Risk Criterion—In general, assessing proliferation risk is a complex and challenging endeavor, primarily because it involves both technical and socio-political considerations, with the dominant factor being facility location. Since most of these factors are beyond the scope of the E&S Study, efforts focused only on evaluation of technical differences between fuel cycle options at the physics-based functional level that may influence proliferation risk (this study did not consider any specific implementing technologies as described in the following section).

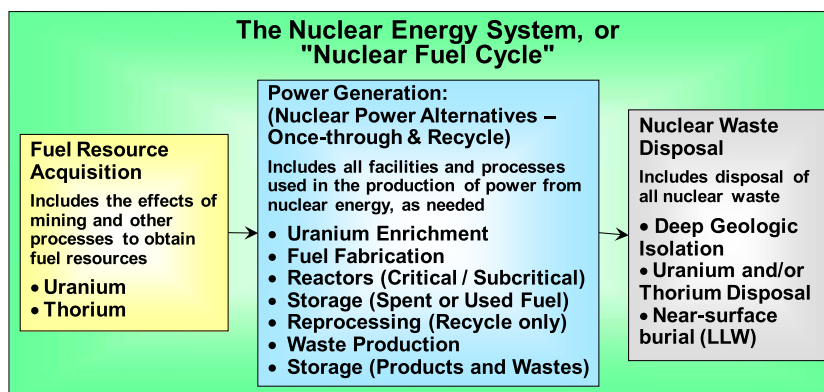


Fig. 1 The Nuclear Energy System, also known as the Nuclear Fuel Cycle.

Nuclear Material Security Risk Criterion—The E&S Study could only inform on the materials available from the fuel cycle that would require physical protection. The other aspects of physical protection relevant to nuclear material security risk are a function of specific facility designs and operations, including physical barriers and assumptions made about the protective force and adversary force capabilities. These were not considerations in the E&S Study of fuel cycles.

Safety Criterion—The E&S Study considered whether a fuel cycle could be safely deployed and the relative challenges in addressing safety hazards for an alternative fuel cycle in comparison to the current U.S. fuel cycle for all of the facilities required for each fuel cycle. The E&S Study did not consider general questions on the acceptability of the current safety of nuclear power as deployed in the U.S.

Environmental Impact Criterion—The E&S Study considered the environmental impacts from the routine operations of a nuclear fuel cycle focusing on impacts from fuel acquisition and nuclear power generation. Environmental impacts from accidents at fuel cycle facilities are not included in this criterion since these are part of the Safety criterion. Similarly, the E&S Study did not consider the environmental impacts of waste disposal under this criterion since they are represented, directly or indirectly, in the Nuclear Waste Management criterion. The information in the E&S Study is only about the relative changes in such impacts between fuel cycle options, and not about whether such impacts are ultimately acceptable.

Resource Utilization Criterion—The E&S Study only considered the natural resources required for nuclear fuel (i.e., uranium and thorium), not resources in general.

Development and Deployment Risk Criterion—The E&S Study considered technology development needs for fuel cycle options including what would be necessary for maturing the technologies and factors that would affect deployment of a first-of-a-kind facility and integration of all parts of the entire fuel cycle.

Institutional Issues Criterion—The E&S Study considered issues such as the existing infrastructure, current regulations, and market conditions and any different supporting needs that alternative fuel cycles would have as potential challenges to the deployment of a fuel cycle.

Financial Risk and Economics Criterion—The E&S Study considered the relative differences in financial risk and economics among nuclear fuel cycle options. However, the E&S Study did not consider the overall economic viability of nuclear power in the U.S.

Evaluation Metrics

Evaluation Metrics were developed for the nine specified Evaluation Criteria, coordinated with input from DOE, industry, universities, and others through collaborations, meetings and reviews, as directed by the Study Charter. This activity resulted in one or more Evaluation Metrics for each Evaluation Criterion along with the justification for each metric and the methods to be used to calculate or estimate the metrics (Wigeland et al., 2014).

Table 1 shows the Evaluation Metrics for the nine Evaluation Criteria. The first six Criteria represented opportunities for improvement (or “benefit”) when compared to the current U.S. fuel cycle, while the other three are related to the “challenges” associated with developing and implementing any new fuel cycle. As a consequence, the current U.S. fuel cycle would likely perform best for the “challenge” criteria relating to development and deployment since this fuel cycle is already in use, with the exception of spent nuclear fuel (SNF) disposal, but it should be noted that disposal capabilities such as those required for SNF are needed by all fuel cycles for managing SNF and high level waste (HLW).

The possible data range for each Evaluation Metric was divided into sub-ranges, called data “bins”, with the result for each Evaluation Metric being a data bin value rather than a specific numerical value. This approach enabled a comprehensive assessment of all possible fuel cycles by removing the need for detailed specification of every part of a fuel cycle.

Comprehensive set of fuel cycle options

To allow the development and performance comparison of the comprehensive set of fuel cycle options, the E&S Study analyzed, evaluated, and screened nuclear fuel cycles only at what is termed the “functional” level, using the fundamental physics characteristics of each step in a fuel cycle (i.e., the physics principles defining what happens at each fuel cycle step, not the technologies for how a step is accomplished), resulting in a “technology-neutral” assessment. For example, a pressurized-water reactor (PWR) is a specific technology for implementing the function of thermal neutron irradiation. Similarly, reprocessing using PUREX to isolate and recover uranium and plutonium is a specific technology for implementing the function of recovering uranium and plutonium from irradiated fuel for reuse. For the E&S Study to be comprehensive in consideration of the relative performance of all possible fuel cycles, the E&S Study only evaluated fuel cycle options using such functions as thermal neutron irradiation, or reprocessing to recover uranium and plutonium, and did not evaluate or screen either specific technology options or implementation/deployment options.

Representing the comprehensive set of fuel cycle options with a finite set of groups

- Although there can be an almost endless set of fuel cycle options when all technology and fuel cycle option choices are considered, there is only a relatively smaller set of fuel cycle option groups based on fundamental physics principles.
- Defining the finite set of groups requires consideration only of the physics principles.
- Accounting for the similarities among options, 40 distinct groups were identified as sufficient to represent possible fuel cycle options for the Evaluation Criteria specific to this study.

Table 1 Evaluation Criteria and Evaluation Metrics.

<i>“Benefit” Criteria</i>	
Nuclear Waste Management	Mass of SNF + HLW disposed per energy generated
	Activity of SNF + HLW (@100 years) per energy generated
	Activity of SNF + HLW (@100,000 years) per energy generated
	Mass of DU + RU + RTh disposed per energy generated ^a
	Volume of LLW per energy generated
Proliferation Risk	Material attractiveness—normal operating conditions
	Material attractiveness—normal operating conditions
Nuclear Material Security Risk	Activity of SNF + HLW (@10 years) per energy generated
	Challenges of addressing safety hazards
Safety	Safety of the deployed system
	Land use per energy generated
	Water use per energy generated
	Radiological exposure—total estimated worker dose per energy generated
Environmental Impact	Carbon emission—CO ₂ released per energy generated
	Natural Uranium required per energy generated
Resource Utilization	Natural Thorium required per energy generated
<i>“Challenge” Criteria</i>	
Development and Deployment Risk	Development time
	Development cost
	Deployment cost from prototypic validation to FOAK commercial
	Compatibility with the existing infrastructure
	Existence of regulations for the fuel cycle and familiarity with licensing
	Existence of market incentives and/or barriers to commercial implementation of fuel cycle processes
	Compatibility with the existing infrastructure
	Existence of regulations for the fuel cycle and familiarity with licensing
Institutional Issues	Existence of market incentives and/or barriers to commercial implementation
Financial Risk and Economic	Levelized Cost of Electricity at Equilibrium

^aDU, RU, and RTh are depleted uranium, recovered uranium, and recovered thorium, respectively.

Fundamental fuel cycle characteristics

The definition of a nuclear fuel cycle used in the E&S Study begins with mining, and ends with the generation of wastes requiring disposal.

The study considered the entire possible range for each physics principle, including once-through and recycle fuel cycles; thermal, intermediate and fast neutron reactors; critical and sub-critical (externally-driven systems, or EDS) reactors; and uranium and/or thorium for natural fuel resources, along with other distinguishing fuel cycle features. All possible combinations of the physics principles resulted in the comprehensive set of possible fuel cycles, as follows:

1. “Once-Through” or “Recycle” where recycle includes limited recycle and continuous recycle
 - Limited recycle fuel cycles are those that recycle only once or a limited number of times before SNF is disposed along with high level waste generated from the recycle processes
 - Continuous recycle fuel cycles are those that always reprocess irradiated fuel for recycle and only high-level waste (HLW) is disposed
2. Irradiation system
 - Self-sustaining (critical reactor)
 - Externally-driven (sub-critical reactor)
3. Neutron spectrum (as defined in Appendix B of [Wigeland et al. \(2014\)](#))
 - Thermal
 - Intermediate
 - Fast
4. Nuclear fuel
 - Uranium
 - Thorium and thorium/uranium
5. Isotopic enrichment
 - Uranium enrichment
 - No Uranium enrichment
6. Recycled elements
 - One or more of the following: U (includes ²³³U bred from Th); Pu; minor actinides (MA); All transuranic elements (TRU); Th; fission products (FP)

The permutations of these functional characteristics resulted in about 4400 potentially viable Fuel Cycle Option Groups. Starting with this set of Fuel Cycle Option Groups, many of these groups were combined using a series of operations based on the similarity of their expected physics-based performance with respect to the Evaluation Criteria (discussed in Wigeland et al. (2014)), such as waste generation or resource use, with the principle of ensuring that promising options would be identified by the Evaluation and Screening, but that no promising option would be inadvertently screened out by being placed in a lesser performing Evaluation Group. At the end of this process, 40 groups of fuel cycles called Evaluation Groups were obtained that provide a comprehensive representation of the performance of all possible nuclear fuel cycles to inform on their potential for providing substantial improvement to the current U.S. fuel cycle:

- 8 Once-through Evaluation Groups
- 10 Limited recycle Evaluation Groups
- 22 Continuous recycle Evaluation Groups

Even though the fuel cycles were collected based on similarity of physics-based performance, the overall performance would not necessarily be equal for all fuel cycles in an Evaluation Group. Since the fuel cycles collected within each Evaluation Group potentially span a range of overall performance, it was recognized that some of the collected fuel cycles in each Evaluation Group may actually be relatively poorer performers overall when compared to the best fuel cycles in the group. To appropriately inform on the potential of each group, the best performance potential for each Evaluation Group and each Evaluation Metric was evaluated, as described in Wigeland et al. (2014), even though all members of the Group would not necessarily be able to achieve such performance. Table 2 contains short descriptions of the 40 Evaluation Groups indicative of the fuel cycle options included in each group, but are not complete descriptions of the Evaluation Group in most cases; see the final report for complete details (Wigeland et al., 2014). Evaluation Group EG01 served as the “Basis of Comparison,” representing the current U.S. fuel cycle.

Best performance potential of each evaluation group

“Analysis Examples,” were developed for each Evaluation Group by specifying the technical functions and specifications of the irradiation environment for a representative fuel cycle in a Fuel Cycle Option Group in each Evaluation Group to provide an example that represents the focus of the Evaluation Group, e.g., specifying a PWR as the thermal reactor in a fuel cycle. The Analysis Example was used for calculating detailed reactor physics-based material mass balance information and other information as appropriate for determining the result for each Evaluation Metric, i.e., which data bin represented the Analysis Example performance.

The results were reviewed to determine if any characteristic of the Analysis Example resulted in performance that could be less than the best performance for all fuel cycles in the Evaluation Group, and revised if necessary. (Wigeland et al., 2014).

Evaluation Criteria weighting scenarios

Following completion of the determination of the Evaluation Metrics, the performance comparison between the Evaluation Groups was conducted. Eleven multiple-criteria scenarios were created to represent a variety of different perspectives about the relative importance (weight, or tradeoff factor) of changes in four of the six benefit criteria (Nuclear Waste Management, Resource Utilization, Environmental Impact, and Safety) and to investigate the sensitivity of the results to changes in those perspectives. The scenarios were grouped into three categories:

- Scenario 1—Changes in the four listed benefit criteria were of equal importance.
- Scenarios 2 through 5—These four scenarios each emphasized changes in a single benefit criterion with respect to a balance of the three remaining criteria.

Six other scenarios were defined, each exploring an emphasis on a sub-set of these four benefit criteria, each defined to reflect one of a variety of perspectives. These scenarios were defined as:

- Scenario 6—emphasize the importance of differences between Evaluation Groups on the Nuclear Waste Management, Resource Utilization, and Environmental Impact criteria—to emphasize the importance of differences in the physical impacts of producing nuclear power and the potential to reduce the impacts by choice of fuel cycle.
- Scenario 7—de-emphasize the importance of differences between Evaluation Groups on the Environmental Impact criterion, focusing instead on the potential for improvement in Nuclear Waste Management, Resource Utilization, and Safety Criteria based on choice of fuel cycle.
- Scenario 8—de-emphasize the importance of differences between Evaluation Groups on the Resource Utilization criterion—to explore the potential impact of expanded fuel resource availability (such as uranium from seawater) and its effect on the relative benefits of fuel cycles. This scenario also provides insight on whether Resource Utilization as a separate criterion adds a different perspective to the results.

Table 2 The 40 Evaluation Groups.

<i>Evaluation Group</i>	<i>Short Description Indicative of Fuel Cycles in the Evaluation Group (Detailed Description of Each Evaluation Group is in Appendix B of Wigeland et al. (2014))</i>
<i>Once-through</i>	
EG01	Once-through using enriched-U fuel in thermal critical reactors
EG02	Once-through using enriched-U fuel to high burnup in thermal or fast critical reactor
EG03	Once-through using natural-U fuel in thermal critical reactors
EG04	Once-through using natural-U fuel to very high burnup in fast critical reactors
EG05	Once-through using enriched-U/Th fuel in thermal or fast critical reactors
EG06	Once-through using Th fuel to very high burnup in thermal EDS
EG07	Once-through using natural-U fuel to very high burnup in thermal or fast EDS
EG08	Once-through using Th fuel to very high burnup in fast EDS
<i>Limited Recycle</i>	
EG09	Limited recycle of U/TRU with new natural-U fuel to very high burnup in fast critical reactors
EG10	Limited recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in fast and/or thermal critical reactors
EG11	Limited recycle of $^{233}\text{U}/\text{Th}$ with new enriched-U/Th fuel in fast or thermal critical reactors
EG12	Limited recycle of U/Pu with new natural-U fuel in fast and/or thermal critical reactor
EG13	Limited recycle of U/Pu with new enriched-U fuel in thermal critical reactor
EG14	Limited recycle of U/Pu with new natural-U fuel in both fast and thermal critical reactors
EG15	Limited recycle of U/Pu with new enriched-U fuel in both fast and thermal critical reactors
EG16	Limited recycle of U/Pu with new enriched-U fuel in thermal critical reactors and fast EDS
EG17	Limited recycle of Pu/Th with new enriched-U/Th fuel in thermal critical reactors
EG18	Limited recycle of $^{233}\text{U}/\text{Th}$ with new enriched-U/Th fuel in thermal critical reactors
<i>Continuous Recycle</i>	
EG19	Continuous recycle of U/Pu with new natural-U fuel in thermal critical reactors
EG20	Continuous recycle of U/TRU with new natural-U fuel in thermal critical reactors
EG21	Continuous recycle of U/Pu with new enriched-U fuel in thermal critical reactor
EG22	Continuous recycle of U/TRU with new enriched-U fuel in thermal critical reactors
EG23	Continuous recycle of U/Pu with new natural-U fuel in fast critical reactors
EG24	Continuous recycle of U/TRU with new natural-U fuel in fast critical reactor
EG25	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new enriched-U/Th fuel in thermal critical reactors
EG26	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in thermal critical reactors
EG27	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new enriched-U/Th fuel in fast critical reactors
EG28	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in fast critical reactors
EG29	Continuous recycle of U/Pu with new natural-U fuel in both fast and thermal critical reactors
EG30	Continuous recycle of U/TRU with new natural-U fuel in both fast and thermal critical reactors
EG31	Continuous recycle of U/Pu with new enriched-U fuel in both fast and thermal critical reactors
EG32	Continuous recycle of U/TRU with new enriched-U fuel in both fast and thermal critical reactors
EG33	Continuous recycle of U/Pu with new natural-U fuel in both fast EDS and thermal critical reactors
EG34	Continuous recycle of U/TRU with new natural-U fuel in both fast EDS and thermal critical reactors
EG35	Continuous recycle of U/Pu with new enriched-U fuel in both thermal critical reactors and fast EDS
EG36	Continuous recycle of U/TRU with new enriched-U fuel in both thermal critical reactors and fast EDS
EG37	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new enriched-U/Th fuel in both fast and thermal critical reactors
EG38	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in both fast and thermal critical reactors
EG39	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new enriched-U fuel in both thermal critical reactors and fast EDS
EG40	Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in fast EDS and thermal critical reactors

Note: EDS = externally-driven systems (subcritical reactors), and $^{233}\text{U}/\text{Th}$ indicates recycle of uranium that is predominantly ^{233}U with thorium.

- Scenario 9—de-emphasize the importance of differences between Evaluation Groups on the Nuclear Waste Management criterion—to explore, in combination with Scenarios 1, 7 and 8, any potential overlap between the Nuclear Waste Management and the Resource Utilization criteria and the potential impact on the choice of fuel cycle.
- Scenario 10—emphasize the importance of differences between Evaluation Groups on the Nuclear Waste Management and Resource Utilization criteria—to focus on long-term and large-scale sustainability issues and the potential impact of the choice of fuel cycle.
- Scenario 11—emphasize the importance of differences between Evaluation Groups on the Nuclear Waste Management and Safety criteria—to explore a perspective reflecting the most prominent current concerns and the potential impact of the choice of fuel cycle.

The same process was used to identify the potentially promising Evaluation Groups for each scenario and for all scenarios. The performance improvement of the Evaluation Groups provided information on the benefit potential for each scenario. As with the criteria, an additional step related the potential improvement in performance for the benefit criteria with the challenge of developing and deploying the fuel cycle, as represented by the Development and Deployment Risk criterion. The sensitivity of the

identification of the promising Evaluation Groups to the range of value judgments was explored by subsequent parametric variation of the functions and tradeoff factors, ultimately resulting in the sets of promising Evaluation Groups (Wigeland et al., 2014).

Key results on relative fuel cycle performance

The E&S Study final report (Wigeland et al., 2014) provides performance data for the 40 Evaluation Groups of fuel cycles on the 25 Evaluation Metrics, the nine Evaluation Criteria, and for 11 multiple-criteria Scenarios, assigning varying degrees of relative importance to the changes possible for each criterion to explore a potential range of policy guidance. The Evaluation Groups were identified where improvement with respect to the current U.S. fuel cycle is possible. Recognizing that what constitutes a “substantial improvement” as stated in the Study Charter is a judgment that may vary considerably among decision-makers, the report identifies multiple sets of potentially promising Evaluation Groups based on the amount of improvement. Using this approach, decision-makers can find the appropriate promising Evaluation Groups corresponding to their view of what constitutes a substantial level of improvement. The report also provides the corresponding development and deployment challenges for all Evaluation Groups, allowing decision-makers to consider both the potential benefits and the associated challenges. The report then identifies the associated R&D that would be required to develop the fuel cycles in the promising Evaluation Groups.

For the six benefit criteria from a fuel cycle alternative to the current U.S. fuel cycle, the alternative fuel cycles were evaluated by comparing them to the current U.S. fuel cycle assuming successful implementation of all disposal paths, and identified several promising Evaluation Groups that have the potential for improved performance relative to the current U.S. fuel cycle for the six benefit criteria as follows:

- *Nuclear Waste Management Criterion:* On a per unit energy generated basis, reduction in generation of fuel cycle waste materials requiring geologic disposal by as much as a factor of 10 or more, reduction in long-term activity corresponding to a reduction in long-term radiation hazard by as much as a factor of 10 or more, and reduction in uranium (depleted from the enrichment process or recovered from reprocessing) and/or thorium (recovered from reprocessing) disposal needs by a factor of 100 or more, and without a large increase in low-level waste generation (up to about 50% higher).
- *Resource Utilization Criterion:* On a per unit energy basis, reduction in the amount of fuel resources needed by a factor of 100 or more.
- *Environmental Impact Criterion:* On a per unit energy basis, reduction in the amount of land required and in the amount of CO₂ emitted (always much lower for nuclear power than for fossil-based generation) by about a factor of 2.
- *Proliferation Risk Criterion:* All of the Evaluation Groups were evaluated as capable of being comparable to the current U.S. fuel cycle at the physics-based functional level as far as material attractiveness is concerned.
- *Nuclear Material Security Risk Criterion:* All of the Evaluation Groups were assessed as comparable to the current U.S. fuel cycle at the physics-based functional level as far as material attractiveness for usefulness in improvised nuclear devices (INDs) is concerned. All Evaluation Groups also contain highly radioactive spent fuel and/or HLW, providing targets with activity comparable to the current U.S. fuel cycle in usefulness for radiation dispersal devices (RDDs)/radiation exposure devices (REDs).
- *Safety Criterion:* It was concluded that it would be possible to safely deploy at least one of the fuel cycles in each Evaluation Group. It was also concluded that the ability to provide enhanced safety compared to the current U.S. facilities was not affected by the choice of fuel cycle but depended on decisions that would be made for implementing the fuel cycle, such as technology choices and facility design decisions.

For the three Evaluation Criteria related to the challenge of developing and deploying a fuel cycle alternative to the current U.S. fuel cycle, the following summarizes the challenges identified by the Study:

- *Development and Deployment Risk:* Alternatives to the current U.S. fuel cycle in the promising Evaluation Groups require R&D to bring the enabling technologies up to the level of successful engineering demonstration, including pilot-scale facilities, estimated as requiring several billion dollars over 10–25 years. Similarly, further development up to the first-of-a-kind commercial facilities were estimated to require an additional several billion dollars. Any transition to a new fuel cycle would take decades to achieve, although some fuel cycle performance benefits such as wastes destined for deep geologic disposal could accrue more quickly. Fully deploying an alternative fuel cycle would likely require several hundred billion dollars or more, comparable to the cost of continuing with the current U.S. fuel cycle as new reactors replace existing reactors.
- *Institutional Issues:* Any of the alternative fuel cycles in the promising Evaluation Groups faces several institutional issues, including lack of supporting infrastructure, lack of regulations and licensing experience, and market barriers to commercial implementation.
- *Financial Risk and Economics:* Estimates of the electricity production cost for fuel cycles in the most promising Evaluation Groups and many of the promising Evaluation Groups are similar to, or close to, those for continuing with the current U.S. fuel cycle.

Promising fuel cycles for performance improvement

Promising fuel cycle options were defined as those that offer the potential for significant improvement over the currently deployed fuel cycle in the United States. Different fuel cycle options might be considered promising by decision-makers or stakeholders who

have different priorities or values. A wide range of perspectives was used to represent this variability in decision-maker preferences, and the sensitivity analyses for each scenario identified Evaluation Groups that are robust to different perspectives, i.e., those that exceed the promise threshold for many of the perspectives considered. The promising options were organized into three sets: most promising options, additional potentially promising options, and other potentially promising options.

An example of the results is shown in Fig. 2 for the scenario where the four benefit criteria of Nuclear Waste Management, Resource Utilization, Environmental Impact, and Safety were given equal criteria tradeoff factors. The potential for improved performance is represented by a non-dimensional benefit utility on the y-axis, and the relative challenge of development and deployment of fuel cycles providing this improved performance is represented by a non-dimensional utility on the x-axis. As the arrows on Fig. 2 indicate, higher utility on the y-axis indicates higher benefit, while lower utility on the x-axis indicates greater challenge. The current U.S. fuel cycle is plotted with a red symbol on the right of the figure, with benefit utility slightly greater than 0.5 and challenge utility of 1.0 (no challenge for development since this fuel cycle is already implemented). The two orange lines indicate thresholds of performance improvement that might be considered as providing substantial, or significant, improvement by decision makers. The figure shows 13 Evaluation Groups above the higher orange line, and four more Evaluation Groups above the lower orange line, showing varying levels of potential improvement and varying degrees of challenge in developing and deploying fuel cycles in these groups. Complete details for the other Scenarios, as well as the sensitivity analyses demonstrating the robustness of the identification of the promising fuel cycle options are provided in the E&S Study final report (Wigeland et al., 2014).

The most promising fuel cycles

Four groups of fuel cycles, described in Table 3, consistently provided the highest improvements compared to the current fuel cycle in the U.S., regardless of the perspective on the relative importance of the benefit criteria (Wigeland et al., 2014; Wigeland, 2015). The descriptions in Table 3 are only indicative of the fuel cycles included in each group, and the term “new natural uranium fuel” indicates that enrichment is not needed for either new uranium fuel resources or for the recycled uranium, in the equilibrium state.

Well-implemented fuel cycles in all of these groups were evaluated as being capable of providing substantial improvements compared to the current U.S. once-through fuel cycle using enriched uranium in light water reactors (LWRs) by reducing the amount

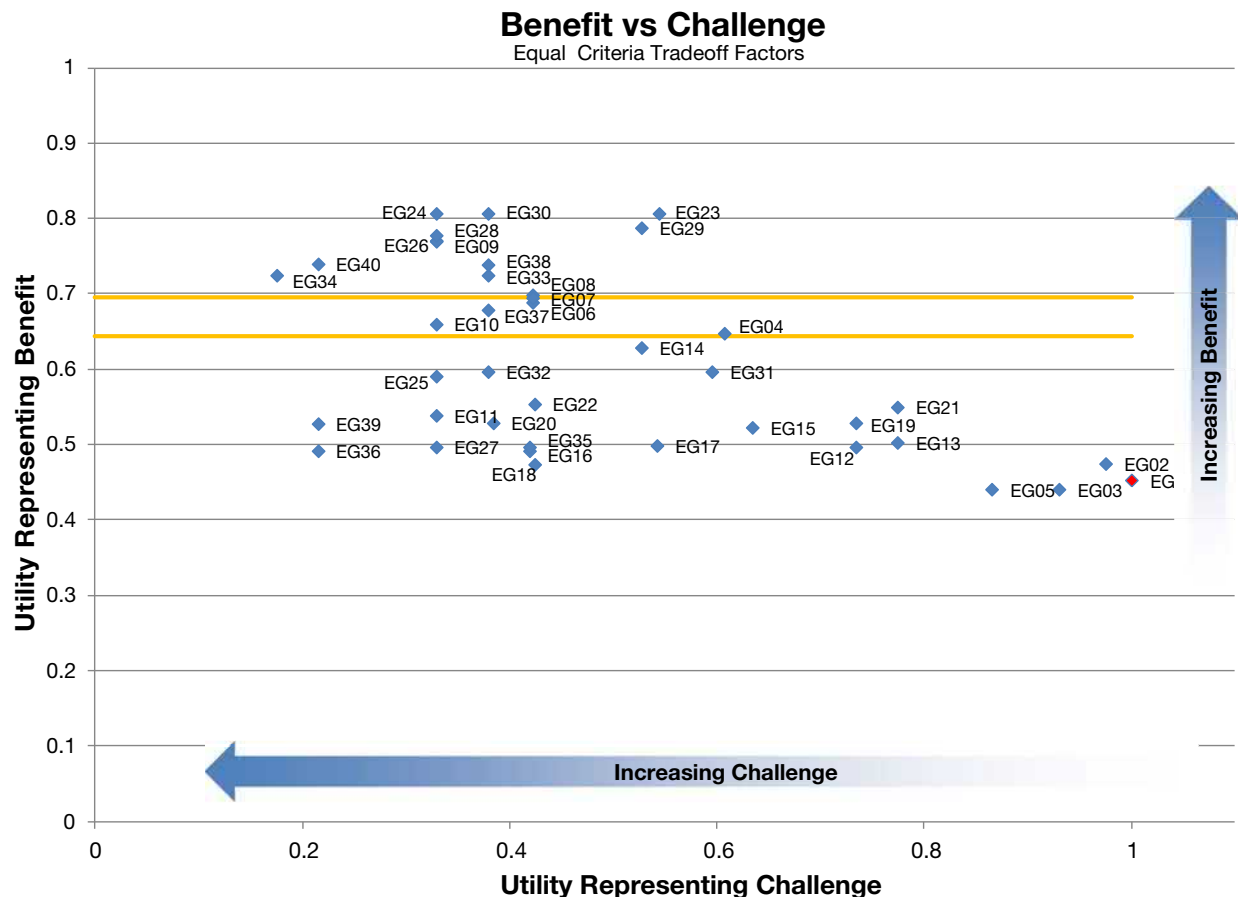


Fig. 2 Benefit versus challenge for the 40 Evaluation Groups for Scenario 1.

Table 3 Most Promising Fuel Cycle Groups.

<i>Evaluation Group</i>	<i>Most Promising Fuel Cycle Groups</i>
EG23	Continuous recycle of U/Pu with new natural-U fuel in fast critical reactors
EG24	Continuous recycle of U/TRU with new natural-U fuel in fast critical reactors
EG29	Continuous recycle of U/Pu with new natural-U fuel in both fast and thermal critical reactors
EG30	Continuous recycle of U/TRU with new natural-U fuel in both fast and thermal critical reactors

Note: U = uranium; Pu = plutonium; TRU = transuranic elements, i.e., atomic number higher than uranium (neptunium, plutonium, americium, curium, etc.); the term "U/Pu" indicates that uranium and Pu are recycled together, similarly the term "U/TRU" indicates that uranium and TRU are recycled together.

of waste generation and increasing the efficiency of using uranium fuel resources that would result if these fuel cycles were used, as follows (US Department of Energy, 1980):

- Greater than a factor of 10 reduction in the hazardous waste mass requiring geologic disposal
- Greater than a factor of 1000 reduction in uranium or thorium disposal
- Greater than a factor of 100 improvement in resource utilization

Part of the considerations in the E&S study that characterize implementing a fuel cycle "well" are developing and choosing technologies that allow safety requirements to be met, waste generation is minimized, safeguards and physical security can be implemented efficiently and effectively, and land and water resources are used efficiently (US Department of Energy, 1980). The fuel cycle characteristics that make these performance improvements possible are:

- Continuous recycle of actinides (U/Pu or U/TRU)
- Fast neutron-spectrum critical reactors with high internal conversion of fertile to fissile material
- Uranium-based fuel cycle with no uranium enrichment required once the fuel cycle is established

Table 4 summarizes the metric data from the E&S study for the Nuclear Waste Management and Resource Utilization metrics for the current U.S. fuel cycle (represented by EG01), and these four most promising fuel cycle groups, where metric data are given by a range of values rather than a single value to reflect the group nature of each fuel cycle group, and the metric data is intended to indicate the best performance potential by fuel cycles in each group. The E&S study results for the metric data show that these four groups of fuel cycles had the same performance improvement when compared to the current once-through U.S. fuel cycle using enriched uranium fuel in light-water reactors. The two metrics where EG29 has metric data in a different bin was caused by the much larger percentage of thermal reactors in the Analysis Example for this case (39% of total power generation) as compared to EG30 (13% of total power generation). It was recognized that if the ratio of thermal to fast reactors was the same in EG29 as it was in EG30, the metric data for EG29 would have been in the same bins as for EG30, prompting identification of EG29 as a most promising option (see footnotes for **Table 4**).

The E&S study also considered the challenge metrics, with the data for the most promising fuel cycle groups summarized in **Table 5** as additional background information for this study. For the metrics where the challenges were estimated as being the same for U/Pu and U/TRU recycle, all of these fuel cycles would face similar challenges, e.g., the time to develop the technologies is in the range of 10–25 years. However, in the E&S study, differences in challenge were noted for the development cost for U/Pu recycle which was estimated to be lower than that for U/TRU recycle, the deployment cost from prototypic validation

Table 4 Benefit Criteria Metric Results from the E&S Study for the Most Promising Options.

<i>Metric</i>	<i>EG01</i>	<i>EG23 (U/Pu)</i>	<i>EG24 (U/TRU)</i>	<i>EG29 (U/Pu)</i>	<i>EG30 (U/TRU)</i>
Mass of SNF + HLW disposed per energy generated, t/GWe-yr	12 to <36	<1.65	<1.65	<1.65	<1.65
Activity of SNF + HLW (@100 years) per energy generated, MCi/GWe-yr	1.05 to <1.60	0.67 to <1.05 ^a	0.67 to <1.05 ^a	1.05 to <1.60 ^a	0.67 to <1.05 ^a
Activity of SNF + HLW (@100,000 years) per energy generated, kCi/GWe-yr	1.0×10^{-3} to < 2.3×10^{-3}	5.0×10^{-4} to < 1.0×10^{-3}	5.0×10^{-4} to < 1.0×10^{-3}	5.0×10^{-4} to < 1.0×10^{-3}	5.0×10^{-4} to < 1.0×10^{-3}
Mass of DU + RU + RTh disposed per energy generated, t/GWe-yr	120 to <200	<1	<1	<1	<1
Volume of LLW per energy generated, m ³ /GWe-yr	252 to <634	252 to <634 ^b	252 to <634 ^b	634 to <1592 ^b	252 to <634 ^b
Natural Uranium required per energy generated, t/GWe-yr	≥145.0	<3.8	<3.8	<3.8	<3.8

^aThe values for the Analysis Examples for each Evaluation Group is as follows: EG23, 1.03; EG24, 1.04; EG29, 1.13; EG30, 0.95.

^bThe values for the Analysis Examples for each Evaluation Group is as follows: EG23, 550; EG24, 560; EG29, 660; EG30, 600.

Table 5 Challenge Criteria Metric Results from the E&S Study for the Most Promising Options.

<i>Metric</i>	<i>EG01</i>	<i>EG23 (U/Pu)</i>	<i>EG24 (U/TRU)</i>	<i>EG29 (U/Pu)</i>	<i>EG30 (U/TRU)</i>
Development Cost, \$B	–	2 to < 10	10 to < 25	2 to < 10	10 to < 25
Development Time, year	–	10 to < 25	10 to < 25	10 to < 25	10 to < 25
Deployment Cost from Prototypic Validation to FOAK					
Commercial, \$B	–	10 to < 25	25 to < 50	25 to < 50	25 to < 50
Compatibility with Existing Infrastructure	–	< 10%	< 10%	10% to < 50%	10% to < 50%
Existence of Regulations for the Fuel Cycle and Familiarity with Licensing ^a	–	No U.S.	None	No U.S.	None
Existence of Market Incentives and/or Barriers to Commercial Implementation of Fuel Cycle Processes	–	<i>b</i>	<i>b</i>	<i>b</i>	<i>b</i>
Levelized Cost at Equilibrium (LCAE)	–	<i>c</i>	<i>c</i>	<i>c</i>	<i>c</i>

^aMetric data for the overall fuel cycle indicate the lowest value for any facility. These cases mainly show the difference between reprocessing and use of U/Pu as compared to U/TRU, where there is no experience with U/TRU regulation.

^bMarkets and market mechanisms are weak or exhibit distortions, requiring Federal government intervention in the form of direct investment or changes in law in order to establish and sustain market drivers. In addition, the fuel cycle option exhibits challenges with respect to the magnitude of capital required and payback prospects.

^cLikely to be “Similar” to the Current U.S. Fuel Cycle – The LCAE for the Evaluation Group is likely to be within about 30% of the LCAE for EG01. Placement in this bin requires the Analysis Example to have a mean LCAE that is no larger than 30% greater than EG01 and the uncertainty difference includes (or is close to including) the mean LCAE for EG01.

to first-of-a-kind (FOAK) commercial which was estimated to be lower for U/Pu recycle in fast reactors only, and the lack of any regulatory experience with facilities supporting U/TRU recycle.

These groups are the most promising options if the amount of reduction provided by these fuel cycles in the amount of waste generated and fuel resources needed is considered to be both important and substantial (since as noted above, choice of fuel cycle for most of the remaining criteria did not result in any improvement or differentiation), a judgment made by decision-makers and others.

None of these fuel cycles are ready to be deployed today. R&D must be conducted to develop the appropriate implementing technologies. The current state of knowledge and experience was examined to identify the R&D needs for each part of the fuel cycle. The R&D required to support these fuel cycles, along with some requirements for implementing technologies in order to achieve the benefits attributed to the fuel cycles, are as follows:

Reactor development

- R&D on fast critical reactors because these reactors facilitate effective consumption of actinide elements and efficient use of uranium fuel resources (which may also include intermediate spectrum reactors since these were grouped with fast reactors in the Study; fast subcritical reactors are also in this Evaluation Group with similar physics-based performance but face greater challenges than critical reactors, including for LCAE, so the supporting R&D is not listed)
 - The reactors must have conversion of fertile materials to more fissionable materials sufficient to sustain operation without the need for ongoing supplies of enriched uranium

Separations/reprocessing development

- R&D on separation of U/Pu or U/TRU from irradiated fuel to make them available for recycle
 - The separations processes should have total nominal product losses of about 1% or less to waste disposal; smaller losses may not be of further benefit

Fuel development

- R&D on recycle fuel development to facilitate use of recovered U/Pu or U/TRU as fuel
 - The fuel should have irradiation capability comparable to or greater than today’s fuel

In addition, for any fuel cycle, an R&D goal should be to reduce waste generation throughout the fuel cycle, including developing waste forms that reduce the volume of any HLW since HLW volume can be an important factor for deep geologic disposal.

Minor actinide recycle

Both U/Pu and U/TRU recycle fuel cycles could each provide benefits with respect to the other depending on where the minor actinides were in the fuel cycle. For U/Pu recycle fuel cycles, whether only fast reactors are used (EG23) or if fast reactors provide the fissile materials to operate thermal reactors (EG29), the minor actinides are intentionally part of the HLW and would only potentially be present in the rest of the fuel cycle as a contaminant from reprocessing activities. Implementing a fuel cycle that recycles U/Pu allows the use of fuels with a significant experience base, and these fuels could be fabricated and handled with less shielding than is needed when the

minor actinides are in the fuel as is the case with U/TRU recycle. The irradiated U/Pu fuel also has lower decay heat than when U/TRU is recycled, allowing handling and shipping sooner with fewer shipments to processing facilities. Reprocessing technologies for recovering U/Pu are available that are similar to those that have been deployed industrially, potentially reducing R&D needs. However, at the same time, even though the presence of the minor actinides in the HLW is accommodated by existing waste form technologies, the minor actinides could complicate advanced waste form development and affect the durability of an advanced waste form due to the higher α radiation, potentially increasing R&D needs in this area. In addition, the HLW minor actinide content, especially Am241, adds decay heat that needs to be managed during handling and storage, and which may increase the size of a geologic repository due to the potential for requiring an increased area for disposal of HLW with higher decay heat.

In contrast, recycle of the minor actinides in U/TRU recycle fuel cycles, whether only fast reactors are used (EG24) or if fast reactors provide the fissile materials to operate thermal reactors (EG30), keeps the minor actinides out of the HLW except for processing losses. This would allow the use of existing waste form technologies and has the potential to simplify advanced waste form development. The lower HLW decay heat may enable more compact disposal, reducing the size requirements for a geologic repository. However, these benefits also come with the development challenges associated with keeping the minor actinides within the fuel cycle. As mentioned above, U/TRU recycle fuels will require more shielding for handling and fuel fabrication, there is little to no experience with such U/TRU recycle fuels in a reactor, and the handling and storage of the irradiated fuel must deal with the higher decay heat. Reprocessing technology needs to be developed, and R&D has been ongoing to achieve U/TRU recovery.

It is also possible to consider recycle of just one or more of the minor actinides rather than all of them together. For example, if americium is recycled along with the U/Pu, the decay heat from the americium is kept out of the HLW, potentially benefitting the space required for waste disposal, but there may still be complications for advanced waste form development due to α damage, mainly from curium. At the same time, the presence of americium in the recycle fuel increases the decay heat, affecting storage and handling, and there is little to no experience with such fuels although challenges related to the recycle of americium have been identified in other studies. This example illustrates that such partial recycle of the minor actinides may move specific benefits and challenges between the different parts of the fuel cycle but they will always be present.

Either U/Pu and U/TRU recycle fuel cycles has both benefits and challenges. It appears that the benefits are associated with the parts of the fuel cycle that do not contain significant amounts of minor actinides, and the R&D challenges are associated with the parts of the fuel cycle where the minor actinides are present.

Additional potentially promising fuel cycles

Recognizing that organizing the promising Evaluation Groups into sets of similar potential benefit is somewhat arbitrary, the thresholds on Fig. 2 (Scenario 1, equal criteria tradeoff factors) and sensitivity analyses were used as guides to identify 11 additional *potentially promising groups of fuel cycles* that provide somewhat lower, but still potentially substantial, beneficial improvements than the four discussed above. While it is again a matter of judgment by decision-makers and others whether the improvements offered by these groups are considered both important and substantial, each of these groups perform better than the current U.S. fuel cycle when almost any, but not all, combinations of criteria are considered. (Evaluation Groups are listed in numerical order, with a short description indicative of the fuel cycles included in each Evaluation Group):

- EG06—Once-through using Th fuel to very high burnup in thermal EDS
- EG07—Once-through using natural-U fuel to very high burnup in thermal or fast EDS
- EG08—Once-through using Th fuel to very high burnup in fast EDS
- EG09—Limited recycle of U/TRU with new natural-U fuel to very high burnup in fast critical reactors
- EG26—Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in thermal critical reactors
- EG28—Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in fast critical reactors
- EG33—Continuous recycle of U/Pu with new natural-U fuel in both fast EDS and thermal critical reactors
- EG34—Continuous recycle of U/TRU with new natural-U fuel in both fast EDS and thermal critical reactors
- EG37—Continuous recycle of $^{233}\text{U}/\text{Th}$ with new enriched U/Th fuel in both fast and thermal critical reactors
- EG38—Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in both fast and thermal critical reactors
- EG40—Continuous recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in fast EDS and thermal critical reactors

While the R&D listed for the four most promising options would also support some of these fuel cycles, other, different or additional, R&D is needed to support development of some of these additional potentially promising options, as follows:

Reactor development

- R&D on thermal critical neutron reactors because these reactors facilitate efficient conversion of thorium to usable fuel
 - The reactors must have conversion of fertile thorium materials to more fissionable materials sufficient to sustain operation without the need for ongoing supplies of fissile materials such as enriched uranium
- R&D on externally-driven systems (EDS) because subcritical reactors can provide an external source of neutrons to facilitate conversion of fertile materials to more fissionable materials (although the greater challenges for such systems, including LCAE, should be recognized)

- The reactors must have conversion of fertile thorium and/or uranium materials to more fissionable materials sufficient to support operation without the need for ongoing supplies of fissile materials such as enriched uranium
- The additional safety challenges identified for EDS compared to critical reactors should be addressed

Separations/reprocessing development

- R&D on separation of $^{233}\text{U}/\text{Th}$ from irradiated fuel to make it available for recycle
 - The separations processes should have total nominal product losses of 1% or less to waste disposal; larger losses may be acceptable while still preserving sufficient benefits, but smaller losses may not be of further benefit

Fuel development

- R&D on recycle fuel development to facilitate use of separated $^{233}\text{U}/\text{Th}$ as fuel
 - The fuel should have irradiation capability comparable to or greater than today's fuel
- R&D on very high burnup fuel to facilitate greater resource utilization
 - The fuel should have irradiation capability several times higher than today's fuel and even well beyond experimentally-demonstrated capability for advanced fuel

Other potentially promising fuel cycles

In addition to the Evaluation Groups listed above, using the lower threshold in [Fig. 2](#) and the sensitivity analyses described in Appendix F of [Wigeland et al. \(2014\)](#), a few additional lesser performing fuel cycles were identified that may be potentially promising depending on the relative importance of the criteria and the underlying metrics, again if the improvements are considered both important and substantial by decision-makers and others (Evaluation Groups are listed in numerical order, with a short description indicative of the fuel cycles included in each Evaluation Group):

- EG04—Once-through using natural-U fuel to very high burnup in fast critical reactors
- EG10—Limited recycle of $^{233}\text{U}/\text{Th}$ with new Th fuel in fast and/or thermal critical reactors
- EG14—Limited recycle of U/Pu with new natural-U fuel in both fast and thermal critical reactors

The R&D requirements already listed above are sufficient to support development of these fuel cycles.

Insights about fuel cycles

Based on the promising fuel cycles identified by the evaluation and screening process, certain fuel cycle characteristics were identified that provide beneficial performance improvements with respect to the Evaluation Metrics, criteria and scenarios:

- Continuous recycle of actinide elements—the actinide elements (thorium, protactinium, uranium, plutonium, neptunium, americium, curium, and so on) are major contributors to the long-lived hazard from irradiation of nuclear fuel, and can be a source of energy, some directly as fuel and others by conversion to usable fuel. Recycling the actinide elements benefits two of the Evaluation Criteria: efficient use of fuel resources and reduction of nuclear waste generation.
- Fast neutron irradiation—fast neutron fission has a much more favorable fission-to-capture ratio for neutrons for certain isotopes, greatly increasing fissioning of isotopes such as ^{240}Pu and enhancing fissioning of ^{239}Pu , reducing the buildup of long-lived highly radioactive higher actinide isotopes.
- Critical reactors—use of reactors that are capable of sustaining fission without the need for an external source of neutrons lowers development risk, lowers safety challenges, and lowers overall costs as compared to externally-driven systems.
- High-internal conversion—efficient conversion of fertile fuel materials to more easily fissionable isotopes allows efficient use of fuel without the need for uranium enrichment for continued operation, increasing resource utilization and reducing waste generation. Fast neutron fission also produces more neutrons per fission than thermal fission, facilitating the high-internal conversion needed for self-sustaining fuel cycles.
- Nuclear fuels—irradiating uranium-based fuels in the fast spectrum provides higher internal conversion capability than thorium-based fuels in either a thermal or fast spectrum, facilitating effective resource utilization as long as uranium enrichment is not required for continued operation, as evidenced by the four most promising options identified in the Study. However, even though uranium may be more readily used to achieve greater resource utilization, potentially promising options were also identified for fuel cycle options using thorium-based, and uranium/thorium-based fuels, as listed above.
- Safety—promising fuel cycles are capable of safe deployment, with many having safety challenges comparable to the current U.S. fuel cycle. Enhanced safety is not provided by choice of fuel cycle, but may be provided by the choice of implementing technologies and facility design

In addition to these specific fuel cycle characteristics, other more general concepts applicable to many or all fuel cycles were examined for potential benefit:

- Extended decay storage (SNF and/or UNF, (*Note:* SNF denotes spent nuclear fuel, which is irradiated fuel destined for disposal. UNF, used nuclear fuel, is used to denote irradiated fuel that is going to be reprocessed.) products, or wastes) can:
 - slowly lower radiation level by radioactive decay to potentially reduce worker exposure or shielding requirements, but the remaining radiation is sufficient to still require remote handling of the materials
 - favorably affect recycle of some actinide elements such as curium, but may adversely affect recycle of other actinide elements such as plutonium
 - slowly lower decay heat at the time of disposal for SNF, facilitating handling and emplacement, but is most effective for the HLW from recycle fuel cycles where most of the content is fission products with a relatively short radioactive half-life.
- Processing of spent fuel prior to disposal, not for recycle (for once-through or limited recycle fuel cycles only) to separate the uranium or thorium from the fission products and any long-lived highly-radioactive elements may also greatly reduce the amount of materials requiring isolation such as that provided by deep geologic disposal. However, the uranium and/or thorium separated and recovered during processing still need to be disposed as waste, in contrast to recycle fuel cycles where such materials can be reused to reduce overall waste generation from the fuel cycle and increase utilization of fuel resources.
 - If the separated uranium or thorium can be disposed with much lower isolation requirements, then processing of SNF prior to disposal can reduce the amount of waste requiring geologic isolation, e.g., HLW, by a factor of 10 or more.
 - If the disposal requirements for the recovered uranium and/or thorium are comparable to those for SNF or HLW, then there would appear to be no benefit from processing SNF prior to disposal.
- Minor actinide separation and transmutation
 - While minor actinide separation and recycle in addition to uranium/plutonium recycle (i.e., TRU recycle) may further reduce waste generation and increase resource utilization as compared to uranium/plutonium recycle, no improvement was noted for the metrics used in this Study for the most promising options of continuous recycle in fast reactors. Additionally, at the fuel cycle level, it was noted that there would be no difference in potential benefit for TRU recycle whether all of the minor actinide elements are recycled individually in different fuels, as a single group in one fuel, or in combination with fuel containing plutonium. Thorium-based fuel cycles have little minor actinide content in the irradiated fuel and therefore would not have substantial benefits from minor actinide separation and transmutation.

Challenges for fuel cycle development and deployment

The criteria indicating the challenges associated with developing and deploying an alternative fuel cycle identified several commonalities among essentially all of the promising fuel cycle options:

- Two of the most promising fuel cycles have estimated total development costs in the range of \$2 - \$10 billion (EG23, EG29), while the other two (EG24, EG30) are in the range of \$10–\$25 billion, as are most of the other promising fuel cycles, and estimated development times in the range of 10–25 years to bring all enabling implementing technologies and facilities to successful demonstration at engineering scale. The government has historically been the major source of funding for such R&D activities.
- Following completion of the technology development, the promising options have an estimated initial total deployment cost in the range of either \$10–\$25 billion (EG06, EG07, EG08, EG23) or \$25–\$50 billion (the remaining promising options) to continue development from engineering demonstration through the deployment of first-of-a-kind commercial facilities. Cost-sharing between industry and government may be expected for such development. Fully deploying an alternative fuel cycle to replace the current U.S. fuel cycle would likely require several hundred billion dollars or more, comparable to the cost of continuing with the current fuel cycle replacing existing reactors as they are retired with new similar reactors.
- The market disincentives and barriers to commercial implementation of nearly all of the promising options are expected to be very significant, such that in the U.S., Federal government intervention in the form of direct investment, mandates, or changes in law in order to establish and sustain market drivers will likely be required for full-scale implementation of a new fuel cycle. An example of this is the current SNF disposal fee based on energy production that provides a disincentive for waste reduction because for a given amount of energy production, the disposal fee is the same regardless of waste amount.

Based on the Study results for the estimated levelized cost of electricity at equilibrium (LCAE), many of the promising options may be expected to have electricity production costs that are similar to, or close to, the estimated LCAE for the current U.S. fuel cycle as shown in [Fig. 3](#) where the difference is given in mills per kilowatt-hour ([Wigeland et al., 2014](#)). Further information on the economics of nuclear energy is provided in Chapter 00222 of this encyclopedia ([Rothwell, 2021](#)).

The four most promising options, EG23, EG24, EG29, and EG30, all have a mean estimated LCAE within 5 mills per kilowatt-hour of the cost of continuing with the current U.S. fuel cycle, which has an LCAE of about 50 mills per kilowatt-hour. It was observed that more complex fuel cycles could cost more to build and operate, but could have offsetting lower costs elsewhere in the fuel cycle, and the dominant fuel cycle cost contributor is for the reactors. For example, a recycle fuel cycle adds costs for reprocessing and recycling, but may have lower fuel resource costs and may eliminate enrichment costs.

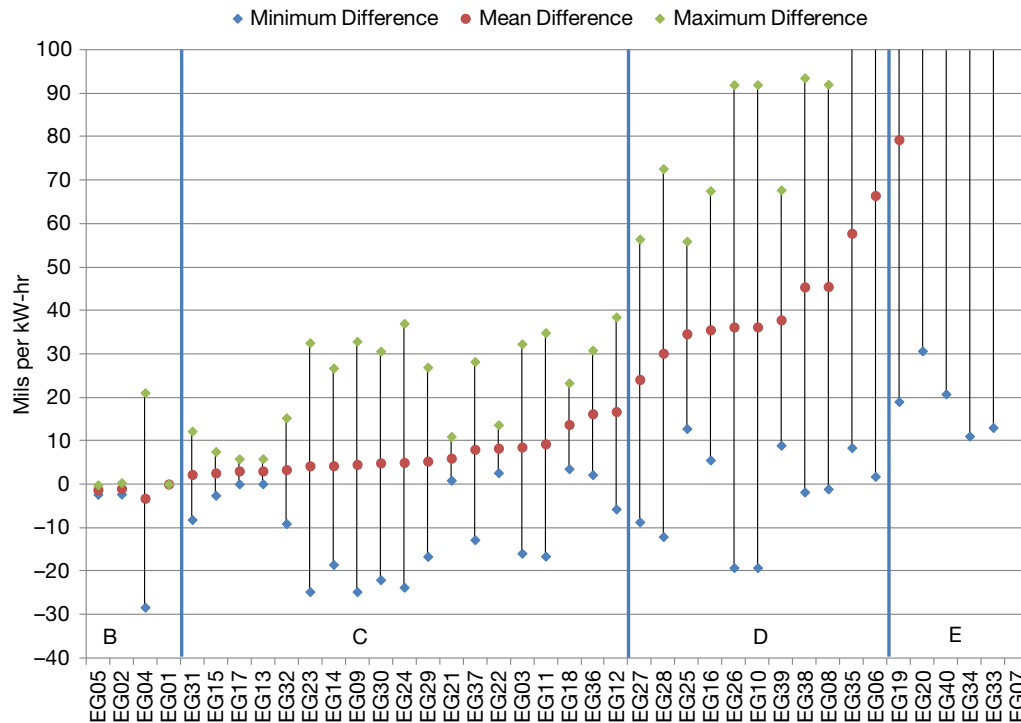


Fig. 3 Difference in LCAE in mils per kilowatt-hour for the analysis example of each Evaluation Group and the analysis example of EG01, ordered by increasing positive difference.

Since the LCAE was not grouped with the other challenge criteria but provided the information separately as additional information, it is instructive to examine the differences in LCAE for the best performing Evaluation Groups. Fig. 4 shows the main cost contributions to the estimated mean LCAE for Analysis Examples EG23, EG24, EG29 and EG30 and compared directly to the cost

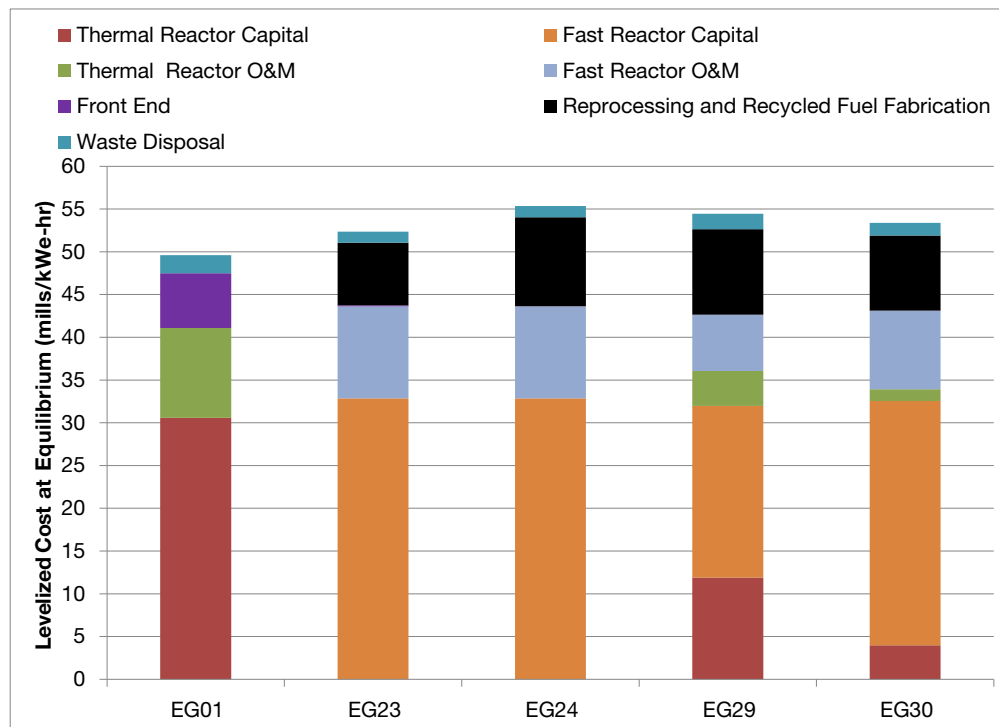


Fig. 4 Comparison of the levelized cost of electricity at equilibrium cost breakdown for EG01, EG23, EG24, EG29 and EG30 using averages of the calculated values.

contributions of EG01. The Analysis Examples EG23 and EG24 are fast reactors operating on a closed U/Pu and U/TRU cycle, respectively, which are self-sufficient due to Pu and TRU production. The Analysis Examples EG29 and EG30 are, respectively, net Pu and TRU producing fast reactors that breed sufficient excess fissile to supply thermal reactors so that uranium enrichment is not needed. While the O&M costs are about the same for all five systems, the charges for capital cost recovery are, as expected, higher for EG23 and EG24 which use only fast reactors. Evaluation Groups EG29 and EG30, which involve a combination of fast and thermal reactors, have capital cost recovery charges that are proportional to the fraction of each reactor type utilized in the system.

See Also: Self-Sustaining Breeding in Advanced Reactors: Characterization of Natural Resources; Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors; Self-Sustaining Breeding in Advanced Reactors: Comparison of Fuel Cycle Performance.

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Self-Sustaining Breeding in Advanced Reactors: Characterization of Natural Resources

Jiri Krepel, Paul Scherrer Institut, Villigen, Switzerland

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Glossary

Ac Actinides, elements with atomic numbers 89 to 103.

Actinides Eigen-composition Stabilized actinides composition, an equilibrium composition which results from a long irradiation of one parent nuclide or their mixture. Necessary condition for the stabilization is that the mass of parent nuclide/s and the irradiation flux do not vary strongly. It is specific and inherent feature for every reactor and every parent nuclide/s and it represents the Eigen-vector of the respective Bateman equation.

B&B Breed-and-Burn; a mode of open cycle operation, where the fissile excess bred during irradiation is equal or higher than the discharged fissile mass after the irradiation. The reactor burns its own bred fuel. The discharged fuel does not need to be recycled and the fresh fuel does not need to contain primordial or synthetic fissile nuclides.

Binding energy per nucleon Energy normalized per nucleon, which would be released when nuclei would be synthesized from protons and neutrons or which would be needed to disassemble the nucleus into protons and neutrons.

Breeding Transmutation of actinide nuclide initiated by neutron capture, which increases fission probability. Typically, primordial non-fissile (fertile) nuclides ^{232}Th and ^{238}U are transmuted into synthetic (man-made) fissile ^{233}U and ^{239}Pu .

Burnup Share of already fissioned actinides. It can be expressed in FIMA % (Fissions per Initial Metal Atom) or as energy released from given mass of actinides in MWd/kg (Megawatt days per kilogram). Since the energy per fission is approximately the same for all actinide nuclides, values expressed in these two units are proportional and roughly 10 times higher for the second unit.

Closed cycle A chain of process steps which is closed for the respective working medium. The medium, in nuclear fuel cycle the actinides, is recycled and does not extensively leave the chain.

Cross-section A measure of the probability that neutron will interact when flying by with the respective nucleus (microscopic cross-section) or when fling through with the respective nuclei concentration (macroscopic cross-section).

Decay series Series of consecutive radioactive decays and intermediate decay products, which are needed to reach the final stable nuclide. All actinides are radioactive and decay through one of the four existing actinides alpha decay series.

Fertile nuclide Nuclide which can be transmuted into fissile nuclide.

Fission barrier It represents the minimal added energy by the interacting particle to cause nuclear fission.

FPs Fission Products, typically two fragments of actinides fission.

Irradiation chain Series of consecutive transmutations and of the respective daughter products induced by neutron irradiation of one parent nuclide, e.g., ^{232}Th or ^{238}U . The radioactive decays of the daughter products are part of the chain.

MCFR Molten Chloride Fast Reactor, unmoderated MSR type based on molten chlorides.

MSR Molten Salt Reactor, a reactor in which the molten salt has substantial function, in this chapter the label is used for graphite moderated fluoride salt based MSR.

Neutron economy Balance between neutron generation and neutron losses in a reactor. Good neutron economy has high neutron generation and small neutron losses by parasitic absorption and leakage from the reactor.

Open cycle A chain of process steps which is open for the respective working medium. The medium, in nuclear fuel cycle the actinides, is not recycled and leaves extensively the chain.

Primordial actinides Actinide nuclides, presumably originated by rapid neutron capture process during a supernova explosion, which have long enough half-life for radioactive decay to be still present on the earth.

Self-sustaining breeding Breeding process where breeding rate $\geq$ fission rate, the mass of fissile synthetic nuclides is conserved in the reactor, and fission chain reaction is sustained solely by these synthetic nuclides. A long time-horizon is being considered in this article.

Synthetic actinides Actinide nuclides originated, typically by neutron-induced transmutation, from primordial actinides. Synthetic nuclides are, by definition, a much broader group than minor actinides or trans-uranic elements. For thorium and uranium elements, both primordial and synthetic isotopes exist.

Waste Waste is an unwanted side product, which is further unusable. In the nuclear fuel cycle, synthetic actinides represent a side product, which is reusable. However, they can be declared as a waste by nuclear fuel cycle policy or when it is too laborious to recover them, e.g., from reprocessing losses of certain fuel forms.

Introduction

This is the first of a series of three chapters that is dedicated to the identification of the advanced reactor technologies that are capable of self-sustaining breeding when fueled with either ^{232}Th or ^{238}U , and to the investigation of equilibrium fuel cycles properties and neutronics performance. This first chapter characterizes the available fuel resources from the nuclear and reactor physics perspective. It presents in detail the features of the ^{232}Th and ^{238}U irradiation chains using two advanced reactors adopted from (Losa, 2016; Krepel and Losa, 2019) as an example and the knowledge of equilibrium nuclides concentration and reaction rates is used for characterization of neutron economy and ingestion radiotoxicity related to these chains. The second chapter (Krepel and Losa, 2021) determines the equilibrium fuel composition in advanced reactors and compares the basic features of the equilibrium fuel cycle and selected reactor neutronic performance characteristics at the equilibrium state. The third chapter (Krepel, 2021) evaluates the fuel cycle performance from a neutronics perspective by means of the earlier obtained equilibrium parameters. It provides an additional insight on why some advanced reactor concepts can act as self-sustaining breeders in closed or even open fuel cycle, while others do not.

Actinides as a resource

In terms of producing energy by atomic nuclei transmutation, the nuclear fission chain reaction is the only alternative to nuclear fusion. From the resources available on earth only actinides can be utilized in nuclear fission reactors. They were presumably originated by rapid neutron capture processes in the core of a collapsing star and expelled into space by gravitational shock waves during its supernova explosion. Hence, the nuclear fuel cycle relies on a resource, which is not renewable. Three dominating primordial actinide nuclides are present on earth: ^{235}U , ^{238}U and ^{232}Th . Their estimated reserves somewhat follow their half-lives for radioactive decay (see Fig. 1). Accordingly, there is probably three to four times more thorium than uranium, and the ^{235}U share in natural uranium is limited to 0.7%. The ^{235}U reserves are the smallest; at the same time, it is the only primordial nuclide fissile by thermal neutrons. Therefore, it can be directly utilized in nuclear fission reactors. The other two primordial nuclides can also be utilized, albeit not directly and not in all reactors. Accordingly, they are foreseen as a resource for advanced reactors (Stanculescu, 2021).

Primordial and synthetic actinides

Actinides nuclides can be divided into primordial and synthetic. The three major primordial nuclides have half-lives of the same order of magnitude as the age of earth (see Fig. 1). The synthetic actinides are originated by primordial nuclides transmutation, typically by neutron capture reactions. Fission reactions also represents nuclear transmutation; however, the resulting Fission Products (FPs) do not belong to the chemical group of actinides.

Fission and transmutation

The ^{235}U reserves are the smallest; at the same time, ^{235}U is the only primordial nuclide suitable for establishing a fission chain reaction. Consequently, vast majority of current nuclear power plants rely on enriched uranium and fission predominantly ^{235}U .

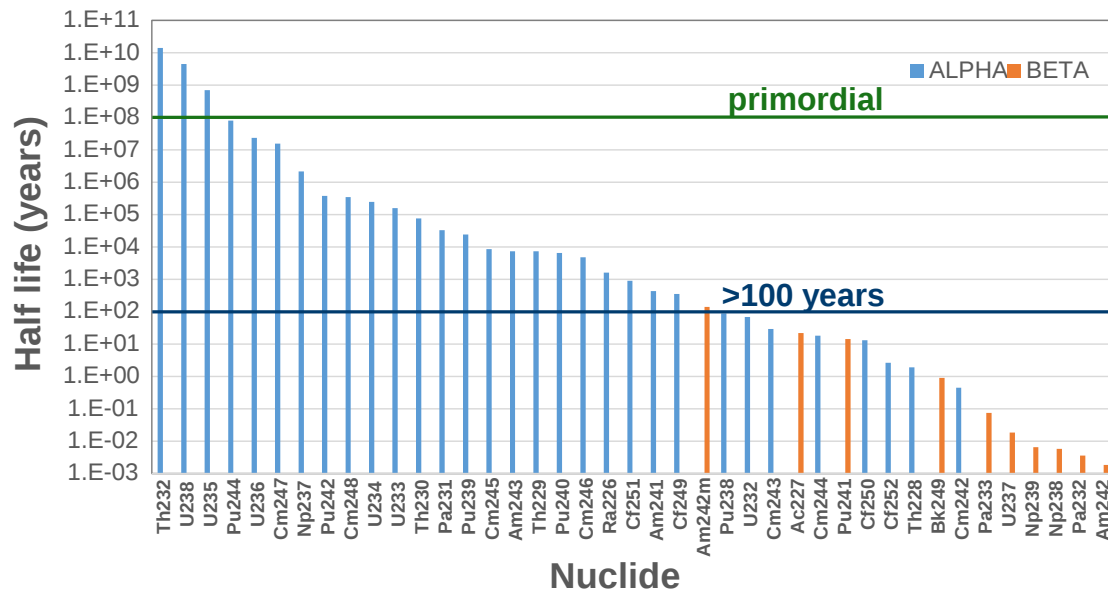


Fig. 1 Half-life and type (alpha or beta) of radioactive decay for primordial and synthetic actinides.

The consumption of natural uranium in these reactors does not exceed 1%. If also the thorium reserves are considered, not more than 0.2% of primordial actinides are utilized for energy production in ^{235}U -fueled reactors. Moreover, the discharged fuel from these reactors contains synthetic actinide nuclides, e.g., ^{234}U , ^{236}U , ^{237}Np , ^{239}Pu , ^{240}Pu , ^{241}Am , originated from ^{235}U and ^{238}U irradiation. Synthetic actinides are the major cause of the long-term stewardship burden related to the final spent fuel repository. Their recycling in ^{235}U -fueled reactors is possible, but only to a limited extent. Since the fuel is usually not recycled, these reactors are operated in so-called open cycle (Crawford, 2021).

Already in 1940 it was discovered that transmutation of fertile ^{238}U results in fissile ^{239}Pu (Seaborg, 1994). Equally, fertile ^{232}Th can be transmuted into fissile ^{233}U . This transmutation process is called breeding and reactors relying on this process are called breeders. Once originated, the synthetic nuclides ^{239}Pu and ^{233}U can sustain the fission chain reaction and the breeding process itself. Accordingly, they practically act as a catalyzer for energy production from the primordial nuclides ^{238}U and ^{232}Th . Even though the first ^{239}Pu was generated by a different (particle accelerator based) nuclear process, the fission chain reaction driven by ^{235}U is the most natural option to generate the initial synthetic nuclides and to launch the self-sustaining breeding. However, not all advanced reactors can sustain this process, and some perform better than others. The aim of this article is to provide insight from a nuclear and reactor physics perspective, as to why it is so.

Natural resources utilization

The amount of primordial actinides on earth is finite and not renewable. Accordingly, the sustainability of the nuclear fuel cycle is determined by its capability to maximize its natural resources utilization. As already mentioned, utilization of natural uranium does not exceed 1% in ^{235}U -based open cycle and it can reach up to 20–35% in the breed-and-burn (B&B) open cycle mode (Qvist, 2021; Krepel and Kramer, 2021; Greenspan, 2021). Nonetheless, the highest resources utilization can be achieved by breeding in a closed cycle (Wigeland, 2021; Krepel et al., 2018). The cycle is called closed, because actinides as the working medium are recycled and do not significantly leave the cycle. Achievable utilization in this fuel cycle is limited solely by the recycling losses (losses in fuel reprocessing and refabrication) and fuel burnup per cycle (see Fig. 2). It can reach up to 95–98% of the resources. Discharged fuel burnup influences the losses, because fuel with higher burnup is less often recycled. In an ideal reactor, the discharged fuel burnup should be 100% FIMA and reprocessing frequency zero. Should this be possible, breeding in closed cycle and in the B&B open cycle will become identical. Even though unrealistic, it may be approached in reactors with liquid fuel from which solely FPs are extracted from the core.

The reason why not all reactor concepts are capable of being self-sustaining (isobreeding/break-even) breeders is the tight neutron economy. Simple fuel balance of a breeder reactor requires that for each fissioned synthetic nucleus at least one transmutation of primordial nuclide takes place. Accordingly, up to two neutrons are needed from each fission: one for sustaining the chain reaction and one for maintaining of the fissile material balance. The neutron economy of a breeder is thus very tight. Less than one neutron per fission should be lost due to the neutron leakage or parasitic neutron capture in the coolant and structural materials or in the fuel itself. Furthermore, once discharged from a breeder, the irradiated fuel must be recycled to maintain the fissile material balance. The only exception from this requirement are very strong breeders, where the fissile excess bred during the irradiation would be equal or higher than the discharged fissile mass after the irradiation. Since the bred excess is not necessary for the fissile

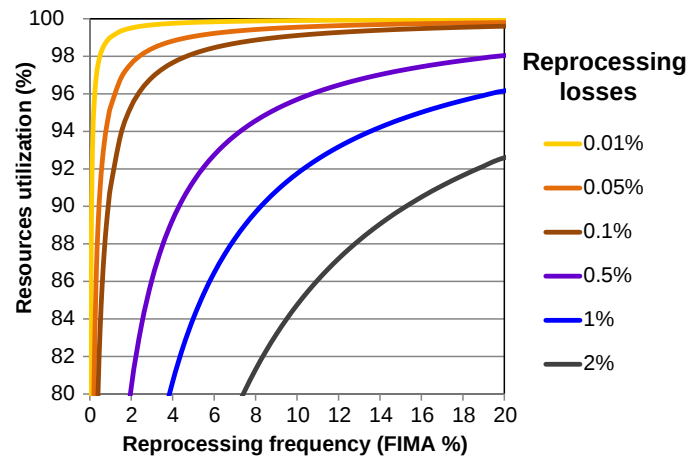


Fig. 2 Natural resources utilization as a function of reprocessing frequency (inversely proportional to fuel burnup per cycle) and losses in recycling. From Krepel J, Hombourger B and Losa E (2018) Fuel cycle sustainability of Molten Salt Reactor concepts in comparison with other selected reactors. *PHYTRA4, Marrakech, Morocco, September 17-19*.

mass conservation, these very strong breeders can self-sustain the breeding in open cycle in the B&B mode (Qvist, 2021). The name is derived from the fact that the loaded fuel does not need to contain fissile actinides; they are bred during the irradiation in the B&B reactor and suitable fraction of them fission in that reactor.

Catalyzer and waste, simultaneously

Synthetic actinide nuclides are the necessary catalyzer for the primordial actinides utilization. However, in the open fuel cycle practiced today, or as losses of recycling in a closed cycle, they cause a long-term stewardship burden. Their half-lives are too short to be primordial, but too long to disappear swiftly once originated (see Fig. 1). Furthermore, synthetic actinides decay in chains and one fast decaying nuclide can be a source of another slow decaying nuclide and vice-versa. The social and environmental sustainability considerations require minimal long-term stewardship burden; hence, the amount of synthetic actinides in the waste stream should be minimized. The burden should be measured in terms of their long-term radiotoxicity per unit of produced energy. Indefinite recycling in closed cycle provides the ultimate solution not only for maximizing resources utilization, but also for minimizing waste production. The recycling losses define the achievable limit for both measures. In the open cycle, all discharged actinides are considered as a waste and the policy of many countries is to seal them forever in a spent fuel repository without recycling. However, synthetic actinides are in reality a resource and they are treated as such in this article.

Irradiation chains of primordial actinides

Since any synthetic actinide, as a transmutation product, can undergo another transmutation, the irradiation of primordial actinides by neutrons results in a chain of synthetic actinides. The irradiation chains of ^{232}Th and ^{238}U are especially relevant for natural resources utilization.

Irradiation chains

For each reactor and for each primordial nuclide, the composition of generated synthetic actinides tends to stabilize and reaches equilibrium concentration after sufficiently long neutron irradiation. Necessary condition for the composition to converge to its equilibrium state is that the mass of irradiated nuclide and the neutron flux do not vary strongly. Hence, when the same fuel is repetitively irradiated in closed cycle, soon or later the equilibrium composition of synthetic actinides will be established. This stabilized actinides composition is actually the Eigen-composition of the respective Bateman equation (Bateman, 1910) and it is specific and inherent feature of given reactor and parent primordial nuclide. For the illustrative purposes, the ^{232}Th and ^{238}U irradiation chains in thermal and fast neutron spectra are presented in Figs. 3–6. The respective reactors used for the simulation are graphite moderated Molten Salt Reactor (MSR) with fuel in form of eutectic mixture of lithium fluoride and actinides tetra-fluoride salts and Molten Chloride Fast Reactor (MCFR) with fuel in form of eutectic mixture of sodium chloride and actinides tetra-chloride salts. For more detail about the selected reactors and modeling assumptions refer to (Krepel and Losa, 2021).

Even in an open cycle, the fuel composition tends to converge to equilibrium. The ^{238}U mass does not vary strongly in the ^{235}U -fueled reactors. Already at the end of an open-cycle irradiation, the immediate daughter product ^{239}Pu is practically at equilibrium concentration and its daughter ^{240}Pu close to it. The concentrations of higher plutonium isotopes are not yet converged, and

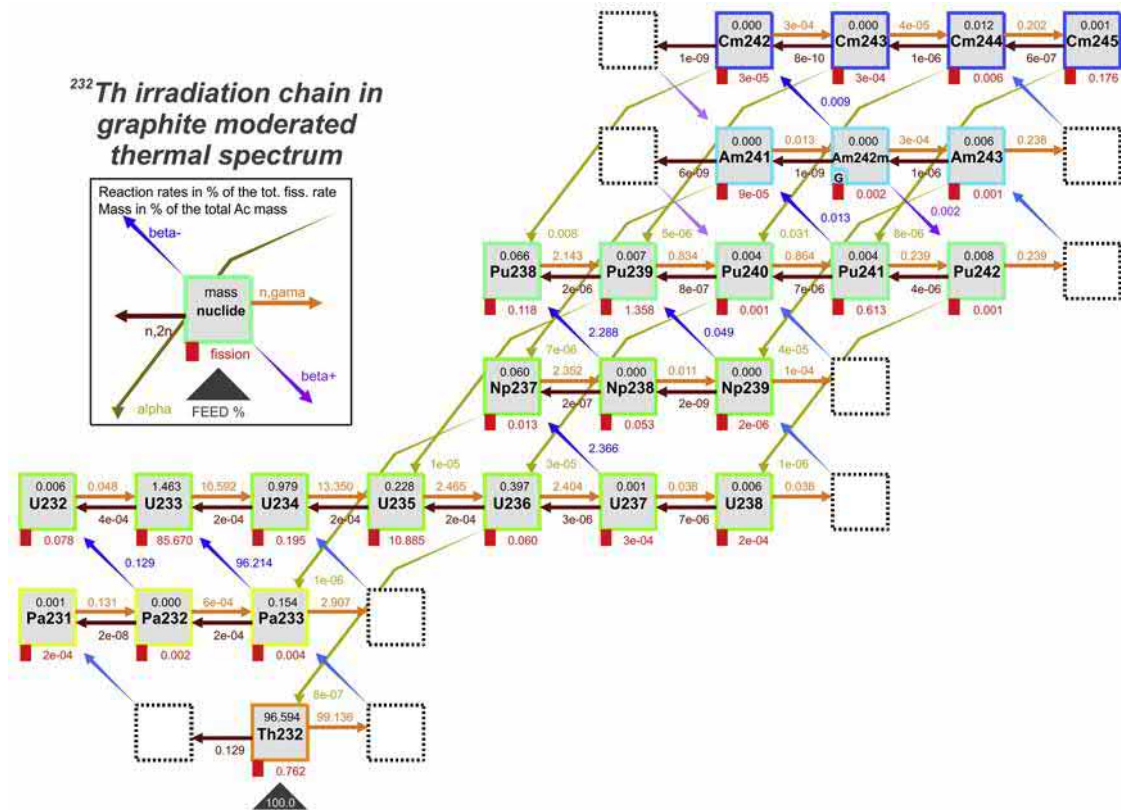


Fig. 3 Example of the ^{232}Th irradiation chain with relative masses and reaction rates in graphite moderated MSR. It partly illustrates also the ^{235}U irradiation chain. From Losa E (2016) *U-Pu and Th-U Fuel Cycle Closure*, PhD thesis. Prague: Czech Technical University in Prague, Department of Nuclear Reactors.

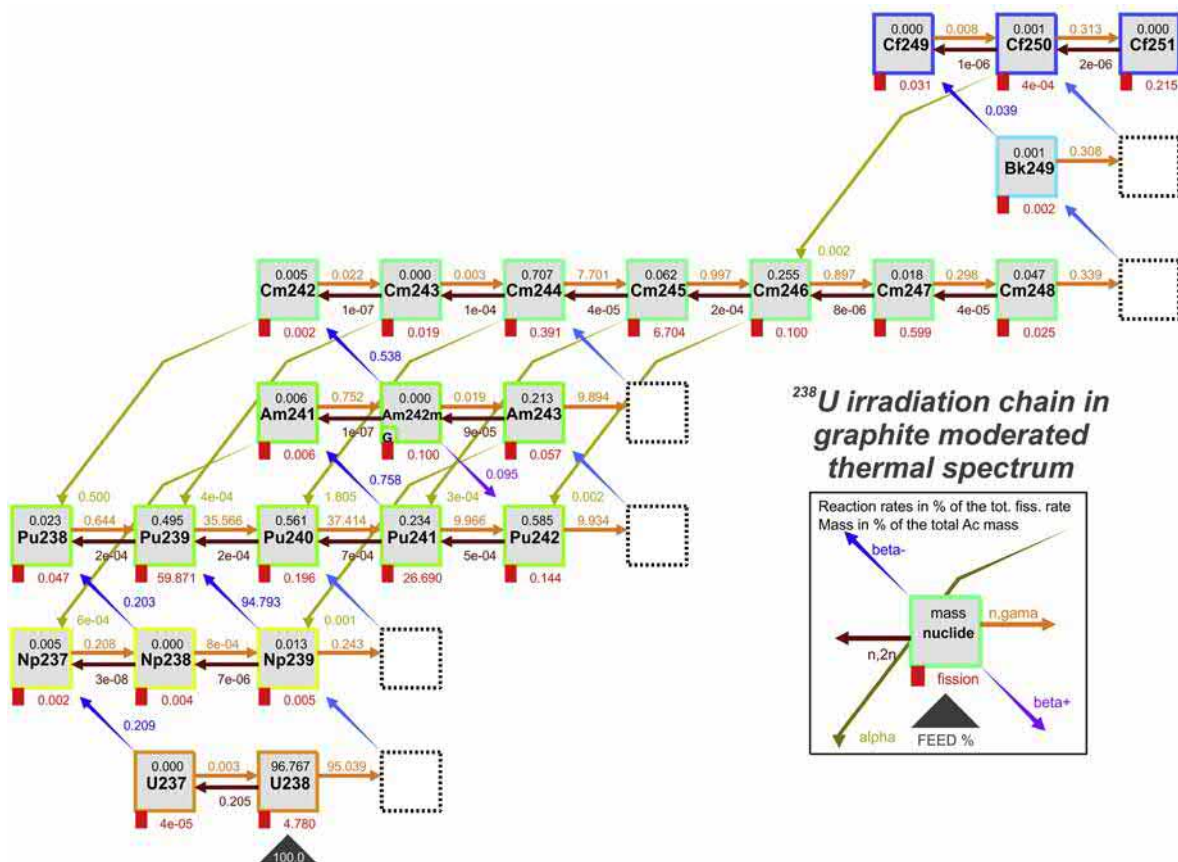


Fig. 4 Example of the ^{238}U irradiation chain with relative masses and reaction rates in graphite moderated MSR. From Losa E (2016) *U-Pu and Th-U Fuel Cycle Closure*, PhD thesis. Prague: Czech Technical University in Prague, Department of Nuclear Reactors.

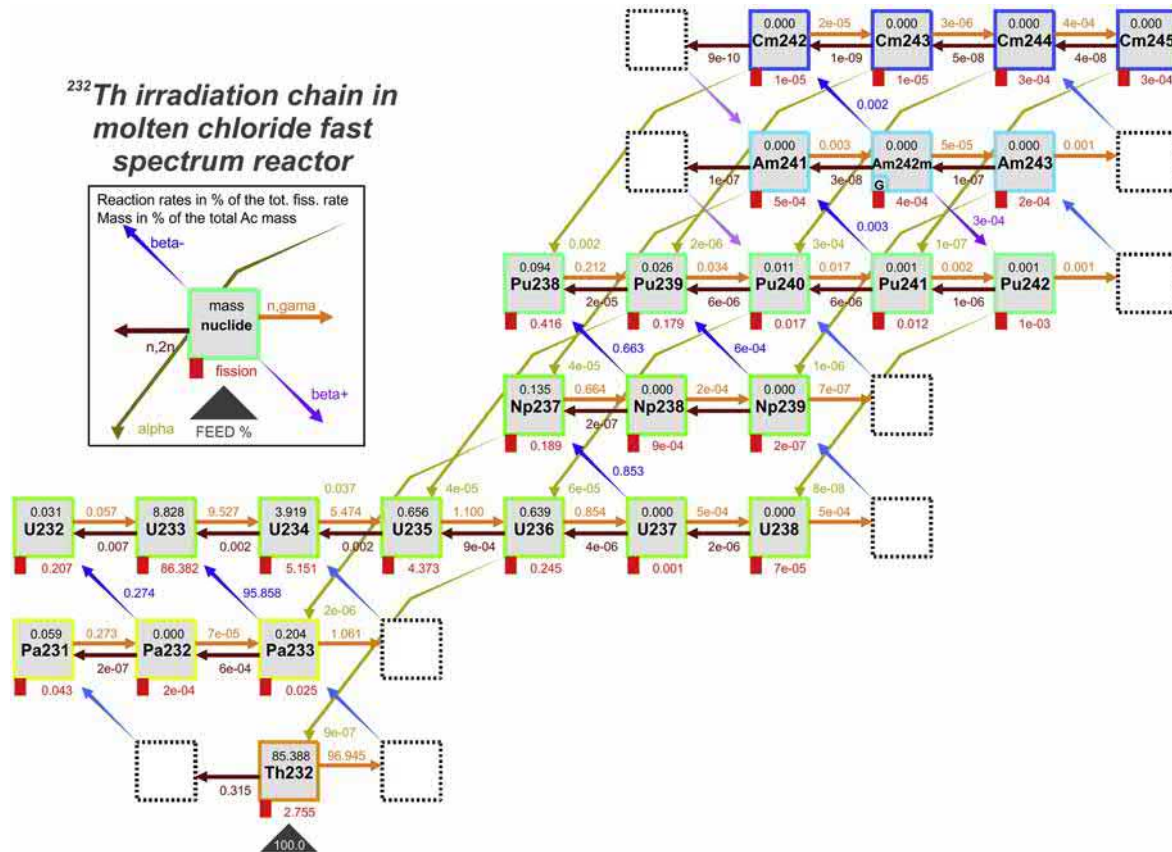


Fig. 5 Example of the ^{232}Th irradiation chain with relative masses and reaction rates in MCFR. It partly illustrates also the ^{235}U irradiation chain. From Losa E (2016) *U-Pu and Th-U Fuel Cycle Closure*, PhD thesis. Prague: Czech Technical University in Prague, Department of Nuclear Reactors.

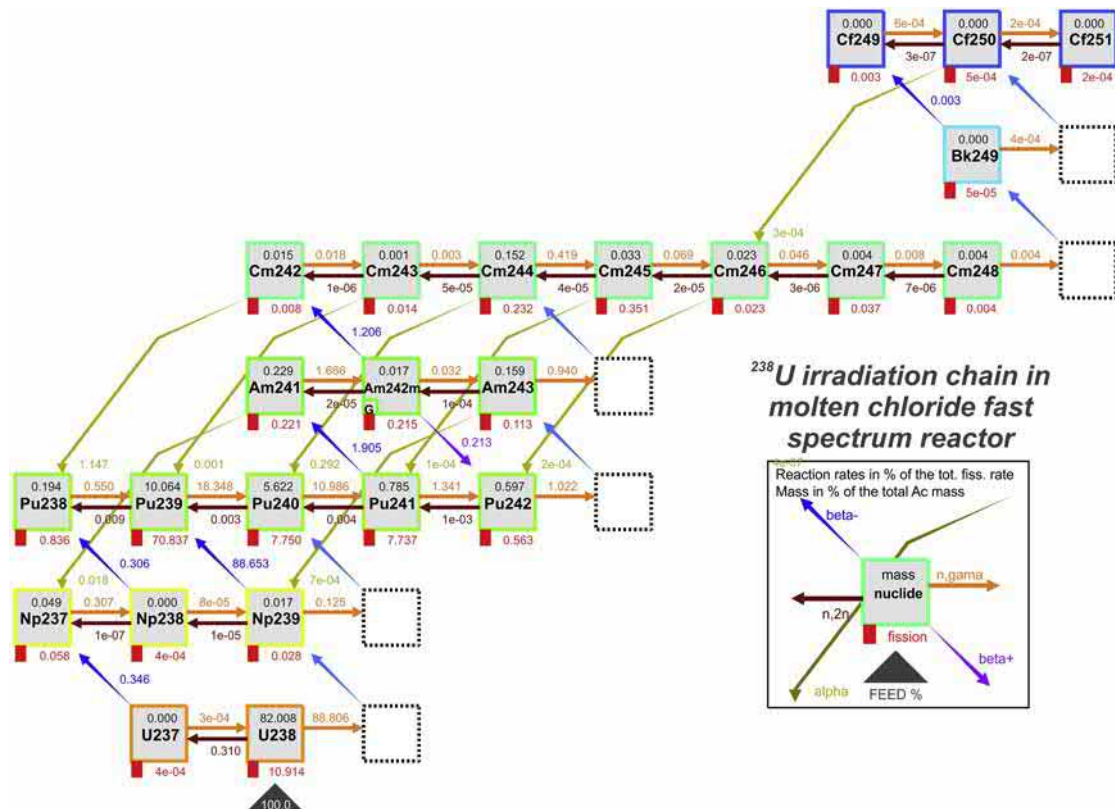


Fig. 6 Example of the ^{238}U irradiation chain with relative masses and reaction rates in MCFR. From Losa E (2016) *U-Pu and Th-U Fuel Cycle Closure*, PhD thesis. Prague: Czech Technical University in Prague, Department of Nuclear Reactors.

americium, curium, and their daughters will need substantially longer time for stabilization. The convergence process of higher plutonium isotopes is also the reason why plutonium is recycled optionally only once in ^{235}U -fueled reactors (although a second and third plutonium recycling is being considered in France (Guillaume et al., 2018)). The plutonium multi-recycling in ^{235}U -fueled reactors will not substantially increase the resources utilization; at the same time, it surely helps to minimize the synthetic actinides amount in the waste stream.

During the irradiation of ^{235}U , concentrations of daughter products like ^{234}U , ^{236}U and ^{237}Np also tends to equalize quite fast. Nonetheless, ^{235}U is fissile and its own concentration strongly decreases during the irradiation. Hence, equilibrium as such is never reached. Even though it is not accurate, the basic features of the ^{235}U irradiation chain can be understood from the ^{232}Th chain, because it is its sub-set. Based on the example in Fig. 3, for instance, the equilibrium concentration of ^{236}U seems to be higher than the ^{235}U concentration in a thermal spectrum reactor. However, also in fast neutron spectrum (see Fig. 5) the ^{236}U concentration is similar to ^{235}U concentration. The irradiated uranium, with left-over ^{235}U and its daughter products ^{234}U and ^{236}U can be also recycled in ^{235}U -fueled reactors. In this case it needs to pass again through the enrichment process to reduce the ^{238}U amount.

Nuclear physics implications

There is some obvious symmetry in the irradiation chains presented in Figs. 3 and 4 or in Figs. 5 and 6. This symmetry, but also all other features of nuclides such as: type of radioactive decay, half-lives, absorption cross-section, fission-to-capture probability, or even the energy released per fission can be explained by nuclear physics and, in particular, by the respective binding energy per nucleon and by the fission barrier.

Binding energy and line of stability

The binding energy per nucleon (protons and neutrons in the nucleus) is well described by empirical liquid drop model or actually the Weizsäcker formula (Weizsäcker, 1935). This formula has five terms: volume, surface, Coulomb, asymmetry and pairing. Balance of these terms define a line of stability. At this line or close to it the binding energy per nucleon is the highest and the nuclei are most stable. The line is defined by an interplay of only three terms of the Weizsäcker formula, because the volume term has no impact on the stability and the pairing term causes only local perturbations. The surface term is important for light nuclei and plays a role in fusion reactions but is less relevant for heavy nuclei and their fission. Furthermore, for nuclei, with the same number of nucleons, which are located on a diagonal (roughly orthogonal to the stability line in Fig. 7), the surface term is the same. The highest binding energy on this diagonal is determined by the balance of the Coulomb and asymmetry terms. These two terms quantify the reduction of binding energy caused by proton repulsive forces and by asymmetry between total proton and neutron count, see also Napolitano (2021) and Klay (2021).

Repetitiveness of nuclei properties

The Coulomb repulsive forces in actinide nuclei, with more than 89 protons, are enormous. The higher neutron count reduces the proton repulsive force. At the same time, it weakens the binding energy through the asymmetry term. Maximal stability of actinide nuclei is reached at 1.55–1.6 neutrons per proton. Yet they are all unstable and undergo radioactive decay. The approximate line of stability shown in Fig. 7 follows a trend where the same stability is obtained when addition of one proton is accompanied by addition of two neutrons. Nonetheless, with increasing proton number, the stability of actinides generally decreases. Moreover, due to the pairing effect, nuclei with even number of neutrons and/or protons have higher binding energy. Accordingly, properties of the nuclei along the stability line are repeating with steps of 2 protons and 4 neutrons ($+2p, +4n$). This repetition, illustrated in Table 1, is the major reason why exactly ^{232}Th and ^{238}U are present on earth and why there is a symmetry of their irradiation chains. Based on this repetition, ^{244}Pu could have been the third primordial actinide nuclide. Unfortunately, its half-life for radioactive decay is 56 times shorter than for ^{238}U . Nonetheless, for illustration of the repetitiveness, it is included in Table 1.

Nuclides in the ^{232}Th irradiation chain have less protons and are generally more stable. The respective longer half-lives are the reason why the estimated ^{232}Th reserves are higher and why the initial radiotoxicity of the respective synthetic actinides seems lower (refer to the later discussion). Nonetheless, the longer half-lives also result in a 11 times higher ^{233}Pa concentration than the respective ^{239}Np concentration in the ^{238}U chain and increase so the parasitic neutron capture probability. The second major difference between the ^{232}Th and ^{238}U irradiation chains is the very short half-life of ^{241}Pu and its different type of decay. At the corresponding position in the ^{232}Th chain, there is α decaying ^{235}U with the third longest half-life. As illustrated in Fig. 7, the major irradiation chain can branch at the ^{241}Pu position. Dominant direction after this split depends on the ratio of decay and capture rates. In a thermal neutron spectrum, in which the ^{241}Pu capture cross-section is larger and its resulting concentration is thus smaller, the capture rate dominates over the decay rate (see Figs. 3 and 4). In a fast neutron spectrum it can be the opposite (see Figs. 5 and 6).

Fission probability and cross-section amplitude

The fission barrier, as the second nuclear physics parameter, is not presented here graphically, but in the area of interest it has slight slope (Petermann et al., 2012); it decreases with increasing number of protons and decreasing number of neutron in the nucleus.

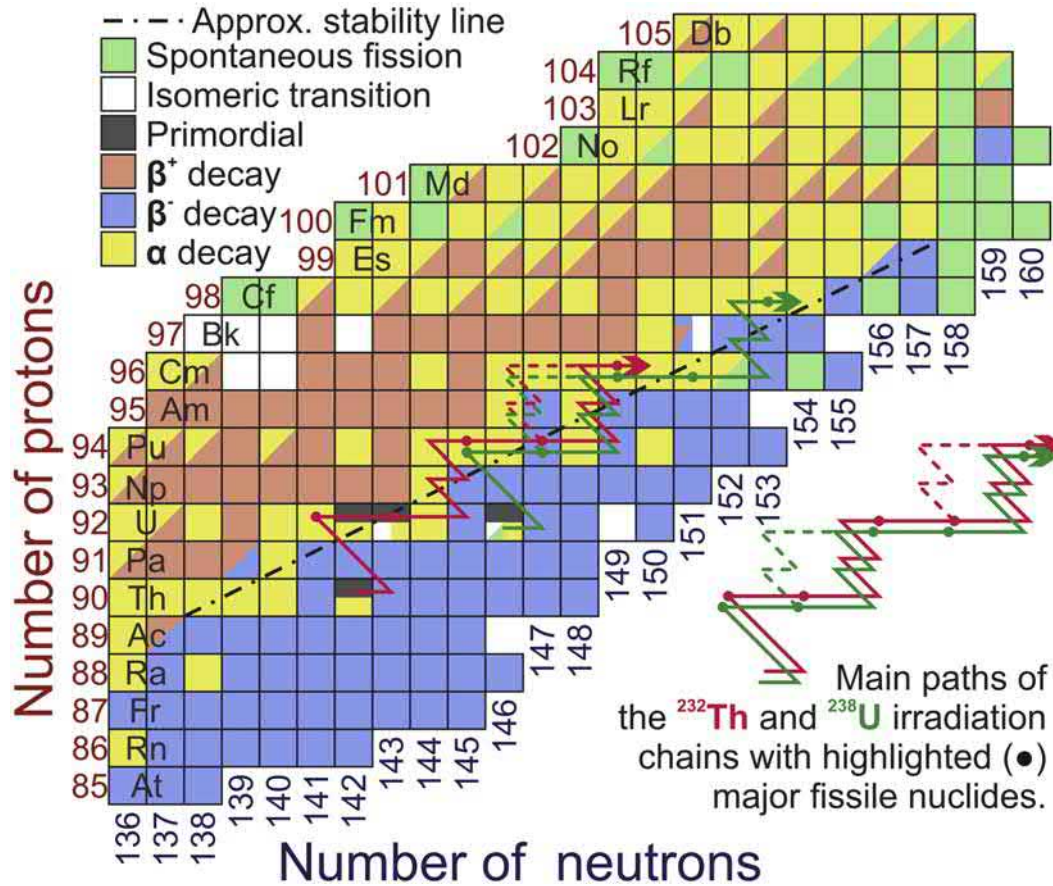


Fig. 7 Local nuclide chart of the actinides area with highlighted approximated line of stability and major paths for the ^{232}Th and ^{238}U irradiation chains with highlighted major fissile nuclides.

The Coulombian repulsive forces get stronger and the fission barrier weaker, when more protons are closer to each other (less neutrons). This is partly the reason why ^{233}U (see Fig. 7) has higher fission-to-capture probability than ^{235}U or ^{239}Pu . At the same time, ^{233}U and ^{235}U as the major fissile nuclides in ^{232}Th chain have lower number of nucleons than the respective ^{239}Pu in ^{238}U chain and release so, in average, less neutrons per fission.

To fission, each nucleus needs to overcome the fission barrier. The required energy is provided by the interacting neutron in potential (its binding energy) and kinetic form (its collision speed). Practically all actinide nuclides can be fissioned by fast neutrons with a certain probability. From this perspective, ^{238}U can be fissioned at lower neutron speed than ^{232}Th (see the respective fission rates in Table 3), what represents certain advantage for the ^{238}U irradiation chain. Possibility of fission by thermal neutrons is determined solely by the balance between fission barrier and binding energy. Since the binding energy depends also on the pairing effect, nuclei with odd number of neutrons are usually fissile by thermal neutrons.

The pairing effect is also relevant for the absolute size of neutron absorption cross-section. Nuclei with even number of neutrons have typically lower cross-sections for interaction with neutrons. This is actually the reason why the concentration of fissile nuclide in self-sustaining breeders is one or two orders of magnitude lower than the concentration of fertile nuclide.

Neutron economy

In the equilibrium irradiation chain, see Figs. 3–6, the actinides concentrations and reaction rates are stabilized. As a result, the equilibrium chain provides many perspectives and options allowing for estimating its neutron economy. The basic question is whether the reactor core could be critical with such an equilibrium fuel composition. Since all reaction rates in the presented chains are normalized to the fission rate and shown in percent, they can be interpreted as the history of one neutron generation, which was born from 100 fissions.

Table 1 Illustration of actinides properties repetitiveness along the line of stability with the steps of 2 protons and 4 neutrons for ^{232}Th , ^{238}U and ^{244}Pu irradiation chains.

Nuclide chains			Position	Protons (p)	Neutrons (n)	Fissile ^a	Decay	Longest half-life	Half-life ratio 1:2	Half-life ratio 2:3	Comment
1	2	3									
^{232}Th	^{238}U	^{244}Pu	Starting points	even	even	not	α	1.4E+10 years	3.1	56	b
^{233}Th	^{239}U	^{245}Pu	+0p,+1n	even	odd	–	β^-	< day	1	0.03	
^{233}Pa	^{239}Np	^{245}Am	+1p,+0n	odd	even	not	β^-	27 days	11	28	
^{233}U	^{239}Pu	^{245}Cm	+2p,-1n	even	odd	yes	α	1.6E+5 years	6.6	2.8	
^{234}U	^{240}Pu	^{246}Cm	+2p,+0n	even	even	not	α	2.5E+5 years	37	1.4	
^{235}U	^{241}Pu	^{247}Cm	+2p,+1n	even	odd	yes	α , β^- , α	7E+8 years	1E-6	45 (1:3)	c
^{236}U	^{242}Pu	^{248}Cm	+2p,+2n	even	even	not	α	2.3E+7 years	62	1.1	d
^{237}U	^{243}Pu	^{249}Cm	+2p,+3n	even	odd	–	β^-	7 days	32	4.6	
^{237}Np	^{243}Am	^{249}Bk	+3p,+2n	odd	even	not	α , α , β^-	2.1E+6 years	291	8152	e
^{238}Np	^{244}Am	^{250}Bk	+3p,+3n	odd	odd	–	β^-	2 days	4.8	3.1	
^{238}Pu	^{244}Cm	^{250}Cf	+4p,+2n	even	even	not	α	88 years	4.8	0.02	f
^{239}Pu	^{245}Cm	–	+4p,+3n	–	–	–	–	–	–	–	g

^aSimplified statement about fission probability by thermal neutrons.

^bIn these cases, the pairing term shifts the balance between Coulomb and asymmetry terms to the right from the stability line. ^{232}Th , ^{238}U and ^{244}Pu belong to the top four actinides with longest half-lives (see Fig. 1).

^cThese nuclides are close to the stability line. However, their behavior is irregular. ^{235}U and ^{247}Cm , as the third and sixth most stable actinide nuclides decay through α decay. Nevertheless, ^{241}Pu decays through β^- and its half-life is by 6 orders of magnitude lower than for ^{235}U . This nuclide causes the major irregularity between the ^{232}Th and ^{238}U irradiation chains.

^d ^{248}Cm can already undergo spontaneous fission.

^e ^{249}Bk decays already by β^- .

^f ^{250}Cf is more stable than ^{244}Cm .

^gBecause of the repetitiveness, refer to the previous occurrence.

Neutron capture rate

The most trivial estimate of the neutron economy is represented by the simple sum of all capture rates in the irradiation chain. This sum can be compared with the average number of neutrons per fission $\bar{\nu}$, reduced by one neutron needed for next fission, to obtain the respective neutron excess or deficiency. It is only an approximate value, its accuracy can be increased by accounting also for the (n,2n) transmutation reaction, which is a source of extra neutrons.

Costs of synthetic actinides fission

Alternatively, the neutron economy can be assessed from the knowledge of the fission rate distribution between the nuclides in the irradiation chain. Each synthetic actinide in the irradiation chain is originated by consecutive transmutations of the primordial nuclide ^{232}Th or ^{238}U . Accordingly, the neutron costs of its creation can be obtained as the difference between its nucleon number and the original nucleon number (232 or 238). The ^{241}Pu creation in ^{238}U irradiation chain costs 3 neutrons ($241-238=3$). Therefore, 4 neutrons are at the end needed for its fission (see Table 2). For ^{237}Np in the same chain – 1 neutron is needed for creation ($237-238=-1$). The cost for its fission is thus 0 neutrons. Based on the knowledge of the fission rate distribution within the irradiation chain the total neutron costs can be obtained. This method is also only approximate. Since many of the nuclides are decaying by α decay, the four nucleons lost in each decay should be accounted for in such a balance.

Neutron balance of the irradiation chain

The overall neutron balance n_{B1} of the actinides chain can be obtained from the average number of neutrons per fission $\bar{\nu}$, reduced by one neutron needed for next fission, and the sum of normalized capture rates C_i and $(n,2n)_i$ rates of a given nuclide i :

$$n_{B1} = \bar{\nu} - 1 - \sum_i C_i + \sum_i (n, 2n)_i$$

An alternative neutron balance n_{B2} can be estimated from the fission rate distribution F_i between the nuclides in the irradiation chain and from the normalized α decay reaction rate:

$$n_{B2} = \bar{\nu} - \sum_i F_i n_i - 4 \sum_i \alpha_i$$

Table 2 Neutron cost of creation and fission for selected actinide nuclides in ^{232}Th and ^{238}U irradiation chains.

Nuclides chain		Position	Protons (<i>p</i>)	Neutrons (<i>n</i>)	Fissile by thermal neutrons	Absolute neutron cost to create	Absolute neutron cost to fission
1	2						
^{232}Th	^{238}U	+0p,+0n	even	even	not	0	1
^{233}U	^{239}Pu	+2p,-1n	even	odd	yes	1	2
^{234}U	^{240}Pu	+2p,+0n	even	even	not	2	3
^{235}U	^{241}Pu	+2p,+1n	even	odd	yes	3	4
^{236}U	^{242}Pu	+2p,+2n	even	even	not	4	5
^{237}Np	^{243}Am	+3p,+2n	odd	even	not	5	6
^{238}Pu	^{244}Cm	+4p,+2n	even	even	not	6	7
^{239}Pu	^{245}Cm	+4p,+3n	even	odd	yes	7	8
^{240}Pu	^{246}Cm	+4p,+4n	even	even	not	8	9
^{241}Pu	^{247}Cm	+4p,+5n	even	odd	yes	9	10
^{245}Cm	^{251}Cf	+6p,+7n	even	odd	yes	13	14

where n_i is the number of neutrons needed for fission of nuclide i (see Table 2) and the product $F_i n_i$ represents the fission costs, neutron costs to create and fission the nuclide. The trivial neutron balance for the four irradiation chains in Figs. 3–6 is illustrated in Table 3. Here it must be stressed, that the result in Table 3 refer only to the neutron balance of the actinides. The overall neutron economy of a reactor will depend also on the capture rate of structural materials, coolants and moderators and on the neutron leakage from the reactor core. Nonetheless, some basic statements can be made, based on the results in Table 3.

The ^{232}Th irradiation chain provides less neutrons per fission, around 2.5 neutrons. At the same time, its neutron capture rate is smaller. Hence it provides neutron excess in both thermal (0.12 neutrons) and fast (0.37 neutrons) spectra. The ^{238}U irradiation chain provides more neutrons per fission, around 2.9 neutrons. However, also the neutron capture rate is higher. The overall neutron excess is positive only in the fast spectrum (0.68 neutrons). In the thermal spectrum there is neutron deficiency (−0.18 neutrons) and the reactor cannot be operated with this fuel composition. Hence, self-sustaining breeding is not possible in thermal spectrum with ^{238}U . The major weakness of the ^{238}U irradiation chain is the lower fission rate of ^{239}Pu . Its major advantage, the higher average number of neutrons per fission, cannot compensate it in thermal spectrum. In ^{238}U uranium chain in graphite moderated MSR 1.08 and 0.54 neutrons are lost by ^{241}Pu and ^{245}Cm fission respectively. The capture rate of ^{239}Pu (35.8%), as a gateway for higher isotopes build-up, is $3 \times$ higher than the ^{233}U capture rate (10.6%) in the same reactor.

Radiotoxicity of the irradiation chain

Actinide alpha decay series

Synthetic actinide nuclides, in the discharged fuel from an open cycle practiced today or as the recycling losses in a closed cycle cause a long-term stewardship burden. They have very long half-lives (see Fig. 1) and the daughter products are also radioactive. All nuclides relevant for nuclear fuel cycle undergo either α or β^- decay. Alpha decay reduces number of nucleons by 4 whereas beta decay does not change it. This is the reason why all actinides decay in four separate decay series, which differ by number of nucleons i ($4i$, $4i + 1$, $4i + 2$, and $4i + 3$). These actinide alpha decay series are called: Thorium ($4i$), Neptunium ($4i + 1$), Uranium ($4i + 2$) and Actinium ($4i + 3$) series and they end with the first available stable nuclide, i.e., ^{208}Pb , ^{209}Bi , ^{206}Pb , and ^{207}Pb . The radiotoxicity should account for the contribution of all the nuclides in the entire decay series.

Since actinides with odd proton number are very unstable, especially when also having odd neutron number; practically all fissile nuclides are in Neptunium and Actinium series. The ^{232}Th and ^{238}U irradiation chains are shifted by 6 nucleons (+2p, +4n); hence, the nuclides in the same position within each chain decay through different series. The pairing term of the Weizsäcker formula cause some irregularity; however the half-lives for radioactive decay generally decrease with increasing nucleon number. The nuclides in the ^{238}U irradiation chains have thus shorter half-lives (see Table 1). At the same time, nuclides from this chain often decay through one of the ^{232}Th irradiation chain nuclides. Finally, evaluation of radiotoxicity is complex, because slowly decaying nuclide, e.g., ^{233}U , can provide low initial radiotoxicity; however, secondary peak can occur once the concentration of daughter product, e.g., ^{229}Th , will be stabilized. To illustrate this behavior, the radiotoxicity of selected nuclides caused by 10^{30} atoms is shown in Fig. 8. It represent the radiotoxicity caused by ingestion of the radioactive materials by human. The necessary coefficients have been adopted from (Soti et al., 2019). This figure shows radiotoxicity normalized to 10^{30} atoms. Accordingly, the values for some minor nuclides are exaggerated. However, it is identifying nuclides with substantial secondary peak: ^{233}U (^{233}Th), ^{234}U (^{238}Pu , ^{242}Am , $^{242\text{m}}\text{Am}$), and ^{237}Np (^{241}Pu). The case of ^{237}Np is particular because it decays through ^{233}U ; however, it has long half-life, which delays the ^{233}U secondary peak occurrence. It is obvious that nuclides with secondary peak are predominantly located in the ^{232}Th irradiation

Table 3 Neutron cost of creation and fission of selected actinide nuclides in ^{232}Th and ^{238}U irradiation chains.

	^{232}Th chain	^{233}U	^{235}U	^{239}Pu	^{241}Pu	^{245}Cm	Sum fissile	^{232}Th	^{234}U	^{236}U	^{237}Np	^{238}Pu	Sum fertile	Sum others	Sum total	$\bar{\nu}$	η_{B1} - η_{B2}
Graphite moderated MSR	Fissile	Y	Y	Y	Y	Y	—	N	N	N	N	N	—	—	—	2.50	0.12
	Capture rate %	10.6	2.46	0.83	0.24	0.03	14.2	99.1	13.3	2.40	2.35	2.14	119	4.81	138	2.50	—
	Fission rate %	85.6	10.9	1.36	0.61	0.18	98.7	0.76	0.19	0.06	0.01	0.12	1.15	0.17	100	2.53	0.12
MCFR	Fission cost	1.71	0.44	0.11	0.06	0.02	2.34	0.01	0.01	0.00	0.00	0.01	0.03	0.01	2.38	2.53	0.37
	Capture rate %	9.53	1.10	0.00	0.00	0.00	10.6	96.9	5.47	0.85	0.66	0.21	104	1.45	116	2.53	—
	Fission rate %	86.4	4.37	0.18	0.01	0.00	90.9	2.76	5.15	0.24	0.19	0.42	8.76	0.30	100	2.53	0.37
	Fission cost	1.73	0.17	0.01	0.00	0.00	1.92	0.03	0.15	0.01	0.01	0.03	0.23	0.00	2.16	2.53	0.37
	^{238}U chain	^{239}Pu	^{241}Pu	^{245}Cm	^{247}Cm	^{251}Cf	Sum fissile	^{238}U	^{240}Pu	^{242}Pu	^{243}Am	^{244}Cm	Sum fertile	Sum others	Sum total	$\bar{\nu}$	η_{B1} - η_{B2}
Graphite moderated MSR	Fissile	Y	Y	Y	Y	Y	—	N	N	N	N	N	—	—	—	2.95	—0.18
	Capture rate %	35.8	10.2	1.01	0.30	0.11	47.5	95.5	37.8	10.2	10.1	7.80	161	3.46	212	2.95	—
	Fission rate %	60.0	27.1	6.79	0.61	0.23	94.7	4.28	0.14	0.15	0.05	0.40	5.03	0.31	100	2.93	—0.17
MCFR	Fission cost	1.20	1.08	0.54	0.06	0.03	2.92	0.04	0.00	0.01	0.00	0.03	0.09	0.02	3.02	2.93	0.68
	Capture rate %	18.3	1.34	0.07	0.01	0.00	19.8	88.8	11.0	1.02	0.94	0.42	102	2.85	125	2.93	—
	Fission rate %	70.8	7.74	0.35	0.04	0.00	78.9	10.91	7.75	0.56	0.11	0.23	19.6	1.50	100	2.93	0.68
	Fission cost	1.42	0.31	0.03	0.00	0.00	1.76	0.11	0.23	0.03	0.01	0.02	0.39	0.03	2.18	2.93	0.68

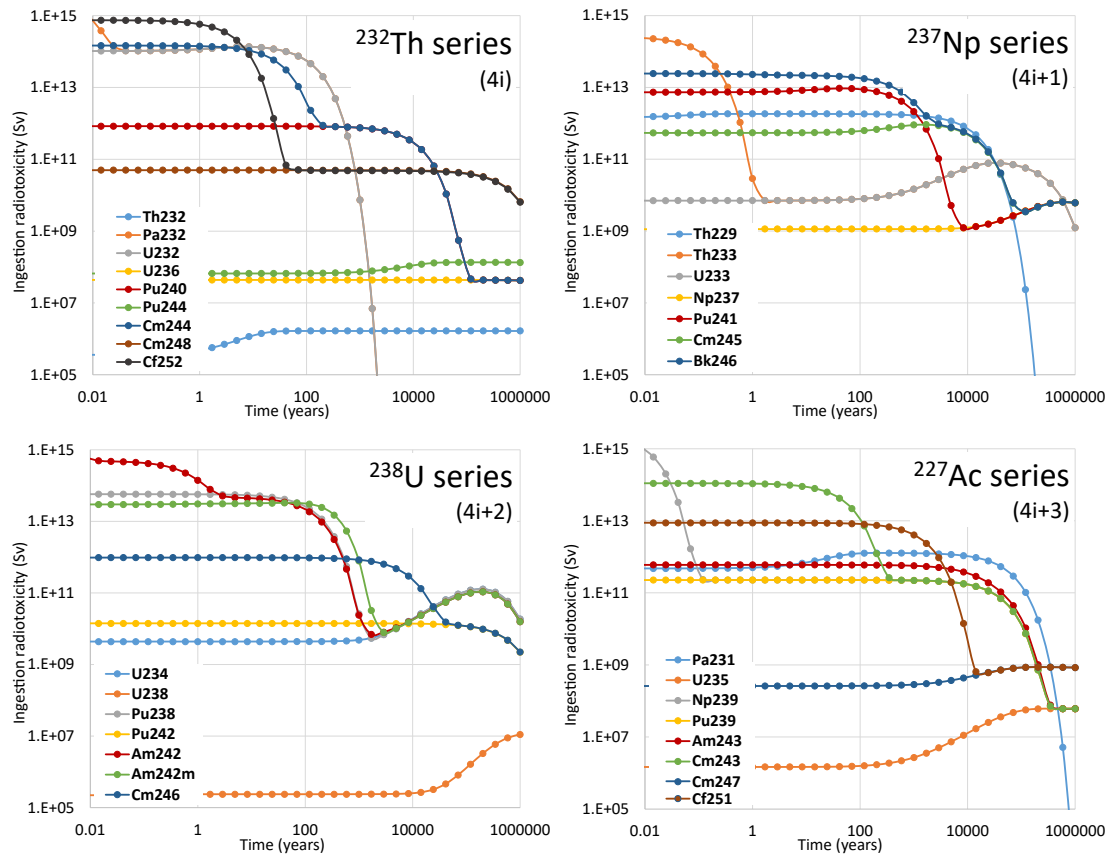


Fig. 8 Evolution of ingestion radiotoxicity originated from 10^{30} atoms of selected nuclides in the Thorium (top-left), Neptunium (top-right), Uranium (bottom-left) and Actinium (bottom-right) series.

chain. For ^{235}U , ^{238}U , and ^{244}Pu , there is a late radiotoxicity increase caused by longer half-life of the daughter products ^{231}Pa , ^{234}U , and ^{240}Pu . However it is rather plateau than secondary peak. Similar radiotoxicity increase appears earlier for ^{232}Th , because its daughter product ^{238}Th has relatively short half-life. In all these cases, the radiotoxicity level is low and, with exception of ^{244}Pu related to the natural ore radiotoxicity.

Radiotoxicity of the ^{232}Th and ^{238}U irradiation chains

The ingestion radiotoxicity of 10^{30} atoms can be presented also for the actinides compositions from the four irradiation chains presented in Figs. 3–6. In the ^{232}Th irradiation chain, actinides composition in thermal spectrum provide slightly lower radiotoxicity. This is caused by lower absolute share of synthetic actinides. The two curves follow the same trend. In both cases a secondary peak occurs (see left Fig. 9). The radiotoxicity of ^{238}U irradiation chain is initially higher in thermal spectrum. The absolute share of synthetic actinides is lower in thermal spectrum also in this case. However, the difference is smaller and the thermal spectrum chain includes larger share of higher synthetic actinides nuclides. Since these decay faster, already earlier than 100 years after discharge the fast spectrum chain provides higher radiotoxicity. The secondary peak is absent in ^{238}U chain and the radiotoxicity continuously decreases (see right Fig. 9).

The results from Fig. 9 can be used to compare the relative radiotoxicity of the ^{232}Th and ^{238}U irradiation chains. Their ratio is presented in left Fig. 10 using logarithmic time scale and in right Fig. 10 using linear time scale. As expected, ^{232}Th chain provides initially up to $40\times$ lower radiotoxicity. Nonetheless, at longer time periods the secondary peak results in up to $20\times$ higher radiotoxicity. Accordingly, the final conclusion would be determined by time period of interest.

Synthetic actinides from the irradiated fuel become waste once they cannot be recycled and reused. In the open cycle they become waste by political decision not to recycle and reuse them. In the closed fuel cycle it is rather the recycling frequency and technology efficiency. There will be always some fraction of synthetic actinides in reprocessing losses. Therefore, the maximal resources utilization depends on the reprocessing losses as shown by Fig. 2. The radiotoxicity of synthetic actinides per unit of produced energy in the closed cycle depends on the resources utilization level. To provide an example, the initial radiotoxicity, just after shut down of the equilibrium reactor, per unit of produced energy is plotted in left Fig. 11 as a function of natural resources utilization. The radiotoxicity per unit of energy linearly decrease with increasing resources utilization. However, the

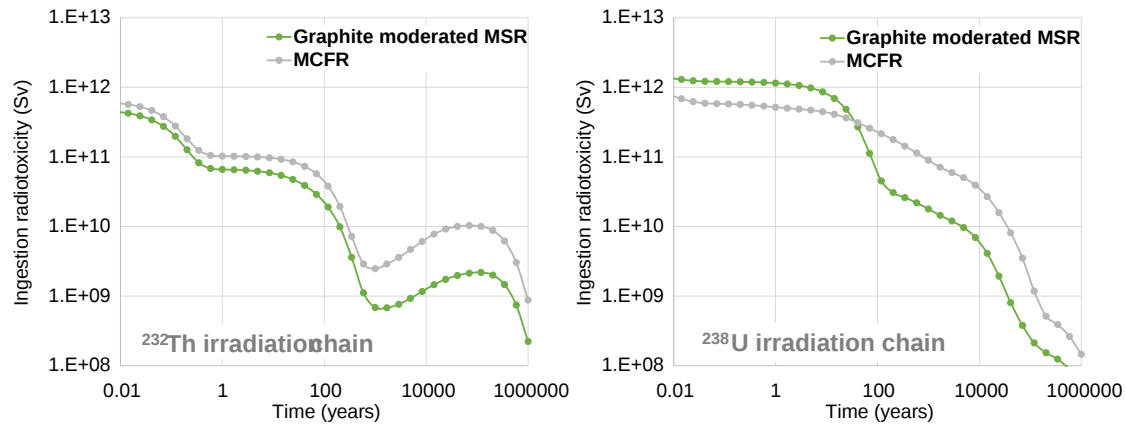


Fig. 9 Evolution of ingestion radiotoxicity originated from 10^{30} atoms of the ^{232}Th (left) and ^{238}U (right) irradiation chains in the graphite moderated MSR and MCFR.

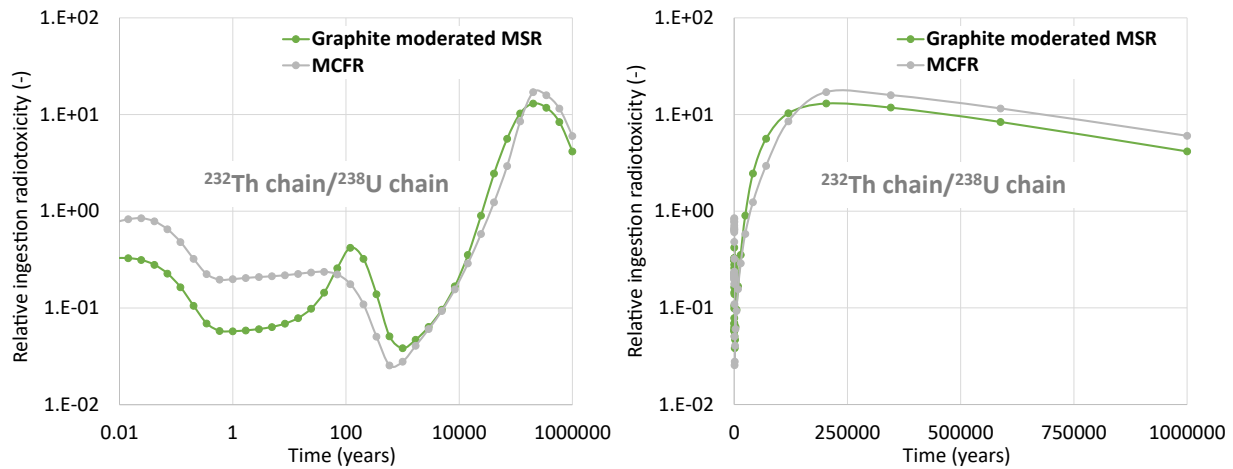


Fig. 10 Evolution of the ratio between the ingestion radiotoxicity in ^{232}Th and ^{238}U irradiation chains in logarithmic (left) and linear (right) time scale.

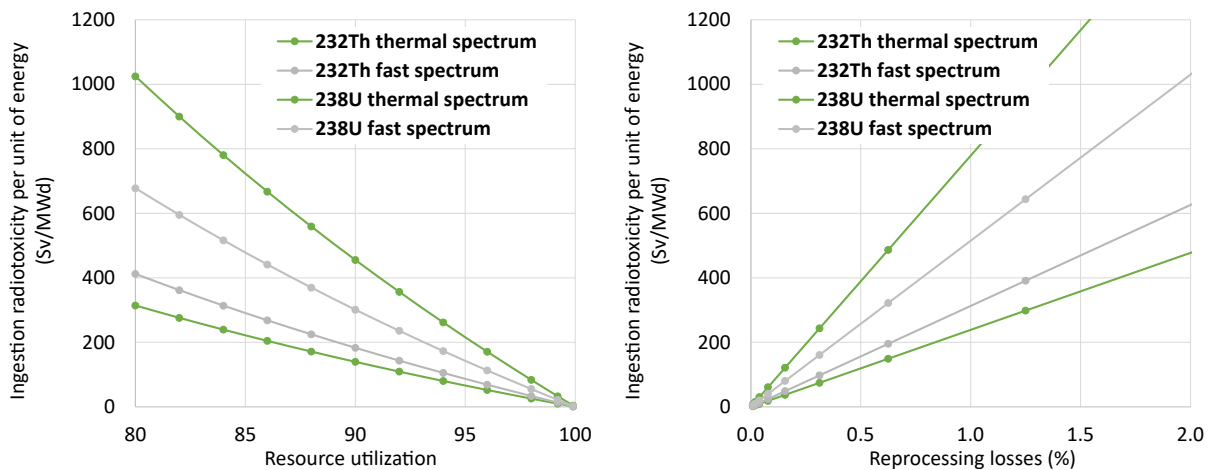


Fig. 11 Dependency of the ingestion radiotoxicity per unit of produced energy on resources utilization (left) and reprocessing losses (right) for ^{232}Th and ^{238}U irradiation chains.

utilization depends on reprocessing losses and reprocessed fuel burnup in FIMA %. Hence, in the right Fig. 11 the same initial radiotoxicity per unit of energy is plotted as a function of reprocessing losses, assuming 5% FIMA burnup of the reprocessed fuel. Since the burnup is fixed in this example, the radiotoxicity per unit of produced energy grows linearly with increased reprocessing losses.

Summary

The nuclear fuel cycle relies on resources, which are not renewable. To fully utilize their potential self-sustaining breeding in closed cycle is needed, which is demanding from the neutron balance perspective. Not all reactor concepts are capable to achieve self-sustaining breeding and fully utilize the natural resources. Yet, many advanced reactors are capable of self-sustaining breeding and thus to produce energy from primordial ^{232}Th and ^{238}U nuclides, using synthetic fissile nuclides ^{233}U and ^{239}Pu as a kind of catalyzer. The fuel composition in a self-sustaining breeder operated in a closed cycle tends to converge to equilibrium composition. The type of irradiation chains and the reactor neutron spectrum determine the equilibrium composition.

These chains follow the line of stability and show a certain symmetry. This symmetry is explained in this article by the binding energy per nucleon in each of the chain nuclides. The properties of actinides nuclides are repeating in the nuclide chart with the step of 2 protons and 4 neutrons. The major difference between these chains is that almost all nuclides in the ^{232}Th irradiation chain have longer half-life for radioactive decay than their respective nuclides in the ^{238}U chain. This is an advantage from several perspectives. The available ^{232}Th resources are larger than those of ^{238}U thanks to the longer half-life of ^{232}Th . By the same token, the stewardship burden caused by high-level waste of the ^{232}Th irradiation chain is smaller in the short term. However, it also results in 11 times higher ^{233}Pa concentration than the respective ^{239}Np concentration. The increased parasitic neutron capture probability of ^{233}Pa slightly deteriorates the neutronic performance of ^{232}Th irradiation chain. Another difference is the irregular behavior of ^{241}Pu in the ^{238}U chain relative to ^{235}U , which is at the respective position in ^{232}Th irradiation chain: ^{241}Pu has a six orders of magnitude shorter half-life and decays by β^- decay, versus alpha decay of ^{235}U . The last important difference is that ^{233}U and ^{235}U as the major fissile nuclides in ^{232}Th chain have lower number of nucleons than the respective ^{239}Pu and ^{241}Pu in the ^{238}U chain, and therefore release, on average, less neutrons per fission.

The ^{232}Th and ^{238}U irradiation chains are presented in thermal and fast reactor spectra. Based on the nuclide concentrations and reaction rates, neutron economy is quantified and long term ingestion radiotoxicity assessed. The existing knowledge was confirmed that self-sustaining breeding in closed cycle is not possible in thermal spectrum reactors when fed with ^{238}U and questionable even when fed with ^{232}Th . In fast neutron spectrum the ^{238}U irradiation chain profits more from neutron spectrum hardening and performs better than the ^{232}Th irradiation chain.

The ingestion radiotoxicity is lower for ^{232}Th irradiation chain on relatively short time scale but higher on the long time scale. The longer half-lives provide initial advantage to the ^{232}Th irradiation chain, but this chain features a radiotoxicity peak several 10s of 1000s years following discharge. Beyond $\sim 20,000$ years the ^{238}U irradiation chain radiotoxicity is lower due to the shorter half-lives of its nuclides.

The general conclusion of this article is that self-sustaining breeding in closed cycle can be achieved with both primordial nuclides ^{232}Th and ^{238}U . The differences between the respective irradiation chains are relatively small and both nuclides represent valuable natural resource.

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Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors

Jiri Krepel^a and Evzen Losa^b, ^a Paul Scherrer Institute, Villigen, Switzerland; and ^b Department of Nuclear Reactors, Czech Technical University, Prague, Czechia

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Glossary

Ac Actinides, elements with atomic numbers 89 to 103.

AcCl Label for tetra-chloride salt of actinides AcCl_4 .

Actinides Eigen-composition Stabilized actinides composition, an equilibrium composition which results from a long irradiation of one parent nuclide or their mixture. Necessary condition for the stabilization is that the mass of parent nuclide/s and the irradiation flux do not vary strongly. It is specific and inherent feature for every reactor and every parent nuclide/s and it represents the Eigen-vector of the respective Bateman equation.

B&B Breed-and-Burn; a mode of open cycle operation, where the fissile excess bred during irradiation is equal or higher than the discharged fissile mass after the irradiation. The reactor burns its own bred fuel. The discharged fuel does not need to be recycled and the fresh fuel does not need to contain primordial or synthetic fissile nuclides.

BG Breeding gain, $\text{BG} = \text{BR} - 1$.

BR Breeding ratio, it is ratio between breeding rate and fission rate. In a self-sustaining breeder $\text{BR} \geq 1$.

Breeding Transmutation of actinide nuclide initiated by neutron capture, which increases fission probability. Typically, primordial non-fissile (fertile) nuclides ^{232}Th and ^{238}U are transmuted into synthetic (man-made) fissile ^{233}U and ^{239}Pu .

Burnup Share of already fissioned actinides. It can be expressed in FIMA % (Fissions per Initial Metal Atom) or as energy released from given mass of actinides in MWd/kg (Megawatt days per kilogram). Since the energy per fission is approximately the same for all actinide nuclides, values expressed in these two units are proportional and roughly 10 times higher for the second unit.

Closed cycle A chain of process steps which is closed for the respective working medium. The medium, in nuclear fuel cycle the actinides, is recycled and does not extensively leave the chain.

Cross-section A measure of the probability that neutron will interact when flying by with the respective nucleus (microscopic cross-section) or when fling through with the respective nuclei concentration (macroscopic cross-section).

Equilibrium reactivity Reactivity provided by actinides Eigen-composition.

FPs Fission products, typically two fragments of actinides fission.

FLI Label for eutectic mixture of lithium fluoride salt and actinides tetra-fluoride salt LiF-AcF_4 .

FLIBE Label for eutectic mixture of lithium fluoride and beryllium di-fluoride with addition of actinides tetra-fluoride salt $\text{LiF-BeF}_2\text{-AcF}_4$.

Irradiation chain Series of consecutive transmutations and of the respective daughter products induced by neutron irradiation of one parent nuclide, e.g., ^{232}Th or ^{238}U . The radioactive decays of the daughter products are part of the chain.

MA Minor actinides, synthetic actinides originated by uranium transmutation (Np, Am, Cm and higher elements) without Pu, which is some time called major actinide.

NaCl Label for eutectic mixture of sodium chloride salt and actinides tetra-chloride salt NaCl-AcCl_4 .

Neutron economy Balance between neutron generation and neutron losses in a reactor. Good neutron economy has high neutron generation and small neutron losses by parasitic absorption and leakage from the reactor.

Open cycle A chain of process steps which is open for the respective working medium. The medium, in nuclear fuel cycle the actinides, is not recycled and leaves extensively the chain.

Primordial actinides Actinide nuclides, presumably originated by rapid neutron capture process during a supernova explosion, which have long enough half-life for radioactive decay to be still present on the earth.

Reactivity Reactivity is related to the effective (k_{eff}) or infinite (k_{inf}) neutron multiplication factor. Reactivity denotes the deviation from the self-sustaining fission chain reaction ($k = 1$), for which it is equal to zero. Reactor with negative reactivity ($k < 1$) cannot sustain the fission chain reaction. Positive reactivity ($k > 1$) means that there are more neutrons than needed for the self-sustaining fission chain reaction. From operational perspective, neutron absorber should compensate positive reactivity to control the chain reaction. From fuel cycle perspective, positive reactivity indicate that there are excess neutrons available for additional utilization by breeding or transmutations.

Self-sustaining breeding Breeding process where breeding rate $\geq$ fission rate, the mass of fissile synthetic nuclides is conserved in the reactor, and fission chain reaction is sustained solely by these synthetic nuclides. A long time-horizon is being considered in this article.

Subcritical/critical core A core that cannot/can sustain a fission chain-reaction.

Synthetic actinides Actinide nuclides originated, typically by neutron-induced transmutation, from primordial actinides. Synthetic nuclides are, by definition, a much broader group than minor actinides (MA) or trans-uranic elements. For thorium and uranium elements, both primordial and synthetic isotopes exist.

Waste Waste is unwanted side product, which is further unusable. In nuclear fuel cycle, synthetic actinides represent a side product, which is reusable. However, they can be declared as a waste by nuclear fuel cycle policy or when it is laborious to recover them, e.g., from reprocessing losses of certain fuel forms.

Abbreviations

FHR Fluoride high temperature reactor

GFR Gas cooled fast reactor

HPLWR High performance light water reactor

HTR High temperature reactor

LFR Lead cooled fast reactor

LWR Light water reactor

MCFR Molten chloride fast reactor

MFBR Metal fueled fast breeder reactor

MSFR Molten salt fast reactor

MSR Molten salt reactor (graphite moderated)

PHWR Pressurized heavy water reactor

RBMK Реактор Болшой Мощности Канальный (high-power graphite moderated water cooled channel-type reactor)

SCWR Supercritical Water-cooled Reactor

SFR Sodium cooled fast reactor

Introduction

This is the second of a series of three chapters that is dedicated to the identification of the advanced reactor technologies that are capable of self-sustaining breeding when fueled with either ^{232}Th or ^{238}U , and to the investigation of equilibrium fuel cycles properties and neutronics performance. The first chapter (Krepel, 2021a) characterizes the available fuel resources from the nuclear and reactor physics perspective. This second chapter determines the equilibrium fuel composition in advanced reactors and compares the basic features of the equilibrium fuel cycle and selected reactor neutronic performance characteristics at the equilibrium state. The third chapter (Krepel, 2021b) evaluates the fuel cycle performance from a neutronics perspective by means of the earlier

obtained equilibrium parameters. It provides an additional insight on why some advanced reactor concepts can act as self-sustaining breeders in closed or even open fuel cycle, while others do not.

Primordial actinide and self-sustaining breeding

The nuclear fuel cycle resources are not renewable and consist of three dominating primordial actinide nuclides: ^{235}U , ^{238}U and ^{232}Th . To fully utilize the potential of fertile ^{232}Th and ^{238}U , the self-sustaining breeding in closed cycle is needed. In such fuel cycle the primordial actinides are transmuted into synthetic fissile actinides ^{233}U and ^{239}Pu . This transmutation process is called breeding and reactors relying on this process are called breeders. Once originated, synthetic nuclides ^{239}Pu and ^{233}U can sustain the fission chain reaction and the breeding process itself. Accordingly, they practically act as a catalyzer for energy production from the primordial nuclides ^{238}U and ^{232}Th .

Simple fuel balance of the self-sustaining breeder requires that for each fissioned synthetic nucleus at least one transmutation of primordial nuclide takes place. Accordingly, up to two neutrons from each fission are needed; one for sustaining the chain reaction and one for maintaining of the fissile material balance. For each reactor and for each primordial nuclide, the composition of generated synthetic actinides tends to stabilize and reaches equilibrium concentration after sufficiently long neutron irradiation.

Irradiation chain and equilibrium cycle

Stabilized irradiation chain is obtained, when primordial nuclides ^{232}Th or ^{238}U are irradiated for sufficiently long time (Krepel, 2021a). The condition, necessary for this convergence, is that the primordial nuclide concentration and the irradiation flux do not vary. This condition is practically fulfilled by the operation of a breeder in closed fuel cycle, where all actinides are recycled and solely ^{232}Th or ^{238}U is refilled to the reactors as the feed material. The stabilized actinides composition is therefore also the Eigen-composition of the respective Bateman equation (Bateman, 1910). The so called equilibrium fuel cycle is a state, at which the self-sustaining breeder operated in closed cycle reaches the stabilized actinides composition. This equilibrium state is characterized by three variables: equilibrium fuel composition, equilibrium neutron spectrum and equilibrium multiplication factor.

The equilibrium fuel composition depends on the neutron spectrum and feed nuclide, here ^{232}Th or ^{238}U . The basic neutron spectrum type is determined by neutron scattering by the coolant and structural materials. However, it is also influenced by the equilibrium composition. Equilibrium spectrum and equilibrium composition together determine the equilibrium reactivity or actually equilibrium neutron multiplication factor (see Fig. 1). It is important to understand here that the multiplication factor does not affect the composition or spectrum. At the same time, reactivity excess or deficiency in equilibrium is the major indicator of the neutron economy and breeding capability.

From a historical perspective, the existence of an equilibrium state was revealed already during the pioneering time of nuclear energy, and the history is well described in Walter and Reynolds (1982). In a typical solid fuel fast breeder reactor, it takes five or six recyclings for plutonium vector stabilization (Journet et al., 1993). To enable concepts cross-comparison, a generalized Bateman equation was proposed and solved by a simplified approach in Salvatores et al. (2004). The term “equilibrium method” was also introduced in this reference. Inspired by this work, the EQL3D routine was developed (Krepel et al., 2008, 2012), which is a predecessor of the tool applied in the studies described in this article. The existence of an equilibrium state was obvious also for the developers of MSR concepts (MacPherson, 1985). Since it has liquid fuel, which enables continuous refueling and fission products removal, the knowledge of equilibrium was crucial for MSR. Several dedicated tools have been developed for MSR analyses in recent years, e.g., (Nuttin et al., 2005; Aufiero et al., 2013; Fiorina et al., 2013a; Seifried et al., 2013). The same is valid for bebble-bed HTR reactor, which also enables continuous removal and refueling of the fuel pebbles (Fratoni and Greenspan, 2007, 2010). Nowadays, there are many similar tools.

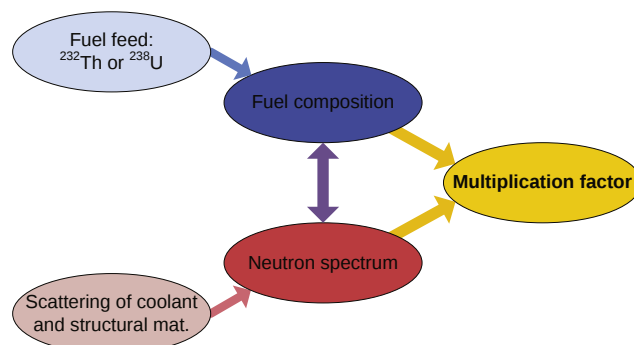


Fig. 1 Interdependency between neutron spectrum, fuel composition, and multiplication factor with indicated impact of the feed and scattering of coolant and structural materials. From Krepel J and Losa E (2019) Closed U-Pu and Th-U cycle in sixteen selected reactors: Comparison of major equilibrium features. *Annals of Nuclear Energy* 128: 341–357.

Objectives of this article

The objective of this article is to characterize the equilibrium fuel cycle for both the ^{232}Th and ^{238}U irradiation chains in 16 selected advanced reactors. The knowledge of the equilibrium state is of paramount importance for reactor performance evaluation, because it provides information:

- About the composition towards which the initial fuel tends to converge during irradiation;
- About the expected waste composition in both open and closed fuel cycles;
- Whether fuel parasitic neutron capture allows for establishing self-sustaining breeding;
- Whether there is sufficient neutron excess in the self-sustaining breeder for the utilization of legacy synthetic actinides (say, from LWR once-through fuel cycle) or for surplus fissile fuel breeding;
- About the rate of decrease of the legacy synthetic actinides concentration during irradiation.

This article is an extension and simplification of Krepel and Losa (2019), where details are provided. Selected aspect of this study have been published in Losa (2016), Krepel and Losa (2016, 2017) and Krepel et al. (2018).

Applied tool, assumptions and simulated reactors

The tool applied to more than 70 equilibrium calculations in this study is the Paul Scherrer Institut in-house procedure EQL0D (Hombourger et al., 2015, 2019). The simulations were relying on several simplifying assumptions, the major being Fission Products (FPs) neglect. The major impact of this simplification is an overestimation of the equilibrium reactivity. The selection of simulated reactors was aiming at covering all major neutron spectra and all major advanced reactor concepts. Accordingly, all Gen-IV reactor concepts described in this encyclopedia were included (Stanculescu, 2021).

Applied tool

The EQL0D procedure for burnup calculations is based on a MATLAB script. Historically there are three versions of the procedure: v1 is based on a MATLAB script and the deterministic ERANOS code (Rimpault, et al., 2002), and versions v2 and v3 that are based on a MATLAB script and the Monte Carlo code SERPENT 2 (Leppänen, 2013). The most advanced version v3 (Hombourger et al., 2016, 2019) adopts directly the burn-up matrix from the SERPENT code and applies the same method for the Bateman equation solution, including FPs. However, in this study the EQL0D v2 was applied (Hombourger, 2013; Losa, 2016), which relies on the MATLAB coupling with the SERPENT (see Fig. 2) code only through the actinide reaction rates. Hence, it cannot simulate FPs. As mentioned, the absence of FPs will result in equilibrium reactivity overestimation and slight spectrum softening. Whereas the spectral change can be neglected, the reactivity overestimation should be kept in mind.

The simulation of the ^{238}U and ^{232}Th irradiation used the ENDF/B-VII.0 nuclear data library (Chadwick, et al., 2006) and the SERPENT 2 code version was 2.1.23. The convergence of fuel composition was obtained after several iterations between the MATLAB burnup solver and the SERPENT code that generated spectrum-averaged cross-section for use in the burnup calculations. 20,000 source particles, 2000 active cycles, and 100 inactive cycles were used in the SERPENT calculations. The Monte Carlo nature of the code introduced issues with data noise. Its impact was not extensively evaluated for the EQL0D v2 version. Nevertheless, to minimize the oscillations relaxation between iterations was applied and relative mass change of ^{246}Cm smaller than 10^{-6} was used as a convergence criterion.

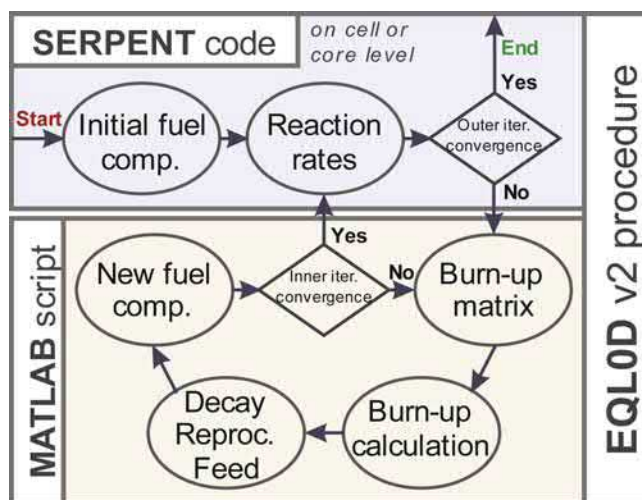


Fig. 2 Flow chart of EQL0D v2 routine. From Krepel J and Losa E (2019) Closed U-Pu and Th-U cycle in sixteen selected reactors: Comparison of major equilibrium features. *Annals of Nuclear Energy* 128: 341–357.

Assumptions

With the objective of this article being the comparison of the basic equilibrium features of the ^{232}Th and ^{238}U irradiation chains in advanced reactors, the results assembled here are based on three major assumptions:

- (1) The amplitude of the neutron flux, in which the fuel is irradiated, is determined by the specific power. It is unimportant for the fuel composition convergence if the core is subcritical or supercritical.
- (2) Each fission results in refilling of one respective ^{238}U or ^{232}Th atom, and FPs, as mentioned, are neglected. With this assumption, the usual composition variations during irradiation, cooling, and reprocessing stages are avoided.
- (3) Neutron leakage is not accounted for. The core is simulated only on the assembly level with reflective boundary conditions.

The first assumption assures that equilibrium state is achieved independent from the multiplication factor. The second and third assumptions simplify the modeling; however, they also result in equilibrium reactivity over-prediction. Presence of FPs and of neutron leakage will definitely reduce the equilibrium reactivity. It will also have an impact on the reactor spectrum and thus on the equilibrium fuel composition. However, this impact is secondary and the fuel compositions obtained by these assumptions are sufficiently representative for the needs of this study.

Simulated reactors

The advanced reactor concepts evaluated in this study follow the Gen-IV systems SFR, VHTR, LFR, GFR, MSR, and SCWR introduced in this encyclopedia in [Stanculescu \(2021\)](#), [Heidet \(2021\)](#), [Fütterer \(2021\)](#), [Alemberti \(2021\)](#), [Hatala \(2021\)](#), [Ignatiev \(2021\)](#) and [Huang and Zang \(2021\)](#). However, this study includes also alternative concept such as the FHR ([Fratoni, 2021](#)), chlorides salt based MCFR ([Krepel and Kramer, 2021](#)), and advanced LWR ([Shwageraus, 2021](#)). Moreover, to ensure that the examined reactor concepts are providing maximal variability of the neutron spectrum, at least one example is adopted from concepts considered in this encyclopedia, specifically: pressurized water cooled reactors ([Brown, 2021](#)), heavy water cooled reactors ([Nichita, 2021](#)), gas cooled reactors, water cooled graphite moderated reactors, and liquid metal-cooled reactors. The boiling water cooled reactors ([Hamon, 2021](#)) is not included as such, but covered by a parametric moderation study for LWR. In general, three basic principles were used for the selection:

- (1) All Gen-IV solid fuel reactors.
- (2) Most realistic MSR concepts.
- (3) Majority of existing power reactors.

As a result 16 reactor types were selected: 8 thermal and 8 fast; 6 based on liquid fuel in form of molten salts, and 10 featuring 4 different solid fuel types. The selected reactors and their labels used in results section are presented in [Table 1](#). It also includes the coolant density steps for the LWR parametric study. The assumed specific powers in [Table 1](#) are determined by three fuel cycle parameters: average irradiation time, burnup and fuel density, and refer to the ratio of volumetric power and specific actinides density, with unit W/g_{Ac} . The specific power for the MSR cases, where the core size is not always known, was adopted from LWR for fluoride salts and from LFR for chlorides salts, respectively. The higher specific power proposed for MFBR is adopted from [IGCAR \(2004\)](#). The lattices of the selected reactors are illustrated in [Fig. 3](#). Basically, there are five types of fuel: oxide pellets (LFR, SFR, LWR, RBMK, PHWR, HPLWR), TRISO coated particles in graphite pebbles in FHR and HTR, carbide pellets in GFR, zirconium based metallic fuel in MFBR, and molten salts in MSRs. The salts are further subdivided into chlorides and fluorides (see [Table 1](#)).

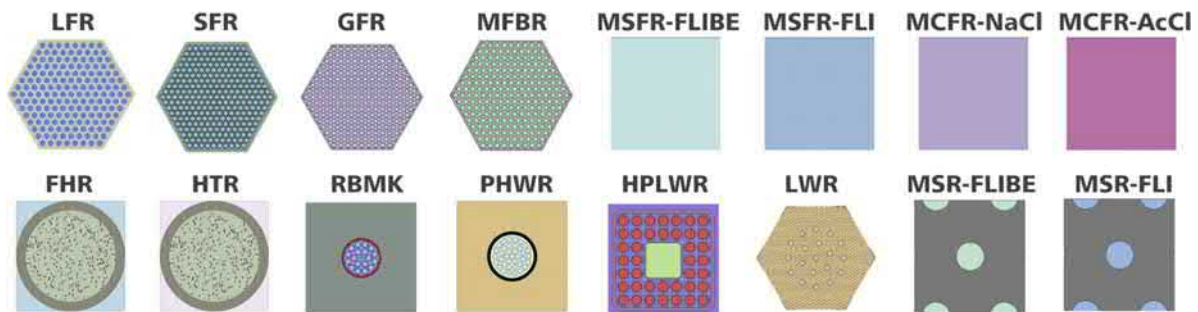
The solid fuel Gen-IV reactors include the sodium cooled reactor with both ceramic fuel (SFR) and metallic fuel (MFBR) options. The HTR is included together with its fluoride salt cooled version FHR. The other three reactors LFR, GFR, and SCWR, are included using one representative designs. Especially in case of SCWR, where HPLWR concept was selected, the choice is crucial, because the cladding material has strong impact on the core performance.

The MSR is rather a category of reactors than one single design; hence, the choice was driven by feasibility of the structural material, moderator, and fuel salt combinations. There are three major MSR classes: Graphite based MSRs, Homogeneous MSRs and Heterogeneous MSRs ([IAEA, 2021](#)). The latter class was excluded from this study. The reactors with heterogeneous layout rely on structural material to separate the fuel from dedicated coolant or non-graphite moderator. The major reason for their exclusion was the absence of publicly available data and some doubts about their technological maturity. However, a parametric study relevant to non-graphite moderated MSRs is presented in [Hombourger \(2018\)](#). The performance of heterogeneous fast MSRs is not addressed, as it is probable that the dedicated coolant and separation structural materials will increase both neutron scattering and capture reaction rates, and thus deteriorate the neutronics performance of this reactor type at the infinite lattice level considered in this study.

Only three MSR families were included in this lattice study: graphite moderated MSR, homogeneous fluoride fast MSFR, and homogenous chloride fast MCFR. Two fluoride and two chloride salts have been considered as the fuel carrier: $^7\text{LiF-AcF}_4$ and $^7\text{LiF-BeF}_2\text{-AcF}_4$ labelled as FLI and FLIBE and $\text{Na}^{37}\text{Cl-Ac}^{37}\text{Cl}_4$ and $\text{Ac}^{37}\text{Cl}_4$ labelled as NaCl and AcCl, respectively. The practical applicability of AcCl salt is rather hypothetical. The UCl_4 has reasonable melting temperature but is unstable; on the contrary, the stable UCl_3 has a high melting point. Since it is still much lower than the melting point of metallic actinides, AcCl was included in the study as a hypothetical actinide carrier delivering maximum performance. The selected salts are the best performing representatives from the neutronics perspective ([Hombourger, 2018](#)). For the graphite moderated MSR, the salt channel radius of 5 cm and the lattice geometry was adopted from [Hombourger et al. \(2016\)](#) using the case with 15% salt fraction ([Rosenthal, et al., 1971](#); [Krepel, et al., 2014](#)).

Table 1 Parameters of 16 simulated reactors and cases for the parametric study (Krepel and Losa, 2019).

Reactor name	Label	Fuel design adopted from (reference)	Fuel type	Specific power (W/g _{Ac})
<i>Solid fuel fast reactors</i>				
European lead system	LFR	Consortium of EU FP7 LEADER project (Artioli et al., 2009; Krepel et al., 2012)	Oxide	54.8
European sodium fast reactor	SFR	Consortium of EU FP7 ESFR project (Buiron, et al., 2007; Krepel et al., 2012)	Oxide	48.8
Gas cooled fast reactor	GFR	Consortium of EU FP7 GoFastR project (Bosq et al., 2006; Perkó et al., 2015)	Carbide	40.1
Metal fueled fast breeder reactor	MFBR	IGCAR, Kalpakkam (IGCAR, 2004; Mohapatra, et al., 2013)	Metallic	178.6
<i>Liquid fuel fast reactors</i>				
Fast MSR: LiF-BeF ₂ -AcF ₄ salt	MSFR-FLIBE	MSBR fuel salt properties (Rosenthal et al., 1971)	Fluoride salt	41.1
Fast MSR: LiF-AcF ₄ salt	MSFR-FLI	Consortium of EU FP7 EVOL project (Heuer et al., 2014)	Fluoride salt	41.1
Fast MSR: NaCl-AcCl ₄ salt	MCFR-NaCl	Salt eutectic comp. and density from Desyatnik et al. (1975) and Taube (1977)	Chloride salt	54.8
Fast MSR: AcCl ₄ salt	MCFR-AcCl	Salt density adopted from Desyatnik et al. (1975) and Taube (1977)	Chloride salt	54.8
<i>Solid fuel thermal reactors</i>				
Fluoride high temperature reactor	FHR	University of California, Berkeley PB-AHTR (Fratoni et al., 2007)	Coated6particles	192.0
High temperature reactor	HTR	PBMR (Pty) Ltd. (IAEA, 2013)	Coated particles	96.8
Reaktor Bolshoi Moshnosti Kanalnyj	RBMK	NEA benchmark LEU-COMP-THERM-060 (OECD, 1995)	Oxide	13.7
Pressurized heavy water reactor	PHWR	CANDU benchmark (Pounders et al., 2011)	Oxide	32.1
High performance light water reactor	HPLWR	KIT Germany (Schulenberg, 2012)	Oxide	25.3
Light water reactor	LWR	VVER-1000 Tvel Russia (Emmett, 2000)	Oxide	41.1
<i>Liquid fuel thermal reactors</i>				
Thermal MSR: LiF-BeF ₂ -AcF ₄ salt	MSR-FLIBE	Graphite share 85% (Hombourger et al., 2016) and MSFR salt (Rosenthal et al., 1971)	Fluoride salt	41.1
Thermal MSR: LiF-AcF ₄ salt	MSR-FLI	Graphite share 85% (Hombourger et al., 2016) and MSFR salt (Heuer et al., 2014)	Fluoride salt	41.1
<i>Parametric study</i>				
Light water reactor	LWR	VVER-1000 Tvel Russia (Emmett, 2000)	Oxide	41.1
Cases: water density in % of nominal value	100, 95, 90, 85, 80, 75, 70, 65, 60, 55, 50, 45, 40, 35, 30, 25, 20, 15, 10, 5, 0			

**Fig. 3** Illustration of 16 simulated reactor lattices. From Krepel J and Losa E (2019) Closed U-Pu and Th-U cycle in sixteen selected reactors: Comparison of major equilibrium features. *Annals of Nuclear Energy* 128: 341–357.

Since the study is limited to lattice calculations, the multi-zone cores in this assessment are represented only by one zone. This is also the reason why only one representative zone was selected for HPLWR and why Boiling Water Reactor (BWR) was excluded. It has flooded part, which performs similarly as Pressurized Water Reactor (PWR) and voided part with much harder spectrum. Nonetheless, the parametric moderation study provides certain insight for possible BWR performance. The third most abundant reactor type, PHWR, was included in this study, but the Gas Cooled Heavy Water Reactor (GCHWR) was not. The graphite moderated light water cooled RBMK was included; however, the CO₂ cooled Advanced Gas cooled Reactor (AGR) was not included. Whereas GCHWR is not operated anymore, the AGR exclusion is not scientifically justified and it is arguably the major reactor spectrum type missing in this comparison.

Equilibrium neutron spectrum

Equilibrium neutron spectrum is one of the three characteristics of the Eigen-state (the solution of the Bateman equations). The spectrum and the equilibrium fuel composition mutually influence each other (see Fig. 1). The initial and equilibrium fuel compositions provide different spectra. For dedicated U-Pu breeders the initial fissile load consists of plutonium from irradiated LWR fuel and the difference is small. For dedicated Th-U breeders it is more complex, because the initial fissile fuel, plutonium or enriched uranium, is not extensively present in the thorium irradiation chain. The difference between initial and equilibrium spectrum is thus not negligible. Nevertheless, for the purposes of this study it is still quite small, because the spectrum is mostly shaped by scattering materials and major fertile nuclide. In reactors, initially using ^{235}U fissile fuel, the equilibrium spectrum differ. However, the differences are smaller than expected and it seems that the effect of scattering and capture cross-sections of the coolant and structural materials as well as of the fertile fuel play a dominant role in spectrum determination. Hence, the equilibrium neutron spectrum is discussed here as the first result, because it is rather spectrum which influences fuel composition than vice versa.

To understand the spectrum differences between the 16 selected reactor types and between the LWR parametric cases, each spectrum is graphically represented as a point in a 2D plot; where the x coordinate represents the relative share of thermal neutrons (0–1 eV) and the y coordinate the relative share of fast neutrons (0.1–20 MeV). The resulting plots for both the Th-U and U-Pu cycles are shown in Fig. 4. This figure is used for general spectrum discussion. For a more detailed discussion, refer to Krepel and Losa (2019).

All fast reactors included in this study have zero share of thermal neutrons in their spectra. Hence they are vertically aligned on a line $x = 0$ according to their share of fast neutrons. The MSFR with softest fast spectrum is the lowest point on this line. The softening is enhanced by beryllium presence. Surprisingly, the share of fast neutrons around 30–35% in MSFR spectrum is comparable to that in LWR. It seems that ^7Li and ^{19}F , even though not thermalizing extensively the neutrons, are slowing the neutrons out of the fast energy range due to their scattering cross-section resonances at relatively high energies. For the respective cross-sections and spectra, please refer to Fig. 2 from Krepel and Kramer (2021) and to Fig. 7 from Krepel and Losa (2019). Other fast reactors have at least 55% of fast neutrons in their spectra. The hardest spectrum with more than 70% of fast neutrons has MCFR based on the AcCl salt. There are no structural materials in this homogeneous reactor and the only non-fuel nuclide, ^{37}Cl , has quite a low scattering cross-section with narrow resonances and features a relatively high mass. Hence, it is not softening the spectrum like, for instance, ^{23}Na having lower mass and higher scattering cross-section with broader resonance. The three classical fast reactors, SFR, GFR, and LFR are located practically at the same spot. Of the fast reactors with solid fuel, only the MFBR has somewhat harder spectrum. Its fuel in metallic form, without light elements like O_2 in oxide fuel, poses higher actinides density; the heterogeneity effect is thus pronounced and Na scattering relatively weaker. The difference between the Th-U cycle and U-Pu cycle in this graphical representation is negligible for fast reactors. Only the MCFR core seems to have harder spectrum in the U-Pu cycle. It is probably related to the higher rate of direct ^{238}U fission, which reduces the overall absorption rate of actinides. For more detailed discussion, refer to Krepel and Losa (2019).

The thermal reactors included in this study are located in the bottom half of Fig. 4. They are scattered along the diagonal, because moderation simultaneously decreases the fast neutron share and increases the thermal neutron share. The thermal spectrum share ranges from 25% to 80%. The hypothetical diagonal line, connecting bottom left and upper right corners in Fig. 4, represents reactors with 20% share of neutrons in the intermediate energy range (1 eV–0.1 MeV). Close to this line are reactors with very hard spectrum as MCFR or with very soft spectrum as FHR and HTR. It seems that the majority of the neutrons in these reactors have one preferable energy range and the intermediate range is thus suppressed. Reactors using graphite, heavy water and light water as a moderator and light water as a coolant (RBMK, PHWR, LWR, and HPLWR) are almost equally distant from the diagonal

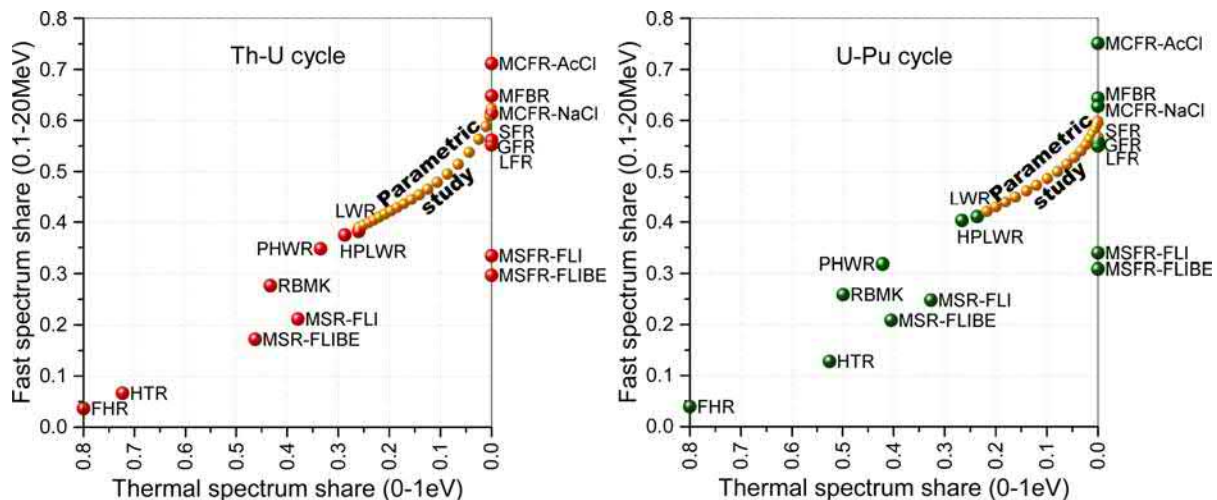


Fig. 4 Equilibrium spectra distribution in Th-U (left) and U-Pu (right) cycles.

line. The worst from this perspective are fluoride salt based MSR and MSFR. The high share of intermediate neutrons in MSR and MSFR can be caused by the above discussed features of ^7Li and ^{19}F . At the same time, it is codetermined by homogeneous dilution of actinides in the salt. The heterogeneity effect is thus strongly suppressed and fast neutrons fission is higher in LWR than in MSFR or MSR.

The FHR and HTR are the two reactors with the lowest specific fuel density and fuel-to-moderator ratio in this study. In addition, they are designed for uranium with enrichment between 10–20%; but the equilibrium fuel composition includes only 0.5% or 1.5% of fissile nuclides for the U-Pu or Th-U fuel cycle, respectively. The total cross-section of actinides is thus very low and more neutrons are in the thermal equilibrium with the moderator than in the slowing down process. The presence of cooling salt in FHR case introduces additional parasitic neutron captures. As a consequence, FHR is less sensitive than HTR to the remarkable difference of synthetic actinides share between the Th-U and U-Pu cycles. The FHR and MSR-FLIBE are composed from the same elements; however, they differ in salt volumetric share and actinides specific density.

The LWR parametric study practically connects the solid fuel fast reactors with the LWR point. Light water moderated reactors (LWR and HPLWR) have specific fuel density comparable to fast reactors, and the fast neutrons share in their spectra is slightly higher than the thermal neutrons share. In heavy water moderated and light water cooled PHWR the proportions are the same. The remaining reactors are moderated by graphite, and the thermal neutron share dominates. The position of these reactors in the chart depends on fuel-to-moderator ratio, where the lower specific fuel density results in softer spectrum. In the RBMK case it is partly biased by the light water coolant. In [Hombourger et al. \(2015\)](#) and [Hombourger \(2018\)](#) another parametric study was performed with EQLOD v3 for graphite moderated MSR operated in the Th-U closed fuel cycle. The results of this study practically linearly span from MSFR through MSR towards FHR. The highest reactivity in the moderated area was obtained for the 15% salt share in the core and 5 cm channel radius; the case which was adopted for the present study.

Equilibrium fuel composition

The second major feature of the equilibrium state is the fuel composition. It represents a state to which every non-equilibrium fuel composition evolves during irradiation. It also determines the waste composition from reprocessing losses. Furthermore, together with the equilibrium spectrum, it determines the equilibrium reactivity.

Sixteen reactors comparison study

The equilibrium fuel composition for each of the 16 selected reactors and for each fuel cycle is presented in [Fig. 5](#) using masses relative to the major fissile nuclides, i.e., ^{233}U or ^{239}Pu . Accordingly, these two nuclides are not plotted in [Fig. 5](#) and the related discussion refers only to other synthetic actinides. The results for both cycles in [Fig. 5](#) are further divided into thermal and fast reactors charts. This division is driven solely by the different scale needed for thermal reactors in the U-Pu cycle. Since the simulation assumes continuous refueling of fertile feed, all nuclide masses are constant. For each nuclide the creation and destruction rates are in balance. The reactor spectrum type determines the reaction probabilities and these probabilities dictate the equilibrium mass of each nuclide. In all fast reactor cases, the synthetic actinides share is below 1. The only exception is MSFR with the softest fast spectrum in the U-Pu cycle.

In the Th-U cycle the synthetic actinides share is close to 1 also for thermal reactors. This is actually the reason why the Th-U cycle can be considered for thermal breeder reactors. It is the consequence of ^{233}U low capture probability, which stays around 10% of total fission rate also in thermal spectrum. The respective ^{239}Pu capture probability in thermal spectrum is roughly 3x higher, refer to [Krepel \(2021a\)](#). Hence, also the creation rate of higher nuclides is 3x higher in the ^{238}U irradiation chain than in the ^{232}Th irradiation chain.

In the Th-U cycle, a slightly higher share of synthetic actinides can be seen for HTR and FHR. These two reactors have the highest specific power. Accordingly, rate of synthetic actinides creation is influenced by increased fraction of ^{233}Pa and its transmutation into ^{234}U . Since the ^{233}Pa amount is proportional to specific power, this effect can be seen in smaller scale also for MFBR, where the synthetic actinides share is one of the lowest between fast reactors in the U-Pu cycle, but one of the highest in the Th-U cycle. Obviously, ^{239}Np with 11x shorter half-life for radioactive decay, refer to Table 1 from [Krepel \(2021a\)](#), does not affect the ^{240}Pu mass in the same manner as ^{233}Pa affects the ^{234}U mass.

In the U-Pu cycle the relative synthetic actinides amount in thermal reactors ranges from 4 to 12. Due to the energy dependent ^{239}Pu capture probability, this fuel cycle is much more sensitive to spectrum variation. Accordingly, for majority of thermal reactors the equilibrium fuel composition alone (not accounting for parasitic absorptions by FPs, coolant and structural materials) has too high parasitic capture and results in negative reactivity.

The relative masses of other synthetic actinides in [Fig. 5](#) are useful for the neutron balance considerations ([Krepel and Losa, 2019](#)). However, for other fuel cycle features, as for instance the radiotoxicity due to reprocessing losses, the absolute share of main fissile nuclide and of other synthetic actinides are more relevant. According to [Fig. 6](#), the main fissile nuclide share of all actinides is $\sim 10\%$ in all fast reactors and in both fuel cycles. This ratio is determined by the respective capture and fission cross-sections in the fast neutron spectrum of the main fertile and fissile nuclides, respectively. The same ratio in thermal spectrum strongly differs between the fuel cycles. It is slightly less than 1.5% in Th-U cycle and 0.5% in the U-Pu cycle. The lower ratio in the U-Pu cycle is a consequence of higher ^{239}Pu fission cross-section and its resonance in the thermal energy range.

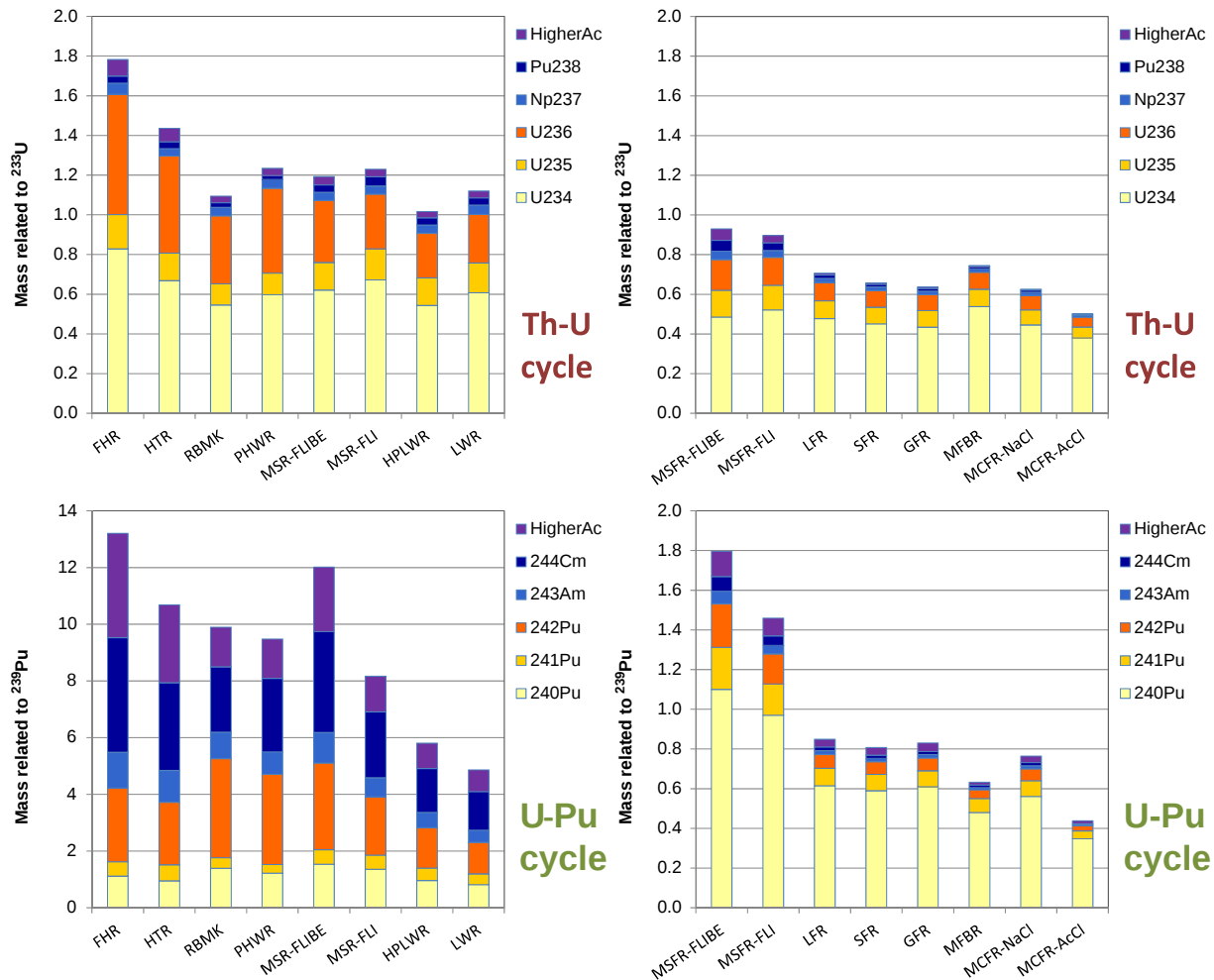


Fig. 5 Actinides mass related to the main fissile nuclide in Th-U (top) and U-Pu (bottom) cycles. From Krepel J and Losa E (2019) Closed U-Pu and Th-U cycle in sixteen selected reactors: Comparison of major equilibrium features. *Annals of Nuclear Energy* 128: 341–357.

The actinides parasitic neutron capture usually does not allow for breeding in thermal spectrum. Furthermore, these reactors are very sensitive to FPs accumulation (Krepel, 2021b). This is valid especially in the U-Pu fuel cycle, where the fissile actinides share is 3 times lower than in the Th-U fuel cycle. The share of fissile nuclides is given by the respective cross-sections, which are actually strongly co-determined by the neutron pairing effect. The ^{233}U and ^{239}Pu nuclei with odd neutrons are more eager to interact with neutrons (have bigger cross-sections and bigger fission chance) than ^{232}Th and ^{238}U nuclei with even neutrons (Krepel, 2021a).

Fig. 6 also shows that the absolute amount of other synthetic actinides is much lower in thermal spectrum, around 1.8% in the Th-U and 2.5% in the U-Pu fuel cycle. Only HTR and FHR, as the outliers, have somewhat higher synthetic actinides share, 2.5–3.5% in the Th-U cycle and 5.5% in the U-Pu cycle. For fast reactors, MSFR-FLI and MSFR-FLIBE are the outliers. The second named has the highest synthetic actinides share of 10% and 18% in the Th-U and U-Pu cycle, respectively. In other fast reactors this share is around 7% in the Th-U and 9% in the U-Pu cycle. Only the MCFR-AcCl spectrum is sufficiently hard so that synthetic actinides share of around 5% is the same in both fuel cycles. In all other cases the Th-U cycle feature a smaller synthetic actinides share than the U-Pu cycle. In general, fast reactors have 4x higher absolute share of synthetic actinides share than thermal reactors. Accordingly, should the same absolute share of legacy actinides be added to the equilibrium composition in all reactors, thermal cores will transmute it up to four times faster (Krepel and Losa, 2016). However, as Fig. 5 or (Krepel, 2021b) indicates, it will cost more neutrons in the thermal spectrum.

The synthetic actinides share in the fuel determines their amount in recycling losses and together with the half-lives also the radiotoxicity. The share of synthetic actinides in both fuel cycles is up to four times lower in thermal reactors, and in all reactors it is also smaller for the Th-U cycle. Furthermore, this cycle profits from the fact that the synthetic actinides have longer half-lives than the respective nuclides in the U-Pu cycle. Thus the radiotoxicity is typically lower at the early stage for the Th-U cycle. However, as a consequence of the longer half-lives, a secondary peak may occur for the Th-U cycle and the U-Pu cycle may provide lower radiotoxicity in the longer term (Krepel, 2021a).

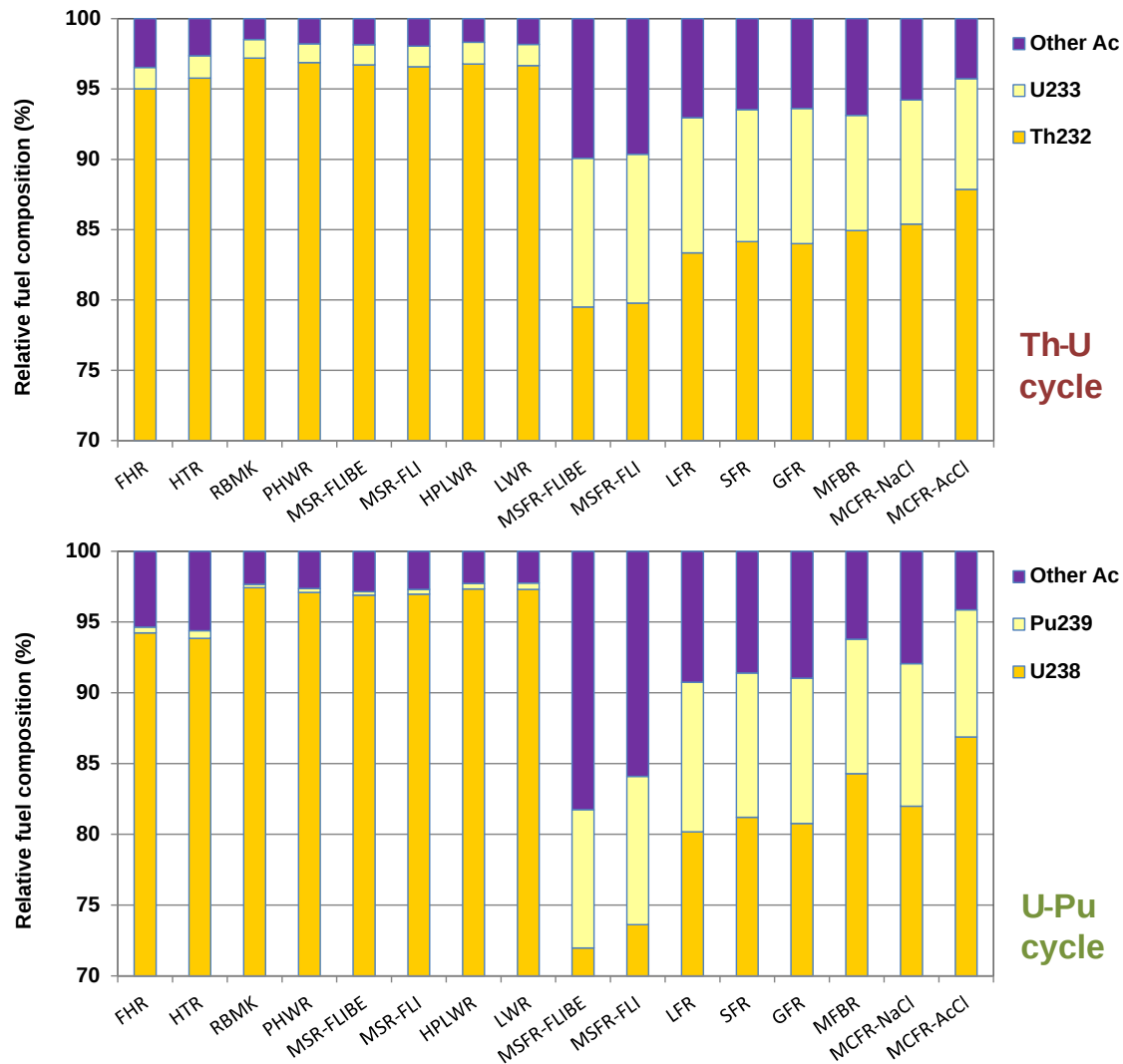


Fig. 6 Actinides relative composition in Th-U (top) and U-Pu (bottom) cycles.

LWR parametric moderation study

The high fission-to-capture probability of ^{233}U over a wide energy range in the Th-U cycle is well illustrated by the LWR moderation study. In this study, the water moderator is step-by-step removed from the reactor. The synthetic actinides share related to the ^{233}U mass stays constant at 1.1 level (see top Fig. 7) up to about 30% moderator density. From about 20% moderation, with water being almost completely removed, and the neutron spectrum getting very hard, the synthetic actinides share related to ^{233}U drops down to 0.6. In the U-Pu cycle, the variable ^{239}Pu fission-to-capture probability causes gradual drop of the ratio between synthetic actinides and the main fissile nuclide ^{239}Pu from 5 at full moderation to 1.1 at 50% moderation. Then there is a plateau, or even slight increase, and only the unmoderated case (0% water density) drops below this ratio being equal to unity (see bottom Fig. 7).

The relative masses in Fig. 7 mirror the fission-to-capture probability of ^{233}U and ^{239}Pu . The absolute shares of main fissile nuclides and of other synthetic actinides, presented in Fig. 8, are rather determined by the respective capture and fission cross-sections of the main fertile and fissile nuclides. The variation of fissile share with moderation has a similar tendency for both the Th-U and U-Pu fuel cycles, but it is not the same. The share of main fissile nuclide is $\sim 10\%$ in fast spectrum (0% moderation) for both cycles. It drops to 3% at 25–30% moderator density in the Th-U cycle, and at 40–45% moderator density in the U-Pu cycle. At 100% moderation the share is around 1.5% in Th-U cycle and 0.5% in the U-Pu cycle. In the thermal spectrum (70–100% moderation), the total amount of synthetic actinides is the same for both cycles, or even slightly lower for the U-Pu cycle. However, the split between the main fissile and the other synthetic actinides is different in the two fuel cycles.

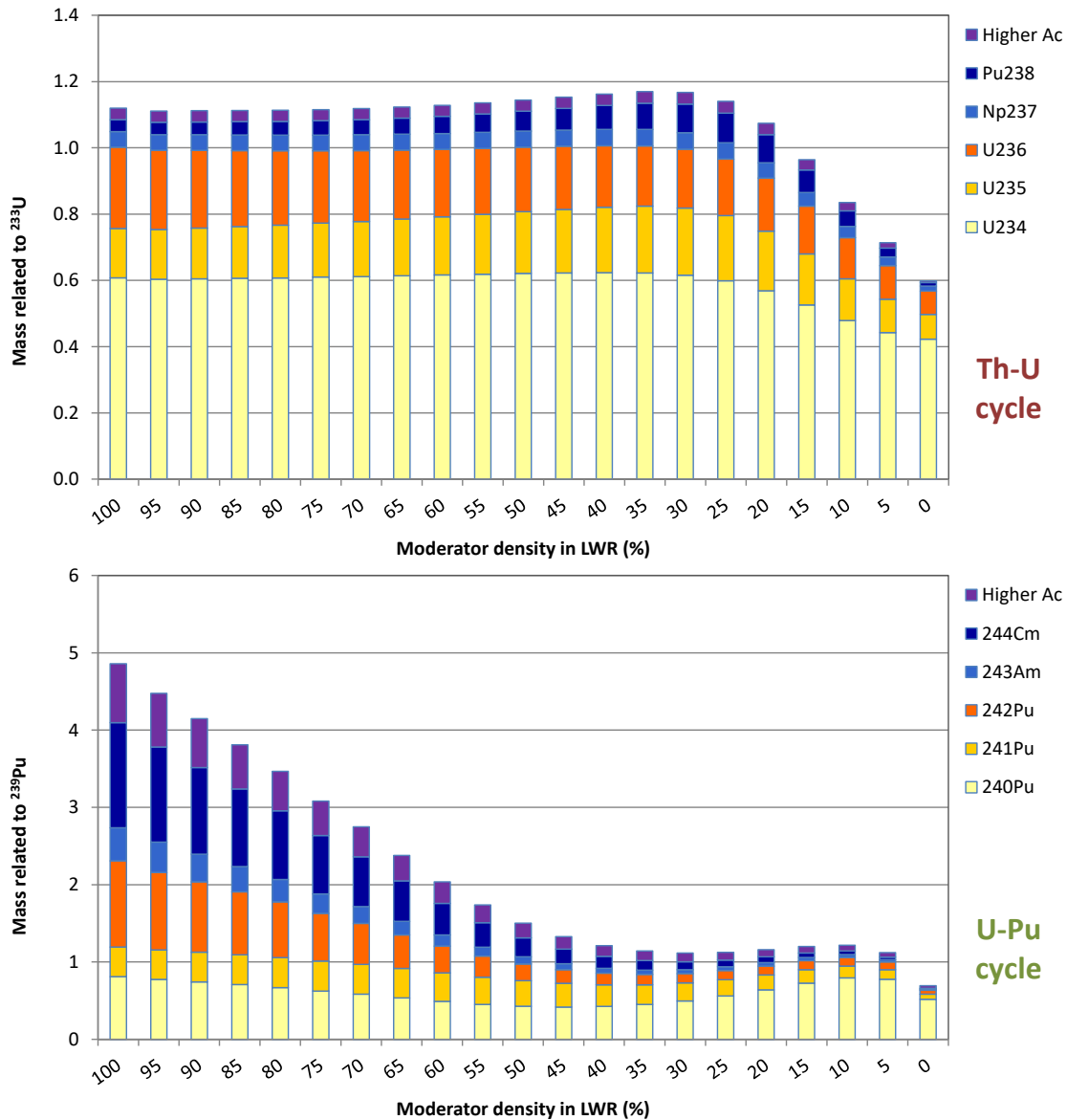


Fig. 7 Actinides mass related to the main fissile nuclide in Th-U (top) and U-Pu (bottom) cycles for the LWR parametric moderation study.

Equilibrium multiplication factor

The equilibrium multiplication factor is the major indicator for the neutron economy. Since the fuel composition is stabilized and does not evolve during irradiation, all simulated reactors act as self-sustaining breeders. Nonetheless, not all reactor can reach criticality with the equilibrium fuel composition. Accordingly, the respective equilibrium reactivity is indication of the capability of the reactor to be operated as a self-sustaining breeder. In [Krepel \(2021b\)](#) the reactors are classified based on this reactivity.

The infinite multiplication factor for the 16 simulated reactors follows the expected trend. Since the reactors are roughly sorted in [Fig. 9](#) by their spectrum, for both fuel cycles, there is an overall increase in slope from left to right as the reactivity grows with spectrum hardening. The increase is higher for the U-Pu cycle, due to its higher sensitivity to spectrum variation and its effect on synthetic actinides build up (see [Figs. 5 and 7](#)). The trend in [Fig. 9](#) is, however, perturbed by varying parasitic neutron captures of coolants and structural materials.

As expected, practically all thermal reactors do not have sufficient equilibrium reactivity to be operated as self-sustaining breeders. For the Th-U cycle, in which the synthetic actinides mass stays low even in thermal spectra, there are few thermal reactors with positive or close to zero reactivity. These reactors are moderated by low capture moderators like heavy water or graphite. However, it should be recalled that the values of neutron multiplication factors in [Fig. 9](#) were derived from infinite lattice calculations and with the neglect of FPs. Accordingly, the excess reactivity will be lower in reality. Potentially, also beryllium could act as

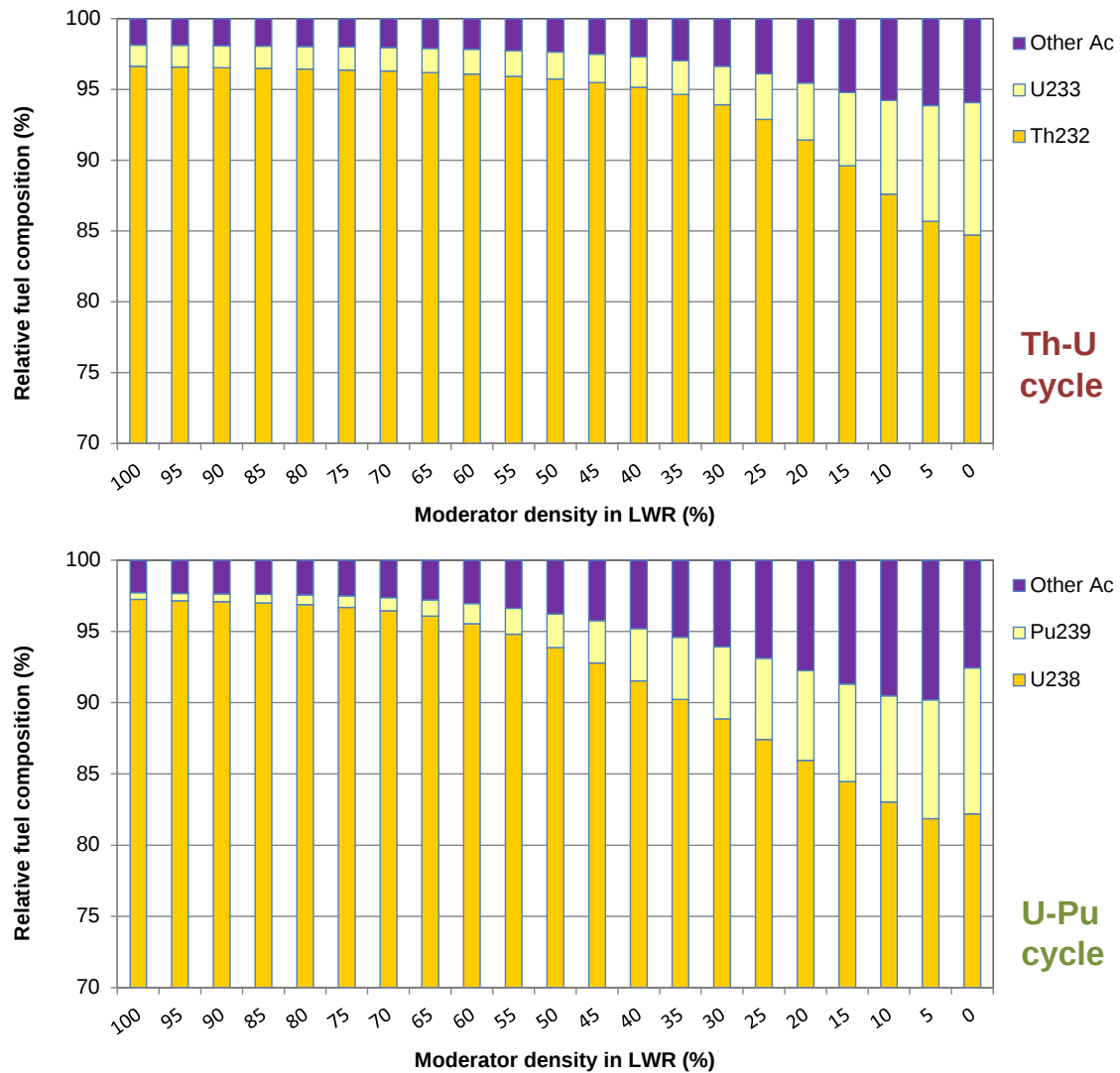


Fig. 8 Actinides relative composition in Th-U (top) and U-Pu (bottom) cycles for the LWR parametric moderation study.

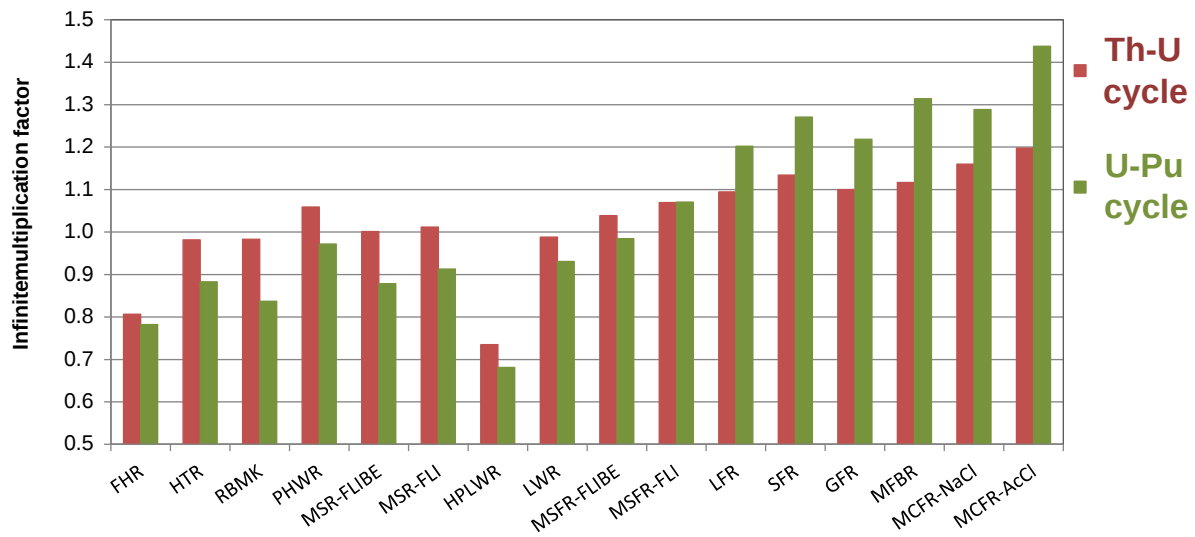


Fig. 9 Equilibrium multiplication factor for Th-U and U-Pu cycles and 16 selected reactors.

low absorbing moderator, e.g., for MSR. Nonetheless, both heavy water and beryllium needs to be separated from the MSR fuel by structural materials, which may deteriorate the performance (Hombourger, 2018). Furthermore, even if breeding would be possible in thermal reactors, their fissile share (see Fig. 6) is small. Hence, they will be very sensitive to FPs buildup (Krepel, 2021b). It may result in prohibitively high fuel reprocessing frequency. The only reasonable option could be the MSR, in which the FPs can be continuously extracted from the liquid fuel. However, even in this case the need of fast FPs removal may prove being prohibitive.

In the thermal spectrum cases, there are two outlier systems, FHR and HPLWR, with very low multiplication factors. In HPLWR case it is caused by parasitic neutron capture of the stainless steel cladding, which is not suitable for thermal spectra. In FHR it is due to the interplay between the parasitic neutron capture of the fluoride salt coolant, its higher volumetric share in the core, and the very low specific fuel density. The same salt in MSR-FLIBE, with higher specific fuel density and lower salt volumetric share, provides reasonable multiplication factors.

The Th-U cycle performs better than the U-Pu cycle in thermal, and worse in fast neutron spectra. The turning point is the MSFR-FLI, where the performance is the same in both cycles. It features a spectrum, in which the lower average number of neutrons per fission and the lower synthetic actinides concentration in the Th-U cycle provide similar reactivity as the higher average number of neutrons per fission and the higher synthetic actinides concentration in the U-Pu cycle. The detailed decomposition and analysis of the Fig. 9 results is presented in Krepel and Losa (2019).

The LWR parametric study provides insight for the equilibrium reactivity transition between thermal and fast reactors. Fig. 10 confirms that the reactivity changes with spectrum variation are stronger for the U-Pu cycle. At the same time the Th-U cycle is very close to $k = 1$ even for nominal water density and become critical already around 80% of the moderator density. However, for water moderator densities below 30%, the reactivity increase in the U-Pu cycle is steeper, yielding reasonable high reactivity earlier than in the Th-U cycle. Fig. 10 also illustrates the option to breed in reactor cores cooled by steam or in the upper part of BWR cores in which the exit water density is around 5% of the nominal LWR water density. It may be possible to design a water-cooled breeder (Huang and Zang, 2021) using water densities between 35% and 5% of the nominal LWR value. Optionally, a breeder with water density below 5% can be proposed solely for the heat production. Since the required steam pressure and accordingly the temperature would be lower, also the heat exchange and maximal power density would be limited.

Core size estimate

By means of Fermi's theory of the bare thermal reactor the critical core size can be estimated from the knowledge of equilibrium multiplication factor and material properties of each reactor (Losa, 2016; Krepel and Losa, 2017). The latter are a measure of how transparent the reactor core is for the neutrons. Application of this theory provides only very rough estimate, because the impact of neutron reflector or blanket is not accounted for. At the same time, it is based on an overestimated equilibrium multiplication factor, with neglected FPs parasitic absorption. Accordingly, it can provide certain insights for the interplay between equilibrium reactivity and reactor transparency for the neutrons.

The estimated core radii of bare self-sustaining breeders are presented in Fig. 11. The graphite moderated MSRs have quite substantial core radius. To obtain realistic core size, blanket and/or reflector should surround the core and the FPs must be continuously removed from the core. Furthermore, the neutron balance should be improved by ^{233}Pa extraction from the molten salt. Its radioactive decay outside of the neutron flux will increase the ^{233}U and reduce the ^{234}U production rates (Krepel, 2021a).

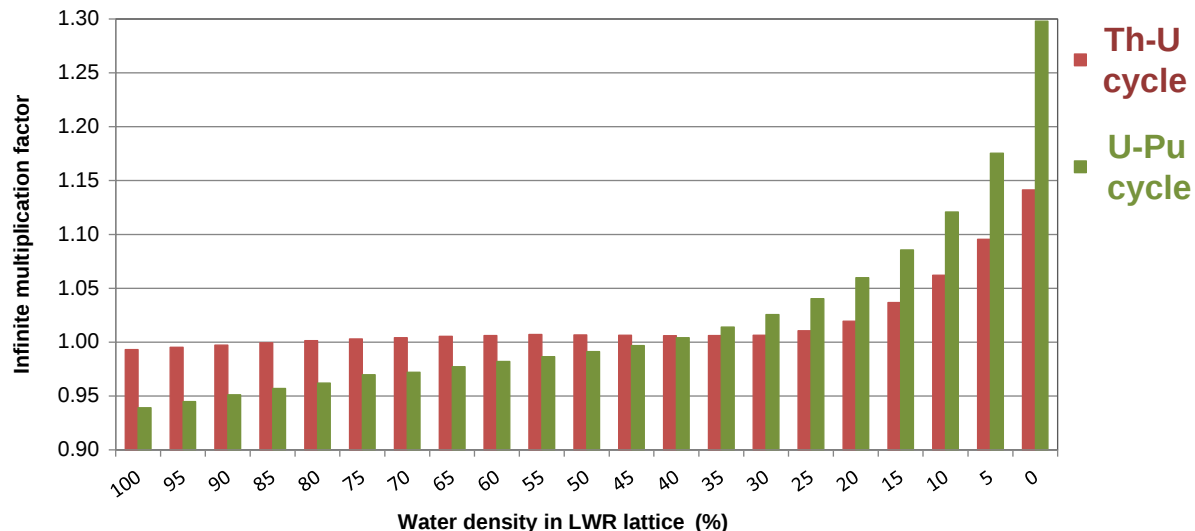


Fig. 10 Equilibrium multiplication factor for Th-U and U-Pu cycles and the LWR parametric moderation study.

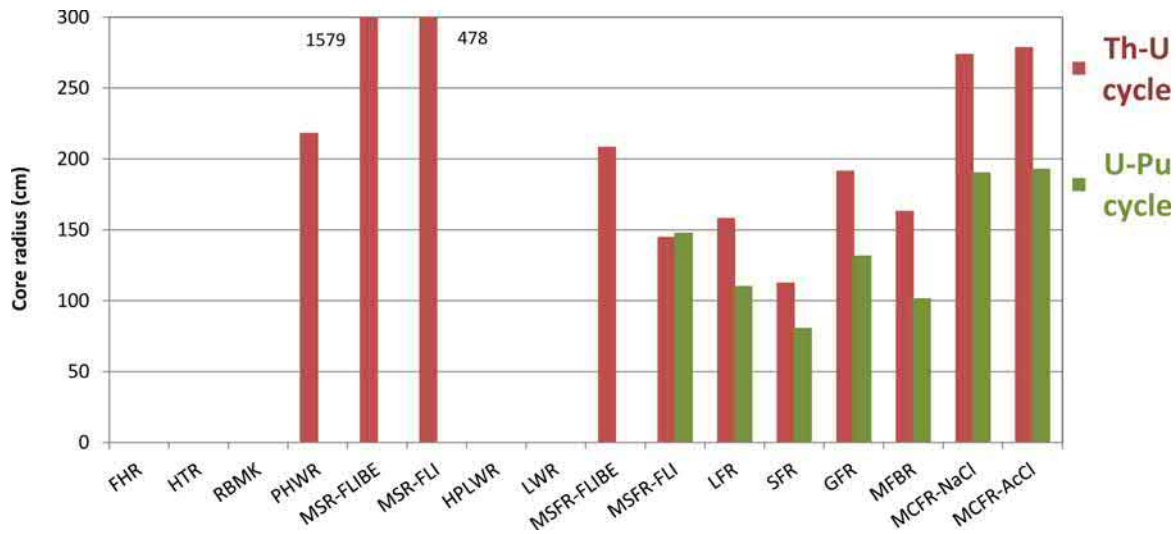


Fig. 11 Estimate of bare critical core radius for Th-U and U-Pu isobreeding fuel cycles.

Optionally, some of the synthetic actinides (e.g., ^{237}Np) may not be recycled to decrease their contribution to parasitic neutron captures. The PHWR displays a smaller core radius than the MSR, but it has a solid fuel. Continuous FPs separation and ^{233}Pa removal is thus not possible. The fuel reprocessing frequency would need to be quite high to minimize the FPs captures and achieve breeding.

The SFR seems to provide the best combination of equilibrium multiplication factor and core material properties for a compact breeder. Especially in the U-Pu cycle the core is quite small. On the other hand, the MCFR represents an example of imbalance between the parameters. It has the highest equilibrium multiplication factor (see Fig. 9) and (together with the HTR and FHR) is the most transparent core for the neutrons. As a result, the core radius of the self-sustaining MCFR breeder is the largest of the fast reactors.

The estimate of minimum core radius for the LWR parametric study shows an interesting trend. For the Th-U cycle there are two minima, one at 55% moderation and the second at 0% moderation. The equilibrium multiplication factor between 100% and 30% moderation in Fig. 10 is almost constant, with very tiny local maxima at 55% moderation; nonetheless, the core radius is very sensitive to the moderation values (Fig. 12). The second minimum at 0% is due to the steep increase in multiplication factor as water density decreases below $\sim 20\%$. However, the core radius at 55% moderation is still too big for practical application. Interesting cases with smaller core radius < 200 cm are obtained for the U-Pu cycle in the range of 20–0% moderation and for the Th-U cycle in the range of 5–0% moderation (which is, probably, not practical from the heat removal perspective).

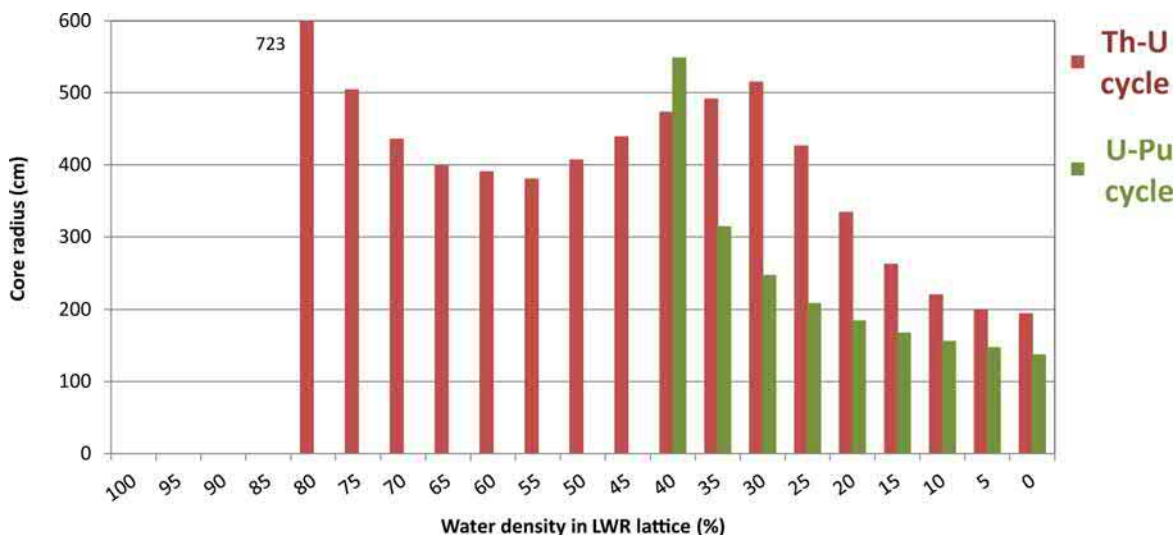


Fig. 12 Estimate of bare core radius for Th-U and U-Pu isobreeding fuel cycles in the LWR moderation parametric study.

Equilibrium temperature coefficients

The equilibrium state can be evaluated also from a safety perspective. In this article, only the fuel and coolant temperature coefficients of reactivity have been evaluated. To obtain these coefficients, the respective material properties are perturbed as follows: temperature of the fuel is increased by 300 °C at constant density, and density of the coolant is reduced by 10% at constant temperature. This approach allows for separation of the temperature and density effects for cases with fluid fuel.

Fluid density change

In all cores, the fluid density reduction results in decrease of the fluid capture and scattering probabilities. Compared to other core materials, its relative importance is thus reduced. The only exceptions are the homogeneous MSFR and MCFR, where the core is exclusively filled by the fluid (that functions as both fuel and coolant). In this case, the fluid density change represents a global density reduction, and since an infinite lattice is simulated, it has no impact on the reactivity (results absent in Fig. 13). In reality fluid density reduction will increase neutron leakage from a finite core and provide a negative reactivity effect.

The reduced capture probability results always in positive reactivity. The reduced scattering probability induces spectrum hardening that can introduce positive or negative reactivity. In fast reactors it introduces a positive reactivity (Krepel et al., 2011). It reduces the neutron capture rate in the resonance area and it increases the fission rate above the resonance area. The induced positive reactivity is higher in the U-Pu cycle (see LFR, SFR, GFR, and MFBR in Fig. 12), because the slope of the respective fission cross-section is steeper with increasing neutron energy as shown by bottom Fig. 17 from Krepel and Losa (2019) and because the resonances of the capture cross-section start at higher energies in the U-Pu cycle (Shusterman, 2021). Consequently, the positive coolant density effect for fast reactors is stronger in the U-Pu cycle (Fiorina et al., 2013b).

In thermal reactors, where the fluid also moderates the neutrons, the reduced scattering probability introduces negative reactivity. Interplay between this negative reactivity and positive reactivity introduced by capture probability reduction is crucial for the safety of moderated reactors. The overall effect should be negative. Hence, moderated cores are usually designed to be under-moderated, so that the moderator density reduction results in negative reactivity as for instance in PHWR and LWR. In many thermal reactors: FHR, HTR, RBMK, PHWR and MSR, dedicated coolant fluid is used, which has only minor moderation function. Its density effect is thus dominated by the capture component and results in positive reactivity. For PHWR coolant the capture component only slightly dominates above the scattering component and the effect is slightly positive. Nonetheless, in reality the increased leakage rate will somewhat compensate for the reduced capture rate. In MSR the fluid includes fuel and its density reduction reduces the fuel-to-moderator ratio (Krepel et al., 2014). In some cores there are two fluids with separate moderation and cooling functions. For such cores two independent results are presented in Fig. 13.

In the HPLWR the results differ qualitatively between the Th-U and U-Pu cycles. However, a big difference between the fuel cycles can be seen also for FHR, PHWR, MSR-FLI and LWR. In the U-Pu cycle, ^{239}Pu fission cross-section has a large resonance at 0.3 eV. A similar but weaker resonance of ^{233}U fission cross-section in the Th-U cycle is located at 1.8 eV, as indicated by top Fig. 17 from Krepel and Losa (2019) and Shusterman (2021). Nonetheless, the ^{239}Pu resonance has higher amplitude, it is broader, preceded by cross-section suppression, and followed by larger cross-section than in the ^{233}U case. Accordingly, in the U-Pu cycle the negative moderation effect is stronger. This can be seen particularly by HPLWR, where the moderator is located in a dedicated central tube (see Fig. 3) and its density effect is dominated by capture rate in the Th-U cycle and the moderation effect in the U-Pu cycle. In

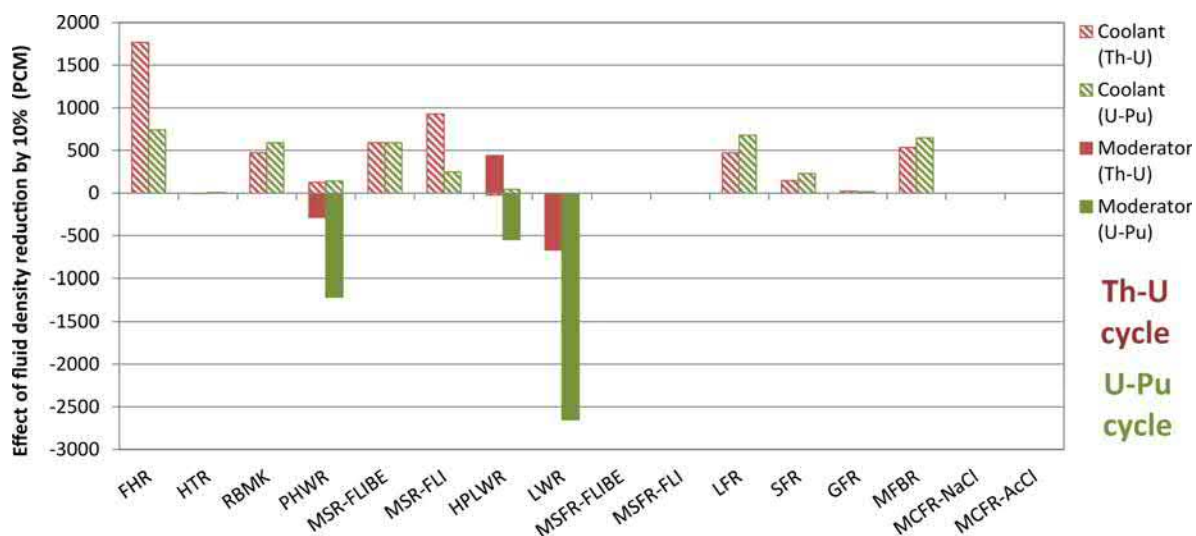


Fig. 13 Reactivity induced by 10% reduction of the fluid density for Th-U and U-Pu cycles. In PHWR and HPLWR cases the moderator and coolant densities were reduced independently. From Krepel J and Losa E (2019) Closed U-Pu and Th-U cycle in sixteen selected reactors: Comparison of major equilibrium features. *Annals of Nuclear Energy* 128: 341–357.

PHWR the coolant and moderator density effects have opposite sign. It seems that the reactor is under-moderated from moderator perspective and over moderated from coolant perspective.

The results for LWR moderation study shown in Fig. 14 combine all above discussed effects. At 0% moderation there is no fluid in the core and the respective fluid density effect is zero. In thermal spectrum the negative reactivity effect is much stronger in the U-Pu cycle. This effect is primarily dictated by the ^{239}Pu fission cross-section characteristics that is also responsible for the relatively small ^{239}Pu fraction in Fig. 8. To a certain extend there is a correlation in the thermal region (100–65% of nominal water density)—the bigger the fissile share presented in Fig. 8, the smaller is the fluid density effect in Fig. 14. Higher fissile share implies a lower fraction of the fissions in the thermal energy range. It becomes unimportant at moderation levels corresponding to 45–5% of nominal water density. In this region the U-Pu cycle features less negative and eventually positive coolant density effect, because of the enhanced fission probability induced by spectrum hardening.

Doppler effect

The fuel temperature increase results in so called Doppler effect, or actually broadening of cross-section resonances. Since the cross-section resonances of the fertile fuel dominate neutron capture, the Doppler effect always results in negative reactivity. The only exception, PHWR in the U-Pu cycle, will be discussed separately. Since the respective mechanisms differ between fast and thermal reactors, the absolute reactivity change is presented in Fig. 15.

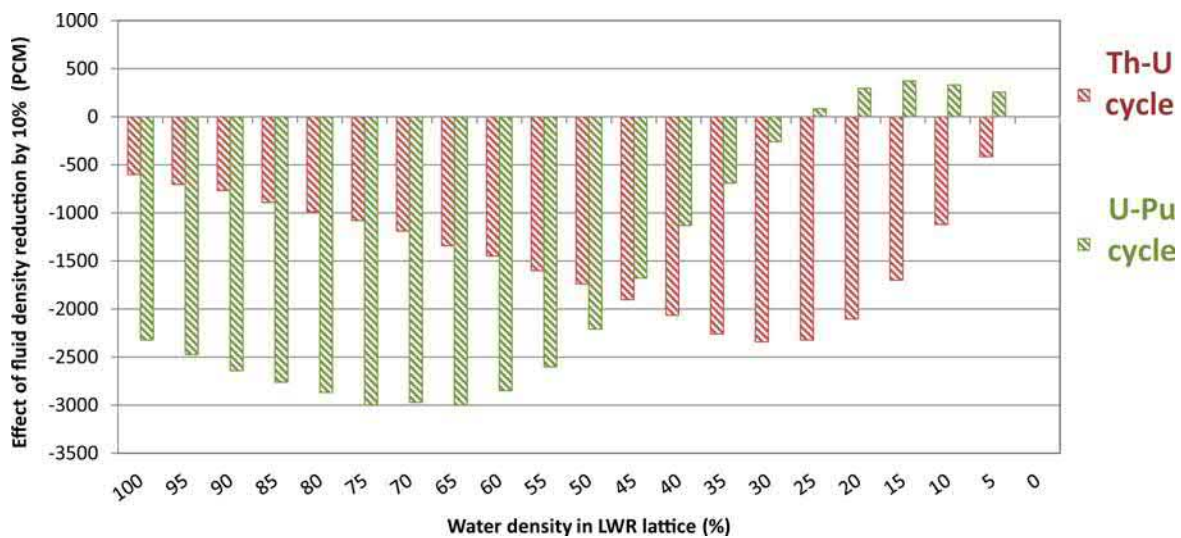


Fig. 14 Reactivity induced by 10% reduction of the water density in the LWR parametric moderation study for Th-U and U-Pu cycles.

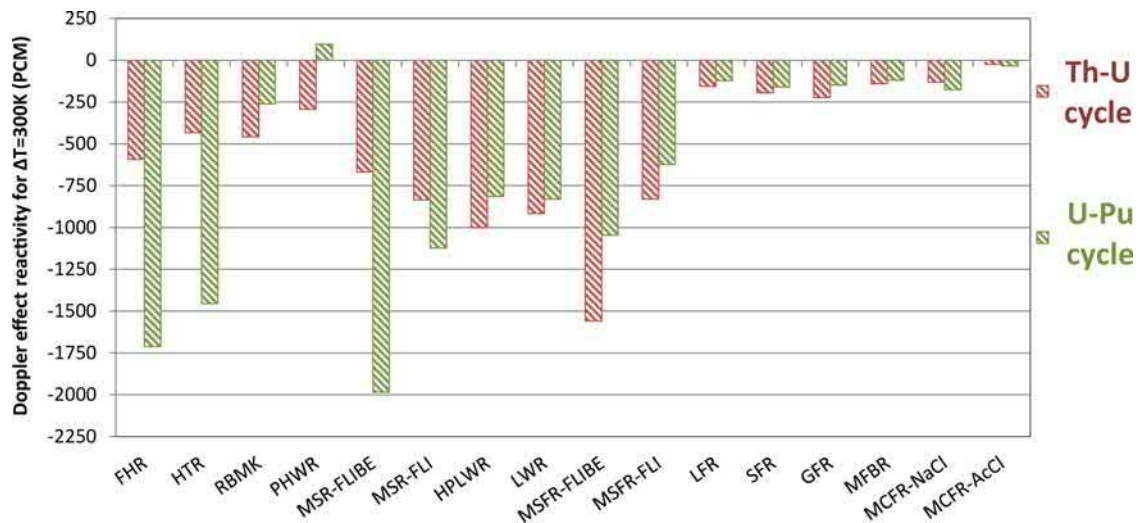


Fig. 15 Doppler reactivity effect induced by 300 °C fuel heat-up in Th-U and U-Pu cycles. From Krepel J and Losa E (2019) Closed U-Pu and Th-U cycle in sixteen selected reactors: Comparison of major equilibrium features. *Annals of Nuclear Energy* 128: 341–357.

In fast reactors only the low-energy tail of the neutron spectrum is in the resonance area. Moreover, the ^{232}Th resonances start later at lower energies than for ^{238}U . The Doppler effect in fast reactors is, in simplified way, composed of negative reactivity caused by the enhanced capture in resonances and of positive reactivity from spectrum hardening induced by this enhancement (Krepel et al., 2011). Since the U-Pu cycle profits more from spectrum hardening, its resulting Doppler effect is less negative in fast reactors. The only exception is the MCFR core which features the hardest spectrum. In this reactor there are almost no neutrons in the energy range of the ^{232}Th resonances so the respective Doppler effect is smaller than in the U-Pu fueled reactors.

In thermal reactors the Doppler effect reduces the neutrons resonance escape probability during the slowing-down process resulting in a reduction of thermal spectrum component. Both the increased resonance capture rate and the suppression of thermal spectrum result in negative reactivity. The magnitude of the Doppler effect in thermal reactors depends on the moderator properties and on the core heterogeneity. The latter effect is strong in all water moderated or cooled reactors including PHWR and RBMK. In these reactors the Doppler effect has medium amplitude and is stronger for the Th-U cycle. In reactors with very weak heterogeneity (FHR, HTR and MSR) the resonance escape probability for the neutrons during the slowing-down process plays much more important role and the respective Doppler effect is very strong especially for the U-Pu cycle. Nonetheless, it is rather a complex phenomenon which is also effected by the concentration of other synthetic actinides in the core.

The U-Pu cycle in PHWR core is an exception. It features a positive Doppler effect because the dedicated heavy-water moderator does not have cooling function and its temperature is kept low. The respective Maxwellian spectrum peaks at the lowest energy, see Fig. 7 from Krepel and Losa (2019). The large resonance of ^{239}Pu fission cross-section is close to this peak. At the same time, the heterogeneity effect is very strong in PHWR and the resonance broadening in epithermal area has thus a lower impact on neutron capture. As a result, the fission resonance broadening dominates in this core.

The variation of the Doppler effect with the water density for LWR parametric study is shown in Fig. 16. The dominant effect is resonance capture probability that peaks at intermediate water densities which feature epithermal neutron spectrum. For very hard spectrum, the Doppler effect is the weakest due to very small resonance capture probability. It is the most negative for the Th-U cycle around 25% water density because spectrum hardening has a smaller positive reactivity effect than for the U-Pu fuel cycle. For the Th-U cycle it is the most negative around 60–75% water density because of the ^{239}Pu fission cross-section characteristics.

Summary

To fully utilize natural resources in nuclear fission reactors, self-sustaining breeding in closed cycle is necessary. In this cycle the synthetic fissile nuclides ^{233}U and ^{239}Pu acts as kind of catalyzer for energy production from primordial ^{232}Th and ^{238}U nuclides. The repetitive fuel irradiation in a closed cycle results in the equilibrium fuel composition. When only the ^{232}Th or ^{238}U primordial nuclides are used as a feed fuel, the respective equilibrium fuel composition corresponds to the ^{232}Th or ^{238}U irradiation chain. In this article the basic features of the ^{232}Th and ^{238}U irradiation chains were compared for 16 selected reactors and certain insight was provided, why some of the advanced reactors are not capable to sustain the breeding process.

EQLOD v2 MATLAB script coupled to the SERPENT 2 code was used to obtain the equilibrium state by iterative procedure on an infinite lattice. The specific power was fixed at nominal value, and the irradiation of the actinides was independent from the multiplication factor. Hence, the equilibrium state was obtained also for subcritical reactors. Since the fuel composition slightly oscillates during different phases of the closed fuel cycle, one simplifying assumption was applied to obtain the constant equilibrium vector. The FPs were neglected and each fission directly resulted in addition of ^{232}Th or ^{238}U atom. With this assumption the results become

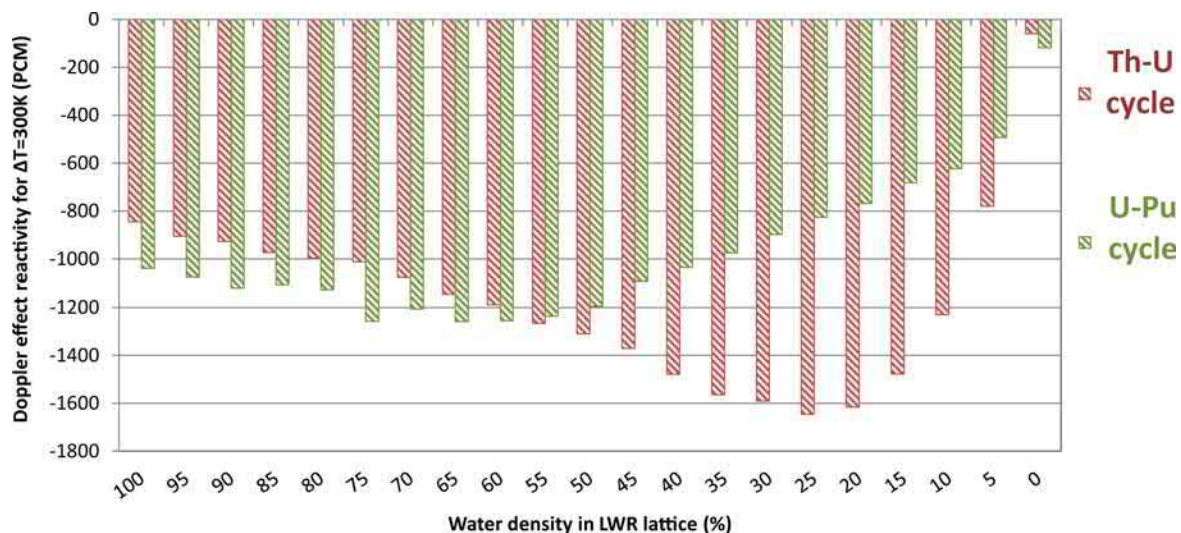


Fig. 16 Doppler reactivity effect induced by 300 °C fuel heat-up in the LWR parametric moderation study for Th-U and U-Pu cycles.

insensitive to the discharged fuel cooling time, irradiation time, reprocessing intensity and reprocessing losses. Hence, the equilibrium state depends only on the reactor specific power and core lattice geometry, composition, and temperature. In spite of the simplifications, the obtained equilibrium reactivities and actinides eigenvectors are representative for the respective reactor and many useful conclusions can be drawn.

Three major equilibrium properties were compared: excess reactivity, fuel composition, and neutron spectrum. The reactor types included in this study were selected so that all major spectrum types are covered. A LWR parametric moderation study was included to illustrate the evolution of equilibrium parameters between thermal and fast spectra. Even though the majority of the presented results is well known, this article offers a slightly different perspective for their interpretation. Moreover, the LWR parametric study provides valuable insights into the evolution of the equilibrium properties between thermal and fast reactors. Self-sustaining breeding is generally possible in fast reactors. In thermal reactors only the Th-U cycle provides positive reactivity, which, however, may not be sufficient.

The assessment of critical core radii, included in this article, is only indicative, because the FPs as well as blankets or reflectors are neglected. At the same time, this assessment nicely illustrates that self-sustaining breeding in closed cycle is nearly impossible in thermal reactors. Their neutron economy is so tight, that they can afford only negligible leakage. The resulting cores are thus very large. Furthermore, to minimize neutron losses due to FPs, prohibitively high fuel reprocessing frequency may be needed.

Among the fast reactors, the SFR seems to provide the best combination of equilibrium multiplication factor and core material properties for compact breeder. Especially in the U-Pu cycle the core is quite small. In contrary, MCFR represents an example of imbalance between the parameters. It has the highest equilibrium multiplication factor, on the one hand side, and it is the most transparent core for the neutrons, on the other. As a result, the core radius of self-sustaining MCFR breeder is the largest of the fast reactors.

Two safety related parameters, i.e., the fluid density and the fuel Doppler reactivity effects, have been evaluated. Both these effects compose of two phenomena. The fluid density effect consists of reduced fluid absorption and of spectral hardening caused by the reduced neutron scattering. The Doppler effect consists of increased absorption in resonances and of the resulting spectral shift. These effects have different amplitude and direction in different reactors.

The general conclusion of this article is that self-sustaining breeding in closed cycle can be achieved with both primordial nuclides ^{232}Th and ^{238}U practically in all fast reactors. The U-Pu cycle profits more from neutron spectrum hardening and outperforms the Th-U cycle in fast spectra. The self-sustaining breeding in closed cycle is nearly impossible in thermal reactors. Due to the lower ^{239}Pu fission probability in thermal reactors, the creation rate of higher synthetic actinides is roughly 3 times higher in the U-Pu cycle and the equilibrium fuel composition alone has prohibitively high parasitic capture. The respective fission probability of ^{233}U in the Th-U cycle is high in all spectra. It may allow for self-sustaining breeding in closed cycle in some particular thermal reactors.

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Self-Sustaining Breeding in Advanced Reactors: Comparison of Fuel Cycle Performance

Jiri Krepel, Paul Scherrer Institut, Villigen, Switzerland

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Glossary

Ac Actinides, elements with atomic numbers 89 to 103.

AcCl Label for tetra-chloride salt of actinides AcCl_4 .

Actinides Eigen-composition Stabilized actinides composition, an equilibrium composition which results from a long irradiation of one parent nuclide or their mixture. Necessary condition for the stabilization is that the mass of parent nuclide/s and the irradiation flux do not vary strongly. It is specific and inherent feature for every reactor and every parent nuclide/s and it represents the Eigen-vector of the respective Bateman equation.

B&B Breed-and-Burn; a mode of open cycle operation, where the fissile excess bred during irradiation is equal or higher than the discharged fissile mass after the irradiation. The reactor burns its own bred fuel. The discharged fuel does not need to be recycled and the fresh fuel does not need to contain primordial or synthetic fissile nuclides.

BG Breeding gain, $\text{BG} = \text{BR} - 1$.

BR Breeding ratio, it is ratio between breeding rate and fission rate. In a self-sustaining breeder $\text{BR} \geq 1$.

Breeding Transmutation of actinide nuclide initiated by neutron capture, which increases fission probability. Typically, primordial non-fissile (fertile) nuclides ^{232}Th and ^{238}U are transmuted into synthetic (man-made) fissile ^{233}U and ^{239}Pu .

Burnup Share of already fissioned actinides. It can be expressed in FIMA % (Fissions per Initial Metal Atom) or as energy released from given mass of actinides in MWd/kg (Megawatt days per kilogram). Since the energy per fission is approximately the same for all actinide nuclides, values expressed in these two units are proportional and roughly 10 times higher for the second unit.

Closed cycle A chain of process steps which is closed for the respective working medium. The medium, in nuclear fuel cycle the actinides, is recycled and does not extensively leave the chain.

Cross-section A measure of the probability that neutron will interact when flying by with the respective nucleus (microscopic cross-section) or when fling through with the respective nuclei concentration (macroscopic cross-section).

Equilibrium reactivity Reactivity provided by actinides Eigen-composition.

FPs Fission products, typically two fragments of actinides fission.

FLI Label for eutectic mixture of lithium fluoride salt and actinides tetra-fluoride salt LiF-AcF_4 .

FLIBE Label for eutectic mixture of lithium fluoride and beryllium di-fluoride with addition of actinides tetra-fluoride salt $\text{LiF-BeF}_2\text{-AcF}_4$.

Irradiation chain Series of consecutive transmutations and of the respective daughter products induced by neutron irradiation of one parent nuclide, e.g., ^{232}Th or ^{238}U . The radioactive decays of the daughter products are part of the chain.

MA Minor actinides, synthetic actinides originated by uranium transmutation (Np, Am, Cm and higher elements) without Pu, which is some time called major actinide.

NaCl Label for eutectic mixture of sodium chloride salt and actinides tetra-chloride salt NaCl-AcCl_4 .

Neutron economy Balance between neutron generation and neutron losses in a reactor. Good neutron economy has high neutron generation and small neutron losses by parasitic absorption and leakage from the reactor.

Open cycle A chain of process steps which is open for the respective working medium. The medium, in nuclear fuel cycle the actinides, is not recycled and leaves extensively the chain.

Primordial actinides Actinide nuclides, presumably originated by rapid neutron capture process during a supernova explosion, which have long enough half-life for radioactive decay to be still present on the earth.

Reactivity Reactivity is related to the effective (k_{eff}) or infinite (k_{inf}) neutron multiplication factor. Reactivity denotes the deviation from the self-sustaining fission chain reaction ($k = 1$), for which it is equal to zero. Reactor with negative reactivity ($k < 1$) cannot sustain the fission chain reaction. Positive reactivity ($k > 1$) means that there are more neutrons than needed for the self-sustaining fission chain reaction. From operational perspective, neutron absorber should compensate positive reactivity to control the chain reaction. From fuel cycle perspective, positive reactivity indicate that there are excess neutrons available for additional utilization by breeding or transmutations.

Self-sustaining breeding Breeding process where breeding rate $\geq$ fission rate, the mass of fissile synthetic nuclides is conserved in the reactor, and fission chain reaction is sustained solely by these synthetic nuclides. A long time-horizon is being considered in this article.

Subcritical/critical core A core that cannot/can sustain a fission chain-reaction.

Synthetic actinides Actinide nuclides originated, typically by neutron-induced transmutation, from primordial actinides.

Synthetic nuclides are, by definition, a much broader group than minor actinides (MA) or trans-uranic elements. For thorium and uranium elements, both primordial and synthetic isotopes exist.

Waste Waste is unwanted side product, which is further unusable. In nuclear fuel cycle, synthetic actinides represent a side product, which is reusable. However, they can be declared as a waste by nuclear fuel cycle policy or when it is laborious to recover them, e.g., from reprocessing losses of certain fuel forms.

Abbreviations

FHR Fluoride High temperature Reactor

GFR Gas cooled Fast Reactor

HPLWR High Performance Light Water Reactor

HTR High Temperature Reactor

LFR Lead cooled Fast Reactor

LWR Light Water Reactor

MCFR Molten Chloride Fast Reactor

MFBR Metal fueled Fast Breeder Reactor

MSFR Molten Salt Fast Reactor

MSR Molten Salt Reactor (graphite moderated)

PHWR Pressurized Heavy Water Reactor

RBMK Реактор Балшой Мощности Канальный (high-power graphite moderated water cooled channel-type reactor)

SFR Sodium cooled Fast Reactor

Introduction

This is the third of a series of three chapters that is dedicated to the identification of the advanced reactor technologies that are capable of self-sustaining breeding when fueled with either ^{232}Th or ^{238}U , and to the investigation of equilibrium fuel cycles

properties and neutronics performance. The first chapter (Krepel, 2021) characterizes the available fuel resources from the nuclear and reactor physics perspective. The second chapter (Krepel and Losa, 2021) determines the equilibrium fuel composition in advanced reactors and compares the basic features of the equilibrium fuel cycle and selected reactor neutronic performance characteristics at the equilibrium state. This third chapter evaluates the fuel cycle performance from a neutronics perspective by means of the earlier obtained equilibrium parameters. It provides an additional insight on why some advanced reactor concepts can act as self-sustaining breeders in a closed or even an open cycle, while others do not.

Nuclear fuel cycle

The purpose of the nuclear fuel cycle, and of nuclear reactors as such, is energy production in two major forms: electricity and heat. The nuclear fuel cycle can be perceived from different perspectives. From a cosmological perspective, the nuclear fuel cycle consists of: first generation star formation following the big-bang (Garcia, 2021), supernova explosions of such first generation stars, ensuing second generation stars and planets formation, and finally primordial actinides utilization on planet Earth by mankind. Hence, the nuclear fuel resources are not renewable, and our aim should be to maximize their utilization with minimal waste production. From the more classical perspective, the nuclear fuel cycle includes stages before the irradiation: exploration, mining, milling, conversion, enrichment and fabrication (Crawford, 2021), the irradiation stage itself, and stages after the irradiation: interim storage, transportation, reprocessing, transmutation and waste disposal (Glatz, 2021). This article focuses solely on the irradiation phase. However, it has implications also for the other phases. For instance, the product of exploration, mining, milling and conversion is natural uranium. Based on the current irradiation scheme, we utilize less than 1% of the mined material, while 80% is stored in form of depleted uranium as the residuum from enrichment process, and 19% is discharged with the irradiated fuel. Higher utilization of the natural uranium resource will reduce the environmental impact, per unit of produced energy, of both uranium mining and enrichment.

Fuel cycle performance parameters

From the irradiation perspective, the major fuel cycle performance parameters are:

- breeding capability,
- achievable burnup,
- initial fissile mass,
- means of criticality maintenance, and
- transmutation capability.

The neutron economy is not listed as a standalone performance parameter. Nonetheless, since it effects all other parameters, it is analyzed in depth. Natural resources utilization is also not listed as a performance parameter. There is a big difference between the ^{235}U -fueled open cycle, where the natural uranium utilization does not exceed 1%, and the self-sustaining breeding in closed cycle (Wigeland, 2021), where the utilization of thorium and natural uranium can reach up to 95–98%. However, within these two fuel cycles types, the differences in terms of natural resources utilization are minimal. Accordingly, the breeding capability suffices as indicator for resources utilization, and the Breeding Ratio (BR) is used here as its measure.

However, there is one type of open cycle, where the utilization can be substantially larger than 1%. In so called breed-and-burn (B&B) open cycle mode, the natural uranium utilization can reach up to 20–35% (Qvist, 2021; Krepel and Kramer, 2021; Green-span, 2021). For this type of self-sustaining breeding in open cycle only the basic criteria for B&B operation are evaluated and discussed. Similarly, there are also safety related fuel cycle parameters. While these are not part of this study, the temperature coefficients of reactivity are addressed in Krepel and Losa (2021).

Compared reactor concepts

In this article the fuel cycle performance of 16 selected reactors is assessed. For details of their selection and their basic description, please refer to chapter (Krepel and Losa, 2021). The major criterion for the selection was to cover the majority of the advanced reactor concepts (Stanculescu, 2021) and the highest possible variety of the core neutron spectra. For readers' convenience the reactor labels are listed here in the same order used in the charts and tables:

FHR	Fluoride High temperature Reactor
HTR	High Temperature Reactor
RBMK	Reaktor Balshoi Moshnosti Kanalnyj (high-power graphite moderated water cooled channel-type reactor)
PHWR	Pressurized Heavy Water Reactor
MSR-FLIBE	Graphite moderated Molten Salt Reactor with fuel in form of eutectic mixture of lithium fluoride and beryllium di-fluoride with addition of actinides tetra-fluoride salts
MSR-FLI	Graphite moderated Molten Salt Reactor with fuel in form of eutectic mixture of lithium fluoride and actinides tetra-fluoride salts
HPLWR	High Performance Light Water Reactor

(Continued)

LWR	Light Water Reactor
MSFR-FLIBE	Molten Salt Fast Reactor with fuel in form of eutectic mixture lithium fluoride and beryllium di-fluoride with addition of actinides tetra-fluoride salts
MSFR-FLI	Molten Salt Fast Reactor with fuel in form of eutectic mixture of lithium fluoride and actinides tetra-fluoride salts
LFR	Lead cooled Fast Reactor
SFR	Sodium cooled Fast Reactor
GFR	Gas cooled Fast Reactor
MFBR	Metal fueled Fast Breeder Reactor
MCFR-NaCl	Molten Chlorides Fast Reactor with fuel in form of eutectic mixture of sodium chloride and actinides tetra-chloride salts
MCFR-AcCl	Molten Chlorides Fast Reactor with fuel in form of actinides tetra-chloride salts

The order of the reactors in the presented charts and tables was selected based on spectral index (Losa, 2016). For more discussion about equilibrium spectrum but also about assumptions and simulation tools refer to Krepel and Losa (2019, 2021).

Methods of fuel cycle performance assessment

Direct irradiation simulation

The basic method of fuel cycle performance evaluation is the direct irradiation simulation of an open cycle and comparison of the BR, achieved burnup, and differences between initial and irradiated fuel compositions. The major disadvantage of this method is that the initial fuel composition itself determines the BR and the respective reactor type (see Fig. 2). Furthermore, different initial compositions result in different achievable burnups. It is therefore hard to make general conclusions about the fuel cycle performance and inter-compare reactor types by means of the open cycle irradiation. It does not necessarily show where the fuel composition tends to converge, and whether the reactor would be critical at equilibrium. The BR obtained by direct irradiation simulation is often not sustainable, because not all daughter products concentrations are fully equalized. Unlike in the open cycle, explicit simulation of repetitive irradiation in a closed cycle is one of the most accurate methods to evaluate the ultimate fuel cycle performance (Krepel et al., 2012). Nonetheless, depending on the applied code, it is usually also the most time demanding simulation.

D-factors

Two more general methods have been proposed in OECD (2006) to evaluate actinides impact on the fuel cycle performance. Even though they were predominantly designed for synthetic actinides transmutation, they are also valid for the breeding. Both these methods rely in one or another way on the stabilized irradiation chain. The first method is based on so called D-factors. These factors represent the number of neutrons which are needed to ultimately fission one atom of a given nuclide. Accordingly, the knowledge of the respective irradiation chain is needed, because the D-factors account also for the neutron consumption by daughter products (Salvatores et al., 1994). The D-factors provide direct information about the neutron cost for the ultimate transmutation of given nuclide. Depending on the spectrum and nuclide type, D-factors can be positive (neutron consumption) or negative (neutron production). The D-factors thus quantify additional burden or gain for the neutron economy in a reactor loaded by external synthetic actinides. Since breeding is nothing else than a ^{232}Th or ^{238}U transmutation, the D-factors of ^{232}Th and ^{238}U actually indicate the neutron balance of a breeder. Since negative D-factor means that neutrons are produced, the ^{232}Th or ^{238}U D-factors should be negative to enable breeding (Krepel and Losa, 2016).

There are four basic Reactions (R) ongoing in nuclear reactor, where each of them can result in neutron production or consumption C^R :

1. $(n, 2n)$, producing one neutron ($C^1 = -1$)
2. neutron capture (n, γ) , consuming one neutron ($C^2 = 1$)
3. radioactive decay, no interaction with neutron ($C^3 = 0$)
4. fission (n, f) , consuming one and producing ν neutrons ($C^4 = 1 - \nu$)

The other interactions with neutrons were neglected in this evaluation. The D-factor for certain nuclide I is calculated from the knowledge of the irradiation chain and rely on the reaction probability P of the reaction type R which is multiplied by the above listed neutron consumptions C . These products should be summed up for all reactions, accounting also for the daughter products:

$$D_I = \sum_{R1=1}^4 P_{I \rightarrow I_{R1}} C_I^{R1} + \sum_{R1=1}^4 P_{I \rightarrow I_{R1}} \left(\sum_{R2=1}^4 P_{I_{R1} \rightarrow I_{R1,R2}} C_{I_{R1,R2}}^{R2} \right) + \sum_{R1=1}^4 P_{I \rightarrow I_{R1}} \left(\sum_{R2=1}^4 P_{I_{R1} \rightarrow I_{R1,R2}} \left(\sum_{R3=1}^4 P_{I_{R1,R2} \rightarrow I_{R1,R2,R3}} C_{I_{R1,R2,R3}}^{R3} \right) \right) + \dots \quad (1)$$

where, for instance, $I \rightarrow I_{R1}$ represent reactions R on a nuclide I of a type 1, 2, 3, or 4 as listed above, which leads to a daughter product I_{R1} . The fission reaction $R = 4$ terminates the transmutation chain. Reactions of the original nuclide I are presented by the first term in Eq. (1), the second term stands for the reactions of its daughters I_{R1} , and the third term for reaction of the daughters of daughters $I_{R1,R2}$ (see Fig. 1). Since one of the later daughter products $I_{R1,R2}$, $I_{R1,R2,R3}$, $I_{R1,R2,R3,R4}$, etc. may be identical with the

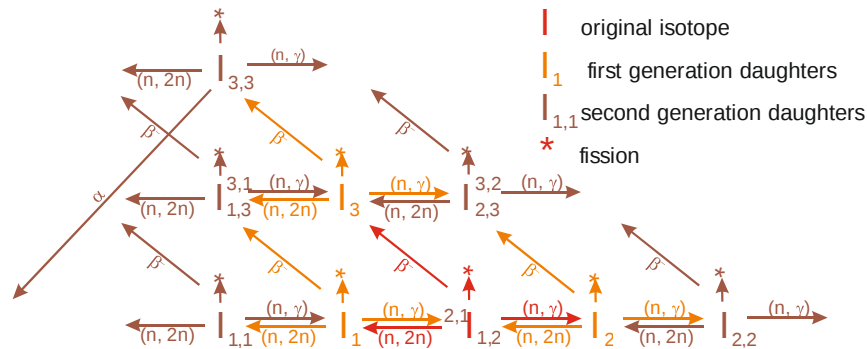


Fig. 1 Example of the D-factor algorithm application. Losa E (2016). *U-Pu and Th-U Fuel Cycle Closure*, PhD thesis. Prague: Czech Technical University in Prague, Department of Nuclear Reactors; Krepel J and Losa E (2016) Enumeration of static and dynamics neutron consumption D-factor for several selected reactors at equilibrium closed fuel cycle, *Proceedings of ICAPP 2016, San Francisco, USA*.

original nuclide I the equation represents an infinite series. For instance, the $(n, 2n)$ and (n, γ) reactions (see Fig. 1) cause that the nuclides I , $I_{2,1}$, and $I_{1,2}$ are identical. Fig. 1 also illustrates the plenitude of possible transmutation paths and the difficulty to compute the D-factors. Nonetheless, since all reactions are properly accounted for, the D-factors of ^{232}Th and ^{238}U accurately describe the neutron balance of a breeder.

Equilibrium method

The second method, proposed in OECD (2006) is actually the equilibrium method. It relies on the fact that if all fuel cycle parameters are fixed, recycling in closed cycle results in the equilibrium fuel composition. One of the fixed parameters is the composition of the fuel feed (make-up fuel). In OECD (2006) the primary aim was synthetic actinides transmutation, and thus the feed included them. This article focuses rather on the resources utilization, and the primordial actinide nuclides ^{232}Th or ^{238}U are exclusively used as the fuel feed. In this case a single irradiation chain represents the whole equilibrium fuel composition, and it is, therefore, the Eigen-vector of the respective Bateman equation (Bateman, 1910). For more detail about equilibrium fuel cycle simulation refer to Krepel and Losa (2021). This reference was also used to adopt the equilibrium fuel composition (stabilized actinides vector or actinides Eigen-composition) and the equilibrium multiplication factors.

One distinguished feature of the equilibrium fuel composition is that the concentrations of all actinides are constant. Per definition, the breeding and fission rates must be in balance for this composition. The rate of ^{232}Th or ^{238}U transmutation is thus equal to the fission rate of synthetic actinides. Hence, for each reactor there exists an equilibrium fuel composition for which $\text{BR} = 1$. However, not all reactors are critical with this fuel composition. Since there is a straightforward relation between BR and reactivity (Krepel et al., 2018), the equilibrium reactivity is used in this study to evaluate the breeding capability.

Breeding capability

Reactor classification

Breeding capability, a foremost fuel cycle performance parameter, defines whether a reactor can or cannot be operated as a self-sustaining breeder in closed cycle, which relies solely on fertile ^{232}Th or ^{238}U feed. It is determined by the neutron economy of the equilibrium state, particularly by the parasitic neutron capture of the equilibrium actinides composition. The breeding capability can be quantified as the ratio between breeding and fission reactions. This ratio is called Breeding Rate (BR), or alternatively also conversion ratio. In some cases the breeding capability is expressed as Breeding Gain (BG), where $\text{BG} = \text{BR} - 1$. In a self-sustaining breeder, where $\text{BR} \geq 1$, the fissile mass remains constant or slightly grows ($A \leq B$ in Fig. 2). In a reactor, where fission ($\text{BR} < 1$) or breeding ($\text{BR} > 1$) dominates, the fissile mass decreases or increases, respectively. However, only the equilibrium BR is relevant for the breeding capability assessment. BR of non-equilibrium fuel compositions does not need to be sustainable. Especially in the Th-U cycle, initial fissile load often consists of ^{233}U , ^{239}Pu , or highly enriched uranium, and $\text{BR} > 1$ can be obtained during the first irradiation cycle. Nonetheless, it does not mean that it is sustainable. Typically, in thermal spectrum systems, $\text{BR} > 1$ during the first irradiation cycle underestimates the parasitic neutron capture of synthetic actinides, e.g., ^{234}U , ^{236}U or ^{237}Np , which obviously did not yet converge to their equilibrium concentrations.

Equilibrium BR can be used to classify reactors into four categories: burner, converter, breeder and B&B (see Fig. 2). In a breeder it is at least equal to 1, in a converter it is below 1, and in a burner it should be at best 0. This would mean that there is no breeding process at all, what is nearly impossible. The BR border between burner and converter is not well defined and ranges from 0.5 to 0.9. Nevertheless, the major difference is that the fertile isotopes ^{232}Th and ^{238}U are usually avoided in burners dedicated to Minor Actinides (MA) transmutation. The minimal BR for a B&B reactor is around 1.5 (see Fig. 6), and thorium fueled reactors practically cannot be operated in B&B mode.

Neutron economy

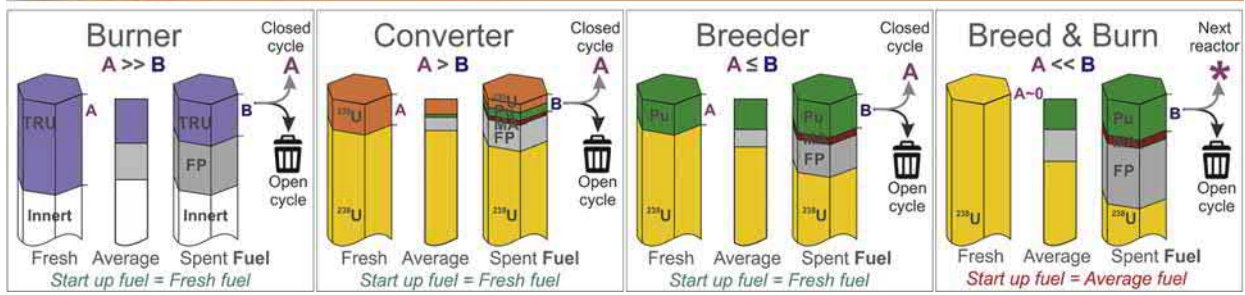


Fig. 2 Illustration of reactor classification by breeding ratio for ^{238}U as a fertile isotope. A is the initial and B the discharged fissile actinides share.

Breeding ratio

The equilibrium multiplication factor k obtained in Krepel and Losa (2021) is smaller than one for some reactors. Using the simple relation from Krepel et al. (2018), BR or BG can be calculated from k and the average number of neutrons per fission $\bar{\nu}$ as follows:

$$BR \cong 1 + \bar{\nu} \frac{k - 1}{k} \quad \text{or} \quad BG \cong \bar{\nu} \frac{k - 1}{k} \quad (2)$$

The approximate relation in Eq. (2) was derived for the equilibrium state and its accuracy decreases with increasing distance of k from 1. For convenience the breeding capability will be discussed using BG. From the 16 selected concepts, there are only few thermal Th-U cycle reactors with positive or close to zero BG (see Fig. 3). Nonetheless, it should be recalled that the equilibrium multiplication factors, applied for this BG estimate, correspond to an infinite reactor without Fission Products (FPs). The infinite BG presented in Fig. 3, thus, represents hypothetical maximum and not realistic values. The neutron losses by leakage from finite core can be reduced by blanket and/or reflector addition. However, the parasitic neutron absorption by FPs cannot practically be suppressed, since it would result in prohibitively high fuel reprocessing frequencies in thermal breeders. The achievable burnup as the second fuel cycle parameter would be discussed in the next sub-chapter. However, MSR concepts with liquid fuel could potentially be considered as a reasonable thermal spectrum breeders. Since their fuel is liquid, continuous FPs removal strategies might offset high reprocessing frequency penalties. Furthermore, this would allow for removal of higher actinides and questionable separation of ^{233}Pa and its decay outside of the neutron flux to improve the neutron balance (see 3rd and 4th term of Eq. 4).

All fast reactors have positive BG. Only the MSFR-FLIBE with the softest fast spectra, see Fig. 4 from Krepel and Losa (2021), has close to zero but negative BG in the U-Pu cycle. For all other fast reactors BG is almost twice higher in the U-Pu cycle. BG as well as equilibrium multiplication factor are integral parameters and encompass also the impact of non-fuel materials in the core. Even though, non-fuel materials introduce a certain perturbation, it is clear that the U-Pu cycle profits more from the spectrum hardening as compared to the Th-U fuel cycle. Once the ^{239}Pu capture probability decreases in a hard spectrum, the higher average number of neutrons per fission starts to dominate. The Th-U fuel cycle is also performing better in a harder neutron spectrum. However, the

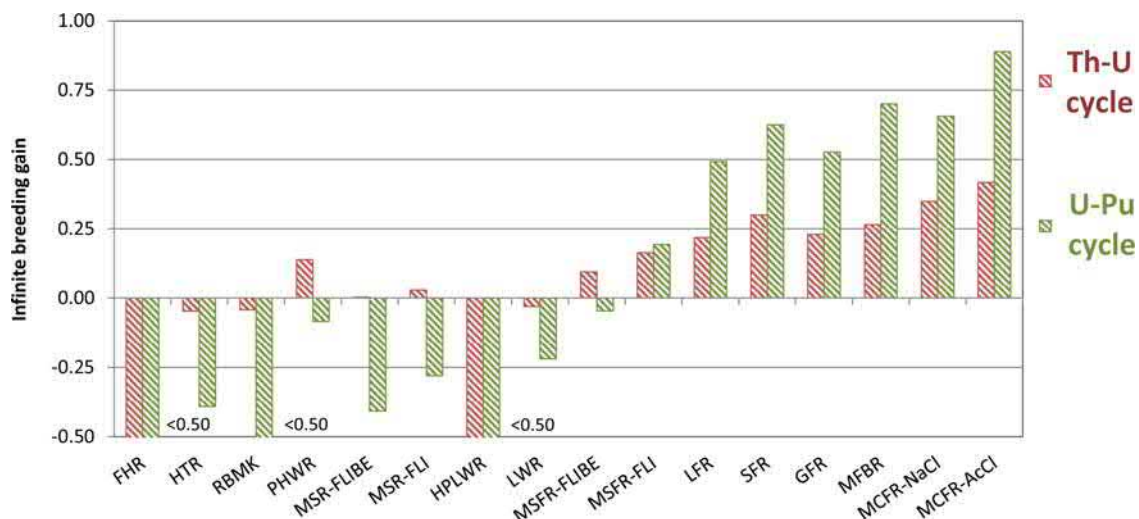


Fig. 3 Infinite breeding gain for Th-U and U-Pu cycles and 16 selected reactors.

trend is much milder, because the ^{233}U capture probability is generally low and less dependent on neutron spectrum. The BG in the Th-U fuel cycle is lower in fast reactors because of the ^{233}U lower average number of neutrons per fission.

Achievable burnup

Once the reactors are classified according to their breeding capability, the second most important performance parameter for the fuel cycle is the achievable burnup. In both open and closed cycles, higher burnup reduces the core reloading frequency and the need of fuel fabrication, transportation and optional reprocessing. Hence, it reduces the fuel cycle costs. Since the amount of synthetic actinides tends to stabilize during the irradiation, higher burnup in an open cycle results in lower waste amount per unit of produced energy. In a closed cycle, it decreases the reprocessing frequency and thus also reduces the waste amount, see Fig. 2 in Krepel (2021). From neutron economy perspective, the achievable burnup is determined by BR, initial fissile mass, and equilibrium fissile mass.

Fission products cumulation

The major issue of burnup is that the neutron economy, but also other fuel mechanical and chemical features, deteriorate during irradiation. In every reactor, FPs accrue during irradiation and increase the parasitic neutron capture. Ultimately, every reactor will become sub-critical if FPs would not be regularly removed. FPs impact on neutron economy depends on their concentration and neutron capture cross-section. Its relative strength is associated with the fissile actinides concentration and fission cross-section. Hence, the deterioration caused by FPs can be pronounced or suppressed by evolution of fissile actinides share and depends also on the absolute value of this share. For equilibrium fuel composition $\text{BR} = 1$, and the share of fissile actinides is constant. Should the initial fissile share be above or below the equilibrium one, the irradiation will result in its decrease or increase, respectively. This has strong implication for actinides parasitic capture itself, but also for the relative FPs strength. Majority of ^{235}U -fueled reactors use thermal spectrum and low enriched uranium. Hence, BR is around 0.2–0.5 and the fissile actinides share is strongly decreasing during the irradiation. Consequently, the relative importance of FPs concentrations and capture cross-sections is growing faster than the respective absolute values.

Impact of absolute fissile share

The absolute share of fissile actinides also indicates how sensitive the neutron economy will be to FPs buildup. The fissile share in the ^{232}Th irradiation chain is roughly 1.5% in thermal and 10% in fast reactors (see Fig. 4). A burnup (BU) of 5% FIMA implies that 5% of the actinides are fissioned. At the same time, the FPs mass corresponds to 300% of the fissile actinides mass in thermal and only to 50% in fast reactors. In the U-Pu cycle it is even more pronounced, burnup of 5% FIMA means that the FPs mass can be up to 14 times higher than the fissile mass (see Fig. 5). The higher fissile share is actually the major reason why fast reactors are less sensitive to FPs build up. However, it partly depends also on the ratio between FPs capture cross-section and actinides fission cross-sections, which is spectrum dependent. Even though, it is rather a hypothetical example, 1.5% of fissile share in discharged fuel and burnup of 5% FIMA represent reasonable values for typical LWR.

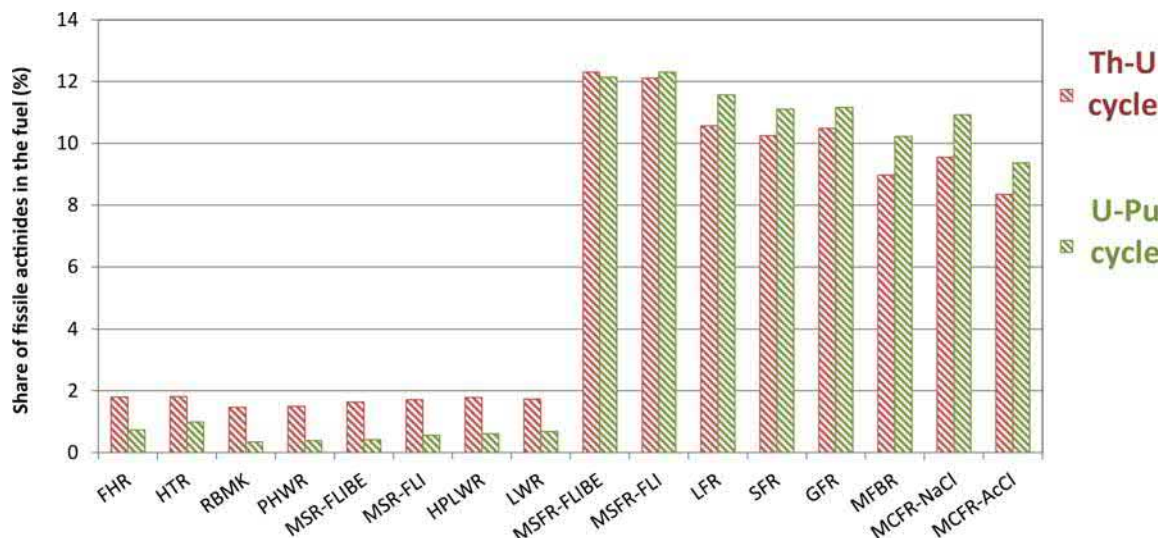


Fig. 4 Share of fissile actinides in the fuel for Th-U and U-Pu cycles and 16 selected reactors.

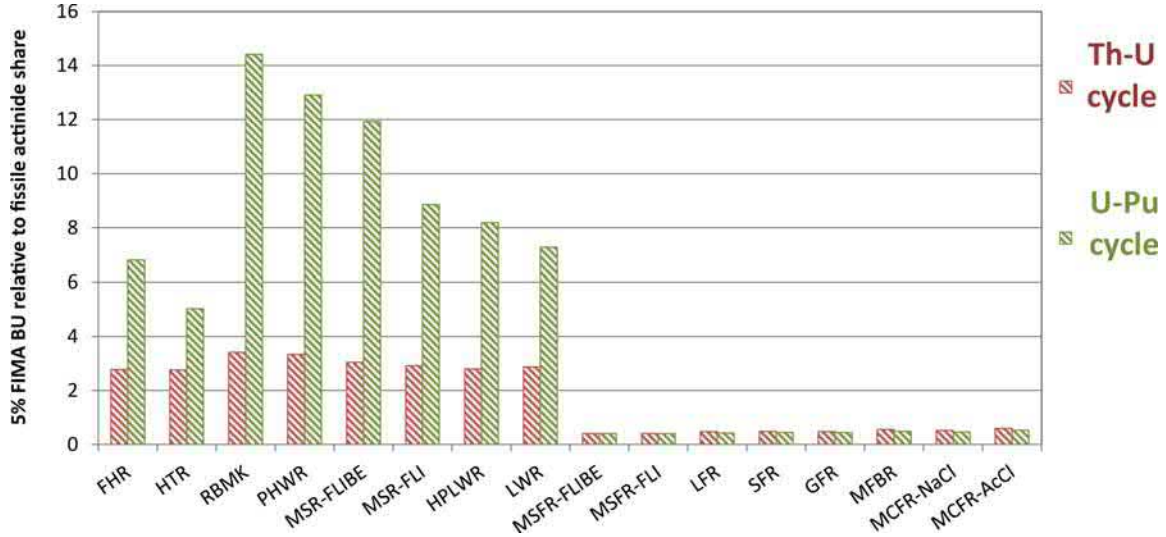


Fig. 5 Ratio between 5% FIMA burnup and fissile actinides share in the fuel for Th-U and U-Pu cycles and 16 selected reactors.

This study does not evaluate achievable burnup as such and it even neglects FPs. Nonetheless, for each reactor, the BR is being assessed and fissile actinides share is provided. In general, the foreseen burnup for fast breeder reactors is usually similar to their fissile actinide share in the core, i.e., around 10% FIMA. Nonetheless, 15–20% is achievable from a neutron economy perspective. Breeding in thermal reactors would likely not even allow for reaching burnup equal to the fissile fuel share.

It should not be forgotten, however, that a reactor core usually consists of several groups of fuel assemblies with different burnups to minimize the reactivity oscillations (see Fig. 8). The average FPs share in every core is thus roughly the half of final discharged burnup. Accordingly, for discharged burnup of 20% the average FPs share in the reactors would be 10%. Hence, it would be equal in this example to the fissile actinides share in fast reactors. The exceptions from this rule, are the reactors with liquid fuel (MSR, MSFR, and MCFR) where the fuel is homogeneously mixed, the effective FPs share in the core is averaged, and the final discharged burnup is equal to the average burnup. The achievable burnup in these reactors can be lower. At the same time, the liquid fuel allows for continuous FPs removal. Hence, it partly compensates the averaging effect.

Basic assessment of the B&B cycle

The neutron economy of a breeder is very tight, and the bred synthetic actinides must be recycled to maintain the fissile material balance. The only exception from this requirement are very strong breeders, where the fissile excess bred during the irradiation would be equal or higher than the discharged fissile mass after the irradiation. Since the bred excess is not necessary for the fissile mass conservation, these very strong breeders can self-sustain the breeding in open cycle in the B&B mode (Qvist, 2021). The name is derived from the fact that the loaded fuel does not need to contain fissile actinides; they are bred during the irradiation in the B&B reactor and suitable fraction of them fission in that reactor. The trivial condition for B&B reactor operation is based on the balance of bred synthetic actinides (Krepel et al., 2018). The amount of extra bred fissile synthetic actinides, in a period between two refuelings, should be equal or higher than the amount of fissile synthetic actinides (F_F) removed during the actual refueling:

$$F_F \leq BU \cdot BG \quad (3)$$

where BU is the burnup in absolute FIMA share and the product of BU and BG represent the extra bred amount of fissile synthetic actinides. The presence of burnup in Eq. (3) is also the reason why the B&B cycle assessment is placed after the achievable burnup subsection. This trivial condition for B&B reactor operation can be graphically presented for three particular fissile actinides contents in the core. These are 1%, 2% and 10% and correspond roughly to the equilibrium fissile actinides share in Fig. 4. In thermal reactors, with 1% or 2% of fissile actinides share in the respective U-Pu or Th-U cycle, the overall neutron economy does not allow for B&B operation. The required combination of BG and burnup is not achievable in thermal spectrum. The B&B cycle is possible only in fast reactors. Even though, the fissile actinides share of 10% shifts the respective curve in Fig. 6 higher, the required combination BG and burnup is still more realistic. Assuming burnup of 20% FIMA, BG of 0.5 would be needed for reactor with 10% share of fissile actinides. There are only four concepts with sufficiently high BG in Fig. 3, these are the two sodium cooled concepts and the two MCFR concepts. Since the MCFR concept with liquid fuel allows for partial FPs removal from the core, it can theoretically allow for higher burnup in the B&B open cycle mode (Krepel and Kramer, 2021).

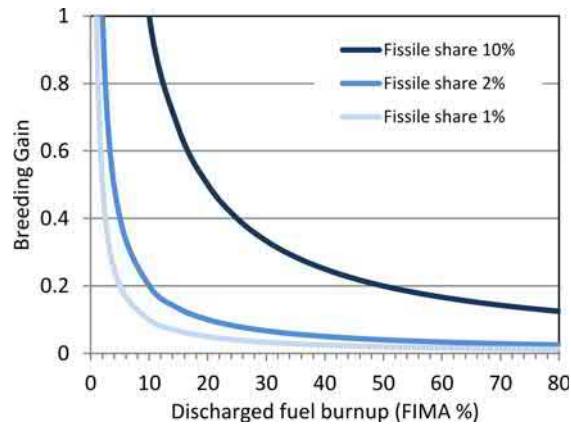


Fig. 6 Relationship between breeding gain, discharged fuel burnup, and fissile share in discharged fuel to enable B&B reactor operation.

Initial fissile mass

As mentioned, nuclear reactors can be divided according to BR into four categories presented in Fig. 2. The breeder and the B&B reactor will, once started, maintain their own fissile mass constant and use only ^{232}Th and ^{238}U (in form of depleted or natural uranium) as the fertile feed. Since the self-sustaining breeder maintains its own fissile share, no matter if in a closed cycle with recycling or in an open B&B cycle, the initial fissile mass does not strongly differ from the equilibrium fissile mass.

The burner reactor is fueled with synthetic actinides, and its objective is maximum reduction of their amount. It should be at best operated in closed cycle. Hence, synthetic actinides represent its initial fuel in open cycle, as well as the feed in closed cycle.

Only converters are extensively operated in an open cycle. Their initial fuel is typically enriched uranium, and their purpose is energy production. The vast majority of ^{235}U -fueled converters has thermal neutron spectrum and cannot be operated as a breeder. The equilibrium share of fissile actinides in thermal reactors is low. In the ^{238}U chain, it is below 1% (see Fig. 4). Hence, it is quite similar to the ^{235}U share in natural uranium. Even though there are exceptions, neither natural uranium nor the ^{238}U irradiation chain can make thermal reactors critical. To be able to operate these reactors, but also to reach a reasonable burnup, the capture rate of ^{238}U needs to be suppressed by uranium enrichment. Since the equilibrium fissile share in ^{238}U chain is below 1%, the 5% ^{235}U enrichment will result in a more than five times lower ^{238}U capture rate. Accordingly, the BR in this simplified example will be roughly 0.2 for 5% enriched uranium.

The initial fissile mass, as well as the transition scenarios towards the equilibrium cycle, are not an objective of this study. For all reactors operated in a closed cycle, the initial fissile mass is determined by the equilibrium fissile share. To provide some insight, the specific density of the main fissile nuclide in the equilibrium state is shown in Fig. 7. It is only an indication of the initial fissile mass requirement. To assess the total masses, it should be multiplied by the approximate core volume (Krepel and Losa, 2021). Furthermore, for a converter operated in an open cycle, the initial fissile mass is correlated with the achievable burnup.

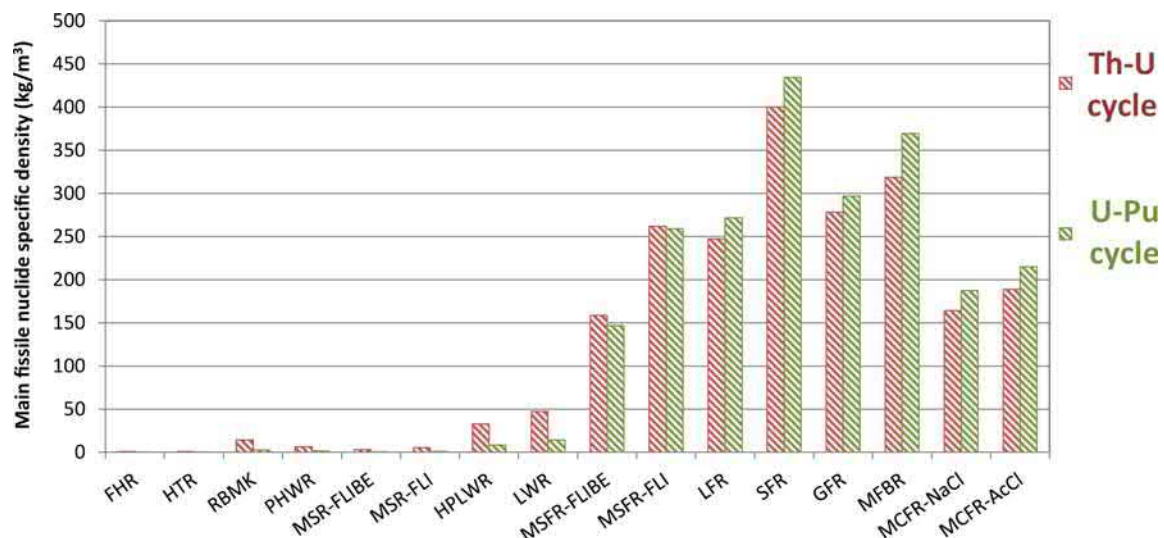


Fig. 7 Main fissile nuclide, i.e., ^{233}U or ^{239}Pu , specific density for Th-U and U-Pu cycles and 16 selected reactors.

Means of criticality maintenance

Means of criticality maintenance are relevant predominantly for the open cycle. In the ^{235}U -fueled reactors, the fission rate strongly dominates over the breeding, and the fissile mass is decreasing during the irradiation. As a consequence, the FPs impact on neutron economy is growing in both absolute and relative terms. To achieve reasonable burnup, the initial ^{238}U capture rate must be suppressed by uranium enrichment. This suppression exceeds the initial criticality needs and causes a huge reactivity excess. This excess needs to be compensated by neutron absorbers, which are either burnable or removable, e.g., control rods. They compensate the initial reactivity excess needed for high burnup and are gradually burned or removed. They are called burnable, because their capture cross-section is very high and their mass low. Hence, their capture rate is strongly decreasing with their transmutation. The primordial nuclides ^{232}Th and ^{238}U can be theoretically also considered as burnable absorbers. However, their capture cross-sections are rather small and absolute mass rather high. Accordingly, the capture rate change is too slow. Nonetheless, ^{232}Th performs slightly better from this perspective and in some reactors it may indeed act as burnable absorber, which is balancing the increasing mass of FPs (Krepel et al., 2015).

In fast reactors operated in a closed cycle or in an open B&B cycle, the fissile fuel share is roughly constant and usually around 10% (see Fig. 4). Nonetheless, the FPs buildup and the ^{232}Th or ^{238}U refilling induces need for regular fuel reloading. The FPs buildup and the actinides composition changes result in reactivity variation during irradiation. Hence, control rods are needed for their compensation. To illustrate these variations the results of a GFR sensitivity study Krepel et al. (2010) are presented in Fig. 8. In this simulation, the GFR was operated as a self-sustaining breeder in closed U-Pu cycle, and different stages of the fuel cycle (irradiation, cooling time before reprocessing, and reprocessing) were explicitly simulated. In the sensitivity study, the cooling time before reprocessing was altered. The objective of the study was to evaluate the impact of ^{241}Pu decay during the cooling period. Since the fissile nuclide ^{241}Pu decays into the fertile nuclide ^{241}Am , the cooling time was actually altering the ratio between fissile and fertile nuclides. Accordingly, the longer cooling time resulted in higher BG at the beginning of irradiation. Fig. 8 is thus prototypical for reactivity variations in a fast breeder reactor as a function of the initial BG.

The left hand side of Fig. 8 shows the reactivity variation caused by FPs buildup and actinide composition change during the irradiation using an one-batch approximation. In this case all assemblies are loaded, irradiated and unloaded together. On the right hand side of Fig. 8, the assemblies are divided into three groups and only one third is removed at a time. In both cases, the reactivity variations are minimal, when the initial BG is positive, and the reactivity firstly grows. This fact is underlined on the right hand side of Fig. 8, where the three-batch operation practically averages the curves from the left hand side. For the cases with highest initial BG, or actually longest cooling time, where the reactivity initially grows (left hand side of Fig. 8), the variation is minimal (bottom curves in right Fig. 8).

The reactivity variations presented above can be avoided in reactors where the FPs are being continuously removed and actinides continuously added to the core. This is the case for MSR, MSFR, MCFR with liquid fuel and for PHWR and HTR (FHR) with small fuel assemblies and online refueling strategy (during power operation). While criticality maintenance is not an objective of this study, it is worthwhile noting that self-sustaining breeders are expected to have much smaller reactivity variation during irradiation than an open cycle operated ^{235}U -fueled reactors. At best, the reactors should have small breeding ratio excess, so that the fissile mass is slightly increased at the beginning of the cycle and balances the subsequent decrease caused by FPs parasitic neutron capture.

Transmutation capability

Some of the advanced reactor concepts are dedicated to the minimization of legacy synthetic actinides. Their performance is thus measured by the transmutation capability. This is an ambivalent term, because even the breeding is a transmutation. Nonetheless, the transmutation refers here rather to the fission, which is the major option to reduce the long-term stewardship burden related to

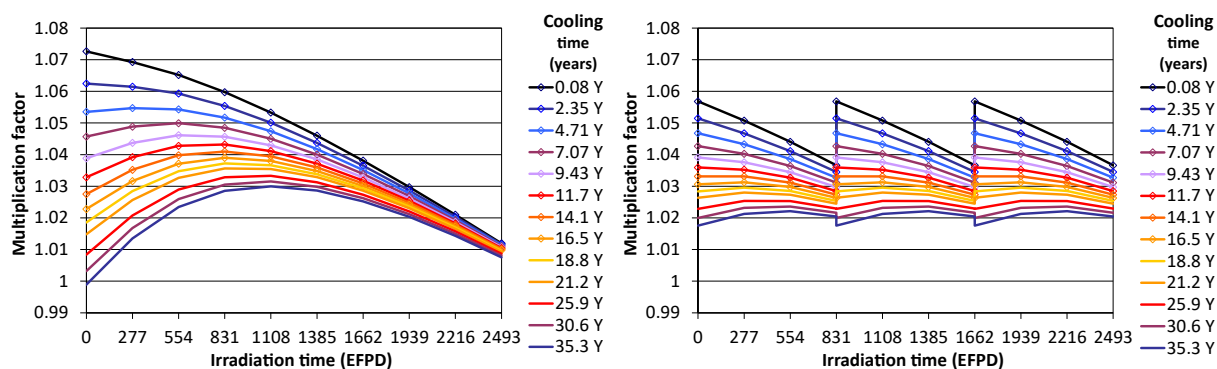


Fig. 8 Neutron multiplication factor evolution during the irradiation in GFR in one-batch approximation (left) and three-batch simulation (right) as a function of fuel cooling time before reprocessing, which changes the initial BG.

synthetic actinides. Furthermore, some concepts are meant to combine both fuel breeding and legacy synthetic actinides transmutation objectives. The usual measure of the transmutation capability is the concentration reduction of selected actinides and the related neutron costs. The reduction can be expressed as the partial fission share of the legacy actinides. The neutron costs are evaluated here by the D-factors.

The fissioned share of legacy actinides in open cycle can be deduced from the difference between the loaded and discharged fuel. Necessary condition for successful transmutation is that the concentration of a given legacy actinide, assuming that it is irradiated together with either ^{232}Th or ^{238}U , is above its equilibrium concentration in the respective irradiation chain. The equilibrium fuel composition thus provides certain information about transmutation capability. For instance, based on the example in Fig. 3 from Krepel (2021), the ^{239}Pu share in the ^{232}Th irradiation chain is very small (0.007% of the fuel) and comparable to the ^{238}Pu and ^{242}Pu concentrations. The share of other plutonium nuclides or of americium and curium nuclides is even lower. This feature makes the ^{232}Th irradiation chain attractive for utilization of synthetic actinides originated from ^{238}U irradiation. In some reactor concepts, thorium is thus proposed as a semi-inert fuel matrix (Shwageraus 2021; Wigeland, 2021). The term semi-inert refers to the fact that ^{232}Th irradiation does not extensively generate synthetic nuclides, which are typical for the ^{238}U chain. Nonetheless, it produces its own synthetic nuclides, and the benefit of this solution should be carefully evaluated, refer to Fig. 10 from Krepel (2021).

In a closed cycle, where all fed actinides are ultimately fissioned, the transmutation rate of legacy actinides is equal to their share in the feed. There are concepts, where MA act as the sole feed in a closed cycle, and their irradiation partly results in fissile Pu buildup (Artioli et al., 2007). The BR can be above one for the respective MA feed (Krepel et al., 2009). This is the reason why only the equilibrium BR related to ^{232}Th or ^{238}U should be considered for the reactors classification given in Fig. 2.

Transmutation generates neutron excess or shortfall, which is represented by the D-factors. As shown by the example given in Fig. 9, the D-factors do not differ strongly between the Th-U and U-Pu cycles. However, they vary strongly with the neutron spectrum. Transmutation of ^{242}Pu can cost around 1 neutron in a thermal spectrum while providing 0.8 neutrons excess in a fast spectrum. The dynamics of transmutation depends on the specific power of the reactor and on the absorption cross-section of the transmuted nuclide. In Fig. 10 the absorption rate per 10^9 atoms of ^{242}Pu is shown. Obviously, the rate is higher in thermal spectra, because of the generally larger cross-section. At the same time, it is quite high also for the MFBR due to its high specific power. In general, the transmutation of legacy actinides costs less neutrons in fast neutron spectrum systems; however, it can be faster in thermal neutron spectrum.

Neutron economy

All fuel cycle performance parameters discussed above depend on the neutron economy. Nonetheless, the equilibrium multiplication factor or BG provide only one integral value for the overall neutron economy. To provide some insight, the partial neutron economy of the actinides in the ^{232}Th and ^{238}U irradiation chains is analyzed in this subchapter.

The neutron economy of the equilibrium cycle is a particular case because all nuclides creation and destruction rates are in balance, and all actinides concentrations are constant. The neutron economy, also referred to as neutron tallying, has usually two pillars. The first one is the neutron generation, which is determined by the average number of neutrons emitted per fission, labelled as $\bar{\nu}$. This number is either related to the main fissile nuclide ^{233}U or ^{239}Pu or can be a weighted average over all fissile nuclides in the core. The second pillar relies on the fact that 1 neutron is needed for sustaining the chain reaction and one for the conservation of synthetic actinides concentration. Accordingly, to start with, two neutrons must be subtracted from the balance.

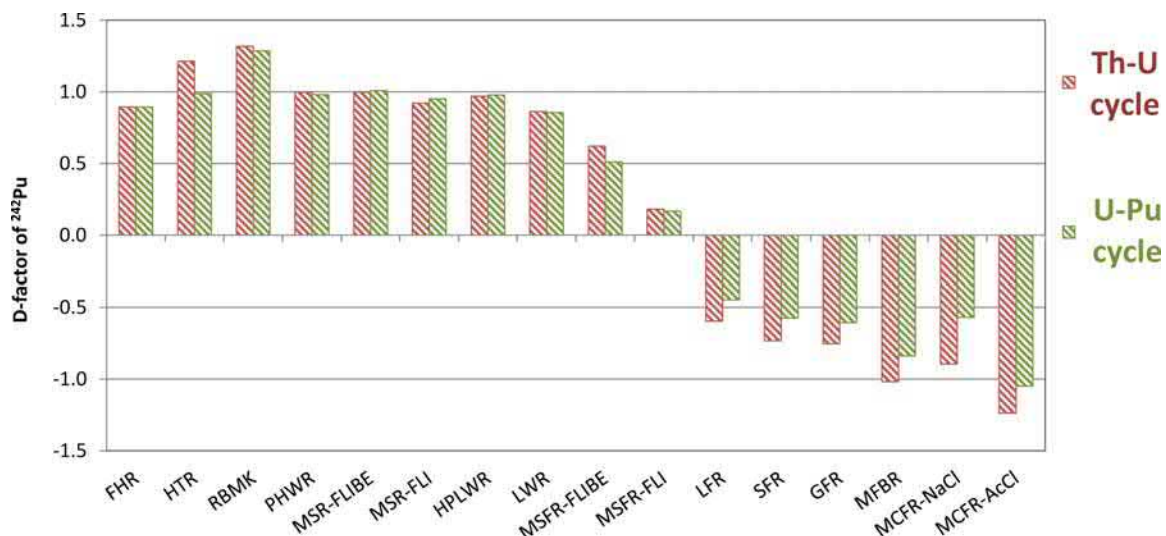


Fig. 9 D-factor of ^{242}Pu for Th-U and U-Pu cycles and 16 selected reactors.

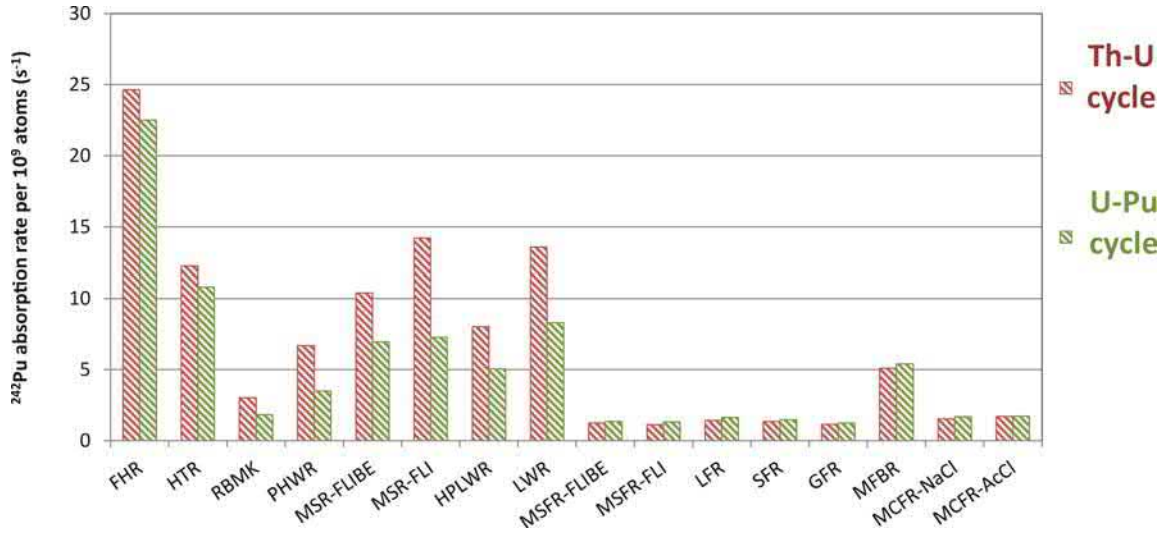


Fig. 10 Neutron absorption rate for 10^9 atoms of ^{242}Pu for Th-U and U-Pu cycles and 16 selected reactors.

η -2

The most traditional measure of the neutron economy is the η factor of the main fissile nuclide, i.e., of ^{233}U and ^{239}Pu . It is a product of $\bar{\nu}$ of the main fissile nuclide and its fission probability. The respective η -2 provides reasonable first estimate of the neutron excess. It is oriented on the conservation of the main fissile nuclide concentration. It assumes the need of two neutrons: one to be absorbed on the main fissile nuclide itself and other on the main fertile nuclide. Since η includes the fission probability it already accounts for the neutron capture by the main fissile nuclide. However, it is only approximate neutron tallying. Only main fissile and fertile nuclides and only selected reactions are considered. The impact of other nuclides and reactions in the irradiation chain is neglected. The accuracy of this estimate can be improved by:

1. Accounting for the relative share of fast Fission $F_{232\text{Th}}$ or $F_{238\text{U}}$ of the main fertile nuclide.
2. Accounting for the other synthetic actinides contributions to the neutron balance. This impact can be at best expressed by the D-factor of the respective ^{234}U and ^{240}Pu nuclides multiplied by the main fissile nuclide capture rate.
3. Accounting for the bypass of the main fissile nuclide by ^{233}Pa and ^{239}Np capture rate. Since this capture results in ^{234}U and ^{240}Pu buildup, this term is composed from own ^{233}Pa and ^{239}Np relative capture rate and from the ^{234}U and ^{240}Pu impact on the neutron balance.

The overall neutron balance n_{B1} , for instance for the ^{232}Th can be expressed as:

$$n_{B1} = \eta_{233\text{U}} - 2 + F_{232\text{Th}} - \frac{C_{233\text{U}}}{C_{233\text{U}} + F_{233\text{U}}} D_{234\text{U}} - \left(2 \frac{C_{233\text{Pa}}}{C_{233\text{U}} + F_{233\text{U}}} + \frac{C_{233\text{Pa}}}{C_{233\text{U}} + F_{233\text{U}}} D_{234\text{U}} \right) \quad (4)$$

where F, C, and D represent the fission rate, capture rate and the D-factor. The notation for the ^{238}U irradiation chain will be similar; however, the reaction rates for corresponding nuclides ^{238}U , ^{239}Pu and ^{239}Np should be applied.

Neutrons produced per fission

A very similar method relies on the neutron balance in a self-sustaining breeder, where the average number of neutrons produced per fission is reduced by 2, assuming one neutron for fission and one for breeding. This method was actually applied in form of reactivity break-down in Krepel and Losa (2019). In this case the $\bar{\nu}$ represents weighted average over all fissile nuclides in the core. The $\bar{\nu} - 2$ provides a less precise first estimate, because it completely neglects the actinides parasitic neutron captures. The accuracy of this estimate can be increased by:

1. Accounting for the relative share of fast Fission $F_{232\text{Th}}$ or $F_{238\text{U}}$ of the main fertile nuclide.
2. Subtracting the Capture rate C_{Ac} of all synthetic actinides.
3. Accounting for the (n, 2n) reaction on main fertile nuclide. It should be accounted twice, because it produces an extra neutron and it reduces number of neutrons needed for main fertile transmutation.

The overall neutron balance n_{B2} , for instance for the ^{232}Th can be expressed as:

$$n_{B2} = \bar{\nu} - 2 + F_{232\text{Th}} - \frac{\sum_i C_i}{\sum_i F_i} + 2R_{232\text{Th}}^{(n,2n)} \quad (5)$$

Costs of synthetic actinides fission

Alternative neutron tallying was introduced in Krepel (2021). It relies on $\bar{\nu}$ as a weighted average over all fissile nuclides in the core, knowledge of fission rate distribution between the nuclides in the irradiation chain, and neutron cost for each nuclide creation and fission (see Table 2 in Krepel (2021)). This method is also only approximate, and more accurate values are obtained when:

1. The nucleons lost by α decay reaction are accounted for.

The (n, 2n) reaction is already included in the respective nuclides creation cost from Table 2 in Krepel (2021). The overall neutron balance n_{B3} , for instance for the ^{232}Th can be expressed as:

$$n_{B3} = \bar{\nu} - \sum_i F_i n_i - 4 \sum_i \alpha_i \quad (6)$$

D-factors of ^{232}Th and ^{238}U

The D-factor of the main fertile nuclide represents the number of neutrons that are required for its ultimate transmutation. Hence, the D-factors of ^{232}Th and ^{238}U cover the neutron balance of the whole respective irradiation chains. However, the D-factors are defined as the neutron cost for ultimate transmutation. Hence, if there is a neutrons excess, the D-factors are negative. The equation for neutron balance n_{B4} has thus the trivial form:

$$n_{B4} = -D_{232\text{Th}} \quad (7)$$

Neutron economy overview

The above four neutron tallying method should provide exactly the same neutron balance. However, since not all necessary corrections are applied, the values in Tables 1 and 2 slightly differ. Usually only the main fertile nuclide (n, 2n) reaction is accounted for. Similarly, there are several minor paths through the chain, which are not always properly considered in the neutron tallying. The values averaged over these four methods are plotted in Fig. 11.

The results in Fig. 11, even though they do not account for parasitic neutron absorption by the structural materials, coolants and moderators, illustrate the difficulty to breed in the U-Pu cycle in thermal reactors. The PHWR is the only reactor in this study, which has a slightly positive neutron balance of the actinides chain in the U-Pu cycle. Nonetheless, it is not sufficient for realistic breeding. Fig. 11 also shows that the HPLWR, FHR, and RBMK performance in the U-Pu cycle does not differ dramatically from other thermal reactors and confirms that the very low BG in Fig. 3 is caused by parasitic neutron captures by non-fuel materials.

The values in Tables 1 and 2 can be also used for a general discussion about the U-Pu and Th-U closed cycles. The improved breeding in harder neutron spectra, surprisingly, is not related to the larger average number of neutrons emitted per fission. For each particular nuclide neutron economy relevant parameters are increasing with spectrum hardening. Nonetheless, the equilibrium fuel composition changes and enhances compensating effects. Thus, in the U-Pu cycle there are 2.9 neutrons emitted per fission, while in the Th-U cycle only about 2.5 neutrons. The higher number of available neutrons is the biggest advantage of the U-Pu cycle in terms of neutron economy. Another advantage of the U-Pu cycle is that up to 15% of the fissions are fast fissions of the fertile ^{238}U . Unlike the number of neutrons emitted per fission, the direct fission strongly depends on the spectrum. The strongest disadvantage of the U-Pu cycle is that the ^{239}Pu fission probability steeply decreases down to 60% with spectrum softening, whereas ^{233}U fission probability stays constant around 90%.

The dominance of the U-Pu cycle in all fast reactors ends with the MSFR, which is considered a fast reactor system. However, its spectrum has the same share of fast neutrons as the LWR (Krepel, 2021). In the Th-U cycle many thermal spectrum system have the same performance as MSFR. Unfortunately, it is the parasitic neutron capture by non-fuel materials which deteriorates the balance.

Summary

The nuclear fuel cycle relies on resources, which are not renewable and not every planet in our solar system or in the universe has them. The nuclear fission chain reaction is so far the only known alternative to nuclear fusion for producing energy from nuclear transmutation processes. The self-sustaining breeding in closed cycle is the best option to fully utilize the potential of the primordial actinides ^{232}Th and ^{238}U . The neutron economy of this process is tight, and not all advanced reactors can be operated as self-sustaining breeders. This article tried to explain the reasons for this.

Two methods have been applied for the fuel cycle performance evaluation. The first method relies on D-factors. This method was originally proposed to quantify the transmutation capability of waste burning reactors. Nonetheless, it can be applied also to ^{232}Th and ^{238}U utilization in nuclear fission reactors. The D-factors represent the number of neutrons which are needed to ultimately fission one atom of a given nuclide. Accordingly, knowledge of the equilibrium irradiation chain is needed to quantify them. The second method applied here is the equilibrium method. Similarly, like the D-factors, this method was proposed for waste

Table 1 Neutron economy of ^{238}U irradiation chain.

^{238}U chain	FHR	HTR	RBMK	PHWR	MSR-FLIBE	MSR-FLI	HPLWR	LWR	MSFR-FLIBE	MSFR-FLI	LFR	SFR	GFR	MFBR	MCFR-NaCl	MCFR-AcCl
$\bar{\nu}$ average	2.96	2.96	2.93	2.93	2.95	2.95	2.93	2.94	2.95	2.95	2.93	2.94	2.94	2.93	2.93	2.92
$\bar{\nu}_{239\text{Pu}}$	2.86	2.86	2.87	2.87	2.86	2.86	2.87	2.87	2.91	2.92	2.93	2.94	2.95	2.94	2.94	2.95
^{238}U fission prob. ($F_{238\text{U}}$)	0.00	0.01	0.05	0.06	0.03	0.04	0.09	0.09	0.05	0.07	0.09	0.13	0.14	0.13	0.11	0.15
^{239}Pu fission prob. ($F_{239\text{Pu}}$)	0.63	0.62	0.66	0.68	0.63	0.63	0.64	0.65	0.62	0.66	0.78	0.78	0.77	0.83	0.79	0.87
^{239}Pu capt. prob. ($C_{239\text{Pu}}$)	0.37	0.38	0.34	0.32	0.37	0.37	0.36	0.35	0.38	0.34	0.22	0.22	0.23	0.17	0.21	0.13
^{238}U D-factor ($D_{238\text{U}}$)	0.26	0.29	0.07	-0.04	0.21	0.18	0.04	-0.01	-0.01	-0.24	-0.62	-0.70	-0.70	-0.80	-0.69	-0.90
^{240}Pu D-factor ($D_{240\text{Pu}}$)	0.12	0.26	0.18	0.07	0.15	0.13	0.13	0.07	-0.32	-0.60	-1.04	-1.09	-1.10	-1.27	-1.10	-1.37
Synthetic Actinides Rel. capture (C_{Ac})	1.22	1.25	1.06	0.96	1.19	1.17	1.08	1.03	1.00	0.79	0.41	0.38	0.40	0.27	0.36	0.19
Neutron balance from Eq. (4): $\eta-2$																
$\eta-2$	-0.15	-0.17	-0.07	-0.01	-0.15	-0.15	-0.13	-0.10	-0.17	-0.06	0.28	0.29	0.25	0.43	0.33	0.55
+ 1 correction term	-0.14	-0.16	-0.02	0.05	-0.13	-0.11	-0.04	-0.01	-0.12	0.01	0.37	0.42	0.39	0.56	0.43	0.70
+ 1&2 correction terms	-0.25	-0.32	-0.12	0.00	-0.24	-0.21	-0.13	-0.07	-0.03	0.20	0.60	0.66	0.66	0.79	0.67	0.90
+ 1&2&3 correction terms	-0.26	-0.32	-0.12	-0.01	-0.24	-0.22	-0.13	-0.08	-0.03	0.20	0.60	0.66	0.66	0.79	0.67	0.90
Neutron balance from Eq. (5): $\bar{\nu}-2$																
$\bar{\nu}-2$	-0.21	-0.20	0.12	0.15	-0.11	-0.08	0.08	0.09	0.26	0.44	0.70	0.77	0.78	0.82	0.75	0.93
+ 1 correction term	-0.25	-0.28	-0.08	0.04	-0.21	-0.18	-0.04	0.01	-0.01	0.23	0.62	0.69	0.69	0.80	0.68	0.90
+ 1&2 correction terms	-0.25	-0.29	-0.08	0.04	-0.22	-0.18	-0.05	0.00	-0.01	0.23	0.61	0.68	0.68	0.79	0.68	0.89
+ 1&2&3 correction terms	-0.25	-0.28	-0.07	0.05	-0.21	-0.17	-0.03	0.02	0.00	0.24	0.63	0.70	0.70	0.82	0.69	0.91
Neutron balance from Eq. (6): $\bar{\nu}$ - fission cost																
$\bar{\nu}$ - fission cost	-0.21	-0.20	0.12	0.15	-0.11	-0.08	0.08	0.09	0.26	0.44	0.70	0.77	0.78	0.82	0.75	0.93
+ 1 correction term	-0.25	-0.28	-0.08	0.04	-0.21	-0.18	-0.04	0.01	-0.01	0.23	0.62	0.69	0.69	0.80	0.68	0.90
Neutron balance from Eq. (7): D-factor																
$-D_{238\text{U}}$	-0.26	-0.29	-0.07	0.04	-0.21	-0.18	-0.04	0.01	0.01	0.24	0.62	0.70	0.70	0.80	0.69	0.90

Table 2 Neutron economy of ^{233}Th irradiation chain.

^{233}Th chain	FHR	HTR	RBMK	PHWR	MSR-FLIBE	MSR-FLI	HPLWR	LWR	MSFR-FLIBE	MSFR-FLI	LFR	SFR	GFR	MFBR	MCFR-NaCl	MCFR-AcCl
$\bar{\nu}$ average	2.50	2.50	2.50	2.50	2.50	2.50	2.50	2.50	2.52	2.53	2.53	2.53	2.54	2.53	2.53	2.53
$\bar{\nu}_{233\text{U}}$	2.50	2.50	2.50	2.50	2.50	2.50	2.50	2.50	2.51	2.52	2.53	2.53	2.54	2.53	2.53	2.54
^{232}Th fission prob. ($F_{232\text{Th}}$)	0.00	0.00	0.01	0.01	0.00	0.01	0.02	0.02	0.01	0.01	0.02	0.03	0.03	0.03	0.03	0.04
^{233}U fission prob. ($F_{233\text{U}}$)	0.90	0.90	0.90	0.90	0.89	0.89	0.89	0.89	0.87	0.87	0.89	0.89	0.89	0.90	0.90	0.91
^{233}U capt. prob. ($C_{233\text{U}}$)	0.10	0.10	0.10	0.10	0.11	0.11	0.11	0.11	0.13	0.13	0.11	0.11	0.11	0.10	0.10	0.09
^{232}Th D-factor ($D_{232\text{Th}}$)	-0.06	-0.14	-0.22	-0.22	-0.15	-0.12	-0.18	-0.16	-0.12	-0.20	-0.32	-0.36	-0.36	-0.35	-0.37	-0.43
^{234}U D-factor ($D_{234\text{U}}$)	0.27	0.30	0.25	0.21	0.32	0.36	0.30	0.28	0.35	-0.02	-0.58	-0.65	-0.65	-0.80	-0.75	-0.96
Synthetic Actinides Rel. capture (C_{Ac})	0.44	0.37	0.29	0.30	0.36	0.39	0.35	0.37	0.42	0.35	0.23	0.22	0.22	0.22	0.19	0.15
Neutron balance from Eq. (4): $\eta-2$																
$\eta-2$	0.25	0.25	0.25	0.26	0.24	0.22	0.23	0.23	0.18	0.19	0.25	0.26	0.26	0.29	0.28	0.31
+ 1 correction term	0.25	0.25	0.26	0.27	0.24	0.23	0.25	0.25	0.19	0.21	0.28	0.29	0.29	0.32	0.31	0.35
+ 1&2 correction terms	0.22	0.22	0.24	0.25	0.21	0.19	0.21	0.22	0.14	0.21	0.34	0.36	0.36	0.40	0.38	0.44
+ 1&2&3 correction terms	0.06	0.14	0.22	0.22	0.15	0.12	0.17	0.15	0.11	0.19	0.32	0.35	0.35	0.35	0.37	0.42
Neutron balance from Eq. (5): $\bar{\nu}-2$																
$\bar{\nu}-2$	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.50	0.52	0.53	0.53	0.53	0.54	0.53	0.53	0.53
+ 1 correction term	0.50	0.50	0.51	0.51	0.51	0.51	0.52	0.52	0.53	0.54	0.55	0.56	0.57	0.56	0.56	0.57
+ 1&2 correction terms	0.06	0.13	0.22	0.21	0.14	0.12	0.17	0.15	0.11	0.19	0.32	0.35	0.35	0.34	0.36	0.42
+ 1&2&3 correction terms	0.06	0.13	0.22	0.22	0.15	0.12	0.19	0.16	0.12	0.20	0.33	0.37	0.37	0.37	0.38	0.44
Neutron balance from Eq. (6): $\bar{\nu}$ - fission cost																
$\bar{\nu}$ - fission cost	0.06	0.14	0.23	0.22	0.15	0.12	0.18	0.16	0.13	0.21	0.33	0.36	0.36	0.35	0.37	0.43
+ 1 correction term	0.06	0.13	0.22	0.22	0.15	0.12	0.18	0.15	0.12	0.19	0.32	0.35	0.36	0.35	0.37	0.43
Neutron balance from Eq. (7): D-factor																
$-D_{232\text{Th}}$	0.06	0.14	0.22	0.22	0.15	0.12	0.18	0.16	0.12	0.20	0.32	0.36	0.36	0.35	0.37	0.43

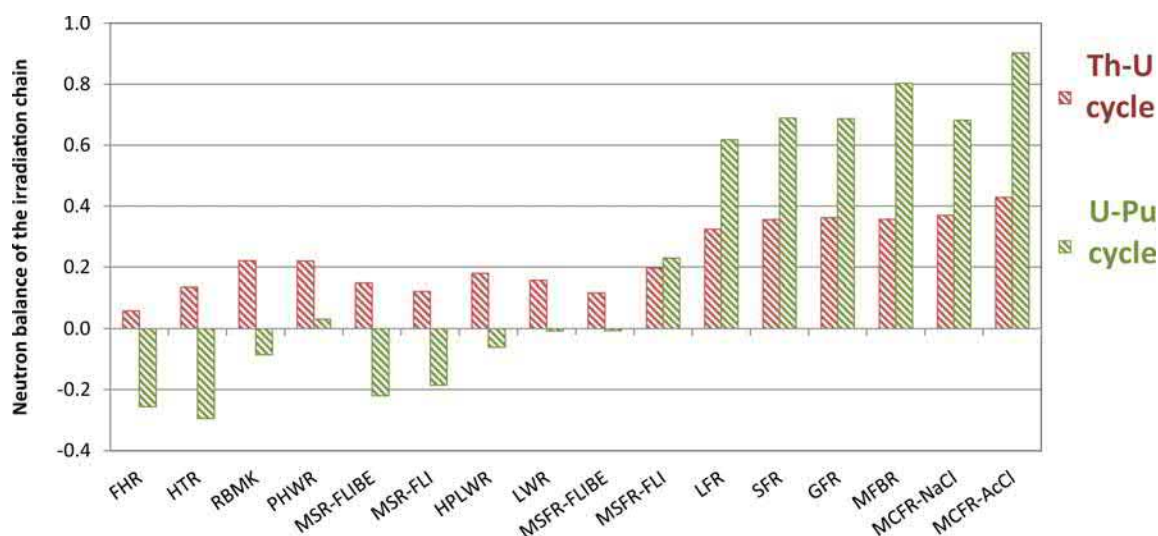


Fig. 11 Neutron balance of the ^{232}Th and ^{238}U irradiation chains. Positive numbers represent neutron excess, negative numbers neutron deficit.

burning reactors, which use synthetic actinides as a feed. However, also here if solely ^{232}Th or ^{238}U is used as a feed the resulting equilibrium represents the self-sustaining breeder. It is a characteristic of this method that the equilibrium isobreeding state is achieved also for reactors, which are subcritical in equilibrium and cannot act as a breeder in reality. The equilibrium state should be understood as an Eigen-state of the reactor and the equilibrium fuel composition as the Eigen-vector of the Bateman burnup matrix.

The knowledge of equilibrium parameters was used to assess the breeding ratio, as the foremost fuel cycle performance parameter, in each simulated reactor. The reactors can be classified into four categories based on their equilibrium breeding ratio. The second fuel cycle performance parameter, achievable burnup and its relationship with the breeding ratio, was discussed and the importance of absolute fissile fuel share was stressed. The lower achievable burnup in thermal spectrum reactors is often assigned to the higher FPs microscopic cross-sections. This is, nonetheless, only a minor contributor to the issue in comparison to the impact of the absolute fissile fuel share.

The initial fissile load for the open and closed cycle was addressed together with its relation to the achievable burnup in an open cycle. The means of criticality maintenance were not addressed as such; however possible reactivity variations in different reactors and fuel cycle types were discussed and the interconnection with breeding ratio illustrated. It is obvious that positive initial breeding gain can partly compensate for the neutron economy deterioration by FPs at the end of cycle. The transmutation capability was quantified by D-factors and by the reaction rate per atom. As expected, the transmutation of legacy synthetic actinides costs more neutrons in a thermal reactor. At the same time, the more efficient transmutation process in fast reactors can be slower.

Finally, the neutron economy of the actinides equilibrium composition, or actually of the ^{232}Th and ^{238}U irradiation chains is analyzed using four different neutron tallying methods. They all rely on the fact, that the fuel composition and reaction rates are stabilized in equilibrium. The general conclusion is that: (1) the self-sustaining breeding in closed cycle offers better neutron economy and higher achievable burnup in fast reactors; (2) the U-Pu cycle profits more from spectrum hardening and has better neutron economy, in which more neutrons are generated per fission but also parasitically captured; (3) the Th-U cycle is less sensitive to the spectrum hardening and has better neutron efficiency, in which the lower neutron generation per fission is compensated by lower parasitic captures; (4) breeding in thermal spectrum may result in prohibitively high fuel reprocessing frequency.

One potentially surprising conclusion of this study is that the better breeding in harder spectra is not related to the higher average number of neutrons emitted per fission. Even though, individual neutron yields are increasing with spectrum hardening, the changing fuel composition compensates for this effect. Hence, no matter in which reactor, in the U-Pu cycle there are roughly 2.9 neutrons emitted per fission, while in the Th-U cycle only about 2.5 neutrons are emitted per fission. The higher number of neutrons per fission, together with up to 15% of the fissions of the fertile ^{238}U that are induced by fast neutrons, is the biggest advantage of the U-Pu cycle. Unlike the number of neutrons emitted per fission, the direct fertile fuel fission probability as well as ^{239}Pu fission probability strongly depends on the spectrum. The ^{238}U fission is a threshold reaction, and the fission probability goes to zero below the threshold energy. The ^{239}Pu fission probability decreases steeply down to 60% with spectrum softening, whereas the ^{233}U fission probability stays constant around 90%. The rate of other synthetic actinides creation is thus three times higher for the U-Pu cycle in thermal reactors.

This is the third of a series of three chapters that is dedicated to the identification of the advanced reactor technologies that are capable of self-sustaining breeding when fueled with either ^{232}Th or ^{238}U and to the investigation of properties of the resulting equilibrium fuel cycles. The trivial conclusion of this study is that (1) both primordial nuclides ^{232}Th and ^{238}U can be utilized by self-sustaining breeding in closed cycle; (2) in case of ^{238}U , the B&B mode in an open cycle is also possible; (3) synthetic actinides, until

they are reasonable recoverable, should be considered as resource and catalyzer of the ^{232}Th and ^{238}U utilization; and (4) engineering and technological issues should be overcome and not allowed to discredit this natural resource.

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Advanced Modeling and Simulation Methods

Carlo Fiorina, Laboratory for Reactor Physics and System Behaviour, EPFL, Lausanne, Switzerland

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Glossary

CFD Computational fluid dynamics
CHF Critical heat flux
CMFD Computational multi-fluid dynamics
DNB Departure from nucleate boiling
DNS Direct numerical simulation
FEM Finite element method
FV Finite volume
HPC High-performance computing
LES Large eddy simulation
LWR Light water reactor
MSR Molten salt reactor
M&S Modeling and simulation
RANS Reynolds average Navier-Stokes
SFR Sodium fast reactor
SP3 Simplified spherical harmonics of third order
URANS Unsteady RANS
VLES Very large eddy simulation
V&V Verification and validation
1-D One dimensional
2-D Two dimensional
3-D Three dimensional

Introduction

The field of modeling and simulation (M&S) is dedicated to devising the mathematical models that best describe a physical phenomenon, implementing them into a computer code, deploying them on suitable computer architectures and applying them to specific problem simulation.

In terms of models, the field of nuclear engineering has experienced a quick progress in its early decades (Ronen, 1986). However, the growing availability of computational resources, together with the decreasing number of available experimental facilities, is increasingly leading the community towards a design-by-simulation paradigm. In turn, the higher fidelity and the reduced uncertainty margins required by this design strategy are gradually transforming the field of M&S. In addition, significant progress is observed in relatively new fields such as multiphysics, and in cross-disciplinary fields like Computational Fluid Dynamics (CFD) and Machine Learning.

In terms of computer codes, M&S of advanced nuclear reactors has been closely related to that of conventional reactors. In the past, tools were mainly developed for the traditional light water reactors (LWRs) that constitute the majority of the Gen I – III reactors (Brown, 2021; Hamon, 2021; Shwageraus, 2021; Stanculescu, 2021), and subsequently adapted to the analysis of reactors relying on other coolants than water, as well as on innovative nuclear fuels and fuel cycles. This approach allows strongly limiting development efforts, as well as verification and validation (V&V). However, the relatively old programming languages and paradigms employed in legacy codes often make their modification time consuming. In addition, code extension often results in tools with only limited capabilities for the analysis of non-LWR systems. Nowadays, progress in the field of Computational Science has made the effort of developing new dedicated codes comparable to that of modifying legacy codes, and the two strategies tend to go hand in hand. The choice has become a trade-off between flexibility and computational performances on one side, and robustness and reliability on the other. It often depends on the availability of in-house legacy codes, on the type of expertise of the developing team, on the importance of high-performance computing (HPC) in the target reactor technology, on the target reactor technology itself, and on the scope of the work (research or licensing).

This chapter focuses on models and codes employed for nuclear reactor design and for the analysis of its steady-state, operational transients and design-basis accidents.¹ No explicit discussion is provided about models for probabilistic risk assessment (Modarres and Kim, 2021), fuel-coolant interaction, core melting, containment analysis, integrity, release of radionuclides (Singh, 2021), transport of radio-nuclides in the atmosphere, radiation protection (Pevey, 2021), specific material performance, etc. Such extended scope would vastly transcend the space allotted to this Chapter. Additional relevant information is provided by Martin (2021), and Kirkland (2021).

Models and algorithms

M&S of nuclear reactors is a multidisciplinary field that requires the investigation of several different and interrelated phenomena. Many phenomena, such as neutron transport, fuel cycle, and fuel behavior, are specific to nuclear reactor analysis. In other cases, cross-disciplinary M&S methodologies are employed, sometimes tailored to the needs of the nuclear sector.

Neutron transport

The objective of neutron transport studies is most often to predict the spatial distribution and temporal evolution of fission power in a reactor core. Other typical applications are the evaluation of irradiation damage in structural materials, neutron shielding and neutron noise analyses.

The transport of neutrons and their interaction with matter is governed by the Boltzmann transport equation (Stacey, 2007). It can be modeled with minimal approximations by using Monte Carlo methods, i.e., by simulating a large number of random neutron histories (Stacey, 2007). The complex energy and angular dependences associated with the interaction of neutrons with matter challenge instead the use of deterministic methodologies. Several approximations have been proposed in the past to tackle this problem. A first important approximation applies to neutron energy and consists in subdividing the energy spectrum in one or multiple groups, each one with its own average energy and effective cross-sections (multi-group approach). A second approximation is then necessary to treat the angular dependence of the neutron flux and of its interaction with matter. In the method of discrete ordinates, neutrons are subdivided in a certain number of groups based on their direction. In the method of spherical harmonics, the neutron flux is instead expanded in a series of anisotropic functions (the spherical harmonics). A limiting case for spherical harmonics consists in assuming an isotropic neutron flux, which results in the diffusion approximation. In the method of characteristics, the multi-group transport equations are written in terms of linear characteristics curves, traditionally in two dimensions (2-D). Finally, the collision probability method, which is mainly employed for the generation of effective cross-sections, builds an algebraic matrix by evaluating the probability for a neutron born (isotropically) in any of the regions of the core lattice to undergo its first collision in another region.

All these methods have been successfully applied to both traditional and advanced nuclear reactors. Neutron diffusion remains the workhorse for the nuclear community, especially for transient analyses. It provides reasonable approximations for most reactor concepts, with very limited computational requirements. However, the availability of large computational resources is driving the research efforts towards the use of more complex models. For instance, new codes are being developed for the 3-D application of the

¹A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.

method of characteristics (Tada et al., 2011; Kouchunas et al., 2015), and significant efforts are being spent towards more efficient and scalable algorithms for discrete ordinate equations (Ragusa and Wang, 2010).

Relevant advances are observed in the field of Monte Carlo analysis, whose intrinsic parallel scalability allow fully benefiting from large HPC facilities. Mainly employed in the past for criticality and shielding calculations, Monte Carlo methods are gradually becoming: an important tool for the generation of multi-group cross-sections; a viable tool for analyzing short transients, such as super-prompt-critical power excursions, with ongoing efforts aimed at modeling delayed neutron precursors in longer transients; and a powerful tool for sensitivity analyses, also thanks to the development of a dedicated perturbation theory (Aufiero et al., 2015a). Finally, while Monte Carlo codes have been historically based on combinatorial geometries, methodologies have been developed to allow employing general unstructured meshes (Leppänen and Aufiero, 2014).

Beside the evident trend towards computationally expensive solutions, new methods have also been developed for specific applications. A good example is that of Molten Salt Reactors (MSRs) (Křepel and Kramer, 2021; Ignatiev, 2021), whose liquid fuel requires modeling the transport of the delayed neutron precursors. To this purpose, many legacy codes have been modified (Křepel et al., 2007), and new codes created (Aufiero et al., 2014a), that include a transport term in the precursor equations. Dedicated methods have also been developed for Monte Carlo codes to be able to evaluate the kinetics parameters with moving precursors (Aufiero et al., 2014b). Another example of development of reactor-specific methodologies relates the modeling of the neutronic impact of core deformations in fast reactors. While traditionally this effect was modeled by parametrizing the cross-sections, and in some cases by uniform radial and axial expansions of the core, modern moving mesh algorithms are currently being developed to allow for a detailed modeling of deformations (see Section “Sodium fast reactors” of this chapter). Another approach to the problem is based on a fixed mesh, but with perturbed equations (Jing and Yang, 2019).

Fuel cycle analysis

Methods for fuel cycle calculations can be subdivided into two main categories. Burnup methods aim at predicting the isotopic composition of the nuclear fuel during irradiation. Based on these results, a second category of methods aims at predicting the evolution of nuclear materials in a reactor fleet, possibly optimizing the deployment of the fleet based on some constraints and figures of merit.

The Bateman equations (Cacuci, 2010) describe the evolution of isotopes in a nuclear reactor. Burnup calculations are based on the assumption that nuclide concentrations can be assumed as constant when solving for the neutron density distribution, and vice versa. As a result, neutron transport and Bateman equations are solved sequentially and iteratively. One has to perform time-independent neutron transport calculations to obtain the various average isotopic production and removal rates, depending on the fuel batch and position in the core. Based on these reaction rates, one can construct and solve the Bateman equations (burnup step). This iterative strategy can be refined by means of predictor–corrector methods for improving the evaluation of the constant reaction rates employed during the burnup step (Yamamoto et al., 2009; Josey et al., 2017).

Unfortunately, the presence of isotopes with very different production and removal rates tends to make the system of Bateman equations extremely stiff, which rules out the use of common methods for ordinary differential equations. The method of matrix exponential is then commonly employed, with several different approximations. A traditional one involves simplifying the Bateman equations by excluding the shortest-lived isotopes, and solving the resulting matrix exponential by using a truncated Taylor series. More recently, the Chebyshev rational approximation method has been demonstrated to work well for large burnup problems while relaxing the need for decay chain simplifications (Pusa and Leppänen, 2013).

A specific case for burnup calculations is that of MSRs. On the one hand, thanks to the liquid fuel, average fluxes and cross-sections do not need to be calculated in several different regions. On the other hand, MSRs often feature complex on-line refueling and/or reprocessing. To simulate this, one needs to modify the Bateman equations to include fictitious production/removal decay constants (Nuttin et al., 2005). Another option is that of employing traditional codes for fuel cycle analysis, but modifying fuel compositions at each burnup step (Powers et al., 2013).

Fuel cycle analyses of pebble-bed reactors (Reitsma, 2021) is another domain that has experienced significant recent advances. The double heterogeneity of the fuel (particles in a pebble, and pebbles in the core) requires special treatments for cross-section generation when employing deterministic methodologies (Rütten et al., 2012). In addition, the flow of the pebbles requires a modified burnup routine. Traditionally this was modeled by radially subdividing the core into “channels,” and the channels into axial zones with different batches of pebbles moving from one axial zone to the next. In recent years, cross-section generation is more often performed by directly applying Monte Carlo methods based on the Woodcock delta-tracking method and employing a stochastic treatment for double heterogeneities, with discrete-element methods for the generation of realistic pebble beds (Rintala et al., 2015).

Fuel behavior

Fuel behavior analyses have two main objectives: investigating the fuel performance at nominal conditions during its residence time in a reactor; and, starting from a certain fuel state, predicting the behavior of the fuel during operational and accidental transients. Although separate codes have often been developed in the past for these two types of analyses, there is a recent tendency to develop tools that are suited for both (Van Uffelen et al., 2018).

Fuel behavior investigations require simulating: temperatures, strains and stresses in fuel and cladding; gap conductance; the thermo-mechanical contact between fuel and cladding; the evolution of major isotopes in the fuel; the generation and release of fission gases; the chemical interaction among fuel, fission products, cladding and coolant; and a myriad of specific phenomena like thermal and irradiation creep, fuel swelling, fuel densification, cladding axial growth, cracking, relocation, formation of high-burnup structures, etc.

Historically, fuel behavior analysis has been performed by following a so-called 1.5-D approach (Van Uffelen et al., 2018). According to this approach, the fuel is subdivided into axial sections, each one featuring a 1-D radial discretization for the thermo-mechanical equations. These 1-D sections are then axially coupled in a loose way by means of a common model for the pressure in gap and plena. The reason for these simple models resides only partly in their limited computational requirements. From an engineering perspective, the nuclear fuel is characterized by such a complex phenomenology that reliable experimental correlations acting on simple mathematical models can turn out to be preferable over complex models with limited empirical content. In addition, experimental data are often limited to fuel centerline temperatures and gas pressures, thus not providing a calibration and validation base for multi-dimensional codes.

Legacy 1.5-D tools have often been developed for LWR analysis. However, the computational framework is applicable to general pin-based nuclear reactors and several legacy codes have then been extended to advanced reactor systems. A major challenge remains in the development of the related experimental correlations and models.

In addition to 1.5-D tools, some efforts have been spent in the past towards simple 2-D models (Rashid et al., 2014). More recently, the growing computational resources are pushing for the development of multi-dimensional codes (Sercombe et al., 2009; Williamson et al., 2012; Soba and Denis, 2015; Scolaro et al., 2020). An objective is to complement legacy codes for the analysis of non-symmetric cases (viz. missing pellet surface, or non-symmetric power/cooling, notably in boiling water reactors). The multi-dimensional capabilities also allow for more mechanistic simulations, which in turn provide scientists with a better understanding of the various phenomena at play.

Multi-dimensional codes tend to relax the need for experimental correlations, thus facilitating their application to unconventional fuels and advanced reactors. In this sense, some of the recent multi-dimensional codes (Williamson et al., 2012; Scolaro et al., 2020) have been based on general libraries for the solution of partial differential equations, which allows for an easier adaptation to non-traditional fuels like the TRISO-based ones (Hales et al., 2013; Helmreich, 2021).

Thermal-hydraulics and system behavior

The thermal-hydraulic analysis of a reactor is a challenging task (Todreas and Kazimi, 2001; Kirkland, 2021). The geometrical complexity of a reactor core often forces to employ significant simplifications, which combines with the well-known difficulties in modeling turbulence and, for certain reactors, two-phase flows. Different methods can be used depending on the level of detail required by the simulation and on its objective.

System codes

By system code one normally means a primarily one-dimensional (1-D) thermal-hydraulic code capable of modeling the entire reactor system, including primary, intermediate and secondary loops. These codes are normally based on 1-D finite volume (FV) or finite difference discretizations of the Navier-Stokes equations, and/or of the Euler-Euler multiphase flow equations (typically 6 equation models). The reactor core is often modeled using multiple parallel 1-D channels. These codes can be used for both steady-state and transient analysis, including accidental situations. For this reason, they normally include relatively simple models for neutron kinetics and fuel temperatures.

System codes have been among the first tools being developed for nuclear reactor analysis, and they still represent today the workhorse of the community. In particular, the decades-long calibration, verification and validation and the heavy use of experimental correlations makes them the preferred choice for reactor licensing. Recent activities have focused on improving their multi-physics capabilities via inclusion of improved models for fuel (US NRC, 2007), as well as via a coupling with external codes for neutron transport. In addition, simplified three-dimensional (3-D) capabilities have gradually been introduced.

System codes have typically been developed for LWRs, and several activities have been carried out to extend their capabilities, notably for fast reactors (Mikityuk et al., 2010; Kruessmann et al., 2015; Nikitin and Fridman, 2018). More recently, system codes have started to be adapted to MSRs and the movement of delayed neutron precursors (Leandro et al., 2019).

Porous-medium approach

Porous-medium modeling (Todreas and Kazimi, 2001) is a common engineering approach for simulating the flow of a fluid in complex structures. Porous-medium equations can be obtained by volume-averaging the Navier-Stokes and energy equations. This results in: an additional porosity term, indicating that only a fraction of the volume is occupied by the fluid; and additional source terms that describe the interaction of the fluid with the subscale structure (viz., the fuel pins in a reactor core). The porous-medium equations revert to 1-D “system code” equations when restricted to one dimension. In this sense, porous medium models can be considered as a multi-dimensional extension of system codes. As a matter of fact, energy and momentum sources are often modeled based on the same correlations for Darcy friction factors and Nusselt numbers that are used in system codes.

The porous medium approach, in its general form, has experienced a limited success in the nuclear community. However, this approach has the advantage of being applicable to general unstructured meshes, which makes it a suitable option for recent

multiphysics codes based on FV or finite element (FEM) discretizations and standardized mesh treatments. In addition, the porous-medium equations reduce to standard Navier-Stokes equations in clear fluids, when porosity is equal to one and the source terms to zero. This allows for a unified, implicitly coupled treatment of pool (or plena) and core thermal-hydraulics. This possibility is particularly interesting for simulating the setup of a natural circulation regime in sodium fast reactors (SFRs) (Heidet, 2021).

Another field of application for the porous-medium approach is the analysis of pebble-bed reactors. In these reactors, the core structure is essentially isotropic and homogeneous, which makes the porous-medium treatment a natural choice for their thermal-hydraulic analysis.

Sub-channel codes

The sub-channel approach (Moorthi et al., 2018) is a special case of the porous-medium approach where the control volumes are standardized to a specific layout. This approach is normally employed for pin-based nuclear reactors. Radially, sub-channels are typically the size of a single fuel rod and associated coolant. Different sub-channels can be used for different positions in an assembly, as shown in Fig. 1 for the specific case of an SFR assembly. In the case of square pin lattices, either rod-centered or coolant-centered square sub-channels can be employed. The use of standardized and properly selected nodes has several benefits over the use of possibly unstructured control volumes in a general porous-medium approach. For instance, use of a triangular control volume in a hexagonal lattice (Fig. 1) allows employing one single expression for radial momentum and energy exchange between control volumes, while the use of an unstructured mesh and Cartesian coordinates would require at least two different expressions. In addition, one can choose volumes that correspond to physical sub-channels with different properties. Finally, and notably for rod-centered treatments, one can associate the power of each single rod to its own channel, thus fully resolving the heterogeneities inside an assembly. As a drawback, sub-channel codes are often based on dedicated algorithms, instead of standardized treatments for unstructured meshes. This requires a time-consuming set-up of code-specific interfaces when coupling them to other codes.

Computational fluid dynamics - single phase

CFD is a vast domain that extends far beyond nuclear energy. It can in principle be referred to the any computational methodology for fluid flows. Often however, and especially in the nuclear field, CFD refers specifically to fine-mesh multidimensional algorithms for detailed fluid flow analyses. This allows clearly distinguishing it from 1-D system codes and from the porous-medium or sub-channel approaches.

Laminar flows can be solved without relevant approximations for problems of engineering interest. On the other hand, the Direct Numerical Simulation (DNS) of Navier-Stokes equations in turbulent cases requires special numerical treatments and it is extremely requiring from a computational perspective. Two main approximations have been devised to simulate turbulent flows, namely: Reynolds Average Navier-Stokes (RANS) and Large Eddy Simulation (LES). Both approaches aim at filtering out part of the turbulent time and length scales. The RANS approach is based on a time average of the Navier-Stokes equations. The resulting

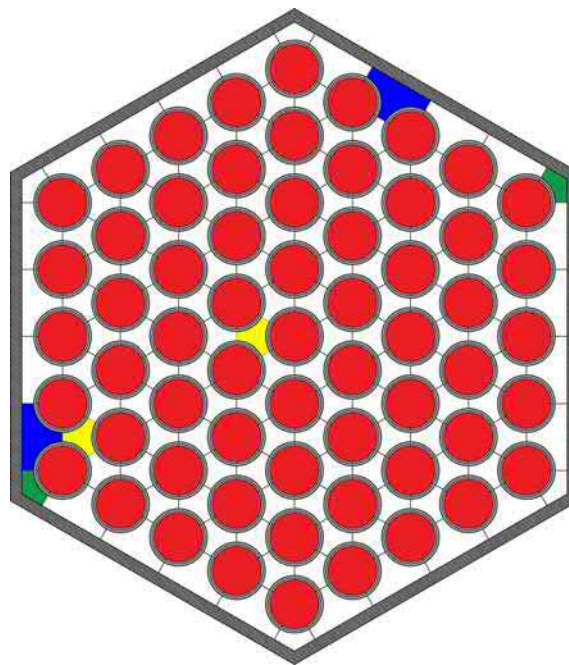


Fig. 1 Example of sub-channels for a SFR fuel assembly. Different types of sub-channels are marked in *blue*, *green*, and *yellow*. Courtesy of Chirayu Batra, IAEA.

approach has limited computational requirements but also significant shortcomings, for instance in case of flow detachments. In addition, due to the time averaging, the RANS approach is in principle unsuited for unsteady flows. However, a time dependence is often retained for simulating transients that are slow compared to turbulent time scales (viz., average velocity and temperature changes during a loss-of-flow accident). The LES approach is instead based on a spatial filtering of the equations and allows simulating unsteady flows. It normally features a higher accuracy, but at the cost of a much higher computational footprint, often calling for special numerical treatments such as the spectral element method (Fischer et al., 2008).

The available computational resources are making CFD calculations common in the nuclear engineering community. RANS analyses are employed for the analysis of plena and pools (Moreau et al., 2019), spacer grids, fuel assemblies (Roelofs et al., 2019), as well as for standard CFD applications like pipes, pumps, and heat exchangers. RANS is also becoming the reference approach for modeling the fluid flow in fast-spectrum MSRs.

LES analyses are typically aimed at smaller application like T-junctions, cross-flow over obstacles and parallel shear flows (Simoneau et al., 2010). However, some studies for entire fuel assemblies are already being performed on large HPC facilities (Merzari et al., 2017). Other LES applications include the local fluid flow in pebble-bed reactors (Shams et al., 2014) and the salt flow in the core of fast-spectrum MSRs. In the case of MSRs, the use of LES can help identifying potential neutronic effects of fluid flow instabilities, as well as improving predictions on flow detachment and stagnation points. Thanks to the capability to investigate unsteady flows, LES is also a reference choice for fluid-structure interaction studies aimed at predicting local vibrations, such as for the fretting of space grids (Bakosi et al., 2013).

DNS analyses are mainly limited to simple geometries such a single backward facing step. They often aim at creating reference solutions for benchmarking RANS and LES solutions, notably for cases that are specific to the nuclear engineering field. An example is that of low-Prandtl number flows (Oder et al., 2019). Another typical application is the investigation of boundary layers, including for instance the effect of surface roughness (Chatzikyriakou et al., 2015).

Computational fluid dynamics - multiphase

Similar to CFD, multiphase CFD (sometimes referred to as Computational Multi-Fluid Dynamics, or CMFD) is employed in several fields of science and engineering. In nuclear, the application of multiphase CFD is mainly restricted to two-phase flow simulations. There is an impressively large set of available methods (Bestion, 2012; Kharangate and Mudawar, 2017), but one can broadly distinguish between Eulerian and Lagrangian methods, as well as between single- and multi-fluid methods. An additional category is that of lattice Boltzmann methods. These broad methodologies include several sub-methods. For instance, single-fluid Eulerian methods can further be distinguished in the way of tracking and reconstructing the interface between liquid and vapor. Multi-fluid methods can simply model the liquid and vapor fractions, or include multiple “fields” to model continuous and dispersed phases, as well as multiple groups of fields depending e.g., on the size of bubbles. Multiphase CFD method can be further subdivided based on the turbulence treatment: RANS, LES, and DNS. A general classification of Eulerian multiphase CFD models has been proposed in (Bestion et al., 2014) and is reported in Table 1.

There are several possible applications of interest for multiphase CFD in the nuclear field (Bestion et al., 2014). An important example is the prediction of Critical Heat Flux (CHF) for both the Departure from Nucleate Boiling (DNB) and the dryout in LWRs (Yadigaroglu, 2014). Single-fluid and lattice Boltzmann methods are both used to investigate the details of bubble growth and coalescence. Two-fluid models are less computationally expensive and are typically employed for larger-scale predictions at a channel level (Brewster et al., 2019). Some major advances in the field of multiphase CFD are associated to CHF predictions, with activities on the improvement of closure relations for multi-fluid models (Baglietto et al., 2018), as well as for the creation of improved and larger-scale models for single-fluid prediction of local phenomena (Sato and Niceno, 2018).

Table 1 Time and space resolution in the various modeling approaches of Eulerian two-phase CFD.

	Pseudo DNS	LES with simulated interfaces	Hybrid LES with filtered & statistical interfaces	LES with statistical interfaces	RANS URANS with statistical interfaces	Porous medium approach with statistical interfaces
<i>Medium</i>	Open	Open	Open	Open	Open	Porous
<i>Time & space filtering</i>	No filter, No averaging	Space Filtering	Space Filtering	Space Filtering	Time averaging	Time averaging
<i>Turbulence</i>	DNS	LES	LES, VLES ^a	LES, VLES	RANS, URANS ^b	RANS, URANS
<i>Interfaces</i>	Simulated	Simulated	Filtered Statistical	Statistical	(Filtered) Statistical	Statistical
<i>Number of fields</i>	1	1	1, 2, n	1, 2, n	1, 2, n	1, 2, n

^aVery Large Eddy Simulation.

^bUnsteady RANS.

Modified table from Bestion D, Anglart H, Mahaffy J, Lucas D, Song CH, Scheuerer M, Zigh G, Andreani M, Kasahara F, Heitsch M, Komen E, Moretti F, Morii T, Mühlbauer P, Smith BL, and Watanabe T (2014) *Extension of CFD Codes Application to Two-Phase Safety Problems – Phase 3*, NEA/CSNI/R(2014)13.

Multiphysics

Multiphysics refers to the coupled simulation of different physical phenomena. In the field of M&S, different physical phenomena are sometimes referred to as “physics.” A typical example in nuclear engineering is the coupling of neutron transport with core thermal-hydraulics. Multiphysics simulations can be categorized in several ways:

- One- or two-ways coupling. In one-way coupling, it is assumed that one phenomenon impacts the other, but not vice versa. This is for instance the case of zero-power reactors: a change in temperature in the moderator will affect the neutron balance, but the corresponding power variation will not significantly affect the temperature of the moderator. In two-ways coupling instead the two phenomena affect each other, like neutronics and thermal-hydraulics in nuclear power plants.
- Internal or external coupling. In the case of internal coupling, the same code is employed for simulating two different phenomena. This is for instance the case of system codes with built-in models for point kinetics and fuel behavior. In external coupling, two or more existing codes are coupled via an interface. This allows employing existing codes, but with a higher risk of computational bottlenecks. The coupling interface can be part of one of the two codes, or an additional independent code. It can operate at the level of the computer RAM, or of the hard disk (for instance by creating and exchanging text files).
- Explicit or implicit coupling. In explicit coupling, one single and separate calculation (or one per time step) is performed for the single-physics models. It is suitable for one-way couplings, or for transient simulations with very small time steps. In case of implicit coupling, models are instead solved so that the interdependence of the coupling variables is fully resolved (at each time step, in case of time-dependent simulations). Implicit coupling can be obtained by iteration, or by directly constructing and solving a block coupled linear matrix. It should be preferred over explicit coupling in two-ways couplings.

System codes often allow for two-ways, internal, implicit couplings of thermal-hydraulics, neutronics and fuel behavior ([US NRC, 2007](#)). However, these neutronic and fuel behavior models are often highly simplified and significant efforts have been dedicated in the past to the coupling of system codes with models for spatial neutron kinetics ([US NRC, 2020](#)). In general, coupling of legacy codes has represented the most common methodology to achieve multiphysics simulations. It allows employing validated, reliable codes while limiting potential sources of errors to fairly simple interfaces. Today, it represents the reference methodology for the safety analysis of LWRs ([Chauliac et al., 2011](#); [Michal and Kothe, 2011](#); [Valtavirta, 2018](#)). It is also employed sometimes for first-of-a-kind implementations of novel methodologies, such as the coupling of Monte Carlo codes with finite-volumes codes ([Aufiero et al., 2015b](#)), sub-channel codes ([Ferraro et al., 2019](#)), and fuel behavior codes ([Valtavirta et al., 2017](#)).

However, external code coupling has some disadvantages: it is prone to computational bottlenecks; it makes it difficult to implement advanced coupling methodologies such as high-order time integration, sub-stepping, block-coupled matrix solution, etc.; and it limits the overall computational performance to that of the least-performing single-physics code. For these reasons, the last decade has witnessed a flourishing of activities dedicated to the development ab initio of new numerical platforms for the multi-physics analysis of nuclear reactors ([Gaston et al., 2009](#); [Clifford, 2013](#); [Fiorina et al., 2015](#)). Considered prohibitive in the past, the development of new codes is nowadays streamlined by some important advances in the field of Computational Science. Modern programming languages (e.g., C++, Python, Java) and paradigms (viz., object-oriented programming) allows for a quick implementation and an effective structuring of very large codes, while strongly limiting maintenance and simplifying code modification. In addition, the availability of modern numerical libraries (e.g., OpenFOAM, PETC, libMesh) allows focusing on high-level equations and algorithms, while employing available routines for discretization, matrix solution, mesh-to-mesh projection, mesh deformation, mesh search, etc. This approach carries with itself the complimentary benefit of employing highly optimized, massively-parallel, and verified routines for the most complex and computationally requiring parts of a code (viz., mesh management, discretization, matrix solution).

Transversal trends

Some important trends are observed throughout the different modeling domains, namely: the development of codes that can exploit the potential of HPC; the development of surrogate models; the increasing use of machine learning and data science methodologies; and the growing role of open-source software.

The growing computational resources have been driving significant research activities aimed at improving the parallel scalability of legacy codes, as well as at writing new codes aimed at exploiting the full potential of HPC. When writing new codes, a clear tendency is observed to make use of available numerical libraries for scientific computing, with an increasing use of methodologies based on FEM and FV discretizations on unstructured meshes. A recent challenge when developing tools for HPC applications is the growing use of vector and graphical processing units. To exploit these hardware components, numerical algorithms should be developed to allow for vectorization, i.e., for performing the same operation simultaneously on a certain number of processing units. However, many algorithms employed in the nuclear field are not suitable for this kind of computation. Typical examples are history-based transport algorithm for Monte Carlo codes, as well as common matrix preconditioners and solvers used in FEM and FV methods.

Despite the growing computational resources, the intrinsic limits of the integrated circuit transistor technology are expected to gradually lead to higher costs for increasing computing power. For this reason, the nuclear engineering community is also looking at developing light-weight (surrogate) models that could be used for the several calculations required e.g., for reactor design, sensitivity analyses, uncertainty quantification, data assimilation, etc. Different from the past, where faster models were mainly obtained via

simplifications in the physics of the problem, the community is now increasingly looking at reducing the computational footprint by applying general mathematical models. An example is the use of projection-based reduced order models (Lorenzi, 2019), that have experienced a good success in the CFD community and that are currently becoming popular e.g., in neutronics, fuel behavior and multiphysics. Another notable example is the use of machine learning techniques, most typically deep artificial neural networks.

In general, data science and machine learning methodologies are gaining increasing attention. Main reasons for this are: the impressively fast progress that these fields are experiencing; their accessibility in terms of theoretical background; and the availability of several user-friendly libraries and tools that allow inexperienced users to quickly develop relatively sound and efficient models. Fields of applications include surrogate models, regression models for physical and chemical properties, closure relationships, fault detection and forecast.

Another important trend in the field of nuclear science and technology is that towards open-source software. Following successful examples e.g., in the fields of CFD (OpenFOAM, 2020) and machine learning (Sonnenburg et al., 2007), open-source and collaborative software development are increasingly perceived as a way to accelerate innovation by enlarging the bandwidth for productivity, limiting work duplication, and multiplying V&V efforts (Fiorina et al., 2020).

Examples of applications

The methods discussed in the previous section are routinely used for the analysis of existing reactor systems. Two types of advanced reactors are presented in this section as boundary cases to illustrate how different methods and approaches can be employed for specific applications. In particular, SFRs represent a reactor type with a relatively traditional design and a very high technology readiness level. MSRs on the other hand are a good example of a non-traditional reactor concept that challenges the use of standard methodologies and codes.

Sodium fast reactors

SFRs are arguably the advanced reactor technology that is closer to large scale commercial exploitation. Several methodologies and codes have been developed in the past by various institutions worldwide for investigating their steady-state and transient behavior. Recent efforts have been focused on both the advanced coupling of legacy codes and on the development of new platforms. In addition, new tools have been developed for a higher-fidelity investigation of some safety relevant phenomena.

Two examples of advanced code systems based on legacy codes are the FAST code system (Mikityuk et al., 2010) employed at the Paul Scherrer Institut (Switzerland), and the MATHYS code (Gerschenfeld et al., 2017) developed at the CEA (France). The FAST code system results from the integration and coupling of several state-of-the-art codes, as shown in Fig. 2. It includes the ERANOS code system for static neutronics calculations (Doriath et al., 1993), as well as a coupling of TRACE, PARCS and FRED (Joo et al., 1998; US NRC, 2007; Mikityuk and Fomitchenko, 2000) for neutron kinetics, thermal hydraulic, and fuel transient analysis, respectively. Besides coupling and interfacing a full set of codes for fast reactor analysis, many of these codes have been extended or modified. As an example, the EQL3D procedure (Křepel et al., 2009) has been developed on top of the ERANOS code for 3-D equilibrium fuel cycle calculations, while the TRACE code has been extended to two-phase sodium flow (Chenu, 2011). The FAST code system has been successfully employed for the analysis of several SFR design, including Phenix, Superphenix, ASTRID and the European Sodium Fast Reactor.

The MATHYS code is an example of integration of thermal-hydraulics codes aimed at an improved simulation of the SFR primary pool in specific scenarios. An example of such a scenario is the passive decay heat removal by natural circulation, where complex natural convection patterns set up that include thermal stratification in the pool and inverse flow in the inter-assembly gaps. To properly model the interaction between the flow in the assemblies, in the inter-assembly gaps and in the pools, the MATHYS code couples the CATHARE system code, the TrioMC sub-channel code, and the TrioCFD code for RANS-based CFD. In particular, TrioMC and TrioCFD are coupled through their boundaries, while both codes are coupled with CATHARE by domain overlapping. In typical applications, the overall dynamics of the system is handled by CATHARE, with TrioMC that is used for a detailed modeling of the assemblies, and TrioCFD that is used for detailed simulations of part of the pool (e.g., the outlet jets of core and heat exchangers) and/or of the flow in the inter-assembly gaps.

Beside its integration in MATHYS, the TrioMC code has also been provided with a new scheme for general polyhedral meshes (Gerschenfeld et al., 2019). This extension allows employing it as a CFD code, which in turns allow for a coupled simulation of assemblies, inter-assembly gaps, and pools without the need of an external coupling with TrioCFD. Fig. 3 shows an example of such simulation for an SFR: in the same simulation, a sub-channel approach on a structured mesh is employed for the core while standard CFD on an unstructured mesh is employed for the pools. Similar efforts have also been carried out for adapting CFD codes to porous-medium or sub-channel simulations to obtain unified coarse/fine mesh treatments (Fiorina et al., 2015).

Another field of development in the analysis of SFRs is related to the simulation of reactivity variations associated with core deformations. Prediction of these effects is traditionally obtained by assuming a uniform radial and axial expansion of the core based on average diagrid and fuel temperature, respectively. Multi-group cross-sections are then parametrized based on the average expansion of fuel and diagrid. An example of this state-of-the-art treatment of reactivity effects is the above-mentioned FAST code system. Recent research activities have tried to step beyond this approach by employing thermo-mechanical solvers, and by using

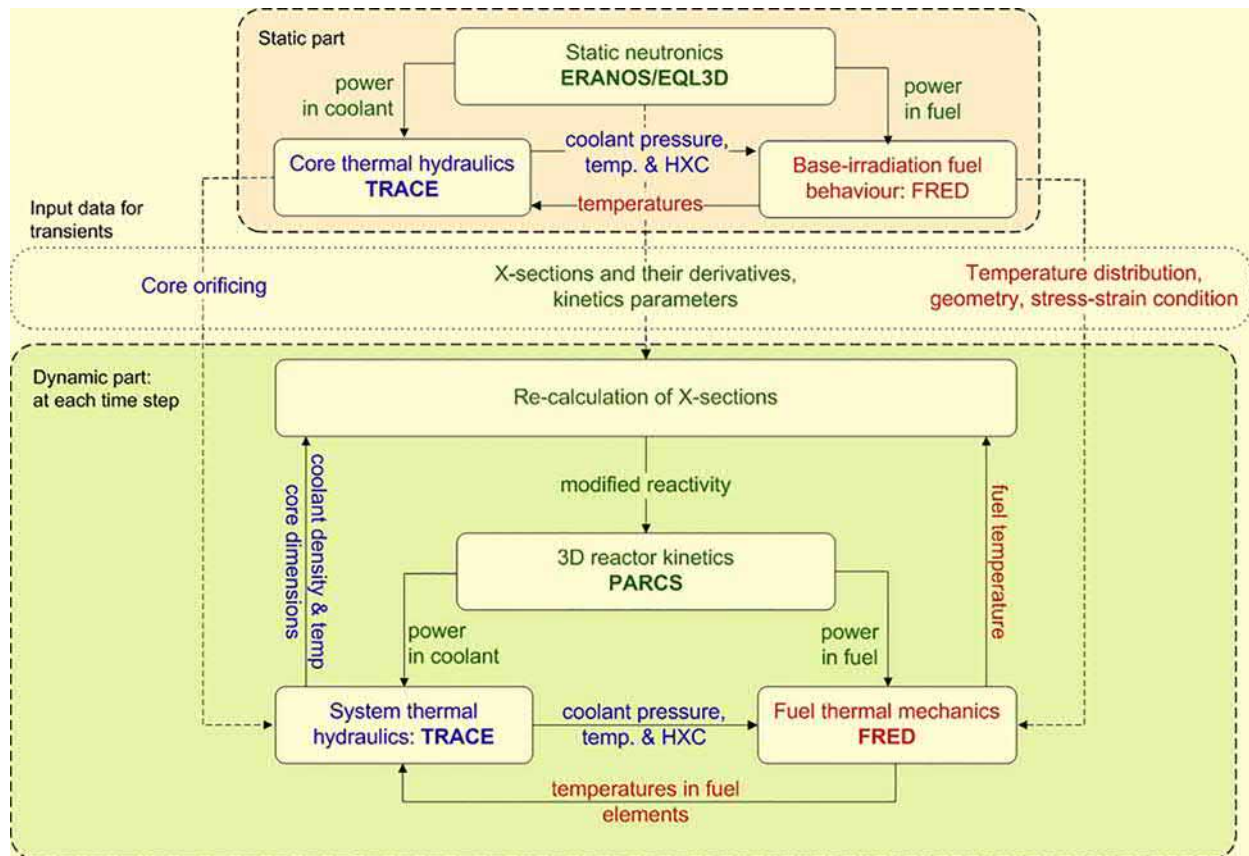


Fig. 2 FAST code system flowchart (Mikityuk et al., 2010).

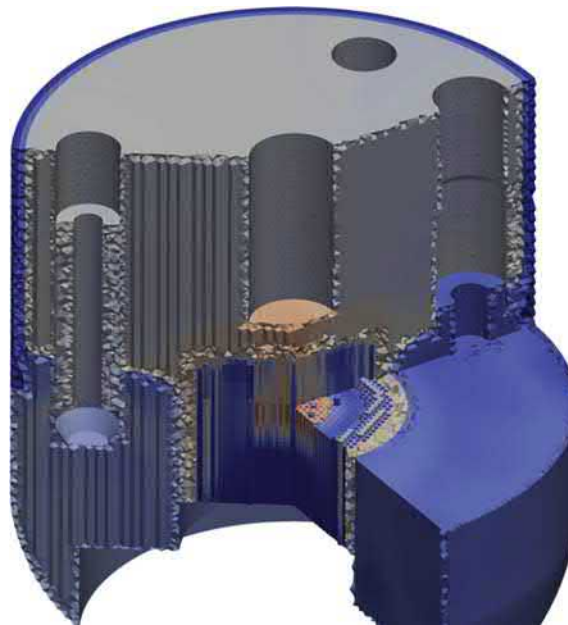


Fig. 3 Mixed sub-channel/CFD simulation of a SFR obtained using TrioMC (Gerschenfeld et al., 2019).

the resulting deformation field to modify the geometry and mesh in a neutronic solver. A development effort in this sense has been carried out based on the SHARP toolkit, which allows to couple advanced codes for thermal-hydraulics, neutronics and thermal-mechanics. It has been used to achieve a detailed simulation of expansion reactivity effects in a SFR mini-core (Merzari et al.,

2015). A similar work has been performed by using a modern numerical library for PDE solution as a means to develop a single code that includes both neutronic and thermo-mechanical capabilities (Fiorina et al., 2015, 2019). The code has been used to predict core flowering, and a hypothetical core compaction, in the European Sodium Fast Reactor (Fig. 4).

Molten salt reactors

Liquid-fuel MSRs present several modeling challenges, including the continuous salt mixing, the possibility of online reprocessing and refueling, the internal heat generation in the fluid, and the transport of the delayed neutron precursors.

The continuous salt mixing and the online reprocessing/refueling require special tools for fuel cycle analysis. An advanced example of such tools has been presented in (Aufiero et al., 2013) and was obtained by modifying the well-known Serpent 2 Monte Carlo code (Leppänen, 2007). The use of a Monte Carlo code is favored in MSRs by the continuous mixing of the fuel salt, which allows using one single set of reaction rates, without the need of deriving multiple sets for different axial and radial zones. To allow for the continuous fuel reprocessing and refueling, the burnup routine in Serpent 2 has been modified for the use of fictitious decay constants. In addition, a reactivity control mechanism has been included by adjusting the fissile-to-fertile ratio in the feed material. The modified Serpent 2 code has been employed in (Aufiero et al., 2013) for a complete characterization of the fuel cycle characteristics of the Molten Salt Fast Reactor, including equilibrium fuel cycle isotopics, transition to equilibrium, breeding ratios, reprocessing rates, decay heat, radio-toxicities, and irradiation damage in structural materials.

The internal heat generation in the liquid fuel requires instead special treatments for thermal-hydraulics calculations. New correlations have been developed for the prediction of Nusselt number correlations (Di Marcello et al., 2010; Fiorina et al., 2014), showing a significant impact of the internal heat generation, notably for low salt velocities and large channel diameters. In these studies, the interesting possibility has also been pointed out of a negative Nusselt number. Similar to Nusselt number correlations, modified thermal wall functions² have been developed for RANS-based CFD calculations (Fiorina, 2019). Also in this case, the impact of the internal heat source has been shown to be significant. Fig. 5 shows the effect of a modified wall function on the temperature profile in a channel of the Molten Salt Breeder Reactor (Robertson, 1971), and the resulting impact on the reactor behavior in case of a loss of flow transient. In Fig. 5 (left), it can be noted that salt temperatures tend to grow towards the periphery of the channel. To understand this trend, one should keep in mind that in MSRs a fraction of the power is released in the graphite by gammas and fast neutrons, and that this power is evacuated towards the salt. The temperature gradient is accentuated by the internal heat generation, due to the lower salt velocity close to walls. It is the latter effect that is normally neglected in traditional wall functions and Nusselt number correlations, resulting in an underestimation of the wall temperature jump.

Another research area of the MSR community has been the development of tools for the reactor transient behavior. In MSRs, the drift of the delayed neutron precursors creates a very strong coupling between thermal-hydraulics and neutronics. In addition, fast-spectrum MSRs cannot be analyzed using simplified 1-D or coarse-mesh thermal-hydraulic treatments, but require CFD analyses to properly simulate the flow field, the possible detachment of the flow at the core wall, and the possible set-up of recirculation and stagnation zones. In view of these characteristics, the analysis of MSRs has represented a formidable playground for the

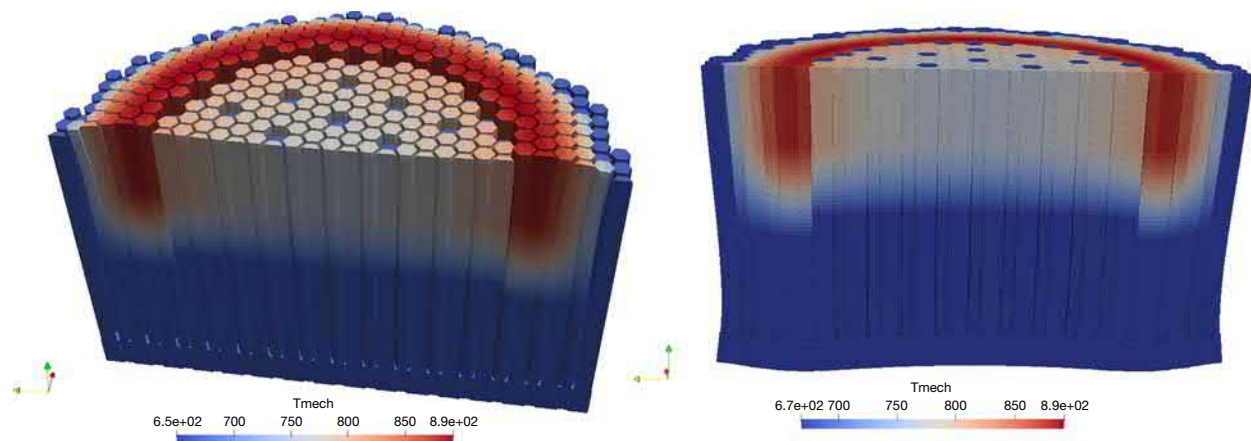


Fig. 4 Core flowering (left) and core compaction (right) for the European Sodium Fast Reactor (deformation is magnified).

²A wall function, or law of the wall, describes the velocity of a fluid in the vicinity of a wall and in case of turbulent flows. Similarly, a thermal wall function is employed to evaluate the temperature in the same region and conditions. Wall functions represent the boundary conditions that are most commonly used to represent walls in turbulent flow simulations. They determine the difference in velocity and temperature between a boundary and the first node or cell in the computational domain. Their use allows avoiding extremely refined meshes and it replaces the turbulence models in a region where their validity is not guaranteed.

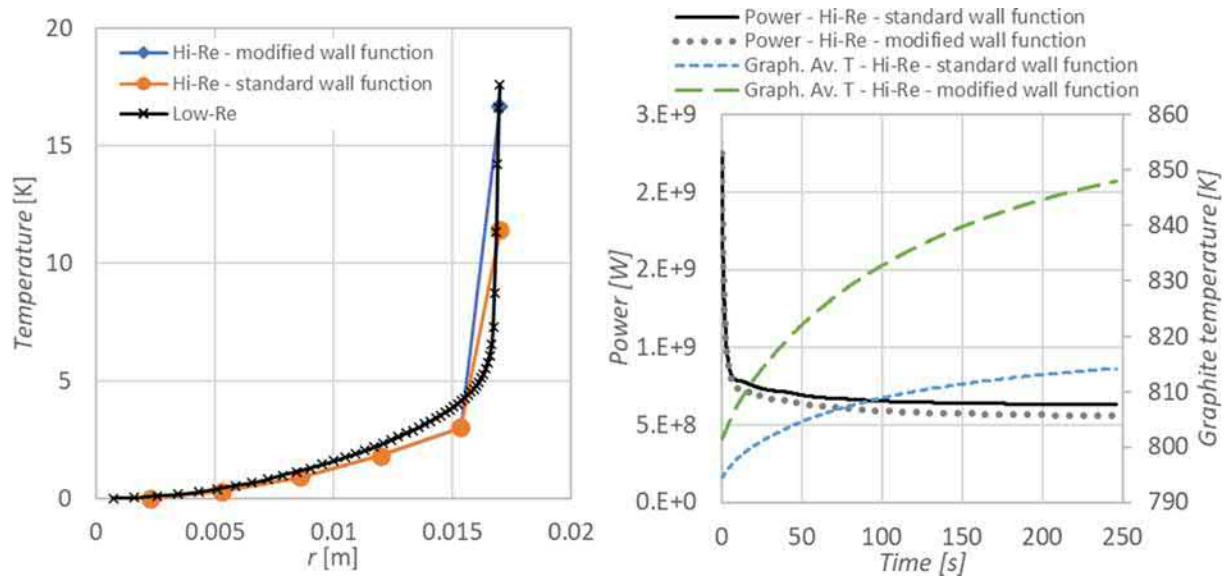


Fig. 5 Fuel salt temperature in a channel of the Molten Salt Breeder Reactor (left) and corresponding transient behavior in case of a loss of flow (right). Hi-Re indicates high-Reynolds k - ϵ models, while Low-Re indicates a low-Reynolds k - ϵ model that does not require the use of wall functions. The temperature in the left figure is rescaled to the temperature at the channel centerline (Radius = 0).

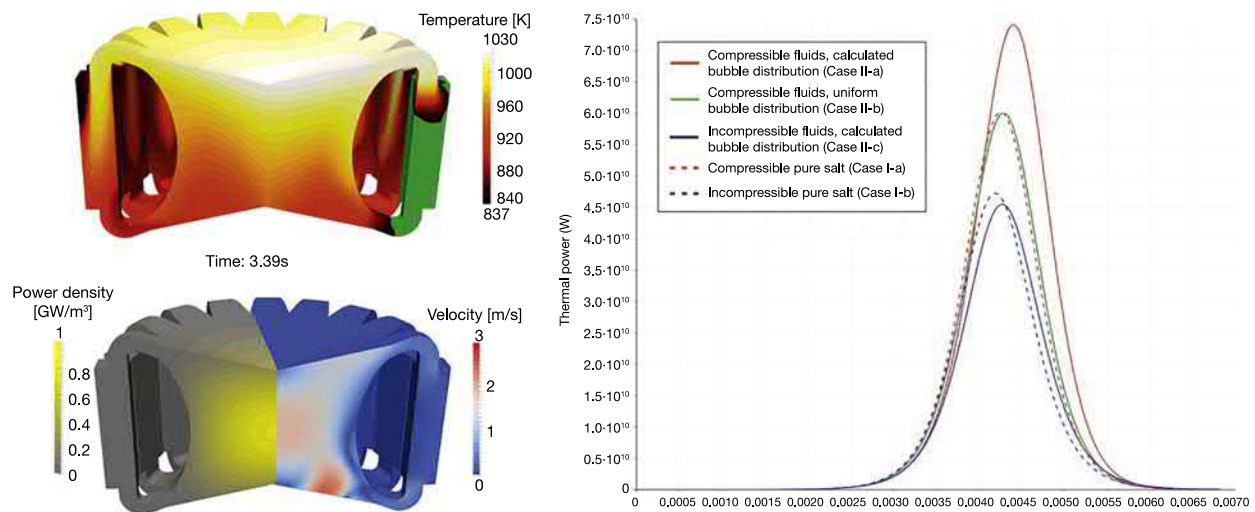


Fig. 6 Temperature (top-left), velocity and power density (bottom-left), and response to a super-prompt critical reactivity insertion (right) in the MSFR (Aufiero et al., 2014a; Cervi et al., 2019b).

development of new methodologies and codes. As a noteworthy example of these activities, work has performed aimed at the use of OpenFOAM for a complete multi-physics code for the transient analysis of MSRs (Aufiero et al., 2014a; Cervi et al., 2019a,b). The code implicitly couples a RANS-based CFD treatment for thermal-hydraulics with diffusion and SP³ models for neutron transport. It also includes a model for simulating helium bubbling and the corresponding compressibility. An example of the results of this code is reported in Fig. 6.

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³Simplified spherical harmonics of third order.

transport and fuel cycle analysis in Japan. The author also wishes to thank Stefan Radman and Alessandro Scolaro for their help in reviewing the chapter.

See Also: Self-Sustaining Breeding in Advanced Reactors: Characterization of Selected Reactors.

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Additive Manufacturing in the Nuclear Reactor Industry

Benjamin R. Betzler, Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Abbreviations

3D Three-dimensional
AGR Advanced gas-cooled reactor
ATR Advanced test reactor
BWR Boiling water reactor
CASL Consortium for advanced simulation of LWRs
CFD Computational fluid dynamics
DOE-NE US Department of Energy Office of Nuclear Energy
HFIR High flux isotope reactor
IRT Research reactor thermal
LWR Light water reactor
MITR MIT Nuclear Research Reactor
MTR Materials testing reactor
MURR University of Missouri Research Reactor
NBSR National Bureau of Standards Reactor
PWR Pressurized water reactor
TCR Transformational Challenge Reactor
TRIGA Training, research, isotopes, General Atomics
VVR Water water reactor

Introduction

Additive manufacturing, which is also known as 3D printing, is a method for manufacturing parts that is often automated. Additive manufacturing creates a part layer by layer based on the specifications provided in a three-dimensional (3D) digital model (ISO/ASTM52900-15, 2015). Conversely, subtractive manufacturing is the removal of material from a bulk solid to create a part. When compared to traditional manufacturing processes, additive manufacturing technologies boast reduced product development times, integrated design and manufacturing efforts, and the ability to create complex geometries once unattainable using traditional fabrication approaches (Babu and Goodridge, 2015). For these reasons, additive manufacturing is being implemented in many

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industries, including the automotive (Leal et al., 2017), aerospace (Lyons, 2014; Shapiro et al., 2016), and biomedical (Gunbay et al., 2019; Murr et al., 2010) fields. In many applications, additive manufacturing technologies are cited for their ability to provide rapid prototyping and a time-compressed design feedback loop (Jared et al., 2017; Thompson et al., 2016).

Within the nuclear industry, additive manufacturing is being applied to manufacture obsolete replacement parts for operating light-water reactors (Thomas, 2016c), for subcomponents on light-water reactor fuel assemblies (Thomas, 2017), and manufacturing small numbers of experimental and replacement components for operating research reactors (Hehr et al., 2017). Additional research and development areas of opportunity include leveraging instrumented data-rich manufacturing processes for component qualification, realizing complex geometries for increased reactor performance, and embedding instrumentation to monitor components or to enhance reactor performance characterization (Tobin et al., 2018). Advanced reactor developers are applying additive manufacturing technologies to create complex geometries, and to fabricate materials or mixtures of materials that are not fabricable through traditional means (Betzler et al., 2020d). In particular, microreactor and space reactor concepts have stringent size, temperature, and power density requirements (Nuclear Energy Institute, 2019) for which rapid prototyping, complex geometries, and improved fabrication processes can yield distinct benefits (Miller et al., 2018; Sterbentz et al., 2017).

Nuclear reactor design and engineering is a multidisciplinary, highly regulated industry, not unlike the biomedical and aerospace industries, which are seeing many successful applications of additive manufacturing technologies. In the aerospace industry, additive manufacturing is being used to create flight-critical components (ASM International, 2020). Additively manufactured components first entered nuclear reactor service in 2017 (Ford, 2017), and reactor core components are being employed for the first time in a reactor entering service in 2020 (Day, 2020a). Increased adoption of these technologies over the next decade is expected, but the extent in which they will be adopted are yet to be defined (Thomas, 2016a).

This article provides an overview of the current state of knowledge on additive manufacturing technology applications in nuclear reactor engineering, including:

- Applicable additive manufacturing methods and technologies, including materials, limitations, and fundamental guidelines for design
- Applications for nuclear reactors, including the manufacture of existing component designs for operating reactors and applications to advanced reactors
- Agile reactor design for rapid prototyping and testing akin to early nuclear reactor engineering reactor demonstrations, including analysis of how agile design can yield the most value from applications to advanced reactor concepts
- Modeling and simulation of complex geometries in reactor physics, thermofluidics, and thermomechanical tools, including applications of topology optimization
- The future role of additive manufacturing in the nuclear industry, including potential future application areas
- Discussion of how all of these elements come together for advanced reactor R&D, with a focus on the US Department of Energy (DOE) Office of Nuclear Energy (NE) Transformational Challenge Reactor program (Terrani, 2020), which is exploring many of these concepts in order to accelerate deployment of additive manufacturing technologies in the nuclear industry

The following sections discuss these topics, drawing on current research efforts and the ongoing developments in advanced manufacturing technologies. Due to the rapid advancement in additive technologies, the focus here is on providing general design guidelines instead of a review of specific technologies.

Additive manufacturing

To accommodate the thin fuel claddings, large pressure vessels, and heat exchange machinery, nuclear reactors employ multiple materials at multiple length scales. As such, most additive manufacturing processes (ISO/ASTM52900-15, 2015) are being pursued for nuclear reactor applications, including binder jetting, powder bed fusion, directed energy deposition, and sheet lamination. The feasibility of a technology for a given component depends on the material, the minimum feature size, the overall geometry, and the component function. This section provides a brief overview of these technologies, along with general design considerations for nuclear reactor applications (Simpson et al., 2019). Outside of prototyping, the polymers typically used in additive manufacturing are not useful in the high-irradiation, high-temperature environments characteristic of nuclear reactors. For parts to perform in these challenging conditions, the following methods are being studied:

- *Binder jetting* selectively binds powder with a solvent-polymer solution to build a low-density part (Fig. 2). Geometric flexibility, feature size, and resulting surface roughness are dependent on the material type and powder size (Fig. 1). Ceramics and metals are possible materials. Follow-on techniques for densification are commonly used to meet performance requirements for the part.
- *Powder bed fusion* melts thin layers of powder to build parts layer by layer. Geometries with complex overhangs require additional support structures that accommodate postmachining (Fig. 2). Feature sizes and roughness depend on material type and powder size. Powder bed fusion is a common method for additive manufacturing with metals. Follow-on techniques for selective machining are typically implemented according to the performance requirements of the part.
- *Directed energy deposition* adds material to a small melt pool that is formed with focused thermal energy to build parts volume by volume (Zhang et al., 2018). Feature size and surface roughness depend on the material feed. Metals are typically used for



Fig. 1 A binder jetted silicon carbide part demonstrating geometric flexibility of the manufacturing method. Credit: Kurt Terrani. Terrani K, Jolly B, Trammell M (2020a) 3D printing of high-purity silicon carbide. *Journal of the American Ceramic Society* 103: 1575–1581.

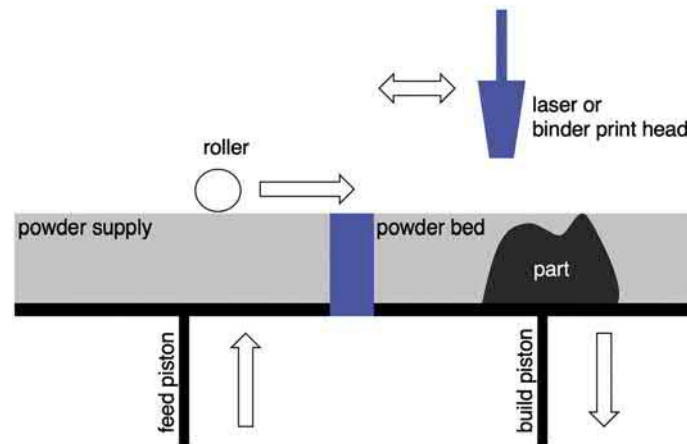


Fig. 2 A simplified diagram of a powder bed additive manufacturing process. Powder spread in an even layer is then selectively bound together using a laser (powder bed fusion) or a binder (binder jetting). The bound layer is covered by another layer of powder from the movement of the build and feed pistons and the roller. Then, the powder in the next layer is selectively bound and the process is repeated.

directed energy deposition. Follow-on techniques for selective machining are commonly used according to the performance requirements of the part.

- *Sheet lamination* bonds sheets of material together to form a large part. This method is ideal for planar-type geometries with metal foils, and it facilitates embedding compatible materials or objects.

Additive manufacturing technologies offer reduced lead times and cost for high value and low-volume production (Pereira et al., 2019) that are characteristic of the complex and customized parts emerging from advanced nuclear reactor designs. Many applications will selectively deploy additive, subtractive, and traditional methods to manufacture and assemble economically competitive components. Additional technologies can be used to combine manufacturing methods. Multimaterial additive manufacturing combines more than one material within the same machine, with or without the same method. Hybrid manufacturing combines additive and subtractive methods within the same machine to achieve the desired surface finishes (Gibson et al., 2015).

Mechanical and thermophysical properties of additively manufactured materials may meet or exceed those of traditionally manufactured materials, but this is dependent on the material type and manufacturing methods. Testing of steel and SiC for reactor applications show comparable properties, with some differences in steel creep behavior due to a high porosity and fine-grained microstructure (Byun et al., 2020b) and some observable impacts from the anisotropic character of SiC (Byun et al., 2020a).

Because of the layer-by-layer approach for building parts, additive manufacturing processes are potentially data rich. Video and thermal imaging may be used to detect abnormalities during manufacturing. Recording this information generates a very large data set that contains the history of the part as a function of space and time. Correlating specific characteristics within this data set (e.g.,

a temperature history at a given location) to defects or failures in the manufactured part (e.g., from testing) is a component of the qualification process for additively manufactured parts (Babu et al., 2018). This type of problem, with very large space- and time-resolved data sets (e.g., imaging and temperature), is ideal for the deployment of artificial intelligence and machine learning approaches (Scime et al., 2020). This is a significant departure from traditional testing and qualification approaches, which rely on statistics and defined processes (Seifi et al., 2017).

Some general guidelines are applied when considering additive manufacturing technologies for nuclear reactor applications:

- An economically viable application of additive manufacturing will likely involve a part redesign process: simply replacing a conventional manufacturing process with an additive manufacturing process to build an identical part is not guaranteed to add value. Incorporating knowledge of the additive manufacturing process in the design process will likely yield a more optimized part for the process while maintaining performance requirements.
- The material type and material density requirements impact the applicable additive manufacturing technology and material maturity.
- The minimum feature size, tolerance, and overall dimension of the part impact the applicable additive manufacturing technology.
- There is an inverse relationship between deposition rates and dimensional resolution, and very large parts with fine detail could take several weeks of machine time to build.
- If a postprocessing step such as densification or machining is required, then the preprocessed geometry must support this functionality.

Additive manufacturing is rapidly developing, so literature on methods, technologies, and capabilities should be reviewed periodically to understand the state of the art for nuclear reactor applications.

Nuclear reactor applications

Objectives for nuclear industry investments in new manufacturing technologies include recapturing lost manufacturing capabilities (Kinsey and Jessup, 2018), reducing manufacturing costs and timelines (Thomas, 2016b), enhancing component performance, and deploying advanced materials (Müller et al., 2019). These technologies include hot isostatic pressing for small modular reactor pressure vessel manufacturing (Gandy et al., 2018) and advanced ceramic-metallic materials for space reactor applications (Houts et al., 2012). While these efforts do not use additive manufacturing methods, they represent new manufacturing approaches to which additive manufacturing may also be applied. For example, efforts focused on wire additive direct energy deposition have yielded large metal structures (Greer et al., 2019; Sridharan et al., 2018) that are directly applicable to large nuclear reactor structural components such as reactor pressure vessels and piping (Sridharan et al., 2019).

The first additive manufacturing applications for operating nuclear reactors started in auxiliary plant components and has slowly migrated to metallic reactor and core components. Many of these components are traditionally manufacturable and are not safety critical components, as outlined below:

- Siemens used powder bed fusion to replace the obsolete design of a fire protection system pump impeller and installed it in a Slovenian reactor in 2017 (Ford, 2017).
- Oak Ridge National Laboratory used ultrasonic additive manufacturing for aluminum alloy control elements to be used in the High Flux Isotope Reactor to study an alternative to costly, manual-intensive manufacturing processes (Hehr et al., 2017); these elements included embedded absorber materials.
- Westinghouse used powder bed fusion for a steel fuel assembly thimble plugging device to demonstrate the manufacturing, qualification, and regulatory approval process for an additively manufactured part (Cleary et al., 2019) that was installed in a US reactor in 2020 (Day, 2020a).
- Westinghouse used powder bed fusion to create a steel fuel assembly bottom nozzle for optimized debris filtering and pressure drop; this effort included using additively manufactured plastics for prototyping and flow testing (Cleary et al., 2019).
- Korea Electric Power Corporation additively manufactured fuel assembly grid spacers to test the applicability of methods and post machining requirements (Nam et al., 2019).
- Oak Ridge National Laboratory and Kairos Power additively manufactured a molten salt pump impeller (Fig. 3) to leverage the iterative design and economic high-rigor low-quantity production characteristics inherent to additive manufacturing technologies (Bumpus, 2020a; Kairos Power, 2020).
- Oak Ridge National Laboratory, Tennessee Valley Authority, and Framatome additively manufactured fuel assembly brackets to demonstrate the iterative design, manufacturing, and qualification process of a safety-related component with a fully advanced digital manufacturing history (Bumpus, 2020b).

In addition to the examples, there are many efforts to apply additive manufacturing technologies for heat exchangers and turbomachinery that are potentially relevant to the reactor industry.

The success of these applications and the deployment of the resulting additively manufactured components in operating nuclear reactors are critical steps for adopting these manufacturing methods in advanced reactor design and deployment. Under the DOE-NE Transformational Challenge Reactor program (Oak Ridge National Laboratory, 2019), Oak Ridge National Laboratory is actively



Fig. 3 An additively manufactured Kairos Power molten salt pump impeller. Credit: Kairos Power. Bumpus KL (2020a) *As ORNL Builds Novel Reactor*. Nuclear Industry Benefits From Technology.

deploying additive manufacturing technologies to rapidly mature an advanced microreactor design (Betzler et al., 2020d), leveraging the geometric flexibility, iterative design, rapid prototyping, and advanced materials possibilities (e.g., high-purity SiC) that additive manufacturing technologies provide. These advanced reactor development applications more fully leverage the inherent characteristics of additive manufacturing technologies by incorporating them within the design process (Betzler et al., 2019, 2020c).

Agile reactor design with additive manufacturing

The term *agile* refers to software development practices that emphasize iterative design, response to change, customer collaboration, and delivery of working software (Shore et al., 2008). These beneficial agile practices are adopted in manufacturing and other technical fields (Nagel and Dove, 1991). Additive manufacturing technologies uniquely enable a rapid iterative design cycle (Fig. 4) that facilitates application of these practices and the realization of their benefits. Agile design characteristics are not typical attributes of the nuclear reactor industry, but these design practices are well suited for advanced nuclear reactor developers working within a more open design space and continuously responding to changing design requirements and rapid technological advances (Day, 2020b).

Agile processes are characterized by four core values (Beck et al., 2001a) and 12 principles (Beck et al., 2001b) as defined for agile software development. The four core values presented below emphasize certain values over others, and they naturally fit within an advanced nuclear reactor design process. In this context, working software applications are equivalent to working prototypes.

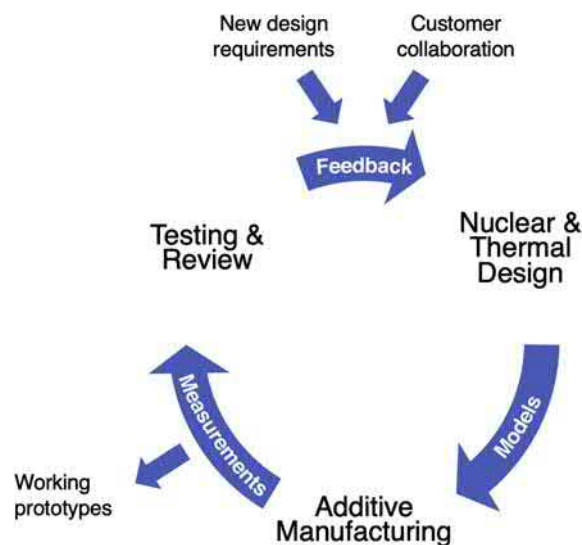


Fig. 4 Iterative nuclear reactor design loop with additive manufacturing.

Interdisciplinary collaboration

Individuals and interactions over processes and tools

Nuclear reactor additive manufacturing applications naturally encourage collaboration between mechanical and material engineers familiar with the manufacturing processes and nuclear engineers familiar with nuclear reactor physics, thermal hydraulics, and safety. These collaborations inform the nuclear design process regarding the geometric flexibility of additive manufacturing processes, as well as manufacturing and material constraints, properties, and tolerances. This requires individual engineers to establish the technical constraints and tolerances within their respective areas of discipline.

The resulting informed design approach immediately considers manufacturability and material compatibility before significant design progress or optimization in any specific concentration (e.g., reactor physics or thermal hydraulics). Feasibility in manufacturing is demonstrated as the design progresses. This differs from more traditional nuclear reactor design activities that either do not consider manufacturability or avoid deviating from existing designs to ensure manufacturability.

Working prototypes

Working software over comprehensive documentation

Within this interdisciplinary collaboration, additive manufacturing is critical for rapidly developing and delivering working prototypes. This provides for manufacturing-informed decision-making, and it reduces the risk associated with proposing new designs. This rapid prototyping enables the adoption of a *fail fast* philosophy that preferentially evaluates designs through extensive testing and incremental development. The follow-on learning within this philosophy becomes a necessary and essential part of the agile design process. The longer lead times required by traditional manufacturing approaches would not typically allow this type of expedited design feedback.

Consistently developing working prototypes tests the manufacturing processes and provides design feedback in quantitative and qualitative ways. Exercising the manufacturing process quickly determines geometric characteristics and constraints such as minimum thicknesses, surface roughness, and tolerances. Advanced characterization techniques provide even more information regarding material properties, impurities, and as-assembled geometries. The in-person examination of physical prototypes with a multidisciplinary team is conducive to identifying issues and considering design changes. When this design approach is implemented, the final design is immediately manufacturable if the prototypes were sufficiently characteristic of the final design.

Customer communication

Customer collaboration over contract negotiation

Throughout this design process, reactor physics, thermal hydraulic, thermomechanical, safety analyses, and manufacturing generate refined technical constraints that lead to design changes. This is particularly true for advanced reactor development, which typically has more open design space than traditional incremental nuclear design processes. Technical information is rapidly harnessed from this interdisciplinary, rapid prototyping design process; this information may be used to inform and involve the customer in the decision making process. Additively manufactured prototype processes are particularly effective for communicating technical information, design constraints, and overall progress to obtain customer feedback.

Changing requirements

Responding to change over following a plan

Flexibility and responsiveness to changing requirements is particularly useful for advanced reactor developers who are working in more open design spaces and are responding to consistently evolving technical requirements and new developing technologies. The iterative design process inherent to additive manufacturing applications directly provides the mechanism to respond to a different or new design requirement: as each design iteration ends with a working prototype, the requirements are reviewed and updated, allowing for increasingly refined design requirements through each iteration. With close customer collaboration, these requirement changes are made according to customer feedback. The inherent capabilities of additive manufacturing for rapid prototyping and iterative design provide for the adoption of these agile design characteristics. While the agile design methodology is not necessarily better than a traditional design approach, it is more responsive to changing design requirements and more flexible when incorporating developing technological advancements. For these reasons, it is well suited for advanced reactor development, in which the design space is more flexible, and the requirements are more negotiable. In order to emerge effectively from this agile

design process to deliver a final design, a change in the design approach is necessary to fit the qualification requirements of a final nuclear core design while maintaining technical rigor.

Computational analyses of additively manufactured geometries

The major modeling and simulation components for nuclear reactor design include reactor physics, thermal hydraulics, system dynamics, and thermomechanical analysis, including fuel performance. During steady-state operation, the core design problem is a coupled reactor physics–thermal hydraulics problem: fission heat generated within the fuel is transferred to the coolant and out of the core. Thermomechanical tools predict material behavior and stresses under relevant conditions. System dynamics tools quantify peak temperatures and heating rates during transient scenarios. Reactor physics methods predict core operational and safety characteristics as conditions change during irradiation (e.g., fuel depletion). The physical and chemical changes in fuel material during depletion and its impact on the cladding is a fuel performance problem.

Decades of code development have yielded the methods and procedures currently deployed for operating reactors that predict conditions during steady-state and transient scenarios (Duderstadt and Hamilton, 1976; Smith, 1980). Many of these methods leverage (1) the simplified pin- or plate-type fuel geometries of most operating reactors (Table 1) to use one-dimensional (1D) or two-dimensional (2D) models, (2) leverage axial coolant channel uniformity for thermal hydraulic models, and (3) adopt simplified methods such as point kinetics for dynamic modeling. In some cases, tools are more safety specific—as when applying uncertainty factors in calculating the margin to fuel failure—than predictive. The simplicity of these tools is useful for adhering to the software quality assurance rigor required for nuclear safety basis calculations.

The geometric complexity and design flexibility proffered by additive manufacturing are not amenable to the modeling and simulation used for more traditional reactor methods and tools. Predictive simulation is critical in leveraging geometric complexity and heterogeneity to improve performance over designs constrained by conventional processes. Ongoing advancements in nuclear modeling and simulation have already provided for consistent incremental changes in light-water reactor fuel assembly geometries to improve fuel cycle economy and reduce fuel failures. These improvements include more complex gadolinia rod loading patterns (Rearden and Jessee, 2016), axial fuel heterogeneity (Franceschini et al., 2015), and more complex water box geometries (Gustafsson et al., 2013). One example of these advanced methods is the main product of the Consortium for Advanced Simulation of LWRs (CASL), which targeted enhanced operational performance, efficiency, and safety of LWRs (CASL, 2019). This product uses a 2D/1D method with the assumption that the majority of the assembly heterogeneities are in the radial dimension, and axial heterogeneities may be captured using a simpler 1D solution (Kochunas et al., 2017).

The most high-fidelity methods for core design include high-fidelity Monte Carlo transport and computational fluid dynamics (CFD). These methods are most relevant for modeling the geometric complexities of additively manufactured components and generating best-estimate results for nuclear design activities (Fig. 5). While seen as intractable for core design applications due to computational expense, these methods have recently been applied in operating reactors (Betzler et al., 2017). Providing validation data for CFD simulations is necessary to verify the method for the application, and additive manufacturing serves well in rapidly generating components for testing. For thermomechanical analysis, a host of finite-element–based tools is available. For nuclear fuel applications, the MOOSE-based BISON tool developed by the DOE-NE Nuclear Energy Advanced Modeling and Simulation (NEAMS, 2019; Williamson et al., 2012) and maintained by Idaho National Laboratory, uses generalized equations and methods that are suitable for additively manufactured geometries.

Table 1 Power densities, fuel geometries, and active core height for selected reactor designs.

Fuel type	Typical core power density (kW/L-core)	Active core height (m)	Fuel geometry type
<i>Commercial reactor fuels</i>			
PWR	100	3.66	pin
BWR	50	3.05–3.66	pin
AGR (Nonboel, 1996)	3	8.30	pin
<i>Research reactor fuels</i>			
MTR (Miller and Sindelar, 1995)	varies	0.61	plate
TRIGA (Cammi et al., 2016)	varies	0.36	pin
VVR (Pond et al., 1998)	varies	0.60	plate
IRT (Hanan et al., 2006)	varies	0.60	plate
<i>High-performance research reactor fuels</i>			
ATR (Stanley and Marshall, 2008)	1000	1.18	plate
MITR (Massachusetts Institute of Technology, 2020)	70	0.57	plate
HFIR (Adamson and Knight, 1968)	1700	0.51	plate
NBSR (Diamond et al., 2015)	40	0.56	plate
MURR (University of Missouri, 2020)	300	0.61	plate

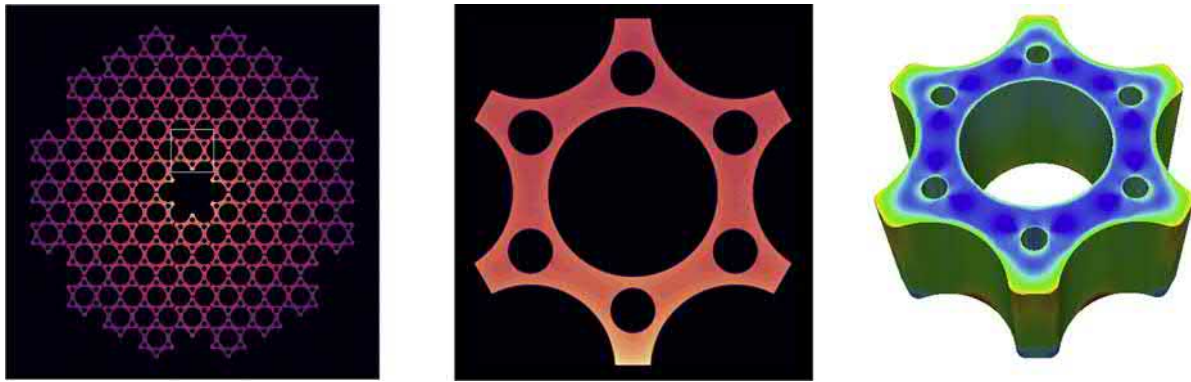


Fig. 5 Fine mesh power distribution within the Transformational Challenge Reactor core (left) and central fuel elements (center) and resulting stress profile (right) from a BISON simulation. The fuel cross section area coincides with the color areas. Credit: Brian Ade and Danny Schappel. Ade BJ, Schappel DP, Betzler BR, Helmreich G (2020b) Particle packing characteristics in dense TRISO/SiC fuel elements and the impact on neutronics and thermomechanics. *Transactions of the American Nuclear Society* 123.

Optimization approaches that efficiently leverage these predictive high-fidelity simulations are critical in fully realizing the benefits that additive manufacturing technologies offer for a given application. Commercial optimization tools exist, with successful applications in optimization for fluidic or mechanical part performance (See et al., 2020). Application to the coupled nature of nuclear reactor multiphysics is more complex, requiring effective communication across several simulation tools. Some successes using emulators in place of high-fidelity physics to rapidly evaluate thousands of potential designs have been achieved (Hiscox et al., 2020; Sobes et al., 2019, 2020). However, more generic topology optimization for nuclear reactor core design applications is an open design problem characteristic of the advanced reactor development industry, which works with fewer design constraints relative to incremental fuel assembly design.

Future roles

Defining the qualification and regulatory processes for additive manufacturing technologies provides for more widespread adoption in highly regulated environments like those for critical nuclear reactor and core components. Efforts to apply additive manufacturing in the defense, aerospace, and medical fields are addressing these same issues (Babu et al., 2018; Horst et al., 2019; Seifi et al., 2017). The earliest proponents of nuclear reactor additive manufacturing applications exercised this process with regulator involvement to address unknowns and reservations associated with these new manufacturing practices (Hull et al., 2019). Demonstration of consistent material performance over long-term irradiations is critical for economically viable core-adjacent components.

Future prospects for additive manufacturing applications in the nuclear industry have been recognized and addressed through many activities under the Transformational Challenge Reactor (Cetiner and Ramuhalli, 2019; Nelson, 2019; Tobin et al., 2018) program:

- Developing manufacturing surveillance processes to record data that may be used to help understand material properties, predict performance, and meet specifications
- Deploying additive manufacturing technologies to facilitate the rapid manufacturing of replacement parts
- Redesigning components with additive manufacturing to improve performance and/or reduce cost
- Deploying additive manufacturing technologies for rapid prototyping to shorten development times for new designs
- Reimagining nuclear core design through the agile design process with additive manufacturing to optimize nuclear performance
- Embedding sensors inside of components to provide additional data during operation to monitor component performance over the lifetime of a part
- Developing heterogeneous fuel monolithic structures to improve performance (e.g., fuel utilization)
- Developing advanced matrices for particle and dispersion fuels free from conventional processes to accurately control distributions, increase matrix quality and packing fractions, and reduce design uncertainties

The data-rich processes and advanced characterization available with modern technologies (Fig. 6) decrease the timeline to deliver on additive manufacturing applications and to realize their benefits. Applications will continually gravitate toward higher performing parts in more challenging physics environments, including critical fuel components. Continued development of new materials, additive manufacturing methods, and advanced computation will only increase the applicability of these technologies to nuclear reactors.

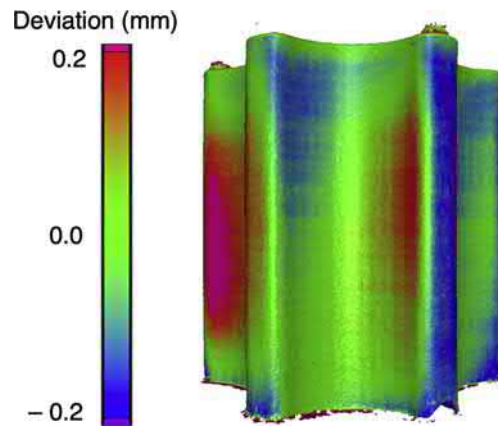


Fig. 6 As-built deviation in an additively manufactured Transformational Challenge Reactor fuel element. Credit: Andres Marquez Rossy, Terrani KA, Jolly BC, Trammell MP, Vasudevamurthy G, Schappel DP, Ade BJ, Helmreich GW, Wang H, Marquez Rossy A, Betzler BR, and Nelson AT (2021) Architecture and properties of TCR fuel form. *Journal of Nuclear Materials*. <https://doi.org/10.1016/j.jnucmat.2021.152781>.

Implementation under the Transformational Challenge Reactor program

The US DOE-NE Transformational Challenge Reactor (TCR) program is the best representation of the integration of these different elements to drive additive manufacturing applications in the nuclear industry (Oak Ridge National Laboratory, 2019). These elements include the development of manufacturing processes and approaches, the collection of advanced in situ manufacturing data, the implementation of an agile design process, the use of high fidelity modeling and simulation tools for optimization, and the exercising of the qualification and regulatory processes. This leverages advances in numerous scientific areas—including materials, manufacturing, sensors and control systems, data analytics, and high-fidelity modeling and simulation—to accelerate design, manufacturing, qualification, and deployment of advanced nuclear energy systems. A core component of the program is to design, build, and operate a small nuclear reactor to fully integrate and rapidly demonstrate these elements in 4 years.

Following assessment of additive manufacturing technologies (Nelson, 2019; Simpson et al., 2019) and their applicability to nuclear reactor engineering problems (Tobin et al., 2018), the TCR program implemented an agile design process (Betzler et al., 2020b) to design and downselect a reactor design in less than 12 months of effort (Betzler et al., 2020a,e). The following maturation of the design with iteratively manufactured prototypes provided the as-built information necessary to further mature the design (Ade et al., 2020a,b). Advanced in situ characterization during prototype manufacturing provides the data necessary to support qualification activities (Scime et al., 2020). Design information provided to irradiation efforts ensure that design conditions are bounded in burnup and temperature (Coq et al., 2020; Kile et al., 2020; Vasudevamurthy et al., 2020).

Many of these design elements are effectively integrated in the agile design process. Under this process, data and constraints from manufacturing processes (e.g., minimum thicknesses, actual material densities, moderator impurities and stoichiometry) are incorporated immediately into the design activities to determine their impacts and sensitivities in relation to reactor performance. Collocation of key staff members at the Oak Ridge National Laboratory Manufacturing Demonstration Facility (ORNL, 2020) allows for consistent communication between materials, manufacturing, nuclear, thermofluidic, and thermomechanical experts and analysts. This collocation ensures direct access to additively manufactured prototypes and test components, as well as the experts operating the machines and performing characterization on manufactured parts.

Initial preconceptual design activities started with two simple guidelines: (1) to design, build, and operate an advanced nuclear reactor within several years, and (2) to leverage the fundamental characteristics of additive manufacturing to accelerate this process. Instead of establishing a preferred reactor technology early in the design process, technologies were assessed based on these guidelines, and technical decisions made heavily emphasized viability and simplicity in design. Processes and materials for powder bed fusion, laser-directed energy deposition, and binder jetting were assessed to determine their feasibility for nuclear applications (Simpson et al., 2019).

Drawings of the earliest preconceptual designs of the TCR core were sent to additive machines and were printed using candidate materials such as steel, SiC, and Inconel. The printed parts included the initial monolithic core structures, different types of stackable fuel elements, fuel element retention structures, and insulated-moderator core concepts (Betzler et al., 2020c). Design sprints focused on determining the feasibility of specific design features and testing additively manufactured materials (Field et al., 2019; Schappel and Terrani, 2019). Feedback from monitoring and characterization were implemented into physics models to determine sensitivities to tolerances and more detailed impacts from material distributions (Ade et al., 2020a,b).

Candidate designs were quickly developed and continually refined as requirements, constraints, and available technologies continued to evolve (Fig. 7). This includes key design decisions, including moving on from the earliest unmoderated preconceptual designs due to fuel constraints (Ade et al., 2020c; Betzler et al., 2020d), selecting a yttrium hydride moderator (Bergeron et al., 2020; Hu et al., 2020) and a TRISO in SiC fuel form (Terrani et al., 2020a,b; Trammell et al., 2019) as manufacturing and characterization



Fig. 7 Transformational Challenge Reactor fuel assemblies incorporating binder jetted silicon carbide fuel elements and powder bed fusion metallic components. Credit: Andrew Sproles and Phillip Chesser. Betzler BR, Ade BJ, Wysocki AJ, Chesser PC, Terrani KA (2020b) Transformational challenge reactor agile design enabled by additive manufacturing. *Transactions of the American Nuclear Society* 123.

rapidly progressed during the design process (Byun et al., 2020a,b), and selecting steel encapsulation for the yttrium hydride moderator to fit with the demonstration timeline. Flexibility and responsiveness of the TCR program design thrust was critical in accommodating these changing requirements and taking on these new technologies to advance this design in less than 12 months (Betzler et al., 2020a). This has produced a viable, simple design that incorporates novel fuel and moderator forms (Table 2) with embedded sensors (Petrie et al., 2019).

Constraints and working prototypes have been immediately shared with TCR program technical and programmatic leadership to help make technical design decisions. This close collaboration and tight feedback loop ensures that the developed design meets high-level requirements while also ensuring that the requirements are consistent with project scope, schedule, and other programmatic constraints, such as realistic availability of the type and quantity of fissile material. In addition, interested parties such as the regulator are kept apprised of new design developments. In the manufacturing area, representatives from potential future regulators are collocated to learn and about additive manufacturing processes and the advanced capabilities in characterization and monitoring.

High-fidelity Monte Carlo-based reactor physics, CFD, and finite element thermomechanical simulations are used in design efforts for optimizing core and fuel element geometries (Popov et al., 2020; Schappel and Terrani, 2019; Weinmeister et al., 2020). Safety analyses are performed with more traditional tools to determine conditions during postulated accidents and sensitivities to design parameters (Betzler et al., 2020e; Wysocki et al., 2020). The fidelity of these simulations has been iteratively refined

Table 2 General Transformational Challenge Reactor conceptual design characteristics.

Attribute	Value
Total power	3 MWt
Inlet coolant temperature	300 °C
Outlet coolant temperature	500 °C
Fuel	UN TRISO
Fuel enrichment	19.75%
Fuel element type	TRISO in SiC
Moderator	Yttrium hydride
Moderator encapsulation	Type 316L stainless steel
Coolant flow direction	Downward
Coolant type	Helium
Coolant pressure	5 MPa
Pressure vessel	Type 304H stainless steel
Control	External segmented shroud
Shutdown	Central rod

through the design process as more as-built information and tolerances have been fed back into the design. This iterative refinement provides the flexibility in the design process to continue to respond to feedback and changing design requirements. Observations from applications of artificial intelligence for more generic nuclear core optimization are incorporated into the design elements as they become available (Hiscox et al., 2020; Popov et al., 2020).

Conclusion

Continued development of additive manufacturing technologies will only widen their applicability in the nuclear reactor industry. Critical additively manufactured components are being deployed in many highly regulated industries, and core components are in use at operating reactors. Additively manufactured components are currently under irradiation in several reactors.

Despite these varied research and development activities, there are some inherent limitations of these technologies. A value proposition for the application of additive manufacturing does not exist for all reactor components. For example, additively manufacturing a long, thin fuel cladding tube is likely neither economically feasible nor possible. Specific technologies quickly become outdated due to the rapid progress made in additive manufacturing methods and technologies. In addition, these applications mainly address in-core reactor components which often do not represent a large cost relative to the entire reactor itself. For these reasons, applications of additive manufacturing in construction (Paolini et al., 2019) may provide the greatest economic impact for large reactors that require high quality construction processes.

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New Reactor Concepts Deployment Challenges and Opportunities

Ramesh Sadhankar*, Canadian Nuclear Laboratories, Chalk River, ON, Canada

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Abbreviations

CEM	Clean Energy Ministerial
CfD	Contract for Difference
EON	Energy Options Network
EPRI	Electric Power Research Institute
ETI	Energy Technologies Institute
FOAK	First-Of-A-Kind
Gen IV	Generation IV
GFR	Gas fast reactor
GHG	Green house gases
GIF	Generation IV International Forum
GWe	Giga watt electric
HTSE	High temperature steam electrolysis
IEA	International Energy Agency
LCOE	Levelized cost of electricity
LMR	Liquid metal reactor
LWR	Light water reactor
MED	Multi-effect distillation
MIT	Massachusetts Institute of Technology
MSR	Molten salt reactor
MWe	Mega watt electric
MWth	Mega watt thermal
NEA	Nuclear Energy Agency
NOAK	Nth-Of-A-Kind
OECD	Organization of Economic Cooperation and Development

*Retired.

PCI Pellet cladding interaction
 PV Photo voltaic
 PWR Pressurized water reactor
 RAB Regulated asset base
 SCWR Super critical water reactor
 SMR Small modular reactor
 USDOE United States Department of Energy
 VHTR Very high temperature reactor
 VRE Variable renewable energy sources
 WNA World Nuclear Association

Introduction

Many countries agreed to reduce their green-house gas (GHG) emissions to limit the global average temperature rise to less than 2 °C compared to pre-industrial levels under the Paris Agreement ([United Nations, 2015](#)). This would require major efforts to decarbonize the entire energy sector, given that the energy demand will continuously increase. International Atomic Energy Agency ([2019a](#)) estimates that the growth in electricity demand will outpace the estimated growth in total energy demand, which would imply major transformation of the electricity sector to carbon-free generating sources. The average GHG emissions from electricity production in OECD countries would require reduction from current level of 430 gCO₂ per kWh to 50 gCO₂ per kWh.

Technology improvements in wind turbine and solar photo voltaic (PV) have reduced their investment costs to the point that they are nearly competitive with natural gas generation, while the cost of building nuclear plants, one of the main low carbon alternatives, has remained high ([Massachusetts Institute of Technology \(MIT\), 2019](#)). With the falling costs and various policy incentives, wind and solar (so called “renewable”) technologies have dominated investment in the power sector globally in recent years. The incentives for the renewables ranged from tax incentives and renewable portfolio requirements as in the United States, to feed-in tariffs or zero emission credits ([MIT, 2019](#)). The main limitation on the renewable sources is the challenge of matching their daily and seasonal variable pattern of supply with the demand, as such these sources are also called as variable renewable energy (VRE) sources. The increasing use of intermittent renewables creates grid management challenges requiring the power system to be more flexible. Flexible operations in current nuclear plants lead to lower capacity factors and affect the economics adversely. Hence, it would not be in a utility’s interest—be it in a regulated or deregulated electricity market—to currently invest in a new nuclear reactor if it expects that it will be operating at relatively low capacity factor in order to meet flexibility needs of the power systems. The challenges of integration of advanced nuclear reactors with VRE sources, flexibility requirements and economic considerations for building new nuclear reactors are discussed in this article. This article also explores opportunities for deployment of advanced reactors through innovative approaches and policies that would be conducive to nuclear deployment.

Current challenges

The advent of abundant wind and solar energy sources is one of the main factors driving the transformation in the power systems. This transformation is compounded by the deployment of decentralized energy resources, including rooftop solar and smart loads such as electric vehicles and smart appliances ([International Energy Agency \(IEA\), 2018](#)). These changes are influencing the way the power systems are planned and operated.

Over the past two decades, nations introduced climate change policies to pursue lower GHG emissions, which included the adoption of variable renewables ([Nuclear Energy Agency \(NEA\), 2019](#)), incentivized in part by subsidies. Some countries, for example, Canada, France, Germany, United Kingdom and the United States, experienced an increasing share of renewable sources for electrical power generation as part of their climate change policies, as shown in [Fig. 1](#) based on data from [US Energy Information Administration \(2019\)](#).

Over the same timeframe of the last two decades, the share of nuclear power generation declined in Germany and South Korea, while Canada, France, the United Kingdom and the United States experienced relatively stable shares of nuclear power generation as shown in [Fig. 2](#), based on data from the [US Energy Information Administration \(2019\)](#).

The increasing share of renewable sources of electrical power generation has led to surplus baseload electricity generation, which has resulted in baseload sources operating at a lower capacity. In France, the nuclear reactors fleet of the Électricité de France (EDF) fleet provides more than 70% of the total electricity supply of France. Therefore, the entire French nuclear fleet operates flexibly to some extent, with all reactors participating in frequency services (up to $\pm 5\%$ of the rated power), and two-thirds of the plants used for load-following (with 5% per minute power ramp rates). The combination of load-following capabilities with the size of the nuclear fleet allows EDF to achieve significant shifting of the nuclear load that can reach more than 12 GWe of nuclear capacity over a few hours (i.e., 20% of French installed nuclear capacity).

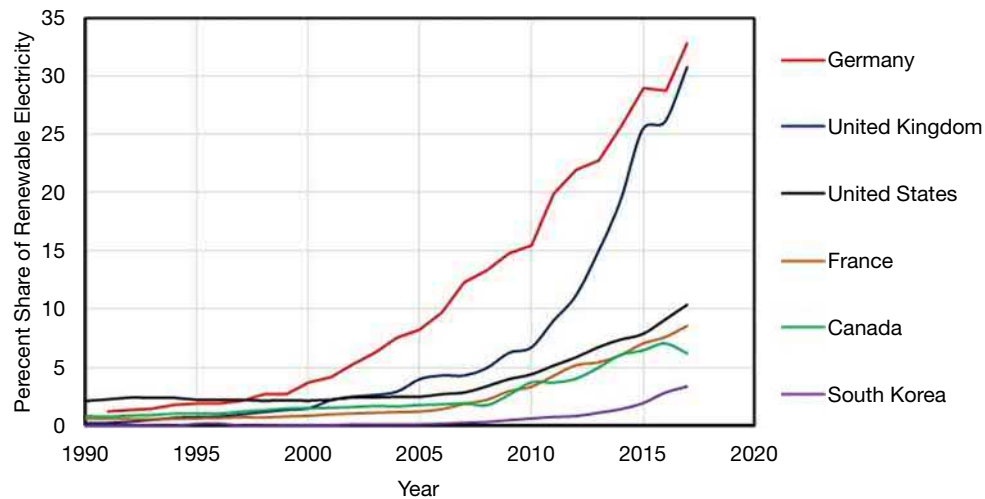


Fig. 1 Increasing share of non-hydro renewables in electricity supply.

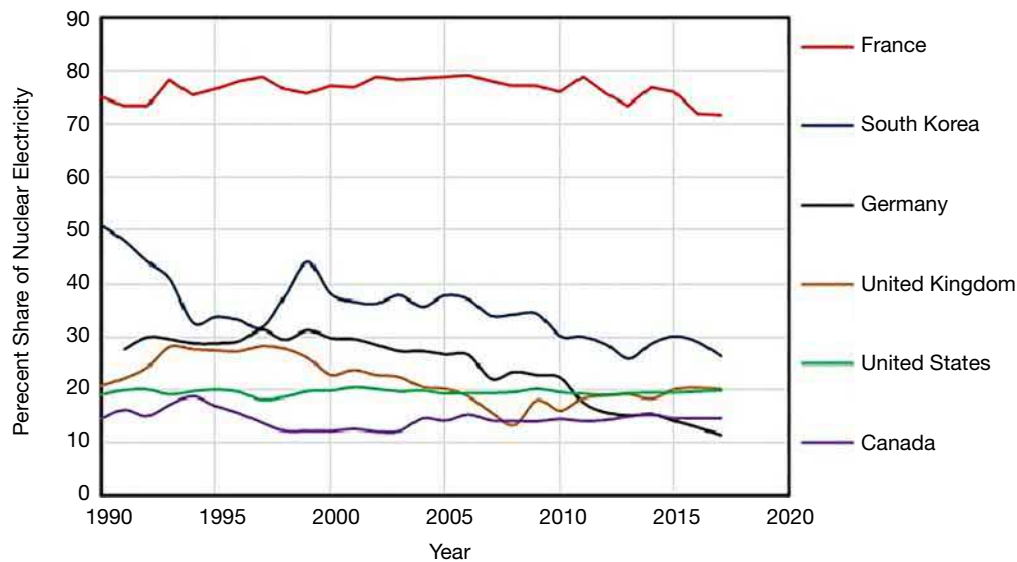


Fig. 2 Share of nuclear power in electricity supply.

In Germany, the renewable sources generate about 30% of the total electricity. Large variations in VRE generation requires nuclear plants to have enhanced load-following capabilities, which adversely affects the economics of nuclear power generation. The load-following capabilities were built-in at the construction of the German nuclear fleet but have only started to be used in response to renewables intermittency.

While the nuclear plants in France and Germany have been operating in flexible mode, not all currently operating nuclear plants can provide flexibility services required to accommodate high shares of VRE sources on the grid. Operational flexibility of nuclear plants not only depends on the original design considerations but also is dependent on the regulatory restrictions and electricity market structure aspects of generating power. Flexible operation of nuclear power plants, resulting from priority dispatch of renewable electricity result in reduction in capacity factors and thus affects the nuclear plant economics adversely.

In some instances, surplus generation from renewable resources has led to negative prices, where the nuclear plants had to pay to dispatch the power on the grid to keep the plants running.

Nuclear power plants are capital intensive and have lower fuel costs compared to fossil fuelled- plants and thus are more economical when operated at full power output rather than at modulated output. Reduced load factors due to flexible operation increases the average electricity production cost.

Integration of the renewable sources with baseload generators such as nuclear power plants makes the grid management more challenging. Forecasting errors in variable renewable output also requires carrying on a higher amount of spinning reserves in the system. Provision of physical inertia in a system with a high share of intermittent VRE sources is a topic of emerging research since the inertia is implicitly provided by the dispatchable plants and not by VRE sources. Hybrid nuclear-renewable energy systems,

large-scale energy storage and possibilities of using thermal energy as a coproduct from the advanced Gen IV reactors would mitigate the challenges discussed in the previous paragraphs and make the new nuclear reactors viable as a climate change solution.

Economic considerations for new nuclear

Levelized cost of electricity (LCOE) is often used to compare the cost of electricity generation from various sources and it is based on the plant-level costs (Rothwell, 2021). However, the cost of electricity provision to the consumer and to the broader society also includes the electricity system costs of transmission and distribution and maintaining grid reliability, as well as the cost of impacts on air pollution and climate change. Thus, the full cost of electricity provision includes three categories: plant-level costs, grid-level system costs and the external social and environmental costs. The NEA report (2018) concluded that the grid-level system costs associated with VRE generation are large and increase over-proportionally with the penetration level of renewables. In comparison, system costs of dispatchable technologies, such as nuclear, are at least one order of magnitude lower. However, the share of VRE in electricity generation has continued to grow globally. It must be noted that the plant-level costs for VRE tend to be lower because of the tax credits and these low costs are often passed on to the consumers; however, the consumer ultimately bears the cost in terms of higher deficits and higher taxes to compensate for the tax credits for renewables (MIT, 2019).

Integration of VRE sources into the existing grids with dispatchable electricity generating sources has been widely studied. MIT report (2019) compared the possible scenarios under the US electricity-sector policy to reduce CO₂ emissions by 90% in 2050 compared to 2005 level. The MIT study concluded that the additional system costs of wind and solar are minimal until they reach about 40% of power supply, but after that level these extra costs rise, making room for other power technologies such as nuclear, which can significantly reduce the carbon price needed to achieve deep decarbonization of the electricity-sector in the United States. In the absence of low-carbon, low-cost advanced nuclear alternative, it will require substantial carbon price to achieve the emissions target because the intermittency of renewables will require a combination of redundant renewable capacity with significant curtailments, energy storage and flexible gas capacity. However, the availability of advanced nuclear reduces the needed carbon price in the power sector from near \$120 per ton (2006 \$) of CO₂ without nuclear option to under \$40 per ton (2006\$) of CO₂ to achieve 90% emission reduction by 2050. Another NEA study (2019) also assessed the costs of alternative low-carbon electricity systems capable of achieving strict carbon emission reductions to 50 gCO₂ per kWh with different shares of renewable and nuclear resources. The NEA study shows that the system costs of electricity provision increases with increasing share of VRE in the electricity mix. Since the load factor of VRE is significantly lower than the conventional thermal (nuclear and fossil) power plants, a significantly higher total generation capacity is required to produce the same amount of electricity. The NEA study also found a striking effect of the deployment of low marginal cost VRE sources on the electricity market with increase in the volatility of the electricity prices. The number of hours with zero electricity prices increase with increasing penetration of VRE, starting with 60 hours per year of zero price at 30% VRE penetration increasing to 3750 hours per year of zero price at 75% VRE penetration. Yet another market effect of increase in the share of VRE is the decrease in the average price of electricity generated by the VRE, because all wind and PV solar generation will be concentrated in limited number of hours, producing disproportionately more electricity.

Electrical Power Research Institute (EPRI) analyzed (EPRI, 2018) the economic drivers and barriers for the advanced nuclear reactors in the future energy markets in the United States. The analysis found that the deployment of the new advanced reactors in the United States would require lower cost technology options and favorable policy environment (for example, carbon pricing), supplemental revenue streams (such as process heat sales), or a combination of these factors. EPRI analysis considered various scenarios around a range of advanced reactor capital costs, natural gas prices, environmental policy (carbon pricing) and additional non-electricity revenues (process heat). Capital costs ranged from \$2000 per kW to \$5000 per kW, although the high value is still considered lower than the cost of new light water reactors. The scenarios favoring deployment of advanced nuclear reactors generally included lower capital costs, particularly in the absence of new environmental policies with low natural gas price environment in the United States. Strong environmental policy and additional non-electricity revenues could tolerate higher capital costs for nuclear.

Another recent MIT study (MIT, 2018) found that new nuclear plants are not a profitable investment in the United States and western Europe. The high cost of new nuclear plants is a significant constraint to the growth of nuclear power under the scenarios of “business-as-usual” and modest carbon emission constraints. The MIT report shows that the overnight capital cost (\$ per kW) for recent Generation III+ reactors in the United States and western Europe are 30–70% higher than the 2009 benchmark of \$5000 per kWh. World Nuclear Association (WNA) also reported similar significant overnight cost increases for the PWR and BWR type reactors in North America and western Europe compared to East Asia (WNA, 2017).

The MIT report (2018) also recommended Government funding programs around prototype testing and commercial deployment of advanced nuclear reactors. The MIT report also analyzed attributes for successful nuclear build projects and cost drivers for those projects. The report recommends focusing on proven project and construction management practices to increase the probability of success in the execution and delivery of new nuclear plants. It further notes that the advanced reactors are expected to utilize compact and simpler designs to reduce cost and capture new markets such as process heat.

Generation IV International Forum (GIF) issued comprehensive cost estimation guidelines (GIF, 2007) for nuclear reactors. Advanced nuclear reactors represent dramatic evolution from the conventional reactors and there is a great uncertainty in the cost estimates in the absence of any commercial projects. Energy Options Network (EON) used the cost estimation framework of the GIF and worked with the leading advanced reactors companies to obtain reliable, standardized cost projections of their Nth-Of-A-Kind (NOAK) plants (EON, 2017). The reactor types being developed by the advanced reactors companies included

the reactor types being developed under the GIF collaborations. The average total capital cost in 2016 dollars for the advanced nuclear reactors was estimated to be \$3782 per kW installed capacity compared to the benchmark PWR cost of \$6755 with uncertainties ranging from $\pm 10\%$ to $\pm 50\%$. It should be noted that the benchmark PWR is NOAK type, while the new PWRs built as units 3 and 4 at Vogtle in the United States are considered as First-Of-A-Kind (FOAK) and could have significantly greater costs. While acknowledging the uncertainties in the cost, the report highlights the strategies being pursued by the advanced reactors companies to reduce the capital cost including:

- simpler and standardized plant designs
- factory fabrication
- lower materials requirements
- modularization
- reduced scope for engineering, procurement and construction
- shorter construction time
- higher power density
- higher efficiency.

The EON study found the average LCOE for advanced reactors to be US\$60 per MWh compared to the benchmark PWR LCOE of \$97 per MWh.

Although there is no specific definition of NOAK plant, it is supposed to reflect improvements and cost decline with each doubling of experience until the capital cost reaches an asymptotic value for the plants supplied by the same vendors and contractors. According to GIF cost estimation guidelines (GIF, 2007), the NOAK costs are achieved for the next plant after 8 gigawatts (GWe) of capacity have been constructed of a nuclear energy system. While 8 GWe criteria could have been based on the historical trend for conventional reactors, the learning factors could be different for the advanced reactors which could use factory-fabrication and modularization techniques. Learning factor represents capital cost decline with each doubling of experience in construction of nuclear plants and is related to the capital cost by the following equation.

$$C_N = C_1 \left(Lf^{\frac{\ln(N)}{\ln(2)}} \right)$$

where, C_N = capital cost of Nth plant, C_1 = capital cost of FOAK plant, Lf = learning factor.

Fig. 3 shows the evolution of capital cost of nuclear plants with learning factors of 0.95 and 0.975 representing cost declines of 5% and 2.5%, respectively, for each doubling of the experience.

Energy Technologies Institute (ETI) analyzed the cost drivers for the nuclear plants to better understand the schedule delays and cost increases for recent new projects in North America and Europe in contrast to those built elsewhere in the world (ETI, 2018). Reducing the capital cost requires reducing the project risks by increasing certainties in project schedule and budget. ETI report makes several recommendations to reduce the capital costs, including completion of the design prior to starting the construction. The ETI report also identified the potential step-reduction in the cost of advanced reactor technologies and small modular reactors (SMR), some of which are like those listed above.

In summary, a major consideration for the successful deployment of advanced reactors would be to reduce the capital costs which consists of two main components.

- *Capital (overnight) cost*: which can be potentially reduced through modularization, factory fabrication, design simplification and improved project management. Overnight capital cost includes all costs incurred before the plant is ready to start earning

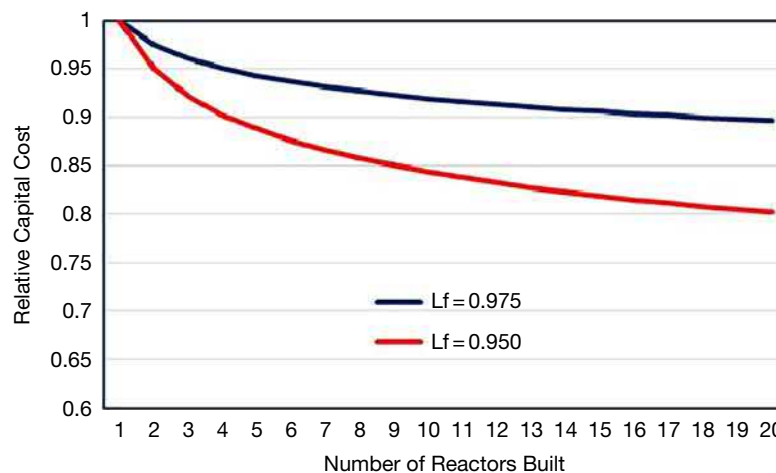


Fig. 3 Cost evolution of nuclear plants with learning factors.

revenue, except the time-value cost of the capital (interest during the construction). It can be considered as the cost incurred if the plant was constructed overnight without accrual of interest.

- *Cost of capital*: which requires favorable policies to finance nuclear projects to reduce the interest during the construction, as discussed later in this paper.

Flexibility requirements for new nuclear

Utility expectations

Electricity demand on a power system varies with time and the changes could occur on a short duration (seconds to minutes) or long duration (daily, weekly, seasonal). These demand changes are caused by the demand variation with time (day versus night, working day versus weekend), and weather conditions (summer versus winter). Typically, power system or grid operators manage the electrical supply to the consumers by forecasting the demand. They then schedule ramp rates needed to match the forecasted demand. Finally, they provide regulation service to correct discrepancies between scheduled and actual load, and load and generation, while maintaining frequency and voltage on the grid.

In addition to the demand variation, increasing integration of VRE sources also add challenges of supply variation which must be managed by the system operator. For the systems that allow priority dispatch for the VRE sources; the other dispatchable generating stations (nuclear, fossil, etc.) are required to have enhanced ramping and load following capabilities.

The Clean Energy Ministerial (CEM), a multinational initiative with current membership of 26 nations, launched the advanced power plant flexibility campaign in 2017 to commit the governments to make the power generation more flexible to allow integration of VRE sources into the power systems. As part of this campaign IEA published a report (IEA, 2018) that highlights the role of power plants in system flexibility and policies required to advance power system flexibility. Although not specific to the nuclear plants, the IEA study looked at the limitations of the current generation nuclear plant and options to enhance the flexibility but did not consider the advanced reactor concepts that are under development. Recently, the CEM also launched flexible nuclear campaign in 2019 with the objective of modeling revenue opportunities for flexible nuclear plants in various parts of the world to inform the government stakeholders as well as the designers of Gen IV reactors.

The utilities in Europe and the United States issued requirements for the advanced light water reactors (Gen III+) (EPRI, 2014) that demand the new nuclear plants to be capable of providing flexibility to the system. These utility requirements are mainly focused on operational flexibility of the nuclear plants. A typical expectation of daily power variation of new nuclear plant is depicted in Fig. 4.

The advanced Gen IV reactors are significantly different compared to Gen III/III+ reactors and use different fuels and different coolants and operate at higher temperatures: making it suitable for applications beyond electricity production. Therefore, to evaluate the flexibility of Gen IV type reactors, EPRI proposed expanded flexibility criteria (EPRI, 2017). EPRI's expanded flexibility criteria consists of a set of three sub-criteria, each having specific attributes as follows:

- operational flexibility
- deployment flexibility
- product flexibility.

The above criteria of flexibility are discussed in further details below.

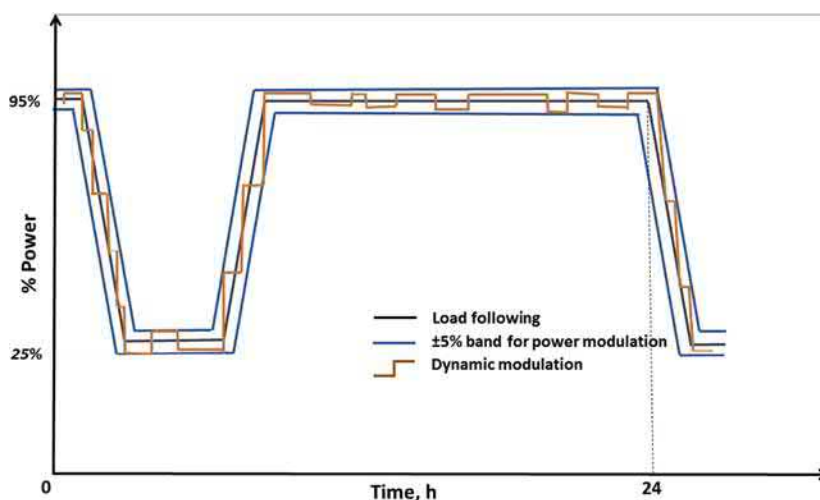


Fig. 4 Typical daily power variation of a flexible nuclear power plant.

Operational flexibility

The operational flexibility is generally understood as the ability of the nuclear power plant to respond to the variability of demand from the grid while maintaining the power quality. In addition to maneuverability, EPRI (2018) proposed other attributes of flexibility for advanced reactors systems, listed below.

1. *Maneuverability*: the ability to meet the ramping requirements for the grid and providing frequency control.
2. *Compatibility with hybrid systems*: The ability is considered important particularly for integration with VRE sources through hybrid energy systems which may include energy storage and other energy applications.
3. *Diversified fuel use*: The ability to operate using different types of fuel.
4. *Island mode operation*: The ability of the advanced reactor system to operate in isolation from electricity distribution networks.

The ability of a light water reactor (LWR) to provide flexibility services depends on the fuel condition. Typically, the LWRs do not participate in ramping for a few days (depending on the reactor and fuel design) after refueling until the fuel is conditioned. Fuel rods are manufactured with a prescribed gap between the fuel pellet and the inside cladding wall. During the reactor operation, the gap may reduce and lead to pellet-cladding interaction (PCI) which could lead to cladding failure. Load following operation could change local power density and cause PCI. Fuel vendors generally provide guidance on restricting flexible operation early in the fuel cycle until the fuel is conditioned to avoid PCI induced cladding failure. The reactors also do not operate in flexible mode towards the end of the fuel cycle due to limited excess reactivity to overcome xenon poisoning. Advanced fast reactors will not have xenon limitations experienced by the LWRs. In the case of gas-cooled reactors, the thermal fluctuations are avoided by controlling the coolant flow with the required power fluctuations, thus preventing ageing of components due to thermomechanical stresses. Advanced reactors are also flexible in terms of fuel; for example, the liquid fuels for molten salt reactors provide inherent fuel flexibility.

Deployment flexibility

The deployment flexibility is the ability of an advanced nuclear reactor to be licensed, financed, sited, and built under a range of external conditions. EPRI (2016) describes three attributes of deployment flexibility.

- *Scalability*: The scalability is the ability of an advanced reactor system to be sized to match energy demand, and to meet other local and regional requirements or to have the ability to be resized to increase energy output to meet changes to accommodate any growth in demand.
- *Siting*: Siting flexibility is described as the compatibility of an advanced reactor system with a variety of host locations, environments and conditions.
- *Constructability*: Constructability is described as the relative ease of building an advanced reactor system on schedule and within budget.

Product flexibility

Product flexibility refers to ability of the advanced reactor systems to be used for multiple missions. Most of the current nuclear plants produce electricity, although over 70 reactors world-wide have been used in cogeneration mode for various applications including district heating, providing industrial steam and for water desalination. Cogeneration is the term used for simultaneous production of electricity and thermal energy. In the case of nuclear power plants, the thermal energy use to date accounts for a small fraction (<1%) of the cumulative output of the nuclear power plants. Nonetheless, the individual thermal energy applications varied from 5 to 240 MWth.

Type of potential applications of nuclear thermal energy depend on the outlet temperature of the nuclear reactor. Past and current experience with nuclear cogeneration relates to lower-temperature applications such as district heating, sea-water desalination and process steam for industrial applications. The higher temperature advanced reactors would enable many more industrial applications including hydrogen production and petroleum refineries. Fig. 5 shows the advanced reactor temperatures and potential industrial applications. Cogeneration opportunities for advanced reactor systems are discussed later in this article.

Innovative approaches

Hybrid energy system

Hybrid system concepts were proposed to integrate nuclear with VRE sources to improve both the reliability of power and economics of integrated system. A hybrid energy system may be broadly defined as a single facility or cooperatively controlled system that integrates two or more energy inputs and produces one or more products, with an energy commodity such as electricity or transportation fuel (Ruth et al., 2014). Hybrid energy systems, which have a common grid for both variable renewable and dispatchable sources with storage possibility, to generate both heat and electricity, have been proposed as a way to integrate renewables with nuclear reactors by seeking new outlets in order to compensate for lower load factors.

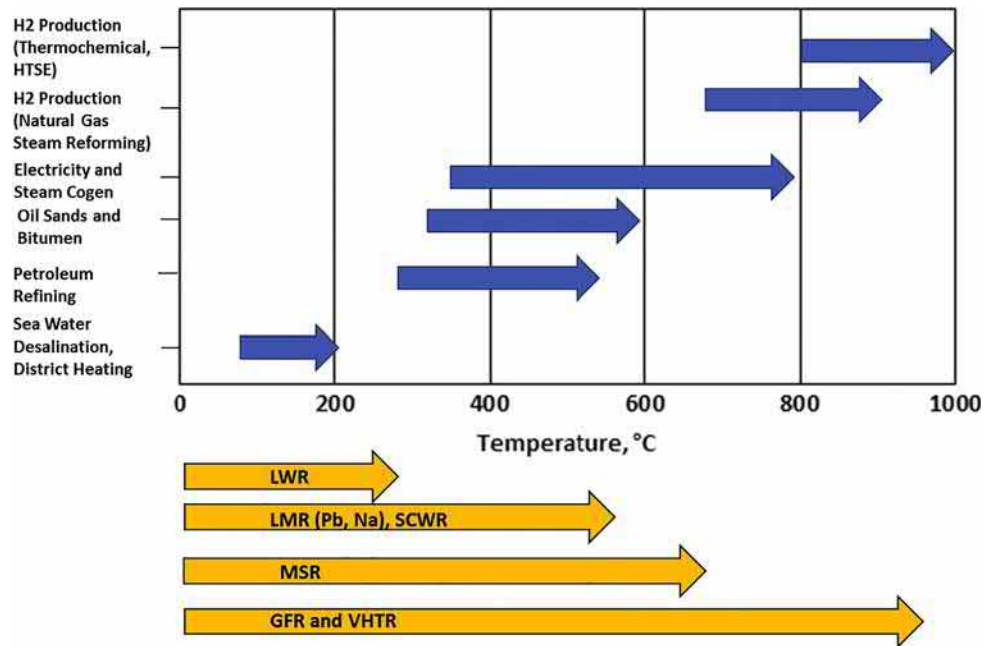


Fig. 5 Potential cogeneration applications of advanced reactors.

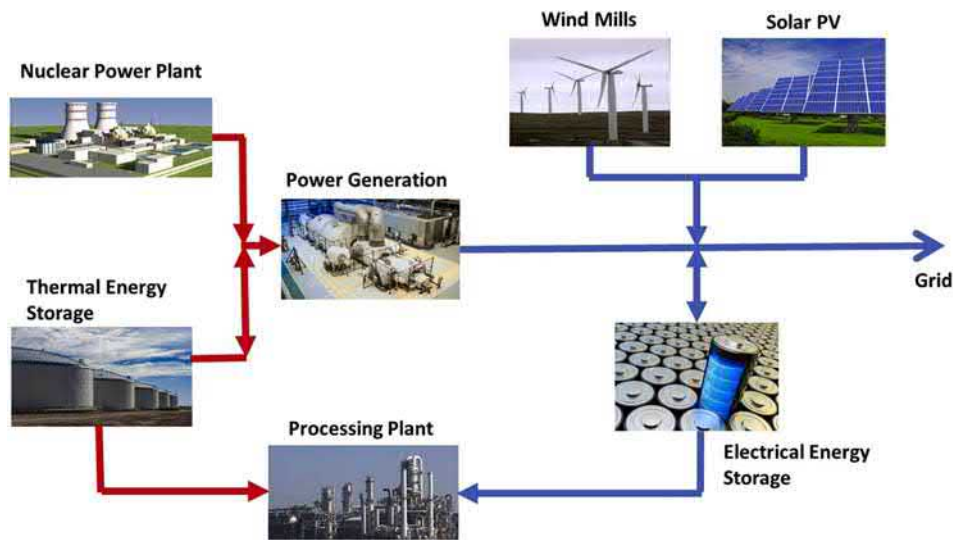


Fig. 6 Nuclear-renewable hybrid energy system.

Hybrid energy systems are comprised of multiple sub-systems, as depicted in Fig. 6, which may or may not be geographically colocated (Bragg-Sitton et al., 2016):

- nuclear heat generation source,
- a turbo-generator for conversion of thermal energy to electricity,
- at least one renewable energy source (VRE),
- an industrial process that utilizes heat and/or electricity from the energy sources to produce commercial-scale product, and
- energy storage (thermal and/or electrical).

Hybrid energy systems can be configured in various ways, for instance, as a tightly coupled system which acts as a single financial entity where nuclear generation and VRE sources and industrial production are controlled upstream of a single connection point to meet the flexible demand from the grid while operating the nuclear plant at capacity.

The key benefits of hybrid energy systems are:

- provide dispatchable, flexible, and carbon-free electricity generation

- provide synchronous electromechanical inertia to the grid (frequency control), and
- reduce carbon footprint of industrial production.

Although fully integrated hybrid systems have not yet been demonstrated, the component technologies are mature. There has been an increased interest in exploring the synergies of nuclear and renewable energy sources and hybrid systems are promising to provide the required flexibility for integration. The secondary output of the hybrid system, such as thermal energy for industrial use or hydrogen as a commodity, is highly dependent on the location. Several national approaches for nuclear-renewable hybrid energy systems were discussed in a technical meeting organized by IAEA (2019b). The viability of a hybrid system will depend on the geographical location and the business model.

Cogeneration

Use of nuclear thermal energy for non-electricity applications have been widely studied and published (IAEA, 2009, 2012, 2017a,b) with main motivation to reduce GHG emissions. More recently, nuclear cogeneration is also being considered for integration of nuclear plants with VRE sources on the grid for improving economics of nuclear operations.

As shown earlier, in Fig. 5, type of potential applications depends on the temperature of the thermal energy delivered by the nuclear reactor.

Low-temperature non-electric applications

Low-temperature non-electric applications are typically those requiring thermal energy at less than 300 °C, which can be supplied by most of the existing reactors and Gen III+ reactors.

Desalination

Sea-water desalination capacity around the world is increasing rapidly to meet the increasing demand of fresh water in both developed and developing countries. Most of the desalination plants depend on fossil fuel and thus are subject to fuel price fluctuations and supply reliability, which could undermine its economic viability. Nuclear desalination has been demonstrated, can be economically competitive and meet the product water quality through a meticulously designed coupling system between a nuclear reactor and a desalination plant. Although the current generation reactors can meet the energy demand and temperature requirements for all sea-water desalination technologies, some of the advanced reactors are also being proposed for this service.

In Aktau, Kazakhstan, 10 units of multi-effect distillation (MED) plants were coupled to a 1000 MWth sodium cooled, fast breeder reactor (BN-350) (IAEA, 2017b). It produced 14,500 m³ per day of very high-quality water for industrial and potable needs and ran for 26 years before shutting down in 1999. Sodium-cooled fast reactors are also being developed as Gen IV advanced reactors.

District heating

District heating systems exist in many countries in Europe and in North America that have harsh winters. Many of these systems are fossil-fuelled. There has also been a gradual shift to decentralized heating for individual houses or buildings because of low natural gas prices. District heating systems are usually based on either hot water or low-pressure steam and range in size from 600–1200 MWth for large cities down to 10–50 MWth for smaller communities. With increasing environmental concerns over the use of fossil fuels, the alternate district heating is being considered. Total heat demand for district heating is significant and could require hundreds of nuclear reactors (IAEA, 2017b).

As of 1998, 46 commercial nuclear plants in 12 countries were being used or have been used for heating purposes with a heat output between 5 and 240 MW, demonstrating a safe and reliable operation (IAEA, 2017b). One example of pressurized water reactor which is run in a cogeneration mode for district heating is Beznau nuclear plant in Switzerland. Beznau plant began to supply nuclear district heating in early 1980s and continues to do so today, serving a population of nearly 20,000. The peak district heat load is about 80 MWth. Experience in Switzerland shows that nuclear-based district heating is economical, safe, reliable and can be acceptable to public.

Other low-temperature applications

Unlike desalination and district heating, there has been limited use of nuclear heat for industrial applications. The largest such application was the use of medium pressure steam from the Bruce nuclear power plant for heavy water production in Canada (IAEA, 2017b). Bruce-A nuclear plants had the largest bulk steam system ever built, with a capacity of 5350 MWth and supplied ~750 MWth for heavy water production, 15 MWth for on-site building heating and about 72 MWth to an industrial park with food processing, ethanol and plastic film manufacturing plants. The heavy water production plants were located near the nuclear power plant and were licensed by the Canadian regulator and operated from 1973 until 1997. In Switzerland, Gosgen nuclear plant supplies about 45 MWth thermal energy to a cardboard factory and a paper mill, using medium pressure steam (1.2–1.5 MPa).

Advanced Gen IV reactors that are under development present additional opportunities for industrial applications because of higher outlet temperatures compared to the existing reactors. A comprehensive study of European industrial heat market (Bredimas, 2004) focused on application of high temperature reactors for industrial cogeneration, identified significant low-temperature thermal energy demand that can be provided by either Gen III+ or other types advanced Gen IV reactors.

High-temperature non-electric applications

High-temperature non-electric applications require thermal energy supply at temperatures above 300 °C and therefore are non-existent due to the limitation on reactor outlet temperature of existing reactors. Ongoing development of advanced reactors with significantly higher temperatures has created possibilities of using nuclear heat for additional industrial applications.

Hydrogen production

Hydrogen has long been considered as an energy carrier but has not made much impact in the total world energy supply because of the practical difficulties of introducing hydrogen in the energy system. To be used as energy carrier, hydrogen must be produced using a primary, low-carbon source of energy and the energy efficiency of the production process must be such that hydrogen can be economically deployed as an energy carrier. Over 90% of the hydrogen currently produced is from fossil raw materials such as natural gas, using fossil fuel as source of energy; and is primarily used in chemical production plants. Only a small fraction of hydrogen is currently used in the energy sector, mainly in fuel cells for electricity production or in vehicles.

High-temperature water-splitting processes for hydrogen production are being developed alongside the development of high-temperature reactors. GIF has a special project for development of high-temperature hydrogen production processes as part the development of the Very High Temperature Reactors (VHTR) (GIF, 2002). The high-temperature processes include thermochemical cycles (such as sulfur-iodine and copper-chlorine) and high-temperature water electrolysis (HTSE) (IAEA, 2009).

Japan Atomic Energy Agency (IAEA) has operated a 30 MWth, helium-cooled high temperature test reactor (HTTR) and has also developed sulfur-iodine thermochemical process for hydrogen production (Yan et al., 2018). One of the main uses of hydrogen in Korea is envisaged to be for steel making using direct reduction of iron ore. At USDOE's Idaho National Laboratory, a high-temperature steam electrolysis (HTSE) process is being developed for hydrogen production using high-temperature reactor (O'Brien et al., 2010). Much research has also been done in exploring large scale hydrogen storage in salt caverns.

Ability of converting hydrogen back to electricity using fuel cells offers additional flexibility for nuclear plant operation as the hydrogen can be used both as energy storage as well saleable product.

Other high-temperature applications

High-temperature cogeneration opportunities for high-temperature reactors were investigated as part of the European Union-supported EUROPAIRS project (Bredimas, 2004; Futterer et al., 2014). The EUROPAIRS study found that almost one third of the European industrial heat market is "externalized" in the form of cogeneration, limited to 550 °C, particularly in the chemical, plastic, pulp and paper industries. This market can be captured by a "plug-in" high-temperature reactor, replacing the existing fossil-fuelled operations. The study also found a significant potential "pre-heating" market potential of nuclear energy for high-temperature industrial applications such as, iron and steel, glass and cement manufacturing. The EUROPAIRS study also looked at "Polygeneration" sub-market involving production of base raw materials such as industrial gases (hydrogen, nitrogen, oxygen) in addition to cogeneration of heat and power. Polygeneration could provide additional flexibility for nuclear power plant and improve its economics particularly for integration with VRE sources, while easing the siting restrictions for the plant. Various studies of potential cogeneration of high-temperature reactors for petrochemical and steel sectors are also summarized in an IAEA report (IAEA, 2012).

Although various cogeneration possibilities for high-temperature reactors have been studied; the challenges of economic competitiveness, licensing aspects of colocation and coupling of nuclear plant to a production plant and public acceptance have not been explored in detail.

Energy storage

Energy storage systems have been used over the past several decades and have helped the reliability and flexibility of the power systems. Pumped-hydro energy storage has been widely deployed throughout the world and represents more than 99% of the worldwide bulk storage capacity (Luo et al., 2015). One of the main contributions of the bulk energy storage systems has been in meeting the peak load demand. During the peak periods, power supply from base-load generators must be complemented by more-flexible but less economic oil- or gas-fired generators. The bulk energy storage systems provide the utilities with a means of providing economical alternative to meet the peak load demand without additional GHG emissions. Pumped-hydro systems have also been used for frequency control purposes by providing a large amount of power when generated electricity is in short supply. Apart from the pumped-hydro systems, other types have also been successfully deployed, including the batteries, compressed-air system, flywheels, and thermal energy storage.

The increasing penetration of VRE sources have created emerging market needs for energy storage systems. Although pumped-hydro systems are matured, the potential for further deployment is restricted by the availability of suitable geographic locations. There have been advances in the alternate technologies for energy storage. As discussed in the previous section, technologies for GHG-free production of hydrogen using nuclear energy are under development and present yet another option for energy storage in conjunction with hydrogen fuel cells. Thermal energy storage is claimed to be significantly less expensive compared to the electrical energy storage. Thermal energy storage using steam accumulators and sensible heat storage using oil and molten salt have already been deployed and other options for large-scale sensible heat storage are being investigated (MIT, 2017).

The available energy storage technologies are shown in Fig. 7.

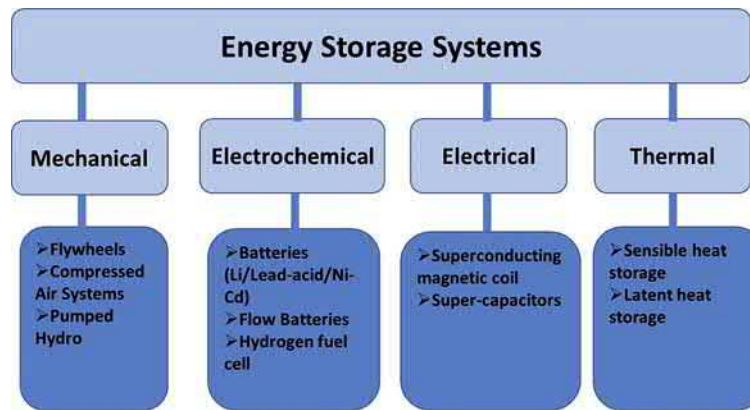


Fig. 7 Energy storage system technologies.

The energy systems described in [Fig. 7](#) span across a range of capacities and applications. Selection of energy storage for an application is based on the following considerations:

- power rating (MW) and power density (MW per m³)
- energy content (MWh)
- response time
- storage duration (minutes to days)
- energy loss (% per day)
- cycle efficiency (%)
- lifetime (years)
- maturity of technology
- energy cost (\$ per kWh) and capital cost (\$ per kW).

A comparison of various energy storage systems is presented in by [Luo et al. \(2015\)](#).

Large-scale energy storage would enable integration of advanced nuclear reactors with the VRE sources and would provide the required grid flexibility and reliability. Such energy storage systems would also enable deployment of hybrid energy systems discussed earlier. Thus, the innovative approaches discussed in this section, including hybrid energy systems, cogeneration applications using nuclear thermal energy and simultaneous development of large-scale energy storage would facilitate deployment of advanced reactor systems in the future energy markets.

Policies required for nuclear deployment

Various policies have been tried out and/or implemented in different jurisdictions to encourage GHG reduction ([Mendoza et al., 2018](#)). As discussed earlier in this article, some of the policies such as market liberalization and priority dispatch for renewables could adversely affect the economics of nuclear generation. However, some other policies may improve the economics of nuclear generation and could be conducive to future nuclear deployment. Policy issues to enable nuclear deployment include taxes and credits, electricity pricing, and financial arrangements.

Carbon taxes and zero emission credits

In response to the possibility of a premature closure of nuclear power plants some jurisdictions considered implementing zero emission credits which are like renewable energy credits ([Nuclear Energy Institute, 2018](#)). For example, New York, Illinois, Connecticut, New Jersey, Ohio, and Pennsylvania in the United States allow utilities with nuclear power plants to apply for such credits ([Tsai and Gülen, 2017](#)). The value of the zero emission credits varies among the jurisdictions. In the case of New York, the zero emission credits are first purchased by the load serving entity, and the cost of the credit is recovered from customer bills. Such programs have been challenged, and in some cases, the challenges have been dismissed, though there continues to be opposition to such programs.

A potential issue for nuclear is that zero emission credits are based on power generation each year. If the share of renewables increases, then the total value of the zero-emission credit from nuclear power generation may decline over time, especially in jurisdictions that provide priority for renewables. Another potential issue is that the timing of receiving the credit may not come quickly enough to offset the rising costs of generating power as the renewable share of electricity generation rises, which may require additional debt load if revenues are declining and there is insufficient working capital.

A carbon tax is a tax set by government in terms of dollars per ton of carbon dioxide (CO₂) emitted. The carbon tax is part of annual costs incurred by a utility. The introduction of a carbon tax would likely make nuclear more competitive (EPRI, 2018; NEA, 2011). Different carbon taxes may be adopted by different regions to enable nuclear power more competitive. Regional differences can reflect different manufacturing culture, environments for regulation and public acceptance of nuclear plants. Furthermore, break-even carbon tax will depend on the prices of fossil fuels in different regions of the world.

As discussed earlier, EPRI analysis (EPRI, 2018) of the economic drivers and barriers shows that strong environmental policy and carbon taxes would favor deployment of advanced reactors in the future energy markets in the United States.

Electricity pricing and market structure

Nuclear power plants struggle in the liberalized markets, firstly with competition from low-natural gas prices and secondly with the over-production from VRE sources which tend to drive down the market price. In order to offset the volatility of wholesale market prices and potential decline in prices, a fixed contract may be used when an electric utility with nuclear power generation must use load following mode. Guaranteed long term electricity contractual agreements, for example, strike price in the United Kingdom, and Power Purchase Agreement for Bruce Power in Canada (Barkatullah and Ahmad, 2017; Lucet, 2015) would ensure economic viability of nuclear and would spur further investment in new nuclear plants.

The appropriate wholesale electricity price should account for both the energy market and the capacity market to provide an incentive for utilities to meet the demand for energy and reserves. Liberalized markets consider energy-only electricity pricing. Under an energy-only pricing scheme, however, wholesale energy market pricing does not account for the value of reliability.

Financing of new nuclear

As discussed earlier, building a new nuclear plant is capital intensive project and to ensure successful deployment requires both reducing the overnight capital cost and the cost of the capital. Financing nuclear power project remains a challenge today. Raising capital for new nuclear plants will be more challenging in markets where nuclear is required to be integrated with renewable sources on the grid which may cause low capacity utilization of the plant.

Even though the government's financial role still dominates the deployment of new reactors, governments are looking for private sector participation. Private financing schemes used are corporate financing, investor financing, and vendor financing. Private sector financing has gained some momentum but is still constrained by the consequences of the 2017–18 global financial crisis, which limits financing, still seeks government support, and puts a risk premium on nuclear projects based on past performance (Barkatullah and Ahmad, 2017). Reducing project and liquidity risks are, therefore, important factors to consider for future research. New financing mechanisms will be vital for the deployment of future nuclear builds. In this regard, the regulated asset base (RAB) model has been proposed in the United Kingdom as an alternative to using a strike price and direct investment for financing a new nuclear reactor deployment (Helm, 2018). Strike price is the agreed price of electricity under a long-term Contract for Difference (CfD) between the nuclear plant owner and the electricity distributor, usually publicly owned. If the market price is lower than the agreed strike price, the government pays that difference to the plant owner and passes on the cost to the consumers. If the market price is above the strike price, generator pays the difference to the electricity consumers by reducing the tariffs. RAB models are used in the United Kingdom for funding monopoly infrastructures, where a regulator grants license to a company to charge a regulated price to the users of the infrastructure. RAB model has potential to reduce the cost of raising private finance for new nuclear projects, thereby reducing costs to the electricity consumers.

The Expert Finance Working Group (EFWG) set up by the UK's Department of Business, Energy and Industrial Strategy looked at what was needed to attract private financing to small modular reactor projects (EFWG, 2020). The EFWG recommends that the Government should help to de-risk (perceived and real risks) the small nuclear market in order to enable private sector to develop and finance projects. Reducing the risks of the projects and making the consequences of the risk manageable are fundamental to creating the best environment for the private sector to be able to invest in small nuclear.

Canadian Small Modular Reactor Roadmap, 2018 also recommended government funding to cost share with the industry in one or more SMR demonstration projects for advanced reactor designs. It also recommends that the government share risk with the private investors to incentivize the first commercial deployment in Canada.

Summary

Main challenges for deployment of advanced nuclear reactors in the future low-carbon energy market would be integration with the VRE sources and economic competitiveness. The developers of the advanced reactor systems are already exploring the means of reducing the investment cost through innovative fabrication and construction techniques. The Gen IV system developers are also taking into consideration the flexible operation requirement and associated thermal cycling and fatigue, reactivity control and fuel optimization at the design stage.

Various innovative approaches are also being explored to enable deployment of advanced reactors. Nuclear-renewable hybrid energy system concepts would provide successful integration of nuclear with renewables while ensuring economic viability of the project. The higher outlet temperatures of Gen IV reactors present opportunities for using thermal energy as an alternative

product to electricity to replace use of fossil fuels for process heat. Ongoing development of large-scale energy storage and technologies for producing hydrogen using nuclear energy would also enable Gen IV reactors to meet the flexible demand from the grid while operating the power generators to full capacity, thus, ensuring overall economically viable operation.

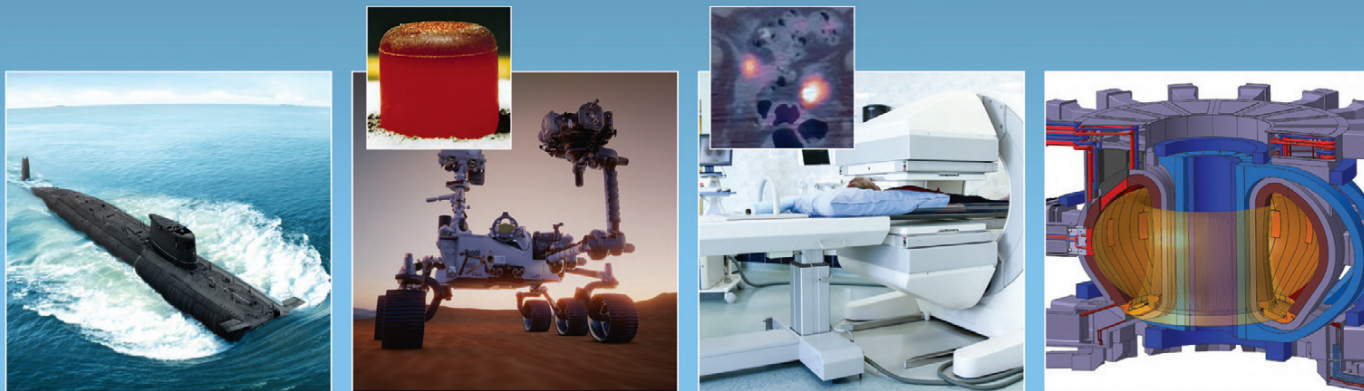
On the other hand, policy- and market-driven challenges could be more formidable compared to technical challenges of designing the reactors for flexible operation. Existing market designs do not always value reliability aspects of supply. Significant policy changes will be required to recognize and value reliability of nuclear generation as part of a resilient and reliable power grid. Favorable policies related to carbon taxes, electricity pricing and financing would ensure successful deployment of advanced nuclear reactors in the future energy market and would significantly help towards meeting the climate change goals.

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Ehud Greenspan

*Retired Professor, Nuclear Engineering Department,
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Bottom: View of a nuclear power plant by the sea in Vandellos, Spain. It is a 1043 MWe Pressurized Water Reactor.
Top, from left to right (sample applications of nuclear energy):

- a submarine powered by nuclear energy
- an exploration vehicle/lab on MARS powered by a Plutonium-238 nuclear heat source (insert)
- a nuclear medical diagnostic machine for humans with a sample of PET-CT scan (insert)
- an artist vision of a future Tokamak-type fusion energy demonstration power reactor

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Associate Laboratory Director, Idaho National Laboratory, Idaho Falls, ID, United States

Alan E. Waltar

Retired Professor and Head, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

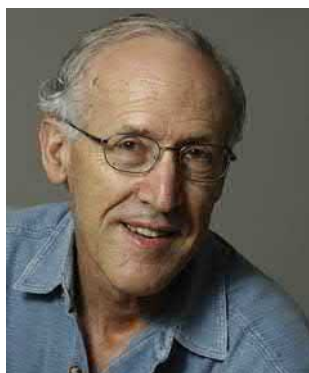
David J. Campbell

Munich, Germany

Andrew C. Kadak

Kadak Associates, Inc., Port Saint Lucie, FL, United States; Former Professor of the Practice, Massachusetts Institute of Technology, Cambridge, MA, United States

EDITOR-IN-CHIEF



Ehud Greenspan

University of California at Berkeley
gehud@nuc.berkeley.edu

Professor Greenspan (Ph.D., 1966, in Nuclear Science and Engineering from Cornell University, Ithaca, N.Y., USA) retired from the Department of Nuclear Engineering of the University of California at Berkeley (UCB) in 2014 but served as a Professor in the UCB Graduate School until 2019. He has nearly 60 years of nuclear energy related research experience starting in 1961 as an M.Sc. student at the Department of Nuclear Engineering of the Israel Institute of Technology. His research spans a wide range of nuclear engineering related disciplines including: theoretical and experimental reactor physics; computational methods development including development of new optimization methods for determining the minimum critical mass and for finding optimal design of radiation shields, fusion reactor blankets and medical installations; conception and analysis of advanced fusion fuel cycles and of fusion-fission hybrid reactors; as well as advanced fission reactors and nuclear fuel-cycle conception, design and analysis. The latter activity was the focus of his research during the last twenty-five years. The thrust of this

research is improving sustainability of nuclear energy along with improving economics, safety, and proliferation resistance. He has been the Principal Investigator on studies of dozens of advanced nuclear systems, including those corresponding to Generation-IV reactors and to Small Modular Reactors (SMRs). He was one of the early advocates of SMRs and led a multi-institutional international team that developed a preliminary conceptual design of the Encapsulated Nuclear Heat Source (ENHS)—an innovative nuclear battery-type SMR that is free of pumps, pipes and valves in the primary reactor system. The advanced reactor technologies addressed in his research include: heavy water reactors, pressurized and boiling water reactors, prismatic and pebble-bed liquid-salt cooled reactors, molten-salt reactors, sodium and lead-alloy cooled fast reactors, breed-and-burn reactors, heat-pipe reactors as well as fusion-fission hybrid reactors and accelerator driven nuclear reactors. His research is documented in over 400 refereed publications, over 200 non-refereed publications, 5 book chapters, and a number of patents.

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Introduction to the Encyclopedia of Nuclear Energy

Ehud Greenspan, Department of Nuclear Engineering, University of California, Berkeley, Berkeley, CA, United States

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Introduction

From the beginning of the Universe through to the present day, nuclear energy, nuclear reactions, and nuclear radiations have played major roles in the evolution and shaping of our universe and planet Earth. All the chemical elements we find on Earth were produced by nuclear processes in the Universe. Planet Earth is continuously exposed to nuclear radiation from primordial radioactive isotopes produced in cosmological processes that followed the Big Bang. A number of primordial radioactive isotopes are still among the Earth's constituents, including thorium ^{232}Th , the uranium isotopes ^{235}U and ^{238}U as well as potassium ^{40}K . The spontaneous radioactive decay of the nuclides of these isotopes releases nuclear energy that provides a major contribution to the heat balance of planet Earth. The internal heat source is responsible for keeping the outer core of the Earth in a molten magma state and played a major role in the geologic evolution of the Earth's continents and oceans. The primordial radiation and Earth's heat source may have also contributed to the creation as well as evolution of life on Earth. Since its birth, the Earth has been also constantly bombarded by galactic cosmic nuclear radiation, which primarily consists of protons (hydrogen nuclei) and alpha particles (helium nuclei). Hence, mankind as well as any organism on Earth, were and are being continuously exposed to background nuclear radiation throughout their evolution. The background radiation level as well as the radioactive heat source due to the decay of the primordial radioactive isotopes were significantly higher at the time of Earth formation and are slowly decreasing with time.

It is only within the last hundred and twenty years or so that mankind discovered the structure of the atom and its nucleus and learned how to harness the enormous energy potential contained within the nucleus as well as the different types of radiation nuclei and atoms can emit. Nuclear radiation was discovered in 1896, when Henry Becquerel and Marie Curie discovered that salts of uranium emitted radiation that could be detected with photographic plates. In the 125 years since this discovery, mankind deciphered the structure of the atom and its nucleus, learned how to produce artificial radioactive isotopes (Irène Joliot-Curie and Frédéric Joliot in 1934), discovered how to fission certain heavy isotopes (Otto Hahn, Fritz Strassmann, Lise Meitner and Robert Frisch, 1938/39) and learned how to establish a fission chain reaction (Leo Szilard, Enrico Fermi et al., 1942) and developed numerous applications of nuclear radiation and nuclear energy. The feasibility of releasing nuclear energy by fusing nuclei of light isotopes was postulated already in 1920 (Arthur Eddington) who speculated that such nuclear reactions provide the enormous energy released in the sun and stars. Although research of harnessing nuclear fusion for controlled generation of nuclear energy is being pursued since the late 1940s, mankind has not yet succeeded in producing sufficiently more energy from fusion reactions than the energy invested in inducing these reactions. Additional information on the history and fundamentals of harnessing nuclear energy can be found in (Norman, 2021).

Encyclopedia scope

The Encyclopedia of Nuclear Energy (The Encyclopedia) provides a comprehensive authoritative coverage of a wide range of topics related to the generation, utilization and safe handling of nuclear energy, nuclear radiation and associated technologies, for the benefit of mankind and planet Earth. It covers past developments, the present status, and research and development of advanced innovative technologies that may enable improvements in the generation and utilization of nuclear energy (including nuclear radiation). The Encyclopedia consists of 272 thematic articles written by experts and arranged in 13 Sections; each Section covers the subject area identified in Table 1.

The fundamental concept of this encyclopedia is to provide all this vast and very diversified information in convenient to read cross-referenced articles within a single publication. Each article provides an effective solid foundation of its topic and references for readers interested in deeper exploration of the subject matter.

Table 1 Encyclopedia sections, number of articles per section and the section editors.

Sec. #	Section title (# of articles)	Section editor	Section editor affiliation
1	Fundamentals and history of development of nuclear energy (17)	Eric B. Norman	Professor in the Graduate School, Nuclear Engineering Department, University of California at Berkeley
2	Commercial nuclear power reactors and their design (22)	Russell E. Stachowski	Chief Consulting Engineer, Global Nuclear Fuel
3	Advanced nuclear reactor concepts under R&D (28)	Alexander Stanculescu	Retired Generation Four International Forum Technical Director; Retired Idaho National Laboratory Division Director
4	Safety, regulation, and decommissioning of commercial nuclear power reactors (24)	Joy L. Rempe	Rempe and Associates, LLC
5	Fuel cycle front-end: from mining through utilization (17)	Douglas C. Crawford	Director, Transient Reactor Test Facility, Idaho National Laboratory.
		Colby B. Jensen	National Technical Lead for Fuel Safety Research, Idaho National Laboratory
6	Fuel Cycle back-end: from recycling to deep geological disposal (19)	Christophe Poinssot	Deputy General Director (CEO) and Scientific Director of the French Geological Survey (BRGM) ^a
7	Radiation protection (21)	Laurence Miller	Professor Emeritus, Department of Nuclear Engineering, the University of Tennessee
8	Non-electric applications of nuclear energy (15)	Shannon M. Bragg-Sittou	Lead for Integrated Energy Systems, Idaho National Laboratory ^b
9	Nuclear power for space (16)	David I. Poston	Chief Reactor Designer, Los Alamos National Laboratory
10	Research reactors (23)	Sean O'Kelly	Associate Laboratory Director, Idaho National Laboratory. Chair of IAEA Technical Working Group on Research Reactors
11	Medical, industrial and agricultural applications of nuclear technology (26)	Alan E. Waltar	Retired Professor and Head, Department of Nuclear Engineering, Texas A&M University, College Station
12	Nuclear fusion R&D (26)	David J. Campbell	Munich, Germany. Former Director for Science and Operations, ITER Organization
13	Social issues of nuclear energy (18)	Andrew C. Kadak	Kadak Associates, Inc. Former Professor of the Practice Massachusetts Institute of Technology

^aAlso: Prof. of Nuclear Chemistry at the National Institute of Nuclear Science and technology (INSTN).

^balso National Technical Director for the DOE-NE Integrated Energy Systems program.

Intended readership

The articles are written at a level that users, with a bachelor degree in engineering, scientific and medical disciplines, especially in nuclear engineering, will find understandable and helpful. The Encyclopedia is intended to be an invaluable resource for students and will make an important contribution to the preservation of the vast amount of knowhow developed since the dawn of the nuclear era in the nuclear energy related fields and to its transfer to the new generation.

The Encyclopedia will also be very useful to industry and academia researchers and practitioners in the many scientific and engineering fields related to the generation and use of nuclear energy, including nuclear radiation. It will enable researchers and practitioners to refresh their memory in their field of specialization or expand their knowhow on less familiar subject areas.

The Encyclopedia will also be useful to members of the public with education in engineering or sciences who are interested in learning about specific nuclear energy or nuclear radiation related topics without having to struggle with textbooks or journal articles.

On the need for this Encyclopedia

The publication of this Encyclopedia is very timely for a number of reasons, including the following:

- Nuclear energy can play a very important role in drastically reducing the amount of fossil fuels burned for generation of electricity and, hence, the amount of greenhouse gas (GHG; primarily CO₂) emissions while, at the same time, making more electricity available for developing countries and for supporting the expected growth in the world population. Moreover, nuclear energy can make an important contribution to drastically reducing the amount of GHG emission from the transportation, industrial, residential and agricultural sectors of the economy. Hence, nuclear energy can make a vital contribution to meet the goal set by the 2015 Paris Agreement on climate change and ratified by many nations—to limit global warming to well below 2, preferably to 1.5 degrees Celsius, compared to pre-industrial levels. This point is elaborated upon below in the section: “On the Importance of Nuclear Energy in Avoiding Climate Change Catastrophe”. The state of the art in nuclear energy generation is covered in (Stachowski, 2021; Crawford, 2021).

- The Generation IV (GEN-IV) International Forum (GIF) was created in 2001 as a co-operative international endeavor of research, development, demonstration and deployment of advanced fourth generation innovative nuclear reactors that, hopefully, will be available for commercial deployment by 2030. The objectives of the innovative GEN-IV nuclear energy systems include improved sustainability, economics, safety and reliability, as well as proliferation resistance and physical protection. This encyclopedia provides an update on the state of the GEN-IV R&D efforts and explains the benefits the innovative GEN-IV reactors may provide. The Encyclopedia also describes (Stanculescu, 2021) the advanced nuclear fuel cycles that offer highly improved resource utilization, greatly reduced nuclear waste, and reduced proliferation risks.
- Over the past decade or so, there has been a surge in the number of private companies, mostly supported by venture capital, that are developing advanced innovative Small Modular Reactors (SMRs); reactors designed to generate between 10 to 300 MW of electricity per unit (Stanculescu, 2021). The SMRs offer a new paradigm in the generation and delivery of nuclear power; they promise to improve all aspects of design, engineering, construction and operation of commercial nuclear power plants, including enhanced safety performance, better economic affordability, and reduced financial risk. The components (modules) of SMRs are to be of a standard design, shop fabricated in assembly lines subjected to high quality control, and then transported as complete modules to the power plant site for rapid installation. The SMRs also offer flexible power generation for a wider range of users and applications—central power stations can consist of multiple units installed at a rate that best matches the growth in power demand; alternatively, a single SMR unit can be installed in the vicinity of the power customer using a so-called distributed (localized) power grid. It is even being proposed that the hundreds of shuttered coal power plant sites could be repurposed for SMRs. This encyclopedia describes (Stanculescu, 2021) a sample of the SMRs under development and explains the incentives for their development and the benefits they offer. The Encyclopedia also describes the motivation for, and examples of, so called “microreactors”—reactors producing an electrical power of less than 10 MW. Such reactors are primarily intended for deployment in remote areas which are not connected to the grid and to which the transport of fossil fuel is very expensive, as well as to provide reliable power to sensitive installations even in case of failure of the central power grid.
- The very ambitious space explorations being planned by leading industrial countries cannot succeed without nuclear energy. Space Fission Power (SFP) systems offer the potential to enable truly ambitious exploration of outer space, and ultimately a permanent human presence in space. Most of the benefits of SFP is tied to the very high amount of energy that can be extracted per unit weight of fissile material, which can allow high powers and/or long lifetime at a relatively low mass. SFP systems also offer great flexibility in terms of being able to operate in diverse environments, including environments with no or limited solar radiation, and the ability to provide abundant electrical power and heat from kilowatts to megawatts. Applications of SFP include propulsion and/or power for spaceships, electricity and heat for exploration rovers for landing on Mars (other planets and moons) or for human colonies on the Moon or Mars. This encyclopedia describes (Poston, 2021; Bennett, 2021) the many benefits nuclear energy offers to space exploration and the many challenges that need to be overcome.
- Controlled thermonuclear fusion is the focus of major international research programs that aim to demonstrate the scientific and technological feasibility of large-scale generation of nuclear energy by the fusion of light atomic nuclei. The widely distributed fusion fuel resources together with the favorable anticipated safety and environmental characteristics of future fusion power plants provide a significant motivation for the continued development of the challenging science and technology towards the exploitation of fusion energy for the benefit of mankind. In addition to the description of the fundamental elements of the fusion process and of the principal approaches to the controlled and sustained generation of fusion energy being pursued, this Encyclopedia presents the state-of-the-art for fusion research and development and the progress towards commercial fusion energy generation (Campbell, 2021). The advantages and challenges of nuclear fusion as an energy source are summarized.
- This encyclopedia also describes the many ways nuclear radiation can be harnessed to provide enormous benefits in the fields of medicine, agriculture, modern industry, environmental protection, and public safety (Waltar, 2021). Described in the Encyclopedia are more than a dozen unique properties of radiation technology that allow for these benefits to be achieved. Even though enormous and widespread, the benefits to society from these technologies are largely unknown and unappreciated. The global markets for these technologies exceed USA \$500 billion annually and are growing rapidly.
- Also provided in this Encyclopedia is information on dozens of research reactors (O’Kelly, 2021) that are operating or under construction around the world and provide unique research capability for scientists pursuing advanced research in physics, chemistry, biology, material and other sciences. Another invaluable utilization of research reactors is the production of special radioactive isotopes for numerous applications in medical, industrial and agricultural applications, and scientific research. Some research reactors are also used for demonstrating the performance of evolutionary as well as advanced nuclear fuels and of advanced nuclear power reactor components.
- Public perception of nuclear energy varies from country to country. It tends to be most negative in wealthy democracies like Germany, Australia, and Japan. Contributing to the negative perception is a fear of reactor accidents, a fear of nuclear radiation, a fear of long-lived radioactive waste, and a fear of the nuclear technology misuse for weapons proliferation. The encyclopedia addresses these concerns and provides authoritative objective information that, hopefully, will contribute to dispelling the public misconceptions (Kadak, 2021).
- The Encyclopedia describes measures implemented to ensure the health and safety of the public from nuclear power (Rempe, 2021). These measures evolved as new lessons were learned from operating experience and unplanned events, in particular events with core damage and radionuclide release (including in Three Mile Island, Chernobyl, and Fukushima). Clearly, the

experience gained from the design, evaluation, operation, decommissioning, and regulation of nuclear power plants has led to significant safety improvements. Future advanced reactor designs are expected to incorporate innovative inherent safety features (Stanculescu, 2021).

- The encyclopedia provides a detailed description of the biological effects of nuclear radiation, of how we reliably measure radiation levels and ensure that they do not exceed the permissible levels (Miller, 2021). The Encyclopedia describes the theory and practice of radiation protection that provide the basis for regulatory requirements that are essential for the safe and beneficial use of the many advanced technologies that involve exposure to nuclear radiation. The Encyclopedia explains that everyone is exposed to natural background radiation on a daily basis; that radiation is routinely in use for diagnostics of health conditions, such as for dental and medical X-rays, and that this is the largest source of radiation exposure for almost all human population in developed countries and that the benefits from these practices far exceed their risk. The regulatory permissible level of nuclear radiation is harmless much like a moderate exposure to sunshine is harmless (and may even be beneficial, whereas over-exposure can cause sunburns). In other words, the regulatory limits of doses induced by exposure to nuclear radiation are already substantively protective, and there is no evidence of health improvements by reducing the doses below these limits. Insisting to reduce radiation exposure to as low as possible below the regulatory limits may result in harm.
- Also provided in this Encyclopedia is detailed information about the nuclear waste and how it can be disposed of safely (Poinssot, 2021; Kadak, 2021). Using advanced fuel cycles (Stanculescu, 2021; Poinssot, 2021) the amount and activity of the long-lived nuclear waste can be reduced to few percent of the amount that needs to be disposed of per unit of energy generated using the contemporary “once-through” fuel cycle.

On the importance of nuclear energy in avoiding climate change catastrophe

The world’s population is expected to grow from current ~6.7 billion people to over 9 billion by 2050. As recognized in the United Nations Resolution 70/1 “Transforming our world: the 2030 Agenda for Sustainable Development” (Stanculescu, 2021; United Nations, 2015), providing “universal access to affordable, reliable and sustainable energy” and in particular to reliable electrical energy, is central to human development, social stability and even world peace. Currently, about 2.8 billion people are still lacking modern energy services (International Energy Agency, 2017), and more than one billion do not have access to electricity. Not only the growth of worldwide population, but also the economic growth in developing countries will lead to a continuing increase in the demand for energy.

The tremendous increase in energy generation required for the benefit of the global population has to be accomplished while reducing the rate of GHG (primarily CO₂) emission. Already, “human activities are estimated to have caused approximately 1.0 °C of global warming above pre-industrial levels” according to the Intergovernmental Panel on Climate Change (IPCC, 2018). In order to limit the global temperature increase to around 1.5 °C, total CO₂ emissions need to be reduced to net zero by around 2050.

Germany, for example, had decided to achieve this goal by installing renewable energy generators made of solar photovoltaic panels and wind turbines. Germany also decided to close down its nuclear power plants. As a result, according to Goldstein et al. (2019), “Germany, which went all-in for renewables, has seen little reduction in carbon emissions, and, according to our calculations, at Germany’s rate of adding clean energy relative to gross domestic product, it would take the world more than a century to decarbonize, even if the country wasn’t also retiring nuclear plants early. A few lucky countries with abundant hydroelectricity, like Norway and New Zealand, have decarbonized their electric grids, but their success cannot be scaled up elsewhere; the world’s best hydro sites are already dammed.”

Goldstein et al. (2019) are also saying “we actually have proven models for rapid decarbonization with economic and energy growth: France and Sweden. They decarbonized their grids decades ago and now emit less than a tenth of the world average of carbon dioxide per kilowatt-hour. They remain among the world’s most pleasant places to live and enjoy much cheaper electricity than Germany to boot. They did this with nuclear power. And they did it fast, taking advantage of nuclear power’s intense concentration of energy per unit weight of fuel. France replaced almost all of its fossil-fueled electricity with nuclear power nationwide in just 15 years; Sweden, in about 20 years. In fact, most of the fastest additions of clean electricity historically are countries rolling out nuclear power.”

In the words of World Nuclear Association Director General Agneta Rising in 2019 (Bisconti, 2021): “The 445 nuclear reactors in 30 countries are the low-carbon backbone of electricity systems, operating in the background, day in day out, often out of sight and out of mind, capable of generating an immense amount of clean power. They are the silent giants upon which we rely today.” Table 2 lists the fraction of electricity generated during 2018 in selected countries and globally from energy sources that do not emit greenhouse gases (Pioro et al., 2021). According to Poinssot and Bourg (2021), nuclear energy is within the top three of the most environmental-friendly energy sources; in the same range as hydropower and wind-power. Measured by CO₂ emissions, nuclear energy is even the most environmentally friendly energy source.

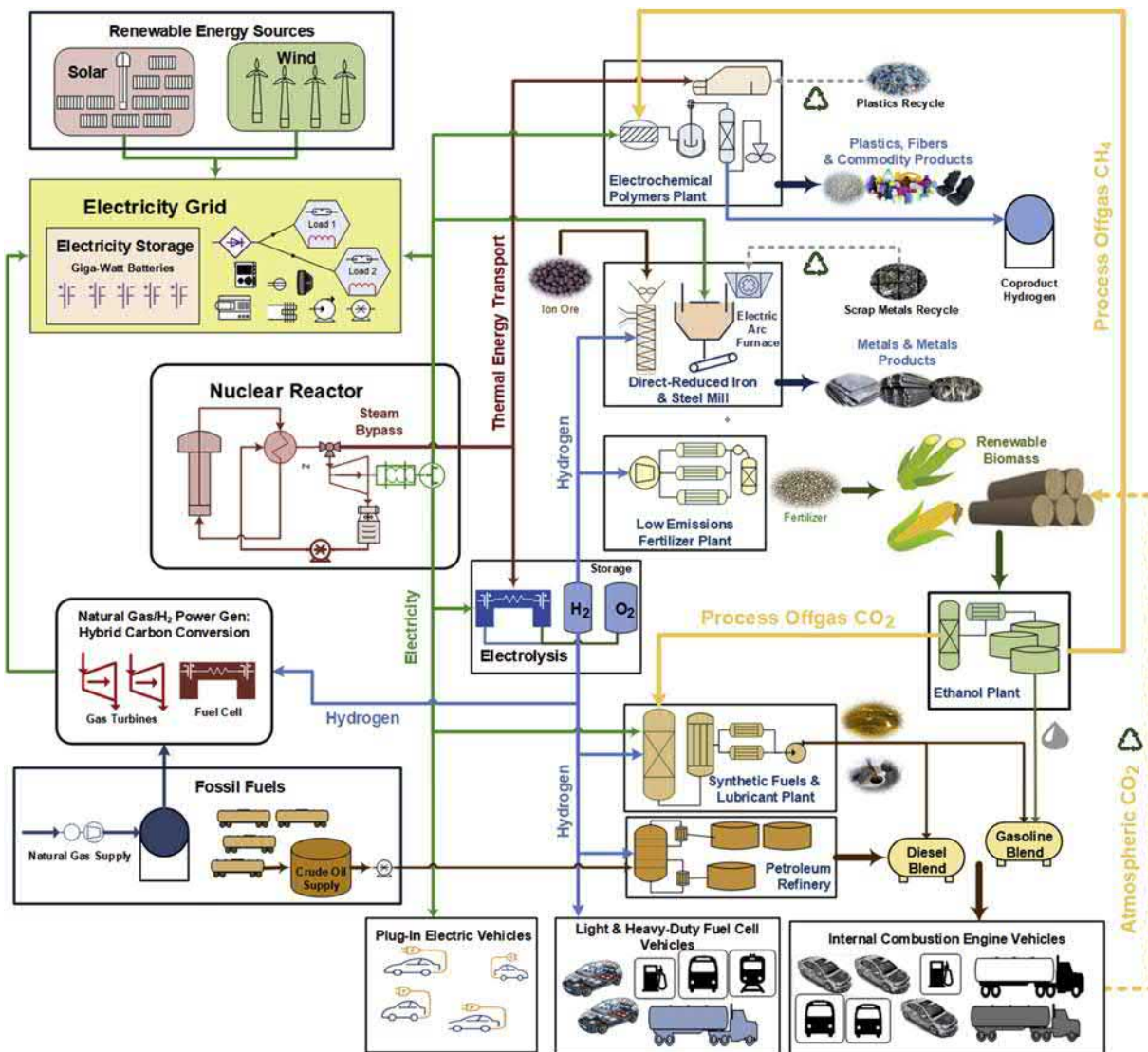
It is being suggested (Conca, 2021) that recent research has revealed the potential for extracting practically unlimited supplies of uranium from the oceans at a cost that may come close to that of land mining. If successful, this may make fission nuclear energy practically as renewable as wind and solar (Conca, 2021).

It is not sufficient to decarbonize the electricity generation sector. In the U.S., for example, 33% of CO₂ emissions are attributed to energy generated to support the electricity sector but an additional 36% of emissions result from transportation and 19% result

Table 2 Fraction of electricity generated by clean energy technologies in selected countries.

Country	Nuclear	Renewable	HYDRO
France	72	3	11
Ukraine	53.5	0.7	4.2
Sweden	41	10.2	39
UK	25	25	-
Korea	23.4	0	1.3
Russia	19.0	0	17
USA	19.7	10.9	6.6
Germany	11.6	22.2	-
World	10.2	6.6	16.3

from industrial processes (Bragg-Sitton and Boardman, 2021). There will be no global environmental benefits from switching transportation to use all electric vehicles if the electricity used for charging the batteries is generated in fossil fuel power plants (without sequestration of the CO₂ these plants emit). The realization of the above-mentioned facts is recently leading to a paradigm shift in the approach to the use of nuclear energy that is likely to bring about a restructuring in the entire energy market.

**Fig. 1** Illustrative examples of the near-term integration of existing LWRs with various industry applications (Bragg-Sitton and Boardman, 2021).

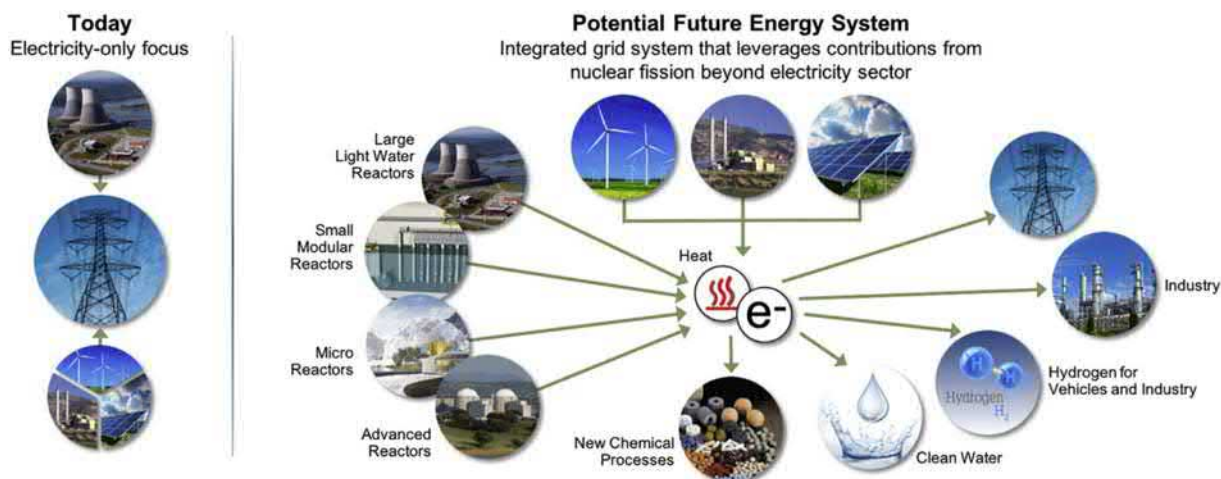


Fig. 2 Conceptualized coordination of energy generation sources and demand to maximize flexibility and economic performance while ensuring grid reliability and resilience.

Most energy experts around the world are advocating the development of a portfolio of low-carbon multi-energy technologies including, in addition to hydro-power plants (with very limited potential for expansion in capacity) and the currently favored “renewable” electricity generators, nuclear power plants. Because the power output of the renewable generators greatly fluctuates over the day and over the year, these generators need to be supplemented by effective energy storage devices that significantly add to the cost of energy.

Nuclear power plants, on the other hand, can generate power at a constant rate throughout the day and the year (excluding a shutdown period of ~1 month for refueling and maintenance once every 18–24 months), practically independent of weather conditions, while requiring a smaller land area per unit energy generated relative to the area required by renewable energy generators.

Moreover, nuclear energy can be generated in the form of high temperature heat (Stanculescu, 2021; Bragg-Sitton and Boardman, 2021) that is the most economical form of energy to store and to use as a backup for the intermittent nature of the renewable energy generators. Therefore, a suitable mix of nuclear and renewables is likely to be the optimal strategy for carbon-free energy generation.

Fig. 1 (Bragg-Sitton and Boardman, 2021) is a simplified illustration of the evolving perception of integrated energy systems using contemporary nuclear power reactors (primarily Light Water Reactors, LWR) in combination with solar and wind renewable energy generators to provide electricity and heat energy for transportation and many industries including non-grid energy users.

Fig. 2 (Bragg-Sitton et al., 2021) is a simplified conceptual perception of the proposed transition away from the use of contemporary large central nuclear power plants from their traditional electricity-only production mode to their exploitation for the provision of a variety of outputs. Future energy systems that seek to maximize energy utilization, generator profitability, and grid reliability and resilience must necessarily consider all available carbon-free energy sources, using each available resource in a way that maximizes its benefits while minimizing less-desirable attributes. The future energy systems should include advanced nuclear reactors including Gen-IV reactors, SMRs and even microreactors and should use distributed in addition to centralized electricity grids.

In summary, the need to meet the objective of zero net CO₂ emission by 2050 combined with the need to greatly increase the total global energy generation rate is an enormous challenge that can be achieved only by making use of all available clean energy generation technologies including nuclear in addition to solar and wind electricity generators. At the 2021 Earth Day climate summit, U.S. president Biden called (WSJ, 2021) for slashing U.S. greenhouse gas emissions by 2030 by 50%–52% compared with the baseline year of 2005, as he seeks to put America at the forefront of world-wide efforts to combat climate change. According to WSJ (2021), meeting this objective would require dramatically reshaping key sectors of the economy, from energy to transportation to agriculture.

Concluding remarks

This Encyclopedia of Nuclear Energy provides a very comprehensive, up-to-date and authoritative information related to nuclear energy, including nuclear radiation—from the discovery of the structure of the atom and its nucleus; through the discovery of three different ways to extract energy from nuclei (fission of heavy nuclei, fusion of light nuclei and radioactive decay); through the state-of-art of the nuclear industry for energy generation and for exploitation of several nuclear technologies for medical, industrial, agricultural and other applications; to the research and development of innovative nuclear technologies that are expected to further

increase mankind benefits from nuclear energy. The reader can find in the Encyclopedia a wide range of topics ranging from the contribution of nuclear energy to the evolution of the continents and oceans of planet Earth to the enabling of very ambitious space explorations including the provision of electricity and heat to human colonies on the Moon, Mars and beyond. The Encyclopedia explains why it is essential to greatly increase the nuclear energy generation rate in order to prevent a global warming catastrophe and, at the same time, provide sufficient energy to the growing world population and to the billions of people in underdeveloped countries.

It is hoped that this Encyclopedia will make an important contribution to transferring of the huge amount of technical knowledge accumulated in the field to the young generation, will inspire students to join research and development of innovative nuclear technologies and will contribute to better understanding of the nuclear technologies, their merits and safety by the general public and policy makers. The appreciation of the enormous benefits in the fields of steady reliable clean energy generation, medicine, agriculture, modern industry, environmental protection, and public safety that mankind enjoys every day and which is described in this Encyclopedia will, hopefully, contribute to much greater public acceptance and support of enhanced utilization of nuclear energy in the mix of clean energy technologies.

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CONTRIBUTORS TO VOLUME 2

Sherry Adadi

Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

Nick A Altic

ORAU, Oak Ridge, TN, United States

A Iulian Apostoaiei

Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States; and Oak Ridge Center for Risk Analysis, Inc. (ORRISK, Inc.), Oak Ridge, TN, United States

John Arnish

Knoxville, TN, United States

Florence Bart

Department of Dismantling and Waste Conditioning, Energy Division, CEA French Atomic Commission, Marcoule Center, Bagnols-Sur-Cèze, France

Lewis R Blackburn

NucleUS Immobilisation Science Laboratory, Department of Materials Science and Engineering, University of Sheffield, Sheffield, United Kingdom

V Blet

CEA, Marcoule, France

Dennis C Bley

Buttonwood Consulting, Inc., Albuquerque, NM, United States

Bernard Boullis

CEA, Saclay, France

S Bourg

CEA, DES, ISEC, DMRC, University of Montpellier, Montpellier, France

Stéphane Bourg

Commissariat à l'énergie atomique et aux énergies alternatives, Institut des sciences et technologies pour une économie circulaire des énergies bas carbone, CEA Marcoule, Bagnols sur Cèze, France

Douglas G Bowen

Oak Ridge National Laboratory, Oak Ridge, TN, United States

Jordi Bruno

Amphos 21 Group, Barcelona, Spain

Thomas Byrne

Radiation Oncology Department, Cumberland Medical Center, Crossville, TN, United States

Xiangzhou Cai

Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai, China

Paul Cantownine

Core and Fuel Engineering, Global Nuclear Fuel – Americas, Wilmington, NC, United States

Guoping Cao

Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

Albert Casagrande

Computational Mechanics and Materials, Idaho National Laboratory, Idaho Falls, ID, United States

Christine Chabert

CEA-Cadarache, Saint-Paul-lez-Durance, France

James I Cole

Idaho National Laboratory, Idaho Falls, ID, United States

James L Conca

UFA Ventures, Inc., Richland, WA, United States

Douglas C Crawford

Idaho National Laboratory, Idaho Falls, ID, United States

Shaheen Azim Dewji

Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

Donald A Dube

Risk Consultant, Unionville, CT, United States

Keith F Eckerman
Oak Ridge National Laboratory, Oak Ridge, TN, United States

Rodney C Ewing
Geological Sciences, Center for International Security and Cooperation, Stanford University, Stanford, CA, United States

Karl N Fleming
KNF Consulting Services LLC, Spokane, WA, United States

Guy L Fredrickson
Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

Ruchi Gakhar
Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

Laura J Gardner
NucleUS Immobilisation Science Laboratory, Department of Materials Science and Engineering, University of Sheffield, Sheffield, United Kingdom

RO Gauntt
Gauntt Technical Safety Associates, LLC, Edgewood, NM, United States

Tyler J Gerczak
Particle Fuel Forms Group, Oak Ridge National Laboratory, Oak Ridge, TN, United States

Stéphane Gin
Department of Dismantling and Waste Conditioning, Energy Division, CEA French Atomic Commission, Marcoule Center, Bagnols-Sur-Cèze, France; and Long Term Behavior Laboratory, Bagnols-sur-Cèze Cedex, France

J-P Glatz
Retired from European Commission, Joint Research Centre, Karlsruhe, Germany

Bernd Grambow
Subatech, IMT-Atlantique, CNRS, University of Nantes, Nantes, France

Alexandra C Hackett
Oak Ridge National Laboratory, Oak Ridge, TN, United States

Nils H Haneklaus
Danube University Krems, Krems an der Donau, Austria; and Freiberg University of Mining and Technology, Freiberg, Germany

TW Hansen, Jr
Ameriphsics LLC, Knoxville, TN, United States

Lawrence Heilbronn
Nuclear Engineering Department, University of Tennessee, Knoxville, TN, United States

Grant W Helmreich
Oak Ridge National Laboratory, Oak Ridge, TN, United States

Patrick Hogan
Idaho National Laboratory, Idaho Falls, ID, United States

Ian Hore-Lacy
World Nuclear Association, Blackburn North, VIC, Australia

Michael Howard
Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

John D Hunn
Oak Ridge National Laboratory, Oak Ridge, TN, United States

Neil C Hyatt
NucleUS Immobilisation Science Laboratory, Department of Materials Science and Engineering, University of Sheffield, Sheffield, United Kingdom

Wade P Ivey
Oak Ridge National Laboratory, Oak Ridge, TN, United States

Colby B Jensen
Idaho National Laboratory, Idaho Falls, ID, United States

Warren F Jones
Idaho National Laboratory, Idaho Falls, ID, United States

Jan-Fong Jue
Idaho National Laboratory, Idaho Falls, ID, United States

Toni Karlsson
Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

Masato Kato
Fuel Cycle Design Department, Japan Atomic Energy Agency, Ibaraki, Japan

Dennis D Keiser, Jr
Idaho National Laboratory, Idaho Falls, ID, United States

John H Kessler
J Kessler and Associates, LLC, Charlotte, NC, United States

Inn Seock Kim
ISSA Technology, Inc., Germantown, MD, United States

David A King
ORAU, Oak Ridge, TN, United States

David C Kocher
Oak Ridge Center for Risk Analysis, Inc., Oak Ridge, TN, United States

Robert Lutz
Lutz Nuclear Consulting, Hendersonville, NC, United States

Marty Malsch
Egan, Fitzpatrick Malsch and Lawrence PLLC, Washington, DC, United States; and Egan, Fitzpatrick, Malsch & Lawrence PLLC, Washington, DC, United States

Craig M Marianno
Texas A&M University, College Station, TX, United States

Jean-Michel Marin
Recycling Business Unit, Orano, Paris, France

Christopher Matthews
Materials Science and Technology, Los Alamos National Laboratory, Los Alamos, NM, United States

Rod McCullum
Organization Nuclear Energy Institute, Washington, DC, United States

Soufiane Mekki
ANDRA—French National Radioactive Waste Management Agency, Administrator of IAEA and OECD/NEA relations with Andra, International Relations Department, Chatenay-Malabry cedex, France

Daniel S Metlay
Institute for International Science and Technology Policy, George Washington University, Washington, DC, United States; and B. John Garrick Institute of Risk Sciences, University of California, Los Angeles, CA, United States

Mitchell K Meyer
Idaho National Laboratory, Idaho Falls, ID, United States

Manuel Miguiriditchian
CEA, DES, ISEC, DMRC, University of Montpellier, Marcoule, France

Laurence Miller
Department of Nuclear Engineering, The University of Tennessee, Knoxville, TN, United States; and

Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

Shinya Mizokami
TEPCO Holdings Company, Tokyo, Japan

Mohammad Modarres
University of Maryland, College Park, MD, United States

Glenn A Moore
Walsh Engineering Services, Idaho Falls, ID, United States

Trent L Nichols
Department of Nuclear Engineering, Nuclear Engineering Building, The University of Tennessee, Knoxville, TN, United States

Stephen Novascone
Computational Mechanics and Materials, Idaho National Laboratory, Idaho Falls, ID, United States

Hakan Ozaltun
Idaho National Laboratory, Idaho Falls, ID, United States

Sulgiye Park
Geological Sciences, Center for International Security and Cooperation, Stanford University, Stanford, CA, United States

Giovanni Pastore
Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States; and Computational Mechanics and Materials Department, Idaho National Laboratory, Idaho Falls, ID, United States

Sylvain Peugot
Department of Dismantling and Waste Conditioning, Energy Division, CEA French Atomic Commission, Marcoule Center, Bagnols-sur-Cèze, France; and Highly Radioactive Material Laboratory, Bagnols-sur-Cèze Cedex, France

Ronald Pevey
Nuclear Engineering Department, University of Tennessee, Knoxville, TN, United States

William C Phillips
Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

Ch Poinssot
BRGM, French Geological Survey, Orléans, France

Christophe Poinssot
French Geological Survey (BRGM), Orléans, France

Douglas L Porter

Idaho National Laboratory, Idaho Falls, ID, United States

Barry Rabin

Idaho National Laboratory, Idaho Falls, ID, United States

Bob Rand

Global Nuclear Fuel – Americas, Wilmington, NC, United States

Harold B Ray

Public Utility Executive Vice President, Chief Nuclear Officer (Retired), San Marino, CA, United States

Joy L Rempe

Rempe and Associates, LLC, Idaho Falls, ID, United States

Céline Roussel

ORANO group, Montigny-le-Bretonneux, France

Gérald Senentz

Orano, Châtillon, France

Alexander R Sich

Department of Physics, Kennesaw State University, Marietta, GA, United States

Vikram Singh

Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

Gordon R Skillman

Skillman Technical Resources, Inc., Hershey, PA, United States

Lisa M Snapp

Y-12 National Security Complex, Oak Ridge, TN, United States

Robin Taylor

National Nuclear Laboratory, Central Laboratory, Sellafield, Seascale, United Kingdom; and National Nuclear Laboratory, Central Laboratory, Sellafield, United Kingdom

Aysenur Toptan

Computational Mechanics and Materials, Idaho National Laboratory, Idaho Falls, ID, United States

Ken G Veinot

Y-12 National Security Complex, Oak Ridge, TN, United States

Timothy J Vitkus

Oak Ridge Associated Universities (ORAU), Oak Ridge, TN, United States; and ORAU, Oak Ridge, TN, United States

Samuel A Walling

NucleUS Immobilisation Science Laboratory, Department of Materials Science and Engineering, University of Sheffield, Sheffield, United Kingdom

Zhiguang Wang

Institute of Modern Physics, Chinese Academy of Sciences, Lanzhou, China

Alexander M Wheeler

Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

Laurence G Williams

Centre for Nuclear Engineering, Imperial College London, London, United Kingdom; and Centre for Nuclear Engineering, Department of Materials, Imperial College London, London, United Kingdom

Bill Williamson

TVA, Nuclear Plant Road, Athens, AL, United States

Michael E Woods

Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

Nicolas E Woolstenhulme

Idaho National Laboratory, Idaho Falls, ID, United States

Tae-Sic Yoo

Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

Adam Zabriskie

Computational Mechanics and Materials, Idaho National Laboratory, Idaho Falls, ID, United States

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Safety, Regulation, and Decommissioning of Commercial Nuclear Power Reactors—Introduction

Joy L. Rempe, Rempe and Associates, LLC, Idaho Falls, ID, United States

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Glossary

Criticality The normal operating condition of a reactor, in which nuclear fuel sustains a fission chain reaction. A reactor achieves criticality (and is said to be critical) when each fission event releases a sufficient number of neutrons to sustain an ongoing series of reactions.

Defense-in-depth (DiD) A design and regulatory principle that relies on several consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. When properly implemented, DiD ensures no single technical, human or organizational failure could lead to harmful effects and the combinations of failures that could give rise to significant harmful effects are of very low probability.

Design Basis Accident (DBA) One category of postulated accidents used to set criteria and limits for the design and sizing of safety-related structures, systems, and components (SSCs).

Design Basis Event (DBE) An event considered in the range of conditions, including normal operation, DBAs, and natural phenomena, for which the plant must be designed to ensure acceptable performance of SSCs.

Engineered Safety Feature (ESF) A SSC relied upon during or following a DBE to prevent or mitigate the associated consequences from such an event.

Nuclear Facility Regulation The inspection, assessment, permissioning and enforcement activities, which are applied to siting, design, construction, commissioning, operation and decommissioning of nuclear installations, to provide reasonable assurance of adequate protection of public health and safety.

Reactivity A measure expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Regulatory Framework for Nuclear Facilities Relevant regulations, criteria, guidance, standards, or other documents used to regulate nuclear facilities.

Safety Evaluation The study, examination and description of a nuclear installation's expected behavior throughout its life, under normal and transient conditions and under postulated events, in order to determine: (a) safety margins provided in normal operation and in transient regimes; and (b) the adequacy of measures to prevent and mitigate the consequences of accidents that may occur.

Introduction

Nuclear power has unique attributes compared to other energy-producing technologies. These include a relatively high energy density, decay heat production after shutting down the fission chain reaction, and the potential to release radionuclides. These attributes have led to special measures being instituted to ensure the health and safety of the public. These measures evolved as new lessons were learned from operating experience and unplanned events, in particular events with core damage and radionuclide release. Clearly, the experience gained from the design, evaluation, operation, decommissioning, and regulation of nuclear power plants has led to significant safety improvements.

Chapters in this section, “Safety, Regulation, and Decommissioning of Commercial Nuclear Power Reactors”, focus on the safety, regulation, and decommissioning of commercial nuclear power plants. These chapters emphasize five major topic areas: **Unplanned Events affecting Reactor Safety**; **Ensuring Safety by Design and Evaluation**; **Regulation and Regulatory Framework**; **Probabilistic Risk Assessment**; and **Decommissioning**. Experts with unique knowledge about these topic areas were invited to author these chapters. Authors were encouraged not only to provide factual information but also to present their perspective regarding the significance of past events, on-going activities, and actions to be performed in the future. Although some readers may wish to solely focus on a single topic and chapter, others may wish to explore all chapters pertaining to a particular topic area. To assist the latter readers, this chapter has been included to introduce these topic areas and provide a roadmap of the relationship between chapters within a topic area and other topic areas included in this section. Suggestions to assist the reader in gaining important insights about each topic area are also provided.

Unplanned events affecting reactor safety

As with other facilities that introduce new technologies, nuclear power plants were designed and licensed to minimize the likelihood of accidents that could adversely affect public health and safety. Nevertheless, unplanned events have occurred at commercial nuclear power plants, some with core damage and radionuclide release. Such events provided important lessons and ultimately, led to improvements that enhance the safety of the operating fleet and new designs.

This section includes four chapters describing unplanned events that led to significant safety measures being implemented. [Rempe \(2021a\)](#) describes two highly scrutinized events: the 1975 Browns Ferry fire and the 1983 Salem Anticipated Transient Without Scram (ATWS) events. Although neither event resulted in core damage or radionuclide release, each resulted in important changes to reduce the potential for their reoccurrence at these and other nuclear reactors. The remaining three chapters review severe accidents that resulted in core damage at: Three Mile Island Unit 2 ([Skillman and Rempe, 2021](#)), Chernobyl Nuclear Power Plant Unit 4 ([Sich, 2021](#)), and Fukushima Daiichi Units 1, 2, and 3 ([Mizokami and Rempe, 2021](#)). Substantial efforts were, or in some cases continue to be, completed by the regulator and industry to understand the progression, root causes, and lessons associated with each event. These efforts have also resulted in important measures being required to prevent and mitigate such events and improve reactor safety.

As emphasized with these and other chapters within this section (see **Ensuring safety by design and evaluation** and **Regulations and regulatory framework**), unplanned events resulted in modifications to existing plant structures, systems, and components (SSCs) as well as changes to the methods used to evaluate SSC performance. Significant safety lessons were learned and continue to be discovered from events with and without core damage. Many of these lessons are independent of the reactor technology and country in which the event occurred. In identifying these lessons, it is important to understand the root causes of these events. There are similarities in the root causes, such as inadequate procedures for operation and maintenance, insufficient operator training, design deficiencies, regulatory deficiencies, and inadequate safety culture. Likewise, there are similarities in the types of lessons learned and the types of changes implemented to address these lessons. Although significant advances have been implemented to improve reactor safety, the industry must continue to remain vigilant in identifying safety lessons and taking appropriate corrective actions.

Ensuring safety by design and evaluation

Nuclear power plant safety is assured by maintaining three key safety functions: control reactivity, control cooling, and control radioactivity release. These key safety functions, often referred to as “Critical Safety Functions” are ensured by various measures, such as inherent and passive safety features, multiple independent barriers, engineered safety systems, instrumentation for plant monitoring, and procedures and guidance for operators. A Defense-in-Depth (DiD) principle is used to ensure these safety functions are maintained for a range of challenges and operating conditions. This principle, which is enforced by a regulatory framework, is applied in reactor design and evaluation activities.

This section includes seven chapters emphasizing how critical safety functions are maintained by ensuring appropriate plant SSCs are selected. [Rempe \(2021b\)](#) provides an overview of this area, highlighting interactions between efforts to achieve critical safety functions, apply the DiD principle, and develop an appropriate regulatory framework. Chapters are also provided that describe representative Reactor Protection Systems ([Lutz and Williamson, 2021a](#)), Engineered Safety Features ([Lutz and Williamson, 2021b](#)), and Instrumentation and Control Systems ([Lutz and Williamson, 2021c](#)) found in U.S. Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). These reactor systems are selected considering the following objectives: maintain critical safety functions, address safety principles such as DiD, and adhere to regulatory requirements and guidance. [Lutz and Williamson \(2021d,e\)](#) call attention to the importance of guidance provided to operators for operational and beyond design basis events. This guidance is also essential in assuring that critical safety functions are maintained. [Gauntt \(2021\)](#) provides an overview of analyses required for evaluating the performance of plant systems, operator actions, and measures to prevent and mitigate accident release. These analyses are used to demonstrate compliance with regulatory requirements and in some cases, to modify the regulatory framework.

Several chapters emphasize prior events that occurred at operating nuclear power plants, such as the events described in **Unplanned events affecting reactor safety**, and landmark analysis efforts, such as major PRAs described in **Probabilistic risk assessment**. These events and analysis efforts influenced system designs, guidance, and regulation. Requirements for plant design and evaluations have evolved and continue to evolve as new insights are gained from operating experience.

Regulation and regulatory framework

Society demands governments regulate industrial applications that present potential harm to people or the environment. The amount of regulatory oversight should be commensurate with the magnitude and likelihood of the potential hazard. The application of nuclear energy for commercial power production is no exception; because of the high hazard potential, it is subject to a stringent nuclear regulatory framework. Although licensing and regulation of nuclear facilities is, in general, a national responsibility, severe accidents with core damage and radiation release reinforce the international aspects associated with commercial nuclear power applications.

This section includes five chapters focussing on reactor regulation. [Rempe and Williams \(2021\)](#) provides an overview of nuclear reactor regulation and regulatory frameworks, highlighting key concepts associated with effective regulation and the objectives, requirements, scope, roles, functions, and attributes of an effective regulatory body. Because of its lead in deployment and regulation of commercial nuclear power plants, four chapters emphasize the U.S. experience with commercial reactor regulation, which evolved over time. [Bley and Malsch \(2021a\)](#) provides a historical review of the evolution of the U.S. licensing process. Current requirements found within U.S. regulation that affect commercial power reactor operation are reviewed in [Bley and Malsch \(2021b\)](#). [Bley and Malsch \(2021a,b\)](#) highlight factors that led to significant changes in the current U.S. Nuclear Regulatory Commission (NRC) regulatory framework. [Ray \(2021\)](#) describes the two licensing pathways currently provided by the NRC to allow operation of a commercial nuclear power plant, identifying the advantages and disadvantages of each pathway. In recent years, there has been renewed interest in advanced reactor systems with different fuels, materials, and coolants than found in current generation Light Water Reactor (LWR) systems. [Rempe \(2021c\)](#) provides insights gained from prior experience with two first-of-a-kind non-LWRs (a gas-cooled reactor and a sodium-cooled fast reactor) authorized for startup by the predecessor of the NRC, the U.S. Atomic Energy Commission. Insights emphasize the importance of factors, such as prior operating experience from similar plants, an appropriate regulatory framework with review and oversight by an independent regulator, initial testing and operation, and complexity and technology advances. As described by [Bley and Malsch \(2021a\)](#) and [Fleming \(2021\)](#), recent actions are being taken by the NRC to establish an optional new regulatory framework to address issues identified in these two and other non-LWR licensing activities. As this new process is explored, it is important to consider difficulties encountered during prior non-LWR licensing, startup, operating, and decommissioning activities. In recognition that operation of commercial nuclear power plants has the potential to transcend borders, international guidance has been developed to outline expectations for assuring public health and safety. [Williams \(2021\)](#) provides an overview of this international guidance and compares regulatory frameworks found in several countries with significant nuclear power programs. Despite differences in legal systems and cultures, these descriptions indicate each country provides nuclear safety regulation and licensing consistent with international expectations.

Together, these chapters emphasize desirable attributes of a regulator and a regulatory framework, the principles of effective regulation, and consequences when regulatory frameworks possessing the desired attributes are not in place. Within this section, several chapters (see **Unplanned events affecting reactor safety**) identify factors, such as the lack of an independent regulator and the need for an adequate regulatory framework, as root causes of unplanned events.

Probabilistic risk assessment

Probabilistic Risk Assessment (PRA) provides a systematic method for answering three questions:

- What can go wrong?
- How likely it is?
- What its consequences might be?

Risk assessments results identify expected outcomes of accident sequences, sensitivities in analysis results, areas of importance, plant system interactions, and areas of uncertainty in plant response. PRAs have been performed to gain important insights about operating reactors as well as new reactor concepts in various phases of design, licensing, and construction. Appropriate delineation of accident sequences, as well as accurate quantification of their frequencies, consequences, and associated uncertainties, is essential for deriving these insights; and appropriate use of these insights separates PRA from deterministic approaches to reactor safety. Risk-informed decision-making, using PRA insights, has been increasingly applied to assess plant safety, enhance plant design, improve operations, optimize maintenance, and optimize resources. In addition, these insights have been used to establish regulatory requirements that better focus licensee and regulator attention on design and operational issues commensurate with their importance to public health and safety.

This section includes three chapters on the topic of PRA. [Modarres and Kim \(2021\)](#) describes the basic concepts associated with risk assessment and presents the overall process used to perform a nuclear power plant PRA. [Dube \(2021\)](#) highlights several

examples in which the NRC has effectively used a “risk-informed approach” in regulatory decision-making and oversight. Fleming (2021) provides several examples of risk insights derived from early PRAs and describes how these insights have been and continue to be used to support risk-informed decision making for existing plants and advanced reactors, including a recent effort to select licensing basis events and design requirements.

Of special interest in these chapters are descriptions of risk metrics and importance measures as well as methods used to derive risk insights and support risk-informed decision-making. To avoid misleading conclusions, however, it is important to understand the limitations of PRAs and associated risk metrics and importance measures. Prior risk assessments have significantly influenced reactor design, operation, and regulation (see chapters in *Ensuring safety by design and evaluation* and *Regulation and regulatory framework*).

Decommissioning

Unlike other industrial facilities, a nuclear power plant requires an owner plan for the facility’s entire life cycle, including decommissioning, prior to receiving an operating license. In the U.S., the NRC requires plant owners establish and maintain a trust fund or another financial mechanism to ensure coverage of all costs for the ultimate decommissioning of the facility. Although some countries initially lacked the strong regulatory governance for decommissioning funding that exists in the U.S., it is now common worldwide to require that plant owners provide financial assurance for the plant’s entire life cycle.

This section includes three chapters related to the topic of decommissioning. McCullum (2021a) presents an overview of U.S. and international decommissioning strategies, regulatory requirements, status, and practices. Currently, there are three strategies through which decommissioning is accomplished—SAFSTOR, DECOM, and ENTOMB. These three strategies are described, and details are provided about activities conducted within each strategy. U.S. nuclear plants that have completed, are undergoing, or are soon to enter the decommissioning process are identified. McCullum (2021b) and McCullum (2021c) provide examples of two U.S. plants that followed two of the more commonly used strategies, SAFESTOR and DECON. These examples illustrate how additional aspects associated with decommissioning, such as public involvement and public policy, influence the ultimate land use of the site.

Information within these chapters demonstrates that initial plans required and overseen by the regulator are being implemented to complete decommissioning successfully. It is also of interest to note that efforts are underway to complete decommissioning within shorter timeframes and at lower costs.

Summary

This section focuses on the safety, regulation, and decommissioning of commercial nuclear power plants. Chapters within this section emphasize the following topic areas: Unplanned Events affecting Reactor Safety, Ensuring Safety by Design and Evaluation, Regulation and Regulatory Framework, PRA, and Decommissioning. As described within these chapters, the experience gained from the design, evaluation, operation, regulation, and decommissioning of nuclear power plants led to significant safety improvements. This introductory chapter provides the reader with a roadmap for this section; special insights are highlighted from each topic area.

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See Also: Commercial Nuclear Power Plant Statistics.

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Safety Lessons From Two Events Without Core Damage

Joy L. Rempe, Rempe and Associates, LLC, Idaho Falls, ID, United States

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Glossary

Advisory Committee on Reactor Safeguards (ACRS) Independent advisory committee to the U.S. Nuclear Regulatory Commission that is statutorily mandated by the Atomic Energy Act of 1954, as amended.

Anticipated Transient An event expected to occur one or more times during the life of a nuclear power plant.

Anticipated Transient Without SCRAM (ATWS) An anticipated operational occurrence involving a failure of the reactor trip portion of the reactor protection system. If not mitigated, the event could result in reactor fuel melting.

Backfitting Modification of or addition to systems, structures, components, or design of a facility, commonly understood to be required by regulatory direction.

Common Cause Failure (CCF) The simultaneous failure or unavailability of more than one similar or identical component due to an underlying common cause.

Reactor Protection System (RPS) System designed to monitor key plant variables to detect off-normal plant conditions arising from anticipated transients and to automatically initiate whatever safety action is needed.

Reactor Trip System (RTS) Critical portion of the RPS that includes the power sources, sensors, initiation circuits, logic matrices, bypasses, interlocks, racks, panels, control boards, actuation devices, and actuated devices, required to initiate reactor shutdown.

SCRAM or Trip A manual or automatic command to insert control rods with the intention of shutting down a nuclear reactor by immediate termination of the fission chain reaction.

Introduction

At the time when nuclear plants were deployed for commercial power production, there was strong awareness about potential hazards associated with the release of radioactive materials. As with other industries that introduce new technologies, nuclear power plants were designed and licensed to minimize the likelihood of accidents that could adversely affect public health and safety. As described in [Rempe and Williams \(2021\)](#), defense-in-depth measures, such as remote siting, containment buildings, and engineered safety features, were implemented to compensate for this lack of experience. Unplanned events led additional safety measures to be implemented.

In Section 4, “Safety, Regulation, and Decommissioning of Commercial Nuclear Power Reactors”, three chapters are devoted to descriptions of reactor accidents that resulted in core damage at nuclear power plants: Three Mile Island Unit 2 ([Skillman and Rempe, 2021](#)), Chernobyl Nuclear Power Plant Unit 4 ([Sich, 2021](#)), and Fukushima Daiichi Units 1, 2, and 3 ([Mizokami and Rempe, 2021](#)). Significant safety lessons were and continue to be gained from these events.

This chapter describes two highly scrutinized events that resulted in no core damage: the 1975 Browns Ferry fire based on information provided in ([TVA, 1975](#); [US NRC, 1976](#); [Haskin et al., 2002](#); [Okrent, 1981](#); [Scott, 1976](#); [Williamson, 2016](#)) and the 1983 Salem Anticipated Transient Without SCRAM (ATWS) events based on information provided in ([Haskin et al., 2002](#); [Okrent, 1981](#); [US NRC, 2008](#); [INPO, 1980](#)). Although there were no adverse effects to public health and safety, important measures were implemented to reduce the potential for reoccurrence of such events and to enhance the safety of operating commercial nuclear reactors.

1975 Browns Ferry fire

On March 22, 1975, a major fire burned for 7 h at the Tennessee Valley Authority Browns Ferry Nuclear Plant (BFNP) located near Decatur, Alabama. The fire and its aftermath revealed several significant inadequacies in design and procedures related to fires at that plant. Some equipment did not function correctly. Furthermore, some response actions were either incorrect or not as effective as they should have been. The fire caused extensive damage to electric power and control systems, impeded the function of normal and standby cooling systems, and degraded the capability to monitor the plant status. The reactors were shut down and cooled successfully. No one on site was seriously injured, and there was no radiological impact on the public. Post-event evaluations, however, identified many safety lessons. Important changes were implemented by the plant owner, the regulator, and the industry.

Background

Historical information

Fire had been recognized as a potential safety concern of importance for at least a decade before the 1975 Browns Ferry fire occurred ([Okrent, 1981](#)), but there were large differences in opinion concerning what measures were needed to make the probability of a serious accident from fires acceptably low. The significant fires that occurred at the San Onofre and Indian Point plants led to some requirement modifications. Evolutionary improvements were implemented into LWRs under construction, and an industry standard was developed. However, these measures were later deemed inadequate.

BFNP

The BFNP included three boiling water reactors (BWRs) designed by General Electric. Each unit was a BWR/4 design with an original licensed thermal power of 3293 MWt and housed in a Mark I containment (see [Fig. 1](#)). Unit 1 was issued its initial operating license in 1973, and Unit 2 was issued its initial operating license in 1974. At the time of the event, Units 1 and 2 were at 100% power and Unit 3 was under construction.

The BFNP is designed such that air flows toward areas of possible higher radiation. Hence, the BFNP reactor buildings and a shared refueling floor are maintained at a negative pressure to ensure that airflow is inward. As part of efforts to connect the Unit 3 reactor building, tests were required to ensure this negative pressure could be maintained. This testing indicated that Unit 1 reactor building air-in-leakage should be reduced, and an extensive program was underway for resealing.

Units 1 and 2 share a common control room with a cable spreading room located beneath the control room. Electrical cables between the control room and plant equipment pass through the cable spreading room. The cable-tray penetrations through the reactor building walls of each reactor were sealed to minimize in-leakage. This was accomplished by filling a penetration with a polyurethane foam to form a seal, and then applying a 3–6 mm layer of flameproofing compound over the foam and over the cables on both sides of the penetration (for 30 cm) to form a fire barrier. Additional cables had been pulled through some of the penetrations after they were originally installed. In order to make an opening for these cables, holes were punched through the sealant and flameproofing materials.

To check the effectiveness of penetration sealing, it was commonplace to hold a lit candle near the penetration opening. If the opening was not fully sealed, the lower pressure in the reactor building would cause air (and the candle flame) to be pulled through the opening, providing visual indication of leakage. Two days earlier, minor fires had occurred at BFNP due to the use of candles for leak testing. In these cases, however, workers were able to quickly extinguish the fire without any damage.

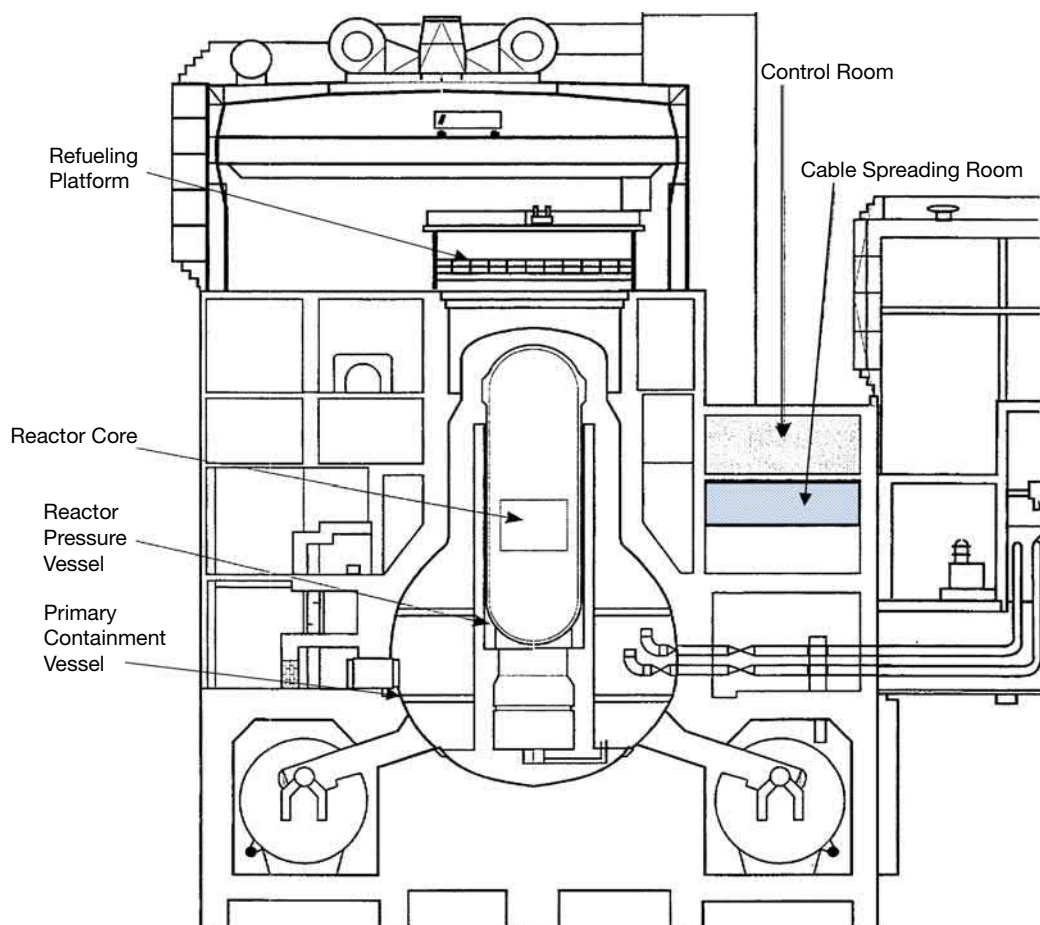


Fig. 1 Layout of BFNP Unit 1. Adapted from Haskin FE, Camp AL, Hodge SA, and Powers DA (2002). *Perspectives on Reactor Safety*. NUREG/CR-6042, Rev 2 March 2002. ML021080026. Graphic found on page 1.6-8 (page 102/333 of pdf).

Event sequence

Tables 1 and 2 summarize the sequence of events that occurred on March 22, 1975. During the early stages of the fire, operators were too busy to log events and actions. Hence, some of the listed times, which were established by reviewing available data and interviewing plant staff, differ in the cited references.

Table 1 Timing of events on March 22, 1975 associated with BFNP cable spreading room fire. Based upon information in (TVA, 1975; US NRC, 1976; Williamson et al., 2016; Haskin et al., 2002).

Time	Event
~12:20	Fire started in penetration.
~12:20–12:30	Portable CO ₂ and dry chemical fire extinguishers discharged into the hole.
~12:30	Two construction workers leave spreading room for Unit 1 reactor building to fight the fire.
~12:35	Plant fire alarm sounded. Fire logged by the Shift Engineer. ^a
~12:35–12:45	CO ₂ alarm sounded in spreading room; permanently installed spreading room CO ₂ (Cardox) system, which had been isolated because of leak repair activities, was discharged.
~12:37	Portable dry chemical fire extinguisher discharged in reactor building.
~12:40	Alarms sound in the control room; smoke from wiring under control panel observed.
~12:40–13:10	Spreading room CO ₂ system operated second time.
~12:40–13:00	Assistant Shift Engineer ^a assumes direction of fire brigade in fighting fire.
~15:00	Spreading room CO ₂ system operated third time.
~15:00	Shift Engineer assumes charge of spreading room firefighting.
~16:20	Spreading room fire extinguished (primarily using dry chemicals).

^aAt that time, a Shift Engineer was a senior reactor operator serving in the position that is today known as the Shift Manager. At that time, an Assistant Shift Engineer was a senior reactor operator serving in the position that is today known as the Unit Supervisor or Control Room Supervisor.

Table 2 Timing of events on March 22, 1975 associated with BFNP reactor building fire. Based upon information in (TVA, 1975; US NRC, 1976; Haskin et al., 2002; Williamson et al., 2016).

Time	Event
~12:30	Two construction workers leave spreading room for Unit 1 reactor building to fight the fire.
~12:37	Portable dry chemical fire extinguisher discharged in reactor building.
~12:40	Assistant Shift Engineer ^a reports exact location of fire in reactor building.
~12:40–13:00	Portable CO ₂ and dry chemical fire extinguishers discharged in the reactor building.
~13:00–13:10	Because smoke was so dense, air-breathing equipment was requested and received within 5 min.
~13:09	Assistant Shift Engineer ^a requested that Athens Fire Department come to the plant.
~13:30	Lighting lost in reactor building.
~16:45	Temporary d.c. lighting installed.
~18:35–19:00	Water applied to fire.
~19:30–19:45	Reactor building fire extinguished.

^aAt that time, a Shift Engineer was a senior reactor operator serving in the position that is today known as the Shift Manager. At that time, an Assistant Shift Engineer was a senior reactor operator serving in the position that is today known as the Unit Supervisor or Control Room Supervisor.

An electrical inspector and an electrician were sealing air leaks in the Unit 1 cable spreading room (see Fig. 1). While one particular penetration was being tested using this open flame technique, a hole (5 × 10 cm) allowed the candle flame to be drawn into the open void where it ignited the foam sealant (~12:20). The fire progression was not stopped by initial attempts, such as beating it with a flashlight, using rags to smother the fire, and discharging CO₂ and dry-chemical fire extinguishers.

The electrician called for someone to notify the reactor operations Shift Engineer that there was a fire in the cable spreading room. Meanwhile, airflow pulled the fire more deeply into the hole until it reached the reactor building side of the wall. At 12:34, the shift engineer was notified there was also a fire in the reactor building. Thus, firefighting activities had to consider two locations: the cable spreading room and the reactor building.

The fire department in Athens, Alabama, was notified and arrived at 13:45. At about 14:00, the Fire Chief recommended the use of water on the fire. However, the Plant Superintendent rejected this recommendation because of the potential that water could introduce short circuits, adversely impacting equipment to control, shutdown, and cool the reactor. Furthermore, the fire was progressing slowly (2–3 cm/min). The use of CO₂ and dry chemical fire extinguishing systems kept the cable room fire suppressed. On several occasions, the fire was reported to be extinguished, but the high energy content in the cables caused the fire to flare up. The fire in the cable room was finally extinguished at about 16:20.

The fire continued to burn in the reactor building. It was located about 6 m above the floor, making access difficult. Attempts to discharge dry-chemical and CO₂ extinguishers were limited due to workers experiencing difficulties in breathing and poor visibility. When the reactor building ventilation system was lost, excessive smoke reduced visibility and necessitated the use of air-breathing equipment. Eventually, lighting was lost in the reactor building. Air-breathing equipment for workers to fight the fire was limited because such equipment was also required by workers that were manually aligning valves for reactor shutdown cooling.

As noted above, water was not initially used because of reactor safety concerns. Between 18:35 and 19:00, the Plant Superintendent agreed to use water on the fire because the reactor had been shutdown for several hours with several options available for decay heat removal. Then, efforts to use water were initially delayed due to a nozzle from the Athens fire truck not fitting the threads on the hose to the second floor of the reactor building. By 19:45, the fire was declared out.

BFNP response

Although operator actions averted core damage, the plant sustained significant damage during the 7 h the fire burned. Approximately 1600 electrical cables that control Units 1 and 2 were damaged with approximately 630 of these being designated as safety related. The electrical cables, after insulation had been burned off, shorted together and grounded to their supporting trays or to the conduits, with the result that control power was lost for much of the installed equipment such as valves, pumps, and blowers. In Unit 1, all the Emergency Core Cooling System (ECCS) was rendered inoperable and operators lost the ability to monitor power from the control room. Portions of the Unit 2 systems were likewise affected.

Enough equipment remained operational throughout the event to shut down the reactors and maintain the reactor cores in a cooled and safe condition.

Unit 1

At approximately 12:42, the Unit 1 operator noted anomalous behavior of instrumentation and control systems designed to provide emergency cooling of the reactor core. Operators reacted to several events, such as the automatic starting of pumps and equipment associated with the low pressure coolant injection (LPCI), core spray (CS), high pressure coolant injection (HPCI), and reactor core isolation cooling (RCIC) systems. After it was determined these systems weren't needed, this equipment was stopped but automatically restarted. At 12:48, both recirculation pumps automatically reduced their flow outputs, dropping the reactor power level from 100% to 70%.

At 12:51, both recirculation pumps stopped, and the operators manually scrammed the reactor. However, the reactor continued to produce heat from the radioactive decay of fission products. At approximately 12:55, several electrical boards that supplied control voltages and power to many of the systems used to remove decay heat became unavailable. Many of the instruments and indicating lights were also lost due to damaged cables. In addition, smoke was observed coming from beneath the ECCS panel. Shortly after 13:00, the main steam isolation valves (MSIVs) closed automatically, presenting two additional challenges. First, the steam generated by the decay heat could not be transferred to the condenser, eliminating an important heat sink. Second, steam from the core was no longer available to drive the feedwater pumps, eliminating another method of providing high-pressure cooling water to the core. The fire eventually disabled several shutdown cooling systems, including the LPCI, CS, HPCI, and RCIC systems. Even though a control rod drive (CRD) system pump was supplying flow at around 400 L/min, decay heat induced boiling of water within the pressure vessel. Condensate booster pumps were operable, but these pumps can only inject water into the pressure vessel at pressures of 2.4 MPa or less. Given these conditions, the operator depressurized the reactor, which was at 7.4 MPa, to allow the use of available low-pressure systems.

The pressure-relief valves were manually opened from the control room at ~13:20, and the steam was transferred from the pressure vessel to the pressure-suppression pool (see Fig. 1). This method reduced the vessel pressure to about 1.8 MPa in 20 min. During the depressurization, the water level in the core decreased but did not drop below a point 1.2 m above the top of the fuel. Once the reactor pressure was reduced below 2.4 MPa, one condensate booster pump and one condensate pump provided makeup water to restore the water level to its normal height by 13:30. This mode of core cooling was adequate until about 18:00, when loss of control air prevented further manual control of the remaining (4 out of 11) operable pressure relief valves. The valves closed, and pressure in the RPV started to rise. As pressure increased above 2.4 MPa, the condensate booster pumps could no longer inject water into the vessel and thus only the CRD system pump was adding water.

After the fire was declared out at 19:45, the smoke began to clear. After about 3.5 h (~21:50), control of the relief valves was restored, the reactor was depressurized, and the condensate booster pump again pumped water into the reactor. With low-pressure operation now secured, adequate makeup water could be supplied by one of the condensate pumps. Alternate modes of cooling also became available.

Unit 2

The effect of the fire on Unit 2 was less pronounced. Nine minutes after Unit 1 was shut down, Unit 2 operators responded to several abnormal events, such as reactor power decreasing, many alarms sounding, and several indicating lights failing. The reactor was scrammed at 13:00.

The turbine was immediately tripped, along with the reactor feedwater pumps. In approximately 4 min after scram, the MSIVs closed, isolating the reactor steam from the condenser and the reactor feedwater pump steam supply. The RCIC and HPCI systems were immediately initiated to control reactor water level. These two systems tripped several times over the next hour. At approximately 13:45, the HPCI system became unavailable, but the RCIC system continued to supply high pressure water to the reactor.

When the MSIVs closed, the reactor pressure was relieved by manual operation of the relief valves. Manual operation of the relief valves was lost at 13:20. However, the relief valves lifted intermittently on pressure until 14:15. At that time, manual operation was restored, and the reactor was depressurized using the relief valves. At 22:10, the MSIVs were reopened, making the condenser heat sink available. At 22:20, the RHR system could be used for long-term cooling of the reactor.

Safety lessons

Post-event evaluations, such as the one conducted by the Special Review Group established by Executive Director of Operations of the U.S. Nuclear Regulatory Commission (NRC), (US NRC, 1976) emphasized several important safety lessons from the Browns Ferry fire. Reviews emphasized the importance of fire protection measures and common-cause failures that could adversely affect the ability of operators to safely shutdown the reactor and take actions to preclude core damage. In addition, evaluations emphasized the importance of proper maintenance procedures, quality assurance programs, and safety culture. The licensee, the regulator, and the industry have implemented important measures in response to these lessons.

Fire protection

Post-event reviews identified the need for better fire prevention and protection criteria, codes, and standards for nuclear power plants. Revisions were suggested to address topics, such as separation of redundant control circuits and power cables, the use of combustible materials within the plant, the placement of systems and equipment to detect and fight fires, the use of water to extinguish a fire, and adequate reporting procedures.

For example, concerns about shorting circuits that would make the plant more difficult to control and shutdown led the Plant Superintendent to make the conscious decision to not initially use water to extinguish the fire. This position reflected a widespread reluctance on the part of licensees to use water on a fire involving electrical cables. However, failures caused by the fire as it continued to burn were largely responsible for the difficulties encountered by operators as they took actions to restore the plant to a safe and stable condition. The fire was quickly extinguished when water was finally applied. Although the Special Review Group agreed that reactor safety takes precedence over extinguishing a local fire, they recommended that water be considered if initial attempts to extinguish a cable fire with non-water means are unsuccessful. Investigations also identified other issues such as the

existence of the fire not immediately being detected by installed smoke detection systems and the exact location of the fire not being identified when construction workers initially reported the fire.

The Special Review Group urged the NRC staff and industry standards groups to accelerate efforts to develop standards and requirements for instrumentation needed to inform operator actions. Instrumentation that failed or gave erroneous or erratic indications did not result in any incorrect operator actions at BFNP. However, operators had to use indirect and inferred methods to obtain needed information and confirmatory indicators weren't available. For example, the control rod position indicators and on-line radiation monitoring systems became inoperative during the event. The Special Review Group recommended that special attention be given to instrumentation required for operator actions and that additional emphasis be placed on the concept of defining a minimum set of safety equipment.

Common mode failures

Despite the independence and separation criteria applied in the plant's design, damage to electrical power and control circuits resulted in the loss of redundant subsystems and equipment. These failures stemmed from two principal reasons: (1) potential failure sources (e.g., the interconnection of safety equipment and non-safety circuits such as indicator light circuits) were not recognized; and (2) the conduit used to isolate cables from their redundant counterparts did not adequately protect the cables (see Fig. 2). The damage inflicted by the fire resulted in the loss of Unit 1 cooling and depressurization systems (e.g., LPCI, HPCI, CS, RCIC, and the automatic depressurization systems). Although loss of the ECCS was beyond conditions for which the plant was designed, there were numerous diverse alternatives still available to the operators to provide cooling throughout the event. In addition to the Unit 1 CRD pump used during this event, there was also the potential for cooling to have been provided by the standby liquid control system and if actions were taken to allow it to be driven by steam from an auxiliary boiler, the Unit 1 RCIC.

Maintenance work control, quality assurance, and safety culture

Except for notations on design drawings, there were no written procedures or work instructions covering the sealing and testing of penetrations during their original installation or subsequent modifications. For example, the engineering aide with the assigned responsibility to find the air leaks was actually plugging the leaks (instead of the journeyman electrician).

Several references (e.g., Okrent, 1981; TVA, 1975; US NRC, 1976; Williamson, 2016; Lewis, 2010) emphasized that the BFNP fire illustrates a failure of the plant owner to institute adequate quality assurance practices, the staff or industry to develop adequate fire protection codes or regulations, and the regulatory staff to respond to an identified problem. Thus, Okrent (1981) observed that this fire is a perfect example of difficulties associated with the "thorny decision-making process" involved with modifications to equipment, procedures, or practices to an operating plant.

Regulatory actions

The BFNP fire was an important challenge to the NRC which began operation in January 1975. As a result of this fire, the NRC issued Branch Technical Position Paper (BTP) 9.5-1 and Appendix A to BTP 9.5-1 to provide licensees guidance about required fire



Fig. 2 Damage observed from BFNP Cable Spreading Room. Note melted aluminum electrical cable conduit. TVA File Photo published in U.S. Nuclear Regulatory Commission (2009) *The Browns Ferry Nuclear Plant Fire of 1975 and the History of NRC Fire Regulations: DVD*. NUREG/BR-0361, Supplement 1. Available from <https://www.nrc.gov/docs/ML0912/ML091250195.pdf>, accessed September 29, 2019.

protection programs at commercial nuclear power plants. The NRC also required each licensee to conduct a Fire Hazards Analysis (FHA) at each power plant and to respond to the requirements of the BTP. In 1980, the NRC issued fire protection regulation [Appendix R to 10 CFR Part 50 or 10 CFR50.48(b); see (U.S. Government Publishing Office, 2019)], which further codified the minimum levels of fire protection that must be provided by the licensees. Appendix R is a highly prescriptive regulation with which existing nuclear power plants must comply or from which they must request exemptions. It requires physical separation of safe shutdown equipment so that failure of one train would not affect the other train. The industry adopted compensatory fire protection measures to address times when there are non-conformances, such as the use of fire watches during maintenance activities.

The NRC approved the final rule incorporating the use of the National Fire Protection Association (NFPA) Standard 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants 2001 edition (NFPA 805) as 10CFR Part 50.48(c). NFPA 805 allows licensees to use tools such as fire modeling and risk analysis to develop a fire protection program. The NRC has indicated that NFPA 805 can be used as a performance-based method to demonstrate that fire protection safety requirements have been met, subject to certain conditions, rather than compliance with prescriptive Appendix R fire protection requirements.

1983 Salem Unit 1 Anticipated Transient Without SCRAM (ATWS) events

On two occasions (February 22 and 25, 1983), Unit 1 of the Salem Nuclear Generating Station (Salem Unit 1) located near Lower Alloways Creek Township, New Jersey, failed to scram automatically due to failure of both reactor trip breakers (RTBs) to open on receipt of an actuation signal. In both cases, the unit was successfully tripped by manual action. The failure of the breakers was attributed to excessive wear from improper maintenance of the under voltage (UV) relays which receive the trip signal from the protection system and result in the breakers opening mechanically. The Salem Unit 1 events, by themselves, were mild transients, but the potential existed for more serious events. Thus, these events provide several important lessons for improving and assuring the safety of commercial nuclear power plants.

Background

Historical information

Early concerns about ATWS in the commercial fleet were raised in Advisory Committee on Reactor Safeguards discussions with the NRC staff and reactor designers (Okrent, 1981). The primary concern, which related to potential interactions (i.e., common mode failures) between reactor control and protection systems, stemmed from an event that occurred at the High Temperature Reactor Experiment (HTRE-3) (GE, 1959). In this reactor, the control system and the protection system took inputs from the same neutron flux instruments, which did not function as designed due to incorrect installation, ultimately leading to a core damage accident.

In September 1973, the regulatory staff of the Atomic Energy Commission (AEC) documented their position on ATWS in WASH-1270 (US AEC, 1973), stating that separation of control and protection functions should be achieved to a reasonable degree, either by physical separation or by electrical isolation. Design changes necessary to either address the consequences of anticipated transients would vary depending on when an applicant submitted their construction permit application. Earlier submittals would need to demonstrate that consequences were acceptable, while later applications would need to incorporate design changes that improve the reliability of the reactor shutdown systems. Operating plants would be considered on a case-by-case basis. Subsequent evaluations (US NRC, 1978) concluded that corrective measures to reduce the probability or consequences of ATWS events were required because the reliability of current reactor trip systems (RTSs) cannot be shown to be adequate because of the rate at which they were challenged. There was concern that the RTS and other equipment required to function during an ATWS event weren't sufficiently invulnerable to common cause failures. Concerns about ATWS events were exacerbated by failures of reactor scram systems observed within the U.S. and abroad; most notably, an event at Browns Ferry Unit 3 in which 76 of the 185 control rods failed to fully insert in response to operators initiating a manual scram when the reactor was at about 35% power on June 28, 1980 (INPO, 1980; Williamson, 2016; Haskin et al., 2002).

On November 24, 1981, 15 months before the Salem Unit 1 ATWS events, the NRC invited comments on three proposed ATWS rules. Each alternative featured a different approach to reduce the risk from ATWS events. In July 1982, a group of NRC personnel from several offices was formed to consider comments received on the three proposals and to develop a final rule on ATWS. Ultimately, the ATWS Steering Group opted to evaluate generic plants using an approach that defined the various fixes and estimated the reduction in probability for ATWS sequences as each additional requirement was added. This gave a value (a benefit associated with the reduction in risk) that could be compared to the impact (cost in dollars) of each incremental requirement. Although there are large uncertainties in such analyses, they proved useful in evaluating various modifications proposed for resolving ATWS concerns.

Salem Unit 1

The Salem Nuclear Power Plant includes two four-loop pressurized water reactors (PWRs) designed by Westinghouse. Unit 1, which received its initial operating license in 1976, had a rated thermal power of 3133 MWt. On the two dates when the ATWS events occurred (February 22 and 25, 1983), Unit 1 was operating at low power levels and Unit 2 was under construction.

As typical for Westinghouse PWRs, the Salem Unit 1 RTS design uses circuit breakers that open to trip the reactor. During normal operation, the circuit breakers are closed and supply power to the CRD mechanisms. Each reactor trip breaker (RTB) undervoltage

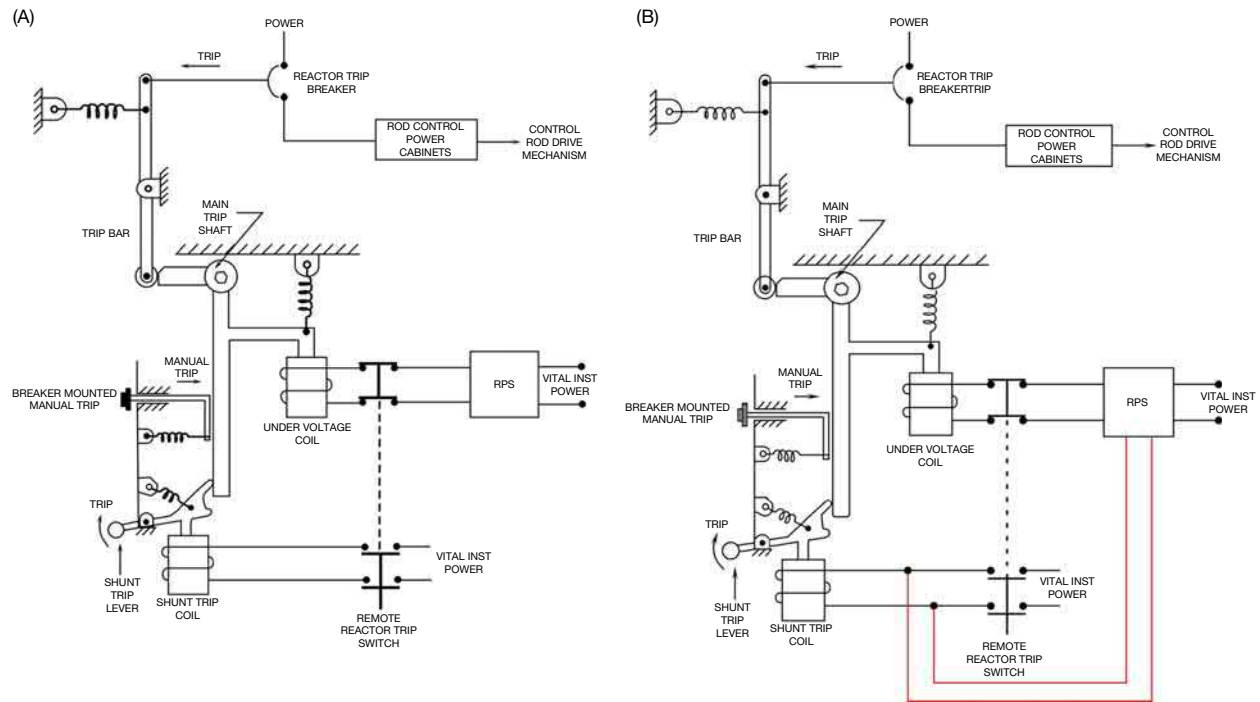


Fig. 3 Salem Unit 1 Reactor Trip Breaker (RTB) design (A) at the time of the event and (B) after modifications were implemented. US Nuclear Regulatory Commission (US NRC) (2008) *0523-R504P-Westinghouse Advanced Technology—04.7—Anticipated Transient Without SCRAM (ATWS)*. ML11216A089, July 2008. Available from <https://www.nrc.gov/docs/ML1121/ML11216A089.pdf>, accessed October 2019 (Images from pages 17 and 19/26 of pdf).

coil is energized and holds the main trip shaft in the position shown in Fig. 3A. When a trip signal is generated, the RPS de-energizes the undervoltage coil and releases the main trip shaft. This allows the trip spring to pull the top portion of the trip bar to the left and opens the RTB. Opening the RTB removes power from the power cabinets and allows the control rods to fall into the core. The undervoltage coil provides a fail-safe feature of the RPS; if a loss of power to the RPS occurs, the undervoltage coils de-energize and the RTBs open. In addition to the undervoltage coil, each RTB has an associated shunt trip coil. Energizing this coil pulls the shunt trip lever down. As this lever moves down, it contacts the main trip shaft and causes a reactor trip. Salem Unit 1 used two redundant RTBs in series to minimize the potential for spurious reactor trips (US NRC, 2008).

Event sequences

At 12:21 on February 25, 1983, a low-low water level condition in one of the four steam generators at Salem Unit 1 initiated a reactor trip signal. At the time, the reactor was at 12% of its rated thermal power in preparation for power escalation after a recently completed refueling outage. Upon receipt of the reactor trip signal, both RTBs failed to open (as noted above, at least one of these circuit breakers was required to open in order for the control rods to drop into the core and trip the reactor). About 25 s later, operators manually initiated a reactor trip from the control room and shutdown the reactor. Following the manual trip, the plant was stabilized in the hot standby condition. All other systems functioned as designed. The cause of the failure to trip was determined to be failure of the UV trip attachment devices to function as designed. The plant was placed in cold shutdown at the request of the NRC.

On February 26, 1983, NRC investigators discovered that a similar failure had occurred in Salem Unit 1 at 09:55 on February 22, 1983. With the reactor at 20% power, operators attempted to transfer the 4160-V group electrical busses from the station power transformers to the auxiliary power transformers; this was routine during power escalation. During the transfer attempt, a limit switch failed, causing the loss of one of the nonvital busses and the immediate loss of associated equipment, including one reactor coolant pump, control power to the operating main feedwater pump, control room lighting, and a 125-Vac miscellaneous distribution panel. At 09:56, a low-low level condition occurred in one steam generator (due to the loss of the main feed pump), initiating a reactor trip signal. Due to the abnormal conditions created by the loss of the 4160-V bus and in anticipation of loss of steam generator water levels, the operator was directed to manually initiate a reactor trip at about the same time. However, it was understood by plant personnel (and reported to the NRC) that the automatic reactor trip signal due to the low-low level in one steam generator had caused the reactor to trip. On February 26, 1983, as a result of NRC queries, a detailed review of the sequence of events in a computer printout for February 22 revealed that the RTBs opened in response to the operator's manual trip signal. Thus, it became evident that on February 22 (as on February 25), the two RTBs failed to open upon receipt of an automatic trip signal from the RPS.

Safety lessons

Because the operators initiated a manual reactor trip shortly after receipt of the automatic trip signals on both February 22 and February 25, no adverse consequences occurred. However, as the first observed ATWS events at a commercial nuclear power plant, the Salem Unit 1 events were of major safety concern. Safety lessons learned from these events emphasize the importance of maintenance, operating procedures, and safety culture.

Post-event testing of the RTBs indicated that both failed because of mechanical binding in the UV trip mechanisms (Thompson, 2016; Marshall, 1983). The breaker failures were attributed to excessive wear from improper maintenance of the UV relays. Root causes included: (1) inadequate attention to the importance of vendor-supplied and NRC-supplied information, (2) absence of an adequate preventive maintenance program, and (3) an inadequate supply, control, and verification of information by the vendor.

While the Salem Unit 1 RTB failures occurred as a result of several possible contributors, evaluations concluded that the predominant cause was excessive wear accelerated by lack of lubrication and improper maintenance. Although other PWRs had experienced RTB failures, none involved an ATWS event.

Regulatory actions

Several NRC actions were initiated in response to these events. One action was plant-specific and addressed before Salem Unit 1 could be restarted. Another was an investigation into the Salem events and the circumstances leading to them. In addition, a task force was formed to study the overall generic implications of this event. Several short-term and long-term actions related to RPS design and testing were taken.

Plant-specific actions

Because of previously identified problems at Salem Unit 1 and the licensee's failure to recognize that an ATWS event had occurred on February 22, 1983, the NRC did not permit restart until technical and management corrective actions were satisfactorily addressed. Detailed reviews of the event and the licensee's corrective actions were performed prior to authorizing restart of Salem Unit 1. As shown in Fig. 3A, the original UV trip attachment design provided only one means by which the RPS would automatically open the RTBs, de-energizing the undervoltage coils. In the aftermath of the Salem events, the RTB device was modified to provide redundant means of automatically opening the breakers. With the modification shown in Fig. 3B, the RPS energizes the shunt trip coil and de-energizes the undervoltage coil for each breaker when the trip logic is satisfied.

On April 26, 1983, the NRC agreed that Salem Unit 1 could be returned to service; however, on May 5, 1983, the NRC forwarded to the Salem licensee a Notice of Violation and Proposed Imposition of Civil Penalties. Violations included operation of the reactor when the RPS could not be considered operable and several significant deficiencies which contributed to the inoperability of the RTBs. An augmented inspection program was instituted to monitor the licensee's progress toward completion of longer-term corrective actions.

Short-term actions

Subsequent initiatives identified and corrected potential deficiencies in RTBs and related maintenance procedures at other plants. Due to the serious nature of these events, the NRC immediately issued Inspection and Enforcement Bulletin No. 83-01 (US NRC, 1983a) to all PWR licensees for action and to other nuclear power reactor facilities for information. IE Bulletins 83-01 and 83-04 (US NRC, 1983a,b) required testing of all circuit breakers in RTSs with an undervoltage trip attachment. IE Information Notice 83-08 (US NRC, 1983c) described the failures of the RTBs discovered in testing and provided recommendations to assure proper operation of RTBs with UV trip attachments being used in other safety-related applications.

In addition, an NRC Task Force was established to determine the generic implications of the Salem Unit 1 events. This work is documented in NUREG-1000 (US NRC, 1983d). As a result, Generic Letter 83-28 (US NRC, 1983e) was issued describing intermediate-term actions to be taken by licensees. The actions address issues related to RTS reliability and general management capability; specifically: (1) post-trip review (the licensee must have adequate procedures, data, and information sources to understand the cause(s) and progression of the trip such that an informed decision on the acceptability of a reactor restart can be made), (2) equipment classification and vendor interface to ensure programs are in place to identify and maintain safety-related components, (3) post-maintenance testing of safety-related equipment, and (4) RTS reliability improvements (US NRC, 1983e).

Long-term impact

The Salem Unit 1 ATWS events contributed to several actions with far reaching effects on the industry (Thompson, 2016). Evaluations of the Salem events were considered in NRC deliberations regarding the 1984 ATWS rule now outlined in 10 CFR 50.62 (U.S. Government Publishing Office, 2019). The Salem events were also considered prior to issuance of Generic Letter 83-28, which directed the industry to establish formal vendor interface programs with an inferred wider scope covering nearly all safety-related equipment. PWRs were required to have equipment that is diverse, reliable, and independent from the RTS to automatically initiate the auxiliary feedwater system and to initiate a turbine trip under conditions indicative of ATWS. BWRs were required to have a diverse recirculation pump trip, alternate rod insertion circuitry, and upgraded emergency operating procedures; or

alternatively, BWRs could have installed high capacity standby liquid control systems. In addition, the staff raised several issues that were closely related to RPS design and testing.

One of the principal findings regarding the Salem Unit 1 ATWS events was the lack of adequate attention being paid to the reliability of the RTS. The Salem Generic Issues Task Force recommended to the Commission that a reliability assurance program be included in the final ATWS rule. While the ATWS rule did not require such a program, the Commission strongly urged the voluntary development of this reliability assurance program. The Commission stressed that ATWS risk reductions can also be achieved by reducing the frequency of transients requiring RPS operation. Challenges to the RPS may arise from unreliable components, inadequate post-trip reviews, poor testing, or tolerance of inadequate or degraded control systems. While not specifically required by the ATWS rule, the Commission urged licensees to analyze challenges to the plant safety systems, particularly the RTS, and determine how improvements could be made. Industry response to this challenge included several major actions (e.g., see Haskin et al., 2002).

The NRC staff in its review of responses to Generic Letter 83-28 approved the installation and operation of the automatic shunt trip in many operating PWRs. As discussed above, the shunt trip attachment was designed to improve the reliability of the circuit breaker, as its function is redundant to the function of the existing UV trip attachment. The staff encouraged each licensee to install the automatic shunt trip at the first refueling following staff review and approval of their design. In addition, the reliability of circuit breakers improved due to more effective maintenance, testing, and inspection programs (Subudhi et al., 1990).

Summary

The 1975 fire in Browns Ferry Nuclear Plant Units 1 and 2 and the 1983 ATWS events at Salem Unit 1 did not result in any adverse effects to public health and safety. However, important safety lessons were gained, and measures were implemented to enhance the safety of the commercial operating fleet. It is important that the industry remain vigilant in identifying safety lessons from future events and in implementing appropriate corrective actions.

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The Three Mile Island Unit 2 Accident

Gordon R. Skillman^a and Joy L. Rempe^b, ^a Skillman Technical Resources, Inc., Hershey, PA, United States; and ^b Rempe and Associates, LLC, Idaho Falls, ID, United States

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Glossary

Core uncover The situation in which the water level within the reactor pressure vessel drops below the top of the fuel elements within a reactor core.

Loss of feedwater In Babcock and Wilcox (B&W) pressurized water reactors (PWRs), a loss of all main feedwater to once-through steam generators.

Pilot operated relief valve (PORV) The power operated relief valve mounted on top of the pressurizer. The PORV is also referred to as the electrometric relief valve.

Severe accident An accident resulting in significant damage to a reactor core with subsequent release of large quantities of radioisotopes beyond the primary coolant system.

Introduction

On March 28, 1979, a loss of feedwater event ultimately led to the first core melt accident in a Pressurized Water Reactor (PWR) deployed for commercial power production. After a brief overview of the plant design, this chapter provides an overview of the accident evolution, activities required for initial stabilization, followed by post-accident analyses and examinations required for long-term stabilization, cleanup, and developing insights. Actions taken to address the insights learned from the TMI 2 accident ultimately increased the safety of the global nuclear enterprise.

TMI 2 design

The Three Mile Island Nuclear Station is a two-unit nuclear power plant located on Three Mile Island in the Susquehanna River, approximately 10 miles southeast of Harrisburg, Pennsylvania (see Fig. 1). Three Mile Island Unit 1 (TMI 1) was licensed to begin operation in 1974 at a rated power of 2535 MWt. Three Mile Island Unit 2 (TMI 2) was licensed to begin operation in 1978 at a rated power of 2772 MWt. TMI 2 had been in commercial operation for less than 90 days when the accident occurred on March 28, 1979.



Fig. 1 Three Mile Island Nuclear Station. Cooling towers and structures for TMI 1 are on the *left* (with steam plume) and for TMI 2 are on the *right* (without steam plume). Courtesy FirstEnergy (current owner/operator of TMI 2 and historical information of TMI 2 images).

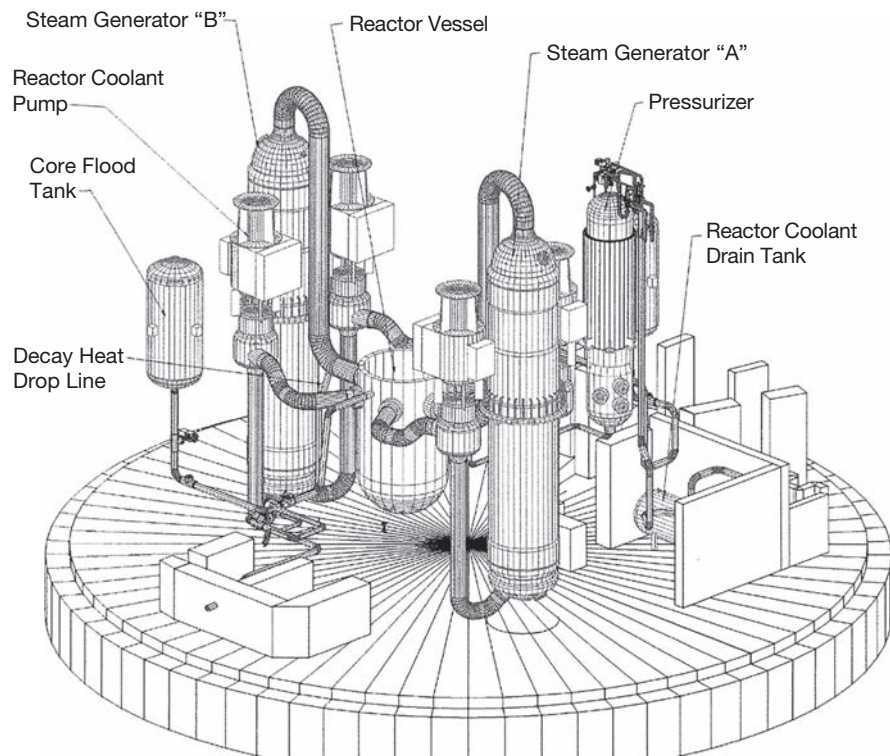


Fig. 2 Isometric depiction of TMI 2 NSSS showing one of the two CFTs and the RCDT. Courtesy FirstEnergy (current owner/operator of TMI 2 and historical information of TMI 2 images).

Nearly identical to TMI 1, the TMI 2 Nuclear Steam Supply System (NSSS) included a reactor pressure vessel (RPV) with a core containing 177 fuel assemblies, two coolant loops (often designated as Loops A and B), one reactor core drain tank (RCDT), one electrically heated pressurizer, and connecting hot leg, cold leg, and surge line piping. Each coolant loop included a once-through steam generator (OTSG) and two reactor coolant pumps (RCPs). Connected on opposite sides of the RPV, and discharging directly into the RPV downcomer, are the two core flood tanks (CFTs). The pressurizer included one pilot operated relief valve (PORV) and two American Society of Mechanical Engineers (ASME) code-required safety relief valves. The Babcock and Wilcox (B&W) NSSS design is unique in that its Makeup and Purification system includes three identical centrifugal pumps, each serving identical dual functions. These pumps provide normal makeup and RCP seal injection flow as well as High Pressure Injection (HPI)/Emergency Core Cooling (ECCS) flow. The physical configuration and geometric layout of the NSSS are shown in **Figs. 2 and 3**. **Fig. 4** shows the layout of TMI 2 buildings, including the large dry containment (e.g., the reactor building housing the NSSS), the auxiliary building, and the turbine building.

Accident evolution

Plant status

Prior to the accident, TMI 2 was at 97% power with all four reactor coolant pumps (RCPs) operating ([EPRI, 1979](#)). The MU-P-1B pump was in service providing normal makeup and RCP seal injection flow. RCS leakage was approximately 22.7 L/m (6 gpm),

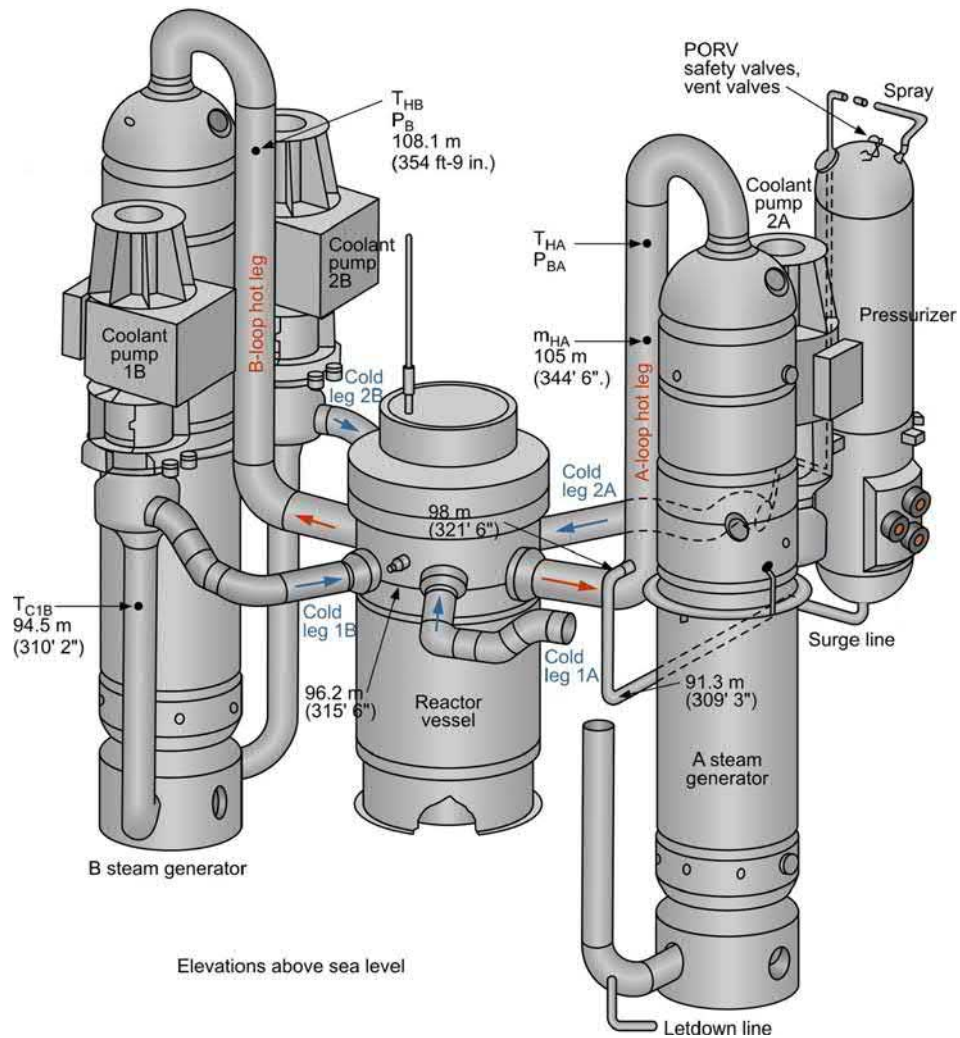


Fig. 3 Isometric showing Coolant Loops A and B of TMI 2 RCS without Core Flood Tanks (CFTs). Rempe JL and Knudson DL (2013) TMI-2—A case study for PWR instrumentation performance during a severe accident. *Idaho National Laboratory Report INL/EXT-13-28043*, p. 16/118.

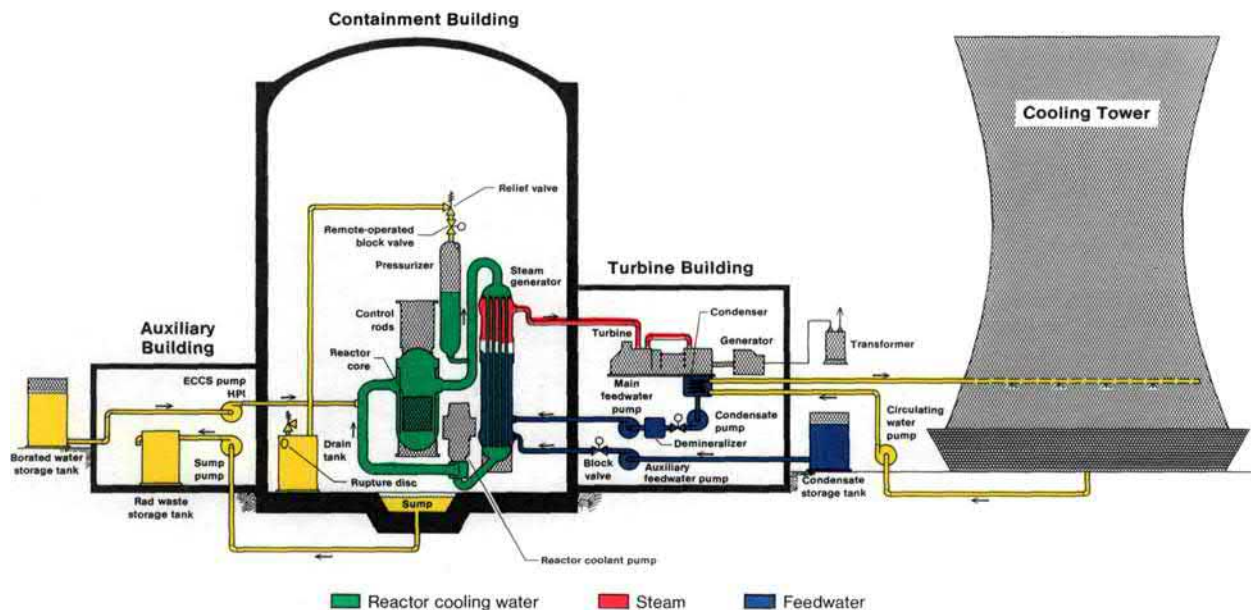


Fig. 4 TMI-2 structures. EPRI (1979) Analysis of Three Mile Island-Unit 2 accident. *Electric Power Research Institute Nuclear Safety Analyses Center Report NSAC-80-1*, p. 28/531.

which was deemed acceptable for TMI 2 at that time. The condensate system was operating with two condensate pumps and two condensate booster pumps in service. Both turbine driven feedpumps were in service. The spray in the pressurizer was operating to equalize the boron concentrations between the pressurizer and the RCS. This equalization was necessary because of steam leakage that was suspected to be from either the relief valve (PORV) or the pressurizer safety valves (ASME valves). This leakage was evidenced by periodic safety valve discharge header temperature alarms. In addition, the plant staff was continuing to perform maintenance on a plugged condensate polisher transfer line and had been doing so for the previous approximately 11 h.

TMI 1 was in hot shutdown for low power physics testing following refueling. There was a vacuum on the condenser, and auxiliary steam was being supplied from TMI 2.

Sequence of events

At approximately 04:00 on March 28, 1979, auxiliary operators were transferring resin from a condensate polisher tank to a resin regeneration tank, encountering difficulty that was attributed to resin blockage in the transfer line. While these operators were clearing the resin blockage, the plant experienced a loss of condensate flow followed by total loss of feedwater to the steam generators. Condensate flow loss was due to the unexpected shutdown of the condensate polisher system pumps in conjunction with a bypass valve failure. Steam generator feedwater was lost because the main feed-water pumps were automatically tripped in response to cutoff of water from the condensate polishers and because flow from the auxiliary feed-water pumps was blocked by two valves that had been inadvertently left closed during prior maintenance. The combined loss of condensate and feedwater flow resulted in the main turbine being tripped at 04:00:37. The emergency feedwater pumps started as designed, and the reactor continued to operate at full power. The PORV opened at its setpoint of 15.5 MPa-g (2255 psig). The reactor tripped when the RCS high pressure trip setpoint of 16.2 MPa-g (2355 psig) was reached.

On reactor trip, coolant in the RCS contracted due to reductions in RCS temperature and pressure. To compensate for RCS coolant contraction, operators took appropriate action to reduce letdown flow and to start a second HPI/Makeup Pump. About 13 s later, the PORV failed to close when RCS pressure decreased to its closure setpoint of 15.2 MPa-g (2205 psig). This was not the first time that a PORV had stuck open. In fact, the same sequence of events occurred in 1977 at a similar B&W plant, Davis-Besse. In that event, an operator recognized that the PORV remained open and immediately blocked it. At TMI 2, this failure of the PORV-to-close was not recognized for about 2 h and 20 min.

About a minute later, the pressurizer water level began to increase. This level increase continued until approximately 6 min after the initiation of the event, at which time the pressurizer level indication went off scale "high" indicating that the pressurizer was completely filled with water or "solid." This was of particular concern to the operators for several reasons. Under normal operating conditions, the pressurizer contains both water and a generous steam cushion that controls RCS pressure, and the steam cushion assures that water is not introduced into the PORV or the ASME valves. Furthermore, because the RCS is filled with water during startup and operation, there is no indication of water level within the core. Operators infer core water level based on the pressurizer water level. Hence, operators were concerned about the pressurizer going "solid" because it impairs their ability to control RCS pressure and they would not be able to prevent primary coolant water from being introduced into the ASME valves and the PORV.

Operators attempted to reduce pressurizer water level by throttling the HPI flow rate and by increasing the letdown flow rate to its maximum value. These efforts were largely unsuccessful. RCS pressure began to increase moderately as the pressurizer went solid; the RCS temperature also increased. The high pressurizer level was caused by voids that formed in the RCS coolant coupled with the open PORV (effectively venting the steam space of the pressurizer to the RCDT and allowing a rapid fluid insurge into the pressurizer).

At approximately 8 min after the start of the accident, an operator discovered that the block valves on the emergency feedwater headers were closed. Upon opening the block valves, the emergency feedwater pumps discharged water into the OTSGs rapidly lowering RCS temperature and associated pressure.

Approximately 14 min after the start of the accident, the rupture disc of the RCDT (see Fig. 2) burst, discharging steam and reactor coolant from the open PORV, combined with original RCDT quench fluid, into the containment.

Reactor coolant mass inventory continued to decrease due to the combination of fluid loss through the open PORV and due to the sustained low coolant injection rate of the throttled HPI system. Although the RCPs continued to operate, their flow rate decreased because conditions were outside their operating limits. The void fraction within the RCS increased to such an extent that it caused cavitation in the RCPs. The operators tripped the RCPs in the B loop at 74 min, and the RCPs in the A loop at 101 min.

When the RCPs in Loops A and B were tripped, the operators expected that natural circulation would initiate. Plant parameters, however, were outside of the operating envelope for natural circulation, and natural circulation did not initiate. Accumulated vapor in upper locations of each loop precluded natural circulation. At this time, RCS pressure continued to decrease but RCS temperature started to increase. Abundant heating due to decay heat generation, combined with the continued unrecognized reactor coolant loss through the PORV and the throttled HPI system flow, contributed to this temperature increase. The throttled HPI flow was considered acceptable by the operators because of the apparently satisfactory, but misunderstood, pressurizer water level.

The RCS pressure decreased to its lowest pressure of 4.6 MPa-g (660 psig) at 2 h and 19 min after the initiation of the accident. At 2 h and 20 min, the operators recognized that the PORV (RC-RV2) was open and proceeded to close its isolation (block) valve, RC-V2. Upon closure of RC-V2, RCS pressure began to increase. This pressure increase was likely caused by heatup from core decay heat and zirconium-water reactor and by additional RCS inventory associated with higher HPI flow rates. The operators believed

that, up to this time, no substantive inventory loss from the RCS had occurred. This belief, however, was also based on the misunderstood pressurizer level without any consideration of the accompanying low RCS pressure.

At approximately 2 1/2 h after the start of the accident, substantial fractions of the reactor core were uncovered and were experiencing sustained high temperatures. These conditions resulted in significant fuel damage, substantial releases of core fission products, and hydrogen generation. The operators recognized neither the magnitude nor extent of these conditions. Additional unsuccessful attempts to establish sustained forced cooling with one or more RCPs were made.

The operators continued unsuccessfully to attempt to collapse the voids in the RCS loops for several hours by opening and closing the PORV block valve, RC-V2, thus relieving the pressure by discharge to the RCDT through the still stuck open PORV. The operators became concerned over the ability of the PORV to remain functional under this high use rate. As a preliminary step for initiating the installed decay heat removal system, the decision was made to reduce the RCS pressure, which would allow water from the CFTs to enter the RCS and assure coverage of the core. This action was largely unsuccessful. Water levels in the CFTs changed only slightly, indicating that only a small amount of CFT fluid entered the reactor vessel. Failure of the CFTs to inject substantial fractions of their fluid volume was interpreted as indication that the core was covered. The CFTs are designed to supply water to the reactor vessel in the event of a large break loss of coolant accident (LOCA) with its accompanying large, rapid depressurization, and for medium sized breaks where CFT fluid injection precedes Low Pressure Injection (LPI). Neither the large and rapid depressurization that accompanies a Large Break LOCA, nor a slower depressurization that accompanies a medium sized break, occurred during the “essentially” small break LOCA associated with TMI 2’s PORV failure to close. Consequently, the CFTs were not effective for core reflood during this scenario.

The extended period of low RCS pressure allowed hydrogen gas, which was generated due to oxidation of zirconium in the fuel cladding, to be released from the RCS. Some of this hydrogen burned in the reactor building at about 10 h after the initiation of the accident producing a rapid pressure spike (at least 0.19 MPa G/28 psig) in the containment. This rapid pressure spike, which was indicated by a single pressure data point, received little attention from most of the plant staff; many were unaware that it had taken place.

Hydrogen release from the RCS may have contributed to the later apparent success of the operators in collapsing the voids in reactor coolant Loop A, to which the pressurizer is connected. This apparent success in establishing what appeared to be some degree of natural circulation in spite of continuing high temperatures in portions of the RCS system likely led the operators to conclude that the RCS had attained stable thermal hydraulic conditions.

In mid afternoon of March 28, 1979, corporate management directed the operators to raise RCS pressure with the objective of collapsing the remaining voids and to start a RCP to establish forced circulation. This was successfully achieved at 1950 h on March 28, 15 h 50 min after the start of the accident.

TMI 2 stabilization

Reactivity management

The as-designed reactor trip on high pressure at approximately 0400 on March 28 resulted in successful insertion of all 61 scrammable control rods ensuring immediate reactor shutdown. Eight axial power shaping rods that were not designed to scram remained in their pre-scram location, 27% withdrawn. No further adjustment of the control rods occurred subsequent to the accident. The normally operating makeup pump MU-P-1B continued to inject water into the RCS injecting from the letdown-storage tank at then current RCS boron concentration levels. The HPI pumps delivered water from the borated water storage tank to the RCS cold legs during their brief use on March 28, injecting several hundreds of gallons of ~2400 ppm borated water into the RCS through nozzles located in the RCS cold legs (Fig. 3). In the weeks after the accident, uncertainty regarding the condition of the core increased. Specifically, uncertainty increased about the core’s geometry, the potential for significant core damage based on the extraordinary radiation levels emerging throughout the plant, and the ability of the control rods to maintain the core subcritical. To address these uncertainties, the RCS boron concentration was first increased to the refueling concentration of 2400 ppm, then, with subsequent intermediate increases, to 6000 ppm.

Long-term cooling

From the afternoon of March 28, 1979, until approximately noon on April 27, 1979, TMI 2 was successfully cooled by manual adjustment of the feedwater flow rate to the Loop A OTSG using a small bypass valve around the main feedwater control valve. Steam created in the Loop A OTSG was drawn to the main condenser by a vacuum through the turbine bypass valve, which was set to control the rate of steaming and hence, the rate of heat removal.

As described by [Elam and Skillman \(1980\)](#), the decision to transition from forced cooling to natural circulation was made in accordance with procedure after the third (and last) pressurizer transmitter failed at 1:07 a.m. on April 27, 1979, approximately 30 days after the initiation of the accident. Approximately 12 h later, the last operating RCP (RC-P-2A) was manually tripped, reducing heat input from the pump into the RCS and resulting in a cool down rate of approximately 6 °C/h (10 °F/h) until a new equilibrium RCS temperature was reached (i.e., when the heat removal rate matched the decay heat generation rate). The transition was made with both steam generators steaming, but the B-Loop OTSG was isolated due to high iodine activity at approximately 1:10 a.m. Thus, the A-Loop OTSG was used exclusively to remove decay heat and promote natural circulation. Prior to this

planned transition to natural circulation, several independent analyses evaluated plant performance. These evaluations concluded that natural circulation would provide adequate capability to remove both decay heat and RCS sensible heat until a long-term decay heat removal system could be installed. Data taken after stopping the last RCS pump confirmed the predicted performance.

RCS pressure control

During the initial hours and days after the accident, RCS pressure was controlled by maintaining a pressurizer steam bubble using the installed electrical heaters (Heater resistance measurements, which were easily accessible because heater distribution panels and circuit breakers were located in the cable spreading room outside the containment, were used to determine that 1100 kW of the 1628 kW of pressurizers heaters were still dependable.) (Elam and Skillman, 1980). Furthermore, reliable pressurizer water level instrumentation, was no longer available. After the transition to natural circulation, the pressurizer was filled with water, and RCS pressure was maintained by adjustment of inflow from the centrifugal makeup pumps. By positioning the letdown valve to balance net in-flow across the normal makeup valve and one of the RCP seal injection valves, RCS pressure was successfully controlled without requiring knowledge of pressurizer water level. This method of pressure control proved to be stable and reliable.

TMI 2 remained in its damaged state from March 29, 1979 until fuel removal began in the fall of 1985. TMI 2 began natural circulation heat removal on April 27, 1979. This method of cooling remained effective first through the steam generators, then through quiescent heat exchange from the reactor vessel to the containment until the last amount of fuel was removed from the reactor vessel years later. Pressure control as described above was effective for the entire duration of natural circulation heat removal and ceased when the RPV head was removed in 1985.

Post-accident damage assessments

Damage assessments, which were required for plant stabilization and cleanup, primarily relied on three sources of information: plant instrumentation data, analysis predictions, and examination observations.

Instrumentation data

Plant data were the primary source of information initially available to operators for assessing the status of TMI 2 and the effects of mitigating actions. The small break LOCA and hydrogen burn events, however, exposed instrumentation to harsh conditions, including direct radiation, radioactive contamination, and high humidity with elevated temperatures and pressures. As discussed in Lutz and Williamson (2021), the TMI 2 accident highlighted the importance of understanding and focusing on key system status information in an environment where operators can be overwhelmed with superfluous and sometimes conflicting data (e.g., sensors allowed over 3000 measurements to be obtained at TMI 2). In order to assess and learn from the TMI 2 accident, an effort was completed to evaluate sensor survivability and qualify instrumentation data and associated uncertainties (Rempe and Knudson, 2013, 2014). This effort used several techniques, including comparisons with data from other TMI 2 sensors, analytical calculations, laboratory testing, comparisons with sensors subjected to similar conditions in large-scale integral tests, and other insights available from post accident examinations. Several examples are highlighted here to illustrate important insights gained from TMI 2 sensor and data evaluations.

Although post-accident examinations provided insights about the endstate location of core materials, such insights were not available until years after the accident and they provided no indication of the timing of accident progression. However, the simultaneous increase in source range monitor count rate, RCS pressure, and cold leg temperatures (see Fig. 5) provided a much earlier indication that a major relocation of core materials occurred between 224 and 226 min after the start of the accident.

A sharp pressure spike in the containment from wide range pressure measurement (see Fig. 6) was initially dismissed as electrical noise. Calculations assuming this peak containment pressure yielded peak containment temperatures of 650 °C (1200 °F), much higher than the measured 93 °C (199 °F) peak temperature data. However, photos of damaged components (e.g., a melted phone, collapsed storage drums, etc.), supported the accuracy of this single data point indicating a hydrogen burn occurred in the TMI 2 containment. Further investigations revealed that the sampling rate for obtaining containment temperature was limited. Hence, experts confirmed the accuracy of the measured peak containment pressure.

Post-accident evaluations also attributed data unavailability to computational limits, such as storage memory, inadequate paper or ink, insufficient sampling rates, and “preset” limits associated with anticipated operating ranges. For example, the lower pressure limit prohibited data being obtained below 11 MPa-gauge (1600 psig) by the RC-3B-PT1-R RCS pressure sensor (see Fig. 7).

Analyses

Analyses initially relied on limited TMI 2 instrumentation data and available knowledge of severe accident phenomena to predict core damage. Results from chemistry evaluations of RCS water confirmed that much of the fuel cladding had failed, resulting in the release of billions of curies of fission products to the RCS and the containment. High radiation levels in the containment and in many areas of the auxiliary building initially delayed forensics investigations. RCS water streamed prohibitive radiation levels that prevented access to any location where reactor coolant resided. Early analyses, such as GEND (1981) and LANL (1983),

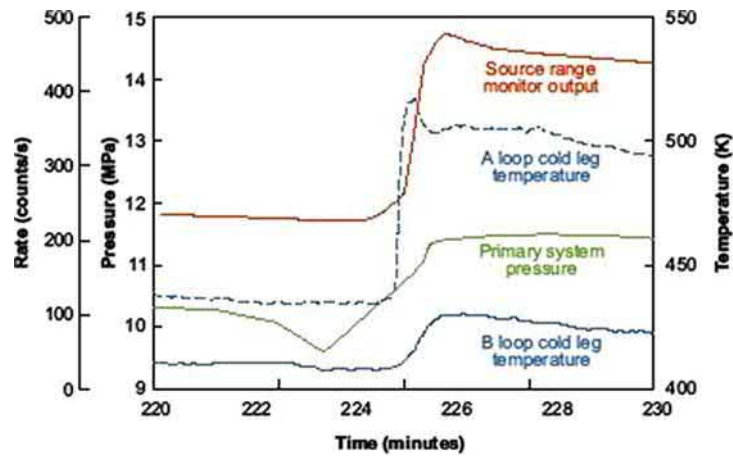


Fig. 5 Overlay of SRM count rate, RCS pressure measurements, and cold leg temperatures. Rempe JL and Knudson DL (2013) TMI-2—A case study for PWR instrumentation performance during a severe accident. *Idaho National Laboratory Report INL/EXT-13-28043*, p. 22/118.

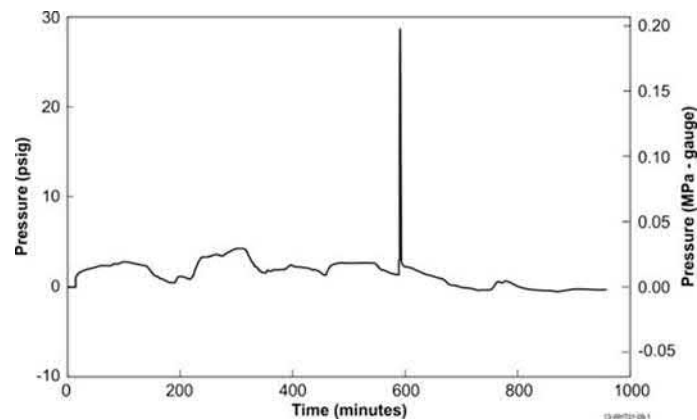


Fig. 6 Combined TMI 2 containment narrow range and wide range pressure transmitter data. Rempe JL and Knudson DL (2014) Qualification of Daiichi units 1, 2, and 3 data for severe accident evaluations—Process and illustrative examples from prior TMI-2 evaluations. *Idaho National Laboratory Report INL/EXT-14-32628*, p. 59/68.

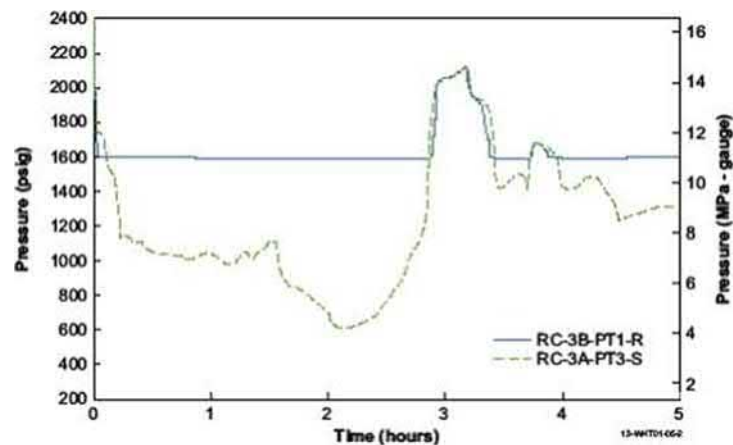


Fig. 7 RCS pressures at selected locations. Rempe JL and Knudson DL (2013) TMI-2—A case study for PWR instrumentation performance during a severe accident. *Idaho National Laboratory Report INL/EXT-13-28043*.

were essential in developing an understanding of when the core became damaged, the extent of such damage, and what hazards might be encountered during future activities after the vessel head and upper plenum structures were removed.

GEND (1981)

Results from GEND (1981), which compared results from several early evaluations, provided minimum and maximum bounds for core damage to establish a “reference” description for the core endstate. This comparison provided the basis for developing tooling and procedures for subsequent TMI 2 inspections, sample acquisitions, and defueling. Key insights from such evaluations included:

- *Fuel failure:* During the period of core uncover (from about 100 to 174 min), virtually all the rods were predicted to reach temperatures in excess of 757 °C (1394 °F), and more than 90% were expected to have failed. Estimated times when cladding failure occurred at temperatures from 757 °C to 877 °C (1394 °F–1610 °F) ranged from 137 to 142 min after the start of the accident. This coincides well with an estimated 3-min time for fission products to be transported to the containment radiation monitors, which responded at 145 min.
- *Cladding oxidation:* The exothermic cladding oxidation reaction contributed to rapid core heatup. Steam rising from lower core regions carried energy to upper core regions, increasing fuel rod temperatures. Hot zircaloy in upper core regions may have been fully oxidized, and the upper half of the core was estimated to be embrittled.
- *Liquefied fuel formation:* During the period of core uncover, essentially all the reviewed investigations predicted liquefied fuel would have been produced in small quantities. Estimates for peak fuel rod temperatures varied from 2127 °C to 2627 °C (3860 °F to 4760 °F). At temperatures above 1902 °C (3455 °F), GEND (1981) observed that experimental evidence indicated UO₂ fuel pellets in contact with the cladding can be dissolved by zircaloy, forming a liquid phase of Zr-U-O termed “liquefied fuel.”
- *Core component damage:* Analyses indicate that axial power shaping rods, control rods, burnable poison rods, spacer grids, and guide tubes would have experienced damage. Many of these materials were predicted to contribute to formation of an in-core debris bed with fused portions of relocated core rubble. Because the zircaloy cladding of the burnable poison rods oxidized over the same region as the cladding on the fuel rods, these rods were also predicted to fragment. GEND (1981) also estimated that upper plenum temperatures would melt Beryllium-Nickel material that attaches the control rod guides (brazements) to the structural support plates in the upper plenum. Stainless steel components, such as the fuel assembly upper end fittings, were also predicted to have melted or fused at their contact points within the plenum. Control rod spider failure and leadscrew distortion would also be likely.
- *Cladding fragmentation:* At 174 min, with the coolant mixture level at about 1.5 m above the bottom of the core, one RCP in the B loop was turned on for 19 min. The sudden influx of cold water to the core thermally shocked the embrittled cladding, causing fragmentation of the ZrO₂ and UO₂. These fragmented materials were predicted to either form a debris bed above the axial mid-plane of the core or increase the size of any existing debris bed. Although substantial quenching of the rods could occur, the debris bed would have remained hot and in contact with steam.
- *Major core material relocation:* The next major change in core condition occurred between 222 and 226 min into the accident. As illustrated in Fig. 5, source range monitor data showed a sharp increase in activity, the primary system pressure increased even though the block valve was open, and the cold leg temperatures of both the A and B loops increased. Estimates based upon thermocouple and Self Powered Neutron Detector (SPND) data indicate that temperatures of 530 °C (980 °F) were reached at locations 25–75 cm above the bottom of the fuel rods. GEND (1981) speculated that additional core damage occurred during this time. Given the existence of oxidized and embrittled cladding prior to about 225 min, it is possible that “unstable thermal hydraulic conditions” developed, fractured additional cladding, and led to some additional slumping of the core and densification of the debris bed. A steam blanket may have formed below the lower debris bed crust, blocking coolant ingress but permitting additional zircaloy oxidation and hydrogen formation. On the basis of available instrumentation data, no apparent change in the condition of the core occurred after about 226 min.

LANL (1983)

LANL (1983) provided an updated core damage assessment using an integrated system of computer codes, MIMAS and TRAC-PF1, to simulate heat transfer and fluid dynamic processes. This accident progression analysis, which considered observed events, initial examinations, and sample analysis results, was performed to aid development of equipment and procedures for cleanup and recovery of TMI 2 and to identify important data that should be collected in future examinations to reduce phenomenological uncertainties. The accident was analyzed from the initial loss of main feedwater to a point in time just after HPCI and subsequent core reflood occurring at 3 h and 20 min. Results indicated that the Inconel grid spacers melted first, followed by melting of the upper 80% of the control rods, and then fragmentation of the upper 80% of embrittled fuel rods. Degraded material relocated to lower core locations where a debris bed formed with a packing fraction of 30% between fuel rod stubs and 44% above these stubs. The calculated debris generation and packing fractions, in conjunction with the TMI “Quick Look” observations, indicate that 22% of the fuel rod debris would be relocated from the core region. The hydrogen production during the accident transient was estimated to be about 400 kg (880 lb_m), with the majority (about 75%) being generated during reflood. As noted below, defueling efforts in 1986 revealed that the amount of core materials predicted to relocate by LANL several years earlier was fairly accurate.

Examinations

Quick Look examinations and ultrasonic surveys

The first direct visual observations of TMI 2's core were the three "Quick Look" examinations conducted on July 21, 1982, August 5, 1982, and August 12, 1982, over 3 years after the accident. Each of these examinations was performed by inserting a small camera into the reactor vessel through one of several control rod drive lead-screw locations near the center of the reactor vessel head, thus near the center of the core. This was accomplished by removing the control rod lead screw and then lowering, by gravity, an approximately 1.5" (3.8 cm) diameter camera with lighting into the lead screw opening. Personnel introduced the camera into the lead screw opening while standing on a work platform located on top of the reactor vessel service structure.

On the first examination on July 21, 1982, "Nothing was observed until the camera encountered a rubble bed approximately 1.5 m (5 feet) below the top of the core region." (GEND, 1983 from Volume 1, Page 1–3) The discovery of this void instantly communicated the gravity of the accident. In 1983, ultrasonic scanning surveys were used to determine the shape and dimensions of materials remaining in the core cavity (US NRC, 2016). These surveys indicated that the voided cavity volume was approximately 26% of the original core region. Three years later, in 1985, when the upper plenum was removed from the reactor vessel, the extent of this damage was observed.

The most fully exposed and vulnerable component in the reactor vessel internals is the upper core support grid. It is the lowest, or "bottom," structural member of the 55-ton reactor vessel upper plenum structure. The design functions for the upper plenum and the upper grid are to index the fuel assemblies by maintaining design clearance between the upper end fittings of each of the 177 fuel assemblies, to sheath and protect the withdrawn or partially withdrawn 69 control rods, to prevent fuel assembly lift due to upward flow forces by providing hold-down force by fuel assembly spring compression, and to accomplish these functions simultaneously on all 177 fuel assemblies and all 69 control rods. The "Quick Look" examinations revealed the extent of structural melting and voiding on the upper grid (see Fig. 8). Data from Quick Look examinations were critical for designing defueling equipment tools and fixtures.

Later examinations and endstate

Images obtained prior to and during defueling, which was an essentially vertical mining process, provided evidence where damage sustained by internal support structures and penetrations was more severe (e.g., Fig. 9). Using examination information, along with results from analyses based on qualified instrumentation data, a debris endstate configuration figure (see Fig. 10) was developed that identified the location of fuel containing materials within the reactor vessel, including granular debris, crust, partially intact fuel assemblies, and "corium," a term used to describe previously molten material containing fuel, clad, and structures. As additional examination information was obtained, this figure was updated.

Examinations also provided insights about the damage induced by relocating core materials to the lower head and its penetration nozzles (see Figs. 10 and 11, Rempe et al., 2012). The most severe damage was observed in nozzles located within the "hot spot" where vessel steel examinations indicated peak temperatures occurred (e.g., the nozzle located at position H8). As discussed in Wolf et al. (1994), examinations indicated that the observed damage was not related to the embedded debris height (e.g., nozzle L6 was submerged in debris, but remained undamaged) and that lower portions of the nozzles appeared to have been protected by crusts that rapidly formed near vessel surfaces (e.g., note undamaged region at the base of nozzle H8).

As noted above, earlier investigations concluded that approximately 20% of the core materials relocated as a liquid phase and solidified in lava-like formations in the core bypass region, the core support assembly, and the reactor vessel lower head. Later evaluations (Russell and McCardell, 1989) estimated that uncertainties for relocated masses at each location ranged from 5% to 40% (regions in which defueling was completed had less uncertainty).

Fuel removal from TMI 2, which was initiated on November 12, 1985, continued for 5 years. A total of 133,000 kg (293,000 lb_m) of the 140,000 kg (310,000 lb_m) of fuel, cladding, structural and control materials was removed, including upper

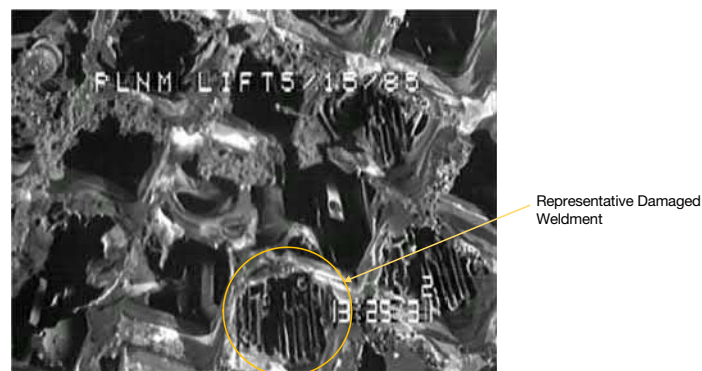


Fig. 8 Image of upper core support grid taken when the upper plenum structure was removed—May 15, 1985; note degraded grid structure and weldments. Courtesy FirstEnergy (current owner/operator of TMI 2 and historical information of TMI 2 images).

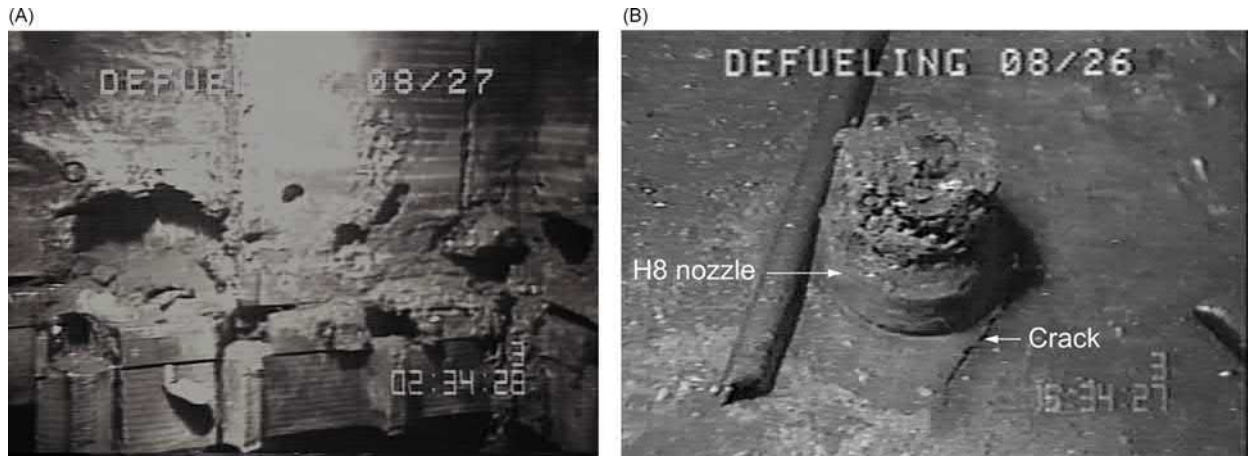


Fig. 9 TMI-2 video examinations revealed damage to the (A) core barrel (B) H8 instrument guide tube nozzle, and crack in vessel cladding (Subsequent evaluations revealed crack was limited to cladding and did not penetrate into vessel.). Courtesy First Energy (current owner/operator of TMI 2 and historical information of TMI 2 images).

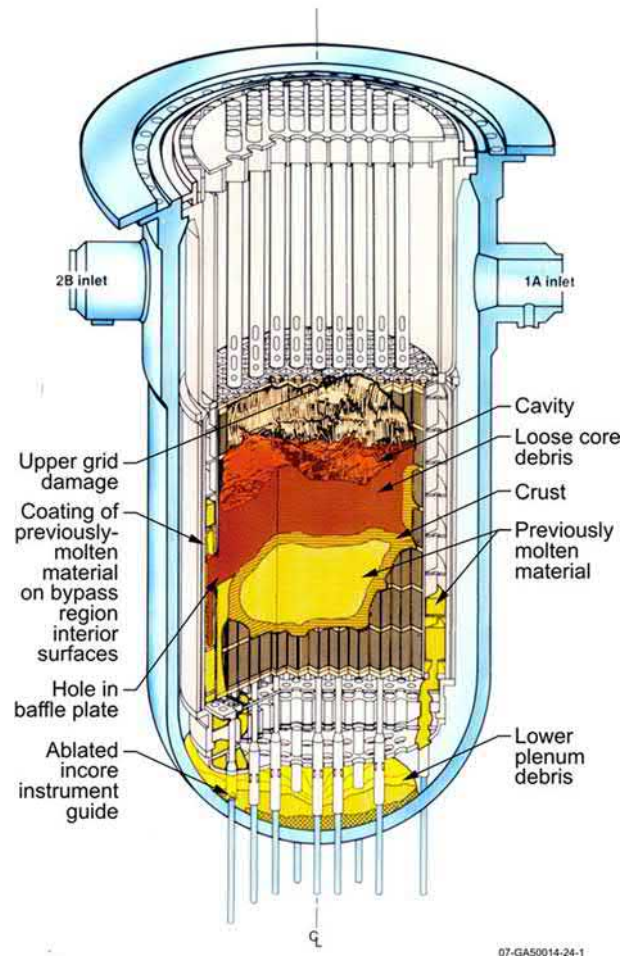
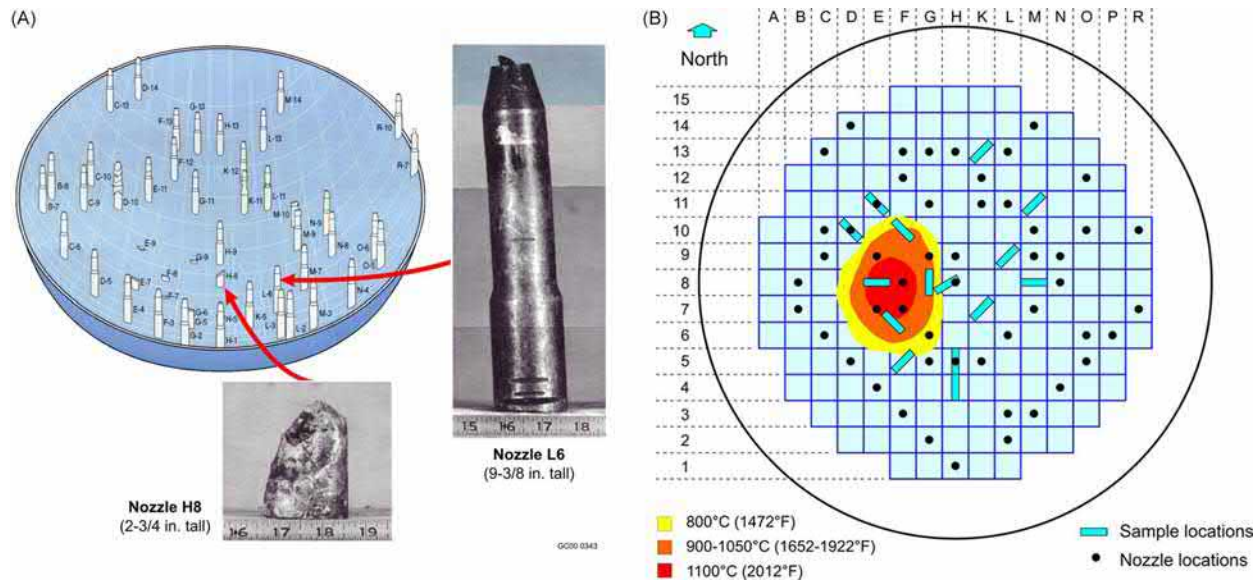


Fig. 10 Endstate of fuel and structures within the reactor vessel. Rempe J, Farmer M, Corradini M, Ott L, Gauntt R, and Powers D (2012) Revisiting insights from three mile island unit 2 post-accident examinations and evaluations in view of the Fukushima Daiichi accident. *Nuclear Science and Engineering* 172: 223–248.



Lessons learned

Numerous lessons were learned from the TMI 2 accident. Actions were taken by the global nuclear community—both by the industry and the regulators—to address these lessons. These lessons included issues discovered, or encountered, from the first seconds of the sequence of events to those that were identified much later through fine scrutiny of the event.

Examples of the first issues to be encountered were those that were highly obvious, such as the obscure and confusing valve status indication for the PORV block valve's (RC-V2's) indicator light location (behind the main control panel, on a waste system indicator panel, not readily visible to the operators), to those less obvious, such as the operator focus to prevent the pressurizer from "going solid." The first, regarding the location of indication lights for the status of the PORV block valve and other similar issues, led to changes in the layout of instrumentation panels and consideration of human factors in control room layouts not only at TMI 2 but throughout the global nuclear industry. Operator actions to throttle ECCS flow were based on prior training that emphasized avoidance of a solid pressurizer. Thorough scrutiny of these actions led to industry-wide training requirements to ensure that the operators understand basic principles of the specific plant design and operation on which they were licensed, combined with engineering measures and procedures to prevent undesirable plant conditions or status.

Careful scrutiny of the first 24 h of the accident highlighted issues regarding the preparedness of plant radiological personnel for unanticipated extreme radiological conditions, the complexity of plant owners' accountabilities for emergency preparedness along with responsibilities of ensuring that local, state and federal agencies and their responders were able to conduct their own specific accountabilities in coordination with the plant operators if a future accident should occur.

The nuclear industry implemented actions to ensure that lessons learned from TMI 2 were identified and addressed, including establishing the Institute of Nuclear Power Operations (INPO), an independent organization dedicated to operational safety in nuclear power plants. Established in 1979, INPO activities to enhance nuclear plant safety and reliability are reflected through key programs: periodic on-site evaluations of each nuclear plant and corporate support organizations, training and accreditation, events analysis and information exchange, and technical assistance (INPO, 2007).

NUREG-0600 (US NRC, 1979) documents results from the inspection effort undertaken by NRC after the TMI 2 accident. In this NUREG, which was published 5 months after the accident, the NRC Office of Inspection and Enforcement identified six areas with inadequacies:

1. Equipment performance (failures and maloperation)
2. Transient and accident analyses
3. Operator training and performance
4. Equipment and system design
5. Information flow, particularly during the early hours of the accident
6. Implementation of emergency planning.

NUREG-0660 (US NRC, 1980a) provided a comprehensive list of actions to address inadequacies for each of these areas. The Commission, however, only approved specific items from NUREG-0660 for implementation, and NUREG-0737 (US NRC, 1980a,b) provided guidance that each plant owner could utilize for implementation. This far-reaching NUREG addressed the higher priority of the lessons-learned from the TMI 2 accident. As discussed above, these lessons issues led to changes in system design, system hardware and instrumentation, operator training, control room layout to address human factors, philosophies for plant management and leadership, and emergency preparedness and planning.

NRC also identified and took actions to address their own inadequacies, including insufficient knowledge about severe accident phenomena, inadequate infrastructure to monitor an accident and to provide timely emergency response actions, insufficient attention to prior plant operating experience, and inadequate monitoring of licensee performance. For example, to address concerns about oversight, the NRC resident inspector program was expanded to provide full-time staff presence at all operating plant sites. The NRC also increased its use of probabilistic risk assessment, and in response to comments raised by the Advisory Committee on Reactor Safeguards, the agency adopted a safety goal—a quantitative objective in terms of acceptable risk to which plants could be assessed (US NRC, 1986). This latter action became increasingly important as it became apparent that some of the newly imposed requirements on licensees might actually hinder safety. Ultimately, a backfit rule was implemented to provide a basis for developing and evaluating new requirements (Ahearn, 1989).

Conclusion

TMI 2's fate was sealed in the first 120 seconds of the accident due to a combination of equipment failure (failure of the PORV to reclose), inadequate instrumentation system design (no indication to alert the operators to the unrecognized "open" PORV for the subsequent 140 minutes), and inadequate operator training. Only during the 'Quick Look' inspection, on July 21, 1982, over 3 years after the initiating events on March 28, 1979 was the extent of damage to the TMI 2 core discovered. The severe accident that followed the initiating events resulted in extensive core uncover, cladding failure, fuel melting, structural degradation, radioisotope release, and more than a decade of clean up, and defueling activities. Despite the serious damage to the reactor, studies have concluded that radiation releases during this event had negligible effects on the physical health of individuals or the environment. As discussed within this chapter, investigations following the accident led to changes that addressed multiple institutional issues

throughout the nuclear power industry. Changes were implemented in control room design, operator training, emergency preparedness, and the regulation of commercial nuclear power plants. Ultimately, the actions taken to address the TMI 2 accident enhanced the safety of the operating fleet.

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The Chernobyl Nuclear Power Plant Unit-4 Accident

Alexander R. Sich, Department of Physics, Kennesaw State University, Marietta, GA, United States

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Glossary

Becquerel (Bq) A unit used to measure radioactivity, which refers to the amount of ionizing radiation released when an element (such as uranium) spontaneously emits energy as a result of the radioactive decay (or disintegration) of an unstable atom. Radioactivity is also the term used to describe the rate at which radioactive material emits radiation, or how many atoms in the material decay (or disintegrate) in a given time period. As such, 1 Bq represents a rate of radioactive decay equal to 1 disintegration per second, and 37 billion (3.7×10^{10}) Bq equals 1 curie (Ci).

China Syndrome A fanciful term—not intended to be taken literally—that describes the fictional result of a nuclear meltdown scenario, whereby reactor components melt through their containment structures and into the underlying earth, “all the way to China.”

Corium A lava-like material created in the core of a nuclear reactor during a meltdown accident; also called fuel containing material (FCM) or lava-like fuel containing material (LFCM).

Curie (Ci) A unit used to measure radioactivity, which refers to the amount of ionizing radiation released when an element (such as uranium) spontaneously emits energy as a result of the radioactive decay (or disintegration) of an unstable atom. Radioactivity is also the term used to describe the rate at which radioactive material emits radiation, or how many atoms in the material decay (or disintegrate) in a given time period. As such, 1 Ci is equal to 37 billion (3.7×10^{10}) disintegrations per second, so 1 Ci also equals 37 billion (3.7×10^{10}) Becquerels (Bq).

Design Basis Accident One category of the postulated accidents used to set criteria and limits for the design and sizing of safety-related systems, structures, and components.

Exclusion Zone The approximately 30-km radius ($\approx 2,600 \text{ km}^2$) area surrounding ChNPP managed (since April 2011) by the State Agency of Ukraine on Exclusion Zone Management (SAUEZM). This Agency is administratively within the State Emergency Service of Ukraine, itself managed by the Ukrainian Ministry of Internal Affairs.

Group of Seven (G7) International intergovernmental economic organization consisting of seven major developed countries: Canada, France, Germany, Italy, Japan, the United Kingdom and the United States, which are the largest International Monetary Fund advanced economies in the world.

International Nuclear and Radiological Event Scale (INES) A tool for communicating the safety significance of nuclear and radiological events to the public. Events are rated at seven levels: Events without safety significance are rated as Below Scale/Level 0. The scale is logarithmic—that is, the severity of an event is about 10 times greater for each increase in level of the scale.

Operating Reactivity Margin (ORM) The extra reactivity that would arise if all control and safety rods were withdrawn; expressed as the number of “equivalent” or “effective” control rods of nominal reactivity worth remaining in the reactor core. This number is a calculated value; it is not equivalent to the number of actual control rods in the reactor core but depends on several variables, such as the power density distribution (the degree of neutron flux where the control rods are positioned which affects how many neutrons the control rods absorb).

Reactivity A measure expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Roentgen (R) A unit of exposure to ionizing radiation. It is the amount of gamma or x-rays required to produce ions resulting in a charge of 0.000258 coulombs/kg of air under standard conditions.

Introduction

At approximately 00:23:40 (Kyiv-time) on April 26, 1986, a safety test led to the most severe accident in the history of commercial nuclear power occurred at Unit 4 of the Vladimir I. Lenin Nuclear Power Station at Chernobyl [Ukr: Chornobyl] (the Chernobyl Nuclear Power Plant or ChNPP). After a brief overview of the ChNPP4 design, this chapter summarizes key events of the accident evolution and Accident Management Actions (AMAs) taken during the initial days and months, including (see Fig. 1) an initial domestic effort to confine radiation releases in a “sarcophagus” (formal name “Object Shelter” or OS) followed by an international effort to provide a more robust building, known as the New Safe Confinement (NSC). Root causes of this event are discussed. In addition, this chapter summarizes measures taken to enhance the safety of nuclear power plants—in particular, RBMKs.



Fig. 1 Photos of ChNPP: (A) Prior to the accident; (B) Original “sarcophagus”; (C) NSC + OS in 2016 (OECD, 2018).

RBMK design and Chernobyl plant overview

The development of nuclear power as a large-scale provider of electrical energy received significant support from the Soviet government. The advent of what would become the “high power reactor – channeled” or RBMK (РБМК: Реактор Большой Мощности – Канальный) was heralded on June 27, 1954 by the startup of the 5 MWe (30 MWt) Obninsk experimental prototype; a graphite-moderated, light-water cooled reactor. The Obninsk reactor was the first of a series of test reactors built during the 1950s and 1960s, leading to the first commercial large-scale RBMK-1000 at Sosnovy Bor (the Leningrad NPP) achieving full-power operation in January 1974.

RBMK design

The RBMK is a heterogeneous, thermal neutron, graphite-moderated, channel-type, boiling-water nuclear reactor with on-line refueling. As originally designed, the light-water coolant flowed upward through 1661 zirconium – 2.5 niobium alloy (Zr-2.5 Nb) pressure tubes (fuel channels) containing slightly-enriched (1.8%–2.0%) UO_2 fuel assemblies within square graphite stacks of blocks (250 mm $\times$ 250 mm of various lengths) acting as the moderator. As shown in Fig. 2, each fuel channel contains two 3.5 m long fuel assemblies stacked one on top of the other providing an “active” fuel length of approximately 7 m. Each assembly consists of 18 Zr-1Nb clad fuel rods distributed in two concentric rings of 6 and 12 rods about a central hollow Zr-1Nb support rod and held in place by stainless steel spacer grids along the assembly length.

The core (see Fig. 3) is arranged in the form of a large vertically-oriented cylinder (equivalent diameter of 11.8 m and height of 7.0 m). The axially central fuel portion is surrounded by an annular graphite reflector (columnar stacks of blocks with equivalent thickness of 0.88 m), and the core is covered on the top and bottom with an additional 0.5-m graphite reflector—making the effective overall dimensions of the graphite stack approximately 13.8 m in diameter and 8.0 m in height. Radial core support is provided by 156 water-cooled support tubes situated at the outer periphery.

The graphite structure included a total of 2488 channels for fuel rods, control rods, and instruments (at the core periphery). In the case of the fuel channels, between the outer wall of the pressure tube and the inner wall of the hollowed-out graphite block are graphite spacer rings of two different diameters fitted alternately to the tube and graphite channel. These provided a path for heat flow from the graphite stack to the pressure tube, while additionally permitting pressure tube expansion. Nominal average operating temperature for the graphite was 600 °C, while the maximum permitted temperature was 750 °C. A mixture of nitrogen and helium gas was circulated within the core between blocks in the graphite stack to prevent oxidation and increase heat transfer to the coolant.

The in-core control system channels were similar to the Zr-2.5Nb clad fuel channels. During operation, control was accomplished using 211 rods with detectors arranged into five functional groups: manual regulation rods (RR), emergency power reduction or SCRAM rods (AZs), overall automatic power regulation rods (OAR), local automatic regulation (LAR), and axial power

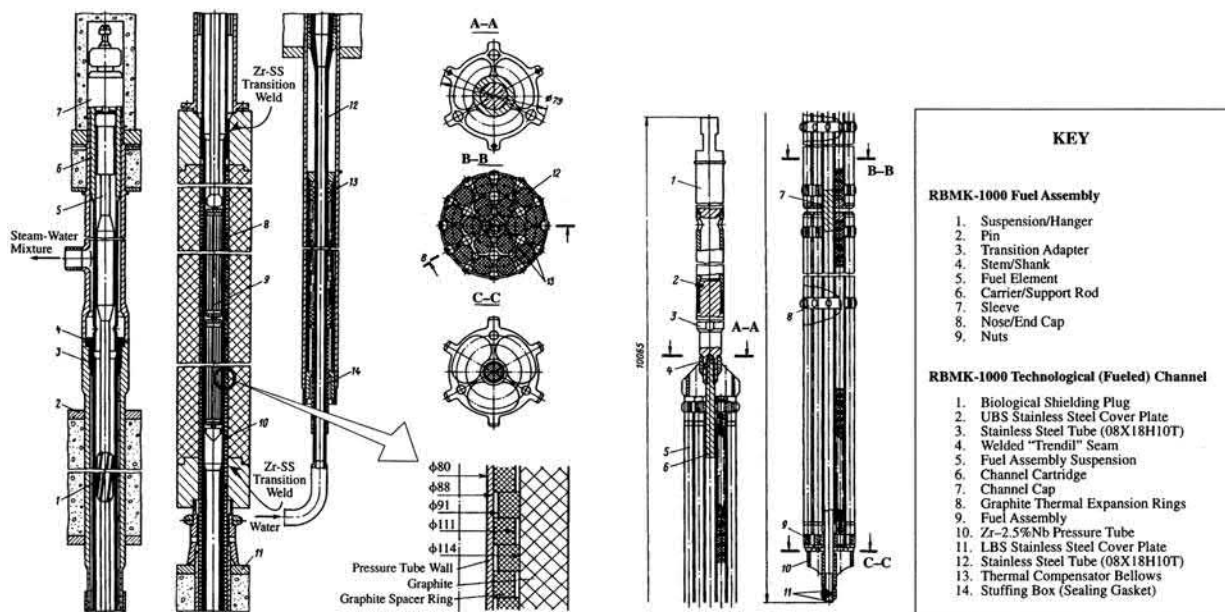


Fig. 2 Cross-sectional views of RBMK Fuel Channel and Assembly (Sich, 1994a).

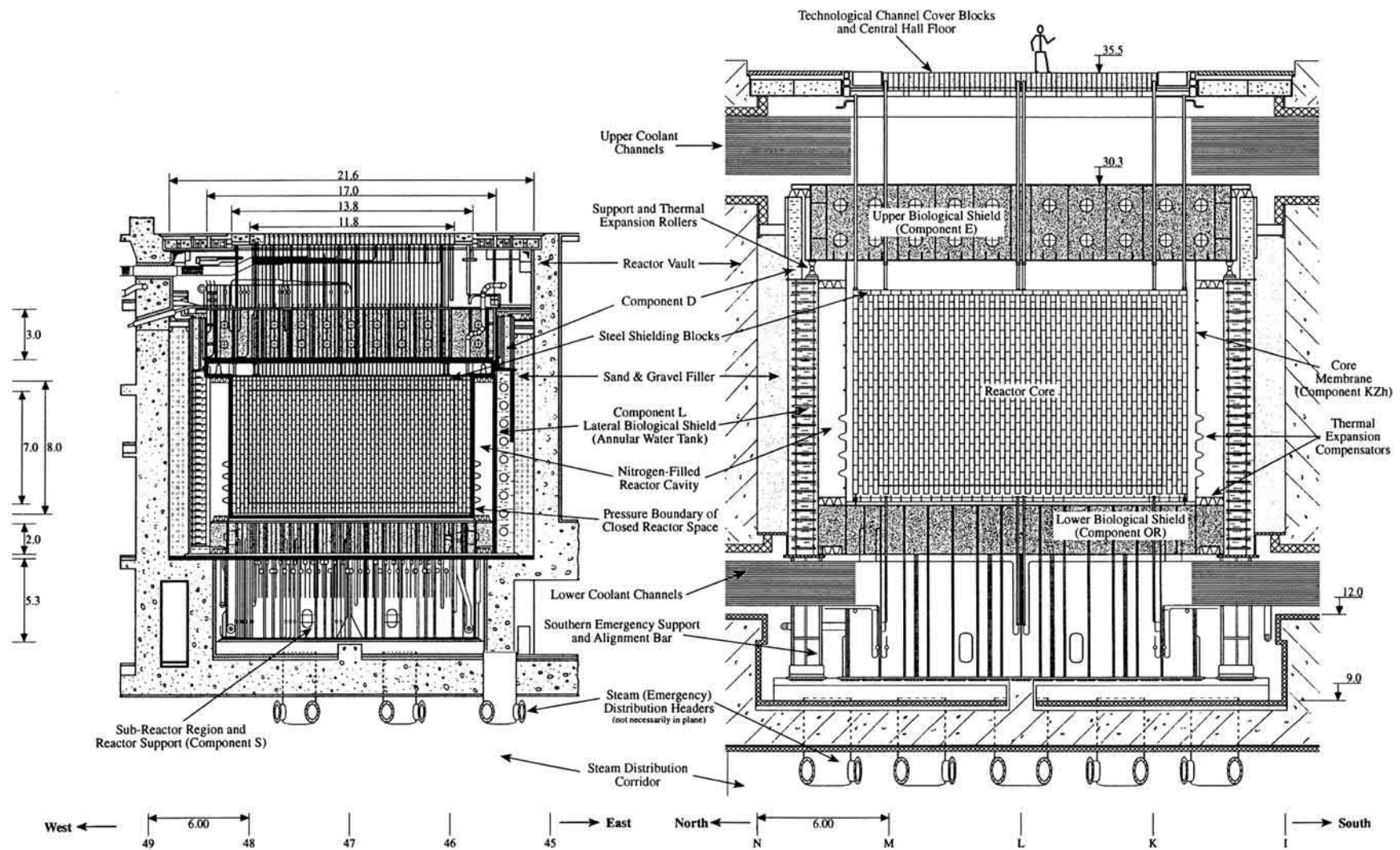


Fig. 3 Cross-sectional views of the ChNPP4 Reactor Vault (Sich, 1994a).

regulation rods (APRs) with shortened absorbers. Except for the rods used for automatic power regulation (e.g., the OARs and LARs), top-inserted control rods (the RRs and AZs) had “followers” or “displacers” connected below the neutron absorber section (for the bottom-inserted APRs, the follower had a displacer as a leader that was located above the neutron absorber section). As a top-inserted control rod is raised out of the core, the follower replaces it—helping to smooth out the local neutron flux. When fully withdrawn, the end of the follower was located 1 m above the bottom of the fuel portion of the core. Hence, in the lowest one meter of the core, control rod followers were too short: they did not extend to the bottom of the core. Those channels were filled with water that acted as a neutron poison relative to the graphite. Hence, if fully withdrawn control rods were reinserted, positive reactivity was also initially inserted into the lowest regions of the core, decreasing the margin to boiling-induced voiding.

As shown in Fig. 4, coolant was circulated through two primary loops with a 14.7% steam two-phase mixture fed from the fuel channels to four steam-separator drums—located in pairs above and on each side of the reactor—then, coolant flowed directly to two 500 MWe turbines (TG7 and TG8 for ChNPP4). The separated water, together with fresh feed water, was returned to the bottom of the core through downcomers by six 4 MWe main recirculation pumps (MCPs). Each MCP increased the coolant pressure by about 2 MPa to an operating pressure of approximately 9 MPa.

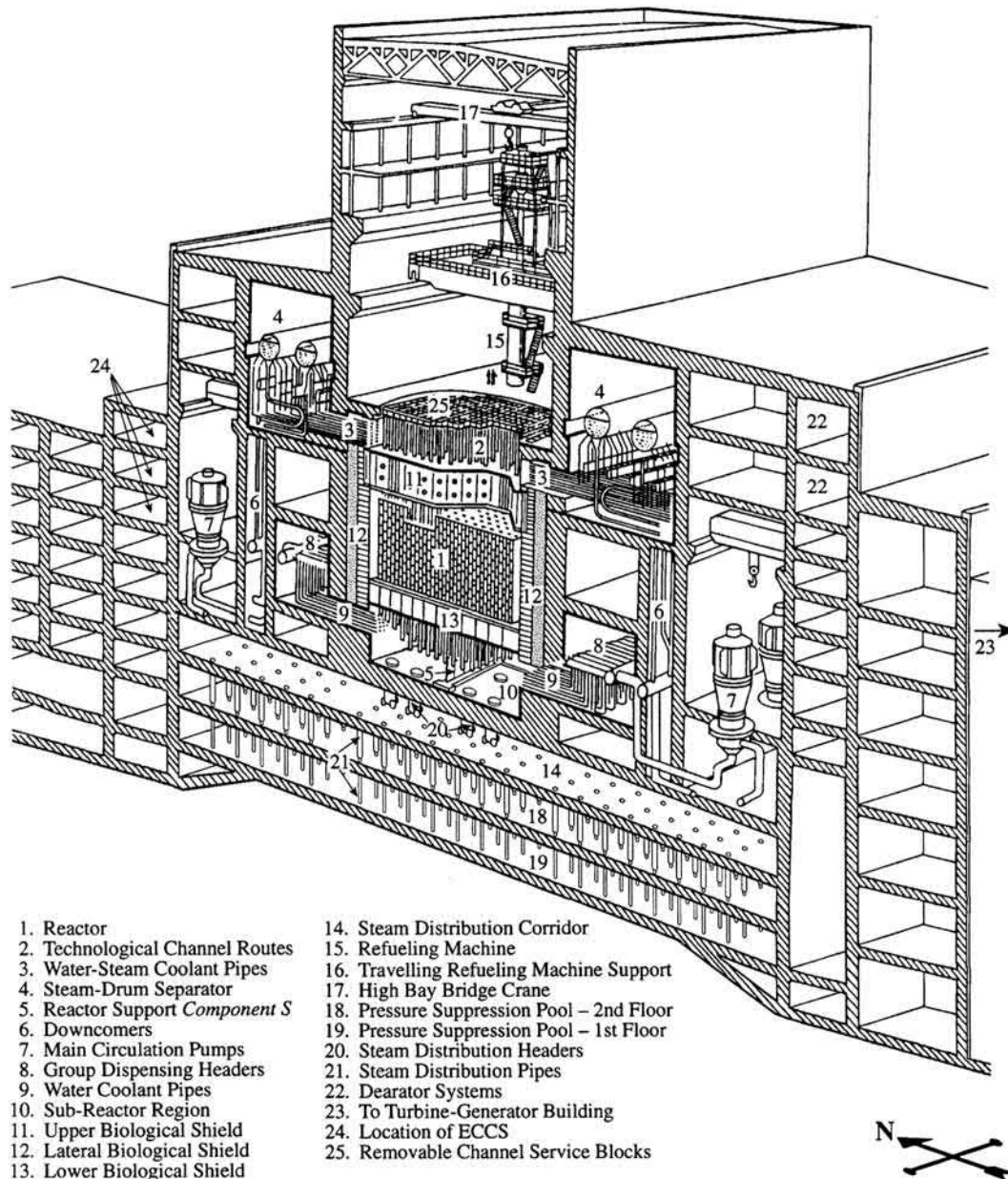


Fig. 4 Cross-sectional view of the RBMK-1000 main building (Sich, 1994b).

Advantages

The RBMK, considered the Soviet Union's "national reactor" (Medvedev, 1991), has several important and rather significant advantages compared to Light Water Reactors (LWRs). Operational advantages of the RBMK include the following:

- Uranium-graphite reactor systems cooled by boiling light water are the most accessible for a country with a moderate economic base and technological infrastructure. Although the RBMK has a more complex structure than a typical pressure vessel-based reactor, similar designs were not without precedent in fossil fuel plants and the absence of complex and expensive steam generators proved to be a distinct advantage.
- Besides not requiring an outer structure surrounding the reactor core (such as a pressure vessel for LWRs or pre-stressed concrete vessels for gas reactors), the graphite channel reactor offers flexibility in reactor design. Because of the lack of restriction on core size, the use of standard structural elements was possible for RBMKs. Its modular construction allowed major reactor components to be fabricated at existing manufacturing plants with relatively simple transport and assembly at the plant site.
- RBMKs use lower enrichment uranium and could be used for the production of U-233 from Th-232.
- The RBMK can be refuelled online without interrupting reactor operations, resulting in a higher capacity factor.
- The RBMK permits superheating of steam and channel-by-channel control of flow rates (implying higher thermodynamic efficiency and reliability). Each channel's integrity and operational parameters could be monitored separately; damaged channels could be replaced under operation.
- The use of small-diameter pressure tubes precluded the large break loss of coolant accidents possible with western-designed LWRs.

Disadvantages

Some RBMK features, which were intended to make it more efficient, actually introduced significant disadvantages. Features identified as design faults include the following:

- The over-moderated core was designed to enhance neutron economy required for low fuel enrichment operation. Because graphite is utilized as a moderator, the water coolant effectively behaves as a slight to moderate neutron poison. Operational experience led to a recognition that if the water voids through boiling ("positive void coefficient" manifested mainly at low-power in "mature" cores with high burnups) or if coolant in control rod channels is displaced by graphite followers (a major design flaw), positive reactivity is introduced (Dyatlov, 1996).
- Although the RBMK can be considered a "modular" system scalable to larger core sizes, this also results in a neutronic loosely-coupled core that is difficult to control—particularly at low power levels when the positive void coefficient was more prevalent. There is a high sensitivity of the neutron field to reactivity disturbances, necessitating a complex, computer-based, control system for stabilizing the distribution of energy generation in the core. Moreover, the large core size gives rise to a potential for local critical masses to form within the core (USNRC, 1987; Sich, 1994a,b).
- The graphite followers attached to the bottom of the control rods were designed to improve neutron economy while in the core and to maximize reactivity change upon control rod insertion. However, the followers and control rods fit closely in the channels—hence their motion was damped significantly by the surrounding water, resulting in a very long insertion time (~20 s) compared to typical insertion times of 2 s [for Pressurized Water Reactors (PWRs)] and 4 s [for Boiling Water Reactors (BWRs)]. In addition, the length of the followers was too short: this left a water gap at the bottom of the core, causing a "positive scram" effect that supplemented activity insertion due to voiding water.
- Besides the complexity of the inlet-outlet piping as the heat-transfer agent of each channel and the large amount of thermal energy accumulated in the metal structures, fuel elements, and graphite stack, RBMKs have a limited capability to accommodate structural effects of temperature transients (stresses and strains)—namely in the zirconium-to-stainless steel transition welds at the inlet and outlet to the fuel channels in the core (US DOE, 1989).
- No emergency or warning signals were provided to generate alarms when key parameters for the reactor reach points requiring emergency protection system actuation. This was a "clear violation" of Article 3.1.8 of NSR-04-74 (Atomizdat (1976); INSAG, 1993; Mosey and Varley, 2006).
- The design basis accident (DBA) for an RBMK-1000 was the near-simultaneous rupture of a maximum of two-to-four fuel channels (of 1661 total). In such a case, the Emergency Core Cooling System (ECCS) and the Accident Localization System (ALS—the emergency pressure suppression system comprised of steam distribution headers and corridors as well as the two-level steam-water bubbler pool) underneath the reactor were designed to withstand the associated over-pressurization of the thin stainless-steel "skin" (so-called Component KZh) that surrounds the core. If more channels fail, it was postulated that the 2500 t-equivalent Upper Biological Shield (UBS) (see Fig. 3) would be lifted upwards—shearing more of the channels and eventually leading to catastrophic failure of the entire system. The Zr-Nb2.5%-to-stainless steel transition welds (diffusion bonding process) at the top and bottom of each of the 1661 fuel channels were considered to be the weakest points in the main coolant circuit and the most likely to fail according to safety analyses (USNRC, 1987).
- Fuel channel and graphite block integrity was also a concern. Due to long-term irradiation over the course of operations, the size and physical properties of these graphite moderator blocks change, and the Zr-2.5Nb pressure tubes suffer from creep, oxidation and hydrogen absorption. Stress corrosion cracking had been observed in the transition welds.

- The lack of an RBMK pressure vessel, while less expensive in terms of construction and maintenance associated with creep, neutron embrittlement, and pressurized thermal shock encountered by other LWRs, is a safety deficiency. In the context of defense-in-depth, the pressure vessel represents a major line of defense against a release of radiation in case of an LWR accident. The RBMK graphite stack (including the core) is located in a sealed cylindrical container formed by a 14.52 m diameter and 9.75 m high steel shell (“Component KZh” or “skin” in Fig. 3) 16 mm thick with close to 4000 penetrations, including primary coolant piping. This container serves as a gas barrier as well as primary structural restraint and support for the graphite stack. It is this shell that, if compromised, will release steam that could raise the UBS and shear coolant channels (IAEA, 1993a,b).
- RBMK reactors have no “containment” structure: there is no “final” robust engineering safety barrier between the reactor and environment. At the time of the accident, RBMKs incorporated a “confinement” pressure-barrier system based on the (then) DBA basis noted above: the rupture of a maximum of two-to-four channels to mitigate against the breach of Component KZh. The ChNPP4 reactor hall (the region above the UBS, with dimensions approximately 24 m wide × 80 m long × 35 m high)—similar to those at other RBMK units—was constructed of reinforced 1.5 m-thick concrete walls for the lower bay and a steel frame roof with precast concrete panel sheathing for the walls of the high bay. This reactor building was only designed to withstand normal building structural loads (USNRC, 1987).

ChNPP

ChNPP4 is located in the eastern part of a large, sparsely forested region called the Belarusian-Ukrainian *Polissia* (woodland) along the banks of the Pryp'yat' (Pripiat) River. Just prior to 1970, Soviet central planners decided this remote region of rural Ukraine, would be suitable for the location of Ukraine's first nuclear power plant. At the start of the plant construction in 1970, the population density was approximately 24 inhabitants per square km, and the population of the entire Chernobyl region was only about 47,000. At the beginning of 1986, the population within the later-defined 30-km Exclusion Zone surrounding the plant was close to 100,000 people, of which 49,000 lived in the newly-built “atomic” city of Prip'yat—located 3 km to the northwest of the plant industrial site.

The first set of reactors at ChNPP, Units 1 and 2, were the third pair of “first generation” RBMK-1000s, (completed in 1972 and 1978); Units 3 and 4 were second-generation reactors (completed in 1977 and 1983); and Units 5 and 6, which were under construction at the time of the accident (85% and 15% completed), were third generation reactors and located approximately 1.5 km southeast of Units 1 through 4. A reservoir, 21.4 km² in area and 5 m deep, along the Prypiat River was built to serve the cooling needs of the first four units, while two cooling towers were built on the site of Units 5 and 6 to serve their needs.

Rated at 3200 MWt (1000 MWe), Unit-4 was “physically started” on November 11, 1983, achieved criticality on November 25, 1983, was connected to the grid on December 21, 1983, and began commercial operation on March 25, 1984. By April 25, 1986 (1 day prior to the accident), ChNPP4 had reached the end of its first operating cycle with a core-average burnup of 10.9 MW · d/kgU for its initial 190.3-t UO₂ fuel load (Borovoi and Sich, 1995). Following a relatively routine reactor safety experiment, a maintenance outage was scheduled for the following day.

The accident

At approximately 00:23:40 (Kyiv-time) on April 26, 1986, after 865 calendar days and 715 effective full-power days in operation, the most severe accident in the history of commercial nuclear power occurred at Unit-4—characterized as a Level 7 Major Accident per the International Nuclear Event Scale (IAEA, 2008). The accident significantly damaged Unit 4 (see Fig. 5). It led to several explosions in which significant quantities of reactor materials (in the form of large chunks of graphite, pressure tubes, and fuel assemblies) were hurled up to 150 m radially from the core onto the buildings and grounds of the plant industrial site. These materials started more than 30 fires on the roofs of the Unit 3 and 4 Auxiliary Building, the deaerator roof of Unit-3 and the “machine hall” above turbine generator TG7. Much of the roofing of these buildings was pitched with tar (bitumen) that exacerbated the fires. Within ChNPP4, other fires started due to the presence of large quantities of lubricating oil pipes and containers damaged as a result of the explosions. The immediate task was to address the fires. By 02:15, fires on the roof of the Machine Hall were contained, by 02:35 fires on the reactor and auxiliary buildings were put out, and by 06:35 all fires in the vicinity had been extinguished (Medvedev, 1991).

Sequence of events

Precise details regarding some of the events that occurred during this accident may never be known. Sich (1994a) reviews available references to develop a detailed chronology: significant events during the first few hours of the accident include the following:

- At 01:06 on 25 April, operators started reducing power in preparation for the TG8 coastdown test.
- At 03:47, reactor thermal power is 1600 MWt.
- At 07:10 on 25 April, the ORM was 13.2 control rods equivalent (in violation of the ORM limit of 15 equivalent rods that requires reactor shutdown (INSAG, 1993a,b)]—likely due to operators compensating for xenon buildup from the descent to half



Fig. 5 A mid-August 1986 view of the remains of the Chernobyl NPP Unit-4 from the south-southeast at an elevation of 250 m. Alexander Sich, personal collection, unsourced photograph courtesy of Valentin Ivanovich Obodzinskiy of the Kurchatov Institute.

power). More violations (procedural overrides and control rod manipulations) continued throughout the day. These violations occurred because the Kyiv load dispatcher imposed an additional nine-hour delay for starting the test when the reactor was at slightly less than half power (1600 MWt).

- At 14:00, the ECCS was disconnected as part of the test (to avoid possible actuation that would require repeating the test).
- At 23:10, the load dispatcher permitted continuation of power reduction. Because of the imposed delay in power reduction, a new shift—which was not briefed on details of the safety test—took over operations.
- At 00:25 on 26 April, the power dropped to 650 MWt in direct violation of normal safety test procedures. The test was designed for a power level of 700 MWt, so the reactor should have been shutdown immediately. This is considered the official “start” of the accident.
- At 00:28, the local automatic control rod positioning system, which monitors and maintains local power in 12 core zones, is switched off; but the operating staff also makes an error by switching to fully automatic control without entering the hold power at the level required for the safety test. As a result, the reactor power drops to below 30 MWt and xenon poisoning “forces” operators, at multiple times, to lower the ORM (drastically attempting to “save” the experiment). No fewer than six times were trip interlocks overridden (including ORMs) to permit the experiment to continue.
- By 01:03, the power level was stabilized at approximately 200 MWt—well below the 700 MWt required for the test. Note, however, the risk of running the experiment at low power was not de-scribed in the operating instructions, so operators proceeded with the test.
- From 01:07 to 01:23 the ORM hovered between five and eight equivalent rods in a difficult attempt to compensate for xenon buildup in a neutronically-decoupled core by (a) balancing feedwater flow rate (the coolant was barely below saturation temperature and pressure—just below voiding—and actually exceeded, by four times, the balanced flow rate), (b) exceeding vibrations limits due to cavitation concerns, (c) hovering steam drum separator levels near the emergency level (operators blocked the emergency trip signals related to low steam pressure followed by manual replenishment of water in the steam drums), and (d) operators removing some of the manual control rods entirely. In short, the core was quasi-stabilized in such a state that any insertion of positive reactivity would lead to a loss of control of the reactor.
- At 01:23:04, the TG8 safety experiment started with the command “Oscillograph on!” The TG8 stop valve was closed leading to an increase in pressure (hence a decrease in reactivity). At the same time, the flow rate began to fall due to MCPs being powered by spinning-down TG8. As a result, voiding (steam) increased—inserting reactivity.
- At 01:23:40 the AZ-5 scram signal was initiated because power was rapidly increasing (at that point, already 320 MWt) and strong and frequent shocks are felt. As control rods slowly descend into the core, the flawed follower design inserted reactivity at the bottom of the core. By 01:23:45, the calculated power peak reaches approximately 470 times nominal full power or 1500 GWt. By 01:23:48–49 the core was destroyed.

Accident management actions

In response to this accident, which had not been previously hypothesized, actions were taken during the initial days and months. To this date, activities continue to monitor conditions at the site.

Actions during the active phase

The “Active Phase” of the Chernobyl accident (see Fig. 6) is defined as the time period between the following:

- the initial destruction of the core, caused by the violent interactions of fragmented fuel as it came into direct contact with the primary coolant resulting in a massive steam explosion, to
- a sharp drop (factor of $\sim 10^3$ – 10^4) in radionuclide releases to the environment about 9.5 days later.

During the morning of 26 April 1986, a Government Commission was formed and charged with coordinating accident “liquidation” (mitigation) tasks, along with the mobilization of resources necessary to carry out such tasks (Medvedev, 1991). Initially, Serhiy Vasylovych Shirokov (Head of the Division of Nuclear Power Stations in the Ukrainian Ministry of Energy and Electrification), who was assigned coordinator of mitigation efforts in Kyiv, indicated that it was very unclear what had happened—mainly because Anatoliy Diatlov (Deputy Chief Engineer of Units 3 and 4) and Viktor Briukhanov (Plant Director) refused to believe what had occurred. Although some plant personnel risked their lives to climb to near the roof to observe the core, they were either afraid to describe what they witnessed or were not believed (Read, 1993). The first official recognition of what occurred was around 15:00 on April 26 (Haynes and Bojcun, 1988). An attempt to cool the reactor with water seemed to fail, although some station engineers still believed the core was being at least partly cooled by water during most of the first day. It was not until 22:00 on April 26 (more than 20 h after the explosions), when a specialist from NIKIET (the RBMK design institute) flew over the plant and confirmed Unit 4 was destroyed. It is likely that the extent of the damage was not fully recognized or understood by members of the Governmental Commission until sometime in the evening following the NIKIET specialist report.

During the Active Phase, the Soviets attempted a number of rapid-response AMAs to minimize radionuclide releases to the environment—primarily focused upon (a) the burning core and its potential impacts on surrounding structures, and (b) heavily-contaminated land in the immediate area of the plant. INSAG-1 (1986) noted these mitigation efforts:

... were, generally, quite successful... [the] dumping [of] materials into the reactor well..., supplying nitrogen to bring down the temperature in the core space and to reduce the oxygen concentration, and the construction of a flat heat exchanger beneath the foundations of the reactor building, stabilized the situation at an early stage (about nine days after the initiating power surge) of the accident.

Indeed, the reality of the situation was markedly different from the Soviet account presented in Vienna to the IAEA in August 1986. The AMAs included the following efforts (see Fig. 6):

- Approximately 1250 helicopter sorties to “bomb” or smother the burning core with 5020 t of materials. Sorties continued through the end of May 1986 primarily aimed at suppressing dust from highly contaminated areas: over 1800 total sorties were flown involving approximately 600 pilots and crews dumping 16,640 t of the following materials:
 - i. boron carbide—to prevent recriticality;
 - ii. dolomite—to generate CO_2 for a smothering gas blanket and heat dissipation;

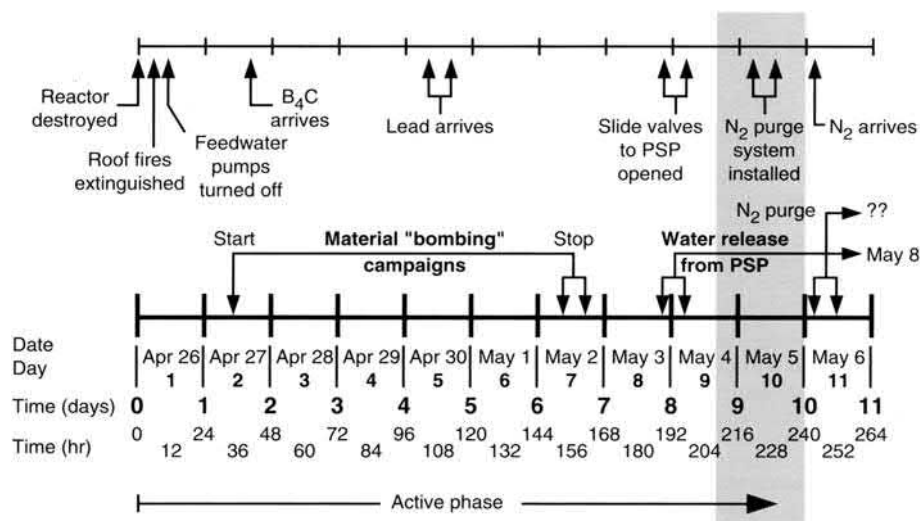


Fig. 6 Summary of AMAs during the active phase of the ChNPP4 accident (Sich, 1994b).

- iii. sand and clay—to quench the fire and provide a filtration layer to inhibit radioisotope releases;
- iv. lead—to absorb heat, eventually cooling to help seal the core (note that vaporized lead would have led to another ecological disaster).
- Nitrogen purge gas into the reactor through coolant channels in an attempt to cool the core and limit oxidation;
- Water release from the ALS pressure-suppression pool to mitigate steam formation from potential core-water interactions;
- Construction of a sub-foundation flat-bed heat exchanger below ChNPP4 to mitigate potential core contact with the soil and ground water;
- A large 8.4-km long 30-m deep below-grade vertical concrete shield around the entire plant (dubbed “Casa Grande”) to limit contamination of ground water flowing into the Prypiat River from surface contamination and to limit any potential “China Syndrome” effects;
- Cutting down and plowing under massive amounts of contaminated woodlands (mostly conifers) located immediately to the southwest of the plant.

At the August 1986 IAEA Meeting of Experts, the official Soviet account stressed the AMAs successfully contained the burning core and eventually stopped radioactive releases into the environment:

...the reactor shaft was covered by a layer of loose mass which intensely absorbed aerosol particles... [and that] the problem of reducing fuel heat-up was solved. To reduce temperature and oxygen concentration, nitrogen from a compressor station was sent into the space under the reactor shaft... The lower slab of the reactor shaft [LBS] had been preserved and fuel was localized mainly (roughly 96%) in the reactor shaft and in compartments of the steam water and lower steam service lines.

Subsequent investigations (Sich, 1994b) revealed intervention measures were either ineffective or not implemented as portrayed by the Soviets:

- Very little, if any, of the 5020 t of materials dropped from helicopters made it into the reactor shaft, confirmed by radiochemical analyses of “frozen” corium or fuel containing material (FCM) in the lower regions of the reactor building (see Fig. 7). Therefore, the core was exposed to ambient conditions—readily releasing radioactive contaminants to the environment (Begichev et al., 1990).
- The gaseous nitrogen purge of the core was initiated after the Active Phase: the first tankers of N_2 arrived at the plant about 01:00 on May 6, 1986, coinciding with the start of nitrogen pumping. This was almost exactly 10 days following the accident (after the end of the Active Phase). The effort was soon abandoned as “a waste of time.”
- The Soviets were concerned the initial 5000 t of material dropped from helicopters could compromise the integrity of structures immediately below the core—possibly forcing the core itself into the 3200 m³ (nominal water volume) ALS pressure-suppression pool as an “assisted China Syndrome.” Core-water interactions would generate tremendous amounts of steam—increasing contamination releases to the environment. The lower regions of the reactor building were known to be flooded with water from

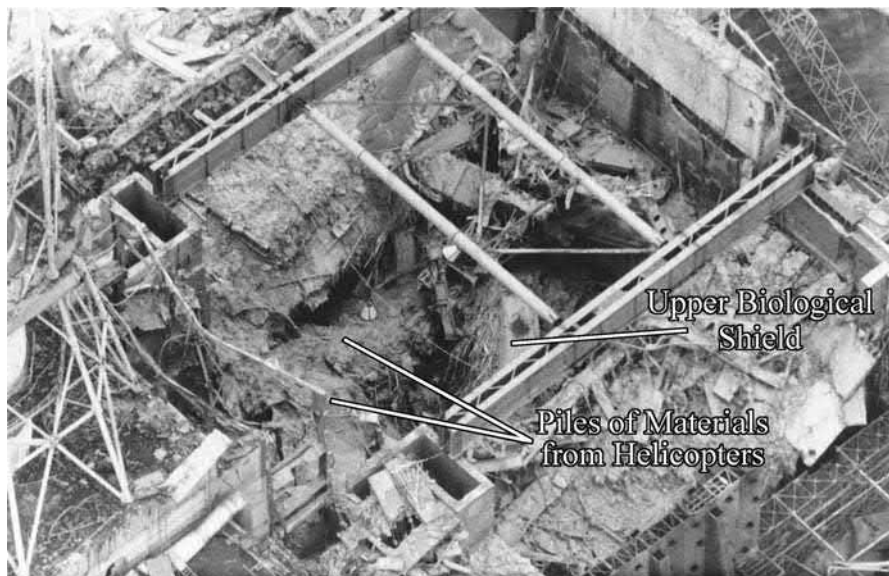


Fig. 7 View above destroyed Unit-4 from the northeast showing the UBS and a gaping hole downward to the reactor shaft. Also visible are piles of materials thrown from helicopters attempting to smother the “red glow.” Adapted from Alexander Sich, personal collection, unsourced photograph courtesy of Valentin Ivanovich Obodzinskiy of the Kurchatov Institute.

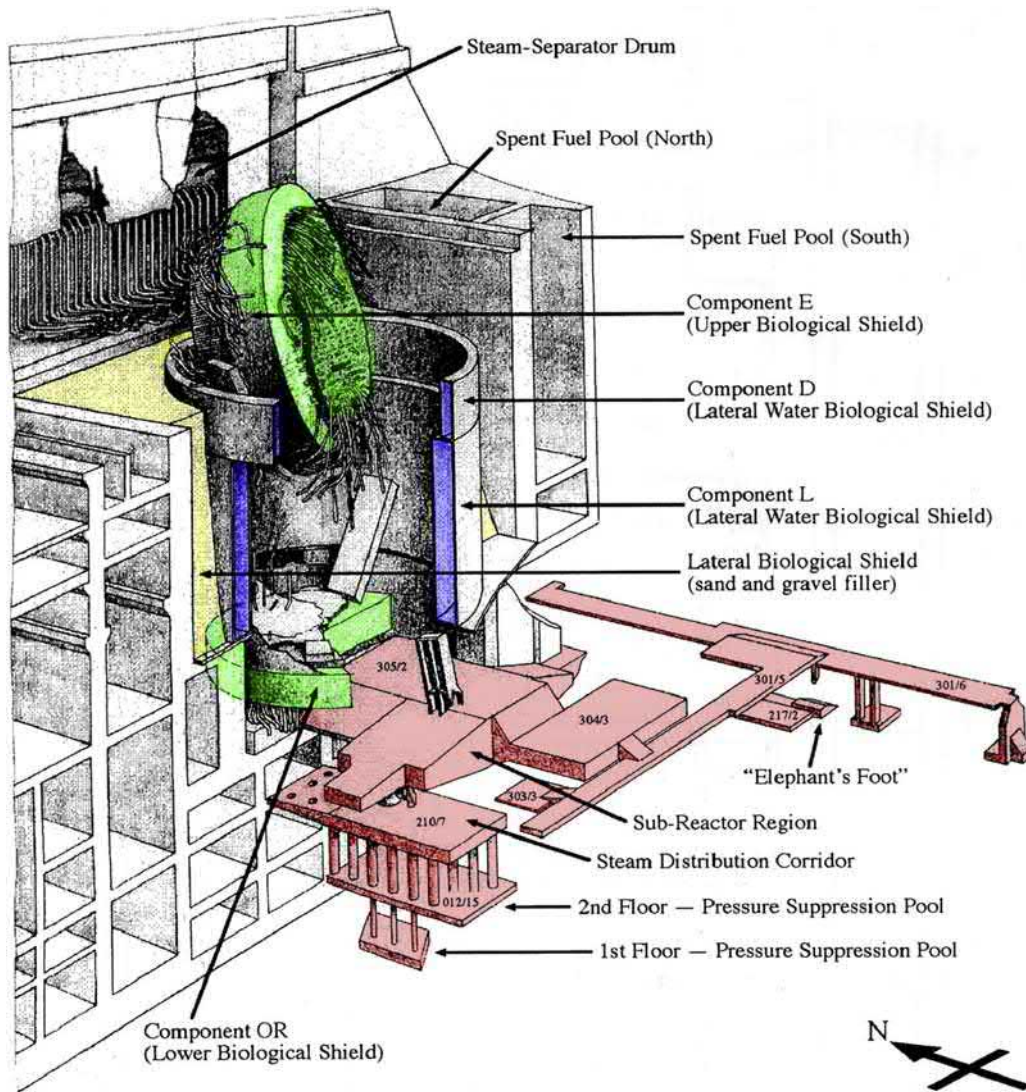


Fig. 8 Sectional View of ChNPP-4 showing possible corium flow paths (red). Virtually no fuel or graphite is found in the reactor shaft region. Approximately one-quarter pie-shaped section of the LBS is missing (Borvoi and Sich, 1995).

the primary system and from water pumped into core in response to the accident. The release of the water (estimated to be $20,000 \text{ m}^3$) was completed on 8 May, or 23 days after the end of the Active Phase. Releasing the water did little to nothing to mitigate releases.

- An over one-quarter pie-shaped section of the two-meter-thick, 17 m diameter, 1200 t LBS (see Fig. 8) was completely eaten away by a molten admixture of a major portion of the fuel ($135 \pm 27 \text{ t}$ or roughly 71% of the initial fuel load), the LBS, and surrounding structures flowed into the lower regions of the reactor building ALS. Only a small portion of the solidifying corium contacted the remaining water to form mounds of corium pumice.
- The corium admixture did relatively little damage to components of the ALS bubbler pool. By the end of the Active Phase, the corium admixture had broken through the LBS—this increased the potential for it to solidify (which decreased future radiation releases) because of several factors: (a) the flow extended to over 41 m from the remains of the LBS—greatly increasing surface area for cooling, (b) decay heat had fallen roughly 10–15 times since the accident (to about 50 kWt/tU), (c) freezing point depression (mixture and chemical complexification) resulted from the uptake of various materials (in particular stainless steel and serpentine) by the admixture.
- The first design version of a sub-foundation flat-bed heat exchanger, whose construction rationale was to avoid a “China Syndrome,” foresaw pumping liquid nitrogen into an excavated cavern roughly $30 \text{ m} \times 30 \text{ m} \times 2.5 \text{ m}$ in contact with the foundation. The design was later altered to employ water as the working fluid. Construction was completed on June 28, 1986 or almost 8 weeks after the end of the Active Phase. Subsequent evaluations found that the heat exchanger played no role in mitigating the accident consequences, but it did expose many workers to very high levels of radiation. The project was abandoned, and the area sealed.

- Only 6.1 km of Case Grande's proposed 8.4 km design was built. Severe surface contamination in the area of the "Red Forest," through which the underground wall was foreseen to be constructed, delayed construction... and the work was eventually abandoned. Unfortunately, the shape of the completed portion of the wall and its location between the plant and the Prypiat River effectively made it function like an underground dam for ground water naturally flowing toward the river. As a result, ground water levels rose significantly—eventually reaching the foundation of the turbine generator building.

In conclusion, little to no direct human interventions aided "natural" processes during the Active Phase: the accident stopped itself.

The sarcophagus

The "Sarcophagus" (formally called the "Object Shelter") (Fig. 1(b)) was commissioned on November 30, 1986—6 months following the accident. It was described by the Soviets as a "concrete cube" because, it was claimed, between 300,000 and 410,000 m³ of concrete was used for its construction. However, taking the cube root of these numbers yields the equivalent lengths of sides of a solid cube of concrete to be between 67 and 74 m. This is larger than the actual sarcophagus, which, in fact, is mostly empty space (Borovoi and Sich, 1995). According to structural drawings, the amount of concrete used was about 161,000 m³. Even so, much of this concrete leaked through holes in the reactor building onto the grounds of the station or was used to cover the ground to shield workers (Sich, 1996a,b). As noted in Sich (2011), "... the sarcophagus's structural integrity is questionable. Better characterized as a 'steel tent' hastily erected upon ruins rather than a 'concrete encasement,' the sarcophagus was never meant to be hermetically sealed."

The sheer size of the structure (roughly 85 m long × 140 m wide × 75 m high, constructed upon the ruins of the destroyed reactor building) and the extremely hazardous working conditions (necessitating the use of unique remote-construction methods in high radiation fields) plus a short time span of 6 months make it a unique accident clean-up and isolation achievement. After considering 18 different project proposals, a design was chosen to take advantage of the reactor building's foundation and remaining superstructure—significantly reducing costs and construction time. There were also significant disadvantages:

- structural integrity of the remaining portions of the reactor building was unknown;
- remote methods for pouring concrete did not always permit observation and review of the work;
- high radiation fields made welding key structural components impossible.

As a result, two essential deficiencies were understood to potentially threaten longer-term stability of the Sarcophagus: the unknown strengths of massive buttresses that formed the upper portions of the structure and the openings and breaks in the walls and roof (estimated to have an area of 1200 m²). In 1987–88, projects to strengthen damaged internal support structures were undertaken to prevent a potential collapse that could further weaken the Sarcophagus. Structural monitoring throughout the 1990s registered no noticeable long-term movements of internal support members, even though the area experienced an earthquake measuring 4.0 on the Richter scale in May 1990. In 1998, the US Department of Energy, in collaboration with plant personnel, successfully completed strengthening the 75 m-tall ventilation stack above Units 3 and 4, which was deemed to be "in [an] emergency condition" (Gore et al., 1998). This was an important short-term stabilization project (that supported the Shelter Implementation Plan) to prevent collapse of the stack onto the Sarcophagus prior to the start of construction of the New Safe Confinement plus Object Shelter (NSC + SO) in 2010. On February 12, 2013, a 600 m² portion of the roof of the turbine-generator building failed near the Sarcophagus due to a truss weakness—resulting in a portion of roof panels installed in 1988 to collapse. Relatively little radioactivity was released, but it necessitated short term evacuation of area workers (IAEA, 2013).

As managed by the Shelter Implementation Plan for the Safety Account of the European Bank for Reconstruction and Development (EBRD), the NSC + SO (162 m long × 257 m wide × 108 m high, 36,000 t mass, lattice construction of tubular steel members supported by two longitudinal concrete beams) (see Fig. 1C) was completed in 2018 to confine the contaminated remains of ChPP4 for the next 100 years—allowing for at least a partial demolition of the original sarcophagus, including localization, characterization, and managed removal of the core of the destroyed reactor.

Post-accident surveys, analyses, international reaction

Disposition of the nuclear fuel within the sarcophagus

Apart from the melt-through of a section of the LBS, the lower regions of the reactor building which form the ALS suffered little if any damage from their interactions with the roughly 135 t (about 71%) of the melted core's initial fuel load (see Figs. 8 and 9). Striking photographs of corium stalactites and "frozen waterfall" (see Fig. 10) as well as results of radiochemical analyses, support the notion that a "China Syndrome" scenario is virtually impossible. The corium lost much of its ability to interact with surrounding structures and quickly solidified—greatly diminishing releases and marking the end of the Active Phase due to the following (Sich, 1996a,b; Borovoi and Sich, 1995):

- Decay heat dropped to about 50 kW/t U after 9 days, and the dilution of the corium by the uptake of surrounding materials further lowered its specific heat generation.
- The chemistry of the corium admixture was strongly affected by the continued uptake of surrounding materials—especially contributions from the LBS's stainless steel and serpentine (a hydrous magnesium silicate)—which likely drove the corium's solidus temperature higher while it formed various stratified layers.

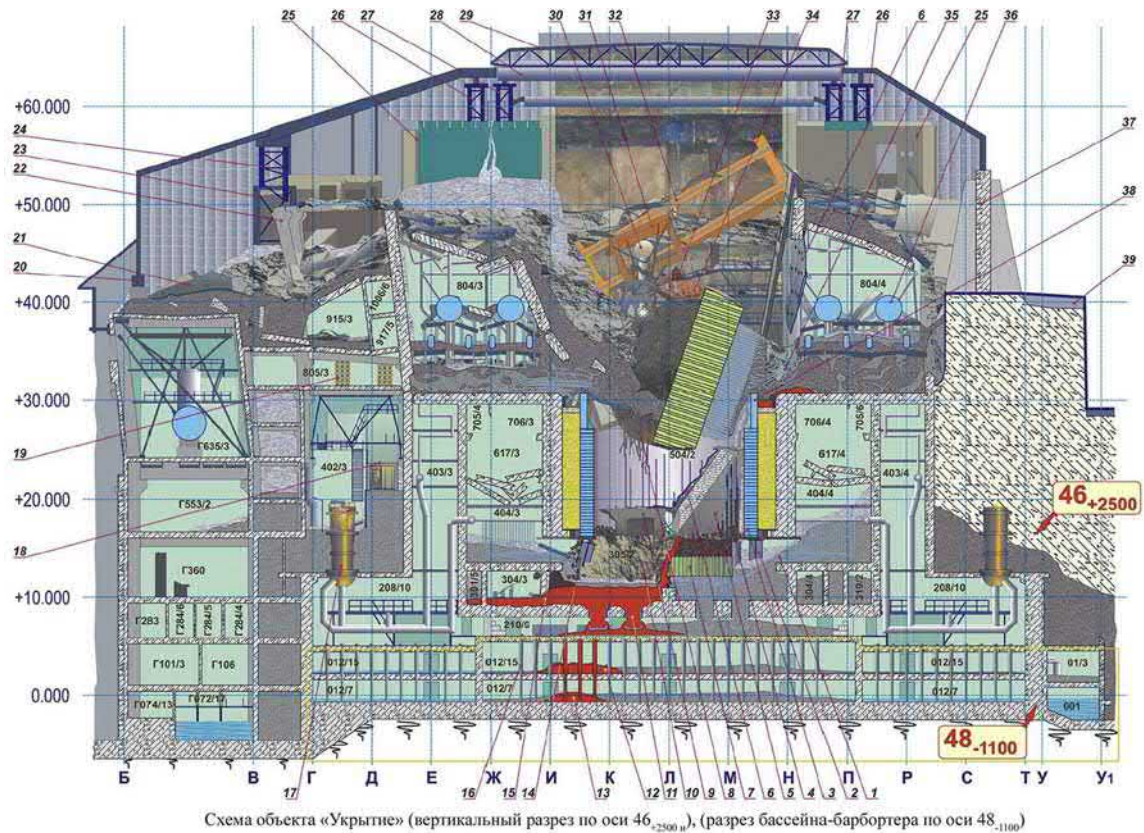


Fig. 9 Cross-sectional view of the Sarcophagus showing conditions of major components and locations of relocated FCMs (in red). Courtesy of V. Krasnov, ISPNPP.

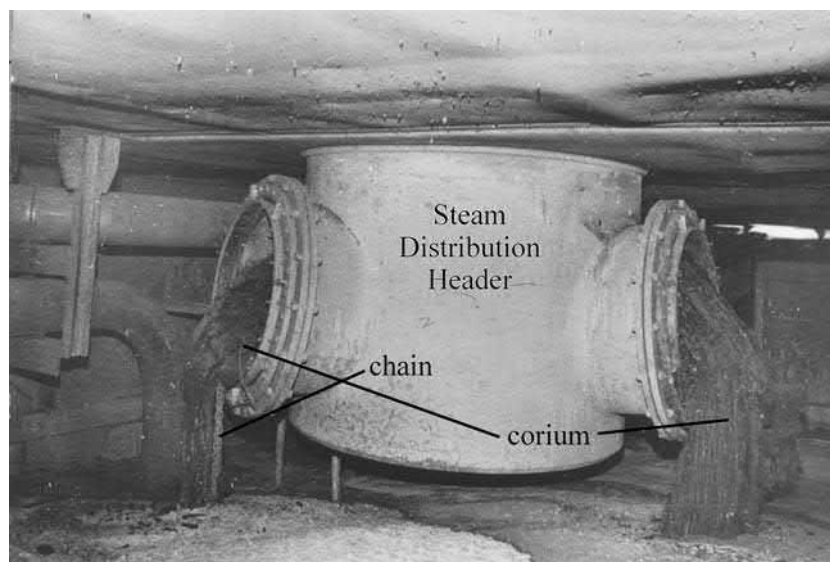


Fig. 10 Steam distribution header below Room 305/2 (sub-reactor region) showing solidified corium with little damage to the header. Below this level are two floors containing the ALS pool. Adapted from Sich AR (1994b) *The Chernobyl accident revisited, part I: Accident management action. Nuclear Safety* 35(1); personal collection of Alexander Sich, unsourced photograph courtesy of Aleksandr Borovoi of the Kurchatov Institute.

- Rapid spreading over a distance up to 40 m from the epicenter just below the reactor shaft occurred immediately after the corium melted through the LBS—facilitating surface cooling and further suppressing heat generation effects.
- The formation of solidified, ceramic-like corium as well as formation of a pumice-like material when the corium contacted water. This indicates a relatively rapid cooling process ensued once the corium penetrated the LBS and flowed into the lower regions of the reactor building.

- There is little indication of blast damage below the nominal level of the LBS. Moreover, channel coolant piping which enters the core from below the LBS seems to have been pulled smoothly downward around its periphery. Rather than an explosive downward-driven motion, this appears to indicate a slow creep-induced collapse of the cruciform reactor support structure occurred due to heat generated from the approximately 75 t of solidified corium that settled around this support once a portion of the LBS melted through.
- The reactor shaft contains concrete slabs identified as sections of wall from above the Central Hall floor near the steam drum separators. The approximately 2000 tonne UBS currently rests on one of these slabs at 15 degrees from vertical with its center at 5 m above nominal (see Fig. 8). This suggests the UBS must have been hurled at least several meters into the air by the violent release of steam before landing in the mouth of the reactor shaft—pinning the wall slab beneath itself.
- The infamous “red glow” spotted and subsequently bombed by helicopter crews was actually located on the floor of the Central Hall about 15 m from the reactor shaft. This glow was not, as previously believed, located within the reactor shaft. To this day, investigations have not been able to confirm whether this is fuel-graphite rubble ejected from the core.
- Results of radiochemical analyses of close to 200 samples of corium indicate the retention of Cs-137 is higher than expected given the highly oxidizing conditions—about 35% of the initial inventory at the time of the accident.
- It is estimated (see Table 1) approximately 95% of the original 190.3 t UO₂ fuel charge at the time of the accident remains within the Sarcophagus: roughly 75% below the reactor in various forms of corium, while the balance is scattered throughout the upper portions of the structure as pulverized parts of the core and dust, including under the “Pioneer Wall” erected on the north side of the original reactor building. (Borovoi and Sich, 1995). The corium manifests in various forms—roughly characterized as black or brown pumice-like and slag-like ceramic materials as well as metallic admixtures of fuel and surrounding structural material.

Table 1 ChNPP4 fuel distribution and mass balance.

<i>Location and description</i>	<i>Fuel mass subtotal, tonne</i>	<i>Fuel mass total, tonne</i>
Upper levels of the reactor building (level ≥ 35.5 m)		37.7 ± 5 +?
• Central hall (estimate of core fragments)	15	
• Central hall (hot particles and dust)	5 ± 2	
• Central hall (under pile of materials thrown from helicopters)	?	
• Core fragments attached to UBS	5 ± 3	
• Steam separator drum areas	2 ± 1	
• Other areas above reactor	5	
• Fragments within the cascade wall of the sarcophagus	1.9	
• Turbogenerator hall (estimate of core fragments)	1.9	
• Fuel fragments beyond reactor building but under concrete	1.9	
4th floor—subreactor region (level 9.0 m)		95.7 ± 26
• Core fragments scattered upon the LBS	0.5 ± 0.2	
• Subreactor region (room 305/2)	75 ± 25	
• Room 304/3	14 ± 5	
• Room 303/3	0.2 ± 0.1	
• Corridor 301/5	3 ± 1	
• Corridor 301/6	3 ± 1	
3rd floor—steam distribution corridor (level 6.0 m)		27 ± 8
• Steam distribution headers (3 units)	1.8 ± 0.6	
• Steam distribution corridor (rooms 210/5, 210/6, 210/7)	23 ± 8	
• Corridor 217/2 (stalactites)	0.2 ± 0.1	
• Corridor 217/2 (“Elephant’s Foot”)	2 ± 0.8	
1st and 2nd floors—pressure suppression (levels 0.0 and 3.0 m)		12.5 ± 3
• PSP—(2nd floor)	(11 ± 3)	
– Open “piles”	6 ± 1	
– Piles covered by concrete or in piping	5 ± 3	
• PSP—(1st floor)	(1.5 ± 0.5)	
– Open piles	0.9 ± 0.2	
– Piles covered by concrete or in piping	0.6 ± 0.5	
Total fuel contained within reactor building		172.9 ± 28 (?)
Fuel released beyond the Chernobyl station ($3.5 \pm 0.5\%$)		6.7 ± 1
Total core load at time of accident		190.3
Fuel unaccounted for		>10.7 (?)

? reflects the additional uncertainty.

Courtesy of Borovoi A and Sich AR (1995) The Chernobyl accident revisited, part II: The state of the nuclear fuel located within the Chernobyl sarcophagus. *Nuclear Safety* 36 (1).

Reconstruction of events during the active phase

The August 1986 Soviet Report to the IAEA (USSR, 1986) concluded that AMAs to smother the core with materials dumped from helicopters were effective. Four primary elements support an accident progression hypothesis quite different from that presented by the Soviets to the IAEA in August 1986:

- Reassessment of effectiveness of AMAs—indicates that AMAs were ineffective: none played significant roles in the Active Phase;
- Radiochemical analyses of FCMs—indicate no material other than proximate structures were taken up in the corium; little to no carbon compounds have been observed in the FCM—in particular, uranium carbide—seemingly precluding carburization as an activity transport mechanism;
- Radiochemical analyses of releases—release patterns vary significantly between volatiles, refractory metals, and refractory oxides; higher than expected amounts of entrained Cs-137 in corium indicate filtration-limited releases and varying transport mechanisms/paths;
- Photographic images and thermal surveys indicate that some fraction of the corium remains in the sub-reactor region and upper regions of Sarcophagus with little damage to surrounding structures by corium. There is no blast damage evident below the reactor shaft.

First proposed by Sich (1995), a peer-reviewed and un-challenged accident progression scenario, which considers the above evidence, is outlined in Fig. 11.

Immediate damage, contamination, evaluations, and health effects

The reactor core was destroyed. Approximately 135 ± 30 t of fuel (~ 70 –85% of the initial load) eventually melted and flowed downward into the reactor building's lower regions and pressure suppression pool—the latter which had been almost completely drained of water by the time of its breach. About 3.5% of the fuel (~ 6.7 t) was blown upwards out of the reactor core “shaft” and released into the atmosphere where aerosol-sized particles were eventually carried over the northern hemisphere. An estimated 25–30 t of fuel, in the form of “dust” or large, broken pressure tubes imbedded in graphite, were scattered through the upper levels of the destroyed reactor building and up to 150 m from the reactor building—the latter subsequently either “scraped” toward the reactor building and interred in the “sarcophagus” (either pushed or carried into the building by hand under extreme radiological conditions) or covered by meters of concrete. The balance of fuel (~ 11 t), most of which is thought to be located inside the reactor building and sarcophagus, has yet to be positively located (see Table 1 above).

The reactor building itself, especially above the floor of the Central Hall, suffered extensive damage and resulted in the complete destruction of the roof and upper building structures (high bay)—reducing the height of the building from an initial 71.5 m to an average 46 m (Sharovarov, 1990). Other major components, including the ECCS, deaerator system, four steam drum separators, eight MCPs, and primary piping were heavily damaged or destroyed. The explosion's pressure wave significantly displaced those reactor building support columns not completely destroyed, including the 75 m-tall main ventilation stack—compromising the building's structure integrity and seriously complicating the subsequent construction of the “sarcophagus.”

Hot core materials (fuel, graphite, and other components) expelled during the explosions started fires in and around the destroyed unit. Particularly heavy damage was suffered by the roof of the “machine hall” covering ChNPP4's twin 500 MWe turbine generators (TG7 and TG8), which collapsed in many places—causing further side-spread damage to equipment below. Large quantities of flammable seal and bearing cooling oil were located at intervals along the external walls and below the main turbine hall level along the entire 750 m-long “machine hall.”

The initial explosions and fires wafted radioactive materials high into the air, much of which were carried away in the form of gases and aerosols by normal air currents. During the Active Phase of the accident, the Soviets officially estimated a total of 50 MCi of noble gases (100% assumed released) and a total of about 50 MCi of other radionuclides (rough 3.5% of core inventory) were released to the environment (Aleksakhin et al., 1991). The currently accepted total radioactivity releases from Chernobyl are 14,000 PBq or 378 MCi (Devell et al., 1995; IAEA, 2006; Sweet, 1999; Devell and Johansson, 1994)—almost four times that reported by the Soviets. (USSR State Committee on the Utilization of Atomic Energy, 1986). For comparison, approximately 17 Ci of Iodine and 2 MCi of noble gases were released from Three Mile Island Unit-2, roughly 1 MCi total activity released from Windscale, and about 1020 PBq (27.6 MCi) from Fukushima-Daiichi (see Mizokami and Rempe, 2021—updated based on WANO).

As a consequence of these releases, within a few months after the accident over 4 km² of pine forest immediately adjacent to ChNPP reddened, died, were plowed under, covered with fresh soil, and sprayed with dust-suppressing solution (Asmolov et al., 1987). The 1438 km² of land within the 30-km Exclusion Zone was declared unfit for any agricultural activity in perpetuity. Of this, 864 km² are located in Belarus and 574 km² in Ukraine. In addition, 69 km² of tillage and pastureland are contaminated to over 80 Ci/km² of Cs-137, 601 km² between 40 and 80 Ci/km², and 16,670 km² to over 5 Ci/km². For perspective, the Soviet lower-bound standard for land considered “exclusionary and designated for evacuation” is 15 Ci/km² of Cs-137—an equivalent planar mass density of about 1/6th gm/km².

Thirty-six and one-half hours after the accident, evacuation of the 49,614 residents of Prypiat began and was completed within 3 h. By this time, the radiation levels outdoors in the city ranged between 400 and 1500 milliroentgen/h (mR/h). Starting between 2

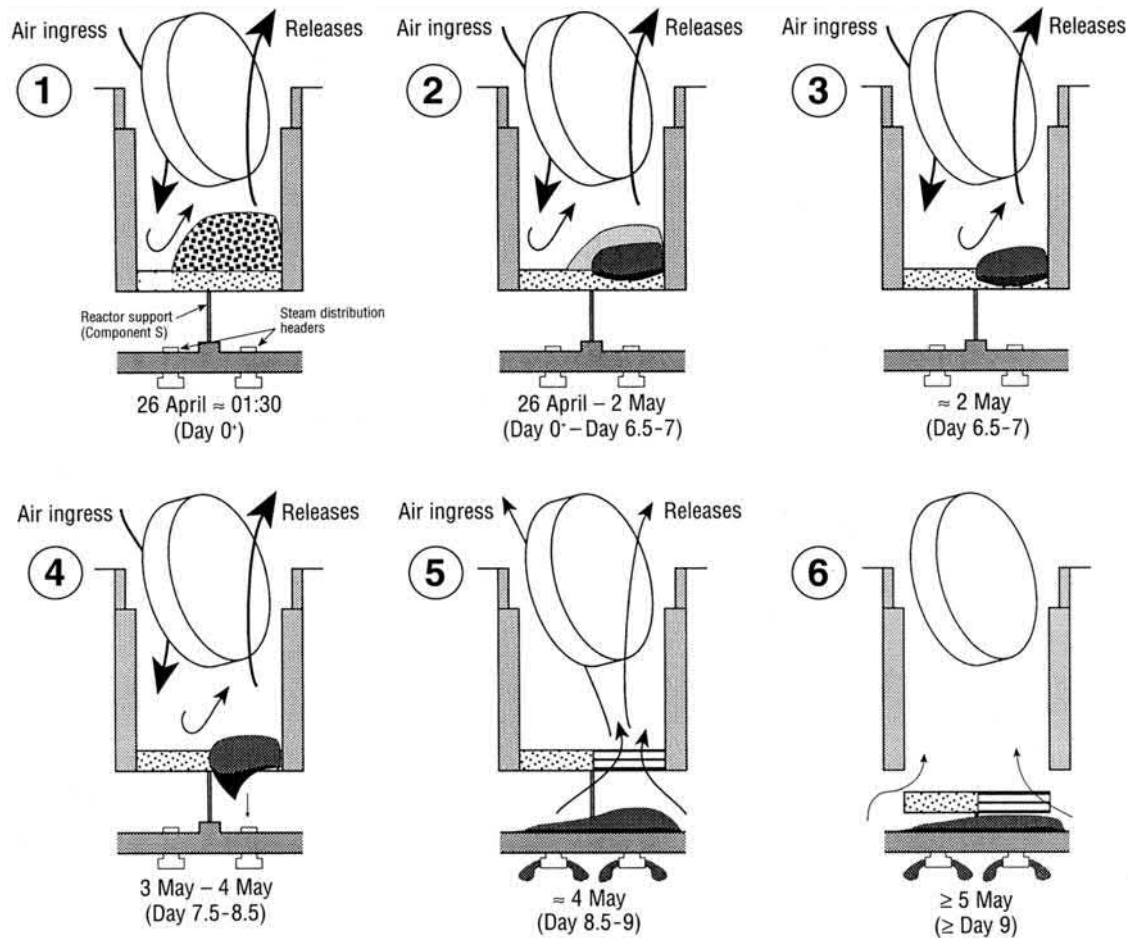


Fig. 11 Hypothesized accident progression scenario during the active phase (looking north).

Accident clock	Time and/or date	Event(s) description
① 0	~01:26:45 26 April	Explosions destroy core; fuel tubes severed at transition joints; UBS thrown several meters upwards
0 ⁺	~ 01:27 26 April	Pulverized fuel assemblies, control rods, graphite, other core components and debris settle on top of the LBS
② 0 ⁺ → 6.5-7.0	26 April-2 May	Denser, melted, fuel layer forms below graphite and other debris; UO ₂ likely oxidized to U ₃ O ₈ ; fuel corium interacts with LBS steel shell and serpentine filler; upper layer of debris/graphite systematically degrade—forming particulates that serve as mechanical transport path
③ 6.5 → 7.0	02 May	Upper layers of debris and graphite melt or otherwise degrade until corium below exposed—increasing release effectiveness for radionuclides; still unclear what happened to the graphite which does not readily oxidize and is not significantly observed as part of FCMs
④ 7.5 → 8.5	3-4 May	Corium melts through LBS and flows downward into the ALS region—spreading out with declined decay heat and rise in corium solidus—enhancing solidification
⑤ 8.5 → 9.0	4 May	End of Active Phase: releases drop dramatically; corium solidifies in mid-flow; little damage to ALS structures
⑥ >9.0	>5 May	Remaining portion of LBS slowly descends 3.85 m as steel core support structure yields under ~800-900 t load and high temperature-induced creep

Adapted from Sich AR (1995) The Chernobyl accident revisited, part III: Chernobyl source term release dynamics and reconstruction of events during the active phase. *Nuclear Safety* 36 (2): 1-24.

and 5 May (6.5–9.5 days after the accident) and extending through August 16, 1986, 90,784 people were evacuated from 75 population centers in Ukraine. By the end of 1986, almost 25,000 people from 107 population centers in Belarus were evacuated. Evacuation of residents from significantly contaminated areas in Russia took up to 3 years, affecting 6805 people in 44 population centers. By the end of 1991 (the time of the collapse of the Soviet Union) approximately 163,000 people had been resettled from Ukraine, Belarus, and Russia, with significant and very expensive evacuation and medical-support efforts continuing for years afterwards (Sivintseva and Kachalova, 1992).

On the night of the accident two Chernobyl plant personnel died, and 28 more persons (operators and firemen) died within a few weeks as a result of acute radiation poisoning—this out of a total of 237 persons originally diagnosed with acute radiation sickness onsite, including subsequent clean-up efforts (to which later 124 more cases were confirmed).

Due to several factors, including the lack of an adequate radiological baseline and the “scattering” of approximately 650,000 workers engaged in accident mitigation efforts (including construction of the sarcophagus), retrospective dosimetry and estimates of deaths attributed to the accident are difficult. Nonetheless, tremendous work has been completed since the accident to assess health effects. UNSCEAR (2012) notes that apart from an increase in thyroid cancers (especially in children), “there is no evidence of a major public health impact attributable to radiation exposure 20 years after the accident.”

Causes and reactions

The accident occurred during a safety experiment (test) to determine adequacy of on-site power during a loss of offsite power. The safety test investigated whether power from spinning-down turbine-generators was sufficient to run the MCPs until the emergency diesel generators could power up. It was an electrical engineering test—one not intended to impact directly any nuclear aspects of the reactor. Similar tests had been attempted at ChNPP3, but power from the spinning-down turbines decreased too rapidly. New voltage regulator designs were to be tested at ChNPP4.

In their January 1993 report, the IAEA’s International Nuclear Safety Advisory Group (INSAG) identified the root technical cause of the accident as the RBMK design and firmly shifted blame away from the operators to the designers (INSAG, 1993). The Group’s original report (INSAG, 1986) based on partial and slanted material provided by Soviet officials in Vienna in August 1986, laid much of the blame on the operators. The design faults identified in the 1993 INSAG report were associated with RBMK features intended to make it a more efficient reactor. The most serious indictment was the revelation that identified design problems had been recognized by the Soviets before the accident.

As discussed below, a confluence of root causes (or precursors), technological and non-technological, led to the destruction of ChNPP4.

Technological causes and corrective actions

Following the accident, a number of contributing technological factors were evaluated to determine root causes. Most broadly these root causes may be classified as (a) inadequate or flawed design, and (b) excessive reliance on administrative controls to fulfill critical safety functions (see Rempe (2021) for discussion of critical safety functions). Indeed, design deficiencies and inadequate training, combined with inadequate safety evaluations and multiple operator errors during a turbine-generator safety test, placed the reactor into an unstable state that led to the accident.

Four design deficiencies, which were identified as technological causes of the accident, are listed below with corrective actions eventually taken by the Soviets to resolve them:

- Positive reactivity insertion for control rods scrambled from an initially fully withdrawn position. To improve the moderation capability of the reactor core and increase effectiveness of control rods, the design incorporated a graphite extension (or “follower”) at the bottom of the rods. However, the “follower” was too short: with rods fully withdrawn, the bottom of the rod channels within the core were filled with water—which in an RBMK acts as a poison. Therefore, the initial displacement of the water with graphite added reactivity to the core (Kurchatov Institute, 1991).

Short-term corrective actions: The graphite followers were lengthened.

- Positive void coefficient is manifested under conditions of relatively high burnup and at relatively low reactor power. The design of RBMKs results in certain low-power operating ranges manifesting over-moderation. In other words, the graphite-moderated RBMKs are “over-poisoned”; the water is more significant as a poison than as a moderator. If water density decreases (e.g., due to boiling-induced voiding), the effect of water poisoning decreases—effectively increasing reactivity.

Short-term corrective actions:

- the number of fixed control rods in the core was increased by 80–90 rods;
- the number of bottom-inserted control rods was increased by 8–16 rods;
- fuel enrichment was increased to 2.4% to compensate for control rod modifications and introduction of additional absorbers (this also increased the potential for higher local peaking factors);
- automatic calculation of operational reactivity margin (ORM) was increased from every 30 min to every 5 min with a control room display provided;

- the limit for the calculated ORM was increased from 26–30 to 43–48 rods.
- changes to identify (using improved analysis methods) and to avoid (using administrative controls and design changes) operating conditions with a large positive steam (void) coefficient of reactivity.
- Slow control rod insertion times. As noted above, typical control rod full-insertion times for LWRs are less than 4 s whereas the insertion time for control rods at ChNPP4 was nearly 20 s.

Short-term corrective actions: (a) the speeds for drive mechanisms were increased to reduce RBMK control rod insertion times to 12 s; (b) a fast-acting automatic emergency protection system which drops 21–24 control rods into the core within 2.5 s was installed.

- Insufficient automatic backup shutdown systems. During the course of the turbine-generator safety test, operators were able to override interlocks several times—including switching off the emergency shutdown system. This imposed a manual rapid-response burden directly upon the operators in their attempt to shut down the runaway power increase.

Short-term corrective actions: in addition to the installation of a fast-acting automatic emergency protection system noted above, double key interlocks were installed to limit operators' abilities to disable emergency protection systems.

Other improvements implemented at RBMKs include the following:

- installation of automated control room displays for disabled protection systems;
- incorporation of interlocks that automatically limit the number of control rods that can be manually withdrawn from the core;
- addition of redundant power supplies for drivers and control circuits of the control rod management system;
- integration of automatic emergency protection shutdown systems under conditions of core coolant low-flow or low-pressure situations;
- improved operational procedures were instituted, including administrative controls to prohibit disconnection of reactor protection systems.

Non-technological precursors

Important societal (historical, ideological, political, cultural, institutional) precursors contributed—not as root or direct causes, but to establishing conditions for an increased likelihood for the accident. Soviet attitudes toward scientific advancement, technological modernization (“nuclear positivism”) and safety were driven by years of ideological commitments, political expediency, and national pride. From its very inception, the Soviet Union paid inordinate attention to the expansion of power production to support the material and technical base of communism as envisioned by Lenin. Compounding this were other factors, including siting considerations near population centers with uneven distribution of water resources, transportation costs, short-cutting environmental concerns, and hard currency generation (primarily through the sale of oil) (Feshbach and Friendly Jr., 1992).

As emphasized by Bernstein and Kushment (1987), “Large-scale engineering systems are more than a collection of technological instruments; they are a reflection of their societies and of the management practices and bureaucratic procedures... Chernobyl and the Challenger explosion had a surprising symmetry in some respects.” The similarities identified included the following:

- the momentum of long-term successful programs;
- military involvement in design objectives and control;
- eventual transition from “mission oriented” to “ordinary repetitiveness”;
- economies of scale spurred “routineness” in shuttle missions and RBMK mass production;
- for the Soviets, adherence to party loyalty through adherence to bureaucratic procedures took precedence over technical ability; for the U.S., the flow of information was compartmentalized.

The Soviets were inheritors of a technocratic progressivism characteristic of the economic policies of capitalist societies prior to when limits on growth and environmental consciousness are imposed. In part, this emerged naturally from Soviet ideological commitments to and a boundless faith in technology and the beneficial effects of technological developments upon mankind's welfare. Soviet nuclear energy policy, in particular, was driven by a desire to demonstrate conspicuous technological achievements in a large-scale program the Soviet Union regarded as urgent to its superpower status. A thriving nuclear power program was seen as a means of enhancing the prestige of the country and providing visible evidence of the superiority of socialism. As emphasized by Josephson (1986), the Soviets developed a cavalier attitude toward safety: nuclear power plants were sited relatively close to population centers for convenience in power transmission, the need for redundant core cooling systems was disregarded, and costly containment structures were eliminated.

An interesting aspect of Soviet society was the unofficial workplace (institutional) practice called *shturn*, or the periodic, frantic attempt to meet plan fulfillment quotas. The most important plant quotas were those imposed by yearly and five-year demands for production. However, there were also quarterly and monthly quotas that became especially important just before State holidays such as the May 1st Socialist Labor Day celebration and vacation period, which informally welcomed the summer period and was second only to the October Revolution celebrations. The importance of this *shturn* period for meeting production quotas cannot be over-emphasized. Indeed, it is quite likely the pressure was great to produce electricity and successfully complete the safety test in order to obtain holiday bonuses. This, in turn, would have influenced workers to inadvertently relax safety practices—a contributing factor to the accident (Medvedev, 1991). Indeed, ChNPP4 operators were anxious to complete the safety experiment because they would lose their bonuses and would not have another chance for over a year... or not at all because another plant

might be requested to complete it. For this reason, operators were not pleased when, after the reactor was dropped to a little less than one-half power (1600 MWt per normal experimental procedure) on 25 April, an additional nine-hour delay was imposed by the grid dispatcher. This was done to support last-minute production quotas for heavy industry prior to May Day. The only thing that seemed to matter was to meet or exceed quotas so that bonuses could be had. The demand imposed by the grid dispatcher was not so much an order by the Ministry of Energy and Electrification as it was haranguing by factory managers to squeeze out as much power as possible prior to the holiday.

The sequence of events, when considered comprehensively, provides a deeply disturbing account of multiple violations driven by political, economic, and personal expediencies... with safety considerations missing in action. There was a lack of emphasis on safety culture, which as noted by Vargo (2000), "refers not only to the defense in depth design but also to the personal responsibility of all of the staff involved in safety-related activities. Implementation of a safety culture means, among other things, that personnel training should place primary emphasis on understanding reasons behind safety practices as well as the safety consequences for failure to adequately comply with personal obligations. Safety culture presumes an overall psychological bias toward safety, as established by organization managers involved in NPP construction and operation."

International reaction: Support for safety upgrades and push for closures

The ChNPP4 accident raised significant concerns among many countries and international organizations regarding the safety of all Soviet-designed reactors (USNRC, 1989; GAO, 2000; IAEA, 1993a,b; PNNL, 1997). These concerns included inadequate means for fire-protection, substandard materials and poor-quality construction, and inadequate separation and redundancy of safety systems. Significant generic RBMK safety issues have been addressed with international assistance based upon safety reviews led by the IAEA, yet skepticism remains. Safety upgrade projects have been sponsored by individual countries (United States, United Kingdom, Canada, Finland, Sweden, Japan, etc.) and international organizations such as the IAEA, the Nuclear Safety Account of the EBRD, the European Union, and the World Association of Nuclear Operators (WANO). Safety upgrade areas included (a) core monitoring and control, (b) instrumentation and control, (c) pressure boundary integrity, (d) accident analysis, (e) safety and support systems, (f) fire protection, (g) operational safety.

A real concern was that another accident at a Soviet and post-Soviet nuclear facility would affect the West directly. First, a major accident would have the potential to undermine the stability of new democratically elected governments, which would be a blow to international efforts aimed at strengthening open markets and free trade. Second, Western investments and aid to former Soviet countries could be placed at risk with potential repercussions on international economies. Third, it would require billions in international assistance to support recovery efforts. And, fourth, such an event could negatively impact the continued viability of nuclear power throughout the world.

A keen interest in developing and strengthening independent nuclear regulatory oversight in former Soviet countries also arose following the accident. Very significant post-Soviet economic-transition difficulties eroded the ability of already weak regulatory bodies to provide adequate safety oversight. In particular, western experts strongly agreed that, among other Soviet-designed reactors, RBMKs fell well below accepted international safety standards and that they could not be upgraded. In particular, there was concern about the lack of a containment structure, inadequate safety systems, insufficient safety backup equipment, unreliable controls on reactor operations, and deficient systems for emergency reactor core cooling.

Soviet-designed reactors were built from a perspective that emphasized power production over safety and assumed timely human interventions would prevent accidents. In contrast, western designs and operations stressed safety over production. This spurred the development of highly automated safety and shutdown systems with minimal reliance on operator involvement. Moreover, because of operational and design priorities, Soviet designers, builders, operators, and regulators did not accept the wisdom of adhering to best international safety practices.

Also, "pride of ownership" on the part of the Russian Federation (expertise in both the design and operation of Soviet-designed reactors was primarily located in Russia during the Soviet period) complicated the situation for newly independent countries emerging from their Soviet past. These countries had to quickly establish their own technical and nuclear regulatory infrastructures. Furthermore, relations between these countries and the Russian Federation deteriorated—exacerbating issues such as the following:

- the difficulty in obtaining replacement parts resulted in makeshift arrangements
- payments to nuclear utilities for electricity production were largely carried out through in-kind markets, sometimes delayed, and often insufficient to cover operational costs—not to mention safety improvement costs
- salaries for nuclear power plant operators were not competitive with other jobs, and salary payments often came late
- regulators earned even less, making the hiring and retention of qualified specialists very difficult.

Indeed, while Chernobyl was the focus, the West took upon itself a mission to assist in reducing risks at Soviet-designed nuclear power plants and to assist host countries in developing a self-sustaining nuclear safety improvement program (operation and regulation) eventually capable of reaching internationally accepted best safety practices without encouraging long-term operation of most risky reactors. The West strongly promoted the importance of safety above all other priorities—to be reflected in the regulation, operation, and professional attitudes of the designers, operators, and regulators of nuclear facilities. The West also pledged to support host countries in the development and strengthening of infrastructure, expertise, and resources in order to participate actively, effectively, and responsibly within the international nuclear power community. That is, relationship-building and enhancement of trust animated western efforts to support and strengthen the technical area goals of operational safety, training, maintenance, safety systems, safety evaluation, as well as legal and regulatory capabilities.

Post-accident shutdown and decommissioning activities

At the time of the accident, 17 RBMKs were operating in Lithuania, Russia, and Ukraine. Since then, the remaining ChNPP Units 1–3 (Ukraine), Ignalina Units 1–2 (Lithuania), and Leningrad Units 1 & 2 (Russian Federation) were closed permanently. Construction on eight units and plans for one unit were cancelled. The Russian Federation currently operates 9 RBMK units at the Smolensk, Leningrad, and Kursk plants with the last unit (Smolensk-3) due to shutdown in 2050 (WNA, 2019a,b).

Following the accident, the situation in and around the ChNPP region was significantly affected. The initial phase involved managing the consequences of the accident, including evacuating the neighboring population. This was followed by a series of daunting tasks, which included restoring power production at the remaining units, cleaning areas where workers had to continue their activities, collecting and storing contaminated wastes spread over the Exclusion Zone, preventing major leakages of contaminants into the water system, encasing the remains of ChNPP4 in a “sarcophagus” designated as “Object Shelter,” (OS) and constructing a new city (Slavutych) to house workers and families evacuated from Prypiat.

Construction of Unit 5 and 6 was abandoned after the accident. During the political changes in the former Soviet Union, the Ukrainian Parliament passed a 02 September 1990 moratorium on the construction of all nuclear power units in Ukraine as well as deciding to eventually close ChNPP. After a major fire in the turbine-generator building in October 1991, Unit-2 was closed and the date for the shutdown of the remaining Units 1 and 3 was fixed by Parliament for the end of 1993. However, the dire economic situation experienced by the country in transitioning from Soviet rule led the Parliament in October 1993 to reverse its original closure decision. Then President of Ukraine, Leonid Kravchuk, also issued a decree in February 1994 requesting restart of Unit-2.

Closing Chernobyl’s remaining Units 1–2 and Ignalina’s Units 1–2 required the combined efforts and support from the Group of Seven (G7) countries, the European Union, and Ukraine, which were implemented through several international programs. The EBRD administers a number of funds under the general rubric of nuclear safety—including the Nuclear Safety Account (NSA) and the Chernobyl Shelter Fund (CSF) which deliver direct, project-based support to Ukraine. In 1995 the NSA initiated nuclear safety projects at Chornobyl—focusing on nuclear safety upgrades and security projects at Unit-3, followed in 1998 by projects to construct two important pre-decommissioning facilities: an Interim Spent Fuel Facility (ISF-2) and the Liquid Radioactive Waste Treatment Plant (LRTP). The CSF was established in 1997 to support the Chernobyl Shelter Implementation Plan (SIP) by assisting Ukraine in transforming the existing sarcophagus into an environmentally safe system known as the NSC + SO (EBRD, 2020).

The West’s early-closure policy developed after the 1992 G7 Munich Summit was directed at the most dangerous Soviet-designed reactors. However, in large part, the policy failed to meet expectations: the “countries in transition” following the collapse of the Soviet Union and Soviet Bloc faced almost insurmountable challenges in reforming their economies. Pressure was applied: as a condition for accession to the European Union for Bulgaria, Slovakia, and Lithuania, eight Soviet-designed reactors were eventually closed. However, 16 reactors of earlier Soviet design (deemed most risky) still operate—including 10 RBMKs in Russia (WNA, 2019a).

In 1994 the G7 Naples Summit proposed an “Action Plan” for helping Ukraine to reform its energy sector that also foresaw the closure of Units 1 and 3 with no restart for Unit 2. Activities comprising the Action Plan included the following:

- Closure of Unit-1 by 1997;
- Decommissioning of Unit-2 without restart;
- Short-term safety improvements to Unit-3, pending its closure by 2000 (IAEA, 1993a,b)
- Provision of facilities necessary to support decommissioning Units 1, 2, and 3;
- Decommissioning of Units 1–3;
- A solution to observed Sarcophagus degradation.

The West applied particular strong political pressure to shutdown ChNPP Units 1 and 3. In December 1994, Nur Nigmatullin, Chief of Ukraine’s nuclear energy utility, *EnergoAtom*, in remarks to G7 representatives reflected the general frustration of countries operating Soviet-designed reactors being pressured to close the most risky reactors—with a barely-veiled barb at the Russian Federation (Sich, 1996a,b)

Of course, we want Chernobyl shut down. What civilized country would want such a reactor? You have no idea what an *ulcer* Chernobyl is for us. But you must understand that in the midst of our current economic crisis, we cannot do so as quickly, inexpensively, and as simply as your governments would like to believe.

On 19 April 1995, then Ukrainian president Leonid Kuchma pledged to close the Chernobyl plant by 2000 if enough resources to do so were provided. In January 2008 the Ukrainian government announced a four-stage decommissioning plan. In February 2014 a new stage of this was approved for Units 1–3, involving dismantling some equipment and putting them into a safe condition by 2028. The plan foresees that by 2046 further equipment will be removed, and by 2064 the units will be dismantled.

Summary—Insights and lessons learned

On April 26, 1986, a safety test led to the most severe accident in the history of commercial nuclear power occurred at Unit 4 of the Chernobyl nuclear power plant (ChNPP4). Root causes of this event, which include reactor design deficiencies, inadequate operator training and plant staff safety culture, are discussed. Over three decades, post-accident recovery measures continue.

Numerous insights were gained and lessons were learned from this accident, related to accident progression and mitigation, RBMK design and operation, and international safety. Some significant insights and lessons, which ultimately enhanced the safety of the operating fleet, are summarized here.

Accident progression and mitigation

- An inextricably linked confluence of root technical causes (design, construction, inadequate training, procedural violations, administrative deficiencies, etc.) and societal/institutional precursors (attitudes, political expediencies, etc.) leading up to and during the safety test destroyed ChNPP4.
- Little to no direct human interventions aided “natural” processes during the Active Phase: the accident stopped itself. During the Active Phase, needed AMAs did not occur and taken actions were ineffective.
- The accident demonstrated conclusively that the postulated “China Syndrome” is impossible—even with a fully melted core exposed directly to the environment.
- The Soviet account of the accident delivered to the IAEA in Vienna 4 months later, not only intentionally misdirected the causes away from fundamental deficiencies in the design (attributing fault solely to the operators) but also misinformed the international community regarding the alleged effectiveness of the AMAs and radioactivity releases (Varley, 1993; Sich, 1996a,b):
- The Soviets did not share post-accident results of investigations with the international community. Only after the dissolution of the Soviet Union did information regarding the progression of the accident slowly emerged strongly countering the initial Soviet account.
- To this day, on the international level, crucial updated information regarding the scope and progression of the accident (doubts about the scope of radioactivity releases soon after the accident and true accounts of the AMAs) remains either ignored or downplayed—muddling not only further appraisals of deficiencies of the RBMK design but also opening the door to popular media myths on both sides of the debate.

RBMK design and operation

- The Soviets knew well before the accident of the potential for serious consequences to arise from design deficiencies and shoddy construction practices.
- The West knew of flawed design and deficient operational issues of Soviet-designed nuclear reactors, and the West also knew of the weak safety culture in the Soviet Union—one that emphasized production over safety (Sich, 2011)
- The effect of the Chernobyl Accident beyond the former Soviet Union accentuated the already significant concerns among many countries and international organizations regarding the safety of all Soviet-designed reactors—leading to a strong effort on the part of the West to pressure for closure of the most-risky reactors and the provision of very significant funds to help upgrade the safety of less-risky Soviet-designed reactors.

International insights

- Chernobyl resulted in changes in the industry’s (within and beyond the USSR) perception about several known challenges:
 - For owners/operators, there was the realization that severe accidents can occur in RBMK designs (there was significant fuel melting and relocation, steam explosion(s), combustion, and significant radioactivity release).
 - Increased recognition of the importance of external stakeholder communication (timely and accurate notifications to domestic and international agencies).
 - The potential for releases from nuclear accidents to transcend national borders (the USSR had to admit what occurred because releases were detected in other countries; other countries became aware of the importance of international nuclear safety).
- Generally, the West understood that no significant design, operational, or regulatory changes would be necessary for their own reactors. For example, the US Nuclear Regulatory Commission (NRC) concluded ChNPP4 lessons fell short of requiring immediate changes in NRC regulations (USNRC, 1989):

Except for the use of a graphite moderator and gravity-driven control rods, these HTGRs (High Temperature Gas-Cooled Reactors and the Chernobyl reactor have no other features in common. The staff assessed the other areas at issue—operations, design, containment, emergency planning, and severe-accident phenomena—and found that the implications of the Chernobyl accident have generated no new licensing concerns for HTGRs and that general conclusion and those pertaining to specific areas are the same as those for light-water reactors. In performing these assessments, the staff reviewed the existing information related to these areas and concludes that programs underway or being considered adequately satisfy any concerns that could be generated because of the Chernobyl accident.

Nevertheless, the NRC staff identified and acted upon several lessons learned from the ChNPP4 accident. In USNRC (1989), the staff identifies several areas of concern, such as low power and shutdown risk, appropriate fire protection measures, and operator training. For example, there was increased recognition of conditions in which positive reactivity coefficients can occur in LWRs:

- In a BWR, unvoided, lower temperature conditions may approach over-moderation when few control rods are inserted.
- In a PWR, a decrease in moderator density results in a decrease in core boron, which, under some conditions, may give rise to a positive moderator coefficient.

The conditions in which this can occur are limited and when present, there is limited reactivity insertion. In addition, LWRs have fast acting scram systems. However, efforts have been made to understand, document, monitor, and take appropriate actions for such conditions, such as modifications to Instrumentation and control systems to detect and suppress instabilities, procedures, administrative controls, and training to ensure that operators understand the issue. In addition, there was increased emphasis on the importance of documenting and sharing lessons from events that had previously occurred in other plants.

- Internationally, there was significantly increased international engagement in existing organizations, such as the Organization of Economic Cooperation and Development (OECD) and International Atomic Energy Agency (IAEA), increased activities to provide nuclear safety support; international conventions on nuclear safety (emphasizing independent regulatory bodies, operator responsibilities, early notification, international system on severity (including development of the INES scale), liability amounts were increased by Paris and Vienna and adopted. An international industry organization, the World Association of Nuclear Operators (WANO) formed in 1989 that offers peer review, performance analysis, member support, and training and development.

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The Events at Fukushima Daiichi

Shinya Mizokami^a, ^a TEPCO Holdings Company, Tokyo, Japan
Joy L. Rempe^b, ^b Rempe and Associates, LLC, Idaho Falls, ID, United States

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Glossary

Beyond Design Basis Event (BDBE) An event less frequent than and with potentially more severe consequences than a design basis accident.

Design Basis Accident (DBA) Accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

Extended loss of all AC Power (ELAP) An event sequence that results in the failure of all AC power sources at the reactor facility so that only the reactor facility's battery power is available.

Moment of Magnitude Scale A measure of an earthquake's magnitude ("size" or strength) based on its seismic moment. It is expressed in terms of the magnitudes of the original "Richter" magnitude scale.

Introduction

On March 11, 2011 at 14:46(GST+9), the Great East Japan Earthquake (GEJE) and a subsequent tsunami occurred as a result of the seismic slip-rupture, whose length was more than 500 km in the rim of Pacific plate. The moment magnitude scale for the GEJE was evaluated as 9.0. A seismic scram signal was immediately activated at Fukushima Daiichi Nuclear Power Station (FDNPS), which includes six Boiling Water Reactors (BWRs). Units 1, 2, and 3, which were operating at their rated power level, were automatically shut down. Units 4, 5, and 6 were in periodic inspection outages at the time of the earthquake. Due to the collapse of the transmission line tower, FDNPS lost off-site power. However, the emergency diesel generators (EDGs) started, and the electric power necessary to support the required safety systems for the reactors was maintained. Later, a tsunami of up to 15 m in height (well above the design flooding height of 6 m) inundated the FDNPS at about 15:37 (GST+9), flooding electrical equipment such as the EDGs. Accident response measures were hindered by the loss of off-site power and the EDGs, along with a loss of DC power due to batteries being rendered inoperable by flooding or by running out of energy due to extended use. This hindered the ability of operators to monitor the reactor and cooling operations in the main control room.

Plant design

As noted above, FDNPS included six units, designated 1F1 through 1F6 in Fig. 1. Unit 1, which is a BWR/3 design housed in a Mark I containment, is rated at 460 MWe and began operation in 1971. Units 2 through 5, which are BWR/4 designs housed in Mark I



Fig. 1 Fukushima Daiichi Nuclear Power Station. Between each numbered unit is a shared venting tower. Courtesy of TEPCO Holdings.

containments, are rated at 784 MWe. Unit 6, which is a BWR/5 design housed in a Mark II containment, is rated at 1100 MWe and began operation in 1979. For decay heat removal, the earlier Unit 1 design included an Isolation Condenser (IC) system rather than the Reactor Core Isolation Cooling Systems (RCICs) present in Units 2 through 6 [See [Hamon \(2021\)](#) for additional details regarding BWR system design differences].

Accident progression and endstate at Units 1, 2, and 3

While the accident progression differed among the units, the loss of DC power resulted in severe accidents in Units 1, 2, and 3. **Fig. 2** compares the estimated endstate of the reactor building, water, and fuel containing materials in Units 1, 2, and 3 ([Mizokami, 2019](#)). These interim endstate figures, which were developed in December 2018, are based on information obtained from muon tomography and robotic investigations, dose surveys, systems analysis code evaluations, and plant instrumentation data. As additional investigations are completed, these images will be updated.

Unit 1

Unit 1 lost AC and DC power at almost the same time after the tsunami arrived. Because the protective door facing seaside of the turbine building was kept open after the earthquake, a massive amount of tsunami water entered the turbine building. The switchgears were located near the front of this door, and their function was lost soon after the tsunami intrusion. Then, the DC batteries located on the basement floor were lost due to flooding. **Fig. 3** ([Hara, 2017](#)) presents a schematic showing the availability of systems and components after the tsunami inundated Unit 1 at 15:36 on March 11, 2011.

After the reactor tripped due to the seismic accelerations, the operators tried to establish cold shutdown (defined by TEPCO to occur when reactor cooling-water temperature falls below 100 °C) in Unit 1 by removing decay heat and controlling the system pressure by cycling the isolation condenser (IC) on and off within prescribed limits. In the normal shutdown procedure, it is required to control the pressure to ensure a gradual temperature decrease of 55 °C per hour. At the time when AC/DC power was lost, the IC was not in operation and DC power was needed to restart the IC. Therefore, Unit 1 lost its cooling function for about 1 h after the earthquake. The loss of cooling resulted in a decrease in the Reactor Pressure Vessel (RPV) water inventory due to evaporation by decay heat. Fuel assemblies and support structures experienced heat up and exposure to high temperature steam, resulting in hydrogen generation due to vapor–cladding zirconium chemical reaction. By 14:30 on March 12, 2011, Unit 1 was experiencing a severe accident with core damage, core melt, and radioactive material release. Venting actions by the operators (see **Fig. 4**) which were confirmed at 14:30 on March 12, lowered the Primary Containment Vessel (PCV) pressure and radioactive releases were scrubbed by suppression chamber (SC) water. However, the hydrogen explosion at ~15:36 on March 12, and images of the Unit 1 reactor building (see **Fig. 5**) ([TEPCO, 2011a](#)) suggest that PCV boundary failure already occurred, allowing hydrogen to leak from the PCV and accumulate in the reactor building causing the observed explosion.

Unit 1 lost cooling earlier than Units 2 or 3, and the restart of water injection was later. Therefore, it is estimated that damage to Unit 1 is the most severe among the three operating units. The investigation results obtained by muon tomography measurements ([TEPCO, 2015](#)) indicate that almost all the fuel material relocated from the original core location, and analyses predict that molten

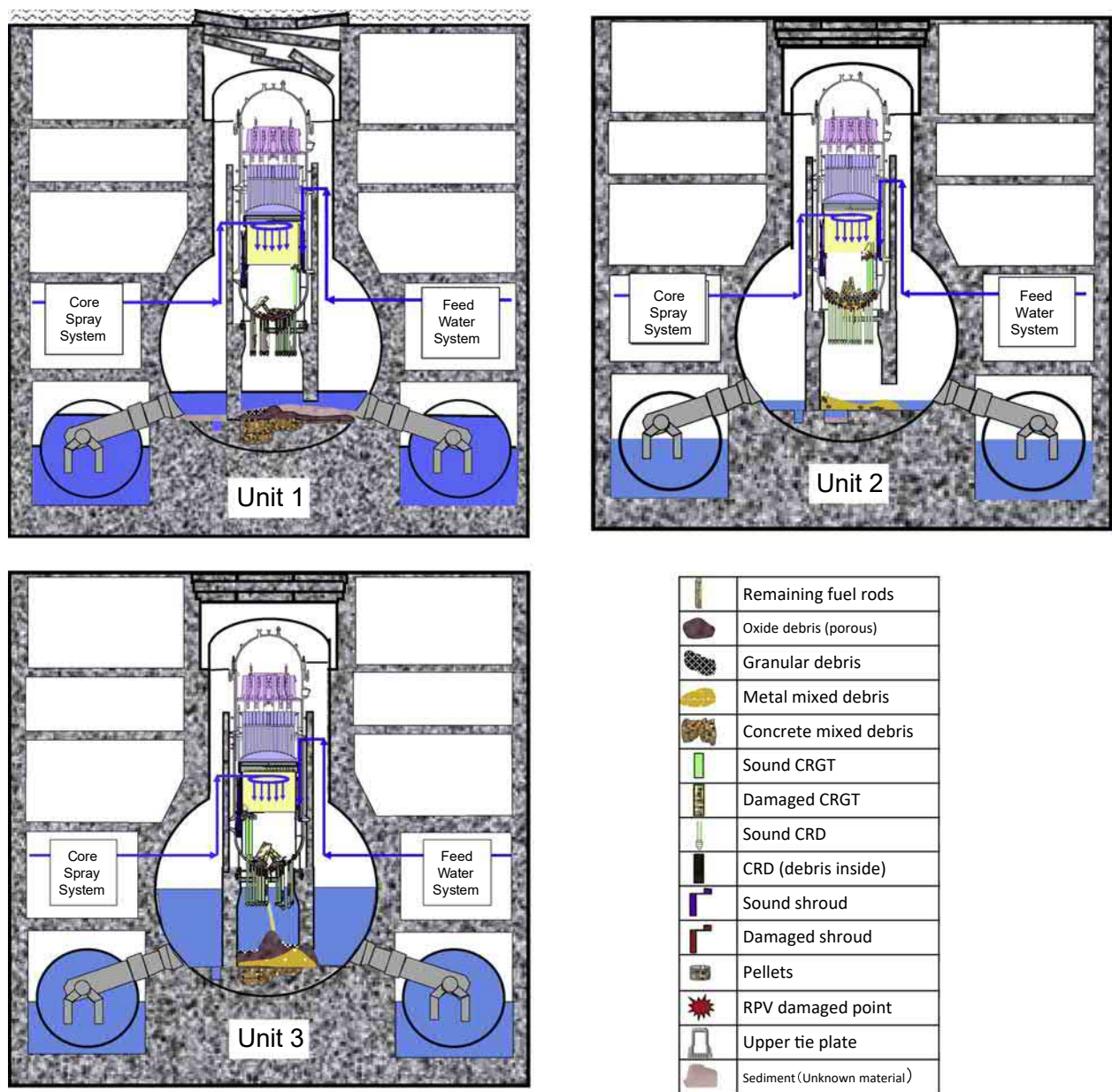


Fig. 2 Interim endstate figures (based on December 2018 update) for Units 1, 2, and 3. Courtesy of TEPCO Holdings; Mizokami S. 2019. Current Status and Recent Investigation Result of Fukushima Daiichi. Presented at 9th Conference on Severe Accident Research (ERMSAR 2019), March 2019.

core materials slumped from the RPV to the primary containment vessel (PCV), ablating PCV floor concrete. Images and dose survey information obtained using robots (Fig. 6) (TEPCO Holding, 2017j) and results from water sample evaluations confirm that fuel containing material (FCM) relocated from the RPV with accumulated deposits heights from 20 cm to 100 cm observed external to the pedestal region of the PCV (TEPCO Holdings, 2017i). At present (July 2019), there are no images from robotic examinations inside the pedestal. However, it is estimated that the debris and other materials accumulated similarly to what has been observed in Unit 3.

Unit 2

Unit 2 also lost AC and DC power at about the same time after the arrival of the tsunami. Although one Unit 2 EDG was air cooled and located in a separate building - the common spent fuel storage building, flooding of switchgear in the basement floor of this separate building resulted in the loss of AC power at Unit 2. Fig. 7 presents a layout of the FDNPS plant systems and components and indicates the availability of Unit 2 systems and components after the tsunami inundated the plant on March 11, 2011.

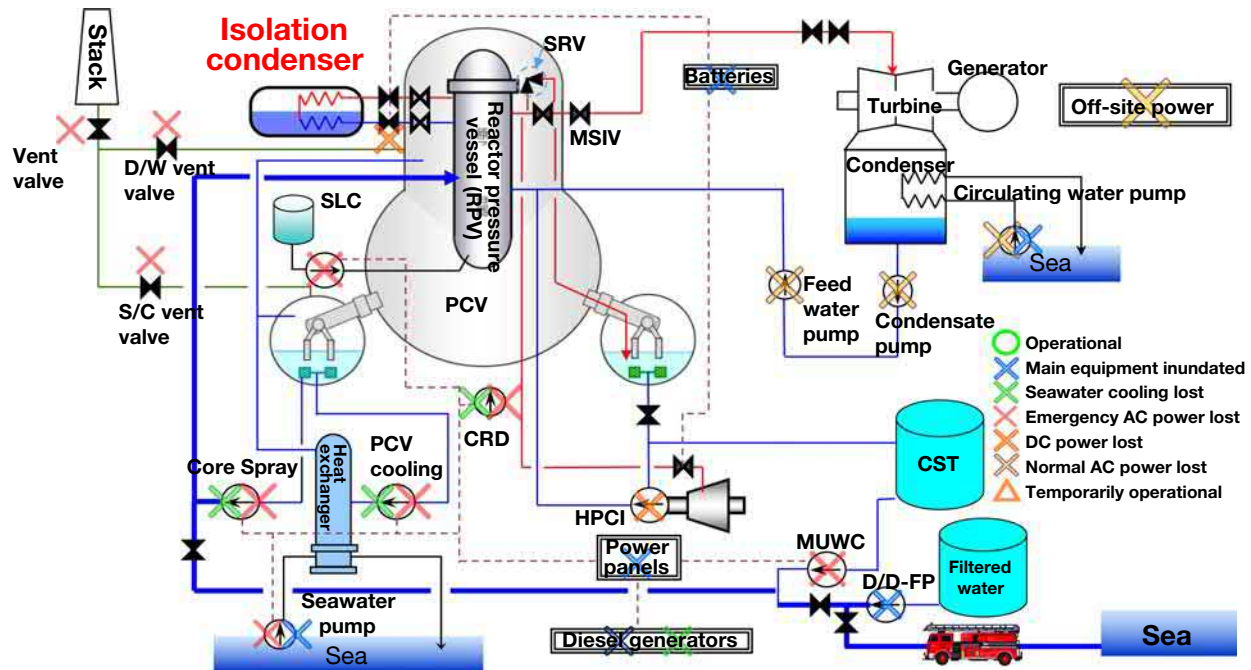


Fig. 3 Status of Unit 1 systems and components after inundation by the tsunami. AC, Alternating Current; CRD, Control Rod Drive; CST, Condensate Storage Tank; DC, Direct Current; DDFP, Diesel Driven Fire Pump; DW, Drywell; HPCI, High Pressure Coolant Injection; MSIV, Main Steam Isolation Valve; PCV, Primary Containment Vessel; RPV, Reactor Pressure Vessel; SC, Suppression Chamber; SLC, Standby Liquid Cooling; SRV, Safety Relief Valve. Courtesy of TEPCO Holdings; Hara T. 2017. Fukushima Japan Accident and Current Situation. Presentation at MIT Nuclear Plant Safety Course, June 21, 2017.



Fig. 4 Unit 1 venting (at 14:39 on March 12, 2011). Note plume from the Units 1 and 2 shared venting stack. This picture is provided by Ishikawa Bon.

After the reactor tripped due to seismic accelerations, operators took actions to place Unit 2 in cold shutdown by removing decay heat and controlling the RPV water level by injecting water using the Reactor Core Isolation Cooling (RCIC) System. The RCIC turbine pump is driven by steam so that it is required to keep RPV water level within a certain range to maintain high steam quality. Although the RCIC system was automatically tripped due to high RPV water level at 14:51 (GMT+9) and 15:28 on March 11, 2011, the RCIC system was manually restarted 15:39 just 2 min before loss of DC power. The RCIC system relies on DC power for critical functions, such as controlling the water injection rate, activating its trip logic sequence, cooling the lubricating oil, etc. It is not fully understood why the RCIC system continued to operate and cool the RPV until the morning of March 14, 2011. Conditions in Unit 2 were outside the RCIC system design envelop, but it continued to operate with low quality steam, high back pressure, and high temperature lubricant oil. It is also not understood what ultimately caused the RCIC system failure.

After loss of the RCIC system, the RPV water level gradually decreased. At about 18:00 on March 14, 2011, the operators depressurized the RPV by opening the Safety Relief Valves (SRVs). The RPV water level at that time was near the top of active fuel (TAF), so the fuel had not yet heated up. When the system pressure suddenly lowered, the water inventory decreased due to flashing; RPV water level suddenly decreased from near the TAF to below the bottom of active fuel (BAF). Thus, the fuel heated up by decay heat rather than interactions between steam and cladding. Water injection from a fire truck should have been conducted just after depressurization; however, injection was delayed due to a lack of fuel for the fire truck. When water was injected onto the preheated fuel assemblies, the exothermic chemical reaction between steam and zirconium-containing cladding generated significant amounts of heat. Thus, Unit 2 experienced a severe accident with core damage, core melt, and radioactive material release on the night of March 14, 2011.



Fig. 5 Unit 1 explosion (at 15:36 on March 12, 2011); (A) image when plume was largest (hiding the reactor building) and (B) damage to the reactor building. (A) This picture is provided by Fukushima Central Television Company 2019. Y. Sakurai, General Manager, News Department, Fukushima Central Television Company, Ltd; (B) TEPCO 2011a Unit 1 of Fukushima Daiichi Nuclear Power Station, photo collection. <https://photo.tepcoco.jp/en/date/2011/201103-e/110313-01e.html>, uploaded march 13, 2011, last accessed July 1, 2019. [used image: 110313_1f_1]).

The Unit 3 hydrogen explosion that occurred at noon on March 14, 2011 hindered efforts to vent Unit 2, failing the valve of the venting line. Therefore, PCV venting was not conducted, but high pressures and temperatures within the PCV are believed to have led to seal degradation, and the PCV pressure decreased on the morning of March 15, 2011. Because releases were not reduced by suppression water scrubbing, Unit 2 released more radioactive materials to the environment than earlier releases from Units 1 and 3.

As shown in Fig. 8, the Unit 2 reactor building was not damaged by any hydrogen explosion. Post-accident evaluations indicate that the absence of any hydrogen explosion was due to “unintentional” early venting of the reactor building. In other words, the Unit 1 explosion at 15:36 on March 12, 2011 (Fig. 5) caused the blowout panel to open, releasing steam and possibly hydrogen. There was no hydrogen explosion within the Unit 2 reactor building. However, it is estimated that the amount of the radioactive materials discharged from Unit 2 is largest among the three units.

Quantification of the generated energy by chemical reaction during core heat up is important to estimate the fuel damage and endstate distribution of FCM in the RPV and PCV. However, there is much uncertainty in the amount of steam available during core heat up because it depends on the amount of water injected by the fire truck. Images obtained using muon tomography measurements (TEPCO Holdings, 2017f) show little damaged or melted material in the original core position with more relocated materials at the bottom of the RPV. Images from robotic investigations show considerable holdup on cabling beneath the RPV, thermally—induced damage to structures beneath the vessel and accumulation of previously molten materials deposited on the PCV floor (with heights ranging from 40 to 70 cm) (TEPCO Holdings, 2018b). As shown in Fig. 9 (TEPCO Holdings, 2017a,e, 2018a), the tie handle from the upper part of a fuel assembly can be seen on the surface of the PCV floor.

Unit 3

Fig. 10 presents a schematic of Unit 3 systems and components available after the tsunami inundated the plant on March 11, 2011 (the layout and floor elevations are similar to that shown in Fig. 6A for Unit 2). Although Unit 3 lost AC power at the time of the tsunami arrival, it didn’t immediately lose DC power because the batteries were at a higher elevation that wasn’t inundated by tsunami flooding. This allowed the operators to initially rely on the RCIC system and when the batteries were depleted, the High Pressure Coolant Injection (HPCI) system.

After reactor trip due to seismic scram, operators worked to place the unit in cold shutdown by removing decay heat and controlling the RPV water level using the RCIC system. The RCIC system continued to operate using DC power until 11:36(GMT+9) on

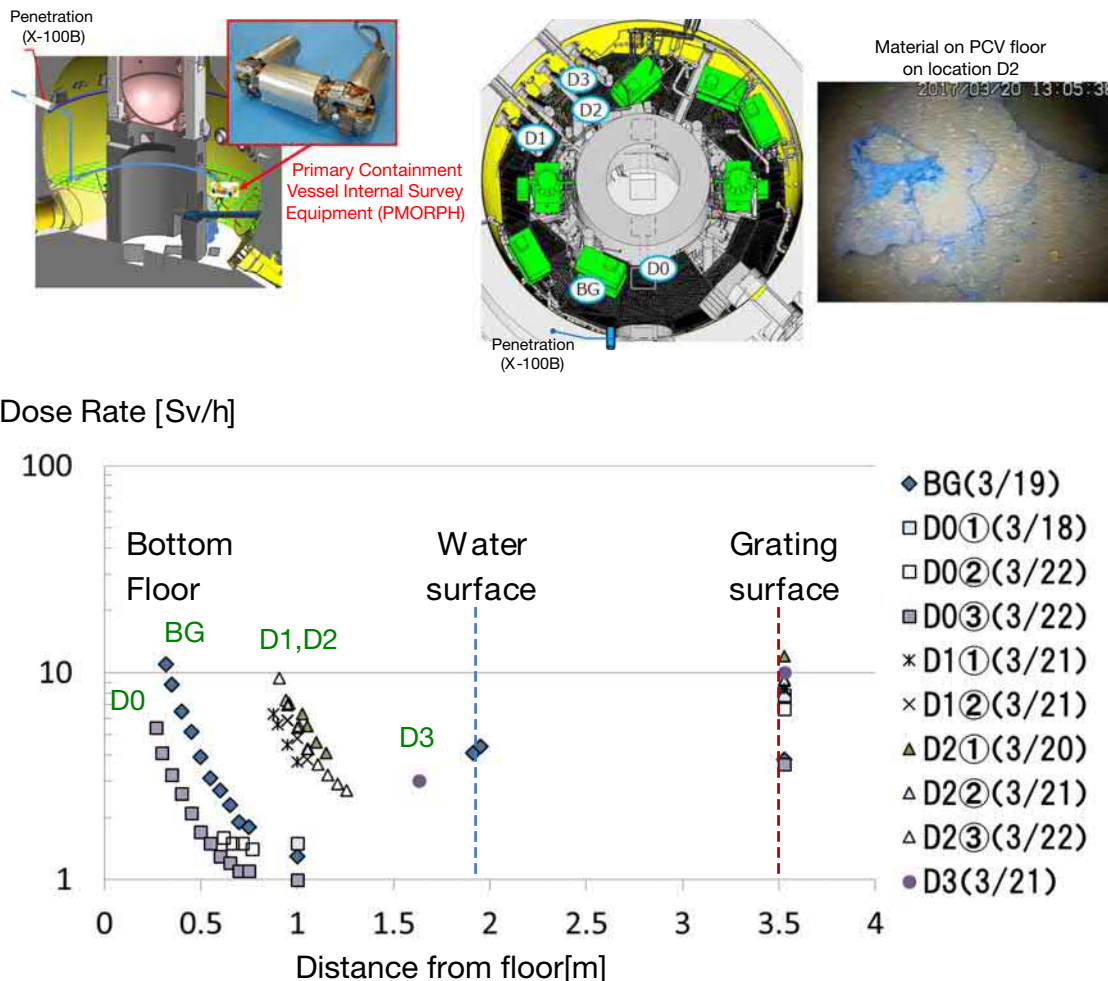


Fig. 6 Unit 1 robotic investigation. Courtesy of TEPCO Holdings 2017j Unit 1 Primary Containment Vessel Internal Investigation. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170327_01-e.pdf, last accessed March 2018.

March 12, 2011. It is estimated that the RCIC system tripped when the back pressure reached the trip setpoint (TEPCO Holdings, 2017b). After RCIC system trip, the HPCI system automatically started due to RPV water level decrease. The HPCI system was able to inject water and depressurize the RPV. At 20:36 on March 12, the RPV water level gage data became unavailable due to loss of DC power. The water injection rate by the HPCI system under low pressure was insufficient, and the RPV water level gradually decreased. At 2:42 on March 13, 2011, the operator manually terminated the HPCI system. At this time, the RPV water level had already decreased to the TAF, and the HPCI was stopped to allow low pressure injection using the Diesel Driven Fire Pump (DDFP). However, the SRV failed to open, resulting in an increase in the RPV pressure and a loss of any cooling. Unit 3 fuel assemblies exposed to steam started to heatup, leading to a severe accident with core damage, melting, and radioactive material release. PCV venting through the suppression chamber wetwell with water scrubbing of radioactive material was conducted at around 9:10 on March 13, 2011 (see Fig. 11).

A hydrogen explosion occurred in Unit 3 at 11:01 on March 14, 2011 (see Fig. 12) (TEPCO, 2011c). Post-accident investigations indicate that the Unit 3 hydrogen explosion obstructed recovery actions for water injection and venting at Unit 2. Furthermore, post-accident evaluations found that Unit 3 venting allowed combustible gases to flow through the standby gas treatment system (SGTS) piping and cause a hydrogen explosion in Unit 4 (TEPCO, 2016).

Examinations indicate that core damage incurred at Unit 3 is less than Unit 1 but more than Unit 2. Evaluations indicate that core heat up occurred during the phase where RPV water level was decreasing. Therefore, the Unit 3 endstate is similar to Unit 1. Results obtained by muon tomography measurements (TEPCO Holdings, 2017f) indicate that no large pieces of damaged or previously molten fuel material remain in the original core location and suggest the possibility that some FCMs have relocated to the bottom of the RPV. Investigations by a submersible robot (see Fig. 13) (TEPCO Holdings, 2017g,h) show that a large amount of previously melted materials relocated to the PCV floor with an accumulated height of more than 2 m (TEPCO Holdings, 2018c). Several Control Rod Guide Tubes (CRGTs) and a speed velocity limiter of the control rod, which were initially within the RPV, were identified in the debris bed on the surface of the PCV floor.

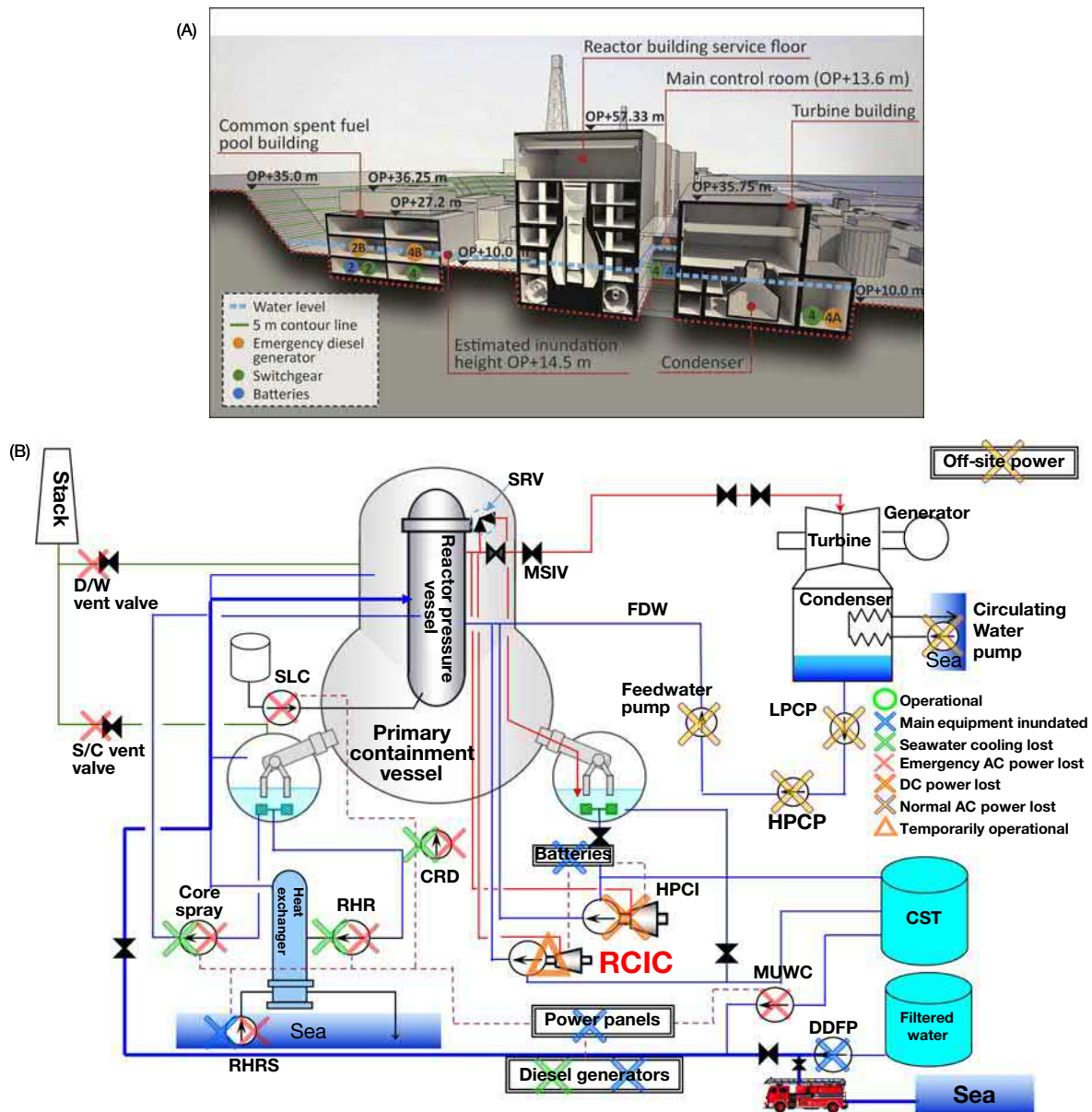


Fig. 7 FDNPS plant systems and components [although this image was generated for Unit 4 in (IAEA, 2015), the elevations and layout of systems is identical to that in Unit 2]; (A) layout, and (B) status of Unit 2 systems and components after inundation by the tsunami. AC, Alternating Current; CRD, Control Rod Drive; DC, Direct Current; DDFP, Diesel Driven Fire Pump; FDW, Feedwater; DW, Drywell; HPCP, High Pressure Coolant Pump; LPCP, Low Pressure Coolant Pump; MSIV, Main Steam Isolation Valve; MUWC, Makeup Water Condensate; OP, Sea level at Onahama Port; PCV, Primary Containment Vessel; RHR, Residual Heat Removal; RPV, Reactor Pressure Vessel; SC, Suppression Chamber; SLC, Standby Liquid Cooling; SRV, Safety Relief Valve. Courtesy of TEPCO Holdings; IAEA 2015 The Fukushima Daiichi Accident Technical Volume 1/5 description and context of the accident. <https://www-pub.iaea.org/MTCD/Publications/PDF/AdditionalVolumes/P1710/Pub1710-TV1-Web.pdf>, last accessed December 11, 2019, August 31, 2015; Hara T 2017. Fukushima Japan Accident and Current Situation. Presentation at MIT Nuclear Plant Safety Course, June 21, 2017.

Long term effects on health and nuclear power

TEPCO estimates (World Nuclear Association, 2016) indicate a total of about 1020 PBq was released to the atmosphere between March 12 and 31, 2011. After this time, very little additional radiation was released. Apart from noble gases, this release was comprised of 500 PBq Iodine-131, 10 PBq Cs-137, and 10 PBq Cs-134. Of the total release, it is estimated that about 20% came from Unit 1, 40% from Unit 2 (which peaked on 15 March), and 40% from Unit 3 (which peaked on 16 March). Relatively little radioactive material was released by venting (because it was routed through steam and water before it was released through the exhaust stacks) or by the hydrogen explosions. Based on measurements of residents and evacuees, Tokonami (2012) reports that the median thyroid equivalent dose is 4.2 and 3.5 mSv for children and adults, respectively, and that maximum thyroid doses to

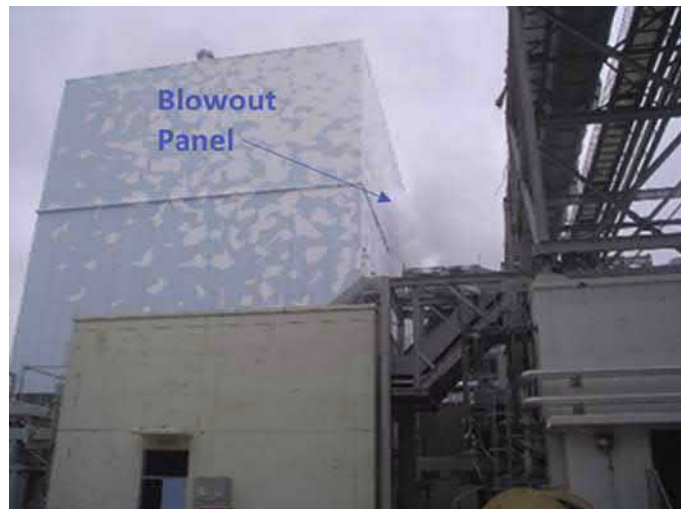


Fig. 8 Steam (and possible hydrogen) release from the Unit 2 building with an open blowout panel. Courtesy of TEPCO Holdings 2017b The 5th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident. https://www7.tepco.co.jp/newsroom/press/archives/2017/1485273_10469.html, December 25, 2017, last accessed July 2019.

children and adults were 23 and 33 mSv, respectively. This is much smaller than the average thyroid dose in the Chernobyl accident (490 mSv in evacuees) reported in [UNSCEAR \(2008\)](#).

Radiation released from the affected reactors at FDNPS exceeded official safety guidelines, but there were no immediate deaths caused by these releases. None of the six FDNPP worker deaths were due to radiation exposure ([UNSCEAR, 2014](#)). Given the uncertain health effects of low-dose radiation, cancer deaths cannot be precluded. The United Nations and World Health Organization studies ([UNSCEAR, 2014](#); [WHO, 2013](#)) indicate that no discernible increase in the rate of cancer deaths is expected ([UNSCEAR, 2014](#)). Several deaths, however, are attributed to the evacuation and subsequent long-term displacement caused by the nuclear disaster.

The ultimate impact on the nuclear enterprise is still unknown. All 60 reactors within Japan were shutdown after the accident. Prior to restarting, plants are now required to pass a safety review by the regulator and gain consent by local governments that would be affected by the plant's operation. At the time that this chapter was written, only 9 PWRs are operating and 25 are either being or are set to be decommissioned ([JAIF, 2019](#)).

Lessons learned /impact of post-accident investigations

Uncertainties remain regarding the events after the GEJE earthquake at FDNPS. As post-accident and recovery efforts continue, more information will be obtained regarding the accident progression at each of the affected reactors. Nevertheless, information available from these reactors already provides many lessons for enhancing the safety of nuclear power plants worldwide. This section identifies some of the actions taken or underway by the nuclear industry and regulators to address lessons learned from the events at FDNPS.

Containment venting and filtration

Available instrumentation data and dose survey information indicate that the three operating FDNPS units experienced PCV leakage, but there are uncertainties regarding the actual leakage locations and the timing of such leakages. Many countries have or are in the process of implementing revised severe accident guidance that places a high priority on early venting to ensure that the primary containment is maintained at pressures and temperatures well below values that challenge containment integrity ([Aleza, 2018](#); [Steinrötter, 2018](#); [O'Neill, 2018](#); [Duspiva, 2018](#); [TEPCO Holdings, 2017b](#)). As discussed in [Lutz and Williamson \(2021\)](#) containment performance will be enhanced in the United States by anticipatory venting using the hardened vent systems mandated for Mark I and Mark II BWRs ([NEI, 2015](#)). In some countries, measures are being considered to add or upgrade filtered containment venting systems for CANDU, PWR, BWR, and VVER units as a severe accident mitigation strategy. However, there are reservations regarding their effectiveness and reliability, as well as the instrumentation required to support such systems (e.g., [LaBarge, 2018](#); [Steinrötter, 2018](#); [Cenerino, 2018](#)).

Severe accident water addition

Available information (see [Fig. 2](#)) indicates there are differences in the debris end-states at Units 1, 2, and 3. These differences stem from differences in their accident progressions which were affected by the availability of water for cooling. Many countries have

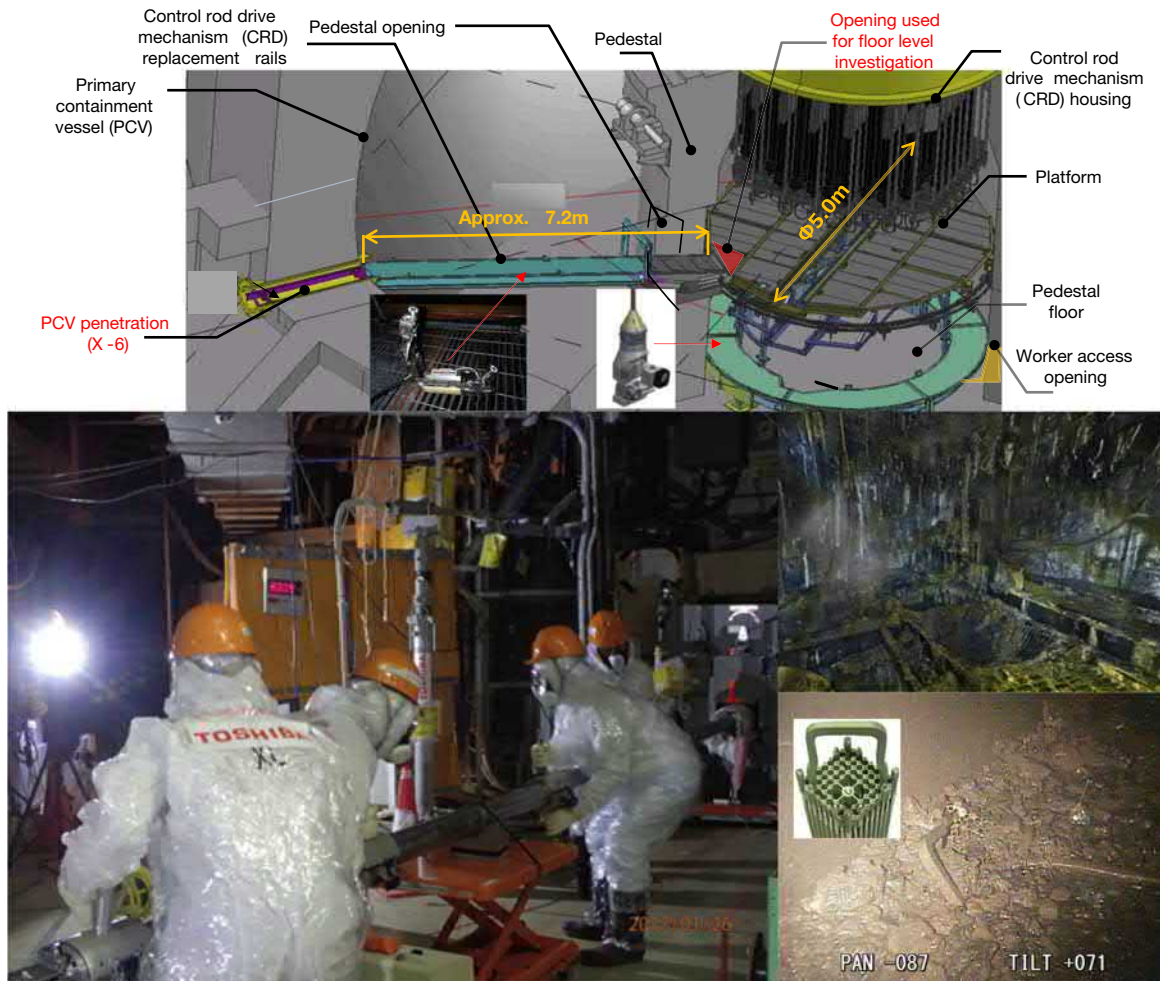


Fig. 9 Investigations through the Unit 2 X-6 penetration. Courtesy of TEPCO Holdings 2017a Pre-investigation Results of X-6 Penetration for the Unit 2 Primary Containment Vessel Investigation at Fukushima Daiichi Nuclear Power Station. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170126_01-e.pdf, January 26, 2017; last accessed July 2019; TEPCO Holdings 2017e Pre-investigation results of the area inside the pedestal for the Unit 2 primary containment vessel investigation at Fukushima Daiichi Nuclear Power Station (examination results of digital images). http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170202_01-e.pdf, February 2, 2017, last accessed July 2019; TEPCO Holdings 2018a Fukushima Daiichi Nuclear Power Station Unit 2 Primary Containment Vessel Internal Investigation Results (Bulletin). http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2018/images/handouts_180119_01-e.pdf, last accessed July 2019.

implemented measures regarding water addition based on insights from FDNPS (NEI, 2015; Steinrötter, 2018; TEPCO Holdings, 2017b). As discussed in Lutz and Williamson (2021), revised US BWROG guidance based on examination information includes severe accident water addition and water management strategies to enhance the effectiveness of fission product scrubbing in the suppression pool during venting through the hardened vent system. The PWROG guidance also includes water addition (possibly via containment spray operation for fission product mitigation) during venting. The guidance considers the impact of water addition on containment integrity and contains guidelines for venting during water addition to prevent over-pressurization of the containment.

Infrastructure degradation

Recovery efforts at FDNPS were hindered by damage to on-site and off-site infrastructure. The loss of all on-site and off-site power along with the inability to bring replacement equipment to the site due to traffic disturbances, such as road damage from the earthquake, hindered the ability of plant staff to restore cooling to the reactors and spent fuel pools at Daiichi. To address such challenges, additional measures have been implemented in many countries (LeBerge, 2018; O'Neill, 2018; Gilbert, 2018a; Gilbert, 2018b; Patel, 2018; Steinrötter, 2018; Suslov, 2018; TEPCO Holdings, 2017b; NEI, 2016) to ensure that there is additional equipment on-site and in some cases, to ensure that additional equipment can be transported on-site even when roadways may not be available. For example, the US industry has instituted the FLEX program that requires that all plant sites maintain additional

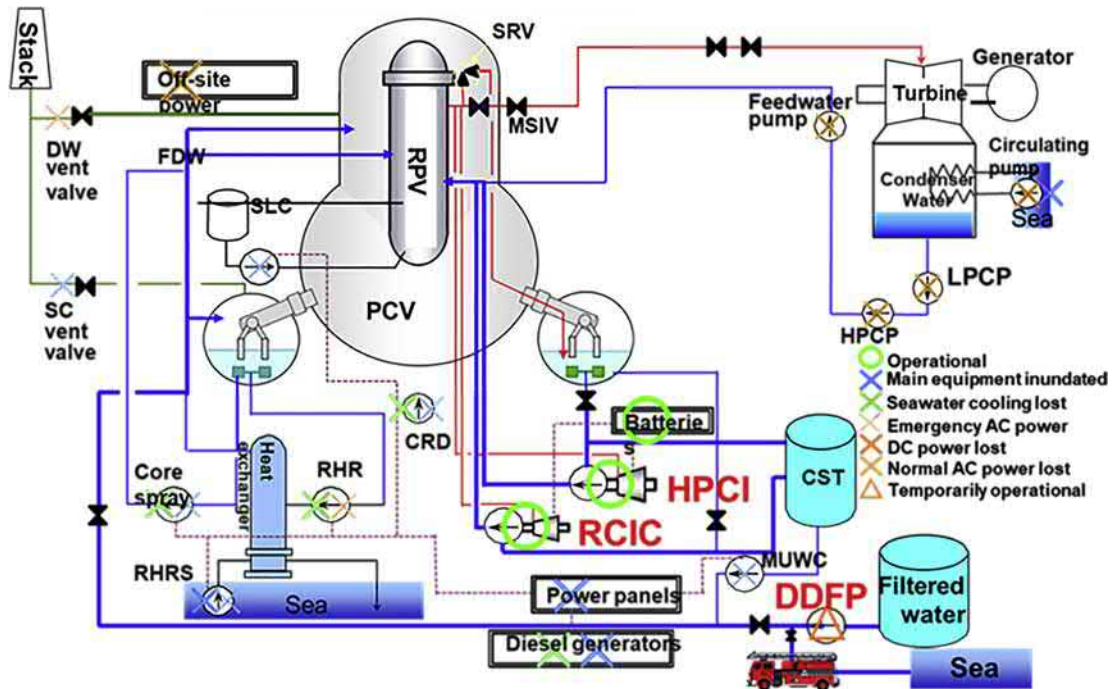


Fig. 10 Status of Unit 3 systems and components after inundation by the tsunami. AC, Alternating Current; CRD, Control Rod Drive; DC, Direct Current; DDFP, Diesel Driven Fire Pump; FDW, Feedwater; DW, Drywell; HPCP, High Pressure Coolant Pump; LPCP, Low Pressure Coolant Pump; MSIV, Main Steam Isolation Valve; MUWC, Makeup Water Condensate; PCV, Primary Containment Vessel; RHR, Residual Heat Removal; RHRS, Residual Heat Removal System; RPV, Reactor Pressure Vessel; SC, Suppression Chamber; SLC, Standby Liquid Cooling; SRV, Safety Relief Valve. Courtesy of TEPCO Holdings; Hara T 2017 Fukushima Japan Accident and Current Situation. Presentation at MIT Nuclear Plant Safety Course, June 21, 2017.



Fig. 11 Unit 3 venting (Left: plume from 1F3/1F4 vent stack at 10:00 on March 13, 2011, Right: plume from 1F3/1F4 vent stack at 13:00 on March 13, 2011). Courtesy of TEPCO Holdings 2017b The 5th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident. https://www7.tepco.co.jp/newsroom/press/archives/2017/1485273_10469.html, December 25, 2017, last accessed July 2019.

equipment for water injection, power restoration, and debris removal as well as similar equipment that are maintained at two national response centers for transportation to plant sites (NEI, 2016). Within Japan, similar measures have been instituted to ensure that such equipment is available to plant sites (Masui, 2019). New requirements, which were instituted by the newly formed Nuclear Regulatory Authority (NRA), also include measures such as enhanced radiation monitoring, effective communication with local governments as well as domestic and international agencies, strengthened emergency response capabilities, consideration of common-cause failures due to fires, flooding, or large scale disasters, and adoption of safety goals (NRAJ, 2011, 2013, 2017).



Fig. 12 Unit 3 explosion (11:01 on March 14, 2011); (A) image when plume was largest (hiding the reactor building); (B) damage to the reactor building. (A) This picture is provided by Fukushima Central Television. (B) TEPCO, 2011c. Unit 3 of Fukushima Daiichi Nuclear Power Station. <https://photo.tepco.co.jp/en/date/2011/201103-e/110321-01e.html>, Uploaded March 21, 2011, last accessed July 1, 2019. [Used image: 110321_1f_kokuen1]

Multi-unit effects

The events at FDNPS highlighted the need to assess a multi-unit site against multiple concurrent and correlated hazards. There were some advantageous multi-unit effects. For example, even though Unit 5 lost its EDGs, one of the surviving DGs at Unit 6 was available to supply AC power to a neighboring facility, Unit 5 (TEPCO, 2012). In addition, the Unit 1 explosion precluded an explosion within the Unit 2 building due to "unintentional venting". However, most multi-unit effects were disadvantageous. The Unit 3 hydrogen explosion delayed recovery actions at Unit 2, and Unit 3 venting allowed combustible gases to flow through the SGTS piping and cause a hydrogen explosion in Unit 4. Actions to eliminate the potential for backflow in SGTS piping and emphasize early PCV venting address some of the observed adverse multi-unit effects (TEPCO Holdings, 2017b; NEI, 2015). In addition, several countries have re-evaluated, and if needed, implemented enhancements to systems for detecting and mitigating the effects associated with hydrogen accumulation (e.g., see Gilbert, 2018a). Finally, the multi-unit effects observed at FDNPS have increased awareness of the need to consider multiple sources of radiation release; the new FLEX equipment is designed to ensure that there are backup systems to provide power and cooling to protect the integrity of fuel in each reactor and spent fuel pool at a site (NEI, 2016; Gilbert, 2018a; Desaulniers, 2018).

Component and system performance under beyond design basis conditions

Several systems and components, such as the RCIC and IC systems and selected plant instrumentation, were able to survive under beyond design basis conditions at FDNPS. Post-accident investigations have provided some insights regarding the performance of such systems and components (TEPCO Holdings, 2017b), but additional information is desired. For example, a joint US-Japan effort is underway to test RCIC systems under beyond design conditions to determine if it is possible to gain regulatory credit for RCIC system performance, and results from evaluations of plant instrumentation have been incorporated into revised US BWROG guidance for severe accident conditions (Rempe et al., 2019). In addition, several countries have or are considering additional instrumentation for detection and mitigation of severe accident conditions (Cenerino, 2018; TEPCO Holdings, 2017b).

Up-to-date comprehensive assessments of external hazards

The events and associated consequences of the events at FDNPS underline the need to assess the potential for external hazards and combinations of such hazards. In recent years, there have been several other examples of flooding or seismic events that exceed those considered within a site's design basis such as the 1999 flooding event in the Blayais Nuclear Power Plant in France and the 2011 seismic event at the North Anna plant in the United States. After the events at FDNPS, the need for updated assessments of external events received increased attention. At this time, updated evaluations have been completed in several countries; and where warranted, additional measures to address increased challenges have been taken (NEI, 2015; O'Neill, 2018; Duspiva, 2018; Patel, 2018).

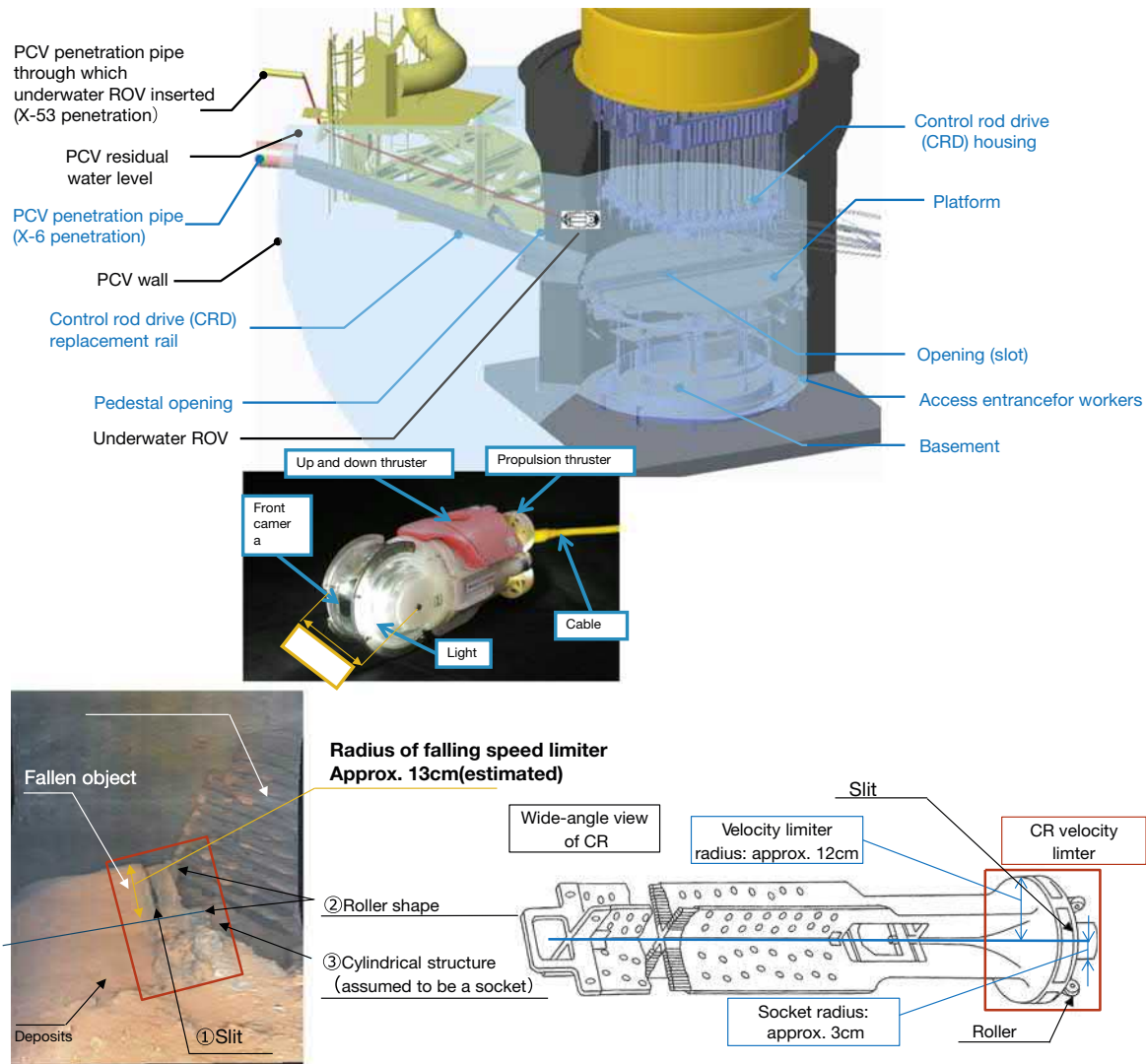


Fig. 13 Investigations through the Unit 3 X-6 penetration revealing lower end of relocated control rod. Courtesy of TEPCO Holdings 2017g Unit 3 Primary Containment Vessel Internal Investigation. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_171130_03-e.pdf, November 30, 2017, last accessed July 2019; TEPCO Holdings 2017h Progress of unit 3 PCV internal investigation(1/2)(preliminary report of July 19 investigation). http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170719_01-e.pdf, July 19, 2017, last accessed July 2019.

Importance of regulatory independence and safety culture

As discussed by Rempe and Williams (2021), the events at FDNPP emphasized the importance of regulatory independence and safety culture. Prior to the Fukushima Daiichi accident, the primary Japanese regulatory agency, Nuclear and Industrial Safety Agency (NISA), was part of the Ministry of Economy, Trade and Industry (METI), which was responsible for regulation and promotion of nuclear energy in Japan. Following the Fukushima Daiichi accident, NISA's association with METI was believed to have compromised its independence. As emphasized in Rempe and Williams (2021), the establishment, implementation, and maintenance of a robust nuclear safety culture is dependent on a technically strong and independent regulator. Several reports (Government of Japan, 2011a,b; NAAIIC, 2012) indicate that NISA failed to monitor or supervise nuclear safety and had lost its technical expertise. Hence, the regulator was not independent and subject to regulatory capture. TEPCO also acknowledges that the reluctance to enhance plant safety against external events using a "step-by-step process" contributed to the severity of the events at Daiichi.

To address these issues, the Government of Japan (NRAJ, 2011, 2013, 2017) reorganized the regulatory framework and established a new regulator independent of METI, the Nuclear Regulatory Authority (NRA) as part of the Environment Ministry. The NRA is responsible for promulgating rules and regulations for nuclear plants and evaluating whether current Japanese plants can resume operations (Approval from local governments is also required prior to plants being allowed to resume operations in Japan). Following its inauguration on September 19, 2012, the NRA carried out a complete review of safety guidelines and regulatory requirements and developed a set of new regulations to protect people and the environment. The new regulations consider the

lessons learned from the events at Fukushima Daiichi, updates regarding harsh natural conditions unique to Japan, and current international safety standards and guidelines. On July 8, 2013, the new regulatory requirements for commercial power reactors were enforced. However, a five-year deferment period is allowed for implementing some safety measures, such as the use of filtered venting systems in PWRs.

Summary

As evidenced by these examples, many actions have already been taken to address insights from the events at FDNPS. As defueling efforts continue, additional insights will be gained, which are of interest to the international community for enhancing the safety of the global nuclear enterprise.

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Ensuring Safety by Design and Evaluation

Joy L. Rempe, Rempe and Associates, LLC, Idaho Falls, ID, United States

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Glossary

Advisory Committee on Reactor Safeguards (ACRS) Independent advisory committee to the U.S. Nuclear Regulatory Commission (NRC) that is statutorily mandated by the Atomic Energy Act of 1954, as amended.

Criticality The normal operating condition of a reactor, in which nuclear fuel sustains a fission chain reaction. A reactor achieves criticality (and is said to be critical) when each fission event releases a sufficient number of neutrons to sustain an ongoing series of reactions.

Defense-in-Depth (DiD) A design and regulatory principle that relies on several consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or engineered barrier were to fail, the subsequent level or barrier would be available. When properly implemented, DiD ensures no single technical, human or organizational failure could lead to harmful effects and the combinations of failures that could give rise to significant harmful effects are of very low probability.

Design Basis Accident (DBA) One category of the postulated accidents used to set criteria and limits for the design and sizing of safety-related systems, structures, and components.

Design Basis Event (DBE) An event considered in the range of conditions, including normal operation, DBAs, and natural phenomena, for which the plant must be designed to ensure acceptable performance of structures, systems, and components.

Emergency Core Cooling System (ECCS) A separate and independent cooling system for the reactor core in the event of a "loss-of-coolant accident".

Engineered Safety Feature (ESF) A system, structure, or component relied upon during or following a DBE to prevent or mitigate the associated consequences from such an event.

Final Safety Analysis Report (FSAR) Document, submitted by an applicant, providing information to support the NRC safety evaluation of a proposed facility. It is based on the finalized detailed design of the facility, and it must periodically be updated throughout the life of the facility.

General Design Criteria (GDC) The set of minimum requirements for the design and safety analysis of water-cooled nuclear power plants.

Reactivity A measure expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Safety-Related Equipment Systems, structures, and components relied upon to remain functional during and following DBEs. Their functionality ensures key regulatory criteria, such as levels of radioactivity released, are met.

Safety Margin In engineering, the difference between the predicted value for a parameter and the value at which failure occurs. In the nuclear industry, safety margins are ensured by setting a conservative safety limit for a parameter and maintaining conditions below this limit by an amount that is at least commensurate with the uncertainty in predicted values for this parameter.

Single Failure Criterion The criterion that no single technical, human, or organizational failure should lead to undesirable consequences. In reactor design, the criterion that the failure of a single system or piece of equipment will not adversely affect public health and safety.

Introduction

From the onset, it was recognized consequences of accidents from a nuclear power plant could have more impact than other types of industrial accidents. As noted by Smyth (1945) in his efforts to design a plant for production of plutonium: “The problem of dissipating thousands of kilowatts of energy is by no means a small one.” To assure safety functions within a nuclear power plant are maintained, Garrick (2013) emphasizes two fronts — the design of the physical system and the evaluations of the physical system design.

On the physical system front, designers emphasize inherent features, such as negative moderator temperature reactivity feedback, which causes an overall decline in reactor power when the moderator temperature increases. Remote plant siting provided additional protection of the public. Designs incorporated independent, redundant, and diverse barriers, such as a containment building, and independent, redundant, and diverse systems to provide emergency core cooling and backup electric power. With time, commercial nuclear power plant designs evolved, not only to maximize economics but also to enhance safety. The need for additional safety measures to address new challenges was illuminated by events, such as the Browns Ferry Fire and the Salem Anticipated Transient without Scram (ATWS) events without core damage (Rempe, 2021) and the Three Mile Island Unit 2 (Skillman and Rempe, 2021), Chernobyl Unit 4 (Sich, 2021), and Fukushima Daiichi (Mizokami and Rempe, 2021) events with core damage. Additional safety measures have been incorporated into the spectrum of nuclear power plant activities, including emergency planning, siting criteria development, design, manufacturing, construction, commissioning and operation, as well as decommissioning and closure.

On the evaluation front, the critical role of safety analyses has been emphasized throughout the development and deployment of commercial nuclear power plants. Results from safety evaluations have been compared with acceptance criteria to assess the ability of barriers and equipment to provide critical safety functions during normal and off-normal (accident) conditions. After events with core damage occurred, it became clear such evaluations required a better understanding of their likelihood of occurrence. Worst-case and maximum-credible accident evaluations were no longer sufficient for ensuring plant safety. Everyday transients, followed by multiple failures of equipment and mistakes by operators, were more likely to occur than the hypothetical design basis accidents required for licensing. Hence, the use of more realistic evaluations and probabilistic risk assessment (PRA) became important in assuring critical safety functions were maintained.

This chapter provides an overview of areas important to ensuring reactor safety by design and evaluation, highlighting the topics of critical safety functions, defense-in-depth (DiD), and regulatory framework. In addition, an example is provided to illustrate that results from evaluations can affect the design of reactor systems and the regulatory framework. Additional information related to reactor systems, operator guidance for prevention and mitigation of accidents, and evaluation of plants with such systems and guidance is found in Lutz and Williamson (2021a,b,c,d,e) and Gauntt (2021).

Critical safety functions

Nuclear power plant safety is assured by maintaining three key reactor safety functions: control reactivity, control cooling, and control radioactivity release (see Fig. 1). Several references (WANO, 2019; IAEA, 2016) emphasize these three “Critical Safety Functions” must be preserved during normal operation and accident conditions. Some references provide a more extensive list; for example, the function to control release of radioactive materials may be separated into functions performed by individual barriers (e.g., the reactor vessel and the containment) or the function to control cooling may be separated into different cooling functions (e.g., maintain core cooling, maintain containment cooling, and maintain the coolant heat sink). However, such additional functions can be consolidated into the three critical safety functions depicted in Fig. 1.

These functions guide reactor design, and evaluations are performed to assess the adequacy of the design to ensure these functions are maintained. Critical safety functions also play an important role in selecting design basis events, events considered in PRAs,

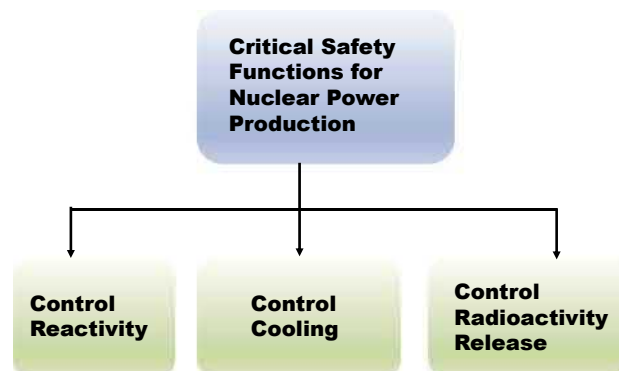


Fig. 1 Critical safety functions.

safety-related equipment, and the minimum number of plant parameters that should be monitored. For example, there are several standards by the American National Standards Institute (ANSI) and the American Nuclear Society (ANS), such as ANSI/ANS 58.14-2011 (ANS, 2017), that stress the importance of these critical safety functions in evaluating the performance of safety-related components. Reviews of prior events with significant core damage at commercial nuclear power plants (see Skillman and Rempe, 2021; Sich, 2021; Mizokami and Rempe, 2021) emphasize root causes associated with one or more of these critical safety functions not being maintained. Risk assessments (Modarres and Kim, 2021) often classify accident progressions in terms of safety function failures. As discussed in Lutz and Williamson (2021d,e), the need to ensure “critical safety functions” underlie development of “symptom-based” procedures. By relating these safety functions to specific plant/control room instrumentation, operators can monitor a relatively few pieces of information to ascertain the plant status.

After the success of initial prototype demonstrations, designers explored several options to make nuclear power production more economical (Walker, 1992; Mazuzan and Walker, 1997; Okrent, 1981). One option was to build larger generating plants, decreasing the number of sites, foundations, control rooms, auxiliary equipment, and other expenses required to complete an individual power plant. Another option was to lower transmission costs, locating plants nearer to residential areas. To ensure critical safety functions were preserved with these economic benefits, designs were required to include additional safety measures, such as engineered safety features (ESFs) and instrumentation to monitor the status of these ESFs. Today, there is a combination of preventive measures to avoid accidents and mitigating measures to decrease the adverse consequences of accidents that cannot be prevented.

Defense-in-depth (DiD)

As discussed in Rempe and Williams (2021), a DiD principle is applied in reactor design, and evaluations are completed to provide confidence the design incorporates adequate measures to maintain critical safety functions. The DiD principle requires a combination of several consecutive and independent levels of protection (barriers) fail before any harmful effects to people or the environment occur. Levels of protection include a combination of factors, such as remote site selection, ESFs, safety margin in design, diverse and redundant protection systems, operating procedures and practices, management commitment to safety and security, and a robust nuclear safety and security culture. This principle is also applied in evaluations of external hazards and severe accidents. No single technical, human, or organizational failure should lead to an uncontrolled release of radioactivity (often referred to as the single failure criterion). When properly implemented, the DiD principle ensures the combination of failures that could give rise to significant harmful effects has a sufficiently low probability of occurrence.

The number of levels and specific measures incorporated into each level of DiD have evolved (NRC, 2016). More recent international references (INSAG, 1996; IAEA, 2016) identify five levels of safety for DiD. Implementation of this approach considers design, construction, and operation activities as well as accident management procedures for unexpected events. For each level of DiD, prevention and mitigation measures and the required analyses to demonstrate adequacy are highlighted in the following descriptions (IAEA, 2016).

Defense Level 1 - Prevention of abnormal operation and/or failures

The purpose of the first level ... is to prevent deviations from normal operation and the failure of items important to safety. This leads to requirements that the plant be soundly and conservatively sited, designed, constructed, maintained and operated in accordance with quality management and appropriate and proven engineering practices. To meet these objectives, careful attention is paid to the selection of appropriate design codes and materials, and to the quality control of the manufacture of components and construction of the plant, as well as to its commissioning. Design options that reduce the potential for internal hazards contribute to the prevention of accidents at this level of defense. Attention is also paid to the processes and procedures involved in design, manufacture, construction, and in-service inspection, maintenance and testing, to the ease of access for these activities, and to the way the plant is operated and to how operating experience is utilized. This process is supported by a detailed analysis that determines the requirements for operation and maintenance of the plant and the requirements for quality management for operational and maintenance practices.

Defense Level 2 - Control of abnormal operation and detection of failures

The purpose of the second level ... is to detect and control deviations from normal operational states in order to prevent anticipated operational occurrences at the plant from escalating to accident conditions. This is in recognition of the fact that postulated initiating events are likely to occur over the operating lifetime of a nuclear power plant, despite the care taken to prevent them. This second level ... necessitates the provision of specific systems and features in the design, the confirmation of their effectiveness through safety analysis, and the establishment of operating procedures to prevent such initiating events, or otherwise to minimize their consequences, and to return the plant to a safe state.

Defense Level 3 - Accident control within design basis

For the third level ..., it is assumed that, although very unlikely, the escalation of certain anticipated operational occurrences or postulated initiating events might not be controlled at a preceding level and that an accident could develop. In the design of the plant, such accidents are postulated to occur. This leads to the requirement that inherent and/or engineered safety features, safety

systems, and procedures be capable of preventing damage to the reactor core or preventing radioactive releases requiring off-site protective actions and returning the plant to a safe state.

Defense Level 4 - Control of severe plant conditions

The purpose of the fourth level ... is to mitigate the consequences of accidents that result from failure of the third level of defense-in-depth. This is achieved by preventing the progression of such accidents and mitigating the consequences of a severe accident. The safety objective in the case of a severe accident is that only protective actions that are limited in terms of lengths of time and areas of application would be necessary and that off-site contamination would be avoided or minimized. Event sequences that would lead to an early radioactive release or a large radioactive release are required to be 'practically eliminated'.

Defense Level 5 - Mitigation of radiological consequences of significant radioactive releases

The purpose of the fifth and final level ... is to mitigate the radiological consequences of radioactive releases that could potentially result from accidents. This requires the provision of adequately equipped emergency response facilities and emergency plans and emergency procedures for on-site and off-site emergency response.

A combination of systems, structures, and components (SSCs), as well as procedures and analyses, are required to achieve each DiD level. The greatest emphasis is placed on the first level of defense so plant performance is predictable during normal and abnormal conditions (NRC, 2016). As discussed in Lutz and Williamson (2021a,c), the second level of defense is achieved by the reactor protection and the instrumentation and control systems. As reported in Lutz and Williamson (2021b,d,e), ESFs and guidance for operational and beyond design basis events provide the third, fourth, and fifth levels of defense. Gauntt (2021) provides additional details regarding the analyses required for evaluating the performance of plants systems, operator actions, and measures to prevent and mitigate accident releases for each level of defense.

Regulatory framework

The regulatory framework influences the design and evaluation of ESFs. In the U.S., this regulatory framework has evolved; it includes regulatory requirements and design criteria in Title 10 of the Code of Federal Regulations (10 CFR) (U.S. Government Publishing Office, 2019) and other components, such as technical specifications, regulatory guidance, and quality assurance programs (see Bley and Malsch, 2021a,b).

After the passage of the 1954 Atomic Energy Act, the U.S. Atomic Energy Commission (AEC) confronted the task of developing regulations and licensing procedures with sufficient rigor to protect public health and safety but sufficient flexibility to allow for rapid changes in nuclear technology. Within a short time, the staff drafted rules on radiation protection, distribution and safeguarding of fissionable materials, and the qualification of reactor operators. The AEC established regulations implementing the two-step licensing process currently outlined in 10 CFR 50 (see Ray, 2021). To gain approval to proceed with construction from the AEC, applicants submitted a "Hazards Summary Report" (later renamed a "Safety Analysis Report") with available design information (which at the construction permit stage would be incomplete), a site description (including seismic, atmospheric condition, and nearby population and industrial hazard information), a detailed plan for plant operation, a schedule for chemical processing and disposal of reactor fission products, proposed methods for disposal of radioactive effluents, and a description of the reactor safety mechanisms. As described in Bley and Malsch (2021a), specific information was required about potential hazards. The applicant listed all the known potentially hazardous features and included the experimental information, calculations, and assumptions used in evaluating those hazards. The report required information on steps taken to minimize the risks and an estimate, if a failure should occur, of any release of radioactive material and the damage to be expected.

In the first decade of commercial nuclear power, the regulatory staff and their independent advisory committee, which eventually became the Advisory Committee on Reactor Safeguards (ACRS), used a case-by-case approach to evaluate construction permit applications. The technology was in its early stages and undergoing continual change; it was difficult to establish general guidelines. By early 1965, several factors, such as the expected increase in the number of new plant applications, the higher power ratings of the plants being reviewed, and recommendations from an outside review panel, led to a more formal review approach. It was hoped better documentation of AEC expectations would simplify and accelerate the licensing process. In the next several years, the AEC and later, its successor, the U.S. Nuclear Regulatory Commission (NRC), established several components of a regulatory framework that influenced the number and design of required ESFs and evaluations of their performance. This chapter describes several components within the NRC regulatory framework that significantly affect ESF design and safety analyses. Bley and Malsch (2021a,b) provide additional information about components of the NRC regulatory framework, the process used to modify this framework, and the evaluation process used to implement such modifications. Of particular relevance to this chapter, are discussions in Bley and Malsch (2021a) and Dube (2021) about the evolution of NRC regulation and guidance that limit ESF modifications or "backfits."

General design criteria (GDC)

Until 1965, there were no written criteria against which the various designs could be compared. In 1965, the AEC staff began drafting what would eventually become the General Design Criteria (GDC) initially published in 1971 as Appendix A of 10 CFR 50

(U.S. Government Publishing Office, 2019). The GDCs state broad requirements but leave the means for achieving them to the applicant. In licensing submittals, applicants must present their plans and sufficient supporting technical data to demonstrate their design would meet each of the GDC (or submit requests for exemptions with supporting information).

The GDC have been and continue to be updated as insights are gained from operating experience and analyses. The current version of the GDC address 64 broad issues grouped into six major categories:

- overall requirements
- protection by multiple fission product barriers
- protection and reactivity control systems
- fluid systems
- reactor containment
- fuel and reactivity control

The GDC are an important part of the NRC's regulatory framework. For Light Water Reactors (LWRs), they provide minimum requirements related to design, fabrication, construction, testing, and performance requirements for SSCs important to safety. The GDC can be adapted to establish principal design criteria (PDC) for non-LWRs (USNRC, 2018). The GDC serve as the fundamental criteria for the NRC staff when reviewing the ability of SSCs to perform their intended safety functions during design basis events. All production and utilization facilities licensed under 10 CFR 50 or 10 CFR 52, including both LWRs and non-LWRs, are required to describe their PDC.

Technical specifications

The GDC itemize broad objectives that applicants should address. They did not, however, spell out the nature and extent of the technical data the AEC expected applicants to supply. By listing safety information the AEC required from applicants, the Technical Specifications (or "tech specs") sought to accelerate and rationalize the licensing process while at the same time improving safety. Today, Technical Specifications are part of the facility design bases; hence, they are part of the NRC license authorizing the facility's operation. Technical Specifications establish requirements for items such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, SSC surveillance, design features, and administrative controls. Technical Specifications include: "nuclear limits" for amount of energy extracted from the fuel, value and sign of reactivity feedbacks, reactivity worth per control rod, and core axial and radial power distribution; "reactivity control limits" for shutdown margin, control rod withdrawal or insertion speeds, and emergency shutdown provisions; "thermal and hydraulic limits" for peak fuel and clad temperatures, coolant flow velocities, and thermal hydraulic stability; and "mechanical limits" for maximum stresses, fatigue, and seismic shock loads.

Guidance

The GDC do not provide quantitative bases for establishing the adequacy of any particular design. The development of more detailed regulatory guidance began in the late 1960s when the regulatory staff started generating internal documents outlining acceptable approaches for specific designs. In 1970, the AEC began publishing regulatory guides. The first published regulatory guide dealt with the concern that a loss of containment integrity should not compromise the ability of the emergency core cooling system (ECCS) to ensure its critical safety functions. This regulatory guide required sources of ECCS water be at sufficiently high pressure (provide sufficient net positive suction head) to avoid pump cavitation. Within a short time, the regulatory staff prepared regulatory guides on a wide variety of issues, ranging from thermal shock to assumptions used for estimating the radiological consequences of a loss-of-coolant accident (LOCA). These regulatory guides supplemented existing safety requirements on siting, radiation protection, and other matters. In addition, the staff issued an initial version of the Standard Review Plan (see NRC, 2019), which provides guidance for reviewing the content of regulatory applications and license amendments for LWRs. Although not regulatory requirements, these guidance documents alerted applicants of topics that had to be considered in their submittals. Applicants, as well as plant owners/operators found it easier to follow a design or analysis approach that had been prejudged by the regulator as acceptable than to defend an alternative approach. As appropriate, the NRC staff worked with professional societies to develop standards that could be endorsed by the staff in regulatory guides. As an example, consider guidance related to **Quality Assurance**.

Quality Assurance

The need to assure quality in the manufacture of SSCs and in plant construction is emphasized in the first level of **Defense-in-Depth (DiD)**; the need for this assurance took on greater importance when plant size increased. Within 10 CFR 50 can be found several Quality Assurance (QA) program requirements:

- Criterion 1: "Quality Standards and Records," in Appendix A to 10 CFR 50 requires a QA program be established and implemented.

- Appendix B “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” outlines guidance for this QA program.
- 10 CFR 50.34(a)(7) requires a description of the QA program to be applied in the design, fabrication, construction, and testing of the facility SSCs, and a discussion of how the applicable requirements of 10 CFR 50 Appendix B will be satisfied.

Similar QA program requirements were later adopted for 10 CFR 52 (see [Ray, 2021](#)), which references applicable sections of Appendix B to 10 CFR 50. Similar to the GDC, the QA criteria told applicants what to do but not how to do it. Guidance the NRC finds acceptable for a QA program, however, can be found in several regulatory guides (e.g., Regulatory Guides 1.33 ([NRC, 2013](#)) and 1.28 ([NRC, 2017](#)) and the Standard Review Plan ([NRC, 2019](#))). The regulatory staff works with professional societies to establish equipment, fabrication, and inspection codes. The staff consults with organizations, such as the ANS, the ANSI, the American Society of Mechanical Engineers (ASME), and the Institute of Electrical and Electronics Engineers (IEEE), to prepare guidelines on the reliability and testability of nuclear pressure vessels, pipes, valves, pumps, reactor protection systems, and in-service inspection of safety equipment. Special criteria were developed for fluid boundaries, for radiation containments, and penetrations to assure required safety functions are maintained. Where appropriate, testing and surveillance requirements are provided.

Safety evaluation importance example

As emphasized in [Gauntt \(2021\)](#), safety evaluations play an important role in the development and deployment of commercial nuclear power plants. Safety evaluations are essential in demonstrating **Critical Safety Functions** are maintained and in providing assurance various **Defense-in-Depth (DiD)** measures exist. As knowledge has been gained from operating experience and unexpected events, regulatory requirements and guidance for such evaluations evolved. These changes affected the design of plant SSCs. However, there are also examples in which analyses completed with updated models, based on additional experimental data, were used to revise the regulations. One such example related to ECCS regulatory requirements is highlighted here.

Background

Evaluation of the suitability of a site is an important aspect of licensing. The effectiveness and reliability of safety features, in conjunction with the distance to population centers, are important in determining the adequacy of proposed sites. Credit for ESFs was sought to allow siting of reactors at locations where, without such features, protection of the public would not be adequate (e.g., 10 CFR 100 criteria would be exceeded). Applicants attempted to get maximum credit for ESF reductions in containment pressure and radionuclide concentrations during a postulated large break LOCA (LB LOCA). The ESFs for which credit was routinely given include the leak-tight containment, the pressure suppression pool in Boiling Water Reactors (BWRs), containment building sprays in Pressurized Water Reactors (PWRs), containment heat removal systems, and containment air-cleaning systems (see [Lutz and Williamson \(2021b\)](#) for more details about ESFs found in PWRs and BWRs).

As power levels of proposed plants increased significantly, however, ACRS began to worry that a core melt could lead to a breach of containment [Okrent, \(1981\)](#). This became more of a concern not only because of the greater decay heat associated with higher power level reactors but also because the containment building volume (which partially controlled increases in temperature and pressure) was not proportionally increased with reactor power level. Hence, there was the potential the safety margin in newer plants was decreased.

Although plants are designed to withstand a spectrum of postulated accidents, the design-basis LB LOCA, whose consequences were not to be exceeded by any other credible accident, became the focus of siting evaluations. In November 1964, the ACRS discussed various safety features and their evaluations of them, emphasizing the need for additional confidence in ECCS performance as units with higher power levels were proposed for metropolitan sites (i.e., these safety features had to compensate for the lack of protection associated with distance from a population center). As discussed in [Lutz and Williamson \(2021b\)](#), one of the primary functions of the ECCS is to provide cooling (by water injection and alternate cooling systems) following a LB LOCA. Although deemed essential for prevention of core melting, the lack of data to provide confidence in ECCS performance led ACRS to recommend ECCS failure be assumed in estimating radionuclide releases postulated for siting.

New test data

In response to concerns about ECCS performance, the AEC initiated a testing and analyses program to be primarily conducted at what is now known as the Idaho National Laboratory (INL). This test program included several smaller scale preliminary tests at the Semiscale facility followed by larger scale, integral tests at the Loss-of-Fluid Tests (LOFT) facility.

In the summer of 1970, small-scale ECCS test results and analyses prompted concern among researchers ([Walker, 1992](#)). Several small-scale experiments conducted in Idaho and other locations indicated that in the event of a LB LOCA, core temperatures could rise higher than previously believed and the coolant from the ECCS might not reflood the core as rapidly as expected. In November and December of 1970, results from the initial Semiscale tests caused more concern. These tests were performed using a ~23 cm high core (compared to reactors with ~366 cm high cores) in a vessel with one set of inlet and outlet pipes. Their purpose was to study the effectiveness of ECCS under accident conditions. The experiments were run by electrically heating the simulated core,

allowing cooling water to escape, and then injecting emergency cooling water. Rather than cooling the core, the high steam pressures within the vessel resulted in about 90% of the injected water directly exiting through the outlet pipe. In January 1971, Idaho researchers reported: “Preliminary analysis of the results of these tests indicates little or no cooling by the emergency coolant” (INC, 1971).

The size, scale, and design of the Semiscale test, in particular the channels directing the flow of coolant in the test model, differed from those of an actual reactor. Unlike commercial plants, which had two to four coolant loops, the test equipment used a single loop. This increased the possibility of emergency coolant bypassing the core and flowing out the break in the single loop rather than reaching the core through other paths. Nevertheless, the unanticipated Semiscale test results raised safety concerns. To initially address these concerns, the AEC published “interim acceptance criteria” in 1971 (USNRC, 2002) until further research could be completed. These criteria required additional maintenance and monitoring in addition to changes in the ECCS designs of some operating reactors. These interim ECCS criteria were similar to what would later be promulgated in 10 CFR 50.46 (see **Regulatory framework modifications**).

In 1973, a second series of experiments was completed. These later tests were called 1½ Semiscale because a loop simulating the unbroken loops of a reactor was added to the ½ (broken) loop. This time, water was injected through the unbroken loop, as would occur during ECCS injection within actual power reactors with two, three, or four loops. As predicted by computer models, the simulated core was successfully cooled in all 1½ Semiscale tests with steam escaping through the broken loop.

Regulatory framework modifications

In 1974, the NRC set up safety requirements based on DiD in 10 CFR 50.46 and provided guidance in 10 CFR 50 Appendix K for models and simulations to ensure conservatism in predictions. These methods resulted in both conservative safety margins as well as economic and operational penalties. In the late 1970s and early 1980s, a large number of experiential programs provided additional data for developing best estimate models. Eventually, these models were used to justify additional changes to ECCS regulatory requirements.

First promulgated in 1974, ECCS requirements include 10 CFR 50.46(b) acceptance criteria:

- (1) *Peak cladding temperature (PCT)*. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F (1200 °C).
- (2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- (3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding surrounding the fuel were to react.
- (4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- (5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period required by the long-lived radioactivity remaining in the core.

The AEC also provided guidelines, which are now found in Appendix K of 10 CFR 50, for ECCS evaluation models for satisfying 10 CFR 50.46(b) acceptance criteria. These guidelines were created to ensure the predicted safety parameters were conservative. Furthermore, if simulations demonstrated acceptance criteria were met, the guidelines provided assurance there would be large margins to actual failure limits. The AEC and its successor, the NRC, reviewed ECCS designs of every operating plant. When necessary, ECCS retrofits and upgrades were required or the operating power level was reduced to assure compliance with the criteria. Indian Point 1 was shut down in October 1974 because of an inadequate ECCS. All new plants and plants under construction were required to meet the criteria. Although the Appendix K methods were very conservative, these methods also led to predictions of non-physical behavior during the transition from depressurization to reflood phases occurring during a LB LOCA (Hochreiter et al., 1992; Rohatgi and Kaizer, 2020). There was concern evaluations were predicting an entirely different sequence of phenomena than what would be expected during an actual event.

To understand how the use of an Appendix K approach affects calculation results, it is helpful to consider Fig. 2 (Rohatgi and Kaizer, 2020). There is a safety limit (such as PCT) above which the fuel cladding will become damaged. A regulatory requirement (or acceptance criterion) is placed below this safety limit; if this regulatory requirement is met, damage is precluded. For a specific safety limit, there is a predicted value for what would occur during an accident (e.g., LB LOCA). However, it is recognized estimates for this value are uncertain (due to uncertainty in analysis models and their input). The Appendix K approach is to quantify these uncertainties using engineering judgment and to choose bounding values for model input parameters. Thus, the Appendix K approach should bias analysis to make predictions conservative with respect to the regulatory requirement (i.e., leading to the “Conservative Margin to Acceptance Criterion” shown in Fig. 2).

There can be complications with an Appendix K approach. First, conservative biases are dependent on the selected safety limit. If the figure of merit changes (e.g., from PCT to peak containment pressure), initially conservative assumptions may become non-conservative. Second, it may be difficult to discern the conservative direction for some model input parameters. While some figures of merit may linearly increase or decrease with certain parameters (e.g., decay heat), reactor analysis tends to be non-linear and the “worst case” for many parameters may not be easily understood. Third, results from a conservative analysis may be

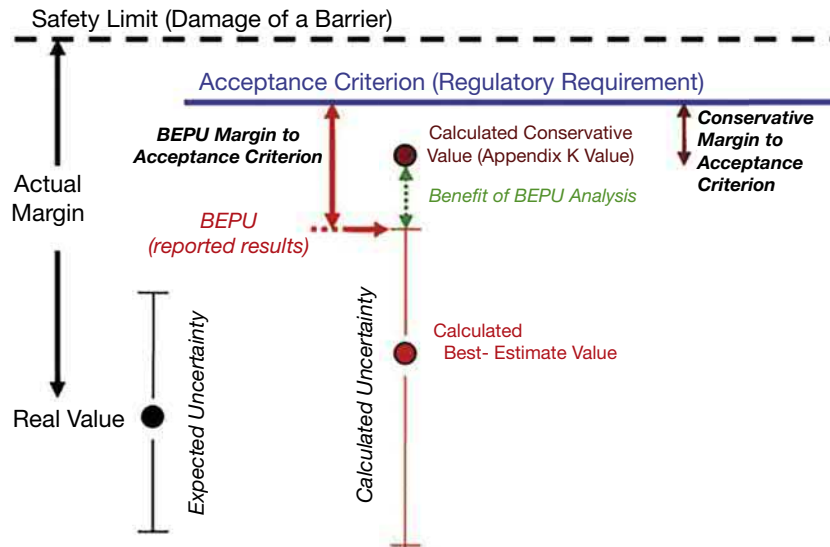


Fig. 2 Comparison of uncertainties and safety margin obtained with Appendix K and Best Estimate Plus Uncertainty (BEPU) approaches. From Rhatgi and Kaizer (2020) Historical Perspectives of BEPU research in US. *Nuclear Engineering and Design*, 358, March 2020.

non-physical, thus limiting any obtained insights. Hence, results showing less margin to regulatory limits may lead to unnecessary penalties on reactor operation.

After results were obtained from the Semiscale tests, several independent assessments of ECCS criteria were completed. For example, the NRC sponsored additional experiments to investigate individual phenomena and system performance and funded development of advanced computer codes for improved simulations of LOCAs. These experimental and computational efforts provided the technical basis for a revised rule for ECCS performance. The rule was approved by the NRC in September 1988. The revised rule retains the acceptance criteria based on PCT, cladding oxidation, and hydrogen generation; however, it allows the use of best-estimate computer codes for evaluating these parameters. If a Best Estimate plus Uncertainty (BEPU) analysis is performed, the revised rule requires calculation uncertainties be quantified and included when comparing predicted results with the acceptance limits provided in 10 CFR 50.46(b). As shown in Fig. 2, results from a Best Estimate Plus Uncertainty (BEPU) analysis provide a larger, more realistic estimate of plant safety margins (the “BEPU Margin to Acceptance Criterion”), leading to additional margin compared to results obtained with the Appendix K approach (the “Benefit of BEPU Analysis”). This additional margin can be used to reduce unnecessary financial penalties, such as limits on fuel discharge burnup, and allow more flexible plant operation, such as relaxations in equipment technical specification requirements.

Summary

Nuclear power plant safety is assured by maintaining three critical safety functions. This chapter provides an overview of the DiD approach used to ensure these safety functions are maintained. This approach, which is enforced by a regulatory framework, emphasizes measures implemented in reactor design and evaluation activities. This chapter also includes an example illustrating the use of evaluations to justify changes in the regulatory framework. In this example, regulations were modified to allow more realistic calculations for demonstrating compliance with core cooling requirements.

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Reactor Protection System

Robert Lutz^a and Bill Williamson^{b, a} ^a Lutz Nuclear Consulting, Hendersonville, NC, United States; and ^b TVA, Nuclear Plant Road, Athens, AL, United States

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Glossary

Analog A continuous signal that represents a time varying quantity. An example is the signal from an instrument sensor which indicates the parameter value is continuously transmitted as it varies with time.

Anticipated Transient Without Scram (ATWS) The failure of the control rods to fully insert in response to a condition in which automatic reactor scram is expected to occur.

Bistable The bistable is an electronic switch with an adjustable on/off setpoint. Within the bistable, a signal from the process sensor is compared to a preset, adjustable setpoint. When the process signal equals or exceeds the setpoint value, the bistable's output is turned off (de-energized), and its output voltage goes to zero (the bistable is tripped). This electronic device or switch thus converts the analog (variable) input signal into a digital (on-off) output signal.

Logic The comparison of input signals, such as signals from instrumentation sensors, to determine whether an actuation of a protective device should occur. An example of logic is 2-out-of-4 inputs indicating that a given plant parameter exceeds a setpoint required for actuation; if only one-out-of-four signals exceeds a setpoint, no actuation will occur.

Moderator Temperature Coefficient (MTC) The effect of a change in the moderator temperature (water in case of light water reactors) on the core reactivity. A positive MTC implies an increase in reactivity due to water temperature increase—an unstable and therefore undesirable situation.

Reactivity A term expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Reactor Scram Automatic insertion of all control rods into the reactor core.

Setpoint A predetermined value at which an action is expected to occur.

Solid State An electronic device in which electricity flows through solid semiconductor crystals rather than through vacuum tubes.

Introduction

In many plant transients, the control system acts to return the reactor to its normal operating condition. However, the more severe transients require that the reactor be shut down by the Reactor Protection System (RPS). The RPS is designed to provide a highly reliable means of achieving reactor shutdown in the event of one of these severe transients. This is accomplished by the RPS monitoring certain parameters and either automatically responding or the operators manually shutting down the reactor. These operator

actions may be accomplished in the control room or from a different location within the plant in which a remote shutdown panel is located. Specific plant parameters monitored for Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs) are identified in **Reactor Protection System**. Plant technical specifications are developed for the RPS instrumentation to assure their operability and accuracy by means of various surveillance requirements.

Throughout this article, the words trip and scram are used to describe the process of automatically shutting down the reactor in off-normal or accident conditions. “Trip” is typically used for PWR plant, while “Scram” is typically used for BWRs.

This article first describes the regulatory requirements for reactor protection, including changes required to address concerns related to events with failures to trip or scram the reactor. Next, representative descriptions of the mechanics of operation of the control rod system in three PWR vendor designs and a General Electric BWR design are provided. This is followed by a discussion of scram failures and anticipated transient without scram (ATWS) events. In response to failures of reactor scram systems at operating plants, additional regulatory requirements were mandated by the US Nuclear Regulatory Commission (NRC) to further assure that the reactor could be shutdown safely. Finally, RPS designs for typical PWR and BWRs, including the logic and instrumentation associated with reactor protection, are described.

Regulatory requirements

As discussed in [Rempe \(2021a\)](#), the predecessor of the NRC, the Atomic Energy Commission, issued a set of General Design Criteria (GDC) for light water-cooled nuclear power plants in what is now found in Appendix A of 10 CFR50 ([NRC, 2020](#)). The GDC establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety. That is, those SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. GDC 20 through 29 are particularly relevant to reactivity control which is undoubtedly the highest priority to assure safe operation of nuclear power plants. These criteria include requirements for:

- Automatic actuation of appropriate systems, including reactivity control systems,
- High reliability and testability so that no single failure results in the loss of system function,
- Effects from natural events, plant operation and accidents to not cause a loss of protection system function,
- Any failure of a protection system to place the plant in a safe state,
- Control systems and protection systems to be separated to the extent possible,
- Protection systems to accommodate a single failure of the reactivity control system,
- Provision for two independent means with different design principles for reactivity protection systems,
- Safe shutdown during accidents by a combination of reactivity control systems and poison injection,
- Safe fuel limits to be maintained by considering the potential amount and rate of reactivity insertion during an accident, and
- A high probability of success of protection systems and reactivity control systems to accomplish their function for anticipated operational occurrences.

In addition, reactivity control systems are subject to regulatory requirements imposed by the NRC in response to events occurring at operating plants (see **Anticipated Transient Without Scram (ATWS)**).

Mechanics of reactor scram system operation

PWR

In PWRs, the control rods are inserted and withdrawn from the top of the reactor vessel (RV) ([Brown, 2021](#)). The control rods are separated into two functional categories: shutdown rods and control rods. Each category consists of several individual rods arranged in banks which are moved together. The shutdown banks are always in the fully withdrawn position during power operations and are moved into this position as the reactor approaches criticality. These shutdown banks provide a large negative reactivity insertion upon reactor trip to ensure the reactor achieves and maintains subcriticality. During plant operation, the position of the control rod banks is changed using the rod control system. A control rod drive (CRD) mechanism is used to withdraw or insert the control rods.

Westinghouse PWRs: The control rod drive mechanism consists of latching assemblies (grippers), the rod drive shaft assembly, and the magnetic coil stack. Every single rod has its own drive mechanism. The primary CRD mechanism is an electromagnetic jack. Movable and stationary gripper latches engage each rod drive shaft when its respective coils are energized. The movable grippers are used to raise or lower the CRD shaft when the lift coil is energized or de-energized. The stationary grippers are used to hold the drive shaft in position when rod motion is not required. A reactor trip is achieved by simply removing the electrical power from the coils, which allows springs to disengage the grippers from the rod drive shafts, and the rods fall by gravity into the core ([NRC, 2009a](#)).

Babcock & Wilcox (B&W) PWRs: The primary components of the CRD mechanism consist of an electrically driven, roller nut assembly, a stator, and a leadscrew that converts the rotary motion of the roller nut assembly to the linear travel of the leadscrew and control rod assembly. As the power to the stator coils is sequenced, the roller nut assembly rotates to a new position. The drive is designed to trip whenever stator power is interrupted. The power loss causes the rotor assembly segment arms to pivot, disengaging

the roller nut assembly from the leadscrew and allowing the leadscrew and control rod assembly to fall into the reactor core (NRC, 2011b).

Combustion Engineering (CE) PWRs: The control element drive mechanism consists of a series of gripper latches that fit into grooves in the control element assembly drive shaft. The grippers are magnetically operated, and they position the control element assembly during withdrawal and insertion. Electromagnetic coils, located outside the pressure housing containing the control element assembly, are used to hold and move the control element assembly. The electromagnetic coils become inactive when electrical power is interrupted. The power loss causes the spring-loaded grippers to disengage from the grooves and the control rod assembly to fall into the reactor core (NRC, 2008a,b).

BWR

For a BWR, the control rods are inserted and withdrawn from the bottom of the reactor pressure vessel (RPV) (Harmon, 2021). During plant operation, the position of the control rods is changed using the reactor manual control system. This system opens and closes valves in the CRD system, which provides pressurized water to the Hydraulic Control Units (HCU) on each control rod. The HCU opens and closes directional control valves that control rod movement.

Each of the individual control rods is equipped with two scram solenoid valves. Fig. 1 provides a schematic of the operation of the BWR RPS. When no full scram condition is present, the scram air header is pressurized from the plant instrument air system to maintain the spring loaded HCU scram inlet and outlet valves closed and the spring loaded scram discharge volume (SDV) vent and drain valves open.

When a reactor scram is initiated by the RPS, the scram pilot air header is isolated and depressurized. This allows spring pressure to open the inlet and outlet scram valves on each HCU and close the (SDV) vent and drain valves. Pressure from the reactor vessel and from the scram accumulator is applied to the area below the CRD while the water above the control rod is allowed to flow to the SDVs. The large differential pressure applied to the control rod drive rapidly inserts the control rod upward into the core.

In addition to the scram pilot solenoid valves, there are two backup scram solenoid valves. The backup scram valves control the air supply to the entire scram pilot air header. When energized the valves reposition, isolating and venting the header. Venting of the scram air header would open any individual scram pilot valve that otherwise fails to open. This provides a redundant means of venting the scram air header and causing rod insertion.

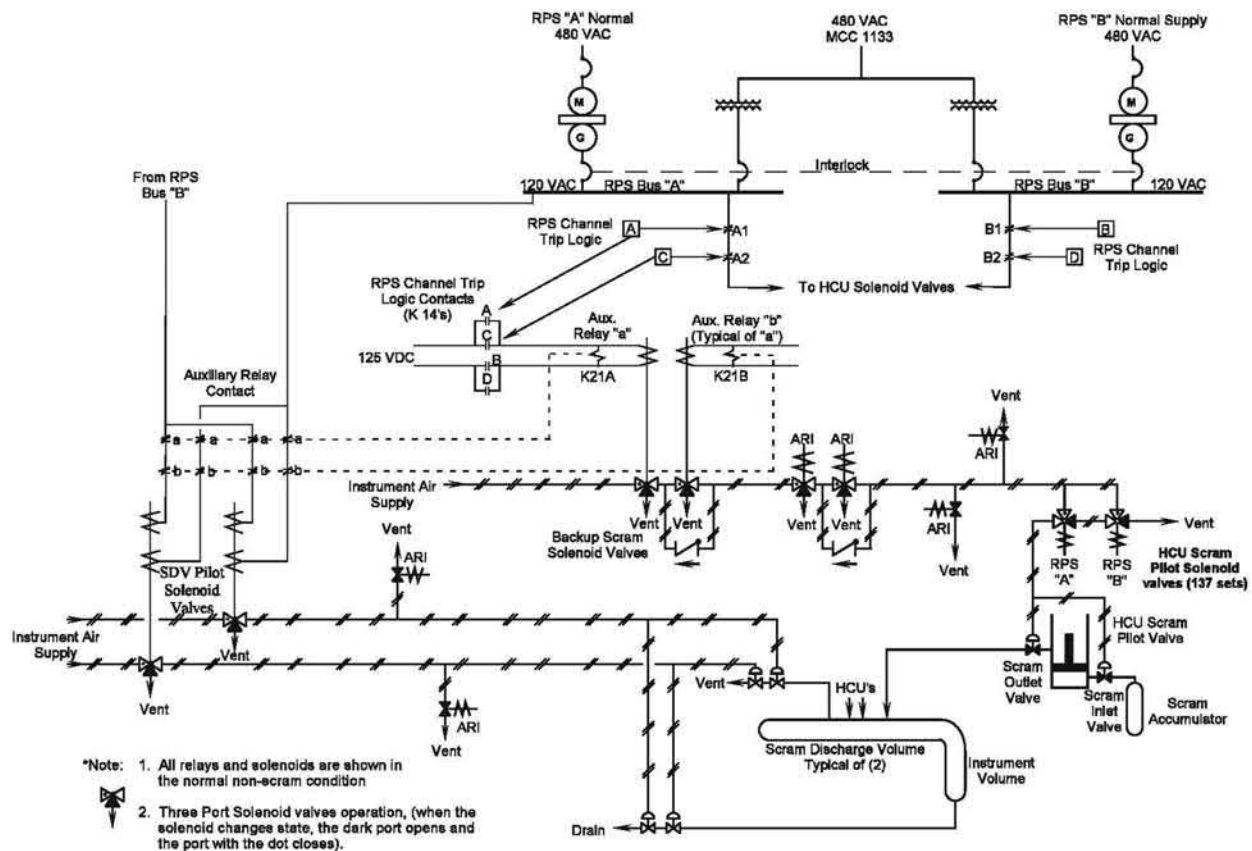


Fig. 1 Simplified BWR reactor protection system. Taken from Fig. 7.3-5 of NRC (2011a) General Electric Systems Technology Manual. Chapter 7.3: Reactor protection system. US NRC Human Resources Training Development (HRTD). Available from <https://www.nrc.gov/docs/ML1125/ML11258A345.pdf>

There are also two SDV pilot solenoid valves that function in much the same way as the HCU scram pilot solenoid valves to close the SDV vent and drain valves on a full scram signal. Upon a full scram condition, they are de-energized to vent the air holding the SDV vent and drain valves open.

Anticipated Transient Without Scram (ATWS)

An Anticipated Transient Without Scram or ATWS event is important to safety because the failure to trip can result in damage to the reactor core due to overheating when the coolant is unable to provide sufficient heat removal. Another concern is the potential for failure of the RPV, and in BWRs, the containment, because of the excessive pressure increase associated with core overheating.

Prior events

ATWS events occurred at Salem Unit 1 in 1982 (two nearly identical events that occurred 3 days apart) (NRC, 1983; Rempe, 2021b). Previously, a failure of a manual scram occurred at Browns Ferry Unit 3 in 1980 (INPO, 1980; NRC, 1980). In 1988 (NRC, 1988), a thermal hydraulic instability event occurred at LaSalle Unit 2. In each of these events, the plants were put into a safe configuration by the licensed operators, and fuel failures were prevented.

Following the failure to scram at Browns Ferry Unit 3, BWR plant operators were required to establish two SDV tanks (one for each bank of control rods) to mitigate the possibility of a hydraulic lock. A hydraulic lock could prevent control rods from inserting if the SDV tank is filled with water and causes blockage. This was the cause of the failure to scram at Browns Ferry Unit 3.

Following the Salem event, Westinghouse PWR plant operators were required to provide redundant means of automatically opening the trip breakers. The original Westinghouse design provided only one means by which the RPS would open the reactor trip breakers. In the aftermath of the Salem events, trip breaker operation was modified so that the RPS de-energizes both the trip breaker and the bypass breaker when trip logic is satisfied (see [Reactor Protection System](#)). The bypass breaker is used to allow testing of the trip breaker in each train during power operation without actually causing a reactor trip.

Following the LaSalle Unit 2 instability event in 1988, the BWR industry added measures to protect the reactor from power and flow instabilities and the possibility of fuel damage due to unmitigated power and flow oscillations. Some of these measures include additional trip signals as part of the Average Power Range Monitor (APRM); for example, addition of an Oscillation Power Range Monitor (OPRM) which scrams the reactor on indications of power oscillation. Also, the operating domain was limited, depending on the power density of the reactor. Additional modifications to these systems have also been required as part of licensee requests for increases in rated power and for additional flexibility with respect to the range of power and flow conditions in which the reactor may be operated.

As discussed in [Gaunt \(2021\)](#), licensees are now required to analyze ATWS events. These beyond-design basis scenarios are analyzed to evaluate the adequacy of proposed additional countermeasures to assure that the event can be safely terminated, and the plant brought to a safe, stable condition.

ATWS rule and plant modifications

In response to the failure to scram event at Browns Ferry Unit 3, the ATWS events at Salem Unit 1, as well as analyses of transients considered in probabilistic risk assessments, several NRC actions were initiated (Rempe, 2021b). In 1984, the NRC published 10CFR50.62, which is commonly referred to as “the ATWS rule” (NRC, 1984). This rule imposed new equipment requirements for all light water reactors, several which directly impacted the RPS design. The basis for the changes required by the ATWS Rule were based on analyses that showed a disproportionate risk for some plant designs. The basis for these changes, as well as typical modifications implemented within PWRs and BWRs are described here.

PWR

The ATWS rule required that some hardware modifications and operating limitations be implemented at all PWRs. However, the specific required changes varied amongst the three PWR reactor vendor designs discussed in this article (Westinghouse, CE and B&W).

CE and B&W designed PWRs were required to have a second, diverse trip system. The system was required to be independent of the RPS from the sensor outputs to the control rod drive mechanisms (CRDMs) power supplies. The diverse trip system has been imposed as a defense-in-depth measure for CE and B&W plants because of the relatively high percentage of fuel cycle time those plants are expected to operate with positive or only slightly negative Moderator Temperature Coefficient (MTC). This is also referred to as Unfavorable Exposure Time (UET). This was done in accordance with analyses performed by the reactor designers at the time of the ATWS rule. If an ATWS occurs during a period of UET, analyses predicted that existing mitigation systems are likely to be ineffective to limit the reactor power and the associated increase in reactor coolant pressure. Westinghouse designed plants have been excluded from this requirement because of their larger pressurizer safety valves and relatively smaller expected percentage of fuel cycle time with a positive or only slightly negative MTC (NRC, 2008a).

The original Westinghouse design provided only one means by which the RPS would automatically open the reactor trip breakers: de-energizing of the undervoltage coils. As noted in [Prior Events](#), trip breaker operation was modified to provide redundant means of automatically opening the breakers after the ATWS events at Salem Unit 1.

For all PWR plants, an ATWS Mitigation System Actuating Circuitry (AMSAC) was installed that starts the auxiliary feedwater pumps and trips the main turbine if indications of a loss of feedwater are present (i.e., steam generator levels drop below a low

level setpoint or the main feedwater flow drops below a predetermined setpoint). The AMSAC circuitry is only functional when the power level is greater than 40% of nominal.

The pressure relief capability was found to be important in preventing reactor coolant system (RCS) overpressurization during periods of UET for Westinghouse plants. Therefore, a Technical Specification was developed that limited operation with pressurizer power operated relief valves blocked during periods of UET.

BWR

The ATWS rule also required a number of changes to BWR systems and equipment (NRC, 2010). To increase the reliability for reactor scram, the rule required that an Alternate Rod Insertion (ARI) system be added that is diverse from the reactor trip system. This ARI system vents air from redundant scram valves to ensure that control rods are inserted.

A transient with failure to scram from full power is most severe for scenarios in which the reactor is isolated from the main turbine (e.g., after the turbine trips or after the main steam isolation valves (MSIVs) close). If the recirculation pumps continue to run, the power level will remain high and a severe pressure excursion will take place. Even if the RCS survives the pressure surge, the very high steam flow will rapidly heat the suppression pool and pressurize the containment. In addition, the High-Pressure Coolant Injection (HPCI) System (BWR/2–4 plants) or High Pressure Core Spray (HPCS) System (BWR/5–6 plants) may not suffice to cool the core: overheating and fuel damage may follow. Ultimately, the containment is expected to rupture due to over pressure while the core sustains damage. Continued core coolant replenishment is questionable after containment rupture. A large radiological release is a plausible outcome. Therefore, changes were mandated.

One necessary mitigating feature identified is a prompt automatic trip of the recirculation pumps to avoid the pressure excursion and diminish the power associated steam flow to the suppression pool. The rule required that additional logic be added to trip the recirculation pumps under ATWS conditions (high RPV pressure, low RPV water level). Given a trip of the recirculation pumps with no (or only a few) control rods inserted, the reactor power will stabilize at roughly 30% power until the reactor coolant boils down and steam bubbles (voids) form in the core. Thereafter, an oscillatory equilibrium will be maintained in which the reactor sustains the average power (up to about 30% power) necessary to boil off as much reactor coolant as is delivered. Analysis shows that HPCI/HPCS or main feedwater can adequately cool the core to avoid extensive fuel damage. However, if the reactor is not tripped, the energy delivered to the suppression pool will be greater than its cooling system can dissipate, resulting in an increase in pool temperature. If the suppression pool temperature increase is not mitigated, core overheating or containment over-pressurization could occur.

Containment over-pressure failure remains a distinct possibility unless the reactor is shutdown, either by control rod insertion or by liquid reactivity poison injection. Well before the containment is significantly pressurized, the suppression pool will approach saturation and steam condensing will begin to cause oscillations (also referred to as chugging or dynamic instabilities) (Marks and Andeen, 1979). These condensation oscillations create pressure impulses and dynamic loading of structures which may threaten containment integrity or pressure suppression structures, shortening the time available to safely shutdown the reactor. Also, because the HPCI/HPCS is a single-train system, the fault or human error that precipitates the initial transient might disable the HPCI/HPCS resulting in a loss of all emergency coolant injection capability.

In addition, system reliability analyses have indicated that HPCI may fail or be unavailable in as many as 1% of the cases in which a demand is made of the system. This may be insufficient reliability for mitigating a potentially serious accident having a frequency of occurrence that might be as high as once in a thousand reactor years. A second diverse system, the Reactor Core Isolation Cooling (RCIC) System should be expected to auto start and run, delivering coolant to the reactor. If RCIC is the sole operative means of replenishing reactor coolant, the adequacy of core cooling, rather than the heat deposited in the suppression pool, is likely to be the factor limiting the time allowed to safely shut down the reactor.

The RCIC can successfully cool the reactor once it is shut down, and it can slow reactor coolant boiloff. The two trains may be the HPCI/HPCS and RCIC. BWR/2 designs and older BWR/3 designs do not have RCIC systems. Instead, these designs have isolation condensers (either one or two depending on the design), which through pressure and gravity feed, can remove energy from the RPV on the order of what RCIC injection can provide.

The ATWS rule required that each BWR has a Standby Liquid Control (SLC) system with procedural actions for manual initiation to ensure injection of poison at a satisfactory rate (equivalent to 86 gpm/19.5 m³/h of naturally occurring Boron-10). The NRC has concluded that the liquid reactivity poison injection system in BWRs must have a start time and poison injection rate such that either of two redundant trains of high pressure reactor coolant replenishment systems, either of which may be expected to be available under ATWS conditions, can successfully mitigate ATWS transients. Some BWRs had SLC pumps that could inject at that rate, while other BWRs increased the Boron-10 enrichment of the solution used in the SLCS system to provide the equivalent rate of boron injection.

Reductions in unplanned scrams

Finally, for both PWRs and BWRs, a scram reduction program was initiated in 1985 by the industry (under the auspices of the Nuclear Utility Management and Reliability Committee) to reduce the annual number of unplanned plant scrams in order to reduce the demands for automatic scram features (NRC, 2000a,b). This program was successful in reducing the annual average annual number of unplanned scrams from over 5 to less than 0.5 between 1985 and 1998 (NRC, 2000a). As stated in NRC (2000b), the BWR Owners Group goal is less than 0.7 unplanned scrams per year. This was primarily achieved through improved

maintenance policies and training to reduce scrams caused by plant maintenance and surveillance activities while the plant is at power and by moving some maintenance and surveillance to when the plant is shutdown.

A decrease in unplanned scrams reduces challenges to the safety equipment and increases plant availability for electricity production. A major benefit is a reduced chance of damage to plant equipment due to operation in undervoltage conditions. Unplanned scrams create a transient on the offsite power grid which can result in significant voltage changes; undervoltage is a significant threat to operating electrical equipment.

Reactor protection system

Designs for the RPS, which include the scram systems have evolved with time. Descriptions of typical designs, which include the modifications discussed in **Anticipated Transient Without Scram (ATWS)**, found in Westinghouse PWRs and General Electric BWRs are provided here.

PWR

RPS designs differ amongst the Westinghouse designed plants. The most common designs are the relay protection system and the solid-state protection system. Either of these systems performs the same functions, with the solid-state protection system being more recent. Some plants have upgraded portions of their solid-state protection systems to include an innovative feature, an on-line, self-testing protection system. Using solid-state devices, this system checks the entire protection system continuously, regardless of the operational condition of the reactor.

The RPS solid-state design has significant improvements over earlier designs. Approximately 750 relays with 4000 contacts connected in various matrices were contained in the earlier RPS designs. Around fourteen 30-in. (76 cm) wide by 30 in. (76 cm) deep cabinets were required to accommodate these relays and contacts. The newer solid-state design only requires six of these cabinets. The addition of the self-testing circuit reduces the test time for the logic section of the protection system, the portion of the system that controls its response, from 4 h per train to approximately 1 h per train.

Fig. 2 shows a simplified diagram of a typical solid-state RPS for actuating a reactor trip or an engineered safety feature (ESF). Note that while every set of signals that generates an ESF actuation also generates a reactor trip, not every reactor trip generates an ESF actuation. This system, similar to the analog RPS, is comprised of two redundant, identical trains (A and B) that are physically separated and electrically independent. Inputs into this system are derived from various nuclear and non-nuclear sensors located both inside and outside of the containment building. Most of these signals are processed in the analog cabinets and result in bistable (B/S) outputs (i.e., the device can be in either an “on” or “off” position; 125 V ac when on, or zero volts when tripped) to the solid-state logic cabinets. Other protection signals are derived directly from the status of contacts at sensors or components (examples are oil pressure switches on the turbine, auxiliary contacts on circuit breakers, and limit switches on valves).

Logic

As shown at the top of **Fig. 2**, power to the CRDMs is supplied by motor generator sets through two series of reactor trip circuit breakers (“Train A” and “Train B”). The opening of either reactor trip breaker de-energizes all CRDMs, and the reactor trips by allowing the control rods to fall into the core by gravity. This trip scheme is an example of one-out-of-two logic; it requires the actuation of only one protection train and the opening of only one reactor trip circuit breaker, at a minimum, to trip the reactor.

While this logic provides redundant means for generating a reactor trip, reactor trip circuit breaker testing would scram the reactor without some compensating measure. Because testing is required, the RPS design includes bypass breakers for each train (“A” and “B” bypass breakers in **Fig. 2**) that are manually racked in (i.e., connected to the trip circuit) during testing. For example, if a test of the Train A reactor trip breaker is required, the “A” bypass breaker is racked in to provide a circuit path parallel to that of the Train A trip breaker to ensure continuity of power when the trip breaker is opened. The Train A bypass breaker is opened by the Train B protection train; therefore, during testing, the RPS is reduced to one-out-of-one logic.

As shown on the left of **Fig. 2**, the protection system consists of a number of analog channels for a given parameter. The analog section receives input signals from transmitters in each channel that sense process parameters. Each process signal for each channel is compared to a setpoint in this bistable (B/S) device. If the monitored parameter’s input signal is equal to or exceeds the setpoint, a trip signal for that channel is generated by the bistable. The bistable trip signal is sent to redundant logic cabinets (“A” and “B” trains), where the reactor trip actuation signals are generated.

Each reactor trip function (i.e., the parameter that can generate a reactor trip) has a coincidence network. Coincidence refers to the condition that more than one bistable trip signal for a given parameter needs to be present. In the center left of **Fig. 2**, only one such coincidence network is shown in each protection cabinet. In these cabinets, there are four channels monitoring a plant parameter, and the coincidence network has a two-out-of-four logic (i.e., input from at least two out of the four bistables/channels must be zero volts a.c.). For example, assume that the instrument transmitters are supplying pressurizer pressure signals. If instrument channel 1 (shown in red) senses a high-pressure condition (greater than the high-pressure trip setpoint), its associated bistable trips (i.e., supplies zero voltage to the logic cabinets). Both logic cabinets receive this signal and open the contacts associated with this channel. If no other contacts are open in this logic matrix, the undervoltage coils remain energized, and no trip occurs. If another

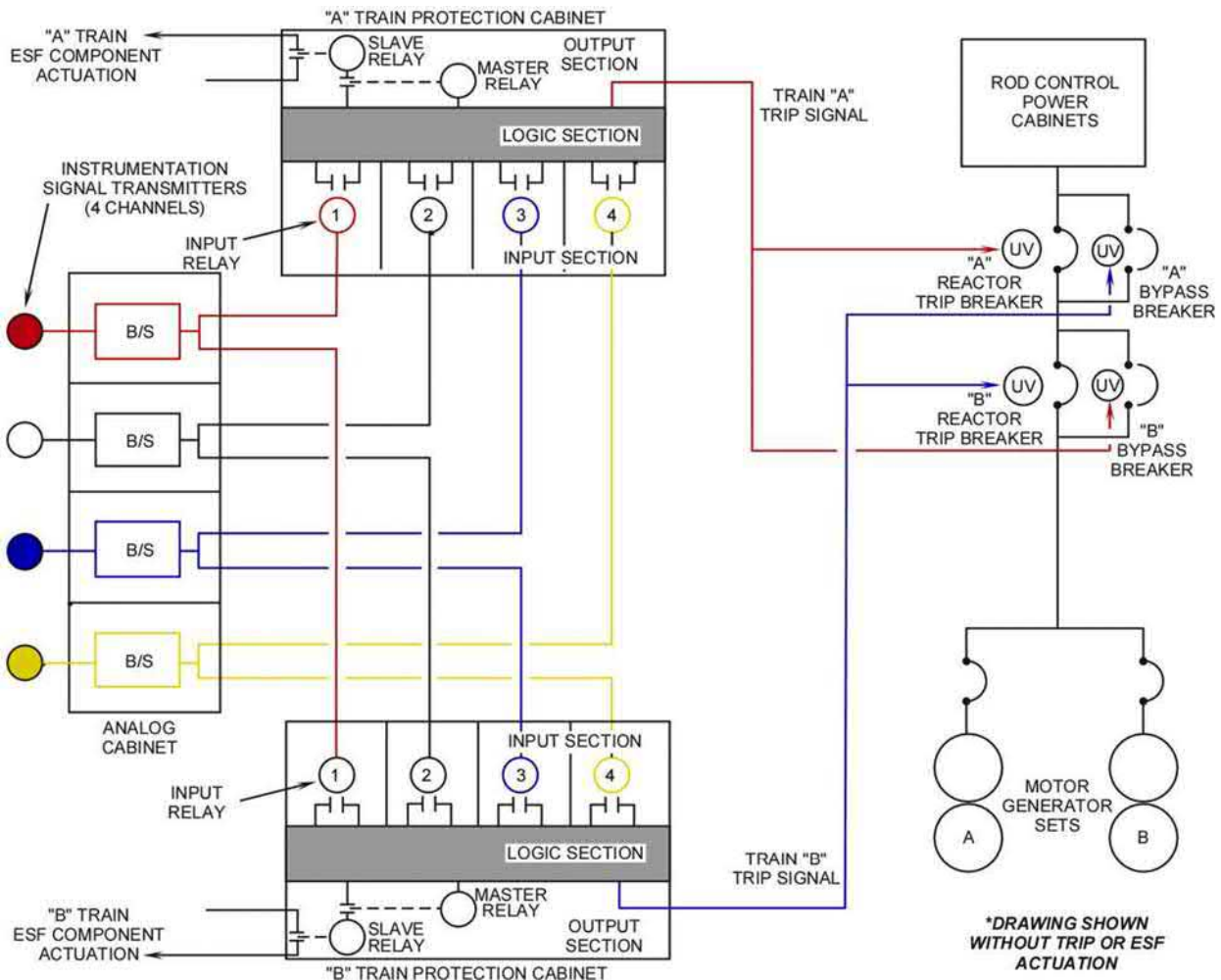


Fig. 2 Typical solid-state Westinghouse PWR reactor protection system. Taken from Fig. 12.1-2 of NRC (2009a) Westinghouse Technology Systems Manual. Section 12.1: Reactor protection system. US NRC Human Resources Training Development (HRTD). Available from <https://www.nrc.gov/docs/ML1122/ML11223A401.pdf>

transmitter also indicates a high-pressure condition, its associated bistable trips; thus, the necessary two-out-of-four logic is satisfied. When this occurs, vital power is interrupted to the undervoltage coils, allowing the reactor trip breakers to open.

Fig. 3 shows a simplified diagram of a typical relay protection system. This system features a significant number of relays for each protection function input as well as added complexity due to the test features. In **Fig. 3**, the red channel (Red I) is shown from the sensor of some parameter to the inputs to the logic cabinets. If this channel senses a pressure in excess of the setpoint value, its associated bistable trips, causing the bistable's output voltage to go to zero. With the bistable output at zero, the red input relays de-energize in both the Train A and Train B logic cabinets. When the input relays de-energize, the red contact labeled "1" opens in the logic matrix of Train A, and the red contact labeled "A" opens in the Train B logic matrix. The logic coincidence for this particular trip function is two out of four. Therefore, two channels must de-energize to produce a reactor trip. Note that even with the 1 and A contacts open, power is still delivered from the 125-Vdc battery buses to the undervoltage (UV) coils for the reactor trip and reactor trip bypass breakers.

A reactor trip does not occur unless at least one of the other three channels also senses the same condition, its associated bistable trips, and the associated input relays de-energize. With any two sets of logic matrix contacts open, power is interrupted to the undervoltage coils of the reactor trip breakers, causing them to open.

The logic cabinets receive the signal inputs from the protection bistables (either on or off). The bistable output signals provide the protection system inputs for all reactor trips and ESF actuations. Energized input relays 1, 2, 3, and 4 (for Train A) or A, B, C, and D (for Train B) hold their associated contacts closed, thereby maintaining continuity of power to the undervoltage coils of the reactor trip breakers. If an undervoltage coil de-energizes as the result of bistable trips, incorrect testing, or any other cause, one of the series connected reactor trip breakers opens, allowing all shutdown and control rods to fall into the core. The relay RPS is tested in a segmented fashion; the "push buttons" (e.g., PB1, PBA, etc.) shown in **Fig. 3** are a means to test individual parts of the logic when the reactor is at power without generating a reactor trip signal.

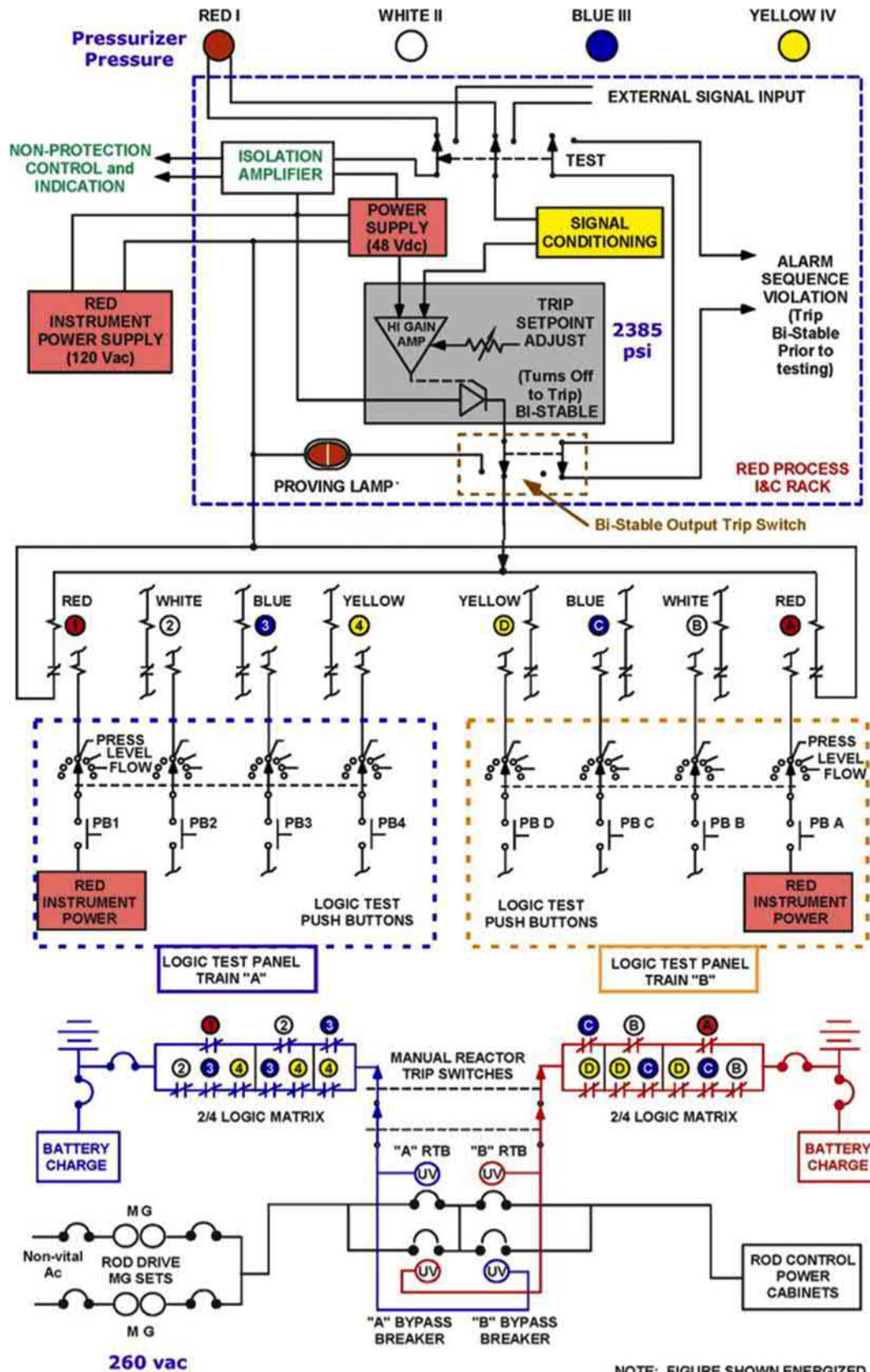


Fig. 3 Typical relay Westinghouse PWR reactor protection system. Taken from Fig. 12.1-1 of NRC (2009a) Westinghouse Technology Systems Manual. Section 12.1: Reactor protection system. US NRC Human Resources Training Development (HRTD). Available from <https://www.nrc.gov/docs/ML1122/ML11223A401.pdf>

Instrumentation inputs

The philosophy of the RPS is to define an area of permissible operation in terms of total power, flow, axial power distribution, and primary coolant temperature and pressure. The reactor is tripped when the limits of a selected area of concern are approached (NRC, 2009b). When the protection system receives signals indicative of an approach to an unsafe operating condition, the system actuates alarms, prevents control rod withdrawal if applicable, initiates a turbine runback if applicable (controlled decrease in turbine speed), and/or opens the reactor trip breakers. The instrumentation inputs to the RPS for a Westinghouse PWR are summarized in Table 1.

The overpower ΔT (OP ΔT) and overtemperature ΔT (OT ΔT) reactor trips provide core protection for situations in which:

- The transient is slow with respect to coolant piping delays from the core to the temperature sensors in the reactor coolant hot legs, and
- Pressure is within the range between the high- and low-pressure reactor trips.

The OP ΔT trip is designed specifically to ensure operation within the fuel temperature design basis. This is done through the OP ΔT by correlating core thermal power with the temperature difference across the reactor vessel hot leg and cold leg (ΔT). Because thermal power is not precisely proportional to ΔT (due to the effects of changes in coolant density and heat capacity), a compensating term, which is a function of average coolant temperature, is factored into the overpower trip setting. The OT ΔT trip is designed to ensure operation within the fuel rod heat transfer design basis and the hot leg boiling limit. Because both of these limits are functions of coolant temperature and pressure as well as core thermal power, the overtemperature trip is correlated with vessel ΔT , vessel average temperature, and primary system pressure.

The OP ΔT and OT ΔT protection function will trip the reactor when the compensated ΔT in any two channels exceeds the setpoint (three- and four-loop plants have one channel per loop; two-loop plants have two channels per loop). The temperature information for each channel (ΔT and T_{avg}) is obtained from independent pairs of resistance temperature measuring devices in the hot and cold legs of each loop. The setpoint for each channel is continuously calculated by analog circuitry programmed to evaluate the compensated OP ΔT and OT ΔT (Westinghouse, 1986).

Other reactor trips shown in Table 1, such as the low coolant flow and high neutron flux trips, provide core protection for accidents in which the loop ΔT signal does not respond quickly enough. Because the neutron flux measurement can span eight decades, three separate instruments are used: source range, intermediate range and power range. The source range is typically used to measure sub-criticality whereas the intermediate range is used during startup and shutdown; the power range is used during power operation (Young, 2021). Additional reactor trips, such as the high pressurizer water level and low feedwater flow trips, are provided primarily for equipment protection. Finally, some reactor trips, such as those produced by a turbine trip or reactor coolant pump circuit breaker opening, are provided in anticipation of potential plant transients and to minimize the resulting RCS thermal response of such transients.

The RPS design includes protection-grade interlocks (designated as P-n), also known as permissives, and control-grade interlocks (designated as C-n). An interlock blocks certain actions and permits others. Rod withdrawal stops are examples of control-grade interlocks; they are provided to prevent abnormal conditions which could result in a trip in response to excessive control rod withdrawal initiated by either a control system malfunction or a violation of administrative procedures by the operator.

Table 1 Representative Westinghouse instrumentation input to RPS (NRC, 2009b).

<i>Trip signal</i>	<i>Reason for trip</i>
Source Range High Neutron Flux	Prevents an inadvertent power increase
Intermediate Range High Neutron Flux	Prevents an inadvertent power increase
Power Range High Neutron Flux—low setpoint	Prevents an inadvertent power increase
Power Range High Neutron Flux—high setpoint	Limits maximum power to prevent damage to fuel rod cladding and fuel pellet centerline melting
High Positive Rate (of increase) Neutron Flux	Limits power excursions and prevents power distribution exceeding design basis
High Negative Rate (of decrease) Neutron Flux	Prevents power distribution exceeding design basis
Overtemperature ΔT (OT ΔT)	Prevents operation in excess of the fuel rod heat transfer design basis and the hot leg boiling limit
Overpower ΔT (OP ΔT)	Prevents operation in excess of the fuel temperature design basis
Pressurizer Low Pressure	Prevents operation in excess of the fuel rod heat transfer design basis and the hot leg boiling limit
Pressurizer High Pressure	Protects integrity of RCS pressure boundary
Pressurizer High Water Level	Prevents loss of pressure control due to solid water operations
Low Reactor Coolant Flow	Ensures adequate coolant flow to remove core heat and prevent fuel rod damage
Reactor Coolant Pump Bus Undervoltage or Underfrequency	Redundant to low reactor coolant flow trip
Reactor Coolant Pump Breaker	Redundant to low reactor coolant flow trip
Steam Generator Low-Low Level ^a	Prevents loss of heat sink
Low Feedwater Flow	Anticipates loss of heat sink
Turbine Trip	Prevents continued operation if steam load is lost to SGs
Safety Injection Actuation	Anticipates loss of reactor coolant

^aThere are two low-level setpoints for steam generator level. The first, called low level, is to alert the operators to an abnormal condition. The low-low-level is a potential loss of heat sink and generates a trip signal.

BWR

The most common BWR RPS design is an analog protection system with the addition of some solid-state protection system features, such as found in OPRMs and digital additions added to the APRMs logic.

The BWR RPS system is comprised of two redundant, identical trains (A and B) that are physically and electrically independent. Inputs into this system are derived from various nuclear and non-nuclear sensors located both inside and outside of the primary and secondary containment. Protection signals are derived directly from the status of contacts at sensors or components (examples are lubrication oil pressure switches on the turbine, auxiliary contacts on circuit breakers, and limit switches on valves).

Logic

The RPS is arranged in a one-out-of-two-twice de-energized-to-function logic scheme. Each trip system is completely independent of the other. The RPS logic consists of two separately powered trip systems (System A and System B). Each trip system contains two trip channel logics. The trip logics of system A are designated channels A1 and A2 and both receive at least one input signal from the same monitored parameter ("**Instrumentation Inputs**"); both trip channels are identical. Thus, any monitored parameter supplying an automatic trip signal has a minimum of four sensors. System B is identical to System A.

A signal from a sensor that exceeds the predetermined setpoint, either scram channel A1 or A2, causes a trip in system A. A trip condition in system A de-energizes the A scram valve solenoid for all 137 control rod HCUs. This condition is called a half scram. Likewise, any sensor in trip system B exceeding its setpoint (trip condition) causes a half scram. To produce a full reactor scram, both trip systems, A and B, must be in the trip (deenergized) condition.

In addition, two other methods are provided for manually causing a reactor scram. Two pushbuttons (one for System A and one for System B) are provided in the control room (one pushbutton per channel), each of which will de-energize its respective channel when pushed. De-energizing one channel in each trip system causes a manually actuated reactor scram. Placing the reactor mode switch into the "SHUTDOWN" position will also cause a manually actuated reactor scram by de-energizing all four RPS channels.

The six ARI solenoid valves are provided to redundantly depressurize the scram air header. These valves are divisionally separated with three assigned to the "A" division and three assigned to "B"; their logic is independent of RPS. Any condition resulting in a high reactor pressure (1120 psig/7.72 MPa-g) or low RPV level (−38 in./ −96.5 cm) will energize these solenoid valves to depressurize the header (the values given above are representative of smaller BWR/4's, values very slightly for other designs). The operator is also provided with capability for manual initiation of the logic.

Instrumentation inputs

Instrumentation input to the RPS for a representative BWR/4 is summarized in [Table 2](#). Similar to the PWRs, the goal of the majority of the instrumentation input to the RPS is to protect the integrity of the fuel rods in the reactor core and the integrity of the reactor vessel and connecting piping. However, several RPS inputs differ from those in a PWR owing to differences in BWR design features.

- Because of the small volume of the drywell, a reactor trip on high drywell pressure assures that primary containment integrity can be maintained in the event of a design basis accident.

Table 2 Representative BWR/4 instrumentation input to RPS (NRC, 2011a).

Scram signal	Reason for scram
High Drywell Pressure	Protect containment integrity (minimize energy containment must absorb)
Low Reactor Water Level	Protect fuel cladding from inadequate cooling (prevent fuel rod failure)
High Reactor Pressure	Protect reactor coolant boundary integrity (prevent loss of coolant)
Main Steam Line High Radiation	Limit further fission product release (prevent fuel rod failure)
Turbine Stop Valve Closure	Protect fuel cladding against positive reactivity insertion following void collapse in the core (prevent fuel rod failure)
SDV High Level	Allows RPS to scram while sufficient free volume exists in SDV tank to complete the scram
MSIV Closure	Protect fuel cladding against positive reactivity insertion following void collapse (prevent fuel rod failure)
Average Power Range Monitor (APRM) High-High Fixed Trip ^a	Protect fuel cladding against excessive power (prevent fuel rod failure)
APRM "Inoperable" Trip	Prevent continued operation with failed instrumentation
Intermediate Range Monitor (IRM) High-High Trip ^a	Protect fuel cladding against excessive power and short time intervals between power oscillations (Young, 2021) (prevent fuel rod failure)
IRM Inoperable Trip	Prevent continued operation with failed instrumentation
Source Range Monitor (SRM) High-High ^a	Protect refueling personnel from inadvertent criticality
SRM Inoperable	Protect refueling personnel from inadvertent criticality

^aThere are two high level setpoints for neutron flux monitors. The first, called "high," is to alert the operators to an abnormal condition. The "high-high" is a potential power excursion and generates a trip signal.

- Because the main steam lines provide a direct path between the reactor vessel and the turbine/condenser, a reactor trip on high radiation limits fission product releases to the atmosphere.
- Because the loss of voids in the coolant in the reactor core is a positive reactivity insertion event, reactor trips for loss of the turbine (stop valve or MSIV closure).
- Because the primary reactor trip strategy involves discharging water into the SDV, a reactor trip for high level in the SDV tank ensures that sufficient free volume still exists for this reactor trip strategy.

The setpoints for the reactor trips for these instruments depends on the mode of operation of the plant. The four principal modes of the RPS are: "SHUTDOWN," "REFUEL," "STARTUP/HOT STANDBY," and "RUN." The selection of the various modes is accomplished via a key lock selector switch appropriately called the reactor mode switch. The reactor mode switch interlocks the various control functions depending on the plant operating condition. The reactor mode switch is manually placed in one of the four positions which places into effect the corresponding trip functions for that mode.

Placing the reactor mode switch in "SHUTDOWN" initiates a reactor scram, disables the MSIV closure scram, and enables the capability to bypass of the SDV high level scram.

Placing the reactor mode switch in "REFUEL" disables the MSIV closure scram. The refuel position also enables the bypassing of the SDV high level scram and allows single control rod withdrawal to support refueling activities.

Placing the reactor mode switch in "STARTUP/HOT STANDBY" allows the operator to withdraw control rods for a plant startup and disables the ability to bypass the SDV high level scram. The MSIV closure scram is still bypassed. Some neutron monitoring trip functions are enabled while others are bypassed.

Placing the reactor mode switch in "RUN" selects the APRM thermal power scram and fixed scram trip level while removing the 15% power APRM scram. The IRM system scram signals are disabled except for the IRM Hi-Hi and companion APRM downscale scram. The MSIV closure scram is enabled.

Summary

Severe plant transients require that the reactor be shut down by the RPS. The RPS is designed to provide a highly reliable means of achieving reactor shutdown in the event of one of these severe transients by monitoring certain parameters; a means for the operators to manually scram the reactor from the control room or from the remote shutdown panel is also provided. Plant technical specifications are developed for the reactor scram system instrumentation to assure their operability and accuracy by means of various surveillance requirements.

The RPS designs for PWRs and BWRs are comprised of two redundant, identical trains that are physically and electrically independent. Inputs into this system are derived from various sensors located both inside and outside of the primary containment building. The RPS contains a logic section that requires instrumentation input from more than one instrument channel to generate a signal to automatically shutdown the reactor. This prevents a failure in a single instrument channel from shutting down the reactor. Automatic shutdown of the reactor is accomplished by removing electric power from a device that then passively inserts the control rods into the core. The PWR and BWR RPS designs meet all requirements of the NRC.

With respect to the RPS design, special consideration is given to their ability to prevent and mitigate ATWS events. In response to ATWS events and several failures of reactor scram systems at operating plants, the NRC imposed new requirements to further assure that reactors could be shutdown safely.

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Engineered Safety Features

Robert Lutz^a and Bill Williamson^b, ^a Lutz Nuclear Consulting, Hendersonville, NC, United States; and ^b TVA, Athens, AL, United States

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Glossary

BWR Boiling Water Reactor.

Design basis accident The range of events taken explicitly into account in the design of structures, systems and components and equipment of a facility such that the facility can withstand them without exceeding authorized limits.

Emergency operating procedure A set of instructions that are to be implemented following an unintentional reactor trip or safety injection initiation.

Emergency power bus The a.c. busses that are powered from offsite power but are also automatically energized by the emergency diesel generators when offsite a.c. power is not available.

Engineered safety feature (ESF) Those systems, structures and components (SSCs) that have a function to mitigate the consequences of an accident.

General design criteria The set of minimum requirements for the design and safety analysis for water-cooled nuclear power plants.

Guidance Any set of instructions that are recommended to be implemented under prescribed plant conditions.

Procedures A set of instructions that are *required* to be implemented verbatim under prescribed plant conditions.

Maximum hypothetical accident A major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Owners group A group of plant owners with similar plant designs (usually by reactor vendor or reactor type) that fund development programs common to their plant designs.

PWR Pressurized water reactor.

Reactivity A measure expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Reactor trip/reactor scram An automatic actuation of systems to terminate the fission process in the reactor. The term, "trip," is typically applied to PWRs and the term, "scram," to BWRs.

Single failure criterion Requirement that a system which is designed to perform a defined safety function must be capable of meeting its objectives assuming the failure of any major component within the system or in an associated system which supports its operation.

System, structure and components (SSC) A term typically reserved for those design features of the plant that are important to safety.

Introduction

Engineered safety features (ESFs) and associated safety systems and procedures are the third level of defense-in-depth to prevent damage to the reactor core and releases of radioactivity requiring offsite protective actions and to return the plant to a safe state (IAEA, 2018; Rempe, 2021). In other words, any system, structure or component (SSC) that is specifically designed to function to mitigate the consequences of an accident can be classified as an ESF of the plant. Generally speaking, the term ESF is only applied to those systems and components used to mitigate design basis accidents. The ESFs are part of the overall Reactor Protection System (RPS) that is discussed in Lutz and Williamson (2021c). This section focuses on the SSCs; the procedures and guidance are discussed in Lutz and Williamson (2021a).

The very first commercial nuclear power reactors did not have ESFs as we know them today. At that time, there was a safety focus on the maximum hypothetical accident, which was then considered to be a catastrophic break of a large pipe in the system that provides cooling water for the reactor. This accident was considered to be of such low probability that dedicated systems with redundant components were not included in the design. In some cases, systems used to support normal operation of the plant were designed to serve double-duty in the unlikely event of an accident.

The development of General Design Criteria (GDC) for power plant construction brought about the first significant change in requirements for ESFs as we know them today (NRC, 1967). As discussed in Rempe (2021), the GDC were finalized in 1971 (NRC, 1971). Published as Appendix A of the NRC regulations in 10 CFR 50 (U.S. Government Printing Office, 2019), the GDC call for redundancy and diversity in engineered safety systems designed to protect the core in the event of a circumferential break in a main coolant pipe (i.e., the above-mentioned maximum hypothetical accident). The GDC also include a requirement for ESFs to prevent the containment from failing in the event of any loss of coolant accident (LOCA). The “single failure” criterion was introduced in the GDC as one of a set of tools for ensuring highly reliable SSCs are provided for safe shutdown and cooling as well as mitigation of design basis accidents (NRC, 1977).

The safety significance of the ESFs of that time, and the single failure criterion were demonstrated in the Reactor Safety Study, also known as WASH-1400 (NRC, 1975). The Reactor Safety Study lists five functions of ESFs for LOCAs:

1. Rapid shutdown of the reactor to limit core heat production,
2. Core cooling to prevent release of radioactivity from fuel,
3. Containment isolation to prevent dispersal of radioactivity to environment,
4. Removal of heat from containment to prevent overpressurization, and
5. Removal of radioactivity from containment atmosphere.

These ESFs were designed to reduce the amount of radioactivity released to the environment should either a LOCA or transient event result in a significant release of radioactivity from the reactor core.

The Reactor Safety Study notes that in early power reactors, the power level was about one-tenth that of the new larger reactors. Because the decay heat was low enough to be readily transferred through the steel containment walls to the outside atmosphere, it could not over-pressurize and fail the containment. Thus, if a LOCA were to occur, and even if the core were to melt, the low leakage containments that were provided would have permitted the release of only a small amount of radioactivity (see Addendum for a description of ESFs in earlier plants from the 1960s). The decay heat levels for the newer plants in the 1970s were so high that the heat could not be dissipated through the containment walls. Thus, new sets of requirements were imposed.

The Reactor Safety Study further confirms that dedicated Emergency Core Cooling Systems (ECCSs) were needed to prevent core melting which could, in turn, cause failure of all barriers to the release of radioactivity. Systems were needed to transfer the core decay heat from the containment to the outside environment in order to prevent the heat from producing internal pressures high enough to rupture the containment. Finally, systems were needed to remove radioactivity from the containment atmosphere in order to reduce the amount that could leak from the containment into the environment. The major goal behind these changes was to attempt to provide ESFs designed so that the failure of any single barrier would not be likely to cause the failure of any of the other barriers.

The Reactor Safety Study concluded that the probability of a large pipe break causing a core melt was much less likely than a small LOCA or a transient (without a pipe break), particularly the loss of offsite power followed by the failure of the auxiliary feedwater (AFW) system for decay heat removal. However, the Reactor Safety Study showed that the ESFs that were required by the GDC were, in part, responsible for the low probability of the maximum hypothetical accident leading to core melt. This result, in conjunction with the recently completed Emergency Core Cooling System Rulemaking (NRC, 1973; NRC, 1974; Rempe, 2021), shifted attention from the large break loss of coolant accident to the more probable small LOCA and transients in safety analysis.

The next major event that changed the course of ESFs was the core melt accident at Three Mile Island Unit 2 (TMI-2). This accident was initiated by a transient (loss of feedwater) that led to a small break LOCA but operator actions to prematurely terminate safety injection resulted in core damage (Rogavin and Framptom, 1980; Kemeny et al., 1979; Skillman and Rempe, 2021). As a result, the U.S. NRC issued new requirements for licensing nuclear power plants (NRC, 1980a,b). Of the numerous new requirements, those dealing with the AFW system and relief valves would fall under the broad classification of ESFs. Other new requirements involved instrumentation and emergency procedures, as discussed in Lutz and Williamson (2021a,b).

For PWRs, the new requirements for AFW included: automatic initiation of feedwater injection on either low SG level or reactor trip; loading motor-driven AFW pumps onto emergency power busses; and that no single failure would result in a loss of AFW function. There were also new requirements for indication of open safety or relief valves.

For BWRs, the new requirements included diverse and redundant indications of the position of safety overpressure relief valves and logic changes to Reactor Core Isolation Cooling (RCIC) to automatically restart after trip on high Reactor Pressure Vessel (RPV) water level.

Based on the results of the Reactor Safety Study, as well as operating experience at nuclear power plants, the NRC imposed an additional set of requirements for the AFW system in 1988 that required the capability to cope with a loss of all a.c. power to the plant (NRC, 1988). The duration of the loss of a.c. power for each plant was based on a number of factors including previous events resulting in a loss of offsite a.c. power.

Engineered safety features (ESFs)

The NRC requires certain SSCs to be identified as ESFs, including the primary containment structure, containment systems, residual heat removal (RHR) systems, ECCSs, containment heat removal systems, containment atmosphere cleanup systems, and certain cooling water systems (NRC, 2007, 2019). Other systems that have design and testing requirements similar to these ESFs are covered by other regulatory requirements. These include the AFW system, emergency power supply and cooling water supply. They are included in this discussion based on the similarity of requirements to the ESFs.

The SSCs included in this discussion are listed in Table 1. Acronyms in the table are defined within this chapter.

Core subcriticality

Establishing and maintaining core subcriticality is not always thought of in terms of ESFs, but there is redundancy and diversity in the systems used to establish and maintain core subcriticality following an accident as required by the U.S. NRC's GDC (NRC, 1971).

In LWRs, the first line of defense in establishing core subcriticality is the negative reactivity supplied by prompt reactor control rod insertion and borated water injection.

Pressurized water reactors

A combination of boric acid in the reactor coolant system (RCS) and control rod position is used to control reactor power during full and partial power operation. For planned shutdowns, a combination of insertion of all control rods and increased boric acid

Table 1 Systems, structures and components considered as engineered safety features.

Function	Method	PWR SSCs	BWR SSCs
Core Subcriticality	Negative Reactivity	PWRs (Control Rods, Boron)	BWRs (Control Rods, Standby Liquid Control)
Core Heat Removal	Water Injection	PWRs (ECCS-HPSI, LPSI, Accumulators)	BWRs (ECCS-HPCI, LPCI, Core Spray, LPCF)
	Heat Removal	PWRs (Heat Exchangers - for cooling water recirculation, and AFW)	BWRs (Suppression Pool, SRVs, ADS)
Emergency Power	Pressure Control	PWRs (PORVs and ARVs)	BWRs (SRVs, ADS logic)
Prevent Radioactivity Release from Containment	Pressure	PWRs (DGs, Batteries)	BWRs (DGs, Batteries)
	Suppression	PWRs (Primary Containment, Volume, Sprays, Ice Condensers)	BWRs (Drywell Volume, Drywell Sprays, Suppression Pool sprays and cooling)
	Heat Removal	PWRs (Sprays, Fan Coolers)	BWRs (Drywell Sprays -some BWRs do not have Drywell sprays, fans, coolers)
	Isolation	PWRs (Isolation Valves)	BWRs (Isolation Valves and MSIVs)
	Fission Product Cleanup	PWRs (Sprays, Filters)	BWRs (Drywell Sprays, Suppression Pool Sprays)
	Hydrogen Control	PWRs (Igniters, Recombiners)	BWRs (Inerting, Igniters, Vent)
Heat Removal	Ultimate Heat Sink Water	PWRs (Component Cooling, Service Water)	BWRs (Closed Cooling Water System, RHR Service Water, Emergency Equipment Cooling Water)

concentration is used to establish and maintain subcriticality. In the event of a LOCA, a combination of the insertion of control rod clusters (the failure of one control rod cluster to insert is assumed in safety analyses) and borated water addition via the ECCS pumps is used to establish and maintain the reactor subcritical. Control rod insertion depends only on disconnecting the electrical power to the control rod drive grippers; boric acid injection is via borated water from one of two independent trains of ECCS (see “[Water injection and core heat removal](#)” section). For other accidents insertion of control rods combined with rapid boration establishes and maintains core subcriticality.

The anticipated transient without scram (ATWS) event is a special accident that is now considered after the issuance of the ATWS Rule (NRC, 2020). Changes to the plant ESFs to reduce the risk from ATWS events are discussed in [Lutz and Williamson \(2021c\)](#).

Boiling water reactors

A combination of adjusting the control rod position and core coolant flow rate is used during reactor power operation to control core power. For planned shutdowns, a combination of fully inserting control rods and reducing core flow is used to establish shutdown conditions. For a BWR with all control rods fully inserted, the reactor is shutdown and subcritical under all conditions.

The anticipated transient without scram (ATWS) event is a special accident that is now considered after the issuance of the ATWS Rule (NRC, 2020). Changes to the plant ESFs to reduce the risk from ATWS events are discussed in [Lutz and Williamson \(2021c\)](#).

Water injection and core heat removal

The water injection and core heat removal functions are served by the same ESFs. Water injection assures that the core is covered with water so that core decay heat can be effectively removed from the reactor fuel rods. Core heat removal assures that the decay heat removed from the fuel rods by the water is transferred to an ultimate heat sink outside of the primary containment structure.

For both PWRs and BWRs, the water injection and decay heat removal requirements vary over the range of LOCA break sizes, as well as for non-LOCA events. Large LOCAs are typically defined as breaks in reactor coolant pipes (or pipes connected to the reactor coolant pipes that cannot be isolated) greater than about 6 in. (15 cm) equivalent diameter, including the design basis double ended guillotine break of a large diameter reactor coolant pipe. Medium LOCAs are typically defined as coolant piping breaks between about 6 in. (15 cm) and 2 in. (5 cm) equivalent diameter. Small LOCAs are coolant piping breaks less than about 2 in. (5 cm) equivalent diameter. There is also a class of Very Small LOCAs which are coolant piping breaks less than about 0.5 in. (1.25 cm) equivalent diameter; these behave similar to non-LOCA events for the purposes of water injection and decay heat removal.

Breaks in the reactor vessel are not considered in the design basis because of the rigorous surveillance to detect and remedy cracks in these thick wall vessels. However, core cooling can be maintained for breaks in the reactor vessel up to the capability of the safety injection system to fill the reactor vessel above the top of the fuel.

Pressurized water reactors

Removal of decay heat from the reactor core involves different combinations of systems and components, depending on the event and the stage of the event. In general, there is an early “injection” phase that relies on the heat of evaporation of injected water to remove decay heat from the core to the containment and a later “recirculation” phase that relies on heat exchangers to cool water as it is recirculated from containment back into the core. The descriptions below pertain to Westinghouse PWR designs (NRC, 2006); differences for the other PWR designs are addressed in “[Other PWR designs](#)” section.

Safety injection system designs for LOCA events have both diversity and redundancy. There are two independent trains of low-pressure, intermediate-pressure and high-pressure safety injection to provide cooling water to the reactor core. Each train of safety injection is powered from a separate emergency power bus that is fed from redundant emergency diesel generators. Accumulators are passive components that inject their water when the RCS pressure drops below a predetermined setpoint. The safety injection system valves in these systems that are required to change position during the injection phase are check valves. The only motor operated valves are those required to switch from injection phase to recirculation phase; these valves are powered from the same emergency busses as the pumps in their respective trains.

Injection phase

Large LOCAs - For large LOCAs, the initial decay heat removal is accomplished with a combination of accumulators, high pressure safety injection (HPSI) pumps and low-pressure safety injection (LPSI) pumps (some plants also have intermediate pressure injection pumps). Water is injected into the RCS cold legs where it removes decay heat as it flows through the core and is discharged to the containment. The flow rates are not only sufficient to remove all decay heat from the reactor core but also to restore the water height in the reactor vessel to a level above the top of the core to maintain core cooling.

For the largest LOCAs, accumulators are needed to provide initial injection because of the delay in loading the injection pumps onto emergency busses if offsite power is lost. These accumulator tanks begin to inject when a single check valve opens as the RCS pressure decreases below a predetermined value (typically between 200 and 600 psig, or 1.4 to 4.1 MPa-g) and the pressure of the cover gas (nitrogen) forces the accumulator water into the cold legs. This water is followed by water injection from the HPSI and LPSI pumps.

Medium LOCAs - For medium size LOCAs, the RCS pressure may not drop below the shutoff head of the LPSI pumps and core cooling is accomplished using the intermediate pressure and HPSI pumps. This flow is sufficient to maintain core cooling while the break flow removes all decay heat from the reactor core and restores water level above the top of the core.

Small LOCAs - For small LOCAs, the RCS pressure remains high enough that only the HPSI pumps will discharge into the cold legs. In this case, the break flow is not sufficient to remove all of the core decay heat and refill the reactor coolant system; auxiliary feedwater to the steam generators is necessary to supplement the break flow in removing core decay heat. Without auxiliary feedwater flow for heat removal, the RCS pressure would steadily increase, and injection flow would not be able to maintain core cooling (injection flow from centrifugal safety injection pumps decreases as system pressure increases).

Transient Events - For the very small LOCAs and for transient (non-LOCA) events, heat is removed from the reactor core by the AFW system. Upon reactor trip, the main feedwater pump is tripped, and the AFW system is automatically started. The AFW system has the same redundancy (separate trains) and diversity (motor driven and a.c. independent pumps). The AFW system generally consists of two motor driven pumps and one turbine driven pump that take suction from a common tank (Condensate Storage Tank) and inject into the steam generators. The turbine driven pump is powered by steam from at least two steam generators (one steam generator supply source can be isolated in the event of a steam generator tube rupture to prevent releases of radioactivity).

For the response to an ATWS event, emergency operating procedures instruct the plant operators to fully open the AFW throttle valves (the throttle valves are manually set at a predetermined flow rate to prevent overcooling of the RCS for steam line break events).

Recirculation phase

When the water stored in a tank for safety injection is depleted, the suction for the safety injection pumps is switched to the containment sump, which collects water released from the break during a LOCA. The LPSI pumps take suction from this sump and either inject the water directly into the reactor coolant cold legs if the reactor pressure is low enough or they act as a booster pump for the HPSI pumps. The water drawn from the containment sump is passed through a heat exchanger before being injected back into the RCS. This is the primary means of long-term heat removal for LOCAs; for some plants, the heat exchanger is on the containment spray recirculation system but acts similarly in removing core decay heat.

In the recirculation phase for large break LOCAs, if the LOCA is in one of the cold legs it is possible for a portion of the injected water to flow around the core barrel and be discharged through the break directly to the containment without traveling through the core. The portion of the safety injection flow that does reach the core would remove decay heat by boiling with the resulting steam flowing out the break location in the cold leg. In this case, the boric acid in the injected flow that reaches the core would be continually concentrated in the reactor vessel as a result of the boiling process (i.e., the product of boiling is primarily steam with other additives left in the reactor vessel). To prevent the boric acid from reaching the saturation concentration where it could precipitate and potentially cause blockages in the core that would interrupt long term heat removal, a portion of the recirculation phase injection is routed to the hot legs where it would flow downwards through the core and dilute any concentrated boric acid. The timing of the switchover to hot leg recirculation is based on conservative analyses because there is no means of measuring boric acid concentration in the reactor vessel following an accident.

Other PWR designs

There are a few unique design details of ESFs amongst the reactor designs offered by the three original U.S. PWR vendors. The descriptions above pertain to the designs offered by Westinghouse Electric Corporation.

The Babcock and Wilcox safety injection system design (NRC, 2008a) includes HPSI and LPSI pumps as well as accumulators as discussed above for the injection phase. However, in the recirculation phase, only the HPSI pumps are used. This is different from the Westinghouse and Combustion Engineering designs in which the required Net Positive Suction Head (NPSH) for the HPSI pumps cannot be met with the containment sump water. For the Babcock and Wilcox design, the containment sump water can satisfy the required NPSH for the HPSI pumps.

The Combustion Engineering design (NRC, 2008b) does not include a heat exchanger in the safety injection recirculation phase. The heat exchanger is in the containment spray recirculation line. Therefore, the Combustion Engineering design requires continual operation of the containment spray in the recirculation phase to provide core decay heat removal. However, the heat exchanger can be used by the LPSI pumps if the alignment is changed (during shutdown cooling, this alignment is normally used). Westinghouse and Babcock and Wilcox designs can terminate containment spray in the recirculation phase if predetermined conditions for containment pressure and temperature are satisfied.

Boiling water reactors

Removal of decay heat from the reactor core involves different combinations of systems and components, depending on the event and the stage of the event. In general, there is an early "injection" phase that relies on the heat of evaporation of injected water to remove decay heat from the core to the containment and a later "recirculation" phase that relies on heat exchangers to cool water as it is recirculated from containment back into the core. The descriptions below pertain to General Electric BWRs (NRC, 2011).

Safety injection during LOCA events is accomplished with diverse and redundant systems. There are two independent trains of low-pressure, and high-pressure safety injection to provide cooling water to the reactor core. Each train of safety injection is powered from a separate emergency power bus that is fed from redundant emergency diesel generators. The valves in these systems that are

required to change position during the injection phase are check valves. The only motor operated valves are those required to switch from injection phase to recirculation phase; these valves are powered from the same emergency busses as the pumps in their respective trains.

For LOCAs, the BWR/2 through BWR/4 designs use a High Pressure Coolant Injection (HPCI) and Low Pressure Coolant Injection (LPCI), while BWR/5 and BWR/6 designs use High Pressure Core Spray (HPCS) and Low Pressure Core Spray (LPCS) which distributes the injected water over the top of the core. For transient events, the BWR/2 and some BWR/3 designs utilize the Isolation Condenser (IC) for response to transients. The remainder of the BWR/3 and all of the BWR/4 through BWR/6 designs utilize a RCIC system which employ turbine driven feedwater pumps. Both the IC and RCIC systems are single train standby systems for safe shut down of the plant. These systems are not considered part of the safeguards core cooling system and do not have a LOCA function.

Injection phase

Large LOCAs - For large loss of coolant accidents, the initial decay heat removal is accomplished LPCI/LPCS pumps. Water is injected into the RPV and removes decay heat as it flows through the core and is discharged to the primary containment. The flow rates are high enough to remove all decay heat from the reactor core and to restore the RPV water level to the top of the jet pumps to maintain core cooling. Some earlier BWR's (BWR/2's) do not have jet pumps and rely on Core Spray systems to remove all decay heat. For the largest LOCAs, pressure control is not needed as the RPV depressurizes due to the Large Break LOCA.

Intermediate Break LOCAs - For medium size LOCAs, the RPV pressure may not drop below the shutoff head of the LPCI / LPCS pumps and core cooling is accomplished using the HPCI /HPCS pumps in conjunction with the LPCI/LPCS pumps or using the Automatic Depressurization System (ADS) in conjunction with the LPCI or LPCS pumps. This flow is sufficient to remove all decay heat from the reactor core. The ADS functions as a backup for the HPCI/HPCS system.

Small LOCAs - For small LOCAs, the RPV pressure remains high enough that only the HPCI/HPCS pumps will inject water. In this case, the HPCI/HPCS flow is sufficient to remove all of the core decay heat. HPCI/HPCS also serves to depressurize the RPV so that the LPCI/LPCS pumps can provide injection. For the case where HPCI is not available, the ADS acts as a back up to rapidly lower RPV pressure to permit injection by the LPCI or the LPCS pumps.

Transient Events - For the Very Small LOCAs and transient (non-LOCA) events, heat is removed from the reactor core by the IC or the RCIC system and on the HPCI/HPCS. Upon the reactor scram, the IC or RCIC starts. HPCI or HPCS is the backup for IC and RCIC under these conditions. ADS is the backup for HPCI or HPCS.

Recirculation phase

Once the initial injection phase is completed, suppression pool cooling using the RHR system is started to remove the heat from the primary containment while injection at a lower rate continues. The suppression pool cooling flow passes through RHR heat exchangers where the energy is transferred to the ultimate heat sink. Some BWRs have a containment or drywell spray, which also uses the RHR heat exchangers to transfer heat to the ultimate heat sink. While Mark I and Mark II BWRs have this spray capability, only some of the Mark III containments have containment spray; those Mark III BWRs without spray capability rely on large fan cooler units in the primary containment to transfer heat to the ultimate heat sink. For station blackout events in which all a.c. power is lost, a hardened vent can be used to remove heat from the suppression pool as long as water levels can be maintained by fresh water injection by turbine driven pumps (e.g., RCIC or HPCI in Mark I and II containments); a new requirement following the Fukushima accident resulted in a new dedicated vent by the suppression pool and the wetwell. For station blackout events in which all A.C. power is lost, Mark III containments use a feed and bleed method to remove heat from the suppression pool.

Pressure control

Effective core heat removal requires that the pressure in the reactor vessel and associated coolant systems be controlled for several reasons.

If the reactor vessel pressure exceeds the design limit of the reactor vessel and attached piping, failures of the vessel or piping are likely to occur which could make an event worse. While a double ended rupture of piping attached to the reactor vessel is a design basis accident, the rupture of the reactor vessel itself is not a design basis event. The catastrophic rupture of the reactor vessel is considered a very low frequency event due to the stringent design and inspection requirements of the ASME code.

If the reactor vessel pressure exceeds the shutoff head of the safety injection pumps, those pumps will not be able to inject water in the case of an emergency event.

For both PWRs and BWRs, the ability to remove heat from the reactor using safety injection pumps and to efficiently transition to a cold shutdown state requires intentionally dumping steam to the atmosphere by opening relief valves on the steam lines.

Pressurized water reactors

Overpressure of either the reactor vessel and associated piping or the steam generator is mitigated by spring-loaded safety relief valves. These are designed and maintained in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME, 2019). There are typically two safety relief valves capable of relieving pressure from the reactor vessel for any event involving a successful reactor trip. The pressure relief valves are situated in the gas space at the top of the pressurizer and are set to open just below the reactor vessel design basis pressure.

Pressure control is normally performed by the pressurizer spray in PWRs. However, this system is not safety related and does not have sufficient capacity to respond to accidents that require pressure relief of the RCS. Pressure relief for accident mitigation is provided by power operated relief valves (PORVs), except as noted below for some Combustion Engineering PWRs. These relief valves are separate from the spring-loaded safety valves for overpressure control except for some European PWRs that use a dual-purpose valve manufactured in France.

Plants licensed prior to about 1979 do not have safety related PORVs; for these plants, the PORV valve operators and electrical control systems are normally designed to non-safety related standards while the remainder of the valve is designed as a RCS pressure boundary. Westinghouse PWRs have two (and sometimes three) large capacity PORVs; CE plants have either 2 PORVs or no PORVs; B&W plants have one small PORV. For CE plants without PORVs, the safety related RCS pressure control function is performed by the Auxiliary Pressurizer Spray (APS) system (NRC, 1989).

For PWRs, pressure control in response to an accident is implemented by the licensed operators from the main control room. Specific procedures prescribe the use of these for reducing reactor vessel pressure following reactor trip from power operation.

Boiling water reactors

Overpressure of either the RPV or its associated piping is provided by spring-loaded safety relief valves. These are designed and maintained in accordance with the ASME Boiler and Pressure Vessel Code. The number of SRVs are plant-specific and capable of relieving pressure from the reactor vessel for any event involving a successful reactor trip. In some BWR plants, there are additional safety valves set to automatically open sequentially at different pressures if the SRVs do not limit RPV pressure satisfactorily. A sufficient number of SRVs exist that even if one is inoperable, BWRs are able to demonstrate margin to the design limit.

The SRVs are spring loaded for automatic overpressure control and can also be operated by the licensed operators from the main control room to manually reduce reactor vessel pressure. Specific procedures prescribe the use of these for reducing reactor vessel pressure following reactor trip from power operation.

Containment functions

There are two principal containment functions: containment integrity and containment fission product control. Containment integrity refers to leak-tightness during an event by controlling containment pressure and temperature, isolation of containment of piping carrying liquid or gas through the containment pressure boundary (i.e., containment penetrations), and containment hydrogen control to prevent a catastrophic containment failure. Containment fission product control refers to reduction in containment airborne fission products to minimize the potential for atmospheric releases.

Pressurized water reactors

Containment leak-tightness

All PWR primary containment structures (except the early PWRs mentioned in the Addendum) are low leakage structures made of steel and concrete. Fig. 1 compares features of typical PWR containments. A few PWR containments are a free-standing cylindrical steel shell, particularly the ice condenser containments. The steel shell is encompassed in a concrete structure with an annular volume between them. The more traditional PWR containment is a re-enforced concrete structure with a steel liner directly inside the concrete structure. The concrete structure provides protection for the reactor vessel and other equipment inside the containment from missiles (externally generated as well as internal missiles) and provides radiation shielding in the event of an accident that releases fission products to the containment. In both cases, the steel shell provides the low-leakage characteristics of the containment. The large dry containment design for a Westinghouse 4-loop PWR typically has a 2.5 million cubic foot or 70,800 cubic meter internal volume with leakage rates on the order of 2 cubic feet per minute (0.06 cubic meter per minute) at the containment design pressure of 50 psig (or 0.34 MPa-g). The internal volume is largely determined by the licensed reactor power level because design pressure must not be exceeded during the initial phase of a design basis large LOCA.

The subatmospheric containment is maintained at a strong vacuum during normal operation (e.g., 10 psia or 0.07 MPa). As a result, the containment internal volume can be about 10% smaller than the large dry containment design while leakage characteristics and design pressures are similar. Following an accident, the containment spray system, using chilled water at about 50 degrees F/10 degrees C is capable of quickly (e.g., within an hour) returning the containment to a vacuum state thereby stopping any leakage from containment.

The ice condenser containments rely on the ice in the ice chest to limit the peak containment pressure during the initial phase of a design basis large LOCA. This allows a design pressure of about 12 psig/0.18 MPa-g and an internal volume of approximately 1.8 million cubic feet/50,970 cubic meters.

Containment pressure and temperature control

Containment pressure and temperature control is accomplished by either removing heat from the containment atmosphere to the ultimate heat sink or by absorption of heat from the containment atmosphere by water sprayed into the containment or melting ice in an ice condenser containment.

The "large dry" containments typically have two diverse methods of removing heat to the containment: the containment spray system and the fan cooler system. In each case, there are two safety related trains of heat removal equipment, each powered from a different emergency a.c. power bus and a different cooling water train.

PWR Containment Designs Differ

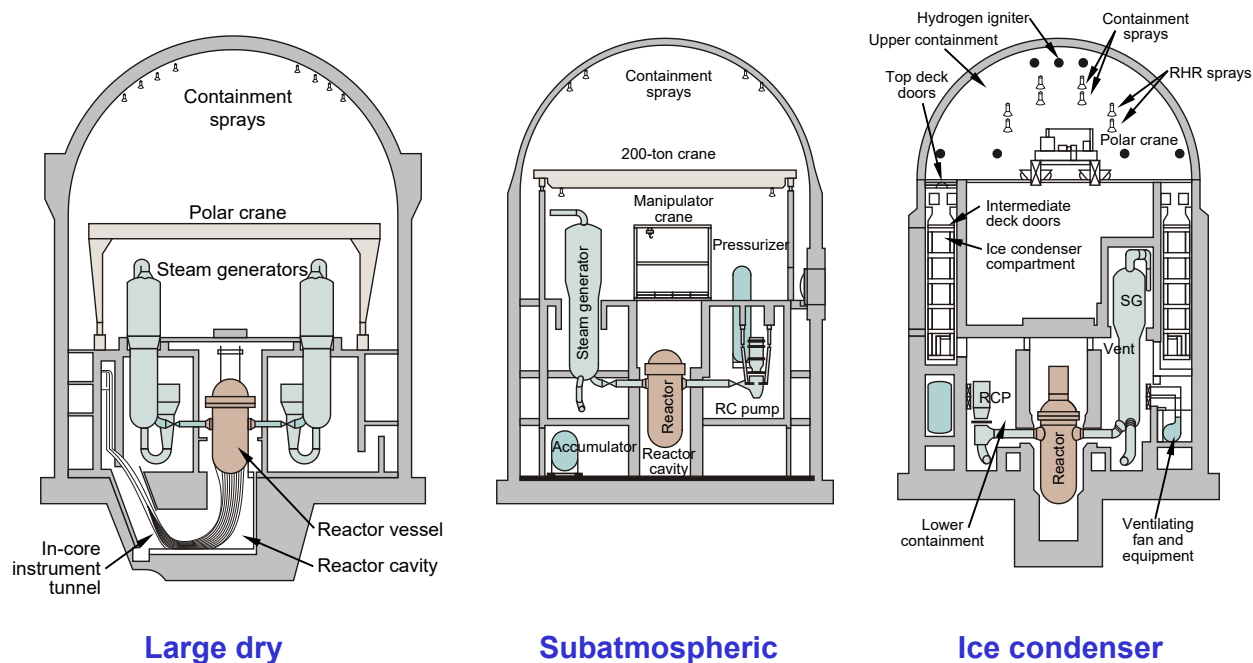


Fig. 1 Comparison of typical PWR containments. Galyean W and Rempe J (2005) *Accident Progression Analysis (P-300)*, U.S. NRC, <https://www.nrc.gov/docs/ML1204/ML12044A213.pdf>, May 2005.

- The containment spray system has both an injection phase and a recirculation phase. In the injection phase, water is sprayed into the containment from a storage tank. The ambient temperature water absorbs heat from the containment atmosphere and falls into the containment sump. In Westinghouse and B&W designs, heat is removed from the sump water by the ECCS recirculation heat exchanger; in Combustion Engineering designs, the heat exchanger is part of the containment spray recirculation alignment rather than the safety injection recirculation alignment. In all three designs, the containment spray water is maintained at a reduced temperature compared to the containment atmosphere and effectively continues to remove heat from the containment atmosphere.
- The containment fan coolers are large ventilation units with a. c. powered fans drawing air past cooling coils that are supplied with water from the service water system. There are normally 4 to 5 of these units in the containment with 50% of the units capable of removing all of the decay heat from containment following a design basis accident.

The subatmospheric containments, which are held at about 5 psig/0.135 MPa-g vacuum during normal operation, have only containment spray available for controlling containment pressure and temperature. The original design of the subatmospheric containments used chilled water (e.g. 50 degrees F/10 degrees C) for containment spray injection. When combined with the initial pre-accident vacuum state of the containment, the chilled spray injection could return the containment to a vacuum condition within 1 hour. The containment spray system in recirculation mode would maintain the containment in a vacuum condition.

The ice condenser containment has an ice bed with a large amount of ice to condense the steam for a loss of coolant accident and maintain the containment pressure below the design basis pressure, which is only on the order of 12–15 psig/0.07–0.08 MPa. This is effective until the ice is depleted. The ice condenser containments also include containment spray and containment recirculation spray to maintain containment pressure below its design limit. The ice condenser has multiple doors that open when a pressure differential is created by the loss of coolant accident steam discharge, so it satisfies the redundancy requirements for ESFs.

Boiling water reactors

Containment leak-tightness

All BWR primary containment structures are low leakage structures made of concrete with an embedded steel liner for leak-tightness. The primary containment is composed of the drywell and the volume containing the water suppression pool. That volume is called the torus in the Mark I design and the wetwell in the Mark II and Mark III designs. Fig. 2 compares features of typical BWR containments.

BWR Containment Designs Differ

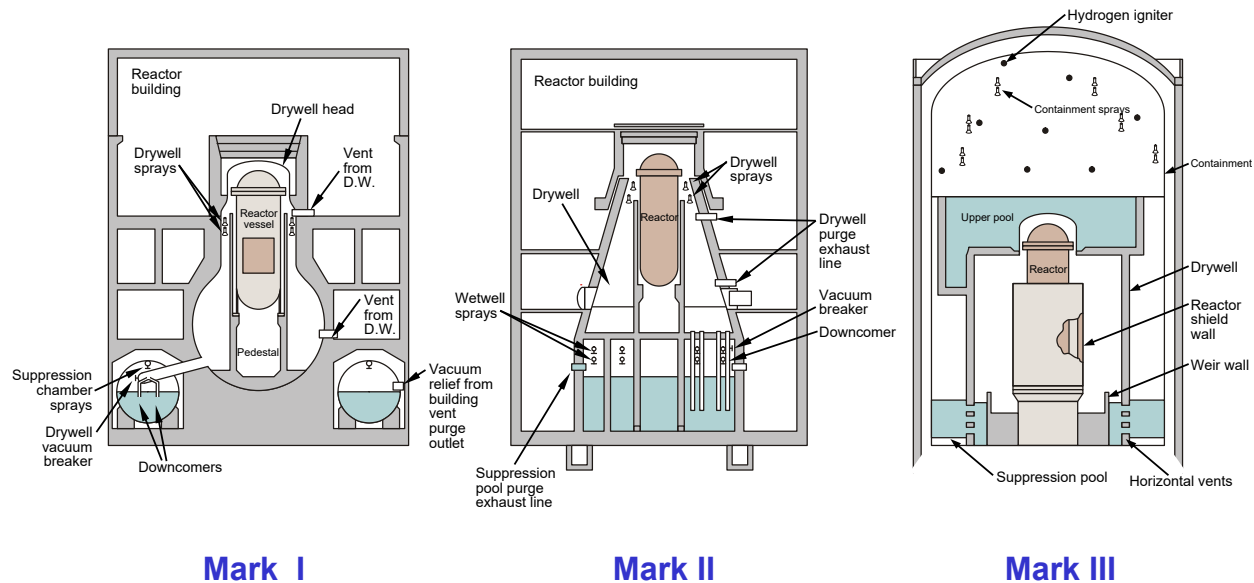


Fig. 2 Comparison of typical BWR containments Galyean W and Rempe J (2005) *Accident Progression Analysis (P-300)*, U.S. NRC, <https://www.nrc.gov/docs/ML1204/ML12044A213.pdf>, May 2005.

Containment pressure and temperature control

Any mass and energy released into the drywell, from either pipe breaks or relief valve discharge, can result in an increase in drywell pressure. The increased drywell pressure forces the steam and air mixture into the suppression pool water where the steam is condensed, thereby reducing the drywell pressure. In the Mark I and Mark II designs, drywell and the volume above the suppression pool water also has containment spray capability to further reduce pressure and temperature.

In the long-term, the containment pressure and temperature are controlled by transferring heat to the ultimate heat sink by the drywell and wetwell sprays. The sprays take suction from the suppression pool water and spray into the drywell or wetwell after passing through heat exchangers to remove the heat. In the event of a LOCA, drywell and/or wetwell sprays are manually initiated, according to emergency operating procedures, within 10 min of initiation of the event.

The Mark I and Mark II plants have a small drywell volume that can rapidly pressurize during an accident and therefore have a relatively high-pressure capability. The Mark III has a much larger containments volume but with a smaller pressure capability. The Mark III containment has large safety grade ventilation units with a.c. powered fans drawing air past cooling coils supplied with water from the service water system. Some Mark III containments also have containment sprays to aid in heat removal. Suppression pool cooling, using the RHR system, is used to remove decay heat once containment or drywell pressure is lowered by sprays.

Emergency power

All ESFs are powered from emergency a.c. busses. The emergency a.c. busses are divided into at least two independent divisions. All engineered safeguards equipment, such as pumps and motor operated valves in a safeguards train, is powered from the same emergency bus of the same division so that a failure of one emergency a.c. bus will not disable both trains for safeguards. The source for the emergency a.c. power busses are either offsite power or emergency diesel generators. The emergency a.c. power busses are always energized from the offsite power source. Upon loss of the offsite a.c. power source, a signal is sent to start the emergency diesel generators. Once the emergency diesel generators are running, safeguards equipment is sequentially loaded onto the busses by automatic sequencer to avoid an overload due to simultaneous starting of various safeguards pumps.

All safety-related instrumentation and control systems are powered from emergency d.c. busses. The emergency d.c. busses are divided into at least two independent divisions. As in the case of a.c. power, control power for all safeguards equipment in a safeguards train is powered from the same emergency bus to prevent failure of one d.c. bus from causing failure of safeguards functions. The emergency d.c. busses are normally powered from inverters that draw from the emergency a.c. busses. In some instances, emergency d.c. busses may draw directly from the station batteries which are kept at full charge by battery chargers powered from emergency a.c. busses. In the event of a loss of all a.c. power, the station batteries are available to power critical d.c. busses that were previously powered from inverters.

Cooling water

There are two key cooling water systems that support the ESFs: Component Cooling Water and Essential Service Water. Component cooling water is used to cool equipment and heat exchangers from which leakage of radioactive materials into the cooling water could result in a release of radioactivity to the environment in the event of an accident. Service water is used to remove heat from the component cooling water system using a heat exchanger between the two systems, as well as other safeguards equipment such as motors and room coolers. Thus, the component cooling water system acts as an intermediate cooling loop which gives up its heat absorbed from components potentially carrying radioactive fluids to the essential service water system. For example, the cooling water used for the heat exchangers in the ECCS is component cooling water. The essential service water system transfers heat to the ultimate heat sink. The service water system also has a non-essential service water loop that removes heat from non-safety grade equipment such as the main turbines.

There are at least two trains of essential service water and component cooling because they serve ESFs. The cooling water trains and the electrical power for these trains is arranged in such a way that failure of one train of cooling water will not result in the failure of an ESF.

Compressed air

If compressed air is used in the functioning of any ESF components (e.g., pneumatic valves), then safety related compressed air trains are required. As in the case of electrical power and cooling water, at least two compressed air trains are provided and arranged so that the failure of one train would not fail an ESF.

Summary

SSCs that have a function to mitigate the consequences of a design basis accident are classified as plant ESFs. These SSCs are subject to more stringent design requirements, as well as regulatory scrutiny, including the effects of a failure of a key component or interfacing system. These ESFs are also subjected to rigorous surveillance and testing to assure they can reliably perform their function.

The ESFs for LWRs include those SSCs that are required to return and maintain the plant in a safe, stable state following a design basis accident and prevent the release of radioactivity to the environment. These SSCs perform the functions of core subcriticality, core cooling, RCS/RPV pressure control, and primary containment pressure and temperature control.

Addendum

Early plants did not have many of the ESFs that we think of today as integral to nuclear safety. Three events had a profound influence on today's basis for ESFs:

- Licensing of second generation plants with higher power levels between 1965 and 1975
- WASH-1400, Reactor Safety Study that was published in 1975 ([NRC, 1975](#))
- The Three Mile Island Unit 2 (TMI-2) accident in 1979 ([Skillman and Rempe, 2021](#))

Two early plants in particular can be used to illustrate the differences in ESFs in the first generation of plants (the original designs of Yankee Rowe and San Onofre Unit 1). Also included is a brief discussion of the Zion and Indian Point Units 2 and 3 designs, which were completed in the 1960s. Note that this information is based on personal experience of the authors of this chapter. A detailed search of the open literature could not identify any sources to validate this information. Please use this at your own risk.

Yankee Rowe (Operated 1961 to 1981 – Construction began in 1958): The original design had a single train low pressure ECC system (100 psi shutoff head) to guard against a large break LOCA (See Maximum Hypothetical Accident identified in WASH-740 ([AEC, 1957](#))) and two high pressure make-up pumps of about 30 gpm each. There was no provision for ECCS recirculation. There was also no provision for emergency feedwater (EFW) until after TMI-2. After TMI-2, EFW was implemented with three 50% capacity trains. The plant was continually upgraded in a piecemeal fashion so that it met regulatory requirements until it was shut down in 1981.

SONGS -1 (Operated 1968 to 1992 – Construction began in 1964): The original design did not have a dedicated ECCS to guard against large pipe breaks. Main feedwater could be connected (single train) to the RCS and a method of boration devised to provide protection against large pipe breaks in the RCS. Like Yankee Rowe, it did not have an AFW system until after the TMI-2 accident. It is not known if upgrades were made for RCS water injection. SONGS-1 was permanently shut down in 1992.

Zion (Operated by Commonwealth Edison from 1973 to 1998/ Construction began in 1968), Indian Point Unit 2 (Operated by Consolidated Edison from 1974 to 2020/Construction began in 1966), and Indian Point Unit 3 (Operated by the Power Authority of the State of New York from 1976 and presently operating in 2020/Construction began 1968): Zion and Indian Point Units 2 and 3 were really the fore runners of the second generation of nuclear plant designs in the USA. Due to their proximity to large population centers in Chicago and New York City, several additional safety features were required. Redundant and diverse safety systems were included in the design basis, some of which were not required by the NRC for plants in less densely populated areas. Following

the TMI-2 accident, the plant operators of Zion and Indian Point Units 2 and 3 commissioned plant specific full PRAs that included external events, such as earthquakes and fires (ComEd, 1981; ConEd and PASNY, 1981) to compare the “risk” from these plants to that for the reference plant used in WASH-1400 (NRC, 1975).

Results from these PRAs indicated that the ‘risk’ from each of these units was less than the reference plant evaluated in WASH-1400 (NRC, 1975). The designs for ESFs for the Zion and Indian Point plants were significantly different than the ESF design for the reference plant used in WASH-1400. There were also advances in PRA methodologies for these studies that led to more realistic results. The NRC and their contractors performed an in-depth review of the methodologies used (NRC, 1982, 1984) and concluded that there were no significant omissions or errors. In the case of Indian Point Units 2 and 3, the NRC subsequently concluded that that neither unit was a risk outlier and therefore neither shutdown nor imposition of additional remedial actions was warranted (NRC, 1985). The NRC came to the same conclusion with regard to the Zion plant although no reference could be found.

These units had two unique features to reduce fission product leakage from the primary containment that are not found on most other PWRs:

- A dedicated ESF to assure containment isolation by pressurizing the pipe volume between the inside and outside containment isolation valves (Isolation Seal Water System for liquid penetrations and Penetration Pressurization System for air and gas penetrations). These systems provided an extra measure of isolation in the event that the isolation valves would leak following an accident. These systems were unique to the Zion and Indian Point Units 2 and 3 due to their proximity to high population areas.
- A fission product removal mechanism on each of the two diverse containment heat removal systems. A chemical additive was included in the containment spray system to raise the pH of the spray water to enhance removal of radioactive iodines by the spray droplets and retain them in the containment sump. The containment fan coolers included High Efficiency Particulate Air (HEPA) and charcoal filters for removal of radioactive iodines, including organic iodines. While many PWR primary containment buildings included these diverse containment heat removal systems, the charcoal filters for iodine removal were unique to the Zion and the Indian Point Unit 2 and 3 plants due to their proximity to high population areas.

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Instrumentation and Control

Robert Lutz^a, ^a Lutz Nuclear Consulting, Hendersonville, NC, United States

Bill Williamson^b, ^b TVA, Nuclear Plant Road, Athens, AL, United States

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Abbreviations

BWR Boiling Water Reactors
IAEA International Atomic Energy Agency
IEEE Institute of Electrical and Electronic Engineers
NEI Nuclear Energy Institute
NRC United States Nuclear Regulatory Commission
PWR Pressurized Water Reactors
SPDS Safety Parameter Display System
TMI-2 Three Mile Island Nuclear Plant Unit 2
TSC Technical Support Center
WENRA Western European Nuclear Regulators Association

Introduction

Few people work under the 5-acre (2 ha = 20,000 m²) roof of a nuclear power plant. Beyond the confines of the control room, one's first impression might well be that the place is deserted; to a major extent, the plant runs itself. Most of the human effort required to monitor and adjust the plant systems is exerted in the control room, which can be either disconcertingly quiet or very much alive with signals and alarms. During plant operation, small groups of workers go about their business of performing surveillance and routine maintenance on the myriad of plant equipment, including calibration of instrumentation. Operators in the control room stay in touch with anyone out in the plant by means of a paging system.

There are two basic functions of instrumentation in a nuclear power plant—those needed for production of electricity and those needed for safe operation. The former might include turbine generator parameters and electrical distribution parameters for non-safety systems and equipment. The later include all of those associated with the reactor and its cooling systems as well as operation of engineered safety equipment and of systems to monitor radiation levels and maintain radioactive barriers. This article is focused on those needed for safe operation of the plant.

Nuclear physics, as all science, is logical, and those instruments are speaking the truth to anyone logical enough to recognize it. Defeat awaits the individual who even subconsciously ascribes human qualities like crankiness to functioning meters and thermocouples simply because they are bringing bad news. The answer in science is never to shoot the messenger (Rogavin and Frampton Jr., 1980).

The operators have all the information needed for safe operation of the plant by monitoring various parameters (pressures, temperatures, flows, levels, etc.) of various plant systems via instrumentation. This gives an indication of the plant "health";

operators can initiate many actions from the control room to maintain these parameters within predetermined ranges. When plant parameters exceed a predetermined value (either too high or too low), alarms sound and lights flash on the alarm panel in the control room to alert the operators of an unacceptable condition. These are driven by what the instruments sense is happening in the plant. The plant operators continuously observe information presented by the instrumentation and make note of trends that might permit them to take action before these alarms sound. In some cases, multiple alarms may announce nearly simultaneously making it difficult to immediately draw any sort of rational picture of something happening out in the vast plant.

Under abnormal or accident conditions, the instrumentation also provides input to automatically actuate protection systems if predetermined values for some parameter values are exceeded. The automatic actuation of the Reactor Protection System (RPS) provides rapid shutdown of the reactor. The RPS actuates a reactor trip by causing the control rods to be inserted into the core; this is sometimes called the Reactor Trip System (RTS). The RPS may also cause the automatic actuation of the Engineered Safety Features (ESF) to provide rapid starting of emergency water injection and cooling to protect the reactor core. The RPS and ESF are discussed in detail in [Lutz and Williamson \(2021a,b\)](#).

The components and systems that make up the instrumentation and control (I&C) function provide defense in depth which includes the use of redundancy, diversity, separation and fail-safe behavior. In some references, the control function refers only to those functions not related to safety; the control function of instrumentation related to safety is referred to as the protection function. In this article, I&C refers to the safety related protection function of instrumentation.

Redundancy provides protection against independent single failures and prevents a random independent failure in one component or system from disabling the desired function. Several similar instrument channels may be used to measure the same physical variable, and a logic decision is then applied to the redundant signals to identify and discard an erroneous channel. Each channel has its own independent power supply to ensure that a common power supply fault will not prevent the measurement of the variable.

Diversity is the use of two or more physically or functionally different means of performing the same function. It protects against certain types of common mode failures, such as those arising from design or maintenance errors. Diversity can be provided by the use of equipment and software of different designs or origins to carry out the same function. Failures due to errors in the design of one system thus do not affect the performance of the other.

Separation refers to the use of physical separation such as barriers or distance. Separation provides protection against common mode or cross-linked effects, such as fires and missiles, and prevents a fault from migrating from one system to another.

The failure-to-safety or fail-safe principle is widely used in I&C systems important to safety. In principle, equipment is selected and designed with a strong bias towards a certain output signal in the event of failure. A classic example of this principle is applied to motor and pneumatically operated valves in which failure of the electrical or pneumatic system causes the valve to either open or close.

However, electronic and computer-based components and equipment can have unknown software errors that might not be identified for all possible situations during software verification and must be used with care.

Historical events

The instrumentation in a modern nuclear power plant is a product of regulatory and industry standards that have been developed as a result of historical events in which instrumentation deficiencies have played a role. Regulatory standards refer to rules that have been developed by national regulators such as the US Nuclear Regulatory Commission (NRC); this includes standards developed by the International Atomic Energy Agency (IAEA) and the Western European Nuclear Regulators Association (WENRA). Industry standards refers to mandatory standards developed by industry groups such as the Institute of Electric and Electronic Engineers (IEEE). In some cases, regulatory standards endorse industry standards with clarifications when appropriate. For example, NRC Regulatory Guide 1.97 ([NRC, 2019](#)) endorses IEEE-497 ([IEEE, 2016](#)).

A concise history of the role instrumentation played in nuclear plant accidents can be found in [EPRI \(2015\)](#). Details for several key events that have had a significant impact on instrumentation systems follows.

Three Mile Island Unit 2 accident

On March 28, 1979, a loss of feedwater event occurred at the Three Mile Island Unit 2 (TMI-2) resulting in a core meltdown that was partially caused by deficiencies in instrumentation as well as the operators' understanding of the feedback provided by instrumentation ([Skillman and Rempe, 2021](#)). In this case, a pressurizer relief valve failed to re-close, but the instrumentation indicated that it was indeed closed (the instrumentation was based on control power and not valve position). The pressurizer water level started to increase as the gas bubble in the top of the pressurizer was lost and the coolant pressure started to drop as a result of the open relief valve. Safety injection was automatically initiated by the instrumentation indicating low coolant pressure, but the operators turned off safety injection because the indicated pressurizer water level was too high and rising. The core continued to overheat as the operators tried to make sense of the various instrument readings and missed several opportunities to conclude that the core had been badly overheated. High reactor coolant system (RCS) hot leg temperatures and high core exit thermocouple temperatures indicated a severely overheated core, but this information was "dismissed." Similarly, a rising core neutron flux indicated that the core was slumping downward, but no one was able to assimilate this information to a correct diagnosis. Other instrumentation data

including containment pressure, containment radiation levels, containment sump water levels and radwaste radiation levels were not considered in the diagnosis until the accident was essentially arrested.

The ways that instrumentation contributed to the accident are detailed in NUREG-0660 (NRC, 1980a) and NUREG-0737 (NRC, 1980b). Significant changes in the requirements for instrumentation were made as a result of the requirements in both Regulatory Guide 1.97 Revision 2 (NRC, 1980c) and IEEE Standard 497-1979 (IEEE, 1977). There were also significant changes made in the display of certain post-accident plant parameters (called the Safety Parameter Display System or SPDS) in the main control room (NRC, 1980d). Additionally, there were requirements for new facilities, a Technical Support Center (TSC) and an Emergency Offsite Facility (EOF), in which the SPDS would be available for plant emergency response staff to perform independent assessments of the plant conditions and trends.

In conclusion, while there were some deficiencies in instrumentation itself, the more important deficiency was in understanding what the instrumentation was trying to tell the operators. This led to improved emergency operating procedures (called symptom-based procedures) to guide the operators through a process of understanding all the information available from the instrumentation. The new guidance is discussed in detail in Lutz and Williamson (2021c).

Note that a similar loss of feedwater event occurred in 1978 at the Davis-Besse plant, but the operators correctly diagnosed that the RCS had become saturate did not terminate safety injection. This information was never shared with other plant operators, and proposed changes to emergency operating procedures were not finalized (Rogovin and Frampton Jr., 1980; Skillman and Rempe (2021)).

Oconee loss of power

On November 10, 1979, an event occurred at the Oconee Power Station, Unit 3, that resulted in loss of power to a panel that supplied power to the Integrated Control System and the Nonnuclear Instrumentation System. As described in NRC Information Notice 79-29 (NRC, 1979), this loss of power resulted in control system malfunctions and significant loss of information to the control room operator. Although this event resulted in the loss of some instrumentation and control, no significant changes to instrumentation itself were required; it is included here for completeness because it resulted in a loss of instrumentation. The event resulted in investigations into power supplies for instrumentation, as well as guidance for recovery from a loss of instrument power.

Crystal River DC bus failure

On February 26, 1980, an event occurred at Crystal River Unit No. 3 that involved a partial loss of non-nuclear instrument and control power (NRC, 1980e). Although this event resulted in the loss of some instrumentation and control, no significant changes to instrumentation itself were required; it is included here for completeness because it resulted in a loss of instrumentation. The event resulted in investigations into power supplies for instrumentation, as well as guidance for recovery from a loss of instrument power.

World Trade Center attack

Although no nuclear power plants were threatened, the terrorist attack at the World Trade Center in New York City on September 11, 2002 played an important role in enhancing the capabilities of using all available indications from plant instrumentation to determine the plant status and plan a response. Following the World Trade Center attack, the US NRC issued an Order that, in part, resulted in industry developing alternate methods to determine the status of key plant parameters when there is a simultaneous loss of all a.c. and d.c. power distribution in a plant as a result of a terrorist attack (NEI, 2006). Although this did not result in any new instrumentation, it led to the development of guidelines for the use of portable instruments and instruments that do not require electrical power to retrieve key plant parameters (Lutz, 2015).

Fukushima accident

On March 11, 2011, an earthquake followed by a tsunami of great magnitude struck the Fukushima Daiichi nuclear power site in Japan that severely crippled the ability to understand the status of the plants as well as the ability to control plant equipment from the control room (Mizokami and Rempe, 2021). The earthquake severed the main offsite electrical supply to the plant, but the emergency diesel generators responded as designed to maintain critical power to equipment and instrumentation. The following tsunami flooded the diesel generators and eventually, the station batteries, ultimately resulting in a complete loss of both a.c. and d.c. power at Units 1 and 2 while Unit 3 had d.c. power for 20–30 h. Without instrumentation, the operators were “flying blind” as they struggled to maintain control of the plant. After some delay, batteries from the plant were connected to provide power to critical instrumentation.

Incorrect and inadequate instrumentation data compounded the diagnosis of the accident progression. For example, the reference leg of the primary set of the reactor vessel water level instrumentation boiled off resulting in an erroneous indication of reactor vessel level in the normal range while a secondary set of level instrumentation indicated a more realistic very low water level; this is further discussed in “Uncertainty and error” section. The operators were confused by the differences but relied on the primary indication. They had never been exposed to this scenario with disagreeing indications in their training. During the event, there was also

concern (that proved to be unfounded) about fuel overheating and melting in the spent fuel pools caused by boil-off of water as a result of the lack of cooling and makeup to the pools.

Based on insights from the events at Fukushima Daiichi, new instrumentation requirements were ordered to monitor plant conditions. Specifically, the US NRC determined that all power reactor licensees must have a reliable means of remotely monitoring spent fuel pool levels. The NRC issued an order (NRC, 2012) to all licensees requiring wide-range spent fuel pool level instrumentation; the existing spent fuel pool instrumentation could only measure within the normal pool level.

The nuclear industry recognized the importance of instrumentation in preventing an accident caused by external events (e.g., earthquakes and floods) from progressing to a severe accident. As a result, all components of instrumentation (power supplies, cables, sensor, etc.) were reviewed to assure that the power supplies (station batteries and battery chargers) are protected from extreme events and that a diverse means is available to power the battery chargers (NEI, 2016).

The events at Fukushima have also shown that severe accidents can subject electrical and instrumentation and control (I&C) equipment to environmental conditions exceeding the equipment's original design basis assumptions. Severe accident conditions can then cause rapid degradation or damage to various degrees up to complete failure of electrical and I&C equipment. Subsequently, the IEEE issued IEEE Std. 497-2016 (IEEE, 2016), which broadened the scope of the standard to include consideration of the reliability of instrumentation potentially required for coping with severe accidents. In other words, those instruments that would be useful in recovering from a severe accident should provide reliable indication when exposed to severe accident pressure, temperature and radiation conditions. In developing this new standard, the IEEE worked closely with the International Atomic Energy Agency (IAEA) and the US NRC. This resulted in new regulatory requirements (NRC, 2019; IAEA, 2017), similar to the IEEE requirements, for reliable instrumentation that are useful in mitigating a severe accident. While these requirements are only applicable to new plant designs, some existing plants are following the requirements when instrumentation is replaced.

Finally, as a result of the operator performance during the accident related to instrumentation indications, the generic severe accident guidance developed by both the BWR and PWR Owners Groups was updated to include a new Technical Support Guideline (TSG) related to instrumentation performance during a severe accident (Lutz, 2015; Ellison and Williamson, 2015). This is described in more detail in [Application to Procedures and Guidelines](#).

Instrumentation

This section focuses on the various aspects of instrumentation including its classification, treatment of uncertainties and errors, application to procedures and guidance, and measurement of key plant parameters.

Instrument classification

Instruments can be classified by their function and by their quality requirements to ensure the desired level of reliability. The first classification, which is the function, is referred to as "Type" (NRC, 1980c, 2019).

- TYPE A Variables: Those variables that provide the primary information needed to permit the control room operating personnel to take the specified manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events.
- TYPE B Variables: Those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining RCS integrity, and (4) maintaining containment integrity (including radioactive effluent control).
- TYPE C Variables: Those variables that provide information to indicate the potential for being breached or the actual breach of barriers to fission product release. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.
- TYPE D Variables: Those variables that provide information to indicate the operation of safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in operation of systems important to mitigating the consequences of an accident.
- TYPE E Variables: Those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and for continually assessing such releases.
- TYPE F Variables: Those variables that provide primary information to indicate fuel damage and the effects of fuel damage (not applicable to currently operating plants).

Design and qualification criteria for the instrumentation used to measure these variables can be separated into four groups or categories that provide a graded approach to requirements depending on the importance to safety of the measurement of a specific variable.

- Category 1 provides the most stringent requirements and is intended for key variables.
- Category 2 provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status.

- Category 3 is intended to provide requirements that will ensure that high quality off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation.
- Category 4 is intended to provide requirements for measurement of key variables when the plants conditions (e.g., pressure, temperature and radiation) are beyond design basis conditions.

A key variable is that single variable (or minimum number of variables) that most directly indicates the accomplishment of a safety function (in the case of Types B and C), the operation of a safety system (in the case of Type D), or radioactive material release (in the case of Type E). It is essential that key variables be qualified to the more stringent design and qualification criteria. The design and qualification criteria category assigned to each variable indicate whether the variable is considered to be a key variable for system status indication or for backup or diagnosis, i.e., for Types B and C. Key variables are Category 1; backup variables are generally Category 3. For Types D and E, the key variables are generally Category 2; backup variables are Category 3.

Category 1 instrumentation has the following design and qualification requirements:

- The instrumentation should be qualified through testing to be reliable in the environmental conditions that would exist for a design basis accident, including seismic effects; see for example (NRC, 1984).
- The instrumentation should have redundancy in indications and power supplies such that no single failure would prevent plant operators from being presented with the information.
- The instrumentation should be powered from standby power sources and should be backed up by batteries.
- The instrumentation should have continuous readout and at least one channel should be recorded; the frequency of parameter sampling for display and recording is not specified in NRC (1980c).
- The instrumentation should be maintained and calibrated on a periodic basis.
- The instrumentation should receive inputs from sensors that directly measure the desired variables, to the extent possible.

Examples for BWR and PWR instrumentation in each of the types is provided in Table 1.

Uncertainty and error

Errors in instrumentation readouts can be attributed to calibration errors, instrument failure errors, known instrument errors, and errors induced by environmental conditions. Calibration errors are typically small, and procedures are designed to ensure that the same calibration errors are not made on multiple channels of instruments reading the same variable. Multiple channels of an instrument are designed to ensure that errors caused by instrument failure are detectable (e.g., one of four channels reading very different from the other three).

Table 1 Example BWR and PWR variables for each type (not all inclusive).

Type	Example BWR variable	Example PWR variable
A	Drywell Pressure	Steam Generator Water Level
B	Neutron Flux	Neutron Flux
	Reactor Vessel Coolant Level	Pressurizer Water Level
	Reactor Vessel Pressure	Core Temperature ^a
		RCS Pressure
C		Containment Pressure
	Radiation Level in Circulating Coolant	Radiation Level in Circulating Coolant
	Primary Containment Radiation	Containment Radiation
	Suppression Pool Water Level	Refueling Tank Water Level
D	Drywell Oxygen and Hydrogen Level	Containment Hydrogen Level
	Main Feedwater Flow	Main Feedwater Flow
	Condensate Storage Tank Level	Condensate Storage Tank Water Level
	Emergency Core Cooling System Flows	Containment Spray Flow
E	Containment Temperature	Containment Temperature
	Safety Injection Flows	Safety Injection Flows
	Area Radiation	Area Radiation
	Radiation in Exhaust Lines	Radiation in Exhaust Lines
F	Vent Flow Rates	Vent Flow Rates
	Meteorological Conditions	Meteorological Conditions
	Core Temperature ^a	Core Temperature ^a
	Reactor Vessel Pressure	RCS Pressure
	Drywell Pressure	Containment Pressure
	Drywell Temperature	Containment Temperature

^aThere is no direct indication of "Core Temperature"; it must be inferred from other parameters such as core exit thermocouples, reactor coolant superheat, etc.

Every instrument has a known error band (e.g., plus or minus 1%). This error is usually referenced to the maximum range of the instrument. Therefore, a 1% error on reactor vessel pressure [which has a maximum range of 1500 psig (10.34 MPa-g)] would be 15 psig (0.1 MPa-g). This is quite acceptable in the normal operating range of 1200 psig (8.27 MPa-g) but could be quite large at the lower end of the range (e.g., 20 psig/0.137 MPa-g), where the error could be nearly as large as the measured value. This likely only to occur when the instrument readout is used as a Type F variable to determine if the reactor vessel and drywell pressures are equalized. An example is the use of a narrow-range instrument when making a comparison between reactor pressure vessel (RPV) pressure and drywell pressure.

The final error is that induced by environmental conditions. The most likely instrument to be subject to this is level instrumentation for components inside the primary containment or drywell. Level instruments typically compare the weight of a column of water in a vessel to a known pressure that is near atmospheric pressure. A reference leg inside the primary containment is used for this comparison. If the pressure in the primary containment is high during an accident, the reference leg used for comparison would result in a bias in the instrument readout. Correction factors for environment conditions within the design basis are pre-calculated for steam generator level (PWRs) and reactor vessel level (BWRs).

An extreme case occurred in all three of the units that were operating at Fukushima Daiichi on March 11, 2011 (see [Mizokami and Rempe, 2021](#)). As shown in [Fig. 1](#), when water in the reference leg for the reactor vessel level instrumentation boiled-off due to high temperatures in the drywell (i.e., primary containment); instrumentation (i.e., a water level gauge using a differential pressure, dP, cell) will indicate that water is above the top of the core when in fact, the water level is below the top of the core.

Application to procedures and guidelines

Instrument errors need to be addressed in Emergency Procedures and Guidelines. For normal instrument error, the setpoints for operator action are adjusted for the maximum expected error in PWRs. For BWRs, the as-is indication (corrected for any off-calibration) is used for operator actions because it is not possible to identify the conservative bias in all cases.

For errors caused by environmental conditions in PWR emergency procedures, two setpoint values (directing operator actions) for steam generator level are provided—one for low containment pressure and another for high containment pressure (called adverse conditions).

For severe accident guidelines, special Technical Support Guidelines ([Lutz, 2015](#); [Ellison and Williamson, 2015](#)) were developed to assist the emergency response staff for several reasons:

- If the plant conditions are beyond the design basis and instrumentation qualification envelope, the instrument reliability could be impacted,
- The instrument response to some operator actions during a severe accident may be outside the expected instrument response for design basis accidents,
- The instrument response during a severe accident may provide some diagnostic assistance to the emergency response staff if the expected response to various actions are identified,
- The indications from the primary set of instrumentation for most key plant parameters can be validated using a diverse set of instrumentation, including the use of computational aids to translate unrelated parameters to the key parameter.
- Alternate means of obtaining most parameter values are available if power to instrumentation is unavailable ([Lutz et al., 2016](#)).

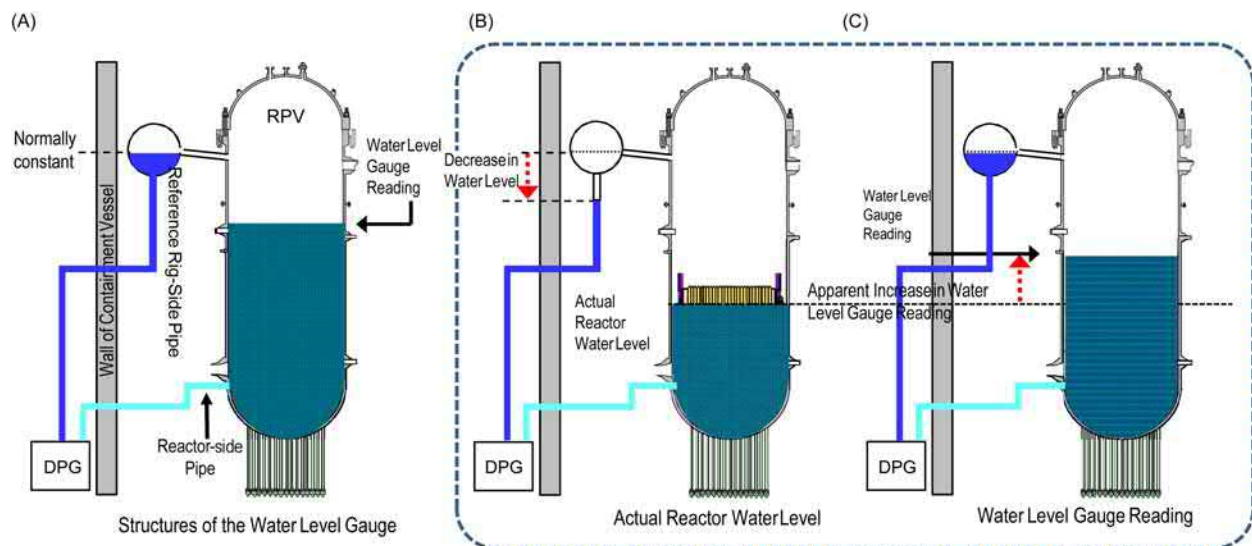


Fig. 1 Water level instrument diagram. Courtesy of TEPCO Holdings, taken from TEPCO (2017) *The 5th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident*. TEPCO Holdings.

Measurement of key plant parameters

This section provides examples of instrumentation used to measure key plant parameters in a nuclear power plant. More detailed information is provided in Hashimian (2014) and Upadhyaya and Wood (2020).

- Water Temperatures: Water temperature is typically measured by either thermocouples (TCs) or resistance temperature detectors (RTDs). The choice depends on the expected temperature range of the water.
 - RTD sensors are thin film or wire wound resistor elements that use the ohm resistance value to indicate a temperature value. The resistance simultaneously increases with temperature increase and vice versa. RTDs are typically used for measuring water temperature when the range is less than about 700 °F (370 °C).
 - TC sensors consist of two dissimilar metal wires joined at a hot junction. As temperature changes, a millivolt signal is generated. The most common alloy combinations used in PWRs and BWRs are known as Type J for applications less than about 1500 °F (800 °C) and Type K for applications less than about 2000 °F (1100 °C). They are typically used for temperature measurements requiring faster response times.
- Pressure: Liquid or gas pressure is typically measured with either an elastic element gauge such as a bellows or diaphragm, or electrical transducers such as strain gauges.
- Water Level: Liquid water levels are typically measured by a differential pressure (dP) cell. As discussed in “Uncertainty and error” section (see Fig. 1), a dP cell measures the difference in pressure between the bottom of the column of water and the pressure of a column of water in a “reference leg” that is exposed to a known pressure. Adjustments for temperature of the measured fluid and that of the water column in the reference leg are needed for an accurate measurement if temperatures are expected to vary significantly.
- Neutron Flux: Neutron flux is typically measured using three separate instruments with varying shielding because the range of energy of thermal neutrons that can cover eight order of magnitude. Source range, intermediate range and power range neutron detectors are normally used in PWRs and BWRs with the source range used to detect subcriticality, the intermediate range is used during startup and shutdown and the power range is used during normal operation.
- Hydrogen: Advances in hydrogen measurement results in two different detector methods in use today. The older method, which draws a sample of gas containing air, steam and hydrogen, splits the sample and burns all hydrogen in the first sample. The difference in electrical conductivity of the two samples is a measure of the hydrogen concentration because hydrogen molecules have a significantly different conductivity compared to air. The newer method measures the temperature of a hydrogen passive recombiner surface. That temperature is correlated to the hydrogen concentration because higher hydrogen concentrations have more rapid recombination.
- Radiation: The most common radiation measurement is for gamma radiation. The ionizing effect of radiation is measured using various methods including scintillation detector, ionization chamber and gamma counter (Han, 2000).
- Flow Rates: Flow rates are most commonly measured by determining the pressure drop through an orifice or nozzle using common pressure measurement devices.

Control (protection) function

This section focuses on the safety related control function of instrumentation. Control systems are designed to maintain the reactor within operational limits during power operation of the plant. For example, a control system monitors the water level in the steam generators of a PWR and adjusts the incoming feedwater flow to maintain that level within prescribed limits. These limits are well below the safety limits to ensure safety during operational disturbances, including accidents. Another part of the plant control system is designed to shut the reactor down, keep it shut down and ensure core and system cooling as well as prevention of radioactivity releases if plant parameters exceed certain plant safety limits.

The safety related control systems used to maintain plant systems within acceptable bounds in the event of an off normal event or accident event can be classified as those having either an automatic control function or a control function initiated by the plant operators. The automatic control functions are typically the RPS actuation or the ESF actuation. These are controls that are needed to be actuated quickly to protect the core and prevent radioactivity releases. Control functions initiated by the plant operators during an accident are generally those to terminate operation of safety systems or to perform longer term actions following ESF actuation. In addition, all of the ESF systems can be manually actuated by the plant operators as instructed in the plant EOPs. There are also a limited number of operator initiated control actions for which no automatic actions are available. The instrumentation to guide the operators for these actions are classified as Type A instrumentation (see section Instrumentation Classification). Examples of PWR and BWR control functions for off-normal and accident conditions are provided in Table 2.

As a measure of defense in depth, safety limits for RPS (see Lutz and Williamson, 2021a) actuation are set below calculated allowable limits to ensure that any unanticipated delays in protection system response will not cause unacceptable consequences. The setpoints for ESF (see Lutz and Williamson, 2021b) actuation are chosen based on safety analyses to prevent damage to the reactor core and to prevent the release of radioactivity to the environment. The introduction of digital systems in place of analog systems in some areas of the protection systems has been done as analog systems become obsolete. The improved reliability of digital systems, simplified surveillance testing and reduced duration of plant shutdown for testing are also benefits of the digital systems.

Table 2 Example control systems required for emergency conditions.

Function	PWR	BWR
RPS	Reactor Trip	Reactor Trip
	ATWS Trips	ATWS Trips, Alternate Rod Insertion
ESF	Safety Injection Initiation	Core Spray Initiation, Low- and High-Pressure Coolant Injection Initiation, Automatic Depressurization Initiation
	Containment Spray Initiation	
	Containment Isolation	Primary Containment Isolation including Reactor Water Cleanup Isolation Initiation
	Steam Line Isolation	Main Steam Line Isolation
	Turbine Trip and Feedwater Isolation	Turbine Trip
	Auxiliary Feedwater Initiation	Reactor Core Isolation Cooling Initiation
		HPCI Isolation Initiation, RCIC Isolation Initiation
	Safety Injection Automatic Switchover to Containment Sump	HPCI Auto Switchover from Condensate Storage Tank to Suppression Pool
Post-Accident Manual Control	ESFAS Interlocks	
	RCS Boration to Shutdown	Manually initiate Standby Liquid Control (SLC) and Alternate Rod Insertion (ATWS)
	Safety Injection Termination	Control Safety Injection and RCIC Termination
	Steam Generator Level Control with Auxiliary Feedwater	Initiate Containment Sprays and Suppression Pool Cooling
	Defeat Containment Isolation	Initiate Containment Inerting

Control systems typically rely on inputs from several independent monitors of a particular plant parameter, called instrument channels. In some cases, input for more than one plant parameter is needed for actuation (i.e., coincident signals). These inputs are then “auctioneered” to ensure that a single erroneous input (e.g., from a failed instrument) does not result in actuation of the control function. For example, PWR reactor trip actuation requires a low steam generator water level input from two-out-of-four instrument channels from any single steam generator. Power supplies for the control function are subject to the same requirements as those for the instrumentation providing input to the control system.

Summary

Reliable and accurate instrument is key to safe operations of a nuclear power plant. Instrumentation connects the plant operations staff to the plant conditions by providing feedback regarding the value of key plant parameters. The term instrumentation includes the sensors which measure a plant parameter and convert that measurement to a signal (voltage, resistance, amperage); cables that carry the signal, transmitters that amplify the signal; processors that convert the signal into a useful indication; displays (meters or readouts) that allow the plant operations staff to observe (and take actions); and recorders that preserve the information for further use.

Nuclear power plant instrumentation has been enhanced over time to provide plant operators and emergency responders with more reliable means to understand the plant conditions during an event. Lessons learned from instrumentation issues during plant events (including but not limited to the TMI-2 and Fukushima accidents) have been considered to improve instrument reliability as well as development of guidance for plant operators for interpreting instrumentation feedback.

Control systems are designed to maintain the reactor within operational limits during power operation of the plant. These limits are well below the safety limits to ensure safety during operational disturbances, including accidents. Another part of the plant control system is designed to shut the reactor down, keep it shut down and ensure core and system cooling, as well as prevention of radioactivity releases, if plant parameters exceed certain plant safety limits. Control systems used to maintain plant systems within acceptable bounds in the event of an off normal event or accident event can be classified as those having either an automatic control function or a control function initiated by the plant operators. The automatic control functions are typically the RPS actuation or the ESF actuation. Control functions initiated by the plant operators during an accident are generally those to terminate operation of safety systems or to perform longer term actions following ESF actuation. These control systems are subject to the same redundancy, diversity, separation and fail-safe requirements as the instrumentation that provide input to the control systems.

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Guidance for Operating Within Design Basis Conditions

Robert Lutz^a and Bill Williamson^b, ^a Lutz Nuclear Consulting, Hendersonville, NC, United States; and ^b TVA, Athens, AL, United States

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Glossary

ATWS Anticipated Transient Without Scram.

BWR Boiling Water Reactor.

Emergency Operating Procedure A set of instructions that are to be implemented following an unintentional reactor trip or safety injection initiation.

Guidance Any set of instructions that are *recommended* to be implemented under prescribed plant conditions.

Procedures A set of instructions that are *required* to be implemented verbatim under prescribed plant conditions.

PWR Pressurized Water Reactor.

Owners Group A group of plant owners with similar plant designs (usually by reactor vendor or reactor type) that fund development programs common to their plant designs.

Special Events Upset conditions or accidents whose safety analysis may not be part of the plant design basis but are approved by the regulator as part of plant licensing in the United States (e.g., Station Blackouts (SBO) and Anticipated Transients Without Scram (ATWS)).

Introduction

Procedures and guidance have always been a part of the operation of nuclear power stations to assure that the plant operation remains within regulatory limits and to protect the plant operator's investment in the facility. The earliest procedures and guidance were focused on operation of the plant as well as responding to likely events that could disrupt normal operation. The procedures were based on recognizing an event and following procedures to respond. The set of events for which procedures were developed were a limited set of design basis events and did not include possible deviations in the event progression. There was no process for incorporation of lesson learned from operating events into the operator procedures. As a result, there was little sharing of knowledge amongst the operating plants.

A significant change came about as a result of the accident at Three Mile Island Unit 2 (TMI-2) in 1979. As emphasized in [Skillman and Rempe \(2021\)](#), investigations into the causes of the accident ([Rogavin et al., 1979](#)) highlighted weaknesses in the plant operating and emergency procedures and guidance, as well as training, in several areas:

- Poor operator procedures that offered no useful assistance,
- A lack of diagnostic skill because operators had never been trained to understand plant parameters,
- Confusion caused by misleading or erroneous instrumentation.

In addition, the investigation by the Kemeny Commission ([Kemeny et al., 1980](#)), which President Carter formed to investigate the accident at the TMI-2 nuclear power plant, recommended the following:

- The (nuclear power) industry should establish a program that specifies appropriate safety standards including those for management, quality assurance, and operating procedures and practices, and that conducts independent evaluations.

- There must be a systematic gathering, review and analysis of operating experience at all nuclear power plants, coupled with an industry wide international communications network to facilitate the speedy flow of this information to affected parties.

This resulted in three significant events that transformed procedures and guidance from the pre-TMI era to today:

- The U.S. Nuclear Regulatory Commission (NRC) upgraded regulatory requirements in the area of emergency procedures and training (NRC, 1980, 1982). Most significant was the requirement for development of symptom-based emergency procedures.
- The nuclear power industry established the Institute of Nuclear Power Operations (INPO), as an autonomous organization to promote the highest levels of safety and reliability in the operation of commercial nuclear power plants. INPO has been particularly responsible for excellence in training as well as a systematic gathering, review and analysis of operating experience at all nuclear power plants, coupled with an industry wide international communications network to facilitate the speedy flow of this information to affected parties.
- The nuclear power created Owners Groups to facilitate the development and maintenance of symptom-based emergency operating procedures that fulfill the new regulatory requirements. A separate Owners Group was created for each American reactor vendor design (Westinghouse, General Electric, Combustion Engineering and Babcock & Wilcox). The Owners Groups were controlled by the plant operators for each reactor vendor design and provided a common source of funding for the development and maintenance of a set of generic symptom-based emergency procedures that could easily be implemented by each reactor operator.

As a point of clarification, at some point after the year 2000, the three Owners Groups from the PWR reactor vendors were combined to form the PWR Owners Group. However, some products of the previous Owners Groups, such as generic Emergency Operating Procedures, are still maintained and updated in their original content and format. It is also noted that in addition to the four Owners Groups in the United States, other Owners Groups were also established (e.g., Framatome/AREVA plants, Canadian CANDU plants and the Nordic BWR plants).

While much of these efforts were aimed at emergency procedures, the fruits of these labors led to similar improvements in all other procedures and guidance, including training.

Plant conditions

The IAEA Safety Glossary (IAEA, 2018) defines five plant states as shown in Fig. 1. These plant states indicate the degree of safety threat associated with that state, ranging from operation states to accident conditions. The three left-most states constitute operation within the plant design basis. The right-most two states constitute design extension conditions. This chapter focuses on the guidance provided for the plant staff when the plant state is within its design basis. Lutz and Williamson (2021) focuses on the guidance provided for design extension conditions (also sometimes called guidance for beyond design basis conditions).

Regulatory guidance (IAEA, 2004; NRC, 2017) specifies that procedures and guidance shall be developed and used by the operating organization to ensure the safe management of the plant. The processes to develop, approve, revise and implement plant procedures should be described in the Safety Analysis Report. A list of the main plant administrative procedures used to control the plant procedures should be also provided.

The “plant design basis” is defined in the IAEA Safety Glossary as the range of conditions and events taken explicitly into account in the design of structures, systems and components and equipment of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits. Therefore, guidance for design basis conditions would refer to

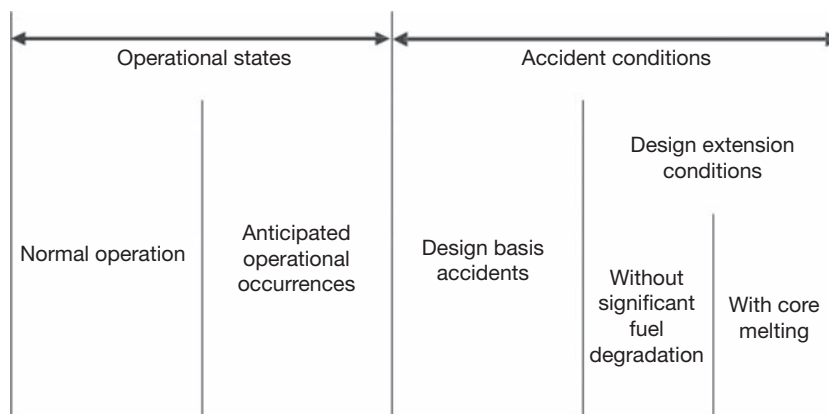


Fig. 1 Five plant states identified by IAEA. Courtesy of IAEA (2018) *IAEA Safety Glossary: Terminology Used In Nuclear Safety and Radiation Protection, 2018 Revision*, International Atomic Energy Agency.

that set of guidelines and procedures that are used when the plant state is either in Normal Operation, Anticipated Operational Occurrences, or Design Basis Accidents.

Several points of clarification, based on acceptable practices in the United States and Europe, are needed here. These are also consistent with regulatory requirements of the IAEA (IAEA, 2016).

- In the United States, there is a difference in terminology between “design basis” and “current licensing basis”. In Europe there is a 10-year periodic review where the plant design basis is updated per regulatory requirements before permission to operate for another 10 years is given. However, because there is no “periodic review” of the plant design basis in the United States, the “design basis” is set at the time of initial licensing of the plant. The “current licensing basis” includes the collection of documents or technical criteria that provides the basis upon which the NRC issues a license and subsequent enhancements to satisfy new regulatory requirements after initial licensing. For simplicity in this chapter, design basis accidents are all of those studied as part of the design basis as well as the licensing basis of a plant.
- Guidance is generally referred to as a set of instructions that are recommended to be implemented under prescribed plant conditions. Procedures are generally referred to as a set of instructions that are to be implemented verbatim under prescribed plant conditions.
- The format and content of guidelines and procedures can vary, as long as they follow human factors principles. Current guidelines and procedures can be arranged as step-wise text, flow diagrams, logic diagrams, etc. Guidelines and procedures are typically thought of in terms of printed paper copies but recently they have begun to be written as electronic media that can be used interactively.
- Plant administrative procedures, while not covered in this chapter, control the development, updating, training and usage of plant guidelines and procedures. These aspects are just as important as the content of the guidelines and procedures themselves because they ensure the quality and consistency of the guidelines and procedures.
- The development of guidelines and procedures should include both verification and validation phases. Verification refers to the process used to ensure that the instructions are appropriate (i.e., technically accurate for their prescribed plant conditions). This is typically done using simulation tools, including the full-scale plant training simulator. Validation refers to the process used to ensure that the instructions can be executed properly (i.e., clearly written and understood) by the appropriate plant staff. This is typically done by presenting a simulated event to the end-user and observing that user’s activities.

Operational states

Operational States shown in Fig. 1 are those plant states wherein continued operation of the plant is permitted based on meeting all regulatory requirements. In the case of anticipated operational occurrences, the plant technical specifications may limit the time that the plant can remain in operation if the plant cannot be returned to normal operating conditions.

Normal operation

Procedures and guidance for the first “Operational State” are for “normal” operation of the plant and typically have two purposes. First and foremost, procedures and guidance ensure that plant operation meets all regulatory requirements for safe operation. Those regulatory requirements for safe operation are contained in the plant “Technical Specifications.” Secondly, procedures and guidance ensure that the plant and all its components are operated in an efficient manner to maximize plant output. Procedures (as opposed to guidance) are typically provided that require the user to execute each step as written.

There are many procedures for normal operation, including:

- System Operating Procedures that provide instructions for operation of each plant system,
- Start-up Procedures that provide instructions for start-up of the plant to full power operation,
- Shutdown Procedures that provide instructions for shutdown of the plant from full power operation to other operating modes such as Hot Standby, Hot Shutdown, Cold Shutdown and Refueling. The operating modes for light water reactors in the United States are defined in their plant Technical Specifications as shown in (NRC, 2012a,b,c,d,e),
- General Operating Procedures that provide instructions for safe operation of the plant when at full or partial power,
- Maintenance Procedures that provide instructions for routine maintenance of plant components as well as repairs for faulted components,
- Instrument Calibration Procedures that provide instructions for ensuring that instrumentation is accurate and within its design basis, and
- Surveillance Procedures that provide instructions for inspection of plant components on a regular basis.

These procedures are used by various staff members as prescribed in the plant administrative procedures (see later for qualifications of staff members including training). For example, the plant Operating Procedures, Start-up Procedures and Shutdown Procedures are typically only used by licensed plant operators. System Operating Procedures may be used by Auxiliary Operators. The remainder of the above listed procedures would be used by specialists in specific disciplines.

Anticipated operational occurrences

Procedures and guidance for the second “Operational State” address “Anticipated Operational Occurrences” and provide instruction for responding to system anomalies for which regulatory requirements allow the plant to remain operational. Again, these are typically developed as procedures wherein the user must execute each step without deviation. The use of these procedures is to return the plant to a “normal” operating state and prevent the plant from entering a state that exceeds regulatory requirements.

There are many procedures in this plant state, including:

- Alarm Response Procedures that provide instructions for responding to plant parameters that are beyond the expected range of values as indicated on the plant control room alarm board,
- Abnormal Operating Procedures that provide instructions for responding to plant events that do not cause the plant to exceed its design basis. These are sometimes called Off-Normal Operating Procedures. Examples of these procedures range from response to the loss of a redundant train of a plant system, response to fires, or response to excessive temperatures in a system of component.

Design basis accidents

The design basis accident state consists of procedures for events that initiate a reactor trip signal and/or a safety injection signal (to start emergency core cooling water flow). These procedures, generally called Emergency Operating Procedures (EOPs), have the singular purpose of returning the plant to a safety stable state that may either be a shutdown condition or a return to power operation. These are typically procedures (as opposed to guidance) wherein the user must execute the procedure steps as written.

Prior to the accident at TMI-2, the EOPs were a series of procedures for each design basis accident that is analysed in the plant Safety Analysis Report. These are referred to as event-based procedures because the user was required to diagnose the event and then start in the correct procedure. As discussed in [Skillman and Rempe \(2021\)](#), an important lesson learned from the TMI-2 accident ([NRC, 1980, 1982](#)) is that EOPs need to be made symptom-based rather than event-based so that the user can respond to indications on the main control board (instrumentation as well as indication of valve positions).

To provide a cost-effective means to address the guidance in [NRC \(1980\)](#), each of the four reactor vendors in the United States (Westinghouse, Combustion Engineering, Babcock and Wilcox, and General Electric) formed an “Owners Group” to develop symptom-based EOPs for staff to operate their reactors. For the most part, the reactor vendor provided technical support to the Owners Group while plant staff provided input from a plant operation perspective. Owing to differences in plant design within each reactor vendor’s Owners Group, these new emergency procedures were developed as generic guidelines that would be applicable to all plants within the Owners Group but did not include specifics of any plant design that were not common to all designs (e.g., system components, instrumentation, etc.). These generic guidelines included those symptoms that should cause a user response as well as the appropriate response. The individual plants would then be expected to provide plant specific information related to symptoms and responses (e.g., instrument readings that should cause a response and the equipment that should be manipulated as a result). International plant owners for each reactor vendor were also invited to become members of the appropriate Owners Groups.

In addition to being symptom-based, the EOPs covered all design basis accidents and Special Events, as required by the NRC. These symptom-based procedures also included instruction on recovery steps if one or more pieces of design basis equipment was unavailable (e.g., out-of-service at the time of the event or fail to start or run). For the first time, the new procedures also incorporated results from generic probabilistic risk assessment (PRA) studies such as WASH-1400 ([NRC, 1976](#)).

Because the generic EOPs were developed by each Owners Group with little or no interaction with each other, the format of the resulting generic procedures was significantly different. However, the content of the generic procedures was quite similar because all of the generic procedures were reviewed and approved by the U.S. NRC. Note although the three Owners Groups for the American PWRs were combined, vendor specific EOPs remain for each of the three PWR reactor vendors. This is primarily due to the enormous costs for retraining plant operating staff for entirely new procedure sets.

Scope of generic emergency operating procedures

The EOPs were designed for use by the control room operators, including shift supervisors and station technical advisors (STA), which were recommended in the United States by the NRC ([NRC, 1980](#)). Features of the EOPs for all plants in the United States and western Europe include:

- Entered upon reactor trip or safety injection signal
- Entry procedure has two purposes: ensure appropriate safety equipment has automatically actuated and diagnosis of accident ‘type’ based on symptoms as read from the main control board
- Procedures pass from one to the next – PWR operators rely on one procedure at a time; BWR operators can be using several EOP flow charts simultaneously based upon the reactor pressure vessel (RPV) and containment control parameter indications.
- Procedures include checking symptoms related to Critical Safety Functions (reactivity, core cooling, heat sink, inventory, pressure and containment) and response procedures when one or more symptoms are out of their acceptable range.

The content of the EOPs for each reactor vendor’s Owners Group are somewhat different in terms of breadth (range of events specifically covered) and depth (degree of reliance on knowledge, skills and abilities) of the procedures. The development of the initial

EOPs by the Owners Groups, as well as subsequent enhancements, was controlled by a group of experts from plant operators and consultants with experience in operations and accident analysis. As a result, each of the four Owners Groups addressed issues differently (this separation continued after the consolidation of the PWR Owners Groups).

The scope of the EOPs, in terms of the accident sequences addressed, is much enhanced from the rigorous definition of design basis accidents and special events. Design basis accidents typically include: the definition of an initiating event; a worst-case single failure (NRC, 1979); and an event mitigation required to achieve a safe stable state. For all credible initiating events, the EOPs first provide guidance for the operating staff to verify that automatic actions have successfully initiated, and safety systems are performing their functions. In the event that one or more safety systems or components fail to initiate or perform their required functions, the EOPs provide guidance to restore the “failed” function. Two important points are worth noting:

- The EOPs verify that all redundant safety systems and components are initiated and operating as intended even though only a single subsystem or component is required for successful mitigation.
- For Westinghouse PWRs, the scope of the EOPs is based on probabilistic safety analyses (see Fleming, 2021) to determine credible failures that should be addressed. A frequency of about once in 1 million years (i.e., 1E-06 per year) was typically used for EOP development. Event sequences (i.e., initiating event combined with equipment failures) whose predicted frequency of occurrence are less than 1E-06 per year are typically excluded from EOPs in order to focus operator training on the more likely events. Thus, the EOPs address a spectrum of conditions including those more severe, as well as those less severe, than were considered in developing the plant design basis, including multiple equipment failures and operator errors.
- BWRs have attempted to identify any mechanistically possible plant conditions for which generic operational guidance can be provided and address them in the generic emergency procedures to minimize the impact on public health and safety, irrespective of the probability of occurrence. Similar to the Westinghouse PWRs, the EOPs address a spectrum of conditions including those more severe as well as those less severe than were considered in developing the plant design basis, including multiple equipment failures and operator errors.
- The frequency of an accident initiated by a catastrophic failure of the reactor vessel is typically estimated to be once in 1 million to once in 10 million years (NRC, 1976) and is therefore not specifically addressed in the EOPs because there are no possible recovery actions; this event was assumed to lead directly to core damage and a severe accident.

A detailed description of the EOP structure for the Westinghouse PWRs, General Electric BWRs, Siemens PWRs (Germany), Electricite de France PWRs, CANDU (Canadian Pressurized Heavy Water Reactors) and Combustion Engineering PWRs is provided in Appendix II of IAEA Safety Report Series No. 48 (IAEA, 2006).

Format of generic emergency operating procedures

For the same reasons that the content of the generic EOPs developed by each Owners Group is somewhat different, the format of the generic EOPs for each Owners Group is quite different.

All of the Owners Groups for the PWRs developed stepwise generic procedures wherein the user would execute a specific step and then move to the next step in the procedures.

- The Westinghouse Owners Group developed their generic EOPs in a two-column format where the left-hand column contains a setpoint for a symptom and the instruction(s) if the setpoint is satisfied; this is called the expected response. The right-hand column contains instruction(s) if the setpoint is not satisfied; this is called Response Not Obtained. For example, the left-hand column may state “At least one Safety Injection pump running” with an instruction to go to the next step if that is satisfied; if it is not satisfied, the right-hand column may state “Start at least one Safety Injection Pump”.
- Other PWR Owners Groups developed their generic EOPs in a one-column format where the instructions are provided for both cases (satisfies the setpoint for the symptom and does not satisfy the setpoint).

The BWR Owner’s Group developed their generic EOPs as a guideline addressing the different radiation barriers: RPV control, Primary Containment Control, Secondary Containment Control and Offgas and Radiation Release Control. Each guideline addresses one of the barriers for release of radiation. Within each guideline, monitor and control of important parameters, such as RPV water level and reactor pressure, is accomplished by implementing a series of actions to return the barriers to their Technical Specification conditions. Within each guideline, provision is made for additional actions if the previous action was not effective or was not available. Originally BWR utilities tried to implement these guidelines as text-based EOPs. However, plant owners began to develop flowcharts to implement the actions in place of text-based guidelines.

As a result, the BWR Owners Group generic EOPs now consist of a series of flow charts based on controlling parameters in the RPV, the primary containment, the secondary containment and offsite radioactive release and restoring those parameters to preferred operating band. Within each flow chart are decision boxes depending on whether a setpoint value was satisfied and then high-level steps that can be taken. The flow charts are backed-up by more detailed procedures for responding to a critical safety function that is beyond its setpoint. The BWROG generic EOPs are referred to as “strategic,” which work to restore the plant to the preferred operating band. In turn, they call upon text-based support procedures for tactical actions, such as starting a pump or using a means to control RPV pressure.

Procedure implementation

Implementation of the generic Owners Group procedures at a nuclear generating station is a multi-step process that includes development of plant specific procedures; verification and validation; training; and qualification. Implementation also includes site specific interface issues such as procedure usage; command and control; interface with the Technical Support Center (TSC); and emergency planning. Each of these is discussed below.

- Plant Specific EOPs – The first step in implementing the generic Owners Group EOPs (or any significant revision to them) is converting the generic procedures to plant specific procedures. The generic EOPs were developed based on a reference plant design that includes most of the applicable features (e.g., equipment, layout, etc.) of the plants in each Owners Group. Therefore, each plant may have features that deviate from the reference plant for which plant specific instructions need to be developed.

There are several major steps involved in the conversion including: development of plant specific setpoint values (control room indications that might require response actions); replacement of generic equipment names with plant specific equivalent equipment; and addition or deletion of instructions based on plant specific equipment. There are also format issues to ensure that the plant specific EOPs fit into the overall procedure schemes, such as numbering systems and denoting “continuous action” steps.

- Verification and Validation (V&V) – The generic EOPs are verified and validated by the Owners Group before they are distributed to member plant owners. This provides a level of assurance that the EOPs are technically accurate and procedures are clear and easy to use. As a result, the plant operator only needs to verify that the plant specific EOPs are a technically correct implementation of the generic material or prepare supporting justification for deviations. Each plant operator then validates the plant specific EOPs with one or more of their operating crews, typically using the plant full-scope simulator, by exercising each procedure pathway that has changed.
- Procedure Usage – The final phase of EOP development and implementation deals with usage. This refers to specific roles and responsibilities of the persons in the plant control room as well as interfaces with other plant emergency personnel. It also includes communication amongst three licensed operators within the control room while using the procedures (i.e., 3-way communication). Typically, one licensed control room operator would be responsible for reading the EOPs and making callouts for various symptoms and equipment status. A second licensed operator would be at the control board(s) reading instrumentation and equipment status information displayed on the control board; this task could be with another licensed operator when staffing permits. A third licensed operator would be responsible for monitoring the event progression and checking critical safety functions to assure that the appropriate procedures and actions are being taken. All communication in the control room in the United States uses three-way communication (INPO, 2006) wherein the receiver of a command or request will repeat back the command or request and then the originator will verbally confirm that the receiver correctly understands the command or request. While this can slow progress through multi-step procedures, it has been shown to be effective in preventing errors caused by various communication issues.

There are a few exceptions to the three-way communication requirement for certain fast-moving events such as anticipated transient without scram (ATWS). For example, once the failure of the control rods to insert is confirmed when a scram signal is present (i.e., an ATWS), the plant operators can take certain actions that do not involve any decision making. When they finish, the actions are reported to the plant supervisor. At this point, the three-way communication is resumed.

- Training – Training for use of the plant specific EOPs should follow the appropriate guidance (IAEA, 1996; INPO, 1993). This training process is called Systems (or Systematic Approach to Training (SAT). SAT is organized into distinct phases of Analysis, Design, Development, Implementation, and Evaluation, and relies on Feedback as a process for continuous improvement. Analysis refers to analysing training to determine the knowledge, skills and abilities that each job requires. Design refers to strategies for training course design including key learning points trainees must grasp. Development refers to the actual development of training modules (self-study, classroom, etc.). Implementing training refers to putting the training into practice. Evaluation refers to the assessment of trainee’s ability to use the procedures and includes feedback for enhancement of the procedures.
- Certification (Qualification) – Following the completion of training nuclear plant operators must undergo a strenuous test of their ability to properly use the EOPs during simulated scenarios that include routine operation, abnormal occurrences, and accidents. In the United States, there are two levels of certification: reactor operator and senior reactor operator; the latter requires more extensive experience as a reactor operator.
- Command and Control – As part of the required planning for responses to emergency conditions (as specified in the plant Emergency Plan), each licensee has established a command and control structure for their Emergency Response Organization (ERO). Within this structure, there are position(s) with the assigned authority and responsibility for providing overall direction on the implementation of EOPs (as well as other procedures and guidance). Typically, the Shift Manager is the licensed person in the control room who has ultimate responsibility for the actions taken in the control room.
- Emergency Planning – Each plant has an Emergency Plan that specifies certain interfaces between the EOP user and other organizations that need to be aware of the plant conditions when the EOPs are in use. Some of these include the regulatory agency, local law enforcement and emergency medical service organizations, and any civil defense organizations. In the United States, there are Emergency Action Levels (NEI, 2012) that prescribe the interfaces based on plant conditions during the use of the EOPs. The EOP user needs to understand the requirements to provide timely notification of plant conditions to organizations outside of the nuclear station.

Procedure review and updating

The Generic EOPs are maintained by each Owners Group. Each Owners Group has a dedicated group of individuals that have operations and analysis expertise to maintain these generic EOPs to state-of-the-art standards. These groups review feedback from EOP users, new research on equipment behavior under transient events or accidents, and new regulatory requirements.

Because the EOPs are exercised regularly by plant operations staff during training drills, exercises as well as certification exercises, they are frequently used for a wide range of scenarios by a wide range of personnel. Any question that arises during these training and certification scenarios, as well as during an actual transient or accident, are referred to the Owners Group for resolution. Feedback from the plant operator experts and consultants regarding new research findings, including new regulatory requirements for new events, such as containment recirculation sump blockage (NRC, 2002), are also referred to these groups, which determine the safety importance of each feedback item and prioritize investigations leading to resolution. If the dedicated group determines that a change to the generic EOPs is recommended, the Owners Group will issue a revision to the generic EOPs (typically only the revised step(s) and not the entire EOP set).

Once the Owners Group issues a revision to the generic EOPs, it becomes incumbent upon each plant to update their plant specific EOPs in a *timely manner*. The term *timely manner* needs to be clarified because the change in plant specific EOPs is a multi-step process. Therefore, timely is interpreted based on safety significance and may vary from a few months to a year or more. The multi-step process involves drafting the change to the plant specific procedure, performing V&V (see earlier discussion of V&V) on the change, making changes to training materials and ensuring that the trainers are well versed in the change, and training the certified operators on the new procedure step(s). Only then will the new procedure step(s) be included in the EOPs in the plant control room. Typically, any EOP changes are introduced into the plant licensed operator training cycle which may extend over a five- or six-week period.

Summary

Procedures and guidance are key to the safe and efficient operation of nuclear power stations. Every activity that is carried out in a nuclear power station is controlled by procedures and guidance that are expected to be followed. This ranges from plant operation and maintenance activities to accident emergency response activities. These procedures and guidance assure a consistent approach for each activity, allowing it to be carried out safely and avoiding any errors.

The approach to procedures and guidance has evolved over the six decades of nuclear power operation as a result of a continued focus on excellence in nuclear power operation. This resulted in the implementation of human factor engineering in the development of procedures and guidance as well as a systematic approach to training plant personnel on these procedures. Procedures and guidance have been developed for a wide range of possible events to ensure that the plant staff has a resource to respond to any event that may occur.

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Beyond Design Basis Event Considerations

Robert Lutz^a and Bill Williamson^b, ^a Lutz Nuclear Consulting, Hendersonville, NC, United States; and ^b TVA, Nuclear Plant Road, Athens, AL, United States

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Glossary

BDBA Beyond Design Basis Accident is an event that is outside of the plant design basis and is similar in meaning to design extension conditions.

BWR Boiling water reactor.

Design extension conditions A set of events that are beyond the design basis of the plant for which mitigative actions should be taken to ensure safety.

Emergency operating procedure A set of instructions that are to be implemented following an unintentional reactor trip signal or a signal for injection of emergency core cooling water.

Guidance Any set of instructions that are recommended to be implemented under prescribed plant conditions.

Owners group A group of plant owners with similar plant designs (usually by reactor vendor or reactor type) that fund development programs common to their plant designs.

Procedures A set of instructions that are *required* to be implemented verbatim under prescribed plant conditions.

PWR Pressurized water reactor.

Severe accident guidance (SAG) A set of recommended actions that are to be implemented if significant damage to the reactor fuel is diagnosed.

Severe accident guidelines (SAGs) Individual recommended actions for a specific goal; collectively, the guidelines form the guidance.

Introduction

One of the first efforts to assess the potential impact of an accident beyond the plant design basis was reported in the WASH-740 report, sometimes referred to as the Brookhaven Report (AEC, 1957) in early 1957. In that report, the radiological consequences of an accident more severe than the design basis loss of coolant accident, called the maximum hypothetical accident, were evaluated. The report evaluated three cases in which the worst case assumed no mitigation by a containment building. The report emphasized that distance between a nuclear facility and the nearest population could mitigate the hazards associated with a major radiological release.

Later in 1957, an accident in the United Kingdom at the Windscale facility (Arnold, 2007) led to a large radiation release. Even though the Windscale facility, which was used for production of fissile material for military applications, differed significantly from commercial light water reactors, this accident pointed out that large radiation releases from nuclear facilities were possible.

As discussed in Bley and Malsch (2021a,b) a technical basis, referred to as TID-14844 (AEC, 1962), was developed in 1962 as guidance for demonstrating compliance with the siting criteria found in 10 CFR 100 (US Government Publishing Office, 2020a). The TID-1844 guidance was based on very simple and very conservative calculations assuming a large release associated with an unlikely maximum credible accident but giving some credit for reductions associated with engineered safety features, including the containment. Using this guidance, as reactor operating power level for proposed new plants increased or the proximity to population centers decreased, the need for additional safeguards to reduce the potential radiation releases from the maximum hypothetical accident increased.

After the Three Mile Island Unit 2 accident (Skillman and Rempe, 2021), significant research was undertaken to improve the understanding of the chemical and physical processes that can occur during an accident in which the core is severely damaged. This new knowledge permitted the development of guidance for the nuclear plant operator actions to mitigate the consequences of such events. Technology advancements also enabled the development of new equipment that could also be used to mitigate the consequences of these accident that needed to be considered in the new guidance. This article provides details of guidance for beyond design basis accidents.

Plant conditions

The IAEA Safety Glossary (IAEA, 2018) defines five plant states as shown in Fig. 1. These plant states indicate the degree of safety threat associated with that state, ranging from operation states to accident conditions. The left-most three states constitute operation within the plant design basis. The right-most two states constitute design extension conditions. As noted in the Introduction, this chapter focuses on the guidance provided for the plant staff when the plant state is outside of its design basis. These conditions are called design extension conditions (DECs) in the IAEA Glossary because the range of possible events is virtually limitless, but certain events outside the design basis can be reasonably mitigated by informally extending the design basis but under a different set of regulatory requirements. Lutz and Williamson (2021) focuses on the guidance provided for operation within the plant design basis. In the United States, DECs are more commonly referred to as beyond design basis events.

There are two distinct set of procedures and guidance to provide assistance to the plant staff in responding to these DECs (also known as BDBAs). The first set of procedures and guidance is applicable to accidents that are progressing toward significant core damage and are beyond the scope of traditional emergency operating procedures, such as an extreme external event. For these types of events, additional guidance beyond the emergency operating procedures is needed. The goal of this BDBA guidance is to arrest the accident progression before significant core damage occurs. The Extensive Damage Mitigation Guidelines (EDMG) and the FLEX guidance discussed later are examples of this type of guidance. The second set of procedures and guidance is applicable to accidents that have already progressed to significant core fuel damage. The goal of this severe accident guidance is to prevent or minimize challenges to the remaining fission product barriers. The Severe Accident Guidelines are an example of this type of guidance.

DECs are postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with a best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. In the United States, these conditions are frequently called Beyond Design Basis Accidents (BDBA). A discussion of design basis versus licensing basis in the United States is provided in Lutz and Williamson (2021); in this chapter, the term design basis is used exclusively although it may be the licensing basis for some plants.

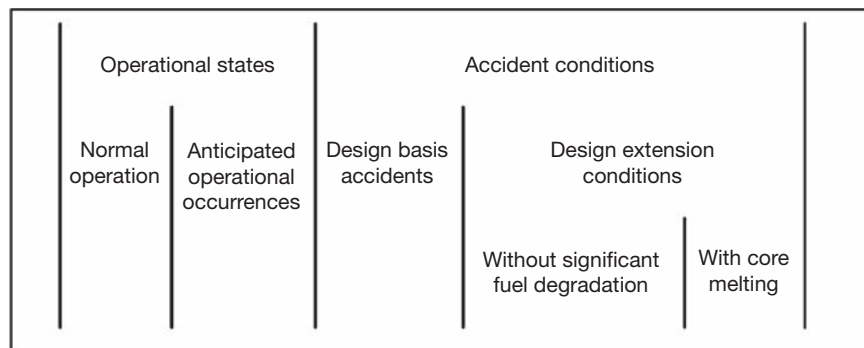


Fig. 1 Five plant states identified by IAEA (2018). IAEA (2016) *IAEA Safety Glossary: Terminology Used In Nuclear Safety and Radiation Protection, 2016 Revision*. International Atomic Energy Agency.

History of BDBA and DEC

The process for creating new regulatory requirements for mitigating accidents not considered in the original design basis of the plant differs significantly between the United States and regulatory bodies in Europe and elsewhere (e.g., China, Japan, Russia, etc.). However, the end result can be remarkably similar in many cases. While the European regulators are attentive to the regulators in the United States, in some cases, they have developed approaches that are different from the regulators in the United States as discussed in the following.

In the United States, the Nuclear Regulatory Commission (NRC) can require safety enhancements for plants through the use of an Order if the majority of the Commissioners can justify a finding that there is not adequate protection of the health and safety of the public (Bley and Malsch, 2021a,b). As the word implies, an Order is a requirement handed down by the Commissioners with instructions for the NRC staff to implement. If that finding cannot be made, then the Commissioners can direct the NRC staff to initiate rulemaking to require safety enhancements. In the latter case, any safety enhancements would need to be shown to be cost beneficial using established processes; this is known as the Backfit Rule (US Government Publishing Office, 2020b). This is one instance in which the regulatory process in the United States and that in western Europe differ significantly as discussed below. In the United States, such safety enhancements have come to be known as beyond design basis accident (BDBA) requirements.

In western Europe, typically, the regulatory body and the plant owner/operator work more collegially in determining enhancements to plant safety. The European regulators are not required to consider cost versus benefit, only the safety benefit itself. In Europe, these same safety enhancements have become known as DEC requirements.

There are three instances of importance to BDBA and DEC enhancements:

- After the accident at Three Mile Island Accident Unit 2 in 1979 to correct numerous weaknesses exposed by the accident;
- After the terrorist attacks on the New York City World Trade Center and the Pentagon in 2001 to add additional protections for prevention and mitigation of a terrorist attack on a commercial nuclear power plant; and
- After the accident at Fukushima in 2011 to add additional protections against extreme external events that are beyond the plant design basis.

In addition, after the wave of terrorist attacks in Europe in the 1970s and 1980s, the regulators in Belgium and Switzerland worked with plant operators to develop “bunkered” or “hardened” capabilities to maintain core and containment cooling in the event of a terror attack by airplane or explosives. Each of these is discussed below.

Three Mile Island Unit 2 accident

After the Three Mile Island Unit 2 accident (see Skillman and Rempe, 2021), the US Nuclear Regulatory Commission required a number of changes in the “design basis” of each plant in the United States (NRC, 1980, 1982). Although the design basis was enhanced through these requirements, there was no requirement for beyond design basis enhancements (e.g., severe accident mitigation) at that time. The design basis requirement for maintaining adequate cooling of the reactor core and the containment remained the central focus. The only requirement that could be considered beyond design basis was to provide procedures and training for mitigating core damage accidents. The strategies developed by the Owners Groups to satisfy this requirement were based on the boundary condition that the core remained in the reactor vessel and in a coolable geometry using design basis systems and equipment.

It was only during the 1980s that beyond design basis accidents began to be studied by regulatory bodies and industry worldwide, but especially in the United States and western Europe. The studies included both prevention and mitigation aspects of beyond design basis accidents which were also called severe accidents.

Severe accident prevention studies

The study of prevention of severe accidents focused on identifying the vulnerabilities of commercial nuclear power plants using Probabilistic Risk Assessments (PRA), also frequently called Probabilistic Safety Analyses (PSA) in Europe (see Modarres and Kim, 2021). PRA studies used the same fault tree and event tree methods pioneered in the commercial aviation industry to identify all of the scenarios in which the reactor core could be damaged. The first such study using this methodology was actually completed before the accident at Three Mile Island Unit 2 (NRC, 1975), but it used a generic plant design so that it did not truly represent any individual plant. The first detailed plant PRAs were completed for the Zion and Indian Point plants in the USA in response to an Order by the US NRC to show why these two plants, whose surrounding population density was greater than any other plants in the United States, should be allowed to continue to operate. Following completion of these studies and a finding that the plants should be allowed to continue operation, similar studies were required by regulators in the United States (NRC, 1988) and western Europe. Plant operators made safety enhancements to their plants when significant vulnerabilities were identified.

Severe accident mitigation studies

Significant resources were expended by both the regulators and the industry to understand and construct simulation models of the physical and chemical processes that could occur during an accident where the reactor core was severely overheated and melting. These research programs were sometimes carried out quite independently in the United States (see [IDOR, 1984](#); [NRC, 1983](#)) and in Europe (see [Micaelli et al., 2005](#)) although there was significant knowledge sharing between the United States and European programs. The United States and European sponsored research has been carried on continuously since the 1980s; the references provided are only a snapshot in time of the programs. This research was used to develop and refine computer-based severe accident simulation codes, such as the Modular Accident Analysis Program (MAAP)¹ ([EPRI, 1991](#)), Methods for Estimation of Leakages and Consequences of Releases (MELCOR) ([NRC, 1986](#)) and more recently Accident Source Term Evaluation Code (ASTEC) ([Chatelard et al., 2014](#)). Although the MAAP and MELCOR simulation codes were developed in the United States, they were also widely used in Europe and Asia. Organizations updated their analytical tools as usage expanded beyond their original focus and as new research information becomes available.

In the 1980s, as severe accident simulation progressed into modeling the behavior of a degraded reactor core, it was judged to be sufficiently robust to begin developing emergency response guidance. At this time, there were large uncertainties in some phenomena (chemical and physical processes). The first attempt to develop severe accident guidance (SAG), or severe accident management guidance (SAMG) as it is sometimes called, was carried out in the late 1980s in coordination with development of a technical basis ([EPRI, 1992](#)) that could be used by each of the Owners Groups in the United States. Each of the Owners Groups then developed generic severe accident guidance (see [Lutz and Williamson \(2021\)](#) for a discussion of Owners Group activities in Emergency Procedures). While the format of the SAG for each Owners Group differed, the overall strategies were identical. The format differences were primarily due to the oversight provided by each Owners Group. The initial version of these was made available to plant operators in the 1994–96 time frame. Similar to the generic EOPs that were discussed in [Lutz and Williamson \(2021\)](#), these were also made available to all international members of each Owners Group. More detail on the Owners Group SAG is provided below.

The IAEA has recently updated guidance for development and implementation of severe accident programs ([IAEA, 2019](#)) which is based on their original 2003 guidance ([IAEA, 2003](#)). The Annex in [IAEA \(2019\)](#) provides additional detail on severe accident management guidance in France, Germany, Japan, and the United States. The severe accident guidance for CANDU Pressurized Heavy Water Reactor (PHWR) plants is similar to the PWR approach in the United States.

In the United States, each plant operator committed to implementing the Owners Group generic SAG by the end of 1998 as a condition to closing the regulatory severe accident issue ([NEI, 1991](#)). In western Europe, various agreements were reached between the plant operators and their regulatory authority concerning implementation of SAG for their plants. It is noteworthy that, with the exception of France and Germany, the European plant operators chose to adopt the Owners Group SAG methodology, including the Russian VVER PWR plant owners in eastern Europe ([Ludovit, 1999](#); [EC, 2014](#)).

Containment hydrogen control

As discussed in [Skillman and Rempe \(2021\)](#), the accident at Three Mile Island Unit 2 revealed that hydrogen generated from zirconium-water reactions in an overheated core could accumulate in the containment in sufficient quantities to become flammable. Although the hydrogen combustion event at TMI-2 did not challenge its containment, there was concern that the resulting pressure spike from hydrogen combustion events could challenge the integrity of containments during other severe accident scenarios, particularly scenarios occurring in other containments with lower design pressures. As a result, the US NRC published, 10 CFR 50.44 “Interim Requirements Related to Hydrogen Control” in 1981 ([US Government Publishing Office, 2020c](#)). This regulation had different effects on different reactor/containment designs. The major impact of this regulation on Mark I BWR Containments was to require containment (drywell) inerting (with nitrogen) and to eliminate containment purging as a post-accident combustible gas control scheme. Subsequent studies led to requirements for inerting of Mark II BWR containments and a requirement for post-accident hydrogen control systems for Mark III BWR containment as well as PWR ice condenser containment plants. Both the BWR Mark III plants and the PWR ice condenser plants chose to use hydrogen igniters connected to an a.c. power emergency bus.

Subsequent studies using PRA methods identified the risk significance of station blackout events without a.c. power recovery prior to core damage. In these scenarios, the hydrogen igniters were not operable due to the loss of a.c. power emergency buses which led to failure of the primary containment due to hydrogen burns. This led to a requirement to provide an alternate highly reliable a.c. source to power at least one train of the igniters for all PWR ice condenser containment and BWR Mark III plants ([NRC, 2008](#)).

Other changes required by the NRC and in the 1985 update of the 10CFR 50.44 rule included the containment hydrogen (PWRs and BWRs) and oxygen (inerted BWR containments only) monitoring capability as well as hydrogen recombiners capable of controlling hydrogen generation for design basis accidents.

Hydrogen combustion in large PWR containments was also found to challenge containment integrity under certain low probability conditions. Development of passive autocatalytic hydrogen recombiners, started in the 1980s and reached a state where they were offered commercially in the mid-1990s ([AECL, 1996](#)). At the recommendation of some national regulatory bodies in western Europe, they were installed in PWR containments starting in the mid-1990s (e.g., at Tihange and Doel in Belgium per the

aforementioned reference). Note that some additional western European regulatory bodies recommended these passive autocatalytic hydrogen recombiners after the Fukushima accident in 2011 (NENE, 2013).

EPRI technical basis report (TBR)

The EPRI TBR (EPRI, 1992) was a groundbreaking report in that it was the first attempt to identify all of the potential impacts of mitigative actions by the plant emergency response staff during a severe accident. Up to this time, a large number of probabilistic safety analyses (PSAs) (see Fleming, 2021), had been performed to identify and quantify the probability of a large release of radioactivity from the plant following a core damage event. For the most part, these PSAs did not model interventions but merely modeled an unabated event progression. The reason for this was the absence of a sound technical basis for modeling possible interventions as well as the absence of methods for modeling human reliability of such actions.

The starting point for the TBR was the myriad of PSA analyses performed with computer simulation codes (primarily MAAP and MELCOR) for the various PWR and BWR plant designs. At each point in the accident progression, the best estimate results of possible operator actions on the accident scenario were identified based on severe accident experiments performed world-wide, first principles of the various physical and chemical processes, and information from other industrial processes.

The report was divided into two volumes. The first volume contained diagnostics to determine the state of the reactor coolant system (actually the reactor core) and the containment during a severe accident and major effects of various Candidate High Level Actions (CHLAs) that could be taken for each core and containment state. Because of uncertainties in severe accident phenomena, both the positive and negative impacts of implementing each CHLA were identified. Volume 2 contained the technical basis for each severe accident phenomena postulated during a severe accident as well as the effect of implementing mitigative actions (i.e., severe accident management strategies).

Three “RCS States” were identified as:

- “Oxidized” (OX)—an intact core with significant oxidation,
- “Badly Damaged” (BD)—a partially relocated core but still in-Vessel, and
- “Ex-Vessel” (EX).

Four “Containment States” were identified as:

- “Closed and Cooled” (C),
- “Challenged” (CH)—but still intact,
- “Impaired” (I) and
- “Bypassed” (B).

Fifteen CHLAs were presented in the report including:

- Inject into the RPV/RCS
- Depressurize the RPV/RCS
- Spray within the RPV (BWRs)
- Restart the RCPs (PWRs)
- Depressurize the Steam Generators (PWRs)
- Inject into (Feed) the Steam Generators (PWRs)
- Spray into Containment
- Inject into Containment
- Operate Fan Coolers
- Operate Hydrogen Recombiners
- Operate Hydrogen Igniters
- Inert Containment with Noncondensibles (BWRs)
- Vent Containment
- Spray Secondary Containment
- Flood Secondary Containment

Two special considerations were also presented: External cooling of the RPV/RCS and Steam Inerting/De-inerting of the Containment.

B&W owners group (B&WOG) and combustion engineering owners group (CEOG)

The B&WOG and CEOG used the CHLAs directly from the EPRI TBR along with the positive and negative benefits of implementing those actions directly from the EPRI TBR to develop their basic generic SAG. Special considerations for their plant design features (e.g., B&W once-through steam generators and CE heat removal capability with containment spray recirculation) were included in their respective SAG. Table 1 provides an example diagnostic matrix for these plants.

For each combination of RCS and Containment states (see Table 1), a set of applicable CHLAs is presented in the SAG. There is some prioritization of the applicable CHLAs within each combination of RCS/Containment states. For example, containment

Table 1 Example diagnostic matrix for CE or B&W plant (EPRI, 1992).

<i>Example diagnostic matrix</i>				
<i>Core state description</i>	<i>Containment state description nomenclature</i>			
	<i>Closed/cooled (CC)</i>	<i>Challenged (CH)</i>	<i>Impaired (I)</i>	<i>Bypassed (B)</i>
Oxidized (OX)	OX/CC	OX/CH	OX/I	OX/B
Badly damaged (BD)	BD/CC	BD/CH	BD/I	BD/B
Ex-vessel (EX)	EX/CC	EX/CH	EX/I	EX/B

venting would only be applicable for the “CH” (Challenged) containment state in combination with either the “BD” (Badly Damaged) or “EX” (Ex-Vessel) core states. Additional detail regarding specific SAG from these two Owners Groups is not available in the open literature.

This generic SAG was intended only for use by the plant Technical Support Center (TSC) staff. For both of these plant types, the main control room staff would continue using their plant EOPs. If any differences in guidance were encountered, they would be discussed between the main control room staff and the TSC. For example, the SAGs may provide guidance to reduce or terminate containment spray to steam inert the containment to prevent a hydrogen burn, whereas the EOPs would continue use of containment spray until the containment was reduced to near atmospheric pressure. The CEOG SAG also included some computational aids to assist the user in identifying the applicable core and containment states. The B&WOG SAG and the CEOG SAG transition from the EOPs to the SAGs based on a high core exit thermocouple indication that is indicative of inadequate core cooling.

Westinghouse Owners Group (WOG)

The Westinghouse Owners Group developed a unique symptom-based diagnostic process (Lutz et al., 1995) based on seven plant parameters: SG Level, RCS Pressure, Core Temperature, Containment Water Level, Containment Pressure, Containment Hydrogen and Fission Product Release Rate. These parameters were then arranged in order of importance from an accident progression perspective and numerical values defined for each parameter that represented a need to implement a mitigating strategy. Strategies were incorporated into step-wise guidelines corresponding to each of the seven symptom based parameters.

A unique feature of the WOG SAG was the development of a separate guideline for control room staff that recommended continuation of steps already attempted in the EOPs that were unsuccessful in arresting the accident before core damage occurred. The remainder of the guidelines were for use by the TSC staff. This approach clearly separated the design basis training for plant operators (restore core cooling) with the beyond design basis focus of maintaining fission product barriers intact (or mitigating fission product releases if a barrier is breached). Like the B&WOG and the CEOG, the WOG transitions from EOPs to the SAGs based on high core exit thermocouple indications.

Another unique feature of the WOG SAG was the symptom based Computational Aids to assist the user in cases where direct measurement might not be available. Key aids included: Required Water Flow Rates to Restore Core Cooling, Containment Flammability (based on hydrogen concentration) and, Containment Volume versus Water Level.

The WOG SAG was updated in 2001 based on issues identified during plant specific implementation as well as new research findings between 1994 and 2001.

BWR Owners Group (BWROG)

Initially, the BWR Owners Group chose to integrate their SAG into their EOPs. The BWROG used an expert team to develop the operator actions from the TBR and other documents. These actions were reviewed by the BWROG and eventually approved for the SAG.

The BWROG designed a transition from the generic EOPs to the generic SAGs based on loss of adequate core cooling. The transition was established because severe accident conditions were not as easy to determine as were conditions within EOP space. It was expected that help from the TSC would be available to determine the appropriate plant conditions for initiating use of the SAGs.

The CHLA from the TBR and insights from other documents were used to develop the SAG guideline changes. SAG Rev. Zero (0) was prepared in 1996 and following some demonstrations Rev. One (1) was used in the US to implement SAGs at the end of 1998.

Terrorist activities in Europe

In the 1970s and 1980s, a significant number of terrorist attacks took place in Europe (Walker et al., 2010). As a result, the regulatory agencies in many European countries studied methods to keep commercial nuclear power facilities safe from bombings and takeovers. Only the Belgian and the Swiss regulators recommended that plant operators in those countries develop hardened facilities where key systems for plant shutdown and the plant operating staff would be protected from such attacks. These became known as

hardened or bunkered systems in Belgium and Switzerland. The following two paragraphs describe how such systems were implemented at the Tihange plant in Belgium and the Beznau plant in Switzerland.

Besides regular primary level safety systems, the Belgian Tihange has secondary level safety systems that can autonomously keep the power plant safe during large external accidents, such as the crash of an aircraft, external explosions or loss of the water level in the reactor coolant system (FANC, 2011). The primary level systems for core cooling and steam generator heat removal have two or three redundant trains of safety. The secondary level systems are housed in a hardened structure that can withstand aircraft or bombs and contain similar systems for core cooling and heat removal. The secondary level systems also use water from underground water tables as the primary heat sink instead of the normal heat sink which is the Meuse river.

In Switzerland, each reactor unit of the Beznau station has been equipped with an emergency structure containing an independent set of emergency systems called the NÄhrüstung NOTstand, or NANO emergency retrofit system. The systems housed in the structure include additional water injection systems for reactor emergency shutdown, feeding the steam generators, a 50 kV emergency power line, and a diesel generator. They are heavily protected (bunkered) from external hazards and, if needed, are able to cool and shut down the power plant without human intervention for 72 h. With at least 1.5 m thick concrete-steel housings, these buildings protect critical systems from external agents such as earthquakes or plane crashes (Cecchi, 1993).

Chernobyl

As discussed in Sich (2021), the 1986 accident at the Chernobyl Unit 4 power station was a catastrophic event initiated by an ill-timed experiment to try to use latent energy in the main turbine during coastdown. The timing of the experiment would have been in direct contradiction to regulations governing the conduct of nuclear power plants in the United States and western Europe. In response to the event, US regulators (NRC, 1987) and the European regulators emphasized the importance of operator training, maintaining the plant within the bounds of the safety analyses while performing tests and experiments at operating reactors, and promoting safety culture. The Institute for Nuclear Power Operations (INPO) developed voluntary guidance that required a reactor engineer to be involved in any tests that could affect core reactivity. Canadian regulators also made several changes to constrain positive void coefficients in their CANDU PHWR plants.

However, the catastrophic release of fission products from the accident caused European regulators to have concerns about the overall health and safety of their citizens. As a result, many regulators in western Europe recommended that plant operators install filtered vents on the containment to reduce the amount of fission product released if containment venting should become necessary. One of the first installations of filtered containment vents was at the plants in Sweden (Kärnkraftsäkerhet, 1997). As these filtered vents were added to plants, it became necessary for the SAG to be updated to reflect this change. It is known that the Westinghouse Owners Group developed an enhancement to their generic SAG for these European plants (Milhalina and Jalovec, 2013). However, little information is available in the open literature.

World trade center and pentagon attacks

On September 11, 2001, terrorists crashed two commercial aircraft in the World Trade Center in New York City, as well as the Pentagon Building in Washington, DC. The US NRC developed new regulatory requirements to enhance the safety of nuclear plants in the United States due to terrorist activities. The NRC issued an Order requiring all plants to upgrade security, provide mitigative strategies and capabilities for spent fuel pools, provide strategies for loss of command and control of the plant and, provide strategies for maintaining core and containment cooling in the event of a terrorist attack at the plant (NRC, 2002).

In response, the industry developed a set of guidance for implementation of the Order that was approved by the NRC (NEI, 2006). One of the major tenants of the guidance was the capability to maintain spent fuel pool, core, and containment cooling in the event of a terrorist attack that destroyed a quadrant of the plant (often referred to as a Loss of Large Area event) and resulted in a loss of all internal power distribution (loss of all a.c. and d.c. power distribution within the plant).

The net result of this guidance was the procurement of portable equipment for maintaining spent fuel pool, core and containment cooling at a remote location on the plants site that could be rapidly connected in the event of such an event. This portable equipment became known as the B.5.b equipment. A high-level regulatory requirement for the B.5.b equipment and associated programmatic topics (e.g., staffing, guidance, training, etc.) was developed as 10CFR50.54(hh) (U.S. Government Publishing Office, 2020d). While this regulation is specifically focused on aircraft crashes into a nuclear power plant site, the guidance and equipment would also be useful for other terrorist-initiated events that cause physical damage to the plant.

There were also procedures and guidelines developed that became known as the Extended Damage Mitigation Guidelines (EDMGs). These guidelines included instruction on the hookup and use of the portable equipment and also methods to determine the measurement of key plant parameters. The loss of internal power distribution made instrument readouts unavailable. The event could also make the control room unavailable for operating equipment and as a focal point for command and control.

This B.5.b equipment and EDMG differs from the Diverse and Flexible Coping Strategies (FLEX) safety enhancements (NEI, 2012) described in the following section in several important ways:

- Only one set of B.5.b equipment was required for the entire plant site,

- The hook-ups for the portable equipment did not require consideration of beyond design basis external events (e.g., floods, tsunami, etc.),
- It was assumed that the plant could be tripped (scrammed) several tens of minutes before systems became unavailable,
- No internal power distribution (a.c., power or d.c. power) is available for EDMGs.

After the programs were implemented at each plant, the NRC did not develop inspection guidance. The readiness of the equipment and training was left to each plant owner. Following the Fukushima accident in 2011 (see below), NRC audits of maintenance and training of the B.5.b equipment by some plant operators revealed weaknesses at some plants. As a result, in mid-2011, the NRC required a detailed description of each plant operator's B.5.b program in relation to the 10CFR50.54(hh) requirements (NRC, 2011). In 2007, the BWROG addressed issues to add the B.5.b capability to both SAG and EOPs where appropriate. For PWRs, the guidance for use of the B.5.b capability is a separate guideline from the EOPs; reference was made in the SAG to the B.5.b guidelines in the 2012 SAG upgrades after Fukushima.

Fukushima

As discussed in Mizokami and Rempe (2021), the events at the three operating units of Fukushima Daiichi revealed weaknesses in the plant coping capability to respond to a Beyond Design Basis External Event (BDBEE). The weaknesses included plant design features, accident management procedures and guidance, and offsite emergency response. As discussed in Mizokami and Rempe (2021), significant changes have been made to nuclear reactors worldwide to enhance the coping capabilities for beyond design basis accidents. The global response resulted in many common changes in coping capability but also many differences. This section describes how insights from the Fukushima accident affected SAG in selected countries (Lutz et al., 2016).

United States response

The response to the Fukushima accident in the United States started immediately after the accident and initially took two separate paths. The first path was led by the regulator, the US NRC, while the second path was led by the nuclear power industry. The regulator and the industry worked together to formulate a set of enhancements to the capability of plants to respond to a BDBEE.

The focus in the United States was clearly on enhancements to guarantee continued core, containment and spent fuel pool cooling in the event of beyond design basis accidents, particularly those resulting from extreme external events. The FLEX initiative (NEI, 2012) added another layer to the beyond design basis mitigation capability previously implemented in response to the 2001 Terrorist Attacks (NEI, 2006).

The FLEX concept involves strategies to maintain core, containment and spent fuel pool cooling for a wide range of BDBEES that result in the loss of all a.c. power (onsite and offsite) as well as access to the ultimate heat sink for an indefinite period of time. The strategies rely upon a combination of fixed, in-place and portable equipment that would be protected from the BDBEE. The FLEX equipment includes portable high- and low-pressure pumps, diesel generators, fire trucks and debris clearing equipment. In addition to the FLEX equipment, the enhancements include: procedures and guidance for using the FLEX equipment (called FLEX Support Guidelines, or FSGs), staffing and communications plant walkdowns to ensure adequate flood protection, seismic evaluations, and enhanced offsite emergency response preparedness. Unlike the B.5.b enhancements, it was required to have one set of FLEX equipment per reactor unit plus one complete spare. The FLEX hook-ups for the FLEX equipment had to consider beyond design basis flood levels.

Each Owners Group developed generic FLEX Support Guidelines (FSGs) that were then made plant-specific by each plant operator. Most of the FSGs focused on diagnosing plant conditions for use of FLEX equipment and guidance on hook-up of FLEX equipment. However, one important FSG focused on alternate methods of determining key plant conditions when normal instrumentation in the control room is not available (Lutz et al., 2016). Those alternate methods include the use of mechanical measuring devices (e.g., local gauges), using portable read-out devices at specific locations and inferring parameters from other available indications.

New regulatory requirements in 10 CFR 50.155 (U.S. Government Publishing Office, 2020e) for procedure/guidance enhancements (including FSGs) focused on maintaining core, containment and spent fuel pool cooling following an extreme external event that is beyond design basis. However, no regulatory requirements for mitigation of an event once core damage occurs (e.g., severe accident guidance) were initiated. The requirement in the Backfit Rule, 10CFR50.109, for cost beneficial enhancements of proposed new regulatory requirements could not be justified for SAG. Nevertheless, the industry voluntarily undertook an update of the Owners Group generic SAG to incorporate lesson learned from the Fukushima accident. The two most notable enhancements were guidance for detecting and mitigating hydrogen accumulation in buildings adjacent to the primary containment and guidance for water addition and venting for Mark I and Mark II BWRs (NEI, 2013).

The Fukushima accident occurred in the same time frame as the consolidation of the three Owners Groups from PWR vendors into a single Owners Group called the PWR Owners Group (PWROG). The enhanced SAG developed by the PWROG includes Fukushima lessons learned and was made available to all PWROG members. A key feature of the updated PWROG SAG is the development of Technical Support Guidelines (TSGs). Key TSGs included a "Plant Capabilities" guideline that list all of the resources (including B.5.b equipment and FLEX equipment) at the plant that could be used in an emergency and an "Instrumentation"

guideline (Lutz et al., 2015) that can be used to validate instrument readings. Due to the proprietary nature of the new PWROG SAG, only limited knowledge of their contents is available in the open literature (Labarge and Lutz 2013; Prior, 2016).

The BWROG enhanced SAG was also updated to include Fukushima lessons learned. In the SAGs, the concept of Severe Accident Water Addition (SAWA) and Severe Accident Water Management (SAWM) were integrated into the SAG. SAWA guidance focuses on ensuring that any ex-vessel core debris is stabilized and SAWM guidance focuses on preserving the use of the suppression chamber vent during a severe accident. Another example is improvement in the generic Technical Support Guidelines, which support SAGs, for determination of control parameters, and of plant conditions, and SAG support actions. These improvements include some generic best estimate calculations to help validate (or refute) parameter indications and determine the best estimate of the plant condition. Like the PWROG SAG, little information is available in the open literature because the BWROG SAG is proprietary.

European response

The regulatory situation in Europe differs from that within the United States. Within Europe, each country has its own regulatory body. There is also a wide range of reactor designs, including those of US design origin, but also Swedish Boiling Water Reactors, German and French Pressurized Water Reactors, Russian VVERs, Canadian PHWRs, and British Gas-cooled Reactors. Prior to the Fukushima accident, significant efforts had been made to harmonize regulatory requirements, led by international organizations such as the International Atomic Energy Agency (IAEA), the Western European Nuclear Regulators Association (WENRA) and the European Nuclear Safety Regulators Group (ENSREG).

Following the Fukushima accident, both WENRA and ENSREG acted quickly to specify a “stress test” to be performed by all European Union (EU) member state plants (Switzerland and Ukraine, who were not EU states, also participated). Each national regulator required their licensees to perform the assessments (sometimes called “complementary safety assessments”) using WENRA and ENSREG guidance (ENSREG, 2011). These assessments covered three areas:

- Initiating events (earthquake, flooding and other extreme natural conditions);
- Consequential loss of safety functions (prolonged total loss of electrical power supply, prolonged total loss of the primary ultimate heat sink, combination of the two);
- Accident management issues for core melt scenarios and degraded conditions in the spent fuel pool

The results of the stress tests (European Union, 2014) were used to identify weak points and develop solutions, increasing the plant’s overall robustness in these areas. A number of safety enhancements were already being considered before the Fukushima event, but different plants and countries were at different stages of implementing needed enhancements. Many plants had also improved severe accident management capabilities by installing fixed mitigation systems such as passive hydrogen recombiners and filtered vent systems. In addition, as discussed in Mizokami and Rempe (2021), Fukushima brought new lessons: the possibility of simultaneous severe accidents in multiple units on a site, the possibility of loss of all instrumentation early in an event, and others, which now needed to be addressed. In Europe, both mobile/temporary and fixed equipment backfits were implemented. These backfits include both preventive (pre-core damage) and mitigative (severe, post-core damage) measures.

As an example, the new Reference Safety Levels issued by WENRA (WENRA, 2014) include requirements that events inside and outside the design basis must be analyzed and appropriately addressed (including the use of additional installed or temporary equipment, where appropriate). The Design Extension Conditions (DECs) should consider those events not leading to core damage (DEC-A), and, importantly, also those events where core damage occurs - severe accidents (DEC-B).

Differences between the United States and Europe

Based on the requirements recommended by a majority of regulators in Europe and in the United States, there are three important differences in the approach to BDBE/Design Extension Conditions (Lutz and Prior, 2016):

1. In Europe, the majority of regulators now require filtered containment vents and autocatalytic hydrogen recombiners capable for mitigating consequences of an accident that results in significant core damage. Neither of these is required in the United States.
2. In Europe, SAG is required by the majority of regulators and is reviewed and approved by the regulator. In the United States, SAG is a voluntary effort that plant operators have committed to implement and maintain; the level of regulatory oversight to ensure that the commitments are maintained is under development.
3. In Europe, many of the regulators are requiring that instrumentation be installed that is qualified for operation under accident conditions resulting from a severely damaged core. In the United States, plant operators use guidelines developed by the Owners Groups to validate indications from existing instrumentation.

The BWROG also demonstrated that existing plant instrumentation, when properly validated, could provide sufficient information to arrest a BDBE with core damage without need of additional instrumentation. Both the BWROG and the PWROG provided information for validation of plant instrumentation for BDBEs in Technical Support Guidance to their SAG.

Japanese response

The immediate response of the Japanese regulator was to shutdown all plants in Japan until a thorough safety review could be conducted. Investigations were initiated to upgrade plant capability to withstand extreme external events (preventive measures), including earthquakes and tsunamis (i.e., prevent an event resulting from an extreme external event). Other investigations have been carried out to develop capability to maintain core, containment and spent fuel cooling in the event of an extreme external event that overwhelms the preventive measures. Finally, mitigation equipment and guidelines have been developed to mitigate challenges to fission product boundaries in the event of a core melt event, such as portable pumps for water injection to the containment spray system, hydrogen control systems using passive autocatalytic recombiners and filtered containment vents.

Summary

Accidents beyond the plant design basis have been considered from the earliest days of commercial nuclear power. Modeling of this class of accidents was quite simplistic and, as a result, the earliest considerations focused on siting plants away from population centers and providing systems and structures to reduce the amount of radioactive materials that might be released during such an accident. After the Three Mile Island Unit 2 accident, significant research was undertaken to improve the understanding of the chemical and physical processes that can occur during an accident in which the core is severely damaged. This new knowledge permitted the development of guidance for the nuclear plant operators to take actions to mitigate the consequences of such events. Technology advancements also enabled the development of new equipment that could be used to mitigate the consequences of these accident. Capabilities to mitigate beyond design basis accidents was further enhanced as a result of terrorist attacks throughout the world and from lessons learned from the accident at Fukushima in 2011.

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Safety Evaluations

R.O. Gauntt, Gauntt Technical Safety Associates, LLC, Edgewood, NM, United States

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Glossary

Advisory Committee on Reactor Safeguards (ACRS) Independent Advisory Committee to the US Nuclear Regulatory Commission (NRC) that is statutorily mandated by the Atomic Energy Act of 1954, as amended.

Anticipated Operational Occurrences (AOOs) Operational events that constitute a perturbation in normal operations that are anticipated and considered in the design basis of the plant.

Anticipated Transient without Scram (ATWS) One of the “worst case” accidents, consideration of which frequently motivates the NRC to take regulatory action. Such an accident could happen if the scram system (which provides a highly reliable means of shutting down the reactor) fails to work during a reactor event (anticipated transient). The types of events considered are those used for designing the plant.

Atomic Energy Commission (AEC) The Federal agency created in 1946 to manage the development, use, and control of atomic (nuclear) energy for military and civilian applications. The AEC was subsequently abolished by the Energy Reorganization Act of 1974 and succeeded by the Energy Research and Development Administration (now part of the US Department of Energy) and the NRC.

Beyond Design Basis Accidents (BDBAs) Credible but unlikely accidents that are outside the design basis of the plant and fail the success criteria set out in 10CFR20, 10CFR50, and 10CFR100.

Boiling Water Reactor (BWR) A common nuclear power reactor design in which water flows upward through the core, where it is heated by fission and allowed to boil in the reactor vessel. The resulting steam then drives turbines, which activate generators to produce electrical power.

Code of Federal Regulations (CFR) Regulations promulgated by US government agencies. Commercial nuclear energy facilities are governed by 10CFR Parts 1–199 promulgated by the NRC.

Conditional Containment Failure Probability (CCFP) Containment failure probability conditional on a core damage event.

Defense-in-Depth (DiD) A design and regulatory principle that relies on several consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or engineered barrier were to fail, the subsequent level or barrier would be available. When properly implemented,

DiD ensures that no single technical, human or organizational failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability.

Design Basis Accidents (DBAs) One category of the postulated accidents used to set criteria and limits for the design and sizing of safety-related systems, structures, and components.

Emergency Core Cooling System (ECCS) Reactor system components (pumps, valves, heat exchangers, tanks, and piping) that are specifically designed to remove residual heat from the reactor fuel rods in the event of a failure of the normal core cooling system (reactor coolant system).

Engineered Safety Feature (ESF) A system, structure, or component that is relied upon during or following a DBE to ensure the capability to prevent or mitigate its consequences such that potential off-site exposures remain below values specified in applicable regulations.

Extended Loss of AC Power (ELAP) An event sequence that results in the failure of all AC power sources at the reactor facility so that only the reactor facility's battery power is available.

Final Safety Analysis Report (FSAR) Document submitted by an applicant that provides information to support the NRC safety evaluation of a proposed facility. It is based on the finalized detailed design of the facility, and it must periodically be updated throughout the life of the facility.

General Design Criteria (GDC) The set of high-level requirements for the design and safety analysis for water-cooled nuclear power plants established to ensure safe reactor operation and to minimize radiological releases.

Individual Plant Evaluation (IPE) A risk analysis (requested in NRC Generic Letter 88-22) that considers the unique aspects of a particular nuclear power plant, identifying the specific vulnerabilities to severe accidents within that plant.

Light Water Reactor (LWR) A term used to describe reactors using ordinary water as coolant, including boiling water reactors (BWRs) and pressurized water reactors (PWRs).

Loss of Coolant Accident (LOCA) Postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including Large Break LOCA (LB-LOCA), which is a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

Maximum Credible Accident (MCA) A major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

National Fire Prevention Association (NFPA) An international nonprofit organization devoted to eliminating death, injury, property and economic loss due to fire, electrical and related hazards.

Operational Basis Earthquake (OBE) Earthquake of magnitude such that plant operations can continue to operate without shutting down for damage inspections.

Pressurized Water Reactor (PWR) A common nuclear power reactor design in which very pure water is heated to a very high temperature by fission, kept under high pressure (to prevent it from boiling), and converted to steam by a steam generator (rather than by boiling, as in a BWR). The resulting steam is used to drive turbines, which activate generators to produce electrical power.

Probabilistic Risk Assessment (PRA) An approach to safety analysis that identifies what can go wrong and quantifies the likelihood and consequences of each scenario. Risk of a facility is presented in terms of the probability of various consequences, including rankings of contributors to the risk and a formal treatment of uncertainty.

Reactor Trip/Reactor Scram An automatic actuation of systems to terminate the fission process in the reactor. The term, "trip," is typically applied to PWRs and the term, "scram," to BWRs.

Safe Shutdown Earthquake (SSE) The maximum earthquake potential for which certain structures, systems, and components, important to safety, are designed to sustain and remain functional.

Safety Analysis Report (SAR) Document submitted by applicant that provides information to support the NRC safety evaluation of a proposed facility. It is based on the detailed design of the facility, and it must periodically be updated throughout the life of the facility.

Single Failure Criterion Requirement that a system, which is designed to perform a defined safety function, must be capable of meeting its objectives assuming the failure of any major component within the system or in an associated system which supports its operation.

Station Blackout (SBO) A class of anticipated operational events involving loss of onsite and offsite AC power.

US Nuclear Regulatory Commission (NRC) The body established by the Atomic Energy Act, as amended, to regulate civilian applications of nuclear power in the United States (formerly, the Atomic Energy Commission).

Introduction

The safety analysis of commercial light water nuclear power plants has been under continuous evolution and improvement since the first reactors were constructed in the 1950s. Practical and operational experience has been gained over the years from knowledge

advancements, experimental investigations and several landmark real-world accidents; this improved understanding is reflected within the rules and guidance set forth in Regulatory Guides and Safety Standards by the US Nuclear Regulatory Commission (NRC), the International Atomic Energy Agency (IAEA) and other international oversight agencies. This article provides an overview of the types of safety analyses that are required to demonstrate design safety and approval for operations with a principal objective of maintaining adequate health and safety to the population. In addition to an overview of the required safety analyses, this article identifies guidance available for performing these analyses and criteria are compared with analyses results. Additional information on this topic may be found in [Martin \(2021\)](#), [Bley and Malsch \(2021a,b\)](#), [Lutz and Williamson \(2021a\)](#), and [Rempe \(2021a\)](#). After providing some historical perspective, the article describes evaluations required for various types of events.

Historical

The Atomic Energy Act of 1946 opened the way for the application of peaceful uses of nuclear energy and for commercial applications to begin to employ the nuclear technology previously used only in defense applications during World War II, and the Atomic Energy Commission (AEC) was established to both promote and regulate this emerging civilian application as well as defense applications. At the time, there was no formalized process for establishing regulatory requirements for the design, construction, operation of nuclear power plants. As discussed by [Bley and Malsch \(2021a\)](#), this would evolve over the following decades as industrial experience was gained and a need was developed for streamlining the licensing review of a rapidly growing industry by the 1960s. Selected historical aspects of regulatory development in the United States that affected reactor safety analyses are highlighted below.

Siting criteria and 10CFR100

Initially, in recognition of the inherent danger to the public of radiation releases, the safety emphasis of the first reactors was focused on siting and proximity to population centers. A siting rule was developed for the relatively low power reactors of the time. This was called the rule of thumb calculation for the distance “R” (in miles) required to separate local population from the site where a reactor of power level “P” (in kilowatts or kW) was located; it was expressed as $R(\text{miles}) = 0.01 * \sqrt{P(\text{kW})}$ such that the dose to a person at a 2.24 mile exclusion radius around a 30 MW reactor would be less than 300 rem ([AEC, 1950](#); [NRC, 1993](#)). This recommendation was made in 1947 by the Reactor Safeguards Committee, which ultimately became the Advisory Committee on Reactor Safeguards (ACRS) (see [Bley and Malsch, 2021a](#)). In their recommendation, the committee discussed siting practices, airborne hazard control, loss of reactor cooling and prevention of energetic events (nuclear and chemical). These concepts would eventually be reflected in modern formalized safety measures and regulatory requirements for the much higher-powered reactors of today.

Defense in depth

One of these measures (beyond remote siting) was the potential use of a “containment” that would in addition to the attenuation of releases from the site offered by distance provide an additional layer of protection from uncontrolled releases of radioactive materials from a presumed accident. For this reason, the Shippingport Atomic Power station, ostensibly the first American commercial power reactor, was situated within a robust containment building. This second layer of protection, a containment building in addition to remote siting, evolved into a principle known today as “Defense in Depth” or DiD in which additional layers of physical and administrative safety measures aimed at prevention as well as mitigation are employed to make commercial nuclear power as safe as possible and radiation exposure “As Low as Reasonably Achievable” often referred to as ALARA (see 10CFR20.113 in [U.S. Government Publishing Office, 2019](#)).

Accidents with radiological consequences

Growing public awareness of the expanding commercial nuclear industry and a concern for adequate federal insurance underwriting through the 1957 Price-Anderson Act (Section 170 of the AEA, as amended) in the event of a large consequential accident, led to the commissioning of a landmark study known as the Brookhaven Report or more commonly today known as WASH-740 ([AEC, 1957](#)), that explored “plausible or credible” potential accidents and associated radiological source terms and consequences. This study examined, among other things, the impact of a loss of cooling accident (LOCA) in which the ability to remove the decay heat has been lost or compromised and found that unless cooling could be restored, melting of the reactor core and melt-through of the vessel releasing radioactive materials to the containment were possible, with eventual penetration of the concrete basement of the containment building (i.e., containment failure). This scenario was later coined “the China syndrome” owing to the notion that molten core materials could continue to move downwards toward China. This growing awareness of potential threats to safe operation of reactors eventually led to the characterization of realizable accidents with consequences, variously referred to as “maximum credible accidents (MCAs),” “limiting accidents” or “design basis accidents (DBAs),” a first step in the goal of eliminating or mitigating unlikely but potentially severe accidents. Eventually, distinction would be made for accidents that are accommodated in the design basis of the reactor and those more extreme accidents that fall outside of the design basis of the plant. Today, the success criteria for addressing a DBA-LOCA, or other such challenges to fuel integrity, are spelled out in 10CFR50.46 ([U.S. Government Publishing Office, 2019](#)), where calculated maximum temperature limits are placed on fuel cladding for such events. As described in Appendix K of 10CFR50, methods for performing such calculations are required to have either prescribed biasing conservatisms to account for uncertainties in the state of knowledge and in actual conditions in the accident (see [Rempe, 2021a](#)).

The “rule of thumb” siting calculation was later replaced by the regulatory requirements of 10CFR100 in which site boundary maximum dose limits from credible but potentially low frequency accidents are prescribed. In meeting requirements for siting, industry was permitted to consider Engineered Safety Features (ESFs) that could mitigate accidents and any resultant releases to the environment by use of containment sprays and safety systems such as the Emergency Core Cooling System (ECCS) as well as protecting containment performance during accidents by specifying maximum containment leakage under accident conditions necessary to meet site boundary requirements of 10CFR100. In this way, regulatory requirements aimed at maintaining “adequate protection” to the public from nuclear power operations were gradually aggregated to be applied to the licensing of subsequent reactor designs.

Standardized design processes and general design criteria

Until the mid-1960s the license reviews of commercial power reactors were considered on a case-by-case basis by the AEC, but the growing numbers of new applications made such a case-by-case safety review process increasingly burdensome and impractical. To alleviate this situation a set of General Design Criteria (GDC) were developed so that designers and reviewers could follow best and approved practices in preparing the safety basis for the plant. These general criteria are described in Appendix A of 10CFR50 (U.S. Government Publishing Office, 2019). These criteria are general, providing high level goals to be met and prescribed design aspects that should be addressed, such as quality control and documentation requirements for safety systems and components; protection against fires and natural phenomena (flooding and seismic events); and sharing of systems and components. These criteria are an accrual of knowledge in reactor design fostering standardization of the safety design process and its documentation in the Safety Analysis Report (SAR) and supporting efficient and consistent regulatory review of submitted license requests (Bley and Malsch, 2021a; Rempe, 2021a).

Safety analyses and the SAR

The modern safety and performance analyses conducted today reflect the lessons learned from accumulated operating experience of reactors since the 1950s. As noted previously, these lessons are reflected in the US Code of Federal Regulations (CFR) Parts 20, 50, and 100 (U.S. Government Publishing Office, 2019), NRC Regulatory Guides as well as in various IAEA guidance documents aimed at promoting best safety practices around the world. It is noteworthy that US regulatory practices strictly follow US laws and regulations and are not always completely harmonious with the equivalent IAEA-recommended standards and practices. US NRC Regulatory Guide 1.70 Rev 3 (NRC, 1978) provides an outline and format of the Safety Analysis Report (SAR), which documents the various analyses required to meet the GDC that are done by the reactor designers and plant owners/operators to demonstrate compliance with all laws and regulations. A similar document, referred to as “The Standard Review Plan” (NUREG 0800) (US NRC, 2020a), serves to guide NRC technical reviewers through the SAR review process.

In particular, SAR Chapter 15, as described in NRC Reg. Guide 1.70 (NRC, 1978) and with more detail in IAEA SSR-2/1 (IAEA, 2016), IAEA SSG2 (IAEA, 2009), and IAEA Safety Guide GSG-4.1 (IAEA, 2004) using a different SAR Chapter numbering system, is dedicated to safety analyses associated with the design basis of the plant and to address the relative likelihood of these accidents according to the following categories: (a) incidents of moderate frequency that are expected to occur several times over the lifetime of the plant (suggesting a frequency of once per year), (b) infrequent events postulated within the design basis of the plant that might or might not occur over the plant lifetime (suggesting a frequency on the order of once in ~40 years or more, and (c) limiting but credible faults that are not expected over the life of the plant but are nevertheless considered because of their potential for a release of a significant amount of radioactive material from the plant.

Conservative versus best estimate and uncertainty quantification approaches

Safety analyses over the years have evolved with respect to the state of scientific knowledge, capabilities of computational resources, sophistication, and accuracy of the underlying models. Early efforts to quantify the risk associated with a core damage accident, such as the Reactor Safety Study (Rasmussen, 1975), relied on bounding analyses and adopted conservative modeling assumptions to quantify the consequences of radiological releases from such events due to uncertainty and knowledge gaps. The use of conservative assumptions has also been reflected in evolving regulatory practices concerning conduct of safety analyses, especially those analyses associated with licensing activities. Analyses associated with the performance requirements of the ECCS during a LOCA are a prime example illustrating the refinement of calculational approaches over time. Such licensing analyses are made using what is referred to as an Evaluation Model, which generally is a computer software model of the reactor system. In this example, the Evaluation Model considers the reactor fuel, coolant pumps, and physics and phenomena required to simulate numerous processes, such as fuel decay heat, coolant flow pressure drops, heat transfer, and chemical reactions (e.g., cladding oxidation in steam). The success criteria for ECCS functions require that the fuel remains coolable following a LOCA with a temporary loss of coolant covering the reactor core. Research gained from experimental investigations led to development of success criteria that ensure fuel coolability, such as during a LOCA transient the peak fuel cladding temperature should remain below 1204 °C (2200 °F) and the local cladding oxidation should be limited to 17%. These success criteria are prescribed in 10CFR50.46. Appendix K to 10CFR50 describes the requirements of the Evaluation Model for such analyses and identifies several prescriptive modeling approaches that assure the analyses are “conservative,” meaning that analyses performed in this way have a “high likelihood” of meeting the success criteria given

uncertainties in the physics and modeling of the system response. Some examples of the prescribed conservative biases include assuming that the reactor decay heat level is 1.2 times the value specified in the ANS Decay Heat Standard (ANS, 1971), that a known high-biased cladding oxidation model be used (Baker and Just, 1962), as well as other similarly conservatively biased assumptions with respect to heat transfer and fluid flow. In this way, uncertainties which could lead to a more pessimistic result can be factored into the licensing analysis, increasing the likelihood that the success criteria are met in all cases. The impact of this approach is that operating power might ultimately be limited to values lower than optimal for energy production in normal operating conditions in order to not exceed the success criteria for the ECCS under accident conditions.

In recognition that unnecessary conservatism in regulatory success criteria can place unnecessary regulatory burden and costs to the licensee, the NRC has been moving toward more risk informed regulatory approaches that attempt to reduce the regulatory burdens to those only necessary to achieve the safety objectives. The original conservative prescriptive approach outlined in Appendix K has been augmented with an alternative “best estimate” approach in which unnecessary prescriptive conservatisms are relaxed to allow for appropriately reduced safety margins relative to the success criteria of 10CFR50.46, provided that uncertainties in the reduced margins are characterized and shown to be within the margins of the regulatory limits.

The performance requirements of a best estimate evaluation model are outlined in NRC Regulatory Guides 1.157 (NRC, 1989) and 1.203 (NRC, 2005). Fig. 1 illustrates the reduction in predictions for peak cladding temperature associated with the use of best estimate methods for calculating peak cladding temperature during a LOCA accident for a PWR. The figure compares the peak cladding temperature estimated with different modeling assumptions to the 10CFR50.46 peak cladding temperature limit. Of these analyses, the conservative modeling guidance of Appendix K yields the highest value, which in this example is below the regulatory limit of 1204 °C (2200 °F). This result follows from assuming 1.2 times the best estimate decay heat and the use of conservative (high biased) cladding oxidation kinetics prescribed in Appendix K. Using a more realistic cladding oxidation model results in a somewhat lowered peak cladding temperature and additional operating margin that could be used by a licensee in requests to reduce unnecessary financial penalties, such as limits on fuel discharge burnup, and allow more flexible plant operation, such as relaxations in equipment technical specification requirements. The use of realistic oxidation modeling and best estimate decay power produces an even lower estimate for peak cladding temperature. Also shown in the figure are the 95% uncertainty bounds on the best estimate analysis; licensees must calculate these upper 95% bound to demonstrate that there is sufficient operating power margin relative to the regulatory limit. In addition to requests for reductions in unnecessary financial penalties, the use of best estimate analyses with characterization of the uncertainties, has been successfully used (in conjunction to changes in plant equipment), to justify higher operating powers, without violating regulatory limits (see NRC, 2020b).

This regulatory approach allows for reduction of unnecessary regulatory burden associated with overly conservative computational approaches. Although this approach requires well validated models and codes as well as additional computational effort to demonstrate regulatory compliance, it can be well worth the effort in terms of cost verses benefit. While this example was specific

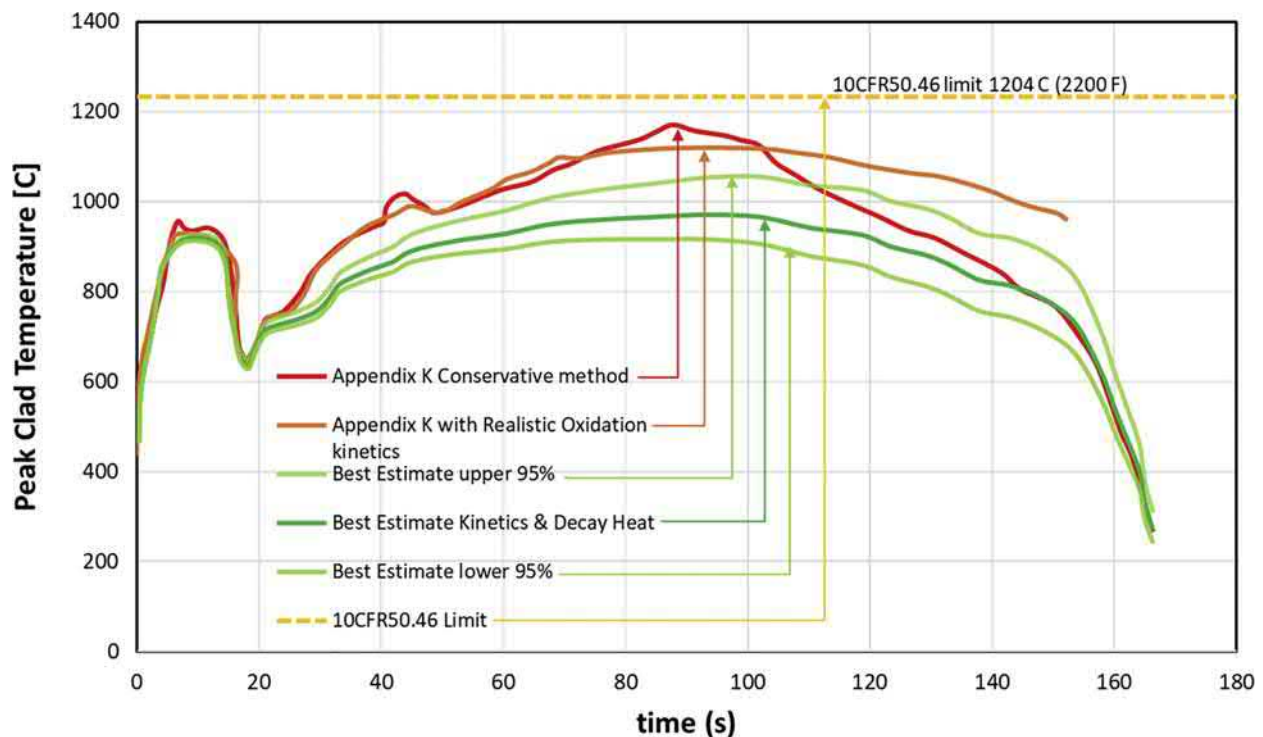


Fig. 1 Illustrative example of peak clad temperature analysis for LOCA using Appendix K conservative prescriptive methods, more realistic analysis and best estimate analysis with quantified uncertainty bands. R.O. Gauntt, Gauntt Technical Safety Associates, LLC.

to demonstrating the success criteria for ECCS performance, an informed approach using best estimate analyses with characterization of uncertainty is general and can be applied to many other licensing analyses such as evaluations of pressurized thermal shock to a reactor pressure vessel (see [Pressurized Thermal Shock](#) section) or estimates for peak containment pressure during accidents.

Analyses required to demonstrate safe operation

The remainder of this article discusses analyses performed in each of the three event categories cited above, beginning with normal operations and transient upsets to normal operations that must demonstrate safe operation within the technical specifications of the reactor fuel, core and any primary or secondary side cooling systems.

Normal operations and anticipated operational occurrences (AOOs)

During normal operation, anticipated operational occurrences (AOOs) are expected to be encountered many times over the plant lifetime. The operating nuclear power plant is comprised of a great many connected and interacting systems, which include the reactor core where heat is generated, control rods to raise or lower the reactor power, the coolant pumps that move the heat to turbines or waste heat rejection systems, safety relief valves to prevent over-pressure events as well as countless numbers of valves and piping. These systems and components interact as the system operates such that, for example, a reduction in electrical load of the power grid requires that the reactor systems respond by inserting control rods to reduce reactor power to match the reduced load on the electrical generation system and restore the overall power and flow equilibrium of the plant. Failure to maintain this operational balance could result in system parameters, such as reactor coolant system pressure, exceeding vessel safety limits leading to an automatic shutdown of the nuclear reactor and plant. Such events that upset the normal balance of the plant systems are called “transients” because they upset the normal steady equilibrium of the plant. It is required to demonstrate that the plant can be safely brought to safe shutdown or operational equilibrium through operator actions or the functioning of either active or passive safety systems. The USNRC Reg. Guide 1.70 ([NRC, 1978](#)) describes many such events and scenarios that must be addressed through analyses and documented in the SAR for the plant in the licensing approval process. The IAEA provides equivalent guidance in their Safety Standards series, such as IAEA SSG-2 ([IAEA, 2009](#)). Some of the scenarios include transient situations such as:

- Increase or decrease in heat removal from the reactor coolant system
- Increase or decrease in reactor coolant systems flow
- Core reactivity or power distribution anomalies
- Increase or decrease in reactor coolant inventory
- Radioactivity releases from plant subsystems, and
- Anticipated transients with failure to (or without) scram.

These potential system states could arise by different possible initiators, leading to the same system transient. For example, a decrease in heat removal from the reactor coolant system could arise from a decrease in coolant flow rate due to a pump failure or from a loss of heat rejection to a waste heat rejection system clogged by infestation of aquatic species such as mussels. This loss of heat removal would cause the reactor coolant system pressure to rise, potentially opening safety relief valves resulting in coolant loss. Analyses of such events make use of computer models of the reactor system that treat the thermal hydraulics of the reactor coolant system (RCS) and may be coupled with physics codes that model the nuclear response of the reactor core to changes in the plant state. The code analyses predict the transient response of the reactor and plant to the accident. These analyses are required to demonstrate that system stability and equilibrium can be restored while avoiding damage to the fuel or release of radioactivity above regulatory limits and that plant operational parameters such as power levels, coolant temperatures and system pressures remain within the design technical specifications. As mentioned earlier, these requirements are referred to as “success criteria” and can be expressed quantitatively in terms of limits on peak fuel cladding temperature or radioactivity dose limits at the site boundary. In the SAR, these safety analyses must also address the extent to which critical safety systems, engineered safety systems, or operator actions must occur in meeting the success criteria as well as a required consideration of a single failure of any given safety grade system, called the “single failure criterion” described in NRC Reg. Guide 1.53 ([NRC, 1973](#)) and SECY-77-0439 ([NRC, 1977](#)).

Discussed up to now are events and deviations from full power operation; however, over the course of a fuel cycle period of from between 1 and 1.5 years between refueling, the plant can be in a range of different operating states (full power operation, low power operation, startup, cold or hot shutdown or refueling, [NRC \(2012a,b\)](#)). The operational states present different conditions for performance analyses as the decay power may be at a very low level during cold shutdown and during refueling, the reactor pressure vessel may be open to the containment, and the containment itself may be open to the environment with various safety systems that are normally operative at full power but temporarily disabled to allow refueling. These various operational states must be considered in assessing safety of normal operations and meeting regulatory dose limit requirements.

Design basis accidents

The next class of accidents to be discussed fall into a less frequent category, one that might (or might not) occur over the course of the plant lifetime but are nevertheless accommodated in the design of the plant. These accidents are called Design Basis Accidents

(DBAs). A DBA is one that presents a significant challenge to the fuel integrity, usually following a temporary loss of cooling water level in the reactor core, that may result in limited fuel damage and limited release of radioactivity to the containment building, but one for which the plant safety systems are designed to accommodate without violating any of the regulatory requirements concerning radiological releases to the environment. The concept of the DBA came out of the early days of assessing potential risks from reactor accidents in which a significant radioactive release might occur and the accident in question was termed the MCA to note that anything more extreme might be dismissed as non-credible.

In WASH-740 (AEC, 1957), it was realized that if the reactor lost the flow or inventory of coolant such that the core could melt owing to the residual decay energy from fission products, releasing radioactivity to the containment and potentially to the environment. This led to the designation of a Loss of Coolant Accident (LOCA) as a realizable credible accident that should be considered in the design capability of the plant. The limiting example of such an accident was designated to be a double ended break of the largest primary system piping, called a Large Break LOCA (LB-LOCA). It was determined eventually that the LB-LOCA accident should be one that is accommodated within the design basis of the plant, requiring such safety components as fast acting high volume flow “accumulator systems” for pressurized water reactors (PWRs) or rapid startup high volume coolant injection pumps for boiling water reactors (BWRs) found in the turbine driven High Pressure Coolant Injection (HPCI) system to quickly inject large amounts of coolant into the reactor coolant system and mitigate uncovering of the reactor core during a large coolant pipe break accident. Based on GDC requirements described in the Appendix A of 10CFR50, the design basis of the plant should ensure that the fuel remain coolable (i.e., in essentially rod-like and non-debris geometry). As noted in **Conservative versus Best Estimate and Uncertainty** section, 10CFR50.46 (U.S. Government Publishing Office, 2019) defines success criteria for LOCA safety system response, including the need to ensure peak fuel cladding temperatures remain below a maximum value of 1204 °C (2200 °F) and that local cladding oxidation be less than 17%.

Experimental evidence from the Loss of Flow Test (LOFT) and Semiscale facilities at Idaho National Laboratory (Walker, 1992) showed that under LOCA conditions with actuation of pressurized core reflooding systems called accumulators, fuel rod geometry could be maintained, and the core could be adequately cooled. As described by Martin (2021), the 10CFR50.46 success criteria must be demonstrated analytically with qualified computer codes with some inclusion of thermal margins between calculated peak clad temperatures and the 1204 °C (2200 °F) limit to account for uncertainties. In addition to demonstrating safety margin with respect to peak clad temperatures, it was also necessary to consider the environmental loads of the containment system due to sudden containment pressurization and heating from the pipe break itself as well as the potential impact of radiation and high temperature steam released to the containment on other safety systems such as safety relief valves and electronic systems that are also key to maintaining integrity of plant systems. These are all success criteria that must be analytically demonstrated in the SAR safety analyses.

Two landmark probabilistic safety studies of nuclear power plants were established with WASH-1400 and NUREG-1150 (Rasmussen, 1975; NRC, 1990). As discussed in Fleming (2021), both WASH-1400 and the later NUREG-1150 showed that small break LOCAs could also constitute significant challenges to 10CFR50.46 cladding performance criteria and Appendix A GDC. Typically, a plant SAR will consider small and large break LOCAs in the DBA realm of analyses. It is important to note that meeting all the regulatory requirements of the DBA, such as peak clad temperature or containment performance effectiveness, does not imply that the plant is not damaged by the accident or that the reactor is not a complete financial loss—only that the regulatory requirements for fuel damage and maximum dose rate at the plant boundary are not violated.

External events and fire

The discussion up to now has focused on accidents associated with DBAs, such as large coolant piping breaks that lead to core uncovering and fuel damage. These were among the first accident initiators examined in early studies such as WASH-740 (AEC, 1957). In time, other accident classes have been identified through early PRA studies such as WASH-1400 (Rasmussen, 1975) that could arise from a series of safety system failures that ultimately produce conditions where the core cannot be cooled, arising from fires which can disable electrical systems that could prevent safe reactor shutdown, by a fault in the control rod drive system, or by disabling power to coolant pumps. Similarly, seismic events could damage reactor safety systems or containment integrity through physical damage or initiation of pipe breaks, and flooding of critical areas in the plant through internal pipe breaks from perhaps fire protection systems or auxiliary water systems. These considerations are captured in the GDC. In addition, Criterion 2 requires that designs be protected against threats from natural phenomena (storms, floods, earthquakes, etc.) and Criterion 3 requires protection against fires. Some aspects of safety analyses or simply regulatory requirements that are performed to ensure that these threats are managed within the design basis of the plant.

External events

Criterion 2 of the GDC described in Appendix A of 10CFR50 requires consideration of site-specific natural phenomena such as earthquakes, floods, and natural disasters.

Seismic challenges

Appendix S of 10CFR50 provides additional specifications relative to seismic design considerations. The seismic design basis for the plant begins with siting evaluation and consideration of the regionally specific seismic potential for the purposes of assessing

appropriateness of the site and to establish the ultimate seismic loads that the plant must be designed to accommodate. In the United States, the US Geological Survey provides geographical maps characterizing the seismic hazards across the United States as shown in Fig. 2.

Here, the most seismically active regions are indicated in red showing notably the New Madrid seismic region in the mid-western United States and the Pacific coast earthquake fault region running from Southern California extending up to Alaska. Also characterized by the hazard map are the estimated probability of exceeding a given ground acceleration that can be used to relate seismic maximum design loads to the estimated frequency of occurrence. Recall that Design Basis Events are those that may happen over the duration of the plant lifetime. The design basis for earthquakes rely on historical evidence for the largest recorded earthquakes in the region. Specific plant systems and structures in this way can be designed with sufficient robustness to survive anticipated seismic loads. In this sense, two types of design basis earthquakes are defined for regulatory purposes, the Operational Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE). Earthquakes come in all sizes, ranging from frequent earthquakes that are barely perceptible to exceedingly rare earthquakes that can cause great damage to the plant site and regional infrastructure such as roads, buildings, or dams.

The OBE is the largest earthquake that the plant can encounter without a required shutdown for damage inspection. The plant is designed to operate continuously without shutdown if the earthquake is below the OBE. On the other hand, the SSE defines the seismic loads that the plant must be designed to withstand while still maintaining safe shutdown capability and preventing any damages that result in radiological releases outside of 10CFR50 and 10CFR100 dose limits. Earthquakes larger than the SSE are outside of the plant seismic design basis and are expected to be exceedingly rare (i.e., below 10^{-5} to 10^{-6} per year). Although the SSE is larger than the OBE, it is possible to spend greater effort meeting OBE design requirements than SSE design requirements if the magnitude of the OBE is set too high relative to the SSE. Appendix S of 10CFR50 allows that the OBE can be set to a value of 1/3 or less than the SSE, and shutdown and plant examinations are not required providing the earthquake is below the OBE. If the earthquake exceeds the OBE, the plant must shutdown and perform plant examinations and assessments before restarting. Problematic is that there is considerably uncertainty associated with the frequency of occurrence of very large earthquakes, owing to their rare occurrence; and revisions to the seismic hazards characterizations have a tendency to increase in frequency estimation, potentially causing plant licensing concerns and backfitting if the margins between plant design and the latest hazards-frequency curves are too low. The seismic design of the plant includes estimating the design basis seismic loads and performing structural and vibrational analyses of key plant safety components and the secondary effects of pipe breaks and structural damage.

Flooding challenges

Similar to the seismic design basis, flooding caused by events external to the plant such as heavy rain, river flooding, and dam breaks, are defined based on maximum observed events historically or upon probabilistic models considering historical data and estimated seismic frequency distribution functions (NRC, 2011, NUREG/CR 7046), and similar GDC requirements for plant survivability of key safety systems and components are also found under 10CFR50 Appendix A, GDC Criterion 2. As with seismic hazard analyses, the analyses consider the impact of damage to plant systems in flooded areas and the impact of such damage on the

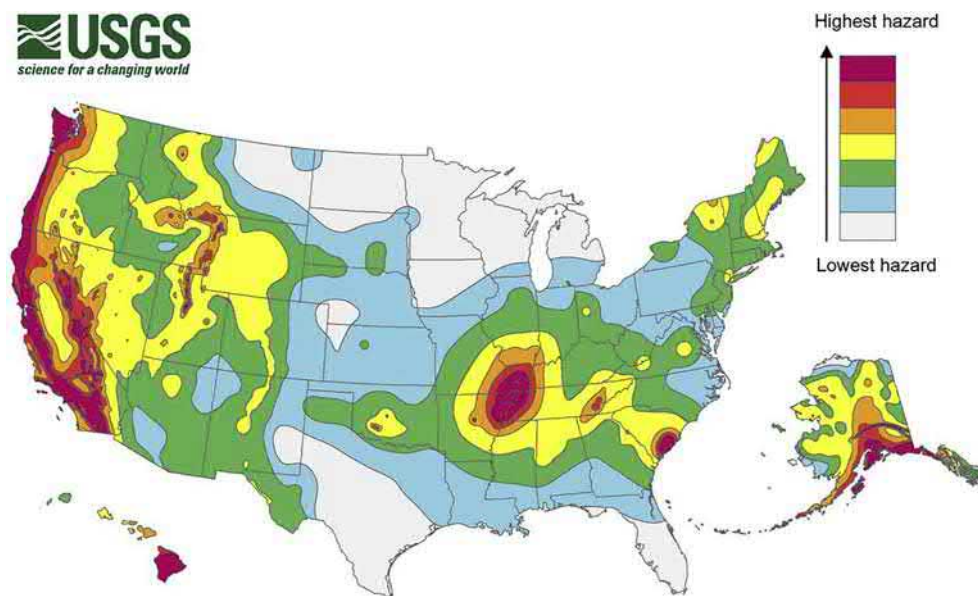


Fig. 2 2018 Long-term national seismic hazard map US Geological Survey. USGS (2018) Available at: <https://www.usgs.gov/media/images/2018-long-term-national-seismic-hazard-map>

performance of systems located outside of directly flooded areas (e.g., loss of electrical switch gear from flooding that prevents operation of safety systems located elsewhere).

In 2011, extreme flooding threats occurred at the Fukushima Daiichi nuclear power station in Japan and at the Fort Calhoun nuclear plant in Nebraska (see Fig. 3). The beyond design basis tsunami flooding accident at Fukushima Daiichi resulted in fuel melting and containment failures at the Unit 1, 2 and 3 reactors with significant offsite releases (see Mizokami and Rempe, 2021); whereas the flooding of the Missouri river at Fort Calhoun was anticipated for days prior to the event and inflatable dykes were placed around the reactor site to prevent flooding of the plant (New York Times, 2011). In 2019, the Missouri river flooded again, threatening the plant where mitigating measures were taken to avoid plant damage. As with seismic threats, contemporary estimates of flooding potential at nuclear power plants can increase with passage of time owing to climate change and rising sea levels. In the case of the tsunami flooding of the Fukushima Daiichi site, the original estimates of maximum tsunami potential were found to underestimate the observed event where the 14-m tsunami height surpassed the site protective sea wall by more than 6 m. These examples illustrate the importance of careful consideration of rare events such as tsunami flooding, or more frequent than anticipated extreme seasonal flooding events such as the Fort Calhoun incidents (Fig. 3).

Fires

Unlike rare tsunami flooding events or infrequent river flooding events, fires are common events in industry, requiring fire protection programs to anticipate and mitigate their severity and consequences. A significant fire occurred at the Browns Ferry Nuclear Power Station that threatened the safety of the plant and had profound effects on the nuclear industry with respect to fire protection (NRC, 1993; Rempe, 2021b). The regulatory oversight of fire safety prior to the Browns Ferry fire in 1979 was addressed by GDC Criterion 3, requiring that the high-level objectives: *Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in such locations as the containment and control room.*

Again, the GDC of Appendix A of 10CFR50 require that fires be addressed within a design. Criterion 3 is focused on fires and requires consideration of potential sources of fire initiation, protection measures such as managing sources of combustible materials and consideration of the damage to critical safety systems and components that might threaten safe operation of the plant and safety systems. General industry-wide fire safety practices are applied, such as minimizing combustibles in fire-sensitive regions of the plant where electrical cables or safety critical components reside and inspections. The potential for unintended damage caused by fire suppression systems is also considered in terms of water causing electrical shorts or damaging equipment and effects of smoke.

The Browns Ferry fire was a serious challenge to the aims set forth in GDC 3; a fire was initiated due an inspection of cable pass-through insulation by a procedure involving a lit candle to detect air leakage. The lit candle ignited the insulating material which resisted efforts to extinguish owing to the organic nature of the polyurethane insulating material and the high temperature of the burning insulation. Water was initially avoided due to fears of electrical shorts but was eventually used to extinguish the fire after other fire suppression measures failed but not before serious damage to some of the cables relied upon for shutdown cooling.

Following this fire, the USNRC set up additional prescriptive requirements that were codified in Appendix R of 10CFR50 in 1981. These requirements were aimed at meeting Criterion 3 objectives but also included specific prescriptive measures that plants must meet such as: fire protection programs and inspections; fire hazards analysis demonstrating that a serious fire which disables

(A)



(B)



Fig. 3 Important Flooding Events at Fort Calhoun power station and Fukushima Daiichi Nuclear Power Plant in 2011. (Right) Flooding of Missouri River at Fort Calhoun Nuclear Power Station; (Left) Fukushima tsunami flooding. From (A) New York Times (2011) *Flooding Brings Worries Over Two Nuclear Plants*. Available at: <https://www.nytimes.com/2011/06/21/us/21flood.html?searchResultPosition=1> (accessed June 20, 2011); (B) TEPCO (2011) Available at: <https://photo.tepco.co.jp/en/date/2011/201105-e/110519-02e.html>

one train of shutdown and cooling capability will leave another protected and redundant train available; specifications on fire barrier duration times; and detailed and prescriptive requirements on training, drills, and administrative procedures. So-called deterministic analysis of fires and their potential consequences could be used to, for example, show that if a fire destroys an entire room that there remains an alternative success path to achieving Criterion 3 objectives.

Appendix R requirements are still used today for many plants, although more “risk informed” practices are allowed that focus on performance-based requirements and probabilistic considerations. Currently, fire regulation is based on NFPA 805, the fire standard developed by the National Fire Protection Association. The NFPA 805 standard offers performance-based requirements which can best be demonstrated through a Probabilistic Risk Assessment (PRA) of the spectrum of fire threats. The fire PRA, similar to the severe accident PRA, is a systematic and somewhat exhaustive analysis of the detailed and interacting systems that come into play during a fire such as co-located systems, common mode failure issues, impact on unrelated safety systems, fire intensity and duration, performance of fire barriers, all considered in an event tree/fault tree analysis construct. This type of risk-informed approach can identify which systems are key to meeting GDC 3 and NFPA 805 objectives and prioritize where resources are aimed, potentially increasing safety and reducing maintenance costs. The cost of performing a fire PRA is substantial; and although it is the current state of art in fire safety and consequence assessment, not all plants embark on the NFPA 804 PRA approach. Plants originally licensed under Appendix R (or earlier) requirements are not obligated to adopt NFPA 805 standards and practices but may be required for certain license amendments. As noted in [Dube \(2021\)](#), it has been implemented in about half of the US plant sites (as of 2019).

Additional considerations

Several additional considerations, that are important elements of modern safety analyses, were added to the regulatory framework to accommodate gained operational experience. Three will be discussed here: Pressurized Thermal Shock, Station Blackout and ATWS.

Pressurized thermal shock

Licensees are now required to perform analyses demonstrating adequate protection of the reactor pressure vessel, which is a major element in the reactor’s layers of defense in depth ([NRC, 1993, 2015](#)). This requirement was imposed after a phenomenon known as “Pressurized Thermal Shock (PTS)” was observed in a non-nuclear pressure vessel due to cold water coming into sudden contact with a hot pressure vessel wall. The cold water generated thermally induced stresses that fractured the vessel wall, resulting in a catastrophic failure of the pressure vessel. In this scenario, the cooled vessel wall becomes brittle and subject to fracture under the high tensile stresses at the vessel wall inner surface. It was recognized that in nuclear reactor pressure vessels, radiation damage by neutrons on the inner pressure vessel wall can reduce the ductile-to-brittle transition temperature. Over decades, radiation damage to the vessel wall can increase the likelihood of PTS. This realization led to the so-called Pressurized Thermal Shock Rule, an added requirement from the codified 10CFR50 and 10CFR100 legislation ([NRC, 1993, 2015](#)). Computational methods, backed up with samples of irradiated pressure vessel wall materials to measure the reduced ductility, are now required to evaluate PTS. The aim of the analyses is to demonstrate adequate margins from catastrophic vessel failure arising from anticipated transient overcooling events. Evaluations include principally detailed transient thermal analysis of overcooling vessel wall events to determine the peak vessel wall stress levels and use of fracture mechanics analysis to demonstrate acceptably low frequency of catastrophic fractures. If margins to failure are low, possible remedies include decreased operating power to reduce neutron damage, vessel annealing, or even vessel replacement.

Station Blackout

Another such after-construction regulatory accommodation relates to the vulnerability of safe plant operations to the loss of offsite and onsite electrical power referred to as Station Blackout. First recognized in the WASH-1400 probabilistic risk study ([Rasmussen, 1975](#)) and later quantified in terms of frequency of occurrence in the NUREG-1150 study ([NRC, 1990](#)), it was found that roughly 50% of the core damage frequency (CDF) for many existing power plants was due to Station Blackout, a risk profile not appreciated until modern PRA methods were available to root out such vulnerabilities. Essentially, the SBO rule requires that power plants be capable of maintaining adequate core cooling and containment integrity during SBO for a plant-specific duration of time generally between 2 and 16 h, often determined by on-site DC battery capacity. This duration is referred to as the “coping time” for SBO in which appropriate mitigations can be implemented to prevent plant damage. However, if mitigations cannot be implemented within the plant-specific coping time, long term loss of reactor cooling and heat rejection can lead to core damage, and severe accident systems analysis computer codes such as MELCOR ([Humphries, 2015](#)) and MAAP ([EPRI, 2015](#)) can be used to evaluate core damage and radiological releases. Aspects of plant systems affecting the success of this strategy include ensuring high reliability of onsite diesel-powered AC power and system redundancy. This requirement received additional scrutiny following the 2011 accidents at the Fukushima Daiichi power plant, where extended loss of AC power (ELAP) persisted well beyond 8 h leading to three sequential core melting events (see [Mizokami and Rempe, 2021](#)).

Anticipated transient without SCRAM (ATWS)

Prior to 1969, the accident mode commonly referred to as ATWS was not well appreciated and not generally considered in safety analyses ([NRC, 1993](#)). While much of the topic concerning transient analyses is in the realm of design basis accidents—that is those

accidents we have anticipated and accommodated by design features in the plant, an additional system failure to scram the reactor was not originally considered. One such anticipated transient is a loss of feed water transient in which loss or decrease in feedwater to the steam generators of a PWR would call for an automatic scram of the reactor while safety systems engage to address the problem. Not considered in this event, however, would be a failure to scram the reactor. Such an event was generally thought to be extremely unlikely and on the order of 10^{-7} per year in frequency. In February 1969, a consultant to the ACRS, E.P. Epler, made the observation that if anticipated transients were estimated at a frequency of something on the order of one per year and that failure to scram might be a 10^{-4} event based on reliability analyses, then the ATWS scenario might be more on the order of 10^{-4} per reactor year and not 10^{-7} per year, putting the NRC on a course for rulemaking on the issue (NRC, 1993). A controversial report, WASH-1270 (1973) explored how to accommodate protection from ATWS with some consideration given to whether the plant was easily modified to accommodate ATWS or whether an older plant was not so easily modified. The final ATWS Rule placed additional requirements on the independence and diversity of shutdown capability. However, before rulemaking could be completed, two real world events at the Browns Ferry BWR plant (1980) and the Salem PWR plant, occurring in 1980 and 1983, respectively. In the Browns Ferry event, a shutdown from low power suffered a failure of the control blades to insert on one side of the reactor that was found to be due to a hydraulic problem with the blade insertion system (see Lutz and Williamson, 2021b). In the Salem events, an electrical problem prevented scram for an automatically generated shutdown due to low steam generator water levels (see Lutz and Williamson, 2021b; Rempe, 2021b). Although these incidents had no ultimate safety impact (perhaps because the reactors were at very low power), they highlighted the frequency significance of ATWS events and led to greater industry focus on reducing the frequency of reactor scram at full power. Due to improvements in reactor protection systems between 1980 and 1995, the frequency of scram in the US fleet was reduced by about a factor of 10 and at-power scrams today are extremely infrequent. This of course reduced the overall frequency of ATWS by at least this factor, but interestingly brought great savings to the industry by reducing the lost revenue due to undesired scrams (NUREG/CR-6042 Rev. 2).

Severe accidents beyond the plant design basis

As discussed earlier, potential health effects of severe accidents were explored in landmark studies such as WASH-740 (AEC, 1957) and WASH-1400 (Rasmussen, 1975) based on postulated maximum credible events that could result in radioactive releases from a damaged or melted reactor core which could subsequently escape from the containment and threaten public health. In NUREG-1150 (NRC, 1990), probabilistic risk assessments were performed for five US BWR and PWR power plants, focusing in what are called “internal events,” in which a spectrum of accident sequences and outcomes are explored based on fault-tree/event-tree analyses of the “things that can go wrong internal to the plant.” Briefly, internal events reflect the spectrum of pipe breaks (considering their sizes and locations), Station Blackout scenarios with loss of power or loss of heat rejection, and many other accidents with nuanced variations. In the PRA, thousands of scenarios are examined probabilistically. In most cases, core damage or radiological releases are arrested by the successful activation of safety systems. However, in some cases in which core melting occurs with subsequent release of radioactivity to the containment followed by containment failure, releases or radioactivity to the environment with potential health consequences can occur.

One of the metrics adopted in the NUREG-1150 study was called the “conditional containment failure probability” (CCFP), which is the probability for a given plant that the containment ultimately fails conditional on a core melting accident having taken place. While the “core melt frequency” determined in the PRA is generally a low frequency event on the order of 10^{-4} per reactor year or lower, the CCFP for the NUREG-1150 plants generally ranged between 0.1 and 1.0. This ultimate containment failure was further characterized as an early containment failure or late containment failure depending on whether the radiological release would take place before or after protective measures such as evacuations or shelter in place could be implemented. Accidents of this magnitude have the potential to significantly exceed the regulatory requirements of 10CFR50 concerning core coolability and 10CFR100 concerning radiological dose to the public. Such accidents are deemed in the US regulatory framework as being beyond the plant design basis, or Beyond Design Basis Accidents (BDBAs) as happened in 2011 at the Fukushima Daiichi plant. Typically, such severe accidents require the failure of multiple safety system failures such as failure of the ECCS to effectively reflood and cool the core following a DBA LOCA, for ELAP conditions to persist beyond the plant-specific coping time, or any number of other system failures that push the accidents outside of the plant design basis.

In the United States, there are no specific success criteria for BDBAs other than a regulatory requirement imposed by the NRC following the accident at Three Mile Island requiring plants to perform a PRA or an industry version of a PRA, known as an Individual Plant Examination (IPE), and include the PRA findings in an added Chapter 19 to the plant SAR (US NRC, 1988). The idea was to encourage licensees to assess potential plant vulnerabilities and make cost effective plant modifications that would reduce specific vulnerabilities or improve the overall plant core damage frequency (CDF), without placing onerous backfit requirements on the plant owner/operator. In meeting these requirements, plants must perform at least a Level 1 PRA (or IPE) to identify the plant sequences that lead to core damage events, although the plants can optionally perform a full scope PRA that includes a Level 2 analysis that further determines the radiological source terms resulting from core damage accidents and a Level 3 analysis that estimates the health effects from offsite radiological releases (see Modarres and Kim, 2021). Radiological consequences are expressed in terms of potential prompt fatalities from large and early releases mainly associated with Iodine releases or longer-term latent health effects such as cancers and lymphomas due to chronic exposure to low level radiation, such as from Cs-137/Cs-134 land contamination. The Level 1 analysis codes model the plant accident from its initiation up to the point of fuel damage that exceeds the 10CFR50.46 temperature and cladding oxidation limits and are aimed mainly at defending the DBA success criteria for fuel performance and core

coolability. Those accidents that fail the 10CFR50.46 success criteria and proceed to core melting accidents are deemed beyond the design basis in the US (BDBA) with radiological releases. Another class of Severe Accident Analysis computer codes is used to predict plant state responses, such as fraction of core damaged or melted, rate and timing of radiological releases, generation of hydrogen and its behavior in the containment, and the containment response to increasing pressure and temperatures that can lead to offsite releases. The Level 3 consequence and health effects use the radiological source term information from the Level 2 analyses together with the site weather characteristics as boundary conditions to an atmospheric transport code which calculates the downwind atmospheric transport of radioactive releases and the fallout to the ground to estimate ground contamination and near or long term health effects on humans.

Post 9/11 measures and Fukushima lessons learned

Following the September 2001 attacks on the World Trade Center in which commercial aircraft were used to destroy two towers, an urgent study was initiated by the NRC to explore the vulnerabilities of nuclear power plants to commercial aircraft attack. The scenarios examined considered the loss of large areas of the plant due to the damage from an aircraft impact, but in many ways resemble Station Blackout conditions ELAP and resultant loss of decay heat removal to move heat via pumps as well as loss of cooling function from large area damage. The studies showed that on-site availability of alternative means of generating AC power with portable generators and on-site availability of portable pumps could be highly effective in mitigating the ELAP conditions. This finding led to the NRC requirements of plants maintaining portable emergency equipment, such as generators and pumps, to mitigate such events.

Also identified in the aircraft impact studies was the vulnerability of spent fuel pools of certain plant designs that are outside of the containment and more vulnerable to aircraft attack—here a sudden loss of water from the spent fuel pool could result in damage to spent fuel from a sudden loss of cooling water. Until this time, there were no guidelines on the loading and distribution of off-loaded fuel in the spent fuel pools. However, research showed that optimal fuel dispersing practices in which very hot, recently off-loaded, fuel was interspersed among colder, earlier-offloaded, fuel assemblies could significantly reduce the likelihood of a severe spent fuel pool accident. Optimal fuel distribution guidance was developed, and the NRC Commission issued orders requiring that the US fleet adopt such optimized storage practices.

A decade later, the accidents at the Fukushima Daiichi plant reinforced the need to maintain portable equipment to respond to loss of large area, this time from natural disasters as opposed to terrorist initiators. This further reinforced the industry commitment to maintaining portable equipment and ensuring that logistical concerns, such as length of hoses, compatibility of hose connections, and the time required to implement mitigation measures were addressed. Both the 9/11 (2001) terrorist events and the 3/11 (2011) Fukushima accidents have led to increased DiD and mitigation options for extreme plant damage scenarios.

Emerging issues

New reactor designs under consideration include non-LWR concepts, many of which may be small modular designs because of their potential to increase safety and decrease manufacturing and construction costs. The current generation of LWRs is based on decades of cautious research into a developing nuclear technology. The research base in non-LWR concepts, which include novel designs using molten salts or liquid metals as working fluids and new fuel forms from the UO₂ fuel used for decades in the LWR industry, will require development of new regulatory guidelines. Development timelines for these advanced reactors cannot afford the decades invested in the regulatory oversight and research that the LWR industry enjoys. Advanced analysis methods are being developed to demonstrate safety in the design of the next generation nuclear power concepts, and validation of these methods will be critical in advancing the strategic impact of nuclear power in an ever more dangerous world to address climate change and the need for safe and secure power. As with LWRs, advanced analytical methods must meet regulatory requirements to satisfy health and safety objectives and high level design requirements.

Summary

The safety analysis of commercial light water nuclear power plants has been under continuous evolution and improvements since the first reactors were constructed in the 1950s. Practical and operational experience has been gained over the years from knowledge advancements, experimental investigations, and several landmark real-world accidents. Regulatory practice is reflected within the rules and guidance's set forth in Regulatory Guides and Safety Standards by the NRC, IAEA, and other international oversight agencies. The analyses described in this article are aimed at assuring safe nuclear shutdown capability and core cooling capability under all possible accident conditions including normal operations, anticipated transients, design basis accidents, and the minimization of risks from severe accidents that are outside the design basis of the plant. This article has provided an overview of the types of safety analyses required to demonstrate design safety and approval for operations with the principal objective of maintaining adequate health and safety to the population. In addition, this article identified what guidance is available for performing these analyses, and what criteria are compared with analyses results.

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Overview of Nuclear Regulation and Regulatory Framework

Joy L. Rempe^a and Laurence G. Williams^b, ^a Rempe and Associates, LLC, Idaho Falls, ID, United States; and ^b Centre for Nuclear Engineering, Department of Materials, Imperial College London, London, United Kingdom

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Glossary

Defense-in-Depth A design and regulatory principle that relies on several consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or engineered barrier were to fail, the subsequent level or barrier would be available. When properly implemented, Defense-in-Depth ensures no single technical, human or organizational failure could lead to harmful effects and the combinations of failures leading to significant harmful effects are of very low probability.

Graded approach to regulation An approach in which the level of regulatory oversight is balanced against the hazard potential of the nuclear installation; i.e., the greater the hazard potential of the installation and hence the complexity of the engineered barriers to maintain the safe working envelope, the greater the regulatory requirements and depth of safety assessment, regulatory inspection, and enforcement powers.

Inspection The core activity performed by the regulator to verify compliance with its regulatory requirements, including regulations, license conditions and permits.

Nuclear facilities and activities Facilities and activities that may result in radiation risks arising from the use of nuclear or radioactive materials.

Nuclear materials Plutonium, uranium 233, uranium enriched in the isotopes 233 or 235, and any material containing one or more of the foregoing. Nuclear material does not include source material (i.e., uranium containing the mixture of uranium isotopes contained in nature; uranium depleted in the isotope 235; thorium; or any of the foregoing in the form of metal, alloy, chemical compound or concentrate) (Adapted from IAEA, 1980).

Nuclear regulation The inspection, assessment, permissioning and enforcement activities necessary to control the siting, design, construction, commissioning, operation and decommissioning of nuclear installations.

Nuclear safety All the activities required to be undertaken to protect workers, the public and the environment from the effects of ionizing radiation as a consequence of operating nuclear facilities during both normal and accident conditions.

Nuclear security All the activities required to be undertaken to prevent the theft of and illegal transfer of nuclear or radioactive materials and to prevent unauthorized access to or sabotage of nuclear facilities.

Regulatory capture Process in which a regulatory agency becomes dominated by the commercial or political concerns of the industry or sector it is charged with regulating. Eventually a captured public-interest agency operates essentially as an advocate for the industries it regulates. Such cases may not be directly corrupt, as there is no quid pro quo; rather, the regulators simply begin thinking like the industries they regulate.

Safety analysis The study of the behavior of a nuclear installation under both normal operation and postulated accident conditions in order to determine: (a) safety margins provided in normal operation and in transient conditions, and (b) the adequacy of the engineered barriers needed to detect the departure from normal operating conditions and initiate protection systems to prevent or limit the consequences of accidents.

Safety assessment Carried out by the regulator, the systematic and independent evaluation of the safety analysis set out in a safety case for a nuclear installation. It is used to enable the regulator to judge the adequacy of the licensee's safety case and support licensing decisions on the plant acceptability, once the risk associated with its operation is known.

Safety culture Culture that encourages a questioning and learning attitude and discourages complacency with regard to safety.

Introduction

Society demands that industrial applications having the potential to cause harm to people or the environment should be subject to regulation by their Government. The amount of regulatory oversight is dependent upon the scale of the potential hazard. The use of nuclear energy is no exception; and hence, because of its high hazard potential, it is subject to a stringent nuclear regulatory framework. It is the Government's responsibility to make the laws that establish the nuclear regulatory framework. The nuclear operators/licensees have a responsibility to comply with the legal requirements set out in the regulatory framework, and it is the role of the regulators to enforce the law. Nuclear regulation is, therefore, essentially law enforcement.

There is no universal legal system that governs nuclear safety. Hence, the responsibility for establishing a nuclear regulatory framework rests with each Nation State. As nuclear regulation is an integral part of a Nation State's legal system, there is no single template for nuclear regulation and licensing. However, Member States of the United Nations International Atomic Energy Agency (IAEA) have developed a number of international conventions in the area of nuclear safety and nuclear security in order to provide the international community with the necessary confidence that nuclear safety and nuclear security is being delivered to consistent standards. The key conventions are the Nuclear Safety Convention (IAEA, 1994), the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA, 1997), and the Convention on the Physical Protection of Nuclear Materials (IAEA, 1980, 2011, 2016).

To support these Conventions, the IAEA has produced extensive guidance on nuclear law (Stoiber et al., 2003) and established a comprehensive set of nuclear safety standards and guides. The IAEA safety standards, principles, and guidance are intended for use by Member States, national authorities, and other international organizations. The Fundamental Safety Principles (IAEA, 2006) provide the high-level aims that are necessary for Member States wishing to make use of nuclear energy. These principles define the roles and responsibilities of the licensee, the regulator, and the government. They also address the need to consider topics, such as justification of facilities and activities that give rise to radiation risk, balancing the optimization of protection with individual risk limitations, the protection of both current and future generations, and prevention and mitigation of accidents. These principles provide a basis for Member States to establish safety requirements for protection against exposure to ionizing radiation under the IAEA's safety standards program and provide the rationale for its wider safety related program. However, the application and enforcement of such standards remain a national obligation.

The nuclear safety conventions and their associated standards clearly place the responsibility for safety on the operator/licensee of the facility. In the case of nuclear security, responsibility is placed on the Government of the Member State. However, the consistent theme of the international nuclear safety and nuclear security framework is the protection of people and the environment from the harmful effects of ionizing radiation. Hence, most regulators indicate their primary mission or safety objective is the protection of people and the environment. Regulators provide assurance to their Governments that nuclear energy is being used safely and securely and in accordance with the legal requirements.

Although it is recognized that nuclear regulation is, in general, a national responsibility, Governments have long recognized the transboundary risks associated with the release of radioactive materials into the environment. The accidents of Three Mile Island (Skillman and Rempe, 2021), Chernobyl (Sich, 2021), and Fukushima Daiichi (Mizokami and Rempe, 2021), reinforced the benefits of improved international policy coordination (Lamm, 2017). The IAEA has played a significant role in the development of international safety standards covering such things as: Government responsibilities to establish a regulatory framework; the siting, design, construction, commissioning, operation and decommissioning of nuclear facilities; the transport of nuclear materials; and the management of radioactive waste. The IAEA has also established programs to assist countries in the evaluation of the effectiveness of their regulation of nuclear facilities and activities. In addition to the IAEA, the Nuclear Energy Agency (NEA) of the Organization of Economic Cooperation and Development (OECD) plays a role in providing guidance on nuclear safety and regulation

in order to share good international and national practice with regulators and the nuclear operating organizations. In the area of nuclear security, the World Institute for Nuclear Security (WINS) provides a forum for promoting best practice.

This chapter provides an overview of nuclear safety regulation and regulatory frameworks. More detailed information, with specific examples, is found in [Bley and Malsch \(2021a,b\)](#), [Ray \(2021\)](#), [Rempe \(2021\)](#), and [Williams \(2021\)](#). Because of the early role the U.S. played in the deployment of nuclear technologies for non-military applications, the U.S. experience is emphasized in these chapters. The focus of this chapter is on the regulation of nuclear facilities for commercial power applications. Hence, statements do not apply to the limited number of research and nuclear fuel cycle facilities located on government sites in some countries. Within the U.S., these research and fuel cycle facilities are subject to internal regulatory oversight by the U.S. Department of Energy. Whilst some countries have followed the U.S. approach, other countries such as the United Kingdom developed different regulatory approaches. These are discussed in [Williams \(2021\)](#).

Safety objective and scope

Internationally, the primary mission or safety objective of nuclear regulatory organizations (often called the Regulatory Body) is the protection of people and the environment from the harmful effects associated with the use of radioactive materials (e.g., [US NRC, 2018](#); [IAEA, 2006](#); [OECD, 2014](#)). Regulators operate within the powers granted to them in a Nation's nuclear laws, which often require an operator of a nuclear installation to be licensed. Regulators aim to achieve this objective via the application of a licensing process that may include conditions attached to a License and the application of Regulations. The licensing process often imposes measures to enable the Regulatory Body to be assured of the following: nuclear facilities are sited, designed, constructed, commissioned, operated, and decommissioned to the appropriate safety standards; radioactive waste is effectively managed; and the risks to workers and the public are as low as reasonably achievable (ALARA). In some countries, risks are required to be as low as reasonably practicable (ALARP). As discussed in [Risk justification and assessment](#), the concepts of ALARA and ALARP differ but are typically assumed to have equivalence.

As noted in [IAEA \(2006\)](#), the regulatory process has the following objectives:

- controlling radiation exposure of people and release of radioactive material to the environment;
- restricting the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, and radioactive source or any other source of radiation;
- mitigating the consequences of such events if they were to occur.

Regulation applies to all facilities and activities and for all stages of the life of such facilities including the associated transport of radioactive material and management of radioactive waste. Regulation is applied to operating facilities during normal operating conditions and off-normal events (e.g., accidents). A graded approach is applied to regulation to ensure the regulatory burden is proportionate to the hazard.

Roles

It is usual for a Regulatory Body to grant a nuclear license to construct or operate a nuclear facility to a corporate organization when it is satisfied that the organization is fit to hold a nuclear site license. The Regulatory Body can "license" particular individuals to undertake specific activities, e.g., to operate a nuclear reactor. Once the Regulatory Body has granted a license to an organization, it becomes the licensee and has the primary responsibility for safety throughout the lifetime of the facility and any associated activities.

As discussed above, regulating nuclear and radiation safety is a national responsibility. Under the Nuclear Safety Convention, Governments must establish a Regulatory Body and ensure it is effective. To be effective ([IAEA, 2006](#); [IAEA, 2010a](#)), the Government must ensure the regulator has the following attributes:

- adequate legal authority, technical and managerial competence, and human and financial resources to fulfil its responsibilities;
- effective independence from the licensee and any other body, so it is free from any undue pressure from interested parties;
- appropriate means for informing those in the vicinity (e.g., the public and other interested parties) as well as the information media about the safety aspects (including health and environmental aspects) of facilities and activities and about regulatory processes in an open and inclusive process.

In the field of nuclear safety, radiation protection and environmental protection, the Government and the Regulatory Body often share the responsibility for establishing appropriate standards and regulatory processes to achieve the primary mission and safety objective. However, the prime responsibility for safety rests with the licensee. The Regulatory Body is not responsible for safety.

In the field of nuclear security, the Regulatory Body is often referred to as the "Competent Authority"; however, it is the Government that is responsible for nuclear security. The Government and the Regulatory Body, again, often share the responsibility for establishing appropriate standards, regulations and regulatory processes to achieve the primary mission and security objective. The Government has the responsibility of defining the "Design Basis Threat," i.e., the threat against which the licensee must provide protection. Protection may include the use of an armed guard force which may be under the control of the licensee or the Government. The licensee has the responsibility for complying with the nuclear security regulations.

The Convention on Nuclear Safety emphasizes not only the operator's ultimate responsibility for safety but also the importance of an independent, technically competent regulator. (IAEA, 1994) Among the core lessons learned from the Fukushima Daiichi accident, the Director General's report (IAEA, 2015) listed several attributes characteristic of an effective regulatory framework, such as the need for an independent Regulatory Body with the necessary legal authority, technical competence, and a strong safety culture. IAEA (2015) also emphasized the importance of having a questioning attitude, continuously challenging or re-examining prevailing assumptions about nuclear safety and the implications of decisions and actions that could affect nuclear safety.

The Regulatory Body has an important role to play in the delivery of nuclear safety and nuclear security; however, the regulator is only one part of a "triple lock" needed to protect society (Williams, 2013a,b). This triple lock comprises three key components. The first lock is the technically competent licensee. The second lock is the Licensee's internal (independent of operational responsibilities) nuclear safety assurance function. The third of the three required locks is an effective Regulatory Body. When all three locks are in place, society can be confident that the use of nuclear energy is safe and secure.

Further analysis of the Fukushima Daiichi accident indicated that a network of institutions and interfaces or "Institutional Strength-in-Depth" (ISiD) is also required (INSAG, 2017; Meserve, 2016). To prevent a nuclear accident, the ISiD philosophy relies on three independent institutional sub-systems:

- a strong nuclear industry
- a capable and effective nuclear Regulatory Body
- stakeholders who reinforce and ensure a robust institutional framework

It is argued that Government should ensure each sub-system has the authority and responsibility to fulfil its clear and distinct role and should link them together so each sub-system strengthens and reinforces the others. For example, the U.S. Government, by law, establishes the obligations of the licensee, creates the Regulatory Body, and through law, governs access to information, public hearings, and legal processes for challenging the Regulatory Body (enabling the public to oversee the entire process). Meserve (2016) observes that to be effective, it is important for each sub-system to have internal ISiD that incorporates elements or layers, such as a strong technical capability, a management system with multiple checks, internal independent oversight, external expert panels, a strong quality control and assurance program, and a vibrant safety culture. Although several references (e.g., see IAEA, 2016; INSAG, 2017) emphasize that the primary responsibility for safety lies with the licensee, the Regulatory Body has the prime responsibility for oversight of the licensee. Thus, the Regulatory Body must have the authority, technical knowledge, capacity and capability to ensure the operator has robust ISiD in place and the protection of the public, society and the environment is always secured. To do this effectively, the Regulatory Body must have its own "ISiD" firmly established within its own organization. Linkages among the various sub-systems depend on transparent and open communications and accountability. Leaders in industry and the Regulatory Body must welcome challenges from stakeholders, listen, respond openly, learn, and improve. Fig. 1 depicts the ISiD model of a robust national nuclear system (INSAG, 2017).

In the U.S., a license is granted for a limited time period. Upon expiration, the licensees must apply for a new license or license renewal. As described in Ray (2021), the regulator also certifies reactor designs for a period of time within the U.S. This is a process in which the public can again provide input and the regulators can review the safety of the licensed facility. As elaborated upon in Williams (2021), licenses are valid over different time periods. In Canada, licenses are valid for 2–5 years. In the U.S., licenses are valid for 40 years, and renewals are granted for up to 20 years. In several countries, regulators use a periodic safety review



Fig. 1 Graphic depiction of a robust national nuclear system ("Regulation" is regarded as the interactions between the Regulatory Body and industry; it includes all regulatory activities and controls, but a prime method of interaction and feedback is the use of regulatory inspection activities.). Courtesy of International Nuclear Safety Advisory Group (2017) *Ensuring Robust National Nuclear Safety Systems—Institutional Strength in Depth*. INSAG-27, IAEA Publication 1779, Vienna.

(PSR) process, with a reassessment usually every 10 years. In the United Kingdom, the license is not time-limited, but each facility is required to undertake a PSR and continued operation is conditional on the Regulatory Body accepting the licensee's safety case to support its PSR. Time limitations on licenses can, depending on the national legal framework, be beneficial. For example, time limits can be used to force up-to-date best practices and periodic scrutiny of licensee performance. In addition, a licensee may be seen to be more accountable if public input is sought frequently over the lifetime of the nuclear power plant.

Regulatory functions

Typical actions conducted by an effective Regulatory Body are identified in several references (e.g., [OECD, 2014](#); [INSAG, 2017](#)) and include the following:

- define safety goals
- set policies and standards
- produce regulations
- license nuclear installations
- review licensee (or applicant) safety submissions during siting, design, construction, commissioning, operation and decommissioning
- carry out independent safety assessments
- permit licensee activities such as construction, commissioning, operation, modifications to the licensing basis, etc.
- sponsor safety research
- review operating experience
- interact with stakeholders and inform the public on radiation protection and nuclear safety
- conduct inspection activities to verify compliance with regulations or license conditions
- carry out enforcement activities such as issuing letters, notices, fines, prosecutions, and shutdown
- oversee/advise on emergency preparedness

Safety assessments and inspections

Safety assessments involve the systematic analysis of the behavior of a nuclear facility under both normal operation and accident conditions. Its aim is to substantiate the claims made in the Licensee's safety case in order to verify the necessary safety margins and the consequences of postulated accident conditions. Safety inspections are required to verify licensee compliance with regulatory requirements and commitments. The Regulatory Body does not have unlimited resources; and hence, it has to sample both the safety documentation provided by the Licensee and prioritize its inspection activities. The selection of appropriate samples to investigate and the prioritization of inspection activities are based on a combination of experience and the application of a graded approach. For those facilities with high hazard potential and complex engineered barriers needed to maintain the safe working envelope, the Regulatory Body would expect to carry out more in-depth assessment over a larger sample of the safety documentation, and prioritize its inspection activities on those aspects of the licensee's operations that maintain the integrity of the key nuclear safety and security systems.

A facility may only be constructed and commissioned, or an activity may only be commenced, once it has been demonstrated to the satisfaction of the Regulatory Body that the safety case for the design of the facility, including the proposed safety measures, is adequate. Safety assessments, in whole or in part, must be repeated as necessary throughout the lifetime of the facility in order to consider changes in circumstances (such as new standards or new scientific and technological developments), lessons learned from operating experience and accidents, modifications, and aging effects. Inspection is a continuous process commencing with construction. The Regulatory Body's inspection plans are tailored to the different phases over the lifetime of the facility.

International guidance emphasizes the importance of learning from accidents and operating experience ([INSAG, 1996](#); [OECD, 2014](#)). The licensee is expected to have processes for the feedback and analysis of operating experience, including initiating events, accident precursors, near misses, accidents and unauthorized acts, so lessons may be learned, shared, and acted upon. The Regulatory Body should monitor the Licensee's activities in these areas through its inspection plan to ensure the Licensee is taking the necessary actions and learning from experience. As discussed in [Lutz and Williamson \(2021a,b\)](#), effective emergency procedures and response actions must be comprehensive, prepared in advance, and practiced. For facility-based emergencies, the licensee is responsible for producing the necessary emergency plans and the Regulatory Body is expected to approve the plans, which normally involve annual demonstration exercises. The Regulatory Body, as part of its facility inspection plan, is expected to assess the licensee's performance during these exercises and require improvements should the Licensee fail to demonstrate its ability to successfully manage the event being tested in the exercise. In some countries, the Regulatory Body contributes to the development and testing of off-site emergency plans.

Stakeholder interactions

Public acceptance of nuclear power is not a matter for the Regulatory Body; public acceptance is a matter for Government and society. The Regulatory Body, whilst being neutral on the deployment of nuclear energy, has a role to play in enabling the public and other

stakeholders to have confidence in the safety and security of nuclear facilities (Here, the term, “stakeholders,” refers to all interested parties, including the public, non-governmental organizations, media, and shareholders. Views from stakeholder constituents may differ.). Public trust and confidence in the Regulatory Body is therefore essential; to gain and maintain this trust, the Regulatory Body must not only be independent but also be seen to be independent. The public and other key stakeholders must have trust in the people making the regulatory decisions. The Regulatory Body’s approach to inspection of nuclear facilities, its methods of assessing the nuclear safety cases, and its approach to enforcement should be transparent and be in the public domain. The Regulatory Body should produce regular reports that allow the public to see not only what the Regulatory Body has been doing, but also to see the Regulatory Body’s assessment of the licensee’s nuclear safety and security performance. Regular presentations to the media and public hearings to explain the actions of the Regulatory Body also contribute to the development of this trust.

Several publications emphasize the important role of stakeholders. As observed in (INSAG, 2006), the nuclear industry and the Regulatory Body have an obligation to explain nuclear safety provisions and plant performance in an open, comprehensible and transparent way and to respond to legitimate questions and challenges.

The Regulatory Body should develop and maintain its relationships with key stakeholders throughout the lifetime of the facility from the initial siting of a facility through construction, commissioning, operation and decommissioning. In many countries, the siting of a nuclear facility is the responsibility of either central or regional Government and controlled via planning legislation. The Regulatory Body is often required to provide information on the nuclear regulatory system at planning inquiries to give the public confidence the facility will be independently regulated at all stages throughout its lifetime.

Processes for public participation must allow for all stakeholders to be free to comment on the Regulatory Body’s activities and be sufficiently robust to resist forces seeking to undermine proper policy processes. The licensing system must be efficient and not result in undue economic burdens on the licensee. In the case of nuclear power, investor concerns focus on administrative procedure costs, a fear of technical difficulties, and project delays incurred by long public inquiries. The Regulatory Body can address some of these concerns through the implementation of an efficient licensing process, having clear and transparent regulatory requirements and expectations, and having clear criteria against which safety cases will be assessed. The Regulatory Body should also work with the licensee to produce an agreed “Regulatory Schedule” for the construction, commissioning and initial operation of the nuclear facility. The Regulatory Schedule should align the project program with required “hold points.” It should also set out the dates for submission of the licensee’s nuclear safety and security documentation required for each “hold point.” Sufficient time should be allowed to enable the Regulatory Body to assess the documentation and give the necessary permission to proceed past the “hold point.”

The INSAG (2017) also recommends that Government and the Regulatory Body consult with independent advisory bodies; for example, the Advisory Committee on Reactor Safeguards supports the U.S. Nuclear Regulatory Commission (see Bley and Malsch, 2021a).

The adoption of the Convention on Nuclear Safety in 1994 legally binds participating countries to prepare and submit National Reports on the safety of their nuclear programs. The Regulatory Body usually contributes to the production of these reports and often leads on the presentation of the National Report at IAEA “peer review” meetings of the contracting parties.

Application of Defense-in-Depth

To ensure that the likelihood of an accident having harmful consequences is acceptably low and to mitigate any releases from such accidents, the Regulatory Body applies the Defense-in-Depth principle (INSAG, 1996). This principle requires the failure of several consecutive and independent levels of protection (barriers) before any harmful effects to people or the environment occur. Levels of protection include a combination of factors, such as remote site selection, engineering safety features, safety margin in design, diverse and redundant protection systems, operating procedures and practices, management commitment to safety and security, and a robust nuclear safety and security culture. When properly implemented, Defense-in-Depth ensures that the combination of failures that could give rise to significant harmful effects has a sufficiently low probability of occurrence. The Regulatory Body will also normally require that no single technical, human, or organizational failure should lead to undesirable consequences such as an uncontrolled release of radioactivity (often referred to as the single failure criterion).

The Regulatory Body may require the Licensee to apply the Defense-in-Depth principle in the assessment of external hazards and severe accidents. Table 1 lists the objectives and typical implementation methods for a Defense-in-Depth concept with five levels of safety. (INSAG, 1996) Methods identified in Table 1 consider design, construction, and operation activities as well as accident management procedures to mitigate unexpected events and arrangements for preparedness and response for a nuclear or radiation emergency. As noted above in the discussion of the “Triple Lock” and the “ISiD” philosophy, the Regulatory Body ensures wider application of the Defense-in-Depth principle by setting safety objectives, by conducting technical assessments of the safety justifications provided by the Licensee, and by performing inspections to check compliance with regulatory requirements.

Risk justification and assessment

International expectations require all new practices involving the use of radioactive materials be “Justified.” For facilities and activities to be justified, their benefits should outweigh potential radiation risks. It is not the role of the Regulatory Body to assess the benefits of a facility or practice; and hence, it is normally the Government that makes “Justification” decisions. The Regulatory Body may provide input to this decision by performing an independent review of the risks.

Table 1 Levels of Defense-in-Depth.

Level	Objective	Essential means
1	Prevention of abnormal operation and/or failures	Conservative design High quality in construction and operation
2	Control of abnormal operation and detection of failures (protection)	Control systems Limiting systems Protection systems
3	Accident control within design basis (protection)	Engineered safety features Accident procedures
4	Control of severe plant conditions (protection)	Complementary measures Accident management
5	Mitigation of radiological consequences of significant radioactive releases	Off-site emergency response

Courtesy of International Nuclear Safety Advisory Group (1996) *Defence in Depth*. INSAG-10, IAEA Publication 1013, Vienna.

The design of a nuclear facility or activity is considered to be optimized if it can be demonstrated the highest level of safety that can reasonably be achieved throughout its lifetime is obtained without unduly limiting its utilization. This approach requires the risk to either workers or the public arising from the exposure to ionizing radiation be reduced to ALARA or ALARP. The ALARA and ALARP concepts are not the same but are usually considered to have equivalence. The evaluation of both ALARA and ALARP requires some understanding of costs required to reduce the risk and the benefits of that reduction. The latter requires a value being placed upon a life. The Licensee is required to demonstrate that the design/modification/operating procedure/activity meets the ALARP/ALARA requirement, and the Regulatory Body is expected to assess the Licensee's case.

All radiation risks from normal operation, abnormal or accident conditions must be assessed (using a graded approach) prior to granting a license and periodically reassessed throughout the lifetime of the facilities and activities.

Radiation protection

Measures to control radiation risk aim to ensure that no individual bears an unacceptable risk of harm. The Licensee is required to apply ALARA/ALARP to all activities involving exposure to ionizing radiation. However, the Regulatory Body may impose annual limits to exposure. The Regulatory Body's inspection program should check exposure records on a regular basis. Also, the Licensee should be required to report any incidents resulting in an exposure that exceeds the limits.

Radiation risks may transcend national borders and persist for long time periods. In judging the adequacy of measures to control radiation risk, possible current and future consequences should be considered. The effects of radiation on the environment have been less thoroughly investigated and regulated. The general intent of measures taken for environmental protection have been to protect ecosystems against radiation exposure that would have adverse consequences for populations of species. Regulation can affect management of radioactive waste, which has the potential to place an undue burden on future generations, ensuring that safe, practicable, and environmentally acceptable solutions are employed for long term management.

Integration of nuclear safety and nuclear security

Safety measures and security measures have the common aim of protecting human life and health and the environment (WINS, 2011). As observed in (IAEA, 2006, 2010b), some safety principles concern the security of facilities and activities to the extent that they apply to measures contributing to both safety and security. Examples include the following:

- appropriate security provisions in the design and construction of nuclear facilities and installations, which can also support safety;
- controls on access to nuclear facilities and installations for safety purposes, which can also be used to prevent the loss of, and the unauthorized removal, possession, transfer and use of, radioactive material;
- arrangements to mitigate the consequences of accidents and failures, which can also be appropriate for dealing with breaches in security that give rise to radiation risks;
- security measures for managing radioactive sources and radioactive material, which can also support safety.

Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security (Williams, 2013b).

Regulatory attributes

Independence

The Regulatory Body must be independent of the nuclear industry and independent of those with policy responsibility for the promotion of the nuclear industry. Whilst the Regulatory Body cannot be entirely separate from other governmental bodies, the

Government must ensure the Regulatory Body is able to make regulatory decisions “without undue pressure or constraint.” To enable the Regulatory Body to exercise this independence, it must be technically competent and have adequate financial resources to ensure the recruitment and retention of high quality staff (INSAG, 2017; IAEA, 2016; OECD, 2014, 2016).

As discussed in Williams (2021), some Regulatory Bodies report to their Governments and some report to their national or regional parliaments. INSAG (2017) recommends Governments ensure independence by delegating to the Regulatory Body the formal authority to license nuclear installations. Some populations might prefer Government organizations to take the final and formal licensing decisions because elected and publicly accountable persons are responsible for final licensing decisions. However, this would run counter to INSAG recommendations because it could undermine the effective independence of the Regulatory Body. In addition, INSAG (2017) contends it is prudent for a specialized expert agency to issue licenses because of the complex political, sociological, and technical issues associated with nuclear power.

In order to maintain its independence, a Regulatory Body must avoid regulatory capture, such that it is not dominated by the commercial or political concerns of the industry or sector it is charged with regulating. Difficulties encountered by the U.S. Atomic Energy Commission, which had the responsibility for both promoting and regulating the nuclear industry, led to a 1957 decision that required public hearings on all licensing cases and that made the existing Advisory Committee on Reactor Safeguards a statutory body with the responsibility to independently produce publicly-available reports on licensing cases, and ultimately, a 1974 decision to establish a separate regulatory agency, the U.S. Nuclear Regulatory Commission (Walker and Wellock, 2010; Bley and Malsch, 2021a).

While the Regulatory Body must be independent from the nuclear industry, this does not mean it should operate in isolation. As emphasized in OECD (2014), the Regulatory Body needs to have frequent and open discussions with all its stakeholders. Although financial independence is a necessary attribute, the Regulatory Body is accountable to the national Government or other political authorities that authorize its funding. Systems for financial accountability vary and continue to evolve as discussed in Williams (2021).

Independence will only be achieved when the Government is prepared to accept regulatory decisions without interference. Government should not use the Regulatory Body as a vehicle to deliver its nuclear policy, nor should it try to control the Regulatory Body or align it with Government policy objectives.

Effectiveness

Key attributes of an effective regulator are provided in OECD (2014) and INSAG (2017). For example, INSAG (2017) displays the ISiD regulatory sub-system attributes as a series of layers, including internal capability, technical knowledge, organizational structure, processes, and external peer review (see Fig. 2).

The Regulatory Body must demonstrate its ability to establish and apply coherent requirements for nuclear safety and nuclear security, including those for human performance and quality. As the Regulatory Body has the prime responsibility for oversight of the licensee, it must have the authority, technical knowledge, capacity and capability to ensure the licensee has necessary technical competence, nuclear safety and security management structure, a robust nuclear safety and security culture and the necessary financial resources to deliver its responsibilities.

Leadership

Effective leadership is a vital attribute of any organization, and experience has shown that inadequate leadership has been a contributing factor in many accidents and failing organizations. The Regulatory Body is a key component in the delivery of safe nuclear power. Hence, strong and competent leadership is a prerequisite for a successful Regulatory Body. The public (who put their trust in the Regulatory Body for their protection), the nuclear industry (which needs a clear understanding of who is ultimately responsible for regulatory decisions), and the Government (which needs to know where regulatory accountability lies) need to know who is responsible for decision-making. There is a need for the senior nuclear regulator or regulatory panel, which has the ultimate power to take regulatory decisions, to be clearly identified.

Culture

The Regulatory Body must have a culture that promotes openness, transparency and accountability. Several references (INSAG, 2017; IAEA, 2016; OECD, 2014, 2016) indicate that the promotion of “safety culture” is reinforced by conducting safety performance assessments or inspections and the use of a structured lessons learned process.

Regulatory approaches

The Regulatory Body may consider the use of various regulatory approaches. These include prescriptive based, goal-setting based, risk- and hazard-informed based, process based, self-assessment based, and education/influence based (OECD, 2014). Summary information for the first three types of approaches are provide here to illustrate their benefits and disadvantages. More detailed descriptions may be found in OECD (2014).

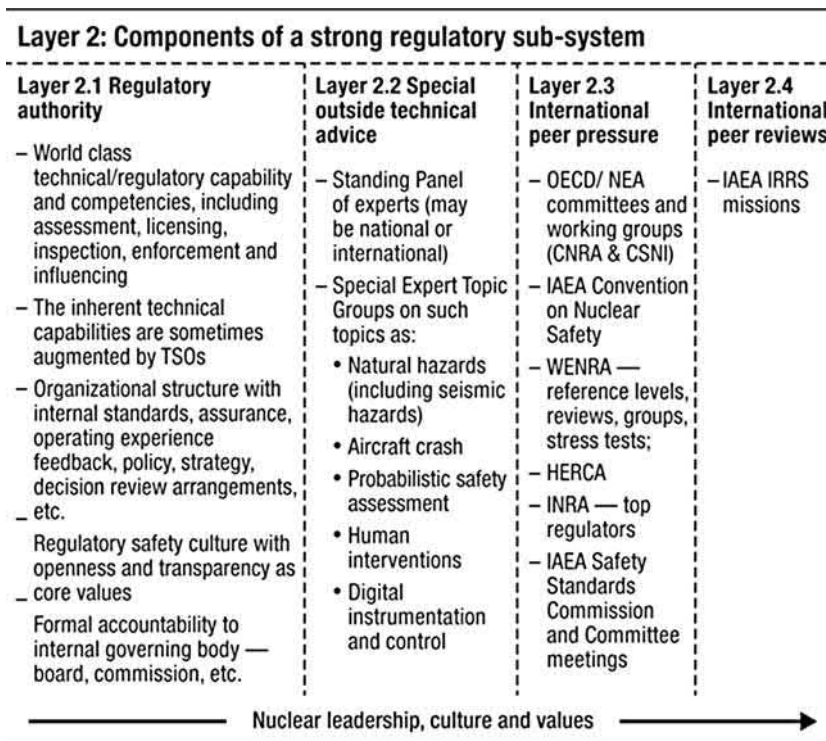


Fig. 2 Examples of the components of a strong regulatory sub-system [CNRA, Committee on Nuclear Regulatory Activities; CSNI, Committee on the Safety of Nuclear Installations; HERCA, Heads of European Radiological Protection Competent Authorities; IAEA, International Atomic Energy Agency; IRRS, Integrated Regulatory Review Service; INRA, International Nuclear Regulators Association; OECD/NEA, Organization of Economic Cooperation and Development Nuclear Energy Agency; TSO, Technical Support Organization; WENRA, Western European Nuclear Regulators Association]. International Nuclear Safety Advisory Group (2017) *Ensuring Robust National Nuclear Safety Systems—Institutional Strength in Depth*. INSAG-27, IAEA Publication 1779, Vienna.

Prescriptive approach

In this approach, the requirements on methodologies, standards, and quality assurance are prescribed by the Regulatory Body. The licensee must demonstrate the plant complies with these regulatory requirements. In addition, regulatory guidelines, describing methods acceptable to the Regulatory Body, may be provided. As discussed in [Bley and Malsch \(2021a,b\)](#), the U.S. Nuclear Regulatory Commission regulations were initially developed as a prescriptive approach.

A major benefit of a prescriptive approach is the level of clarity for both the regulator and the licensees. An additional benefit in some situations is that the approach can be used to exert specific regulatory authority. This approach, however, has several difficulties. The main difficulty is that by defining specific requirements against which the Licensee must comply, it can be argued that the Regulatory Body is assuming responsibility for plant safety. The prescriptive approach also requires the Regulatory Body to have sufficient technical competence to match the regulations with the nuclear safety, security and environmental protection requirements. It also places a great responsibility on the Regulatory Body to get it right as the industry will in general aim to comply with the required regulations. The approach can also be resource-intensive for regulators and inflexible for licensees and operators.

Goal-setting approach

This approach is used by the Regulatory Body to establish specific performance goals or outcomes for licensees to attain but does not specify how they must be attained. Licensees determine how they will conduct their work activities. The United Kingdom licensing system provides an example of a non-prescriptive approach. As discussed in [Williams \(2021\)](#), the Office for Nuclear Regulation (ONR) sets goals, but the Licensee can undertake whatever action it chooses to meet these targets. This approach puts the responsibility for nuclear safety, security and the protection of the environment clearly on the Licensee. The Licensee is required to demonstrate that its facility is safe, nuclear materials are secure, and the environment is protected. The Regulatory Body will assess the adequacy of the Licensee's safety case against its own safety, security, and environmental principles to ensure that the goals have been met.

A major benefit of a goal-based approach is that it allows licensees to select optimal methods to meet safety goals. This approach is more flexible for new technologies and encourages licensees to improve plant performance. However, a goal-based approach

presents some difficulties for the Regulatory Body in identifying good performance goals and quantifying outcomes. It can also present difficulties for the Licensee in that regulatory expectations are less clear than found in a prescriptive approach.

This approach requires a technically competent Licensee. It also requires a technically competent Regulatory Body to be capable of independently assessing the arguments made and the information presented before permissioning specific Licensee activities.

Risk-informed and hazard-informed approaches

Risk-informed and hazard-informed approaches are used by the Regulatory Body to determine the risk or hazard associated with an issue to determine the appropriate level of regulatory attention. A risk-informed approach considers the probability and potential for harm to identify areas of greatest risk. A hazard-informed approach uses specific criteria to identify areas of greatest potential for harm. As discussed in [Bley and Malsch \(2021b\)](#), the U.S. Nuclear Regulatory Commission is moving toward a more risk-informed regulatory approach.

Risk- and hazard-informed approaches are beneficial because regulatory attention is focused on prioritized safety issues. However, Regulatory Bodies using these approaches must consider risk and hazard assessment limitations, such as the lack of completeness, which could result in some areas receiving inadequate or too much attention.

Future considerations

Regulation continues to evolve. Several factors that may impact current regulation are harmonization of regulatory requirements, advanced reactor regulation, and interfaces between nuclear safety, nuclear security, and non-proliferation safeguards.

Harmonization of regulation

Despite similarities in the objectives, attributes and functions, there are differences in regulatory requirements across the world. As noted in the [Introduction](#), the main reasons for this revolve around the fact that regulation is law enforcement and there is no universal legal system covering all countries that are using nuclear energy. Differences also relate to: national energy policy; national industrial tradition (e.g., giving more credit to redundancy or diversity, or crediting a software-based system as opposed to hard-wired controls); consistency with national regulatory or legislative system (e.g., compliance with probabilistic safety criteria on individual and societal risk as applicable to the environmental policy); country-specific conditions (e.g., differences in geography such as potential flooding and seismic hazards); and consideration of uncertainties associated with severe-accident phenomena.

The Nuclear Safety Convention and the supporting IAEA safety standards may be used by Regulatory Bodies to benchmark their approaches and promote a measure of equivalence to be achieved. Efforts by Regulatory Bodies to work together through organizations, such as the International Nuclear Regulators Association (INRA), the Western European Nuclear Regulators Association (WENRA) and the OECD/NEA, have led to some harmonization of safety requirements and regulatory practices.

Regulation for advanced reactors

The regulatory framework and the Regulatory body should be capable of regulating any nuclear technology. As discussed in [Bley and Malsch \(2021a\)](#), [Rempe \(2021\)](#), and [Williams \(2021\)](#), some regulatory frameworks have been effectively regulating non-LWRs for many years. Most nuclear reactor regulatory requirements are, however, focused on large LWR designs. Attempts to license other designs (e.g., Small Modular Reactors and non-LWRs) are often hindered by existing regulatory requirements. Some existing regulatory requirements aren't applicable to specific advanced designs. [Bley and Malsch \(2021b\)](#) describes on-going activities by the U.S. Nuclear Regulatory Commission to develop new "technology inclusive" regulatory approaches better suited for such designs. Of particular interest are the so-called advanced reactor technologies with enhanced margins for safety such that a regulator could approve smaller emergency planning zone sizes, different engineered safety feature requirements (including the elimination of large leak-tight containment buildings and different approaches for calculating source terms) and exemptions from technology-specific design criteria. As discussed in [Bley and Malsch \(2021b\)](#), the use of risk-informed, performance-based approaches is emphasized to develop a more flexible licensing approach applicable to diverse technologies.

Interfaces between safety, security and non-proliferation requirements

To ensure that the public is adequately protected, it is now recognized that Regulatory Bodies must also address nuclear security and non-proliferation safeguards. Such issues increased in importance in the 1970s because of growing concerns about terrorist groups ([Walker and Wellock, 2010](#)). This led Regulatory Bodies to strengthen regulatory requirements for the transportation of nuclear materials and for nuclear plant security. Some Regulatory Bodies also implemented measures that would limit the export of nuclear materials and technologies associated with the proliferation of nuclear weapons. Following the terrorist attacks in the U.S. on September 11, 2001, increased security measures and practices at nuclear facilities were implemented. The result was significant enhancements to the design and operation of nuclear plants that strengthened Defense-in-Depth against malevolent behavior. Additional effort is needed to integrate regulatory requirements for safety, security, and non-proliferation safeguards.

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U.S. Regulation and Licensing—Historical Perspective

Dennis C. Bley^a and Marty Malsch^b, ^a Buttonwood Consulting, Inc., Albuquerque, NM, United States; and ^b Egan, Fitzpatrick Malsch and Lawrence PLLC, Washington, DC, United States

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Glossary

Atomic Safety and Licensing Board (ASLB) Three-member atomic safety and licensing boards conduct public hearings on license applications and challenges to licenses for the U.S. Nuclear Regulatory Commission (NRC).

Backfitting A backfit occurs when the NRC changes its mind about the safety of a licensed facility and wants a plant to add additional protection; i.e., make modifications to installed equipment.

Code of Federal Regulations (CFR) Regulations promulgated by U.S. government agencies. Commercial nuclear energy facilities are governed by 10 CFR Parts 1–199 promulgated by the NRC.

Engineered Safety Features Plant systems designed to reduce the likelihood or consequences of possible accidents.

FLEX An industry-developed set of standard flexible adaptations to plant equipment and operating procedures to allow plant operators to quickly attach portable mechanical or electrical components in case of failure of installed equipment—known as “Diverse and Flexible Coping Strategies.”

General Design Criteria A set of broad plant design criteria defined in Appendix A to 10 CFR Part 50. The intent was to define criteria such that, if applicants could demonstrate that their design would meet them, it could expedite the licensing process and make it more predictable.

Joint Committee on Atomic Energy Joint committee of the U.S. Senate and House of Representatives that oversaw the nuclear stockpile and technology following World War II and promotion and regulation of commercial applications of nuclear energy until 1977.

kw kilowatts.

Manhattan Project The World War II government agency that developed nuclear weapons.

mw megawatts.

Probabilistic Risk Analysis An approach to safety analysis that identifies what can go wrong and quantifies the likelihood and consequences of each scenario. Risk of a facility is presented in terms of the probability of various consequences, including rankings of contributors to the risk and a formal treatment of uncertainty.

Safeguards Committee In this chapter, this refers to the original Reactor Safeguards Committee and its successor, the Advisory Committee on Reactor Safeguards. It is an advisory committee established by statute to advise the NRC (formerly the Atomic Energy Commission) on matters of nuclear safety.

Regulatory Guides Guidance documents from the NRC that provide applicants with acceptable ways to meet the regulations.

Safety Analysis Report A formal document that must be submitted by applicants to obtain licenses for commercial nuclear plants.

Source Term the radionuclide content, release locations, and energy content of possible releases following accidents.

Standard Review Plan The guidance for NRC staff on how to review applications.

Technical Specifications A set of plant specifications that define what technical information is vital for evaluating nuclear safety and provide an acceptable range of operating limits.

U.S. Nuclear Regulatory Commission (NRC) The body established by the Atomic Energy Act, as amended, to regulate civilian applications of nuclear power in the United States (formerly the Atomic Energy Commission).

Introduction—Institutional evolution and technical issues

Nuclear energy was introduced to the world through weapons developed by the Manhattan Project during World War II. The first nuclear device, 'Trinity,' was detonated at Alamogordo, New Mexico on July 16, 1945. Less than 1 month later, the bomb, 'Little Boy' was dropped on Hiroshima and 3 days later 'Fat Man' was dropped on Nagasaki. This beginning, as the most terrifying weapon in history, strongly influenced the administration and application of nuclear science in the early years and has colored popular concerns ever since.

The institutional milieu

Following the war, the reactors, laboratories, and material of the Manhattan Project were transferred to a new agency, the U.S. Atomic Energy Commission (AEC) with the passage of the Atomic Energy Act of 1946. It focused on the military applications of nuclear energy and the associated need for secrecy. While the act acknowledged a growing recognition of possible peaceful uses of atomic energy, applications of the technology were restricted to U.S. Government activities. The act also created the Joint Committee on Atomic Energy (the "Joint Committee"), a unique and powerful oversight body with members from both the Senate and the House of Representatives, and the unique authority to report proposed legislation simultaneously to both houses of Congress. Developments were moving quickly. 1954 marked the U.S. Navy launch of the first nuclear powered submarine, U.S.S. Nautilus. Political pressure for developing civilian applications grew, both for envisioned technical benefits and to retain the U.S. position as the leader in the field. In 1953, President Dwight D. Eisenhower offered that "this greatest of all destructive forces can be developed into a great boon, for the benefit of all mankind" (Walker and Wellock, 2010).

A new law, the Atomic Energy Act of 1954, opened the possibility of peaceful, civilian use of nuclear energy. It also marked the beginning of nuclear regulation in the U.S., by directing the AEC to prepare regulations to protect the public. The AEC would now have three functions: manage the weapons program, promote civilian commercial uses of nuclear power, and protect the public health and safety from associated radiation hazards. Jumping ahead several decades, the conflict among these competing responsibilities, eventually damaged AEC's credibility on regulatory and safety issues. The Energy Reorganization Act of 1974 separated those responsibilities, with the military and promotional activities assigned to a new agency, the U.S. Energy Research and Development Administration (which later became the U.S. Department of Energy) and the safety regulatory activities assigned to a new agency, the Nuclear Regulatory Commission (NRC).

Returning to the first years of the AEC, the Commissioners formed the Reactor Safeguards Committee, out of concern for reactor safety issues. The Committee's first task in 1947 was to survey the nuclear reactors already operating to evaluate problems peculiar to design and siting. In 1950, the AEC formed another advisory committee, the Industrial Committee on Reactor Location Problems, with representatives from the electric-power industry and industrial-research corporations. They were tasked to examine the hazards associated with AEC production facilities, especially reactor-siting questions, and to consider classified agency activities for possible commercial development. The two committees were combined in 1953 as the Advisory Committee on Reactor Safeguards (ACRS).

Commercial nuclear power was a new field and the Atomic Energy Act of 1954 gave the AEC broad authority on how the promotional and regulatory aspects should be managed. While the AEC and the Joint Committee agreed on the high-level policy issues, the details were often contentious. Within a few years, Congress passed a 1957 amendment to the Atomic Energy Act of 1954. Among other things, this act established the ACRS as a statutory committee and required that ACRS review every construction permit or operating license application for a new commercial or test reactor and publish a public report on their review. This role continues to the present day.

In 1954, the AEC was facing the beginning of a new commercial industry and with respect to licensing, they were starting with a nearly blank slate. There was guidance from the National Committee on Radiation Protection (NCRP) on radiation protection guidelines. However, no other aspects of possible regulatory concern were in place. In 1955 the AEC created the Division of Licensing and designated a Reactor Hazards Evaluation Staff. During the same time period, the Joint Committee was looking for ways to allow greater public confidence in the AEC's regulatory process. The 1954 Atomic Energy Act had required public hearings on reactor construction permit and operating license applications only if potentially affected persons requested them. In order to promote public confidence in the licensing process, the 1957 amendment to the Atomic Energy Act required the AEC to hold mandatory public hearings on license applications for commercial and test reactors. These requirements proved more difficult than expected, when opposition to nuclear power projects increased. During the period 1957–62 extensive interactions among the Joint Committee, the AEC, and expert groups from government, industry, and academia led to several attempts to implement the hearing requirements. The AEC established a formal legal separation of the licensing staff from the hearing staff. In 1958, they set

up an Office of Hearing Examiners, which later, in 1962, became the source of the Atomic Safety and Licensing Board (ASLB) members. In 1959, based on the recommendation of the AEC, another amendment to the Atomic Energy Act allowed the AEC to enter into agreements with states, giving them certain regulatory responsibilities. In 1962 the Atomic Energy Act was amended to allow three-member ASLBs to conduct hearings and, in 1969, the AEC established an ASLB Appeal Panel. The 1974 amendment to the Atomic Energy Act finally directed separation of the weapons and promotional activities of the AEC from their regulatory responsibilities. The new NRC was established in 1975.

In parallel with this rapid organizational evolution, commercial nuclear power plant designs and applications for licenses proceeded at a rapid pace. In 1953, the Shippingport pressurized light water breeder reactor project was initialized as a combined effort among the AEC, Naval Reactors, Westinghouse, and Duquesne Light Company. In 1955, the AEC announced the Power Demonstration Reactor Program (Wiley, 1955) (later called the Cooperative Power Reactor Demonstration Program [AEC, 1963]). It was a government/industry cooperative effort to develop, construct, and operate experimental power reactors. Some new reactors were proposed to be developed using private funds alone. During 1955, applications were received for a number of reactors: Dresden, a BWR by Commonwealth Edison Company and General Electric Company; Vallecitos, a BWR experimental reactor by General Electric Company; Yankee Rowe, a PWR by Yankee Atomic Electric Company and Westinghouse; Hallam, a sodium graphite reactor by Consumers Public Power District of Nebraska; Fermi, a fast breeder reactor by Power Reactor Development Company; Elk River, a closed cycle BWR by Allis Chalmers; Piqua, an organic-cooled and moderated reactor by Atomics International. In 1956, the AEC issued construction permits for Indian Point, a PWR by Consolidated Edison, and Dresden, a BWR by Commonwealth Edison. There were even more in the following year. See Remppe (2021b) for a more complete discussion of the government/industry cooperative programs.

Regulatory process

As the AEC embarked on developing a system of regulation, a 1955 statement by the acting Chairman of the Commission provided a safety philosophy that was marvelously prescient (Libby, 1956): “the ultimate safety of the public is dependent upon three factors:

1. Recognizing all possible accidents which could release unsafe amounts of radioactive materials;
2. Designing and operating the reactor in such a way that the probability of such accident is reduced to an acceptable minimum;
3. By appropriate combination of containment and isolation, protecting the public from the consequences of such an accident, should it occur.”

The evolving regulatory process can be seen as a series of attempts to meet these goals, first by creatively applying engineering judgment, then by attempts to bound each criterion, and eventually finding a way to quantify each of them. The last approach has become known as probabilistic risk assessment (PRA) and it provides quantification of the ‘risk triplet’ (Kaplan and Garrick, 1981):

1. What can go wrong?
2. How likely is it that it will happen?
3. If it does happen, what are the consequences?

which is remarkable similar to the three factors described by Libby.

The evolution of regulatory practice at NRC has much to do with developing and refining approximations to Libby’s three criteria, even when many of those involved would not have expressed it that way. There were no regulations in the late 1940s and early 1950s, and the AEC had a very small staff. During that time, the AEC relied heavily on the expertise of the Reactor Safeguards Committee (later the ACRS). The physicist, Edward Teller, who chaired the Safeguards Committee for 6 years, pointed out that the historic industrial practice of trial and error was not sufficient for nuclear reactors, because of the nature and extent of possible consequences of an accident. He described a ‘simple procedure’ the Committee had developed to structure its reviews of safety (Teller and Brown, 1962): for each reactor, they asked the designers to “imagine the worst possible accident and to design safety apparatus guaranteeing that it could not happen... The committee reviewed each reactor plan, trying to imagine an accident even worse than that conceived by the planner. If we could think of a plausible mishap worse than any discussed by the planner, his analysis of the potential dangers was considered inadequate. In most cases, the required discussion created a reasonable spirit of caution, and we could advise the Atomic Energy Commission that the reactor would be sufficiently safe” (Mazuzan and Walker, 1984). The Safeguards Committee was exercising their judgment to approximate Libby’s first and third principles.

The AEC institutionalized and systematized this ‘simple procedure’ in the early 1950s by requiring a hazards-summary report (later renamed the safety analysis report) on each planned facility. “The report had to include a description of the reactor and the site, a detailed plan of operation, a schedule of chemical processing and disposal of reactor fission products, the methods of disposal of radioactive effluents, and a description of the safety mechanisms of the reactor. More specific information was required on potential hazards, incorporating the data used in the safety-evaluation procedure described by Teller. The designer had to list all the known potentially hazardous features and include the experimental information, calculations, and assumptions used in evaluating those hazards. The report required information on steps taken to minimize the risks and an estimate, if a failure should occur, on the extent of any release of radioactive material and the damage to be expected... Regarding the site of a reactor, the report required consideration of hydrological data including the expected drainage of liquids in case of a major accident, seismic data including estimates of potential damage that might occur in the event of earthquakes of various intensities, and atmospheric conditions including an assessment of dangers to the surrounding population and industries. In addition, contractors had to provide

information on the distribution of population and list vital industrial, defense, and public-service installations within the possible hazard radius of the facility... This extensive report became a standard document by which the AEC staff and the Safeguard Committee judged the hazards of a reactor" (Mazuzan and Walker, 1984; AEC, 1951). These reports were based on data and analysis to the extent possible; they also relied heavily on engineering judgment, by applicants, AEC staff, and advisors.

The AEC and the Joint Committee negotiated specific kinds of facilities to be licensed: 'Class 103' licenses (named for Section 103 of the 1954 Atomic Energy Act) for commercially viable facilities and 'Class 104' licenses for test and research reactors, as well as demonstration power reactors (104b). Early power reactors received Class 104b licenses. By the 1960s commercial nuclear power was established and applications for large power reactors were coming to the AEC; however, Class 104b licensing continued. The AEC argued that it could not judge if reactors were commercially viable until long-term operating experience was available. Congress disagreed and in 1970 amended the Atomic Energy Act of 1954, removing the requirement from Section 102 that a reactor must be sufficiently developed to be of practical value, before issuing a Class 103 license. It required licenses issued to commercial facilities to be issued under Section 103, unless already licensed under 104b or developed under the Cooperative Power Reactor Demonstration Program (Walker, 1992).

The institutional milieu again

By 1954 the application review load was becoming more than the part-time Safeguards Committee could support. In 1955, the AEC established the Reactor Hazard Evaluation Staff, which was to take on more of the application review process, with oversight by the Safeguards Committee. Over the years, as AEC staff grew, the agency passed through several reorganizations. By this time, quantitative criteria for radiation exposure to workers and the general public had been established by the NCRP. During 1955 the AEC published their own proposed radiation protection standards and have revised them several times since, based on new information and recommendations by NCRP.

Through a series of efforts, the agency and the Safeguards Committee negotiated issues of reactor siting, hypothetical accident source terms (the radionuclide content, release locations, and energy content of possible releases), the probability of a major accident, and emergency planning. In parallel, there evolved a process of formalizing accident analyses, improving quantification, and addressing uncertainty. AEC and later NRC regulations for civilian facilities are tabulated in Title 10 CFR Parts 1-199 of the Code of Federal Regulations (GPO, 2019, 2020).

Siting and source terms

The first attempt to address siting issues was the Safeguards Committee's report (AEC, 1950) that recommended for AEC reactors that the general public be excluded from a circular area of *adius (in miles)* = $0.01 \sqrt{\text{power in kw}}$, based on calculations of maximum possible releases. As the AEC began to consider power reactors for electrical distribution, this simple isolation criterion became impractical (for a 100 kw research reactor the exclusion radius would be 0.1 miles, but for a 3000 mw power reactor, the radius would be over 17 miles). For economic electric power distribution, reactors would need to be much closer to population centers. Therefore, focus began to shift from isolation to engineered safeguards to prevent and mitigate possible accidents. For example, emergency core cooling systems were developed to reduce the likelihood of the release of fission products from the core and containments were designed to block any release from the plant, should the core be damaged. The first nuclear power plant containment was built at West Milton, NY, in the early 1950s and it set a precedent that has been followed by all light-water reactors (LWRs) in the U.S. All have included some type of containment system and physical structure, designed to withstand the most severe over-pressure events identified for the specific reactor design (Okrent, 1981; Kerr, 1989; ACRS, 2018).

Siting requirements and evaluation of potential source terms evolved from the bounding approach of WASH-3, to the siting criteria of 10 CFR 100, which established the following distance definitions for commercial reactors:

1. An 'exclusion area' completely controlled by the licensee, such that an individual located anywhere on its boundary for 2 hours following a release would not receive a dose to the whole body in excess of 25 rem or to the thyroid of 300 rem
2. A 'low population zone,' such that an individual located anywhere on its boundary for the entire period of the release would not receive a dose to the whole body in excess of 25 rem or to the thyroid of 300 rem
3. A 'population center distance' of at least $1 \frac{1}{3}$ times the distance from the reactor to the outer boundary of the low population zone.

Calculation of these siting distances was based on conservative (in the sense that calculated required distances are greater than if more accurate calculations could be made) assumptions and approximations of the physical and chemical phenomena that control the release and dispersion of fission products, as well as the dose received by individuals at various distances from the plant (DiNunno et al., 1962). The authors of this report, TID-14840, make the point that consideration of the consequences of an accident cannot be considered independent from the probability of the accident. Their recommendations are based on stylized scenarios for the types of accidents they considered. They are not offered as acceptable doses in general, but as reasonable reference values for well-designed plants with very low probability of a severe accident. Their results became known as the 'TID source term' and remain the acceptable basis for evaluating radiological consequences of reactors whose owners applied for licenses before January 1997 and choose to continue its use (US NRC, 2003a,b).

Plants that applied for licenses after January 1997 (and older plants that choose to change their licensing basis) use more recent “alternative source terms” (Soffer et al., 1995) based on new information and experiments providing a more thorough understanding of the timing, magnitude, and chemical form of fission product releases from reactor accidents. Use of the alternative source terms changes the dose criteria for the exclusion area and low population zone, from separate whole body and thyroid doses to 25 rem total effective dose equivalent (TEDE) (US NRC, 2000).

The institutional milieu again

Let us return to Libby’s three criteria. The AEC has by now addressed the issue of recognizing all possible accidents primarily by engineering judgment. As for designing and operating the reactor in such a way that the probability of an accident is reduced to an acceptable minimum, qualitative approximations for ensuring high reliability have been developed, such as the single failure criterion, which requires that any engineered safety system must be able to carry out its safety function despite of the failure of any single component within the system or within an associated support system (US NRC, 2003a, 1977b).

The industry has made great strides in designing emergency core cooling systems that can respond to the accidents judged most important, and containment systems to withstand anticipated overpressure events. Coming to grips quantitatively with the probability of a severe accident and understanding containment capability for a range of core damage accidents will take some years. Finally, a balance between isolation and engineered safety features has been addressed in an approximate, but conservative way.

Technical specifications, general design criteria, and backfits

The AEC (and later the NRC) published an extensive catalog of regulations. However, demonstrating that an application met these regulations became cumbersome. Even modest design changes led to new rounds of hearings. The AEC staff looked for ways to set clear criteria that applicants and licensees could apply unambiguously. The first approach, developed during the period 1962–68, was to publish a set of technical specifications. They were to define what technical information was vital for evaluating nuclear safety and provide an acceptable range of operating limits. If the plant remained within those limits, a licensee could modify the design or operation of the plant without AEC approval. Specific criteria were provided describing remedies should the plant not maintain operations within the technical specifications. Another problem had developed—applicants expressed uncertainty about the depth of technical information needed to support a construction permit—that led to delay following an Atomic Safety and Licensing Board hearing and review. It appeared that a set of design criteria could expedite the licensing process and make it more predictable (AEC, 1965). By 1965 the AEC staff proposed a set of design criteria to the Commission and the Safeguards Committee. Applicants would need to demonstrate that their design would meet these broad criteria. After several years of negotiation with ACRS and stakeholders from industry, AEC issued the ‘general design criteria’ (GDC) as Appendix A to 10 CFR Part 50 in 1971. The criteria told applicants what was needed, not how to accomplish it.

During this same time period, the issue of ‘backfitting’ became contentious. Essentially, a backfit occurs when the NRC changes its mind about the safety of a licensed facility and wants a plant to add additional protection; i.e., make modifications to installed equipment. This became troublesome when applicants and the AEC disagreed about newly raised issues after work under a construction permit was complete and the application for an operating license was being reviewed. Industry was concerned with the high costs of backfitting; ACRS was concerned that important safety issues might not be addressed if backfitting were proscribed; and the AEC, still having promotional responsibilities, was concerned with the economic attractiveness of nuclear generation compared to coal. The AEC staff had been addressing this issue on a case by case basis. In 1970, the Commission adopted an amendment to 10 CFR 50 that required backfitting only when it provided substantial, additional protection. The burden to show substantial safety improvement was on the AEC staff (AEC, 1969; Okrent, 1981). The new general design criteria, discussed in the previous paragraph, helped avoid the backfitting controversy: once an applicant showed that they would meet the GDC, the likelihood of the AEC requiring a backfit following construction turned out to be much lower.

For many years the rule (10 CFR Part 50.109) provided simply that backfitting may be required if the Commission finds that “such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security.” On its face, this was a high standard, implied by the use of “substantial, additional protection” that is “required.” In practice, the rule was rarely invoked, and it failed to address some key issues, including whether economic costs to be incurred by the licensee (and electricity consumers) in accomplishing a backfit could be a relevant factor in deciding whether the backfit was “necessary.” In 1983 the NRC began a rulemaking to address these and other issues and, in 1988, issued a new backfit rule that remains essentially the same today. The 1988 rule embodies a two-step safety philosophy as follows. The Commission must assure an “adequate” level of protection regardless of economic costs, as required by Section 182 of the Atomic Energy Act and will order a backfit to a design (or a procedure) to accomplish this. However, once an adequate level of safety protection is achieved, the Commission may under Section 161i of the Atomic Energy Act consider and take economic costs into account in ordering further safety improvements (backfits). It was soon recognized that probabilistic risk assessments can play an important role both in defining what level of protection is “adequate,” and in deciding whether a proposed backfit adds “substantial additional protection.”

Another issue was a long-running problem for the AEC and NRC, the growing list of ‘unresolved safety issues’ and ‘generic safety issues’ that were identified during license application reviews for specific plants. Such issues were often put aside for generic resolution, because they applied to a class of plants, not just the current application. How this problem was eventually resolved is discussed in Bley and Malsch (2021).

Guidance for meeting the regulations

Starting in the early 1970s, the AEC published a variety of guidance documents supporting its regulations. The Standard Review Plan (SRP) (US NRC, 2007) provides guidance to the regulatory staff for reviewing applications. Also, an extensive set of Regulatory Guides provides applicants with approved methods and approaches for meeting the regulations. Two of the Regulatory Guides (US NRC, 2018b, 1978) provide guidance for the content of applications and safety analysis reports respectively.

The earliest of these Regulatory Guides provided acceptable methods for addressing many of the issues that had concerned the ACRS and the regulatory staff, including some that led to the GDC; e.g., net positive suction head for engineered safeguards pumps, thermal shock to reactor pressure vessels, evaluating radiological consequences following accidents, control of combustible gas concentrations, independence between electric power sources, periodic testing, quality assurance, seismic design, and the single failure criterion.

The identification and review of potential operational events—transients and accidents—became more systematic as well. The SRP Chapter 15 (US NRC, 2007) now categorizes events by both their frequency of occurrence and by ‘type’ based on their effect on the plant. The frequency categorization includes:

- Anticipated Operational Occurrences (AOO), which are events expected to occur one or more times in the lifetime of the plant.
- Accidents, which are postulated, but not expected to occur in the lifetime of the plant.

The AEC/NRC has different criteria for evaluating AOOs and accidents. Other earlier guidance (US NRC, 2018b, 1978) separates AOOs into events of moderate frequency (i.e., they are expected to occur several times over the plant lifetime—also called Condition II events in ANS standards) and infrequent events (i.e., they may occur in the plant lifetime—Condition III events). Postulated accidents are called Condition IV events.

Design-specific specializations of AOOs are addressed in separate sections of the SRP. Very rare accidents with frequencies less than approximately 1×10^{-5} per year are called beyond-design-basis events and are not analyzed in the safety analysis report (US NRC, 2007; Apostolakis et al., 2012).

Operational events are categorized by the following seven types, based on departures from steady-state operation:

- Increase in heat removal by the secondary system (e.g., steam line break)
- Decrease in heat removal by the secondary system (e.g., turbine trip)
- Decrease in reactor coolant system flow rate (e.g., reactor coolant pump trip)
- Reactivity and power distribution anomalies (e.g., control rod drop)
- Increase in reactor coolant inventory (e.g., inadvertent safety injection)
- Decrease in reactor coolant inventory (e.g., loss of coolant accident)
- Radioactive release from a system or component (e.g., unisolated steam generator tube rupture)

This type of categorization can assist the applicant in conducting a thorough search for possible AOOs and accidents. Within each type, the applicant must identify events that challenge diverse operating limits; e.g., challenges to over-pressure or thermal margin.

The likelihood of an accident

The importance of the probability of an accident was discussed during the development of source terms and is more directly considered in the treatment of operational events. Some of the concepts here have evolved with expanded use of PRA. The idea of risk is imbedded in the GDC. They are structured to prevent relatively frequent events from generating serious consequences, while unlikely events are allowed to incur more severe consequences. Specifically, AOOs should not challenge the integrity of the reactor coolant system, main steam piping, or fuel cladding. A postulated accident should not result in off-site doses in excess of the guidelines of 10 CFR Part 100, but there could be substantial damage to the plant. There are also a number of analysis assumptions that must be used in the safety analysis, as well as requirements on calculated results.

The design basis accidents to be analyzed in the reactor safety analysis were not intended to be specific event sequences as would be analyzed in a PRA. They were intended to be stylized composites or surrogates to allow deterministic bounding (i.e., conservative) evaluation of the reactor’s engineered safety features—will they fulfill their functions under a variety of challenges, with margin. These analyses should ensure that the level of defense-in-depth can compensate for uncertainties in accident progression and analysis data (US NRC, 2000, 2003b; Drouin et al., 2016). The analyses are characterized by the use of prescribed and conservative methods and data, and, for each credited protection or safety system, by the assumption of the most limiting single active failure.

This approach has served the regulator and the U.S. industry well. However, it fails to address the question of the probability of a serious accident. In 1965, Glenn Seaborg, Chairman of the AEC, sent a letter to the Joint Committee stressing the low probability of a catastrophic accident, but acknowledging that the consequences could be severe (Seaborg, 1965). Unfortunately, at the time, we still had no convincing way to calculate the probability of a severe accident and our consequence calculations compounded many strongly conservative and sometimes conflicting assumptions. Following 1965, there were many unsuccessful attempts to solve the problem. Among the most important issues were the difficulty of imagining all the possible paths to failure and the need for practical mathematical methods to calculate the reliability of complex systems. In 1972, the AEC commissioned the Reactor Safety Study (Rasmussen and Levine, 1975) to quantify the probability of a severe reactor accident. The study was led by Prof. Norman Rasmussen of MIT and Saul Levine of the AEC staff.

Rasmussen and Levine turned to the aerospace and chemical processing industries for expertise on new methods of reliability analyses (Vessely et al., 1981). One of their intermediate products was a gigantic fault tree with core damage as the 'Top Event'. This tree modeled all the combinations of failures that could lead to the Top Event. It was so big that it covered the ceiling and walls of a large conference room. This made technical review of the work nearly impossible and use of the structure of the model difficult, at best. The team searched for a way to add simplifying structure. Rasmussen and Levine visited the Sloan School at MIT and talked with decision analysts; the decision trees used in that work sparked a breakthrough (Bley, 1975). Rasmussen and Levine imagined a similar tree structure, one they called an event tree on critical safety functions. Each event tree would begin with an 'initiating event' followed by a series of questions about the success or failure of the safety functions that could prevent fuel damage or mitigate it. Paths through these trees became known as event sequences or scenarios and represented, usually in proper time order, how the plant sequentially responds to the initiating event. This restructuring provided significant advantages:

- The overall PRA model could be more easily reviewed, and its results interpreted
 - It is now organized into digestible pieces
 - Initiating events are separated from the response of engineered systems to those events
 - Interactions among plant systems are clearly displayed
- The analysis was organized similar to how the plant responds, which
 - Allows specialization of thermal-hydraulic and neutronic response to each specific pathway (or scenario) through the analysis; importantly, when one tries to layout all possible scenarios, each leading to different levels of challenge, many more engineering calculations are required than for more simple bounding approximations.
 - Makes clear the path dependence of safety system, physical barrier, and human performance
 - Provides better understanding of the importance of systems, components, and operators
 - Allows coupling of similar plant scenarios with phenomena that can lead to core damage and fission product release

The Reactor Safety Study was a massive undertaking involving something like 70 staff years by AEC staff and contractors, as well as a like amount donated by vendors, licensees, and interested parties (Bley, 1975). Much of this effort occurred because it was a first of a kind study that involved the development of new methods, many trial approaches, and substantial manual effort. In the years that followed the Reactor Safety Study methods were extended and refined, and computer codes were developed to automate much of the manual effort. In addition to providing the first convincing estimate of the risk from commercial nuclear power plants, important insights included the fact that the risk was higher than previously believed, several non-safety grade systems useful when secondary heat removal is lost were shown to be important to risk, loss of electric power was especially important because it affected nearly all plant systems, a number of failure sets previously described as common-cause failures were identified as support system to front line system dependencies that could be modeled explicitly. The study marked the first time the chain of events following core damage were explicitly modeled to estimate the likelihood of various kinds of release from the containment. Thus, given a core melt, some pathways lead to containment failure and release, some bypass the containment leading to release, and some lead to an intact containment and no substantial release. For further insights from the Reactor Safety Study, see Fleming (2021). Despite its successes, a variety of criticisms and unfavorable press releases led to NRC withdrawing support for PRA. It was not until the Three Mile Island Unit 2 (TMI-2) accident (Walker, 2004) and reviews of that accident (Kemeny et al., 1979; Rogovin et al., 1979), which chastised NRC for not using a tool that could have helped avoid the accident, that NRC returned to PRA. In the years that followed, the Commission issued a policy statement (US NRC, 1995) supporting 'risk-informed' regulation, PRA methods were further improved and extended, a consensus PRA standard was issued to assure PRA quality (ASME/ANS, 2009), and NRC issued guidance documents on PRA quality (US NRC, 2009).

PRA—Probability and consequences

PRA (Modarres, 2021) takes a different tack from that developed by AEC over the 1950–70 timeframe. Rather than a representative set of stylized scenarios, analyzed 'conservatively' to test the margin of safety systems, PRA attempts to model the plant realistically. Recall the risk triplet. PRA attempts to model all scenarios that could lead to core damage and release. It acknowledges that some single failures can be more likely than double failures and models all the interactions among system components, allowing their individual reliabilities to be used to calculate the overall system reliability. The supporting engineering calculations (e.g., thermal-hydraulics) are performed on a best-estimate basis rather than a conservative one. However, they must account for uncertainties in data and modeling.

One of the first industry-sponsored PRAs, the Indian Point Probabilistic Safety Study (Pickard, Lowe, and Garrick, Inc. et al., 1982) was instrumental in resolution of the Indian Point ASLB hearings on the risk from that plant (Palladino et al., 1985). Having been designed and built in the 1960s, these units had not been designed to withstand external events (such as earthquakes and fires) to the same degree as more modern plants. During the course of the PRA, improvements were proposed to reduce risks in three general areas—earthquakes, fires, and hurricanes. They included:

- The installation of a bumper between the control building and the superheater building of Unit 2 to dampen interactions in the event of an earthquake.
- The addition of alternate power cables for the component cooling water and charging pumps at Units 2 and 3 to increase the reliability of these pumps in case of fires.
- A modification of the technical specifications for the operation of Unit 2 to require plant shutdown in the event of a hurricane.

These improvements significantly reduced the likelihood of core melt at the Indian Point Units 2 and 3, and their cost effectiveness was recognized by the NRC as well as by the plant licensees. In a meeting of the U. S. Nuclear Regulatory Commission ([US NRC, 1984](#)), Frank Rowsome, Assistant Director for Technology of the NRC, stated that:

The [pre-PRA] mitigation conceptions, both those that were subject to contention [in the ASLB hearings on the safety of the plants] and those that had been developed in the [NRC] staff action plan...cost more than they were worth... The prevention improvements volunteered by the licensees were worth roughly a quarter of a billion dollars in averted risk, a whopping big value. Their cost ...was of the order of \$1 million or thereabouts, for about 100 to 1 value to cost ratio. In fact, if you count in the whole cost of the hearing and the inquiry and the PRAs and everything in the balance, the value accorded the fixes we found was about ten times the cost of the entire enterprise.

Over the next 20 years PRA models were developed for most plants by both the owners and the NRC. These models were used to identify risk outliers and improve them, optimize technical specifications, and to address specific regulatory issues. By 1986 the U.S. Department of Energy ([DOE, 1986](#)) submitted an application for the General Atomics modular high temperature gas-cooled reactor (MHTGR) with a safety case based on extensive use of its PRA ([Williams et al., 1989](#)). Some years later, NRC developed a similar approach known initially as the technology neutral framework ([Drouin et al., 2007](#); [ACRS, 2007](#)) that included parallel considerations of defense-in-depth as a means to protect against uncertainties in the PRA model.

Defense-in-depth

A paper, "On the Role of Defense in Depth in Risk-Informed Regulation," was issued earlier as an attachment to an ACRS letter report ([ACRS, 1999](#)). The letter stated that improved capability to analyze nuclear power plants as integrated systems is leading us to reconsider the role of defense-in-depth. Two different perceptions of defense-in-depth were identified. In one view (the "structuralist" view), defense-in-depth is considered to be the application of multiple and redundant measures to identify, prevent, or mitigate accidents to such a degree that the design meets the safety objectives. Until recently, this has been the general view taken by plant designers. The other view (the "rationalist" view) sees the proper role of defense-in-depth in a risk-informed regulatory scheme as compensation for inadequacies, incompleteness, and omissions of risk analyses.

As stated in the paper, the structuralist and rationalist models need not be in conflict. Both can be construed as a means of dealing with uncertainty. In the final analysis, they both depend on knowledgeable people discussing the risks and uncertainties, and ultimately agreeing on the provisions that must be made in the name of defense-in-depth.

The difference is that the structuralist model accepts defense-in-depth as the fundamental value, while the rationalist model would place defense-in-depth in a subsidiary role. Adopting a high-level structuralist view and a low-level rationalist view is a pragmatic approach to reconciling defense-in-depth with risk-informed regulation. There can be little doubt, however, that the rationalist model will ultimately provide the strongest theoretical foundation for risk-informed regulation. As more experience has been gained with the application of PRA in the design and regulation of nuclear power plants, when PRA models can adequately treat most of the phenomena of interest, the role of defense-in-depth can and should be changed to one of supporting the risk analyses. This transition will need to be supported by the development of subsidiary principles from which necessary and sufficient criteria could be derived.

At the present time, the traditional approach to licensing remains and applicants are free to temper it with risk-informed applications. However, a new generation of reactors is being developed including small modular reactors, micro-reactors, passive designs, and simplified designs. For many of these a simpler approach to licensing might be possible. NRC staff has been working with stakeholders to develop a vision and strategy for licensing the new designs. One key aspect is an extension of the MHTGR and NUREG-1860 approach laid out in an industry document that NRC may adopt ([NEI, 2019](#); [US NRC, 2018a](#); [ACRS, 2019](#)). This technology-inclusive, risk-informed, and performance-based methodology for informing the licensing basis and content of applications for non-light-water reactors marks the next evolution of a licensing approach that has been developed over the past 30 years. Its use is confined to three objectives: to select licensing basis events; to classify structures, systems, and components; and to assess the adequacy of defense-in-depth for new designs. The licensing basis events include AOs, design basis events, and beyond design basis events, all now defined unambiguously by the PRA frequency results. For classifying structures, systems and components, the paper extends and makes operational the concepts that were expressed earlier in NUREG-1860 ([Drouin et al., 2007](#)). They are selected from important risk contributors in the PRA, with special treatment assigned based on importance to risk. For defense-in-depth, the paper provides an operational structure for evaluation. It bridges the gap between frameworks as described in NUREG-1860 and NUREG/KM-0009 ([Drouin et al., 2016](#)), and viable regulatory actions. As discussed in [Fleming \(2021\)](#), the report ([NEI, 2019](#)) describes a defense-in-depth approach that includes probabilistic and deterministic assessment techniques using a combination of plant capabilities and programmatic controls. As part of the evaluation, each licensing basis event is evaluated to confirm that risk targets are met without exclusive reliance on a single element of design, single program, or single defense-in-depth attribute. One of the primary motivations of employing defense-in-depth attributes is to address uncertainties, including those that are reflected in the PRA estimates of frequency and consequence, as well as other uncertainties which are not sufficiently characterized for uncertainty quantification and are not amenable to sensitivity analyses. The designer is to convene an Integrated Decision Panel to make judgments involving quantitative and qualitative factors.

The effect of accidents on regulatory process

We consider three accidents at nuclear power plants that have had substantial impact on the design, operation, and regulation of nuclear plants in the U.S.: the Browns Ferry Fire in 1975 ([Gonzalez et al., 2014](#)), the Three Mile Island 2 accident in 1979, ([Walker, 2004](#); [Skillman and Rempe, 2021](#)), and the Fukushima earthquake and tsunami 2011 ([Miller et al., 2011](#); [Mizokami and Rempe, 2021](#)). As a result of these accidents, the NRC dramatically revised regulations and industry refined operating practices—e.g., fire protection requirements, the development of symptom-based procedures, a requirement for on-site simulators, and establishment of the Institute for Nuclear Power Operations (INPO). Following the Three Mile Island accident, there was a long delay in licensing plants in the regulatory review pipeline that burdened several utility companies with massive debt. The net effect of the response of the industry and NRC to these accidents has been a substantial improvement in the safety of our nuclear power plants. Additional accidents that impacted reactor design, operation, and regulation are described in other chapters ([Rempe, 2021a](#); [Sich, 2021](#)).

The Browns Ferry fire

The Browns Ferry fire ([Gonzalez et al., 2014](#); [Rempe, 2021a](#)) started on March 22, 1975 by a worker using a lit candle to check for air leaks around cable seals. It propagated quickly then continued to burn for many hours because plant personnel did not want to put water on an electrical fire. Before it ended, many of the engineered safety features were disabled. NRC published recommendations ([US NRC, 1976](#)), new guidance ([US NRC, 1977a](#)), and a new regulation, 10 CFR 50.48(b) in 1980. The approach was deterministic and tried to ensure safety margin to fire events ([US NRC, 1981](#)). However, many plants had to seek temporary exemptions because of specific design issues. Some of the fire-related issues were not resolved until recently as part of the Alternate Fire Protection Rule 10 CFR 50.48(c) which incorporates the National Fire Protection Association (NFPA) Standard NFPA 805 ([NFPA, 2001](#)) by reference, a risk-informed, performance-based standard.

The Three Mile Island accident

The Three Mile Island-2 PWR accident ([Walker, 2004](#)) began at 4:00 am on March 28, 1979 as a simple secondary system transient when pumps in the condensate polishing system tripped. Bypass valves that should have allowed continued feedwater flow failed to open. Due to the loss of a condensate flow path, the main feedwater pumps tripped, as they should. Unfortunately, the remaining system that could have provided feedwater, the auxiliary feedwater system, also failed because two valves in the lines to the steam generators were inexplicably left closed. Now, with no secondary heat removal, pressure in the primary system increased. The reactor tripped, but with no secondary cooling, pressure continued to rise causing a power operated relief valve to open. It failed to reclose, but because the control room indication was based on the demand signal to open the valve, it erroneously indicated closed. Because the valve had previously leaked, the operators failed to see the high downstream temperature as an alternative indication of valve position. So, because of multiple failures, the condition of the plant was masked to the operators, who did not realize that a small loss of coolant accident was in progress. This is exactly the sort of multiple failure situation that is modeled in PRA.

The plant was still in relatively good condition and had built-in capability to survive the accident. Unfortunately, when the pressurizer level dropped below the heaters and the system lost overpressure, the reactor coolant became saturated and a steam bubble developed in the hottest location in the system—at the outlet of the core in the top of the reactor vessel. The pressurizer was no longer an indicator of the volume of water in the system; its level began to rise as the steam bubble in the reactor vessel expanded. Fearful of over-filling the pressurizer and going ‘water solid’ the operators turned off the safety injection system, which was providing makeup water to ensure core cooling. (Actually, going water solid is a dangerous condition, because slight changes in temperature can lead to large pressure variations.) There were additional issues later, but the fate of the plant was already sealed.

Following the accident, there were a great many lessons learned and NRC issued a number of orders and short- and long-term plans. There were a number of reviews of the accident sponsored by the government and industry. Among them, the two most far-reaching were the President’s Commission Report ([Kemeny et al., 1979](#)) and the Rogovin Report ([Rogovin et al., 1979](#)) commissioned by the NRC. Both reports drew a similar overall conclusion: the primary problems were not with the equipment, but with management; fundamental changes are needed in the organization, procedures, practices, and attitudes of the industry and the NRC. Quoting the Kemeny Report:

We are convinced that if the only problems were equipment problems, this Presidential Commission would never have been created. The equipment was sufficiently good that, except for human failures, the major accident at Three Mile Island would have been a minor incident. But, wherever we looked, we found problems with the human beings who operate the plant, with the management that runs the key organization, and with the agency that is charged with assuring the safety of nuclear power plants... In the testimony we received, one word occurred over and over again. That word is “mindset”... they concentrated on equipment, assuming that the presence of operators could only improve the situation – they would not be part of the problem... [T]he belief that nuclear power plants are sufficiently safe grew into a conviction... The Commission is convinced that this attitude must be changed to one that says nuclear power is by its very nature potentially dangerous, and, therefore, one must continually question whether the safeguards already in place are sufficient to prevent major accidents. A comprehensive system is required in which equipment and human beings are treated with equal importance... We note a preoccupation with regulations... [O]nce regulations become as voluminous and complex as those regulations now in place, they can serve as a negative factor in nuclear safety... This Commission believes that it is an absorbing concern with safety that will bring about safety – not just the meeting of narrowly prescribed and complex regulations.

Both reviews found deficiencies in training, particularly with respect to understanding unusual conditions. They also found the emergency operating procedures deficient and confusing. Both the NRC and the industry had focused almost entirely on design, with little focus and little expertise in operations. The lessons from previous accidents did not result in new, clear instructions being passed on to the operators; there were precursors to this accident, but the operators and many others were unaware of them. Generic safety issues had been open for years. The lessons of WASH-1400 were forgotten and the NRC should put more emphasis on risk, by emphasizing those accident sequences that contribute significantly to risk. The design review can then focus on those systems that contribute to risk, identify weak points, and make improvements to eliminate them.

There were an enormous number of recommendations in the two reports. Rather than focus on all of those, here is a list of some of the many changes that came about in response to the accident:

- Symptom-based procedures organized to emphasize the most risk important response first. Earlier procedures were based on high level response to very specific events and did not help in complex situations or those that deviated from the stylized scenarios of the safety analysis.
- Improved monitoring of key accident parameters such as reactor water level, development of the safety monitor to support symptom-based procedures and grouping displays to better support operator needs.
- The establishment of realistic simulators at every plant. The models were improved to included details that operators would see, even if those were not crucial in safety analysis.
- Requiring a graduate engineer on shift as the Shift Technical Advisor, to provide oversight and advise the operating crew of unanticipated safety issues.
- The establishment of NRC resident inspectors at all plants.
- The establishment of INPO. It fulfilled many recommendations: the industry needs to take more ownership of safety, the industry should establish a training institute, and the industry should track, evaluate, and share operational experience.
- The establishment of a Technical Support Center separate from the control room both to minimize traffic in the control room and to provide guidance during unusual situations, and communications with vendors, regulators, and other experts.
- The establishment of the Office of Analysis and Evaluation of Operational Data at NRC to review operating experience and share important observations throughout the NRC.

Finally, quoting the Rogovin report's balancing of their criticism:

In an investigation like this, the very purpose of which is to focus on what went wrong and what needs changing, it is inevitable that less attention than is deserved will be given to what "went right"-the strong points in the system. Chief among these is the fact that the "defense in depth" concept worked to protect public health and safety. In spite of multiple equipment malfunctions, human failures, and the creation of conditions in the reactor and auxiliary buildings that were never contemplated in the design of the plant's safety systems, the utility and its engineering support staff were able to bring the system to a stable condition without releases of radioactive materials to the atmosphere that could have resulted in significant health effects to those living near the plant.

Thereafter, a massive response by industry, the national laboratories, and government greatly assisted the plant operator, Met Ed, both in bringing the plant to a safe shutdown and in establishing recovery operations. Over a thousand people, from reactor operators and health physics technicians to top executives from every corner of the industry, dropped their everyday work and went to the TMI site. Thousands more were active in performing supporting analyses and experiments, and in procuring and dispatching needed supplies.

The Fukushima accident

The Fukushima accident (Neureiter et al., 2014) occurred on March 11, 2011 when the Great East Japan Earthquake was followed by a massive tsunami that damaged the Dai-ichi nuclear power plant and wreaked havoc on Japan destroying property over a wide area and killing more than 15,000 people. At Dai-ichi, the operating Units 1, 2, and 3 were responding to a loss of offsite power and associated reactor trips, when the tsunami rolled over the site flooding some areas of Units 1, 2, 3 and 4 (Unit 4 was in an outage with all its fuel removed, so it required no cooling). The most important damage was the flooding of lower areas of the plants including the diesel generator and battery rooms causing a loss of all AC electrical power and leading to the loss of DC power, almost immediately in Units 1 and 2, and after about a day and a half in unit 3. The three operating reactors were BWR designs. Unit 1 included two passive emergency heat exchangers called the isolation condenser system. The shell side of each condenser could be replenished from outside the plant. Varied reasons for their failure and the inability of operators to restore isolation condenser cooling have been suggested in different accounts of the accident. Unit 2 had a turbine driven cooling system that was able to run for about 3 days. Unit 3 had two turbine driven cooling systems that continued to function for about a day and a half. At each operating unit, the loss of all core cooling led to fuel damage and subsequent release of fission radioactive materials and hydrogen. After fuel damage, it was important to vent the containment before catastrophic failure on over-pressure. A number of problems delayed venting. Units 1, 3, and 4 suffered structural damage from hydrogen explosions.

Working conditions were extremely difficult due to high temperatures, high radiation fields, lack of lighting, the need for bulky personnel protective equipment, and the failure of communication devices. Additionally, for weeks there were inadequate accommodations for sleeping, eating, and bathrooms, and workers could not communicate with family members to ascertain their safety following the tsunami. In these conditions, operators jury-rigged ways to inject water to cool damaged cores and limit the off-site health effects. As described below, numerous reports provide additional details of the accident progression at each unit.

Following the accident there were numerous investigation reports by the Japanese utility company, the Japanese government, and international government, academic, and industrial organizations. A long list is provided in the U.S. National Academy report (Neureiter et al., 2014). There were also articles and books by individuals who were on-site during the accident. We will focus here on the response in the U.S., by industry, NRC, and interested parties. The first report was by the ‘Near-Term Task Force’ (NTTF) established by the NRC to investigate implications for regulation in the U.S. They were given 90 days to complete their review and provide recommendations to the NRC. Their report (Miller et al., 2011) surprised many people by its thoroughness and the strength of its recommendations. After review of that report and discussions with the NRC staff, the ACRS (ACRS, 2011a) reported that,

While complete understanding of the Fukushima Dai-ichi accident will take many years, the NTTF Report and the staff’s recommended actions to be taken without delay are appropriately focused on lessons learned from what is currently known. We believe that none of the recommendations enumerated...will be negated, or rendered inappropriate, by the acquisition of new information. Hence, timely initiation of the staff’s recommended actions to be taken without delay, along with corresponding additions or modifications included herein, is appropriate.

The ACRS recommended specific expansion of a few of the NTTF recommendations. The staff described a three-tiered approach for addressing these issues, because it was not possible to do them all at once. They set priorities based on risk significance, the need for additional assessment, available resources and critical skills, dependence on resolution of higher tier results (ACRS, 2011b).

Over the next decade, NRC and industry took many actions that should help plants survive very rare events. The enhancements included: adding capabilities to maintain key plant safety functions following a large-scale natural disaster; updating evaluations on the potential impact from seismic and flooding events; providing new equipment to better handle potential reactor core damage events; and strengthening emergency preparedness capabilities. Combined, these actions enhance the likelihood that the nuclear industry and the NRC are prepared for the unexpected. Among the most important of these is the industry FLEX approach to provide backup equipment at plants and in regional supply centers, with rapid delivery capability to any plant in the nation, called SAFER. This equipment is provided with rapid hookup capability, now also installed in all plants and a set of mitigation strategies for using the equipment. Many details for these improvements can be found on the NRC’s public website (nrc.gov), by searching for Post-Fukushima Safety Enhancements there.

Summary

This chapter has described problems associated with developing regulations for a new industry. As the AEC embarked on developing a system of regulation, acting Chairman Libby provided a safety philosophy based on three goals—recognize all possible accidents, minimize the probability of an accident, and use a combination of containment and isolation to protect the public from the consequences of such an accident (Libby, 1956). The evolving regulatory process can be seen as a series of attempts to meet these goals, first by creatively applying engineering judgment, then by attempts to bound each criterion, and eventually finding a way to quantify each of them.

By the early 1970s, the AEC had established an extensive catalog of regulations and it began to publish a variety of guidance documents supporting those regulations. The Standard Review Plan provides guidance to the regulatory staff on conducting review of applications and an extensive set of Regulatory Guides provides applicants with approved methods and approaches for meeting the regulations.

In 1972, the AEC commissioned the Reactor Safety Study to estimate the probability of a severe reactor accident, something that had eluded its scientists and contractors for years. With its publication the Commission had, in principle, the tools to implement risk-informed regulation that accounted for both the probability and consequences of accidents. It took many years to gain confidence in this approach and, in 2020, the agency is on the cusp of making more extensive use of this approach (Bley and Malsch, 2021).

Regulations also evolved in response to operational events, including accidents. Three accidents discussed in this chapter had far-reaching impact on regulations and operating practice in the U.S. industry.

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U.S. Regulations for Commercial Nuclear Power Plants

Dennis C. Bley^a and Marty Malsch^b, ^a Buttonwood Consulting, Inc., Albuquerque, NM, United States; and ^b Egan, Fitzpatrick, Malsch & Lawrence PLLC, Washington, DC, United States

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Glossary

Advisory Committee on Reactor Safeguards (ACRS) An advisory committee established by statute to advise the US Nuclear Regulatory Commission (formerly the Atomic Energy Commission) on matters of nuclear safety.

Backfitting A backfit occurs when the NRC changes its mind about the safety of a licensed facility and wants a plant to add additional protection; i.e., make modifications to installed equipment.

Code of Federal Regulations (CFR) Regulations promulgated by US government agencies. Commercial nuclear energy facilities are governed by 10 CFR Parts 1–199 promulgated by the US NRC.

General design criteria A set of broad plant design criteria defined in Appendix A to 10 CFR Part 50. The intent was to define criteria such that, if applicants could demonstrate that their design would meet them, it could expedite the licensing process and make it more predictable.

Generic safety issue Safety issues that arise during license application reviews by NRC or from other sources, such as accidents and requests from the public, that apply to all reactors or to a broad class of reactors and are not appropriate to resolve as part of a particular license application process.

Probabilistic risk analysis (PRA) An approach to safety analysis that identifies what can go wrong and quantifies the likelihood and consequences of each scenario. Risk of a facility is presented in terms of the probability of various consequences, including rankings of contributors to the risk and a formal treatment of uncertainty.

Regulatory guides Guidance documents from the NRC that provide applicants with acceptable ways to meet the regulations.

Rulemaking The process by which new regulations are introduced and processed by NRC, including being issued as formal changes or additions to 10 CFR Parts 1–199. The process is governed by NRC Management Directive 6.3.

US Nuclear Regulatory Commission (NRC) The body established by the Atomic Energy Act, as amended, to regulate civilian applications of nuclear power in the United States (formerly the Atomic Energy Commission).

Introduction

US regulations for commercial nuclear facilities are issued by the US Nuclear Regulatory Commission (NRC) and tabulated in Title 10 Parts 1–199 of the US Code of Federal Regulations (GPO 2019, 2020), as discussed in Bley and Malsch (2021). These regulations were initiated under the US Atomic Energy Commission (AEC). Harold Price, an AEC lawyer, was named to head a task force on regulation at the AEC in 1954. The task force laid out a plan to organize all licensing activities in one office, rather than the then current division of the responsibilities among multiple offices. The Commission approved the plan and Price headed up the new office; he would remain the agency's chief regulator until his departure in 1971. His group was devising a scheme of regulation from scratch. They had to decide what kinds of production and utilization facilities would need to be licensed, because the Atomic Energy Act of 1954 left such details up to the AEC. Available guidance indicated that regulation should be as minimal as would be consistent with protecting the health and safety of the public (Bley and Malsch, 2021). Regulations needed to be clear, providing the necessary legal, safety, and technical issues to protect the public, and be sufficiently flexible for a rapidly developing industry—a difficult task. Nevertheless, they issued key rules during 1955 and early 1956. For example, they provided a draft rule for the licensing of reactor operators in 1955 (Mazuzan and Walker, 1984).

This section provides a list of all the relevant parts of the regulations with some explanation of how they were developed.

Organization of Title 10 of the US Code of Federal Regulations (CFR)

The many parts of 10 CFR 1–199 are tabulated in [Table 1](#). The most directly relevant parts for commercial nuclear power reactors are:

- Part 20—Standards for protection against radiation, which describe the requirements for protecting workers and the public, as well as associated administrative requirements.
- Part 50—Domestic licensing of production and utilization facilities and describes the ‘two-step’ licensing process used for most existing plants, i.e., first a Construction Permit, later followed by an Operating License for a specific power plant.
- Part 51—Environmental protection regulations for domestic licensing and related regulatory functions.
- Part 52—Licenses, certifications, and approvals for nuclear power plants, which describes the optional certification and licensing process developed to support hoped for standardization of plant designs. Full implementation of Part 52 would involve design certifications, early site permits (ESPs), and combined licenses (COLs). A first step would be a design certification that approves a power plant design expected to be built at multiple sites. A prospective owner may obtain an ESP, which approves of a prospective site for an enveloped range of potential designs; with an ESP, the owner can later construct any certified design that fits within the approved envelop, without re-review of site suitability. The owner may also apply for a COL. The COL gives approval both to construct and to operate the facility, with the latter conditioned on satisfaction of a set of license conditions called acceptance criteria. As described by [Ray \(2021\)](#), Part 52 includes the flexibility to obtain an ESP with or without a design certification, and a COL with or without either a design certification or an ESP.

A rulemaking initiative began in 2020 that should harmonize Parts 50 and 52 such that the requirements of the two rules are synchronized ([US NRC, 2019a](#)).

- Part 53 does not yet exist. The NRC staff has begun work on the new Part 53 and plans to issue an Advanced Notice of Rule-making late in 2020 ([US NRC, 2020a,b](#)). A public meeting with the Advisory Committee on Reactor Safeguards (ACRS) was held on July 20, 2020 ([ACRS, 2020](#)). The new rule had been essentially directed by Congress ([US Public Law, 2019](#)) and is expected to apply to non-LWR designs that opt to use an approach based on the technology-inclusive, risk-informed, and performance-based methodology for informing the licensing basis and content of applications for non-light-water reactors. The methodology is used to accomplish three objectives—to select licensing basis events; to classify structures, systems, and components; and to assess the adequacy of defense-in-depth ([NEI, 2019](#); [US NRC, 2018a,b](#); [ACRS, 2019](#); [Bley and Malsch, 2021](#); [Fleming, 2021](#)).
- Part 54—Requirements for renewal of operating licenses for nuclear power plants.
- Part 55—Requirements for operator licenses.
- Part 100—Reactor site criteria (which was discussed in Section 4.13).

The regulations for licensing commercial nuclear power plants fall under 10 CFR Parts 50 and 52 (and, possibly in the future Part 53). These regulations are similar, so we use Part 50 as an example, which includes the following sections:

- General Provisions—Sections 1–9
- Requirement of License, Exceptions—Sections 10–13
- Classification and Description of Licenses—Sections 20–23
- Applications for Licenses, Certifications, and Regulatory Approvals ...—Sections 30–39
- Standards for Licenses, Certifications, and Regulatory Approvals—Sections 40–49
- Issuance, Limitations, and Conditions of Licenses and Construction Permits—Sections 50–68
- Inspections, Records, Reports, Notifications—Sections 69–76
- US/IAEA Safeguards Agreement—Section 78
- Transfers of Licenses—Creditors’ Rights—Surrender of Licenses—Sections 80–83
- Amendment of License or Construction Permit at Request of Holder—Sections 90–92
- Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission—Sections 100–103
- Backfitting—Section 109
- Enforcement—Sections 110–111
- Additional Standards for Licenses, Certifications, and Regulatory Approvals—Sections 120–155

Other parts of 10 CFR 1–199 apply legal, procedural, and process issues with which the owner must comply.

Broad requirements of Title 10 of the Code of Federal Regulations Parts 1–199

Generally, the broad requirements incorporated into the regulations were anticipatory in nature and intended to reflect good practice. They usually came about in response to issues raised by the ACRS, the staff, and other stakeholders and the development of most were discussed in [Bley and Malsch \(2021\)](#). A list of the most significant regulations would include:

- 50.34 Contents of applications; Technical Information

Table 1 Title 10 Code of Federal Regulations Parts 1–199.

<i>Title 10 Code of Federal Regulations Parts 1–199</i>	
Part 1	Statement of Organization and General Information
Part 2	Agency Rules of Practice and Procedure
Part 4	Nondiscrimination in Federally Assisted Programs or Activities Receiving Federal Financial Assistance from the Commission
Part 5	Nondiscrimination on the Basis of Sex in Education Programs or Activities Receiving Federal Financial Assistance
Part 7	Advisory Committees
Part 8	[Reserved]
Part 9	Public Records
Part 10	Criteria and Procedures for Determining Eligibility for Access to Restricted Data or National Security Information or an Employment Clearance
Part 11	Criteria and Procedures for Determining Eligibility for Access to or Control over Special Nuclear Material
Part 12	Implementation of the Equal Access to Justice Act in Agency Proceedings
Part 13	Program Fraud Civil Remedies
Part 14	Administrative Claims under Federal Tort Claims Act
Part 15	Debt Collection Procedures
Part 16	Salary Offset Procedures for Collecting Debts Owed by Federal Employees to the Federal Government
Part 19	Notices, Instructions and Reports to Workers: Inspection and Investigations
Part 20	Standards for Protection Against Radiation
Part 21	Reporting of Defects and Noncompliance
Part 25	Access Authorization
Part 26	Fitness for Duty Programs
Part 30	Rules of General Applicability to Domestic Licensing of Byproduct Material
Part 31	General Domestic Licenses for Byproduct Material
Part 32	Specific Domestic Licenses to Manufacture or Transfer Certain Items Containing Byproduct Material
Part 33	Specific Domestic Licenses of Broad Scope for Byproduct Material
Part 34	Licenses for Industrial Radiography and Radiation Safety Requirements for Industrial Radiographic Operations
Part 35	Medical use of Byproduct Material
Part 36	Licenses and Radiation Safety Requirements for Irradiators
Part 37	Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material
Part 39	Licenses and Radiation Safety Requirements for Well Logging
Part 40	Domestic Licensing of Source Material
Part 50	Domestic Licensing of Production and Utilization Facilities
Part 51	Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
Part 52	Licenses, Certifications, and Approvals for Nuclear Power Plants
Part 53	[Reserved]
Part 54	Requirements for Renewal of Operating Licenses for Nuclear Power Plants
Part 55	Operators' Licenses
Part 60	Disposal of High-Level Radioactive Wastes in Geologic Repositories
Part 61	Licensing Requirements for Land Disposal of Radioactive Waste
Part 62	Criteria and Procedures for Emergency Access to Non-Federal and Regional Low-Level Waste Disposal Facilities
Part 63	Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada
Part 70	Domestic Licensing of Special Nuclear Material
Part 71	Packaging and Transportation of Radioactive Material
Part 72	Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste
Part 73	Physical Protection of Plants and Materials
Part 74	Material Control and Accounting of Special Nuclear Material
Part 75	Safeguards on Nuclear Material—Implementation of Safeguards Agreements between the United States and the International Atomic Energy Agency
Part 76	Certification of Gaseous Diffusion Plants
Part 81	Standard Specifications for The Granting of Patent Licenses
Part 95	Facility Security Clearance and Safeguarding of National Security Information and Restricted Data
Part 100	Reactor Site Criteria
Part 110	Export and Import of Nuclear Equipment and Material
Part 140	Financial Protection Requirements and Indemnity Agreements
Part 150	Exemptions and Continued Regulatory Authority in Agreement States and in Offshore Waters under Section 274
Part 160	Trespassing on Commission Property
Part 170	Fees for Facilities, Materials, Import and Export Licenses, and Other Regulatory Services under The Atomic Energy Act of 1954, As Amended
Part 171	Annual Fees for Reactor Licenses and Fuel Cycle Licenses and Materials Licenses, including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by the NRC
Parts 172–199	[Reserved]

- 50.36 Technical Specifications (Safety License Conditions)
- 50.36a Technical Specifications on Effluents from Nuclear Power Reactors
- 50.69 Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors
- Appendix A to Part 50—General Design Criteria for Nuclear Power Plants
- Appendix B to Part 50—Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

Particular issues that extended the regulations

A number of regulations were reactive, issued in response to concerns about specific issues. Some of these issues arose during NRC staff and ACRS reviews of license applications, some were identified as a result of operational events and accidents, and some were identified from the results of safety research. Some were relatively simple to address, others eventually arose from or led to investigative hearings. For example, 50.46 and 50.46(a) address emergency core cooling systems and were issued originally in response to research questioning the adequacy of then current systems and a long and contentious rule-making hearing.

Many of the issues that extended the regulations originated as so-called unresolved safety problems. The origin of this concept was in the earliest license application reviews in which the AEC staff argued that construction could continue when they found that “there was reasonable assurance that the reactor could be operated safely or at least that any unresolved safety problems could be mitigated over the intervening construction period ... Excessive caution, though ... might also have a negative impact on development of the industry. The problem was to find the appropriate balance despite many unanswered questions about atomic energy” (Mazuzan and Walker, 1984). The issues that were specific to the design under review were straightforward to address. Others really applied to all reactors or to a broad class of reactors and were not appropriate to resolve as part of the license application process. These became known as generic safety issues. More pressing design-specific issues received staff attention, while others did not. Eventually, the list of unresolved generic safety issues grew to an embarrassing length. The Commission finally directed the staff to develop a program to resolve these generic safety issues and a 1977 amendment to the Energy Reorganization Act of 1974 included a new Section 210 (US NRC, 2015):

The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter.

The NRC staff proposed a generic issues program planned to identify all existing generic issues, set priorities among them, and lay out plans for resolving the most important of them that identified the associated estimated costs (US NRC, 1978). They initially ranked the issues based on importance into one of four categories from significant to of little or no importance using engineering judgment (US NRC, 2011a,b). This was followed by developing a semi-quantitative approach using results from probabilistic risk assessment (PRA) (Bley and Malsch, 2021; Dube, 2021; Fleming, 2021; Modarres, 2021) and later by a more fully quantitative use of PRA to identify the projected risk reduction by resolving each generic issue, along with an estimate of the costs. Following the accident at Three Mile Island Unit 2 (Bley and Malsch, 2021; Skillman and Rempe, 2021; Walker, 2004) many new generic issues were identified (US NRC, 2011a), and the NRC staff requested Commission approval to use the new methodology (US NRC, 1981). After discussion and revision the Commission approved its use (US NRC, 1983).

A summary of some of the regulations that were issued in response to concerns about specific issues is provided in the following list:

- 10 CFR 50.44 Combustible Gas Control
- 10 CFR 50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors
- 10 CFR 50.46a Acceptance Criteria for Reactor Coolant System Venting Systems
- 50.47 Emergency Plans
- 50.48 Fire Protection
- 50.49 Environmental Qualification of Electric Equipment Important to Safety For Nuclear Power Plants.
- 50.61 Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.
- 50.61a Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events
- 50.62 Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants
- 50.63 Loss of All Alternating Current Power (Station Blackout)
- 50.150 Aircraft Impact Assessment
- 50.155 Mitigation of Beyond-Design-Basis Events
- Appendix E to Part 50—Emergency Planning and Preparedness for Production and Utilization Facilities
- Appendix G to Part 50—Fracture Toughness Requirements
- Appendix H to Part 50—Reactor Vessel Material Surveillance Program Requirements
- Appendix K to Part 50—ECCS Evaluation Models
- Appendix R to Part 50—Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

Legal, procedural, and processes issues

Legal, procedural and process issues are addressed primarily in 10 CFR Part 2, which addresses how public hearings on license applications, contested enforcement actions (like license suspensions), and monetary civil penalties are conducted, and how members of the public may ask the NRC to revoke or suspend licenses. But 10 CFR Part 50 also includes some process-related provisions, e.g., provisions addressing license transfers (50.80), financial security interests (50.81), license termination (50.82), decommissioning (50.83), grounds for license termination (50.100–50.103) and grounds for criminal prosecution for willful violations of the Atomic Energy Act and NRC regulations (50.110–50.111). The NRC may, by itself, initiate legal proceedings to suspend, revoke and amend licenses, and to impose monetary civil penalties for violations of its regulations, but alleged criminal violations must be referred to the US Department of Justice for consideration of criminal prosecution.

Summary

NRC regulations were developed from a blank slate following passage of the Atomic Energy Act of 1954. They are tabulated in 10 CFR Parts 1–199. Rules were developed to address broad requirements that were anticipatory in nature and intended to reflect good practice; to address particular issues in response to accidents, generic safety issues, and new research; and to fulfill legal, procedural and process issues. New regulations are introduced through the rulemaking process ([US NRC, 2019b](#)).

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U.S. Options for Licensing a New Commercial Power Plant

Harold B. Ray, Public Utility Executive Vice President, Chief Nuclear Officer (Retired), San Marino, CA, United States

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Glossary

Advisory Committee on Reactor Safeguards (ACRS) Independent Advisory Committee to the US Nuclear Regulatory Commission that is statutorily mandated by the Atomic Energy Act of 1954, as amended.

Atomic Safety and Licensing Board (ASLB) Independent trial level adjudicatory body of the NRC. Public hearings for individual applications are typically conducted by three designated panel members.

Combined License (COL) Authorization for a licensee to construct and operate a facility using a certified standard design.

Construction Permit (CP) Authorization for a licensee to construct a facility.

Design Certification (DC) A Commission approval, issued under subpart B of 10 CFR Part 52, of a standard design for a nuclear power facility. This design may be referred to as a certified standard design.

Design Control Document (DCD) Document submitted by the applicant that provides information to support the NRC's approval and certification of a standard design, under the provisions of 10 CFR Part 52.

Early Site Permit (ESP) A Commission approval, issued under subpart A of 10 CFR Part 52, for a site for one or more nuclear power facilities. The early site permit will address site safety issues, environmental protection issues, and plans for coping with emergencies, independent of the review of a specific nuclear plant design.

Final Safety Analysis Report (FSAR) Document submitted by applicant that provides information to support the US NRC safety evaluation of a proposed facility. It is based on the finalized detailed design of the facility, and it must periodically be updated throughout the life of the facility.

Operating License (OL) Authorization for a license to operate a facility.

Preliminary Safety Analysis Report (PSAR) Document submitted by applicant that provides information to support the US NRC safety evaluation of a proposed facility. It is based on the preliminary design of a facility and limited to use in the two-step 10 CFR Part 50 process.

Safety Evaluation Report (SER) NRC document summarizing the anticipated effect of the proposed application on public health and safety.

Standard Design Approval (SDA) An NRC staff approval, issued under subpart E of 10 CFR Part 52, of a standard design for a nuclear power reactor of the type described in 10 CFR 50.22. The approval may be for either the design for the entire reactor facility or the design of major portions thereof.

Introduction

As described in detail, US Federal law ([U.S. Government Publishing Office, 2019](#)) requires that an operating license (OL) be issued by the US Nuclear Regulatory Commission (NRC) specifically for each commercial nuclear power plant prior to its entering service. Issuance of the license is based on both safety and environmental impact reviews of the proposed plant, and the initial term of the license is limited to 40 years. Broadly, there are two alternative pathways available for obtaining an OL, and each pathway requires one or more prior approvals to be obtained before the plant can be constructed and the OL can be issued.

Requirements for the first alternative pathway are set forth in 10 CFR Part 50. This provides for a two-step process to obtain an OL (see [Fig. 1](#)). First, prior to commencing construction, a Construction Permit (CP) must be obtained by the operator/licensee based on submittal and review of a Preliminary Safety Analysis Report (PSAR). Then, as construction is underway, an OL must be sought based on submittal and review of a Final Safety Analysis Report (FSAR), and the OL must be issued prior to fuel loading.

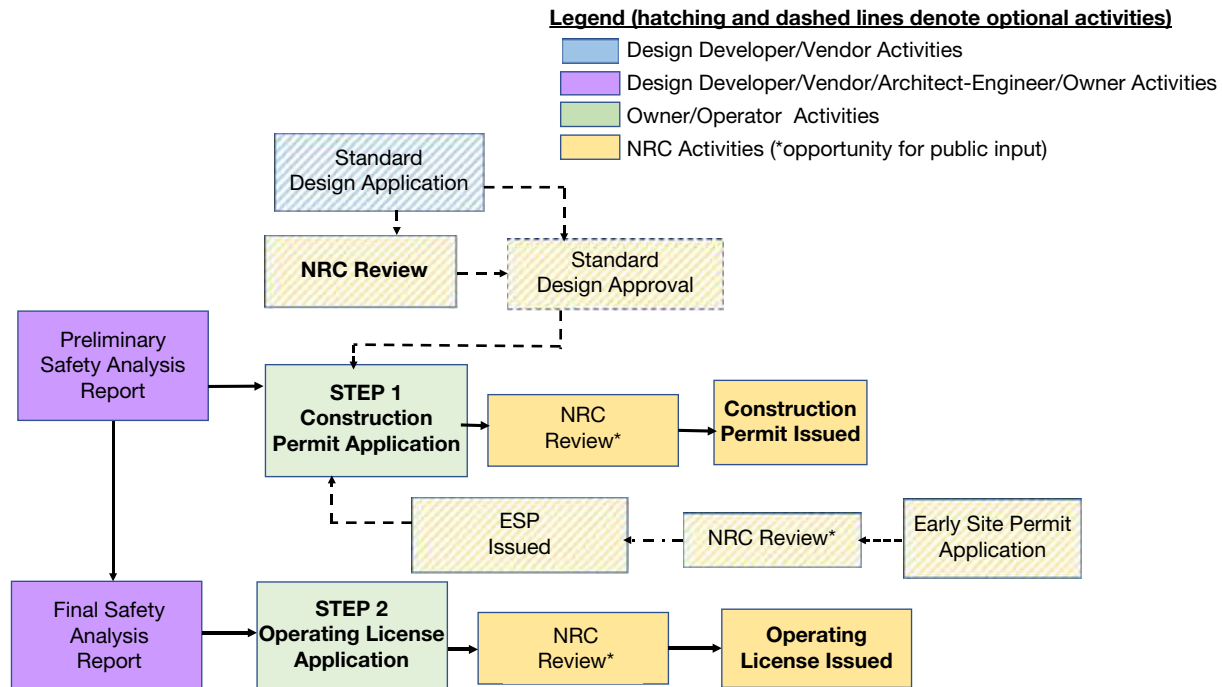


Fig. 1 10 CFR Part 50 licensing process.

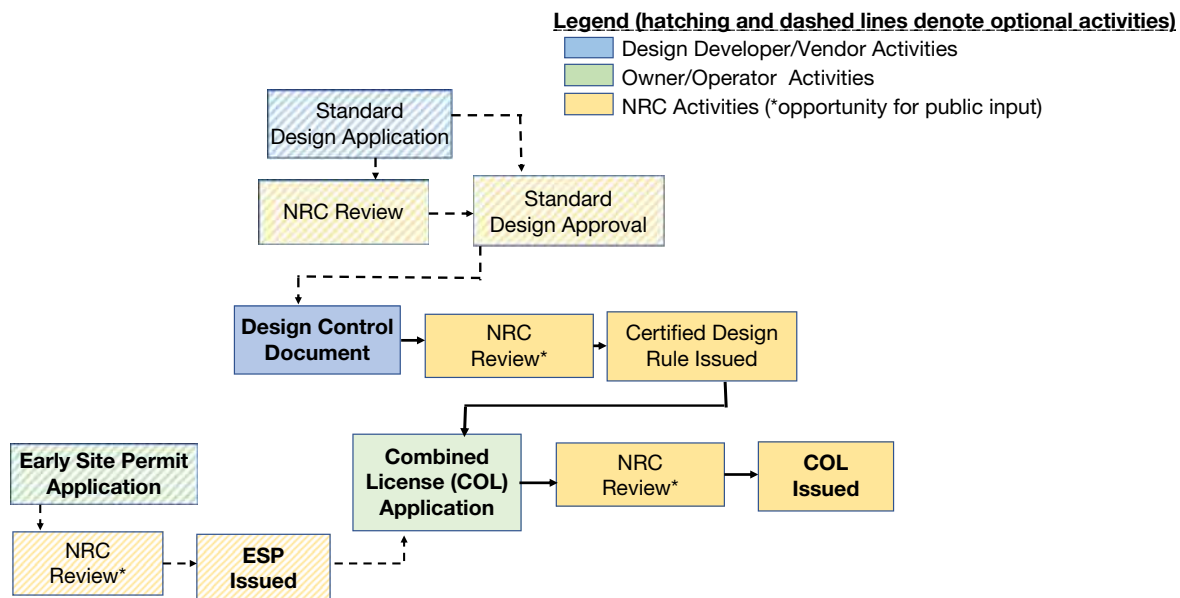


Fig. 2 10 CFR Part 52 licensing process.

Since 1989, and as updated in 2007, a second alternative pathway has been provided in 10 CFR Part 52 (see Fig. 2). For those plants which incorporate a certified standard design, as described in a specified design control document (DCD), this alternative provides for a combined license (COL) to be obtained in a single step prior to commencing construction. Since the DCD can be referenced by multiple COL applicants, this second alternative pathway allows the cost of the design and its certification ultimately to be recovered from multiple plant owners. More importantly, it promotes standardization of the certified design, thus reducing plant OL uncertainty and the associated financial risk to the plant owner. Design certification (DC) does require significant up-front investment by the design vendor prior to sale of the certified design to plant owners following which operator/licensees then can apply for their respective COLs. The risk of recovering this required up-front investment is a significant hurdle for the design vendor. (Note: Plant ownership may be shared among several entities. The plant operator/licensee is limited to a single entity which usually also is one of the plant owners.)

Thus, under the Part 50 two-step process the plant owner bears more financial risk due to licensing uncertainty, which remains until the OL is issued to the operator/licensee following completion of construction, than it does under the Part 52 one-step process where the certified design vendor has separately obtained the certification for use of a standard plant design. However, the certified design vendor may have to bear the entire risk of the investment required to obtain DC prior to its first sale to a plant owner. To summarize, Part 52 effectively allows transfer of a portion of a plant owner's investment risk under Part 50 to the design vendor. Although this can become a major burden, a successful design vendor can market a DC to multiple plant owners.

Instead of design certification, a standard design approval (SDA) may be obtained for a new design and then referenced in either a Part 50 CP application or in a Part 52 COL application. An SDA does not provide the regulatory finality of a design certification, but it can cover a lesser design scope and require less initial investment, while resolving selected regulatory issues.

Either the two-step Part 50 CP/OL pathway or the one-step Part 52 COL pathway can be preceded by an Early Site Permit (ESP), as described in 10 CFR Part 52, Subpart A. The optional addition of an ESP to a plant development allows site-related matters to be addressed and decided prior to proceeding further with plant investment, and separately from obtaining either the CP or COL. However, obtaining an ESP prior to taking other licensing actions can add to the total time required.

Finally, regardless of the alternative pathway chosen, a comprehensive assessment of the environmental impact of plant construction and operation is separately required.

Current alternatives for licensing commercial nuclear power plants

Plant license application *not* based on certified standard design—10 CFR Part 50

If a proposed plant does not include a standard design which has been certified pursuant to 10 CFR 52, Subpart B, then the two-step licensing process described in 10 CFR Part 50 is required to be followed. More than 100 operating nuclear plants have used this licensing process, and its advantages include that it does not require as much investment in design development prior to obtaining the permit required to begin plant construction. Its disadvantages include that design development following CP issuance lacks the standardization of a certified design, and there is the potential for contested public hearings being required at both the CP and OL approval steps.

Principal elements of this process are as follows:

Submittal and review of the PSAR

10 CFR 50.34 describes in subsection (a) the minimum information required to be provided to support issuance of a CP. The information provided includes a description of the preliminary design, a safety assessment of the site and a safety assessment of the proposed plant.

Public meetings

An initial public meeting is held by the NRC upon submittal of the plant license application. A subsequent public hearing and findings by the Atomic Safety and Licensing Board (ASLB) is also required prior to issuance of the CP. A second public hearing is held prior to issuance of the OL only if its issuance is contested. (The ASLB conducts hearings and reaches findings relative to compliance with all regulatory requirements as a legal proceeding. Parties may then appeal findings to a separate Atomic Safety and Licensing Appeals Board.)

Elements of NRC safety review

The NRC review of safety analyses follows a standard review plan (SRP) applicable to the reactor type. For light water reactors, the SRP is provided in NUREG-0800. A review is also conducted by the NRC's independent Advisory Committee on Reactor Safeguards (ACRS). When the NRC completes its review, it issues a safety evaluation report (SER) summarizing the anticipated effect of the proposed plant on public health and safety.

Elements of NRC environmental review

In accordance with requirements of the National Environmental Protection Act, a full environmental impact statement (EIS) must be submitted and reviewed, relative to potential impacts and benefits of the proposed plant, prior to issuance of the CP. Regulatory requirements for the environmental review are provided in 10 CFR Part 51. (A supplemental EIS, addressing any changes which occurred following issuance of the CP, must also be submitted and reviewed prior to issuance of the OL.)

Construction permit issuance—Step one completed

At this point, following issuance of the CP, construction may commence. Typically, design details not previously completed are then finalized as necessary in advance of affected equipment procurement and as required to support ongoing construction. Deferral of investment in completing these design details until following CP issuance is an advantage of using the two-step 10 CFR Part 50 licensing process and avoids placing this investment at risk earlier in the process. Also, it allows the design details to reflect site specific (e.g., seismic, flooding, etc.) characteristics as well as the current selected equipment detailed information.

While use of the two-step licensing process allows deferral of investment in detailed design, it does not preclude use of existing designs, as appropriate, including designs certified in accordance with 10 CFR 52, Subpart C, standard designs approved in

accordance with 10 CFR 52, Subpart E, and designs used in plants previously approved under Part 50. The disadvantage of the two-step licensing process is that the first step does not authorize operation of the completed plant, and this may therefore be contested at the second step.

Submittal of final safety analysis report (FSAR)

The FSAR is submitted based on the finalized detailed design and in support of the second step, which is issuance of the OL. Information required to be included in the FSAR is provided in subsection (b) of 10 CFR 50.34.

Following issuance of the OL, the FSAR must periodically be updated throughout the life of the plant in accordance with 10 CFR 50.71(e). Guidance for this is provided in Regulatory Guide 1.181.

Operating license issuance—Step two completed

The OL is issued for an initial period of 40 years and is subject to renewal in 20 year increments.

Standard design certification—10 CFR Part 52, subpart B

The vendor of a new plant design must decide whether to pursue its development independently of prospective plant owners to the point that the design has been certified, or to develop the design with support from prospective plant owners. In the latter case, a design vendor/plant owner partnership can take the form of the two-step 10 CFR Part 50 process described above, in which the plant operator/licensee seeks a CP and then an OL, supported throughout by the plant design vendor. The completed design can then subsequently be certified for future use at other plants. Alternatively, prospective plant owners can support the plant design vendor as it pursues a DC and separately pursue the one-step 10 CFR Part 52 process described below, in which the plant operator/licensee seeks a COL based on the certified standard design.

Each standard DC requires a notice and comment rulemaking which is conducted in response to an application submitted by the design vendor. (The NRC may also elect to hold a legislative-style hearing to review comments received.) The DC review does not address site-specific design features, operational programs and environmental impacts of building the design at a particular site. Through 2019, DC has been finalized for five designs (As of August 2019, (NRC, 2019a), the NRC has issued the following design certifications: [Advanced Boiling Water Reactor \(ABWR\)](#); [ABWR Design Certification Rule \(DCR\) Amendment](#); System 80+; [Advanced Passive 600 \(AP600\)](#); [Advanced Passive 1000 \(AP1000\)](#); [Economic Simplified Boiling-Water Reactor \(ESBWR\)](#). A proposed rule for certification of the [Advanced Power Reactor 1400 \(APR1400\)](#) was issued in May 2019).

The certification is held by the design vendor and has an initial term of 15 years, which may be renewed. All major design features must be covered by the certification, and the required contents of the application are specified in 10 CFR 52.47. Sufficient detail is required to reach a final conclusion that there is reasonable assurance of adequate protection of the public health and safety and of the environment. As for the first step of the 10 CFR Part 50 process, a public hearing is required, and the design, when certified, must be as described in detail in the DCD referenced in the NRC rule.

If the certification is sought for an existing or evolutionary design, additional design development or testing prior to certification may not be required. However, advanced designs will require significant design development and may require testing to verify important design assumptions, either prior to certification or as a condition of certification. The NRC states that the design must be “essentially complete” in order to be certified.

The complete design description in the DCD is categorized as Tier 2 information, and it is equivalent to the information contained in a Part 50 FSAR. DCD Tier 2 information is approved, but it is not certified. A procedure is provided to allow changes to Tier 2 information without prior approval by the NRC, provided the change meets specific criteria. However, the DCD also includes select information which is categorized as Tier 1 and which is based entirely on the Tier 2 information. Due to its safety significance, Tier 1 information cannot be changed, or deviated from, except by first revising the rule which established the DC.

A third category of DCD information is Tier 2*. Due to its safety significance similar to Tier 1 information, Tier 2* information does require NRC approval prior to making a change, but the approval does not require a revision of the DC rule since it has not been certified as is Tier 1 information. The Tier 2* category exists in order to allow some added flexibility.

Tier 1 information in the DCD also provides the basis for, and includes, required inspections, tests, analyses and acceptance criteria (ITAAC) which must be satisfied prior to fuel loading. ITAAC include design acceptance criteria (DAC) which provide the processes and acceptance criteria for completing design detail which was not complete at the time of certification. To some extent, investment in design and testing can be deferred beyond certification by increasing use of ITAAC, including DAC, but this increased use reduces the certainty in what will ultimately be necessary to demonstrate that requirements have been met in order to allow fuel loading.

In all cases, a probabilistic risk assessment, including uncertainties, is required as part of the DC application. A framework and guidance is provided by NUREG-1860.

The investment in design detail required to obtain a standard design certification grows significantly as the departure from previously licensed designs increases. A standard design certification is valid for inclusion in a COL application for an initial term of 15 years, and this may be renewed.

Plant license application based on certified standard design—10 CFR Part 52, subpart C

If a proposed plant does use a vendor's standard design which has been certified pursuant to 10 CFR Part 52, Subpart C, then the one-step licensing process described in Subpart C may be followed by the plant operator/licensee to obtain a combined license (COL). That is, the separate CP and OL steps required in 10 CFR Part 50 are combined into a single step. The COL is a combined CP and conditional OL, it is issued for an initial term of 40 years, and the COL is subject to renewal. (The OL is conditional based on demonstrating that all requirements of the COL have been met prior to fuel loading.) Through 2019, COLs have been issued for eight sites (As of August 2019 ([NRC, 2019b](#)), the NRC has issued the following COLs: Detroit Edison Company for the Fermi Unit 3 ESBWR; Duke Energy Florida, LLC (DEF) for the Levy Nuclear Plant, Units 1 and 2 AP1000s; Dominion Virginia Power (Dominion) for the North Anna, Unit 3 ESBWR; Nuclear Innovation North America, LLC (NINA) for the South Texas Project, Units 3 and 4 ABWRs; Florida Power & Light Company (FPL) for the Turkey Point, Units 6 and 7 AP1000s; South Carolina Electric & Gas (SCE&G) for the Virgil C. Summer, Units 2 and 3 AP1000s; Southern Nuclear Operating Company (SNC) for the Vogtle, Units 3 and 4 AP1000s; and Duke Energy for the William States Lee III, Units 1 and 2 AP1000s. Per [NRC \(2019c\)](#), only the Fermi Unit 3, North Anna Unit 3, Vogtle Units 3 and 4, and William States Lee 1 and 2 COLs have not been terminated).

Principal elements of this process are as follows:

Submittal and review of the COL application (COLA)

Requirements concerning information to be included in the COLA are provided by 10 CFR Part 52.79 and associated guidance is provided in Regulatory Guide 1.206. In addition, design specific review standards may be created by the NRC to expedite the application submittal and review process. For multiple COLAs using the same certified design and submitted by multiple plant owners, the NRC encourages the use of what is referred to as a design-centered review approach to facilitate standardization of applicant reviews. (i.e., Multiple COL applicants form a design center and jointly interface with the NRC on design matters in common.) COLA reviews may proceed concurrently with review of a referenced DC application and do not need to wait until the DC rule is issued.

Public meetings

An initial public meeting is held by the NRC upon submittal of the plant license application. A subsequent public hearing and findings by the ASLB is also required prior to issuance of the COL. There is an opportunity for a hearing prior to authorization of operation of a completed facility, but at this point a petitioner must first show that the plant has not met, or will not meet, the acceptance criteria for operation established in the COL.

Elements of NRC safety review

The NRC review of safety analyses follows a standard review plan (SRP) applicable to the reactor type. For light water reactors, the SRP is provided in NUREG-0800. A review is also conducted by the NRC's independent ACRS. As in the case of a Part 50 review, when the NRC completes its review, it issues a SER summarizing the anticipated effect of the proposed plant on public health and safety.

Elements of NRC environmental review

To the extent that they have not been resolved in an earlier ESP approval for the plant site, environmental issues must be addressed in an environmental report as described above for the 10 CFR Part 50 process.

Combined license issued—Construction may commence

COL issuance authorizes construction of the plant, and it also authorizes operation subject to NRC approval of ITAAC compliance. ITAAC, including DAC, are provided in the referenced design certification, in any referenced ESP, and they are also developed during the COL review in accordance with 10 CFR Part 52.80 to verify that the facility has been constructed in accordance with its certified design.

Because the certification necessarily requires sufficient design detail to support concurrent approval of both construction and operation, changes to this detail may become necessary as construction proceeds. This includes changes to information requiring a rule change (Tier 1) or prior NRC approval (Tier 2*). Processing resulting amendments to the COL and DCD can delay construction, and the NRC has therefore implemented a preliminary amendment request procedure to reduce this potential impact. However, changes can remain a burden, especially for the initial applications of a design certification.

Inspections, tests, analyses and acceptance criteria met—Operation may commence

ITAAC, including DAC, can be established as part of: (1) design certification, (2) the optional ESP approval and (3) the COL approval. The NRC must review and accept in accordance with 10 CFR Part 52.103(g) that ITAAC have been met prior to affected structures, systems and equipment being considered operable for purposes of fuel loading or power production. This requires a public hearing near completion of plant construction.

The FSAR submitted in the COLA is updated at the time operation is authorized and it is maintained during the plant life, the same as for a FSAR submitted in support of a Part 50 OL.

Standard design approval (not certification)—10 CFR Part 52, subpart E

SDA, as described in 10 CFR Part 52, Subpart E, provides for NRC staff review and conclusions on a proposed design, or on only portions thereof. When referenced in subsequent Part 50 or Part 52 licensing applications, the SDA is not binding to the same extent as is a certified standard design. However, it may require far less investment and time to obtain, and it can therefore be a valuable part of a staged licensing process. As of August 2019, SDAs (previously termed to as final design approvals) have been finalized for all of the designs certified by the NRC.

Although detailed procedures may differ, the SDA may be considered comparable to the Vendor Design Review process of the Canadian Nuclear Safety Commission and Generic Design Assessment process of the United Kingdom Office for Nuclear Regulation insofar as it provides for clear and early feedback on potential licensing and safety issues. However, because SDA lacks the certainty of DC, it has not been favored as a licensing path. SDA applications are required to include information as described in 10 CFR Part 52.137.

Early site permit (ESP) approval prior to submittal of license application—10 CFR Part 52, subpart A

An optional ESP resolves issues of site suitability (including security issues), environmental impact (including consideration of alternative sites) and emergency planning. It includes a plant parameter envelope which is intended to bound actual plant designs which could be proposed using either the two-step Part 50 or one-step Part 52 licensing processes following ESP approval. Because it resolves issues of site suitability in accordance with the finality provisions of 10 CFR Part 52.39(a), it does require its own public hearing. The permit may be granted for a period of 10 to 20 years prior to plant licensing, and it is subject to renewal. Through 2019, ESPs have been finalized for four sites (Per [NRC \(2019d\)](#)), the NRC has issued the following ESPs for the following applicants: Clinton ESP Site for Exelon Generation Company, LLC; Grand Gulf ESP Site for System Energy Resources Inc.; North Anna ESP Site for Dominion Nuclear North Anna, LLC; Vogtle ESP Site for Southern Nuclear Operating Company; and PSEG Site for PSEG Power, LLC, and PSEG Nuclear, LLC (PSEG). The NRC is reviewing the ESP applications for the Clinch River Site for Tennessee Valley Authority).

Use of an ESP resolves significant issues and removes them from the plant licensing process itself. This can provide a major benefit by reducing risk to the investment which is made prior to and during the plant licensing process. However, it can extend the overall time involved in bringing a new plant into service if it takes place prior to submittal of the CP or COL application.

Limited work authorization

Regardless of the alternative path chosen, or whether an ESP approval is sought, a Limited Work Authorization (LWA) may be sought in accordance with 10 CFR Part 50.10 to permit certain construction activities to commence at a site prior to obtaining either a Part 50 CP following the two-step, or a Part 52 COL following the one-step, alternative.

Summary and future activities

Plant developers and prospective plant owners and operator/licensees currently have two alternative pathways available to enable new commercial power plants to be built and operated in the United States. The pathway used for all plants currently in operation provides for construction to be authorized based on a preliminary plant design and subsequently for operation to be authorized based on the final plant design, as constructed. The pathway used more recently for a few plants provides for an essentially complete standard plant design first to be certified for use and then to be referenced in applications for combined construction and operation approval at specific sites. Both pathways must separately also satisfy environmental impact reviews for the specific sites.

The first pathway allows investment in design development to be deferred until authorization to construct a plant at a specific site has been granted based on a preliminary design. It also allows for non-standard design details to be included, based on owner and operator/licensee preferences. Because the second pathway is based on use of a certified, standard plant design, the owner and operator/licensee uncertainties during construction and pending operation are significantly reduced, as compared to the first pathway. Also, the second pathway enables a plant vendor to invest in and obtain a design certification without having first to partner with a prospective plant owner. The certified design can then be marketed to multiple prospective owners.

There has not been sufficient experience with the second licensing pathway to determine which is preferable in most cases. Each pathway has advantages and disadvantages that can only be weighed in specific cases after the parties involved reach and implement detailed agreements.

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Regulation of Two U.S. Non-LWRs—Lessons Learned

Joy L. Rempe, Rempe and Associates, LLC, Idaho Falls, ID, United States

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Glossary

Advisory Committee on Reactor Safeguards (ACRS) Independent advisory committee to the U.S. Nuclear Regulatory Commission (NRC) that is statutorily mandated by the Atomic Energy Act of 1954, as amended.

Construction Permit (CP) Authorization for a licensee to construct a facility.

Final Safety Analysis Report (FSAR) Document, submitted by an applicant, providing information to support the NRC safety evaluation of a proposed facility. It is based on the finalized detailed design of the facility, and it must periodically be updated throughout the life of the facility.

Operating License (OL) Authorization for a license to operate a facility.

Introduction

Polymakers run the risk of "reinventing the wheel" when they make judgments on problems they face unless they are well informed about the context in which previous decisions of a similar nature were made, what alternatives were considered and why certain ones were chosen, and what personal and impersonal forces shaped a particular policy. ... The point is not to belabor the often misleading aphorism that "those who fail to study the past are condemned to repeat it," but rather to recognize that both continuity and change in history need to be understood to deal effectively with the present.

-Mazuzan and Walker, 1997

Advanced reactor designs, which may be cooled by water, gas, liquid-metal, and molten salt, often incorporate different fuels, materials, and fission product release barriers than found in current generation Light Water Reactors (LWRs) [defined in [US NRC \(2008\)](#) as LWRs licensed before 1997]. In recent years, some advanced reactor design developers assert that their selected fuel and coolant combinations offer increased safety margins, even in the absence of emergency power or operator intervention. In short, these developers contend their designs achieve safety by relying on inherent physical processes rather than engineered safety features. In addition to funding from the U.S. Department of Energy (DOE) to accelerate research and deployment of their technologies, advanced nuclear reactor proponents ([Martin, 2015](#); [Geist, 2015](#); [CRS, 2019](#)) advocate that successful deployment of non-LWR technologies requires significant changes in the current regulatory framework, which is burdensome and inflexible. Some are requesting a technology-neutral regulatory framework with provisions for a "test-then-license" approach, similar to the system used within the U.S. during the early years of nuclear. In 2017, the 115th U.S. Congress enacted two bills related to such concerns: the Nuclear Energy Innovation Capabilities Act of 2017 (NEICA), which required the DOE to develop a versatile fast neutron test reactor (VTR); and the Nuclear Energy Innovation and Modernization Act (NEIMA), which requires the U.S. Nuclear Regulatory Commission (NRC) to develop an optional regulatory framework suitable for advanced nuclear technologies. Both acts have been signed into law. The DOE currently plans to self-regulate the startup and oversight of the VTR ([INL, 2020](#)).

It is reasonable for applicants to request reductions in the time and costs required to gain regulatory approval for a design with significantly different characteristics than current-day LWRs. However, it is important to recognize the increased uncertainty associated with the performance of such reactor systems. As emphasized by the Advisory Committee on Reactor Safeguards (ACRS), "Decision makers should regard with caution quantitative claims of high safety performance for reactor systems still at the conceptual design stage" ([ACRS, 1989](#)). In this letter report, ACRS also emphasized issues that must be resolved, including the calculation and use of a source term for siting, the need for a containment building, and offsite emergency planning requirements. In a more recent letter, ACRS continues to emphasize the need for caution, noting that for non-LWRs, "Creative thinking will be required to identify such unique situations, to thoroughly identify the scenarios that will be the basis of the safety analysis and the source of releases, and to evaluate the suitability of sites" ([ACRS, 2019](#)).

In considering an alternative regulatory framework, important regulatory lessons can be gained by reviewing the startup and operating experience, as well as the regulation, of prior non-LWR examples. This chapter concentrates on two examples of commercial reactor prototypes built and operated in the U.S.—the Fort Saint Vrain (FSV) high temperature gas reactor (HTGR) and the Enrico Fermi Unit 1 (EF1) sodium fast reactor (SFR). This chapter also summarizes recent actions taken by the NRC to establish a new regulatory framework to address issues raised during these and other non-LWR licensing activities. Evaluation of the examples discussed in this chapter provides insights for avoiding difficulties with this new framework.

Background

The two non-LWRs focussed upon in this chapter were initially licensed by the U.S. Atomic Energy Commission (AEC), the predecessor for the NRC. At that time, the economic and political environment was significantly different than today. In his 1954 message to Congress advocating that the Atomic Energy Act (AEA) of 1946 be amended, President Eisenhower called for the AEC "to establish minimum safety and security regulation" that would encourage the use of nuclear technology for various commercial

Table 1 Power Reactor Demonstration Program (PRDP) rounds, applicants, and plants [based on information in Gillespie et al. (2016) and Allen (1977)].

	<i>Round I</i>	<i>Round II</i>	<i>Round III</i>	<i>Modified Round III</i>
Opened	January 1955	September 1955	January 1957	August 1962
Purpose or goal	Stimulate construction of prototype commercial reactors	Engage public utilities in construction of small, experimental reactors (< 40 MW)	Provide continuing assistance for development of power reactors; focused on large-scale commercial reactors	Support the construction of large baseload plants using proven technologies
Applicant: Plant Name (if available)	<i>Successful:</i> - Yankee Atomic Electric Company: Yankee Rowe - Nuclear Power Group: Dresden - Consumers' Public Power Group: Hallam - Power Reactor Development Corporation (PRDC): Enrico Fermi Unit 1	<i>Successful:</i> - City of Piqua, Ohio: Piqua - Rural Cooperative Power Association: Elk River - Dairyland Power Cooperative: LaCrosse <i>Unsuccessful:</i> - Chugach Electric Association - Wolverine Electric Cooperative - Holyoke Gas and Electric Company - Orlando Utilities Commission - University of Florida	<i>Successful:</i> - CVNPA: Carolinas-Virginia Tube Reactor - Consumer's Power Company: Big Rock Point - Northern States Power Company: Pathfinder - Philadelphia Electric Company (PECO)^a Peach Bottom Unit 1 (PB1) - Southern California Edison (SCE): San Onofre Unit 1 <i>Unsuccessful:</i> - East Central Nuclear Group/Florida-West Coast Nuclear Group	<i>Successful:</i> - Connecticut Yankee Atomic Power Company: Haddam Neck <i>Unsuccessful:</i> - City of Los Angeles Department of Water and Power: Corral Canyon
Approved and completed plants (Initial Criticality)	- Yankee Rowe PWR (1961) - Dresden BWR (1959) - Hallam Sodium Graphite Reactor (1962) - Fermi Fast Breeder Reactor (1963)	- Piqua (Organic Cooled/moderated reactor) - Elk River BWR (1962) - La Crosse BWR (1967)	- Carolinas-Virginia Tube Reactor Heavy Water Reactor (1963) - Big Rock Point BWR (1963) - Pathfinder BWR w/integrated nuclear superheat (N/A)^b - Peach Bottom HTGR (1967) - San Onofre PWR (1967)	Haddam Neck PWR (1968)

^aPECO was operator and major owner; however, PB1 was supported by a large consortium of more than 50 organizations.

^bAlthough Pathfinder was constructed, it never entered commercial operation due to complications associated with superheat production.

applications (Eisenhower, 1954). After the 1954 amendment to the AEA became law, the AEC initiated a three-round Power Reactor Demonstration Program (PRDP) [also referred to as the Cooperative PRDP, the Program for Demonstration Reactors, or the Power Demonstration Reactor Program] with incentives, such as funding for research and development and fuel at a reduced cost, to encourage deployment of a diverse collection of commercial nuclear power plant designs that were privately owned and operated. As summarized in Table 1, the PRDP resulted in 13 units being successfully completed, several of which were non-LWRs (Gillespie et al., 2016; Allen, 1977). Although this chapter focuses on issues encountered with deployment and operation of FSV and EF1, their experiences during licensing, startup, and operation are representative of what may occur with other first-of-a-kind (FOAK) demonstration reactors.

The two non-LWRs focussed upon in this chapter were licensed as Research and Development (R&D) facilities under Section 104(b) of the Code of Federal Regulations, Title 10—Energy, Part 50, Section 21 (10 CFR 50.21), “Class 104 Licenses; for Medical Therapy and Research and Development Facilities” (U.S. Government Publishing Office, 2020). After a plant had operated sufficiently to demonstrate its safety and commercial viability, the owner was expected to convert its Section 104(b) license to a Section 103 license (a commercial power plant license). As discussed by Bley and Malsch (2021) and Mazuzan and Walker (1997), this requirement was dropped in 1970, and the 104(b) process was eventually eliminated. Today, if data are insufficient to demonstrate the performance of reactor safety features, the applicant may pursue an NRC license using provisions of 10 CFR 50.43(e) for a prototype plant. Similar to the Section 104(b) approach, 10 CFR 50.43(e) provides a licensee more flexibility, but the NRC maintains the right to impose additional requirements on siting, safety features, or operational conditions for such plants. As noted by the NRC (see US NRC, 2007), insights from operation of a prototype may then be used in either the 10 CFR 50 or 10 CFR 52 processes described by Ray (2021).

In this chapter, insights gained from the FSV and EF1 experiences are highlighted in four areas: relevant operating experience; regulatory framework, review, and oversight; initial testing and operation; and complexity and technology advances. The importance of each area is first discussed.

Relevant operating experience

Domestic and international reactor operating experience provides important insights into component and system interactions and the importance of integrating requirements for design, construction, and operation. Relevant operating experience may be used by applicants to provide confidence that new aspects of their proposed design will accomplish required safety functions. Experience gained from the first commercial LWRs had limited application to non-LWRs. Operating experience from a prototype or test reactor, however, was used as a basis to request that the regulator reduce some of the required testing (see [Initial Testing and Operation](#)).

Regulatory framework, review, and oversight

Several references ([Moses, 2006](#); [Brey, 2014](#)) emphasize that it would have been advantageous if an established licensing framework (including regulatory guidance and design criteria) were in place prior to seeking regulatory approval. U.S. regulations and requirements evolved in response to emerging safety issues, such as the Browns Ferry fire ([Rempe, 2021](#)) and the Three Mile Island Unit 2 accident ([Skillman and Rempe, 2021](#)). Such experience is limited for non-LWRs. Without tools, such as a Standard Review Plan, Regulatory Guides, General Design Criteria, and regulator-endorsed standards, to evaluate the acceptability of a system or program, the regulator was forced to rely on analysis using tools with limited validation and on the limited relevant operating experience available. Although a regulator may allow more latitude in such cases, this latitude may be offset by the delays and additional resources needed to determine appropriate regulatory requirements.

An independent reviewer's perspective can be invaluable in efforts to avoid designing, building, and operating an unsuccessful plant. As part of their independent review of a new proposed reactor concept, the regulator should ask probing questions, such as the following ([Fuller, 1989](#); [Moses, 2006](#); [Moses and Lanning, 1985](#)):

- What parameters should be measured to demonstrate the plant can operate safely and how will these parameters be measured?
- Is this design consistent with regulatory requirements (as reflected in agreed upon design criteria, equipment qualification requirements, and technical specifications)?
- Are data and logic presented in the probabilistic risk analyses appropriate, especially information used to characterize the performance of safety and non-safety-related systems?
- Is there the potential for a system or component to cause unintended consequences that can adversely impact plant safety?

Although it may be expensive to address issues raised by an independent reviewer, it costs less than building and operating an unsuccessful plant. Experience indicates that proactive engagement with the regulator can reduce unnecessary delays and costs.

Initial testing and operation

Without relevant operating experience, the regulator will require extensive testing, including zero power, power ascension, and transient testing, to demonstrate the performance of the reactor and new systems incorporating FOAK equipment. Prior experience demonstrates that startup and operation of a non-LWR requires significant technical support and funding. In the FSV and EF1 examples, modifications were required to address issues identified during initial testing.

Complexity and technology advances

Experience indicates that it is wise to limit complexity and adoption of technological advances and if possible, to rely on proven systems and components. Although such advances are alluring, the FSV and EF1 experiences illustrate that plant unavailability overcomes any perceived benefits. Unnecessary problems associated with deployment and use of new component and system designs should be avoided by proactively identifying potential issues and by obtaining relevant test and operating data to address such issues. Operating and accident conditions, as well as maintenance activities, should be considered in the quest to identify potential issues with new components and systems. Again, an independent regulator may be beneficial in identifying potential issues.

Example 1—The Fort St. Vrain (FSV) high temperature gas reactor

Designed by General Atomics (GA), FSV was operated by Public Service of Colorado (PSC). After the AEC issued a permit, construction was initiated in September 1968 to build FSV near Platteville, CO.

Design

The FSV reactor, which operated between 1974 and 1989, was rated at 842 MW(t) with an electrical output of 330 MW(e). The reactor core used prismatic (hexagonal) graphite (H-451) blocks for fuel elements surrounded by a graphite moderator and reflector. The fuel contained high enriched uranium (originally, greater than 93 wt% U-235) and thorium carbide fuel particles with TRISO coatings (i.e., three coating layers: an inner layer of high-density pyrolytic carbon (PyC), a layer of silicon carbide, and an outer layer of high density PyC) (see Helmreich, 2021).

High-temperature helium was used as the primary coolant to produce steam at $\sim 538^{\circ}\text{C}$. Helium (at 4.8 MPa and $\sim 400^{\circ}\text{C}$) was discharged from four helium circulators and passed down through the core, where it was heated to a temperature of $\sim 775^{\circ}\text{C}$. The coolant then passed through 12 steam generator modules. The entire primary coolant system, including the active core, steam generators and helium circulators (shown on the left of Fig. 1), was contained within a pre-stressed concrete reactor vessel (PCRV). A filtered, vented confinement building enclosed the PCRV with essential piping and cabling. Table 2 summarizes FSV plant characteristics.

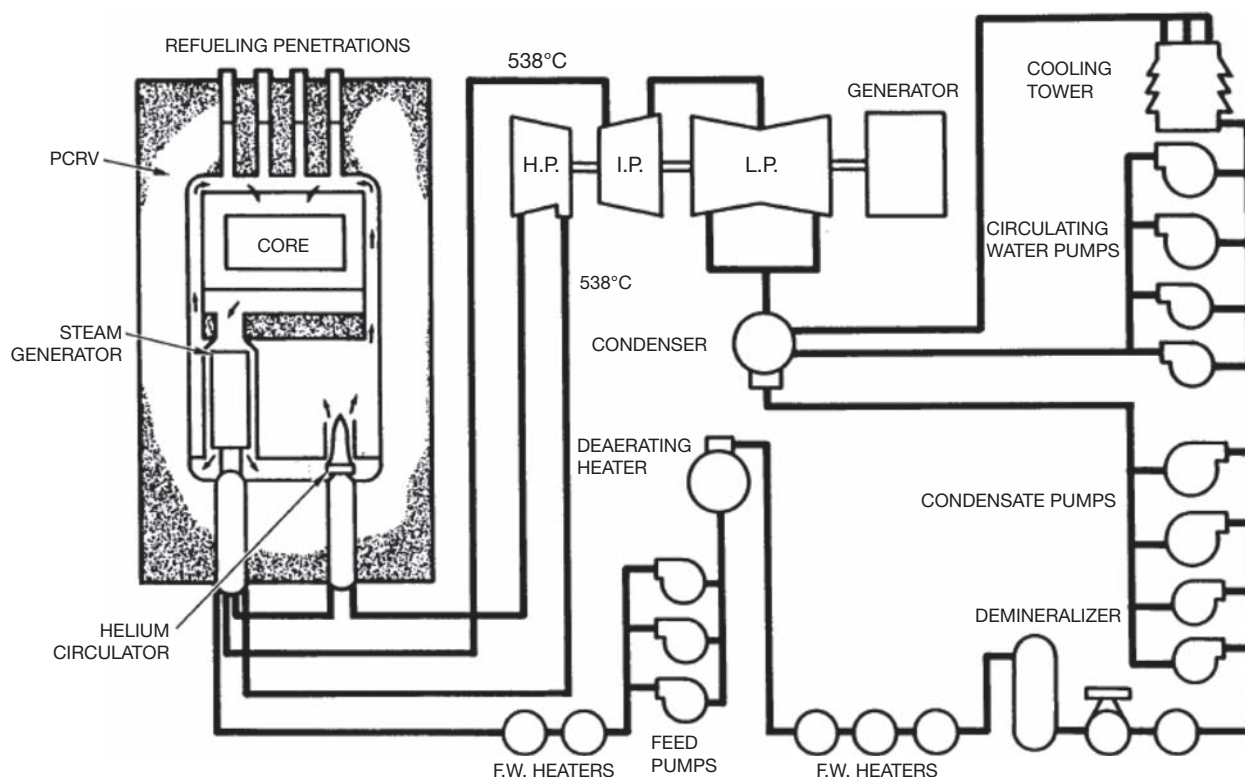


Fig. 1 Fort St. Vrain simplified flow diagram. Brey HL (2014) Initial start-up and testing of the Fort St. Vrain HTGR: Lessons learned which may be useful for the HTR-PM. In: *Proceedings of the HTR 2014 Conference*, Weihai, China, 27–31 October 2014, Paper HTR2014-71312. Available from <https://www.osti.gov/etdeweb/servlets/purl/20198167>.

Table 2 Fort St. Vrain design characteristics [based on information in Brey (2014) and Habush and Walker (1974)].

Parameter	Value
Thermal power	842 MWt
Net electric power	330 MWe
Power density	6.3 MW/m ³
Primary coolant	Helium
Pressure (inlet)	4.8 MPa
Temperature (inlet/outlet)	$\sim 400^{\circ}\text{C}/\sim 775^{\circ}\text{C}$
Steam temperature (main/reheat)	$538^{\circ}\text{C}/538^{\circ}\text{C}$
Fuel type	Hexagonal block
Fuel material	TRISO-coated high enriched uranium—thorium carbide
Primary system vessel	PCRVR with carbon steel liner
Total lifetime electric generation	~ 5500 GWh

After leaving the steam generator modules, steam entered the high-pressure (HP) turbine (see Fig. 1). After the HP turbine, the steam passed through a single stage turbine, which provided the driving force for the four helium circulators. This steam was then reheated to 538 °C in the upper (reheat) section of the 12 steam generator modules. The hot reheat steam was directed back to the intermediate-pressure (IP) turbine, then to the low-pressure (LP) turbine, and finally, to the condenser. Apart from the unique once-through steam generators, the secondary side of the plant was similar to a conventional fossil unit with 16.7 MPa main steam pressure, 538 °C main steam temperature, and 538 °C reheat steam temperature. The condensate system featured a full-flow demineralizer, three low pressure heaters, and a deaerator. Three boiler feedwater pumps directed the feedwater through two HP heaters and back to the steam generator modules.

With respect to deployment of a nuclear technology that could reap the economic and safety benefits associated with HTGRs, FSV operation led to several notable accomplishments:

- exceptionally good radiological cleanliness with low personnel exposures;
- outstanding safety features such as handling of temporary loss of forced cooling events without approaching core temperature limits;
- high cycle efficiencies (at that time, approaching highest in the power industry);
- generation of high-quality steam providing an excellent energy source for a multitude of process heat applications.

However, FSV experienced several operational issues associated with its FOAK systems and components (see [Initial Testing and Operation](#)).

Relevant operating experience

Although many features differed, FSV significantly benefitted from the design, construction, and operation of an earlier PRDP plant (see [Table 1](#)), the Peach Bottom Atomic Power Station Unit 1 (PB1). Using the PB1 experience, the design of FSV was modified to be a “proof-of-concept” demonstration that would provide the technical, regulatory, and operational bases for a larger commercial power plant design [Despite several potential plant owners using the FSV experience to prepare preliminary safety reports for 10 CFR 50 license applications (see [Ray, 2021](#)), all the FSV follow-on projects were canceled and GA withdrew the large HTGRs from the commercial market in 1975 (see [Moore et al., 1982](#))]. Significant aspects of the PB1 and other selected international HTGR experiences are highlighted here; more detailed information about HTGR demonstrations may be found in [Tsevekot \(2020a\)](#).

Operated by PECO, PB1 was a 40 MW(e) HTGR demonstration plant situated 80 miles southwest of Philadelphia on the Susquehanna River. The PB1 Nuclear Steam Supply System, which was designed by GA, included a helium-cooled, graphite moderated, 115 MW(t) reactor core operating at high temperature on a uranium-thorium fuel cycle. The reactor core consisted of 804 graphite fuel elements oriented vertically in a closely packed circular pattern within the reactor vessel. Each fuel element contained 30 annular fuel compacts comprised of fuel particles in a graphite matrix material. Similar to FSV, the fuel kernels in these particles were mixed uranium-thorium carbide. For the first PB1 core (Core 1), fuel particles were coated with a single layer of PyC to prevent hydrolysis; for the second core (Core 2), fuel particles had BISO coatings (an inner layer of low density PyC and an overlayer of high density PyC) to retain fission products and preclude hydrolysis. Core 1 was designed for continuous operation until its end-of-life (approximately 900 full-power days), but it only accumulated 452 effective full power days before Core 2 was installed. The premature installation of Core 2 was required because cracks had formed in the graphite sleeves surrounding the fuel compacts in around 90 of the fuel elements. The cracked sleeves were caused by swelling of the fuel compacts due to irradiation-induced dimensional changes in the fuel particle coatings ([Steward, 1978](#)).

PB1 went critical in March 1966 and shutdown in October 1974. FSV benefitted from several notable PB1 accomplishments:

- data for validating core physics methods;
- demonstration of reliable reactor control system operation;
- demonstration of steam generator operation and performance;
- demonstration of the performance of several unique gas reactor systems;
- demonstration that the plant could be operated in a load-following manner;
- demonstration of station availability (excluding planned shutdowns for R&D programs, the station availability over its lifetime was 88%).

FSV also benefitted from international licensing and operating experience regarding materials and fuel performance. For example, there was experience related to graphite cracking phenomena from the United Kingdom and related to TRISO fuel performance and dust generation and transport phenomena from Germany.

Regulatory framework, review, and oversight

Both the construction permit and operating license granted to FSV were issued under Section 104(b) of the Atomic Energy Act (AEA) of 1954, as amended. FSV was under construction when the AEA was amended in 1970 (see [Bley and Malsch, 2021](#)). Consistent with a Class 104(b) operating license, the NRC allowed a generous amount of latitude in regulating FSV. However, this latitude was offset by delays associated with determining appropriate regulatory requirements.

The ACRS review of the FSV construction permit was controversial. In fact, the ACRS did not reach a unanimous position in their letter report (ACRS, 1968). Member Hendrie dissented because FSV did not have a containment building; although Member Okrent approved FSV construction, he expressed concern about having a larger reactor, similar to the FSV design, without a containment at more populated sites unless additional features were added to mitigate major accidents.

Applicable regulatory guidance was limited, particularly in the areas of system design and plant safety attributes. There were often unnecessary delays when attempting to apply LWR regulatory requirements to the FSV design. Consequently, interactions between PSC and NRC on licensing issues often required significant negotiations and considerable (often third-party) analyses to obtain resolution. This was particularly evident with new regulations related to fire protection and environmental qualification of electrical equipment. Brey (2014) emphasizes that such negotiations resulted in added cost, delays in plant operation, diminished credit for the unique HTGR safety features of FSV, and lack of confidence between the regulator and the licensee. Because the burden of proof of compliance with the intent of the regulation lies with the plant owner/operator, Fuller (1989) observed that it is better to anticipate, rather than react to, regulatory changes. Hence, PSC established a licensing group with the explicit purpose of evaluating and actively participating in proposed rulemakings, thereby facilitating future implementation of regulatory requirements.

Several references (Moses, 2006; Fuller, 1989; Beck and Pincock, 2011) emphasize the importance of independent regulatory review and oversight. Reviews of the FSV licensing basis detected several incorrect or inadequately documented items:

- The design, which included steam generators and helium circulators located in large-sized PCRV penetrations, was developed assuming these penetrations had double closures. Hence, the accepted frequency (10^{-7} per year) for a pressure-induced failure of a large penetration on an LWR pressure vessel was assumed. However, the frequency for such LWR vessel penetration failures was estimated assuming the vessel is inspected to requirements specified within the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. This wasn't the case for these large FSV PCRV penetrations.
- The design was developed assuming that the probability of turbine trip could be based on British Magnox reactor experience. However, the British Magnox reactors used multiple small turbine generators as opposed to the one large turbine generator used at FSV.
- The FSV FSAR included a base reactivity curve to estimate the large reactivity change observed following a major water ingress event in 1974. This base reactivity curve was not explained in any documentation nor in any other topical report, despite technical specifications recording it as a limiting condition for operation.

An independent regulator should ensure that the licensing basis is clearly understood and documented and that industry codes and standards are appropriately applied. Exceptions should be acknowledged, explained, and documented to avoid inaccurate safety evaluations or an unrecognized vulnerability.

Initial testing and operation

The initial startup and continued operation of FSV required significant instrumentation, technical support, and funding. Because of its unique design, special testing was initially performed to confirm design assumptions. Throughout its operation, additional testing was required to address unanticipated phenomena, resulting in a significant amount of lost electrical generation (Fuller, 1989; Brey, 2014; Habush and Walker, 1974). Training and licensing of plant operators also required special attention.

Initial testing

The initial testing program consisted of four independent phases. The first phase, Cold Check-Out testing, included validation of systems and associated instrumentation to assure conformance with design requirements. The second phase involved preliminary operating tests on systems and components as they were completed. Although it did not include the use of nuclear heat, it was a formal testing program to assure system and component operation was in accordance with design values. This phase of testing culminated with performance of a Hot Flow Test, which utilized the helium circulators to heat up the primary coolant system to operational temperatures. The latter two phases, which were part of the plant start-up testing program, were divided into two groups, the low power physics tests (Group A) starting at 2% power and the Rise-to-Power tests (Group B) cumulating at 100% power. The FSV testing program was the principle means for assuring plant nuclear safety using a License-by-Test basis with hold points at power levels of 2%, 5%, 30% and 70%.

Significant resources were required to complete this testing. For example, steam generator performance testing was performed at many steady state power levels. Within the steam generator, temperature and pressure measurements for the helium, feedwater, and steam were logged along with strain gauge readings. One of the 12 steam generator modules was highly instrumented with strain gauges and thermocouples in order to assess its performance throughout startup. Similar parameters were logged during and after nine transient conditions, which included trips and load shedding. The performance of an instrumented steam generator module with simulated plugging of one feedwater tube was evaluated. The steady state and transient performance of the turbine-generator was also tested. In addition, safety valve lifting and reseating pressures were tested.

During Rise-to-Power testing, FSV experienced several temporary loss of forced coolant events and reactor operational excursions (due to an unanticipated ingress of moisture into the primary coolant and movement of a portion of the reactor core). As the reactor approached 70% power, individual core region outlet temperatures fluctuated by as much as 40 °C, and power channel indications varied (one channel showed prompt power changes of up to 8% over 8–20 min cycles). After an extensive three-year testing and

analyses program, the cause of the fluctuations was determined to be thermal-induced movements of core components accompanied by periodic changes in bypass flows and crossflows of primary coolant helium. These fluctuations were eliminated by placing metal locking devices between the columns of fuel elements at the top of the core.

Training

Training of the FSV plant staff began in 1968 and was closely governed by AEC requirements (Brey, 2014; Habush and Walker, 1974). Because the reactor system had not previously undergone initial core physics and start-up testing, operators were required to secure a “Cold Senior License,” which involved more stringent requirements than licensing of operating personnel in a standard nth-of-a-kind nuclear power plant. The requirements set by the AEC on the FSV staff for achieving this license included training at a similar facility (PB1), training at FSV, and five eight-hour exams (three written and two oral) administered by AEC specialists.

Operating experience

The FSV operational history can be characterized by low availability and inconsistent power production. Several lessons were learned from this experience (Fuller, 1989; Brey, 2014).

FSV faced several challenges due to moisture ingress, not only from out-gassing of moisture from graphite as it was heated during plant startup but also from water-lubricated bearings in its FOAK helium circulators. Moisture ingress during shutdown conditions led to moisture being absorbed into graphite and insulation. One large water ingress event required the FSV helium purification system to remove large quantities of water, overwhelming the system’s capacity and leading to corrosion of helium circulating system components.

The FSV experience also provided important lessons regarding graphite performance. Changes in length, bowing, and cracking observed in graphite components were attributed to high fluence and temperature gradients.

The event that led to the premature FSV closure was severe cracking of the Incoloy-800 steam generator super-heater headers. Replacement of the headers was deemed too expensive to justify a plant restart. As observed by Moses (2006), one cannot assume that components will stay dry or not be exposed to corrosive chemicals used in the reactor system. Detailed metallurgy on all boundaries of the primary coolant system should be completed prior to installation and continuously evaluated during operation to avoid unnecessary extrapolation. Challenges to component integrity, such as fatigue, corrosion, and creep rupture, must be identified and addressed.

Complexity and technology advances

Although the FSV reactor design employed many of the fundamental principles that formed the basis for PB1, there were several significant differences: (1) PB1 had a 40-MWe power rating versus FSV had a 330-MWe power rating; (2) PB1 used a steel reactor vessel versus FSV had a PCRV; (3) PB1 used long, annular fuel elements with a solid graphite spine versus FSV used prismatic-block graphite fuel elements; and (4) PB1 used electric-motor-driven helium circulators with oil-lubricated bearings versus FSV used steam and water turbine-driven helium circulators with water-lubricated bearings. Ultimately, the added FSV design complexity led to its unavailability. In retrospect, several FSV references (e.g., Brey, 2014; Fuller, 1989) conclude it is prudent to limit complexity and adoption of technology advances associated with FOAK components and systems.

For example, consider difficulties associated with the helium circulator auxiliary system. Transients in this system were initiated by improper or inadequate response to control signals. Frequently, this resulted in relatively large quantities of circulator bearing water entering the primary coolant system. The subsequent cleanup of moisture from the primary coolant was a principle contributor towards low plant availability. In addition, failures in circulator bolts due to chloride stress corrosion cracking required a significant maintenance outage.

The FSV experience emphasizes the importance of using systems and components that are ruggedly simple in design and have a history of reliable operation and minimal maintenance. Where possible, safety-significant systems should be tested in a prototypic setup prior to installation. This is especially important for FOAK equipment for which there is limited availability of spare parts. This led Brey (2014) to recommend that advanced designers incorporate a generous over-build capability into systems and components. If there is extra margin in components, such as compressors, pumps, and motors, testing can be completed during operation with less chance for spurious trips.

Example 2—The Enrico Fermi Unit 1 (EF1) sodium cooled fast reactor

The world’s first commercial liquid-metal fast breeder reactor, which was developed as part of the AEC PRDP, was the Enrico Fermi Unit 1 (EF1) Nuclear Generating Station. In 1955, the Power Reactor Development Company (PRDC) was chartered to design, construct, and operate the reactor portion of EF1. The Detroit-Edison Company owned and operated the turbine-generator portion of the plant. Upon issuance of an AEC permit in 1956, construction was initiated to build EF1 near Monroe, Michigan (approximately 30 miles from Detroit, Michigan and 25 miles from Toledo, Ohio).

Design

The EF1 reactor was cooled by sodium and operated at essentially atmospheric pressure. Its first core (Core A) contained 26 wt% enriched metallic uranium fuel with zirconium cladding; with Core A, EF1 was designed to reach power levels of 200 MWt (67 MWe). The enriched uranium section of the core contained 105 driver fuel subassemblies (the term, subassembly, in a sodium-cooled reactor is analogous to the term, assembly, typically used in LWRs). The core was surrounded by 531 additional subassemblies containing depleted uranium. The core also contained two control rods and eight safety rods. With uranium and mixed plutonium-uranium oxide fuel (Core B and subsequent cores), EF1 was designed for operation at power levels of up to approximately 430 MWt (150 MWe).

The primary system was filled with sodium in December of 1960, and criticality was achieved in August 1963. The reactor was a loop design in which the liquid sodium primary coolant transferred its heat to secondary sodium in an external intermediate heat exchanger (see Fig. 2). The secondary sodium then passed through the tube side of three parallel steam generators and transferred heat to water on the shell side. The once-through steam generator design produced superheated steam, which turned the main turbine-generator. The core and primary loop were enclosed in a steel containment building, while the steam generators, turbine generator, and associated equipment were housed in a conventional building next to the containment. Table 3 summarizes EF1 plant characteristics.

With respect to deployment of a nuclear technology that could reap the economic and safety benefits associated with SFRs, there were several notable EF1 accomplishments:

- data for validating core physics methods;
- demonstration of performance and maintenance of sodium systems and components (including necessary design retrofits), such as large mechanical sodium pumps, the under-the-plug fuel handling system, sodium-heated steam generators, and intermediate heat exchangers;
- data for validating characteristics of thermal-hydraulic systems;
- demonstration of steam generator operation and performance;
- data to confirm predictions for subassembly radiation relaxation in high flux conditions.

However, EF1 also experienced several operational issues associated with its FOAK systems and components (see Initial Testing and Operation).

Relevant operating experience

SFR experience for EF1 was gained from several demonstrations: Experimental Breeder Reactor Unit 1 (EBR-I), Southwest Experimental Fast Oxide Reactor (SEFOR), and Experimental Breeder Reactor Unit 2 (EBR-II). In addition, the 6.5 MWe power generating Sodium Reactor Experiment provided experience with using sodium in a thermal nuclear reactor power system. Clearly, EF1 benefitted from prior and contributed to future SFR efforts.

Sited at what is now known as the Idaho National Laboratory (INL), EBR-I was designed to both breed plutonium and to produce electric power. On December 20, 1951, the 1.2 MWt reactor went critical and lit four 200-watt light bulbs, becoming the world's first fast reactor to generate electricity. EBR-I was fueled with weapon-grade (94 wt%-enriched) uranium. On June 4, 1953, the AEC announced that EBR-I had become the world's first reactor to demonstrate breeding of plutonium from uranium. Unfortunately, the reactor was designed with a prompt positive power coefficient of reactivity (fuel bowing led to increases in reactivity and power). On November 29, 1955, during an experiment to obtain information about this instability, the reactor had a partial (40–50%) core meltdown. The damaged core was removed, and the reactor was repaired and restarted in 1957 using a modified core design to preclude rod bowing. EBR-I continued to operate until it was shutdown in 1963.

SEFOR was an experimental fast breeder reactor located in Cove Creek Township near Strickler, Arkansas. Operating from 1969 to 1972, SEFOR was privately operated by General Electric and funded by the U.S. government through Southwest Atomic Energy Associates, a nonprofit consortium formed by 17 U.S. and European organizations. Using mixed plutonium-uranium oxide fuel, SEFOR generated 20 MWt of heat but no electricity. It was constructed to provide reactor physics data (including a demonstration of the Doppler response mechanism during initial operation) and to test inherent safety features of the oxide fuel/sodium cooling configuration. SEFOR provided thermal expansion data for characterizing the core response during accidents, confirming that such thermal expansion would stabilize the core. SEFOR was also used to obtain physics and engineering data at fuel compositions, temperatures, and states characteristic of power reactor operating conditions. In addition, SEFOR provided data for determining the Doppler coefficient of reactivity at temperatures up to fuel melting.

EBR-II was a 62.5 MWt (20 MWe), sodium-cooled, “pool-type” reactor, i.e., the heat exchangers for transferring heat to a secondary loop of liquid sodium were submerged in the reactor vessel. Also sited at what is now known as INL, EBR-II construction started in 1958. Criticality at low power without coolant was achieved on September 30, 1961; criticality with sodium coolant was achieved on November 11, 1963; and design power was achieved on September 25, 1969. EBR-II demonstrated the feasibility of a sodium-cooled fast breeder reactor operating as a power plant. Initially operating with metallic high enriched uranium fuel, EBR-II featured an adjoining facility that permitted continuous reprocessing and recycling. From 1964 until 1969, this facility processed spent fuel and fabricated fresh fuel.

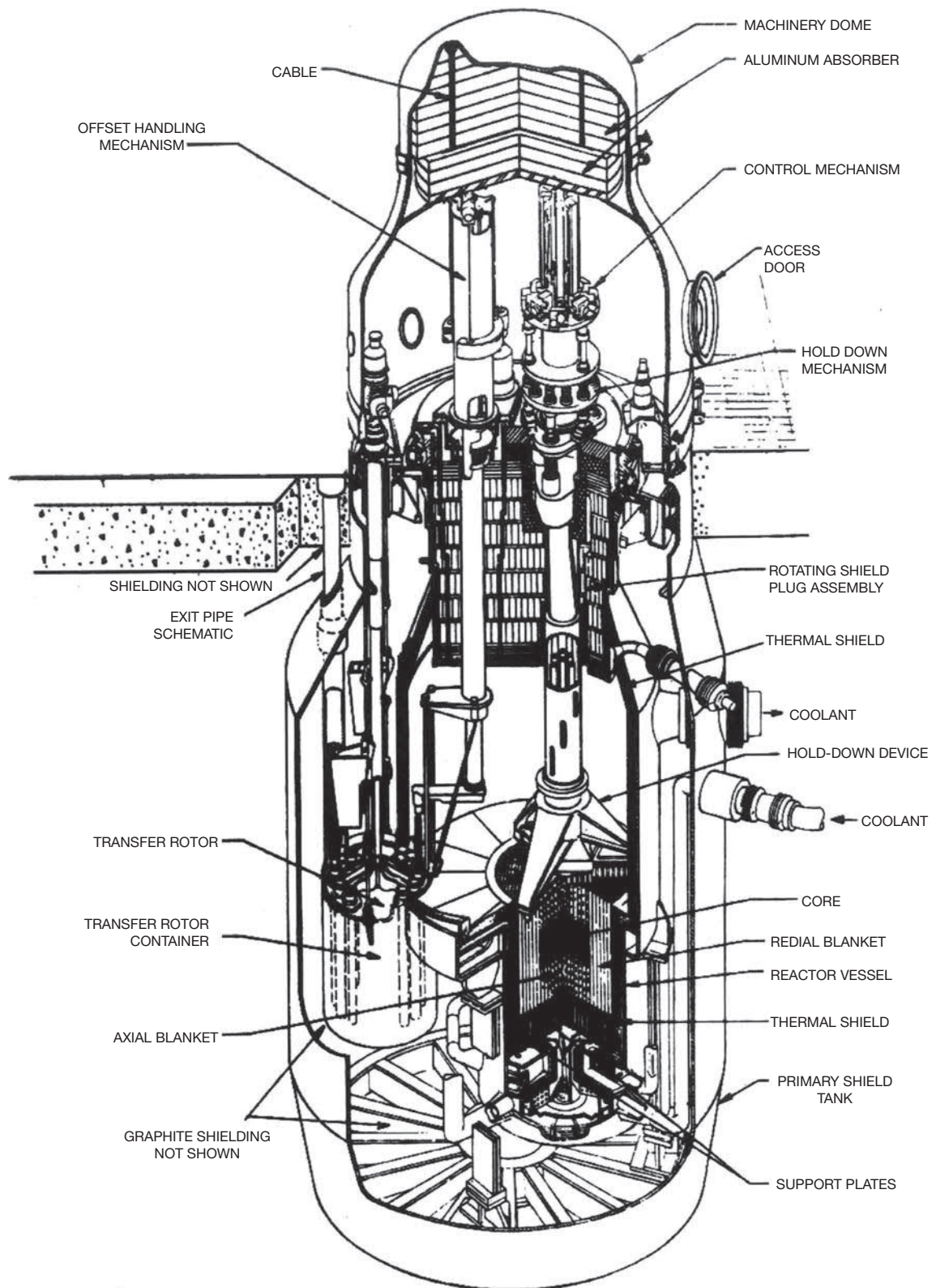


Fig. 2 Enrico Fermi Unit 1 reactor. DTE Energy Company (2014) *Submittal of Revision 8 to the Fermi 1 Safety Analysis Report*. Letter from V. A. Kaminskis, Site Vice President, to U.S. Nuclear Regulatory Commission Document Control Desk, p. 17 of Enclosure 1.

Table 3 Fermi Unit 1 design characteristics [based on information in PRDC (1970, 1971) and US AEC (1974)].

Parameter	Value
Reactor power	~430 MWt (final design); 200 MWt (Core A ^a)
Net electric power	~150 MWe (final design); 67 MWe (Core A ^a)
Primary coolant (final design)	Sodium
Pressure (inlet/outlet)	0.83 MPa/0.14 MPa
Temperature (inlet/outlet)	316 °C/482 °C
Fuel type	Pin
Fuel	U-10% Mo (Core A ^a); UO ₂ and PuO ₂ -UO ₂ (Core B, etc. ^a)
Primary system configuration/housing	Loop/stainless steel reactor pressure vessel
Total lifetime electric generation	32 GWh (as of November 1972)

^aCore A was installed at startup (1963); proposed cores with oxide fuels were never installed.

Regulatory framework, review, and oversight

When the EF1 construction permit application was reviewed, no formal design or siting criteria and little preliminary design information were available. Relevant SFR operating experience was limited.

The January 1956 application for a construction permit to build the EF1 plant proved particularly contentious. In their review of the application, the ACRS concluded “there is insufficient information available at this time to give assurance that the PRDC reactor can be operated at this site without public hazard” (US NRC, 2002). The ACRS expressed concern that questions regarding the reactor’s safety could be resolved within PRDC’s proposed schedule for obtaining an operating license. The ACRS urged the AEC to expand fast breeder reactor experimental programs and obtain data to address issues raised during its review.

Public controversy that arose during congressional hearings regarding the EF1 application for a construction permit had far-reaching impact. In meetings with the chairman of the Joint Committee on Atomic Energy, AEC Commissioner Thomas Murray disclosed ACRS concerns about the EF1 application to support a request for additional R&D funds. The Joint Committee promptly requested the ACRS report, claiming the AEC had failed to keep them “fully and currently informed” as required by the 1954 Atomic Energy Act (US NRC, 2002; Okrent, 1981). Based on a more optimistic review by their staff, the AEC commissioners issued a construction permit by a vote of three to one (Murray was the lone dissenter). This and other on-going activities at the time formed the basis for 1957 legislative action to create a more open licensing process. The existing ACRS was made a statutory body with the responsibility to independently produce public reports on licensing cases. In addition, this incident led the Joint Committee to start questioning whether AEC regulatory responsibilities should be separated from developmental and operational functions (Mazuzan and Walker, 1997).

Approval for startup and operation of EF1 was also delayed because of several other factors: a sodium/water reaction incident due to steam generator testing (Stone et al., 1981); an unsuccessful lawsuit to halt the plant because of its siting in a populated area (Okrent, 1981); and as discussed in **Initial Testing and Operation**, a flow blockage incident that led to melting in two fuel assemblies. This latter incident was attributed to zirconium plates, which were added late in the design process and not shown on EF1 construction drawings. If an independent regulator had been involved, this event (see **Initial Testing and Operation**) might have been avoided because the design change would have been subjected to additional scrutiny and quality control requirements.

Initial testing and operation

EF1 first achieved criticality on August 23, 1963. During its first years of operation, an extensive testing program was conducted at low power levels to confirm reactor physics and thermal hydraulic method predictions. Power ascension testing above 1 MWt didn’t commence until December 29, 1965, when a license was received from the AEC to operate EF1 at power levels up to 200 MWt. Successful completion of the low-power testing program enabled operators to increase reactor power level to 10% (and later to 50%) for additional testing programs.

During a June 1966 test, operators observed that outlet temperatures for one fuel element were 20–25% higher than other fuel elements but values were dismissed as faulty thermocouple readings. During an August 1966 run at 50% power, operators observed that coolant outlet temperatures above two subassemblies were 40–47% above outlet temperatures from other subassemblies. During an October 1966 test, the operators observed several inconsistencies in plant instrumentation readings when the reactor power level reached 15% power. In addition to high outlet temperatures in several fuel elements, the control rods were more withdrawn than expected for this power level. At 15:09 on October 5, radiation monitors in the ventilation exhaust ducts from the reactor building alarmed and automatically shut dampers to isolate flow to the environment. Radiation monitors in four other plant areas were also indicating high values. The operators shut down the reactor, which was at 31 MWt, precluding any abnormal releases to the environment.

In September 1967, investigators lowered a periscope into the reactor and discovered an unknown metallic object had blocked sodium coolant from reaching the fuel. Later, the metal object was identified as a zirconium plate that had been installed in the reactor as a safety measure (as noted in **Regulatory framework, review, and oversight**, these zirconium plates were added late in the design process). During reactor operation, the plate became unfastened and obstructed sodium coolant flow to some fuel subassemblies. Investigations revealed damage had occurred in four fuel subassemblies, with significant melting in two subassemblies.

Damage to the reactor and fuel assemblies took approximately four years to repair. In May 1970, the reactor was ready to resume operation, and the reactor finally reached a power level of 200 MWt in October 1970. During 1971, EF1 only generated 19.4

gigawatt-hours (GWh) of electricity, corresponding to an average capacity factor of 3.4%. PRDC made the decision to decommission the plant in November 1972.

Complexity and technology advances

The EF1 differed from and was much more complex than first-generation LWRs. This added complexity was accepted because of potential economic benefits associated with breeder reactors; there was the potential to sell two products: the generated electricity and plutonium by-products. Hence, the increased complexity of EF1 was viewed as necessary to demonstrate the practical and economical use of the breeder reactor program.

During initial testing, several components, such as the check valves on the primary sodium pumps and the steam generators, were modified. As described by [Stone et al. \(1981\)](#), the steam generators, however, continued to be plagued with leaks at the tube-to-tubesheet weld joints. In addition, instrumentation indicated instability within the steam generators. Ultimately, all tube-to-tubesheet joints in the water header of all three steam generators were rewelded to make the units leak-tight; and flow restrictors were inserted into each water header tube to equalize the flow through the tubes and stabilize their performance.

In response to the 1966 flow blockage and fuel melting event, flow guards were added to the nozzles of all core and inner radial blanket subassemblies. In addition, PRDC incorporated improved neutron detectors with faster response times and an improved data acquisition and control system.

Recent actions to address non-LWR regulation issues

Over the past several decades, the NRC has engaged in several pre-application interactions with prospective applicants for non-LWR designs. These engagements included pre-application reviews of high-temperature gas cooled and sodium-cooled reactors. In 2010, the NRC staff identified several potential advanced reactor policy, licensing, and technical issues, such as: principal design criteria selection; licensing basis event selection; accident radiological source term development; containment functional performance criteria; emergency preparedness requirements; staffing requirements; safety-related systems, structures, and components selection; and physical security requirements (see [US NRC, 2010](#)). Since that time, the NRC staff has worked with stakeholders to address these issues and to establish NRC policy when enough information (i.e., clear industry proposals with supporting design detail, data, and analysis) is available to support decision-making. In more recent years, NRC advanced reactor activities are focussed on actions required by NEIMA ([CRS, 2019](#)) and to prepare for anticipated near-term non-LWR submittals. NEIMA Section 103 requires that the NRC “complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications.” [Bley and Malsch \(2021\)](#) and [Fleming \(2021\)](#) discuss recent actions by the NRC staff and industry to address non-LWR issues. As these actions are implemented, it is important to consider difficulties encountered during prior non-LWR licensing, startup, and operating activities.

Summary

This chapter reviews two examples of FOAK non-LWRs that were authorized by the predecessor of the NRC, the AEC, as R&D facilities under Section 104(b) under the provisions of 10 CFR 50. In deploying and operating a non-LWR under this more flexible Section 104(b) process, the FSV and EF1 examples provide several important lessons in four areas:

- relevant operating experience
- regulatory framework, review, and oversight
- initial testing and operation
- complexity and technology advances

This chapter also summarizes recent actions taken by the NRC and industry to address issues identified during these and other non-LWR licensing activities. These actions support NRC activities to establish a new regulatory framework for optional use in future commercial advanced nuclear reactor license applications. Careful evaluations of issues encountered during licensing, as well as during startup and operation of FOAK reactors, offer insights that may be used to improve this new framework.

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Overview of International Nuclear Safety Regulation and Licensing

Laurence G. Williams, Centre for Nuclear Engineering, Imperial College London, London, United Kingdom

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Glossary

Nuclear Licensing The activities undertaken by a Regulatory Body in order to grant a license to permit the construction, commissioning, operation, and decommissioning of a nuclear installation.

Nuclear Regulation The inspection, assessment, permissioning and enforcement activities necessary to control the siting, design, construction, commissioning, operation and decommissioning of nuclear installations.

Nuclear Safety All the activities that are required to be undertaken to protect workers, the public, and the environment from the effects of ionizing radiation as a consequence of operating nuclear facilities during both normal and accident conditions.

Nuclear Security All the activities that are required to be undertaken to prevent the theft of and illegal transfer of nuclear or radioactive materials and to prevent unauthorized access to or sabotage of nuclear facilities.

Nuclear Site License A legal document to permit specified activities on a designated nuclear site.

Radioactive Waste Radioactive materials that are of no further use and destined for interim storage and eventual disposal.

Regulatory Body The body within a State with the legal authority to conduct the regulation of nuclear installations including the inspection, assessment, permissioning and enforcement activities necessary to control the siting, design, construction, commissioning, operation and decommissioning of nuclear installations.

Spent Fuel Nuclear fuel that has been removed from a reactor when it has come to the end of its useful life and is no longer required for power generation.

Introduction

It is clear that the international community regards the regulation of nuclear safety (and nuclear security) to be a national responsibility (IAEA, 1994, 1997, 1980, 2016a). Nuclear safety regulation and nuclear security regulation are in essence nuclear law enforcement; and because there is no universal legal system that has jurisdiction in all countries, there can be no single nuclear safety or nuclear security regulatory system or regulatory body. However, the radioactive fallout from a nuclear accident can have international consequences; and hence, the international community via the International Atomic Energy Agency (IAEA), the OECD/Nuclear Energy Agency (NEA) and other international organizations have developed international requirements and guidance to help countries with, or aspiring to have, nuclear power programs to develop a consistent approach to nuclear safety and security regulation and nuclear facility licensing.

This article provides an insight into the international legal framework for nuclear safety regulation and the licensing of nuclear installations. It also provides an overview of the regulatory approaches taken in a number of countries with significant nuclear power programs. The regulatory approach in the United States is excluded from this article because it is described in Bley and Malsch (2021a,b) and Ray (2021). In addition, more detailed information on nuclear safety regulation can be found in Rempe and Williams (2021).

Nuclear Law

Through the IAEA, the international community has developed a legally based international nuclear law architecture to not only promote the importance of nuclear safety and its regulation, but also to assist states in the development of robust legal systems to adequately control the peaceful use of nuclear energy. Nuclear law can be regarded as “the body of special legal norms created to regulate the conduct of legal or natural persons engaged in activities related to fissionable materials, ionizing radiation and exposure to natural sources of radiation” (IAEA, 2003). The objective of nuclear law is to provide a legal framework for conducting activities related to nuclear energy and ionizing radiation in a manner that adequately protects individuals, property, and the environment.

The IAEA identify a number of fundamental principles that can be used to distinguish nuclear law from other aspects of national law. The 11 principles of nuclear law are:

- The safety principle;
- The security principle;
- The responsibility principle;
- The permission principle;
- The continuous control principle;
- The compensation principle;
- The sustainable development principle;
- The compliance principle;
- The independence principle;
- The transparency principle; and
- The international co-operation principle.

While all these principles are important, those that are key to the delivery of effective nuclear safety regulation and licensing are outlined below.

The Safety Principle

The safety principle reflects the fact that safety is paramount and unless the use of nuclear energy is safe its use should be prevented. Therefore, given the special nature of the risks that arise from the use of nuclear energy, the primary objective of nuclear law is to promote the exercise of caution and foresight to prevent damage. In addition, the fundamental purpose of any regulatory regime is to balance the social risks and benefits. It is therefore important that the risks associated with nuclear energy are well understood. It is accepted that the stringency of legal arrangements should be related to the potential for harm, i.e., a graded approach.

The Security Principle

The security principle reflects the potential for harm to society that could arise from the theft or diversion of nuclear materials or technologies. For these reasons special legal measures are required to protect nuclear materials and installations. This principle relates to both physical security and safeguards issues.

The Responsibility Principle

This principle primarily reflects the need for someone to be responsible for ensuring safety. In the case of nuclear law this responsibility falls on the licensee that has been granted authority to carry out nuclear related activities. The licensee also has the responsibility to carry the burden of financial liability for any damage that may be caused by its activities. In the context of the "responsibility principle," the licensee is responsible for safety, those aspects of security that are within its control, and the protection of the environment.

The Permission Principle

Where an activity has the potential to cause harm to people or the environment, it is appropriate for the law to require that permission to carry out such an activity is obtained by the licensee before the activity is undertaken. Given the special risks associated with the use of nuclear energy, nuclear law normally requires permission to be obtained before activities involving fissionable material are undertaken. Permission can be in a variety of forms: such as a license, an authorization, a permit, an approval, a consent, etc.

Continuous Control Principle

This principle reflects the fact that a permission to do something is not a one-off activity. The person giving the permission, normally the Regulatory Body, will need to be convinced that the licensee's activities are being continuously conducted in a safe and secure manner. It is through this principle that the nuclear law requires that the Regulatory Body is given the right of access to nuclear facilities in order to effectively monitor the licensee's activities.

The Independence Principle

Nuclear law places particular emphasis on the fact that regulatory decisions relating to nuclear safety are not subject to undue influence or interference from those entities that have a responsibility for the promotion and use of nuclear energy. The Regulatory Body must therefore be independent of the nuclear industry and parts of government that have a role in the promotion of the nuclear industry.

The Transparency Principle

In some countries, the nuclear industry grew out of military programs; and hence, the users of nuclear energy have therefore often been seen as secretive. The transparency principle requires that those who are involved in the regulation, development or use of nuclear materials should make available information on how nuclear materials are being used and on accidents that have occurred that could have an impact on human health or the environment.

Nuclear Related Conventions

The IAEA has developed a number of international conventions in the area of nuclear safety and nuclear security in order to provide the international community with the necessary confidence that nuclear safety and nuclear security is being delivered to consistent standards. The key Conventions are: the Nuclear Safety Convention (IAEA, 1994), the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA, 1997), and the Convention on the Physical Protection of Nuclear Materials (IAEA, 1980, 2011, 2016a).

Nuclear Safety Convention

The Convention on Nuclear Safety was adopted in Vienna on 17 June 1994 and came into force in the autumn of 1996. The Convention was introduced following the 1986 accident at Chernobyl (Sich, 2021) with an aim to raise nuclear safety standards worldwide.

The Convention aims: to achieve and maintain a high level of nuclear safety worldwide through national measures and international cooperation; to establish and maintain effective defenses in nuclear installations in order to protect individuals, society and the environment from the harmful effects of ionizing radiation; and to prevent accidents [*Safety Principle*]. The Convention applies to the safety of nuclear installations.

Contracting Parties to the Convention are required to implement the obligations set out in the Convention through their legal and regulatory frameworks. This framework should provide for: the establishment of national safety requirements and regulations; a licensing system for nuclear installations [*Permission Principle*]; the prohibition of the operation of a nuclear installation without a license; a system of regulatory inspection and assessment of nuclear installations [*Continuous Control Principle*]; and the enforcement of regulations and license conditions.

The Contracting Party is also required to establish a Regulatory Body to implement its national legislative and regulatory framework. The Contracting Party is required to ensure that the Regulatory Body is provided with adequate authority, competence, and financial and human resources, in order to fulfill its responsibilities. The Contracting Party is also required to ensure that there is an effective separation between the Regulatory Body and any other organization that is concerned with the use or promotion of nuclear energy [*Independence Principle*].

The Convention requires the Contracting Party to ensure that the responsibility for nuclear safety rests with the holder of the relevant license [*Responsibility Principle*] and that all organizations engaged in activities directly related to nuclear installations give due priority to safety.

The requirements of the Convention are based to a large extent on the principles contained in the IAEA Safety Fundamentals document "The Safety of Nuclear Installations" (IAEA, 1993).

Joint Convention on the Safety of Spent Fuel and the Safety of Radioactive Waste Management

The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA, 1997) was adopted on 5 September 1997, and it entered into force on 18 June 2001. The key aims of the Joint Convention are: to achieve and maintain a high level of safety worldwide in spent fuel and radioactive waste management; to ensure that during all stages of spent fuel and radioactive waste management there are effective defenses against potential hazards so that individuals, society and the environment are protected from harmful effects of ionizing radiation, now and in the future, in such a way that the needs and aspirations of the present generation are met without compromising the ability of future generations to meet their needs and aspirations; and to prevent accidents with radiological consequences.

Contracting Parties to the Joint Convention are required to implement the obligations set out in the Convention through a legal and regulatory framework. The framework should provide for: national safety requirements and regulations for radiation safety; a system of licensing of spent fuel or radioactive waste management activities; the prohibition of the operation of spent fuel or radioactive waste facilities that do not have a license; appropriate institutional control; regulatory inspection; enforcement powers and allocation of responsibilities. The regulatory framework is expected to ensure that at all stages, both spent nuclear fuel and radioactive waste is managed in such a way that individuals, society and the environment are adequately protected against radiological hazards.

A Contracting Party is required to establish a Regulatory Body to implement its national legislative and regulatory framework. The Contracting Party is required to ensure that the Regulatory Body is provided with adequate authority, competence, and financial and human resources, in order to fulfill its responsibilities. The Contracting party is also required to ensure that there is an effective independence of the Regulatory Body in relation to any other organizations that are involved in both spent fuel and radioactive waste management. The national Regulatory Body is required to have powers to license spent fuel and radioactive waste management facilities to enable the control of the siting, design, safety assessment, construction, commissioning, operation and decommissioning of such facilities. In relation to radioactive waste or spent fuel disposal facilities, the Convention requires a Contracting Party to preserve records of the location, design and radioactive inventories of such facilities.

Convention on the Physical Protection of Nuclear Materials and Facilities

The Convention on the Physical Protection of Nuclear Materials was adopted in 1980 and came into force in 1987. However, it was limited to the physical protection of nuclear materials during international transport. It did not cover the protection of nuclear facilities or nuclear materials in domestic use, storage or transport. In 2005 an amendment to the Convention (IAEA, 2005a,b) was adopted to make it legally binding on States to establish, implement and maintain an appropriate physical protection regime for nuclear materials and nuclear facilities under their jurisdiction. This amendment came into force in 2016 (IAEA 2016a).

Guidance (IAEA, 2011) on how to implement the revised Convention outlines 12 fundamental principles relating to the security of nuclear materials and nuclear facilities. It is the responsibility of the State to establish, implement and maintain a physical protection regime. This regime should include establishing and maintaining a legislative framework to govern physical protection;

establishing a Competent Authority which in effect is similar to the nuclear safety Regulatory Body, ensuring that the prime responsibility for implementing physical protection of nuclear material and nuclear facilities rests with the licensee; and ensuring that priority is given to the establishment of an effective security culture. However, the responsibility for nuclear security is placed on the Government of the Member State.

IAEA Requirements and Guidance

To support the nuclear safety Conventions, the IAEA has produced extensive guidance on nuclear law (IAEA, 2003) and established a comprehensive set of nuclear safety standards and guides [see the IAEA website (www.iaea.org) for further information]. The IAEA Safety Standards have three main tiers: Fundamentals; Requirements; and Guides. The Fundamental Safety Principles (IAEA, 2006) provide the high-level aims that Member States using or aspiring to use nuclear energy should comply with. These principles define the roles and responsibilities of the licensee, the regulator, and the government. In the second tier there are General Safety Requirements (GSRs) (7 parts) and Specific Safety Requirements (SSRs) (6 parts) such as SSR 2 that addresses the safety of nuclear power plants (NPPs). In relation to regulation and licensing GSR 1—Governmental, Legal and Regulatory Framework for Safety (IAEA, 2010) is the key document.

The consistent theme of the international nuclear safety and nuclear security framework is the protection of people and the environment from the harmful effects of ionizing radiation. As a result, most Regulatory Bodies indicate that their primary mission or safety objective is the protection of people and the environment. The Regulatory Bodies also provide assurance to their Government that nuclear energy is being used safely and securely and in accordance with the legal requirements.

European Union—Nuclear Safety Directives

Nuclear safety

The European Union has competence in relation to nuclear safety; and hence, EU Member States are subject to The Nuclear Safety Directive (EU, 2009, 2014). The objectives of the Nuclear Safety Directive (EU, 2009) are to establish a European Community wide framework to maintain and promote continuous improvement of nuclear safety and its regulation; and to ensure that EU Member States have appropriate national arrangements to provide a high level of nuclear safety to protect workers and the public from the dangers arising from ionizing radiations from nuclear installations. The Directive places a number of obligations on EU Member States including the requirement to establish and maintain a national legislative and organizational framework for nuclear safety at nuclear installations. This “national framework” is required to establish responsibilities for the adoption of national nuclear safety requirements; and the provision of a licensing regime, a system of nuclear safety supervision, and enforcement actions.

The Directive in many ways mirrors the IAEA Nuclear Safety Convention in that it requires EU Member States to establish and maintain a competent regulatory authority, ensure that the regulatory authority is functionally separate from bodies associated with the promotion or use of nuclear energy and that the responsibility for nuclear safety rests with the licensee of the nuclear installation.

In 2014, the Nuclear Safety Directive was amended (EU, 2014). This amending Directive provided more substantive technical rules in the field of nuclear safety including safety culture. It also added more detail on the obligations of EU Member States including additional specific obligations relating to an EU Member State’s national nuclear safety framework. These new obligations provided more focus on the nuclear safety objectives for nuclear installations, how the new obligations should be achieved, requirements for more detailed assessment of nuclear installations including periodic safety review, and on-site emergency preparedness. In addition to the peer reviews required by the initial Nuclear Safety Directive (EU, 2009), the amending Directive requires EU Member States to conduct topical peer reviews every 6 years on a safety issue that is selected jointly by the EU Member States, the intention being to provide a mechanism for shared learning. Although the new Directive, is more detailed than its predecessor, it again mirrors the intent of the IAEA Nuclear Safety Convention.

The amending Directive was adopted on the 8 July 2014 and entered into force on 14 August 2014. Member States were required to transpose the Directive into their National laws by 14 August 2017 and report to the European Commission on its implementation by July 2020.

Safe management of spent fuel and radioactive waste

In 2011 the EU introduced a Directive on the safe management of spent fuel and radioactive waste (EU, 2011). This Directive establishes a Community framework for ensuring responsible and safe management of spent fuel and radioactive waste. It ensures that EU Member States provide for appropriate national arrangements for a high level of safety in spent fuel and radioactive waste management. The scope of the Directive is similar to that of the IAEA Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management.

The Directive requires EU Member states to establish a national framework for spent fuel and radioactive waste management that includes a system of licensing of spent fuel and radioactive waste facilities and activities; appropriate regulatory controls and enforcement actions. EU Member States are required to establish a competent regulatory authority and to ensure that this authority has the necessary legal powers and human and financial resources to fulfill its obligations; and that the regulatory authority is independent of organizations concerned with the promotion or utilization of nuclear energy or with the management of spent fuel and radioactive waste. Like the IAEA Joint Convention (IAEA, 1997), the Directive requires that the prime responsibility for the safety of

spent fuel and radioactive waste management rests with the license holder. EU Member States are required to make and implement national policies for the responsible and safe management of spent fuel and radioactive waste. EU Member States are also required to have a national program with clear milestones to show spent fuel and radioactive waste will be managed over time, including concepts for the post closure period of a disposal facility.

Similar to the IAEA Joint Convention, EU Member States are required to produce national reports every 3 years. The Directive requires EU Member States to submit the national report to the European Commission.

Case Studies

The following case studies are used to illustrate how international requirements are implemented at a national level. These case studies focus on countries, other than the United States, that have substantial nuclear power programs. The case studies are structured to give an insight into each country's legal framework, regulatory body, regulatory approach, and licensing, inspection and enforcement methods.

Case Study 1: Nuclear Regulation in the United Kingdom

The United Kingdom's (UK) civil nuclear power program grew out of the UK's post-war military imperative of producing plutonium for nuclear weapons. The UK built two plutonium-producing reactors, the Windscale Piles, between 1947 and 1950. In 1953 the UK commenced the construction of the first of eight gas-cooled reactors that had been designed to produce both plutonium and electricity for the national grid. HM Queen Elizabeth opened the first of these reactors, located at Calder Hall in October 1956. The 1955 White Paper entitled "*A programme of nuclear power*" identified nuclear power as a strategically diverse source of electricity (UK, 1956), and the UK embarked on a nuclear power program that resulted in the construction and operation of 41 nuclear power reactors. The UK also developed the full fuel cycle capability including uranium enrichment, nuclear fuel manufacture, spent fuel reprocessing, and radioactive waste treatment and storage facilities. In addition to this civil nuclear power program, the UK carried out a major nuclear research program with Europe's first atomic pile opening in 1946. Five major nuclear research sites housed a variety of research reactors, including fast breeder reactors and experimental fuel cycle facilities. In the early years of nuclear development, the UK's nuclear activities were not subject to statutory regulation.

Legal framework

The UK's nuclear safety legislation has developed over the years (Salter, 2010; Tromans, 2010). Following the accident in 1957 at one of the Windscale Piles and with the expansion of the civil nuclear power program, the UK introduced its first "Licensing and Insurance" Act, the Nuclear Installations Act of 1959. This Act was subsequently amended in 1965 to take on board new international treaties relating to nuclear liability and insurance. The 1965 Nuclear Installations (as amended) Act (UK, 1965) was a licensing and insurance Act.

The Nuclear Installations (as amended) Act 1965

The key feature relating to nuclear safety and licensing of the 1965 Act was to provide regulatory control over the construction and operation of NPPs and the associated radioactive waste. The Act states that no person may use a site for the purpose of installing or operating a nuclear reactor or any other installation that has been prescribed in law, unless a nuclear site license has been granted. The power to grant licenses was initially given to Ministers but subsequent amendments transferred this power to the independent Regulatory Body, currently this is the Office for Nuclear Regulation (ONR). The 1965 NI Act also enables the government to prohibit certain operations except under permit. Under the Act a nuclear site license can only be granted to a corporate body and once granted, it is not transferable. A nuclear licensed site cannot be de-licensed and hence be removed from ONR's regulatory control, unless the licensee can demonstrate that there is "no danger" from ionizing radiations from anything on the site. ONR has the power to vary a nuclear site license and to attach conditions at any time that are necessary or desirable in the interests of safety with respect to the design, siting, construction, installation, operation and maintenance of any plant installed on the site. ONR also has the power under the 1965 Act to attach conditions to a nuclear site license in relation to the handling, treatment and disposal of nuclear matter, which includes radioactive waste.

ONR has the power to revoke a nuclear site license at any time. A nuclear site licensee may at any time surrender the license to ONR. However, even if the nuclear site license is revoked by ONR, or the licensee surrenders the license, the licensee retains responsibility for a "period of his responsibility." At any time during this period ONR has the power to give the licensee directions in order to prevent risk of injury to any person or damage to any property from ionizing radiations from anything remaining on the site. Hence, the licensee remains responsible for the site until ONR has granted a new license for the site to a new licensee, or the licensee can demonstrate that there is no danger from ionizing radiation from anything on the site.

Health and Safety at Work etc. Act 1974

The primary legislation governing Health and Safety in the workplace in the UK is the Health and Safety at Work Act etc. 1974 (HSWA) (UK, 1974). The HSWA introduced the concept of goal setting and proactive approaches to safety management. The objectives of the 1974 Act are: to secure the health, safety and welfare of persons at work; to protect others against risks to health and

safety arising from work activities; and to control dangerous substances and preventing the unlawful acquisition, possession and use of such substances. The common law principle of a “Duty of Care,” is enshrined in law with the HSWA. In the HSWA specific roles, responsibilities and duties are placed on the employer, the employee (this is applied very broadly—and may include some contractors and agency staff) and others (e.g., the emergency services and local authorities). It defines the enforcement powers and arrangements for prosecution, and the applicability of the legislation to the crown.

Under Section 2 of the HSWA, the employer is required to ensure, so far as is reasonably practicable, the health, safety and welfare at work of all his employees. Section 3 requires employers and the self-employed to ensure so far as is reasonably practicable the safety of persons other than employees, for example contractors, visitors and the general public. A key phrase in the Act is “so far as is reasonably practicable,” often abbreviated to SFAIRP. SFAIRP involves comparing the risk reduction achieved by a safety measure (or range of measures) with their cost (money, time, trouble). The measure(s) should be implemented unless the cost is grossly disproportionate to the risk averted. The burden of proof as to what is reasonably practicable rests with the employer.

The Ionizing Radiations Regulations 2017

The Ionizing Radiations Regulations (IRR17) (UK, 2017) implements the worker safety aspects of the Basic Safety Standards Directive (96/29/2013/59/Euratom). IRR17 covers radiation protection requirements in the UK. The regulating authority is the Health and Safety Executive (HSE). IRR17 identifies requirements to obtain authorization from and provide notification to HSE; to perform prior risk assessment; to provide and maintain personal protective equipment and engineering controls and to adhere to the dose limits specified in the regulations. These regulations also discuss arrangements for the management of radiation protection, for the designation of areas and classification of personnel, and for the control of radioactive substances. ONR Inspectors enforce these regulations on nuclear licensed sites and on defense nuclear sites that are not licensed under the NI Act.

The Radiation (Emergency Preparedness and Public Information) Regulations 2019

The Radiation (Emergency Preparedness and Public Information) Regulations UK (2019) implements the Euratom Basic Safety Standards Directive (2013/59/Euratom). The regulations establish a framework of emergency preparedness measures to ensure that members of the public are properly informed and prepared, in advance, about what to do in the unlikely event of a radiation emergency occurring and given information if a radiation emergency actually occurs.

The regulations require operators to identify all hazards that have the potential to cause a radiation emergency, and then evaluate the range of possible on and offsite consequences for a reflective range of all hazards identified. The key outcome is a consequence report to the local authority, with a recommended technical distance for the detailed emergency planning zone. The local authority uses this output to develop and implement the off-site emergency response plan that is specific to the nuclear site.

The regulations allow for a graded approach to planning for a site that is appropriate, proportionate and flexible to the risks identified in the hazard evaluation process. The regulations also require a review of hazard evaluation at least every 3 years and for a review to be undertaken where there is a material change to the work with ionizing radiation on the site. Emergency plans are required to be tested at least every 3 years. For nuclear licensed sites ONR is responsible for oversight of the regulations.

Energy Act 2013—Part 3 Nuclear Regulation

The Energy Act 2013 Part 3 (UK, 2013) sets up the Office for Nuclear Regulation (ONR). ONR is the UK’s nuclear safety Regulatory Body and the nuclear security Competent Authority. The Act defines the principle function of ONR, gives it the power to enforce relevant statutory provisions of the Nuclear Installations Act relating to licensing, reporting of and inquiries into dangerous occurrences, the appointment of inspectors, the provision of powers to inspectors, and recovery of expenses. The Act enables the Secretary of State to make regulations for the purposes of nuclear safety, nuclear site health and safety, and nuclear security, safeguards and transport of nuclear and radioactive materials. The Secretary of State has the power to ensure that sensitive nuclear information relating to activities carried out on or in relation to civil nuclear premises is protected in the interests of national security.

Regulator—Office for Nuclear Regulation (ONR)

The Office for Nuclear Regulation (ONR) is responsible for the regulation of nuclear safety and security across the UK (see www.onr.org.uk). ONR is a statutory corporation that is independent of the nuclear industry and independent of those parts of the UK Government that are responsible for the promotion of nuclear energy and the management of radioactive waste. ONR’s sponsoring department in the UK Government is the Department for Work and Pensions. ONR has a Board of Directors that has corporate responsibility for ensuring that ONR fulfills the aims set out in the “Framework Document between the Department for Work and Pensions and the Office for Nuclear Regulation” (ONR, 2018). ONR’s mission is to provide efficient and effective regulation of the UK’s nuclear industry, holding it to account on behalf of the public. ONR has five regulatory divisions: Operating Facilities; New Reactors; Sellafield Decommissioning, Fuel and Waste; Civil Nuclear Security and Safeguards; and Technical. ONR’s resources are set out in its 2016–20 Strategy (ONR, 2016a); at the end of 2020, ONR expects to have around 500 specialist staff in post and have a budget of £85.4 m. Of this £85.4 m, some £84 m is recovered from charging the nuclear industry and £1.4 m is a government grant.

Regulatory approach

The UK’s civil nuclear program is diverse, covering not only nuclear research facilities and activities, but also activities and facilities associated with the whole nuclear fuel cycle including the storage and disposal of radioactive waste. Hence, as the regulatory

framework developed, it became difficult to apply a prescriptive approach to regulation. In a prescriptive regime, the Regulatory Body prescribes the rules or regulations that the licensee must comply with. The range of types of facilities and technologies made it impossible for the Regulatory Body to develop a comprehensive set of safety requirements and associated regulations for all the facilities and activities. The UK therefore developed a “goal-setting” approach to the regulation of its nuclear facilities. In this approach, the legal framework sets the “nuclear safety goals” and it is the responsibility of the licensee to demonstrate compliance with the goals. This approach enabled the licensees to be flexible in determining how to meet the nuclear safety goals and hence tailor their nuclear safety arrangements to fit the nuclear activity and its associated hazards. Because this approach requires the establishment of specific safety goals and targets, it is necessary for the Regulatory Body to have highly professional and competent staff not only to develop meaningful safety goals and targets, but also to effectively evaluate the Licensee’s nuclear safety arguments.

Licensing and enforcement

ONR’s approach to licensing is set out in “Office for Nuclear Regulation—a Guide to Nuclear Regulation in the UK” (ONR, 2016b) and “The Licensing of Nuclear Installations” (ONR, 2019b). ONR regulates 37 licensed nuclear sites associated with operating nuclear power stations, NPPs under construction, fuel cycle facilities, waste management facilities and some defense-related activities. To obtain a nuclear site license for a new site to construct and operate an NPP or any other prescribed facility, the applicant is required to demonstrate to ONR that it is a fit and proper corporate body. In essence, it has to demonstrate that it is not only the “controlling mind” but also an “intelligent customer” for the services it will procure. It also has to show that it has sufficient financial resources to undertake the intended activities and liabilities. The applicant is also required to produce a “management prospectus” that outlines its organization, quality management arrangements and its arrangements for compliance with the conditions attached to the nuclear site license (ONR, 2017). In the case of an NPP, the license applicant is also required to produce a “Pre-Construction Safety Report” (PCSR) for the NPP to be built on the site.

ONR will judge the adequacy of the license applicant’s documentation against its Safety Assessment Principles (ONR, 2014); when it is satisfied with the documentation, ONR will grant the applicant a nuclear site license. The Government introduced an optional process for the certification of new nuclear reactor designs. This Generic Design Assessment (GDA) (ONR, 2019a; EA, 2019) process is a contractual arrangement between a reactor vendor and two regulators—ONR and the Environment Agency for England and Natural Resources Wales. At the end of the process, if the regulators are satisfied with the adequacy of the design, the environment regulator will issue a Statement of Design Acceptability (SODA) and ONR will issue a Design Acceptance Certificate (DAC). These documents are given to the reactor vendor; should a license applicant propose to construct an NPP design with a DAC and SODA, the license applicant is still required to produce a site specific PCSR. In assessing the PCSR, the regulators will take account of the work done during the GDA process.

Once permission to commence construction on the nuclear licensed site has commenced, the licensee requires permissions from ONR and where appropriate, the environment agencies to commence inactive commissioning, the introduction of fuel onto the site for the first time, active commissioning, commencement of routine operation, periodic safety reviews, and eventual defueling and decommissioning. At each stage the licensee is required to produce appropriate documentation to accompany the request for permission to proceed. This documentation is required to substantiate the licensee’s safety, security and environmental protection arguments. The major documents produced are the Pre-commissioning Safety Report (PCmSR), the Pre-operation Safety Report (POSr) and the Pre-decommissioning Safety Report.

License conditions

At the heart of the UK licensing regime is the power to attach Conditions (ONR, 2017) in the interests of nuclear safety and radioactive waste management. There are currently 36 standard Conditions attached to all nuclear site licenses. These Conditions are a mixture of prescriptive requirements and goal setting objectives. In the main, the Conditions aim to envelope all the activities that are needed to effectively manage nuclear safety and ensure the safe design, construction, commissioning, operation, maintenance and decommissioning of a nuclear facility or facilities on the site. The Conditions also cover the production, minimization, conditioning, treatment and storage of radioactive waste. The goal setting Conditions generally require the licensee to make and implement adequate arrangements to address specific issues such as the control of construction of a new plant, modifications of an existing plant, or the training of people with safety related responsibilities. Hence, although there is a standard set of Conditions for each nuclear licensed site, the licensee’s arrangements will differ depending upon the activities being undertaken on the licensed site. ONR is responsible for assessing the adequacy of the licensee’s arrangements and can, if appropriate, “freeze” the licensee’s arrangements so that the licensee cannot change the arrangements without the permission of the ONR.

Regulatory inspection and enforcement

In the UK, inspection and enforcement activities are carried out through the use of designated site inspectors. The UK does not have inspectors who are resident on a nuclear licensed site. Typically, a designated site inspector will spend 30% of his or her time at the site. Much of the site inspector’s remaining time is spent writing up visit reports, reviewing the licensee’s documentation submitted to seek permission to undertake specific activities on the site, and working with other site inspectors and technical assessors (ONR, 2019b).

The site inspector will follow a site inspection program that enables ONR to check the licensee’s compliance with the license condition arrangements for the site. Inspections may also take place at the licensee’s corporate headquarters and elsewhere,

including contractor's premises and manufacturing sites. ONR also uses "team inspections" on particular topics such as witnessing the annual demonstration emergency exercise or the readiness inspection prior to the commencement of active commissioning.

ONR has a range of enforcement tools in the case of licensee non-compliance. These range from formal letters expressing concern, legally enforceable improvement or prohibition notices that can be served on the licensee, and the option of increased regulatory scrutiny. ONR also has powers to direct the shutdown of an NPP and prevent its re-start without ONR's consent. For serious cases of non-compliance, ONR can submit evidence of its investigations to the UK's national prosecution services for prosecution in the courts. In the UK, failure to comply with nuclear laws is a criminal offence.

Case Study 2: Nuclear Regulation in Canada

Canada has a mature and comprehensive nuclear power program with around 15% of its electricity coming from nuclear power (WNA, 2019). Canada is a leader in nuclear research and technology, has developed and exported NPPs, and has uranium mining, nuclear fuel manufacture and radioisotope production facilities. Nuclear energy has been regulated in Canada since 1946 when the Canadian Government established the Atomic Energy Control Board (AECB). The AECB was charged to provide for the "control and supervision of the development, application and use of atomic energy and to enable Canada to participate effectively in measures of international control of atomic energy." In 1954 the AECB was given responsibility for the regulation of Canada's atomic research and development activities that were under the control of the newly formed Atomic Energy of Canada Limited (AECL). The Canadian Nuclear Safety Commission (CNSC) replaced the Atomic Energy Control Board on 31 May 2000.

Legal framework

The OECD Nuclear Energy Agency provides a comprehensive overview of nuclear legislation in Canada in its document: Nuclear Legislation in OECD and NEA Countries—Regulatory and Institutional Framework for Nuclear Activities—Canada (OECD, 2011a). A list of relevant Acts and regulations is given in <http://nuclearsafety.gc.ca/eng/act-and-regulations/index.cfm>.

Nuclear Safety and Control Act

The Nuclear Safety and Control Act (Canada, 1997) is a comprehensive piece of legislation governing the use of nuclear energy in Canada. The Act established the Canadian Nuclear Safety Commission to regulate the development, production and use of nuclear energy and the production, possession and use of nuclear substances, prescribed equipment and prescribed information. The Act covers Canada's international obligations relating to the use of nuclear energy, national security, health, safety and environmental protection, and non-proliferation.

The Act gives the CNSC extensive powers to regulate the nuclear industry and make regulations in a wide range of activities involving nuclear energy and materials including the transport of nuclear materials, other prescribed substances, nuclear equipment and nuclear technology.

General Nuclear Safety and Control Regulations

The General Nuclear Safety and Control Regulations (Canada, 2000a) support the Nuclear Safety and Control Act. These regulations set out the requirements relating to the licensing process including exemptions. The regulations describe the obligations of licensees including the requirements: to have sufficient qualified staff; to protect people, to protect the environment; to maintain the security of the site; and to publish health and safety information. The regulations also impose obligations on workers including the requirement to comply with licensee's arrangements to protect people, the environment and national security.

The regulations set out what are prescribed nuclear facilities, equipment and information; give details on reports and records to be produced by the licensee; and details of inspector's certificates.

Radiation Protection Regulations

The Radiation Protection Regulations (Canada, 2000b) set out the obligations of licensees and workers. A licensee is required to implement a radiation protection program; keep records of radiation exposure; designate nuclear energy workers; and provide information to workers on radiation exposure levels. Dose limits are described along with the obligations on licensees.

Regulator—Canadian Nuclear Safety Commission (CNSC)

The CNSC is an independent organization that reports to the Canadian Parliament via the Minister of Natural Resources. It is independent from the Canadian nuclear industry. The CNSC is responsible for the regulation of nuclear energy and materials: to protect health, safety, security, and the environment; to implement Canada's international commitments on the peaceful use of nuclear energy; and to provide information on regulatory matters to the Canadian public (CNSC, 2019). The CNSC may have up to seven permanent members including the President and Chief Executive. Decisions made by the CNSC are not subject to review by the Minister. The CNSC recovers almost 70% of the cost of its activities from fee-paying applicants and licensees.

The CNSC consist of the President, members of the Commission and ~800 staff members. It has four main branches. The Regulatory Operations Branch is headed by the Executive Vice President and Chief Regulatory Operations Officer. The other three Branches, namely the Technical Support Branch; the Regulatory Affairs Branch; and the Corporate Services Branch are each headed by a Vice-President. The Regulatory Operations Branch has four directorates that are responsible for Power Reactor Regulation,

Nuclear Cycle and Facilities Regulation, Nuclear Substance Regulation, and Regulatory Improvement and Major Projects Management. The Directorate of Power Reactor Regulation regulates the development and operation of Canada's NPPs.

Regulatory approach

The CNSC operates within a regulatory framework with a hierarchy that has the Nuclear Safety and Control Act at the top and supported by regulations, licenses and certificates, and regulatory documents. Regulatory documents are key to the CNSC approach, and they explain to licensees and applicants what they must achieve in order to meet the requirements set out in the Nuclear Safety and Control Act and its supporting regulations. In the development of its regulatory documents, the CNSC consults the public and other stakeholders.

The CNSC uses licensing as its main regulatory approach. The CNSC can only issue a license for a nuclear installation if it is satisfied that the applicant is qualified to carry out the activity that is to be licensed and that the applicant will, *inter alia*, make provision for the safety, security, and the protection of the environment. The CNSC can also impose any term or condition on the license that it considers necessary for the purpose of the Nuclear Safety and Control Act. Regulations are used to place requirements on the licensee (Canada, 2000a). Inspectors have the powers to enter and inspect nuclear facilities, nuclear powered vehicles, vehicles carrying nuclear substances, prescribed equipment, or prescribed information. Inspectors also have the power to order a licensee to take measures that are considered necessary, *inter alia*, for the protection of health and safety, the protection of the environment and the maintenance of national security (Canada, 1997).

The CNSC established a "vendor-optional" process to assess reactor facility designs prior to the formal submission of a license application. The aim of this process is to give a vendor the opportunity to evaluate whether its reactor design will meet Canadian regulatory requirements. The process is not part of the formal licensing process and unlike other countries, the CNSC does not certify the design. However, it is useful in identifying fundamental barriers to the licensing of new designs in Canada.

A new NPP requires licenses from the CNSC for: site preparation, construction, and operation. The CNSC is not involved in the site selection process but the license applicant must demonstrate that the proposed site will not pose an unacceptable risk to health, safety, security, and the environment. The applicant is also expected to consider the facility's entire lifecycle. When applying for a license to construct, the applicant must demonstrate that the proposed NPP design meets the regulatory requirements. In addition, the applicant must demonstrate that the proposed design will be safe over the intended lifetime of the plant. In the case of a license to operate, the applicant is required to demonstrate that, *inter alia*, it has an appropriate safety management systems that will ensure the safe and secure operation of the plant. The applicant is also required to show that the inactive commissioning program has been satisfactorily completed and the reactor is ready to accept its first fuel load.

Information on decommissioning plans and financial guarantees are required early in the licensing process in the initial application. A licensing basis is used to define the set of requirements and documents that set the boundary conditions for acceptable performance and include such things as the safety control measures and the documents needed to support the license. Once a license has been granted by the CNSC, the information and commitments submitted with the license application have a legal status in relation to the requirements placed on the licensee. The CNSC uses a "permissioning" approach that requires the licensee to seek approval or consent from the CNSC in certain circumstances such as permission to proceed past a pre-defined hold point.

The CNSC uses a flexible approach when it comes to specifying the license period for a particular facility. In the case of NPPs, the CNSC is moving to aligning the operating license period with the 10-year periodicity of the periodic safety reviews. At the end of life for an NPP, the CNSC's licensing strategy is to require the licensee to apply for a 10-year decommissioning license that will be subject to renewal. At the end of the decommissioning/dismantling of the NPP, the licensee will be required to apply for a license to abandon the site.

Licensing and enforcement

Throughout the licensing process the CNSC is required to carry out assessments of the licensee's activities, and in order to define the scope of these assessments the CNSC uses a risk-informed approach.

Licensees are responsible for on-site emergency planning. The licensee is required to prepare a detailed emergency plan that includes the provision for emergency exercises to test the readiness of staff at the NPP to respond effectively to emergencies. The CNSC observes these exercises to be satisfied that the licensees are managing and implementing their emergency response plans effectively. For emergencies that have offsite effects, the licensee is expected to work with the appropriate municipal government, the principal/territorial governments and the federal government. Offsite plans are exercised, and in some cases the CNSC participates. The CNSC also observes these offsite exercises to confirm that the emergency response plan has been adequately implemented.

The CNSC's regulatory framework is largely non-prescriptive with clear goals, objectives and requirements together with associated regulatory guidance. The CNSC's regulatory documents are continuously updated and aligned with international standards. The regulatory framework for siting, design and construction of NPPs aligns with the IAEA safety standards (CNSC, 2016). Within the Directorate of Power Reactor Regulation, there are four regulatory program divisions (RPDs), each division being focused on an NPP site. In each RPD, there are permanently situated CNSC resident inspectors who carry out the day-to-day site inspection activities to check compliance with the licensing basis for the facility (CNSC, 2016). Technical staff that are located at the CNSC's Head Office are assigned to each RPD.

Inspectors are authorized to carry out inspections to check that the licensee is complying with the CNSC's regulatory requirements which may include license conditions. The CNSC's compliance program is informed by risk, looks at the implementation

of international agreements, and takes into consideration the previous record of the licensee in relation to regulatory compliance. Various inspection techniques are used including individual and team inspections on specific topics. Team inspections are usually led by the site office inspectors and include multi-disciplinary specialists from the CNSC head office. The results of inspections are given to the licensee. The baseline inspection program for an NPP is scheduled over a 5-year period; but if the licensee's performance falls below CNSC expectations, risk management is used to identify areas where additional inspection activities should be focused. Inspectors have the power to undertake unscheduled inspections.

The CNSC has a range of enforcement tools that includes written notices, increased regulatory scrutiny, requests from the Commission for information, orders, licensing actions, prosecutions, and administrative monetary penalties. The choice of the regulatory action is guided by the safety significance of the non-compliance.

Case Study 3: Nuclear Regulation in France

France has the largest nuclear power program in Europe with some 58 reactors on 19 sites and which generates over 70% of France's electricity production. In 2015 France introduced a new Act on energy transition and green growth (TECV Act 2015-992), this Act aims, *inter alia*, to reduce the nuclear share of electricity production to 50% by 2025. In addition to its nuclear power program, France has a full fuel cycle capability with facilities for fuel manufacture, uranium enrichment, fuel reprocessing, plutonium recycling and radioactive waste management. Through the CEA (Commissariat à l'énergie atomique et aux énergies alternatives), France has a large nuclear research capability (OECD, 2011b).

Legal framework

The OECD Nuclear Energy Agency provides a comprehensive overview of nuclear legislation in France in its document: Nuclear Legislation in OECD and NEA Countries—Regulatory and Institutional Framework for Nuclear Activities—France (OECD, 2011b). Further details are given in France's Seventh National Report for the 2017 Review Meeting of the IAEA Convention on Nuclear Safety (France, 2016) and France's Eighth National report for the 2020 Review Meeting (France, 2019).

The hierarchy of the French nuclear safety legal and regulatory framework flows down from European Union Directives, laws enacted by Parliament, decrees and orders given by Government, and technical regulatory and individual resolutions from the nuclear safety regulator. The safety regulator also produces non-legally binding guides. The Acts specifically relating to Basic Nuclear Installations (BNIs) are codified in the Environmental Code. The BNI licensing and regulation system is underpinned by some of the provisions of titles I, IV and the provisions of title IX of book V of the Code underpin the BNI licensing and regulation system (France, 2016, 2019).

The nuclear transparency and safety Act, TSN Act 2006 made extensive changes to the BNI legal system. This Act created a legislative framework for BNIs, allowing the regulation of each life cycle phase, and enabling monitoring, and the application of sanctions. It also allowed the safety regulator, *Autorité de Sûreté Nucléaire* (ASN) to become an independent authority. Article 4 of the TSN ACT 2006 gives ASN responsibility for monitoring nuclear safety and radiation protection, particularly monitoring compliance with the general rules and specific requirements relating to nuclear safety. The TSN Act requires BNIs to be licensed.

The licensing system for BNIs was set in Decree Number 2007-1557. For a license to be given, ASN is required to give its opinion; and after a public enquiry has been held, a decree must be issued by the Ministers responsible for Nuclear Safety. The person responsible for the operation of the nuclear installation lodges the application for a license with the Ministers. The operator is then required to send copies of the license application and the supporting documents to ASN. The supporting documents are expected to include such things as: an impact assessment, a preliminary safety report and a risk management assessment. A plan for decommissioning of the installation and reinstatement of the site is also required.

The Ministers responsible for nuclear safety send this information to the Prefect in the "Département" in which the public enquiry is expected to take place. The Prefect will notify neighboring foreign states in the vicinity of the proposed nuclear installation of the opening of the public enquiry, and foreign authorities may participate if they wish. At the conclusion of the public enquiry, the Prefect sends the report of the public enquiry and his opinion to the Ministers responsible for nuclear safety. When the Ministers are satisfied a draft decree is sent to the operator. Once in receipt of the draft decree the operator may submit its observations within 15 days.

Order 7 February 2012

The Order of 7 February 2012 which entered into force on 1 July 2013, incorporated the "reference levels" developed by the Western European Nuclear Regulators Association (WENRA). This order significantly reinforced the BNI regulatory framework. It also provided the legal basis for ASN requirements relating to the "stress tests demanded of nuclear licensees following the Fukushima Daiichi accident." This order applies to BNIs from design through to de-licensing and sets out the concept of "integrated safety." The key areas covered by the Order are: the licensee's organization and responsibilities including technical capabilities, monitoring of contractors, policy on integrated safety, integrated management systems and public information; the demonstration of safety via IAEA standards and ASN technical directives; control of detrimental effects and impact on health and the environment; pressure equipment designed specifically for BNI; waste management; and emergency preparedness.

Energy Transition and Green Growth Act 2015–922 (TECV Act)

The introduction of the 2015 TECV Act made substantial modification to the nuclear legislative framework. The main provisions of the Act were to reinforce transparency and information to the public; reinforcement of the regulation of BNIs to include the regulation of the use of sub-contractors; and clarification of the organization of the ASN and IRSN roles in the oversight of nuclear safety.

Regulator—Nuclear Safety Authority (ASN)

The ASN is an independent administrative authority responsible for the regulation of nuclear safety and radiation protection in France (France, 2019). Whilst the French Government retains the power to stipulate general regulations relating to nuclear activities by decree or order, ASN has the power to issue regulatory resolutions to clarify these decrees and orders. ASN is an administrative authority and it therefore has to rely on the expertise of the French Institute for Radiation Protection and Nuclear Safety (IRSN) and various Advisory Committees for advice on technical issues.

At the head of ASN is the Commission that has five Commissioners including the Chairman. The Chairman and two Commissioners are appointed by the French President; one commissioner is appointed by the president of the National Assembly and the other Commissioner is appointed by the president of the Senate. Each Commissioner is appointed for 6 years. The role of the Commission is to define ASN's strategy and develop general policies relating to such things as regulation, inspection, transparency, emergency management and international relations. The Commission submits ASN opinions to the Government and issues ASN resolutions.

The Director General (DG) of ASN reports to the Chairman of the Commission. The DG manages ASN's Head Office and 11 Regional Divisions. Three Deputy Director Generals, the Head of Cabinet, and the Chief Inspector report to the DG. Within the Head Office there are 9 Thematic Departments, an Office of Administration, a Management and Expertise Office, and a Control Support Office. The 9 Thematic Divisions cover: Nuclear Power Plants; Nuclear Pressure Equipment; Transport and Sources; Waste, Research and Fuel Cycle Facilities; Ionizing Radiation and Health; Environment and Emergencies; International Relations; Communication and Public Information; and Legal Affairs. The 11 Regional Divisions conduct most of the direct oversight of nuclear facilities, radioactive material transport, and small-scale nuclear activities.

At the end of 2018, ASN had a total workforce of 516 staff, 289 in the Head Office and 289 in the Regional Divisions and 1 abroad. In 2018 ASN's total budget was €84.45M. ASN's costs are covered by the State's general budget. ASN relies on IRSN for technical expertise and hence, there is an additional regulatory cost of €84.3M to cover IRSN's regulatory support activities.

Technical Support Organization

IRSN is the ASN Technical Support Organization. A 5-year support agreement between ASN and IRSN is used to set out the technical support arrangements that are needed to support ASN's activities. Each year there is an agreed protocol that provides the more detailed nuclear safety and radiation protection priorities. IRSN dedicates around 400 staff to provide the technical support for ASN. The French Government funds IRSN's technical support activities that are dedicated to ASN. IRSN is an independent public and commercial organization. It is regarded as the public expert in the field of nuclear and radiological risks. IRSN contributes to the development of public policies in the areas relating to its expertise. IRSN also carries out research programs to enable it to maintain its technical expertise.

ASN Advisory Committees of Experts

As noted above, ASN relies on technical and scientific advice from a number of advisory committees. These committees report to the ASN Director General.

Other relevant stakeholders

There are a number of stakeholders that impinge on the activities of ASN.

The Parliamentary Office for the Evaluation of Scientific and Technical Choices (OPECST), informs the French Parliament on scientific and technical choices. The OPECST is assisted by a Scientific Council. Each year ASN presents a report on the state of nuclear safety and radiation protection in France to the OPECST.

The High Committee for Transparency and Information on Nuclear Security (HCTISN) is a body that was set up to enable matters relating to the risks associated with nuclear activities to be discussed in an open and transparent way. Questions concerning information about nuclear safety, its regulation or oversight can be referred to the HCTISN by the ministers responsible for nuclear safety, and others including the chairmen of the National Assembly and the Senate and the chairman of the OPECST. BNI licensees can also put questions to the Committee.

The Nuclear Safety and Radiation Mission (MSNR)

The MSNR prepares general regulatory texts in conjunction with ASN and under the jurisdiction of the ministers and coordinates individual administrative procedures. It also provides the secretariat for the HCTISN committee, and ensures consistency of regulatory texts between BNIs and other Installations that are classified for the protection of the environment (ICPE).

The French High Council for Technical Risk Prevention (CSPRT)

This Council assists ministers responsible for ICPE, nuclear safety and industrial safety. The Council considers draft regulations and questions submitted to it by ministers or ASN. The CSPRT gives its opinion where the law or regulations require.

Regulatory approach

The BNI legal system provides an integrated approach aimed at preventing or mitigating all risks to the public or the environment. It adheres to the four principles of environmental protection applicable to nuclear installations, namely: prevention, precaution, polluter-pays, and public participation.

ASN is responsible for ensuring that the licensee's responsibility for nuclear safety is delivered in compliance with regulatory requirements. ASN sets out the safety and radiation protection objectives to be met by the licensees. The licensee is responsible for producing the safety documentation that demonstrates how the objectives will be delivered. When satisfied ASN approves the adequacy of the licensee's means for achieving these objectives. The licensee is required to implement the approved measures. ASN conducts inspections to check the licensee's compliance with the approved safety measures at the BNI.

ASN may, on request from the licensee, provide advice on proposed safety options. ASN will generally ask an advisory committee to review the licensee's proposal. ASN requires certain activities to be authorized and for such activities, ASN carries out a review and assessment of documentation submitted by the licensee before the licensee is permitted to carry out the required activity.

ASN operates a graded approach to its regulatory activities. ASN appoints inspectors and the duties and functions of inspectors are set out in Article 40 of the TSN ACT and Decree 2007-831.

Licensing and enforcement

An applicant wishing to operate a BNI requires an authorization from the minister responsible for nuclear safety. The applicant can ask ASN for an opinion on the safety options for the plant. These options must then be presented in the form of an application that includes a preliminary version of the safety analysis report (RPS). The application which is known as the BNI creation authorization, is then sent to the minister responsible for nuclear safety. ASN reviews the application and when satisfied, it proposes a draft decree to the Government. The application may be subject to a public inquiry and consultation with the Commission of the European Union. As soon as the BNI creation authorization application is submitted, a local information committee can be created. However, this committee must be operational by the time the authorization is issued. On receipt of the RPS, ASN submits it to one of its advisory committees for review.

A BNI creation authorization decree (DAC) is signed by the Prime Minister and the minister responsible for nuclear safety. The DAC sets out the key characteristics of the facility, the period of the authorization, and the maximum time until facility commissioning. ASN defines the nuclear safety requirements relative to BNI design, construction and operation. Once the facility is constructed, the licensee requires an authorization from ASN to commence commissioning of the facility with nuclear materials. The application to commence active commissioning is sent to ASN and includes the updated safety analysis report for the facility. When satisfied with the licensee's application, ASN issues the authorization to commission the BNI.

Substantial modifications to the facility as defined in Article R.593-47 of the Environment Code must undergo a similar procedure to the BNI creation authorization application. For other modifications, ASN has the power to control the licensee's actions via authorization or notification depending upon the significance of the proposed modification.

ASN can adopt regulatory resolutions to clarify decrees and orders relating to nuclear safety. These resolutions are subject to ministerial approval. ASN has developed basic safety rules and guidelines on various technical subjects relating to PWRs and other BNIs. These are guides, and the licensee need not follow them provided the licensee can demonstrate alternative approaches to attain the safety objectives.

ASN's regulation and oversight of the licensee's activities at nuclear power plant BNIs includes oversight of nuclear safety, radiation protection, the protection of the environment and labor inspectorate duties. ASN undertakes regulation and oversight on a number of levels, namely, review and analysis of the licensee's documentation prior to authorizing activities and inspections during operation. Inspections are done on a sampling basis with the overall aim of checking that the licensee is delivering its responsibilities for nuclear safety, health and environmental protection.

If ASN inspectors observe non-compliance with safety regulations, enforcement measures are available to ASN including financial penalties up to a maximum administrative fine of €10 m for non-compliance with the provisions applicable to BNIs. ASN also has the power to provisionally suspend the operation of a BNI and require the implementation of necessary improvements to the licensee's activities. In addition, criminal infringements identified by inspectors are sent to the public prosecutor's office for review and action. Penalties on conviction can range from €7500 to €150,000 plus between 1 and 3 years imprisonment.

Case Study 4: Nuclear Regulation in Japan

Japan has a large, mature nuclear power program covering NPPs (20 PWRs and 31 BWRs) fuel fabrication facilities, spent fuel reprocessing facilities, radioactive waste treatment and storage facilities. Japan also has a substantial nuclear fission and fusion research program. As discussed in [Mizokami and Rempe \(2021\)](#), the events at Fukushima, Daiichi has significantly impacted Japan's nuclear program. At the time that this article was written, Japan has, or is in the process of decommissioning 27 reactors including

two research reactors (JAIF, 2019). Japan's long-term policy for nuclear power is to reduce the contribution from nuclear power to around 20% in 2030 and to reduce dependency on nuclear power as much as possible by 2050. The source of much of the information in this case study is the "Convention on Nuclear Safety National Report of Japan for 8TH Review Meeting," August 2019 (Japan, 2019) and the Nuclear Regulation Authority (NRA) website (NRA, 2020).

Legal framework

Nuclear law in Japan started with the Atomic Energy Basic Act (Japan, 1955). The Reactor Regulation Act was introduced in 1957 (Japan, 1957). More detailed descriptions of nuclear legislation in Japan are given in Japan (2015, 2019) and OECD (2017).

Atomic Energy Basic Act 1955 (as amended)

This Act had a number of objectives relating to the security of Japan's future energy resources; progress in science and technology and the promotion of industry through engagement in research and development. Its basic policy was to ensure the safe use of nuclear energy for peaceful purposes. The Act initially established Japan's Atomic Energy Commission (AEC) under the Prime Minister's Office (now the Cabinet Office). The AEC's role related to research and development and the exploitation of nuclear energy. The Science and Technology Agency (STA) provided the Commission's secretariat. This Act set the framework for government control of certain activities relating to nuclear energy including the control of nuclear reactors.

Act on the regulation of nuclear source material, nuclear fuel material and reactors (Reactor Regulation Act) as amended

The purpose of this Act was *inter alia* to provide the regulations for the installation and operation of nuclear reactors. It set out the licensing procedure and requirements for NPPs including the requirement for operational safety programs, and identified the minister and the government department responsible for licensing decisions. It specifies the requirements for the control of design, construction, operation and modification. It also provides for regulatory inspection and gives ministers the powers to rescind permissions.

Revisions to the legal framework

Following a review in 1978, several laws were amended including: the Atomic Energy Basic Act; the Law on the Establishment of the Atomic Energy Commission known as the Establishment Law; and the Reactor Regulation Act. The main changes to these laws were the creation of the Nuclear Safety Commission (NSC) and NPP licensing procedures as set out in the Reactor Regulation Law. The NSC had five Commissioners appointed by the Prime Minister, and the STA provided its secretariat. These new arrangements required Ministers when issuing permissions for nuclear facilities, to consider the opinions of the NSC. The licensing of commercial NPPs subsequently came under the control of the Nuclear and Industrial Safety Agency (NISA) part of the Ministry of Economy, Trade and Industry (METI). The Department of Transport regulated marine nuclear reactors. Research and fuel cycle facilities were regulated by the STA.

The 1999 accident at the JCO Tokaimura facility precipitated changes in the nuclear safety regulatory framework. In 2000, the NSC was strengthened, transferred to the Prime Minister's Office, and given additional regulatory oversight responsibilities. The Reactor Regulation Act was amended to strengthen nuclear safety requirements, and the Act on Special Measures Concerning Nuclear Emergency Preparedness (Nuclear Emergency Act) was adopted.

Following the accident at the Fukushima Daiichi NPP, the legal framework for nuclear energy and its regulation was reviewed. In 2012, the Atomic Energy Basic Act, the Reactor Regulation Act, and other related legislation were amended. A new Act that established a new nuclear regulatory body (Japan, 2012) and the Nuclear Regulation Authority came into force in 2013.

Act for Establishment of the Nuclear Regulation Authority as amended

This Act set up the independent Nuclear Regulation Authority (NRA) for the regulation of nuclear safety and safeguards for nuclear reactors and of other activities and facilities associated with refining of nuclear materials, fuel fabrication, enrichment, interim storage, reprocessing, and radioactive waste disposal. The NRA was set up under the Ministry of Environment to ensure its independence from the nuclear industry and those parts of government responsible for the promotion of nuclear energy. The Act provides the organizational structure of the NRA, the appointment and dismissal of Commissioners, the duty to report to the Japanese parliament, the National Diet, and the disclosure of information.

Regulator—Nuclear Regulation Authority (NRA)

The NRA is an independent regulator with the power to establish NRA ordinances to execute laws. It exercises control using its powers to grant, permit and approve nuclear related activities. The NRA also has the powers to carry out inspections and issue orders. A Commission comprising of a Chairman and four Commissioners sits at the top of the NRA. The Prime Minister with the consent of the Japanese Parliament (Diet) appoints the Chairman and the Commissioners. The term of office for the Chairman and Commissioners is 5 years. The Chairman of the Commission appoints the staff of the NRA.

The Commission is supported by a secretariat headed by the Secretary General. There are two Deputy Secretary Generals, one heading the Human Resource Development Centre and the other being the Deputy Secretary General for Technical Affairs (the Chief Engineering Officer). Within the Secretary General's Secretariat there is a Director General for Emergency Response and a Director General for Radiation Protection Strategy and Security. This secretariat hosts the:

- Divisions for Policy Planning and Coordination, Personnel, Budget and Accounting, and Legal Affairs;
- Departments for Regulatory Standards and Research and Radiation Protection which includes safeguards and security.

The Nuclear Regulation Department is separate from the Secretary General's Secretariat and is headed by the Director General for Nuclear Regulation. The Nuclear Regulation Departments hosts the:

- Division for Nuclear Regulation and Planning;
- Division for Licensing Nuclear Power Plants;
- Division for Licensing Research Reactors;
- Division for Licensing Nuclear Fuel Facilities;
- Division for Licensing for Earthquakes and Tsunami Measures;
- Division for Planning Coordination and Oversight;
- Division for Oversight of Nuclear Power Plants;
- Division for Oversight of Specified Oversight; and
- Division for Oversight of Nuclear Fuel Facilities and Research Reactors.

In addition to the above Headquarter functions, the NRA has 22 Regional Offices located near nuclear sites that undertake oversight activities relating to nuclear safety, emergency preparedness and radiation protection and the Rokkasho Safeguards Center carries out inspection of reprocessing facilities.

The Act that established the NRA provided for a number of Councils and Committees to be established under the NRA to provide additional sources of advice to the NRA. The Reactor Safety Examination Committee provides advice on nuclear safety matters; its members are appointed by the NRA on a part-time basis for a period of 2 years and can be reappointed. The Nuclear Fuel Safety Examination Committee provides advice on the safety of nuclear fuel and associated facilities. The NRA appoints its members on the same basis as the members of the Nuclear Fuel Safety Committee. The Radiation Council provides advice on radiation hazard. The National Research and Development Agency Council provides the NRA with advice on issues concerning research and development.

The NRA has a staff of around 1000. In the financial year for 2019 the NRA had a budget of 54.7 billion yen ([Japan, 2019](#)). NRA's funding is provided by the national budget.

Regulatory approach

The NRA's regulatory approach is based on regulatory requirements and licensing authorizations with the objective of ensuring the protection of the public and the environment. The NRA produces the regulatory requirements that industry must comply with and carries out oversight inspections to check that licensees are complying with the requirements. Regulatory requirements are developed on a facility specific basis. NRA sets the safety goals for facilities to achieve, and the regulatory requirements are intended to deliver the goals. To develop regulatory requirements, the NRA will normally establish "Study Teams" and make use of technical agencies and academic experts. For example, NPP regulatory requirements are set out in the "NRA Ordinance Prescribing Standards, for the Location, Structure, and equipment for Commercial Power Plants and their Auxiliary Facilities," known as the "Reactor Establishment Permit Ordinance." This Permit is part of the licensing process.

Licensing and enforcement

In the case of an NPP, a licensee is required to submit an application to NRA for the construction of a new NPP ([Japan, 2015](#)). This application is required to address, *inter alia*, the location of the proposed facility, the type, power and number of the NPPs to be installed, a construction plan, type of fuel to be used, and method for spent fuel disposal. Typical documents required to support the application include description of the safety design. The NRA will carry out a "Conformity Review" before granting the permit; and the NRA must seek the opinion of the Atomic Energy Commission to confirm that the facility will only be used for peaceful purposes.

Once an Establishment Permit has been given, the licensee is required to obtain approval for its construction plan. The construction plan will be expected to give a detailed description of the design for the reactor. NRA has powers to control modifications during the construction process. NRA inspectors also examine key components, systems and structures prior to operation of the NPP, including nuclear fuel assembly fabrication. The licensee requires approval from the NRA for its "Operational Safety Programme" (OSP). The licensee's OSP is expected to address such things as: compliance with legislation, compliance with the OSP; safety culture; quality assurance; duties carried out by key operating personnel; training; operation; safety review of operation; environmental discharges; accident management; and maintenance. Before handling nuclear fuel, the licensee is required to produce a physical protection program and obtain NRA approval of its program. During the lifetime of the facility, the licensee is required to produce Periodic Safety Assessments (PSAs) at least every 5 years ([Japan, 2015](#)) and submit them to NRA. Licensees are required to report accidents and incidents on their facilities to the NRA. At the end of a facility's life, the licensee is required to produce a decommissioning plan and submit it to the NRA for approval.

In relation to emergency preparedness, the licensee must prepare a Nuclear Operators Emergency Preparedness and Response Plan (NOEPRP) before the commencement of NPP operations. The licensee is required to consult local governments in the vicinity of the facility when developing the NOEPRP and notify the NRA before operations begin. The licensee must also establish on-site emergency preparedness arrangements that include an emergency preparedness manager.

NRA inspectors undertake operational safety and physical protection inspections to check licensee compliance with regulatory requirements during the lifetime of the facility.

The NRA has the power to revoke a reactor installation permit if the licensee does not start operation within 5 years of obtaining the permit without good cause. NRA can also revoke a reactor installation permit or order the shut-down of a power reactor for a period of up to 1 year if the licensee violates the provisions of the Reactor Regulation Act. The NRA also has the power to require the licensee to improve safety if there is clear non-compliance with regulatory requirements.

There are penalties for non-compliance with the Reactor Regulation Act. If a reactor is constructed without a permit or a reactor is not shutdown when ordered to do so by the regulator, the person responsible can be imprisoned for up to 3 years or fined not more than 3 million Japanese Yen. There are other penalties for non-compliance with the NRA Establishment Act concerning persons who interfere with NRA investigations. Under these circumstances, a person may be fined up to 300,000 Japanese Yen. In addition, the licensee can also be fined a similar amount.

Case Study 5: Nuclear Regulation in Russia

Russia has a major nuclear power program and operates several types of nuclear fuel cycle facilities. As of 2019, the Russian federation had 36 operating NPPs, including two fast breeder reactors, giving some 29 GWe of installed nuclear capacity (IAEA, 2019).

Legal framework

The legal framework for the regulation of nuclear safety in the Russian Federation is carried out on the basis of Federal Laws and by-laws (IAEA, 2016b). The principal laws relating to the regulation of nuclear energy is Federal Law No. 170-FZ on 21 November 1995—“On the Use of Atomic Energy” as amended, and Federal Law No. 3-FZ of 9 January 1966—“On the Radiological Safety of the Public” as amended. However there are numerous other laws that control the use of nuclear energy in the Russian Federation.

Federal Law No 170 of 21 November 1995 (as amended)—On the use of atomic energy

The main principles for the regulation of the use of atomic energy, as set out in the 1995 Act On the Use of Atomic Energy, as amended, are *inter alia*: to protect individuals, the public and the environment from radiation hazards; to establish the responsibilities of the state safety regulatory authorities; and the independence of the state safety regulatory authorities from the authorities that use nuclear energy (Russia, 1995). The main purpose of the act is to create a legal framework for the system of state management of the use of atomic energy and for the system of state regulation of safety in the use of atomic energy and to establish the rights, obligations and responsibilities of state authorities, local government authorities, organizations and other juridical persons, and citizens. The operating organization is held solely responsible for the safety of a nuclear facility. The 1995 Act as amended has a wide scope including NPPs and fuel cycle facilities. It applies to the design, construction, operation and decommissioning of nuclear facilities including NPPs and radioactive disposal facilities.

The 1995 Act, as amended, sets out the procedures for drafting rules and regulations relating to the safety of atomic energy facilities. Under this Act, the President of the Russian Federation has powers to make decisions on safety issues relating to the use of atomic energy and approve those juridical persons that can own nuclear materials and nuclear facilities. The Federal Assembly of the Russian Federation is given powers to adopt Federal laws concerning the use of atomic energy and approve budgetary allocations in order to finance atomic energy related activities. The Act also gives powers to the government of the Russian Federation to, *inter alia*, enact federal laws; establish the functions, working procedures, rights and duties of the state safety regulatory authorities; and take decisions on the design, construction, operation and decommissioning of nuclear facilities.

Government of Russia Ordinance No. 280 of 29 March 2013

This ordinance covers the regulation and licensing of activities in the field of the use of atomic energy.

Rostechndzor Order of 8 October 2014—Administrative Regulation on Providing the State Service of Licensing the Activity in the Field of the Use of Atomic Energy by the Federal Environmental, Industrial and Nuclear Supervision Service

This Regulation defines the administrative actions the regulator must carry out, including interactions with the license applicants, licensees and other state bodies during the licensing process.

Federal Standards and Regulations in the Field of the Use of Atomic Energy

The Russian nuclear safety regulator has endorsed numerous regulations and standards that relate to the use of atomic energy. These address such things as: the general safety provisions for nuclear power plants (NP-001-15); safety rules for the management of NPP radioactive waste (NP-002-15); procedures relating to emergencies (NP-005-16); requirements for physical protection systems for nuclear materials and nuclear installations (NP-083-15); and general provisions for the decommissioning of nuclear facilities (NP-091-14).

Regulator—Federal environmental, industrial and nuclear supervision service (Rostechndzor)

The nuclear safety Regulatory Body in Russia is the Federal Environmental, Industrial and Nuclear Supervision Service (Rostechndzor). Rostechndzor has the authority to endorse legal regulatory acts related to the uses of atomic energy such as:

- Federal standards and regulations;
- Procedures for work licenses;
- Requirements for the composition and contents of safety related documents;
- Procedures for the submission of safety related documents to Rostechndzor;
- Procedures for the review of safety cases for nuclear facilities;
- Procedures for permits for the discharge of radioactive substances; and
- Procedures for compliance assessment of products and processes during design, manufacture, construction, installation, and operation.

It also has the authority to *inter alia*:

- License activities;
- Grant permits to people to work in nuclear installations;
- Establish guidelines for the release of radioactive materials into the environment;
- Carry out compliance inspections at nuclear facilities;
- Ensure oversight of nuclear facilities in emergencies;
- Oversee physical protection at nuclear installations; and
- Oversee control of nuclear material accountancy.

The organizational structure of Rostechndzor that is related to nuclear safety (IAEA, 2016b) is headed by a Chairperson with a Deputy Chairperson responsible for each of the four Headquarter (HQ) Departments. The HQ Departments cover:

- Safety Regulation of NPPS and Research Nuclear Installations;
- Safety Regulation of Nuclear Fuel Cycle Facilities, Ship Nuclear Installations and Radiation Hazard Facilities;
- Special Security; and
- International Cooperation.

In addition to its HQ Departments, Rostechndzor has six interregional territorial departments for supervision of nuclear and radiation safety. It is also supported by two technical support organizations, namely the Federal Budgetary Institution “Scientific and Engineering Center for Nuclear and Radiation Safety,” and Federal State-Owned Unitary Enterprise for Safety. Russia’s Seventh National Report to the Convention on Nuclear Safety Review Meeting showed that Rostechndzor had around 150 HQ staff and some 850 staff in the interregional territorial departments. The activities of Rostechndzor are funded by the federal budget.

Regulatory approach

The regulatory approach in Russia is based upon the application of laws, ordinances, orders, standards, and regulations. Ordinances of the Russian Government set out detailed requirements. A revision of the federal standards and regulations relating to NPPS was carried out in 2015 to better align the Russian requirements with those of the IAEA safety standards. These standards and regulations are detailed and focused on a wide range of nuclear related activities, such as lifetime performance management, rules for the layout and safe operation of equipment, monitoring of welds, safety rules for the management of radioactive waste, and requirements for the justification of the strength and thermal mechanical behavior of fuel assemblies.

Rostechndzor uses orders for enforcement of: licensing actions, including suspensions and annulment; actions to be taken following inspections; reviews of safety cases; and qualification requirements of experts used by Rostechndzor. Rostechndzor also produces detailed guides, such as recommendations for writing a Level 1 probabilistic safety analysis (see Modarres and Kim, 2021) relating to external hazards, recommendations for conducting reliability analysis, and severe accident management.

Licensing and enforcement

Rostechndzor controls and monitors a licensee’s activities via granting licenses and permissions to undertake nuclear related activities, assessing documentation, inspections, and enforcement (Russia, 2014).

To obtain a license to construct an NPP, the applicant must submit a suite of safety justification documents. The suite of documents is expected to include a:

- Preliminary Safety Analysis Report;
- Quality Assurance Program (QAP) for the NPP;
- Detailed QAP for construction;
- Description of the reactor design and significant safety systems.
- Probabilistic safety assessments (Level 1);
- Description of the physical protection arrangements for the NPP; and
- Commissioning and initial start-up program.

The information provided should be sufficient to enable Rostechndzor to be able to gain an adequate understanding of the design and safety features of the NPP. A license will only be granted when Rostechndzor is satisfied with the adequacy of the safety justification documentation. The NPP design organization requires a design and engineering license from Rostechndzor.

Following the completion of construction, the licensee is required to submit documents to demonstrate nuclear and radiation safety prior to the application for an operating license. The submission is expected to include documents such as:

- Preliminary Final Safety Analysis Report;
- Commissioning program, including first criticality and initial start-up;
- Results of the first criticality and power start-up;
- Final Safety Analysis Report (updated with results of initial power operations);
- Level 1 and Level 2 probabilistic safety assessments;
- Accident mitigation instructions and beyond design basis accident management guidance;
- Quality assurance program for operation;
- Selection and training of NPP workers;
- Maintenance arrangements; and
- Physical protection.

Rostechndzor inspectors carry out readiness inspections of the NPP prior to the first criticality and prior to initial power operations. Rostechndzor will only grant an operating license when it is satisfied that the submitted documents demonstrate the safety of the NPP during routine operation.

During the lifetime of an NPP, Rostechndzor has the power to conduct continuous inspections, annual inspections and unscheduled inspections (IAEA, 2013). Continuous inspections are performed by the site resident inspectors. Comprehensive inspections are organized by the headquarters personnel and are carried out once every 3–4 years. Targeted inspections carried out by the Interregional Territorial Departments.

Rostechndzor is required to exercise its powers in a way that is commensurate with the potential danger of the facility (Russia, 1995). It has the power to issue instructions to the licensee to rectify identified breaches of mandatory requirements. Rostechndzor has the power to gather information on violations in connection with breaches of regulations and to forward this information to the appropriate authorities to consider the institution of criminal proceedings. Penalties for violations have a ratio of 1:10 for individuals and legal entities (IAEA, 2013).

Summary

Individual countries have developed their own regulatory and licensing approaches. The five case studies described in this article illustrate that in spite of different legal systems and cultures, each country, in their own way, delivers nuclear safety regulation and licensing in ways that are consistent with international expectations set out in the Nuclear Safety Convention and supporting IAEA standards.

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Probabilistic Risk Assessment

Mohammad Modarres^a and Inn Seock Kim^b, ^a University of Maryland, College Park, MD, United States; and ^b ISSA Technology, Inc., Germantown, MD, United States

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Introduction

Consequences of failures of complex systems, such as nuclear power plants, are significant and could lead to adverse effects on the public, environment, and the economy. Although failures of complex engineering systems are very rare, they are unavoidable. Probabilistic Risk Assessment (PRA), which is also known as Probabilistic Safety Assessment (PSA), is a model-based methodology to systematically identify scenarios of failure events that end with significant consequences, and then, quantify the frequencies and consequences for those scenarios to obtain risk estimates. While initially focused on estimating the likelihood of adverse outcomes such as reactor core damage and release of radioactive material to the environment, the PRA methodology has more recently become a life-cycle tool for informing regulation, design, manufacturing, construction, operation, maintenance, security, and decommissioning of nuclear facilities. Today, a broad spectrum of organizations in defense, energy, aerospace, health, and security industries rely on the PRA methodology for devising risk-informed life-cycle, regulatory, and policy decisions that are commensurate with the risks of the complex engineering systems. This chapter defines key PRA terms, the overall methodology steps, and the uses of its results.

Definition

Probabilistic Risk Assessment is a systematic procedure for studying the safety and security of complex systems during design and operation. The PRA models interactions and failure events involving human errors, software failures, and hardware malfunctions that could potentially lead to undesirable states of the system. It also assesses the most significant contributors to undesirable outcomes of the system (i.e., consequences including human, environmental, and economic losses). The fundamental questions a PRA intends to address are defined by [Kaplan and Garrick \(1981\)](#) as:

1. What can go wrong in the system that could lead to exposure of hazards?
2. How likely is this to happen?
3. If it happens, what consequences are expected?

These are commonly referred to as the risk triplet. The PRA addresses the first question by identifying scenarios of events (a combination of failures of humans, software, and hardware) leading to exposure of radiological, physical, or chemical hazards. A hazard is any agent that can cause harm or damage to humans, properties, or the environment. In nuclear power plants, the primary hazard of

concern is the radiological sources contained in the reactor core and includes undesirable events such as core damage or containment failure that potentially can lead to a large release of radiation. To address the second question, the PRA uses probabilistic and statistical methods to estimate the likelihood of each scenario. Finally, to address the last question, the PRA also includes an assessment of the magnitude of health, safety, environmental, and economic consequences resulting from the hazard exposure for each scenario. The risk value (i.e., expected loss) of each scenario is often measured as the product of the scenario frequency and its consequence. The main result of the PRA is not the estimated risk value, but rather the identification of the nuclear plant components and failure events that significantly contribute to the risk value, including the uncertainties associated with such estimates. In the remainder of this chapter, major steps in conducting a PRA will be discussed.

Basic elements of PRA

The PRA evaluates risks associated with complex engineering systems by delineating a large number of event scenarios that lead to undesirable consequences (or end states) defined by the analyst. An event scenario consists of the following elements in general:

- Initiating event: An event that triggers abnormal condition in the systems, and that if not correctly responded to, may lead to undesirable consequences.
- Hardware events: Failure or unavailability of hardware components.
- Software events: Failure or unavailability of software components.
- Human error events: Errors of human operators or plant personnel.
- Nonrecovery events: Failures to recover failed or unavailable components.

If an initiating event occurs inside the plant or for a reason related to the plant systems, it is called an internal event, and the PRA for internal events is referred to as an *internal event PRA*. If it is caused by a reason outside the plant or beyond the plant systems, it is called an external event, and the PRA for external events is referred to as an *external event PRA* (e.g., tsunami PRA). Also, depending on the operating state of the plant when the initiating event occurs, the PRA is called a full-power PRA or a low-power/shutdown PRA.

Steps in conducting a PRA

Fig. 1 shows the steps in performing a full-scale PRA. Each step is further discussed in the sections that follow.

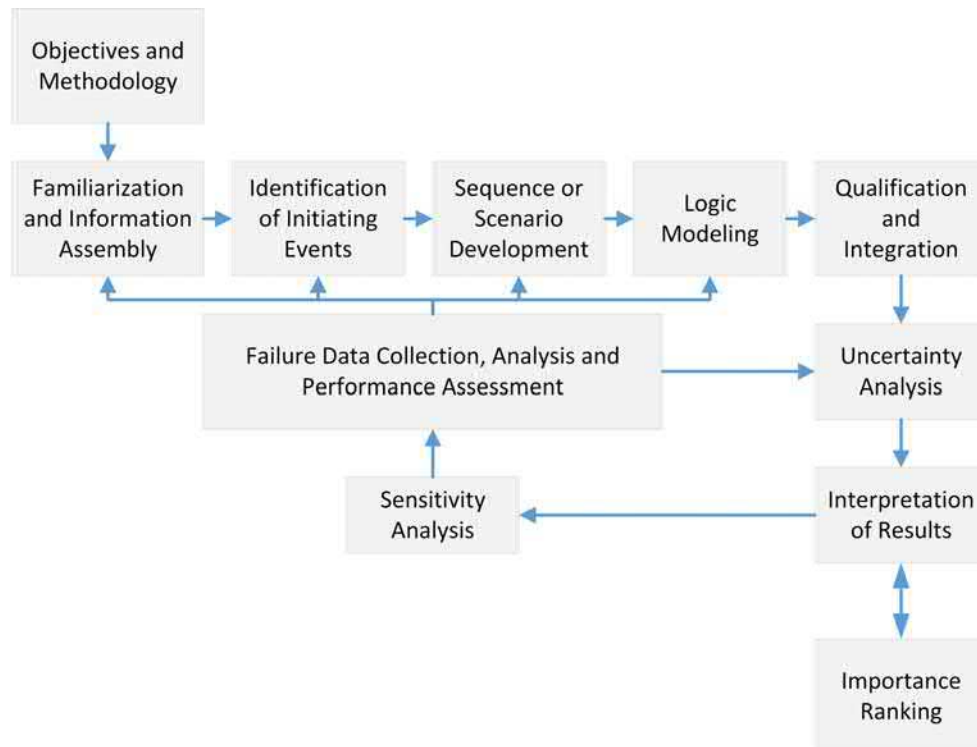


Fig. 1 Components of the overall PRA process. Adapted from Stamatelatos M and Dezfuli H (2011) *Probabilistic Risk Assessment Procedures Guide for NASA Managers and Practitioners*, 2nd edn. Washington, DC: National Aeronautics and Space Administration, NASA Headquarters, Office of Safety and Mission Assurance.

Objectives and methodology

The first step in a PRA involves a definition of the objectives of the analysis. There are varied objectives for completing a PRA, and among them, the most common ones include design improvement, demonstration of risk acceptability, design and operational decision support, regulatory and oversight support, and aging and system health management. Once the objectives are clarified, an inventory of possible techniques and scopes that are consistent with the specific objectives should be selected. This step provides the framework and road map for the analysis. For the inventory of different PRA techniques and scopes, see [Modarres \(2006\)](#) and [Rausand \(2013\)](#).

Familiarization and information assembly

It is crucial to assemble all the relevant information and data about the physical layout of the overall system (e.g., plant description and design blueprints), administrative controls, and maintenance and test procedures, as well as subsystems and barriers of any form embedded in the system design (e.g., physical barriers, human interventions to prevent a further degradation of the plant condition). A detailed inspection or walkthrough of the plant should be performed in the areas expected to be of interest and importance to the analysis (e.g., relevant to protection, prevention or mitigation of hazard exposure conditions). The following items should be performed in this step:

1. Identify major critical barriers in the form of structures, emergency systems, and human-based or other interventions.
2. Identify and explicitly describe dependencies in the form of direct and indirect physical connections among critical subsystems (or parts of the system). The result should be summarized in a dependency matrix (see [Modarres \(2006\)](#)).
3. Gather historical data and information in the form of past failures and abnormal events that have been observed in the plant being analyzed or other plants of a similar design. Such information would help determine the likelihood and frequency of events and ensure the inclusion of essential scenarios.
4. During the analysis, it is crucial to have a well-documented process of information assembly and update, including any PRA models and analyses previously performed for the plant.

Identification of initiating events

Initiating events are the first events that take the nuclear plant out of its normal operating condition and potentially start a scenario of events leading to hazard exposure. Those abnormal events or conditions that, if not correctly and timely responded to, could ultimately result in the event of core damage or a large release of radioactive materials. During the normal plant operation, failures of equipment, natural events, or other events from outside the plant, such as an external fire or a flood, may cause the plant to enter an off-normal or transient state. When a transient starts, there are two possibilities. First, the state of the plant could be such that no other function is needed to maintain a safe condition. Here, the term safe refers to a mode of operation where the chance of exposing hazards is negligible. In this case, inherently or passively provided barriers prevent any hazard exposure. The second possibility is a state wherein certain protective functions by system operations, or human actions are required to prevent hazard exposure. PRAs are mostly concerned with the second class of events.

One method for determining initiating events is to develop a functional block diagram of the systems involved in the normal operation of a plant. Each function can then be decomposed into its subsystems and components of the systems that, individually or in combination, realize those functions. The functional block diagram is used to trace the effect of failure events that lead to the loss of the critical normal operating functions, whereby a transient in the plant would follow.

An alternative method is the use of Failure Mode and Effect Analysis (FMEA) (see [Stamatis \(2003\)](#)). The FMEA is an inductive method of identifying initiating events and is mostly an experiential technique. In both of these methods, one can always supplement the set of identified initiating events with generic initiating events, if known. For example, see [Sattison and Hall \(1990\)](#) for these generic initiating events for nuclear power plants. In particular, the choice of external initiating events (e.g., earthquake, high wind, and external flood) are based on generic sources. The following should be followed as part of this step in the PRA:

1. Select a method for identifying initiating events. The aforementioned functional block diagram and FMEA are representative methods. For other methods such as master logic diagram, refer to [USNRC \(1983\)](#).
2. Using the method selected, identify a set of initiating events.
3. Group the initiating events that require the same mitigating functions to avert hazard exposure.
4. Add generic initiating events that are judged by the analysts to be important.

In nuclear power plants, two classes of initiating events are generally possible: transients and loss of coolant accidents. Transients represent initiating events that trigger an abnormal condition where the reactor coolant system is intact, and the reactor pressure boundary is not affected. Loss of coolant accidents are abnormal conditions where the reactor pressure boundary is compromised due to a break or an opening through which the coolant will be lost. A PRA typically involves dozens of transients (such as loss of electric power due to a grid failure or loss of the feedwater system due to a pump failure) and several loss of coolant initiating events (such as a small, medium or large loss of coolant accident depending on the size of the opening or break). Other initiating events

such as internal fires and floods that start inside the plant, or external events such as earthquakes and high winds that start from outside of the plant, are also important in nuclear plant PRAs and will be discussed later in this chapter.

Development of event scenarios

Development of event scenarios addresses one of the risk triplet questions discussed earlier, namely: what can go wrong? Its purpose is to develop a complete set of scenarios following the occurrence of an initiating event. The scenarios that describe the functional response of the plant to the initiating events are frequently displayed by event trees. In a PRA, two types of event trees can be developed: functional and systemic. The functional event tree uses mitigating functions as its heading (sometimes referred to as pivotal events). The primary purpose of the functional tree is for better understanding of the event scenarios at an abstract level. The functional event tree also guides the PRA analyst in the development of a more detailed systemic event tree. The systemic event tree reflects the scenarios of specific events (in terms of loss of barriers, human errors, failures of protective or mitigative systems). Examples of mitigative systems in nuclear power plants are emergency core cooling systems and containment cooling systems. An example of a protective system is the reactor shutdown system. Therefore, a systemic event tree fully delineates the overall plant response to an initiating event and serves as the primary tool for further analyses in the PRA. For detailed discussions on specific tools and techniques used for this purpose, see [Modarres \(2006\)](#). The following procedure should be followed in this step of the PRA:

1. Identify the mitigating functions for each initiating event.
2. Identify the human actions and hardware operations associated with each function, along with the specific circumstances of success or failure.
3. Develop a functional event tree for each initiating event.
4. Develop a systemic event tree for each initiating event, delineating the success conditions, failed conditions, initiating event progression phenomena, and end effects (i.e., outcomes or consequential losses) of each scenario such as core damage or a large release of radioactivity.

For specific examples of scenario development, see [Modarres et al. \(2017\)](#), [Rausand \(2013\)](#), and [Stamatelatos and Dezfuli \(2011\)](#).

Logic modeling

Event trees commonly involve approximate chronological events developed from left to right from the initiating events. The branch points in the event tree show occurrence of events (known as pivotal events) where a barrier either successfully works or not. Since these barriers (e.g., human actions, subsystem operations) rarely fail in general, a sufficient record of the barrier failure data may not be available. In such cases, logic-based system analysis methods such as fault trees can be used to estimate the frequency of barrier failures represented as pivotal events. The fault tree analysis is a deductive modeling approach and involves developing a logic model in which the barrier is systematically broken down into its components or segments (referred to as basic events) for which adequate failure or unavailability data exist. Once the fault trees are developed, the Boolean expressions representing their binary logic in terms of Boolean operators, such as OR, AND and NOT, are obtained. The failure probabilities of the pivotal events can then be quantified from the Boolean expression representing the minimal cut sets of the top event, see [Vesley et al. \(1981\)](#).

Fig. 2 conceptually displays integration (or merging) of the event tree and fault tree logic models. The event tree depicts scenarios of events following the initiating event of a large loss of coolant accident. In order to avoid undesirable consequences, the reactor should be successfully shut down, the lost coolant replenished by an emergency core cooling system, and the decay heat removed by a residual heat removal system. In turn, these pivotal events are modeled by fault trees, as mentioned earlier. These fault trees show the important combinations of hardware failures, human errors, or software faults that cause the corresponding pivotal events to fail. For more details about fault trees associated with the event headings of an event tree, see [Vesley et al. \(1981\)](#), [Rausand \(2013\)](#), and [Modarres et al. \(2017\)](#).

In principle, a fault tree can be developed as follows:

1. Develop a fault tree for each pivotal event included in the event tree headings for which actual historical failure data does not exist and with which a subsystem consisting of multiple components, or a combination of human actions is associated.
2. Explicitly model dependencies of a subsystem on other subsystems and dependencies among components. Common cause failures (CCFs) discussed later in more detail in this chapter involving implicit dependent failures of multiple redundant components can also be explicitly modeled in the fault trees. This topic will be further discussed later.
3. Include all potential and probabilistically quantifiable causes of failure in terms of hardware, software, test and maintenance, or human error in the fault tree.

Failure data collection, analysis and performance assessment

The second question of the risk triplet (i.e., how likely is it?) can be addressed by evaluating the frequency of scenarios. To perform this assessment, the reliability and availability of components and subsystems that contain or mitigate the hazards are estimated from data gathered as part of the first step of the PRA. One of the best means for predicting future reliability or availability of barriers

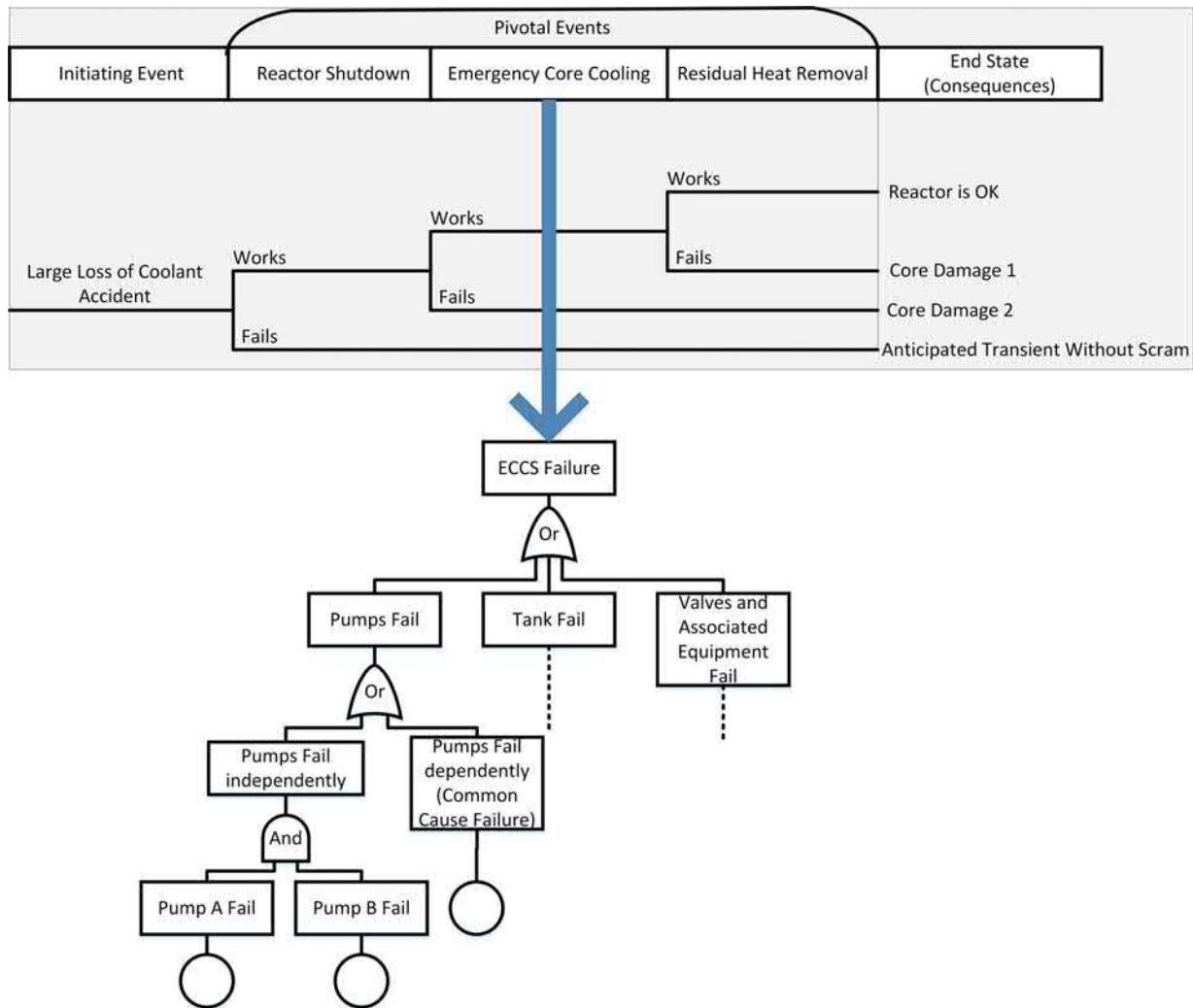


Fig. 2 A simple integrated event-tree and fault-tree model.

is past field experiences, and sometimes coupled with reliability life tests when limited, or no field data are available. For methods of how to assess reliability and availability of components, subsystems and human actions (as barriers in risk) and for generic failure probabilities, see [Atwood et al. \(2003\)](#) and [Modarres et al. \(2017\)](#).

The analysis of failure data consists fundamentally of the following steps: (1) collect generic data, (2) assess generic data, (3) statistically evaluate facility- or system-specific data, and (4) develop failure probability distributions using the facility- or system-specific data. Three types of events identified during the risk scenario definition and system modeling must be quantified for the event trees and fault trees to estimate the frequency of occurrence of sequences: initiating events, component failures, and human errors.

The essential data include the time of failures, repair times, test frequencies, test downtimes, and common-cause failure events and parameters. Further, uncertainties associated with such data must also be characterized. [Modarres et al. \(2017\)](#) discuss available methods for analyzing such data. [Mosleh \(1988\)](#) discusses the data analysis focusing on dependent failures. Establishment of the database to be used will generally involve the collection of some plant-specific data combined with the use of generic failure data when specific data are absent or sparse. For example, [Modarres et al. \(2017\)](#), [AIChE \(1989\)](#), [Atwood et al. \(2003\)](#), and [IEEE \(1983\)](#) describe generic data for electrical, electronic, and mechanical equipment. When events are so rare that no prior experience exists and no reliability tests such as accelerated life tests are possible, then the likelihood of such events should be estimated using expert elicitation. The following procedure may be carried out as part of the data analysis task:

1. Determine generic values of material strength or endurance, applied load and stress, failure times, and failures on demand for each item (hardware, human action, or software) identified in the PRA event tree and fault tree models.
2. Gather plant-specific data on hazard barriers, along with test and maintenance data.

3. Assess the frequencies of initiating events and the probabilities of failure events based on experience, expert judgment, or generic sources.
4. Determine CCF probabilities for similar items, primarily from generic values. However, when significant specific data are available, they should be preferably used.

Common cause failures

Dependent failures are significant in nuclear power plant PRAs and must be given adequate treatment to avoid gross underestimation of such failures in protective or mitigative systems. These failures could be due to common root causes and coupling mechanisms such as the same design deficiencies, failure mechanisms, or poor maintenance that increase the chance of the near-simultaneous failure of components. A class of dependent events called common cause failures have been shown to contribute significantly to the overall plant risk. There is no unique and universal definition for CCF. However, a fairly general definition of CCF is given by Mosleh (1988) as “a subset of dependent events in which two or more component fault states exist at the same time, or in a short time interval, and are direct results of a shared cause.”

The primary causes of dependency among a set of systems or components are explicitly described and modeled in the logic models of a PRA. However, CCFs are implicit dependencies. For example, functional and shared equipment dependencies are explicitly modeled in the system fault tree analysis, but other dependencies among identical barriers or equipment are collectively modeled using CCF models.

Several different parametric models have been used in estimating the CCF probabilities. A single-parameter model, i.e., the β -factor model, is the first parametric model applied to CCF events in risk and reliability analysis. Multiple-parameter models are used to obtain a more accurate assessment of CCF probabilities in systems with a higher level of redundancy. The Multiple Greek Letter (MGL) model and the α -factor model are two popular methods for evaluating CCF probabilities in PRAs. For more information, see Mosleh (1988).

Human reliability analysis

It has long been recognized that human error has a substantial impact on the reliability of complex systems. Accidents at Three Mile Island and Chernobyl clearly show how human errors can defeat safety systems and play a dominant role in the progression of accidents. According to Gertman et al. (2002), past PRA studies have revealed that more than 60% of the potential accidents in the nuclear industry are attributed to human errors.

To obtain an accurate measure of system reliability or risk, human error must be taken into account. Analysis of system designs, operational procedures, and post-accident reports shows that human error can be an immediate accident initiator or can aggravate the accident situation. Without incorporating human error probabilities (HEPs), the results of system reliability and failure analysis are incomplete and often underestimated.

To estimate HEPs, one needs to understand human behavior. However, it is challenging to fully understand and model human behavior. Literature shows that there is not a strong consensus on the best way to model all types of human actions, and thereby, quantify HEPs. The assumptions, mechanisms, and approaches used by any one specific human model cannot be applied to all human activities.

Human reliability analysis (HRA) is an essential element of the PRA process to determine HEPs. HRA can be generally performed as follows:

1. Definition: Ensure that all different types of human interactions are considered.
2. Screening: Select the human interactions that are significant to system reliability.
3. Qualitative analysis: Develop a detailed description of the significant human actions.
4. Representation: Select and apply techniques to model human errors.
5. Impact assessment: Explore the impact of the significant human actions identified in the preceding step on the system reliability model.
6. Quantification: Apply appropriate data to suitable human models to calculate probabilities for various interactions under consideration. The quantification includes assessment of the dependent human failure events.
7. Documentation: Document all information necessary for the assessment to be understandable, reproducible, and traceable.

A large number of different HRA modeling methods have been developed and applied to PRAs, and can be classified into the following categories:

1. Simulation methods: Maintenance personnel performance simulation (MAPPS), and Cognitive environment simulation (CES).
2. Expert judgment methods: Paired comparison, Direct numerical estimation (absolute probability judgment), Success likelihood index method (SLIM), and Integrated Human Event Analysis System (IDHEAS).
3. Analytical methods: Technique for human error rate prediction (THERP), Standardized Plant Analysis Risk-Human Reliability Analysis (SPAR-H) method, Human cognitive reliability (HCR) correlation, and Time-reliability correlation (TRC).

These HRA modeling methods have their own pros and cons. For more details on these techniques, refer to Modarres et al. (2017) and Kolaczowski et al. (2005).

Quantification

Quantification of risk requires that fault trees and event trees be integrated (merged) into a single Boolean logic model, and their events quantified to determine the frequencies of event scenarios and associated uncertainties in the final risk values. This integration depends somewhat on how system dependencies have been handled.

The quantification requires the generation of the so-called minimal cut sets that represent the combination of events which lead to the end states of the PRA model. Starting with fault tree models for the various systems or event headings in the event trees, and using probabilistic estimates for each of the events modeled in the event trees and fault trees, the probability of each event tree heading (or pivotal event often representing failure of a hazard barrier) is calculated (if the heading is independent of other headings). The fault trees for subsystems and support components (e.g., lubricating, cooling, or electric power) are merged where needed, and the equivalent Boolean expression representing each scenario in the event tree model is developed. The Boolean expressions are reduced to arrive at the smallest combination of basic failure events (the so-called minimal cut-sets) that lead to the undesired outcome (e.g., core damage) or hazard exposure (e.g., large release). If possible, all minimal cut sets must be generated and retained during this process. However, in large PRA models, this leads to an unmanageably extensive collection of terms and a combinatorial explosion. Therefore, the collection of cut sets is often probabilistically truncated (i.e., the cut sets are discarded based on the number of terms in them or based on the probability or frequency of the cut set).

To perform Boolean manipulation of the PRA logic models, software tools may be needed. The following steps are used to integrate and quantify the PRA models:

1. Merge corresponding fault trees associated with each failure or success event modeled in the event tree scenarios (i.e., combine them in a Boolean form).
2. Develop a reduced Boolean function for each scenario (i.e., truncated minimal cut sets).
3. Calculate the total frequency of each scenario.
4. Calculate the total frequency of all scenarios in all the event trees.

Three levels of PRA for nuclear power plants

A nuclear power plant typically consists of not only a nuclear reactor core where nuclear chain reactions occur for energy generation but also a containment enclosing the nuclear reactor. The design purpose of the containment building is to contain the escape of radioactive materials to the environment in case of an emergency.

Because of the defense-in-depth features of a nuclear power plant, the PRA is usually performed at three levels as follows (refer to Fig. 3):

- A Level 1 PRA estimates the frequency of accidents that cause damage to the nuclear reactor core. This is commonly called core damage frequency (CDF).
- A Level 2 PRA, which starts with the Level 1 core damage accidents, estimates the frequency of accidents that release radioactivity from the nuclear power plant.
- A Level 3 PRA, which starts with the Level 2 radioactivity release accidents, estimates the consequences in terms of fatalities or injury to the public, and undesirable consequences to the environment.

In particular, Level 2 PRA models various severe accident phenomena that could occur after core damage, such as hydrogen combustion, core concrete interaction, or steam explosion. Further, Level 2 PRA estimates source terms, namely, types and amounts of radioactive or hazardous material released to the environment following an accident.

Level 2 PRA is typically conducted in the following steps:

1. The accident scenarios identified in Level 1 PRA as leading to core damage are grouped into plant damage states (PDSs) based on similarities in the plant conditions following core damage.
2. Next, the containment event tree analysis is performed to identify the scenarios that challenge the integrity of the containment and lead to the possible releases of radioactive material to the environment.
3. For each scenario, source term analysis is carried out to determine the quantities of radioactive materials that escape to the environment.
4. Releases are grouped into the release categories, each having the same impact on the consequences (i.e., human, environmental, or economic losses).

From the results of Level 2 PRA, one can gain insights into the level of robustness of the containment in providing protection for severe accidents and the adequacy of the accident management systems in protecting the containment and mitigating a large release of radioactive materials. Since the full-scope Level 2 PRA requires considerable effort, a limited-scope Level 2 PRA is frequently performed instead focusing on the assessment of the so-called large early release frequency (LERF). LERF is defined as the sum of the frequencies of those accidents leading to the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there

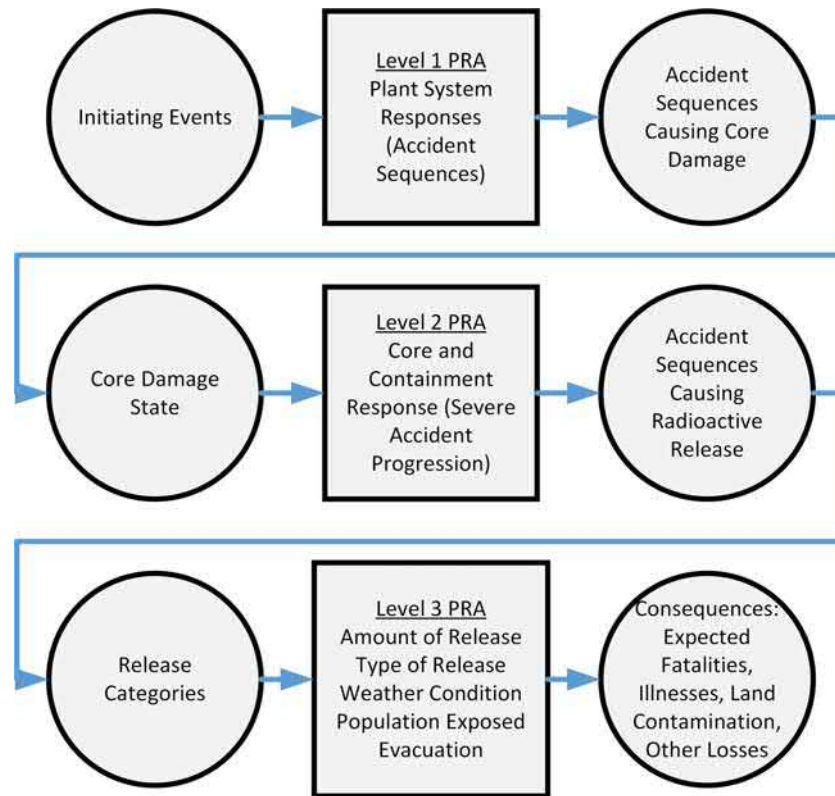


Fig. 3 Three levels of PRA for a nuclear power plant.

is the potential for early health effects (USNRC, 2009). LERF is determined similar to the Level 2 PRA process, by only focusing on accident sequences leading to large early release. In addition to the LERF, a broader metric called large release frequency (LRF) is also sometimes used as a risk metric depicting all large release scenarios of the Level 2 PRA.

Uncertainty and sensitivity analysis

In all facets of the PRA (e.g., frequencies of initiating events, probabilities of failure of components, human error probabilities, common cause failure models and parameters, and severe accident phenomena), uncertainties need to be addressed. In PRAs, uncertainties are primarily shown in the form of probability distributions. For example, the probability of failure of a subsystem or a hazard barrier may be represented by a beta or gamma distribution.

The process involves characterization of the uncertainties associated with the frequencies of initiating events, the probabilities of failure of subsystems (or barriers), the probabilities of all event tree headings, the strengths or endurances of barriers, the applied loads or incurred damages associated with the barriers, the amounts of hazard exposures, the consequences of exposures to hazards, and the sustained total amounts of losses. Other sources of uncertainties are in the models used, for example: the fault tree and event tree models; the stress-strength and damage-endurance models used to estimate failure or capability of some passive barriers; the probabilistic failure models for hardware, software and human actions; and the models to treat inter- and intra-barrier failure dependencies. Another important source of uncertainty is incompleteness of the risk models and other failure models used in the PRAs.

Once uncertainties associated with hazard barriers have been estimated and assigned to models and parameters, they must be “propagated” through the PRA model to find the uncertainties associated with the results of the PRA. Propagation is done using one of several available techniques, in which the most popular method used is Monte Carlo simulation. The results are then shown and plotted in the form of probability distributions. Steps in uncertainty analysis include:

1. Identify models and parameters that are uncertain and the method of uncertainty estimation to be used for each.
2. Describe the scope of the PRA and the significance and contribution of those elements that are not modeled or considered in the PRA.
3. Assess the probability distribution for each basic event modeled in the PRA.
4. Propagate uncertainties associated with the hazard barrier models and parameters to find the uncertainty associated with the risk value.

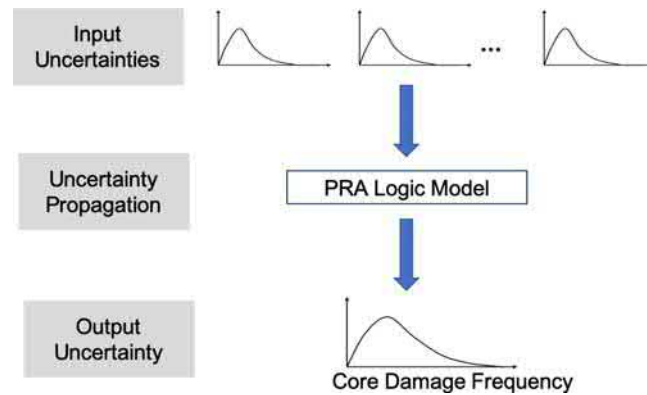


Fig. 4 Treatment of uncertainties in PRAs.

5. Present the uncertainties associated with risks and contributors to risk in an easy way to understand and in a manner visually straightforward to grasp. **Fig. 4** illustrates the uncertainty associated with the overall core damage frequency as a result of Level 1 PRA based on the input uncertainties which were propagated through the PRA logic model.

Sensitivity analysis is aimed at determining the significance of choice of a model or its parameters, assumptions, and significance of specific uncertain input parameters or variables to the final risk value calculated. The process of sensitivity analysis is straightforward. The effects of the input variables and assumptions in the PRA are measured by modifying them by several folds, factors or even one or several orders of magnitude one at a time, and measure relative changes observed in the PRA's risk results. Those models, variables, and assumptions whose change leads to a relatively significant change in the final risk values are regarded as "sensitive." A proper sensitivity analysis strengthens the quality and validity of the PRA results. Interested readers are referred to [Modarres \(2006\)](#) for more detailed discussions on sensitivity analysis.

Risk ranking and importance analysis

Ranking the elements of the system with respect to their risk or safety significance is one of the most important outcomes of a PRA. The ranking is merely arranging the elements of the system based on their increasing or decreasing contribution to the final risk values. There are several different importance measures in PRAs used for risk ranking in PRAs. For example, Fussell-Vesely, Birnbaum, Risk Achievement Worth (RAW), and Risk Reduction Worth (RRW) ([Modarres et al., 2017](#)) are identified as appropriate measures for use in PRAs, and all show the level of contribution of various parts of the PRA model to the final risk values. For additional discussions on the risk ranking methods, see [Azarkhail and Modarres \(2004\)](#). The following are the significant steps of importance ranking:

1. Determine the purpose of the ranking and select an appropriate importance measure that has a consistent interpretation for the intended uses of the ranked results.
2. Perform risk ranking and uncertainty ranking as needed.
3. Identify the most critical and vital elements of the system with respect to the final risk values and the total uncertainty associated with the calculated risk values.

Consideration of internal flooding and internal fire initiating events

Internal flooding refers to floods originating within the plant boundary. The internal flood PRA for a nuclear power plant evaluates the impact of internal flood as the cause of either an accident or a system failure in such a way that: (1) the fluid sources within the plant that could flood plant locations or create adverse conditions (e.g., spray, elevated temperature, humidity, pressure, pipe whip, jet impingement) or that could damage mitigative plant equipment are identified, and (2) the internal flood scenarios that could lead to hazard exposure are identified and quantified.

The following is a concise description of each technical element included in the internal flood PRA process ([AMSE/ANS, 2009](#)).

1. Define the physical boundaries of the analysis (i.e., the locations within the plant where flood scenarios are postulated), and divide the various volumes within that boundary into physical analysis units referred to as flood areas.
2. Identify and characterize the various sources of floods and equipment along with the mechanisms resulting in a flood or spray from these sources (e.g., amount of liquid and flowrates).
3. Develop flood scenarios involving flood source, propagation path(s), and the affected equipment.
4. Identify the expected plant response(s) to the selected set of flood scenarios and quantify the frequency of core damage or any radioactive release frequency.

The internal fire PRA analyzes accident scenarios associated with fires that might occur within the plant boundary. The following briefly describes those technical elements conducted as part of the internal fire PRA ([AMSE/ANS, 2009](#)).

1. Define the physical boundaries of the analysis (i.e., the locations within a plant where fire scenarios are postulated) and divide the various volumes within that boundary into physical analysis units generally referred to as “fire areas” or “fire compartments” including equipment located in them. Screen out areas and equipment which are not important.
2. Identify cables (and the equipment to which the cables are connected) that are required to support the operation of fire PRA equipment selected, and cables whose failure could adversely affect systems needed to respond to this accident.
3. Develop scenarios that reflect the plant response following a fire. Moreover, select scenarios that contribute to plant risk.
4. For each fire area, find the frequency of fires (expressed as fire ignitions per year) and screen out areas with low fire risk contribution.
5. Trace fire-induced cable failures and determine their impact on the plant equipment, systems, and functions included in the fire PRA response model.
6. Analyze operator actions needed for safe shutdown, including those called out in the relevant plant fire response procedures.
7. Quantify and present the resulting fire-induced CDF and LERF risk results. For further readings on this topic, see [EPRI and USNRC \(2005\)](#).

Consideration of external initiating events

As described earlier, external events initiate accidents from outside of the plant boundary. Examples of external events include earthquakes, high winds, external floods, external fires, lightning, and volcanic activity. Typically, a demonstrably conservative bounding analysis is performed for these external events to screen out insignificant ones. When the risk impact is judged to be potentially significant by the bounding analysis (e.g., conservatively estimating the hazard- occurrence frequency or the risk impact of the hazard, and then comparing the bounding estimate to a specific predetermined criterion), a detailed PRA may be performed for those selected or screened-in external events. Because seismic events are often found to be significant for nuclear plants, the way to conduct a PRA for this external event is briefly described herein.

The seismic PRA process consists of the following three steps ([AMSE/ANS, 2009](#)):

1. Estimate the frequency-seismic magnitude relationship (probabilistic seismic hazard analysis) associated with the plant site.
2. Determine the strength of structures, systems, and components (SSCs) concerning various ground acceleration levels (i.e., seismic magnitude) in terms of the conditional probability of failure given a specific acceleration level.
3. Develop seismic scenarios using the plant risk model for internal events with the application of seismic failure probabilities or frequencies for SSCs as well as human error probabilities in consideration of seismic situations. Calculate the risk values and contributors to each seismic scenario.

Interpretation of results

When the risk values are calculated, they must be interpreted to determine whether any revisions are necessary to refine the results and the conclusions. There are two main elements involved in the interpretation process. The first is to understand whether or not the final values and details of the scenarios are logically and quantitatively meaningful. This step verifies the adequacy of the PRA models and the scope of analysis. The second is to characterize the role of each element of the plant in the final results. This step highlights additional analyses of data and information gathering that would be considered necessary. The interpretation step is a continuous process with the acquisition of information from the quantification, sensitivity, uncertainty, and importance analysis activities of the PRA. The process continues until the final results can be best interpreted and used in the subsequent risk management activity.

Summary

Probabilistic Risk Assessment is a systematic procedure for studying and potentially improving the safety and security of nuclear power plants during design and operation. This chapter outlines the essential elements of the PRA technique used in the nuclear power industry and increasingly in other industries involving complex engineering systems. Engineers and safety inspectors can gain insights from the PRA to better focus resources on the hardware, software, and human elements of the system, maintenance, operations, and organizational factors that are most important to safety. Using the PRA results, safety, health and environmental regulators and other stakeholders can confirm the safety significance of enhancements and proposed changes to complex system design and operation. The PRA methodology has more recently become a life-cycle tool for informing regulation, design, manufacturing, construction, operation, maintenance, security and decommissioning of nuclear facilities.

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Regulatory Applications of Risk Assessment

Donald A. Dube, Risk Consultant, Unionville, CT, United States

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Risk-informed regulation

Definitions

The question of “What is risk?” can be answered by the risk triplet:

- What can go wrong?
- How likely is it?
- What are the consequences?

The responses to these three questions may be qualitative or quantitative depending on the specific application and the availability of operational data and experience.

A “risk-informed approach” to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety (US NRC, 1999).

Principles of risk-informed regulation

The US Nuclear Regulatory Commission (US NRC) has led the way internationally in the application of risk assessment and risk-informed regulation since the late 1970s and early 1980s. Specifically, the US NRC established Safety Goals for nuclear power plant operations in 1986 (US NRC, 1986). In brief, the Qualitative Safety Goals are to:

- Protect nuclear plants such that individual members of the public bear no significant additional risk to life and health.
- Ensure that plants do not cause significant societal risk to life and health and have risk comparable to or less than risks of other electrical generation technologies.

For practical reasons, these qualitative goals are translated into the following Quantitative Health Objectives (QHOs):

- For people living near a nuclear power plant, the ‘nuclear risk’ should be less than one-tenth of a percent (1/1000) of the overall risk of death from accidents and cancer experienced by the average member of the US population.

In subsequent implementation of the Commission policy statement, the NRC Staff demonstrated that core damage frequency (CDF) of 10^{-4} /year and large early release frequency (LERF) of 10^{-5} /year are acceptable surrogates for the early and latent QHOs (US NRC, 2007b).

In August 1995, the NRC issued a final Commission policy statement on the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities (US NRC, 1995). The statement adopted the following policy:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, Regulatory Guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109, Backfitting. Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. This policy intends compliance with existing rules and regulations unless these rules and regulations are revised.
- PRA evaluations in support of regulatory decisions should be as realistic as practicable, and appropriate supporting data should be publicly available for review.
- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties when deciding on the need to propose and backfit new generic requirements on nuclear power plant licensees.

In its approval of the noted policy statement, the Commission stated its expectation that implementation of the policy statement will improve the regulatory process in three ways: (1) foremost, through safety decision-making enhanced by the use of PRA insights, (2) through more efficient use of agency resources, and (3) through a reduction in unnecessary burdens on licensees.

Defense in depth

Defense in depth is an element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility ([US NRC, 2018a](#)). The defense-in-depth philosophy has traditionally been applied in plant design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance and, in particular, to account for the potential for unknown and unforeseen failure mechanisms or phenomena that, because they are unknown or unforeseen, are not reflected in either the PRA or traditional engineering analyses. Nuclear power plant defense in depth is taken to consist of independent layers of defense (i.e., successive measures) to protect the public:

- Robust plant design to survive hazards and minimize challenges that could result in an event occurring,
- Prevention of a severe accident (core damage) if an event occurs,
- Containment of the source term if a severe accident occurs, and
- Protection of the public from any releases of radioactive material (e.g., through siting in low-population areas and the ability to shelter or evacuate people, if necessary).

Types of applications

A broad list of the types of risk-informed applications that have been employed in the United States as well as internationally include ([EPRI, 2019](#)):

- Evaluating the acceptability of a change to the plant design or operating practices,
- Comparing design alternatives,
- Optimizing maintenance strategies, including evaluating and managing risk for different plant configurations,
- Deciding what, if any, actions should be taken when new information comes to light regarding a plant's capabilities or the nature of the hazards to which a plant might be exposed,
- Evaluating the need for changes to improve plant safety during periodic safety reviews,
- Accounting for plant performance in determining appropriate regulatory oversight activities.

The above risk-informed applications can be more conveniently categorized into those applications affecting operating reactors, those impacting the design and licensing of new reactor concepts, and those processes common to both currently operating and new reactor designs.

Regulation and oversight of operating reactors

Regulations—Mandatory

Since the mid-1980s the US NRC has promulgated a number of regulations that have a strong risk basis. For example, the Anticipated Transient Without Scram (ATWS) rule under 10 CFR 50.62 was prompted by concerns regarding the reliability of the reactor protection system (RPS) and the consequences of anticipated operational occurrences without reactor scram ([US NRC, 2003a](#)). Precursor ATWS events at Brown's Ferry Unit 2 in 1980 and Salem Unit 1 in 1983 led to regulatory analysis which concluded

that additional ATWS safety requirements were justified. The ATWS rule required the installation of hardware to lessen the probability of ATWS events and to mitigate their consequences. The current fleet of Pressurized Water Reactors (PWRs) is now required to have equipment that is diverse, reliable, and independent from the RPS, to automatically initiate the auxiliary feedwater system, and to initiate a turbine trip under conditions indicative of ATWS. Boiling Water Reactors (BWRs) are required to have a diverse alternate rod injection system and a diverse recirculation pump trip that are reliable and independent from the RPS. BWRs have a standby liquid control (SLC) system to inject borated water into the reactor vessel, and the ATWS rule also specified a minimum SLC system injection rate and boron concentration.

The Station Blackout (SBO) rule under 10 CFR 50.63 was implemented based on several plant-specific probabilistic safety studies; operating experience; and reliability, accident sequence, and consequence analyses completed between 1975 and 1988 (US NRC, 2003b). SBO is defined as the complete loss of ac electric power to the essential and nonessential electric switchgear buses in a nuclear power plant. The SBO rule requires that nuclear power plants be capable of withstanding an SBO for a specified duration and of maintaining core cooling during that period. The specified duration is determined for each plant by comparing the individual plant design with factors that have been identified in NRC technical studies as the main contributors to the risk of core damage resulting from an SBO.

The NRC published 10 CFR 50.65 (commonly referred to as the Maintenance Rule) on July 10, 1991. The NRC's determination that a maintenance rule was needed arose from the conclusion that proper maintenance is essential to plant safety (US NRC, 2018e). The Maintenance Rule is based on the premise that there is a clear link between effective maintenance and safety as it relates to such factors as the number of transients and challenges to safety systems and the associated need for operability, availability, and reliability of safety equipment. Specifically, 10 CFR 50.65(a)(4) requires that before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems and components (SSCs) that a risk-informed evaluation process has shown to be significant to public health and safety.

Finally, 10 CFR 50.44 provides requirements for the mitigation of combustible gas generated by a beyond-design-basis accident (US NRC, 2007a). In existing light-water reactors, the principal combustible gas is hydrogen. In an accident more severe than the design-basis loss-of-coolant accident (LOCA), combustible gas is predominately generated within the containment as a result of the following factors:

- (1) fuel clad-coolant reaction between the fuel cladding and the reactor coolant, and
- (2) molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel.

If a sufficient amount of combustible gas is generated, it may rapidly react with oxygen and increase pressure sufficiently to lead to a containment breach or a leakage rate in excess of technical specification limits. Additionally, damage to systems and components essential to continued control of the post-accident conditions could occur. Requirements are provided in 10 CFR 50.44 for combustible gas control in containments including atmosphere mixing, equipment survivability, oxygen and hydrogen monitoring, and other provisions depending on reactor type (e.g., BWR Mark I containment inerting) and operating versus future reactor designs.

Regulations—Voluntary

The US NRC has also issued guidance and policy statements that are voluntary in nature meaning that the licensee is free to implement the measure (or not) as an alternative to existing regulation.

Regulatory Guide (RG) 1.174 (US NRC, 2018a) describes the key principles and the overall process for risk-informed decision-making as it relates to changes to the plant licensing basis. This guidance may be augmented by application-specific guidance as necessary. Fig. 1 illustrates the relationship of the five key principles. These principles include:

- The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption.
- The proposed licensing basis change is consistent with the defense-in-depth philosophy.
- The proposed licensing basis change maintains sufficient safety margins.
- When proposed licensing basis changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.
- The impact of the proposed licensing basis change should be monitored using performance measurement strategies.

RG 1.174 provides Acceptance Guidelines for what constitutes a small increase in risk level using surrogate measures of CDF and LERF from the plant PRA model. These Acceptance Guidelines are reproduced in Figs. 2 and 3.

An important application is RG 1.178 which provides the guidance for risk-informed changes to the ASME-required in-service inspection of piping within nuclear power plants (US NRC, 2003c). Specifically, RG 1.178 can enhance overall safety by focusing inspections of piping at risk-significant locations and locations where failure mechanisms are likely to be present, and by improving the effectiveness of inspection of components by focusing on personnel qualifications, inspection for cause, and the use of multi-discipline review teams.

Another key risk-informed application is that of the surveillance frequency control program (SFCP) under the Risk-Informed Technical Specifications Initiative 5b (NEI, 2007). The SFCP uses a risk-informed, performance-based approach for establishment of surveillance frequencies, consistent with the philosophy of RG 1.174. PRA methods are used to determine the risk impact of the revised intervals. Sensitivity studies are performed on important PRA parameters. A multi-disciplinary plant Independent Decision-

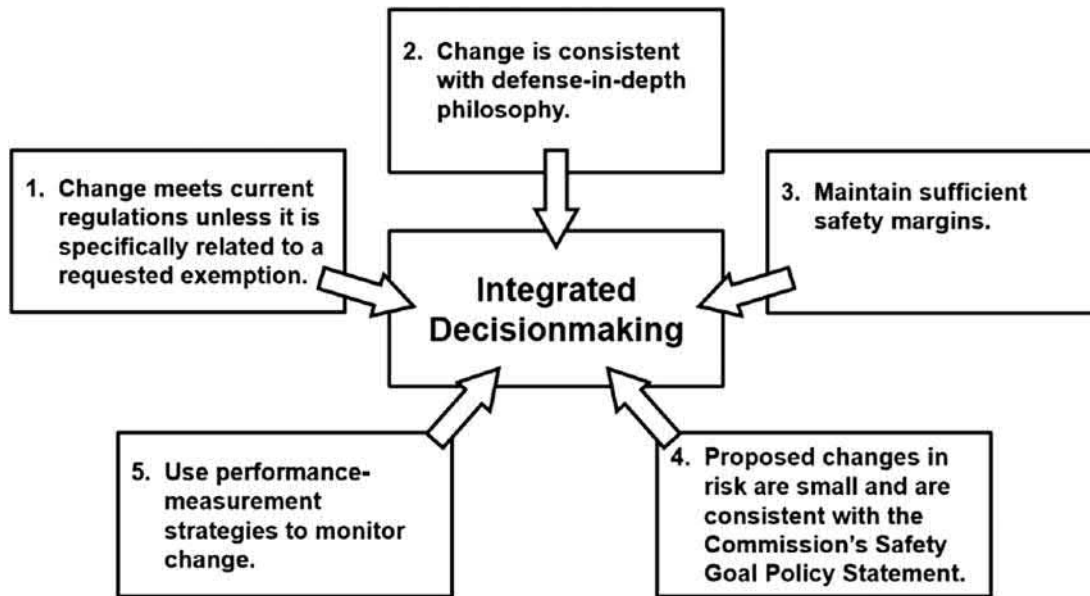


Fig. 1 Key principles of risk-informed decision-making. US NRC (2018a) *Regulatory Guide 1.174. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*. Revision 3, p. 9. Washington, DC: US Nuclear Regulatory Commission.

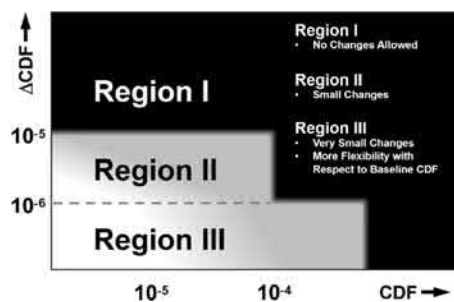


Fig. 2 Acceptance guidelines for core damage frequency. US NRC (2018a) *Regulatory Guide 1.174. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*. Revision 3, p. 27. Washington, DC: US Nuclear Regulatory Commission.

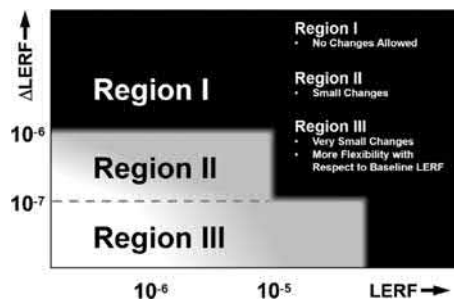


Fig. 3 Acceptance guidelines for large early release frequency. US NRC (2018a) *Regulatory Guide 1.174. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*. Revision 3, p. 27. Washington, DC: US Nuclear Regulatory Commission.

making Panel (IDP) is utilized to evaluate determinations of revised surveillance frequencies, based on operating experience, test history, manufacturer's recommendations, codes and standards, and other factors, in conjunction with the risk insights from the PRA. Use of an IDP-like decision-making forum can be seen in numerous risk-informed applications.

A growing regulatory application is that of risk-managed technical specifications (RMTS) under Risk-Informed Technical Specifications Initiative 4b (NEI, 2006). This program includes individual changes to technical specifications under RG 1.177 (US NRC,

2011). RMTS permits extension of the completion times, also referred to as the allowed outage times (AOTs), provided risk is assessed and managed within a configuration risk management program. Such a program is intended to maintain and improve safety through the incorporation of risk assessment and management techniques in technical specifications, while reducing unnecessary burden. Plants implementing this initiative are expected to use the same PRA analyses to support their Maintenance Rule (a)(4) programs. A deterministic backstop value is imposed to limit the completion time extension regardless of low risk impact. Results of implementation are monitored, and cumulative risk impacts are compared to specific risk criteria. Corrective actions are implemented should these criteria be exceeded.

The national standard, NFPA 805, specifies the minimum fire protection requirements for existing light water nuclear power plants during all phases of plant operation, including shutdown, degraded conditions, and decommissioning in order to protect the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations (NFPA, 2015). Fundamentally, NFPA 805 implements 10 CFR 50.48(c) as a risk-informed alternate fire protection rule and as a means of resolving long-standing fire protection issues. As of 2019, it is being implemented at about half of the US nuclear power plant sites.

RG 1.230 provides guidance on alternate requirements regarding pressurized thermal shock (PTS) in reactor pressure vessels (RPV) (US NRC, 2018b). The RPV in a nuclear power plant is exposed to neutron radiation during normal operation. Over time, the RPV steel becomes progressively embrittled in the region adjacent to the core. If an RPV were to have a pre-existing flaw of critical size and certain severe system transients were to occur, this flaw could propagate rapidly through the RPV, resulting in a through-wall crack. The severe transients of concern, known as PTS events, are characterized by rapid cooling (i.e., thermal shock) of the internal RPV surface that may be combined with repressurization. The simultaneous occurrence of critical-size flaws, embrittled steel, and a severe PTS transient is a low probability event.

The original PTS rule was implemented when there were limited embrittlement data, and analytical tools were overly conservative. The alternate requirements are based on more comprehensive and realistic analysis methods, while still maintaining adequate safety margins. The application of the alternate requirements may help to extend plant lifetime or reduce unnecessary burden on plant operations.

NEI 00-04 (NEI, 2005b) provides guidance for the implementation of 10 CFR 50.69 regarding risk-informed categorization of SSCs and the application of NRC special treatment requirements. Special treatment refers to requirements beyond normal industrial practices that ensure the SSC can perform its design-basis functions. The 50.69 process makes use of the multi-disciplinary IDP as noted above. The approach allows plants to focus on those SSCs that have the most nuclear-safety significance rather than treating all safety-related SSCs as equally important. Fig. 4 illustrates the risk-informed safety categories (RISC). The benefits of implementing the program accrue, in part, by savings in the procurement process for RISC-3 components that no longer need to be subjected to the normal safety-related quality assurance program. However, RISC-3 SSCs must still be capable of performing their design-basis functions.

Oversight

The US NRC regulatory framework for reactor oversight is a risk-informed, tiered approach to ensuring plant safety (US NRC, 2018c). To measure plant performance, the reactor oversight process (ROP) focuses on seven specific “cornerstones” which support the safety of plant operations in broad strategic areas. These cornerstones include such focus areas as initiating events, mitigating systems, barrier integrity, and others. Additionally, there are cross-cutting elements such as human performance. Within each cornerstone, a broad sample of data on which to assess licensee performance in risk-significant areas is gathered from performance indicator (PI) data submitted by licensees and from the NRC’s risk informed baseline inspections.

	Safety-Related	Nonsafety-Related
	NEI 00-04 Categorization Process	
Safety Significant	RISC-1	RISC-2
Low Safety Significant	RISC-3	RISC-4

Fig. 4 Risk-informed safety classifications (RISC). NEI (2005b) 10 CFR 50.69 SSC Categorization Guideline. NEI 00-04, Revision 0, p. 4. Washington, DC: Nuclear Energy Institute.

An example of a risk-informed PI is the mitigating systems performance index (MSPI) (US NRC, 2005). The MSPI monitors the performance of selected nuclear power plant systems based on their ability to perform risk-significant functions. It provides a surrogate measure of change in CDF (Δ CDF) resulting from departures of component unreliabilities and train unavailabilities from established baselines.

Another risk-informed approach for systems and events not covered by PIs is the significance determination process (SDP) for inspection findings (US NRC, 2019). The SDP uses risk insights, where appropriate, to assist NRC staff in determining the safety or security significance of inspection findings identified within the seven cornerstones of safety at operating reactors. The inspection findings use a color-coded scheme for safety importance as follows:

- *Green* indicates a finding of very low safety or security significance,
- *White* indicates a finding of low to moderate safety or security significance,
- *Yellow* indicates a finding of substantial safety or security significance,
- *Red* indicates a finding of high safety or security significance.

When it is possible to quantitatively assess a change in risk resulting from a performance deficiency, the change can be associated with this color coding as depicted in Fig. 5.

The resulting safety or security significance of findings, combined with the results of the risk-informed performance indicator program, is used to determine a licensee's level of safety performance and the level of NRC engagement with the licensee.

The NRC defines 'incident investigation' under Management Directive (MD) 8.3 as a formal process conducted for the purpose of accident prevention (US NRC, 2014). The process includes gathering and analyzing information; determining findings and conclusions, including the cause(s) of a significant event; and disseminating the investigation results for the NRC, industry, and public review. Fig. 6 illustrates how quantitative information regarding the conditional probability of core damage resulting

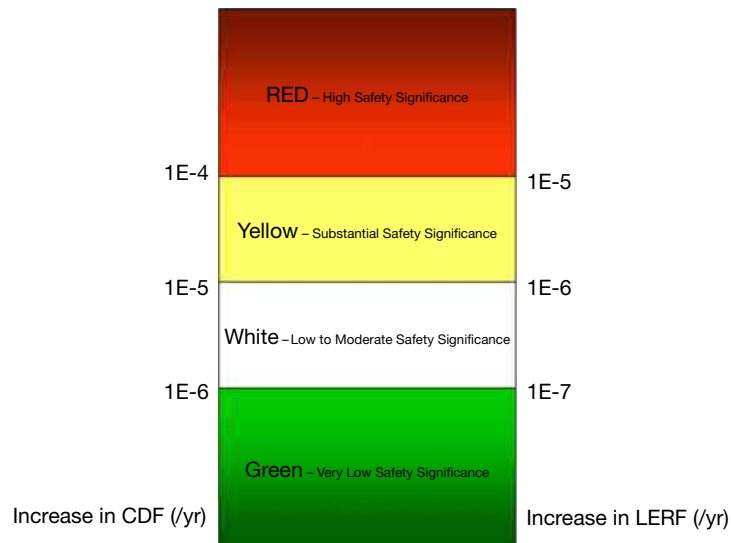


Fig. 5 Color-coding significance based on increase in CDF or LERF. US NRC (2019) *Inspection Manual Chapter (IMC) 0609, Significance Determination Process*, Exh 1-1. Washington, DC: US Nuclear Regulatory Commission.

Estimated Conditional Core Damage Probability (CCDP)				
CCDP < 1E-6	1E-6 – 1E-5	1E-5 – 1E-4	1E-4 – 1E-3	CCDP > 1E-3
No additional inspection				
	Special inspection			
		AIT		
			IIT	

Fig. 6 Use of risk by NRC to set inspection priority for significant events. US NRC (2014) *Management Directive 8.3, NRC Incident Investigation Program*, p. 6. Washington, DC: US Nuclear Regulatory Commission.

from a reactor event is used to set the priority for inspections. (Note: AIT stands for augmented inspection team, while IIT represents incident investigation team.)

Design and licensing of new reactor concepts

New light-water reactor safety goals

The NRC has established expectations that new reactor designs achieve a higher standard of severe accident safety performance than currently operating designs (US NRC, 1985). As noted in the SRM on SECY-90-016 the Commission supports a goal of 10^{-4} /year CDF and 10^{-6} /year for large release frequency (LRF) for use by the staff in its assessment of evolutionary light-water reactors (LWRs) (US NRC, 1990). It should be noted that LRF used in the licensing of new reactors is closely related to LERF used in regulatory applications for operating reactors in that both metrics are associated with large radionuclide releases of the order of a percent or more of core inventory. However, LERF is associated with releases before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects (ASME, 2009) whereas LRF does not specify timing of release.

Additionally, the Commission approved a conditional containment failure probability (CCFP) objective of 0.1 as a basis for establishing regulatory guidance for evolutionary designs.

Advanced reactor safety goals

The safety goals for new LWR designs are not necessarily applicable to many advanced non-LWR concepts such as high-temperature gas cooled reactors, liquid metal fast reactors, and molten salt reactor designs. In this regard, the 2008 NRC Policy Statement on the Regulation of Advanced Reactors (US NRC, 2008) provides more generalized Commission expectations:

‘...that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.’

More specifically, the 2008 Policy Statement identifies attributes that could assist in establishing the acceptability of a proposed advanced reactor design or the ability to license it, including:

- Reliable and less complex shutdown heat removal systems,
- Longer time constants before reaching safety system challenges,
- Simplified safety systems that reduce required operator actions,
- Reduced potential for severe accidents, and
- Considerations for safety and security requirements together in the design process.

Risk-informed applications for new reactors

The NRC staff engaged with external stakeholders in a series of tabletop exercises regarding the appropriateness of regulatory applications of risk assessment that were developed for the current fleet of reactor designs (US NRC, 2012). Specifically, the tabletop exercises tested various realistic performance deficiencies, events, modifications, and licensing bases changes against NRC policy, regulations, guidance and other requirements (e.g., technical specifications) that are or will be relevant to the licensing bases of new LWRs.

The exercises evaluated many of the widely applied regulatory applications of risk assessment including:

- Risk-informed in-service inspection of piping,
- Risk-managed technical specifications,
- Risk-informed surveillance frequency control program,
- 10 CFR 50.69, risk-informed categorization and treatment of SSCs,
- 10 CFR 50.65(a)(4), maintenance rule, and
- The acceptance guidelines and key principles in RG 1.174.

The NRC staff did not identify any potentially significant decreases in the enhanced safety margins for new LWRs resulting from the implementation of the above regulatory applications. For several of these programs, the staff identified some potential regulatory and implementation issues that would need to be addressed, such as the lack of plant-specific operating experience.

The NRC staff and industry also engaged in a series of tabletop exercises related to the ROP. A broad cross-section of well-vetted cases from actual SDP findings, MSPI data, and event response per MD 8.3 from the current fleet of reactors was assessed. The SDP tabletops indicated that the existing risk thresholds for determining significance of inspection findings are generally acceptable for new LWR designs. However, the existing process would not always ensure an appropriate regulatory response for degradation of passive components and barriers. Similarly, the tabletop exercises for event response demonstrated that the existing risk thresholds for invoking reactive inspections are adequate for new LWRs, and that thresholds could be crossed to invoke reactive inspections.

The case studies developed for the MSPI tabletops showed that the basic MSPI formulation would be largely ineffective in determining an appropriate regulatory response for new LWR designs with active safety features. Furthermore, a meaningful MSPI may not even be possible for passive safety systems using the current formulation of the indicator. Subsequently, the NRC staff has recommended to the Commission that the MSPI not be implemented for the AP1000 in the near future given the limited reactor performance data available for this relatively new design (US NRC, 2018d).

Construction SDP

The NRC established an ROP for new reactors under construction, presently, only the AP1000 reactor design. The construction significance determination process (SDP) under the ROP uses risk insights, where appropriate, to help NRC inspectors and staff determine the safety or security significance of inspection findings identified within the six cornerstones of safety at nuclear reactors that are under construction (US NRC, 2013). The construction SDP is a risk-informed process and the resulting safety significance of findings is used to define a licensee's level of safety performance in constructing the facility and to define the level of NRC engagement with the licensee.

Like the ROP for operating reactors, inspection findings during construction are assigned a color representing the significance of the finding. The color thresholds for the construction SDP were risk informed through the assignment of systems and structures by an expert panel based on risk achievement worth (RAW) values and other risk importance considerations. In addition, finding color thresholds are based on a qualitative measure of construction quality, which was defined through expert staff judgment. Thresholds for non-reactor safety SDPs were similarly developed using either quantitative risk evaluation methods or were risk informed through expert judgment of the staff.

The final determination of significance is similar in nature to that of operating reactors although for reactor construction the significance has more to do with issues regarding the quality of construction of key structures and systems as opposed to actual risk configuration and events during operations.

Licensing of advanced non-LWR reactors

As of July 2019, the US NRC continues to implement a long-term strategic plan for the licensing of advanced non-LWR designs including the development of guidance for a flexible non-LWR regulatory review process (US NRC, 2017).

The principles and methodology in NEI 18-04 (NEI, 2018) provide one acceptable method for informing the licensing basis and determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs. Specifically, NEI 18-04 "presents a modern, technology-inclusive, risk-informed, and performance-based (TI-RIPB) process for selection of Licensing Basis Events (LBEs); safety classification of SSCs and associated risk-informed special treatments; and determination of DID adequacy for non-LWRs." As discussed in NEI 18-04, LBEs consist of the entire collection of event sequences considered in the design and licensing basis of the plant, which may include one or more reactor modules. They include Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Design Basis Accidents (DBAs), and Beyond Design Basis Events (BDBEs). Non-LWRs include, but are not limited to, molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors (e.g., sodium-cooled, lead-cooled, gas-cooled, etc.).

The TI-RIPB process described in NEI 18-04 is:

- Risk-informed to fully utilize the insights from systematic risk assessment in combination with structured prescriptive rules to address the uncertainties which are not addressed in the risk assessment. This approach can provide reasonable assurance that adequate protection is provided for public radiological protection.
- Performance-based to evaluate effectiveness relative to realizing desired outcomes that are achieved by using quantifiable performance metrics for LBE frequencies and consequences and performance requirements for SSC capabilities to prevent and mitigate events. This is an alternative to a prescriptive approach specifying particular features, actions, or programmatic elements to be included in the design or process as the means for achieving desired objectives.

Comparing design alternatives

Risk insights can be used to inform the design of new reactor concepts, or to enhance the safety performance of operating reactors. For example, the design of a new reactor concept may begin with no detailed specifications of the design beyond the reactor core and the heat transport mechanism. Existing regulations may dictate general requirements and, in some instances, more specific requirements, but the designer is generally at liberty to decide how to meet aspects of the DID philosophy and safety margins. This flexibility often includes the degree of diversity and redundancy of mitigating systems for anticipated operational occurrences, design basis events, design basis accidents, and beyond design basis events.

For operating reactors, new requirements in the form of Orders and regulations continue to be issued, e.g. response to the Fukushima accident. While meeting the minimum requirements of the regulation, the licensee may choose to integrate the required change with discretionary changes to more effectively enhance plant safety. Similarly, plant improvements to mitigate the consequences of fire implemented under NFPA 805 (NFPA, 2015) may also improve the risk profile of the nuclear power plant for other types of hazards and events.

Risk insights and reliability studies are often used to ensure that plant modifications do not result in undesired consequences. Thus, mitigating systems should be designed with sufficient reliability to perform their intended function when required, but not spontaneously actuate when not required (e.g., 2-out-of-4 actuation logic).

Finally, risk-informed approaches may be utilized to optimize plant operational performance through the appropriate balancing of capital, operating, and maintenance costs. Cost-benefit analysis may be utilized and is a key aspect of severe accident mitigation alternatives (SAMA) analysis (NEI, 2005a) for license renewal.

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Risk Assessment Insights and Impact

Karl N. Fleming, KNF Consulting Services LLC, Spokane, WA, United States

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Abbreviations

AFW Auxiliary feedwater
ANS American Nuclear Society
AOO Anticipated Operational Occurrence
ASME American Society of Mechanical Engineers
ATWS Anticipated transient without scram
BDBE Beyond Design Basis Event
BWR Boiling water reactor
CCDF Complementary cumulative distribution function
CDF Core damage frequency
CFR Code of Federal Regulations
DBA Design Basis Accident
DBE Design Basis Event
DG Diesel-generator
DID Defense-in-depth
EAB Exclusion Area Boundary
ECCS Emergency core cooling system
F-C Frequency-consequence
F-V Fussell-Vesely
IAEA International Atomic Energy Agency
IE Initiating event
KV Kilovolt
LBE Licensing Basis Event
LERF Large early release frequency
LMP Licensing Modernization Project
LOCA Loss of coolant accident
LPSD Low power and shutdown
LWR Light water reactor
MOV Motor operated valve
mrem Millirem
NEI Nuclear Energy Institute
NRC Nuclear Regulatory Commission

NSRST Non-Safety-Related with Special Treatment
 NST Non-Safety-Related with No Special Treatment
 PB* Performance-based
 PRA Probabilistic risk assessment
 PWR Pressurized water reactor
 QHO Quantitative Health Objective
 RAW Risk achievement worth
 RCP Reactor coolant pump
 rem Roentgen-Equivalent Man
 RI Risk-informed
 RIPB Risk-informed and performance-based
 RSS Reactor Safety Study
 RT Reactor trip
 SR Safety-Related
 SSC Structures, systems, and components
 TD Turbine driven
 TMI-2 Three Mile Island Unit 2

Introduction

The purpose of this chapter is to discuss the development of risk insights in analyzing the results from a PRA and their use in risk-informed decision making. This section includes a brief review of the role of risk metrics and risk importance measures to identify the principal contributors to risk and to evaluate the sensitivity of the risk levels to variations in input parameters and models. Representative examples of some of the most noteworthy risk insights about risk contributors and how those insights were different than insights from traditional deterministic safety analyses are provided. A second use of insights covered here is that in risk-informed decision making associated with changes to plant designs and operations to improve safety and to prioritize resources on areas most important to managing risk. Additional examples of risk insights derived from risk-informed decisions are included. The third and final category of risk insights covered in this chapter is their use to define and evaluate the safety case for advanced reactors by risk-informed decisions in the selection of licensing basis events, safety classification of SSCs, and evaluation of defense-in-depth adequacy. The chapter concludes with a perspective on the impact of use of risk insights derived from PRA and PRA applications on the level of safety exhibited by operating and advanced reactors.

Evaluating the results of a PRA for risk insights

Examples of risk insights from PRAs

During the first two decades of PRAs on nuclear power plants, starting with the Reactor Safety Study (RSS) (USNRC, 1975), a significant body of work was accumulated in the performance of Level 3 PRAs.¹ Some key insights from this early work were reviewed in (Fleming, 2003).

Many of the methods and risk insights that we still use today were introduced in that landmark study (WASH-1400), and shortly thereafter in industry, led to PRAs being performed for such plants as Zion, Indian Point, and Oyster Creek (Garrick, 1988). The accident sequences that were found to make the most important contributions to the risk of a severe accident were found not to be correlated to the design basis accidents (DBAs).² Hence, small break loss of coolant accidents were found to be much more significant to risk than those initiated by large pipe breaks. Moreover, the concept that conservative safety analyses of design basis accidents can establish an upper bound on the risk to public health and safety was refuted. In fact, the risks calculated in the RSS were found to be completely determined by severe core damage accidents that are more severe than the DBAs. Although it was obvious that the consequences of a severe core damage event would exceed those of a design basis event, a key insight here was that the frequency of severe core damage events was much higher than expected using traditional deterministic approaches to safety.

¹A Level 1 PRA is limited in scope to determine the core damage frequency (CDF); A Level 2 PRA expands the scope to estimate the frequencies and radioactive material release source terms from a severe accident; and a Level 3 PRA extends the risk assessment to include a quantification of the offsite consequences of releases on public health and safety and property damage (USNRC, 1983).

²DBAs are accident sequences that are analyzed in the reactor license applications. They are postulated in a conservative manner so as to bound the consequences of accidents deemed credible. They are selected based on qualitative judgments of likelihood and their consequences are evaluated using conservative assumptions. In PRA, accidents more severe than the DBAs are defined and assessed in terms of quantitative estimates of their frequency and consequences using realistic assumptions.

Another key insight from the RSS and the early industry studies was that human errors make a significant contribution to the risk of a severe accident.

Traditional deterministic approaches to safety suggested that accidents require the postulation of failure of several independent systems and fission product barriers following an initiating event. Such thinking supported the qualitative judgments made in the deterministic framework of safety analysis that accidents more severe than the design basis accidents were so unlikely as to be negligible in the definition of the design basis and associated general design criteria. With the benefits of insights from application of PRA technology as well as lessons learned from various incidents and accidents such as the Browns Ferry Fire (Rempe, 2021) and the Three Mile Island accident (Skillman and Rempe, 2021), it is now clear to most PRA practitioners that severe core damage events are in fact much more likely than the DBAs. If one were to go back and assess the frequency of the DBA initiating events and realistic probabilities of the specific success and failure combinations that are assumed in the safety analysis reports, frequencies much lower than currently calculated core damage frequencies would result.

In the first PRA on LWRs, the Reactor Safety Study (RSS) was limited to internal events initiated at full power with only a qualitative evaluation of external hazards. While the concept of PRA levels was not introduced until publication of the PRA Procedures Guide (USNRC, 1983), the RSS can be classified as a Level 3 PRA. Almost immediately thereafter the scopes of PRAs performed on Zion, Indian Point, and Oyster Creek was expanded to address internal fires and floods, and seismic events. The first seismic PRA ever performed was that on Oyster Creek (Garrick, 1988). The next two decades of PRA development were primarily devoted to Level 3 PRA. This phase of PRA development was responsible for incorporating external hazards and eventually non-full power plant operating states. Risk insights that were developed in this phase of PRA have been summarized in Apostolakis et al. (2017), Fleming (2003), Garrick (1988), and USNRC (1990).

Examples of some of the most noteworthy insights are summarized in Table 1. The list of insights in this table is not intended to be comprehensive, but rather representative of the key insights from the extensive body of work in PRAs that have been completed

Table 1 Examples of risk insights regarding the principal contributors to risk in LWRs.

<i>Insight category</i>	<i>Insight</i>	<i>References</i>
Significant Contributions to Risk of Severe Core Damage	Risk of severe accidents is not bounded by that from the design basis accidents	Fleming (2003)
	Small loss of coolant accidents (LOCAs) exhibit much greater risk significance than large LOCAs	USNRC (1975)
	Support system faults, e.g., loss of offsite power and station blackout, loss of service water, component cooling water and safety system dependence on these systems are important risk contributors from the internal events hazard group	Apostolakis et al. (2017) and Garrick (1988)
	Common cause failures dominate the risk for accident sequences involving failure of redundant systems, e.g. anticipated transients without scram (ATWS), station blackout; independent failures in these system seldom if ever make significant contributions to risk	Mosleh et al. (1997)
	Significant risk importance identified in modeling dependencies of passive components on active systems, e.g. reactor coolant pump seal LOCAs resulting from cooling system faults	Fleming (1996)
	Importance of risk contributions identified from human errors of omission (e.g. failure of ECCS recirculation switchover following small LOCA) and commission (e.g., turning off high pressure injection during TMI-2 accident)	Apostolakis et al. (2017) and Kemeny et al. (1979)
	Significant risk contributions identified from accident sequences caused by seismic events on many sites	USNRC (1990)
	Significant risk contributions of accident sequences caused by internal fires and floods identified for selected plants	NSAC (1984)
	Unexpected finding that accidents initiated during low power and shutdown states more important than initially assumed	NSAC (1985)
	In general the level of risk and contributors to risk are plant and site specific	Garrick (1988) and Apostolakis et al. (2017)
Significant Contributions to risk of large early release and Level 3 risk metrics	Containments found to be effective in reducing the risk of a large release for current operating LWR designs	Apostolakis et al. (2017) and USNRC (1990)
	Dominance of containment bypass sequences involving interfacing systems LOCAs and steam generator tube ruptures for PWRs with large dry containments	PLG (1983, 1986a)
	The dominant accident progression bins for Mark I BWRs are early containment failure and drywell failure caused by drywell failure and loads at vessel breach (due to direct containment heating, steam blowdown, or quasistatic pressure from steam explosion)	USNRC (1990)
	Source terms for severe accidents found to be significantly less than those estimated in earlier PRAs for selected accident sequences	USNRC (2012)
	On sites with multiple reactor units, the risk of accident sequences involving two or more reactor units is significant for all sites and dominant for plants with shared systems and large risk contributions from external hazards	IAEA (2019), CNSC (2014), Fleming (2005), and PLG (1983)
	Risk reduction benefits of evacuation localized to 1-mile from the site boundary; extending the emergency planning zones from 1-mile to 10 miles achieves insignificant risk reduction benefit	PLG (1986b)

for operating LWR plants. These insights caused a major change in identifying the most important issues that contribute to nuclear safety in comparison with that which was associated with deterministic approaches to safety and inspired the movement toward risk-informed regulation culminating in many plant design and operational changes and associated license amendments using the decision making process described in USNRC Regulatory Guide 1.174 (USNRC, 2018).

One of most interesting insights from this early work was that the level of risk and principal contributors to risk was highly design and site specific. Indeed, the risk levels and contributors were not driven that much by reactor type or containment type, but more by details in how the plant was designed to protect against hazards, site characteristics, the vintage of the design, the engineering firms that built the plants, and other factors that determined the dependencies between front line safety systems and support systems, etc. For example the PRAs reviewed in Garrick (1988) and Apostolakis et al. (2017) included those on Zion, Indian Point, Diablo Canyon, South Texas, Seabrook, Sequoyah, Beaver Valley, and Watts Bar, all of which were Westinghouse PWRs with large dry containments. Yet the risk levels, risk contributors, and risk insights from these PRAs are all different and not highly correlated to reactor and containment type. Such insights were not available in NUREG-1150 for example because that landmark study selected only one plant from a specific reactor and containment type.

Another important insight from these early PRAs is that a full appreciation of the insights from a PRA requires the consideration and quantification of uncertainties (Kaplan and Garrick, 1981). It is important to sort out the role of the uncertainty in producing a PRA result so that the most effective risk management strategy can be devised. For risk contributions from many internal events, the uncertainties in the reliability performance of SSCs supported by both generic and plant specific data will tend to be very small. In these cases there is some confidence that strategies can be devised to improve the SSC availability or reliability to reduce the risk of these sequences. However, if the risk is dominated by a seismically induced failure of an SSC, such risk is most likely driven by uncertainties in the seismic hazard for that site in which case equipment reliability strategies would be ineffective in management of the risks.

As PRA technology evolved over the next two decades, additional important risk insights were obtained. The first efforts to assess the risk of accident sequences initiated during low power and shutdown (LPSD) plant operating states indicated that the risk of such sequences was significant and much greater than initially assumed (NSAC, 1985). This resulted from a combination of factors including the general relaxation of technical specifications during shutdown, extensive maintenance activities being performed during refueling outages, and difficulties in achieving containment isolation during such activities. With the benefit of risk insights from the early LPSD PRAs and the implementation of risk monitoring tools to track risk profile changes due to changes in plant configurations, many changes were made to manage the risks in all modes of operation. In parallel with these efforts, risk-informed changes to technical specifications were made using the decision making process in RG 1.174 to manage the risks of performing maintenance with the plant at power. As a result, an overall improvement in risk management was achieved in all modes of operation.

One of the limitations of practically all the PRAs that have been performed is application of a “one reactor at a time mindset” in which each reactor unit is assessed under the assumption that the other reactor units on the same site are safe. The first PRA that was performed that explicitly addressed the risks from accidents involving multiple reactor units concurrently was that on Seabrook Station (PLG, 1983). Even though the two reactor units that were originally intended to be built (the second unit was never completed) had dedicated containments and safety systems and sharing of systems was limited to the use of a common switchyard and ultimate heat sink, the risk of multi-unit accidents was found to be significant. Later some PRAs on sites with extensive sharing of systems found that multi-unit accident sequences dominated core damage frequency (CDF) and large early release frequency (LERF) (IAEA, 2019). The “one reactor at a time mindset” has been used to develop conclusions about satisfaction of the USNRC safety goals, even though the wording of the safety goal policy (USNRC, 1986) suggests otherwise. This mindset has been used to support single unit PRAs on multi-unit sites in which all the units not being analyzed are assumed, besides the one being investigated, to be safe. Based on the wording used in the safety goal policy, a case can be made that the safety goals should be applied to the site as a whole and that conclusions about the satisfaction of the safety goals would require the performance of a multi-unit PRA (Fleming, 2005). Hence, it can be easily seen that interpretation of safety goal policy is dependent on risk insights and the nature and scope of the PRA whose results are interpreted to develop those insights.

Risk metrics and methods for deriving risk insights

Risk insights are developed by presenting, interpreting, and analyzing the results of a PRA. In the decade or more that followed the RSS, most of the published PRAs were Level 3 PRAs; and by the end of the first decade, the PRAs were typically expanded to include a full range of internal and external hazards as well as both full power and low power and shutdown operating states (Garrick, 1988; Apostolakis et al., 2017).

In the most recent decade of PRA development for operating LWR plants, the scope of the PRAs has been focused on estimating CDF and LERF and adhering to the technical requirements in the ASME/ANS PRA Standard (ASME/ANS, 2013a). This standard has requirements for developing the risk models needed to estimate CDF and LERF for single reactor PRAs. These risk metrics are discussed more fully in Modarres and Kim (2021). Although much of the current focus in risk-informed applications of PRA makes use of these risk metrics, a more complete set of risk insights requires a full scope Level 3 PRA that addresses both internal and external hazards, all risk significant plant operating states, and accidents involving two or more reactors and sources.

The principal risk metrics and methods that are typically used to derive risk insights from PRAs are summarized in Table 2. The ASME/ANS PRA Standard focuses on the following risk metrics for determining what is risk significant and for delineating the risk contributors to CDF and LERF.

Table 2 Risk metrics and metrics used to develop risk insights.

Category	Metrics and methods	Role in developing risk insights	References
Level 3 PRAs	Exceedance frequency versus consequence levels ^a	Absolute statements of the integrated risks of various consequence metrics including population dose exposures, early and latent health effects, property damage, etc.	USNRC (1975) and USNRC (1983)
	Individual risks of early and latent fatalities	Metrics for evaluating risks against the NRC Safety Goal Quantitative Health Objectives (QHOs)	USNRC (1986)
	Matrix formalism for analysis of Level 3 PRA results	Systematic method for identifying the initiating events, accident sequences, plant damage states, and release categories that contributed to the overall risk	Kaplan (1982) and PLG (1983)
Level 1/LERF PRAs	Accident sequences grouped by common elements including initiating events, plant damage states, dependencies, and classes of contributors	Relative risk contributions of classes of accident sequences grouped by a common PRA model element	Garrick (1988) and Apostolakis et al. (2017)
	Rank ordered contributions from individual accident sequences and cutsets	Relative risk contributions of sequences and cutsets to CDF and LERF	Garrick (1988) and Apostolakis et al. (2017)
	Risk importance measures normalized against baseline risk measures	Relative risk contributions of basic events to CDF and LERF; sensitivities of CDF and LERF to changes in basic event probabilities and uncertainties	Van der Böst and Shoonakker (2001)
	Risk importance measured against fixed risk targets.	Quantifies increase in basic event, system, or other PRA modeling element necessary to reach a defined risk target	Johnson and Apostolakis (2009)
	Changes in CDF and LERF due to plant design and operational Changes	Basic metrics used to demonstrate that plant and operational changes due not result in significant changes to CDF or LERF	USNRC (2018)

^aThese are often referred to as Complementary Cumulative Distribution Functions (CCDFs).

- Percentage contributions to CDF and LERF from the accident sequences modeled in the PRA.
 - The collection of the accident sequences that make at least 95% of the CDF and LERF are required to be modeled using realistic assumptions so as to avoid the masking of risk insights using conservative assumptions. Meeting this requirement makes it possible to define the risk significant parts of a PRA model.
 - Individual accident sequences that contribute at least 1% of the CDF and LERF are defined as risk-significant accident sequences.
- Risk importance measures are used to determine which basic events³ make risk significant contributions to CDF and LERF.
 - If the accident sequences involving a basic event contribute at least 0.5% to either CDF or LERF, the basic event is defined as risk-significant
 - If either CDF or LERF increases by a factor of at least 2.0 when the basic is assumed to be failed, the basic event is classified as risk-significant.

By combining the basic events associated with specific SSCs and human failure events, the risk importance of SSCs and operator actions can be derived from the basic event importance measures.

Because the ASME/ANS PRA standard is limited to the above risk metrics, current PRAs and their associated risk insights are largely confined to those. Some pitfalls and potential misleading conclusions that be obtained by relying on these risk metrics are explored in the next section.

Pitfalls in developing useful risk insights from risk importance measures

Definitions, uses, and potential pitfalls in deriving risk insights from risk important measures have been reviewed in Van der Böst and Shoonakker (2001) and Fleming (1996). Risk importance measures such as F-V and RAW are defined for elements of the PRA model such as basic events. The basic events in turn are associated with systems, structures, and components (SSCs) and operator actions whose failures may contribute to risk. In recent PRA applications such as those to implement 10 CFR 50.69, (2008) these measures are used to establish the safety significance of SSC in which special treatment requirements for safety related non-safety significant SSCs may be relaxed. However, this practice of directly linking the risk importance of PRA model elements to ranking and

³Each accident sequences in a PRA is modeled as a sequence of basic events. The basic events represent initiating events and failures of SSCs and human actions that designed to mitigate the consequences of the initiating events. A given basic event may, and often do, appear on more than one accident sequence.

prioritizing activities associated with SSCs must be done with care so as to avoid certain pitfalls and derive the appropriate risk insights. These pitfalls are briefly discussed below with an illustrative example.

Lack of independence of importance measures

The risk importance measures for a set of SSCs or other PSA model elements are not independent parameters, e.g., the F-V importance for component X is not independent of the Fussell-Vesely parameter for component Y. This property is obvious from the mathematical definitions of the importance measures, but in many applications they are treated as though they were independent parameters. The importance assigned to each component is derived from the accident sequences in the PRA model in which that component, or the basic events assigned to that component, are modeled as failed. The basic event probabilities on these accident sequences may be reasonably assumed to be independent of other basic event probabilities. However, a given accident sequence involves other basic events not associated with that component that may belong to another component or an initiating event or operator action shared with other components along the same sequences. In addition, another key factor that contributes to any given sequence are the functional dependencies of one component on another. The lack of independence means that the risk importance measure for a component does not fully “belong to” that component and that there may be single issues that determine the risk importance measures for a whole host of components that are involved on the same accident sequences.

Individual character of importance measures

Risk importance measures examine the risk contributions and risk sensitivities one model element at a time. The combined impacts of changing the probability of two or more model elements cannot be reliably inferred from individual risk importance parameters. Any change made in the design, operation, or maintenance of the plant that can impact the risk is not accounted for in risk importance parameters computed for a baseline PSA. If care is exercised, one may be able to infer the risk impact of a few isolated changes from the risk importance measures. However, for many PSA applications, it is necessary to update the PSA to reflect the proposed change so that the cumulative effects of the change on two or more elements of the PSA model can be evaluated.

Risk importance of SSCs and PSA model elements that cause initiating events

There are several problems here; one already noted is the contribution that initiating events make to risk importance. In principle, fault tree models might be used for initiating event frequencies, and when this is done, the corresponding basic events can be ascribed a F-V importance. Indeed, one can easily ascribe importance to the event as a whole and this has been performed. When initiating events are modeled only as a basic event with an assigned frequency, the definition of the underlying causes become obscured as these causes are “buried” in the data. An additional problem is that RAW is not defined for initiating events since basic event frequencies are unbounded quantities, i.e., there is no technical basis for limiting the initiating event frequency to 1/year.

A frequent interpretation of the RAW for a given SSC is that it provides an indication of the risk impact of taking an equipment item out of service. While this interpretation may be reasonable in many cases, the full impact of the out of service condition is not always captured in the RAW measure. Taking a component out of service for testing or maintenance may imply a much higher likelihood of an initiating event due to human errors in performing these tasks, whose risk impact is not captured in the component RAW value.

Computational issues

The numerical values obtained in attempts to assign risk importance to equipment may be in error due to a variety of problems such as judgmental exclusion of the equipment from the PSA model, model truncation during the quantification process, implicit representation of the equipment that obscures the true risk impact, simplifying assumptions in the PSA model, scope limitations of the model, and other non-physical aspects of the PSA. Many of these issues are a bit more problematic for the RAW in comparison with F-V, since the argument to exclude the item from the model, however reasonable, might have been based on a low probability or low frequency argument. The possibilities for truncation need to be considered lest there be too sweeping a conclusion made from the results of a table of RAW values.

Masking risk insights regarding dependencies

One of the most important issues in deriving risk insights from risk importance measures is that the methods used to evaluate them seldom permit the identification of the risk importance of dependencies that are typically hardwired into the event tree/fault tree models. The following example described in [Fleming \(1996\)](#) provides an illustration of this pitfall.

For this example, the top ranking SSCs according to Fussell-Vesely (F-V)⁴ importance from an internal events full power PRA for an 3 loop PWR are shown in [Table 3](#) and some important information about the underlying accident sequences responsible for these F-V values in [Fig. 1](#).

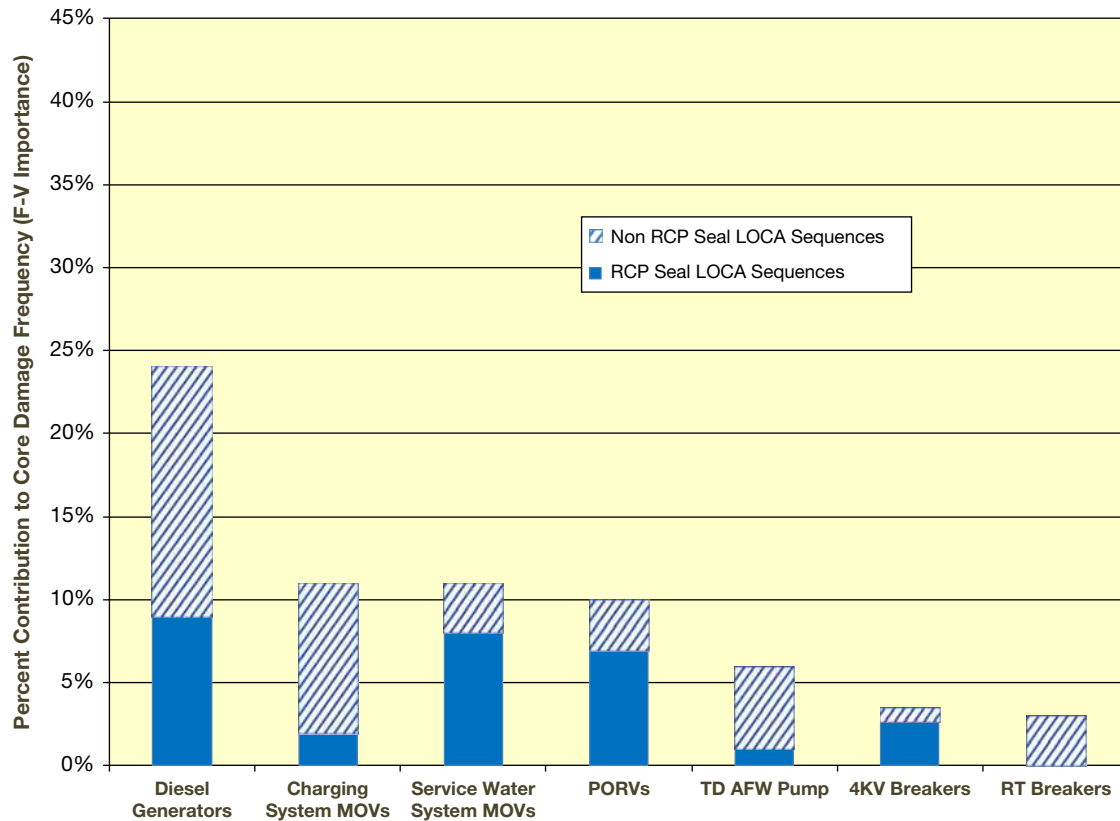
There is a tendency in PRA applications to take results in the form of [Table 1](#) and conclude that such results can be used to prioritize actions to improve SSC reliability and availability that help determine the underlying basic event probabilities. If the F-V values

⁴Fussell-Vesely (F-V) importance is the fraction of the CDF associated with accident sequences in which the identified component is failed.

Table 3 SSC importance from a selected PRA.

Component	F-V importance
Diesel Generators (DGs)	0.24
Charging System Motor Operated Valves (MOV)s	0.11
Service Water System MOVs	0.11
Pilot Operated Relief Valves (PORVs) ^a	0.10
Turbine Driven Auxiliary Feedwater (TD AFW) Pump	0.059
4 Kilovolt Electrical Breakers	0.035
Reactor Trip (RT) Breakers	0.030

^aPORVs are valves that can be actuated by a pilot valve to reduce the pressure in a reactor coolant system.

**Fig. 1** Contributions to SSC importance by accident sequences involving reactor coolant pump seal LOCAs.

in **Table 1** are used for this purpose, there is a strong tendency to look at SSC availability and reliability issues that contribute to the basic event probabilities associated with these components. It turns out in this example that much of the risk importance listed in **Table 1** is associated with accident sequences involving reactor coolant pump seal LOCAs. This insight is gained by examining the accident sequences that contribute to the top ranking SSCs on the F-V importance table. While failures of the SSCs in question contributed to these sequences, the dependence of the integrity of the reactor coolant pump (RCP) seals on those SSCs is key to understanding why these sequences resulted in core damage.

Another risk insight that is gained from examination of the accident sequences underlying the F-V importance results is that the two MOV examples and the reactor trip breaker examples are associated with common cause failure of redundant MOVs and breakers. In addition, the TD AFW pump importance and much of the DG importance is from station blackout sequences initiated by loss of offsite power which overlap the RCP seal LOCA sequences. The DG importance is also dominated by common cause failures.

In **Fig. 2**, the importance information is rearranged to isolate out the risk importance of the RCP seal LOCA dependency, and revising the SSC values to exclude the contributions from RCP seal LOCA sequences (Fleming, 1996). This figure is much more valuable for developing risk management strategies for the top ranking F-V SSCs. It clearly shows how strategies to eliminate the RCP seal integrity dependence on active cooling systems would have a much greater impact in reducing CDF than trying to rely

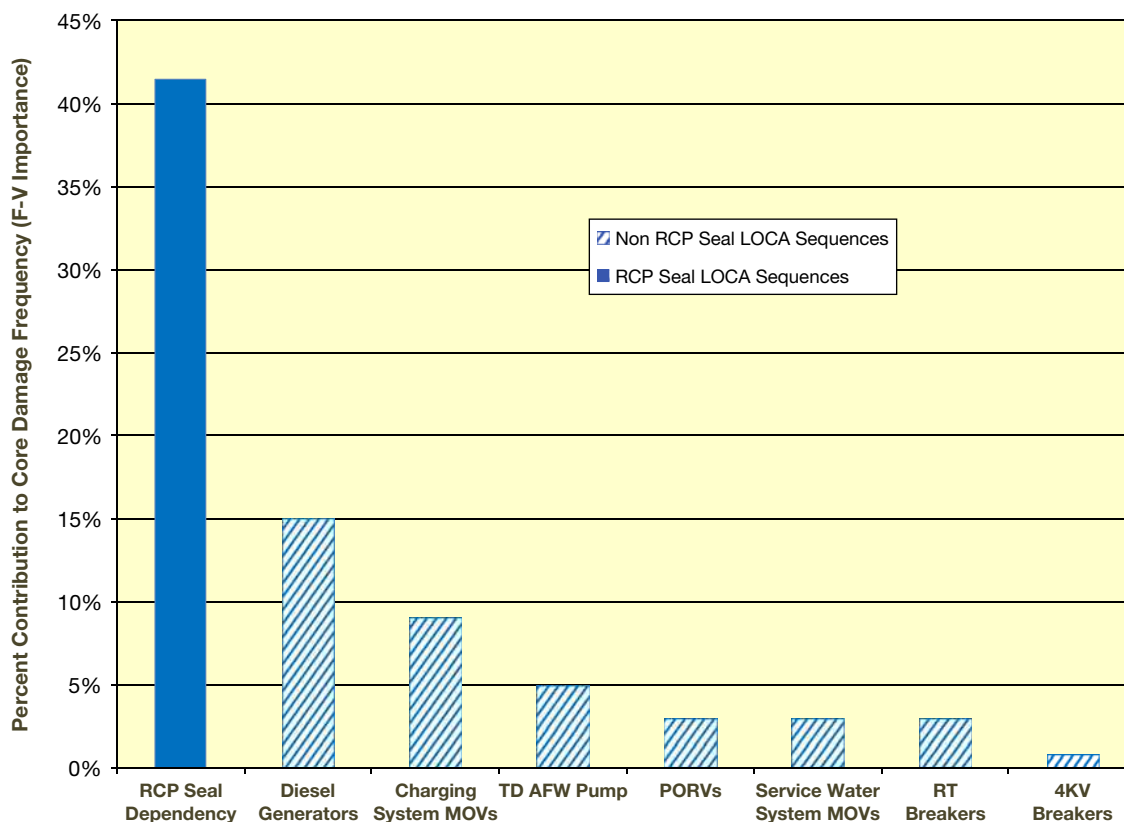


Fig. 2 Revised SSC importance determined by isolating importance of RCP seal LOCA dependency.

on equipment reliability and availability priorities derived from [Table 3](#). In addition, the examination of the sequences contributing to the remaining importance results in [Fig. 2](#) would lead to the additional risk management strategies that would be effective in reducing CDF.

With a more complete understanding of the accident sequences and dependencies underlying the F-V values in [Table 3](#) the following risk management strategies would have the greatest opportunity to reduce CDF:

- Eliminate the dependence of the RCP seal integrity on active cooling system.
- Identify strategies to reduce the likelihood of common cause failures for
 - the charging system and service water system MOVs,
 - DGs, and
 - reactor trip breakers
- Identify strategies to reduce the frequency of loss of offsite power and transient events that challenge the ability of the plant to prevent station blackout and ATWS

Any of the above strategies is likely to be more beneficial to reducing CDF than by improving the reliability and availability of the SSCs in [Table 3](#). Such strategies would be second order effects.

The above example illustrates the pitfalls of the use of importance measures to directly associate them with equipment reliability and availability performance, without a detailed review of the underlying accident sequences and the system dependencies that are often hard wired into the PRA model and reflected in the sequences.

Use of risk insights in risk-informed regulation

An area where use of risk insights has arguably the greatest impact on the safety of operating LWRs is risk-informed regulation. Industry initiatives such as the EPRI PSA Application Guide ([True et al., 1995](#)) proposed that changes to the plant's operating licenses be permitted based in part on a demonstration that such changes would not have adverse impacts on risk. The now commonly used risk metric, Large Early Release Frequency (LERF), was proposed in this Guide as a means of addressing risk using CDF and LERF and obviating the need for a full scope Level 2 or Level 3 PRA. However, the key regulatory initiatives that made risk-informed regulation a reality were the NRC Commission policies on safety goals ([USNRC, 1986](#)) and use of PRA ([USNRC, 1995](#)), and the issuance of a regulatory guide, RG 1.174 ([USNRC, 2018](#)) and supporting guides that defined a risk-informed decision making framework for evaluating proposed changes to the licensing basis. This framework included an evaluation of the changes that would pose no significant risk increase, and would maintain defense-in-depth, safety margins, and adherence to the

regulations. RG 1.174 included numerical guidelines for demonstrating that changes in CDF and LERF brought about by the proposed license amendments would be sufficiently small. These guidelines were tied to the baseline results of the plant PRAs. One of the supporting regulatory guides was RG 1.200 (USNRC, 2009) which sets forth NRC guidance on the technical adequacy of the PRAs used for this purpose including the endorsement of industry PRA standards. More information on the use of this guide is found in Dube (2021).

The types of changes originally addressed in risk informed regulation included risk-informed changes to technical specification, inservice inspection of piping systems, inservice testing of safety related SSCs, containment leak-rate testing, and graded quality assurance. The last item on this list was expanded to an application involving a risk-informed reclassification of SSCs and relaxation of special treatment requirements for safety related SSCs that were found not be safety significant. This concept of safety significance was incorporated into the regulations in 10 CFR 50.69 and includes consideration of risk significance and preservation of defense-in-depth. Risk significance criteria for SSCs are based on those identified in the PRA standard (ASME/ANS, 2013) including those based on F-V and RAW importance. SSCs with F-V importance exceeding .005 and RAW exceeding 2.0, for either CDF or LERF, are regarded as risk significant in the standard.

Incorporating risk insights into advanced reactor design and licensing

For advanced non-LWRs under development, a technology inclusive, risk-informed, and performance-based methodology for supporting design and licensing has been developed as part of the industry led Licensing Modernization Project (LMP) (NEI, 2019). The LMP includes a risk-informed and performance-based decision making process for the following applications:

- Selection and evaluation of Licensing Basis Events (LBEs)
- Safety classification of SSCs and selection of SSC performance requirements
- Evaluation of defense-in-depth adequacy.

The LBEs that are selected and evaluated in this methodology address event sequences involving multiple reactors and non-reactor radionuclide sources.

The NRC has proposed to the Commissioners that the LMP methodology be approved to inform the content of license applications for future advanced non-LWRs (USNRC, 2019).

The LBEs in the LMP methodology are broadly defined to include all the events considered in the design, licensing, and formulation of the safety case for the reactor plant. Each LBE is a family of event sequences from a design specific PRA where event sequences are grouped according to similarity of the initiating event, plant response to the initiating event, and mechanistic source term for sequences involving a release of radioactive material from any source of radionuclides that has the potential for producing

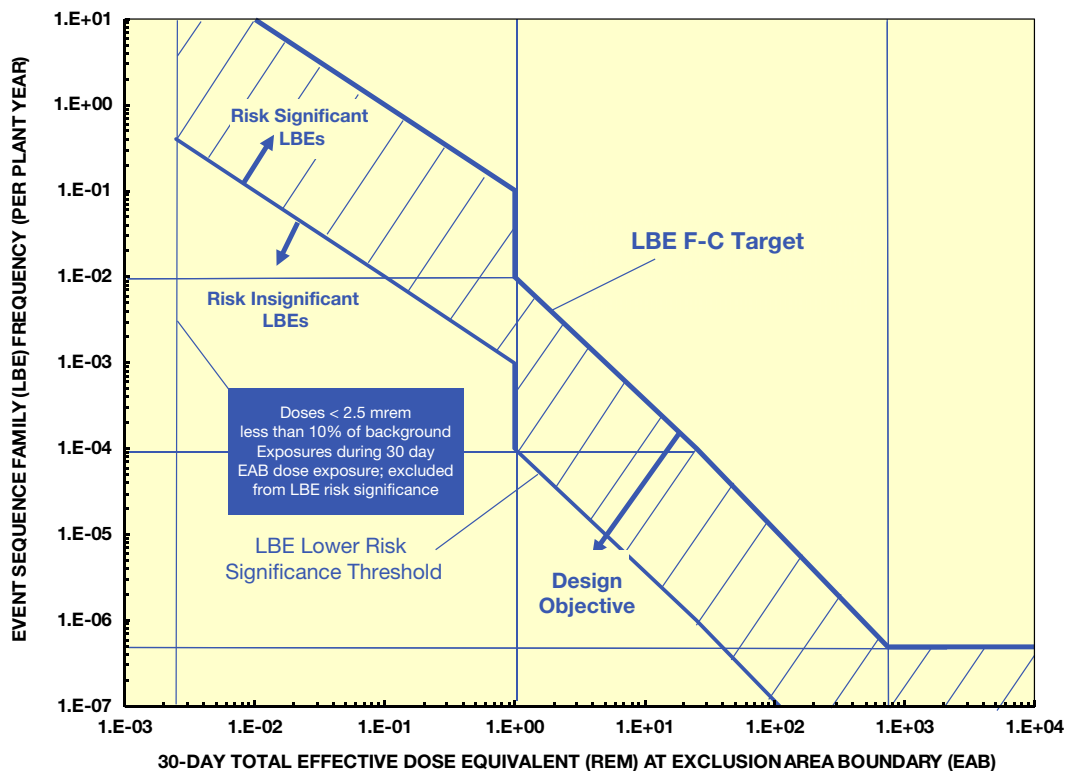


Fig. 3 LMP frequency-consequence targets for evaluating LBE risk significance © 2019 Nuclear Energy Institute (NEI, 2019).

a risk significant LBE. Risk significance of an LBE is determined by comparing the frequencies, consequences, and uncertainties about the frequency and consequence estimates against a Frequency-Consequence (F-C) target. The risk significance criteria for LBEs are shown in Fig. 3.

The basis for the F-C Target in Fig. 3 is an interpretation of the NRC regulations and policies that relate to limits on the frequency and consequences of anticipated events and accidents. Details are discussed in NEI (2019).

There are four categories of LBEs in the LMP methodology as defined in Table 4. Importantly, LBEs may involve event sequences involving two or more reactors or radionuclide sources within the scope of the plant being evaluated. Hence, the frequencies are expressed on a plant-year basis where a plant may be comprised of multiple reactors and radionuclide sources. This requires the inclusion of these types of event sequences within the scope of the supporting PRA. A trial use PRA standard for advanced non-LWRs (ASME/ANS, 2013b) has been developed and is planned for a major upgrade in 2020 to incorporate feedback from pilot PRAs that exercised the trial use standard.

LMP also includes risk targets for the cumulative risks evaluated in a PRA that account for all the individual LBEs. These cumulative risk targets are selected to assure that annual dose limits from Anticipated Operational Occurrences do not exceed 100 mrem per year, and the NRC goal QHOs for individual risk of fatality from early and latent health effects (USNRC, 1986). Specifically the cumulative risk targets in NEI 18-04 include (NEI, 2019):

- “The total mean frequency of exceeding a site boundary dose of 100 mrem from all LBEs should not exceed 1/plant-year. This metric is introduced to ensure that the consequences from the entire range of LBEs from higher frequency, lower consequences to lower frequency, higher consequences are considered. The value of 100 mrem is selected from the annual cumulative exposure limits in 10 CFR 20.
- The average individual risk of early fatality within 1 mile of the exclusion area boundary (EAB) from all LBEs based on mean estimates of frequencies and consequences shall not exceed 5×10^{-7} /plant-year to ensure that the NRC safety goal QHO for early fatality risk is met.
- The average individual risk of latent cancer fatalities within 10 miles of the EAB from all LBEs based on mean estimates of frequencies and consequences shall not exceed 2×10^{-6} /plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.”

SSC risk significance in the LMP methodology makes use of risk significance criteria that are anchored to fixed targets (NEI, 2019).

- “The total mean frequency of exceeding a site boundary dose of 100 mrem should not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded based on the mean estimates of frequencies and consequences. An SSC makes a significant contribution to this cumulative risk metric if the total frequency of exceeding a site boundary dose of 100 mrem associated with LBEs with the SSC failed is greater than 10^{-2} /plant-year.
- The average individual risk of early fatality within 1 mile of the EAB shall not exceed 5×10^{-7} /plant-year based on the mean estimates of frequencies and consequences to ensure that the NRC safety goal QHO for early fatality risk is met. An SSC makes a significant contribution to this cumulative metric if the individual risk of early fatalities associated with the LBEs with the SSC failed is greater than 5×10^{-9} /plant-year.

Table 4 LBE categories in the LMP methodology © 2019 Nuclear Energy Institute (NEI, 2019).

Event type	LMP definition
Anticipated Operational Occurrences (AOOs)	Anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of 1×10^{-2} /plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification
Design Basis Events (DBEs)	Infrequent event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than AOOs. Event sequences with mean frequencies of 1×10^{-4} /plant-year to 1×10^{-2} /plant-year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification
Beyond Design Basis Events (BDBEs)	Rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. Event sequences with mean frequencies of 5×10^{-7} /plant-year to 1×10^{-4} /plant-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification
Design Basis Accidents (DBAs)	Postulated event sequences that are used to set design criteria and performance objectives for the design of Safety Related SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of Safety-Related SSCs needed to mitigate and prevent event sequences, respectively. DBAs are derived from the DBEs by prescriptively assuming that only Safety Related SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits
Licensing Basis Events (LBEs)	The entire collection of event sequences considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include AOOs, DBEs, BDBEs, and DBAs

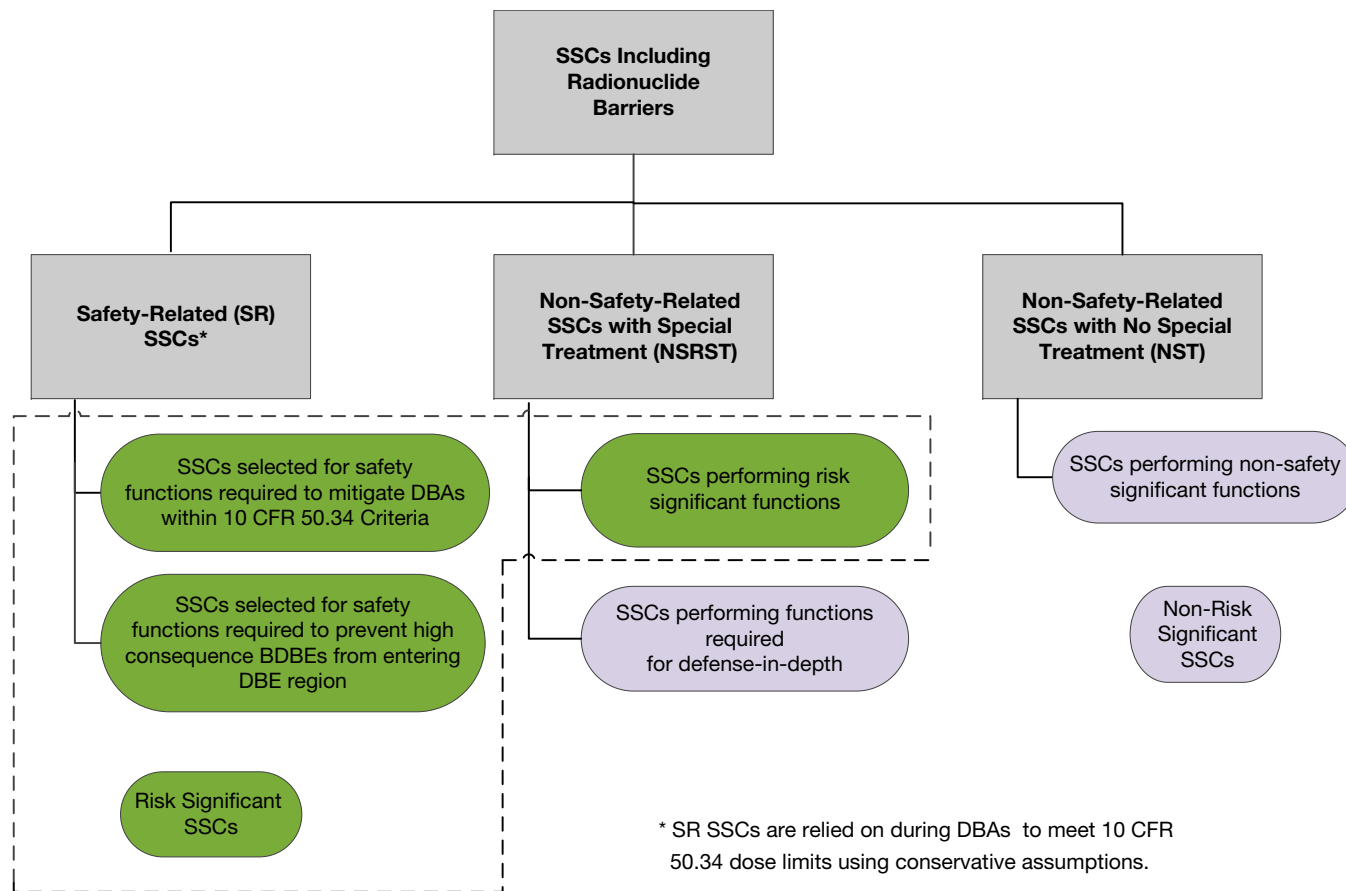


Fig. 4 LMP SSC safety categories and safety significant SSCs © 2019 Nuclear Energy Institute (NEI 18-04).

- The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed 2×10^{-6} /plant-year based on the mean estimates of frequencies and consequences to ensure that the NRC safety goal QHO for latent cancer fatality risk is met. An SSC makes a significant contribution to this cumulative risk metric if the individual risk of latent cancer fatalities associated with the LBEs with the SSC failed is greater than 2×10^{-8} /plant-year."

The use of mean values of frequency and consequence to establish the risk significance requires a full quantification of uncertainties. The uncertainty analysis is a key input to the identification of defense-in-depth protective strategies that are part of the process for establishing the adequacy of defense-in-depth. The defense-in-depth approach also addresses limitations of the PRA to achieve a balance consideration of deterministic principles and risk insights in demonstrating the safety case.

The risk significance criteria as well as guidelines for evaluating the role of the SSC functions in supporting the adequacy of defense-in-depth are used to classify the SSCs according to the safety significance of their functions as shown in Fig. 4.

An important element of the safety classification approach is that risk inputs and defense-in-depth evaluations are linked to the underlying event sequences in the PRA that are grouped into families represented in the LBEs.

The LMP methodology emphasizes that risk insights used for safety classification, and the evaluation of defense-in-depth adequacy, are fully integrated into the event sequence models in the PRA that addresses dependencies and interactions among the SSCs that cause initiating events and resulting event sequences as well as the characterization and analysis of uncertainties in the estimation of LBE frequencies and consequences.

This methodology has been subjected to several pilot applications that have confirmed its usefulness and applicability to a spectrum of advanced reactor designs including a high temperature gas cooled reactor, a fluoride salt high temperature reactor, a sodium cooled fast reactor, a molten salt reactor, and two microreactor designs involving the use of heat pipes. (USNRC, 2020).

Summary

In this chapter, the process of deriving risk insights from the results of a PRA has been described. The description has included examples of risk insights that were derived from early PRAs and a summary of how these insights have been and continue to be used to support risk-informed decision making for existing and advanced reactors. Risk metrics and methods of using these risk metrics to derive risk insights and support risk-informed decision making have been summarized. Limitations and pitfalls that must be appreciated to avoid misleading conclusions from the interpretation of risk importance measures such as F-V importance have been highlighted using examples.

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Nuclear Power Plant Decommissioning

Rod McCullum, Nuclear Energy Institute, Washington, DC, United States

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Introduction

Nuclear power plants are like no other industrial facilities in the world in that the entire facility life cycle, including decommissioning is planned for from the very beginning of the plant's existence. In the United States, which has the world's largest operating commercial nuclear fleet, the Nuclear Regulatory Commission (NRC) requires – before a plant even begins operation – that the owner must establish a trust fund or other financial mechanism to ensure that there will be sufficient money to pay for the ultimate decommissioning of the facility. These same regulations require that the plant owner report to NRC every 2 years on the status of decommissioning funding. NRC's regulations are described in detail within this chapter. While practices vary in other countries, and initially some countries lacked the strong regulatory governance of decommissioning funding that exists in the United States, the concept of life of plant financial assurance has now become common worldwide.

What is meant by “ultimate” decommissioning of a nuclear power plant is also very clearly defined. National regulators, such as the US NRC, set requirements for the safe removal of a nuclear power plant from service including very stringent specification of the extent to which residual radioactivity levels must be reduced before the property can be released and its license terminated. At license termination, the site is remediated to a very small amount of radioactive material, with radiation levels at least 14 times less than natural background dose rates. After a plant's license is terminated, the site is no longer governed by the national regulator; however, State and local authorities often impose additional requirements for further site restoration which can include reducing radiation levels to an even smaller fraction of natural background dose rates.

There are three chapters related to the topic of Decommissioning in Section 4 of the Encyclopedia. This chapter presents an overview of US and international decommissioning strategies, regulatory requirements, status, and practices. The next two chapters provide examples of two U.S. plants that have proceeded with two of the more commonly used strategies: SAFSTOR (McCullum, 2021a) and DECON (McCullum, 2021b). The examples provided in these chapters illustrate some additional aspects associated with the decommissioning process, such as public involvement and public policy that influences the ultimate land use of the site.

Decommissioning strategies

The US NRC allows three strategies to achieve decommissioning (USNRC, 2018). These are listed as follows:

- DECON – The plant owner begins dismantling the plant and removing materials containing radioactive contaminants shortly after the facility ceases operations
- SAFSTOR – The plant owner defers dismantling and maintains the facility in a safe condition for several years after the facility ceases operation. This allows radioactivity to decay and the trust fund to grow before most of the work begins
- ENTOMB – The plant owner permanently encases radioactive contaminants on site in a structurally sound containment materials and monitors until radioactive decay allows the site to meet regulatory requirements for the license to be terminated

NRC regulations allow up to 60 years to complete the decommissioning process, this is based on up to 50 years in SAFSTOR followed by 10 years of DECON. To date, most plants in the US have pursued a combination of SAFSTOR and DECON with an initial decision being made to move into SAFSTOR with some DECON activities being conducted followed by a subsequent decision to move fully into DECON after some period of time in SAFSTOR. Often this decision is made concurrently with the transfer of either the license or the license and ownership from the utility that operated the plant to a company that specializes in decommissioning (see “U.S. decommissioning marketplace evolution” in this chapter). Recently there has been a trend, commensurate with these transfers, toward accelerating the transition into DECON.

Under the current regulatory framework, no US plant has ever opted for the ENTOMB option. However, prior to the formation of NRC, the Hallam Nuclear Station in Nebraska was decommissioned under the auspices of the Atomic Energy Commission using a similar strategy. The reactor vessel and other components remain buried under a grassy knoll at that site today awaiting site release in 2070 after sufficient radioactive decay has occurred (ANS, 2019a). Globally, the only unit currently using ENTOMB is Chernobyl Unit 4 in the Ukraine where that option was determined to be necessary due to the special circumstances that have existed since the reactor accident at that site in 1986 (Sich, 2021).

US decommissioning status

As of the date of this publication, in the United States 10 commercial nuclear power reactors have completed the decommissioning process¹ while an additional 10 have permanently ceased operations and are either in SAFSTOR or DECON. Tables 1 and 2 provide a summary of the status of these plants.

US decommissioning regulations

US NRC regulations are seen as the gold standard by regulatory authorities around the world. Because the US has also completed¹ the most decommissioning projects – approximately half of the power reactors that have reached the point of license termination are in the United States – the US regulatory framework for decommissioning is also the most tested. US decommissioning regulations have been effective at assuring safety and environmental protection throughout the life of each of the 10 completed projects listed in Table 1 above. As mentioned above, most other countries either already have or are moving towards requirements for decommissioning practices that mimic those in the United States.

In the US, the steps for commencing the decommissioning process are defined in 10 CFR 50.82 of NRC's regulations. In accordance with 10 CFR 50.82(a)(1), the process formally begins when a reactor licensee notifies the NRC that it has determined that it will permanently cease operations by submitting a written certification to NRC, within 30 days of having made this determination, in accordance with 10 CFR 50.4(b)(8). The licensee's first immediate task is to remove all fuel from the reactor vessel and place it in the spent fuel pool. Once this task has been completed, the licensee is required to submit a second certification – of permanent removal of fuel – in accordance with 10 CFR 50.4(b)(9). After NRC has formally docketed (e.g., accepted) both of these certifications, the licensee is no longer legally authorized to operate the reactor or return any fuel to the reactor vessel. At this point the decision to shut down the reactor has become irreversible.

The certification of permanent cessation of operations also starts the 60 year time period during which decommissioning must be complete. This is specified in 10 CFR 50(a)(3). The licensee may use any combination of SAFSTOR, DECON, or ENTOMB to accomplish this. 10 CFR 50(a)(3) also provides that, in certain special circumstances, NRC may consider extending the 60 year period only when necessary to protect public health and safety. Shutdown reactors which are co-located with operating reactors are generally considered an acceptable circumstance for extending the 60 year period as the conduct of demolition activities in proximity to an operating reactor could have adverse safety consequences.

Table 1 US Plants that have completed decommissioning (July 2019) (USEIA, 2017; USNRC, 2019).

<i>Plant</i>	<i>Location</i>	<i>Period of Decommissioning</i>
Pathfinder	Minnehaha County SD	1968–2007
Fort St Vrain ^a	Platteville CO	1989–1997
Rancho Seco ^a	Herald CA	1989–2009
Saxton	Altoona PA	1972–2005
Shoreham	East Shoreham, NY	1989–1992
Yankee Rowe ^a	Rowe MA	1992–2007
Trojan ^a	Columbia County OR	1993–2005
Haddam Neck ^a	Haddam Neck CT	1996–2008
Maine Yankee ^a	Wiscasset ME	1996–2005
Big Rock Point ^a	Charlevoix MI	1997–2007

^aSpent fuel remains stored at an Independent Spent fuel Storage Installation (ISFSI) on these sites. All land at the site is ready for release, but first the fuel must be removed and the ISFSI decommissioned.

¹The decommissioning process is considered complete when all plant structures and systems have been decommissioned such that the NRC license can be terminated. Spent fuel may remain on the site after this point in a separately licensed Independent Spent Fuel Storage Installation (ISFSI). In cases where an ISFSI remains on site the land can not be released for other purposes. So, in these cases, while the decommissioning is considered complete from a technical and regulatory point of view, from a public perception point of view it is not considered complete due to the continued presence of radioactive material on the site.

Table 2 US Plants^a currently undergoing decommissioning (August 2019) (NEI, 2019a; USNRC, 2019; Energy Daily, 2019).

<i>Plant</i>	<i>Location</i>	<i>Ceased operations</i>	<i>Current status</i>
Humbolt Bay	Eureka CA	1976	Initially placed in SAFSTOR, moved to DECON in 2008, scheduled for completion 2020
Lacrosse	Genoa WI	1987	Initially placed in SAFSTOR, moved into DECON in 2016 upon license transfer in 2016, expected completion 2020
Zion 1&2	Zion IL	1998	Initially placed in SAFSTOR, moved into DECON upon license transfer in 2010, expected completion 2020
San Onofre 1,2, & 3	San Clemente CA	2013	DECON
Crystal River Unit 3	Crystal River FL	2013	SAFSTOR, ownership transfer pending. Plant will move to DECON upon transfer
Kewaunee	Kewaunee WI	2013	SAFSTOR
Vermont Yankee	Vernon VT		Originally placed in SAFSTOR, ownership transferred in 2019 at which point plant entered DECON
Ft Calhoun	Ft Calhoun NE	2016	SAFSTOR
Oyster Creek	Forked River NJ	2018	Ownership transferred in 2019 at which point plant entered DECON
Pilgrim	Plymouth MA	2019	SAFSTOR, Ownership transfer pending, plant will move to DECON upon transfer
Three Mile Island	Middletown PA	2019	Unit 1 placed into SAFSTOR upon shutdown. Unit 2 Ownership transferred in 2019

^aDoes not include shutdown reactors on sites where there are still currently one or more operating reactors. Decommissioning of these reactors will be done in coordination with the future shutdown and decommissioning of the remaining operating unit(s).

A key requirement that must be met before decommissioning can begin is the completion, and submission to NRC, of the licensee's Post Shutdown Activities Report (PSDAR). In accordance with 10 CFR 50.82(a)(4), this report must describe planned decommissioning activities, the schedule for completing them, an estimate of costs, and a discussion of the environmental impacts associated with decommissioning (USNRC, 2019). The PSDAR must be submitted prior to or within 2 years following permanent cessation of operations. Although NRC approval of the PSDAR is not required, NRC does solicit public comments on the report and may ask questions of the licensee on its content (USNRC, 2019).

10 CFR 50.82(a)(5) specifies that the licensee may conduct no major decommissioning activities (as defined in 10 CFR 50.2) until both NRC has received the PSDAR and both the certification of permanent cessation of operations and permanent reactor defuel. 10 CFR 50.82(a)(6) and (7) place additional restrictions on the conduct of decommissioning and sets requirements for the reporting of any changes to the PSDAR (USNRC, 2019).

Throughout the decommissioning process, and until the license is terminated, safety is assured under the regulations of 10 CFR Part 20 (for radiation protection), 10 CFR Parts 50 and 73 (for everything else pertaining to the plant), and 10 CFR Part 72 (for spent fuel stored onsite in an Independent Spent Fuel Storage Installation (ISFSI)). The transportation of spent fuel away from the plant site is governed by 10 CFR Part 71 (USNRC, 2019).

NRC has recognized that there are some regulations in 10 CFR Parts 50 and 73, particularly in the areas of emergency preparedness and security, which should not be applicable at a decommissioning site because the potential for a reactor accident no longer exists. To date, relief from these unnecessary requirements has been obtained by the licensees applying for and NRC approving exemption and license amendment requests. However, this review and approval process is expensive. At the time of this publication, NRC had a rulemaking underway to codify an orderly transition from the full scope of operating reactor requirements to a more limited set of regulations appropriate for decommissioning, without the need for the licensee to submit these requests (USNRC, 2015).

NRC also carefully regulates both the formation of the decommissioning trust fund – while the plant is operating – and its use once the plant is shut down. Decommissioning financial assurance requirements which must be met over the entire life of the plant are spelled out in 10 CFR 50.75 (US, 2019). This regulation includes requirement for determining the sufficiency of the fund to provide for decommissioning at end of plant life and for biennial reporting to NRC on the result of this determination. Use of the trust fund upon shut down is governed by 10 CFR 50.82(a)(8) (US, 2019). This regulation allows initially for up to 3% of the fund to be used for decommissioning planning and 20% of the fund to be used once the licensee has submitted the PSDAR. Use of any funds above these amounts is not allowed until after the licensee has submitted to NRC a detailed site-specific cost estimate for the project (US, 2019).

Once decommissioning is complete, licensees must submit, to NRC, an application to terminate the license which must include a license termination plan at least 2 years prior to the date on which approval of termination is being sought. This submittal is defined in 10 CFR 50.82(a)(9). The radiological criteria which must be met in order for the license to be terminated are spelled out in 10 CFR 20 Subpart E (US, 2019).

US decommissioning practices

The process of decommissioning a nuclear power plant actually begins long before a decision has been made to shut the plant down. Three important activities take place while the plant is still operating;

1. The accumulation of funding sufficient to complete the decommissioning project. The plant owner sets funds aside in accordance with the requirements described above.
2. The radiological and environmental characterization of the plant. Records of plant conditions are collected throughout the operating life of the plant. How these records are organized and maintained will determine how well defined the task of decommissioning can be. Ideally, at the time of shutdown, a clear picture of the extent to which radioactive and hazardous materials are present and where they exist will be in place to guide the decommissioning effort.
3. The development of a decommissioning plan and strategy. At the time a plant shuts down, there will be many options for the path ahead. Based on what is known about available funding and plant conditions at the time of shutdown (the results of activities 1 and two above) the plant owner will then lay out the path from certification of permanent reactor defuel to termination of license. This path can involve both periods where the plant is in SAFSTOR and plans where active DECON is underway. Optimizing activities along this path is essentially a complex project management challenge. The success of this project will be determined by the skill of the project planners.

Once the radiological and environmental conditions of the plant are sufficiently characterized, the decommissioning project can be planned in earnest. As spent fuel is, by far, the most significant source of radioactivity on site, priority is placed on getting all of the fuel out of the plant and into a co-located dry storage facility – known as an Independent Spent Fuel Storage Installation (ISFSI) – before beginning active DECON of the plant. Even if the plant is to be placed in SAFSTOR for a period of time, getting all of the spent fuel into an ISFSI at an early stage will significantly reduce the risk profile of the plant and allow a number of plant systems to be taken out of service, hence reducing costs.

After spent fuel is removed from the plant's spent fuel pool and into an ISFSI, the decommissioning focus will shift to the reactor vessel internals and primary systems – steam generators (PWR), turbine/condenser (BWR), reactor coolant pumps, pressurizer (PWR), and connecting piping (ANS, 2019b). There are many options for decontaminating, segmenting, and preparing these components to be shipped offsite to existing disposal facilities which are designed and approved to accept these materials. Skilled project managers must consider both technical and business considerations in determining which options to pursue, in what order to pursue them, and whether work can be performed in parallel or must be performed in series. The results of the radiological and environmental characterization of the plant will inform this decision-making. Use of plant space, radwaste volume, worker protection, and cost must all be considered.

Reactor vessel internals have the highest radioactivity levels of anything in the plant except for the spent fuel (ANS, 2019b). Once characterized, decisions must be made about how to segment, package, and, if possible, ship these components offsite. Segments will fall into one of 4 classifications depending on level of radioactivity – Class A, B, or C low level waste or Greater than Class C waste (GTCC). Multiple options exist for packaging and disposing of Class A through C low level radioactive waste, however, there is currently not a disposal path in the US for GTCC. GTCC waste is placed into dry cask storage and stored in the ISFSI with the spent fuel. Once the GTCC has been separated out, Class A through C wastes can be packaged for disposal. In some cases, this is accomplished by placing these components back into the reactor vessel (which is typically categorized as Class A Waste) and shipping the entire vessel offsite for disposal (ANS, 2019b).

Before the reactor vessel and its internals can be segmented, accessible space must be created in the plant for the conduct of this work – including room for shielding to protect workers. This means that the strategy for addressing the reactor vessel will be affected by the strategy for decontaminating and dismantling the primary system components surrounding it. Here, as with the vessel itself, characterization is the key to success. Based on what is known, a chemical decontamination strategy can be deployed to facilitate the optimal dismantling of primary systems with a minimum volume of low-level waste generation. An effective decontamination strategy will simplify access for workers to dismantle the systems, minimize the spread of contamination, and reduce the amount of low-level waste that must be disposed of (ANS, 2019b).

While the reactor vessel internals and primary systems represent the most significant aspects of radiological decommissioning, the status and condition of the entire plant is carefully considered and factored into planning throughout the decommissioning process. For example, some support systems – such as water cleanup, ventilation, and electric power – will need to remain operational. In some cases, it will be prudent to maintain or modify existing plant systems and in other cases temporary systems will be brought in to take the place of permanent equipment that is being taken out of service.

Upcoming decommissioning work in the US

During the decade between 2020 and 2030 the US is expected to see a significant increase in the amount of active decommissioning work being conducted as a number of recently shutdown plants (see Table 2) transition from SAFSTOR into DECON and a number of currently operating plants (see Table 3) cease operations and enter decommissioning. The measures that the US industry is taking to address this increased workload are next discussed.

U.S. decommissioning marketplace evolution

After a number of nuclear power plant decommissioning projects had been completed in the US, it became apparent that the organizational skill set best suited to decommissioning a nuclear power plant was, in fact, very different from the organizational skill set

Table 3 US Plants projected^a to enter decommissioning before 2025 (NEI, 2019b).

<i>Plant</i>	<i>Location</i>	<i>Expected date to cease operations</i>
Duane Arnold	Palo IA	2020
Indian Point 1, 2, & 3	Buchanan NY	2020–2021
Beaver Valley 1&2	Shippingport PA	2021
Palisades	Covert MI	2022
Diablo Canyon 1&2	San Luis Obispo County CA	2024–2025

^aThis projection is based on shutdowns announced as of August 7, 2019. The actual list of plants that will shut-down may vary depending on electricity market conditions and future policy decisions at the State or Federal level that could affect market conditions in ways that either incentivize or dis-incentivize the continued operation of nuclear power plants.

best suited for operating one. This has led to a trend away from operating utilities “self-performing” plant decommissioning and to the rise of highly experienced and skilled decommissioning companies who specialize in managing these projects under one of three different business models. These are described as follows:

- **Contract Oversight** – responsibility for the conduct of decommissioning is placed in the hands of an expert company under contract to the utility. The utility oversees the contract and retains ownership of the facility (including fuel) as well as possession of the NRC license. The expert company’s success is determined by their ability to satisfy the terms of the contract.
- **Shared Stewardship** – The NRC license for the facility and a fixed amount of capital is transferred to an expert decommissioning company that conducts the decommissioning project. Upon termination of the NRC license, the site and the ISFSI containing the used fuel are transferred back to the utility. The expert company’s success is determined by their ability to achieve license termination on time and under budget.
- **Asset Acquisition** – An expert decommissioning company purchases the facility (including the fuel) and the license along with the decommissioning trust fund is transferred to the new owner. The expert company’s success is determined by their ability to complete the decommissioning with the resources of the existing trust fund, either move the fuel offsite or recover the costs of maintaining the ISFSI from the federal government, and recover the value of the land once the project is complete and the fuel is moved off site.

The trend away from utilities self-performing decommissioning towards business models that place the projects increasingly in the hands of decommissioning expert company is illustrated in **Table 4** which categorizes active and completed projects by business model.

Table 4 Business approach for active or completed major US decommissioning projects^a (USNIC, 2019; ANS, 2019c; Wyman, 2017).

<i>Business model</i>	<i>Plant</i>	<i>Utility</i>	<i>Expert company</i>	<i>Date plant ceased operations</i>	<i>Date decom. complete</i>
Self-perform	Rancho Seco	Sacramento Municipal Utility Dist.	NA	1989	2009
	Shoreham	Long Island Lighting Company	NA	1989	1992
	Yankee Rowe	Yankee Atomic	NA	1992	2007
	Trojan	Portland General Electric	NA	1993	2005
	Haddam Neck	Yankee Atomic	NA	1996	2008
	Maine Yankee	Yankee Atomic	NA	1996	2005
	Big Rock Point	Consumers Power	NA	1997	2007
Contract Oversight	San Onofre	Southern California Edison	Energy Solutions (plant) & Holtec (ISFSI)	2013	Ongoing
	Ft Calhoun	Omaha Public Power District	Energy Solutions	2016	Ongoing
Shared Stewardship	Lacrosse	Dairyland Power	Energy Solutions	1987 ^b	Ongoing
	Zion	Exelon	Energy Solutions	1998 ^c	Ongoing
	Crystal River	Duke	Accelerated Decommissioning Partners	2013	Ongoing
Asset Acquisition	Vermont Yankee	Entergy	Accelerated Decommissioning Partners	2014	Ongoing
	Oyster Creek	Exelon	Holtec	2018	Ongoing
	Pilgrim	Entergy	Holtec	2019	Ongoing
	Three Mile Island Unit 2	First Energy	Energy Solutions	1979	Ongoing

^aDoes not include plants currently in SAFSTOR (Three Mile Island 1, Kewaunee) or smaller scale reactors (Pathfinder, Ft St Vrain, and Saxton).

^bIn SAFSTOR with some DECON until 2016 when NRC license was transferred to Energy Solutions.

^cIn SAFSTOR until 2011 when NRC license was transferred to Energy Solutions.

International decommissioning

In general, the decommissioning marketplace is not as well defined outside the US. There are considerable uncertainties as to the timing and nature of decommissioning processes going forward. Also, the degree to which project funding is established through a decommissioning trust fund which was accumulated during plant operation varies (although in many cases this because the financing of the project is backed by the government). However, in one respect, the decommissioning marketplace is often better defined outside the US – in many countries centralized, off-site, interim storage capabilities exist to facilitate the removal of spent fuel from the site so that when decommissioning of the plant is complete, the land may also be released.

To date, approximately half of the completed decommissioning projects in the world have been accomplished in the United States (those listed in Table 1). However, worldwide, approximately 140 commercial power reactors have permanently shut down and either are or will be undergoing decommissioning (IAEA, 2017). There are three primary factors affecting the scope of the global decommissioning marketplace.

1. National decisions to phase out nuclear power, such as those taken in Germany and Taiwan (Reuters, 2019; World Nuclear News, 2019).
2. Aging of the global nuclear fleet (over 50% of the world's 448 operating nuclear power reactors are more than 30 years old) (IAEA, 2017).
3. The extent to which Japan may, or may not, be successful in regaining sufficient public confidence to restart its reactors in the wake of the Fukushima accident (Mizokami and Rempe, 2021) All of Japan's 57 operating reactors were shutdown following the accident. At the date of this publication, only 9 of these have gained approval to restart. Twenty-four reactors have entered decommissioning or are set to be decommissioned, and the fate of the remaining 24 has yet to be decided (Japan Electric Power Information Center, Inc., 2019).

As decommissioning becomes more common around the world, a number of challenges will be encountered. Legal and regulatory frameworks are incomplete in some countries. Funding uncertainties are common and waste disposal paths are often unknown. Finally, the global workforce will be stretched to provide skilled and experienced labor.

Summary

U.S. nuclear plant owners have safely completed the decommissioning of 10 commercial nuclear power plants. This has been accomplished through careful advance planning – utilizing the resources of trust funds established during plant operations and guided by a well-established regulatory framework. The pace of decommissioning in the U.S. is accelerating with 15 reactors at 11 sites currently undergoing decommissioning and 9 reactors at 5 sites poised to enter decommissioning between 2020 and 2025.

This experience has led to a refinement in techniques for decontaminating and dismantling nuclear plants as well as for the management of spent fuel and low-level radioactive waste. The deployment of innovative new business models is building on the lessons learned from past experience to enable the U.S. nuclear industry to improve efficiency as demand for decommissioning services increases. Although international experience varies, this experience is also informing global efforts to meet an expected increase in decommissioning activity worldwide.

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SAFSTOR Decommissioning Example (Kewaunee)

Rod McCullum, Nuclear Energy Institute, Washington, DC, United States

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Introduction

On May 7, 2013, after 39 years of operation, the Kewaunee nuclear power station ([Fig. 1](#)), which includes a single unit Westinghouse pressurized water reactor, permanently ceased operations. On May 14, 2013 Dominion Energy (Dominion), the plant owner, certified that all fuel had been permanently removed from the reactor vessel and placed in the spent fuel pool ([Dominion Energy Kewaunee, Inc., 2013a](#)). At that time, the plant owner/operator, Dominion Energy Kewaunee Inc. announced that it intended to place the plant in SAFSTOR. [McCullum \(2021a\)](#) describes the SAFSTOR option. Kewaunee is one example of plants at which this option is being pursued.

Decommissioning progress

In February of 2013, Dominion completed a Post Shutdown Activities Report (PSDAR) ([Dominion Energy Kewaunee, Inc., 2013b](#)). As described in [McCullum \(2021a\)](#), this document laid out a schedule for SAFSTOR and the eventual decommissioning of the plant and included a cost estimate. The schedule called for all SAFSTOR preparations to be completed by 2020 and for the plant to then enter a period of dormancy with all fuel in dry storage until 2050. Dominion estimated that spent fuel would be removed from the site in 2050 (an uncertain date given the US Department of Energy's long history of delay in fulfilling its contractual obligations to remove used fuel from commercial reactor sites) and then the site would remain dormant until 2067.

According to Dominion's plan, workers and equipment will return to the site in 2067 to begin dismantlement of plant structures, removal of plant systems, and site decontamination. This process is expected to be completed by 2072. Then, in 2073, site restoration work will restore the site to a greenfield condition. During this time the natural process of radioactive decay and the interest earned on the decommissioning trust fund (not to mention potential technological advances that occur over that time) will make



Fig. 1 Kewaunee Nuclear Station in 2013 (located in Carlton, Wisconsin). Wald ML (2013) As price of nuclear energy drops, a Wisconsin plant is shut. *New York Times*, 7 May 2013.

their job easier. There will be less low-level radioactive waste that will need to be shipped off site and, based on conservative financial projections, the trust fund will grow to be more than adequate to pay the \$920 million cost that Dominion estimates for this entire process. In 2013 the US Nuclear Regulatory Commission reviewed Dominion's cash flow analysis and found that the existing trust fund would be adequate to cover all decommissioning, spent fuel management, and site restoration costs.

Of course, a lot can change over 60 years. Dominion's estimated schedule takes a purposefully long view in order to demonstrate that it is, in fact, capable of meeting NRC's requirements for exiting SAFSTOR and completing the decommissioning of the plant—up to license termination—within the allowable 60 years. At any point in the future, Dominion may decide to accelerate the project if it proves prudent to do so. In fact, the plant is already moving into dormancy ahead of schedule. All fuel was moved into dry storage in July of 2017, 3 years and 5 months earlier than originally planned.

In 2019, the Kewaunee site still looks pretty much the same on the outside—with most major structures still standing. But inside things could not be more different. A skeleton crew carefully watches over physical, environmental, and radiological conditions in the dormant facility with only a minimal amount of equipment operating to maintain the condition of the plant in anticipation of the arrival of its destruction decades in the future. Conditions in the plant are adjusted to prevent any undue degradation of plant structures that would complicate the task of future destruction workers when they arrive and to assure that no hazardous materials are released beyond the facility fences. In some cases, where it is economical to do so, existing plant equipment has been modified to provide ventilation flow, electricity, potable water, and fire protection in those areas where it is still needed. In other cases, where continued reliance on existing systems would be inefficient, these systems have been taken out of service and replaced with smaller more economical new systems to provide necessary utility services. The plants control room (Fig. 2) is largely a relic, albeit a neat and tidy one, with all but a few panels—those still linked to the existing systems that have been retained—de-energized and long inactive.

Inside Kewaunee's dormant rooms, there is a notable scarcity of radiation zone postings that would typically be found at many locations in an operating plant. The radiation protection offices, where teams of health physicists used to be busy around the clock assuring that workers were protected and that no contamination was taken offsite, are dark and empty. Their sophisticated monitoring and survey equipment is long gone. Before entering dormancy, plant workers removed readily accessible radioactive contamination. These materials were shipped offsite to low level waste disposal facilities in a manner similar to what operating plants do on a routine basis. The bulk of the radioactive materials remaining at the site are entrained inside the still-standing structures and systems along with the spent fuel on the ISFSI site. It is a testament to the meticulous efforts of those health physicists during plant operation that, upon plant shutdown, this was not a difficult task. For example, after the spent fuel pool was drained, it was simply hosed down for decontamination with a negligible amount of waste generation. In 2019, one can stand directly over the empty pool (Fig. 3), look down into the now dry expanse through which passed every single spent fuel assembly—containing the radioactive byproducts of nearly four decades of splitting uranium atoms—and receive no radiation exposure.



Fig. 2 Kewaunee Nuclear Station control room in 2019. Only few panels are functioning (*lights on*). McCullum R (2019) Notes and photos taken from tour of Kewaunee site, 25 September 2019.

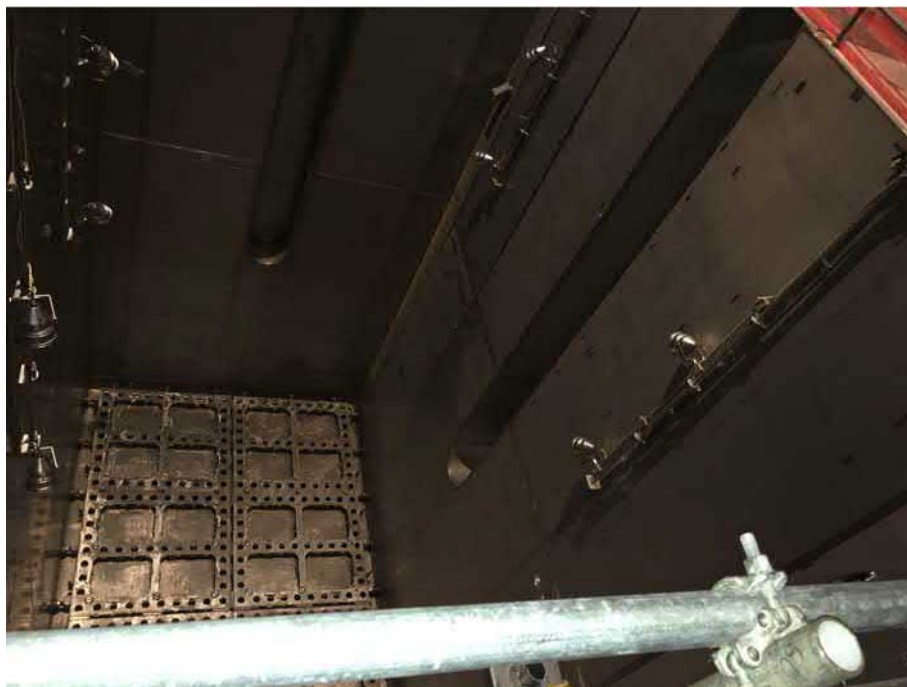


Fig. 3 Kewaunee Nuclear Station spent fuel pool in 2019; without fuel and water. McCullum R (2019) Notes and photos taken from tour of Kewaunee site, 25 September 2019.



Fig. 4 Kewaunee Nuclear Station Independent Spent Fuel Storage Installation (ISFSI) in 2019. McCullum R (2019) Notes and photos taken from tour of Kewaunee site, 25 September 2019.

The fission products contained in the spent fuel assemblies remain stored in dry casks (Fig. 4) on a concrete pad just north of the plant awaiting the anticipated 2050 arrival of the US Department of Energy (DOE) to transport them off to final disposal. The concrete pad and the dry cask storage systems that sit on it are fenced off from the rest of the plant, inside a unique security perimeter, in an Independent Spent Fuel Storage Installation (ISFSI) that is regulated under a separate NRC license in accordance with 10 CFR Part 72. The separation of the ISFSI from the rest of the plant prevents the continued presence of spent fuel on the site from

interfering with the decommissioning of the plant. These storage systems also provide a high degree of safety protection for the spent fuel without reliance on plant systems. They have been designed and tested to withstand severe events such as earthquakes, floods, tornados, airplane crashes, and explosive fires. Even if the DOE does not remove the spent fuel in 2050—which could happen if political controversy continues to delay the US spent fuel disposal program—the active decommissioning of the plant can still proceed on schedule beginning in 2067. It is, in fact, quite common for plant decommissioning to be completed¹ while the separate ISFSI remains in place. This is illustrated by the Maine Yankee example detailed in [McCullum \(2021b\)](#).

Summary

Dominion Energy permanently ceased operations of its Kewaunee Nuclear Power Station in 2013. The plant was immediately placed in SAFSTOR. Although the fundamental concept of SAFSTOR is to wait on decommissioning until some future time after levels of radioactivity have decayed and decommissioning and trust fund resources have grown through accumulated interest, several activities have been conducted to assure that industrial and radiological safety is maintained in all areas of the plant throughout the SAFSTOR period. Accessible areas of the plant have been decontaminated, plant systems have been modified, and all spent fuel has been placed in an onsite ISFSI. Dominion anticipates completing the NRC required decommissioning process in 2072 and begin restoring the site to a greenfield condition in 2073. The total cost of this project at completion is expected to be \$920 million. ([Dominion Energy Kewaunee, Inc., 2013b](#)).

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¹The decommissioning process is considered complete when all plant structures and systems have been decommissioned such that the NRC license can be terminated. Spent fuel may remain on the site after this point in a separately licensed Independent Spent Fuel Storage Installation (ISFSI). In cases where an ISFSI remains on site, the land cannot be released for other purposes. So, in these cases, while the decommissioning is considered complete from a technical and regulatory point of view, from a public perception point of view it is not considered complete due to the continued presence of radioactive material on the sites.

DECON Decommissioning Example (Maine Yankee)

Rod McCullum, Organization Nuclear Energy Institute, Washington, DC, United States

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Introduction

On August 6, 1997, amidst an extended shutdown for repairs, the Maine Yankee Atomic Power Company Board (the Board) voted to decommission the Maine Yankee Nuclear Power Plant or Maine Yankee ([Fig. 1](#)), which included a single unit Combustion Engineering pressurized water reactor, after 25 years of operation. The next day, the Board submitted to the US Nuclear Regulatory Commission (NRC), a certification of permanently ceasing power generation operations and a certification of permanent removal of fuel from the reactor. In making this decision, the Board determined that prompt dismantlement was the most economically advantageous approach for electricity ratepayers ([EPRI, 2005](#)). Therefore, Maine Yankee went immediately into DECON upon termination of operations.

[McCullum \(2021\)](#) outlines the steps in the DECON process until the point at which the technical and regulatory definition of complete is met, even though spent fuel may remain on the site in a separately licensed Independent Spent Fuel Storage Installation (ISFSI). Maine Yankee is one example of this situation.

Decommissioning progress

Decommissioning planning activities commenced in 1996, even as the Board was still contemplating the decision to cease operations. These activities included: drafting of the Post Shutdown Activities Report (PSDAR) ([Maine Yankee Atomic Power Co, 1997](#)); the advance preparation of a number of exemption requests that would need to be submitted to the NRC to transition from operating plant regulatory requirements to the more limited set of regulations that would remain applicable to a shutdown plant site;



Fig. 1 Maine Yankee Nuclear Plant in 1998 (located in Wiscasset, Maine) ([Bangor Daily News, 2013](#)).

estimating plant decommissioning costs for all available options; evaluating decommissioning approaches; and planning necessary interactions with the community surrounding the plant.

Between August of 1997 and July of 1998, Maine Yankee entered into a transition period in which decommissioning plans were finalized and a Decommissioning Operations Contractor (DOC) was selected to perform the work. In August of 1998, Stone and Webster Engineering Corporation (SWEC) was awarded a turnkey, fixed-priced DOC contract. However, this arrangement proved problematic to implement. By July of 2000, the DOC contract had been terminated, and the Maine Yankee Atomic Power Company took over management of the work to self-perform the decommissioning (EPRI, 2005).

One of the most important decisions that needed to be made early in the decommissioning process was the selection of a spent fuel management strategy. When Maine Yankee ceased operations in 1998, the nuclear industry was just beginning to come to terms with the US Department of Energy's (DOE) default on its statutory and contractual obligation to begin removal of spent fuel from reactor sites in January of 1998. With the Yucca Mountain Repository Project significantly delayed, there was considerable uncertainty as to when DOE would arrive at the Maine Yankee site to begin removing fuel. Maine Yankee had not yet implemented dry cask storage in response to spent fuel pool capacity limitations as had a number of other operating plants because available space for future reactor offloads still remained at this point.

The Board initially chose to adopt a spent fuel island approach in which spent fuel pool related systems would be isolated from the rest of the plant and operated in a self-contained manner until DOE would arrive to remove the spent fuel. Obviously, this would limit the extent to which the plant could be dismantled. A business case conducted early in the decommissioning process indicated that, assuming DOE did not remove the spent fuel prior to 2019, it would be economically advantageous to deploy dry cask storage. Between 2000 and 2002, an Independent Spent Fuel Storage Installation (ISFSI) was constructed and the transfer of all spent fuel to the ISFSI pad was completed by February 2004 (EPRI, 2005).

In parallel with the spent fuel decision-making, three early projects were undertaken to prepare the plant for dismantling – asbestos abatement, radioactive hot spot reduction, and reactor coolant system decontamination. As was the case with most reactors built in the 1970s, asbestos was widely used at Maine Yankee for insulation, fire protection, and in various paints and tiles. An estimated 28,500 cubic feet of asbestos would be removed from the plant (EPRI, 2005). One third of this material was radioactively contaminated and required disposition at a licensed low-level radioactive waste disposal site. The asbestos remediation project began in March of 1998 and was completed in December of 1998. Radioactive hot spot reduction¹ and reactor coolant system decontamination activities were conducted to reduce the radioactive source term in the plant as a worker protection measure. Radiation surveys identified hot spots so that affected equipment could be carefully cut out of affected systems. This project is estimated to have reduced total project radiation exposure by approximately 150 person-rem (EPRI, 2005). A similar dose reduction was achieved via a through chemical decontamination of the reactor coolant system. All source term reduction activities were completed by May of 1999 (EPRI, 2005).

By December 1998, Maine Yankee had reached a “cold and dark” state where liquid containing systems had been drained and power to electrical components had been removed (EPRI, 2005). From this point, major component dismantlement could begin. Throughout 1999, major components such as reactor coolant pumps were removed and shipped off site for disposal in licensed



Fig. 2 Maine Yankee Reactor Vessel ready for shipment to Barnwell LLW site in 2003 (EPRI, 2005).

¹Radioactive hot spots are areas where, in accordance with 10 CFR Part 20, special worker protection measures are required.

low-level radioactive waste facilities. The most significant work internal to the plant was the segmentation of reactor vessel internals (the most highly radioactive components other than the spent fuel). This activity, which commenced in November 2000, was performed in the flooded refueling cavity using abrasive water jet and mechanical cutting techniques (EPRI, 2005). The segmented internals were separated into three categories of waste – low level waste that would be shipped offsite inside the reactor vessel (Fig. 2), low level waste that would be shipped offsite in separate casks, and Greater than Class C (GTCC) waste – a category designated by NRC as between low and high level requiring geologic disposal. About 10% of the materials resulting from the complete dismantling of the nuclear plant were classified as GTCC (See McCullum, 2021). These materials were placed in the same type of dry casks as the spent fuel and placed on the ISFSI pad along with the fuel in April of 2003. The reactor vessel was shipped offsite in August of 2003 (Fig. 2).

With radioactive components removed, focus shifted to demolition of the concrete structures on site. An approach to demolition using explosives was established, tested and used to bring down the turbine building, the 330-ton polar crane inside the containment building, and finally, in September of 2004, the containment building itself (Fig. 3).

Throughout the project, stakeholder input was a significant factor in all decision-making. A Community Advisory Panel was formed in 1997 and functioned throughout the project as a conduit for public involvement in the decommissioning process. Having an informed public was vital to successfully navigating both the State permitting and federal regulatory processes. A total of three federal agencies (NRC, Environmental Protection Agency, and Federal Energy Regulatory Commission) and seven state agencies had authority over various aspects of the process. In a number of these areas, public interest groups intervened in regulatory matters.

Stakeholder input was probably most valuable in reaching decisions about the final end state to which the Maine Yankee site would be restored. By 2019, aside from the ISFSI and a remaining active switchyard, a visitor to the site would find it difficult to believe that a commercial nuclear reactor had stood on the green field along the banks of the Back River (Fig. 4). Maine Yankee's NRC license was terminated in October 2005, all State agency commitments were also met. The site, except the ISFSI, is now ready to be released for other purposes. The total cost of the project to that point was approximately \$500 million (EPRI, 2005).

Summary

The Maine Yankee Atomic Power Company permanently ceased operations of its Maine Yankee Power Station in 1997. The plant was immediately entered DECON. As one of the earlier commercial reactor decommissioning projects in the U.S., Maine Yankee's experience would become the source of many lessons learned that would be applied over the next two decades at numerous other commercial reactors that would enter decommissioning over the next two decades. Although the project would have to make a number of course corrections in its early stages – refining the overall management and spent fuel strategies – it ultimately became a success story. DECON was completed, in accordance with NRC regulations in 2005 at a total cost of approximately \$500 million (EPRI, 2005).



Fig. 3 Maine Yankee Containment Building ready for final demolition in 2004 (EPRI, 2005).



Fig. 4 Maine Yankee Site in 2015 with all plant structures removed, ISFSI remaining ([Planet Forward, 2019](#)).

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Introduction to Fuel Cycle Front End: From Mining Through Utilization

Douglas C. Crawford, Idaho National Laboratory, Idaho Falls, ID, United States

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Glossary

Fuel cycle front end The term given to the range of nuclear fuel cycle activities necessary to prepare nuclear fuel for utilization in nuclear reactors.

Fuel cycle back end The term given to the range of nuclear fuel cycle activities employed to handle and treat used nuclear fuel upon discharge utilization in nuclear reactors.

Nuclear fuel A mixture of elements, including fissile isotopes, used to establish a sustainable fission chain-reaction and release energy in a nuclear reactor. The form can be solid in any of a number of chemical compounds or metal alloys, or liquid with fuel constituents in solution such as a molten salt or an aqueous solution. Still other forms have been proposed and considered.

Nuclear fuel cycle (or simply “fuel cycle” in this context) The lifecycle of nuclear fuel from extraction of a natural resource, such as uranium or thorium, through mineral processing concentration, sale on the market, isotopic enrichment, conversion to desired chemical compound, fabrication into a usable form, loading into the reactor core to generate energy, recovery and, optionally, recycle of re-usable constituents (if deployed), treatment of waste constituents for disposal, and disposal. In a narrower context, engineers might also use the term “fuel cycle” when referring to the planning and optimizing of fuel loading and management in the core of an operating reactor.

Introduction

Nuclear fuel is a mixture of elements and isotopes used to establish a fission chain reaction and release fission energy in a nuclear reactor. The supply chain for light water reactor (LWR) fuel is mature and configured to provide fuel for nuclear reactors at the lowest possible cost. Some advanced reactor designs, with prospect for improved safety and sustainability, reduced waste and improved economics, call for different fuel cycle strategies and for different fuel forms and designs. Whether those reactor systems are successfully deployed in energy markets will depend in part on whether and how quickly the fuel supply chains for those systems mature to sufficient economic efficiency. Deployment requirements for research and test reactors are different; however, motivations for new fuel designs for research and test reactors include reduction in operating costs, improved operating reliability, and reduction of ^{235}U content.

Purpose of this chapter

This chapter introduces the Fuel Cycle Front End: Mining through Utilization Section of this *Encyclopedia*, providing context for the chapters that follow. It presents an overview of the nuclear fuel cycle and describes design considerations important to fuel technology.

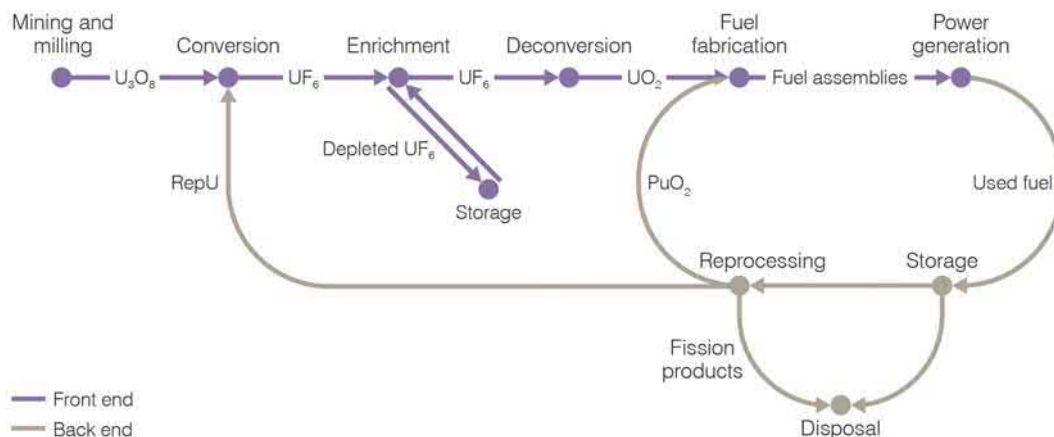


Fig. 1 Illustration of the light water reactor nuclear fuel cycle. All the activities prior to power generation (fuel utilization) are considered the front end of the fuel cycle, while those after power generation, including activities that recycle fissile material back into the front end, are considered the back of the fuel cycle. Courtesy World Nuclear Association.

Overview of the nuclear fuel cycle

The industry that provides nuclear fuel is considerably more complex than that for fossil fuel. Unique aspects of the nuclear fuel industry include the need to convert uranium and uranium oxides to uranium fluoride needed for enriching the natural uranium with the fissile isotope ^{235}U and then converting the enriched uranium fluoride gas to a solid metallic uranium or a uranium compound, the removal of hafnium (a neutron poison with chemical characteristics similar to zirconium) from zirconium (the metal used in LWR fuel cladding), and the rigor applied to quality controls.

The fuel cycle for LWRs is considerably more developed than those for other reactor types, due to the size of the LWR fuel market. Variations from the LWR fuel cycle for application to other reactor types are also indicated. The main elements of the nuclear fuel cycle are depicted in Fig. 1 and described below.

Mining and milling

Uranium is mined in a manner similar to other mineral resources using hard-rock techniques, such as underground mining, surface mining, and open pit mining. See Hore-Lacy (2021) for an overview of uranium mining practices. In-situ leaching or in-situ recovery is also used where economically warranted. Uranium ore is crushed and ground to reduce particle size, increasing the specific surface area of the rock to improve the kinetics of dissolution into acidic or alkaline solutions for concentration of the uranium into “yellowcake,” or U_3O_8 . Uranium milling produces a considerable volume of waste products, or tailings, per unit of yellowcake. Remediation of tailings, not always addressed in the past but now included in modern milling operations, ensures that abandoned tailings piles don’t become sources of wind-blown radioactive dust particles or ground-water contamination.

Although Thorium is considerably more plentiful than uranium, uses of thorium are not currently significant enough to motivate commercial-scale production of thorium. Development and deployment of nuclear power using a thorium-based fuel cycle could change that.

Conversion

Yellowcake is converted to UF_6 , primarily because UF_6 can be shipped at ambient temperatures as a solid but can be heated to relatively low temperature (to or above 57°C) to form UF_6 gas, which is suitable for enrichment (Hogan, 2021). Conversion to UF_6 can be accomplished using a wet process, beginning with dissolution in nitric acid followed by solvent extraction to purify the uranium oxide and series of steps that fluorinate the oxide. Alternatively, UF_6 can be formed from U_3O_8 in a dry process using a series of reactions with hydrogen, HF, and fluorine (Benedict et al., 1981, pp. 272–274).

Enrichment

Natural uranium is comprised of, by mass, 0.72% ^{235}U , 99.27% ^{238}U , and trace amounts of ^{234}U . With the exception of heavy-water-moderated reactors (HWRs), such as the CANDU reactor (Nichita, 2021), and graphite-moderated reactors, such as the Magnox reactors in the United Kingdom, most reactor types require uranium enriched in the ^{235}U isotope to 3% or greater to achieve criticality and enable reactor operation for a practical length of time. Although the atomic masses of ^{235}U and ^{238}U are very similar, the 1.3% mass difference is sufficient to allow segregation of uranium by mass to produce a product that is enriched in ^{235}U and a tailings by-product that is depleted in ^{235}U (see Hogan, 2021).

Processes used to enrich uranium include centrifugal separation and gaseous diffusion through a membrane. Each process requires a large number of stages in succession, because the mass difference between ^{235}U and ^{238}U is small. Laser enrichment processes, which operate on uranium in atomic or molecular form, continue in development but have not yet been commercially

deployed. The processes offer prospect for much greater separation efficiency with associated improvements in capital cost, tailings assay, and required energy input.

The energy invested in uranium enrichment is measured in Separative Work Units (SWU), which is a complex derivation reflecting energy input per amount of uranium input, degree of enrichment of ^{235}U in the product stream, and the degree of depletion of ^{235}U in the tailings stream. The Separative Work Units of separating a mass F of feed with ^{235}U enrichment x_f into a mass P of product with ^{235}U enrichment x_p and tailings of mass T with ^{235}U enrichment x_t , can be expressed as the following (U.S. E.I.A., 2020; Benedict et al., 1981, p. 669):

$$\text{SWU} = TV(x_t) + PV(x_p) - FV(x_f) \quad (1)$$

$$V(x) = (1 - 2x) \ln \left(\frac{1 - x}{x} \right) \quad (2)$$

Note that the units for SWU will be the same as the units used for the values of F , P , and T , which can be expressed as kg, moles, or atom fraction, provided the values for enrichment (x_f , x_p , and x_t) are expressed on the same basis (weight fraction, mole fraction, or atom fraction).

Deconversion

After enrichment, the enriched uranium product is chemically converted into the form best suited for the desired fuel, usually UO_2 powder given the large market for UO_2 fuel in LWRs. Conversion of UF_6 to UO_2 can be accomplished using a wet process, through reactions with water and ammonium (or ammonium carbonate) followed by filtering and heating in a reducing environment. The alternative dry process heats UF_6 to the gaseous state, for contact with steam and then H_2 -rich steam to produce UO_2 . The dry process has some advantages over the wet process, producing less waste and keeping the uranium in a solid form that can be easier to contain than a uranium-bearing solution. For processes that subsequently sinter UO_2 powder into pellets, the powder characteristics are very important, so process steps are often adjusted to result in powder that leads to the best fuel quality. Some fuel production processes, such as those used to make tri-structural isotropic (TRISO) fuel for high temperature gas-cooled or molten-salt-cooled reactors, use sol-gel precipitation processes to produce highly regular ceramic particle shapes of UO_2 or uranium oxycarbide (UCO) suitable for coating (Helmreich and Hunn, 2021). Production of other chemical and physical forms of the uranium material, such as uranium metal, uranium nitride or carbide, or uranium salt, is also possible, though UO_2 is the only form in industrially significant use today.

Fuel fabrication

Fuel fabrication includes all the steps needed to process uranium into form necessary for a specific reactor design. Ceramic powders such as UO_2 , $(\text{U,Pu})\text{O}_2$, UN, and $(\text{U,Pu})\text{N}$ are compressed into pellets for use in LWRs, HWRs, or fast reactors (see Kato, 2021). Those pellets are then loaded into cladding tubes, made of zirconium alloys or stainless-steel alloy to form fuel rods that are assembled into square (for most designs) fuel bundles and assemblies. However, TRISO particle fuel for high-temperature gas reactors or molten-salt-cooled reactors is fabricated in very different manner, which coats individual powder particles which are then consolidated into a graphite matrix shaped as cylindrical compacts or spherical “pebbles,” as described by Helmreich and Hunn (2021).

Metallic uranium might be alloyed with molybdenum or zirconium and cast into ingots or fuel slugs suitable for production of research reactor fuel or fast reactor fuel. Research reactor fuel is commonly encapsulated in cladding to form plates which are then assembled into fuel elements (Cole et al., 2021a,b). Metal fast reactor fuel is placed into a cylindrical cladding tube (Porter and Crawford, 2021), usually with metallic sodium filling the annular gap between the fuel and the cladding, to form fuel rods that are assembled into hexagonal fuel assemblies (or subassemblies, depending on terminology).

Nuclear fuel is produced to very exacting quality requirements, which makes nuclear fuel production very specialized. Descriptions of the more common fuel designs in use or under consideration today appear later in this chapter.

Service

Fuel cycle steps in the progression up to fuel service, or use, are considered to be the front end of the fuel cycle. Nuclear fuel is placed into service when it is loaded into a reactor core and the core is made critical. The great majority of the reactors in the world are used for generating power, but there are other reactors intended to provide neutrons for irradiation testing or production of isotopes. The in-service lifetimes of solid nuclear fuel are usually limited by the degradation of material properties that reduces the ability of the fuel to perform its safety and design functions. In some specialized reactors with very high power density or reactors proposed to operate with molten salt fuel, the fuel lifetime will be limited more by depletion of fissile isotopes than by reduction in fuel integrity. At the end of the fuel's lifetime, fuel assemblies (or fuel salts, in the case of molten-salt-fueled reactors) are removed from the reactor and replaced with fresh fuel.

Used fuel management and storage

The subsequent steps in the fuel cycle progression after use, with used nuclear fuel, are considered the back end of the fuel cycle, as described by Glatz (2020) and Kessler (2021). Upon discharge, used fuel assemblies are usually cooled to allow fission products to decay to lower heat generation rates. This cooling often is carried out in fuel pools, for water-cooled reactors, or in other facilities

configured for efficient removal of decay heat. After decay heat output is reduced sufficiently, used fuel can be removed from cooling locations and placed into dry storage to await final disposition. Even in post-service storage, fuel integrity is important for continued containment of radionuclides in the fuel. Provisions are made for storage and containment of failed or breached fuel, and their storage requires additional attention.

Reprocessing and/or disposal

The final disposition of used nuclear fuel varies by country. For nations that implement or otherwise allow fuel recycling, many options are available (Shwageraus, 2021; Wigeland et al., 2021). Some nations utilize large fuel reprocessing facilities in which used fuel is dissolved into aqueous solution and plutonium is (or all transuranic elements are) separated and consolidated for reuse in new fresh fuel. In those facilities, fission products and the higher actinides are consolidated into waste forms intended for geologic disposal. In other nations, economics or policy decisions discourage commercial investment in used nuclear fuel reprocessing facilities, so the reference plan for disposition of used fuel is direct disposal in a geologic repository. Regardless of the disposition strategy, the radiotoxicity, chemical toxicity, and safeguards associated with used nuclear fuel brings a set of technological challenges unlike those experienced in most other industries. Glatz (2020) and Kessler (2021) describe the various options being considered for the back-end of the fuel cycle, and the specific technologies are addressed by other authors (Senentz, 2021; Marin, 2021; Boullisand Chabert, 2021; Miguiditchian, 2021; Gonzales, 2020; Grambow, 2021; Ewingand Park, 2021).

Fuel design

As a component of a nuclear reactor, nuclear fuel is key to establishing the fission chain reaction for generating nuclear energy and for ensuring safe containment of the fission products. Many configurations of nuclear fuel might be used successfully, but those used or considered today reflect experience gained from the earliest developments of nuclear energy. Solid fuel designs with optimal surface-to-volume ratios are preferred for most reactor technologies. However, technological limitations with solid fuel designs motivated reactor types that use homogeneous liquid forms of nuclear fuel, and molten salt fuels are a favored liquid form due to the desirable thermal and chemical properties of salt compounds. Whereas all solid fuel forms in use are encapsulated in some manner to provide a first barrier to contain fission and activation products (i.e., encased in cladding tubes or in deposited layers), for liquid fuel reactors the analogous first barrier is the reactor vessel and primary coolant loop.

Evolution of nuclear fuel technology has been driven by economics of nuclear plant operation, to (1) decrease plant operating cost by increasing fuel utilization, or (2) increase revenue by increasing power generation.

1. The economics of nuclear reactor operation depends in part on efficient utilization of the nuclear fuel. For example, in the United States for the years 2010–17, nuclear fuel cost comprised 18–21% of total operating cost of commercial LWRs, even as total operating costs have decreased (see Nuclear Energy Institute, 2018). As reactor fuel cycles call for increasing fissile enrichment to allow more energy generation per unit weight of fuel, the proportion of fuel cycle-related cost will increase. However, that additional cost is compensated by increased fuel utilization (fuel efficiency) if more energy, or more fissions, can be obtained from a given amount of fuel before the fuel must be replaced. Fuel utilization is typically limited by one or both of two things: in-service degradation of the fuel components that reduces the reliability of the fuel to contain radionuclides and the reduction of the reactivity value of the fuel as the fissile isotopes are depleted and the fission products accumulate. So, much of the development of nuclear fuel technology seeks to increase fuel utilization by decreasing and managing rates of fuel component degradation and by increasing the concentration of fissile material in the fuel.
2. Nuclear reactors are designed for efficient energy production operate at temperatures as high as the components and structures will allow. Even when reactor operating temperatures are not increased, changes in plant operation can challenge fuel designs and fuel reliability. Changes to operation of existing reactor include plant power uprates (allowing the plant to generate more power than originally designed) and increases in operating cycle length (the length of time that plant operates between fuel reloads), which can expose fuel to greater flow-induced vibrations, greater corrosion, and greater radiation-induced degradation. LWR power uprates include higher average power densities in the cores and higher coolant flow rates to safely remove the increased power. Because efficiencies of energy conversion systems are improved at higher coolant temperatures, most of the advanced nuclear reactor technologies under development feature higher coolant outlet temperatures than LWRs (Stanculescu, 2021). For a given reactor type, increases in fuel operating temperatures can increase rates of degradation and increases in coolant flow rates (which might be necessary for higher power density) can increase flow-induced vibrations that cause fretting wear on fuel components.

Fuel functional and safety requirements and fuel qualification

Fuel systems must fulfill certain functions, some of which ensure the safety of the nuclear reactor system and some of which are necessary for the nuclear reactor to operate efficiently.

Specific requirements are determined for specific nuclear reactor designs, but in general, the functional and safety requirements for solid fuel systems reflect the following:

1. The fuel system must position the fissile material in a stable and predictable manner to allow a controlled fission reaction.
2. The fuel system must allow effective transfer of nuclear fission heat and radioactive decay heat from the fuel to the coolant (heat transfer medium).
3. The fuel system must provide containment of radionuclides (fuel and fission products) for operational convenience and as a first barrier for safety. For most solid fuel systems this containment includes structural containment as well as chemical containment (i.e., some containment contribution provided by the chemical form of the fuel and fission products). In liquid fuel systems (such as molten salt) this containment is provided by the reactor vessel and the primary coolant system.
4. The fuel system must provide or allow a practical means of loading fresh fuel into the core and removing and managing used fuel. For solid fuel this can be accomplished through suitable design of the fuel assembly structures and the equipment used for fuel loading and unloading, but for liquid fuel this is accomplished primarily online by fuel processing systems.
5. The fuel must not compromise the ability of other reactor components to fulfill their safety functions; for example, fuel components must not interfere with the function of reactor control systems (e.g., in-core control rod motion), must not impair the performance of heat removal systems, and must not adversely impact other radionuclide containment barriers such as the primary system coolant boundary.

As with other nuclear reactor components, fuel designs are typically *qualified* for their purpose. Although what is needed to qualify a fuel design is not well documented, *fuel design qualification generally demonstrates, through a combination of testing and analysis, that a fuel design as-specified will perform its design-basis functions under design-basis conditions*. Other aspects of fuel qualification, such as qualification of the fuel production process or fuel handling procedures in a reactor plant, are intended to demonstrate that product and process variability are sufficiently controlled by the prescribed quality assurance regimen.

Qualification of new fuel designs is analogous to those applied to mechanical and electrical components to be used in a nuclear reactor (such as pumps, valves, circuit breakers, and switchgear): as-specified fuel systems and designs are demonstrated through analysis and testing to perform their design-basis functions.¹ The definition of the design basis includes expected fuel operating conditions: fuel, cladding, and coolant temperatures; fission power density in the fuel; rates of power and temperature change; and coolant conditions (such as flow rate and chemistry). The qualification case provides assessment of prior experience and testing and comparison of current design with previous designs. It will also include analysis of expected fuel phenomena and the comparison of those phenomena against limits (e.g., maximum fuel cladding strain or temperature margin to melting temperature). Confidence in fuel designs is typically reached with compilation of fuel material properties and sufficient experience to identify and quantify the degradation of fuel component properties and behavior that accompanies fuel use in service. If the understanding of fuel phenomena is not sufficiently established to allow confident prediction using modeling and simulation, then prior operating experience and testing is additionally important to demonstrate expected fuel reliability and margin to failure.

For a new fuel design, gathering properties and performance information can require 20 or more years, given the time needed to evaluate and demonstrate fuel behavior during in-service residence times of several years; e.g., Crawford et al. (2007). This development and qualification time varies with the maturity of the design and its similarity to other designs. The long duration and considerable investment needed to develop and qualify new fuel designs discourages introduction of innovative designs into operating, or even new, nuclear reactors. Initiatives within nuclear technology development centers, such as US national laboratories, seek to improve the innovation timeline by reducing the time and investment needed to gather the information and experience necessary to qualify a new fuel design and by establishing modeling and simulation tools sufficiently validated to allow accurate fuel behavior prediction with less test data (Terrani et al., 2020). The development of new testing techniques and validation of predictive models are underway (Tonks et al., 2017; Beausoleil II et al., 2020).

Fuel designs

The specific design objectives and optimization challenges for each reactor design or concept have motivated specific fuel design features. This is true of molten salt compositions proposed for different molten salt reactor (MSR) concepts, for example, with chloride salts preferred for some fast-spectrum systems and fluoride salts preferred for thermal-spectrum systems (Gakhar et al., 2021). But the variety of fuel design choices is particularly evident in solid fuel systems. For solid-fueled reactors, the fuel design is determined through optimization of reactor power density (which typically is preferred to be as high as possible to reduce reactor size and cost), heat removal and temperature margins needed to ensure the fuel performs its design-basis functions, and fuel production and handling cost. Heat removal generally drives designs with higher specific surface area, so rod-shaped fuel is most common, as with LWRs and most fast reactor designs, and thin plate-shaped fuel is used in most research reactors. Objectives to provide extremely robust and resilient containment of fission products has led to the TRISO particle fuel design that places the fuel material in encapsulated spheres that are then embedded in graphite spheres (or pebbles) or graphite cylindrical compacts (Electric Power Research Institute, 2019; Helmreich and Hunn, 2021).

¹For LWR component qualification in the United States, the criteria are provided in accordance with US Nuclear Regulatory Commission (NRC) *Standard Review Plan for Licensing of Light Water Reactors*, (USNRC, NUREG-0800).

Table 1 Key properties for oxide, metal, and nitride fuels for fast reactor application.

	Mixed oxide (U, Pu)O ₂	Metal fuel U-Pu-Zr	Mixed nitride (U, Pu)N
Composition	(U _{0.8} Pu _{0.2})O ₂	U-19Pu-10Zr	(U _{0.8} Pu _{0.2})N
Theoretical density (g cm ⁻³)	11.1	15.9	14.3
Metal atom density (g cm ⁻³)	9.75	14.3	13.5
Thermal conductivity (W m ⁻¹ K ⁻¹)			
at 773 K	4.1	18	15
at 1273 K	2.9	31	18
Melting/solidus temperature (K)	3083	1330	3053
			at 1 MPa N ₂ pressure

Data from Arai Y (2012) Nitride fuel. In Konings R and Stoller R (eds) *Comprehensive Nuclear Materials*, pp. 41–54. Amsterdam: Elsevier.

Fuel compositions

Compositions used or proposed for many fuel systems are briefly described here. Values of key properties of different solid fuel compositions are included in **Table 1** (Arai, 2012), illustrating some of the characteristic trade-offs that must be considered when selecting a fuel type for a reactor application. Molten salt fuel compositions are introduced in Gakhar (2021).

LWR fuel: UO₂ is the most common, with ²³⁵U enrichment usually ranging from 3% to 5%. Recent developments call for fuel enrichments greater than 5% ²³⁵U, to allow longer reactor operating cycles or to compensate for design features that increase tolerance to reactor accident conditions (e.g., use of Fe-based cladding rather than Zr-based cladding). Mixed oxide fuel, or MOX, (U,Pu)O₂ is also used in countries where recovery and recycle of plutonium into reactor fuel is allowed and implemented. The design and fabrication of oxide fuels are described by Kato (2021), the irradiation performance of LWR fuel are described by Cantonwine (2020) and the properties of ceramic fuels are described by Pastore and Toptan (2021).

Research reactor fuel: Research reactors present a different set of needs, and although those reactors operate at less challenging (i.e., lower) temperatures than reactors intended for efficient power generation, research reactor fuel technology is challenged by international objectives to reduce fuel uranium enrichment to less than 20% ²³⁵U and to reduce fuel production cost. Research reactors for many years have been fueled with fuel plates comprised of uranium aluminide particles (UAl₂, UAl₃, and UAl₄) dispersed in an aluminum matrix, with ²³⁵U enrichments varied as needed to meet the performance requirements of each specific reactor, and clad in aluminum. Efforts to reduce ²³⁵U enrichment to less than 20% motivated implementation of uranium silicide fuel particles (U₃Si₂) in the aluminum matrix, with its higher uranium density compensating for the reactivity impacts of reducing the ²³⁵U enrichment. Currently, plate-shaped U–Mo alloy fuel clad in aluminum is being developed and qualified in the United States to achieve an even higher uranium density to allow high-performance research reactors (high neutron flux, implying high power density) to be converted to fuel with enrichment less than 20% ²³⁵U. Another fuel common to TRIGA research reactors is uranium-zirconium hydride (U–ZrH_x), which is a form of metallic uranium particle dispersed in ZrH_x with the overall H/Zr stoichiometry nominally around 1.6. This fuel composition is unique in that a moderator is intimately mixed with the fissile material. Research reactor fuels are described in greater detail by Cole et al. (2021a,b).

Fast reactor fuel: Fast reactors have been fueled predominantly by MOX, (U,Pu)O₂ with Pu contents of 25–30 wt% of the heavy metal, but also by UO₂ and metallic uranium alloys such as U–Mo and U–Zr. Other fuel compositions that have been proposed and studied to varying extents include U-Pu-Zr, uranium nitride (UN), uranium-plutonium nitride (U,Pu)N, uranium carbide (UC, U₂C₃), and uranium-plutonium carbide (U,Pu)C/(U,Pu)₃C₂. Each of the fuel compositions brings different fuel performance considerations, with fuel stoichiometry of the ceramic fuel types and the interdiffusion of the metallic fuel alloys with the metallic cladding being important aspects to address. More detail is available on ceramic fast reactor fuel (Kato, 2021; Pastore and Toptan, 2021; and Jensen and Woolstenhulme, 2021) and on metallic fast reactor fuel (Porter and Crawford, 2021; Novascone et al., 2021; and Jensen and Woolstenhulme, 2021).

High-temperature gas reactor fuel: TRISO fuel resulted from development efforts seeking to reduce requirements for a reactor containment building by incorporating robust fission product containment into the fuel itself. TRISO fuel makes use of spherical UO₂ or uranium oxy-carbide, UCO, fuel particles coated by layers of carbon, pyrolytic graphite and silicon carbide. The containment is being proven through testing to be sufficiently robust to allow the fuel to remain at reactor-accident temperatures for extended periods of time without significant release of fission products. More information on TRISO fuel appears in Helmreich and Hunn (2021) and Gerczak (2021).

Fuel geometries

An example of a boiling water reactor (BWR; see Hamon, 2021) fuel design appears in Fig. 2. For convenience, the fuel rods are assembled as a bundle of rods to facilitate fuel handling (i.e., handling bundles of many rods is more efficient than handling very large number of separate fuel rods). A lower tie plate in the fuel assembly affixes the fuel rods into position at the bottom of the bundle, and grid spacers placed at different elevations along the length of the bundle hold the fuel rods in their respective positions, ensuring the rods maintain spacing for coolant flow. The lower tie plate also includes features for the fuel assembly interface with the reactor core grid, while the upper tie plate anchors the fuel rods at their upper end and includes design features that allow handling of the fuel assemblies. A square box, or channel, is placed around the fuel bundle to provide a defined coolant



Fig. 2 Rendering of the GNF-2 fuel assembly, which is shown in a horizontal orientation rather than the vertical in-service orientation. LWR fuel assemblies such as this range in length from roughly 3.5 to 4.5 m. Courtesy Global Nuclear Fuel.

geometry through the fuel bundle, to add structured load paths to the fuel assembly and the assembled reactor core, and to reduce any deleterious impact of one fuel bundle on another. Additional design features and components were introduced into the fuel assembly to optimize fuel utilization and improve reactor stability in the boiling water environment; those features include incorporation of water rods (hollow tubes) in place of selected fuel rods to allow water to be directed into the top of the fuel bundle to improve neutron moderation in the upper part of the fuel bundle, thereby improving fuel utilization. Partial-length fuel rods are used in selected bundle positions to replace fuel volume with water volume in the upper part of the bundle, to improve fuel utilization and fuel bundle pressure drop. Other features include design of the grid spacers to improve boiling transition behavior in the fuel bundle (by improving boiling characteristics at the fuel rod surface) and incorporation of debris filters into the fuel bundle coolant inlet in the lower tie plate to reduce fuel rod debris fretting. Finally, optimization of thermal-hydraulic performance and pressure drop within the fuel bundle improves reactor stability of modern BWR fuel designs. [Cantonwine and Rand \(2021\)](#) provides detail on the irradiation performance of LWR fuel, including BWR fuel.

Sodium-cooled fast reactor (SFR) fuel is similarly designed as bundles of rods, as seen in [Figs. 3 and 4](#). Most SFRs use hexagonal bundle geometry, rather than square bundle arrays as used in LWRs, primarily to reduce coolant-to-fuel volume ratio (and thereby harden the neutron spectrum), so the fuel rod bundles are enclosed in a hexagonal duct (or “hex can”) to form fuel assemblies. The hexagonal duct functions in a manner similar to the square channels used for BWR fuel: to provide structured load paths for the fuel bundle, to define the coolant flow geometry for the fuel bundle, and to mitigate or reduce any deleterious impacts between fuel bundles. Similarly, SFR fuel assemblies are equipped with components at the top of the assembly to allow fuel handling and at the bottom of the assembly to interface with the reactor core grid and to allow the right amount of coolant into the fuel bundle under design-basis conditions. SFR fuel differs, however, in the means used to ensure spacing of fuel rods, which is accomplished using wires helically wrapped around the fuel rods. The helical wrapping allows coolant to flow upward between the rods and induces coolant mixing between fuel rods to reduce variation in coolant temperature across the section of the fuel bundle. Because SFR coolant is intended to stay in the liquid phase and there is no need to moderate the neutrons, unlike the coolant in BWRs, and because SFR fuel rods are shorter (~ 1 m) than in LWRs (3–4 m), fewer flow-optimizing features are needed in the fuel bundle. However, the higher neutron energy in fast-spectrum reactors induces more neutron damage in reactor components and structural materials than is typical of LWRs, so SFR fuel assemblies often include stainless steel reflectors (or shields) above and below the fuel rods to reduce neutron flux incident on structures outside the fuel assemblies. Information on irradiation performance of fast reactor fuel can be found in [Jensen and Woolstenhulme \(2021\)](#).

An example of a TRISO fuel particle is shown in [Fig. 5](#), in which the coated layers (described in detail by [Helmreich and Hunn, 2021](#)) are clearly visible. The fuel kernel, UCO in [Fig. 5](#) but UO_2 has been used in other fuel designs, lies in the center of the sphere and is coated with layers of pyrolytic carbon and SiC to provide chemical and physical accommodation for and containment of fission products. The size of typical TRISO particles and their consolidation into a cylindrical compact can be seen in [Fig. 6](#). Another design uses spherical graphite compacts of TRISO fuel particles, to form a ball or “pebble” that is coated with a graphite “crust.” The density of the fuel particles within the compact or pebble is a key design feature, with higher densities favored for increasing reactor power density and reducing core size but difficult to achieve due to degrading structural and dimensional integrity of the compact. For some high-temperature gas reactor designs ([Futterer, 2021](#)), the compacts are stacked into fuel channels drilled into hexagonal graphite blocks which are assembled to form the core assembly. Other reactors use beds of pebbles, with the interstitial space between pebbles sufficient for coolant flow and heat removal. Reactor concepts that use solid TRISO fuel with molten salt coolant offer the prospect for higher reactor power density, because the higher heat capacity of molten salt coolant over helium provides more efficient heat transfer from the core. [Gerczak \(2021\)](#) provides information on the behavior of TRISO fuel in high-temperature gas reactors.

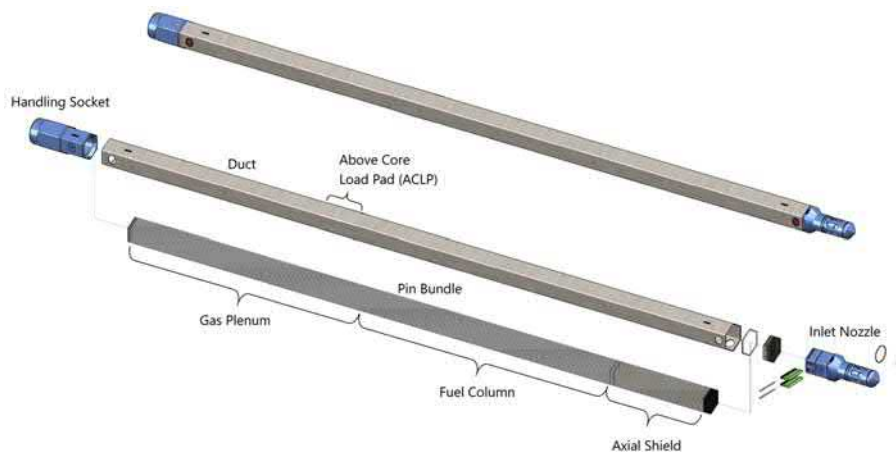


Fig. 3 Rendering of the Traveling Wave Reactor fuel assembly showing the fuel bundle, the hexagonal duct, the inlet nozzle and handling socket, and the final configuration of the components into a finished fuel assembly. Depending on the size of the Traveling Wave Reactor, these assemblies are projected to range from 3.5 to 6 m in length. Courtesy TerraPower, LLC.

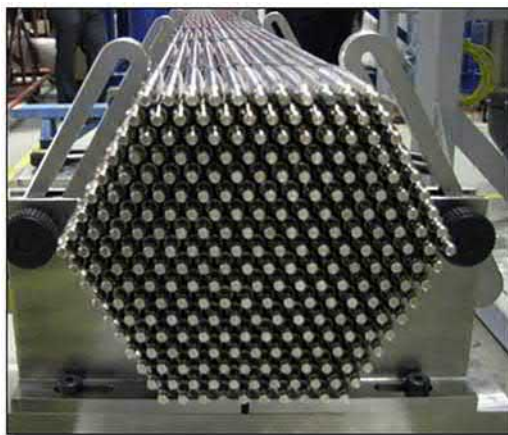


Fig. 4 Photo showing the top end of a mock-up fuel bundle for a Traveling Wave Reactor fuel assembly. This rod bundle is composed of 217 fuel rods, each of which is helically wrapped with a spacer wire. Courtesy TerraPower, LLC.

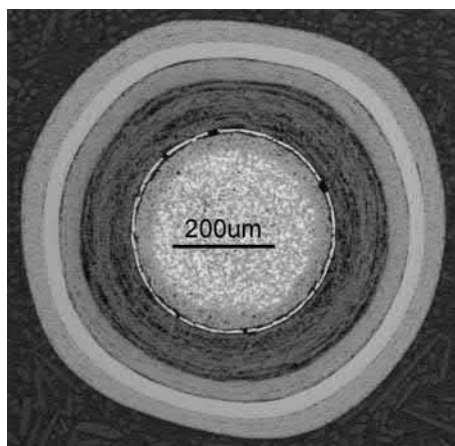


Fig. 5 Cross sectional ceramographic image of a TRISO fuel particle, showing the layers of pyrolytic carbon and SiC that contain fission products within the particle. Courtesy U.S. Department of Energy.



Fig. 6 Photo showing individual TRISO fuel particles, TRISO fuel particles as seen in the cross section of a cylindrical compact, and an intact cylindrical compact. Courtesy U.S. Department of Energy.

Section overview

This section of the Encyclopedia presents an overview of technologies important to the front end of the nuclear fuel cycle and to utilization of nuclear fuel in reactors. The technologies addressed have varied maturity. For example, technology and practice for mining, extracting, converting, and enriching uranium and for producing, transporting, and using LWR fuel are commercialized and deployed by a global industry. In fact, many of the products in the LWR fuel supply chain are commodities, with value obtained at the margins for sellers and buyers. These commercialized technologies determine the fuel cycle economics of LWR power generation and set the standard to be improved upon by any new fuel cycle or reactor technology.

Some fuel technologies are less mature. It remains to be seen whether new LWR fuel designs intended to improve resistance to reactor accidents and off-normal conditions will offer operating cost reduction (by reducing the need to maintain specialized safety equipment in the plants) to compensate for anticipated increase in fuel cycle cost. Although SFRs are not deployed on an industrial scale, the economic potential of oxide or metallic fuels in SFRs is relatively well known. The economic prospects of molten salt fuel technology for MSRs, based on principles of molten salt chemistry and experience with experimental reactors operated in the 1960s (Gakhar et al., 2021), are encouraging but somewhat speculative at this point.

Chapters in this section will provide the reader with some background on the technology and market considerations for the front-end (i.e., the extraction and processing of uranium and thorium for subsequent fabrication of specific fuel products). Most of the discussion and examples will address uranium-based fuel, because the uranium fuel market is the most developed. Content on uranium and thorium resources and extraction is provided by Hore-Lacy (2021), Conca (2021), and Haneklaus (2021). Uranium enrichment and conversion are addressed by Hogan (2021).

Fabrication of specific fuel designs is addressed in chapters that address ceramic fuel (Kato, 2021), metal fuel (Porter and Crawford, 2021), research reactor fuel (Cole et al., 2021a,b), and TRISO fuel (Helmreich and Hunn, 2021). Other fuel types, designed for use in power generating reactors, with significant development and demonstration, though not yet adopted commercially, include TRISO coated particle fuel for use in high-temperature gas reactors (and recently-proposed salt cooled reactors; see Gakhar et al., 2021), uranium-plutonium dioxide ($\text{UO}_2\text{-PuO}_2$) fuel and uranium alloy or uranium-plutonium alloy fuel for fast-spectrum reactors. Each of these fuel types has required unique fabrication approaches, particularly as conventional fabrication processes were adapted to nuclear fuel, and these fabrication approaches are described in this section.

Finally, because processing and handling of fissile material in the fuel cycle brings nuclear safety considerations not encountered in other industries, a chapter is provided to address nuclear criticality safety (Bowen, 2021).

Because the fuel cycle theme is so broad, some associated topics are addressed in other sections of this Encyclopedia. Specifically, the approaches used to design nuclear reactor cores for operating cycles (i.e., the duration of operation time between fuel reloading, or replacement of used fuel with fuel) are described by Shwageraus (2021) and Wigeland et al. (2021) as is the technology for addressing used nuclear fuel to prepare its constituents for disposal (e.g., Glatz, 2020). Other authors address the technology available to remove useable constituents from used nuclear fuel for reintroduction back into the nuclear fuel cycle (Senentz, 2021; Marin, 2021; Boullis and Chabert, 2021; Miguiditchian, 2021), some of which has been commercially deployed in nations such as France and United Kingdom, but which has not yet reached commercial deployment in other nations, such as the United States and Japan.

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Uranium Resources

Ian Hore-Lacy, World Nuclear Association, Blackburn North, VIC, Australia

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Geology and mines

Uranium is a relatively common element in the crust of the Earth (very much more than in the mantle). It is a metal approximately as common as tin or zinc, and it is a constituent of most rocks and even of the sea ([Herring, 2021](#)).

Uranium deposits worldwide are grouped into 15 major categories of deposit types based on their geological setting. The most recent geological classification has been defined by the International Atomic Energy Agency ([Bruneton et al., 2014](#)) and is adopted in the following edition of *Red Book* (2014). The deposit types have fundamental characteristics and recognition criteria and in that respect, while mainly named by host rock, the types are essentially empirical models, based on observable characteristics.

A comprehensive listing of world uranium deposits from a purely geological perspective includes much low-grade mineralization which is subeconomic. There is very little recovery of uranium from the three deposit types with the largest estimated totals—lignite, black shale and phosphate. Three of the next most-abundant comprise most of the 6.1 million tonnes of economic resources listed in the 2018 *Red Book* ([OECD NEA & IEA, 2018](#)), and contribute much of the world's supply of uranium today: sandstone, iron-oxide breccia, and Proterozoic unconformity.

Sandstone deposits

Sandstone uranium deposits occur in medium to coarse-grained sandstones deposited in a continental fluvial or marginal marine sedimentary environment. Impermeable shale/mudstone units are interbedded in the sedimentary sequence and often occur immediately above and below the mineralized sandstone. Uranium is precipitated under reducing conditions caused by a variety of reducing agents within the sandstone including: carbonaceous material, sulfides (pyrite, H₂S), hydrocarbons, and interbedded

basic volcanic ash with abundant ferro-magnesian minerals (e.g. chlorite). There are several sub-types of sandstone deposits, often mixed.

Sandstone deposits constitute about 18% of world uranium resources and 41% of known deposits, and are of major economic importance in Kazakhstan, Uzbekistan, Australia, the United States and Niger. Orebodies of this type are commonly low to medium grade (0.05–0.35% U) and individual orebodies are small to medium in size (ranging up to a maximum of 50,000 tU). Imouraren (Niger) and some Kazakh deposits are larger. Several sub-types are mined by in situ leach (ISL) methods. The main primary U minerals are uraninite and coffinite.

The United States has large resources in sandstone deposits in the Western Cordillera region, and most of its uranium production has been from these deposits, recently by ISL mining. The Powder River Basin in Wyoming, the Colorado Plateau and the Gulf Coast Plain in south Texas are major sandstone uranium provinces.

Polymetallic iron-oxide breccia complex deposits

Olympic Dam is the world's largest deposit of uranium, and accounts for the major part of Australia's uranium resources, though it is only recovered as a by-product with copper. Both it and Carrapateena occur in a hematite-rich granite breccia complex in the Gawler Craton, overlain by approximately 300 m of flat-lying sedimentary rocks of the Stuart Shelf geological province.

The central core of the Olympic Dam complex is barren hematite-quartz breccia, flanked by zones of intermingled hematite-rich breccias and granitic breccias. These zones are approximately 1 km wide and extend almost 5 km. Virtually all the economic copper-uranium mineralization is hosted by these hematite-rich breccias. The deposit contains iron, copper, uranium, gold, silver, rare earth elements and fluorine. Only copper, uranium, gold, and silver are recovered. Uranium grades average from 0.07 to 0.035% U, the higher-grade mineralization being pitchblende. Copper grades average 2.7% for proved reserves and 2.0% for probable reserves. Details of the origin of the deposit are still uncertain, though the principal mechanisms which formed the breccia complex are considered to have been hydraulic fracturing, tectonic faulting, chemical corrosion, and gravity collapse.

Proterozoic unconformity deposits

Unconformity-related deposits arise from geological changes occurring close to major geological unconformities from the Proterozoic era. Below the unconformity, the metasedimentary rocks which host the mineralization are usually faulted and brecciated. The overlying younger Proterozoic sandstones are usually undeformed. Unconformity-related deposits constitute about one third of the western world's uranium resources and they include some of the largest and richest deposits. Minerals are uraninite and pitchblende associated with strong quartz dissolution. The main deposits occur in Canada (the Athabasca Basin, Saskatchewan and Thelon Basin, Northwest Territories); and Australia (the Alligator Rivers region in the Pine Creek Geosyncline, Northern Territory and Rudall River area, Western Australia).

All of Canada's uranium production in Saskatchewan over the last 40 years is from unconformity-related deposits—Key Lake, Cluff Lake, Rabbit Lake, McClean Lake, McArthur River and Cigar Lake deposits, with some ore around 20% uranium. The deposits in the Athabasca Basin occur below, across and immediately above the unconformity, with the highest grade deposits situated at or just above the unconformity (e.g. Cigar Lake and McArthur River). In the Alligator Rivers region, the known deposits (Ranger, Jabiluka, Koongarra and Nabarlek) are hosted in the basement structure below the unconformity and are generally lower grade. Strata-bound structure-controlled unconformity deposits include the low-grade Chitral and Lambapur deposits, Cuddapah Basin, India.

Most uranium is mined as the primary mineral in any mining project, but a significant amount is obtained as by-product, notably at Olympic Dam in South Australia, which is in fact the world's largest known orebody for uranium, though it is primarily a copper mine. In addition, there is potential production from so-called unconventional resources. This article is mainly focused on situations where uranium is primary, or a significant by-product.

Major uranium mines

Mining methods have been changing. In 1990, 55% of world production came from underground mines, but this reduced dramatically to 33% in 1999. From 2000 the new Canadian mines increased it again. In situ leach (ISL, or in situ recovery (ISR)) mining has been steadily increasing its share of the total, mainly due to Kazakhstan, and in 2019 accounted for over half of production. In 2019 production was as follows:

<i>Method</i>	<i>Tonnes U</i>	<i>%</i>
In situ leach (ISL)	30,339	57%
Underground and open pit (except Olympic Dam)	19,607	36%
By-product (including Olympic Dam, underground)	3710	7%

Table 1 The largest-producing uranium mines in 2019.

Mine	Country	Main owner	Type	Production (tonnes U)	% of world
Cigar Lake	Canada	Cameco/Orano	Underground	6924	13
Husab	Namibia	Swakop Uranium (CGN)	Open pit	3400	6
Olympic Dam	Australia	BHP Billiton	By-product/underground	3364	6
Moinkum & Tortkuduk	Kazakhstan	Orano/Kazatomprom	ISL	3209	6
Inkai, sites 1–3	Kazakhstan	Kazatomprom/Cameco	ISL	3209	6
Budenovskoye 2	Kazakhstan	Uranium One/Kazatomprom	ISL	2600	5
Rossing	Namibia	Rio Tinto	Open pit	2076	4
Central Mynkuduk	Kazakhstan	Kazatomprom	ISL	1964	4
SOMAIR	Niger	Orano	Open pit	1912	4
South Inkai (Block 4)	Kazakhstan	Uranium One/Kazatomprom	ISL	1601	3
<i>Top 10 total</i>				30,032	56%

Unconventional resources

In addition to the 6.1 million tonnes of uranium in the world's known recoverable resources, there are substantial amounts comprising what is known as "unconventional resources." Such unconventional resources, from which uranium can be produced in conjunction with other metals, as by-product, have accounted for over 11% of historical uranium production.

The main unconventional resource for uranium is *rock phosphate*, or phosphorite, and some 20,000 tU has been recovered as a by-product of agricultural phosphate production to the 1990s, but it then became uneconomic. Estimates of the amount available range from 9 to 22 million tonnes of uranium, though the 2018 *Red Book* tabulates only about 8 million tonnes.

With uranium as a minor by-product of phosphates, the potential supply is tied to the economics of phosphate production, as well as uranium price, coupled with the environmental benefits of removing uranium from the waste stream and/or product. World phosphorous pentoxide (P_2O_5) production capacity from about 250 Mt. of rock phosphate is about 50 million tonnes per year. From 1981 to 1992 US production from central Florida's phosphate deposits as by-product averaged just over 1000 tU per year, comprising up to 20% of US total, then fell away sharply and finished in 1998. Morocco has by far the largest known resources of uranium in phosphate rock (Phosenergy, n.d.).

Rare earth element (REE) deposits are another such unconventional resource. REEs play a critical role in the application of many modern technologies, including magnetic resonance imaging (MRI) machines, satellites, batteries, LED screens and solar panels. China is the leading supplier of REEs, giving rise to commercial pressure for development of deposits elsewhere.

REEs are a set of 17 chemical elements in the periodic table, specifically the 15 contiguous lanthanides plus the lighter scandium and yttrium. Scandium and yttrium are considered REEs since they tend to occur in the same ore deposits as the lanthanides and exhibit similar physical and chemical properties. REEs are in fact relatively abundant in the Earth's crust, but are rarely found in concentrations that are economically exploitable. REE resources occur in four primary geological settings: carbonatites, ion-absorption clay deposits, igneous systems and monzanite-xenotime placer deposits. REE resources are usually reported as rare earth oxides (REO).

Kvanefeld in Greenland is the main REE deposit with major potential for uranium production, with Sorensen, Zone 3 and Steenstrupfeld orebodies in the same Ilimaussac intrusive complex. Those four deposits have a total of 228,000 tU (May 2015, JORC-compliant), nearly half of it measured and indicated resources. Greenland Minerals (<http://www.ggg.gl>) has an agreement with Chinese company Shenghe Resources which will enable development to proceed.

Black (Alum) shales are another unconventional resource with some attempts being made to exploit them. The 2018 *Red Book* tabulates about 300,000 tU in Sweden and mentions 24,000 tU in Finland at the Sotkamo mine of Terrafame Oy, for which the government granted a permit in 2020 for uranium recovery as by-product by heap leaching. Uzbekistan has 32,900 tU in black shales, attracting Chinese interest.

Exploration: Identifying resources

This section is drawn substantially from Kyser (2016).

Exploration for uranium deposits, as with any type of geological search, requires the integration of regional geology, structural geology, geophysics and geochemistry and must embrace new technologies and research results, particularly ore genesis models, to be effective. Political risks must also be factored in.

Exploration effort is focused on areas that have the potential to host economic uranium deposits. The geological assessment of the uranium mineral endowment is a knowledge-based activity that is based on understanding deposit models and the probability of their occurrence. A great deal depends on developing understanding of how orebodies were formed millions or billions of years ago. This knowledge then enables interpretation of evidence collected as described below.

A variety of methods are normally used for exploration, including a balance of geological mapping, remote sensing, geophysics, and geochemistry, followed by actually drilling to investigate anomalies which may be possible orebodies.

Geological mapping

Some broad-scale mapping is normally done by governments. Further, more detailed, mapping is required to supplement what is available on the public record. It is focused on structures that are evident and what the structural evolution of the area is understood to be. Structures are controlling many types of uranium deposits related to hydrothermal fluid circulation.

Regional-scale structural results from mapping can be integrated with geophysical methods that identify possible structures at depth based on physical anomalies of the rocks. Although structure is critical in the formation of the deposits, the vast majority of structures with seemingly appropriate characteristics are not mineralized. Mapping is also used to characterize the cover, including Quaternary geology, soil character and type, and character of vegetation.

Remote sensing

This involves acquiring, processing and interpreting spectral information about the Earth's surface, and recording interactions between matter and electromagnetic energy, mainly from satellites. Among the remote sensing data to be integrated in evaluating prospective areas are radiometry as part of airborne geophysics, and other means of identifying radiometric anomalies. These techniques detect energy reflected and emitted from the Earth's surface from minerals, vegetation, soils, etc., in selected wavelengths.

Geophysical methods

These vary with the type of uranium deposit because different aspects of the mineralisation are targeted. The mainstay of exploration geophysics for uranium deposits is gamma ray radiometry and spectrometry used from airborne regional to local detailed surveys. This is followed up with down-hole logging of gamma radiation and measurement on outcrops using hand-held units. Gamma data is normally collected as part of airborne magnetic surveys with magnetic contrasts used to reveal the general geology. Airborne gamma-ray spectrometry directly indicates U, K, and Th in surficial material.

Magnetics are used to map basement lithologies in basin-hosted systems. Various electromagnetic (EM) techniques are used to image graphitic conductors that are occasionally associated with unconformity-related deposits, paleochannels in sandstone-hosted systems, and alteration haloes in basin related systems. Ground EM surveys are used routinely to follow-up airborne surveys with deeper penetration through any sandstone cover, and they often provide detailed assessments of targets of interest.

Gravity measurement reflects structures and alteration because of changes and breaks in lithology. Seismic techniques can reveal structures, areas of alteration, stratigraphy and depth to basement.

Measuring radon emanation can indicate buried mineralisation, due to the ability of radon, a gaseous progeny of from decay of U-238, to migrate to the surface. Most instrumentation relies on alpha particle detection.

In areas such as sedimentary basins where the deposits tend to be well covered, geophysical techniques are powerful tools to see physical anomalies beneath.

Prompt Fission Neutron (PFN) borehole logging technology directly measures the content of uranium in boreholes, overcoming the problem of disequilibrium which limits the interpretation of uranium concentrations using gamma logging tools. This is particularly important in placer deposits such as in palaeochannels.

Although geophysics costs can be substantial, drilling is much more so, and a great deal of information about hidden geology can be obtained.

Geochemical methods

There are two distinct processes that expand the geochemical footprint of a deposit: (1) primary dispersion, which provides information on alteration and primary element dispersion associated with the original formation of the ore; and (2) secondary dispersion, which provides clues about element migration from alteration and ore zones well after emplacement.

During primary dispersion, components in the mineralizing fluids permeate into the country rock, which alters primary minerals and elevates the concentrations of "pathfinder" elements. These enrichments relative to background concentrations are sometimes evident in the geochemistry of rocks up to several kilometers away from the uranium mineralisation. Hydrothermal alteration minerals associated with mineralizing fluids can be detected.

The levels of uranium concentration that can be considered anomalous vary according to the deposit type and its environment. In U-rich granites, significant anomalies should be above some tens of ppm, the granites having themselves a background uranium concentrations of 10–30 ppm, whereas for unconformity related deposits, anomalies of a few ppm in the sandstone may be meaningful because most of the sandstone has an average uranium content below one ppm.

In contrast to primary dispersion that occurs with the ore-forming process, secondary dispersion occurs subsequently via the mobilization of ore or alteration elements into the environment around the deposits. Tracing element migration in the near-surface environment involves understanding the secondary dispersion of elements.

Most ore deposits are potential havens for microbes that can mobilize elements during secondary dispersion. Microbe-mobilized elements can migrate to the surface, particularly along fractures and faults, and make their way into the biosphere. Their specific element and isotope signatures can reflect the deposit at depth.

Mineralogy is an excellent exploration guide when specific alteration minerals are developed in association with ore deposit formation. The mineralogy of the uranium is also critical for evaluating the processing costs for uranium extraction.

Chemical analysis of groundwater can be a useful strategy for regional exploration for uranium in reduced sediments in palaeochannels.

Groundwater samples collected from boreholes tens of meters from unconformity-related uranium mineralisation have consistently high levels of uranium, radium, radon and helium. Biogeochemistry of plants indicate that tree roots can extract anomalous uranium from groundwater that may reflect deposits at 300 m depth.

Lake water and sediment geochemistry and radiometric prospecting are significant tools in early regional exploration for uranium deposits in Canada due to the mobility of uranium in surface waters, which allows the element to disperse widely from its source.

Drilling

After targets have been identified by integrating favorable geophysical and geochemical results, the next stage of exploration is drilling. Most alteration halos associated with deposits are barren. Directional drilling allows several intersections to be made from a single pilot hole, reducing drilling costs and improving target precision.

Given that drill hole samples represent an actual physical sample of the subsurface and are among the costliest aspects of exploration, interpreting them needs to be linked with the geological, geophysical, and geochemical data. Core or cuttings are logged to reveal lithology and stratigraphy. In addition, down hole electromagnetics, radiometrics, and neutron tomography should be done if possible.

At each step, the property should continually be re-evaluated so that informed decisions about investing in further steps can be made.

Quantification of resources

Total world resources of uranium, as with any other mineral or metal, are not known exactly. The only meaningful measure of long-term security of supply is the known reserves in the ground capable of being mined.

An orebody is, by definition, an occurrence of mineralization from which the metal is economically recoverable. Orebodies, and thus measured resources—the amount known to be economically recoverable from orebodies—are therefore relative to both costs of extraction and market prices. For example, at present neither the oceans nor any granites are orebodies, but conceivably either could become so if prices were to rise sufficiently. At 10 times the current price, seawater, for example, might become a potential source of vast amounts of uranium. Thus, any predictions of the future availability of any mineral, including uranium, which are based on current cost and price data, as well as current geological knowledge, are likely to prove extremely conservative (Table 2).

Uranium has been successfully mined since the 1940s. Historical uranium production is generally well known, though uncertainties remain about the amount mined in the Soviet Union between 1945 and 1990. Table 3 summarizes historical production. World Nuclear Association has estimated production in countries where data is unavailable.

Reporting uranium resources

The following are internationally-recognized categories based on Australia's JORC code, which the Canadian NI 43-101 code follows.

A "mineral resource" is a known concentration of minerals in the Earth's crust with reasonable prospects for eventual economic extraction. Mineral resources are sub-divided, in order of increasing geological confidence, into inferred, indicated and measured categories.

- An "inferred" mineral resource is that part of a mineral resource for which tonnage, grade and mineral content can be estimated with only a low level of confidence. The information on which it is based is limited, or of uncertain quality and reliability.
- An "indicated" mineral resource is that part of a mineral resource for which tonnage, grade and mineral content can be estimated with a reasonable level of confidence. It is based on exploration, sampling and testing information which is adequate to assume but not confirm geological and/or grade continuity.
- A "measured" mineral resource is that part of a mineral resource for which tonnage, physical characteristics, grade and mineral content can be estimated with a high level of confidence. It is based on detailed and reliable exploration, sampling and testing information with locations spaced closely enough to confirm geological and grade continuity.

A "mineral" reserve (or ore reserve) is the economically mineable part of a measured and/or indicated mineral resource. It allows for dilution and losses which may occur when the material is mined. Appropriate assessments and studies will have been carried out,

Table 2 Uranium resources by country in 2017.

	<i>Tonnes U</i>	<i>Percentage of world</i>
Australia	1,818,300	30%
Kazakhstan	842,200	14%
Canada	514,400	8%
Russia	485,600	8%
Namibia	442,100 ^a	7%
South Africa	322,400	5%
China	290,400	5%
Niger	280,000 ^a	5%
Brazil	276,800	5%
Uzbekistan	139,200 ^a	2%
Ukraine	114,100	2%
Mongolia	113,500	2%
Botswana	73,500 ^a	1%
Tanzania	58,200 ^a	1%
USA	47,200	1%
Jordan	43,500	1%
Other	280,600	4%
<i>World total</i>	6,142,600	

Identified resources recoverable (reasonably assured resources plus inferred resources), to US\$130/kg U, 1/1/17. The total recoverable identified resources to US\$260/kg U is 7.989 million tonnes U. Identified resources in situ to US\$130/kg U are 8.122 Mt., and to US\$260/kg U, 10.653 Mt.

^aIAEA estimate.

From OECD NEA & IAEA (2018) *Uranium 2018: Resources, Production and Demand ('Red Book')*. OECD NEA & IAEA.

Table 3 Historical uranium production, 1945–2018.

	<i>Cumulative production (tU)</i>
Canada	531,608
Kazakhstan/Uzbekistan	493,164
United States	374,838
Germany	217,161
Australia	219,693
Russia	170,870
South Africa	164,697
Niger	146,379
Namibia	135,572
Czech Republic	111,183
France	77,015
Ukraine	67,845
China	52,337
Others	147,230
<i>Total</i>	2,909,592

and include consideration of realistically assumed mining, metallurgical, economic, marketing, legal, environmental, social and governmental factors. Mineral or ore reserves are sub-divided in order of increasing confidence into probable mineral/ore reserves and proved mineral/ore reserves.

- A “probable” mineral reserve (or probable ore reserve) is the economically mineable part of an indicated mineral resource. Studies to at least pre-feasibility level will have been carried out, demonstrating that extraction could reasonably be justified.
- A “proved” mineral reserve (or proved ore reserve) is the economically mineable part of a measured mineral resource. Studies to at least pre-feasibility level will have been carried out, demonstrating that extraction is justified.

Using uranium resources

The world’s power reactors, with combined capacity of some 400 GWe, require about 65,000 t of uranium from mines or secondary sources each year. While this capacity is being run more productively, with higher capacity factors and reactor

power levels, the uranium fuel requirement is increasing, but not necessarily at the same rate. The factors increasing fuel demand are offset by a trend for higher burn-up of fuel and other efficiencies, so demand is steady. (Over the years 1980 to 2008 the electricity generated by nuclear power increased 3.6-fold while uranium used increased by a factor of only 2.5.)

The world's present measured resources of uranium (6.1 Mt) in the cost category less than three times present spot prices and used only in conventional reactors, are enough to last for about 90 years. This represents a higher level of assured resources than is normal for most minerals. Further exploration and higher prices will certainly, on the basis of present geological knowledge, yield further resources as present ones are used up.

Reducing the tails assay in enrichment reduces the amount of natural uranium required for a given amount of fuel. Reprocessing of used fuel from conventional light water reactors also utilizes present resources more efficiently, by a factor of about 1.3 overall. The 2018 *Red Book* recognized these trends, along with more efficient plant operation, and as a result of which, the report's generic reactor fuel consumption was reduced from 175 tU per GWe year at 0.30% tails assay (2012 report) to 160 tU per GWe year at 0.25% tails assay (2016 and 2018 reports). The corresponding U_3O_8 figures are 206 t and 189 t. Note that these figures are generalizations across the industry and across many different reactor types.

Today's nuclear reactors use less than 1% of the mined uranium. About 89% is put aside as depleted uranium from enrichment, and 11% is fed into the reactor as actual fuel. Of this, 2.5% is fissioned (i.e. 0.275% of original) and a similar amount is converted into plutonium, about half of which is fissioned, so total of about 0.4% of original uranium. In a fast neutron reactor potentially all the mined uranium can be used, and a figure of 50 times the energy output is often quoted. In any kind of reactor a limitation of fuel use is the build-up of neutron poisons in the fuel.

So a very significant change in uranium use is likely if and when many countries replace present reactor fleets with fast neutron reactors. Instead of using less than 1% of the uranium that is mined, they will be able to use a majority of it by converting U-238 to plutonium-239 in the reactor core on a much larger scale than at present (where about one third of the power typically comes from Pu-239). This will potentially make uranium mining redundant, given that there will be over two million tonnes of depleted uranium conveniently available as potential fuel by mid-century.

About half of the Generation IV reactors being developed internationally are fast neutron reactors. However, practically all the reactors planned and under construction today are Generation III models with an expected operating life of 60 years, and they will be predominant at least until mid-century.

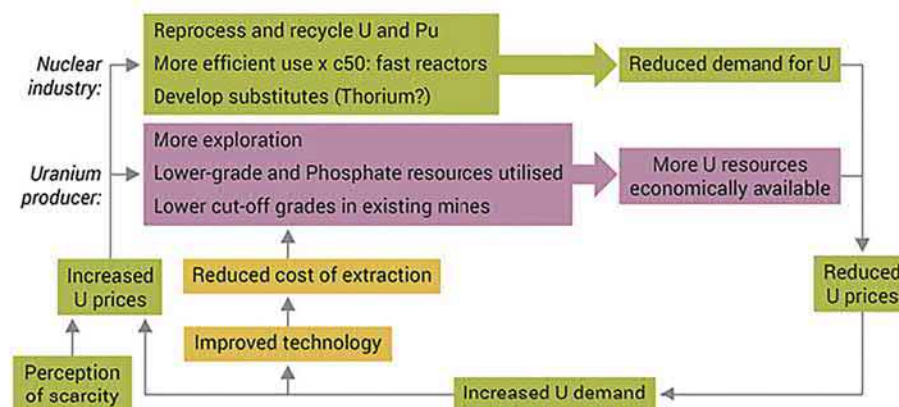
Sustainability of uranium supply

The sustainability of uranium supply for world nuclear power requirements is affected by three things:

- gains in geological knowledge and understanding of mineral deposits;
- new technologies utilized to discover, process and use them;
- economic principles, particularly where resources are thought of only in present terms, not in terms of what will be economic through time, nor with concepts of substitution in mind.

The following diagram summarizes the basic dynamics involved:

Economic Adjustments in Uranium Supply and Use



Today's reactor fuel requirements are met almost entirely from primary supply (direct mine output) with some from secondary sources: commercial stockpiles, nuclear weapons stockpiles, recycled plutonium and uranium from reprocessing used fuel, and some from re-enrichment of depleted uranium tails (left over from original enrichment).

These various secondary sources make uranium unique among energy minerals.

Mining and milling uranium ore

Uranium mines operate in some 20 countries, though in 2018 some 51% of world production came from just 10 mines in four countries (see [Table 1](#)).

Most of the uranium ore deposits at present supporting these mines have average grades in excess of 0.10% of uranium—that is, greater than 1000 ppm. In the first phase of uranium mining to the 1960s, this would have been seen as a respectable grade, but today some Canadian mines have huge amounts of ore up to 20% U average grade. Other mines however can operate successfully with very low-grade ores, down to about 0.02% U.

Some uranium is also recovered as a by-product with copper, as at Olympic Dam mine in Australia, or as by-product from the treatment of other ores, such as the gold-bearing ores of South Africa, or from phosphate deposits such as Morocco and Florida. In these cases the concentration of uranium may be as low as a tenth of that in orebodies mined primarily for their uranium content. An orebody is defined as a mineral deposit from which the mineral may be recovered at a cost that is economically viable given the current market conditions. Where a deposit holds a significant concentration of two or more valuable minerals then the cost of recovering each individual mineral is reduced as certain mining and treatment requirements can be shared. In this case, lower concentrations of uranium than usual can be recovered at a competitive cost.

Generally speaking, uranium mining is no different from other kinds of mining unless the ore is very high grade. In this case special mining techniques such as increased dust suppression, and in extreme cases remote handling techniques, are employed to limit worker radiation exposure and to ensure the safety of the environment and general public.

Searching for uranium is in some ways easier than for other mineral resources because the radiation signature of uranium's decay products allows deposits to be identified and mapped from the air. See above section on Exploration.

Open pit and underground mining

Where orebodies lie close to the surface, they are usually accessed by open cut mining, involving a large pit and the removal of much overburden (overlying rock) as well as a lot of waste rock. Where orebodies are deeper, underground mining is usually employed, involving construction of access shafts and tunnels but with less waste rock removed and less environmental impact. In either case, grade control is usually achieved by measuring radioactivity as a surrogate for uranium concentration. (The radio-metric device detects associated radioactive minerals which are decay products of the uranium, rather than the uranium itself—see later footnote.)

At Ranger in north Australia, Rössing and Husab in Namibia, and most of Canada's Northern Saskatchewan mines through to McClean Lake, the orebodies have been accessed by open cut mining. Other mines such as Olympic Dam in Australia, McArthur River, Rabbit Lake and Cigar Lake in Northern Saskatchewan, and Akouta in Niger are underground, up to 600 m deep. At McClean Lake and Ranger, mining was planned to be completed underground.

In situ leach (ISL) mining

Some orebodies lie in groundwater in porous unconsolidated material (such as gravel or sand) and may be accessed simply by dissolving the uranium and pumping it out—this is in situ leach (ISL) mining (also known in North America as in situ recovery (ISR)). It can be applied where the aquifer in which the orebody lies is confined vertically and ideally horizontally. Buried paleochannels (ancient river channels) are typical. It is not licensed where potable water supplies may be threatened.

ISL mining means that removal of the uranium minerals is accomplished without any major ground disturbance, so it is certainly the mining method with least environmental impact. Groundwater from the aquifer with a lot of oxygen added is circulated back through the enclosed underground aquifer which holds the uranium ore in loose sands. This leaching solution is weakly acidified or alkaline and it dissolves the uranium before being pumped to the surface treatment plant where the uranium is recovered as a precipitate. Most US and Kazakh uranium production is by this method.

In Australian ISL mines the oxidant used is hydrogen peroxide and the complexing agent sulfuric acid to give a uranyl sulfate in the pregnant liquor. Kazakh ISL mines generally do not employ an oxidant but use much higher acid concentrations in the circulating solutions. ISL mines in the United States use an alkali leach to give a uranyl carbonate due to the presence of significant quantities of acid-consuming minerals such as gypsum and limestone in the host aquifers. Any more than a few percent carbonate minerals means that alkali leach must be used in preference to the more efficient acid leach, though the cost is often double.

In either the acid or alkali leaching method the fortified groundwater is pumped into the aquifer via a series of injection wells where it slowly migrates through the aquifer, leaching the uranium bearing host sand on its way to strategically placed extraction wells, where submersible pumps bring the liquid to the surface for processing.

For very small orebodies amenable to ISL mining, a small satellite plant can be set up. This does no more than provide a facility to load the ion exchange (IX) resin/polymer so that it can be trucked to the central plant some distance away in a bulk trailer for stripping. Hence very small deposits can become viable, since apart from the wellfield, little capital expenditure is required at the mine and remote IX site.

Heap leaching

Some ore, usually very low-grade (below 0.1% U), may be treated by heap leaching. Here the broken ore is stacked about 5–30 m high on an impermeable pad and irrigated with acid (or sometimes alkaline) solution over many weeks. The pregnant liquor from this is collected and treated to recover the uranium, as with ISL, usually using ion exchange. After the material ceases to yield significant further uranium, it is removed and replaced with fresh ore. Recoveries are typically 50–80%. The depleted material has the potential to cause pollution so must be emplaced securely so as not to affect surface water or groundwater. Usually this will be in mined-out pits.

Milling and processing

Conventional mines have a mill where the ore is crushed and ground to liberate the mineral particles, then leached in tanks with sulfuric acid to dissolve the uranium oxides. The solution is then processed to recover the uranium. With some South African uranium recovery from gold tailings, a pressure leach is necessary.

Sometimes a physical beneficiation process is used to concentrate the ore and increase the head grade before chemical treatment. This may be radiometric sorting as at Ranger, screening/gravity, or a new process called ablation.

Most of the ore is barren rock or other minerals which remain undissolved in the leaching process. These solids or “tailings” are separated from the uranium-rich solution, usually by allowing them to settle out. The remaining solution is filtered and the uranium is recovered in some form of ion exchange (IX) or solvent extraction (SX) system. The pregnant liquor from ISL or heap leaching is treated similarly. The uranium is then stripped from this and precipitated. The final chemical precipitate is filtered and dried.

Peroxide products can be dried at ambient temperatures to produce a product containing about 80% U_3O_8 . Ammonium or sodium diuranate ($(\text{NH}_4)_2\text{U}_2\text{O}_7$ or $\text{Na}_2\text{U}_2\text{O}_7$) products are dried at high temperatures to convert the product to uranium oxide concentrate— U_3O_8 —about 85% uranium by mass. This is sometimes referred to as yellowcake, though it is usually khaki.

In the case of carbonate leaching the uranyl carbonate (UO_2CO_3) can be precipitated with an alkali, e.g. as sodium or magnesium diuranate ($\text{Na}_2\text{U}_2\text{O}_7$ or MgU_2O_7).

The product is then packed into 200-L steel drums which are sealed for shipment. The U_3O_8 is only mildly radioactive (the radiation level 1 m from a drum of freshly-processed U_3O_8 is about half that—from cosmic rays—on a commercial jet flight). In ISL mills the process of uranium recovery is very similar, without the need for crushing and grinding.

Tailings management and mine rehabilitation

From open cut mining, there are substantial volumes of barren rock and overburden waste. These are placed near the pit and either used in rehabilitation or shaped and revegetated where they are.

Uranium minerals are always associated with more radioactive elements such as radium and radon in the ore which arise from the radioactive decay of uranium over a few million years. Therefore, although uranium itself is barely radioactive, the ore which is mined, especially if it is very high-grade such as in some Canadian mines, is handled with some care, for occupational health and safety reasons.

Mining methods, tailings and run-off management and land rehabilitation are subject to Government regulation and inspection. For instance in Australia the Code of Practice and Safety Guide: Radiation Protection and Radioactive Waste Management in Mining and Mineral Processing was published in 2005, and updated in 2015.

Solid waste products from the milling operation are tailings, ranging in character from a very fine material to coarse sands. They comprise most of the original ore and they contain most of the radioactivity in it.¹ In particular they contain all the radium present in the original ore. At an underground mine they may be first cycloned to separate the coarse fraction which is returned underground and used for underground fill. The balance is pumped as a slurry to a tailings dam, which may be a worked-out pit as at Ranger and McClean Lake, or an engineered structure.

When radium undergoes natural radioactive decay one of the products is radon gas. Because radon and its decay products (daughters) are radioactive and because the ground rock comprising the tailings is now on the surface, measures are taken to minimize the emission of radon gas. During the operational life of a mine the material in the tailings dam is often kept covered by water to reduce surface radioactivity and radon emission (though with lower-grade ores neither pose a hazard at these levels). This water needs to be recycled or evaporated since it contains radium, which is relatively soluble. Most mines adopt a “zero discharge” policy for any pollutants.

¹About 95% of the radioactivity in uranium ore is from the U-238 decay series, totaling about 150 kBq/kg in ore with 0.1% U_3O_8 (Shusterman, 2021). The U-238 series has 14 radioactive isotopes in secular equilibrium, thus each represents about 11 kBq/kg (irrespective of the mass proportion). When the ore is processed, the U-238 and the very much smaller masses of U-234 (and U-235) are removed. The balance becomes tailings, and at this point has about 85% of its original intrinsic radioactivity. However, with the removal of most U-238, the following two short-lived decay products in the uranium decay series (Th-234 and Pa-234) soon disappear, leaving the tailings with a little over 70% of the radioactivity of the original ore after several months. The controlling long-lived isotope then becomes Th-230 which decays with a half-life of 77,000 years to radium-226 followed by radon-222 (Supervising Scientist Group, Australia).

On completion of the mining operation, it is normal for the tailings dam to be covered by some two meters of clay and topsoil with enough rock to resist erosion. This is to reduce both gamma radiation levels and radon emanation rates to levels near those normally experienced in the region of the orebody, and for a vegetation cover to be established. Where possible, tailings are eventually returned to the mine pit or underground. In Canada, ore treatment is often remote from the mine that the new ore comes from, and tailings are emplaced in mined-out pits where possible, and engineered dams otherwise.

At established ISL operations, after mining is completed the quality of the remaining groundwater must be restored to a baseline standard determined before the start of the operation so that any prior uses may be resumed. Usually this is potable water or stock water (usually less than 500 ppm total dissolved solids). Contaminated water drawn from the aquifer is either evaporated or treated before reinjection.

In contrast to the main US operations, the water quality at the Australian ISL sites is very poor to start with, and it is quite unusable. At Beverley the original groundwater in the orebody is fairly saline and orders of magnitude too high in radionuclides for any permitted use. At Honeymoon the original water is even more saline, and high in sulfates and radium. When oxygen input and leaching are discontinued, the water quality reverts to its original condition over time.

Upon decommissioning, ISL wells are sealed or capped, process facilities removed, any evaporation pond revegetated, and the land can readily revert to its previous uses.

Mining is generally considered a temporary land use, and upon completion the area with any waste rock, overburden, and covered tailings needs to be left fit for other uses, or its original use. In many parts of the world governments hold bonds to ensure proper rehabilitation in the event of corporate insolvency.

Secondary sources of uranium

Most uranium used for nuclear power comes directly from mines, via the fuel cycle facilities which prepare it for use. Some, however, comes from what are known as secondary sources.

The most obvious secondary source is *civil stockpiles* held by utilities and governments. The amount held here is difficult to quantify, due to commercial confidentiality. At the end of 2018 some 279,000 tU total inventory was estimated for utilities—USA 43,000 t, EU 45,000 t, China 120,000 t, other East Asia and India 71,000 t (World Nuclear Association, 2019). These reserves are expected to be maintained at a fairly high level to provide energy security for utilities and governments.

Military warheads have been an important source of nuclear fuel since 1987. The United States and countries of the former USSR signed a series of disarmament treaties to reduce the nuclear arsenals of the signatory countries by approximately 80%, which enabled repurposing of their fissile inventory.

The weapons contained a great deal of uranium enriched to over 90% U-235 (i.e. up to 25 times the proportion in reactor fuel). For two decades to 2013, up to 10% of electricity produced in the United States was generated from fuel fabricated using uranium that had been converted from bomb-grade material under the so-called “Megatons to Megawatts” programme. Under the programme, enough bomb-grade material for 20,000 nuclear warheads was eliminated.

Worldwide, until 2013, the conversion of military high-enriched uranium was providing about 15% of the world’s reactor requirements.

For more information, see World Nuclear Association (<https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/uranium-resources/military-warheads-as-a-source-of-nuclear-fuel.aspx>).

Recycled uranium and plutonium from reprocessing used fuel is another source, and currently saves about 2000 tU per year of primary supply, depending on whether just the plutonium or also the uranium is considered. In fact, plutonium is recycled as MOX fuel, whereas the reprocessed uranium (RepU) is mostly stockpiled, but some 16,000 t of RepU from Magnox reactors in United Kingdom has been used to make about 1650 t of enriched AGR fuel. In Belgium, France, Germany and Switzerland over 8000 t of RepU has been recycled into nuclear power plants.

For more information, see World Nuclear Association (<https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/fuel-recycling/processing-of-used-nuclear-fuel.aspx>).

Re-enrichment of depleted uranium (DU, enrichment tails) is another secondary source. There is over 1.6 million tonnes of depleted uranium available, from both military and civil enrichment activity since the 1940s, most at tails assay of 0.25–0.35% U-235. Non-nuclear uses of DU are very minor relative to annual accumulation of over 50,000 tU per year. This leaves most DU available for mixing with recycled plutonium for MOX fuel or as a future fuel resource for fast neutron reactors.

However, some DU that has relatively high assay can be fed through under-utilized enrichment plants to produce natural uranium equivalent, or even enriched uranium ready for fuel fabrication. Russian enrichment plants have treated 10–15,000 t per year of DU assaying over 0.3% U-235, stripping it down to 0.1% and producing a few thousand tonnes per year of natural uranium equivalent. This Russian programme treating Western tails has now finished, but a new US one is expected to start when surplus capacity is available, treating about 140,000 t of old DU assaying 0.4% U-235.

Underfeeding at enrichment plants is a significant source of secondary supply, especially since the Fukushima accident reduced enrichment demand for several years. This is where the operational tails assay is lower than the contracted/transactional assay, and the enricher sets aside some surplus natural uranium, which it is free to sell (either as natural uranium or as enriched uranium product) on its own account. This source will provide an estimated 3500–7000 tU/year to the mid-2020s.

International fuel reserves

There have been three major initiatives to set up international reserves of enriched fuel, two of them multilateral ones, with fuel to be available under International Atomic Energy Agency (IAEA) auspices despite any political interruptions which might affect countries needing them. The third is under US auspices, and also to meet needs arising from supply disruptions. All have low-enriched uranium (LEU), typically 3.5–5% U-235.

IAEA LEU fuel bank

In December 2010 the IAEA board resolved to establish a guaranteed reserve of low-enriched uranium, the IAEA LEU Bank. It comprises a physical stock of UF₆ owned by the IAEA, is responsible for storing and protecting it. According to international norms, such a “fuel bank” must be located in a country with no nuclear weapons and be fully open to IAEA inspectors. It is located at the Ulba Metallurgical Plant in Kazakhstan.

The fuel bank hosts a potential supply of 92 t LEU (as UF₆) nominally at 4.95% enrichment for the production of fuel assemblies for nuclear power plants. The Kazakh government in April 2015 approved a draft agreement with the IAEA for this and the IAEA board then approved plans for the IAEA LEU Bank to be located at the Ulba Metallurgical Plant (UMP) at Ust-Kamenogorsk (aka Oskemen) and operated by Kazakhstan. A transit agreement with Russia for shipping LEU was also approved. An agreement between the IAEA and UMP was signed in May 2016 and the facility was formally opened at the end of August 2017. It became operational in 2019, and it awarded contracts to Orano and Kazatomprom to supply it.

Russian LEU reserve

In November 2009 the IAEA Board approved a Russian proposal to create an international “fuel bank” or guaranteed reserve of low-enriched uranium under IAEA control at the International Uranium Enrichment Centre at Angarsk. This Russian LEU reserve was established a year later and comprises 123 t of low-enriched uranium as UF₆, enriched 2.0–4.95% U-235 (with 40 t of latter), available to any IAEA member state in good standing which is unable to procure fuel for political reasons. It is fully funded by Russia, held under safeguards, and the fuel will be made available to IAEA at market rates, using a formula based on spot prices. Following an IAEA decision to allocate some of it, Rosatom will transport material to St Petersburg and transfer title to IAEA, which will then transfer ownership to the recipient. The 120 t uranium as UF₆ is equivalent to two full fuel loads for a typical 1000 MWe reactor.

American assured fuel supply

In 2005 the US government announced plans for the establishment of a mechanism to ensure fuel supply for use in commercial reactors in foreign countries where there has been supply disruption. The fuel would come from downblending 17.4 t of high-enriched uranium (HEU) to LEU. In August 2011 US Department of Energy announced an expanded scope for the program so it would also serve US utility needs, and now be called the American Assured Fuel Supply (AFS). At that point most of the downblending of the HEU had been completed, and the scheme was ready to operate. The AFS comprises 286 t of LEU (with another 60 t from downblending being sold on the market to pay for the work). Additionally, the USA has completed the process of downblending another 20.1 t of HEU. The AFS program is administered by the US National Nuclear Safety Administration, any foreign access must be through a US entity, and the fuel will be sold at current market prices. The 286 t amount is equivalent to about six reloads for a 1000 MWe reactor.

Uranium markets

All mineral commodity markets tend to be cyclical, i.e., prices rise and fall substantially over the years, but with these fluctuations superimposed on long-term trend of decline in real prices, as technological progress reduces production cost at mines. In the uranium market, however, high prices in the late 1970s gave way to depressed prices in the whole of the period of the 1980s and 1990s, with spot prices below the cost of production for all but the lowest cost mines. Spot prices recovered from 2003 to 2009, but have been weak since then.

The quoted “spot prices” through to about 2007 applied only to marginal trading from day to day and represented a small portion of supply, though since 2008 the proportion has approximately doubled, to about one quarter in the last decade. Most trade is via 3–15 year term contracts with producers selling directly to utilities at a significantly higher price than the spot market, reflecting the security of supply.² The specified price in these contracts is, however, often related to the spot price at the time of delivery. However, as production has risen much faster than demand, less long-term contracts are being written.

In 2000, primary market participants - utilities and producers—accounted for 95% of the spot market. That share decreased to two-thirds by 2005 and one-third by 2011 and it has remained 30–40% since. The rest comes from the financial community, namely traders and financiers who have moved in on the market, bringing greater liquidity and efficiency.

The reasons for fluctuation in mineral prices relate to demand and perceptions of scarcity. The price cannot indefinitely stay below the cost of production, nor will it remain at very high levels for longer than it takes for new producers to enter the market and anxiety about supply to subside.

²In May 2020 the spot price was \$33.40 per pound U₃O₈, the 5-year forward price published by UxC (www.uxc.com) was \$39.75.

See Also: Origins and Resources of Uranium and Thorium.

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Uranium Extraction From Seawater Makes Nuclear Power Virtually Renewable

James L. Conca, UFA Ventures, Inc., Richland, WA, United States

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Abbreviations

AAPG UCOM American Association of Petroleum Geologists Uranium Committee

CNNC China National Nuclear Corporation

dpm Disintegrations per minute (measure of radioactivity)

DOE United States Department of Energy

EIA United States Energy Information Administration

MW Megawatt of electricity

ORNL Oak Ridge National Laboratory

PNNL Pacific Northwest National Laboratory

REE Rare earth element

SMORE Symbiotic Machine for Ocean uRanium Extraction

U Uranium

Introduction

America, Japan, and China are racing to be the first nation to make nuclear energy completely renewable. The hurdle is making it economic to extract uranium from seawater, because the amount of uranium in seawater is practically inexhaustible.

New technological breakthroughs from the U.S. Department of Energy's (DOE's) Pacific Northwest (PNNL) and Oak Ridge (ORNL) national laboratories have made removing uranium from seawater within reach and the main question is - when will seawater extraction become the favored source of uranium for our nuclear power plants?

The world's almost 400 reactors produce about 2.5 trillion kWhs per year, about 10% of the global generation, using about 100,000 tons of U each year to do so. Given that there is approximately 4 billion metric tons of uranium in seawater now, this would produce about 40 quadrillion kWhs of electricity if extracted completely.

If humanity's electricity use stabilizes at about 40 trillion kWhs per year, then this would provide all of the world's electricity needs for 1000 years if nuclear were 100% of our source and we used the presently predominant design (Light Water Reactors; LWR) that operate with the once-through fuel cycle (and which utilizes less than 1% of the energy value of the natural uranium). New reactor designs, particularly fast reactor designs that utilize more of the energy in the fuel, will extend even further into the future the contribution of uranium to energy supply.

Because uranium extracted from seawater is replenished continuously through geochemical interactions with the Earth's crust deposited in seawater mainly via river flow, the potential for uranium supply goes well beyond the 4 billion metric tons currently in the oceans. This article provides an overview of the potential impact of seawater extraction of uranium, the technology used for seawater extraction of uranium, and the economic considerations that will determine whether or when benefits of seawater uranium will be realized. Although the context for this article is the U.S. uranium market, the considerations presented should be applicable to other uranium market situations.

Present sources of uranium

In their 2020 Annual Report, the Uranium Committee of the Energy Minerals Division of the American Association of Petroleum Geologists noted that the prices and supplies of uranium should remain favorable to resource and supply development for the near-future.

The United States' 96 reactors produce 20% of the nation's electricity, and require about 25,000 tons of U each year to do so. In contrast, 624,000,000 tons of natural gas is used to generate 34% of our electricity, and 750,000,000 tons of coal for 30%. 1000,000,000 tons of petroleum (7,200,000,000 barrels) fuels our transportation sector, the energy equivalent of 1,500,000,000 tons of coal.

The U.S. Navy also operates more than 40 ships and submarines using (the original) small modular nuclear power plants.

Since the 1990s, most of the uranium used in the United States is from other countries like Canada and Australia. This is a good thing, as the uranium ores in these countries are much higher grade than in the United States, which requires a lot less mining and refining to get the same amount of clean uranium.

There have been numerous discoveries of new high-grade uranium deposits in Canada and new low-grade deposits reported to be under development in Argentina and Peru. The main Australian uranium mine in South Australia (Hore-Lacy, 2021) has resumed operations (Figs. 1 and 2).

The world produces about 63 million kg (140 million lbs) of uranium as U_3O_8 each year. According to the United States Energy Information Administration, the U.S. produced a total of 77,000 kgs (170,000 lbs) of U_3O_8 of uranium concentrate from all domestic sources in 2019, 89% less than in 2018, which itself was 33% less than in 2017. 2018 production was primarily from six facilities: five in-situ leach plants in Nebraska and Wyoming (Crow Butte Operation, Lost Creek Project, Ross CPP, North Butte, and Smith Ranch-Highland Operation) and one underground mine.

The United States developed deposits in Wyoming and New Mexico that were sources of U for weapons programs during the Cold War. But known deposits and some new discoveries have occurred in 13 states. A new U.S. uranium district has been identified in the eastern Seward Peninsula of Alaska on the eastern margins of McCarthy Basin. Thorium and rare-earth elements have been discovered in the surrounding igneous rocks. Even Virginia has a huge untapped U deposit.

On the other hand, China will be entering the U supply arena this year with their Husab mine in Namibia. This huge deposit is expected to add 6,800,000 kg (15,000,000 lbs) of U to the market each year once it reaches full production. However, mining costs at the Husab site are above the current U price of \$67/kg (\$30/lb); however, the world is in oversupply, so the economic viability of the Husab mine remains to be seen.

In addition to those developments, the U industry in the U.S. has been buffeted by the U.S. government itself, which has sold excess U (as yellowcake or UF_6 from weapons inventories) directly into the market. It's a great way to get rid of weapons U, but with impacts to producers. In response, the Uranium Producers of America recommended that the DOE should stop all sales when the U spot market price is below about \$70/kg (\$32/lb).

The bottom line is that there is a relative abundance of U in the world, and more is discovered continually. U resources increased about 25% over the last decade, which has kept prices low.

Many hard-rock uranium deposits also contain associated REEs to the extent that co-production of raw rare earth elements (REEs), thorium, and other critical metals are underway for stockpiling, awaiting shipment to processing sites around the world.

According to the American Association of Petroleum Geologists Uranium Committee (AAPG UCOM), we have more U than we need for hundreds of years of nuclear power supplying a big portion of our energy generation. In this author's opinion, this is critical, since we need to double nuclear power to address climate change and replace coal, even as we ramp up renewables.

Fortunately, nuclear technology has not been stagnant. Newer designs for nuclear reactors include fast reactors that can burn ^{238}U and other actinides, getting well over 10 times the energy from the same amount of fuel (Stanculescu, 2021; Wigeland et al., 2020). That will extend present U supplies for thousands of years by making use of spent fuel from old reactors and depleted uranium left from previous activities including enrichment and weapons production.

Extraction of uranium from seawater

Uranium is present in seawater dissolved at very low concentrations, only about 3 parts per billion (3 micrograms/L, $\mu g L^{-1}$, or 0.00000045 oz per gallon).

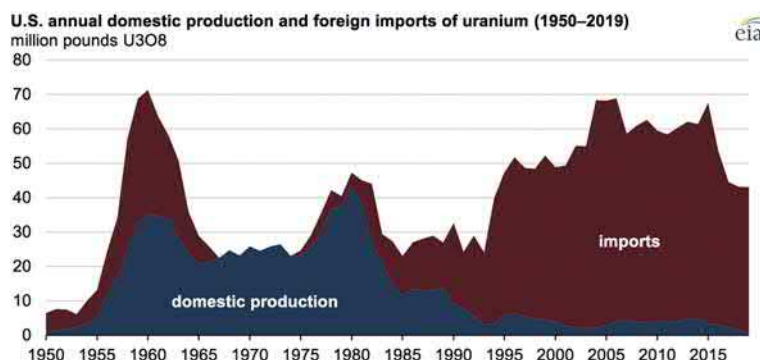


Fig. 1 U.S. Domestic Uranium Production versus Imports 1950 to 2019 (US EIA, 2020b).

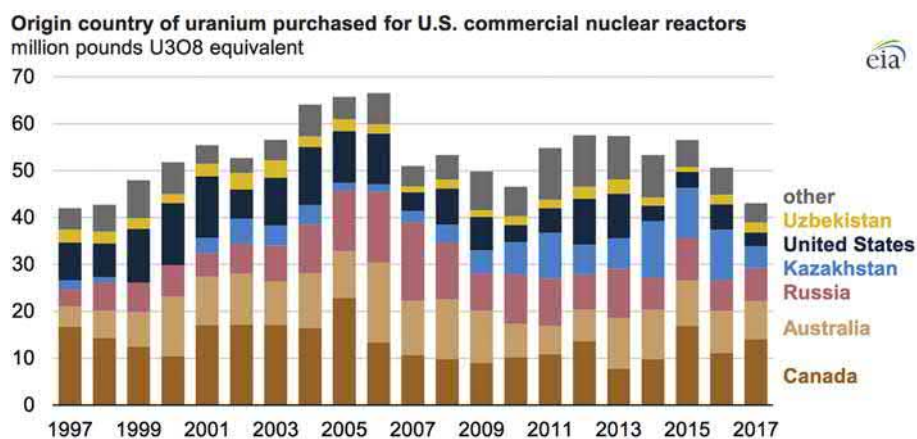


Fig. 2 Country Origins of U.S. Uranium for Nuclear Fuel. Note the small input from the United States (US EIA, 2020a).

At present, the approximately 350 quintillion gallons (1.4 sextillion liters) of seawater on Earth contains about 4 billion metric tons of uranium at any one time. The primary dissolved species are various uranyl carbonate complexes of U(VI), although water in confined basins such as the Black Sea generally have more in solution as reduced U(IV). The isotopic ratio of $^{235}\text{U}/^{238}\text{U}$ in seawater is the same as in the crust, about 0.007.

According to Ku et al. (1977), "Current estimations of the average world river uranium concentrations ($0.3\text{--}0.6\ \mu\text{g L}^{-1}$) and $^{234}\text{U}/^{238}\text{U}$ [radioactivity] ratios (1.2–1.3) and of the diffusional ^{234}U influx from sediments ($0.3\ \text{dpm cm}^{-1}\ 10^{-3}\ \text{dpm yr}^{-1}$) [used to estimate diffusion of ^{238}U] are essentially consistent with a model which depicts a steady state distribution of uranium in the ocean."

Passive adsorption and electrochemical processes are presently the two different methods to extract elements and minerals directly from seawater. Seawater contains large amounts of valuable minerals including 17 REEs, precious metals, lithium, and uranium.

Whereas land-based minerals are concentrated in specific geologic formations, seawater minerals are distributed evenly in seawater, although higher concentrations can occur near continents as a result of runoff from the crustal rocks from which these elements come. Extracting minerals from seawater is more environmentally-friendly than mining on land as it does not require fresh water or create volumes of contaminated water and mining tailings.

The most economical methods use passive adsorption technology, thereby avoiding the energy needed to process or pump large volumes of seawater. In a passive extraction system, the natural ocean currents deliver seawater to the adsorbent for extraction of the elements of interest. Passive systems are envisioned as farms resembling a kelp forest that are planted and harvested like crops (Fig. 3). Fig. 4 shows the retrieval of one harvested acrylic fiber sorbent from an ocean demonstration.

The latest technologies for extracting U from seawater build on work by researchers in Japan and use polyethylene fibers coated with various amine compounds (Fig. 5), such as amidoxime, to pull in and bind uranium dioxide from seawater (Fig. 6). In seawater, amidoxime attracts and binds uranium dioxide to the surface of the fiber braids, which can be on the order of 15 cm in diameter and run multiple meters in length depending on where they are deployed.

After a month or so in seawater, the length of braided fibers are remotely released to the surface and collected. An acid treatment recovers the uranium in the form of a uranyl complex, regenerating the fibers that can be reused many times. The concentrated uranyl complex then can be enriched to become nuclear fuel.

Another method has been researched by Haji and co-workers. Haji et al. (2017a,b) built on the previous systems described by Picard et al. (2014) and Haji and Slocum (2016) to design a mechanical exposure system they call Symbiotic Machine for Ocean uRanium Extraction (SMORE) that uses adsorbent shells that are incrementally spaced along a continuous moving roller chain that washes the sorbent with fresh seawater (Fig. 7). The energy needed to operate these systems depends on the type of material and approaches used. Absorbents can be electrochemically enhanced for better sorption and desorption, or just mechanically driven by belts or roller chains into and out of the seawater and into and out of extraction baths.

This procedure, along with the global effort, involves synthesizing and characterizing uranium adsorbents and marine testing of these adsorbents at facilities like PNNL's Marine Sciences Laboratory in Sequim, Washington (Alexandratos and Kung, 2016; LiVecchi et al., 2019). In addition to marine testing, PNNL assessed how well the adsorbent attracted uranium versus other elements, how durable the adsorbent was, how buildup of marine organisms might impact performance, and which adsorbent materials are not toxic (Gill et al., 2016).

This marine testing shows that these new fibers have the capacity to hold at least 6 g of uranium per kilogram of adsorbent after only about 50 days in natural seawater. A video of U extraction from seawater can be seen on the University of Tennessee Knoxville website.

The first uranium yellowcake extracted from seawater by the United States occurred in 2018 during 3 month-long tests at the Sequim facility. It used inexpensive sorption-coated acrylic fibers developed by an Idaho-based clean energy company LCW

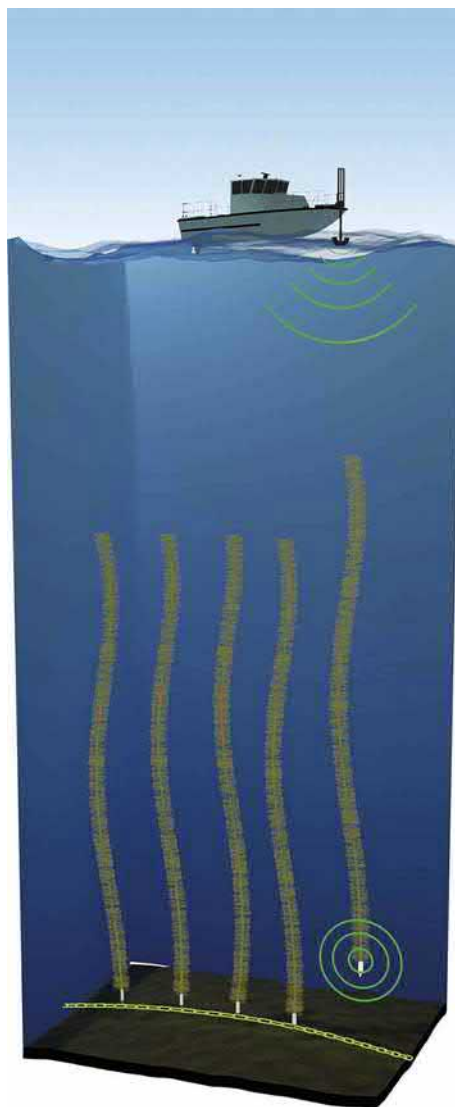


Fig. 3 Scientists envision anchoring hundreds of lengths of U-extracting fibers in the sea for a month or so until they fill with uranium. Then a wireless signal would release them to float to the surface where the uranium could be recovered and the fibers reused. Fiber length can be up to 60 m, a chain over a kilometer-long set on the sea floor up to 200 m below the surface with at least 40 m of seawater above the top of the fibers. It doesn't matter where in the world the fibers are floating. Courtesy Oak Ridge National Laboratory.

Supercritical Technologies with early support from PNNL through DOE's Office of Nuclear Energy. Seawater was pumped through about a kilogram of fiber and the uranium was then extracted, producing about 5 g of uranium. Calculations indicate this method would be cost-compatible with mining if scaled up.

Of course, other elements can be extracted from seawater as well, either by the same process or others acting simultaneously. **Fig. 8** shows the relative abundance of elements extracted by the amidoxime-based sorbents from ORNL.

When nuclear becomes renewable

Because uranium extracted from seawater is replenished continuously from the Earth's crust in a thermodynamically-controlled steady-state ([Ku et al., 1977](#)), it will be theoretically impossible to drop the amount of U in seawater significantly even as you're extracting it. Therefore, nuclear energy effectively can become as endless as solar and wind.

However, seawater concentrations of uranium are controlled by steady-state, or pseudo-equilibrium, chemical reactions between waters and rocks on the Earth, where it is transported by rivers to the ocean. Although mixing rates and equilibria reaction rates are generally unknown, extrapolations from how other elements are replenished by dissolution of crustal rocks suggest they are fast enough to keep seawater U concentrations about the same no matter our rate of removal ([Ku et al., 1977](#)).



Fig. 4 Uranium acrylic fiber sorbent deployment and retrieval by Texas A&M University at Galveston researchers. Courtesy Chen Xu.



Fig. 5 Researchers around the world have been working frantically to develop an array of materials and fibers able to economically extract uranium from seawater. They have succeeded, as discussed at a conference devoted to the topic. Courtesy Oak Ridge National Laboratory.

Crustal rocks contain 100 trillion tons of uranium ([LiVecchi et al., 2019](#)). Whenever uranium is extracted from seawater, more is leached from crustal rocks to replace it. So it is unreasonable to think that humans could extract enough uranium to significantly lower the overall seawater concentrations of uranium over the next billion years, even if nuclear provided 100% of our energy and our species lasted a billion years.

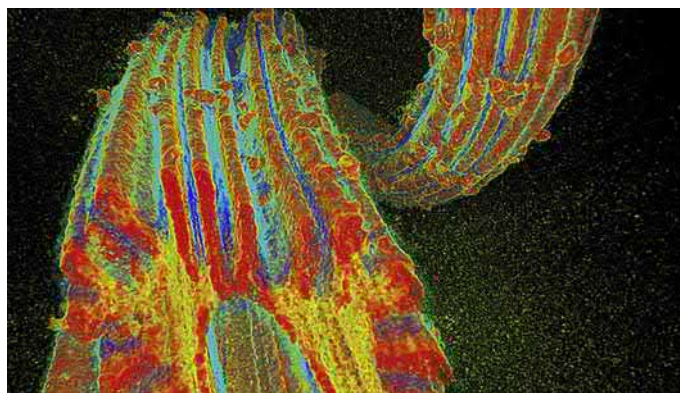


Fig. 6 Researchers at the Pacific Northwest National Laboratory exposed this special uranium-sorbing fiber developed at ORNL to *Pseudomonas fluorescens* and used the Advanced Photon Source at Argonne National Laboratory to create a 3-D X-ray microtomograph to determine microstructure and the effects of interactions with organisms and seawater. Courtesy Pacific Northwest National Laboratory.

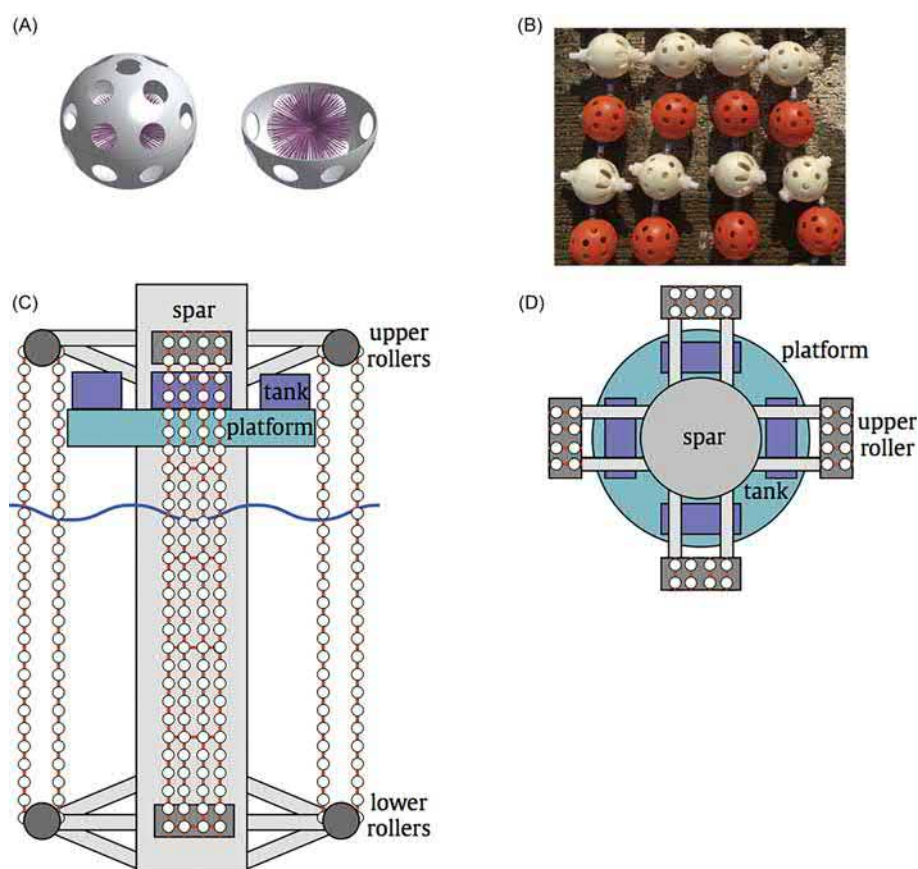


Fig. 7 Design of the Symbiotic Machine for Ocean uRanium Extraction (SMORE). (A and B) Hard permeable shell enclosure encapsulating the polymer adsorbent. (C and D) Side and top views of a SMORE 1:10 physical scale adsorbent ball-chain ladder which uses rollers to move the ball-chain lengths of adsorbent through the water column (Haji et al., 2017a). Courtesy American Nuclear Society.

In other words, uranium in seawater is renewable. Possibly as renewable as solar energy. Yes, uranium in the crust is, strictly speaking, finite. But so is the Sun, which will eventually burn out. But that won't begin to happen for another 5 billion years. Even the wind on Earth will stop at about that time as our atmosphere boils off during the Sun's initial death throes as a Red Giant.

Cost

Anecdotally, observers note that alternatives to the current means of mining uranium ore would help, and are possibly necessary for, the future of nuclear energy. Advances by PNNL and ORNL have reduced the cost of seawater uranium extraction

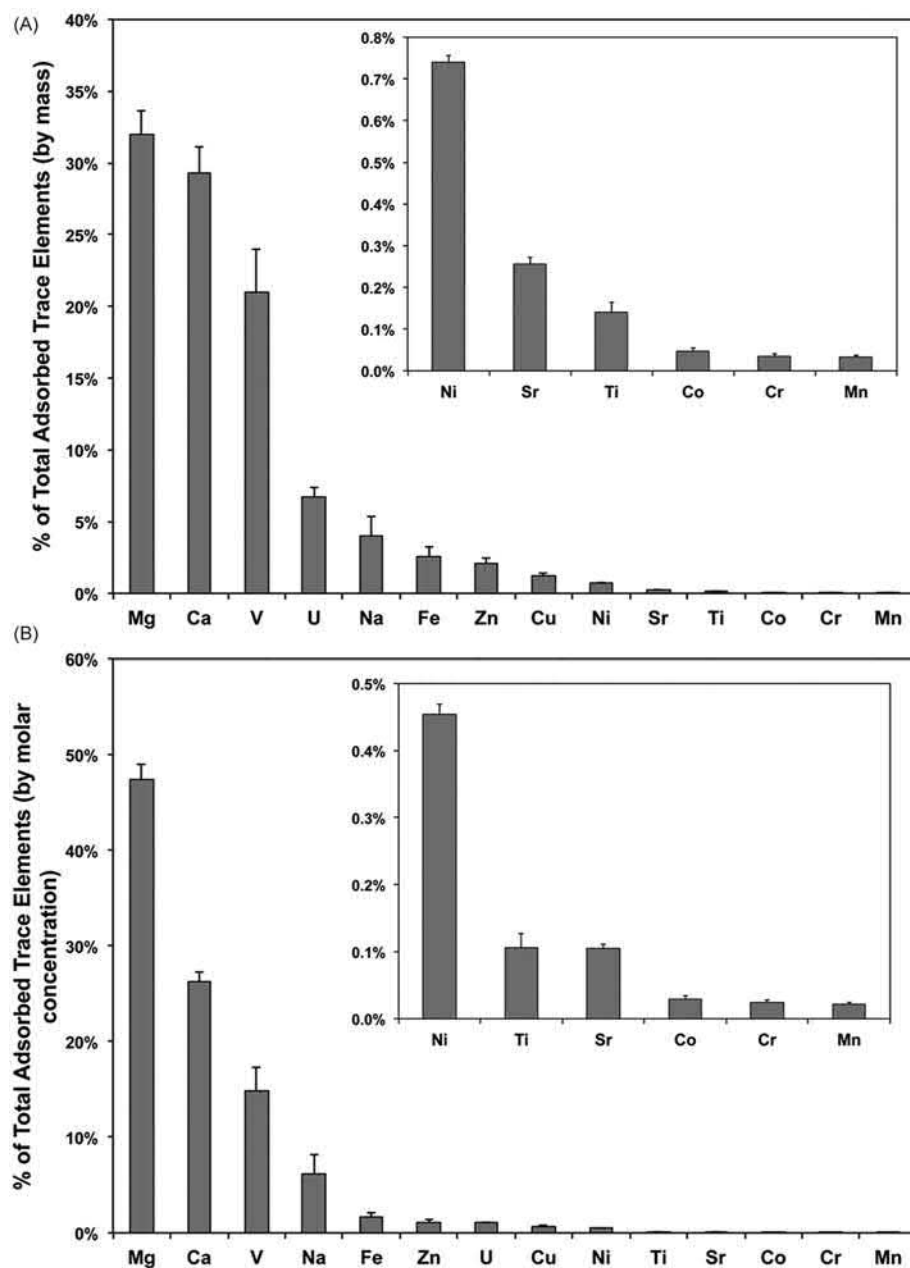


Fig. 8 Relative abundance of elements absorbed by the ORNL AF1 adsorbent after 56-day seawater exposure in a column flow-through system: (A) by mass percentage; (B) by molar percentage. The insets are the expanded view for Ni, Ti, Sr, Co, Cr, and Mn. Courtesy U.S. Pacific Northwest Laboratory.

by a factor of four in just 5 years. But at more than \$200/lb. of U_3O_8 , the cost remains twice as much as needed to replace mining uranium ore.

The cost of uranium is a small percentage of the cost of nuclear fuel, which is itself a small percentage of the cost of nuclear power. Over the last 20 years, uranium spot prices have varied between \$22 and \$267/kg (\$10 and \$120/lb) of U_3O_8 (Fig. 9), mainly from changes in the availability of weapons-grade uranium to blend down to make reactor fuel, followed by the shutting of Japan's nuclear fleet after Fukushima, and now from changes in energy use during the coronavirus pandemic (Fig. 10).

So, as the cost of extracting U from seawater falls to below \$225/kg (\$100/lb), it will become a commercially viable alternative to mining new uranium ore. But even at \$450/kg (\$200/lb) of U_3O_8 , the addition to the cost of nuclear power is no more than a fraction of a cent per kWh (World Nuclear Association, 2008). Research by Dr. Haji at Cornell indicates that the economics are even better if the mineral harvester shares its infrastructure with another offshore system such as a wind turbine or oil platform.

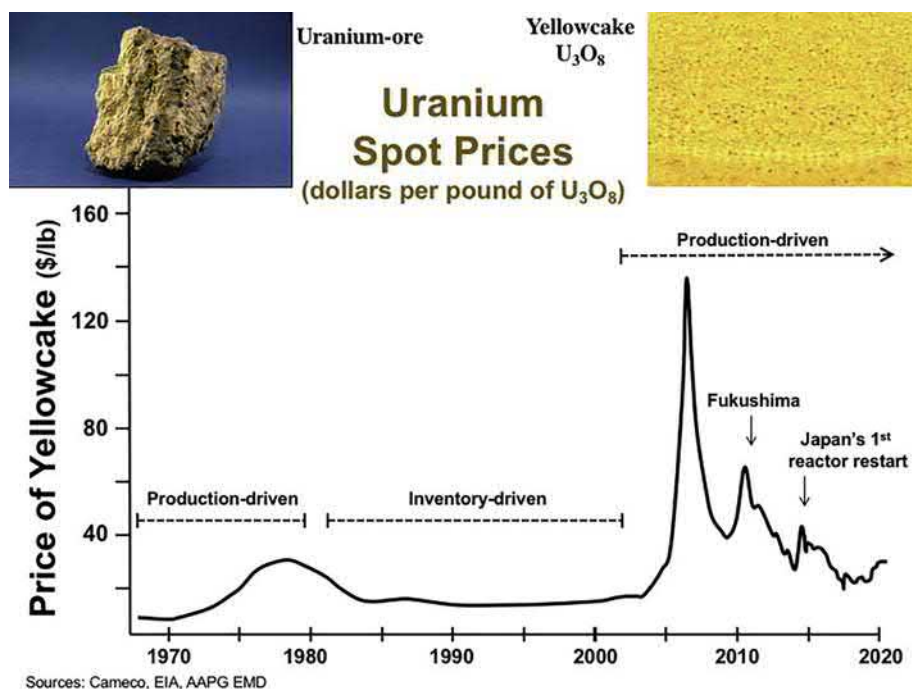


Fig. 9 Historical Spot Price of Uranium from 1988 to 2019 in US\$/lb. of U_3O_8 . As of late April, 2019, the price remains between \$25.00 to \$30.00 as a result of long-term uranium oversupply. But Japanese re-starts with Chinese and other new reactor start-ups will diminish the oversupply and catalyze a rise in uranium spot prices over the next decade. Before 2006, spot prices remained very low as cheap weapons-grade U was blended down to make commercial fuel. Spot prices are per pound of yellowcake (upper right) that is processed from uranium ore rock (upper left).

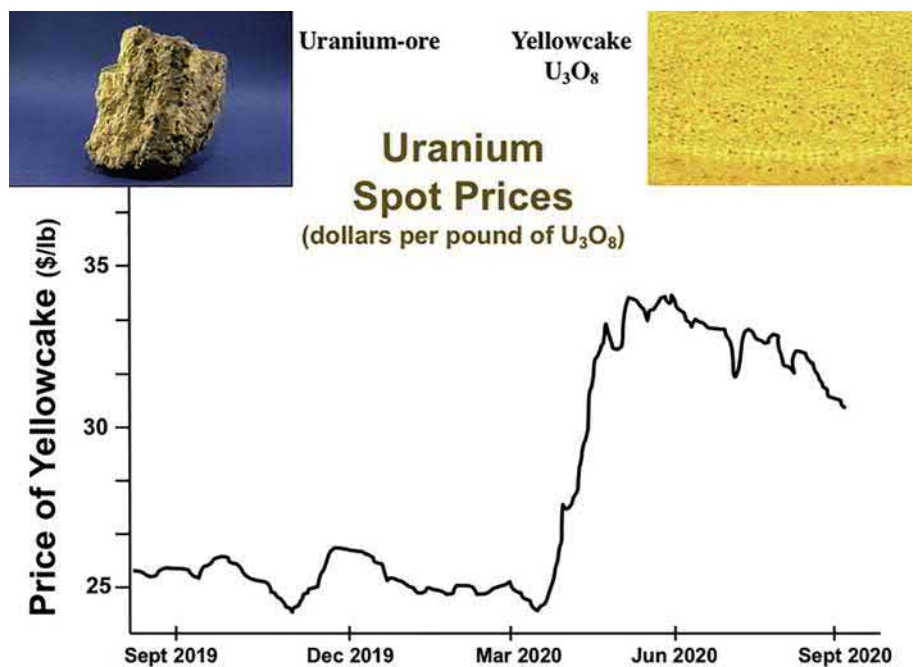


Fig. 10 Uranium spot prices over the last year, noting the big surge in price after the pandemic began, from about \$25/lb to \$34/lb. These are still quite low compared to the over \$100/lb prices from about 10 years ago after cheap weapons stocks of U had been exhausted. Spot prices are per pound of yellowcake (upper right) that is processed from uranium ore rock (upper left).

Conclusion

Extraction of uranium from seawater has become viable technically and should become viable economically within a relatively short time. The leading methods use passive adsorption technology, where the natural ocean currents deliver seawater to an adsorbent, particularly amidoxime-coated polyethylene fibers. Passive systems are envisioned as farms resembling a kelp forest that are planted and harvested like crops.

Importantly, nuclear fuel made with uranium extracted from seawater makes nuclear power completely renewable. According to Professor Jason Donev from the University of Calgary, “Renewable literally means “to make new again”” (Donev, 2020). Any resource that naturally replenishes with time, like the creation of wind or the growth of biological organisms for biofuels, is certainly renewable. Renewable energy means that the energy humans extract from nature will generally replace itself. And now uranium as fuel meets this definition.”

Nothing is renewable forever. All humans need are sources that are *practically* renewable for as long as humans exist.

So, for all practical purposes, solar, wind, hydro and nuclear are all renewable.

See Also: Environmental Protection: The Oceans—Formation and Global Climate Change; Environmental Protection: The Oceans—Implications of Manmade Radiation.

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Unconventional Uranium Resources From Phosphates

Nils H. Haneklaus, Danube University Krems, Krems an der Donau, Austria; and Freiberg University of Mining and Technology, Freiberg, Germany

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Introduction

Phosphate rock or phosphorites are minerals that contain phosphorus (P), measured by its phosphorus pentoxide (P_2O_5) content at elevated concentrations. In phosphate ore the concentrations of P_2O_5 usually range from 5–25%. To be marketed the P_2O_5 is concentrated to roughly 30%. Simple screening and more elaborate techniques such as flotation and calcination are used for the concentration or beneficiation of P_2O_5 . P is an essential building block for plants and animals. To feed the world approximately 250 million t of phosphate rock are mined annually. This makes phosphate rock the fourth most mined material on earth. The majority (>90%) of the mined phosphate rock is used for mineral fertilizer production. As of now there is no substitute for P in fertilizer production and large-scale recycling of P is still in its infancy. Phosphate rock is thus directly tied to food security.

Phosphate rock can contain a number of accompanying valuable elements in elevated concentrations that could justify commercial recovery (depending on specific market conditions and production costs). Most notably, rare earth elements (REE) and uranium could be industrially recovered (Chen and Graedel, 2015). Uranium can be found in some phosphate rock in relevant average concentrations of 70–200 mg/kg that could justify recovering it. Uranium is a relatively common element in the earth crust appearing with an average concentration of 2.8 mg/kg. In comparison, the average concentration of uranium in seawater that is, as a result of the tremendous overall quantities of uranium (approximately 4 billion t), often considered as a future uranium resource is as low as 0.003 mg/kg or 3 parts per billion. Uranium resources can be differentiated by concentration (WNA, 2020a,b) into very high-grade ores, high-grade ores, low-grade ores and very low-grade ores as indicated in Table 1.

Average uranium concentrations in phosphate rock deposits in Algeria (Djebel Kouif, 100 mg/kg), Angola (Cabinda, 260 mg/kg), Brazil (Araxa, 182 mg/kg and Catalao, 220 mg/kg), Burkina Faso (Kodjari, 125 mg/kg), Egypt (Hamrawein, 110 mg/kg, Safaga, 120 mg/kg, and West Mahamid, 100 mg/kg), Israel (Arad, 150 mg/kg), Mali (Tilemsi, 123 mg/kg), Morocco (undifferentiated, 130 mg/kg), Tanzania (Minjingu, 390 mg/kg), United States (Central Florida, 141 mg/kg and Idaho, 107 mg/kg) as reported by Van Kauwenbergh (1997) could thus also be classified as very low-grade uranium ores.

Ironically, several phosphate rock producers mine ores with natural uranium concentrations exceeding those of commercial uranium mines. Uranium from phosphate is considered an unconventional resource, since mining phosphate rock for its uranium content alone or recovering uranium as a by-product is presently considered uneconomic. Should uranium prices increase, or technologies allow for more cost-effective recovery, uranium from phosphates could become a conventional uranium resource again.

It is noteworthy that radiotoxic uranium from phosphate rock, not recovered during phosphate fertilizer production will largely transfer to the final fertilizer product and ultimately be distributed on agricultural soils. There is an active scientific discussion regarding the potential harm of uranium brought out with mineral fertilizers on agricultural soils (Bigalke et al., 2017; Liesch et al., 2015; Schnug and Haneklaus, 2015). While there is presently no legal limit for uranium in fertilizers, the German Commission for the Protection of Soils for instance, proposed setting it to 50 mg uranium per kg P_2O_5 (Kratz et al., 2016) or 167 mg uranium per kg fertilizer with a P_2O_5 content of wt30%.

Past uranium recovery from phosphates

Phosphate rock is present as apatite, most commonly fluorapatite ($Ca_5(PO_4)_3F$ or $Ca_{10}(PO_4)_6(F,OH)_2$), that is soluble on only very acidic soils as they are found in some tropical areas. To make P accessible to plants the majority of phosphate rock is processed to soluble mineral fertilizers. The most common process for this is the wet phosphoric acid (WPA) process. During WPA production the concentrated ore is digested with either hydrochloric acid (HCl) (Eq. 1), nitric acid (HNO_3) (Eq. 2) or sulfuric acid ($H_2SO_4 \cdot nH_2O$) (Eq. 3).

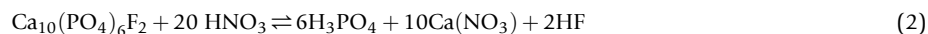
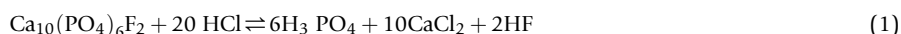
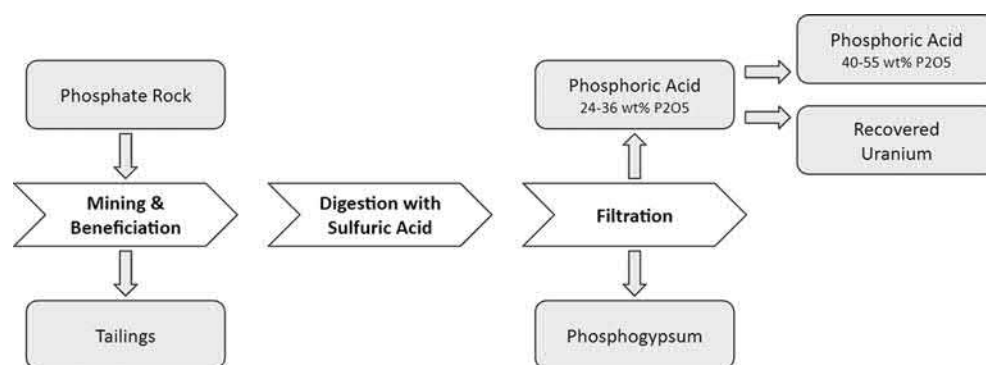
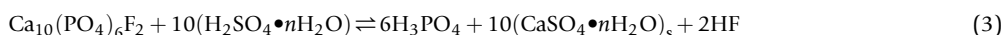


Table 1 Uranium resources by concentration.

Classification	Uranium concentration	
Very high-grade ore	20%	200,000 mg/kg
High-grade ore	2%	20,000 mg/kg
Low-grade ore	0.1%	1000 mg/kg
Very low-grade ore	0.01%	100 mg/kg

WNA (World Nuclear Association) (2020a) *Supply of Uranium*. <https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/uranium-resources/supply-of-uranium.aspx>.

**Fig. 1** Schematic overview of the wet phosphoric acid (WPA) process with uranium recovery.

Phosphate rock digestion with sulfuric acid is the most common process, used in about 75% of the roughly 400 fertilizer plants operating worldwide. The use of sulfuric acid is attractive from an industrial viewpoint as calcium is simultaneously precipitated as solid calcium sulfate known as phosphogypsum ($\text{CaSO}_4 \cdot n\text{H}_2\text{O}$) that can be separated using simple filtration. The use of hydrochloric acid and nitric acid do in contrast form water-soluble calcium nitrate ($\text{Ca}(\text{NO}_3)_2$) or calcium chloride (CaCl_2) that require more costly purification techniques. Besides, the majority (80–90%) of the radiotoxic radium transfers to the phosphogypsum replacing Ca in the chemical structure if sulfuric acid is used instead of being liberated as is the case with hydrochloric- and nitric acid digestion. Depending on the crystallization temperature, gypsum dihydrate ($\text{CaSO}_4 \cdot 2\text{H}_2\text{O}$), hemihydrate gypsum ($\text{CaSO}_4 \cdot 0.5\text{H}_2\text{O}$) and gypsum anhydrite (CaSO_4) that are usually all referred to as phosphogypsum are produced. For every 1 t phosphoric acid approximately 5 t phosphogypsum (dry weight) are produced. As a result of its low radioactivity most of the produced phosphogypsum (100–280 million t/year) is stacked next to fertilizer plants and not further processed (Saadaoui et al., 2017; Tayibi et al., 2009). Even the relatively low radioactivity concentrations (0.2–3 Bq/g for ^{226}Ra) present in most phosphogypsum does not permit further using it under most national regulations today. Recent successful case studies on the use of phosphogypsum as construction material and soil amendment are slowly changing the practice of stacking (Hilton et al., 2020). Reducing the radioactivity even further by removing radium decay products could make this phosphogypsum available as an inexpensive resource for agriculture and construction. The WPA process for fertilizer production with uranium recovery is schematically shown in Fig. 1.

After the phosphogypsum is filtered out uranium can be recovered from phosphoric acid. Phosphoric acid must be purified in the WPA process depending on its further use, as other impurities in addition to uranium such as iron, magnesium, arsenic, aluminum, etc. are dissolved in the digestion stage and transfer into the phosphoric acid. Food and pharmaceutical industries require high-grade phosphoric acid whereas low-grade phosphoric acid is sufficient for fertilizer production.

While the majority (60–90%) of the uranium transfers into the phosphoric acid stream the majority of the REEs will be found in the phosphogypsum, making recovery of the REEs more challenging. For uranium recovery from phosphoric acid, mostly solvent extraction (SX) and to a much lesser extent ion exchange (IX) techniques were used in the past (Haneklaus et al., 2017a). Particularly, solvent extraction of uranium from phosphoric acid can be considered an industrially mature technology that has also been reviewed in the scientific literature (Beltrami et al., 2014; Bunus, 2000; Singh et al., 2016) while new ion exchange processes are currently investigated for future uranium recovery from phosphates that will still need to be tested on industrial scale.

Besides, it is also possible to leach uranium directly from phosphate rock using weak acids and conventional leaching techniques (Al-Khaleedi et al., 2019). Lab-scale experiments that leached uranium and REE from phosphate rock showed that such an approach can be beneficial in a way that uranium is removed early on in the process and at least the radioactivity in phosphogypsum caused by ^{238}U can be dramatically reduced.

Assuming uranium recovery from WPA, the amount of uranium that can be recovered from a certain phosphate rock can be estimated using Eq. (4) for phosphate rock and Eq. (5) if mass and uranium concentrations of the relevant phosphoric acid are provided.

$$m_U = (m_{PR} * c_{U,PR} * f_{U,PA}) * \eta_{U,PA} \quad (4)$$

$$m_U = (m_{PA} * c_{U,PA}) * \eta_{U,PA} \quad (5)$$

m_U = Mass uranium, m_{PR} = Mass phosphate rock, m_{PA} = Mass phosphoric acid, $c_{U,PR}$ = Concentration of uranium in phosphate rock, $c_{U,PA}$ = Concentration of uranium in phosphoric acid, $f_{U,PA}$ = Fraction of uranium that transfers from the phosphate rock to the phosphoric acid, $\eta_{U,PA}$ = Extraction efficiency of uranium from phosphoric acid.

The amount of uranium that could thus be recovered from 1 t phosphate rock from Central Florida (m_{PR}) that shows an average uranium concentration ($c_{U,PR}$) of 141 mg/kg (Van Kauwenbergh, 1997), if 80% ($f_{U,PA}$) of the uranium transfers to the phosphoric acid and 90% ($\eta_{U,PA}$) can be extracted from the phosphoric acid is a bit more than the mass of a chocolate bar or 102 g.

Larger phosphate fertilizer plants process 500,000 thousand to 2 million t phosphate rock per year resulting in notable potential for uranium recovery. Table 2 summarizes past commercial uranium recovery activities in the United States, Belgium, Canada, Taiwan and Iraq. Different extractant solvents were used for uranium recovery that gave the respective process its name. Bis(2-ethylhexyl) phosphoric acid (DEPA) and tri-n-octylphosphine oxide (TOPO) was the most popular extractant used for the DEPA-TOPO or ORNL process named after Oak Ridge National Laboratory (ORNL) in the United States where the process was developed.

Uranium recovery activities are sometimes classified in a 1st and 2nd wave of uranium recovery from phosphates (Haneklaus et al., 2015a). In the 1st wave (1951–61) uranium mining was still in its infancy and various materials, including phosphate rock, were considered to be potential sources of uranium. With the discovery of higher-grade uranium deposits the 1st wave and uranium recovery from phosphates stopped. The 2nd wave (1976–98) was commercially driven. Since the majority of costs for excavation, mining infrastructure, etc. are covered by the phosphate industry already (Reitsma et al., 2018) and uranium prices were favorably high as a result of the oil crisis in 1973, operating commercial uranium extraction plants next to fertilizer plants was economic for some time and done in the United States, Belgium, Canada, Taiwan and Iraq. In addition, uranium was recovered from phosphates in Kazakhstan behind the iron curtain, and many other countries operated pilot plants investigating potential uranium recovery from domestically mined or imported phosphate rock.

Particularly impressive was uranium recovery from phosphates in the United States, where the technology at its height contributed up to 20% of the domestic uranium production in the 1980s before decreasing uranium prices made this process uneconomic. Fig. 2 compares historic uranium imports and historic uranium mining with historic and potential uranium recovery from phosphates in the United States.

Like many other countries that rely on nuclear power for a considerable share of their electricity production the United States is importing the bulk of the required uranium from abroad. Interestingly, since the 1990s the amount of uranium that could theoretically be recovered from phosphates as a byproduct in fertilizer production could be larger than actual domestic uranium mining. In this context, recent concerns about uranium supply security in the US may indeed *make uranium recovery from phosphates great again* as speculated by Steiner et al. (2020).

Table 2 Historic uranium recovery from phosphates by country and plant.

Operating period	Country	Plant	Uranium extraction process	Capacity (t U/year)
1951–62	United States	Joliet, IL	Precipitation	31
1952–56	United States	Texas City, TX	OPPA	20
1954–59	United States	Nichols, FL	–	–
1955–61	United States	IMCC Plant, Bonnie, FL	OPPA	31
1957–61	United States	Tampa, FL	OPPA	62–138
1976–80	United States	W.R. Grace Plant, Bartow, FL	OPAP	109–127
1978–81	United States	Farmland, Pierce, FL	DEPA-TOPO	153–173
1978–99	United States	Uncle Sam, Convent, LA	DEPA-TOPO	265
1979–82	United States	Riverview Plant, East Tampa, FL	OPPA	138–164
1980–85	United States	CF Industries, Bartow, FL	DEPA-TOPO	231–243
1980–92	United States	CF Industries, Plant City, FL	DEPA-TOPO	231–243
1980–92	United States	IMC, New Wales, FL	DEPA-TOPO	265–288
1980–98	Belgium	Engis, Liège	DEPA-TOPO	50
1980–87	Canada	WCFL, Calgary, AB	OPAP; DEPA-TOPO	38
1981–85	Taiwan	China Phosphate, Lung Tan	DEPA-TOPO	10
1981–98	United States	Sunshine Bridge, Donaldsonville, LA	DEPA-TOPO	162
1984–91	Iraq	Al Qaim	DEPA-TOPO	87

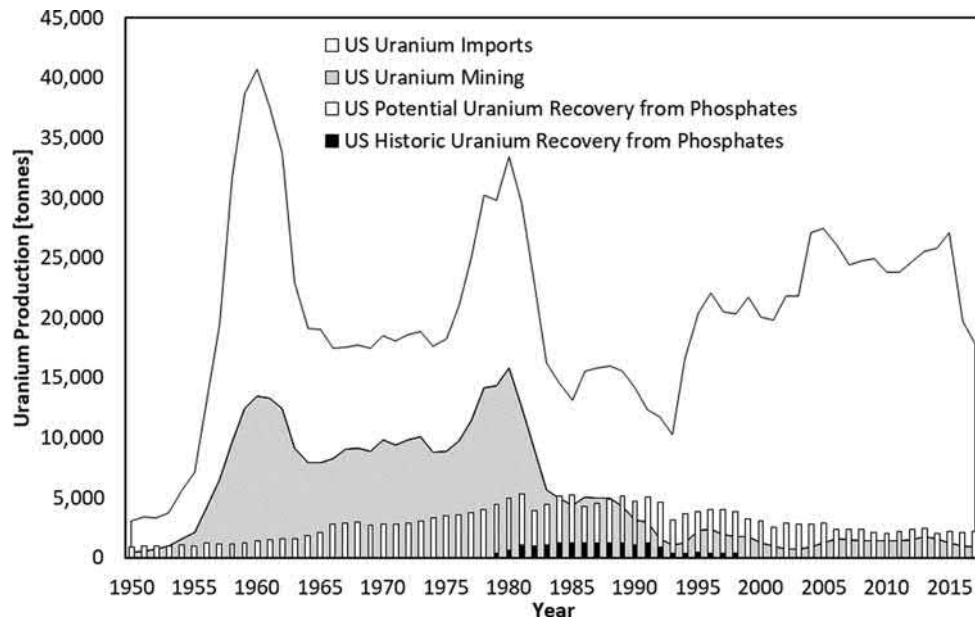


Fig. 2 Historic US uranium imports, uranium mining, past- and potential uranium recovery from phosphates.

Potential uranium recovery from phosphates

The potential of uranium from phosphates can be estimated using annual production data and phosphate rock reserves as they are published annually by the [USGS \(2020\)](#) as well as average concentrations of uranium in these phosphates. [Table 3](#) shows the amounts associated with phosphate rock of the largest 23 producing countries in 2018 as well as the quantity of uranium associated with phosphate rock reserves in these countries. Production data was taken from the [USGS \(2020\)](#) and average uranium concentrations by country were taken from [Van Kauwenbergh \(1997\)](#).

The very simple quantitative assessment in [Table 3](#) indicates that there are roughly 14,000 t uranium associated with phosphate rock production annually. Of these 14,000 t approximately 8800 t of uranium are associated with phosphate rock that show elevated concentrations of 90 mg uranium per kg that may justify economic recovery. The brief estimate presented here corresponds well to more elaborate studies by [Gabriel et al. \(2013a,b\)](#) and [Ulrich et al. \(2014\)](#) that estimate the potential uranium from phosphate rock to be as high as 10,000 t and 11,000 t respectively.

In this context, world uranium production for the same year (2018) was approximately 53,000 t ([WNA, 2020b](#)). Equally impressive are the total amounts of uranium associated with phosphate reserves that are with 6.25 million t as high as recoverable identified uranium resources (6.15 million t) ([WNA, 2020b](#)). Just like the amount of uranium associated with seawater, the quantity of uranium from phosphates is impressive, but will only be relevant if it can be recovered economically.

The economics of phosphate production, the uranium price and environmental benefits of recovering uranium are the most important factors to determine whether uranium recovery from phosphates is economic or not ([Haneklaus et al., 2017b](#)). In addition, strategic considerations such as uranium supply security may be considered.

To estimate quantities of uranium that may be recovered economically one day a cut-off grade of 90 mg uranium per kg phosphate rock is often used. [Kim et al. \(2016\)](#) and [Steiner et al. \(2020\)](#) estimated that the quantities of unconventional uranium that could be recovered during phosphate rock processing in the United States could theoretically cover 10% of the national uranium requirements (2116 t of 20,386 t uranium in 2014). Similar assessments were conducted by [Shang et al. \(2021\)](#) for China, indicating that China could have recovered some 648 t uranium during phosphate fertilizer production in 2016. This amount could have covered nearly 10% of the country's reported uranium requirements for that year. The Chinese case study is particularly interesting since the average uranium content in Chinese phosphate rock (27 mg/kg) provided by [Van Kauwenbergh \(1997\)](#) in [Table 3](#) would not justify uranium recovery. More recent investigations by [Ye et al. \(2019\)](#) found that there are indeed phosphate rock locations in China with uranium concentrations as high as 480 mg/kg. Even if these deposits are not in the majority, uranium recovery from them should be considered and could add relevant amounts of heavy metal to the country's uranium supply.

The Chinese case study is further relevant as it highlights the importance of relatively small deposits with elevated uranium contents that could actually profit from recovery of most uranium from those phosphates and could be the next locations where this technology may be relevant. Another example of this is the Minjingu phosphate rock deposit in Tanzania that shows elevated average uranium concentrations of 390 mg/kg with peak concentrations reaching nearly 800 mg/kg. The high uranium (and also REE) concentrations of the Minjingu deposit may justify multi-element mining as is already done at Olympic Dam in Australia.

Besides the case studies for the United States and China, [Tulsidas et al. \(2019\)](#) and [López et al. \(2019\)](#) conducted case studies for the European Union where 2% of the uranium requirements or 334 t uranium of 16,084 t total demand could have come from

Table 3 Potential quantities of uranium associated with phosphates.

	<i>Phosphate rock production in 2018 (million t)</i>	<i>Estimated phosphate rock reserves in (million t)</i>	<i>Estimated average uranium content (mg/kg)</i>	<i>Uranium associated with phosphate production in 2018 (t)</i>	<i>Total amount of uranium associated with phosphate reserves (t)</i>
Algeria	1.200	2200	63	76	138,600
Australia	2.800	1200	84	235	100,800
Brazil	5.740	1700	201	1154	341,700
China	120.000	3200	27	3240	86,400
Egypt	5.000	1300	90	450	117,000
Finland	0.989	1000	37	37	37,000
India	1.600	46	23	37	1058
Israel	3.550	62	120	426	7440
Jordan	8.020	1000	84	674	84,000
Kazakhstan	1.300	260	100	130	26,000
Mexico	1.540	30	100	154	3000
Morocco	34.800	50,000	97	3376	4,850,000
Peru	3.900	210	72	281	15,120
Russia	14.000	600	28	392	16,800
Saudi Arabia	6.090	1400	100	609	140,000
Senegal	1.650	50	67	111	3350
South Africa	2.100	1400	23	48	32,200
Syria	0.100	1800	75	8	135,000
Togo	0.800	30	94	75	2820
Tunisia	3.340	100	44	147	4400
United States	25.800	1000	99	2554	99,000
Uzbekistan	0.900	100	47	42	4700
Vietnam	3.300	30	30	99	900
Other countries	0.970	770	–	–	–
Total	249.489	69,488		14,353	6,247,288
Total uranium with 90 mg/kg cut-off grade				8798	5,560,960

phosphates in 2017 and Argentina where 8% of the uranium requirements or 19 t uranium of 235 t uranium could have come from phosphates in 2017. Both case studies assume a cut-off grade of 90 mg/kg, and in both case studies phosphate rock imports are of significant importance.

All countries need phosphate fertilizer for their food production and several countries have phosphate rock processing facilities without actually mining phosphate rock. Detailed material flows are for instance available for a phosphate rock processing plant in the Philippines [Haneklaus et al. \(2015b\)](#) that processes phosphate rock from different parts of the world.

Morocco has by far the largest phosphate rock reserves in the world (73% in [Table 3](#)) and this phosphate rock shows elevated uranium concentrations of 97 mg/kg that are often reported to actually be as high as 130–160 mg/kg ([Azouazi et al., 2001](#)). Orano SA (previously Areva SA), the large multinational nuclear fuel cycle company headquartered in France, signed a memorandum of understanding with Morocco's state-owned phosphate rock mining company (OCP Group) in 2006 for uranium recovery from Moroccan phosphates. Unlike the United States and China that are the other major phosphate rock producing countries in the world (see [Table 3](#)), Morocco does not have a large national market for phosphates and exports most of the mined ore to other countries.

Given the current phosphate rock reserves, Morocco will most likely supply various regions in the world with phosphate rock, phosphoric acid and phosphate fertilizers in the near future ([Cooper et al., 2011](#)). The increased use of Morocco phosphate rock with elevated uranium concentration could favor uranium recovery. Mineral resources are dynamic in a way that innovations and exploration efforts continuously discover “new” materials deemed economic for mining and processing ([Scholz and Wellmer, 2013](#)). Seabed phosphates, such as the Chatham Rise deposit offshore of New Zealand with 240 mg/kg uranium content may become a resource soon, changing the current assumptions about resources, reserves and geological potential one more time.

While one day phosphate rock may be mined not only for its P content but for all its constituents such as uranium, REE and its rock matrix that can be used in construction after radiotoxic uranium and its radioactive decay daughters have been recovered ([Zhang, 2014](#)), this is clearly not the case yet. The latest push towards commercial uranium recovery from phosphates by PhosEnergy was put on hold in 2019 as a result of low uranium prices. These low uranium prices are presently forcing some commercial uranium mines into a state of permanent maintenance. Facilities to recover uranium from phosphates can be erected fairly quickly (2–3 years) compared to the time required to set-up a new commercial uranium mine (5–10 years). Should uranium prices rise again though, it will most likely be uranium mines currently shut down that will be online first to fill the demand gap.

Conclusions and outlook

Quantities and concentrations of uranium associated with phosphates are considerable, and uranium from phosphates is probably the most important unconventional uranium resource that is not mined today. Although the majority of the costs for excavation, infrastructure development, mining, and processing are covered by the phosphate industry, uranium recovery from phosphates is not profitable today and it is unlikely that this will change as long as uranium prizes are low. However, uranium supply security and environmental concerns may lead to uranium recovery from phosphates at selected locations.

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Uranium Conversion and Enrichment

Patrick Hogan, Idaho National Laboratory, Idaho Falls, ID, United States

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Introduction

After mining, milling, and associated chemical processing, the resulting product usually consists of a dry powder that is still impure but concentrated in uranium oxides, especially U_3O_8 or UO_3 (Hausen, 1998). This material is neither chemically nor isotopically suited for direct use in nuclear fuel material and is instead subjected to subsequent chemical processing referred to as conversion. This process purifies the incoming concentrates but also, as the name implies, chemically converts the concentrates into the form required by the subsequent processing step, which is most typically an enrichment process requiring uranium hexafluoride, UF_6 . The term is occasionally used to include the direct conversion of concentrates to UO_2 , such as for CANDU reactors, essentially skipping an enrichment step in the nuclear fuel cycle, however conversion to UF_6 is the far more predominant process. A small number of large facilities provide all the conversion services for the world's nuclear energy industry, with only three facilities supplying 93% of global demand in 2019.

After conversion, the isotopic composition of uranium is too low to use as fuel for most types of nuclear reactors. The fissile isotope, ^{235}U , makes up only 0.71% of all naturally occurring uranium, with the balance consisting almost entirely of the non-fissile isotope ^{238}U . While ^{238}U is fertile and some fuel cycles, such as that used by the previously-mentioned CANDU reactors, can use uranium with this naturally occurring isotopic composition (called "natural uranium"), most reactors require a higher concentration of ^{235}U to perform as designed or even to achieve criticality at all. The process of increasing the concentration of ^{235}U in uranium is called enrichment.

However, achieving this increase is notoriously technically demanding. As is the case with all isotopes of a single element, ^{235}U and ^{238}U have nearly identical chemical and physical properties, thus viable enrichment processes are limiting to exploiting the mass difference between the target isotopes, a requirement made even more difficult by the fact that the two uranium isotopes of interest exhibit a mass difference of only just over 1%.

The technical challenges are further complicated by the nuclear weapons proliferation concerns associated with uranium enrichment. Although enriching uranium to 3–5% ^{235}U yields material sufficient to fuel nearly all the world's nuclear energy generation capacity, the same enrichment technologies can yield even higher enrichments, up to the 90% or more typical of a nuclear weapon. Few technological barriers exist to developing a rudimentary but powerful nuclear weapon once enough uranium of sufficient enriched is obtained, and for that reason uranium enrichment technology is closely controlled by the nations that deploy it, classified among the most closely guarded national security secrets any nation possesses.

These aspects combine to make uranium enrichment technology uniquely difficult to develop and expensive to deploy. Nevertheless, the drive for ever-improving enrichment technologies continues because the unique challenges are only exceeded by the unique properties of enriched uranium and the essential role it plays in the global nuclear energy industry.

Conversion processes

The process of uranium conversion adapts a variety of well-established chemical processing technologies and is now a mature manufacturing industry benefitting from decades of optimization focused on every detail of the process. The process itself must yield a high purity product with a uniform chemical and physical form as dictated by their customer, generally conforming to a standard such as ASTM C996-20. While any number of chemical processes are potentially suitable, the current industry has settled on a chemical processing route known as the "wet" process. U.S. general partnership ConverDyn employs an alternative "dry" process, however that conversion facility ceased production in 2018 with restart planned for 2023 at the earliest.

The wet process starts with dissolving the incoming uranium concentrates in nitric acid, yielding a yellow-colored solution of uranyl nitrate, $\text{UO}_2(\text{NO}_3)_2$. An organic solvent extraction process is then utilized, which may vary slightly between facilities but nonetheless produces a precipitant that is then dried into a purified uranyl hexahydrate, $\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ and thermally decomposed to yield uranium trioxide, UO_3 (Cameco, 2021). The UO_3 is then processed in a furnace or kiln and reacted with hydrogen gas to reduce the contents to uranium dioxide, UO_2 . At this point, the product may be further processed to UF_6 for subsequent enrichment or used as-is if no enrichment is required. To convert to UF_6 , solid uranium tetrafluoride (UF_4) is first produced via the reaction $\text{UO}_2 + 4\text{HF} \rightarrow \text{UF}_4 + 2\text{H}_2\text{O}$, which occur with aqueous hydrofluoric acid (HF) in solution, or completely in the gas phase. Finally, gaseous fluorine (F_2) is reacted with the UF_4 , generally in a fluidized bed reactor, to produce the UF_6 final product, which is then condensed and stored in purpose-built containers for shipment to the customer.

The dry process forgoes the dissolution, solvent extraction, and drying steps and directly reduces the incoming oxide uranium concentrate with hydrogen in a furnace to produce UO_2 , which is then subject to the same process described above to yield UF_6 . Although this produce does not realize the purification benefits of the solvent extraction steps of the wet method, purification is instead realized by distilling the UF_6 , drawing off light fraction gases and impurities and leaving UF_6 of over 99.99% purity (ConverDyn, 2020).

Enrichment process overview

While enrichment processes vary, they share some fundamental commonalities. All modern commercial enrichment processes use natural UF_6 as the input (“feed”) and yield enriched UF_6 as the product. The target enrichment depends upon the application; however the vast majority of modern enrichment operations are aimed at civilian nuclear energy generation and yield product in the 3–5% range. Higher enrichments are used in smaller quantities, frequently up to 20%, the upper limit that can be considered Low Enriched Uranium (LEU). Enrichments above 5% but less than 20% are often referred to as High Assay, Low Enriched Uranium or HALEU, and are of growing interest beyond research reactors for advanced reactors intended for widespread deployment. Beyond 20% enrichment is considered High Enriched Uranium, or HEU, and used for a small number of research reactors as well as many naval reactors and for nuclear weapons.

In addition to the product, the enrichment process necessarily produces another output stream consisting of UF_6 depleted in ^{235}U , called “tails.” These three quantities and their respective ^{235}U to U ratios (called assays) are related by a series of two equations consisting of the feed to product ratio,

$$\frac{F}{P} = \frac{x_p - x_t}{x_f - x_t}$$

and the tails to product ratio,

$$\frac{T}{P} = \frac{x_p - x_f}{x_f - x_t}$$

Where F, P, and T are the feed, product, and tails masses, respectively; and x_f , x_p , and x_t are the respective assay of the feed, product and tails. These equations illustrate several important considerations in enrichment, but one that stands out is that a low tails assay is required to yield the greatest product quantity while minimizing the use of feed. The amount of feed is an important economic driver for enrichment processes because its mining, milling, and conversion represent a substantial expense. However, enrichment processes are not perfectly efficient and, in practice, the UF_6 can pass through as many as thousands of individual enrichment stages to reach the desired enrichment. The work required to achieve product assay x_p is quantified as separative work units, or SWU, and defined as

$$\text{SWU} = P \cdot V(x_p) + T \cdot V(x_t) - F \cdot V(x_f)$$

where the value function is, in turn, defined as

$$V(x) = (2x - 1) \ln\left(\frac{x}{1-x}\right)$$

These constraints define a balance between the work required for enrichment and the resources required to produce the feed material, and any calculation of an optimum set of parameters incorporates these constraints. Typically, tails assays are in the range of 0.2–0.4% ^{235}U , and the product is enriched to 3–5% for light water reactors (LWRs), meaning that only approximately 10% of the feed is converted into product, with the remaining 90% resulting in tails with limited or no economic utility. These tails are often referred to as “depleted uranium,” or DU, because the tails are depleted in the ^{235}U isotope. While DU does have some useful applications, these demands are dwarfed by DU production as a byproduct of enrichment. A modern, large LWR requires on the order of 100,000 kgSWU/year, and the large enrichment plants responsible for most of the world's SWU produce on the order several million kgSWU/year.

These costs constitute a significant driver of the economics of nuclear plant operations and are governed by a complex set of supply and demand relationships between mining, milling, conversion, enrichment, fuel fabrication, and power generation. These market-driven costs may fluctuate substantially over a period of years, with enrichment at times making up most of the total cost of

fuel production. Although fuel makes up only approximately 20% of the overall cost of nuclear power (Nuclear Energy Institute, 2020), enrichment costs are still significant in a nuclear power industry with tight profit margins, where \$0.01/kWh can make a meaningful difference in decisions to retire legacy plants, or in financing new reactor builds.

Enrichment technologies

Enrichment technologies have followed an evolutionary trajectory that can be traced through historical technology development. While there have been many suggestions for novel techniques, some of which have been demonstrated at laboratory scale, the principal technologies can be divided into a few categories. The earliest enrichment technologies were developed by the United States during World War II as part of the Manhattan Project for the purpose of nuclear weapons. They were enormously expensive but justified by a wartime sense of urgency. Over time, other technologies were developed and improved, until the economics of commercial nuclear energy drove a convergence to a single technology, the centrifuge. Nevertheless, while centrifuges have performed most of the world's enrichment for many decades, the costs and complexity of the technology continue to motivate deployment of a new generation of enrichment technologies as well as evolutionary improvements to the venerable centrifuge.

Electromagnetic

The first practical method for enriching uranium was demonstrated in 1941 using a device called a Calutron. The technique was later improved and deployed on a grand scale in 1943–44 to enrich uranium for the Manhattan Project.

The technique relied upon the mass difference between the different isotopes of uranium, ionizing atoms of metallic uranium (rather than UF_6), accelerating them through an electrical field, then deflecting their trajectory with an orthogonal magnetic field. The lighter ^{235}U atoms experienced greater deflection than did the heavier ^{238}U atoms. A collector specially separated the stream of ions, resulting an enriched product stream accompanied by a depleted tails stream.

The technique was quickly supplanted by the more efficient gaseous diffusion process and the calutrons ceased enriching uranium shortly after the war ended, at a total cost of approximately \$9B in 2020 USD. Electromagnetic separation techniques are no longer used to support the nuclear energy industry, although it remains a concerning route for potential nuclear weapons proliferation (Fig. 1).

Diffusion

The gaseous diffusion technique also was first deployed at a practical scale as part of the Manhattan Project, with the first pilot plant operational in 1944. However, with evolutionary improvements, the gaseous diffusion technique grew into and sustained a role as an important contributor to enrichment capacity, mainly in the United States and France, contributing as much as one third of global production as late as 2008 and continuing in large-scale commercial enrichment operations until 2013. Unlike electromagnetic separation, gaseous diffusion utilizes UF_6 throughout. The technique relies on carefully-engineered barriers through which the UF_6 diffuses. Because the diffusion rate depends on the molecular mass of the gas, lighter molecules diffuse through the barrier at a higher rate, resulting in a slight ^{235}U enrichment of the gas on the lower-pressure side of the barrier. By using large areas of such

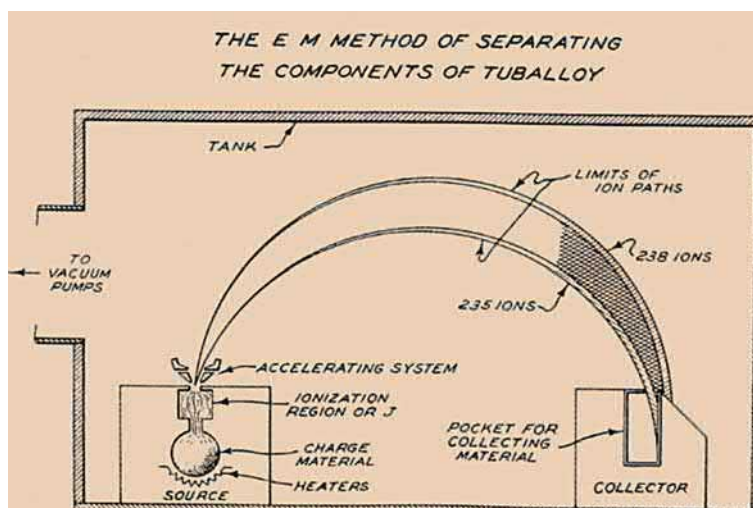


Fig. 1 Circa early 1949s diagram of a calutron. Courtesy Westinghouse.



Fig. 2 Georges Besse gaseous diffusion plant after decommissioning. Courtesy Orano.

barriers and rapidly exchanging the gas across them repeatedly (and typically via thousands of stages), this enrichment effect can be multiplied to yield commercially significant throughputs.

However, the operating costs of this technique, particular the electrical demand, is surpassed only by that of electromagnetic separation. The last operational gaseous diffusion plant, located in Paducah, KY in the United States, consumed approximately 3 GW of electricity before it ceased operations in 2013 (Fig. 2).

Thermal diffusion was also deployed during the Manhattan Project at scale and utilized liquid UF_6 to exploit the fact that a thermal gradient yields differential diffusion along the gradient in an otherwise well-mixed solution. However, it was not competitive with gaseous diffusion and the plant was shut down in 1945. The technique has not since been commercially deployed.

Centrifuge

Centrifuge technology is currently the only commercial enrichment process utilized to support the global nuclear energy sector. A centrifuge plant consists of cylinders into which feed UF_6 is flowed. The cylinders are spun at very high speeds, generating large centripetal forces that serve to concentrate the heavier molecules nearer the periphery of the cylinder. The lighter UF_6 , enriched in ^{235}U , is concentrated nearer the center of the centrifuge and can be drawn off to yield a slightly enriched product, while the heavier UF_6 , depleted in ^{235}U , can be drawn off from the periphery of the cylinder. The enriched product stream can then be flowed into a subsequent centrifuge to yield a further enriched product, and the process repeated until the target enrichment is obtained. This system of units that combine small enrichment increments is called a cascade and is ubiquitous in centrifuges and other enrichment technologies.

The performance can be enhanced by reaching higher rotation speeds (either through larger diameters or higher RPM), or employing more modern “Zippe-type” technology that utilizes an axial temperature gradient to increase the efficiency of separation in each stage.

Although simple in principle, the high rotational speeds required for efficient separation require advanced materials, precision machining, and uniquely high-performing components to prevent catastrophic and destructive failure of each centrifuge. Centrifuge performance is varied, distinguished both by the evolution of technology over time as well as the nation deploying it. The technical details of centrifuges are carefully protected by the nations that utilize them (Fig. 3).

A similar but nonetheless unique technique is the vortex process developed in South Africa. In this process, the centripetal forces are achieved not by rotating a cylinder, but by tangentially injecting the gas at high speeds into a carefully designed cylindrical chamber. Though more technically simple than the gas centrifuge process, it is more energy intensive. It was deployed to enrich uranium in South Africa at commercially significant rates, but production was halted when South Africa ended its nuclear program in 1989. Vortex-based enrichment does not currently constitute a significant contribution to global uranium enrichment output.

Laser

Numerous laser techniques for uranium enrichment have been evaluated and tested but are not yet commercially significant. Laser based techniques rely on slightly different spectroscopic properties of the uranium atoms or uranium-based molecules to achieve some spectral selectivity between the isotopes, although the details vary. Molecular Laser Induced Separation (MLIS) was the first technique demonstrated at an engineering scale, investigated in the United States in the 1970s. It used UF_6 gas and selectively excited one isotope with laser radiation until the UF_6 dissociated into UF_5 which precipitated and could be collected. Another technique, Atomic Vapor Laser Induced Separation (AVLIS), was also first developed in the United States and reached closer to commercial deployment. AVLIS used atomic (metallic) uranium as the medium, heated to a vapor. The laser then selectively excited one isotope to ionize it, then the ionized product stream could be collected electrostatically. The technology was developed in the



Fig. 3 Gas centrifuge cascade, Piketon, OH. Courtesy U.S. Department of Energy.

1980s, and eventually was licensed to United States Enrichment Corp. (USEC) in 1994 for eventual commercial operation. However, in 1996 USEC abandoned its pursuit of AVLIS in favor of yet another laser enrichment method called SILEX (Selective Isotope Enrichment by Laser Excitation). Like MLIS, this process also uses UF_6 but has confirmed few technical details. Global Laser Enrichment LLC (GLE) currently controls the technology and continues development with plans to deploy a commercially significant operation in the United States, claiming that their technology can disrupt the industry with reduced deployment cost compared to incumbent technologies.

Downblending

Although the practice does not enrich uranium, in recent years downblending HEU from weapons stockpiles has been an important contributor to SWU needs in the nuclear energy sector. The best known such program has been “Megatons to Megawatts,” an agreement between the United States and Russia to reduce excess Russian HEU inventory originally produced for nuclear weapons. This material was blended with other, low enriched uranium to reduce the enrichment to the level required by nuclear reactor operators, then shipped to the United States to provide enriched uranium for nuclear fuel. Between 1994 and 2013 this agreement resulted in the equivalent of approximately 90 million kgSWU and constituted up to approximately 20% of overall U.S. enrichment demand.

Global production summary

Uranium conversion technology has been widely demonstrated but with the 2018 idling of the U.S. ConverDyn conversion plant due to contracting global demand, only a handful of producers currently supply the world’s needs (see Fig. 4).

Company	Country	Location	Production (2019)
Orano	France	Pierrelatte & Malvési	2500
CNNC	China	Lanzhou & Hengyang	10,000
Cameco	Canada	Port Hope	10,000
Rosatom	Russia	Seversk	12,000
ConverDyn	USA	Metropolis	0
Total			34,500

Fig. 4 2019 Global Conversion Output in metric tons uranium per year (World Nuclear Association, 2019).

Country	Company and plant	2015	2020
France	Areva, Georges Besse I & II	7000	7500
Germany-Netherlands-UK	Urenco: Gronau, Germany; Almelo, Netherlands; Capenhurst, UK.	14,400	14,900
Japan	JNFL, Rokkaasho	75	75
USA	USEC, Piketon	0	0
USA	Urenco, New Mexico	4700	4700
USA	Global Laser Enrichment, Paducah	0	0
Russia	Tenex: Angarsk, Novouralsk, Zelenogorsk, Seversk	26,578	28,663
China	CNNC, Hanzhun & Lanzhou	5760	10,700+
Other	Various: Argentina, Brazil, India, Pakistan, Iran	100	170
Total		58,600	66,700

Fig. 5 Global Enrichment Production in thousands of SWU per year (World Nuclear Association, 2019).

The global enrichment landscape has simplified in recent years as production has shifted essentially entirely to centrifuge technology, although the industry continues to be dynamic, with several large countries accounting for most of the enriched uranium production, while a number of other countries expand their capabilities (see Fig. 5).

In addition, the U.S. company USEC continues to develop and deploy their advanced centrifuge technology and GLE continues to pursue a potentially disruptive generational advancement.

The future enrichment landscape is particularly uncertain given today's growing demand for HALEU. Due to various technical constraints characteristic of uranium enrichment, it is difficult or impossible to convert existing enrichment plants to produce enrichments much beyond 5%. A number of advanced nuclear reactors require nearly 20% enriched uranium, and widespread deployment of such reactors will require new enrichment capacity to be operational well before such reactor deployments in order to produce sufficient fuel.

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<http://astm.org/Standards>—ASTM Standards and Publications.

Fuel Design and Fabrication: Pellet-Type Fuel

Masato Kato, Fuel Cycle Design Department, Japan Atomic Energy Agency, Ibaraki, Japan

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Introduction

This article presents a summary of fuel performance for fuel pellets made of uranium dioxide or UO₂, uranium-plutonium mixed oxide (MOX) or (U,Pu)O₂, uranium and uranium-plutonium nitrides or UN and (U,Pu)N, and uranium or uranium-plutonium carbides or (U,Pu)_xC_y. All these fuel types have been developed as fuels for sodium-cooled fast reactors, while UO₂ is commonly used in light water and heavy-water reactors as well. The design attributes of these fuel types are presented first. The fuel pellet temperature, which is a function of the pellet radius, the linear heat rate induced by fission, and the cladding and coolant temperature, is a key factor in the fuel performance phenomena, so the approach to analyzing for temperature distribution in a fuel pellet is presented. Fabrication methods are summarized in the latter section of this article.

Fuel design

As the first barrier to hazardous release of fission products and fuel constituents, fuel rods are designed to prevent fuel failure (typically defined as cladding breach). An important design objective and reactor operational requirement is to prevent fuel melting which could allow the fuel to slump axially, with adverse effects on core reactivity, or could cause liquid fuel penetration of the cladding leading to cladding breach. The maximum temperature of the fuel is limited to below the melting temperature of the fuels in early irradiation conditions, usually by limiting the linear heat rate (the power generated per unit length of fuel rod) that can be reached in the fuel rods. Fission products (FPs) accumulate in fuels during irradiation. The accumulation of FPs causes interactions between the fuel and the cladding (termed pellet-cladding interaction, or PCI) including FP gas release, pellet swelling, corrosion of cladding inner walls and so on. In addition, decreases in cladding strength are caused by neutron irradiation damage, which reduces the ability of the cladding to withstand internal pressure and other loads induced during service, possibly leading to fuel failure. It is important to evaluate in the fuel design stage the effects of creep damage of cladding during service. In this section, temperature analysis, pellet properties, and irradiation behavior are summarized, which are all necessary for establishing fuel design.

Temperature analysis in the radial direction of pellets

Power per unit length of fuel rods is called the linear heat rate, represented here by P , and is described with units of W/cm. UO₂ used in light water reactors operates at about 300 W/cm. MOX, nitrides, and carbides used for fuels in sodium-cooled fast reactors operate over 360 W/cm, with maximum values exceeding 400 W/cm.

The linear heat rate L is described by Eq. (1) as a function of thermal conductivity (λ) using the cylindrical coordinate system.

$$L = P/4 \cdot r_1^2 = \lambda(T_0 - T_1) \quad (1)$$

where P is the power per unit fuel volume, r_i is radial direction distance, and T_i is temperature at r_i , as shown in Fig. 1. According to Eq. (1), if we know the temperature dependence of the thermal conductivity of a fuel pellet, we can evaluate the temperature profile in the radial direction. Fig. 1 shows a schematic diagram of temperature profile in the radial direction of a fuel pin. The temperature profile can be described by Eqs. (1)–(4).

$$P = h_G \cdot 2 \cdot \pi \cdot r_1 (T_1 - T_{c1}) \quad (2)$$

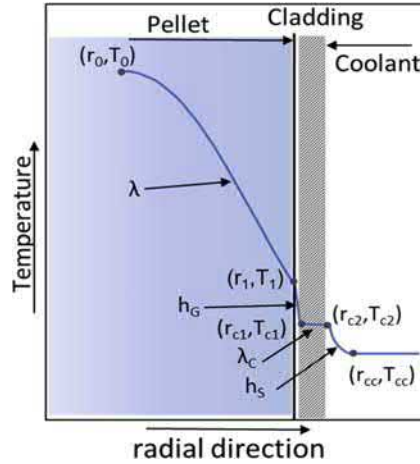


Fig. 1 Schematic diagram of temperature profiles in the radial direction of a fuel pin.

$$P = -2 \cdot \pi \cdot \lambda_c (T_{c1} - T_{c2}) \ln \left(\frac{r_{c1}}{r_{c2}} \right) \quad (3)$$

$$P = h_S \cdot 2 \cdot \pi \cdot r_{c2} (T_{c2} - T_{cc}) \quad (4)$$

where λ_c is thermal conductivity of the cladding, h_G is the heat transfer coefficient between the pellet and cladding, called gap conductance, and h_S is heat transfer coefficient between cladding and coolant.

Properties

The physical and thermo-physical properties of fuel materials are essential for evaluating fuel performance in fuel design (Table 1). Crystal structures of the fuel materials are shown in Fig. 2. Oxide fuel has the CaF_2 -type crystal structure, and the other ceramic fuel compounds have the NaCl-type structure. The lattice parameter, a , which is a basic property describing a crystal structure's unit cell is compared in Table 1.

The nitride and carbide fuels have higher density and thermal conductivity than do the oxides. The melting temperature for all materials is about 3000 K. Chemical stability at high temperatures can be a problem for nitrides because the nitride can decompose to metal and nitrogen gas, depending on the nitrogen partial pressure, and this behavior limits the maximum temperature in the fuel design (Uno et al., 2012).

In the stoichiometric composition of oxide fuel compounds, the oxygen-to-metal (O/M) ratio is equal to 2.00, as shown in Fig. 2a; however, oxide fuel is known to be a non-stoichiometric compound stable over wide range of O/M ratio. Specifically, MOX stability expands to hypo-stoichiometric compositions, with O/M ratio lower than 2.00. The stoichiometry significantly affects various properties. Therefore, it is important to understand O/M ratio in property evaluation of MOX fuel.

Morimoto et al. (2008a) measured thermal conductivity of Am-bearing MOX as shown in Fig. 3. The phonon conduction mechanism results in decreased thermal conductivity with increasing temperature. MOX shows slightly lower thermal conductivity than that of UO_2 , and the effect of Am content on thermal conductivity is small. Because the Pu and Am atoms behave as scattering factors in phonon conduction, it is understood that the solid solution of their atoms reduces thermal conductivity. This relationship can be written with the following equation.

Table 1 Property comparison (Carbajo et al., 2001; Solntceva et al., 2016; Sengupta et al., 2012).

Properties	UO_2	MOX ^a	(U,Pu)N ^a	(U,Pu)C ^a
Crystal structure type ^b	CaF_2	CaF_2	NaCl	NaCl
Lattice parameter, a (nm) ^b	0.5471	0.5952	0.489	0.496
Density (g/cm ³)	10.95	11.48	14.4	13.5
Melting temperature (K)	3128	3061	3070	2750
Thermal conductivity ^c (W/m-K)	3.84	3.54	18	19

^aPu/(Pu + U) = 0.2.

^bSee Fig. 2.

^cData at 1000 K.

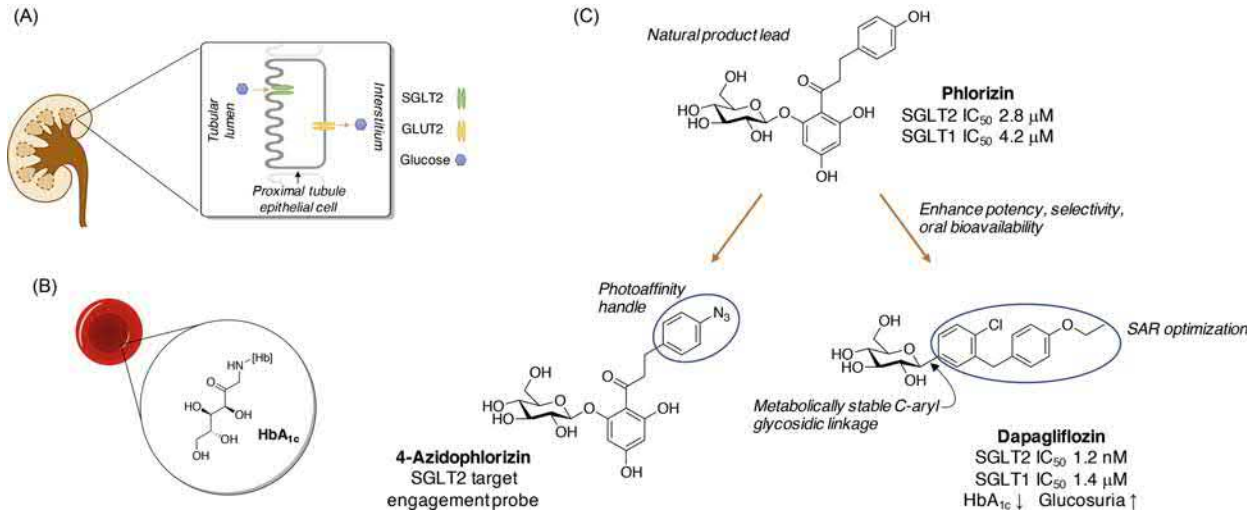


Fig. 2 Atomic positions in the CaF₂ and NaCl unit cells.

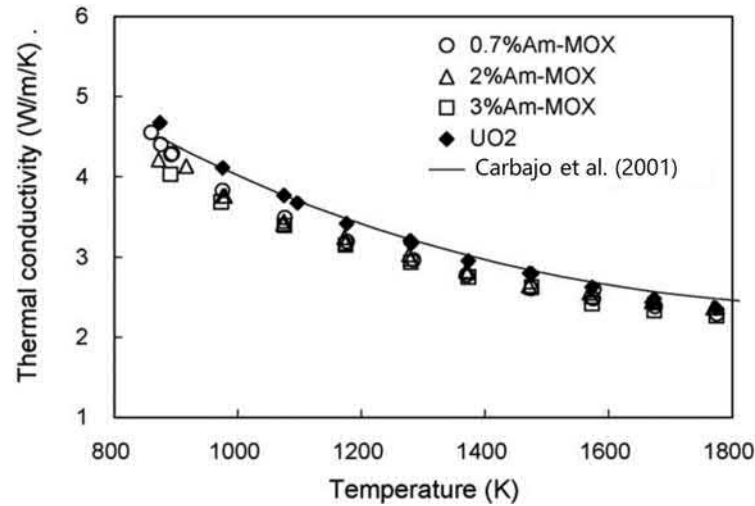


Fig. 3 Thermal conductivity of UO₂ and Am-containing MOX. Morimoto K, Kato M, Ogasawara M, Kashimura M, and Abe T (2008a) Thermal conductivities of (U, Pu, Am)O₂ solid solutions, *Journal of Alloys and Compounds* 452: 54–60.

$$\lambda = (A + BT)^{-1} + \frac{C}{T^2} \cdot \exp\left(-\frac{D}{T}\right) + C \cdot T^3 \quad (5)$$

where, A, B, C, and D are constants. The terms of $(A + BT)^{-1}$, $\frac{C}{T^2} \cdot \exp\left(-\frac{D}{T}\right)$ and $C \cdot T^3$ are phonon conduction, electrical conduction, and radiation contribution, respectively, in Eq. (5). MOX is stable in the hypo-stoichiometric composition. Therefore, thermal conductivity has been investigated as a function of O/M ratio as shown in Fig. 4 (Morimoto et al., 2008b). Decreases in O/M ratio leads to decreased thermal conductivity because oxygen vacancies lead to phonon scattering. Various correlations have been proposed to describe thermal conductivity (Carbajo et al., 2001; Philipponneau, 1992; Kato et al., 2011).

$$\lambda = \frac{(1-p)}{(1+2p)} \left[\frac{1.158 \cdot ((2.85x + 0.035) + (-0.715x + 0.286)T)^{-1}}{\left(\frac{T}{1000}\right)^{\frac{5}{2}}} \cdot \exp\left(-16.35 / \left(\frac{T}{1000}\right)\right) \right] \quad (6)$$

$$\lambda = \frac{1}{0.864} \frac{(1-p)}{(1+2p)} \left[\frac{\left((1.528\sqrt{x + 0.00931} - 0.1055) + (2.885 \cdot 10^{-4} \cdot T)^{-1} \right)}{+76.38 \times 10^{12} \cdot T^3} \right] \quad (7)$$

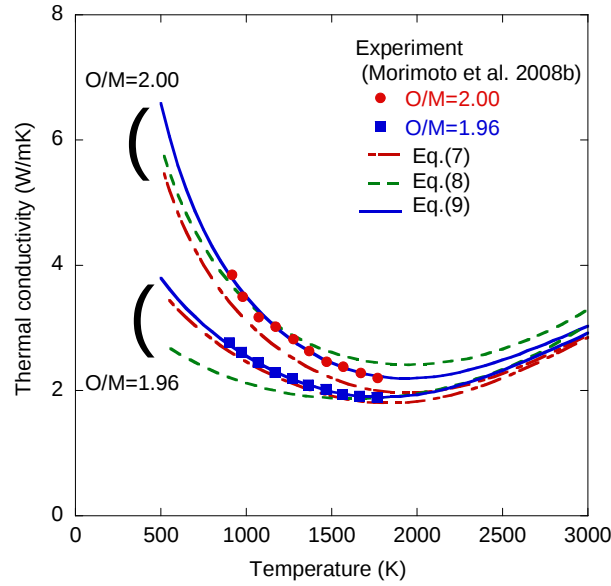


Fig. 4 Comparison between measured and calculated data of thermal conductivity of $(U_{0.7}Pu_{0.3})O_{2-x}$ with 95% TD.

$$\lambda = \frac{(1-p)}{(1+0.5p)} \left[\frac{(2.713x + 0.3583z_1 + 0.0317z_2 + 0.01595 + (-2.625x + 2.493) \cdot 10^{-4} \cdot T)^{-1}}{+ \frac{1.541 \times 10^{11}}{T^2} \cdot \exp(-1.522 \times 10^4/T)} \right] \quad (8)$$

where x is the substoichiometric deviation in MO_{2-x} , z_1 is Am content, z_2 is Np content in heavy metal, and p is porosity. Values calculated by the equations are shown in Fig. 4 for comparison. Eq. (8) was derived with use of the latest dataset.

Fig. 5 shows the thermal conductivity of nitrides as a function of Pu content. The value increases with increasing temperature and uranium content. Electrical conduction leads to higher temperature dependence of thermal conductivity as compared with oxide fuels.

In Fig. 6, the data are compared for various materials. Thermal conductivity of nitrides and carbides is similar with respect to temperature variation. This relationship in nitrides is described by Ross et al. (1990):

$$\lambda = (1.37 - 1.6C + 1.142C^2) \cdot T^{0.41} \cdot \frac{(1-p)}{(1+2p)} \quad (9)$$

where C is Pu content in $(U + Pu)$.

These correlations describing thermal conductivity are used to evaluate fuel performance using Eq. (1).

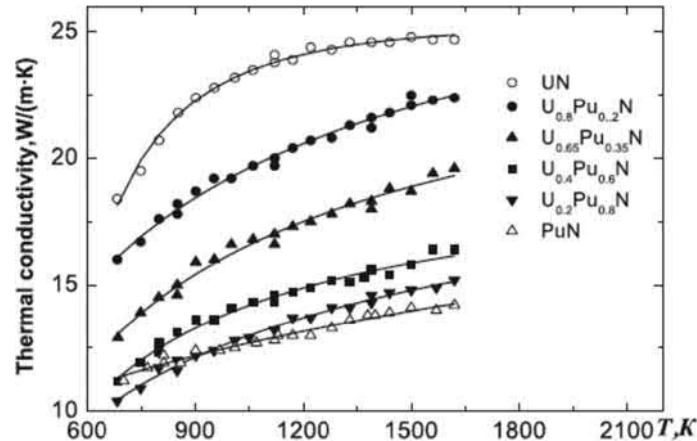


Fig. 5 Thermal conductivity of uranium and plutonium mixed nitrides. Solntceva ES, Taubin ML, Vybyvanets VI, Galyov IE, Baranov VG, Homiyakov OV and Tenishev AV (2016) Thermal conductivity of perspective fuel based on uranium nitride. *Annals of Nuclear Energy* 87: 799–802.

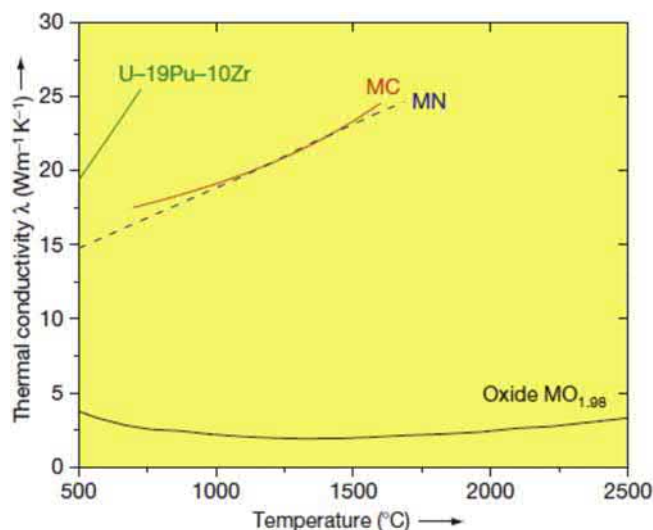


Fig. 6 Thermal conductivity of $(\text{U}_{0.8}\text{Pu}_{0.2})\text{O}_{1.98}$, indicated as $\text{MO}_{1.98}$, $(\text{U}_{0.8}\text{Pu}_{0.2})\text{N}$, indicated as MN, $(\text{U}_{0.8}\text{Pu}_{0.2})\text{C}$, indicated as MC, and U-19Pu-10Zr, a metallic fuel alloy. Sengupta AK, Agarwal R and Kamath HS (2012) Carbide fuel. In *Comprehensive Nuclear Materials Carbides*, pp. 55–85.

Irradiation behavior

Many irradiation tests have been carried out to investigate performance of these fuel types. Microstructure of the cross sections of irradiated pellets are shown in Fig. 7. Cracks are observed in the radial and circumferential directions of the pellets. It can be seen that there is significant restructuring in the irradiated MOX pellet. The MOX pellet in the figure was irradiated at a high linear heat rate of about 420 W/cm, and the high temperature and thermal gradient caused restructuring, which is shown in Fig. 8 (Tanaka et al., 2013). In the radial direction of the pellet, an unchanged region, an equiaxed grain region, a columnar grain region, and a central void are observed (Parrish and Aitkaliyeva, 2018). Pore migration through the fuel pellet plays a major role in restructuring in the radial direction of the pellet. High linear heat rate irradiation tests of minor actinides (MA)-bearing MOX pellets with varying O/M ratio were performed. Larger restructuring was observed in stoichiometric MOX pellets. This O/M dependence of restructuring is significantly affected by the vapor pressure in MOX. Ikusawa et al. (2017) reported that vapor pressure of UO_3 species becomes high in the near-stoichiometric composition, and forms the columnar grain and central void very quickly. The great thermal gradient causes actinide redistribution in the radial direction of the MOX pellet. Pu and Am content increased around the central void as shown in Fig. 9. This compositional change must be considered in fuel design, and the maximum linear heat rate is limited due to the melting temperature decrease caused by enriched Pu in center of the pellet.

An irradiated nitride pellet is shown in Fig. 7c after irradiation at a linear heat rate of about 710 W/cm. However, restructuring in the radial direction, like in the MOX pellet, was not observed because the nitride pellet has a high thermal conductivity and therefore is irradiated at lower temperature.

Many kinds of FPs are generated and accumulated during fissioning. Kleykamp (1985) reported the chemical states of FPs in oxides in detail. The ratio of FPs is different between Pu and U. Fission of Pu or U atoms generate two FPs and two oxygens liberated from molecular MO_2 . FP can be classified into four types according to their chemical behavior as follows:

- Type 1: gases and volatile elements such as Xe, Kr, Br, I.
- Type 2: FPs forming metallic precipitates such as Tc, Pd, Rd, Ru, Ir, Ag.
- Type 3: FPs forming oxide precipitates such as Rb, Cs, Ba, Zr.
- Type 4: FPs dissolved in U and Pu sites such as Nd, La, Ce, Zr.

The behavior of type 1 elements, Xe and Kr, is important in fuel design. Those elements are release as gases from the pellet to the fuel rod plenum (i.e., the empty space inside the fuel rod), increasing the internal fuel rod pressure and PCI. The evaluation of gas release rate is an important factor considered in the fuel design for LWR fuels and fast reactor fuels, because the fuel rod cladding must withstand the pressure of released fission gas. The behavior of FPs in nitrides and carbides is unclear because Post Irradiation Examination (PIE) data is limited.

In oxide fuel, Type 3 and 4 FPs react with liberated oxygen, and some of the surplus oxygen atoms accumulates during irradiation. The O/M ratio increase with burn-up is estimated to be $\text{O/M} = 0.001$ for a burn-up of 10 GWd/t (Matzke, 1995). This means that the chemical potential inside the fuel changes, and the chemical form of FPs, such as Mo and Cs, can change. Oxygen potentials of MOX were investigated by gas equilibrium methods and determined as functions of Pu content, O/M ratio, and temperature, as shown in Fig. 10 (Kato et al., 2017). In the near-stoichiometric region, oxygen potential is drastically changed with O/M ratio.

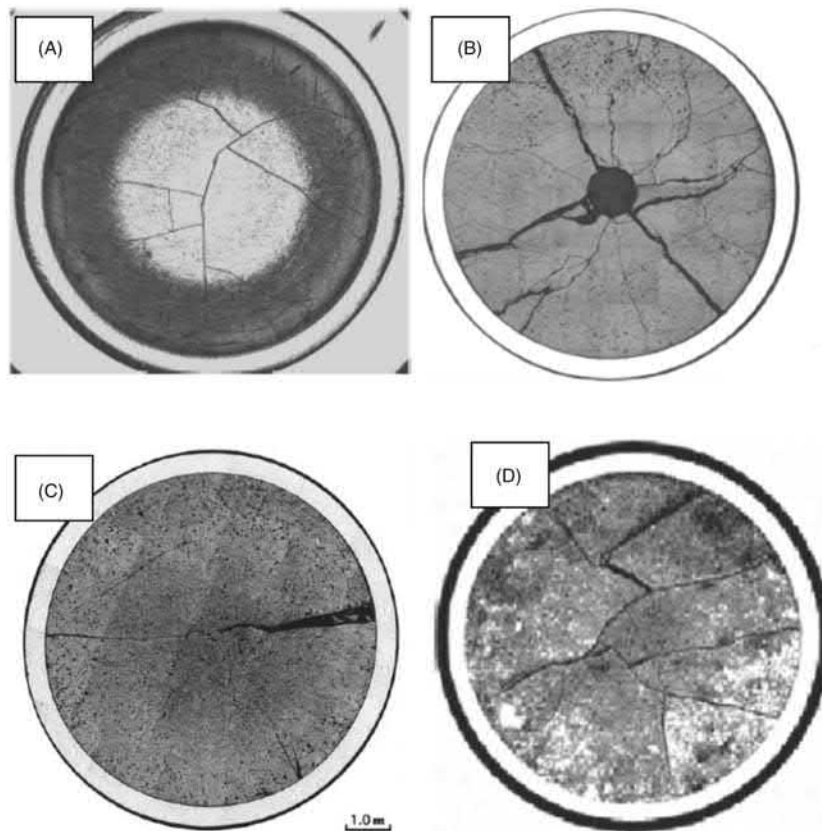


Fig. 7 Microstructure of the cross section of irradiated fuel pellets: (A) UO_2 pellet irradiated at a maximum linear heat rate of about 320 W/cm; (B) MOX pellet irradiated at a maximum linear heat rate of about 430 W/cm; (C) $(\text{U}_{0.8}\text{Pu}_{0.2})\text{N}$ pellet irradiated at a maximum linear heat rate of about 710 W/cm; (D) $(\text{U}_{0.8}\text{Pu}_{0.2})\text{C}$ irradiated at a maximum linear heat rate of about 450 W/cm. (A) Restani R, Horvath M, Goll W, Bretsch J, Gavillet D, Hermann A, Martin M and Walker CT (2016) On the condition of UO_2 nuclear fuel irradiated in a PWR to a burn-up in excess of 110 MWd/kgHM. *Journal of Nuclear Materials* 481: 88–100; (B) Tanaka K, Miwa S, Sekine S, Yoshimochi H, Obayashi H, and Koyama S (2013) Restructuring and redistribution of actinides in Am-MOX fuel during the first 24 h of irradiation. *Journal of Nuclear Materials* 440: 480–488; (C) Tanaka K, Maeda K, Katsuyama K, Inoue M, Iwai T and Arai Y (2004) Fission gas release and swelling in uranium-plutonium mixed nitride fuels. *Journal of Nuclear Materials* 327: 77–87; (D) Suzuki Y, Arm Y, Handa M and Shiba K (1988) *Research and Development of Uranium Plutonium Mixed Carbide and Nitride Fuels at JAERI*. IAEA TECDOC-466. IAEA: Vienna, pp. 73–82.

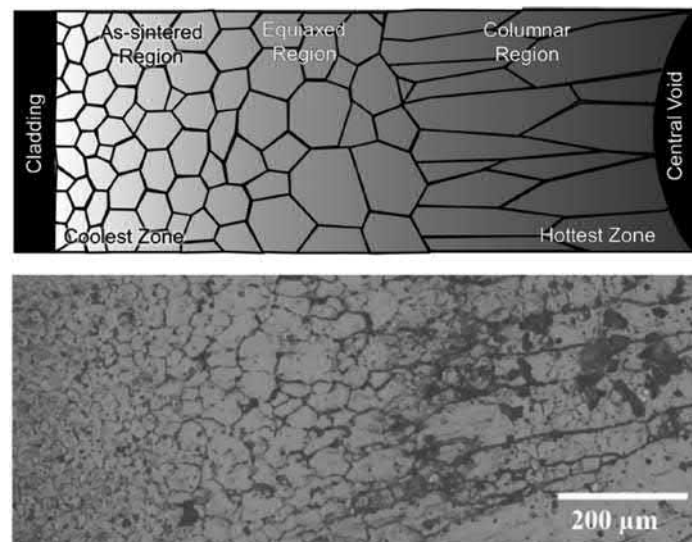


Fig. 8 Microstructural change in the radial direction of a MOX pellet irradiated in a fast reactor.

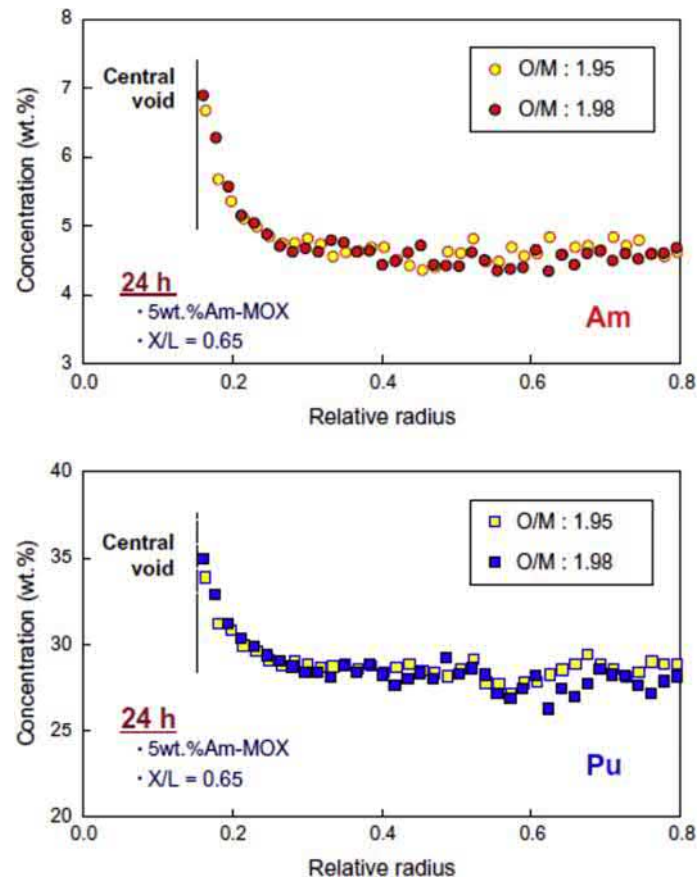


Fig. 9 Redistribution of actinide elements in the radial direction of MOX pellets irradiated in a fast reactor.

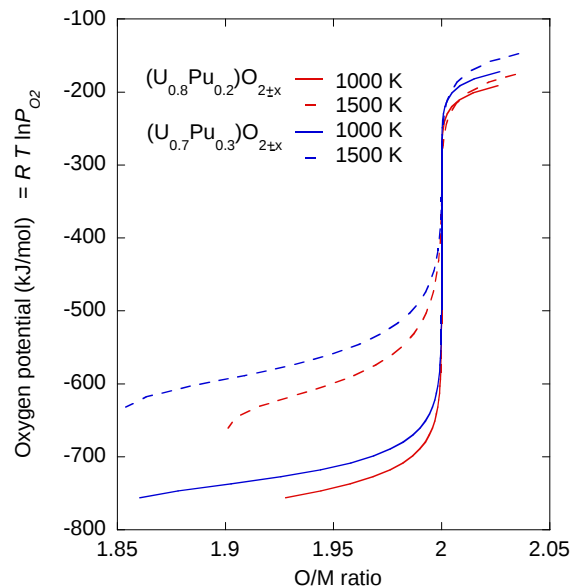


Fig. 10 Oxygen potential of $(U_{0.8}Pu_{0.2})O_{2.00 \pm x}$ and $(U_{0.7}Pu_{0.3})O_{2.00 \pm x}$. The oxygen potential of MOX as indicated was measured as functions of temperature, Pu content and O/M ratio. The value increases with increasing temperature, Pu content and O/M ratio. Kato M, Watanabe M, Matsumoto T, Hirooka S, and Akashi M (2017) Oxygen potentials, oxygen diffusion coefficients and defect equilibria of nonstoichiometric $(U,Pu)O_{2 \pm x}$. *Journal of Nuclear Materials* 484: 424–434.

Increased Pu content increases the oxygen potential. Oxygen potential (or the partial molar Gibbs free energy of oxygen in a thermodynamic system), ΔG_{O_2} , is an important thermodynamic property that can be described by the following equation.

$$\Delta G_{O_2} = RT \ln P_{O_2}$$

Here R is the gas constant, T is temperature and P_{O_2} is oxygen partial pressure. The oxygen potential of MOX can be evaluated from O/M ratio, Pu content, and temperature, and then the chemical stability of FPs can be known. Fig. 11 (an Ellingham diagram) shows the temperature and oxygen potential conditions of stability of for oxide compounds of fission products together with the oxygen potential of MOX. The figure which FPs will be stable in MOX as metal and which as oxide; at a given temperature, fission product oxide reactions with values of oxygen potential less negative than that for a MOX compound will be stable in MOX as a metal, while those with oxygen potentials more negative than a MOX compound will be stable as the oxide. For example, the oxygen potentials of $(U_{0.8}Pu_{0.2})O_{2.00}$ and $(U_{0.8}Pu_{0.2})O_{1.97}$ were shown to be about 650 and 400 kJ/mol, respectively, at 1000 K. Therefore, according to the diagram, molybdenum should be stable as MoO_2 and Mo(Metal), respectively, in $(U_{0.8}Pu_{0.2})O_{2.00}$ and $(U_{0.8}Pu_{0.2})O_{1.97}$. This highlights the importance of MOX O/M ratio to the chemical state of fission products, which affects the behavior of those fission products in the fuel.

Fabrication

Fuel fabrication necessarily must produce pellets that meet fuel specification values of pellet density, diameter, appearance, fissile ratio, impurities, and so on, to ensure proper fuel performance. The characteristics of raw material powders and the production process have been improved from conventional methods obtain stable, good pellets (Kato et al., 2020).

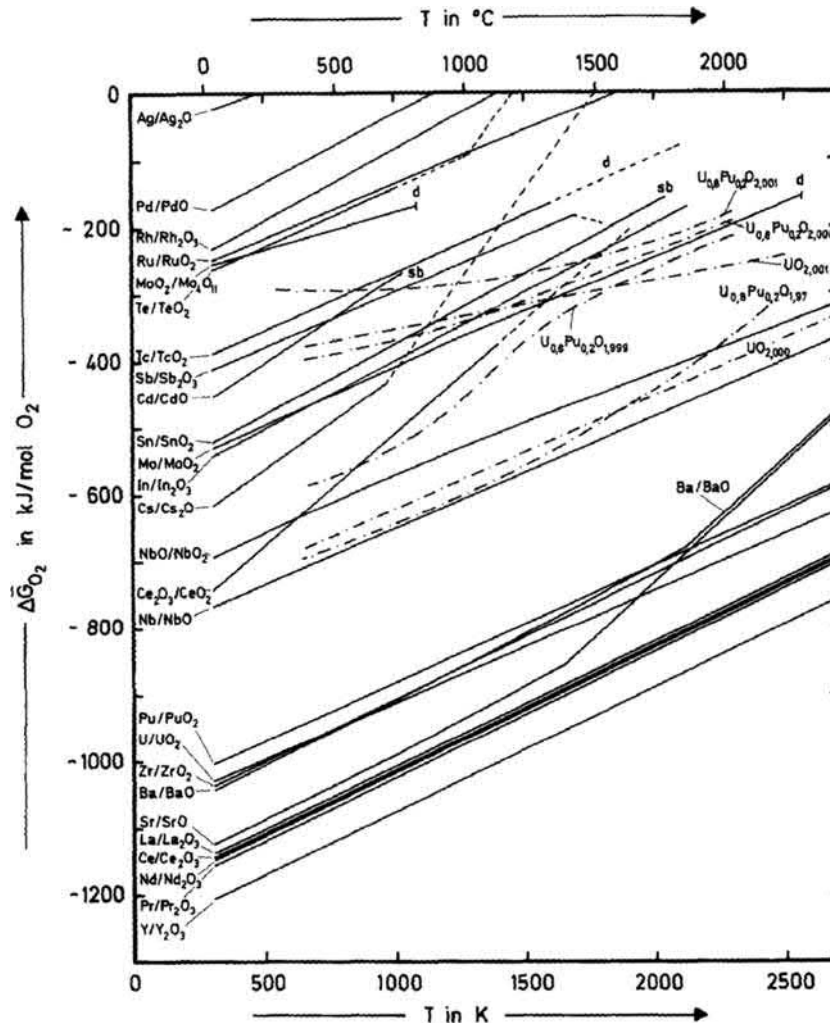


Fig. 11 Relative partial molar Gibbs free energies of oxygen of the fission products. The temperature dependence of partial molar Gibbs energies of oxidation reactions in various materials is shown with lines, which is called Ellingham diagram. Chemical stability of FP's can be evaluated by comparison with oxygen potential of oxide fuels (i.e., at a given temperature, FPs with values less than those of a MOX compound are stable as metals, while those with values less than those of MOX compounds are stable as oxides). Kleykamp H (1985) The chemical state of the fission products in oxide fuels. *Journal of Nuclear Materials* 131: 221–246.

UO₂ pellets

The LWR UO₂ pellet is produced with 95% of theoretical pellet density (TD), and a dish and chamfer are formed on the ends of the pellet, as shown in Fig. 12. These features are incorporated to suppress Pellet-Clad Mechanical Interaction (PCMI) due to thermal expansion and swelling of pellets. In the pellet production process, raw powder is granulated, blended with additive, pressed, and sintered into pellets. Then the outer periphery of the pellet is ground to final pellet dimensions. UO₂ pellet sintering is carried out at about 1973 K in pure gaseous hydrogen and the O/M ratio is adjusted to 2.00. Raw UO₂ powder with excellent flowability and sinterability has been developed and used successfully in UO₂ fabrication. Pellets of higher than 95%TD can be obtained with the use of powders with higher specific surface area.

MOX pellets

MOX pellets for fast reactor fuels are adjusted to the low O/M ratio of 1.98 to prevent oxidation of inner surface of the stainless steel cladding tube, improving fuel and cladding compatibility. The O/M adjustment is accomplished in the sintering process, which is performed in Ar and H₂ gas mixture. Plutonium spots or inclusions in the pellet are an important feature and are limited in the fuel specifications. Plutonium inclusions cause localized hot spots in the fuel pellet and may melt the fuel because a localized higher-than-average fission density is caused by a localized concentration of fissile plutonium atoms. Therefore, the size of the Pu inclusions in the cross section of the pellet as measured by alpha-radiography and Electron Probe Micro Analysis is strictly specified in the fuel specification. The sintering rate of the pellet is also affected by O/M ratio during heating (Nakamichi et al., 2020, Takeuchi et al., 2011). Fig. 13 shows pellet density change with O/M = 2.00 and 1.99 during the sintering process (Nakamichi et al., 2020). The

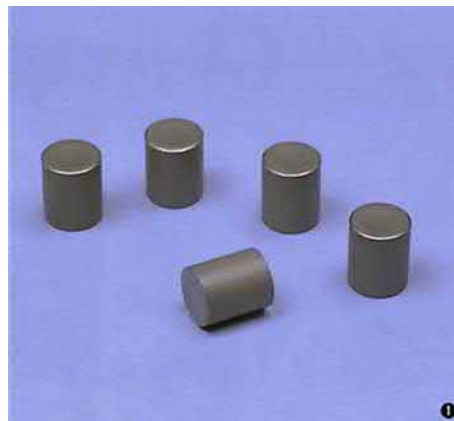


Fig. 12 Appearance of UO₂ pellets. Image available at https://www.ndc-tokai.co.jp/enterprise/index_fuel.html

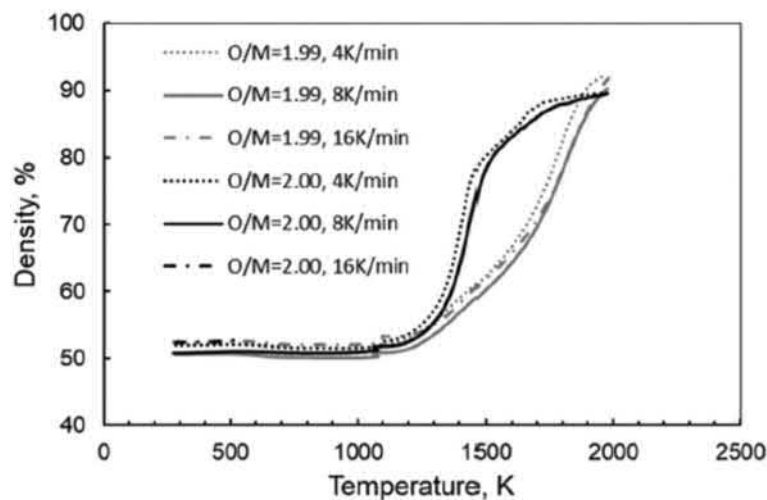


Fig. 13 Theoretical density ratio (%TD) of MOX pellets for different temperatures and heating rates. Nakamichi S, Hirooka S, Kato M, Sunaoshi T, Nelson AT, and McClellan KJ (2020) Effect of O/M ratio on sintering behavior of (Pu_{0.3}U_{0.7})O_{2-x}. *Journal of Nuclear Materials* 535: 152188.

sintering rate of stoichiometric MOX is fast and can result in a homogeneous pellet. The activation energy of sintering was evaluated to be 919 and 482 kJ/mol for MOX pellets with O/M = 2.00 and 1.99, respectively. The analysis results from the isothermal sintering test showed that grain boundary diffusion contributes to the sintering of stoichiometric MOX, whereas the sintering mechanism of low O/M pellet was unclear (Takeuchi et al., 2011).

Nitride and carbide pellets

In manufacturing of nitride and carbide pellets (Solntceva et al., 2016; Sengupta et al., 2012), carbo-thermic reduction is generally employed to produce nitride and carbide, as described by the following reactions, respectively.



Uranium and plutonium mixed oxide powders are blended with carbon and heated in N₂ and Ar to obtain nitride and carbide, respectively, and then pressed and sintered into pellets. The materials can be also produced by a hydriding method:



Both materials easily react with oxygen to provide high heat generation. Therefore, the materials must be produced in a glove box filled with Ar gas to prevent ignition by reaction with air.

See Also: Fuel Rod, Fuel Assembly, and Reactivity Control Assembly Design.

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Fuel Design and Fabrication: Fast-Reactor Metal Fuels

Douglas L. Porter and Douglas C. Crawford, Idaho National Laboratory, Idaho Falls, ID, United States

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Introduction

The topic “Design and Fabrication of Fast Reactor Metal Fuels” is quite broad, so the approach used here is to start this article with a short history of the use of metal fuels. Using that as a background, the design and fabrication of a metallic nuclear fuel element is discussed by describing the process used for one particular fuel type, a driver fuel pin (or rod) for a sodium-cooled fast reactor. The 40-year evolution of this fuel type provides a helpful illustration of the technical issues addressed in fuel design as needed to achieve acceptable fuel operation. Although a number of early nuclear reactor designs used metal fuel, including research reactors, the designs described here are for fast reactors.

There is a process to choose the basic fuel type, based on reactor operating conditions (dictates choices for fuel and cladding materials), potential fuel/coolant incompatibilities, required life expectancy (fuel burnup, cladding irradiation exposure, etc.), and other consideration. Knowledge of the reactor operating conditions and any special restrictions as to size or shape influenced by the reactor design will influence the choice of fuel alloy composition and/or fabrication methods. Availability of fabrication facilities and vendors, feed materials, and hardware also influence design choices when dealing with nuclear fuels. Requirements for unique materials or fabrication processes can limit options considerably. These matters, along with experience gained from using a particular fuel design constrain, or even dictate, fuel design characteristics as much as performance requirements.

Design: Fuel alloy and fuel pin design to optimize performance

A subsequent section of this chapter describes the design and fabrication process with a uranium-alloy fast-reactor metallic fuel. Other metals, plutonium and thorium, have also been used in the past as the base metal for the alloy, but most prior experience is with uranium-based alloys. Within that description, the fuel performance features which influence the design and fabrication choices are outlined.

But first, presenting a brief history of metal fuel usage is helpful, followed by fast-reactor fuel designs and use. The final sections of this chapter describe the developments expected to come next—new designs, fabrication technologies, tailored fuel compositions, concern over waste issues, etc.

Metal fuel use, a short history

Originally, unalloyed metallic uranium was used as fuel, but too many applications had problems because of its unpredictable behaviors in lower temperature operation. The alpha phase of uranium is prone to swelling problems, so alloying elements were added to stabilize the gamma phase, which has more-predictable behavior due to its highly symmetric cubic crystal structure. Molybdenum was the most popular alloying element, because it provided the desired gamma-phase stabilization and resulted in suitable solidus and liquidus temperatures that were low enough to facilitate fabrication and high enough to provide in-reactor temperature margin to melting. A variety of reactor types utilized U-Mo metallic alloy fuel in the past. Many of the early pulsed

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reactors used U-Mo, two with low-alloy alpha-phase U-1.2Mo (alloy compositions are expressed in weight percent), but another five reactors were fueled with an alloy with stabilized gamma-phase (U-10Mo) (Horak et al., 1973). The low alloying of U-1.2Mo was used to produce a small grain size, to reduce the effects of material property anisotropy prevalent in the alpha phase fuels. The higher alloying of U-10Mo eliminated any anisotropic behavior.

Other applications of alloyed fuel included a power reactor, the Organic Moderated Reactor was cooled and moderated by an organic liquid at the Piqua Nuclear Generating Station in Ohio, which was built and operated between 1963 and 1966 as a demonstration by the Atomic Energy Commission (Baumeister and Wilde, 1960). Another, the Hallam Nuclear Power Facility (HNPF), a graphite-moderated sodium-cooled reactor, which operated from 1962 to 1964 in Nebraska, used fuel that consisted of U-10wt% Mo cylindrical rods sodium-bonded to stainless-steel cladding. Fuel development and testing were carried out by Atomics International in the Materials Test Reactor (Willard and Schmitt, 1964).

Some early fast reactors employed U-Mo alloys as the standard driver fuel, and these included the Dounreay Fast Reactor (DFR) in the United Kingdom and the Enrico Fermi Fast Breeder Reactor (EFFBR) in the United States. One of the central reasons to use alloyed uranium in these applications was to control the fuel dimensional changes caused by gas swelling and phase transformations. This was very important for fast reactor applications where fuel temperatures were high (500–800 °C), so the fuel could operate with and be cycled between the low-temperature alpha and high-temperature gamma phases (Cottrell et al., 1964).

The EFFBR was the first commercial fast reactor, located in Monroe, Michigan. A 200-MWt (60-MWe) three-loop design, it was approved by the Atomic Energy Commission in 1955. The fuel was U-10Mo, sodium-bonded to Zircaloy cladding. The geometric design of the fuel assemblies was a 12×12 array of 4-mm diameter fuel elements (or fuel rods). The arrays were held in 67.2-mm square stainless-steel ducts, which are analogous to the square channels used around boiling-water reactor fuel bundles.

The EFFBR fuel prototypes were tested thoroughly before use in Argonne's CP-5 reactor to demonstrate that the U-10Mo alloy could perform to the reactor design specifications. The studies indicated that the primary concern was to ensure that the fuel would maintain the gamma phase during operation. U-10%Mo is now known to accomplish this. The goal of the studies was to provide the best dimensional stability of the fuel.

Alloy choice

The initial, early choice was unalloyed uranium. Uranium exists as several major phases depending on T — α , β , and γ . The low-temperature crystal structures have much less symmetry than the highest temperature form, the gamma (γ) phase. Gamma uranium has a body-centered cubic structure, so most all of its properties, physical and mechanical, are isotropic. The thermal expansion as a function of temperature is the same in all orthogonal dimensions. The alpha and beta phase do not have this symmetry.

It is easier to predict the performance of the cubic phase, so alloying was used to bring the stability of the γ phase at lower operating temperatures. As mentioned earlier, molybdenum was the most common alloying ingredient (Horak et al., 1973; Baumeister and Wilde, 1960; Willard and Schmitt, 1964; Cottrell et al., 1964; Shoudy et al., 1963).

Early studies showed that the fuel growth caused by fission gas generation, or due to thermal cycling, could be very surprising (see Fig. 1) if the fuel pin had an anisotropic crystal structure (from rolling, as shown in the figure) (Chiswick, 1956). In Fig. 1, the material had been rolled to varying degrees of deformation (3–74%) at 300 °C (in the alpha regime), resulting in a crystallographic texture. When thermally cycled in the alpha temperature regime, where the thermal expansion coefficients vary with the crystal orientation, a preferential growth was seen. This was in the absence of irradiation. When irradiated, the growth caused by fission gas bubbles showed similar anisotropy. Because of these phenomena, the isotropic gamma phase was desired, and the early choice of alloying elements were all gamma phase stabilizers, such as Mo.

Note that in Fig. 1, increased deformation (reduction in area) created a crystallographic texture. Smaller grain size enhanced the anisotropic growth. Note also that the decrease in grain size created no such effect if there were no preexisting texture induced by rolling.

After the early transitions of unalloyed uranium to U-10Mo, the emergence of fast reactors influenced alloy design. Mo remained as an option, as shown by the choice of Mo for EFFB. Other options were Zr or Ti, and also a fictitious element, fission (Fs), which was a combination of fission products that were expected to remain after reprocessing. Fs is dominated by Mo and Ru. U-Fs and U-Pu-Fs were studied, and Experimental Breeder Reactor-II (EBR-II) operated for many years using U-5%Fs fuel, including reprocessed fuel (Walters et al., 1984).

In later years, and with the advent of the Integral Fast Reactor program, the reprocessing scheme changed and design changes improved fuel performance, where higher temperature operation and enhanced fuel pin lifetimes were expected. The use of Fs was abandoned. With high-temperature operations, fuels with other characteristics now have more popularity. Uranium-zirconium alloys, specifically U-10wt%Zr, have been used, because Zr increases the melting temperature of the fuel and also seems to provide more protection against fuel-cladding chemical interactions (FCCI). FCCI was expected to be one of the most important factors influencing fuel pin lifetime and therefore influenced fuel alloy design.

FCCI is the mixing, or interdiffusion, of cladding components with the fuel alloy and attendant rare earth fission products (La, Ce, Nd, Sm, and Pr) (Matthews et al., 2017; Keiser, 2012). The zirconium alloys also have an increased melting temperature compared to unalloyed uranium. The things to most avoid in fuel performance are fuel melting and a large number of cladding breaches, and both phenomena are aggravated by FCCI. FCCI creates interdiffusion regions in the fuel and in the cladding, layers of intermetallic compounds. The layers effectively reduce the load-bearing thickness of the cladding by producing weak and brittle

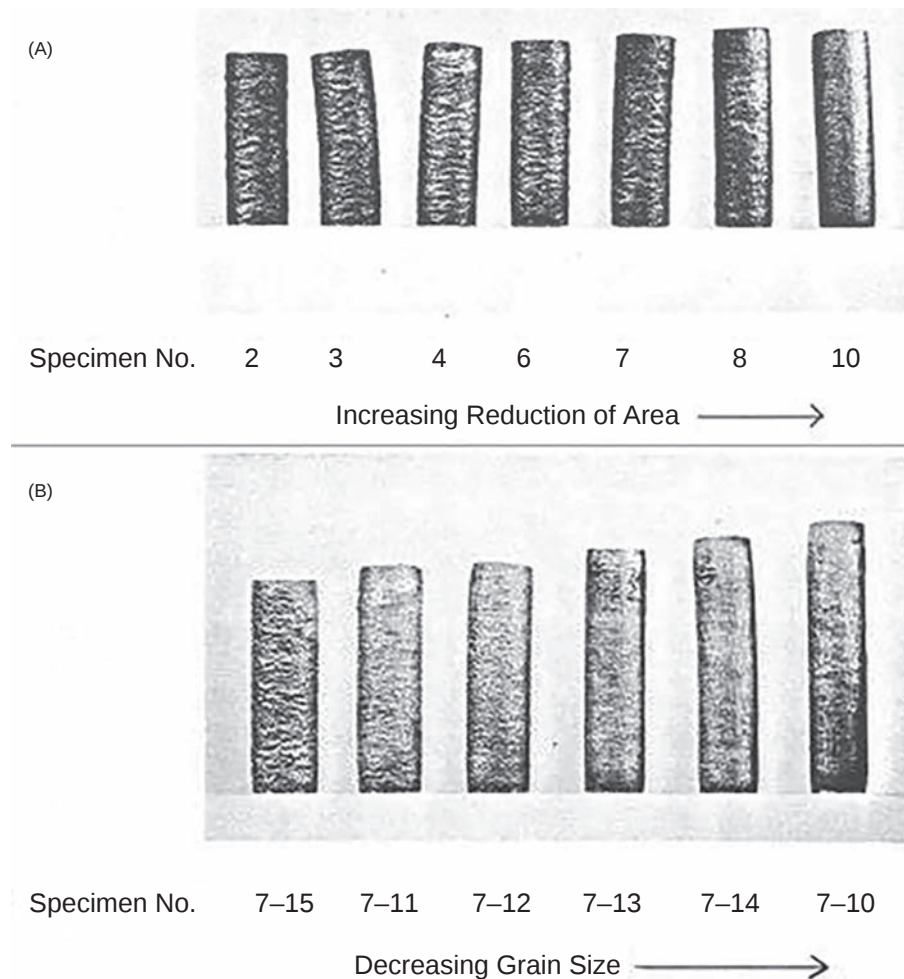


Fig. 1 Photographs from a 1956 study (Chiswick, 1956) of cylinders of uranium thermally cycled 500 cycles from 50 °C to 550 °C after being rolled at 300 °C to a given area of reduction (A) 3–74%, (B) 70%. In (B) the reduction in area was a fixed 70%, but the grain size varied. All samples began at the same 2.5 cm length.

layers at the fuel/cladding interface. Alloying the fuel with zirconium seemed to help mitigate the formation of these layers. Future methods for the mitigation of FCCI are discussed later in this article.

Design and fabrication of sodium-cooled fast reactor (SFR) metal fuels (EBR-II example)

Fuel pin design

As mentioned in the discussion of fuel alloy choice, engineers and scientists responsible for the EBR-II reactor experimented for many years, choosing a fuel alloy that contained a mixture of elements representative of the fission products carried over when using the fuel reprocessing method of the time. The alloy was U-5Fs, where the 5wt%Fs represented alloying contents 2.4%Mo, 1.9%Ru, 0.3%Rh, 0.2%Pd, 0.1%Zr, and 0.01%Nb. Fabrication methods were improved to avoid fuel texturing. The pin design was changed to incorporate a 75% smeared density (calculated by the cross-sectional area of the cylindrical fuel slug divided by the area inscribed by the inner diameter of the cylindrical cladding). The lower smeared density allows the porosity forming in the fuel to connect into a network before the fuel swells to contact the cladding. Fission gas is then easily released, reducing the stresses applied by the swelling fuel. Dimples in the cladding, designed to prevent fuel slug lift-off inside the cladding, were found to shorten the lifetime of fuel pins (they were locations of enhanced stress in the cladding) and were removed. Eventually, lift-off was proved to not be a problem. This history is described in (Walters et al., 1984) and (Crawford et al., 2007). The resultant design, emerging in the final decades of EBR-II operation, is shown in the sketch in Fig. 2. The smeared density concept is shown in Fig. 3.

Fig. 3 shows the cross-section of the fueled area. The smeared density is defined as the cross-sectional area of the fuel slug divided by the area inscribed by the cladding (fuel slug area + bonded-sodium area). In this case, it is drawn to represent 75%, the usual cross-section for EBR-II and Fast Flux Test Facility (FFTF) metal fuel pins.

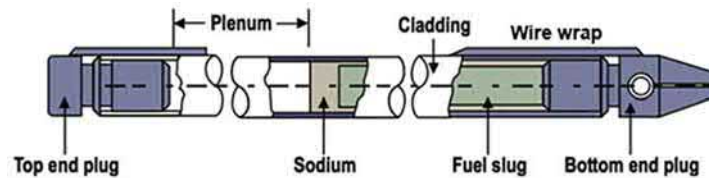


Fig. 2 A sketch of a sodium-cooled fast-reactor metal fuel pin.

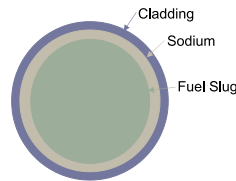


Fig. 3 A cross-section of fuel pin in the fueled area. This design has a 75% smeared density.

Sufficient volume is provided in the plenum space above the fuel column to accommodate the fission gases that are released from the fuel during operation. The fuel is “bonded” to the cladding with sodium (i.e., sodium fills the gap between the fuel slug and the inner cladding surface), to provide for the thermal transfer from the fuel to the cladding until the fuel has swollen to make physical contact with the cladding. The sodium coolant in the reactor flows through a bundle of the pins contained within the hexagonal duct of the assembly. The wire wrap is spiraled around and down each pin. The wire diameter and pitch are designed to maintain spacing between pins and mix the reactor coolant radially as it flows axially from bottom to top of the fuel assembly. The fuel pins were attached to the fuel assembly in a hexagonal grid, allowing close packing but minimizing constraint by the single point of attachment. The slotted lower end plug is slipped onto grid bars (see Fig. 4). This design not only held the pins securely but made the disassembly and reassembly of the fuel bundle possible. Test pins could be examined after a period in the reactor, and the assembly could be rebuilt in new assembly hardware and continue irradiation. The fuel pin design proved to be simple but effective both for fabrication efficiency and experiment functionality. The efficacy of the design was confirmed in larger geometries by a set of irradiation experiments in the FFTF in the United States (Pitner and Baker, 1993).

Fabrication

With the simple design chosen, all that remains is prescribing the best methods to use in the fuel pin fabrication process. Of course, the fabrication method development progressed along with the design evolution. Eventually, a process involving induction melting

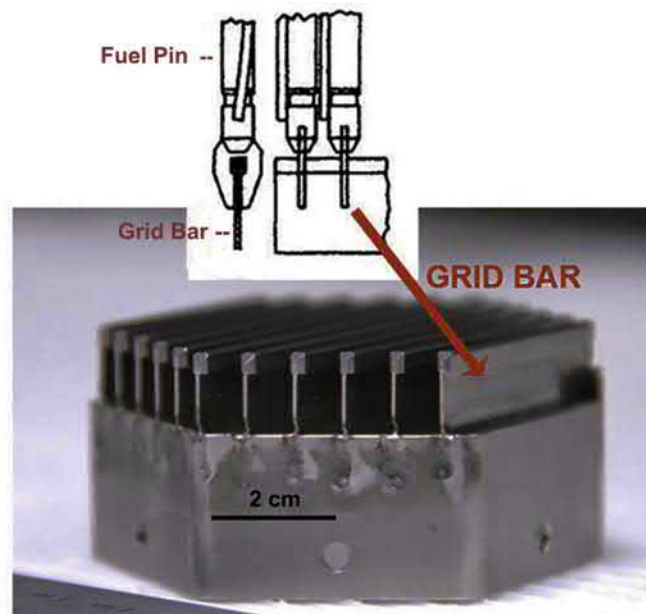


Fig. 4 An assembly of fuel pins on the assembly grid bar, allowing for the easy construction of a tightly packed hexagonal bundle of pins. It also allowed the interim examination of test pins (Burkes et al., 2009).

(in a single stage) was chosen, followed by injection casting (Burkes et al., 2009). The injection casting is demonstrated in the sketches shown in Fig. 5. The charge materials are loaded into the crucible and placed in the furnace. The furnace is evacuated, then the charge is melted and stirred both by the induction fields generated by the induction heating coils and also, periodically, manually with a tantalum stirrer. Finally, the quartz molds, coated internally with zirconia powder and sealed at one end, are immersed into the molten uranium alloy, and the furnace is quickly pressurized with argon to quickly push molten alloy into the evacuated molds.

The slugs were demolded and sheared to length. One slug (for EBR-II use, two or three slugs for FFTF metal fuel experiments) was placed in each cladding tube (called a cladding jacket) that was sealed at one end with a bottom end plug and contained an extruded length of sodium. The sodium was briefly melted, and the fuel “settled” to the bottom of the jacket. A top end plug was welded to the upper end, sealing the fuel in the jacket.

The next step was sodium bonding, a process to heat the jacket and its contents at a temperature high enough (550 °C) to wet the outer surface of the fuel and the inner surface of the cladding. While at temperature, the fuel pins were mechanically impacted (dropped a short vertical distance) to settle the sodium further toward the bottom of the fuel rod to force out any gas bubbles trapped between the fuel and cladding. This ensured a good thermal contact between fuel and cladding. Heat generated by the fuel slug was efficiently transferred to the reactor coolant.

While there were some important design and fabrication method improvements made along the way, there were nearly 35,000 pins made from irradiated and reprocessed EBR-II driver fuel; all produced remotely in a hot cell and assembled into fuel pins and assemblies in the 1960s (1964–69). The processes used were very similar to those discussed here (Stevenson, 1987). Of course, there

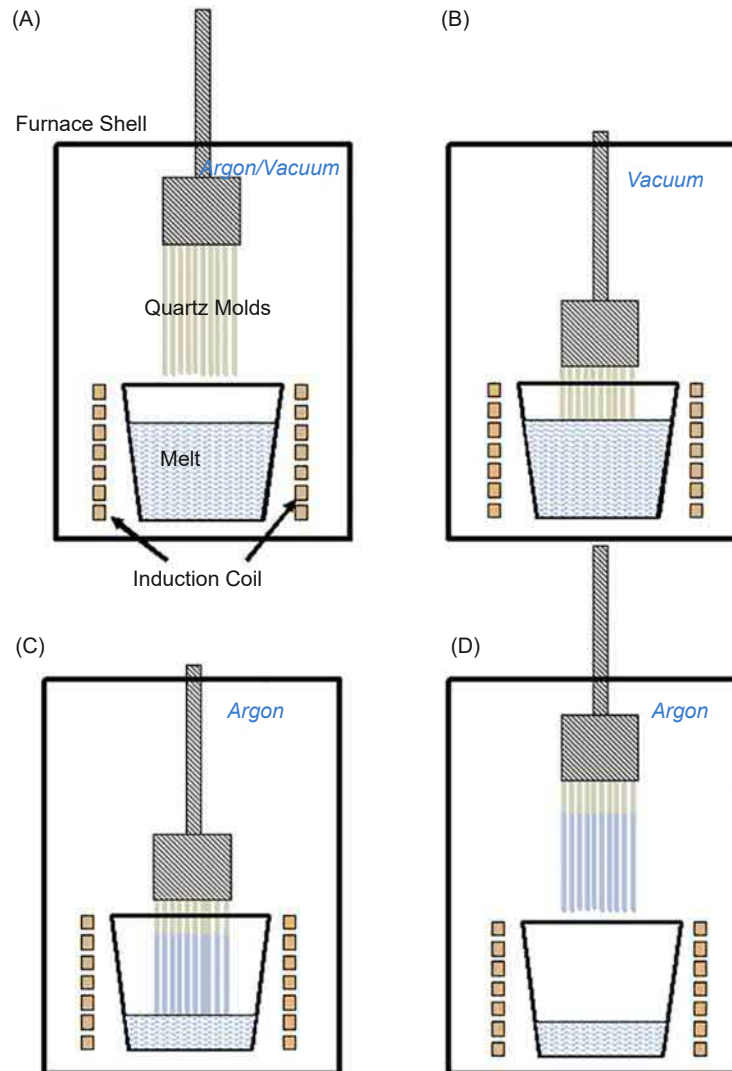


Fig. 5 The injection casing of metal fuel slugs. (A) A bundle of closed-end molds are suspended over the melt crucible while the charge is melted, (B) the furnace is evacuated and molds are inserted into the melt, (C) the furnace is re-pressurized, filling the molds, and (D) filled molds are held above the melt to cool.

were also many pins fabricated from unirradiated feedstock (167,000 U-5Fs pins as of 1987 at Argonne and 47,000 more at vendors, and another 16,000 pins of U-10Zr at Argonne) until EBR-II was shut down in 1994 (Wilkes et al., 1987; Burkes et al., 2009).

In the future

Designing to mitigate fuel/cladding chemical interaction (FCCI)

FCCI occurs when the fuel (U or Pu) and/or rare earth (RE) fission products (Nd, Ce, Sm, La, and Pr) interdiffuse with cladding constituents (Fe, Cr, and Ni). The intermetallic compounds that result form in layers in the cladding at the fuel/cladding interface. Because FCCI limits the useful lifetime of the metallic SFR fuel, there have been a series of proposed design modifications to mitigate FCCI. One approach inserts a diffusion barrier between fuel and cladding. Zirconium and other materials have been used, either as a liner to the cladding or a coating bonded to the inside diameter of the fuel cladding. Critics of this approach assert that there may be no certain means to ensure the 100% reliability (no breaks in the barrier) of these barriers and that they cannot be considered a mitigation.

Another approach is to use subtle changes in fuel composition to reduce FCCI. There are alloying elements (Sn, Pd, Sb) that form very stable intermetallic phases with the RE fission products (Mariani et al., 2011; Xie et al., 2018), the elements that dominate the FCCI effect in U-Pu-Zr fuels (Keiser, 2012). This effectively reduces the amount of REs available to penetrate the cladding. The formerly used U-5Fs alloy, for which the 5 wt% Fs was a mixture of noble metal fission products (2.4%Mo, 1.9%Ru, 0.3%Rh, 0.2%Pd, 0.1%Zr, and 0.01%Nb), demonstrated some of these effects, showing the FCCI to be dominated by uranium diffusion into the cladding and less RE penetration. REs formed intermetallic compounds with Pd in the U-5Fs (Harp et al., 2020).

The challenge with alloy additions as a means to mitigate FCCI is to perfect the delivery of these elements to allow for the formation of the intermetallic phases through prescribed fabrication techniques and quantities of additives, without allowing an interaction with the alloy, which would minimize their effectiveness and remove Zr from solid solution and thereby reducing the fuel solidus temperature. In addition, it remains to quantify the alloying addition's beneficial effect on FCCI. The use of stronger cladding, such as oxide dispersion strengthened alloys, may also be beneficial, because it is the loss of strength as the cladding thins due to FCCI that allows an early approach to a strain or cumulative damage fraction limit (Briggs et al., 2018). The benefit of stronger cladding material can be easily incorporated into the design limit analyses. Preventing any liquid phase formation with the fuel remains important in this case as well.

That kind of FCCI (melting) also limits fuel operation. That FCCI is eutectic melting, which occurs when the fuel materials, U and Pu, interact/interdiffuse with the cladding, forming low melting compositions. Pu bearing fuels are often limited to a 650 °C peak cladding temperature to prevent this melting behavior. Alloying additions to the fuel and stronger iron-based cladding materials will not prevent this reaction. So perhaps the best, most efficient barrier to FCCI of all kinds is a barrier material inserted between fuel and cladding. Zirconium and vanadium have been effective in preventing the penetration of both fuel and fission products into the cladding (Cole and Keiser, 2007). The challenge for the use of barriers (whether incorporated as tubes, foils, or coatings) is their fabrication with the cladding to ensure a continuous, unbroken barrier layer. So, the co-fabrication of the coatings and liners with the cladding is very important. There have been a number of recent studies to select effective coatings or liners and to find effective fabrication techniques (Ryu and Lee, 2009; Firouzdor et al., 2011, 2013a,b).

Spent fuel storage

One of the other problems with current SFR metallic fuel designs is the disposal of sodium-bonded used fuel. Evaluation by the US Department of Energy determined that neither indefinite storage of sodium-bonded fuel stored at the Idaho National Laboratory nor its direct disposal into a high-level waste repository was recommended because of the chemical reactivity of the sodium and uranium. It is therefore difficult to provide for a way to dispose of the used fuel. Electrochemical treatment to separate sodium from actinide constituents and to produce disposable waste form is the preferred disposition path in the United States for sodium-bonded used fuel. The most recent designs of SFR metal fuels use are 75% smeared-fuel density and are sodium bonded, so pre-disposal treatment is likely to remain a requirement for use of metal fuel.

However, if the sodium bond were not part of the used fuel then it would be less risky to store, and more disposal options could be considered. The problem is that the larger fuel-cladding gap filled with sodium is the design feature adapted for accommodating the swelling of the metal fuel without severely stressing the surrounding cladding. The sodium bond is therefore necessary, at least early in life, to transfer the heat from the fuel slug to the cladding and prevent the fuel slug from overheating and potentially melting.

The obvious alternative to the current design is, therefore, to use another thermally conductive liquid, or gas, something that will not transfer stress to the cladding as the fuel swells. Alternatives to sodium as the thermal bonding agent have been suggested and dismissed because they may cause cladding embrittlement or corrosion (Kamdar, 1983; Lynn et al., 1975). Potential transmutations of the alternative liquid metal, which could cause species to form that may attack the cladding, must also be investigated.

Another possible solution is to fabricate the fuel slug with a tight fit to the cladding and provide fuel space to accommodate fuel swelling elsewhere in the fuel slug. Two suggested alternatives are (1) slotted fuel where the empty space falls on the periphery of the fuel but only at specific areas around the circumference and (2) annular fuel with a fuel centerline void. Fig. 6 illustrates this first concept, using a computer model of fuel swelling to predict how the fuel would swell and contact the cladding.

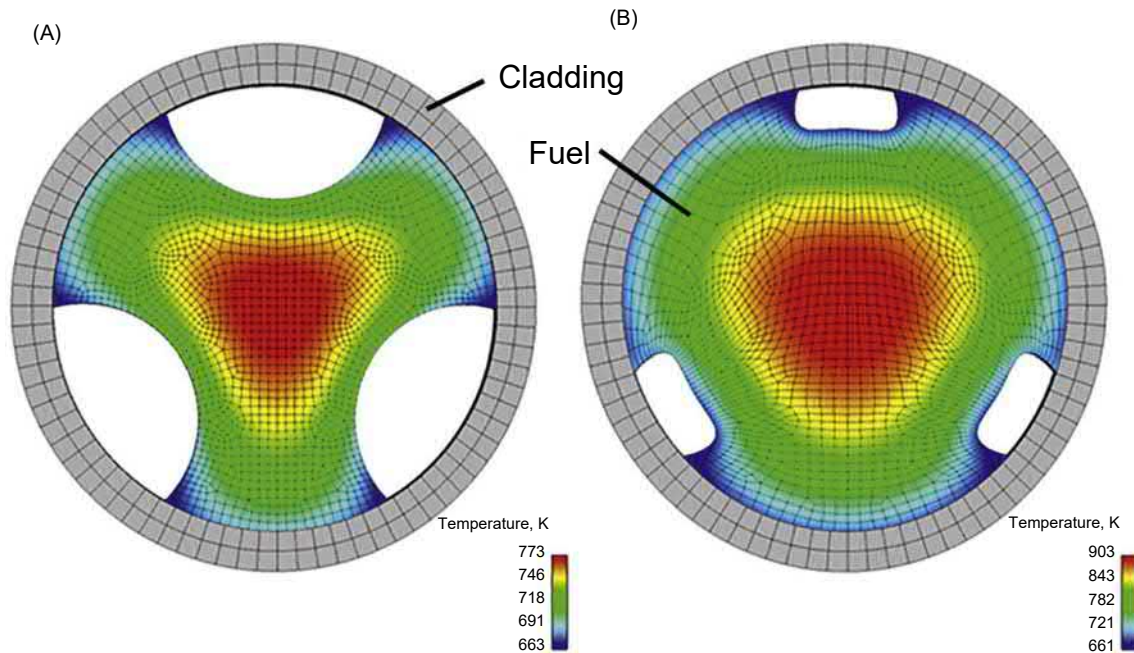


Fig. 6 A diagram of slotted fuel being modeled to show the temperature distribution and swelling as a function of fuel burnup. (A) Near the beginning of life and (B) after significant swelling. Modeling images courtesy of Dr. Pavel Medvedev, INL.

However, earlier work demonstrated that the fuel can cause the cladding to deform and become unround (Golovchenko, 2011). So further work is needed to test variants of the design for acceptability.

The second concept of a fuel design with no thermal bond material between fuel and cladding is the annular design. Again, the fuel slug must fit tightly within the cladding, but the smeared fuel density is reduced by placing a central void down the centerline of the fuel slug, sized sufficiently to allow for the formation of open porosity (to release fission gas to the fuel pin plenum) by the time the fuel swells to fill in the central void. But the central void is relatively difficult to produce in the fuel by casting techniques, because the small fuel cross-sections challenge mold construction. Fig. 7 shows a sketch of the cross-section of this type of fuel pin, with a 75% design, the same as shown in Fig. 3 for the current, sodium-bonded design.

One barrier to using this design is the difficulty in fabricating fuel slugs in long sections through traditionally used casting methods. Note also that, because the bond-free fuel designs require tight-fitting fuel slugs, the assembly of these designs (placing the fuel in the cladding) is also challenging. The annular fuel could be produced by extrusion or perhaps continuous casting. These alternative fabrication schemes, also suggested to minimize fabrication waste, will therefore be discussed in the next section as one of the new advances in fabrication.

Fabrication solutions

There are a variety of new reactor designs, some using very small fuel components, some very large with long fuel columns. All of these can contribute to new challenges in metal fuel fabrication. So, the development of new designs motivates rethinking the basic fabrication methods to address these new designs and minimize or simplify fuel manufacturing waste disposition.

Even the current methods of fuel fabrication, which may still be utilized in the future, need improvement. The current practice for injection casting leaves a significant heel in the bottom of the crucible, which must be recycled in a future casting. The heel also

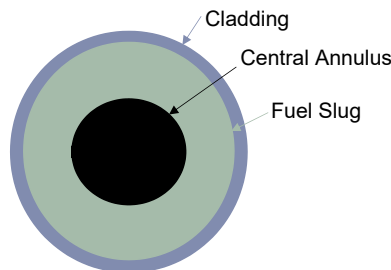


Fig. 7 A sketch of annular style fuel with a central void to allow for fuel swelling. This fuel is also represented at 75% smeared density.

picks up impurities, and, after re-melting several times using charges where about half of the feedstock is from the recycled heel, impurities in the fuel charge accumulate to approach impurity limits. At that point, the remaining heel, without costly processing to remove impurities, must be disposed as waste.

There are a number of ways to reduce casting heel waste. One possibility is to redesign the melting and casting process to minimize the casting heel, or even eliminate it with an alternative casting process, such as bottom-pour or perhaps even tilt-pour. A more drastic change would be to cast billets (bottom-pour) and then extrude them into the fuel slugs. Recall that this also eliminates the waste created when quartz molds are broken to retrieve the as-cast fuel slugs.

In fact, extrusion of uranium-based metal fuel for an SFR, EBR-I, was done in the late 1950s (Sawyer, 1958; Kittel, 1959). The Mark-III core was made up of U-2wt%Zr coextruded into Zircaloy-2 cladding. The compound billet (U-2Zr in Zircaloy-2) was extruded. But the mechanical working of the billet into a long cylindrical rod produced a crystallographic texture in the fuel, so the coextruded material was heat treated to remove as much texture as possible after extrusion. The texture can cause anisotropic swelling of the fuel (Sturcken and Walter, 1974).

Recently, extrusion development is underway again, working out a way to produce fuel from cast U-10Zr billets (Pace and Mackowiak, 2018). Similar work is investigating the coextrusion of a UZr alloy in zirconium tubing to produce U-10Zr surrounded by unalloyed zirconium. The goal this time, however, is not to use the coextruded product as fuel and cladding, or, as was studied in the early 1990s (Crawford et al., 1993), to use the co-fabricated product (UPuZr cast into a zirconium mold) as a fuel slug with stainless-steel cladding in a classic solid slug, 75% smeared density design. In that effort, the zirconium tubes were used to eliminate the need for quartz molds, and the resulting Zr layer around the fuel acted as an FCCI retardant. However, the zirconium cracked as the fuel swelled, so areas of fuel and cladding were exposed for interaction. The current efforts intend to extrude a multilayered annular fuel slug that will closely fit into the stainless-steel cladding. This also allows for the removal of bonded sodium and reduces casting slug mold waste. Fig. 8 shows a coextruded U-10Zr/Zr cross-section, and Fig. 9 shows an annular extruded fuel slug. The fuel pieces in the figures have not been heat treated or straightened, and the rough surface is an artifact caused by the use of a worn extrusion die. The best result would be to eventually combine these two design/fabrication features to produce an annular, tight-fitting fuel slug comprised of U-Zr fuel with a thin covering of zirconium. A thinner, uniform zirconium covering than that shown in Fig. 8 would be necessary.

The last waste difficulty that engineers/scientists are addressing is casting mold waste from traditional metal fuel manufacturing. The removal of quartz molds from as-cast slugs leaves the quartz broken into shards with small amounts of adherent fuel. The adherent fuel material complicates the disposal of this waste stream. Studies to find a reusable mold have thus far been unsuccessful (Chen et al., 1992; Shapiro, 1994). In one of these studies, various oxides, carbides, nitrides, and sulfides were used to coat refractory metal molds. Y_2O_3 was the only coating material that effectively prevented the interaction of the molten fuel with the refractory metal mold.

Bottom-pour casting is an idea worth perfecting because it eliminates the heel left in the injection casting crucible after the injection casting molds are filled. Furthermore, a reusable mold might be easier to design for a bottom-pour application.

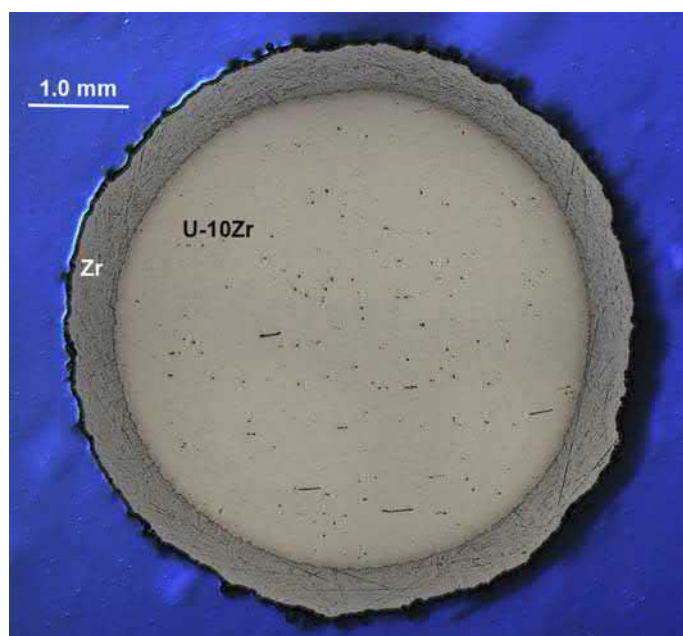


Fig. 8 A cross-section of coextruded U-10wt%Zr with zirconium. The studies are intended to find a way to simply to add a layer of Zr to inhibit fuel/cladding interactions.

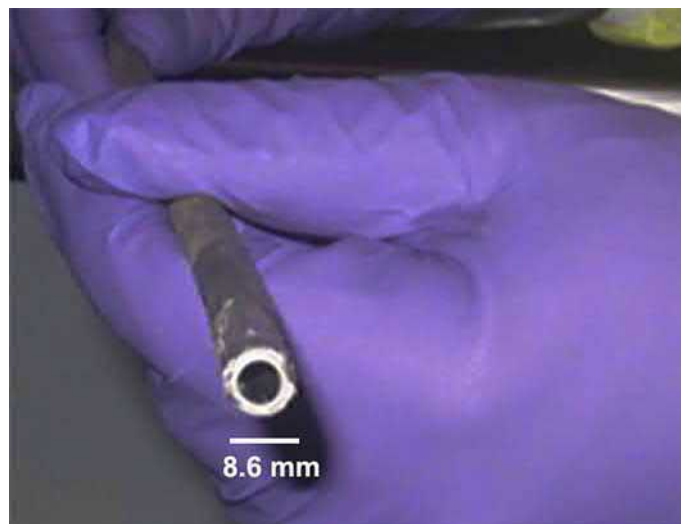


Fig. 9 The end of an extruded rod of annular U-10Zr fuel.

Some very early fuel designs, such as one of the EBR-I fuel designs, used coextrusion of the fuel into the cladding, but these were 100% smeared density and therefore had very low fuel burnup potential. These low-burnup designs are not practical today, but an annular coextruded design shown to be fabricable in usable diameters could be attractive. Particularly so, if the coextrusion process can place a zirconium FCCI barrier between the coextruded fuel and cladding. The tight tolerances on diameter and straightness and the inclusion of a central void are difficult for the small fuel diameters of interest today. In addition, the application of the technology to Pu-bearing or TRU-bearing fuel alloys, is also challenged by the need to operate and maintain the extrusion press within a radiation containment, such as a glovebox.

Conclusions

Experience with the design and fabrication of metallic fuels for sodium-cooled fast reactors now exceeds 60 years. That experience resolved initial barriers to high-burnup fuel use (addressing fuel swelling, fuel/cladding interaction), resulted in designs with relatively large tolerances on dimensions and fuel impurities, and established relatively simple fabrication methods. But the reduced interest in fast reactors in the 1990s removed the impetus to further improve fuel designs and fabrication methods.

But with renewed increased in the applications of fast-reactor technology with reduced waste and increased fuel utilization, significant changes appear possible to:

1. Minimize fuel (and RE fission product) interaction with stainless-steel (austenitic and ferritic/martensitic) claddings. This FCCI has proven to be the effect that limits the operating temperature and burnup of these fuel types.
2. Reduce the waste generated during fuel manufacturing by changing the casting methodology (smaller heels, reusable molds and crucibles, no molds—extrusion, continuous casting).
3. Simplify spent fuel storage, fuel treatment, or reprocessing technologies by removing the sodium bond from the fuel design.
4. Adapt the methods above to fabricate fuel containing plutonium, where the process containment is much more imperative.

See Also: Sodium-Cooled Fast Reactors.

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Fuel Design and Fabrication: TRISO Particle Fuel

Grant W. Helmreich and John D. Hunn, Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Introduction

The modern TRISO particle fuel design is the result of over 60 years of development spanning multiple countries, reactor designs, and research programs (Demkowicz, et al., 2019; Verfonderm et al., 1997). Fundamental features of the fuel design are the encapsulation of fission products in a robust barrier immediately surrounding the fuel and the consolidation of very small encapsulated fuel particles into a matrix that provides a heat transfer path from the fuel particles to the reactor coolant. The intended result is a fuel and fission product barrier that is so reliably robust that reduces reliance on subsequent barriers (fuel cladding, containment building) typical of other reactor and fuel systems.

TRISO fuel design

The concept for a microsphere dispersion fuel with coatings for fission product retention during operation was first developed as a part of the Dragon project in the 1950s (Price, 2012). Evolving from the initial design with a single pyrolytic carbon coating, various additional coating layers were added to the fuel design to improve the consistency with which the particle coatings retained fission products within the fuel (Dayton, et al., 1964; Goeddel, 1964, 1967; Carlsen Jr., et al., 1964; Stansfield, 1991). These innovations culminated in the 1970s with the emergence of the modern TRISO particle, featuring ceramic uranium kernels encapsulated in four distinct coating layers (Verfonderm et al., 1997).

At the center of TRISO particles is a roughly spherical uranium-bearing kernel. The composition and size of these kernels varies with reactor design; however, they are typically composed of either UO_2 , UCO (a heterogeneous mixture of uranium oxide and uranium carbide), or UN with a diameter of 350–800 μm (Stinton, et al., 1982; Phillips, et al., 2012; Lu, et al., 2018). Each of these kernel materials features unique advantages and drawbacks, such as enhanced immobilization of rare earth fission products (UO_2), reduced internal pressure due to reduction of carbon monoxide gas formation (UCO), or increased density of fissile material (UN). As such, kernel composition and size may be tailored for the specific goals of specific reactor designs for which TRISO particle fuel is considered. The kernel itself functions as the first line of containment for solid fission products, in addition to providing the fissile material necessary for the generation of nuclear power, as many fission products are thermochemically stable within the ceramic uranium systems.

While there is significant variation in kernel size and composition between various TRISO particle designs, the thickness and composition of the four coating layers are much more uniform. The nominal layer thicknesses from the reference particle design of the 1970s German AVR program (given in Table 1) have been largely adopted by later designs (Kania, et al., 2013).

The first layer encapsulating the kernel is the buffer layer, which consists of approximately 50% dense pyrocarbon and serves two primary functions. First, the buffer layer provides void space to accommodate fission gases released from the kernel and kernel swelling due to solid fission products. Second, the buffer layer provides distance between the kernel and the other coating layers to prevent concentrated damage in those layers from high-energy fission product recoils. Next is the inner pyrolytic carbon (IPyC) layer. Unlike the buffer layer, the IPyC layer is highly dense so that it functions as an effective barrier to gasses and diffusion

Table 1 Nominal layer thicknesses of German AVR reference TRISO particle design.

Layer	Thickness
Buffer	95 μm
IPyC	40 μm
SiC	35 μm
OPyC	40 μm

of many of the actinides and fission products. The IPyC layer is responsible for contributing to retention of fission gasses during irradiation and protecting the kernel from HCl vapor, which is a byproduct of the silicon carbide (SiC) layer deposition process. The overlaying SiC layer critically serves as both the primary source of structural strength for the particle and the primary barrier to the release of fission products which are not retained within the kernel and IPyC layer. Finally, the outer pyrolytic carbon (OPyC) layer protects the SiC layer from handling damage and serves as a final barrier to fission product release. An image of a TRISO particle with a portion of the coating layers broken away to show the particle's interior is provided in Fig. 1.

The final TRISO particle is roughly spherical, with a design-dependent diameter of 750–1200 μm . These particles are typically compacted along with a matrix material into fuel forms containing thousands of particles. The shape and matrix material depend on the specific reactor design, but most commonly they are spheres or cylinders, as shown in Fig. 2. Various research programs have considered further adaptations of TRISO particles by substituting different layer materials or adding entirely new layers (Verfondern et al., 1997); however, this overview focuses on the standard TRISO design.

TRISO fuel fabrication

Fuel particle fabrication

The first step in TRISO fuel fabrication is to produce the kernels. This is most commonly performed using a sol-gel process with either internal or external gelation (Haas, et al., 1980; Beatty, et al., 1979). In this process, an actinide or actinide mixture is dissolved into nitric acid to form a precursor solution. This solution is generally made with an acid deficiency to allow for increased uranium concentration and to promote optimal gel-sphere properties. As such, in the most common case in which uranium is the sole actinide, this solution is commonly referred to as *acid-deficient uranyl nitrate* (ADUN). Gelation is triggered by the addition of ammonia ions to the ADUN solution either from an external source such as a vapor (external gelation) or from an internal source mixed with the ADUN, in which the reaction is controlled by pH and temperature (internal gelation). In either case, hydrated actinide oxide precipitates are formed into gel (semi-solid) spheres while passing through a heated liquid medium. Gel spheres are then washed with ammonium hydroxide to remove any residual gelation medium or reaction byproducts and to push any incomplete

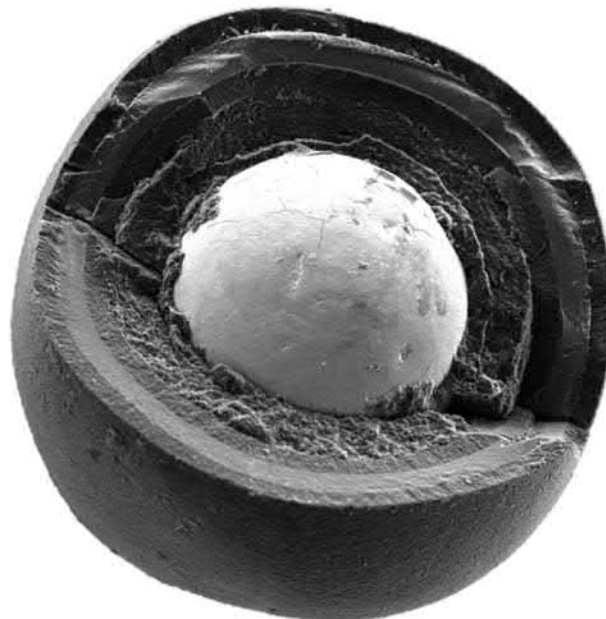


Fig. 1 Micrograph of a cracked TRISO particle with the kernel and coating layers exposed. Image supplied courtesy of X Energy LLC.

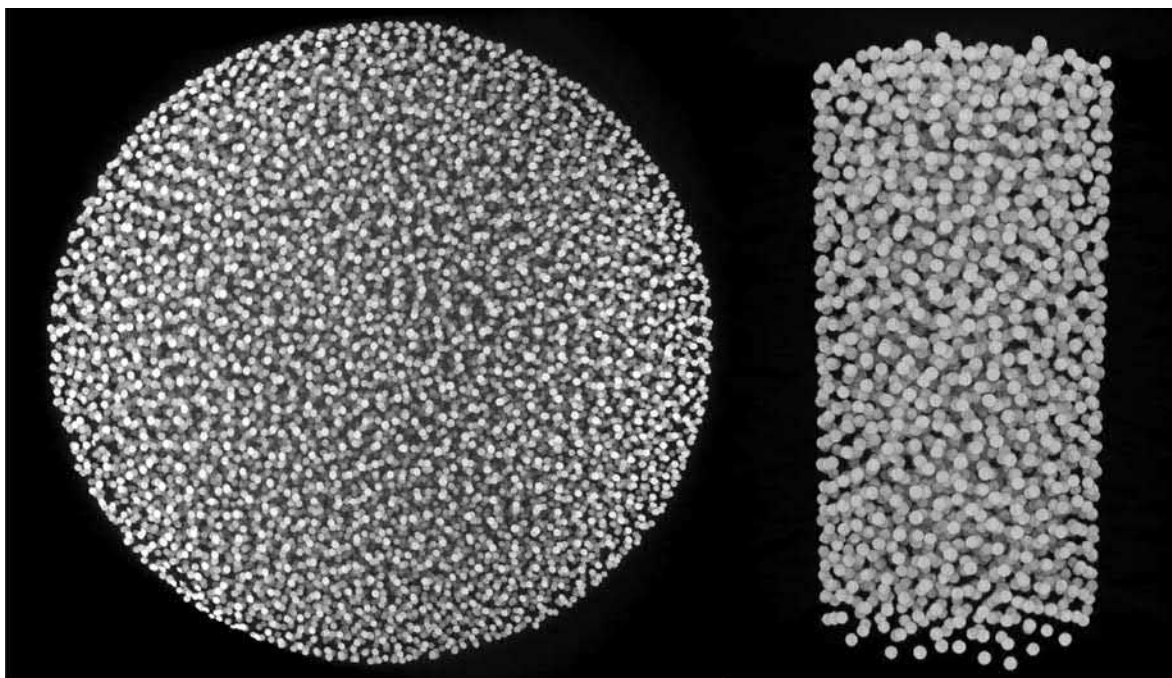


Fig. 2 3D image from x-ray computed tomography showing TRISO particles distributed within a pebble (left) and a compact (right) (Helmreich, et al., 2018a,b).

reactions to completion. After being washed, gel spheres are dried in either air or steam to remove water, and then they are subjected to thermochemical processing to reach their final desired composition. In the case of UO_2 kernels, this step involves calcining and sintering to convert from UO_3 . For UCO or UN kernels, carbon is dispersed in the initial ADUN solution to provide additional thermochemical reactions with process gasses to convert some or all of the UO_3 to UC, UC_2 , or UN (Stinton, et al., 1982; Hunt, et al., 2014). In all cases, significant shrinkage occurs during conversion from gel spheres to final kernels, so the target gel sphere's diameter must be scaled appropriately to result in a given target kernel diameter.

In general, the kernels fabricated by the sol-gel process using modern methods are round and near the targeted diameter; however, the stochastic nature of the process and the large number of kernels produced in a batch naturally result in some fraction of undesirable kernels. These low-quality kernels are commonly removed from the batch by upgrading based on size and shape (Hunn, et al., 2014). Kernels outside of the desired size range may be removed from a batch by either sieving or by using a roller sorter. If sieving is used, care must be taken not to chip or fragment the kernels by overloading or violently shaking the sieves. Aspherical kernels may be removed by tabling, in which kernels are rolled onto a tilted table in a direction perpendicular to the table's slope. Round kernels roll readily toward the tilt direction in a parabolic arc, while elliptical or faceted kernels will lose momentum and prematurely slide down the table.

After kernel fabrication and upgrading, TRISO coating layers are applied using a fluidized-bed chemical vapor deposition (FB-CVD) system. These systems use an inert gas (commonly argon or hydrogen) to fluidize a bed of kernels, causing them to churn and flow like a liquid, while reactive gasses are introduced to deposit layers onto the kernels. Continuous mixing of the bed promotes even deposition of coating layers by decomposition of reactive gasses. Various properties of the coating layers, such as density, deposition rate, and microstructure, depend upon the fractions of the reactive gasses used and the temperature of the FB-CVD system (Ford, et al., 1973; Hunn, et al., 2008).

The buffer layer is generally deposited using acetylene (C_2H_2), although methane (CH_4) has also been used. Since the buffer layer must be low density to accommodate kernel swelling and fission gasses, buffer deposition is performed at a high rate. Although the IPyC and OPyC layers are composed of pyrolytic carbon (PyC) like the buffer layer, they are required to be near fully dense. As such, differing coating gasses and conditions are used for these layers. A wide range of hydrocarbon gasses have been studied for dense PyC deposition. These gasses include methane, acetylene, propylene (C_3H_6), propane (C_3H_8), and butane (C_4H_{10}); however, modern PyC deposition is commonly performed with a mixture of acetylene and propylene. The primary process parameters controlling PyC properties are the deposition temperature and the partial pressures of reactive gasses (Lowden, et al., 2005). Finally, the SiC layer is deposited using a mixture of methyltrichlorosilane (CH_3SiCl_3) and H_2 , often with the addition of an argon diluent. Key properties of the SiC layer, including density and microstructure, are controlled by the interplay between deposition temperature and the ratio of Ar/H_2 . As with kernels, batches of coated TRISO particles are commonly upgraded for both size and shape using methods such as sieving, roller sorting, and tabling (Hunn, et al., 2014).

Fuel compact fabrication

TRISO particles may be consolidated into a wide range of fuel form geometries within a variety of matrix materials. The most common TRISO fuel forms are cylindrical compacts, which are used in prismatic reactor designs, and spherical pebbles, which are used in pebble-bed reactor designs (Pappano, et al., 2008; Nabielek, et al., 1984; Fütterer, 2021). In both cases, TRISO particles are typically overcoated with a blend of graphite flake and either thermoplastic or thermosetting resin. These overcoated particles (and potentially additional matrix material) are then pressed into the desired shape and heat treated to carbonize the resin, resulting in a compact of TRISO particles in a cylindrical or spherical graphite matrix. Additional features may also be added depending on the specific reactor design are being developed, such as a fuel-free zone on the outer surface or at the center, or an additional coating on the outside of the spherical pebbles to protect the pebbles from handling and collision damage. Alternative TRISO fuel form concepts have been developed, such as SiC matrix compacts (Lu, et al., 2018) and geometrically complex additively manufactured structures (Terrani, et al., 2019; Betzler, 2021).

TRISO fuel characterization and quality control

Statistics of TRISO quality control

Statistical analysis is central to quality control of TRISO particle fuels due to the enormous number of TRISO particles contained within a reactor and the destructive nature of many characterization methods. There are two statistical distributions which are applied to TRISO particle properties: the Gaussian distribution, which is applied to variable properties, and the binomial distribution, which is applied to attribute (pass/fail) properties (Kercher & Hunn, 2005; Hunn, et al., 2009). In both cases, representative random sampling is key to obtaining an accurate indication of quality. For this reason, characterization samples are generally taken from TRISO particle lots using highly randomized methods which sample the full material lot, such as chute or rotary riffing.

Variable particle properties such as kernel diameter are represented by the Gaussian distribution and are characterized by a mean and standard deviation. Determination of these values requires a sufficient number of measurements of the property in question to define a normal distribution. Once this criterion is met, the mean and standard deviation may be used to apply specifications on the mean, as well as the dispersion, with given confidence intervals based on the Student's t-test and tolerance intervals (Hogg & Craig, 1978). For example, the mean kernel diameter could be specified as $350 \pm 10 \mu\text{m}$ with 95% confidence with an additional specification on the dispersion of less than 1% below $300 \mu\text{m}$ and less than 1% above $400 \mu\text{m}$. A randomly representative sample of kernels from each composite batch would then need to be characterized to demonstrate compliance with this specification. Unless otherwise specified, TRISO particle properties described herein are variable.

Attribute particle properties such as defective IPyC are represented by the binomial distribution and are characterized by the number of defects observed in a given sample size. These two values may be used to calculate a maximum defect fraction with a given confidence based on the cumulative value of the binomial distribution (Hirsch, 1957). For example, the fraction of particles with defective IPyC could be specified to be less than $1\text{E-}4$ with a 95% confidence. Given this type of specification, the required sample size to pass may be estimated based on the binomial distribution and an assumed true defect fraction for the material. Due to the long-tailed nature of the binomial distribution and the statistical penalty that accompanies a high degree of confidence, the required sample size is typically much larger than the inverse of the targeted defect fraction. In the previous example of proving a $1\text{E-}4$ (1 in 10,000) defect fraction with 95% confidence, a sample size of nearly 30,000 particles would be required even in the best case of high-quality fuel with no observed defects. In the more likely case that some defects are observed, the required sample size increases even further (e.g., to 119,000 particles if six defects are observed).

Kernel characterization methods

Commonly specified properties for TRISO kernels may be divided into two categories: physical (e.g., kernel size, shape, and density) and compositional (e.g., isotopic and elemental content).

Measurement of physical kernel properties

Kernel diameter and aspect ratio may be measured using shadow imaging, in which a single layer of thousands of kernels is backlit and imaged with an optical microscope. The diameter and aspect ratio of each kernel may then be measured either manually or using automated image analysis software as shown in Fig. 3 (Kercher, et al., 2008). The shadow imaging method relies on the assumption that imaged kernel cross sections are representative of the full three-dimensional kernel geometry, which is reasonable if the kernels are either approximately spherical or are at least randomly oriented. While both kernel diameter and aspect ratio are variable particle properties, aspect ratio is mathematically limited to a minimum value of one. This means that aspect ratio cannot be accurately modeled by the Gaussian distribution, so it must be treated as an attribute property with a critical upper limit.

Density, the other physical property commonly specified for kernels, may be measured by mercury porosimetry (Hunn, et al., 2014). In this method, a kernel sample of known mass is placed in a cell under vacuum and infiltrated with mercury up to a pressure to envelop each kernel without intruding into any surface pores (approximately 25 psi). The volume of the kernel sample is then calculated based on the mass of displaced mercury, allowing for calculation of the average density of the kernel sample based on its

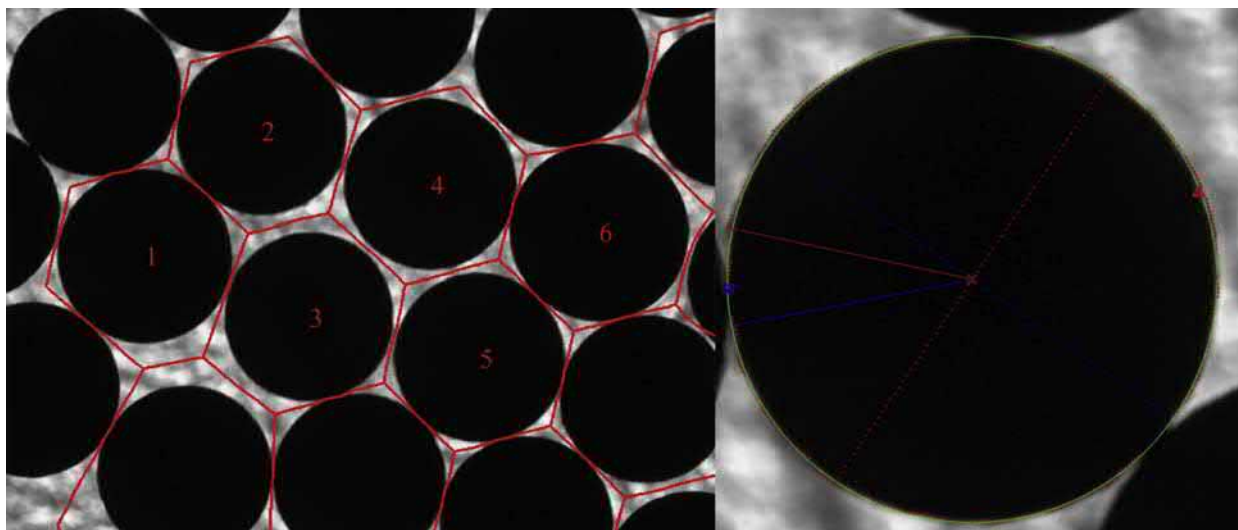


Fig. 3 Segmented shadow image of kernel monolayer (left) and sub-image of individual kernel with automatic measurements of mean diameter, aspect ratio, local curvature, and other geometric parameters (right).

weight. Since this method of density calculation is based on the envelope volume of the kernels (i.e., the volume without intrusion into pores), the resulting density is sometimes referred to as the *envelope density*.

Measurement of compositional kernel properties

Regardless of the type of kernel used in a given TRISO design, an acceptable range of kernel composition is commonly specified as a variable property. This may be given in terms of elemental ratios (e.g., O/U for UO_2 kernels) or in terms of phase ratios (e.g., uranium carbide/uranium oxide for UCO kernels). Elemental ratios for kernels may be determined by a combination of combustion and chemical analyses (Price, et al., 2010). In combustion analysis, kernels are burned, and the mass of elements such as carbon, nitrogen, and oxygen within the kernels are measured based on the resulting combustion gasses. In chemical analysis, kernels are dissolved in nitric acid, which is then analyzed by Davies-Gray titration to assay the total uranium content (Bickel, 1997). Phase ratios for kernels with multiple phases present may be determined by optical microscopy of near-midplane kernel cross sections based on image analysis coupled with the known densities of the phases present (Helmreich, et al., 2018a,b).

The other commonly specified components of kernel composition are the maximum levels for various impurities such as phosphorous, sulfur, and various metallic elements. The allowable quantities of these impurities are typically very low, on the order of hundreds to thousands of parts per million. Impurity levels in kernels are typically measured in the same general fashion as uranium content: kernels are dissolved into a solution which is then used for chemical analysis using methods such as mass spectrometry.

TRISO layer characterization methods

Commonly specified properties for TRISO layers may be divided into variable properties, including the thickness, density, and structure of each layer, and attribute properties, which generally characterize specific defects that may arise during TRISO fabrication.

Variable layer properties

The mean thickness of each of the four TRISO coating layers is generally specified, along with limitations on the dispersion. The thickness of all the coating layers may be measured simultaneously using automated image analysis of optical microscopy images of ceramographic cross sections (e.g., see Fig. 4) near the particle's midplane (Kercher, et al., 2008). This process relies on two key components: the ability to accurately segment layers within the cross-sectional images, and the ability to correct for geometric distortion in the image plane due to distance from the particle's midplane. Several image processing approaches may be applied for accurate layer segmentation. One such method is coupled Bayesian snakes, in which initial estimates of the boundaries for each layer are iteratively updated based on local image intensity, estimated layer thicknesses, and the position of adjacent boundary points until convergence criteria are met (Price, et al., 2007). Correction of the initial measurement of layer thicknesses to account for geometric distortion due to distance from the particle's midplane may be performed by applying a polish-down correction. This correction assumes a near-spherical particle and uses the polish-down distance for the image and the outer diameter of the particle in the imaging plane to estimate the distance to the particle's midplane and project the measured layer thicknesses to the midplane (Helmreich, et al., 2018a,b).

The aspect ratio of TRISO particles may be measured in the same manner as bare kernels, by shadow imaging and image analysis, as previously described. As with kernels, although particle aspect ratio is a variable property, the asymmetric nature of its

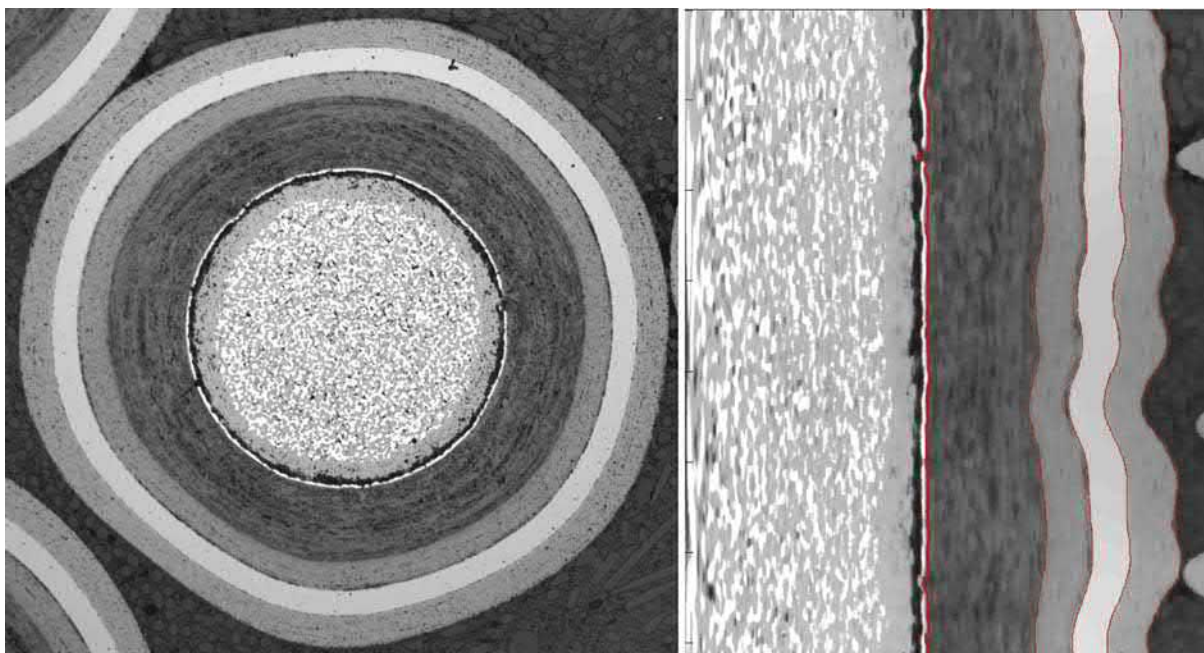


Fig. 4 Optical image of a UCO TRISO polished to near midplane (left) and the same image transformed to radius and angle about the kernel's center with layer boundaries marked by automatic segmentation for measurement of mean layer thickness.

distribution precludes the use of Gaussian statistics. As such, the TRISO aspect ratio is treated as an attribute property, with a critical upper limit.

Another TRISO layer property measurable by imaging particle cross sections is the anisotropy of the IPyC and OPyC layers. The anisotropy of PyC represents the degree to which graphite basal planes are either aligned or randomly oriented. Since PyC is optically active, the anisotropy of the IPyC and OPyC layers may be measured by using a polarized light microscope equipped to measure diattenuation: the preferential absorption of light along a specific axis (Jellison & Hunn, 2008). Since an anisotropic microstructure may result in undesired anisotropic behavior under irradiation, specifications on IPyC and OPyC anisotropy are typically one-sided, setting a maximum value only.

Another key variable property commonly measured for TRISO layers is density. This measurement presents a challenge for the buffer layer, as its low strength makes removal from the kernel and other layers infeasible. A solution is to (1) perform an interrupted coating run in which only the buffer layer is deposited, (2) measure the total mass and volume of the buffer-coated kernels by mercury porosimetry, (3) divide the total buffer-coated kernel mass and volume by the number of particles in the sample to convert to averages, (4) subtract out the average mass and volume of the kernels, and (5) divide average buffer mass by average buffer volume to calculate average buffer density (Hunn, et al., 2014). There are several drawbacks to this approach. First, it requires the assumption that the properties of the buffer-coated kernels from the interrupted run will be the same as the full TRISO particles. Second, subtraction of the average mass and volume of the kernels results in a relatively larger uncertainty in the calculated buffer density. Third, this approach only measures the average buffer density without providing any data regarding the statistical distribution.

Density measurement for the other coating layers is more straightforward. Under sufficient pressure, TRISO particles will crack, and the IPyC, SiC, and OPyC layers will separate into fragments. Fragments from each of these layers may then be collected for direct measurement of their density. One highly accurate approach to measuring the density of these layer fragments is the use of a linear-gradient density column (Hunn, et al., 2014). Density columns are constructed by mixing liquids of two different densities and slowly filling a transparent vessel with the mixture while gradually shifting the ratio of the liquids. The result is a column with a density gradient from top to bottom that is dependent on the densities of the liquids used and the mixing ratios (ASTM International, 2018). Layer fragments and calibration floats of known density are added to the column and allowed to settle before their heights within the column are measured. The column calibration is calculated based on the density and heights of the calibration floats, allowing the density of each layer fragment to be determined based on its height within the column.

Attribute layer properties

The stochastic nature of TRISO fabrication processes may result in a low population of various particle anomalies (e.g., coating anomalies caused by over fluidization which ejects particles from the bed during coating layer deposition). Anomalies that may impact fuel performance are regarded as defects, and their allowable populations are specified and quantified to a given level of

confidence based on the reactor design. Characterization of other anomalies that may or may not be directly linked to specified defect fractions can be helpful in providing feedback for process optimization and reduction of the specified defect fractions.

The most critical TRISO defects are failed layers which will allow actinides and fission products to be released from particles during and after irradiation. These defects are commonly identified by performing leach-burn-leach analysis on particles from deconsolidated fuel forms to ensure that any particle failures due to the compacting process are identified (Baldwin, et al., 2014). Particles are first leached in near-boiling nitric acid for a long period (on the order of 24 h). The leachate is then analyzed for uranium content to identify the defect fraction of exposed kernels based on the known uranium content of a single kernel (assuming the uranium kernels of failed TRISO dissolve fully in the solution). The OPyC layer on the particles and any other exposed carbon is then burnt away by air oxidation at high temperature ($\sim 750^\circ\text{C}$). A second leach is then performed in the same manner as the first, but uranium in the second leachate is attributed to particles with a defective SiC layer.

One of the roles of the IPyC layer is to prevent infiltration of HCl gas generated as a byproduct of the SiC deposition process. If unusually high permeability of the IPyC layer allows for HCl infiltration to the kernel, then subsequent exposure to high temperatures will result in dispersion of uranium from the kernel into the buffer layer due to leaching. Identification of defective IPyC in TRISO particles requires them to be heated to trigger uranium dispersion and then examining them by radiography to identify particles with excessive uranium dispersion into the buffer layer (Helmreich, et al., 2017a,b). Counting the number of particles with excessive dispersion out of the total number of particles imaged allows the maximum defect fraction to be determined with a given confidence as described in the “Statistics of TRISO quality control” section. Due to the penetrating nature of radiography and the large number of particles which must be analyzed to determine the defective IPyC fraction, this analysis presents an excellent opportunity for the identification of other particle anomalies as a feedback mechanism to the fabrication process (Helmreich, et al., 2017a,b).

The inclusion of carbon and silicon soot within the SiC layer during coating is known to result in the formation of lenticular flaws which may reduce the strength of the SiC and could result in particle failure (Kinsey, et al., 2010). These defects may be identified by optical analysis of the SiC layer in large numbers of cross-sectioned particles.

Another attribute specification on the SiC layer targets its microstructure. Data indicates that large columnar grains within the SiC layer may reduce fission product retention, as well as layer strength (Petti, et al., 2003). As such, the SiC layer microstructure may be specified to be finer than a given visual standard of undesirable SiC microstructure based on backscattered electron images (BSE) which show grain structure due to the dependence of electron penetration and backscatter on grain orientation. More quantitative methods for SiC microstructure analysis may be implemented based on automated grain size measurement of BSE images (Helmreich, et al., 2020).

Impacts during particle handling may result in cracking and removal of the OPyC layer for some particles. This defect removes the protective layer from around the SiC, which may result in further damage to the particle during handling and compaction into a final fuel form. Due to the significant difference in surface appearance for OPyC and SiC, particles with this defect may be identified by visual inspection of particles with an optical microscope.

TRISO fuel form characterization methods

While TRISO particles are generally similar regardless of specific reactor design, the fuel forms into which TRISO particles are fabricated are diverse, as are their required characterization methods. Specifications on overall shape, the geometry of internal features such as fuel-free zones, the density and geometric distribution of TRISO particles in each unit, and matrix impurity limits are all common. For some reactor designs in which fuel forms are non-static (e.g., pebble bed reactors), additional specifications on fuel form strength and abrasion resistance may be required. However, a detailed description of the various fuel form fabrication and characterization methods which may be applied for differing reactor designs is beyond the scope of this overview.

Summary

Fabrication of TRISO particle fuels is complex, involving gelation of uranium-bearing beads, thermochemical conversion to kernels, deposition of multiple coating layers, and compaction into a final fuel form. Characterization of TRISO particle fuels is even more complex, with multiple variable and attribute properties which must be measured with a high degree of statistical rigor. However, more than 60 years of particle fuel research and development have produced a robust body of knowledge surrounding TRISO fuel fabrication and characterization, enabling the development and deployment of a wide range of reactor designs using this fuel.

See Also: Fluoride-Salt-Cooled High-Temperature Reactors; Pebble Bed Gas Cooled Reactors; The High Temperature Gas-Cooled Reactor.

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Fuel Design and Fabrication: Research Reactor Fuel

James I. Cole,^a Jan-Fong Jue^a, and Glenn A. Moore^{b, a} Idaho National Laboratory, Idaho Falls, ID, United States; and ^b Walsh Engineering Services, Idaho Falls, ID, United States

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Introduction

Research and test reactors serve a vital role in the scientific community and society in general. Contributions include the production of medical isotopes, fundamental scientific studies into the nature of materials, nuclear fuels testing for advanced reactor and commercial applications, and a variety of national security missions. Typically, “research” reactors are defined as small, lower-power reactors under 5 MW thermal (5 MW_{th}), commonly used for education and training, while “test” reactors are larger reactors with missions ranging from isotope production, the accelerated testing of materials and fuels for a variety government and commercial nuclear energy development missions, and the production of intense neutron beams for basic scientific research. Test reactors operate up to several hundred MW_{th}. Due to the expansive range of missions and operational envelopes (e.g., neutron flux, duty cycle, test position size), the fuel systems used for these reactors are also quite varied, including different geometric configurations (plates, pins) and fuel material (oxides, intermetallics, and alloys) as particles dispersed in a thermally conductive matrix or in a solid monolithic fuel form. Fig. 1 provides an illustration of the wide range of operational performance envelopes for several U.S. high-power research reactors (HPRR) that a fuel system must accommodate.

Fuel design considerations

Fuel systems developed for research and test reactors are designed to meet a unique set of requirements and, in many cases, for the highest performance reactors, a high specific power density is required to generate large neutron fluxes in a small spatial volume (on the order of $1 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$). Compact, large specific power density cores require the fuel to rapidly eject heat, thus the fuel systems for large HPRRs are plate type (with a large surface-to-volume ratio) and are mainly composed of metallic constituents that have a superior thermal conductivity. These reactors operate at much lower temperatures than commercial power reactors, with inlet coolant temperature under 150 °C, therefore the cladding materials can be composed of alloys with a lower melting temperature, such as aluminum.

Because many of the world’s research reactors were originally designed with highly enriched uranium (HEU) fuels (²³⁵U content greater than 90%), there has been a global effort to remove HEU research reactor fuels from civilian commerce, which entails the conversion of existing research and test reactors from HEU fuel to low enriched uranium (LEU) with <20% ²³⁵U content. This effort supports U.S. government and international nuclear nonproliferation goals. As a result, several new fuel systems have been developed, or existing fuel systems altered, to accommodate LEU. This nearly fivefold decrease in enrichment requires an offsetting increase in the fuel uranium density to maintain the overall operational performance metrics (such as power density and fuel dimensions), which reactor owners require for the conversion from HEU to LEU. To maintain the operational performance requirements and the duty cycle of the research reactors, developers of these new fuels must address a variety of challenges, including the development of new fabrication methods and new testing to ensure the changes that have been made to accommodate the higher density fuel do not compromise the fundamental ability of the fuel to meet irradiation performance requirements (Cole, et al., 2021).

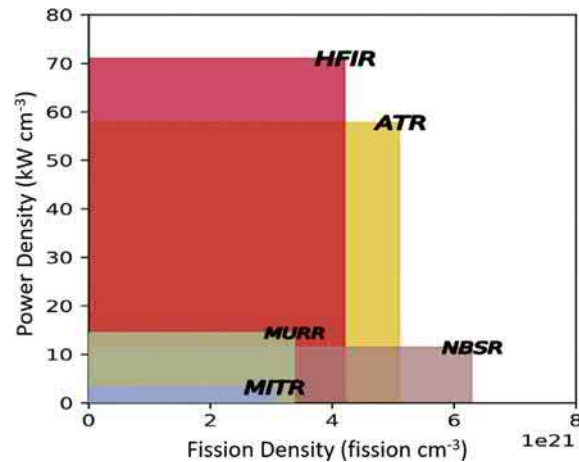


Fig. 1 Fuel operational envelopes for several U.S. HPRRs, including the High Flux Isotope Reactor (HFIR), Advanced Test Reactor (ATR), Missouri University Research Reactor, The National Institute of Standards and Technology Reactor and the Massachusetts Institute of Technology Reactor. Power density is directly related to the available neutron flux, while fission density impacts overall fuel and operational cycle management. Adapted from National Academies of Sciences, Engineering, and Medicine (2016) Reducing the Use of Highly Enriched Uranium in Civilian Research Reactors. Washington, DC: The National Academies Press.

The physical form, or geometry, the fuel takes will impact a variety of design considerations and serves several functional objectives. Plate-type fuels are typically made of fuel and cladding layers with high thermal conductivity to aid in rapid heat rejection. The fabricated layers of these composite fuel plates are composed of multiple constituents, with each constituent playing a specific role related to ensuring the fuel can effectively produce high neutron flux, transfer heat, and retain fission products during its entire life-cycle. HPRR plate-type fuel systems are typically based on the dispersion of particles in a matrix or as a monolithic fuel form. **Fig. 2** illustrates the differences between a monolithic fuel and a dispersion type fuel form. Frequently, a barrier interlayer or coating is introduced between the cladding and the fuel to prevent a fuel-cladding chemical interaction. The interlayers are selected to inhibit interdiffusion and have a low chemical reactivity with both the fuel and the cladding.

Dispersion fuels

The original concept of a dispersion fuel was a system with fuel particles surrounded by a matrix that inhibits the transport of fission products and isolates the particles to prevent the interconnection of fission-gas-induced porosity across the fuel bearing region, termed “fuel core,” or “meat” (impacting mechanical integrity and thermal conductivity). The ductile, high-conductivity matrix (typically pure aluminum) also accommodates dimensional changes in the fuel particles that result from the formation of gaseous and solid fission products (e.g., fuel swelling). The cladding provides the main mechanical structure, acts as a barrier to fission product release, and acts as a conduit for the transfer of heat to the coolant. Interlayers (barrier layers or coatings) serve to inhibit the chemical interaction between the constituents’ dissimilar layers that might otherwise compromise the overall fuel system irradiation performance. This design enables the fuel to function up to a very high fission density (i.e., high exposure or high burnup). The intimate contact between the fuel and the cladding and the absence of a fission-gas plenum (as incorporated into light-water reactor and fast reactor fuel systems) means that research reactor fuel must maintain the fission gas in the fuel material itself and accommodate this gas while retaining adequate heat conduction and structural integrity to high burnup levels.

The density of the fuel particles and distribution of those particles within the matrix material will influence the behavior of the fuel during irradiation and its ability to meet reactor performance requirements (because an agglomeration of the fuel particles

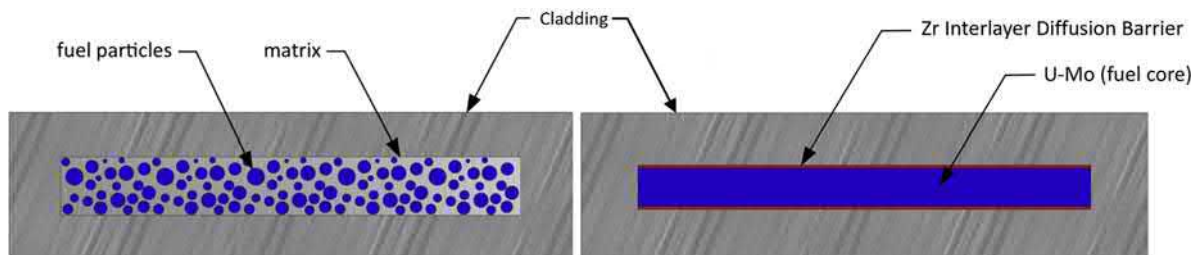


Fig. 2 Schematics of a plate-type fuel composed of (A) particle dispersions and (B) of a solid monolithic fuel material. The density of uranium in the monolithic fuel is significantly greater for the same fuel material and the surface area to fuel volume is less. Modified from Rabin B, et al. (2017) Preliminary report on U-Mo monolithic fuel for research reactors. INL/EXT-17-40975 Rev. 3. Idaho Falls, ID: Idaho National Laboratory.

could result in localized hot spots). Historically, as HEU fuels were used, the number of fuel particles per unit volume was markedly lower, but, due to the fuel's high enrichment, reactor power and burnup targets could be achieved. However, converting research reactors to LEU requires an increase in the fuel uranium density (achieved using a combination of high-density compounds and increased fuel particle density) to maintain the same fuel operational performance attributes. Maintaining the same neutron flux, while reducing the enrichment from more than 90% to less than 20% fissionable ^{235}U , requires maintaining the same ^{235}U density by increasing the overall uranium fuel density in the fuel core. In fact, the total amount of uranium needed in the fuel core is even higher than for HEU fuel due to the parasitic neutron absorption from an increased amount of ^{238}U in the fuel.

Monolithic fuel

The monolithic foil fuel form is used to achieve the highest possible density of uranium in the fuel core. This is particularly important for the highest power reactors converting to LEU, which would otherwise require dispersion fuel particle densities beyond those that can be practically fabricated. For the LEU conversion effort, the U-10Mo alloy demonstrates the best combination of uranium density and irradiation performance properties (stable, isotropic swelling behavior due its bcc crystal structure). The surface area per fuel volume in contact with the aluminum matrix and cladding is less for monolithic fuel than for dispersion fuels, which is advantageous in managing fuel-aluminum chemical interactions. The results from the initial irradiation testing of the U-10Mo fuel led to the introduction of a Zr foil interlayer to prevent chemical interaction between the fuel and the cladding. A barrier layer thickness of 25 μm bounds the fission fragment recoil distance, thus preventing fission products from being injected into the cladding.

In summary, after extensive development and testing, an LEU monolithic U-10Mo has been designed for use in U.S. HPRRs, comprised of a foil-shaped U-10Mo fuel core clad in the aluminum alloy 6061 (AA 6061). The U-10Mo fuel alloy is co-rolled with a Zr foil layer on each flat outer surface to obtain a fuel foil. The fuel foil is slit or sheared to size and clad in aluminum using hot isostatic pressing (HIP). This resulted in the foil edges being free of the Zr barrier; however, because the contact surface with the cladding is lower on the edges, the interaction with the cladding is of minimal concern. Alternative methods of applying Zr layers to the fuel foils were explored during development, but these methods did not produce Zr layers with adequate adhesion to the fuel.

Fuel element design

After fabrication, the fuel plates are curved (for reactors using curved fuel plate geometries) and assembled into an element. The plate configurations and elements optimize the neutron flux for a specific core design and application, and curved fuel elements allow the formation of flux traps in the reactor core (irradiation regions of high flux). But the plate curves also improve mechanical stiffness and geometric stability in the high coolant flow conditions needed to adequately cool the plates (ATR maximum flow is $3.09 \text{ m}^3 \text{ s}^{-1}$ (49,000 gal min^{-1}) (Stanley and Marshall, 2008)). Fig. 3 provides a three-dimensional rendering and cross section of a representative element used in the ATR along with an image of the HFIR reactor core composed of an inner and outer fuel element. ATR is designed as a high-power material test reactor requiring large volume test positions and extended irradiation cycles. Because HFIR is designed to generate extremely intense neutron beam lines, the HFIR core is highly compact to optimize the flux in the center flux position of the core. This requires all the HFIR fuel plates to be of the same dimension and uniformly spaced within a cylinder (uniform coolant channels). Geometrically, this can only be achieved with plates formed in an involute geometry

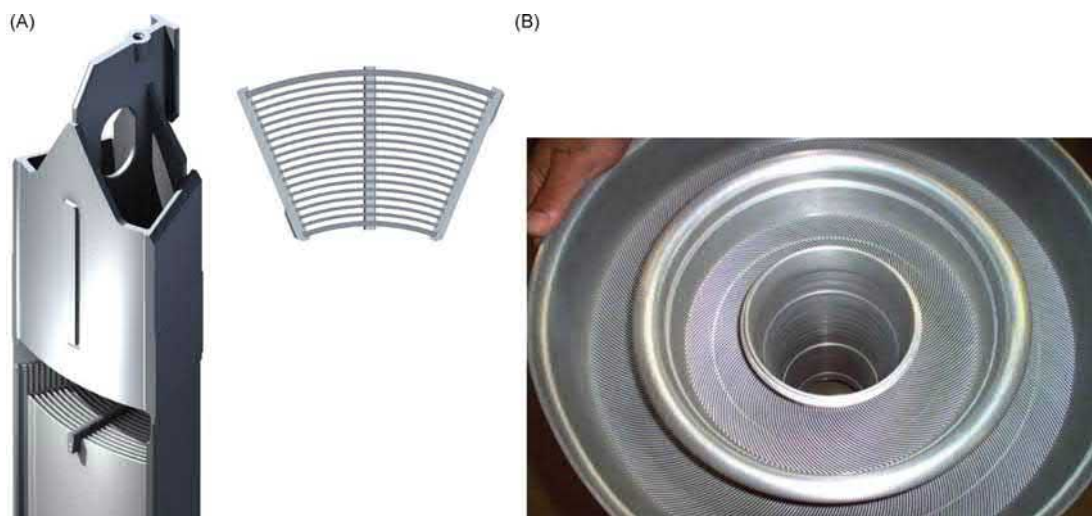


Fig. 3 (A) Three-dimensional rendering and cross section of ATR element and (B) image of HFIR Core (Betzler et al., 2019). The ATR core is composed of 40 fuel elements, each with 19 plates, arranged in a serpentine core with four individual power lobes. The HFIR core is composed of two nested cylindrical elements, inner and outer, containing 369 and 171 involute plates, respectively.

(the shape a string takes when tautly wrapped around another curve). While each plate in an ATR element is exposed to slightly different irradiation conditions, all plates within a HFIR element (inner or outer) encounter nearly identical irradiation conditions.

Fuel compounds and associated properties

The compounds used for research reactor fuel materials include oxides (UO_2 , U_3O_8), intermetallics (U_3Si , U_3Si_2 , UAl_x), and alloys (U-Al, U-Mo). The need for stable fuel behavior up to a high burnup has limited the choice of compounds to only a few, with the selection even further limited to high uranium density compounds when replacing HEU with LEU. **Table 1** lists the most common fuel compounds in use or being proposed for use and their associated uranium densities. As mentioned earlier, a high-density fuel compound or alloy in a monolithic form has the greatest uranium density, permitting a decreased enrichment while maintaining ^{235}U loading. This permits the fuel to achieve the same operational objectives as HEU. U-Mo alloy fuels, while not currently qualified for use in operating reactors, are the subject of a large effort to convert existing HPRRs to low enriched fuel. An additional fuel system composed of a U-Zr-H compound is found in training, research, and isotope-production reactors—General Atomic (TRIGA) reactors—that operates at substantially lower powers and will not be discussed further.

Material properties are needed by fuel designers and reactor operators to evaluate the thermal and mechanical performance of various designs under steady-state and transient operational conditions. In-service performance attributes of interest regarding the fuel system include temperatures, deformations, and stresses—strains that facilitate the identification of limiting operational parameters (e.g., power density) or fuel conditions (e.g., maximum burnup) for use in various reactor applications.

The density and melting temperature of the fuel and cladding are important physical properties that designers must know to determine the limits in operating conditions to prevent the melting or excessive oxidation of the cladding by the coolant (phenomena that could lead to the release of fission products from the fuel). Knowing the thermal conductivity of the fuel core is important to ensuring that the fuel remains below its melting temperature and that temperature dependent processes within the fuel stay within a predictable range. Heat capacity influences the fuel response to rapid power increases during transient or accident conditions. Mechanical properties of the cladding (and dispersion fuel matrix material) impart the ability to mechanically resist or accommodate fuel swelling as fission gases accumulate and exert pressure that can induce plastic deformation or cracking. Understanding the degradation of these properties during irradiation is critical to the evaluation of accident scenarios that could lead to a fuel breach and subsequent fission product release.

Fabrication of plate-type dispersion fuel

Dispersion fuel systems have several unique advantages resulting from the powder metallurgical processes used to fabricate them. The careful selection of fabrication processing parameters can beneficially introduce porosity into the fuel that will accommodate the increased volume of fission gas produced at higher burnup. Burnable poisons can also be introduced into the fuel by mixing them with the powders and can be spatially graded to aid in shaping the neutron flux profiles (allowing the attainment of uniform neutron flux profiles to high fission density). In the absence of a gas plenum, the more open structure of a powder metallurgy produced dispersion fuel accommodates larger quantities of fission gas without a coalescence of gas bubbles that might reduce the temperature threshold for surface blister formation, which is a fuel failure mechanism of concern for aluminum-clad, plate-type fuels (Cole et al., 2021; Beeston et al., 1980).

Powder fabrication

The process used to form the powders used in fabricating dispersion fuels depends on the nature of the fuel compound (Hofman and Snelgrove, 1994; Meyer et al., 2020). For brittle solids, like intermetallic fuel compounds, powders are produced by comminution. This process uses jaw crushers and hammer mills to mechanically break down the brittle cast alloy into a powder, followed

Table 1 Properties of HPRR fuel compounds (Kim, 2012; Meyer et al., 2020; Hofman & Snelgrove, 1994).

Fuel compound	Density (g cm^{-3})	Fuel uranium loading (U g cm^{-3})	Typical enrichment	Typical fuel form (fuel core/fuel system)
U	19.1	19.1	N/A	N/A
U-7Mo	18.4	19.1	LEU	Dispersion/plate
U-10Mo	18.2	16.4	LEU	Monolithic foil/plate
U_3Si	15.6	15.0	LEU	Dispersion rodlet/pin
U_3Si_2	12.2	12.2	LEU	Dispersion/plate
UAl_x^a	N/A	4.5	HEU	Dispersion/plate
U_3O_8	8.4	7.1	HEU	Dispersion/plate

^a UAl_x is a mixture of UAl_2 , UAl_3 , and UAl_4 .

by a sieving of the powders using metal screens to generate the desired distribution of particle sizes. These processes are carried out in inert gas environment to prevent the excessive oxidation of the powders.

Powders from cast alloys of ductile metals are produced through a melt atomization process. In this process, the molten fuel alloy is either poured onto a rotating disk or turned into liquid droplets by rotating an electrode of the fuel material as it is heated. Through centrifugal force and surface tension, small droplets of the fuel material form and rapidly solidify into spherical particles. The fabrication of the metal alloy powders is also carried out in an inert atmosphere because fine metal powders are highly pyrophoric (ignite readily when exposed to oxygen). As with the comminution process, powders can be sieved using metal screens to produce the desired particle size distributions.

Oxide powders, such as U_3O_8 , are produced through inorganic precipitation processes. This is followed by a multiple stage calcining processes at an elevated temperature to produce the desired high-density powders. Following these steps, the fuel material is ground to produce the final powder size distribution.

The images shown in Fig. 4 are of as-fabricated dispersion fuels illustrating the differences in particle geometry between powder fabrication methods. Tailoring the particle size distributions enables an increased packing density of the particles; however, there is a practical limit of $\sim 50\%$, by volume of fuel particles, where fabrication becomes impractical.

Roll bonding of fuel plates

Powder components of the fuel core (fuel powders, Al matrix powders, and, if needed, burnable poisons) are placed into a die and compacted to form the consolidated fuel core. Fabrication of dispersion-based research and test reactor fuel plates uses a picture frame technique to produce the fuel. This technique was used for a variety of fuels, including $U-Al_x-Al$, U_3Si_2-Al , and U_3O_8-Al , and has been in practice for more than 50 years to produce research reactor fuel. The fuel core compact is assembled inside of a rectangular opening machined into a sheet of Al alloy fuel cladding, resembling an open picture frame. The top and bottom plates of the fuel-cladding material are assembled with the fuel-containing layer to form the complete assembly (Fig. 5 provides a schematic of the fuel plate constituents). The entire assembly is then hot rolled and cold rolled to bond the layers together.

One of the significant challenges in producing fuel plates with a high volume fraction of fuel particles is a phenomenon termed “dogboning” (Beeston et al., 1980). Regions of the fuel compact thicken during rolling (typically the leading and trailing edge of the

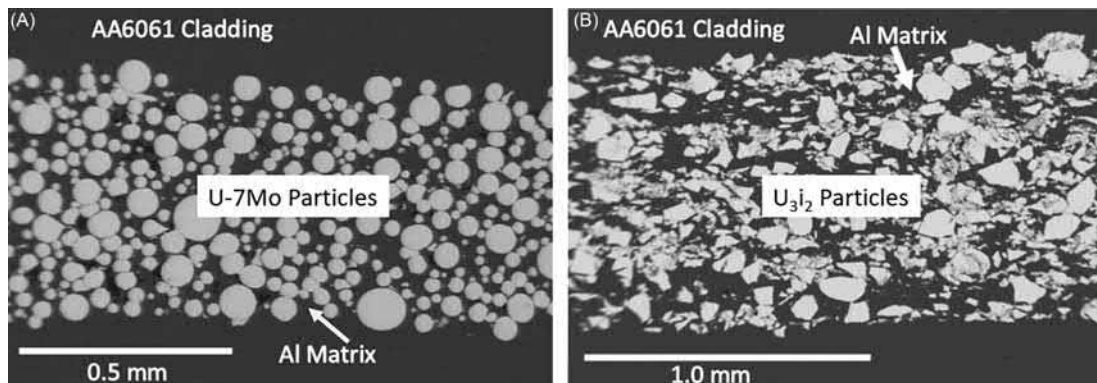


Fig. 4 Scanning electron microscope images of (A) atomized particles of U-7Mo and (B) comminuted particles of U_3Si_2 fuel. Adapted from Keiser DD, Jr., et al. (2020) The use of U_3Si_2/Al dispersion fuel for high power research reactors. *Journal of Nuclear Materials* 528: 151820.

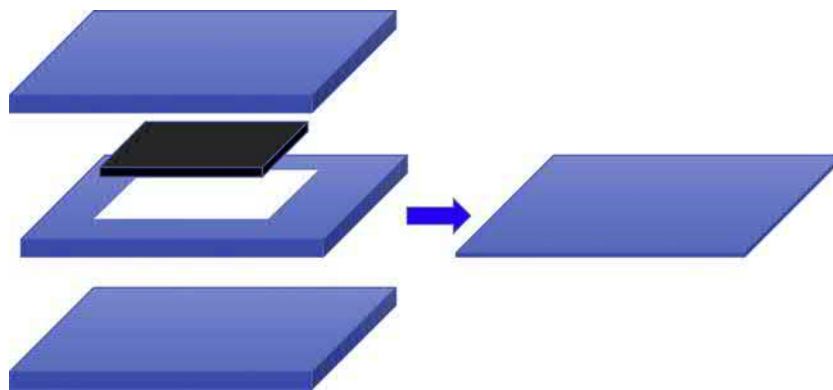


Fig. 5 Schematic of the picture frame method of roll bonding research reactor fuel. The dark rectangle represents the powder compact of fuel particles dispersed in a matrix with or without burnable poisons.

compact), which can lead to a thinning of the cladding to unacceptable levels over these thickened fuel core regions. To remedy this issue, nonrectangular compacts with tapered ends enable the production of fuel plates with more uniform fuel cores and adequate cladding thicknesses.

Following rolling, the plates are blister annealed (i.e., heated and held at an elevated temperature to ensure the assembly does not delaminate due to the expansion of entrapped gas). Blister annealing is followed by rough shearing, by an ultrasonic inspection for debonded interfaces and to measure the minimum cladding thickness, by final sizing, and by radiographic inspection for uranium homogeneity.

After fuel plate fabrication, the plates are mechanically formed into the final geometry (typically curved) and assembled into the elements. Many of the elements for the high-power reactors using plate-type fuels are fabricated by mechanically swaging the plates into side rails, although some elements (notably HFIR) are assembled by welding the plates into the element side rails.

Fabrication of plate-type monolithic fuel

The high-density LEU U-10Mo monolithic fuel system has been developed to enable the conversion of HPRRs. The general fabrication process of the U-Mo monolithic fuel system is (1) alloying and casting, (2) foil rolling, (3) bonding of the cladding using a HIP, and (4) finishing and inspection for quality assurance. The following descriptions provide examples of the overall generic process developed at the lab-scale for fabricating LEU monolithic fuel in the U.S. The adaptation of the lab-scale process to the commercial scale has led to some variation in specific details, some of which are proprietary to a particular commercial fuel vendor. However, the basic four-step process described above is the same. Fig. 6 provides a flow diagram with more details of the processes within each step.

Casting and alloying

Alloying is performed through the vacuum induction melting of U and Mo metals at a temperature between 1400 °C and 1500 °C. Because the melting point of Mo is 2623 °C and that of U is 1132 °C, Mo dissolution and alloying into molten U takes place. To expedite alloying, a Mo powder, foil, or sheet is often used to increase the contact surface area. To obtain an LEU alloy having a 19.75% ²³⁵U enrichment, HEU, depleted uranium, and natural uranium metal source material is utilized in the alloying process. Alloying is performed using graphite crucibles and molds with a protective coating (e.g., erbium oxide) that prevents the interaction of the molten alloy with the graphite crucible. To establish the desired alloy compositions and homogeneity, alloy melts may be broken up, sampled, and remelted several times, and compositional adjustments made as needed by foundry personnel. The composition of the fuel alloy utilized for monolithic fuel plates is designated as U-10Mo, with ± 1 wt% Mo content being acceptable. In preparation for U-10Mo fuel foil fabrication, rectangular ingots are cast. Once cooled, cast ingots are machined to remove surface imperfections and any alloy-mold interaction products to produce the starting coupons used for foil fabrication. Cast coupons are radiographed to ensure inclusions or voids that can lead to unacceptable foil rolling are not present.

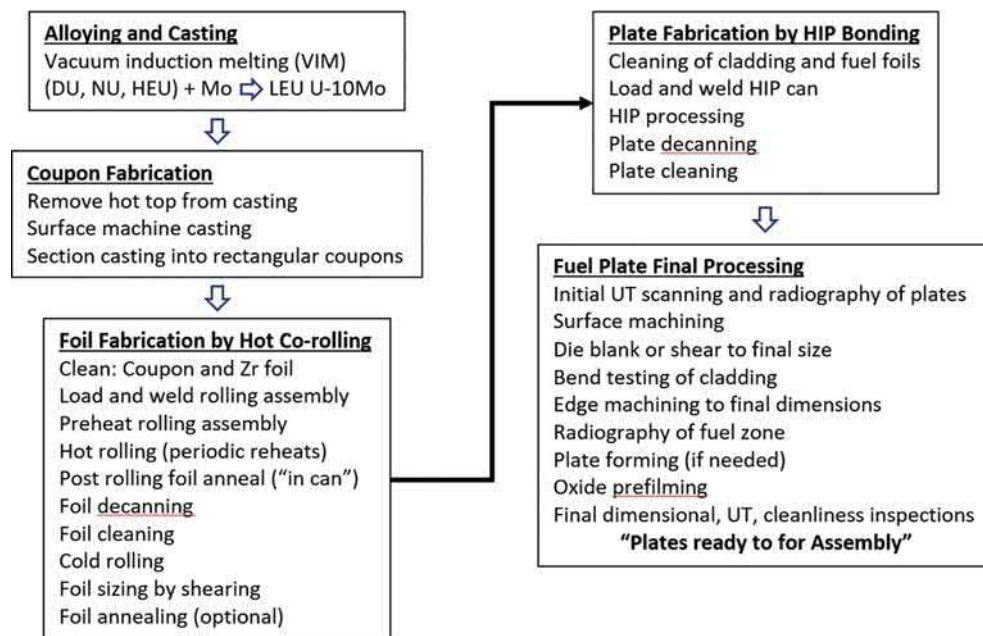


Fig. 6 Flow diagram illustrating the processes used to fabricate U-10Mo monolithic fuel plates. Adapted from Rabin B, et al. (2017) Preliminary report on U-Mo monolithic fuel for research reactors. INL/EXT-17-40975 Rev. 3. Idaho Falls, ID: Idaho National Laboratory.

Foil fabrication

The coupons are homogenized at 900 °C for 144 h under a vacuum to prevent the oxidation of the reactive uranium. Following homogenization, two Zr foils are placed on the outside flat surfaces of the coupons to form a stack and the stack is hermetically sealed in a carbon steel can to prevent oxidation. The ingots are individually hot rolled after preheating the can to 650 °C. Cans are passed through a rolling mill several times to incrementally reduce the foil thickness. The number of rolling passes varies based on the target thickness reduction planned for cold rolling. As needed, the cans are placed back into the box furnace and reheated to maintain the hot rolling temperature. After hot rolling to ensure the Zr layer is bonded, the foils are removed from the cans and cold rolled directly. This is to bring the final thickness of each cold rolled foil to the desired final thickness. Finished cold rolled foils are examined for edge cracking. If edge cracking is observed, cracks are removed via shearing to prevent cracks advancing during storage.

Plate fabrication—HIP bonding of Al cladding

The HIP process for bonding involves the application of pressure and temperature to produce a solid-state joint formed through a combination of mechanical deformation and interdiffusion at the bond line. The joining and forming of metals by HIP is a common fabrication process used for a variety of industrial applications. One of the main difficulties in bonding aluminum alloys is the native oxide that forms on the surface. Joining processes that break up the oxide tend to improve bonding and includes an application of pressure or roughening of the surface. A fairly extensive amount of literature exists on roll bonding to produce composited sheets of aluminum to improve strength, while HIP technology as applied to joining of aluminum sheet is less mature.

During the fabrication development of the U-10Mo fuel, extensive effort went into ensuring adequate and reproducible bonding between the fuel foil and the Al alloy cladding. It is essential to maintain intimate contact between the interlayers to make sure heat transfer from the fuel to the cladding is not perturbed. The HIP bonding process was developed precisely to accomplish this. The temperature of the HIP process produces an interaction layer (evidence of metallurgical bonding) while limiting its extent so as to not negatively impact overall fuel performance. Adequate cladding-cladding bond strength is established through fabrication process controls and a demonstration that the bonding of the Al alloy cladding is adequate through extensive irradiation testing.

In the HIP fabrication process originally developed at Idaho National Laboratory (Jue et al., 2010; Rabin et al., 2017), the surfaces of two sheets of Al cladding are mechanically roughened. A rectangular recess is then machined into to one sheet, which is thicker than the other, to position and hold the U-10Mo foil. The sheets are then chemically cleaned and rinsed in preparation for bonding. HIP bonding entails the fabrication of a stainless steel “can” to hold the fuel plate assemblies. A steel plate, referred to as a “strong back,” and Grafoil sheets separate each assembly. The HIP can and fuel assemblies are then baked at an elevated temperature to remove entrapped gas prior to being seal-welded. The optimum time and temperature for HIP processing was determined to be 90 min at 560 °C, which minimizes the extent of the interaction between the fuel and the Zr diffusion barrier. The finished plates are then sheared to size. The excess material sheared from a plate typically undergoes a bend test, bending the material back and forth over a radius, to further ensure the Al cladding is adequately bonded. For final inspections, the fuel plates undergo ultrasonic testing to ensure the adequate bonding of the cladding interfaces, radiography to ensure the fuel location and uranium loading meet specification, and dimensional and visual inspection.

A subset of cladding material trimmed during sizing operations or from archival unirradiated fuel plates may also undergo scanning electron microscopy (SEM) to ensure a good metallurgical bond has been produced. A typical cladding-cladding bond-line analysis on fresh fuel is illustrated in Fig. 7. The intent of the analysis is to examine the bond line, ensuring there are no gross defects

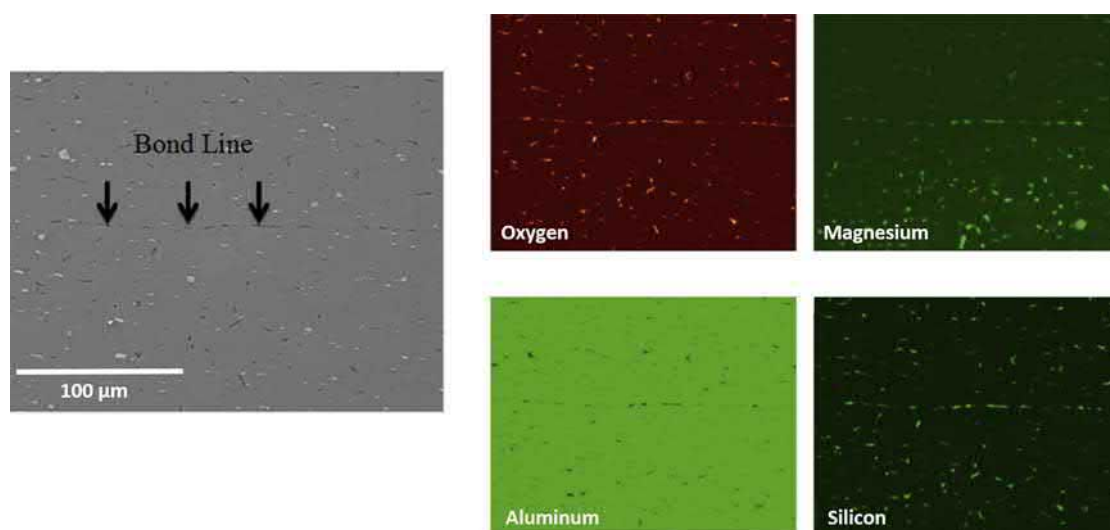


Fig. 7 Secondary electron image taken by SEM along with elemental EDS maps illustrating the cladding-cladding bond line and observed formation of magnesium- and silicon-rich particles at the interface, along with an elevated oxygen concentration.

or cracks, and to perform chemical analysis, such as energy dispersive x-ray spectroscopy (EDS) mapping, inspecting the character of the precipitates in the matrix and along the bond line. As these precipitates tend to be brittle intermetallics, their density must be known and controlled to ensure the strength of the bond line is not compromised. Because HIP bonding involves less plastic deformation than the roll-bonding process described for powder metallurgy dispersion fuels, the extent of grain growth across the bond line (resulting from alloy recrystallization during hot processing) is substantially lower. Roll bonding produces growth of 20–80% of the grains across the interface (a metric used to verify adequate bonding), while the HIP bonding typically produces growth of no more than 10% of the grains across the interface, thus making this measurement unreliable in assessing cladding-cladding bond integrity. Efforts to develop fabrication inspection criteria and process controls to ensure adequate bond integrity across all constituent interfaces is an important aspect of commercializing the overall fuel fabrication process.

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See Also: An Introduction to Research Reactors; Research Reactor Conversion to Low Enriched Uranium (LEU) Fuel.

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Nuclear Criticality Safety

Douglas G. Bowen, Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Glossary

Critical mass The fissile or fissionable material mass at which criticality is achieved for a given configuration.

Criticality accident The release of energy as a result of accidentally producing a self-sustaining fission chain reaction (Paxton, 1989). A criticality accident occurs essentially instantaneously and is a hazard mainly to workers within about 4.6 m (15 ft) of the accident.

Criticality safety index A dimensionless number assigned to and placed on the label of a fissile material package to designate the degree of control of the accumulation of packages during transportation. This is derived as the number 50 divided by a number of packages, N, that can be shown to be subcritical for packages subjected to normal conditions of transport or hypothetical accident conditions.

Fissile nucleus A nucleus capable of fission by thermal neutrons, provided the effective neutron production cross section, $\nu\sigma_f$, exceeds the effective absorption cross section, σ_a (Paxton, 1989). The fissile nuclides of interest in Nuclear Criticality Safety are ²³³U, ²³⁵U, and ²³⁹Pu.

Fissionable nucleus nuclides capable of sustaining a neutron chain reaction with neutrons of some energy (Paxton, 1989).
hydrogen-to-²³⁵U ratio or hydrogen-to-U ratio (H/²³⁵U or H/U) The atomic ratio of hydrogen to ²³⁵U or to uranium, respectively, in a solution or hydrogenous mixture.

K-effective (k_{eff}) For a chain-reacting system, the mean number of fission neutrons produced by a neutron emitted in a previous fission during its life within the system. It follows that that $k_{eff} = 1$, if the system is critical, $k_{eff} < 1$, if the system is subcritical; $k_{eff} > 1$, if the system is supercritical (Paxton, 1989).

Nuclear criticality safety protection from the consequences of a criticality accident, preferably by prevention of the accident (Paxton, 1989).

Introduction

Nuclear Criticality Safety (NCS) is a field of nuclear engineering dedicated to precluding criticality accidents during operations with fissionable materials outside of a reactor. For NCS the fissionable nuclides of practical interest are ²³³U, ²³⁵U, and ²³⁹Pu. For establishing a fission chain reaction, the fissionable material should contain a sufficient fraction of fissile isotopes. Fissile materials have

been available in significant quantities since the Manhattan Project in World War II as a result of uranium enrichment and plutonium production operations to support defense programs. Since then, large quantities of fissionable materials have been processed, handled, transported and stored. During hands-on operations with fissionable materials, NCS is necessary to ensure a criticality accident does not occur. Since 1943, there have been 22 process accidents that have resulted in 9 deaths, 33 over-exposures to radiation, and 2 significant over-exposures resulting in physical worker harm, e.g., limb amputations. (McLaughlin, et al., 2000) Criticality accidents have occurred in the United States (7), former Soviet Union (13), United Kingdom (1), and Japan (1) (McLaughlin, et al., 2000). Process criticality accidents are less frequent and tend to be less severe than other industrial hazards, e.g., mechanical and electrical accidents, trips, falls, etc. According to the United States Department of Labor, there were 4779 worker fatalities in process accidents compared to zero process criticality accidents in 2018 (Anon, 2019).

The purpose of this article is to discuss the applicability of NCS to the Light Water Reactor (LWR) Nuclear Fuel Cycle (Fuel Cycle). The LWR fuel cycle in the U.S. currently does not include spent fuel reprocessing or a high-level waste repository for long-term disposal of spent fuel. NCS concerns for these activities are similar to other processing and storage activities. Further, NCS for fissionable material operations in nuclear defense activities are not discussed.

NCS fundamentals and background

As previously discussed, the nuclear fuel cycle involves performing hands-on operations with fissile materials, mainly ^{233}U , ^{235}U , and ^{239}Pu , during handling, processing, storage, and transportation. Based on the aforementioned process criticality accident history (McLaughlin, et al., 2000), there are definite hazards for workers performing these activities. The criticality hazards for the nuclear fuel cycle vary as a result of the characteristics of the facilities and locations where these materials are processed, handled, stored or transported. For these situations, the following characteristics are important to consider for precluding a criticality accident:

- Fissile material nuclide (^{233}U , ^{235}U , or ^{239}Pu) mass present (mass),
- Neutron absorption by non-fissionable nuclides present with fissionable materials (absorption),
- The shape of fissionable material, i.e., spherical, cylindrical, etc. (geometry),
- Neutron interaction with other adjacent masses of fissionable material (interaction),
- The concentration of fissile nuclides in a solution or solid material (concentration),
- Presence of moderating compounds, e.g., water, oil, plastic, with the fissile nuclides, which can significantly reduce the critical mass needed for a criticality accident (moderation),
- The enrichment of the fissile nuclide or quantity of the fissile nuclide in uranium (^{235}U) or plutonium (^{239}Pu) (enrichment),
- Available neutron reflecting materials, e.g., water, concrete, steel structures, worker bodies, etc., outside the fissile materials that are capable of scattering back to the materials neutrons that would have otherwise escaped (reflection) (Paxton, 1989), and
- Volume of the fissionable material system (volume).

As an example, the critical mass of enriched uranium varies significantly depending on these characteristics and what the form of the fissile/fissionable material is in: solution or slurry, powder, or solid metal. To help manage the risk of criticality accidents in nuclear facilities throughout the nuclear fuel cycle, consensus standards have been developed domestically (American National Standards Institute, 2020) and internationally (International Standards Organization (ISO), 2020) for nuclear facilities to implement to assist with the generation of NCS controls and limits.

NCS computational tools and validation of computational methods

Producing NCS controls and limits requires NCS staff to work with process operations staff to determine the normal and credible abnormal conditions for the process. Normal conditions are those operations performed in accordance with an operating procedure. Obviously, the fissile process shall be designed such that these operations remain safe or subcritical ($k_{\text{eff}} \leq 1.0$) at all times during normal operations. Occasionally, equipment failures or personnel errors occur during process operations. Those failures or errors that are credible are considered in NCS analysis and are used to derive appropriate controls on NCS parameters, e.g., mass, moderation, etc., and control limits to ensure the consequences of a credible abnormal condition is subcritical. The consequences of a credible abnormal condition rely on the formality of operations, safety culture health, and process characteristics to determine the magnitude of the credible abnormal condition, which might consist of full water flooding rather than partial flooding of a fissile process or an over-accumulation of fissile material of a credible magnitude, such as accumulating 5000 g of plutonium for a process with a mass limit of 4500 g of plutonium.

NCS engineers perform NCS analysis using a computer code system, such as SCALE (Rearden, 2016), using the most recent neutron cross sections to calculate the k_{eff} of the normal or credible abnormal process configuration. A consensus standard is available (American National Standards Institute, 2008) to provide guidance to those performing k_{eff} calculations to ensure normal and credible abnormal conditions are subcritical. Validation is the process of quantifying the suitability of a computer code system for use in NCS analysis by comparing calculation results with the results of critical benchmarks ($k_{\text{eff}} = 1.0$) to establish the appropriate bias and bias uncertainty (American National Standards Institute, 2008). Further, uncertainties in the data, such as benchmark

descriptions, calculational models for the benchmarks, and the calculational methods (code bugs, nuclear data errors, etc.), need to be considered in the validation to ensure the upper subcritical limit is not exceeded during process operations.

It should be noted that the k_{eff} is not a measure of safety margin of a fissile process but, instead, represents the neutron balance of the process (neutron production and losses). The upper subcritical limit (USL) limit on a calculated k_{eff} value is established to ensure that calculated values represent a subcritical system. Consensus standards recommend the USL be less than unity ($k_{eff} < 1.0$). The USL is typically lower for fissile material configurations with few or no critical experiment benchmarks and higher when there is more experimental benchmark data to use for validation purposes, to provide additional safety margin to compensate for uncertainty of lesser-known behavior. To ensure conservatism, some regulators will mandate lower k_{eff} USLs, e.g., $k_{eff} \leq 0.95$, without a technical basis. NCS engineers will tend to apply administrative margin to reduce the USL further when experimental benchmark data is sparse, nuclear data is uncertain, or operational parameters are not well constrained to ensure NCS for certain fissile processes, such as with solution operations for which concentrations or other operational factors are not well known.

Criticality accident concerns in the light water reactor fuel cycle

The Light Water Reactor (LWR) fuel cycle involves the processing, handling and storage of fissionable materials. The LWR fuel cycle is summarized in this section with an emphasis on those fuel cycle steps where accidental criticality is credible. Although accidental criticality has significant worker safety concerns for some fuel cycle steps, NCS concerns must be balanced with other worker safety concerns, overall worker risk, and cost versus benefit. NCS practice includes reviewing fuel cycle processes and implementing effective engineered and administrative controls in accordance with facility procedures and the NCS consensus standards. Fig. 1 illustrates the LWR fuel cycle with the blue boxes indicating those activities that do not require NCS controls (because criticality accident is not credible) and the black boxes indicate activities that require NCS controls (criticality accident is credible) (Knief, 1985). The credibility of a criticality accident depends upon health of the formality of operations and safety culture, location of the facility, age of the facility, worker experience, and other factors. The dashed lines connecting the boxes indicate the front end of the fuel cycle and the solid lines show the back end of the fuel cycle. Reactor operations are not considered to be within the purview of NCS consensus standards, because other nuclear safety controls are in place to ensure criticality in reactors is controlled. The NCS concerns in each LWR fuel cycle step will be discussed in subsequent sections. Safety considerations in the design, licensing, operation and decommissioning of nuclear reactors are addressed by Rempe (2021).

Uranium mining, milling, and processing operations

Uranium ore extracted from mining activities has an enrichment of about 0.711 weight (wt.) percent (%) ^{235}U and 99.3 wt. % ^{238}U ; this is natural enrichment. Ores are milled and then leached with chemicals to produce yellow cake (U_3O_8). Mill tailings leached of uranium (but consisting of more than 99% of the ore mass) are disposed. The yellow cake is then processed at conversion plants to chemically react the U_3O_8 with fluorine to generate uranium hexafluoride (UF_6) gas that is then solidified for transportation to uranium enrichment facilities (Freiwald, LASL-75-15, 1978). Due to the low concentration of ^{235}U in the uranium (i.e., at natural enrichment), there are no credible NCS concerns for these materials in nominal fuel cycle processing. Natural uranium can be made critical only under conditions involving heterogeneous arrangements (appropriate diameter and spacing) of uranium metal rods in water. The minimum possible enrichment of uranium in a homogeneous mixture with water under optimum conditions of neutron moderation and neutron reflection is about 1 wt. % ^{235}U (Thomas, 1978). Mining, milling, and processing operations are limited to natural enrichment, 0.711 wt. % ^{235}U , and will remain subcritical during these activities under any foreseen conditions.

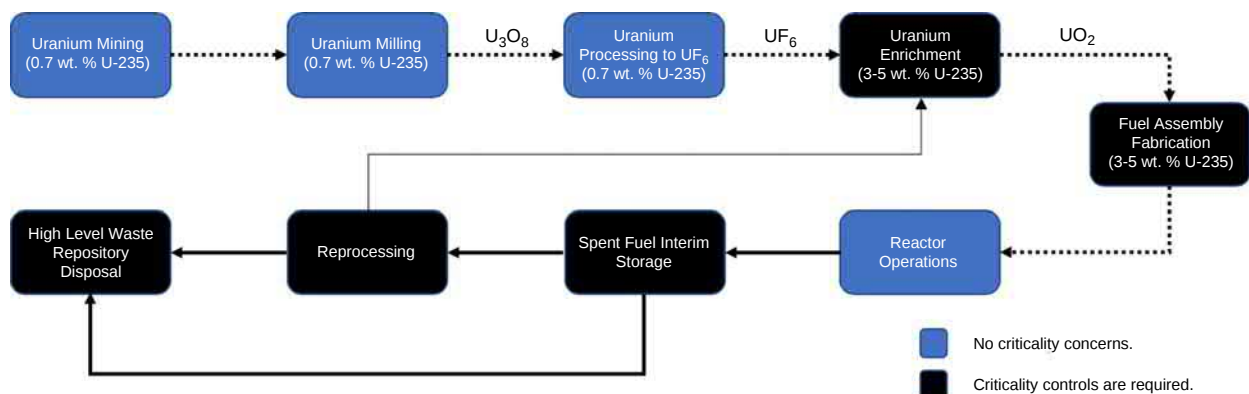


Fig. 1 Light water reactor fuel cycle.

Uranium enrichment operations (3–5 wt. % ^{235}U)

Enriched uranium is needed to fuel LWRs for electrical power generation. Uranium is enriched by increasing the content of ^{235}U in the natural uranium from 0.711 wt. % ^{235}U to about 3–5 wt. % ^{235}U . In the U.S., most uranium was enriched at the Oak Ridge Gaseous Diffusion Plant in Oak Ridge, TN, the Portsmouth Gaseous Diffusion Plant in Piketon, OH, and the Paducah Gaseous Diffusion Plant in Paducah, KY. These enrichment facilities were closed down in 1985, 2001, and 2013, respectively. In 2004, a Gas Centrifuge enrichment facility began production in Eunice, NM, to continue enrichment operations in the United States for LWR operations.

Gaseous diffusion enrichment process

The Gaseous Diffusion Process involves forcing uranium hexafluoride gas, UF_6 , through a converter containing a diffusion membrane that slightly increases the enrichment the lighter ^{235}U isotope. This process requires large compressors that maintain a high pressure on the feed side of the diffusion membranes to ensure sufficient flow rates to make an efficient process (Reed, 2011). Enriching ^{235}U to 3–5 wt. % for the LWR fuel cycle requires the process to be repeated many times due to the low enrichment efficiency of each diffusion stage. The Gaseous Diffusion cascade is safe from an NCS perspective by controlling moderator (humid air) inside the cascade piping. The water in humid air can react with UF_6 gas to form hydrated uranyl oxyfluoride compounds ($\text{UO}_x\text{F}_y \cdot z\text{H}_2\text{O}$) that can settle and deposit locally. These deposits, commonly UO_2F_2 , can grow over time in the vicinity of a breach in the gaseous diffusion cascade that can result in a condition (critical mass of ^{235}U that is moderated) that can support a criticality accident. The approximate minimum critical mass of ^{235}U is 7.6 kg for 3.0 wt. % ^{235}U enrichment and 3.0 kg for 5.0 wt. % ^{235}U enrichment at a hydrogen-to- ^{235}U (H/ ^{235}U) ratio, or moderation level, of about 500 (Jordan and Turner, 1992). In the 1990s, two significant uranium deposits (479 kg U and 98.5 kg U with enrichments of about 3.34 wt. % ^{235}U) were found in cascade process gas piping at the K-25 Gaseous Diffusion Plant in Oak Ridge, TN, resulting from the accumulation of uranium from a humid air in-leakage over a long period of time. Although a criticality accident was not possible due to the mass, deposit geometry, and level of moderation (H/U of 3.5), the unanticipated and undetected accumulation of enriched uranium underscores the importance of NCS principles to fissionable material processing. Gaseous Diffusion plant operations require large, powerful, pumps and support equipment that are unfavorable from an NCS perspective.

Gas centrifuge enrichment process

The Gas Centrifuge Enrichment process is currently the enrichment method in the United States. Unlike the Gaseous Diffusion Process, the Gas Centrifuge Enrichment Plant has been designed to be safer from an NCS perspective. The plant design assumes a maximum plant enrichment of 6 wt. % ^{235}U , although the maximum planned enrichment will not exceed 5 wt. % ^{235}U under normal and credible abnormal conditions. In addition to controlling the uranium enrichment, process equipment dimensions are also credited for safety based upon the geometrical dependencies identified in NCS criticality safety analysis. For example, the gas centrifuges and process gas piping containing gaseous UF_6 are designed to be less than the critical diameter (at which $k_{\text{eff}} = 1$) of an infinite cylinder with full water reflection, which is 24.4 cm (9.6 in.) (Louisiana Energy Services, 2005). A safety factor of 0.9 is applied to this critical diameter to derive a safe diameter: the maximum diameter of gas centrifuges, chemical traps, process cold traps, and process gas piping, then, is 21.9 cm (8.6 in.). Vacuum cleaners, oil containers, and UF_6 pumps are designed based on a safe volume for 6 wt. % enriched uranium: 18.0 L (4.8 gal) (Louisiana Energy Services, 2005). UF_6 feed cylinders (identified as 48X and 48Y cylinders) and UF_6 product cylinders are safe from an NCS perspective with a limitation on available moderator inside the cylinder with solid UF_6 present. Passive engineered features considered early in the plant design makes NCS less reliant on administrative controls and human interaction over the life of the plant/facility. It is not possible to totally engineer out the possibility of human error, but NCS is significantly enhanced with passive and active engineered controls. Active engineered controls, such as control valves, require more human interaction over time for operation and maintenance and allow more opportunity for error, while passive engineered controls, such as equipment dimensions, provide more robust protection against human error. Administrative controls, such as limits to fissile mass or fissile concentration allowed in a process, are also used for NCS when engineered controls cannot be used.

Laser-based uranium enrichment

A proposal for a uranium enrichment considers a laser-based enrichment technology to enrich natural (0.711 wt. % ^{235}U) UF_6 gas. Natural UF_6 gas is separated into a product UF_6 gas stream that can be enriched up to 8 wt. % ^{235}U and a tails stream that is depleted in ^{235}U (United States Nuclear Regulatory Commission, 2012). The cascade or gas handling area of the facility will utilize lasers sufficiently tuned to excite the gaseous UF_6 molecules to separate ^{235}U from ^{238}U in the UF_6 feed stream. The tails stream that is depleted in ^{235}U , is treated in a similar way as depleted UF_6 in gaseous diffusion or centrifuge enrichment processes. NCS is ensured through limiting uranium enrichment, limiting moderation and controlling the size of the equipment (geometry).

Fuel assembly fabrication

High-purity, solid UF_6 is shipped from the enrichment facilities to fuel fabrication facilities, typically in 30B containers. The 30B containers can be used to ship UF_6 up to an enrichment of about 5 wt. % ^{235}U , with NCS limits on volume, interaction (spacing

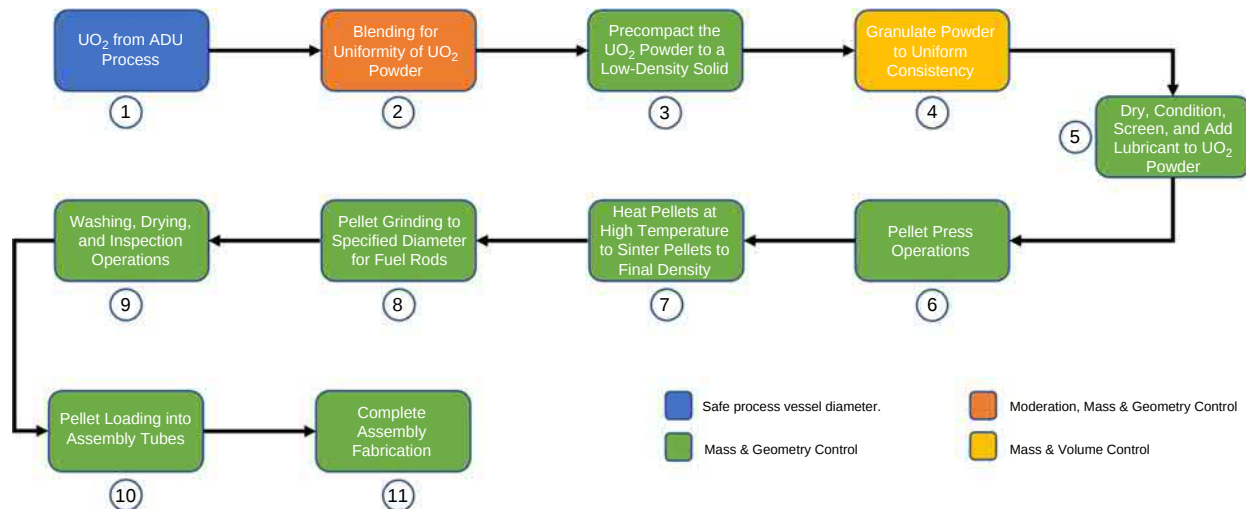


Fig. 2 LWR fuel fabrication process and NCS controls needed for subcriticality. Adapted from Knief RA (1992) *Nuclear Engineering—Theory and Technology of Commercial Nuclear Power*. Washington, DC: Hemisphere Publishing Corporation.

between containers), and moderation (Knief, 1985). Fig. 2 illustrates the LWR fuel fabrication process and identifies the NCS controls required to maintain subcriticality during normal and credible abnormal conditions.

UF₆ to UO₂ conversion processes

The conversion of solid UF₆ feed to UO₂ powder is more challenging from an NCS perspective because the uranium is present as aqueous solution. The critical mass of uranium under optimum water moderation and reflection conditions is significantly reduced from tens of kilograms for solid materials (metal, oxides, and oxide powders) to a kilogram or less when the uranium and moderator (water, acid, etc.) are intimately mixed. There are two main UF₆-to-UO₂ conversion methods utilized in the United States: 1. The Ammonium DiUranate (ADU) process and the 2. Ammonium Uranyl Carbonate (AUC) process. The ADU process involves the hydrolysis of UF₆ with water to form UO₂F₂ and hydrofluoric acid in an aqueous solution (DeLuca and Maas, 1977). This solution requires NCS controls on process vessel and product solution tank diameters. Aqueous solutions are assumed to be optimally moderated, so control of the solution geometry is necessary for NCS to ensure desired material throughput can be maintained. The subcritical tank diameter (of infinite length and fully water reflected) for 5.0 wt. % ²³⁵U in aqueous UO₂F₂ solution is 20.1 cm (7.9 in.) (American National Standards Institute, 2014). For uranium enriched to 3.0 wt. % ²³⁵U, the subcritical tank diameter increases to 37.4 cm (14.7 in.) (American National Standards Institute, 2014). Aqueous ammonia is added to the aqueous UO₂F₂ solution to create ammonium diuranate precipitate that is collected, dried and heated to produce UO₂. The AUC process is similar, except that ammonium carbonate is used as the precipitating agent, so similar approaches to control geometry to safe configurations are used.

UO₂ powder processes

After the UF₆-to-UO₂ conversion process is complete, the UO₂ powder is pressed into pellets for fuel assembly fabrication. Fig. 2 shows each step in the conversion of UO₂ powder to fuel assemblies. During this process, the UO₂ powder must be kept dry (moderation control). Fissile powders can easily mix with moderators, e.g., water, oil, etc., and flow into unfavorable configurations, resulting in NCS concerns. Keeping UO₂ powder dry precludes any reduction in the critical mass below that of the subcritical limit for UO₂ powders at full density, spherical geometry, less than 1.5 wt. % water, and full water reflection, which is 32.3 kg of ²³⁵U (American National Standards Institute, 2014). The UO₂ powder is then blended to ensure uranium enrichment uniformity, compensating for the mixing of different batches with slight variations in enrichment (Knief, 1992). NCS at this stage involves limits on the mass, moderation and geometry parameter controls (blender shape/geometry) during processing. Hydrogenous “pore former” additives are blended with the powders to provide for porosity control during the final sintering process later on, and this hydrogenous material is taken into account for determining mass limits and geometry controls. NCS for the step to precompact the UO₂ powder to a low-density solid involves controlling the mass and geometry parameters while NCS for the next step, powder granulation to ensure uniform consistency, relies on mass and volume (hopper size) control.

Pellet operations

Fig. 2, steps 6-11, involve pressing powders into pellets to prepare for fuel assembly fabrication (pressing, sintering, grinding, washing/inspection, pellet loading and assembly fabrication). NCS control is maintained during the pellet manufacturing process

by using safe geometry transfer hoppers and powder lines and also by using passive engineered controls for conveyors and furnace boats which limiting the pellet stacking below a subcritical slab thickness. This subcritical slab thickness is about 10 cm (4-in) thick and is the primary NCS limit applied to processing, transport and in-process storage. A 10-cm-thick water-reflected slab is subcritical for dry UO_2 full density powders up to an enrichment of 5.0 wt. % ^{235}U (see Fig. 25 by Paxton and Pruvost (1986)). One of the implementation methods for this slab thickness control is a molybdenum furnace boat that is open on the top and filled with UO_2 pellets before and after the sintering process and will remain subcritical even flooded with water during a credible abnormal condition. Pellet sintering consolidates the powder particulate to result in a denser pellet. The resulting dimensions are intentionally larger, and the pellet diameters must be ground to the exact dimensions, which results in the generation of UO_2 powder, which remains subcritical due to favorable geometry equipment. Engineered controls preclude the stacking of pellets during operations (limitation on mass), ensuring pellet tray physical spacing (limitation on neutron interaction) and perforating the trays so water cannot accumulate in an abnormal condition (limitation on moderation). After the pellets are fabricated and ground to final dimensions, the pellets are washed, dried, and inspected. Water or air is used to remove impurities that may be present and then the pellets are dried in an oven to remove excess moisture. During inspection, each pellet is carefully inspected for defects prior to moving to the fuel rod fabrication process. For this step, NCS is ensured by using favorable geometry equipment and spacing (interaction) controls on pellet trays.

Loading UO_2 pellets into fuel rods and assemblies

The completed fuel pellets are loaded into tubes made of suitable cladding materials, such as zirconium alloy or stainless steel, then the tubes are sealed, charged with helium, and inspected. Subcritical slab geometry ensures subcriticality during fuel rod fabrication activities, even fully flooded with water. Collections of rods that need to be stored over time are put into metal horizontal storage boxes with covers in spaced arrays limited to a subcritical slab height (Knief, 1985). Completed fuel rods are about 4.3 m (~14 ft.) long (Knief, 1985).

Fuel assembly handling, storage, and transportation

Fresh fuel assembly handling, storage, and transportation require NCS controls derived from an NCS evaluation to ensure normal and credible abnormal conditions are subcritical. An NCS consensus standard is available that provides guidance on these operations with LWR fuel assemblies outside reactors (American National Standards Institute, 2009).

Rods are assembled into rectangular arrays of various sizes, ranging from 7×7 for older boiling water reactor designs to 17×17 for some pressurized water reactor designs. Once assembled, the fuel assemblies, or bundles, are handled carefully and require NCS controls to prevent water flooding (moderation control) and neutron interaction (spacing requirements). Several fuel bundles, two for some PWR bundle designs, can achieve criticality with optimum side-by-side spacing and full water flooding (Knief, 1985). A prudent NCS control for fuel assembly operations (assembly, cleaning, and inspection) typically restricts operations to one fuel bundle at a time, as a single bundle is they are subcritical when flooded by water. After the fuel assemblies are inspected, the fresh fuel assemblies are placed in storage racks designed to provide safe spacing between fuel assemblies (interaction control) and to preclude the accumulation of water near the fresh assemblies (moderation control). The transportation of fresh LWR fuel assemblies is accomplished by loading them into shipping containers for transport to a nuclear power plant. The shipping containers used for transportation activities in the U.S. are certified via U.S. Nuclear Regulatory Commission and US Department of Transportation regulations, 10 CFR Part 71 (Office of the Federal Register National Archives and Records Administration, 2020a) and 49 CFR Part 173 (Office of the Federal Register National Archives and Records Administration, 2020b). The packaging certification process in the U.S. requires the shipping package to be designed to withstand all Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) during transport operations. Based on the package design, package testing, number of packages, and proposed payload (in this case LWR fuel assemblies), a Criticality Safety Index can be derived to ensure packages will remain subcritical under NCT and HAC conditions. At the reactor site, LWR fuel assemblies are handled and stored as they were at the fuel fabrication facility. They are removed from the shipping containers and placed into storage racks or into water pool storage. The storage racks or pool storage locations may contain neutron poisons (absorption), such as gadolinium or boron, to provide additional safety margin during storage and handling. When the reactor core is loaded, fuel assemblies are removed from their storage racks one at a time and loaded into their core positions. Subcriticality of the core is assured by insertion of control and safety rods and, in PWR, adding soluble boron to the water.

Spent fuel reprocessing

Spent fuel can be reprocessed to recover unused fissile constituents from the spent fuel and to separate and remove fission products. Recovering the unused ^{235}U and the plutonium (^{239}Pu) produced by in-reactor neutron absorption in ^{238}U from the fuel allows reuse of these materials. Recovered uranium can be sent to an enrichment facility for further enrichment. The recovered ^{239}Pu can be blended with the uranium to make mixed oxide fuel (MOX). The separated fission products are disposed as waste. Spent fuel reprocessing is not currently part of the fuel cycle in the U.S. but is utilized in France, United Kingdom, Russia, and Japan. Spent fuel reprocessing involves the following steps, all of which require NCS control and guidance:

- Spent fuel transportation to a reprocessing facility.
- Spent fuel receipt, handling and interim storage.
- Fuel assembly disassembly and shearing operations.
- Dissolution of the spent fuel rod segments in nitric acid.
- Separation of fissionable materials (U and Pu) from aqueous solutions.
- Separation of heavy metals, fission products, and transuranic waste from the solution.
- Product preparation, handling, storage and transportation.
- Waste preparation, handling storage and transportation.

Spent fuel is transported in specially designed shielded casks that meet transportation safety regulations for transporting hazardous materials. These casks have been designed to prevent a criticality accident due to water intrusion and neutron interaction with other fuel, prevent the loss of any of the radioactive material, and to provide shielding and heat dissipation during transportation activities. NCS during spent fuel receipt, handling and interim storage is assured by handling assemblies in much the same manner as if they were fresh fuel, although the radiological hazards tend to outweigh the NCS hazards with spent fuel.

Fuel disassembly and shearing operations involve removing the individual fuel rods from the fuel bundles and shearing the rods into small segments (cladding and fuel pellet fragments) to facilitate efficient dissolution. Typically, NCS is considered with a limit of one spent LWR fuel assembly at a time for disassembly and shearing operations. After the fuel cladding is removed, the fuel segments are then dissolved in nitric acid to create an aqueous solution of spent fuel constituents (fissile and nonfissile actinides and fission products) and cladding materials. NCS control is essential during the dissolution stage because the combination of heterogeneous fuel pieces and fuel solution can be more reactive than the fuel assembly or aqueous solution separately (Knief, 1985). To allow a high throughput of sheared spent fuel dissolution the dissolver must be large enough to ensure complete and efficient dissolution operations can be performed, which is counter to NCS practice to reduce vessel diameter wherever possible. Because dissolver size is an NCS concern with fissile materials present, some dissolver tanks have been constructed with an annular geometry to increase neutron leakage and, consequently, to increase critical mass. Dissolved neutron poisons, such as cadmium, gadolinium, or boron either in the fuel or added to the dissolver will increase the critical mass, consequently increasing the safety margin, of the system.

After the fuel is dissolved, the fission products are separated from the uranium and plutonium present in the aqueous solution via various processes discussed by Taylor (2021). The product stream containing uranium and plutonium is sent for uranium and plutonium separation processing and the fission products and soluble poison are sent for waste processing. Soluble poisons in the aqueous solutions are no longer required for criticality control in the product stream. Criticality controls in the product stream are typically limitations to fissile material concentration and favorable equipment geometry. Further, tanks are typically small diameter with annular features to increase neutron leakage in the system. Further, the columns of these small diameter tanks for solution storage are typically spaced from each other such that neutron interaction between tanks is minimized. Uranium and plutonium product materials are converted to uranium oxide, plutonium oxide, or mixed oxide and can be shipped using a certified shipping container to fuel fabrication or enrichment facilities for further processing to recycle the material into LWR fuel.

High-activity, intermediate-activity and low-activity wastes from reprocessing are typically staged or stored in large volume vessels with fixed neutron poisons (neutron absorbing plates or tubes) or soluble neutron poisons, small-diameter tanks (favorable geometry), slab tanks (favorable geometry), or annular tanks (favorable geometry). These waste materials usually have no further use and need to be processed for safe long-term storage as liquid waste or dried and vitrified in stable form.

Spent fuel handling and storage

Fuel assemblies are resident in the core of an LWR for about 5–6 years. At end of life, the fuel assemblies are removed from the core and transported under water one assembly at a time to a spent fuel pool for interim storage for a few years. This time period allows the fission products to decay and for the assemblies to cool down sufficiently for other storage options. Typically, about 80% of the ^{235}U in the fuel assemblies is consumed during reactor operations; however, other fissile isotopes, such as ^{239}Pu , created from neutron absorption by ^{238}U are present in the spent fuel. Spent fuel pools are large, steel-lined, concrete water pools that are designed specifically to cool the spent fuel stored there. They are about 13.3 m (40 ft) deep to allow for attenuation of radiation. NCS is challenged if the fuel assembly is dropped (loss of fuel assembly geometry) or if the assembly is misplaced in the pool (storage array geometry change).

Because the U.S. has no commercial reprocessing facility, there has been a need to increase spent fuel storage capacity. NCS concerns with increasing spent fuel storage capacity in pools can be addressed by double tiering or stacking the fuel assemblies vertically (interaction as primary NCS control). Or boric acid can be added to the water (neutron absorption) allowing capacity increase by reducing the necessary spacing between the fuel assemblies in the pool. Solid neutron absorbers (neutron absorption) can be added to spent fuel storage racks to significantly increase storage capacity as well.

Because spent fuel is less neutronically reactive than fresh fuel, NCS evaluations for these systems can take advantage of a consensus standard for burnup credit for LWR fuel that accounts for reactivity effects of fuel burnup in a UO_2 -fueled LWR (American National Standards Institute, 2008). Burnup credit can be applied by estimating the fission product inventory in the spent fuel using computational tools such as ORIGEN (Rearden, 2016) and including that inventory in criticality calculations. Spent fuel contains fission products that can absorb neutrons from the overall system and increase nuclear criticality safety.

After the spent fuel has been stored for a few years in the spent fuel pool, the fuel assemblies can be removed from the pool storage racks and loaded into dry-storage casks at the reactor site. The fuel is loaded into a sealed steel cask that has an inert atmosphere. There are various dry storage cask system designs. With some designs, the steel cylinders containing the spent fuel assemblies are placed vertically in a concrete vault; other designs orient the cylinders horizontally. The concrete vaults provide the radiation shielding and are designed to ensure cooling sufficient to keep fuel at safe storage temperatures. Other cask designs orient the steel cylinder vertically on a concrete pad at a dry cask storage site and use both metal and concrete outer cylinders for radiation shielding. (United States Nuclear Regulatory Commission, 2019) Each cask contains 24–72 spent fuel assemblies, depending on the type of the fuel assembly, and NCS is ensured by spacing the individual casks, limiting water (moderation control), the use of fixed neutron poisons such as boron (absorber control), and limiting the number of assemblies in the storage design (mass control). Fuel assemblies can be stored long term in casks, or the fuel can be moved via approved shipping containers to long-term cask storage facilities.

The Onkalo spent nuclear fuel repository in Finland is the only high-level spent fuel repository to be operational in the near future (early 2020s) (Posiva Oy, 2020). Despite the lack of spent fuel repositories, NCS is a major factor in the transportation and storage strategy for a national repository. For spent fuel transportation, a variety of highly robust shipping containers for over-the-road and rail transport have been designed for shipping spent fuel to a repository. These shipping containers are designed in accordance with applicable regulations to withstand both NCT and HAC during transportation activities. NCS is also considered for pre-closure repository activities that include shipping and receiving operations, repackaging into long-term storage containers, and placement into the repository storage locations. Pre-closure NCS controls involve mass controls on the number of casks or fuel assemblies allowed to be handled at one time or limits on the number items stored in a location. NCS for cask storage involves limits on the spacing between casks to keep neutron leakage as high as practical (interaction). Post-closure activities include the long-term storage of the spent fuel and potential long-term retrieval operations. NCS for long-term storage considers burnup credit for the presence of fission product neutron absorbers to allow for more dense fuel assembly storage (interaction control). The use of robust container degradation over the years requires a robust storage container design to preclude the intrusion of water (moderation control) over time to minimize the risk of criticality after a repository has closed.

Nuclear criticality safety of research reactor fuels

There are no NCS concerns for an operating research reactor. However, NCS concerns arise when fresh fuel must be handled or removed from the core. Cole, et al., (2021) provides some background on research reactor fuel materials. Some research reactors are fueled with high enriched uranium (uranium with 93 wt. % ^{235}U), although the ^{235}U enrichment in most research reactors is 20% or lower. NCS is applied for fresh fuel and spent fuel handling. The fresh research reactor fuel would be analyzed for NCS and would consider controls on interaction (fuel spacing) and mass (one fuel assembly, fuel plate, or fuel item to be handled at a time). Spent fuel also has criticality concerns in the storage, handling and transportation steps of life and would likely involve NCS controls interaction (spacing), absorption (introduction of neutron poisons or burnup credit due to fission product absorbers present), and mass (number of fuel assemblies in a pool or storage cask). If stored fuel is to be spaced closer together for some reason, fixed poisons can be used in the storage locations to decrease neutron interaction, which makes the configuration safer from an NCS standpoint.

NCS with GEN-IV reactors

NCS with respect to the latest GEN-IV reactor designs (Stanculescu, 2021) such as Very High Temperature Reactors, Pebble Bed Modular Reactor, Molten Salt Reactors, Gas-cooled Fast Reactor (GFR), Sodium-cooled Fast Reactors (SFR), and Lead-cooled fast reactors (LFR) is applied similarly as for LWRs. The GFR, SFR, and LFR concepts use a predominately fast neutron spectrum, rather than a thermal neutron spectrum. In any case, NCS for the fuel cycles supporting these concepts is handled in the same way as to those processes for the LWR fuel cycle. The ANSI/ANS-8.1 standard (American National Standards Institute, 2014) requires that process analysis be performed on any process operation that involves fissionable materials outside of reactors. This process analysis requires close identification and analysis of the normal and credible abnormal conditions for processes involved with the processing, handling, storage, and transportation of the fissile materials present. The processes of interest (e.g., enrichment, fuel fabrication and handling), will each require NCS limits for controls on the NCS parameters (mass, absorption, geometry, interaction, concentration, moderation, enrichment, reflection, and volume) that will be similar to those needed for other, similar fuel cycle activities. For example, if a new reactor design considers a higher fuel enrichment of 20 wt. % ^{235}U rather than ~ 5 wt. % ^{235}U (typical for LWRs), this will result in more restrictive limits (larger spacing between fissile materials to increase leakage and decrease interaction) on NCS controls. Regardless, NCS is applied to a particular process in a similar way despite the differences in fissile materials and process environments in different fuel cycles.

Summary

NCS is inherently necessary for all fuel cycles involving the handling, processing, transportation, and storage of fissile isotopes (^{233}U , ^{235}U , and ^{239}Pu) to protect workers and the public. The discussion in this article has focused on those parts of the LWR

fuel cycle and described similar applications to new fuel cycle activities proposed with GEN-IV Reactor concepts. Those operations with very low fissile content, very small concentration or ^{235}U enrichment of less than about 1.0 wt. % ^{235}U have fewer NCS concerns than with operations with higher concentrations and enrichments for the LWR fuel cycle, research reactor fuel, or proposed GEN-IV advanced reactor fuel cycles. NCS fundamentals and background information were provided to understand the NCS hazards associated with fuel cycle processes to ensure all operations, normal and credible abnormal, remain subcritical to ensure facility workers and the public are safe from the consequences of a criticality accident. Consensus standards for NCS are available for use to assist NCS engineers and operations staff during the design, operation, and modification of hands-on fissile processes.

See Also: Physics of the Fission Chain Reaction in Thermal and Fast Reactors.

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Fresh Fuel Properties: Ceramic Compounds

Giovanni Pastore^{a,b} and Aysenur Toptan^b, ^a Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States; and ^b Computational Mechanics and Materials Department, Idaho National Laboratory, Idaho Falls, ID, United States

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Introduction

Oxide ceramic nuclear fuels were introduced in the 1950s and are still the most widely used fuel types in present-day commercial reactors. Among the unique features that qualify these compounds for nuclear fuel applications are their high melting point, high retention of fission products thanks to the open crystal structure, and highly variable oxygen-to-metal ratio that can shift to accommodate fuel chemistry changes with burnup, resulting in excellent resistance to radiation damage. Other advantages of oxide nuclear fuels include inertness to many coolants, prolonged irradiation with low swelling, and a relatively low fabrication cost. Oxide ceramic fuels include uranium dioxide (UO_2) and uranium-plutonium mixed-oxide ((U, Pu) O_2 , or MOX) fuels. UO_2 has been used in the light-water reactor (LWR) fuel rods (Reeve, 1975; Olander, 1976, 2009) and high-temperature gas-cooled reactor (HTGR) fuel particles (Reeve, 1975; Kania et al., 2013). MOX has been used in LWR and fast breeder reactor (FBR) fuel rods (Reeve, 1975; Olander, 1976, 2009; Waltar et al., 2012). Mixed thorium-uranium oxide ((Th, U) O_2) and thorium-plutonium oxide ((Th, Pu) O_2) are also ceramic compounds that have been used as nuclear fuels in HTGRs and LWRs (Reeve, 1975; IAEA, 2005).

The notable shortcomings of oxide ceramic fuels are their low uranium density and low thermal conductivity. UO_2 has a theoretical density (TD) of about 10,970 kg/m³, and its uranium density of about 9670 kg/m³ is approximately half that of uranium metal. The lower the uranium density, the lower the amount of fissile material and the maximum attainable length of the reactor operation cycle. Poor thermal conductivity results in high fuel centerline temperatures and high temperature gradients across the fuel element to extract sufficient fission heat per unit of fuel to make electric power production economical. At high temperatures, solid-state reactions proceed at rates adequate to produce significant changes in material properties during the fuel lifetime in the reactor. Thermal stresses resulting from the steep temperature gradient across the fuel pellet cause the fuel to crack in low-temperature regions or deform plastically in high-temperature zones. The temperature gradient also drives pore migration and redistribution of fuel constituents, such as oxygen, plutonium, and fission products, from their initial concentration profiles. However, pellet cracking can be allowed inside a structural metal cladding. The low thermal conductivity is compensated for by a high melting point of about 3100 K with an absence of phase changes up to the melting point and by the acceptable irradiation behavior of oxide ceramic fuels (Reeve, 1975; Olander, 1976).

Non-oxide ceramics, such as uranium and mixed uranium-plutonium carbides (UC and (U,Pu)C) and nitrides (UN and (U,Pu)N), have also been investigated for nuclear fuel applications (Reeve, 1975; Blank, 1988; Streit and Ingold, 2005; Terrani et al., 2020), motivated by advantages over oxide fuels that include higher uranium density and higher thermal conductivity. The uranium density of uranium monocarbide (UC) is 12,970 kg/m³, and that of uranium mononitride (UN) is 13,520 kg/m³. These values are about 35–40% higher than the uranium density of UO_2 . For this reason, non-oxide ceramic fuels are also referred to as dense ceramic fuels (Blank, 1988). Both uranium carbides and nitrides are candidates for application in FBRs (Crawford et al., 2007; Chandar et al., 2020). Moreover, uranium nitride is considered for application as nuclear fuel for future space reactor systems, as it exhibits good thermal and irradiation behavior as well as compatibility with many candidate space reactor coolants and structural materials (Hayes et al., 1990a). Finally, a solid solution of uranium carbide and nitride (U(C,N)) that is rich in the latter is being used to fabricate fuel particles currently under testing for use in LWRs and advanced reactors (Terrani et al., 2020).

UC exhibits a metallic electronic structure due to the overlaps of f-orbitals. Hyperstoichiometric UC_{1+x} contains a precipitate of UC_2 or U_2C_3 . For hyperstoichiometric uranium carbides, the metallic character persists, and the C–C bonds are covalent like in

graphite. The U-4f core levels do not change strongly with increasing carbon content, and the U-5f electrons are predominantly itinerant (Manara and Agarwal, 2020). Hypostoichiometric UC_{1-x} contains a grain boundary distribution of uranium metal, which leads to unfavorable in-reactor swelling behavior. The in-reactor behavior of (U,Pu)C is relatively similar to that of UC (Reeve, 1975).

In this article, we provide an overview of the main properties of oxide ceramic fuels UO_2 and MOX, as well as dense ceramic fuels UC and UN. In particular, thermal properties are dealt with in “Thermal properties” section, and mechanical properties are summarized in “Mechanical properties” section. For brevity, mixed uranium-plutonium carbides and nitrides, and thorium-based oxide ceramic fuels are not covered here in detail. Finally, this work is limited to fresh (i.e., zero burnup) fuel properties. Therefore, changes in properties with burnup and burnup-dependent phenomena, such as fuel densification and fission-product swelling, are not included in this article. Ceramic fuel irradiation performance is discussed by Cantonwine and Rand (2021).

Thermal properties

The thermal properties of ceramic fuel compounds addressed in this section include thermal conductivity, melting point, and specific heat capacity. Each subsection consists of an introduction and definition for the corresponding property, followed by examples of correlations used in computational analyses. A detailed discussion on the physical understanding of material properties is beyond the scope of this article. However, to lead into the mathematical formulations presented in the following, we mention here that the thermal conductivity of the considered materials has main contributions due to lattice vibrations (phonon component) and free electrons (electronic component), and the heat capacity is mainly a consequence of the vibrations of the atoms in the lattice.

Thermal conductivity

The thermal conductivity of UO_2 and MOX depends on temperature, fuel density, oxygen to metal (O/M) ratio, and additive content, in addition to burnup. The basic form used to represent the thermal conductivity of ceramic oxide fuels includes a lattice vibration (phonon) term and an electronic term, i.e.,

$$k_p = k_{ph} + k_{el} \quad (1)$$

where k_p (W/m K) is the thermal conductivity at a certain fuel density (ρ), k_{ph} the phonon term, and k_{el} the electronic term. The first term is typically of the form

$$k_{ph} = \frac{1}{A + BT} \quad (2)$$

where A (mK/W) is a parameter that relates to phonon scattering from lattice impurities and defects, B (m/W) a parameter that takes into account phonon-phonon interactions, and T (K) the temperature. The deviation from stoichiometry and dopants such as gadolinia (Gd_2O_3) affect the thermal conductivity in a manner that can be taken into account through additional terms in Eq. (2). Moreover, burnup accumulation creates fission product impurities in the lattice, which affect the phonon term. The electronic term is generally of the form

$$k_{el} = f(T)\exp(-C/T) \quad (3)$$

where f (W/m K) is a function of temperature and C (K) is an empirical constant.

Correlations for the thermal conductivity of UO_2 and MOX used in fuel behavior calculations are available, e.g., in Harding and Martin (1989), Massih et al. (1992), Lucuta et al. (1996), Pihlatie (1999), Ronchi et al. (1999), Duriez et al. (2000), and MATPRO (2003). An expression for the thermal conductivity of fresh UO_2 , given by Harding and Martin (1989) and also used by Lucuta et al. (1996), is

$$k_{100} = \frac{1}{0.0375 + 2.165 \times 10^{-4}T} + \frac{4.715 \times 10^9}{T^2} \exp\left(-\frac{16361}{T}\right) \quad (4)$$

where k_{100} (W/m K) denotes the thermal conductivity for fuel at 100% TD. Other correlations account for the effect of Gd_2O_3 content in UO_2 , such as the one from Massih et al. (1992).

$$k_{100} = \frac{1}{0.1149 + 2.48 \times 10^{-4}T + 1.1599g} + 1.216 \times 10^{-2} \exp(1.867 \times 10^{-3}T) \quad (5)$$

where g (wt. fraction) is the Gd_2O_3 content.

A correlation for the thermal conductivity of fresh MOX fuel is the one from Duriez et al. (2000).

$$k_{95} = \frac{1}{0.035 + 2.85x + (2.86 - 7.15x)10^{-4}T} + \frac{1.689 \times 10^9}{T^2} \exp\left(-\frac{13520}{T}\right) \quad (6)$$

where k_{95} (W/m K) denotes the thermal conductivity for fuel at 95% TD and $x = 2 - O/M$ is the deviation from stoichiometry, with O/M being the oxygen to metal ratio. Eq. (6) is based on data from MOX samples with low Pu content in the range 0.03–0.15 wt. fraction (Duriez et al., 2000).

Furthermore, a correction for the effect of sintering pores (and for fission gas bubbles in irradiated fuel) is introduced. In case of the closed porosity formed by isolated spherical pores, the Maxwell-Eucken formula can be used

$$k = k_{100} \left(\frac{1-p}{1+\beta p} \right) \quad (7)$$

where k (W/m K) is the thermal conductivity, p (–) the fractional porosity, and $\beta = 0.5$.

The thermal conductivity of UC involves dependencies that include temperature, carbon to uranium ratio (C/U), oxygen impurities, and grain size (Manara and Agarwal, 2020). A simple analytical relation proposed by Matzke (1986) and reported by Manara and Agarwal (2020) that can be used for modeling purposes is

$$k = \begin{cases} 20 & \text{for } T < 773K \\ 20 + 1.3 \times 10^{-3}(T - 773) & \text{for } T \geq 773K \end{cases} \quad (8)$$

where k (W/m K) is the thermal conductivity. Oxygen impurities lead to a decrease of the thermal conductivity of UC that can reach up to a factor of 2 for 15 at.% O (Holleck and Kleykamp, 1987). Also, the thermal conductivity of UC was reported to increase with the grain size at equal porosity (Holleck and Kleykamp, 1987).

The thermal conductivity of UN can be estimated with the correlation (NASA GRC, 2019)

$$k = \left(\frac{1-p}{1+p} \right) \begin{cases} 22.55T_r^{0.3845} & \text{for } 10 \leq T \leq 2025K \\ 29.58 & \text{for } 2025 < T \leq 2500K \end{cases} \quad (9)$$

where k (W/m K) is the thermal conductivity, p (–) the fractional porosity and $T_r = T/1000$ the reduced temperature. Eq. (9) is valid for $0 \leq p \leq 0.2$ and $10 \leq T \leq 2500$ K. Experimental evidence shows that the increase in the proportion of plutonium content in a mixed uranium-plutonium nitride fuel significantly reduces its thermal conductivity (Arai et al., 1993).

Thermal conductivity correlations for UO_2 , MOX, UC, and UN corresponding to Eqs. (4), (6), (8), (9), respectively, are plotted in Fig. 1 (a semi-log plot). Experimental data (Vlahovic et al., 2018; Duriez et al., 2000; De Coninck et al., 1975; Kurosaki et al., 2000) are also reported for comparison. Only undoped fuel is considered in Fig. 1. The plot highlights the significantly higher thermal conductivity of dense ceramic fuels UC and UN compared to oxide ceramic fuels UO_2 and MOX. Also, there is considerable

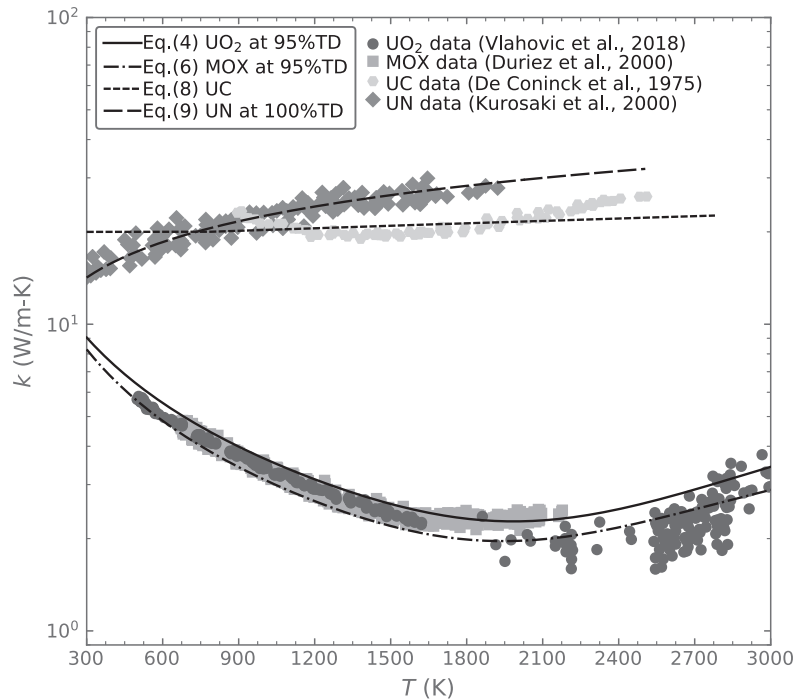


Fig. 1 The thermal conductivity of fresh stoichiometric UO_2 , MOX, UC, and UN with experimental data from Vlahovic et al. (2018), Duriez et al. (2000), De Coninck et al. (1975), and Kurosaki et al. (2000). For UO_2 , Eq. (4) has been used along with Eq. (7) to obtain the thermal conductivity at 95% TD, consistent with the experimental data from Vlahovic et al. (2018). For UC, the experimental data refer to samples with approximately 99–100%TD (De Coninck et al., 1975), while Eq. (8) is not associated with a specific density in (Manara and Agarwal, 2020).

dispersion in the experimental data for UO_2 and MOX, even in the limited database shown in Fig. 1. Significant dispersion can also be appreciated in the reported data for UC and UN. The uncertainty in the fuel thermal conductivity is one of the most important sources of uncertainty in the thermal analysis of nuclear fuel (Bouloré et al., 2012). Uncertainties can arise from three main sources in engineering modeling: (1) model input uncertainties due to the nature of the experiments providing the data (e.g., uncertainties in the physical geometry, experimental settings, biases, etc.), which are typically reported by the experimentalists with error bars; (2) model solution uncertainties, corresponding to discrepancies between the numerical approximations and the expected results; and (3) model form uncertainties, related to the accuracy of the mathematical model in representing the real-life behavior of the parameter of interest. An analyst can perform uncertainty quantification studies by propagating those uncertainties in the system to establish confidence in the response values.

Melting point

The solidus in the phase diagram of the fuel material represents the lowest temperature with an indication of melting and is loosely referred to as the melting point (Olander, 1976). The melting point of the fuel material is needed to define the limiting power of a fuel element.

The melting point of fresh stoichiometric UO_2 is in the range of 3073–3153 K (Hausner, 1965; Lyon and Baily, 1967; Brassfield et al., 1968; Latta and Fryxell, 1970; Olander, 1976). However, it is lower in non-stoichiometric $\text{UO}_{2\pm x}$. Also, mixed uranium-plutonium oxides, i.e., MOX fuels, melt at lower temperatures than pure UO_2 . Since UO_2 and PuO_2 form nearly ideal solid solutions, the melting point of the mixture varies smoothly from that of UO_2 to that of PuO_2 (Olander, 1976). Also, irradiation and the consequent buildup of fission-product impurities affect (reduce) the melting point of the fuel material (Olander, 1976). Correlations for the melting point of UO_2 and MOX as a function of PuO_2 content and burnup are given in the MATPRO material property library (MATPRO, 2003). According to MATPRO (2003), the melting temperature (solidus) of fresh stoichiometric UO_2 and mixed UO_2 and PuO_2 (MOX) is given as

$$T_m = 3113 - 5.414 \times 10^2 \omega + 7.468 \times 10^1 \omega^2 \quad (10)$$

where ω (wt. fraction) is the PuO_2 content. For irradiated fuel, a negative term proportional to burnup is included (MATPRO, 2003). Eq. (10) is based on the value of 3113 K for the melting temperature of UO_2 (Lyon and Baily, 1967; Brassfield et al., 1968) and the phase diagram for stoichiometric (U,Pu) O_2 from Brassfield et al. (1968). Note that, according to Lyon and Baily (1967), an uncertainty of ± 20 K is associated with the measurement for the value of 3113 K for the melting temperature of UO_2 .

The melting temperature of stoichiometric UC has been reported as $T_m = 2780$ K (Manara and Agarwal, 2020). Stoichiometric UC is stable from room temperature to its melting point. However, at temperatures greater than 1400 K UC can exist in both hypo-stoichiometric and hyperstoichiometric forms (Manara and Agarwal, 2020).

The vapor or decomposition pressure of UN is often experimentally studied rather than the melting point in the literature because UN melts only under high nitrogen partial pressures, and its decomposition occurs at lower pressures (both fuel melting and decomposition are undesirable conditions for nuclear fuel applications). To determine safety margins for UN fuel elements in engineering analyses, the temperature-pressure relationship for the melting or decomposition of UN is required. A correlation for the melting or decomposition temperature of UN valid for nitrogen pressures between 1×10^{-13} and 7.5 atm was developed in Hayes et al. (1990c), and reads

$$T_{m,d} = 3035.0(P_{N_2})^{0.02832} \quad (11)$$

where $T_{m,d}$ (K) is the melting or decomposition temperature and P_{N_2} (atm) is the nitrogen vapor pressure over UN. The latter is estimated by (Hayes et al., 1990c)

$$\log_{10}(P_{N_2}) = 1.8216 + 1.882 \times 10^{-3}T - 23543.4/T \quad (12)$$

for $1400 \leq T \leq 3170$ K. Eq. (12) is solved for temperature at a given nitrogen vapor pressure to estimate the temperature at which either melting or decomposition of UN occurs.

Specific heat capacity

Accurate knowledge of the specific heat capacity of the fuel material is required for assessment of fuel behavior under transient conditions, where the thermal diffusivity (which depends on the thermal conductivity, fuel density, and specific heat capacity) determines the time dependence of the temperature (Olander, 1976). The stored energy or enthalpy is also important in reactor transient analysis (e.g., reactivity-initiated accidents, or RIAs) because the severity of the transient is greatly affected by the initial stored energy of the fuel.

In the MATPRO library (MATPRO, 2003), an empirical correlation for the specific heat capacity at temperature T of solid UO_2 and PuO_2 is given in the form

$$c_p = \frac{K_1 \theta^2 \exp(\theta/T)}{T^2 [\exp(\theta/T) - 1]^2} + K_2 T + \frac{Y K_3 E_D}{2RT^2} \exp\left(-\frac{E_D}{RT}\right) \quad (13)$$

Table 1 Values for the parameters in the specific heat capacity and enthalpy correlations from MATPRO (2003), Eqs. (13) and (14).

	K_1 (J/kg K)	K_2 (J/kg K ²)	K_3 (J/kg)	θ (K)	E_D (J/mol)
UO ₂	296.7	2.43×10^{-2}	8.745×10^7	535.285	1.577×10^5
PuO ₂	347.4	3.95×10^{-4}	3.860×10^7	571.000	1.967×10^5

and the specific enthalpy at temperature T is derived by integrating Eq. (13) with respect to temperature, yielding

$$h = \frac{K_1 \theta}{\exp(\theta/T) - 1} + \frac{K_2 T^2}{2} + \frac{Y K_3}{2} \exp\left(-\frac{E_D}{RT}\right) \quad (14)$$

where c_p (J/kg K) is the specific heat capacity, h (J/kg) the specific enthalpy, $Y = O/M$ (–) the oxygen-to-metal ratio, R (8.314 J/mol K) the universal gas constant, θ (K) the Einstein temperature, and K_1 (J/kg K), K_2 (J/kg K²), K_3 (J/kg), E_D (J/mol) are constant parameters. The values for the parameters in Eqs. (13) and (14) relative to UO₂ and PuO₂ are reported in Table 1.

For a mixture of UO₂ and PuO₂ (MOX fuel), the specific heat capacity and specific enthalpy are evaluated as weighted averages of the contributions from each component (MATPRO, 2003)

$$c_{p,(U,Pu)O_2} = (1 - \omega)c_{p,UO_2} + \omega c_{p,PuO_2} \quad (15)$$

$$h_{(U,Pu)O_2} = (1 - \omega)h_{UO_2} + \omega h_{PuO_2} \quad (16)$$

where ω (wt. fraction) is the PuO₂ content.

The specific heat capacity of solid UC can be calculated as a function of temperature as (Manara and Agarwal, 2020)

$$c_p = 1.973 \times 10^2 + 1.012 \times 10^{-1} T - 7.352 \times 10^{-5} T^2 + 2.250 \times 10^{-8} T^3 - 2.435 \times 10^6 T^{-2} \quad (17)$$

Most of the U and Pu carbides show steep increase in heat capacities at temperatures above $0.6 T_m$, which is attributed to the formation of defects (Holley et al., 1984; Manara and Agarwal, 2020).

The specific heat capacity of UN can be evaluated by (NASA GRC, 2019)

$$c_p = \begin{cases} 2.784 \times 10^5 T_r^{2.168} / (1 + 91.91 T_r + 756.8 T_r^2 + 408.6 T_r^3) & \text{for } 5 \leq T \leq 293 \text{ K} \\ 2.14 \times 10^2 + 9.746 T_r + 1.718 \times 10^1 T_r^2 - 2.607 / T_r^2 & \text{for } 293 < T \leq 3000 \text{ K} \end{cases} \quad (18)$$

where $T_r = T/1000$ is the reduced temperature.

In Fig. 2, correlations for the specific heat capacity of UO₂, PuO₂, UC, and UN are compared to each other and to experimental data from (Kruger, 1968; Ronchi et al., 1993; Amaya et al., 2004; Moser and Kruger, 1967; Hayes et al., 1990c). Additionally, in Fig. 3, correlations for the enthalpy of UO₂ and PuO₂ are plotted along with experimental data from (Szwarc, 1969; Kruger, 1968). At moderate temperatures, the specific heat capacity of UO₂ is relatively similar to PuO₂. At higher temperatures (above ~2000 K), the specific heat capacity of UO₂ significantly differs from PuO₂. The specific heat capacity and enthalpy of (U,Pu)O₂ lie between the values for UO₂ and PuO₂. Lastly, specific heat capacities of UC and UN are comparable to each other and lower than the specific heat capacities of UO₂ and PuO₂, at least at temperatures below 2200–2600 K.

Mechanical properties

Mechanical properties of ceramic fuel compounds addressed in this section include thermal expansion, elasticity, plasticity, fracture strength, and creep. Each subsection consists of an introduction and definitions for the relevant properties, followed by representative values and correlations used in computational analyses.

Thermal expansion

With reference to thermal expansion, the thermodynamic property considered for the material is the coefficient of linear thermal expansion (CTE), which is defined as

$$\alpha = \frac{1}{l} \frac{dl}{dT} \quad (19)$$

where α (1/K) is the CTE and l (m) the length of the specimen.

The following empirical correlation has been recommended by Martin (1988) for the CTE of stoichiometric UO₂ and stoichiometric (U,Pu)O₂ in the solid phase

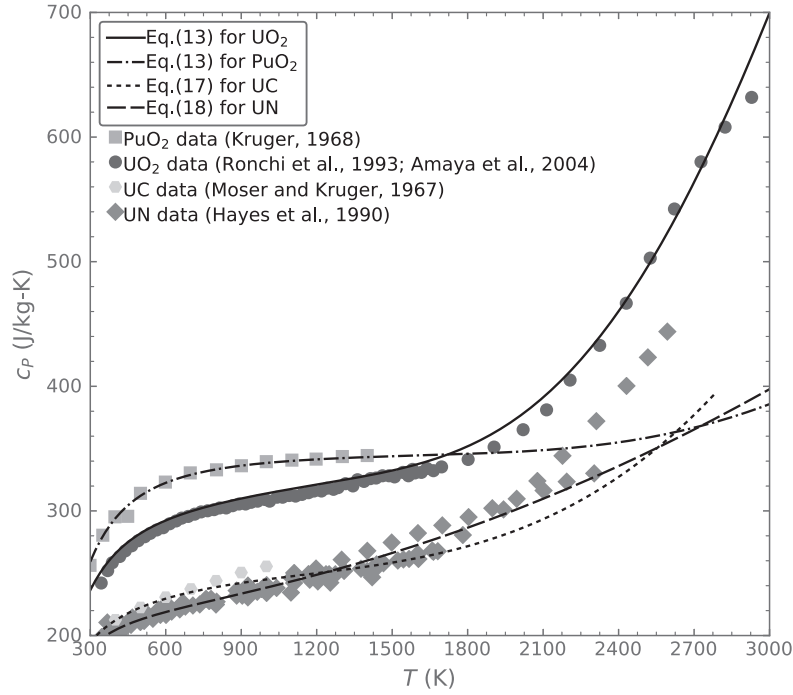


Fig. 2 The specific heat capacity of UO_2 , PuO_2 , UC , and UN with experimental data from Kruger (1968), Ronchi et al. (1993), Amaya et al. (2004), Moser and Kruger (1967), and Hayes et al. (1990c).

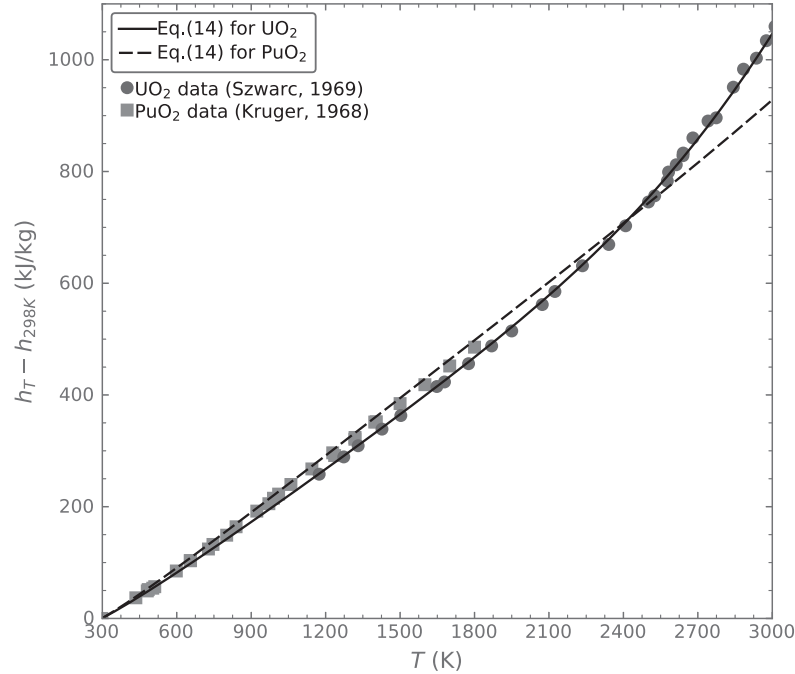


Fig. 3 The specific enthalpy of UO_2 and PuO_2 with experimental data from Szwarc (1969) and Kruger (1968). Data are given in terms of relative difference between the enthalpy values at temperature T and at 298 K.

$$\alpha_{\text{UO}_2} = \begin{cases} 9.828 \times 10^{-6} - 6.390 \times 10^{-10}T + 1.330 \times 10^{-12}T^2 - 1.757 \times 10^{-17}T^3 & \text{for } 273\text{K} \leq T \leq 923\text{K} \\ 1.1833 \times 10^{-5} - 5.013 \times 10^{-9}T + 3.756 \times 10^{-12}T^2 - 6.125 \times 10^{-17}T^3 & \text{for } 923\text{K} < T \leq 3120\text{K} \end{cases} \quad (20)$$

The effect of non-stoichiometry on the CTE of UO_2 is low, at least for $|O/M - 2| < 0.2$ (Martin, 1988; Baldock et al., 1966; MATPRO, 2003). It was established that Eq. (20) also describes the thermal expansion of solid hyperstoichiometric UO_{2+x} with

x values in the ranges 0–0.13 and 0.23–0.25 (Martin, 1988). Eq. (20) is valid for stoichiometric MOX with Pu content up to ~30% (Martin, 1988). For hypostoichiometric MOX, the CTE of $(\text{U,Pu})\text{O}_{2-x}$ can be evaluated as (Martin, 1988)

$$\alpha_{(\text{U,Pu})\text{O}_{2-x}} = \alpha_{\text{UO}_2} [1 + 3.9(\pm 0.9)x] \quad (21)$$

where α_{UO_2} is the CTE of stoichiometric UO_2 at the same temperature. For hyperstoichiometric MOX, Roth et al. (1967) found the following dependency of the CTE of $(\text{U,Pu})\text{O}_{2+x}$ on the deviation from stoichiometry, also taken up by Olander (1976).

$$\alpha_{(\text{U,Pu})\text{O}_{2+x}} = \alpha_{(\text{U,Pu})\text{O}_2} (1 - 5.1x) \quad (22)$$

where $\alpha_{(\text{U,Pu})\text{O}_2}$ is the CTE of stoichiometric $(\text{U,Pu})\text{O}_2$ of the same plutonium content at the same temperature. This relation was established for $1.94 < O/M < 2.01$ and only for a 20% PuO_2 – UO_2 mixture (Roth et al., 1967; Olander, 1976).

The following expression for the CTE of UC has been proposed in Manara and Agarwal (2020).

$$\alpha_{\text{UC}} = 10.08 \times 10^{-6} + 5.802 \times 10^{-9} \times (T - 273.16) \quad (23)$$

The CTE of UN in the range of $298 \leq T \leq 2523$ K has been given by Hayes et al. (1990a) as

$$\alpha_{\text{UN}} = 7.096 \times 10^{-6} + 1.409 \times 10^{-9} T \quad (24)$$

Correlations for the CTE of the fuel types considered here are plotted in Fig. 4.

Elastic constants

Like most other ceramics, polycrystalline UO_2 is a brittle material under normal conditions, i.e., at temperatures lower than about one half of the melting point (Olander, 1976). As stress is applied, a brittle substance deforms elastically, according to Hooke's law, until the critical stress value for fracture (fracture strength) is exceeded. For elastic deformation, the stress-strain curve is a straight line. The slope of the curve in a uniaxial tensile test is equal to Young's modulus of the material. The ratio of transverse contraction strain to longitudinal extension strain in the direction of the stretching force is Poisson's ratio of the material. The ratio of shear stress to shear strain is the shear modulus.

Young's modulus of oxide fuels is affected by the temperature, density and, to a lesser extent, oxygen-to-metal ratio and burnup of the fuel. Although published $(\text{U,Pu})\text{O}_2$ mixed oxide data are limited, it has been indicated that the addition of PuO_2 to UO_2 causes an increase in Young's modulus (MATPRO, 2003). The following correlation is given in MATPRO (2003) for Young's modulus of stoichiometric UO_2 below the melting temperature

$$E_s = 2.334 \times 10^{11} [1 - 2.752(1 - D)] [1 - 1.0915 \times 10^{-4} T] \quad (25)$$

where E_s (Pa) is Young's modulus and D (fraction of TD) the fuel density. For nonstoichiometric fuel or fuel that contains PuO_2 (MOX fuel), according to (MATPRO, 2003), Young's modulus below the melting temperature is described by

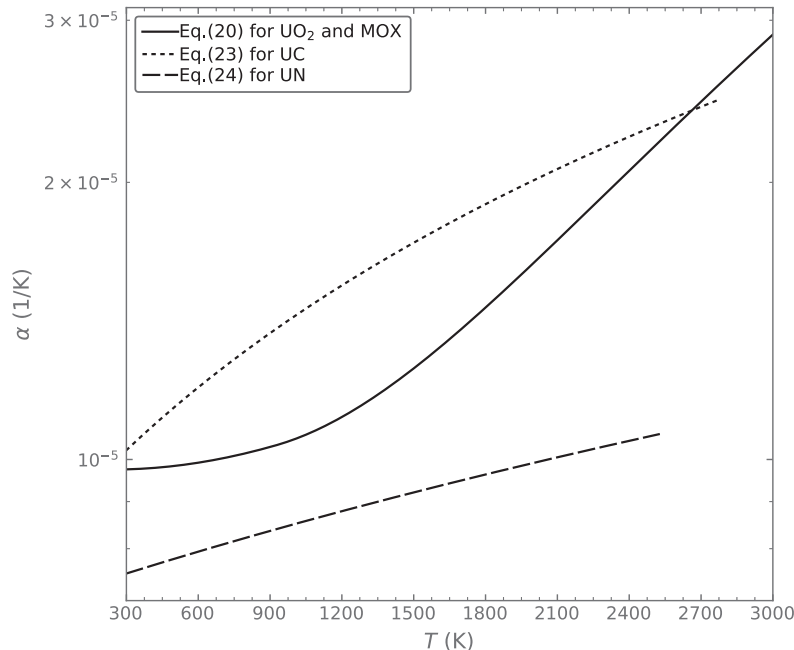


Fig. 4 Correlations for the linear thermal expansion coefficient of UO_2 and MOX, UC, and UN.

$$E = E_s \exp(-bx)[1 + 0.15\omega] \quad (26)$$

where E (Pa) is Young's modulus, b (–) is a parameter equal to 1.34 for hyperstoichiometric fuel or 1.75 for hypostoichiometric fuel, $x = 2 - O/M$ (–) is the deviation from stoichiometry, and ω (wt. fraction) is the PuO_2 content.

Correlations and experimental data for Young's modulus of UO_2 and MOX are reported in Fig. 5.

Poisson's ratio can be related to Young's modulus and the shear modulus as follows

$$\nu = \frac{E}{2G} - 1 \quad (27)$$

where ν (–) is Poisson's ratio, and G (Pa) is the shear modulus. Values between 0.306 and 0.320 have been reported for Poisson's ratio of UO_2 at room temperature (Wachtman et al., 1965; Padel and de Novion, 1969; Olander, 1976; MATPRO, 2003). Based on the effect of fuel porosity on Young's and shear moduli of UO_2 as determined by Marlowe and Kaznoff (1969), Olander (1976) gave the following formula for Poisson's ratio of UO_2 at room temperature as a function of porosity

$$\nu = 1.32(1 - 0.26p) - 1 \quad (28)$$

where p (–) is the fractional porosity.

For MOX fuel with 20% Pu content, a Poisson's ratio of 0.276 ± 0.094 at room temperature has been reported (Nutt et al., 1970; MATPRO, 2003; Sobolev, 2005).

Expressions for Young's modulus and Poisson's ratio of UC and UN at room temperature as a function of porosity have been determined experimentally by Padel and de Novion (1969). For UC, the expressions read

$$E = E_0(1 - 2.717p) \quad (29)$$

$$\nu = \nu_0(1 - 0.993p) \quad (30)$$

where $E_0 = 2.249 \times 10^{11}$ Pa and $\nu_0 = 0.288$ are the values for Young's modulus and Poisson's ratio, respectively, of UC at 100% TD (Padel and de Novion, 1969). For UN, the expressions read

$$E = E_0(1 - 2.307p) \quad (31)$$

$$\nu = \nu_0(1 - 1.313p) \quad (32)$$

where $E_0 = 2.668 \times 10^{11}$ Pa and $\nu_0 = 0.284$ are the values for UN at 100% TD (Padel and de Novion, 1969).

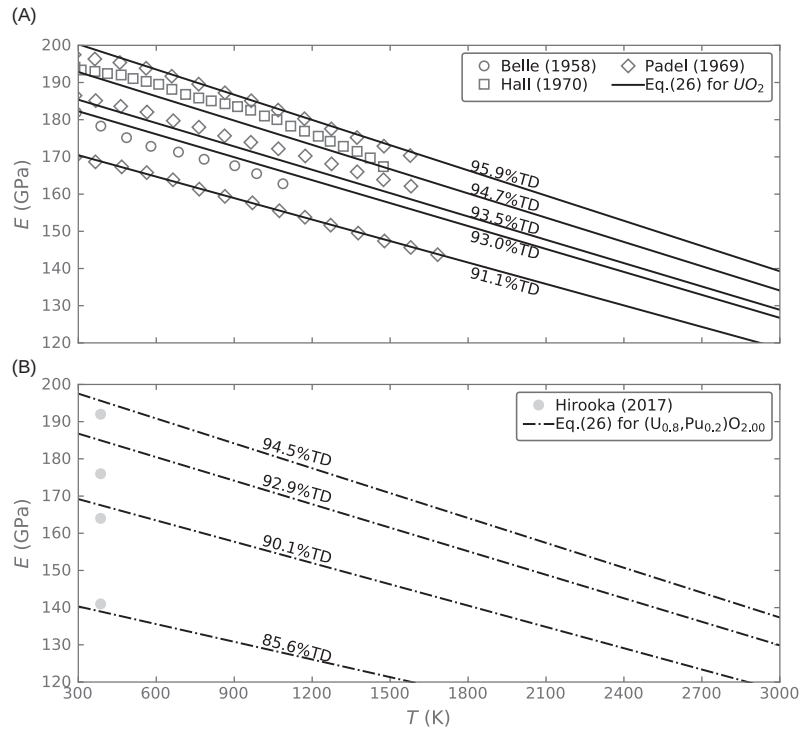


Fig. 5 Young's modulus of (A) UO_2 and (B) $(\text{U,Pu})\text{O}_2$ with a 20% Pu content with experimental data from Belle and Lustman (1978), Padel and de Novion (1969), Hall (1970), and Hirooka and Kato (2017).

Plasticity

At sufficiently high temperatures, even normally brittle materials such as UO_2 exhibit measurable amounts of unrecoverable or plastic deformation under applied stress before failure occurs (ductile behavior). The point on the stress-strain curve that indicates the limit of elastic behavior and the beginning of plastic behavior is the yield point, and the corresponding stress is the yield stress. The strains beyond the yield point are not recovered when the load is removed. The transition from brittle to ductile behavior occurs at a temperature characteristic of the material, called the ductile-brittle transition temperature or nil ductility temperature. In this regard, Olander (1976) identified three temperature regions for UO_2 :

1. A *brittle region* from room temperature to the ductile-brittle transition temperature. The latter is dependent on the strain rate imposed on the specimen during the test. It can be identified at about 1473 K for a strain rate of about 0.1 h^{-1} , based on test results by Canon et al. (1971).
2. A *semibrittle region* from the ductile-brittle transition temperature to the temperature above which purely ductile behavior is observed. The latter is estimated as 1673 K (Olander, 1976). A measurable plastic strain occurs before fracture, and yield stress can be determined. The ultimate strength of the material remains high.
3. A *region of complete ductility*, where the ultimate strength decreases rapidly with temperature and appreciable plastic deformation occurs before ductile fracture.

The distinction above has led to a model for the mechanical state of the fuel pellet during irradiation, with (i) an outer, colder region that is brittle and subject to cracking due to the thermal stresses set up by the temperature gradient; (ii) an intermediate region, which is strong and moderately ductile; and (iii) an inner, hot region that is easily subject to plastic deformation and is therefore not cracked (Olander, 1976).

UC is known to have good plasticity under load and at high temperature, which allows its fabrication by extrusion. The brittle-ductile transition occurs between 1273 and 1473 K, depending on deformation and grain size (Manara and Agarwal, 2020).

Fracture strength

The critical stress at which a specimen fails via fracture is called the fracture strength. Ductile materials have a fracture strength lower than the ultimate tensile strength (UTS) whereas, in brittle materials the fracture strength is equivalent to the UTS. The fracture strength and the yield stress are considered as independent temperature-dependent properties of the material. At temperatures where the fracture strength is lower than the yield stress, failure occurs in a brittle fashion, with no plastic flow prior to failure. At higher temperatures, yielding occurs at a lower stress than that required to induce fracture and, as a result, plastic deformation and ductile fracture are observed (Olander, 1976).

The correlation proposed in the MATPRO (2003) for the in-pile fracture strength of UO_2 as a function of temperature and fuel density reads

$$\sigma_f = 1.7 \times 10^8 \sqrt{1 - 2.62(1 - D)} \begin{cases} \exp[-1590/(8.314T)] & \text{for } 273 < T \leq 1000\text{K} \\ \exp(-1.590/8.314) & \text{for } T > 1000\text{K} \end{cases} \quad (33)$$

where σ_f is the fracture strength (Pa), and D (fraction of TD) is the fuel density. According to this model, a constant value for the fracture strength is used above 1000 K. This is based on the consideration that 1000 K is the lowest temperature at which plasticity has been observed in-pile (MATPRO, 2003). The transition from brittle to ductile material is accompanied by a discontinuity in the fracture strength and occurs at temperatures below the usual out-of-pile brittle-ductile transition temperature due to fission-induced plasticity (MATPRO, 2003). The value for the fracture strength above 1000 K in Eq. (33) is based upon theoretical considerations and out-of-pile data and applies to plastic UO_2 .

Fig. 6 shows the fracture strength of UO_2 as a function of temperature at varied fuel densities, according to Eq. (33). Experimental data are also shown. The significant scatter in the experimental data is evident.

Note that, when subjected to purely compressive loads, ceramics such as UO_2 are found to be far stronger than they are under tension. The compressive fracture strength of UO_2 is nearly an order of magnitude greater than the ultimate tensile strength obtained from bending tests (Olander, 1976). For application to fuel mechanical analysis, the tensile fracture strength is shown in Fig. 6 is the important one (Olander, 1976).

For UC, data are available for the transverse rupture strength and indicate a value of the order of 100 MPa for stoichiometric UC at room temperature (Rough and Chubb, 1960).

The fracture strength of UN has been reported as ~ 370 MPa at 1073 K (Feng et al., 2012). This value is about three times larger than that of UO_2 at the same temperature.

Creep

Creep is time-dependent plastic deformation. In general, the steady-state creep strain rate of oxide fuels depends on time, temperature, stress, fission rate, fuel grain size, density, and oxygen to metal ratio. The steady-state creep rate correlation for UO_2 given in MATPRO (2003) reads

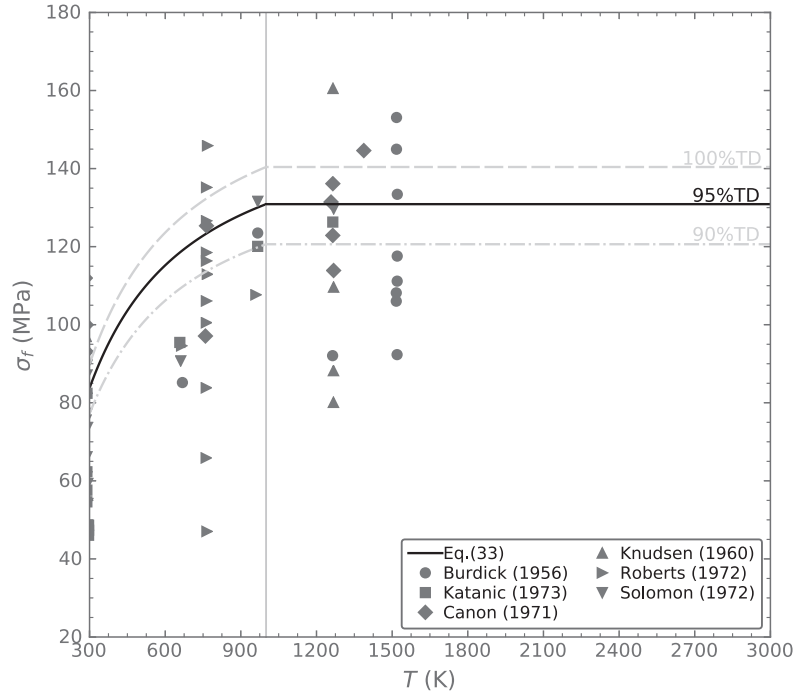


Fig. 6 The fracture strength of UO₂ at 95% TD with experimental data from Burdick and Parker (1956), Roberts and Ueda (1972), Knudsen et al. (1960), Canon et al. (1971), Solomon (1972), and Katanic-Popovic and Petrovic (1973).

$$\dot{\epsilon}_S = \frac{(A_1 + A_2 \dot{F})}{(A_3 + A_4 D) G^2} \sigma e^{(-Q_1/T)} + \frac{(A_5 + A_8 \dot{F})}{(A_6 + A_4 D)} \sigma^{4.5} e^{(-Q_2/T)} + A_7 \sigma \dot{F} e^{(-Q_3/T)} \quad (34)$$

where $\dot{\epsilon}$ (s⁻¹) is the creep strain rate, $\dot{F}$ (fissions/m³ s) the fission rate, σ (Pa) the stress, R (J/mol K) the universal gas constant, D (fraction of TD) the fuel density, G (μ m) the grain size, and Q_1 and Q_2 are the activation energies given as (in cal/mol)

$$Q_1 = \frac{9000.0}{\exp\left(\frac{-20}{\ln(O/M-2)} - 8\right) + 1} + 36294.4 \quad (35)$$

$$Q_2 = \frac{10000.0}{\exp\left(\frac{-20}{\ln(O/M-2)} - 8\right) + 1} + 56431.8 \quad (36)$$

where O/M is the oxygen-to-metal ratio.

The correlation from MATPRO (2003) for the steady-state creep rate of MOX fuel reads

$$\dot{\epsilon}_S = \frac{(B_1 + B_2 \dot{F})}{G^2} \sigma e^{(-Q_3/T + B_3[1-D] + B_4 \omega)} + (B_5 + B_6 \dot{F}) \sigma^{4.5} e^{(-Q_4/T + B_7[1-D] + B_4 \omega)} \quad (37)$$

where ω denotes the PuO₂ content (wt. fraction).

Note that, in MATPRO (2003), a transition stress $\sigma_t = 1.6547 \times 10^7 / G^{0.5714}$ is introduced such that for $\sigma > \sigma_t$, σ is replaced by σ_t in the first term on the right hand side of Eqs. (34) and (37). The values for the parameters in Eqs. (34) and (37) are given in Table 2. Fig. 7 shows experimental data from the literature for the steady-state creep rates of UO₂ and MOX compared to model predictions corresponding to Eqs. (34) and (37), respectively.

For both UC and UN, the steady-state creep rate at high temperatures can be expressed in the form (Hayes et al., 1990b; Manara and Agarwal, 2020)

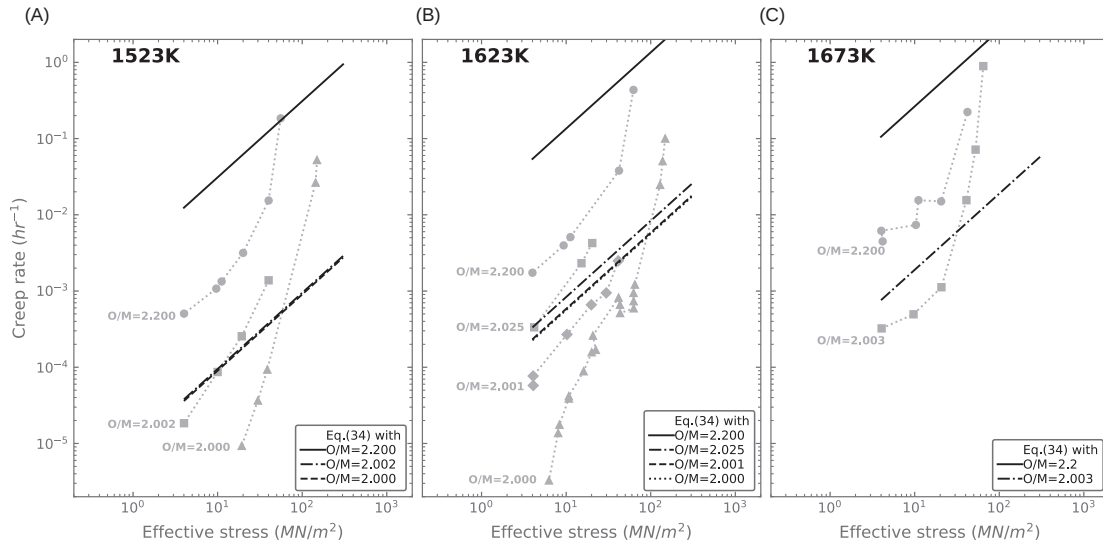
$$\dot{\epsilon}_S = A G^{-m} \sigma^n \exp\left(-\frac{Q}{RT}\right) \quad (38)$$

where A (1/s), m (–) and n (–) are constants, G (m) is the grain size, σ (MPa) the stress, Q (J/mol) the apparent activation energy, R (J/mol K) the ideal gas constant, and T (K) the temperature.

For UC, values $A = 4.36 \times 10^7$, $n = 2.4$, and $Q = 5.06 \times 10^5$ were reported in (Manara and Agarwal, 2020) with reference to Eq. (38). It was also noted that these values seem too uncertain to be used in fuel performance models (Manara and Agarwal, 2020).

Table 2 Empirical coefficients for Eqs. (34) and (37).

UO_2 (Eq. 34):									
A_1	0.3919	A_2	1.3100×10^{-19}	A_3	- 87.7	A_4	100.0	A_5	2.0391×10^{-25}
A_6	- 90.5	A_7	3.72264×10^{-35}	A_8	0.0	Q_3	2.6167×10^3		
$(U,Pu)O_2$ (Eq. 37):									
B_1	0.1007	B_2	7.57×10^{-20}	B_3	33.3	B_4	3.56	B_5	6.469×10^{-25}
B_6	0.0	B_7	10.3	Q_3	55.354×103	Q_4	70.451×10^3		

**Fig. 7** The steady-state creep rates of UO_2 at various temperatures with experimental data from [Burton and Reynolds \(1973\)](#).

The C/U ratio, porosity, grain size, and impurities can independently change the creep rate by several orders of magnitude. Stoichiometric and hypostoichiometric carbides ($C/U < 1.01$) are characterized by a higher creep rate than hyper-stoichiometric carbides due to UC_{2-x} precipitation ([Manara and Agarwal, 2020](#)). Finally, the stress exponent for single crystals is much higher ($n > 5$) than for polycrystalline UC ([Manara and Agarwal, 2020](#)).

For UN, values $A = 2.054 \times 10^{-3}$, $n = 4.5$, and $Q/R = 3.94 \times 10^4$ were reported in ([Hayes et al., 1990b](#)) with reference to Eq. (38). Also, the correlation in ([Hayes et al., 1990b](#)) is independent of grain size (i.e., $m = 0$) and is valid in the ranges $1770 \text{ K} \leq T \leq 2083 \text{ K}$ and $20 \text{ MPa} \leq \sigma \leq 34 \text{ MPa}$. Moreover, Eq. (38) with the aforementioned parameter values is valid for the creep of UN at 100% TD. Finally, the following porosity correction factor can be applied to the UN creep rate ([Hayes et al., 1990b](#))

$$f(p) = \frac{0.987}{(1-p)^{27.6}} \exp(-8.65p) \quad (39)$$

where p (vol. fraction) is the porosity.

Summary

In this article, an overview of the thermal and mechanical properties of the ceramic nuclear fuels UO_2 , MOX, UC and UN was given, with a focus on fresh fuel. In particular, the thermal conductivity, melting point, specific heat capacity, coefficient of thermal expansion, elastic constants, plasticity, fracture strength and creep rate were discussed. Examples of correlations from the literature for the evaluation of each property as a function of fuel state variables such as temperature, density and composition were given. Such correlations are used in fuel thermo-mechanics analysis. Furthermore, a collection of experimental data for fuel properties from various sources was included. It is worth noting that, for several properties, significant uncertainties are associated with the available experimental data and, consequently, with the correlations used in the fuel thermo-mechanics analysis. Uncertainties for specific properties were pointed out in the discussion. More details on ceramic nuclear fuel properties and the associated uncertainties can be found in the references provided.

Acknowledgments

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Zr-Alloyed Metallic Fuel Properties

Stephen Novascone^a, Christopher Matthews^b, Albert Casagrande^a, Aysenur Toptan^a, and Adam Zabriskie^a, ^aComputational Mechanics and Materials, Idaho National Laboratory, Idaho Falls, ID, United States; and ^bMaterials Science and Technology, Los Alamos National Laboratory, Los Alamos, NM, United States

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Introduction

We start this high-level study of zirconium-alloyed metallic fuel by reviewing the work accomplished at the experimental breeder reactor (EBR-II) facility (Westfall, 2004; Walters, 1999; Koch, 2008; Sackett, 1997). EBR-II operated for approximately 30 years (1964–94) and achieved several important objectives related to metallic fuels. The first objective of EBR-II was to demonstrate that a closed fuel cycle was possible, meaning that the reactor would generate power while breeding fissionable plutonium from U-238. Additionally, the fuel would be reprocessed (separating fissionable material from fission products), re-fabricated (re-melted and cast into fuel slugs), and re-inserted into the reactor for additional power production. All of these processes were performed on-site, thus eliminating the need for transportation of radioactive materials off-site or requiring fresh fuel to be delivered. Due to the inability to remove all radioactive fission products and transuranics, the reprocessed fuel form was extremely radioactive. While this provided natural proliferation resistance due to handling difficulties, it also forced all reprocessing to be performed remotely. In 1969, ~35,000 fuel pins had been remotely recycled back into EBR-II.

Besides providing a working example of a closed nuclear fuel system, the collection of irradiated pins and associated post-irradiation examinations throughout the EBR-II irradiation testing mission provided an unmatched understanding of fuel behavior during irradiation. To set the stage for discussing material properties and the enlightening information that came out of EBR-II in the *Metallic Fuels Handbook* (MFH, 1988), a description of the fuel pin is required. A sketch of the fuel pin is presented in Fig. 1, which shows the solid cylindrical fuel encased within a cylindrical tube, referred to as cladding. The fuel is uranium-zirconium alloy (U-Zr) or uranium-plutonium-zirconium alloy (U-Pu-Zr). The cladding material is stainless steel; many were tested in the United States in EBR-II and HT9 was eventually selected in the 1990s as the preferred cladding alloy for that time.

There's a large gap between the fuel and cladding to accommodate the large swelling characteristic of metallic fuel. Liquid sodium, referred to as the sodium bond, is added into the fuel pin to fill the fuel/cladding gap and enhance heat conduction. The level of the liquid sodium is just above the top of the as-fabricated fuel column. The space inside the cladding, above the liquid sodium is called the plenum. The outside of the fuel pin is cooled by flowing liquid sodium.

This description of the basic metal fuel design provides context for this article, which describes the properties of metal fuel and how those properties contribute to fuel performance.

The aim for this article is to visually present the material models used by a fuel performance analyst using graphs to help the reader develop a basic and intuitive understanding of fuel behavior in a reactor. In simple terms, the fuel in a reactor heats up during operation, swells due to thermal expansion and the formation of fission gas, contacts the cladding, then releases the fission gas as the pores that contain the fission gas grow and interconnect forming a path to the fuel surface. The fission gas is (by design) contained inside the cladding, which isolates the fission gas and its damaging effects from the coolant and other reactor components. The material models required to describe and quantify these phenomena are thermal conductivity, heat capacity, thermal expansion, mechanical elasticity, creep, swelling, and constituent diffusion. Each of these properties is described in the following sections with graphs to show material property changes in response to changing temperature, composition, and porosity. Because it's important to view these material property models in relation to reality, material property measurements are shown along with the models. In this article, we present models and measurements for uranium-zirconium (U-Zr or binary) and uranium-plutonium-zirconium

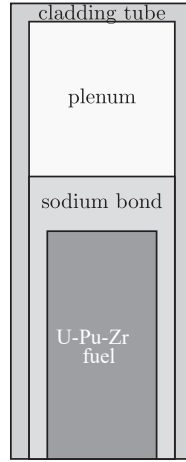


Fig. 1 Simplified illustration of EBR-II metallic fuel pin (*not to scale*).

(U-Pu-Zr or ternary) fuel. Henceforth, the term metallic fuel refers to these fuel types. When a number is presented in reference to fuel alloy, such as U-10Zr, this means the fuel is 90 wt% U and 10 wt% Zr.

In this article, we provide an overview of the main properties of metallic fuels. In particular, thermal properties are dealt with in section **“Thermal properties,”** and mechanical properties are summarized in section **“Mechanical properties.”**

To make the work presented here clear, a brief description of porosity is required. As the fuel resides in a fission reactor, fissionable atoms react with neutrons to release energy, fission products, and additional neutrons. Fission products at fuel temperatures (~ 300 – 1500 kelvin (K) for illustration purposes in this article) can be solid, liquid, or gaseous, and the gaseous fission-product atoms (inert gases, in particular) will segregate and combine with other fission gases to form fission gas bubbles. These gas bubbles (porosity) form throughout the fuel. Herein, porosity is defined as the fraction of the small fission gas-filled volume over total fuel volume. The process of how materials react to irradiation is complicated, and the subject of ongoing research. For this introduction to metallic fuel properties, it's sufficient to understand that porosity forms, grows, and coalesces as fission occurs generating additional fission gases, and is an indication of how much fissioning has occurred.

Thermal properties

Thermal conductivity

The energy released via fission heats the fuel. The heat conduction equation describes well how this heat traverses through the fuel. An essential component of this equation is thermal conductivity, a material property that describes how much heat (thermal energy) can pass through a material in a given time. Its units are power per length-temperature. A model for thermal conductivity of U, U-Zr, and U-Pu-Zr alloys is provided in the *Metallic Fuels Handbook* (MFH, 1988), which was developed from many sets of measurements compiled in Kittel et al. (1977). Thus, the thermal conductivity is correlated as:

$$k_0 = A + BT + CT^2, \quad (1)$$

with

$$\begin{aligned} A &= 17.5 \left[\frac{(1 - 2.23W_{Zr})}{(1 + 1.61W_{Zr})} - 2.62W_{Pu} \right] \\ B &= 1.54 \times 10^{-2} \left[\frac{(1 + 0.061W_{Zr})}{(1 + 1.61W_z)} + 0.920W_{Pu} \right] \\ C &= 9.38 \times 10^{-6} (1 - 2.70W_{Pu}) \end{aligned} \quad (2)$$

where k_0 is the thermal conductivity of non-porous material (W/m/K), W_{Zr} is the zirconium weight fraction (wt/ W_{Pu}) is the plutonium weight fraction (wt/), and T is the temperature (K).

A correction for irradiation effects through the introduction and growth of porosity is calculated as follows: In case of closed porosity formed by isolated spherical pores, the Maxwell-Eucken formula can be used

$$k = k_0 \left(\frac{1 - p}{1 + \beta p} \right) \quad (3)$$

where k_0 is the thermal conductivity of non-porous material (W/m/K), p is the fractional porosity (–) (note that throughout this article (–) is used to denote a unitless quantity, such as porosity or strain), and β is the porosity constant that is typically taken as 2.5

for conservatism as recommended by Billone et al. (1968). The porosity p is calculated through additional material models for volumetric swelling due to fission gas production (see section “Swelling”).

Fig. 2 shows this model to demonstrate how fuel thermal conductivity changes as temperature is increased, for different fuel alloy compositions, and values of porosity. Fig. 2 shows that as the temperature within the fuel increases, the thermal conductivity increases. In other words, as the temperature increases, energy can flow more readily through the fuel. On the other hand, the fuel that includes plutonium (e.g., an alloy with uranium, plutonium, and zirconium; ternary fuel) has a lower value of thermal conductivity than fuel with only uranium and zirconium (binary fuel). This means that for the same energy generated within the fuel, a higher fuel temperature in ternary fuels is expected. Just as importantly, Fig. 2 shows that thermal conductivity decreases significantly for larger values of porosity, which is expected due to increased thermal resistance of pores. It's important to point out that temperature impacts all aspects of fuel performance, and emphasize that thermal conductivity directly influences fuel temperature. For context, measurements of thermal conductivity documented in Bauer and Holland (2017) are plotted along with the model in Fig. 2. These measurements were taken while the fuel was fissioning. The burnup (a measure of duration and intensity of fissioning) was 0.8 fissions per initial metal atom, or at.%, which is low.

Specific heat capacity

Specific heat capacity is defined as the amount of heat (or energy) required for a given mass to produce a unit change in its temperature. The units of heat capacity are energy per mass-temperature. Heat capacity is essential when describing how fast a material can change temperature when energy is deposited in or removed from the material. As such, it's especially important when energy is rapidly deposited in removed from the fuel (e.g., quick rises in power, or loss-of-flow events).

The correlation for specific heat capacity from Savage (1968) is split into the low temperature $\alpha + \delta$ region, and the high temperature γ region:

$$\begin{aligned} c_p^{\alpha+\delta}(T) &= 6.36 + 0.00636T, \quad \text{for } 25^\circ\text{C} < T < 600^\circ\text{C} \\ c_p^\gamma(T) &= 3.79 + 0.00623T, \quad \text{for } T > 650^\circ\text{C} \end{aligned} \quad (4)$$

where c_p is the specific heat capacity (cal/mol/°C) and T is temperature (°C).

A simple linear interpolation is employed in the $\beta + \gamma$ region:

$$c_p^{\beta+\gamma}(T) = \left(\frac{c_p^\gamma(T_{\text{high}}) - c_p^{\alpha+\delta}(T_{\text{low}})}{T_{\text{high}} - T_{\text{low}}} \right) (T - T_{\text{low}}) + c_p^{\alpha+\delta}(T_{\text{low}}), \quad (5)$$

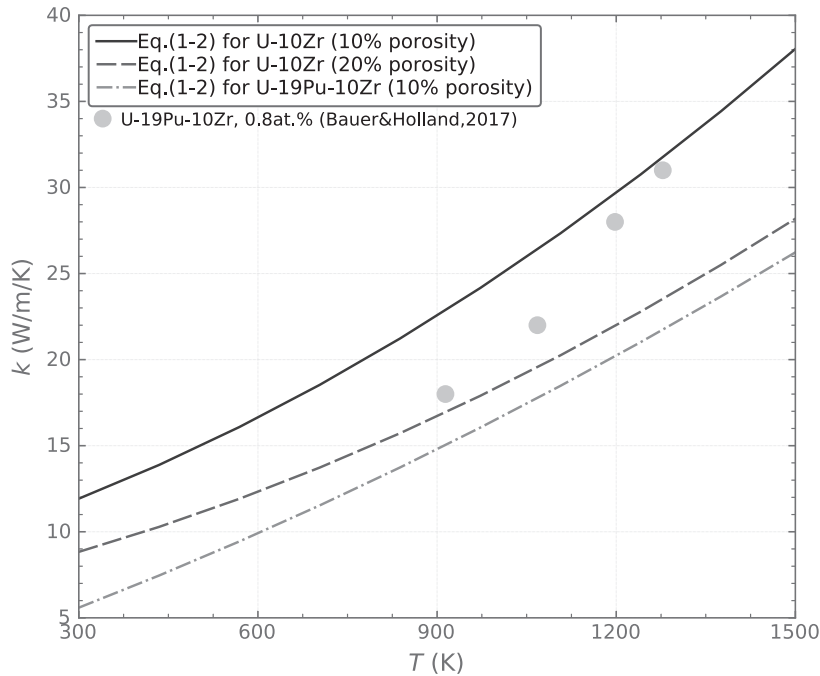


Fig. 2 The thermal conductivity of zirconium-alloyed metallic fuel at varied compositions and porosity fractions (experimental data from Bauer and Holland (2017)). Note that binary fuel (U-Zr alloys) thermally is more conductive than ternary fuel (U-Pu-Zr) and pores within the fuel act to decrease thermal conductivity.

where T_{low} and T_{high} are the transition temperatures 600 °C and 650 °C, respectively. The designations $\alpha + \delta$, γ , and $\beta + \gamma$ denote material phase. Similar to the familiar solid/liquid/gas phases of matter, solid metallic alloys can go through material phase changes (i.e., transformations at the atomic level in the arrangement of the atoms which make up the alloy) in the presence of heat and fission. The result is significant changes to the material properties. Publications such as [Savage \(1968\)](#) have attempted to describe the effect of this phase change in material models for heat capacity, and that is reproduced here in [Fig. 3](#). It's important to note that all material properties are affected by phase change.

To limit the complexity of this article, and due to the fact that material property measurements during phase change are scarce, the effects of material phase are only shown for heat capacity. The measurements ([Savage, 1968](#)) used to develop this model are also presented in [Fig. 3](#).

Mechanical properties

Thermal expansion

The thermal response of the fuel affects the mechanical response via thermal expansion. Thermal expansion is simply uniform volume change due to heating or cooling. A relevant example is heating a bearing prior to placing it on a shaft in a mechanical system as part of the shrink fitting process. The bearing expands during heating, placed on the shaft, then shrinks during cooling, contacting the shaft to provide a strong mechanical bond. As the temperature in the fuel increases due to fission, the fuel expands in all directions. We show a few models for thermal expansion in [Fig. 4](#). The simplest model is a constant linear thermal expansion coefficient (α) multiplied by the difference between the current temperature and reference temperature ($\alpha\Delta T$). The reference temperature is where measurements begin. An average value of the fuel thermal expansion coefficient in the temperature range of 293 K (reference temperature) to 1200 K from [Saller et al. \(1956\)](#) is $\alpha = 1.88 \times 10^{-5} \text{K}^{-1}$. Additionally, models from [Touloukian et al. \(1975\)](#) and [Basak et al. \(2009\)](#) are shown in [Fig. 4](#) along with measurements from [Saller et al. \(1956\)](#). The change in trend between 923 K and 998 K, is due to a change in phase. Also note that thermal expansion is a volumetric phenomenon, but is usually expressed linearly, as in change in length divided by original length.

Elastic correlations

Elasticity, or modulus of elasticity, is the property of the fuel that quantifies how resistant it is to an applied load. The units are force per area. Due to the redistribution of zirconium in the fuel and the need to model both U-Zr and U-Pu-Zr fuel, correlations included in the elastic correlation for the fuel must be applicable to any concentration of plutonium or zirconium. Unfortunately, the

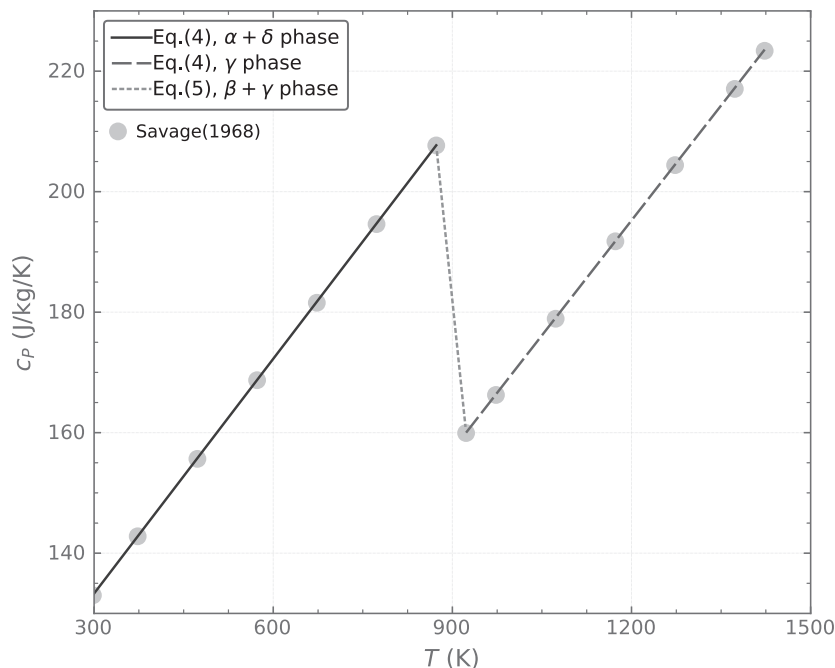


Fig. 3 The specific heat capacity of metallic fuel (*experimental data from Savage (1968)*). The fuel experiences a thermal gradient with high temperatures in the center and lower temperature on the surface that cause different material phases going from the center of the fuel to the surface. Three material phases are shown here; the $\alpha + \delta$ phase for the coolest section (*near the edge*), γ in the hottest section (*fuel center*), and $\beta + \gamma$ for the section in-between. See [MFH \(1988\)](#) for more information about material phases.

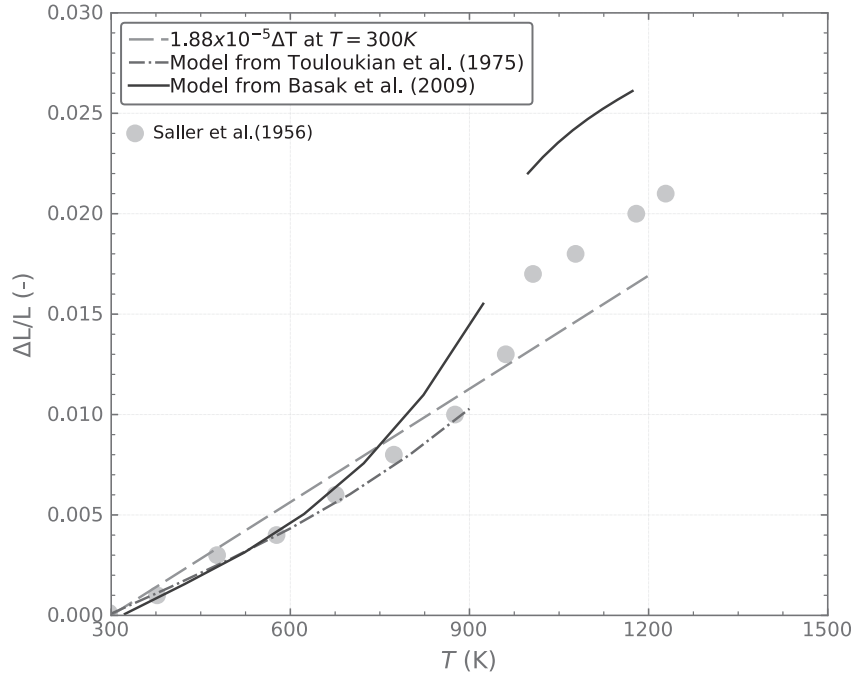


Fig. 4 The fuel thermal expansion for a constant thermal expansion coefficient and aforementioned models (Touloukian et al., 1975; Basak et al., 2009) (experimental data from Saller et al. (1956)).

available models for the elastic properties of U-Pu-Zr alloys are generally incomplete. Regardless, some correlations can be made given the sparse data, as described in MFH (1988).

The correlation for Young's modulus can be derived using data from U-11Pu-6.3Zr cast fuel pins as:

$$E = E_u(1 - \beta_E p) \left(\frac{1 + 0.17W_{Zr}}{1 + 1.34W_{Zr}} - W_{Pu} \right) \left(1 - 1.06 \left[\frac{T - 588}{T_{mu}} \right] \right) \quad (6)$$

where $E_u = 1.6 \times 10^5$ MPa is the Young's modulus for pure uranium at the reference temperature of 588 K, p is the fractional porosity (-), W_{Zr} is the zirconium weight fraction (wt/), W_{Pu} is the plutonium weight fraction (wt/), T is the temperature (K), and $T_{mu} = 1405$ K is the melting temperature of pure uranium. β_E is the Young's modulus degradation factor due to porosity and is typically 1.2.

The models in Fig. 5 display the value of elasticity as temperature increases for binary (U-10Zr) and ternary (U-19Pu-10Zr) fuels and the effect of porosity on binary fuel. The amount of zirconium and plutonium is shown in weight percent. Fig. 5 shows that the elastic modulus decreases as the temperature increases; in other words, the fuel becomes softer as the temperature gets higher. Furthermore, the presence of plutonium and/or increases in porosity result in softening the fuel. Measurements for pure uranium from Holden (1957) and ternary fuel from Harbur et al. (1970) are included in Fig. 5.

Another material property related to the modulus of elasticity is Poisson's ratio, which is a measure of a materials dimensional response to an applied load. Rather than a direct resistance to the applied load, like the elastic modulus, Poisson's ratio reflects how a material expands or contracts when a load is applied. As an example, when a tensile load is applied to a long circular bar in the direction of the bar's length, the diameter of the rod changes. Expressed formally, Poisson's ratio is the ratio of lateral strain to a longitudinal strain when subject to a longitudinal force.

The correlation for Poisson's ratio can be derived from the U-Zr correlation (MFH, 1988) as:

$$\nu = \nu_u(1 - \beta_p p) \left(\frac{1 + 3.4W_{Zr}}{1 + 1.9W_{Zr}} \right) \left(1 + 1.2 \left[\frac{T - 588}{T_{mu}} \right] \right) \quad (7)$$

where $\nu_u = 0.24$ is the Poisson's ratio of pure uranium at the reference temperature of 588 K, p is the fractional porosity (volume fraction, (-)), W_{Zr} is the zirconium (weight fraction, wt/), T is the temperature (K), and T_{mu} is the melting temperature of pure uranium (1405 K). β_p is the Poisson's ratio degradation factor due to porosity and is typically 0.8.

Fig. 6 shows Poisson's ratio as temperature increases for binary fuel (U-Zr) for two values of porosity. This graph indicates that as temperature increases, metallic fuel will strain more. Additionally, as porosity increases, the fuel strains more with the applied load. Not many measurements exist for this material property. As such, only the model is shown in Fig. 6.

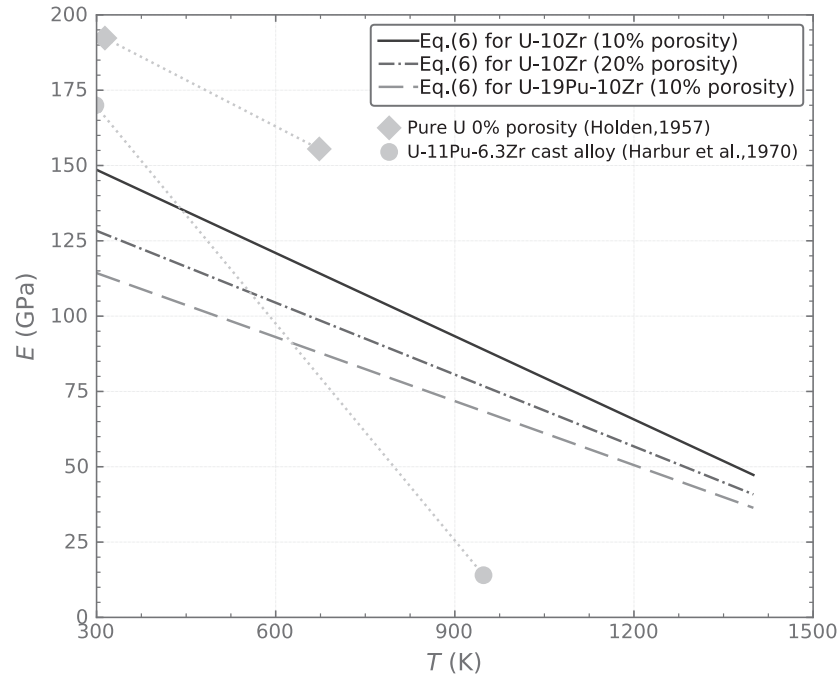


Fig. 5 The elastic modulus at different alloy compositions, porosity fractions, and temperatures (*experimental data from Holden (1957) and Harbur et al. (1970)*). An observation from this plot is that binary fuel (U-Zr) is stiffer (more resistant to mechanical load) than ternary fuel (U-Pu-Zr). Also note that pores within the fuel act to decrease elastic modulus, i.e. soften the fuel.

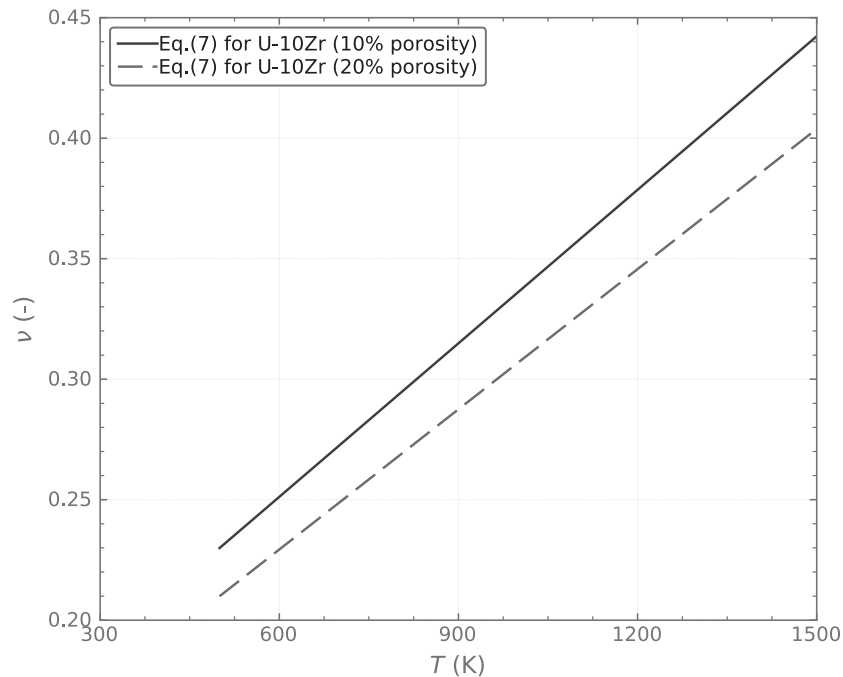


Fig. 6 The Poisson's ratio of U-10Zr at varied porosity fractions.

Creep

In the simplest terms, elastic modulus, Poisson's ratio, and thermal expansion are used together to calculate the mechanical response or movement/deformation of the fuel. However, this is not sufficient to describe how fuel deforms while in the reactor. Another mechanism, called creep, must be included for a more representative description. Creep is defined as slowly evolving, irrecoverable strain due to applied load. Creep strain accelerates at high temperatures and due to stress induced by the presence of fission products.

A model for the combined thermal and irradiation creep of U-Pu-Zr is available with the creep rate modeled as a function of time, fuel porosity, effective stress, and fission rate (MFH, 1988) and is given as

$$\dot{\epsilon} = A_1(1 + 7.9p + 470p^2) \exp\left(\frac{-Q_1}{RT}\right) \sigma + A_2(1 - p^{0.67})^{-4.5} \exp\left(\frac{-Q_2}{RT}\right) \sigma^{4.5} + A_3 \dot{F} \sigma \quad (8)$$

where $\dot{\epsilon}$ is the creep rate (1/s), σ is the effective (Mises) stress (MPa), T is the temperature (K), p is the porosity fraction (—), $\dot{F}$ is the volumetric fission rate (fissions/cm²/s), Q_i are the activation energies (cal/mol), R is the universal gas constant (1.987/cal/mol/K), and A_{1-3} are empirical constants given as $A_1 = 5 \times 10^3$, $A_2 = 6$, and $A_3 = 7.7 \times 10^{-23}$.

The terms on the right-hand side of Eq. 8 respectively represent diffusional creep, power-law creep, and irradiation creep as it depends upon the fission rate. The activation energies for the thermal creep terms (Q_1 and Q_2) are given as $Q_1 = Q_2 = 52\,000$ cal/mol.

These U and Pu alloys have a phase change temperature of 923.15 K. Above this temperature, the creep rate equation changes to

$$\dot{\epsilon} = A_4(1 - p^{0.67})^{-3} \exp\left(\frac{-Q_3}{RT}\right) \sigma^3 + A_3 \dot{F} \sigma \quad (9)$$

where $A_4 = 8.0 \times 10^{-2}$ and $Q_3 = 28\,500$ cal/mol.

Fig. 7 shows a creep strain rate model from MFH (1988) that is a function of temperature and porosity at constant stress (load) of $\sigma = 9.8$ MPa. An essential point about creep is to note how it accelerates *nonlinearly* as the temperature increases and even more so as porosity increases (due to fission). This has a profound effect on the mechanical response of metallic fuel. Measurements from Saller et al. (1953) for binary fuel and from Argonne (Foote et al., 1965) for ternary fuel are also shown in Fig. 7, both in unirradiated conditions. Due to the nonlinear nature of creep, Fig. 7 uses a log scale on the y-axis, which clearly shows the significant differences between the models for different values of porosity.

Swelling

As the fuel is fissioning, fission gases and voids form within the fuel. The gas diffuses through the crystalline atomic structure and collects in the voids, referred to as pores or porosity. As the temperature increases and fission gases commence to form, these now filled voids grow and rapid fuel swelling occurs. The volume change in the fuel is similar to thermal expansion except the strain is driven by the increased pressure in the pores. The pores eventually coalesce into a smaller number of larger pores, and when the porosity reaches a value of about 20%, the gas-filled spaces form an interconnected path through the fuel to the sodium bond and plenum. Further swelling in the fuel decreases significantly because the gas escapes the fuel.

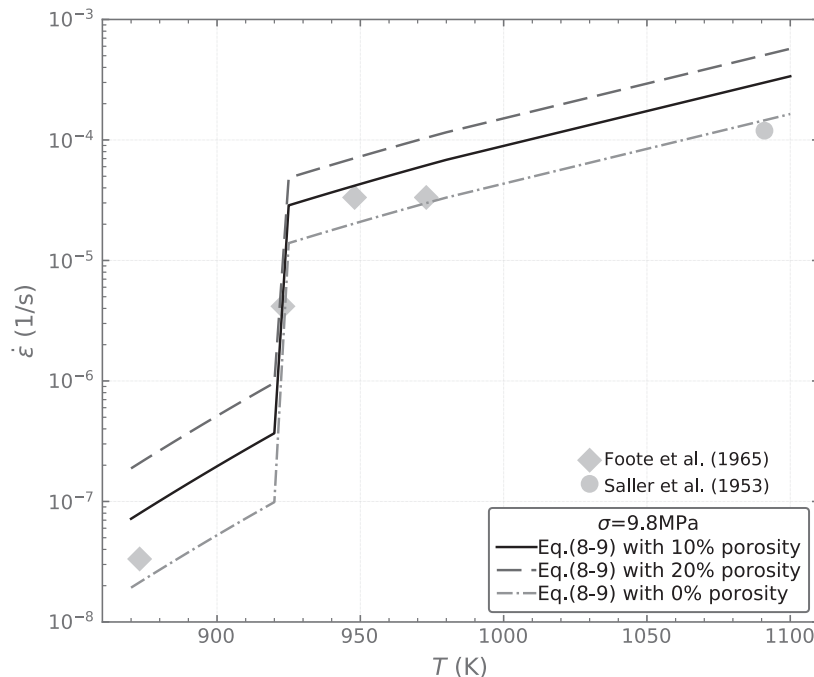


Fig. 7 The steady-state creep rates as a function of temperature and porosity (experimental data from Saller et al. (1953) and Foote et al. (1965)).

The derivation for the mechanistic fuel swelling model for U-Pu-Zr systems used here was originally presented in Olander (1976). The model is based on the mechanical force balance for an equilibrium fission gas bubble (note that bubble and pore are used interchangeably here) and the equation of state:

$$P = \frac{2\gamma}{r_b} - \sigma_h + \sigma_{cr} \quad (10)$$

where P is the pressure of the fission gas in a bubble (Pa), γ is the surface tension of the fuel (N/m), r_b is the fission gas bubble size (m), σ_h is the hydrostatic stress in the fuel (Pa), and σ_{cr} is the creep strength stress of the fuel (Pa). Assuming that the gas in the bubble is governed by the ideal gas law and rearranging to solve for the volume of the bubble leads to an expression for the volumetric strain due to swelling:

$$\left(\frac{\Delta V}{V_0}\right)_g = \left(\frac{3}{4\pi}\right)^{1/2} \frac{[(kT/2\gamma)Y_{Xe}\dot{F}t]^{3/2}}{N^{1/2}} \quad (11)$$

where k is the Boltzmann constant (1.38×10^{-23} m²/kg/s²/K), T is the temperature (K), Y_{Xe} is the gaseous fission product yield (atoms/fission), $\dot{F}$ is the fission rate density, (fissions/m³/s), t is time (seconds), N is the number density of bubbles (number of bubbles), V is the current volume (m³), and V_0 is the original volume (m³).

Once the gaseous fuel swelling is calculated, it is used to calculate the porosity:

$$p = \frac{\left(\frac{\Delta V}{V_0}\right)_g}{1 + \left(\frac{\Delta V}{V_0}\right)_g} \quad (12)$$

Models that describe the swelling and formation of porosity are shown in Fig. 8 from Olander (1976). The plots show nonlinear porosity formation and swelling as temperature increases for two fission rates. Porosity formation and swelling almost overlap, but there are slight differences. Measurements of fission gas-induced swelling for these binary and ternary fuels as a function of burnup and temperature aren't available, so measurements are not included in Fig. 8. Swelling measurements for other uranium alloy materials can be found in McDonnell and Sturcken (1975).

Constituent redistribution

Another phenomenon that is important to understanding fuel behavior is the composition of the fuel. As the fuel fissions and heats up, the uranium and zirconium migrate from a uniform distribution to a distribution with a higher concentration of zirconium and

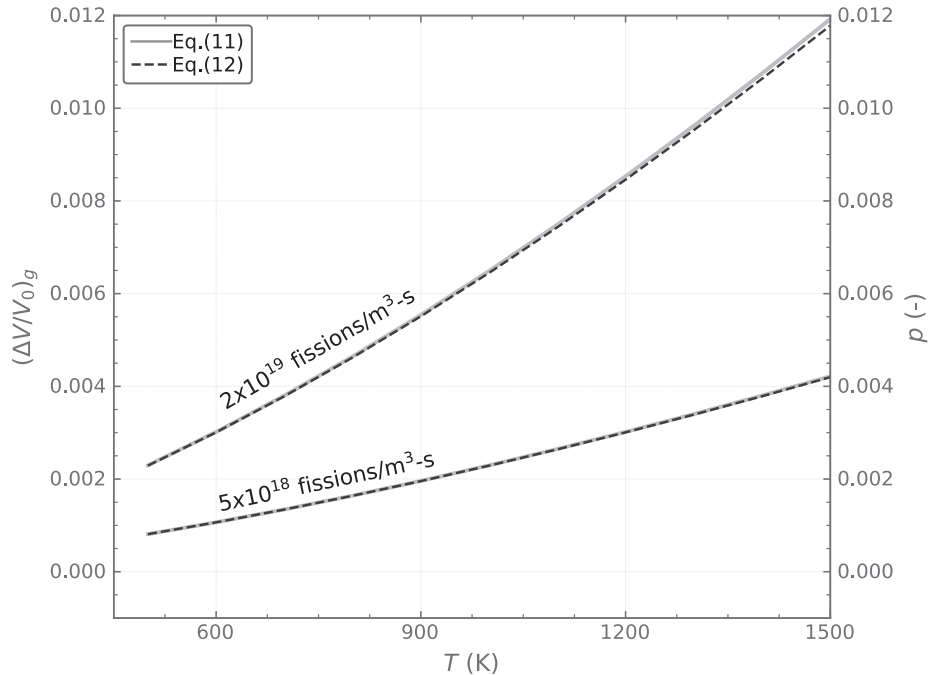


Fig. 8 Fuel swelling as a function of temperature and fission rate.

a correspondingly lower concentration of uranium in the fuel center via a process called diffusion. This is important because the concentration of the alloy constituents influences other material properties and will affect the spatial intensity of fissions as the uranium concentration changes. The material property that governs this process is called the diffusion coefficient. It is analogous to thermal conductivity. The units are in length squared per time. The Metallic Fuels Handbook shows measurements of diffusion coefficients for uranium and zirconium in binary (U-Zr) fuel from [Adda et al. \(1957\)](#) and [Müller \(1959\)](#) at various temperatures. These measurements are shown in [Fig. 9](#). For more detail in how these diffusion coefficients are modeled, see [Galloway et al. \(2015\)](#).

Summary

From observations of [Figs. 2–8](#), one can understand how the fuel properties and characteristics change as temperature and fission increase. The energy released by fission heats the fuel, and the fuel's thermal conductivity and heat capacity define how heat conducts through the fuel. As the temperature of the fuel increases, it expands based on thermal expansion, elastic modulus, Poisson's ratio, creep, and fission-gas-driven swelling. Porosity generated by fission eventually interconnects, and fission gas is released from the fuel, which reduces the driving force for swelling and thereby slows swelling. Although more detail is necessary for thoroughly describing how the fuel behaves, especially during transients, early in fuel life, and after long exposure to irradiation, the compilation of the property models described here provides insight on how changes in the metallic fuel material lead to phenomena that impact cladding integrity and performance. A useful review of binary (U-Zr) and ternary (U-Pu-Zr) properties can be found in [Janney and Hayes \(2018\)](#) and [Janney et al. \(2019\)](#), respectively.

The fuel's cladding is a vital component of the metallic fuel system. Cladding is the structural barrier that retains fission products and is the thermal medium to the sodium coolant. The fuel properties discussed here significantly affect the cladding; this relationship cannot be over-emphasized. Due to the large fuel swelling and concerns about the fuel contacting and stretching the cladding, many experiments were conducted during EBR-II to determine the fuel rod features necessary such that fission-gas-driven swelling would essentially stop before contacting the cladding. It was determined that the ratio of fuel cross-sectional area to total cross-sectional area available within the cladding should be 75%. This so-called smear density of a fuel rod is illustrated in [Fig. 10](#), which shows a schematic of a fuel rod cross section. This ratio was determined such that by the time the fuel swelling slows down (due to porosity interconnection and fission gas release), the fuel softly contacts the cladding. The force applied to the cladding by the fuel depends very much on all the material properties presented in this article. While there are other features beyond the scope of this article that are also important to the structural integrity of the cladding, such as fuel cladding chemical interaction, the properties and description presented here make the connection and emphasize the relationship between fuel properties, cladding mechanical response, and containment of fission products and, specifically, fission gas within the cladding—a critical design feature.

The information presented here introduces how zirconium-alloyed metallic fuel responds to elevated temperature and fission. Furthermore, it forms the basis for progressive understanding as more detailed sources are explored.

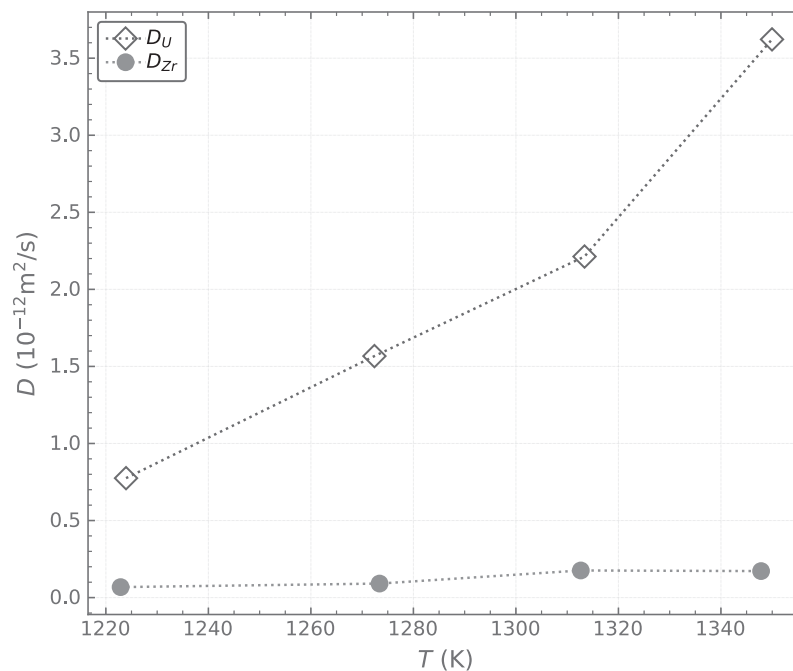


Fig. 9 Uranium and zirconium diffusion coefficients in binary (U-Zr) metallic fuel alloy (experimental data from [MFH \(1988\)](#), [Adda et al. \(1957\)](#), and [Müller \(1959\)](#)).

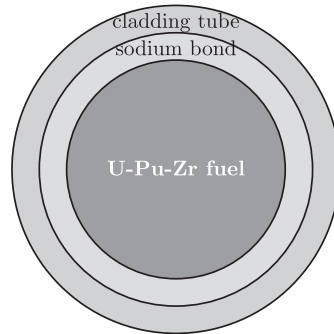


Fig. 10 Simplified top-down illustration of the concentric EBR-II metallic fuel pin, smear density (*not to scale*).

Table 1 Material property model, corresponding reference, and nominal value.

Material property model	Reference	Nominal value
Thermal conductivity	MFH (1988) and Bauer and Holland (2017)	20 W/m/K
Specific heat capacity	Savage (1968)	100 J/kg/K
Thermal expansion coefficient	Touloukian et al. (1975) and Basak et al. (2009)	1.88×10^{-5} m/m/K
Elastic modulus	MFH (1988)	80 GPa
Poisson's ratio	MFH (1988)	0.30
Diffusion coefficient	Adda et al. (1957) and Müller (1959)	2(U), 0.1(Zr) 10×10^{-12} m ² /s

As a quick reference for typical values of the Zr-based metallic fuel properties presented here, see Table 1.

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Molten Salt Fuels: Properties, Purification, and Corrosion Control^{*,**}

Ruchi Gakhar, William C. Phillips, Guoping Cao, Tae-Sic Yoo, Michael E. Woods, Guy L. Fredrickson, and Toni Karlsson, Idaho National Laboratory, Originating Organization C420, Idaho Falls, ID, United States

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Abbreviations

DSC Differential Scanning Calorimetry
GC-MS Gas Chromatography Mass Spectroscopy
ICP-MS Inductively Coupled Plasma Mass Spectroscopy
LWR Light Water Reactors
MSR Molten Salt Reactors
MSRE Molten Salt Reactor Experiment
MWe Mega Watt Electrical
MWt Mega Watt Thermal
ORNL Oak Ridge National Laboratory
RedOx Reduction Oxidation
TGA Thermogravimetric Analysis

Introduction

Molten salt reactors (MSRs) are a nuclear reactor concept that use molten salts as fuel and coolant. MSRs were first studied in the U.S. shortly after World War II with intended application for aircraft propulsion and civilian power production. These efforts began at Oak Ridge National Laboratory (ORNL) with the Aircraft Nuclear Propulsion Program in 1949. The nuclear propulsion programs

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were terminated in 1961 and interests in MSRs for civilian power production continued into the early 1970s. Worldwide, only two MSRs have operated: the Aircraft Reactor Experiment (2.5 MWt) from November 3–12, 1954, intended to lead to a design for aircraft propulsion; and the Molten Salt Reactor Experiment (MSRE) (8 MWt) between June 1, 1965 and December 12, 1969, which demonstrated technologies intended for civilian power production. A detailed review and overview of these ORNL experiments can be found in literature (Fredrickson et al., 2018).

A common feature of MSRs for steam generation of electrical power is the use of molten salts as heat transfer media within two separate molten salt loops: primary and secondary. The primary salt loop extracts heat from the reactor's core, and the secondary salt loop extracts heat from the primary salt loop. Steam is produced when a water loop extracts heat from the secondary salt loop. Several different designs of MSRs have been proposed and are most easily differentiated by the type of fuel. The designs for which fissile inventory is dissolved in molten salt are referred to as salt-fueled MSRs. For example, enriched uranium is added directly to the salt as uranium fluoride or uranium chloride. As the salt flows through the reactor's core, the heat of fission is generated within the salt itself. The designs for which the fissile inventory is kept separate from the salt are referred to as molten salt-cooled reactors. For example, enriched uranium fuel is contained in cladding and stationary within the reactor's core. As the salt flows through the reactor's core the heat of fission is generated within the uranium fuel and transferred to the salt coolant.

More complex MSR designs include breeder reactors that use a $^{232}\text{Th}/^{233}\text{U}$ fuel cycle (Fredrickson et al., 2018; Heuer et al., 2014) and are capable of producing excess fuel materials, and burner reactors whose function is to consume nuclear materials recovered from spent nuclear fuels or nuclear material produced for nuclear weapons (Gat et al., 1991). MSRs have several promising, yet unrealized features that make them attractive for nuclear reactor applications. These include high coolant temperatures, low pressure cooling systems, enhanced intrinsic safety features, and greater thermal-to-mechanical efficiencies than, for example, Light Water Reactors (LWRs). Fluoride, and to a lesser extent chloride, salt systems have received the most attention (Gen IV Forum, 2018; Serp et al., 2014). Table 1 lists a variety of fuel and coolant salt systems historically proposed for different MSR applications.

Nuclear material safeguards and monitoring are important considerations as are long-term waste treatment and disposal (Bauman et al., 1977). Radioactive wastes from MSRs will require treatment and disposal technologies that are different than those developed for LWR fuel waste management. There are several technical requirements placed on molten salt fuels and coolants. The physical-chemical properties are important to basic engineering design aspects, including heat capacity, thermal conductivity, viscosity, vapor pressure, density, and radiation performance.

Many of the technical challenges of MSR design are related to the chemical and electrochemical properties of the molten salts, particularly regarding the chemical compatibilities of the molten salts with the materials of construction of the reactor components. In the case of salt-fueled MSRs, fission events generate fission products that bond with the halides to form new and complex salt species. Corrosion in MSRs is strongly affected by the molten salt chemistry and purity of reagents; therefore, salt purification is an important aspect of consideration for reactor applications. The next section discusses the molten salt properties important for their selection as candidate fuel or coolant salts for reactor applications including isotopes, neutronics, thermophysical properties, and materials corrosion. The purification methods for both fluoride and chloride salts are also reviewed in the later sections.

Properties of molten salts for reactor designs

Though most literature considers fluoride and chloride salt systems for MSR applications, there are no specific reasons deterring the use of iodine and bromine other than the lack of experience and knowledge (Holcomb et al., 2011). In fact, heavier halides can improve burning and breeding by providing a harder neutron spectrum, which can be advantageous for fast reactor applications. Since the purpose of this article is to review the existing knowledge base on MSRs, we leave the consideration of the heavier halides as a front-line research endeavor and limit the exposition to chloride and fluoride systems. Chloride and fluoride systems have distinctive characteristics that are relevant to MSR applications. Understanding these characteristics is important to comprehending the rationale and consequence of selecting one system over the other for specific applications.

Table 1 Summary of historic and proposed salt systems and compositions for MSR applications (World Nuclear Association, n.d.; Fredrickson et al., 2018).

Salt composition	Use in reactor	MSR application	Years
NaF-ZrF ₄ -UF ₄ (53.09-40.73-6.18) mol%	Fuel Salt	Aircraft Reactor Experiment 2.5 MWt	1954
LiF-BeF ₂ -UF ₄ (68.5-31.3-0.2) mol%	Fuel Salt	Two-fluid MSBR, 1000 MWe	1961–1970
LiF-BeF ₂ -ThF ₄ (71-27-2) mol%	Blanket Salt		Design only
LiF-BeF ₂ -ThF ₄ -ZrF ₄ (68-29.2-0.8-5) mol%	Fuel Salt	MSRE, 8 MWt	1965–1969
LiF-BeF ₂ -ThF ₄ -UF ₄ (68-22-9-1) mol%	Fuel Salt	Molten Salt Converter Reactor, 1000 MWe	1961–1965
LiF-BeF ₂ -ThF ₄ (68-23-9) mol%			Design only
LiF-BeF ₂ -ThF ₄ -UF ₄ (71.5-16-12-0.5) mol%	Fuel Salt	Molten Salt Breeder Experiment, 150 MWt	1969–1970
KNO ₃ -NaNO ₂ -NaNO ₃ (44-49-7) mol%	Coolant Salt		Design only

Isotopes

Fluorine is the lightest halide with the average atomic mass of 18.998 g/mol. Chemically, it is the simplest among the halides since it has only one oxidation state available (-1). It is also the simplest isotopic composition because ^{19}F is the only stable isotope among its 17 known isotopes. All the other fluorine isotopes have relatively short half-lives and are not found naturally. The simple chemistry and absence of long-lived isotopes favor fluorine as the top halide candidate for the MSR application. Chlorine is the second lightest halide with the average atomic mass of 35.45 g/mol. Chemically, it is more complex than fluorine because five oxidation states (-1 , 1 , 3 , 5 , and 7) are possible, though it usually exhibits a -1 oxidation state. It has 25 known isotopes and two of them are stable in nature, ^{35}Cl (76% natural abundance) and ^{37}Cl (24% natural abundance). All the other isotopes have relatively short half-lives except ^{36}Cl (301,000-year half-life). The long half-life of ^{36}Cl can be a concern for long-term nuclear waste management.

Neutronics

Lithium fluoride is a commonly adopted constituent for salt-fueled MSRs based on the fluoride system (e.g., FLiBe). Lithium has two stable isotopes, ^6Li (7.59% natural abundance) and ^7Li (92.41% natural abundance). A problem of adopting lithium fluoride to an MSR system is the production of tritium from the fission of ^6Li . Tritium emits beta radiation with a half-life of 12.32 years, which is sufficiently long to require that it be carefully managed to prevent release to the environment. Thus, most proposed molten salt reactor concepts call for salts enriched in ^7Li to minimize tritium production from ^6Li . Literature reports various means to separate lithium isotopes (Hoshino and Terai, 2011; Arisawa et al., 1982; Symons, 1985; Black et al., 2009).

Stable and naturally abundant ^{35}Cl can capture a neutron and become ^{36}Cl . ^{36}Cl is a long-lived isotope and long-term reactor operation with abundant ^{35}Cl can accumulate ^{36}Cl . ^{36}Cl is a high-energy beta emitter and is soluble in water as the salt making it very mobile in the natural environment. Thus, the disposition of the used salt can be problematic for a long-term repository. As discussed earlier, chlorine has two stable isotopes, ^{35}Cl (76% abundance) and ^{37}Cl (24% abundance). Separation of ^{35}Cl and ^{37}Cl can be achieved by various methods such as liquid-phase thermal-diffusion-based separation (Kranz and Watson, 1953) and gas centrifuge (Ragheb, 2012). By minimizing the partition of ^{35}Cl to the reactor, ^{36}Cl production can be minimized. Another source of ^{36}Cl is the $^{39}\text{K}(n,\alpha)^{36}\text{Cl}$ reaction. Thus, adopting KCl is not advisable for the MSR application. Even with this shortfall, the chloride salt system has received attention primarily for fast reactor applications including TerraPower's Molten Chloride Fast Reactor (Kramer, 2019). Chloride salts are advantageous in fast reactors because they have higher solubilities for actinides and are harder to neutrons (less neutron moderating) compared to the fluoride salts.

Thermophysical properties

Fundamental thermophysical properties, such as melting temperatures, specific heat capacity, density, viscosity, thermal conductivity, and thermal diffusivity are critical salt characteristics that impact the design of MSRs, for both fuel and coolant salts (Samuel, 2009; Capelli, 2015; Beneš and Konings, 2012; Fredrickson et al., 2018; Korin and Soifer, 1997). The composition and chemical properties of the fuel and coolant salts vary with time primarily due to fuel burnup (i.e., the fission process that converts fissile isotopes to fission products) and materials corrosion. This leads to changes in actinide, fission product, and corrosion product concentrations within the salts. Consequently, relating the changes in salt composition to changes in the thermophysical properties is important to ensure safe and efficient reactor operations. Because small changes in composition can lead to significant changes in the associated thermophysical properties of the salt, rigorous methods must be developed to prepare high-purity samples used for measuring salt properties, and rigorous analytical practices are needed to make reproducible measurements.

A useful analogy for the physical behavior of molten salts is to compare the properties of molten salts to those of liquid water at room temperature. Most alkali or alkaline earth halides above their melting points have viscosities, heat transfer characteristics, and optical properties similar to those of liquid water, with the obvious difference of temperature. However, just as with aqueous chemistry, changes in the chemical composition of a molten salt solutions can cause large changes in thermophysical properties. Due to the complexity of the composition fluctuations as a function of operating time, the amount of experimental data needed to optimize a molten salt mixture for a particular application is extremely large. Many of these properties are sensitive to impurities, including dissolved metals, moisture, and oxygen-containing impurities, such as oxides, hydroxides, sulfates, carbonates, and oxyhalides. The wide array of salt compositions being investigated increases complexity, as fuel and coolant salts vary by design and neutronic needs for specific MSR concepts. Some of the most important properties needed for fuel or coolant salt selection, reactor design, and computational modeling are listed in Table 2 (Diamond et al., 2018; Magnusson et al., 2020; Williams and Britt, 2017). Although Table 2 is not an all-inclusive list of properties and measurement techniques, it provides an overview of techniques available to determine the properties most important to molten salt performance. The importance of these properties is described in greater detail in the sub-sections below.

It is important to note that salt compositions must be selected not only for neutronic characteristics, but also for properties that ensure robust reactor operation. Computational models validated with limited and select experimentally measured properties data enable the prediction of thermophysical properties over a much broader range of temperature and chemical compositions than realistically can be measured experimentally. High temperature measurements for accurate and reproducible thermophysical properties of molten salts are difficult due to the unique experimental conditions required.

Table 2 Thermophysical and thermochemical properties needed for MSR applications and some experimental techniques for determining each property.

<i>Properties</i>	<i>Application</i>	<i>Experimental technique</i>	<i>References</i>
Density	Thermal Hydraulics, Neutronics	Neutron Radiography, Bubbler Dilatometer, Pycnometer, Thermal Mechanical Analyzer, Archimedes Method	Fredrickson et al. (2018) and Magnusson et al. (2020)
RedOx Behaviors, Precipitation and Plating of Metals	Temperature Distribution, Power Distribution, Liquid Composition Distribution, Gas Transport and Composition, Velocity	Thermogravimetric Analysis (TGA), Inductively Coupled Plasma—Mass Spectroscopy (ICP-MS), X-Ray Diffraction (XRD)	McFarlane et al. (2018), Flanagan et al. (2018), and Beneš (2008)
Phase Stability, Melting Point, Solubility	Operating Parameters, Salt Composition	Differential Scanning Calorimetry (DSC), Thermogravimetric Analysis (TGA), Drop Calorimetry	Beneš and Konings (2009), Beneš and Konings (2012), and Magnusson et al. (2020)
Specific Heat Capacity	Heat Transfer, Energy Transfer	Drop Calorimetry, DSC	Sohal et al. (2013), Beneš and Konings (2009), and Magnusson et al. (2020)
Thermal Conductivity	Heat Transfer, Energy Transfer	Concentric Cylinder, Forced Rayleigh Scattering, Parallel Plate Method, Transient Hot-Wire, Variable Gap, Transient Laser Flash	Beneš and Konings (2012), Sohal et al. (2013), Beneš and Konings (2009), Magnusson et al. (2020), and An et al. (2015)
Thermal Expansion	Heat Transfer Energy, Transfer Velocity, Liquid Composition and Distribution	Transient Laser Flash, Thermal Mechanical Analyzer, Dilatometer	Brown et al. (2020) and Diamond et al. (2018)
Viscosity	Thermal Hydraulics, Temperature Distribution, Power Distribution	Dynamic Neutron Radiography, Viscometer, Rheometer, Falling Ball	Ingersoll et al. (2007) and Magnusson et al. (2020)
Volatility	Temperature Distribution, Power Distribution, Liquid Composition and Distribution, Gas Transport and Composition	TGA, Gas Chromatography—Mass Spectroscopy (GC-MS), Knudsen Cell	Brown et al. (2020), Diamond et al. (2018), and McFarlane et al. (2018)
Wetting, Surface Tension	Interfacial Chemistry, Mass and Energy Transport Across Interface	Sessile Drop, Bubble Pressure	Grimes (1964), Briggs (1964a), Rosenthal et al. (1969); Yuan and Lee (2013), and Ziesing (1953)

There are currently no benchmarks or data collection standards for molten salt samples, and very few commercial instruments are designed to work with molten salts in an off-the-shelf configuration. Salts must be synthesized, purified, and blended to composition with care because of their highly hygroscopic and corrosive nature, and materials compatibility must be at the forefront when determining the experimental design and atmospheric control of instruments.

Phase diagrams

Phase diagrams that describe the solubility limits and melting temperatures of salts as a function of composition are important tools for MSR design. To allow the reactor the widest operating temperature range possible, it is desirable to select a salt system with the lowest melting point possible, while simultaneously maintaining high temperature stability and optimizing other properties. This is typically accomplished by selecting a eutectic composition of two or three higher-melting-point salts; however, in some instances, it is desirable to deviate from eutectic composition for one or more reasons. The commonly used FLiBe composition, 2LiF:BeF₂, is one such example where the composition is chosen to optimize both the melting point and the viscosity. The phase diagram of the binary LiF-BeF₂ system is shown in Fig. 1 (Beneš and Konings, 2009). Addition of more BeF₂, up to the 52 mol% BeF₂ eutectic point, lowers the melting point further; however, due to the complex intramolecular interactions in FLiBe, the viscosity of the eutectic composition is much higher than at the peritectic 2LiF:BeF₂ composition (i.e., 66 mol% LiF, 34 mol% BeF₂; Beneš and Konings, 2009). Additions of other components to the base binary salt composition, such as actinides, fission products, corrosion products, and impurities also affect the phase relations. Consequently, it is necessary to know how the additions of these various species affect the liquidus temperature and solubility limits of other species.

Density and thermal expansion

For salt-fueled MSRs, the inverse relationship between density and temperature is one of the primary passive safety features, as any sudden increase in salt temperature will quickly cause a decrease in fuel density. As the fuel density directly affects the reactivity of the core, an increase in temperature has a net impact of lowering the reactivity coefficient ($k_{\text{effective}}$) and thus the power generation rate. This negative reactivity feedback mechanism is one of the most attractive features of salt-fueled MSRs, but to realize this feature, it is necessary to accurately understand how density fluctuates with temperature and composition. Densities for molten salts are typically between 1.62 g/cc for molten LiCl-KCl eutectic at 500 °C (Sridharan, 2012) to 3.1 g/cc for the NaCl-UCl₃ eutectic at 600 °C (Desyatnik et al., 1975).

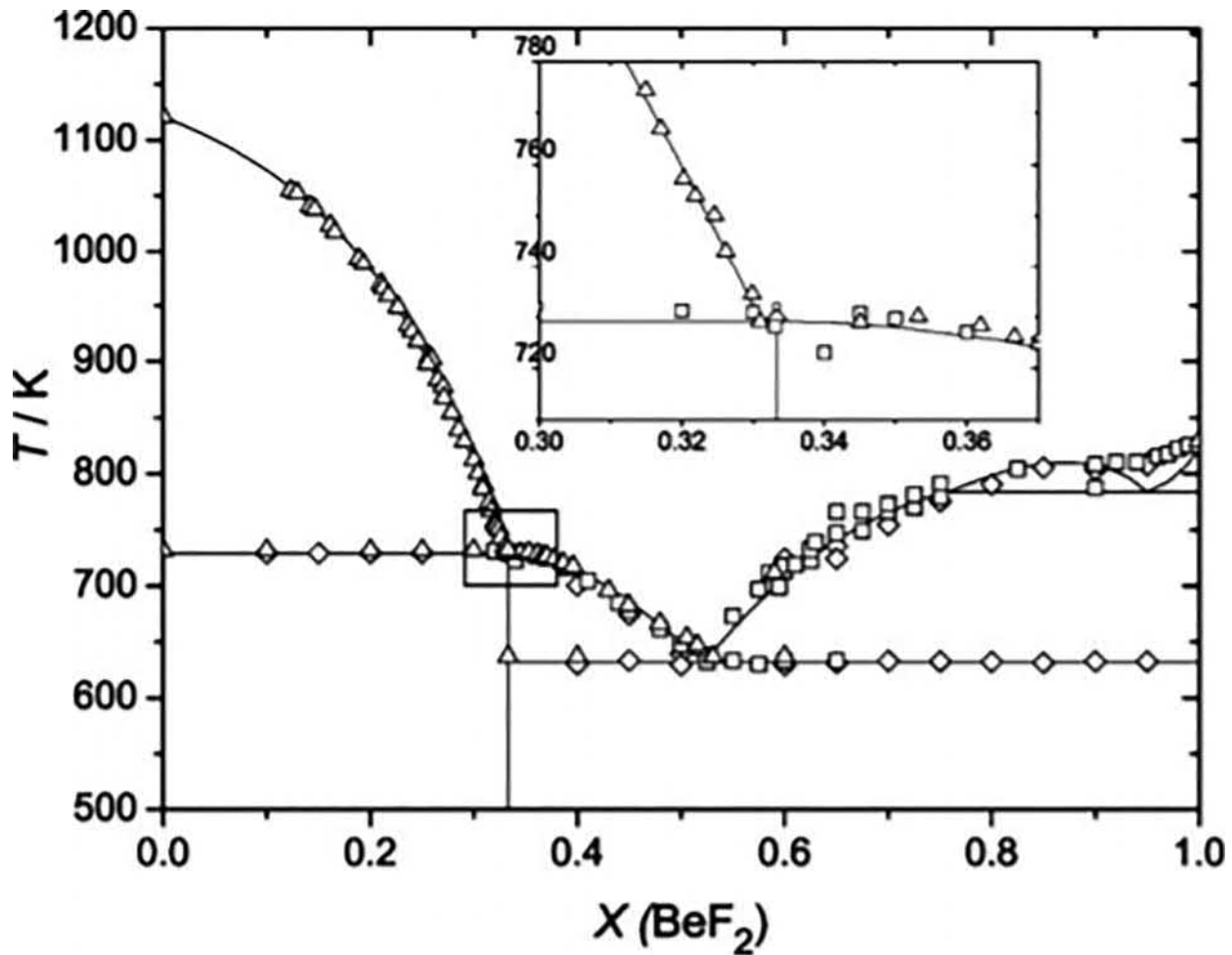


Fig. 1 LiF-BeF₂ binary system phase diagram (Image reproduced from Beneš and Konings, 2009).

Specific heat capacity

The primary purpose of an electrical power producing reactor is to generate heat. Because the properties of a reactor's structural materials are temperature dependent, one of the key safety functions of a reactor is to transfer heat effectively from the core to the heat removal system. The specific heat capacity of a substance is a measure of the amount of thermal energy required to elicit a temperature change. Typically this is measured in terms of J/(kg K), with typical values for molten salts being between approximately 1200 and 2100 J/(kg K) (Sohal et al., 2013). When compared to the specific heat of water of 4185 J/(kg K) (at 20 °C), salts have approximately half the specific heat capacity. However, given the higher density of salts, the volumetric heat capacity of most molten salts is comparable to that of water, meaning that molten salts are very effective heat transfer fluids.

Viscosity and rheology

When considered together, rheology and viscosity provide an understanding of how liquids flow in an unconstrained system and how they behave under the influence of an applied force. An accurate understanding of these parameters is critical for describing the flow of molten salt through the system, including through pumps, piping, and in-core components. Molten salts are Newtonian fluids that follow an Arrhenius-type equation describing their exponential decrease in viscosity with reciprocal temperature (Wakeham et al., 1991; Ingersoll et al., 2007). Due to this behavior, viscosity varies more with temperature than other thermophysical properties. As mentioned above, viscosity is also impacted by variations in composition. For salts containing cations with typical oxidation states greater than 1, such as BeF₂, increasing concentration of the cation can cause an increase in viscosity (due to edge- and corner-sharing interactions between adjacent halide anion coordination shells, which are the loosely organized atomic species that exist in a liquid phase). This behavior has also been observed in melts containing MgCl₂ (Wu et al., 2020) and UCl₃ as well (Okamoto and Ogawa, 1999; Okamoto et al., 2005). Consequently, the effect of the buildup of fission products in salt-fueled MSRs on viscosity is important to understand, as most salt-soluble fission products have an oxidation state greater than 1.

Wetting and surface tension

Wetting properties, such as surface tension and contact angle, influence interfacial behavior (foaming, intrusion into pores, and wetting of surfaces). Determination of interfacial behavior enables the estimation of mechanistic details pertaining to the corrosion of structural materials by molten salts. Under the influence of the forces acting on molecules in the bulk liquid and at the surface, the liquid contracts its surface area to maintain the lowest surface free energy, as shown in Fig. 2 (top image). If the interfacial energy of the liquid-solid interface (i.e., the free energy of the interface) is less than the free energy of the bulk liquid, then the liquid will more easily wet the solid surface; a wetted condition allows more chemical interaction between the liquid molecules and the surface, such as corrosion. The term free energy describes the propensity of a chemical system to spontaneously change. Generally, chemical reactions tending toward equilibrium occur in ways that minimize free energy. Measuring the angle at which the drop contacts the surface provides an understanding of the forces at play within the fluid and in relation to the material. Contact angle measurements performed at ORNL during MSRE operations demonstrated that the contact angle fluctuated with the redox chemistry of the salt and the presence of moisture (Briggs, 1964b; Rosenthal et al., 1969; Ziesing, 1953; Yuan and Lee, 2013). (*Redox chemistry or redox potential*: The term is defined as the tendency of a chemical species to acquire electrons and become reduced. It is a measurable voltage that is related to the concentration of the reduced and oxidized species in the molten salt solution.)

Molten salt purification

The need for the purification of commercial salts was identified early on during MSRE operation at ORNL (Kelleher et al., 2015). Among the impurities present in salts, moisture and oxides are the main species of concern due to their prevalence and their effects on salt chemistry and corrosion. Other, minor impurities include alkaline earth and transition metal fluorides and chlorides. Most salts are highly hygroscopic, with some having such an extreme affinity for water that they tend to deliquesce in presence of humidity, necessitating their handling and experimentation under inert and dry atmospheres (gloveboxes, hotcells, etc.). Absorbed water in salts cannot always be removed by simple heating and drying (Kelleher et al., 2015), as the hydrated salts undergo hydrolysis upon transition to the liquid phase if the moisture is not removed prior to melting, leading to the formation of hydroxides and/or oxides through the Reactions (1) and (2) (X represents F or Cl):

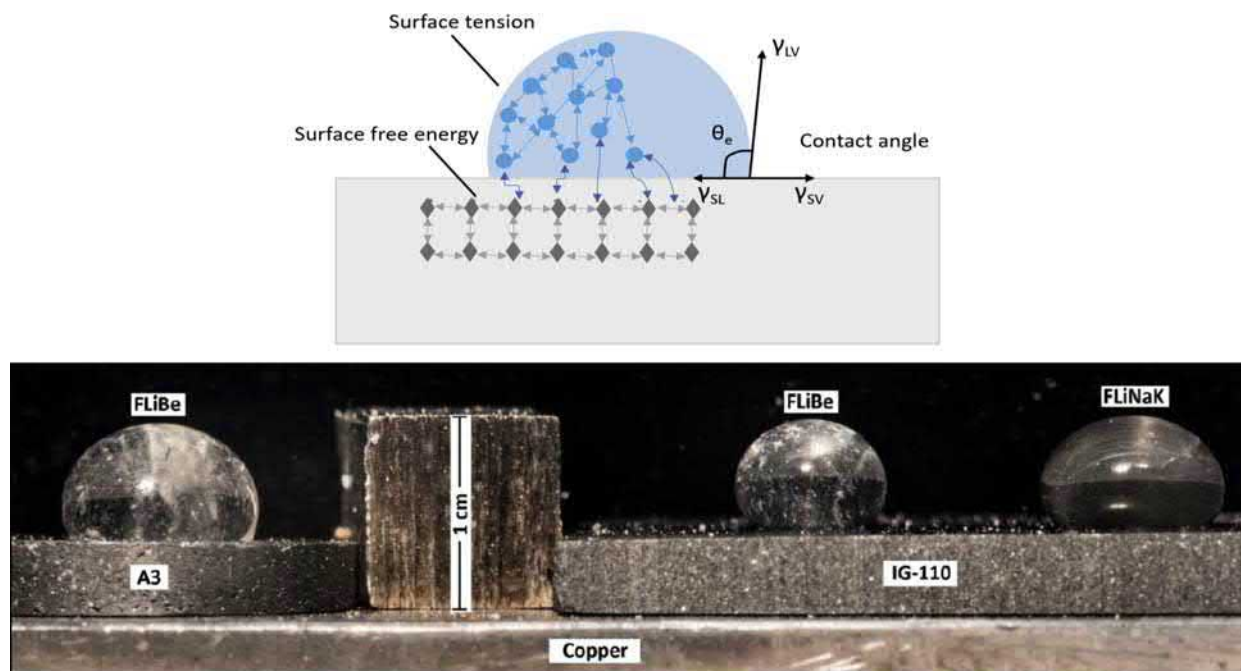


Fig. 2 (Top) Schematic of a sessile drop and the balance of surface tensions at the three-phase contact line. (Below) Contact angle of FLiBe and FLiNaK on two different grades of nuclear graphite—IG110 and Matrix A3. A metallic cube (1 cm side) is included in the picture for scale reference. Both FLiBe and FLiNaK exhibit low wetting on graphite surface. (Below) Image reproduced from Delmore AR, Derdeyn W, Gakhbar R, Scarlat RO (2018) Wetting of graphite by molten fluoride salts: initial experiments. In: Transactions of the American Nuclear Society, Vol. 118, Philadelphia, Pennsylvania, June 17–21, 2018.

The hydroxide ions thus produced results in precipitation of metal ions, attack ceramics, metals, or other container materials, and are more easily reduced compared to other components of the salt; leading to severe corrosion of the container materials. The difficulty of removing the last traces of residual moisture from the salts has often necessitated rather arduous drying procedures. Several techniques have been considered and applied for molten salt drying and purification, including the removal of cationic and anionic impurities, and are summarized in [Table 3](#).

Table 3 Various purification methods considered for fluoride and chloride salt systems.

<i>Techniques</i>	<i>Application/Details</i>	<i>References</i>
Vacuum Drying	Typically used to drive off the majority of moisture present in commercial salts	Kurley et al. (2019)
Distillation, Vacuum Distillation	Common purification technique for separation of impurities from salt	Young and Mamantov (1990)
Gas Sparging	<i>Sparging with an inert gas:</i> Removes volatile components including HCl, H₂S, sulfur and hydrocarbons <i>Cl₂ or HCl for chloride salts:</i> • A common purification technique for removal of hydroxide impurities • Cl₂ has the advantage of being much easier to dry than HCl; however, both gasses at elevated temperatures are highly corrosive <i>Carbochlorination:</i> The oxide impurity in some chloride salts (e.g., MgCl₂) is more difficult to remove by HCl sparging. In such cases, a chlorocarbon compound, such as phosgene or CCl₄, can be used in conjunction with chlorination <i>Mixture of HF/H₂ gas for fluoride salts:</i> • In addition to removing moisture and oxide impurities, this method also removes anionic contaminants, such as chloride and sulfur • Contaminants not removed by the H₂/HF treatment include corrosion products from the materials of construction and neutron activation products <i>NF₃ gas as an alternative to HF:</i> • Removes oxide and water contaminants; to convert contaminant graphite and corrosion products to soluble or volatile fluorides • NF₃ is significantly less toxic compared to HF • Significant global warming potential impact	Johnson et al. (1969) Maricle and Hume (1960) Sun et al. (1991) and Kurley et al. (2019) Kelleher et al. (2015) Scheele et al. (2017)
Dry Chemical Treatment	As an alternative to HF gas, solid NH ₄ HF ₂ was used for conversion of U and Th oxides to tetrafluorides and for subsequent removal of oxygen-containing impurities from fluoride salt constituents	Ignatiev and Surenkov (2017) and Ignatiev et al. (2014)
Zone Refining	• Zone refining is a recrystallization technique for the preparation of ultra-pure materials, commercially • A major advantage of this process is the elimination of the need to separate pure from impure fractions, as in fractional crystallization • Process time can be reduced by having the zones closer, and throughput can be improved with a setup with multiple heaters. • Impurities with distribution coefficient equal to or higher than 1 (Fe, Ni, Ca, Co) are difficult to remove by zone refining, necessitating additional purification steps	Ghosh et al. (2009) , Pamplin (2013) , and Shim et al. (2017)
Reduction by Active Metals or Alloys	Involves the reduction of the salt with a constituent active metal such as an alkali metal, beryllium, or zirconium. An example is the use of a Mg-Cd alloy for removal of metallic impurities from MgCl ₂ . Oxides, hydroxides, and water react with Mg to form hydrogen and insoluble MgO	Johnson et al. (1969)
Multi-step Purification	Due to differences in the separation characteristics of various ionic impurities, utilization of multiple purification procedures in series may be warranted An example is the multi-step procedure purification of chloride salt (LiCl) following various steps:	Hinks and Susman (1974)
	a. Solvent extraction: to chemically remove ions that cannot be removed by zone refining b. Vacuum dehydration: to remove bulk moisture c. Halogen treatment: to remove residual hydroxide species d. Zone refining: for removal of cationic impurities	

Corrosion in molten salts and redox control

Corrosion in molten salt

Molten salts are corrosive to MSR structural materials at high temperatures, particularly through a reactor's service life of 30 years or more. Therefore, successful design and operation of an MSR requires a thorough understanding of corrosion behavior of materials in molten salts. In contrast to traditional high temperature oxidizing environments, the formation of a passivating oxide layer on the surface of alloys is typically prevented due to the solubility of the corrosion products in the molten salt (Sridharan and Allen, 2013; Zhang et al., 2018). As a result, the corrosion resistance of alloys for MSR applications largely depends on the thermodynamic inertness of the alloy in molten salts.

The MSRE generated a large amount of data and experience associated with salt chemistry and materials corrosion in molten fluoride systems. More recently, chloride salts have been considered as top candidate fuel salts for some fast spectrum MSRs, but currently corrosion data is less available. Molten fluoride and chloride salts both share some features of salt chemistry and corrosion behavior in MSR environments, meaning that experience with fluoride salts can inform the analysis of data for chloride salts. However, the thermodynamic property differences between fluoride and chloride salts leads to some differences in corrosion performance of materials and salt chemistry control. Several comprehensive review papers on salt chemistry and materials corrosion in these salts are available (Guo et al., 2018; Wang et al., 2018; Ignatiev and Surenkov, 2017; Zhang et al., 2018). Generally, there are several different corrosion mechanisms (Williams and Britt, 2017; Sohal et al., 2013): (1) intrinsic corrosion, (2) impurity driven corrosion, (3) temperature gradient-driven corrosion, (4) dissimilar metal-driven corrosion, and (5) stress corrosion cracking.

Intrinsic corrosion

Intrinsic corrosion is caused by chemical reactions of the material constituents with the salt itself. When the salt is perfectly clean and pure and the metal surface is free from any oxide layers, the corrosion is a result of redox reactions between the alloying elements in the metal and the ions in the salt. Because alkali and alkaline earth halide salts are very chemically stable, the reaction between these salts and the alloying elements in stainless steels and nickel-based alloys (e.g., Cr, Ni, Mo, W etc.) are thermodynamically unfavorable, meaning that intrinsic corrosion rates should be minimal.

Impurity driven corrosion

Impurities including moisture, oxides, and other oxidants in molten salts can come from different sources during manufacturing, handling, storage, and operations. The impurities can react with the salt melt, form corrosive agents, and then attack the structural materials. Reactions (3) through (6) (where X represents F or Cl) show the typical impurity induced corrosion reactions.



Chromium is used here as an example as it is the most active alloying element in most structural alloys, meaning that it will be preferentially depleted from the alloy before other alloying elements are attacked. Similar reactions can occur with other alloying elements depending on the thermodynamic equilibrium of the salt-metal interface. It is important to note that impurity driven corrosion causes accelerated corrosion initially, and as impurities are consumed, the corrosion rate will decrease assuming no further impurities are added (Sridharan and Allen, 2013).

Temperature gradient-driven corrosion

The equilibrium constant for the reaction between salt and the main active alloying element, typically chromium, is temperature dependent. Considering uranium fluoride salts as an example, at higher temperatures, the equilibrium constant for Reaction (7) is higher, which means higher corrosion rates, while at lower temperatures, the equilibrium constant is lower, and the corrosion rate is lower.



In a system with large temperature gradients, such as in an operating MSR or thermal convection loop (*Thermal convection or natural circulation loop*: Natural convection flow results from the difference in density of the fluid (molten salt in this case) in the hot and cold portions of the loop. Thus, the flow is generated and maintained without the use of any external source like pumps, fans, suction devices, etc.), this temperature dependence for the equilibrium concentration of CrF_2 in the salt drives mass transport of CrF_2 from the hot sections to the cold regions. At the colder areas, when the concentration of CrF_2 from the hot sections is greater than the equilibrium concentration of CrF_2 in Reaction (7), the equilibrium can shift to the left side and lead to the deposition of chromium. Fig. 3 represents an example of the corrosion difference in hotter and colder sections of a schematic thermal convection loop with test samples inserted: more severe corrosion in hotter sections and chromium deposition in colder sections (Koger, 1972). This temperature gradient-driven corrosion is considered the primary corrosion mechanism for sustained corrosion in MSRs.

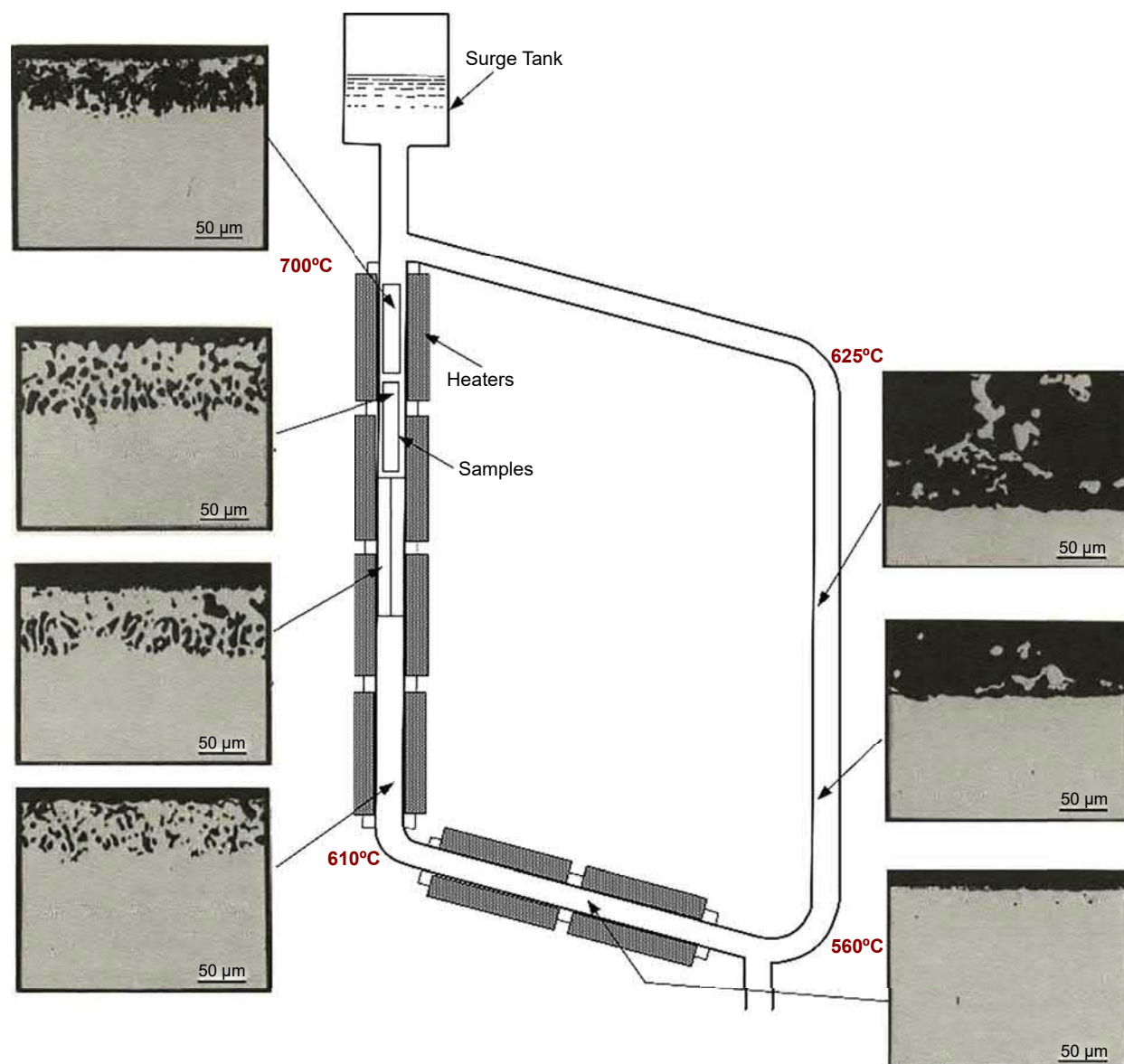


Fig. 3 Micrographs of Hastelloy N exposed to $70\text{LiF}\cdot 23\text{BeF}_2\cdot 5\text{ZrF}_4\cdot 1\text{ThF}_4\cdot 1\text{UF}_4$ molten salt in a thermal convection loop after 9.2 years. Image reproduced from Koger JW (1972) "Evaluation of Hastelloy N Alloys after nine years exposure to both a molten fluoride salt and air at temperatures from 700 to 560°C". Oak Ridge National Lab Report, ORNL-TM-4189.

Dissimilar materials effects

When dissimilar materials are present in the molten salt, the corrosion rate can be accelerated. The dissimilar materials do not necessarily have to be electrically contacted with each other, particularly when one of the materials has a chemical affinity for the corroded species in the salt and forms a compound, as is the case for carbide-forming alloying elements in the presence of graphite, such as chromium (Sridharan and Allen, 2013). When the two dissimilar materials are in electrical contact, galvanic corrosion occurs, leading to accelerated corrosion of the more active material. For example, Alloy 800H tested in a graphite crucible was found to have a much higher corrosion rate than when tested in an Alloy 800H crucible (Sridharan and Allen, 2013). It should be noted that a similar galvanic corrosion effect can occur on the micro-scale in alloys due to microstructural dissimilarities between different regions of the material, such as precipitates and grain boundaries and the composition differences between welds and the base material (Sohal et al., 2013).

Stress corrosion cracking

In the MSRE, it was found that tellurium-assisted stress corrosion cracking occurred in Hastelloy N in FLiBe. Tellurium was found to diffuse into the Hastelloy N and form brittle intergranular precipitates. A concerted effort to control this effect led to the discovery that tellurium-assisted stress corrosion cracking can be avoided by maintaining the salt under more reducing conditions by using

a lower ratio for UF_4/UF_3 (for example 95:5 or lower) and by using a modified Hastelloy N alloy with additions of Ti or Nb to preferentially react with Te (McCoy, 1978).

Redox control

Transmutation and fission reactions in the reactor deplete fissile materials and lead to accumulation of fission products in fuel salt, changing the composition and redox state of the salt to generate a net-oxidizing environment. It is necessary to control the redox properties of the salt to minimize the corrosion of materials. There are various strategies for redox control of salt, depending on the thermodynamic properties of the halide salts, including H_2/HF sparging, active metal insertion, active electrochemical means, and the strategic use of salt components with multiple possible valence states to buffer the redox state of salt (Zhang et al., 2018).

During operation of the MSRE, the redox control of fuel salt was achieved by maintaining a UF_4/UF_3 ratio of about 100:1, leading to a redox state that is not unacceptably corrosive to the non-ferritic materials of construction (i.e., nickel-based alloys). It should be noted that the reason for maintaining such a high UF_4/UF_3 ratio is that in molten fluoride salts UF_3 is not stable unless a large excess of UF_4 is present due to a disproportionation reaction shown in Reaction (8) (Grimes, 1970).



If the UF_4/UF_3 ratio is not sufficiently high, metallic U can deposit on surfaces within the reactor and subsequently react with the moderator graphite and structural alloys. Maintenance of a proper UF_4/UF_3 ratio for redox control was achieved in MSRE by periodic immersion of a beryllium rod, an active reducing element, with the salt, via Reaction (9) (Thoma, 1971).



Due to the slight solubility of Be metal in FLiBe, this control strategy had the benefit of being able to control the redox state of the salt in a distributed manner, rather than in a discrete location that may occur when using reducing metals that are insoluble in salts.

For redox control in chloride salts, the $\text{U}^{3+}/\text{U}^{4+}$ ratio is also considered as one of the primary control mechanisms. However, because there is higher relative stability of UCl_3 in comparison with UF_3 , much lower $\text{UCl}_4/\text{UCl}_3$ ratios of roughly 0.003–0.05:1 are expected to minimize the corrosion of nickel-based structural alloys (Holcomb, 2017). The presence of a small amount of UCl_4 can help inhibit the disproportionation reaction of UCl_3 to U and UCl_4 . However, experimental validation is needed to verify and identify the ideal $\text{UCl}_4/\text{UCl}_3$ ratio range for long-term chloride salt-fueled MSR operation.

Conclusions

The discussion above on the various neutronic, thermophysical, chemical and materials compatibility properties of molten salt systems is intended to serve as an overview of the physical phenomena that must be understood for the design and operation of an MSR, and the state of knowledge in the fields as of the date of publishing. Much more detail on each of these properties is available in the literature sources cited, as well as in many other journal articles, book chapters, and reports that were not included here. Recent renewed interest in advanced nuclear reactors, and MSRs specifically, has significantly accelerated the rate at which the knowledge base for the topics discussed in this report is expanding, necessitating those interested in MSR development to remain diligent about maintaining an up-to-date understanding of the research that has been recently conducted. The reader is thus encouraged to use this report as a jumping off point to explore the intricacies and nuances of the details of molten salt systems and molten salt reactors.

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Irradiation Performance: Light Water Reactor Fuels

Paul Cantownine^a and Bob Rand^{b,*}, ^a Core and Fuel Engineering, Global Nuclear Fuel – Americas, Wilmington, NC, United States; and ^b Global Nuclear Fuel – Americas, Wilmington, NC, United States

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Glossary

Burnup A term used to quantify how much ^{235}U has fissioned. The commercial industry quantifies burnup in terms of power (MWd/kgU or GWd/MTU).

Dislocation A line defect within the atomic lattice of crystalline materials. It is most often thought of as an extra partial plane of atoms in the lattice. Movement of dislocations are the underlying mechanism for plasticity and high strain-rate creep.

Lattice The periodic arrangement of atoms in crystalline materials.

LHGR Linear heat generation rate—power generated per unit length of fuel rod.

Reactivity Reactivity is a measure of deviation from criticality—the state in which the rate of neutron generation from fissioning equals the rate of neutron losses.

SAFDL Specified acceptable fuel design limit (SAFDL) is a term used by the Nuclear Regulatory Commission to communicate a fuel performance limit that is acceptable for showing regulatory compliance. For example, one fuel performance SAFDL is that fuel melting will not occur.

Washout When fuel degradation occurs after a fuel failure, large open axial cracks can occur exposing surface area of the fuel pellets to the coolant. When this happens, the fuel oxidizes further, and small pieces become entrained in the coolant. This process of losing fuel to the coolant is called washout. Fuel degradation is uncommon today.

Abbreviations

AOO Anticipated operational occurrence

DBA Design basis accident

HBS High burnup structure

LOCA Loss of coolant accident

*Retired.

PCMI Pellet cladding mechanical interaction
RIA Reactivity initiated accident

Introduction

Light-water reactor (LWR) fuel is elegant in its simplicity. The fuel is in the form of ceramic uranium dioxide (UO_2) cylindrical pellets with an approximate 1:1 length-to-diameter ratio that are contained in a thin-walled tube made of a zirconium alloy called the cladding (see Fig. 1). The relative content of fissile ^{235}U ranges from natural (0.71% ^{235}U) to as high as 5% enriched. These fuel rods are then assembled into square-array bundles (or assemblies). Over time the fuel rod diameters have decreased, which decreased the linear heat generation rate (LHGR) for given rods while the number of rods per assembly has increased, maintaining bundle powers and enabling increased bundle powers. In boiling water reactors (BWRs), the fuel bundles were initially a 6×6 square array of fuel rods but the number of fuel rods in an assembly increased over time to improve performance. Modern fuel is 10×10 with some fuel vendors introducing 11×11 fuel designs. Unlike BWRs, the array size of pressurized water reactor (PWR) fuel is static for each specific reactor design. In some of the older plant designs, the array size is 14×14 or 15×15 while the newer designs use 17×17 fuel. For perspective, the fuel rod diameters of BWR 11×11 fuel is comparable to PWR 17×17 fuel (~ 9.5 mm), so the PWR fuel assemblies are larger with a greater number of comparably-sized fuel rods. Another difference between BWR and PWR fuel is the channel box or fuel channel surrounding each BWR fuel bundle. The channel box eliminates cross flow and creates a gap between the fuel assemblies where water does not boil, providing neutronic benefits.

The process of manufacturing a nuclear fuel rod starts with fabricating UO_2 fuel pellets. The first step in this process is the conversion of uranium hexafluoride (UF_6) gas into UO_2 powders. The powders are then blended to the desired enrichment, pressed into a cylindrical pellet and sintered to a very high density (95–98% theoretical density). The physical differences between PWR and BWR fuel pellets (diameter, density, enrichment and microstructure) have only a minor effect on fuel performance.

One important lesson learned from the early days of manufacturing LWR fuel is that open porosity should be minimized. Open porosity refers to connected pores that provide pathways to the surface of the pellet, while closed porosity refers to isolated pores inside the pellet. The problem with open porosity was that it created large surface areas where moisture could be adsorbed prior to pellet insertion into the cladding tubes. This moisture was then released and/or radiolyzed during irradiation, reacting with the inner surface of the cladding, releasing hydrogen that was then absorbed in the cladding, causing primary hydride failures. Today, pellet specifications limit the amount of open porosity and moisture content in the pellets.

Traditionally LWR fuel cladding has been fabricated from a zirconium alloy, although there were some very early fuel rod designs that used stainless steel as the cladding material. Zirconium alloys were developed as structural materials for in-reactor applications within the Navy Nuclear Propulsion program in the United States during the early 1950s. In 1952, Westinghouse patented

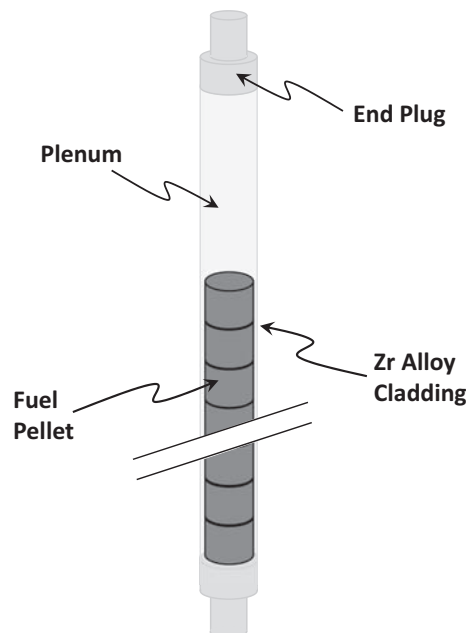


Fig. 1 Schematic of a typical LWR fuel rod.

a zirconium alloy they called Zircaloy-2, which is ~98% zirconium with additions of tin, iron, chromium and nickel. Soon after, various zirconium-niobium alloys were also developed in Russia and England. The use of zirconium as a cladding material was driven by the low thermal-neutron absorption cross section of zirconium and the good corrosion resistance and structural performance in LWR conditions.

The development of zirconium cladding materials since the 1950s has been different for BWRs (Hamon, 2021) and PWRs (Brown, 2021). PWR fuel primarily used an alloy called Zircaloy-4 until the late 1990s when various zirconium-niobium alloys were introduced, while BWR fuel vendors still universally use Zircaloy-2 for the cladding (though with optimized chemistry and processing compared to early Zircaloy-2). The primary reason for the difference between BWR and PWR cladding is that the water chemistry in a BWR is oxidizing while in a PWR it is reducing. However, while there have been different solutions to cladding development, the development in both BWRs and PWRs was driven mainly by the desire to improve corrosion resistance and decrease the pickup of corrosion-generated hydrogen into the cladding. Looking ahead, cladding development is moving toward adding coatings to further improve corrosion and hydrogen pickup performance of zirconium alloys and even non-zirconium materials are being considered.

Although simple in concept, one reason LWR fuel is such an interesting technology is the diversity of knowledge needed to understand its irradiation performance. From the nuclear engineering perspective, understanding the fission process within the fuel is important because it is the fundamental phenomenon that drives all other considerations. There is also the need to understand the plant operations that define the performance requirements, including normal steady-state operations, anticipated transient events (or anticipated operational occurrences, AOOs), and design basis accidents (DBAs). From the mechanical engineering perspective, nuclear fuel is a problem of getting heat from the ceramic UO_2 fuel through the metal zirconium alloy cladding to the hot and pressurized water coolant while maintaining the mechanical integrity of the fuel rod throughout life. And from the materials science perspective, one must understand the effect of the fission process on the properties of UO_2 fuel, the effect of the neutron flux on the properties of the zirconium alloy cladding and other structural components, and the effect of the highly corrosive water on the component surfaces. These issues have filled the careers of many engineers over the last 70 years.

Given the vast data and knowledge regarding LWR fuel, the following can only be an introduction to the important aspects of the irradiation performance of LWR fuel. In the next section the performance requirements are discussed from a regulatory perspective. Then for each type of operation, the fuel performance requirements and phenomena are discussed in detail: normal steady-state operation is covered in section “**Normal operation performance**,” anticipated transient events in section “**Anticipated transient performance**” and design basis accidents in section “**Design-basis accident performance**.” Section “**Fuel performance analyses**” provides a short discussion of the fuel performance analyses needed to show compliance to regulatory requirements, and the final section provides a summary.

Performance requirements

The primary objective for commercial reactors is safe generation of power. Safe operation is always defined in the context of safety to plant workers and the public living and working around the plant. The fuel rod is the first of three barriers that protect workers and the public from the highly radioactive isotopes in the fuel. The two other barriers are the reactor pressure vessel and the containment structure. Therefore, regulators either define or provide guidance on the fuel performance requirements to ensure fuel failure (i.e., breach of the cladding) does not occur or to explicitly account for fuel failures to ensure there is no safety impact (Rempe, 2021). As an example, the US Nuclear Regulatory Commission (NRC) has issued NUREG-800 Standard Review Plan (SRP), Section 4.2 (USNRC, 1987), which defines Specified Acceptable Fuel Design Limits (SAFDLs) that can be used to confirm a fuel design will operate safely. SAFDLs of primary importance for fuel rods during normal operation and anticipated transients are those that address fuel rod temperature, cladding strain and fuel rod internal pressure.

The first and most important fuel performance requirement is to prevent cladding failure and the potential release of gaseous fission products or fuel washout. This is done by limiting the cladding strain under low strain-rate conditions that occur during normal operation and high strain-rate conditions during the anticipated transient events and design basis accidents. As discussed in the following sections, the strain limits are not challenged under normal operating conditions but are during the transient and accident events.

The second requirement is to avoid fuel melting, particularly during normal operation and anticipated transients. Thus, reactor operation must be controlled to ensure the temperature of the center of the fuel pellet remains less than the melting temperature of the UO_2 fuel, which is a function of the specified as-fabricated chemistry and the changes in chemistry due to accumulation of fission products. Because fuel centerline temperature will follow power level, the avoidance of fuel melting limits the fuel power (i.e., the LHGR) that can be experienced in anticipated transient events, which in turn limits LHGR during normal operation.

The third requirement is to limit fuel rod internal pressure. During normal operation, gaseous fission products (primarily Xe and Kr) are released to the fuel rod void space, resulting in an increase in rod internal pressure. The pressure is limited to prevent large cladding diametral strains and possible cladding failure. The limit on fuel rod internal pressure is a concern during normal operation. The size of the plenum region in the fuel rod is a variable that helps manage rod internal pressure.

For DBAs, fuel rod failure is permitted so the fuel performance requirements are defined to minimize the number of failed rods and their effects. In the loss-of-coolant accident (LOCA), the purpose of fuel performance limits is to ensure that when water is added to the reactor to replace lost coolant the fuel rods can be effectively cooled. For example, the NRC limits cladding

temperature, cladding oxidation, and hydrogen generation from cladding oxidation (see 10 CFR Part 50.46) to avoid flow blockage because of clad ballooning when temperatures increase and fuel rod fragmentation (i.e., the fracture of a fuel rod into pieces) when temperature decreases rapidly during quench. The other accident that impacts the fuel is the reactivity-initiated accident (RIA). This accident is a special kind of power transient, and thus, similar to the anticipated transients, the performance requirements are defined relative to mainly cladding strain and fuel temperature.

In addition to meeting the regulatory requirements, fuel rods are designed to assure high reliability and to meet specific performance and operation requirements, including those on power density, neutron flux range, temperature range, lifetime, and fuel cycle economics.

Normal operation performance

Normal operation includes steady-state performance where the power is generally held constant (or at steady state), start up, cool down and any power changes that are initiated in the control room. During start up, the reactor pressure is increased to operating pressure and the water is pre-heated to nearly operating temperatures. The control rods are then removed slowly until fissioning starts to occur in the fuel pellet, adding heat to the system and increasing the intensity of the neutron cloud (or flux) that feeds further fissioning. When the reactor is at power, there are axial and radial power distributions across the core. The radial distribution is such that the bundles in the outer periphery are at a lower power than bundles in the center region of the core. Axially in BWRs the power distribution is initially peaked at the bottom but transitions to top peaked over the operation time in the cycle. In PWRs the axial distribution is not as peaked and does not change as significantly during operation. Lastly there can be power distributions within the fuel rods of a single bundle. This is more prevalent in BWRs because of the water gaps between the bundles, which increase moderation near the edge rods. An effect of this variability in power is that there will be a small number of locations in the core that experience limiting fuel performance conditions during operation, and these locations may change during the cycle.

The main difference between the PWR and BWR plant designs, with regard to fuel performance, is the pressure and temperature of the reactor water. BWRs are operated at a reactor pressure of ~ 7 MPa, and the coolant temperature along the length of the fuel is a constant 288°C . In contrast, the PWR is operated at approximately twice that pressure (~ 14 MPa) and the temperature increases from $\sim 280^\circ\text{C}$ at the inlet to $\sim 315^\circ\text{C}$ at the outlet. The constant axial temperature in BWRs is a result of meeting the boiling condition at all elevations except for inlet region at the lower portion of the bundle where the water temperature is somewhat lower. In PWRs, the high pressure suppresses boiling creating a thermal hydraulic condition allowing the water temperature to increase axially along the length of the fuel rod. The impact of these differences is that the cladding temperature is higher in a PWR than a BWR. Therefore, cladding phenomena driven by temperature (like cladding corrosion and hydrogen pickup) are more severe in a PWR than a BWR.

As discussed in section “**Performance requirements**,” the one fuel performance requirement that is challenged during steady-state operation is fuel rod internal pressure. The purpose of this performance limit is to avoid excessive creep strain in the cladding, which could cause rupture. There are two common ways to ensure the cladding does not rupture during normal operation. The most conservative approach is to limit the rod internal pressure to the coolant pressure of the reactor vessel, making large creep strain deformation impossible. The second approach is to limit the rod internal pressure to a value that assures the cladding will not creep out faster than the fuel is swelling. This limit prevents the outward creep of the cladding away from the fuel surface (i.e., cladding liftoff), which increases heat transfer resistance and raises fuel temperature, causing further fission gas release and creep out.

Fuel performance phenomena in steady state

Before fuel-rod performance can be predicted for steady-state operations, the underlying physical phenomena that occur during operation must be understood. This understanding then allows mechanistic, phenomenological or empirical models to be incorporated into an integrated fuel-rod model that will predict both the thermal and mechanical response of the fuel rod. The fuel pellet is the more complex part to model. The thermal stresses that are generated during start up cause the pellet to expand and fracture. Then over time, the accumulation of fissions (referred to in the industry as fuel burnup) causes changes in the fuel pellet with the introduction of fission products (both solid and gaseous). The fission products cause pellet swelling. The high temperatures inside the pellet result in fuel pellet densification, particularly for pellets with high initial porosity. In addition, the gaseous fission products eventually collect into fission gas bubbles and are partially released over time to the void space in the fuel rod. The expansion of the pellet from thermal expansion and fission-induced swelling combined with the internal gas pressure eventually causes pellet cladding mechanical interaction (PCMI), which in turn causes expansion of the cladding diameter. Finally, distribution of fission rate through the fuel pellet results in a distribution in burnup within the fuel. This causes greater changes in the microstructure of the pellet rim (where the power and burnup are highest) creating what is commonly referred to as a high-burnup rim structure. Fuel burnup is the main source of physical changes to the pellet, and effects of the fast neutron flux combined with PCMI induce the greatest cladding geometry and material properties changes. The fast neutron flux causes irradiation hardening, irradiation growth and irradiation creep and enhances corrosion, which has the result of increasing hydrogen pickup. The following sections provide a detailed description of these pellet and cladding phenomena during steady-state operations.

Fuel pellet performance phenomena

A fuel rod is manufactured by inserting sintered UO_2 pellets into a zirconium alloy tube, and then welding zirconium alloy plugs on both ends to hermetically seal the tube. Generally, there is an unfueled length at the top of the rod, referred to as a plenum, intended to accommodate released fission gases. The environment inside the rods is pressurized helium. To ensure that fuel pellets can be loaded into the cladding tube by automated rod loaders there is a gap between the pellets and the tube (~ 0.1 mm). This is the condition of a fuel rod at the start of operation.

The fission process

There are really two phenomena to consider when thinking about the fission process occurring within UO_2 fuel pellets. The first is the fission event itself, where a thermal neutron is absorbed by a ^{235}U atom causing ~ 200 MeV energy to be released—most as kinetic energy when the uranium atom splits into fission products. The second phenomenon is the conversion of the fission-product kinetic energy to heat energy within the crystalline lattice of a ceramic fuel pellet. This process is illustrated using the simplified UO_2 lattice of Fig. 2. After the fission event occurs (see Fig. 2A), the fission products are ejected from the original lattice site creating a damage path in their wake, eventually settling into a substitutional or interstitial uranium lattice site (see Fig. 2B). The damage to the lattice is quickly “healed” as most of the atoms knocked out of their lattice sites return. However, at the end of this process (which is on the order of picoseconds), some point defects will remain (see Fig. 2C).

The distribution of fission products varies mainly between elements with mass numbers of 75 and 160. In addition, because of radioactive decay of most fission products, the distribution is constantly evolving. For the purposes of fuel performance, the fission products can be characterized as either solid or gaseous. All fission products contribute to solid fuel swelling initially; the gaseous fission products may combine into bubbles which further drive swelling and can be released from the fuel to increase the rod internal pressure or cause a chemical degradation of the cladding.

Fuel pellet cracking and relocation

When the power increases in a fresh rod, the pellets heat up. A strong temperature gradient inside the pellet forms with the center of the pellet having the highest temperature (see Fig. 3). Because thermal expansion will follow temperature, the pellet volume near the center of the pellet will increase more than the volume near the pellet surface, which creates radial and axial gradients of thermal strain and stress, ultimately causing the pellets to fracture into smaller pieces due to radial and transverse cracks. The shapes of these pieces are idealized as wedges in Fig. 4A, considering the thermal strains and resulting stresses run in both the hoop and axial directions. The rapid energy release associated with this pellet cracking can result in relative motion of the pellet pieces filling the gap between the pellet and cladding. Such motion may also occur due to forces from the cladding being applied to the pellet pieces. This motion is referred to as “relocation.” Because sintering during normal operation may heal cracks created during the initial power rise, this phenomenon of cracking and relocation may also occur during subsequent power rises. As described above, relocation is assumed to transfer gap volume to internal pellet cracks. Thus, relocation is assumed to occur only at fuel burnups where a theoretical gap may exist.

Fuel densification and swelling

Fuel densification during normal operation can be thought of as a continuation of the sintering process used to fabricate the pellets to 95–98% of theoretical density. Sintering is a specific type of heat treatment of materials—usually ceramics—which is used to densify packed-powder structures into nearly fully dense components. The driving force for this densification is the transformation of the high-energy surface areas of the powders being sintered to low energy grain boundaries. The rearrangement of the atoms occurs via thermal diffusion, thus the need for high temperatures. During in-reactor operation, diffusion of uranium and oxygen continue because of temperature. At 100% power the range of temperatures from the outer surface to the center is ~ 500 °C to

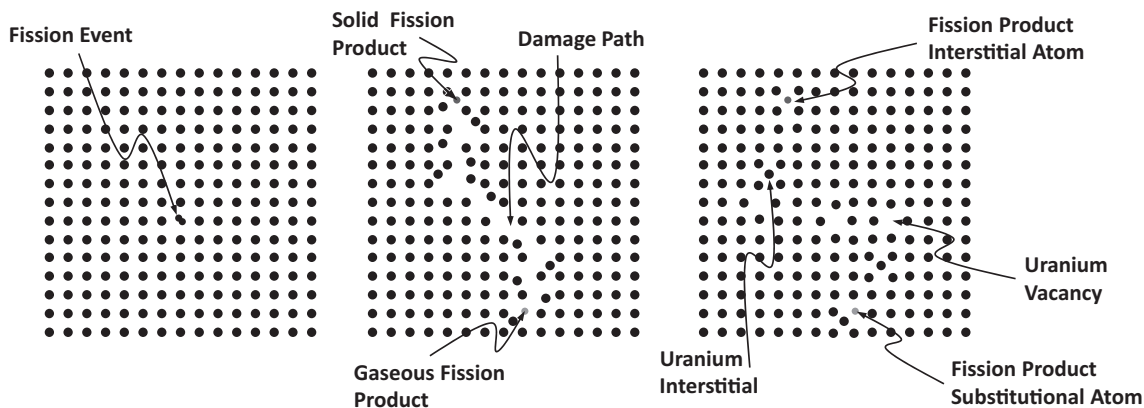


Fig. 2 A schematic illustration of the fission process in a UO_2 lattice. (For simplicity, the oxygen atoms are not shown.) (A) Fission, (B) fission damage, and (C) point defects created after “healing” of fission damage.

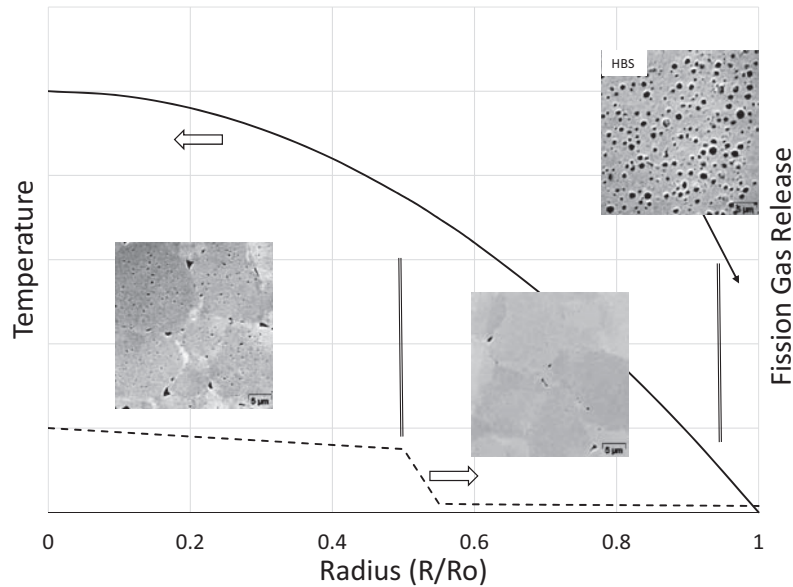


Fig. 3 Schematic of the profile of temperature and fission gas release in a fuel pellet during normal operations. Fuel microstructure images from Garcia P, Martin G, Sabathier C, Carlot G, Martin P, Dorado B, Freyss M, Bertolus M, Skorek R, Noirot J, Noirot L, Kaitasov O, Maillard S (2012) Nucleation and growth of intragranular defect and insoluble atom clusters in nuclear oxide fuel. *Nuclear Instruments and Methods in Physics B* 277, 98–108.

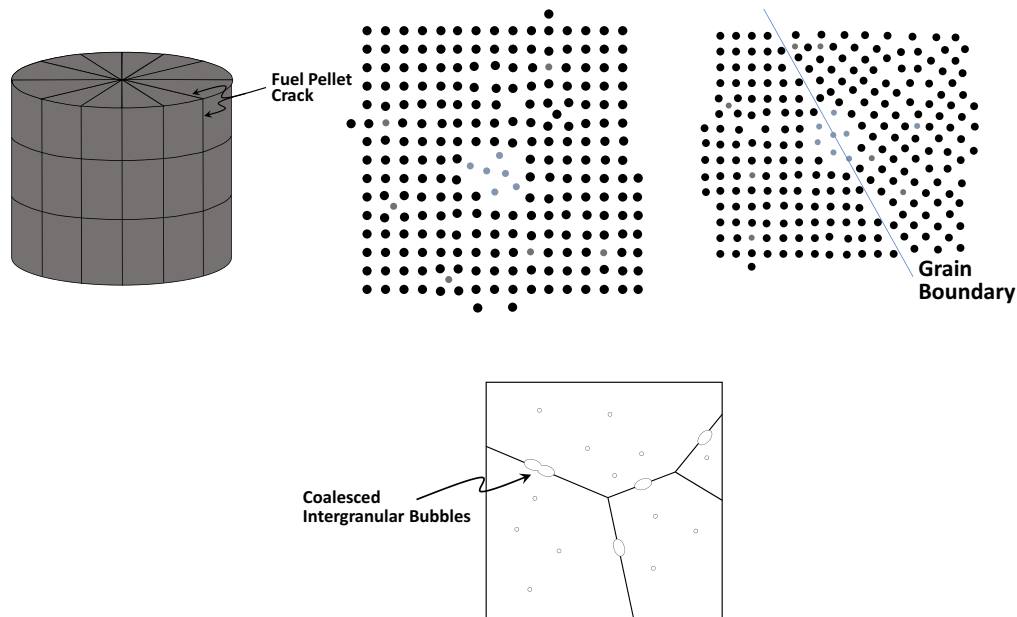


Fig. 4 Simplified schematic representation of various fuel pellet phenomena. (A) Fuel pellet cracking, (B) intragranular fission gas bubble at atomistic level, (C) intergranular fission gas bubble at atomistic level, and (D) intergranular and intragranular fission gas bubbles at microscopic level.

~1800 °C. But diffusion will also occur when atoms are displaced by fast neutron bombardment or by recoiling fission fragments. This nuclear energy is then added to the thermal energy in the pellet making densification via in-reactor sintering easier than out of reactor. In addition to sintering causing fuel densification, it will also “heal” cracks created during power rises; that is, the crack surfaces may rebond.

However, sintering is not the only mechanism for densification. Fuel densification will also occur because of hot-pressing which is the simultaneous application of high temperature and stress. During in-reactor operation when fission occurs and a ^{235}U atom splits, the resulting fission products will occupy a greater volume than the original ^{235}U atom. As shown in Fig. 2C, the result of a fission event is one more atom in the lattice and additional point defects. The reality is of course much more complex than shown in Fig. 2 as the various fission products behave differently (some being stable within the UO_2 lattice while others form second phase

particles) and point defects are also being created in the oxygen lattice. These lattice defects cause a local volume increase, resulting in a local stress field around the defect. As this internal stress builds up, it increases the driving force for diffusion into as-fabricated porosity causing further densification.

Once the as-manufactured porosity is eliminated by either sintering or stress-induced diffusion, the volume change associated with the creation of fission products in the UO_2 lattice causes the macroscopic dimensions of the pellet to increase. This apparent swelling is usually referred to as solid swelling. Two important fission products are Xenon and Krypton, which are noble gases. Like other fission products, these elements will contribute to solid swelling when they are created. But unlike other fission products, these elements can eventually collect together to form fission gas bubbles—either inside grains as intra-granular bubbles (Fig. 4B) or between grains as inter-granular bubbles (Fig. 4C). However, because solid swelling models are generally empirical and based on macroscopic measurements of pellet swelling, any volume change associated with the precipitation of fission gas into bubbles is already accounted for in the data. This volume change is also a function of the hydrostatic stress state on the pellet due to the PCMI between the pellet and cladding, which is extremely complex and difficult to model. Thus, most fuel performance models will not explicitly account for the formation of fission gas bubbles in their swelling models. Fission gas swelling separate from solid swelling is treated empirically in some models. Fission gas swelling will be considered again when discussing transients where burnup-driven solid swelling is not a factor.

Fission gas release

In addition to contributing to burnup-driven swelling, whenever Xenon or Krypton atoms reach a free surface, the gas atoms will be released to the fuel rod volume causing an increase in the internal gas pressure of the fuel rod. To reach free surfaces, which could be the outer surface of the pellet or a crack created during a power increase, the fission gas atoms must diffuse from their creation sites through the UO_2 lattice. (When a fission event occurs near a free surface, the resulting fission gas atom may be ejected directly into the free volume. This mechanism accounts for a very small percentage of fission gas release and is not discussed further.) While some fission gas is released by diffusing directly to a free surface, the majority of fission gas is released through a two-step process. Step 1: fission gas diffuses through the lattice to grain boundaries where inter-granular fission gas bubbles precipitate and grow. The lattice diffusion step is highly dependent on temperature—increasing as temperature increases. Step 2: over time as burnup increases, these inter-granular bubbles will eventually coalesce (see Fig. 4D). When significant coalescence occurs and fission gas bubbles grow large enough to interconnect, a continuous path to a free surface can be created causing a small burst of fission gas release. This process of fission gas release through interconnected inter-granular bubbles is dependent on temperature, burnup, grain boundary strength, hydrostatic stress on the pellet due to PCMI and the diffusion coefficient of the gas atoms in the solid pellet. Based on empirical data, there appears to be an burnup-dependent temperature threshold for the release of fission gas by this coalescence mechanism. The implication of these observations is that fission gas is not released uniformly throughout the pellet. Because there is a large temperature distribution in the pellet, the majority of fission gas will be released in the higher temperature parts near the center of the pellet. As the temperature decreases toward the outer radius of the pellet, diffusion rates decrease, and it becomes less likely that fission gas will find grain boundaries and more likely that fission gas atoms will find like atoms and precipitate into intra-granular fission gas bubbles, effectively trapping the gas atoms from release to a free surface.

The process of fission gas release described above is shown in Fig. 3 as a function of relative radius. Near the center of the pellet, both the inter-granular and intra-granular bubbles are μm -scale in size and visible at typical magnifications used in a scanning electron microscope (SEM). Coalescence of these inter-granular bubbles produces the largest fission gas release. As the temperatures in the pellet decrease, there is so little diffusion that the bubbles remain nanometer size and are not visible at typical SEM magnifications. Most of the fission gas will be trapped in small intra-granular bubbles and there is little fission gas release. Near the periphery of the pellet, where the temperatures are the lowest and the burnup is the highest, due to enhanced creation of plutonium (via the self-shielding phenomenon), there is a transformation to the so-called high-burnup structure (HBS) (Garcia et al., 2012). The process of transforming from an intra-granular dominated microstructure to a high-burnup structure is still not well known. What is known is that a larger-grain structure with nanometer-size intra-granular bubbles transforms into a subgrain structure with large fission gas bubbles, where the large fission gas bubbles are surrounded by many subgrains. There is very little fission gas within the lattice of HBS; all the fission gas has been swept into the large fission gas bubbles during the transformation process. Interestingly, the transformation process that causes a large redistribution of fission gas does not result in a significant increase in fission gas release; like the intra-granular dominated region, little fission gas is released in the HBS region during normal operation. The release of fission gas from the HBS region will be discussed again in the context of design basis accident scenarios (see section “Design-basis accident performance”).

Cladding performance phenomena

During normal operations, the zirconium-alloy cladding must maintain integrity while being exposed to high-temperature, high-pressure water and bombarded by neutrons. The impacts of neutron bombardment are phenomena called irradiation hardening and irradiation growth. When the neutron bombardment is added to high-temperature and high-pressure conditions, irradiation creep occurs. Because all this happens within a water environment, corrosion of the cladding and the subsequent absorption of hydrogen from the oxidation process are important considerations for the performance of the cladding. These phenomena also affect the cladding response to applied stress due to coolant pressure, rod internal pressure and pellet-cladding mechanical interaction.

Irradiation hardening

Irradiation hardening and irradiation growth are material phenomena that are unique to a nuclear reactor environment. The high-energy neutrons that are emitted by the fission process will also interact with any surrounding structural material like the cladding. The high-energy neutrons collide with atoms in these materials creating what are called collision cascades—similar to the fission damage cascade represented in Fig. 2C. The atom that is hit careens through the lattice knocking other neighboring atoms out of their normal lattice sites and into interstitial sites. Each time an atom is knocked out of its lattice site, one vacancy and one interstitial defect is created. Subsequent collisions decrease the energy of the original atom making the next collision less energetic. Once the process stops, the original atom will be in an interstitial site and there is a path of vacancies and interstitials in the wake.

The majority of the new interstitial atoms will recombine with a neighboring vacancy, “healing” the defected structure. But some percentage of the defects become permanent (see Fig. 5A). These new point defects will diffuse through the lattice and eventually find other like defects to combine with to form plate-like defects called dislocation loops (either as vacancy or interstitial loops, see Fig. 5B) or deposit onto grain boundaries. The dislocation loops are effectively “hard” spots in the lattice that limit dislocation motion due to slip, which results in an increased yield strength. Over time as the fluence (or time-integrated neutron flux or neutrons/cm²) increases the dislocation loop population, the yield strength will increase (i.e., the material hardens), eventually saturating once the defect structure saturates. Another common quantity of neutron damage is displacements per atom (dpa), which is a measure of the number of atomic displacements each atom might receive upon an amount of neutron exposure. This saturation in irradiation hardness of the cladding occurs during the first cycle of operation when the neutron damage is ~ 3 dpa. A typical neutron damage in the cladding at the end-of-life in an LWR is between 10 and 12 dpa. What this means is that every atom in the cladding is displaced by a neutron 10–12 times in the lifetime of the fuel—highlighting how challenging the environment is in a nuclear reactor.

Irradiation growth and irradiation creep

The creation of point defects from neutron collisions followed by redistribution is also the source for irradiation growth and irradiation creep. Irradiation growth occurs because diffusion is faster along certain planes within the hexagonal closed packed (HCP) crystal structure. Interstitial atoms prefer the prism planes while vacancies prefer the basal planes (i.e., the energy barriers to atomic jumps are lower for interstitials in prism planes and for vacancies in basal planes). What this means is that the plate-like defects that are the source of irradiation hardening will preferentially form on certain types of planes in the crystal structure: interstitial loops on prism planes, which expand the lattice, and vacancy loops on basal planes, which contract the lattice. Thus, the differential diffusion causes the grains to change shape, expanding perpendicular to the prism planes and contracting perpendicular to the basal planes. If the orientation of grains was randomly distributed in space, the expansion and contractions would cancel each other out and there would be no observed macroscopic growth. However, because zirconium alloys have an HCP crystal structure, the normal processing routes for cladding cause most of the grains in the microstructure of the cladding tube to have a similar orientation; the material is said to have a texture. In the case of zirconium-alloy cladding, the texture is such that the material exhibits irradiation growth in the axial and transverse directions but contracts in the radial or thickness direction. Because irradiation growth occurs without any applied stress it is sometimes referred to as stress-free growth.

Irradiation creep is the time-dependent deformation of zirconium alloys under an applied stress that is below the yield stress during neutron radiation exposure; deformation at stresses beyond the yield (i.e., plastic deformation) is considered time-independent deformation. The most common analytic model for creep in unirradiated materials is the power-law model. In this model, the steady-state creep rate ($\dot{\epsilon}$) is $\dot{\epsilon} = K(T)\sigma^n$ where $K(T)$ is a temperature-dependent function (usually an Arrhenius-type function), σ is stress and n is the stress exponent. Typically, at low stress, n is 1, while at higher stress it increases to 4–7. The change in the exponent indicates a change in the dominant mechanism. At high stress (approaching the yield stress), the mechanism of irradiation creep is generally thought to be similar to out-of-reactor thermal creep. That is, creep at high stresses is driven by diffusion-controlled dislocation motion. However, because of irradiation hardening, irradiation creep in the high-stress regime is lower than thermal creep of unirradiated cladding.

In contrast, at low stress, irradiation creep is enhanced compared to thermal creep. In this stress regime the creep rate is linearly proportional to stress (i.e., $\dot{\epsilon} = K(T)\sigma$). The mechanisms of thermal creep in this stress regime might be referred to as grain shape-

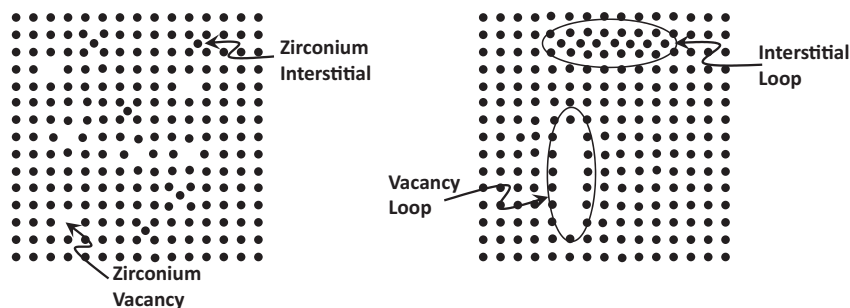


Fig. 5 Simplified representation of irradiation damage in a zirconium lattice. (A) Vacancy and interstitial point defects in a zirconium lattice and (B) vacancy and interstitial dislocation loop defects.

changing mechanisms; grains change shapes because of diffusion driven by internal stress distributions in the grain. For thermal creep, when diffusion is dominant along grain boundaries, the creep mechanism is referred to as Coble creep. When diffusion is through the grain volume, the mechanism is known as Nabarro-Herring creep. Irradiation creep at low stress is generally thought to be a grain shape-changing mechanism like Coble creep or Nabarro-Herring creep, but by a different mechanism—most likely a stress-enhancement of the irradiation growth mechanism. That is, stress enhances the growth of the interstitial and vacancy loops that are a result of irradiation damage.

Corrosion and hydrogen pickup

One of the most important performance criteria for LWR cladding is a general resistance to corrosion during in-reactor operation. Corrosion is the oxidation of the cladding via the decomposition of water. The chemical reaction is given by $\text{Zr} + 2\text{H}_2\text{O} \leftrightarrow \text{ZrO}_2 + 2\text{H}_2$. The zirconium oxide becomes a thin corrosion film on the outer surface of the cladding, while the hydrogen is either released into the coolant or absorbed into the cladding. Hydrogen pickup (absorption) by the cladding is typically a small fraction (0.1–0.2) of the hydrogen generated.

Corrosion resistance (i.e., resistance to oxidation) is important for two reasons. First, in the mechanical design of a fuel rod, the metal loss from corrosion must be considered, and second, the predicted fuel temperature must account for the resistance of the corrosion film to heat transfer. Thus, minimizing corrosion maximizes the cladding thickness available to resist internal gas pressure and minimizes the fuel centerline temperature at a given power (which maximizes the power associated with the prediction of the temperature to fuel melt).

Absorption of hydrogen (i.e., hydrogen pickup) is a side effect of the corrosion process. The problem with hydrogen is that it has an embrittling effect on the cladding. This occurs because of the formation of zirconium hydrides, which are generally considered a brittle phase. Another negative impact of hydrogen pickup is the volume increase in the formation of zirconium hydride. Concentration of hydrogen can thus cause a hydride bulge in the cladding. The volume change causes large internal cladding stresses and these locations become weak spots that are susceptible to failure (cladding breach) during normal power increases. Additionally, the volume change associated with hydrogen absorption can result in assembly bow if hydrogen occurs non-uniformly within the assembly. In BWRs, such bow by fuel channels has the potential to impede control rod insertion (Cantonwine et al., 2009).

More than anything, the development of cladding materials for LWR applications has been driven by the desire to improve corrosion resistance and hydrogen pickup. The whole development of the Zircaloy family of alloys in the 1950s was linked to improvements in both corrosion and hydrogen pickup; Zircaloy-2 for corrosion and Zircaloy-4 for both corrosion and hydrogen pickup. Decades of work on the effects of irradiation, cladding chemistry and microstructure are largely documented in the ASTM series on Zirconium in the Nuclear Industry. The result of these decades of work today is the excellent performance of cladding in both BWRs and PWRs. Looking ahead, one of the main advantages for potential coatings on zirconium-alloy cladding is the improved corrosion and hydrogen pickup performance.

Interactions between the pellet and cladding

The most important function of the fuel rod is to transfer heat generated in the fuel pellet to the coolant outside the cladding. The more effectively heat is transferred, the lower the fuel temperatures. At the beginning of life, the pathway of the heat is through the fuel pellet, across the ~ 0.1 mm gap between the fuel pellet and the cladding, through the cladding and out into the coolant. The initial thermal conductivity of the fuel is a function of the as-sintered density but changes with burnup because of lower thermal conductivity fission products and the development of fission gas bubbles. The burnup dependence of the thermal conductivity is often expressed as a function of swelling through its dependency on porosity. The result is a reduction in thermal conductivity with burnup that can be as large as 20% at end of life. Cracking and relocation introduce crack surfaces and internal gaps that can also degrade thermal conductivity especially when oriented circumferentially. The thermal resistance of the gap is a function of the gap size and the thermal conductivity of the gas atoms occupying the gap. At beginning of life, the thermal resistance of the gap is minimized by filling the rod with pressurized helium, which has a much higher thermal conductivity than air. However, the thermal resistance increases with the release of fission gas with increasing burnup but decreases as the gap closes via swelling of the fuel and creep down of the cladding. Like the degradation in thermal conductivity in the fuel with burnup, the changes in thermal conductivity of the gap are accounted for in the thermal analysis of the rod.

After the gap is closed during steady-state operation, further expansion of the pellet from fission-induced swelling causes the cladding diameter to increase with burnup. This type of interaction is referred to as Pellet Cladding Mechanical Interaction (PCMI). Because the fission-induced swelling rate is low, the strain capacity of the cladding is large, and failure of the cladding by PCMI during steady-state operation does not occur. However, there is a chemical interaction that can occur and cause failure during normal operation. When there is a localized power increase in an area of the reactor core, due mainly to pulling out a control rod, there is an immediate increase in temperature of the pellet and some fission gas is released. While the fission gas is mostly Xe and Kr, it also contains iodine, which is known to cause stress corrosion cracking in zirconium alloys. Thus, when the cladding stress is high enough and the power change and rate of power change is large enough, iodine stress corrosion cracking can cause fuel failure. Because this failure is always associated with a normal power change, it is generally referred to as a “duty” failure. In addition, to distinguish this phenomenon from PCMI, these duty failures are said to be caused by Pellet Cladding Interaction (PCI) or sometimes Pellet Cladding Chemical Interaction (PCCI). These failures typically occur at low cladding plastic strain and well below the 1% strain limit used for PCMI in anticipated transients (see section “PCMI failure”).

Duty failures were a significant issue in the 1970s and 1980s (mainly in BWRs) until an understanding of PCI was developed and mitigating actions were taken. BWRs were more susceptible to PCI because control rods are used more to control power through an operating cycle than in PWRs, which use boron in the primary coolant to manage reactivity through the operating cycle. Because of this there is more control rod motion during normal operation in BWRs than in PWRs. The mitigation of PCI has entailed both design changes to the fuel and operational changes at the plants. The development of barrier fuel, which adds a thin layer of less-susceptible, dilute zirconium alloy to the inside diameter of the cladding, significantly decreased failures attributed to PCI. In addition, operating guidelines have been developed for plant operations to better control the rate of local power changes when moving control rods; therefore, decreasing the amount of fresh iodine released and the stress applied to the cladding.

Anticipated transient performance

Fuel rods are designed and licensed to withstand the effects of transients during normal operation. These transients are sometimes called anticipated operational occurrences (AOOs). The frequency of performance limiting anticipated transients is low (~ 1 per reactor lifetime). There are two types of anticipated transients, either core-wide or localized. The most common core-wide anticipated transient is due to a pressurization transient. A pressurization transient is a rapid increase in coolant pressure that causes a short duration rapid power increase. The power increase can be several times the normal operating power, and the duration is typically 1–2 s, which is much shorter than the fuel rod thermal time constant (< 10 s for modern LWR fuel). With these time constants, it takes ~ 30 s to reach the equilibrium temperature distribution at the maximum power in the transient. Thus, the maximum fuel temperature at the center of the pellet will lag the power increase. This type of anticipated transient is often referred to as a fast transient and a generic example is shown in Fig. 6. Localized anticipated transients are typically due to operator actions taken to achieve specific operational targets. In BWRs, these anticipated transients are often referred to as rod-withdrawal errors (RWEs). Localized events typically have longer duration (larger than the thermal time constant) and smaller power increases than core-wide events such that the temperature increases coincidentally with the power.

By their nature, anticipated transients are more severe than power increases during normal operation. Depending upon operation, the peak power allowed in an anticipated transient is limited either by the fuel rod temperature or cladding strain. When temperature is limiting, the allowable peak power is defined such that the temperature at the center of the pellet is less than the melting temperature of the UO_2 fuel. When strain is controlling, the allowable peak power increase is limited to maintain a cladding strain increment below a failure threshold, which is typically 1%.

Performance phenomena in an anticipated transients

Two features of an anticipated transient event drive what phenomena need to be considered in predicting performance. Those two features are the change in power and the time over which the power change occurs. The fuel pellet expands with the increase in power mainly by thermal expansion, but at high burnups fission gas swelling may also be a contributor. The expansion of the pellet results in a corresponding loading of the cladding material by PCMI. The strain rates in the cladding during a transient are

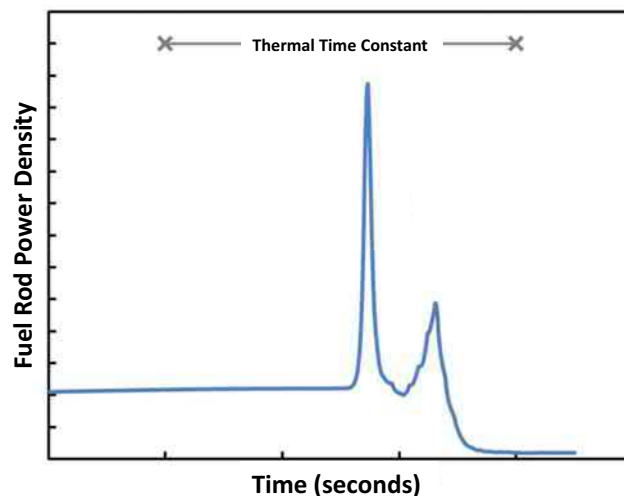


Fig. 6 An example of a fast transient where the event is shorter than the thermal time constant of the fuel rod—meaning the temperature increase in the fuel and the cladding will lag the power increase.

considered to be consistent with time-independent deformation mechanisms (i.e., plasticity). Transient response is determined by the duration of the transient relative to the fuel time constant. For fast transients, when there is a delay in the temperature response relative to the peak power, the full fuel thermal expansion corresponding to the peak power is not applied to the cladding because the equilibrium temperature distribution associated with the peak power is not reached.

Fuel melting

Although the phenomenon of melting is well understood, the conditions of melting in a fuel rod are unique because the heating is internal rather than external. Melting can only occur when the temperature exceeds the melting temperature, which for an anticipated transient is reached first in the center of the fuel pellet. The melt volume would be limited to this area of the pellet because of the radial and axial temperature gradients. That is, there would be no migration of the melted fuel radially or axially—meaning melting does not propagate. The axial temperature gradients are a result of the axial power distributions in the fuel rods. Thus, even if fuel melting occurred it would be confined to the center of the fuel pellets at the axial location of maximum power and would not interact with the cladding directly. The main fuel performance concern is the increased PCMI due to the increased volume of the small core of fuel that transforms from solid to liquid.

Thermal expansion

Thermal expansion of a material occurs because the bond length between atoms changes with temperature—increasing as the temperature increases. Because thermal expansion is assumed to be instantaneous relative to the fuel rod thermal time constant and transients are limiting at burnups where the pellet and cladding are in contact, the increase in pellet temperature results in a simultaneous increase in pellet diameter that must be accommodated by straining the cladding.

Fission gas swelling

Fission gas swelling during a transient event is considered secondary to fuel pellet thermal expansion. It is a secondary expansion mechanism because it isn't active at lower burnups and thus, is only a concern at higher burnups where the transient events are less limiting. Fission gas swelling has only recently been considered in fuel performance modeling.

The process of fission gas swelling in a transient is as follows. During a transient the temperatures in the pellet can increase to near the thermal limits of the fuel (i.e., the melt temperature). At temperatures near the melting point, the fuel is softened and deforms either by time-independent plastic deformation or by time-dependent creep deformation. The result is fission gas bubbles expand during the transient as much as the resistance of the cladding allows. The effect is for fission gas swelling to cause some increase in cladding strain in addition to that caused by thermal expansion of the pellet.

PCMI failure

The concept of pellet cladding mechanical interaction or PCMI was introduced in section “Interactions between the pellet and cladding” for normal operation. As discussed there, PCMI isn't active until there is gap closure between the pellet and the cladding ($\sim 25\text{--}30$ GWd/MTU at normal operating powers). During steady-state operation, the pellet increases in diameter from swelling and thermal expansion while the cladding creeps down because the differential pressure across the cladding causes irradiation creep. In an anticipated transient, the power increase causes an increase in fuel temperature, which causes the pellet diameter to expand. If there is still a gap between the pellet and the cladding in steady state, this temperature-induced expansion due to the anticipated transient will eventually close it and initiate PCMI. If the gap has already been closed during steady-state operation, the thermal expansion of the pellet in the anticipated transient causes the cladding to strain accordingly. The strain rates in the cladding during an anticipated transient are $> 10^{-4}/\text{s}$, meaning the deformation is time-independent plasticity. Thus, failure by PCMI is predicted to occur when the predicted strain in the cladding reaches a strain limit; 1% being the most common limit used for licensing assessments.

Design-basis accident performance

Anticipated transient events are events that might happen once in the lifetime of a plant, while DBAs are not expected to occur but could have a significant consequence if they did occur. Because of the potential severe consequences, these events are considered in the design of a reactor and the licensing of a core design to ensure such an event would not impact the health and safety of the plant operators or the public.

There are two specific types of design-basis accidents that are accounted for explicitly in licensing of a fuel assembly or an operating plan for a nuclear power plant. The first is the LOCA. This event is postulated as a removal of a section of large pipe (the so-called “double-end break”) causing the loss of coolant at a higher rate than can be replaced by the coolant make-up system. The accident at the Three-Mile Island plant in the United States, which caused the core to melt, was a LOCA-like event ([Skillman and Rempe, 2021](#)). The second is the RIA. The limiting RIA occurs when a control rod is suddenly removed from a reactor at zero power. In this accident event, prompt criticality occurs over milliseconds in a local region of the core, potentially shattering fuel rods and creating a pressure pulse that, if unaccounted for, can damage the reactor internals or pressure vessel. The SL-1 accident in the early 1960s, which killed three US Army soldiers, was an RIA event.

These events are addressed in reactor and core design in accordance with regulatory requirements to ensure core coolability is maintained and off-site dose is acceptable. Fuel failures may occur but must be accounted for in licensing analyses. However, licensing analyses are often performed to show fuel failures do not occur, to ensure there is significant margin against harmful radio-nuclide release. The following sections provide a description of the fuel performance phenomena that are expected in the LOCA and RIA events and considered in analyses.

LOCA

The time-temperature history for the cladding in a LOCA event is illustrated in Fig. 7. The first thing that happens in a LOCA is that the reactor pressure drops, which changes the coolant environment from water in the PWR and boiling film in the BWR to a steam environment resulting in less effective heat transfer between the cladding and coolant. The reactor then scrams (that is, automatic insertion of control rods shuts down the reactor). When the reactor power decreases, the temperature in the fuel follows. After about 10–20 s, the less effective heat transfer at the cladding surface is overwhelmed by the decay heat in the fuel causing the fuel temperatures to again increase. The rate of temperature increase in each rod will depend on the extent of the depressurization and the rod power prior to the event—with rods operating at higher power having more decay heat. The effect is that the cladding temperature increases well above its normal operating temperature while the peak temperature in the pellet will remain below normal operating conditions (except for in an accident severe enough to result in core melting). Therefore, the performance phenomena of interest in a LOCA event are related to how the cladding responds to the fission gas pressure and temperature from the interior of the rod (fuel rod ballooning and burst) and the low-pressure steam on the exterior of the rod (high temperature oxidation). In addition, for fuel at high burnup there is the concern of potential fuel fragmentation and relocation, which could result in fuel expulsion if burst occurs. After coolant has been added back into the reactor vessel, the temperatures start to decrease. Eventually, the cooling environment transitions back from steam to a boiling condition and the temperature drops dramatically, quenching the fuel. The limitations on high-temperature oxidation ensure the cladding maintains ductility to withstand the thermal stresses that occur during this quenching phase of a LOCA—avoiding potential breakage of the fuel rod into pieces and loss of rod geometry.

Performance phenomena in a LOCA

Fuel rod ballooning and burst

As the temperature of the cladding increases above the operating temperature, the lattice defects that caused irradiation hardening start to anneal out. That is, the dislocation loops created during irradiation “dissolve.” As the cladding anneals, the mechanical properties revert to those of the unirradiated condition. Yield strength and creep resistance decrease while ductility increases. The increasing temperature also causes the internal pressure in the fuel rods to increase. The result of these changes is that the cladding diameter increases at strain-rates more consistent with plastic deformation rather than creep ($> 10^{-4}/s$). Because of variations in the temperature, cladding thickness and axial burnup in the rods, there will likely be a local area where the cladding diameter increases at a higher rate than other regions. This results in a region of cladding ballooning. Then, like in a tensile test, the cladding will burst when the ultimate tensile strength is reached. After the cladding bursts, the deformation of the cladding stops at all elevations because the internal pressure driving the deformation is released.

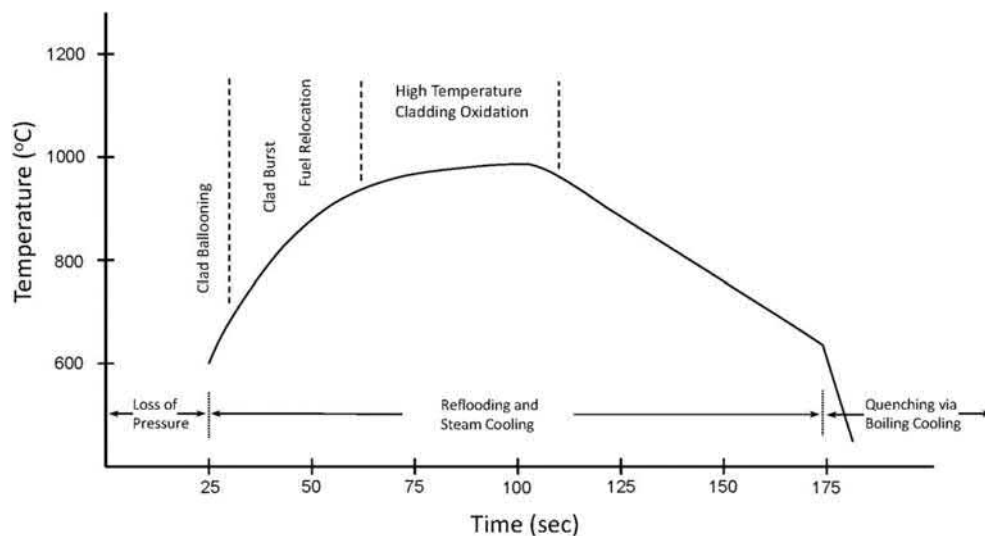


Fig. 7 Representation of the cladding temperature as a function of time in a LOCA event; the heat-up rate, maximum temperature, and time will all depend on the actual event and the operating power of the rods prior to the event.

Fuel fragmentation and relocation

It is expected that during normal operation the fuel pellets are cracked into wedge-like pieces as illustrated in Fig. 4A. In addition, when fuel cracking occurs during the initial power rise, cracked fuel can relocate to fill the gap between the fuel pellet and cladding (relocation). Fuel fragmentation and relocation during the LOCA are extensions of these phenomena.

When the cladding diameter increases during a LOCA, a gap between the fuel and the cladding is created. This decreases the constraint on the fuel, decreasing friction between the fuel fragments and allowing potential relocation. Initially that relocation would be limited to the gap—meaning relocation occurs in the radial direction (similar to what might happen during the initial rise to power). When the size of the gap becomes larger than the size the fuel fragments, relocation can occur axially within the gap. This axial relocation is of greater concern than radial relocation because it can increase UO_2 mass at a given elevation—changing the neutronic characteristics of that location in the core. The concern is especially relevant where ballooning occurs because of the significant increase in gap size that fuel can occupy.

While axial relocation is a concern, it doesn't pose a significant safety hazard. The more significant concern is the potential expulsion of fuel into the coolant upon cladding burst, which could result in relocation of the fuel throughout the reactor system. The size of the fuel fragments relative to the size of the burst failure is an important factor when considering potential expulsion. It is now known that at high burnup, there is a post-LOCA fuel fragmentation mechanism that appears to be driven by fission gas release rather than thermal stresses, which drive fuel fragmentation under normal conditions. The regions of high-burnup structure (HBS) discussed in section “Fission gas release” are particularly susceptible to this mechanism, and could result in a high-energy expulsion of powder-size fragments through the burst cladding opening. In addition, there is some indication that this fuel fragmentation phenomenon may be triggered by the decrease in constraint of the fuel when the rod internal pressure drops with clad burst, which allows the full force of the fission gas pressure to act on the internal ligaments between fission gas bubbles. The consideration of this new fuel fragmentation mechanism has been the topic of potential new regulatory requirements for high burnup fuel.

High temperature oxidation

At the highest temperatures in the LOCA, high temperature oxidation is a concern. The potential for oxidation is limited by fuel design and core design. Fundamentally, high temperature oxidation and corrosion are the same phenomenon; the chemical reaction, $\text{Zr} + 2\text{H}_2\text{O} \leftrightarrow \text{ZrO}_2 + 2\text{H}_2$, is the same in both cases. The difference is the gaseous phase of the high-temperature H_2O in a LOCA oxidizes cladding at a much higher rate than does the liquid H_2O phase at normal operating temperatures; in other words, the oxidation rate in a LOCA is significantly higher than in normal operation.

High temperature oxidation can impact the performance of the cladding in two ways. First as oxidation occurs cladding is consumed. Effectively, the load-bearing thickness of the cladding is decreased as oxidation occurs. Given the difference in density between ZrO_2 and Zr metal alloys, the cladding thickness decreases 63% of the oxide thickness. That is, if the oxide thickness is 10.0 μm , 6.3 μm of metal was consumed. Second, because of the affinity of zirconium for oxygen, significant amounts of oxygen are absorbed into the metal lattice below the oxide film. Oxygen strengthens and embrittles zirconium alloys and thus can significantly decrease ductility. After burst occurs, high temperature oxidation on both the outside and inside surfaces of the cladding are considered.

To minimize the change in ductility, the specific performance metric often used by regulators is the percentage of the total cladding thickness consumed into the oxide film and the oxygen embrittled zone (termed the Equivalent Cladding Reacted, or ECR). Traditionally the ECR is calculated for a postulated LOCA based on correlations of steam oxidation data like Baker-Just (Baker and Just, 1962) and Cathcart-Pawel (Cathcart et al., 1977) and compared to regulatory limit like 17% ECR, which has been traditionally used in the United States. The expectation is that, by maintaining the calculated ECR to less than the regulatory limit, ductility in the cladding is maintained to withstand the thermal stresses that occur during the quenching stage, avoiding fuel rod fracture and rearrangement. Oxygen embrittlement is implicitly addressed by the ECR limit.

RIA

The RIA is a very short event (less than 100 ms) and occurs when a single control rod is quickly removed from a reactor core at zero power. In a PWR, the control rod is ejected when the drive mechanism at the top of the reactor vessel malfunctions and pulls the rod out or allows vessel pressure to push the rod out. In a BWR, the control rod is envisioned to become stuck in the core, subsequently disengage from the drive mechanism at the bottom of the reactor vessel and then later releases, dropping downward out of the core. In both cases the removal of the control rod causes a local prompt criticality event that is terminated by the Doppler reactivity feedback phenomenon. The control rod is then reinserted by a scram. The associated change in fuel enthalpy for both the PWR and BWR RIA events are provided in Table 1 (NEA, 2010) with the range in event times, and a representation of the fuel and cladding temperatures evolution is provided in Fig. 8 (USNRC, 2015).

The response of the fuel rod in an RIA has similarities to both the anticipated transient and the LOCA but over a much shorter time period. The power pulse causes the fuel temperature to increase rapidly. Like the anticipated transient there is a power increase with a subsequent increase in fuel temperature that results in PCMI. Like the LOCA, the local coolant environment changes to steam (a less effective heat transfer condition than pressurized water in a PWR or boiling film in a BWR), which subsequently causes the cladding temperature to rise with the fuel temperature. But because the coolant is never lost and reactivity feedback mechanisms terminate the power excursion, the event ends quickly, and the temperatures start to decrease. Thus, when considering potential failure mechanisms in an RIA, PCMI is possible as the fuel temperature rises and cladding burst is possible when the cladding is at the maximum temperature and the fuel temperature is decreasing.

Table 1 Estimated pulse widths and change in fuel enthalpies.

Reactor accident scenario	Pulse width (ms)	Maximum change in fuel enthalpy (J/gUO ₂)
PWR, rod ejection accident at hot zero power	25–65	80–250
BWR, rod drop accident at cold zero power	45–75	130–450

Modified from Nuclear Energy Agency (2010) Nuclear fuel behavior under reactivity initiated accident conditions, NEA-6847. OECD.

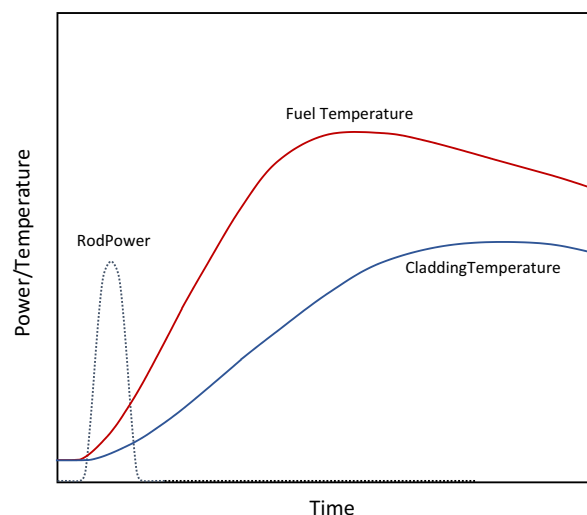


Fig. 8 Representation of the power change in a rod during an RIA and the subsequent impact on fuel and cladding temperature. Modified from US Nuclear Regulatory Commission (2015) Technical and regulatory basis for the reactivity-initiated accident acceptance criteria and guidance. Revision 1, NRC Memorandum, March 16, 2015, ML14188C423.

While the RIA could be analyzed considering the fuel performance phenomena explicitly, the regulatory requirements are universally based on empirical fuel rod test data that are used to define limits based on the applied fuel enthalpy. However, regulators have considered different fuel performance failure mechanisms by defining different fuel enthalpy limits for PCMI failure and for cladding burst. These limits are used analytically to show that failure doesn't occur or to quantify how many failures do occur for subsequent dose calculations, which is ultimately the concern in DBAs.

Fuel performance analyses

The thermal-mechanical response of fuel rods in normal operating and accident conditions is complex. The transfer of heat from the fuel rod to the coolant involves the heat generation within the pellet by nuclear fission, the transfer of that heat through the pellet, across the pellet-cladding gap, through the zirconium based cladding and then from the cladding outer surface to the coolant. Modeling this process over the entire lifetime of the fuel rod must account for material property changes in the pellet and cladding, geometry changes due to gap changes, and pellet diameter changes due to thermal expansion, densification and swelling. The heat transfer process is especially complicated by the reduction of pellet thermal conductivity and the evolution of gap conductance with burnup.

To predict the thermal-mechanical response of fuel rods, fuel vendors and regulatory agencies developed fuel performance models and codes that address the performance mechanisms noted above. These codes generally include a mechanics response module and a thermal response module. The mechanics model is often based upon the finite element method and the thermal model is generally based upon either the finite element or finite difference method. These codes generally address both steady-state and transient (AOO) response.

The vendor codes are used to assure safe operation by confirming compliance with licensing requirements as specified by the SAFDLs (e.g., the US NRC SRP 4.2) and to assure reliable and economic operation. The regulatory codes are used by regulatory agencies such as the US Nuclear Regulatory Commission to independently confirm vendor analyses and safety conclusions. For these reasons, codes are extensively qualified by (1) material properties tests on unirradiated and irradiated specimens and (2) integral on-line and post-irradiation examination (PIE) fuel performance data, including measured fission gas release.

After thermal-mechanical codes have been qualified and approved by regulators, vendors will use them to define steady-state operating limits with constraints on overpowers to account for anticipated transients. The steady-state operating limits are usually

expressed as power versus burnup or power versus time, and overpower limits are generally expressed in terms of percent overpower relative to the steady-state operating limit. Some vendors use a single overpower limit to protect against temperature and strain violations, while others specify both a thermal overpower limit to protect against temperature violations and a mechanical overpower limit to protect against cladding strain violations.

As discussed in section “**Design-basis accident performance**,” fuel rods are designed to meet different requirements during DBAs. The RIA is a fast energy deposition event that can be analyzed using a suitably-equipped thermal-mechanical code, but is often conservatively analyzed as an adiabatic event without a detailed prediction of the thermal-mechanical phenomenon described herein. Rather the maximum fuel enthalpy of the event is predicted using approved nuclear core/plant models and compared to the empirically based regulatory limits.

The LOCA is slow relative to the RIA and most anticipated transients, and the concerns are different. During the LOCA event, fuel rod internal pressure increases due to fission gas release and the cladding is subjected to high temperature due to limited heat transfer from the cladding outer diameter to the steam environment. The major concerns are cladding ballooning and possible perforation. Most thermal-mechanical codes are not capable of predicting large cladding deformation response that are typical of ballooning or do not have fission gas release models that are valid for the fuel annealing conditions in a LOCA. Therefore, LOCA analyses are generally performed with codes that predict the plant responses of such events. The event conditions are used as inputs to the high-temperature oxidation correlations that predict ECR, which is then compared to a regulator limit, or the event conditions are compared to empirically based cladding burst limits.

Like the thermal-mechanical analysis of steady-state and anticipated transient conditions, the intent of plant-based RIA and LOCA analyses is to confirm the licensing requirements for these design basis accidents are satisfied.

Summary

The performance requirements for LWR nuclear fuel are challenging because the LWR operating environment is challenging. The fuel must withstand elevated operating temperatures, a highly corrosive environment, the damaging effects of uranium fission and neutron irradiation, and stresses from both external and internal pressure and PCMI. This necessitates understanding the performance phenomena of both the UO_2 fuel pellet and the zirconium alloy cladding under all operating, transient and accident conditions so that performance models can be utilized to show compliance to regulatory requirements. To learn more about the details of the issues discussed herein, a list of further reading has been provided.

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Irradiation Performance: Fast Reactor Fuels

Colby B. Jensen and Nicolas E. Woolstenhulme, Idaho National Laboratory, Idaho Falls, ID, United States

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Glossary

MOX Mixed oxide of uranium and plutonium, synonymous with (U,Pu)O₂.

U-Zr, U-Pu-Zr Fuel alloys of uranium, plutonium, and zirconium.

UO₂, (U,Pu)O₂ An oxide of uranium and/or plutonium.

UC Uranium monocarbide.

UN Uranium mononitride.

Abbreviations

BDBA Beyond design basis accident

DBA Design basis accident

dpa Displacements per atom

EBR Experimental breeder reactor

F/M Ferritic/martensitic steels

FCCI Fuel-cladding chemical interaction

FCMI Fuel-cladding mechanical interaction

FFTF Fast flux test facility

JOG Joint oxide gain

LOF Loss of flow

LWR Light water reactor

NRC Nuclear Regulatory Commission
 O/M Oxygen/metal ratio
 ODS Oxide dispersion strengthened
 SAFDL Specified acceptable fuel design limits
 SFR Sodium-cooled fast reactor
 TOP Transient overpower
 TREAT Transient reactor test facility

Introduction

Fast-spectrum nuclear reactors (or “fast reactors”) are distinguished from thermal-spectrum reactors by their lack of neutron moderating materials, enabling high-energy neutrons sustain the fission chain reaction. Because fissions are less likely to occur at higher neutron energies, fast reactors require higher neutron populations than thermal spectrum reactors in order to achieve the same core power density. The abundance of high-energy neutrons is useful for breeding fissile isotopes from fertile isotopes relative to the fission rate and for splitting minor actinides into shorter-lived isotopes to greatly reduce the radiotoxicity of spent nuclear fuel. However, this high quantity of higher-energy neutrons, however, also increases neutron-induced atom displacement damage to structural materials. Crucial components such as reactor pressure vessels can be designed to reside far enough from the core to reduce incident neutron intensity and, thus, fast neutron damage, but some structural materials (e.g., cladding and coolant ducts) must reside very near the fuel itself and thus be able to survive the challenges of fast neutron damage and other harsh environmental conditions.

Fast reactor physics favoring the breeding of fissile isotopes have historically been emphasized as a major advantage point—especially at a time when uranium-rich natural resources were not so well-known—and they remain of interest in countries with fewer uranium resources relative to population demands. The physics conducive to minor actinide splitting, as well as compatibility with plutonium fission, continue to be notable for creating more advantageous fuel cycles and disposition strategies (Wigeland et al., 2014).

While several different types of fast reactor designs are likely viable, most experience to date has been gained via cores cooled by liquid sodium (see Heidet, 2021; Hejzlar, 2021). The remainder of this article focuses primarily on the irradiation performance of the most common fuel systems used in sodium-cooled fast reactors (SFRs) as well as modern reactor designs. Other fast reactor systems are discussed elsewhere: lead-cooled (Alemberti, 2021), gas-cooled (Halata, 2021), and molten-salt-cooled (Krepel and Kramer, 2021). These fuels include oxide fuels—(U,Pu)O₂ and UO₂—and metallic alloy fuels—U-Zr and U-Pu-Zr. In this article, light water reactor (LWR) fuel technology is used as a known point of reference. For more information on fast reactor technology see (Waltar et al., 2012).

Fuel design

Overview

Fast reactor fuel technology is dominated by relatively simple designs consisting of a cylindrical nuclear fuel column encapsulated in a cladding tube hermetically sealed at both ends. Though appearing simple, this fuel design was optimized over decades of R&D, and it contains a host of subtle features enabling reliable long-term performance in environments that are extremely challenging for material performance. For most SFR designs, the residence time of fuel in operation is 1–3 years though the fuel in some designs is proposed to last decades, even the entire lifetime of a reactor plant. Core reactivity management strategies involving long irradiation lifetimes and high fuel burnup are achievable due to the higher initial fissile loading and fissile isotope breeding in fast reactor fuel systems. Thanks to the efficient heat removal capability of liquid metal systems employed in fast reactors, a high-power density is also achievable compared to water-cooled systems. Table 1 offers a comparison of typical SFR and LWR performance conditions, for perspective on the dramatic contrast in the fuel performance demands for each.

Table 1 Representative LWR and SFR service conditions illustrating performance comparisons relevant to fuel considerations.

	Core power density (W/cm ³)	Peak power density (W/cm ³)	Peak linear heat generation rate (W/cm)	Peak heat flux (W/cm ²)	Peak rod average burnup (operation, demonstrated) (GWd/t)	Neutron flux (n/cm ²)	Typical cladding irradiation damage (dpa)	Coolant temperature (K)
LWR	100	400	250	80	62, 100	10 ¹⁴	10–15	573
SFR	300	2000	450	260	90, 200	10 ¹⁶	100–150	873

Design limits

The overarching goals of fuel design are to achieve economic fuel cycle performance and reliable operation in service, and to ensure that offsite dose limits for all reactor design-basis conditions (including postulated accidents) are minimized and maintained within known margins. More practically, fuel designs are evaluated to identify every potential degradation mechanism and quantify the failure thresholds for relevant operating parameters. Predictive fuel performance models play a crucial role in this effort as the ultimate product of R&D programs for demonstrating acceptable fuel behavior has an adequate margin to fuel failure (see Section “**Fuel performance modeling**”). (Note that fuel failure includes not only fuel cladding breach but also other conditions such as unacceptable fuel rod or fuel assembly deformation.)

To illustrate how design limits for fuel might be applied it is helpful consider the specified acceptable fuel design limits (SAFDLs) established by the U.S. Nuclear Regulatory Commission (NRC). Adherence to these SAFDLs ensures acceptable performance of the fuel under normal operational and unanticipated transient conditions. For LWR fuel systems, the NRC established fuel damage criteria in its Standard Review Plan (NUREG-0800), Section “**Mid-to-high burnup evolution (<10 at% burnup)**”. The primary purpose of these criteria is to ensure that the fuel system is:

- 1) Not damaged as a result of normal operations and anticipated operational occurrences (e.g., any transient power or coolant condition)
- 2) Never damaged to the point that control rod insertion is prevented when needed
- 3) Conservatively assessed in regard to the number of failed fuel rods in accidents
- 4) Always maintained coolable (“coolability”).

For fast reactor fuels, no such explicit SAFDLs exist under a unified standard such as that developed by NRC for LWR fuels. However, many SFRs have been built and operated using defined limits that serve similar purposes. Recent work has proposed revisions for specific application to SFR fuels (Belles et al., 2017). For a better understanding of fuel performance under the most relevant conditions, note the following four defining criteria, which are common to the oxide and metallic fuel systems described in this article.

1. Limit cladding strain/stress—The cladding is the first barrier against fission product release and criteria are imposed to ensure cladding integrity. Frequently, such criteria are associated with concerns related to fuel-cladding mechanical interaction (FCMI) and over-pressurization due to fission gas release.
2. Maintain an adequate temperature margin to fuel melting—For various reasons, this criterion is common to all fuel systems, as explained in greater detail in coming sections. In short, ensuring an adequate temperature margin to fuel melting has been sufficient to avoid quantifying the uncertainties in molten fuel behavior and the resulting consequences.
3. Limit fuel-cladding chemical interaction (FCCI)—FCCI can threaten the cladding, degrading its structural integrity by effectively reducing the load-bearing thickness of the cladding wall. (This is also referred as cladding “thinning” or “wastage.”)
4. For certain accident conditions, post-failure behavior must preclude loss of coolability. This encompasses a wide range of considerations—primarily, maintaining fuel and core component geometries within the expected thresholds.

It is helpful to consider the complexity of fuel performance in the context of these criteria. One significant implication is that fuel performance must be understood in order to ensure cladding integrity while efficiently transferring heat to the reactor coolant. Therefore, the complexity of fuel performance is sometimes simplified to interpret its impact on cladding performance as a priority.

Common SFR fuel designs and key properties

Background

Several fuel systems and designs have been considered, tested, and developed to varying levels of maturity. Four primary fuel systems have been developed to significant maturity: oxides, metal alloys, carbides, and nitrides (Crawford et al., 2007; Delage et al., 2013). However, international experience is dominated by the use of oxide and metallic fuels, thus they are the focus of the detailed performance discussion in this article.

Mechanical design and function

The existing knowledgebase for fast reactor fuels is dominated by fuel designs in which a reactor core is composed of many assemblies (or “subassemblies” when related to the core assembly) of fuel pin (or rod) bundles. Frequently these fuel pin bundles are shrouded within a duct that has fittings attached to each end for mechanical interfaces and coolant inlet/outlet paths. Within the duct, the pins are arrayed on a hexagonal pitch and typically spaced apart using wires wrapped helically around the cladding of each pin. The vast majority of fuel designs have featured relatively similar design specifications.

As with traditional LWR fuels, each fuel pin is composed of a cladding tube that comprises an outer structural shell, along with an internal volume filled with the fissile fuel. The cladding and fuel are cylindrically shaped. In ceramic designs, the fuel typically takes the form of a stack of individual pellets (like those in LWR fuel rods), while metallic fuel designs generally employ monolithic columns or “slugs.” The required number of slugs is dependent on the fuel column length and fabrication method limits on slug length, attempting to minimize the number of slugs. Fuel smeared density (the areal fraction of cladding’s internal cross-section that is comprised of fuel) is a crucial parameter for any fuel type to achieve acceptable performance at high burnups and under transient conditions. The free volume within the fuel pin (i.e., a plenum located above or below the fuel column, the annular gap between the

fuel and the cladding, and in some designs a central hole in the fuel) is used to accommodate fuel thermal expansion, irradiation-induced swelling, and swelling due to fission gases and solid fission products. Metallic fuel designs include a sodium fill to provide good thermal transport from the fuel to the cladding. Annular fuel pellets (containing a central axial hole) are now common in oxide fuel to provide the desired smeared density while also producing benefits in terms of accommodating FCMI and improving transient behaviors (including providing a wider margin to fuel melting). Annular geometries for metallic fuel have received growing interest as a design solution that allows for eliminating the sodium from the pin while still maintaining good heat transfer between the fuel and cladding. Despite its performance-enhancing design features in metallic fuels, the in-pin sodium requires unfavorable disposal considerations when used in a once-through fuel cycle (e.g., U.S. regulations restrict the amount of reactive metal that can be disposed of with radioactive material in a geologic repository) (Fig. 1).

In some cases, fast reactor fuels include high concentrations of transuranic actinides (recycled from LWR or fast reactor fuel). These are typically referred to as minor actinide (MA)-bearing fuels, and their express purpose is to leverage the fast neutron spectrum in order to transmute the long-lived MAs into isotopes with significantly shorter half-lives (by a factor of approximately 300×). For a detailed overview of MAs' impact on fuel performance, see OECD-NEA, 2014.

Fuel types and design

Metallic and ceramic fuels have both been proven viable by in-reactor experiments and use in prototype plants. In both metallic and ceramic designs, the addition of plutonium is known to create some performance differences. Besides the in-reactor performance of these fuel systems (including safety-related performance), other engineering tradeoffs exist in terms of fuel supply/fabrication and post-irradiation processing in recycle and disposition (Wigeland and Cahalan, 2009).

Mixed oxide (MOX) fuels (U,Pu)O₂ have become the dominant fuel used in fast reactors internationally, thanks to established fabrication techniques, demonstrated performance, and operational experience mostly established through the LWR industry in the 1950s and 1960s (Waltar et al., 2012). Despite some disadvantages (e.g., poor thermal conductivity and lower density), oxide fuels benefit from having no allotropic changes, a high melting point, and low swelling rates. A summary of MOX fuels' important performance-related properties is found in Table 2. Though fuel designs vary, typical fuel parameters include a low oxygen/metal (O/M) ratio to mitigate fuel chemical interaction with the cladding. Fuel smeared densities typically range from 80–90%, with lower smeared densities showing greater resilience at higher burnups and under transient conditions. To achieve the desired smeared density, common fuel designs include both solid and annular (central hole) fuel pellets, with the annular pellets showing notable improvement in FCMI behaviors, especially during transients. Fuel pin diameters have typically been around 1/2–2/3 those of LWR rods with fissile fuel columns ~1 m long. Over decades of development, the fuel pin plenum was enlarged to incorporate an increasingly large free volume for accommodating increased fission gas release at higher burnups. Table 3 presents an overview of representative fuel design parameters relevant to discussions in later sections.

For metallic SFR fuels, an irradiation database exists to support fuel licensing under performance limits comparable to those of oxide fuel (Walters et al., 2011). The important characteristics of metallic fuels are (1) relatively high thermal conductivity and material density, as shown in Table 2 (see Novascone (2021) for more information on metallic fuel properties); (2) relative insensitivity

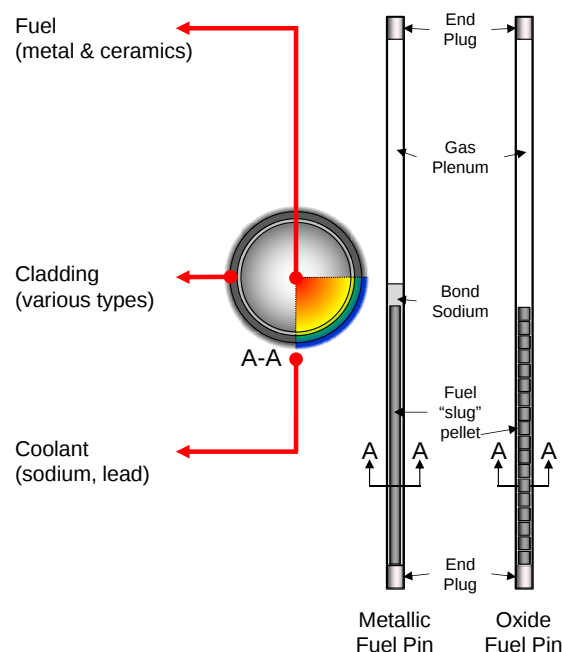


Fig. 1 Simplified overview of fuel pin designs typically used for liquid metal cooled reactor designs.

Table 2 Important performance properties of MOX and metallic fuels.

	MOX (20% PuO ₂)	Metal (U-20Pu-10Zr)
Theoretical density at 293 K (g·cm ⁻³)	11.1	16.2
As-fabricated density at noted temperatures (g·cm ⁻³)	10.3 → 9.70 ^a	15.3 → 15.1 ^b
Melting temperature (K)	3050	1350
Thermal conductivity (W·m ⁻¹ ·K ⁻¹)	3.5 → 2	20 → 25
Specific heat capacity (J·kg ⁻¹ ·K ⁻¹)	300 → 450	170 → 150
Thermal expansion (10 ⁻⁵ ·K ⁻¹)	1.0 → 2.0	1.7 → 2
Young's Modulus (GPa)	225 → 150	60 → 10
Centerline temperature (K), homologous temperature (T/T _{melt}) at 40 kW/m	~2373, 0.8	~1073, 0.8

Where applicable, values given are for typical fuel temperature distributions during operation: the MOX fuel range is 873–2373 K; the metallic fuel range is 873–1073 K.

^a95% theoretical density.

^bDensity accompanying injection casting; recent fuel extrusion shows about 3.2% increased density (very near theoretical density).

Adapted from Wigeland R and Cahalan J (2009) Fast reactor fuel type and reactor safety performance. In: Proceedings of the GLOBAL 2009 congress—The Nuclear Fuel Cycle: Sustainable Options and Industrial Perspectives, Paris, France, 2009, (p. 567).

Table 3 Representative fuel design parameters included for reference (Crawford et al., 2007; Waltar et al., 2012; Guerin., 2012).

	Mixed oxide	Metal
Nominal composition	(U,Pu)O ₂	(U-Pu-Zr)
%Pu	15–30	0–26
O/M ratio	1.95	n/a
Fuel density (% theoretical)	95	100
Fuel height (mm)	100	100
Fuel diameter (mm)	5–7	5–7
Central hole diameter (mm)	0–2	0–1.5
Cladding outer diameter (mm)	6–9	6–8
Smeared density (%)	80–85	75
Plenum/fuel volume ratio	1	1.5
Fuel-cladding annular gap fill	He	Na

to most design variables, (3) reprocessing and refabrication options that demonstrate superior security and economic benefits as compared to other fuel types, and (4) inherent attributes that reduce the complexity of the core design and control systems. Metallic fuel presents complex allotropic evolution dependent on alloy composition and temperature, which becomes a dominant theme in its performance. The modern reference alloys are U-Zr and U-Pu-Zr, with Zr concentrations of up to ~10 wt.% to improve chemical compatibility with steel cladding alloys, raise the fuel solidus temperature, and provide a workable fuel fabrication temperature. Typical fuel smeared densities are ~75%, virtually eliminating the fuel-swelling-induced failures seen in early metallic fuel designs with higher smeared densities. Fuel pin diameters are varied to optimize different performance parameters, but (as with MOX fuel) are roughly 1/2–2/3 the diameter of LWR rods with fissile fuel columns ~1 m long. As with oxide fuels, metallic fuel designs include a large plenum to reduce internal pressurization that accompanies the desired high fission gas release. Typical fuel performance properties and design parameters for both fuel types are given in Table 3.

Fuel assembly structural material performance

Most irradiation experience involving fast reactor fuel cladding, flow ducts, and other components that must reside in the reactor core during irradiation (i.e., wire wraps, pin end caps, axial reflector blocks, etc.) largely pertain to stainless-steel alloys (Fe-based alloys with Cr, Ni, and other metals). Compared to zirconium-based alloys (used in LWR plants because of their relatively lower neutron capture), stainless-steel alloys are more attractive for SFR applications because of their improved strength at high temperatures at which the abundant high-energy neutrons in fast reactors are less susceptible to parasitic neutron capture in the cladding. Due to their essential functions, much of the focus of alloy development has been on cladding tubes and the flow ducts surrounding the pin bundles. Cladding materials serve to retain fission products in the fuel pin, isolate fuel materials from the coolant, restrain fuel swelling, and retain fission gas pressure in a coolable geometry. The ducts serve to maintain the structural functions of the fuel

assemblies and establish a well-defined coolant flow path. Their stable geometric behavior is important for thermal hydraulic functions and to enable fuel removal, shuffling, and reinstallation. Conveniently, the primary corrosion and hydrogen pickup issues that drive LWR alloy development do not apply to stainless-steel alloys in sodium cooled plants. However, both the cladding and ducts are challenged beyond LWR conditions with higher temperatures and fast neutron damage rates. As with LWRs, fast reactor cladding is also challenged by chemical and mechanical interaction with the fuel material.

Apart from performance phenomena relevant to fuel-cladding interaction, (covered in the next section), the principal challenge for fast reactor structural materials is dimensional stability at high fast-neutron damage. A tremendous number of irradiations have been performed on conventional stainless-steel austenitic alloys 304 and 316, which show excessive swelling at irradiation doses of above 50 displacements per atom (dpa) (in solids, dpa is used to measure the irradiation effect on material properties). In stainless-steel alloys the fundamental issue of void swelling drives dimensional changes in which kinetic collision with fast neutrons displaces metal atoms from their crystallographic lattice. The process creates atom interstitials and vacancies (point defects), as well as dislocations in the atomic lattice. Most of these defects self-annihilate quickly at reactor temperatures, though some evolve and interact with one another. Swelling results from the preferential attraction of interstitials towards dislocations. Excess vacancies become supersaturated and coalesce, initiating void nucleation and growth at reactor operating temperatures (Chen, 2013). Alloy modifications such as titanium additions and cold working extend this incubation period in advanced austenitic alloys (e.g., D9, 15-15Ti, PNC 1520), making them viable for doses of up to around 160 dpa, depending on the specific alloy type and design details (Cheon et al., 2009). Due to these limitations austenitic alloys have not been preferred for applications intended to support very high-burnup fuels with cladding doses of 200 dpa and beyond.

More recent SFR cladding designs have employed ferritic/martensitic (F/M) stainless steels in lieu of austenitic alloys, thus decreasing the swelling behavior (especially at high doses), attributed to their bcc crystal structure and complicated defect-sink interactions (Chen, 2013). Various F/M alloys have been developed and tested in different SFRs worldwide including notable alloys HT9, EM12, DIN 1.4914, and PNC-FMS. These alloys exhibit behavior similar to that of austenitic alloys, but with markedly reduced steady-state swelling rates. F/M alloys have been irradiated up to 200 dpa and can, in principle, withstand even higher doses. Accordingly, F/M alloys are preferred over austenitic alloys when it comes to high-dose applications, but can be more challenging from a tube fabrication and welding standpoint. The most notable disadvantage of F/M alloys is their relatively low creep strength at above ~ 925 K. F/M alloys are a logical choice for high dose duct materials in modern SFR core designs, which typically experience temperatures below 925 K. Ongoing developments for other industries suggest that constituent modifications may yield F/M alloys with improved creep strength for nuclear application (of particular interest in regard to SFR fuel cladding), but these advanced alloys have yet to be extensively tested in neutron irradiation environments.

Advanced fast reactor plants designed to maximize economic power production through increased coolant temperatures and very high fuel burnups may be better served by other advanced cladding alloys. Among these, ferritic oxide dispersion strengthened (ODS) steels stand out as potentially offering both low swelling behavior and superb mechanical properties at high temperatures, likely enabling reliable cladding operation at up to 700 °C. Some irradiation data on ODS alloys exist, but a more robust irradiation performance database is needed to foster ODS cladding deployment in fast reactors. Since ODS alloys are also more difficult to fabricate via conventional methods, fuel assembly designs using ODS cladding will likely be well-served by using F/M alloys as materials for the duct, which operates at lower temperatures (Cheon et al., 2009). Finally, other advanced alloys have been proposed and are in development for SFR application, in an attempt to retain the high swelling resistance of F/M alloys while also affording improved high-temperature strength and creep resistance (Hoezler et al., 2020).

Fuel performance under normal and anticipated conditions

This section describes the known behaviors of both ceramic and metallic SFR fuels during sustained normal reactor operation from beginning of life to plant discharge. This mode is typically referred to as “steady-state operation,” though it includes deliberate transients such as startups, shutdowns, and power adjustment maneuvers as well as core power distribution shifts resulting from control element adjustments and the spatial effects of burnup/breeding evolutions. These expected operations can also include minor transients resulting from occasional equipment malfunctions and cladding breaches on a limited amount of fuel elements. Section “Fuel performance under off-normal conditions” covers off-normal conditions including cladding breaches and general transient behaviors. Many excellent works provide extensive and detailed summaries of the irradiation performance of SFR oxide fuels (Olander, 1976; Boltax, 1994; Guerin, 2012; Papin, 2012; Waltar et al., 2012; Crawford et al., 2007). For metallic fuels, similar detailed overviews may be found in (Hofman et al., 1997; Ogata, 2012; Chang, 2007; Seidel et al., 1986; Crawford et al., 2007). For uranium nitride and carbide, see (Manara et al., 2012; Sengupta et al., 2012).

Early life evolution

Fuel lifetime begins with the insertion of fresh fuel. Fuel fabrication is described in Porter and Crawford (2021) and Kato (2021) for metallic and oxide fuels, respectively. For oxide fuels, beginning-of-life irradiation behavior is marked by the rise to power achieving high centerline temperatures and steep temperature gradients in the fuel. In metallic fuels, early-life behavior is most notably affected by fission gas induced swelling, though significant thermally induced evolutions begin as well. Due to the fuel swelling, both fuels are expected to have some degree of interaction with the cladding, and thus fuel design incorporates optimized fuel

smeared densities specific to each composition. In either case, significant thermal/chemical/mechanical changes begin to take place during early life at full power.

Mixed oxide fuel (U,Pu)O₂

The first rise to full power for newly manufactured fuel pins marks the introduction of cooling and internal nuclear heating, creating significant temperature gradients across the radius of the fuel. In fresh fuel, peak power locations typically drive MOX fuel centerline temperatures to as high as 2500 K, contrasted by coolant temperatures of 800–900 K. The resulting temperature gradients in the fuel may exceed 600 K/mm. In brittle ceramics at low temperatures, the thermal stresses easily exceed the fracture strength of the oxide fuel, inducing radial cracks in the pellets as the hotter fuel center expands more than the periphery. After achieving full-power conditions, the fuel microstructure quickly begins to undergo dramatic changes—many occurring within hours—driven by the thermal extremes experienced by the fuel. Across the fuel radius, the initially uniform hypostoichiometric O/M ratio of the fuel becomes radially distributed with a minimum towards fuel center as oxygen redistributes down the temperature gradient in the fuel, impacting material properties (e.g., melting point, thermal conductivity, species diffusion coefficients, etc.). The fuel develops a well-known multilayered ring structure consisting of a central void region, a ring of columnar grains with radial orientation, a layer of equiaxed grain growth, and the as-fabricated microstructure (e.g., see [Olander \(1976\)](#)). [Fig. 2](#) illustrates these developing behaviors.

In the center region of the fuel where temperatures can exceed 2075 K, pores created during fabrication migrate up the temperature gradient to form a single central hole (or “void”) within hours of operation. This results in the creation or expansion of a central hole in the fuel, which also conveniently vacates what would otherwise be the peak temperature location. The mechanism of pore migration is thought to be fuel evaporation/condensation across the pore (due to the difference in fuel vapor pressure between the hot and cold sides of the lenticular pores) as well as the diffusion of the oxide surrounding the pores ([Olander, 1976](#)). This process develops near crystalline grains with higher density and thermal conductivity, healing cracks formed during the initial rise to power. In Pu-bearing fuel, the fuel vapor transport mechanism favors U constituents, such that the columnar grain region results in higher Pu content towards the fuel’s central region. In the middle radial region, classical high temperature grain growth occurs resulting in a region of larger equiaxed grains. At the colder fuel periphery, the microstructure remains largely in an as-fabricated state.

Though the cladding has a higher coefficient of thermal expansion than the fuel, the higher temperatures in the fuel cause a net reduction of the fuel-cladding gap. The gap is additionally reduced due to fuel cracking, which also causes “relocation” of the outer diameter of the fuel towards the cladding. Temperature and irradiation-enhanced creep in the fuel also begin contributing to radial expansion in the fuel.

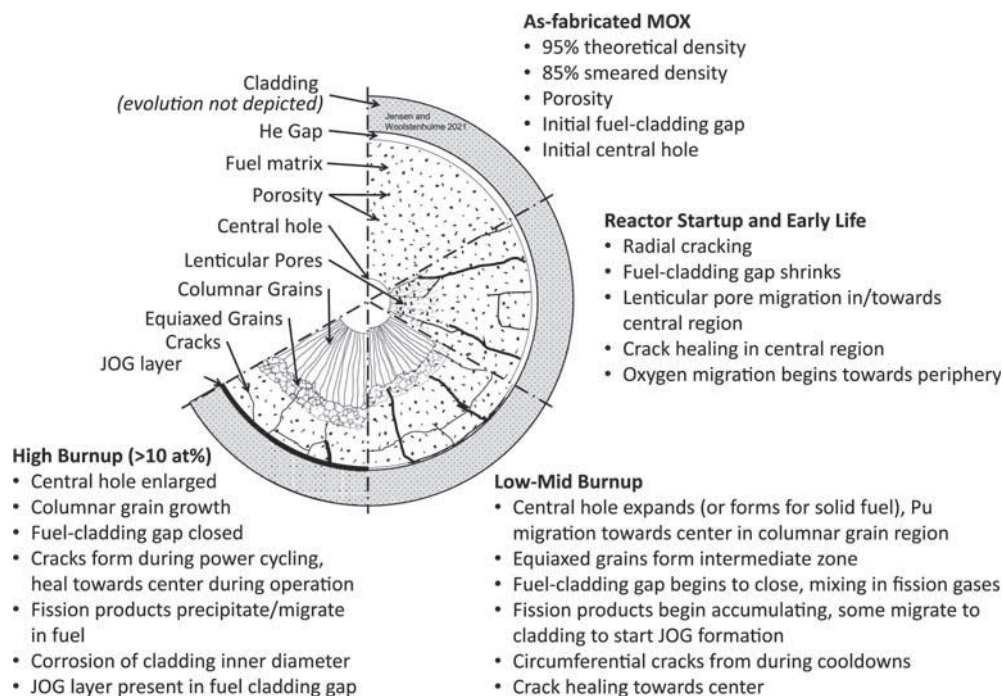


Fig. 2 Schematic overview of the irradiation lifetime of MOX fuel.

Metallic fuel (U-Pu-Zr)

In contrast, metallic fuels exhibit much lower intrinsic porosity, higher thermal conductivity, and ductile mechanical behaviors early in life. Thus, metallic fuels do not exhibit such marked thermal gradients, thermomechanical cracking, or pore migration during the first rise to power. Peak fuel centerline temperatures may approach 1200 K, resulting in smaller peak temperature gradients in the fuel (up to 120 K/mm). As-fabricated metallic fuel “slugs” contain a relatively uniform distribution of alloy constituents throughout. At power, the alloy constituents begin to migrate under driving forces established by the local temperature and local fuel composition, resulting in a redistribution across the radius that directly aligns (and is visibly obvious in post-irradiation examination (see Fig. 3)) with the isotherms within the fuel. The rate of constituent migration remains largely unquantified though it has been observed in the early life of the fuel and is driven by the temperature distribution in the fuel. Zr tends to migrate towards the higher-temperature center of the fuel and towards the periphery, both of which are beneficial outcomes in terms of providing a high centerline solidus temperature and more FCCI-resistant Zr content near the cladding. Uranium moves in the opposite directions while the Pu in Pu-bearing fuels largely remains stationary. This redistribution of alloy constituents plays an important role in the behavior of the fuel, affecting properties in each region of the well-known two- or three-layered structure visible in Fig. 3. The redistribution is somewhat altered as the fuel cools down later in life at lower powers and the radial position of phase fields move. Fig. 4 illustrates the evolution of changes in a metallic fuel during service.

As irradiation proceeds, the growing number of fission gas bubbles is largely retained as nano bubbles in the fuel matrix; they then begin to coalesce into larger pores depending on local zonal properties. Elevated temperatures and atom vacancies formed from fission and fast neutron collisions accelerate the fuel's creep behavior in response to the internal fission gas pressure. These behaviors cause metallic fuels to swell significantly during early-life burnup accumulation. They also are the primary cause of degraded thermal conductivity in the fuel. This behavior continues until $\sim 2\text{--}3$ at.% burnup, when the $\sim 30\%$ volume increase allows pore interconnection to enable fission gas transport more freely into the rod's internal volume. As porosity interconnects, the sodium fill in the pin begins to migrate into the fuel, helping restore the thermal conductivity of the dense fuel matrix. Thus, the temperature evolution of the fuel also undergoes notable variations during this process (Bauer and Holland, 1995).

In the case of metallic fuels, FCCI is an important consideration at temperatures relevant to operation (even for early-life conditions). In this context, two primary mechanisms affect fuel performance: (1) lanthanide (fission products) attack of the cladding material causing wastage at the inner cladding wall and (2) Fe-U diffusion forming U-Fe compositions susceptible to eutectic liquefaction of the fuel and/or cladding (cladding being the obvious concern) at high temperatures. The former plays an increasing role with increasing burnup while the latter is typically of greater concern in regard to transient overtemperature conditions discussed later.

Mid-to-high burnup evolution (<10 at% burnup)

As irradiation proceeds, the fuel evolution increasingly relates to the creation and behavior of fission products along with continuation of the same early-life behaviors. Fission products come in the form of gaseous and solid constituents. These additions to the fuel system alter material properties and potentially introduce additional loading to the cladding (which undergoes its own evolution as overviewed in Section “Fuel assembly structural material performance”). After a few at% burnup, the fuel has expanded to contact the cladding and effectively create important fuel-cladding mechanical and chemical interactions. In most SFR designs, fuel burnups have achieved burnup limits of 5–10 at%, with deformation of structural materials (e.g., duct and cladding) being the most life-limiting factor, in regard to austenitic stainless-steel claddings.

For many nuclear fuels, fission gas behavior has crucial ramifications for fuel performance. Introduction of fission gas to the fuel pin increases the internal pressure loading on the cladding and degrades the thermal conductivity of the gas gap between the fuel and cladding. The solid fission product impurities in the fuel matrix degrade the thermal properties of the fuel reducing both the thermal conductivity and melting point. These new constituents also migrate within the fuel, driven by high temperatures and the gradients thereof to become a consideration for cladding chemical interaction and potential cladding breach.

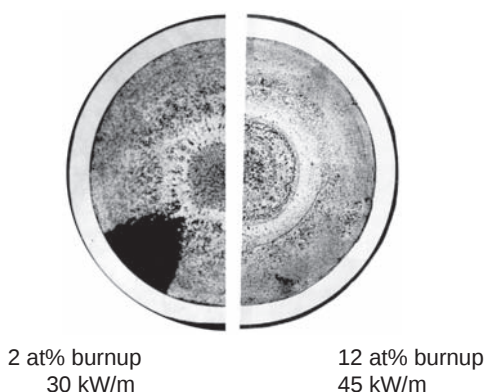


Fig. 3 Examples of U-19Pu-10Zr irradiated to low and high burnup.

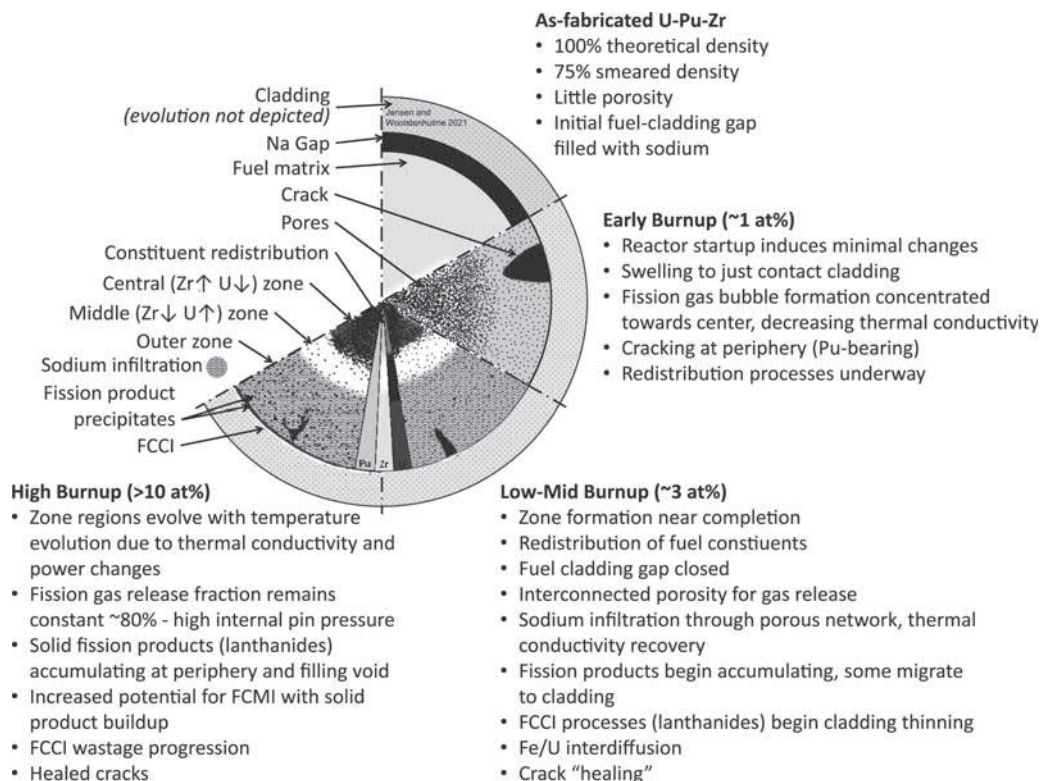


Fig. 4 Schematic overview of the irradiation lifetime of metallic fuel (U-Pu-Zr).

Mixed oxide fuel (U,Pu)O₂

As the fuel begins to accumulate burnup, the fuel structure undergoes continued cracking as a result of thermal cycling including additional circumferential cracks formed primarily while the fuel cools during shutdown periods. Cracks in the fuel can provide migration pathways to enhance diffusion rates in local regions (for example, via diffusion along crack surfaces, which is faster than diffusion via the bulk fuel material). In hypostoichiometric fuel, the fission process frees up oxygen atoms increasing O/M ratios, with the favorable effect of improving thermal properties. Solid fission products within the fuel matrix can alter the chemical properties of the fuel and may form metallic precipitates at higher burnups. Fission products that migrate down the temperature gradient eventually reach the fuel-cladding interface forming a layer called joint oxide gain (JOG). The JOG layer has a reduced thermal conductivity that is lower than that of the fuel but higher than that of fission gas. Furthermore, the JOG layer has relatively viscous properties, which allows it to serve as a mechanical buffer to the cladding and allows it to be extruded to other regions of the fuel in the event of fuel-cladding gap shrinkage. However, its fission product composition imposes deleterious corrosive effects on the cladding. Volatile constituents may exist in all phases across the radius and axis of the fuel and are likely to accumulate in colder regions of the fuel, producing a corrosive effect on the cladding. **Fig. 5** presents cross-section examples of MOX fuel irradiated at different power levels.

Fission gases formed within the fuel matrix precipitate into bubbles located within and between the grains of the fuel (termed intergranular and intragranular, respectively). The gas produced within the fuel induces local swelling and primarily diffuses through the fuel matrix to form bubbles, which may coalesce, interconnect, and continue migrating to cracks and the free volume, until finally venting to the fuel plenum. Fission gas release into the plenum increases with higher local temperature and fuel burnup. Fission gas release reaches nearly 100% at temperatures above ~1400 °C. At a certain burnup, the capacity to retain fission gases becomes saturated, and typical fission gas release reaches and sustains an 80–90% total release, at higher than even modest burnups. **Fig. 6** shows representative measured fission gas release for SFR MOX fuel as a function of pin average burnup. Fission and decay processes increase the total number of various smaller atoms in the system ultimately resulting in net positive fuel swelling. Relative to other fast reactor fuel types, the swelling rates are low and have proven constant over the explored burnups. The total swelling contribution is due to a combination of solid fission product and gaseous additions into the fuel.

Metallic fuel U-Pu-Zr

Coming from early-life burnup, metallic fuel undergoes a relatively dramatic thermal evolution as porosity in the fuel builds to the point of interconnection. The previously described temperature-driven characteristics governing material phase and compositional zones continue to play out with the addition of fission products, further complicating these behaviors. By 3–5 at.% burnup, zone formation is complete, with the aforementioned initial redistribution of alloy constituents also impacting local thermal properties.

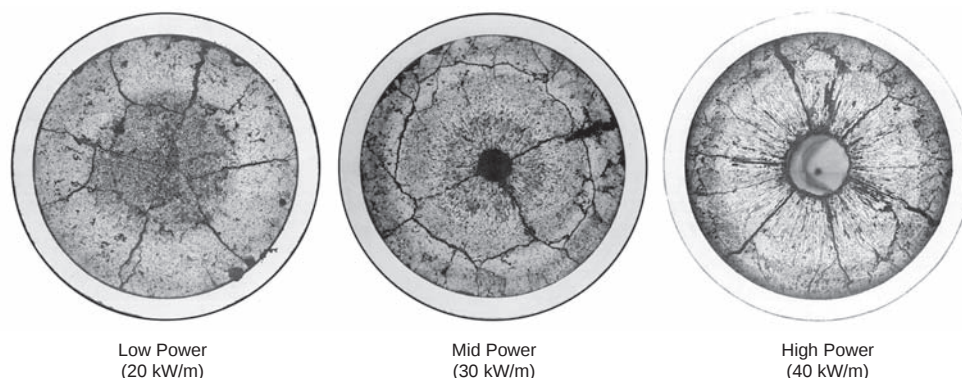


Fig. 5 Examples of MOX irradiated at various power levels (~ 5 at%, 85% smeared density).

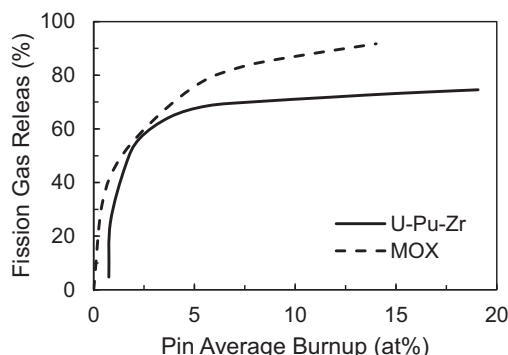


Fig. 6 Comparison of fission gas release for MOX and metallic fuels. Based on data from Hofman GL and Walters LC (1994) Metallic fast reactor fuels. R.W. Cahn, P. Haasen, E.J. Kramer (Eds.), *Materials Science and Technology, A Comprehensive Treatment*, vol. 10B, Frost BRT (ed.), Nuclear Technology\ VCH Verlagsgesellschaft mbH, p. 1.; Boltax A (1994) Mixed oxide fuel pin performance. In: Frost BRT (Ed.) *Nuclear Materials, Part II*, Cahn RW, Haasen P, Kramer EJ (Eds.), *Materials Science and Technology, A Comprehensive Treatment*, vol. 10B, Frost BRT (Ed.), Nuclear Technology\ VCH Verlagsgesellschaft mbH, p. 339; Guerin Y (2012) Fuel performance of fast spectrum oxide fuel. In: *Comprehensive Nuclear Materials*, vol. 2, pp. 547–578. Elsevier, ch. 2.21.

The thermal conductivity, reduced as a result of porosity growth, continues to recover as sodium backfills the growing network of connected pores. Solid fission products in the fuel matrix act as impurities and inclusions, ultimately decreasing thermal conductivity as they accumulate.

As with oxide fuels, fission products take various forms within metallic fuels. Such forms include gaseous, dissolved in the internal sodium, and in solution or precipitated within the fuel. Lanthanide fission products that collect in the fuel-cladding interface are an important performance consideration in terms of causing cladding wastage, which effectively weakens the cladding and forms low-melting-temperature eutectic compositions. This is exacerbated by elevated temperatures and burnup.

Fission gas release has been shown to be directly linked to the process of porosity interconnection and, thus, fuel swelling. After a few at.% burnup, the fuel typically reaches a near-constant fission gas release fraction of close to 80% (for smeared densities of $<75\%$), as illustrated in Fig. 6. Once again, the pin plenum is designed with a relatively large volume to accommodate this gas while maintaining manageable pressure within the cladding.

As fuel life progresses, a primary concern for fuel integrity is the combined impacts of reduced cladding strength due to FCCI and increased plenum pressure caused by fission gas release. Zonal microstructure evolution and gas porosity/sodium infiltration complicate accurate determinations of power to melt, due to uncertainties in the local thermal conductivity and solidus temperature of each radial zone. Precise quantification of these effects on thermal properties in metallic fuels is relatively nascent compared to the effort expended to understand oxide fuels.

Late life evolution (>10 at% burnup)

As fuel progresses to very high burnups, the increasing accumulation of fission products becomes an additional concern that must be accommodated volumetrically within the cladding, along with protecting the cladding from corrosive attack. Continued addition of “impurities” through fission products further degrades the thermal properties of the fuel. The impact of these changes is mitigated by decreased power generation and reduced fuel peak temperature due to the depletion of fissile isotopes in the fuel.

Mixed oxide fuel (U,Pu)O₂

As burnup becomes very high, dominant themes in fuel behavior resemble, from a thermomechanical standpoint, those found in earlier life, but with increasing impacts from accumulating fission products. Percent fission gas released remains relatively constant with continued increase of pressure in the pin plenum. Tellurium and cesium, which have migrated to the cladding, increase the internal corrosion of the cladding, and become increasingly problematic as time at temperature integrates with increasing burnup and with the continued addition of fission products. Cs migrates and accumulates in colder regions of the fuel (e.g., the ends of the fuels stack in low temperature regions and near UO₂ insulator pellets) where increased FCMI may occur and has been shown to manifest especially in very high burnup fuels (Guerin, 2012). Fuel designs must account for effects present at design-peak burnup in order to provide sufficient room for fuel swelling (including transients, as discussed later), depends on several factors, including cladding type. Typically, an initial smeared density of 80–85% is adequate to support experimentally validated burnups (around 20 at.% or more) and perform well under transient conditions.

Metallic fuel U-Pu-Zr

Progressing to very high burnups, solid fission products begin to fill the fuel's void space and will induce FCMI at some point, depending on the initial smeared density (for a burnup approaching 20 at.% with 75% smeared density). Still, experimentally measured cladding strain at 19 at.% has been accounted for by the effect of internal pin pressure alone. Regions of maximum FCCI tend to be in the upper portion of the fuel column, though not always at the region of highest fuel-cladding interface temperature (i.e., the top of the fuel column) for pins ~ 1 m in length. The peak FCCI location likely arises from high fission product availability via relatively high local fission rates, combined with a high interface temperature (Porter and Tsai, 2012). Fig. 3 includes an image of a high burnup metallic fuel specimen.

Fuel performance under off-normal conditions

The potential for fuel compaction via melting or dispersal scenarios is an important consideration in regard to SFR fuel in postulated accidents due to the resulting potential for reactivity insertion, power excursions, and propagation of damage to the core. Fuel-coolant compatibility is another important reactor design consideration. Of the primary fuels discussed herein, only oxide fuels have a meaningful chemical reaction with sodium coolant. The following sections provide an overview of important (and relevant) off-normal conditions such as operation with breached cladding, transient conditions such as overpower, loss of flow conditions, and severe accidents. Descriptions of hypothetical reactor conditions that could lead to these events is reactor-design-specific and thus beyond the scope of this paper.

Fuel performance with breached cladding

A variety of fuel defect types have been observed and their performance effects studied. Although fabrication and inspection methods are intended to minimize such defects, a core with thousands of pins is likely to contain defects that occasionally result in fuel pin failure. In this context, “fuel failure” refers to a breach of cladding hermeticity. For SFR oxide and metallic fuels, significant efforts were expended to understand the system response to breached cladding and its effects on operation. These studies focused on fission product release, fuel-sodium reaction behavior, and pin geometric stability.

When oxide fuel breaches, it contacts and reacts with the sodium coolant, forming reaction products that further degrade the fuel, sparking concerns over failure propagation by blocking coolant channels (loss of coolability). Experimental results revealed the volume increase within the fuel due to the formation of fuel-sodium compounds to be the primary mechanism of fuel pin degradation. The reaction product has lower thermal conductivity and decreases the fuel's O/M ratio, thus increasing fuel temperatures. The resulting effects of oxide fuel breach were successfully mitigated through proper fuel design (smeared density of <85%) and operation of the pins.

Metallic fuels are compatible with sodium (allowing for the internal sodium fill in the fuel pins). This compatibility leads to benign effects from metallic-fuel breaches and is an important feature regarding a variety of operational and accident conditions. Experiments involving the use of artificially defected cladding to instigate breach showed that the fuel can operate for extended periods (experimentally verified for up to 200 days) with minimal consequences, and they confirm the occurrence of negligible fuel-sodium reactions. Continued operation of breached pins all the way to the end of a reactor's operating cycle is manageable, the primary consideration being fission gas release into the primary coolant. A comparison of representative metallic and oxide fuel pins irradiated after cladding breach is shown in Fig. 7.

Transient conditions

Transient conditions of interest are classified as “operational transients.” These include reactor startup or shutdown, Anticipated Operational Occurrences, Design Basis Accidents (DBA), and Beyond Design Basis Accidents (BDBA). As implied by their names, categorization of these events is based on their probability of occurrence. Here, the events are listed in order of decreasing probability and reflect an increasing allowable uncertainty for quantified transient fuel behaviors. The resulting behavioral complexity also increases significantly under severe conditions, potentially leading to manifestations of material phase change and bulk

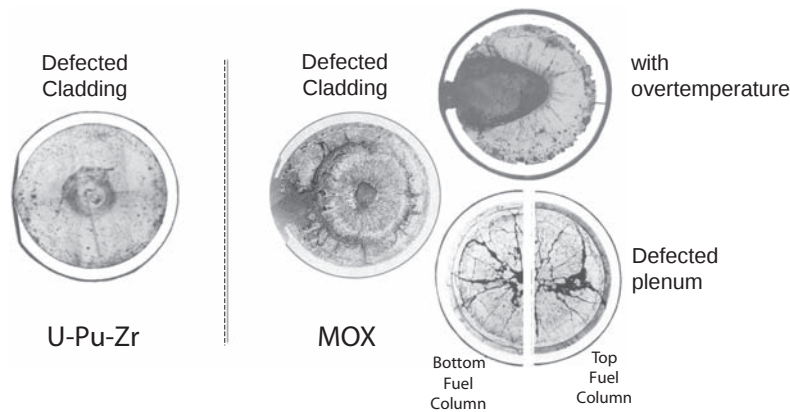


Fig. 7 Examples of artificially defected metallic and oxide SFR fuels run beyond cladding breach (Hofman and Walters, 1994, Strain et al., 1992, Chang, 2007).

relocations. As noted in Section “Design limits”, fuel design criteria directly apply to these events so behaviors must be well-enough understood to ensure compliance. The primary conditions discussed here include loss of flow (LOF) and transient overpower (TOP) conditions. For both these categories, “protected” and “unprotected” events (DBA and BDBA, respectively) are typically considered, based on whether active reactor shutdown systems are credited.

In each of these transient scenarios, the primary performance concern for the fuel relates to the effects of overheating conditions, which vary in timescales and severity for each event considered. Typical TOP events are relatively slow power ramps in the fuel induced by an unplanned control rod withdrawal and are considered local faults (impacts in the core localized near the control rod). In an LOF condition, fuel generates heat at a higher rate than the coolant can remove it (rates depend on the specific event classification), because cooling is reduced or lost. Fuel temperature gradients typically remain the same or decrease, but overall fuel and cladding temperatures increase. The dominant concern during a severe condition is progression to coolant overheating and localized coolant boiling (channel voiding), resulting in a positive reactivity insertion and subsequent fast overpower transient, thus leading to greater concern over additional reactivity insertions and overpower extremes. Still, even BDBA conditions in most modern SFR designs result in a rather benign fuel response. The following discussion briefly emphasizes the dominant phenomena during transient power-cooling mismatch conditions, which results in overheating (TOP and LOF) in the context of temperature rate effects, where protecting cladding integrity and understanding the margin to melting are dominant concerns.

Oxide fuel performance during transient conditions

A significant experimental database exists for oxide fuel transient performance (Papin, 2012). In a TOP event, cladding failure is primarily driven by the fuel cavity pressure loading overheated cladding. In this context, a “cavity” refers to a fuel region where temperatures may exceed the fuel solidus temperature, causing increased coalescence of high-temperature porosity. The resulting molten fuel-gas mixture forms a porous structure called “foam.” The lower-density fuel and high-temperature gases increase localized loading on the cladding, which could balloon depending on cladding temperature evolution and the kinetics of the internal accommodation of gaseous swelling and molten fuel expansion. Representative examples of transient behaviors in MOX fuel are given in Fig. 8.

In a pure LOF condition, the effects of fuel thermal evolution are generally less challenging are in an overpower event. Reduced cooling in LOF conditions causes the cladding surface temperature to rise while the power input decreases due to reactor shutdown and/or negative temperature reactivity feedback effects. Overheating within the fuel triggers phenomena similar to those described for TOP conditions. Stored energy (high temperatures) in the fuel can have a dominant effect on the margin to coolant boiling, resulting in positive void reactivity effects and added TOP conditions. Although, typically speaking, such conditions are highly improbable, LOF + TOP events with fast power ramp rates have been studied and show failure results from fuel thermal expansion and molten cavity pressure. Slower power ramp rates tend to show increased effects from gaseous swelling. Cladding breach during such an event is also influenced by the reduced strength or potential melting of the cladding due to increased temperature at the start of the power excursion, after LOF conditions have already progressed. In all cases, the increased cladding loading caused by the formation of a molten fuel cavity mixed with released fission gas is mitigated by lower smeared densities and in-pin fuel motion whenever a central hole is present in the fuel.

Metallic fuel performance during transient conditions

One of the most beneficial features of metallic fuels is that they possess inherent attributes that benefit core design and control systems by exhibiting helpful behaviors during off-normal conditions (Seidel et al., 1986; Chang, 2007; Wigeland and Cahalan, 2009; Sofu, 2017). Perhaps the most notable demonstration of the relationship between metallic fuel behavior and plant response under accident conditions was performed in EBR-II as part of the Inherent Safety Demonstration tests. A whole-plant, unprotected LOF initiated from full power was performed in the metallic-fuelled EBR-II facility (Planchon et al., 1987). In these demonstrations,

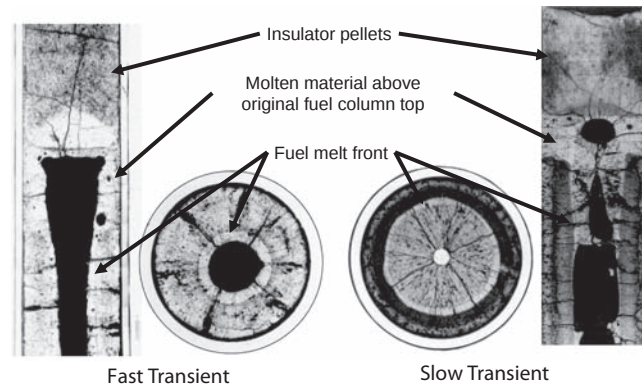


Fig. 8 Overview of observed MOX fuel responses to transient conditions showing fuel cross sections as well as the upper end of the fuel column, showing clear signs of fuel motion that occurred during the transient.

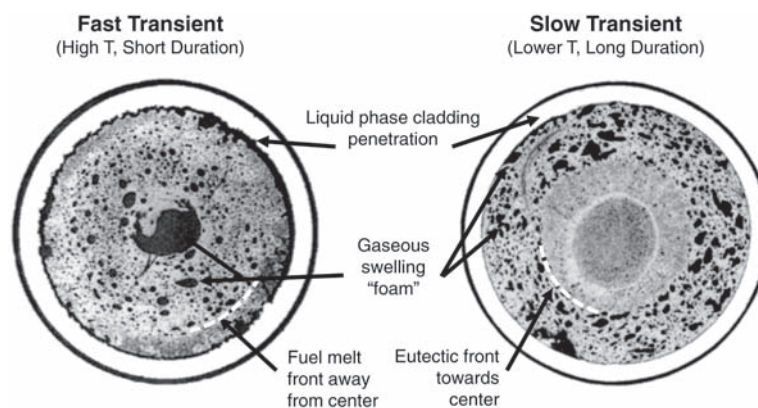


Fig. 9 Overview of the observed metallic fuel response to transient conditions.

the metallic fuel core exhibited no failures during the tests, nor afterwards during subsequent extended operations. In metallic fuels, reduced fuel temperatures yield less stored energy, thus diminishing Doppler reactivity effects and allowing LOF + TOP conditions to be more easily mitigated. The higher fuel density and harder spectrum core also require less excess reactivity to offset burnup (i.e., lower control rod worth), lessening the severity of accidental control-rod-induced TOP events.

More generally, the high thermal conductivity and high creep rate properties at relevant temperatures are key to good resilience of metallic fuels during transient off-normal conditions. In any transient scenario, cladding failure is caused by cladding rupture induced by the combined effects of cladding wastage from FCCI along with pin internal overpressure. Overpressurization effects are naturally exacerbated at increased fuel burnup from high accumulated fission gas release, whereas cladding eutectic penetration dominates the failure behavior in lower-burnup pins. Thus, fuel susceptibility to these two failure mechanisms is predictable from low to high burnups (Cohen et al., 1993).

The dominant FCCI behavior of concern in regard to off-normal performance is the eutectic interaction between Fe in the cladding and U in the fuel, with interdiffusion of each element between the fuel and the cladding. A eutectic temperature is found near 1000 K for high U-to-Fe concentrations. The net effect is greater induced melting in the fuel than within the cladding as illustrated in Fig. 9. Thus, the integrity of the cladding is not greatly imperilled until higher temperatures or long times at temperature are reached. Fuel liquefaction results in significant fission gas release and the coalescence of smaller gas bubbles from the matrix, inducing significant axial fuel swelling. Core-wide, this swelling serves to elongate the fuel column, leading to a beneficial, less-reactive condition. The relatively stable structure seen in Fig. 9 is sometimes called “foam.” As in normal operations, FCMI during transients shows a considerably less significant contribution (for smeared densities of <75%), due to significant fuel creep rates at elevated temperatures. Thus, the relatively ductile properties of the fuel mean that cladding loading is primarily driven by pin overpressure. Fig. 9 also includes a photomicrograph of fuel that underwent a fast transient resulting in melting of the fuel’s internal region, then extending towards the periphery. In fast power ramps, liquid phase eutectic penetration in the cladding becomes a dominant failure mechanism at high temperatures (>1350 K, a lower temperature eutectic for low U-to-Fe concentrations).

Post-failure behaviors in accident conditions

Fuel/core designs (and associated limits) for SFRs typically preclude the potential for conditions that lead to significant fuel failure, even during very low-probability transients. These conditions have been evaluated rather extensively for oxide (more) and metallic

fuels (less) in regard to the potentially significant consequences associated with them. Notable post-failure behaviors include a strong tendency for failure in the upper or top of the fuel column where cladding temperatures tend to be at a maximum both before and during a transient. The notable exception is for faster power ramps in oxide fuel systems, where peak temperatures may occur towards the core axial center, near maximum fuel power locations. In metallic fuels, peak temperatures more closely follow the coolant temperature in all cases (rising from the bottom to the top of the core). In metallic fuels, the top of the fuel column tends to also experience the highest FCCI effects caused by normal operation.

Transient experiments show that both metallic and oxide fuels expelled from the fuel column move upwards and out of the core towards regions with a lower neutron flux (i.e., into a less reactive state). Fuel-coolant compatibility plays a role in the potential formation of coolant blockages that reduce flow and may induce the failure of other nearby fuel pins (failure propagation and a threat to core coolability). Still, with large uncertainties in the evolution of extremely low-probability events and the resulting phenomena, the risk of recriticality after fuel dispersal has not been ruled out for oxide-fueled sodium fast reactor designs (Papin, 2012), while the metallic fuel system displays inherent characteristics (e.g., compatibility with sodium, low temperatures) more favorable for avoiding such concerns (Sofu, 2017; Tentner et al., 2019).

Fuel performance modeling

A nuclear fuel technology package includes the fuel design, knowledge of fuel performance and limits, and modeling tools to support development and evaluation. Fuel performance codes are typically based on finite-volume or finite-element methods for evaluating the thermo-chemical-mechanical behavior of the fuel—a truly multi-physics problem whose complexity benefits greatly from computational frameworks.

Fuel performance computer codes have been instrumental in the nuclear fuel R&D, commercialization, and licensing since the invention of computers. A fuel performance code essentially functions as a repository of integrated fuel performance knowledge used for predicting fuel performance for various purposes. Fuel developers use fuel performance codes to design and evaluate behaviors in new fuel designs and experiments intended to push the boundaries of current human understanding. Thus, this process relies on the incorporation of state-of-the-art models to minimize the level of extrapolation. One commercial or regulatory interest is to evaluate fuel performance in the context of the SAFDLs for a given fuel/reactor system. Developers and commercial industry also use the codes to verify that fuel performance is within nominal targeted operational limits, with the goal of maximizing performance and economic benefits.

Several fuel performance codes have been developed to address relevant performance criteria for SFR fuels and include some capability for application to other fuels and/or reactor types. These codes rely on an experimental database that supports model development in order to address material properties and relevant behaviors. The experimental database is also used to validate models ranging from simple property relationships to full integral system behaviors.

Behaviors described in this paper that are important to performance and safety have been sufficiently modeled to allow successful operation of many SFRs. Still, the development and refinement of models continues for complex phenomena such as fission gas migration and release, JOG formation and behavior in oxide fuels, and complex FCCI mechanisms in metallic fuels. In some cases, the evolution of the technology has led to adequate, but imperfect descriptions of material behaviors. For example, FCCI in metallic fuels has sufficient supporting experimental data over a variety of material types and burnups to validate semi-empirical models shown to conservatively predict the extent of FCCI in cladding. Yet, the opportunity exists to further develop understanding by expanding the current experimental database, along with developing improved models that reduce uncertainties and justify performance increases. This evolution in our understanding of fuel and increasing performance is seen in early SFRs (and especially LWRs) in which burnup and temperature were initially restricted to benign conditions—restrictions later relaxed as experience led to improved designs and reduced uncertainties.

Even with a rich historical knowledgebase, fast reactor fuel technology is making significant strides forward, thanks to the increasing availability of advanced experimental characterization and computational tools. As improved understanding of fuel performance translates into highly predictive capabilities, the true potential of the fuel technology may be unleashed to perform safely and with maximum utility.

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Irradiation Performance: High-Temperature Gas Reactor Fuels[☆]

Tyler J. Gerczak, Particle Fuel Forms Group, Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Glossary

AGR US Department of Energy Advanced Gas Reactor Fuel Qualification and Development Program.

Abbreviations

BISO Bistructural isotropic
EDS Energy dispersive X-ray spectroscopy
FIMA Fission per initial metal atom
HTGR High-temperature gas reactors
IPyC Inner pyrolytic carbon
KMC Kernel migration coefficient
MTR Materials test reactors
OPyC Outer pyrolytic carbon
PIE Post-irradiation examination
PyC Pyrocarbon
R/B Release to birth ratio
SEM Scanning electron microscopy
SiC Silicon carbide
TAVA Time-average volume-average
TRISO Tristructural isotropic
μXCT Micro-X-ray computed tomography

Introduction

The tristructural isotropic (TRISO) coated particle fuel form enables the realization of high-temperature gas reactors (HTGRs) and their attractive attributes such as passive safety and high outlet temperatures up to 950 °C, which facilitate industrial process heat applications and high thermal efficiencies for power production. The TRISO fuel concept has evolved considerably from the first coated particle fuel concepts developed in the 1950s during the initiation of the DRAGON Project in the United Kingdom (Price, 2012; Demkowicz et al., 2019). The first designs focused on carbide fuel kernels coated with a single layer of dense carbon in

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a carbonaceous monolithic fuel form. These early designs allowed fission product release and pyrocarbon fracture, but they established a starting point to improve on the early coated particle fuel designs. Improvements included adding another carbonaceous layer—termed the *bistructural isotropic (BISO)* (Goeddel, 1967; Jaye and Goeddel, 1968) or *buffer isotropic* (Kania et al., 2015) layer—to accommodate phenomena such as fuel kernel swelling, pyrocarbon dimensional changes, and fission gas pressure (Demkowicz et al., 2019). The decades that followed saw significant evolution of the coated particle fuel form for HTGR concepts, ultimately leading to adoption of the modern TRISO-coated particle fuel form which evolved to its current form in the late 1970s and early 1980s. Multiple comprehensive reviews of TRISO fuel design and development have been published over the decades of TRISO fuel and HTGR research, development, and demonstration (Demkowicz et al., 2019; Zhou and Tang, 2011; Verfondern et al., 2007; Petti et al., 2012; Stansfield, 1991; Kania et al., 2015). These reviews serve as an excellent background for the overview of HTGR fuel irradiation performance presented herein, which focuses on the mechanistic behavior of TRISO fuel.

Fuel configurations for HTGRs depend on reactor design. The most prominent fuel forms are spherical pebbles for pebble bed reactor designs and cylindrical compacts for prismatic reactor designs. While construction of each fuel form varies according to specific reactor requirements, most systems share common design features (see Helmreich and Hunn, 2021). Generically, the fuel form consists of modern TRISO fuel particles embedded in a matrix of natural and synthetic graphite flake and carbonized resin. The modern TRISO fuel particle consists of five primary components: the fuel kernel and successive layers of low-density carbon buffer, inner pyrolytic carbon (IPyC), silicon carbide (SiC), and outer pyrolytic carbon (OPyC). Each coating layer is deposited in succession using a fluidized bed chemical vapor deposition process (Sawa, 2012). Fig. 1 shows a cross section of a modern TRISO fuel particle (Hunn et al., 2015a).

Each component of the TRISO fuel design provides specific functionality during fabrication and/or operation. The fuel kernel provides fissionable material to produce heat and sustain reactivity in the core. The kernel also plays a key role in controlling internal gas pressure, as well as stabilization and retention of fission products and actinides during irradiation. The fuel kernel in modern TRISO fuel is UO_2 or a heterogeneous mixture of UO_2 and UC_x which is commonly referred to as “UCO.” The kernel’s enrichment level in modern fuel is typically low enriched uranium below 20% ^{235}U , depending on the intended application and reactor design. Some early efforts used highly enriched uranium, but a shift was made to low enriched uranium due to proliferation concerns (Verfondern et al., 2007). Over the course of the TRISO design history, many other kernel compositions have been explored and considered such as thorium- or plutonium-based kernel compositions, to provide different fuel cycle options for HTGRs (Verfondern et al., 2007; Besmann, 2010). The low-density carbon buffer layer serves as an effective fuel plenum. The buffer attenuates fission product recoils, and it provides excess volume to accommodate fission gas production, as well as space to accommodate fuel kernel swelling. The IPyC layer provides structural support and retains select gaseous fission products. The IPyC layer also protects the kernel from chlorine attack during fabrication. The SiC layer provides the primary structural support of the particle, resisting internal fission gas pressure and fuel swelling, and is a barrier to metallic fission products released from the kernel. The OPyC layer provides another “gas-tight” layer, and it protects the SiC layer when the TRISO fuel is being handled prior to compacting into its final fuel form. The OPyC layer also provides a surface for overcoating to facilitate particle compaction. Particles are then overcoated, pressed, and heat treated to create the final form of TRISO fuel particles in a matrix of graphite and carbonized resin.

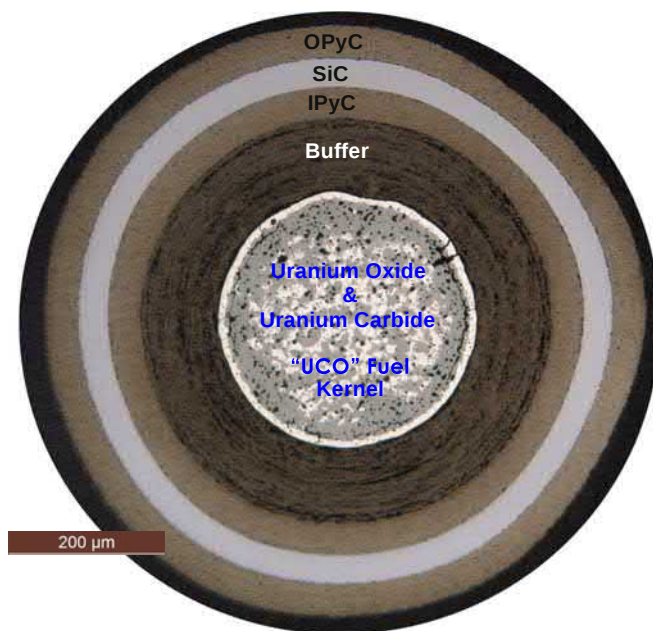


Fig. 1 Cross section of a modern TRISO fuel particle from the DOE AGR Fuel Qualification and Development Program. Reproduced with permission, courtesy of John Hunn, Oak Ridge National Laboratory.

The matrix material provides structural integrity, heat conduction, and neutron moderation. **Table 1** identifies the nominal kernel and layer dimensions and provides details of (1) the selected fuel form of the German reference fuel which has been adapted for multiple fuel development programs around the world (e.g., the HTR-PM in China and the PMBR in South Africa) and (2) the UCO-TRISO fuel produced in the United States through the Department of Energy Advanced Gas Reactor Fuel Qualification and Development Program (AGR Program) (Kania et al., 2013; Phillips et al., 2010).

Although each TRISO fuel component provides specific functionality as described in the preceding paragraph, post-irradiation examination has shown clearly that a key aspect to the holistic performance of TRISO fuel is the synergy between the components. For example, the functionality of the kernel to retain actinides and fission products is only effective when gas is retained by the outer coating layers. Particles whose coatings have defective or failed gas retention properties can exhibit substantive kernel degradation due to unrestrained reactions with the surrounding carbon, this behavior is especially evident after safety testing (Hunn et al., 2018a,b, 2019b). Another good example is the fact that, while the Buffer and IPyC layers in the BISO particle exhibit poor retention for many metallic species, in the TRISO particle the Buffer and IPyC layers act as a barrier to prevent high concentrations of actinides and fission products from contacting the SiC. Fracture of these inner two coatings has been observed to result in degradation of the SiC by localized high concentrations of species such as palladium and carbon monoxide (Minato et al., 1991; Hunn et al., 2016).

HTGR fuel performance has been established through decades of irradiation testing in demonstration and prototype reactors and materials test reactors (MTRs) (Petti et al., 2012). The HTGR fuel operational envelope varies for prismatic and pebble designs. **Table 2** shows relevant irradiation conditions for select irradiation programs targeting both prismatic and pebble reactor concepts. Generally, irradiations to support pebble bed designs have lower particle volume packing fractions with lower power densities, burnups, and temperatures, compared with prismatic designs. Comprehensive discussions of the irradiation conditions for TRISO fuel can be found in the literature (Petti et al., 2003, 2012; Kania et al., 2013; Demkowicz et al., 2019; IAEA, 1997).

Irradiation performance

A primary metric for fuel performance is the retention of fission products and actinides within the coated particle. Retention of fission products within the particles is of importance as the TRISO particles are a primary reactor component which define the functional containment for HTGRs and supports the potential for reactor co-location with industrial processes such as H₂ production, desalination, and process heat (INL, 2011). The pathway for release from the HTGR fuel form requires release from the kernel, coating layers, matrix material, and finally reactor components in the coolant flow path, with each component mitigating release depending on temperature, diffusion, and absorption properties. The release of fission products from intact fuel at design-basis operating conditions is inherently low relative to the release from particles with as-fabricated defects or particles which fail during service (Hunn et al., 2016). For that reason, a primary emphasis has been placed on reducing the initial defect fraction in as-fabricated particle fuel and ensuring that particle failure in service is controlled to limit the source term.

Kernel release

Several behavior models have been developed to predict the release of fission products from the kernel, and summaries are presented in IAEA-TECDOC-978 and by Powers and Wirth (IAEA, 1997; Powers and Wirth, 2010). The kernel is the first barrier to fission product release from the fuel form, governing release from particles with exposed kernel defects. The fractional release from the kernel, sometimes defined as *release to birth ratio* (R/B), of long-lived and stable fission products is presented in Eq. (1) (Nabielek, 1976). In general, the R/B is the fraction of a specific radioisotope that has been released from a component (kernel, particle, fuel element, etc.) relative to the total inventory that was produced in the component. The R/B from the kernel depends on the diffusivity of the fission product or fission gas in the UO₂ kernel. It is generally expressed as a function of the irradiation time, t (s), and reduced diffusion coefficient D' from Eq. (2), where D (m²/s) is the diffusion coefficient for the fission product species in the kernel and a (m) is the radius of the kernel. D' follows an Arrhenius relationship. Data supporting release from UCO kernels are limited since the effective diffusivity is not explicitly known. Therefore, UO₂ diffusion coefficients are usually used to predict release (Collin et al., 2015).

$$\frac{R}{B} = 1 - \frac{6}{D't} \sum_{n=1}^{\infty} \frac{1 - \exp(-n^2\pi^2 D't)}{n^4\pi^4} \quad (1)$$

Table 1 Nominal kernel and layer dimensions of modern reference TRISO particles for HTGR applications.

Org.	Kernel	Buffer	IPyC	SiC	OPyC	Fuel form
U.S. AGR (Phillips et al., 2010) ^a	UCO-425 μm	100 μm	40 μm	35 μm	40 μm	Compact
German (Kania et al., 2013)	UO ₂ -500 μm	95 μm	40 μm	35 μm	40 μm	Pebble

^aDimensions are from AGR-2, the second AGR irradiation experiment in the U.S. AGR Program.

Table 2 Summary of relevant conditions for select TRISO irradiation experiments.

Experiment	Fuel form	Packing fraction (%)	Burnup (% FIMA) ^a	Fast fluence ($\times 10^{25}$ n/m ²) ^b	Temperature (°C) ^a
U.S. AGR-1 and AGR-2 (Demkowicz et al., 2019) ^b	Compact	37%	7.3–19.6	2.2–4.3	995–1296 °C ^a
German HFR-K5 and K6 (Kania et al., 2013)	Pebble	6.2% ^c	7.8–10.9	3.2–5.9	800–1140 °C ^d
China (Knol et al., 2018)	Pebble	5.0% ^c	11.6–13.7	3.79–4.95	1050 $\pm$ 50 °C
Japan (Ueta et al., 2007)	Annular Compact	30%	5.5–7.0	2.2–2.7	1300 °C

^aTemperature reported in time-averaged, volume-averaged temperature, Burnup is reported in fissions per initial metal atoms (FIMA).

^bE > 0.18 for U.S. and Japan and E > 0.10 for all others.

^cFrom (Seeger et al., 2016).

^dIncluded transient above listed temperature.

$$D' = \frac{D}{a^2} \quad (2)$$

For short-lived fission products, Nabielek's work discusses that the release reaches steady state (Nabielek, 1976) and is then dependent on the competition between diffusive release and radiological decay. The Booth model for a spherical element is generally used to determine the R/B of short-lived fission products from the kernel (Booth, 1957; IAEA, 1997). Eq. (3) shows the steady-state R/B dependence for short-lived radioisotopes, where λ is the half-life of the radioisotope of interest. In the Booth model, when the condition $\sqrt{\lambda/D'} \gg 1$ is satisfied, then conditions for steady state have been met and it is appropriate to apply Eq. (3) to determine R/B of short-lived fission products.

$$\frac{R}{B} = 3\sqrt{\frac{D'}{\lambda}} \quad (3)$$

Diffusion is not the sole mechanism for release from the kernel: recoil release from fission occurring near the outer radius of the kernel also contributes. Fission recoils are responsible for ~2–3% of fission product release from the kernel for typical TRISO kernel sizes. These fission product recoils generally deposit in the buffer and are free to diffuse through the system. Eq. (4) characterizes fission product recoil release from the kernel (Ogawa et al., 1985). A primary design variable impacting fission product recoil release is the ratio of the kernel's surface area to its volume. In Eq. (4), z is the recoil range in the kernel.

$$\frac{R}{B} = \frac{3z}{4a} \quad (4)$$

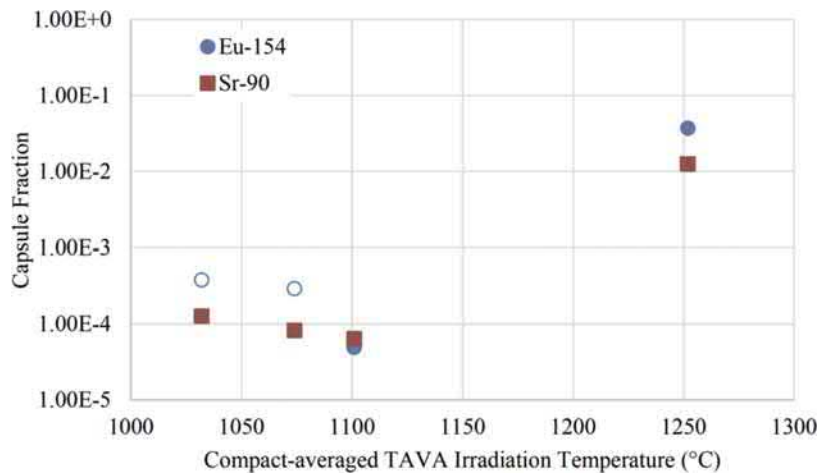
The kernel's retention of fission products is also influenced by its chemistry. In particular, the introduction of uranium carbide in the kernel reduces its ability to retain metallic fission products by reducing its oxygen potential. The lower oxygen potential limits the kernel's ability to form stable oxides (Homan et al., 1977). This was highlighted in a study of the irradiation of bare uranium carbide kernels, which showed near complete loss of ¹³⁷Cs, ¹²⁵Sb, and ⁹⁰Sr (Nabielek, 1976). Similarly, the second irradiation experiment from the U.S. AGR program, AGR-2, included compacts with both UO₂ and UCO kernels that were irradiated to similar conditions. Table 3 lists the compact average irradiation conditions from the AGR-2 irradiation (Collin, 2018). Destructive post-irradiation examination (PIE) by scanning electron microscopy (SEM) with energy dispersive x-ray spectroscopy (EDS) of UO₂ kernel particles from Capsule 3 showed primarily palladium in the IPyC and SiC layers. Comparatively, the UCO kernel particles from Capsule 5 showed not only palladium, but also a consistent presence of uranium co-located with the palladium, suggesting lower U retention relative to the UO₂ kernel. As would be expected for thermal diffusion-related migration, the higher temperature UCO kernel fuel particles (Capsule 2) showed a greater concentration and diversity of fission products and actinides released from the kernel than did lower temperature UCO kernel fuel particles (Gerczak et al., 2020a). The release of fission gases and noble metals such as silver and palladium is not expected to be strongly impacted by kernel chemistry, because they do not readily form an oxide. However, noble metal inclusions were observed in UO₂ kernels and not in UCO kernels. Similarly, a significant release of lanthanides was reported from carbide kernel fuel (e.g., UC_x) if there was insufficient oxygen (O/U ≤ 1.1) to stabilize the fission products as oxides (Tiegs, 1982).

Fission product release through TRISO layers

Most fission products are well retained in modern, high-quality TRISO particles. Historically, biologically relevant fission products such as cesium have been observed to be released, along with silver, from both UO₂ and UCO kernel fuel (IAEA, 1997). However, modern TRISO fuel from the AGR-1 irradiation experiment has shown excellent retention of cesium, with a capsule average R/B < 3 × 10⁻⁶ for ~49,500 particles per capsule, which is much less than a single particle's inventory (~2 × 10⁻⁴) (Demkowicz et al., 2015). The U.S. AGR program has explored TRISO fuel irradiation behavior over a broad range in conditions. Irradiation temperature has been shown to be a primary driver for fission product release (Demkowicz et al., 2015; Harp et al., 2018a,b; Stempien and Demkowicz, 2020). The AGR program's recent PIE has provided insight on the release of select fission products

Table 3 Summary of compact average irradiation conditions for AGR-2 capsules.

Capsule ID	Kernel type	Burnup (%FIMA)	Fast neutron fluence (10^{25} n/m^2) ^a	TAVA ^b temperature
Capsule 6	UCO	7.26–10.81	1.94–2.73	987–1129 °C
Capsule 5	UCO	10.07–12.88	2.91–3.42	1040–1141 °C
Capsule 3	UO ₂	9.01–10.69	3.05–3.53	996–1062 °C
Capsule 2	UCO	10.80–13.15	2.88–3.47	1194–1296 °C

^aE > 0.18 MeV.^bTime-average volume-average.**Fig. 2** Capsule's average fission product release fraction of ⁹⁰Sr and ¹⁵⁴Eu for the AGR-2 experiments, open symbols for ¹⁵⁴Eu are values derived from minimum detectable activity. Image courtesy of John Stempien, INL.

by providing an effective mass balance of the fission products in the capsule, capsule components, and compacts to determine the extent of release over a range of irradiation conditions (Demkowicz et al., 2015; Stempien and Demkowicz, 2020). The AGR-2 irradiation experiment explored a range of Time-Average Volume-Average (TAVA) temperatures (i.e., the irradiation temperature of a fuel compact and constituent particles is characterized by the calculated temperature averaged over time and through the entire compact). The AGR-2 PIE focused on four capsules (Capsules 2, 3, 5 and 6) with Capsule 2 representing the highest temperature, with compact average TAVA temperatures of 1194–1296 °C compared to average TAVA temperatures of 987–1141 °C for all other capsules. The AGR-2 capsule-level release behavior of select fission products is described by Stempien and Demkowicz (2020). Outside of high-temperature Capsule 2, most radioisotopes remain well retained. Cesium is shown to be well retained, even in Capsule 2, with ¹³⁴Cs release fractions $\leq 8.74 \times 10^{-5}$ where a single particle's inventory is $\sim 2\text{--}3 \times 10^{-5}$. However, a temperature dependence is observed for capsule-level fractional release of ⁹⁰Sr, ¹⁵⁴Eu, and to a lesser extent, ^{110m}Ag. Fig. 2 shows the capsule average fission product release fraction of ⁹⁰Sr and ¹⁵⁴Eu for the AGR-2 experiments (Stempien and Demkowicz, 2020). There is a clear increase in release fractions for ⁹⁰Sr and ¹⁵⁴Eu at higher temperature (Capsule 2, capsule average TAVA temperature is 1252 °C), indicating a temperature-dependent release process through intact TRISO layers. Several models have been developed to predict fission product release through the TRISO layers. These models are dependent on both the diffusivity of the radioisotopes of interest, reflecting the temperature dependence, and the neutron fluence for the irradiation. The uptake of fission products into the SiC layer was shown to be proportional to the neutron fluence (IAEA, 1997). Because concentration gradients are drivers for diffusion, it is reasonable to hypothesize that the increased uptake with increased SiC layer radiation damage increases the effective diffusivity in the SiC layer. This would be expected to impact overall release, because diffusion in the SiC layer is rate limiting. For example, the diffusivity (diffusion coefficient) of silver, cesium, and strontium in PyC is 100–1000 × greater than in the SiC layer at 1100 °C (IAEA, 1997).

While ^{110m}Ag is not a high-yield fission product, the release of silver has been a point of much interest over the history of the TRISO fuel development due to its unique release: in some cases, a majority of the silver inventory is released (Harp et al., 2018a,b; Stempien and Demkowicz, 2020). Despite the extent of release, silver is considered an occupational hazard and not a hazard to the public (NRC, 2004). Silver does not form a stable oxide or carbide and is readily released from both UO₂ and UCO kernels. Therefore, silver is available to diffuse through the TRISO layers. The release of silver from intact particles during operation has been suggested to be impacted by the microstructure of the SiC layer, with more-equiaxed, fine-grained SiC layers retaining silver better than columnar-grained SiC layers with grains extending greater than half the thickness of the SiC layer (Petti et al., 2003). Observations of microstructure-related differences in silver release have been observed in safety-tested compacts from the AGR-1 PIE effort. However, a difference in grain size relationship is observed during safety-testing at 1800 °C relative to historic behavior during operation as TRISO particles with SiC layers of similar grain shape but a finer grain size showed greater additional release relative to

larger-grained SiC layer variants (Gerczak et al., 2016). Ultimately in pile silver release is tied to irradiation temperature. The irradiation experiments highlight an apparent critical temperature of $\sim 1100^\circ\text{C}$ above which silver retention decreases significantly (NRC, 2004) this is visualized in Fig. 3. Fig. 3 shows the compact-average retention for $^{110\text{m}}\text{Ag}$ from AGR-2, highlighting the perceived irradiation temperature dependence on silver release taken from compact gamma analysis (Harp et al., 2018a). The complete temperature dependence is yet to be resolved.

Particle failures

The typical behavior of TRISO fuel during irradiation is formation of fission gas bubbles in the kernel and radiation-induced kernel swelling as well as radiation-induced shrinkage of the buffer and associated internal tearing of the buffer away from the IPyC layer. The ideal TRISO particle response is shown in Fig. 4 with the IPyC, SiC, and OPyC layer remaining intact (Hunn et al., 2015a). This response leads to successful retention of most radionuclides during operation and during safety testing (Demkowicz et al., 2015). However, this behavior is not always realized and, in some situations, particle failure is observed.

The nature of a TRISO particle failure dictates the release of specific radionuclides. Typical particle failures can be described by two general classes: failed-TRISO particles and failed-SiC particles. Failed-TRISO particles are defined by the failure of the normal retention properties of the IPyC, SiC, and OPyC layers, providing a direct pathway for radionuclide release; this can happen either prior to or during irradiation. A major manufacturing defect or a particle whose TRISO layers are cracked during post-fabrication handling is known as an exposed kernel defect and is an example of TRISO particle failure. Overpressurizing a particle during irradiation or excessive radiation-induced layer shrinkage can also lead to this type of failure. This failure type is generally indicated by concurrent release of fission gases, volatile fission products, metallic fission products, and other radionuclide species from the kernel. Failed-SiC particles are generally described as particles with a compromised SiC layer and at least one intact pyrocarbon (PyC) layer to remain gas tight. Failed-SiC particles release volatile and metallic fission product species that diffuse through PyC; they generally have good retention of the noble fission gases Kr and Xe. Release of fission products from failed particles is notably higher than release from intact particles. However, the release can be variable and gradual, particularly as the kernel stabilizes a measurable fraction of many fission product species (depending on the local oxygen potential), thus inhibiting their release.

Holistic measurement of fuel performance and quality is represented by R/B on larger length scales such as the capsule or fuel element level (i.e., from a pebble or from a compact) and is a measurement used to assess the performance of the fuel during irradiation. The R/B from a fuel element is similarly defined as the released fraction of a specific isotope vs. the amount produced during irradiation. In-pile R/B measurements are frequently made of noble fission gases such as ^{85}Kr , which are not condensable in sweep gas or coolant. The R/B signal can indicate any of a number of phenomena during irradiation. In particular, elevated baseline R/B signals can imply a high level of dispersed uranium contamination outside the SiC layer in the matrix and OPyC. Fission of dispersed uranium outside the SiC layer contributes to release of the fission products and fission gas into the primary circuit of the reactor, as they only have to diffuse through the OPyC and/or matrix and are not held up by the rate-limiting SiC layer. The online monitoring of fission gas R/B can also indicate complete TRISO layer failure due to fabrication, failure during irradiation, or the presence of other exposed kernel defects after manufacturing. During online fission gas monitoring, acute increases in R/B of fission gases can indicate the presence of complete TRISO failures where no gas-tight layers remain intact.

The R/B analysis provides confidence in confirming complete TRISO failure and estimating the number of exposed kernels during service. So, it is significant that modern TRISO fuel irradiations have had very low in-pile R/B measurements, indicating

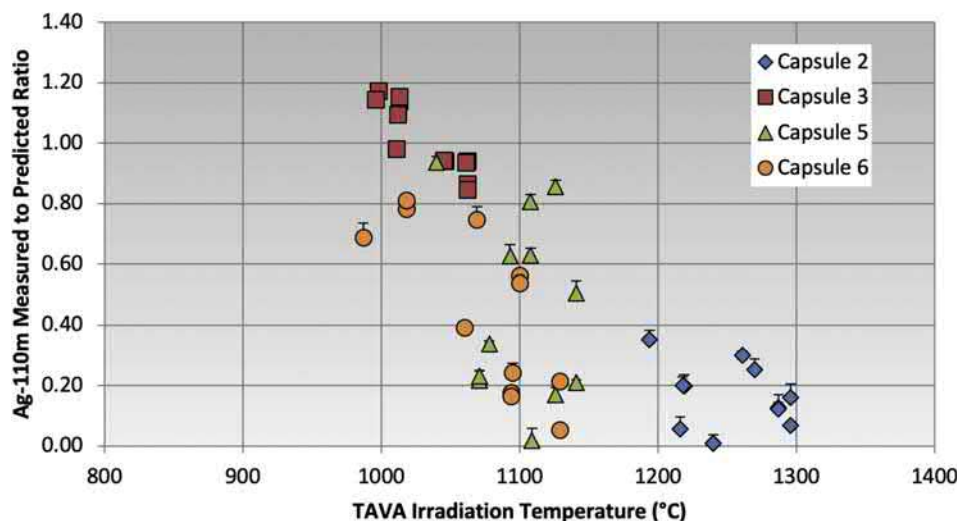


Fig. 3 Decay-corrected, measured-to-predicted $^{110\text{m}}\text{Ag}$ activity ratios of compacts from each AGR-2 capsule as a function of the TAVA irradiation temperatures. Lower values indicated less $^{110\text{m}}\text{Ag}$ retention and greater release from the compacts. Image courtesy of J. Harp.

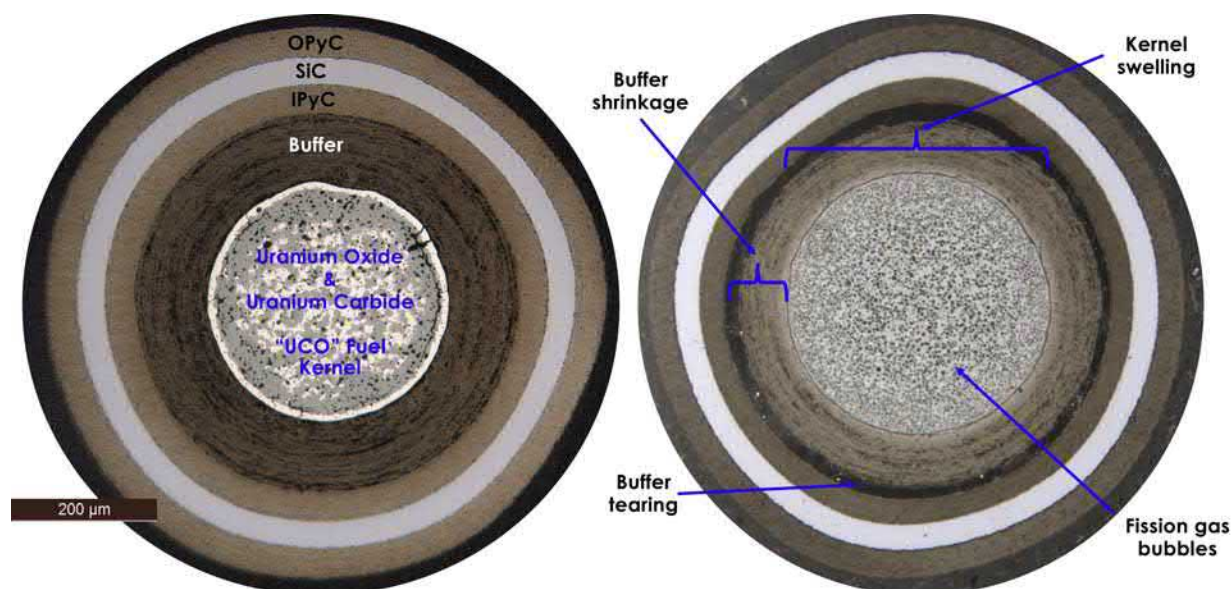


Fig. 4 Typical response of UCO TRISO fuel particle architecture to irradiation in the AGR-1 irradiation experiment. An as-fabricated particle cross section is shown on the left and an as-irradiated particle cross section is shown on the right.

exceptional performance. For example, the Chinese fuel HTR-PM irradiation had a $^{85\text{m}}\text{Kr}$ R/B of $2.4\text{--}3.3 \times 10^{-9}$ (single particle equivalent 1.1×10^{-7}) with an in-pile failure fraction of $\leq 5.0 \times 10^{-5}$ at 95% confidence. The US AGR-1 irradiation showed $^{85\text{m}}\text{Kr}$ R/B = $0.2\text{--}2 \times 10^{-7}$, demonstrating an in-pile failure fraction of $\leq 1.1 \times 10^{-5}$ at 95% confidence, and the German HFR-5/6 irradiations showed $^{85\text{m}}\text{Kr}$ R/B = $1.7\text{--}9.0 \times 10^{-7}$, demonstrating an in-pile failure fraction of $\leq 5.1 \times 10^{-5}$ at 95% confidence. For all these irradiations, the R/B analysis indicates that no complete TRISO layer failures occurred during irradiation (Demkowicz et al., 2019; Kania et al., 2013).

Particle failures have the potential to significantly contribute to the accident source term. High-quality fuel is defined as having “very low particle failure rates during irradiation with failure fractions $\sim 10^{-5}$ (upper bound at 95% confidence)” (Demkowicz et al., 2019). As-fabricated defects can contribute directly to the in-pile failure fraction. Acceptable as-fabricated defect fractions (which are low, but are not zero) vary based on reactor design and regulatory expectations. Of particular issue are exposed kernel defects, which provide an immediate pathway for release of fission products not retained in the Kernel. Defective particles can be present due to numerous factors. Observed as-fabricated defects include soot inclusions, metallic inclusions, missing coating layers, cracked coating layers, and other anomalies. With the exception of cracked coating layers, most defects are attributed to the particle coating process, although cracked particles can be generated also during the compacting stage or during particle handling. Cracked particles are considered exposed kernel defects and can release fission products readily. Historically, the compacting process has been observed to fracture particles (Minato et al., 1998). This is not unexpected since the particles are subjected to elevated pressures—up to ~ 9000 lb./in² in AGR-2 (Pappano and Hunn, 2008)—during compacting, and the particle bed has limited opportunities to reconfigure itself for stress relief during the compacting process. However, particle fracture can be avoided or limited, as was shown in AGR-2, in which the as-fabricated defect fraction for particles before and after compacting were roughly equivalent (Hunn et al., 2010).

Processing defects such as soot and metallic inclusions may not fail immediately or at all during service. However, these defects have been shown to lead to particle failure, with the resultant release of fission products and actinides (Hunn et al., 2018b). Fig. 5 shows an example of an unusually exaggerated as-fabricated soot inclusion in a TRISO particle which resulted in a porous SiC layer. While this compromised SiC layer survived throughout irradiation, a small crack developed during safety testing at 1600 °C. The kernel was removed during the post examination task of deconsolidation-leach-burn-leach (Hunn et al., 2016). Fig. 6 shows an example of a molybdenum inclusion in the SiC layer of a TRISO particle (from fabrication) (Hunn, 2009), along with a suspected particle failure due to molybdenum attack of the SiC layer from the AGR-2 irradiation experiment after safety testing at 1800 °C (Hunn et al., 2018b). Both inclusion defect particles shown here maintained gas tightness during irradiation and did not fail until being subjected to the severe conditions of a high-temperature safety test.

In-pile particle failure

Although modern TRISO fuel has shown that the observed failure rates in pile are within acceptable limits, understanding the failure mechanism associated with the few observed failures contributes to optimization of fuel fabrication and quality control methodology which help to ensure safe, efficient operation, as well as prediction of safe operating conditions. Over the decades of coated particle fuel development, multiple failure mechanisms have been identified, and designs and operating conditions have evolved to

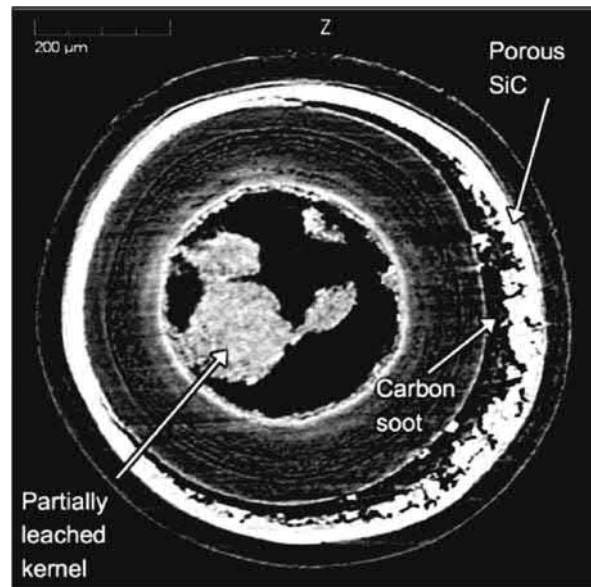


Fig. 5 AGR-1 particle that released cesium during 1600 °C safety testing of Compact 3-3-2. Image courtesy of John Hunn, Oak Ridge National Laboratory.

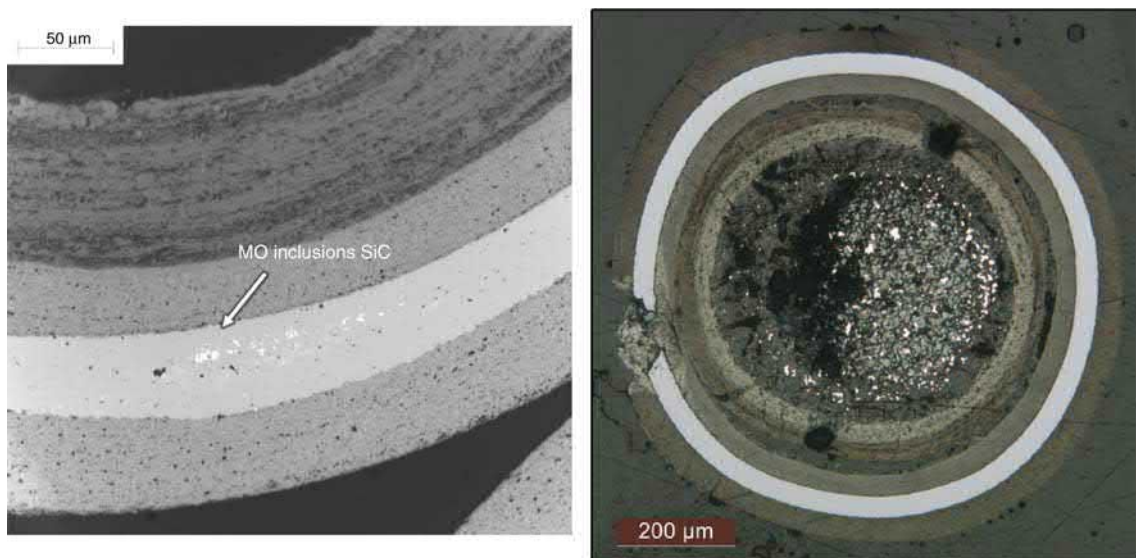


Fig. 6 Optical micrograph of AGR-2 UO₂ TRISO showing molybdenum inclusion in SiC (left) and optical micrograph of AGR-2 UCO TRISO Particle 232-SP01 planar section prior to final polish, showing through-layer SiC failure from Mo attack (right). Images courtesy of John Hunn, Oak Ridge National Laboratory.

mitigate certain failure mechanisms. This highlights the continued need to understand particle failure mechanisms, both in-pile and under abnormal accident conditions, so that failure rates can be limited and appropriate operation can be ensured. Irradiation testing and safety testing (in core or post-irradiation) are important for determining the conditions and likelihood for failure. The particle failure mechanisms responsible for appreciable fission product and radionuclide release are understood by (1) determining the nature of the fission product release (e.g., fission gases only vs. fission gases and metallic fission products) and (2) through destructive PIE to observe the specific failure mechanisms.

Pressure vessel failure

Noble fission gases (Kr, Xe) and carbon monoxide (CO_(g)) are the primary contributors to internal gas pressure in the standard TRISO-coated oxide kernel particle. Fracture of the PyC and SiC layers can occur if the internal pressure exceeds the yield strength of the coating layers. This fracture can severely degrade particle integrity, leading to abrupt failure of the TRISO layers and exposure

of the kernel, providing a direct path for fission product and radionuclide release. This failure mechanism is known as *pressure vessel failure*. Fig. 7 shows an example of this failure mode in a UO_2 kernel TRISO fuel particle (Homan and Long, 1976).

The fission gas pressure is dependent on kernel chemistry (e.g., oxide vs. carbide), and irradiation conditions (primarily burnup and temperature). Release of Kr and Xe fission gases from the kernel is described above and in Eq. (1). For UO_2 kernel fuel, the $\text{CO} + \text{CO}_2$ gas pressure dominates at burnups below $\sim 9\%$ fission per initial metal atom (FIMA) (Barrachin et al., 2009). The equilibrium $\text{CO} + \text{CO}_2$ gas pressure is tied to the initial oxygen-to-metal ratio (O/M) (e.g., uranium, plutonium) (Besmann, 2010). In general, in the absence of a gettering phase, the total $\text{CO} + \text{CO}_2$ gas pressure increases with increasing O/M ratio as the fuel becomes hyperstoichiometric at higher burnups and fissile material is consumed liberating oxygen (Besmann, 2010). An empirical relationship for oxygen released from the kernel in UO_2 was developed by Proksch et al. (1982) and is given in Eq. (5), where O/F is the fraction of oxygen released in atoms per fission, t is irradiation time (s), and T is time-averaged particle surface temperature (K) (Proksch et al., 1982; Hales et al., 2013). In Eq. (5), the fraction of released oxygen is assumed to convert to $\text{CO}_{(g)}$ by reacting with the carbon buffer layers (Hales et al., 2013). When equilibrium conditions due to high temperatures and long irradiation times are met, the O/F reaches a thermodynamic limit of 0.4 (Proksch et al., 1982). Fission gas buildup is primarily accommodated through gas migration into the excess void space in the buffer. The buildup of $\text{CO}_{(g)}$ gas pressure limits or influences the design envelope (burnup, temperature, temperature gradient) for UO_2 kernel TRISO. Control of the $\text{CO}_{(g)}$ gas pressure is one of the primary drivers for the use of UCO kernel fuel, particularly when higher burnups are sought, as in prismatic fuel designs (Petti et al., 2003; Maki et al., 2007). The addition of the uranium carbide phase, or the use of oxygen getters such as SiC or ZrC, buffers the formation of $\text{CO}_{(g)}$ and reduces overall $\text{CO}_{(g)}$ pressure by effectively lowering the O/M ratio so that it is much less dependent on irradiation conditions. For context, in the PuO_2-x fuel system, the total gas pressure can be reduced by ~ 3 orders of magnitude through oxygen getters (Besmann, 2010). This approach mitigates the risk of failure by internal pressurization and allows for higher burnups to be achieved.

$$\frac{O}{F} = \frac{t^2}{(1.211 \times 10^{10}) 10^{\frac{8500}{T}}} \quad (5)$$

Pressure vessel failure is an example of mechanical failure. The TRISO particle architecture is also intended to provide some resiliency to SiC fracture. Radiation-induced shrinkage of the OPyC layer has been modelled to put the SiC layer under compression when under irradiation. This compressive stress state is expected to decrease the probability of SiC mechanical failure (Miller et al., 2006; Powers and Wirth, 2010). However, recent PIE efforts suggest that this effect may not be applicable in all conditions experienced by the fuel, as separation has been observed between the SiC and OPyC layers, suggesting the OPyC layer does not always keep the SiC under compression (Hunn et al., 2014, 2015b, 2019a).

Kernel migration

Kernel migration, or the *amoeba effect*, is the net movement of the fuel kernel along the temperature gradient within a particle. Excessive migration can lead to kernel interactions with the IPyC or SiC layers, compromising the integrity of the fuel particle. Fig. 8 shows an example of excessive kernel migration in a UO_2 TRISO fuel particle (Homan et al., 1977). Work by Wagner-Löffler (1977) provides an overview of the mechanisms driving the kernel migration phenomenon. Kernel migration occurs due to the transport of carbon from the hotter side of the kernel to the cooler side. In carbide kernel TRISO fuel, the transport is thought

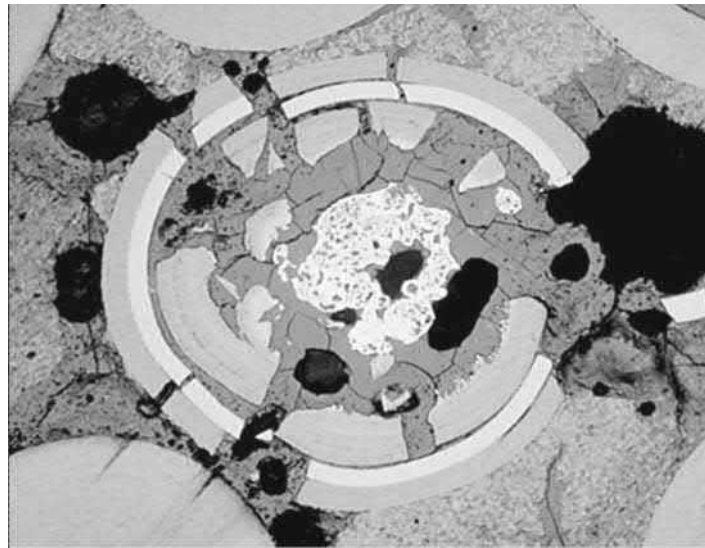


Fig. 7 Example of a pressure vessel failure in UO_2 TRISO fuel.

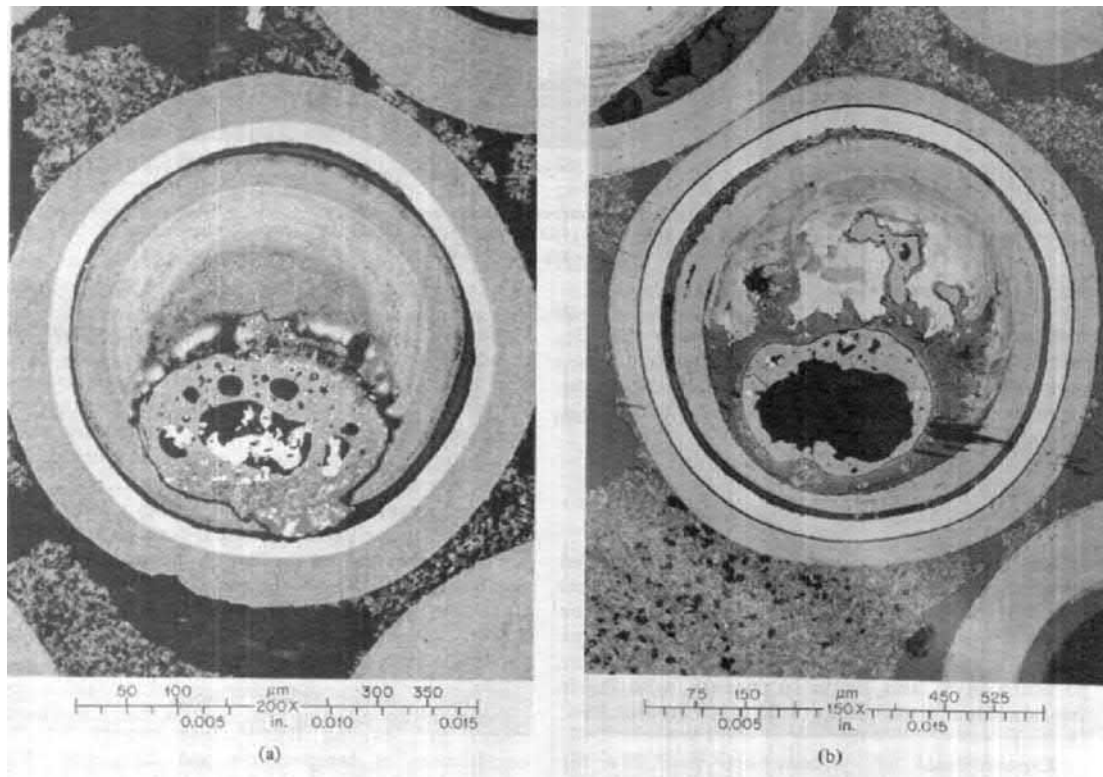


Fig. 8 Examples of the amoeba effect on kernel migration.

to be due to solid-state diffusion of carbon across the kernel. For oxide fuel particles, the mechanism is thought to be dependent on a higher concentration of $\text{CO}_{(g)}$ forming on the hotter side of the particle and diffusing in a gaseous state around the kernel to the cooler side. A complimentary mechanism supporting net transport of oxygen to the hot side must occur. Possible pathways are (1) solid-state diffusion of oxygen back through the kernel (oxygen liberated from $\text{CO}_{(g)}$ decomposition) or (2) gas phase $\text{CO}_{2(g)}$ transport back to the cool side. The transport of oxygen back to the hot side is expected to be the rate-limiting step. This mechanism depends on the oxygen potential in the systems, the control of which can mitigate the potential for kernel migration (Besmann, 2010; Bullock and Kaae, 1983). This has been confirmed in UCO kernel TRISO fuel, in which the oxygen potential is low enough that kernel migration is not observed (Homan et al., 1977), consistent with oxygen transport being the rate-limiting mechanism for this phenomenon. The minimum amount of carbide phase (UC_x) in UCO TRISO fuel needed to limit kernel migration by suppressing $\text{CO}_{(g)}$ production has been calculated at 5.1–5.5 mol% for 15.5% enriched fuel taken to 16.1% FIMA (McMurray et al., 2017).

An empirical relationship for kernel migration in UO_2 TRISO is shown in Eq. (6) (Maki et al., 2007), where δ_{MIG} is the migration distance (m), KMC is the kernel migration coefficient ($\text{m}^2\text{-K/s}$), T is the temperature, and ∇T is the associated temperature gradient. Comprehensive models to predict the amoeba effect have been developed by Özdere Güllol et al. (2008) and Choi and Lee (2006). The KMC follows an Arrhenius relationship, and the term KMC/T^2 varies from 1.00–1.70 over the range 1100–1300 °C (Maki et al., 2007). The higher power densities and related thermal gradients in prismatic fuel elements relative to pebble fuel designs can lead to a greater propensity for kernel migration. This is one motivation for using the UCO kernel (with its lower oxygen potential and reduced kernel migration tendency) for prismatic fuel designs, in which thermal gradients tend to be higher than in pebble-bed designs.

$$\frac{\delta_{\text{MIG}}}{T^2} = \int \frac{\text{KMC} \cdot \nabla T}{T^2} d\tau \quad (6)$$

SiC corrosion by fission products and $\text{CO}_{(g)}$

The SiC layer is a structural component in the TRISO particle design and the primary barrier to the release of metallic fission products not retained in the fuel kernel. The SiC layer has been observed to be susceptible to corrosion by rare earth fission products, palladium, and $\text{CO}_{(g)}$, and this can result in particle failure (IAEA, 1997; NRC, 2004; Grübmeier et al., 1977). The nature of the interaction of fission products and gases with the SiC layer can depend on the internal structure of the TRISO particle, specifically, whether or not the SiC layer is exposed due to IPyC cracking (Minato et al., 1991; Hunn et al., 2016).

Palladium is a high-yield fission product that does not readily form a stable oxide or carbide, as such it is released from both UCO and UO_2 TRISO fuel particles. Though in low-enriched UO_2 TRISO fuel, the formation of noble metal inclusions was observed

to stabilize a small fraction of the palladium inventory in the kernel (Tiegs, 1982). The palladium released from the kernel has been observed to interact with the SiC layer in different ways depending on the internal structure of the TRISO particle.

The depth of palladium penetration into the SiC for general SiC interactions has been observed to be temperature dependent, following an Arrhenius relationship (Tiegs, 1982; Stansfield et al., 1983), and related to the amount of palladium released from the kernel (Minato et al., 1990). The Pd/SiC interaction can occur grossly over the circumference of the IPyC/SiC interface, or it can be isolated to specific regions of the SiC layer (which is most damaging). Gross palladium interaction effectively thins the SiC. Temperature gradients across the TRISO particles can also influence general palladium interactions with a preference of palladium pile up and penetration on the “cold” side of the particle subjected to a temperature gradient (Minato et al., 1990).

Through-layer SiC failures are frequently associated with localized palladium corrosion (Minato et al., 1990; Hunn et al., 2016). A holistic understanding of the relationship between IPyC cracking and localized corrosion was developed from the multiscale PIE approach adapted by the U.S. AGR program to confirm the nature of particle failures. The multiscale approach leverages capsule-level analysis, compact-level analysis, and individual-particle gamma analysis in conjunction with micro-X-ray computed tomography (μ XCT) and informed destructive materialography techniques (Hunn et al., 2016). This approach has been effective in isolating failed particles, finding localized failure(s) in the particles, and explicitly determining the responsible failure mechanisms (Hunn et al., 2016; Demkowicz et al., 2015). In the AGR-1 irradiation experiment, this PIE approach confirmed that local through-layer SiC attack by palladium was associated with particle failure and served as a fission product release pathway. The multiscale PIE approach linked incomplete buffer separation from the IPyC to the formation of IPyC cracks that exposed a small region of the SiC layer to higher palladium concentrations and promoted accelerated and localized palladium attack (Hunn et al., 2016). After this sequence occurs and a through-layer SiC failure appears, and the release of fission products typically limited by the SiC layer is possible. Fig. 9 visualizes this failure sequence (Hunn et al., 2016).

Corrosion of the SiC layer by high-yield rare earth elements is observed in TRISO particles with low oxygen content (O/U ratios ≤ 1.1). This O/U ratio is found in TRISO fuel with carbide kernels (e.g., UC_2). However, rare-earth SiC layer corrosion is not readily observed in UO_2 or UCO TRISO fuel compositions of current interest for HTGRs. In high carbide kernels there is insufficient oxygen to stabilize the rare-earth fission products (neodymium, cerium, praseodymium, and lanthanum) as oxides. These rare-earth elements are often co-located with palladium at the SiC layer in carbide kernel TRISO, suggesting similar transport behaviors within the TRISO particle (Homan et al., 1977; Tiegs, 1982).

Corrosion of the SiC layer by $CO_{(g)}$ is a potential failure mechanism in UO_2 TRISO and is not expected in UCO TRISO fuel due to $CO_{(g)}$ gettering. The SiC layer is generally protected by the IPyC layer that prevents excessive $CO_{(g)}$ exposure and corrosion under certain operating conditions. However, the SiC becomes susceptible to global $CO_{(g)}$ corrosion at higher temperatures associated with loss-of-cooling accidents or when the IPyC layer is compromised. Compromise of the IPyC layer protective properties can happen when it debonds or detaches from the SiC layer, possibly generating microcracks, or when macro IPyC cracks are present. The $CO_{(g)}$ corrosion mechanisms preferentially occur at SiC grain boundaries (Gerczak et al., 2020b; Minato et al., 1991), and the $CO_{(g)}$ /SiC reaction results in the production of volatile $SiO_{(g)}$, which transports to the cool side of the particle (Minato et al., 1991).

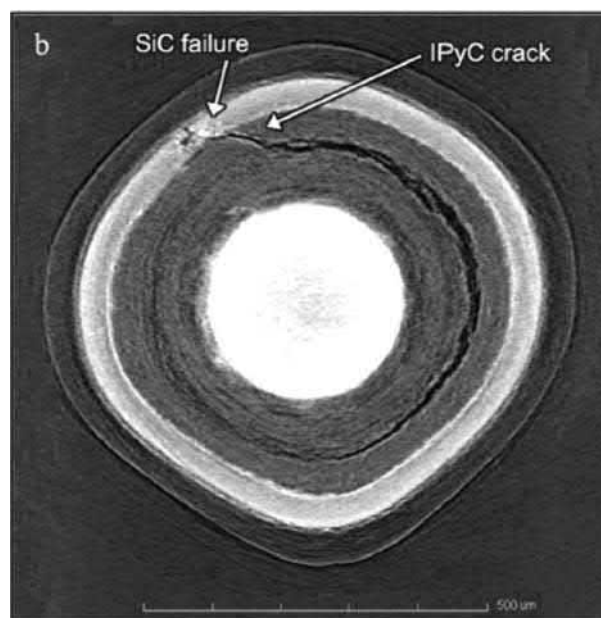


Fig. 9 X-ray tomography of AGR-1 particle with failed SiC from Compact 3-3-1 Particle 1 after 1700 °C safety testing following completion of AGR-1 irradiation test. The image reveals the failure sequence leading to a through-layer SiC failure. Image courtesy of John Hunn, Oak Ridge National Laboratory.

Localized CO_(g) corrosion often precedes mechanical failure of the SiC layer due to weakening and thinning of the SiC (Minato et al., 1991; Morris et al., 2018). The corrosion of SiC by CO_(g) is dependent on temperature, as evidenced by the increase in CO_(g) corrosion driven particle failures observed at increasing safety testing temperatures (Hunn et al., 2018b; Morris et al., 2018).

Thinning of SiC by palladium or CO_(g) corrosion increases the risk of particle failure via stress from internal pressurization (Miller et al., 2006). This has been observed empirically (Minato et al., 1991). The mechanistic fuel performance model, PARFUME, has been used to predict SiC layer fracture probability in response to different thinning scenarios which may occur by palladium and CO_(g) corrosion (Miller et al., 2006).

Summary

The identified particle failure mechanisms described in this article highlight the integrated nature of the TRISO fuel design. The many potential particle failure mechanisms described herein are well-understood. The efficacy of incremental design improvements made during the evolution of TRISO fuel have been verified in worldwide test irradiations and post-irradiation examinations. The irradiation testing highlights the importance of (1) controlling kernel chemistry to limit gas pressure and fission product transport, (2) ensuring the debonding of the buffer layer from the IPyC during irradiation, and (3) providing strong and crack free bonding of the IPyC to the SiC layer. In addition, as-fabricated particle defect fractions and particle damage during the fuel forming process must be held to an absolute minimum, and the irradiation stability of the matrix material must be high with minimum potential for cracking that may lead to externally-induced particle failure.

The understanding of the interconnected response and role of each TRISO particle component during irradiation has led to particle designs and fabrication approaches that have allowed modern TRISO fuel for HTGR applications to exceed performance targets for prescribed operational ranges. Failure rates better than of $\leq 5.1 \times 10^{-5}$ at 95% confidence have been obtained, thus proving the attainability of a goal of high-quality fuel performance under irradiation (Demkowicz et al., 2019). The achieved reduction of the fraction of particle failures in-pile elevates the importance of manufacturing high-quality TRISO fuel with low-defect fractions to ensure appropriate performance metrics are met (i.e., limit fission product release). The decades of TRISO fuel development and demonstration have shown the possibility for acceptable performance of TRISO fuel in HTGR reactors designed to realize the benefits of passive safety, industrial process heat applications and high thermal efficiencies for power production.

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Irradiation Performance: Research Reactor Fuels

James I. Cole, Hakan Ozaltun, Jan-Fong Jue, Warren F. Jones, Dennis D. Keiser, Jr, Barry Rabin*, and Mitchell K. Meyer, Idaho National Laboratory, Idaho Falls, ID, United States

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Introduction

This article serves as a companion to the article on research reactor fuel design and fabrication (Cole et al., 2021). That companion chapter defines and describes terms and fuel design features referred to in this article, such as plate-type fuel, monolithic fuel, dispersion fuel, fuel matrix, and fuel forms used in research reactor fuel (U_3O_8 , U_3Si_2 , U-Mo). The purpose of this article is to inform the reader of the unique set of requirements that research reactor fuels must meet and the synergistic relationship between fuel system irradiation performance, reactor performance, and operational safety. These requirements and their criteria constrain the fuel system choice to a handful of design options having specific characteristics and properties that enable research reactor designers and operators to successfully execute their unique missions. In particular, this article will focus on the performance attributes of plate-type fuel systems in high-power research reactors where the fuel margin to failure is reduced and understanding irradiation performance and failure behavior is most critical. Many lower power research reactors utilize the same fuel systems but operate significantly below the fuel safety limits. Reviewing fuel performance considerations for the high-power applications will best illustrate the phenomena of interest and challenges for research reactor fuel designs in general.

To this end, rather than organizing this discussion around specific fuel materials and cataloguing the irradiation behavior of each specific fuel type, which has been done in several excellent and comprehensive published reviews (Hofman and Snelgrove, 1994; Meyer et al., 2020; Kim, 2012), generic performance requirements will be emphasized, and the fuel system attributes needed to meet these requirements will be described. These descriptions are provided in the context of fuel system development and qualification to obtain regulatory approval for subsequent deployment to new or existing research reactors in the United States. Throughout the discussion, the impact of fuel degradation under irradiation on fuel integrity will be emphasized. Because the recent emphasis in the United States has been to qualify a new low-enriched uranium (LEU) U-10Mo (all alloy compositions are in weight percent) monolithic plate-type fuel, many of the examples provided will address that fuel design; but the principles illustrated will apply generally to the issues to be addressed in qualifying any fuel design (demonstrating acceptable fuel performance under design-basis operation and demonstrating sufficient knowledge of phenomena to adequately predict lifetime irradiation behavior). Finally, a general overview of the methodology used to test and qualify new fuel systems for use in research reactors will be described.

Research reactor fuel performance requirements

A fuel system must meet a set of performance requirements to ensure that a given reactor can effectively and safely meet its mission envelope. In the development and regulatory qualification of fuels for research reactors, data is collected to demonstrate that the fuel system meets these established performance requirements (Anon., 1988, 2009, 1996). Specific requirements are established to

*Retired.

ensure that the fuel maintains the following overall irradiation performance attributes throughout its service life. The requirements identified for research reactor fuel, in broad terms, are the following:

- Mechanical integrity—no cladding breach occurs under normal operating conditions or anticipated transients such that the cladding retains fission products.
- Geometric Stability—ability to maintain adequate system cooling under normal reactor operating conditions and expected transients, mainly by maintaining the coolant-flow channel gap (spacing between fuel plates).
- Stable and predictable behavior—the fuel exhibits consistent, predictable fundamental irradiation behavior over the anticipated range of operating conditions.

The fabrication processes utilized must produce a fuel system that encapsulates the fuel material and prevents the release of fission products to the environment (Cole et al., 2021). The cladding itself must resist corrosion and an excessive buildup of oxide on the surface under operational conditions (oxide will impede heat transfer from the fuel plate and could lead to exceeding fuel design temperatures). Additionally, the stress state of the fuel system and its implication on overall fuel system performance must be understood. Compressive and tensile residual stresses can build up in the fuel plate due to the heating and cooling of materials with different thermal expansion behavior, resulting in the fuel plate being nearer to the failure limits than anticipated. The following sections consider each of the major irradiation performance requirements in more detail and provide examples of testing, examinations, and modeling that are utilized to demonstrate that the fuel system meets those requirements.

Mechanical integrity

The robust structural integrity of the cladding ensures that the fuel does not breach under normal operating conditions and design basis transients, ensuring the fuel plate retains hazardous radionuclides. Mechanical integrity under normal operating conditions is demonstrated primarily through the successful testing of fuel plates up to the limiting conditions established by the specific reactor's fuel performance and safety analyses. This includes an adequate margin to cladding failure (or breach) supported by analysis using data from comprehensive microstructural characterization and post-irradiation examination (PIE). Maintaining mechanical integrity under anticipated transients is primarily demonstrated through blister threshold testing (details will be described later), which demonstrates the limiting fuel temperature to be used in safety analyses of reactors. The mechanical integrity of the fuel-plate system is evaluated both experimentally and with simulations using finite element analysis (FEA). Five criteria are used to demonstrate that the fuel system maintains adequate mechanical integrity under irradiation:

- Physical and mechanical properties related to fuel integrity for all constituents are known (melting point, modulus, strength, etc.).
- The fuel plate will not delaminate during normal operating conditions and anticipated transients.
- Mechanical responses of fuel core (uranium-bearing regions of fuel system), cladding, and any interlayers during normal operations and anticipated transients is established.
- Limits for fabrication defects are established.
- Water corrosion limits for the cladding are established.

Fuel performance modeling of thermo-mechanical behavior

It is widely recognized that fuel performance modeling can provide valuable information that would be infeasible, difficult, or cost prohibitive to obtain experimentally (Rest, 2012; Rabin et al., 2017). For instance, fuel performance modeling allows investigation of the fuel's sensitivity to variations in key geometric, operational, and materials property variables to improve the overall understanding of fuel behavior. Fuel performance modeling also provides valuable input for establishing manufacturing defect limitations, produces information useful for interpreting characterization and PIE results, and allows an assessment of important fuel phenomena, such as geometric changes, operational limits, and fuel failure mechanisms.

This understanding of the fuel mechanical response is established through thermomechanical modeling, incorporating the best available materials properties measurements and behavioral models. Behavioral modeling of fuel plates is a multi-physics problem, involving the integration of several tightly coupled phenomena. This multi-physics coupling inherently necessitates a consideration of many parameters to assess the overall fuel performance. Stresses, strains, deformations, and temperatures are evaluated together to provide an accurate performance assessment of the fuel plate throughout its service life.

Sensitivity studies are conducted to isolate the effect of individual variables on overall irradiation performance. Parameters representing various geometric, operational, and property-related variables consider a wide range of design and operational constraints required by the specific reactor. Geometric parameters include fuel thickness, shape of fuel corners, fuel offset, fuel curvature, etc. Operational parameters include cooling rate, fission profile, mechanical constraints, etc. And finally, property related parameters include thermal properties degradation, swelling, creep, etc. These parameters are comparatively evaluated to determine the most influential parameters, while studying the performance of plates for what-if scenarios.

To accurately assess the integrity of the fuel, the thermal and mechanical responses of the monolithic U-10Mo fuel have been modeled over the entire lifecycle of the fuel plates to account for cumulative effects. Lifecycle stages include foil fabrication, plate

bonding, irradiation, and post irradiation testing. The fidelity of model output can be further refined through continual validation using experimental fuel performance data derived during testing and PIE (e.g., fission gas bubble distributions and dimensional changes as a function of burnup). The reduction of uncertainties with a growing body of critical parameter measurements increases the ability of the models to evaluate conditions under which fuel plate failure is likely to occur. Focused evaluation of critical properties identified by the modeling provides an efficient method of streamlining irradiation performance assessments.

As an example of the utility of fuel performance modeling, Fig. 1 illustrates the formation of a pillow in an irradiation-tested U-10Mo fuel miniplate (i.e., a small but prototypic fuel-plate design used for irradiation testing). Along with the image of the fuel plate are images that were produced using the FEA simulation at end of life (EOL), following reactor shutdown, demonstrating that temperatures stresses and fission-gas-induced porosity are greatest adjacent to the region exhibiting the pillowing behavior, which suggests that the genesis of the behavior is a combination of factors including stress state, and porosity distribution (Medvedev et al., 2018). Comparative assessment of modeling results with experimental observations can provide a more comprehensive and mechanistic understanding of irradiation performance that experiments alone may fail to reveal.

Cladding corrosion

Limits on fuel fabrication defects are based on experimental irradiation performance data and theoretical calculations of oxide growth and cladding corrosion resulting from an assumed single-sided debond condition (i.e., separation of the cladding plate from the fuel core) that greatly reduces heat transfer from the debonded face of the plate. The use of aluminum-clad research reactor fuel in water leads to the formation of a corrosion product layer with low thermal conductivity. The growth rate of the layer varies with reactor operational parameters (plate power and coolant flow) and reactor-specific water chemistry. If the fuel plate power and coolant channel flow are not properly matched in high-performance reactors, the corrosion layer thickness can increase rapidly. If the corrosion layer thickness is allowed to exceed a point where the temperature gradient across the layer is greater than approximately 119 °C, oxide spallation is likely to occur, exposing fresh aluminum to further rapid oxidation (Pawel et al., 1990; Rabin et al., 2017). Extended exposure to conditions where spallation occurs may result in the complete oxidation of the aluminum cladding. If an oxide surface layer does not spall and grows locally to a sufficient thickness to increase the fuel temperature beyond the blister threshold temperature, blistering may occur.

Geometric stability

The geometric stability of the fuel is essential to preventing coolant channel blockages through an increase in fuel plate thickness or through the mechanical collapse of a coolant channel. Five criteria must be met to demonstrate geometric stability.

- Geometry of the fuel must be maintained during normal operating conditions and anticipated transients.

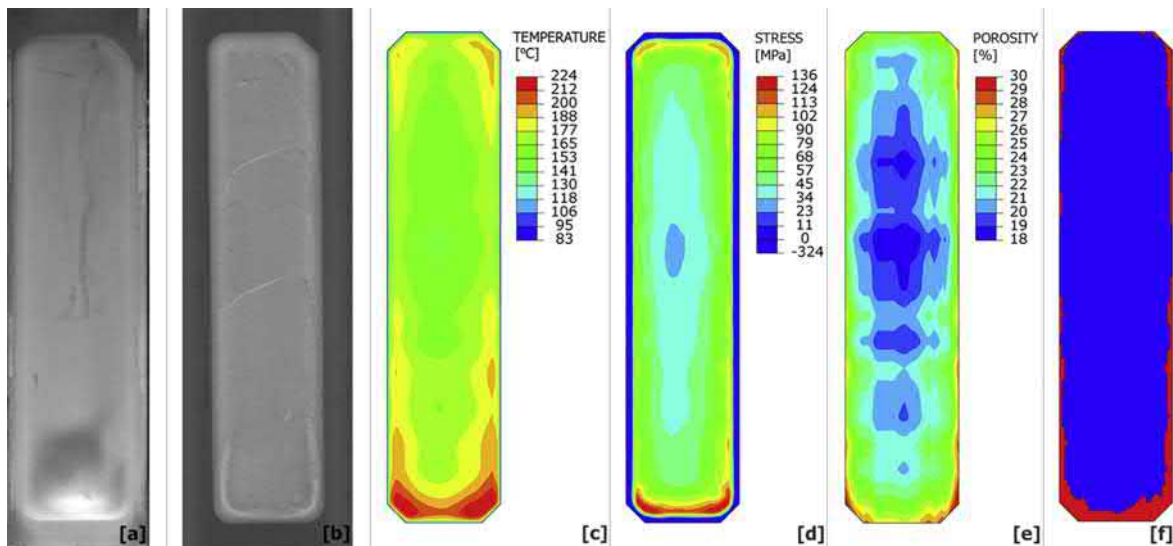


Fig. 1 Irradiated LEU U-10Mo monolithic miniplate images showing: (a) a pillow-type failure (b) a neutron radiograph of the failed plate along with the FEA model output of (c) the fuel centerline temperature at end of life (d) the maximum principal stresses at shutdown (e) the end-of-life fuel porosity, and (f) regions with porosity interconnection. Adapted from Medvedev PG, Ozaltun H, Robinson AB and Rabin BH (2018) Shutdown-induced tensile stress in monolithic miniplates as a possible cause of plate pillowing at very high burnup. *Nuclear Engineering and Design* 328: 161–165.

- Data must be documented for material properties that affect geometric stability and thermal-hydraulic performance as a function of increasing fuel burnup.
- Fuel performance and structural stability must be maintained so that reactor-coolant flow maintains fuel plate heat transfer and/or temperatures within the reactor safety envelope.
- Plate movement caused by the hydraulic pressure differential must not compromise ability to cool the fuel.
- Changes in channel gap dimensions must not compromise the ability to cool the fuel plate.

The fuel plates are typically assembled in highly compact fuel assemblies with narrow coolant channels and high flow rates. For this reason, any features that develop in the fuel plate that restrict this flow can lead to plates overheating (resulting in excessive oxide buildup) and breaching. Thus, it becomes critical that the fuel maintains geometric stability under these design conditions and resists any tendency to undergo unanticipated dimensional changes due to imposed thermo-mechanical forces.

During irradiation, coolant channel gaps must be maintained within as-designed tolerances to ensure that fuel plates are adequately cooled. Changes in coolant channel gap width may be caused by flow-induced stress, relaxation of stress introduced into fuel plates during fabrication, blistering, pillowing/delamination, or breakaway fuel swelling. These behaviors are to be avoided because they can compromise the ability to cool the fuel (in addition to leading to cladding beach).

Aluminum-clad, plate-type research reactor fuel fails at elevated temperatures through the formation of small, raised pockets of fission gas termed as “blisters,” which result in a geometry change that can impede coolant flow. Therefore, blister threshold data can be used to establish safe operating temperature limits during both normal and off-normal operating conditions. Blister threshold testing is a destructive process, usually performed in conjunction with PIE of tested fuel plates. The test is typically performed on full, intact plates and can indicate the presence of defects, high levels of porosity in the fuel, or an agglomeration of fission gas bubbles. During the testing, the fuel plate is heated to an elevated temperature (typically between 300 and 550 °C) in specific increments, holding at each temperature for a fixed period of time (~20 min) and then visually examined to detect the formation of blisters on the fuel plate surface. Results of this test are used to assess margins to failure for various accident scenarios analyzed in reactor-specific safety bases and to ensure that the geometry is maintained under anticipated transient conditions.

A more detailed, quantitative examination of out-of-pile fuel plate stability can be accomplished utilizing a testing apparatus such as a generic test-plate Assembly (GTPA) (Rabin et al., 2017). The GTPA is a modular mechanical assembly that can be used to house different types of instrumented test plates, in a thermal-hydraulic test apparatus, such as the Hydro-Mechanical Fuel Test Facility (HMFTF) located at Oregon State University in the United States (see Fig. 2). Test results can be used to compare the structural characteristics and dynamic response of plates produced through different fabrication methods (e.g., roll bonding and hot isostatic press (HIP) bonding), different cladding materials or alloy heat treatments, and different fuel types (e.g.,

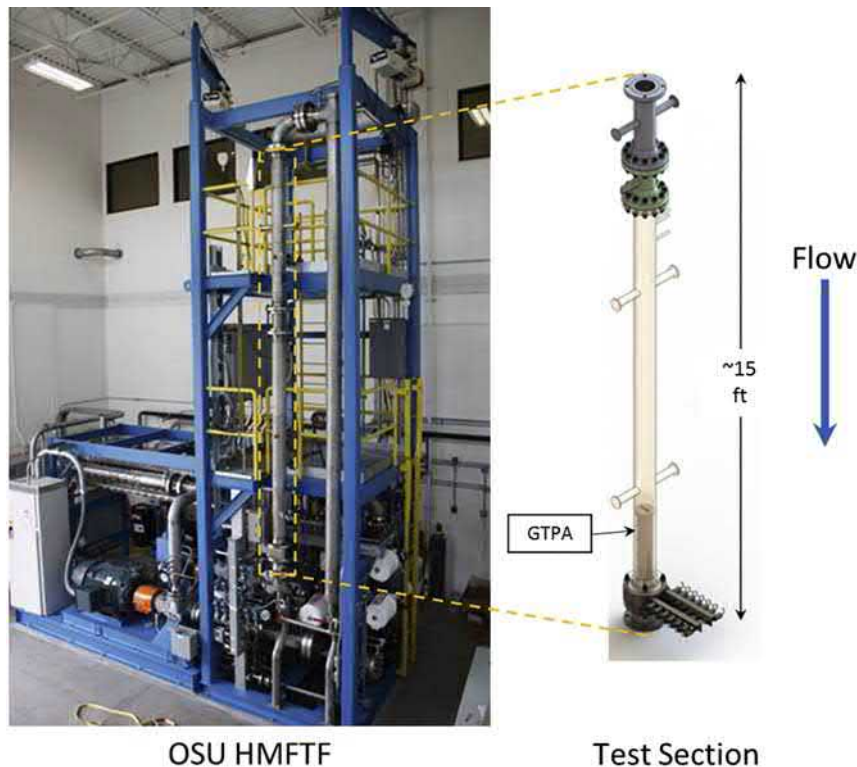


Fig. 2 Oregon State University's Hydro-Mechanical Fuel Test Facility (HMFTF) and test section with the GTPA location indicated. Courtesy Professor Wade Marcum, Oregon State University.

monolithic or dispersion). The plates are tested at selected flow rates under sub-cooled fluid-state conditions that represent reactor-coolant temperatures and pressures.

The objectives of the tests are to:

- Identify the flow conditions that induce the onset of elastic plate deformation for each fuel plate material.
- Identify the approximate flow conditions necessary to produce an ultimate plate failure in either a static (large plastic deformation) or dynamic (plate vibration) state.
- Provide test data for validation of computational models of the static and dynamic response of plates and plate assemblies to flow, enabling the modeling of more complex systems.

Understanding the hydraulic flow behavior of research reactor fuel in prototypic elements is also critical to ensure that the fuel system behaves in a predictable manner and is not susceptible to vibration-induced resonance and gross thermo-mechanical deformations when placed in service. The models established to predict this behavior can also be used to aid in the design of prototypic experiments for the regulatory qualification of new fuel systems as they are developed and tested.

Stable and predictable behavior

Research reactor fuel plates must operate in a stable and predictable manner, despite perturbations in operating conditions. The key phenomena that can impact the dimensional stability and robustness of a fuel plate during irradiation are fuel swelling (due to fission product accumulation, phase transformation to less-dense phases, or radiation-enhanced diffusion) and fuel relocation resulting from irradiation-induced creep. Breakaway swelling occurs in some fuels when fission gas bubbles coalesce to form large gas pockets, which under decreased surface tension easily expand and cause a rapid increase in fuel-plate thickness upon further use. Large areas on the fuel plate with increased thickness due to “blistering” or “pillowing” are unacceptable due to the potential to impact mechanical integrity or coolability. Pre- and post-irradiation dimensional measurements and detailed microstructural characterization are the primary tools used to document the evolution of fuel condition during irradiation and the formation of microstructural features that indicate the onset of unstable behavior.

Because the fuel core is physically bonded to the cladding (or to fuel-cladding interlayers, for some designs) to improve heat conduction, it is critical for the fuel to behave in a predictable manner. Pure uranium has highly anisotropic properties at room temperature due to its orthorhombic crystal structure. The impact of this anisotropic structure can manifest with increasing fuel burnup as non-uniform dimensional changes. The addition of alloying elements to the uranium can stabilize a body-centered cubic (bcc) fuel phase (e.g., Mo) that has isotropic behavior; such stabilization is preferred in monolithic fuel designs, which do not have the surrounding Al matrix of a dispersion fuel to reduce the impact of individual fuel particle swelling on the behavior of the bulk fuel core.

Five requirements have been established to demonstrate stable and predictable behavior:

- Fuel performance is known and predictable for the expected range of fuel composition and impurities, processing parameters, and microstructure for all credible environmental and irradiation conditions.
- Physical properties and fuel attributes that are important for the analysis of fuel stability under irradiation are documented.
- Fuel swelling behavior is within the stable swelling regime (linear and predictable).
- Compatibility of fuel with reactor coolant is established to cover the case where a small cladding breach may occur under normal operation conditions.
- Irradiation behavior on scale-up (geometric and fabrication) is predictable.

In terms of quantifying the accumulated irradiation damage within fuel specimens, the primary parameter of interest is the percent burnup (i.e., the relative number of initial fissile atoms consumed), which is frequently and more conveniently expressed in terms of fission density (fissions cm^{-3}). Fission density removes the bias inherent in ^{235}U burnup values due to uranium enrichment or heavy metal atom fissions. For that reason, fission density allows a more direct comparison of burnup levels between highly enriched uranium (HEU) and LEU fuels. Therefore, fission density is often considered a more relevant fuel irradiation performance measure than burnup and is a critical operating parameter that defines the lifetime of the fuel as well as an important consideration for reactor fuel management.

Irradiation-induced dimensional changes

During irradiation, the swelling of the fuel core occurs due to an accumulation of fission products and, to a much lesser extent, irradiation damage. The swelling behavior of the fuel material is a defining attribute of the fuel form that determines its suitability for use in a specific nuclear application. Moderate amounts of swelling can be accommodated in design, provided the swelling behavior is stable and predictable for the range of fuel operating conditions and provide sufficient operating margin.

The fission products responsible for fuel swelling can be broken into two general categories: solid and gaseous. While swelling due to solid fission products is relatively easy to project and is relatively stable, swelling due to gaseous fission products can manifest in a variety of ways and is often the source of nonlinearity in the swelling behavior. For dispersion fuels, the swelling is largely accommodated by the presence of a ductile Al matrix. In the case of U-Mo alloy fuels (both monolithic and U-Mo dispersion in

an Al matrix), the incorporation of fission gases within U-Mo leads to a precipitation of discrete nano-scale fission gas bubbles dispersed throughout the fuel. Gamma-phase U-Mo exhibits unusual behavior wherein these nano-scale fission gas bubbles form a three-dimensional, highly ordered fission gas bubble superlattice that imparts this phase with excellent fission gas retention characteristics and stability against fission gas bubble coarsening. As burnup increases, microstructural changes due to polygonization/recrystallization, lead to the formation of discrete μm -scale gas bubbles, initially located along grain boundaries, and later dispersed through the fuel.

As fission gas production in the fuel increases with burnup, most fuel materials have a tendency towards an accumulation of the gas into the μm -scale bubbles that eventually interconnect and lead to swelling that increases more rapidly with further increases in fission density. Breakaway swelling is a term that has been used historically to define a rapid or unpredictable transition to high swelling behavior following an initial regime of stable fuel behavior. If the fuel system exhibits breakaway swelling, which has sometimes been seen in dispersion fuels, a fission density threshold that defines the onset of this breakaway swelling must be identified during the fuel development effort. Examples of fission gas bubble development in a U-10Mo and U_3Si_2 fuel respectively are illustrated in Fig. 3. The as-fabricated U_3Si_2 fuel particles shown in Fig. 3b have become amorphous during irradiation.

The readers should note that although frequently used interchangeably, “fuel swelling” and fuel plate “thickness change” are not the same and involve different phenomena. Swelling is an increase in volume that occurs isotropically (in its thickness, length and width) due to the formation of gaseous and solid fission products. However, in plate-type fuel, fuel deformation occurs primarily in its thickness due to the mechanism of irradiation-induced creep, which results in strains due to mass relocation away from the constrained edges as the fuel swells. The observed plate thickness changes are a superposition of both volumetric fuel swelling and irradiation-induced creep strains. Measured local thickness changes historically used to calculate strains are often referred to as fuel swelling; however, the measured thickness change always contains contributions from both irradiation-induced creep and volumetric swelling.

The stresses that develop due to constraints imposed by the cladding produces fuel creep. The resultant material relocation and stress relaxation can influence the microstructural evolution including recrystallization, grain subdivision, and gas bubble growth, as the internal stress state is altered. This synergistic interaction between the evolving microstructure and the creep deformation behavior has an impact on all fuel performance parameters, including mechanical integrity, geometric stability, and stable and predictable behavior.

Microstructural changes

A large part of functional fuel design is to allow heat removal as efficiently as physically possible. Changes in the fuel microstructure can reduce the effectiveness of heat transfer (by reducing thermal conductivity through the fuel), contribute to an embrittlement of the fuel, consequently affecting its mechanical integrity, and introduce phases that have unstable and unpredictable behavior (e.g., break-away swelling). The fuel temperatures in research reactor fuels is low enough that many of the microstructural processes occurring in the fuel are being driven by the creation and evolution of the radiation-induced defect structures in the absence of a significant thermal component. The fuel microstructure evolves with the creation of a high density of fission gas filled gas bubbles, solid fission products and crystal defects such as radiation-produced dislocation structures. As more damage accumulates, fuel materials can lose long-range order and become amorphous, which is most typical for intermetallic compounds. Amorphous materials inherently have a greater free volume, and gas diffusion in this structure is accelerated. So, as a general rule, the resistance to bubble formation and unstable growth in an amorphous fuel is lower than in a fuel that remains crystalline. Amorphization occurs at lower fission density in oxide (U_3O_8) and intermetallic fuels (UAl_x , U_3Si and U_3Si_2) compared to an alloy fuel, such as U-Mo, and the loss of crystallinity can be more rapid with an increasing volumetric power density. This characteristic, along with a relatively high uranium density, has motivated the development of U-Mo fuel for research reactors.

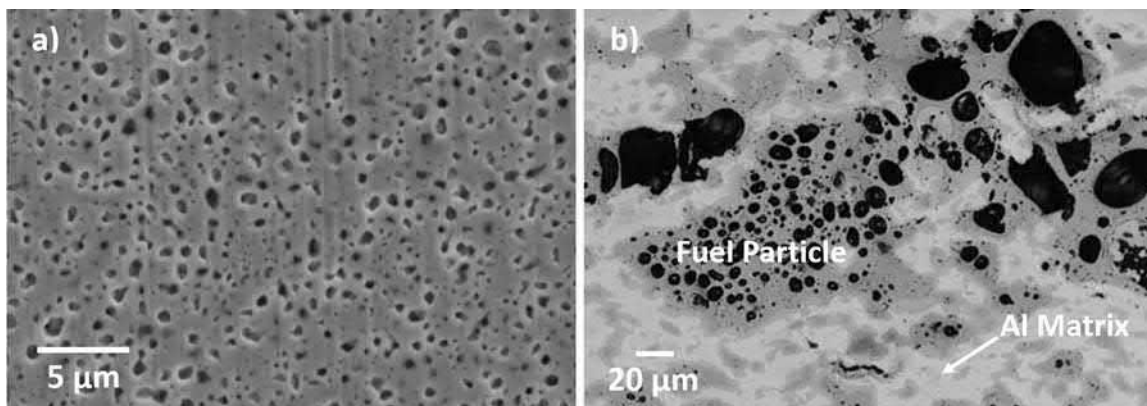


Fig. 3 Scanning electron microscope images of fission gas bubble formation in (a) a monolithic U-10 Mo fuel and (b) a U_3Si_2 fuel at moderate burnup levels ($< 5 \times 10^{21}$ fissions cm^{-3}).

For U-Mo alloys, as the number of crystalline defects increase with fission density, the thermodynamic driving force is for the microstructure to reconfigure into a lower energy state. This results in the subdivision of the grains into smaller units or “cells” with the rearranged dislocation structures forming the grain boundaries, similar to behavior commonly seen in materials that are heavily cold-worked (i.e., mechanically deformed) in metal fabrication applications. This refining of the microstructure is often correlated to a loss of stable microstructural evolution because the increase in grain boundaries enhances the nucleation rate of fission gas bubbles and subsequent growth.

Fuel-chemical interactions

Nearly all the fuel compounds have some type of interaction with an Al matrix material and the Al alloy cladding. Under irradiation, particularly at highest powers and temperatures, this leads to the formation of distinct interaction phases that can consume a significant fraction of the Al matrix surrounding the fuel particles. For HEU dispersion fuels, with lower fuel particle density than is necessary for LEU dispersion fuels, Al matrix can accommodate a much larger fraction of the interaction phase without compromising the overall irradiation performance of the fuel at high fission density. For high-density LEU fuels, that have volumetric particle densities approaching 50%, the consumption of the Al matrix with a phase that is frequently brittle and susceptible to high rates of bubble growth can lead to an interconnection of porosity across the entire fuel zone; ultimately this phenomenon can lead to fuel failure as large voids are formed that reduce the mechanical integrity of the fuel core and inhibit heat transfer across the fuel. Images of a U-7Mo fuel before and after irradiation are shown in Fig. 4. The post irradiation microstructure shown in Fig. 4b reveals that, during irradiation, spherical fuel particles (light contrast) react with the aluminum matrix (dark contrast) to form an interaction phase (gray contrast) between the fuel and the aluminum matrix.

Research reactor fuel testing

Irradiation

As indicated previously, during the development and qualification of new fuel systems, testing to bounding conditions and beyond is conducted to ensure that the irradiation performance of the fuel meets the requirements necessary for use in research reactors. This testing establishes fuel failure limits, provides data to reactor safety analysts concerning the overall irradiation performance envelope, and generates data needed to develop and validate fuel performance codes. For the qualification of the fuel, a typical test series will encompass subsize or mini fuel specimens (miniplates) to obtain the basic irradiation performance data of the fuel material. This is followed by scaled-up fuel plate testing that is more prototypic of the actual fuel plate dimensions and is used to discern significant differences in irradiation performance in going from the smaller to larger fuel samples. Finally, the fuel is tested in a prototypic element configuration to ensure that fuel plates, mechanically constrained in an element, do not undergo unforeseen thermo-mechanical or thermal-hydraulic behavior under prototypic operating conditions.

The size of the fuel plate specimen is an important parameter in determining the test objectives. A larger quantity of miniplates can be irradiated more economically using a wider range of reactor operational parameters. Miniplates have generally been used for tests when several design and environmental variables are under investigation, when emerging fuel fabrication technologies or equipment have not yet been scaled up to prototypic size, or when a large quantity of specimens needs to be tested to suit data objectives. A schematic of a typical irradiation capsule for testing multiple miniplates in the Advanced Test Reactor (ATR) is provided in Fig. 5. Several capsules are stacked in a vertical column inside of a basket that mates to a specific reactor test position.

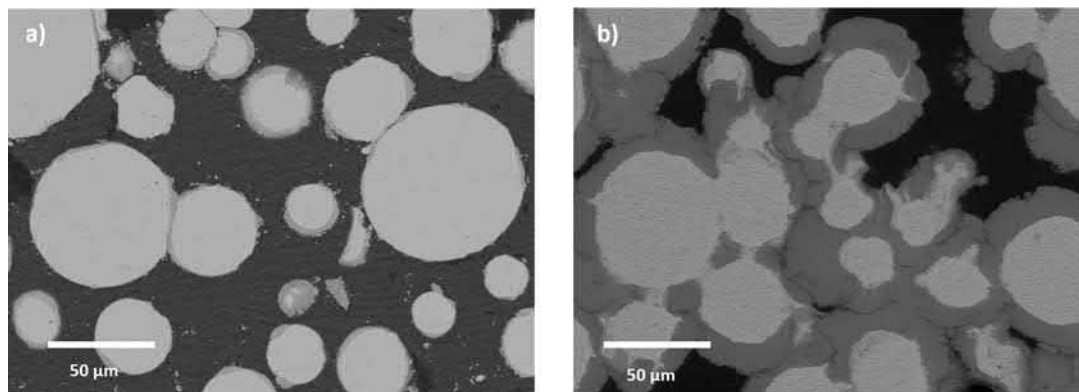


Fig. 4 Scanning electron microscope images of a U-7Mo fuel (a) as-fabricated and (b) following irradiation, demonstrating the formation of extensive interaction phases between the fuel and the Al matrix. The light contrast spherical features are fuel particles, dark phase is the Al matrix and the gray regions are interaction phase.

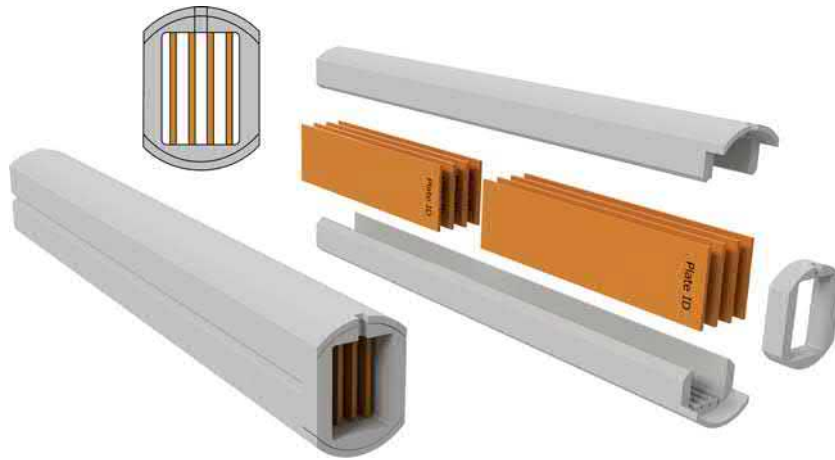


Fig. 5 Miniplate test capsule assembly containing eight plates per capsule.

Because the larger engineering-scale behaviors of fuel plates may not be readily exhibited at the miniplate scale, larger-scale tests are used to determine how performance differs with plate size. Fabrication processes necessary for larger fuel plates may result in microstructural characteristics that differ from those of miniature fuel plate, and larger specimens also provide a vehicle for determining fuel behavior and reliability for fuel specimens produced by scaled-up fabrication processes. Large-size specimens are generally irradiated in smaller quantities due to space constraints in test reactors.

After substantial data has been collected using large-size plates and miniplates prior to fuel element demonstrations to ensure a low risk of failure, plates are assembled into fuel elements for demonstration and qualification of the scaled-up fuel design. Fuel element demonstration testing requires that a repeatable pilot-scale fabrication process is in place and qualified. Element testing provides a prototypic environment for testing fuel elements that represent, as closely as possible, axial, radial, and azimuthal power shape and flow conditions of the test reactor in which the elements are being tested. Element testing for research reactor fuel is similar to a lead test assembly (LTA) used in light-water reactors to demonstrate performance of a new fuel design in a specific reactor. However, frequently when developing conversion elements to convert from HEU to LEU, it is not possible to test a developmental element in the end-use research reactor. In this case, design demonstration elements (DDE) are tested in a separate reactor that has prototypic design features but may not represent the exact dimensions of the actual fuel element. Test conditions are matched to the reactor being converted as closely as possible at the element level. Test conditions for each plate are defined by the element test environment, so it is not possible to have independent test conditions for each plate within the element.

Post-irradiation examination

PIE is conducted on fuel plates to assess the irradiation performance (as indicated in post-irradiation measurements and inspection) and generate the data needed to support fuel qualification and reactor safety analysis. PIE is typically separated into two phases; non-destructive examination (NDE) and destructive examination (DE). For NDE, fuel plates remain intact and interrogation methods include visual examination, neutron radiography, high-resolution profilometry, immersion density and scanning gamma spectroscopy. The NDE techniques are focused on quantifying the extent of bulk and local dimensional changes; the observation of any evidence of fuel failures, including formation of blisters (visual, profilometry, immersion density), pillows, or cladding delamination; the extent of cladding oxidation and initial estimation of fuel burnup and fission product distribution (scanning gamma spectroscopy). **Fig. 6** provides a representation of the measurements made during high-resolution profilometry to evaluate fuel swelling and other dimensional changes. Neutron radiography is used to evaluate whether the fuel has remained in-tact during irradiation and to identify fuel fracture, relocation, or other anomalous behavior that occurred within fuel material (the Al cladding is effectively transparent to neutrons). In combination, all these techniques are used to identify features of interest that can inform and focus subsequent DE investigations.

DE investigations are conducted to evaluate the fuel microstructure, further refine burnup measurements, obtain blister threshold temperature, and, in the case of monolithic fuel types, measure mechanical property changes of the fuel core. Samples generated during sectioning for destructive examination are also used to make thermophysical property measurements to ascertain changes in its thermal conductivity during irradiation. Because of the highly radioactive nature of irradiated fuel specimens, localized regions of the fuel plates are typically sectioned and reduced in size to perform more detailed microstructural and microchemical characterization utilizing analytical scanning and transmission electron microscopy. In this way, a picture of the microstructural changes can be mapped across length scales from bulk properties to the atomic scale, providing a detailed understanding of how fuel property degradation under irradiation impacts overall irradiation performance. During destructive exams, full sized plates are also tested to determine the blister threshold temperature. If appropriate, fuel plates are heated beyond the melting temperature to

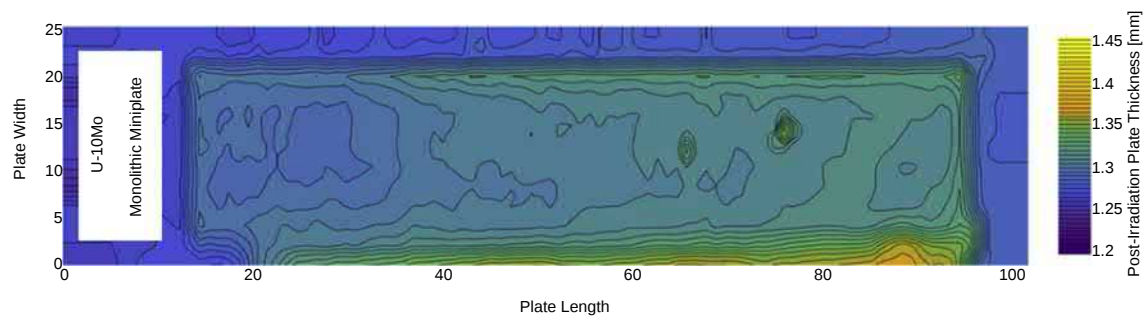


Fig. 6 Topographic representation of an irradiated miniplate showing measured thickness changes compared to the as-fabricated measurements. Lighter yellow regions were indicative of a separation of the fuel cladding in that location. These measurements were taken on a U-10Mo monolithic plate following irradiation testing in the ATR in the United States.

evaluate release rates of fission products from the fuel. This information is critical for larger research reactors (such as the ATR) that have a significant fissile material inventory that must be addressed in safety assessments.

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Key Characteristics of Used Nuclear Fuel

J.-P. Glatz, Retired from European Commission, Joint Research Centre, Karlsruhe, Germany

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Abbreviations

CEA Commissariat à l'énergie atomique et aux énergies alternatives
dpa Displacements per atom
EPMA Electron Probe Micro-Analysis
EPRI Electric Power Research Institute
ESCP Extended Storage Collaboration Program
EURATOM European Atomic Energy Community
GEN-II Second Generation
GIF Generation IV International Forum
GWd/tHM Gigawatt day per Tonne Heavy Metal (Nuclear Fuel Burn-Up Unit)
HBS High Burn-up Structure
HLW High Level Waste
IFR Instant Release Fraction
JRC Joint Research Center
LWR Light Water Reactor
MAs Minor Actinides
MOX Mixed Oxide fuel
P&T Partitioning and Transmutation
PWR Pressurized Water Reactor
RE Research entity
RED-IMPACT European Research Project on Impact of Advanced Recycling
R&D Research and Development
SEM Scanning Electron Microscope
SRIM Stopping and Range of Ions in Matter code
XRF X-ray Fluorescence

Introduction

The generation of energy by means of nuclear power has evolved a lot over the years since the 1950s, especially in the last few decades. It became more and more obvious, that the strategy for the management of used fuel as a major part of the waste generated in the nuclear energy programs, is a key issue mainly in view of public acceptance.

In the early phase of nuclear power production (1960s and 1970s), all used fuel was foreseen to be reprocessed. Recovered uranium and plutonium were supposed to be recycled in fast neutron reactors to avoid potential supply shortages of natural uranium. In the 1980s, some countries have decided to reprocess used fuel and recycle uranium and plutonium as so-called mixed

oxide (MOX) and Uranium Reprocessed Enriched fuels in light water reactors (during the same period, other countries have changed their strategy and implemented an open fuel cycle).

Nuclear energy systems of the future, as they were defined by the Generation IV International Forum (GIF), are supposed to provide a sustainable energy generation for the future (Kelly, 2014). The major benefits for used fuel recycling are the conservation of natural uranium resources, reduced dependence on foreign fossil fuel, the reduction of the nuclear waste radiotoxicity and the heat load of repositories. Other big challenges for the implementation are significant costs, safety and increasing proliferation concerns also affecting the public acceptance of these technologies.

The various activities associated with the production of electricity from nuclear fuels are referred to as the nuclear fuel cycle (see Fig. 1). The nuclear fuel cycle starts with the mining of uranium and ends with the disposal of nuclear waste. When opting for the reprocessing route of used fuel fissile materials will be recycled and remaining waste will be vitrified before final disposal.

The current reactors use less than 1% of the uranium available in nature. With such a low efficiency, the uranium resources identified today worldwide might last only about 100 years with the currently installed nuclear power infrastructure. Depending on the growth rate in the future use of nuclear systems, this time span could also be significantly lower. New energy systems using a technology based on the combination of fast neutron reactors with advanced multi-recycling of the fuel would improve the usage of natural uranium resources by at least a factor of 50 and possibly even higher depending on the scenarios adopted.

A comprehensive management of used nuclear fuel from the operation of nuclear power plants is a multistage process, where all steps need to be carried out in an effective, safe and responsible way. The two main management options are final disposal and reprocessing. In the first case, the so-called “open fuel cycle,” the fuel is disposed of without further treatment. In numerous research programs worldwide, involving underground laboratories in potential geological formations (granite, clay salt or volcanic rock) the long-term evolution has been investigated and the common outcome is that under normal evolution scenarios, the dose rate released to the surface is significantly lower than the natural radiation background.

The second option also called “closed fuel cycle,” involves extraction of the energetic components in the used fuel, mainly plutonium and uranium, for reuse. Consequently, in fully closed cycles based on fast reactors and adapted advanced recycling of the used fuel components by anticipated technologies, 50–100 times more energy can potentially be generated from the uranium mined originally. If at the same time a partitioning and transmutation (P&T) would be deployed, the waste could decay to low levels of radioactivity in less than 1000 years in contrast to > 100.000 years for direct disposal of used fuel. However, all of these steps involve additional dedicated up-scaled facilities, and require substantial further research and development before they would be commercially available.

Worldwide, increasing inventories of surplus plutonium are stored in the main nuclear countries, they need to be secured against the risk of diversion. In order to reduce therefore these inventories, several countries including those with accumulated surplus weapons-grade plutonium, have opted for the so-called mixed oxide MOX strategy where plutonium is used as fissile material in the civil nuclear industry. Globally MOX fuel represents today almost 5% of nuclear fuel used. In France about 10% of the

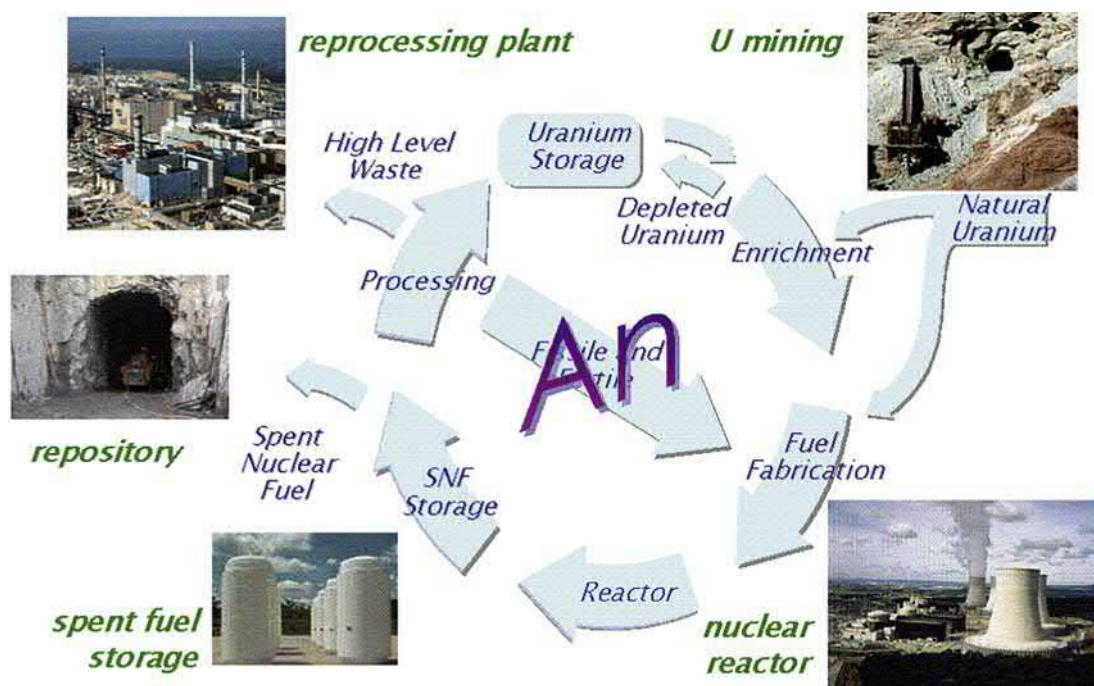


Fig. 1 The nuclear fuel cycle.

fuel in the reactor fleet is MOX. The increased proliferation risk associated with increased separation of plutonium from used fuel leads to controversial public acceptance discussions in many countries.

A common feature of all routes is the need of a more or less long-term intermediate storage of the used fuel discharged from the reactor. A thorough characterization of the fuel and its evolution during storage is essential to guarantee stability and thereby safety for often undefined storage duration. Also for the final disposal option to be implemented later, the characterization helps to understand the evolution of the waste package in the disposal site, including all kind of parameters affecting possible interactions with the corresponding geological formation. This chapter aims to give a detailed description of the key physical and chemical characteristics of fuel during irradiation, and the evolution of used nuclear fuel after irradiation and discharge from reactors. The main focus is of course on UO_2 fuel, however also the MOX fuel case will be discussed.

Nuclear fuel irradiation

Especially at the beginning of the irradiation process, when the fission event density is the highest, leading to a maximum linear power and central temperature, a significant relocation of fission products, depending on their volatility, takes place in the fuel. In fact, in an oxide fuel, temperature gradients of at least 500°C between the fuel periphery ($\sim 500^\circ\text{C}$) and the fuel center ($> 1000^\circ\text{C}$) lead to strong migration and diffusion to the fuel pellet rim. The grain structure of the fuel initially pressed from UO_2 powder, induces a precipitation of some of the fission products at the grain boundaries, noble elements partially form metallic precipitates. The most volatile elements, i.e. in addition to the fission gases xenon and krypton, mostly cesium, tellurium and rubidium partially migrate towards and outside of the fuel pellet edge where they are deposited as elements or potentially form compounds, also with the cladding material. They can also be integrated in an oxide layer formed inside the zircaloy cladding since the very beginning of irradiation. Parts of the volatiles, above all fission gases are found in the fuel rod plenum.

The irradiation process leads to the formation of numerous radionuclides by fission of ^{235}U . As irradiation goes on, more and more fissile ^{239}Pu is formed by neutron capture of ^{238}U and higher actinide isotopes are consequently formed by a cascade of neutron absorptions. At the same time the high temperature and a steadily increasing radiation field building up upon progressing irradiation is responsible for significant thermophysical modifications of the fuel (Kleykamp, 1985). In the very high temperature gradient of typically at least 500°C between the pellet center ($> 1000^\circ\text{C}$) and the pellet periphery ($\sim 500^\circ\text{C}$), many fission product elements, depending on their volatility and thermos-physical properties migrate to the colder outer regions of the fuel and exhibit totally different behaviors.

In the course of years nuclear industry has steadily increased the utilization efficiency of fuel, especially by increasing the burnup. This leads to increased radiation fields and consequently enhanced impact on fuel structure also interaction with cladding, itself supposed to be the first barrier against radioactivity release. Fig. 2 shows an exploded schematic drawing of an irradiated fuel rod segment. The different regions marked indicate locations with potential fission product and higher actinide accumulation.

The fuel pellet cracks during irradiation due to temperature gradients, being aware that micrographs made in a hot cell laboratory do not represent the fuel status at the end of irradiation because also the cool-down process induces cracking of the ceramic fuel. Volatile fission products are accumulated in the cracks and in the gap between the fuel and the cladding. Metallic and oxide precipitates are formed inside the grains of the fuel and at the grain boundaries.

Fuel composition after irradiation

Used fuel is among the most complex materials known worldwide. After irradiation nuclear fuel contains a large number of radioisotopes of most elements listed in the Mendelev table of elements (see Fig. 3).

Starting from the commonly used pure UO_2 fuel, the fission and neutron capture reactions lead upon irradiation to the formation of considerable amounts of different isotopes of elements from all eight element groups in the periodic table. Gen-II reactors were typically designed to achieve a burn-up of about 40 GWd/t. With an improved fuel technology, these same reactors are now reaching up to 60 GWd/t. The goal is to produce a given amount of energy with a lower number of fresh nuclear fuel elements and to generate less used nuclear fuel elements. Furthermore the downtime of the reactor for refueling is reduced. At some stage however, the build-up of fission product neutron poisons (mainly rare earth fission products) reach values which necessitate the reactor to be shut down and refueled.

Used fuel is a highly radioactive material and contains at an average burnup of 45 GWd/t about 94% U-238, about 1% U-235 that has not fissioned, almost 1% plutonium and 4.5% fission products with the following approximate composition: Rare Earths, Y: 24%, Ru, Tc, Rh, Pd: 16%, Kr, Xe: 15%, Zr, Nb: 14%, Mo: 13%, Cs, Rb, I, Te: 11%, Ba, Sr: 7%.

All these new chemical elements are either included in the UO_2 fluorite structure if soluble, or segregated out of the matrix in solid or gaseous forms. A detailed classification of fission products was published by Kleykamp back in 1985 (Kleykamp, 1985), as follows:

- Dissolved in the matrix: Rb, Sr, Y, Zr, Nb, Te, Cs, Ba, La, Ce, Pr, Nd, Pm, Sm, Eu
- Partly precipitated at grain boundaries (oxides): Rb, Sr, Zr, Nb, Mo, Se, Te, Cs, Ba
- Metallic precipitates: Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb, Se, Te
- Volatiles: Br, Kr, Rb, I, Xe, Cs, Te

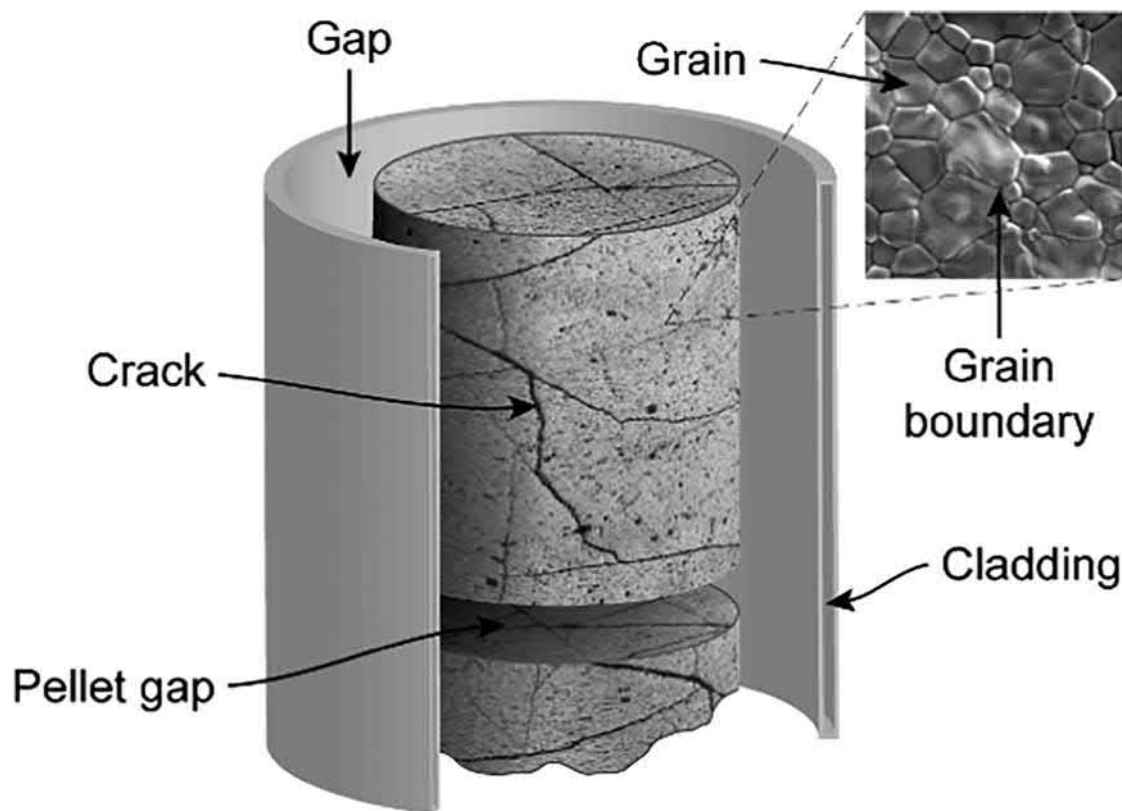


Fig. 2 Schematic exploded illustration of an irradiated fuel rod segment with areas of potential fission product and actinide accumulation.

Some of these elements are in a single physical form (see list above), whereas others are present under different forms depending in a proportion influenced by the fuel temperature, the fuel oxidation state, interactions with the cladding, the fuel matrix and also the other elements formed upon irradiation.

The above mentioned burn-up has also a considerable impact on the content of transuranic elements which are formed by neutron capture of ^{238}U (see Fig. 4).

It is obvious, that also the actinide content is increased with higher burnups, if, at the same initial ^{235}U enrichment, more and more higher actinide isotopes, formed by neutron capture, are the fissile material. In Table 1 the content of transuranium actinides in LWR UOX and LWR MOX at different burnups are compared.

The Cm content is almost a factor of 10 higher if the burn-up is raised from 33 to 60 Gwd/t U. The same is true for MOX fuel where ^{239}Pu is the fissile material in comparison to an LWR fuel at the same burnup of 45 Gwd/t U.

Element redistribution upon irradiation

In discharged irradiated fuel, the above mentioned overall content inventories of fission products and actinides are not homogeneously distributed. In the strong temperature gradient as described in the previous chapter, fission products migrate to the colder fuel periphery regions. Pu and consequently minor actinides are formed by neutron capture preferentially at the fuel periphery by incident neutrons. Various experimental techniques are used to quantify the distribution of fission products and actinides. Those are mainly, scanning electron microscopy (SEM), X-ray, X-ray fluorescence (XRF) (Mogensen et al., 1999) and most commonly Electron Probe Microanalysis (EPMA) (Fisher et al., 2002; Walker et al., 1996a).

In Fig. 5, EPMA line-scans show the redistribution of 4 key elements in a UOX fuel with a burn-up of 62.8 Gwd/tHM.

The radial distribution of uranium is obviously almost constant along the whole radius of the fuel pellet. A slight decrease at the pellet periphery is due to the above mentioned conversion of uranium into plutonium by neutron capture. This observation is of course confirmed by the plutonium increase in the outer region of the fuel pellet.

The fission product neodymium is commonly regarded as good burn-up indicator, because it does not move away from its point of formation and therefore allows in the present case to determine the radial burnup variations. Its line-scan displays a quite homogeneous distribution of fission events along the pellet radius, obviously correlated with the homogeneous distribution of fissile ^{235}U in the fuel at the beginning of irradiation.

At the fuel periphery the neodymium content increases exponentially, due to the fact that as the irradiation process goes on, more and more ^{239}Pu formed through neutron capture of ^{238}U replaces ^{235}U as the fissile material (see Fig. 6).

Periodic table of the elements

Period	Group**																18	
	1	2											13	14	15	16	17	18
	IA	IIA											IIIA	IVA	VA	VIA	VIIA	VIIIA
	1A	2A											3A	4A	5A	6A	7A	8A
1	<u>H</u> 1.008																	<u>He</u> 4.003
2	<u>Li</u> 6.941	<u>Be</u> 9.012											<u>B</u> 10.81	<u>C</u> 12.01	<u>N</u> 14.01	<u>O</u> 16.00	<u>F</u> 18.998	<u>Ne</u> 20.18
3	<u>Na</u> 22.99	<u>Mg</u> 24.31	3 3B	4 4B	5 5B	6 6B	7 7B	8 ----- ----- 8 -----	9 ----- ----- 8 -----	10 ----- ----- 8 -----	11 IB	12 IIB	<u>Al</u> 26.98	<u>Si</u> 28.09	<u>P</u> 30.97	<u>S</u> 32.07	<u>Cl</u> 35.45	<u>Ar</u> 39.95
4	<u>K</u> 39.10	<u>Ca</u> 40.08	<u>Sc</u> 44.96	<u>Ti</u> 47.88	<u>V</u> 50.94	<u>Cr</u> 52.00	<u>Mn</u> 54.94	<u>Fe</u> 55.85	<u>Co</u> 58.93	<u>Ni</u> 58.69	<u>Cu</u> 63.55	<u>Zn</u> 65.39	<u>Ga</u> 69.72	<u>Ge</u> 72.59	<u>As</u> 74.92	<u>Se</u> 78.96	<u>Br</u> 79.90	<u>Kr</u> 83.80
5	<u>Rb</u> 85.47	<u>Sr</u> 87.62	<u>Y</u> 88.91	<u>Zr</u> 91.22	<u>Nb</u> 92.91	<u>Mo</u> 95.94	<u>Tc</u> (98)	<u>Ru</u> 101.1	<u>Rh</u> 102.9	<u>Pd</u> 106.4	<u>Ag</u> 107.9	<u>Cd</u> 112.4	<u>In</u> 114.8	<u>Sn</u> 118.7	<u>Sb</u> 121.8	<u>Te</u> (127.6)	<u>I</u> 126.9	<u>Xe</u> 131.3
6	<u>Cs</u> 132.9	<u>Ba</u> 137.3	<u>La*</u> 138.9	<u>Hf</u> 178.5	<u>Ta</u> 180.9	<u>W</u> 183.9	<u>Re</u> 186.2	<u>Os</u> 190.2	<u>Ir</u> 192.2	<u>Pt</u> 195.1	<u>Au</u> 197.0	<u>Hg</u> 200.5	<u>Tl</u> 204.4	<u>Pb</u> 207.2	<u>Bi</u> 209.0	<u>Po</u> (210)	<u>At</u> (210)	<u>Rn</u> (222)
7	<u>Fr</u> (223)	<u>Ra</u> (226)	<u>Ac~</u> (227)	<u>Rf</u> (261)	<u>Db</u> (262)	<u>Sg</u> (266)	<u>Bh</u> (264)	<u>Hs</u> (277)	<u>Mt</u> (276)									

Lanthanide Series*

<u>Ce</u>	<u>Pr</u>	<u>Nd</u>	<u>Pm</u>	<u>Sm</u>	<u>Eu</u>	<u>Gd</u>	<u>Tb</u>	<u>Dy</u>	<u>Ho</u>	<u>Er</u>	<u>Tm</u>	<u>Yb</u>	<u>Lu</u>
140.1	140.9	144.2	(147)	150.4	151.9	157.3	158.9	162.5	164.9	167.3	168.9	173.0	175.0

Actinide Series~

<u>Th</u>	<u>Pa</u>	<u>U</u>	<u>Np</u>	<u>Pu</u>	<u>Am</u>	<u>Cm</u>	<u>Bk</u>	<u>Cf</u>	<u>Es</u>	<u>Fm</u>	<u>Md</u>	<u>No</u>	<u>Lr</u>
232.0	(231)	(238)	(237)	(244)	(243)	(247)	(247)	(249)	(254)	(251)	(256)	(254)	(257)

Fig. 3 Elements of the Mendeleev table contained in used fuel (in red boxes).

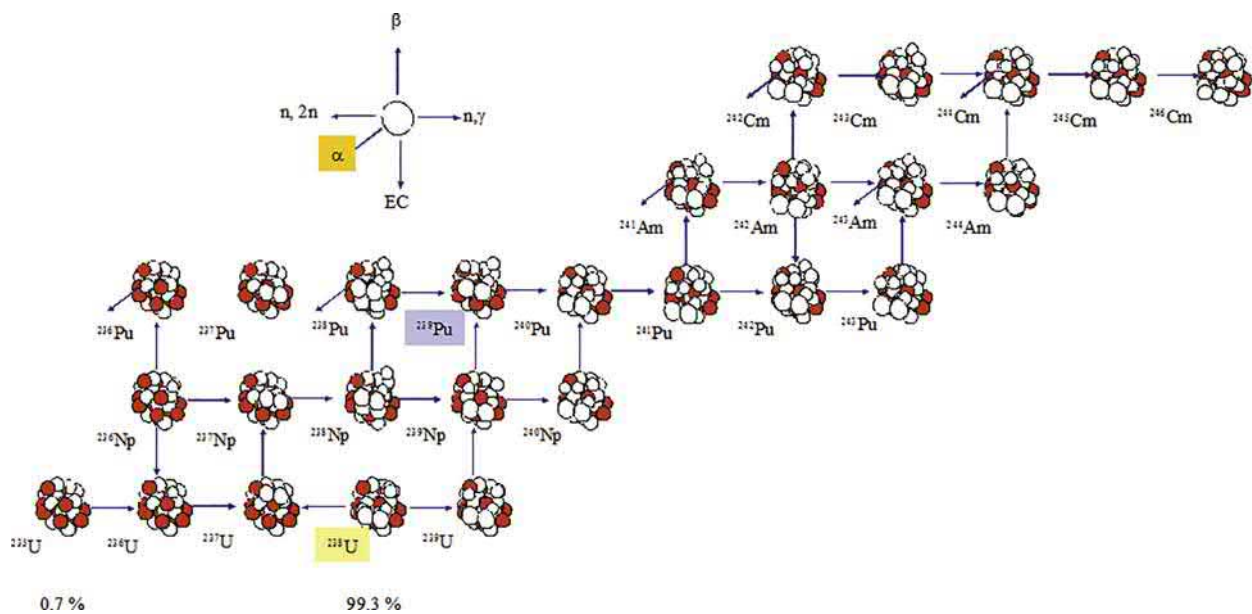
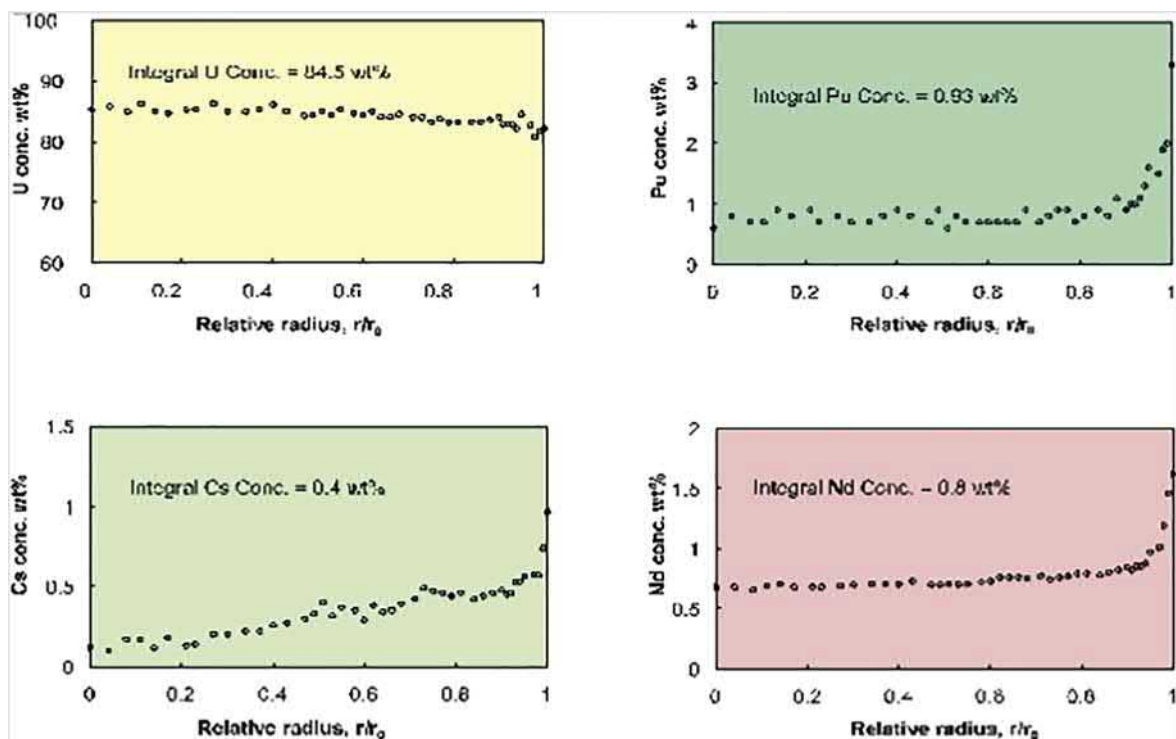


Fig. 4 Built-up scheme of higher actinides in irradiated fuel.

Table 1 Influence of the burnup on the composition of key major transuranium elements in g/t U of LWR UOX fuels in comparison to MOX fuel.

Fuel type	LWR-MOX		LWR-UOX	
Burnup, Gwd/t U	45	33	45	60
Pu	48,850	9740	11,370	12,990
Np	161	433	611	887
Am	4480	325	521	765
Cm	810	23	92	213

**Fig. 5** EPMA line-scans for U, Pu, Cs and Nd of a UO_2 fuel at 62.8 GWd/tHM burnup.

Cesium is a major fission product with a high volatility at temperatures exceeding 1000 °C and thus a high mobility towards the colder fuel periphery. The large increase in the outer part of the fuel pellet are due to the fission of fissile plutonium in this region. Gamma scans of entire fuel rods show increasing accumulation of cesium at the pellet-pellet interface at high fuel burn-up (Walker, 1997). The fission gases xenon and krypton have a migration behavior similar to cesium (Manzel and Walker, 2002; Lassmann et al., 1995; Nogita and Une, 1995).

The above findings mean, that in case of a failure of the fuel rod during storage or disposal, e.g. a defect in the cladding, the pellet rim zone will be the first in contact with the environment and the inventories present in this zone will determine the immediate release to the biosphere, the so-called instant release fraction (IFR).

Nuclear fuel restructuring

Using SEM and EPMA, Manzel and Walker have thoroughly investigated the distribution of used fuel constituents in connection with microstructure changes, such as significant recrystallization processes especially in the rim zone, for fuels with high burnups up to 100 GWd/t (Walker et al., 1996a; Walker, 1997) and in the plutonium rich particles of MOX fuel (Fig. 7).

At about 50 GWd/tHM the UO_2 grains in the outer region of the fuel pellet begin to recrystallize due to accumulation of irradiation damage, leading to an increase in fission gas release to the rod free volume (Lassmann et al., 1995; Nogita and Une, 1995). The transformed microstructure, also called high burnup structure (HBS), consists of small grains of average size 0.15 μm and a high concentration of pores of typical diameter 1–2 μm (Spino et al., 1996; Ray et al., 1997; Xiao and Long, 2016). As the burn-up increases the grain subdivision progresses towards the center of the fuel and further grain subdivision in the outer pellet zone is observed (Wiss and Rondinella, 2010; Walker et al., 1997; Ronchi et al., 2004; Sari et al., 1979). Two mechanisms of formation

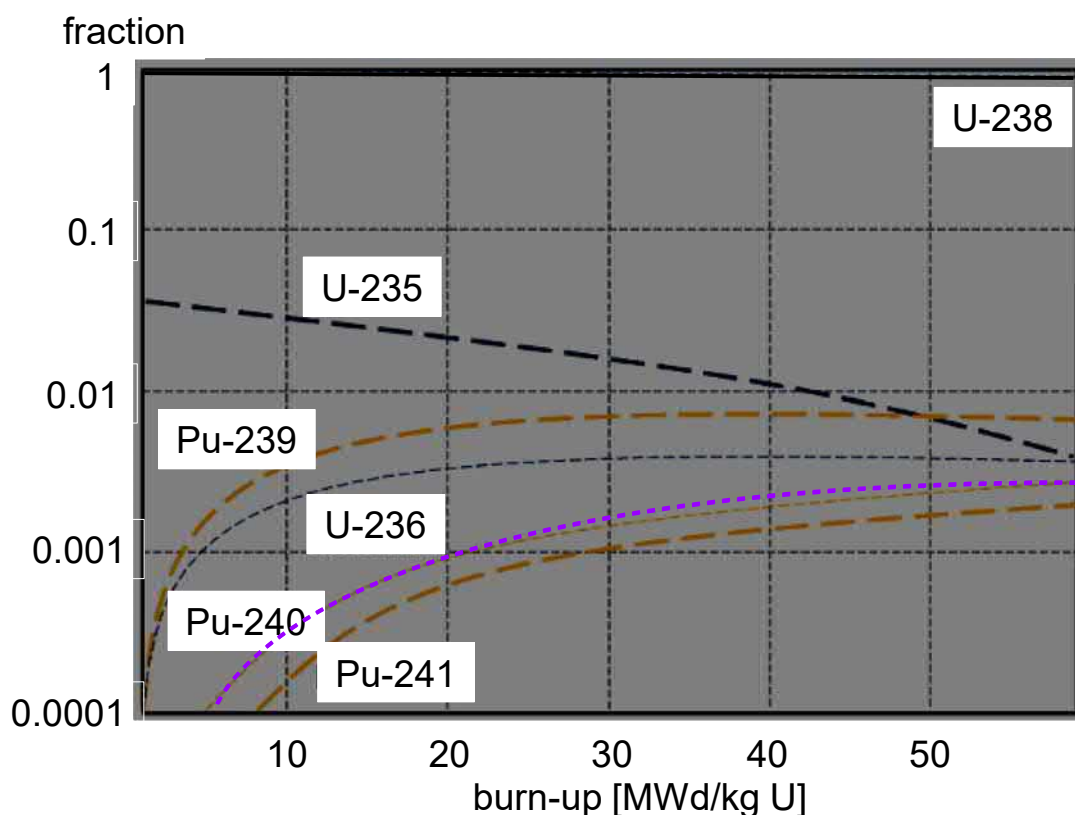


Fig. 6 Content of main uranium and plutonium isotopes in nuclear fuel at increasing burnup.

for the HBS structure. Spino et al. assume that migration of fission gas out of the UO_2 lattice induce the formation of faceted gas pores, overpressurisation of the pores and grain sub-division around the pores due to the local build-up of mechanical stress (Spino et al., 1998). Nogita and Une (Nogita and Une, 1995) postulate, that the accumulation of irradiation damage leads to the formation of dislocation tangles, with increasing burnup these unravel to produce sub-divided grains and those with high-angle boundaries act as nuclei for recrystallisation.

MOX fuel often produced as master-mix by comilling processes such as OCOM or MIMAS (Staicu et al., 2011). Pu agglomerates up to 200 μm in size are dispersed in a matrix of natural or depleted UO_2 . For LWR, the plutonium content in the agglomerates often ranges between 6% and 10%. During irradiation to a normal burn-up of ~ 45 GWd/tHM, about 75% of the fissions occur in the Pu-rich agglomerates (calculation based on data from Lassmann et al., 1995), leading to a high local burnup in the agglomerates, up to 270 GWd/tHM (Walker et al., 1996b; Bailly et al., 1999). This results after irradiation in a microstructure of fine grains and large pores with a diameter of several microns, similar to the high burn-up structure (HBS) formed at the periphery (rim) of UO_2 pellets with local burnup above 60 GWd/tHM (Govers et al., 2019).

Fission gas behavior

The increased fission at the pellet periphery must also be the origin of the high fission gas released to the rod free volume. The release mechanisms and pathways for volatile fission products is not fully clear. Puncturing of PWR rods in the burn-up range of 15–100 GWd/tHM shows that up to 50 GWd/tHM only 5–6% fission gas is released (see Fig. 8).

With increase in burnup from 50 to 100 GWd/t the percentage of fission gas released to the rod free volume increases exponentially to about 25%. This finding goes along with a significant restructuring of the fuel across the pellet radius, but mainly at the pellet periphery (see Fig. 2). It is quite obvious, that, because as burnup progresses more and more fission events occur in the outer pellet rim, where the actual process of recrystallisation leads to gas release from the fuel. Nogita and Une (Nogita and Une, 1995) have shown that prior to recrystallisation, the fuel grains contain a high density of nanometer size gas bubbles, which are absent after grain sub-division has occurred. Bremier et al. (2000a) have calculated that at the temperature of the pellet rim (400 °C), radiation-enhanced diffusion results in the release of large fractions of fission gas from the recrystallised grains. At the same time, they found (Bremier et al., 2000b) that for an un-recrystallised grain of diameter 10 μm , it requires 572 days to obtain a fractional release of 0.01 using an apparent diffusion coefficient of $1023 \text{ m}^2/\text{s}$. This is the diffusion coefficient that gives the equilibrium xenon concentration of 0.2–0.3 wt% measured by EPMA in the grains of the high burn-up structure (Nogita and Une, 1995).

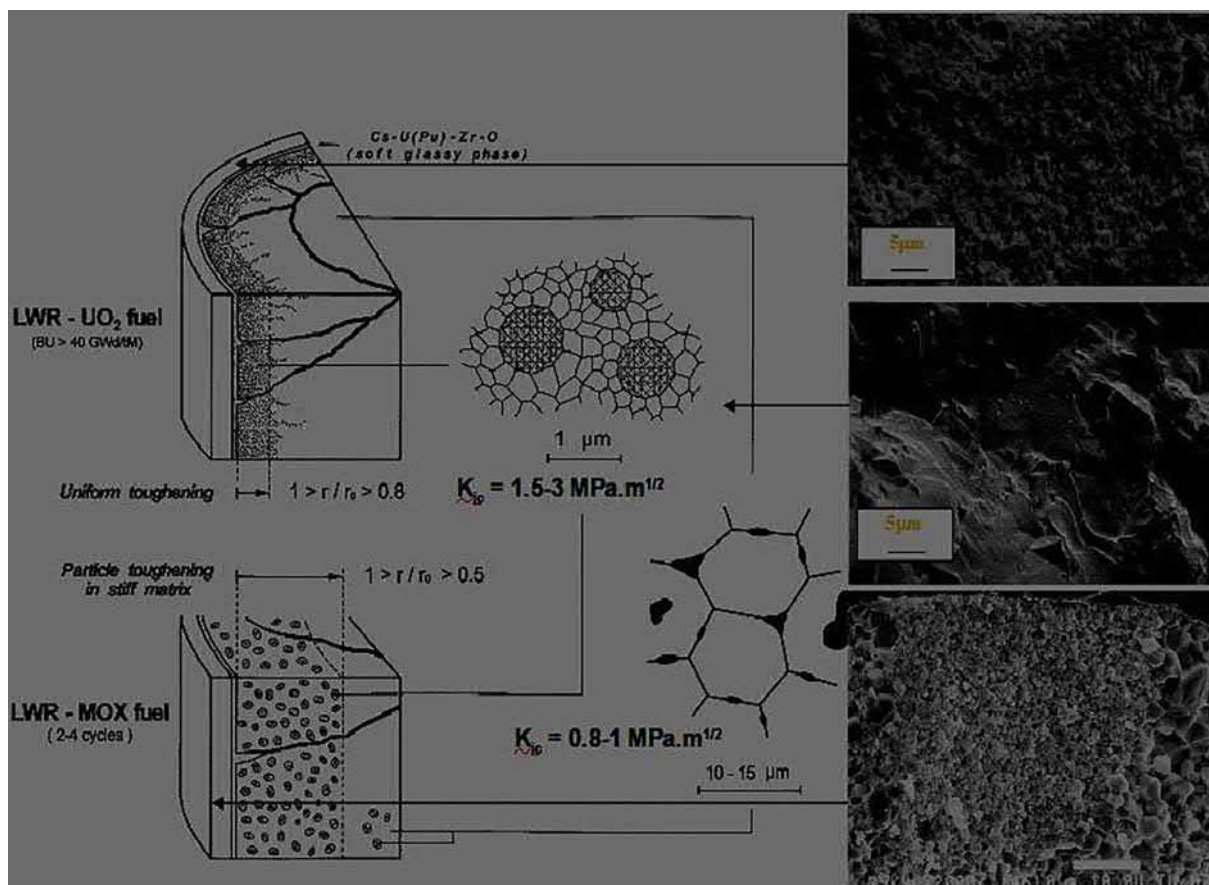


Fig. 7 Restructuring of nuclear fuel upon irradiation showing high burn-up zones at the pellet periphery for LWR UO₂ fuel and in the plutonium-rich agglomerates for LWR-MOX fuel.

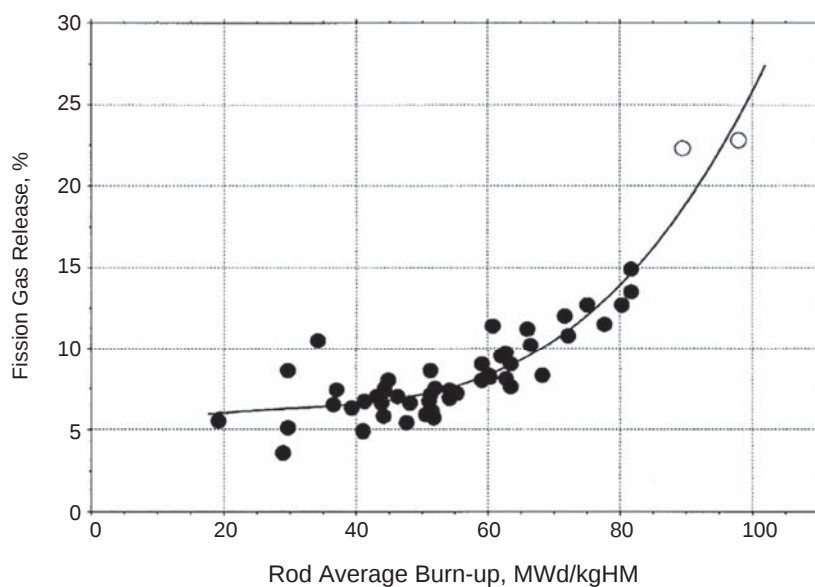


Fig. 8 Fission gas release to the fuel rod plenum as a function of burnup.

Aging mechanisms of used fuel

Aging processes occurring during storage of used nuclear fuel rods are safety-relevant both for storage and for handling/transportation processes. In particular, it is important to assess all mechanisms that may cause severe degradation of the mechanical integrity of the used fuel rods, mainly affected by helium gas buildup, dislocations, grain disintegration, hardness, heat capacity, swelling, and microstructural changes of the spent fuel matrix. Such mechanisms are strongly affected by the radioactive decay power generated in used fuel, which determines also the temperature level experienced by the rods in the dry storage container. The build-up of alpha-decay damage and helium may become a determining factor in the evolution of the mechanical properties of the used fuel (Bruno et al., 2007; He et al., 2007).

Hydrogen formed by corrosion of zircaloy cladding of the fuel rods, is dissolved or precipitated as hydrides in the metal matrix and affects the status of the cladding. Nevertheless, the experience of more than 40 years of wet storage of several million light water reactor rods shows that used nuclear fuel with zirconium alloy cladding have excellent durability (IAEA, 2006). Destructive and nondestructive examinations of PWR Zircaloy-clad rods showed no detectable changes in the cladding characteristics after 20–30 years of wet storage. Indicative techniques show that corrosion rates in wet storage should be <0.007 mm/year. The cladding also retained significant creep ductility after dry storage.

Composition evolution with time

After irradiation, the elemental and chemical composition changes of the fuel are only minor. On the other hand the radioactivity emitted from the fuel changes by orders of magnitude over the repository timescale. Fig. 9 shows the evolution of the total α - and β -activity of a PWR fuel at 40 GWd/tHM over a storage time up to 1 million years.

The main difference between discharged UOX and MOX fuel is that the total α -activity in the MOX is nearly 7 times larger, this factor decreases to 3 after 100,000 years. The fact that α -particles are emitted by quantum tunneling out of the nucleus with extremely low probabilities and decrease with the particle, low-energy α -emitters are generally much more long-lived than $\beta(\gamma)$ -emitters (Krane, 1988).

The very intense $\beta(\gamma)$ -field emitted by discharged fuel is due to the decay of mainly the short-lived radionuclides, listed in Fig. 10, where also the decay curves of individual isotopes of light (left) and heavy (right) fission products present in a UO_2 fuel with a burnup of 40 GWd/tHM after 1 year cooling are shown.

Most of the fission product isotopes listed in Fig. 10 decay between 1 and 600 years, except ^{99}Tc , ^{99}Tc , ^{79}Se , ^{93}Zr , ^{94}Nb , ^{107}Pd , ^{126}Sn , ^{129}I and ^{135}Cs . This is well in agreement with the evolution of the total $\beta(\gamma)$ activity in Fig. 9.

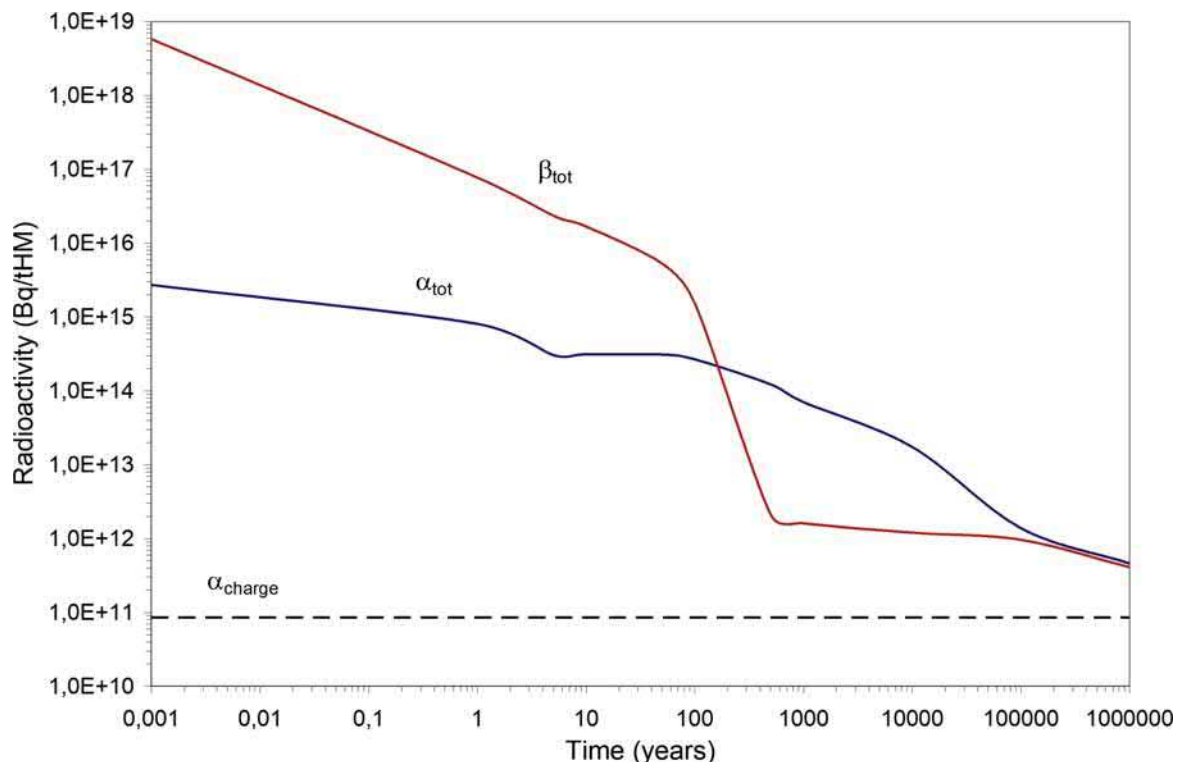


Fig. 9 Evolution of the total α - and β -activity of a PWR fuel at 40 GWd/tHM over a storage time up to 1 million years.

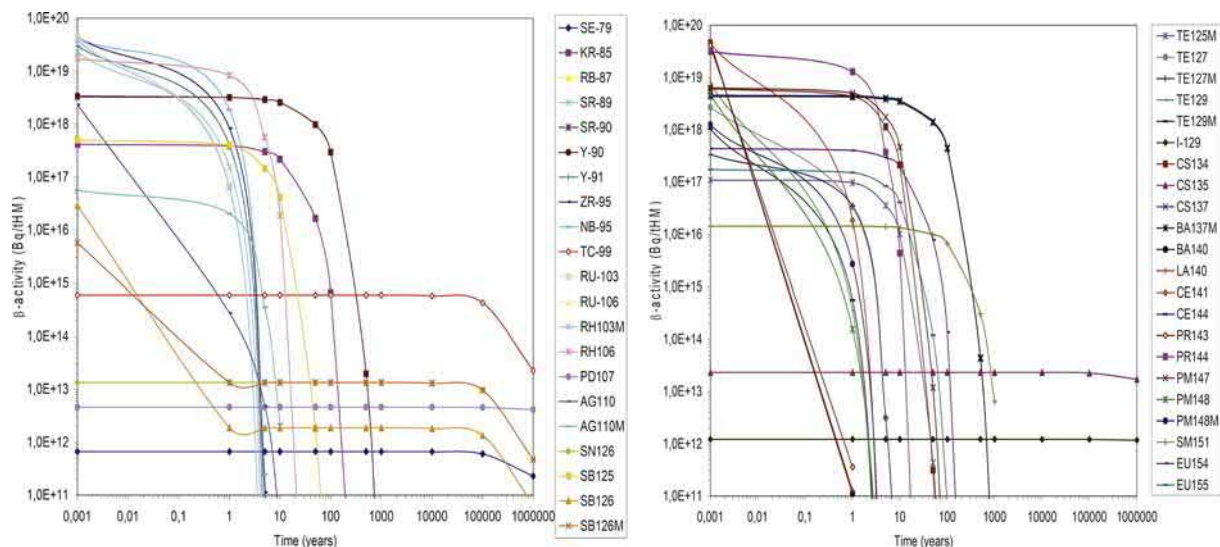


Fig. 10 Decay of lighter (left) and heavy (right) fission products present in a UO_2 fuel with a burn-up of 40 Gwd/tHM after 1 year cooling as a function of storage time.

Radiation damage

Alpha radiation has a typical kinetic energy of about 5 MeV. It can heavily ionize matter and quickly deposit its energy along the short paths. Defects created by displacement cascades preferentially at the fuel periphery are extensively recombined in UO_2 and PuO_2 and hence, no matrix amorphization can be detected in those materials (Matzke, 1982; Van Brutzel et al., 2006). Nevertheless, lattice parameter increase caused by α -radiation damage that displaces atoms in the lattice can lead to microscopic swelling, which is accommodated by the free space between fuel and cladding. Radiogenic helium and alpha-damage can cause early material degradation and enhanced corrosion (Roudil et al., 2004; Fanghaenel et al., 2013; Poinssot et al., 2005).

The β -decay results in formation of elements of different valence state and eventually change the oxidation state of the UO_2 matrix. Matzke et al. could demonstrate that in high burnup UO_2 fuel the oxygen potential in the fuel appears to be controlled by molybdenum (Mo/MoO_2). Mo is among the most abundant light fission product in used fuel (Matzke, 1982).

Many studies were carried out also on tailored specimen to study individual effects. The question how particles from radioactive decay interact with matter was already investigated by the nuclear pioneers M. Curie, J. Thomson, N. Bohr and E. Rutherford (Curie, 1900; Rutherford et al., 1899). The development of ion accelerators allowed extensive studies on radiation damage in all types of materials. Linked to the use of the Monte Carlo code Stopping and Range of Ions in Matter (SRIM) for the estimation of energy loss, range, and damage, the description of the main processes of particles/ions interactions with matter is given (Ziegler et al., 1985).

The equivalent damage is indicated as displacements per atoms (dpa). The investigation of alpha-damage by XRD, TEM and Vickers Hardness showed that the fluorite structure is preserved but with a substantial increase in lattice parameter (up to 0.6%), the formation of dislocation loops (41022 m^3), the formation of helium bubbles and an increase of the Vickers Hardness. For instance annealing studies performed on UO_2 specimen doped with alpha-emitters, show that recovery of the lattice parameter is complete at 1100 K, with the major recovery occurring between 600 and 900 K. (Matzke, 1995).

The special case of SUPERFACT fuel

The SUPERFACT experiment carried as a collaboration project of the French Commissariats à l'Énergie Atomique (CEA) and the European TransUranium Institut (TUI) aimed to study the transmutation of MA's (Walker and Nicolaou, 1995). Several fuels with neptunium and americium contents from 2 to 45 wt were irradiated for 365 electric full power days (EFPD) in the French PHENIX reactor (FR). Post-irradiation examinations, carried in the CEA and TUI hot cell laboratories to assess the transmutation efficiency include in 1991 SEM examinations. Two samples of a fuel with the composition $\text{U}_{0.6}\text{Am}_{0.2}\text{Np}_{0.2}\text{O}_2$ taken at the same radial positions were again examined after 12 years of storage in the hot cell laboratory. Calculations show that the α -dose accumulated after this time period is equivalent to MOX (9% Pu, 50 Gwd/tHM) after 400,000 years storage. The SEM micrographs recorded in 1991 and 2003 are compared in Fig. 11.

Even if of course the samples selected are not identical, but taken at adjacent axial positions, it is obvious that at a magnification of $2000\times$, no difference is visible despite the huge α -dose rate deposited during 12 years of storage.

Also international programs aiming at investigating the technical bases to ensure safe, long-term used fuel storage and future transportability. In the United States, the Electric Power Research Institute (EPRI) has launched the so-called Extended Storage Collaboration Program (ESCP) with the participation of 70 industry, governmental, and academic organizations from at present

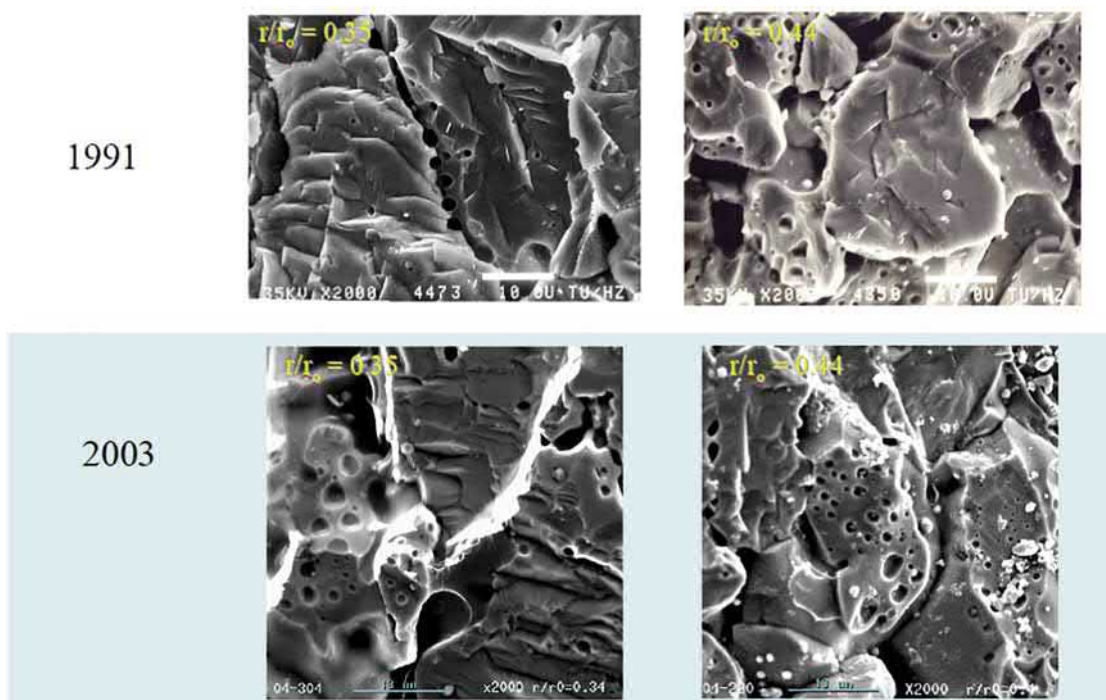


Fig. 11 Evolution of the total α - and β -activity of a PWR fuel at 40 Gwd/tHM over a storage time up to 1 million years.

20 countries to identify research gaps to allow interim storage of used fuel up to 100 years. A major partner is Germany, where the final disposal project Gorleben has been given up, safe long intermediate storage has to be implemented. Also in Belgium, post-irradiation examinations (PIEs) of spent BWR-MOX and PWR-UO₂ fuel rods irradiated in commercial LWRs and stored for 20 years were carried out to evaluate fuel integrity during storage no marked difference in the visual inspection, fission gas release, oxide layer thickness, pellet microstructure, and cladding mechanical properties or hydride orientation after storage (Sasahara and Matsumura, 2008).

Conclusions

The irradiation of nuclear fuel results in an extremely complex material. A strong temperature gradient of ~ 500 °C between the center and fuel periphery leads especially for volatile fission products to an accumulation at the fuel pellet interface, neutron capture is responsible for large quantities of transuranium actinides at the fuel periphery. Strong $\beta(\gamma)$ - and α -radiation fields forms defects and dislocations, however also due to high temperatures an almost complete healing of these defects is observed. Some studies claim, that during storage the much lower temperatures, alpha-damage defects (dislocation loops) could lead to microcracking, starting from 1000 years after discharge especially in MOX fuel with higher α -radiation. The consequence could be on the long term a partial loss of integrity. However the Superfact fuel indicates, that even extremely large α -radiation energy deposits, equivalent to those of MOX for very long storage times (even longer for UO₂), does not give evidence for any disintegration process.

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Fuel Storage (New Fuel/Used Fuel Pool)

John H. Kessler, J Kessler and Associates, LLC, Charlotte, NC, United States

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Commercial spent nuclear fuel (CSNF) locations and amounts

International Atomic Energy Agency (IAEA) member countries regularly report the amount of spent fuel in their country including the type(s) of storage systems used in their countries. The most recent set of reports are included in the proceedings of the *Sixth Review Meeting of the Contracting Parties to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management*, Vienna: International Atomic Energy Agency (2018), available at: <https://www.iaea.org/topics/nuclear-safety-conventions/joint-convention-safety-spent-fuel-management-and-safety-radioactive-waste/documents>.

As of December 31, 2013, the IAEA reported that a total of roughly 370,000 metric tons of uranium (MTU) of spent fuel (commercial SNF and from other sources) was generated (IAEA, 2018). This is summarized in Table 1.

Characteristics of CSNF requiring management during storage, transportation, reprocessing, and disposal

The primary characteristics of CSNF that require safe management are the high levels of radiation and decay heat.

When fresh uranium fuel is placed in a reactor, it typically contains 3–5% U-235 and 95–97% U-238. When it is removed, it contains about 0.7–1% U-235, 93–96% U-238, 1% plutonium, 0.1% minor actinides (e.g., Np-237, Am-241, Cm-244), and 4–5% fission products as shown in Fig. 1. Some of the fission products are long lived, such as I-129 and Tc-99, while others have shorter half-lives.

The primary contributors to both radiation and decay heat during the first few decades of storage are the fission products Cs-137 and Sr-90—both of which have half-lives of approximately 30 years. Both decay heat and the radionuclides decrease with time—an example of which is shown in Fig. 2. The exact inventory of radionuclides and decay heat generation versus time is a function of the burnup of the CSNF: the amount of energy generated per unit mass of the initial uranium in the fresh fuel (expressed in gigawatt-days per metric ton of uranium (GWd/MTU)). Thus, CSNF storage systems must be able to shield from the intense radiation and remove the decay heat during the storage period.

Commercial spent nuclear fuel storage systems

Today, the majority of CSNF generated in water reactors is stored in spent fuel pools (SFPs) adjacent to operating reactors. This general type of storage is called “wet” storage. Two examples of SFPs are shown in Fig. 3.

Due to the absence of operational geological repository or reprocessing (in most countries) for CSNF, many of the SFPs are currently nearing or are at capacity in countries where no reprocessing-recycling is implemented. In such a situation, some CSNF must be removed from the pool and stored elsewhere in order to continue to operate the reactors. Some countries, such as Sweden, have built additional SFPs at locations away from the reactors. The SFP in Sweden has been designed with sufficient capacity to store 100% of the CSNF anticipated to be generated in that country.

There is an increasing use of “dry” storage systems in which the water is and replaced by an inert gas. Individual dry storage systems are capable of storing a few up to several tens of CSNF assemblies. There are two types of such dry storage systems. One employs a set of bolted metal lids, commonly referred to as “casks,” such as those shown in Figs. 4 and 5. The other employs a welded steel “canister” with welded lids inside a concrete “overpack,” such as those shown in Figs. 7 and 8. Most recent casks/canisters are often designed for both storage and transportation, so they are called “dual-purpose casks” (DPCs). However, many of the used casks and canisters were single-use designed for storage, creating a long-term burden associated to the need

Table 1 Spent fuel discharged from nuclear power plants (MTU), as of December 31, 2013.

	Wet storage	Dry storage	Reprocessed	Total
Africa	850	50	n.a. ^a	900
Eastern Europe	28,600	2700	3200	40,000
Western Europe	37,000	4600	108,400	1 54,100
Far East	32,100	5700	8600	46,400
North America	79,300	41,900	n.a. ^a	121,200
Latin America	3000	2000	n.a. ^a	5000
Middle East and South Asia	n.a. ^a	n.a. ^a	n.a. ^a	n.a. ^a
South East Asia and Pacific	n.a. ^a	n.a. ^a	n.a. ^a	n.a. ^a
Global total	180,800	56,900	120,300	367,600
Joint Convention Contracting Parties	177,000	56,900	120,300	363,700
EU Member States ^b	43,300	6800	107,600	162,300
OECD/NEA members	166,100	53,200	119,900	343,800

Note: Possible differences in totals are due to reminding.

^an.a.: not applicable (or none reported).

^bExcludes Austria, Greece, Malta and the Netherlands.

From Table 5 of IAEA (2018) *Status and Trends in Spent Fuel and Radioactive Waste Management*. IAEA Nuclear Energy Series report NW-T-1.14. Vienna: International Atomic Energy Agency.

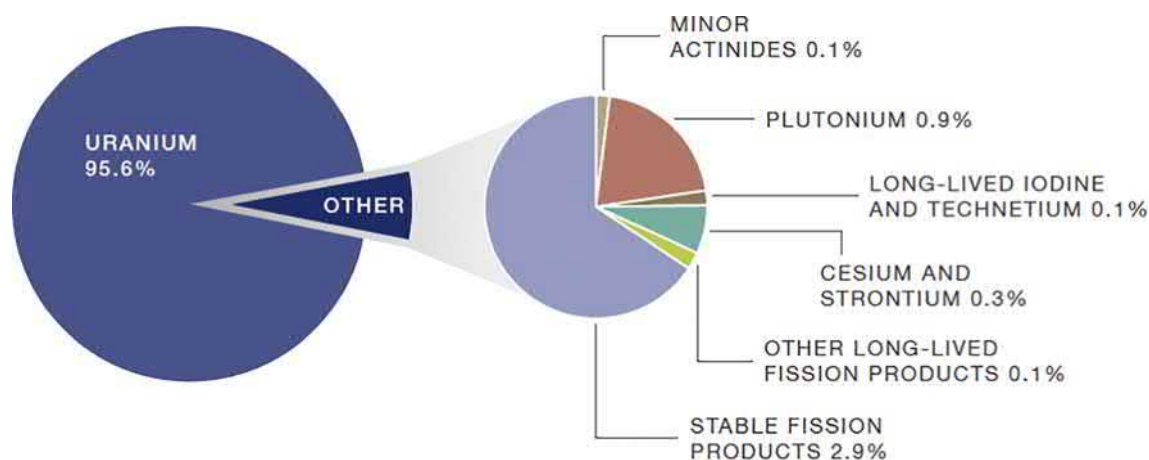


Fig. 1 Composition of spent nuclear fuel after 10 years of cooling. Figure 6 from BRC (2012) *Blue Ribbon Commission on America's Nuclear Future—Report to the Secretary of Energy*. Washington, DC: Blue Ribbon Commission on America's Nuclear Future.

for initially transporting the CSNF and repackaging it. At some locations for which the reactor and associated buildings have been decommissioned, there is no longer a SFP available if it is necessary to reopen the cask or canister for some reason. Furthermore if there is no other place for the CSNF to be transported, then the all that will remain is the CSNF in storage and the site cannot be fully decommissioned.

Cask systems use a double lid design, such as shown in Fig. 6. Each lid uses two O-ring gaskets to ensure the lid provides an adequate seal to limit the release of gas inside the cask body where the CSNF is placed.

The loading procedure for the bolted metal lid cask design is as follows. The empty cask with the lid removed is lowered into the SFP, and the CSNF is loaded into individual storage cells. The cask containing the CSNF is then lifted still almost entirely full of water into a handling bay next to the SFP, and the primary lid that has two O-rings for the seal is emplaced and bolted down. Next, the majority of the water is drained out of the cask interior via a drain orifice. A vacuum is created through the vent orifice to remove remaining water inside the cask via vaporization, after which the cask interior is filled with an inert gas, such as helium. Then a small lid covering the drain orifice is bolted down. The secondary lid is then emplaced over the primary lid again using a double O-ring design. A helium overpressure system is then installed that provides a positive pressure between the two seals in both lids. Maintaining a pressure between the seals that is greater than the pressure inside the cask ensures that if there is a loss of seal, no gas from inside the cask in which the CSNF is stored can escape into the environment. A protective cover is then emplaced, and the cask is transferred onto the concrete storage pad.

The bolted metal systems provide the safety functions via the cask body, seals, and lids. The safety functions include shielding, heat transfer, confinement subcriticality, and structural integrity. Decay heat is transferred through the cask body via metal cooling fins. The cask body, primary, and secondary lids provide gamma and neutron shielding. The pressure between the two seals is

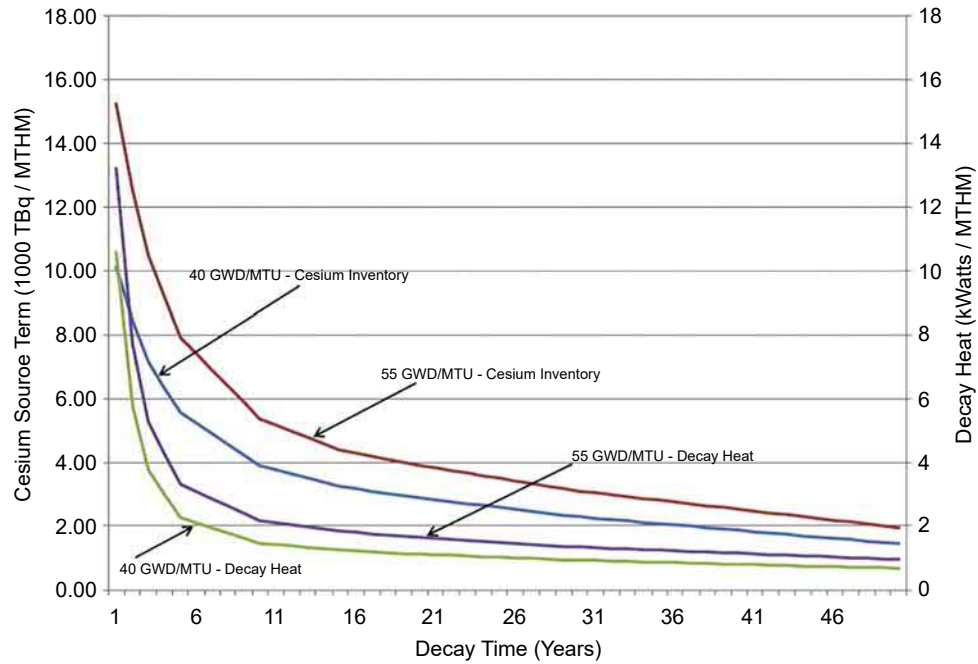


Fig. 2 Pressurized water reactor SNF assembly decay heat (right axis) and cesium inventory (left axis) as a function of burnup and cooling time. Figure 2-2 from EPRI (2012) *Impacts Associated With Transfer of Spent Nuclear Fuel From Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling*. Rev. 1, EPRI Report 1025206. Palo Alto, CA: Electric Power Research Institute.

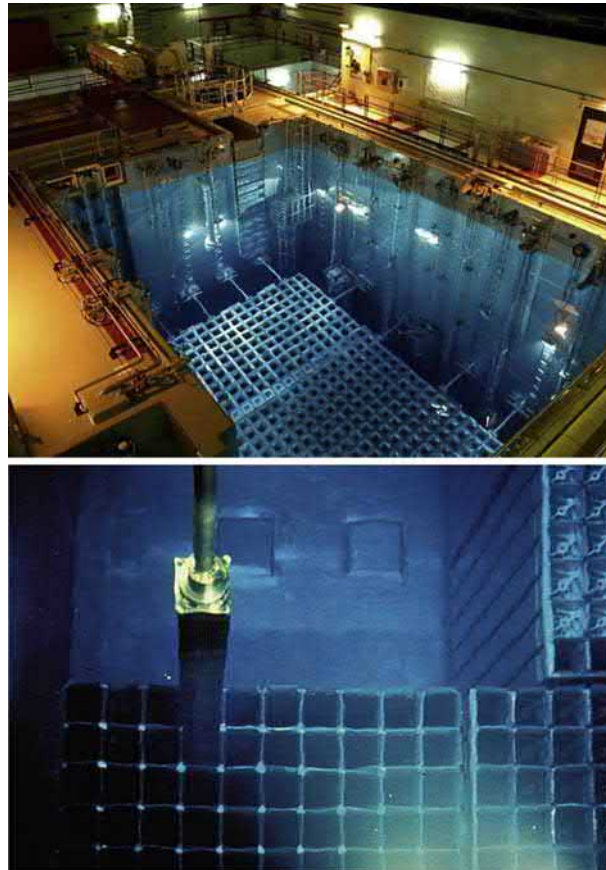


Fig. 3 Examples of spent fuel pools (SFPs) adjacent to reactors. CSNF is discharged from the reactors into these SFPs. The bottom figure shows a spent nuclear fuel assembly being loaded into a storage cell in the spent fuel pool.



Fig. 4 CASTOR V/19 bolted metal storage cask. Courtesy of GNS at <http://www.gns.de/language=en/21551/castor-v-19>

continuously monitored during the entire time the cask is in use. Structural integrity includes loss of cask confinement or shielding due to seismic and atmospheric events that could cause the cask to tip over or to be impacted by projectiles. As will be discussed later, the primary barrier preventing loss of gas or particulates from inside or and the intrusion of air or water into the cask is the double O-ring seal design. Hence, the long-term integrity of the seals is important to maintaining confinement during the entire storage period.

The second type of dry storage system that holds up to a few tens of CSNF assemblies uses welded steel “canisters” inside a concrete “overpack,” such as those shown in [Figs. 7 and 8](#). As opposed to the bolted metal systems that provide the safety functions using the cask itself—along with the seals and lids—the safety functions are partially divided between the canister and the concrete overpack. The welded canister provides the decay heat removal to the outside of the canister, confinement and subcriticality safety functions, and also contributes to the overall structural integrity of the system; the concrete overpack provides the primary structural and shielding functions, and is also designed to transfer the heat from the canister via natural air convection within the air gap between the outer canister wall and the inner concrete overpack wall. For those welded canister systems stored outdoors, the concrete overpack also provides protection from rain and projectiles. [Fig. 7](#) shows the NUHOMS type of system. The NUHOMS systems are designed to store PWR and BWR CSNF assemblies horizontally. Its main components include the stainless steel canister with an internal fuel basket, and a concrete “overpack” that protects the canister and provides radiological shielding. The right image in [Fig. 7](#) is a cutaway view of the canister showing the CSNF assembly storage cells and the two lids. The left image shows the on-site transfer cask with the welded canister inside being mated up to the front of the horizontal concrete overpack just before transfer of the canister into the concrete overpack. One of the air inlet vents with a screen is shown underneath the concrete overpack entry point. There are air outlet vents located on the top of the concrete overpack to provide cooling via natural air convection.

The loading procedure for the NUHOMS type of system is as follows. With both the canister and transfer overpack oriented in the vertical direction, the empty canister without the lids is placed inside an on-site “transfer” overpack. The transfer overpack

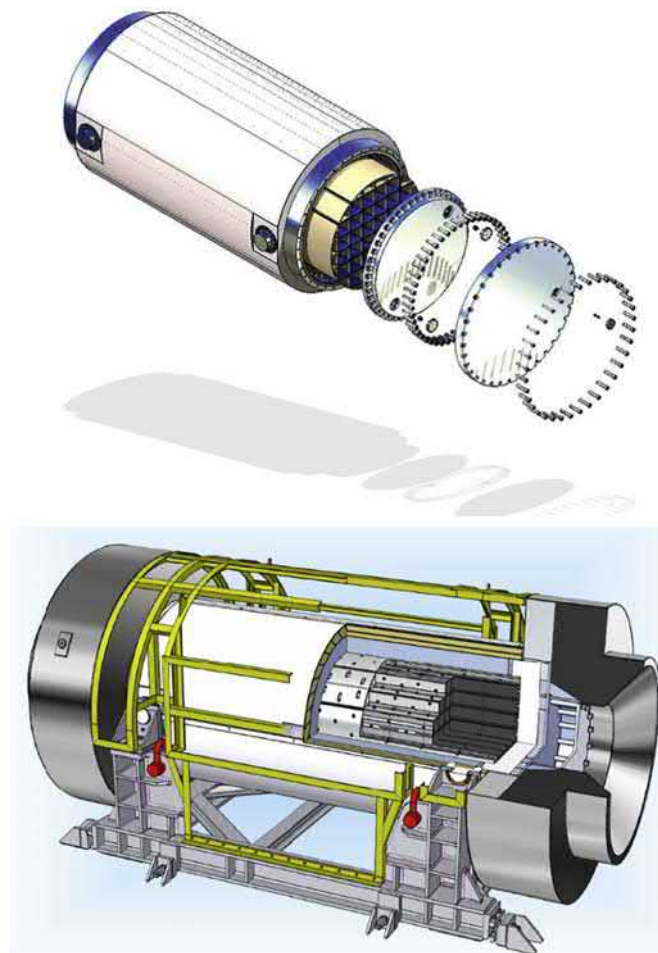


Fig. 5 ENSA Storage and Transportation Casks. Top: ENUN 32P exploded view. Bottom: ENUN-24P and transportation cradle and impact limiters cutaway view. Figures courtesy of ENSA.

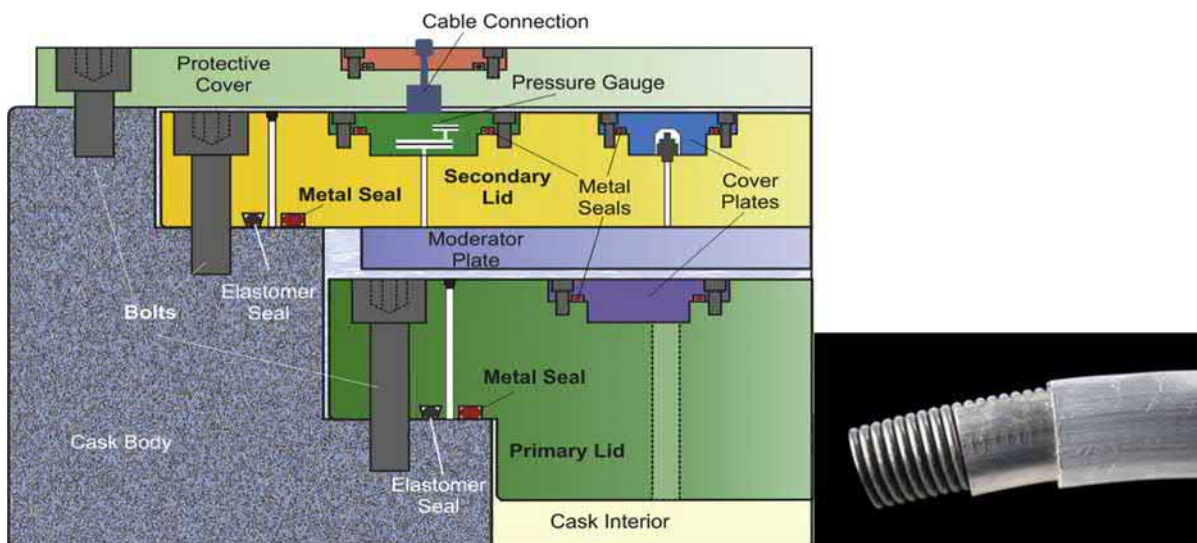


Fig. 6 Typical double barrier bolted closure system of CASTOR® cask types (left) and a metal seal of the Helicoflex® type (right).

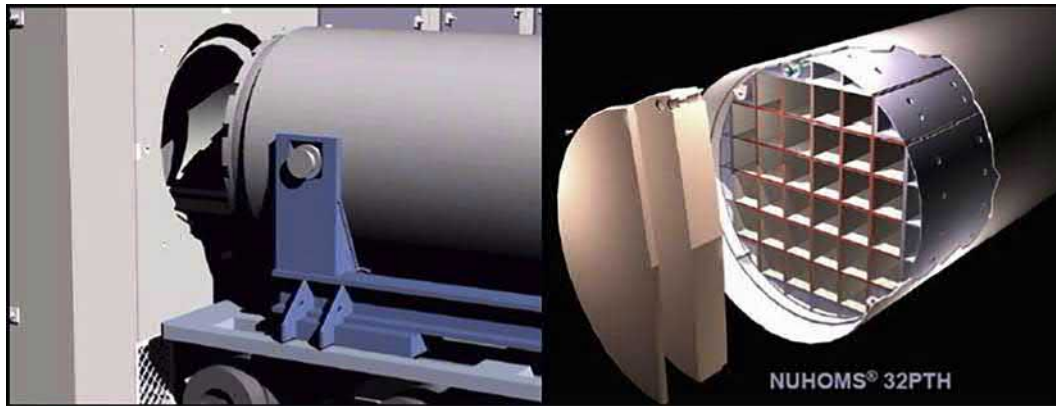


Fig. 7 NUHOMS' horizontal storage module, transfer trailer, and NUHOMS 32PTH DSC (Neider, 2005, 2008).

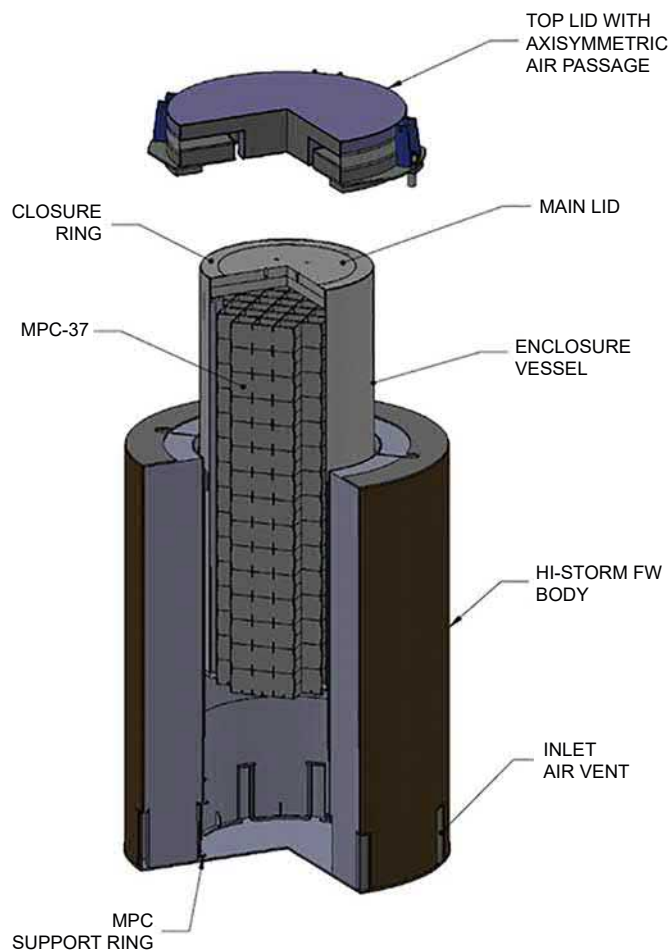


Fig. 8 Cross section of the Holtec MPC-37 Overpack with MPC. From <https://i2.wp.com/holtecinternational.com/wp-content/uploads/2011/06/3-hi-storm-fw.jpg?w=1200&ssl=1> on 22 March 2021.

provides shielding and protection to the canister during on-site loading and handling operations and is used to transport the loaded canister to the concrete overpack. The inner diameter of the transfer overpack is slightly larger than the outer diameter of the canister. The empty canister and transfer overpack are placed into the SFP and the CSNF assemblies are placed into the individual assembly storage cells inside the canister. The canister and transfer cask are then lifted out of the SFP into an adjacent handling bay. The water is partially drained out of the canister via a drain port and the inner lid is then welded onto the top of the canister. The remaining water is then removed from inside the canister through the drain port, and a drain port cover is then welded on. An inert gas is

injected via a similar type of vent port design described for the bolted metal lid cask design above. The outer lid is then welded onto the canister. The canister and transfer cask are then placed onto the transfer trailer and lowered into a horizontal position. The transfer trailer is then moved to in front of the horizontal concrete overpack, and a hydraulic ram system is used to push the canister out of the transfer overpack directly into the concrete overpack. A shielding and structural lid is then placed on the front of the concrete overpack.

Fig. 8 is an example of a vertically oriented welded steel canister inside a concrete overpack. The loading and on-site transfer procedure for the vertical designs is the same as for the NUHOMS design except that the transfer cask and canister are always kept in the vertical position. The transfer cask with the canister are mated to the top of the open concrete overpack, and the canister is lowered into the concrete overpack, the transfer overpack is removed, and the concrete overpack lid is then installed. The concrete overpack air inlet and outlet vents are connected to the gap between the outside of the canister and the inside of the overpack to provide decay heat removal via natural air convection.

Vaults in which CSNF is stored dry in canisters is a second option that is sometimes used. An example of a vault design is shown in Fig. 9. This vault contains CSNF from the decommissioned Fort Saint Vrain high-temperature gas-cooled reactor. The right photo in Fig. 9 shows the shield plugs on top of the filled drywells.

Dry storage licensing and design criteria

General safety functions

In most countries the common safety functions that must be maintained for storage and transportation are: sub-criticality; confinement; shielding; structural integrity; thermal evacuation; and CSNF retrievability at the end of the storage phase.

Subcriticality: subcriticality can be maintained using a variety of design and operational choices, including one or more of the following:

- Ensuring no water can enter the casks or canisters for the dry storage and transportation systems;
- Including neutron poison material in the cask or canister storage cells;
- Maintaining a sub-critical geometry even if the cask or canister is full of water; or
- Ensure the CSNF placed in the cask or canister has sufficiently high burnup at a given initial enrichment.

Confinement (Confinement is required for the entire period of storage and transportation. For geologic disposal, confinement is required for only the period for which retrievability must be maintained: the first few decades to centuries after emplacement.): prevent any radioactive gases or particulates that may be inside the cask or canister from escaping into the environment, using design and operational strategies that include one or more of the following:

- Casks:
 - Use of gaskets (O-rings) to reduce the rate of gas leakage out of the cask to near zero;
 - Maintain a helium gas overpressure between the O-rings and continuously monitor for loss of overpressure;
 - Minimize bolt loosening during storage due to, for example, bolt creep or corrosion
- Welded steel canisters:
 - For austenitic stainless steel canisters:
 - Use of approved body and closure lid welding techniques to minimize the risk of a through-wall crevice or crack;
 - Minimize the likelihood that a stable crack or crevice forms;
 - Minimize the crack or crevice growth rate such that a through-wall crack will not develop during the storage or transportation period;



Fig. 9 US Department of Energy CSNF vault containing CSNF from the decommissioned Fort St. Vrain reactor in the US; vault exterior view (left) and interior charge face (right) (Hain, 2010). DOE (2009): downloaded from <http://www.nwtrb.gov/meetings/2010/june/hain.pdf> (20 March 2016).

- o For carbon steel canisters: canister body and lids are of sufficient thickness to avoid loss of wall thickness due to general corrosion during the storage or transportation period.

Shielding: maintain adequate shielding for normal, off-normal, and accident events during storage, transportation, and disposal at earlier times:

- Casks: minimize shrinkage or loss of the neutron absorber material
- Canisters in storage inside concrete overpacks:
 - o Avoid concrete overpack concrete spalling due to freeze-thaw, or corrosion of embedded rebar;
 - o Avoid development of through-wall cracks of sufficient size to allow radiation streaming.

Structural: ensure the cask or canister/concrete overpack remain sealed in the event of cask or canister/overpack tipover, seismic vibrations, or severe wind-borne missiles:

- Casks (storage and transportation):
 - o For carbon steel casks: cask body and lids are of sufficient thickness to avoid loss of wall thickness due to general corrosion;
 - o Minimize bolt loosening during storage due to, for example, bolt creep or corrosion;
- Concrete overpacks during storage:
 - o Minimize silica-alkali reaction rates;
 - o Minimize concrete rebar corrosion;
 - o Minimize concrete dry-out due to decay heat and low atmospheric humidity.

Thermal: ensure adequate decay heat removal for normal, off-normal, and accident events such that the storage and transportation—including the CSNF—do not lose any of the other required safety functions:

- Casks: ensure continuity of the decay heat removal path through the canister and embedded cask steel;
- Canisters/overpacks (storage): ensure air inlet and outlet vents in the overpack do not become blocked.

In addition, some regulations include a requirement that the CSNF in the storage systems should not suffer “gross failures” due to loss of fuel cladding integrity during storage. If the cladding is already known to have a breach at the time of loading, a special “can” or “bottle” is used to contain potential loss of particulates during storage and subsequent transportation. Since the CSNF cannot be examined once it is sealed inside the cask or canister, experimental data in conjunction with in-reactor irradiation history information is used to predict CSNF integrity.

Regulatory standards

The basis for many regulations governing CSNF storage is the IAEA Specific Safety Requirement SSR-4 (IAEA, 2017). The example of a country-specific set of CSNF storage regulations is that from the USA. The US Nuclear Regulatory Commission (NRC) licenses storage of SNF in dry storage systems and in pools (other than onsite reactor spent nuclear fuel pools) under the regulation of 10 Code of Federal Regulation (CFR) 72, “Licensing Requirements for the Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.” Under Part 72, applicants can pursue a site-specific license or a general license. The regulations for a site-specific license govern both at-reactor ISFSIs and away-from-reactor storage facilities. As required in 10 CFR 72.212, to use casks under a general license, the applicant must demonstrate that the conditions in the casks’ Certificates of Compliance (CoCs) are met and that the site design, site-specific environmental conditions, and Design Basis Accidents (DBAs) including natural phenomena are bounded by the terms of the general license. In cases where the design conditions fall outside the design envelope of the cask CoC, such as high seismic areas, a site-specific license is needed. The initial licenses granted under 10 CFR 72.42 (a) were limited to 20 years. However, effective May 17, 2011, 10 CFR 72.42(a) was officially changed to allow an initial license period of up to 40 years and license extensions of up to 40 years.

Standard review plans

The NRC conducts its review of CSNF storage applications under two review plans: NUREG-1536, “Standard Review Plan for Dry Cask Storage System” (NRC, 2010) for dry storage systems and NUREG-1567, “Standard Review Plan for Spent Fuel Storage Facilities” (NRC, 2000) for storage facilities. NUREG-1927, “Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance” (NRC, 2016) is a review plan for renewal of ISFSI licenses and DCSS CoCs. These standard review plans are officially provided for NRC reviewers of storage license applications. However, it is common for the applicants themselves to closely follow the format laid out in these plans in the applicants’ Safety Analysis Reports (SARs) submitted with a storage license application.

Regulatory guides

Regulatory Guides provide guidance to licensees and applicants on how to implement specific parts of NRC regulations, techniques used by NRC staff to evaluate specific problems, and data needed by NRC staff in its review of applications. Regulatory Guides are reviewed about every 10 years. The Regulatory Guides applicable to dry storage are found in the Fuels and Materials Facilities

(Division 3) section found at <https://www.nrc.gov/reading-rm/doc-collections/reg-guides/fuels-materials/rg/>. The following is a listing of select Regulatory Guides with brief descriptions relevant to extended dry storage:

- Regulatory Guide 3.48, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage).”
- Regulatory Guide 3.50, “Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste.”
- Regulatory Guide 3.54, “Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation.” This revision provides an updated methodology for determining heat generation rates for PWR and BWR fuel with greater flexibility. The scope of the previous revision was restricted to PWR fuel with a maximum burnup of 50 GWd/MTU and BWR SNF with a maximum burnup of 45 GWd/MTU. This revision allows the loading of higher burnup SNF by using more accurate methods for calculating decay heat and covering a wider range of fuel characteristics and operating history.
- Regulatory Guide 3.60, “Design of an Independent Spent Fuel Storage Installation (Dry Storage).”
- Regulatory Guide 3.61, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask.” The NRC recommends revising this Regulatory Guide due to outdated guidance.
- Regulatory Guide 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks.” The NRC recommends revising this Regulatory Guide due to outdated guidance.
- Regulatory Guide 3.72, “Guidance for Implementation of 10 CFR 72.48, Changes, Tests, and Experiments.”
- Regulatory Guide 3.73, “Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations.”

The need for longer-term storage

Due to decades of delays in the availability of geologic disposal in almost all countries, dry storage systems and the SNF contained therein must maintain their safety functions for longer than originally anticipated. Understanding the longer-term behavior of the SNF and the dry storage systems is now an active area of both R&D and licensing. The licensees of existing storage systems must demonstrate that the dry storage systems will continue to comply with the relevant regulations over longer term. If the dry storage system is often licensed for transportation, as well, then it is important that any degradation be considered for its effect on meeting transportation regulations. The regulators in some countries are requiring inspections of the storage systems prior to granting license renewals. Long-term degradation mechanisms include, but are not limited to: failure of the SNF cladding due to creep or embrittlement, corrosion of the steel canister containing the SNF or corrosion or degradation of radiation shielding such as polymeric neutron shielding and concrete neutron and gamma shielding. As of the end of 2019, no failures of dry storage systems have occurred. However, since disposal in some countries may not become available until near or beyond the end of this century, many of the storage systems will need to continue performing their safety functions for decades to come.

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The Chemical Basis for Separating Recycling Materials by Hydro-Processes

Robin Taylor, National Nuclear Laboratory, Central Laboratory, Sellafield, Seascale, United Kingdom

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Glossary

Advanced gas cooled reactor (AGR) UK reactor type.

BUTEX UK process for reprocessing based on dibutyl carbitol as the extractant.

Carbon, hydrogen, oxygen, nitrogen (CHON) Principle for decomposable organic agents added to processes.

Co-extraction of actinides (COEX™) French process for co-processing U and Pu.

Commercial spent nuclear fuel (CSNF) Irradiated fuel from power reactors, also referred to as used nuclear fuel.

Critical aqueous concentration (CAC) Point at which a third phase forms based on species concentration in the aqueous phase.

Decontamination factor (DF) Ratio of an impurity in a feed compared with a product.

Dibutyl phosphate (DBP) TBP degradation product.

Dissolver off gas (DOG) Aerial effluent from dissolution process.

Distribution ratio of species X (D_X) Ratio of concentration of species X in organic phase to concentration in the aqueous phase (usually at O/A of 1).

High efficiency particulate (HEPA) Filter used to remove particles from aerial effluents.

Highly active (HA) Containing very radioactive and heat generating fission products.

Hydrogenated tetrapropylene (TPH) Industrial diluent for the extractant used as the organic phase in reprocessing.

Insoluble fission product (IFP) Isotopes produced by nuclear fission that do not dissolve in nitric acid.

Light water reactor (LWR) Generic description for thermal reactors with H₂O coolant.

Limiting organic concentration (LOC) Point at which a third phase forms based on species concentration in the organic phase.

Mixed oxide (MOx) Usually referring to (U, Pu)O₂ fuels.

Monobutyl phosphate (MBP) TBP degradation product.

Nitrogen oxide gases (NOX) Generic term for gases containing nitrogen and oxygen atoms, primarily NO and NO₂.

Odourless kerosene (OK) Industrial diluent for the extractant used as the organic phase in reprocessing.

Organic to aqueous flow ratio (O/A) Ratio of flowrates in the SX process.

Plutonium purification (PP) A solvent extraction cycle in a reprocessing plant.

Plutonium Uranium Reduction EXtraction (PUREX) Process used in commercial scale reprocessing.

Reduction oxidation process (REDOX) Process for reprocessing based on methyl isobutyl ketone.

Research and development (R&D) Scientific research and technology development.

Solvent extraction (SX) Chemical separation method also referred to as liquid-liquid extraction.

Spent nuclear fuel (SNF) Irradiated fuel, also referred to as used nuclear fuel.

Thermal denitration (TDN) A route for conversion of uranyl nitrate to uranium oxide.

Tri-*n*-butylphosphate (TBP) Extractant used in the PUREX process.

Uranium oxide (UO_x) Nuclear fuel material.

Uranium purification (UP) A solvent extraction cycle in a reprocessing plant.

Introduction

The use of nuclear energy inevitably leads to the generation of used (or “spent”) nuclear fuel. This commercial spent nuclear fuel (CSNF) can either be interim stored before direct disposal in a geological repository (the “open” fuel cycle) or recycled into new fuel for reactors (the “closed” fuel cycle). There are different degrees of recycling possible, leading to variations of the closed fuel cycle, but to date industrial scale implementation has been limited to recycle of uranium (U) and plutonium (Pu) by hydrometallurgical processes (McKay et al., 1990; Wilson, 1996; Lecomte, 2008; Nash and Lumetta, 2011; Taylor, 2015).

Typically, ~97% of CSNF can be recovered and recycled into new fuels; this process, known as reprocessing, has been widely used over the last ~60 years to recover the major reusable actinide elements: U and Pu. However, with concerns around Pu proliferation, the near-term costs of reprocessing by comparison with interim storage (where most of costs are delayed to the longer term) and the wastes generated (Cashmore et al., 2011; Poinssot et al., 2015; Williams, 2015), reprocessing at the commercial scale is currently only undertaken in a small number of countries (WNA, 2018). In conjunction with a lack of progress on geological repositories, this has led to the situation where there is about 266,000 t (te) of SNF in stores around the world (2015 figures), a figure that could rise to about 570,000 t by 2050 (Liu et al., 2005; IAEA, 2016). Eventually, decisions on SNF disposition must be taken and these stored fuels must be either reprocessed or disposed of in geological repositories designed for high-level radioactive wastes (EASAC, 2014).

Currently, nuclear fuel reprocessing at the commercial scale uses the PUREX (Plutonium Uranium Reduction EXtraction) process to recover U and Pu products (Lanham and Runion, 1949). First developed in the United States in the late 1940s, this is a hydrometallurgical process in which SNF is initially dissolved in refluxing nitric acid and then solvent extraction (SX) is used to extract U and Pu from a nitric acid solution containing the fission products into a kerosene diluent containing the extractant tri-*n*-butyl phosphate (TBP) (McKay et al., 1990; Denniss and Jeapes, 1996; Herbst et al., 2011). Separation of U from Pu is then achieved by manipulation of the Pu oxidation state and/or nitric acid concentration (Miles, 1990). Further SX cycles can be used to purify the separate U and Pu products before they are converted to solid oxides (Patterson and Parkes, 1996). The quite remarkable properties of TBP and success of the PUREX process are indicated by the fact that even now it remains almost unchallenged as the process of choice for CSNF reprocessing (Herbst et al., 2011; Netter, 2012; Nash and Nilsson, 2015).

This article provides an overview of the chemical basis to nuclear fuel reprocessing by the PUREX process focusing on the dissolution of CSNF and the separation (SX) process that generates pure U and Pu products. How the chemical separation process has been upscaled to industrial scales will be discussed with reference to second and third generation reprocessing plants operated at the Sellafield site in the United Kingdom. The Magnox Reprocessing Plant was commissioned in 1964 to reprocess uranium metal fuels from the UK’s Magnox reactors. The Thorp plant, (1994–2018) reprocessed higher burn up uranium oxide fuels (Phillips and Milliken, 2000; Benedicic, 2017).

Nuclear fuel reprocessing by the PUREX process

Nuclear fuel reprocessing is generally used to describe the separation and purification of reusable uranium and plutonium products from irradiated nuclear fuel. The recovered U and Pu can then be converted in to new uranium oxide (UO_x) or mixed oxide (MO_x) fuels for recycle to reactors (Poinssot et al., 2012; Behar, 2014). The highly active (HA) fission product wastes are evaporated and then vitrified ready for disposal. By far the most successful reprocessing technology to date has been the PUREX process, which uses SX between aqueous nitric acid solutions and organic solutions of tri-*n*-butyl phosphate (TBP) diluted in a paraffinic diluent, such as Exxsol D-80 or hydrogenated tetrapropylene (TPH) (Denniss and Jeapes, 1996; Lecomte, 2008; Herbst et al., 2011; Netter, 2012).

A brief history of reprocessing

During the Manhattan Project and afterwards, large-scale plutonium processing used a bismuth phosphate precipitation method; this was then superseded by solvent extraction processes that could be run continuously, recover uranium and plutonium and

generated less wastes. The REDOX process, using methyl isobutyl ketone $[(CH_3)_2CHCH_2C(O)CH_3]$ or hexone as the solvent, was used at the Hanford site (United States) and the BUTEX process, based on dibutyl carbitol $(CH_3(CH_2)_3O(CH_2)_2)_2O$, was used in the United Kingdom at Windscale. However, the PUREX process was quickly shown to possess a number of process chemistry and chemical engineering advantages over these other processes, including being more stable, less flammable, salt-free extraction with better separations (Choppin, 2002). The PUREX process was first deployed at Savannah River in 1954 and has remained the dominant technology for processing nuclear fuel since the 1950s/60s (Nash and Nilsson, 2015). As of 2020, large scale PUREX reprocessing plants (~ 800 t/a) were in operation in the United Kingdom (Sellafield), France (La Hague), and Japan (commissioning at Rokkasho) but a number of other countries have experience of PUREX reprocessing at various scales, e.g. United States, Germany, Russia, Belgium, India (Netter, 2012). China, India and Russia are at various stages of developing new reprocessing capacity (Guoan and Taihong, 2015; Natarajan, 2017; WNA, 2020).

Historically, the United Kingdom, used in this article as a case study, has operated four reprocessing plants, although all will be in decommissioning by the early 2020s, see Table 1 for details (Benedicic, 2017). The two commercial scale reprocessing plants operated were the Magnox reprocessing plant which reprocessed uranium metal fuel from the domestic Magnox reactors and Thorp (Thermal Oxide Reprocessing Plant) which reprocessed domestic (AGR: Advanced Gas cooled Reactor) and overseas oxide (LWR: Light Water Reactor) fuels. Characteristics of these two plants are compared in Table 2 (Phillips and Milliken, 2000).

Reprocessing plant layout

A PUREX reprocessing plant comprises a number of facilities or capabilities: (a) ponds to receive and store spent fuel; (b) the Head End plant to prepare the fuel for reprocessing (disassembly) and to convert the fuel to a solution in HNO_3 ready for (c) chemical separation using solvent extraction to produce purified aqueous nitrate products that can be (d) converted to solid oxide products ready for manufacturing into new fuel. A substantial supporting infrastructure is necessary to (e) provide reagents needed for the process and (f) treat solid wastes and liquid and gaseous effluents arising from reprocessing operations.

This article covers the chemistry of the dissolution and conditioning process that prepares the aqueous phase for SX and the SX process itself. PUREX process flowsheets are all broadly similar with a number of solvent extraction cycles to separate and purify U and Pu from all other elements (a single SX cycle covers the extraction of a solute into the organic phase from the aqueous phase and the return of the solute back to the aqueous phase in a stripping or backwashing stage). The radiotoxicity of the elements and isotopes within CSNF means that reprocessing must obtain both very high recoveries of the product elements (U and Pu) and very high purities of these products from other isotopes. In separations chemistry, commonly, these two ambitions of high yields versus product purity are in opposition and this makes the reprocessing of CSNFG quite challenging from a technical perspective. Hence, to achieve U and Pu products of the desired purity the SX process must provide the requisite decontamination from fission products in the spent fuel. Decontamination factors are used as a measure of the process performance and are simply the ratio of the species concentration in the feed to the concentration in the product (Rydberg et al., 2004a,b). Over the entire SX process typical decontamination factors in the order of 10^6 – 10^8 are achieved so that U and Pu products can be used in fuel fabrication or enrichment plants without the need for remote handling in hot cells (Netter, 2012). As can be seen in Section “Magnox and thorp reprocessing” (in the evolution from Magnox to Thorp reprocessing), process development has been in the direction of reducing the number of solvent extraction cycles, adopting an early separation of Pu from U, using salt free reagents to minimize waste arisings and intensifying the contacting equipment, e.g., pulsed columns rather than mixer-settlers.

Table 1 UK reprocessing plants.

Dates	Plant	Description/status
1951–72	Windscale	Reprocessing plant for metal fuel, early commercial reactor fuel and dissolution of prototype oxide fuel
1960–96	Dounreay	Reprocessing of a variety of fuels from the UK fast reactor development programs
1964	Magnox	2nd reprocessing plant at Sellafield, reprocessed commercial Magnox clad uranium metal fuel
1994–2018	Thorp	3rd reprocessing plant at Sellafield, reprocessed commercial thermal oxide fuels (LWR and AGR)

Table 2 Characteristics of the fuels reprocessed and processes implemented in the Magnox and Thorp reprocessing plants.

Magnox	Thorp
Al/Mg clad U metal fuel from Magnox reactors	Stainless steel or zircaloy clad oxide fuel from AGRs and LWRs
Natural enrichment	Enriched U-235
Low burnup and short cooled fuel	High burnup and long cooled fuel
Mixer-settlers and 20% TBP	Pulsed columns and 30% TBP on Pu streams
Four cycle SX—“late split” flowsheet	Three cycle SX—“early split” flowsheet
Continuous dissolver	Batch dissolver
Salt containing flowsheet (ferrous, sulfate, Na ions)	Salt free flowsheet [U(IV) reductant, nitrogen oxide (NOX) gases]
Linked series of plants	Integrated plant

PUREX process chemistry

Dissolution process

Basic concepts

The “head-end” of an aqueous CSNF reprocessing plant converts the fuel into a form that is suitable for the solvent extraction purification process. Firstly, the disassembled CSNF must be transported from the fuel storage pool to a highly active cell where heavy engineering equipment is used to prepare the CSNF for chemical processing. Typically, this is done by mechanically shearing (cutting) the fuel rods into small pieces that can be fed to a dissolver for leaching the fuel out of the chopped pieces into hot nitric acid. The leached zircaloy hulls (cladding) are immobilized into a wasteform for disposal.

After the mechanical processes, chemical processing of CSNF in the head end involves fuel dissolution and liquor conditioning to prepare the feed for the separation process, whilst chemical and radioactive off-gases must also be treated to an appropriate degree. Dissolvers can be operated in either continuous or batch modes and whilst the basic chemistry is the same there are some subtle differences. As well as acid fumes, during dissolution the dissolvers are operated to allow efficient removal of $^{14}\text{CO}_2$, ^{129}I and ^{85}Kr ; a fraction of the tritium is also evolved. Dissolvers have air added to condensers to promote increased re-conversion of nitrogen oxides (NOx) generated from the dissolution to nitric acid thereby decreasing the amount of nitric acid used and the amount of nitrogen oxide fed to the dissolver off-gas (DOG) treatment plant. This technology is termed “fumeless” dissolution when operated at high efficiencies. Dissolvers use nitric acid recombination columns and caustic scrubbers to clean the gaseous streams, which are passed through high efficiency particulate (HEPA) filters before discharge via a stack. The caustic scrubbers efficiently remove ^{14}C which can be precipitated with barium carbonate (Maher, 2015).

There are several steps that are usually carried out after dissolution and before feeding to a solvent extraction process, these include:

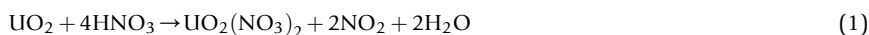
- Iodine removal.
- Conversion of plutonium to the tetravalent state.
- Feed clarification.

Iodine removal is usually carried out during the dissolver cycle. An air sparge can be used to improve the efficiency. During irradiation of oxide fuel some of the fission product metals are not soluble in the fuel oxide structure and migrate to the fuel grain boundaries. These inclusions do not dissolve completely during a standard dissolution cycle and the residues are termed insoluble fission products (IFPs). The IFPs comprise of alloys of ruthenium, rhodium, palladium, molybdenum and technetium and may contain other metals including plutonium. A feed clarification step must remove solids that could interfere downstream with the SX process or block SX equipment—centrifugation or filtration are typical technologies.

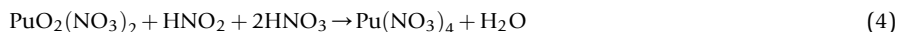
Uranium oxide dissolution and the adjustment of plutonium oxidation states (conditioning) are specifically discussed in the next section.

Dissolution chemistry

The rate of dissolution of uranium oxide fuels is dependent on several factors including: nitric acid concentration, temperature, surface area, nitrous acid, volume, sparging and stirring. In plants where fuel rods are sheared and leached, dissolution efficiency will be affected by the size and shape of the hulls which can cause gas-locking. Overall, dissolution is thought to proceed according to Eqs. (1) and (3) where the key role of nitrous acid is evident (where $0 < x < 1$ in Eq. (3)) (Sakurai et al., 1988).



After dissolution and iodine removal, plutonium is usually present in the solution as a mixture of tetravalent and hexavalent oxidation states, Pu(IV) and Pu(VI). The tetravalent state is optimal for current TBP-based solvent extraction processes. Nitrogen dioxide (NOx gas) can be used to condition the plutonium to the tetravalent state via formation of nitrous acid and Eq. (4).



Solvent extraction (SX) process

Basic concepts

It is the choice of solvent extraction for actinide separation that drives many of the upstream and downstream processes; SX thus lies at the heart of nuclear fuel reprocessing. There are various advantages to SX including:

- Safe, low risk, mature technology.
- Easy to scale up and industrialize.
- High selectivities and high product purities achievable.

- Low temperature, low pressure processes.
- Can recycle and re-use solvent and aqueous phases.
- High heavy metal loading capacity.

Disadvantages include:

- Solvent degradation.
- Entrainment.
- Generation of secondary wastes (organic solvent and aqueous effluents).

An important concept is that industrial SX processes are run counter-current. That is, the aqueous and organic phases flow in opposite directions through a series of contactors, which are usually unit stages with both mixing and settling zones. Across this series of contactors, the feed is progressively depleted of the extractable solute until it exits the last contactor as a lean raffinate. The extracting phase, flowing in the opposite direction, exits the first contactor as a loaded product (Rydberg et al., 2004a,b). Fig. 1 illustrates this concept.

In applying SX to reprocessing of CSNF there are a range of additional problems to overcome compared to its use in more conventional process industries such as hydrometallurgy. These include radiation and radioactivity requiring remote operations, solvent radiolysis, criticality safety and radioactive waste management. The chemistry of the actinide elements and fission products in nitric acid also need to be considered, together with the choice of technology for solvent extraction. There are three types of contacting equipment typically considered for modern reprocessing plants: mixer-settlers, pulsed columns and centrifugal contactors. The principles of how these technologies operate are outside the scope of this paper and interested readers are referred to the references of Arm and Phillips (Arm and Phillips, 2011) and Hanson (2015) who describe in detail the engineering of solvent extraction in the PUREX process. It is sufficient here to note that these “contactors” must intimately mix the two phases (by generating a dispersed phase of very small droplets in a continuous phase) which enables transfer of chemical species to occur across the interface between the dispersed and continuous phases. The contactor must then separate out the phases (by gravity or centrifugal force) so they can be transported to the next stage of the process. Thus, from the chemical viewpoint, one important aspect to consider is whether the kinetics (rates of chemical reaction and/or rates of mass transfer between aqueous and organic phases) are compatible with the residence times in the solvent extraction contactor, which decrease in the order: mixer-settlers > pulsed columns > centrifugal contactors.

U and Pu extraction and separation process chemistry

In the PUREX process the purification of U and Pu from fission products is achieved by extraction of hexavalent and tetravalent metal nitrate complexes into 20–30% TBP (Eqs. (5) and (6)). The uranyl TBP complex is illustrated in Fig. 2 where it is seen that bidentate nitrate ions charge balance the UO_2^{2+} ion enabling two TBP molecules to complex the neutral nitrate species via the oxygen atoms of the P=O group (Lecomte, 2008). This species is then easily solvated in the non-polar organic diluent; that is, it is transferred across an aqueous/solvent interface from the bulk aqueous to bulk organic phase. Fig. 3 illustrates the classical 4-stage mechanism for this process (Rydberg et al., 2004a,b). In SX the amount of solute (e.g., uranium) extracted is usefully described by the distribution ratio (D), i.e., the ratio of a solute in the organic phase to that in the aqueous phase at equilibrium ($O/A = 1$), which itself is related to the equilibrium constant as shown in Scheme 1. In this example, it is apparent that distribution ratios for U(VI) are dependent on the nitrate concentration and uncomplexed or “free” TBP concentration (Rydberg et al., 2004a,b). In practice, this leads to a variation in $D_{\text{U(VI)}}$ with nitric acid as illustrated in Fig. 4.



U and Pu are then separated from each other by reductive stripping of Pu, whereby the similarly extractable tetravalent Pu is reduced to inextractable trivalent Pu and stripped to an aqueous phase, leaving U(VI) in the organic phase (Denniss and Jeapes, 1996). The distribution ratios for Pu(IV) and Pu(III) ions are compared in Fig. 2. Typically, a reductant such as uranous nitrate, $\text{U}(\text{NO}_3)_4$, is used to reduce Pu(IV) (Eq. 7) although ferrous ions, hydroxylamine (NH_2OH) and other reducing agents have been either used or proposed for this purpose, as have electrochemical and photochemical methods (Newton and Baker, 1967; Toth et al., 1977; Petrich et al., 1986; Miles, 1990). Of these other reducing agents, hydroxylamine is particularly relevant since

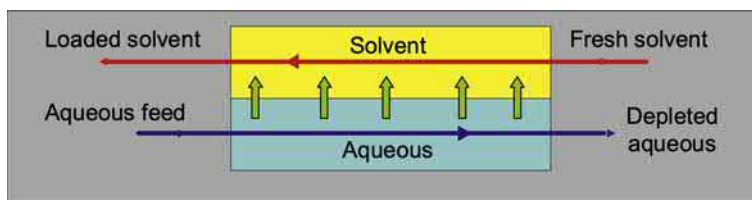


Fig. 1 Principle of counter-current solvent extraction.

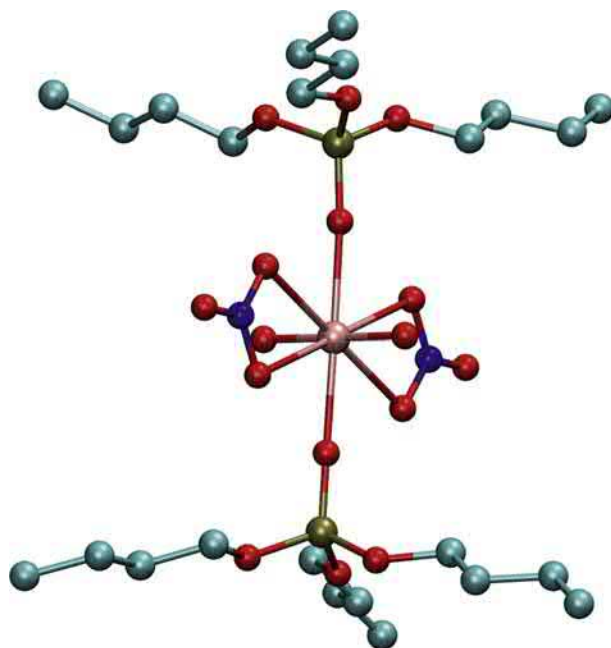


Fig. 2 Structure of $\text{UO}_2(\text{NO}_3)_2 \cdot 2\text{TBP}$ (hydrogen atoms not included).

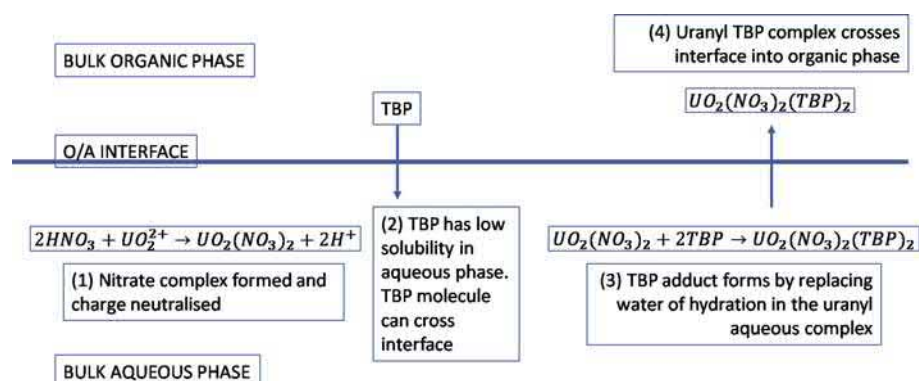
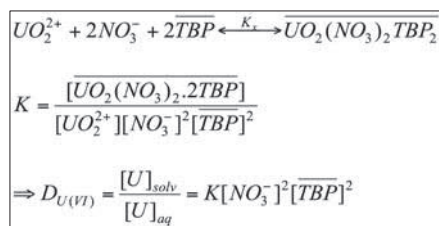
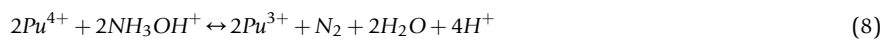
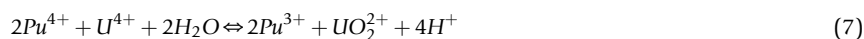


Fig. 3 Proposed mechanism of U(VI) extraction by TBP (Rydberg et al., 2004a,b). Drawn by the author based on figure in Rydberg J, Cox M, Musikas C, Choppin GR (eds.) (2004) Solvent Extraction Principles and Practice. 2nd edn, Revised and Expanded, pp. 1–750, New York: Marcel Dekker, Inc.



Scheme 1 SX equilibria and distribution ratio using uranyl nitrate extraction into TBP.

it is salt-free and is suitable for use in both the uranium and plutonium purification cycles. Pu(IV) reduction by hydroxylamine (Eq. 8, excess hydroxylamine) is a quite complex reaction as it is strongly inhibited by acidity and the product Pu(III) ions (Eq. 9), where k is the rate constant and K_{Pu} is the dissociation constant for $\text{Pu}(\text{NO}_3)^+$ dissociation (Barney, 1976).



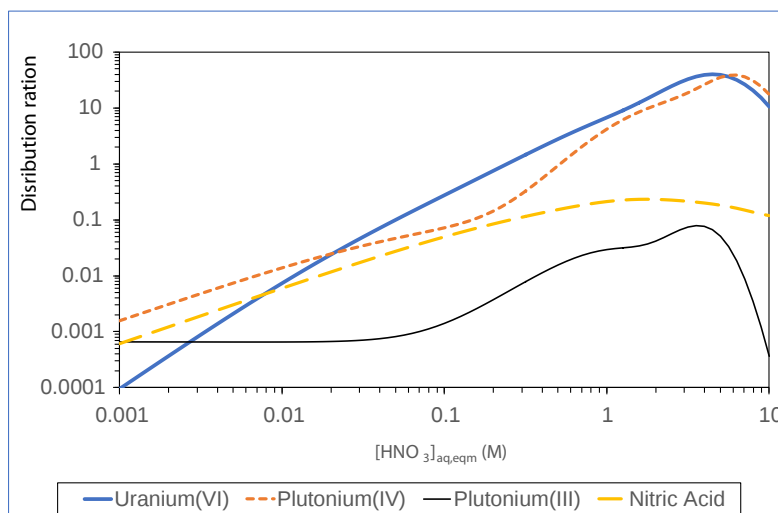


Fig. 4 U and Pu distribution ratios as a function of nitric acid concentration in the aqueous phase at equilibrium extraction of UO_2^{2+} , Pu^{4+} , Pu^{3+} ions and nitric acid into 30% TBP/diluent at 20 °C (modelled fits to experimental distribution ratios).

$$-\frac{d[\text{Pu}^{4+}]}{dt} = \frac{k[\text{Pu}^{4+}]^2[\text{NH}_3\text{OH}^+]^2}{[\text{Pu}^{3+}]^2[\text{H}^+]^4(K_{\text{Pu}} + [\text{NO}_3^-])^2} \quad (9)$$

A significant complication in the PUREX process is that Pu(III) is quite easily re-oxidized to Pu(IV) by nitric acid in a series of reactions that are autocatalyzed by nitrous acid (Eqs. 10 and 11) (Koltunov, 1974; Miles, 1990). To interrupt this process and stabilize Pu as Pu(III), an aqueous phase nitrous acid scavenger such as hydrazine is added to the U/Pu separation stages (Eqs. 12 and 13) (Perrott and Stedman, 1977).

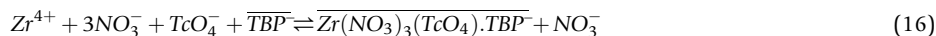
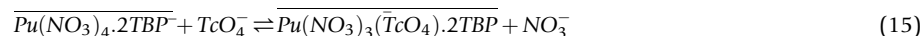
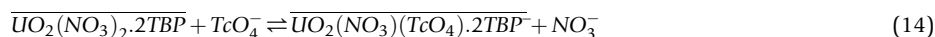


However, although it is nominally considered to be “inextractable,” trace quantities of Pu(III) are extracted in to the solvent phase where, in the absence of a scavenger, re-oxidation to Pu(IV) does occur. In 2-phase mixed systems (i.e., HNO_3/TBP) this process is enhanced as the system tries to equilibrate, the ultimate effect being a requirement to add a substantial excess of reductant to maintain Pu in the trivalent state (Miles, 1990; Fox et al., 2006a,b). Once, U and Pu are separated, U can be backwashed into the aqueous phase by simply reducing the acidity to a low value, typically ~ 0.01 M HNO_3 . As U extraction is exothermic to improve the kinetics and shift the equilibrium of Eq. (6) to the left, backwashing is usually performed at an elevated temperature (Bagawde et al., 1978).

The solvent raffinates from the process, after U and Pu backwashing, contain degradation products due to radiolysis and hydrolysis of TBP. The main degradation products of concern are dibutyl phosphate (DBP) and monobutyl phosphate (MBP) since they can form strong complexes with actinide ions and some fission products (such as Zr), retaining them in the organic phase or forming cruds at the interface. This can have detrimental effects on process performance due to increasing losses of U and Pu to raffinate streams, deterioration of decontamination factors for some fission products, poor phase separation or risk of blockages. To avoid accumulation of degradation products in the solvent it is essential that there is a solvent wash process to remove these species and allow recycle and re-use of the solvent in the process. Typically, a standard solvent wash procedure with alkaline reagents is used to remove the acidic degradation products from the solvent (Lecomte, 2008).

Technetium

Technetium (^{99}Tc) is a fission product with a long half-life ($t_{1/2} = 2.11$ a) that increases in CSNF with increasing burn up. It is stable in acidic solutions as the anionic pertechnetate ion (TcO_4^-) and, as such, can replace nitrate ions in TBP complexes of U and Pu (Eqs. 14 and 15). It also strongly co-extracts with zirconium into TBP (Eq. 16) and, therefore, is a problem in PUREX flowsheets as it can be co-extracted with the (U, Pu)-containing organic phase (Herbst et al., 2011). Garraway and Wilson (Garraway and Wilson, 1985) reported a 1:1 complex for the Zr/Tc complex with TBP although separately Zr and Tc extract as higher complexes.



A further problem for the PUREX process is that technetium reacts with hydrazine which is used in the U/Pu separation stage to stabilize Pu(III) (Miles 1990, Denniss and Jeapes, 1996; Lecomte, 2008). Initially, there is a slow initiation phase in which Tc(VII) is reduced in stages to lower oxidation states leading to the build-up of Tc(IV). Tc(IV) can be re-oxidized either by remaining Tc(VII) ions or by nitrate and this leads to a catalytic cycle in which hydrazine is rapidly consumed. U(IV) enhances this process by initially reducing Tc(VII) quickly to Tc(IV). There is some debate about the exact mechanism but, based on references (Garraway and Wilson, 1984; Koltunov et al., 1986; Boukis, 1988), the general cycle of reactions with hydrazine is summarized in Fig. 5.

To date two main strategies have been employed to address the issues of technetium in PUREX reprocessing, viz the addition of a technetium rejection stage or the adjustment of conditions in the U/Pu separation to reduce the effects of this reaction to tolerable levels by adjusting the operating conditions in the pulsed columns where the U/Pu separation occurs. The basis of the former strategy, as used in the French La Hague UP3 reprocessing plant (Baron et al., 1993; Herbst et al., 2011), is illustrated in Fig. 6 which shows how Tc distribution ratios decrease as nitrate ion is increased.

Neptunium

Current practice in reprocessing plants is to treat the ‘minor actinide’ Np as a waste and to purify U and Pu products from Np, which is routed to high level wastes for vitrification (Fox et al., 2006b; Lecomte, 2008). However, this is not straightforward as Np exhibits a complex process chemistry. Np in solution can exist in oxidation states from III to VII; although in the PUREX process Np is usually present only as a mixture of Np(IV), Np(V) and Np(VI), as the ionic species Np^{4+} , NpO_2^+ and NpO_2^{2+} respectively (Drake, 1990; Yoshida et al., 2006). Acidity, nitrate ion concentration and temperature influence the equilibrium distribution of Np between these three oxidation states in nitric acid which readily inter-convert through reactions with $\text{HNO}_2/\text{HNO}_3$ and disproportionation/re-proportionation reactions (Drake, 1990). In the first extraction section of the PUREX process, the equilibrium between oxidation states (V) and (VI) (given in Eq. 17) is most important in determining the speciation and hence degree of extraction into the solvent phase. This equilibrium is dependent on the respective rates of the forward and reverse reactions which display quite complex kinetics due to the dual role of nitrous acid. From Eq. (17) it is apparent that HNO_2 reduces Np(VI) to Np(V) but it also acts as a catalyst for the oxidation of Np(V). Consequently, the kinetics and equilibria of this reaction have been studied quite extensively over the last 60 years (Siddall and Dukes, 1959; Gourisse and Gautier, 1969; Gourisse, 1971; Koltunov, 1974; Moulin, 1978; Tochiyama et al., 1995a, b; Gregson et al., 2012; Chen et al., 2016, 2017). For instance, Moulin (1978) proposed that the rate of oxidation is dependent on the ratio of nitrous acid to Np(V) and is given by Eq. (18) where k_x – k_e are individual rate constants.

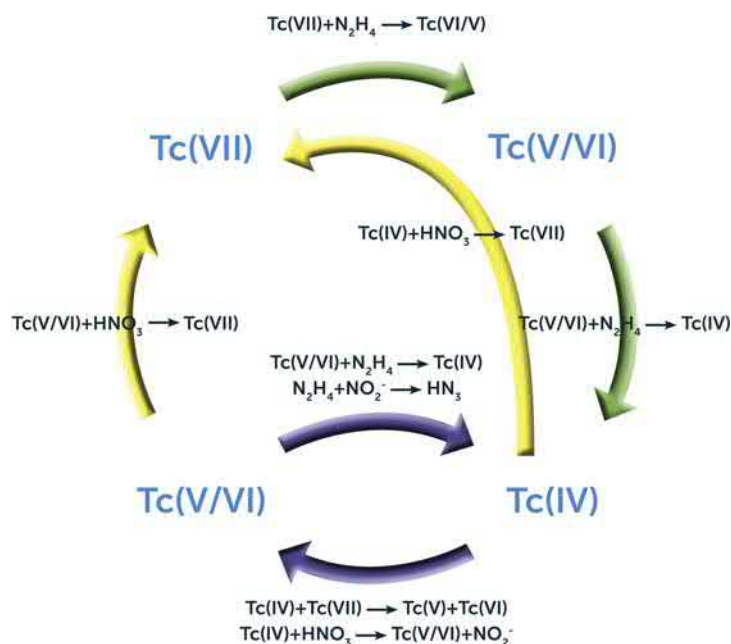


Fig. 5 Technetium—hydrazine reaction cycles. Green arrows are initiation reactions, purple arrows are the catalytic cycle and yellow arrows are termination reactions.

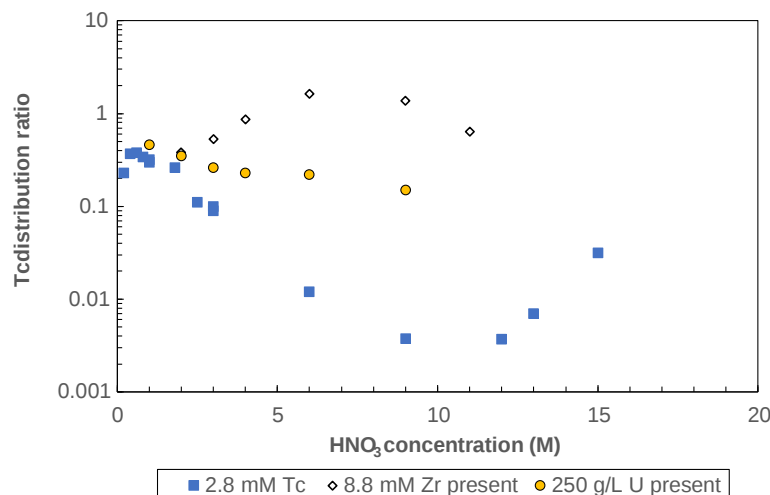
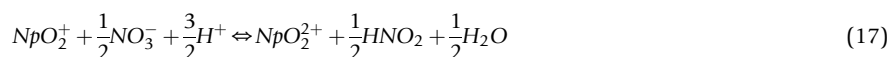


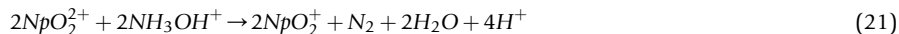
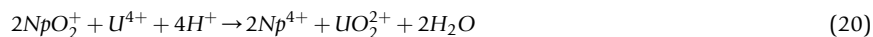
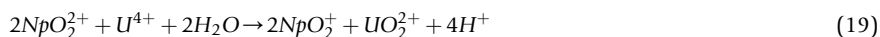
Fig. 6 Tc distribution ratios as a function of nitric acid, also with zirconium and uranium present in solution (35 °C, extraction into 30% TBP/OK). Drawn by the author based on data in Garraway J and Wilson PD (1985) Coextraction of pertechnetate and zirconium by tri-n-butyl phosphate. *Journal of the Less-Common Metals* 106(1): 183–192.



$$-\frac{d[\text{Np(V)}]}{dt} = \frac{(k_\gamma[\text{H}^+] + k_\delta[\text{H}^+]^2 + k_\epsilon[\text{HNO}_2])[\text{NO}_3^-][\text{HNO}_2][\text{Np(V)}]}{k_\alpha[\text{HNO}_2] + k_\beta[\text{Np(V)}]} \quad (18)$$

The extraction of Np into TBP is also not straightforward. Np(IV) and Np(VI) are considered extractable, forming similar complexes to Pu(IV) and U(VI). Np(V) has a very low extractability into the PUREX solvent of 30% TBP/OK (tributyl phosphate diluted in odourless kerosene) and is generally considered inextractable (Drake, 1990).

As NOx conditioning of the dissolved CSNF solution in the Head End plant stabilizes Np as Np(V), partial oxidation to Np(VI) occurs by Eq. (17) in the initial (U, Pu) co-extraction stage in a reprocessing plant with the extent of oxidation depending on solution conditions and residence times in the contactor. This leads to Np being split between the Highly Active (HA) raffinate (Np(V) ions) and (U, Pu) containing TBP phase (Np(VI) ions) (Dinh et al., 2008; Taylor et al., 2013). In the U/Pu split, U^{4+} will reduce neptunium to Np^{4+} (Eqs. 19 and 20) which, depending on acidity and O/A ratio can be co-extracted with U (Newton and Baker, 1967). This necessitates a second solvent extraction cycle in which the U product is decontaminated from Np. This can be achieved by oxidizing Np^{4+} back to Np(V) and Np(VI) and then reducing any Np(VI) formed to inextractable Np(V) with a selective reducing agent. Hydroxylamine is commonly used for this purpose (Eq. (21)) as it is a rapid reductant for Np(VI) and does not reduce Np(V) to Np(IV) or reduce U(VI) (Koltunov and Tikhonov, 1977).



Third phase formation

So far, the basic PUREX process chemistry has been described through the extraction reactions for U and Pu, and their dependence on nitric acid concentration, and the redox chemistry of Pu, including the need for a nitrous acid scavenging agent to stabilize the Pu(III) state. The mechanism of extraction via reactions at the organic/aqueous interface has been discussed for uranyl nitrate as an example. The cases of two specific problematic elements were then considered with respect to their process chemistry: technetium and neptunium. However, there are many other facets of PUREX process chemistry (and chemical engineering) that must be considered when developing the flowsheet or supporting the safety case. One specific example is third phase formation. This is an effect in solvent extraction whereby the organic phase when heavily loaded with nitric acid and/or metal ions splits into two layers: a lighter diluent-rich and a heavier TBP-rich layer (Rao and Kolarik, 1996; Rydberg et al., 2004a,b). This has serious implications for process efficiency and process safety as it can cause phase inversion, leading to misrouting of U or Pu, and the third phase can become heavily loaded with Pu, thus presenting a criticality hazard (Plaue et al., 2006a,b,c). The PUREX process must be operated in regions well below the limits for third phase formation.

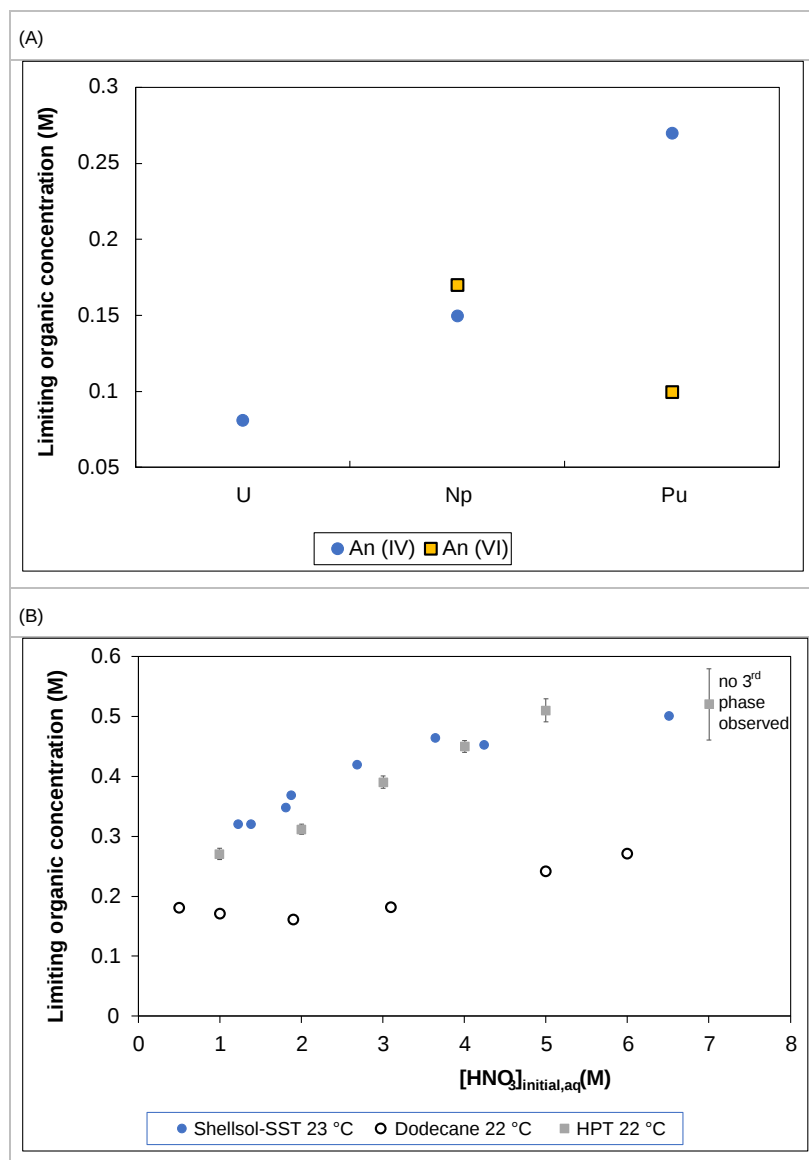
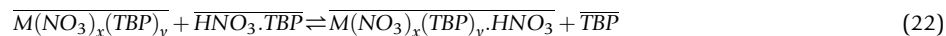


Fig. 7 Example third phase data for actinide extraction into TBP: (A) variation of Limiting Organic Concentrations (LOC) with tetravalent and hexavalent actinide ions (extraction from 7 M HNO_3 into 30% TBP/dodecane) (Plaue et al., 2006a,b,c); (B) Effect of different diluents on LOC (TPH, Shellsol-SST and dodecane) (Plaue et al., 2006a,b,c). (A): drawn by D Whittaker of NNL for the author based on data in Plaue J, Gelis A and Czerwinski K (2006b) Actinide third phase formation in 1.1 M TBP/nitric acid/alkane diluent systems. *Separation Science and Technology* 41(10): 2065–2074.; (B) Drawn by D Whittaker of NNL for the author based on data in Plaue J, Gelis A and Czerwinski K (2006c) Plutonium third phase formation in the 30% TBP/Nitric Acid/Hydrogenated polypropylene tetramer system. *Solvent Extraction and Ion Exchange* 24(3): 271–282.

In determining third phase boundaries, it is usual to find the point at which the third phase forms, this is defined either as the limiting organic concentration (LOC) or critical aqueous concentration (CAC). Example LOC data for the series of hexavalent and tetravalent actinide ions, U to Pu, are given in Fig. 7A (note that U(VI) does not form a third phase under these conditions) (Plaue et al., 2006a,b,c). Generally, in the PUREX process it is the tetravalent actinide ions that present the greater challenge with respect to third phase formation, Pu^{4+} being of primary concern. It is seen here and elsewhere that different metal ions behave quite differently with respect to third phase formation and that the LOC changes over the nitric acid range. For instance, plutonyl (VI) ions, PuO_2^{2+} are much more susceptible to third phase than the analogous UO_2^{2+} ions (Plaue et al., 2006a,b,c). To generalize, increasing temperature, extractant concentration, and ionic strength increase LOC, as does decreasing the carbon chain length of the diluent. In fact, third phase formation is very dependent on the nature of the diluent as can be seen in Fig. 7B (Plaue et al., 2006a,b,c). It is clearly established that branched alkanes reduce the likelihood of third phase and modifiers, such as alcohols or aromatics, that increase the polarity of the diluent also inhibit third phase formation. Industrially, diluents comprising mixtures of branched alkanes with densities similar to *n*-dodecane are used in reprocessing; for example, TPH or Exxsol D-80. Mechanistically, third phase

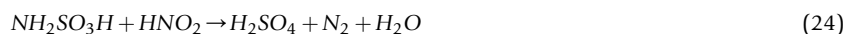
has been described in two basic ways: either by the formation of specific solvates (adducts) that form on high loading in the organic phase (e.g., Eq. 22 (Kumar and Koganti, 2003)) or by the association of small reverse micelles which form in the organic phase. These reverse micelles contain several TBP molecules around a polar core of metal ion, nitric acid and water molecules. As the micelles swell with increased organic phase loading, they interact through van der Waals forces until, at a critical value of interaction energy, phase separation takes place (Chiarizia et al., 2003a,b; Nave et al., 2004; Plaue et al., 2006a,b,c; Antonio et al., 2018). The Baxter sticky spheres model has been widely used to describe the particle interactions. The speciation in the organic phase that leads to third phase formation is a very topical area of fundamental study in the field of solvent extraction of actinides. Despite nearly 70 years of PUREX research, there remain gaps between results from these experimental speciation studies, solvent extraction calculations (Kumar and Koganti, 2003; McLachlan et al., 2016) and molecular modelling (Ivanov et al., 2017; Chandrasekar et al., 2019) that such basic research is now filling.



Magnox and Thorp reprocessing

Magnox reprocessing

The Magnox reprocessing plant was commissioned in 1964 to reprocess spent uranium metal fuel from the UK's Magnox reactors. Fuel rods were mechanically declad and uranium rods fed to a continuous dissolver prior to PUREX reprocessing. The solvent extraction flowsheet was based on a late split; that is, there was an initial co-decontamination cycle of U and Pu from fission products before U and Pu were separated from each other. U and Pu separation used ferrous ions (Eq. 23) to reduce Pu(IV) to Pu(III) and sulfamate ($NH_2SO_3^-$) ions to scavenge nitrous acid (Eq. 24) to prevent re-oxidation of Pu(III). Following U and Pu separation, separate purification cycles were employed to reach high DFs on the U and Pu products (i.e., four SX cycles in total). As Magnox reprocessed short cooled low burn up fuels, decontamination from ruthenium, zirconium and niobium fission products was one of the key issues. Due to low Pu concentrations, solvent extraction used 20% TBP and mixer-settlers. Solvent wash for high active, plutonium active and low active solvents from separate SX cycles were used to enable recycle of solvent within the process. U and Pu products were converted to solid oxides by thermal denitration and oxalate precipitation processes respectively. The use of salt-containing reagents in the process, such as ferrous sulfamate, increased the complexity of downstream waste treatment as well as waste volumes generated.



Thorp reprocessing

The United Kingdom Thorp reprocessing plant exemplified the modern approach to conventional PUREX reprocessing. Following disassembly, fuel rods were sheared into small pieces and leached in a batch dissolver. Following dissolution of the CSNF in nitric acid, U and Pu are purified by SX. As illustrated in Fig. 8 (Baron et al., 2019), Thorp used three solvent extraction cycles with an early split flowsheet, i.e., U and Pu are extracted away from the fission products and then separated from each other in the first or HA cycle of solvent extraction. This was achieved despite Thorp processing higher burn-up, enriched oxide fuel from AGRs and LWRs. Unlike Magnox, Thorp used only salt free reagents in the core separation process. U and Pu were separated from each other by a mixture of U(IV) and hydrazine (N_2H_4), as indicated by Eq. (7) and Eq. (12) above. Due to the co-extraction with U, Pu and Zr (Eqs. 14 and 16) and the reactions with U(IV) and hydrazine (Fig. 5), Tc mainly followed the Pu product. This contrasts with Np which was mainly split between the HA raffinate and the U product due to the equilibrium between inextractable Np(V) and extractable Np(VI) (Eq. 17). Following conditioning of the backwashed aqueous products, there were then two further SX cycles to purify the uranium product stream and the plutonium product stream. Uranium purification (UP) provided an additional decontamination from neptunium and residual plutonium whilst plutonium purification (PP) removed technetium from the product. Both UP and PP increased the overall decontamination factors from fission products such as ^{106}Ru , $^{134,137}Cs$ and ^{144}Ce (Phillips, 1996; Burrows et al., 2006). UP and PP cycles used hydroxylamine nitrate as the reducing agent (see Eq. 8 and Eq. 21).

For improved criticality safety and to reduce solvent degradation, pulsed columns were used for the Pu bearing streams whereas mixer-settlers were used for U-streams post U/Pu separation. As in Magnox reprocessing, solvent wash cycles, based on a carbonate washing process, enabled used solvent to be recycled back into the process. Following SX, U and Pu products were converted into solid oxides suitable for interim storage before manufacture into new fuels. The conversion processes used were a thermal denitration (TDN) process for Uranium Finishing and an oxalate precipitation and calcination process for Plutonium Finishing. The Thorp plant was integrated with the other supporting operations such as CSNF receipt and storage, plutonium product storage, waste management plants and MOx fuel fabrication (Phillips and Milliken, 2000). The French La Hague reprocessing plant is similar in design although technetium (a problematic fission product) is back-washed out of the (U, Pu) solvent product prior to the U/Pu separation stage to avoid the complications described in "Technetium" section (Lecomte, 2008).

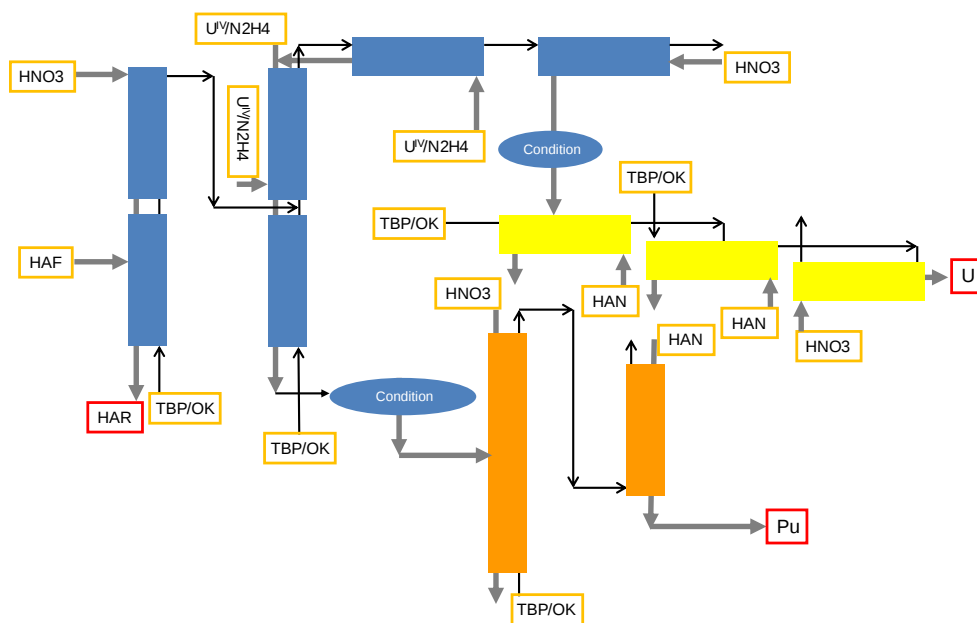


Fig. 8 Thorp solvent extraction flowsheet for commercial scale oxide fuel reprocessing (Sellafield, United Kingdom). Reproduced from ref. Baron P, Cornet SM, Collins ED et al (2019) A review of separation processes proposed for advanced fuel cycles based on technology readiness level assessments. *Progress in Nuclear Energy* 117: 103091.

Future perspectives

Despite the proven utility and established operational experience of the PUREX process, which is underpinned by a large body of knowledge and data, there is still room for improvements in its application for future reprocessing in closed nuclear fuel cycles. Indeed, the development of advanced or modified PUREX processes must be a major component of any program considering aqueous based technology for fuel processing. How radical any changes to the PUREX process can be depends on timescales to deployment. For near term deployment PUREX flowsheets based on conventional designs (such as those used at Thorp or La Hague plants) will have to be considered, possibly with evolutionary modifications such as those proposed in the French COEX™ process which produces a mixed U-Pu product to improve the anti-proliferation characteristics of the plutonium product (Senentz et al., 2009; Herbst et al., 2011; Poinssot et al., 2012).

Longer term R&D programs are generally more ambitious aiming for significant adaptations of the PUREX process that substantially reduce capital costs and address other perceived shortcomings of the conventional PUREX process for applications in future nuclear fuel cycles (Baron, 2010; Cashmore and Koppelman, 2015; Glatz et al., 2015; Guoan and Taihong, 2015; Minato, 2015; Todd, 2015; Baron et al., 2019). Hence, the actinides U, Np and Pu need to be fully controlled, ideally within a single PUREX solvent extraction cycle that is flexible to a wide range of feeds—higher burnup fuels and MOx fuels, for instance. A future process must probably meet similar product specifications on the U stream as currently but display increased barriers to proliferation on Pu streams, such as dilution with U and/or increased radiation fields (lower DFs). There are advantages if this can be achieved within short residence time annular centrifugal contactors, using kinetically fast “CHON” reagents only, producing minimized volumes of easily treated product and effluent streams. (The “CHON Principle” indicates that the reagent contains carbon, hydrogen, oxygen and nitrogen only and so can be completely decomposed without adding to the waste volumes). Greater use of modern technologies for on-line process monitoring will be needed to enable process control and accountability in real time. Waste arisings must be minimized at source and hazardous process reagents such as hydrazine replaced, if possible, with an increased robustness towards process upsets.

Hence, it is seen that recent progress internationally includes, inter alia, the development of single cycle PUREX flowsheets (Birkett et al., 2005; Lumetta et al., 2019); replacement of U(IV)/hydrazine for reductive stripping of Pu with a single complexing agent (Birkett et al., 2005; Carrott et al., 2007); development of real time process monitoring and control through on-line spectroscopic methods (Asmussen et al., 2019; Lumetta et al., 2019); use of TBP analogues that reduce risks of third phase formation (Sreenivasulu et al., 2016; Chandrasekar et al., 2017); improved Np control (Dinh et al., 2008; Taylor et al., 2013), etc. Elsewhere, the replacement of TBP itself by a CHON extractant, e.g., an amide (Manchanda and Pathak, 2004; McCann et al., 2018) or use of crystallization for the initial co-decontamination step (Washiya et al., 2009; Takeuchi et al., 2016) offer even more fundamental changes to the separation process. Lastly, much work has been undertaken in recent decades to extract the trivalent minor actinides (americium and/or curium) from the PUREX HA raffinate to reduce the radiotoxicity and heat loading of the high level waste that is ultimately sent to the geological repository (Baetslé et al., 2004; Grenèche et al., 2008; Baron, 2010; Modolo et al., 2015; Moyer et al., 2015; Collins et al., 2018). This requires the development of completely new separation processes (trivalent actinides are inextractable into TBP). It is

indicative of the success of the PUREX process that most of these are based on solvent extraction from nitric acid, e.g., see (Modolo et al., 2015; Moyer et al., 2015; Sharrad and Whittaker, 2015; Collins et al., 2018).

Conclusions

The PUREX process has proven to be a very successful process for the recovery of U and Pu from CSNF and has been operated at commercial scales in a number of countries. The process is based on the solvent extraction of U and Pu from an aqueous nitric acid phase into an organic phase of tri-butyl phosphate in a paraffinic diluent. There is a very good chemical and engineering basis to the PUREX process generated from over 70 years of research but options still exist to update and improve the process for “advanced” reprocessing applications in future closed nuclear fuel cycles. Specifically, the challenges in controlling neptunium and technetium are highlighted due to their complex process chemistry.

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- <https://www.iaea.org/>—International Atomic Energy Agency.
- <https://nucleus.iaea.org/sites/connect/SFMpublic/Pages/default.aspx>—Spent Fuel Management Network.
- <http://www.oecd-neo.org/>—Organization for Economic Co-operation and Development Nuclear Energy Agency.

The Chemical Basis for Separating Recycling Materials by Pyro-Processes

Stéphane Bourg, Commissariat à l'énergie atomique et aux énergies alternatives, Institut des sciences et technologies pour une économie circulaire des énergies bas carbone, CEA Marcoule, Bagnols sur Cèze, France

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Introduction

Pyrochemical processes have been studied for more than 70 years for recovering and purifying actinides, and in particular plutonium, and later, for recovering all actinides from spent nuclear fuel. Although they are based on a quite simple chemical concept, their implementation and their monitoring reveals to be quite complex as this chapter will demonstrate.

The actinide chemistry in molten salts was first studied in the 1950s for developing the Aircraft Reactor Experiment (ARE) (Bettis et al., 1957), and then in the 1960s for the Molten Salt Reactor Experiment (MSRE) (Robertson, 1965). In both experiments, the molten salt was a molten fluoride salt containing uranium, plutonium and/or thorium. The next step was the development of the Molten Salt Breeder Reactor (Robertson, 1971), with online pyrochemical processing. However, these developments were stopped by 1975 to focus on the development of light water reactors with solid fuels. In the 1980s, there was a renewal interest of pyrochemical processes for the processing of fast reactor metallic fuel in the frame of the Integral Fast Reactor concept (IFR) (Ackerman, 1991). In this case, the molten salt was a chloride salt (LiCl-KCl eutectic). The process was implemented up to the pilot scale for the recovery of uranium, and only to the lab scale for the recovery of the transuranic elements. Again, the developments were stopped in the mid-1990s.

There was a renewal of such studies at the end of the 1990s, with Japan, South Korea, Russia and France, with different perspectives and applications on metallic or oxide fuel, but also a renewal of the studies on molten salt reactor concepts, which remain today the main field for molten salt chemistry developments.

After a description of the chemical basis in molten salt, the main process options will be presented and illustrated by the key demonstration projects.

Finally, the challenges still to be addressed will conclude this article.

Chemical basis for pyroprocesses

The molten salts

Molten salts are ionic liquids composed of anions and cations. The most common anions are the halides (chloride, fluoride), but could be nitrate or carbonate. The cations are alkaline or alkaline earth elements. The most studied molten salt is the LiCl-KCl eutectic (59.2/40.8) with a melting point of 353 °C.

They are very good heat conductors (KCl at 800 °C is 22 times more conductor than water at 20 °C) and excellent ionic conductors. Depending on the choice of the anions and cations, the structure of the salt can range from a fully dissociated mixture of anions and cations to an organized complex liquid with ion pairs or molecules. A mixture of two different salts can present specific properties and have a strong non-ideal behavior, opening the field for more process applications.

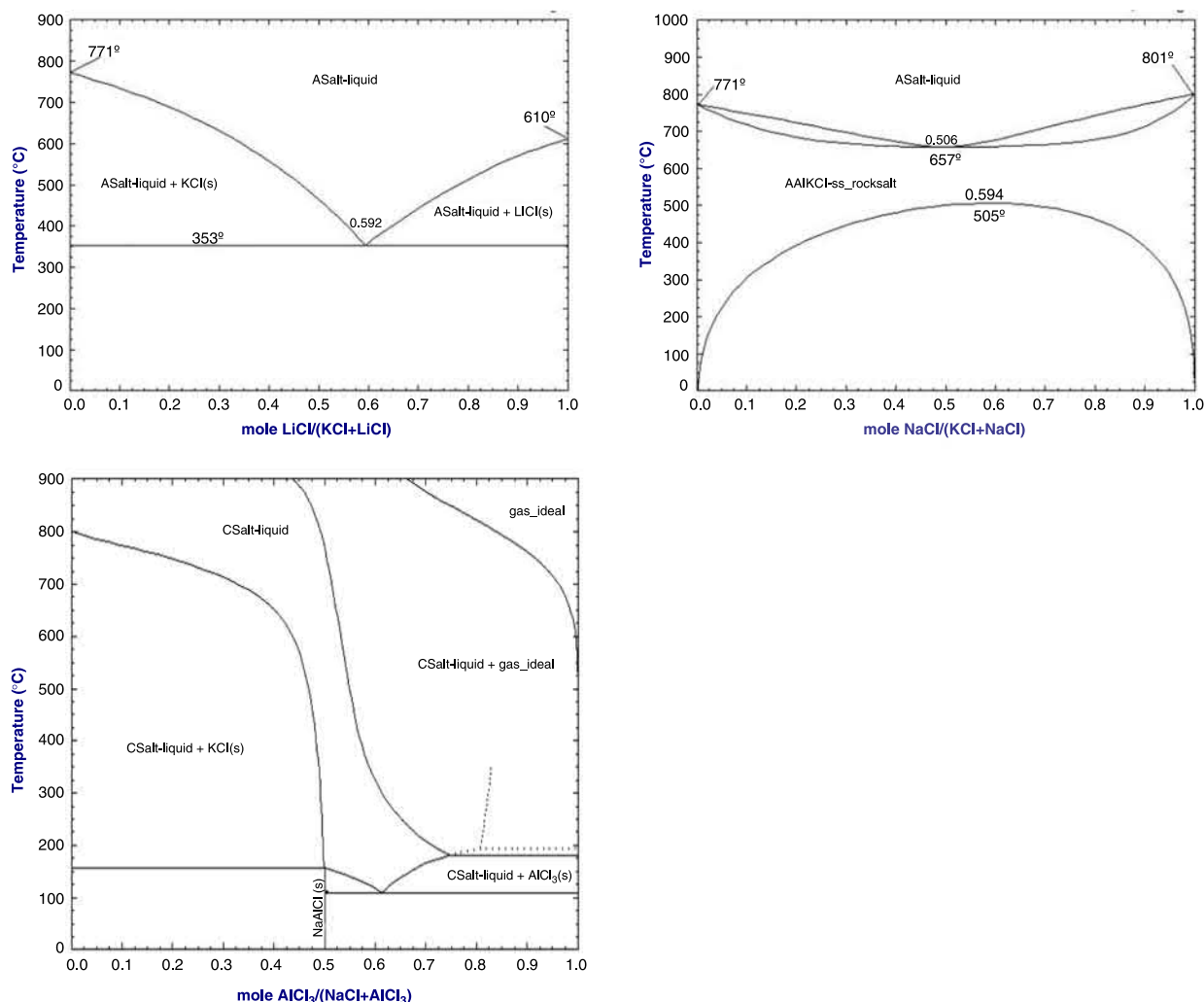


Fig. 1 Phase diagram of LiCl-KCl, NaCl-KCl and NaCl-AlCl₃ salt. Data from FTSalt—FACT Salt Database, http://www.crct.polymtl.ca/fact/documentation/FTsalt/FTsalt_Figs.htm

Generally, alkaline halide mixtures exhibit low deviation to ideality, whereas when one of the cation is not an alkaline (or alkaline earth), such as aluminum, the mixture can present strong deviation. In such a case, the mixture will behave like a Lewis acid, the acidity depending on the ratio of the two salts (**Fig. 1**).

The solubility of actinides and lanthanides can be very high in some salt, when a binary or ternary eutectic composition exists (up to 40 mol%) (**Figs. 2 and 3**).

Thermodynamics

The chemistry in molten salt is driven by the thermodynamics.

For a **reaction (i)**, the Gibbs energy (or free enthalpy of formation) ΔG is defined by Eq. (1), where A,B,C,D are four chemical species, n_i the number of mole of I, ΔG° the free energy of formation of the pure compounds, a_i the activity of I in its phase. The activity is linked to the molar fraction X_i by Eq. (2), where γ_i is the activity coefficient of i in the studied media (when a species is alone in its phase, the activity coefficient is 1). R is the gas constant ($8314 \text{ J mol}^{-1} \text{ K}^{-1}$) and T is the temperature in Kelvin.

$$n_A A + n_B B = n_C C + n_D D \quad (i)$$

$$\Delta G = \Delta G^\circ + RT \ln \frac{a_C^{n_C} a_D^{n_D}}{a_A^{n_A} a_B^{n_B}} \quad (1)$$

$$a_i = \gamma_i X_i \quad (2)$$

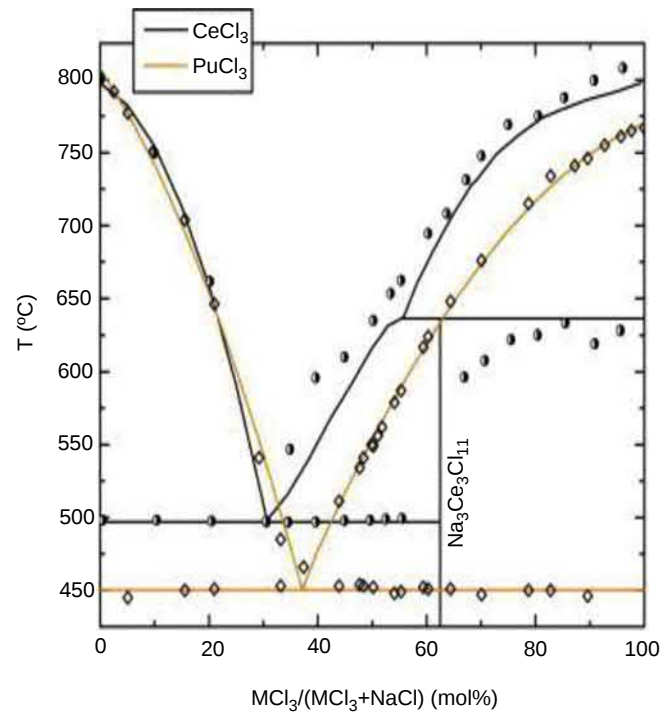


Fig. 2 Binary phase diagrams of the NaCl-CeCl₃ (—) and NaCl-PuCl₃ (—) systems. From Sooby AS et al. (2015) Measurements of the liquidus surface and solidus transitions of the NaCl-UCl₃ and NaCl-UCl₃-CeCl₃ phase diagrams. *Journal of Nuclear Materials* **466**: 280–285.

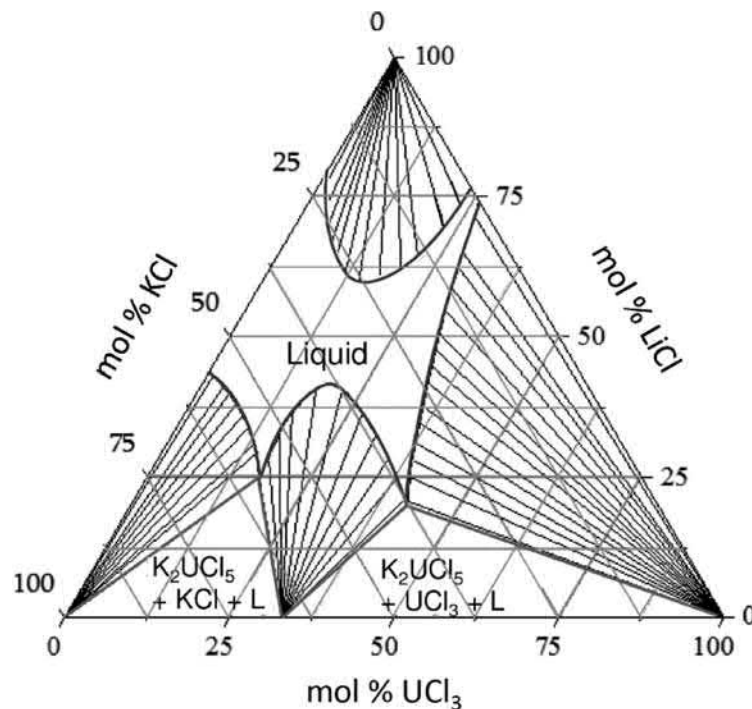


Fig. 3 Isothermal projection of the LiCl-KCl-UCl₃ system at 773 K. From Hames A et al. (2018) Phase equilibria studies of the LiCl-KCl-UCl₃ System. *Nuclear Technology* **203**: 1–10. doi: 10.1080/00295450.2018.1448673.

The redox potential E and the Gibbs energy ΔG are linked by Eq. (3), where n is the number of exchanges electrons and F the Faraday number ($F = 96.487 \text{ } ^\circ\text{C}$).

$$\Delta G = -nFE \text{ or } E = -\Delta G/nF \quad (3)$$

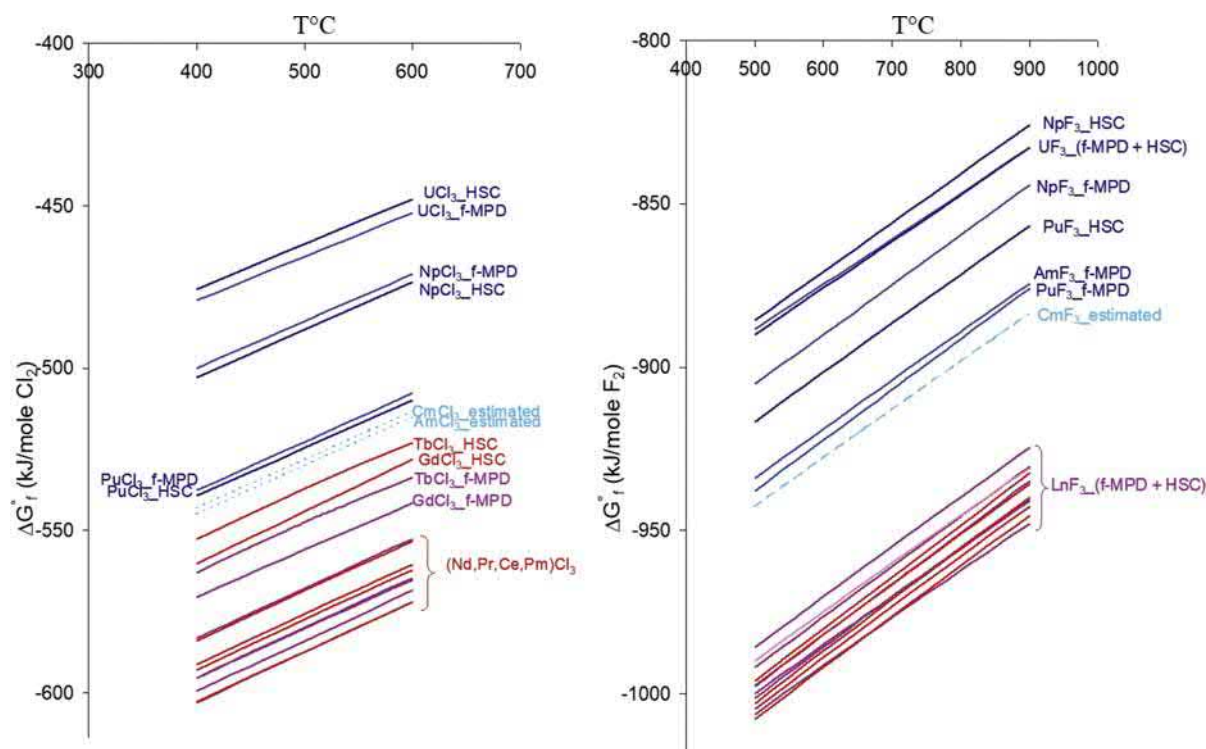


Fig. 4 Free energy of formation of pure compounds, built from HSC and f-MPD databases (own work).

All the chemistry, or electrochemistry, in molten salts is driven by these few equations. The main difficulty is the determination of the activity coefficients, which depend on the media and can only be experimentally determined. Different databases list the free energy of formation of pure compounds (such as Outokumpu HSC), in particular actinide and lanthanide halides (Fig. 4).

These activity coefficients must be determined for the species in molten salts, but also for the reduced species recovered in a metallic solvent or deposited on a cathode made of a metal able to dissolve (alloy formation) or form polymetallic compounds with these species (Fig. 5).

A molten salt has a stability chemical domain (also called electrochemical window) defined by the reduction of the less stable cation on one side and by the oxidation of the anion on the other side. Within this domain, it is possible to perform redox chemistry or electrochemistry on, or between, any dissolved species having a Gibbs energy (or free enthalpy of formation), ΔG , or a redox potential, E , within this window. In the case of LiCl-KCl, the less stable cation is Li^+ (Fig. 6).

Therefore, in molten chloride salt such as the LiCl-KCl eutectic, this can be summarized by Fig. 7.

Similar determination was made for different metals on or in which the species can be reduced (Fig. 8). When the deposited metal can react with the metallic substrate (Cd, Bi, Al), the reduction is easier (depolarization phenomenon) than with an inert substrate (such as tungsten) but the gap between the reduction potential of two species is lower, meaning that the selectivity will be lower too. The highest selectivity between actinides and lanthanides is with tungsten, then aluminum. With Cd and Bi, the selectivity is poor.

It is also important to monitor and control the oxide content in the salts, since it can influence the speciation of the actinides and lanthanides. It allows a specific chemistry based on the selective formation of oxide species. Such an approach was used by the RIAR in Russia to develop the Dimitrovgrad Dry Process. Pourbaix diagrams (or predominance diagrams) can help to determine the potential and the oxide concentration at which either a selective precipitation or a redox reaction is possible (Figs. 9 and 10).

Based on this information, pyrochemical separation processes can theoretically be developed and tentatively industrially deployed.

From the chemical reactions to a potential separation industrial scheme

Based on the previous statements, three kinds of separation processes can be developed. Two are directly based on the redox chemistry by mastering the potential either by electrochemistry or by a chemical reagent. The third one is based on oxide or oxychloride precipitation by controlling the concentration of oxide ions in the salts.

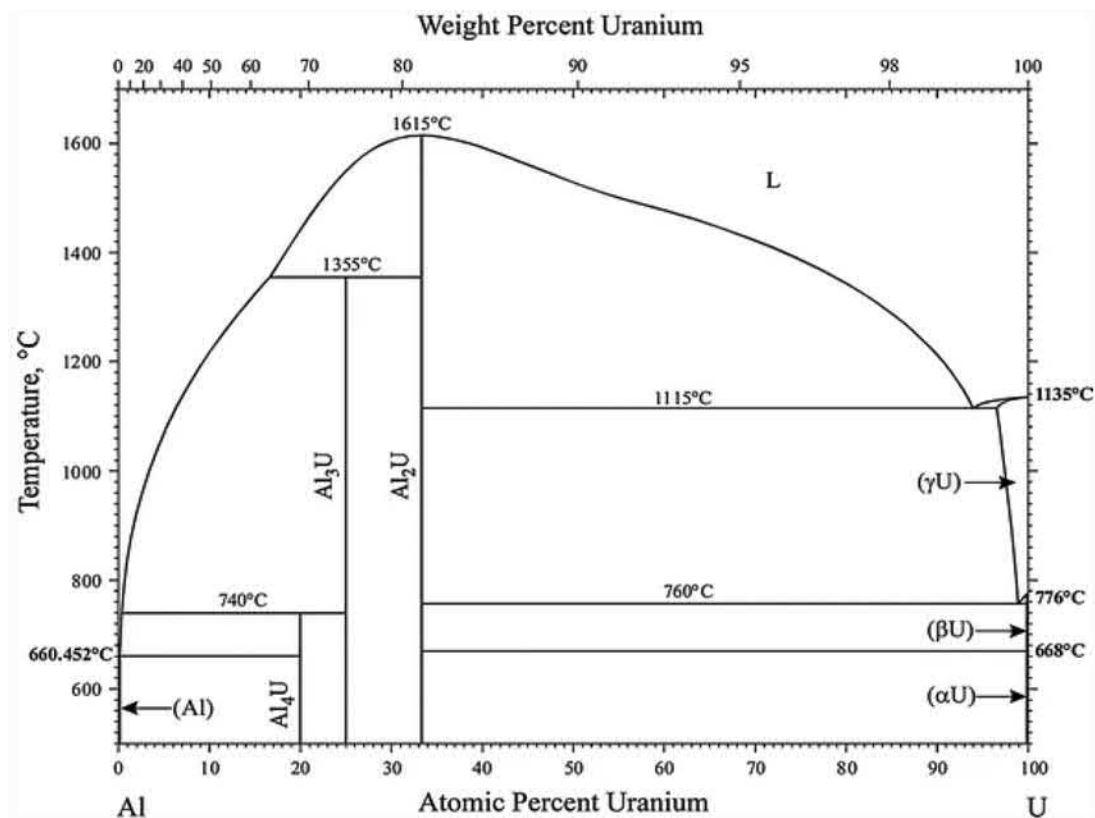


Fig. 5 Al-U phase Diagram. From Okamoto H (2012) Al-U (aluminum-uranium). *Journal of Phase Equilibria and Diffusion* 33: 489–490.

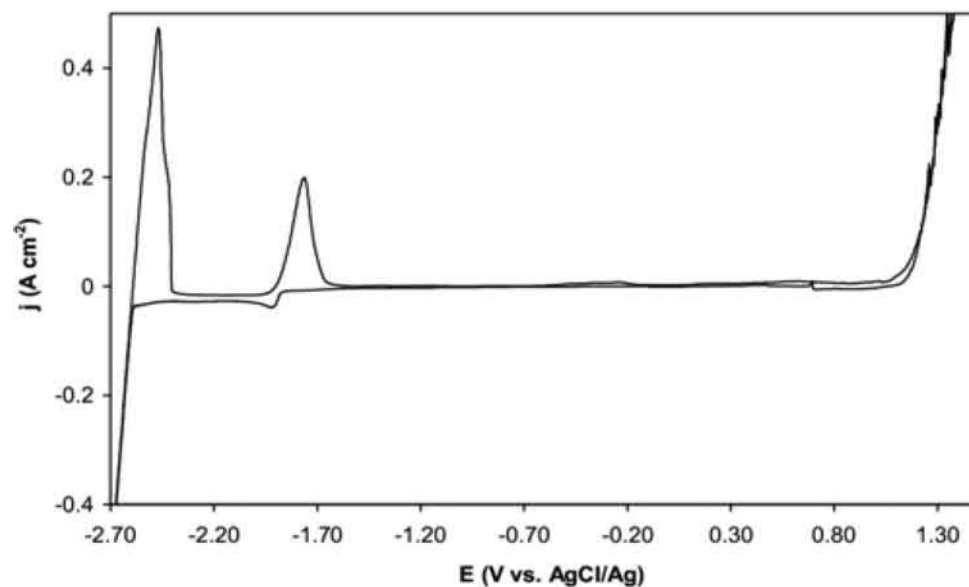


Fig. 6 Cyclic voltammogram of the LiCl-KCl-Pu(III) solution on a tungsten electrode at 450 °C, v scan = 0.1 V/s, $[Pu(III)] = 2.4 \times 10^{-2}$ mol/kg. From Caravaca C et al. (2008) Determination of the E-pO₂-stability diagram of plutonium in the molten LiCl-KCl eutectic at 450°C. *Journal of Nuclear Materials* 377(2): 340–347, doi: 10.1016/j.jnucmat.2008.01.031.

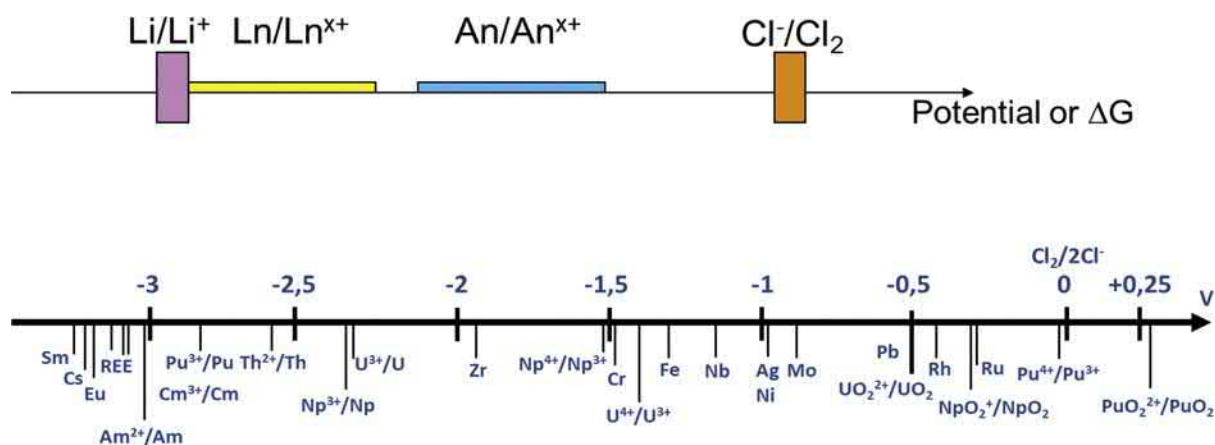


Fig. 7 Potentials row for actinides and fission products in molten 3LiCl₂KCl under 773 K. From Bychkov A (2009) *Current Status of Development in Dry Pyroelectrochemical Technology of Spent Nuclear Fuel Reprocessing*, Joint ICTP/IAEA School on Physics and Technology of Fast Reactors Systems, 9–20 November. <http://indico.ictp.it/event/a08209/session/55/contribution/32/material/0/0.pdf>.

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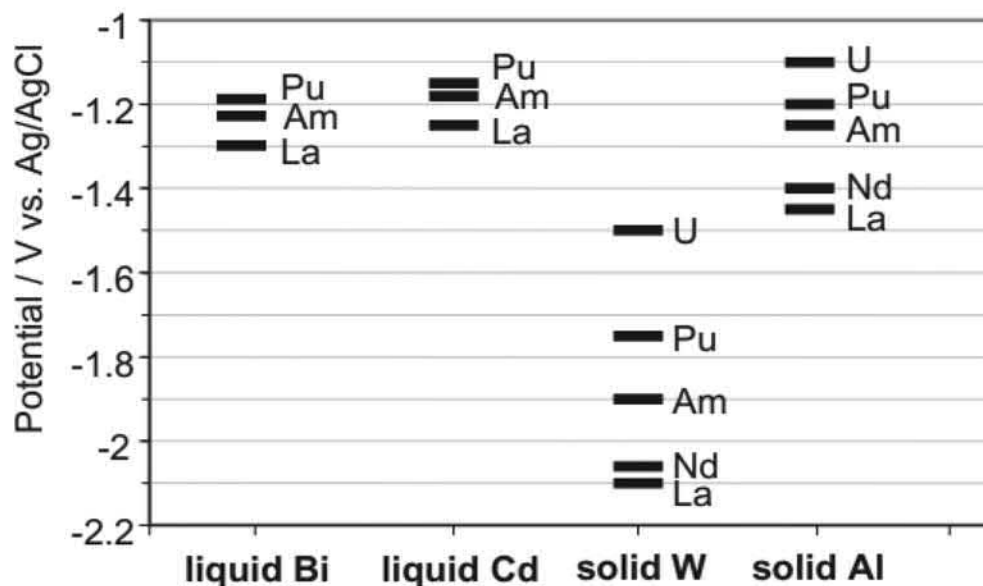


Fig. 8 Reduction potentials of some actinides and lanthanides on different cathodic materials. From Malmbeck R et al. (2011) Advanced fuel cycle options. *Energy Procedia* 7: 93–102, ISSN 1876-6102, doi: 10.1016/j.egypro.2011.06.013.

Spent nuclear fuel dissolution is molten salt

For some separation processes, the spent fuel needs to be first dissolved in the molten salt. This is the case the liquid-liquid extraction process, for a precipitation process or for an electroreduction (or electrowinning) process such as the DIMITROVGRAD Dry Process. This dissolution can be quite challenging.

Oxide fuels can be dissolved by a carbochlorination process following [reaction \(ii\)](#) (see for example [Dobbins and Burnet, 1988](#)).



Excess of chlorine gas must be used to avoid the formation of soluble oxychloride species.

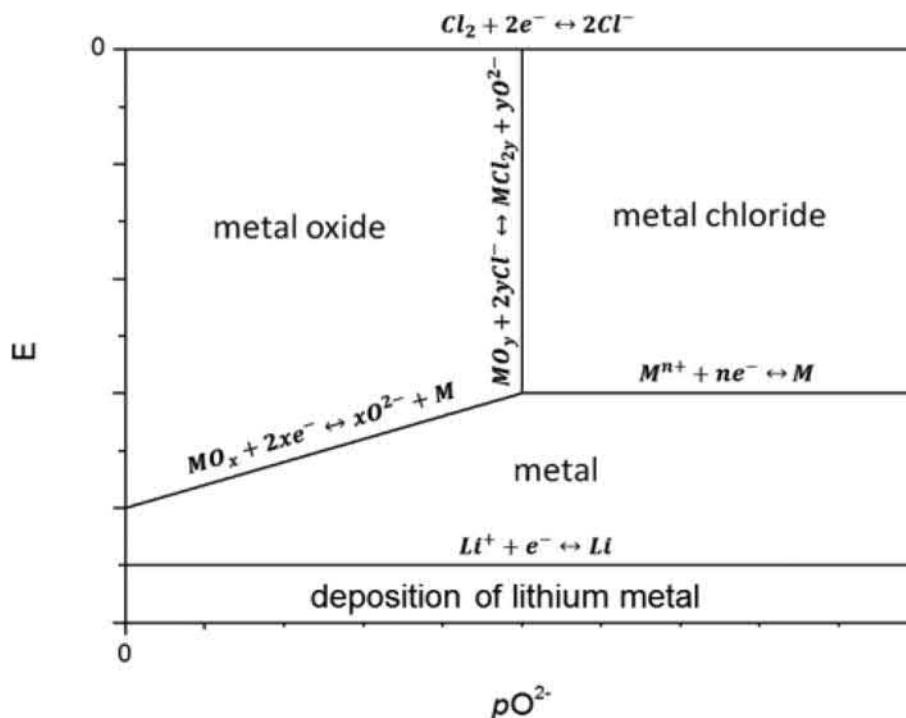


Fig. 9 Example of the three main regions of stability in a predominance diagram (based on a LiCl-KCl salt eutectic) and showing the nature of reactions leading to horizontal, vertical and diagonal lines. From Abdulaziz R et al. (2016) Predominance diagrams of spent nuclear fuel materials in LiCl-KCl and NaCl-KCl molten salt eutectics. *International Journal of Electrochemical Science* **11**: 10417–10435, doi: 10.20964/2016.12.27.

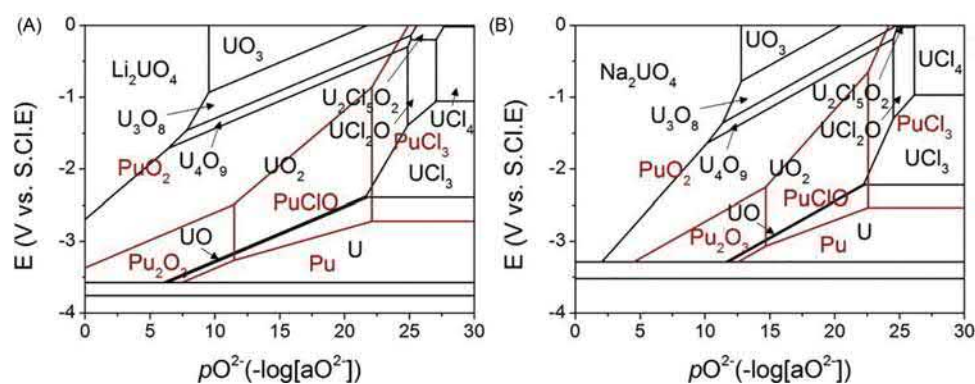
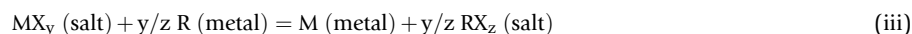


Fig. 10 Predominance diagrams of U and Pu species. (A) In LiCl-KCl at 500 °C, (B) in NaCl-KCl at 750 °C. From Abdulaziz R et al. (2016) Predominance diagrams of spent nuclear fuel materials in LiCl-KCl and NaCl-KCl molten salt eutectics. *International Journal of Electrochemical Science* **11**: 10417–10435, doi: 10.20964/2016.12.27.

Chemical processes

If a metal for which the ΔG of formation of its halide is between the ones of lanthanides and actinides is contacted with the salt, it will selectively reduce the actinides to the metallic form, whereas the lanthanides will remain in the salt under cationic form Conocar et al. (2005a). For instance, lithium metal, which is more reductive than both lanthanides and actinides, will reduce both actinides and actinides to the metallic form.



R is the reductive metal, dissolved in the metallic solvent (the reductive metal can be the solvent)

$$K = \frac{a_M a_{RX_z}^{y/z}}{a_{MX_y} a_R^{y/z}} \quad (4)$$

$$\log D_M = \frac{Y}{Z} \log D_R + \log K + \log \left(\frac{\gamma_{MX_Y} \gamma_R^{Y/Z}}{\gamma_M \gamma_{RX_Z}^{Y/Z}} \right) \quad (5)$$

$$\left\{ D_M = \frac{X_M}{X_{MX_Y}} \right. \quad (6)$$

In chemical reduction processes, the challenge is the identification of the reductive metal that will selectively reduce the actinides and the identification of the metallic solvent. However, as shown in Eqs. (4) and (5), the activity coefficients of species both in the salt and in the metal are key parameters and the selection of the reductive metal cannot be based on the free enthalpy of formation. If the ratios of the activity coefficients of two species are almost always close to 1 in molten salts, they can vary on a broader range of several orders of magnitude in the metallic phase.

When considering the equilibrium between two species (for instance actinides vs. lanthanides) shown in reaction (iv), the selectivity is expressed by the separation factor SF_{ij} . SF_{ij} is defined as the ratio of the distribution coefficients D_{Mi} and D_{Mj} (Eq. 8) (Conocar et al., 2005b).



With

$$K_{ij} = \frac{a_{MjX_3} a_{Mi}}{a_{MiX_3} a_{Mj}} \quad (7)$$

$$SF_{ij} = D_{Mi}/D_{Mj} \quad (8)$$

By combining Eq. (5), Eq. (7) and Eq. (8), it gives Eq. (9)

$$SF_{ij} = K_{ij} \frac{\gamma_{MiX_3} \gamma_{Mj}}{\gamma_{MjX_3} \gamma_{Mi}} \quad (9)$$

Where K_{ij} depends only on the Gibbs energy of formation of M_iX_3 and M_jX_3 , $\frac{\gamma_{MiX_3}}{\gamma_{MjX_3}}$ represents the solvation differential in the salt and $\frac{\gamma_{Mj}}{\gamma_{Mi}}$ represents the solvation differential in the liquid metal.

For M_i being an actinide and M_j a lanthanide, the solvation differential in the salt is close to 1.

The situation is very different for the solvation differential in the liquid metal. This parameter has to be minimized to increase the separation factor.

An illustration of the solvation differential is given for different metals as a function of the temperature (M_i = plutonium and M_j = Cerium) on Figs. 11 and 12. It shows that aluminum is the best metallic solvent.

Indeed, the most efficient process was proposed in molten fluoride salt $LiF-AlF_3$, with the aluminum as both the liquid metal phase and the reductive metal. The use of AlF_3 bring another parameter into the system: the fluoroacidity concept. For instance, in the LiF/AlF_3 mixture, LiF is totally dissociated as Li^+ and F^- whereas AlF_3 is a F^- acceptor, leading to the formation of AlF_3^{x-} depending on its molar fraction. This fluoroacidity also affects the activity coefficients in the salt (Fig. 13).

It has to be noticed that the spent fuel has to be first dissolved in the salt before the extraction process.

Electrochemical processes

The same results can be obtained by electrochemistry, by applying a cathodic potential between the redox potentials of lanthanides and actinides. The reduced metals are deposited on the cathodic material (Laidler et al., 1997). This second option is more flexible since any potential can be applied precisely, but it will require more energy than the chemical method.

The selectivity will depend on the gap between the reduction potential of the two species to be separated. It can be estimated by using the Nernst equations (Eq. 10).

$$E = E^\circ + 2,3RT/nF \log(ox/red) \quad (10)$$

Let's consider the selective recovery of an actinide (An) from neodymium (Nd). For each element, Eq. (10) can be written as:

$$E_{Nd(III)/Nd(0)i=0} = E_{Nd(III)/Nd(0)}^0 + \frac{2,3 \times RT}{3F} \log[Nd(III)] \quad (11)$$

$$E_{An(III)/An(0)i=0} = E_{An(III)/An(0)}^0 + \frac{2,3 \times RT}{3F} \log[An(III)] \quad (12)$$

Then, the gap between the two reduction potential is given by Eq. (13)

$$\Delta\Delta E = E_{Nd(III)/Nd(0)}^0 - E_{An(III)/An(0)}^0 = -\frac{2,3 \times RT}{3F} \log \frac{[Nd(III)]}{[An(III)]} \quad (13)$$

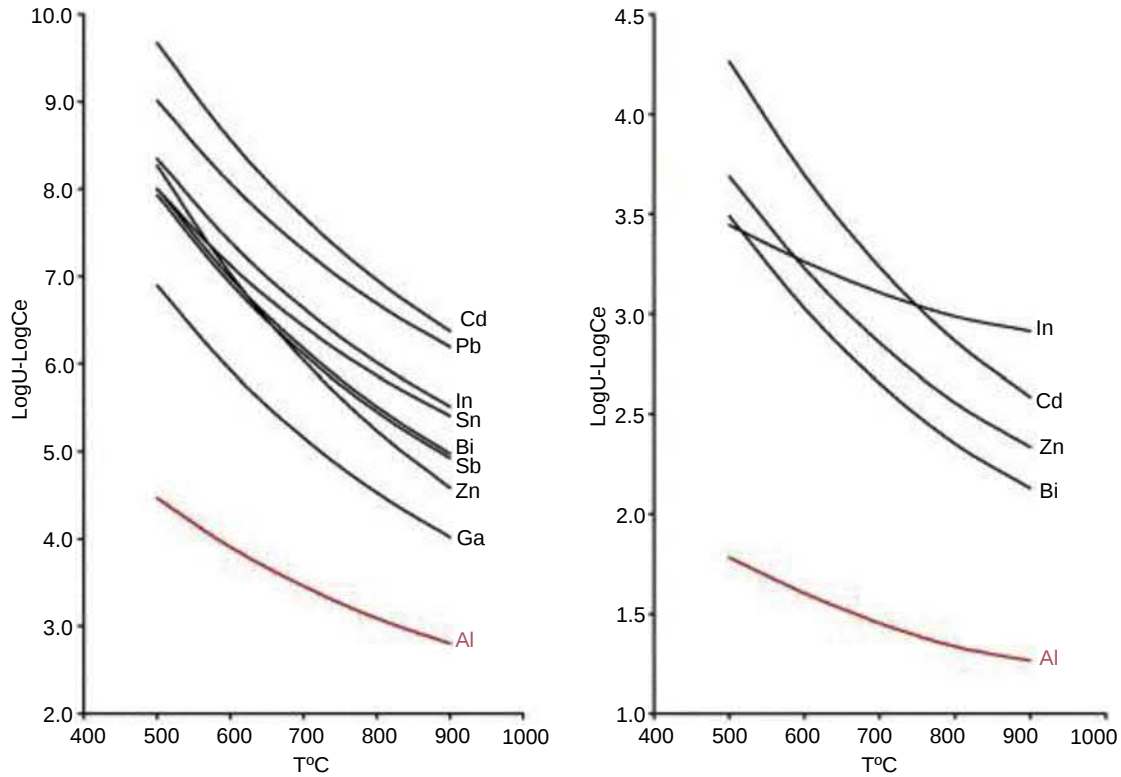


Fig. 11 Variation of $\log \gamma_U - \log \gamma_{Ce}$ and $\log \gamma_{Pu} - \log \gamma_{Ce}$ versus temperature for various metals. From Conocar O et al. (2005b) Extraction behavior of actinides and lanthanides in a molten fluoride/liquid aluminum system. *Journal of Nuclear Materials* **344** (1–3): 136–141.

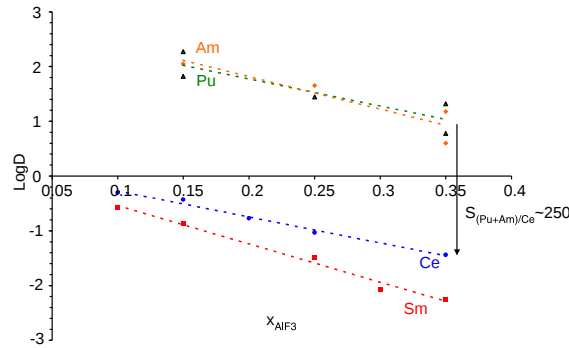


Fig. 12 Pu, Am, Ce and Sm partitioning in the $\text{MIF-AlF}_3/\text{Al-Cu}$ system: effect of the salt composition. From Conocar O et al. (2005b) Extraction behavior of actinides and lanthanides in a molten fluoride/liquid aluminum system. *Journal of Nuclear Materials* **344** (1–3): 136–141.

Where $\log[\text{Nd(III)}/\text{An(III)}]$ is the logarithm of the separation factor of the two elements.

If we want to achieve a separation factor between An and Nd of 10,000, then

$$\Delta\Delta E = E_{\text{Nd(III)}/\text{Nd(0)}}^0 - E_{\text{An(III)}/\text{An(0)}}^0 = -4 \frac{2.3 \times RT}{3F} \quad (14)$$

It gives a minimum gap of 190 mV at 450 °C and a gap of 285 mV at 800 °C.

According to Fig. 7, this is achievable for selectively recovering U and Pu, but much more challenging for americium.

However, when the composition of the salt is complex, the metallic cathodic deposit can reduce other cations which are in the salt and then be re-oxidized and re-dissolved in the salt. This is what happens when plutonium is reduced whereas there is still uranium chloride in the salt. An alternative is to use a reactive cathodic material, which will form an alloy or a solid solution with the reduced metal. The most studied reactive cathode material is the cadmium, but aluminum could be a good alternative since thermodynamics demonstrates that the selectivity is higher on aluminum.

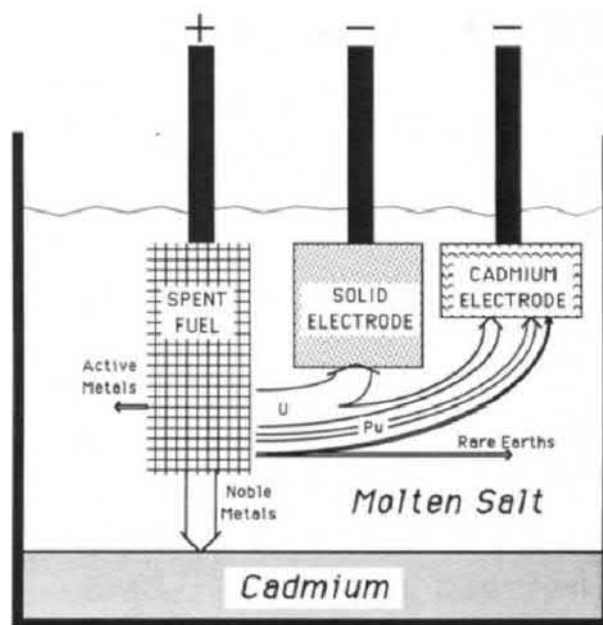


Fig. 13 Electrorefiner conceptual diagram. Ackerman JP (1991) Chemical basis for pyrochemical reprocessing of nuclear fuel. *Industrial and Engineering Chemistry Research* **30**(1), 141, doi: 10.1021/ie00049a022.

Two types of process can be developed: the electrowinning, or electrodeposition, where the spent fuel is first dissolved in the molten salt, and the electrorefining where the treatment is directly applied to the spent fuel and allows a metal to metal direct processing.

An electrorefining process is more efficient than an electrowinning process because the concentration of the species to be reduced remain higher in the salt, allowing a higher selectivity. However, the electrorefining alone does not allow the quantitative recovery of all the actinides. It is particularly well suited for the recovery of the uranium.

This explains why electro-pyrochemical processes in molten chlorides are particularly well suited for the processing of metallic fuels. The actinides, which are the elements to recover from the spent fuel, are the first selectively reduced.

Selective precipitation

Actinides can be removed from a molten by oxidizing them into solid oxide or oxychloride species (Lambertin et al., 2002, 2003; Caravaca et al., 2008). Such a precipitation can be obtained by adding an oxidizing reagent such as calcium carbonate or sodium carbonate (McNeese et al., 1995), oxygen sparging (Cho et al., 2006) or by wet gas sarging (Vigier et al., 2013). However if the concept is simple and good performances seem to be achievable based on Pourbaix diagram analysis, the selectivity observed in experiments is poor (Lambertin, 2001).

Such a process can be used for oxide recovery from molten salt in order to fabricate fresh fuel (Vigier et al., 2013).

Current experimental feedback and key demonstration projects

Based on the chemical properties described above, two kind of salts can be used (chlorides and fluorides) and two types of methods (chemical or electrochemical).

The first and most studied combination is the electrorefining in molten chloride for the processing of metallic spent nuclear fuel. Two examples will be developed: the reference process developed in the frame of the IFR project and the process developed in Europe at the JRC Karlsruhe.

The second option is the Dimitrovgrad Dry Process for fast MOX fuel reprocessing.

The third option is the chemical liquid-liquid reductive extraction process between a molten fluoride salt and liquid aluminum developed at the CEA.

Electrorefining process

In the 1980s, the IFR project was based on a processing of fast reactor metallic fuel after short cooling time, in a facility near the reactor. Pyroprocesses are deemed compact and suitable for small units. The process was based on two steps: first the recovery

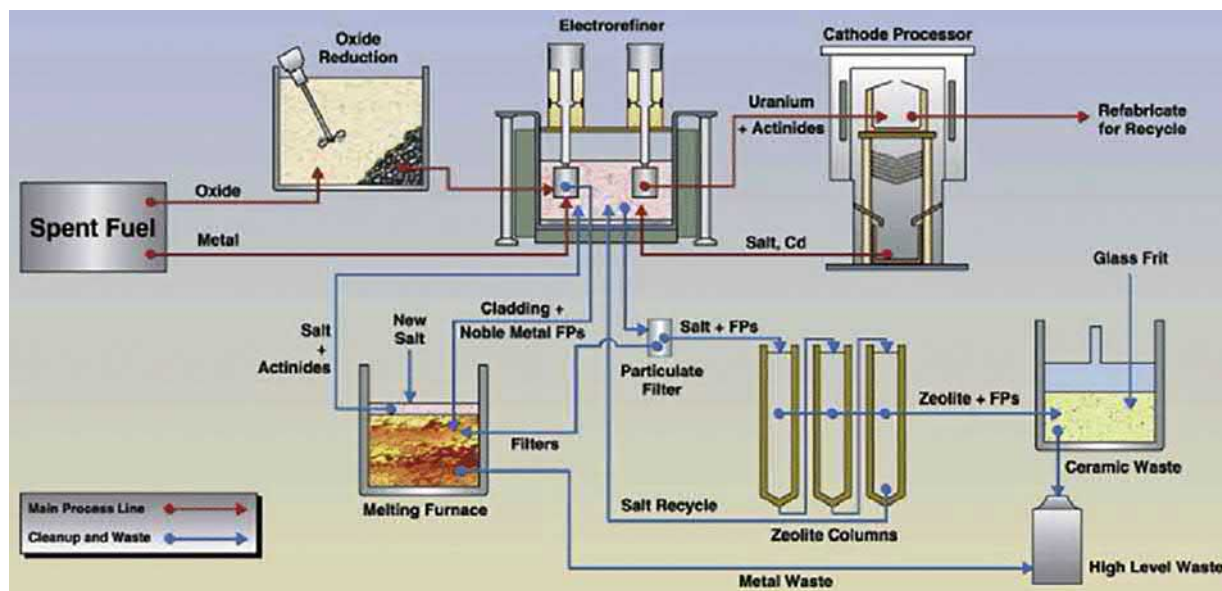


Fig. 14 Process scheme for spent fuel processing in molten salt by electrorefining. From Chang Y (2002) *Advanced Nuclear System for the 21st Century, Workshop on Nuclear Cycle System for the 21st Century, Karlsruhe, Germany, 6–8 May*.

of pure metallic uranium by direct electrorefining of the spent fuel, and second a recovery of the plutonium and the minor actinides by an electrolysis on liquid cadmium (Ackerman, 1991) (Fig. 13). An additional step at the head-end of the processed allowed the processing of LWR oxide fuel through an oxide reduction process prior the electrorefining (Fig. 14) (Chang, 2002).

The first step was extensively studied and developed up to the pilot scale at Idaho Falls National Laboratory. A dedicated electrorefiner was built (Ackerman, 1991) and upgraded up the fifth version, Mark V (Fig. 15). A few tons of EBR II blanket fuel were processed.

However, the recovery of plutonium and minor actinides on the liquid cadmium cathode was mainly developed up to the engineering-scale (a few kilograms) with many technical challenges in the handling of this liquid cathode and limited decontamination factors of the plutonium versus the lanthanides (Li et al., 2009, 2010; Vaden et al., 2008).

In the meantime, similar developments were carried out in Japan (Koyama, 1997; Iizuka et al., 1997; Kato et al., 2006). Decontamination factor of Pu (ratio of neodymium to plutonium concentrations, $[Nd]/[Pu]$, in the salt before extraction to that in the cadmium after extraction.) calculated from the reported experiments are about 25 for the best ones.

In South Korea, KAERI has also carried out technology developments for the scale-up of the pyroprocess proposed by INL. The PRIDE facility was designed to demonstrate engineering-scale pyroprocessing in an integrated manner (Fig. 16). The PRIDE facility included all the technological devices for testing the oxide reduction, the uranium electrorefining and the plutonium and minor actinide electrolysis on cadmium (Lee and Jang, 2017). However, KAERI can only work with uranium and the electrolysis was never tested with plutonium and minor actinides.

In these processes cadmium has been chosen for its high volatility which allows its distillation and therefore an easy recovery of the pure transuranics at the metallic state.

However, very few experiments were carried out on Pu-Am loaded cathodes.

The electrochemical reprocessing of oxide fuels has also been proposed, including a pretreatment in which the oxide fuel is reduced to the metallic state (Williamson and Willit, 2011) (Fig. 17).

Electrochemical European process using an aluminum cathode

Since the early 2000s, an electrorefining process allowing the recovery of both the uranium and the transuranics on a solid aluminum cathode has been studied at the JRC Karlsruhe (Serp et al., 2005). As reminded in the first part of this article, aluminum is the best reactive cathode material for the selective recovery of actinides.

However, such a process has only been demonstrated at the lab-scale and still requires engineering and upscaling developments. The redox potentials of U, Pu and Am on Al show that U could be recovered alone but the gap between Pu and Am potentials makes the recovery of pure plutonium very challenging.

This process has been tested on a METAPHIX fuel in the frame of the JRC/CRIEPI collaboration (Soucek et al., 2019).

- Efficient recovery of actinides from irradiated metallic fuel METAPHIX-2 was demonstrated by electrorefining process.
- High separation factors of actinides (An) over lanthanides (Ln) were achieved in electrolyte with high concentration of Ln.
- Grouped selectivity of the actinides recovery was proven for later and final stages of the electrorefining process.

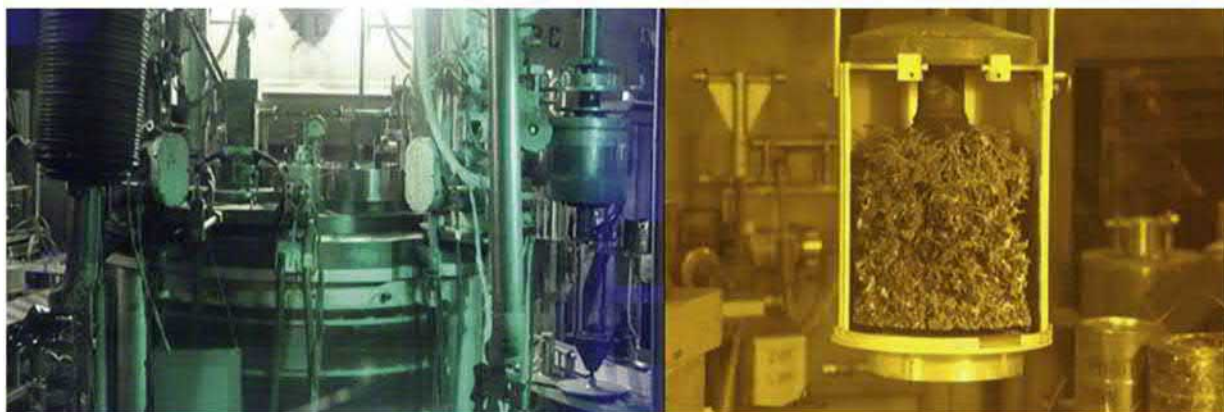
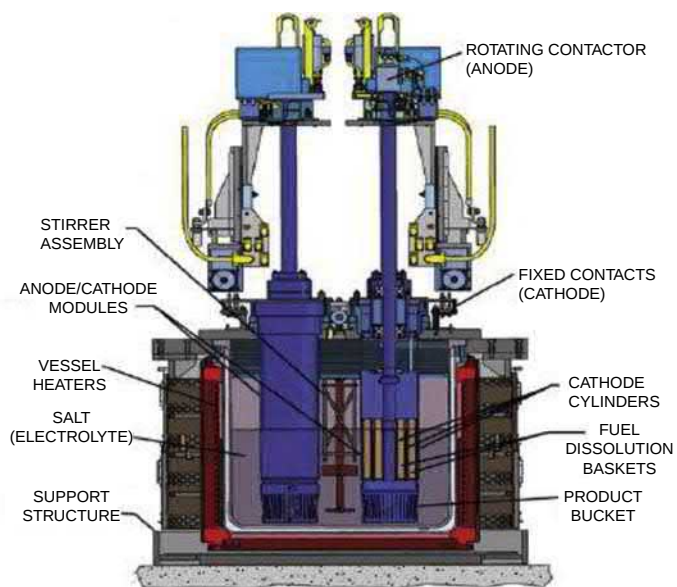


Fig. 15 Mark IV electrorefiner. From Westphal BR et al. (2009) Development of a ceramic-lined crucible for the separation of salt from uranium. *Metallurgical and Materials Transactions A* **40**(12): 2861–2866, doi: 10.1007/s11661-009-9957-3, <https://mfc.inl.gov/SitePages/Instruments/Fuel%20Conditioning%20Facility/Mark%20IV%20electrorefiner.aspx>

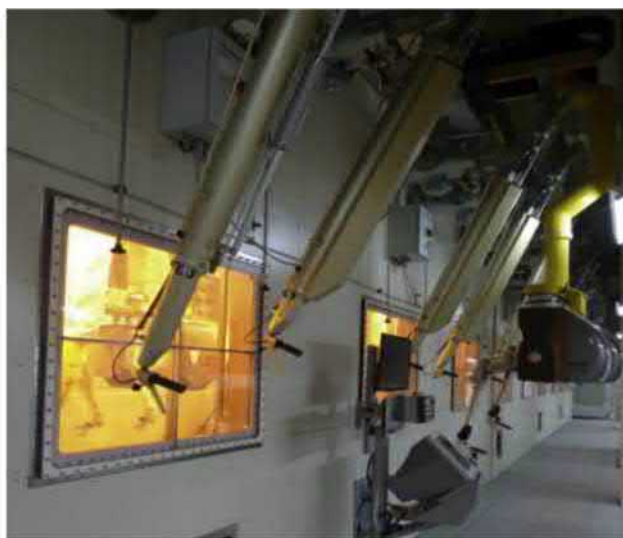


Fig. 16 Process cell for technological demonstration at KAERI. From NEA/NSC/R(2018)4 report, [http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NSC/R\(2018\)4&docLanguage=En](http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NSC/R(2018)4&docLanguage=En)

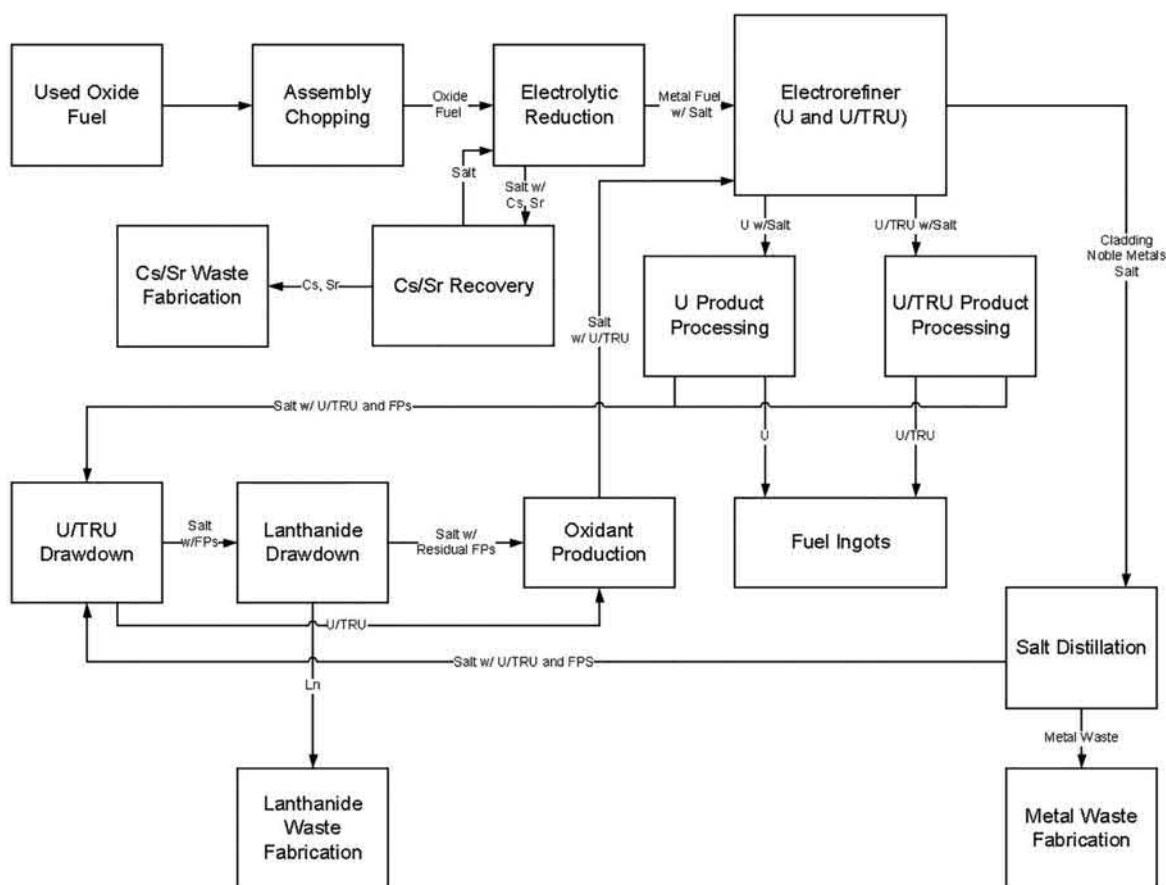


Fig. 17 Conceptual flowsheet for the treatment of used LWR fuel. From Williamson MA and Willit JL (2011) Pyroprocessing flowsheets for recycling used nuclear fuel. *Nuclear Engineering and Technology* 43(4).

- Aluminum cathode was proven suitable for recovery of all actinides, while only uranium was deposited on the inert cathodes.
- Zirconium co-dissolution from the fuel had no effect on the process efficiency and on the structure of the deposits.

A weak point of such a process is the recovery of the actinides from the aluminum matrix. Two options have been studied: the selective chlorination of Al as AlCl_3 under controlled conditions to avoid the formation the volatile actinide chloride species (Cassayre et al., 2011; Soucek et al., 2014) or the chemical back extraction of actinides as chloride salts from the metallic phase to a salt phase containing AlCl_3 as oxidative species (Mendes et al., 2014).

Dimitrovgrad dry process

In the 1970–2000s, the Russian Institute of Atomic Reactors, RIAR, developed a pyroprocess allowing the recycling of MOX fuel from fast reactor and the fabrication of a mixed uranium oxide/plutonium oxide aggregates to be directly used in VIPAC fuel (Bychkov, 2009) (Fig. 18). With such a process, it is also possible to recover separately plutonium oxide and uranium oxide thanks to the different stability domains of the two oxides in the Pourbaix diagram.

The process was only demonstrated at the kilogram scale on spent MOX fuel (40 kg). So far, the DDP was mainly used to manufacture a few tons of MOX VIPAC fuel from a mix of pure civil and weapon grade plutonium and depleted uranium (7.2 t).

This process is very interesting because it combines all the potentialities of the chemistry in molten salt. First, the spent fuel is dissolved in molten salt by carbo-chlorination with uranium as UO_2^{2+} species and plutonium as Pu^{3+} . Then most of the uranyl cations are reduced as uranium oxide at the cathode under argon/chlorine atmosphere.

In the next step, it is possible to precipitate the plutonium as plutonium oxide by introducing oxygen gas in the Ar/Cl_2 atmosphere (DDP MOX $\rightarrow$ PuO_2 flow sheet). Then the remaining uranyl cations can be reduced again as oxide at the cathode.

Instead of precipitating the plutonium, by adjusting both the oxygen potential (by introducing oxygen in the atmosphere) and the electric potential, it is possible to deposit a mix of uranium oxide and plutonium oxide at the cathode (Fig. 19).

At the last step, the minor actinides and the rare earth are precipitated by sodium phosphate to purify the salt.

Good decontamination factors were achieved on different spent fuel samples (Fig. 20).

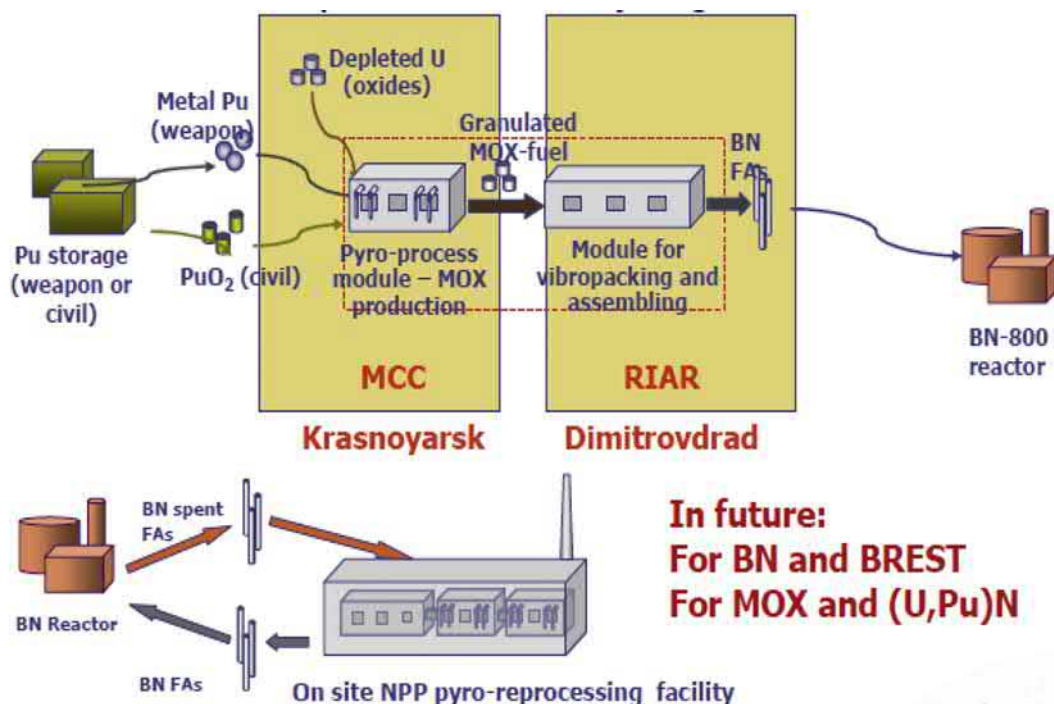


Fig. 18 Implementation Pyroprocess for BN-800 Fuel Cycle. Combination of pyroprocess and vibropacking technology is the basis for BN-type MOX fuel production and recycling in different scenarios. From Bychkov A (2009) *Current Status of Development in Dry Pyroelectrochemical Technology of Spent Nuclear Fuel Reprocessing*, Joint ICTP/IAEA School on Physics and Technology of Fast Reactors Systems, 9–20 November. <http://indico.ictp.it/event/a08209/session/55/contribution/32/material/0/0.pdf>.

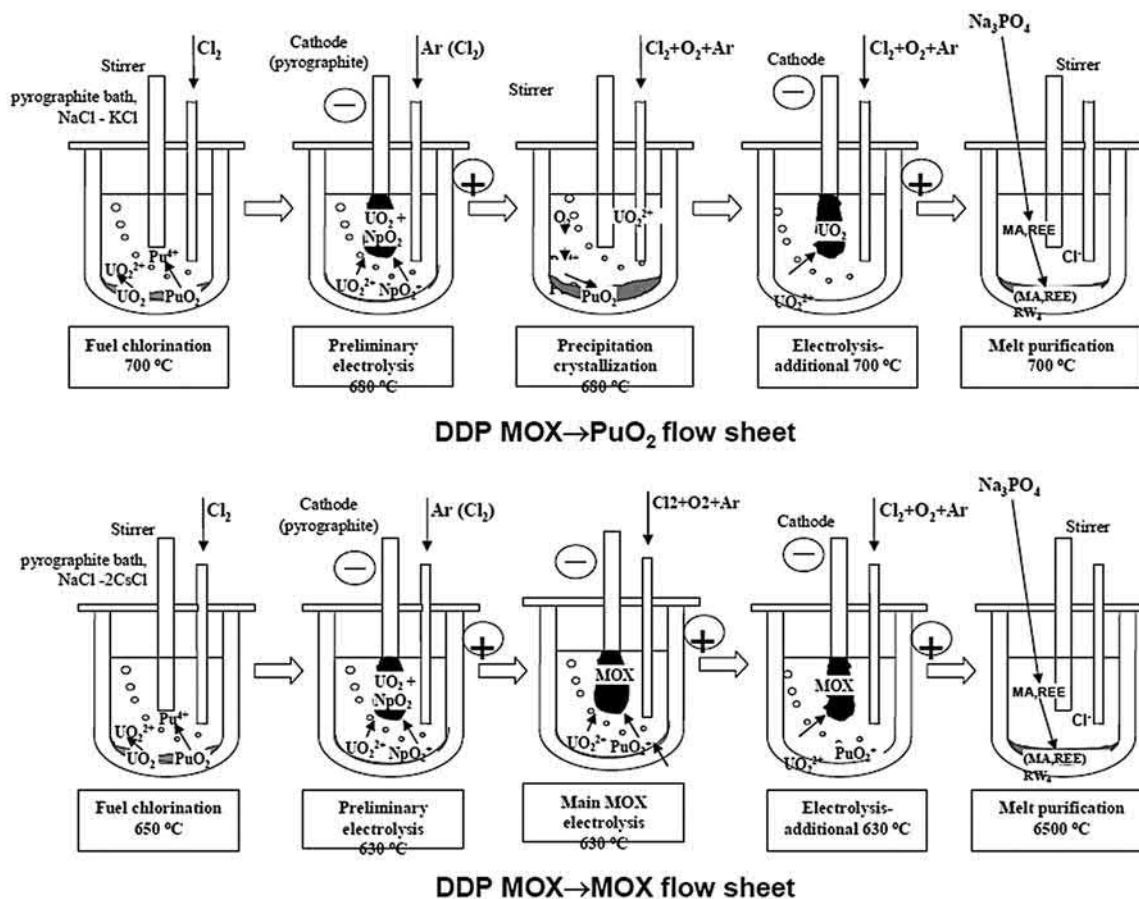


Fig. 19 DDP MOX → PuO₂ and DDP MOX → MOX flow sheets. From Bychkov A (2009) *Current Status of Development in Dry Pyroelectrochemical Technology of Spent Nuclear Fuel Reprocessing*, Joint ICTP/IAEA School on Physics and Technology of Fast Reactors Systems, 9–20 November. <http://indico.ictp.it/event/a08209/session/55/contribution/32/material/0/0.pdf>.

Decontamination factors (DF) from main FPs

Fuel type	Main FPs				
	Ru-Rh	Ce-Pr	Cs	Eu	Sb
PuO ₂ for BN-350 (test, 1991)	50	220	> 3000	40	200
PuO ₂ for BOR-60 (test, 1995)	33	40..50	4000	40..50	120
UO ₂ for BOR-60 (test, 2000)	> 30	~	> 4000	> 200	~
(U,Pu)O ₂ for BOR-60 (test, 2001)	20 - 30	25	~ 10000	> 100	~

Fig. 20 Decontamination factors (DF) from main FPs from RIAR experience in reprocessing of spent fuel of the BOR-60 and BN-350 reactors. From Bychkov A (2009) *Current Status of Development in Dry Pyroelectrochemical Technology of Spent Nuclear Fuel Reprocessing, Joint ICTP/IAEA School on Physics and Technology of Fast Reactors Systems*, 9–20 November. <http://indico.ictp.it/event/a08209/session/55/contribution/32/material/0/0.pdf>.

Liquid-liquid reductive extraction

An alternative to the electrolysis is the liquid-liquid reductive extraction developed at the CEA in the years 2000s (Conocar et al., 2005b). This process, operated in molten LiF-AlF₃ salt requires a dissolution of the spent fuel in the salt and is more relevant for oxide fuel reprocessing since the production of metal would require additional reduction processes. In such a salt, uranium, plutonium and americium are very selectively reduced into liquid aluminum thanks to the reductive properties of aluminum toward these elements in a bespoke contactor (Fig. 21). This is due to the specific activity coefficients of U, Pu and Am in Al. In one extraction batch, it is possible to recover 99% of Pu and Am with decontamination factors toward Ce and Sm higher than 250 (Fig. 22) (Conocar et al., 2006).

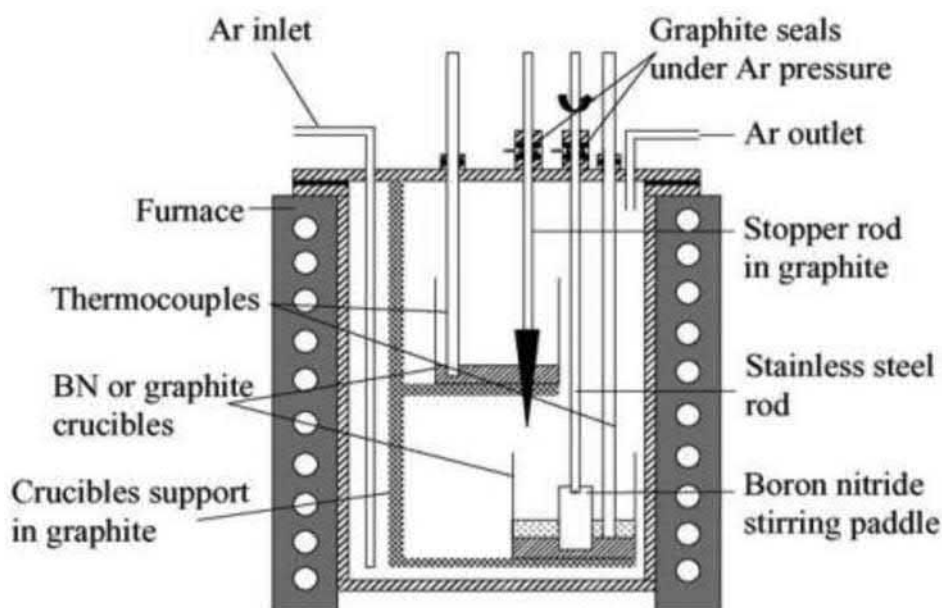


Fig. 21 Schematic cross section through the high temperature liquid-liquid contactor (HTLLC). From Conocar O et al. (2005a) Distribution of actinides and lanthanides in a molten fluoride/liquid aluminum alloy system. *Journal of Alloys and Compounds* **389**: 29–33.

TABLE I

Demonstrative Experiment in Glove Box:
Initial Composition of the Salt

Compound	Weight (g)	Mole Percent	Weight Percent
LiF	10.82	83.03	53.92
AlF ₃	6.18	14.65	30.80
PuF ₃	2.228	1.50	11.10
AmF ₃	0.024	0.02	0.12
CeF ₃	0.502	0.51	2.50
SmF ₃	0.105	0.10	0.52
EuF ₃	0.102	0.10	0.51
LaF ₃	0.105	0.11	0.52
Total	20.066	100.00	100.00

TABLE II

Demonstrative Experiment in Glove Box: Distribution
Coefficients of Pu, Am, CE, Sm, Eu, La, Separation
Factors with Pu, and Equilibrium Compositions

Element	D_M	$S_{Pu/M}$	Equilibrium (mg/g)	
			Salt	Metal
Pu	197 ± 30	1	0.4 ± 0.05	82.9 ± 3.3
Am	144 ± 20	1.4 ± 0.4	0.007 ± 0.0003	1 ± 0.05
Ce	0.142 ± 0.01	1307 ± 308	16.3 ± 0.6	2.25 ± 0.1
Sm	0.062 ± 0.006	3177 ± 760	3.9 ± 0.1	0.24 ± 0.02
Eu	<0.013	>15000	4.2 ± 0.2	<0.05
La	<0.06	>3000	3.85 ± 0.1	<0.2

Fig. 22 Results from a lab scale demonstration of the liquid-liquid reductive extraction. From Conocar O et al. (2006) Promising pyrochemical actinide/lanthanide separation process using aluminum. *Nucleat Science and Engineering* **153**: 253–261.

Once the pure actinides in the aluminum, they have to be back-extracted. This is possible by contacting the liquid metal with a molten chloride salt containing AlCl₃. The actinides are oxidized as AnCl₃ and transferred to the salt as cations (Mendes et al., 2012). They can be precipitated as oxide in order to re-manufacture fresh oxide fuel.

This route is less attractive for metallic fuel processing however, it could be suitable for nitride or carbide fuels.

In the first developments, the fuel has to be dissolved in the fluoride salt by fluorination using HF gas for instance. However, an innovative approach based on the Direct Solubilization Process (DOS) allows a direct processing of the solid oxide spent fuel (Mendes et al., 2016).

This process can also be used to reprocess CERCER transmutation targets (Mendes et al., 2016).

The key issue for developing an industrial pyroprocess

As shown in section “Current experimental feedback and key demonstration projects”, promising options have been developed up to the pilot scale but most of the developments have remained at the lab scale.

An international mobilization was observed between mid-1990s until mid-2010s but the effort has slowed down since then.

The main reason is the lack of industrial perspective for the development of fast reactors with metallic fuel worldwide, and more globally the lack of the development of advanced reactor fuel cycles. There is no clear perspective in Japan, a stop of the program in Korea, the selection of MOX pellet fuel for BN800 in Russia.

A renewal of pyrochemical process development could come from the molten salt reactors. However, the thermodynamics is not favorable since it imposes that first the actinides are removed from the salt before removing the lanthanides, which makes the process flowsheet complex. In addition, the programs led by US or Canadian start-ups focus on concept without processing, or with the minimum level of salt polishing. Moreover, concepts based on chloride salts could easily rely on an aqueous processing.

Finally, today, pyrochemical processes are far to be industrially deployed. Those addressing oxide fuels do not bring real benefit compared to aqueous separation processes, not to say ore drawbacks, in particular in terms of waste management and technological challenges (high temperatures, material corrosion, etc.). If the liquid-liquid reductive extraction core process is efficient and very compact, the whole flowsheet would be much more complex.

However, an opportunity could come from very specific and refractory transmutation targets that would hardly dissolved in aqueous media and therefore not suitable for such processes.

Pyroprocesses addressing metallic fuels could be deployed if a fleet of fast reactors operated with metallic fuel would be deployed. No plan exists today. However, in such a case more efforts should be put on the whole fuel cycle design “from fuel to fuel,” not only focusing on actinide purification and recycling, but also on all the ancillary steps that make a process viable salt fabrication and purification, maintenance, plant operation, analytical techniques and monitoring, material resistance and corrosion control. However, as minor actinides can hardly be separated from plutonium, such processes would lead to a homogeneous recycling strategy. For heterogeneous recycling strategies, in particular involving the sole recycling of americium, aqueous reprocessing would be needed even for metallic fuel, which reduces the interest of pyrochemical processes in this case (Poinssot et al., 2017).

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The Current Industrial Mono-Recycling of U and Pu: Reprocessing Step

Gérald Senentz, Orano, Châtillon, France

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Glossary

- 1CUPu 1st uranium plutonium extraction cycle.
- 2CU 2nd uranium liquid-liquid extraction cycle.
- 2CPu 2nd plutonium liquid-liquid extraction cycle.
- LWR Light Water Reactor.
- MOX Mixed Oxide fuel.
- NOX NO, NO₂.
- TBP Tri-*n*-butyl phosphate.

The implementation of PUREX process at the industrial scale

Process steps

The purpose of fuel treatment is to recover reusable material (uranium and plutonium), with a level of purity compatible with the recycling operations, and to minimize and condition waste arising from non-reusable spent fuel components (fission products and minor actinides, but also metallic structures). To that end, very high recovery rate ($> 99.5\%$) and decontamination factors ($> 10^6$) must be achieved.

Current reprocessing plants use variants of the PUREX process. The main part of this process is based on hydrometallurgy, where a selective solvent, tri-*n*-butyl phosphate, is used with nitric acid in liquid-liquid extraction steps to recover reusable materials (uranium and plutonium), and condition the rest of the spent fuel (fission products, minor actinides, and fuel metallic structural elements) as waste. Process steps are hence organized to dissolve the material, then recover and purify the products. Those products are sent to recycling plants. Liquid effluents arising from those operations are treated before release or solidification, and waste are conditioned (Fig. 1).

Fuel receipt and storage

The very first step is to receive the spent fuel from the nuclear power plant or an interim storage and transfer it to the pool. The fuel assembly being transported in heavy transport casks, the task at hand is to open the cask, unload the fuel, transfer the fuel to the

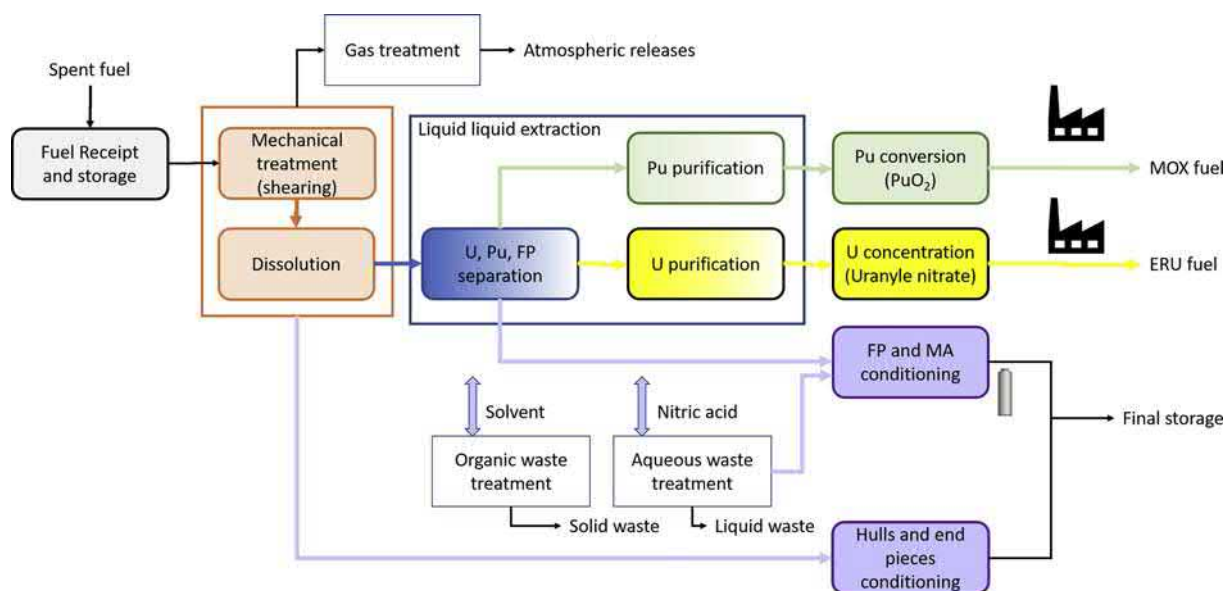


Fig. 1 Fuel Treatment Process Steps.

pool, while managing irradiation and contamination risks from very active nuclear fuels. Additional operations are required to verify the identity of the fuel received, through visual observation of the fuel identification plate and nuclear measurements.

Two different unloading processes can be used.

- Underwater unloading is similar to what is performed at a nuclear power plant. The transport cask is opened under water, water providing both irradiation protection and cooling. The fuel is lifted with a crane, transferred to a basket and moved to the pool.
- Dry unloading on the other hand requires a docking station, and the transport cask design must be compatible with this docking station to provide a leak tight confinement when the cask is opened. The fuel remains dry until it is lifted down to a control well or to a basket for storage in the pool. Irradiation and contamination protection rely on the concrete walls of the unloading cell and the leak tightness of the docking station.

Dry unloading provides distinctive advantages. The contaminated inside cavity of the transport cask does not come into contact with large volumes of water, limiting the generation of radioactive effluents. In addition, damaged spent fuel can be identified and encapsulated in a container to avoid contamination of the pool. However, this dry process is less flexible, since the fleet of transport casks must be compatible with the initial design of the unloading facility.

At the end of the unloading process step, the spent fuel is safely stored under water in the storage pool of the reprocessing plant. There, water is treated and cooled to maintain it at low contamination levels and close to ambient temperature to prevent evaporation. Production programs and contracts determine when the fuel is transferred out of the pool for reprocessing, with a typical total cooling time of about 5 years. Additional cooling time decrease the radioactivity of the fuel, limiting solvent degradation (see “Organic effluents management” section), and may help reduce dose rate and releases. However, with additional cooling time, more ^{241}Am form from ^{241}Pu radioactive decay, decreasing the energy value of plutonium and increasing long term radio-toxicity and decay heat of the waste.

Mechanical treatment and dissolution

The purpose of the mechanical treatment is to prepare the fuel for dissolution and separate parts incompatible with or detrimental to the downstream process.

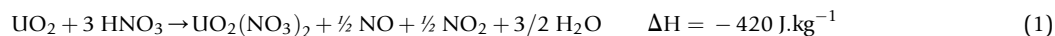
For fuels with a magnesium cladding (Magneox fuels for example), the mechanical process objective is to remove (to peel) the cladding before dissolution, in order to avoid dissolution of large amounts of magnesium that would increase the amount of high activity waste. The metal rod is then transferred to the dissolver.

For UOX fuels from current light water reactors, the cladding is made from zirconium alloys, not soluble in nitric acid. The mechanical process objective in this case is to ensure nitric acid will selectively leach and dissolve the oxide material despite the insoluble cladding. To that end, the zirconium cladded fuel is sheared (chopped) into small pieces of a few centimeter long, called hulls.

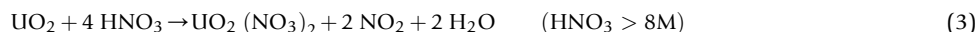
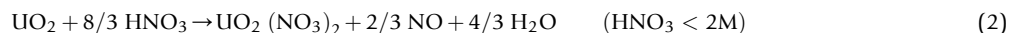
During the shearing process, a hydraulic gag presses and holds the fuel while a hydraulic blade cut through the fuel rods. The hulls fall by gravity through a chute to the dissolver. Between two cuts, the blade is moved back, the pressure on the hydraulic gag is released, and the fuel is pushed forward a few centimeters, exposing that much to the next cut. Those steps are repeated until the whole fuel has been fully cut.

Head and bottom end pieces of the fuel can be separated and directed to metallic waste conditioning through a separate chute, depending on the design of the dissolver and its compatibility with those large pieces.

The dissolution reaction of UOX fuel in nitric acid is described by Eqs. (1) through (3):



The exact proportion of NOx produced during the oxidation of UO₂ depends on the acidity and temperature of the dissolution solution. NO is favored at low acidity, while NO₂ dominate at high acidity and high temperature:



where Eqs. (2), (3) are related through an equilibrium reaction between NO and NO₂:



Plutonium is dissolved according to Eq. (5):



The effective dissolution rate is controlled mainly by nitric acid concentration, temperature, and surface area. Nitric acid concentration is chosen in a range compatible with the following liquid-liquid extraction process step, typically at about 3 mol L⁻¹ after dissolution, with a uranium concentration of 200–250 g L⁻¹. To accelerate dissolution, dissolvers are heated. Dissolution temperature can be up to the boiling point of the dissolution solution, although high temperature favors the formation of precipitates containing MoZr. Surface area is favored by reducing the length of hulls, but doing so also increases the number of cuts to shear the fuel, and hence the time to process it. Hulls length of a few centimeters is classically used.

Dissolution has been initially implemented as a batch process (for instance in Magnox plant in UK, UP1 and UP2 400 plants in France). For efficiency reason, a continuous process is used in more recent plants (UP2 800, UP3, Rokkasho), with a continuous rotary wheel dissolver. The wheel is divided into baskets. Through the rotation of the wheel, each basket moves from a loading position, where it receives hulls from the shearing machine, to dissolution positions, during which the basket is kept submerged in boiling nitric acid solution, and finally to the unloading position where hulls fall towards the metallic waste treatment. The duration of the dissolution process can be adjusted through modifying the rotation speed of the dissolution wheel (Fig. 2).

THORP reprocessing plant presents an intermediate design, with three batch dissolvers working in sequence, fed alternatively through a movable chute. Each dissolver is equipped with a basket to recover and empty the hulls after dissolution.

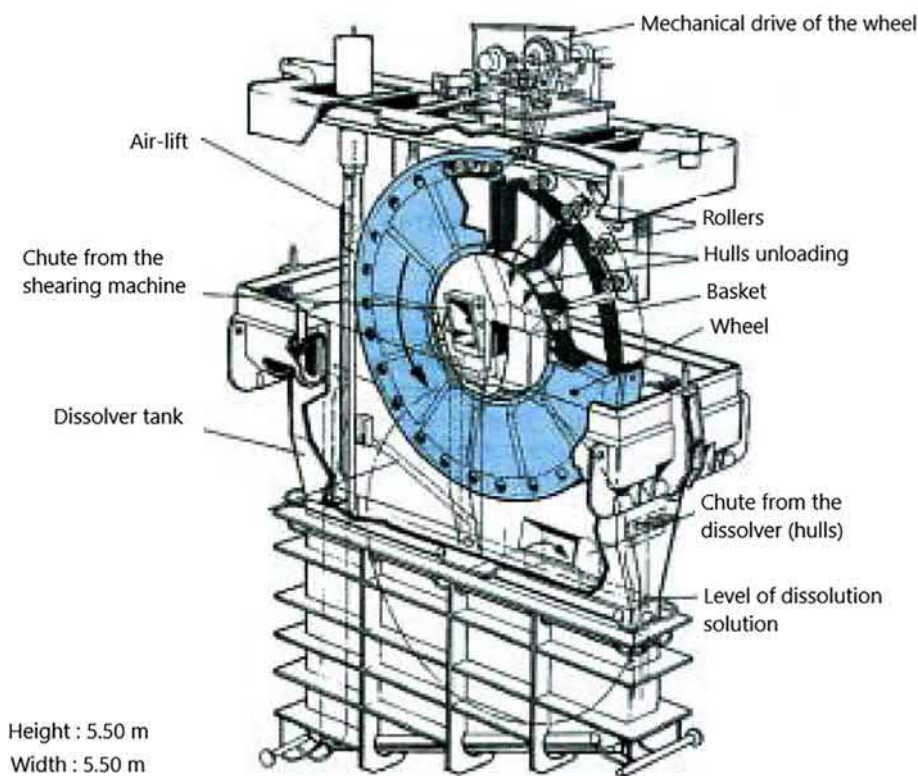


Fig. 2 The rotary wheel dissolver (CEA, 2008). Courtesy CEA.

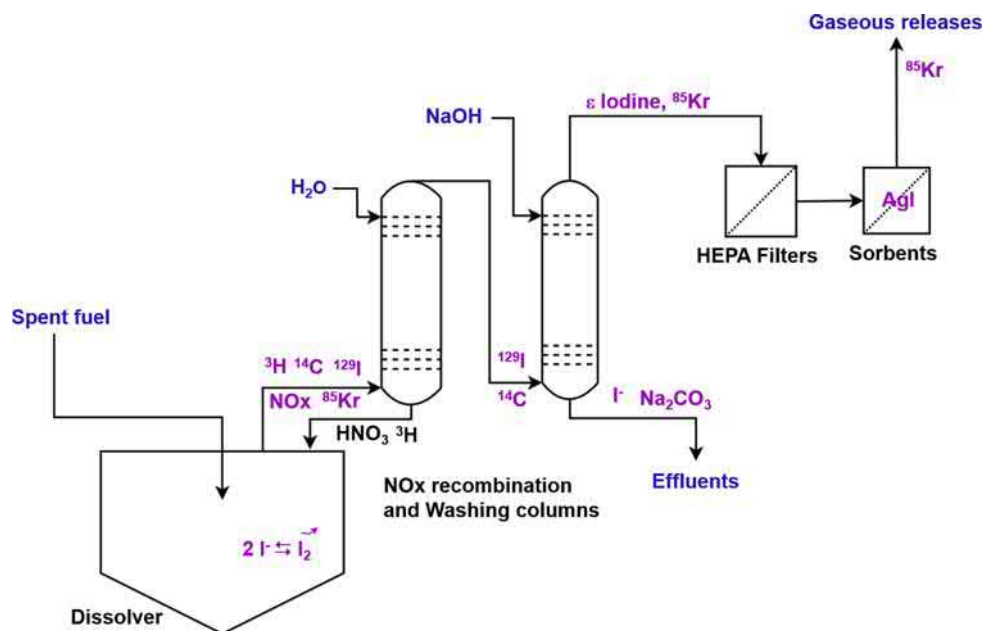


Fig. 3 Treatment of gases from shearing and dissolution.

Three streams come out of the shearing and dissolution step.

- The main process stream is the dissolution liquor, i.e. nitric acid having dissolved the oxide part of the fuel, containing uranium, plutonium, minor actinides, and fission products. Sparging is used to remove iodine from the solution, directly in the dissolver at the end of the dissolution (batch dissolution process), or in a subsequent process vessel (continuous dissolution process). Precipitates and metallic particles are separated, using for example a centrifuge. In a buffer storage tank, solutions are characterized to account for fissile material received in the reprocessing plant. After dissolution, uranium is present as uranyl nitrate U(VI). Plutonium is present as Pu(IV). Some Pu(VI) can coexist, especially at high acidity, or when using a batch dissolver, because of the low amount of nitrous acid at the end of the dissolution. Because of its lower extractability, Pu(VI) in significant concentration could lead to plutonium accumulation in the extraction system. Systematic analysis and, where necessary, sparging with nitrogen dioxide, ensure plutonium is almost entirely under Pu(IV), as required by the following extraction and purification cycle. The dissolution liquor can then be sent to the extraction and purification cycles.
- Hulls and end pieces, i.e. the zirconium-made metallic structure of the spent fuel, are solid waste. They are conditioned in drums filled with concrete (THORP and initial La Hague design) or compacted (La Hague UP3 and UP2-800) to reduce the waste volume.
- Gases from shearing and dissolution contain nitric oxide and nitrogen dioxide produced during the dissolution reactions. They also contain small droplets of dissolution solution entrained with gases, and volatile fission products, especially ^{85}Kr , ^{129}I , ^{14}C , ^3H , and must be treated before release (see Fig. 3). Gases are first contacted with nitric acid and water solutions in washing columns. NOx are recovered as nitric acid. Droplets of radioactive liquid, including most of tritium present as tritiated water, are captured. Tritium is ultimately diluted into the sea in a high current zone, with distillates of the tritiated acid recovery unit (see below).

Gases are then contacted with caustic soda to capture ^{14}C present as CO_2 , and iodine present as I_2 . High efficiency filters and sorbents finally capture remaining traces of non-volatile material and iodine.

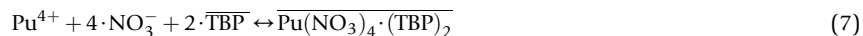
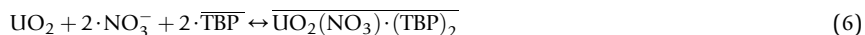
Captured ^{14}C can be either stabilized with concrete, and then stored (Sellafield, UK), or neutralized and released to the sea (La Hague). Captured ^{129}I can be precipitated and conditioned as solid waste (Rokkasho), or released to the sea (La Hague) where its radiological impact is low, thanks to dilution with natural iodine present in sea water.

^{85}Kr is a noble gas and as such does not interact with washing columns and filters. It is released to the atmosphere through a stack. Thanks to its chemical nature, its radiological impact to the population and the environment is low.

Detailed studies demonstrate that the effective dose potentially absorbed by people living near La Hague at $5 \mu\text{Sv yr}^{-1}$ for a fisherman at the nearby village of Goury and at $13 \mu\text{Sv yr}^{-1}$ for a farmer at the nearby village of Digulleville. Both villages were chosen as the most likely impacted nearby sites. Those estimates are consistent with IRSN estimates based on the results of environmental monitoring (IRSN, 2018).

Extraction and purification cycles

The PUREX process uses a solvent, tributyl phosphate (TBP), to selectively extract uranium and plutonium from dissolution liquor, separate then purify them. TBP forms complexes with uranium and plutonium, immiscible in water, effectively extracting those materials from the aqueous solution.



Reaching both a very high recovery rate and decontamination factors are seemingly contradictory objectives, as it means quantitative extraction of uranium and plutonium, and almost no fission product extraction, a level of solvent selectivity impossible to reach in single step. A succession of counter current liquid-liquid extraction steps are necessary to achieve the high level of uranium and plutonium recovery, and to purify them to the extent required by downstream fuel fabrication and recycling steps.

The solvent and the aqueous phase are separated using decantation. Because the density of TBP is very close to the density of an aqueous phase (0.979 at 25 °C, [Chemicalbook, 2021](#)), or even higher when complexed with uranium ([McKibben, 1975](#)), TBP has to be mixed with a light organic diluent, at 20%–30% vol. depending on plant design, ensuring an efficient separation of organic and aqueous phase. Addition of a diluent also decreases viscosity.

Industrial diluent can be odorless kerosene (Thorp reprocessing plant in the U.K.), hydrogenated polypropylene tetramer (La Hague in France) or n-dodecane (Rokkasho in Japan). Hydrogenated polypropylene tetramer is a highly branched 12-carbon alkane, with other C10 to C13 alkanes. It is the most resilient to third phase formation, a dangerous phenomenon where high concentration of plutonium or uranium induce an organic phase splitting, leading to diluent and a dense third phase of (U, Pu, TBP, HNO₃) ([Plaue et al., 2006](#)). The downside of hydrogenated polypropylene tetramer is a lower flash point, similar to n-undecane.

Extraction is sensitive to the following effects:

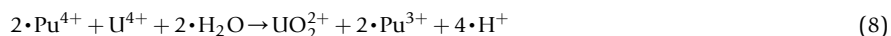
- Salting out: D_U increases with nitrate concentration in the aqueous phase
- Saturation: in consistency with Eq. (6), uranium extraction by the solvent is limited to 1 mol of uranium per 2 mol of TBP.
- Competition: U, Pu, H⁺ are all extractable, but saturation of the solvent by uranium will limit extraction of Pu, and H⁺
- Temperature: D_U decreases when temperature increases

where D_X is the distribution coefficient of component X, defined here as the chemical X concentration ratio between the organic and the aqueous phase.

For a given element, distribution coefficients vary greatly depending on the oxidation state, as illustrated in [Table 1](#). Solvent extraction and actinides separation in the PUREX process relies on the adjustment of the oxidation state of uranium and plutonium to separate those two elements.

The process to separate uranium, plutonium, and fission products, known as the first Uranium, Plutonium Cycle (1CUPu), implement those basic principles as illustrated in [Fig. 4](#).

In the feed of the extraction cycle, U(VI) and Pu(IV) are readily extractable, while most of fission products and minor actinides are not. In those conditions, uranium and plutonium are extracted by the solvent, while most fission products and minor actinides remain in the aqueous phase. A scrubbing operation removes fission products. Neptunium and traces of fission products (Ru, Tc...) still follow the desired products. To strip plutonium out of the organic phase, solvent is contacted with a reductant, uranous nitrate (U(IV)), stabilized with hydrazine (N₂H₄). Plutonium is reduced to Pu(III), non-extractable, and neptunium to Np(IV), extractable ([Fig. 5](#)).



Out of the plutonium stripping operation, the organic phase contains uranium and neptunium, while the aqueous phase has recovered plutonium. A uranium scrubbing step recovers U(IV) that could have followed plutonium and the aqueous phase. Plutonium solution is finally contacted with NO_x to decompose hydrazine and oxidize plutonium back to Pu(IV), ready for the second plutonium liquid-liquid extraction cycle (2CPu). Uranium is finally stripped, along with neptunium and traces of fission products,

Table 1 Valency states and extractability in TBP of uranium, neptunium, plutonium ([Guillaume et al., 1984](#)).

	Oxidation State			
	+III	+IV	+V	+VI
U		U ⁴⁺		UO ₂ ²⁺
Np		Np ⁴⁺	NpO ₂ ⁺	NpO ₂ ²⁺
Pu	Pu ³⁺	Pu ⁴⁺		PuO ₂ ²⁺

: Weak, good and excellent extractability

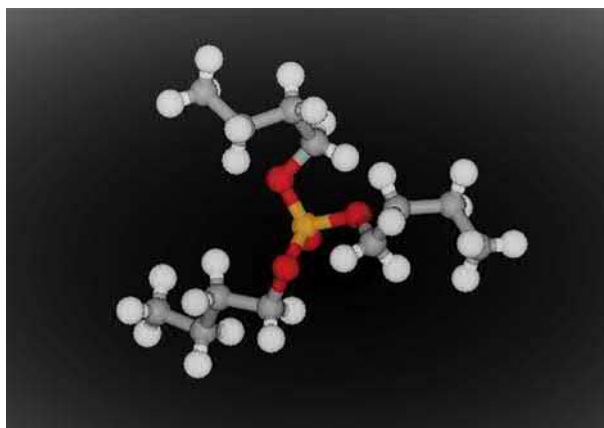


Fig. 4 TBP molecule $O=P(OC_4H_7)_3$.

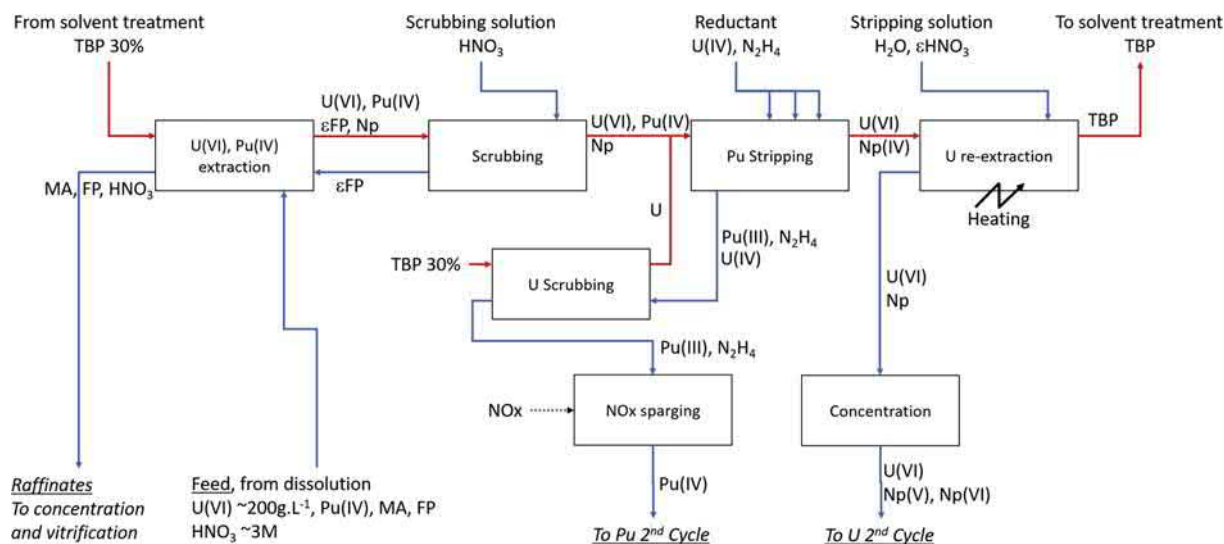


Fig. 5 1CUPu: U, Pu and FP separation.

by contacting the solvent with a very low acidity aqueous solution while slightly heating the solution (around 50 °C). Uranium is then concentrated, ready for the second uranium cycle (2CU).

To reach the end-products specifications, an additional purification cycle is necessary, for both uranium and plutonium, with process conditions targeting the remaining contaminants.

- Uranium from 1CUPu is extracted again by TBP. A scrubbing operation removes traces of fission products still present (Ru...). Addition of hydrazine, a reductant weaker than U(IV), reduces neptunium to Np(V), but does not reduce it to Np(IV). Np(V), non-extractible, is stripped from the solvent. Final uranium re-extraction uses again the effect of temperature and low nitrate concentration to strip uranium from the solvent. Uranium is finally concentrated (up to 400 g L⁻¹) and turned into oxide before being sent to recycling plants.
- Pu(IV) from 1CUPu is also extracted again by TBP, while remaining contaminants stay in the aqueous phase. It is then reduced to Pu(III) using another reductant, a mixture of hydroxylamine nitrate (HAN) and stabilizing hydrazine (N₂H₄), and stripped back to the aqueous phase. The aqueous phase is contacted with NO_x to eliminate the reductants and oxidize plutonium to Pu(IV), ready for the conversion step.

Industrial implementation of the extraction cycles

The equipment used for the liquid-liquid extraction can be mixer settlers, pulsed columns or centrifugal extractor (Drain et al., 2003). All those technologies are used at different process steps, and different plants have different sets of extractors.

- Mixer settlers are simple to operate, especially during short shutdowns. They are very flexible and can accommodate a wide range of capacity. They have a large footprint. They can accumulate interfacial cruds. Because of usually large holdups, they may lead to

more solvent degradation due to radiolysis and hydrolysis. Extra-flat mixer settlers have been designed where subcritical geometry was required, but their capacity is limited. In the context of a reprocessing plant, and to provide protection from irradiation and contamination, mixer settlers are typically installed below a concrete slab. Motors for the mixers are installed above this slab, with a leak tight shaft going through the slab to link motors and mixers. That way, motors can be maintained and replaced when necessary without exposing workers to radiation.

- Pulsed columns with the right packing have low sensitivity to interfacial cruds. They can be designed with no moving part in radioactive areas: the liquid in the pulsation leg is pushed with compressed air, controlled with a valve located out of radioactive areas, and as such easily maintainable. Subcritical geometry can be achieved, even for relatively high throughput, with annular columns fitted with a neutron absorber shaft along the axis of the column. However they are high (up to 8 m), and need tall buildings to be installed in.
- Centrifugal extractors are mechanically much more complex, and therefore more sensitive to maintenance. Their main advantage is their compacity compared with the two other kinds of extractors. Their small holdup and residence time lead to lower solvent degradation. They are not tolerant to interface cruds.

For the 1CUPu, pulsed columns are usually selected, due to criticality constraints and the presence of interphase dross. For uranium cycle, with no criticality constraint and low irradiation risk, mixer settlers are preferred. For plutonium cycle, with high criticality constraints, centrifugal extractor or pulsed columns can be used.

To avoid the maintenance of mechanical moving parts contaminated by radioactive solutions, transfers use airlifts, ejectors, siphons, or just gravity, to the most possible extent, with control valves in non-radioactive areas. Pumps, when necessary, are designed with the motor above a shielding slab, and the core of the pump is such that it can be replaced using a shielded enclosure. Chemical equipment, with no mechanical moving parts, is installed in blind cells and is designed to last for the lifetime of the plant. Despite this initial design, equipment such as evaporators had to be replaced, due to corrosion.

Plutonium conversion

Plutonium conversion involves turning plutonium nitrate, coming from the extraction and purification cycles, into plutonium oxide (PuO_2), the required form for light water reactor fuel fabrication.

Plutonium nitrate is precipitated by mixing plutonium nitrate with oxalic acid. The plutonium oxalate slurry is filtered with a rotary filter and dried. Plutonium oxalic is then turned into plutonium oxide PuO_2 in an electrically heated screw calciner, at high temperature with an oxidizing atmosphere.

The key role of streams management and effluent recycling

Special care is needed during the plant design in order to reach high decontamination factors before release to the environment, and to minimize waste volumes. Aqueous effluents are sent to evaporators, where salts, including radioactive contaminants, are concentrated. Several consecutive evaporation and condensation cycles are necessary in order to minimize radioactive releases. To ensure the efficiency of waste management, effluents are sorted according to their radioactivity (presence or not of tritium, level of activity) and their chemical nature (aqueous, acidic or caustic, organic).

Aqueous effluents management and releases

Aqueous effluents treatments and nitric acid recovery

Acidic aqueous effluents arise from various plant operations: raffinates from extraction and purification cycles, water and nitric acid sent to gas washing columns, laboratory effluents, with activity levels from very high (raffinates from the first extraction cycle, containing all fission products and minor actinides from the spent fuels) to low (effluents from gas washing column of low activity parts of the plant).

Because of its mobility, tritium creates a contamination concerns where its concentration is significant. Therefore, the plant is divided into two broad areas, tritiated and non-tritiated parts. Tritium is present in significant concentration from dissolution down to the fission product scrubbing column of the first extraction cycle. Some tritium is extracted by the solvent in the form of tritiated water or tritiated nitric acid and could follow TBP to the non-tritiated part of the plant. The use of non-tritiated water and nitric acid for scrubbing solutions removes tritium from the solvent before it can reach the non-tritiated area. Each area, tritiated and non-tritiated, have their own aqueous effluents treatment and nitric acid recovery, forming respectively a tritiated acid loop, and a non-tritiated acid loop.

In the case of tritiated nitric acid recovery (illustrated in Fig. 6, actual plant design involving several evaporators running in parallel), the first evaporator receives the most active aqueous effluent, i.e. the raffinates from the first extraction cycle. It also receives concentrates from the medium activity waste concentration. Concentrates from this evaporator feed vitrification. Distillates feed the medium activity waste evaporator. The distillates of this second evaporator feed the tritiated acid recovery. Essentially a distillation column, the acid recovery concentrates nitric acid at ~ 10 M; concentrated nitric acid is recycled and is used to dissolve the spent fuel. Distillates from this column are analyzed to verify their remaining activity, then released to the sea. Tritiated water properties

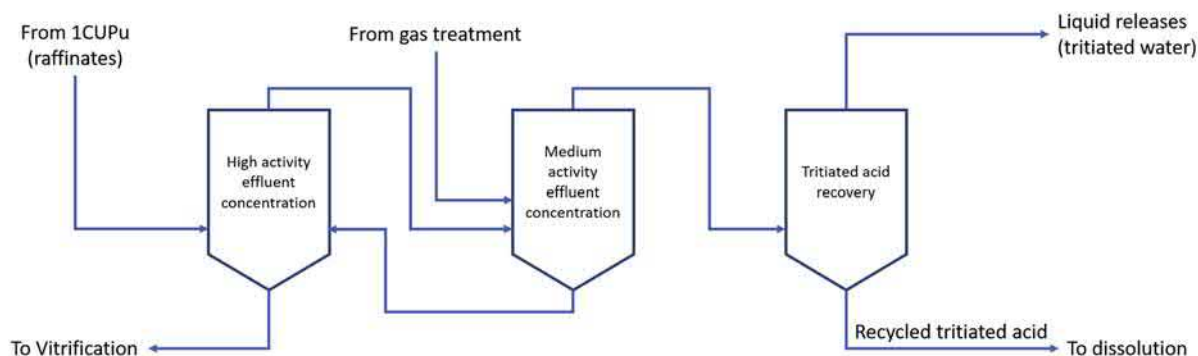


Fig. 6 Principle of aqueous waste management, and tritiated acid recovery.

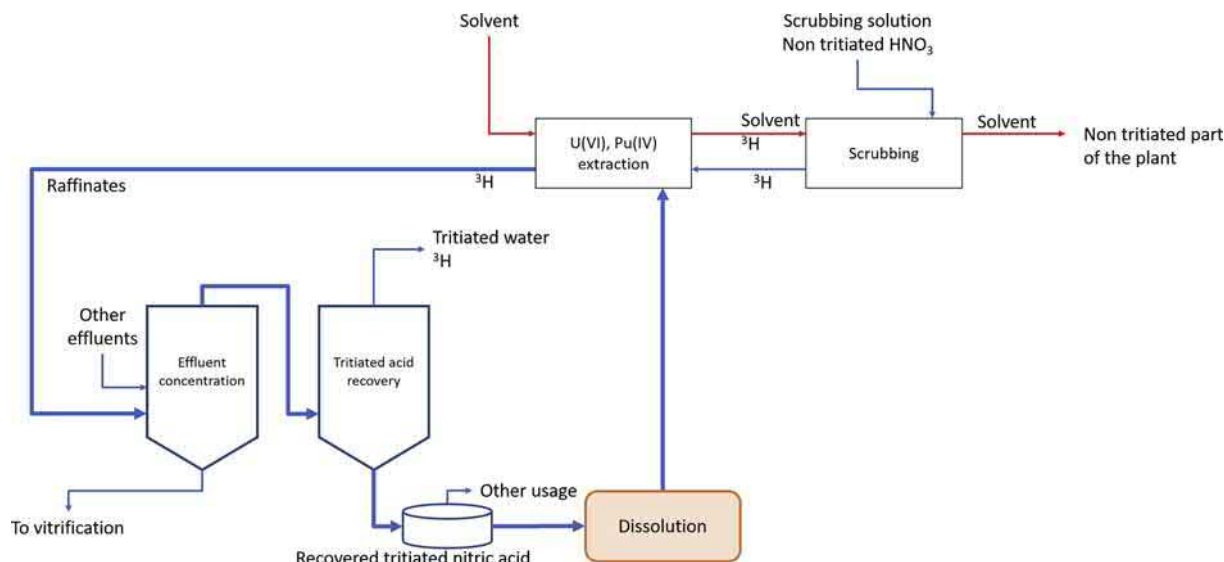


Fig. 7 Tritiated nitric acid loop.

being very close to the behavior of natural water, tritium is not concentrated in either of those streams, and is released with the distillates of the acid recovery unit (tritiated water).

All stable non-volatile salts added to the process end up in the concentrates of the high activity effluent concentration, and from there in glass canisters. Great care is taken to limit such addition; reagents are all made of C, H, O, N atoms, leaving at the end no residue (oxidized with NO_x or by concentrated nitric acid in evaporators), with the notable exception of caustic soda and sodium carbonate (Fig. 7).

The non-tritiated acid loop is similar in principle. However, the activity of the streams is very much different: received effluents come from areas of the plant downstream the fission product scrubbing, i.e. medium activity. Concentrates from the first evaporator is not sent directly to vitrification, due to its lower radioactivity, but is routed instead to the high activity evaporators of the tritiated nitric acid recovery.

Caustic effluents management

Solvent treatment use caustic reagents, caustic soda and sodium carbonates, to clean and recycle the solvent. Once used, those reagents generate caustic effluents. Caustic effluents from the first uranium plutonium cycle are concentrated in a specific evaporator. Distillates from this evaporator are released after control. Concentrates are conditioned as solid waste.

Organic effluents management

Tributyl phosphate sent to the extraction cycles is recovered after uranium stripping (or plutonium stripping, in the case of the 2CUPu). However, radiolysis and hydrolysis of TBP generates degradation products of the solvent (Fig. 8).

These degradation products are very detrimental to the purification cycles performance. They tend to stabilize emulsions. They can extract and maintain plutonium in the organic phase, even after the addition of reductants. Through the design of the plant and

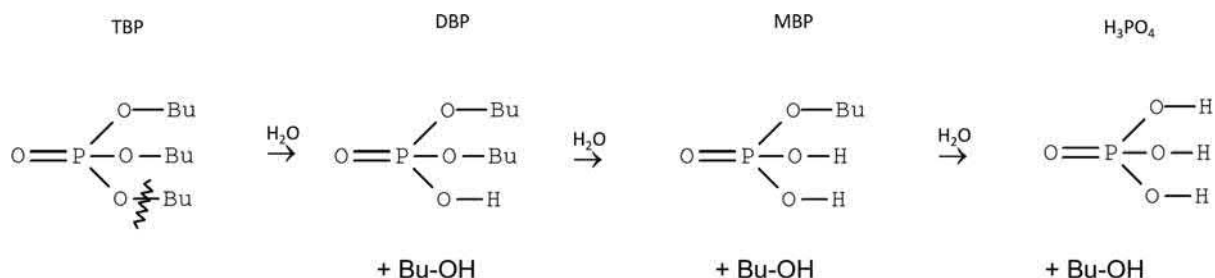


Fig. 8 Main TBP degradation products.

operating rules, solvent degradation is minimized, trying to limit the time of solvent exposure to radiation and nitric acid. Solvent treatment is nevertheless essential for the lasting performance of the purification cycles.

Solvent sent to the extraction and purification cycles is recovered after plutonium or uranium stripping and is washed in mixer settlers with sodium carbonate and caustic soda reagents. This operation removes acidic degradation products and is enough to keep the solvent working. However, with this treatment alone, solvent must be entirely renewed after some time of service (years, depending on the expected performances), to retrieve acceptable performances.

An additional treatment, the Organic Waste Treatment (OWT), based on distillation, has been implemented at La Hague UP3 plant, then on UP2 800 and Rokkasho plant. Solvent is evaporated, and a distillation column separates diluent (the lighter phase), TBP, and a residue containing heavier degradation products and activity (Ginisty, 1991). Distillation is performed at low pressure and temperature, after washing the solvent with water to remove nitrates; otherwise, TBP would be degraded by excessive temperature and nitric acid, utterly defeating the purpose of this unit.

The combination of solvent treatment, using sodium carbonates and caustic soda, and organic waste treatment based on distillation has proved to be very effective at maintaining very high levels of decontamination factors and product purity achieved by the extraction cycles.

Another benefit of this unit is to produce a continuous stream of diluent (Fig. 9). This diluent is used to wash aqueous phase coming out of the extraction cycles, especially those going to evaporators. It extracts traces of TBP otherwise dissolved in the aqueous phase, improving downstream operations.

Used diluent is mixed with the process stream of solvent, effectively reducing TBP concentration below 30%. In the solvent treatment units, the operator adds TBP recovered from the Organic Waste Treatment unit to solvent, bringing its content back to the desired concentration. Keeping this TBP content within acceptable margins and controlling solvent flowrate to extraction cycles is critical for the good operation of the liquid extraction units.

The whole process, with the two complementary treatment, has kept the solvent in excellent conditions for 30 years. It also reduces the amount of waste, with a flowrate of residues equal to a few liters per hour.

Residues from the organic waste treatment unit are mineralized and conditioned as solid waste.

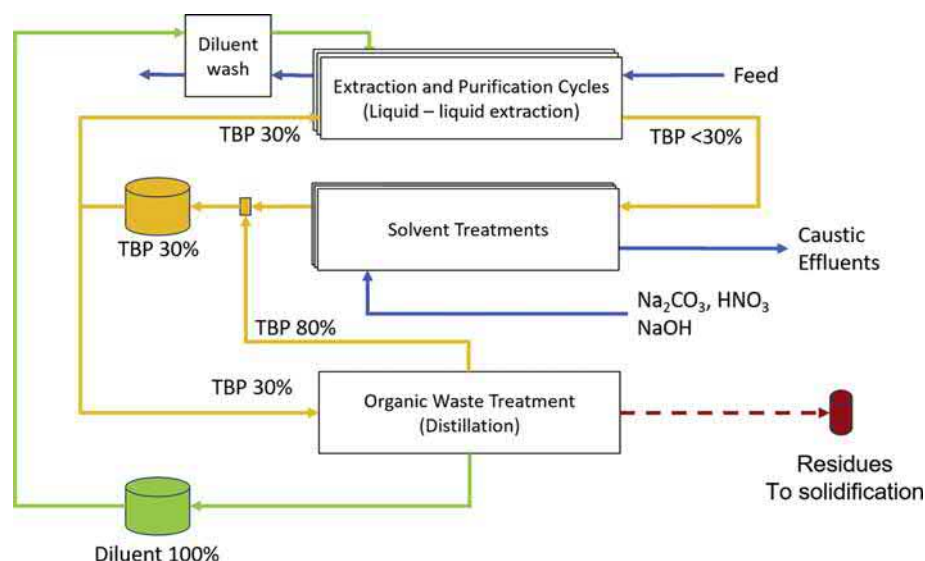


Fig. 9 Solvent management.

Buildings layout considerations

Nuclear risks to be addressed are contamination, irradiation, and criticality. Security of fissile materials requires access and material control. Flammable diluent and reactive reagents are used, so regular industrial risks are also present, such as fire risk, and chemical risks. Buildings are also designed to withstand earthquake, the magnitude of which being defined by the safety authority, taking into consideration historical and tectonic data, to allow for a safe shutdown of the facility and maintain desired confinement.

Initial reprocessing plants (Hanford, WA and SRS, SC in the USA) where built with a canyon type design, with a very long cell (more than 300 m long) housing the high activity process on one side, the medium activity on the other side. Along with versatile connection operated with a traveling crane, this design provides great flexibility. But with a more established flowsheet and emphasis on waste reduction and contamination prevention, individual process cells and welded pipes are preferred. Process cells and their concrete walls are meant to divide nuclear facility into areas with similar risks and constraints. They limit the spread of contamination in case of process leak and are designed to contain fire. Chemical process cells are organized around an “active gallery,” a large cell where pipes run from one cell to another, limiting concrete wall penetrations.

A typical chemical process cell, with no mechanical moving parts, is blind, with no maintenance planned for the life of the facility. They are fitted with drip trays, to detect and recover leaks, and air contamination level monitoring. Chemical equipment and pipes are welded, and chemical process cell contamination level is typically very low; contamination detection is often an early detection of a leak. Components requiring maintenance (pumps...) are specifically designed and fitted in process cells to allow for their replacement.

A mechanical process cell on the other hand is equipped with a range of maintenance tools: shielded windows for visual control, robotic manipulators for light maintenance and operation, traveling crane for heavy duty, and a super-maintenance system featuring a dedicated cell closed with a shielded door to maintain traveling cranes while being protected from irradiation. Because mechanical operations often involve breaching a confinement barrier (ex: shearing of spent fuel, or opening containers), contamination level can be high.

A key feature of a reprocessing plant is ventilation. Buildings are divided in successive zones, reminiscent of Russian dolls, from low contamination and irradiation risk (control rooms, changing rooms...) to higher contamination or irradiation risk (process cells...) located at the inner part of the facility. Control of non-contamination is mandatory when moving to a lower risk zone, to avoid spreading contamination to clean areas. Building ventilation, independent from process ventilation, is designed to sweep air from clean to more at-risk areas and regulate pressure differences accordingly. Process equipment is always at lower pressure than the surrounding cells. Air is filtered before release through building and process ventilation stacks, and contamination is monitored. Air contamination inside the buildings is continuously monitored through air sampling. Local and centralized alarms inform the personnel of any abnormal situations.

Ventilation is designed to mitigate the risk of fire, selectively shutting down air flow when and where necessary. It is also a preventative feature, meant to avoid the accumulation of potentially dangerous substance, such as hydrogen. Because of all the essential functions of ventilation, it is one of the few systems designed to always stay in operation, with the proper ventilator redundancy and proper power back-up.

Buildings are designed of course to accommodate process equipment. For example, the choice of pulsed columns for the extraction leads to tall buildings to house 8-meter-high columns. But buildings' size and process layout are also a consequence of selected transfer technologies. To avoid moving mechanical parts in contact with radioactive material, transferring by gravity, siphon or airlift is preferred to using pumps, when possible. This has a direct impact on possible relative elevation and maximum distance of process vessels.

Remote control and monitoring

Remote and automated operations are very effective at reducing operator exposure to contamination and irradiation. Therefore production activities are for the most part monitored and operated from centralized control rooms, except for maintenance activities and monitoring rounds, so much so that visitors are often surprised not to see any operators in the facility during their tour.

For safety reasons, the control system is organized in three independent systems. The mission of the first system is to pilot industrial production and to monitor the plant, just as any regular industrial process control system would do. It is supplied by the regular electrical grid. A second system is designed to bring and maintain the facility in a safe shutdown state when the first control system fails. From there, operators are able to control essential ventilation and refrigeration systems. Fire, contamination and irradiation monitoring is available. Onsite electrical generators and batteries supply electricity to the control system and local actuators. Finally, a third system is designed to control critical systems to avoid unacceptable consequences in accidental situations.

Many sensors used in the plant to feed the control system with data had to be developed or adapted to accommodate the high level of activity in the reprocessing plant and be virtually maintenance free. For example, bubbling rods are used extensively, to measure pressure, density, and volumes of solutions in tanks. In addition, specific nuclear measurements are implemented to measure plutonium concentrations or to characterize solid waste.

Analysis are often required before authorizing transfer of solutions to downstream process steps. Accountancy of fissile material before extraction cycles is just one example. Buffer tanks are thus necessary to keep the plant running while solutions are analyzed, and they significantly impact building volumes. With the development of online chemical analysis capabilities, it is tempting to

replace those buffer tanks by online analytical systems whenever possible, with an expected benefit on initial investment cost. However, those buffer tanks also provide flexibility and improve the overall reliability and availability of the plant, by decoupling otherwise continuous process steps.

Short industrial feedback from worldwide commercial plants

<i>Reprocessing plants</i>	<i>Design capacity</i>	<i>Start of operation</i>	<i>End of operation</i>	<i>Maximum annual production</i>
UP3 & UP2-800	1700 t yr ⁻¹	1989–1994	–	1700 t yr ⁻¹
THORP	1200 t yr ⁻¹	08/1997	11/2018	900 t yr ⁻¹
Rokkasho-Mura, JP	800 t yr ⁻¹	Supposed to be in operation in 2022	–	–

UK

THORP (Thermal Oxide Reprocessing Plant) is a nuclear reprocessing plant located in Cumbria, England, on the Sellafield nuclear site. Design capacity was never reached, and the plant suffered several outages, most notably about 3 years due to the 2005 leak of 83 m³ of dissolution liquor in the feed clarification cell ([Health and Safety Executive, 2007](#)). Failure to reach production targets led to economic difficulties. Decision to shut down reprocessing operations was taken in 2012. Reprocessing operations ended in November 2018, after fulfilling existing reprocessing contracts.

France

UP3 (Uranium Plutonium 3, names after older plants UP1 and UP2 in France) was built to treat nuclear spent fuels from Germany, Belgium, Switzerland, Italy, Spain, the Netherlands and Japan and started operation in 1989. Compared with previous plants, several major innovations were introduced, including the shearing machine, rotary wheel continuous dissolver, and the Organic Waste Treatment. Aqueous effluent treatment was initially partially based on a precipitation process, precipitates being then mixed and solidified with bitumen. After a very successful start-up of the plant, performances were found to be better than their design value. A new aqueous effluents management was implemented, with the construction of several additional evaporators, leading to the process presented above, reducing the volume of waste and cutting radioactive releases of the plant. Corrosion issues have forced the replacement of several evaporators before their designed end of life. The operator (now Orano) was able to replace them, demonstrating its capacity to modify and retrofit those plants.

UP2-800 started operation in 1994 and was built to reprocess spent fuels from French nuclear power plants. Its process is almost identical to the UP3 process, although the plant layout is very different due to site constraints and start up strategy: UP3 was built from scratch, while UP2-800 facilities were replacing older, UP2-400 equivalents, over a 10-year period.

The two plants combined throughput reached 1700 t yr⁻¹ in 1996. Since then, and with many of its foreign customers reconsidering their nuclear policy or having decided to phase out of nuclear, production has fallen to about 1200 t yr⁻¹, most of it for the French utility EDF. Its design capacity would be difficult to reach again: the plant was initially designed to reprocess spent fuels with burn-up close to 33 GWd t⁻¹, currently reprocessed fuels are now at 45 GWd t⁻¹, with proportional amount of fission products and above proportional amount of minor actinides.

Japan

Rokkasho reprocessing plant in Japan uses technologies from France (UP3), the U.K. (THORP), and technologies developed in Japan. The successful UP3 design was used extensively, with the notable exceptions of the high-activity liquid waste concentration unit, based on THORP design, and the vitrification facility, based on Japanese technology developed in the Rokkasho reprocessing plant. Another significant difference is the management of iodine, released to the sea by other reprocessing plants. Instead, iodine is captured by solid sorbents. Started in 1993, its construction and commissioning suffered several delays. Active tests, using actual nuclear spent fuel, started in 2006. New safety standards enacted after the 2011 Fukushima accident have further delayed the start of operation, now expected in the first half of FY2022.

Cost of reprocessing

The cost of reprocessing and of nuclear waste final disposal is the subject of public debate trying to compare spent fuel management options, i.e. reprocessing and direct disposal. The purpose of this section is to present key findings and conclusion of the study published by OECD on this matter ([OECD, 2013](#)).

- Spent fuel management is a small fraction of the total cost of electricity (about 7 USD MWh⁻¹). However, the absolute cost can be large, depending on the size of the fleet.
- The difference of the total fuel cycle costs between the options considered are within uncertainties (see Fig. ES.1 p16 of the report).

Those results are sensitive to the discount rate accounted for in the estimate. Higher discount rates decrease the contribution of future expenditure, and tend to favor open cycles, with no upfront investment in a reprocessing plant.

With a relative neutrality of the economic argument, the report stresses the ultimate importance of qualitative parameters in the decision-making process, such as security of energy supply, non-proliferation, public attitudes, environmental effects, wastes streams, and perspective of development of advanced fuel cycles.

Poinssot (2014) compares environmental impact of open fuel cycle and mono recycling, and concludes that an open cycle has a bigger environmental impact, spending more natural uranium (+16%), and producing higher volume of high level radioactive waste.

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The Current Industrial Mono-Recycling of U and Pu: Recycling Step

Jean-Michel Marin, Recycling Business Unit, Orano, Paris, France

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Abbreviations

ADU Ammonium Di-Uranate
AGR Advanced Gas-cooled Reactor
AHWR Advanced Heavy Water Reactor
ALARA As Low As Reasonably Achievable
ASTM American Society for Testing and Materials
BWR Boiling Water Reactor
CANDU CANadian Deuterium-Uranium reactor
COCA CObroyage CAdarache
DU Depleted Uranium
EFPD Equivalent Full Power Day
ENU Enriched Natural Uranium
ERU Enriched Reprocessed Uranium
FA Fuel Assembly
FBR Fast Breeder Reactor
FP Fission Product
GCR Gas-Cooled Reactor
GWe Giga Watt electric
HEPA High Efficiency Particulate Airfilter
HEU Highly Enriched Uranium
HLW High Level Waste
IAEA International Atomic Energy Agency
LWR Light Water Reactor

MAGNOX MAGnesium Non-OXidizing gas cooled graphite reactor
 MOX Mixed Oxide
 MIMAS Micronization of a MASTer Blend
 NPP Nuclear Power Plant
 PCI Pellet Clad Interaction
 ppb Parts per billion
 PWR Pressurized Water Reactor
 RepU Reprocessed Uranium
 SBR Single Binderless Route
 TIG Tungsten Inert Gas
 UOX Uranium OXide fuel
 WNA World Nuclear Agency

The rationale and properties of MOX fuels

Rationale for recycling

Development and energy consumption are strongly associated. Energy experts estimate that the demand for primary energy in the world will increase approximately 65% by 2030, and the demand for electricity will double. Correspondingly, the progressive rise in average temperatures as well as Greenhouse gas emissions produced by human activity directly affect global warming and are specifically due to the burning of fossil fuels. Nuclear power is used to generate electricity for base load production free of all external constraints, and with zero greenhouse gas emissions. It is a definite advantage in the fight against global warming. Meeting energy needs, and environmental issues are key challenges and nuclear power constitutes part of the solution especially with its recycling concepts and designs. The objective of this article is to explain the industrial mono-recycling of Uranium and Plutonium in a closed fuel cycle. It is based upon a proven track record of recycling nuclear materials and has already achieved significant savings of earth resources to produce efficient and carbon-free energy. Recycling the energy from used fuel and efficiently managing the waste are key factors for sustainable growth of nuclear energy, bringing lower volumes and radio toxicity of waste, an enhanced proliferation resistance, uranium savings and economic benefits (Poinssot et al., 2016; Nuclear Energy Data 2007, 2007).

Why recycling?

Up to 96% of used nuclear fuel is recyclable. When unloaded from the reactor, the used nuclear fuel contains about 4% of fission products and minor actinides and an estimate of 96% of valuable materials (fissile and fertile materials), as demonstrated in detail in previous sections of the present document and shown in Fig. 1 below.

Recycling used nuclear fuel is interesting for recovering valuable materials, and it is equally a responsible solution for managing nuclear waste.

The purpose of recycling is thus twofold:

- 1) To recover energy from valuable materials as the spent fuel still contains U and Pu;
- 2) To separate the actual waste from valuable materials and to condition them into a safe and compact form suitable for transportation, storage, and final disposal.

The recycling principles

Recycling has been implemented in several countries for more than 45 years, with 40+ reactors using MOX (Mixed Oxide Fuel) fuel with an excellent return of experience. MOX fuel can be defined as the mixture of uranium and plutonium oxide fuel (Fig. 2).

Recycling of nuclear used fuel is a proven industrial process which allow the re-use of uranium and plutonium into new fuels: Enriched Reprocessed Uranium fuel (ERU fuel) and Mixed Oxide fuel (MOX fuel). At the end of the process, both recycled materials (either in the form of fresh fuel assemblies like MOX and ERU fuels or in the form of oxide powder ready to reuse in the case of uranium) and the conditioned waste are returned to the customer (Spent Fuel Reprocessing Options, 2009).

The benefits of recycling

Recycling offers immediate benefits to utilities. Recycling period and duration is a great benefit for utilities: between the unloading of used UO_2 fuel from the reactor and the loading of the fresh recycled fuel assemblies, the length of time is typically 15–20 years. Moreover, recycling globally allows a significant reduction of the quantity of spent fuel in storage pools. The volume reduction is as follows: 8 spent UO_2 fuel are used to produce one fresh MOX fuel, and one fresh ERU fuel through enrichment of Reprocessed Uranium (RepU). So, recycling offers an immediate and efficient mean for utilities to optimize their storage needs and avoid the accumulation of large quantities of spent fuel.

Spent fuel assembly



Fig. 1 Used fuel assembly and its recyclable material ratios.

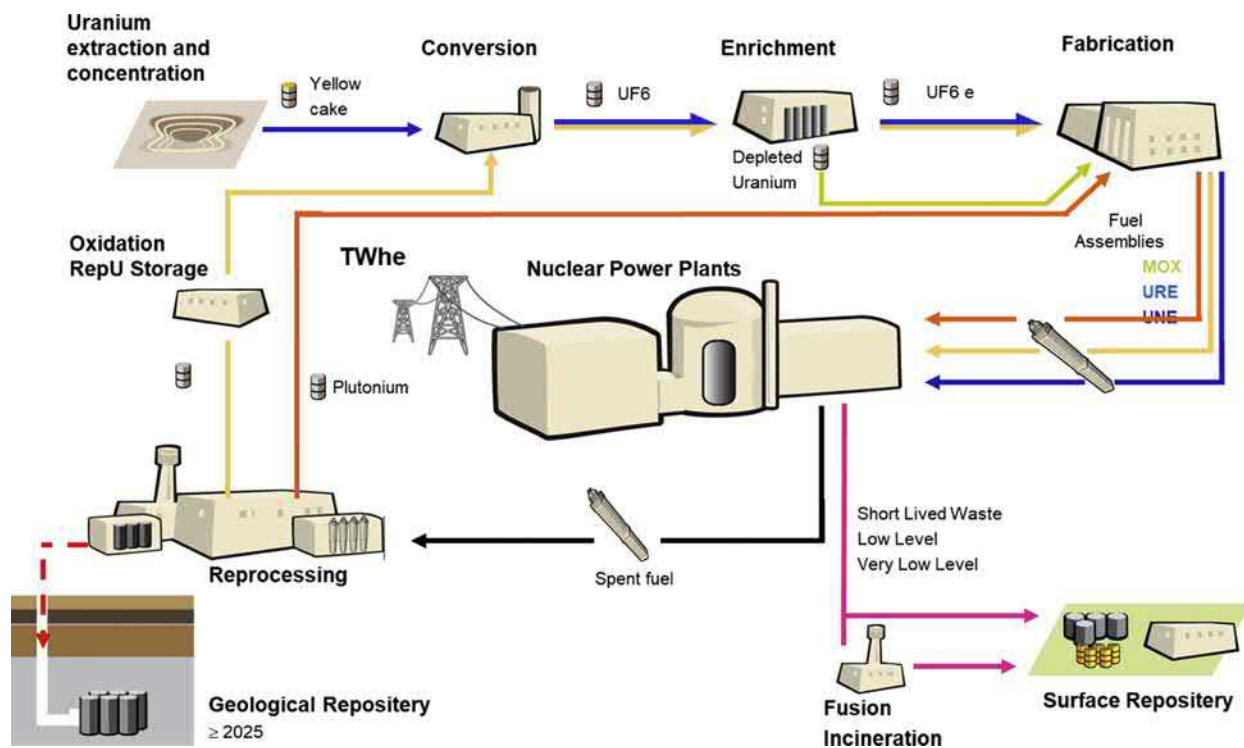


Fig. 2 Nuclear fuel closed cycle.

For example, in France without recycling the used fuel stockpile would reach more than 50,000 tons by 2030. With recycling, this stockpile has been stabilized to about 15,000 tons, and could be reduced further if needed, depending on the strategy chosen by the French utility (for instance, to prepare the feeding of future fast reactors).

Advantages of recycling

Recycling produces energy

Each MOX Fuel Assembly (FA) in current Light Water Reactors (LWRs) supplies about 50,000 Tons of Oil Equivalent (TOE) in energy. This corresponds to the electrical annual consumption of one European city of about 100,000 inhabitants.

Recycling saves resources

Recycling enables up to 25% in natural uranium savings. The comparison is based on the amount of enriched uranium which is needed to produce 100 TWh of electricity in closed (with recycling) and open cycle (without recycling). Without recycling, 280 tons of enriched uranium are necessary, while this quantity can be reduced to 215 tons with recycling, thanks to the additional power brought by using MOX and ERU fuels. Final waste is reduced to the minimum, in the closed cycle, only non-reusable materials are considered as actual waste, and are conditioned for final disposal (Fig. 3).

Recycling favors nuclear acceptance

Without a solution for the management of radioactive waste, the percentage of favorable opinions for nuclear energy is 44% but increases up to 62% with a solution for waste management. By offering a sound and efficient solution to saving earth's resources and waste management, recycling is clearly well-perceived by the public (Barbat and Liberge, 2013, pp. 78).

In addition, proliferation resistance is enhanced by recycling:

- continuous reuse of fissile material for light water reactor or fast reactor.
- no need of safeguarding HLW (they do not contain any significant quantities of plutonium).

Principles and industrial feedback of MOX fabrication

Principles of MOX fabrication

In enriched uranium, the fissile material is inherently present in the fuel. For MOX fuel, the fissile material which is plutonium must be added to the carrier material uranium. Precipitation of plutonium oxalate is the most widely used commercial process for obtaining plutonium dioxide for MOX fuel fabrication, and this technology has reached an advanced level of maturity. It is performed in the treatment facility which can be combined or separated from the proper MOX fuel facility. Three options for carrier material have been investigated for commercial plutonium recycling strategies: natural uranium (U-235 at 0.8%), depleted uranium (U-235 at 0.2%) or reprocessed uranium separated during the treatment of used fuels. Early designs were based on natural uranium. Reprocessed uranium has also been used in test MOX fuel assemblies demonstrating its general suitability as a carrier material. However, the isotopic composition of the reprocessed uranium is mainly dependent on the initial enrichment and the burnup of the used fuel. As a consequence, this will greatly complicate the design and plutonium content of the MOX fuel. Those two materials are not in use anymore and current MOX is fabricated mainly with tails depleted uranium as the carrier for economic reasons. Since MOX fuel assemblies are more expensive to fabricate than UOX uranium fuel assemblies, there are economic incentives to concentrate as much plutonium in the assemblies as possible. This blending of two fissile/fertile materials is the most specific difference between uranium and MOX fuel manufacturing. Plutonium has an energetic, radioactive behavior including alpha, beta and gamma radiations and it must be separated from the environment, the workers, and the public. All production operations must be conducted in strict confinement using glove-box technology and automated processes (Figs. 4 and 5).

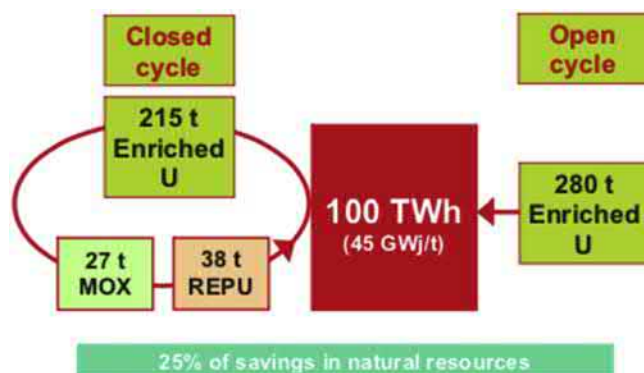


Fig. 3 Closed cycle savings of uranium due to MOX and RepU use.



Fig. 4 Example of a simple glove-box.

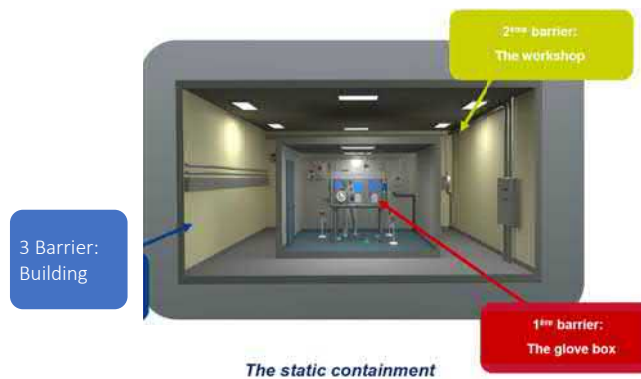


Fig. 5 Three barriers concept.

The first fabrication processes appeared in Germany, France, USA, UK and Belgium in dedicated fabrication plants. Each plant was characterized by specific processing approaches, and influenced by their design basis as well as their licensing limits:

- Isotopic composition of plutonium allowed
- Maximum americium content
- Maximum plutonium content in the MOX fuel
- Maximum allowed personnel exposure
- Confinement rules
- Criticality approach

These parameters greatly influence the equipment adopted, the plant size, and the plant layout.

In USA a few pilot fabrications started in the early 1960s but never attained industrial status.

The first fabrication plant in Germany was in Hanau for the SIEMENS company it operated for FBR and PWR fuels but was shut-down prematurely in the early 1990s by the green party.

In Belgium the plant was operated by BELGONUCLEAIRE in Dessel and developed the first two-steps process.

In France the plant named COGEMA Cadarache was the largest FBR Fuel plant, it then enters in transition to PWR fuels in the 1991 until it stopped of operation after 2010 to be replaced by the more industrial MELOX plant.

In the UK a plant named MDF (MOX Demonstration Facility) was a predecessor to a bigger facility named SMP (Sellafield MOX Plant); both facilities were shut down in 2011.

The conventional fabrication route is a direct application of the most common industrial process used for uranium fuel, the enrichment of uranium being replaced by a mechanical blending of the feed powders: UO_2 and PuO_2 and $(\text{U}, \text{Pu}) \text{O}_2$ for the scrap recycle material.

Three main processes were used:

In the 1970s the French CEA developed the COCA process in its Cadarache facility. It is an acronym for Cobroyage (co-milling) Cadarache and it is based on an optimized ball mill acting as a blender. The powder is then forced through a calibrated sieve to produce a free flowing granulate, adequate for feeding the rotary pellet press. This process was originally developed for Fast Breeder Reactor fuel. It eventually became adapted for PWR MOX fuel.

During the 1980s in the UK, BNFL developed a process named Short Benderless Route (SBR), which is based on the application of alternative process equipment for blending and granulation functions. The ball mill is replaced by an attritor adapted for providing a short processing time. The precompaction granulation is replaced by a spheroidizer equipment working on a powder agglomeration principle.

Today's most common process is the MIMAS (MIconization of a MASTer blend). Invented in the early 1980s, it is an adaptation of earlier fabrication processes that involved a single blending step.

Furthermore, the process evolved into a two blending step process. This evolution occurred when industrial production started to ramp up. The process developed due to the increase of specification demands from the fuel designers, fuel users in the Nuclear Power Plants, and even treatment facilities of spent fuels (Fig. 6).

In the first step, the pure PuO_2 feed and some UO_2 are co-micronized resulting in a master-mix of UO_2 - (typically 30% PuO_2). In the second step, the master mix is blended down with free flowing UO_2 to the specified content of the MOX fuel. Pelletizing is carried out in hydraulic presses with multiple dyes for increased productivity. Sintering is most commonly conducted in continuous electric powered furnaces. The pellets are arranged in large Molybdenum boats and then pushed sequentially inside the furnaces where their temperature will ramp-up to 1750°C after a 24-h cycle in a specific atmosphere. The very close contact between the micronized UO_2 and PuO_2 particles provides for adequate interdiffusion of cations during sintering and therefore the achievement of the required level of homogeneity. Centerless grinding is then performed in dry conditions to adjust very precisely both diameter and perfect cylinder shape of the pellets. Following a visual and dimensional inspection, all conforming pellets are placed on specific trays in order to weigh them. Then the preparation of pre-measured lengths of the pellet columns are ready to be introduced in zircaloy tubes named rods. Pellet samples are randomly sent for various analyses in laboratories for Quality Control (QC).

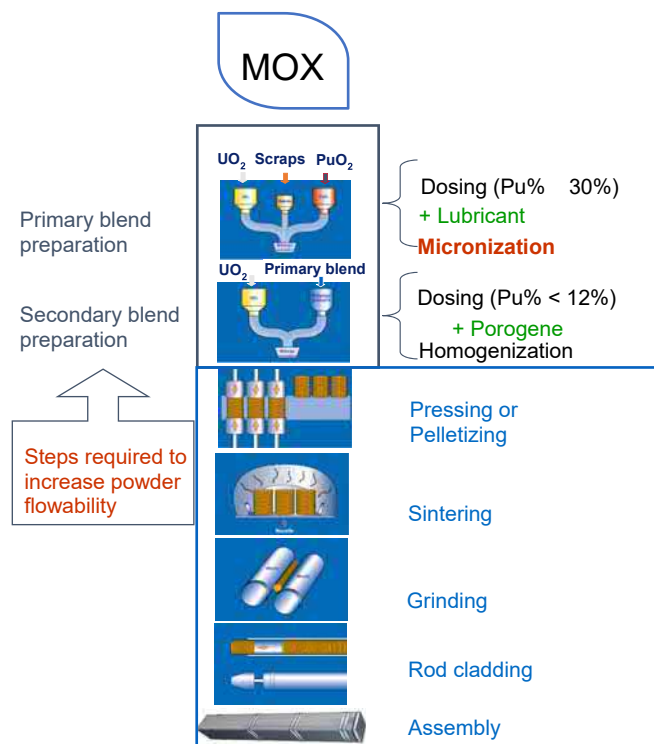


Fig. 6 MOX process using a two-step blending.

Tungsten Inert Gas (TIG) welding is commonly adopted for both the circular weld and the seal weld of the upper end plug of the rod. The seal weld is closed after the introduction of a pressurized helium gas into the rods. After their fabrication, the MOX fuel rods are inspected and tested for their integrity, the quality of their welds, their leak-tightness and overall conformity to the required specifications. Then later being sent to a rod storage, fuel rods of different and adequate Pu contents will be positioned in magazines with the same lattice of the Fuel Assembly (FA). The rods will then be drawn automatically and layer by layer pulled from the magazines through the Fuel Assembly skeleton. The Fuel Assembly (FA) manufacturing and Quality Control (QC) are fully automated and are identical to the UOX Fuel Assembly manufacturing. After storage, to constitute a reload of a given campaign the MOX FAs are handled and placed in dedicated and specific shipping containers before their transportation to the nuclear reactor. MOX FAs can be transported by ship, railway or road to reach their destination (Garwin et al., 1994).

Industrial feedback of MOX fabrication

The technologies of recycling and fuel fabrication have been adapted for plutonium recycling for more than 45 years. Since, plutonium recycling has been focusing on the stabilization of the separated plutonium inventory. Currently, the use of MOX fuel has been established on an industrial scale in a number of countries. In Germany, Belgium, France, Netherlands, Switzerland, and Japan a considerable number of thermal power reactors, both pressurized water reactors (PWRs) and boiling water reactors (BWRs), are either licensed or a license has been applied for. A license is required in order to use MOX fuels at levels up to 30% or more of the reactor core.

The number of "MOXed" reactors is the following:

The "MOXed" reactors represent 10% of the world's nuclear reactors. In Germany, the first loading of MOX fuel was in 1972, in Switzerland it was in 1984, in France it occurred in 1987, then Belgium and Japan in 1995 and finally in the Netherlands in 2014 (Fig. 7).

Ultimately, the industrial quality has been achieved and the performances of MOX fuel are equivalent to the performances of UOX. The MOX fuels have reached high burnup equivalent to UOX fuels (> 60 GWd/tHM) notably in Germany and Switzerland. The UK is investigating the incorporation of its 120 tons of reactor-grade plutonium into MOX fuel which would be used in LWR or CANDU reactors.

In the mid-1990s for disarmament purposes, plutonium from nuclear weapons declared in excess by USA and Russia was also considered for MOX fuel. Weapons-grade plutonium has over 93% of the fissile isotope, Pu-239, and can be used as well as a reactor-grade Pu, in fuel for electricity production.

Options considered for the disposal of weapons-grade plutonium included:

- Immobilization with high-level waste - treating plutonium as waste,
- fabrication with uranium oxide as a MOX fuel for burning in existing reactors,
- fabrication with thorium for existing reactors,
- fuel for fast-neutron reactors.

These considered options are now stalled or in stand-by.

The MOX fuels have presented a high reliability as they have shown no measurable impact on reactor operations (no early outage, no reactor shutdown, no material dissemination into primary water) for 45 years. The impact on the operation of the reactor caused by the very limited number of rod leakages is the same for MOX as for UOX and does not impose new constraints. The safety authorities in the countries operating MOX fuel in commercial reactors have determined that it is safe. The safety authorities have concluded that it meets the highest safety standards and the same rigorous licensing requirements as UOX fuels.

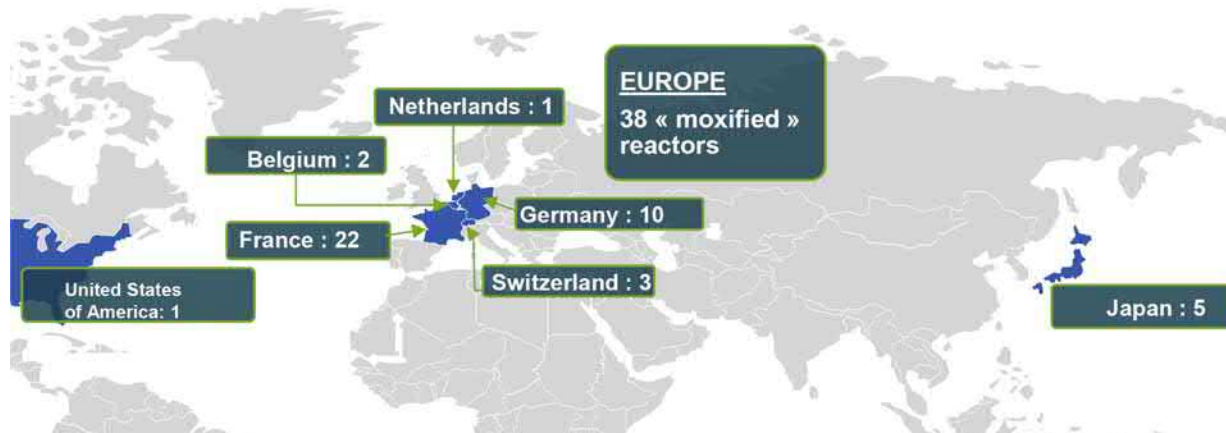


Fig. 7 Reactors having loaded MOX reloads.

MOX is also used in fast neutron reactors in several countries, particularly France and Russia. It was first developed for this recycling, with research and design being done in USA, Russia, UK, France, Germany, Belgium and Japan. Today, Russia has long-term plans to build a new generation of fast reactors fueled by MOX for the BN-800 fast reactor which started up at Beloyarsk, in mid-2014. France also has a long industrial experience with fuel for its PHENIX and SUPERPHENIX reactors, and may reconsider Fast Breeder Reactor (FBR) in the future.

The U recycling: Specificity of fuel fabrication from RepU

RepU Re-use and fabrication feedback

After the treatment of used nuclear fuel, there is the opportunity to fully close the fuel cycle by using the 95% remaining material in the form of reprocessed uranium or RepU. Utilities performing the reprocessing of used fuels are facing a growing RepU inventory, raising the risk of demonstrating an exit for this material. After fission products and plutonium (Pu) have been removed from spent LWR fuel, RepU remains.

A fissile content in the RepU of 235-U of 0.9% wt. to 1.1% wt., two possibilities are available: either direct use in certain type of reactors using natural uranium as the CANDU reactors or re-enrichment for reactors using enriched uranium as their fuel. It is to be noted that China is planning to use RepU in CANDU reactors that could be strictly devoted to this objective ([Use of Reprocessed Uranium, 2010](#)).

RepU in CANDU reactors

These reactors have a sufficiently high neutron economy to use RepU as fuel. RepU from spent LWR fuel can be considered as a lower cost source of enrichment at the optimal enrichment level for CANDU fuel pellets. In Europe the feedstock of RepU is approaching several thousand tons, and would provide enough fuel for numerous CANDU reactors for many years of operation. The use of RepU fuel offers significant benefits to CANDU reactor operators. RepU fuel improves the fuel cycle economics by increasing the fuel burnup, which enables large cost reductions in fuel consumption and in spent fuel disposal. RepU fuel offers enhanced operating margins that can be applied to increase the reactor power. These benefits can be put in place using existing fuel production technologies and practices. Additionally, these benefits can be established with almost negligible changes to fuel receipt and handling procedures at the reactor site. The application of RepU fuel could be an important element in many NPPs around the world. For example, China is contemplating the recycling of RepU on an industrial scale in the nearby future.

ERU in AGR and LWR

Starting in 1976 when the price of natural uranium was high, reprocessed uranium (RepU) recovered from Magnox fuel was re-enriched and fabricated into fuel for the British Advanced Gas-cooled Reactors (AGR). Over 15,000 tons of Magnox RepU was recycled to produce over 1500 tons of enriched reprocessed uranium (ERU) fuel for AGRs. The recycling ceased in 1996 as it became economically unattractive due to a combination of factors, such as low U-235 content of Magnox RepU (with increasing discharge burnups) and falling natural uranium prices.

Utilization of ERU in Light Water Reactors (LWR) began in the German Pressurized Water Reactor (PWR) Obrigheim in 1983, which was then followed in the late 1980s by Japan, France and Belgium. If not utilized, RepU constitutes a liability consisting in the costs of conditioning, interim storage and final disposal as a waste. As a result, in countries with political uncertainty about the future of nuclear power such as Belgium, Switzerland, Germany and the Netherlands, the utilities did recycle their ERU in the LWRs. Countries with a long-term policy of nuclear power generation, like France and Japan, store their RepU as a fissile material reserve. Recycling some of it as ERU is only applied on a reduced scale to ensure the availability and maintenance of the technology. For example in France, for many years, Electricité de France (EDF) has loaded ERU fuels in four 900 MWe reactor units at Cruas. France has a multi-decade experience in producing, managing and using ERU Fuel Assemblies (FA). The first implementation of reprocessed uranium (ERU) reconversion and fuel fabrication was achieved in the Romans, France facility in 1987.

Between 1987 and 2000 for each fabrication campaign, up to four « 900MWe » reloads per year, 20 tons of ERU each were produced. Around 1200 URE Fuel Assemblies have been loaded in 900 MWe Cruas reactors, leading to the saving of 4500 tons of natural uranium. The lessons learned from reactor operations are excellent and ERU fuel behavior was identical to natural uranium. The ERU recycling stopped in 2013, but will start again in 2023. EDF is evaluating the capacity of the industry to re-start ERU FA manufacturing for its 900 MWe fleet as well as for its 1300 MWe reactors. EDF will elaborate its schedule to manage all the needed upgrades and investments in the facilities for the industrial recycling of ERU. Although the experience with RepU and ERU from oxide fuel has been limited, there have been successful demonstration campaigns which will result in an increase in recycling in the future.

Fabrication challenges from RepU to ERU

The RepU contains only 1% wt. of U-235 and thus needs to be re-enriched in a dedicated facility to become Enriched Reprocessed Uranium (ERU) with an enrichment level similar of the Enriched Natural Uranium (ENU) close to 5%wt. First, the separated uranium is purified and concentrated, then the RepU is stockpiled in the form of stable oxides: U_3O_8 or UO_3 .

The RepU specificities are due to its isotopic composition:

- The level of U-235 is around 1%wt.
- Presence of U-232 (some ppb): its radioactive decay leads to gamma emitters with a risk of external exposure requiring biological protection or specific operating procedures
- Presence of U-234: alpha emitter and neutron absorber
- Presence of U-236: neutron absorber but no influence upon radioactivity emission

The fabrication route is as follows: The RepU in its oxide form must be converted to a gaseous state in hexafluoride of uranium: UF_6 . The process is fundamentally the same as for Unat, with the exception that the radiation shielding must be added, and the process must be operated faster. Thus, leading to significant over-cost. The enrichment facilities are designed to process UF_6 and can adjust to the UF_6 from RepU with few modifications. *Re*-enrichment of RepU will create 7/8 of depleted RepU with no possibility to be recycled in the current nuclear fuel cycles. After enrichment to the targeted U-235%wt. content the UF_6 must be reconverted into an oxide form in a reversion facility. The daughter products of the U-232 remain in the waste stream after the last gas vaporization. It authorizes the reduction of dose rate exposure and facilitates the handling and transportation of containers with RepU solid heels. The freshly produced UO_2 powder is therefore free from these daughter products. However, from the date of vaporization, the relatively fast decrease of the U-232 leads to the reappearance of those daughter products in growing quantities. As a result of this process, the product is named Enriched Reprocessed Uranium or ERU.

Fuel fabrication plant

The beta emission of U-232 is about 10% higher for the ERU than for the ENU. Also, the daughter products of U-232 will emit harmful gamma rays. Usually, fuel plants are limited to a U-232 of less than 15 ppb; and therefore need special licenses, specific environment requirements and protective equipment for the workers to operate with U-232 of around 20 ppb or more. Capital Expenditures (CapEx) for protective equipment and production management are needed as well as an increase in Operational Expenditures (OpEx). The fuel fabrication plant needs to be able to master the speed of the process, and the manufacturing steps to decrease the ageing of U-232 (and daughter products). The fuel fabrication plant must also be able to manage ERU FA manufacturing in parallel with ENU FA manufacturing. To avoid gamma dose rate increase, it is also key to limit the time the ERU spends in the installation. In order to avoid ageing, it is essential to supervise the time the material spends in storage and to detect zones prone to accumulation.

Properties of MOX fuels

The neutronic design of MOX assemblies is based on the simple principle that the assemblies must be equivalent to assemblies containing only uranium fuels in terms of power generated, overall reactivity and accumulated burnup. The MOX fuel must meet all safety criteria in manufacturing, transportation, handling, storage and reactor operation. It must also meet all safety criteria, concerning the unloading and later reprocessing of the used elements. As the current operation in reactor is mainly made in hybrid cores (i.e. MOX and UOX fuel together), it must provide no impact on the UOX fuel assembly design. For good handling and compatibility, the mechanical structure is identical to UOX assemblies thus enabling no impact on reactor design and limited impact on core operation. An additional goal for MOX fuel is to reach the same level of recyclability and used fuel management to meet equivalent economic criteria.

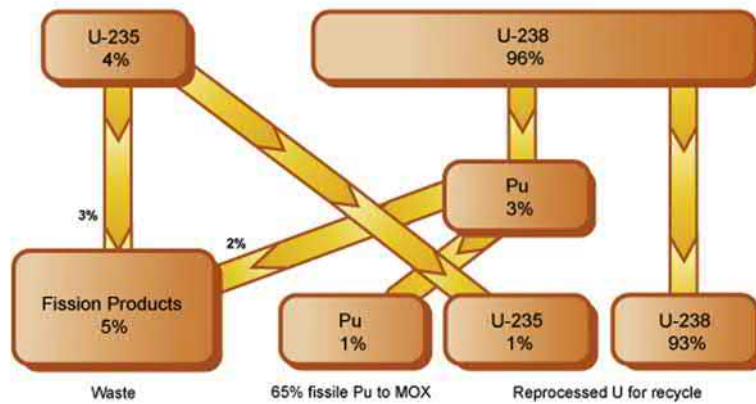
To fully understand the properties of MOX fuels, we must compare them with UOX fuels. During the in-core evolution, the fuel undergoes physical and chemical transformations related to the fission reactions and high temperatures endured. MOX and UOX fuels behave in slightly different ways which consequently influences performance and key safety parameters ([Status and Advances in MOX Fuel Technology, 2003](#)).

Isotopic evolution

The MOX fuel contains plutonium fissile isotopes; Pu-239 and Pu-241 generate more fissions than the U-235 present in the UOX fuel. However, Pu-240 is also present in the fresh MOX fuel; it features a very high neutron capture cross-section, higher than the cross-section of U-238 present in UOX fuel. During the in-core operation, the evolution is that Pu-240 inventory is decreasing and thus its related capture properties also decreases with the burn-up. This results in an increase through absorption of the Pu-241 inventory and an increase in additional fissions. It is also to be mentioned that during irradiation Pu-239 is generated in both MOX and UOX fuel from neutron capture by U-238 and is consequently contributing to additional fissions along the reactor operation ([Figs. 8 and 9](#)).

Reactivity

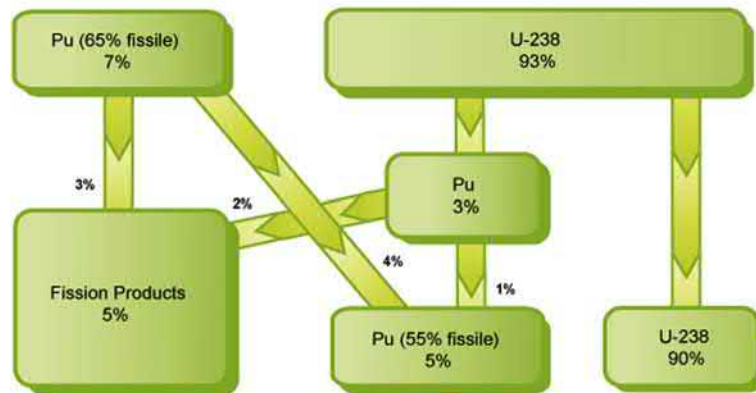
The plutonium composition has an impact on time-dependent neutronic behavior of the MOX fuel. The reactivity of MOX fuel decreases more slowly with burnup than that of UOX fuel. At a higher burnup, MOX fuel generates more power than UOX and experiences a higher transient power for the same increase in neutron flux than equivalent UOX. The MOX fuel presents a harder neutron spectrum based on its energy distribution. For their use in hybrid cores, the design of MOX assemblies has therefore been

Reaction in Standard UO_2 Fuel

Basis: 45,000 MWd/t burn-up, ignores minor actinides

Fig. 8 Isotopic evolution in UOX fuel.

Reaction in MOX Fuel



Basis: 45,000 MWd/t burn-up, ignores minor actinides

Fig. 9 Isotopic evolution in MOX fuel.

optimized to reduce the power peaking at the uranium/MOX interface and as a result has been designed to obtain the flattest possible power distribution (Fig. 10).

This is achieved by zoning the assembly, using three plutonium enrichments (or content zones) in a concentric distribution as shown in Fig. 4 for a PWR fuel assembly.

Thermal behavior

Due to the difference in the pellet composition and structure, MOX fuel possesses different thermal conductivity than UOX. At lower temperatures (below 1200 °C), the thermal conductivity is about 10% lower than of UOX. This leads to different temperature profiles within the fuel and a higher central line temperature for MOX at an equivalent power level. At higher temperatures, the difference reduces. Another thermal characteristic is the melting temperature of the pellets. MOX and UOX melting temperatures are equivalent (higher than 2750 °C) and far above the maximum pellet temperature in the core during normal operating conditions (typically maximum is 800 °C). There are also differences between MOX and UOX in terms of decay heat released after discharge; this is mainly due to the difference in actinides inventory. Just after the end of the chain reaction, decay heat is dominated by delayed neutron fission (which inventory is lower in the case of MOX fuel) and the contribution of short-lived Fission Products (FP). In the hours after shutdown, the decay heat released by MOX is lower than the decay heat released by UOX. Although in the long term, the MOX will still generate more heat due to a difference in actinides inventories resulting in relatively slow decay in comparison to Fission Products decay. The decay heat is dominated in the very long term by Am-241, Pu-239 and Pu-240 for which the inventories are larger in the case of MOX (Figs. 11 and 12).

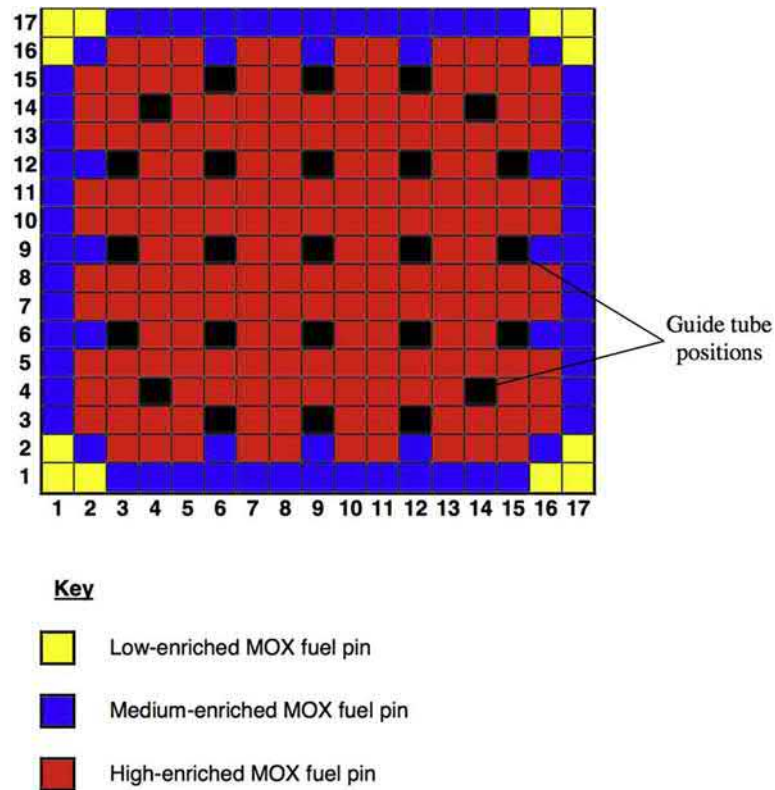


Fig. 10 MOX FA example of plutonium composition: lower Pu fuel rods: 3.7 wt.% Pu, Medium Pu fuel rods: 5.2 wt. % Pu, higher Pu fuel rods: 8.2 wt. % Pu.

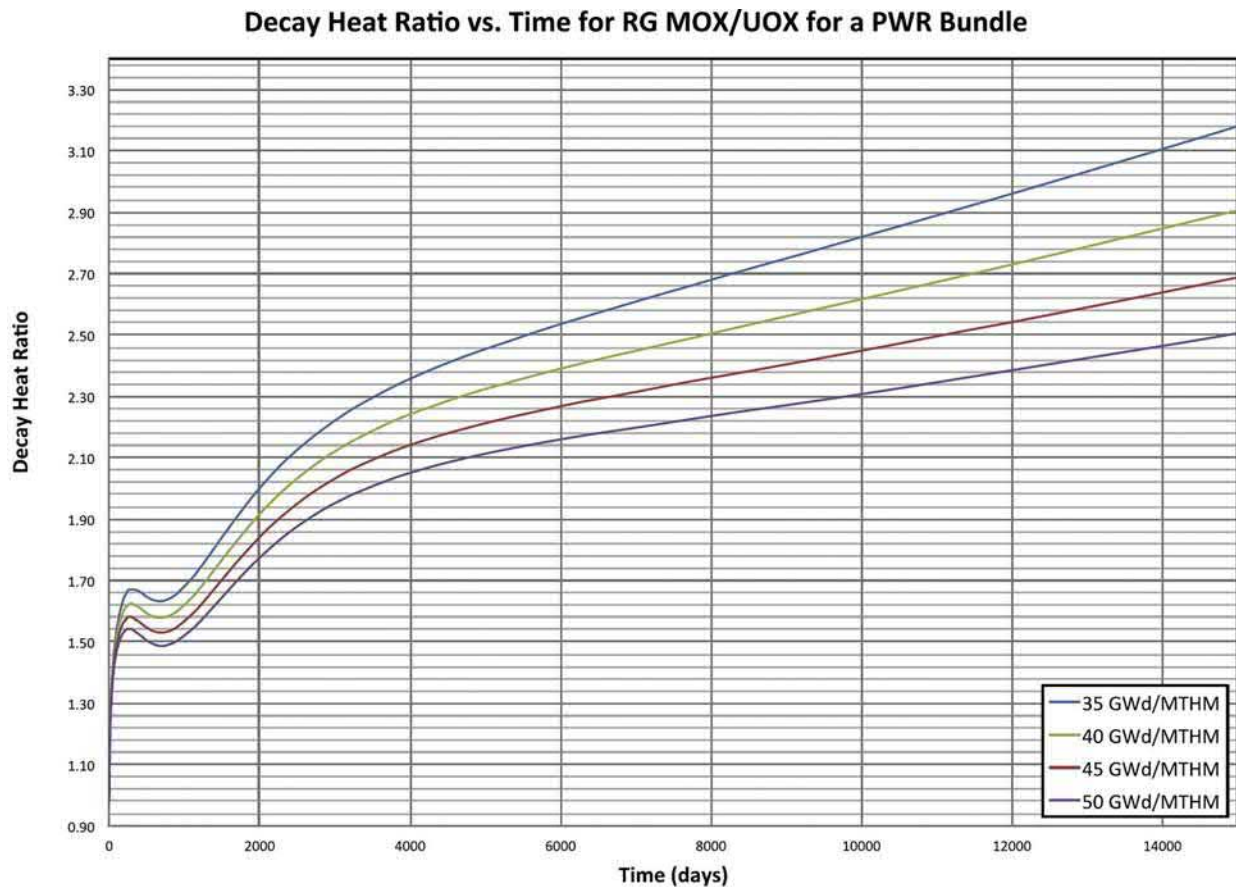


Fig. 11 Comparison of fuel temperatures in relation to burnup.

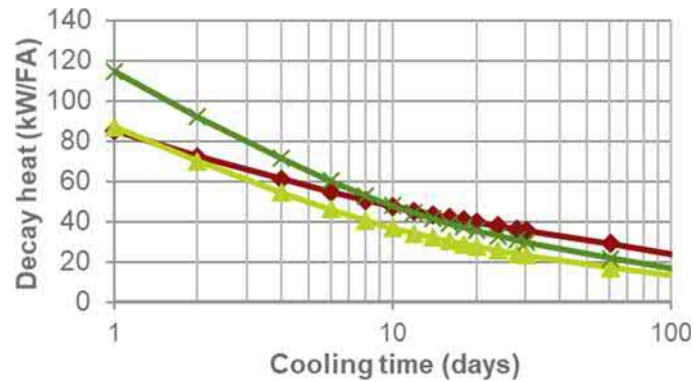


Fig. 12 Heat decay of different types of fuel.

Physical and chemical properties

For both fuels during the in-core evolution, the pellets will undergo physical and chemical transformations relating to the prevailing high temperatures and fission reactions.

The following phenomena will occur:

- **Densification:** it will happen during the early stages of reactor irradiation, caused by disappearance of submicron pores. The densification behavior of MOX fuel is comparable to that of UOX fuel.
- **Swelling:** After the initial densification, a new phase will occur, the swelling due to the accumulation of Fission Products. The gaseous swelling kinetic is slightly different in MOX fuel than in UOX because of the difference in fission gas release of the two fuels.
- **Fuel Creep:** Creeping is the process that causes the cladding material under constant load to deform progressively. The creep rate in fuel is dependent on several parameters including temperature, fission rate, grain size... Accordingly, MOX and UOX feature different behaviors regarding creeping. Higher creep rates in MOX fuel, enables the relaxing of the cladding stress more quickly.
- **Pellet Cladding Interaction (PCI):** It is a complex phenomenon that tends to occur under power ramping conditions in the reactor. The expansion of the fuel pellets due to high internal temperatures, cracking due to thermal stress, and irradiation induced-swelling may lead to mechanical interaction of the fuel with the cladding. In extreme conditions, PCI may lead to cladding failure and disturb the reactor operation. The MOX fuel presents better results in PCI resistance due to enhanced creep rates of MOX as compared to UOX.
- **Fission Gas Release:** The Fission Products resulting from the fission of uranium or plutonium are the same regarding their type, however they differ in their repartition. The non-volatile are embedded in the pellets and are equivalent for both fuels. The volatile ones are in equivalent production quantity but of different types and it is important as they are the major contributor in case of the very unlikely event of an accidental release. In regard to the noble gases, the contribution for MOX fuel is higher toward the higher burnup and leads a higher internal pressure. The MOX fuel rod design integrates this specificity of higher internal pressure by a bigger plenum volume inside the rods to safely accommodate the gas release.

In conclusion, the properties and differences in fuel performances between MOX and UOX may lead to:

- higher MOX centerline temperatures,
- specific microstructures in the Pu-rich agglomerates (Fig. 13)
- greater fission gas release and,
- higher rod internal pressure.

These differences are relatively minor and manageable, falling within fuel design margins. Furthermore, MOX fuel features a higher potential for energy production due to higher production of neutrons from fissions in one generation (Figs. 14 and 15).

Overview of the overall cost associated to U/Pu recycling in MOX fuels

The overall cost of current closed cycle (interim storage, recycling, transportation, waste disposal) is about 6% of the cost of producing electricity (Barbat and Liberge, 2013, pp. 70).

The cost covers transportation of used fuel from reactor storage, reprocessing of used fuel, treatment and disposal of HLW resulting from reprocessing, and construction-operation-decommissioning of the plant. It incorporates saving up to 25% of natural uranium equivalent. Due to the recycling of the spent fuel, the front-end costs for utilities are lower thanks to the recovering of uranium and plutonium. Recycled fuel credits offset up to 40% of recycling costs, which are now comparable to direct disposal costs (Barbat and Liberge, 2013, pp. 76). Recycling also reduces financial uncertainties in the back-end by relying on industrial facilities

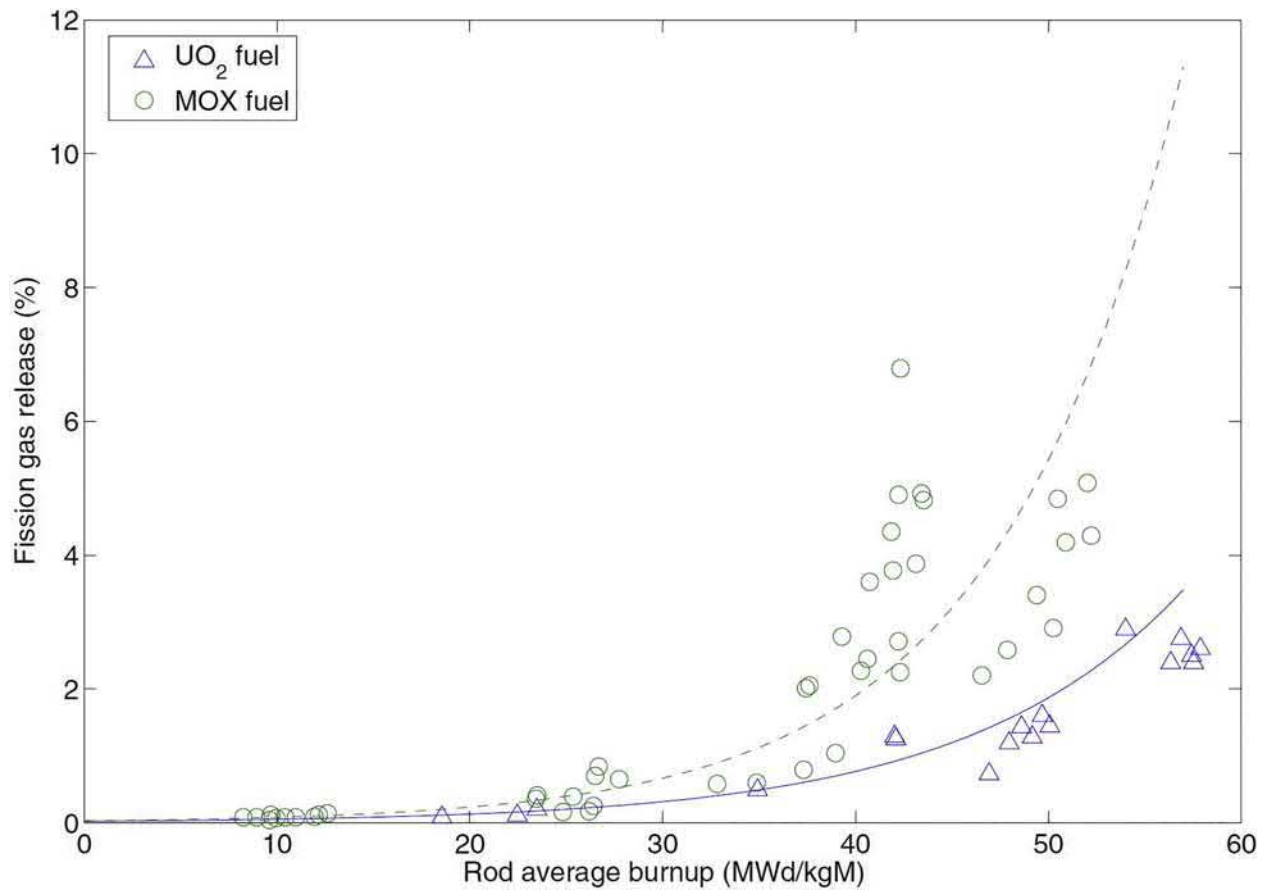


Fig. 13 Comparison of fission gas release for MOX and UOX fuel.

with proven track records, recycling costs, as well as transportation costs. These costs are fully controlled. In closed fuel cycle, only waste disposal presents a certain level of financial uncertainty as no geological disposal for high-level waste (HLW) is already in operation worldwide.

However, in case of closed cycle these disposal costs are reduced compared to open cycle for several reasons:

- Due to volume reduction there is less waste to be disposed of,
- thermal power of recycling waste decreases much faster than thermal power of spent fuel,
- HLW from recycling can be put in interim storage with safe and durable conditions,
- the need for final disposal can be delayed and HLW is more compact than used fuel in the final disposal.

On the contrary, the majority of direct disposal costs (encapsulation and geological disposal) are unproven leading to large uncertainties associated with the spent fuel management route. Recycling is a competitive solution which reduces financial uncertainties compared to direct disposal, in Geological Disposal Facilities (GDF), in which actual costs will only be revealed in the future.

Mixed Oxide (MOX) and Enriched Reprocessed Uranium (ERU) are respectively competitive.

Cost associated to MOX fuels

The respective competitiveness of Mixed Oxide (MOX) and Enriched Reprocessed Uranium (ERU) will be explored further in the sub-chapter below.

MOX fuel costs are not affected by the volatility of the cost of natural uranium supply. MOX fuel represents a secured source of supplies with a very high stability and predictability rate. This is a unique feature for utilities. To support the general, long-term trend (higher fuel burnup) an increase of uranium consumption and enrichment is required in the case of UO_2 fuel leading to cost augmentation. In the case of MOX fuel, the increase of plutonium content is enough to sustain the rise of fuel burnup which has a small impact on the fuel cost. Currently, MOX fuel fabrication is more expensive than UO_2 fuel fabrication. This is partly due to the specific safety and radiation protection measures required in MOX fuel fabrication plants for the safe handling of plutonium. The main reasons for the high cost of MOX fuel fabrication are the small sizes of the facilities and the relatively early stage of development of the recycling industry.

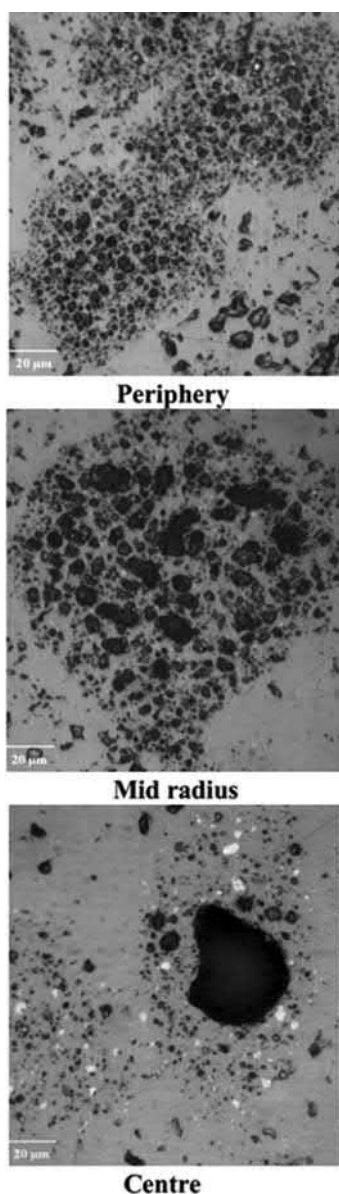


Fig. 14 MOX microstructure in different pellet areas.

The overall cost of production numbers per unit are:

- \$400/kgU for UOX LWR
- \$1000/kgHM for MOX LWR

Reference costs from international sources based on reference capacity in dollars (\$) of the year, 2017 ([Advanced Fuel Cycle Cost Basis, 2017](#)).

Costs associated to RepU and ERU fuels

Reprocessed uranium can be used as a direct substitution in fuel fabrication. Reprocessed uranium's main advantage is the cost predictability due to the absence of price volatility in contrast to natural uranium prices. If natural uranium prices remain high and if suitable enrichment capacities are available, a policy of sustained single cycle of RepU could reduce primary uranium demand by 15% or more. It will also give NPPs a way to diversify their supplies.

For CANDU fuel, the use of RepU can provide a direct economical saving as demonstrated in the [Table 1](#).

For ERU fuel, as the isotopic composition brings more constraints, the overall cost of fuel manufacturing is higher than normal ENU fuel.

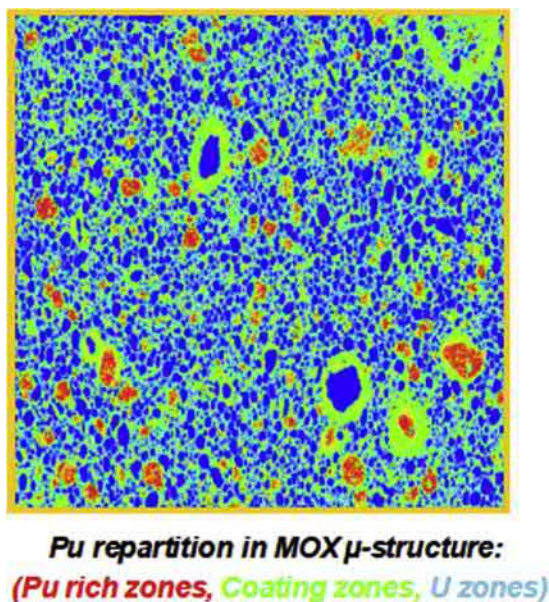


Fig. 15 Pu repartition illustrating heterogeneous structure.

Table 1 Price comparison using RepU.

<i>Economics of the Use of LWR Reprocessed U in CANDU Reactors</i>		
<i>Natural U CANDU fuel from uranium ore:</i>		
Uranium ore price (English)	89.6	\$/lb U308
Uranium mine & mill price (Metric) (as if U308 produced)	233	\$/kgU
Canadian conv of U-mill solutions to pure reactor-grade UO ₂	10	\$/kgU
CANDU fuel fabrication price (from UO ₂ powder)	100	\$/kgU
Total cost	343	\$/kgU
CANDU FUEL FROM LWR REPROCESSED U^a		
Dissolution of REPU308, cleanup of sol'n, and conversion to UO ₂ of right powder morphology	40	\$/kgU
CANDU fuel fab price (adj for higher handling risk)	130	\$/kgU
Total cost	170	\$/kgU

^aNo enrichment step assumed. ~ 0.7%–0.9% U-235 in REPU; some U-236 present.

The cost numbers coming from international sources: (Shropshire et al., 2006).

- \$400/kgU for UOX ENU LWR
- \$435/kgU for UOX ERU LWR

Reference costs based on capacity in dollars (\$) of the year, 2017 (Shropshire et al., 2006).

The current industrial mono-recycling of U and Pu

In conclusion, recycling of recycled uranium and plutonium in MOX fuel has demonstrated its competitiveness throughout the world for over 45 years. Recycling provides a long-term source of fuel supply, independently from uranium markets volatility. MOX fuel and RepU fuel are based on the simple principle that they must be equivalent to natural uranium fuels for power generated, overall reactivity and accumulated burnup. The MOX fuel have met all safety criteria in manufacturing, transport and reactor operation. From the financial point of view, recycling reduces uncertainties and is based on proven industrial technologies.

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Routes for Pu Multi-recycling

Bernard Boullis^a and Christine Chabert^b, ^aCEA, Saclay, France; and ^bCEA-Cadarache, Saint-Paul-lez-Durance, France

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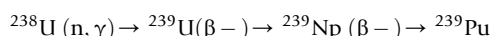
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Goals of Pu multi-recycling

Plutonium formation and amount in used fuels

Plutonium in LWR enriched uranium classical fuel is formed in the reactor core by neutron capture reactions on ^{238}U , which is “not fissile” at low neutron energy (“thermal spectrum,” which prevails in light water reactors cores), but is able to be transformed into ^{239}U . This uranium isotope is unstable and, by β -successive decays, is spontaneously transformed (decay half-time: 23.5 min) into ^{239}Np then (decay half-time: 2.35 days) into ^{239}Pu .

The global mechanism is the following:



From ^{239}Pu many different events can then occur: either fission (^{239}Pu is fissile in thermal spectrum) or successive capture reactions, resulting in higher isotopes successive formation (^{240}Pu , ^{241}Pu , ^{242}Pu mainly). And each of them can then be transformed, either by fission (for the odd isotopes) or radioactive decay α decay with long radioactive periods, but also β -decay leading to trans-plutonian elements (Am, Cm mainly).

^{238}Pu can also be formed: either by capture reactions on ^{235}U and ^{236}U (leading to ^{237}U , then ^{237}Np which can evolve toward ^{238}Np the ^{238}Pu), or by spontaneous α decay from ^{241}Am , previously formed by mechanisms mentioned just above.

The main radioactive periods and absorption and fission cross-sections of uranium and plutonium isotopes are reminded in Table 1 and Fig. 1 hereafter.

The respective amount of plutonium isotopes in typical enriched uranium oxide used fuels are presented in the following Table 2.

It must be noticed that the plutonium amount in used fuel reaches an equilibrium value in the core of the LWR, due to antagonistic mechanisms: on the one hand, Pu is formed through capture on ^{238}U , and on the other hand, it is eliminated by fission, capture or decay. The global amount of plutonium in LWR used fuel is thus about 1% (mass fraction), almost not depending on the residence time of the fuel in the reactor core.

The duality of plutonium: Asset and hazards

- *Residual energy value:* Plutonium odd isotopes (^{239}Pu , ^{241}Pu) are fissile in thermal neutron spectra, and a large part of the formed Pu is fissioned before fuel unloading (about 30% of global recovered energy from the fuel is due to in situ formed plutonium). Plutonium in used fuel represents about 1% of the total mass of initial uranium, and more than half of the Pu isotopes are fissile/odd isotopes. This is why plutonium amount in used fuel can be considered as an asset, for fissile potential of it is about 20% of initial ^{235}U fissile potential.

Table 1 Radioactive decay half-life of uranium and plutonium isotopes.

Isotope	^{234}U	^{235}U	^{236}U	^{238}U	
$T_{1/2}$ (years)	2.454×10^5	7.037×10^8	2.342×10^7	4.468×10^9	
Isotope	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
$T_{1/2}$ (years)	87.71	24.1321×10^3	6.569×10^3	13.355	3.76×10^5

- *Plutonium potential radiotoxicity*: plutonium isotopes are heavy nuclei, which all belong to long decay chains with α -active radionuclides, a large part of them being long-lived. This leads plutonium isotopes to be the main source of long term radiotoxicity of used fuels: the plutonium share in the total radiotoxicity of used fuels is higher than 90% after one century after used fuel discharge from reactor (see Fig. 2).

Finally, plutonium is a source of proliferation concerns: any plutonium amount, in any material, used fuel or waste form, has to be strictly controlled to prevent any diversion.

Plutonium recycling incentives

According to plutonium properties and their potential consequences mentioned above, one can draw several incentives for plutonium recycling in nuclear reactors.

1. *To take advantage of plutonium energy value*: this could extend the use of initial uranium loaded in the reactor; if plutonium recovered in used fuel is re-loaded and fissioned, the part of initial ^{238}U which has been transformed into plutonium can be valorized. The efficiency of such a strategy depends on the efficiency of ^{238}U to ^{239}Pu conversion, and also of the ability of the diverse plutonium isotopes to be efficiently fissioned (for both aspects, fast neutron reactors appear far more efficient than thermal neutron reactors). In an idealistic case, if the conversion of ^{238}U is quantitative, all the initial uranium isotopes could be either directly (^{235}U) or indirectly (^{238}U) fissioned, and thus the energy yield of the fuel can be increased by about 100.
2. *To improve the final waste management*: indeed, by removing plutonium from the waste, recycling drastically decreases the long term "radiotoxic inventory." The gain is however limited by the amplitude of formation of minor actinides (from high plutonium isotopes β -decay), and depends on the nuclear reactor characteristics. As for fission, the highest performance is obtained with fast neutron reactors, which favors plutonium fission (vs. capture then decay) at the expense of minor actinides formation.
3. *The impact on proliferation issues is not to be determined in a general manner*: recycling plutonium leads to plutonium consumption (thus a favorable effect) but the recycling operation may use processes and technologies leading to potentially temporary increase in plutonium accessibility, needing a case per case in depth analysis to be conclusive and adapted safeguards regulations and technologies to prevent any diversion of material.

The physics of Pu recycling in nuclear reactors

Plutonium in thermal neutron spectrum: Potentialities and limitations

The average neutron flux in light water reactor is presented in Fig. 3 (Santamarina et al., 2009).

According to plutonium isotopes cross-sections presented in section "Plutonium formation and amount in used fuels," one can draw the following conclusions:

- (1) The high energy part of the flux, even if the most important, is of far lower importance on plutonium nuclei transformations than the low energy part: this is due to the respective values of absorption cross-sections in these two different zones, far higher for thermal neutrons than for fast neutrons.
- (2) Odd plutonium isotopes (^{239}Pu , ^{241}Pu) present rather high fission versus absorption ratio values in the thermal zone of the neutron flux; in this respect, their behavior is similar to ^{235}U behavior in a thermal reactor.
- (3) Even plutonium isotopes (^{238}Pu , ^{240}Pu , ^{242}Pu) present very low fission versus absorption ratio values in the thermal zone of the neutron flux; in this respect, their behavior can be seen as introducing a "poisoning" effect into the reactor core (that is, absorbing neutrons), that leads to the formation of higher plutonium isotopes (and then minor actinides by spontaneous β -radioactive decay).
- (4) Even plutonium isotopes (notably ^{240}Pu and even more ^{242}Pu) present high fission to absorption ratio value in the high-energy zone of neutron flux ("fast neutrons" zone). This has to be considered in assessing the thermal reactor core safety (effect on the reactivity of a loss of moderator resulting in spectrum hardening), and can limit the effective load of plutonium in light water reactors.

Plutonium in fast neutron spectrum: Potentialities and limitations

The average neutron flux in fast neutron reactor is presented in Fig. 4 (Rimpault et al., 2002).

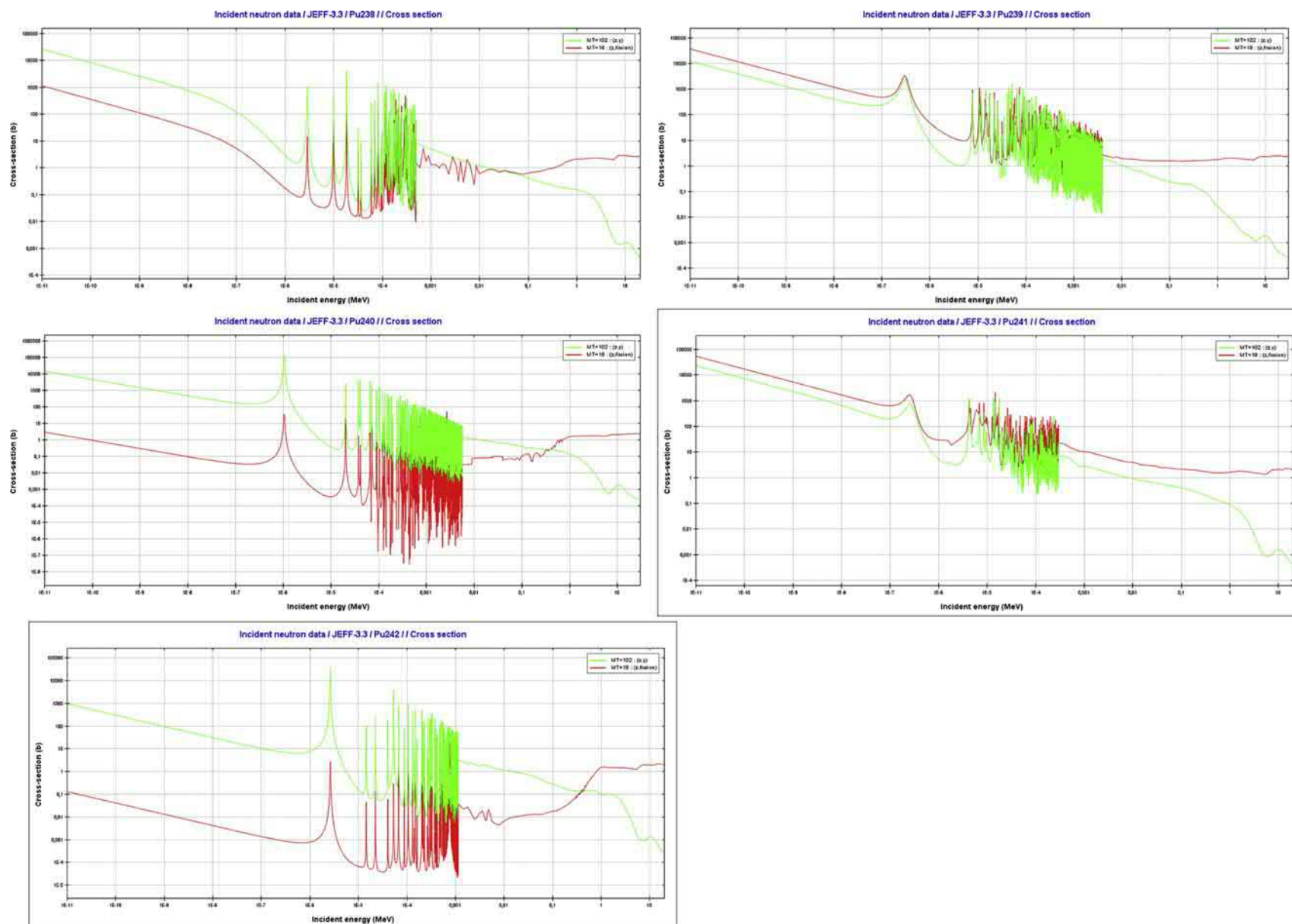


Fig. 1 Fission and capture cross-sections of plutonium isotopes (^{238}Pu to ^{242}Pu , barns) versus incident neutron energy (eV). From JANIS data bank, JEFF3, Nuclear Energy Agency.

Table 2 Plutonium isotopes amount in used fuel^a.

<i>Pu isotope</i>	<i>Kg per TWhe^b at discharge from reactor</i>	<i>Kg per TWhe 5 years after discharge from reactor</i>
Pu ²³⁸	0.74	0.77
Pu ²³⁹	17.12	17.4
Pu ²⁴⁰	7.51	7.54
Pu ²⁴¹	4.8	3.8
Pu ²⁴²	2.00	2.00

^aEnriched natural uranium oxide fuel, initial enrichment 3.7%, average burn-up 46 GW days /t.

^bApproximatively: 2.8 t used fuels at 46 GWd/t needed to get 1 TWhe of electricity.

From French CEA COSI code calculations.

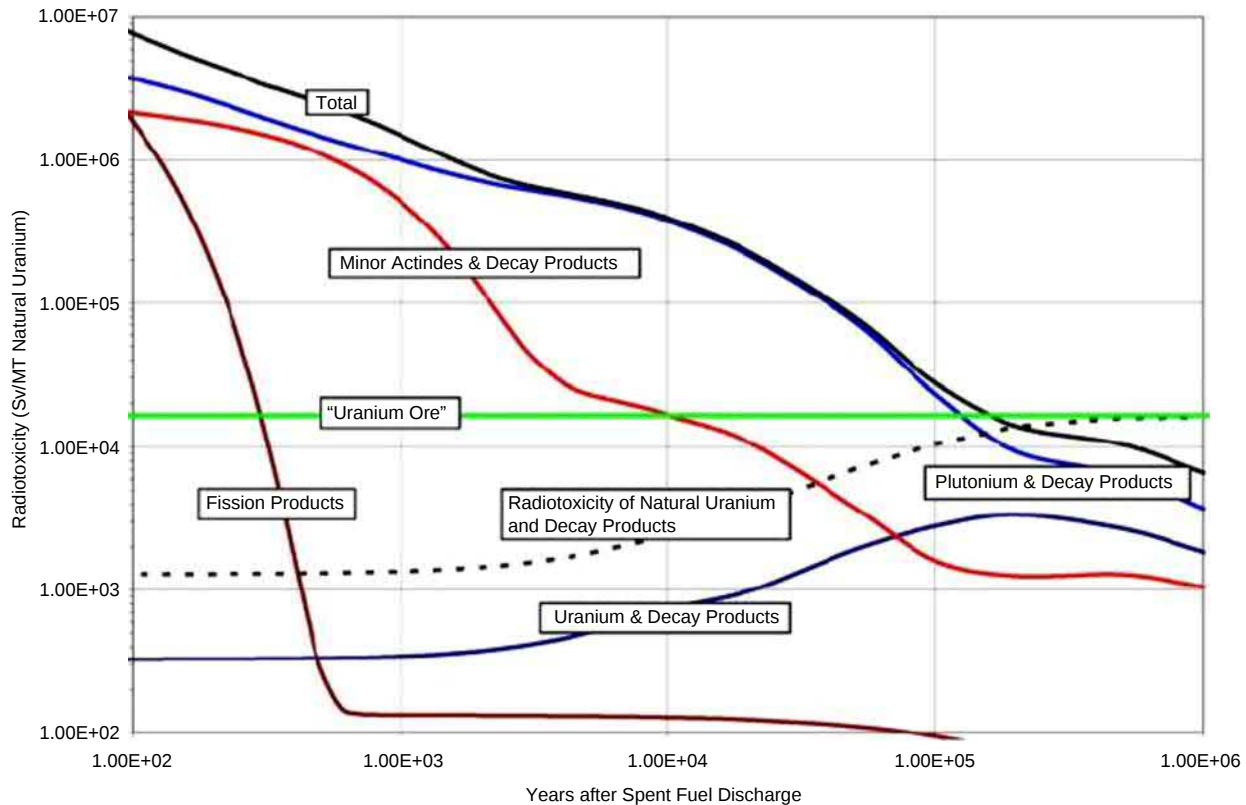


Fig. 2 Contribution of diverse components to the radiotoxicity of used fuel versus time after discharge. From: OECD/NEA (2006) *Physics and Safety of Transmutation Systems: A Status Report* (Report 6090).

According to plutonium isotopes cross-sections presented in section “**Plutonium formation and amount in used fuels**”, one can draw the following conclusions:

- (1) The most important feature is that fast neutron reactors don’t present any significant part of the flux in the thermal zone: Pu nuclei transformations occur in the high neutron energy zone, where absorption cross-sections are far lower than in thermal zone, but fission to absorption ratio values are higher.
- (2) Both odd and even plutonium isotopes present high fission to absorption ratio values: for all plutonium isotopes, the dominant processes is fission.

The average “secondary” neutrons resulting from fission (η = number of fission neutrons generated per neutron absorbed) is presented in **Fig. 5** (for diverse nuclei fission, and for diverse incident neutron energy range).

Typical values are given in **Table 3**, respectively for Light water reactors and fast neutron reactors.

It can be noticed that ²³⁹Pu fission at high neutron energy provides higher η values than ²³⁵U or ²³⁹Pu fission at low neutron energy. This provides in the first case more available neutrons, which can be used to convert ²³⁸U into ²³⁹Pu. Consequently, the conversion ratio of ²³⁸U is higher in a fast reactor than in a thermal neutron reactor because fast neutron reactor uses ²³⁹Pu as the main fissile isotope.

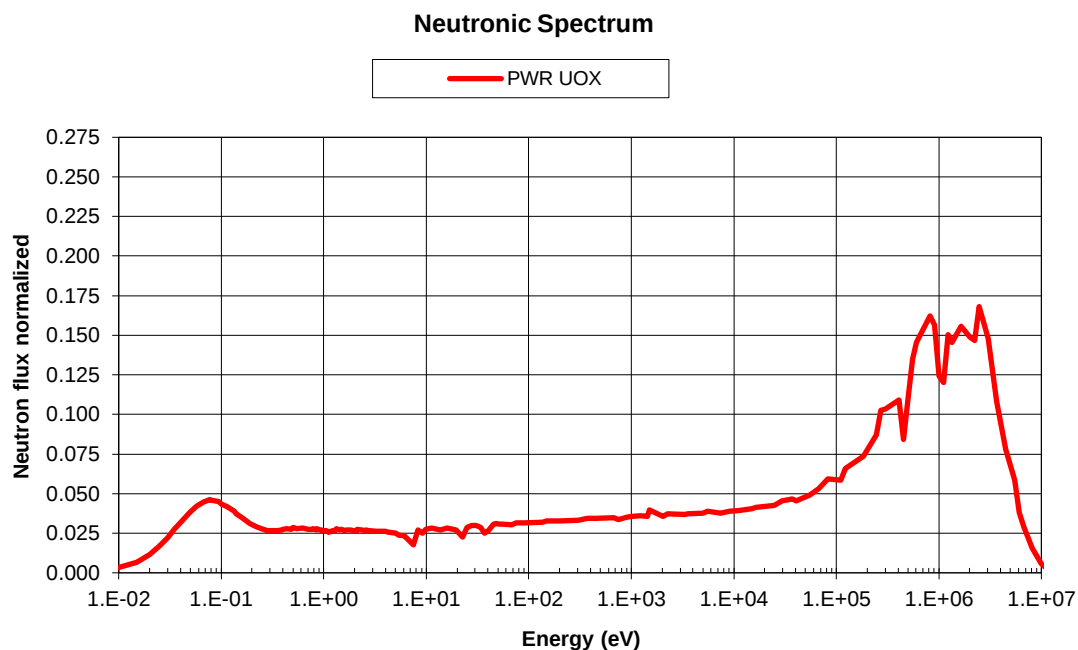


Fig. 3 Neutron flux average distribution versus neutron energy in a light water reactor. From: A. Santamarina A et al. (2009) *APOLLO2.8: A Validated Code Package for PWR Neutronics Calculations—ANFM 2009*, Hilton, Head, Island, USA.

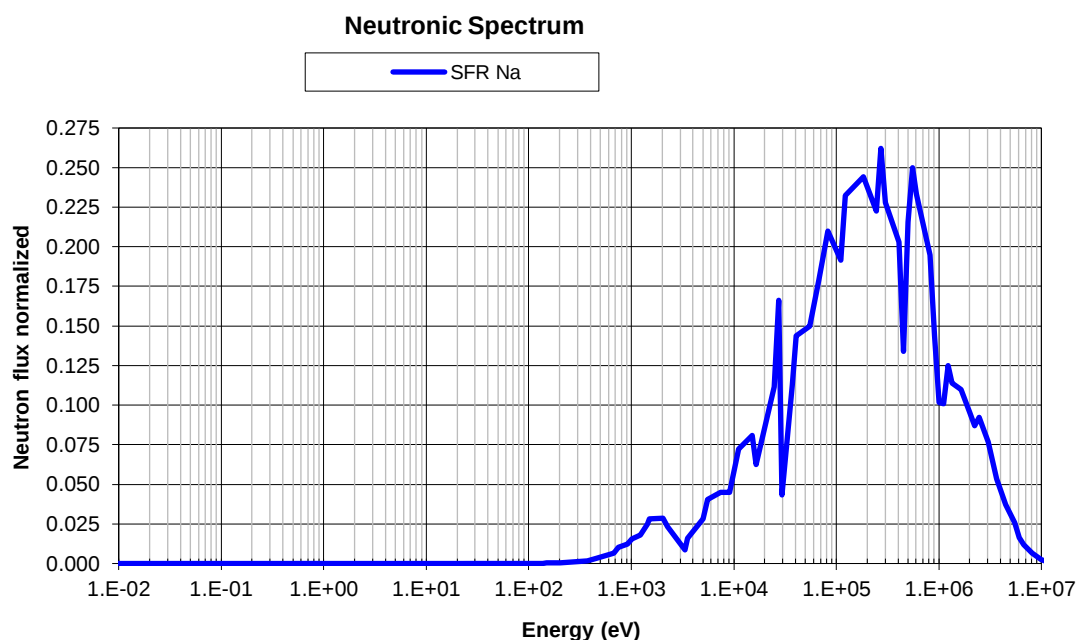


Fig. 4 Neutron flux typical average distribution versus energy in a fast neutron reactor. From: Rimpault G et al. (2002) *The ERANOS Code and Data System for Fast Reactor Neutronic Analyses*, PHYSOR Conference.

Plutonium recycling in light water reactors

Plutonium first recycling in LWR: The MOX option

Plutonium recycle is currently operated in France: Natural enriched uranium used fuel is reprocessed, and the totality of the recovered plutonium is reloaded in the nuclear fleet as Mixed OXide (MOX) fuel. MOX fuel is obtained by mixing recovered Pu (which is here the fissile element, replacing ^{235}U) with depleted uranium (essentially ^{238}U , ^{235}U content 0.1–0.3%). Plutonium content in the fuel depends on the isotopic ratio, and is about 5–10% in average: this provides a reactivity of the fuel analog to enriched uranium fuel (4% ^{235}U).

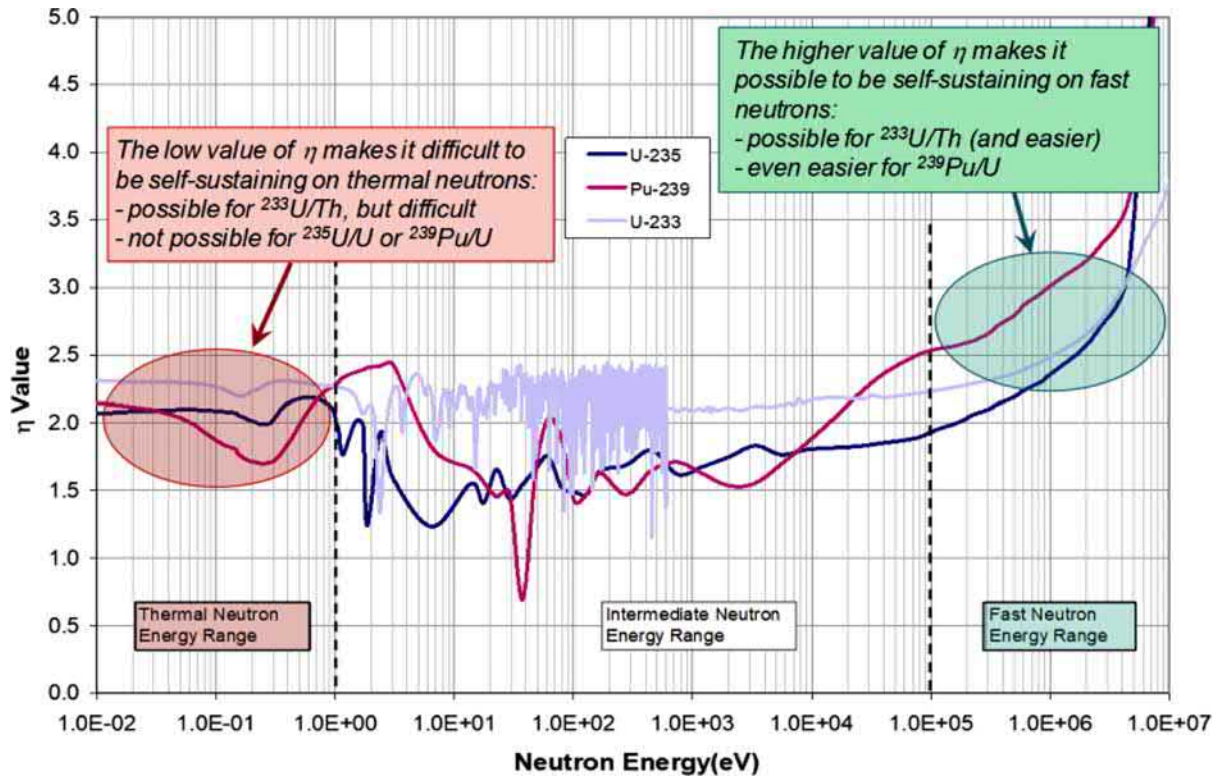


Fig. 5 η values vs neutron energy for diverse fissile nuclei (^{233}U , ^{235}U , ^{239}Pu). From: Wigeland R et al. (2014) *Nuclear Fuel Cycle Evaluation and Screening*, Final Report, I DOE-FCRD-2014/106.

Table 3 η average values in typical light water and fast neutron reactors.

	PWR spectrum	FR spectrum
U^{235}	2.0	1.9
Pu^{239}	1.8	2.3

From: BUSSAC and REUSS (n.d.) *Traité de neutronique*, Hermann ISBN.

The **Table 1** (see above) presents average isotopic content of plutonium recovered from enriched uranium used fuel; ^{239}Pu is the main isotope, but the higher is the burn-up of the fuel, the higher is the content of high plutonium isotopes (^{240}Pu and above) in recovered Pu, and the lower is its average reactivity in thermal spectrum. So, the higher burn-up is, the higher the global plutonium content must be in MOX fuel; and the trend to increase fuel burn-up in reactors in the previous decades (from about 30 GWd/t up to about 50 GWd/t) lead to increase Pu content in MOX fuels from about 6% to almost 10% currently.

MOX fuel form consists in mixed oxide ($\text{UO}_2\text{-PuO}_2$) homogeneous pellets, put into rods sub-assemblies strictly identical to enriched uranium ones; if recovered plutonium is complete, MOX fuel is about 10% of the total fuel in the fleet. The presence of MOX subassemblies in the core leads to some changes: as this can be seen on **Fig. 6**, and by comparing to **Fig. 3**, neutron spectrum is slightly “hardened” by plutonium enhanced thermal neutrons absorption, with consequences on reactor control rods behavior, needing some adjustments.

In the French fleet, MOX assemblies are combined with enriched uranium assemblies up to 30% in some dedicated reactors being MOX assemblies. **Fig. 7** presents the principle of the current French recycling policy: used enriched uranium fuels are systematically reprocessed, recovered plutonium is recycled as MOX fuels (in addition to plutonium systematical recycling, recovered uranium in used fuels can be re-enriched and reloaded too), used MOX fuels are not reprocessed and are currently stored in pools.

MOX fuels are currently not reprocessed not because of a technical impossibility (the specificities and difficulties can be overcome; industrial demonstrative campaigns have demonstrated the feasibility of such an operation on more than 70 t), but because the isotopic content of plutonium recovered from used MOX fuels is not suitable for an efficient use in LWRs (see **Table 4**).

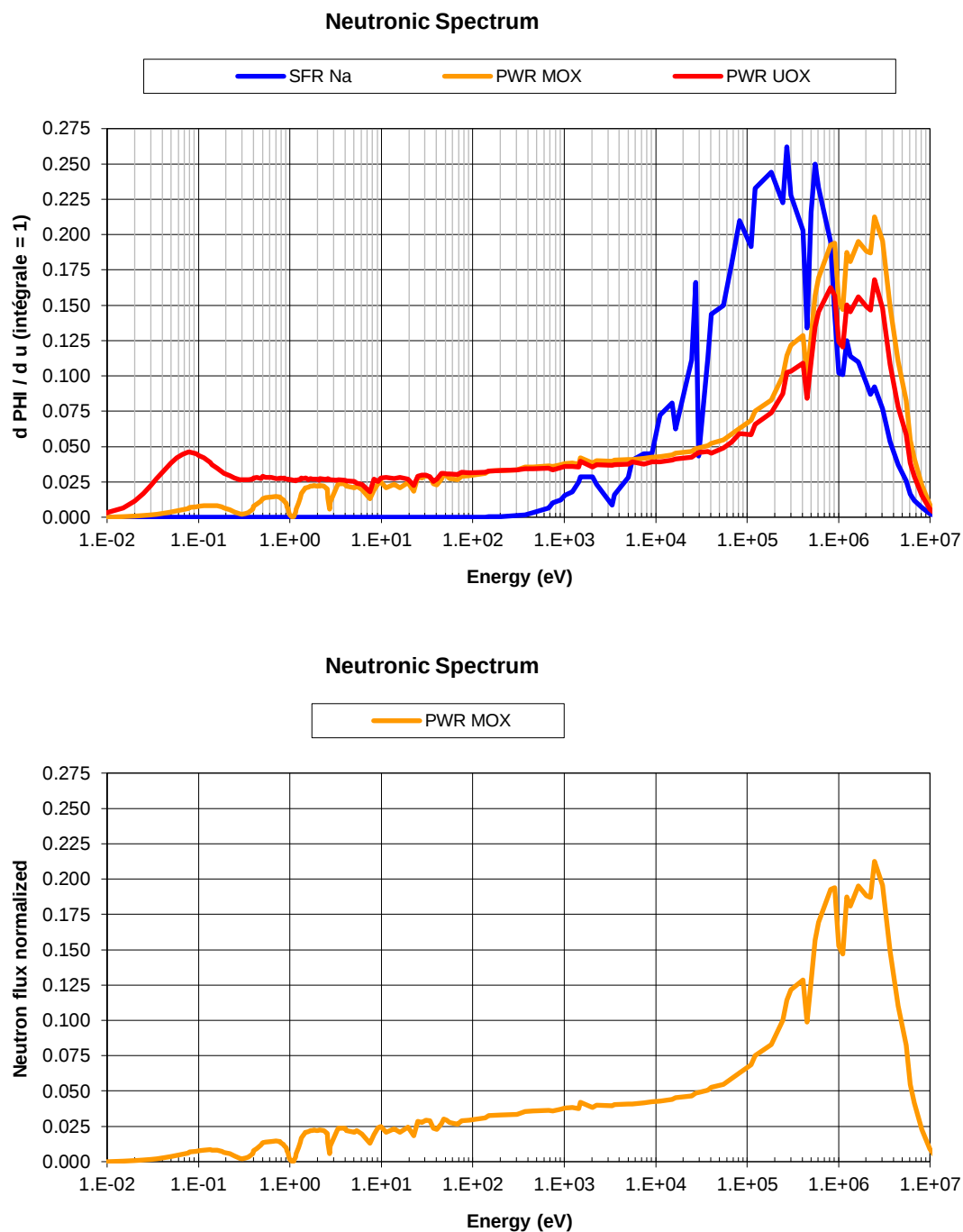


Fig. 6 Neutron flux typical distribution vs neutron energy in a light water reactor fuel with UOX (red) fueled with one third MOX (orange) and for sodium fast neutron reactors (blue). From CEA.

The values presented in Table 4 show significantly lower fraction of odd (fissile) Pu isotopes in the used MOX fuel that effects Pu recycle in thermal reactors:

- The higher relative amounts of ^{240}Pu and ^{242}Pu require an increase of plutonium concentration in MOX fuels to recover the targeted nominal reactivity, but this is limited by safety concerns due to even isotopes neutronic behavior in some incidental conditions (cf. section "Plutonium in thermal neutron spectrum: Potentialities and limitations" (4)).

An additional recycle is possible by combining Pu with different isotopic compositions, but an infinitely recurrent recycling of plutonium in standard (current) MOX form is thus practically not possible in LWRs.

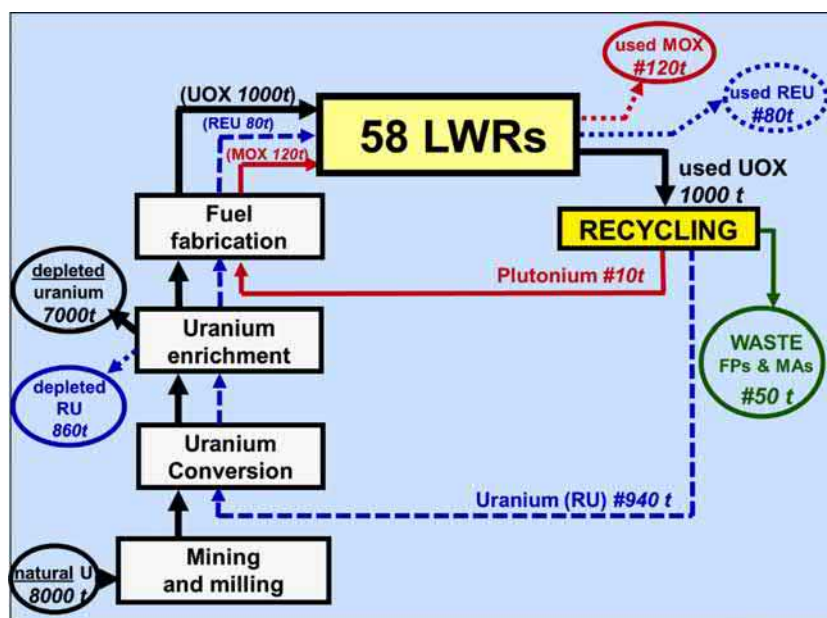


Fig. 7 The current nuclear fuel cycle in France (annual amount). From: CEA (2012) *Sustainable Radioactive Waste Management*, Report to French Government, 2006 Act.

Table 4 Plutonium isotopic content in uranium oxide fuels and in MOX fuels.

Pu isotope	Pu from used UOX	Pu from used MOX
^{238}Pu	2.3%	3.5%
^{239}Pu	51.4%	38.6%
^{240}Pu	24.2%	29.9%
^{241}Pu	15.0%	16.5%
^{242}Pu	7.1%	11.5%

From French COSI code calculations.

Advanced potential recycling concepts

In order to tackle this impossibility to indefinite recycling, two main ideas could be thought about to reach a recurrent recycle of plutonium in LWRs:

- (1) Isotopic separation of even plutonium isotopes, to maintain the “isotopic quality” of recycled plutonium all along the “multi-recycling” process by purging it from its non-fissile isotopes. This seems theoretically possible, but with (i) the technical difficulty related to isotopic separation operation, (ii) the necessity of confining in long lasting wastefoms the non-recycled (rejected) plutonium isotopes, and (iii) high security concerns about the potential increased proliferation risks. This route has never been considered by any country for industrial operation, and does not seem to be considered by any important research effort.
- (2) Add enriched uranium to recovered plutonium to compensate the decrease in fissile plutonium isotopes: this is the basis of several concepts under development today (especially in Russia and in France).
 - First, enriched uranium oxide (^{235}U about 1% or 2%) can be blended with plutonium oxide to increase the reactivity of the mixture, and thus allow an efficient behavior in the reactor within comfortable safety margins. In such conditions, plutonium total content in the fuel remains about 10%, as in current MOX fuel: the difference is that uranium is enriched a little, instead of depleted uranium in standard MOX fuel. This route is referred to as the homogeneous plutonium recycling.
 - Second, a heterogeneous route can also be developed by mixing in a same fuel sub assembly both standard MOX rods (about 10% plutonium and depleted uranium, with a lack of reactivity if plutonium presents large amount of heavy isotopes), and enriched uranium rods (^{235}U content higher than standard value to compensate the presence of the “under reactive” MOX rods).

Both concepts have been under research: for instance, MIX (homogeneous) or CORAIL (heterogeneous) designs have been studied by French teams, and pilot-scale experimentations are scheduled for the coming decade.

Table 5 Estimate of the effects of (hypothetic) CORAIL and MIX implementation in the French fleet (vs. open fuel cycle and current mono-recycle options) considering a French nuclear fleet of 63.2 GWe LWR providing 400 TWh_e per year.

	<i>U and Pu mono-recycle</i>	<i>CORAIL multi-recycle</i>	<i>MIX 9.54% multi-recycle</i>	<i>Fast reactor multi-recycle</i>	<i>Open cycle</i>
Fraction of MIX/CORAIL or FRs in the fleet	0%	87%	35%	100%	0
U _{nat} consumption (t/year)	6000	6300	5500	0	7500
Pu inventory (t/year)	+ 7.2	Stabilized	Stabilized	Stabilized	+ 9.9
Minor actinide inventory (t/year)	+ 3.3	+ 4.2	+ 4.5	+ 2.3	+ 2.7
Increase in spent fuel inventory (t/y)	180	0	0	0	950

From: Chabert C (2019) *Prospective Inventory of Radioactive Materials and Waste Produced by the French Nuclear Fleet According to Various Options*. Global 2019, Seattle.

The impact of these specific designs on LWRs fleets has also been assessed (see Table 5 for the French case): it can be seen that this could effectively allow plutonium multi-recycling and thus avoid used fuel plutonium accumulation. The main drawbacks are, besides complication of fuel cycle operations, the formation of higher amount of minor actinides (americium and curium) by increased capture reactions on plutonium nuclides followed by β^- radioactive decay. In Table 5 are presented the main results of the implementation of plutonium recycling devices (CORAIL or MIX) in the current french fleet (from Chabert, 2019).

Plutonium recycling in fast neutron reactors

Plutonium physics in fast reactors presents principle advantages, due to more favorable physical properties, and notably (cf. section “Plutonium in fast neutron spectrum: Potentialities and limitations”):

- fission to capture cross-sections ratio higher than in thermal reactors;
- easier conversion of ^{238}U to ^{239}Pu , thanks to η values higher than in thermal reactors (cf. section “Plutonium in fast neutron spectrum: Potentialities and limitations”).

These advantages can open the way to several nuclear systems for several diverse purposes.

Self-sustainable nuclear systems

Based upon the two main advantages recalled just above, one can imagine a recurrent plutonium and uranium systematical recycle in fast neutron reactors (see Fig. 8).

In such schemes, the fissile element is plutonium, initially loaded in the core and then continuously produced from neutron capture in the non-fissile ^{238}U . Plutonium can be multi-recycled without any limitation, as all plutonium isotopes (both odd and even isotopes) are fissile in fast neutron flux. This is the consequence of the two main advantages of fast neutron spectra described in the first paragraph.

So, the fuel can be a mixture of plutonium and depleted uranium (both metallic and oxide forms have been experimented worldwide). Inside the core, a part of the plutonium is fissioned (resulting in energy release and in FPs formation), and a part of uranium is converted into plutonium (^{239}Pu). Discharged used fuel is reprocessed, fission products (and also the amount of minor actinides formed by capture reactions in Pu isotopes and their capture reaction products) are removed, and recovered uranium and plutonium are recycled and refueled in a recurrent way, without any physics-based limitation. A complement of

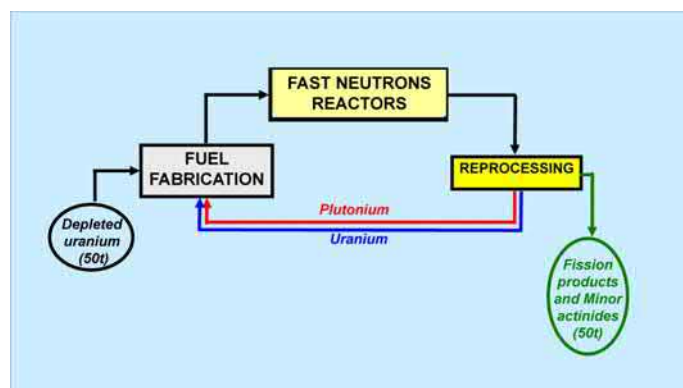


Fig. 8 The principle of U Pu multirecycle in a fleet of fast neutron reactors.

^{238}U (depleted or natural uranium) is added to the reloaded fuel in order to compensate for the amount used primarily by neutron capture (making Pu) by fissions induced by fast neutrons (thus making the neutron economy in fast reactors even better).

Fast reactor cores can be designed in several ways:

- (1) *Plutonium consumption is perfectly balanced by plutonium formation*: the system is “self-balanced,” and can last as long as ^{238}U uranium is available. Considering the fact that ^{238}U amount is rather abundant in earth crust, more than 100 times higher than ^{235}U amount (and also that depleted uranium stockpiles are currently very large worldwide), one can speak of self-sustainable nuclear systems (able for instance for France, to provide electricity for more than 1000 years, with today estimated uranium reserves and current electricity consumption).
- (2) *Plutonium formation is higher than plutonium consumption*: the system is a breeder (overproduction of plutonium): this can allow expansion of the nuclear power plants fleet, by making initial fuel loading for new reactors using the plutonium excess recovered from used fuels.
- (3) *Plutonium formation is lower than plutonium conversion*: the system is not able to be operated in a sustainable way without adding fissile nuclides (i.e., plutonium) regularly to the recycled fuel. This option is not as interesting as the two other ones, except for plutonium inventory reduction purposes as explained hereafter in section “**Plutonium burners**”.

Plutonium burners

Here the purpose is *not* to deploy sustainable nuclear systems based on ^{238}U conversion to produce plutonium; the purpose is to decrease plutonium amount (either separated available plutonium, or plutonium content in used fuel stockpiles). Fast neutron reactors can be used, as plutonium can be efficiently consumed (by fissions, providing energy), without any limitation on number of recycling. The main “principle” issue in this case is that if uranium is also present in the fuel, it will be partly converted into plutonium again, which compensates plutonium consumption.

So, to reach high plutonium consumption rates, it seems natural to think in this case about uranium-free fuels: here one would get pure plutonium burners, but some important difficulties have to be overcome. ^{238}U presence in the core is key for reactor safety, notably thanks to significant “Doppler effect” provided to decrease core reactivity upon fuel temperature increase in incidental conditions. ^{238}U conversion is also useful to maintain the reactivity of the core all along reactor operation with the fuel (in situ formed plutonium compensates plutonium amount decrease due to fissions and neutron captures). These are real issues or concerns, and actually, attempts for designing plutonium consumption fast neutron cores have up to now essentially tried to limit uranium amount in the fuel, not to get rid of it.

One example is the “CAPRA” project in France: in the specific case of a nuclear fleet phase-out decision, the future fast reactors fleet could directly be used to burn plutonium thanks to a significant modification of the fuel. In order to effectively consume plutonium, the CAPRA principle is to increase the plutonium content and decrease the uranium content in the fuel. In an hypothetical fast reactor fleet with plutonium content in the fuel up to 40% (which could lead to some issues regarding diverse items, not considered at this step), the global plutonium inventory within the fleet could be decreased by about 30% in 30 years (CEA, 2012).

Synergic systems (using both LWR and fast neutron reactors)

Self-sustainable fast reactors fleet appears in some respects promising, but could remain (it’s a possibility) more sophisticated, more expensive than LWR fleet. On the other hand, LWR present important advantages, but can’t meet alone long-term sustainability goals. So, symbiotic fleets have been thought to, incorporating fast reactors to improve uranium and plutonium management, but keeping a strong basis of LWRs. This could also be seen as a pathway to evolve from a LWR fleet to future fast reactor fleet, in a progressive stepwise approach.

Plutonium multi-recycling in such schemes is possible, and a comprehensive study has been performed to assess several routes for the evolution of the current French LWR fleet. Table 6 presents, for diverse FR/LWR ratio in the fleet, the values of some key-variables (from CEA, 2015). The main conclusions are that the higher fast reactors in the fleet, the higher plutonium recycling potentialities, and thus the lower natural uranium consumption and the lower used fuels stockpiles.

Table 6 Effect of an increased fraction of FR in a LWR fleet on key fuel cycle attributes.

	Open cycle (0)	UPu monorecycle	UPu multirecycle
Fraction of FRs in the fleet	0	0%	31%
U_{nat} consumption (t/year)	7500	6000	3400
Pu inventory (t/year)	↗ 9.9	↗ 7.2	Stabilized
Minor actinide inventory (t/year)	↗ 2.7	↗ 3.3	↗ 4.4
Unrecycled used spent fuel to be stored (t/year)	950	180	0

The minor actinides issue

Minor actinides in used fuels

Capture nuclear reactions in nuclear reactors lead to formation of transuranic elements: plutonium is the most abundant in spent fuel, but significant build-up (about 0.1–0.5% total mass) of neptunium and of “trans-plutonium” elements are also observed, in particular for high burn-up fuels.

- Neptunium (long-lived ^{237}Np is the only noticeable isotope, as others are short-lived) can be formed either by successive capture reactions in ^{235}U , or by (n,2n) reactions in ^{238}U , followed by β -decay; it can also be formed more indirectly by alpha decay from heavier transplutonic isotopes.
- Americium (mainly ^{241}Am and ^{243}Am , as other isotopes are short-lived) is formed by capture reactions followed by β -decay from plutonium isotopes (^{240}Pu and ^{242}Pu respectively for ^{241}Am and ^{243}Am).
- Curium (mainly ^{244}Cm , ^{245}Cm) are resulting from capture reactions in americium isotopes followed by β -decay; ^{242}Cm is also formed (from ^{241}Am), but is converted relatively quickly (its half-life is $\sim 1/2$ year by alpha-decay into ^{238}Pu).
- Heavier elements (Cf, Bk) are also produced during fuel residence in the nuclear reactor, but their respective quantities are very low (producing, by β -decay, Cf isotopes from Cm, needs several successive neutron captures from ^{244}Cm up to ^{248}Cm , which is of low occurrence).

Table 7 presents the respective amount (Kg/TWhe) of these elements in used fuels.

The following main observations can be noticed:

- (1) The total build-up of other than plutonium transuranic elements is about few ‰, of the total used fuel mass, that is to say about one tenth of plutonium build-up. That is why Np, Am and Cm are called “minor” actinides, as their respective amount in used fuel is rather low.
- (2) All minor actinide isotopes are heavy nuclides, far from stability state and each of them belongs to a long alpha decay chain (with alternate alpha and β -emitters before reaching stable nuclides). All of them are, directly or by their decay products, source of long-lasting alpha radioactivity (in all chains, presence of at least one long-lived alpha active nuclide).
- (3) The higher the fuel burn-up, the higher the minor actinide content (no significant consumption by fission, fission average cross-sections are rather low for the mentioned minor actinide isotopes).
- (4) For plutonium-based fuels (MOX fuels), Am and Cm amounts are higher than for enriched uranium fuels (due to formation mechanisms exposed just above); it's the contrary for neptunium, for a large part coming from ^{235}U neutron capture.
- (5) In fast neutron reactors fuel, minor actinide amounts are lower than in thermal reactors fuel: this is due to plutonium isotopes fission to capture cross-sections ratio (higher for fast neutrons and leading to plutonium fission rather than forming higher plutonium isotopes, potential minor actinide source).

The goals of minor actinide recycling

minor actinides (MA) amounts in used fuels are, because of their radioactive properties (long-lived alpha emitters), the second contributor to long term used fuel radiotoxicity¹ (the very first one is plutonium, 10 times higher due to higher amount). Recycling

Table 7 Minor actinide amount (in kg/TWhe) in used fuels (i) at fuel discharge; (ii) 5 years after fuel discharge.

(i)	Typical enriched uranium fuel (LWR) 46 GWd/t	Typical MOX fuel (LWR) 46 GWd/t	Typical FR MOX fuel 81 GWd/t
Np	1.7	0.4	0.4
Am	0.7	5.0	1.7
Cm	0.3	4.1	0.3
MA ^a	2.7	9.5	2.4
(ii)	Typical enriched uranium fuel (LWR) 46 GWd/t	Typical MOX fuel (LWR) 46 GWd/t	Typical FR MOX fuel 81 GWd/t
Np	1.7	0.4	0.3
Am	1.7	10.9	3.1
Cm	0.2	2.9	0.1
MA ^a	3.6	14.2	3.5

^aTotal MA, total minor actinide content.
From French COSI code calculations.

¹Radiotoxicity: dose (Sv) delivered per unit of intake (Bq)

minor actinides, *in addition to prior plutonium recycle*, can offer a significant reduction of final waste long term radiotoxicity (after three centuries, MA contribution to global radiotoxicity is several orders of magnitude higher than residual fission products contribution). This has been a strong incentive for research in the field of MA separation and recycling; but the following should be noticed:

- (1) Converting a minor actinide nuclide into another minor actinide nuclide (by neutron capture) is inefficient if the goal is to decrease long term radiotoxicity of the final waste; as mentioned in section “**Minor actinides in used fuels**” (2) all of them can be considered as long-lived alpha emitters, with a comparable radiotoxicity. To decrease radiotoxicity, only fission is efficient (to get shorter-lived fission products).
- (2) MA recycling can make sense only if plutonium has been first removed from final waste; it's clearly a “secondary” issue.
- (3) The benefit of MA recycling is still a matter of debate: intrinsic radiotoxicity of final waste must be balanced by the confinement properties of the final repository. For instance, MA are almost not mobile in the deep clay-rich layers as those considered in France for the repository. The long-term impact of the repository is therefore driven by some more soluble and mobile fission products although they are less radiotoxic. In such a case the direct advantage of MA recycling is therefore questionable, and “cost to benefit” approaches has been launched in order to assess (in respect to several diverse attributes and criteria) the relevance of minor actinide recycle.

Another incentive often mentioned is the residual *heat load of final waste*, which is a key parameter for designing the deep repository, with direct influence on the overall cost. After about one century, spent fuel residual heat load is mainly due to ^{241}Am radioactivity (^{241}Am is the only present nuclide with half time of about several centuries). This could be a real incentive for Am recycle, in particular in the case of final deep repository, whose compactness is governed by decay heat emission.

Minor actinide recycling in fast neutron reactors

According to section “**The goals of minor actinide recycling**” (1), minor actinides have to be recycled into reactors able to provide their fission. Fission to absorption cross-section ratios for the minor actinide nuclides of interest (^{237}Np , ^{241}Am , ^{243}Am , ^{244}Am) are very low in thermal spectrum: water-cooled reactors are not suitable to get a significant amount of MA fission, and it would be no use to recycle them into such reactors. It would lead to the production of higher isotopes, without any significant decrease in radiotoxicity.

Fast reactors have therefore to be considered for this purpose: fission to absorption ratio is more favorable (although not so high, far lower than for plutonium isotopes, except for neptunium). Fission routes are more complicated than plutonium ones, not direct but often consisting successive capture reactions, radioactive decay and finally fission (see for instance the ^{241}Am transmutation case). Complete transformation of fed MA requires longer residence times and could hardly been reached without MA multi-recycling (that is to say MA-bearing targets loading into the reactor, processing of unloaded MA-bearing targets for residual MA recovery and reload, and so on...). Experiments have been achieved with “once through” irradiation of MA-bearing targets: about less than 50% of MA initial content have been fissioned in average, which appears far from enough; it seems it would be difficult to reach high fission yields using once through targets, without significant (and unacceptable) neutron induced radiation damage (notably important swelling).

So, diverse routes have been considered for MA multi-recycling in fast reactors:

- (1) *Homogeneous MA recycling*: MA are diluted in the whole core fuel, processed and recycled with plutonium and uranium.
- (2) *Heterogeneous MA recycling*: MA are concentrated in specific targets (MA bearing core fuels or dedicated MA bearing blankets); processing and recycling requires dedicated facilities.
- (3) *Dedicated strata*: MA are recycled in specific fast reactors, dedicated to MA transmutation (the “dedicated strata,” while the main strata is for power generation). A particular option for this route is the “accelerator driven systems” (ADS), with a sub-critical core and extra-neutrons provided to the system (for instance by spallation reactions) (Abderrahim and De Bruyn, 2021).

Each route (which have not yet been studied at higher scale than laboratory-scale) would present advantages and drawbacks (see Wigeland et al., 2014) in brief:

- The homogeneous recycling route presents the disadvantage of “contaminating” the whole fuel of the nuclear fleet, and also leads to some safety issues due to MA unfavorable physical properties;
- The heterogeneous recycling route presents the advantage of “confining” MA to a tiny part of the fleet; but handling high MA concentration in dedicated fuels can be a difficult issue, and if blankets are placed at the periphery of the core, lower transmutation yields will result due to lower neutron flux;
- The dedicated strata with ADS devices still poses diverse physical and technological challenges, concerning both the control of the spallation reaction and the sub-critical core control with high minor actinide loading.

Additional information on multi-recycling in nuclear reactors can be found in Shwageraus (2021), Wigeland and Taiwo (2021) and Krepel (2021).

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The Adaptation of Recycling Processes to Pu-Multi Recycling

Manuel Miguiditchian^a and Robin Taylor^b, ^aCEA, DES, ISEC, DMRC, University of Montpellier, Marcoule, France; and ^bNational Nuclear Laboratory, Central Laboratory, Sellafield, United Kingdom

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Glossary

AHA Aceto-hydroxamic acid.
APM Marcoule Pilot Plant.
ASTM Standard Specification for Nuclear-Grade Uranyl Nitrate Solution or Crystals and Nuclear-Grade Plutonium Dioxide Powder, Sinterable.
BN-600 Sodium cooled fast breeder reactor located in Russia.
BWR Boiling Water Reactor.
CHON concept using molecules containing only C, H, O and N atoms.
DBP Di-butyl-phosphate.
DEHBA N, N-Di-2-éthylhexyl-butanamide.
DEHiBA N,N-Di-2-éthylhexyl-isobutanamide.
DFR Dounreay Fast Reactor.
FBTR Fast Breeder Test Reactor.
FPA Fission products and actinides.
FR Fast reactor.
GEN IV Generation IV reactors. Different designs, mostly neutron fast reactors, currently developed for commercial applications by the *Generation IV International Forum* (GIF).
KNK compact sodium-cooled nuclear reactor developed in Germany.
LWR Light Water Reactor.
MA Minor Actinides.
MBP Mono-butyl-phosphate.
MOX Mixed Oxide Fuel.
NEXT New Extraction system for TRU recovery.
PFBR Prototype Fast Breeder Reactor.
PFR Prototype Fast Reactor.
PUREX Plutonium Uranium Redox EXtraction.
PWR Pressurized Water Reactor.
SFR Sodium Fast Reactor.
TBP Tri-butyl-phosphate.
TCP Specific Fuel Treatment (future workshop for Pu-rich fuels dissolution at Orano La Hague reprocessing plant).
TTC Twice through cycle.
UOX Uranium Oxide fuel.
VVER-440 Water-water energetic reactor, Pressurized Water Reactor developed in Russia.

Introduction

Plutonium multi-recycling is the term applied to nuclear fuel cycles in which spent fuels from reactors are reprocessed to recover uranium and plutonium which are then fabricated into new fuel for re-use in the reactor. Unlike the twice through cycle (TTC), in which the plutonium is only recycled once and the spent mixed oxide (MOX) fuels are then stored ready for disposal, in this scenario the spent MOX fuels are reprocessed and the plutonium re-used in new fuels multiple times. The TTC has been industrially implemented at commercial scales but, while it can be achieved in light water reactors (LWR), plutonium multi-recycling in fast neutron reactors (FR) is considered to be the optimum way to reach a fully closed and sustainable nuclear fuel cycle (see Chapter 7, [Boullis and Chabert, 2021](#)). The usual approach consists first in the reprocessing of spent LWR MOX fuels to recover Pu and produce FR MOX fuels to feed the first “Gen. IV” fast reactors deployed into the fleet. Then, in a second step, used FR MOX fuels are treated to recycle U and Pu and produce new FR fuels. Unlike Pu recycling in LWR fuels, the Pu isotopic composition in fast neutron spectra is not an issue and theoretically allows infinite recycling with no further need of using natural uranium resources.

However, as with the development of fast neutron reactors themselves, scientific and technological challenges have to be addressed to adapt the back-end of the nuclear fuel cycle and the associated facilities to meet this strategy. Indeed, current generation reprocessing plants are designed to reprocess spent uranium oxide or UOX-type fuels that contain approximately 1% plutonium once unloaded from pressurized or boiling water reactors (PWR or BWR). So while there is a substantial degree of feed-back already available from the industrial scale reprocessing of LWR fuels, for the reprocessing of FR MOX fuels industrial experience has been limited to specific campaigns or only performed through pilot units ([Masson et al., 2013](#)).

This chapter will, therefore, summarize some of the international experiences in reprocessing FR MOX fuels using the PUREX process and then describe some innovative approaches that aim to address the limitations of the PUREX process.

MOX reprocessing past experience and limitations of the existing PUREX process

In this section, firstly some past international experiences in reprocessing MOX fuels will be highlighted, particularly those undertaken at industrial or semi-industrial scales, and then some of the lessons that can be learned from these experiences will be summarized.

International experience

In France significant experience has been gained in the reprocessing of MOX fuels as part of the efforts to develop advanced fuel cycles based on plutonium multi-recycling. Indeed, 12 tons of used LWR MOX fuels were first reprocessed between 1991 and 1998 at the Marcoule pilot plant (APM) and the Orano La Hague UP2-400 plant. 63 tons were then treated at UP2-800 through four specific campaigns performed between 2004 and 2008 in order to confirm the feasibility of LWR MOX fuel reprocessing at an industrial scale. A large and unique pre-industrial experience is also available in France on the reprocessing of fast reactor fuels. Approximately 25 tons of FR MOX fuels unloaded from several sodium cooled fast reactors (SFR), including Rapsodie, Phenix (in France) and KNK (in Germany) were treated from 1969 to 1997 firstly at the AT1 workshop at la Hague and then at the Marcoule pilot plant (APM). These campaigns were performed either at a low rate (less than 5 t/y) or in dilution with larger quantities of used UOX fuels to mitigate the impact of MOX fuels on the process performance. Furthermore, up to 25 t of FR MOX were processed at the Marcoule Pilot plant at 1.5 kg of (U + Pu)/h without prior dilution of the MOX fuel liquor. While significant experience has been obtained over this period to demonstrate MOX reprocessing, the results also emphasize the limitations of the PUREX process to reach industrial treatment capacities with these specific fuels. Due to the characteristics of FR MOX assemblies and the higher content of Pu (see [Fig. 1](#)), the limitations were mainly related to the head-end steps; that is the mechanical dismantling and fuel dissolution steps. In fact, the dismantling of used FR MOX fuels was not carried out in the reprocessing plant but close to the reactor in the post-irradiation examination (PIE) cell and this operation was consequently identified to be the main issue for the development of future high-capacity industrial treatment.

The UK fast reactor program at Dounreay included a dedicated plant for the reprocessing of the spent fuels from both DFR (Dounreay Fast Reactor) and PFR (Prototype Fast Reactor) ([Judd and Ainsworth, 1998](#); [Natarajan, 2015](#)). The plant was designed to reprocess 120 day cooled MOX fuel with Pu contents of 17–25% and decay heat 3 kW per subassembly. 5.5 t of MOX fuels from PFR were reprocessed between 1980 and 1996 with burn ups of over 80 GWd/t and cooling times as low as 136 days. The plant used a PUREX based solvent extraction process with 3 cycles but using sulfate ions for plutonium separation. This is a separation based on the preferential complexation of Pu(IV) compared to U(VI) by the aqueous phase sulfate ions rather than the usual reductive stripping (using U(IV) and hydrazine, ferrous sulfamate or hydroxylamine). However, the use of sulfate ions is unattractive for future plants since it adds to the aqueous wastes, causes problems in vitrification and is potentially corrosive.

India has a detailed national program to develop nuclear energy including fast reactors as part of its 3-stage strategy ([Banerjee and Gupta, 2017](#)). Fuels from the Fast Breeder Test Reactor (FBTR) of up to 70% Pu content were reprocessed in the pilot scale CORAL facility with a peak burn up of 155 GWd/t and 2 year cooling period. However, the FBTR fuels were (U,Pu) carbide rather than oxide. The 500 MWe Prototype Fast Breeder Reactor (PFBR) being commissioned will use MOX fuel although subsequent units are planned with metallic fuels. The solvent extraction process was adapted to manage the high Pu content and the higher

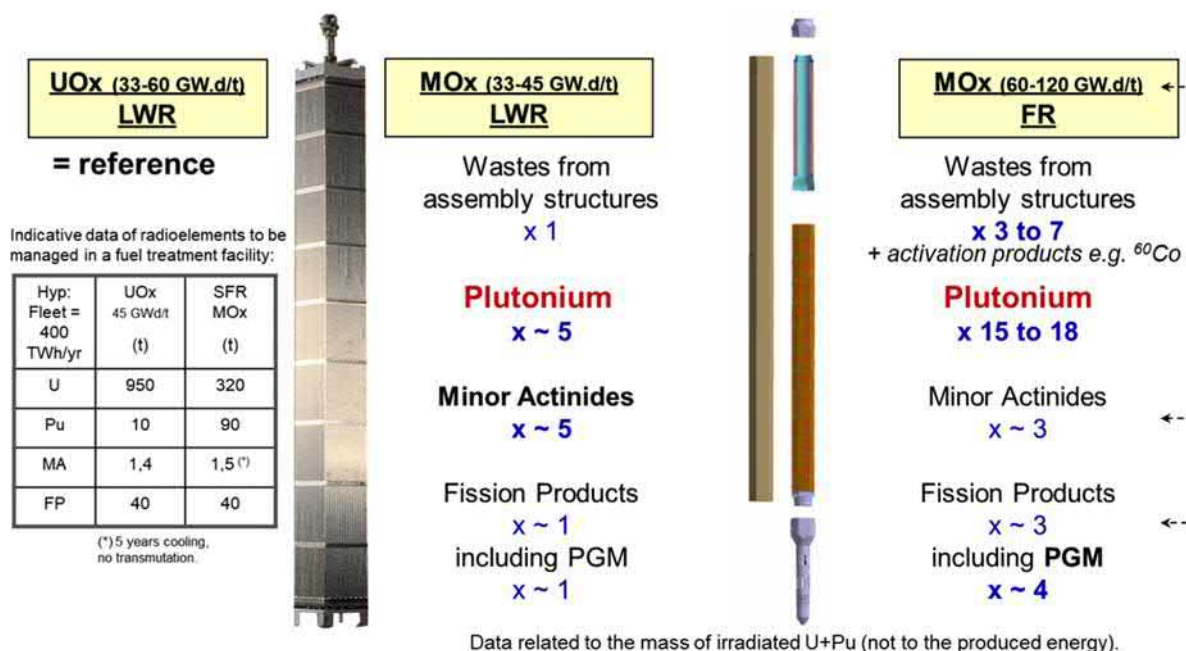


Fig. 1 Main differences between LWR UOX, LWR MOX and FR MOX fuel compositions.

concentrations of fission products compared to the reprocessing of thermal fuels. Centrifugal contactors were installed for reprocessing FR fuels in order to manage the radiation dose to the solvent (Natarajan, 2015, 2017).

A number of tests have been made in the reprocessing of MOX fuel from Japan's JOYO reactor but these are mainly small scale using the Chemical Processing Facility (at JAEA's Tokai site). Some examples are described below in Section "Application of crySTALLIZATION for uranium separation" as part of the development of their NEXT process for FR fuel reprocessing. Kilogram quantities of MOX fuels from the KNK-II fast breeder reactor in Germany were processed in the pilot scale MILLI facility. These fuels had a maximum burn up of 100 GWd/t with 10 month cooling periods (Marth and Koehler, 1998; Natarajan, 2015).

Russia has also a large experience in MOX fuel fabrication and reprocessing. Recently, 8 fuel assemblies unloaded from the BN-600 reactor and with a burn-up of 73–89 and a cooling time from 17 to 21 years have been reprocessed in 2012 and 2014 at the RT-1 plant (Kolupaev et al., 2016). Uranium and plutonium were fully recovered and no increased losses of plutonium were detected compared to VVER-440 fuel reprocessing.

Lessons learned

As noted above, French experience clearly highlighted the challenges in the head end steps that must be addressed in preparing the spent FR MOX fuels for reprocessing. On the other hand, the increase of Pu content from 1% to 5% (in LWR MOX) and up to 30% (in FR MOX) in the spent fuels will impact the efficiency of the Pu recovery at the dissolution step. This is because a fraction of Pu is present as rich PuO_2 fragments in the fuel (that are more refractory to dissolution in refluxing nitric acid). Hence, dissolution under classical PUREX conditions would lead to an increase of residues and potential losses of Pu in the vitrified wastes. Such Pu losses can also be a criticality hazard if they were allowed to build up in the plant (Senentz, 2021).

The reprocessing of several tons of LWR and FR MOX in industrial or pilot scale reprocessing plants was a great opportunity to demonstrate reprocessing of Pu-rich fuels by hydrometallurgy. Nevertheless, it also highlighted the limitations to overcome to reach the industrial reprocessing capacity required for Pu multi-recycling in the future. Even if the main difficulties were focused on the head-end steps, it was clear that plutonium separation and purification from spent MOX fuels, fresh MOX fuel fabrication and wastes conditioning processes will also have to be adapted to take into account the higher content of Pu and the characteristics of spent MOX fuels. Moreover, in addition to the capacity to process high Pu content fuels, these new processes will have to:

- be more efficient (simple and compact) in order to be economically viable;
- be reliable and available (keeping in mind that in a closed fuel cycle the recycled products will be the unique source for feeding the reactors);
- be flexible enough to process LWR and FR MOX fuels as well as UOX fuels during the transition periods where both LWR and FR reactors will be present in the fleet;
- have enhanced global safety especially toward proliferation resistance;
- reduce the generation of wastes and more generally the environmental footprint in order to meet future regulatory and environmental requirements.

Over the last two decades, ambitious R&D programs have thus been launched in France, India, Japan and elsewhere to meet these challenges. While the exact approaches may differ, such R&D programs need to address the fundamental limitations and develop key processes for Pu multi-recycling including (Miguirditchian et al., 2017):

- an industrially scalable head-end process for MOX dissolution and full Pu recovery in solution;
- a new and simplified solvent extraction process for plutonium and uranium separation from Pu-rich solutions;
- a high-capacity process for plutonium and uranium co-conversion into mixed oxide and an automated and flexible MOX fabrication process avoiding powder milling and dust dissemination as far as practicable;
- an optimized management of the effluents and specific wastes generated by spent FR fuels.

The next sections will describe some proposed innovations developed internationally that aim to address these specific challenges.

Adaptation of the headend processes to FR fuel characteristics

The FR fuel assembly (here in the case of SFR fuel) is very different from an LWR fuel (see Fig. 1). In addition to the different cladding material (stainless steel vs zircaloy), the presence of a hexagonal wrapper tube and spacer wires around each pin makes the mechanical dismantling much more complex compared to LWR fuels. For instance, in France, dismantling was first performed at the Marcoule pilot plant by opening the wrapper tube (by mechanical drilling), removing the spacer wires and shifting the pins. Once separated, the pins were sheared mechanically. The number of operations and their complexity explains the difficulty with such approaches to increase the capacity treatment of FR fuels and reach industrially-relevant throughputs.

In the short term, a new industrial unit called TCP (for specific fuel treatment), dedicated to the shearing and dissolution of Pu-rich fuels (irradiated or not), is under development and should be implemented at the Orano UP2-800 La Hague plant (Durand et al., 2015). This unit, which is designed to treat up to 40 t/y of MOX fuels, includes a new shearing step, a dissolution unit and complementary steps to recover the residual plutonium still contained in the fuel cladding after rinsing and in the fine undissolved particles collected after feed clarification (see Fig. 2). The residual presence of undissolved particles is explained on one hand by the higher content of platinum group metal (PGM) elements in MOX fuels which form insoluble intermetallic phases in the presence of Pu. This behavior is particularly enhanced in the case of FR fuels because of higher burn-ups and higher temperatures in the pellets. On the other hand, refractory Pu-rich fragments can also be formed locally in the (U,Pu)O₂ oxide and will contribute to the increase of solid residues after MOX dissolution. R&D studies have recently shown that the quantity of residues is directly linked to the fabrication process and to the initial Pu enrichment (Bertrand et al., 2019).

To ensure quantitative Pu recovery and minimize the loss of Pu in the waste, an additional digestion step must be coupled to the primary dissolution. The digestion process is based on the principle of oxidizing dissolution with generated Ag(II). The process is currently being optimized at the laboratory scale in order to extend its use to a wide range of particles potentially containing Pu (Fig. 3). Similar studies focused around the oxidative dissolution process using Ag(II) are ongoing in the UK and elsewhere (Carrott et al., 2012; Natarajan, 2017). Ag(II) can be regenerated during the process and thus acts as a catalyst in the dissolution process; electrochemical generation or ozonolysis are leading technologies for Ag(II) generation. For comparison, an alternative complexing dissolution with HNO₃/HF was used in Russia for BN-600 irradiated fuel reprocessing (Kolupaev et al., 2016). The fuel was first dissolved in 10 M nitric acid and one or two complementary dissolution steps have been carried out with 1 g/L fluoride ions in HNO₃ to complete Pu dissolution.

In the longer term, new dismantling and dissolution processes should be proposed to ease the access to the fuel and reduce the overall size of the head-end steps (Fig. 4). Innovative shearing techniques avoiding the use of a mechanical blade such as a laser or water-jet are under assessment on mock-up assemblies at CEA. The goal is to reach a full separation of the irradiated oxide from the cladding in order to reduce the size of the dissolver and avoid the partial dissolution of the cladding in nitric acid. Complementary steps such as thermal treatment (voloxidation) might be used to help the oxide/cladding separation by a volume expansion of the fuel due to the phase transformation of the monolithic fluoritic UO₂ to orthorhombic α -U₃O₈ powder (Collins, 2018; Maher,

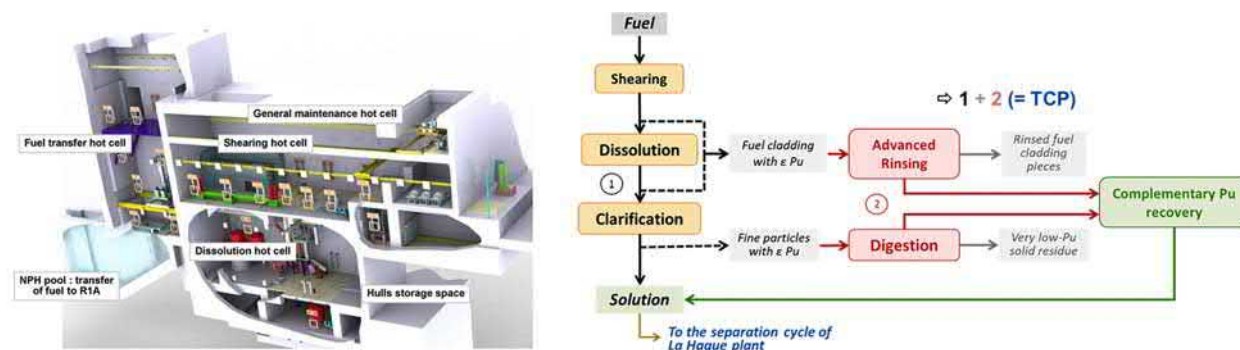


Fig. 2 3D view of Orano's new shearing and dissolution facility (TCP) currently at the project stage and simplified scheme of the TCP process.

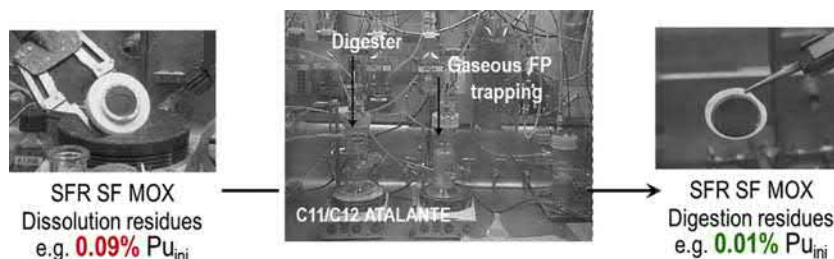


Fig. 3 View of the experimental setup used to digest dissolution-resistant solid residues containing some plutonium, in the ATALANTE facility at CEA Marcoule (France).

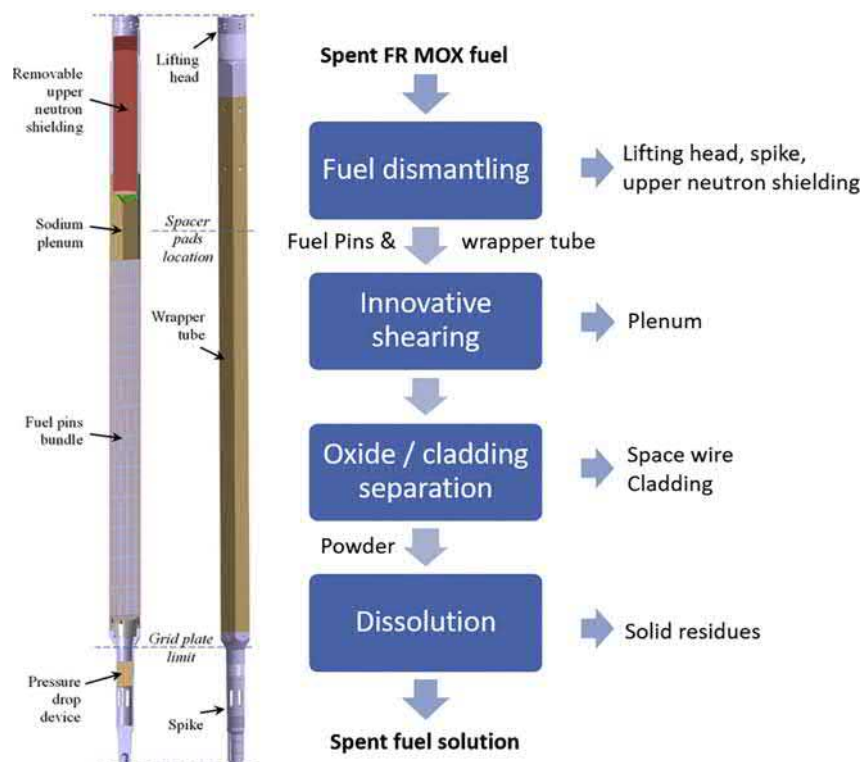


Fig. 4 New head-end operations considered for used FR MOX fuel treatment.

2015). Once separated from the cladding, U-Pu-FP-MA powder will be dissolved in nitric acid in a very compact dissolver unit. A gain up to a factor of 10 is expected on the size of the dissolver compared to the wheel dissolver implemented in current industrial reprocessing plants (Netter, 2012).

Adaptation of the partitioning processes to the specificity of Pu from FNR fuels

While international experiences have demonstrated the application of the PUREX process to FR MOX fuel reprocessing, further process improvements are possible to adapt the separation flowsheet to higher Pu/U ratios. First, as illustrated in Fig. 5, several extraction and purification cycles were required to separate and purify U and Pu when applying historic PUREX reprocessing to FR fuels. Unlike the current PUREX process, a co-decontamination of U and Pu from FP and MA was performed in the first extraction cycle while Pu and U separation occurred in a 2nd extraction cycle. Each element was then submitted to a 3rd purification cycle to meet the purity standards. Into the future, the reduction of the number of cycles should, however, be possible and will improve the process compactness in order to reduce the cost of a future plant.

The main principle of the PUREX process, using tributyl phosphate (TBP) as the extracting ligand, is the exploitation of Pu redox chemistry. Pu/U partitioning is performed in the PUREX process by selective reduction of Pu(IV) to Pu(III) using U(IV) and hydrazinium nitrate ($N_2H_5^+$). If the use of these reducing agents is well managed at the industrial scale with UOX fuels, the increase of Pu content in LWR and FR MOX fuels will drastically increase the consumption of these reagents (up to a 10-fold increase of hydrazine,

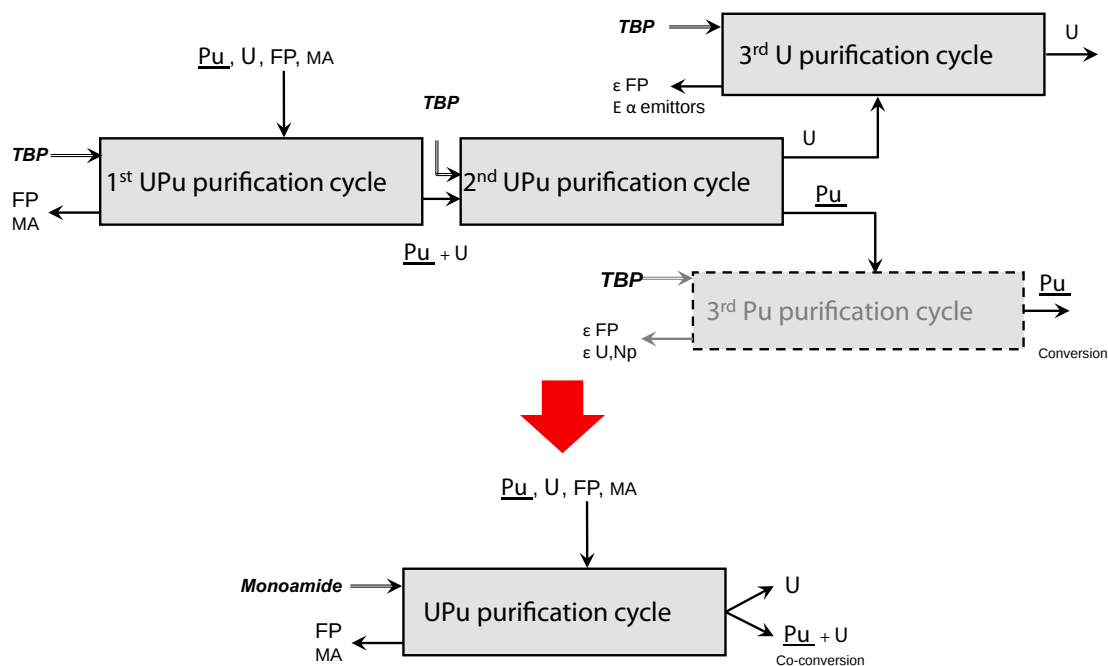


Fig. 5 Toward an innovating separation process of U and Pu; above: the TBP process applied in the past to Phenix FR MOX dissolution liquor, below: the process under development (“one single cycle redox free”).

a hazardous and toxic chemical that presents specific safety issues in the process that must be managed). Furthermore, the increase of the Pu/U ratio in MOX fuel solutions makes the PUREX flowsheet more sensitive to critical parameters reducing the safety operating margins in case of process upsets such as changes in flowrates or critical reagent concentrations (TBP, U(IV), hydrazinium nitrate, etc.).

It is also important to notice that FR MOX fuels are generally irradiated to a very high burn-up and also may be short cooled. The combination of the higher Pu content (α , n radiolysis) and the higher FP content (β - γ radiolysis) will lead to a faster degradation of the organic solvent under radiolysis. Recommendations were made after these industrial campaigns to improve the solvent regeneration (solvent clean-up step) to keep high purification performances toward U and Pu purification or to substitute the extracting molecule to avoid any accumulations of troublesome degradation products (such as HDBP or MBP).

The development of a new 1 cycle extraction process that is redox-free for the separation of U and Pu has thus become a priority objective for the treatment of future spent MOX fuels. Three strategies are described below to illustrate this concept.

Use of a novel extracting ligand

The first strategy requires alternative extracting systems (not based on TBP) allowing U/Pu separation without needing redox control of plutonium. *N,N*-dialkylamides are being studied in France (Acher et al., 2017; Condamines and Musikas, 1992; Moeyaert et al., 2016), India (Gupta et al., 2000; Pathak et al., 2010; Ruikar et al., 1992, 1993) and in Japan (Ban et al., 2014, 2016) for their ability to extract uranium and plutonium from concentrated nitric acid solutions. Among this family, some branched-alkyl monoamides have the advantage to separate U(VI) and Pu(IV) by decreasing the HNO_3 concentration. Monoamides show also high U/FP selectivity giving the opportunity to reduce the number of extraction cycles to only one and are highly stable toward radiolysis. The mixture of DEHBA (*N,N*-di-2-ethylhexyl-butanamide) and DEHiBA (*N,N*-di-2-ethylhexyl-isobutanamide) was selected in France and successfully hot tested in 2015 at the CEA's Atalante facility in laboratory-scale mixer-settlers (Sorel et al., 2017). The scientific feasibility of the “1 cycle process without redox control” was demonstrated by a quantitative recovery of uranium and plutonium, sufficiently purified from the fission products to reach ASTM specifications (ASTM, 1996, 2015). To respect Generation IV directives, a co-management of uranium and plutonium is planned in the process and a well-known fraction of uranium is co-stripped with Pu in the U/Pu partitioning step (Fig. 6). The U/Pu ratio should be defined by proliferation constraints and MOX fabrication input requirements. The U/Pu ratio in the Pu product was carefully monitored during the 2015 hot test and the value of 0.33 measured at the steady state was consistent with the target.

New asymmetrical *N,N*-dialkylamides are currently under assessment (single molecule, less viscous) and the best candidate will be tested in 2021 on an actual LWR MOX feed (Milanole et al., 2019). In Japan, *N,N*-dialkylamides are also studied for U and Pu separation from MOX fuel solutions and a 2-cycle strategy was chosen to recover U and Pu from MOX fuels: a first separation of the main part of uranium with DEHDMPA and the separation of Pu along with the residual U with DEHBA in a second step. A hot test was recently performed and showed good performances for U and Pu separation (Ban et al., 2014, 2016).

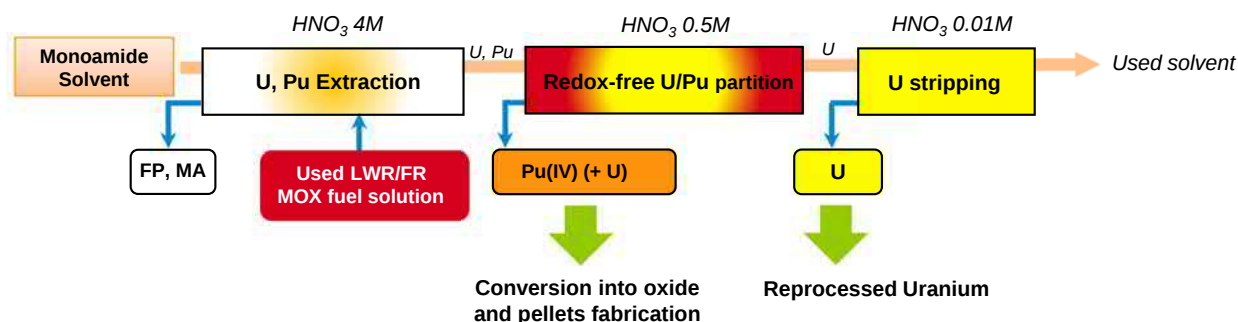


Fig. 6 Simplified flowsheet of an advanced separation process in 1 cycle and redox free for U and Pu separation from MOX fuel solutions.

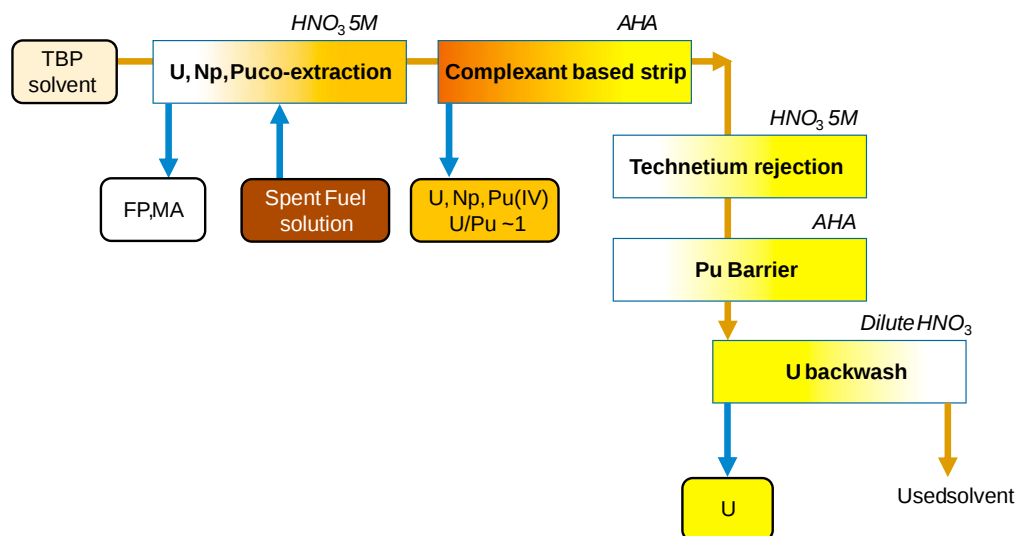


Fig. 7 A 1-cycle 'Advanced PUREX' process tested in centrifugal contactors with co-recovery of a mixed (U,Np,Pu) product from a simulated MOX feed and using complexant based stripping for plutonium separations.

Complexant based plutonium stripping

The second strategy proposes the retention of the TBP (or similar tri-alkyl phosphate) for co-extraction of uranium and plutonium in the primary extract-scrub step but then separates U from Pu by the complexation of Pu(IV) using an aqueous phase ligand that is selective for tetravalent actinides over the hexavalent uranyl ion. Aceto-hydroxamic acid (AHA) has been demonstrated to be a very effective ligand for achieving this 'complexant based Pu stripping' (Birkett et al., 2005). Advantages of complexant based stripping with AHA include fast kinetics, no requirement for hydrazine and ability to process high Pu content feeds. As AHA is a CHON molecule it can be decomposed to gases rather than adding to waste volumes. The high selectivity of AHA for Pu(IV) over U(VI) is demonstrated by their respective stability constants (at 22 °C; ionic strength = 2 mol/L) (Carrott et al., 2007):

$$\text{Log}_{10}\beta_1 (\text{UO}_2^{2+} - \text{AHA}) = 7.94$$

$$\text{Log}_{10}\beta_1 (\text{Pu}^{4+} - \text{AHA}) = 14.2$$

Since it is a substituted hydroxylamine, AHA is also redox active and can rapidly reduce extractable Np(VI) to inextractable Np(V) which presents the opportunity to route the neptunium with the plutonium stream for burning in fast reactor fuels (Birkett et al., 2007). AHA, however, in acid solution is slowly hydrolyzed to hydroxylamine and acetic acid which means it is best used in short residence time solvent extraction equipment, i.e. centrifugal contactors. Pu(IV) can be slowly reduced to Pu(III) but this is related to the hydrolysis of AHA and is not relevant under normal operating conditions (Carrott et al., 2008). A number of flowsheet tests with AHA have been reported as part of 'advanced PUREX' process development in the United Kingdom, India, USA (the UREX process), France and Russia (Collins, 2018; Herbst et al., 2011). Fig. 7 illustrates an example of one of these tests that produced a mixed (U,Np,Pu) product (with a U/Pu ratio of ~1) and a pure bulk U product from a simulated MOX feed (Taylor and Mathers, 2019). A technetium rejection stage was added to manage the specific problem of technetium decontamination in the PUREX process (Taylor, 2021). Recent work has also confirmed that AHA can be used to produce mixed U + Pu products that are likely to be preferred in future reprocessing.

Application of crystallization for uranium separation

The third strategy considered here is to replace part of the solvent extraction-based reprocessing with an alternative aqueous separation technology e.g. ion exchange or extraction chromatography. Perhaps the most promising option of this type that can be applied to high Pu content FR fuel reprocessing is the use of crystallization as proposed in the New Extraction System for TRU recovery or NEXT process that has been developed by JAEA (Japan) for recycling of spent FR fuels (Collins, 2018; Minato, 2015; Nakahara et al., 2007; Yano et al., 2007). Following dissolution, the liquor is cooled to separate a large amount (>70%) of the uranium as uranyl nitrate hexahydrate (UNH), by crystallization, and to increase the Pu/U ratio. A 1-cycle solvent extraction flow-sheet based on the PUREX process, i.e. using TBP as the extractant, then co-recovers the residual uranium, plutonium and neptunium. (The minor actinides, americium and curium, can be recovered by an extraction chromatography step on the aqueous raffinate). Some precautions must be taken to ensure Pu is present as Pu(IV) (since Pu(VI) will co-crystallize with the uranium) and to minimize contamination due to $\text{Cs}_2\text{Pu}(\text{NO}_3)_6$. The U and Pu (and Np) can then be co-extracted away from the fission products in the first stage of the solvent extraction flowsheet and then co-backwashed with careful control of temperature and nitric acid concentration, avoiding conditions that lead to plutonium (IV) hydrolysis reactions. The NEXT process has been tested with FR MOX fuel from the Joyo reactor in centrifugal contactors at the lab scale in the CPF Facility at JAEA.

Adaptation of processes for Pu conversion and FR MOX fabrication

At the end of the partitioning process, the solution of plutonium and uranium nitrates has to be converted into a mixed oxide (U,Pu) O_2 . The addition of uranium and the increase of Pu quantities in the cycle require the adaptation of the conversion step to higher metal flows. The capacity of the conversion process should be increased by approximately one order of magnitude compared to the industrial capacity of the current industrial Pu conversion workshop. To avoid a multiplication of the precipitation lines in a future plant, a compact and high-capacity process has to be designed for the quantitative (U + Pu) co-solidification and their calcination into a mixed oxide. As redox-free processes are mostly encouraged, an innovative co-conversion process of U(VI) and Pu(IV) was proposed. A denitration route based on an advanced thermal denitration of U(VI) and Pu(IV) solution in the presence of organic additives (ADAO process, see Fig. 8) shows promising results and has been selected as the CEA reference route for co-conversion (Leblanc et al., 2019). 15 g of $\text{U}_{0.8}\text{Pu}_{0.2}\text{O}_{2.2}$ were recently synthesized with this technique. The powder was found to be compatible with the pelletizing process after production of dense and homogenous MOX pellets. Direct premixing of Pu and U in solution instead of co-milling PuO_2 and UO_2 powders offers a better homogeneity in the final pellet and limits dust dissemination.

FR MOX fabrication is usually done by classical metallurgical powder processes (Marin, 2021). As depicted in Fig. 9 (left side), depleted UO_2 , PuO_2 and recycled powders (scraps) are first intimately mixed by ball milling. Then, after a granulation step, pellets are obtained by pressing and sintering. More than 100 t of FR MOX fuels were produced at the ATPu in Cadarache using this co-milling process from 1963 to 1999 to feed Phenix and SuperPhenix reactors. Based on this industrial feedback, the fabrication process was recently optimized. In addition to an improved automation and flexibility, this Advanced SFR process was simplified to lower the dust build-up to reduce the workers exposure (suppression of ball milling and scraps crushing milling steps). The process was also adapted in order to accept a mixed (U,Pu) O_2 obtained by oxalic co-precipitation as an input of the fabrication scheme (Vaudez et al., 2013).

To go further and suppress any powder handling in the first steps, innovative fabrication processes such as “wet routes” are under development. The most promising route consists of a freeze granulation of a UO_2/PuO_2 suspension followed by water elimination by lyophilisation (La Lumia et al., 2019) (Fig. 10). The small and homogeneous granules obtained are then pressed to give dense pellets after dedicated sintering. MOX pellets containing 15% of Pu have been recently obtained from this method. Microstructural analysis shows a high U/Pu homogeneity and a high density after sintering (>99% of Theoretical Density was achieved).

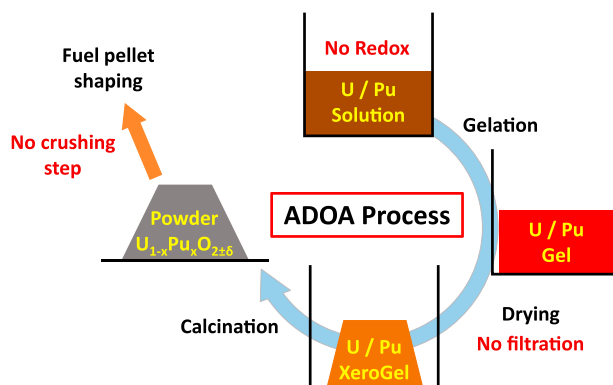


Fig. 8 Description of the advanced thermal denitration of U(VI) and Pu(IV) solution in presence of organic additives (ADAO process).

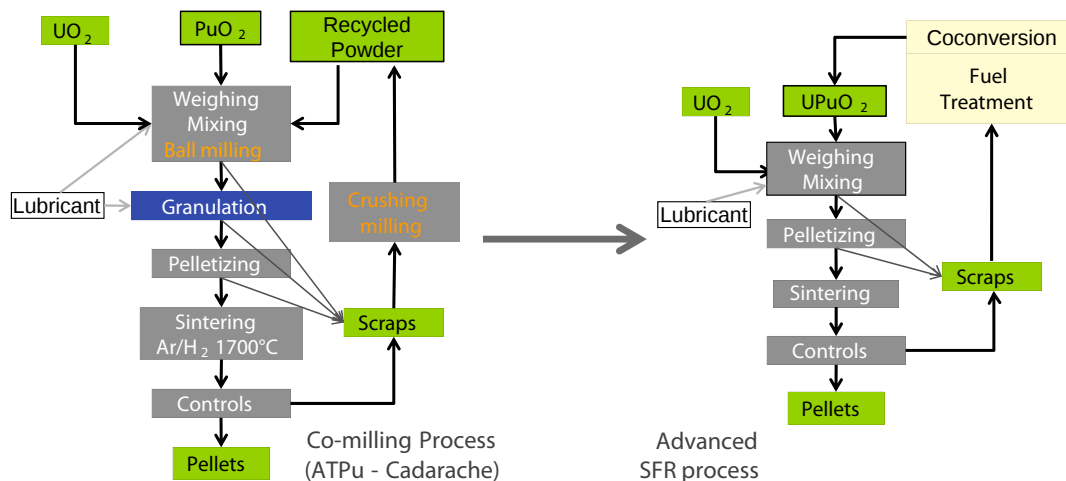


Fig. 9 Comparison between the advanced SFR MOX fuel fabrication process and the co-milling process used for Phenix and SuperPhenix fuel production.

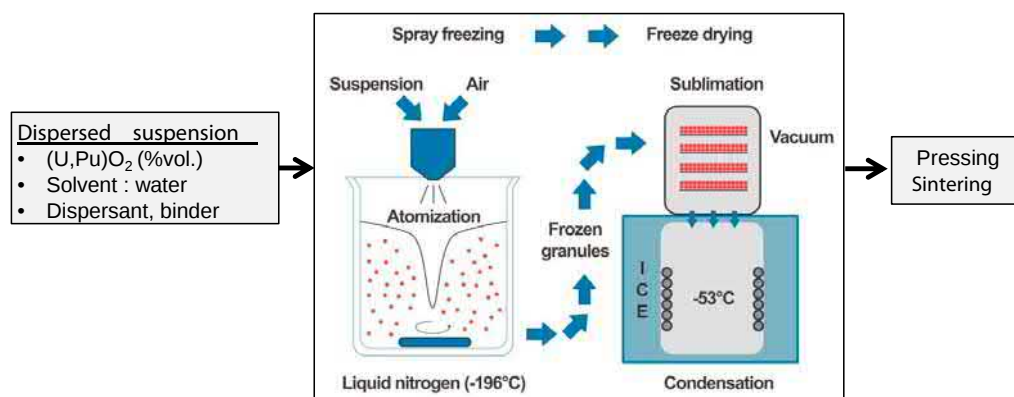


Fig. 10 Schematic principle of the MOX fabrication process by freeze granulation.

The impact of FNR-Pu to waste characteristics and conditioning processes

Characteristics of used FR MOX fuels (summarized in Fig. 1) will impact waste management, and thus strongly influence the technical solutions needed to be implemented for the treatment of the wastes arising from the future generation reprocessing plants. First, the higher quantity of stainless steel pieces in FR MOX subassemblies will increase the amount of intermediate-level waste produced between LWR UOX/MOX and FR MOX spent fuel reprocessing. Then, metallic fission products such as platinum group metals (Ru, Rh, Pd, Mo, etc.) will be more concentrated in the solid residues recovered after dissolution and clarification. These elements being responsible for partial sedimentation in the glass matrices, their incorporation in the glass could involve difficulties at the vitrification step (Bart, 2021). Finally, the higher Pu content will require a much better recovery of Pu from the cladding and the solid residues, in order to limit the losses of Pu in the ultimate wastes (which will go to the deep geological repository) to approximately the same value as obtained from current reprocessing of UOX fuels. To tackle these issues, a melting process has been proposed for the simultaneous conditioning of stainless steel cladding and solid residues (Ladirat et al., 2018) (Fig. 11). Melting allows a better homogeneity and higher density of the metallic waste compared to the compacting process currently used in the reprocessing of LWR fuels. Then, solid residues (containing a large part of metallic elements) could be incorporated with the cladding in the melting process, avoiding any vitrification issues. Eventually, residual plutonium oxide still contained in the fuel cladding and in the solid residues could be recovered by the addition of a specific slag in the melting process.

Laboratory-scale experiments on synthetic FR representative waste (stainless steel with metallic FP residues and Hf and Ce used as U, Pu surrogates) have recently confirmed the good incorporation of solid residues in the melt and the selective recovery of cerium oxide in the slag. Up to 3 wt% of metallic elements (Ru, Mo, Pd) were quantitatively incorporated into the molten metal without any issues. A SAC-type slag (based on Silica, Aluminum oxide, and Calcium oxide) at 1–5 wt% was required to recover the oxides released by the melt (Hf, Ce in particular) (Ladirat et al., 2018).

Increased quantities of fission products and minor actinides (by a factor 3 compared to UOX fuels) as well as the higher contents of iron, nickel and chromium (caused by partial corrosion of stainless steel cladding) require the development of new glass matrices

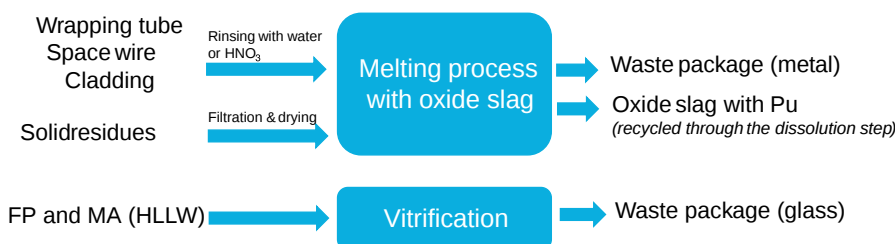


Fig. 11 Example of an optimized flowsheet for FR MOX waste management.

to reduce the number of waste canisters ultimately produced. Among the borosilicate glasses, the peraluminous-type formulations (Piovesan et al., 2017; Gasnier et al., 2014) seem to be of considerable interest to incorporate higher quantities of fission products and to increase the solubility of Cr, Fe in the glass matrix. Incorporation of higher concentrations of Cr and Fe in the glass was indeed pointed out as an issue for FR MOX treatment. Elsewhere, alternative processes to vitrification such as immobilization in ceramic matrices are also under development (Chen et al., 2016; Maddrell, 2013; Vance et al., 2016).

Conclusion

The specificities of FR fuels compared to LWR fuels and the increase of Pu content in the fuel require adaptation of the existing processes to successfully multi-recycle plutonium in future closed fuel cycles. Based on the global experience already acquired on LWR and FR fuels recycling, and on new innovative developments arising from new R&D programs, technical solutions are being proposed for each of the main steps of the future generation treatment/recycling plant. Some of the main innovations that are under investigation in global R&D programs, and that have been introduced in this chapter, are listed below:

- Oxide/cladding separation to optimize head-end operations, improve the dissolver compactness and speed up dissolution kinetics;
- “1-cycle process without redox” for uranium and plutonium separation by solvent extraction with N,N-dialkylamides molecules;
- U(VI) and Pu(IV) conversion into mixed oxides without redox control by direct denitration with organic additives;
- MOX fuel fabrication using “wet routes” avoiding powder milling;
- Melting process for the conditioning of fuel cladding pieces and solid residues in addition with a dedicated slag for Pu recovery and decontamination.

Building on past industrial scale experience of reprocessing MOX fuels, promising results have been obtained on these different steps in France and experiments are in progress to validate the technical option selected for each step. Based on these results, a general system analysis will be made by an integration of the different processes in a future plant in order to check the links between the different steps, to confirm the benefits of each options and to be sure that all the selected options can be integrated together in the same flowsheet.

Similar studies, to a greater or lesser extent, are occurring in India, China, Russia, Japan, the United Kingdom and elsewhere; including multi-national efforts sponsored by EURATOM European Union framework programs; the OECD-NEA and IAEA. These global efforts are building confidence that the FR fuel recycling needed to take full advantage of the benefits presented by Generation IV fast reactors and the fully closed fuel cycle can be realized later this century.

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Effluent Decontamination

V. Blet, CEA, Marcoule, France

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Glossary

Cyanobacterium Any of a major group (Cyanobacteria) of photosynthetic bacteria that produce molecular oxygen and use water as an electron-donating substrate in photosynthesis.

Decontamination factor (acronym DF) Ratio of the initial amount of activity in a stream to the final amount of activity in a stream following the treatment by a given process.

Electro deionization A water treatment technology that utilizes electricity, ion exchange membranes and resin to deionize water and separate dissolved ions (impurities) from water. It differs from other water purification technologies in that it is done without the use of chemical treatments and is usually a polishing treatment to reverse osmosis.

Electrodialysis A combination of electrolysis and dialytic diffusion in which an applied electric potential difference across an ion-exchange membrane drives salt ions from the solution on one side of the membrane to the solution on the other side. This process is usually carried out in a stack, in which alternating anion and cation exchange membranes are placed between two electrodes. Applying a current to this stack will produce alternating streams of permeate and concentrate.

Flocculent A substance added to a suspension to enhance aggregation of the suspended particles in the form of flocs. In the case of Actiflo™-Rad, the flocs are fixed on micro-sand to enhance the sedimentation.

Geopolymer An alternative cementitious material synthesized by combining source materials that are rich in silica and alumina with strong alkali solutions such as potassium or sodium hydroxide.

Heterogeneous photo catalysis Photocatalysis is the acceleration of a photoreaction in the presence of a catalyst. In catalyzed photolysis, light is absorbed by an adsorbed substrate. In photogenerated catalysis, the photocatalytic activity depends on the ability of the catalyst to create electron-hole pairs, which generate free radicals (e.g., hydroxyl radicals: $\bullet\text{OH}$) able to undergo secondary reactions. Heterogeneous catalysis has the catalyst in a different phase from the reactants. Most common heterogeneous photocatalysts are transition metal oxides and semiconductors.

Highly active waste Waste for which management is a priority due to its very high levels of radioactivity above 370 GBq/kg.

Ion exchanger Polyelectrolyte (essentially as granular or bead forms suitable for packed bed) which has ions weakly attached to it; typically used is HOH type resin. When a liquid is passed across this bead, it will readily exchange the ions on the bead with ions of the same charge in the liquid. There are two types of resins, cation and anion resins. The cation resin has a positive ion attached in the hydrogen (the H part of HOH resin) form. The anion resin has a negative ion attached in the hydroxide (the OH part of HOH resin) form.

Low and intermediate level waste Comes from objects that are contaminated by contact with radioactive products but that are not radioactive in themselves. The radioactivity must be below 370 GBq/t (10 curies) for beta emitters and 3.7 GBq t^{-1} for alpha emitters (one hundredth).

MetalHexaCyanoFerrates (acronym MHCF) MHCF is a porous coordination polymer, consisting of metal cations M^{a+} and hexacyanoferrate anions $[\text{Fe}(\text{CN})_6]^{b-}$. A historical pigment, Prussian Blue (PB), $\text{Fe}_4[\text{Fe}(\text{CN})_6]_3$, is one of the MHCs. MHCF

has a porous network in its crystal constituted by two types of pores. One is the interstitial sites surrounded by a cubic cage consisting of eight metals bridged by 12 cyano-groups (CN). The other includes vacancy sites implemented by the introduction of $[\text{Fe}(\text{CN})_6]$ vacancies.

Photo-Fenton reaction Photochemical degradation method for organic contaminants using hydrogen peroxide and ferrous iron as a catalyst.

Reverse osmosis (acronym RO) The most exploited membrane operation. In reverse osmosis the feed water is pressurized (to about two to three times its osmotic pressure in order to overcome friction losses and to maintain an adequate flow rate) and forced through the membrane. Relatively pure water permeates through the membrane while a concentrated solution leaves as reject.

Introduction

A general principle applied in radiological protection and waste management is a preference for options that focus on “concentration and containment” of radioactivity over “dilute and disperse.” Thus liquid effluent decontamination consists—most of the time—in transferring radioactive contaminants more or less diluted in an aqueous stream into a solid concentrated waste. However, it is mandatory to first evaluate the safety of the decontamination process as well as of the disposal of the ultimate waste with regards to potential increase in operator doses and accidental discharges. When its activity (in Bq l^{-1}) is below the authorized limit, the clean effluent may be discharged to the environment possibly after a suitable decay period in tanks (IAEA-TECDOC-1638, 2010). The solid waste is generally immobilized in a stable matrix such as cement or even in glass for long-term storage or disposal.

Following that principle the decontamination of liquid effluents is based on a combination of abatement techniques which constitute the so called Best Available Techniques (OECD, 2003; IAEA-TECDOC-1336, 2003). The application of these BAT depend on the nature of the radioactive effluent (temperature, pH, salinity, presence of organic matter or suspended solids, level of radioactivity) but also on the volume to be treated. BAT are evaluated considering the maturity of the technology, its effectiveness, flexibility, costs, safety, reliability, operability and maintainability, occupational radiation exposure, secondary solid waste production and ease of decommissioning.

Usually the majority of radioactive liquid effluent originate from Nuclear Power Plants (NPP) and Fuel-Cycle Facilities (FCC) but it is worth noting that some are produced by the medical sector (99Tc for example), the research centers and other industrial plants all of them being characterized by a wide variety.

Much work has been done for developing optimized or innovative BAT but some challenges still remain in the management of potentially huge volumes of resulting solid residues (Zhang et al., 2019). Keeping in mind this issue, this article aims at giving an overview on the different abatement techniques used for the decontamination of aqueous effluents. Recent developments are highlighted. A case study based on the management of the huge amount of contaminated water produced after the Fukushima Daiichi NPP accident is presented as an illustration of the challenges faced in an emergency.

State of the art

The main radioactive nuclides which can be found in liquid effluents are relatively long lived activation products (60-Co, 55-Fe, 54-Mn, etc.), tritium, fission products (137-Cs, 90-Sr, 106-Ru, etc.), uranium (U), plutonium (Pu), minor actinides isotopes such as americium, curium.... Due to its extremely low radiological impact, aqueous tritium is released in seawaters in Europe. The difficulty in the removal of the radioactivity lies in the complex (when known) composition of the effluent which is often a mixture of complexing, oxidizing or reducing species, suspended solids, organic matters, salts, detergents...with concentrations often higher by orders of magnitude than the targeted radionuclide. It should also be mentioned that the effluent composition may be highly variable, requiring a flexible decontamination process. In recent decades, many studies have been made into the treatment of radioactive wastewaters in a safe method based on four groups of processes: (1) physico-chemical processes, (2) evaporation based processes, (3) biological processes and (4) electrochemical processes. This effort in R&D has drastically increased since the Fukushima disaster as illustrated by the annual cumulated number of citations of the topic radioactive effluent decontamination rising from less than 28 before 2011 to 147 in 2019 (Source: Web of Sciences). Due to their relatively low decontamination efficiency (decontamination factor ranging from 1 to $10^2 - 10^3$) except for evaporation (decontamination factor ranging from 10^4 to 10^6), these processes are effective methods for treating low and intermediate levels waste. Due to its high-energy consumption ($\sim 2.7 \text{ GJ m}^{-3}$ of purified water), evaporation is more attractive for the treatment of small volumes of highly active waste.

It must be pointed out that many methods rely on the combination of two or more of these particular processes.

Physico-chemical processes

Coprecipitation

Among physico-chemical processes, coprecipitation corresponds to the simultaneous precipitation of a normally soluble component with a macro-component from the same solution by the formation of mixed crystals, surface adsorption, occlusion or mechanical entrapment (also referred as inclusion) (see Fig. 1). The precipitates thus formed are of very fine sizes thereby requiring flocculation for faster settlement. The solids are separated and the remaining clear water can be filtered through a sand bed. Flocculation by either cationic or anionic flocculent (see Glossary) has shown improvement in removal of radioactive strontium (Rout et al., 2006). Due to its unique advantages (see Table 1) that makes it extensively used in nuclear industry despite its main drawback, that is the quantity of produced sludge (about 1% in volume of initial liquid volume). Most R&D efforts concentrated on selectivity through chemicals development and on the chemical engineering of the reactions in order to overcome this limitation (Flouret, 2013; Osmanlioglu, 2018). A combination of microfiltration and coprecipitation at a laboratory scale has shown good performances in strontium removal while indicating good potentialities for overcoming the major problems of solid-liquid separation and subsequent sludge concentration in chemical precipitation method (Wu et al., 2014).

From an industrial point of view, coprecipitation is routine for the cesium, strontium, manganese, cobalt, ruthenium and actinides decontamination. As an illustration, after pH adjustment, cesium may be removed by nickel ferrocyanide, strontium by barium sulfate, ruthenium by cobalt sulfide, cobalt by active charcoal.... Other reactants may be used depending on the nature of the effluent. For instance, taking into account the iron component of the waste, the pH adjustment by sodium hydroxide to 9–10.5 induce the coprecipitation of ferric hydroxide along with any radionuclides present in solution and especially actinides. The flowrate are usually high ranging from 100 to 500 m³ d⁻¹ or even more (IAEA, 1992). The decontamination factors obtained at industrial scale are modest (and far below those obtained at laboratory scale) about 100–1000 for cesium, 50–100 for strontium, 6–30 for ruthenium, 1000 for actinides, 100 for manganese. Despite these modest values, chemical precipitation processes appear to be very flexible and quite easy to control. Moreover, chemical precipitation is considered the cheapest process (20–50 times less expensive than evaporation).

Sorption/ion exchange

Solid sorption processes are based on either adsorption or ion exchange into porous matrixes capable to withstand radiation exposure during functioning and to minimize the dosimetry when handling the material. In the context of the Fukushima Daiichi NPP disaster and the necessity to separate most of the 137-cesium (Cs) and 90-strontium (Sr) release in the surroundings waters, a huge number of sorbents have been tested. Among them, much attention has been paid to mineral exchangers (See Glossary) like zeolite (Fang et al., 2017; Sachse et al., 2012), Prussian Blue (PB, ferric hexacyanoferrate) or PB analogs (Metal HexaCyanoFerrates, MHCF see Glossary) nanoparticles (Ishizaki et al., 2013), sodium nonatitanates (Villard et al., 2015), barium titanate (Guevar et al., 2017), ...thanks to their high selectivity and capacity for strontium or cesium. In particular, the outstanding performance of potassium-copper HCF (KCuHCF) towards adsorption of cesium ions has been highlighted as being due to the difference in free energy of the hydrated state between K⁺ and Cs⁺ in aqueous solution and to the vacancy site ratio (Takahashi et al., 2018). Nevertheless,

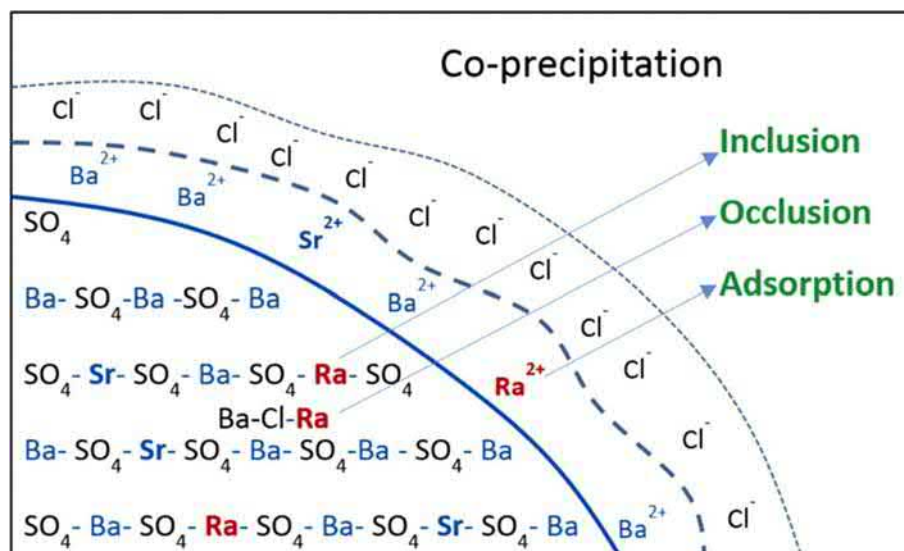


Fig. 1 Three mechanisms (inclusion, occlusion, and adsorption) of radium co-precipitation in binary solution with Ba-SO₄. Cl⁻ stands for any counter anion and radium can be replaced by strontium. Reprinted (adapted) with permission from Zhang T, Gregory K, Hammack RW and Vidic RD (2014) Co-precipitation of radium with barium and strontium sulfate and its impact on the fate of radium during treatment of produced water from unconventional gas extraction. *Environmental Science & Technology* 48: 4596–4603, Copyright (2014) American Chemical Society.

Table 1 Pros and cons of the main liquid effluent decontamination techniques.

Method	Advantages	Limitations	Radionuclide	Industrial scale DF
Precipitation	• Not sensitive to salt content • Adapted to large volume • Good decontamination efficiency • Simple • Relatively low cost operation	• Important volume of flocs/sludge • Dewatering system needed for the sludge • Chemical hazards due to the reagents • Efficiency depends on solid-liquid separation step	Cs, Sr, Mn, Co, Ba Ru Actinides	1–100 (10^4 for Cs) 6–30 1000
Ion exchange/adsorption	• Good decontamination efficiency • Adapted to a wide variety of species	• Suspended solid content • Salt content • Absence of nonionic radioactive species • Possible blocking of the bed • Cost of the material • Relatively low sorption capacity and rate • Regeneration and reuse of material	137-Cs 90-Sr 60-Co 65-Zn	1000–10,000 200–2000 10–2000 100
Advanced oxidation process	• Adapted to low contents of organic molecule and suspended matter • Complete oxidation	• Cost of the operation • Pre-mature technique	Cs Zr 60-Co	8–33 (lab scale) 7–22 (lab-scale) 18–43 (lab scale)
Membrane separation	• Reverse Osmosis (RO): removes all ions, high efficiency • Nano-filtration (NF): removes divalent and trivalent ions • Ultra-filtration (UF): separates dissolved salts from particulate and colloidal materials. Good chemical and radiation stability for inorganic membranes • Micro-filtration (MF): complete recovery	• RO/NF/UF: Fouling • organic RO/NF/UF membrane may be sensitive to radiation • MF sensitive to impurities in waste stream	137-Cs 90-Sr 241-Am, Pu, U 58-Co	200–3000 400 1600 500–57,000
Evaporation	• Very good decontamination efficiency and volume reduction provided that the inactive salt content of the stream is relatively low • Relatively high salt content with a wide heterogenous chemical composition	• Detergent content (foaming) • Salt content (precipitation and corrosion) • Organic compounds (risk of explosion) • Expensive (high capital, energy and maintenance costs) • Scaling (heat transfer) • Occupational radiation exposure of workers		10^4 – 10^6
Electrochemical methods	Electrodialysis (ED): • Specific radionuclide removal. • Relatively high salt content • Reasonable useful lifetime Electrodeionization (EDI): • Simple and continuous operation • Chemicals for regeneration completely eliminated • Non pollution, safety and reliability • Complete removal of dissolved inorganic particles • In combination with reverse osmosis pre-treatment, it removes more than 99.9% of ions from the water	• High resistance for the electric current, especially when the feed ionic concentration is low. • Do not remove uncharged impurities • Can form an explosive admixture • An overflow of electric current or concentration polarization can cause the formation of precipitates • Requires purification pretreatment • Its economic viability need to be further assessed	90-Sr, 137-Cs, 129-I, 106-Ru and Hg 137-Cs 90-Sr 60-Co	10 (lab scale) 200–1000 (lab scale) 50–250 (lab scale) 100–200 (lab scale)
Biological sorption	• Cost effective • No chemicals • Good decontamination efficiency (uptake and selectivity)	• Sensitive to impurities in waste stream • requires further dehydration • Dried organic matter to manage in disposal conditions	^{238}U , ^{137}Cs , $^{110\text{m}}\text{Ag}$, ^{60}Co , ^{54}Mn , ^{65}Zn , ^{226}Ra , ^{90}Sr	

most of these materials suffer from relatively slow sorption kinetics or low mechanical strength for practical applications in column operations. These drawbacks have led some research teams to develop new materials based on the grafting of MHCF nanoparticles on mesoporous silica grains (Michel et al., 2017) or monoliths (Grandjean et al., 2020) or on the precipitation of MHCF nanoparticles into the porous network of a geopolymer (See Glossary) (Petlitchaia et al., 2018). The main interest is to benefit not only from the structural properties of these matrixes but also from their compatibility with vitrification or cementation processes. On another hand, PB nanoparticles have been immobilized in cellulose nanofibers and gave satisfactory results in field studies on soil and seawater decontaminations in Fukushima (Vipin et al., 2016). Other promising development concerns graphene oxide (GO), which is an oxidized derivative of graphene. It has been proved that magnetic PB/GO nanocomposites encapsulated in low-cost calcium alginate microbeads exhibit a very high selectivity to Cs and could extract it in trace amounts from natural water or seawater (Yang et al., 2014). However, further studies are required for optimizing the GO by grafting chemically selective functional groups onto the materials surface to increase the sorption capacity of the material (Carey et al., 2018). More recently, a review on PB-carbon nanotubes-graphene nanocomposites concludes on their strong interest in 137-Cs extraction from aqueous media but points out that further studies on the toxicity and regeneration of these materials, on the mode of operation (batch or continuous)....need to be conducted before industrial applications (Rauwel and Rauwel, 2019). In particular, the recovery of these nanopowders has been validated at lab scale via magnetic extraction on attaching Fe₃O₄ nanoparticles to the graphene surfaces (Namvari and Namazi, 2014). The drawback of Fe₃O₄ is that the saturation magnetization is rather low for large-scale applications this potentially leading to some nanomaterial uncaught by the applied magnetic field.

That R&D aim at reducing the overall cost of the decontamination process while keeping a high selectivity towards competing ions, fast sorption kinetics and high sorption capacity. Nanomaterials, with unique physical and chemical properties, such as the nanosize effect, large specific surface area, high adsorption capacity and selectivity have naturally gained an extensive interest for radioactive wastewater decontamination. Nevertheless, a comprehensive review of such nanomaterials based applications has pointed out that their economic viability has not yet been taken into account or even overlooked in recent investigations (Zhang and Liu, 2020).

Novel 3D printing techniques (see Fig. 2) offers new potentialities for the fabrication of hierarchical zeolite monoliths for Cs and Sr removal from nuclear wastewater (Halevi et al., 2020). The 3D-printing process enables control over the dimensions and shapes of the zeolite embedded polymer, and over the degree of porosity and internal structure of the matrix. The monoliths retain the ion exchange properties of the zeolites powders and possess good mechanical stability. Again, it is worth mentioning that these processes have still to be validated at larger scales to assess their techno-economics interest. In particular, the capability of the materials to be easily regenerated for re-use in an environmentally friendly way should be assessed.

From an industrial point of view, ion exchangers based on organic resins have been used since the 1980s in NPP or FCC. The typical volume of the column reactor is between 0.1 and 10 m³ and the DF for 137-Cs and 60-Co were initially low about 15–20 and 4–24 respectively. More recently inorganic commercial exchangers Cs-Treat® (Fortum, Finland, www.fortum.com) based on ferrocyanide claim a DF for cesium usually between 1000 and 10,000 for evaporator concentrates and higher than 8 million for the Fukushima NPP contaminated waters. Last, titanium oxide based products SrTreat® and CoTreat® (Fortum, Finland) are highly selective ion exchangers respectively for strontium (DF higher than 165 million for the NPP contaminated waters when combined with precipitation and usually between 200 and 2000) and cobalt (DF between 10 and 2000).

Membranes

Membrane processes are pore-size dependent filtration processes including microfiltration (MF), ultrafiltration (UF), nanofiltration (NF) membranes and reverse osmosis (RO, see Glossary) with a dense membrane for ions in solution (see Fig. 3). A membrane is a layer of material which serves as a selective barrier for some components allowed passage into a permeate stream, whereas others, such as the radioactive isotopes are retained and accumulated in the retentate stream. Tubular ceramic MF and UF membranes have been used in the field of low or intermediate-level liquid effluent treatment. UF can be used for removal of all substances that are present in radioactive waste in colloidal or suspended form, which cannot be removed, by evaporation or ion exchange. To improve the efficiency of UF and MF, selective sequestering agents, which are often macromolecular ligands (e.g., polyacryl phosphonic, polyethyleneimine and polyacrylic acid), have been used to bind with radioactive ions. For instance copper hexacyanoferrate acts as a surfactant and form micelles with Cs in water, which adsorbs on UF or MF membrane. Ceramic NF membranes can be used to retain multivalent ions (or organic compounds of moderate molecular weight) from monovalent ion present in the solution (Ferreira-Esmi, 2014). It is worth noting that when combined with selective ligands such as resorcinarenes or calixarene, NF can selectively remove 137-Cs (Chitry et al., 2001). Nevertheless, the synthesis of such molecules is costly and increased attention has been given to reverse osmosis (RO) for cesium removal from wastewater at laboratory and pilot-scales (Wang and Zhuang, 2019). Despite the fact that RO membranes are composed of dense organic layers, recent study shows a limited degradation of a semi-industrial RO module under gamma irradiation with promising performances in Cs and Sr retentions (Combernoux et al., 2017). Nevertheless, organic membrane fouling and ageing under irradiation are still questioning the industrial viability of such processes leading to compromised performance and shortened lifespan of the membranes.

In order to overcome the problem of fouling while keeping a radionuclide removal efficiency as high as possible, adsorption or complexation combined with membrane filtration (NF/UF) has been proved to be more effective than membrane filtration used solely (Zakrzewska-Trznadel, 2003). Recently, a submerged membrane (MF) adsorption reactor system has shown its potential reliability for the Cs removal at large scale (Han et al., 2019).

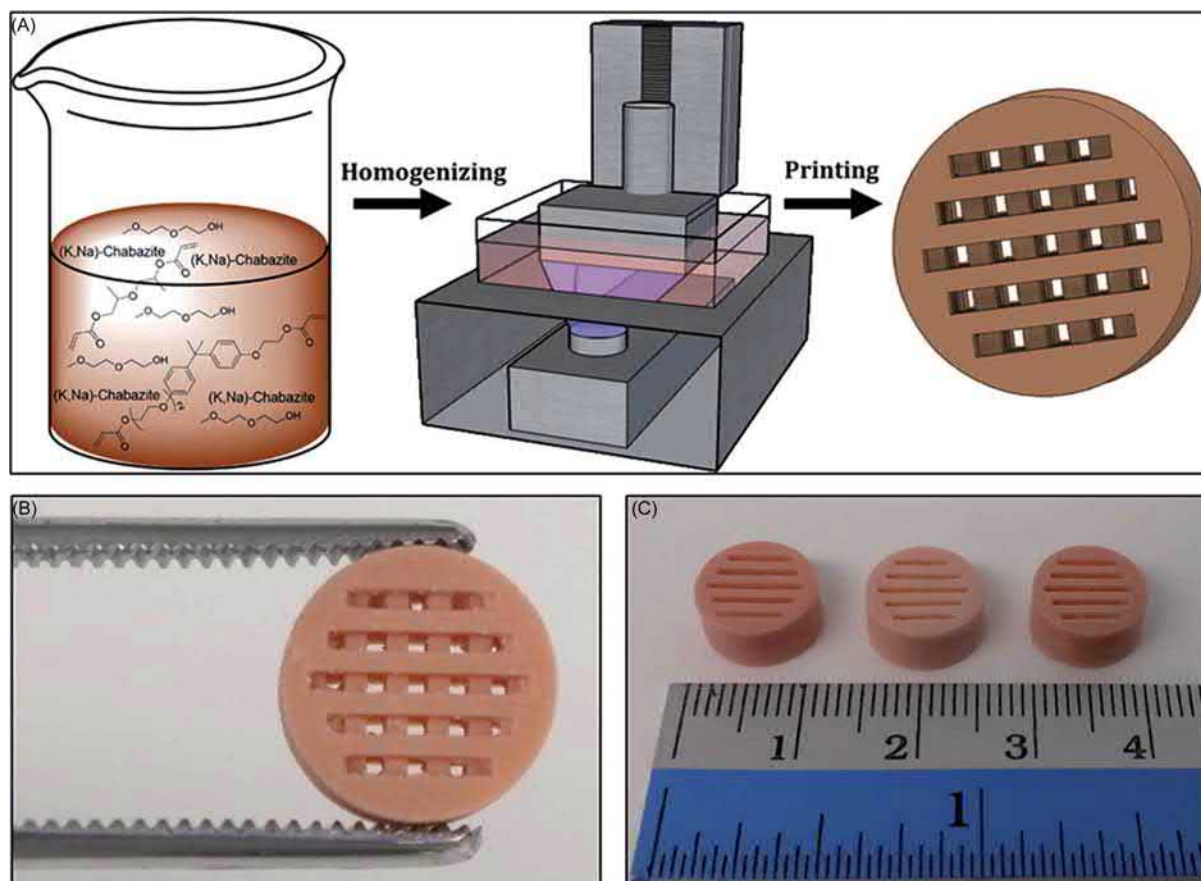


Fig. 2 (A) Schematic overview of the 3D printing process; first a dispersion of the zeolite was formed within the polymerizable monomers and porogenic solvent, then the formulation was 3D-printed by the DLP method. (B and C) The printed zeolite-embedded monolithic structures. Reproduced from Halevi O, Chen TY, Lee PS, Magdassi S and Hriljac JA (2020) Nuclear wastewater decontamination by 3DPrinted hierarchical zeolite monoliths. *RSC Advances* 10: 5766–5776, with permission from the Royal Society of Chemistry.

From an industrial point of view, membrane technologies (reverse osmosis alone or in combination with a ultrafiltration or microfiltration pretreatment, ultrafiltration and microfiltration alone) have been used for the treatment of liquid waste originated from NPP or FCC (IAEA, 2004). Examples of DF ranging from 500 to 57,000 are common, in particular with regards to colloidal 58-Co. These technologies have been proved to be industrially viable for processing alpha contaminated waste from fuel reprocessing, for processing separate batches of spent media transfer liquid (sluice water), for reducing the NPP's liquid effluent mixed fission and activation product activity.

Advanced oxidation processes

Advanced oxidation processes (AOP) involves in the two stages of oxidation process. The first stage of oxidation process is the creation of strong oxidants (hydroxyl radicals, $\text{OH}\bullet$) and the second stage is the reaction of these oxidants with organic contaminants in wastewater. Among different available AOP producing hydroxyl radicals, photochemical processes using hydrogen peroxide, H_2O_2 , and ultraviolet radiation (UV-C) and heterogeneous photocatalysis (see Glossary) using titanium dioxide, TiO_2 , and ultraviolet radiations (UV) are the two most popular technologies for the decomposition of low concentration organic waste (see mechanisms on Fig. 4). However, UV-C photo-Fenton AOP (see Glossary) using ferrous ions is more adapted for the decomposition of relatively high concentrations of oxalic acid produced by the decontamination of nuclear power plant (Kim et al., 2019a). For other types of radioactive effluent like those generated by the nuclear fuel treatment, dissolved radionuclides (like 60-cobalt, Co(II) , or uranium, U(VI)) can be complexed by organic molecules (like ethylenediaminetetraacetic acid, EDTA) at trace concentrations. The $\text{OH}\bullet$ produced in situ by $\text{H}_2\text{O}_2/\text{UV-C}$ or TiO_2/UV methods promote the degradation of the organic molecules leading, in the case of dilute U(VI) -EDTA complex, to the reductive deposition of the uranyl ions on the TiO_2 surface (Chen, 1997). In the case of Co(II) -EDTA complex, the degradation of EDTA leads to either the adsorption of cobalt onto TiO_2 particles and/or to the precipitation of cobalt cation Co^{2+} depending on the pH (Rekab et al., 2015). At a lab scale, the obtained DF values were in the range 18–43 for 60-Co. In almost the same conditions, Seshadri and coworkers obtained DF values ranging from 7 to 22 for Sr and from 8 to 33 for Cs (Seshadri et al., 2008).

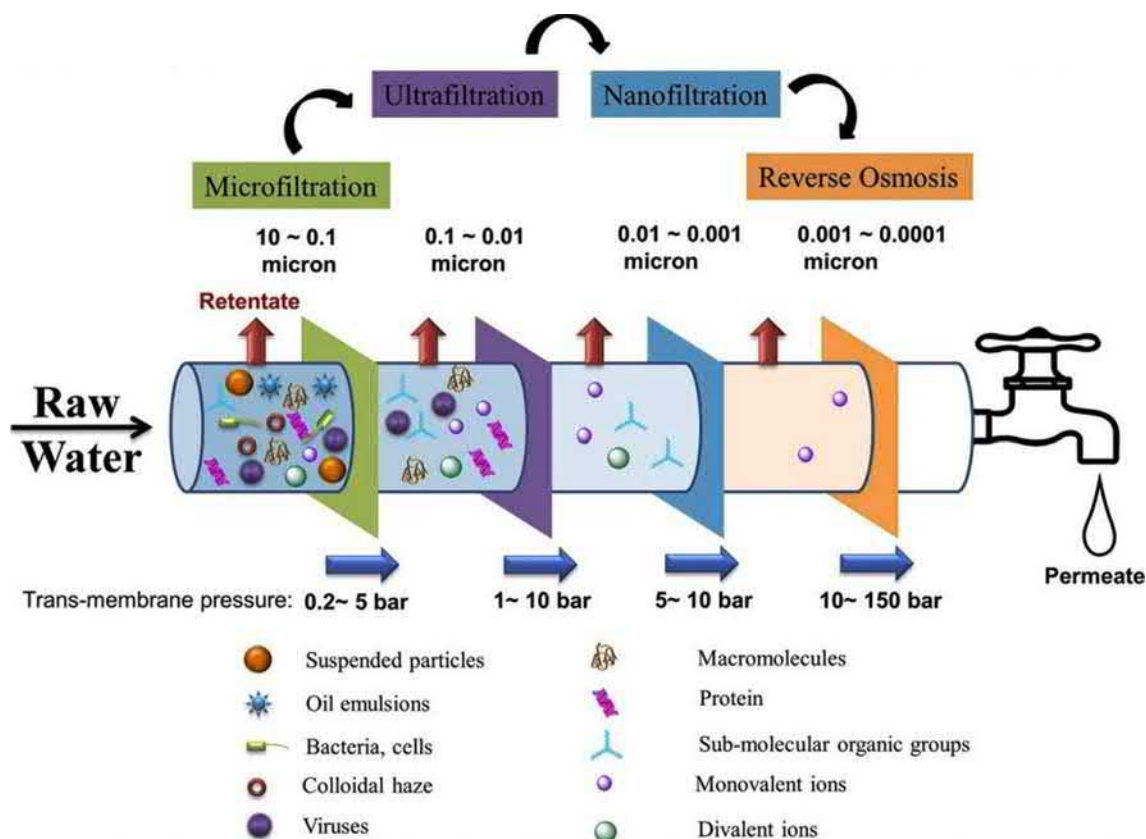


Fig. 3 Various membrane filtration technologies. In membrane filtration, the use of a pressure difference across the membrane, a transmembrane pressure is an essential driving force for separation. From Nordic Membrane Laboratory website (www.nordicmembrane.com).

It should be noted that the advanced oxidation process is still a pre-mature process and process validation at large scale is definitely needed.

Evaporation

Evaporation is a proven method for the treatment of liquid radioactive waste with nonvolatile radionuclides and salts providing both high decontamination factors and good concentration of salts and radionuclides in bottom concentrate (see Fig. 5). Although it has relatively high operating costs (high-energy consumption), its large volume reduction (of about 100), high decontamination factor (10^4 – 10^6), and the stability in operation make it more suitable than other techniques for processing intermediate- and high-level radioactive waste liquid. However, flowrate are usually lower than those used for other physico-chemical methods. As an illustration mature technologies based on two stages evaporator claim that decontamination factors up to 5×10^7 are achievable for non-volatile radionuclide (see NUKEM website). Another innovative technology has been recently developed and validated at relatively large scale; by continuously controlling the drag/entrainment effect due to the vapor production during evaporation, the process is capable of separating the water molecules from the others. This yields to DF for ^{137}Cs as high as 10^6 at pH 7 and 3.3×10^5 at pH < 7 even in the presence of volatile compounds or salts (see WoW website).

Clean water is removed as vapor leaving behind non-volatile components such as salts and most radionuclides. Nevertheless, the entrainment of radioactive droplets with vapor stream is inevitable, which contaminates the condensate and thus reduces the DF. Much work concentrated on that issue and recently the working principle and operational characteristics of a decontamination tower based on sieve trays for washing the vapor have been published (Deng and Huang, 2017). However, it is worth noting that Westinghouse Electric Company LLC, in the frame of the general design assessment of AP1000 nuclear power plant, claimed that using ion exchange and filters offers a simpler and safer option that will still effectively control discharges of radioactivity due to radiation exposure risks linked to evaporation (Environment Agency, 2011).

Electrochemical processes

Electrochemical processes, in which the electric potential is the driving force, encompass electrodialysis (ED, see Glossary) and electrodeionization (EDI, see Glossary) and have been developed in the late 1950s (see Fig. 6). They are carried on with the use

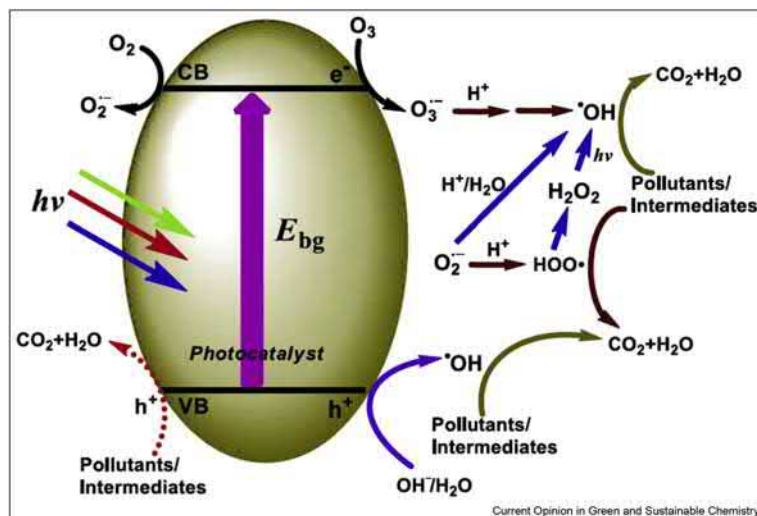


Fig. 4 Illustration of some of the events taking place in UV/visible light-irradiated semiconductor photocatalysts (like TiO_2) in the presence of ozone and molecular oxygen in aqueous media. Pollutants and intermediates degrade into H_2O and CO_2 . Reprinted from Serpone N, Artemev YM, Ryabchuk VK, Emeline AV and Horikoshi S (2017) Light-driven advanced oxidation processes in the disposal of emerging pharmaceutical contaminants in aqueous media: A brief review. *Current Opinion in Green and Sustainable Chemistry* 6: 18–33, Copyright (2017) with permission from Elsevier.

of ion-exchange membranes, which act as selective barriers. The main advantages of ED lie in the fact that the electrochemical elution eliminates the need for addition of chemicals to elute the targeted radionuclide from the ion exchange resin and very small volumes of the eluting solution can be used in a complete recycle mode in order to minimize the total volume of the radionuclide eluate. DF as high as 10^4 were obtained for the removal of cesium from spent ion exchange resins at the laboratory scale (Bontha et al., 1996). The main drawback ED lies in a phenomenon known as concentration polarization (due to Donnan exclusion) that decreases cell efficiency and lead to a very high-energy consumption especially when dealing with huge volume of low radioactive wastewater. In order to eliminate this disadvantage, EDI has been developed in the 1950s but has suffered from a slow advance likely due to its complex functional kinetics. A comprehensive review (Alvarado and Chen, 2014) points out the primary advantages of EDI that include residue free operation, low process cost and exemplary efficacy in ionic separation and pave the way towards new applications. Basically, the working principles of EDI lie in three simultaneous actions, i.e., (i) exchange of salt ions in the solution with mobile ions in the ion exchange resin; (ii) the continuous transport of ions to the concentrate stream through the ion exchangers and membrane layers; and finally (iii) the continuous regeneration of ion exchangers by hydrogen and hydroxyl ions in situ produced as a result of dissociation of water under an electric field. Furthermore, electrochemical treatment processes can be controlled remotely and automatically, are tunable via system parameters such as operating voltage, and avoid direct exposure to radioactive wastes. Thus, it is no surprise that these electrochemical treatments have rapidly been implemented throughout the world to treat aqueous wastes containing 137-Cs, 60-Co, nitrate wastes resulting from nuclear fuel fabrication plants, and organic wastes such as scintillation cocktails (El-Aziz and Khalifa, 2016). As an illustration EDI has been successfully applied for the purification of primary coolant of a NPP reaching DF values as high as 1000 (Yeon et al., 1999).

As an illustration, the electrically-switched ion exchange is a Cs^+ separation technique where the exchange of Cs^+ ion can be controlled easily by electrode potential. It is capable of overcoming a shortcoming of MHCF (and especially copper MHCF, CuHCF) based processes linked to the fact that small sized CuHCF particles will result in second contamination to water and shorten powdery CuHCF's service life. Moreover, pure CuHCF suffer from low surface area and a poor conductivity thus leading to slow Cs^+ exchange rate and small Cs^+ distribution coefficient. To overcome these limitations a recent process has been developed based on copper hexacyanoferrate/multiwalled carbon nanotube (CuHCF/MWCNT) hybrids. In addition to the remarkable Cs^+ selectivity and removal performance, CuHCF/MWCNT hybrids can be recycled and show a high stability during Cs^+ disposal (Zheng et al., 2017).

An active R&D is still in progress around the world. For instance, a very recent method for the decontamination of nuclear plant cooling water has been developed at MIT based on shock electrodialysis (SED) (Alkhadra et al., 2020). Like ED, SED passes an electric current perpendicular to flow in order to guide ions out of a purified stream of water and into a concentrated brine to be disposed of. However, the separation of fresh and brine does not make use of a physical barrier like a membrane. Instead, the streams are separated by a stable concentration "shock" that arises from passing overlimiting current between the electrodes. This shock is stable in flow and possible only because of the SED device's carefully chosen porous media, which must have pores small enough to prevent the formation of vortices and have enough surface charge to promote the surface conduction of oppositely charged ions. This technique is used to selectively, continuously, and efficiently remove cobalt and cesium from a feed of dissolved lithium, cobalt, cesium, and boric acid commonly found in light-water reactors and in other nuclear processes. In general, the energy

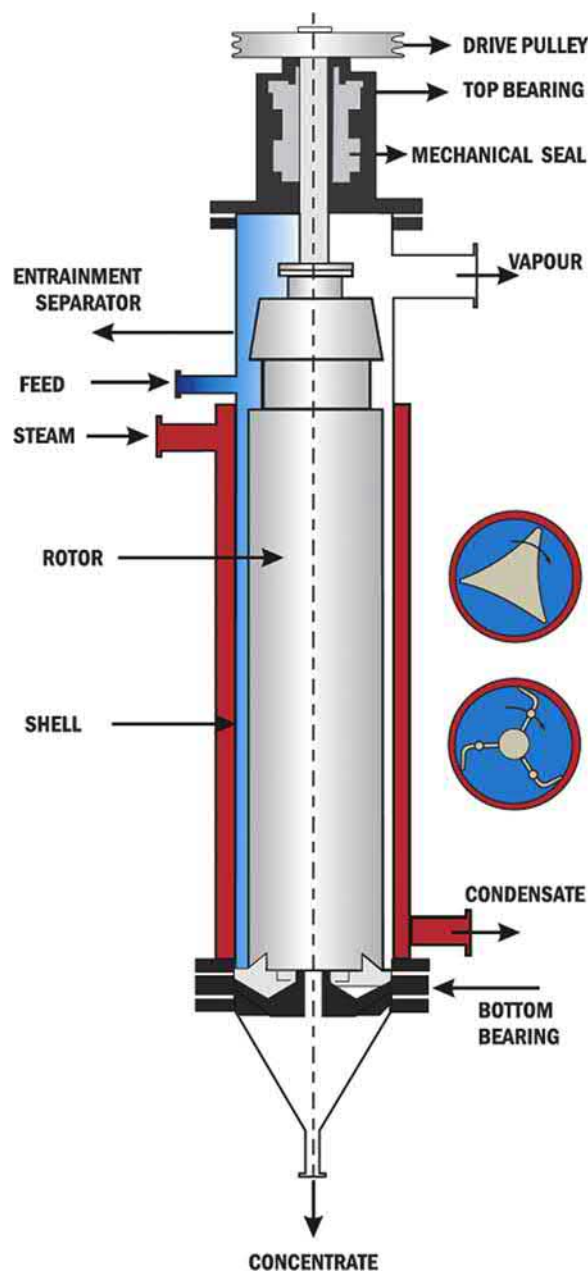


Fig. 5 Principle of an agitated thin film evaporator. From the Concept Process Equipment website (www.conceptprocess.com).

consumed during this process (ranging between 1.76 and 4.8 kW h m⁻³) is low because only charged species are targeted and virtually no energy is expended removing boric acid, the most abundant species in solution.

Although promising that technique still has to be validated at larger scales not only in a technical point of view but also regarding the economics of the whole value chain including the saved cost of disposal or storage.

Biological processes

Bioremediation strategies are gaining significant interest owing to their relative cost effectiveness and eco-friendliness. Various types of microorganisms including fungi, algae, cyanobacteria, and phytoplankton have been reported to retain radionuclides in harsh ionizing environment (McGraw et al., 2018; Chicote et al., 2005). For instance a microalga, *Coccomyxa actinabiotis*, capable of withstanding ionizing radiation doses up to 20,000 Gy, rapidly accumulates most of the metallic radionuclides contained in nuclear effluents including ²³⁸U, ¹³⁷Cs, ^{110m}Ag, ⁶⁰Co, ⁵⁴Mn and ⁶⁵Zn with decontamination rates above 90%. The process feasibility has been demonstrated at real-scale. Once dried, the algae volume was reduced by 90% leading to a volume of radioactive waste at least 100

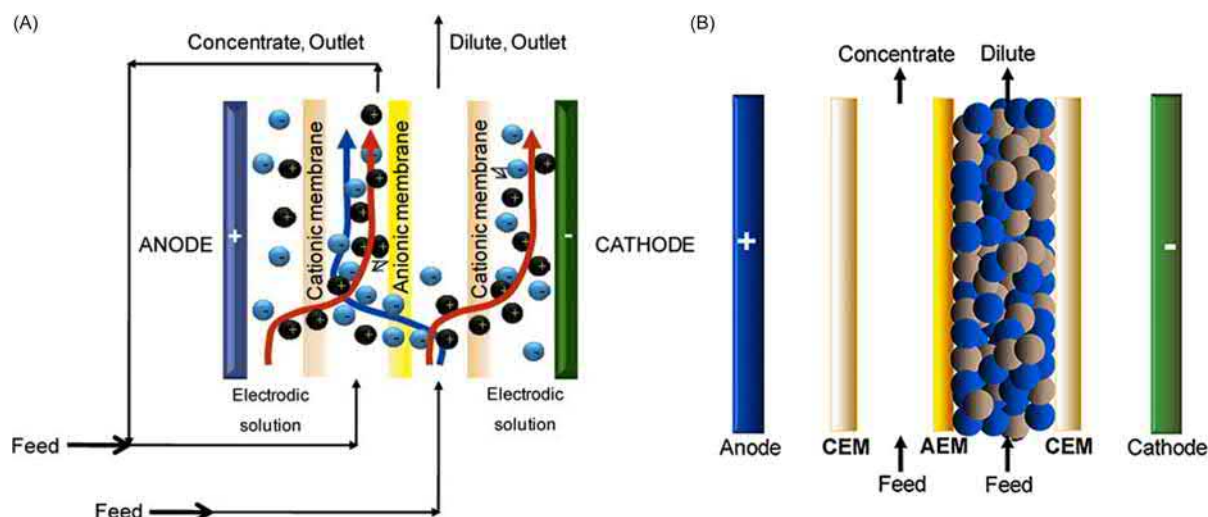


Fig. 6 Representation of (A) an electrodialysis process and (B) an electrodeionization cell. The spheres represent the ion exchange resin; *AEM*, anionic exchange membrane; *CEM*, cationic ExchangeMembrane. Reprinted from Alvarado L and Chen A (2014) Electrodeionization: Principles, strategies and applications. *Electrochimica Acta* 132: 583–597. <https://doi.org/10.1016/j.electacta.2014.03.165> (web archive link,) with permission from Elsevier.

lower than that of resins (Rivasseau et al., 2013). On another hand, the remarkable ability of the cyanobacterium *G. lithophora* to retain high activities of ^{226}Ra and ^{90}Sr in comparison with other organisms makes it an attractive candidate for bioremediation applications in the decontamination of low-salinity radioactive wastewater streams. Nevertheless, complete development of *G. lithophora*-based $^{90}\text{Sr}/^{226}\text{Ra}$ bioremediation need further studies notably on its tolerance to ionizing radiation and to other metal-loids present in the waste streams (Mehta et al., 2019). Recently, the combination of magnetic separation and bioaccumulation in a micro-alga, *D. armatus*, has been proved to be promising for ^{137}Cs removal from radioactive wastewaters containing competing cations (Kim et al., 2019a, b). Nevertheless, validation at larger scale is still lacking.

The following Table 1 summarizes the main pros and cons of these techniques from the above references and from reference reviews (IAEA, 2001; Abdel Rahman et al., 2011; Rana et al., 2013; Barré et al., 2018).

A case study: The decontamination of cooling waters in Fukushima Daiichi nuclear power plant

A brief history of a disaster

At 3:42 pm, Friday 11 March the first tsunami wave hit the Fukushima No. 1 nuclear plant, followed by a second 8 min later. These submerged and damaged the seawater pumps for both the main condenser circuits and the auxiliary cooling circuits, notably the residual heat removal (RHR) cooling system. They also drowned the diesel generators and inundated the electrical switchgear and batteries, all located in the basements of the turbine buildings thus causing three catastrophic meltdowns of the cores of units 1–3. At 7:03 pm a Nuclear Emergency was declared. Efforts to bring the overheating reactor units of Fukushima Daiichi back under control involved the injection of water directly from the sea, containing a range of substances as well as salt. This water flowed across the molten reactor cores and picked up a range of radionuclides before emerging from the damaged reactor vessels to accumulate in the building basements. The result was an enormous and increasing inventory of polluted and highly radioactive water that represented a major hazard in itself. Unable to prevent the water from leaking out of the reactor vessels, Tokyo Electric Power Company (Tepco) reduced the amounts in the basements by pumping to on-site waste management facilities and then establishing a system to decontaminate the water so that it could be re-injected, using the basements as a part of a closed cooling loop.

The solution, designed to respond to the emergency and support ongoing operations was constructed in record time—two and a half months rather than the usual 3 years. This solution came in the form of the Actiflo®-Rad process, developed in collaboration with AREVA, VEOLIA and the CEA and accepted by TEPCO on April 8, 2011 (Prevost et al., 2012). It relies on a combination of proven nuclear coprecipitation technologies and coagulation, flocculation and sedimentation technologies used for over 20 years in the water treatment industry (see Fig. 7). The saline effluent produced by the upstream oil separation and cesium preadsorption stages is mixed beforehand with differently charged reagents to trap the target radionuclides. This effluent is then transferred into four compartments in series, the first of which is the coagulation tank, where iron chloride (FeCl_3) is added and the pH adjusted. The second compartment is the mixer-contact or where the micro sand is added and the third, called the maturation tank, is where a polymer is added to improve flocculation. The solid precipitate and colloid matter settles in the fourth tank, which contains an inclined plate clarifier (filter) to improve the solid-liquid separation process. The sludge produced is pumped out at the bottom of this last tank. Two stages of this type were set up in series at Fukushima Daiichi, with the only difference between the two stages

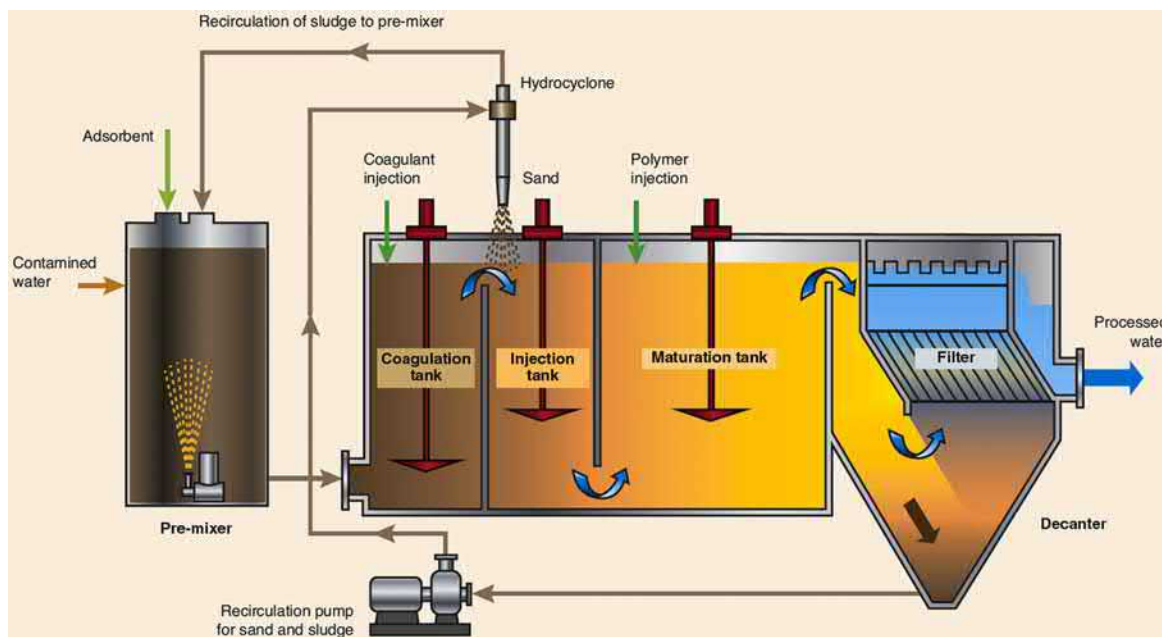


Fig. 7 Actiflo-Rad[®] operating principle.

being the absence of the micro sand injection step and the hydrocyclone in the first stage. The process was integrated into an overall system of treatment processes, including an oil separator and pre-treatment process using cesium adsorption upstream and reverse osmosis and an evaporator to desalinate the decontaminated water downstream. After the feasibility tests, a program was conducted to optimize the operating conditions, during which the reagents in the Fukushima seawater were qualified (nickel and potassium hexacyanoferrate to adsorb the cesium, barium chloride to coprecipitate the strontium with barium sulfate) and the optimum contact times defined. The process effectively started on June 17, 2011 and ran until the end of September of the same year. The Actiflo[®] Rad unit reduced the radioactivity of the wastewater by a factor of 10,000 (DF of 10^4 for cesium over the entire period), even with significant salt content (seawater diluted twice) at a flow-rate slightly below the design value of $50 \text{ m}^3 \text{ h}^{-1}$. Between June and September 2011, the system treated 80,000 m^3 of water with a radioactivity of 1 GBq/kg for a total decontamination of 80,000 TBq. About 600 m^3 of waste sludge was produced in total and stored in shielded containers before possible vitrification (Amamoto et al., 2016).

In August 19, 2011 a supplementary and simplified active water retrieve and recovery system (SARRY) plant to remove cesium was installed and commissioned. The system incorporated adsorption towers with ion-exchange resins to grab and immobilize cesium radionuclides (^{137}Cs , ^{134}Cs) present in the water. Over the following months, the system was improved, and a return loop with desalination added so that treated water could be used to cool the reactors. By the end of May 2012, the system had treated 177,000 m^3 of water, and, by storing the treated water in huge tank farms on site, managed to lower the flooding levels in the buildings to below the water table. But with nearly 160,000 m^3 of highly-active water stored in tanks around the site, and the continuing use of some fresh water for the 500 m^3 of daily cooling injection into reactor units 1, 2 and 3, the situation was clearly unsustainable.

Therefore, in March 2013, TEPCO started to operate an Advanced Liquid Processing System (ALPS) to treat all of the contaminated water by removing 62 radionuclides. These are not all of the radionuclides present—researchers have identified more than 1000—but are the ones judged to approach the closest to regulatory maximum levels for seawater, according to Japanese reactor regulations. The cesium is first removed from the contaminated water—reactor-cooling water and ground water flowing in the buildings—using SARRY II plant, from which the freshwater component for further reactor cooling is separated. The remaining wastewater from the RO process is then treated by ALPS. Two notable nuclides that will not be substantially removed by this system are tritium (see *infra*) and carbon-14 but the maximum concentration of this nuclide in the treated water is only 11% of the regulatory standard required by Japanese law (2000 Bq/l) for discharge into the Ocean (see METI website). Water treatment processes in general do not eradicate every last active molecule; they are designed to conform to discharge limits set by the national regulator (see METI website) that vary by country and radionuclide. These guidelines bring radioisotope concentrations post-treatment close to, or below, the limits of detection. TEPCO has chosen Toshiba as the prime contractor for the so-called “multi-nuclide removal system” (ALPS) created by EnergySolutions. The core of the treatment system consists of adsorption stage, in which ion exchange occurs, with extra pretreatment beforehand to remove precipitates and organics. First, a ferric flocculation process that targets ruthenium isotopes; second, a co-precipitation process that removes calcium, magnesium and strontium as carbonates. In both stages, a crossflow ultrafiltration step forces water through a porous ceramic pipe (hole diameter: $0.02 \mu\text{m}$), trapping solids and organics whilst allowing water to pass through. Flocculates, precipitates and organics, which accumulate as sludge, are pumped into 1.9 m

tall by 1.6 m diameter cylindrical radwaste storage containers. In the adsorption stage, water flows through a series of adsorption towers. Because radionuclide concentrations are relatively low, their media are not worth sluicing out; instead, the entire tower is replaced once the resins are spent (on about a monthly basis, or 15 times a year, according to TEPCO).

On the evening of January 24, 2020, the total volume of contaminated water treated by ALPS at the Fukushima Daiichi NPPs surpassed the milestone of 1 million t. As of March 12, 2020, some 1.19 million cubic meters of treated water are stored within 979 tanks on the plant site. The total tank storage capacity will amount to approximately 1.37 million cubic meters by the end of 2020 and all the tanks are expected to be full around the summer of 2022. Therefore, a decision on the disposition path for the stored treated water—after further treatment as needed—should be taken urgently, considering safety aspects. Two options for controlled disposal- vapor release and discharges to the sea- are under examination since both are technically feasible.

Particular case of tritium contaminated water

It has become common practice for nuclear plants around the world to dump into the sea water tainted with tritium, a radioactive isotope of hydrogen that emits very low beta particles and cannot be separated from water under the current purification system, after dilution, based on standards set by each country. There's no hazard posed by external radiation of tritium, which is similar to water and does not secrete into any particular organs in fish or humans through the food chain (ASN, 2019). In the beginning of 2020, a panel of experts advising Japan's government recommended releasing the tritium-tainted water into the ocean after dilution in order to meet the standards set under the law. A simulation of what would happen to the people in Fukushima Prefecture has shown that if all of the approximately 860 trillion Bq of tritium-tainted water were released slowly over 1 year after dilution, the environmental impact would be between 0.052 and 0.62 μSv a year. This dose is significantly less than the one received from natural background radiation, diagnostic X-rays or air travel. For example, in 2015, the La Hague nuclear fuel reprocessing plant in France discharged 13.7 quadrillion Bq of tritium water into the sea.

All possible measures have been exhausted to contain the water, including building an "ice wall" or drilling wells to keep groundwater from entering the reactor buildings, but almost 9 years after the disaster, those measures alone are not enough. It is probably time to put an end to the issue and gain acceptance from the Japanese people for a sea discharge just like at other nuclear plants. Nevertheless any decision to dispose of the waste water into the sea will require intense dialogue and confidence of local fishermen, residents and neighboring countries such as South Korea.

Concluding remarks

Due to the increasingly stringent regulatory frameworks for radioactive waste management, decontamination is a key step for engendering public confidence in the industry. As an illustration, the French group Electricité de France (EDF) which operates the nuclear reactors in France has reduced the radioactivity of its discharges, with the exception of tritium and carbon 14, by a factor of more than 100 over the past 20 years.

To date, conventional techniques such as chemical precipitation, evaporation, adsorption, ion exchange and membrane separation have been industrially applied with their own advantages and limitations. Among them, adsorption is probably the most widely used technique for removing various soluble matters from wastewater. The current R&D focus on novel techniques that take benefit from the combination of the above processes or from new advances in technologies (3D printing, nanomaterials, etc.). The main objective is to maximize the selectivity and capacity while minimizing the duration (optimize sorption kinetics) of the treatment process for limiting both the volume of final waste and the dose for the working on-site staff. For instance, a combination of membrane separation and (co)precipitation or new nanomaterials like carbon based nanocomposites exhibit promising performances. Unfortunately, most of the studies have not been validated at large scale and their economic viability is then questionable. Moreover the majority of the publications does not consider the capability of the final waste (e.g., used (nano) adsorbent and ion exchange material loaded with high-concentration radioactive nuclides) to be recovered, optionally regenerated and finally properly integrated into the waste management system. There is thus a need to develop a specific Life Cycle Assessment of the whole value chain from the radioactive effluent to the final disposal. This systemic approach must take into account issues as human exposure to radiation (direct exposure of on-site working staff and indirect exposure due to the release of improperly treated radioactive wastewater), safety (classical hazards like aerosols, chemicals, nanoparticles, etc.), environmental impact and costs (treatment, packaging, transportation and disposal). Such a multi-criteria approach should enable the design of novel decontamination techniques capable to be industrially implemented.

In a context where nuclear energy should play a growing role in the future for seriously tackling the challenges of the climate change, the quest for such processes is thus likely to be pursued in the next decades.

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ILW conditioning and performance

Samuel A. Walling, Laura J. Gardner, and Neil C. Hyatt, NucleUS Immobilisation Science Laboratory, Department of Materials Science and Engineering, University of Sheffield, Sheffield, United Kingdom

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Glossary

Cold-cap A cold melt surface that can minimize volatilization of radionuclides.

Skull A protective coating on the internal surface of a crucible.

Waste product Refers to a conditioned waste stream that requires further processing prior to long-term storage and disposal.

Waste streams Refers to an individual waste type.

Wasteform Refers to a conditioned waste stream that would be suitable for long-term storage and disposal.

Abbreviations

ANSTO Australian Nuclear Science and Technology Organisation

AVM Atelier de Vitrification

BADGE Bisphenol A diglycidyl ether

BFS Blast furnace slag (also referred to as ground granulated blast furnace slag (GGBS)).

C-(A-)S-H Calcium (aluminum) silicate hydrate

CBPC Chemically bonded phosphate ceramic

CEA Commissariat à l'Energie Atomique et aux énergies alternatives

CSA Calcium sulfoaluminate cement

C-S-H Calcium silicate hydrate

EDTA Ethylene diamine tetraacetic acid

HIPing Hot isostatic pressing

HLW High level waste

IAEA International Atomic Energy Agency
 ICV In-can vitrification (also used for In-container vitrification).
 IER Ion exchange resin
 ILW Intermediate level waste
 IRIS Incineration Research Installation for Solid waste
 JHCM Joule heated ceramic melter
 LLW Low level waste
 MK Metakaolin
 MPC Magnesium phosphate cement
 N-(A)-S-H Sodium aluminum silicate hydrate
 NNSS Nevada National Security Site
 NPP Nuclear power plant
 PC Portland cement
 PCM Plutonium contaminated materials
 PE Polyethylene
 PFA Pulverized fuel ash (also referred to as fly ash).
 PMF Plasma melting facility (located at Kozloduy NPP, Bulgaria).
 SHIVA Système Hybride d'Incineration Vitrification Avance
 THOR[®] Thermal organic reduction
 UNGG Uranium naturel graphite gaz
 WVP Waste vitrification plant

ILW conditioning and performance

Intermediate level wastes (ILW) typically contain long-lived radionuclides at a concentration, and with a half-life, that will require long-term disposal in a facility which provides more containment than near-surface disposal (e.g. in an engineered repository tens to hundreds of meters below ground). The definition of ILW wastes varies from country to country, however, the International Atomic Energy Agency (IAEA) defines ILW as material with a higher activity than low level waste (LLW), but that does not require active cooling in storage conditions ([International Atomic Energy Agency, 2009](#)). The typical lower limits of activity are 4 kBq g^{-1} for long-lived alphas, and tens of kBq g^{-1} for long-lived beta/gammas ([International Atomic Energy Agency, 2009](#); [Radioactive Waste Management Limited, 2017](#)). Most notably, the United States does not have an ILW category, however, under IAEA classification many wastes defined under US law as either high level, transuranic or low level may be recognized as ILW in other jurisdictions, and, as such, may be cited as examples throughout this article. Intermediate level wastes contain a diverse range of chemistries, physical properties and radionuclide inventories, which has resulted in a wide application of immobilization and encapsulation technologies to these wastes over many decades of successful treatment.

The performance of ILW wasteforms, especially in disposal conditions are highly dependent on the encapsulation or immobilization matrix and the engineered barrier systems to be employed within a repository (geological, or near-surface). The long-term performance of ILW wasteforms are not necessarily primary factors in choice of immobilization matrix in the same manner as highly active wastes (containing high concentrations of long-lived radionuclides). Many of the shorter/medium-lived isotopes predominant in ILWs will have substantially decayed prior to the eventual breakdown of geological repository barriers, reducing the risk of potential long-term radiotoxicity. ILW wasteforms are typically packed within an outer container (e.g. steel drums or concrete boxes), which will degrade before substantial interaction between wasteform and groundwater can occur. Post-closure repository conditions, whether it is a cementitious, clay, or salt backfill, will dominate near-field conditions long into the future, either limiting water ingress, or conditioning groundwater to limit radionuclide mobility. These repository conditions are key to limiting radionuclide mobility for ILW, whereas HLW are expected to remain radiotoxic long after the degradation of repository engineered barriers, with a corresponding greater emphasis on their long-term durability.

Here, we detail some of the most common, operational and emerging treatment technologies for ILW treatment, ranging from conventional cementation and bituminization, through to a wide range of thermal treatment technologies, including a brief indication of their longer-term performance within an engineered waste repository.

Cementation

Cementation of nuclear wastes typically refers to the physical encapsulation and solidification of wastes using an inorganic binder which is capable of setting at ambient temperatures. Most often this refers to the usage of a blended Portland cement grout, either

mixed waste to be immobilized or poured around solid materials (e.g. compacted drums), then left to set, typically within a stainless steel drum or box.

Portland cements

Portland cement (PC) is the most widely used and manufactured cement worldwide, with widespread usage within the construction industry and a relatively well understood chemistry. It is produced via the calcination of clays (alumino-silicates) and limestone (CaCO_3) at $\sim 1450^\circ\text{C}$ before being ground into a fine clinker, with the addition of gypsum ($\text{CaSO}_4 \cdot 2\text{H}_2\text{O}$) or anhydrite (CaSO_4) as setting regulators. To this powder, limestone or other minor components may be added according to national or wider legislation (e.g. European standard EN 197-1). Although many different varieties of Portland cement exist, a typical Portland cement clinker consists principally of alite (Ca_3SiO_5 , 50–70%), belite (Ca_2SiO_4 , 15–30%), ferrite ($\text{Ca}_4(\text{Al,Fe})_2\text{O}_7$, 5–15%) and aluminate ($\text{Ca}_3\text{Al}_2\text{O}_6$, 5–10%). Minor oxides such as Na_2O , K_2O and MgO usual constitute less than 5% of the clinker (Taylor, 1997; Atkins and Glasser, 1992).

Portland cement gains strength primarily through hydration of alite and belite (with belite reacting more slowly than alite), but also ferrite and aluminate phases. These clinker solids are unstable in the presence of water, and within seconds of mixing produce a highly alkaline solution which accelerates dissolution-precipitation reactions. Alite and belite react to form a poorly crystalline calcium silicate phase (typically denoted as 'C-S-H') which is the principal strength giving phase within Portland cements, with excess calcium precipitating as $\text{Ca}(\text{OH})_2$. The chemical composition of this 'C-S-H' phase is somewhat flexible, accommodating a Ca/Si ratio from ~ 0.7 – 1.7 , giving it a unique pH buffering capability and ensuring alkaline conditions prevail within the cement pore solution (along with $\text{Ca}(\text{OH})_2$). The aluminate phase is highly reactive, rapidly forming hydrated calcium aluminate phases which give early stiffening of the paste (prior to C-S-H precipitation), but is controlled by the supply of sulfate into solution from gypsum and anhydrite. In the presence of gypsum, ettringite is formed ($\text{Ca}_6\text{Al}_2(\text{SO}_4)_3(\text{OH})_{12} \cdot 26\text{H}_2\text{O}$), typically converting to monosulfate over time ($\text{Ca}_4\text{Al}_2\text{SO}_4(\text{OH})_{12} \cdot 6\text{H}_2\text{O}$) as more aluminate reacts. Hydration of ferrite typically follows the same pathways as aluminate, forming an iron substituted ettringite, although more slowly.

Various supplementary materials can be combined with Portland cement, finding routine usage in both construction and the nuclear industry. These can have cementitious properties, affecting the hydration of Portland cement in beneficial ways such as altering the chemistry of C-S-H, increasing fluidity, lowering the heat of hydration, and reducing porosity. Industrial sources typically consist of blast furnace slag, fly ash and silica fume, but also include burnt shales. Natural materials include pozzolans such as volcanic tuffs and natural zeolites, along with those which have undergone additional processing including rice husk ash and calcined clays. Most of these materials have a high proportion of glassy aluminosilicates (illustrated in Fig. 1), with varying levels of CaO, MgO, SiO_2 and Fe_2O_3 , which can readily react within the highly alkaline, Ca rich pore solution of Portland cements (Taylor, 1997; Lothenbach et al., 2011). The use of these supplementary cementitious materials often decreases the Ca/Si ratio within the C-S-H, and results in Al substitution, forming C-(A)-S-H gels, while materials richer in MgO (such as some slags) can precipitate hydrotalcite ($\text{Mg}_6\text{Al}_2\text{CO}_3(\text{OH})_{16} \cdot 4\text{H}_2\text{O}$).

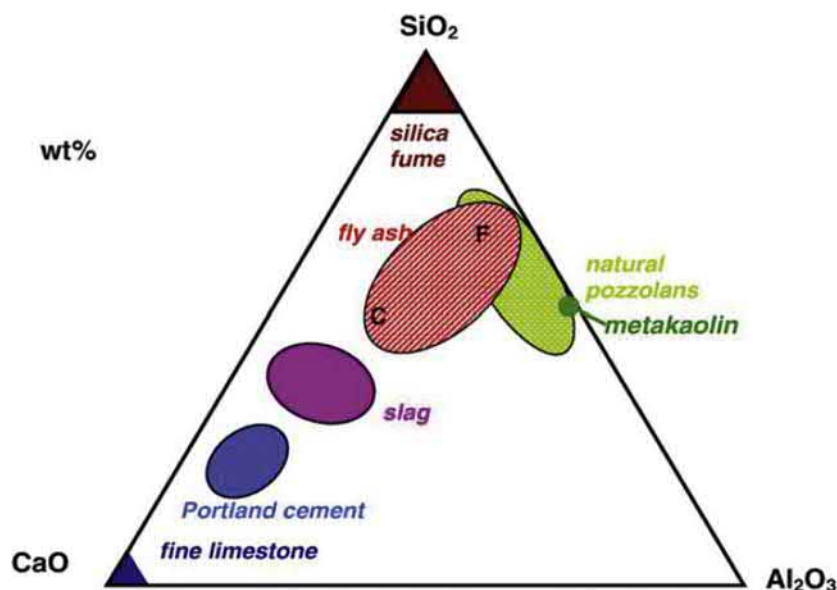


Fig. 1 $\text{CaO-Al}_2\text{O}_3\text{-SiO}_2$ ternary diagram of cementitious materials, represented as oxides in weight %. From Lothenbach B, Scrivener K and Hooton, RD (2011) Supplementary cementitious materials. *Cement and Concrete Research* 41(12): 1244–1256.

Cementation with Portland cements has been favored as a flexible treatment option for ILW wastes worldwide due to the following (Fenton and Holland, 2001; Sharp et al., 2003; International Atomic Energy Agency, 2017; Crossland and Vines, 2001; International Atomic Energy Agency, 2004):

- Ability to process at ambient temperatures, avoiding volatile emissions
- Intrinsic fire resistance due to being an inorganic binder
- Able to solidify liquid or sludge wastes
- High radiation stability, with dense material also providing radiation shielding
- Formed from relatively low cost commercially available materials
- Chemically tolerant of many waste streams
- High pH pore solution limits mobility of many radionuclides
- Limited porosity allows some gas migration, but provides a diffusion barrier against radionuclide release

These waste grouts almost always utilize partial or high level replacement (routinely up to 90%) of Portland cement with blast furnace slag (BFS or GGBS) or fly ash (sometimes known as ‘pulverized fuel ash’, PFA or FA for coal power plants). These blended Portland cements offer a lower heat of hydration along with reduced cracking and porosity, while maintaining adequate fluid characteristics during mixing (Atkins and Glasser, 1992; Glasser, 1992) and forming familiar cement phases (Fig. 2 depicts the fine microstructure of a PC-BFS blend (Taylor et al., 2010)). In the United Kingdom, PFA blends are used as capping grouts (i.e. to infill around supercompacted material) due to the increased fluidity from these blends, while BFS blends are typically used as the main grout for a wide range of wastes.

Cementation is a flexible process, in which liquid or solid wastes are mixed with cement either outside drums (allowing pours into several drums), or in-drum via a lost paddle mixing process. After setting, wasteforms may be capped with a layer of inert grout before package sealing. In-drum grouting is another implementation, where solid items are placed then in-filled with grout, with optional vibration to improve infilling and eliminate voids. Annular grouting is also performed, where multiple compacted drums are stacked inside a larger drum, then surrounded by inactive grout; this has been applied in supercompaction facilities treating low density wastes (Crossland and Vines, 2001; International Atomic Energy Agency, 2004).

Though typically encapsulating, rather than chemically incorporating wastes (unlike vitrification within a glassy matrix), the chemistry of the cement phases formed and the pore solution help to limit radionuclide mobility. The high pH environment (initial pH > 12) will cause many of the transition, alkaline earth, lanthanide and actinide elements to form poorly soluble precipitates—elements which often account for the majority of ILW radionuclide inventory (Crossland and Vines, 2001). Certain elements are not as strongly retained within cement, particularly monovalent cations (e.g. $^{137}\text{Cs}^+$) and anionic species other than inorganic carbonates (e.g. $^{36}\text{Cl}^-$, $^{129}\text{I}^-$, $^{99}\text{TcO}_4^-$). However, Cs^+ mobility is hindered by sorption as a charge compensating ion within a C-(A-)S-H phase,

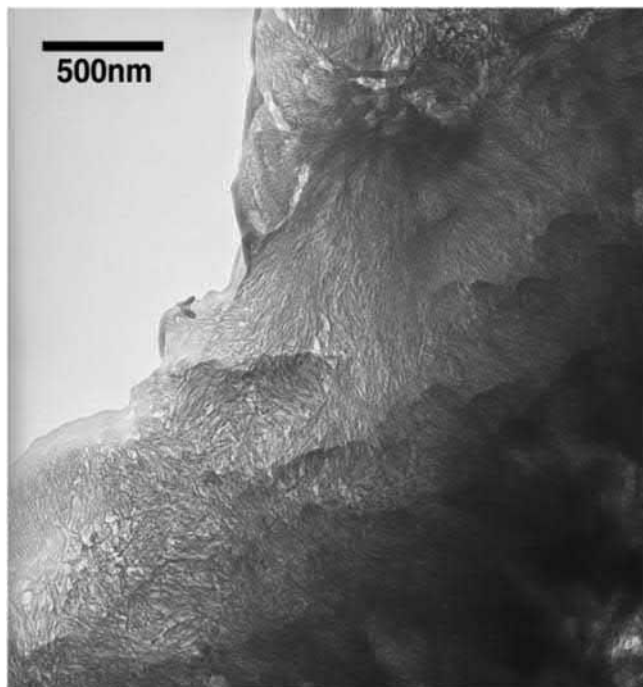


Fig. 2 TEM micrograph of C-S-H in a PC-BFS blend containing 75% slag sample, cured for 20 years. From Taylor R, Richardson IG and Brydson RMD (2010) Composition and microstructure of 20-year-old ordinary Portland cement–ground granulated blast-furnace slag blends containing 0 to 100% slag, *Cement and Concrete Research* 40(7): 971–983.

anionic species exhibit some limited sorption to ettringite, partially replacing SO_4^{2-} , and a lower porosity wasteform, which will necessarily lengthen the time taken for migration of poorly interacting species out of the product (Glasser, 2001).

Although Portland cements are a flexible encapsulation matrix, certain wastes might not be fully compatible with conventional PC blends. Oils are poorly encapsulated, organic materials may degrade over time (resulting in non-aqueous phase liquids, or potentially cracking cement through volume changes), and salts may deleteriously affect cement setting characteristics (either accelerating, or retarding setting) (International Atomic Energy Agency, 2017). Reactive metals such as uranium, aluminum and magnesium alloys can corrode in the high pH or free water conditions of a cement matrix, potentially resulting in expansive corrosion products forming which can crack cements (Sharp et al., 2003). Though not precluding these wastes from cementation, consideration is required for predicting wasteform evolution and maximum waste loadings.

Potential issues with Portland cement blend usage in the future include the significant volume increase from cementitious solidification, as well as any longer-term deleterious waste-cement interactions (reactive metals, organic degradation, etc.). An emerging concern is also that of ensuring continuity of cement supply. The nuclear industry is a very minor consumer of cementitious powders, with industrial production geared towards satisfying demand from construction industries. Changing national and supra-national standards over decades (especially with a move towards lowering embodied CO_2 content of cements) can result in difficulty of supplying cementitious powders to specific particle size, reactivity and chemistries preferred for usage at fixed ILW and LLW cementation plants (which may operate over several decades). This is a particular issue in the United Kingdom due to the projected closure of all coal fired power plants by 2025 (which supply PFA) and concerns over the long-term future of steelmaking (supplying BFS/GGBS), with numerous changes to BFS supplier over the past 20 years as blast furnaces are shuttered (Angus et al., 2010).

Alkali activated cements

Alkali Activated Cements (AACs) consist of a powdered precursor, typically with a high content of glassy $\text{SiO}_2\text{-Al}_2\text{O}_3$ (with or without CaO/MgO) which is 'activated' via addition of an alkaline solution, often sodium (or potassium) hydroxide or silicate, though carbonate and sulfate activation have been studied. Suitable materials available in industrial quantities include blast furnace slag, fly ash, and metakaolin (MK), though natural pozzolans can also be used. The main distinction between the various alkali activated cements is the calcium content, with BFS cements containing high contents of calcium, PFA available as calcium containing or relatively calcium free, and MK which are calcium free (Provis and Bernal, 2014).

Activation of calcium rich precursors results in a set cement product with characteristics and a mineral assemblage similar to that of blended Portland cements. The main binder is a poorly crystalline C-(A-)S-H gel, often with hydrotalcite formed if the precursor contains MgO . These cements have found interest for a variety of reasons. Reduced sulfur present in slags can bring a negative redox potential, which may limit the mobility of certain redox sensitive radionuclides (e.g. ^{99}Tc), and both final pH and porosity can be altered depending on activator and slag chemistry, opening up possibilities for tailored wasteforms and reduced leaching. Resistance to corrosive environments is claimed to be higher, with some studies finding reduced Cs leaching, postulated to be due to lower Ca/Si ration within the binder phase, along with Al substitution in the C-S-H gel (as C-(A-)S-H) helping to retain Cs (Shi and Fernández-Jiménez, 2006).

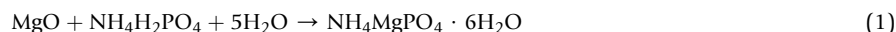
Activation of low calcium precursors results in a very different product, dominated by a highly cross-linked aluminosilicate gel often defined as a 'N-A-S-H' gel (or the potassium equivalent), which lacks long-range ordering. These are typically called geopolymers, and are usually produced through the mixing of metakaolin (a partially dehydrated kaolinite clay) with sodium (or potassium) silicate solutions at varying ratios, molarity and water contents to target specific Na_2O (or K_2O)- $\text{SiO}_2\text{-Al}_2\text{O}_3\text{-H}_2\text{O}$ contents for optimum performance (Provis and Bernal, 2014). Sodium hydroxide cured FA geopolymers are possible, though typically require steam curing due to slow strength development, and as such are not widely utilized.

As geopolymer cements are largely calcium free, and do not rely on a traditional clinker dissolution step, they are potentially more tolerant towards certain waste streams than Portland cements. High salt bearing wastes, which typically affect PC hydration can be targeted, as can oil bearing wastes which can partially saponify within the highly alkaline pore solution, allowing better entrainment within the cement matrix (Cantarel et al., 2015). Formulations can be adjusted to encourage the precipitation of zeolites, which can more readily retain problematic radionuclides such as Cs (Arbel Haddad et al., 2017). They have also been studied for encapsulation of reactive metals, using NaF (which would normally precipitate as CaF_2 in PC systems) to passivate Mg-Zr alloys, which would allow encapsulation of metallic wastes from UNGG reactors in France (Rooses et al., 2013). Commercial geopolymers (e.g. SIAL[®]) have also found useful application in immobilization of radioactive sludges, ion exchange materials and coolant fluid in Slovakia and Czechia, with wide scale usage since 2003 (Lichvar et al., 2013).

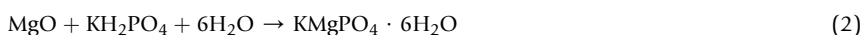
Though not relying on a calcium-based chemistry, these geopolymers maintain a high pH and therefore retain certain benefits of traditional PC-blended systems. These systems have not, however, found widespread adoption in industrial waste cementation. Issues include being a 'two part' system rather than the 'one part' PC powder and water traditionally used. Geopolymer formulations also require careful balancing between reactivity, targeted end-composition, and rheology. The latter, rheology, can be an issue due to the very high surface area of metakaolin and non-standard rheology, along with the water content of wastes affecting calculations. High calcium alkali activated materials such as blast furnace slag can be more easily formulated, have a rheology closer to PC blends, and a chemistry which is more familiar. Unfortunately, these are still calcium based, and are incompatible with many of the same waste streams as PC blends, and also suffer from similar concerns over security of supply into the future, resulting in only limited uptake for bulk waste cementation.

Magnesium phosphate cements

A very different class of cementitious material, which have gained traction for waste immobilization, are magnesium phosphate cements (MPCs). Originally developed for use as refractory cements for application in steel plants, these were popularized as rapid repair cements, finding usage on roads, runways and sidewalks. These cements are in essence a controlled acid-base reaction between magnesium oxide and an acidic phosphate (Walling and Provis, 2016), enabling rapid setting, and the ability to set at sub-zero temperatures. The reaction between MgO with H_3PO_4 is rapid and strongly exothermic, however by calcining MgO until 'dead burnt', this begins to sinter and reduce reactivity, while usage of weakly acidic granular phosphates (e.g. $\text{NH}_4\text{H}_2\text{PO}_4$) further reduce reactivity. This controlled reaction (Eq. 1) results in the formation of a crystalline phosphate mineral called Struvite, which forms a coherent cementitious mass.



A slight modification on these, using KH_2PO_4 instead of ammonium phosphate as a starting material produces analogous Struvite-K (Eq. 2). Magnesium phosphate cements based on Struvite-K were first developed for radioactive waste encapsulation at Argonne National Laboratory, as cements tolerant of ash and salt bearing wastes. Also known as chemically bonded phosphate ceramics (CBPCs), and commercially as 'Ceramicrete', these cements have been trialed both in Russia and the United States for nuclear waste cementation of Pu contaminated ashes, Tc bearing wastes (using reducing agents to reduce Tc to less soluble Tc(IV)), salt wastes at Mayak (Russia) and liquid wastes from the Hanford site (United States) (Wagh, 2004).

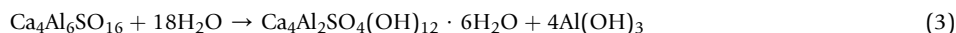


Struvite-K is favored for nuclear applications due to slower reaction times (resulting in slower heat evolution) and no evolution of gaseous ammonia during setting. They are typically blended with a high proportion of diluent, either PFA or BFS, which further reduce the heat output that is especially important as wastes will be solidified in massed drums. Borates are used as highly effective setting retarders, typically applied as borax ($\text{Na}_2\text{B}_4\text{O}_7 \cdot 10\text{H}_2\text{O}$) or boric acid (H_3BO_3) at several weight percentage (wt %) to further extend setting and suppress the maximum heat evolution (Walling and Provis, 2016).

The internal pore solution of MPCs is close to neutral, which makes these cement systems of interest for encapsulation of waste containing reactive metals, such as aluminum fuel cladding, which forms a passive layer at circumneutral pH. The relatively high proportion of crystal bound water within Struvite-K has also drawn research to applying these cements to other reactive metals (uranium and magnesium alloys), which can react in free water. If water is mostly crystal bound after mixing, rather than available in pore solutions this has the potential to reduce corrosion in wasteforms. These cements may also be more tolerant of salt rich wastes, and there is also some interest in structural substitution of K^+ within Struvite-K with Cs^+ , more effectively binding cesium rich waste streams (Leng et al., 2020; Walling and Provis, 2016).

Calcium sulfoaluminate cements

Calcium sulfoaluminate (CSA) cements have garnered some interest for application to radioactive waste immobilization. These cements are made from firing calcium sulfate, limestone and bauxite together (analogous to PC production, though at a slightly lower temperature of $\sim 1250^\circ\text{C}$) to form ye'elimite ($\text{Ca}_4\text{Al}_6\text{SO}_{16}$) and other minor clinker phases. This is typically ground with ~ 15 – 25% gypsum to form a packaged powdered product. Hydration with water results in the formation of ettringite, monosulfate and aluminum hydroxide via Eq. (3)–(4) (Winnefeld and Lothenbach, 2010):



Ettringite formation (and thus water requirements) are highly dependent on the content of gypsum added, resulting in a modifiable cement product. In comparison to Portland cements, the internal pH of a CSA cements are lower, typically ~ 10.5 – 11 , due to the lack of free $\text{Ca}(\text{OH})_2$. Though ettringite has been proposed as a useful immobilization phase for anionic radionuclides, CSA cements are mainly of interest for the encapsulation of reactive metals. Similarly to magnesium phosphate cements, these have been trialed for metallic aluminum encapsulation, including usage in Savannah River p-reactor decommissioning (Hayes and Godfrey, 2007; Stefanko et al., 2010). Ettringite, as the principal hydrate phase in CSAs incorporates a significant bound water concentration, and are expected to result in less free water within the pores, CSA cements may therefore also be suitable for metallic Mg and U wastes.

Performance

Cements utilized for waste cementation are usually high pH materials with a more porous nature than glasses, ceramics or polymers. This results in a greater degree of water ingress and gas migration than other ILW wasteforms. As such, the mechanical integrity of cementitious wasteforms is not as critically important, with radionuclide release largely dominated by chemical conditioning of wastes and waste-wasteform interactions (Glasser, 2001).

Cements are constantly evolving, albeit slowly, with cementitious reactions continuing years after waste emplacement due to unhydrated clinker, unreacted slag or fly ash, and the influence of groundwater or waste materials. These reactions may continue

strengthening the cements, but can also be associated with undesirable reactions. The degradation of cemented wasteforms from groundwater is a key factor in their longevity, with the ingress of chlorides and sulfates deleteriously affecting the cement binder through the formation of expansive mineral phases, such as ettringite or thaumasite, resulting in cracking ([Radioactive Waste Management Limited, 2016](#)). Groundwater migration will also slowly affect the local pH, through removal of free alkali in pore solutions, then dissolution of $\text{Ca}(\text{OH})_2$, followed by slow decalcification of the C-S-H/C-A-S-H gel. Carbonation can also lower the cement pH, either from atmospheric CO_2 during storage, or dissolved carbonates in groundwater. Understanding these changes are key to wasteform longevity, as under alkaline conditions many radionuclides are likely to exist as poorly soluble oxides or hydroxides, severely limiting their migration (reducing conditions from slags also assist with radionuclide retention). The presence of different mineral phases within the cement also support radionuclide sorption, though some radionuclides (notably Cs) do not form insoluble hydroxides, and are only weakly sorbed by C-S-H/C-A-S-H ([Glasser, 2001](#); [Glasser, 1992](#); [Crossland and Vines, 2001](#)).

As these are inorganic materials, radiolytic degradation is much less of a concern, though this may cause reactions with organic material (such as chloride release from PVC degradation), and can cause H_2 formation from pore water radiolysis ([Radioactive Waste Management Limited, 2016](#)). Microbial degradation is also unlikely to affect cements deleteriously, with the high pH also reducing microbial activity. Of greater impact on wasteform integrity are specific waste interactions within the cements, in particular the expansive reaction products of ion exchange resins or encapsulated reactive metal corrosion which may induce cracking, accompanied by gas production. Other wastes such as flocs, slurries or raffinates may slowly react with the cements and may result in shrinking (or expansive reactions) and cracking, making it important to understanding waste-cement interactions.

For the alternative cements, such as geopolymers and MKPCs, their long term performance is less well understood. Geopolymers are highly alkaline, with the N-A-S-H gel binder expected to provide substantial pH buffering, and consequently reduced radionuclide mobility. Alkali activated slag cements will perform closely to blended PC/BFS cements, but likely with a higher pH and more reducing environment. MKPCs are near-neutral, with performance likely to rest upon the slow dissolution of the main hydrated magnesium phosphate phase, and the behavior of encapsulated wastes. Specific dissolution rates, or durability of cements are very difficult to determine, varying greatly by waste, type of cement, and proposed engineered barrier systems within a geological repository. Some countries, such as the United Kingdom, place reduced weight on cement wasteform durability, as proposed cementitious backfills and the predicted long durability of steel containers in these conditions are liable to keep these conditioned until much of the radioactivity has decayed ([Radioactive Waste Management Limited, 2016](#)).

Bitumen

Bitumen is a thermoplastic, which is solid or semi-solid at room temperature, comprising a complex mixture of high molecular weight hydrocarbons. Though bitumens occur naturally, industrial applications utilize material from asphaltic crude oil distillation. In waste immobilization, these can be further categorized into 'straight run' bitumen obtained by direct distillation of crude oils, 'blown' bitumen, in which air is blown through bitumen at 200–250 °C, and 'emulsified' bitumen which has various soaps or emulsifiers added to vary the properties ([Valcke et al., 2014](#); [International Atomic Energy Agency, 1993](#)).

The bituminization process for nuclear wastes is usually either via continuous extrusion, or thin film evaporation. Continuous extrusion comprises pouring both heated bitumen and waste into a heated rotating screw extruder, which intermixes both materials, extruding a homogeneous product out into a drum. During this process, water and low molecular weight hydrocarbons are evaporated and removed, with the final product left to cool after. Thin film evaporation utilizes a cylindrical vessel with rotating blades which pass very close to the heated wall. Waste and heated bitumen are added at the top, and are both spread and thoroughly intermixed on the walls by the blades, with gravity pulling this mixture towards a drum underneath the vessel. Countercurrent hot air removes water and volatile organics, again resulting in a product which is left to cool after ([International Atomic Energy Agency, 1993](#)).

The appeal of bituminization is that of a highly flexible process with good compatibility for organic materials, and ability to achieve high waste loadings with a low permeability, and lower leach rates than cementation. It is a relatively simple process with low operating costs, finding usage with immobilization of chemical precipitates (especially from fuel reprocessing) and low heat/low alpha decay wastes. This process found notable usage in Belgium for high salt content wastes from fuel reprocessing, along with Sweden and Finland for immobilization of ion-exchange wastes, France, and more minor usage in Japan, Germany and Switzerland amongst others ([Valcke et al., 2014](#); [International Atomic Energy Agency, 2017](#); [International Atomic Energy Agency, 1993](#)).

Bituminization does have some severe drawbacks that have limited its widespread adoption worldwide. Though suitable for high salt content wastes (Eurobitum wastes in Belgium are ~60 wt % blown bitumen and ~40 wt % salts, mainly as NaNO_3 and CaSO_4), swelling of these wasteforms from osmotic pressures can cause considerable volume expansion and are of concern in final disposal conditions (with research also ongoing into the effect of salt release on the clay backfill) ([Valcke et al., 2014](#); [Mariën et al., 2013](#)). Waste materials may require pre-treatment as dehydrating salts, strong oxidizers (e.g. sodium nitrate), and tributyl phosphate can be fire hazards; indeed several fires have been documented at bituminization plants, due to poor understanding of wastes prior to immobilization ([International Atomic Energy Agency, 2017](#)). Bitumen products also offer poor radiation shielding due to low density, requiring extra shielding during processing/transport, and are particularly affected by radiolytic degradation.

This radiolysis and associated radio-oxidation can result in gas release (particularly H_2), swelling and embrittlement of the wasteform over time (International Atomic Energy Agency, 1993; Valcke et al., 2014).

The degradation of bituminous wastes is complex and is governed predominantly by type of bitumen used, exposure to radiation, and type and quantity of wastes encapsulated. In general, bitumen is very resistant to chemical action, with a low water diffusion and radionuclide release if carefully produced. Over time bitumen will age, predominantly via oxidation. This can result in a slow hardening, embrittlement and cracking of the wasteform, which may provide pathways for enhanced water ingress and radionuclide release (Valcke et al., 2014). Oxidation typically begins at the surface, slowly migrating inwards and forming lightweight organic degradation products, though only affecting a small proportion of the wasteform as oxygen diffusion is low (Pettersson and Elert, 2001). Wasteform contact with water can result in bitumen swelling, along with leaching of soluble organic molecules from oxidation or radiolytic degradation processes (Valcke et al., 2014). If particularly hygroscopic wastes are encapsulated (e.g. salts or dried organic resins) at high waste loadings, then water uptake and osmotic pressures can result in considerable swelling, exerting pressure on waste containers, resulting in fissures and potential radionuclide release (Hendrix et al., 2016). Radiolysis from encapsulated wastes can result in further swelling and breakdown via gas evolution, along with embrittlement, especially when exposed radiation doses exceeding $> 10^7$ Gy (International Atomic Energy Agency, 1993). The radiolytic gases evolved are predominantly H_2 , with minor contributions from CO, CO_2 and low weight organics (Radioactive Waste Management Limited, 2015; Pettersson and Elert, 2001). The associated fire hazard from hydrogen evolution, and potential organic release within a repository has generated questions about the ultimate disposal route for these wastes.

Microbial degradation of bitumen wasteforms is possible, however, it is reported to be low and unlikely to affect wasteform durability, especially under anoxic conditions likely to be prevalent in geological repositories (Radioactive Waste Management Limited, 2015). Ultimately, the rate of radionuclide leaching is highly dependent on their solubility, with reported values in the range of 10^{-4} to 10^{-1} $g\ m^{-2}\ d^{-1}$ for actinide and rare earth metals, and 5×10^{-2} to $50\ g\ m^{-2}\ d^{-1}$ for alkali and alkaline earth metals (International Atomic Energy Agency, 1993).

Polymers

Polymers have been utilized for the immobilization of problematic wastes that are unsuitable for other routine immobilization methods. The most notable advantage of polymer encapsulation is tolerance of a wide range of waste chemistries at high waste loadings, including organic materials, reactive metals, and high salt wastes, which may preclude efficient cementation (US Department of Energy—Office of Environmental Management, 1999; Radioactive Waste Management Limited, 2015). Polymers used for immobilization are organic synthetic plastic materials typically broadly classified as either thermoplastic or thermosetting (International Atomic Energy Agency, 2002; Radioactive Waste Management Limited, 2016).

Thermoplastics are materials which soften upon heating, and are generally supplied in a solid form. Their chemistry is largely unchanged by heating, allowing re-molding and incorporation of wastes before solidifying upon cooling. The most common thermoplastic studied for ILW encapsulation is polyethylene (consisting of repeating $-CH_2-$ units) and bitumen (discussed in section 'Bitumen'). Polyethylene (PE) processing is undertaken using an extrusion processes with wastes and PE added into a feed hopper, heated (low-density polyethylene (LDPE) becomes fluid $\sim 85^\circ C$) and intermixed through an extrusion screw before entering a drum which is left to cool (Radioactive Waste Management Limited, 2015). This polymer is readily available and less flammable than bitumen. Some water is driven off during the heated extrusion process, though wastes with high water contents require a drying step before addition. As no chemical reactions are required to produce a solid product, this polymer is extremely tolerant of wastes, highly chemical resistant and solidified products exhibit low leach rates (Radioactive Waste Management Limited, 2016).

Thermosetting polymers are liquid at room temperature, and require either the application of heat or various curing/hardening agents to initiate cross-linking polymerization and for a hardened product. Common thermosetting polymers for immobilization include: epoxy resins, polyester resins and urea formaldehyde (International Atomic Energy Agency, 2002).

Solidified epoxy products are typically made from the combination of an epoxy resin with a hardener. A wide range of epoxies are manufactured, with varying physical properties, allowing tailored solutions to varied waste streams. However, the correct ratio of hardener to epoxy is required to control reactions, meaning some epoxies are sensitive to water content and pH of wastes. The most common epoxy resin is based upon the reaction of epichlorohydrin with bisphenol A in the presence of sodium hydroxide, forming bisphenol A diglycidyl ether (BADGE; Fig. 3) (Radioactive Waste Management Limited, 2015). The oxirane group in the BADGE reacts with varying curing agents (amines, anhydrides, phenols and thiols), cross-linking to form a hardened product. Very high strengths can be achieved, though this often results in brittle products. These epoxies exhibit very good leach resistance along with low flammability but are attacked by strong acids. France has successfully used epoxy resins to immobilize ion exchange resins

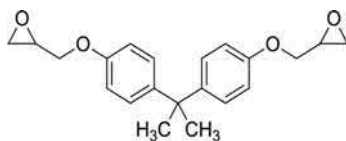


Fig. 3 Bisphenol A diglycidyl ether (BADGE).

since 1996 using the MERCURE process, entailing a lost-paddle mixing procedure once waste, epoxy and hardener have been added (Radioactive Waste Management Limited, 2015).

Polyester resins have found usage, due to high flash points and ability to incorporate water (with an emulsifier), including for ILW in the United Kingdom (Trawsfynydd NPP ion-exchange immobilization) using vinyl ester styrene (Radioactive Waste Management Limited, 2015). Produced via polymerization of linear polyester with a cross-linking initiator, these polymers can achieve high compressive strengths and are in general very chemically resistant (International Atomic Energy Agency, 2002; Radioactive Waste Management Limited, 2016). Urea formaldehyde is not commonly utilized for waste immobilization. Though these are low cost and low temperature while curing, a particular problem is the formation of acidic water (which remains liquid within the waste-form after setting) that can corrode conventional steel containers. Compressive strengths are low (Radioactive Waste Management Limited, 2015), as is their tolerance to radiation (International Atomic Energy Agency, 2002).

Though useful for certain problematic wastes, the drawbacks of polymer encapsulation include their relatively low radiation resistance compared to cementation or vitrification, along with increased cost compared to cementation. Radiolysis can generate hydrogen gas (depending on the polymer), and substantially affect physical properties, including embrittlement and cracking—which may be of concern in a final repository environment (Radioactive Waste Management Limited, 2015).

Polymer stability and degradation largely occurs as a function of the specific polymer utilized. Many of the less suitable polymer types and formulations have been excluded through extensive testing prior to wide-scale waste encapsulation operations, resulting in wasteforms fabricated with stable polymers. In general, engineered polymers degrade and release radionuclides via exposure to high temperature, extensive radiation, and waste-wasteform interactions. Wasteform degradation can lead to reduced compressive or tensile strength (or embrittlement), the formation of non-aqueous phase degradation products, and gas formation (Radioactive Waste Management Limited, 2016). Polymers are extensively resistant to chemical attack (especially PE), and exhibit radiation tolerance up to 10^6 Gy. One notable exception is urea-formaldehyde resins, in which doses $> 10^5$ Gy result in severe damage. Radiolysis of polymers can result in gas formation, the composition being dependent on the polymer (predominantly CO_2 in polyester, H_2 for epoxy and PE) (Radioactive Waste Management Limited, 2015). The presence of high waste loadings with hygroscopic materials (e.g. dried ion exchange resins) can result in deleterious expansion, although the extent of these effects depend on the plasticity of the polymers, with more ductile materials like LDPE being more tolerant of expansive forces (Arnold et al., 1985). Overall leaching rates for radionuclides from polymer wasteforms are considered low in the short term, however, as wastes are encapsulated rather than chemically conditioned, once leaching begins this can provide pathways for enhance leaching, with mobility ultimately dominated by geological repository engineered barriers.

Wet oxidation

Wet oxidation is a low temperature aqueous oxidative process that utilizes hydrogen peroxide coupled with a catalyst (typically iron salts) to produce hydroxyl and hydroperoxyl radicals (as shown in Eq. 5–6) (Babuponnusami and Muthukumar, 2014; Bokare and Choi, 2014). These radicals readily attack organic bonds, with the ultimate aim of degrading organic materials to CO_2 and H_2O leaving a much smaller volume of sludge containing radionuclides within an aqueous solution. This technique is not currently applied to any wastes on an industrial scale, but there have been numerous pilot plants and mobile rigs developed over the past 30 years. These largely focused on degradation of spent ion exchange resins, but also decontamination solutions (containing organic complexants like EDTA and citric acid) and waste reprocessing solvents.



A typical wet oxidation setup for ILW treatment consists of a glass reaction vessel, to which waste (e.g. spent ion exchange resins) and a catalyst are added, heated to ~ 90 – 100°C and slowly dosed with hydrogen peroxide over the course of hours (typically 2–8 h). During this process the resins slowly decompose, first to linear soluble lower molecular weight products and then to CO_2 and H_2O . Resin decomposition releases functional groups (e.g. sulfates and ammonia species), requiring the solution pH to be monitored and maintained at $\text{pH} \sim 2$ – 3 for optimum reactivity (especially for iron catalysts). Decomposition leaves radionuclides within an aqueous solution, which can be neutralized by precipitating many as poorly soluble hydroxides; the resulting sludges may be suitable for cementation (Walling et al., 2021b).

Decomposition of spent ion exchange resins with wet oxidation occurs below 100°C , limiting the risk of volatile element release (e.g. cesium volatilization). This allows for less complex off-gas systems (via retention of decomposition acids within the processing solution and lower radionuclide volatilization), enabling waste volume reduction and allowing degradation of organic complexants (which may preclude acceptance to waste repositories). Trials using pilot plants have been undertaken in the United Kingdom, Europe and the United States, using both iron and copper catalysts, though full implementation of this technology for waste treatment has not occurred (Walling et al., 2021b; Charman and Twissel, 1999). The output wastes generated from wet oxidation are highly unlikely to be directly disposed, and as such may undergo secondary conditioning to form cementitious, glassy or ceramic wasteforms.

Thermal treatment

Thermal treatment embraces a broad range of approaches that include the application of pre-treatment technologies (which convert liquid or solid wastes (e.g. soft organics) into lower volume solids for further treatment), and final-treatment technologies, which produce wasteforms suitable for long-term storage or disposal in a national repository with no further processing required. Thermal treatment is suitable for a wide range of ILW waste streams including organic and inorganic liquids, organic and inorganic solids and metallic wastes. Thermal treatment often converts waste into an integral part of the wasteform, as opposed to cementation where wastes are physically encapsulated (resulting in increased volume). The general advantages of thermal treatment include (but are not limited to) waste volume minimization via the destruction of organics, removal of water, thermal degradation and oxidation of wastes. Thermally treated wasteforms are considered to be more stable with a higher aqueous durability (lower leachability) and radiation tolerance than encapsulated wastes and may be vitreous (glass), glass-ceramic, or ceramic materials, with the phase assemblage controlled by the waste and additive compositions.

A common challenge associated with thermal treatment technologies is the need for off-gas systems to capture and retain radioactivity through a range of functions, such as; off-gas combustion, gas cooling to precipitate particulates and filtration (e.g. filter ash), wet-scrubbing and HEPA filters prior to the emission of the purified off-gas within local regulatory limits. Therefore, thermal treatment is inherently a more complex process, with associated cost implications, when compared to cementation, that also requires the development of suitable process and formulation envelopes to maximize waste loading while controlling criticality and volatility risks. Nevertheless, the production of passively safe products of minimized waste volume, compared to encapsulation technologies, delivers overall lifetime cost savings, arising from downstream storage and disposal, as well as reduced hazard. In this section, thermal treatment technologies for ILW conditioning will be discussed with a range of technological maturity from industrial application to demonstration scale to laboratory scale.

Pre-treatment technologies

Calcination

Calcination of nuclear wastes is a common process for level liquid wastes generated from nuclear fuel reprocessing operations, with the United Kingdom and France utilizing calcination as part of a two-stage calcination-vitrification thermal treatment process for high level liquid wastes, though with the technology applicable to ILW. Typically these exist as spray (350–550 °C), fluidized-bed (500–600 °C) and rotary kiln (500–600 °C) calciners ([International Atomic Energy Agency, 2007](#)). Calcination provides high efficacy volume reduction, elimination of water, salt decomposition, and formation of a stable oxidized waste product ([International Atomic Energy Agency, 2007](#)). The calcination product is a fine powder with concentrated activity that is easily dispersible, and consequently requires further processing.

Incineration

High temperature incineration (1000–1500 °C) promotes the complete combustion of solid and liquid organic wastes in an oxygen rich environment, with the waste products being a solid residue (ash) and an off-gas. Commonly used for the treatment of solid ILW and some ILW, wastes can be solid or liquids (organic, aqueous) but the technology is intolerant to high concentrations of non-combustibles, metallic wastes and plastics ([International Atomic Energy Agency, 2007](#)). An incinerator plant is often composed of primary combustion chamber (and secondary chambers for off-gas oxidation), a boiler to cool the off-gas, off-gas filtration and radiological purification and wet scrubbing prior to discharge ([International Atomic Energy Agency, 2007](#)). Incinerators are a proven technology with worldwide usage for both nuclear and non-nuclear applications including: Austria, Belgium, Canada, France, Germany, India, Japan, Netherlands, Russian Federation, Slovakia, Spain, Sweden, United Kingdom, Ukraine and United States ([International Atomic Energy Agency, 2007](#)). The advantage of incineration in waste management include: it is a proven technology with a high volume reduction, high throughput, and flexible to varying waste streams. The incineration product is a stable oxidized ash that requires additional conditioning, e.g. vitrification, super compaction, or cementation depending upon the activity and waste classification.

Pyrolysis

Pyrolysis is flameless thermal treatment technology that operates upon a dry distillation process where wastes are exposed to external heating in the absence of air/O₂ (inert atmosphere) to promote the fracture of organic compounds (gasification), typically into methane, ethane, benzene, toluene and a carbon-rich char. The product gases undergo off-gas treatment to yield a calorific off-gas, which is subsequently fully combusted and purified ([International Atomic Energy Agency, 2007](#)). The relatively low temperature (500–600 °C) ensures that volatile radioelements are retained with the solid output wastes (e.g. tars and char). Pyrolysis results in a significant waste volume decrease that can be further improved by removal of carbon from the carbon-rich tar. This can be achieved by steam reformation (Eq. 7; not discussed further) or incineration (oxidized) to yield atmospheric carbon (CO₂)



Pyrolysis is deemed a suitable technology for all waste types such as liquids (high in Na, Al, nitrates, nitrites, nitric acid, hydroxides and sulfates), oils, organic solvent, sludges, plastics, cellulose and soils with the exception of inorganic liquids and inorganic ion exchange resins (IERS). There are several examples of pyrolysis plants used for the treatment of ILWs including: Germany

(pebble pyrolysis with $\text{Ca}(\text{OH})_2$ for spent IERs and spent reprocessing solvents); France (IRIS plant, 55% plastics, 35% latex/rubber and 10% cellulose, which is pyrolysed at 550 °C followed by subsequent incineration at 900 °C; O_2 -rich); and in the United States (THOR® process by Studsvik) (International Atomic Energy Agency, 2007). The latter uses additives in the reactor (fluidized-bed) to support the formation of insoluble aluminosilicate minerals (mineralization), which are considered suitable for disposal in high integrity containers (International Atomic Energy Agency, 2007). While in other systems, the final product is typically a granular material that may require further processing (similar to calcination/incineration) to yield a monolith waste product suitable for disposal.

Thermal gasification

An alternative to pyrolysis is thermal gasification (reducing atmosphere) that targets the incomplete combustion of wastes to yield a calorific off-gas, which is subsequently fully combusted and purified. This technology is under development for the treatment of both LLWs and ILWs rich in organic matter (e.g. organic ion exchange resins; IERs), for which cementation of the raw wastes may not be fully suitable (Vehmas et al., 2020).

The VTT thermal gasification design includes a thermal fluidized-bed reactor (bubbling or circulating fluidized-bed reactors) operating at approximately 900 °C, followed by subsequent gas purification processes for gasified wastes; CO , CO_2 , H_2 , CH_4 , O_2 (Nieminen et al., 2020a). Filter dust and bottom-ash (fluidized-bed material; Al_2O_3 and inorganic particles) removed from the reactor constitute the output solid waste. A recent demonstration trial successfully converted Cs-exchanged IER into filter dust, with the solid output wastes equivalent to 1 wt % of the original waste stream (Nieminen et al., 2020b). The solid wastes require additional processing, which is proposed to be accomplished using cementation technology, and more recently geopolymerization (Nieminen et al., 2020a,b; Vehmas et al., 2020). The advantages of thermal gasification include high waste minimization, high conversion of carbon, capture of volatile radionuclides in the filter ash, lower processing temperature and compact design, however, complex off-gas systems are required.

Final treatment technologies

Induction melters

Induction melters use external heating to provide induction heating to a corrosion-resistant metallic reactor vessel (e.g. Inconel alloy crucible) that will contain the waste and glass forming additives, with the resulting product typically a homogenous glass. Induction melters are primarily used in the thermal treatment of HLWs, however, if deemed economically viable, this technology could be applied to a wide range of ILWs, and in particular wastes with a high refractory content.

An example of induction melter is the Atelier de Vitrification (AVM) of La Hague process facility and the Waste Vitrification Plant (WVP; United Kingdom) for the thermal treatment of liquid high level waste (HLW) generated from fuel reprocessing operations (International Atomic Energy Agency, 2007). The UK design is based upon a two-stage process, where the liquid waste is first dried via calcination (600–840 °C) and gravity-fed into the melter at a controlled rate (25 kg h^{-1}) alongside the glass-frit until the melt achieves 1050 °C. The molten glass (waste loading 25–32 wt %) is poured in 200 kg batches into stainless steel canisters (Harrison, 2014), and cooled slowly (annealed). The resulting waste packages are deemed suitable for the long-term storage and disposal.

An alternative system is a Cold Crucible induction Melter (since retrofitted in the La Hague plant (Naline et al., 2010)) where the addition of external crucible cooling leads to the formation of a protective ‘skull’, which essentially provides a protective coating on the internal surface of the crucible, owing to a thermal gradient between the crucible and the molten material (also referred to as the ‘melt pool’). This design allows for processing at higher temperatures (for refractory wastes) and/or more corrosive wastes (Hyatt and James, 2013; International Atomic Energy Agency, 2007). Further alternatives include the in-can vitrification system, which is under inactive commissioning by CEA for processing incineration wastes (e.g. PVC, cotton) and simulant alpha-contaminated effluents, with the can (Inconel alloy) renewed after each cycle (Nieminen et al., 2020b) however, the system can only handle small quantities of organics and metallic waste.

Joule heated ceramic melters

Joule heated ceramic melters (JHCMS) use resistive heating which is directly introduced into the waste/melt by passing an electric current between submerged electrodes in a refractory lined reaction vessel, with a melt formed from glass formers at temperatures reaching ~1150 °C (Goel et al., 2019; Matlack et al., 2010). The waste (in the form of a slurry or calcine) and additional glass formers are introduced during the melt from above, lowering surface melt temperatures (‘cold-cap’) which helps reduce volatilization of certain radionuclides (e.g. ^{137}Cs , ^{99}Tc). The thermal transfer within the melt can be improved through use of bubblers and/or mechanically stirring to actively mix, substantially improving throughput times (Matlack et al., 2010). Final homogeneous glass products are poured into suitable canisters for long-term storage and disposal.

JHCM is a popular choice for the thermal treatment of nuclear wastes with existing plants (or plants under construction) in the United States, Germany, Belgium, Japan with applicability to a wide range of ILW waste streams including sludges, salt rich wastes, and ion exchange materials (both organic and inorganic) (Goel et al., 2019; Matlack et al., 2010; Hyatt and James, 2013). The main technical challenge arises from noble elements, which can accumulate in the melter and impact pouring capabilities (Radioactive Waste Management Limited, 2017; Goel et al., 2019).

In-container vitrification (ICV)

Joule-heated In-container vitrification (ICV) uses resistive heating applied by sacrificial electrodes (typically graphite) with a starter path to initiate melting, due to the low conductivity of waste/glass former. Once initiated, melt conductivity increases, facilitating convective mixing to form a melt pool containing the pre-loaded waste and glass formers. ICV utilizes a refractory lined steel container, which the container forms part of the overall waste package, with no melt pouring required (Fig. 4). The key advantage of ICV technology is its applicability for highly heterogeneous waste streams, including larger complex objects (canisters, sea disposal crates, compacted drums), metallic wastes, along with high processing temperature (up to 1800 °C) (Hyatt and James, 2013; Radioactive Waste Management Limited, 2017). Wastes can either be fully pre-charged into the vitrification container, or partially charged with additional feeding while melting to accommodate volume reduction (Radioactive Waste Management Limited, 2017). The final wasteform may be homogenous or a heterogeneous combination of glass, ceramic and metallic phases (Hyatt and James, 2013).

In-container vitrification has been investigated or utilized for numerous wastes including depleted uranium contaminated soils, metallic sodium wastes, chlorinated organics, and the destruction of asbestos (Walling et al., 2021a). Kurion (now a subsidiary of Veolia Nuclear Solutions) developed the GeoMelt® ICV technology, which ranges in application from full-scale operations to demonstration trials for the vitrification of nuclear wastes. In the United Kingdom, ICV technology targeted plutonium contaminated materials (PCM) and co-mixed Magnox sludge and clinoptilolite/sand wastes with large scale successful demonstrations (Hyatt and James, 2013; Witwer et al., 2010; Nieminen et al., 2020b). In 2018, large scale trials were performed using GeoMelt® technology in the United States for processing metallic sodium contaminated waste from the Fermi 1 reactor, yielding 16 glass waste products, which have since been accepted for disposal at the NNSS (Nevada National Security Site; United States) (Garrett et al., 2020).

Plasma heating

Plasma heating is generated by creating an electric discharge between electrodes, with the electric arc forming a plasma (ionized carrier gas). A plasma torch is designed with concentrically-arranged tubes that are water-cooled and contain a flowing process gas supply, all housed in high-grade refractory liner that can withstand the extremely high temperatures generated. There are two categories of plasma melting systems: non-transferred and transferred. In transferred-torches, the energy is transferred directly from the anode via electric discharge to the molten pool (e.g. slag), which serves as the cathode (International Atomic Energy Agency, 2007). This process is considered more energy efficient than non-transferred torches, where two electrodes, separated by a gas injection chamber, create an electrical arc that ionizes the gas (plasma). The typical plasma temperature for non-transferred torches is 5000 °C, which is lower than transferred-torches but still capable of inducing a melt within the waste material (International Atomic Energy Agency, 2007).

The use of very high temperature is key to the success of plasma facilities, as this enables a wide range of wastes to be treated from organic to inorganic materials, and liquids to solids provided they can be fed into the system, which yields a stable slag/glass waste product. A plasma plant typically consists of four key constituents: waste feeder (batch or continuous), plasma chamber (refractory lined or cold crucible), off-gas system and slag collection system (International Atomic Energy Agency, 2007; Deckers, 2020; Hyatt and James, 2013).



Fig. 4 GeoMelt® technology used in UK ILW demonstrations. Nieminen M, Olin M, Laatikainen-Luntama J, et al. (2020b) Thermal treatment for radioactive waste minimisation, EPJ Nuclear Sciences & Technology, 6: 25.

There are two examples of full-scale plasma plant facilities for the immobilization of nuclear wastes, these are the ZWILAG plasma facility in Switzerland which has operated since 2004 (International Atomic Energy Agency, 2007) and the Kozloduy plasma melting facility (PMF) in Bulgaria operational since 2018 (Deckers, 2020). The ZWILAG plasma facility operates using a transferred-plasma torch capable of reaching plasma flame temperatures of 5000 °C, which equates to a melt temperature of 1200–1500 °C (Heep, 2007). Batch processing sees single 200 L drums containing untreated waste being introduced to the plasma chamber via a horizontal drum feeder, typically totaling 300–400 drums per cycle (Heep, 2007). During processing, the crucible is rotated to utilize centrifugal forces, which prevents the molten slag from nearing the pour port (located at the bottom of the crucible). To enable pour into a slag mold, the crucible rotation is slowed to a suitable speed. Gasified organic wastes are directed through an afterburner (including heat recovery) and off-gas system (International Atomic Energy Agency, 2007; Heep, 2007). Plasma heating is suitable for a wide range of wastes with only minimal pre-treatment (e.g. waste shredding if needed) however, complex off-gas systems are required to process the combusted organic material, volatilities and particulates generated (Hyatt and James, 2013; International Atomic Energy Agency, 2007).

Dual heating vitrification (SHIVA)

The SHIVA (Système Hybride d'Incineration Vitrification Avance) process developed by CEA (Commissariat à l'Énergie Atomique et aux Énergies Alternatives) utilizes dual simultaneous heating (plasma and induction) to achieve the incineration-vitrification of radioactive wastes. The SHIVA technology consists of a water-cooled stainless steel reactor vessel with twin plasma transfer torches (graphite electrodes) located within the reactor chamber and a flat inductor at the bottom (Lemont et al., 2008; Nieminen et al., 2020b). Glass frit is pre-filled into the reaction crucible and heated to form molten glass. During waste processing the plasma torches are sustained with a mix of argon and oxygen (10 L min⁻¹), in contrast to argon-only flow during the start-up phase, which induces an oxidizing atmosphere suitable for the incineration (gasification) of organic wastes. Waste is directly introduced to the plasma zone using a hopper feeder and worm screw with a typical feed rate between 1 and 10 kg h⁻¹ (Lemont et al., 2008; Nieminen et al., 2020b). The gasified wastes are fully oxidized in the plasma chamber, which requires a simpler off-gas system than other thermal treatment technologies (Nieminen et al., 2020b). Once the melt process is complete, the molten glass is drained to yield a homogenous vitrified product, namely an aluminoborosilicate glass.

The SHIVA process is suited to the treatment of organic and inorganic ion exchange materials, in either liquid or solid form, but is incompatible with metallic wastes (Nieminen et al., 2020b; Fournier and Rousset, 2020). The specific advantages include ability to continually feed waste during melting, a cold-crucible design to reduce corrosion, and a compact design which allows operation within hot cells for ILW and HLW (Lemont et al., 2008).

Hot isostatic pressing

Hot isostatic pressing (HIPing) is an alternative advanced thermal treatment process that involves the simultaneous application of induction heating and pressure (typically using Ar(g) as the pressurizing medium) to a hermetically sealed HIP canister (e.g. stainless steel). The processing temperature can be up to 1400 °C with isostatic pressure in the range of 10–200 MPa. The waste streams deemed suitable for HIPing include ILW inorganic wastes, ranging from sludges (e.g. Magnox sludge), spent inorganic ion exchange resins, Idaho calcine and sludges, fissile materials, wastes from medical isotope production to plutonium contaminated materials; however, organic materials are unsuitable. Higher activity wastes are being considered for this technology including potentially immobilization of the UK inventory of civil separated plutonium, which will target a single phase ceramic wasteform (Stewart et al., 2009; Heath et al., 2018; Hyatt, 2020; Radioactive Waste Management Limited, 2017). The advantages of HIPing compared to other technologies include; batch-to-batch processing provides improved inventory accountancy and critically control, lower processing temperature, minimal secondary wastes (fewer off-gas requirements) and high waste loadings (up to 100%) (Stewart et al., 2009).

For HIPing, a certain degree of pre-conditioning is required to prepare the hermetically sealed HIP canister, which can include low temperature calcination (to retain volatile radioelements but drive off water) and waste homogenization (mixing with additives) prior to can filling (Stewart et al., 2009; Heath et al., 2018). During the HIP process, the canister undergoes deformation due to the elevated temperature reactions (e.g. melt or solid-state reaction) and pressure imposed upon the system, which results in consolidation and densification of the waste than can range from a glass, glass-ceramic or ceramic depending on the waste composition and chosen additives, as demonstrated in Fig. 5. One limitation of HIPing is the lack of industrial scale pilot plants to validate the efficacy of for nuclear immobilization; however, the Australian Nuclear Science and Technology Organisation (ANSTO) are currently commissioning a HIP for the immobilization of ⁹⁹Mo/^{99m}Tc bearing wastes arising from medical isotope production (Gregg et al., 2020), while in the United Kingdom, proof of concept radiological HIPing on the laboratory scale is underway (Gardner et al., 2020).

Performance of wasteforms from thermal treatment of ILW

The thermal treatment of ILW may produce either homogenous or heterogeneous glasses and glass-ceramic wasteforms, owing to the complex nature of waste compositions which may be combined in co-mixed waste treatment (e.g. in-can vitrification). It is generally accepted that glasses, ceramics and glass-ceramics have higher chemical durability and thus, higher performance than non-thermal conditioned wastes (e.g. cementation). In ILW glass and glass-ceramic wasteforms, the waste often provides a portion of the fundamental wasteform composition (e.g. clinoptilolite can act as a glass former, as a source of Al and Si, and is fully incorporated within the chemical structure). This is in contrast to cementation, where the waste is physically encapsulated and retains the

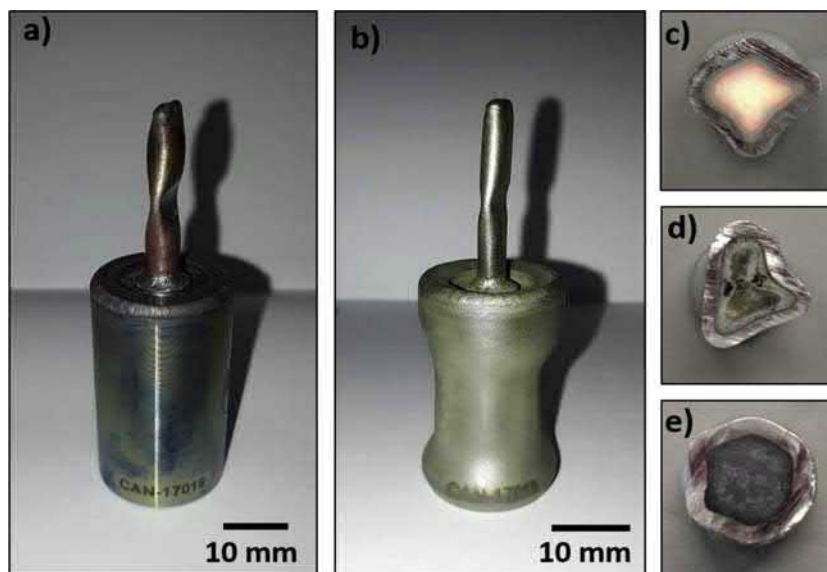


Fig. 5 Example photographs of a stainless steel HIP canister before (A) and after HIPing (B) including sectioned HIPed canisters demonstrating simulant Magnox sludge vitrification in a borosilicate glass matrix with uranium (U_3O_8) contents at 0 wt % (C), 7 wt % (D) and 44 wt % (E).

original waste characteristics, which often includes soluble and/or highly mobile species. Natural analogues of glasses and ceramics, dated from thousands to millions of years, provide much needed confidence in the long-term durability and integrity of these materials, including withstanding natural disposal conditions and weathering.

The performance of glass and glass-ceramic wasteforms relates to the chemical durability (i.e. leachability) and radiation tolerance. In geological repository conditions, wasteform dissolution will only occur upon failure of the wasteform container/overpack (typically stainless steel), allowing groundwater ingress. Due to this delay, a significant decrease in radioactivity associated with short lived radionuclides in intermediate level wastes will inherently reduce the risk to the biosphere resulting from radionuclide migration.

Glass dissolution (typically silicate based glasses for ILW) proceeds in three key stages. Stage 1 (also referred to as initial or forward rate), is defined by inter-diffusion and ion-exchange processes. Following a rate-drop, Stage 2 (characterized by a residual rate) occurs, and is defined by hydrolysis of the glass network. This leads to elemental release and alteration of the glass surface via the formation of an amorphous gel-layer and/or the precipitation of secondary crystalline phases, namely phyllosilicates and zeolites in saturated conditions (Mann et al., 2019). The impact of secondary phases is often complex. Gel-layers, rich in Si (but can form as single element-rich bands) are often thought to be passivating and protect the glass surface from further dissolution, while phyllosilicates and zeolites may stimulate progression to Stage 3 (rate resumption), which promotes accelerated elemental release and dissolution rates (Mann et al., 2019).

The rate of glass dissolution is highly dependent on the pH, in particular the solubility of Si (the main glass former), which is reported to increase by two to three orders of magnitude from near-neutral to pH 12 (Radioactive Waste Management Limited, 2016). This solubility trend is also affected by the temperature, where higher temperatures lead to enhanced dissolution rates over a wider pH range. In post-closure repository conditions, the predominant mechanism for radionuclide release from ILW glass wasteforms will arise from contact with groundwater and will be controlled by the near-field environment, temperature and redox conditions. If a high pH (e.g. cementitious, $\text{Ca}(\text{OH})_2$ rich) backfill is utilized in a repository, this may promote Stage 3 (rate resumption) leading to higher dissolution rates (Mann et al., 2019), whereas, a lower-pH backfill (e.g. Boom clay) and low water transport rates will maintain Stage 2 (residual) dissolution, leading to slower dissolution kinetics (Gin et al., 2013).

Ceramic hosts are typically very durable with respect to radiation tolerance (as are glasses (Radioactive Waste Management Limited, 2016)) and chemical durability (although there is some variation with ceramic type). The dissolution of ceramics is affected by the number of defect structures (e.g. oxygen vacancies) and grain boundaries, where higher occurrences can lead to enhanced dissolution (Corkhill et al., 2016), nevertheless much lower than glass dissolution. Glass-ceramics are heterogeneous wasteforms that can be advantageous for waste management: radionuclides typically partition in the more durable ceramic host, while the glass phase is chemically flexible and can easily accommodate variation in the waste composition. Glass-ceramics would be expected to undergo incongruent dissolution as a result of the different material properties associated with the glass and ceramic fractions, with the durability ultimately dependent on the least durable component.

Estimates for thermally treated wasteform dissolution rates in geological repository conditions vary dependent on specific wasteform chemistry, and groundwater conditions, however in average conditions they can be estimated to be between: 10^{-4} – $10^{-2} \text{ g m}^{-2} \text{ d}^{-1}$ for a borosilicate glass and 10^{-5} – $10^{-4} \text{ g m}^{-2} \text{ d}^{-1}$ for ceramics (Forschungszentrum Jülich and Brenk Systemplanung, 2011). For example, thermally treated ILW co-mixed wastes via in-can vitrification resulted in Si dissolution rates

(7–28 d) between 10^{-4} and 10^{-3} g m⁻² d⁻¹ (Walling et al., 2021a), while spent ion exchange materials converted into a glass-ceramic via hot isostatic pressing resulted in averaged dissolution rates (1–28 d) of 10^{-4} g m⁻² d⁻¹ (Si) and 10^{-5} g m⁻² d⁻¹ for Cs (simulant waste), where Cs was preferentially partitioned into the ceramic phase (Gardner et al., 2021), which broadly agree with the reported best estimate dissolution rates.

Overall, the performance of glass, ceramic and glass-ceramic wasteforms are deemed more durable with respect to aqueous dissolution than non-thermally treated wasteforms, however this enhanced durability should be assessed in conjunction with the higher facility costs of thermal treatment technologies, and the total risk associated with intermediate level wastes.

Summary

Overall, the majority of ILW treatment is currently undertaken by encapsulation using conventional Portland cement based blends, with the occasional exception for particularly challenging wastes. Cementation offers a durable, long-term solution with a high level of reliability, stability and understanding about long-term degradation mechanisms. Cemented products are generally radiation tolerant, have strong fire resistance and are ultimately very compatible with proposed construction methods for final geological repositories (especially with cement backfills). These reasons are likely to result in continued usage of cementation as an ILW treatment technology into the future.

Thermal treatment technologies provide advance processing for a wide variety of wastes, including those that may not be suitable for cementation in their raw format (e.g. organic materials, reactive metals). The variety of thermal treatment technologies vary in both scale and technological maturity, which may impact uptake within the nuclear industry. For example, induction heating and plasma heating are well-developed with existing industrial scale facilities, whereas the comparative maturity of hot isostatic pressing is currently much lower. The overall benefits for thermal treatment involve waste volume minimization to yield a stabilized monolith of either glass, glass-ceramic or ceramic that could be suitable for geological disposal, with very low dissolution rates under proposed repository conditions for both glasses and ceramics. There is an ongoing research and development drive to support the uptake of thermal treatment of ILWs via multi-partner consortiums in collaboration with end-users and the wider nuclear community. These projects are the 'Thermal treatment of radioactive waste minimization and hazard reduction' (THERAMIN; 2016–20) and the 'Pre-disposal management of radioactive waste' (PREDIS; 2020–24) with the overall aim to enhance strategic implementation of thermal treatment technologies for the management of intermediate level wastes.

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HLW Conditioning and Long-Term Performance

Florence Bart^a, Christophe Poinssot^b, Stéphane Gin^{a,c}, Sylvain Peugnet^{a,d}, and Céline Roussel^e, ^a Department of Dismantling and Waste Conditioning, Energy Division, CEA French Atomic Commission, Marcoule Center, Bagnols-sur-Cèze, France; ^b French Geological Survey (BRGM), Orléans, France; ^c Long Term Behavior Laboratory, Bagnols-sur-Cèze Cedex, France; ^d Highly Radioactive Material Laboratory, Bagnols-sur-Cèze Cedex, France; and ^e ORANO group, Montigny-le-Bretonneux, France

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Introduction

The operation of nuclear reactors, as well as their associated fuel cycles: uranium production, enrichment, fuel fabrication and reprocessing in some cases, generate radioactive waste. Radioactive waste also results from the use of radioactive materials in research, medicine, education and industry. In some countries, radioactive waste is also generated in support of defense activities. This means that practically all States have to consider the management of radioactive waste and to make sure it is managed in a safe manner with due regard to the level of radioactivity and in compliance with national regulations, often based on, or in harmony with, IAEA safety standards (AIEA, 2018).

High level waste (HLW) are waste with levels of activity concentration high enough to generate significant quantities of heat by the radioactive decay process and/or waste with large amounts of long lived radionuclides. Disposal in deep, stable geological formations usually several hundred meters or more below the surface is the generally recognized option for disposal of HLW.

Synthesis of the HLW typology and key characteristics

HLW typology

After use, Spent Nuclear Fuel becomes a highly radioactive material, due to the presence of minor actinides and fission products created by nuclear reactions. Spent Nuclear Fuel (SNF) is considered as a high level waste in some countries (United States, Sweden, Finland, Spain, Canada, etc.) or a potential future energy resource in others (France, Japan, United Kingdom, Belgium). In the first case, SNF will be disposed after some decades of interim storage ("once through fuel cycle"). In the second case, is planned to be reprocessed to recover fissile materials for future use ("closed fuel cycle"). The majority of States have adopted the policy of direct disposal of spent nuclear fuel, while several States with the largest nuclear programs (France, India, Japan, the Russian Federation, the United Kingdom and China) have adopted the policy of reprocessing nuclear fuel and recycling the separated material. Germany and the United States of America have changed their policy from reprocessing and recycling to direct disposal and are currently planning spent fuel repositories.

Basic characteristics of spent nuclear fuel (Poinssot and Gin, 2012)

A PWR spent fuel assembly is comprised of four 580-Lm thick Zircaloy fuel rods of 9.5 mm in diameter and 4-m long, inside which are stacked uranium oxide fuel pellets (Johnson et al., 1988). The free volumes of the fuel rods are initially filled with Heat 25 bar. After irradiation, the fuel pellets are fractured in roughly 15 fragments due to the presence of a steep thermal gradient and contain

minor actinides (produced by neutron capture) and fission products (produced by fission reactions), which are either trapped in the UO_2 lattice or remain present as metal or oxide solid phases within the UO_2 .

Part of the fission products are gases, which are for the larger part (95%) trapped within the UO_2 matrix, whereas the remaining gases are released in the free volumes of the rod and contribute to significantly increasing the internal rod pressure (up to 50–60 bar in room conditions). In addition, the distribution of fission reactions is not completely homogeneous within the rods and the zones with very high fission rates have a specific microstructure with smaller grains and a higher porosity. In the case of UOX rods, such zones occur as an external rim, and for MOX fuels, these zones occur in the Pu-rich agglomerates from the initial microstructure.

Furthermore, fuel cladding is also damaged during the in-reactor period due both (i) to the external corrosion by primary circuit water leading to hydrogen uptake, and (ii) to the irradiation-induced mechanical hardening. Finally, it is important to stress that nuclear fuel is not designed to be a durable waste form, but to host the fission reaction inside the reactor and to produce electricity. This means that it is severely damaged after irradiation and, furthermore, not in thermodynamic equilibrium due to the fast quenching upon reactor shutdown.

Therefore, its long-term durability and confinement properties are more difficult to demonstrate compared to a homogeneous and specifically-designed matrix such as nuclear glass.

Basic characteristics of high level liquid waste coming from SNF reprocessing

SNF contains about 2% fissile material (1% remaining ^{235}U , 1% plutonium and minor actinides) that can be isolated (reprocessed) and recycled as a new fuel. The HLW accounts for over 95% of the total radioactivity produced in electricity generation if a reprocessing policy is followed. It originates from a liquid residue from reprocessing spent nuclear fuel to extract the uranium and plutonium for reuse. The liquid contains most of the radioactivity from the original spent fuel.

In the reprocessing option, more waste streams need to be dealt with in addition to the HLW itself. These other waste streams include intermediate level structural waste such as hulls and nozzles from fuel assemblies, and intermediate level process waste, for example sludge from liquid effluent treatment. Typically, 1 t of reprocessed PWR fuel generates about 0.1 m^3 of HLW, 2.5 m^3 of ILW, $1.5\text{--}3 \text{ m}^3$ of LLW (all values are after conditioning, including the waste package/container) and 950 kg of uranium which can be reused for new fuel production (AIEA, 2018). The most important other waste streams produced in reprocessing spent fuel is the long lived ILW composed of the hulls and endcaps of the metal tubes that contained the fuel, and other parts of the fuel elements. These can be embedded in a matrix of cement inside a steel container or can be compacted into steel cylindrical containers similar to those containing the HLW.

Conventional aqueous (PUREX type—plutonium and uranium recovery by extraction) reprocessing generate a complex solution containing some 40 chemical elements which must be continuously stirred and cooled to dissipate its thermal power.

The fission product solution produced by the PUREX process is the initial highly radioactive liquid waste to be considered for conditioning. It is a nitric solution (typically 1–2 N) containing a complex mixture of species (about 100 g L^{-1}), partially precipitated, including mainly Cs, Sr, Mo, Zr, minor actinides (Np, Cm, Am), lanthanides, platinum group metals (Pd, Ru, Rh), plus alkaline and alkaline earth cations, and corrosion products from stainless steel, nearly the whole periodic classification.

Conserving this fission products solution in the liquid state is not a sustainable option considering security and safety concerns on the long-term, and this initial high level liquid waste (HLLW) have to be transformed to produce a chemically durable and heat- and radiation-resistant solid matrix tailored for final disposal nuclear glass wasteforms, as explained later on.

Rationale for selecting a relevant conditioning process

The conditioning of fission products solutions is aimed at (1) turning waste from the liquid to the solid state, (2) reducing the volume to be stored and, then, disposed of, and (3) getting a material which complies with the safety requirements peculiar to storage and long term disposal. More precisely, the conditioning material has to present satisfactory properties of resistance to self-irradiation (alpha, beta and gamma emitters), thermal stability (heat production associated with radioactivity), and chemical durability (taking into account environmental disposal conditions). The wasteform is meant to stabilize the waste during packaging, an interim storage period, transportation, disposal in a geological repository, and post closure disposal period. The deep geologic disposal is considered to be the most suitable option for disposing higher activity radioactive wastes worldwide because of the predicted effectiveness of many natural geological systems in reducing the transport of radionuclides that can be augmented by robust engineered barriers, thereby enhancing disposal facility performance.

Early research routes were first focused on mica- or feldspath-type crystalline materials prior to being re-oriented to vitreous materials in the late 1950s (Vernaz and Bruezière, 2014). During the 1960s, glass was selected by the world's community as the confinement material for fission products solutions, due to the structural flexibility of its disordered structure that enables glass to confine many chemical elements. The aim is not a mere embedding, but an atomic-scale confinement, since radionuclides are intimately incorporated in the glass structure.

Glasses were selected in the 1970s primarily for the following reasons (Jantzen and Ojovan, 2019):

- Vitrification technologies are known and reliable large scale non-nuclear commercial technologies.
- More cost-effective process than ceramics.



Fig. 1 Simulated non-radioactive nuclear glass.

- High solubility of waste components in the glass and tolerance to variation in waste composition.
- Low raw materials costs.
- Sufficiently high durability to ensure long-term performance.
- Continuous, high-throughput, operation of glass melters.
- High resistance to damage from radiation and radioactive decay.
- Reasonably well understood corrosion/release mechanisms for a homogeneous wasteform.
- Existence of natural analogues.

The use of the vitrification process (see [Fig. 1](#)) for conditioning of fission product has been chosen by many countries, such as France, Belgium, the United Kingdom, the United States of America, Japan, the Russian Federation, South Korea, India, China. Except for alkali-aluminophosphate glass used in Russia, alkali-borosilicate glass has been universally selected as the vitreous waste-form convenient for fission products solutions immobilization.

Determining a glass composition means making a compromise between the material properties and the technological feasibility of its industrial-scale fabrication ([Jantzen and Ojovan, 2019](#); [Vernaz, 2009](#); [Hyatt and Ojovan, 2019](#)).

The main following criteria are considered:

- Waste loading: the wasteform must be able to accommodate a significant amount of waste (typically 20 wt%) in order to minimize final volume of waste canisters to be disposed in deep geological;
- Industrial feasibility, considering that the processes need to be remote operated, in hot cells facilities.
- Durability: low rate of dissolution to minimize the release of radioactive and chemical constituents in the biosphere during very long period of time (typically 100,000 years), radiation stability.
- Compatibility with the intended disposal environment—compatible with the near-field environment of the disposal facility.

Description of the HLW conditioning processes, the sole industrially deployed conditioning process for HLW

Two kinds of industrial vitrification technologies are available: JHM, Joule Heated Melter and CCIM, Cold Crucible Induction Melters.

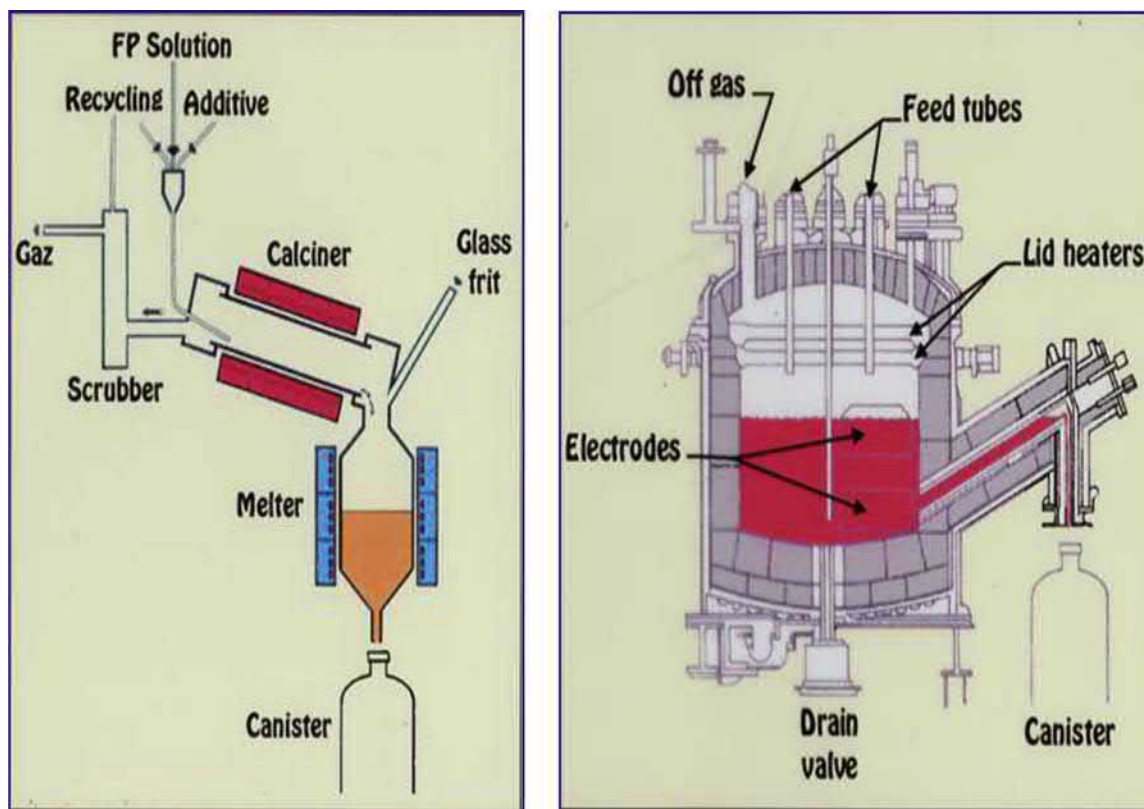


Fig. 2 Schematic drawing of joule Metallic Melter (left) and Liquid-Fed Ceramic Melter (right).

Two kinds of Joule Heated Melters are used in nuclear glass industry (Fig. 2):

- The ceramic melter, known as “LFCM” for liquid fed ceramic melter,
- Joule metallic melters.

The LFSM process consists in heating the glass by an electric current flowing between electrodes that are immersed in molten glass (approximately 1150 °C). The furnaces are big-sized ceramic furnaces, they show extended lifetime compared to metallic pots (several years). The melter is directly fed with the liquid waste, evaporation and calcination steps take place on the molten glass surface. This continuous process is implemented in Belgium (Pamela), United States (DWPF and WVDP), Japan (TVF), Russia (Mayak), India (Trombay) and Germany (CEK).

Joule Metallic Melter technologies for processing nuclear glass are implemented in France (Marcoule Vitrification facility, La Hague Facility), United Kingdom (WVP), India (Tarapur). The Joule heated melters are smaller than ceramic melters (containing typically 400 kg of glass), with a shorter lifetime, and metallic pot need to be changed after typically 1 year. They have a maximum operating temperature around 1150 °C. The vitrification process is a continuous two steps process, on line performed in two separate devices: evaporation-calcination of fission product solutions, and the vitrification of the calcinate previously produced. Heat is induced in the crucible walls, which heat the glass.

The CCIM is selected for processing glass or glass ceramic wasteform when higher melting temperatures (1200–1300 °C) are required. In this case, heat is directly induced in the molten glass and the crucible walls are cooled. There is a self-forming crucible, which is a thin cooled glass liner, a few centimeters thick, protecting the metallic walls of the furnace from molten glass corrosion. This technology is implemented industrially in France (La Hague facility), the induction melter being replaced by a CCIM on Fig. 2, and in Korean for low and intermediate level waste.

Typical nuclear glass compositions are given in Table 1 (Malkovsky et al., 2020) for countries that are industrially producing high level nuclear glass.

Vitrified waste interim storage

Background

The vitrified wastes are produced complying with product quality conditions (process conditions, glass chemical composition, thermal power, dose rate) which are the results of the optimization of production parameters, the interim storage conditions, the canister transportation and the final disposal facilities.

Table 1 Typical nuclear glass compositions.

Country	Facility	Type of glass: Composition in wt. %
Russia	EP500 PA "Mayak"	NAP: 53.3 P ₂ O ₅ · 15.8 Al ₂ O ₃ · 1.6 Fe ₂ O ₃ · 23.5 Na ₂ O · 5.8 miscellaneous ^a
Belgium	Pamela	NAP: 70.7 P ₂ O ₅ · 7.1 Al ₂ O ₃ · 22.2 Fe ₂ O ₃
Belgium	Pamela	ABS : 52.7 SiO ₂ · 13.2 B ₂ O ₃ · 2.7 Al ₂ O ₃ · 4.6 CaO · 2.2 MgO · 5.9 Na ₂ O · 18.7 miscellaneous
France	AVM Marcoule	ABS: 46.6 SiO ₂ · 14.2 B ₂ O ₃ · 5.0 Al ₂ O ₃ · 2.9 Fe ₂ O ₃ · 4.1 CaO · 10.0 Na ₂ O · 17.2 miscellaneous
France	AVH R7/T7	ABS: 45.1 SiO ₂ · 13.9 B ₂ O ₃ · 4.9 Al ₂ O ₃ · 4.0 CaO · 9.8 Na ₂ O · 22.3 miscellaneous
	La Hague	
Germany	Karlsruhe	ABS: 60.0 SiO ₂ · 17.6 B ₂ O ₃ · 3.1 Al ₂ O ₃ · 5.3 CaO · 7.1 Na ₂ O · 6.9 miscellaneous
UK	WVP Sellafield	ABS: 47.2 SiO ₂ · 16.9 B ₂ O ₃ · 4.8 Al ₂ O ₃ · 5.3 MgO · 8.4 Na ₂ O · 17.4 miscellaneous
US	DWPF Savannah River	ABS: 49.8 SiO ₂ · 8.0 B ₂ O ₃ · 4.0 Al ₂ O ₃ · 1.0 CaO · 1.4 MgO · 8.7 Na ₂ O · 27.1 miscellaneous
US	WVDP West Valley	ABS: 45.8 SiO ₂ · 8.4 B ₂ O ₃ · 6.1 Al ₂ O ₃ · 11.4 Fe ₂ O ₃ · 1.4 MgO · 9.1 Na ₂ O · 17.8 miscellaneous
US	WTP Hanford (under construction)	ABS: 50.0 SiO ₂ · 20.0 B ₂ O ₃ · 5.0 Al ₂ O ₃ · 25.0 Na ₂ O
Japan	Tokai Vitrification Facility	ABS: 46.7 SiO ₂ · 14.3 B ₂ O ₃ · 5.0 Al ₂ O ₃ · 3.0 CaO · 9.6 Na ₂ O · 21.4 miscellaneous
India	WIP Trombay	ABS: 30.0 SiO ₂ · 20.0 B ₂ O ₃ · 25.0 PbO · 5.0 Na ₂ O · 20.0 miscellaneous
India	AVS Tarapur	ABS: 34.1 SiO ₂ · 6.4 B ₂ O ₃ · 6.2 TiO ₂ · 0.2 Na ₂ O · 9.3 MnO · 43.8 miscellaneous

NAP, Sodium Alumino Phosphate Glass; ABS, Alumino Boro Silicate Glass.

^aMiscellaneous includes waste oxides.

From Malkovsky VI, et al. (2020) The influence of radiation on confinement properties of nuclear waste glasses. Science and Technology of Nuclear Installations **2020**: 8875723, p. 14.

These high level waste canisters must be kept safe, whatever a decision concerning long-term disposal for HLW, interim storage is therefore worldwide necessary to manage glass canisters during several decades. In France for example, nuclear glass canisters (about 30,000 canisters) are safely placed in interim storage since the beginning of the industrial production (1978 for Marcoule Vitrification facility and 1989 for La Hague Vitrification facility).

The efficiency of the interim storage facility relies on the stability of the glass canisters and more especially of the glass itself, submitted specific thermal conditions (from 400 °C after production to less than 200 °C after 10 years) and submitted to a self-irradiation stress coming from the decays of fission products and minor actinides. Dose rate is around 1.4×10^4 Gy h⁻¹. Thermal heat is approximately 2.5 kW per canister at the production step, 1.6 kW after 10 years, and 0.6 kW after 50 years. After 100 years, ⁴³²Am (half-life period = 430 years) has a more and more important contribution to heat power in comparison to ¹³⁷Cs/^{137m}Ba and ⁹⁰Sr/⁹⁰Y radionuclides.

As an example, the French vitrified HLW canister is a steel canister 43 cm (17 in.) in diameter and 1.4 m (55 in.) in height (Fig. 3). The total volume is about 180 L (47.5 US gallons) and is filled with about 400 kg (882 pounds) of glass containing 65 kg (143 pounds) of HLW (about 15% to 18.5% in weight of FP and actinide oxides). US glass canisters produced at DWPF are larger, 10 ft tall and 2 ft in diameter.

After fabrication, glass canisters are cooled by a forced ventilation system. When thermal power is decreased (typically, < 2 kW for la Hague Facility), canisters are moved to an interim storage with natural air ventilation. The interim storage must guarantee that glass is kept at temperature lower than the glass crystallization temperature, which is depending on glass composition. In the case of borosilicate nuclear glasses, the minimum crystallization temperature is about 600 °C, and that is why the facility is designed to keep the canisters at temperatures lower than 500 °C.

Thermal stability

The nuclear glass are obtained by quenching of a glass melt prepared in the in the vitrification furnace. From a thermodynamic point of view, the glass at solid state is out of equilibrium conditions and could transform into a more stable form, a crystalline material. Crystallization phenomena in glassy materials obey to nucleation and growth laws that are limited by kinetic reasons. As an example, some studies performed on natural basaltic glasses have shown the preservation of the glassy state with no devitrification during the 1 million years of natural ageing after their fabrication (Crovisier et al., 2003).

In the case of the nuclear glass, specific studies have been performed on simulated and industrial nuclear glass to evaluate the risk of devitrification during the interim storage conditions. Long term thermal treatments have been performed, at 550 °C and 450 °C for 1 year, and have shown no crystallization of the glass. Moreover, some scientific studies were also performed to evaluate the nucleation and growth laws of the main crystalline phases that can be induced by thermal treatments in a temperature range of 630–1170 °C, either from isothermal or thermal gradient heat treatments (Orlhac et al., 1999, 2001; Delattre et al., 2013).

Optical microscopy and SEM experimental studies were used to determine the nucleation and growth kinetics of these phases. A modeling approach was developed to evaluate the crystallization rate of the glass with time and then applied in this temperature range. Three main crystalline phases have been observed, i.e., zincochromite, cerianite and powellite phases (Fig. 4). Taking into account the glass chemical composition a maximum of around 4 wt% of crystalline phase could be formed and would require an optimized thermal treatment. From the modeling work, the time necessary to induce a full crystallization of these phases is estimated to be of around 50,000 years at a temperature of 690 °C. This crystallization time increases by a factor of 100 with a decrease of temperature of 100 °C, meaning that at 200 °C, crystallization phenomena are negligible.

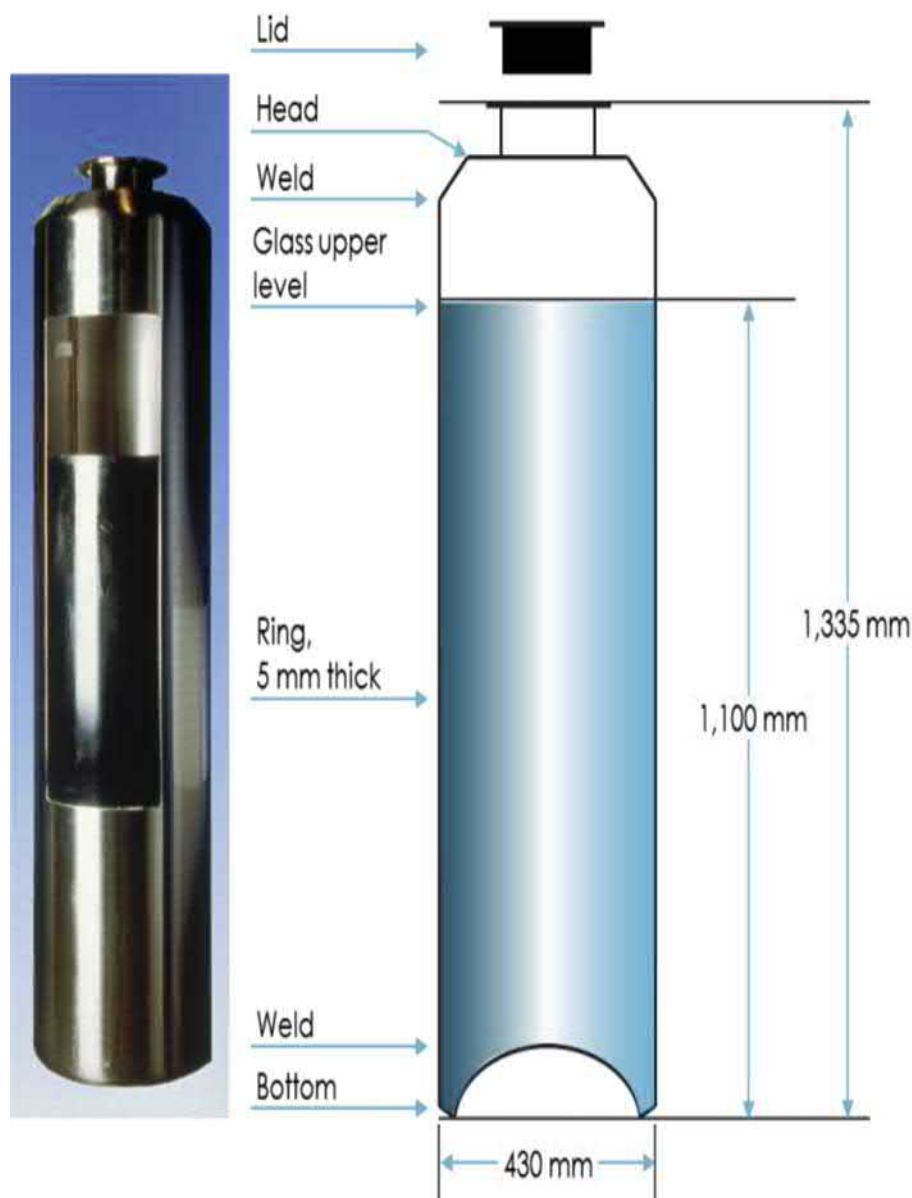


Fig. 3 Vitrified HLW canisters produced by ORANO La Hague (France).

Radiation stability

Nuclear glass are submitted to beta and alpha decays coming respectively from the radioactive decays of fission products and minor actinides, and also to gamma transition emitted by of the excited nucleus (Malkovsky et al., 2020; Mir et al., 2016, 2017; Weber, 2014). The beta decays, gamma transition and the alpha particles emitted in alpha disintegrations loss most of their energy to the glass structure by electronic excitation and ionization of the atoms. The recoil nuclei of alpha disintegrations mainly loss its energy by nuclear collisions with the atoms of the glassy network, inducing displacement cascades.

During the first 100 years of interim storage, the level of absorbed dose form beta decays and gamma rays is of typically around 4–5 GGy, and is called electronic dose. In the same time, the level of absorbed dose from the recoil nuclei of alpha decays is of around 0.02 GGy and is termed nuclear dose (See Fig. 5).

It has been shown that (Mir et al., 2016, 2017), under beta irradiation, simplified borosilicate glasses can undergo significant structural transformations, mainly involving point defect generation, boron coordination changes, modification of the glass polymerization level, changes of the mean angle between silica tetrahedral, and molecular oxygen formation. Nevertheless it was reported that the level of the structural transformation is influenced by the glass chemical composition. For instance, a mixed alkali effect appeared to decrease the changes, and the addition of transition metals or lanthanide elements into the glass can decrease or completely suppress the glass structure modification. It was suggested that the various redox states of transition metals

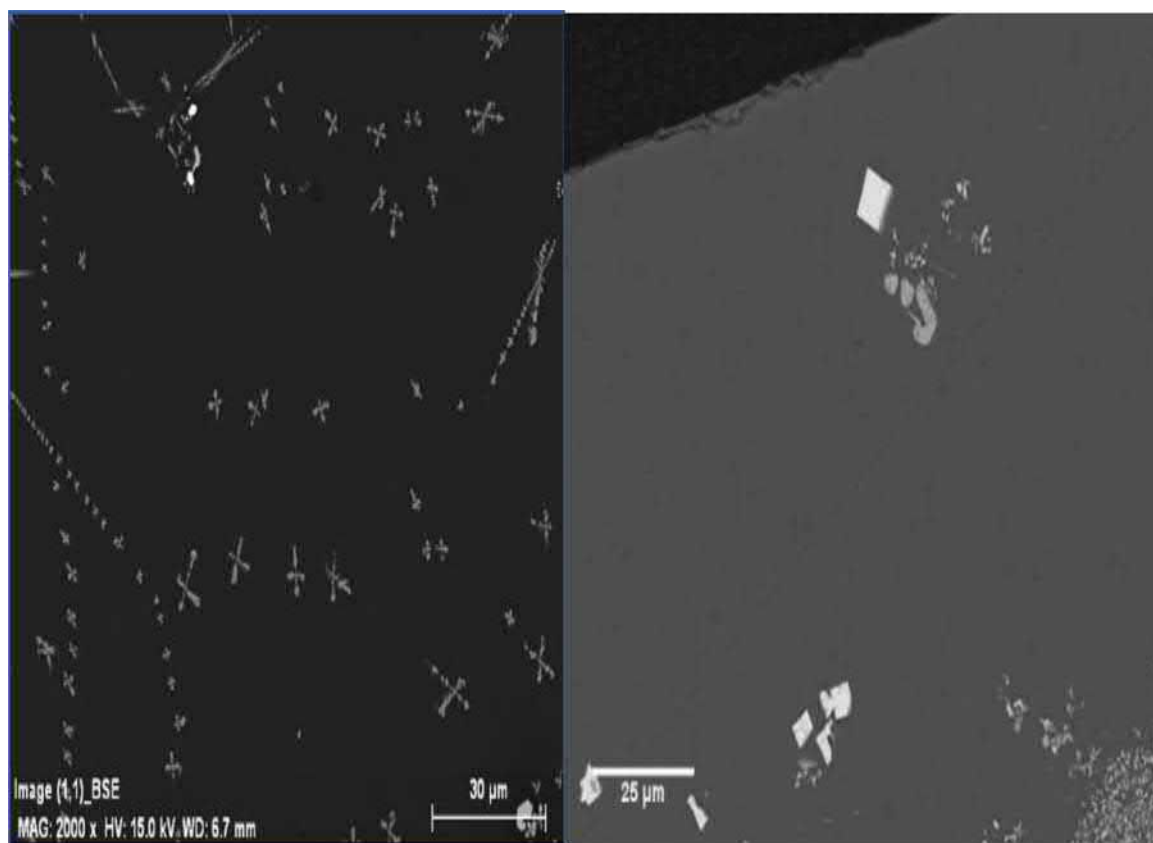


Fig. 4 Example of crystallization observed in a borosilicate nuclear glass after thermal treatment: zincchromite, cerianite and powellite are the white crystals surrounded by the glass which is darker.

or lanthanides act as traps for the electrons and holes generated by the interaction between the electrons and the glassy network, and consequently reduce or prevent the generation of point defects and any subsequent structural transformation. Therefore the chemical complexity of the glass is a positive factor that reduces its sensitivity to ionizing radiation. This certainly explains why few structural changes, with no modification of the glassy state, were observed in the studies of complex nuclear glass irradiated by beta particles up to 4 GGy.

Alpha decay accumulation in nuclear glass induces slight changes of macroscopic properties (density and mechanical properties), no modification of the glassy state with a glass microstructure that remains homogeneous without any phase separation and crystallization, a slight modification of the glassy structure, with modifications of both short and medium range orders (Peuget et al., 2014). The main structural changes are a partial conversion of boron from a tetrahedral coordination number to a trigonal one, an increase of the non-bridging oxygen atoms concentration and a modification of the ring statistic and angles in between the glass polyhedra. The final structure is the result of the ballistic melting in recoil nuclei tracks and the subsequent alpha particles partial damage repair due to a thermal self-healing mechanism in the alpha particles tracks. It results in a new homogeneous glassy state, similar to that obtained by a very high quenching rate of a molten glass, with slightly modified macroscopic properties. Their variations with dose is controlled by the accumulation of recoil nuclei track overlapping, according to a direct impact model. The saturation of the structural changes is therefore observed at a nuclear dose of around 40 MGy (or 4×10^{18} alpha decay/g) (Fig. 6).

The glassy state is preserved all along the duration of the interim storage period. Few structural changes and modification of the macroscopic properties are induced by both beta and alpha decays that are limited in amplitude and for which a saturation of the variation with dose have been observed. These results show that the main function of the glass that is to contain the radioactive isotopes will be preserved during the interim storage period.

Long term performance of HLW in geological disposal

Background

There is presently a broad consensus among technical experts that the preferred method of ensuring long term management of HLW (Spent Nuclear Fuel and Glass) is isolation in a deep geological disposal facility, which provides passive, multi-barrier isolation of radioactive materials from biosphere (AIEA, 2018).

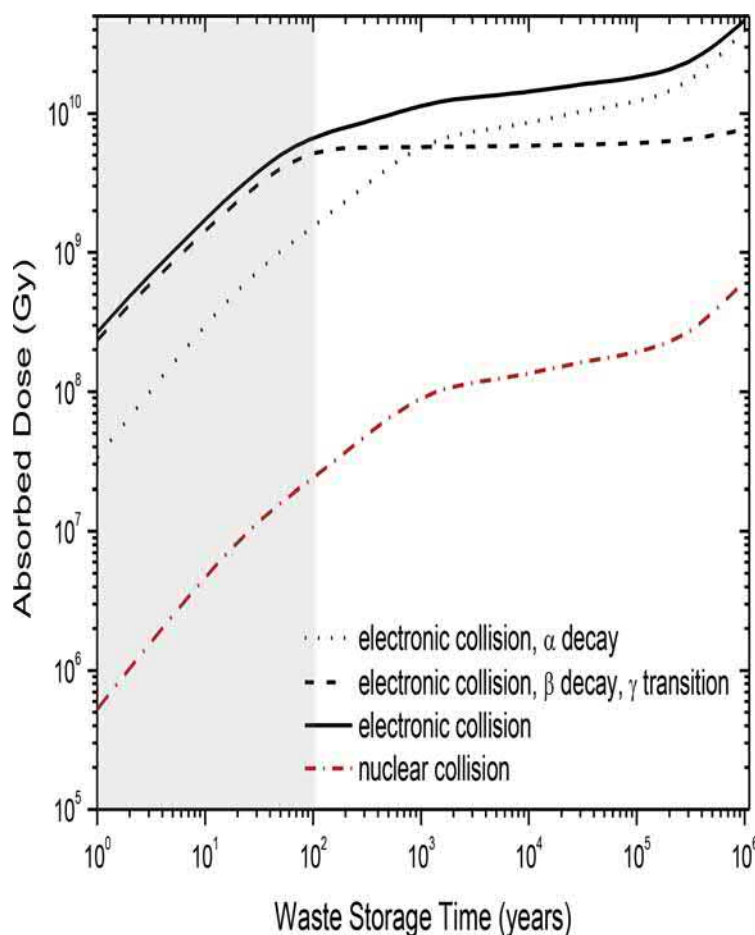


Fig. 5 Evolution of the absorbed dose of a nuclear glass with time. The gray rectangle highlights the interim storage period.

Emplacement in carefully engineered structures buried deep within suitable rock formations provides the long term stability typical of a stable geological environment. At depths of several hundred meters, in a tectonically stable region, processes that could disrupt the disposal facility are so slow that the deep rock and groundwater system remain practically unchanged over hundreds of thousands or even millions of years.

In the geological disposal, high level waste packages will be exposed to underground water after several thousands of years, with ongoing corrosion of over- and primary containers, and that is why long term behavior of glass under leaching is widely described since decades. A schematic representation of spent fuel assemblies and glass package is shown on [Fig. 7 \(Poinssot and Gin, 2012\)](#).

In order to describe the long-term performance of waste forms, it is necessary to consider the different steps of their lifetime, in relation with the environmental conditions of the waste packages (wasteform, metallic containers and surroundings materials such as concrete and host rock, clay or granite).

Two boundary conditions are considered, associated with two scenarios.

- First situation: the waste package remain intact, without any interaction with the surrounding environment. The package undergoes only an internal modification and is exposed to energy exchanges involving heat and radioactivity (residual alpha, beta and gamma radioactivity and neutrons in spent fuel). No mass transfer occurs between the system and the external environment. These conditions are close to interim storage conditions.
- Second situation: the evolution of waste forms in a water-saturated open system. Once the leaktight container is breached, the waste matrix is subjected to aqueous alteration and can exchange matter with nearfield materials, such as cementitious materials and host rock. Migration of the radioactive elements then becomes possible.

A transient state between both of the above-described scenarios corresponding to the evolution of waste forms in an open unsaturated medium must also be considered, when the nearfield environment is not entirely resaturated with groundwater when canister breaches.

As a general rule, a waste form will successively undergo these three different types of boundary conditions. The first period corresponds to the lifetimes of the container and overpack.

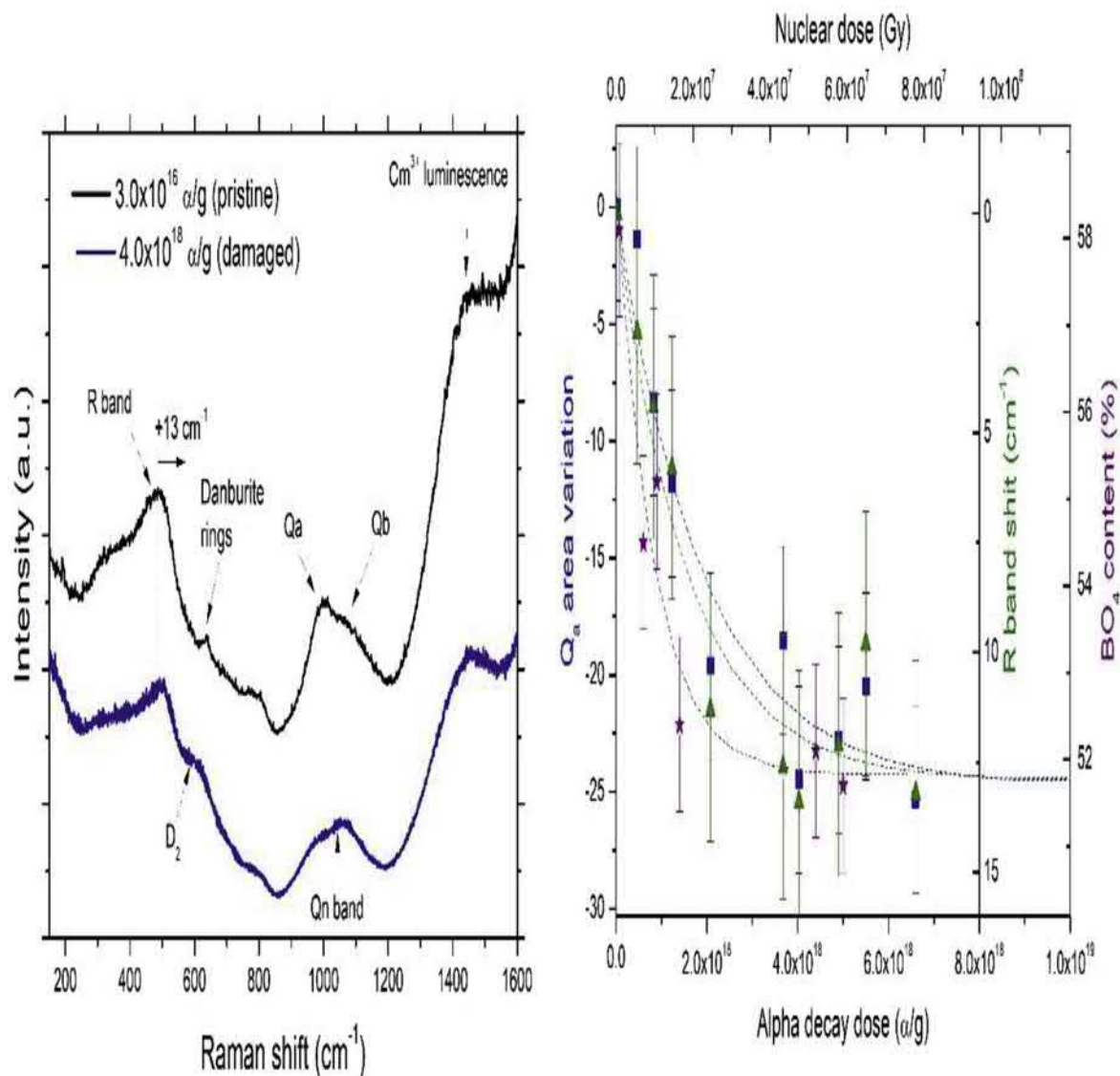


Fig. 6 Structural evolution of the laboratory ISG glass doped with ^{244}Cm . (A) Raman spectra at two alpha decay dose, (B) Evolution of structural parameters with increasing alpha decay dose.

The occurrence of the open system/unsaturated conditions scenario will depend on the water re-saturation rate of the nearfield. It is therefore strongly site-dependent. It may last longer if gas bubbles (i.e., hydrogen) are produced due to metal corrosion (due mainly to the waste containers) and avoid the complete resaturation. At long term, the waste form will be placed in open and saturated conditions.

A simplified representation of the influence of the main physical and chemical parameters on spent fuel and glass source terms is given on Fig. 8 (Poinssot and Gin, 2012).

The water transport and solution chemistry are key factors regarding the long-term durability of both matrices. However, the mechanisms through which spent fuel and glass may release their RN are quite different. First, spent fuel is very sensitive to radiolysis whereas glass is not, second and most important, part of the RN within spent fuel are instantaneously released in contact with water (instant release fraction or IRF) whereas the RN in the glass are only released when glass dissolves.

Keys elements of the evolution of SNF in a repository (ref to be completed)

The RN release from spent fuel in contact with water, such as in a geological disposal site, can conceptually be broken down into two main fractions (Johnson et al., 1988). The first fraction involves the rapid release of the labile activity corresponding to the radionuclide inventory on the free surfaces of the fuel, and the release of the grain boundary inventory, over a period of several months to several years. This fraction is often referred to as the Instant Release Fraction (IRF). Quite interestingly, part of the radionuclides

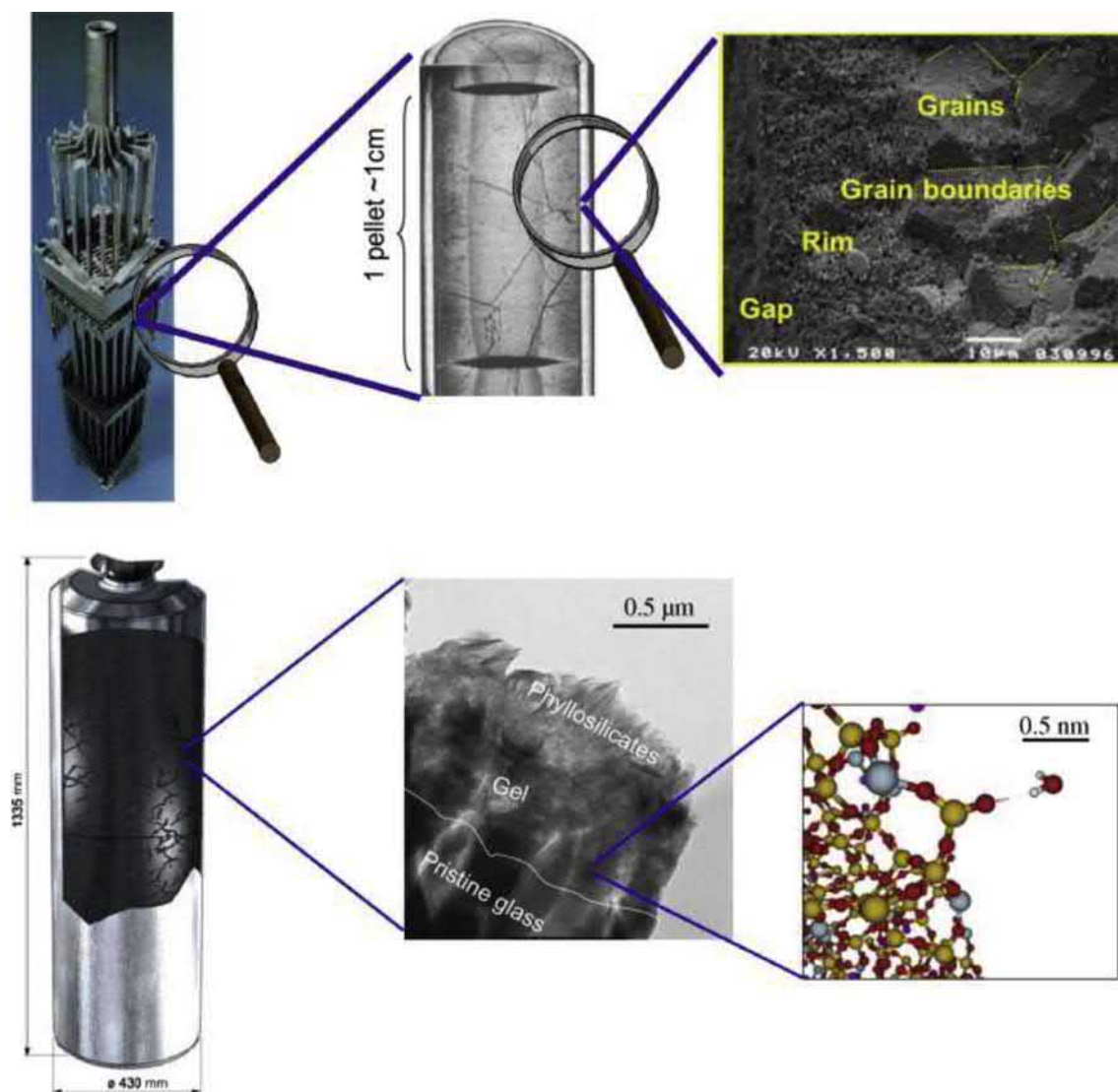


Fig. 7 (Top): Schematic representation of a spent fuel assembly comprised of numerous fuel rods (center), which are made of an external Zircaloy cladding filled with UO_2 sintered pellets. After irradiation, these pellets are strongly modified (right). Bottom: schematic representation of a glass package (left) showing the primary canister made of stainless steel filled with borosilicate glass. Glass is fractured due to rapid cooling after the elaboration and forms a gel and crystalline phases when leached by water (center). At atomic level (right), glass displays an amorphous structure with a short range order. (Down): Schematic representation of a glass package (left) showing the primary canister made of stainless steel filled with borosilicate glass. Glass is fractured due to rapid cooling after the elaboration and forms a gel and crystalline phases when leached by water (center). At atomic level (right), glass displays an amorphous structure with a short range order. Poinssot C and Gin S (2012) Long-term behavior science: The cornerstone approach for reliably assessing the long-term performance of nuclear waste. *Journal of Nuclear Materials* 420: 182–192.

associated to IRF are anions which can be highly mobile in natural environments; Therefore, IRF significantly dominated to overall impact of spent nuclear fuel in geological disposal. The second fraction corresponds to the slow release of activity as the UO_2 matrix dissolves, and may last several thousands of years. The release of most of the activity from the spent fuel rod, i.e., the activity contained in the grains, is thus controlled by the fuel dissolution kinetics (Ewing, 2015; Jegou et al., 2007; Ferry et al., 2006; Kerleguer, 2020).

Key elements of the long term behavior of glass in deep disposal

The vitrification process involves chemically bonding radionuclides to an aluminoborosilicate network, except for a very small amount of them, which are retained in the small undissolved noble metal particles. From a long-term behavior point of view, this means that radionuclides confined in such a material cannot be released in the environment if the glass remains undissolved. More precisely, no instant-release fraction occurs for HLW glass, which is a very significant different with spent nuclear fuels. The

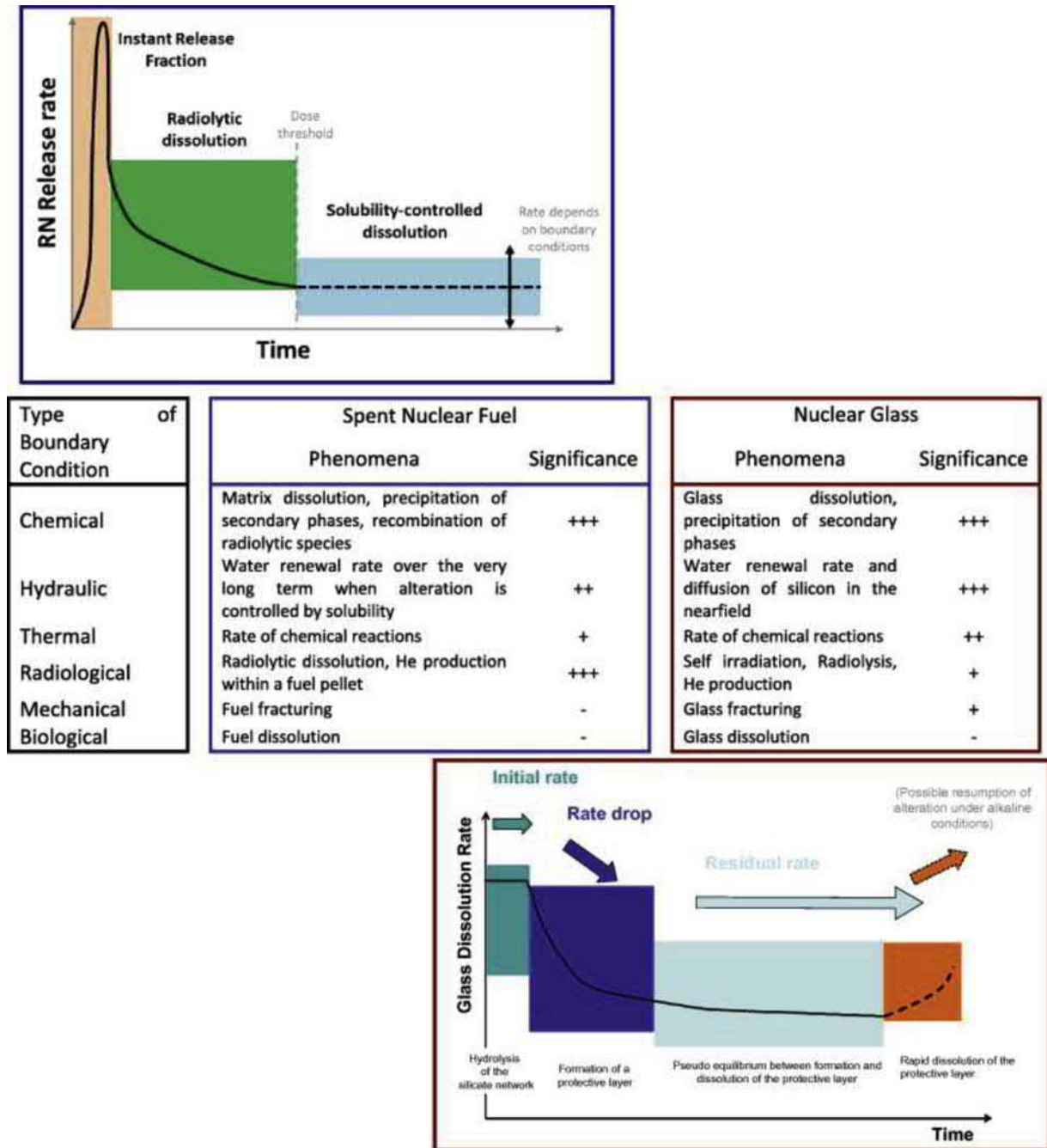


Fig. 8 Synthesis of the direct effects of the main constraints applied to waste forms in geological disposal sites (table) and their relative significance regarding the source term. Schematic representation of the main kinetic regimes of the spent fuel (up) and glass (down). Poinssot C and Gin S (2012) Long-term behavior science: The cornerstone approach for reliably assessing the long-term performance of nuclear waste. *Journal of Nuclear Materials* 420: 182–192.

solid-state diffusion process, even if thermally assisted, remains negligible. Radioactive decay produces heat and radiation damage and its potential effect is another key issue regarding the long-term stability of this type of waste form.

No significant impact of alpha/beta/gamma activity is expected on stage I (initial dissolution rate). For stage II corresponding to the residual rate regime, explored irradiation conditions seem to have very low effect both on the alteration rates and on the morphology of the alteration layer but some secondary effects can be observed such as modifications in the production of radiolytic species (Tribet et al., 2020). During glass alteration, RN are more or less retained in alteration products formed at the glass surface (gel and crystalline phases). The retained fraction mainly depends on the predominant redox form of each RN, which needs to be related to the water composition, the redox potential and the pH. The mobile fraction is made of aqueous species (potentially

promoted by ligands) and colloids. Many data are available in the literature for actinides and fission products (Gin et al., 2017), including some from integrated mockups and in situ tests, showing that under anoxic and reducing conditions expected in most of geological repositories, the solubility of most of long lives RN is very low.

Explaining the Long-Term Behavior approach requires identifying the governing mechanisms and developing models to over-estimate their influence (Poinssot and Gin, 2012; Ribet et al., 2009).

Glass alteration is described, whatever the glass composition, by the different reactions involved that are: interdiffusion (proton—alkali ion exchange), hydrolysis (Si—O—Si bond rupture), dissolved species recondensation so as to form an amorphous alteration gel which constitutes a diffusive barrier between the glass and the solution, and the precipitation of secondary crystalline phases.

These reactions are structured as four simultaneous processes which are likely to be described in terms of kinetics:

- the oxide glass is turned into a porous, hydrated, hydroxide-type phase (the gel);
- this glass hydration reaction is limited by the water diffusion transport into the pristine glass, through the gel layer already formed: in other words; the gel has a passivating effect;
- the gel is dissolved on its external surface at a rate which is a function of the solution composition and renewal conditions;
- secondary phases precipitate, thereby consuming passivating-zone formers.

At laboratory scale, the residual rate can be measured either thanks to static experiments at a high surface-to-volume ratio (S/V), or using a single-path-flow-through test at a low flow rate and a silica doped solution. When these protocols are applied to a large range of glasses, in most cases, a constant or a near-constant rate is observed, which is three to four orders of magnitude lower than the initial rate (the maximum dissolution rate corresponds to pristine glass in contact with pure water) and leads to rates of a few nm per year for UOX-type borosilicate glasses.

Modeling the long term behavior of nuclear glasses is needed to describe the glass alteration kinetics over a very large period of time, including the final residual rate regime, thanks to a mechanistic model such as the French GRAAL model (GRAAL meaning Glass Reactivity with Allowance for the Alteration layer) (Frugier et al., 2018; Poinssot and Gin, 2012; Frugier et al., 2008).

GRAAL is modeling the alteration of the glass, taking into account the evolution with time of the equilibria between the passivating layer, the secondary phases that may precipitate, and the composition of the liquid phase in interaction with the solid phases. Graal is correlated with a reactive transport model (such as HYTEC), and connected with a thermodynamical data base.

The long-term glass dissolution rate also depends on chemical and hydraulic factors that influence the fate of silicon. Silicon is the main glass former and this element is therefore involved in most of the chemical processes governing the glass dissolution process (Grambow and Müller, 2001; Gin, 2018). As the host rock is chosen for its low permeability and diffusivity, it can be expected that if the barriers placed in between the glass and the host rock do not prevent the saturation conditions with respect to the silica-rich passivating layer, the dissolution rate will be low. The low rate achieved when saturation conditions prevail is called the “residual rate” or “final rate.” These terms refer to the fact that no thermodynamic equilibrium between the glass and solution can ever be achieved and that no surface layer can completely protect the glass.

Performance assessment of geological nuclear waste repositories requires also modeling of the long-term evolution of the aqueous alterations of the fractured nuclear glass block, because the time scales under consideration are of several thousands of years and hence beyond the range of any direct experimental perspectives (Repina et al., 2020).

Furthermore, in situ glass corrosion experimentations on radioactive glasses have been carried out, giving some indications of the performances which are obtained for nuclear glass in representative repository conditions (see a review in (Gin et al., 2017)). Tests related to the geological disposal of intermediate and high level glasses have been carried out on three types of glasses with different compositions (borosilicates or aluminosilicates) in different natural environments, over several years. The results show very thin altered glass thicknesses, with the same orders of magnitude (a few microns) for the different glasses and environments, and typical of a process controlled by diffusion and point to the formation of a passivating alteration layer was observed, as observed for non-radioactive simulated glasses. Finally, accurate studies of archaeological or natural analogues demonstrates that the current nuclear glass alteration models are well predicting the order of magnitude of the overall alteration rates.

Development of alternative conditioning processes for HLW

A titanate-based fully crystallized wasteform, SYNROC, was invented by Australian scientists from ANSTO in the late 1970s (Ringwood et al., 1988; Gregg et al., 2020; Lutze and Ewing, 1988). The original form of Synroc, a multiphase ceramic wasteform based on stable and leach resistant titanate minerals was directed toward the immobilization of PUREX wastes from the reprocessing of nuclear fuels. Geologically stable titanate minerals—perovskite, hollandite, zirconolite and rutile are the main phases of Synroc.

The Synroc wasteforms are obtained thanks to HIPing processes. However, a current limitation is the lack of demonstration of this type of process at an industrially relevant scale within a nuclear facility.

Conclusion

The management of high level waste, spent nuclear fuel (SNF) and nuclear glass, is a key point in the nuclear industry, all around the world, since many decades, and for the future also. Two kinds of HLW need to be taken into account: SNF is considered as a high

level waste in some countries (United States, Sweden, Finland, Spain, Canada, etc.) or a potential future energy resource in others (France, Japan, United Kingdom, Belgium). In the last case, vitrification of the fission products coming from fuel reprocessing is the industrial process implemented in these countries, resulting in an industrial maturity which is now clearly demonstrated. Disposal in deep, stable geological formations usually several hundred meters or more below the surface is the generally recognized option for disposal of HLW. The description of the long term behavior of nuclear glass benefits from many international studies, showing that such glassy wasteform is quite relevant for HLW conditioning.

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Spent nuclear fuel long term behavior and performance

Bernd Grambow, Subatech, IMT-Atlantique, CNRS, University of Nantes, Nantes, France

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Glossary

AUC Ammonium Uranyl Carbonate (AUC) Conversion process of powders used for nuclear fuel pellet production.

Burnup Measure, how much energy is extracted from a nuclear fuel. It is a measure of the fraction of fuel atoms that underwent fission in megawatt-days/kg of heavy metal (MWd/kgHM), or similar units. Heavy metals are Uranium or Plutonium.

Dishing volume Dish like void space at the top and bottom of a nuclear fuel pellet to accommodate fission product swelling and axial thermal expansion of hot pellet centers.

Drip shield Massive shield made of corrosion-resistant titanium alloy, planned to be used in the Yucca Mountain nuclear waste repository project in the United States to protect spent fuel containing waste containers from seepage water dripping onto its outer surface.

Fluorite structure This structure refers to crystalline solids of the formula MX_2 like UO_2 , where the X ions occupy the eight tetrahedral interstitial sites and the M ions occupy the regular sites of a face-centered cubic (FCC) structure.

Half life The time it takes for the radioactivity of a specified radioactive isotope to fall (decay) to half its original value.

Ingestion toxicity The toxic effect of radioactive isotopes taken up into the human body for example by food.

IRF Instant release fraction: the fraction of inventory of radioactive isotopes released rapidly (days to month) from the used nuclear fuel upon contact with water.

Linear power rating The heat generation rate per unit length of fuel rod, commonly expressed in watts per m (w/m) of fuel rod.

LWR Light water reactor (nuclear reactor moderated by H_2O of normal isotopic composition).

MOX Mixed oxide fuel: Nuclear reactor fuel composed of UO_2 and PuO_2 , the latter obtained from reprocessing of spent nuclear fuels.

Radiolysis/radiolytic processes The splitting of water molecules by action of ionizing radiation, forming highly reactive radical and molecular radiolysis products such as $OH^\bullet$ or H_2O_2 . The radiolytic yields are different for alpha and gamma radiolysis. In the short time (up to hundreds of years) gamma radiolysis is dominant, in the long term, alpha radiolysis. Principal source for alpha radiation is the decay of actinides.

Source term Term describing the radionuclide release from spent nuclear fuel in case of groundwater access in a repository for nuclear waste.

UOX Nuclear reactor fuel composed of low ^{235}U enriched UO_2 .

Uraninite Uranium(IV) oxide (ideal formula: UO_2).

Vickers hardness The Vickers hardness gives the hardness of materials measured using a diamond indenter and a light load to produce an indentation on the material. The depth of indentation is converted into the hardness value of the object. The smaller the indentation, the harder the object.

Yucca Mountain Site for geological disposal of high-level nuclear waste in Nevada, United States. It is one of the few repository projects worldwide where geochemical conditions of natural waters are expected to be oxidizing. In most other repository concepts, reducing conditions are expected.

ε particle An agglomerate of metal alloy particles from the nanometer scale to the micrometer scale, containing mainly the fission products Mo, Ru, Tc, Rh, Pd and Te.

Introduction

The inventory of non-reprocessed spent fuel worldwide fuel has grown to some 250,000 metric tons corresponding to a total ingestion toxicity $>10^{13}$ Sv from its large contents of highly radiotoxic long-lived fission products such as for example ^{99}Tc

(half-life = 210,000 years), ^{129}I (16 million years) or ^{239}Pu (24,110 years). This stock is awaiting geological disposal to protect thousands of generations to come. Almost 50 years of research and repository planning have allowed characterizing the properties of spent fuel relevant for disposal safety, considering the factors of influence both in the fuel irradiation history as well as in the repository near field.

After reactor operation and 10 years of cooling typically about 4.4% of the initial Uranium content of typical light water reactor (LWR) fuel has been transformed to 0.9% Plutonium, 0.1% minor actinides (Am, Np, Cm), 0.3% radioactive Cs and Sr, and 0.2% long lived fission products like I, Tc and others and 2.9% stable fission products (Blue Ribbon Commission, 2012 for an average burnup of about 36 MWd/kgU for the a mix of PWR and BWR reactors). The decay of short-lived fission products will lead to a change over thousands of years in composition, radioactivity and heat production (Bruno and Ewing, 2006). Heat generation increases with fuel burnup and decreases over time. Evolution of heat generation and radionuclide inventories are typically calculated using nuclear modelling and simulation code packages like SCALE (Bearden and Jesse, 2018), using as input the initial fuel composition, the initial enrichment, considering the trace element contents of used materials, nuclear data libraries, the burnup and the irradiation history and the linear power ratings. Uncertainties can be about 2% for U and Pu contents, 7% for fission products and 11% for minor actinides (SKB, 2010a). Uncertainties in activation product (^{14}C , ^{36}Cl , ...) contents can be much higher as often no measurements are available and upper bounding values are obtained from maximal materials specification values for inactive trace impurities. Detailed burn-up and fuel charge history data are acquired from reactor operators before reliable calculations can be made to correctly dimension the heat load and emplacement distances to optimize space requirement in a nuclear waste disposal site. For equal burnup, heat generation is higher for MOX fuel than for UOX fuel. Cooling remains necessary even after 200 years of decay.

Radionuclide distribution in spent nuclear fuel

The radionuclides and non-radioactive fission isotopes are heterogeneously distributed in the irradiated nuclear fuel, as seen in Fig. 1.

The major amount of radionuclides are bound to the nuclear fuel matrix (UO_2), others accumulate at grain-boundaries, fracture surfaces, fuel plenum, dishing volumes or the gap between cladding and fuel (Ferry et al., 2005):

Fuel matrix: both in solid solution and at inter-lattice positions are located more than 99% of the actinides (Pu, U, Np, Am, Cm), rare earth elements, Ce, Nd, La, Pm, Pr, Sr, and some 80% to 90% of Xe, Kr (Ferry et al., 2005) and as well for He, Cs and I. These latter elements are rather volatile under reactor operation.

Grain boundaries: Many radionuclides accumulate in grain boundaries separating the typically about 10 μm large fuel grains. This includes fission gases and radionuclides of lower solubility in the fuel matrix such as $^{135,137}\text{Cs}$, ^{129}I , ^{90}Sr , ^{99}Tc .

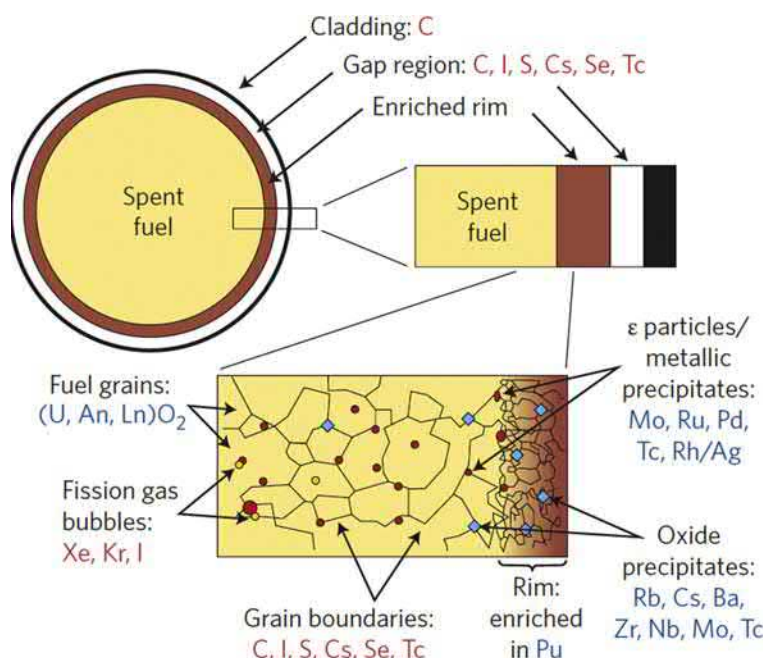


Fig. 1 Schematic representation of the structure of spent fuel. Elements in red are released rapidly on first contact with water, while elements in blue are released more slowly as the UO_2 matrix corrodes (An = actinides and Ln = lanthanides in solid solution in the UO_2 structure). From Bruno J and Ewing RC (2006) Spent nuclear fuel. *Elements* 2 : 343–349. <https://doi.org/10.2113/gselements.2.6.343>, with permission from Mineralogical Society of America.

High burnup structure: this porous structure is formed due to high local burnup of ^{239}Pu , which is formed by resonance capture of epithermal neutrons on ^{238}U and beta decay at the fuel rim at an average burnup of about 60 GWd/tU following recrystallization of UO_2 grains. In case of MOX fuel, the high burn-up structure does not correspond to the rim but to the Pu rich islands within the U rich matrix.

Metallic inclusions like the ϵ particles are composed of the metallic fission products ^{107}Pd , Rh, Ru, Mo, ^{99}Tc alloy or (^{107}Pd , Te, ^{126}Sn) alloy or *oxide inclusions* of BaZrO_3 , formed from the fission products Ba and Zr. These particles have a size of few nm to hundreds of nm depending on the linear power and burnup under reactor operations.

Gap, void and fracture inventories: These include $^{135,137}\text{Cs}$, ^{90}Sr , ^{99}Tc , and ^{129}I . Fission gases in these volumes are released upon cladding failure during handling or disposal. Upon groundwater contact and cladding failure, these nuclides are first released, forming an instant release fraction IRF.

Cladding: These zirconium alloys with a zirconia (ZrO_2) surface and a crud deposit outside of the cladding include neutron activation products like ^{14}C or ^{36}Cl , alpha recoil nuclides, fission fragments as well as radioactive deposits (^{59}Ni , ^{63}Ni) from the primary coolant circuit.

Structural fuel parts: This includes both steel and Inconel, contaminated typically by ^{59}Ni , ^{63}Ni and ^{36}Cl .

Due to the absence of operational repositories for spent nuclear fuel today, an extended interim storage is necessary. Even though spent fuel is able to withstand long-term interim storage without major modification of its disposal relevant properties, some potential effects are of concern. Highest priority items are the stress corrosion cracking (SCC) of welded stainless steel (SS) canisters and the hydride effects (reorientation and embrittlement) of high burnup fuel cladding. Research items include the heat transfer (effects in SCC and temperature peak at drying) and the reorientation of hydrides of the cladding during the initial drying process and its relation with the Ductile Brittle Transition Temperature (DBTT) which is a key parameter regarding the transport after dry storage. Some of the processes might continue during disposal before contact of spent fuel with groundwater.

Alpha-decay damage is the dominating effect modifying the structure of spent nuclear fuel during long term storage prior to water access. The fluorite structure of UO_2 is preserved with an increase in lattice parameter (up to 0.6%), the formation of dislocation loops, the formation of helium bubbles and an increase of the Vickers Hardness. Bubbles are not expected to start forming in conventional spent fuel for several thousand years; however, in MOX fuels, they could begin forming within 1000 years (Wiss et al., 2014). High mobility of He bubbles is confirmed by natural analog uraninite samples of age of many 100 millions of years (Roudil et al., 2008).

In case of cladding failure and wet storage, many of the processes discussed further below for fuel water interaction under repository conditions apply. However, oxidizing conditions are generally expected in this case.

Long term interaction of spent fuel with water: Distinction between the instant release fraction IRF and release from fuel matrix

Most repository concepts for spent fuel worldwide foresee geological disposal in containers beneath the water table, resulting in container corrosion over periods of typically ten to hundred thousands of years (earlier for repositories in salt formations) producing by slow corrosion $\text{H}_2(\text{g})$ and $\text{Fe}(\text{II})$. After container breach, the initially very hot used fuel has cooled down and fuel immersion in water will occur at close to ambient temperature. The cladding is very stable in water and protect the fuel from groundwater contact probably for some additional tens of thousands of years. However, since no statistically relevant study of long term cladding stability has been conducted, current safety assessments do not take credit from cladding tightness and assume cladding breach with container breach exposing the spent nuclear fuel to water. An exception is Yucca Mountain/Nevada, where the protection by the cladding is taken into account (Ahn and Mohanty, 2008). Corrosion by groundwater of cladding and structural materials will lead to leaching of activation products like ^{14}C (partially organic), $^{59,63}\text{Ni}$ or ^{36}Cl . The corrosion of Zircaloy under repository conditions and the associated radionuclide release were discussed in Johnson and McGinnes (2002). After removal from reactor, the cladding (thickness of about 600 μm) contains a thin oxide film of tens of μm of ZrO_2 . Suggested maximal corrosion rates are about 10 nm/year.

The interaction between spent fuel (without or with defect cladding) and groundwater ("leaching") is being studied since more than 45 years. Early results show (1) that the leach rate decreases from $\text{Cs} > \text{Sb} > \text{Sr} + \text{Y} > \text{Pu} > \text{Cm}$ with about 1% of the inventory of Cs^{137} being released (Katayama, 1976), (2) that the fractional release rate of Pu decreases over many orders of magnitude in few days and (3) that release rates are relatively insensitive to temperature (Johnson, 1982). The concentrations of both U and Pu attain saturation rapidly in aqueous solutions contacting spent fuel and the leaching rate for Sr^{90} decreased from about $10^{-5}/\text{day}$ to $10^{-7}/\text{day}$ but remained higher than for U, Pu, Cm, Ce, Eu and Ru. In reducing geochemical conditions, expected in European repository concepts, significantly lower release rates are attained (Forsyth et al., 1986) due to low solubility of many radionuclides including uranium.

Other, more water-soluble radionuclides do not attain solubility. Release kinetics vary over time, from the first groundwater contact after container breach to over hundreds of thousands of years of water contact, distinguishing between radionuclide specific release from the fuel matrix, from gaps, from accessible grain boundaries, from cladding and structural metals. A major conceptual choice traditionally made lies in codifying this heterogeneity by only two sources of release: the fuel matrix and an instant (or fast) release fraction, IRF, representing everything being released faster than the matrix. Schematically the temporal evolution of these sources of release are given in Fig. 2. A fraction of a given radionuclide content of the fuel may be attributed to the matrix, the other

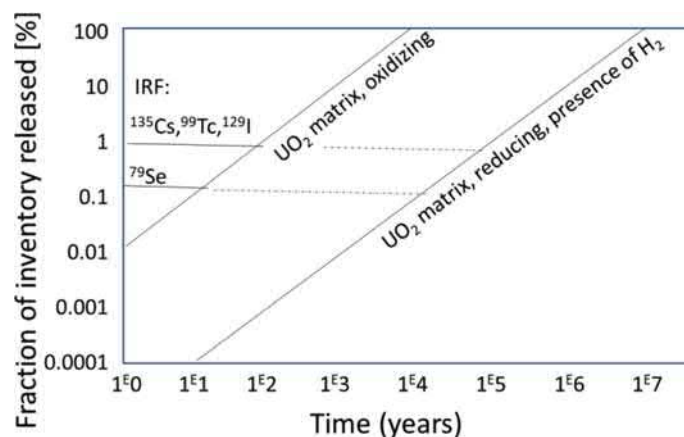


Fig. 2 Typical model for long term radionuclide release with instant release (IRF) and fuel matrix contributions. IRF values are considered independent on environmental conditions but depend on spent fuel irradiation history, matrix contributions depend principally on the oxidation state of the environment.

is conceptualized in the IRF. Two type of IRF constituents are distinguished: those, which under reactor operation have similar volatility as fission gases and, hence, their mobility is linked to fission gas release, FGR, others are independent of FGR. Accessible fractions of grain boundary inventories may count as IRF, inaccessible ones count in the matrix. In some rare cases (Ferry et al., 2005), IRF(t) values for different time scales are given, accounting for diffusion toward low-performance microstructure before water ingresses the canister.

The release from the fuel matrix is controlled by the dissolution rate of the UO_2 (or UO_2/PuO_2 in case of MOX) matrix of the fuel. Release rate control by matrix dissolution is relevant for all radionuclides dispersed in the fuel matrix: those being in solid solution but also those in metallic or oxide precipitates dispersed in the fuel matrix and even those being incompatible with the fuel matrix like gases, trapped after fission or alpha decay (He) in the fuel grains.

Assessment of the IRF inventory as a function of spent nuclear fuel characteristics and irradiation history

The IRF inventory varies significantly with spent fuel characteristics. After Helium as alpha decay product and the fission gases, the most mobile elements in the fuel both under reactor conditions and after cool-down are iodine, tellurium and bromine, the rare gases and cesium. IRF inventories for $^{135,137}\text{Cs}$, ^{36}Cl and ^{129}I are strongly linked to FGR due to diffusion coefficients in the fuel (Walker and Lassmann, 1986). Indeed, transport from grain center to grain-boundaries is governed by thermal diffusion at the hot fuel centerline and by slower radiation assisted athermal diffusion beyond about mid-radius. Grain-boundaries provide further pathways across the fuel. This leads to much higher FGR, $^{135,137}\text{Cs}$, ^{129}I and probably ^{36}Cl release from the hot fuel center and since thermal diffusion in the fuel rod central region is strongly depending on temperature gradients under reaction operation conditions, IRF and FGR are strongly linked to linear power as already observed for ^{137}Cs very early (Eklund and Forsyth, 1978). With increasing burnup and formation of the high burn-up rim structure (Fig. 2) an acceleration of FGR is observed (Mogensen et al., 1999; Brémier et al., 2000). The ratio of IRF to FGR has been observed to be about 1:3 ratio for $^{135,137}\text{Cs}$ and 1:1 for ^{129}I (Johnson et al., 2004, 2012). These ratios are linked to ratios of diffusion coefficients of the fission products in the UO_2 matrix. However, in contrast to fission gases, no accelerated release of IRF nuclides was observed from the high burnup region (Fors et al., 2009; Serrano-Purroy et al., 2012; Johnson et al., 2012). The same absence of accelerated diffusion from the HBR zone (the Pu islands, Fig. 3) is observed for MOX fuel (Carbol et al., 2009b).

The European project “FIRST-Nuclides” (Lemmens et al., 2017; Kienzler et al., 2015) has studied the fast release of fission gases, $^{135,137}\text{Cs}$, ^{129}I , ^{14}C , ^{79}Se , ^{99}Tc , ^{126}Sn , etc. from gap and UO_2 grain boundaries, identifying a clear link to reactor operation conditions such as burn-up, power rating and calculated fission gas releases. Abnormally high fission gas release and IRF values were observed for abnormally low density fuels like ATM-106 as well as for fuels with high linear power ratings.

A mechanism contributing to the IRF is grain-boundary release. The IRF was anticipated to grow in the early 2000s due to diffusion mechanisms, which was finally not supposed to be significant enough. Diffusion coefficients in grain-boundaries are orders of magnitude higher than in the matrix. Distinction of gap and grain-boundary contributions to the IRF values were attempted by Serrano-Purroy et al. (2012).

After about 0.5–1 year the release rates of Cs and I are in the same order of magnitude as observed for matrix dissolution. Due to the operational definition of the IRF value, a clear relation between the observed IRF of redox sensitive elements and elements incorporated in the UO_2 matrix has been observed, demonstrating the effect of the leaching conditions, allowing a better definition of the transition between both release mechanism: Fast release can also be due to fast dissolution of the fuel matrix. Moreover, matrix

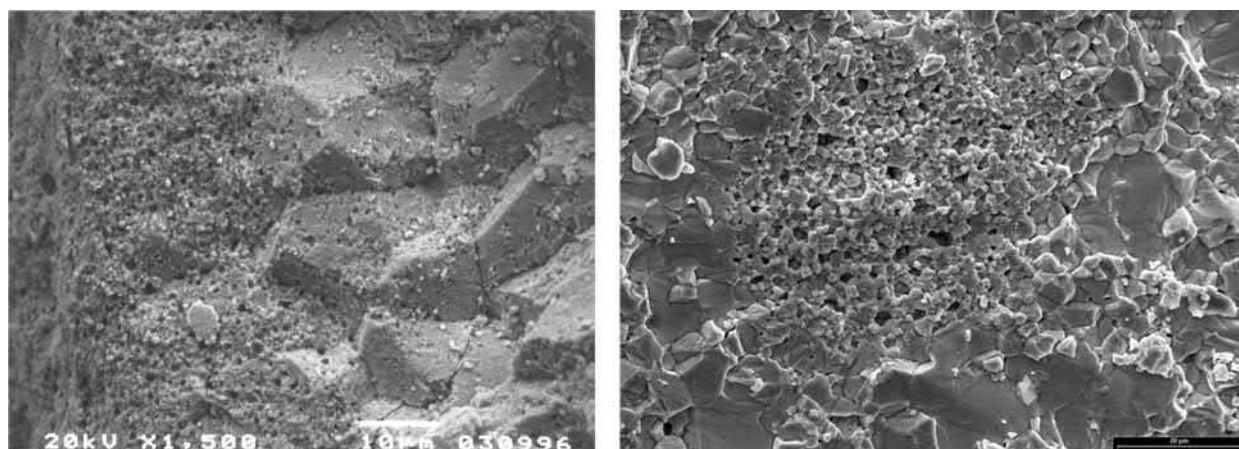


Fig. 3 SEM examination of the periphery ("rim") of a UOX pellet (left) and aggregates in AUC MOX fuel after three cycles (burnup ≈ 30 GWd/t) 1200 μm from the edge of the fuel (right) (Ferry et al., 2005).

Table 1 Estimates for IRF values in [%] for PWR fuel as a function of burnup in GWd/tU.^a

	PWR	PWR	PWR	PWR	PWR
Burnup	37	41	48	60	75
Fission gas	0.3	0.4	1	3.6	6.5
% FG in rim	0	0	2	4	8
¹⁴ C	10	10	10	10	10
³⁶ Cl	0.9	1.2	3	10.8	19.5
⁷⁹ Se	0.2	0.2	0.2	0.2	0.2
⁹⁹ Tc	1	1	1	1	1
¹⁰⁷ Pd	1	1	1	1	1
¹²⁶ Sn	0.1	0.1	0.1	0.1	0.1
¹²⁹ I	1	1	2.5	7.6	10.5
^{135,137} Cs	1	1	2.5	6.1	7.8

^aAfter the publication of Johnson et al. (2005) it became evident by new data that in contrast to gases, other fission products are effectively retained in the high burnup rim zone and no preferential release is observed. The values of Johnson et al. (2005) were initially kept forward in Grambow et al. (2008) as pessimistic values, but in the present table, the pessimistic values for Sn, Tc, Pd, Se are omitted as evidence is clear (see references in this paper) that there is no preferential release from the HB rim zone. As in Johnson (2014), only fission gas released to gap was used as reference for Cl, Cs and I release data, using average fission gas data from figure 3.2 in Johnson et al. (2004). To account for model uncertainties in representing release from grain-boundaries, an additive factor IRF contribution of 4% was added by Johnson (2014) to the IRF values of Cs and I, corresponding to the fission gas inventory in the rim zone at a burnup of 60 GWd/tU. As the grain boundary inventory is much lower at lower burnup, the values in SKB (2010b) were used for lower burnup.

oxidation tends to open grain boundaries and thus causes the continuous exposure of fresh grain boundaries with high concentrations of soluble IRF nuclides (Kienzler et al., 2015).

Various models are described in literature to assess the contribution of IRF values to the source-term of spent nuclear fuel in the performance assessment of nuclear waste repositories. A model considering all known uncertainties at the time was developed (Poinssot et al., 2005a) attributing no confinement properties to grain-boundaries, high burnup structure and gap. Grain boundary inventories are in general lower than gap inventories (Johnson et al., 2004). Using the above-mentioned relations between FGR and Cs and I release, best estimate and pessimistic gap release values were suggested as a function of burnup, as well as grain boundary inventories including the high burnup rim porosity. However, it has been shown that actually release from the rim zone is lower from the rim than from the center of the fuel. The data were corrected with some more recent approaches and insights to calculate IRF values for various nuclides as a function of burnup (see Table 1). For example, recent literature indicates the absence of fast release of Se (Johnson et al., 2012) bounded by a maximal fast release of $<0.2\%$, explained by direct binding of Se to U in the fuel matrix (Curti et al., 2015). For MOX fuel best estimate IRF values are also obtained from fission gas release values (Johnson et al., 2004).

Assessment of the matrix dissolution behavior, role of radiolytic oxidants, hydrogen and near-field buffering

The dissolution of the UO_2 matrix is the condition for the release of matrix bound radionuclides into groundwater and it provides as well an upper limit for the dissolution of nano-crystalline phases like the ϵ phases, finely dispersed in this fuel matrix. In case of

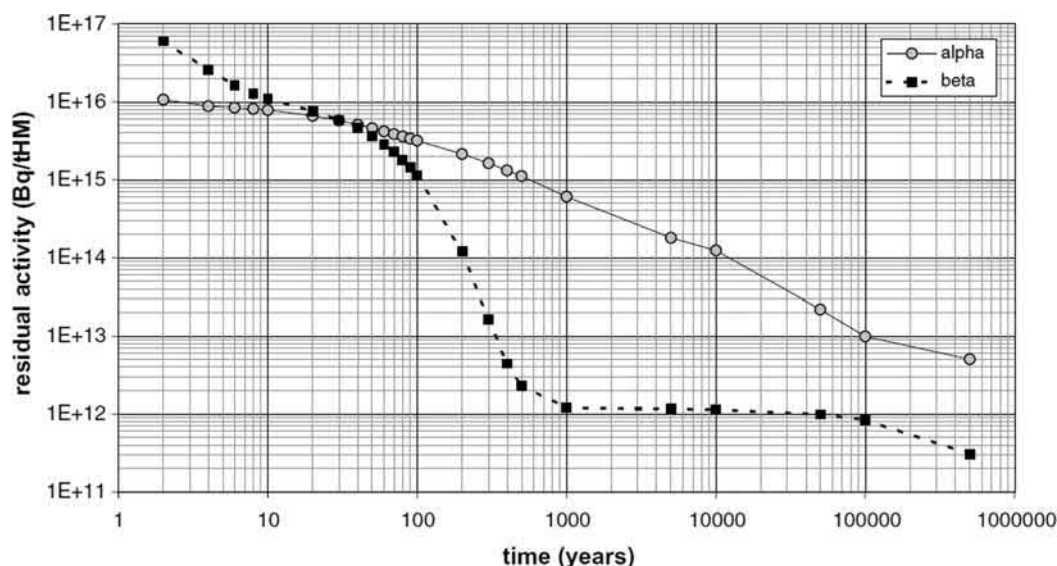


Fig. 4 Evolution of alpha and beta radiations as a function of time for a 55 GWd/t UOX fuel. Data are calculated from the CESAR isotopic evolution code. Reprinted from Poinssot C, Ferry C, Lovera P, Jegou C, and Gras J-M (2005b). Spent fuel radionuclide source term model for assessing spent fuel performance in geological disposal. Part II: Matrix alteration model and global performance. *Journal of Nuclear Materials* 346 : 66–77. <https://doi.org/10.1016/j.jnucmat.2005.05.021>, with permission from Elsevier.

oxidizing conditions there is surface oxidation from U(IV) to U(VI), relatively fast dissolution as described by Schortmann and De Sesa (1958), Grandstaff (1976), and Leider et al. (1991), accelerated dissolution in presence of carbonate, formation secondary reaction products like schoepite $((\text{UO}_2)_8\text{O}_2(\text{OH})_{12} \cdot 12\text{H}_2\text{O})$, studtite $(\text{UO}_4(\text{H}_2\text{O})_4)$, Uranophane $(\text{Ca}[(\text{UO}_2)_2(\text{SiO}_3\text{OH})]_2 \cdot 5\text{H}_2\text{O})$ or becquerellite $(\text{CaU}_6\text{O}_{19} \cdot 11\text{H}_2\text{O})$. Reversely, under reducing conditions expected for European repository concepts, dissolution of the fuel matrix is extremely slow, the solubility being controlled by UO_2 (about 10^{-10} M) and a potential phase transformation to coffinite (USiO_4) (Ewing, 2015). As the solubility and water flow rates are very low the release rates of matrix bound radionuclides shall remain also very low.

However, it became evident (Forsyth et al., 1986; Christensen et al., 1994; Christensen and Sunder, 2000) that the radiation field can lead to oxidative spent fuel dissolution even in globally reducing environment. The interrelation between the electrochemistry of oxidative dissolution, of solution chemistry and radiation field is reviewed in detail by Eriksen et al. (2012). Radiation decomposes water molecules, forming very reactive radical $(\text{OH}^\bullet, \text{HO}_2^\bullet, \text{e}_{\text{aq}}^-, \text{H}^\bullet)$ or molecular radiolysis products $(\text{H}_3\text{O}^+, \text{H}_2, \text{H}_2\text{O}_2)$ depending on the nature of the radiation field (α , β or γ) and on the deposited dose. The radiation dose will decrease over time and α -radiolysis will become dominant in the long term (see Fig. 4).

The radiation field of actual spent fuel is very strong and not representative for the radiation field expected for 1000-year-old fuel. To simulate long-term spent fuel behavior in water, alpha doped UO_2 can be used. Radiolysis provides a strong source for oxidation of UO_2 to UO_{2+x} (Sunder et al., 1990) allowing for oxidative dissolution in presence of groundwater species like carbonate ions. There is a threshold dose, below which alpha radiation is unable to sustain oxidative dissolution (Poinssot et al., 2006), see Fig. 5.

The threshold depends on environmental conditions. A higher dose threshold is obtained in the presence of H_2 gas formed by container corrosion. Indeed, the dissolution rates even of fresh spent fuel dropped in presence of H_2 by four orders of magnitude, when compared with oxidizing conditions (Röllin et al., 2001), see Table 2.

With H_2 concentrations as low as 5.9×10^{-6} M, the oxidizing effect of radiolysis and in particular of formed H_2O_2 is completely blocked by galvanic coupling with the ϵ phases (Wu, 2014) and low dissolution rates are observed, as expected under reducing conditions. Reduction of radiolytic oxidants and of slightly oxidized UO_2 to $\text{UO}_{2.00}$ occurs as well by activation of H_2 on a UO_2 surface without ϵ -particles (Carbol et al., 2009a). Solution concentrations of uranium are controlled by the solubility of UO_2 . For the other radionuclides incorporated in the spent fuel matrix, the solubility of UO_2 is also a boundary limiting maximum solution concentration. Behavior in cementitious environment is similar to that in neutral groundwaters (Lemmens et al., 2019). The interlinked processes leading to spent fuel dissolution are illustrated in Fig. 6.

In the presence of failed cladding, the dissolution rate decreases significantly, probably due to mass transfer resistance (Wilson, 1990), studying the corrosion of cladged spent fuel with cladding defects in form of a slit (0.015 cm diameter/2.5 cm length) or a hole (0.2 mm diameter) in oxic Yucca Mountain water at 85 °C. This lead to a decrease of dissolution rates by up to 2 orders of magnitude for Tc99, Sr90 or I129. No such effect is expected under reducing conditions, as corrosion rates are much slower, implying a lower significance of rate controlling mass transfer.

Various models have been developed describing fuel matrix dissolution for the source term in the context of the performance assessment of nuclear waste repositories. The model uncertainties have been assessed in the MICADO project (Grambow et al., 2011). The models typically consider (1) the generation of oxidants by a radiolytical model considering radiation field and

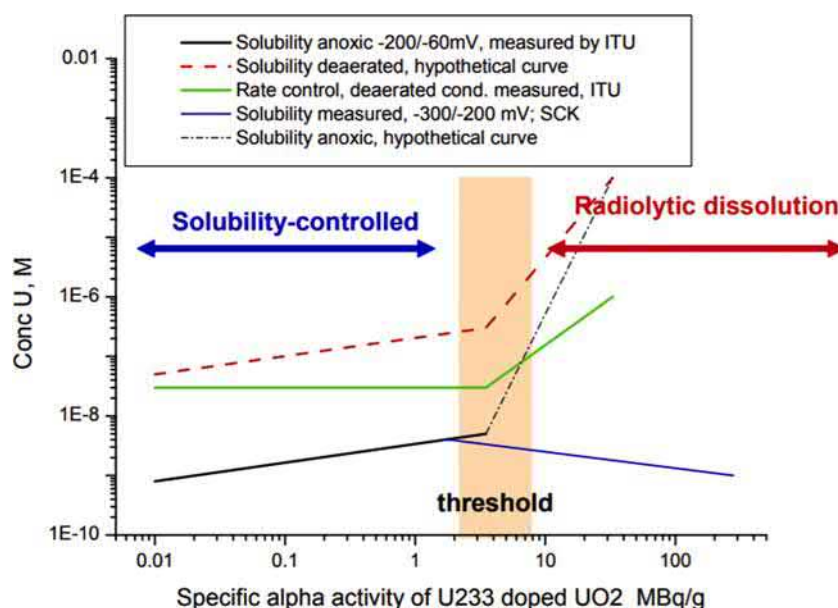


Fig. 5 Threshold activity of the spent fuel for alpha radiolytic enhanced oxidative dissolution (Poinssot et al., 2005a).

Table 2 Typical spent fuel matrix dissolution rates for various environmental conditions, if necessary: fractional dissolution rates converted using a typical external specific fuel surface area of 2 ± 1 cm/g (Grambow et al., 2011).

Dissolution rate (g/m ² /day)	Tracing elements	Conditions	References
$2 \pm 1 \times 10^{-3}$	Sr, Cs	Spent BWR fuel, oxid, static, after 1000 days	Forsyth et al. (1986)
8×10^{-3}	U, Cs, Tc	Oxidized and unoxidized spent PWR fuel, oxid, after 20 days	Gray and Wilson (1995)
$0.2\text{--}8 \times 10^{-3}$	U, Tc, I, Np	Spent fuel, oxid, yucca mountain water, pH 8.0–8.4	Ahn and Mohanty (2008) review
$2 \pm 1 \times 10^{-3}$	Np, Tc, Sr, Rb, Ba, U, Cs	Spent BWR fuel, oxid, carbonate, pH 8–9, flow test	Röllin et al. (2001)
5×10^{-6}	Cs, Sr, U, Pu	Spent PWR fuel, reducing, 3 bar H ₂ , 5 M NaCl brine	Loida et al. (2001)
$1.3 \pm 1 \times 10^{-5}$	U	α doped UO ₂ , dill. NaCl, pH 7, anoxic, isotope dill.	Ollila (2008)
$3 \pm 2 \times 10^{-6}$	Cs, U	Spent BWR fuel, reducing, H ₂ , pH 5–9, flow test	Röllin et al. (2001)
$10^{-7}\text{--}10^{-5}$	U, Pu, Np, Cs, Mo, Tc...	α doped UO ₂ , HB spent fuel, MOX, reducing, H ₂ > 10^{-3} M	Carbol et al. (2005)
$1.5 \pm 1 \times 10^{-6}$	U	α doped UO ₂ , reducing (Fe)/H ₂ , dill. NaCl, pH 7, isotope dill.	Ollila (2008)
$10^{-5}\text{--}10^{-2}$	Fuel matrix	Commercial spent fuel, reference for performance assessment, immersion oxid, Yucca mountain	Ahn and Mohanty (2008)
10^{-6}	Fuel matrix	Spent BWR fuel, reference in performance assessment; granite, reducing, SKB	Werme et al. (2004)

type, (2) oxidation of the spent fuel surface, and (3) dissolution of the spent fuel matrix and radionuclide release according to the uranium speciation and groundwater composition with or without consideration of electrochemical processes and solution complexation. Conservatism was established by ignoring radiolytic reductants (Poinssot et al., 2005b) or only considering H₂O₂ as radiolytic reactant. A small list of typical models is given in Table 3.

Recent simulations (Liu et al., 2017) have demonstrated that H₂O₂ is the primary radiolytical oxidant driving spent fuel corrosion. The simultaneous radiolytical production of H₂ and its activation on ϵ -phases, rapidly lead to the suppression of this rapid corrosion reaction to insignificant radiolytical rate values (i.e. $< 10^{-20}$ mol/m²/s). Radiation dose rate, the H₂O₂ decomposition ratio, and the surface coverage of ϵ particles all influence the time necessary to achieve these very low corrosion rates. As for natural uraninite, one can conclude that non-oxidizing UO₂ dissolution rates become dominant in reducing repository conditions being controlled either by the very small solubility of UO₂ or a dissolution/precipitation or dissolution/sorption reaction involving dissolution of spent fuel and precipitation of a U(IV) solid phase like coffinite. Isotope dilution tests have been used to investigate the role of precipitation or sorption on the dissolution rate of UO₂ under expected reducing conditions (Fe). While ²³⁸U concentrations in static dissolution tests correspond to the theoretical solubility limit of UO₂ in granitic water (10^{-10} M), isotope dilution indicated continued dissolution with a very low rate 10^{-6} g/m²/year, see Table 2 (Ollila, 2008). These data suggest that a UO₂ solubility

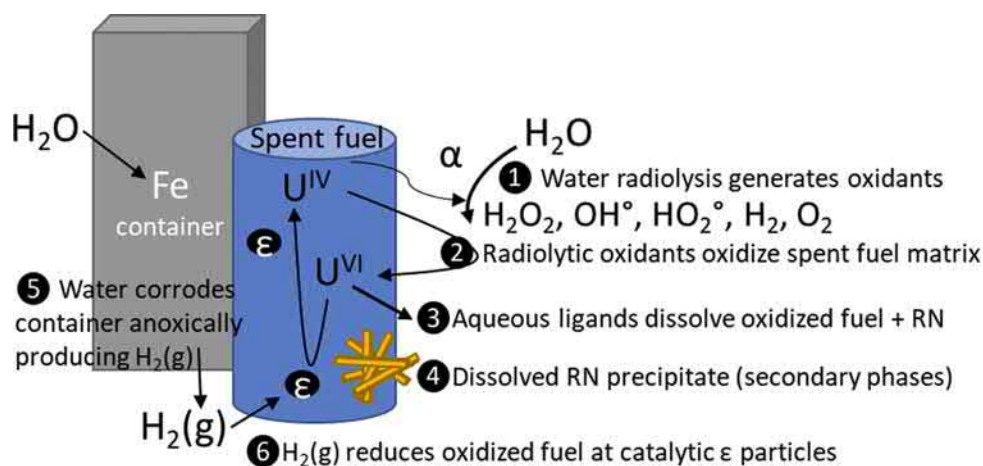


Fig. 6 Schema describing the processes interlinked in the dissolution of the spent fuel matrix (IRF not taken into account).

controlled dissolution model is not conservative when describing the long-term release of matrix bound radionuclides from spent fuel under reducing conditions.

A best estimate of dissolution rate has been obtained (Werme et al., 2004) from the measured very low solution concentration (around 10^{-10} M) at the end of a long-term static tests in reducing waters, assuming that the release rate is linear. This method overestimates the long-term release rate since in all cases, where testing with expected H_2 concentrations has been conducted, the dissolution rate decreases with exposure time. This approach has shown that there is indeed no enhanced dissolution due to alpha radiolysis under the conditions that will pertain in a repository in granite formations. Matrix dissolution rates of relevance to soluble fission products and to actinides from spent UO_2 fuel are given as a triangular distribution with a range from 10^{-6} to 10^{-8} per year with a peak at 10^{-7} per year (Werme et al., 2004). While the dissolution rate of MOX fuel of moderate burnup is about 7 times higher than of UO_2 based fuel, the uranium oxide matrix of the spent MOX fuel appeared to be the main contributor to the measured dissolution in the presence of H_2 , while the corrosion of the high burnup Pu rich islands were negligible (Carbol et al., 2009b). Many radionuclides will have much lower release rates as expected from the fuel dissolution rate due to solution control by solubility or coprecipitation (Metz et al., 2012) in secondary alteration products. There was some discussion as to whether the release from spent fuel under reducing conditions is rate controlled or solubility controlled. However, recently (Grambow et al., 2017) it has been shown that both processes are linked and have the same physical-chemical origin: the kinetic interpretation of solubility equilibrium involves equal sorption/desorption rates, which in case of UO_2 involve much less than the dissolution of an atomic layer on the solid. With a maximum reaction rate of 10^{-6} – 10^{-8} /year for matrix bound radionuclides and a solubility of dissolved U of 10^{-10} M more than 99% of released Uranium will be immobilized again, hence one shall call this process spent fuel conversion and not fuel matrix dissolution as in SKB (2010b). The above mentioned continued isotopic exchange on UO_2 surfaces despite solubility equilibrium (Ollila, 2008) cannot be used as proof signifying that there is a significant solid state transformation despite UO_2 saturation as it may only involves restructuring of few surface sites on UO_2 . This does not rule out a dissolution/reprecipitation mechanism driving release of matrix bound radionuclides, for example by coffinite formation, but since less than a mono-layer is involved in this exchange (Grambow et al., 2017), the distinction between sorption and precipitation has only semantic character. And if the UO_2 or MOX surface is thermodynamically stable under reducing conditions, other elements like Tc, Mo, Np, Pu and Am initially released from the fuel actually are observed to decrease their concentration in solution to values $< 10^{-9}$ M and not to increase it (Fig. 3.29, Carbol et al., 2005).

Summary

Forty years of research have shown the very high kinetic and thermodynamic stability of the spent fuel under reducing geological disposal conditions. One can conclude that due to the high fuel matrix stability even in presence of radiation fields, overall release of radiotoxic elements from the spent fuel in a repository under reducing geochemical conditions is largely dominated by release of the IRF inventories. In repositories in clay-rock, it is in particular the release of ^{129}I which will be dose limiting to future generations living in the vicinity of the disposal location, yet largely inoffensive to any health impact. While reliable estimations of ^{129}I release can be made today as a function of burnup and linear power, it would be useful to have a larger data set for fuels of different irradiation histories to reduce the overall uncertainties when deriving maximal ^{129}I release data from available gas release data. In contrast, for oxidizing repository environments like those targeted for Yucca Mountain in Nevada, United States, while expecting similar release of ^{129}I , much higher release of matrix bound actinides and Tc is expected upon immersion as the dissolution rate of the fuel matrix is about 1000 times faster (see Table 2). In the absence of intrinsic fuel stability, other factors like a protective role of the cladding or a drip shield need to be taken into account to assure high spent fuel stability.

Table 3 Typical models used to describe spent fuel dissolution.

<i>Model description</i>	<i>References</i>
Model 1: The Matrix alteration model (MAM) was developed during the SFS and MICADO project by Amphos21, CIEMAT and ENRESA, considering water radiolysis, geochemical solution and surface complexation reaction (kinetics), calibrated with a large set of data on UO_2 dissolution as a function of pH, H_2O_2 and carbonates concentration, successfully applied α doped UO_2 and spent fuel. The effect of hydrogen on spent fuel dissolution rates is described as a homogeneous effect in bulk solution.	Martinez Esparza et al. (2005)
Model 2: The H_2O_2 model was developed by KTH. It considers water radiolysis and diffusion of species and heterogeneous kinetics at the SF surface, making the hypothesis that only H_2O_2 production leads to spent fuel oxidation, and all oxidized spent fuel was conservatively considered as being dissolved. Hydrogen effects are described by catalytic interaction with the epsilon phases.	Trummer and Jonsson (2010)
Model 3: of SUBATECH considers water radiolysis and diffusion of radiolytic species with the radiolytic transport code Tramo, considering dose gradients at the fuel surface. The water radiolysis model and electrochemical surface reactions are coupled. The effect of H_2 is described by an effect on the corrosion potential.	Grambow et al. (2011)
Model 4: of CEA is the French operational source term model. It considers primary radiolytic species. It assumes that 50% of the oxidizing species will be able to hit the fuel surface leading instantaneously to oxidation and dissolution of the fuel as U(VI) . It is a mass balance approach without fitting parameters.	Poinssot et al. (2005b)
Model 5: Describes the impact of H_2O_2 and of H_2 from container corrosion (activated on ϵ particles) on spent fuel dissolution, considering diffusion of O_2 , H_2 , H_2O_2 and UO_2^{+2}	Liu et al. (2017)

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The Concept of Geological Disposal of Highly Radioactive Nuclear Waste

Rodney C. Ewing and Sulgiye Park, Geological Sciences, Center for International Security and Cooperation, Stanford University, Stanford, CA, United States

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Acronyms

AEC	U.S. Atomic Energy Commission
Andra	French National Agency for Radioactive Waste Management
DoE	U.S. Department of Energy
EPA	U.S. Environmental Protection Agency
HLW	high-level waste
IAEA	International Atomic Energy Agency
NAS	U.S. National Academy of Sciences
NEA	Nuclear Energy Agency of the Organization for Economic Co-Operation and Development
Nirex	A body created by the UK nuclear industry to examine all aspects of geological disposal of radioactive wastes
NRC	U.S. National Research Council
NRC	U.S. Nuclear Regulatory Commission
NWF	Nuclear Waste Fund
NWPA1982	Nuclear Waste Policy Act of 1982
NWPAA1987	NWPA Amendment 1987
NWTRB	U.S. Nuclear Waste Technical Review Board
PSA	Performance safety assessment
SKB	Swedish Nuclear Waste Management Company
SNF	Spent nuclear fuel
TRU	Transuranic waste
WIPP	Waste Isolation Pilot Plant

Introduction

Since the 1950s, there has been an international effort to develop deep-mined geological repositories for the disposal of highly radioactive nuclear waste. At least 10 countries (Canada, China, Finland, France, Germany, Japan, Sweden, Switzerland, United Kingdom and United States) have major efforts underway that have already lasted for decades. Significantly, as of 2020, no country has opened a geologic repository for the disposal of the spent fuel from nuclear power plants used to generate electricity. Among those making the greatest progress are Finland, Sweden and France. Other countries, such as Canada, Germany, United Kingdom and the United States, have had to delay or “reset” their programs, mainly to address political and public opposition (NWTRB, 2015; Metlay, 2016; Ewing et al., 2018).

The earliest recommendations for the disposal of nuclear waste date from the mid-1950s, when a U.S. National Academy of Sciences (NAS) Committee first presented a report (NAS, 1957), *Disposal of Radioactive Waste on Land*. Prior to this, many fanciful solutions, such as deep space disposal or technological solutions, such as accelerator-drive transmutation, were considered. The basic principles from the NAS (1957) report were affirmed again 20 years later (NAS, 1978). Among many ingenious alternative strategies proposed for disposing of some or all of the nuclear waste inventory, geological disposal remains the only one that appears to offer safe, long-term disposal of highly radioactive waste types: spent nuclear fuel and solidified high-level waste from reprocessing. The first thoughtful scientific analysis of requirements for developing a deep-mined geologic repository was by earth scientists (Bredehoeft et al., 1978). Their prescient paper recognized the importance of different waste types, the challenges of site characterization, the perturbations that the emplaced waste would impose on geologic formations, and the time frames over which geologic isolation would be required.

Nuclear nations in western Europe and elsewhere also began to plan for geological disposal in the late 1970s and 1980s. Most countries have interacted through the International Atomic Energy Agency (IAEA), which has established safety requirements for the disposal of radioactive waste (IAEA, 2011a) and developed guidance on how geological disposal facilities should be built (IAEA, 2011b).

The United States was the first country to embark on this journey, and thus, is illustrative of the challenges all countries have faced in implementing programs of nuclear waste disposal. The earliest concern in the U.S., in the late 1940s, was for the fate of the high-level waste from reprocessing fuel from plutonium production reactors. The wastes were part of the Manhattan Project at the Hanford and Savannah River sites in the states of Washington and South Carolina, respectively. A committee of the U.S. NAS recommended disposal in a deep mined geological repository, specifically in salt formations (NAS, 1957). From 1963 to 1972, the Atomic Energy Commission (AEC) conducted studies of a salt dome near Lyons, Kansas, but finally, had to abandon the effort for both technical reasons and political opposition. A three-year effort by an interagency review group culminated in the Nuclear Waste Policy Act (NWPA) of 1982 (Carter, 1987). The NWPA was a comprehensive covenant that outlined the role of three U.S. agencies: Department of Energy (DoE) as the implementer; Environmental Protection Agency (EPA) to set the standard; and Nuclear Regulatory Commission (NRC) to implement the standard and regulations in order to review the application for construction and disposal submitted by the DoE. States were given a lesser role of “consultation and concurrence.” The NWPA 1982 required the characterization of multiple sites and the development of two repositories, east and west of the Mississippi river, and a method of funding derived from a fee levied on electricity generated by nuclear power plants that went into a Nuclear Waste Fund (NWF). The federal government was to have taken ownership of commercial spent fuel in 1998 for transport to and final disposal in a geological repository.

In 1987, the NWPA was amended such that only Yucca Mountain in Nevada would be carried forward for characterization. Despite strenuous objections by the state of Nevada, the Yucca Mountain project went forward, and a license application was submitted in 2008, a decade after the federal government was to have taken ownership of the commercial spent fuel. In 2009, the Obama Administration attempted to withdraw the license application, describing the Yucca Mountain site as “unworkable.” During the past decade the fate of the Yucca Mountain project has been mainly in the hands of Congress and Federal courts, and work has not continued. A more detailed description of these recent events can be found in Ewing et al. (2018).

Other countries have had their own unique histories, mainly starting in the 1970s. A history of 10 national programs was published by the Nuclear Waste Technical Review Board (NWTRB, 2015). In nearly all cases, however, the development of a geologic repository has been very difficult.

For all nuclear waste programs, there have been four fundamental challenges:

- **Models:** The development and application of credible physical and chemical models of the projected behavior of a repository is challenging. Natural systems are notoriously difficult to model. In evaluating a geologic repository, models necessarily include spatial scales that begin at the atomic-level with the corrosion of spent fuel to scales of tens of kilometers in order to consider the effects of hydrologic and tectonic events, such as earthquakes, volcanism and climate change (Nordstrom, 2012; Konikow and Bredehoeft, 1992).
- **Time:** Typically, the projected behavior of a repository extends to hundreds of thousands of years in order to assure safety to the environment and humans. However, these time scales are on an evolutionary time scale, comparable to the time over which the human species has evolved. Additionally, boundary conditions, such as climate, hydrology and human geographic distribution patterns and habits, which change over time, can have a huge effect on the outcome of model simulations.
- **Exposure:** The evaluation of the impact of a geologic repository on the human population out to hundreds of thousands of years is a major technical challenge. The calculation of dose to a population far into the future requires many assumptions, such as the

health effects of low doses of ionizing radiation, as well as the determination of exposure pathways to populations into the distant future.

- *Public acceptance:* Finally, public acceptance of the results of the technical analysis, already difficult enough, depends on the development of a strategy for public engagement and power-sharing that is not accounted for in most national programs or, indeed, the governance structure of most countries.

Despite these challenges, a number of useful approaches have been developed for the evaluation of the long-term performance of a geologic repository over the past decades. In terms of the models, one may use a range of parameter values that are conservative. Regarding future boundary conditions, a variety of scenarios should be investigated. The long periods of the analysis and the complexity of natural systems also imply that studies of natural systems, so called “natural analogue” studies, and experiments in underground research laboratories are useful in providing confirmatory guides of projected performance. The use of exposure as a metric of compliance can be supplemented by using multiple measures of performance as embodied in the safety case approach. Finally, some countries have developed successful strategies of public engagement that involve joint planning and power-sharing.

Types of radioactive waste

The types of waste intended for geological disposal depend on the type of nuclear fuel cycle adopted in each country, whether the country has a nuclear weapons program, and the way different radioactive materials are classified. Waste types are of three broad categories (Ewing et al., 2016): (i) highly radioactive, heat-producing waste, which mainly includes spent nuclear fuel and high-level waste from reprocessing of spent nuclear fuel; (ii) less radioactive waste materials contaminated by long-lived nuclides, such as plutonium. This is generally termed long-lived intermediate level waste in Europe or transuranic waste in the U.S.; (iii) low-level waste, mainly contaminated by short-lived radionuclides, generated by medical procedures and reactor operations. The first two categories are generally destined for deep, mined geological disposal, whereas the third category is suitable for disposal in shallow, landfills.

For geologic disposal, the important issues for developing a repository are: (i) the radionuclide inventory; (ii) the evolution of this inventory over time; (iii) the geochemical mobility and radiotoxicity of the radionuclides in specific disposal environments; and (iv) the thermal output of the waste. The challenge for geoscientists is to understand how the evolving conditions (e.g., thermal and radiation fields, as well as hydrologic conditions) of a repository affect the geochemistry and mobility of the radionuclide inventory, which is also changing over time. The decay of one radionuclide often becomes the source of another (e.g., ^{239}Pu with a half-life of 24,100 years decays to ^{235}U with a half-life of 700 million years).

Most deep-mined geological repositories are for the disposal of heat-producing spent fuel or high-level waste from reprocessing, but there are important variations within each of these waste types. For spent nuclear fuel, the type of fuel, burn-up and age since removal from the reactor all have an impact on the radionuclide inventory and the thermal output (Hedin, 1997; Ewing, 2015). For high-level radioactive waste, the type of chemical processing and the age of the waste are important. Reprocessing generally lowers the content of long-lived actinides, but it leaves high concentrations of shorter-lived fission products, such as ^{137}Cs and ^{90}Sr . Although these shorter-lived fission products have half-lives of approximately 30 years, they are responsible for the large thermal output, which determines how long the waste must be stored at the surface prior to disposal. There are also very long-lived, problematic radionuclides, such as ^{129}I , ^{99}Tc and ^{79}Se . These complexities mean that countries with a long history of nuclear power plants and nuclear weapons programs have a wider variety of wastes that require different treatments and handling; countries with waste from only commercial nuclear power plants have much less complicated waste streams.

The basic tenets of geological disposal

Multiple barriers

A major tenet of radioactive waste disposal is the concept of multiple barriers (Ewing et al., 2016; Yardley et al., 2016). This “belt and suspenders” approach envisions a series of barriers, each with a capacity to prevent or slow the release of radionuclides. The objective is not to fully contain the waste forever, but rather to delay radionuclides reaching the biosphere so that most will have decayed away, or so that dilution and sorption onto the geologic media during transport have lowered concentrations in groundwater, thus lowering subsequent human exposure to well below regulatory limits. The multi-barrier concept is a precautionary approach in the face of the huge uncertainties that result from projections of physical and chemical processes over large spatial (tens of kilometers) and temporal (hundreds of thousands of years) scales. If one barrier fails to perform as expected, other barriers should provide for a margin of safety. Barriers are of two types: engineered and geologic (Ewing et al., 2016).

Engineered barriers

Engineered barriers include the waste form, the waste package, and surrounding backfill. Most of the engineered barriers are physical in that they delay the access of water to the waste package (e.g., the drip shields proposed at Yucca Mountain) or the release of

radionuclides from a breached canister (e.g., the bentonite buffer in the KBS-3 concept for granite repositories (see [Hedin and Olsson, 2016](#))). In some cases, the engineered barrier has a chemical function, affecting the geochemical environment around the waste. For example, MgO is emplaced with the transuranic waste in the bedded salt repository at the Waste Isolation Pilot Plant (WIPP) in New Mexico, and a high pH cement was developed in the United Kingdom to encapsulate intermediate level waste. In both cases, the intention is to remove CO₂ and provide an environment in which the solubility of actinides remains low.

Waste Form: Although the most common waste forms for highly radioactive materials are spent nuclear fuel and vitrified high level waste, there are many other types of waste forms ([Lutze and Ewing, 1988](#); [Weber and Ewing, 2013](#)). A summary of the characteristics and recent developments in waste form research can be found in [Ewing \(2006\)](#). The properties of the waste form can be of critical importance, as the waste form initially contains *all of the radioactivity*. To the extent that the waste form is durable, and the release of radionuclides is low, the remainder of the safety analysis is less challenging. One approach is to tailor waste form properties to match either the composition of the waste stream or the geologic conditions of the waste form in the repository. As an example, the UO₂ in spent nuclear fuel is unstable under oxidizing conditions, but much more stable under reducing conditions. Also, the corrosion rate of the borosilicate glass is slower when the silica concentration in the groundwater is higher, although other chemical components can lead to accelerated corrosion. The properties of the waste form also change as the temperature and radiation field evolve with time ([Ewing, 2015](#)). Radiation damage mainly from the alpha-decay of actinides can lead to radiation-induced phase transformation from the crystalline to amorphous state ([Ewing and Weber, 2010](#)).

Waste Package: The waste form is typically placed in a metal canister, which may consist of steel, copper or more advanced corrosion resistant alloys. In many countries, the waste producers (i.e., utilities) are required to package waste in standard packages that the repository is designed to accommodate. Some countries intend to follow the Swedish KBS-3 concept, in which spent fuel is placed in iron insets that are then placed in copper containers. These are then inserted into holes drilled into crystalline rock and overpacked with bentonite clay. Intermediate level wastes are commonly encapsulated in cement in stainless steel or iron canisters, and a diverse range of cements have been proposed.

Back-fill or overpack: In order to reduce water movement through a repository after closure, the remaining void space must be filled with a material compatible with the engineered and geological barriers. Bentonite backfill for heat-producing waste canisters has been investigated extensively, but for larger waste volumes, host rock materials and cement are likely to be important. Repositories in salt will employ crushed salt, which will gradually be compacted as the excavated rooms close. Uniquely, the proposed repository at Yucca Mountain does not utilize a backfill ([Swift and Bonano, 2016](#)).

Geologic barriers

Geologic barriers rely on the properties of the rock and hydrologic system around the repository and are intended to extend the travel time required for radionuclides to reach the biosphere. The movement of fluid through the rock is controlled by its hydraulic conductivity, which depends on its matrix permeability and the presence or absence of fracture systems. Irrespective of the movement of groundwater, the mobility of the radionuclides in solution may be further retarded by precipitation of radionuclide-bearing phases, sorption onto mineral surfaces, and matrix diffusion (a process by which radionuclides are “trapped” in the micro-sized void spaces in the rock). In other cases, the movement of radionuclides can be accelerated by sorption onto colloids and advective transport or due to the formation of negatively charged chemical species: such is the case for ¹²⁹I, that are repelled, rather than sorbed, from mineral surfaces. Mineral surfaces have a negative charge; hence there is no sorption of negatively charged species in solution.

All proposed repository sites to date are located below the water table, with the exception of the Yucca Mountain site in Nevada, where the proposed repository is in the unsaturated zone ([Swift and Bonano, 2016](#)). The apparent “dryness” of the unsaturated zone is complicated by a high infiltration rate, large volumes of water held in the porosity, and rapid flow along fractures. For disposal below the water table, the host rock must have a low permeability to ensure that radionuclides released from containers over time do not reach the biosphere. Three types of host rock are widely recognized as having the potential to provide a sufficiently low permeability to meet this safety requirement: strong rocks with very low porosity and very few fractures (including igneous, metamorphic and even some sedimentary rocks, see for example, [Hedin and Olsson \(2016\)](#)); relatively weak mudrocks or clays, which will not sustain open fractures for extended periods of time ([Grambow, 2016](#)); and rock salt, which is also self-sealing ([von Berlepsch and Haverkamp, 2016](#)). The rock must also have the necessary physical properties for the construction of the repository. In some cases, as in granite, high mechanical strength favors construction, but the presence of fracture systems may increase the hydraulic conductivity. In other cases, such as salt, a medium level of mechanical strength may be advantageous as plastic deformation and flow of the salt seal the rock that surrounds the waste packages.

Thus, the petrophysical, geochemical and hydrologic properties of the geologic formation have an important impact on the strategy that is adopted for the complementary engineered barriers. Also, there are important interactions among the engineered and geologic barriers. As an example, the geochemistry of the groundwater at the repository horizon may buffer its geochemical properties, which in turn, affect the rates of canister and waste form corrosion.

The physics and chemistry of the safety assessment is essentially an evaluation of the effectiveness of the barrier systems over time, complicated by the coupled and evolving thermal, radiation, hydrologic and geochemical conditions of the repository. The strategy for handling the heat of the spent nuclear fuel and high-level waste will also dictate the disposal strategy and the choice of barriers. Aged-waste in smaller packages can reduce the thermal pulse in the repository, which is important if a clay backfill is used. On the other hand, larger packages and a larger thermal pulse might be used to drive water away from the surface of the waste package. Very high heat, however, can affect the condition of fuel cladding, a thin metal alloy wrapped around the fuel pins, which

in some concepts is considered as a barrier. Additionally, the safety assessment must consider regional aspects of the geology, such as seismicity and volcanism and even climate change and glaciation. The presence of natural resources in a region must also be considered, as the exploration for and extraction of resources can compromise a geologic repository.

Geological repositories

From the scientific perspective, considerable amount of basic research and site characterization have been completed in a variety of rock types—clay, salt, crystalline rock, and volcanic tuff. For each rock type, the scientific and engineering strategy is different—driven by the properties of the different geological formations, the types of waste to be disposed of, and the regulatory requirements of each country. Indeed, there is now a huge literature on all aspects of nuclear waste management and disposal ([Ahn and Apted, 2010](#); [Lee et al., 2013](#)).

Salt

The concept of disposal in bedded salt or salt domes dates back to the 1957 study and recommendations of the U.S. NAS. The earliest effort was at a salt dome near Lyons, Kansas, but this failed due to technical issues (e.g., an incomplete understanding of the previous drilling and mining activity) and political opposition.

Disposal of low and intermediate nuclear waste in Germany has favored salt domes ([von Berlepsch and Haverkamp, 2016](#)). Nuclear waste has been disposed of at the Endlager für radioactive Abfälle Morsleben (ERAM) site, a previous salt mine, in the former East Germany, but disposal of waste was discontinued in 1998. In addition, some 125,000 drums of low and intermediate level waste were disposed of in 13 chambers at depths of approximately 750 m in what had previously been an operating salt mine at Asse II in lower Saxony in the former West Germany. Disposal and research ended in 1995, due to the ingress of water into the mined cavities. Whether to retrieve the radioactive waste or leave it in place remains a major political and difficult technical issue. Another site in lower Saxony, Gorleben, has been the subject of research and development from 1979 to 2000. A moratorium on work at Gorleben was declared in 2000.

The U.S. has successfully developed a deep-mined geologic repository for transuranic defense waste at the Waste Isolation Pilot Plant (WIPP) in southeastern New Mexico. The mine was licensed for the disposal of nuclear waste for a compliance period of 10,000 years and began receiving transuranic waste on March 26, 2019. To date, some 12,700 shipments of waste have been received at WIPP from 22 transuranic sites in the U.S. The repository is in a 910 m thick layer of bedded salt, and the disposal horizon is at 660 m below the surface. In February of 2014, there was a minor release of radioactivity from a breached waste package due to an exothermic reaction of waste package contents. The repository was closed for 3 years. Operations at WIPP resumed on January 9, 2017. In 2020, WIPP is the only operating deep-mined geologic repository in the world.

The advantages of disposal in salt are three-fold: (i) Salt is self-healing via a process of creep, which closes open fractures and porosity, leading to extremely low permeabilities. Waste packages become sealed in the salt, although due to lithostatic pressure, such as at WIPP, the drums of waste will be crushed and breached. (ii) Salt domes and bedded salt deposits have low concentrations of water. The age of many salt deposits, that is hundreds of millions of years, speaks to their stability and the absence of freshwater. (iii) Salt has a high thermal conductivity as compared to typical silicate-based rock formations. For HLW with heat-producing radionuclides, the heat is more quickly dissipated away from the waste package.

The disadvantages of salt include: (i) Salt is soluble in fresh groundwater. (ii) Thermal pulses induce fluid inclusions to move toward and concentrate around the waste package. (iii) Brines are exceptionally corrosive of metal waste packages. (iv) Hydrogen gas evolves from the corrosion of metal waste packages, leading to a buildup of gas pressure. (v) Bedded salt formations are not entirely homogeneous and often contain thin layers of clay or other mineral salts that can open and provide access to water or release of radionuclides. (vi) Salt deposits are often in areas of important mineral resources, such as potash, oil and natural gas. Inadvertent intrusion by drilling programs for resources may penetrate the repository horizon. Thus, the main barrier to release of radionuclides depends on estimates of the drilling rates in the area at distant times and an understanding of the chemistry of brine solutions, particularly the solubility of radionuclides in brines.

The chief barriers to radionuclide release in salt formations are the absence of water, although brines are commonly present in bedded salt deposits, and the physical sealing of the waste by the salt that surrounds the waste. Waste packages and waste forms are not considered barriers to release. The chemistry of the salt plays only a limited role in radionuclide retention. At the WIPP site in New Mexico, thousands of pounds of MgO have been added to the disposal galleries in order to moderate the chemistry of the brine and prevent the formation of CO₂ created by microbial activity in order to reduce the solubility of plutonium in the brine ([Tracy et al., 2016](#)).

Clay

A number of countries pursue disposal in clay, claystone or shale. Clays are very fine-grained, silicate minerals with a sheet structure that has the ability to expand and absorb water or radionuclides. The expansion of a clay leads to a substantial reduction in porosity and permeability. At the present, three countries, Belgium, France and Switzerland, have active site investigations in clay formations ([NWTRB, 2016](#)).

The main advantages of clay include ([Grambow, 2008, 2016](#); [NWTRB, 2016](#)): (i) Expanding clays, such as bentonite, are effective seals to the ingress of water. (ii) The very small grain size (millimeters to micrometers) provides substantial surface area onto which radionuclides may be sorbed. (iii) The corrosion of canister materials, such as iron, will produce a hydrogen gas that slows down the movement

of water toward the canister. (iv) Ground water in clay formations is saturated in silica, which reduces the corrosion rate and radionuclide release from vitrified waste. (v) In case radionuclides are released, some actinides and ^{99}Tc will be sorbed onto the iron corrosion products of the canister or the clay surfaces of the surrounding backfill and rock. Additionally, the presence of organic carbon in shales will maintain reducing conditions that curtail the mobility of actinides and other redox sensitive radionuclides. The release of more soluble radionuclides, such as ^{139}I will depend on the rate of corrosion of the waste form, although the complete dissolution of vitrified “logs” in breached canisters is expected to take many hundreds of thousands of years (Grambow, 2016). Even for radionuclides that do not readily sorb onto clay mineral surfaces, the mobility of radionuclides in a clay formation is diffusion-controlled, a much slower process than transport by moving ground waters, i.e., advection. (vi) Although clay formations can be faulted or experience fracturing, clay formations typically deform plastically, filling open spaces; thus, clay formations generally retain their low permeabilities.

Potential disadvantages of clay include: (i) Clay formations are heterogeneous with a variety of types of clay minerals whose physical and chemical properties vary. (ii) Non-sorbing radionuclides, such as ^{129}I , ^{36}Cl and ^{79}Se can diffuse through the clay formation until reaching other formations with faster groundwater transport. (iii) As with salt, shale formations, particularly organics shales, are often related to the formation of hydrocarbon deposits; hence, there is the possibility of inadvertent human intrusion due to the search for natural resources.

The chief barriers to radionuclide release in clay formation are both physical and chemical. The physical barriers are related to the low porosity and permeability that reduce the mobility of radionuclides. The chemical barriers involve the reduced corrosion rate of glass in silica-saturated groundwater, redox conditions that reduce the solubility of key radionuclides, and chemisorption of radionuclides onto mineral surfaces.

Crystalline rock

Crystalline rock refers to both igneous and metamorphic formations that consist of tightly interlocked grains of minerals that have formed by crystallization from a melt or solid-state reactions at elevated pressures and temperatures. For the purpose of the disposal of nuclear waste, granites, gneisses and basalt formations have been considered in Canada, China, Japan, United Kingdom, Russia and the United States—but the concept has been developed in its most advanced form in Sweden and Finland. The basic KBS-3 concept in Sweden consists of a series of barriers around the spent fuel: metal, zircaloy cladding that surrounds the UO_2 fuel pellets, a copper canister with a ductile iron insert that holds individual fuel assemblies, a bentonite clay backfill around the copper canister, and emplacement in crystalline rock (Fig. 1). The concept has been modified in other countries, such as in Russia, where an iron alloy is envisioned for the waste package (Laverov et al., 2016).

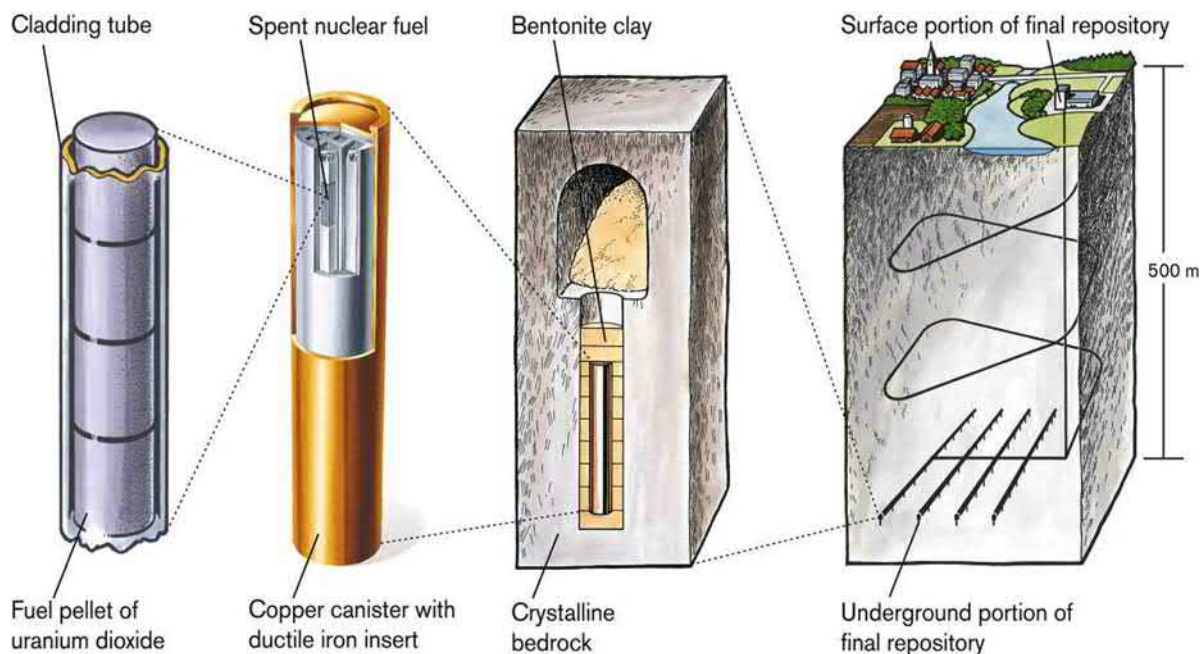


Fig. 1 The Swedish KBS-3 concept for the disposal of spent nuclear fuel (SKB, 2006). The spent fuel assembly is placed in iron inserts that are then placed into copper canisters and the whole package is then surrounded by bentonite clay and deposited at approximately 500 m depth in groundwater-saturated granitic rock. The system of multiple barriers include: (i) low corrosion rate of spent fuel under reducing conditions; (ii) low corrosion rate of the copper canister under anoxic conditions; (iii) the swelling bentonite clay slows the ingress of water to the copper canister and reduces the mobility of any released radionuclides due to the very low permeability of compressed bentonite; (iv) the slow flow of groundwater and sorption of radionuclides onto mineral surfaces delays entry to the biosphere. SKB (2006) *Long-Term Safety for KBS-3 Repositories at Forsmark and Laxemar—A First Evaluation*, Report No. TR-06-09, Swedish Nuclear Fuel and Waste Management Company: Stockholm, Sweden, pp. 27, 613 pages.

The advantages of the KBS-3 strategy in crystalline rock include (Hedin and Olsson, 2016): (i) Because of the high strength and rigidity of crystalline rock, tunnels and rooms remain open, resisting deformation due to glacial epochs or earthquakes. (ii) The disposal level is below the water table, which helps mitigate the effects of climate change on geohydrological conditions. (iii) The redox conditions are generally reducing due to the presence of sulfides in the crystalline rock, hence lowering the solubility and mobility of some key radionuclides, such as the actinides. (iv) Sites can be located where the chance of exploration for economic mineral or energy deposits is low. (v) The waste packages are emplaced at relatively low temperatures, no higher than 100 °C, which maintains the stability of other barriers, such as the bentonite. (vi) The use of a backfill, such as a 35 cm thick mixture of mainly bentonite, can delay the ingress of water to the canister and reduce the mobility of radionuclides that might at some distant time be released from a breached canister. (vii) The engineered barriers are made of natural materials, such as copper, thus providing evidence of their stability over geologic periods.

The principal challenges of the KBS-3 concept include: (i) The need to characterize the flow of groundwater through fractures in the crystalline rock. (ii) The possibility that the bentonite backfill or buffer might be “washed-out” due to flowing groundwater. (iii) The potential for copper canister corrosion, a subject of some recent controversy (King et al., 2010; Taxén and Bergqvist, 2019).

The safety strategy of the KBS-3 concept relies on an evaluation of the compatibility of all of the materials in the various barriers systems and their passive performance in the absence of human intervention. The primary barrier to release of radionuclides is the copper canister, which is expected to have a lifetime of many hundreds of thousands of years. Great care has been taken to preserve the integrity of the copper canisters, such as by using two techniques, electron beam welding and friction stir welding of the lids to the canister body in order to mitigate the possibility of strain-induced corrosion that results from normal welding techniques. The addition of backfill is meant to slow the ingress of water to the canister surface, and the low temperature at the surface of the canister reduces the rates of chemical reaction. The secondary barrier, in the event the first barrier is breached, is the crystalline rock itself and its interaction with any exposed fuel elements. This is not expected to occur until far into the future when the fuel is much cooler and most of the radioactivity has decreased substantially (Hedin, 1997). That said, the challenge of understanding the chemical interactions between the UO₂ in the fuel, the corroded canister, and the surrounding rock is substantial. A number of the relevant processes for the release of uranium are illustrated in Fig. 2.

Volcanic tuff

Volcanic tuff consists of fragments of material and ash ejected during a volcanic eruption. During compaction and cooling from a high temperature, a harder “welded” tuff can form. Tuff typically has a high porosity created by escaping gases during volcanic eruptions. Tuff can be fractured due to cooling and contraction and has a medium structural strength, but still is able to support the construction of tunnels.

The U.S. is unique in its effort to construct a repository in the volcanic tuff at Yucca Mountain, some 145 km northwest of Las Vegas, Nevada. There are a number of unique aspects to the U.S. approach: (i) The repository is “hot”, that is the surface of the waste packages is above the boiling point of water as part of a strategy for delaying the contact of water with the surface of the waste package. (ii) The geochemical conditions are oxidizing. (iii) The disposal horizon is some 200 m above the water table. (iv) There is no backfill, but rather the tunnel space is left open for ventilation and the dissipation of decay heat from the relatively large waste packages; waste packages will be protected from water by a titanium drip-shield.

The advantages of volcanic tuff at Yucca Mountain (Swift and Bonano, 2016) include: (i) Yucca Mountain is located in an arid region and approximately 200 m above the water table in the unsaturated zone. The dry climate and closed hydrologic basins are believed to limit transport of radionuclides. (ii) There are two very important engineered barriers: a corrosion resistant nickel-based alloy, Alloy-22, that is highly resistant to general and localized corrosion in an oxidizing environment; a 1.5 cm thick titanium drip-shield that will be placed over the metal waste packages in order to divert and delay any intermittent seepage from percolating groundwater. (iv) The very limited and distant population reduces the possibility of radiation exposure.

The disadvantages of this strategy (Ewing and Macfarlane, 2002; Macfarlane and Ewing, 2006) include: (i) Yucca Mountain tuff is not dry. The very high porosity, 10–40 vol%, retains water, much as a sponge does. Additionally, rapid water movement along fast paths of transport, fractures, has been identified by the detection of “bomb pulse” ³⁶Cl at the repository horizon. The rapid transport of ³⁶Cl occurred over less than a half-century. (ii) Oxidizing conditions lead to the relatively rapid breakdown and alteration of UO₂ to uranium oxyhydroxides, some of which have low solubility constants or lead to the formation of colloids that can transport radionuclides (Novikov et al., 2006). (iv) A hot repository causes the movement of water around the disposal tunnels and can accelerate the corrosion of the waste packages due to the presence of deliquescence-induced corrosion of the waste package. (v) Yucca Mountain region is located in the Basin and Range province of the western U.S., typified by volcanic and seismic activity.

Conceptually, at Yucca Mountain, the main barrier to the release of radionuclides is the absence of water in this arid region. The secondary barrier consists of the engineered barriers, waste package and the drip shield. Finally, the very long distance, 20 km, of transport slows the time and amount of radioactivity that can reach the nearest inhabited area, Amargosa Valley, a small farming community of less than 2000 people. This strategy means that the entire transport distance is part of the geology barrier, and the principal means of reducing the total radioactivity are radioactive decay, sorption onto mineral surfaces and dilution.

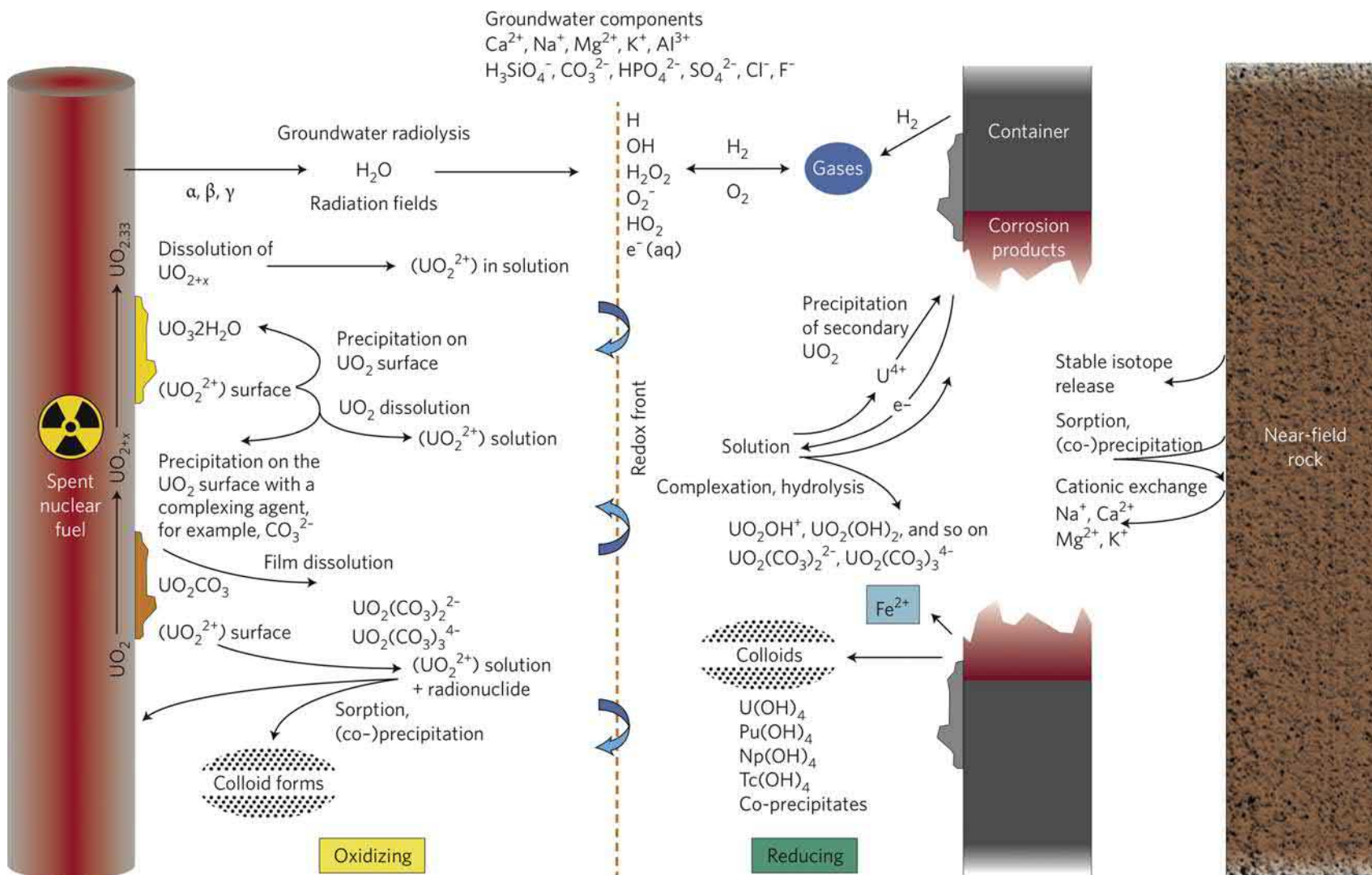


Fig. 2 Chemical processes that may affect the alteration of spent fuel in contact with groundwater. Schematic representation of spent fuel alteration mechanisms and mechanisms controlling radionuclide concentration in solution in the near-field environment of the fuel, including groundwater, corroded waste package and backfill (if present), as well as the surrounding rock (Ewing, 2015). This schematic deals mainly with the fate of uranium from the SNF, but a similar understanding has to be developed for the release of each radionuclide that is released due to the bulk dissolution of the SNF. Each radionuclide will have its unique chemistry and fate over the long-term corrosion process. Ewing RC (2015) Long-term storage of spent nuclear fuel. *Nature Materials* 14: 252–257.

Deep borehole

The concept of deep borehole disposal involves, as the name indicates, the emplacement of waste down a borehole, usually < 50 cm in diameter, to depths as great as 3–5 km. The fundamental strategy is based on the concept that “deeper is better” in that there is geologic evidence for the long-term isolation of very deep groundwater, usually brines, from the surface. The concept requires remote emplacement and the absence of backfill or thick-walled waste packages, due to the need for the waste package to fit into the small diameter hole. There are a number of variations on this strategy, in some cases the emplaced waste packages are stacked in a vertical hole or emplaced horizontally by directional drilling. The temperature of the waste package may be low or high, depending on the age of the fuel and the geologic formations used for emplacement. Critical to the success of the approach is the need to guarantee that the borehole can be sealed over the course of hundreds of thousands of years. One very novel approach to sealing the hole has been the proposal to use the decay-heat of the spent fuel to melt the surrounding silicate rock, such that, once cooled, the waste would be entirely encapsulated in solid rock (Gibb and Travis, 2015).

The earliest discussions for the use of deep boreholes for the disposal of radioactive waste date from the 1950s (NAS, 1957) as a possible solution for the disposal of the HLW created by fuel reprocessing for the production of plutonium at the Hanford site in Washington. Studies in Sweden (SKB, 1992, 2010) and the United Kingdom (NIREX, 2004) reviewed borehole disposal in detail as part of a consideration of alternatives to the disposal of spent fuel in a deep-mined geological repository. The concept has been periodically revived in various forms (Gibb et al., 2012; Brady et al., 2012; Bates et al., 2014; Bracke et al., 2016; Cornwall, 2015; Krall et al., 2020). Two companies, Deep Isolation (<https://www.deepisolation.com>) and NuclearSAFE (<https://nstusa.net>) have been the most recent proponents of deep borehole disposal, both proposing the use of horizontal emplacement of canisters containing spent fuel.

In addition to the great depth of the disposal horizon and depending on the strategy adopted, there are other barriers to limit the ability of the radionuclides to reach the surface from a deep borehole. These include: (i) the density stratification of deep fluids, usually brines, should prevent advective, upward flow; (ii) at depth, permeability of the rock decreases due to the lithostatic pressure; and (iii) geochemical conditions are reducing.

The challenges to deep borehole disposal include: (i) Remote emplacement and retrievability in the event of an accident, such as the breach of a canister; (ii) Even very deeply seated rock units can display considerable heterogeneity in physical, chemical and petrologic properties. The required level of characterization for a safety assessment may not be possible at such great depths; (iii) The approach emphasizes the geologic barriers at the expense of the use of engineered barriers. A truly multi-barrier system may require chemical processing to incorporate the waste into a durable waste form or the use of waste packages that can withstand the pressure, temperature and corrosive effects of brine at depth; (iv) The plug in the borehole must not fail, as it is the main pathway for fluids, which are under pressure, to move upward. There are also important operational concerns with handling, emplacement, inspection and retrieval during the emplacement operation. A detailed discussion of these issues can be found in Krall et al. (2020).

In the final analysis, deep borehole disposal may require the same level of scientific site characterization, engineering innovation and performance analysis as a deep-mined repository. That said, deep borehole disposal may be useful for disposal of unique waste streams, such as radioactive sources or fissile material from dismantled nuclear weapons.

Multi-barrier disposal systems

Based on the brief summaries of the different disposal strategies in different geologic regimes associated with different rock types, one can recognize that there are very different approaches to the barrier systems for different types of geologic repositories. In some cases, the geologic barrier is preeminent, and in others, the waste package is the primary barrier to release. All of these systems have multiple barriers, but the evaluation of safety will depend on the robustness and redundancy of the barriers over appropriate time scales.

Retrievability and reversibility

In response to public concern and skepticism about the strategy of permanent geologic disposal, two new concepts emerged in France in 1991 as part of the French Radioactive Waste Act and were analyzed as part of the Andra report, *Dossier 2005: reversibility and retrievability* (Andra, 2005). The NEA (2001, 2010) has reviewed the factors involved in the application of these concepts.

Reversibility refers to the ability to reverse any one of a series of steps in the development of a geologic repository. This requires an iterative, stepwise review during the process of characterization and construction of a geologic repository. At any specific step, new knowledge, such as that developed during characterization of the disposal horizon, may require changes in the strategy envisioned for the role of the engineered barriers or require more detailed investigation of the geologic barriers. An example of the latter is in the early history of WIPP project in the U.S., for which the site location was shifted in order to avoid disturbed rock units that were revealed by the early drilling program.

Retrievability is a special case of reversibility as it requires the removal of already emplaced nuclear waste. This may be the result of new knowledge or the discovery that waste packages do not meet regulatory requirements. Over the longer term, new policies and standards may justify the need to retrieve the waste: (i) the waste may become a valued resource; (ii) standards of safety may change

and become stricter; (iii) better technical options may be developed; (iv) some waste may be reclaimed to confirm behavior during disposal in order to confirm the models used in the safety assessment.

There is a tension between the concept of permanent geological disposal in which the passive operation of the engineered and geological barriers contains the waste and delays the release and the time of arrival of radionuclides to the biosphere. In contrast, reversibility and retrievability require active engineering interventions at the disposal site. These interventions may disrupt the efficacy of certain barriers and must be considered in the design of the barrier systems (e.g., ensuring that barrier systems and, if necessary, the waste packages can be removed as needed). This judgement will be challenging, as one must weigh the exposure and environmental effects of retrieving waste in the near term against exposure and environmental effects in the long term. Generally, it is understood that with the passage of time, the role of the passive barriers increases, while the possibility of reversibility or retrievability decrease.

Site selection

As illustrated in the description of different strategies in different rock types, the multi-barrier approach is adopted in different ways. In some instances, the primary barrier is the geology, such as in a clay repository, and in others, the primary barriers are the engineered barriers, such as highly corrosion-resistant waste packages that utilize copper or the nickel-based Alloy-22. In all cases, the strategies involve multiple barriers, but the role of engineered versus natural barriers has an impact on the approach to site selection.

Since site selection is one of the most controversial steps in developing a geologic repository, it is important to understand how criteria are developed and applied in this process. From the technical perspective, there are three types of criteria (NWTRB, 2015):

Exclusion Criteria: These criteria can be used to eliminate areas based on broadly applicable concepts (NAS, 1957). Resource-rich areas might be excluded due to the possibility of future exploration for and extraction of these resources, such as the exploration for oil and gas. Regions of geologically recent tectonic activity, such as earthquakes or volcanism, might be excluded in preference for older and more stable geologic terranes.

Host-Rock Specific Criteria: In countries where the types of different host rock types are limited, e.g., granitic gneisses of Scandinavia, very specific technical criteria can be developed. These criteria are highly quantitative. The geometry of a rock unit might be specified, such as the thickness of a geologic formation or the depth of disposal. One would expect to see specifications on the groundwater flow (very low hydraulic conductivity), groundwater chemistry (reducing conditions preferred over oxidizing conditions), and strong sorptive capacity (an abundance of sorbing minerals, such as clays, in the host rock).

Generic Criteria: In countries that can choose among a variety of geologic settings, rock-specific criteria become problematic; hence, generic criteria are preferred. Although difficult to quantify, statements of desired properties include, but are not limited to: (i) low hydraulic gradient and conductivity, (ii) thermal stability or high thermal conductivity; and (iii) high mechanical strength. Generic criteria can become exclusionary depending on how they are applied. As an example, a requirement of high thermal conductivity would favor salt over crystalline rock, or a requirement of high mechanical strength would favor crystalline rock over clay.

Equally important to the technical criteria are the social criteria. The U.S. NWTRB (2015) completed a historical analysis of 24 efforts to site a geologic repository in 10 countries. Only six of these programs remain on track. Nearly all of the programs reached a point of failure that required the programs to restart, e.g., Canada, with greater emphasis on the social aspects of siting a geologic repository. The profoundly important conclusion of this study is that successful siting requires not only a robust technical basis but also requires careful engagement and planning with local communities, indigenous peoples and states (Fig. 3).

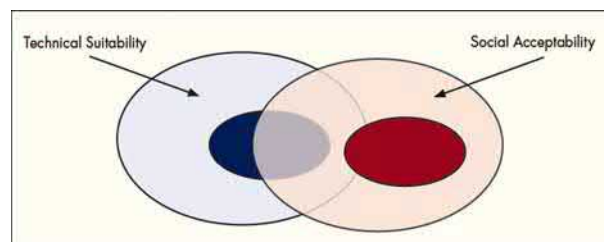


Fig. 3 Selecting a site is an iterative process. There must be a schedule of successive evaluations of technical suitability and social acceptability defined prior to the process of selecting a site. Both activities must be coordinated with clearly defined decision points. Lighter shades denote early-stage judgments, and the darker shades denote later-stage judgments. Clearly, with the passage of time technical suitability and social acceptability must overlap in order to successfully site a repository—or they may diverge (NWTRB, 2015). Nuclear Waste Technical Review Board (2015)

Designing a Process for Selecting a Site for a Deep-Mined, Geologic Repository for High-Level Radioactive Waste and Spent Nuclear Fuel: Overview and Summary, pp. 51, Detailed Analysis, pages 228.

Safety

The safe disposal of radioactive waste in a geological repository depends on three physical and chemical processes:

- *Containment within the near-field* of the repository until radioactive decay has eliminated most of the inventory of radionuclides. Containment is accomplished mainly by engineered barriers, such as a durable waste form, a corrosion-resistant waste package or geochemical and hydrologic conditions that lead to the precipitation of radionuclide-bearing phases or their sorption onto corrosion products of the waste form and waste package.
- *Reduced concentrations of radionuclides in groundwater or on colloids.* This is facilitated by reducing geochemical conditions, sorption onto mineral surfaces along the transport path and dilution by groundwater.
- *Reduced mobility of radionuclides once released from the near-field to far-field geologic formations* such that radionuclide arrival at the biosphere takes a long time. Generally, this is insured by long-travel times for dissolved radionuclides and colloids in the groundwater.

These physical and chemical processes are evaluated in terms of an actual dose to a person or population within or at a specific time and location. This requires a knowledge of future population distributions and human activities, such as farming and irrigation practices with contaminated groundwater, near the area of the geologic repository over very extended periods. Hence, the key elements of regulations, discussed below, are an individual or group dose limit at a specific time and location into the far future.

Performance assessment models

The challenge in determining the safety of a geologic repository relies mainly on the projected results of models of the complex natural and engineered systems that are connected to biosphere models. These models cover large spatial and temporal scales from atomic-scale processes (e.g., corrosion of spent fuel) to global scale processes (e.g., climate change) that evaluate the physical and chemical processes that describe the release and transport of radionuclides to the biosphere. These physiochemical models are then coupled to biosphere models that evaluate exposure pathways for each of the radionuclides. Each radionuclide has a different chemistry, half-life, and radiotoxicity. This modeled behavior is projected over hundreds of thousands of years.

In the broadest sense, these models require: (i) a good knowledge of the source term, that is the radionuclide inventory and the amounts and speciation of the radionuclides. This inventory will change with time due to radioactive decay and the ingrowth of daughter product elements (e.g., the decay of short-lived ^{241}Am to long-lived ^{237}Np); (ii) the behavior of engineered materials in the repository environment; (iii) all relevant geophysical, geochemical and hydrological properties of the repository; (iv) the impact of radionuclide transport through geologic media until they enter the biosphere; and finally, (v) the exposure pathways to individuals or groups of people. This daunting task is made all the more challenging by the fact that over hundreds of thousands of years, the boundary conditions change. The composition and thermal output of the waste change due to radioactive decay. The hydrologic conditions may vary due to climate change or glaciation. The number and distribution of people and communities will be vastly different over time in the vicinity of the geologic repository. As an example of the complexity of such an approach, Grambow et al. (2014) and Grambow and Bretesché (2014) have illustrated the demands of going from long-term (years) experimental data to their use in a safety analysis applied to a geologic repository in clay over many hundreds of thousands of years.

Different countries have approached this challenge with different methods and strategies. In the United States, a total system, probabilistic analysis that was derived from methods to evaluate nuclear reactor safety is used (Ewing et al., 1999). The structure of the analysis is straightforward: (i) Identify numerical measures for the performance of the system. These metrics are usually a calculation of dose to a person or population over time at a specific location. This numerical measure is compared directly from the relevant national standards and regulations. (ii) Characterize the system. This includes all elements of the repository system, including waste form performance, radiochemical properties, geochemical conditions and the hydrology, to name a few; (iii) Create mathematical models of the system whose dependent variables are the performance measures for that system; (iv) Quantify uncertainties in the model relationships, and model parameters using probability distributions; (v) Propagate the uncertainty through the total system model in order to predict the uncertainty in performance measures; (vi) Compare the predicted performance measures to regulatory requirements. Although the process of completing a performance systems analysis (PSA) is straightforward in its description, in practice each step is a challenge. The PSA for the performance of the proposed repository at Yucca Mountain had thousands of input parameters, some represented by probability distributions or “look up” tables, and hundreds of subsystem models, many of which are coupled one to the other (Fig. 4). Perhaps, the greatest value to such an approach is that it helps the analyst understand how the different components of the repository strategy work. In the face of unknown futures, the analyst uses a variety of future scenarios to set the boundary conditions for the models in the analysis. In order to be “conservative,” bounding values of input parameters may be used or the scenarios may include the analysis of the consequences of extreme events such as volcanic eruptions, earthquakes or extreme changes in the climate.

Although such an approach has the appearance of being transparent and quantitative, there are many issues that can be raised when the complexity of the system is great and the uncertainty in the future boundary conditions and calculated performance measures are high. For the public and even for scientific reviewers, the total system analysis is opaque. Few will have the skills or time to examine all aspects of the analysis. Also, in such a complex analysis there are inevitable errors either in input parameters (e.g., solubility limits for radionuclides) or conceptual models (e.g., the relation between climate change and the infiltration rate of

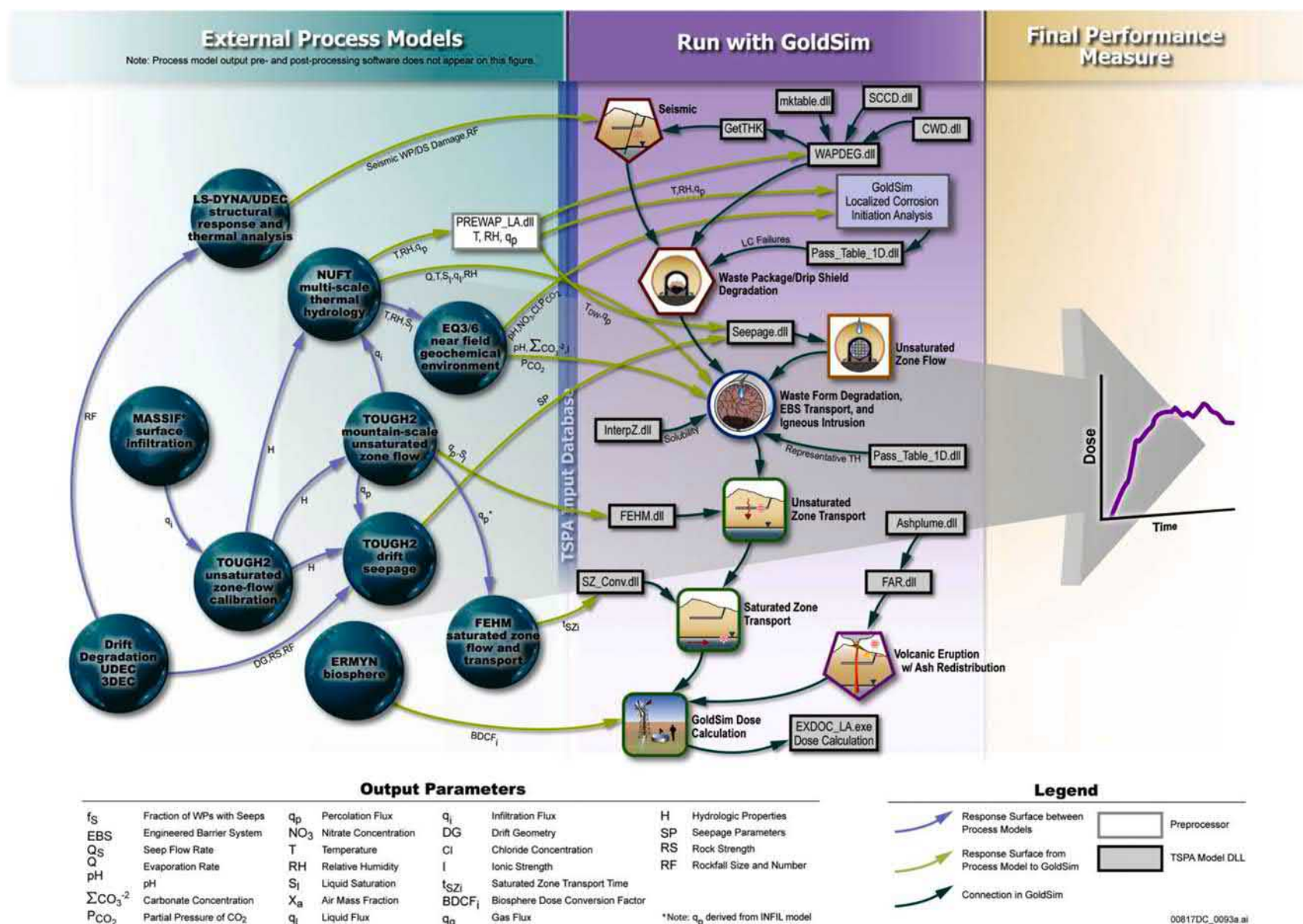


Fig. 4 Schematic of model linkages used for the total system performance assessment of the proposed geologic repository at Yucca Mountain, Nevada. Figure from Swift PN (2018) Deep Geologic Disposal of Radioactive Waste at the Waste Isolation Pilot Plant and the Proposed Yucca Mountain Repository. In: *Presentation at Stanford University, After TRW Environmental Safety Systems (2000) Total system performance assessment for the site recommendation, TDR-WIS-PA-000001, Prepared for U.S. DOE, Las Vegas, Nevada, vol. 214, pp. 225.*

water). Research programs can correct errors with new data and understanding, but in the final analysis one must expect that there will be a gap between the modeled, projected behavior and the future, actual behavior of a repository.

There is also the important issue of the proper use and application of models (Oreskes et al., 1994; Saltelli et al., 2020) and a continuing effort to develop methods that can use expert opinion as part of the model construction (Kerr, 2020). The fundamental issue of whether quantitative model results can or should be used as the only criteria to determine compliance of a repository system remains a subject of debate (Blandford et al., 2011).

Safety case

Most European countries have taken a different approach from a fully quantitative analysis. They have developed the concept of a *safety case*. The phrase “safety case” is used in many different ways. In the most general sense, a safety case is (NEA, 2004).

“The synthesis of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe after closure and beyond the time when active control of the facility can be relied on.”

A safety case is a blend of evidence that “quantif[ies] and substantiates” the argument for safety and draws on a wider variety of data. Quantitative analysis is certainly an important part of a safety case, but safety is also substantiated by other types of evidence such as observations of natural systems that demonstrate the behavior of key radionuclides in a natural and complex geologic settings (e.g., studies of the behavior or radionuclides at the Oklo natural reactors in Gabon or studies of the alteration and corrosion of native copper in geologic deposits, such as those found in upper peninsula of Michigan, United States).

The safety case is developed early in the site selection process and is derived from the regulatory framework. The safety case provides clear expectations for the role of the engineered and geologic barriers and how these barriers will interact with one another over time. As the site selection process narrows to a few sites and site characterization begins, the safety case is refined. If new knowledge of the site violates the original strategy for the safety case, then the site should be abandoned. The safety case can also be developed in a way that makes greatest use of quantitative models when the results are most useful, probably within the first hundred to thousands of years after repository closure. The quantitative analysis relies heavily on laboratory data that have a demonstrated relevance to the time scales being modeled. Qualitative evidence plays a more important role over geologic periods (e.g., the selection of a site that is in a stable craton far from active tectonic and volcanic activity). Observations of natural systems, so called natural analogue studies, can play a key role in evaluating the performance of a repository at more distant times (Côme and Chapman, 1987; Ewing and Jercinovic, 1987). Underground research laboratories can provide fundamental data, a conceptual understanding of coupled processes, and the opportunity to test operational plans under repository-like conditions (NWTRB, 2020).

In summary, the safety case is more like a compelling legal brief that marshals evidence from a variety of sources in order to make the case for safety (Ewing et al., 2018). Key to evaluating the efficacy of the safety case is the answer to the following questions:

- Has the implementer demonstrated a sufficient understanding of how the repository system will “work” over time?
- Have alternative conceptual models of the key processes been considered?
- Have the uncertainties in the quantitative and qualitative analyses been identified?
- Have key issues raised by affected parties, such as nearby communities, been addressed?
- Finally, and of great importance, is the safety case consistent with the early rationale for the original site selection?

Although the quantitative analysis of a calculated dose to a person at some distant time and place may be part of a safety case, the answers to the questions above are the compelling argument for safety.

Regulatory framework

Once the repository is designed and the effects of the physical and chemical processes on the multi-barrier system are understood, the total performance of the system of engineered and geologic barriers has to meet regulatory requirements (Ewing et al., 2016). There are various critical parameters that are measures of compliance. In the United States, early regulations limited the fractional release relative to the radionuclide inventory for the different engineered barriers at 1000 years after emplacement. For geologic barriers, the regulations on minimum travel times, as an example, were defined for the groundwater flow rate. However, these early regulatory requirements were considered to be too restrictive and led to suboptimal performance in the total system analysis; thus, subsystem requirements were replaced by a global risk-based calculation for the entire system. This risk-based approach requires a total PSA that probabilistically calculates a quantitative estimate of radiation dose to a person or population at a specific time and location. Typically, the regulatory period extends to a million years. And while the risk-based calculation of dose may be required by the regulation, the complexity of the biosphere adds to the uncertainty in the analysis of the global safety analysis.

Most countries extend their safety analysis to a compliance period of 1 million years. However, the level of the quantitative analysis varies (Ewing et al., 2016). For example, the Swedish KBS-3 concept intends to provide containment of the radionuclides within

the engineered barriers until the radioactivity level has decayed to that of the original uranium ore from which the waste was derived. Safety analyses may be based on quantitative determinative models early in the disposal period and then be replaced by more qualitative arguments for the rest of the period analyzed. In other words, the safety case approach uses different methods, quantitative and qualitative, as needed over the entire time of the safety assessment.

Another fundamental prospect in the regulatory framework is the location of the compliance boundary, defined as the point at which calculations are made for the level of exposure to the public. The compliance boundary can vary dramatically, even within a single country. At WIPP, the compliance boundary is a square, 4 miles on an edge around the centrally located repository. At Yucca Mountain, the compliance boundary is a rectangle, with its length parallel to the direction of groundwater movement, stretching to 20 km. This boundary is set to increase the effects of dilution and sorption processes along the transport path, effectively making the transport of radionuclides a part of the multi-barrier system. Undoubtedly, all of these assessments require some confidence in the knowledge of where people will live and their daily habits far into the future.

Public engagement and acceptance: The social filter

Even when the total performance of the engineered and geologic barriers meets the regulatory requirements and a regulatory agency declares a geologic repository to be safe, public acceptance remains key to success (Ewing et al., 2016). Public acceptance does not come at the end of the process, but rather is earned throughout the entire process from the initial site selection, site characterization, and through license approval (Ewing et al., 2016). In this regard, the experience in different countries has met with different levels of success (Metlay, 2016, 2021). Notably, there is one aspect of the technical strategy that contributes to a favorable communication with the public, and that is the simplicity of analysis on different repository strategies. A robust strategy based on simple principles that are transparent when communicating with the public and politicians is probably more useful than overly sophisticated, probabilistic analyses that are less transparent. In fact, every safety analysis requires both—a simple, compelling argument for safety, as well as an underlying scientific basis, which can be quantitative or qualitative (Ewing et al., 2016).

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Relevant Websites

<https://www.deepisolation.com>—Deep Isolation, Inc.
<https://nustusa.net>—NuclearSafe.

Radionuclide Migration in the Context of High-Level Nuclear Waste Management

Jordi Bruno, Amphos 21 Group, Barcelona, Spain

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The basis for radionuclide migration in the environment

Introduction

In this article we will present some of the key processes that control radionuclide migration in a nuclear waste repository, from the nuclear waste matrix to the biosphere. These processes are known as near-field and far-field processes, in order to distinguish and categorize processes occurring at the vicinity of the waste package from the ones that control the fate of radionuclides in the geosphere.

The emphasis will be in high-level nuclear waste (HLNW) disposal, and mainly in crystalline rock environments, although we will introduce some of the key processes occurring in clay rock media, as an extension of the processes that control radionuclide migration in the clay buffer material surrounding the waste package. We will also touch upon the quantification of these radionuclide migration processes in the performance assessment of HLNW repositories.

Thermodynamic equilibrium vs. kinetic control

One key concept in the quantification of radionuclide migration is the concept of local thermodynamic equilibrium, which facilitates the systematic quantitative treatment of the chemical processes involved. Most of the chemical processes that we will hereto discuss can be treated as equilibrium processes, chiefly because the groundwater transport (or residence) times in the various parts of the repository are sufficiently long to grant this (Domenico and Schwartz, 1990; Bruno, 1997) (Fig. 1).

Nevertheless, there are two important exceptions to the equilibrium rule: dissolution-precipitation processes and redox reactions. In addition, some solid phase transformations shall be classified as kinetically controlled or as disequilibrium processes. This may include colloidal metastability given that it is an intrinsic disequilibrium process. We will try to discuss some of the implications as we get into the details of the process description. Radionuclide migration from the deep geological nuclear waste repositories to the biosphere will occur mainly by groundwater transport. In this context, radionuclides are only specific isotopes of some chemical elements and therefore their behavior in groundwater is controlled by the same aquatic chemical processes that control metal behavior in the various compartments of the environment. These processes are mainly (Stumm and Morgan, 1992):

- Aqueous complex formation
- Dissolution and precipitation of solid phases
- Redox transformations
- Colloid formation and stability
- Surface complexation/Ion exchange
- Surface precipitation
- Co-precipitation

A schematic of these processes om the context of deep geological disposal is presented in Fig. 2.

Characteristic vs. Residence times

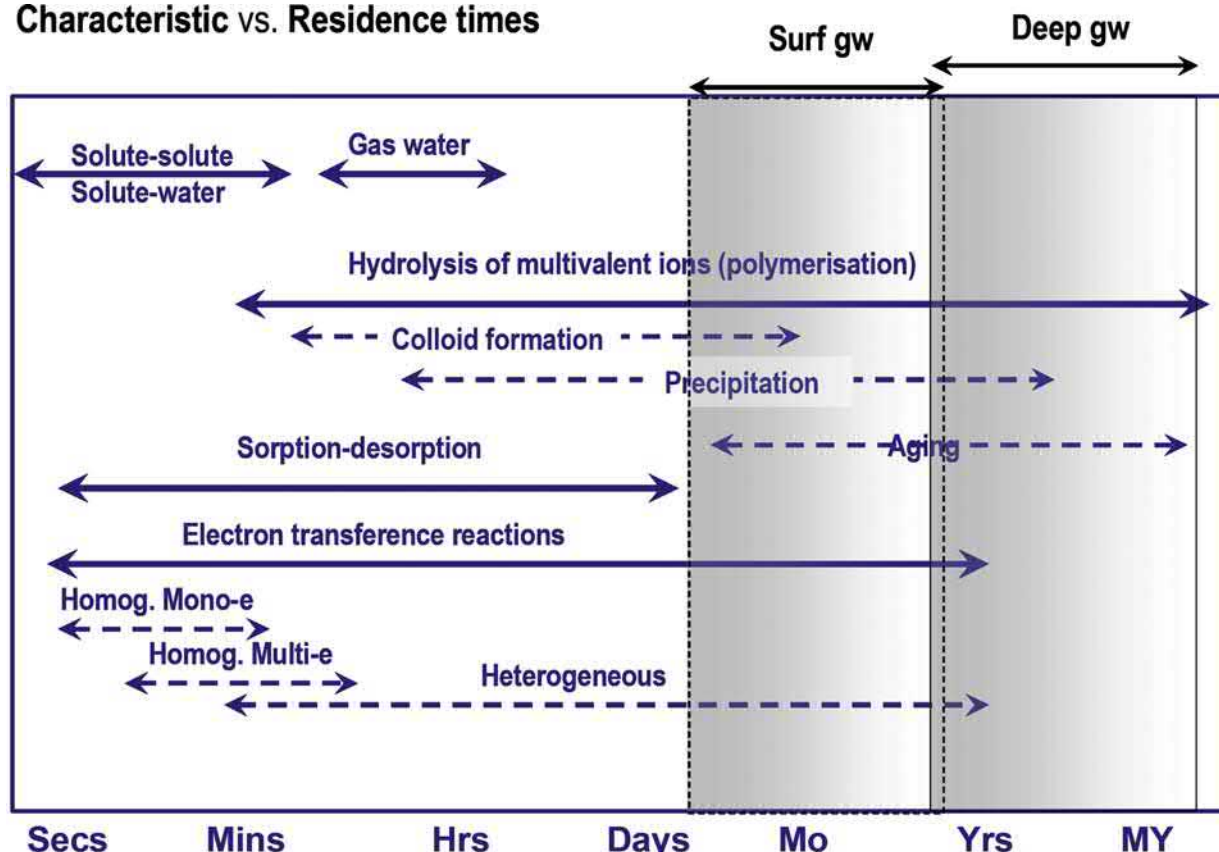


Fig. 1 Characteristic reaction times vs. residence times of groundwaters in the deep geological disposal system. Adapted from Domenico PA and Schwartz FW (1990) *Physical and Chemical Hydrogeology*. John Wiley and Sons, and Bruno J (1997) *Trace element modelling*. In: Grenthe I, Puigdomenech I (eds.) *Modelling in Aquatic Chemistry*. Paris: OECD/NEA.

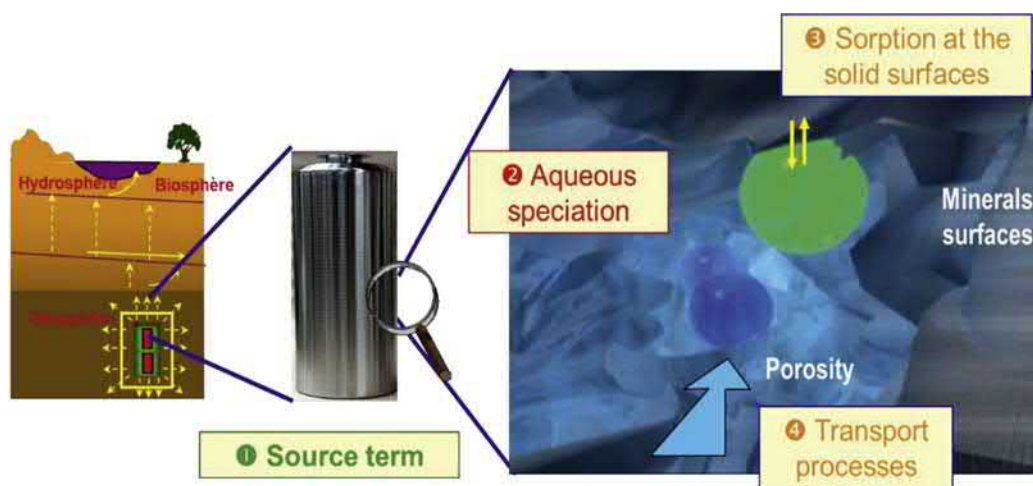


Fig. 2 Schematic view of the three kind of processes for the radionuclide migration in the deep geological disposal of nuclear wastes.

These processes can be classified as homogeneous and heterogeneous processes, depending on, if they occur in a single phase or if they involve phase transformation, including colloidal formation. Although the thermodynamic treatment of colloids as a phase is debatable.

Furthermore, some of the basic processes occur both in a single phase or through phase transformations, i.e., redox processes, therefore, is not very useful to handle them as homogenous or heterogeneous.

These basic aquatic chemical processes control radionuclide speciation and radionuclide distribution between the aqueous, solid, and surface phases. In terms of safety assessment are the solubility constant and the partition coefficient (K_d), two magnitudes that are fundamental in understanding and predicting radionuclide migration. However, radionuclide solubility and adsorption are dependent on the chemical speciation which at the end of the day is controlled by all the processes that we have presented.

Furthermore, the environmental impact of any metal, including radionuclides, is directly related to its bioavailability, which is in turn finally dependent on redox state and electrochemical behavior, and consequently on the radionuclide speciation. Furthermore, the link between the source of radionuclides (the waste) and the target biota occurs through groundwater transport processes, either advective or diffusive, or both, depending on the geological setting.

In the following sections will present a brief account of the basic aquatic chemical processes, with a specific focus on their relevance for radionuclide migration.

Aqueous complex formation

Complexes are stable entities that result from the formation of covalent bonds between a metal ion and the electron donating ligand.

Complex formation processes are the result of the interaction between ions and complexing ligands. Dissolved radionuclides are basically, metal and metalloids that form ions in water solution. These ions have a strong tendency to build complexes with the key complexing ligands present in the repository systems, including groundwater and in some cases, they also build ion pairs with counterions in concentrated media that may occur, for instance, in brines.

Their equilibria reactions are (in simplified form) written as:



And the equilibrium constants are defined as:

$$K_{\text{eq}} = [\text{RNL}]/[\text{RN}][\text{L}] \quad (2)$$

Ion pairs are the result of the electrostatic interaction in a critical distance which results in the total or partial neutralization of the ion charges. Longer-range electrostatic interactions are accounted for by the Specific Interaction Theory, which take into consideration the deviations from the pure Debye-Hückel electrostatic interactions. (The reader is referred to the publications by the NEA-TDB project for a more detailed explanation of these processes. [Grenthe et al. \(1997a\)](#) is an excellent introduction to aqueous chemical processes and in [Grenthe et al. \(1997b\)](#) there is a sound discussion of the effect of electrolyte theory on thermodynamic data.)

Dissolution and precipitation processes

These processes are extremely important in the evaluation of the safety of nuclear waste management options, including deep geological disposal. Dissolution is the key process controlling the release of radionuclides from the various nuclear waste matrices being disposed, while precipitation is a critical process to control the radionuclide behavior in the chemical gradients present in a repository system, particularly in the near field.

The thermodynamic treatment of the dissolution and precipitation processes is rather straightforward by using solubility constants such as:



Being the solubility constant defined as

$$K_{\text{so}} = [\text{A}] \cdot [\text{B}] \quad (4)$$

However, reality is far more complex. Nuclear waste matrices are by definition complex multicomponent systems. For instance, despite that the composition of spent nuclear fuel is up to 95% $\text{UO}_2(\text{s})$, its radionuclide content is present in several phases within or outside the uranium dioxide matrix. Thus, when dealing with spent nuclear fuel solubility and the subsequent radionuclide release, a complex multiphase approach must be applied. We will further address this when discussing near field processes.

Likewise, the reverse process, radionuclide precipitation, is also a remarkably complex process. While it could be argued that the solubility constant (Eq. 4) applies to determine the solubility limit of a certain radionuclide in the near field and the far field environment, the reality is richer and far more complex.

The fact is that radionuclides in the repository environment are normally trace components, and consequently their solubility behavior is determined by the interactions with the main groundwater components through their migration pathway. This means that in a very few cases, the individual solubility of a radionuclide is a relevant parameter to understand its migration behavior. As a matter of fact, the interaction between many of the key components of the repository system with groundwater is controlled by solid solution-aqueous solution processes, particularly in the near-field, but also in the far field.

Many components in UO_2 and MOX spent nuclear fuel are present as solid solutions of the UO_2 matrix, and their dissolution behavior can be explained by using the framework of aqueous solution-solid solution processes ([Bruno et al., 2007](#)). A similar situation has been described for the behavior of trace components in the altered layer of vitrified nuclear wastes ([Grambow and Müller, 2001](#)), the alternative matrix for the closed fuel cycle.

Cement is a central component in many designs. Its alteration forms an array of calcium silicate hydrate solid solution phases, commonly known as CSH.

The corrosion of steel containers and cast iron inlets produces a variety of Fe(III)/Fe(II) oxyhydroxide solid phases that are able to incorporate trace components in their structures. While initially the trace metal interactions are mainly controlled by adsorption processes and described by surface complexation models (Dzombak and Morel, 1990), the evolution of these Fe(III) oxyhydroxide phases with time (aging) and their interactions with contacting fluids may be described also in the context of the aqueous-solid solution systems (Bruno et al., 2007).

We will discuss these processes in more detail as we describe radionuclide migration in the near field and far field environments.

Redox processes

In principle, radionuclide migration occurs through different environmental compartments with a varying degree of oxidation. One of the key safety principles for many high-level nuclear waste repositories is based on the fact that the waste itself and containment system, known altogether as near-field, is kept under reducing conditions. The grounds for this is that deep underground the redox condition varies from anoxic to reducing.

In their migration pathway towards the biosphere, radionuclides will encounter various stages of the redox scale (Stumm and Morgan, 1992) which will have an impact on their chemical speciation, solubility and surface distribution (adsorption). Therefore, the redox state constitutes together with the pH, the two master variables that control radionuclide migration in groundwater systems.

Redox reactions are the result of electron transfer processes from the reduced species to the oxidized one, and vice versa. Contrary to the proton, the free electron is not stable in aqueous solution. Hence, redox reactions are the result of, at least, two simultaneous redox pairs. This brings some additional complexity to the definition of redox systems, mainly because there are important kinetic hindrances that make these reactions to be slow in natural systems. In addition, as we have already discussed, redox processes may occur in a single aqueous phase but also through phase transformation reactions.

However, as explained in Bruno (1997), for most of homogeneous redox reactions equilibrium can be assumed to be reached within the long groundwater residence times expected in the repository system. The exceptions to this principle are the redox reactions involving multielectron transfer and large structural rearrangements, for instance, the reduction of sulfate to sulfide, which is a slow process, unless it is mediated by bacteria.

This brings us to a very important issue when dealing with redox process: living organisms thrive in the energy gradients generated by these redox reactions and therefore they play a critical role defining the biogeochemistry of subsurface systems. Therefore, microbial mediated reactions can play a very relevant role, even in deep groundwater systems (Hallbeck and Pedersen, 2008).

There is a number of relevant processes for radionuclide migration that occur at the frontier of the solid solution interface, these include colloid formation, surface complexation and surface precipitation. We will touch upon them in the following sections.

Colloid formation

This is a critical process for radionuclide migration. Colloids are nanoparticles of size between 1 and 1000 nm of organic or inorganic composition which are found in natural groundwaters (Stumm and Morgan, 1992), and particularly in anthropogenically disturbed systems, such as nuclear waste repositories (McCarthy and Zachara, 1989; Kim, 1991; McCarthy and Degueldre, 1993).

Colloids have been classified as intrinsic (*eigen*) colloids, or pseudo-colloids. Intrinsic colloids are formed by many metal ions when they hydrolyse and particularly by the actinide ions. Actinide (IV) are very prone to build colloids in water systems.

Pseudo-colloids are basically radionuclides attached to external colloidal phases occurring in the various compartments of the repository system. This nomenclature is somewhat misleading, as pseudo-colloids are as real and normally even more critical to radionuclide migration than *eigen*-colloids.

The main process involved in colloid-driven transport is surface complexation and/or surface precipitation. These processes are normally bumped in the general term "adsorption," which includes all reactions responsible for the attachment and immobilization of radionuclides on solid (mineral) surfaces.

Adsorption is a very important process in the safety assessment methodologies used in nuclear waste management, but its handling lacks at present the thermodynamic rigor granted for other migration/retardation processes. We will discuss this point deeply in the performance assessment section.

Transport processes

As we have already indicated, the main mechanism for the migration of radionuclides is groundwater transport. Gas transport may also occur, particularly for gaseous nuclides like Rn, but it has a minor contribution to the overall radionuclide migration process in the HLNW context.

The main hydrogeological mechanisms for groundwater transport are advection and diffusion.

Advection is the process in which the migrating radionuclides are transported at the rate of the flowing groundwater. In crystalline rock systems, advection occurs through fractures and microfractures, where it combines with diffusion processes. Radionuclide transport through advection is mathematically approached by using the mass and energy conservation laws.

Diffusion is the net movement of radionuclides in concentration gradients and it is the main operating mechanism for radionuclide transport in geological media with very low permeability like clay rocks, including the bentonite buffer material. A specific case of diffusion applies to the mechanism operating in bulk rocks at the interface with conductive fractures, which is known as matrix diffusion (Neretnieks, 1980).

Diffusion is mathematically approached by Fick's law although some relevant diffusive transport processes are considered not to follow this law and are considered *non-fickian*.

In general, advective transport is faster than diffusive transport, and therefore it is more efficient in terms of radionuclide migration.

Diffusive transport is the main groundwater transport process in clay rock environments which is the preferred geological disposal option for HLW in countries like France, Belgium and Switzerland, for instance.

Reactive transport models

The integration of chemical and groundwater transport processes affecting radionuclide migration is conducted by applying reactive transport models. These models are fundamental for assessing the potential radionuclide mobility in the various portions of the repository system and are the basis of most safety assessment exercises.

A large part of the development of these models have been the result of the work related to the needs of development of these kind of models as a result of the progression of safety assessment exercises for the geological disposal of HLNW.

This is clearly proved/shown by comparing two of the seminal works concerning Reactive Transport Modeling, which have been collected in the Reviews of Mineralogy and Geochemistry (Lichtner et al., 1996; Steefel, 2019). While in the 1996 version (Lichtner et al., 1996) the emphasis was in the application of these models to the understanding of the evolution of geochemical systems, the issue from 2019 (Steefel, 2019) contains many references to nuclear waste management problems in spite of the fact of the somewhat parochial sight from US scientists which tend to overlook what is being done in Europe and elsewhere which in many respects is far more advanced in the applicability of these models for nuclear waste management (Markelova et al., 2020).

However, in a recent review, Steefel (2019) points out that one of the key limitations of the applicability of Reactive Transport Models is the insufficient approach to multicomponent sorption that the K_d parametrization offers. Something that we consider also to be a weakness in the consistent application of Reactive Transport Models in the safety assessment of nuclear waste repositories.

There is a substantial bibliography concerning reactive transport models, and the reader is referred to these works for additional information. An overview of the historical development of Reactive Transport Models is given in the following picture extracted from a recent report written by Amphos 21 concerning the development of Reactive Transport Models in fractured crystalline media (Markelova et al., 2020) (Fig. 3).

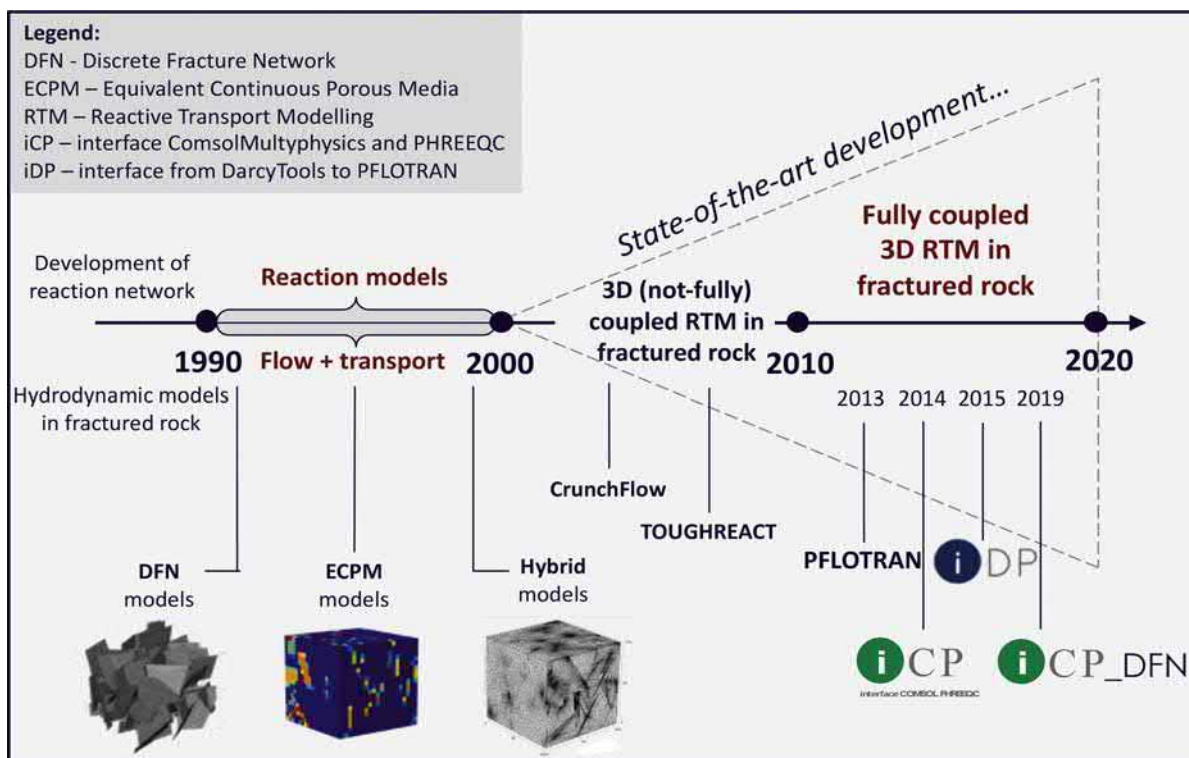


Fig. 3 Historical development of Reactive Transport Models in crystalline, fractured, rock (Markelova et al., 2020).

Nevertheless, the fundamental question remains the same: following the GiGo dilemma, the outcome of these models will only be as good as the input data. If we insist on providing limited chemical information into these models, the output of these models will still be rather limited, no matter how sophisticated the hydrological model is.

Geological disposal of high-level nuclear waste (HLW)

Near field components and processes

As it has been mentioned in the Introduction, the environment surrounding a nuclear waste repository is divided in two main zones, the near and the far field. The near field is the closest area surrounding the waste emplacement and it is by definition affected by the presence of the waste.

In principle, the far field is not disturbed by the waste emplacement, although there can be a transition zone which is indeed perturbed by the placement of the waste.

In terms of radionuclide migration, the near field is a complex system where radionuclide release processes are originated, and where the released radionuclides interact with a number of engineered barriers. This is prior to their continued travel in the bedrock in which the repository is located which constitutes the far field.

The near field of a nuclear waste repository is mainly composed by the waste matrices and their containment which may vary depending on the type of waste which not only depends on the activity but also on the preferred nuclear cycle type, open or close.

Nuclear wastes are classified according to their radiological activity and the decay times of the radioisotopes included in the waste.

For low and intermediate waste, the near field can be a heterogenous mixture of containment materials and waste packages, depending on their origin. A typical example of low and intermediate nuclear waste repository is given in Fig. 4.

Typically, many L + IW waste streams coming from the nuclear power stations are conditioned in concrete and steel containers. In addition, there are some waste streams, particularly arising from former operations, that have been conditioned in bitumen.

HLW streams are diverse but the majority can be classified either as spent nuclear fuel (UO₂-fuel or MOX) or vitrified HLW, in which the waste arising from the reprocessing activities is mixed in melt-glass prior deposition.

Hence, radionuclide migration in the near field is a complex multi-process array which has many intricacies depending on the management concept. We will try to categorize these, depending on the waste type and the disposal concept.

Near field processes in high level nuclear waste (HLW) disposal

As we have previously indicated, there are two main streams of HLW, spent nuclear fuel and HLNW glasses. In both cases, the initial process that destabilizes these waste matrices is their contact with groundwater and the subsequent radionuclide release.

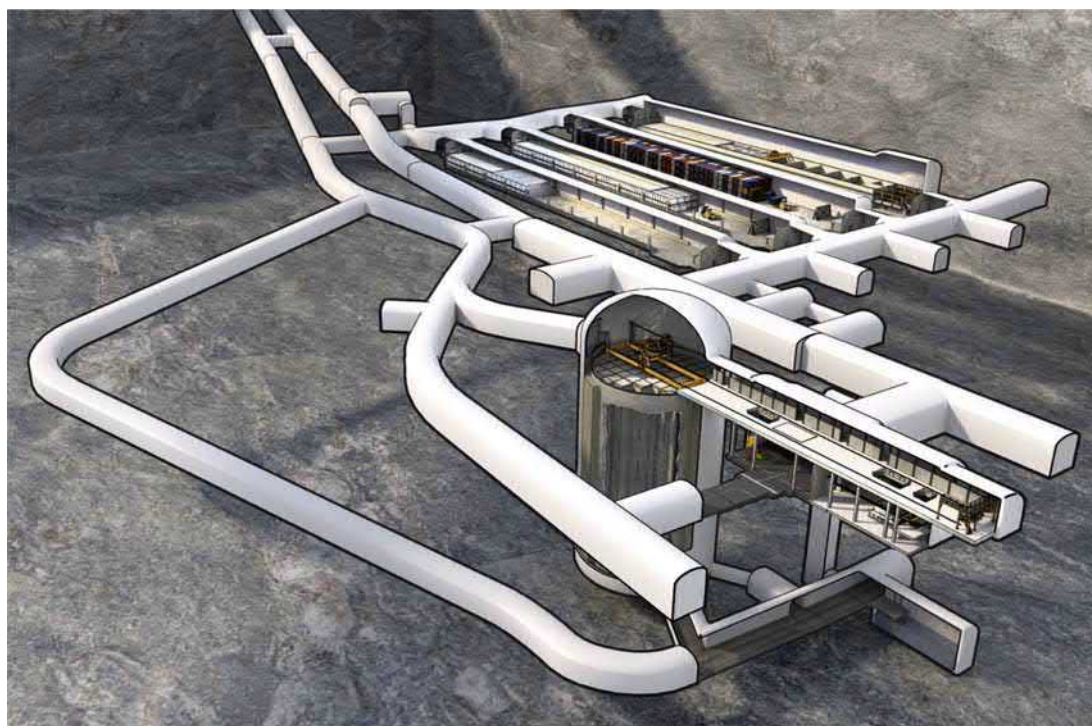


Fig. 4 Layout of the SFR—SKB's Final Repository for Short-Lived Radioactive Waste located at Forsmark in the municipality of Östhammar, Sweden.

The intruding groundwater has gone, in principle, through the various components of the near field, prior to contacting the waste matrix. Therefore, the interactions of groundwater with the near field components are critical to radionuclide release, as well as to radionuclide migration.

In the case of spent nuclear fuel, the waste is surrounded by the zircaloy cladding, a metallic container, and the buffer material. In the following Fig. 5 we present a scheme of the KBS-3 concept for UO_2 spent nuclear disposal that is being implemented in Sweden and Finland. In this case, the metallic container is a copper canister with a ductile iron insert and the buffer material is compacted bentonite clay.

Thus, the key processes that will render possible spent fuel dissolution are the interaction of groundwater with the surrounding bentonite, and the corrosion of the copper canister. If the groundwater corrodes the copper canister, then it will get in contact with the iron inlet. This results in hydrogen generation and the development of highly reducing redox conditions (Duro et al., 2014). This process is also key in controlling the redox condition of the low and intermediate waste repositories, where most of the waste materials are kept in iron containers.

The development of hydrogen in the vicinity of the spent fuel has many implications in the dissolution of spent nuclear fuel and the subsequent radionuclide release (Carbol et al., 2009), imposing reducing conditions at the spent fuel surface, which counterbalances the production of oxidants by water radiolysis.

In any case, several radionuclides will be released from the disposed spent nuclear fuel if they are put in contact with groundwater. These are mainly fission products and are categorized as the Instant Release Fraction. The Instant Release Fraction (IRF) is the fraction of the radioactive inventory that will be released from the waste “immediately” after the fuel rod cladding fails, and the waste containment is compromised.

The waste form consists in a rod containing stacked pellets of polycrystalline UO_2 fuel during reactor operation, concentration and thermal gradients drive redistribution of some of the fission products from intragranular to the intergranular space, and outside the pellet towards the inner-rod void space. It is generally assumed that under repository conditions, breaching of the cladding would rapidly release the inventory accumulated in the void space of the rod. In addition, the radionuclides segregated at the fuel grain boundaries would also be released relatively quickly.

After the instant release of radionuclides from the fuel, a much slower radionuclide release mechanism would occur over long periods of time due to dissolution of the UO_2 matrix (Bruno et al., 1996; Poinssot et al., 2005). The main alteration reactions that may occur at the spent fuel/iron canister/groundwater interface were described as a result of an Euratom collaborative project (Poinssot et al., 2005) and have been depicted in the following Fig. 6.

It is important to realize that the schematic shown in Fig. 6 is not drawn at scale. The oxidizing front extends to some hundred microns from the surface at maximum, while the reducing front covers some decimeters in range. The near field rock, which is placed at meters length and separated from the canister by the backfill, ensures that the overall redox condition remains reducing. The net effect is that H_2 generation at the reducing front is able to offset the surface oxidation (Carbol et al., 2009).

Once radionuclides are released, they encounter a changing groundwater composition and some of the initially released radionuclides, particularly the actinides, will meet reducing conditions which will favor their precipitation as actinide(IV) hydroxides. In the case of Pu, it may trigger the build-up of eigen-colloids, as we have previously discussed.

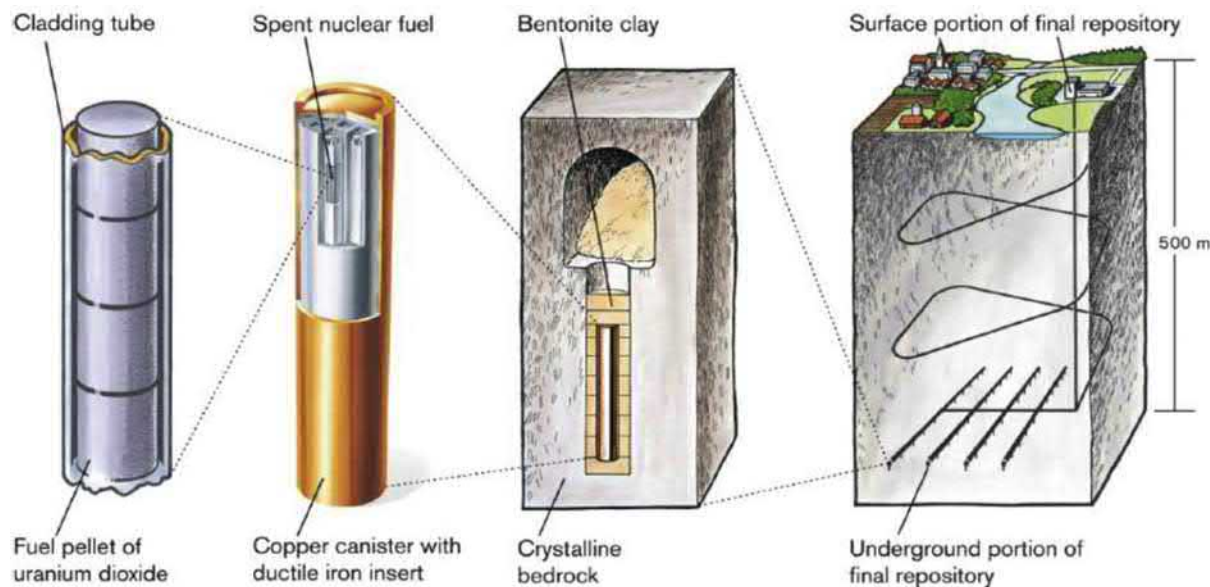


Fig. 5 The KBS-3 concept for spent nuclear fuel disposal (SKB, 2020).

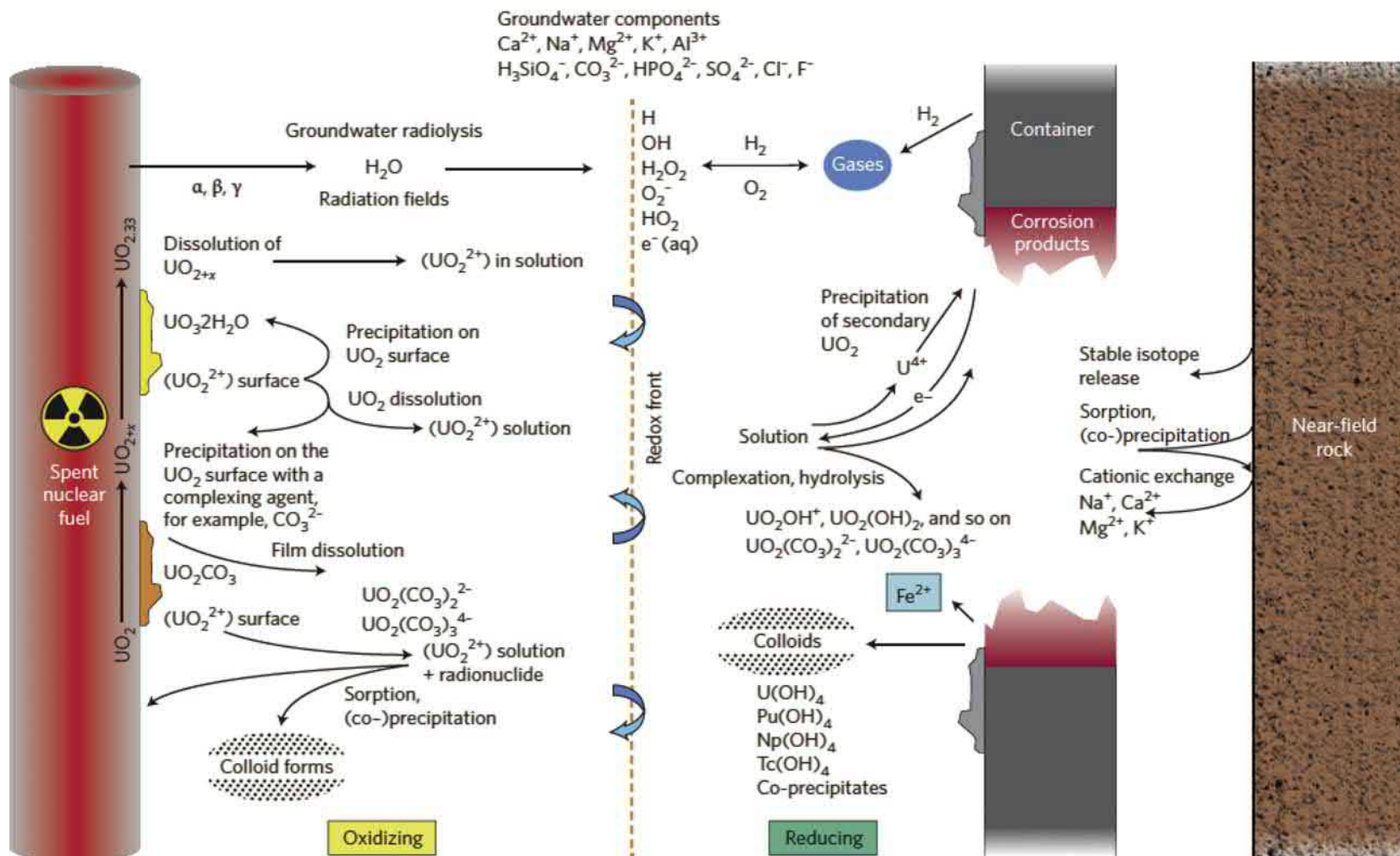


Fig. 6 Scheme of the reactions occurring at the near field of a spent nuclear fuel repository (Ewing, 2015).

In the case of other relevant radionuclides like radium, their migration will be controlled by processes affecting the solubility of barium. The presence of sulfate in the contacting groundwaters will facilitate the precipitation of the insoluble barium sulfate (barite), and radium will co-precipitate with the barite that is formed. There has been a substantial amount of research (i.e., Vinograd et al., 2013) of this co-precipitation process since its consideration in performance assessment models (Grandia et al., 2008).

Further to these chemical processes, the migrating radionuclides will encounter the metallic components, including the zircaloy cladding and the components of the canister, basically metallic iron and copper, in the case of KBS-3 concept. Under the anaerobic conditions expected in the near field, iron will be corroded to magnetite (Duro et al., 2014) and therefore radionuclides will interact with the magnetite surface.

The interaction of radionuclides with the surface of magnetite has been investigated, particularly for redox sensitive elements like uranium (Missana et al., 2003a,b; Singer et al., 2012) and selenium (Martínez et al., 2006); but also, non-redox sensitive elements like Th (Rojo et al., 2009).

An additional component that is not depicted in Fig. 6 is the buffer material surrounding the canister, which has several functions, mainly to guarantee physical and hydrological stability. In the KBS-3 concept, the selected buffer material is compacted bentonite, which is mainly composed by montmorillonite clay (> 80%). It also contains a few accessory minerals like feldspars, quartz, cristobalite, gypsum, calcite, and pyrite. Montmorillonite belongs to the smectite group, in which all the minerals have an articulated layer structure. A model structure of bentonite is given in Fig. 7.

The physico-chemical properties of bentonite have a strong influence in the migration processes that may occur within the bentonite barrier. Water saturated bentonite has an extremely low hydraulic conductivity and therefore radionuclide migration is diffusion controlled and the predominant retention process is sorption, either through ion-exchange and/or surface complexation processes.

Montmorillonite consists of octahedral alumina sheets sandwiched between tetrahedral silica sheets (2:1 clay). These clay minerals feature two distinctly different types of surface, where two main types of sorption take place (Stumm, 1992):

- The siloxane ("layer") surfaces of clay minerals are permanently charged surfaces. These charges derive from isomorphous substitutions, which result in a constant negative surface charge. This charge is partly compensated by cations and bound to the clay surface, and partly by cations simply accumulated in a counter-ion swarm. The latter is equivalent to an electrical double layer extending from the surface. Macroscopically, sorption takes place when compensating ions are exchanged therefore ion exchange models of the site-binding type have are used to describe this process.
- The edge surfaces of clay minerals are variably charged surfaces. They carry a net positive or negative surface charge depending on the sorbed species to their surfaces which often involve surface-bound OH⁻ groups. Surface complexation and ligand exchange models were established by Stumm, Schindler and co-workers in the 1970s by extending proton binding and metal coordination chemistry in a rigorous fashion to surface chemistry. To account for the electrostatic field, the mass laws for surface equilibria, often include an electrostatic correction term. These corrections are based on the Electrical Double Layer (EDL) theory (Stumm, 1992).

In terms of the interaction with migrating radionuclides, ion exchange is the main sorption mechanism for alkali, and alkaline-earth elements, as well as for some transition metals at low pH values, where positive species are predominant. In the pH range of interest, surface complexation is generally the relevant process for all the key radionuclides (transition metals, actinides, lanthanides, and key groundwater anions such as carbonate).

It is important to realize that both ion exchange and surface complexation reactions take place simultaneously, but at different surface locations. The most important factor for both processes is the ion content of the porewater, which is principally determined by the original cations in the exchange positions of the montmorillonite, the readily soluble minerals in the original bentonite material, like calcite, and by the influence of the intruding groundwater.

Protons are bound to the edge sites of clays (as well as to ion exchange positions at very low pH), which means that pH changes can occur due to surface chemical reactions. Given its buffering properties, the montmorillonite can counteract external changes in pH. The presence of pyrite as an impurity has also an effect as redox buffer (Wanner et al., 1992).

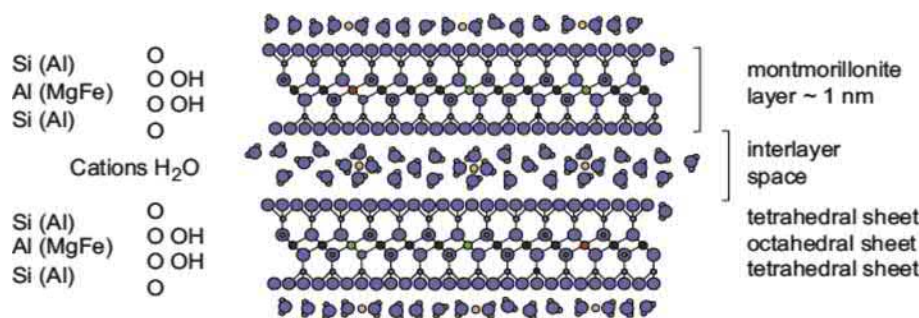


Fig. 7 Layered silicate structure of montmorillonite, the main component of bentonite (Karnland et al., 2006).

In terms of describing radionuclide migration through bentonite, the reversibility of the sorption is a fundamental concept. Ion exchange and surface complexation models have been developed to take into consideration radionuclide sorption in bentonite. The most comprehensive and systematic work in this context has been performed by the PSI group, led by Mike Bradbury and Bert Baeyens. There are a series of publications that establish consistent and reproducible surface complexation models to describe the interaction of most radionuclides with bentonite (Bradbury and Baeyens, 2009).

One specific complication of the radionuclide interaction with bentonite is the concept of anion exclusion. As we have already discussed, the montmorillonite structure has an excess of negative charge that is compensated by cations in the interlayer space and on the outer surface. Anions are therefore, repelled from the negative surface of montmorillonite and thus, occupy only part of the pore space of bentonite. This effect can become small, or even insignificant, when the bentonite is highly compacted and the external solution has low ionic strength (Molera et al., 2003; Muurinen et al., 2007; Van Loon et al., 2007). The reader is referred to these publications as well as to the work of Tournassat and Appello (2011).

Another aspect that is fundamental in terms of radionuclide migration is the fact that compacted clays in general, and bentonite in particular, are very efficient filters for colloidal transport and microbial development (Holmboe et al., 2010; Haydn et al., 2018), this in some respect helps to constrain the effect of microbes and colloids on radionuclide transport through the buffer barrier as well as in clay rock in general.

There is, however, a complication in this context, which is that bentonite colloids can be generated as a consequence of ground-water weathering under certain circumstances (Missana et al., 2003a,b). This is particularly important when diluted groundwater will enter in contact with the bentonite buffer material. In principle, this is a process that is unlikely to happen in contact with groundwater as its cation content is sufficiently high to prevent it, but it might be possible if very diluted melt waters intrude the repository in glaciation/deglaciation cycles. This phenomenon has been largely investigated within nuclear waste management projects located in areas where glaciation/deglaciation cycles are likely to occur through the forthcoming glaciation cycles, like the Swedish, Finnish and Canadian programmes.

Radionuclide migration in the far-field

Most of the processes that we have discussed in the near field are reproduced with a different intensity in the far field. In clay rock repositories the key processes we have established for the bentonite buffer material are acting on the migration of radionuclides. Ion-exchange and sorption are the key chemical retention mechanisms, while diffusive transport is the predominant transport pattern.

In crystalline rock repositories, radionuclide migration occurs through fractures and the transport patterns are mainly advective, although diffusive transport within the rock matrix plays a substantial retention role. Radionuclide migration processes are very dependent on their chemical speciation and therefore the aqueous composition of the flowing groundwater will determine most of the retention processes through the far field.

Groundwater chemistry is mainly determined by the type of rock. Granitic groundwaters have a quite constant composition mainly controlled by the dissolution and precipitation of calcite and feldspars in the fractures. The master variables of the system are pH and pE (redox potential). While pH is rather constant in undisturbed groundwaters, the redox potential varies depending mainly on depth and the potential intrusion of oxygen from the atmosphere.

As a rule, deep groundwaters are reducing while surface waters are oxidic. This clearly determines the chemical speciation of the redox sensitive radionuclides, like Uranium, Neptunium and Plutonium, and consequently drives their sorption interactions.

Aqueous complexation influences the radionuclide sorption behavior, both by stabilizing the radionuclides in the aqueous phase and increasing their surface complexation. An example is given in the study by Bargar et al. (1999), in which surface complexation of bidentate carbonate complexes increases uranium sorption. This is one of the main reasons while aqueous-surface equilibrium models are useful to ascertain these processes, which would be largely overlooked if the K_d approach would be considered.

The salinity of the groundwater also affects the extent of ion-exchange in clay materials, as solutes will compete with the radionuclides for ion-exchange sites. This is particularly important in bentonite, in which ionic strength has a strong effect on pore water composition, but also in the clay rock as well as the clay materials coating the fracture surfaces.

Sorption can take place directly on the outer surfaces of flow-bearing fractures or on grain-boundary surfaces and microfractures within the rock matrix, where the porewater is effectively stagnant. This is the basis of the phenomenon conceptualized as Matrix Diffusion (Neretnieks, 1980).

Radionuclide sorption can also take place on other materials, such as secondary clay minerals in fracture-surface coatings. As a rule, sorption processes result in the retardation of solute transport along a flow path. However, if radionuclides are attached to colloids, then radionuclide migration will not be retarded. Sorption is only the first step towards phase transformations, such as precipitation and co-precipitation, which can give rise to additional radionuclide retardation.

Radionuclide precipitation/dissolution processes and the formation of solid solutions (i.e., coprecipitates) is crucial for establishing relevant background concentration ranges for stable and naturally occurring radioactive isotopes, which can influence the sorption of the radioactive isotopes (the radionuclides), which are originated in the repository (SKB, 2014). This is particularly important for elements such as cesium, strontium and uranium.

Natural background levels of radium are sufficiently high to determine the concentration range in which radium adsorption must be determined. Co-precipitation of radium and barium as a solid solution in barite is a well-known process, and under

most groundwater conditions is expected to exert an important control on the concentration range of radium present in groundwater owing to the low solubility of barite and its ubiquitous nature (Bruno et al., 2007).

The different minerals participating in sorption reactions have different capacities for adsorption. Clay minerals and iron oxyhydroxides, for example, have a large sorption capacity, whereas quartz and feldspar tend to sorb very weakly. In granitic rocks, biotite, or its alteration product chlorite, as well as hornblende, have the strongest sorption affinity for radionuclides. The sorption of cesium and strontium, for example, is known to be strongly concentrated to biotite and chlorite (micas) which, being phyllosilicate minerals, have a large capacity for ion-exchange sorption (e.g., Torstenfelt et al., 1982; Huitti et al., 1998).

Uranium and neptunium also have an affinity for the iron-rich minerals (Kienzler et al., 2009), possibly due to surface complexation sorption on frayed edge sites of micas combined with reduction to the more strongly adsorbing tetravalent state. The surface complexation sorption of nickel, plutonium, and americium, on the other hand, appears to be less specific.

Although sorption of solutes, including radionuclides, is generally reversible in a thermodynamic sense, the incorporation of metal ions into solid phase lattices can result in an effectively irreversible immobilization. However, owing to the dynamic equilibrium of most common mineral phases likely to incorporate radionuclides, irreversible immobilization is not always verifiable, and such retardation processes are usually neglected in safety assessments. This is of course a weak argument difficult to defend because the irreversibility of some of these processes may involve the formation of "radionuclide pumps," which are accumulated in some fracture zones and may be released when the chemical conditions are suddenly changed, for instance, through tunnel excavation and fracture grouting.

Although the master variables pH, redox, and concentration of complexing ligands in the groundwater largely determine the speciation (distribution of an element among different chemical species) of chemical elements, temperature and pressure also play a minor role given the effect that these variables can have upon the thermodynamics of chemical reactions. The importance of chemical speciation in safety assessments lies in its impact on the geochemical retention and immobilization reactions likely to occur, and their consequences for radionuclide mobility.

The existence of an alkaline pH plume as a result of the use of concrete in various parts of the repository may influence the retardation of radionuclide transport in a number of ways. This is particularly important in low and intermediate waste repositories where most of the waste is conditioned in concrete.

The penetration of an alkaline plume will affect the migration of radionuclides in several ways, either by increasing the solubility of some key actinides and lanthanides, which tend to be more soluble at high pH values due to the formation of anionic hydrolysis complexes, but also affecting the rock and their fracture materials and therefore disturbing the retardation properties.

The alkaline plume may influence retardation of migrating radionuclides by altering the material properties of the rock matrix. In the immediate vicinity of the repository, high pH conditions may promote the dissolution of quartz and precipitation of secondary Ca-silicate crystalline or gel phases. Similarly, the dissolution of plagioclase may result in the precipitation of Mg-saponites. Since these secondary minerals typically have a larger molar volume than the primary quartz dissolved from the rock matrix, the net effect is a decrease in porosity, both in the rock matrix and in the fractures, which will be shotcreted.

Although this is an effect which might primarily influence the rock matrix effective diffusivity, it may also influence adsorption, as secondary precipitates would cover the surface of other minerals in such a way that the overall retardation effect would be altered. The net effect of the alkaline plume in fractures will change with time as the alkalinity of the cement-based plume will decrease with time as the cement ages and alkaline cations are depleted.

Natural analogs

In any case, the heterogeneity and complexity of all the migration processes involved in a HLNW repository, together with the fact that these processes have to be assessed for hundredths of thousands of years, has promoted the use of Natural Analog systems as a way of assessing the validity of the key hypothesis and their quantification.

What follows is a brief account of the use of Natural Analog studies in radionuclide migration investigations related to HLNW disposal. For a more detailed account the reader is referred to Duro and Bruno (2012).

Generally speaking, Natural Analog studies have helped to increase the confidence in the qualification and quantification of the key processes that control radionuclide migration. It has also given the possibility to test the Hydrogeochemical Models used to describe radionuclide migration in a repository system. This has been the combination of several testing exercises of these models in Natural Analog systems, which have provided confidence on the tools, but also have helped to improve the thermodynamic data bases associated to these calculational tools.

The key NA studies that were used for this purpose are the following, classified based on the geological media involved:

1. Crystalline media
 - a. El Berrocal (Spain)
 - b. Palmottu (Finland)
2. Sandstone-clay media
 - a. Oklo (Gabon)
 - b. Cigar Lake (Canada)
3. Alkaline-volcanic media
 - a. Poços de Caldas

4. Hyperalkaline media
 - a. Maqarin (Jordan)

A description of these analog systems can be found in [Miller et al., \(2000\)](#), and the main outcome of these studies has been published in [Bruno et al. \(2002\)](#). Additionally, Natural Analogs studies are also used in the Safety Assessment of HLNW repository to bound the key processes involved in radionuclide migration and the models used for their quantification.

Radionuclide migration in safety assessment exercises

The space and time scales involved in the safety assessment of HLNW repositories impose many constraints in the way the radionuclide migration processes we have previously described are handled. This is in spite of the continuous improvement and the enlargement of the computing capacity.

As a matter of fact, the handling of radionuclide migration processes in safety calculations simplifies in a dramatic way the complexity we have been discussing. Two main chemical processes are integrated in these calculations:

1. Radionuclide solubility, to determine the source term from the waste form;
2. Radionuclide sorption, to describe the interactions between radionuclides and the contacting surfaces through the migration pathway.

Furthermore, several simplifications are involved in the interactions that are considered in the migration pathway. For instance, in the case of spent nuclear fuel disposal, the zircaloy cladding is/is considered as transparent in terms of radionuclide migration. This is a hypothesis which can be absurd, as in the reverse it would imply that the radionuclides generated and contained in the fuel matrix would be released freely in the operating reactors.

When it comes to the metallic containers, they are only treated as physical barriers and this applies exclusively to the integrity of the copper canister for which corrosion mechanisms under repository conditions have been extensively investigated ([Hedin et al., 2018](#)).

The role of iron and its corrosion products as a radionuclide retention barrier is largely overlooked based on a pessimistic approximation to the consequences of radionuclide migration. Hence an enormous retention capacity is not considered in HLNW repositories but also in LINW ones, where iron, together with cement, are the main system components.

Finally, the complexity of radionuclide sorption in the various parts of the repository is dumped in partition coefficients, which cannot represent the richness of the surface interactions in a complete manner, and cannot be linked properly to the geochemical variability of the system. Additionally, linear K_d cannot represent properly the competitive effects on sorption among the various radionuclides ([Steeffel, 2019](#)).

The main reason for this can no longer be the lack of computing capacity in a time where fully coupled models are now run routinely (see the “Reactive transport models” section). The main reason behind this is the fact that the complexity of the hydro-geological models hijacks the capacity of transport models to capture the richness (and the beauty) of radionuclide migration chemistry.

Let's hope that in the future the chemical processes behind radionuclide migration will be fully integrated in the radionuclide transport models used for the safety assessment of HLNW repositories, and that the chemical complexity of the repository components is considered in a comprehensive way.

Probably, the transition from conceptual to real repositories will require a more detailed quantitative account of these processes, both in the near and the far fields, as construction and operation of the repository will alter the migration pathways and in some cases in an irreversible manner. In these cases, the hypothesis of linear K_d to represent radionuclide sorption will become less and less defensible.

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Main Repository Projects Worldwide

Soufiane Mekki, ANDRA—French National Radioactive Waste Management Agency, Administrator of IAEA and OECD/NEA relations with Andra, International Relations Department, Chatenay-Malabry cedex, France

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Presentation of the sole geological repository in operation: The WIPP Site (US)

The United States is the first country in the world to have begun operating a deep geological repository for radioactive waste. The Waste Isolation Pilot Plant (WIPP) in Carlsbad, south-eastern New Mexico, is currently the only waste disposal facility of its type in existence. This site for the disposal of defense-related transuranic waste (TRU) in a deep salt layer was commissioned in 1999 (Fig. 1).

Defense-related TRU waste is so called because it is made up of materials that have been contaminated with plutonium. It partly consists of protective clothing, such as coveralls, gloves, shoe covers, etc. It also includes dismantled machine tools, rubble from the demolition of workshops and sludge contaminated with radioactive substances from US nuclear weapon plants.

Under the 1992 Land Withdrawal Act, WIPP is not licensed to accept high-level waste or spent fuel from civilian nuclear activity initially destined for disposal at the Yucca Mountain repository in the Nevada desert.

WIPP is located in a sparsely populated desert area. Waste is disposed of in salt domes. The salt formations, the top of which lies 305 m below the surface, are 650 m thick. Salt is considered to be a good medium for repositories as it is free from water. The geology of the salt formations at the WIPP has been stable for 200 million years. The choice of salt formation for WIPP is justified by the absence of freshwater, the ease with which this material can be excavated, its non-permeability and its geological stability. However, its most important quality for the purpose of radioactive waste containment over a very long time scale is the way in which the saline formation seals all fractures and naturally closes all openings (US DOE, 2016) (Fig. 2).

The underground repository facilities are laid out on a plane at a depth of approximately 650 m. They comprise four shafts: two air intake and extraction shafts, one for mining activities and taking salt to the surface, and one for lowering and managing waste. The site is organized into 10 waste disposal areas (Fig. 3).



Fig. 1 WIPP surface facilities. U.S. Department of Energy.

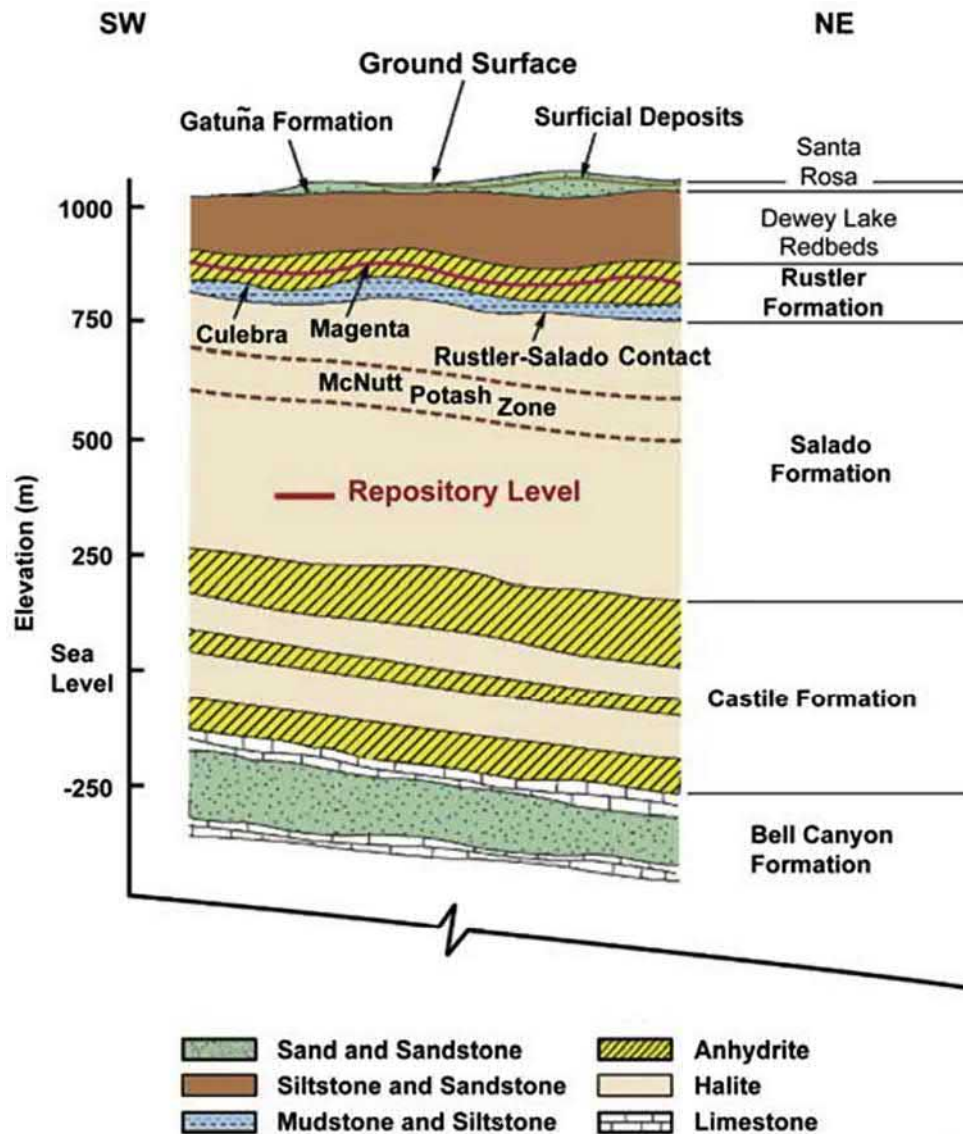


Fig. 2 WIPP geological stratigraphic sequence. U.S. Department of Energy.



Fig. 3 WIPP site architecture. U.S. Department of Energy.



Fig. 4 Disposal of contact handled TRU waste. U.S. Department of Energy.

Two handling methods are used depending on the level of radioactivity of the waste packages concerned. If the gamma emission rate is high, the waste is put in place using a remote handling system. If the rate is low, contact handling may be used (Figs. 4 and 5).

The TRU waste with the highest level of radioactivity requires remote handling. Special-purpose Horizontal Emplacement and Retrieval Equipment (HERE) is used to push packages into disposal cells excavated into the walls of a disposal cavern (Fig. 6).

The Department of Energy (DOE), which operates the facility, is licensed to dispose of up to 173,600 m³ of TRU waste from National Defence-related activities. Operation is planned to continue until 2070. The National Defence argument undoubtedly played a role in overcoming the many procedural barriers that had to be passed before the site could open. Twenty years passed (1979–99) between the initial license for WIPP being granted by Congress and the final operating license being issued by the New Mexico Environment Department.

By the end of 2005, after 6 years of operation, the site was 20% full, with no releases of radioactivity into the environment or any worker contamination. However, operation was interrupted in February 2014 following two events that took place within the space of 10 days:

- 5 February: An underground fire on the engine of a transport vehicle led to many operators being evacuated due to intoxication by smoke inhalation.
- 14 February: Radioactive contamination detected at the surface of the site near the air exhaust at levels that do not pose a risk to the public and the environment. This contamination resulted from an exothermic reaction due to chemical incompatibility between an organic absorbent material introduced into a drum and its contents, a radioactive nitrate salt.



Fig. 5 Disposal of remote handled TRU waste. U.S. Department of Energy.

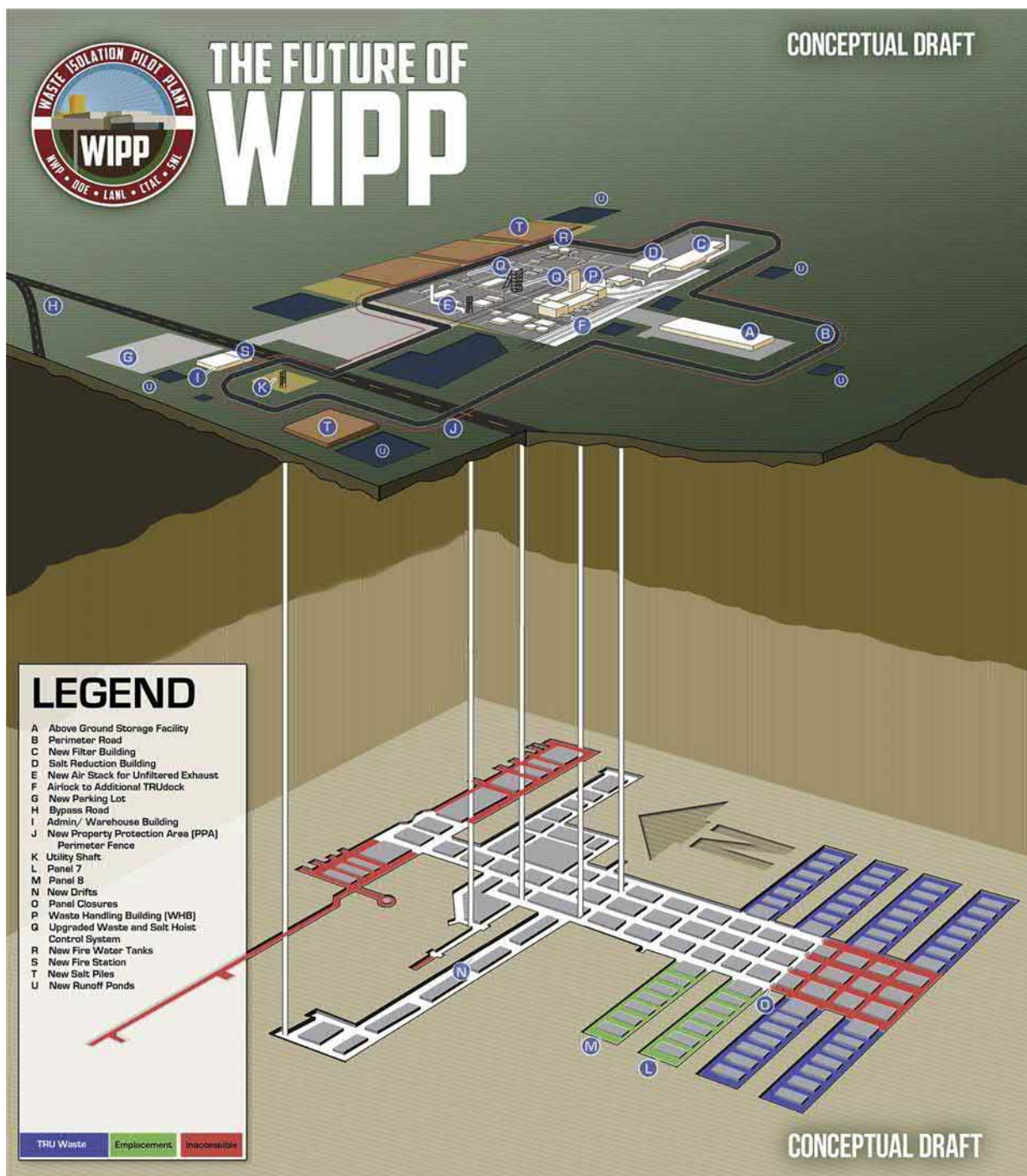


Fig. 6 Conceptual view of the WIPP architecture including the new ventilation shaft. U.S. Department of Energy.

The operations to dispose of transuranic waste, produced by defense-related activities, have gradually resumed since the end of 2016. Adopting a phased approach, the return to operation is described in the WIPP Recovery Plan, published by the DOE at the end of September 2014. In 2019, the US DOE published a four-year strategic plan for the WIPP, including the construction of a fifth ventilation shaft with a rate of air intake from the surface of 900,000 m³/h, in addition to the existing shaft which has a rate of 290,000 m³/h. The construction contract was formalized in August 2019 by the U.S. Department of Energy (DOE) Carlsbad Field Office (CBFO)—(US DOE, 2019).

The KBS-3 repository facility with completed parts of the KBS-3 repository

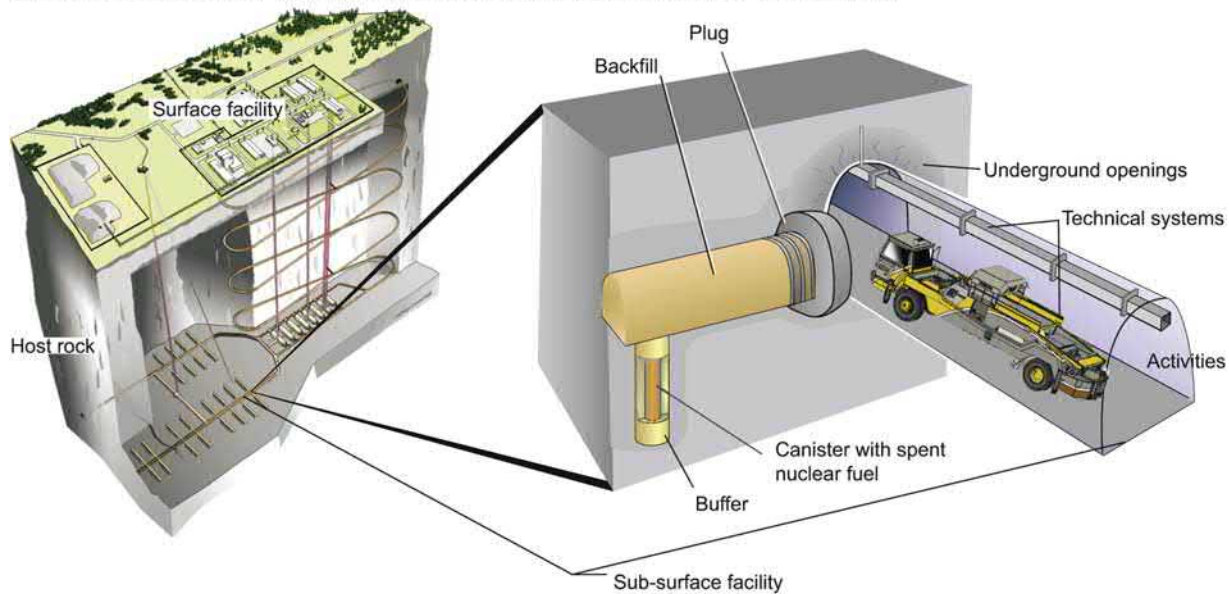


Fig. 7 KBS-3 geological repository concept for spent fuel in Forsmark. Courtesy of SKB.

Summary of the Swedish and Finnish granite repository project

Sweden

The four electricity companies that operate Swedish nuclear reactors—Vattenfall, Forsmarks Kraftgrupp AB, OKG Aktiebolag and E.ON Kärnkraft Sverige AB—created the Swedish Nuclear Fuel and Waste Management Company, SKB, to meet their current and future obligations regarding radioactive waste. SKB is responsible for the handling, transport and disposal of all spent fuel and nuclear waste produced by nuclear power plants and operates the facilities at the back end of the nuclear fuel cycle, which are concerned with the storage of spent fuel.

Principle of the repository

Since work began on the Swedish final repository project in the late 1970s, SKB has defined a number of principles for the design of the final repository. These principles constitute the safety doctrine defining the KBS-3 concept (SKB, 2015). They are summarized below.

- By building the repository deep underground in a geological environment that is stable in the long term, the waste is isolated from the human environment. This means that the repository is not strongly affected by societal changes or the direct long-term effects of climate change taking place at ground level.
- By locating the repository on a site where it can be presumed that the host rock is of no economic interest to future generations, the risk of human intrusion is reduced.
- The spent fuel is surrounded by several natural safety barriers that have been designed.
- The main safety function of the barriers is to contain the fuel in a tank.
- In the event of a containment failure, the secondary safety function of the barriers is to delay a potential release from the repository.
- The artificial barriers must be made of natural materials that are stable in the long term within the repository environment.
- The repository must be designed and constructed in such a way as to prevent temperatures that can have significant adverse effects on the long-term properties of the barriers.
- The repository must be designed and constructed in such a way as to prevent radiation-induced processes that can have significant adverse effects on the long-term behavior of the artificial barriers or the rock.
- The barriers must be passive, that is, they must function without human intervention and without any artificial input of matter or energy.

Combined with many other considerations, such as the geological context in Sweden and the technical requirements for constructing the repository, these principles led to the development of the KBS-3 system for managing spent fuel.

SKB in Sweden and POSIVA in Finland use the same repository concept, KBS-3, developed by SKB. The KBS-3 repository concept is illustrated in the figure below (Fig. 8).



Fig. 8 KBS-3 concept and closure using clay-based materials. Courtesy of SKB.

The geological repository concept for spent fuel

The principle behind disposal in Forsmark is based on placing spent fuel in copper canisters. The packages will be placed at a depth of around 500 m in the granite host rock. A bentonite (clay) engineered barrier is also planned.

The purpose of the granite rock is to isolate the waste. It provides a stable chemical environment and protection against any events that may occur at ground level. However, the granite substrate also contains groundwater that flows through fractures in the rock. If radioactive substances were to escape from a canister and enter the bentonite, they could be picked up by the surfaces and minerals in these fractures as well as by the micropores of the rock. This is the KBS-3 geological repository concept. The construction, operation and closure of the facility will take place over a period of 70 years.

The copper packages will be disposed of in a vertical disposal cell, in which the clay engineered barrier will be installed. Clay barriers are also implemented in the repository access and operation structures. The disposal cells are closed with a “plug,” disposal drifts are closed with “seals” and cavities are backfilled, etc. The backfill consisting of compacted bentonite clay blocks and granules will be installed in the disposal drifts (Fig. 8).

The repository is designed for 12,000 t of spent fuel contained in approximately 6000 copper canisters. Operation is expected to last 60 years. The repository footprint is planned to cover a 3–4 km² surface area and the disposal cells will be situated at a depth of approximately 500 m (Fig. 9).

Current interim storage of spent nuclear fuels

All spent fuel in Sweden is temporarily stored in the centralized temporary storage facility (CLAB) in Oskarshamn. CLAB is an underground pool-based storage facility that will be integrated into the future spent fuel encapsulation facility in preparation for final disposal (CLINK, future facility at the project stage). This encapsulation plant will condition the spent fuel in copper canisters. Approximately 6000 packages of spent fuel will constitute the total reference inventory of the geological repository (SKB, 2011).

Indicative planning

SKB submitted an application for a license to construct the geological spent fuel repository in 2011. In early 2018, the Radiation Safety Authority (SSM) and the Land and Environmental Court both submitted their opinions to the government. SSM recommended that the government grant permission to move forward. However, the Court requested additional information on the properties of the copper canister and its safety in the long term. In April 2019, SKB submitted additional documentation to the Ministry of the Environment concerning the construction license application for the repository.

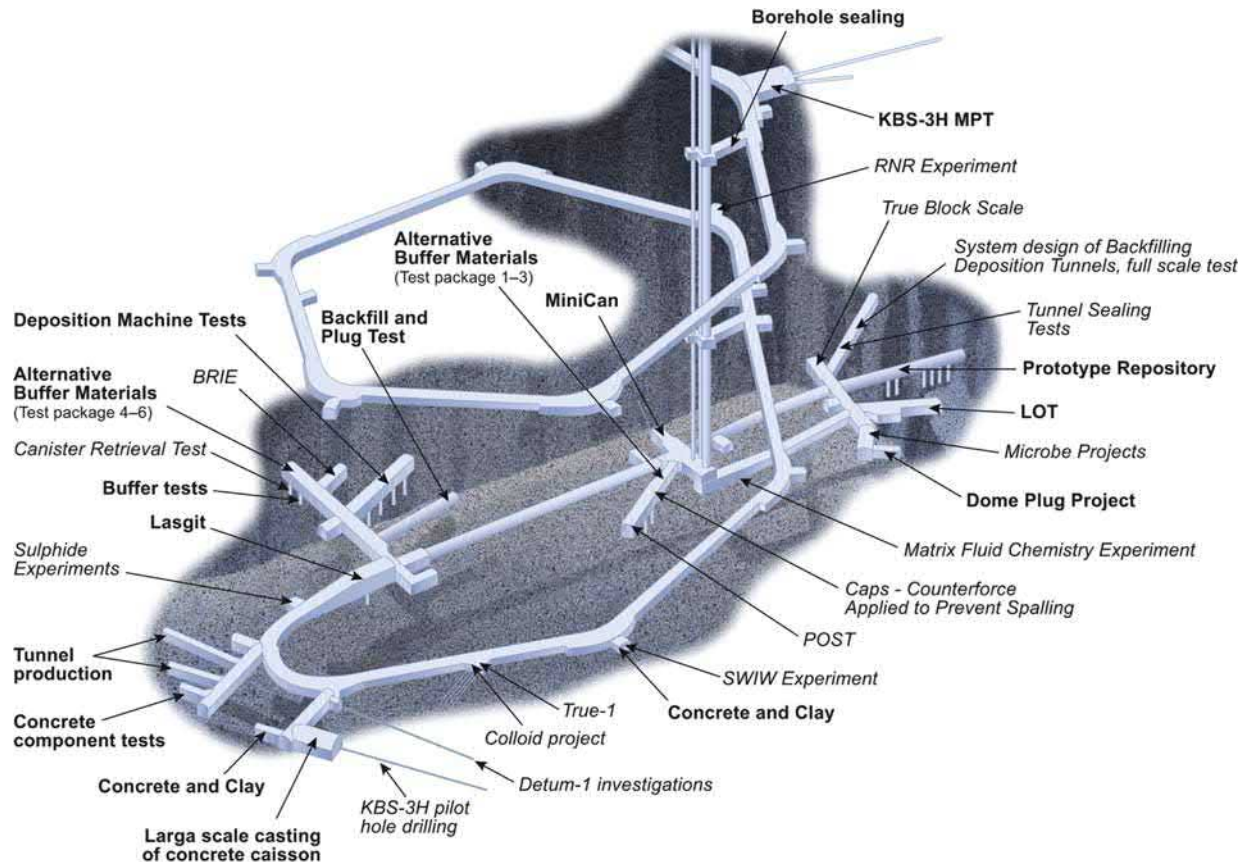


Fig. 9 View of repository site at the end of facility life. Courtesy of SKB.

At present, the Ministry of the Environment is continuing to prepare the government's decision and therefore the process of granting a license for the final disposal system for spent fuel in Forsmark. The repository is expected to be commissioned in 2035, following a pilot phase starting in 2030 (SKB, 2019).

Once fully developed—in the 2080s—the repository site will comprise 66 km of tunnels and disposal cells to hold more than 6000 copper canisters containing spent nuclear fuel. At the end of facility life, the repository will have required the extraction of around 2.3 million cubic meters of rock.

On-going R&D programs

Following a long process that began in the 1980s, SKB has carried out geological research involving, among other things, borehole campaigns to characterize the rock. The process of searching for and selecting a site began in 1992. Field investigations began in 2002 and lasted 5 years. The Forsmark site was selected in 2009 as the location for the geological spent fuel repository in a 1.9 billion-year-old granite formation.

In parallel to the site selection, SKB developed a dedicated underground research laboratory, the Äspö Hard Rock Laboratory, north of Oskarshamn, around 460 m below the ground. This work aims to investigate the behavior of the various barriers in the future repository: canister, engineered barrier, and rock; and to test different pieces of equipment for handling the packages. It should be noted that the Äspö HRL is not located near the Forsmark site. The main objective was to provide an opportunity for research, development and demonstration in a realistic and undisturbed rocky environment up to the depth planned for the future final repository.

The Äspö laboratory was constructed between 1990 and 1995. It has been in operation since construction was completed (Fig. 10).

SKB's RD&D programme includes full-time activities at the Äspö laboratory planned until the end of 2023 (SKB, 2019):

- Around 15 experiments are underway,
- Several new large-scale tests are planned.

The purpose of the Äspö HRL continues to be to develop the methodology used to characterize the geological environment that will be used for the repository in Forsmark, the modeling of flows, the behavior of the various barriers, and the validation of digital models.

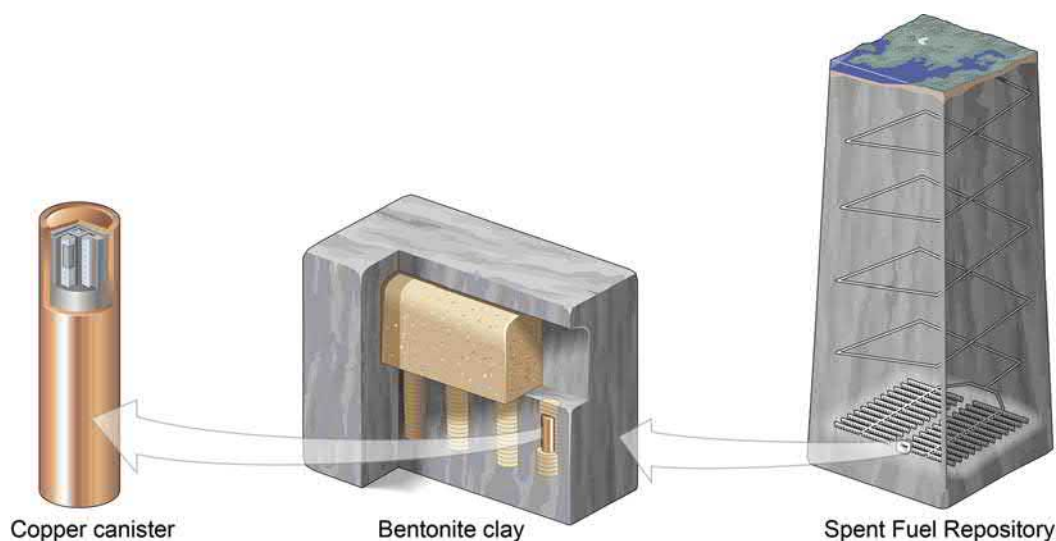


Fig. 10 Diagram of the Aspo HRL laboratory. Courtesy of SKB.

Finland

The Finnish nuclear waste management programme was launched in 1983, shortly after the four reactors entered commercial operation. The Nuclear Energy Act of 1987 selected the option of disposal and set up the Nuclear Waste Management Fund under the supervision of the Ministry of Trade and Industry. There are no nuclear fuel cycle facilities in Finland. Given that in Finnish legislation spent fuel is classed as radioactive waste, the Loviisa and Olkiluoto nuclear power plants are the main producers of radioactive waste.

Posiva Oy was founded in 1995 as a joint venture between TVO and Fortum, the owners of the Loviisa and Olkiluoto power plants. Responsibility for nuclear waste lies with the electricity companies until it is finally eliminated. Most of the radioactive waste and spent nuclear fuel was produced in the nuclear power plants currently in operation in Olkiluoto (OL1 and OL2) and Loviisa (LO1 and LO2). Spent fuel is currently stored in pools in storage facilities located on the power plant sites. Small quantities of irradiated fuel were produced when the research reactor FİR 1 was in operation in Otaniemi.

The spent fuel to be produced during the service life of units OL1 and OL2 at the Olkiluoto power plant, units LO1 and LO2 at the Loviisa power plant and, finally, of unit OL3 (EPR) currently under construction is estimated at 5500 t of uranium (tU), constituting the reference inventory of the Finnish geological repository.

The Finnish repository concept is the same as that adopted by Sweden, namely KBS-3. It consists of drifts excavated in granite at the Olkiluoto site. In these drifts, vertical (KBS-3V, the reference) or horizontal (KBS-3H) cells are constructed and bentonite blocks are installed, leaving space in the center for the disposal package to be inserted. The disposal package is a sealed copper canister with the spent fuel elements inside (Fig. 11).

Finland began the process of searching for a geological repository site for high-level waste in the late 70s, which included the following major steps:

- the study of potential sites with over 100 pre-selected areas in a granite environment;
- following a complex selection process, five then four areas were characterized in a preliminary manner and then in more detail;
- one site, the Olkiluoto site, was put forward for the construction of a geological repository in 1999, based on the conceptual architecture below (Fig. 12).

Research laboratory

In 2004, Posiva laid the foundation stone by building the research center ONKALO on the future repository site. The purpose of ONKALO was to organize underground research work in a site suitable for the disposal of irradiated nuclear fuel and to provide data on the Olkiluoto rocky substrate. From the start, ONKALO was intended to become part of the future repository. In the future, some parts of ONKALO will continue to have the same requirements as regards research as the former research facility. This site is situated in a granite gneiss-migmatite geological formation, which is approximately 1.8 billion years old.

The ONKALO access tunnel reaches a depth of 455 m. ONKALO has one access tunnel and three shafts: one shaft for personnel and two shafts for ventilation. The slope of the tunnel is 1:10. It is 5.5 m wide and 6.3 m high (Fig. 13).

The research conducted at ONKALO provides additional data on the conditions of the rock substrate and groundwater at the repository site, as well as on the impact of construction on these conditions. This research helps to ensure that the Olkiluoto rock mass is suitable for the repository and to identify areas in which the construction of disposal drifts can be optimized. Several categories of studies and experiments are conducted at ONKALO (Posiva, 2003):

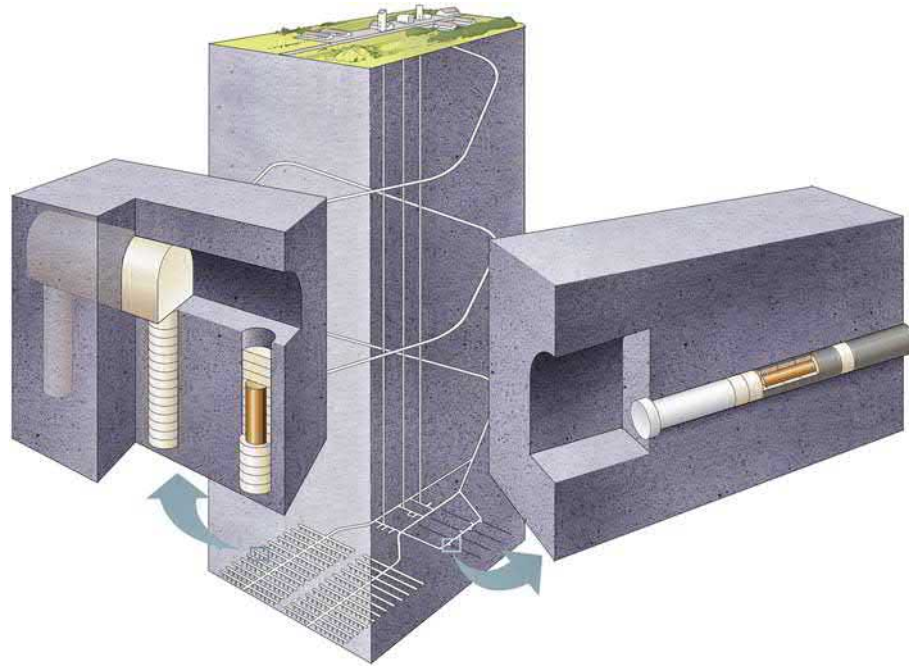


Fig. 11 Posiva repository strategy based on KBS-3 concept. Courtesy of SKB.

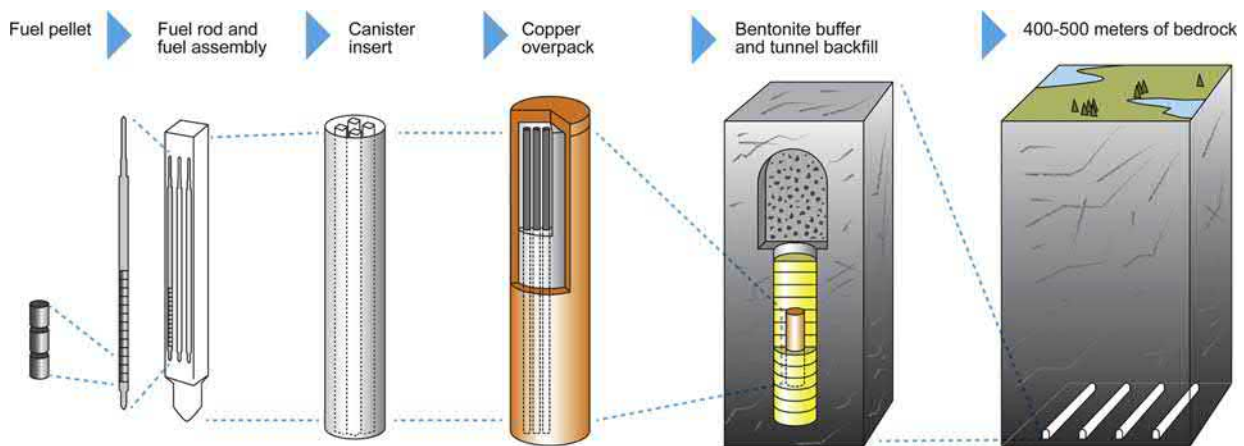


Fig. 12 Posiva deep geological repository system based on the KBS-3 concept. Courtesy of Posiva.

- Geological mapping is used to collect data in order to adjust geological models as well as by experts who design reinforcement and grouting measures. Based on this mapping, forecasts are also drawn up concerning the characteristics of the rock at greater depths.
- Probe holes are bored at ONKALO to gather information on the rock to be excavated;
- Characterization niches are also constructed at ONKALO. The first characterization niche was built 1475 m downstream of the tunnel. In total, there are five characterization niches at ONKALO.
- The volume and characteristics of groundwater are studied. Groundwater studies form part of monitoring the consequences of construction.
- Rock mechanics studies serve to determine the conditions prevailing in the rock, such as the direction and intensity of the stress field as well as the strength of the rock. In order to determine the mechanical characteristics of the rock, various types of measurements are carried out before excavation begins in the area in question. The methods can be used both in drillholes and on the excavated surface.

These studies were used to draw up the construction license application for the geological spent fuel repository. Studies are currently being conducted to optimize the design and will be used to support the operating license application file.

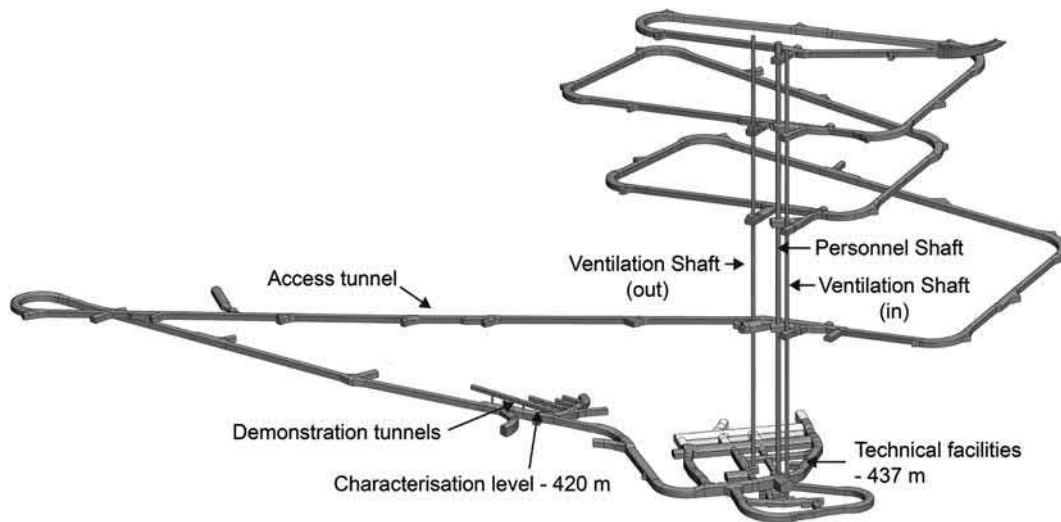


Fig. 13 Diagram of the ONKALO laboratory in Olkiluoto. Courtesy of Posiva.

The repository safety concept

Posiva's safety concept is based on the KBS-3 concept, taking into account the properties of the fuel as well as the characteristics of the Olkiluoto site (Posiva, 2012).

Because of the long-term risks it poses, spent fuel must be isolated from the surface environment for an extended period of time. Safe disposal is therefore achieved by isolating and containing the fuel in the very long term. The characteristics that favor long-term isolation and containment include (Fig. 14):

- the copper container encapsulating the spent fuel,
- the bentonite engineered barrier installed inside the disposal cells. It is composed of bentonite blocks put in place before the package is inserted.

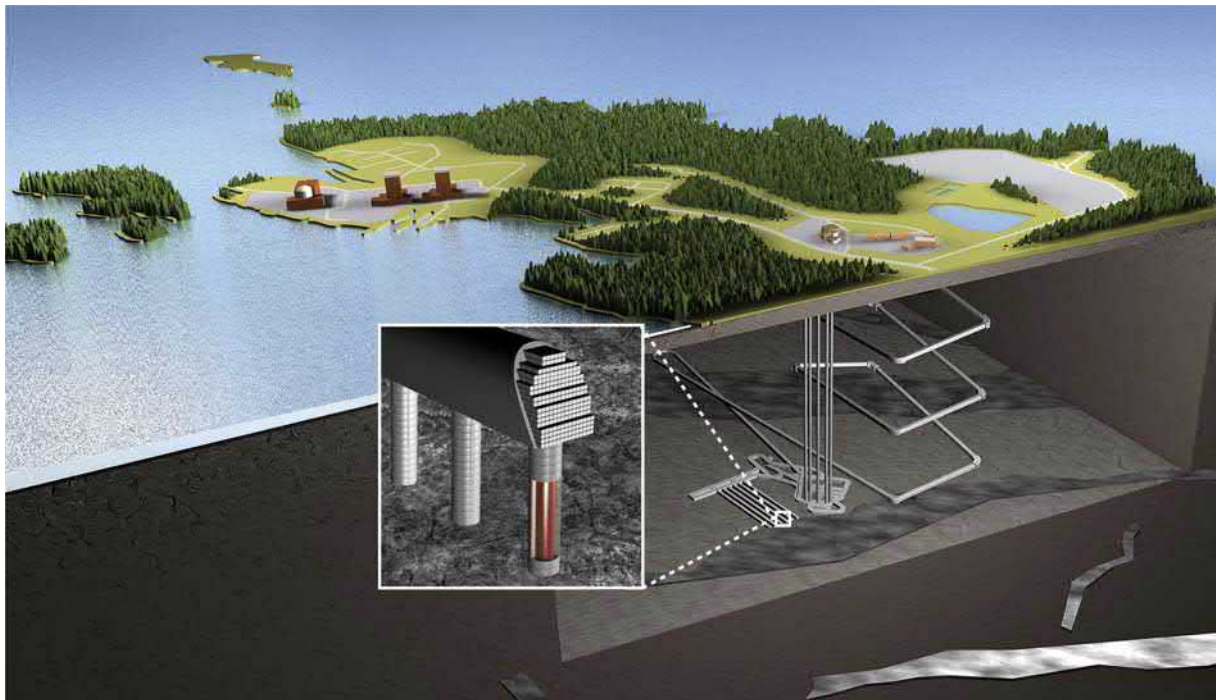


Fig. 14 Diagram of the different scales repository-cell-package. Courtesy of Posiva.

The disposal drifts are backfilled with bentonite blocks and the ends of the tunnels are sealed with a specific seal or plug. The function of all of these barriers is to protect the packages and to isolate them hydraulically.

Repository architecture

The disposal drifts are located at a depth of around 400–450 m inside the Olkiluoto granite massif. They will be excavated in the rock and the disposal packages will then be placed in the drilled holes.

The underground site is divided into three parts:

- the disposal drifts where packages containing used fuel will be placed;
- the central tunnels connecting the disposal drifts, the access tunnel and shafts;
- the underground technical auxiliary facilities.

In accordance with the reference inventory adopted, representing approximately 2800 packages to be disposed of, the volume of rock to be extracted for the facility, excluding the waste package disposal cells, is approximately 1.3 million m³. 137 disposal tunnels are planned. The total length of the tunnels has been estimated at 42 km and they are located in a 2–3 km² area.

An access tunnel and four vertical shafts lead from the surface to the repository. The vertical shafts include one shaft for personnel, one shaft for lowering packages and two ventilation shafts. The access tunnel, the personnel shaft and two ventilation shafts were constructed when the ONKALO research laboratory was built. Fig. 15 shows a summary image of the whole repository.

The construction license application for the repository was submitted in 2012. The construction license was granted in November 2015. This came with a set of requirements that Posiva must meet before the repository can begin operation. The first repository excavation works began in 2017.

A second application to the decision-making authorities is required: the operating license application. It will be submitted in 2020. The geological repository is expected to begin operating around 2024. According to current plans, the disposal site will close around 2120.

Summary of the French repository project in clay-rich rock

In France, HLW is mainly produced through the reprocessing of spent fuel from the 19 nuclear power plants (58 existing reactors and the Flamanville EPR under construction). France has chosen to reprocess spent fuel. Ultimate high level waste (i.e., fission products solutions) is conditioned thanks to a specific vitrification process (see dedicated chapter) into stainless steel containers. ILW-LL is more varied. It includes structural components that surround fuel assemblies and residue from the operation and maintenance of nuclear facilities. It is conditioned in metal or concrete containers (Figs. 16–22).

In 1991, the French parliament began to address the problem of managing radioactive waste by passing the “Bataille Waste Management Act” on 30 December 1991. This Act defined the main priorities for research into the management of the most highly radioactive waste. Three research areas were identified: partitioning and transmutation, long-term storage (the CEA was given responsibility for both of these areas) and deep geological disposal, entrusted to Andra, which has conducted research thanks to its underground research laboratory. Andra and the CEA submitted the results of 15 years of research in these areas to the French

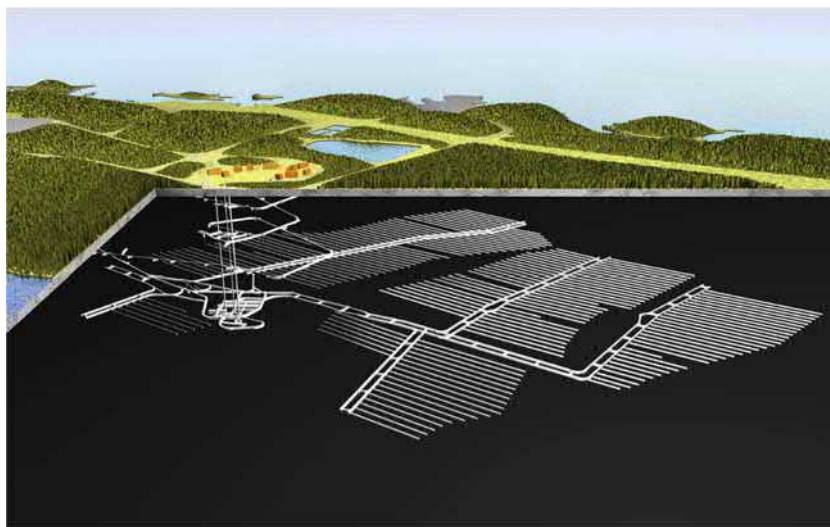


Fig. 15 Illustration of the whole Olkiluoto repository. Courtesy of Posiva.



Fig. 16 Basic illustration of the CIGEO deep geological repository project. Courtesy of ANDRA.

government in 2005. Based on the outcomes of these extended researches, the French Nuclear Safety Authority (ASN) notably concluded that deep geological disposal seems to be the only definitive management solution possible.

In 2006, on the basis of the scientific results, their review by the ASN and a public debate conducted in 2005, the French parliament ratified the choice of deep geological disposal and tasked Andra with designing a disposal facility in the Meuse and Haute-Marne *départements*. Research is ongoing into partitioning and transmutation as well as storage as complementary waste management options to disposal.

The Industrial Centre for Geological Disposal (Cigeo) project is designed to handle all the HLW and ILW-LL that has been produced and that will be produced by existing nuclear facilities before they are dismantled (nuclear power plants, research centers, etc.). Waste that will be produced by nuclear facilities currently being built (Flamanville EPR, ITER and the Jules Horowitz experimental reactor) has also been taken into account. This represents around 10,000 m³ of HLW and 75,000 m³ of ILW-LL, i.e., around 85,000 m³ of radioactive waste in total. This waste is currently stored temporarily in surface facilities at the sites where it was produced (mainly La Hague, Marcoule and Cadarache), pending its disposal in Cigeo (Fig. 16).

Research into geological disposal, entrusted to Andra by the 1991 Act, draws on various disciplines and covers different aspects, beginning with knowledge of the waste and the rock, as well as the design of the repository, monitoring and the transmission of memory. The gradual process of siting the geological repository will need to be supported by in-depth knowledge (geometry, structure, stability, homogeneity, continuity, etc.) of the geological formations. This characterization of the geological environment is based on regional studies, borehole campaigns, geophysical measurements and observations made in the Underground Research Laboratory.

Underground Research Laboratory

In 1994, investigations were carried out at four candidate sites (in Gard, Vienne, Meuse and Haute-Marne) for the purpose of setting up an underground research laboratory for studying the feasibility of deep geological disposal. The preliminary studies showed that the geology of the Meuse and Haute-Marne sites, now merged into a single site, was particularly suitable:

- The Callovo-Oxfordian formation has a very low permeability. Circulation of water is very limited; a molecule of water travels only a few centimeters in 100,000 years.
- Because of this very low permeability, the radioactive elements move preferentially by diffusion through the rock and this diffusion is very slow.
- The rock has a very high capacity of retention for most radioactive elements.
- The rock contains a significant amount of clay minerals including smectite which fixes elements (and therefore traps the radioactive elements).
- Most of the radioactive elements are not very soluble in the water in the rock (due to the chemical composition of that water).

In 2000, the construction of the Underground Research Laboratory began on this site. Situated at a depth of 490 m and consisting of a network of almost two kilometers of drifts as to the end of 2020, this research facility has been used for scientific and technological research carried out directly within the Callovo-Oxfordian argillite stratum and, in 2005, led to the conclusion that deep geological disposal was feasible (Andra, 2006). The Laboratory is still used for research and experiments for designing Cigeo. Since 2004,

experiments have been conducted to collect geomechanical, geochemical and thermal data. These different measurements required instrumenting the rock that is, carrying out more than 1000 drilling campaigns directly from the experimental drifts, and installing more than 12,000 measurement sensors in the drifts, shafts and at the surface (Andra, 2019) (Fig. 17).

Until 2006, these experiments focused on:

- characterizing the water contained in the rock;
- the mechanical behavior of the rock after excavation;
- the thermal behavior of the rock following heating (which would be induced by HLW packages within the argillite);
- the diffusion and retention properties of the radioactive elements (Fig. 18)

Based on this work in particular, in 2005, a 250-km area around the underground laboratory, called the “transposition zone,” was defined as having identical geology to that of the Laboratory: the argillite formation is stable and its properties will ensure the containment of the radioactivity in the very long term.

Following the Act of 2006 entrusting Andra with the task of designing and implementing a repository, the Meuse/Haute-Marne Centre, which houses the Underground Research Laboratory, entered the industrialization phase of the Cigeo project. New technical and scientific experiments have been set up, focusing on:

- studying changes in the behavior of the rock after excavation;
- testing new excavation techniques (road header since 2009 and partial face excavation machine in 2013);
- excavation tests for HLW disposal cells;
- the behavior of materials within the rock (glass, steel, concrete, etc.);
- the interactions between the materials and the rock.

The scientific experiments that began prior to 2006 are continuing and becoming more complex (thermal tests with three spaced out heating resistors to simulate HLW disposal cells, diffusion tests using radioactive tracers in the argillite, etc.).

In 2009, Andra proposed to the French government a 30 km² underground area inside the transposition zone: the zone of interest for detailed survey (ZIRA). This zone was defined on the basis of both scientific criteria related to the safety and geology of the site and criteria identified by elected representatives and the local population during a consultation. The ZIRA was approved by the government after assessment were issued by the ASN and the National Assessment Board, and after consulting elected officials and the Local Information and Oversight Committee (CLIS) for the Underground Research Laboratory. If Cigeo is licensed, the underground facility will be built in this zone.

The Cigeo concept

Cigeo is designed to remain safe during construction and operation, which will last for around a century, and after its closure (Andra, 2016a,b). The safety of Cigeo relies to a large extent on the geological formation within which the underground facilities will be built. This formation, which has been stable for more than 168 million years, has containment properties that can slow down the transfer of radionuclides contained in the radioactive waste to the surface. Its safety also relies on design choices such as:

- the overall architecture of the repository: the separation of nuclear areas from work areas, for example;
- the facilities and structures: the methods used for excavating and lining the drifts and disposal cells, for example;
- the materials used: non-flammable materials and substances, for example;
- the requirements concerning package characteristics and inspections;
- the instruments and sensors used to monitor changes in the repository as well as to detect any malfunctions;
- the organization system that will be put in place for repository operation: the use of automated and remote-controlled equipment and machinery, for example.

As shown in Fig. 19, the Cigeo project will include two surface facilities: the dispatch zone where waste packages will be received, inspected and conditioned, and the shaft zone used for support for the underground activities and construction works. A double waste package transfer ramp (12% slope) and five shafts for the movement of personnel and transfer of equipment connect the surface and underground facilities. Finally, the underground disposal area will consist of two areas:

- one for the disposal of ILW, with around 20 disposal drifts, 500 m in length (Fig. 20),
- the other for HLW packages, with around 900 disposal cells, 150 m in length (Fig. 21).

A small disposal area set up for an industrial pilot phase will contain 20 disposal cells for slightly exothermic HLW, each 80 m in length.

A reversible repository

In France, ethics with respect to future generations means that action is needed. In particular, given the century-long service life planned for the geological repository, France considers that it is the responsibility of our generation to design a safe facility that we will pass on to future generations, which they are able to modify or improve according to their own objectives and constraints,

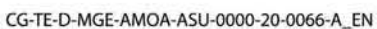


Fig. 17 3D view of Andra's Underground Research Laboratory and its 1.8 km of drifts. Courtesy of ANDRA.



Fig. 18 Picture of an underground drift at the Underground Research Laboratory. Credits ANDRA—A. Daste.

or indeed replace with other management facilities, if other options become available, particularly as a result of technical progress. In practice, reversibility is based on governance systems and technical project management tools:

- Governance systems: Continuous improvement of knowledge regarding radioactive waste management, transparency and transmission of information and knowledge, monitoring by the ASN, the involvement of society, and assessment and supervision by Parliament.
- Project management systems: Incremental development and gradual construction of the Cigeo facilities, flexibility offered by their operation, adaptability of the facilities and recoverability of the packages.

Combined with this principle of reversibility, the Cigeo repository incorporates into its architecture the provisions and tools to ensure that waste packages can be recovered, a technological concept complementary to reversibility (Andra, 2016c).

The initial construction of Cigeo will begin when the construction license decree is granted, for which Andra intends to submit the construction license application file to the ASN in 2021. If the license is granted, construction will include:

- the construction of the structures necessary for the disposal facility to be commissioned, so that the first radioactive waste packages can be disposed of, in particular the nuclear building where the waste packages will be received, inspected, prepared and from which the packages will be lowered;
- the excavation and construction of the ramp (sloping tunnel to transport the waste packages to the underground facility);
- the excavation and construction of shafts (vertical shafts connecting the underground facility to the surface and used for the movement of personnel and transfer of equipment as well as for ventilation).
- the excavation and construction of the first underground structures.

In 2030, Cigeo is set to enter an industrial pilot phase (lasting around 5 years), the main objectives of which are to ensure that Cigeo is properly managed and to verify that it functions correctly; to test the results and concepts developed through R&D work, experiments conducted at the surface and in the Underground Research Laboratory and technological tests on a full scale; to acquire, under operational conditions and on a full scale, all the feedback required to make adjustments for the subsequent phases of construction; and, finally, to identify any necessary additions and works that remain to be carried out and thereby inform subsequent decisions.

At the end of this pilot phase, the repository will operate for more than 100 years and will be built progressively as radioactive waste packages arrive. The total surface footprint of the underground repository will cover 15 km².

The first structures to be built will be the facilities needed to conduct the excavation work and start operations at the repository. Once operation of the repository has begun, the underground facility will be constructed as needs arise and in successive phases. The areas under construction will be physically isolated from operating areas. In the same way, the ventilation shafts for the package transfer drifts will be separated from the ventilation shafts for the working drifts (Fig. 22).

In 2020, Andra submitted an application for the declaration of public utility for the Cigeo project to the government in order to reaffirm the public utility of the project. This application is required for all other regulatory procedures, and specifically for operations prior to the construction of the repository. It also makes it possible to finalize the acquisition of plots of land at the surface. This application is accompanied by an overall environmental analysis for the project.

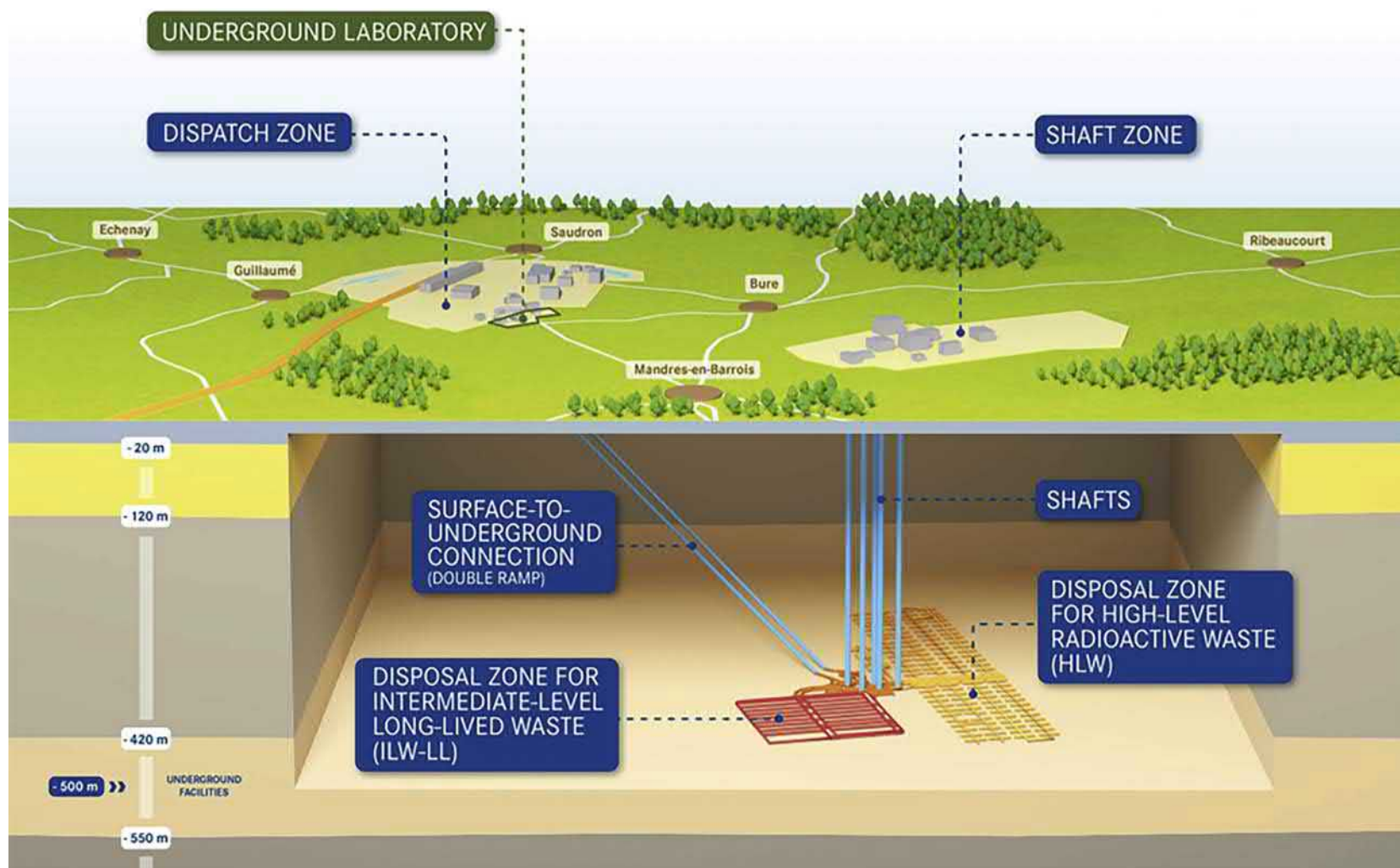


Fig. 19 Diagram of the Cigeo geological repository concept. Courtesy of ANDRA.

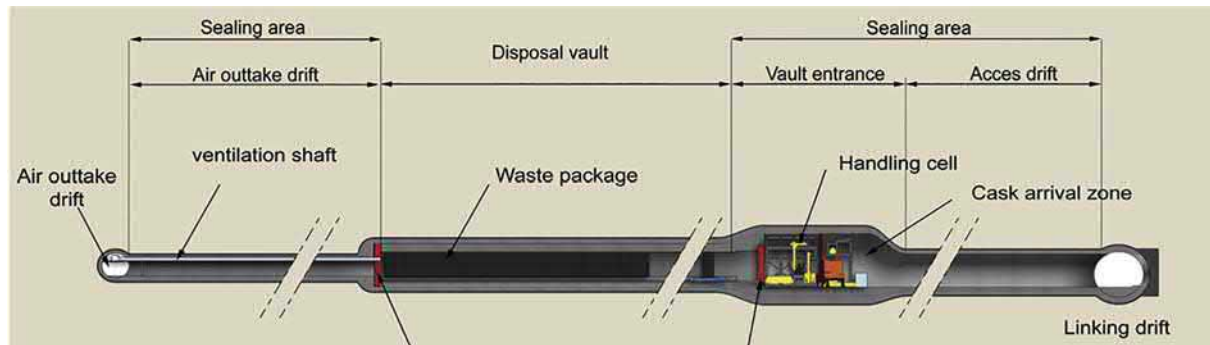


Fig. 20 Typical arrangement of an ILW disposal cell. Courtesy of ANDRA.

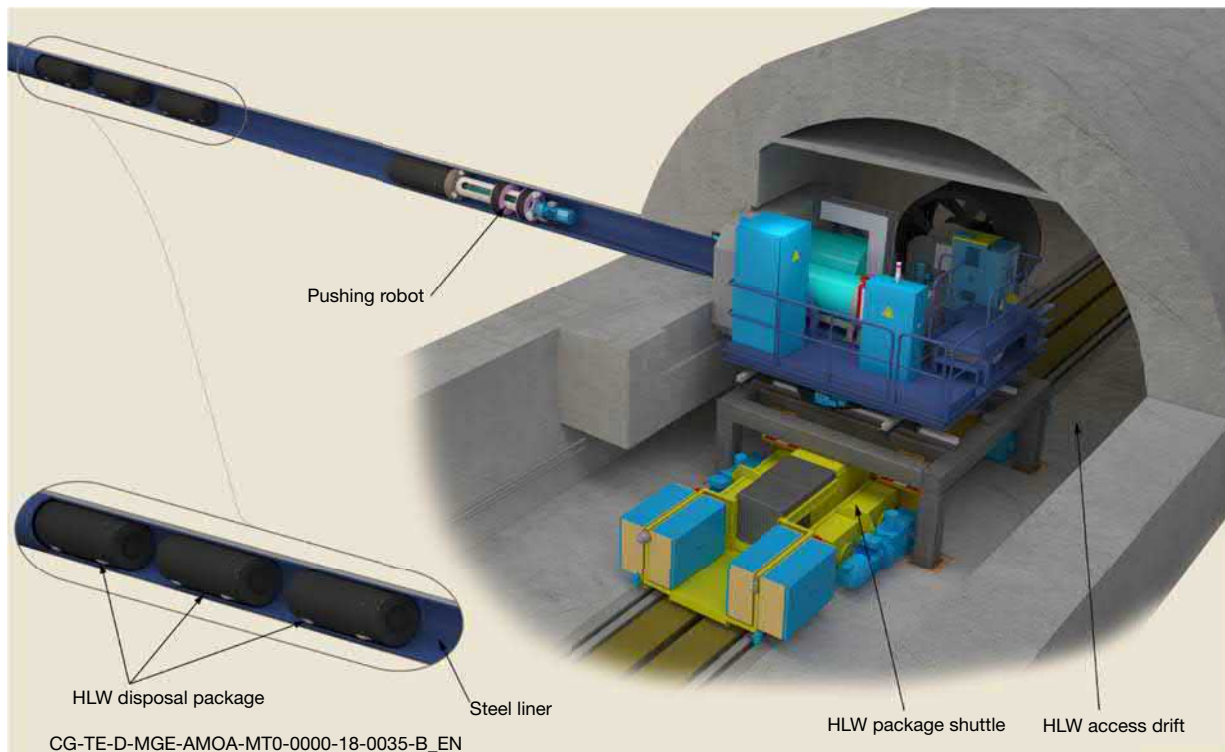


Fig. 21 Typical arrangement of HLW disposal micro-tunnels. Courtesy of ANDRA.

Summary of US repository project in volcanic tuff rock

Prior to 1987, and as required by the 1982 Nuclear Waste Policy Act (NWPA), the U.S. Department of Energy (DOE) selected 10 locations in six states to consider as potential disposal sites for 70,000 t of heavy metals (tHMs) of radioactive waste, including 63,000 tHMs of spent fuel of civil nuclear origin, 2300 tHMs of military origin (managed by DOE) and finally 4700 tHMs of vitrified HLW. This inventory corresponds to approximately 11,500 disposal packages. After detailed studies of the sites, President Ronald Reagan approved three sites for detailed characterization. The three sites were Hanford, Washington; Deaf Smith County, Texas; and Yucca Mountain, Nevada. In 1987, Congress amended the NWPA and ordered the DOE to study only Yucca Mountain. The act provided that if Yucca Mountain were found to be inappropriate, "site characterization studies" would be terminated.

Yucca Mountain is a volcanic ridge with a prominence of nearly 400 m and extending on a north-south axis in Nye County, Nevada, 150 km north-west of Las Vegas on federally controlled lands on the edge of the Nevada Test Site. Yucca Mountain consists of tuff, a rock made from compact volcanic ash formed more than 13 million years ago (Fig. 23).

Yucca Mountain has a desert climate and receives a mean of 125–200 mm of rain per year, depending on elevation a topography. The precipitation in the area presents two seasonal input of meteoric water. In winter, the precipitation is of low

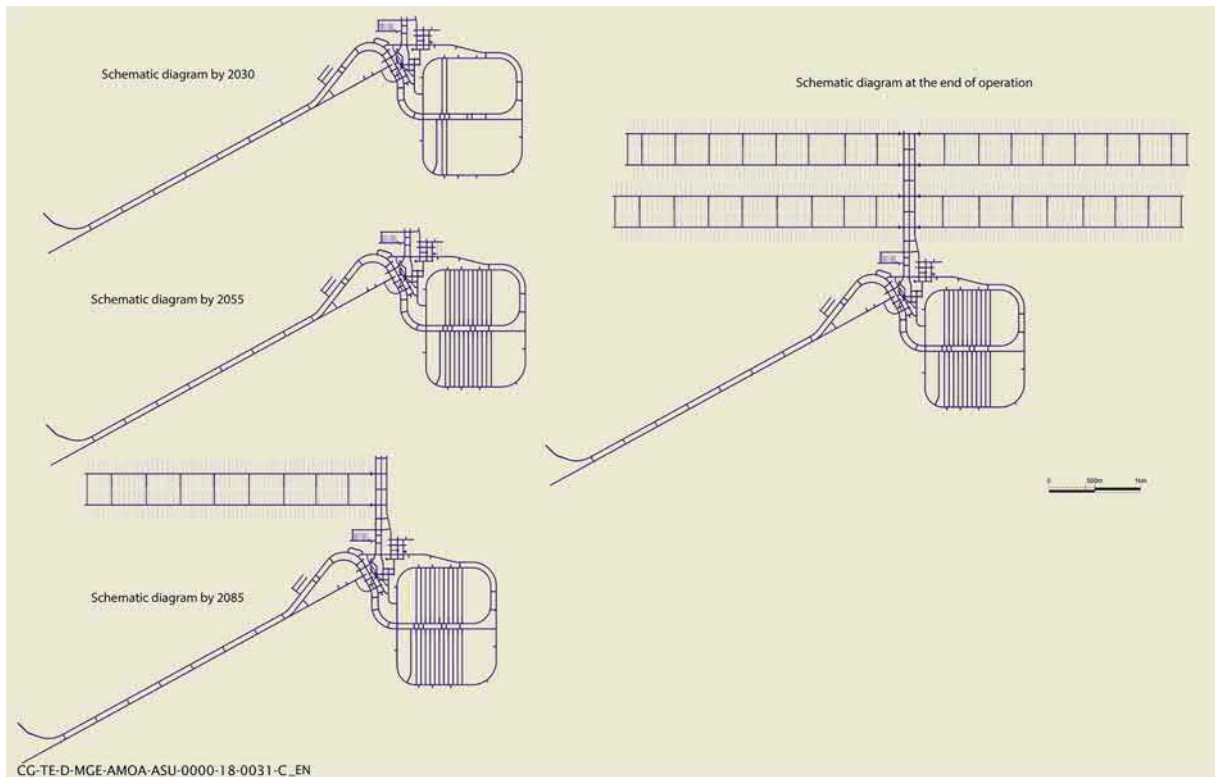


Fig. 22 Illustration of the progressive and incremental deployment of the CIGEO facility from 2030 to end of operation. Courtesy of ANDRA.



Fig. 23 Aerial north view of Yucca Mountain. U.S. Department of Energy.

intensity and long duration. In summer, rainstorms of high intensity and short duration tend to occur (USGS, 2004). The mountain has a deep groundwater table. The repository would be constructed about 300 m below the earth's surface and 300 m above the groundwater table. Despite the arid climate, unsaturated rocks at Yucca Mountain are not dry: hydrogeological analysis on the tuff media have shown porosities ranging from 10% to 40%, with residual water saturations ranging from 2% to 42%. At the repository level, 90% of modeled water flux occurs in fractures and 10% occurs in the rock matrix (Birkholzer et al., 2006). Within the lower-porosity welded tuffs, performance assessment models show an important capillary diversion of water around the openings in the unsaturated rock and the depth of the groundwater under the repository (Figs. 24 and 25).

Due to these conditions, the unique Yucca Mountain repository concept relies only on the engineered barrier consisting of a primary radioactive waste container made of stainless steel (UoS-DoE, Office of Civilian Radioactive Waste Management, 2008). The radioactive waste packages are emplaced in shields—to divert potential drips and seeps of water—made of a corrosion-resistant titanium metal that would have been placed over the waste packages. The repository design has been defined without backfill allowing ventilation for an extended period providing an effective way to remove radioactive decay heat from the repository and associated heated water vapor into the air fed by the infiltrated surface water (Rechard and Voegelé, 2014).

The measures envisaged to maintain operational safety and to facilitate the potential recovery of packages before closure include the installation of a concrete liner that is 5.5 m in diameter (Fig. 24), or alternatively a steel liner in the event pH changes caused by concrete degradation prove significant.

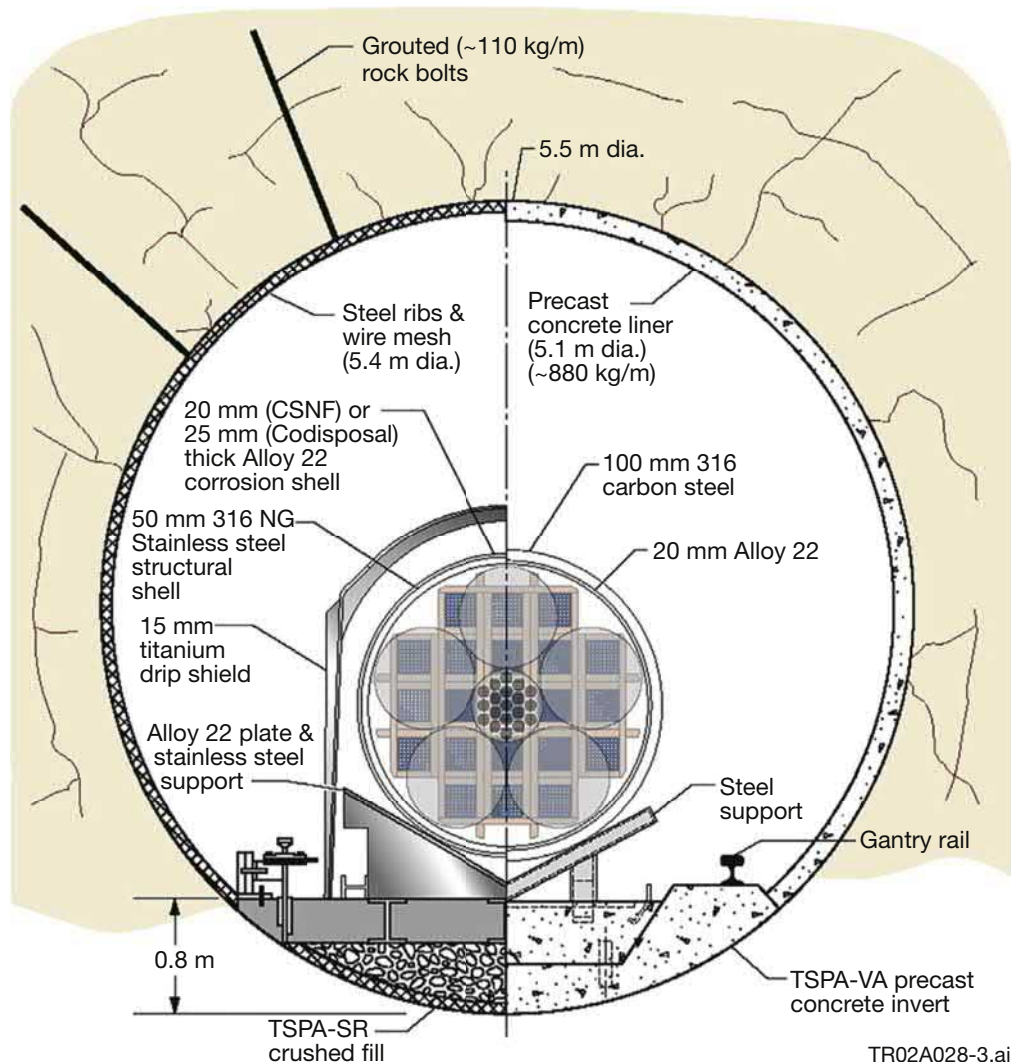


Fig. 24 Illustration of the concept of disposal in drifts. Rechard R and Voegelé M (2014) Evolution of repository and waste package designs for Yucca Mountain disposal system for spent nuclear fuel and high-level radioactive waste. *Reliability Engineering & System Safety Journal* 122: 53–73. (2014), with permission.

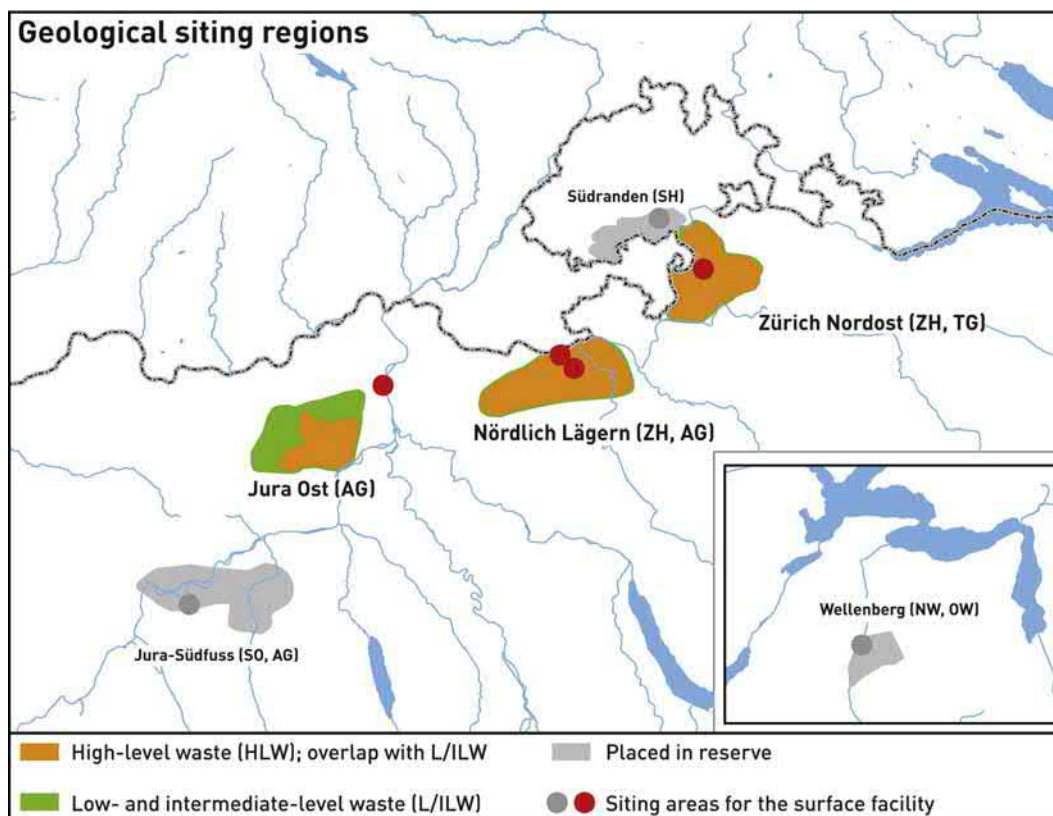


Fig. 25 Geological siting regions. Reproduced with permission of Nagra (2018). *Annual report 2017*. Nagra, Wettingen, Switzerland.

In 1995, a U-shaped tunnel, 7.5 m in diameter and 8 km in length, was dug connecting the surface access portals (North and South). The tunnel is an exploratory shaft facility (ESF) and extends 4 km to the depth of the planned repository (**Fig. 25**).

In 2008, the DOE applied to the Nuclear Regulatory Commission for a permit to build a geological repository at Yucca Mountain.

In February 2009, President Barack Obama's administration, which declared that it was opposed to disposing of spent fuels in Nevada, announced, "Yucca Mountain is not an option for the long term." Funding for the project has been terminated and the Blue Ribbon Commission on America's Nuclear Future has been appointed "to conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle and recommend a new strategy." The commission concluded there are compelling reasons to move as quickly as possible to develop the disposal, storage, processing and transport of spent fuel and radioactive waste from both defense and civilian sectors. It also proposed important structural developments on the financing, organization and approach needed to conduct these programs. In 2010, the DOE withdrew its construction application filed with the NRC.

Alternative concepts have then been studied, such as the deep borehole disposal concept, which consists of drilling a borehole into crystalline rock to a depth of approximately 5000 m. Canisters containing used nuclear fuel or vitrified radioactive waste from reprocessing would be emplaced within the lower 2000 m of the borehole, and the upper 3000 m of the borehole are sealed. Research and development activities carried out at Sandia National Laboratories from 2009–12 have built the technical baseline for performance of a deep borehole for disposal of nuclear waste ([Sandia National Laboratories, 2012](#)).

In 2013, the DOE announced a new strategy for managing spent nuclear fuel and HLW. This included centralized waste storage, then mixing commercial and defense waste into a single repository—which would be operational by 2048—at a site other than Yucca Mountain. The Blue Ribbon Commission recommended the establishment of an independent agency to manage radioactive waste, the Nuclear Waste Administration. It would be entrusted with the responsibilities and missions currently carried out by the DOE in the area of waste, including defense waste.

In 2015, President Obama indicated that a separate repository for defense-related radioactive waste was needed. The DOE stated that defense waste is smaller in volume, less radioactive, and thermally colder than commercial irradiated nuclear fuel, indicating that a defense facility could be easier to develop. Following this announcement, the DOE declared its intention to build two disposal facilities, one for most of the country's defense-related radioactive waste and the other for commercial irradiated nuclear fuel and residual defense waste.

In 2016, the National Defence Authorization Act for Fiscal Year 2017 denied funds for a repository solely for defense waste.

In 2018, President Trump's budget request for fiscal year 2019 included \$120 million for the resumption of the review of the Yucca Mountain repository construction permit application and associated surface storage of the radioactive waste inventory concerned. As of February 2019, the Congress had not allocated funding for the resumption of the permit application.

As to the end of 2020, the waste management policy in the USA is currently in a dead end. Yucca Mountain project did not resume up to now and no alternative projects emerged.

Overview of other projects: Switzerland, Belgium, Japan and Russia

Switzerland

The reference long-term solution for all radioactive waste in Switzerland is deep geological disposal. Since the 1980s, scientific research on geological disposal has been conducted on two host rocks: clay and granite. Two underground laboratories were built:

- from 1983, in Grimsel, in the granitic massif next to the Oberhasli hydroelectric power plant;
- and since 1996, at Mont Terri, in the layer of clay crossed by the motorway tunnel of the same name.

These are "methodological" laboratories that serve to expand knowledge about some of the scientific and technical aspects related to the geological disposal of high and intermediate-level long-lived waste (HLW/ILW-LL).

For spent fuel, vitrified HLW and ILW-LL, Nagra, the National Cooperative for the Disposal of Radioactive Waste in Switzerland, has demonstrated the feasibility of clay disposal at Opalinus in the Zürcher Weinland. The reports documenting the study were submitted to federal authorities in 2002 (Nagra, 2002).

In 2005, Nagra concluded that repositories in the crystalline subsoil were feasible and that long-term safety was possible, but geological field data did not provide a clear conclusion regarding the availability of sufficiently large areas of appropriate crystalline rock. Thus, it was concluded that the feasibility of the implementation had not been fully demonstrated.

In June 2006, the Swiss Federal Council, relying on its competent bodies, confirmed that the proof of feasibility required by law had been established for the waste concerned. The decision was not intended to designate a particular site, but to attest to the feasibility of a deep geological repository in Switzerland, as required by the Nuclear Energy Act. The demonstration of feasibility served as a basis for the Federal Council to determine the continuation of the procedure for the final disposal of these wastes.

Nagra's process of searching for a clay disposal site in Switzerland consists in choosing a site and disposal concept from several sites preselected on geological and socio-political criteria. The process will converge on a (geographic) site with an associated disposal concept in 2022.

The total volume of the radioactive waste inventory taken as reference for geological disposal is 92,000 m³. This includes 9400 m³ of HLW representing 479 containers or packages to be placed in the underground facility.

Site selection process

The process of selecting a deep geological repository site is governed by the Deep Geological Repository Sectoral Plan (Swiss Federal Office of Energy SFOE, 2008) in accordance with the Nuclear Energy Act and Land Use Planning Act. The objective of the Swiss government's process is to identify deep geological disposal sites as framework of a large-scale procedure. Safety is the top priority, but socio-economic aspects and regional planning also play a role.

The search for safe sites is implemented in such a way as to meet all of the objectives for protection formulated by the authorities and which can be achieved from the point of view of regional planning. The Sectoral Plan provides for a comprehensive coordination of all spatial impacts of deep geological disposal and ensures the early participation of cantons, communities and authorities from neighboring countries, as well as interested organizations and associations.

Phase 2 of the site selection process was completed by the Swiss Federal Council in late 2018 and Switzerland is currently in the third and final stage. Nagra studied in detail the other three areas of Jura-Est, Nördlich-Lägern and Zürich-Nordost, and then compared them with each other. The program of investigation includes seismic campaigns already conducted, as well as quaternary and deep exploratory drilling. The results of deep drilling will supplement the overall geological picture of the underground environment in the locations.

Based on the results of the earth science studies, Nagra will announce the site(s) for which it will prepare general permit applications around 2022. Nagra will then prepare the applications and plans to submit them by 2024.

According to the Sectoral Plan, a repository for HLW will not be available before 2060 (Fig. 26).

Geological disposal concepts

The spent fuel/HLW disposal packages would be leaktight for a long time and constructed of thick black steel (12–14 cm) or a copper shell.

According to the current reference concept (Nagra, 2014), each disposal canister will be flooded in an appropriate buffer material consisting of bentonite blocks for placing the package and a (very compacted) granular bentonite mixture (GBM) for

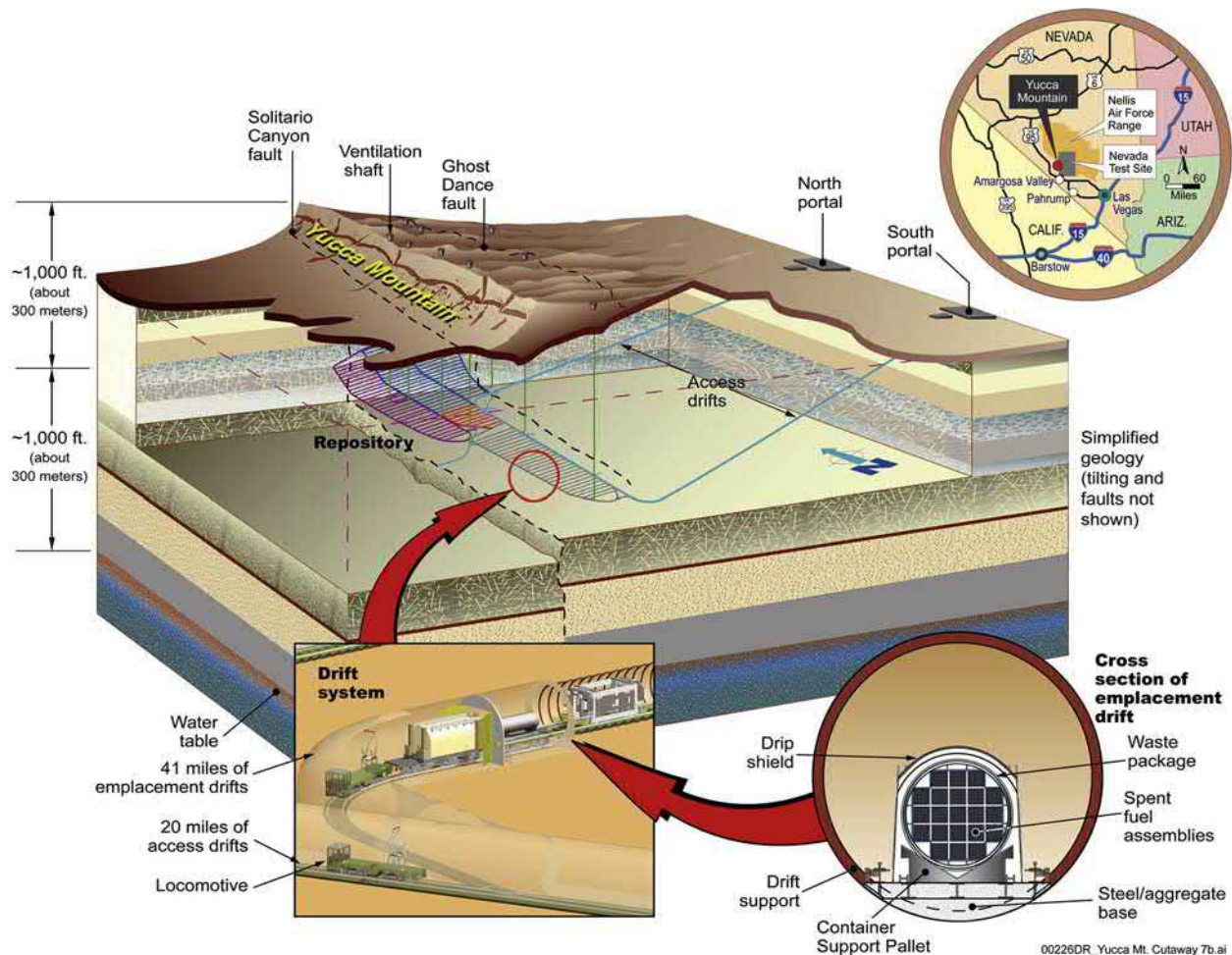


Fig. 26 Schematic of emplacement design at Yucca Mountain. Birkholzer JT, Webb SW, Halecky N, Peterson PF and Bodvarsson GS (2006) Evaluating the Moisture Conditions in the Fractured Rock at Yucca Mountain: The Impact of Natural Convection Processes in Heated Emplacement Drifts. *Vadose Zone Journal* 5: 1172–1193. (2006), with permission.

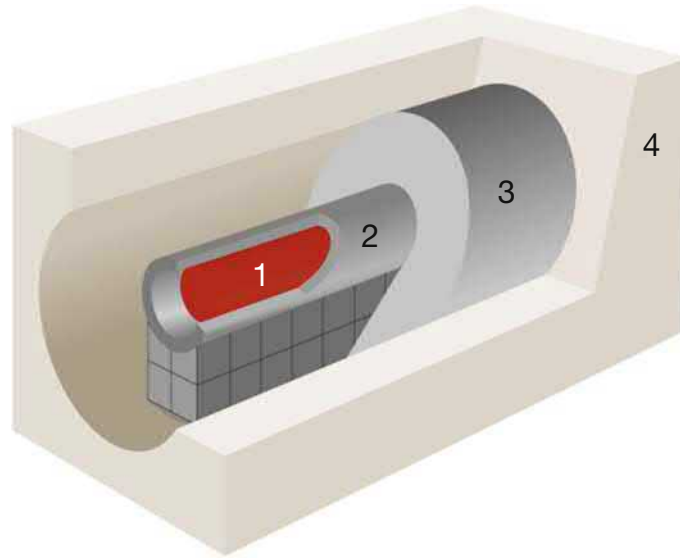
filling the voids. The disposal tunnel will be filled with GBM after each disposal canister is emplaced. The bentonite material and the backfilling procedure must meet several long-term safety and technical feasibility requirements (Nagra, 2019) (Fig. 27).

Access tunnels will also be partially filled with sand/bentonite mixtures. A prototype filling machine was designed and built as part of a test at the Mont Terri Methodological Laboratory in Switzerland.

The architecture of the HLW repository is in the design stage. The depth of the repository is highly dependent on the site. The clay formation at Opalinus is generally more than 100 m thick and is surrounded by clay-rich containment units. The repository is connected to the surface via a shaft and/or ramp. The concept has been engineered in depth, from 400 to 900 m. The disposal tunnels for spent fuel or HLW are up to 800 m long and have an internal diameter of 2.8 m (Fig. 28).

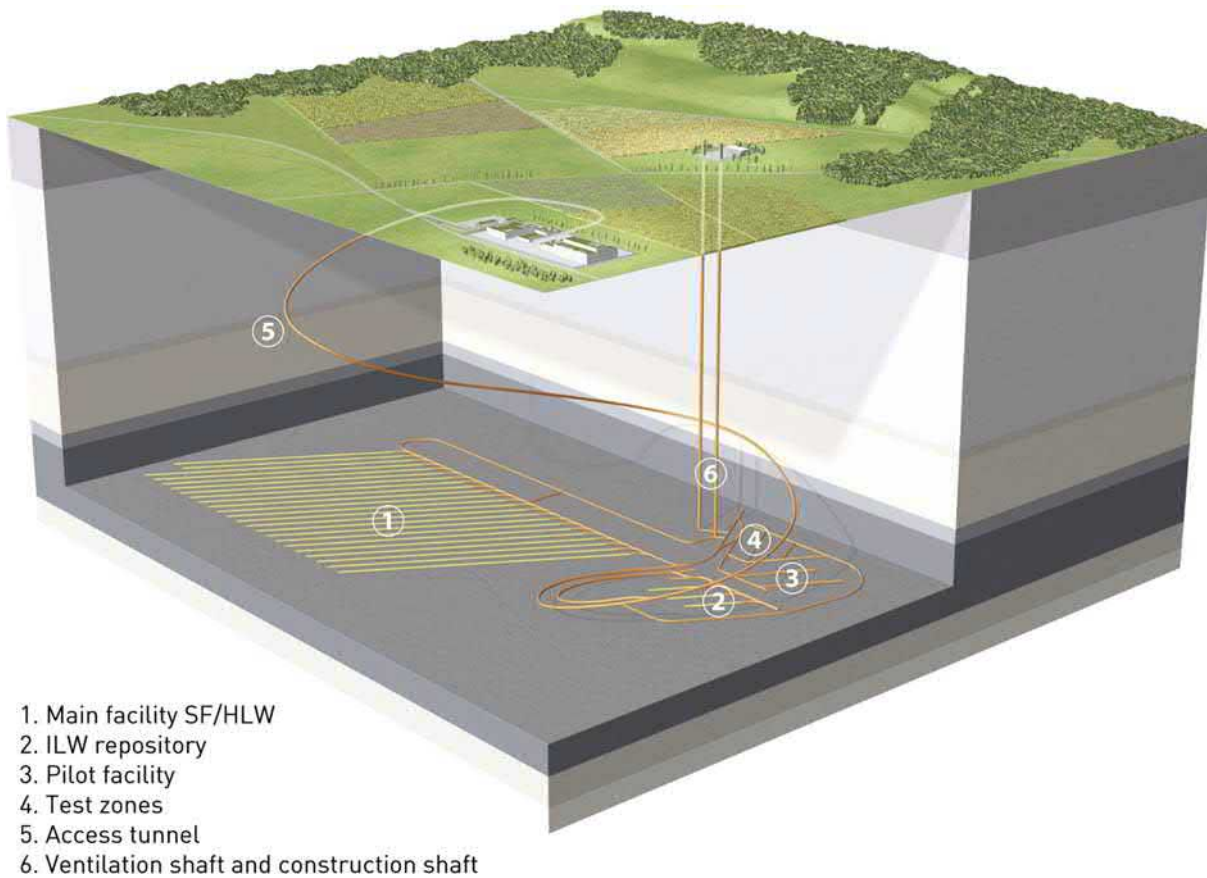
The deep geological repository will consist of an underground laboratory, pilot facility and main repository. Prior to excavation of the future deep geological repository, an underground laboratory is first built at the site. The laboratory performs experiments directly in the host rock to verify the characteristics of the chosen location (in operation between 2032 and 2048, according to the Nagra estimate). Only when the acquired data meets the stringent security requirements can excavation begin on the other parts of the facility that are planned for 2049 and 2059 (first the pilot facility, then the main repository). The pilot facility is used to emplace waste characteristic of the repository to observe its behavior over a long period of time. Both the underground laboratory and pilot facility enable verification of the essential assumptions and parameters that form the basis of the barrier and disposal concept. Long-term disposal in the main radioactive waste repository will begin as soon as the pilot facility is filled.

A concept for disposal of LILW and HLW at the same site is also under consideration (Fig. 29).



1. Glas matrix, containing radioactive material
2. Metal canister
3. Bentonite backfill
4. Host rock

Fig. 27 Disposal concept package and engineered barrier. Reproduced with permission of Nagra (2018) *Long-term safety: The main goal of deep geological disposal of radioactive waste*. Nagra, Wettingen, Switzerland.



1. Main facility SF/HLW
2. ILW repository
3. Pilot facility
4. Test zones
5. Access tunnel
6. Ventilation shaft and construction shaft

Fig. 28 Architectural concept for HLW repository. Reproduced with permission of Nagra (2018) *Long-term safety: The main goal of deep geological disposal of radioactive waste*. Nagra, Wettingen, Switzerland.

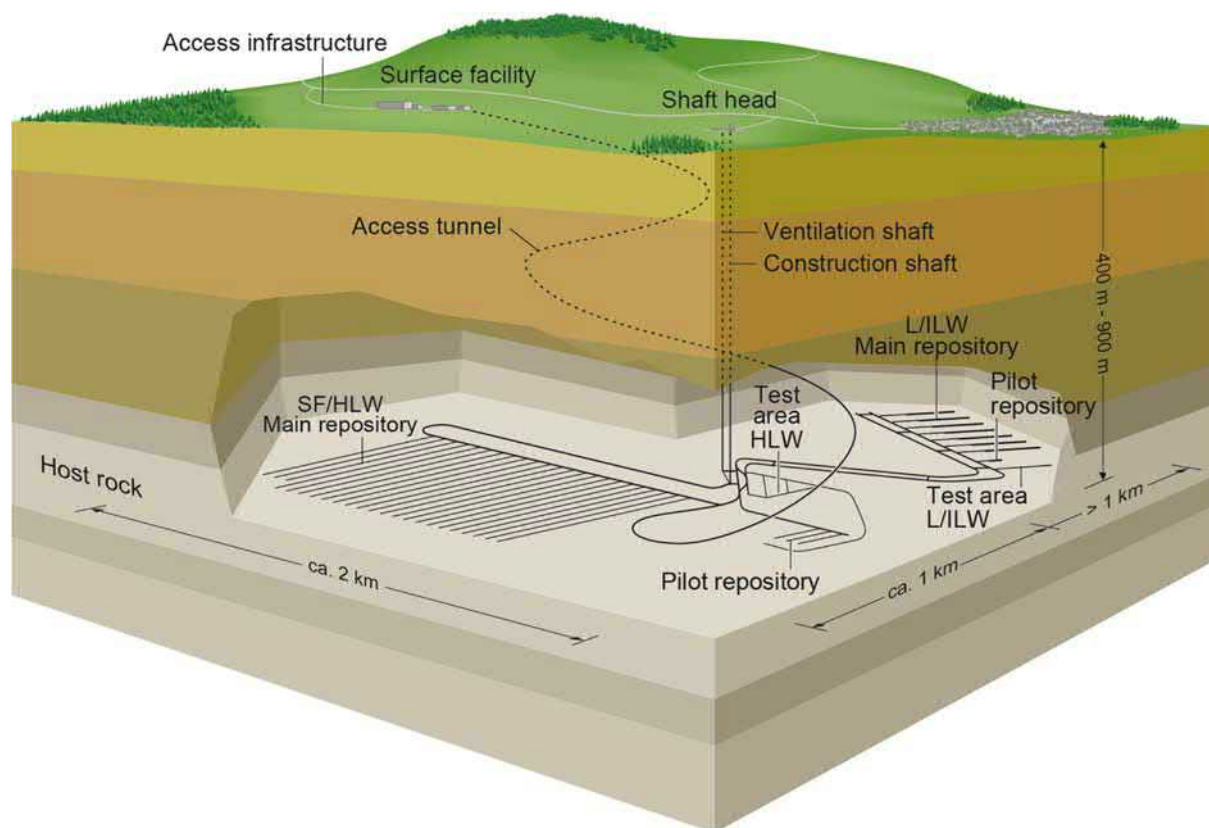


Fig. 29 Disposal concept combining HLW and L/ILW. Reproduced with permission of Nagra (2016) *Waste Management Programme 2016 of the Waste Producers*. Nagra Technical Report. NTB 16-01 E. Nagra, Wetztingen, Switzerland. (Figs. 3–11).

Belgium

In practice, the national program has adopted an approach based on waste management streams, where the streams are defined according to the ultimate destination (planned or not) of the waste. More specifically, the following main types of waste (and their prospective inventory in 100 years) are taken into account for geological disposal, although the Belgian government has not yet taken the decision:

- conditioned LILW-LL, also known as category B waste (corresponds to intermediate activity waste according to the IAEA Classification of Radioactive Waste, 2009), 11,100 m³ over 100 years;
- conditioned HLW, also known as category C waste, which include unprocessed spent fuel from nuclear power plants and certain research reactors declared as waste and vitrified waste from reprocessing spent fuel from nuclear power plants (corresponds to HLW in the IAEA classification), 4500 m³ over 100 years.

The CEN SCK (Center for the Study of Nuclear Energy) launched the research, development and demonstration for geological disposal of HLW and ILW in 1974. Beginning in the early 1980s work continued under the auspices of Ondraf/Niras (the Belgian national agency for radioactive waste and enriched fissile material), which retained CEN/SCK as its main research partner. Having begun work in 1980, CEN/SCK completed construction on the HADES underground laboratory in 1987. Located at Mol at a depth of 225 m, in the heart of the Boom Clay layer, it is the first underground laboratory in Europe, built in a deep clay formation to study the possibility of geological disposal in clay. The layer has an average thickness of about 100 m.

In 1995, researchers launched the PRACLAY project. The purpose of the project is to demonstrate the possibility of using industrial techniques to construct a repository in poorly indurated clay. The project tested the construction methods of an actual disposal facility. During the first phase of the project, a second access shaft was added to HADES. The shaft was then connected to the existing part of the laboratory. In 2007, the PRACLAY gallery was dug perpendicular to the connecting drift to host a large-scale heating experiment: the second phase of the PRACLAY project. The experiment will study the effect of heat released on the deep clay layer. It is also intended to examine the extent to which excavations influence the behavior of the clay layer (Fig. 30).

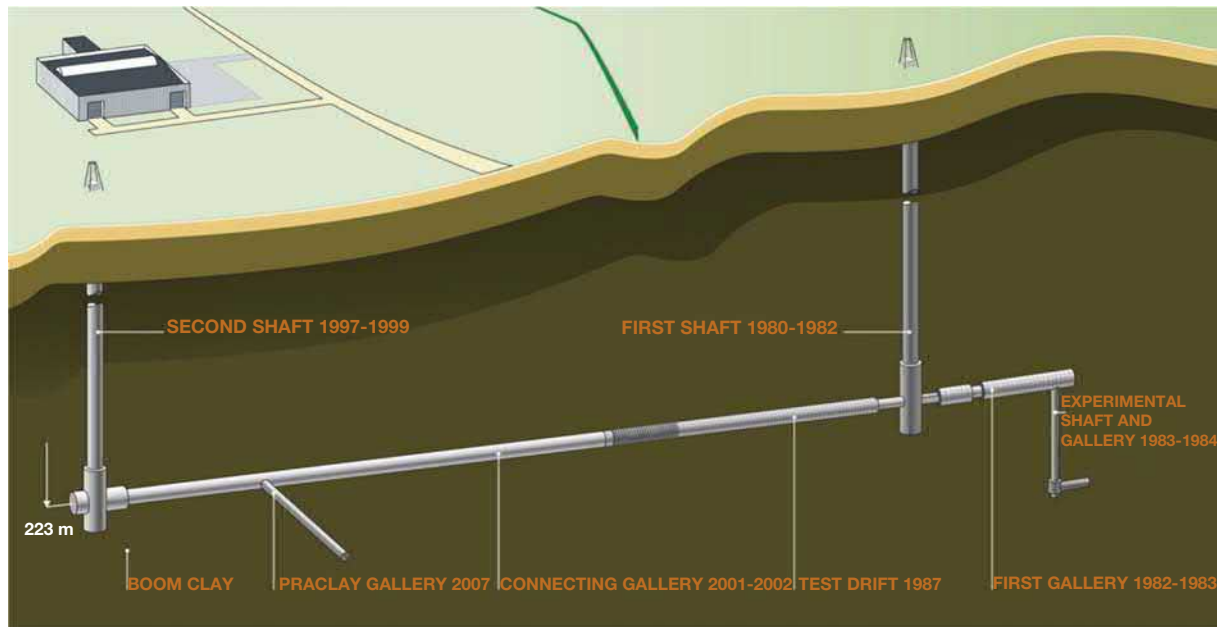


Fig. 30 The HADES underground laboratory was built in two phases. Courtesy of ONDRAF-NIRAS.

HADES was built in a layer of poorly indurated clay at a depth of about 225 m. Two phases of construction have been completed:

- a construction phase by manual excavation after freezing between 1980 and 1987. The first phase included excavation of the first access shaft, two drifts and an experimental shaft.
- a construction phase using tunneling machines between 1997 and 2007. The second phase saw excavation of the second access shaft, the connecting drifts and PRACLAY.

Several canister options are under consideration for vitrified waste and spent fuel:

- the Vitrified waste supercontainer (37,147 tons, 4077 m height and 2144 m diameter)
- the spent UOX supercontainer (variable mass from 46,851 to 69,711 tons, variable height from 4193 to 6122 m and 2165 diameter)
- the spent MOX supercontainer (35,202 tons, 5498 m height and 1724 m diameter) (**Fig. 31**)

Ondraf/Niras took the initiative, in 2011, to bring together in one document, *Plan déchets* (Waste Plan), all the elements necessary to make a “decision in principle” concerning Belgian policy on the long-term management of HLW and ILW-LL (including used fuel if declared as waste). The document, accompanied by a Strategic Environmental Assessment (SEA), was submitted to a SEA Committee, the Belgian Federal Council on Sustainable Development and regional governments, as well as the public. Ondraf/Niras also submitted the documents for an opinion from the Belgian Federal Agency for Nuclear Control (FANC). In 2015, Ondraf/Niras resubmitted the Waste Plan in order to comply with the European directive on radioactive waste transposed into Belgian law.

Studies for the long-term management of HLW/ILW have so far focused on geological disposal in poorly indurated clay (found at Boom or Ypres). The disposal options for this waste have evolved since 2011:

- 2011: Proposal by Ondraf/Niras, in the Waste Plan, to retain geological disposal as a long-term reference solution using poorly indurated clays at a depth of 225 m at the Mol site;
- mid-2018: Following the request of the supervisory ministries and FANC, a new proposal was made without specifying the host rock and the number of sites;
- Late 2018: A new architecture was proposed for geological disposal on a single site located in Belgium with poorly indurated clay at a reference depth of 400 m. Authorization is scheduled for 2050.

There has not yet been a “decision in principle” at the federal level regarding the long-term management of HLW and ILW-LL. Without a national policy for the long-term management of B and C waste in Belgium, there is no preferred rock or site. Seeking to optimize the process, FANC requested that all potential host rocks be considered. There is no dedicated and comprehensive regulation on the site search process or the status of spent fuels. Ondraf/Niras was legally responsible for proposing management policies as well as for concepts for radioactive waste disposal.

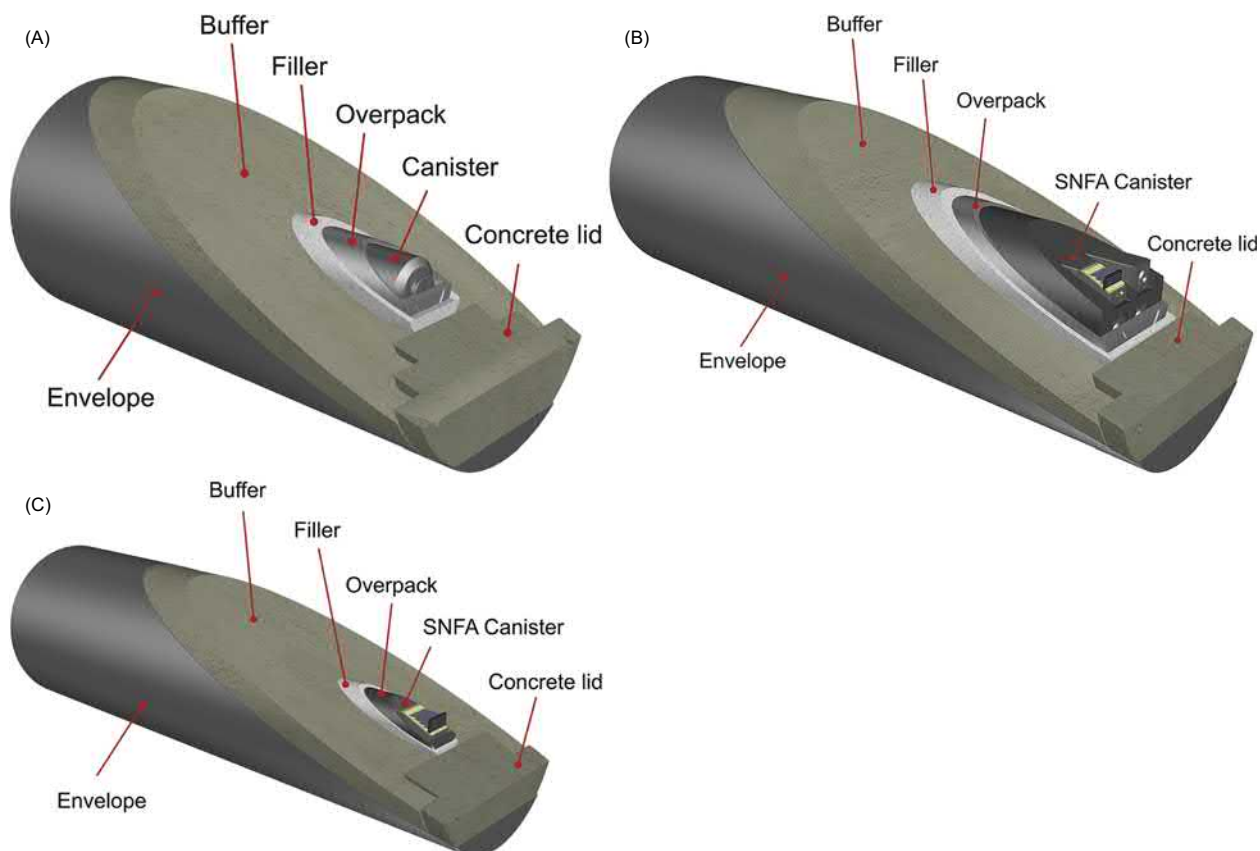


Fig. 31 Concepts of Vitriified waste supercontainers (A), Spent UOX supercontainers (B) and Spent MOX supercontainers (C). (A-C) Courtesy of ONDRAF-NIRAS.

Repository concept for B and C waste

Several options are under consideration for canisters for vitriified waste and spent fuel. Their diameter is approximately 2.1 m and between 4 and 6.1 m in length. The weights vary between approximately 37 and 70 tons. The containers provide radiation protection.

The repository design for B and C waste builds on the combined experience and knowledge gained on Boom clay in HADES. The proposed site for the construction of a geological disposal in 2011 was therefore Mol ([ONDRAF/NIRAS, 2013](#)).

FANC, in its “FANC Opinion on the National Spent Fuel and Radioactive Waste Management Programme” (April 2015), expressed its preference for geological disposal, a term that includes the options of underground drifts and deep drilling. In contrast, FANC considers that it is not possible to make a decision on the host “poorly indurated clay” formation proposed by Ondraf/Niras, due to the lack of systematic comparison of potential host formations, as well as isolation, which is not sufficiently addressed, particularly concerning the presence of usable groundwater reserves (aquifers) adjacent to the proposed host formation.

In 2018, after the opinion was issued, Ondraf/Niras proposed a new concept taking into account the following characteristics:

- site depth between 200 and 600 m;
- measures to support flexible decision making, package recoverability and repository monitoring;
- special attention paid to operational safety.

The reference schedule has thus been changed. It is expected to be operational in 2050. The surface area of the disposal is approximately 4 km². The number of disposal drifts, 400 m long, is 42 for C waste and 37 for B waste. The B waste and C waste sections are separated by the shaft area ([ONDRAF/NIRAS, 2020](#)) ([Fig. 32](#)).

In 2020, Ondraf/Niras submitted to FANC, for its opinion, the revision of the long-term management plan for conditioned HLW and ILW-LL in Belgium, the accompanying environmental impact report and the associated non-technical summary.

Several months later, FANC gave its opinion stating that geological disposal, whether in drifts or deep drilling, is the safest long-term management option for HLW and ILW-LL taking into account current knowledge. This is based on the following technical considerations:

- a repository ultimately based on exclusively passive measures appropriate to such waste;

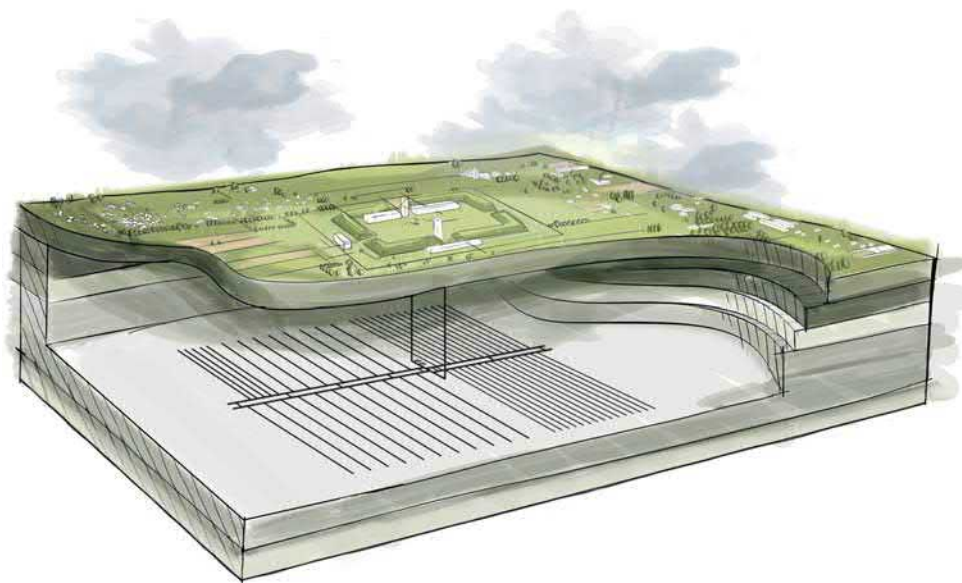


Fig. 32 Conceptual repository architecture during operation. Courtesy of ONDRAF-NIRAS.

- this type of waste requires a level of containment and isolation that can only be provided by geological formation at an adequate depth;
- this option limits burdens on future generations and helps to avoid uncertainty about its context.

FANC notes, however, that in subsequent stages of the decision-making process, the safety of geological disposal in Belgium will still need to be demonstrated on the basis of safety files within the framework of a decision-making process to be defined.

Japan

Within the framework Japan's nuclear power programme, the safe management of the resulting radioactive waste, particularly vitrified HLW from fuel reprocessing, has been a major concern and a focus of R&D since the late 1970s. The *Second Progress Report on R&D for the geological disposal of HLW* (Report H12) was published after two decades of R&D activities and shows that HLW disposal in Japan is feasible and can be implemented at sites that meet certain requirements for geological stability. The report supported the government's decisions that were the basis of the Act on Final Disposal of Specified Radioactive Waste, which came into force in 2000. The Act on Final Disposal specifies that deep geological disposal of HLW must be carried out at depths greater than 300 m, and that a three-step site selection process must be completed: selection of preliminary investigation areas (PIAs), selection of detailed investigation areas (DIAs) and selection of a geological disposal construction site.

Created in the same year, by the Act on Final Disposal of Specified Radioactive Waste, the Nuclear Waste Management Organisation of Japan (NUMO) is the body responsible for geological disposal of vitrified HLW from reprocessing spent fuel used in nuclear power plants. In 2007, the act was revised, extending the inventory of geological disposal to TRU wastes containing over a certain concentration of long-life radionuclides (transuranic type) generated by the operation and dismantling of reprocessing plants and MOX fuel fabrication plants.

This will consist of 40,000 containers of vitrified HLW and 18,000 m³ of TRU waste.

As part of its search for a site, NUMO issued a call for applications from municipalities in 2002. A few sites expressed interest in the project, including the city of Toyo in Kochi Prefecture. However, it withdrew its application in 2007. Given the failures and a social context which rejects nuclear energy following the Fukushima accident in December 2013, the Japanese government officially adopted a new procedure for selecting a geological disposal site. It provides that the competent authorities first propose a selection of appropriate sites, on the basis of scientific investigations, before seeking a local consensus on one of them. In May 2015, the procedure was amended to make the government more proactive in preparing the local consensus.

The final choice of the site to host the repository is planned for 2025, with operation scheduled to begin in 2035.

Underground laboratories

Parallel to site searches, the Japan Atomic Energy Agency (JAEA) continues its research programs. To date, there are two underground laboratories, the first in Mizunami at a depth of 500 m in crystalline rock, and intended to reach 1000 m,

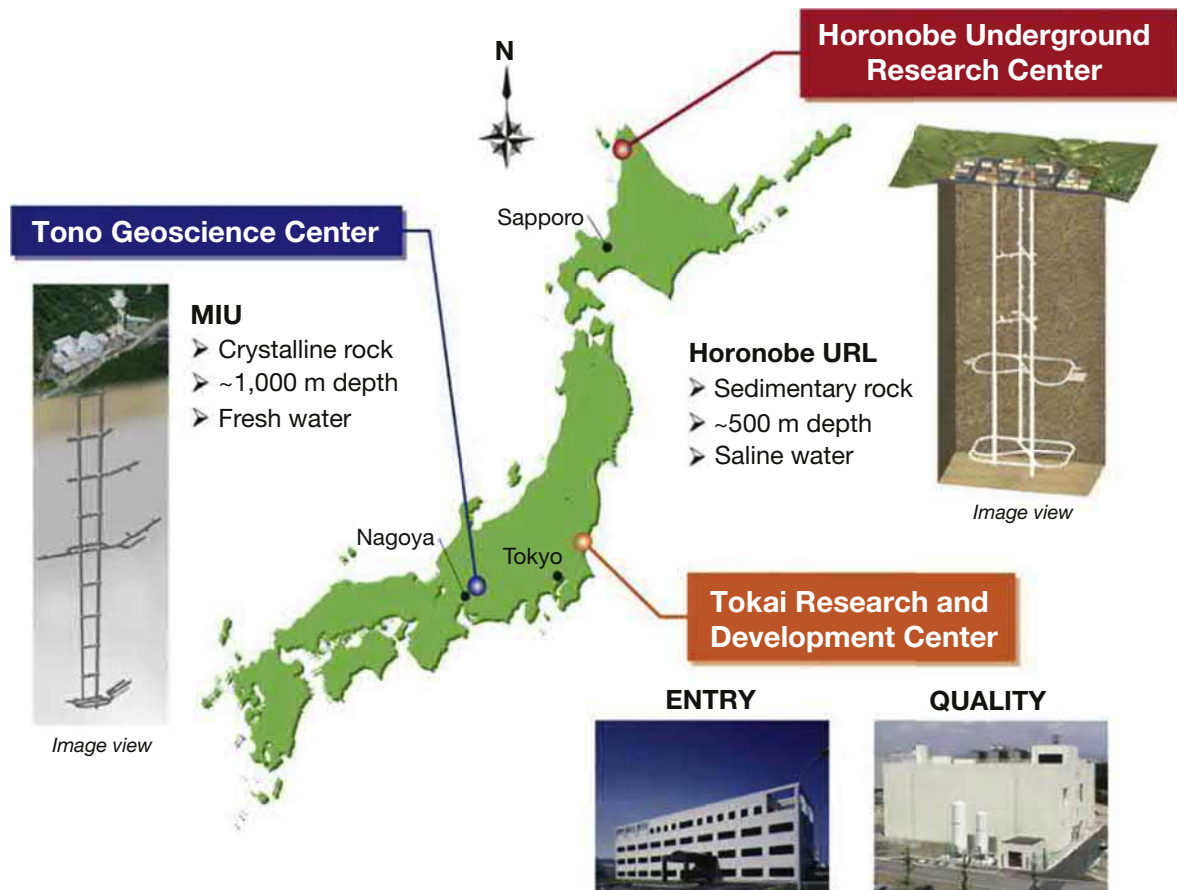


Fig. 33 Locations of the two JAEA underground laboratories and surface R&D facilities. Courtesy of JAEA.

and the second in Horonobe, at a depth of 500 m in clay (Hama et al., 2016). In surface R&D, JAEA operates two major facilities (ENTRY and QUALITY) at its Tokai site. They study geological and engineered materials in repository conditions (Fig. 33).

Repository concept

In Japan, it was decided that the waste will be placed in a stable rock formation at a depth of more than 300 m. A multi-barrier system consisting of artificial and natural (geological) barriers will isolate and contain radioactive waste over a long period of time (NUMO, 2013).

The future repository site will consist of surface installations where vitrified HLW will be encapsulated in steel overpacks and TRU waste in steel primary steel containers, before being grouped into concrete overpacks (Fig. 34).

Underground facilities will include tunnels for the transport and disposal of waste packages and will be built in a stable rock formation more than 300 m underground. The underground project includes construction of disposal drifts, emplacement of waste packages and backfilling of disposal tunnels. There will be two large separate HLW and TRU disposal areas for parallel disposal operations (Fig. 35).

The repository is scheduled to operate for 60 years, after which it will be continuously monitored for 300 years.

NUMO plans to build a disposal site, with surface facilities extending over 1–2 km², with the surface area of the underground facilities:

- for disposal of HLW packages, 3 by 2 km;
- for disposal of TRU packages, 300 by 500 m (Fig. 36).

Russia

A site search for a geological repository has been conducted for more than 20 years. The Radioactive Waste Management Act and the amendment of certain legislative acts (Federal Law 190-FZ of 11 July 2011) constitute the first legislative act regulating the

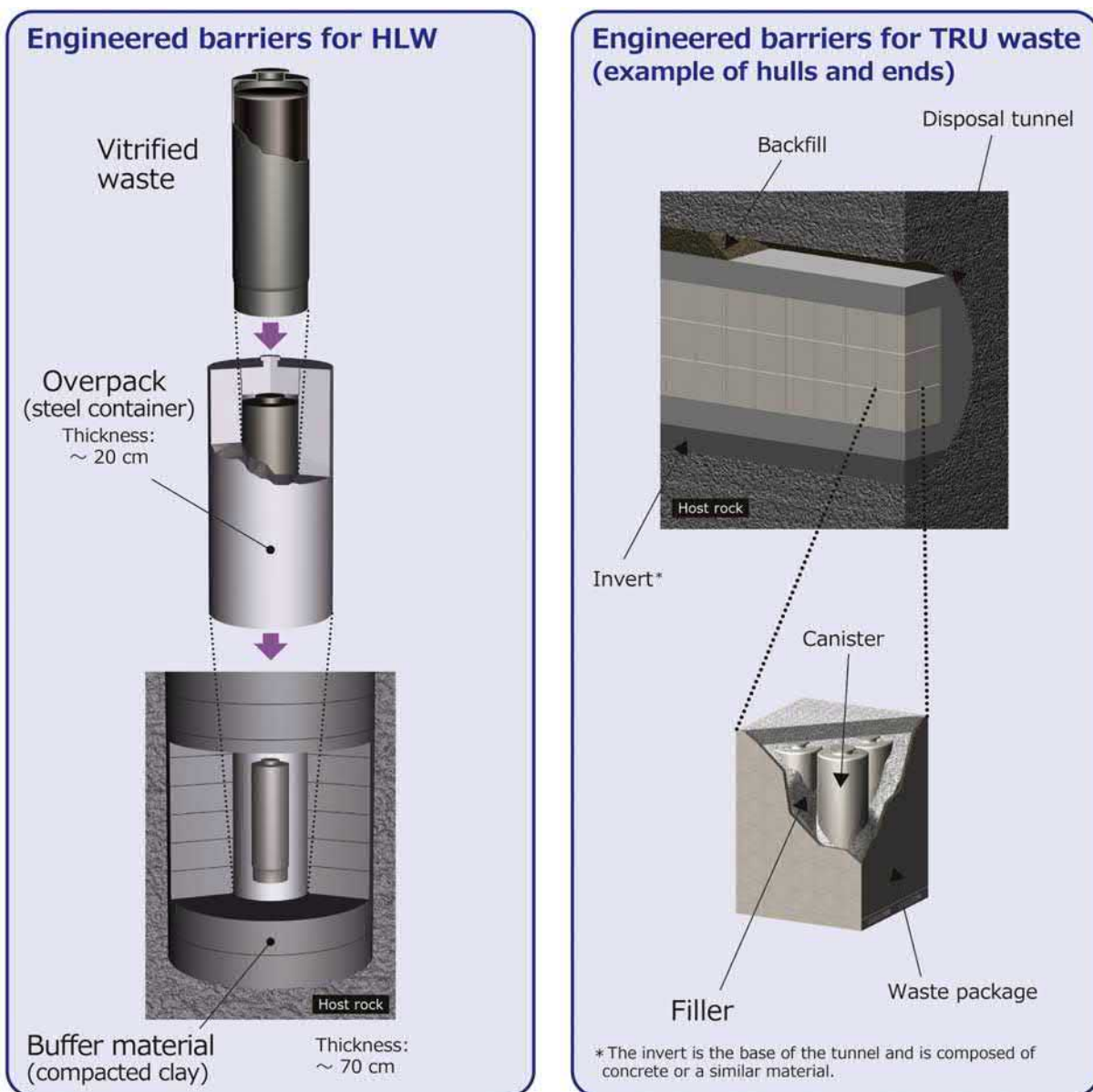


Fig. 34 Illustration of methods for packaging HLW and TRU wastes. Courtesy of NUMO.

end of the fuel cycle. This legislative framework distinguishes radioactive waste from spent fuels and proposes a classification into 8 categories according to lifetime (short or long) and level of activity (VLLW, LLW, ILW, HLW). The law also created a national radioactive waste management operator, NORAO, in charge of developing a geological disposal solution for class 1 and class 2 radioactive waste:

- Class 1 waste: exothermic liquid or solid HLW,
- Class 2 waste: non-thermal liquid or solid HLW, sealed sources and ILW-LL

The results of the investigative studies of the Yeniseisky site in Krasnoyarsk, in the immediate vicinity of ZATO Zheleznogorsk, in the gneiss rocks of the Nizhnekansky massif have made it possible to draw a conclusion on the suitability of the geological environment for disposal of radioactive waste in the Archean gneisses within the target depth range of 450–550 m. The methods of studying the geological environment and the results of deep drilling cannot provide all the necessary and reliable information on both the interaction of deep structures with host rocks and groundwater and the processes occurring in the “waste—technical barriers—geological environment” system.

In 2012, the site was recognized as an ideal candidate for the construction of an underground laboratory.

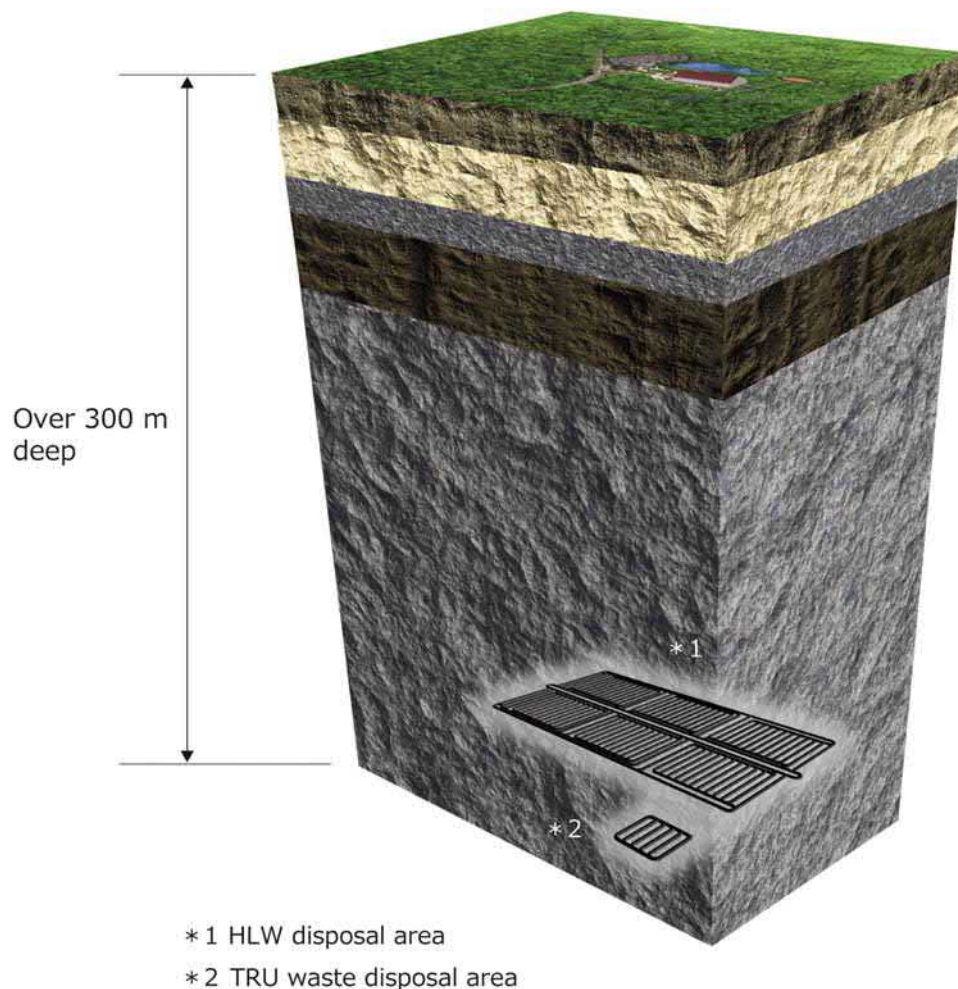


Fig. 35 Conceptual illustration of the future geological repository. Courtesy of NUMO.

In 2016, NORAO received authorization to construct an underground laboratory and published the *Strategy for Creating a Deep Disposal Site for Radioactive Waste* in 2018 (NORAO, 2018). Justification for the possibility of transition from laboratory activities to the industrial operation of the disposal facility will be presented on the basis of the results of the research.

The creation of an underground research laboratory will enable the implementation of a research complex covering the following areas:

- studies of rock mass under natural conditions, determining possible ranges in variation of important parameters to assess containment safety;
- experimental studies of the long-term confining properties of technological barrier materials under conditions including thermal influences and water saturation;
- development of technical solutions for design, construction of technological barriers and disposal of radioactive waste;
- development and testing of equipment, technical resources and methods of operating mine sites;
- verification of mathematical models to assess the behavior of the entire multi-barrier system to justify safety;
- demonstration on a real or reduced scale of the possibility of sealing the disposal chambers (Fig. 37).

Upcoming deadlines set by NORAO include:

- 2019–23: field and laboratory studies of rock mass during the construction of vertical and horizontal structures of the underground laboratory;
- from 2024: pilot research and development of technological operations in the underground laboratory;
- after 2028: review of the results of pilot studies in the underground laboratory.

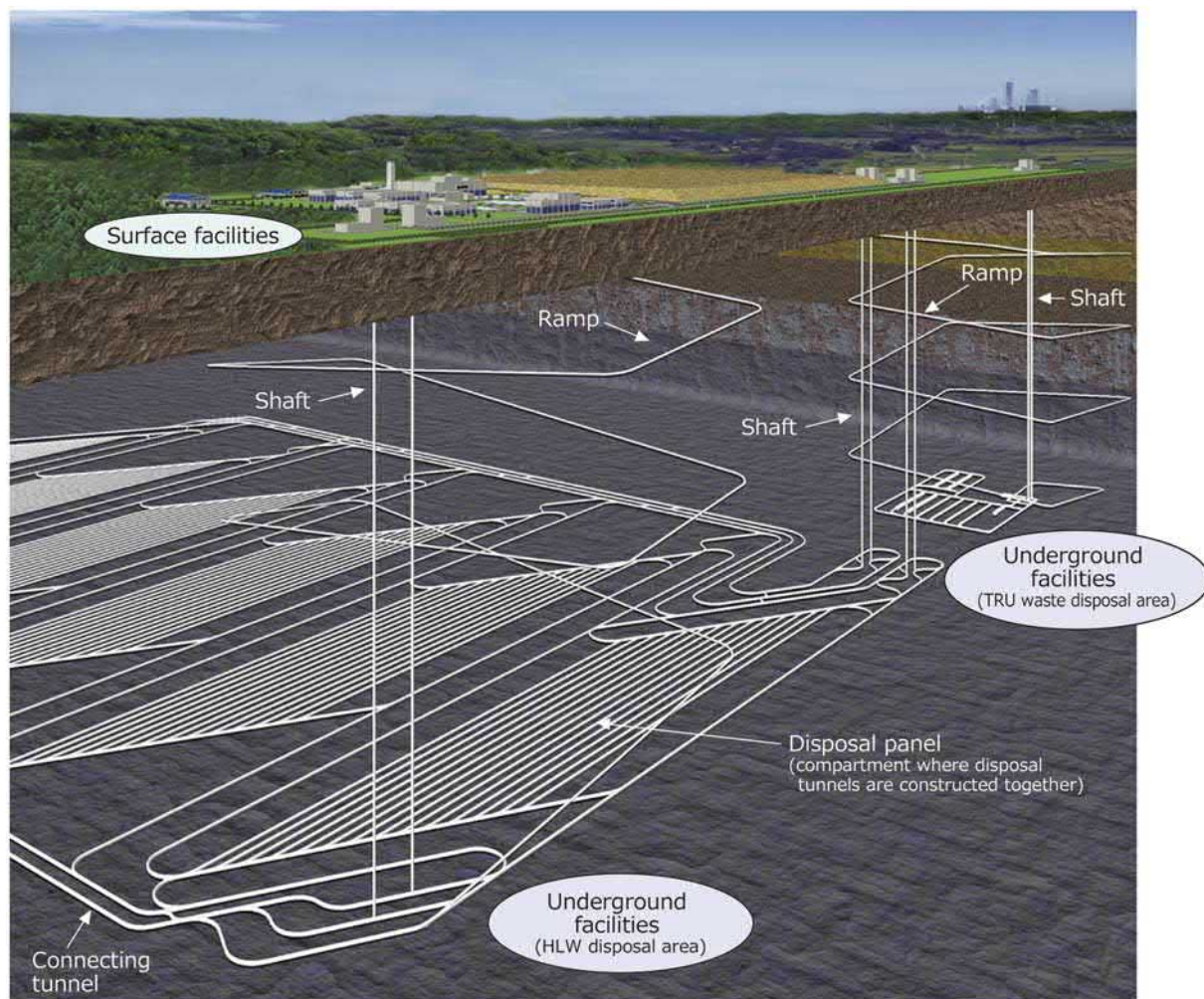


Fig. 36 Detailed conceptual illustration of future geological repository. Courtesy of NUMO.

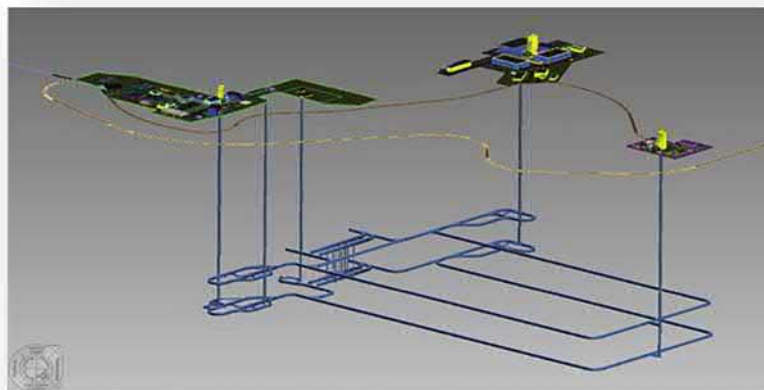


Fig. 37 Basic Concept of the underground research laboratory. Courtesy of NORAO.

Conclusion

In most of the countries that have developed military or electronuclear programs, numerous technical solutions for the SNF and HLW management have been imagined and studied for the past 70 years. Management methods such as extended long-term storage, reducing their dangerousness by transmutation, sending them into space, evacuating them to the seabed or in the magma, disposing them of in the geological environment, have all been thoroughly analyzed and evaluated (IRSN, 2019). Of these, deep geological repository is the only management method whose feasibility has been proven and which can handle all SNF and HLW (OECD/NEA, 2020). It is the international benchmark for the sustainable management of this waste. None of the alternative modes of management currently envisaged would allow a sustainable and secure management of all SNF and HLW inventories, around the world, already existing and to be produced. All the alternative solutions would still require research and very significant investments in the hope, not guaranteed, of removing the technological obstacles to their use today.

Following a universal ethical requirement, their sustainable management aims at permanently securing them to prevent or limit any burden on the future generations. The current generations are indeed benefiting from the nuclear activities. It is their responsibility to deploy technical options insuring the safe and sustainable management of SNF and HLW, without passing on the burden from one generation to the next indefinitely.

As illustrated within this article, the development of a geological repository is a long process, divided in different stages and based on a stepwise approach. Achieving progress in many national disposal programs is difficult, owing to institutional, administrative and economical aspects, reticence among decision makers to deal with controversial issues and the opposition of stakeholders. Over the past 30 years, several geological disposal projects have been impeded or even halted due to vehement or even violent opposition. The ineffectiveness or absence of stakeholder involvement has altered acceptability and resulted in important obstructions to programs progresses. The introduction of national and local public debates along with the development of deep geological repositories around the world in the early 2000s has proven the crucial importance to involve the national public and local stakeholders in the projects' development decision-making and through concertation. Because of changes in society's decision-making environment and increased public sensitivity to all matters connected with environmental protection, nuclear power, radioactivity, and especially radioactive waste, any decision regarding whether, when and how to implement waste management solutions nowadays typically require thorough public examination and the involvement of many relevant stakeholders (OECD/NEA, 2004).

Certain countries facing such public refusal for hosting a national deep geological repository and some of the 30 countries with a small inventory of SNF/HLW (<3000 tHM) have together explored the practicalities and implications of sharing some elements of their radioactive waste management strategies—specifically, sharing one or more deep geological repository sited within one of the interested countries, rather than each developing its own facilities. Since 2004, the national benefits of being part of a multinational shared geological disposal program have been studied (IAEA, 2004; ERDO, 2013). They were identified as achieving a clear demonstration of a credible approach to responsible management of national radioactive wastes; a reduced R&D burden; increased, pooled resources to develop a realistic and timely solution; large economic incentives and infrastructure improvements to the host country and access to wider skills and technology. Multinational shared repositories being seen as a real opportunity for certain countries and thus implementing the “wait and see” approach, International organizations are recommending to develop their own national program following a “dual track” approach that includes working cooperatively with other interested countries toward shared solutions (IFNEC, 2016; IAEA, 2016). As of today, considering the important remaining challenges both at the legal and technical levels (environmental impact liability, integration of important chemical/radiological variability of conditioned SNF or HLW, etc.) multinational waste disposal facilities are criticized as not being credible until such time as a country agrees to host one.

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Actinide Immobilization in Dedicated Wasteforms: An Alternative Pathway for the Long-Term Management of Existing Actinide Stockpiles

Lewis R. Blackburn and Neil C. Hyatt, NucleUS Immobilisation Science Laboratory, Department of Materials Science and Engineering, University of Sheffield, Sheffield, United Kingdom

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Abbreviations

ALSEP	Actinide Lanthanide Separation
FP	Fission Product(s)
GDF	Geological Disposal Facility
HAL	High Activity Liquor
HLW	High Level Waste
IAEA	International Atomic Energy Agency
ILW	Intermediate Level Waste
LLW	Low Level Waste
MA	Minor Actinide(s)
MOX	Mixed Oxide Fuel
NPT	Nuclear Non-Proliferation Treaty
PWR	Pressurized Water Reactor
RTG	Radioisotope Thermoelectric Generator
SNF	Spent Nuclear Fuel
TALSPEAK	Trivalent Actinide Lanthanide Separation with Phosphorus-Reagent Extraction from Aqueous Komplexes
TBP	Tributyl-phosphate
TRU	Transuranic

Actinide production and inventories

Actinide elements, whilst a valuable resource for scientific study due to their rich chemistry, present a unique challenge with respect to radioactive waste management, due to their long half-lives, high radiogenic heat output, and chemical toxicity (see Table 1). With the exception of U and Th, which exist primordially in appreciable quantities, the actinide series is produced synthetically, as the result of either transmutation and decay processes occurring within commercial and experimental nuclear reactors. The production of Pu isotopes poses an aggrandized risk, insofar as these isotopes are separated in considerable quantities, with relatively small critical masses capable of sustaining a fission reaction, and are therefore subject to proliferation safeguards. Under conventional operating parameters, a civilian reactor operating in the burn-up range 35–45 MWd kgU⁻¹ will produce around 1 at.% Pu, among other transuranic (TRU) elements at around 0.1 at.% (Ewing, 2015). It should be noted that the isotope distribution of Pu at discharge is a generally a function of reactor design, fuel type, initial ²³⁵U enrichment, and burn up (Table 2). The remaining spent nuclear fuel (SNF) matrix is comprised of approximately 96 at.% U, with 3–4 at.% entrained fission products (FP). Notable high yield constituents that require consideration for immobilization include

Table 1 Properties of some actinide isotopes in the range ^{227}Ac – ^{246}Cm .

Actinide	Atomic number	Isotopes	Half-life	Specific activity (Ci g^{-1})	Primary decay mode	Decay product
Ac	89	^{227}Ac	21.7 (year)	72	β^-	^{227}Th
		^{228}Ac	6.13 (h)	2.2×10^6	β^-	^{228}Th
Th	90	^{227}Th	18.7 (day)	3.2×10^4	α	^{223}Ra
		^{228}Th	1.9 (year)	820	α	^{224}Ra
		^{230}Th	7.5×10^4 (year)	190	α	^{226}Ra
		^{231}Th	25.5 (h)	5.2×10^3	β^-	^{231}Pa
		^{232}Th	1.4×10^{10} (year)	1.1×10^{-7}	α	^{228}Ra
		^{234}Th	24.1 (day)	2.4×10^4	β^-	$^{234\text{m}}\text{Pa}$
Pa	91	^{230}Pa	17.4 (day)	3.2×10^4	β^+	^{230}Th
		^{231}Pa	3.3×10^4 (year)	4.5×10^{-2}	α	^{227}Ac
		^{232}Pa	1.3 (day)	2.1×10^4	β^-	^{232}U
		^{230}U	20.8 (day)	2.7×10^4	α	^{226}Th
U	92	^{232}U	68.9 (day)	21	α	^{228}Th
		^{233}U	1.6×10^5 (year)	9.5×10^{-3}	α	^{229}Th
		^{234}U	2.5×10^5 (year)	6.2×10^{-6}	α	^{230}Th
		^{235}U	7.0×10^8 (year)	2.1×10^{-6}	α	^{231}Th
		^{236}U	2.3×10^7 (year)	6.5×10^{-5}	α	^{232}Th
		^{238}U	4.5×10^9 (year)	3.3×10^{-7}	α	^{234}Th
		^{237}Np	2.1×10^6 (year)	6.9×10^{-4}	α	^{233}Pa
		^{239}Np	2.4 (day)	2.3×10^5	β^-	^{239}Pu
Pu	94	^{238}Pu	87.7 (year)	17	α	^{234}U
		^{239}Pu	2.4×10^4 (year)	6.2×10^{-2}	α	^{235}U
		^{240}Pu	6.6×10^3 (year)	0.23	α	^{236}U
		^{241}Pu	14.3 (year)	110	β^-	^{241}Am
Am	95	^{242}Pu	3.8×10^5 (year)	3.9×10^{-3}	α	^{238}U
		^{241}Am	432.2 (year)	3.2	α	^{237}Np
		^{243}Am	7370 (year)	0.19	α	^{239}Np
Cm	96	^{242}Cm	182.2 (day)	3.3×10^3	α	^{238}Pu
		^{243}Cm	29.1 (year)	4.2	α	^{239}Pu
		^{244}Cm	18.1 (year)	82	α	^{240}Pu
		^{245}Cm	8.5×10^3 (year)	0.1	α	^{241}Pu
		^{246}Cm	4.8×10^3 (year)	0.4	α	^{242}Pu

Table 2 Plutonium isotope distribution on discharge.

Pu Source	Mean fuel burn up (MWd/tHM)	Pu isotope on discharge				
		^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Magnox	5000	0.1	70.0	24.9	3.9	1.1
AGR	30,000	1.1	53.5	27.9	12.3	5.2
PWR	45,000	2.2	54.3	22.4	14.9	6.2
WG ^a	N/A	~0	<92	>8	~0	~0

^aMoD reporting in 2000.

^{129}I ($t_{1/2} = 15.7$ My) and ^{137}Cs ($t_{1/2} = 30.2$ y), and activation products originating from neutron activation of cladding materials e.g., ^{55}Fe ($t_{1/2} = 2.7$ y). Operating a reactor to a higher burn-up affords a greater yield of minor actinide (MA) species, which form a significant contribution to the overall radioactive output of the SNF matrix, often with considerable half-lives. Notable examples of MA species, for which immobilization options have been considered include ^{241}Am ($t_{1/2} = 433$ y), ^{239}Pu ($t_{1/2} = 21,400$ y), ^{237}Np ($t_{1/2} = 1.2$ My) and ^{237}Cm ($t_{1/2} = 1.56$ My). At fuel discharge there remains the option to implement an aqueous reprocessing procedure, whereby the SNF matrix is dissolved and actinides are targeted for solvent extraction, thus decreasing the radiogenic output associated with the fuel matrix, and allowing the separation and recovery of fissile material. Currently, only the PUREX (plutonium-uranium-reduction-extraction) process has been deployed at the industrial scale, examples of which are reprocessing plants integrated into the back-end of commercial fuel cycles in France (La Hague; capacity ~1600 tHM/year) and the United Kingdom (Magnox; capacity ~1500 tHM/year). During the reprocessing stage, the irradiated UO_2 pellets are sheared of cladding, prior to dissolution in a hot solution of 3–6 M HNO_3 . Pu^{4+} and U^{6+} may then be extracted to an organic phase via complexation with 20–40 vol.% tri-butyl-phosphate (TBP) diluted in inert kerosene, and separated from the residual

Table 3 International inventories of Pu as reported to IAEA under INFCIRC/549 (reporting as of 2018).

Nation	Unirradiated Pu (teHM)	Pu Contained in Spent Fuel (teHM)
United States	49.3	716
Russia	61.3	167
United Kingdom	138.9	26
France	83.2	299.6
China	–	–
Japan	9.0	169
Germany	–	123.1
Belgium	< 50 kg	44
Switzerland	< 2 kg	20

high activity liquor (HAL) containing the remaining MA and FP contaminants. Pu is subsequently separated from U through a reductive backwash, and reclaimed via formation of insoluble $\text{Pu}^{4+}(\text{C}_2\text{O}_4)_3 \cdot 6\text{H}_2\text{O}$ and $\text{Pu}^{3+}_2(\text{C}_2\text{O}_4)_3 \cdot 10\text{H}_2\text{O}$ precipitates with the addition of oxalic acid, prior to calcination and storage as PuO_2 . It should be noted that the PUREX process, and subsequent separation of U & Pu has formed the most significant contribution to existing actinide stockpiles internationally, as this is the only process demonstrated on any industrial scale. Indeed, the development of future nuclear fuel cycles may necessitate the implementation of advanced MA recovery separations, significantly reducing the radiothermal output associated with high level waste (HLW) packages prior to geological disposal. Concepts for integrated MA reprocessing cycles include TALSPEAK (trivalent actinide-lanthanide separations by phosphorous-reagent extraction from aqueous complexes) and ALSEP (actinide-lanthanide separations process). However, as such MA solvent extraction processes only occur on small scale, in particular the separation of ^{237}Np as a precursor for ^{238}Pu production for radioisotope thermoelectric generators (RTG), and are not stockpiled in considerable quantities, it is largely the isotopes of separated Pu that are considered when discussing the stockpiling of actinides. Nevertheless, the development of advanced materials capable of acting as immobilization hosts for MA constituents remains an active area of research. For a comprehensive discussion of advanced reprocessing techniques pertaining to separation and partitioning of actinides, the reader is referred to [Nash and Nilsson \(2015\)](#).

Internationally, over 340 metric tonnes of Pu were declared as holdings to the International Atomic Energy Agency (IAEA) in 2017, reported under INFCIRC/549 obligations of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) ([Table 3](#)). Some nations have implemented a management strategy whereby Pu is reintegrated into the fuel cycle, through the fabrication of mixed oxide (MOX) fuels for reuse in existing reactor fleets. However, it is disconcerting to note that there exist some significant inventories of unirradiated Pu held at surface facilities. For example, the United Kingdom is in a unique position insofar as it holds an inventory of unirradiated Pu forecast to reach approximately 140 teHM (tonnes heavy metal equivalent) currently held under civil safeguards. There is a growing consensus that the indefinite storage of such material is not a satisfactory approach to long term management, as this is in contravention of security and non-proliferation safeguards. Hence, it is necessary to implement a strategy favoring either reuse or prompt disposal. For example, the bulk of the United Kingdom civil inventory could be utilized to support a campaign of energy production, through fueling a fleet of a three 1100 MWe Pressurized Water Reactors (PWR) for around 60 years. For a comprehensive review of Pu management policy in the context of the United Kingdom inventory, see ([Hyatt, 2020](#)). Nonetheless, the implementation of such a strategy is controlled by commercial appetite for MOX fuel, which has not been presented in the United Kingdom. In some instances, the accumulation and long-term storage of Pu presents an additional challenge of materials degradation. For example, it is estimated that ~ 5 t of United Kingdom Pu has succumbed to contamination through radiolytic degradation of the interim storage containment, hence regardless of the chosen pathway to disposition, it is advisable to develop long-term management strategy that can account for some portion, if not bulk inventories, to be designated as waste for disposal. There is a generic consensus that the end point of all ILW and HLW waste-streams, including actinide inventories, is geological disposal, subsequent to the design of a suitable wasteform.

Purpose and design requirements of nuclear wasteforms

As a precursor to the long-term disposition of nuclear waste inventories, it is necessary to convert, encapsulate or solidify the feed-stock into a product with improved chemical and physical stability than the source material, offering improved characteristics with respect to handling, transport and disposal. To convert material in such a manner is termed *immobilization*, with the product referred to as the *wasteform*. Demonstrated practices of immobilization include vitrification, cementation, bitumenization and ceramification ([Ojovan and Lee, 2005](#)). The conditioning of waste in such a manner is a precursor towards ultimate disposition, the foreseen pathway for which is disposal in an engineered facility. The geological disposal facility (GDF) concept is considered to be the only technically mature option for the permanent disposition of intermediate and high level wastes, whereas surface disposal routes are deemed applicable to wastes with low levels of activity (< 4 GBq/t α). The safety case for geological disposal is underpinned through the multi-barrier concept, a system whereby the encapsulation or immobilization matrix acts as the primary layer of containment,

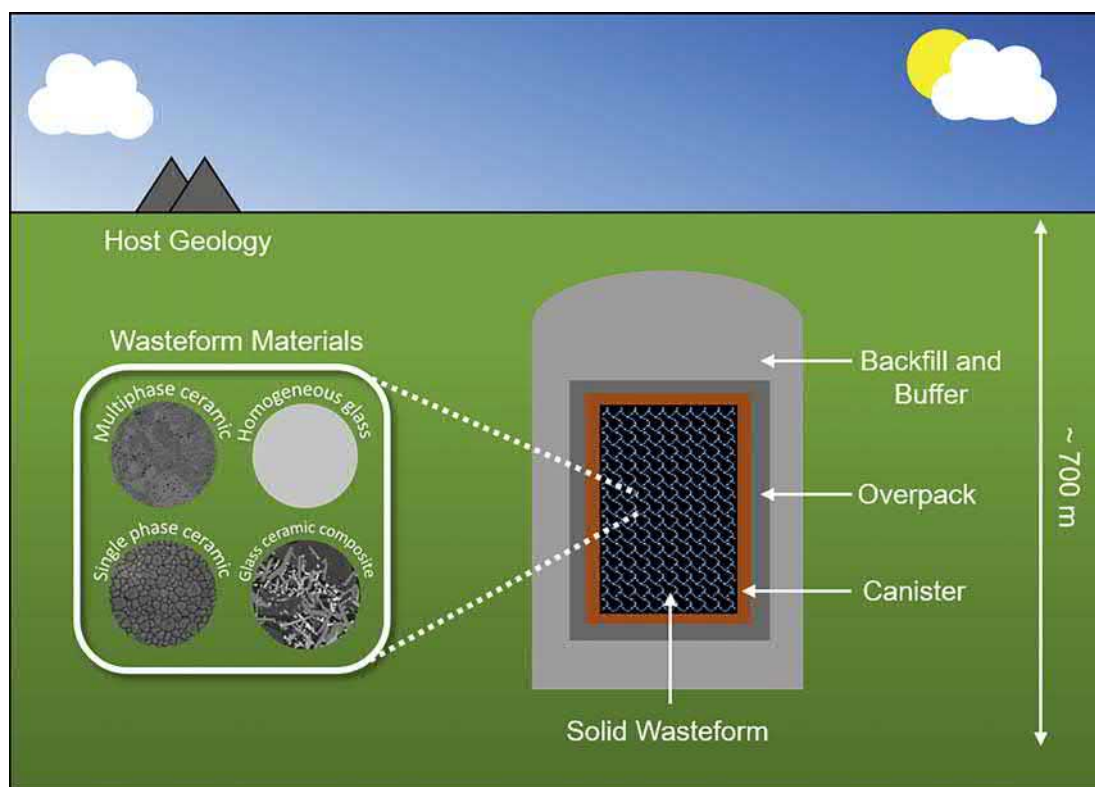


Fig. 1 Multibarrier concept for the immobilization and geological disposal of ILW, HLW and actinide-rich material.

which is subsequently over-packed and backfilled to retard water groundwater ingress, obviate corrosion and extend the lifetime of the package integrity, ultimately mitigating the release of radionuclide material into the near field environment, for such a time period that the overall activity of the waste is comparable to that of the natural uranium ore from which it was derived (Fig. 1). A sound disposal strategy stipulates that the choice of wasteform is tailored to suit the physical and chemical properties of the feedstock, as the immobilization matrix itself is considered to act as the primary containment, with the fundamental objective the reduce the potential for radionuclide dispersion into the near field environment. A set of generic wasteform design criteria is outlined below. For HLW wastes, typically concentrated raffinate derived from the reprocessing of SNF matrices, the thermal treatment of choice is calcination prior to vitrification in a borosilicate glass matrix (Table 4). Glass is a suitable candidate on the basis that a wide array of constituent elements comprising HLW are soluble in the glass matrix, either through substitution within the glass network (Si^{2+} , B^{3+} , P^{5+}) or as network modifiers (Cs^+ , Na^+ , Li^+ , Mg^{2+}). The vitrification process is continuous, utilizes relatively low melting temperatures ($\sim 1050^\circ\text{C}$) and is relatively insensitive to the nature of the wastestream, allowing the glass composition to be easily modified to accommodate small variations in feedstock. Upon cooling, the wasteform is a homogeneous product with a relatively high chemical durability ($\text{NL}_{\text{Si}} = 0.011 \pm 0.004 \text{ g m}^{-2} \text{ day}^{-1}$ based on 1:3 Magnox-ThORP blend with 25% simulant waste loading at $T = 23^\circ\text{C}$ and $\text{pH} = 8$; see (Cassingham et al., 2015). Despite the efficacy of borosilicate glass to accommodate HLW, the solubility of actinide-rich feedstocks in glass is relatively low. Using Ce as a surrogate for Pu, (Lopez et al., 2003) determined that Ce solubility in borosilicate glass was a function of temperature and $\text{Ce}^{3+}/\text{Ce}_{\text{total}}$, increasing from 0.25 wt.% at 1100°C to 15 wt.% at 1400°C , corresponding to an increase in the $\text{Ce}^{3+}/\text{Ce}_{\text{total}}$ ratio from 0.5 to 0.9. This reduction protocol was utilized by (Cachia et al., 2006) to increase the solubility of PuO_2 from $\sim 2 \text{ wt.}\%$ to $\sim 4 \text{ wt.}\%$. However, through the success of the SYNROC wasteform, alternative crystalline wasteform materials are favored for actinide immobilization, as it is considered these will offer superior wasteloading and durability.

Table 4 Compositions of selected nuclear waste glasses (wt.%).

Waste glass	SiO_2	B_2O_3	Al_2O_3	CaO	MgO	Na_2O	Other	Waste loading
R7T7 (France)	47.2	14.9	4.4	4.1	–	10.6	18.8	<28
Magnox (United Kingdom)	47.2	16.9	4.8	–	5.3	8.4	17.4	< 25
DWPF (United States)	49.8	8.0	4.0	1.0	–	8.7	27.1	<33
PAMELA (Germany)	52.7	13.2	2.7	4.6	2.2	5.9	18.7	<30

Ceramic wasteforms for actinide immobilization

Multiphase SYNROC

Despite the suitability of borosilicate glasses for the vitrification of chemically heterogeneous HLW waste streams, single and multi-phase ceramic phase assemblages are generally considered to offer superior durability in the geological environment, alongside, in some instances, greater wasteloading of actinides. Ceramics offer a high performance solution for separated elements, as an alternative to vitrified wasteforms, and pertain to the substitution of the waste cations into a specific lattice sites within a crystalline structure, whilst retaining long-range order. Hence, tailoring ceramic materials for the effective immobilization of radionuclides requires additional considerations with respect to the relative size and accessible valence state compatibility between waste cations, and the targeted substitution site within the host structure. Such a design philosophy essentially allows the waste constituents to adopt the physical and chemical properties of the host material, hence ceramic immobilization is typically informed through the study of natural mineral analogs in the wider environment. Mineral specimens that remain present in the environment, having exhibited resistance to natural weathering processes over 10^6 years, are therefore excellent candidates from which to design synthetic ceramic materials for actinide immobilization. Ringwood et al. (1979) developed and proposed SYNROC (*synthetic-rock*) as alternative candidate ceramic wasteform for HLW and actinides. The SYNROC wasteform is a collection of thermodynamically stable titanate ceramic phases, chosen due to their extensive geochemical stability in the natural environment, and ability to accommodate the vast majority of elements comprising HLW waste streams. The dominant phases present in most variations of SYNROC are hollandite, zirconolite, perovskite and rutile. Hollandite, ideally $\text{Ba}_x(\text{Al,Ti})_{2x}\text{Ti}_{8-2x}\text{O}_{16}$, is predominantly included as a host for processing contaminants and fission products. The hollandite structure is comprised of a framework of corner and edge sharing TiO_6 and AlO_6 polyhedra parallel to the c-axis of the tetragonal unit cell, allowing large mono or divalent A-site cations (Na^+ , Ba^+ , Cs^+ , Sr^{2+} , Pb^{2+}) to reside in the tunnel centers, whilst the B and C sites can accommodate a variety of di- tri- and pentavalent species such as Mg^{2+} , Cr^{3+} , Cu^{2+} , Zn^{2+} and Sb^{5+} (Lutze and Ewing, 1988). Zirconolite, prototypically $\text{CaZrTi}_2\text{O}_7$, is the most durable of the SYNROC phases, and is the primary tri- and tetravalent actinide-bearing host. The anion deficient fluorite superstructure of zirconolite permits substitution of tetravalent actinides U^{4+} , Pu^{4+} , and Th^{4+} for Ca^{2+} in eightfold coordination, alongside moderate substitution for Zr^{4+} and a range of di- tri- and pentavalent substitution for Ti^{4+} (e.g., Mg^{2+} , Fe^{3+} , Al^{3+} and Nb^{5+}). The structure of perovskite (ideally CaTiO_3) permits a wide array of Ca^{2+} substitution, including Sr^{2+} , Ba^{2+} , REE^{3+} , Cm^{3+} , and Pu^{3+} , hence is included in SYNROC primarily to uptake Sr and minor actinides. The partitioning of key HLW elements in the constituent SYNROC phases is summarized in Table 5. Comprehensive discussions of SYNROC mineralogy, fabrication and disposability are provided by (Ringwood, 1985; Lutze and Ewing, 1988).

Wasteloading: Wasteform chemistry should be tailored such that solubility of waste material in the host phase is as extensive as reasonably possible, without promoting the formation of deleterious secondary phases, or breaching criticality limits. This will alleviate GDF volume requirements through the reduction in the number of overall waste packages produced.

Chemical flexibility: The ideal immobilization matrix should demonstrate the ability to accommodate variations in feedstock chemistry without the formation of detrimental accessory phases that may compromise performance in the disposal environment. This may include fluctuations in processing contaminants and additives. Multiphase ceramic, or glass-ceramic composite materials may be preferable to single phase ceramic materials in this respect.

Synthesis: Wasteform processing should favor rugged and established processing techniques, promoting compatibility with existing technology. Where possible, exotic syntheses routes and high temperature thermal treatments should be avoided, so as to prolong the lifetime of equipment, and suppress the potential for volatilization. These processes should also consider the environmental and economic impact of any secondary effluents that may be produced.

Aqueous durability: Leaching of the wasteform matrix in saturated groundwater compositions is considered to be the primary mechanism by which radionuclides begin to migrate into the near field environment. The wasteform should exhibit high chemical durability under such conditions, with the formulation optimized in laboratory trials to ensure the leach resistance is maximized. The aqueous durability of the chosen wasteform may be established through a variety of field tests, establishing a measure of the corrosion rate. However, the selection of wasteforms with existing natural analogs is preferable, as these specimens provide a true measure of corrosion resistance under geological timescales, and therefore act as a useful reference to support the safety case for geological disposal.

Radiation tolerance: The wasteform should exhibit resistance against amorphization due to the effects of self-irradiation. The formation and accumulation of decay products within the bulk wasteform may impact the overall mechanical integrity of the waste package, leading to volume swelling and cracking. This may, in turn, decrease the long-term durability by increasing the available surface area available for leaching.

Table 5 Distribution of key HLW elements in SYNROC phases.

Hollandite	Zirconolite	Perovskite	Metal
Cs^+	U^{4+}	Na^+	Ru
Rb^+	Th^{4+}	Sr^{2+}	Tc
K^+	Pu^{4+}	Pu^{3+}	Mo
Ba^{2+}	Cm^{4+}	Am^{3+}	Ni
Fe^{2+}	Am^{4+}	Cm^{3+}	Pd
Cr^{3+}	Np^{4+}	Np^{3+}	Rh
Ni^{2+}	Sr^{2+}	—	Te
Mo^{4+}	—	—	S

Table 6 Mineralogy of various SYNROC formulations.

Formulation	Ideal mineral composition (wt.%)						
	Zirconolite	Hollandite	Rutile	Perovskite	Alloy phases	Spinel	Nepheline
SYNROC-C	30	30	15	20	5	—	—
SYNROC-D	19	3	—	15	—	46	17
SYNROC-E	7 ^a	5	79	7	—	—	—
SYNROC-F	90	5	5	—	—	—	—
SYNROC-Z	10	35	20	30	5	—	—

^aPlus approximately 2% pyrochlore.

The conventional method of SYNROC fabrication involves the addition of ~2% Ti metal, due to the reducing conditions imposed during hot pressing. As the oxidation of the Ti reduces some portion of the Ti^{4+} inventory to Ti^{3+} , rutile, and related Maggelli phases TiO_{2-n} are included to provide an excess of Ti^{4+} , thus preventing the stabilization of Ti-poor phases such as $CsAlTiO_4$ that may be detrimental to wasteform performance in the disposal environment (Ringwood, 1985). The mineralogy of SYNROC may be varied, such that the relative proportions of the host ceramic phases are tailored to complement specific waste-streams (Table 6).

For example, whereas SYNROC-C was developed for as an alternative to borosilicate glass as an immobilization matrix for HLW waste streams, SYNROC-D is a unique variation developed specifically for the immobilization of defense wastes produced at the Savannah River Plant, favoring the replacement of hollandite with nepheline, and an inert spinel phase. Confidence in the SYNROC approach as a viable alternative to glass was largely supported by field durability studies. Development work by (Oversby and Ringwood, 1981) compared the release rates of key elements from 0.5 g disks of SYNROC and PNL-76-68 borosilicate glass (loaded with 33% simulant HLW), at both 85 °C and 200 °C. Based the concentrations of Ca, Cs and U in the distilled H_2O leaching solution, it was determined that whilst the PNL-76-68 wasteform had average leach rates of 1.4 and 8.9 $g\ m^{-2}\ day^{-1}$ at 85 °C and 200 °C, respectively, the upper limit of the SYNROC leach rate was several orders of magnitude lower at $<0.005\ g\ m^{-2}\ day^{-1}$. Subsequent work by (Ringwood et al., 1981) confirmed that whilst the retention of Cs in SYNROC was 500 times greater than PNL-76-68 at 95 °C, over a period between 10 and 30 days, the leach rate of U from SYNROC was 10^5 times lower than for corresponding borosilicate formulations (Fig. 2). The long term durability of SYNROC has also been established, with monoliths of SYNROC-C containing 10 wt.% simulated PW-4b-D HLW, containing 0.62 wt.% ^{239}Pu subjected to leaching in deionized water at 70 °C for a total of 2472 days. The normalized release rate of Pu from these samples was observed to be $\sim 5 \times 10^{-5}\ g\ m^{-2}\ day^{-1}$ (Smith et al., 1997). The radiation stability of SYNROC wasteforms is largely informed by in situ (i.e., doping with short lived actinides such as ^{238}Pu) and/or ex situ (i.e., charged ion-irradiation or gamma/neutron irradiation) response testing of the constituent actinide

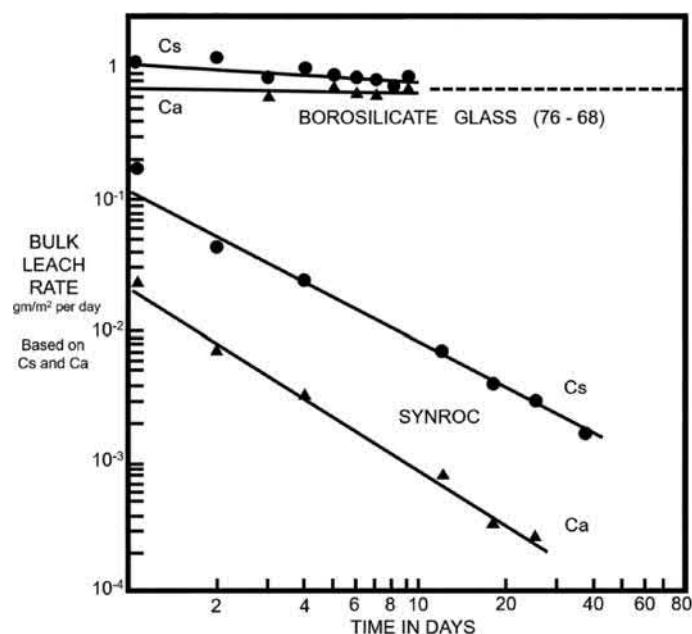


Fig. 2 Bulk leach rates of Cs and Ca from SYNROC and PNL-76-68 borosilicate glass. From Ringwood AE, et al. (1981) Immobilization of high-level nuclear reactor wastes in synroc: A current appraisal. *Nuclear Waste Management* 2: 287–305.

bearing phases, e.g., zirconolite and perovskite, which are discussed in the following sections. However, several pertinent examples of SYNROC radiation testing have been reported. For example specimens of SYNROC-C were subjected to neutron irradiation dose of $4.8 \times 10^{20} \text{ n cm}^{-2}$, corresponding to a simulated aging of 2×10^5 year (Woolfrey et al., 1982). Specimens remained physically intact, yet with micro-cracking observed at volume expansions $\geq 4\%$. The microstructure of SYNROC-C doped with ^{238}Pu and ^{244}Cm , with ages in excess of 20 years and estimated experienced dose of $3 \times 10^{19} \alpha \text{ g}^{-1}$, were characterized by (Hambley et al., 2008). Micro-cracking was present across hollandite and rutile-rich regions of the microstructure yet did not extend across regions of zirconolite-perovskite intergrowths. As Pu was exclusively located, as expected, in the zirconolite and perovskite phases, the resultant swelling of these grains was proposed as the cause for the cracking. For a comprehensive review of radiation effects in SYNROC, and related titanate phases, the reader is referred to (Weber et al., 1998).

Single phase wasteforms

Whilst the original motivation for ceramic wasteform development was to develop alternative matrices to conventional borosilicate glasses, for the conditioning of heterogeneous reprocessing wastes, the past several decades have seen some nations strengthen reprocessing efforts, affording greater quantities of separated Pu and other actinides. Moreover, strategic arms reductions between the United States and the former Soviet Union has afforded some 100 metric tonnes of Pu designated as waste for disposal. Hence, the nature of wasteform development has evolved to favor the study of designated single-phase ceramic materials as immobilisation matrices for high purity actinide inventories. This rationale has seen many crystalline phases proposed for actinides (Table 7), justified by high durability and radiation stability, informed by natural analogs and laboratory field tests. For brevity, we shall provide a brief discussion of the crystallography and chemical properties of the titanate phases: pyrochlore, zirconolite, brannerite and perovskite.

Pyrochlore ($\text{A}_2\text{B}_2\text{O}_7$)

The pyrochlore family of minerals conforming to $^{\text{VIII}}\text{A}_2^{\text{VI}}\text{B}_2^{\text{IV}}\text{X}_6^{\text{IV}}\text{Y}_1$ stoichiometry have been proposed as wasteforms for the immobilization of actinides, including Pu, either through solid solution with the 16(d) B^{4+} site e.g., $\text{Gd}_2\text{Ti}_{2-x}\text{Pu}_x\text{O}_7$ or replacement of the 16(c) A site e.g., $\text{CaPuTi}_2\text{O}_7$. The cubic pyrochlore structure (ideally $\text{A}_2\text{B}_2\text{O}_7$ —space group $\text{Fd}\bar{3}m$) is described as a defect fluorite superstructure, related to the ideal AX_2 fluorite unit cell with one eighth of the oxygen atoms replaced by ordered vacancies on the anion sublattice. Over 500 distinct compositional variations of this phase have been reported through control of A and B-site cations, hence single phase pyrochlore materials certainly offer the requisite structural flexibility to act as nuclear wasteforms for actinides. Typically, the A-site is able to incorporate formal valence states between +1 and +6, and the B-site between +3 and +6, although the mineralogy of pyrochlore typically favors the occupation of the eightfold coordinated A site with large tri- and tetravalent actinide and rare earth cations (e.g., Th^{4+} , U^{4+} , Gd^{3+} , Y^{3+} , Sm^{3+}) and smaller cation modifiers on the B-site (Zr^{4+} , Ti^{4+} , Ta^{5+} , Sb^{5+}). The long term durability of pyrochlore is supported by the existence of naturally altered specimens, many of which exist with primordial U/Th inventories, up to 30 wt.% and 9 wt.%, respectively, although the durability of these analogs is thought to be slightly lower than that of zirconolite (Lumpkin, 2001). Wang et al. (2019) tested the durability of $(\text{Gd},\text{Sm})_2(\text{Zr},\text{Ce})_2\text{O}_7$ pyrochlore materials through static leaching in distilled water for 7 day at 90°C , with the leaching rates of all elements between 10^{-6} and $10^{-7} \text{ g m}^{-2} \text{ day}^{-1}$. Icenhower et al. (2000) investigated the corrosion resistance of $\text{Lu}_2\text{Ti}_2\text{O}_7$ and $\text{Gd}_2\text{Ti}_2\text{O}_7$ through dissolution under dynamic SPFT conditions at $\text{pH} = 2$, confirming that even at under highly acidic conditions, dissolution rates remained low, in the range 1.3×10^{-3} – $7.0 \times 10^{-4} \text{ g m}^{-2} \text{ day}^{-1}$. The stability and radiation response of pyrochlore-structured phases is dictated by the relative size ratio of A and B-site cations, with the cubic-forming region determined experimentally to be $1.46 < r_A/r_B < 1.78$. Compositions for which $r_A/r_B < 1.46$ form destabilize to form the defect fluorite structure, in which A and B-site cations become randomly distributed, and oxygen vacancies become disordered along the anion sublattice (Lang et al., 2010). (Reid et al., 2012) synthesized the $\text{Gd}_2\text{Zr}_{2-x}\text{Ce}_x\text{O}_7$ with the view to simulate Pu^{4+} immobilization, identifying three distinct phases corresponding to pyrochlore ($x < 0.25$), defect fluorite ($0.5 < x < 0.75$) and an incommensurate C*-type structure for $x = 1.00$. The response to radiation is significantly improved for Zr based pyrochlores, as is shown in Fig. 3 for the $\text{Gd}_2\text{Ti}_{2-x}\text{Zr}_x\text{O}_7$ binary system.

Table 7 Candidate ceramic wasteforms for actinide immobilization.

Ceramic phase	Ideal formula	Space group	Unit cell parameters ($\text{\AA}$, $^\circ$)	Calculated density (g cm^{-3})
Zirconolite	$\text{CaZrTi}_2\text{O}_7$	C2/c	$a = 12.44$; $b = 7.27$; $c = 11.38$; $\alpha = \gamma = 90$; $\beta = 100.56$	4.44
Pyrochlore (Zr)	$\text{Gd}_2\text{Zr}_2\text{O}_7$	Fm $\bar{3}m$	$a = b = c = 5.27$; $\alpha = \beta = \gamma = 90$	6.91
Pyrochlore (Ti)	$\text{Gd}_2\text{Ti}_2\text{O}_7$	Fd $\bar{3}m$	$a = b = c = 10.18$; $\alpha = \beta = \gamma = 90$	6.57
Zirconia	ZrO_2	P2 $_1$ /c	$a = 4.15$; $b = 5.21$, $c = 5.32$, $\alpha = \gamma = 90$; $\beta = 99.21$	5.82
Zircon	ZrSiO_4	I4 $_1$ /amd	$a = b = 6.60$; $c = 5.98$; $\alpha = \beta = \gamma = 90$	4.66
Monazite	CePO_4	P2 $_1$ /c	$A = 6.77$; $b = 7.00$; $c = 6.44$; $\alpha = \gamma = 90$; $\beta = 103.42$	5.26
Xenotime	YPO_4	I4 $_1$ /amd	$a = b = 6.89$; $c = 6.03$; $\alpha = \beta = \gamma = 90$	4.26
Brannerite	UTi_2O_6	C2/m	$a = 9.81$; $b = 3.77$; $c = 6.93$; $\alpha = \gamma = 90$; $\beta = 118.96$	6.36
Kosnarite	$\text{KZr}_2(\text{PO}_4)_3$	R3c	$a = b = 8.73$; $c = 23.20$; $\alpha = \beta = 90$; $\gamma = 120$	4.38
Perovskite	CaTiO_3	Pbnm	$a = 5.38$; $b = 5.44$; $c = 7.64$; $\alpha = \beta = \gamma = 90$	4.04

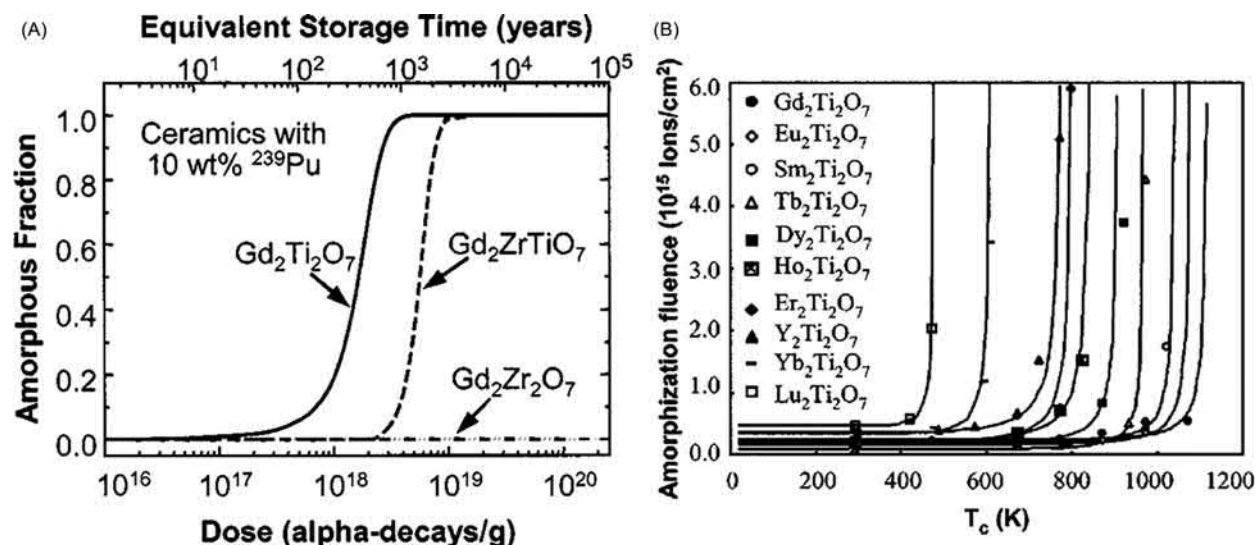


Fig. 3 Dependence of amorphisation dose rate on x parameter in $Gd_2Ti_{2-x}Zr_xO_7$ pyrochlore (A) and choice of REE $^{3+}$ cation (B). Reproduced from Ewing RC, Weber WJ and Lian J (2004) "Nuclear waste disposal-pyrochlore ($A_2B_2O_7$): Nuclear waste form for the immobilization of plutonium and 'minor' actinides". *Journal of Applied Physics* 95(11): 5949–5971, with the permission of AIP Publishing.

Zirconolite ($CaZrTi_2O_7$)

Zirconolite, nominally $CaZrTi_2O_7$, is a relatively rare accessory mineral located in a variety of geological formations, and is included as the primary actinide bearing phase in the SYNROC wasteform. However, in recent years there has been a renewed interest in the development of single phase zirconolite ceramic materials for the immobilization of stockpiled Pu inventories, and zirconolite glass-ceramic composites for the conditioning of Pu residues, in the United Kingdom. Zirconolite has a demonstrated affinity for extensive substitution of several actinides ($Pu^{3+/4+}$, Np^{3+} , $U^{4+/6+}$, Th^{4+}), rare earth (Y^{3+} , Sm^{3+} , Nd^{3+} , Gd^{3+}) and alkaline/transition (Mg^{2+} , Al^{3+} , Fe^{3+} , Cr^{3+}) elements. Zirconolite, related to $A_2B_2O_7$ pyrochlore by a compression along the (111) plane, is an anion deficient fluorite-type structure, with cation ordering on planes parallel to the superstructure c -axis. A projection of the zirconolite unit cell is displayed in Fig. 4. The zirconolite-2M parent structure, stabilized over the compositional range $CaZr_xTi_{3-x}O_7$ ($0.8 < x < 1.3$), crystallizes with monoclinic symmetry in the space group ($C2/c$, $Z = 8$). The unit cell forms a lamellar structure comprised of alternating layers of Ca^{2+}/Zr^{4+} polyhedra, coordinated eight- and sevenfold to oxygen, respectively, interspaced 1:1 along the (001) plane with sheets of Ti^{4+} octahedra arranged in a hexagonal tungsten bronze (HTB) type topology.

Ti^{4+} is present in three distinct crystallographic sites, two of which, Ti(I) and Ti(III), are contained in sixfold coordination, whilst Ti(II) forms an unusual trigonal bipyramidal TiO_5 site, located off-center from the six membered ring, statistically 50% occupied. The chemistry of zirconolite permits solid solution of actinides and charge compensator species across the five distinct cation receptor sites ^{VIII}Ca , ^{VII}Zr , $^{VI}Ti(I)$, $^{VI}Ti(II)$ and $^{VI}Ti(III)$, and hence allows for accommodation of multiple cations in conjunction. Several substitution regimes have been proposed and demonstrated for both tri- and tetravalent actinide species. For example, (Begg et al., 2001)

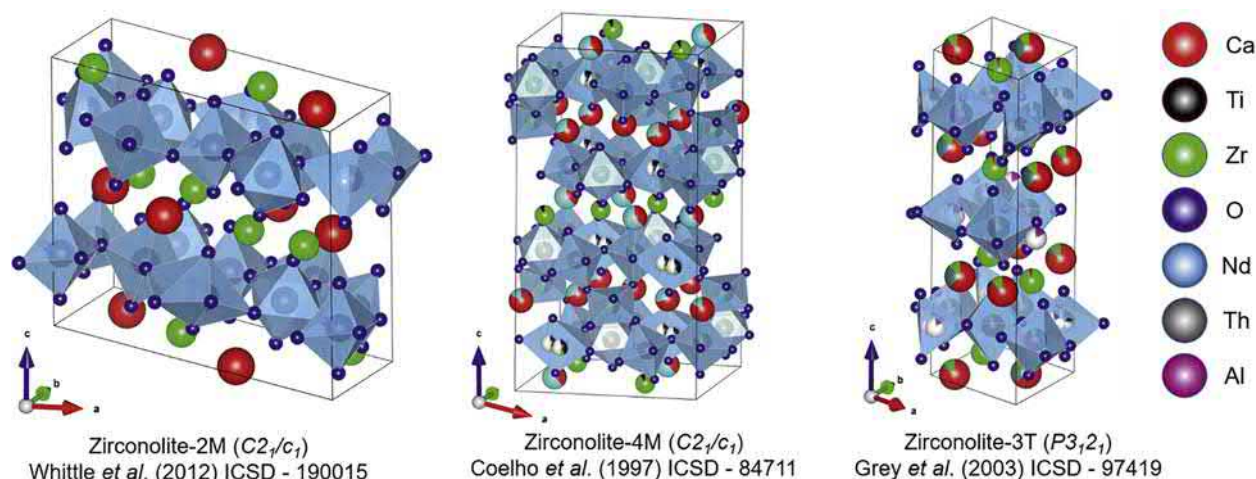


Fig. 4 Crystal structures of zirconolite-2M, 4M and 3T polytypes.

demonstrated that Pu^{4+} may be accepted in solid solution via isovalent substitution for Zr^{4+} , i.e., $\text{CaZr}_{1-x}\text{Pu}_x\text{Ti}_2\text{O}_7$. Alternatively, it has been demonstrated that heterovalent tri- and tetravalent substitution for Ca^{2+} is also feasible, with a lower valence charge balancing species such as Al^{3+} accommodated across Ti^{4+} sites, e.g., $\text{Ca}_{1-x}\text{Pu}_x\text{ZrTi}_{2-2x}\text{Al}_{2x}\text{O}_7$ and $\text{Ca}_{1-x}\text{Pu}_x\text{ZrTi}_{2-x}\text{Al}_x\text{O}_7$ for Pu^{4+} and Pu^{3+} , respectively (Vance et al., 1994). The progressive substitution of foreign elements, depending on the chosen substitution regime and processing conditions, has been shown to stabilize a variety of crystallographically distinct polytype variants of zirconolite, most notably zirconolite-4M and zirconolite-3T (formerly recognized as zirkelite (Bayliss et al., 1989)). Successive zirconolite polytypes are characterized by variations in stacking sequence of Ca/Zr and HTB layers. For example, the structure of zirconolite-4M resembles intergrowths of zirconolite-2M and pyrochlore-type modules, resulting in lattice expansion along the c-axis and stacking of four Ca/Zr—HTB interlayers to comprise the unit cell. Isovalent substitution of U^{4+} , Ce^{4+} and Pu^{4+} for Zr^{4+} targeting $\text{CaZr}_{1-x}(\text{Ce}, \text{U}, \text{Pu})_x\text{Ti}_2\text{O}_7$ has been shown to favor zirconolite-4M as the dominant actinide phase above a threshold of $x = 0.15$, with further substitution to $x = 0.50$ yielding the pyrochlore phase (Begg et al., 2001; Vance et al., 2002; Blackburn et al., 2020a). Interestingly, it was recently reported that the $\text{CaZr}_{1-x}\text{Th}_x\text{Ti}_2\text{O}_7$ did not give rise to the zirconolite-4M intermediate phase, rather a Th-rich pyrochlore phase was preferred in the interval $0.10 \leq x \leq 0.50$, forming a single phase at $x = 0.60$ (Blackburn et al., 2021). Targeting tetravalent substitution for Ca^{2+} , coupled with a charge balancing species has been shown to extend the formation range of zirconolite-2M. For example, (Vance et al., 2002) confirmed the formation of single phase 2M when targeting $\text{Ca}_{0.80}\text{U}_{0.20}\text{ZrTi}_{1.60}\text{Al}_{0.40}\text{O}_7$, with complementary data by (Deschanel et al., 2006) targeting, $\text{Ca}_{0.87}\text{Pu}_{0.13}\text{ZrTi}_{1.73}\text{Al}_{0.30}\text{O}_7$ confirming the 2M structure. Using Ce as a surrogate, (Blackburn et al., 2020b) confirmed the 2M polytype could be stabilized up to $x = 0.35$ in the $\text{Ca}_{1-x}\text{Ce}_x\text{ZrTi}_{2-2x}\text{Cr}_{2x}\text{O}_7$, however, partial Ce^{3+} speciation promoted the formation of a Ce-rich perovskite phase. Furthermore, (Gilbert et al., 2010) reported the formation of the 3T polytype for the corresponding $\text{Ca}_{1-x}\text{Pu}_x\text{ZrTi}_{2-2x}\text{Fe}_{2x}\text{O}_7$ solid solution. Incorporation of trivalent actinide species may be accomplished through equimolar substitution of a trivalent charge balancing species such as Al^{3+} , extending the solid solution range over which the zirconolite-2M polytype is stabilized. For example, the $\text{Ca}_{1-x}\text{Nd}_x\text{ZrTi}_{2-x}\text{Al}_x\text{O}_7$ solid solution was fabricated by Ma et al., confirming that zirconolite-2M remained single phase for $x \leq 0.6$, after which the orthorhombic zirconolite-3O polytype was produced (Ma et al., 2018). The radiation stability of zirconolite is comparable to that of $\text{Gd}_2\text{Ti}_2\text{O}_7$ through doping with 3 wt.% ^{244}Cm (Weber, Wald and Matzke, 1986), with an amorphisation dose corresponding to $0.47 \times 10^{16} \alpha \text{ mg}^{-1}$ causing bulk swelling between 6–6.5 vol.%. Similar results have been reported by ion irradiation with $1.3 \times 10^{21} \text{ keV cm}^{-1} \text{ Au}^+$ (Deschanel et al., 2014). A comprehensive report on the radiation response of zirconolite was provided by (Strachan et al., 2008), concluding that a dose rate of $2.6 \times 10^{18} \alpha/\text{g}$ was necessary to amorphize ^{238}Pu doped zirconolite. Furthermore, the forward dissolution rate at $\text{pH} = 2$ was experimentally determined as $1.7 \times 10^{-3} \text{ g m}^{-2} \text{ day}^{-1}$, showing no dependence on the amount of induced radiation damage. Other studies have concluded that the leaching of elements is not affected by amorphisation, whereas the release rates from fully amorphous pyrochlore specimens, have in some instances, been shown to increase by a factor of 10–15 (Smith et al., 2003). Durability studies have demonstrated that release rates of actinide surrogates from zirconolite specimens are typically of the order $10^{-6} \text{ g m}^{-2} \text{ day}^{-1}$. Zhang et al. (2017) determined that the release of Ce from monoliths of zirconolite, with nominal composition $\text{CaZr}_{0.70}\text{Ce}_{0.30}\text{Ti}_2\text{O}_7$, was as low as $2.3 \times 10^{-6} \text{ g m}^{-2} \text{ day}^{-1}$ over a 42 day period in deionized H_2O . Similar results were reported by (Wen et al., 2015) with Ce leach rates of $2.3 \times 10^{-6} \text{ g m}^{-2} \text{ day}^{-1}$ over a 28 day period. Exceptionally low Gd^{3+} release rates of $4.72 \times 10^{-7} \text{ g m}^{-2} \text{ day}^{-1}$ were recently reported by Zhang et al. (2019) from $\text{Ca}_{1-x}\text{Hf}_{1-x}\text{Gd}_x\text{Ti}_2\text{O}_7$ zirconolite, with Gd included as a trivalent actinide surrogate. A summary of pH dependent U leaching from zirconolite, relative to other candidate wasteforms, was provided by Weber et al. (2009) and is summarized in Fig. 5.

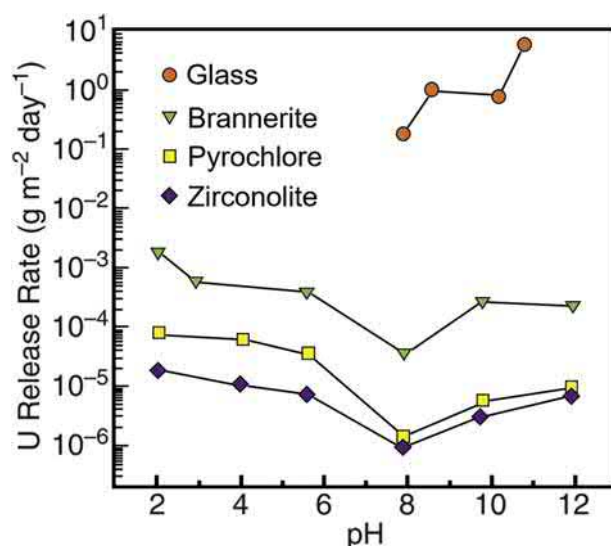


Fig. 5 Summary of U release rates ($\text{g m}^{-2} \text{ day}^{-1}$) from a selection of candidate wasteforms. From Weber WJ, Navrotsky A, Stefanovsky S, Vance ER, and Vernaz E (2009) Materials science of high-level nuclear waste immobilization. *MRS Bulletin* 34 (01): 46–52.

Brannerite (UTi₂O₆)

Brannerite is a common accessory U-bearing accessory mineral, with a prototypical composition of UTi₂O₆, and in some areas is the principle uranium ore. As natural brannerites contain ~55 wt.% U, ceramics based on the UTi₂O₆ system may be suitable hosts for wastes containing a high fraction of actinides, such as MOX residues (Bailey et al., 2018). UTi₂O₆ is a layered structure comprised of anatase-type sheets of edge-sharing distorted TiO₆, with chains of edge-sharing UO₆ octahedra connecting adjacent Ti-O layers. The ideal UTi₂O₆ structure crystallizes in monoclinic symmetry (space group C2/m) and is only formed under low partial oxygen pressure, in order to maintain U⁴⁺ valence. The preparation of UTi₂O₆ in air, to the correct stoichiometry, has been shown to produce a mixture of U₃O₈ and TiO₂. The U⁴⁺ site can be completely replaced by isovalent Ce/Th substitution, e.g., the isostructural thorutite structure ThTi₂O₆, however, the counterpart Ce brannerite was shown to be non-stoichiometric, viz Ce_{0.975}Ti₂O_{5.95}, with cerium and oxygen vacancies, as determined by powder neutron diffraction (Stennett et al., 2012). The incorporation of higher valence actinide species in brannerite is permitted through the co-substitution of lower valence charge balance species. For example, U_{1-x}Y_xTi₂O₆ and Th_{1-x}Y_xTi₂O_{6-δ} solid solutions were synthesized in air at 1400 °C by (James et al., 2003) and it was determined that the accommodation of Y³⁺ was greater in UTi₂O₆, due to the formation of charge balancing U⁵⁺ centers. Y³⁺ accommodation in ThTi₂O₆ is considerably lower, permitted by the formation of oxygen vacancies due to the stability of Th⁴⁺ in air. The chemical durability of brannerite is consistently lower than that of related zirconolite and pyrochlore phases, yet 3–4 orders of magnitude more durable than some glass matrices. Amorphous specimens of natural brannerite from the El Cabril mine (Spain) have been recrystallized by annealing at a range of temperatures by (Zhang et al., 2006) prior to durability testing. U release rates in weakly acid conditions (pH = 4) at 30 °C were observed to be ~1 × 10⁻³ g m⁻² day⁻¹ when annealed at 500 °C. Synthetic UTi₂O₆ was produced and dissolved over the pH range 2–12 by (Roberts et al., 2000) at 25 °C, 50 °C and 75 °C. The dissolution rate increased with temperature, yet the magnitude of the increase was observed to vary with pH, reaching a minimum near pH = 8. Whilst the dissolution rates for brannerite were in some instances a factor of 30 greater than zirconolite under the same conditions, the release rates for U were moderately low, between 10⁻³ and 10⁻⁵ g m⁻² day⁻¹. Nevertheless, brannerite may act as a buffer phase to accommodate excess actinides that may exceed the solubility limit in targeted phases, and its inclusion not expected to significantly impact the overall performance of wasteform materials for actinides in the geological environment.

Perovskite (CaTiO₃)

Perovskite (ABO₃) structured materials are extensively studied, and have a wide field of applications, particularly in the development of photonic functional materials, and as a primary constituent of the SYNROC wasteform. Oxide perovskites have the general structure ABO₃, in which BO₆ octahedra form a three-dimensional corner sharing framework, with large A-site cations (i.e., Ca²⁺, Sr²⁺) residing in the cavity, 12-fold coordinated with oxygen. Although ABO₃ perovskites ideally have cubic symmetry in the Pm-3m space group, most perovskites are distorted due to tilting of BO₆ octahedra, caused by variation in cation radii r_A and r_B, and may undergo reversible, phase dependent transitions. For example, the CaTiO₃ unit cell distorts with orthorhombic symmetry in the space group Pbnm at room temperature, to tetragonal I4/mcm at 1225 °C, with a further transition to cubic Pm-3m observed at 1361 °C. The binary system Ca_{1-x}Sr_xTiO₃ exhibits complete solid solubility, with a structural transformation from Pbnm to the I4/mcm structure at x = 0.6, followed by transformation to Pm-3m with additional Sr²⁺ content (Qin et al., 2002). CaTiO₃ has been proposed as a host phase for Pu³⁺ and Pu⁴⁺ due to high radiation tolerance. (Ca_{0.9}Nd_{0.1})(Ti_{0.9}Al_{0.1})O₃ was synthesized by (Davoisne et al., 2011) with Nd³⁺ employed as a structural simulant for Pu³⁺, and irradiated with a dose of 2 MeV Kr⁺ at 5 × 10¹⁵ ions cm⁻² with the view to simulate Pu recoil damage during α-decay. A well-defined damage region of 0.8–1.1 μm was observed, however only some of this was amorphous; the number of displacements per atom (dpa) was estimated at around 4 dpa (Fig. 6).

The incorporation of Ce³⁺ and Ce⁴⁺ within the CaTiO₃ was investigated by (Begg et al., 1998b) with Ce acting as a structural simulant for Pu. Ce³⁺ was successfully incorporated within the Ca²⁺ site with Al³⁺ charge compensation targeting Ca_{0.9}Ce_{0.1}Ti_{0.9}Al_{0.1}O₃ under both oxidizing and reducing conditions, with the partial formation of Ce⁴⁺ formed in air charge balanced by Ti⁴⁺ vacancies. Ce⁴⁺ was fully accommodated into single phase when targeting Ca_{0.9}Ce_{0.1}Ti_{0.8}Al_{0.2}O₃ under oxidizing conditions, however Ce was distributed as 80% Ce⁴⁺, with the excess negative charge apparently compensated by oxygen vacancies. Moreover, Ce³⁺ was fully accommodated into single phase Ca_{0.9}Ce_{0.1}TiO₃ with lower valence species available for charge compensation, demonstrating the formation of cation vacancies. These solid solutions were later verified by (Begg et al., 1998a), using Pu^{3+/4+}, in all instances minor undigested PuO₂ (< 2%) was evidenced. Investigations of Pu incorporation within Ca_{0.9}Pu_{0.1}Ti_{0.9}Al_{0.1}O₃ (targeting Pu³⁺), Ca_{0.9}Pu_{0.1}Ti_{0.8}Al_{0.2}O₃ (targeting Pu⁴⁺) and Ca_{0.9}Pu_{0.1}TiO₃ (targeting Pu³⁺) yielded near single phase perovskite when sintering in air, with Pu uniformly present as Pu⁴⁺; annealing in 3.5% H₂/N₂ resulted in 80–90% Pu³⁺ and the subsequent formation of a minor CaPu(Ti,Al)₂O₇ pyrochlore, at ~2 wt.% in each instance. The preferential incorporation of Ce within the CaTiO₃ phase is likely the result of increased reduction potential with respect to Pu, although the extent of Pu solubility on the Ca site has not been explicitly determined. CaTiO₃ has also demonstrated the ability to successfully accommodate both Np in both Np³⁺ and Np⁴⁺ valence states through tailoring the oxygen fugacity during synthesis (Begg et al., 1998a). Similar to related Pu compositions, targeting Np³⁺ incorporation in the absence of charge balancing additives did not prevent the formation of near single phase Ca_{0.90}Np_{0.10}TiO₃ when reacting in both air and 5% H₂/N₂. Np⁴⁺ remained the dominant valence state when targeting Ca_{0.90}Np_{0.10}Ti_{0.80}Al_{0.20}O₃ regardless of the sintering atmosphere, yielding >99% perovskite, demonstrating the ability of the phase to successfully immobilize both tri- and tetravalent actinides. The fundamental disadvantage hindering the deployment of single phase perovskite as an actinide wasteform pertains to its aqueous durability. Despite greater radiation tolerance relative to the actinide-designated zirconolite

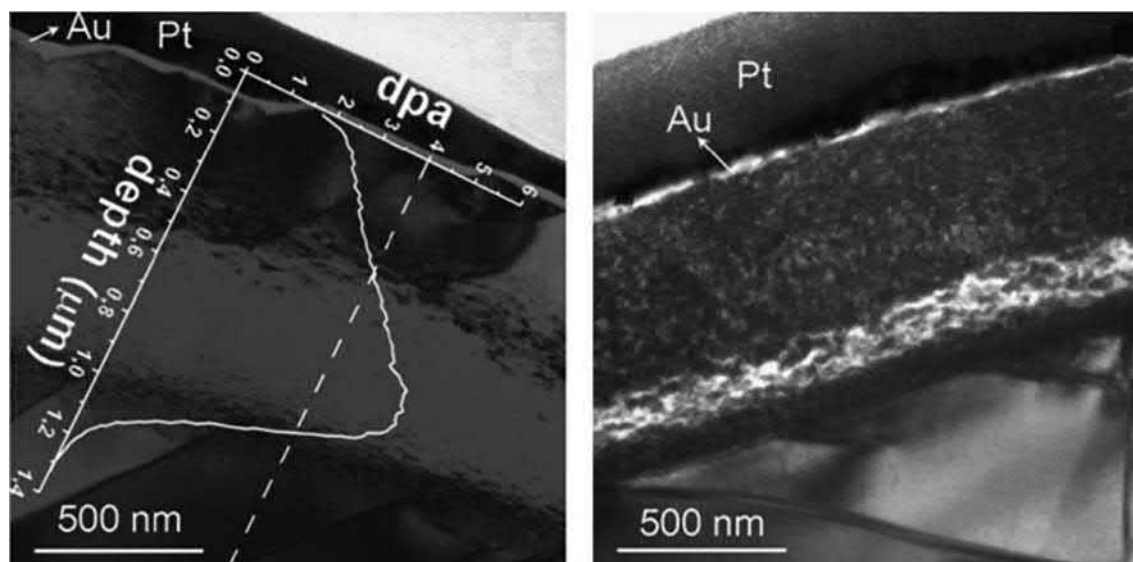


Fig. 6 TEM micrograph of damage region of Nd-doped perovskite irradiated with 2 MeV Kr⁺ at a dose of 5×10^{15} ions cm⁻², detailing a well-defined damage region between 0.8–1.1 μm, in agreement with TRIM calculations. From Davoisne C, Stennett MC, Hyatt NC, Peng N, Jeynes C, and Lee WE (2011) Krypton irradiation damage in Nd-doped zirconolite and perovskite. *Journal of Nuclear Materials* 415 (1): 67–73.

phase, perovskite is considered the most soluble of the SYNROC phases under hydrothermal conditions. Durability tests on disks of well characterized SYNROC-C in de-ionized water, bicarbonate and silicate solutions saturated at 150 °C up to 84 days confirmed that the dissolution rates of the SYNROC phases followed the order perovskite > hollandite > zirconolite, with the percentage of initial Al, Ca and Ce distributed across the SYNROC phases released after 337 days in 150 °C deionized water determined to be 0.10%, 0.068% and 0.045%, respectively (Lumpkin et al., 1991; Smith et al., 1992). Given the lower relative durability of perovskite with respect to zirconolite, and the common presence of CaTiO₃ as an ancillary phase during zirconolite synthesis, it is often desirable to reformulate precursor compositions with an excess of ZrO₂ and TiO₂ to preclude the ingrowth of perovskite.

Summary

Actinides, once chemically partitioned and separated from nuclear reactor wastes, pose an acute security and proliferation risk when stored *en masse* as unirradiated inventories. A credible pathway towards actinide disposition is immobilization in dedicated ceramic matrices, the design of which is largely informed by the study of naturally occurring mineral analogs that are robust against geological weathering. The SYNROC approach to wasteform development, comprised of durable titanate crystalline phases each targeting specific radionuclides present in HLW waste streams, demonstrated that ceramic immobilization is a feasible pathway to disposition, whilst offering superior aqueous durability compared to conventional borosilicate glass wasteforms under a range of standardized corrosion tests. A wide variety of titanate, zirconate and silicate-based mineral species have since been proposed as single phase wasteforms for separated actinides, including zirconolite, pyrochlore, perovskite and brannerite. The chemistry of these materials permits solid solution with actinides, allowing waste cations to adopt the properties of the host lattice, thus stabilizing and solidifying the waste prior to geological disposal, alleviating difficulties associated with storage, transport and nuclear non-proliferation. Wasteform selection trials, commonly utilizing actinide surrogates with similar chemistry, are fundamental in underpinning this approach, with rare-earth (Ce^{3+/4+}, Gd³⁺, Sm³⁺, Eu³⁺) and in some cases U⁴⁺ and Th⁴⁺ commonly deployed as simulants. It should be noted, however, that with the exception of recent advances in the SYNROC program, the deployment of ceramic wasteform technology has not been demonstrated at the industrial scale. For example, whilst it is envisaged that ceramic matrices will be utilized to immobilize some portions of the United Kingdom Pu stockpile, there is still considerable development of industrial-scale processes necessary before such a program can be implemented.

Acknowledgments

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Potential Fuel Cycle Back-End for Thorium Fuel Cycle

Xiangzhou Cai^a and Zhiguang Wang^{b, a} Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai, China; and
^b Institute of Modern Physics, Chinese Academy of Sciences, Lanzhou, China

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Main features of ^{233}U and impact on spent Th-fuel properties

The thorium fuel cycle is a nuclear fuel cycle that uses an isotope of thorium, ^{232}Th , as the fertile material (IAEA, 2005). In the reactor core, ^{232}Th is transmuted into the fissile artificial uranium isotope ^{233}U by absorbing neutrons as shown in Fig. 1. Compared with ^{239}Pu and ^{235}U in U-Pu fuel cycle, main differences arising from ^{233}U in Th-U fuel cycle are:

- **Lacking fissile element ^{233}U in nature.** Being different from natural uranium, which contains $\sim 0.7\%$ 'fissile' ^{235}U isotope, natural thorium is made up only of the 'fertile' ^{232}Th isotope and does not contain any 'fissile' material. Therefore, thorium and thorium-based fuel in the form of metal, oxide or carbide, molten salt, et al., has been utilized in combination with 'fissile' ^{235}U or ^{239}Pu in power reactors for conversion to 'fissile' ^{233}U , thereby enlarging the 'fissile' material resources (IAEA, 2005; Yu et al., 2019; Zou et al., 2018a).
- **Less sensitive to neutron spectrum.** Unlike ^{235}U and ^{239}Pu , the number of neutrons released per neutron absorbed (represented as η) of 'fissile' ^{233}U nuclei is greater than 2.0 over a wide range of neutron spectrum as displayed in Fig. 2. The ^{232}Th - ^{233}U fuel cycle is hence less sensitive to the type of reactors, thermal or fast reactor (IAEA, 2005).
- **Less production of transuranic (TRU) isotopes.** Under thermal neutron flux, the capture cross-section of ^{233}U is much smaller (46 b) than those from ^{235}U (101 b) and ^{239}Pu (271 b), while all the three fissile isotopes have fission cross-section in the same order of magnitude (525, 577 and 742 b for ^{233}U , ^{235}U and ^{239}Pu , respectively). Therefore, it is much less probable to have non-fissile absorption with ^{233}U which would lead to higher isotopes (e. g., transuranic) with higher absorption cross-sections (Wu et al., 2019; Zou et al., 2015).
- **Other long term radionuclides.** ^{232}Th - ^{233}U fuel cycle produces much less plutonium and minor actinides like Np, Am and Cm than that of U-Pu fuel cycle, but is associated with other long term radionuclides such as ^{231}Pa (half-life of $\sim 34,300$ years) (Winkle et al., 1949), ^{229}Th (half-life of ~ 7340 years) (Britannica, n.d.) and ^{230}Th (half-life of $\sim 75,381$ years) (Meadows et al., 1980).
- **^{233}Pa problem.** ^{233}Pa is formed as an intermediate in the conversion chain of ^{232}Th to ^{233}U . It has a relatively longer half-life (~ 27 days) as compared to its counterpart ^{239}Np (half-life of ~ 2.4 days) in the uranium fuel cycle. As a consequence, to complete the decay of ^{233}Pa to ^{233}U and avoid loss of any ^{233}U fissile material, at least 12 months (more than 10 half-lives of ^{233}Pa) is required for cooling prior to reprocessing (OECD-NEA, 2015a).
- **^{232}U problem.** ^{232}U is produced along with the bred ^{233}U in irradiated mixed ^{232}Th - ^{233}U fuel. The $^{233}\text{Pa}(n,2n)^{232}\text{Pa}(\beta^-)^{232}\text{U}$ and $^{233}\text{U}(n, 2n)^{232}\text{U}$ reaction with the neutron energy threshold up to ~ 6.37 MeV are two main ways of generating ^{232}U . ^{232}U decays into ^{228}Th by emitting alpha particles and has a half-life of 73.6 years. Radiation hazard mainly arises due to build-up of the gamma emitting, short-lived daughter products of ^{228}Th , namely ^{212}Pb , ^{212}Bi and ^{208}Tl , particularly ^{208}Tl , which emits 2.6 MeV gamma radiation (OECD-NEA, 2015b).

Table 1 lists the specific radiation rates for main actinides in thorium fuel cycle and uranium cycle. Main differences arising from ^{232}U and ^{233}U in thorium-based fuel cycle would impact the properties of spent Th-fuel, including the radioactivity and toxicity, which in turn influence the nonproliferation, spent fuel reprocessing and disposal (IAEA, 2005).

- **Impact on reprocessing of spent Th-based fuels.** THORium-uranium EXtraction (THOREX) process (as shown in Fig. 3), which is the wet chemical route developed in Oak Ridge National Laboratory (ORNL) in the mid 1950s based on solvent extraction separation of uranium and thorium from fission products by means of tributyl phosphate (TBP) (Gresky, 1956), is so far the most viable route for recovery of uranium from spent Th-based fuels. Up to now, reprocessing of spent thorium fuel has been carried out only in a few countries (Merz, 1982) based on THOREX process mostly on laboratory or pilot plant scale. In the THOREX process, Pa is generally passed into the fission product waste, which could have long term radiological impact due to the formation of ^{231}Pa . High gamma emitters ^{232}U and ^{228}Th cannot be chemically separated from the recovered ^{233}U and thorium respectively. The formed high-energy gamma radiation field requires remote reprocessing and refabrication of Th-based

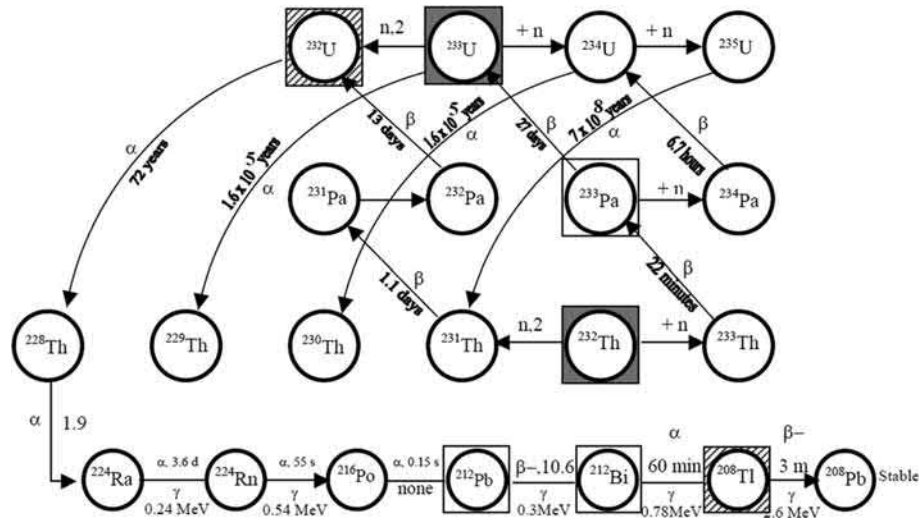


Fig. 1 Main isotopes in ^{232}Th - ^{233}U fuel cycle (IAEA, 2005). From IAEA (2005) *Thorium Fuel Cycle-Potential Benefits and Challenges*. IAEA-TECDOC-1450.

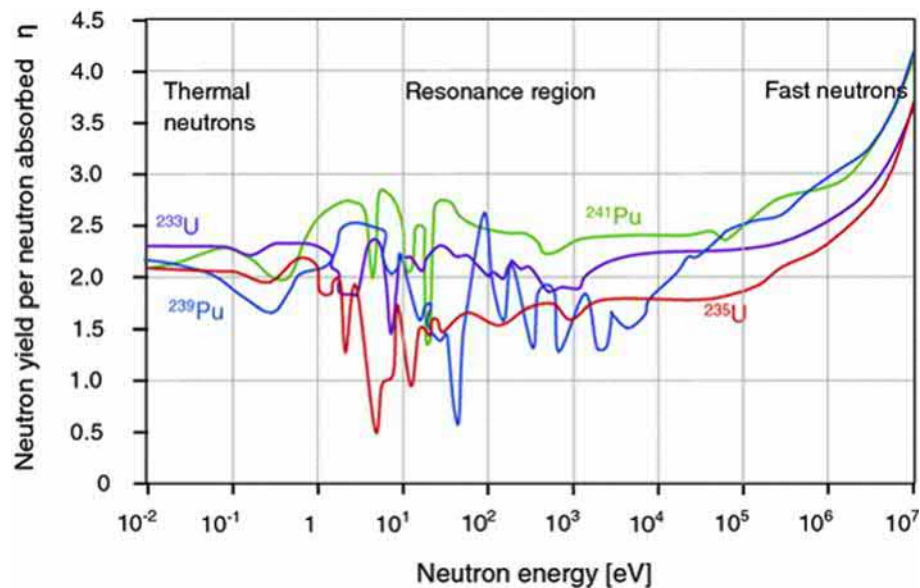


Fig. 2 Neutron yield per neutron absorbed (Brown et al., 2015). From Brown NR, Powers JJ, Feng B, Heidet F, Stauff NE, Zhang G, Todosow M, Worrall A, Gehin JC, Kim TK, and Taiwo TA (2015) Sustainable thorium nuclear fuel cycles: A comparison of intermediate and fast neutron spectrum systems. *Nuclear Engineering and Design* 289: 252–265.

fuels. In addition, off-gas transport of the highly radioactive radon daughters through high-efficiency particulate air (HEPA) filters to the environment should be avoided in reprocessing spent fuel (for recovery and recycle of ^{233}U). It is necessary to withhold the off-gas around 10 min in designing the off-gas system to allow for decay of ^{220}Rn prior to HEPA filtration. The above issues also exist in the pyrochemical route of spent thorium fuel reprocessing, which is still in laboratory level. If thorium is not to be recycled because of its radioactivity, its disposal becomes an additional problem.

- *Impact on non-proliferation of spent Th-based fuels.* The high energy gamma ray emitted by the daughter products of ^{232}U could help to detect ^{233}U , which, from the standpoint of safeguards (i.e. the means to prevent unauthorized use of ^{233}U), is an asset (Wu et al., 2020).
- *Impact on radiotoxicity evolution of spent Th-based fuels.* A pure $\text{Th}/^{233}\text{U}$ cycle will indeed produce a much reduced amount of transuranics than conventional UO_2 fuels. But, decay products from ^{233}U (^{229}Th with the half-life of 7340 years) and ^{234}U which is mainly produced through neutron capture on ^{233}U (^{226}Ra with the half-life of 1600 years) drive radiotoxicity to be higher than that of low enriched uranium or U/Pu fuel at around 10^5 years.

Table 1 Specific radiation rates for selected actinides.

Nuclide	Dose rate (Gamma) at 1 m ($\mu\text{Sv/h/kg}$) ($t = 1$ s)	Heating rate (W/kg)	Neutron (emission rate/kg/s)
Pu-238	1.81E+03	5.65E+02	2.94E+06
Pu-239	1.52E+01	1.93E+00	2.53E+01
Pu-240	2.61E+01	7.05E+00	1.20E+06
Pu-241	1.76E+02	4.00E+00	5.00E+01
Pu-242	3.17E-01	1.16E-01	2.01E+06
Am-241	3.12E+05	1.12E+02	1.19E+03
U-232	1.97E+04	7.06E+02	1.84E-01
U-233	8.16E+00	2.79E-01	–
U-234	2.69E+00	1.78E-01	9.78E+00
U-235	1.30E+00	5.71E-05	4.00E+00
U-236	1.52E-02	1.75E-03	7.17E+00
U-238	8.63E-05	8.45E-06	1.67E+01

From IAEA (2005) *Thorium Fuel Cycle-Potential Benefits and Challenges*. IAEA-TECDOC-1450.

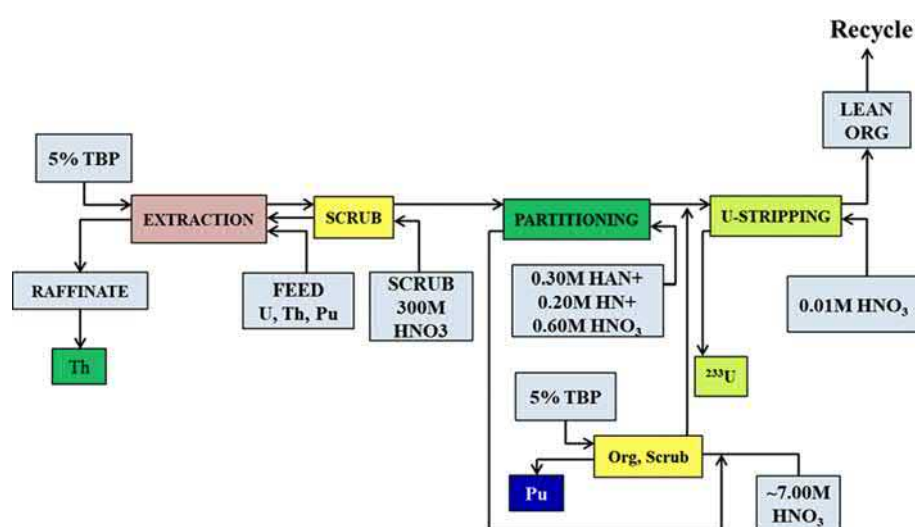


Fig. 3 Aqueous Reprocessing by THOREX Process (Achuthan and Ramanujam, 2013). From Achuthan PV and Ramanujam A (2013) Aqueous reprocessing by THOREX process. In: Das D and Bharadwaj S (eds) *Thoria-Based Nuclear Fuels. Green Energy and Technology*. London: Springer.

Evolution of Th-spent nuclear fuel radioactivity and toxicity with time

Radiotoxicity, caused by the decay of fission products and actinides, is an important parameter in fuel cycle consideration, wherever spent fuel is disposed of (Clement, 2012). The radiotoxicity from the spent fuel is mainly composed of two parts: (1) The fission products whose half-lives vary from days to over 200,000 years dominate the radiotoxicity in the first 500 years; (2) The long-term (of the order of about 10^3 – 10^6 years) radioactive hazard is dominated by plutonium and minor actinides (ICRP, 1995). In general, the period between 500 years and 10,000 years is considered to be an important factor in repository performance, because this period is the point when waste packages are likely to start losing their integrity and subsequently release radionuclides in the near-field environment. After this period, most of the fission products have decayed and the radiotoxicity becomes dominated mainly by transuranic elements.

Typically for the Th-U fuel cycle, the radiotoxicity evolution of spent nuclear fuel can be divided into three main phases due to the radioactive decay of spent nuclear fuel (as shown in Fig. 4) (Brian et al., 2013):

- 1) 1–200 years: the medium-lived fission products dominate the radiotoxicity of the spent fuel. The fission product yields of ^{233}U are similar to ^{235}U and ^{239}Pu , and therefore the total radiotoxicity has little difference in this period. For the fission products, ^{90}Sr accounts for essentially all of the total radiotoxicity during this period for two reasons: (1) the 28.8-year half-life of ^{90}Sr is sufficiently long to allow it to contribute substantially to the radioactivity in this time period; (2) it has a relatively high radiotoxicity. The fission product yield of ^{90}Sr from ^{233}U (6.9%) is larger than the yield from ^{235}U (5.9%) (Croff and Krahn, 2016). Therefore, the radiotoxicity of the Th-based fuel is about twice that of the U-based fuel during this period.
- 2) 200–1000 years: the transuranics (primarily Np, Pu, Am, and Cm) dominate the radiotoxicity of the spent fuel in this period. In the thorium fuel cycle, five captures in ^{232}Th are necessary to produce a transuranic element. Therefore, the transuranic

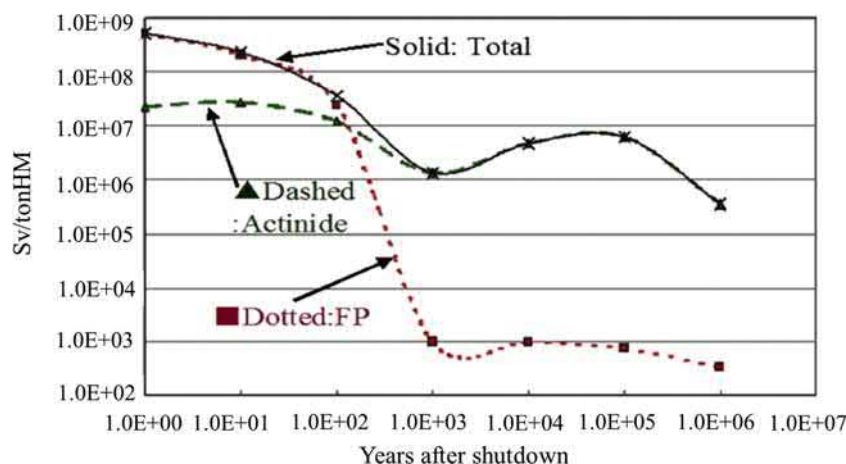


Fig. 4 Ingestion radiotoxicity of Th-spent nuclear fuel (Thomas, 2017). From Yoshioka R, Kinoshita M, Takaki N, and Yoshida T (2015) Radiotoxicity evaluation for spent thorium fuel in PWR. *Transaction of AESJ-2015 Meeting, Hitachi, Japan*.

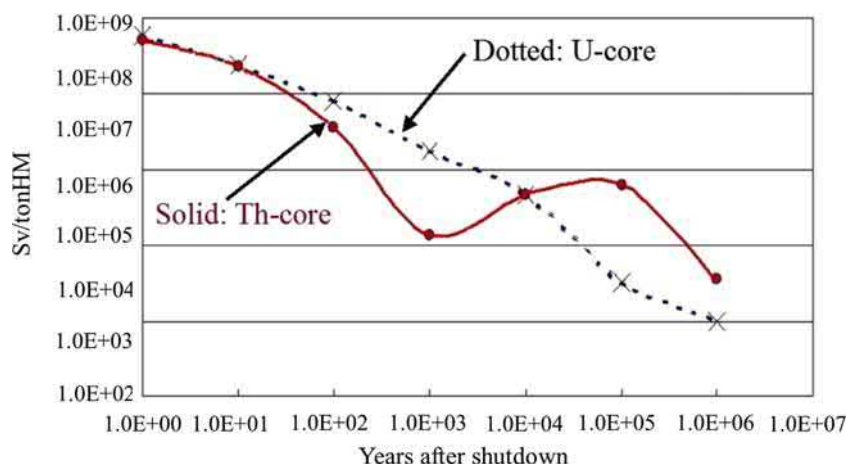


Fig. 5 Radiotoxicity for U-core and Th-core (Thomas, 2017). From Yoshioka R, Kinoshita M, Takaki N, and Yoshida T (2015) Radiotoxicity evaluation for spent thorium fuel in PWR. *Transaction of AESJ-2015 Meeting, Hitachi, Japan*.

production is extremely low. In addition, 98–99% of thorium-cycle fuel nuclei would fission as either ^{233}U or ^{235}U , so much fewer long-lived transuranics are produced (IAEA, 2005; Wikipedia, n.d.). Therefore, there are much lower quantities of plutonium and the longer-lived minor actinides compared with the uranium-plutonium (U-Pu) fuel cycle, typically several orders of magnitude less than the equivalent U-Pu fuel. The radiotoxicity in the Th-U fuel cycle in this period is therefore about 10 times lower than of the U-Pu fuel cycle (as shown in Fig. 5) (Thomas, 2017). In the back end of the thorium fuel cycle, the radiotoxicity of actinides of spent Th-U fuel is determined by ^{232}U and its daughter nuclides, the first of which is ^{228}Th , for the first 300 years. The half-lives of ^{232}U and ^{228}Th are 68.9 years and 1.9 years, respectively. The subsequent daughter nuclides in the decay chain of ^{232}U have shorter half-lives. The most significant of the remaining actinides are ^{238}Pu and ^{234}U , whose contributions are 10–100 times smaller. The contribution of ^{239}Pu , ^{240}Pu , ^{241}Am , and ^{232}Th is 104 times smaller than that of ^{232}U . After 100 years of storage, the total radiotoxicity decreases by a factor of 2.6 (Croff and Krahn, 2016).

- 3) > 1000 years: long-lived ^{233}U daughter products dominate the radiotoxicity in this period. For a disposal period longer than 1000 years, ^{232}U with its daughter nuclides do not contribute to the radiotoxicity, which is determined by ^{234}U and ^{230}Th . After 3000 years, the radiotoxicity increases because of the accumulation of ^{226}Ra , ^{229}Th , and ^{230}Th . When the disposal period is longer than 10,000 years, the radiotoxicity is determined by ^{226}Ra and ^{229}Th , and ^{233}U makes a small contribution. For a disposal period of 10,000 years, the radiotoxicity decreases by a factor of 160 relative to that at 1 year after shutdown (Bergel et al., 2001; Yoshioka et al., 2015). The dominance of these nuclides results in the radiotoxicity of the Th-U fuel cycle being about twice that of the equivalent U-Pu fuel cycle for these extended periods, although this can vary depending on the specific scenarios chosen.

The thorium cycle can fully recycle actinide wastes and only emit fission product wastes, and after a few hundred years, the waste from a thorium reactor can be less radiotoxic than the uranium ore that would have been used to produce low enriched uranium

fuel for a light water reactor of the same power. Therefore, the thorium fuel cycle can minimize the radiotoxicity associated with spent fuel and control radiotoxicity between 500 and 100,000 years (WNA, World Nuclear Association, 2009; Thorium Test Begins, n.d.). For example, in the closed thorium fuel cycle in fast reactors with Pu and minor actinides, the amount of Pu burning will be much larger relative to fast reactors operating with the U-Pu fuel cycle, and appreciable amounts of ^{233}U would be produced which could be utilized for starting new thorium-fueled thermal reactors or Accelerator Driven Systems (ADS) cycle (Croff and Krahn, 2016; Hesketh and Thomas, 2013; Abderrahim and Bruyn, 2021). Molten salt reactors with thorium fuel cycle have the unique advantages of online reprocessing (Zou et al., 2018a, b). If reprocessing is applied for MSR and most actinides such as U, Th, and their daughters are separated and recycled, then radiotoxicity after several hundred years can be greatly reduced (Yu et al., 2019; Zou et al., 2015; Robertson, 1971). The amount of radioactive waste produced by the thermal reactors based on Th-U fuel cycle is much less than that produced by PWRs with an once-through fuel cycle, and it also has obvious advantages compared with the fast breeder reactors based on U-Pu fuel cycle.

Key issues associated with Th-fuel processing and industrial experience

Molten salt reactors are particularly well adapted to the ^{232}Th - ^{233}U fuel cycle which has the advantage of producing less minor actinides than the ^{238}U - ^{239}Pu fuel cycle. Moreover, the Th- ^{233}U fuel cycle can be obtained with a moderated neutron spectrum while a fast neutron spectrum is generally required to breed in the U-Pu cycle (Krepel, 2021). As far as poisoning resulting from fission products is concerned, it is worse in a thermal neutron spectrum than in a fast neutron spectrum, and the rate at which fuel reprocessing is performed can become a major issue. However, continuous extraction of the fission products is a possibility given the liquid nature of fuel in molten salt reactor (Mathieu et al., 2006; Ignatiev, 2021). This has been demonstrated in the conceptual design of the molten salt breeder reactor (as shown in Fig. 6) in the 1970s that aimed at investigating the feasibility of low-cost electricity production with a low specific inventory of fissile material, and simultaneously a reasonably high breeding gain in a molten salt reactor. According to this design, all of the gaseous fission products and most noble metal particles can be removed by a fuel salt off-gas purging system. Another chemical process was designed to continuously remove fission products and ^{233}Pa from the fuel salt, and the bred ^{233}U could be recovered in order to provide necessary amounts of fissile material (Delpech et al., 2009) (Fig. 7).

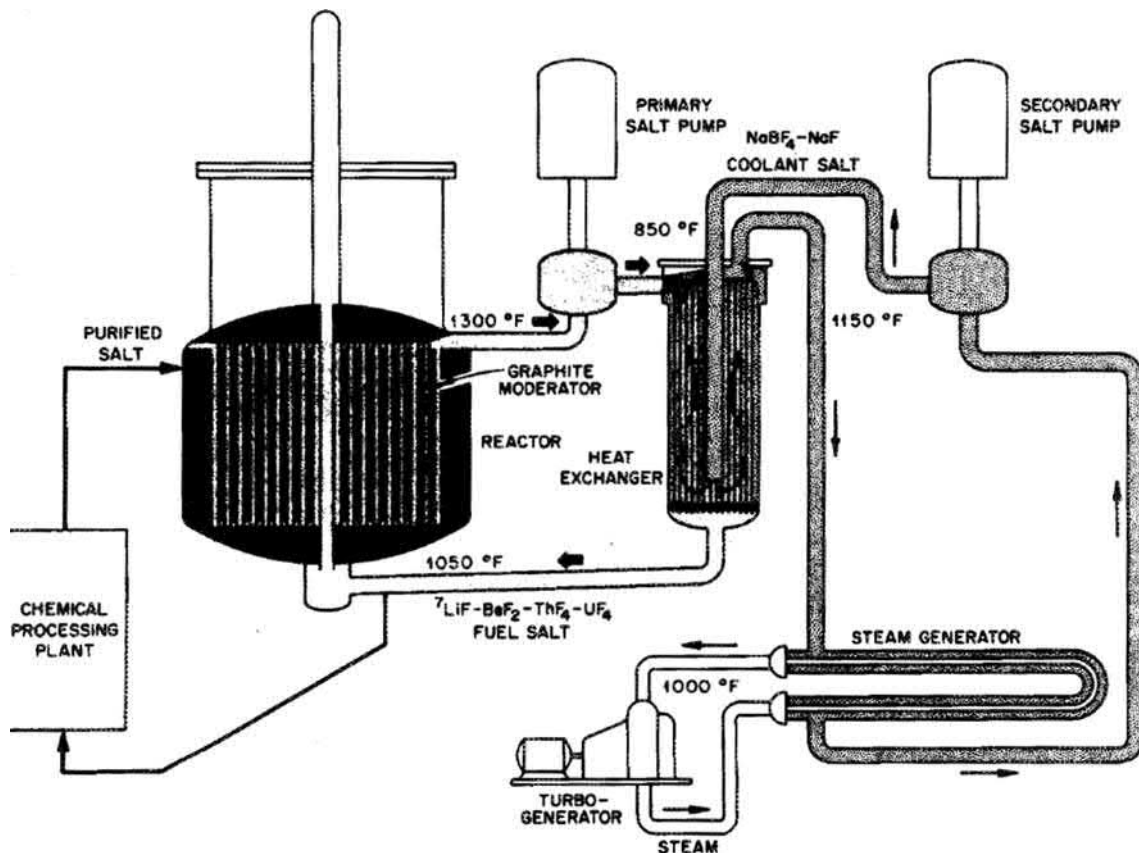


Fig. 6 Molten salt breeder reactor (Rosenthal et al., 1972). From Rosenthal MW, Haubenreich PN, and Briggs RB (1972) *The Development Status of Molten-Salt Breeder Reactors*. ORNL-4812.

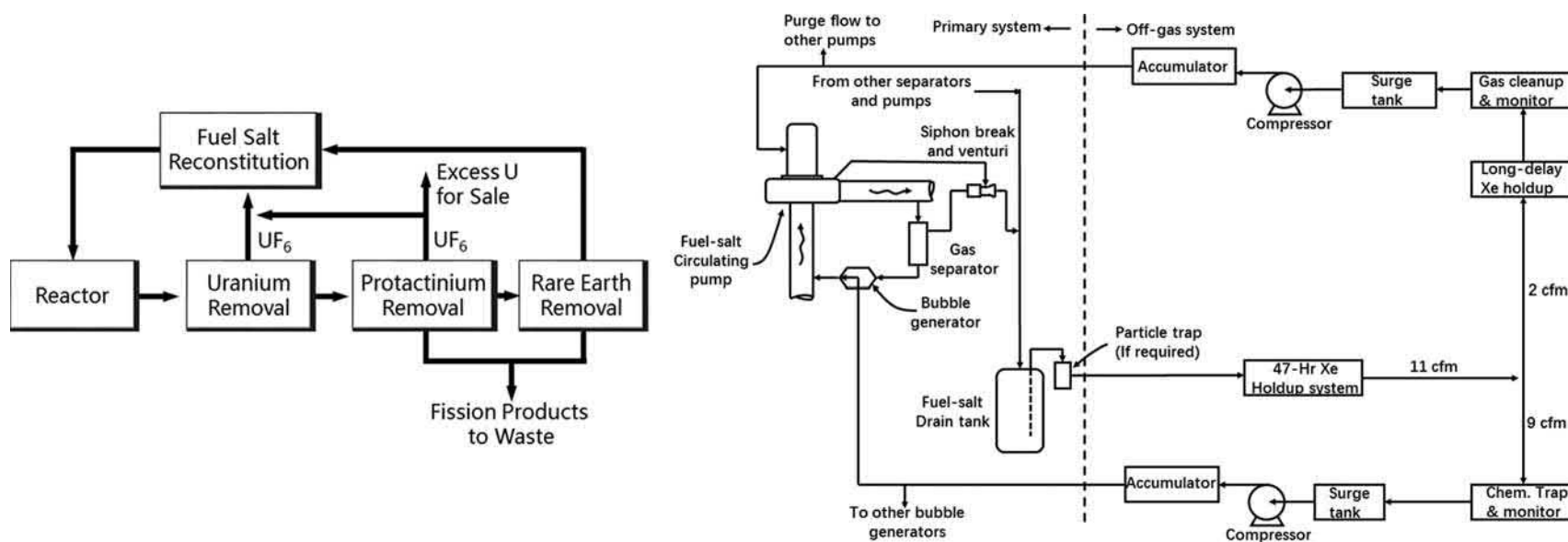


Fig. 7 Simplified flow-sheet for processing a molten-salt breeder reactor (left), and schematic flow diagram of molten-salt breeder reactor off-gas system (right) (Rosenthal et al., 1972). From Rosenthal MW, Haubenreich PN, and Briggs RB (1972) *The Development Status of Molten-Salt Breeder Reactors*. ORNL-4812.

Although thorium fuel reprocessing has been carried out in water and organic solvents, it is more practical to employ pyroprocessing especially for the thorium fuel of molten salt reactor. One of the most obvious advantage is that the fuel used in reactor is essentially the same as that in pyroprocessing, which means no additional pre- or post- treatment of the fuel is required. In spite of this, this kind of thorium fuel reprocessing is not currently in significant use worldwide. The major reason is that reprocessing itself is not currently in favor, and places with this capability already have plants constructed that are based on separation in water and organic solvents. More advanced technology than the presently used fuels based on uranium and plutonium is therefore required. Another problem associated with thorium fuel processing is that many of the daughters of ^{232}Th and ^{233}U are strong gamma emitters (as shown in Fig. 8) which makes it difficult and dangerous to reprocess the used thorium fuel. For example, although ^{232}U itself is not particularly harmful, it quickly decays to the strong gamma emitter ^{208}Tl with significant quantities. The presence of ^{232}U and others significantly increases the requirements of shield and protection when processing thorium fuel, and this could increase the complexity for the development of pyroprocessing technology aiming at thorium fuel reprocessing (Morss et al., 2006). Finally, pyroprocessing separation technologies has not yet been demonstrated to achieve high separation rates, and their industrial implementation remains complex and there is currently no real industrial feedback. However, the separation rate issue is not likely a problem if pyroprocessing is used for the treatment of molten salt fuels since it is possible to partially drain the liquid fuels from the reactor which can be fed by new fuels after pyroprocessing later. This lowers the requirements for the separation rate as long as the drain-feed cycle is maintained at a level that assures the normal operation of the reactor. It might not be so advantageous to pyroprocess the fuels from other types of reactors due to the increased complexity for the front-end and back-end operations (dissolution, conversion et al.).

The key issues of the proliferation-concern

An increasing number of countries have taken nuclear energy as a preferred approach in response to the environment deterioration and energy supply deficit. The rapid expansion of nuclear technologies, however, would pose a great challenge to nuclear non-proliferation, especially for the Generation IV nuclear reactor systems (Stanculescu, 2021) which significantly differ from current nuclear fuel systems. As an alternative to U-Pu fuel cycle, Th-U fuel cycle has also drawn some attention although it is not yet commercially available. The decay path of thorium is well understood. Pure ^{232}Th forms ^{233}Th while bombarded with neutrons, which has a half-life of 22 min and β -decays to ^{233}Pa . ^{233}Pa has a half-life of 27 days and becomes ^{233}U through β -decays. Under IAEA Safeguards Agreements, pure ^{233}U is treated identically to plutonium (i.e. a significant quantity is defined as 8 kg and amounts over 2 kg are treated as Category I materials) (International Atomic Energy Agency, 2002, 2011). The suggested categorization presented in Table 2 defines ^{233}U denatured with ^{238}U to be an equivalent categorization to denatured ^{235}U (Ashley et al., 2012a). Thus, ^{233}U poses proliferation risks.

However, in Th-U fuel cycle, the presence of high energy gamma rays (2.6 MeV from ^{208}Tl) emitted by the daughter products of ^{232}U results in the robustness or physical protection of Th-U fuel cycle (Gangotra et al., 2014). Strong gamma rays not only cause great damage to operators and equipment, but also are difficult to be detected. They also put forward higher requirements for remote operation, which makes ^{232}U a vital measurement for thorium-based fuel cycle proliferation resistance assessment. Jungmin Kang et al. calculated the relationship between the radiation dose of ^{233}U and the ^{232}U content based on the reaction chains to produce ^{232}U (as shown in Fig. 9). High burn-up fuels (e.g. PWR fuel with a discharge burn-up of 70 GWd/t) is analyzed, the radiotoxicity of a 5 kg bare uranium metal, with an assumed ^{232}U component of 0.001% is 130 mSv/h at 0.5 m, after 1 year of decay time (Kang and Frank, 2001).

Mass, isotopic mix, heat and radiotoxicity of the nuclear materials, technical characteristics (detectability, technical level, nuclear facilities, etc.) and institutional arrangements of nuclear energy systems all have an impact on nuclear proliferation performance. For the front-end of the thorium-based fuel cycle, the type of fuel used to initiate the reactor is the main factor to be considered. Unlike uranium, there is no need for enrichment process for Th-U fuel cycle, which reduces the proliferation risks. In the operation stage, it is necessary to focus on the monitoring of gamma rays in the reactor, since the concentration of ^{232}U in the reactor will directly affect the difficulty of ^{233}U extraction, while ^{232}U production is related to the energy spectrum, fuel ratio and burnup of the reactor. The post-processing scheme also affects the technical difficulty of ^{233}U extraction, hence has an influence on the proliferation performance. After decommissioning, nuclear waste needs to be disposed. The fractions of different nuclear materials in

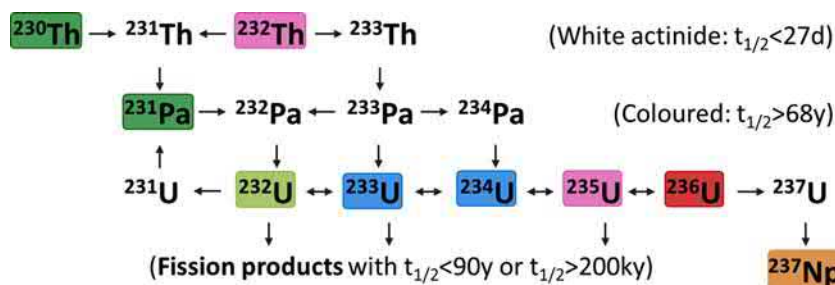
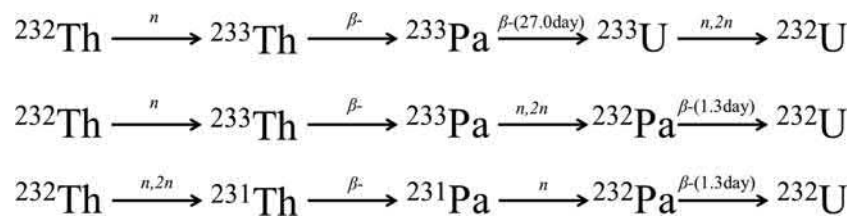


Fig. 8 Transmutation in the thorium fuel cycle.

Table 2 Material definitions for safeguarding unirradiated Pu, ^{235}U and ^{233}U (Ashley et al., 2012a).

Material	Form	Category		
		I	II	III
Pu	Any material form, with exception of plutonium containing $\geq 80\%$ ^{238}Pu	≥ 2 kg	< 2 kg, but > 0.5 kg	≤ 500 g, but > 15 g
^{235}U	U enriched to $\geq 20\%$ ^{235}U	≥ 5 kg	< 5 kg, but > 1 kg ≥ 10 kg	≤ 1 kg, but > 15 g
	U enriched to $\geq 10\%$ ^{235}U , but $< 20\%$		≥ 10 kg	< 10 kg
	U enriched above natural, but $< 10\%$			≥ 10 kg
^{233}U	Any material form	≥ 2 kg	< 2 kg, but > 0.5 kg	≤ 500 g, but > 15 g
	U with $\geq 12\%$ ^{233}U	≥ 2 kg	< 2 kg, but > 0.5 kg	≤ 500 g, but > 15 g
	U With $< 12\%$ ^{233}U , but $\geq 6\%$		≥ 4 kg	< 4 kg
	U with $< 6\%$ ^{233}U , but $\geq 0.66\%$			≥ 4 kg

From Ashley SF, Nuttall WJ, Parks GT et al. (2012) On the proliferation resistance of thorium-uranium nuclear fuel. *UK Poni Conference: Nuclear Stability: from the Cuban Crisis to the Energy Crisis*.

**Fig. 9** Reaction chains to produce ^{232}U (Wojciechowski, 2018). From Wojciechowski A (2018) The U-232 production in thorium cycle. *Progress in Nuclear Energy*, 106: 204–214.**Table 3** Developed non-proliferation assessment methodologies worldwide (Wu et al., 2020).

Method	Analysis category	Characteristics
INFCE and NASAP	Empirical analysis	Qualitative
MAUA	Attribute analysis	Quantitative (+)
TOPS		Qualitative
SAPRA		Qualitative
FLB		Quantitative (+)
INPRO		Qualitative
PR&PP	Scenario analysis	Qualitative
Markov approach		Quantitative (+)
RIPA		Quantitative (+)

Note: the mark “+” in the table has no meaning but is used for telling the difference between “quantitative” and “qualitative” easily.

From Wu J, Ma Y, Yu C, et al. (2020) Nuclear non-proliferation review and improving proliferation resistance assessment in the future. *International Journal of Energy Research* 2020: 1–24.

waste will affect the radiation quantity, heat release, spontaneous neutron generation rate, etc. The amount of spent fuel in Th-U fuel cycle is relatively small and stable, which does not react with water and is favorable for dry and wet storage.

Ashley et al. (Ashley et al., 2012b) conducted an extensive discussion on non-proliferation of Th-U fuel cycle in comparison to U-Pu fuel cycle, and raised several open questions that why the USA stopped pursuing ^{233}U for weapon purposes, how to define the weapon grade uranium vector when ^{232}U and ^{233}U are involved, whether Th-U fuel cycle systems would provide additional pathways to obtain sensitive nuclear materials, whether they are suitable to adopt the same U-Pu fuel cycle non-proliferation criterion (20% low-enriched uranium) and whether they are more difficult to be safeguarded. Although all these questions remain to be answered, some progress have been made in the related studies on Proliferation Resistance (PR) of Th-U fuel cycle, especially for MSR and CANDU since they are both considered most suitable for thorium utilization (Wang, 2016).

In addition, the accumulation of Pa is also one of the proliferation risks of Th-U fuel cycle. Pure ^{233}U can be produced by beta decay from ^{233}Pa , which makes the nuclear fuel cycle containing $^{233}\text{U}/\text{Pa}$ separation recycling process riskier compared to the once-through fuel cycle. It has been demonstrated that around 200 g of thorium metal could produce 1 g of ^{233}Pa and hence 1 g of ^{233}U . However, beta decay of 1 g ^{233}Pa produces 50 W of heat, and separation equipment is exposed to strong radioactivity, which complicates the handling.

Several proliferation assessment methodologies have been put forward world widely as shown in Table 3. In 2000, a PR assessment methodology dedicated for innovative nuclear reactors and fuel cycles was proposed by the International Project on

Innovative Nuclear Reactors and Fuel Cycles (INPRO) which had been launched by IAEA (IAEA, 2003). And the PR&PP framework proposed by GIF (Generation IV International Forum), which divided proliferation risk into national proliferation and threats from terrorisms (Nishimura et al., 2004). A toolbox containing the potential theories of expert elicitation, Multi-attribute Utility (MAU) theory, fuzzy sets and possibility theory, analytic hierarchy procedure, dynamic modeling, two-sided methods and logic diagrams was offered, and some of the models and/or methods in this toolbox were also extensively developed in the later related studies (Charlton et al., 2007). Nevertheless, these methods have some limitations in the assessment of Th-U fuel cycle, and it should be effectively addressed in the future safeguard analysis or design.

Thorium fuel cycles face unique proliferation resistance problems compared with U-Pu fuel cycles as introduced above. And these problems depend very much upon how thorium fuel cycles are implemented. Some measures, including raising ^{232}U content by increasing discharged burnup and/or by denaturing ^{233}U with ^{238}U , were put forward to effectively enhance the proliferation performance (Vanderhaegen and Janssens-Maenhout, 2010).

Comparison of the waste characteristics of the U-Pu and Th-U fuel cycles

The nuclear wastes in the fuel cycle mainly consist of fission products and actinides. In the following, the differences of the waste characteristics between the U-Pu and Th-U fuel cycles are detailed.

1) Fission products

For a given amount of thermal energy produced in a reactor core, the amount of fission products is essentially the same in thorium or uranium fuels because a fission reaction releases about 200 MeV, regardless of the fission nuclide, and gives birth to two fission products. The relative proportion of each of the radioisotopes is slightly different according to the origin of the fission (^{235}U , ^{239}Pu , ^{241}Pu , ^{233}U) and consequently the evolution of the total radiotoxic inventory of these fission products is slightly different between the two cycles. However, as most of the radioactivity from fission products disappears after a few centuries, and there is little difference in the amounts of long-lived mobile fission products that tend to drive repository safety cases. Therefore, there is almost no difference between the U-Pu and Th-U fuel cycles with regard to the fission products (OECD-NEA, 2015a). The fission products account for about 3% of the total mass of the spent fuel, which are considered as radioactive waste or may be separated further for various industrial and medical applications. As shown in Fig. 10, the fission products include every element from zinc through to the lanthanides. Much of the fission yield is concentrated in two peaks: one in the second transition row (Zr, Mo, Tc, Ru, Rh, Pd, Ag) and the other later in the periodic table (I, Xe, Cs, Ba, La, Ce, Nd). Most of the fission products are either non-radioactive or only short-lived radioisotopes. But a considerable number of fission products are medium to long-lived radioisotopes such as ^{90}Sr , ^{137}Cs , ^{99}Tc and ^{129}I (Encyclopedia, n.d.-b). The yield ratios of long-lived fission products (LLFP) for thermal neutron fission of ^{233}U , ^{235}U and ^{239}Pu are listed in Table 4. Large differences can be observed for some LLFPs, especially for ^{129}I which is a mobile long-lived fission product and significantly contributes to the final impact of the repository in reducing environment. The yield ratio of ^{129}I in ^{233}U fission is about triple that in ^{235}U fission, implying larger impact would be imposed by Th-U fuel cycle on disposing high level nuclear wastes from the view of LLFPs.

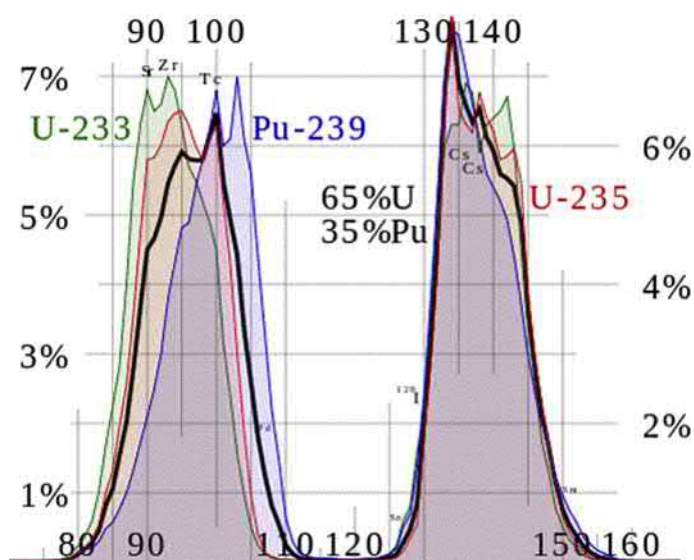


Fig. 10 Fission product yields by mass for thermal neutron fission of U-235, Pu-239, a combination of the two typical of current nuclear power reactors, and U-233 used in the thorium cycle (Encyclopedia, n.d.-a). From Encyclopedia, <https://encyclopedia.thefreedictionary.com/Fission+product+yield>.

Table 4 Yield ratio of long-lived fission products for thermal neutron fission of ^{233}U , ^{235}U and ^{239}Pu .

Nuclide	Half-life (Ma)	Yield- ^{233}U	Yield- ^{235}U	Yield- ^{239}Pu
^{99}Tc	0.211	9.2324E-02	1.1484E-01	1.1678E-01
^{126}Sn	0.230	2.2433E-03	5.6143E-04	1.9861E-03
^{79}Se	0.327	2.8676E-03	8.9447E-04	4.3452E-04
^{93}Zr	1.53	6.9788E-02	6.3463E-02	3.7975E-02
^{135}Cs	2.3	6.2680E-02	6.5392E-02	7.6209E-02
^{107}Pd	6.5	1.1453E-03	1.4619E-03	1.2700E-07
^{129}I	15.7	1.5913E-02	5.4335E-03	1.3714E-02

From Chadwick MB, Herman M, Obložinský P et al. (2011) ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data. *Nuclear Data Sheets*, 112(12) 2887–2996.

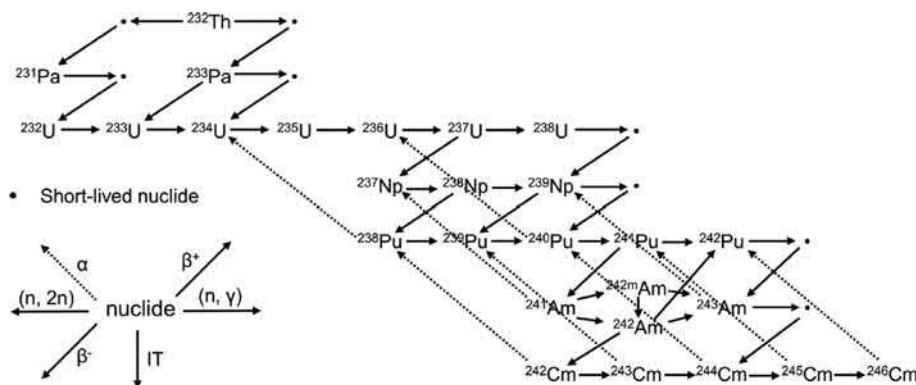


Fig. 11 Reaction chain in the U-Pu and Th-U fuel cycles (Zhang et al., 2020). From Zhang A, Zou C, Wu J et al. (2020) Radiotoxicity of minor actinides in thermal, epithermal and fast TMSRs with very high burnup. *Annals of Nuclear Energy* 137: 107–162.

2) Actinides

The difference between the U-Pu and Th-U fuel cycles may come from the amount of depleted uranium and minor actinides generated in the two cycles.

In the U-Pu fuel cycle, 96% of the mass of spent fuel is the remaining uranium: most ^{238}U and a little ^{235}U . In general, ^{235}U would be less than 0.83%. In addition, the remaining uranium will contain ^{236}U with 0.4%, which is not found in nature and can be used as a fingerprint for spent reactor fuel. About 1% of the mass is ^{239}Pu and ^{240}Pu resulting from ^{238}U conversion, which may be considered either as a useful byproduct, or as dangerous and inconvenient waste (Encyclopedia, n.d.-b). One of the main concerns regarding nuclear proliferation is to prevent this plutonium from being used by states to produce nuclear weapons. If the reactor has been used normally, the plutonium is considered as reactor-grade. The plutonium contains more than 19% ^{240}Pu and less than 80% ^{239}Pu , which makes it not ideal for making nuclear weapons. If the irradiation period is short, the plutonium is considered as weapons-grade (more than 80%, up to 93%). In the U-Pu cycle, significant amounts of three long-lived transuranic elements (other than plutonium) are generated (as shown in Fig. 11): ^{237}Np (with half-life of 2.14 million years), americium (mainly ^{241}Am , with a half-life of 432 years and ^{243}Am , with a half-life of 7380 years) and curium (mainly ^{245}Cm , with a half-life of 8530 years). These nuclides are particularly radiotoxic alpha emitters and the main contributors to the total radiotoxic inventory of final radioactive waste after a few hundred years (with the exception of plutonium, which may be recycled depending on the specific fuel cycle). These nuclides have low solubility limits in water and have very low mobility in geological media, which practically excludes the possibility of their migrating to the biosphere, at least over timescales during which they could deliver significant radiological doses to living species. The partitioning of these elements and their subsequent transmutation in reactors is an option that has been studied nowadays to further minimize this risk. Research has been conducted in this direction for many years in various countries (such as France, Japan and the United States) suggesting that these operations could be carried out industrially. However, this would require the implementation of complex processes and technologies with specific additional costs.

In the pure Th-U fuel cycle, almost none of the above mentioned minor actinides (Np, Am, Cm) are produced because five captures in ^{232}Th are generally necessary to produce transuranics, as displayed in Fig. 11. Americium and curium mainly come from plutonium (via neutron capture and various radioactive decays), while ^{237}Np mainly comes from ^{235}U (via producing ^{237}U and decays to ^{237}Np). In the Th-U fuel with a burn-up of ~ 60 GWd/t, there is about 30 times less ^{237}Np , than in the U-Pu fuel cycle. This ratio is about 60 for ^{238}Pu (with half-life of 88 years) (Lung, 1997). However, the spent nuclear fuel in the Th-U fuel cycle will have U-233, with a half-life of 159,200 years. This will affect the long-term radioactive decay of the spent fuel. When compared with MOX fuel, the activity around one million years in the Th-U cycle will be higher due to the presence of

the not completely decayed U-233. Irradiation of thorium fuel also increases the accumulation of specific long-lived heavy radionuclides, some of which have a much higher radiotoxicity than their counterparts in the U-Pu fuel cycle. The most prominent are (OECD-NEA, 2015b):

- a) ^{233}U with a half-life of 159,200 years. ^{233}U has a daughter product of ^{229}Th , which contributes significantly to the radiotoxicity of the spent fuel. However, this difference becomes much less noticeable if ^{233}U is recycled and ^{233}U is usually necessary to be reused as it is an attractive fissile fuel.
- b) ^{231}Pa with a half-life of 33,000 years, which results from the beta decay of ^{231}Th (with a half-life of about 25 h) which in turn is formed by (n, 2n) reactions of ^{232}Th . The long half-life of ^{231}Pa makes it an important contributor to long-term global radiotoxic inventory. It has a particularly high radiotoxicity, even higher than the principal plutonium isotopes (^{239}Pu and ^{240}Pu).
- c) ^{232}U with a half-life of 70 years. The main burden comes from its decay products (^{212}Bi and ^{208}Tl), which are highly radiotoxic (with gamma-ray radiation of 1.6 and 2.6 MeV) and which would result in heavy shielding requirements in recycling plants.

In conclusion, in terms of the potential reduction of the radiotoxic mass of the nuclear waste compared to the U-Pu fuel cycle, the benefits of a closed Th-U fuel cycle at equilibrium are more modest. One disadvantage of the Th-U fuel cycle arising from reprocessing spent fuel (for recovery and recycle of ^{233}U), is that it is the required to avoid off-gas radon transport of the highly radioactive radon daughters through high-efficiency particulate air filters to the environment. Therefore, retention of about 10 min of the off-gas must be incorporated into the off-gas system design to allow for ^{220}Rn decay before high-efficiency particulate air filtration (OECD-NEA, 2015a, b).

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Assessment of the Relative Environmental Footprint of Nuclear Energy and Its Fuel Cycle

Ch. Poinssot^a and S. Bourg^b, ^a BRGM, French Geological Survey, Orléans, France; and ^b CEA, DES, ISEC, DMRC, University of Montpellier, Montpellier, France

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Glossary

GHG Green-house gases gas that absorbs and emits radiant energy within the thermal infrared range, causing the greenhouse effect. The primary greenhouse gases in Earth's atmosphere are water vapor (H₂O), carbon dioxide (CO₂), methane (CH₄), nitrous oxide (N₂O), and ozone (O₃). They are classically expressed in CO₂ equivalent or CO₂ equivalent.

HLW High-level waste corresponds to the most radiotoxic nuclear waste containing long-lived radionuclides and high radiation fields. They corresponds either to nuclear glasses containing the fission productions, or spent nuclear fuel.

ILW-LL Long-lived intermediate level waste corresponds to intermediate level nuclear waste containing long-lived radionuclides. It basically consists of materials contaminated by actinides in the fuel cycle, in particular the remaining pieces of fuel cladding and hulls and end-pieces.

Life cycle assessment (or LCA) Methodology for assessing environmental impacts associated with all the stages of the life-cycle of a commercial product, process, or service. Environmental impacts are assessed from raw material extraction and processing (cradle), through the product's manufacture, distribution and use, to the recycling or final disposal of the materials composing it (grave).

LLW-LL or SL Long-lived or short-lived low level waste corresponds to waste containing (or not) radionuclides with half-life below 30 years at a low concentration. They mainly comes from the operation of industrial or research facilities and are currently conditioned in drums encased in concrete cells which will finally be protected by additional grounds.

VLW Very low level waste contains a very low radionuclides content, potentially none, but has to be specifically managed due to the absence in France of liberation threshold below which are cleared from any contamination. They are stored in an existing industrial shallow deposit in France.

Introduction

The increasing impact of the overall environmental crisis leads public opinion and stake-holders to take into deeper consideration the actual environmental footprint of the different industrial activities. It not only aims at quantifying green-house gases emissions which contribute to the global climate change, but also other indicators contributing for instance to biodiversity preservation, pollution prevention, resource saving ... Nuclear energy which has always been controversial is hence also questioned on its actual relative impact by comparison to other energy sources, in particular renewables energies. For many opponents, back-end of the fuel cycle is considered as the most controversial part of the nuclear energy system:

- (i) recycling deals with plutonium which is considered by some opponents as the most dangerous radioactive element, and is linked to military applications in people's mind;
- (ii) recycling also requires treating highly radioactive spent nuclear fuel, which they assume to be very polluting;
- (iii) geological repository aims at confining nuclear waste, the lifetime of which is well beyond the human history.

Although it is not scientifically sound, fuel cycle back-end is considered by many opponents as very polluting activities, which should justify the nuclear energy phasing out. Developing reliable and scientifically based environmental indicators is hence of

prime importance not only to have a reliable assessment of the environmental footprint of nuclear energy by comparison to other energies, but also to have a reliable picture of the respective contribution of the various nuclear fuel cycles part.

Numerous sustainability indicators have been developed and proposed by economics or system ecologists to rank given systems by comparing one to each other in terms of sustainability (see for example Gasparatos et al. (2008), Buchholz et al. (2009), Gallego-Carrera and Mack (2010), Evans et al. (2010)). However, many indicators are not yet internationally agreed upon and are for some of them not easily and objectively quantifiable, for instance in the societal field (Evans et al., 2009; Lior, 2010; Piera, 2010). Furthermore, beyond the individual and elementary indicators, there is a real need for composite indicators, named indices, which can be more easily handled by the stakeholders. However, deriving indices is also a matter of real debate on the way to aggregate numerous and heterogeneous criteria in a limited number of upper-level indices (Gasparatos et al., 2008).

Life cycle assessment (or life cycle analysis, LCA) is the main basis for deriving relevant and integrated indicators and has been widely applied to different energy sources, such as wind power, hydroelectricity, biomass, photovoltaic energies (Dones et al., 2009; Lior, 2010; Goralczyk, 2003; Evans et al., 2009; La Rovere et al., 2010). Several studies have also addressed the specific case of a target system (for instance a major city) and attempted to rank the different types of energy sources in order to derive the most “sustainable” one (Jovanovic et al., 2010; Schwenk-Ferrero and Andrianov, 2017). However, very few studies have dealt with the specific case of nuclear energy, and they yield a wide range of results, depending on the chosen nuclear fuel cycle scenario, the type of technology implemented (especially for the uranium enrichment) and the national electrical mix (e.g., Kessler (2002), Lenzen (2008) and Dones et al. (2009)).

Rationale and limitations of LCA

Several methodologies have been described to implement relevant LCA. Among the different approaches and studies, the most critical point concerns the database to be used for the reference primary data. Most of the studies are directly using the well-known Ecoinvent database, which has been developed by several institutions associated to the ETH Zürich (see the webpage <https://www.ecoinvent.org/>). However, some of the data may not be of relevance or are obsolete, in particular for nuclear activities, as very few studies are targeting to update them. For instance, the enrichment process is still considered to be based on gaseous diffusion which is quite obsolete and less and less used. For instance, in France, gaseous diffusion plant (called GB-I plant) was closed in 2012 and replaced by a more modern plant based on ultra-centrifugation which consumes 24 times less electricity (100 kWh/UTS instead of 2400 kWh/UTS). As the enrichment process requires large amount of electricity to be operated, its overall impact strongly depends on the carbon intensity of the country where the process is implemented. Enriching uranium in a country where electricity has a low carbon content as in France yields much lower CO₂ emissions than implementing it in country where electricity has a high carbon content such as China. Extrapolation from one country to the other is therefore all but obvious and requires a lot of precaution. Subsequently, direct use of Ecoinvent database may lead to very significant bias in the overall result, in particular if a careful check of the basic supporting data is not realized. For this reason, results of LCA can be highly scattered from one study to the other, yielding difficulty to get a representative trend without a more in-depth analysis. For instance, CO₂ emissions for nuclear electricity can vary in the literature by almost two orders of magnitude, between a few gCO₂/kWh to more than 100 gCO₂/kWh for the most extreme ones (Lenzen, 2008; Warner and Heath, 2012; Turconi et al., 2013).

In order to overcome this intrinsic difficulty, Poinssot and coworkers developed in a previous study a specific database solely based on factual and published data concerning the actual impact of the nuclear fuel cycle in France (Poinssot et al., 2014; Serp et al., 2017; ASN, 2019). This study was developed by using the Process Chain Analysis (PCA) approach from cradle to grave (construction, operation, dismantling, including transports between all the fuel cycle facilities). The analysis was performed on a yearly basis for the whole French nuclear fuel cycle. For all the facilities (fuel cycle plants or reactors), both the contribution of their construction and dismantling, and that coming from their operation along their lifetime was considered with a lifetime assumed to be 40 years, which is likely a conservative assumption for most of them. In order to simplify the calculations, it was assumed that the French reactors park is only fed by the fuel from Orano (at that time, Areva) front-end plants, which is actually only true for about 50% of the annual fuel input. Finally, input data were directly taken from the actual consumption and release data for year 2010 as described in the yearly environmental and safety report produced by any nuclear facility in France thanks to the Nuclear Safety and Transparency Law of 2006.

This study was therefore a unique opportunity to derive relevant estimate of the contribution of the different fuel cycle steps on the overall environmental footprint, which explains why it is taken as a relevant and reliable case study in this article.

Description of the nuclear fuel cycle

The simulation is based on the French nuclear fuel cycle taken as a representative situation. It considers the whole fuel cycle, from the ore mining to the geological disposal, through the conversion, the enrichment, the fuel fabrication, the electricity production within the reactors, the fuel storage, the fuel recycling and the different types of waste conditioning plant and interim storages. Geological Disposal for ultimate waste planned to be built in France by 2025 is also included. Non-reprocessed spent fuels are

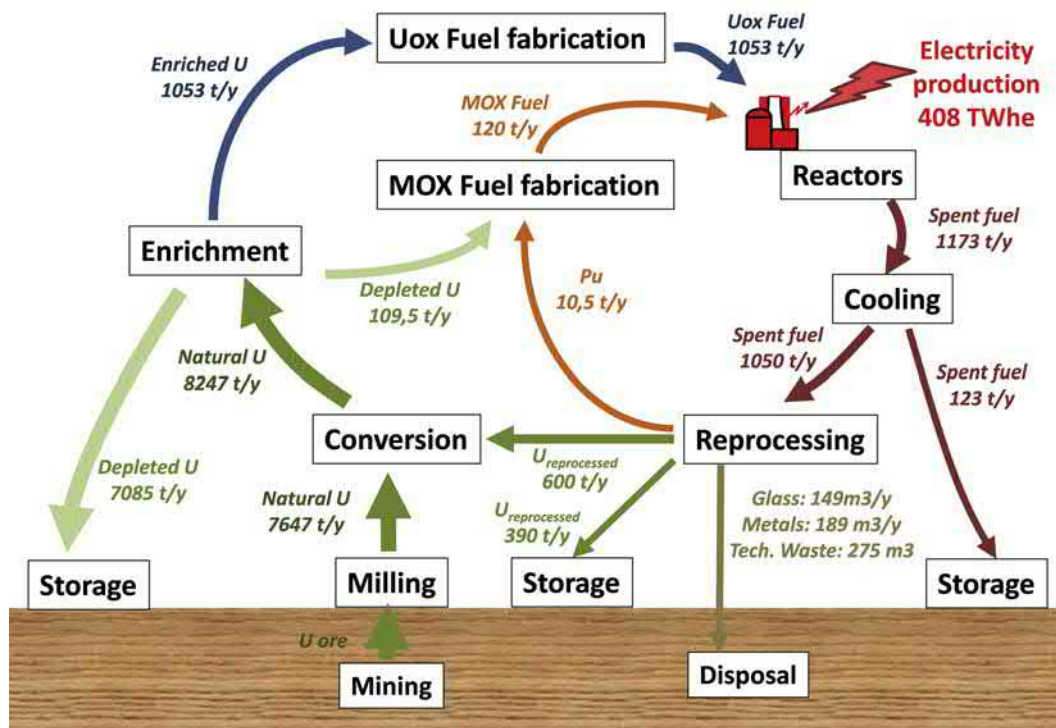


Fig. 1 French reference fuel cycle and its representative streams (Year 2010). From Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J. (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

not considered as waste since they are planned to undergo a delayed recycling for feeding future 4th generation Fast Neutron Reactors FNR (for instance Sodium Fast Reactors (SFR)) (Fig. 1).

Eight key generic environmental indicators have been selected based on their technical relevance and frequency in literature:

- GHG emissions (mass of CO₂ equivalent, g per electrical kWh).
- the atmospheric pollution (mass of SO_x and NO_x, mg per electrical kWh).
- the water pollution (mass of pollutants, mg per electrical kWh).
- the land-use (surface area, m² per electrical GWh).
- the water consumption (water is not released to the environment) and withdrawal (water is released after cooling) (volume of water, L per electrical kWh).
- the production of technological waste (mass of waste, g per electrical MWh).

Three additional indicators were also selected to address the radioactive gaseous and liquid releases (activity, Bq per electrical kWh) and the solid radioactive waste production (mass or volume of waste, g per electrical MWh, or m³ per electrical MWh).

Finally, five potential impact indicators have also been assessed: soil acidification, surface waters eutrophication due to phosphate release, photochemical ozone creation potential (POCP) in the lower atmosphere, overall eco-toxicity and human-toxicity. By definition, they refer to a potential maximum impact that the overall liquid and gaseous release from all the fuel cycles plants could generate and represent boundary overestimation of the environmental impact (Fig. 2).

Comparison of the nuclear energy environmental footprint with other energy sources

Based on this approach, the LCA of the entire French nuclear fuel cycle was performed, where every activity, from the mining uranium extraction down to the spent nuclear treatment and recycling, and the ultimate waste geological disposal were considered. Even with such a comprehensive description of this industry, the overall environmental footprint of the French nuclear energy is very low by comparison to other energy sources as shown in this Fig. 3.

Nuclear energy is within the top-3 of the most environmental-friendly energy sources, in the same range as hydropower and windpower for most of the indicators except the water withdrawal and the technological waste production. For some of them as GHG emissions, nuclear energy is even the most environmental friendly energy source, far from what people have often in mind and about two to three times lower than the common value proposed by EcoInvent (12 g CO₂ equivalent/kWh).

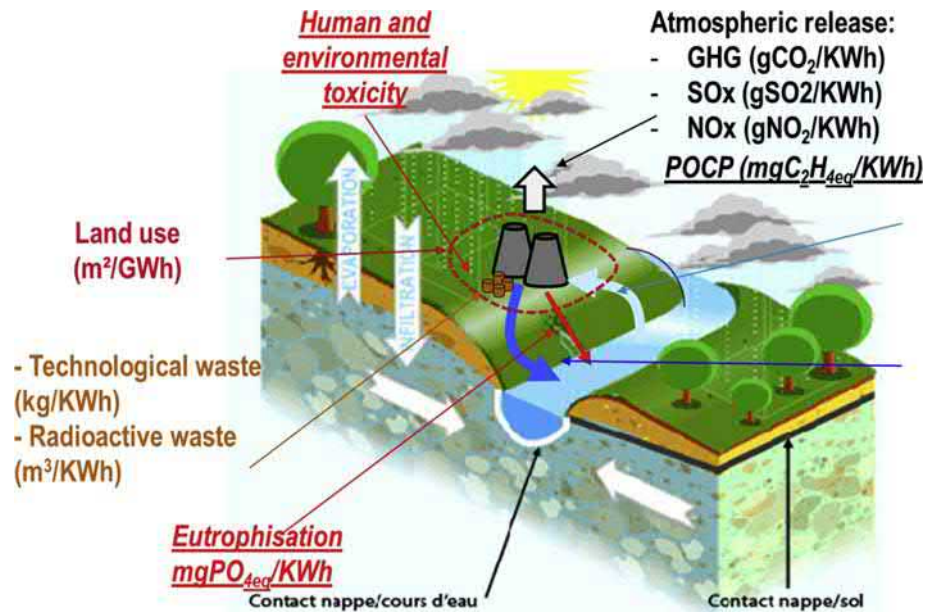


Fig. 2 Key indicators described in this article. From Poinssot Ch, Bourg S and Boullis B (2016) Improving the nuclear energy sustainability by decreasing its environmental footprint. Guidelines from life cycle assessment simulations. *Progress in Nuclear Energy* 92: 234–241.

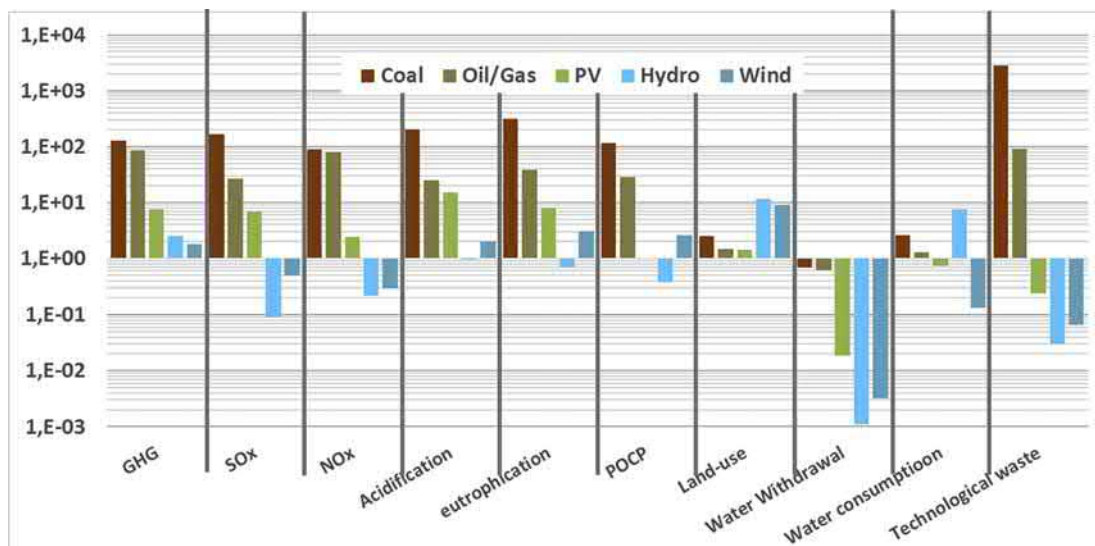


Fig. 3 Comparison of the relative environmental impact of the different energy sources normalized to the nuclear energy impact for the selected impact indicators. Values higher than unity indicate that the impact is higher than that of nuclear energy. Values lower than unity indicate that the impact is lower than that of nuclear energy. Adapted from Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

Place of nuclear fuel cycle within the overall nuclear environmental footprint

Based on the LCA approach and the previous assumptions, the relative contributions of each step of the nuclear fuel cycle has quantitatively been assessed by associating the construction/deconstruction impact of each plant to their respective cycle step (Fig. 4).

This figure clearly illustrates the relatively low impact of the fuel cycle back-end by comparison to the front-end or reactors activities: Fuel cycle back-end activities are all below 10% of the overall environmental impact, except for CO₂ emissions that represent 30%. For water withdrawal or consumption, ecotoxicity, human toxicity, fuel cycle back-end contributions are even negligible and below 1% of the overall impact. Within the back-end activities, geological repository and reprocessing are the most significant contributions depending on the indicators: repository dominates the GHG emissions, the landuse, the NO_x emissions and the waste production, whereas reprocessing dominates the SO_x emission, water pollution and eutrophication.

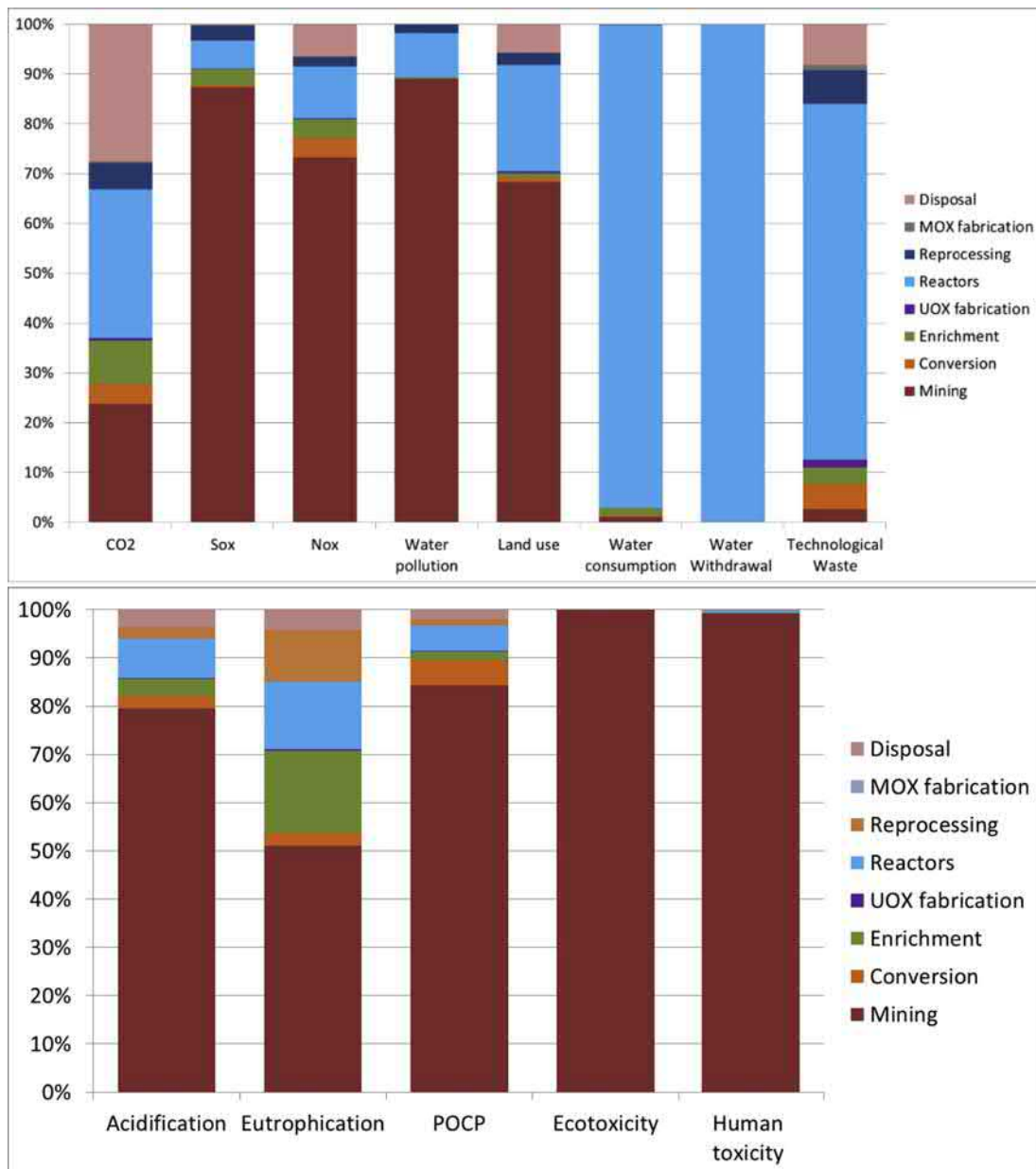


Fig. 4 Relative contribution of each step of the fuel cycle to the environmental and technological impact indicators for the French fuel cycle. From Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

Reversely, fuel cycle front-end plays a major role on the overall impact, with the exception of the water consumption/withdrawal and the technological waste production, usually representing between 50% and >99% of the overall impact. It clearly illustrates the very strong disequilibrium existing between front-end and back-end activities, this latter being far more environmental-friendly than the front-end despite the usual thinking. This strong imbalance supports any option that would allow either reducing the environmental impact of the front-end activities, or decreasing the volume of uranium to be treated within the front-end activities, in particular by implementing an efficient recycling of the spent nuclear fuels.

Impact of mono-recycling in PWR

In order to better assess the relative impact of recycling on the overall environmental footprint of nuclear energy systems, the impact of a theoretical energy system producing the same amount of electricity but without any recycling was estimated by correcting adequately the various material fluxes in the nuclear fuel cycle. Basically, an additional uranium flux in the front-end was to be included to compensate the energy produced by uranium and plutonium recycling. The corresponding overall fuel cycle is given in Fig. 5.

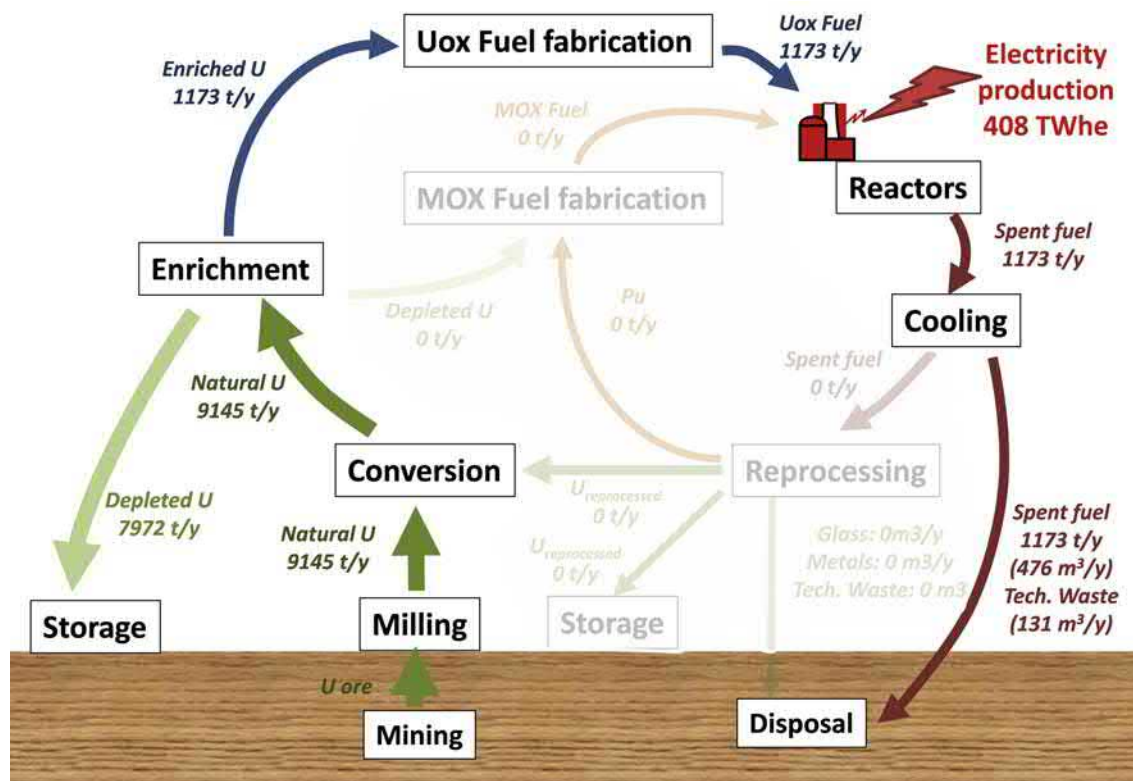


Fig. 5 Description of a theoretical nuclear fuel cycles producing the same amount of electricity without any spent nuclear fuel recycling. The uranium flux in the front-end has therefore to be increased by roughly 1500 t/year (Poinssot et al., 2014). From Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

Comparison of the indicators for the two fuel cycles allowed us to directly assess the impact of the recycling. It clearly demonstrates that recycling has a very positive impact on the overall environmental footprint of the nuclear energy, with the exceptions of rare gases and tritium emissions, and ILW-LL waste.

For a fuel cycle where up to 20% of natural uranium can be saved thanks to U and Pu recycling, the improvement of the environmental footprint ranges between 10% and 20% of the initial open (once-through) fuel cycle footprint (Fig. 6).

Regarding the High-Level Waste which concentrates more than 99% of the initial radioactivity, the gain is much more substantial as it is decreased by a factor of more than two due to the efficient conditioning of the ultimate waste as nuclear glass. Reversely, the spent nuclear fuel processing creates new waste which are contaminated by long-lived radionuclides and are therefore categorized as ILW-LL, in particular CSNF hulls and end-pieces (Fig. 7).

It is also noteworthy that recycling allows a significant reduction of the HLW burden as both the residual thermal power, the residual radioactivity and the volume of waste are significantly decreased by CSNF recycling. Therefore, it has a very positive impact on the geological repository design as the residual heat power is one of the most influencing parameters for establishing the repository volume and surface (Fig. 8).

Finally, although releases of tritium and noble gases into the atmosphere and ocean are increased by recycling activities, their impact has been demonstrated to be well below the health regulation (in the order of 17 $\mu\text{Sv}/\text{year}$ for the surrounding fishermen, and 24 $\mu\text{Sv}/\text{year}$ for the surrounding farmers, to be compared to the average French natural radioactivity of 2400 $\mu\text{Sv}/\text{year}$) and with no impact on the health of the surrounding population (Fig. 9).

As a conclusion, CSNF recycling as an overall positive impact on the environmental footprint of the whole nuclear energy system due to the dominant impact of the front-end activities which are reduced when recycling is implemented.

Impact of multi-recycling in FNR

Similar study was performed to assess the potential impact of implementing CSNF multi-recycling thanks to the use of Generation IV Fast Neutron Reactors (FNR) as Sodium fast reactors. In order to overcome the lack of actual impact data for the Generation IV reactors, past French industrial experience with FNR was used and data from Phenix and Superphenix reactors operation were considered (Serp et al., 2017).

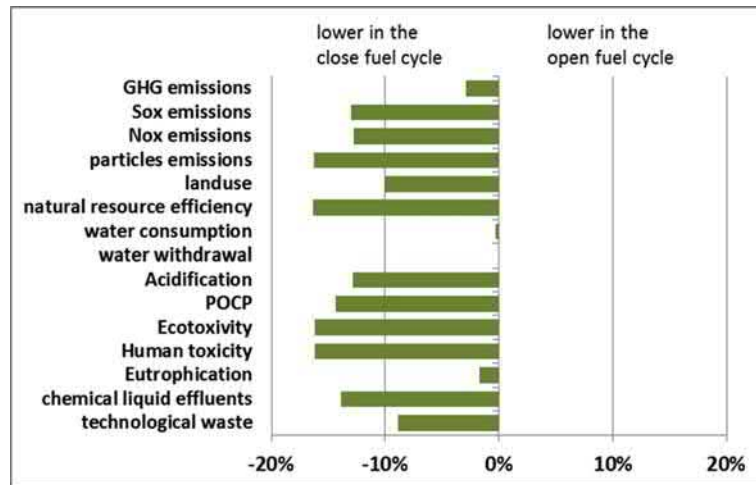


Fig. 6 Relative evolution of the environmental indicators when simulating the implementation of the mono-recycling within the French nuclear fuel cycle. Negative values indicate that the impact is lower with recycling, whereas positive value indicate that the impact is lower without recycling. Adapted from Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

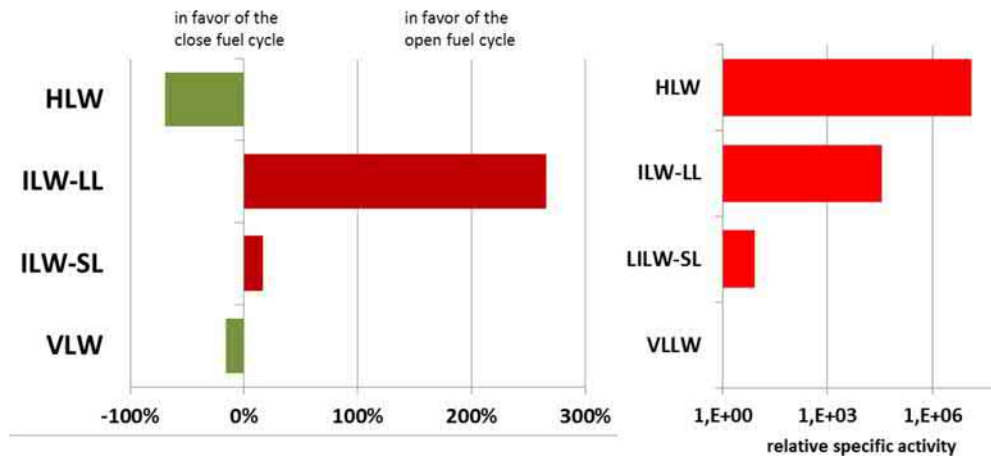


Fig. 7 Impact of CSNF recycling on the waste typology and relative specific activity for each waste type. Adapted from Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

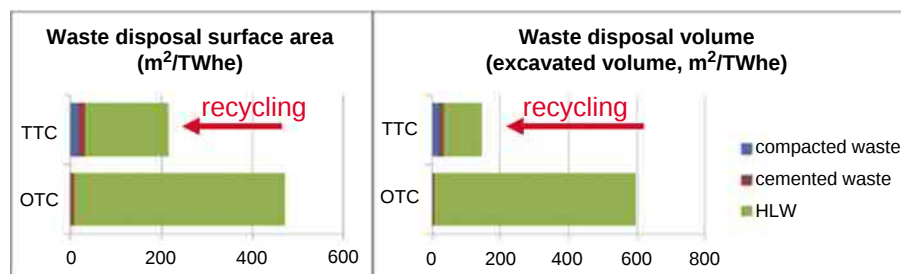


Fig. 8 Impact of CSNF recycling on the repository surface area and volume. Adapted from Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

In such a fuel cycle, since France has stockpiled more than 300,000 Mt. of depleted uranium, the mining activity would not be necessary anymore, the multirecycling of the plutonium provides the fissile material and the depleted uranium the fertile material (Fig. 10).

It yields to similar conclusions than before, i.e., the more intensive the recycling, the lower the environmental impact as evidenced by the following synthetic figure (Fig. 11).

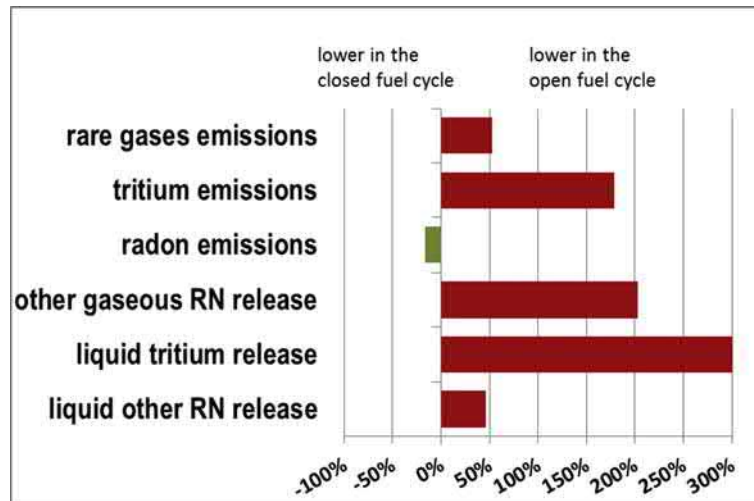


Fig. 9 Impact of CSNF recycling on gaseous and sea release in the back-end related to the impact of the reprocessing plant. RN stands for Radionuclides. Adapted from Poinssot Ch, Bourg S, Ouvrier N, Combernoux N, Rostaing C, Vargas-Gonzalez M and Bruno J (2014) Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles. *Energy* 69: 199–211.

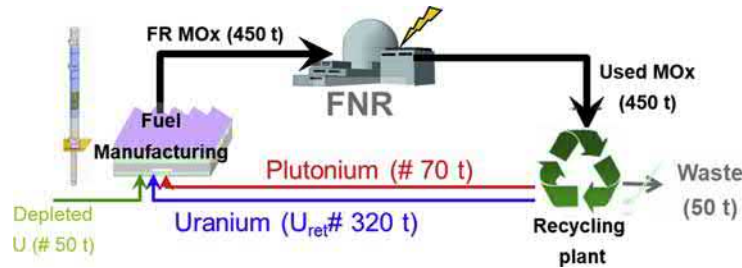


Fig. 10 the equivalent FNR fuel cycle that would produce the same amount of electricity than the current French fuel cycle. From Poinssot Ch, Bourg S and Boullis B (2016) Improving the nuclear energy sustainability by decreasing its environmental footprint. Guidelines from life cycle assessment simulations. *Progress in Nuclear Energy* 92: 234–241.

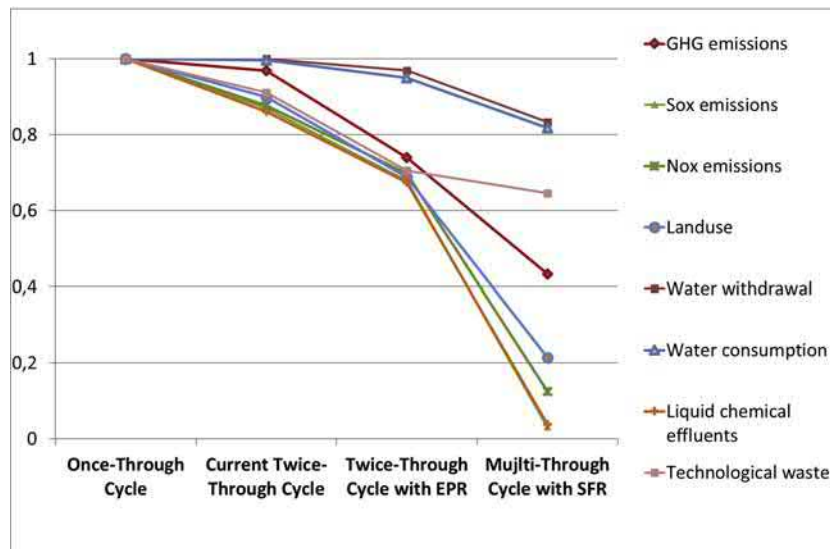


Fig. 11 Impact of CSNF multi-recycling on the overall environmental footprint of nuclear energy. Multi-through cycle stands for indefinite continuous recycling. Twice-through cycle using EPR reactors are more effective than current fuel cycle due to the efficiency increase of EPR reactors and their capacity to use full MOX core. Adapted from Serp J, Poinssot Ch and Bourg S (2017) Assessment of the anticipated environmental footprint of future nuclear energy systems. Evidence of the beneficial effect of extensive recycling. *Energies* 10: 1445.

Conclusion

LCA approaches are relevant and powerful to assess the respective environmental footprint of any human activities as they allow us to consider the integrated contribution all along the activity lifetime, from cradle to grave. Based on this objective approach, environmental footprint of fuel cycle back-end was assessed by considering France as a reference case. It comes out that most of the environmental footprint of nuclear activities is due to the front-end activities rather than the back-end, the impact of which remains rather low. Within the back-end, geological repository and reprocessing are the most contributing activities, depending on the environmental indicators considered. This significant imbalance between front-end and back-end activities strongly support the environmental interest of implementing spent fuel recycling as it increases the back-end activities at the expenses of the front-end activities. Tentative simulations of future fuel cycles where multi-recycling could be implemented thanks to fast neutron reactors enlightens the very positive impact of recycling and the subsequent significant improvement of the environmental footprint since the accumulation of the depleted uranium stockpiled within the PWR operation lifetime would lead to a significant reduction of the uranium mining activity, not to say its full removal (Poinssot et al., 2016). Similar results have been obtained in other recent studies, especially from Schwenk-Ferrero and Andrianov (2017) where they compared different fuel cycle scenarios both on environmental footprint, waste management and economics. Unsurprisingly, nuclear fuel cycles with higher degree of recycling yields significantly improved environmental footprint. In contradiction to the common perception, the nuclear fuel cycle backend has a very little detrimental environmental impact and significantly contributes to improving the overall nuclear energy low environmental footprint by comparison to other energy sources.

See Also: Advanced Fuel Cycles Identification and Evaluation; Self-Sustaining Breeding in Advanced Reactors: Comparison of Fuel Cycle Performance.

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Social Acceptability of Geologic Disposal

Daniel S. Metlay, Institute for International Science and Technology Policy, George Washington University, Washington, DC, United States; and B. John Garrick Institute of Risk Sciences, University of California, Los Angeles, CA, United States

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Introduction and framework

High-level radioactive waste (HLW) and spent nuclear fuel (SNF) are inevitable by-products of nuclear-weapons manufacturing and nuclear-electricity generation. Reliable information about how much HLW and SNF have so far been created worldwide is hard to obtain. But, as an indication of the scale, approximately 90,000 metric tons of commercial SNF has been generated in the United States alone (GAO, 2019). Yet, despite efforts that began more than a half-century ago, none of the world's HLW and SNF has been permanently managed; all of it rests in temporary storage at hundreds of sites across the globe. The critical challenge of nuclear waste management today, then, is to identify both geologically sound and publicly supported locations where this material can be securely isolated for hundreds of millennia. (Similar challenges face efforts to construct a disposal facility for low- and intermediate-radioactive waste and uranium mill tailings. The development of a facility to dispose of low- and intermediate-level radioactive waste in Ontario is a typical example. This chapter, however, exclusively reviews work to dispose of HLW and SNF.)

The task of siting such an isolation facility, called here by convention a repository, can be conceptually understood as an iterative process: many potential sites “pass” through both “technical suitability” and “social acceptability” filters to yield fewer candidate sites. In concrete terms, the technical suitability filter specifies how factors affecting repository performance will be measured and the standards that performance must meet. The social acceptability filter refers to a range of governance processes, plebiscites, formal rules and regulations, and *ad hoc* public actions, such as demonstrations and protests. If all goes well, one of them is developed into a repository. A snapshot of that process is illustrated in Fig. 1. (This conceptualization was slightly modified and later was published in a report the author drafted for the U.S. Nuclear Waste Technical Review Board (NWTB, 2015). The reader is encouraged to consult that volume for a much more comprehensive analysis than space here permits.)

Significantly, the *order* in which potential sites proceed through the two filters depends on national laws, regulations, and practices. In some countries, the implementer passes potential sites first through the technical suitability filter and then the social acceptability one. In others, the process is reversed. But in all nations, *both* filters must be satisfied before a repository site can be successfully developed.

Early on, waste management programs focused almost exclusively on establishing the scientific and engineering foundations for evaluating whether a potential repository site was technically suitable. Implicitly—and sometimes explicitly—the presumption was that only specialist communities working for the implementer needed to be persuaded about the facility's performance. (For instance, it was not until 1975 in the United States that the implementer had to make its performance arguments to an independent regulator. In other nations, such as Germany, this separation was not required until well into the 21st century.) More recently, that presumption has come under strong attack. As a report prepared by the International Atomic Energy Agency (IAEA) emphasized:

Members of the general public and representatives of local communities recognize that they have a clear stake in the outcomes of [siting] decisions and almost always seek to have their views taken into account by the policy elites (IAEA, 2007).

Not surprisingly then, in many nations, interested and affected elements of the public demanded that the social acceptability filter be substantially raised in importance, even to the point that its requirements had to be met *before* the process could move

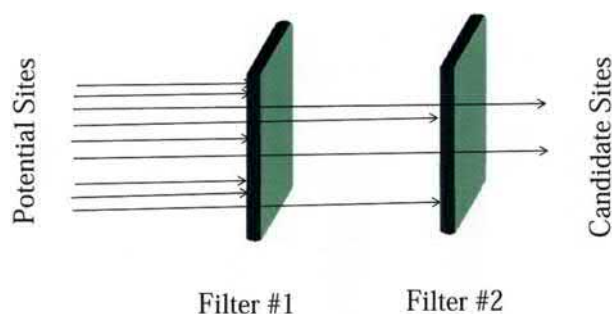


Fig. 1 Siting a Repository. Metlay D (2011) International Experience Developing Deep-Mined Geologic Repositories. *Presentation at AtomEco-2011, Moscow.*

forward. Thus, the history of nuclear waste management in countries with strong pluralist and democratic traditions is truly the history of how the order and the prominence of the filters intrinsic to the siting process were contested. (The author is not unaware of the possibility that the two filters may depend on each other or be influenced by a common third factor. For example, attitudes about whether nuclear power is a valuable energy option may influence views about both technical suitability and social acceptability. Space limitations prevent a discussion of this important question here. The interested reader should refer to (NWTRB, 2015) where it is considered in depth.)

This article explores that history by reviewing how waste-management programs in nine nations—Canada, Finland, France, Germany, Great Britain, Japan, Sweden, Switzerland, and the United States—evolved over the last half century. Although there is some value in telling each country's story separately, such an approach makes it harder to draw comparisons about why some countries appear to be on the brink of successfully developing a repository and why others are still struggling. As importantly, organizing the discussion by examining each nation sequentially masks the observation that each national program (with one exception, Finland) has followed essentially the *same trajectory*.

Thus section **"A strategy for long-term management of HLW and SNF emerges"** discusses how nations initially adopted a strategy that focuses almost exclusively on the technical suitability filter. Section **"Challenging the strategy: Social acceptability emerges as a critical concern"** describes how that strategy was criticized because it failed to give proper consideration to the social acceptability filter. Section **"Why is social acceptability so problematic?"** analyses the underlying bases for that criticism. Section **"Reconstructing the deep-disposal strategy"** considers how the strategy was reconstructed to elevate the importance of the social acceptability filter. Finally, section **"What have we learned? What does the future hold?"** assesses how national experiences inform us about securing social acceptability and what might be expected going forward.

A strategy for long-term management of HLW and SNF emerges

During the first decade of the nuclear age, the HLW from weapons-material production was stored in tanks. The design of the first commercial reprocessing plant at West Valley, New York, adopted the same approach. For the reprocessing plants at Sellafield in Britain and La Hague in France tank storage of HLW was also adopted. For many involved in those efforts, however, the technical and organizational requirements for perpetual care seemed mismatched with the longevity of the wastes' hazards. Could institutions be counted on to secure the material over millennium? Perhaps a less risky method could be found.

It is hard to overestimate the impact of a report prepared under the aegis of the US Academy of Sciences (NAS, 1957). A specially appointed panel proposed burying the material in a conventionally mined repository deep underground. (The panel envisioned disposing of liquid HLW. At the time, leaders of the nuclear power industry expected that commercial SNF like weapons-production SNF would also be reprocessed to extract fissionable plutonium and uranium. The requirement that liquid HLW had to be solidified was not established until 1970 (AEC, 1970). Several years later, a technical consensus emerged that SNF could be just as well disposed of in a repository.) In particular, the NAS report observed that salt formations might prove to be an appropriate host-rock type because of its plasticity and isolation from flowing water.

After a period during which the radioactive waste disposal issue was largely ignored by policymakers, the panel's general strategy gained traction in many nations where HLW and SNF had already been or were about to be created. In the United States, a federal repository was mandated in 1970 (AEC, 1970). After comparing a variety of strategies, the Department of Energy (DOE) formalized the strategy of deep-mined, geologic disposal ten years later (DOE, 1980). Based on an extensive review, the Swedish Government initially approved the deep-disposal strategy (KBS-1 for HLW now replaced by KBS-3 for SNF) in 1979 (Elam and Sundqvist, 2011).

In other nations, disposal in a deep-mined, geologic repository became the default strategy that was not closely scrutinized and evaluated beforehand. For example, in 1965, field investigations began in Germany (BMW, 2008). In 1978, the Swiss nuclear waste legislation set forth requirements for siting a repository (Vomvoris et al., 2013). The French waste-management implementer started its repository research in 1982 (ANDRA, 2019).

It would take many years of dedicated and creative research to fill in the technical details of the strategy advanced by the NAS panel. But based on those studies, a consensus emerged within the scientific and engineering communities that HLW and SNF could be safely disposed not only in salt, but also in clay, crystalline rock, and volcanic formations. (See the chapters in this volume for additional details about the safety cases associated with each of those host rocks.)

What all these nascent national programs had in common was the belief that implementing the strategy involved a relatively simple, straightforward scientific and engineering task—a technical fix (Metlay, 1978). Nowhere in the document authored by the NAS panel, for instance, does it suggest or even imply that affected communities might raise objections or, more generally, that the how and where of disposal represented a broad societal decision. (In fairness, it must be noted that, through the last decades of the 20th century, the panel's perspective was entirely consistent with prevailing practices governing how large-scale technologies were implemented. Small groups of scientists, engineers, industrial leaders, and government officials exercised decisive decision-making power.) By the end of the 1970s and certainly by the end of the 1980s, it was clear that this strategy, which focused narrowly on the technical suitability filter and which hardly recognized the significance of the social acceptability one, would face substantial challenges and doubts on those accounts.

Challenging the strategy: Social acceptability emerges as a critical concern

The Germans were the first to discover the on-the-ground reality of that observation. In 1966, a year after field studies had gotten under way to explore salt formations at Krummendeich and Bunde, the studies had to be terminated because of public opposition (BMW, 2008). In 1976, the State of Lower Saxony began an extensive evaluation of 140 potential locations where the deep-disposal strategy might be executed. This assessment culminated in the selection of a site near the town of Gorleben, which lay adjacent to the East German border. Over the next 40 years, that decision became enmeshed in intense technical and political controversies (Tigemann, 2019). As a result, the social acceptability of the site was never established.

In the United States, the Atomic Energy Commission (AEC) focused its efforts to implement the deep-disposal strategy in a salt formation near Lyons, Kansas during the late 1960s (De la Bruheze, 1992). Technical problems with the site led state officials to successfully campaign against its selection for a repository. Subsequently, governors of Michigan and New York maintained that the federal strategy was socially unacceptable and blocked DOE from conducting surface-based investigations in those states (Carter, 1987). In response, Congress in 1982 passed the Nuclear Waste Policy Act that mandated a site-selection process that ostensibly relied solely on technical considerations. It allowed the *situs* state to disapprove of its selection subject to an override by both houses of Congress. It also expanded the opportunities for interested and affected parties to participate in the process but gave them little formal power to influence its outcomes.

Nonetheless, potentially affected states soon thereafter launched intense campaigns, arguing that the sites DOE had finally settled upon were unacceptable. Five years later, this strong public opposition prompted Congress to amend the 1982 law to exclude all locations except Yucca Mountain in Nevada (Walker, 2009). (The exploding cost of characterizing (investigating) multiple sites was the rationale for exploring a single site. The politics of the process explains why Yucca Mountain was chosen: Nevada's congressional delegation was powerless in the face of strong opposition from Washington and Texas, the two other states then under consideration (Colglazier and Langum, 1988).) Notwithstanding the fact that Congress in 2002 specifically approved the site's selection and that the Nuclear Regulatory Commission's (NRC) staff in 2014 positively reviewed the application to construct a repository, the State of Nevada continues to object.

In 1978, the Swiss implementer, Nagra, started to look at potential repository sites in granite formations. A decade later, the Swiss Government concluded that attempts to execute the deep-disposal strategy in crystalline rock were not likely to be successful (Nagra, 2002). It encouraged Nagra to shift its focus to the Opalinus clay formations that lay in the northern part of the country. Nagra first advanced a proposal to construct a repository for long-lived, intermediate-level waste in the town of Wellenberg located in the Canton of Nidwalden. In 1993, despite support from the community's leaders, nearly 52% of the public voted against the proposal in a cantonal referendum.

In response to this setback, the federal government appointed an independent advisory committee, which recommended monitored long-term geologic disposal and the continuation of the work at Wellenberg (EKRA, 2000). A second cantonal referendum was held in 2002. This time, 58% of the voters rejected the proposal.

In 1979, the Swedish implementer, SKB, began to execute the deep-disposal strategy by drilling and logging dozens of boreholes throughout the country (SKB, 2011). The first investigations, at Finnsjön and Kråkemåla evoked no public resistance. When the studies expanded in 1980 to Kynnefjäll, however, opponents constructed a guard hut obstructing the only path leading to the proposed drill site. The hut is pictured below in Fig. 2. SKB's subsequent efforts in Svartboberget, Klipperås, and Almunge also were frustrated. The national media reported extensively on these protests, leading the Minister of Energy and Environment to reprimand SKB for creating a confrontation and for not properly informing the affected communities in advance of its plans.

It was not until 1987 that Great Britain and France began their search for a deep-mined, geologic repository site (Kemp, 1992) and (Cotton, 2017). The British focused on the disposal of intermediate-level radioactive waste. (The rationale for this focus derived from a belief that by the time it was disposed, HLW would have decayed to ILW.) As part of *The Way Forward* project, the implementer, Nirex, elaborated a nine-step strategy that ultimately winnowed nearly 500 potential sites down to two: Dounreay in Northern Scotland (near an experimental breeder reactor) and the Sellafield complex in west Cumbria, the location of the country's weapons-material production and reprocessing facilities. The selection of those two sites took place in the face of extensive public



Fig. 2 Guard hut at Kynnefjäll. US Nuclear Waste Technical Review Board (2015) *Designing a Process for Selecting a Site for a Deep-Mined Geologic Repository for High-Level Radioactive Waste and Spent Nuclear Fuel: Detailed Analysis*, p. 101. Washington, DC: NWTRB.

consultations that revealed strong opposition in both communities. Based on evaluations of boreholes drilled at both sites in 1989, the implementer concluded that Sellafield was the preferred location. It submitted a planning application for a Rock Characterization Facility four years later.

The Cumbria County Council argued against the application and, relying on its legal authority, called for a public inquiry. Over a period of five months from September 1995 through February 1996, supporters and opponents debated the environmental impact of the surface facilities and examined the projections of long-term safety. The inquiry's moderator recommended that the planning application be rejected. In March, 1997, the Secretary of State for the Environment upheld that recommendation.

The French implementer, ANDRA, sought to carry out the deep-disposal strategy by investigating potential sites in granite, shale, clay, and salt formations. These were located the Deux-Sèvres, Maine-et-Loire, Aisne, and Ain *départements* respectively. Depending on the locality, between 75% and 95% of voters expressed opposition to the studies even before they began (SKN, 1992). At Deux-Sèvres, ANDRA was forced to use aerial surveys after ground survey lines were slashed. At Maine-et-Loire, 400 armed police were called out to quell a demonstration. At Ain, opponents seized excavating equipment and burned documents after entering the local ANDRA office (ERC, 1989).

By 1989, the local and national protests had become so disruptive that the central government declared that the communities must cooperate with ANDRA's efforts. Those pronouncements were of little effect. If anything, they prompted the demonstrators to intensify their campaigns. In early 1990, having just met with politicians and local opponents from Maine-et-Loire, Prime Minister Michel Rocard announced a yearlong moratorium on test drillings at the four sites.

In 1994, the Canadian implementer, Atomic Energy of Canada Limited (AECL), released a detailed environmental impact statement to support the deep-disposal strategy by constructing a repository in the crystalline rock of the Canadian Shield (AECL, 1994). Public hearings began two years later, and a report to the Canadian Government was published in 1998 (Seaborn, 1998). The key conclusion of the report held:

From a technical perspective, safety of the AECL concept has been on balance adequately demonstrated for a conceptual stage of development, but from a social perspective, it has not. As it stands, the AECL concept for deep geological disposal has not been demonstrated to have broad public support. The concept in its current form does not have the required level of acceptability to be adopted as Canada's approach for managing nuclear fuel wastes.

In late 1998, the Canadian Government accepted the panel's advice.

Notwithstanding Japan's deep commitment to nuclear power as an energy source, the Japanese nuclear waste-management program got off to a slow start. Coming belatedly to waste-management policymaking did have one advantage: lessons learned from other national programs, especially the importance of the social acceptability filter could be incorporated into the plans to site a repository.

The Diet passed the Specified Radioactive Waste Final Disposal Act in 2000. Two years later, the implementer, NUMO, invited more than 3000 municipalities to learn more about the possibility of hosting a deep-mined, geologic repository. NUMO proposed a three-stage siting process through which potential sites would be winnowed down to candidate sites. It also set forth conditions that would exclude further consideration of specific sites.

Nine communities initially expressed some interest in accepting NUMO's invitation. All but one, Toyo Town on the southeast coast of Honshu Island, soon lost their enthusiasm. In early 2007, the Mayor informed the relevant governmental body, the Ministry of Economy, Trade, and Industry (METI), that it was willing to proceed. Immediately thereafter, the Governor of the Kochi Prefecture, where Toyo Town is located, travelled to Tokyo to voice his strong disapproval. Within weeks, 60% of the municipality's residents signed a petition to recall the Mayor. As one of his first acts, the new Mayor withdrew the town's acceptance.

In sharp contrast to these experiences, the Finnish implementer, Posiva, has largely been successful in passing its chosen repository site in the Eurajoki Municipality through both the technical suitability and social acceptability filters. The process for finding a repository site in Finland was established in 1983. According to the plan, screening studies throughout the nation would be conducted for two years. Preliminary site investigations would take another six years. Detailed studies would be carried out at least five locations. The final choice would be submitted to the Finnish Parliament and the *situs* municipality for their approval shortly before the end of the century. With little variance, those milestones have been met (Posiva Oy, 2019).

In 2015, the Finnish regulator granted the license to construct the repository (STUK, 2015). Posiva subsequently announced that it expected to commence repository operations in 2025 or 2026. Section “What have we learned? What does the future hold?” below discusses why Finland has proven to be a special case.

Why is social acceptability so problematic?

An extensive literature has documented public attitudes regarding nuclear waste in general and repository siting in particular. (With respect to the United States, see, for example, Slovic (1991) and Dunlap et al. (1993). In 1998, 2001, 2005, and 2008, the European Commission sponsored a series of surveys that focused on the question of nuclear waste management. See, for example, European Commission (2008).) Two pieces of research are especially noteworthy for providing insights into the complex interactions of factors influencing the social acceptability of a repository.

The first study advanced a structural equation model based on the results of a telephone survey of Nevada residents probing their support or opposition to the proposed Yucca Mountain repository (Flynn et al., 1992). The most important path, from “trust” to “opposition,” is shown in red. The reader should note that economic benefits played little role in determining support or opposition to the proposed repository (Fig. 3).

Based on a mail survey of the Swedish population residing near proposed repository sites, the second study examined support for or opposition to a local repository (Sjöberg, 2004). Unlike Flynn and his colleagues, the author did not construct a structural equation model but only reported standardized regression coefficients. Because of the sample’s size (2548 respondents), virtually all of the independent variables examined were statistically significant. The four most important, however, were (in order): dread of nuclear waste, benefits from the repository, attitudes toward nuclear power, and trust in the management of SKB.

In addition to these quantitative studies, investigators have carried out qualitative case studies to understand local reactions to the prospect of developing a repository nearby. One inquiry, for instance, examined local responses to efforts to site a repository in France (Mays, 2004). Another considered how Canadian localities were responding to NWMO’s siting activities (Ramana, 2013).

Both the quantitative and qualitative analyses suggest that a number of factors, singly and in combination, can give rise to challenges to the deep-disposal strategy based on social (un)acceptability. These include:

- Contested technical claims—Are data collected or are the models constructed disputed?
- Public perceptions—What does the public believe about the risks that a repository poses? Does the facility’s presence stigmatize the communities?

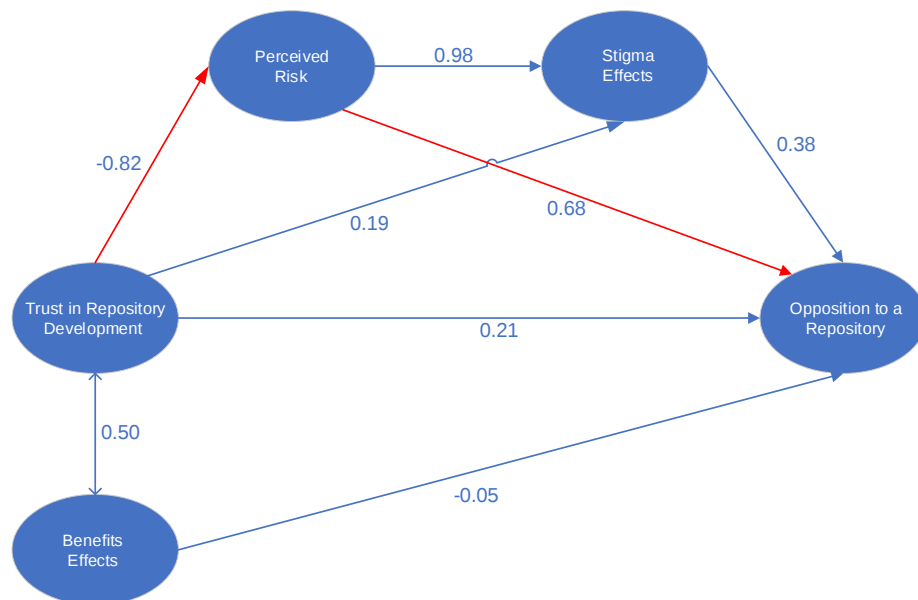


Fig. 3 Opposition to the Yucca Mountain repository. The numbers refer to standardized regression coefficients. Flynn J., et al. (1992). Trust as a determinant of opposition to a high-level radioactive waste repository: Analysis of a structural model. *Risk Analysis* 12(3): 417–429.

- **Mistrust**—Are the implementer and regulatory authorities credible?
 - **Inequitable processes and decision-making power**—Are the decision-making processes considered fair and just? Are interested and affected parties given some decision-making power?
 - **Opposition toward nuclear power**—Should nuclear reactors continue to operate?
- Other, less systematic factors that might contribute to social (un)acceptability are (1) the presence on an intermediate jurisdiction (state or province) between the central government and the locality and (2) a clear prohibition for accepting foreign HLW and SNF.

Contested technical claims

In the United States and Germany, contested technical claims undermined the social acceptability of the deep-disposal strategy. In the early 1970s, the AEC's plans to develop a repository in Lyons, Kansas, ran into two technical complications. Large quantities of water used for solution mining of the salt in an adjacent location could not be accounted for. The locations of some of the boreholes drilled to explore for oil and natural gas were unknown (De la Bruheze, 1992). Fifteen years later, the State of Nevada strongly rejected DOE's claims about the technical suitability of the Yucca Mountain site. When the implementer filed a license application to construct the facility, the state countered with nearly 400 contentions that it sought to adjudicate (Adams, 2010).

Attacks were mounted as well about the technical suitability of the Gorleben site. Critics in particular called attention to the so-called "Gorleben channel," a geologic structure that could degrade and erode the overlaying rock of the salt dome (Hocke and Renn, 2009).

Risk perceptions and stigma

The historian Spencer Weart persuasively documented why the general public looks at things nuclear through a very different lens than it perceives other technological hazards (Weart, 1988). Those perceptions and the possible stigmatization of localities that might arise therefrom are termed "special effects." (These effects stand in contrast to "standard" effects, which can arise from both nuclear and non-nuclear projects. Such impacts include, among other things, changes in the tax base, employment level, water and air quality, and housing stocks.)

Psychometricians and sociologists have provided strong evidence that the general public believes the risks associated with the disposal of radioactive waste are significantly higher than the risk calculated by the scientific and technical communities (Slovic, 1987, 1991; Sjöberg and Drottz-Sjöberg, 2009). Those specialists generally bemoan and grumble that such a difference persists, although recently they have accepted that the difference is an important reality that needs to be acknowledged. Such an acknowledgment appears to be a prerequisite to effective communication with interested and affected parties.

The two quantitative studies noted above suggest the influence of risk perception and stigma on the social acceptability of repositories in the United States and Sweden. But case studies provide support for the claim that those perceptions have influenced to varying degrees the social acceptability of repositories elsewhere. In France, for example, the potential site in Ain was originally called "Bresse." Bresse chickens and cheese, however, are especially valued in France. Local farmers were concerned that a nearby repository would stigmatize those agricultural products and reduce their value (ERC, 1989). Yet, even a change in the name of the potential site did not extinguish fears about the possible taint of the localities. In Germany in the early 1970s, the Lower Saxony Ministry of Agriculture objected to examining salt formations, complaining that releases from a repository would stain the reputation of locally produced foods.

Mistrust

The quantitative studies also highlight how trust in the implementer affects the social acceptability of a repository projects in the United States and Sweden. But once again the case studies suggest the important role trust plays in other countries. In Great Britain, the implementer Nirex, not only failed to obtain planning permission to construct the Rock Characterization Facility, but the company's behavior as revealed in the public inquiry permanently damaged its credibility (Cotton, 2017). ANDRA's then-incorporation into the French Atomic Energy Commission made it harder to secure the trust of the communities when it sought to investigate potential repository sites in the late 1980s. After a series of hearings beginning in 1979, the Prime Minister of the State of Lower Saxony announced the selection of the Gorleben repository site. But he prevented the release of any report documenting the basis for the selection. His expectation that trust in his judgment would be sufficient for social acceptability proved to be unfounded (Hocke and Renn, 2009).

Inequitable processes and distributions of decision-making power

Through the 1980s, efforts to effect the deep-disposal strategy lacked transparency and paid little or no attention to the wishes, views, and opinions of most interested and affected parties. Concerns raised about the fairness of the process and how power was distributed contributed to the (un)acceptability of the deep-disposal strategy as well.

In Germany, France, Sweden, and the United Kingdom, the implementers were at first given free rein to identify and investigate potential sites for a repository. Although they had to keep governmental authorities informed of their activities, those bodies were

not required to approve of the implementers' choices. Nor, of course, were localities given the power to object to any particular site. In fact, often communities first learned about the implementers' interest when drilling crews arrived. Not surprisingly—except perhaps to the implementers themselves—once the plans became public, widespread protests arose in all those countries.

In the United States, inequitable processes and distributions of decision-making power strongly motivated the State of Nevada's opposition to the proposed Yucca Mountain repository. As noted above, the 1982 legislation to structure the site-selection process called for technical studies to compare the viability of at least three possible sites. In 1987, Congress amended that law to exclude investigations anywhere other than Nevada. More than 30 years later, the state's supporters still refer to the amendments as the "screw-Nevada bill" (Adams, 2010)."

Opposition to nuclear power

The inextricable connection between the production of nuclear-weapons material and nuclear electricity on the one hand and the management of HLW and SNF on the other provides grounds for certain groups to reject the deep-disposal strategy. In some nations, activists sponsored demonstrations aimed at blocking the development of a repository in the hope of shutting down nuclear power reactors of all types. The reasoning imputed to those parties suggests that resolving the nuclear waste management issue would reduce or eliminate one rationale for opposing continued production of nuclear electricity. In Canada, Great Britain, Switzerland, and the United States, it is unclear how much anti-nuclear movements, *per se* influenced the social acceptability of the deep-disposal strategy. See, for example, Touraine (1983), Cotton (2017), Carter (1987) and Ebbin and Kasper (1974).

But in Sweden and Germany, those advocates have been joined by large segments of the general public. In the former country, the link between nuclear power production and nuclear waste management was the subject of a national referendum held in 1980 (SIFO, 1980; Sunqvist, 2002). The options presented in that plebiscite did not permit a straightforward "yes/no" vote. One option was to allow the reactors operating or under construction to continue to be used until renewable energy sources came on line; it secured roughly 20% of the vote. A second alternative called for at least a partial phase-out; it received nearly 40% of the vote. The third option required a more rapid total phase out; it also collected nearly 40% of the vote. The Swedish Government chose to interpret the results of the referendum as calling for a gradual phase-out. This sharp division among the general population has become somewhat attenuated over time. Nonetheless, Sjöberg (2004) clearly documents the lasting effects of attitudes toward nuclear power on the social acceptability of the deep-disposal strategy.

In Germany, opposition to nuclear power generation has a long and controversial history (Nelkin and Pollak, 1982). Anti-nuclear activists were well-represented at the demonstrations that took place around the Gorleben site. They also led the legal fight against the Konrad repository for medium- and low-level waste. The bulk of the protestors who blocked the repatriation of German utility-owned HLW from the French reprocessing plant at La Hague were part of the anti-nuclear movement. A general consensus has formed around the idea that the German nuclear waste-management program had been held hostage to the politics of nuclear power for nearly a half century (Hocke and Kallenbach-Herbert, 2015).

Reconstructing the deep-disposal strategy

By the end of the 1980s, it had become clear that the deep-disposal strategy, as formulated by the NAS panel, had become socially unacceptable among countries with strong pluralist and democratic traditions. (Other scholars give the label "deliberative turn" to this period. See, for example, Olliges (2019).) Attempts to locate a potentially technically suitable repository site had been contested and blocked time and time again. An awareness took hold in the halls of governmental authorities and in corporate board rooms that repeating past efforts would likely lead to the same failed result. Thus, over the next 25 years, France (1991), Sweden (1992), Canada (2002), Japan (2002), Great Britain (2006), Switzerland (2008), United States (2010), and Germany (2013) began the process of reconstructing the deep-disposal strategy. Common in all those nations was the recognition that the social acceptability filter had to be strengthened and had to occupy a more central role in the siting process. The effort to reconstruct the deep-disposal strategy in each country is discussed below in roughly the order that it took place.

Shaken by the demonstrations that erupted at each proposed site, the French Parliament passed the 1991 Nuclear Waste Act. The main provisions of that law were (Barthe and Meyer, 2012):

- The exploration of new research options became a requirement. In parallel with studies on irreversible and reversible geological disposal, two other alternatives were explored: processes to destroy the long-lived elements in nuclear waste (separation and transmutation), and solutions for storage above ground.
- Local communities have to agree before the implementer could construct an underground research laboratory (URL).
- New procedures were established to improve information and debate.
- New ethical principles were asserted, such as the protection of nature, the environment and health, as well as the rights of future generations.
- At least two different rock formations had to be characterized for the URL.

Over the next four years, ANDRA focused its efforts to site a URL in the Gard, Vienne, Meuse, and Haute-Marne *départements*. The wine producers' fears of stigmatization ultimately led to Gard's exclusion; Vienne proved to be technically suspect. In Haute-Marne, the regional council and the *département* approved the URL's construction, although the Meuse *département* never took a vote. In

early, 1999, ANDRA submitted a license application to construct the facility adjacent to the town of Bure, located near the border of the Meuse and Haute Marne *départements*. The French Government approved that decision six months later.

In 2005, the implementer prepared a dossier supporting the choice of an argillite formation underlying Bure (ANDRA, 2005). (Since the crystalline rocks near Vienne were technically compromised and no other community sitting atop such formations was prepared to volunteer, ANDRA compared the Bure site with an abstract granitic site.) A public debate was then held, out of which came the recommendation that any deep-mined geological repository must be designed to be “reversible.” (I thank Christophe Poinssot for this comment. “Since the basis for defining the concept of deep geological repository was to find a site suitable forever, what reversibility could mean? I think it is a very good example of the necessary hybridization between technical criteria and suitability, and social criteria and acceptability. For sure, an irreversible deep repository is from the experts’ point of view safer, but from the citizens’ point of view, a repository can hardly be accepted if it is not reversible. Reversibility was therefore introduced by the society as a kind of compromise to find a midway path between both positions.”) In 2006, Parliament ratified this condition, thereby establishing a new deep-disposal strategy. That same law specified that the repository for long-lived intermediate-level waste and HLW would, in fact, be located in the region surrounding Bure. After a contentious public debate, Parliament passed the Act of 25 July 2016, which set the terms of construction for the repository and detailed the conditions required for reversibility.

In 2017, a critical report prepared for the Nuclear Safety Authority expressed reservations about the chemical reactivity of packages of bituminous waste, which would be co-disposed in the repository. A 2021 report from the French Environmental Authority also questioned whether ANDRA’s reversibility strategy was sufficient. Studies to resolve those concerns have pushed back plans to submit a construction authorization from the original 2018 date.

By all accounts, ANDRA has successfully embedded itself within the local communities. It has underwritten major development projects, including those upgrading the transportation infrastructure and improving the attractiveness of the area. Its engagement with local advisory committees (CLIs) has largely been productive. Yet, demonstrations involving as many as 500 “pinprick protesters” continue to take place around Bure. Even so, there is certainly no indication that the French nuclear-waste management program is likely to change course.

After the resistance it encountered from targeted municipalities in the late 1980s, the Swedish implementer, SKB, paused its site-specific geologic studies. However, by then, it had concluded that many locations were “good enough” to host a repository based on the KBS-3 logic (Daoud and Elam, 2012). Moreover, SKB determined that even less-than-perfect rock could be engineered to meet the country’s safety standards. Most importantly, the implementer came fully to understand that under Swedish law, municipalities held the final say over whether a repository could be constructed (Swedish National Council, 2014). The NAS panel-inspired deep-disposal strategy was no longer sustainable. SKB’s focus shifted to ensuring that any location ultimately selected would meet the requirements of the social acceptability filter, that is, avoiding a municipal veto.

In sum, SKB now recognized that the success of its siting project depended decisively on finding a willing host community. Also, it became apparent that the old technocratic organization was ill-matched for the social and political environment in which it would now have to operate. As one study observed (NWTRB, 2015):

New leaders were put in place; individuals were rewarded when they formed bonds of trust with local residents; and, as appropriate, officials maintained a strong presence in the municipalities to answer questions and to reinforce the idea that SKB was not necessarily an outsider.

SKB invited all 286 municipalities to host “feasibility studies.” The response was anemic. Two communities in northern Sweden, Malå and Storuman, far away from the nation’s reactors and waste storage facilities, did volunteer. As before, protests arose in each. After the completion of the feasibility studies, the municipalities’ leaders authorized referenda on whether SKB should follow up those initial studies at the sites. In Malå, 71% voted to shut down the investigations; in Storuman 54% objected (Elam and Sundqvist, 2006).

The municipalities’ negative response prompted SKB to prod five nuclear communities—Nyköping, Oskarshamn, Östhammar, Varberg, and Kävlinge—to come forward. The first and the last two never pursued the matter. But, subsequently, three neighboring municipalities—Älvkarleby, Hultsfred, and Tierp—accepted the implementer’s invitation. For reasons both technical and political, in the end, only Oskarshamn and Östhammar remained in the hunt. It should be noted that Oskarshamn and Östhammar are communities where SKB’s owners operate nuclear power plants. In 2011, SKB selected the latter site for the final repository and the former for the required SNF encapsulation plant. Östhammar’s leadership has tacitly supported the construction of a repository throughout the siting process, unlike Oskarshamn’s, which was more vocal in its support. But the Östhammar Municipality unwilling to make a formal commitment until the central government reviewed SKB’s license application.

That review was two-pronged. The safety authority had to decide whether the application met the requirements of the Nuclear Activities Act; an especially appointed Land and Environmental Court had to determine whether the application fulfilled the conditions of the Environmental Code. In 2018, both bodies submitted their advice to the government. The safety authorities’ recommendation was positive; the court’s was negative, holding that there were unresolved questions regarding the longevity of the waste package. Those uncertainties needed to be better evaluated before the repository could be constructed. (The Land and Environmental Court never prepared an official English translation of its decision. An unofficial one was released by MKG, an advocacy organization that opposes approval of SKB’s license application (MKG, 2018)).

In 2019, SKB submitted to the Swedish Government additional technical studies designed to address the Environmental Court’s concerns. That same year the Oskarshamn municipality formally voted to approve the license application for the encapsulation plant. The Environmental Court then revisited its earlier decision and determined that the final repository would, in fact, satisfy

the requirements of the Environmental Code. In November, 2020, the Östhammar municipality agreed to support to license application. As of early 2021, license approval awaits a decision by the Swedish Government.

When AECL's technology-driven approach failed to gain social acceptance in 1998, Canada's government considered a variety of alternatives for moving forward. Four years later, it brought into force the Nuclear Fuel Waste Act. That legislation transferred the responsibility for selecting a repository site from the Crown Corporation AECL to the nuclear power plant owners. That group was instructed to recommend a new deep-disposal strategy that would merit social acceptability.

The new implementer, the Nuclear Waste Management Organization (NWMO), announced that its mission was to "develop collaboratively with Canadians a management approach for the long-term care of Canada's used nuclear fuel that is socially acceptable, technically sound, environmentally responsible, and economically feasible (NWMO, 2005)." NWMO focused sharply on ways to ensure community well-being, a prerequisite for encouraging localities to come forward and participate in the siting process.

And participate they did. Initially 22 communities expressed an interest in learning more about the implications of hosting a deep-mined, geologic repository. They consented to desk-based studies and preliminary site investigations. By the end of 2019, NWMO had deselected all but two possible locations—South Bruce and the area around Ignace. NWMO ended its involvement in the other 20 sites because of concerns that they could not successfully pass through the technical suitability filter. But in several cases, the implementer also concluded that constructing a repository would create fissures within the locality and thus would detract from community well-being, making it unlikely that those locations could pass through the social acceptability filter. Importantly, the leaders of none of the 20 potential sites withdrew from the process on their own volition (NWMO, 2019).

The situation may have potentially changed as the new decade arrived. The majority owner of NWMO, Ontario Power Generation (OPG), had proposed to construct a repository for low- and intermediate-level waste in Bruce, which was the location of several of its nuclear power plants. In January, 2020, 85% of the Saugeen Ojibway First Nation voted its disapproval. Fulfilling its longstanding promise, OPG announced that it would not develop that repository without a willing and informed host and that it would seek alternative solutions. The implications of this development for NWMO's site-selection process for SNF are not clear at this time; the Saugeen Ojibway are also heavily involved in that activity. In any event, NWMO reports that its interactions with the Ignace community are intensifying.

With the collapse in 2007 of NUMO's initiative to engage a willing community potentially interested in hosting a repository, the Japanese implementer stepped back to reassess its approach. Its first response was to enhance its so-called "public acceptance" activities. When that tact yielded no results, NUMO hinted that it might target specific municipalities.

The 2011 tsunami that destroyed the Fukushima-Daiichi reactors fundamentally changed the political and social environment with respect to nuclear power in general and nuclear-waste management in particular. The Science Council of Japan, the equivalent of the NAS, cautioned that NUMO's approach was fundamentally flawed. "There is a limit to what we can do with currently available scientific knowledge and technological capacities to search out geological formations that will remain stable of tens of millennia." The Japanese Government rejected the Council's advice. METI maintained that it would play an active role in choosing a permanent place. More specifically, it would abandon the current system of waiting for volunteers to raise their hands.

That aggressive stance moderated over the next few years. METI published a map delineating areas that would likely fail to pass through any technical suitability filter. These included locations near active volcanos and faults and adjacent to natural resources. Potentially suitable areas, especially with regards to transportation, were found mostly along the coasts of each Japanese island. Using the map, NUMO conducted a series of symposia and "Dialogue-Based Explanatory Meetings" throughout Japan. By the end of 2020, more than 100 consultations had taken place. Then in late 2020 two communities, Sutto Town and Lamoenai Village, accepted NUMO's proposal to conduct literature surveys.

The failure of Great Britain's implementer, Nirex, to obtain planning permission for a Rock Characterization Facility shattered that nation's waste-management program. In its wake, the government established an independent Committee on Radioactive Waste Management (CoRWM) to review a broad range of long-term strategies. The Committee's advice focused on laying out the requirements for a radically different social acceptability filter (CoRWM, 2006). It stressed the

- Importance of continuous public and stakeholder engagement
- Principle of volunteerism
- Need to develop partnerships between the community and the implementer
- Right of withdrawal from the process
- Requirement that key decisions be ratified by democratically elected local bodies.

Two years later, the British Government published a *White Paper*, in which it accepted those recommendations (DEFRA, 2008). Although feelers were extended to other communities, only the Cumbria County Council and two boroughs in west Cumbria, Allerdale and Copeland, were prepared to form a partnership. That alliance worked hard for nearly four years to evaluate the impacts of hosting a repository. It forwarded its comprehensive report to the three governmental participants in the partnership (West Cumbria Partnership, 2012). Under rules stipulated by the Government, all those local authorities had to agree ("three green lights") to move the process forward. In 2013, the two boroughs did; the county council did not.

The next year the Department of Energy and Climate Change issued another *White Paper* announcing changes to the process for siting a repository (DECC, 2014). Presented as a "lessons-learned" document, it advanced several important new policies.

- A National Geological Screening would be carried out to identify in advance sites that were potentially suitable.
- Local communities would lose their planning-permission authority. Boreholes for investigating sites and the repository itself would be subject to national land-use planning.

- Communities would be allowed to withdraw from the process up until the point when a “test of public support” is taken.
- Specific monetary benefits for participating in the process would be provided.

It is noteworthy that the word “partnership” cannot be found in the report. Moreover, important questions, such as what constitutes a community, what is a test of public support, and what authority should exercise the right of withdrawal, were left to the deliberations of a Community Representation Working Group. That body met regularly but never released a formal report.

In late 2018, the Department of Business, Energy, and Industrial Strategy, released a series of policy papers that attempted to address those lingering questions (BEIS, 2018). Working Groups (community partnerships) once again became the main object of engagement. If established, they would have to include at least one “relevant principal local authority,” be it a county or borough council. Importantly, the Working Groups would have sole responsibility for making recommendations to relevant principal local authority/ies on whether to invoke the right of withdrawal and if and when to hold a test of public support. Further, the Working Groups would determine the mechanism for that test, such as a local referendum, a formal consultation process, or statistically representative polling. If there are multiple relevant local authorities in the group, all must agree to hold the test. No single local authority would be able to unilaterally exercise the right of withdrawal. (The policy papers do not indicate how disagreements among the relevant principal local authorities should be resolved. This was the central stumbling block that doomed the West Cumbria Partnership in 2012.) In late 2020 and early 2021, the implementer (now Radioactive Waste Management Ltd [RWM]) announced that working groups had been formed in Copeland and Allerdale respectively. It remains to be seen whether other communities will join the two west Cumbrian boroughs, which had been active for more than a decade.

For the most part, the Swiss nuclear waste-management program has progressed at a deliberate and measured pace. The implementer, Nagra, explored potential sites in crystalline rock but formally abandoned that approach in 2004 for technical reasons. It then focused its attention on locations overlying Opalinus clay formations in the north, publishing a report that detailed a safety case for a repository constructed in that host rock (Nagra, 2002). The OECD’s Nuclear Energy Agency gave the analysis a strong positive review (Nuclear Energy Agency, 2004). Nagra recommended that a site in the Zürcher Weinland be chosen.

The Federal Council concluded that the safety and engineering feasibility of such a disposal facility had been demonstrated. But it maintained that a choice of one site was no choice at all. It asked the Federal Office of Energy to develop a more comprehensive, transparent, and participatory siting process. That agency released its Sectoral Plan in 2008 (SFOE, 2008).

A multistage effort began. Over time, technically suitable regions were identified. Increasingly, local communities became more involved in selecting the location for the repository’s surface facilities, although at least formally they have no role in shaping the postclosure safety assessment of the repository itself. Moreover, the cantonal veto, exercised in Wellenberg, was replaced with the possibility of a national referendum. To date, five potential regions have been winnowed down to three: Zürich Nordost, Nördlich Lägern, and Jura Ost. Nagra at first proposed drilling test boreholes only in Zürich Nordost and Jura Ost. The regulatory authority recommended that boreholes should be drilled in Nördlich Lägern as well. In early 2018, the Federal Council approved that recommendation. Several boreholes have been drilled to evaluate the subsurface characteristics of the three potential regions; some of them yielded unpleasant surprises about the quality of the rock. Based on the results of the geoscientific investigations, Nagra will announce the site or sites for which it will prepare general licence application sometime around 2022. It expects to submit the application by 2024. Although concerns have been raised at various stages, the new deep-disposal strategy appears to have largely secured broad social acceptability.

In accord with the 1987 Nuclear Waste Policy Amendments Act, the United States Congress in 2002 approved President George W. Bush’s recommendation to select the Yucca Mountain site as the nation’s first repository for HLW and SNF (Congress, 2002). In 2008, DOE submitted to NRC a license application to construct a repository. The regulatory agency’s staff began to conduct a series of adjudicatory hearings to review the application. In 2010, DOE sought to withdraw the application. (The unit within DOE responsible for developing a repository had been disbanded.) Although it cited no technical reasons for this action, the DOE Secretary declared that the Yucca Mountain project was “unworkable,” an unmistakable reference to Nevada’s implacable opposition. Several rounds of litigation followed, both at the NRC and before the Court of Appeals for the DC Circuit. In the end, NRC was ordered to spend the small amount of appropriated funds that remained. This enabled the regulatory staff to complete a *Safety Evaluation Report* for the repository.

President Barack Obama in 2011 directed DOE to empanel a Blue Ribbon Commission on America’s Nuclear Future (BRC) to evaluate policies for managing the back-end of the nuclear fuel cycle. The BRC’s *Final Report* appeared 2 years later (BRC, 2012). Its centerpiece was the recommendation that “a new consent-based approach to siting future waste-management facilities” be adopted. Legislation was introduced to authorize many of the elements found in the BRC’s report. None passed. And, although the administration accepted virtually all of the BRC’s advice, it delayed nearly three years before it seriously attempted to initiate programs to effectuate the Commission’s ideas.

In 2017, with the advent of the Trump Administration, focus shifted away from the BRC’s report. Policymakers (once again) turned their attention to promoting centralized interim storage facilities, either private or public. President Donald Trump continued to request funding to restart the Yucca Mountain licensing hearing. The House of Representatives consistently supported the budget request. Those bills, however, never got through the Senate.

In 2020, the administration stopped asking for money to reconvene the NRC’s review. The President tweeted that he heard the “voices of Nevada’s and that his Administration will RESPECT [sic] you.” Although the President indicated that “new innovative approaches” would be explored, it was never made clear what they might be. Other than announcing its opposition to licensing the Yucca Mountain-repository, the Biden Administration has not yet set forth a new waste-management policy. Thus, lack of social acceptability doomed whatever remained of the nearly half-century old nuclear waste management program in the United States.

In 1999, after the Social Democratic and Green Parties took over the reins of government in Germany, they appointed a Committee on a Site Selection Procedure for Disposal Sites (AkEnd, 2002). The government instructed the committee to start with a blank “white map” of Germany and to develop a process for siting a single repository that would take all forms of radioactive waste. The committee proposed an elaborate set of qualifying and disqualifying conditions to winnow down potential sites. The panel held workshops, public meetings, and numerous meetings with interested and affected parties to obtain their views on how to revise the prevailing deep-disposal strategy. Ultimately, however, the committee failed to convince federal legislators that it was time for a change.

In 2011, after several years of vacillation and in the wake of the tsunami-triggered accident at the Fukushima-Daiichi reactor, the German Government shut down eight nuclear power plants and committed to close the remaining nine reactors by 2022. Once it became evident that this policy would not be reversed, the door was opened for the first time in decades to reconsider the siting of a repository for HLW and SNF.

With substantial multi-partisan support, Parliament passed the Search and Selection of a Site for a Repository for Highly Radioactive Waste Act (StandAG) in 2013. The key provision called for the appointment of a Commission on the Storage of High-Level Radioactive Waste. Its charge was virtually the same one presented to the AkEnd panel. But its conclusions differed in some important respects (NTW, 2017).

- The “best-possible” safety of the repository should be ensured for a period of 1,000,000 years.
- The site-selection process should start with a blank map of Germany. (This language was adopted to allow the Gorleben site to be included in the search. It also meant that sites possessing host-rocks other than salt, such as clays and granites, had to be evaluated.)
- The Bundestag should ratify each of the five steps specified in the site-selection process.
- The repository should be designed to be reversible.
- Extensive and robust participatory procedures should be scrupulously followed, although the selected community would not have the formal authority to veto the choice of site.
- New implementing and regulatory agencies should be established and their credibility should be tightly guarded.

In 2017, the Parliament amended the StandAG to incorporate almost all of the Commission’s recommendations.

Even before then, Germany embarked on the extraordinarily demanding task of fundamentally restructuring its nuclear waste-management program. A new implementer, the Federal Company for Radioactive Waste Disposal (BGE), and a new regulator, the Federal Office for the Safety of Nuclear Waste Management (BsE), were established. The operations of low-level disposal facilities at the Asse II mine, Morsleben, and Konrad were consolidated under BGE. In addition, the implementer created the National Advisory Board, an independent body that would monitor the siting process to ensure that the public’s best interests were protected. BsE started the long process of completely revamping its waste management regulations to accommodate the possibility that a site in clay or crystalline rock might have to be evaluated. In particular, it realized that new regulations that would have to be adopted that permitted the comparison of sites in different host rocks.

The site-selection process gained momentum over the next three years. In late 2020, BGE published an analysis that evaluated the “blank map” against the exclusionary criteria contained in the Commission’s report. It concluded that Germany’s total high-activity waste could be disposed in 15 out of the 16 states in 90 areas covering 54% of the country. Among those locations eliminated was the Gorleben site because of its potential for erosion.

Most observers have concluded that the legally specified deadlines for site selection (2031) and repository operation (2050) are optimistic. It remains to be seen whether pursuing them over the next few decades will sustain or degrade the current reservoir of social acceptability.

What have we learned? What does the future hold?

In the final section of this article, I reflect on the special case of the Finnish nuclear waste-management program and ask why it avoided all of the disruptions that plagued the efforts in so many other nations. I then evaluate the material presented above and use that information to ponder what might lie ahead.

The special case of Finland

Finland has four operating reactors, two built using Swedish technology and the two built using Soviet technology. Initially, the Soviets agreed to accept the SNF from its plants, so, at its earliest stage, the Finnish waste-management program had only to address the disposal of roughly half of its SNF inventory. In 1997, the Russians abrogated the agreement to repatriate the waste. The Finnish program was revised to accommodate this change in circumstances.

In 1983, the Finnish Government adopted the deep-disposal strategy. It set forth the main goals and milestones for the site-selection program. Several potential sites, whose selection would be based on careful studies of the whole country, would be characterized, and the location for the repository would be selected by the year 2000 (McEwen and Äkäs, 2000).

Between 1984 and 1986, the implementer (now Posiva Oy) screened regional blocks of land and recommended that 101 of them be explored further. The government approved 85. Using the technical suitability filter, these were winnowed down to five

in 1987 including Eurajoki, the location of the two Swedish-designed reactors. A decade later, after the agreement with the Russians was terminated, a sixth site was added at Håstholmen, the location of the Soviet-designed reactors. An environmental impact assessment was prepared that down-selected the number of sites from six to two (Posiva Oy, 1999). Relying on this assessment, which explicitly considered the social acceptability filter, the implementer recommended that Eurajoki be chosen. In 2001, having obtained the consent of the municipal government, the national government passed a “Decision-in-Principle,” officially picking Eurajoki. In 2012, Posiva Oy submitted an application to construct the repository to its regulatory authority, STUK. In 2015, the authority approved the application, and construction commenced a year later (STUK, 2015). Initial operation of the repository is expected to begin in 2025.

The concatenation of five factors that often blocked the passage of sites through the social acceptability filter in other nations never materialized in Finland.

Contested technical claims: The safety cases advanced by the implementers in Sweden and Finland are virtually identical, relying as they do on an updated KBS-3 logic. Yet, in Sweden, SKB continuously had to respond to technical attacks directed at its modeling of the corrosion of the copper canister. In Finland, those challenges were weak and intermittent at best.

Public perceptions: Posiva Oy commissioned opinion surveys at all six potential sites. Not surprisingly, the most welcoming locations were the two nuclear sites. The choice of these communities not only softened negative perceptions, but also promised significant material benefits in the form of property tax payments that covered a substantial portion of the municipal budget (Kojo, 2009).

Meriting and sustaining trust: Although data relating to trust in Posiva Oy is scattered and anecdotal, evidence is strong that the citizens of Eurajoki held STUK in high regard and respected its judgments (Choi, 2018).

Inequitable processes and distribution of political power: Notable is the situation in Finland that the plan governing the waste-management program retained its legitimacy for nearly two decades. The implementer never experienced fierce objections to what it proposed. Moreover, by a 20-7 vote, the Eurajoki Municipality’s concurred with the 2001 Decision-in-Principle.

Opposition to nuclear power: Intense parliamentary debates took place in 1993 surrounding the decision to forego building a fifth nuclear power plant. But those fights barely spilled over into the larger public arena, a sharp contrast to the bitter battles over nuclear power a decade earlier in Sweden. Indeed, without much controversy, parliament approved a fifth reactor nearly a decade later (Lammi, 2009).

Prospects

Overall, it is undeniable that national nuclear-waste management programs have made considerable progress in tackling the social-acceptability issue over the last twenty years; efforts to reconstruct the deep-disposal strategy have been largely but certainly not completely successful.

The technical basis for safe disposal of HLW and SNF in crystalline and a variety of clay host-rock formations has been established in principle. (See, for example, these international peer reviews of disposal strategies in a variety of host rocks: Nuclear Energy Agency (2004, 2006, 2012).) To be sure, non-governmental advocacy organizations in Sweden still contest the viability of the KBS-3 method. The State of Nevada maintains its position that the Yucca Mountain site is technically unsuitable and that DOE’s license application to build a repository there ought not to be approved. Nonetheless, in most countries, contested technical claims rarely drive the (un)acceptability of geologic disposal.

In two respects, however, the newly reconstructed German waste-management program probably represents the test case of whether an implementer can avert strongly contested technical claims from undermining the acceptability of its activities. First, it will be closely scrutinized on how it assesses the technical suitability of the controversial Gorleben site. (As noted earlier, the “blank map” assessment excluded it.) Second, the program will also be judged on how it lays a strong technical foundation for evaluating specific sites not only in salt but also in crystalline and clay formations, host rocks that hitherto have not been a focus of research and engineering.

Unquestionably, public perceptions of the risks that HLW and SNF pose and the stigma they evoke will continue to be more negative than the views held by technical specialists. Implementers have generally acquiesced to this fact of life. Their response has been to focus repository-siting efforts, not always successfully, on nuclear communities (Eurajoki, West Cumbria, Yucca Mountain, Östhammar) or economically underdeveloped areas (Bure and northern Ontario) where the negative perceptual beliefs are counterbalanced by strong positive material benefits. So far, in Canada, Finland, France, Switzerland, and Sweden, that tact seems to have served the programs well. In Great Britain and Nevada, the so-called “doughnut effect” has manifested itself; communities near the sites have been strongly supportive, while larger jurisdictions around them, such as a state and or a county, continue to object.

Meriting and sustaining trust remains a work in progress in every nation. As is now commonly understood, trust must be gained over the long-term, but it can be lost in a moment. Historically, the Canadian, Finnish, Swedish, and Swiss implementers and regulators have mostly done quite well in this regard. Those in France, Germany, Great Britain, Japan, and the United States have much more mixed records. The former group of countries explicitly structures its activities to emphasize the centrality of trust. The latter group, while acknowledging the importance of trust, does not appear to make it the core of their actions. Given that waste-management programs unavoidably must endure for many decades, maintaining trust in all countries remains an open question.

In all the nations discussed in this article, except the United States, the legitimacy of the process for selecting repository sites and developing them has generally gone unchallenged. Part of the reason for this acceptance stems from the enlarged role interested and

affected parties have played and will continue to play in reconstructing the deep-disposal strategy. In Finland and Sweden, the affected municipalities have exercised their near-absolute right to veto the license application to construct the repository. (In the United States in sharp contrast, the State of Nevada sought in 2002 to veto the selection of Yucca Mountain, but its objection was overridden by both houses of Congress.) A different approach has been suggested by the Steering Group of the Reset Project (Reset Group, 2018.) In Canada, France, Great Britain and Japan, communities must affirmatively express interest in hosting a repository and can withdraw from the process up until some pre-determined point. Neither Germany nor Switzerland offer localities the right to veto or withdraw from the siting process. But local governments in both nations have strong participatory protections.

Anti-nuclear sentiments clearly affected the early siting activities in Sweden and continued to shape site-selection efforts in Germany for decades. Those attitudes dissipated once decisions were taken to limit or shut down the operating reactors. To be sure, anti-nuclear protests, such as those experienced in the last two years at Bure, are likely to persist. They are unlikely, however, to reemerge as a decisive factor affecting the social acceptability of geologic disposal.

In sum then, progress in Finland, France, and Sweden almost certainly will continue. In Canada and Switzerland, the steady and deliberative efforts are likely lead to the development of a repository. In the remaining four nations discussed in this article, the prospects are much more clouded. Significant regulatory, technical, and political challenges remain in Germany. Successive *White Papers* detailing a siting process in the Great Britain are still struggling to set out a socially acceptable pathway for finding a community willing to host a repository. In Japan, METI's determination to pursue voluntarism appears deep-seated. Whether it will yield positive outcomes remains an open question. More than a decade of political conflict at the highest levels in the United States has resulted in stalemate and programmatic limbo.

As I noted recently, "It is hard to remain sanguine ... But then, working in the nuclear waste disposal business has always been a vocation that has attracted optimists and visionaries, even if they are too often disappointed (Metlay, 2013)." I now have reluctantly concluded that even optimists and visionaries will need greater patience. More than century of the nuclear age will have passed at the very least before we discharge of our long-standing obligation to future generations to securely isolate HLW and SNF for hundreds of millennia.

See Also: International Nuclear Waste Disposal Centers; Nuclear Waste Can Be Disposed of Safely.

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Radiation Protection – Introduction

Laurence Miller, Department of Nuclear Engineering, The University of Tennessee, Knoxville, TN, United States

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The theory and practice of Radiation Protection provide the basis for regulatory requirements that are essential for the safe and beneficial use of many advanced technologies that utilize ionizing radiation. Medical diagnostic protocols, such as Computed Tomography (CT) and Positron Emission Tomography (PET) scans are effective for detecting physiological and biokinetic health concerns, respectively. Electrons, alpha particles, photons, neutrons, and protons are used for both curative and palliative treatment of diseases. Chemical and molecular tracers that are tagged with radioactive isotopes contribute to the understanding of fundamental physical, chemical and biological processes in natural and biological systems. The Spallation Neutron Source (<https://neutrons.ornl.gov/sns>) at Oak Ridge National Laboratory (ORNL) is used for imaging basic structural features of materials, which contributes to improved understanding of material science. Imaging with photons and neutrons is useful for Quality Assurance (QA) procedures that verify manufactured components satisfy specifications. Radiation is used to disinfect spices, to sterilize medical equipment, to cross link polymers, and for various other applications. For example, one of the beam ports of the High Flux Isotope Reactor (HFIR) at ORNL is a source of very low energy neutrons that can be reflected at interatomic distances to investigate molecular structures (<https://neutrons.ornl.gov/hfir>). More detailed information is provided by [Waltar \(2021\)](#) and [Barradas \(2021\)](#).

In addition to medical and scientific benefits derived from the use of radiation, power produced by nuclear fission makes a substantial contribution to the energy infrastructure in the US. According to US Energy Information Administration (EIA; <https://www.eia.gov/outlooks/aeo/>), in 2019, the net US electricity generation was 4161 TWh, which was produced by the following sources: natural gas (1558 TWh; 37%), coal (985 TWh; 24%), nuclear (807 TWh; 19%), wind (296 TWh; 7%), hydroelectric (287 TWh; 7%), solar (73 TWh; 2%), biofuels (31 TWh; ~1%), oil (17 TWh; <1%), and geothermal (16 TWh; <1%). EIA expects an average electricity demand growth of 0.9% per year through 2050. These data show that energy from fission is the largest contributor of carbon-free electrical power in the USA (19%), and it is an established technology that could play a major role in sizable reductions in future emissions of CO₂. Individual coal power plants are reported to generate from 0.58 to 1.59 kg of CO₂/kWh (world-nuclear.org, n.d.), for an average of about 1 kg of CO₂/kWh. Natural gas produces about one-half the amount of CO₂ relative to anthracite coal (epa.gov, 2020); thus, the generation of 807 TWh of energy by fission in the US during 2019 would have displaced about 800 million tons of CO₂ if it had been produced by anthracite coal and about 400 million tons of CO₂ if it had been produced by natural gas. For more detailed information see [Stachowski \(2021\)](#).

All the above-mentioned applications involve the use or generation of radiation and therefore require the theory and practice of Radiation Protection.

Section 7 discusses the main elements of radiological protection that enable the use of radiation emitting sources and facilities for enhancement of quality of life. Articles 00701 “*Ionizing Radiation: Discovery and Dose Quantities*” (Dewji, 2021) and 00702 “*Fundamentals of Health Physics*” (Dewji and Miller, 2021) provide the fundamental quantities and scientific bases for quantifying the impacts of exposure to nuclear radiation and defining radiation doses. Concepts of nuclear physics, radioactive decay, absorbed dose, equivalent dose, and effective dose are introduced and discussed.

Articles 00703 “*Natural and Man-Made Sources of Radiation*” (Miller, Apostolaei, Howard, and Singh, 2021) and 00704 “*Source Terms for Nuclear Reactors Operations, Accidents, and Waste*” (Singh and Wheeler, 2021) describe the sources of exposure to radiation in occupational, environmental and medical settings. These sources include: radioisotopes in naturally-occurring radioactive materials, the nuclear fuel cycle, operations of nuclear reactors, diagnosis and treatment of diseases, industrial and academic facilities, and in radioactive fallout from testing of nuclear weapons.

Methods for monitoring external exposure to radiation are described in Article 00705 “*External Dosimetry*” (Veinot, 2021), and those for internal exposure are presented in Article 00706 “*Internal Dosimetry*” (Snapp and Eckerman, 2021). These articles review standards and regulations applicable to external and internal dosimetry programs, describe the use of human anthropomorphic phantoms commonly encountered in dosimetry programs, and provide a brief summary of algorithms for estimation of radiation doses.

Article 00709 “*Standards to Control Exposures of Workers and the Public*” (Kocher, 2021) focuses on the system of radiation protection developed based on the recommendations of the International Commission on Radiological Protection (ICRP) and National Council on Radiation Protection and Measurements (NCRP). It discusses the justification of exposure (positive net benefit to an

individual or society), optimization of protection (keeping all exposures as low as reasonably achievable [ALARA]), and limitation of dose to individuals. Control of dose to individuals limits the risk of stochastic health effects, such as solid cancers, leukemia, and severe heritable effects, and it prevents non-stochastic (deterministic effects), in organs or tissues.

Practices for controlling exposure to radiation in industrial environments are explained in Article 00713 *“Radiological Control Practices”* (Hansen, 2021). Since actions to further reduce already low radiation doses and risk may introduce or exacerbate other industrial risks, the manner in which target risk reductions are traded for countervailing risk is examined and discussed.

When radiologically contaminated sites previously utilized for medical, defense, or industrial purposes are to be made available for general use by the public, they must be characterized, remediated, and certified for compliance with applicable regulations. These processes are described in Article 00707 *“Instrumentation and Practices for Radiological Assessments”* (Vitkus, King, Altic, Wade, and Ivey, 2021) and in Article 00712 *“Characterization and Release of Contaminated Sites”* (Vitkus, 2021). Computer codes commonly used for calculating radiation doses from residual contaminants are described in Article 00711 *“Computer Codes for Determining Acceptable Levels of Contamination”* (Arnish, 2021). These codes provide the criteria for remediation of contaminated sites.

In some environments, radiation fields are sufficiently high that personnel and equipment must be protected by shielding. General methods for determining the amount and type of shielding in industrial environments are described in Article 00714 *“Radiation Shielding Methods”* (Pevey, 2021), and Article 00715 *“Shielding against Galactic and Solar Radiation in Space”* (Heilbronn, 2021) discusses radiation protection in outer space. Article 00708 *“Emergency Response”* (Marianno, 2021) explains procedures and policies applicable in cases of accidental releases or exposures that exceed regulatory requirements. If a release would be difficult to directly evaluate by personnel, an option for remote surveillance is presented in Article 00718 *“Drones for Radiological Assessment”* (Hackett, 2021).

The U.S. Federal regulations are intended to limit cancer risk via the annual limits on the effective dose and to avoid tissue reactions (deterministic effects) via annual limits on the dose to individual tissues. However, many people are uncomfortable with their perceived risk due to radiation, and this has led to considerable debate about the acceptability of production of power by nuclear fission. One way of addressing this issue is to compare known outcomes from everyday societal activities with known outcomes due to operating various nuclear facilities. Article 00710 *“Risk Tradeoff Analysis”* (Hansen, 2021) analyzes this issue within the framework of risk tradeoff analyses. The health effects based on radiation biology are discussed in Article 00716 *“Health Effects and Radiation Biology”* (Nichols and Byrne, 2021). Data obtained from a collection of cohort studies on the health consequences of exposure to various levels of exposure to radiation, under different conditions, are discussed in Article 00717 *“Estimation of Health Risks from Exposure to Ionizing Radiation”* (Apostoei, 2021). Biological impacts of radiation at the microscopic level are described in Article 00719 *“Microdosimetry”* (Nichols, 2021). Finally, a glossary of terms is presented in Article 00720 *“Glossary”* (Dewji, Kocher and Adadi, 2021).

Article Titles

Introduction to Radiation Protection
Ionizing Radiation: Discovery and Dose Quantities
Fundamentals of Health Physics
Natural and Man-Made Sources of Radiation
Source Terms for Nuclear Reactors Operations, Accidents, and Waste
External Dosimetry
Internal Dosimetry
Instrumentation and Practices for Radiological Assessments
Emergency Response
Standards to Control Radiation Exposures of Workers and the Public
Risk Tradeoff Analysis
Computer Codes for Determining Acceptable Levels of Contamination
Characterization and Release of Contaminated Sites
Radiological Control Practices
 00714: *Radiation Shielding Methods*
Shielding against Galactic and Solar Radiation in Space
Health Effects and Radiation Biology
Estimation of Health Risks from Exposure to Ionizing Radiation
Drones for Radiological Assessment
Microdosimetry
Glossary

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Ionizing Radiation: Discovery and Dose Quantities

Shaheen Azim Dewji, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

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Introduction

The study of the interaction of radiation with matter—i.e., the mechanisms by which energetic waves and particles deposit their energy—along with the associated observed effects of these interactions, have become foundational in informing recommendations, practices, and technologies implemented in radiation protection science.

Radiation is defined as the propagation of energy through matter and space. Radiation can exist in two forms: electromagnetic waves and energetic particles. Radiation with enough excess energy to cause a valence electron to escape the atom, resulting in ionization of the atom, is classified as ionizing radiation. This acts by removing electrons from atoms and molecules of materials that can include air, water, and living tissue. Damage caused by ionizing radiation occurs by the incident wave (i.e., X-ray, gamma-rays) or particle (alpha, beta, neutron) initiating chemical reactions that, in turn, have deleterious effects on tissue. In the context of health physics and radiation protection, the target molecule of primary interest is deoxyribonucleic acid (DNA) in exposed organs or tissues, as well as any intervening materials (e.g., shielding). Ionizing radiation can be classified as directly ionizing radiation, in which charged particles (i.e., protons, alpha, beta, heavy ions) directly transmit energy resulting in an ionization, or indirectly ionizing (e.g., neutrons, gamma-rays, X-rays), which requires a charged particle intermediary to transfer energy to the medium to cause the ionization. The threshold energy for ionization in water is on the order of 4–25 eV (~7.4 eV for liquid water) (Attix, 2008). Ionization can occur with wavelengths shorter than 320 nm, which includes most of the ultraviolet band, X-rays, and gamma rays. Non-ionizing radiation is radiation that lacks sufficient energy to cause an ionization in tissue, thereby limiting the potential damage from exposure and exists around us in many forms and from many sources.

The discovery of X-rays in 1895 by Wilhelm Röntgen initiated studies in the fields of radiation protection and health physics that enabled us to harness the strength of the atom in medicine, industry, and energy, while limiting the harmful effects from unintended exposures of radiation. The beneficial and detrimental effects of Röntgen's rays were quickly noticed, and within a year of this discovery, diagnostic X-ray radiography had become widely employed.

Following the discovery of X-rays, the field radiation protection developed synchronously as subsequent discoveries and the effects of radiation were revealed. Since the early stages of discovery, radiation was proven helpful in the medical field; although, unexamined uses of radiation were followed by serious injuries to the body, including redness on the skin. Several months after Röntgen discovered X-rays in November 1895, the first case of skin injury occurred in the U.S. In many cases, the damage did not appear immediately on the skin. However, in 1905, evidence linking radiation as a causality of cancer was established (Kang, 2016). Soon after realizing the potential harm of unintended use of radiation to the human body, a series of different protection measures were developed.

Management and limitation of radiation exposure has led to the growth of field of health physics, which is defined as the protection of human beings and the environment from the harmful effects of ionizing radiation while permitting its beneficial applications. Health physicists can further be sub-classified as specialists in radiation protection. Regulations and practices have since evolved for establishing exposure limits to members of the public, as well as occupational nuclear workers. Application areas of radiation protection may include the following:

- 1) nuclear energy fuel cycle;
- 2) industry (oil/gas; blood and food irradiation);
- 3) nuclear medicine;
- 4) space exploration;
- 5) consequence management and emergency response; and
- 6) defense and security.

Discovery of ionizing radiation

Early pioneers

Discovery of X-rays

The discovery of the X-ray on November 8th, 1895 was the unexpected result of experiments by Röntgen using a Crooke's tube. In his investigation, a vacuum-enclosed glass tube, much akin to the cathode ray tube of early televisions, was used to transmit electric current—called "cathode rays"—between two electrodes. Operation of the Crooke's tube resulted in fluorescent excitation of the gas in the tube and glass walls encasing the vacuum. While studying the fluorescence in the tube, Röntgen noticed that during operation of the Crooke's tube, a screen located at a distance from the tube fluoresced in a dark room. Realizing that he had discovered a new phenomenon, he referred to these mysterious rays as "X"-rays. Over the course of the following days, Röntgen had amassed a wealth of information regarding the fundamental properties of these X-rays, including the penetrating power through low density, light materials (paper and wood), as well as heavier and higher density materials (aluminum and tin). **Fig. 1** depicts the X-ray of Röntgen's wife, Frau Röntgen's, hand, demonstrating an obvious medical imaging benefit of these new rays, with the hazardous effects



Fig. 1 X-ray picture of the hand of Frau Röntgen made by Röntgen on December 22, 1895. *Reproduced from Turner JE (2008) Atoms, Radiation, and Radiation Protection. John Wiley & Sons.*

yet to be observed thereafter. This image demonstrated the contrasting degrees of skeletal bone and soft tissue as a result of the differing degrees of absorption of the X-rays in the hand. Röntgen further discovered that X-rays, unlike cathode rays, are not deflected by magnetic fields, and that X-rays cause a charged electroscope to lose charge when interacting with photographic plates (later attributed to the ability of X-rays to ionize air, see “**Exposure**” section). The discovery of X-rays marked the first discovery of ionizing radiation. His discovery was subsequently published in a paper entitled, “On a New Kind of Rays” (Röntgen, 1895, 1896).

Soon after the discovery of the X-ray, it became a prevalent topic of research among scientists in different fields. Scientists throughout the world started studying the nature and effect of X-rays and developed multiple applications for it, especially for medical purposes. However, little research was done on the biological impact of X-rays and scientists were unaware of the side effects that could lead to series of health problems.

Less than a year after Röntgen announced his discovery in November 1896, an American engineer, Elihu Thomson, experimented to determine the potentially harmful effects of X-rays. Thomson exposed his finger to the direct radiation from an X-ray tube daily for a few weeks. For the first week, there was no observable effect, but in the second week, he noticed swelling and pain in his hand and had to terminate the experiment immediately and issued a strong warning against overexposure. Although witnessing apparent deleterious effects, some scientists still denied that the effects were due to X-ray exposure. As for the Thomson’s injury, he thought that the injury was due to a chemical effect within the skin tissues (Kathren, 2019).

Another scientist who conducted research on X-rays was William Herbert Rollins. After the discovery of the X-ray, Rollins spent most of his life performing X-ray-related research and was one of the first pioneers in developing radiation protection measures. Rollins approached every important technique of diagnostic X-ray safety, usually calling attention to a particular source of unnecessary exposure and suggesting to lessen that exposure (Kathren, 2019). American engineer, Wolfram Fuchs, provided the first recommendations for radiation protection in 1896, which have become the tenets of radiation protection: (1) Making the exposure as brief as possible (time); (2) standing greater than 12 in. of the X-ray tube (distance); and (3) covering the skin with Vaseline and add an extra layer on the most exposed area (shielding) (Kang, 2016).

Discovery of radioactivity

Only 2 months following the discovery of X-rays, radioactivity was discovered by Henri Becquerel. Using uranium salt on a photographic plate, Becquerel was able to detect the radioactive rays emitted from the uranium salt. In his investigation, Becquerel exposed potassium uranyl sulfate to sunlight. He expected the uranium salt to emit X-rays from the absorption of the sun’s rays once placed on photographic plates that were wrapped in black paper. The hypothesis was not effectively tested due to cloudy weather in Paris. However, he decided to develop the photographic plates, regardless. Much to his surprise he found clear images proving that uranium emitted radiation without the external energy source of the sun, thereby discovering radioactivity.

Although scientists did not show much interest in Becquerel’s discovery, it marked the beginning of the study of natural radioactivity. The term “radioactivity” was coined by Marie Curie (née Skłodowska). In 1898, following the discovery of radioactivity by Becquerel, Marie Curie together with her husband, Pierre Curie, demonstrated that pitchblende ore showed higher activity than pure uranium, concluding the ore contained other radioactive constituents. This led to the subsequent discovery of two new elements, polonium and radium, both more radioactive than uranium. Although radium was known to destroy tissue, inadequate consideration was given to possible adverse consequences from its unrestricted use. Both Becquerel and Pierre Curie were unknowing victims of the effects of radioactivity, each suffering skin erythema on the abdomen from small samples of radioactive materials carried in their pockets (Kathren, 2019).

The Nobel Prize in Physics 1903 was divided among Antoine Henri Becquerel “in recognition of the extraordinary services he has rendered by his discovery of spontaneous radioactivity,” and the other half jointly to Pierre Curie and Marie Curie “in recognition of the extraordinary services they have rendered by their joint researches on the radiation phenomena discovered by Professor Henri Becquerel,” depicted in Fig. 2 (Nobel Foundation, 2020). The Nobel Prize in Chemistry 1911 was additionally awarded to Marie Curie “in recognition of her services to the advancement of chemistry by the discovery of the elements radium and polonium, by the isolation of radium and the study of the nature and compounds of this remarkable element” (Nobel Foundation, 2020). Marie Curie was the first person to receive the prestigious Nobel Prize twice.

Quantities describing ionizing radiation interactions with matter

The fundamental concepts and quantities used to describe radiation dose are defined as a basis to correlate quantification of radiation physics interactions from ionizing radiation with observed effects and consequences of exposure. These physics interactions lead to the concept of “radiation dose,” defined as the amount of energy imparted per unit mass. In addition to the quantity of absorbed dose, associated fundamental calculable quantities of Kerma and exposure have also been defined by the International Commission on Radiation Units and Measurements (ICRU) (ICRU, 2011).

Dose coefficients¹ can be employed to relate the protection quantities to physical quantities. The International Commission on Radiological Protection (ICRP) defines protection quantities as dosimetric quantities specified in the human body. Protection

¹The use of the terms “dose coefficient” and “dose conversion coefficient” are used here and in the literature interchangeably, representing the same quantities. The term, “dose coefficient” represents updated terminology, as the use of “conversion coefficient” was deemed redundant.

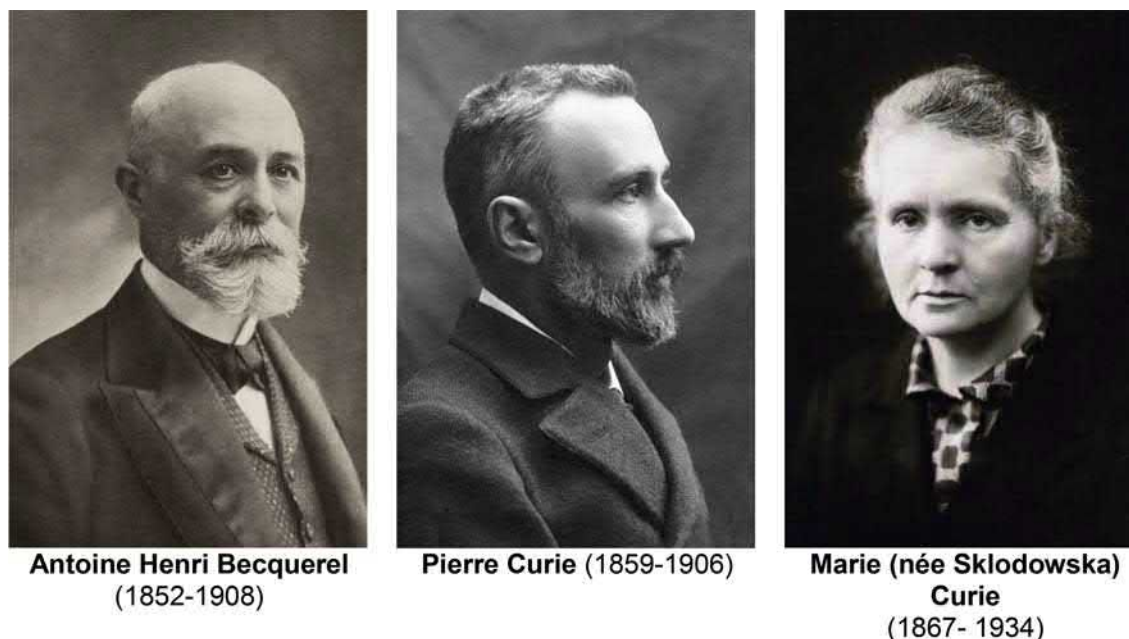


Fig. 2 1903 Nobel Prize in Physics awarded to (Left) Antoine Henri Becquerel, (Middle) Pierre Curie, and (Right) Marie (née Skłodowska) Curie in recognition of the discovery of spontaneous radioactivity.

quantities can be determined by calculation or by a combination of calculations and measurements (ICRP, 1997, 2010). Complementary to the protection quantities prescribed by the ICRP, operational quantities, developed by the ICRU, are intended to demonstrate compliance with protection dose limits by providing calibration quantities for instrumentation measuring or monitoring dose to correlated protective quantities (ICRU, 1998a).

The computation of fluence-to-dose coefficients permits relating a radiation field (operational) quantity, such as fluence (Φ , units cm^{-2}) and its derived quantities (Kerma, exposure), to protection quantities, such effective dose, (E). The relationship between physical, operational, and protection quantities are captured in Fig. 3.

Fluence-to-dose coefficients foundationally relate radiation protection and operational quantities. The recommendations of the ICRP and ICRU, in addition to joint efforts between the two organizations, have led to the computation of dose coefficients aligned with the latest scientific recommendations. The methodologies and tabulations of data of fluence-to-dose coefficients, per the latest international scientific recommendations, are described below and published extensively through a series of reports (ICRU, 1998a, 2011; ICRP, 2007, 2010).

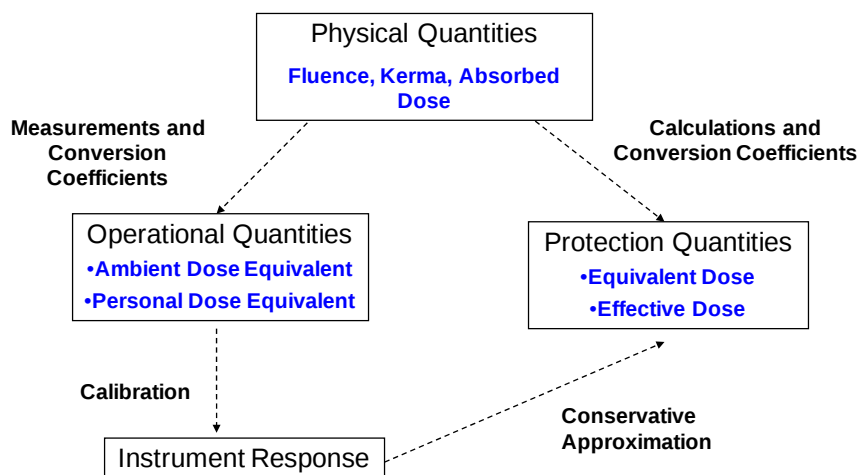


Fig. 3 Relationship between operational and protection quantities. Reproduced with permission from ICRP (1997) *ICRP Publication 74: Conversion Coefficients for Use in Radiological Protection Against External Radiation*. New York, NY: International Commission on Radiological Protection.

Absorbed dose

The absorbed dose, D , is the fundamental unit describing the deposition of energy from ionizing radiation in matter. Absorbed dose can be described in terms of the stochastic quantity, energy imparted (ϵ), as the difference of the radiant energy entering and exiting the volume from uncharged (indirectly ionizing photon and neutron) sources and charged (directly ionizing) particle sources. The absorbed dose is further defined as the quotient of the mean energy imparted by the ionizing radiation into a target volume divided by the mass of the matter in the target volume. The unit of absorbed dose is joules per kilogram (J/kg), which is given the name, gray (Gy) (ICRP, 2007) and sometimes in traditional units of rad ($1 \text{ Gy} = 10^2 \text{ rad}$) or ergs ($1 \text{ Gy} = 10^4 \text{ erg/g}$).

In the system of radiation protection, absorbed dose is the most fundamental quantity and refers to the dose to a specific organ or tissue (T) as (D_T). Modifications to include radiation detriment for different types of source radiation and radiosensitivities are integrated with the fundamental quantity of absorbed dose to evaluate biological effects and establish regulatory exposure limits. It is not possible to write an equation that relates absorbed dose in relation to a deterministic quantity that characterizes the radiation field, as is possible with Kerma, as discussed next. Therefore, fluence-to-dose conversion coefficients can be employed to correlate properties of measurable radiation fields to absorbed dose estimates.

Kerma

Kerma (K) is a deterministic quantity that describes the first step in energy dissipation by indirectly ionizing radiation. The Kerma (kinetic energy release per unit mass) is defined in terms of energy transferred (ϵ_{tr}) to intermediary charged particles in a volume by radiant energies from indirectly ionizing radiation sources (defined only for photon and neutron sources).

Kerma is therefore defined at a point of interest in a volume as the expectation value of the stochastic quantity, energy transferred, to charged particles per unit mass (m) at the point of interest. Kerma can be correlated to the quantities of mass energy transfer coefficient ($\frac{\mu_{tr}}{\rho}$) for photons and Kerma factor (F_n) for neutrons, which are further detailed in (Attix, 2008) and (Veinot, 2019).

The Kerma for X- or γ -rays consists of the energy transferred to electrons and positrons per unit mass of medium. The kinetic energy of fast electrons may be spent in two ways:

- 1) Collision interactions: Coulomb-force interactions with atomic electrons resulting in the local dissipation of the energy as ionization and excitation in or near the electron track.
- 2) Radiative interactions: Coulomb-force interactions with atomic nuclei resulting in the emission of X-ray photons (bremsstrahlung) due to the deceleration of electrons or positron annihilation, that carry energy from the track.

Kerma (K), can be subdivided into two components dependent on whether the energy is dissipated locally through excitations and ionizations, called collisional Kerma (K_c), or whether the energy is carried away by photons, called radiative kerma (K_r):

$$K = K_c + K_r \quad (1)$$

The associated stochastic and deterministic quantities for collisional Kerma and radiative Kerma (K_r being the difference between K and K_c) further described in (Attix, 2008) and (Veinot, 2019).

For monoenergetic photons, K_c is related to the energy fluence (Ψ in MeV cm^{-2}) and mass energy absorption coefficient ($\left(\frac{\mu_{en}}{\rho}\right)_{E,Z}$), which is defined for a given photon energy (E) and the Z of the medium. For neutrons, $K \sim K_c$ because the resulting charged particles from neutron interactions tend to be protons or heavier recoiling nuclei, for which $K_r \sim 0$.

Charged particle equilibrium

Charged particle equilibrium (CPE) is the condition under which each charged particle of a given type, energy, and direction entering a volume is replaced by an identical particle exiting the volume (Attix, 2008; ICRP, 2010). Under CPE conditions at a point in a medium, absorbed dose is equal to collisional Kerma ($K_C =^{CPE} D$), which equates the measurable quantity of absorbed dose with the calculable quantity of collisional Kerma, which is true irrespective of radiative losses.

Transient charged particle equilibrium (TCPE) is said to exist at all points in a region in which $K_C \propto^{TCPE} D$, with a constant of proportionality > 1 . Fig. 4 demonstrates a case where $K_r \approx 0$ (i.e., a case for incident neutrons or for photons up to 3 MeV in a low- Z medium where $K_r < 1\%$ of K) (Maqbool, 2017). The absorbed dose increases with depth near the surface as the number of charged particles flowing into the medium are augmented by increased interactions of indirectly ionizing radiation. The absorbed dose curve reaches a maximum at a depth where the buildup of charged particles is counterbalanced by the descending slope due to attenuation of the indirectly ionizing radiation. Beyond this depth, the absorbed dose curve becomes parallel to the $K = K_C$ curve, where the secondary charged particles starting at the surface penetrate in the direction of the incident rays.

Exposure

Exposure is one of the oldest fundamental quantities in health physics and was introduced before the concepts of Kerma and dose. The units of Röntgen were defined by the ICRU originally in 1928 and are applicable for only X-ray and γ -ray sources. The definition of exposure (X) is the quotient of the absolute value of the total charge (Q) of ions of one sign produced in air when all the electrons (negatron and positron) liberated by photons in air of a given mass (m) are completely stopped in air (ICRU, 2011).

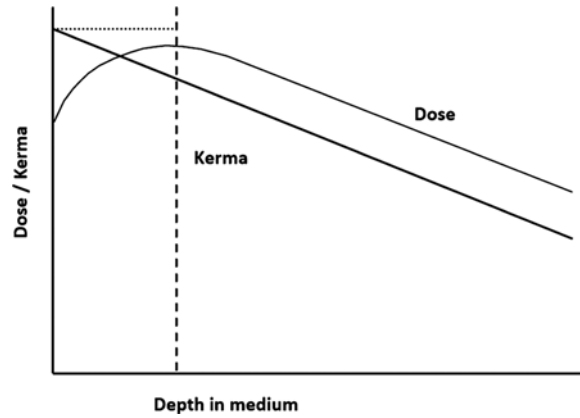


Fig. 4 The relationship between Kerma and absorbed doses for photons and fast neutrons as a function of depth in an absorbing medium under TCPE conditions. Reproduced with permission from Maqbool M (2017) *An Introduction to Medical Physics*. Springer.

Any ionization from bremsstrahlung radiated from the emitted electrons is not included in Q , nor are radiative losses due to positron annihilation. In essence, exposure is the ionization equivalent of K_C for X-ray and γ -ray sources.

To further define exposure, we must introduce the quantity of $\bar{W}$, the mean energy expended in a gas per ion pair formed. This quantity is independent of the incident photon energy. For dry air, $\bar{W}$ is:

$$\bar{W}_{air} = \frac{33.97 \text{ eV/ion pair}}{1.602 \times 10^{-19} \text{ C/electron}} \times \frac{1.602 \times 10^{-19} \text{ J}}{\text{eV}} = 33.97 \text{ J/C} \quad (2)$$

The exposure at a point due to the energy fluence (Ψ) of a monoenergetic photon radiation field of energy, E , can be defined as (Attix, 2008):

$$X = \Psi \left(\frac{\mu_{en}}{\rho} \right)_{E,Z} \left(\frac{e}{\bar{W}_{air}} \right) = (K_C)_{air} \left(\frac{e}{\bar{W}_{air}} \right) = (K_C)_{air} \left(\frac{1}{33.97} \right) \quad (3)$$

where X is given in units in C/kg. To convert from C/kg to units of Röntgen, $1 \text{ R} = 2.580 \times 10^{-4} \text{ C/kg}$.

Therefore, for a given photon energy fluence, the $(K_C)_{muscle}$ per unit exposure is nearly independent of photon energy, with a ratio of $(K_C)_{muscle}/(K_C)_{air} \sim 1.07$ (constant within $\sim 3\%$) for photons between 3 keV and 10 MeV (Fig. 5). This makes it possible to characterize an X-ray field regardless of whether air is located at that point, implying that a photon energy fluence at that point would produce an exposure compared to the energy fluence in Kerma or collisional Kerma for a medium (not necessarily air). This also provides a good approximation to tissue dose using simple measurement of photons.

Furthermore, under CPE conditions, it is possible to equate exposure in air with the absorbed dose:

$$D_{air} \quad (K_C)_{air} = X \left(\frac{\bar{W}_{air}}{e} \right) \quad (4)$$

Protection quantities

Protection quantities are used to establish dose limits for both occupational workers and members of the public from activities involving exposure to radiation. Protection quantities have been derived by ICRP using computational reference anthropomorphic phantoms. Protection quantities have been defined according to recommendations by ICRP, in milestone publications in Publication 26, 60, and 103 (ICRP, 1977b, 1991, 2007). Quantities fundamental to protection are:

- Absorbed dose (described in “Absorbed dose” section);
- Equivalent dose (described in “Equivalent dose and effective dose” section);
- Effective dose (described in “Equivalent dose and effective dose” section)

Equivalent dose and effective dose

Dose limits and administrative controls for delayed stochastic effects are expressed by the quantity, effective dose (E). The latest methodology recommended in ICRP Publication 103 (ICRP, 2007) for the computation of effective dose and its constituent quantities, is captured in Fig. 6.

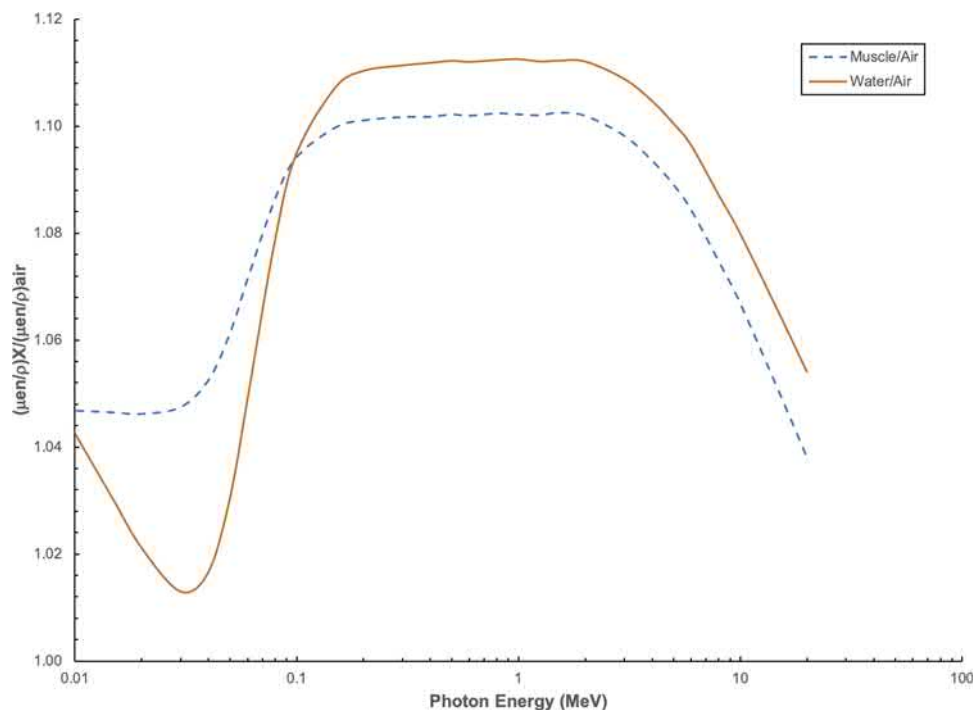


Fig. 5 Ratio of mass energy absorption coefficients for $X = \text{muscle}$ and $X = \text{water}$ relative to air. Generated using data from (Hubbell and Seltzer, 1995).

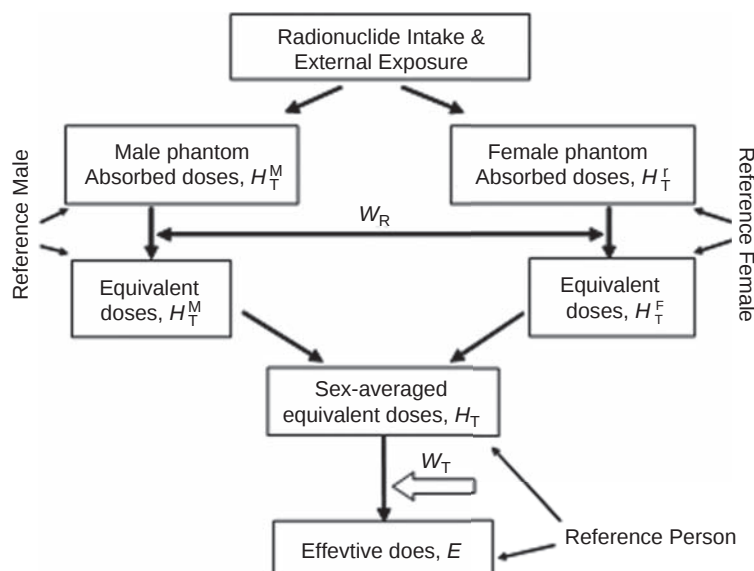


Fig. 6 Methodology for the computation of equivalent and effective dose recommended by ICRP Publication 103. Reproduced from ICRP (2007) ICRP publication 103: Recommendations of the international commission on radiological protection. *Annals of the ICRP* 37: 1–332.

In Fig. 6, the application of a radiation-weighting factor, (w_R) represents the biological effectiveness of different radiations to define the quantity of equivalent dose (H_T) in an organ or tissue (T). The units of the derived quantity, equivalent dose, are fundamentally J/kg with the special name, sievert (Sv). The application of a tissue-weighting factor, (w_T) to the equivalent dose, is the same for all types radiations (e.g., alpha, proton, electron, gamma), but different for each organ based on its associated radiosensitivity risk as evaluated by the ICRP. As tissue weighting factors represent relative radiation detriment based on radiosensitivity of organs and tissues, effective dose is proportional to the radiobiological detriment. The sum of the products of H_T and w_T yield the derived quantity of effective dose, (E), which is used to quantify the radiation dose received by an exposed person in units of J/kg with the special name, sievert (Sv) (ICRP, 2007).

The term of effective dose was introduced in ICRP Publication 60 with recommended set of organs to be used in its computation (ICRP, 1991). In addition, the radiation quality factor was further redefined as the radiation-weighting factor, based on the radiation type and energy incident on the body.

As reinforced in ICRP Publication 103 (ICRP, 2007), the radiation-weighted absorbed dose in the organ or tissue is referred to as the equivalent dose, H_T and is calculated via:

$$H_T = \sum_R w_R D_{T,R} \quad (5)$$

where $D_{T,R}$ is the absorbed dose averaged over the organ or tissue (T) from radiation (R).

The computation of effective dose is the sum of the product of the tissue-weighting factor (w_T) (discussed next in “**Tissue-weighting factor**” section) with the sex-averaged value of the equivalent dose (H_T)

$$E = \sum_T w_T \left[\frac{H_T^M + H_T^F}{2} \right] \quad (6)$$

where H_T^M and H_T^F are the equivalent doses to organ or tissue (T) in the male and female models, respectively.

Radiation quality

Quality factor

The Quality Factor, Q , is a dimensionless variable weighting factor to be applied to the absorbed dose to provide an estimate of the relative human hazard of different types and energies of ionizing radiations. The relative biological sensitivity is defined using a quality factor function $Q(L)$ for a charged particle to be the unrestricted linear energy transfer LET [$L = L_\infty$], which is a smooth function and is equivalent to stopping power in water. As a basis of comparison, photons (X-rays and γ -rays) are used as reference radiation to which other radiations are compared. In ICRP Publication 60 (ICRP, 1991), revised from ICRP Publication 21 (ICRP, 1977a), reflects the $Q(L)$ - L relationship given in Table 1. LET is the integral of stopping power to a cutoff energy that is characteristic of the local distance of ionization of the charged particle.

Radiation weighting (w_R)

The recommendations made by ICRP for biological effectiveness associated with the type of incident radiation have evolved based on the latest available scientific data from recommendations in Publication 60 to current recommendations in ICRP Publication 103. Both Recommendations for w_R from ICRP Publication 60 (ICRP, 1991) and ICRP Publication 103 (ICRP, 2007) are tabulated in Table 2.

The continuous function representing the neutron w_R was developed to serve as an approximation as a function of neutron energy (E) in MeV. ICRP Publication 60 defines this function as (ICRP, 1991):

$$w_R = 5 + 17e^{-\frac{(\ln(2E))^2}{6}} \quad (7)$$

The continuous function representing the neutron w_R was developed to serve as an approximation as a function of neutron energy (E) in MeV. ICRP Publication 103 defines this function as (ICRP, 2007):

$$w_R = \begin{cases} 2.5 + 18.2e^{-\frac{(\ln(E_n))^2}{6}} & E_n < 1 \text{ MeV} \\ 5 + 17.0e^{-\frac{(\ln(2E_n))^2}{6}} & 1 \text{ MeV} \leq E_n \leq 50 \text{ MeV} \\ 2.5 + 3.25e^{-\frac{(\ln(0.04E_n))^2}{6}} & E_n > 50 \text{ MeV} \end{cases} \quad (8)$$

A comparison of the neutron radiation weighting factors are plot in Fig. 7.

Table 1 ICRP Publication 60 Quality Factor $Q(L)$ as function of L from ICRP Publication 21 (ICRP, 1977a) and Publication 60 (ICRP, 1991).

L ($\text{keV } \mu\text{m}^{-1}$)	ICRP Publication 21 $Q(L)$	L ($\text{keV } \mu\text{m}^{-1}$)	ICRP Publication 60 $Q(L)$
≤ 3.5	1	< 10	1
7	2	10–100	$0.32L - 2.2$
23	5	> 100	$300/\sqrt{L}$
53	10		
≥ 175	20		

Table 2 Comparison of radiation weighting factors from ICRP Publication 60 (ICRP, 1991) and ICRP Publication 103 (ICRP, 2007).

Organ/tissue	ICRP Publication 60	ICRP Publication 103
Photons	1	1
Electrons and muons	1	1
Protons and charged pions	5	2
Alpha particles, fission fragments, and heavy ions	20	20
Neutrons ^a	5	Continuous function
< 10 keV		
10 keV–100 keV	10	
> 100 keV–2 MeV	20	
> 2 MeV–20 MeV	10	
> 20 MeV	5	

^aA continuous function serves as an approximation.

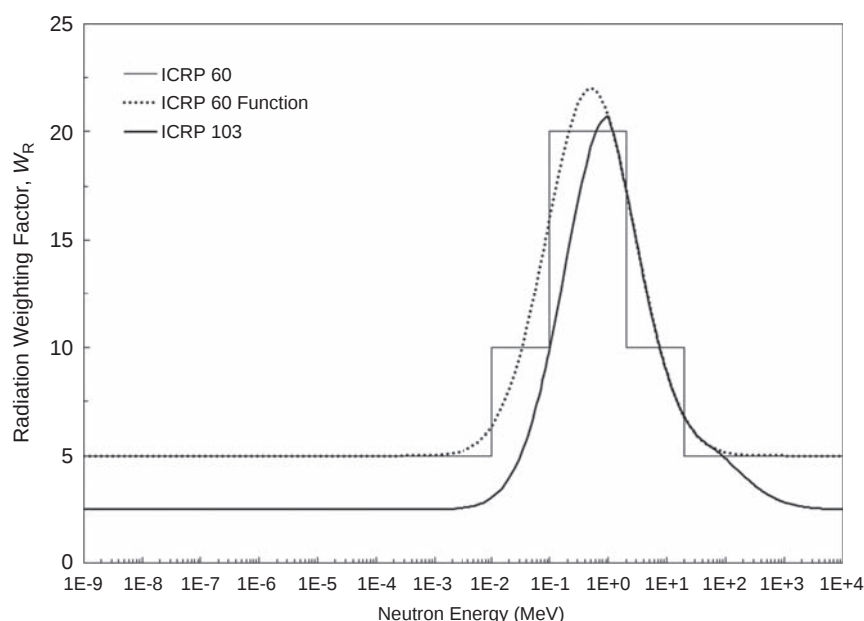


Fig. 7 Neutron radiation weighting factors given in ICRP Publication 60 (ICRP, 1991) and ICRP Publication 103. Reproduced using data from ICRP (2007) ICRP publication 103: Recommendations of the international commission on radiological protection. *Annals of the ICRP* 37: 1–332.

Tissue-weighting factor

The recommendations made by ICRP for radiation risk and detriment associated with constituent tissues and organs of the human body have also evolved with the latest scientific recommendations. The recommendations for tissue-weighting factors (w_T), which are independent of incident radiation type, are provided per the recommendations of ICRP Publication 26, ICRP Publication 60, and ICRP Publication 103 (ICRP, 1977b), ICRP Publication 60 (ICRP, 1991) and ICRP Publication 103 (ICRP, 2007) in Table 3.

Operational quantities

Protection quantities define dose limits to populations of members of the public and occupational workers through the establishment of organ absorbed dose, equivalent dose, and effective quantities. However, these protection quantities cannot be applied to calibration or radiation detection instrumentation. Consequently, operational quantities were developed by the ICRU to correlate measurable quantities with calculated protection quantities. Operational quantities have been developed by ICRU in Report 39 (ICRU, 1985), Report 43 (ICRU, 1988), Report 47 (ICRU, 1992), Report 51 (ICRU, 1993), Report 60 (ICRU, 1998b), and Report 85a (ICRU, 2011). Where protection quantities employ the (w_R) in tissue, operational quantities employ (Q), to differentiate biological effectiveness of radiation.

Operational quantities are used to correlate the protection of effective dose and operational-derived quantities for recommendations of whole body dose, dose to lens of the eye, and dose to skin. The operational quantities applied to area monitoring are ambient dose equivalent and directional dose equivalent; for individual monitoring, the operational quantity of personal dose equivalent is defined.

Table 3 Comparison of tissue-weighting factors per ICRP from ICRP Publication 26 (ICRP, 1977b), ICRP Publication 60 (ICRP, 1991) and ICRP Publication 103 (ICRP, 2007).

Tissue/organ ^a	ICRP publication 26 ^b	ICRP publication 60 ^c	ICRP publication 103 ^d
Gonads	0.25	0.20	0.08
Bone marrow (red)	0.12	0.12	0.12
Colon		0.12	0.12
Lungs	0.12	0.12	0.12
Stomach		0.12	0.12
Bladder		0.05	0.04
Breast	0.15	0.05	0.12
Liver		0.05	0.04
Esophagus		0.05	0.04
Thyroid	0.03	0.05	0.04
Skin		0.01	0.01
Bone surface	0.03	0.01	0.01
Remainder	0.32	0.05 ^c	0.12 ^d
Brain			0.01
Salivary glands			0.01

^aOrgans without tissue-weighting factors defined comprise the remainder.

^bThe five most highly irradiated other organs and tissues.

^cAdrenals, brain, upper large intestine, small intestine, kidneys, muscle, pancreas, spleen, thymus, and uterus (female).

^dAdrenals, extrathoracic tissue, gall bladder, heart, kidneys, lymphatic nodes, muscle, oral mucosa, pancreas, prostate (male), small intestine, spleen, thymus, uterus/cervix (female).

For all types of external radiation, the operational quantities for area monitoring are defined on the basis of a dose-equivalent quantity that would exist in the ICRU sphere (30 cm in diameter) as a theoretical construct of tissue-equivalent material of ICRU four-element tissue with a density of 1 g/cm³, and a composition of 76.2% oxygen, 11.1% carbon, 10.1% hydrogen, and 2.6% nitrogen (ICRP, 2010). The radial depths (*d*), depicted in Fig. 8 (Shannon, 2009), of the sphere correspond to control levels for the whole body dose, lens of the eye doses, and skin dose (Table 4). One key limitation of the ICRU sphere is that it is composed of a material that cannot be manufactured, but can only be simulated.

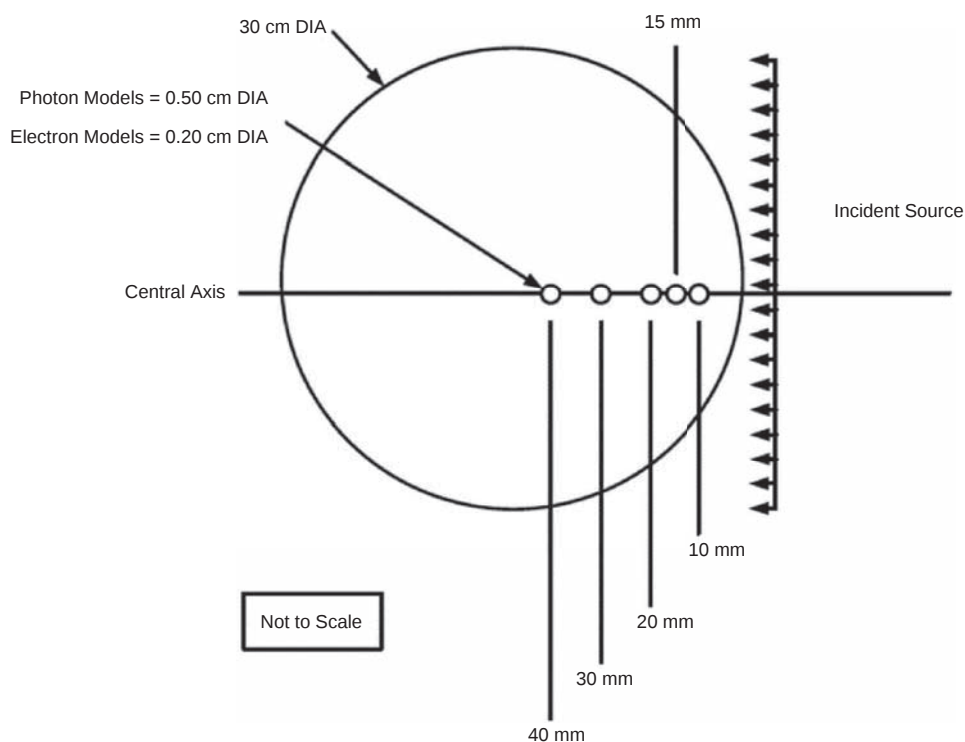


Fig. 8 Schematic of ICRU sphere employed in ambient dose equivalent calculations. Reproduced from Shannon MP (2009) *The Dosimetry of a Highly-Collimated Bremsstrahlung Source in Air*. Doctor of Philosophy. Georgia Institute of Technology.

Table 4 Scheme of operational quantities used for dose monitoring in external exposure situations.

Task	Area monitoring	Personnel monitoring
Control of effective dose	Ambient dose equivalent $H^*(10)$	Personal dose equivalent $H_p(10)$
Control of dose to the lens of the eye	Directional dose equivalent $H'(3, \Omega)$	Personal dose equivalent $H_p(3)$
Control of dose to skin, hands, and feet	Directional dose equivalent $H'(0.07, \Omega)$	Personal dose equivalent $H_p(0.07)$

Reproduced from ICRP (2010) ICRP publication 116: Conversion coefficients for radiological protection quantities for external radiation exposures. *Annals of the ICRP* 40: 1–257.

Area monitoring

Ambient dose equivalent

The ambient dose equivalent, $H^*(d)$, at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded and aligned field in the ICRU sphere at a depth, d , on the radius opposing the direction of the aligned field (ICRU, 2011). As explained in ICRP Publication 74 (ICRP, 1997), an expanded and aligned radiation field is a hypothetical field where the fluence and its energy distribution are the same as an expanded field, but the fluence is unidirectional. Corresponding to effective dose, the ambient dose equivalent is the dose equivalent that would be produced in a corresponding radiation field in the ICRU sphere at a depth of 10 mm. Therefore, the recommended value of d in the ICRU sphere is 10 mm with notation $H^*(10)$ (ICRP, 1997). The units of ambient dose equivalent are (J/kg) as Sv.

Directional dose equivalent

The directional dose equivalent, $H'(d, \Omega)$, is used for assessing the dose to the skin, extremities (hands, wrists, and feet), and lens of the eye. The directional dose equivalent at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded field in the ICRU sphere at a depth, d , on a radius in a specified direction, Ω . For assessing dose to skin and the extremities, a depth of $d = 0.07$ mm is recommended, with notation, $H'(0.07, \Omega)$. For monitoring dose to the lens of the eye, depth $d = 3$ mm, with notation, $H'(3, \Omega)$. The quality factor is employed to assess radiation quality as a function of LET.

Personnel monitoring

Personal dose equivalent

When calibrating the dosimeters of occupational workers, the corresponding operational quantity of interest is the personal dose equivalent, $H_p(d)$, at a depth, d , in units of J/kg as Sv. Similar to the parameters defined in area monitoring in ambient and directional dose equivalent, the same depths are recommended for personal dose equivalent, which are: 10 mm for penetrating radiation, 3 mm for lens of the eye, and 0.07 mm for skin (ICRP, 2010).

In computing personal dose equivalent for dosimeter calibration, the ICRU specifies a 30 cm × 30 cm × 15 cm-thick slab phantom representing the trunk of the body, and constructed of the 4-element ICRU tissue. This measurement is denoted by the corresponding quantity as $H_{p, \text{slab}}(d)$. Since ICRU tissue cannot be fabricated, calibrations frequently use polymethyl methacrylate (PMMA) with thermoluminescent detectors.

Table 4 summarizes the application of the different operational dose quantities with protection values.

Early radiation protection efforts

After noticing severe health problems caused by radioactivity, the original dose limit of approximately 10 rad per day (or 3000 rad per year, where 1 rad = 0.01 Gy), was suggested in 1902. The 10 rad-per-day (0.1 Gy) limits were not based on evaluated biological data but rather on the lowest amount that could be surely detected, particularly, the amount required to create a visible exposure or fogging, on a photographic plate. Animal studies in 1903 had shown that X-rays could produce cancer and kill living tissues. The organs most exposed to radiation injury were the skin, the blood-forming organs, and the reproductive organs (Inkret et al., 1995). Table 5 includes measures of dose rates found by radiation workers in the early part of the 20th century:

By 1905, there was a general acceptance that X-rays could indeed be harmful, and radiation protection was introduced (Kang, 2016). The earlier typical opinion had changed, and the general belief now among physicians, scientists, and those working with X-rays, as well as the general public, was that X-rays were indeed the cause of the harmful effects. Precautions were needed to ensure their safe use in medicine. The change was mostly attributable to a better knowledge of the scientific aspects, particularly physics of X-rays and radioactivity. It was also due to the growing body of publications in the medical and scientific literature describing contrary effects of exposure to X-rays, including deaths among some of the early workers. There had also been higher sophistication and less tolerance in public attitudes regarding potential hazards. The first of these X-ray victims was Clarence Dally, Thomas Edison's assistant, who had suffered significant exposures in 1896 and who subsequently died in 1904.

Table 5 Dose rates for radiation workers in the early part of the 20th Century.

Occupation	Approximate dose rate (rad min^{-1}) ^a
Fluoroscopist	0.6–6 (hands)
X-ray therapy technician	0.006–0.06 (body)
Radium therapist or technician	0.006–0.06 (body)

^a1 rad is defined as 0.01 Gy.

Adopted from Inkret WC, Meinhold CB and Taschner JC (1995) Protection standards. *Los Alamos Science* 23: 117–123.

Along with recognizing the problem came several suggested resources made by some well-known protection pioneers. Although they did not have sufficient knowledge of the nature of X-rays, they helped to minimize the adverse effects. Nevertheless, more information concerning the physical nature of the X-rays was required including a practical, globally accepted means of quantification, specified limitations on exposure, advancements in devices, procedure, and processes. Most of the progress came from the medical applications, for this was where most problems were seen (Kathren, 2019).

William Herbert Rollins had made recommendations for examining X-ray areas by using film in 1902. He suggested that if a photographic plate was not fogged in 7 min, then the radiation field was not dangerous. Rollins, a dentist from the Boston area, should be considered a leading pioneer in the radiation protection field. The basic principles for developing X-ray protection techniques existed early in the gas tube period. Meanwhile, more and more evidence of X-rays' ability to produce cancer continued to be collected. However, between 1900 and 1913, little effort was made toward developing radiation protection systems (McGinley and Miner, 1995).

Early cohort studies

By 1910, many people who used X-ray generating and radioactive material began to develop skin inflammations, which eventually led to early skin cancer. Therefore, scientists presumed that exposure to radiation could cause infertility, bone disease, and cancer. During the early years, there were no regulated systems for the measurement of radiation exposure or the estimation of radiation dose. However, some work had been done to standardize and formalizing the quantities and units related to radioactivity. In 1925, an investigation carried by Harrison S. Martland, a New Jersey Medical inspector, suggesting that evidence related to radium intake by female factory workers could lead to serious illness and fatality (Jones, 2005).

In 1920 radioluminescent paints, made by combining radium with fluorescent materials, were popular in the production of watch and clock dials, gun sights, and other applications. Many firms, including the Radium Luminous Material Corporation (reorganized as a successor company, United States Radium Corporation) purchased the paint, which was applied, almost exclusively by women referred to as the "Radium Girls," to the clock dials with tiny brushes. One company reported that about 4300 dials were painted every day (Turner, 2008). The workers ingested a large amount of radium, a bone-seeking element, by repeatedly sharpening the tip of the brush with their lips. By the 1920s, numerous cases of radiation-related diseases and fatalities were reported in the radium related industries. More than 1000 individuals had the amount of radium in their bodies measured. Bone samples were gathered after death and examined. An official guide of 0.1 μg for the maximum allowable amount of ^{226}Ra in the body was later established. Legal action was against the U.S. Radium Corporation by employees experiencing radiation poisoning (Fig. 9).

It was estimated that this level corresponds to an average dose rate of 0.6 mGy week^{-1} and a dose equivalent rate of somewhere between ~ 1 and $\sim 6 \text{ mSv week}^{-1}$ (Turner, 2008). The large majority of the "Radium Girls" received no compensation from the company. The U.S. Radium Corporation was reorganized into new corporations and ultimately dissolved following 1980. The former radium processing facilities, which included the dial painting studio of Orange, New Jersey, became a U.S. Environmental Protection Agency Superfund cleanup site (Dabbs, 2018).

Uranium miners' experience is another significant source of information on the negative consequences of radiation. These data are particularly relevant to exposure to the daughters of radon everywhere. From the early days of the Medieval period, it was noticed that miners in some regions of Europe had abnormally high numbers of lung diseases, also known as mountain sickness. During the 20th century, miners worked in an environment with high concentrations of dust, containing elements such as arsenic, uranium, and other alloys. The rate of lung cancer was raised among the workers, and consequently, half of the workers died of this disease. It took scientists a while to realize that the radon and its daughters were the leading cause of this disease.

The reports of many miners throughout the world have been studied and investigated by the Biological Effects of Ionizing Radiation (BEIR) VI report, *Health Effects of Exposure to Radon* (National Research Council, 1999), updating the 1988 BEIR IV report (National Research Council, 1988). Further, models have been generated to estimate the exposure of the radon daughters to lung tissues per working level month (WLM). The value of absorbed doses is somewhere between 0.2 and 3.0 mGy. The result of a group of 11 studies that involved 60,606 highly exposed workers from all over the world, shows that the average exposure was 164.4 WLM, and the average time, 5.7 years. There were about 2000 lung-cancer deaths reported among the exposed people. To determine the risks at low doses; the data were compared with other studies that had smaller mean exposures (Turner, 2008).

Another study that was performed on underground miners in New Mexico showed the same results, such as lung cancer and disease due to radon and its decay products. This study also proved that uranium dust has little to no effect on an individual's health, which is

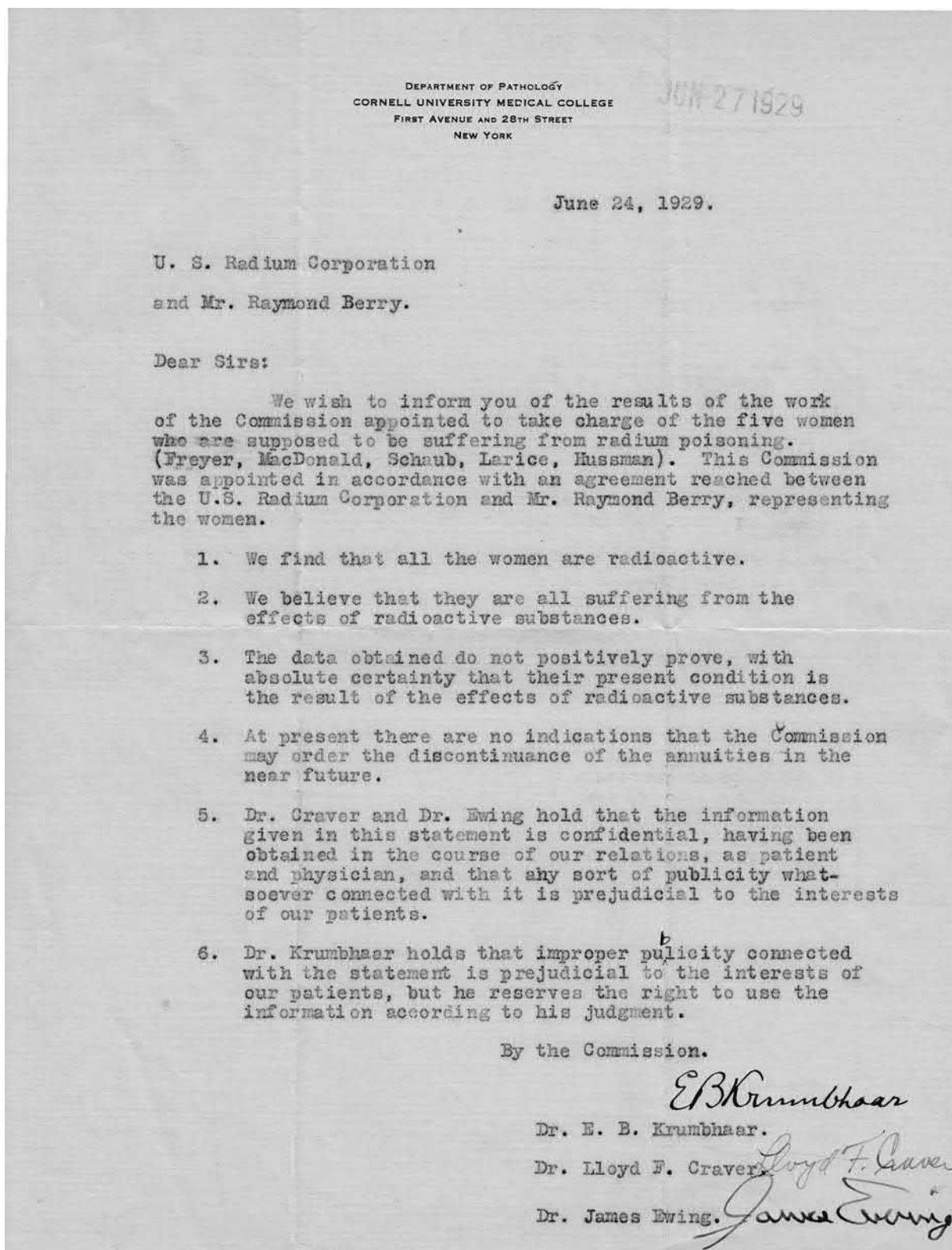


Fig. 9 Letter of complaint to U.S. Radium Corporation documenting radiation poisoning of five workers (National Archives, 1929).

because uranium ore is not very radioactive and gamma radiation from uranium dust is low. Underground miners were exposed to radon, causing serious health problems for them. However, uranium millers were exposed to uranium dust, and no sign of cancer or other diseases were observed. The result of this study proved that unlike radon, uranium dust does not pose a major health hazard (Boice et al., 2008).

Table 6 Major milestones in the discovery of radiation.

Year	Event
1895	Roentgen discovers ionizing radiation
1896	Elihu Thomson experiment, reveals the harmful effect of X-ray
1896	Henri Becquerel discovers alpha and beta particles
1898	Curie discovers and isolates polonium and radium which later caused cancer
1900	American Roentgen Ray Society (ARRS) founded
1902	Ernest Rutherford discovers isotopes
1902	Dose limit of approximately 10 rad per day (or 3000 rad per year), was suggested
1902	Rollins had made recommendations for examining x-ray areas by using film
1903	Animal studies in had shown that X-rays could produce cancer and kill living tissues
1905	Link that radiation can cause cancer
1915	British Roentgen Society adopts X-ray protection resolution; believed to be the first organized step toward radiation protection
1932	James Chadwick discovered the neutron
1932	Cockcroft and Walton produced nuclear transformations by bombarding atoms with accelerated protons
1935	Enrico Fermi found that a much greater variety of artificial radionuclides could be formed when neutrons were used instead of protons
1939	Hahn and Strassman discover Nuclear fission, induced by capture of a slow neutron in ^{235}U
1939	Bohr soon proposed that fission was much more likely to occur in the uranium-235 isotope than in U-238 and that fission would occur more effectively with slow-moving neutrons than with fast neutrons
1939	Szilard and Fermi propose using a “moderator” to slow down the emitted neutrons
1939	WWII begins
1942	An experimental graphite pile constructed by Fermi operated at the University of Chicago established the first man-made fission chain reaction
1945	The first atomic device tested successfully at Alamogordo, New Mexico; detonation of atomic bombs in Hiroshima and Nagasaki, Japan, marking the end of WWII
1953	President Eisenhower proposed his “Atoms for Peace” program, which set the course for civil nuclear energy development in the United States
1979	Three Mile Island reactor accident
1986	Disaster at the Chernobyl nuclear power plant
2011	Earthquake and tsunami prompting Fukushima Daiichi Nuclear Power Plant incident

Milestones in the discovery of radiation and its effects

Table 6 lists a timeline of key milestones in the discovery of radiation, the early evolution of radiation protection practices, and a few notable events.

The future of radiation protection

Management and limitation of radiation exposure has led to the growth of field of health physics, where a broad spectrum of issues require attention and must eventually be addressed to ensure that radiation protection programs in the energy, industrial, medicine, defense, and security sectors continue to meet these emerging needs. Efforts must be focused conducting focused research in critical scientific/mission areas and advancing training and developing of skilled radiation personnel. Recently, a series of future priorities has been established in health physics and radiation protection in the fields of nuclear energy, medicine, dosimetry/risk, space travel, instrumentation/operations, environment, decontamination/decommissioning, emergency response, defense, and effects from exposures to low dose radiation (Dewji et al., 2017; Davis et al., 2019). As these fields continue to study the utility of radiation sciences and technology, radiation protection will continue to evolve in tandem to limit and control radiation exposure.

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See Also: Discovery of Radioactivity; Radioactive Decay.

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Fundamentals of Health Physics

Shaheen Azim Dewji^a and Laurence Miller^b, ^a Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States; and ^b Department of Nuclear Engineering, The University of Tennessee, Knoxville, TN, United States

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Interactions of radiation with matter—including tissue, shielding materials, and radiation detectors—all stem from fundamental physical interactions at the atomic and sub-atomic level of interest in health physics and radiation protection.

Nuclear physics

The structure of the atom

Prior to the postulate of the “Standard Model of Fundamental Particles and Interactions” in 1968, neutrons and protons were generally considered to be fundamental particles. However, results from studies of high energy particles made it clear that the structure of the nucleus was composed of more components than only neutrons and protons. As a result, the “Standard Model” was developed.

In the “Standard Model”, there are 12 fundamental matter particle types (quarks and leptons) and their corresponding antiparticles. Quarks have charges of either $+2/3$ or $-1/3$, and they are held together with force-carrying particles in combinations of three to form particles with charges of $+1$, -1 , and neutral. The force-carrying particles include the following: photon, gluon, Z boson, and W boson. It is interesting to note that the top quark was not discovered until 1995 by Fermilab ([Abe et al., 1995](#)) and that a Higgs boson was not discovered until 2012 ([Aad et al., 2012](#); [Chatrchyan et al., 2012](#)). The Higgs boson is sometimes called the “God particle” since the theory associated with the Standard Model implies that it gives mass to all other particles and that a universe-wide Higgs field exists.

There are three general families of matter, as shown in [Fig. 1](#) ([Ling et al., 2019](#)), but most health physics, radiation protection, and nuclear engineering disciplines are concerned with the first family that include up quark, down quark, electron neutrino, and the electron, which associate to form photons, electrons, positrons, protons, and neutrons. Neutrinos and anti-neutrinos are generally included in discussions of beta decay (see section [Beta decay](#)).

Proton

Protons presumably exist in the nucleus of an atom and govern the identity and chemical properties of the atom. The atomic number, Z , is the number of protons in the atom, and define the elemental properties of the atom. Protons are comprised of two up quarks (each with unit charge $+2/3$) and one down quark (unit charge $-1/3$) resulting in a total net charge of $+1$ for the proton, or 1.602×10^{-19} Coulombs (C) per unit charge. The rest mass of the proton is 1.673×10^{-27} kg (Baum et al., 2002) with the mass-energy equivalent rest mass of 938.272 MeV/c².

Neutron

Neutrons are also presumed to exist in the nucleus of an atom and are comprised of one up quark (+2/3 charge) and two down quarks (-1/3 charge each) resulting in a net neutral charge. The rest mass of the neutron is slightly larger than a proton at 1.675×10^{-27} kg (Baum et al., 2002) and a rest mass of 939.566 MeV/c².

Electrons

Electrons are fundamental particles, known as leptons, that exist outside of the nucleus of the atom and govern the chemical interactions of the atom, as well as some nuclear interactions. They have a net charge of -1 and mass of 9.11×10^{-31} kg and rest mass of 0.511 MeV/ c^2 . Neutral atoms have an equal number of protons (Z) and electrons. Electrons occupy the region outside of the nucleus in shells. The discrete energy states occupied by electrons are dictated by quantum mechanical probabilities, in which electrons can excite and de-excite between energy states.

	I	II	III	
Mass	2.4 MeV	1.27 GeV	171.2 GeV	0
Charge	$\frac{2}{3}$	$\frac{2}{3}$	$\frac{2}{3}$	0
Spin	$\frac{1}{2}$	$\frac{1}{2}$	$\frac{1}{2}$	1
Name	u up	c charm	t top	γ photon
Quarks	4.8 MeV $\frac{1}{3}$ $\frac{1}{2}$ d down	104 MeV $\frac{1}{3}$ $\frac{1}{2}$ s strange	4.2 GeV $\frac{1}{3}$ $\frac{1}{2}$ b bottom	0 0 1 g gluon
Leptons	<2.2 eV 0 $\frac{1}{2}$ ν_e electron neutrino	<0.17 MeV 0 $\frac{1}{2}$ ν_μ muon neutrino	<15.5 MeV 0 $\frac{1}{2}$ ν_τ tau neutrino	91.2 GeV 0 1 Z ⁰ weak force
	0.511 MeV -1 $\frac{1}{2}$ e electron	105.7 MeV -1 $\frac{1}{2}$ μ muon	1.777 GeV -1 $\frac{1}{2}$ τ tau	80.4 GeV ± 1 1 W [±] weak force
				Bosons (forces)

Fig. 1 Families of Matter: Leptons, quarks, and force carriers. Radiation protection and related disciplines focus on: up/down quarks that constitute protons and neutrons; electron neutrinos/anti-neutrinos and electrons/positrons leptons; and photon force carriers. Reproduced from Ling SJ, Sanny J and Moebs W (2019) *University Physics Volume 3*. OpenStax CNX. Available: <http://cnx.org/contents/af275420-6050-4707-995c-57b9cc13c358@12.2> (Accessed 2020).

Nucleus

Together the sum of the atomic number and the number of neutrons (N) is mass number ($Z + N = A$), which represents the integer number of nucleons in the nucleus of an atom. A radionuclide is symbolically represented by A_ZX , where X represents the element. For example, ${}^{12}_6\text{C}$ represents the carbon nucleus with six protons and six neutrons (totaling 12 nucleons). ${}^{17}_8\text{O}$ represents the oxygen nucleus with eight protons and nine neutrons (totaling 17 nucleons).

Isotopes, isotones, and isobars

Elements with different number of neutrons (and hence nucleons) are called isotopes. Certain isotopes are unstable and may undergo nuclear transformations by emitting energetic particles, converting neutrons into protons, converting protons into neutrons, or de-excitation through the emission of photons to reach a more stable energetic state. These will be discussed in further detail in the section, **Radioactive Decay**.

Elements with the same number of neutrons (though not the same number of protons) are referred to as isotones, and those with the same number of nucleons are referred to as isobars.

Atomic mass unit

The atomic mass is the mass of the atom inclusive of the bound protons, neutrons and electrons. If one mole of any material contains 6.022×10^{23} atoms (Avogadro's number), and the weight of one mole is defined by the atomic mass unit, then the weight of a single atom can be calculated. For example, 1 mol of ${}^{12}_6\text{C}$ has a unique mass of 12.0000 g, which results in the mass of a single atom of ${}^{12}_6\text{C}$ as:

$$\frac{12.0000 \text{ g} \cdot \text{mol}^{-1}}{6.022 \times 10^{23} \text{ atoms} \cdot \text{mol}^{-1}} = 1.993 \times 10^{-23} \text{ g} \quad (1)$$

Since the atomic mass of ${}^{12}_6\text{C}$ is a whole, round number, carbon was selected as the reference value for the system of relative masses. The mass of an atom can be given in grams or in a relative quantity called the atomic mass unit (amu), where,

$$1 \text{ amu} = \frac{1.993 \times 10^{-23} \text{ g}}{12} = 1.661 \times 10^{-24} \text{ g} = 1.661 \times 10^{-27} \text{ kg} \quad (2)$$

In mass-energy equivalent units using Albert Einstein's $E = mc^2$ relation, where c = speed of light in a vacuum ($3 \times 10^8 \text{ m/s}$)

$$E(1 \text{ amu}) = \frac{(1.661 \times 10^{-27} \text{ kg} \cdot \text{amu}^{-1})(3 \times 10^8 \text{ m} \cdot \text{s}^{-1})^2}{1.6 \times 10^{-13} \text{ J} \cdot \text{MeV}^{-1}} = 931.5 \text{ MeV} \quad (3)$$

We can define a "mass excess," Δ , as the difference between the atomic mass and mass number ($M-A$) in amu or MeV equivalent. Masses of radionuclides can be defined relative to ${}^{12}_6\text{C}$, in which Δ for ${}^{12}_6\text{C}$ is 0. A compendium of mass excess value for a series of radionuclides are given in [Turner \(2008\)](#) and [Wang et al. \(2017\)](#).

Binding energy

If it was possible to weigh an individual atom, it would be found that the atom weighs less than its constituent parts, inclusive of all its protons, neutrons, and electrons. For example, if the sum of the constituent protons and neutrons of an atom is computed, this yields ([Cember et al., 2008](#)):

$$W = (Z)m_p + (A - Z)m_n \quad (4)$$

where W = mass of sum of constituent nucleons; m_p = proton mass = 1.00728 amu; m_n = neutron mass = 1.00867 amu; Z = atomic number; A = mass number.

For the example of ${}^{17}_8\text{O}$, with $Z = 8$ and $A = 17$:

$$W = (8)(1.00728 \text{ amu}) + (17 - 8)(1.00867 \text{ amu}) = 17.1363 \text{ amu} \quad (5)$$

However, the mass of ${}^{17}_8\text{O}$ atom is $A = 16.9991$ amu. The missing mass is equivalent to the energy binding the nucleus together, which is equal to $(17.1363 - 16.9991) = 0.13717$ amu. This behavior is true not just for the example of ${}^{17}_8\text{O}$, but also for all radionuclides. This difference between the atomic mass of an element A_ZX and W is referred to as the mass defect, δ (in amu), where

$$\delta = W - M_X(A, Z) \quad (6)$$

Therefore, the mass defect is the mass equivalent of the energy that must be expended to separate the nucleus into its individual constituent nucleons, called the binding energy (BE), such that they are separated by a distance large enough so they exert no forces on each other. Conversely, the BE can also be interpreted as amount of energy released if free nucleons were to coalesce into a nucleus ([Fig. 2](#)).

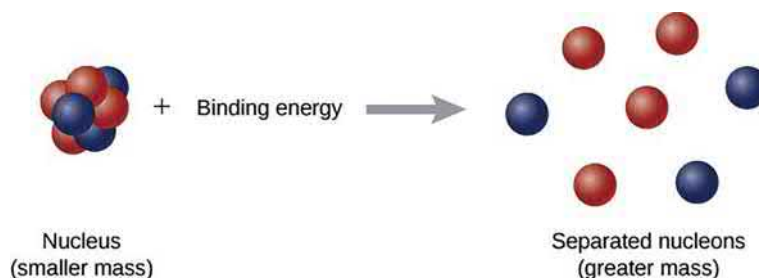


Fig. 2 The binding energy (BE) is the energy required separate all nucleons into its constituent protons and neutrons. The sum of the constituent nucleons is larger than the bound nucleons. Reproduced from Ling SJ, Sanny J and Moebs W (2019) *University Physics Volume 3*. OpenStax CNX. Available: <http://cnx.org/contents/af275420-6050-4707-995c-57b9cc13c358@12.2> (Accessed 2020).

The BE is therefore defined as the energy required to remove a subatomic particle from an atom:

$$BE = \delta \text{ (amu)} \times \frac{931.5 \text{ MeV}}{\text{amu}} \quad (7)$$

An important quantity in experimental nuclear physics is the binding energy per nucleon (BEN), which is defined as:

$$BEN = \frac{BE}{A} \quad (8)$$

BEN values are measured from nuclear scattering experiments. A graph of the BEN as a function of atomic number A is shown in **Fig. 3**. In this graph, the BEN typically ranges from ~ 7 – 9 MeV, with a broad maximum ~ 8.5 MeV/nucleon. It takes millions of electron volts to separate a nucleon from the nucleus, demonstrating that the nuclear force binding the nucleons is indeed strong, giving rise to the specific binding force in the nucleus called the “strong nuclear force.” The highest BEN falls around ${}^{62}_{28}\text{Ni}$ at 8.795 MeV/nucleon, followed by ${}^{58}_{26}\text{Fe}$ and ${}^{56}_{26}\text{Fe}$ at 8.792 MeV/nucleon and 8.790 MeV/nucleon (Veinot, 2019).

An interesting implication of this curve is that elements to the left of the most stable region must undergo fusion of light elements to undergo an exothermic reaction (i.e., release energy), whereas elements to the right must undergo fission of heavy elements to undergo an exothermic reaction. These two interactions will be discussed later in (see section **Neutrons**).

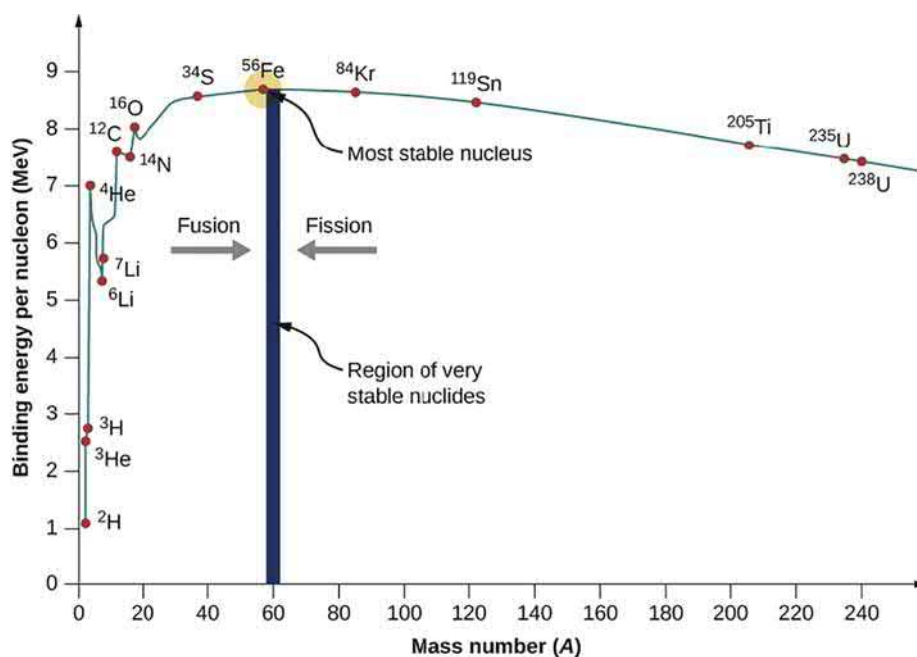


Fig. 3 Binding energy per nucleon (BEN) plot as a function of mass number. The greatest BEN falls around ${}^{56}_{26}\text{Fe}$, making radionuclides in this region the most stable nuclei. Reproduced from Ling SJ, Sanny J and Moebs W (2019) *University Physics Volume 3*. OpenStax CNX. Available: <http://cnx.org/contents/af275420-6050-4707-995c-57b9cc13c358@12.2> (Accessed 2020).

Electromagnetic spectrum

Photons and electrons (as well as any elementary particle) may manifest as particles or waves, depending on how they are observed. For example, if electrons are directed towards a target with a double slit, a wave pattern will emerge. On the other hand, if the electrons are observed with a counting system, they behave as particles, as demonstrated by Clinton Davisson and Lester Germer in their experiments between 1923 and 1927, where Davisson received the Nobel Prize in Physics in 1937 for “experimental discovery of the diffraction of electrons by crystals” (Davisson and Germer, 1927). Low energy photons, at radio frequency, manifest as waves while high energy photons manifest as particles. In order for neutrons to exhibit wave-like behavior they must be cooled to a few degrees Kelvin. Fig. 4 illustrates the utility and the hazards of electromagnetic radiation in society, as well as the relationship between the energy, frequency, and wavelength of electromagnetic radiation according to the equation derived by Max Planck (Planck, 1901) and harnessed by Albert Einstein to explain the photoelectric effect (Einstein, 1905) (see section Photoelectric Effect).

Radiowaves at the left-most side of portion of the spectrum in Fig. 4 have the lowest frequency and the longest wavelength.

- Radio wavelengths vary from ~ 3 m to 100 m.
- Radio waves are also emitted by other sources, such as stars and gases in space.

Microwave radiation falls above the radio spectrum in terms of energy.

- Wavelength is between 1 mm and 30 cm.
- Can be used to study the universe, communicate with satellites in Earth orbit, and cook popcorn.

Infrared (IR) radiation has wavelengths between 700 nm and 1 mm.

- The name means “below red”—red being the color of visible light of longest wavelength.
- Common use of IR is in night vision equipment and television remote controls since it does not penetrate walls and interfere with other devices.
- The heat that we feel from the Sun or from a fireplace is infrared radiation.

Visible Light is electromagnetic radiation at wavelengths which the human eye can see.

- We perceive this radiation as colors ranging from red (longer wavelengths; ~ 700 nm) to violet (shorter wavelengths; ~ 400 nm.)

Ultraviolet radiation is shorter than the violet end of visible light; the atmosphere of the Earth effectively blocks the transmission of most ultraviolet light.

- It is the UV rays that cause our skin to burn! Stars and other “hot” objects in space also emit UV radiation.
- At a wavelength of up to 320 nm, which includes most of the ultraviolet band, these energies can be classified as ionizing radiation.

X-Ray and Gamma ray radiation have the shortest wavelength and therefore the highest frequency at the right-most side of Fig. 4.

- They are also the most energetic part of the spectrum.
- These are classified as ionizing radiation.

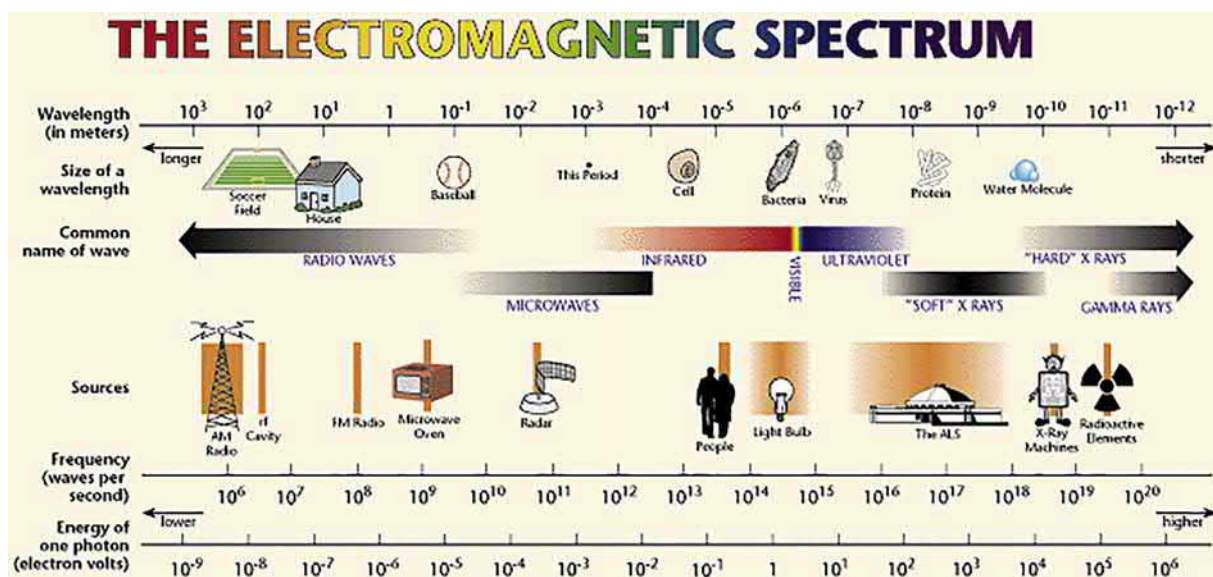


Fig. 4 The electromagnetic spectrum as a function of frequency. As the frequency increases, the energy of the wave increases and as the frequency decreases, the energy of the wave decreases. Reproduced from Lawrence Berkeley Laboratory.

Radioactive decay

Radioactivity denotes the property of spontaneous transformation of state in a nucleus that results in the release of energy in the form of electromagnetic waves or particle emissions. These properties of radionuclides are determined by nuclear considerations only, independent of chemical and physical states of the radionuclide, making these properties unique and characteristic of the respective radionuclide. The specific mode of radioactive transformation is dependent on the available energy for the transition, which further depends on (1) the type of nuclear instability (ratio of protons and neutrons); and (2) mass-energy relationships among the radioactive parent, progeny, and emitted particles.

Fig. 5 shows a “band of stability” that represents a region in which stable species exist. Above the band exists an excess of neutrons, and below the band exists an excess of protons. In regions of excess of either protons or neutrons, nuclei will seek to transform these excess particles to drive towards more stable states aligned with the “band of stability.” For radionuclides with an excess of neutrons, the nucleus will undergo transformations towards more stable state by transforming a neutron into a proton and typically decays by emitting an electron. For radionuclides with an excess of protons, compared to neutrons, the nucleus will undergo transformations towards a more stable state by transforming excess protons into neutrons, and typically decays by emitting a positron, the anti-particle of the electron.

Alpha decay

Alpha decay is a process of radioactive decay in which a heavy parent radionuclide with atomic number Z and mass A spontaneously emits an alpha (α) particle (i.e., a He nucleus with $Z = 2$, $A = 4$, and charge $+2$) producing a daughter radionuclide with atomic number $Z-2$ and mass number $A-4$.

$${}^A_ZP \rightarrow {}^{A-4}_{Z-2}D + {}^4_2\alpha + Q \quad (9)$$

Almost all naturally occurring alpha emitters come from heavy elements with $Z > 83$. The range of an alpha particle is very short, $\sim 2\text{--}10$ cm in air. Alpha particles have low penetrating power and can be easily shielded by paper or by the outer layer of the skin. However, if radionuclides undergoing alpha decay are internalized such as through inhalation, ingestion, or skin absorption, then this potentially poses a significant internal health hazard. For example, radon is a noble gas that can diffuse into environmental systems and homes after being formed by the alpha decay of ${}^{226}_{88}\text{Ra}$:

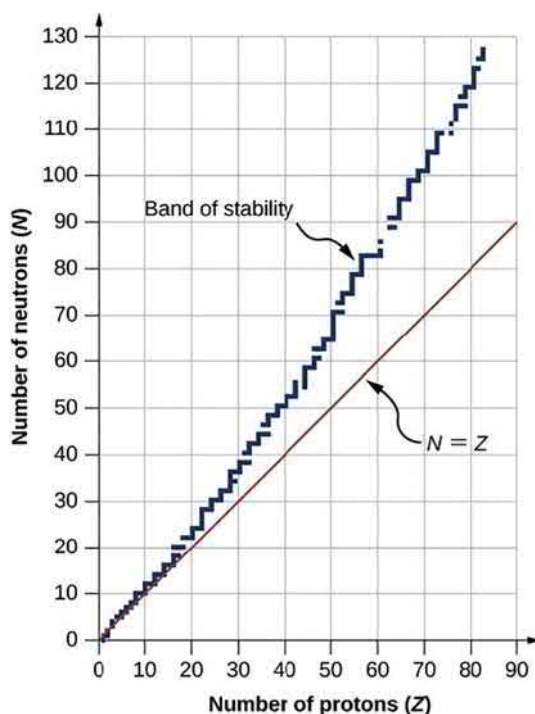


Fig. 5 Plot of number of number of neutrons vs. number of protons for stable atomic nuclei. Stable radionuclides exist along the “band of stability”. Radionuclides above the “band of stability” have an excess of neutrons, whereas those below have an excess of protons. Reproduced from Ling SJ, Sanny J and Moebis W (2019) *University Physics Volume 3*. OpenStax CNX. Available: <http://cnx.org/contents/af275420-6050-4707-995c-57b9cc13c358@12.2> (Accessed 2020).

Radon decays to polonium isotopes that attach to dust particles and subsequently reside in lungs for days, weeks or even years, depending of the solubility class of dust particles. Polonium isotopes decay by alpha emission and are an internal hazard if the polonium particles are inhaled and remain in the lungs. Radon is a noble gas and does not pose a direct health concern compared to its progeny.

In Eq. (10), Q is equal to the total amount of kinetic energy released in radioactive decay, equal to the difference in the total mass of the system. If Q is >0 , the reaction is exothermic, and if $Q < 0$, the reaction is endothermic.

$$Q = M_P - (M_D + M_\alpha) = \Delta_P - (\Delta_D + \Delta_\alpha) \quad (11)$$

where M_P = nuclear mass of parent nuclide; M_D = nuclear mass of daughter nuclide; and M_α = nuclear mass of alpha particle; Δ_P = mass excess of parent nuclide; Δ_D = mass excess of daughter nuclide; and Δ_α = mass excess of alpha particle. The nuclear mass difference is nearly equal to the atomic mass difference, with the exception of the slight difference in the binding energies of the 88 electrons on either side of the arrow, which are neglected when atomic mass loss is equal to nuclear mass loss. In essence, the nuclear masses are needed but atomic masses are much better known, resulting in negligible differences.

To calculate the Q -value of the alpha decay of $^{226}_{88}\text{Ra}$ from Eq. (10) by applying Eq. (11):

$$\begin{aligned} Q &= M_{^{226}_{88}\text{Ra}} - (M_{^{222}_{86}\text{Rn}} + M_{^4_2\alpha}) \\ Q &= 226.025402 - (222.017570 + 4.002603) \text{ amu} \\ &\text{as mass excess : } 23.69 \text{ MeV} - (16.39 \text{ MeV} + 2.4248 \text{ MeV}) \\ Q &= 0.005229 \text{ amu} = 4.87 \text{ MeV} \end{aligned} \quad (12)$$

Alpha decay is an exothermic process, yielding Q -values in the range of 4–9 MeV. The energy released (Q -value) is distributed as kinetic energy between the emitted two particles following decay. If conservation of energy and momentum is applied, the following kinetic energy distribution between the daughter (E_D) and the alpha particle (E_α) can be calculated:

$$E_\alpha = \frac{M_D Q}{M_\alpha + M_D} \quad (13)$$

$$E_D = \frac{M_\alpha Q}{M_\alpha + M_D} \quad (14)$$

In the decay of $^{226}_{88}\text{Ra}$, the kinetic energy of the alpha particle is 4.78 MeV, where the recoil energy of the daughter, $^{222}_{86}\text{Rn}$, is 0.087 MeV. The decay scheme of $^{226}_{88}\text{Ra}$ is given in Fig. 6. Note that 94.4% of $^{226}_{88}\text{Ra}$ transformations decay directly to $^{222}_{86}\text{Rn}$ with an associated alpha particle energy of ~ 4.78 MeV, and 5.5% to an excited metastable state $^{226}_{88}\text{Ra}$ with an alpha energy of 4.60 MeV, accompanied by the emission of a characteristic gamma-ray of 0.186 MeV (gamma emissions will be discussed further in section Gamma emission).

Beta decay

Beta (β) decay is type of radioactive decay in which the nucleus is transformed without changing the number of nucleons. These transitions, called isobaric transitions, include the processes of β^- , β^+ , and electron capture (EC).

β^- Decay

β^- decay occurs in low- Z nuclei that are neutron rich. These neutron rich nuclei gain stability by decreasing the N/Z ratio by converting a neutron to a proton. In β^- decay, an antineutrino and a negative electron are emitted from the nucleus as a result of the

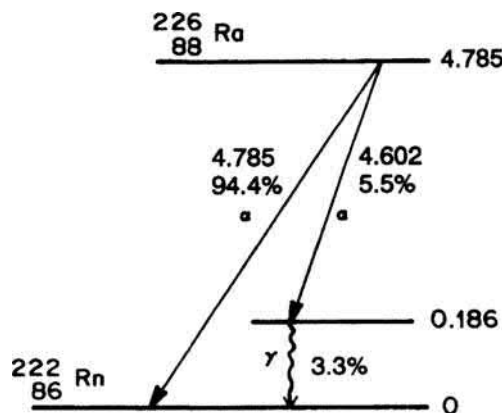


Fig. 6 Alpha decay scheme of $^{226}_{88}\text{Ra}$. Reproduced from Turner JE (2008) Atoms, Radiation, and Radiation Protection. John Wiley & Sons.

transformation of a neutron into a proton, where A remains the same. A general example for the decay of a parent (P) to a daughter (D), an electron (${}^0_{-1}e$), and antineutrino where an energy of Q is imparted to the daughter and beta particle:



In summary,



where an electron is created in the nucleus and ejected from the nucleus. This electron is the β^- particle which has the mass and charge of an electron. β^- particles have a short range (~ 3 m in air). They are more penetrating than alpha-particles and can be shielded by low- Z materials like plastic or thin metals. From a health physics perspective, β^- can be an internal hazard and pose a predominant risk to the skin and eye exposure pathways.

An example of β^- is the decay of ${}^{60}_{27}\text{Co}$



Similar to the computation of Q -value for alpha decay, the energy released is the mass-energy difference between the parent, daughter, and β^- particle.

$$Q = M_P - (M_D + M_e) = \Delta_P - \Delta_D \quad (18)$$

where M_P and M_D are the nuclear masses, and M_e is the mass of an electron, assuming that electron binding energy is neglected. Therefore, for β^- decay to be energetically possible, $M_P > (M_D + m)$. For the decay of ${}^{60}_{27}\text{Co}$, the Q -value energy released is 2.82 MeV. In β^- , the energy released is distributed among three products: the daughter nucleus, the β^- particle, and the antineutrino. Because of the relatively large mass of the recoil daughter nucleus, it receives very little kinetic energy. Therefore, the kinetic energy is distributed primarily between the β^- (E_{β^-}) and the antineutrino ($E_{\bar{\nu}}$). However, depending on the relative direction of the momenta of each of the three decay products, E_{β^-} and $E_{\bar{\nu}}$ vary between 0 and Q , which means E_{β^-} exists as a continuous energy spectrum with an average $E_{\beta^-} \sim Q/3$. This is in contrast to alpha decay, where the alpha particle has a single discrete energy, on account of the two-body product of alpha decay.

The decay scheme for ${}^{60}_{27}\text{Co}$ is given in Fig. 7. Over 99% of ${}^{60}_{27}\text{Co}$ undergoes β^- decay emitting a β^- with $E_{\beta^-} \sim Q/3 = 0.318/3 = 0.106$ MeV. Following the β^- , the daughter ${}^{60}_{28}\text{Ni}$ nucleus is still in an excited state, and de-excites to a stable ground state through the emission of two characteristic gamma-rays at 1.173 and 1.332 MeV. The ${}^{60}_{27}\text{Co}$ β^- , also decays via a rare mode of an excited state of ${}^{60}_{28}\text{Ni}$, with only the single 1.332 MeV gamma-ray emission to de-excite to the ground state.

A number of pure β^- emitters exist, meaning that they have no accompanying gamma rays, including ${}^3\text{H}$, ${}^{14}\text{C}$, ${}^{32}\text{P}$, ${}^{90}\text{Y}$, and ${}^{90}\text{Sr}$. Some mixed beta-gamma emitters include ${}^{60}\text{Co}$, ${}^{137}\text{Cs}$, and ${}^{131}\text{I}$.

β^+ Positron decay

Positron (β^+) decay occurs in particles that are proton rich. These proton rich nuclei gain stability by increasing the N/Z ratio by converting a proton to a neutron. In β^+ decay, a neutrino (ν) and a positron (${}^0_{+1}e$) are emitted from the nucleus, where A remains the same as with β^- decay:

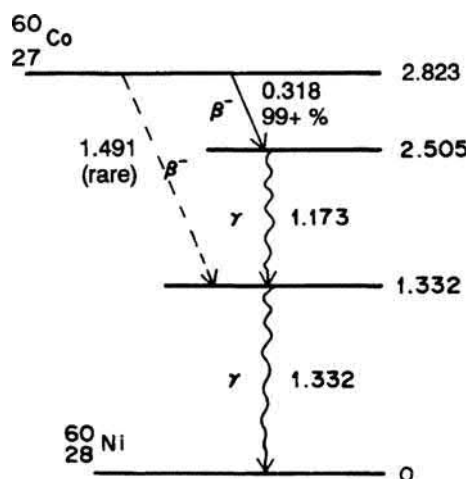


Fig. 7 β^- decay scheme of ${}^{60}_{27}\text{Co}$. Reproduced from Turner JE (2008) Atoms, Radiation, and Radiation Protection. John Wiley & Sons.

In essence,

$${}_1^1p \rightarrow {}_{+1}^0e + {}_0^1n + {}_0^0\nu \quad (20)$$

where a positron (the antiparticle of an electron with the same mass) is created (instead an electron as in β^- decay) and ejected from the nucleus.

An example of β^+ is the decay of ${}_{11}^{22}\text{Na}$:

$${}_{11}^{22}\text{Na} \rightarrow {}_{10}^{22}\text{Ne} + {}_{+1}^0e + {}_0^0\nu + Q \quad (21)$$

Similar to the computation of Q -value for β^- decay, the energy released is the mass-energy difference between the parent and daughter plus electron, but must further account for 2 electron masses (M_e) in the mass-energy calculations.

$$Q = M_P - (M_D + 2M_e) = \Delta_P - \Delta_D - 2M_e c^2 \quad (22)$$

where M_P and M_D are the nuclear masses, and M_e is the mass of an electron, assuming that electron binding energy is neglected. Therefore, for β^+ decay to be energetically possible, $M_P > M_D + 2M_e$. Recall, $2M_e = 2 \times 0.511 \text{ MeV} = 1.022 \text{ MeV}$. For the decay of ${}_{11}^{22}\text{Na}$, the Q -value energy released is 1.821 MeV. One of the decay schemes for ${}_{11}^{22}\text{Na}$ is given in Fig. 8, depicting how the daughter nucleus, following β^+ decay, is still left in an excited state and must de-excite to the ground state with 100% probability through the emission of a 1.275 MeV characteristic gamma-ray. Note that β^+ is one of two possible decay modes of ${}_{11}^{22}\text{Na}$, where electron capture competes with positron decay (see discussion next in section [Electron capture](#)). Although not explicitly sketched in Fig. 8, all β^+ create two accompanying 0.511 MeV photons when they annihilate with electrons, which are emitted essentially back-to-back at 180° , with some dependence on momentum transfers. You may notice in Figs. 8 and 9 that ${}_{11}^{22}\text{Na}$ has energy levels of 2.843 and 1.820 MeV and that this energy difference of 1.022 MeV ($= 2 \times 0.511 \text{ MeV}$) equals the rest mass of two electrons. These annihilation photons originate from a positron undergoing annihilation with an atomic electron.

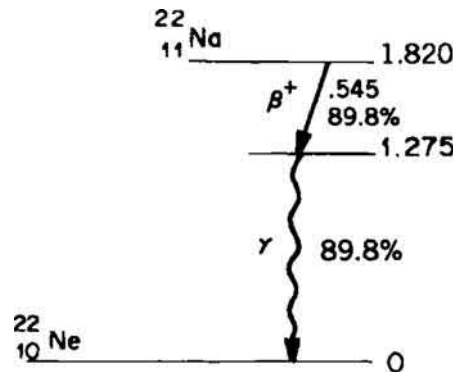


Fig. 8 β^+ Decay scheme of ${}_{11}^{22}\text{Na}$. Reproduced from Turner JE (2008) *Atoms, Radiation, and Radiation Protection*. John Wiley & Sons.

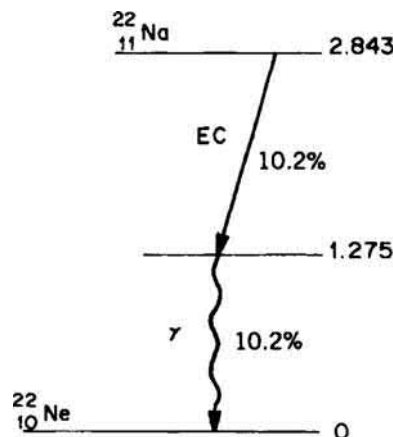


Fig. 9 EC decay scheme of ${}_{11}^{22}\text{Na}$. Reproduced from Turner JE (2008) *Atoms, Radiation, and Radiation Protection*. John Wiley & Sons.

Electron capture

Electron capture (EC) is a competing decay mode to β^+ decay for proton-rich nuclides in circumstances where β^+ decay is not energetically possible, i.e., where the $M_P > M_D + 2M_e$ energy conditions cannot be met. In EC, an atomic electron is captured by the nucleus, which transforms a proton (p) into a neutron (n) and an accompanying neutrino (ν) is emitted. EC has the same effect on the nucleus as β^+ decay:

$${}^A_Z P + {}^0_{-1} e \rightarrow {}^A_{Z-1} D + {}^0_0 \nu + Q \quad (23)$$

In summary:

$${}^1_1 p + {}^0_{-1} e \rightarrow {}^1_0 n + {}^0_0 \nu \quad (24)$$

Positron (β^+) decay occurs in radionuclides that are proton rich. These proton-rich nuclei gain stability by increasing the N/Z ratio by converting a proton into a neutron. In β^+ decay, a neutrino and a positron are emitted from the nucleus as a result of the transformation of a proton into a neutron, where A remains the same as with β^- decay:

$${}^A_Z P \rightarrow {}^A_{Z-1} D + {}^0_{+1} e + {}^0_0 \nu + Q \quad (25)$$

In essence,

$${}^1_1 p \rightarrow {}^0_{+1} e + {}^1_0 n + {}^0_0 \nu \quad (26)$$

In EC, nuclei capture an orbital electron, usually a K -shell electron, and must therefore overcome the binding energy (E_B) of the electron:

$$M_P + (M_e - E_B) \rightarrow M_D + Q \quad (27)$$

Therefore, the Q -value can be written as:

$$Q = M_P + M_e - (M_D + E_B) = \Delta_P - \Delta_D - E_B \quad (28)$$

For EC to be energetically possible, $M_P - M_D > E_B - M_e$.

Two EC decay schemes for ${}^{22}_{11}\text{Na}$ that compete with the β^+ from Fig. 8 are given in Fig. 9. The β^+ emission is the predominant decay mechanism, occurring 89.8% of the time, and EC occurring 10.2% of the time. In both EC cases, a gamma-ray with energy of 1.275 MeV occurs with 100% frequency, indicating that both β^+ and EC leave the daughter nucleus in an excited state, which must decay to a ground state through the emission of this characteristic gamma ray.

Given the mechanism of the capture of a (typically) K -shell electron, characteristic X-rays are also emitted in EC decay. The decay data for ${}^{22}_{11}\text{Na}$ is summarized (Table 1).

Gamma emission

Photon sources from radioactive transformations are ionizing forms of electromagnetic radiation. Photon emissions from radioactive transformations are distinguished as a function of their origin: (1) X-rays originate from de-excitation of electron orbital shells of an atom; (2) annihilation photons originate from positron/electron annihilation with energies of 0.511 MeV each; (3) bremsstrahlung originate from deceleration of electrons in matter (discussed later in detail in section **Light Charged Particles**); and (4) gamma rays emitted from the nuclei of excited daughter nuclides following radioactive transformations. Gamma rays (γ) generally have higher energies than characteristic X-rays. Gamma rays have no mass or charge, travel a long range in air (~ 100 m) before interacting. High- Z materials are required to shield gamma rays, where typical shielding materials employed are lead, steel, or concrete. From a health physics perspective, gamma rays are predominantly hazardous as an external total body source, but can also be internal hazards for gamma rays that deposit their energy within the body.

Nuclei that decay from an excited state with no change in the number of nucleons are called isomeric transitions (IT). Nuclides in the initial and final states are called isomers, and are denoted by (D^*) or (D^m), where " m " denotes a temporary, excited, "meta-stable" state. For each of the decay schemes discussed thus far, isomeric transition from an excited to ground state has occurred

Table 1 Decay data for ${}^{22}_{11}\text{Na}$ from β^+ and EC modes.

Decay mode	Average energy (MeV)	Probability
β^+	0.215 (max 0.545)	89.8%
β^+	0.8350	0.06%
β^+ annihilation	0.511	180%
γ	1.275	100%
Auger (K-shell e^-)	0.0008	9.20%
K-shell-X-ray	0.0008	0.13%

in the decay of $^{226}_{88}\text{Ra}$ accompanying alpha decay (see Fig. 6 in section **Alpha decay**); $^{60}_{27}\text{Co}$ following β^- decay (see Fig. 7 in section **β^- Decay**); and $^{60}_{27}\text{Co}$ following β^+/EC decay (see Fig. 8 in section **β^+ + Positron decay** and Fig. 9 in section **Electron capture**).

Unlike β -particles, which have an associated spectrum of β -energies, gamma rays emitted from the nucleus have discrete energies in an Γ to the ground state:

$$Q_{\Gamma} = M_P - M_D = \Delta_P - \Delta_D \quad (29)$$

Internal conversion

Internal conversion (IC) is an alternative pathway for excited nuclei to de-excite as a process of radioactive decay in which a nucleus in an excited state interacts electromagnetically with one of the atom's orbital electrons, resulting in emission of the electron from the atom. Emission of IC electrons always competes with emission of a photon (gamma ray) in de-excitation of excited states of nuclei. Unlike a β^- decay, which originates due to a transformation in the nucleus, the electron emitted in IC is from the orbital electrons (K- or L-shell, most likely). The energy of the ejected electron (E_e) is therefore equal to the difference between excess energy of the daughter nuclide E_{D^*} and the E_B is the binding energy of the atomic electron:

$$E_e = E_{D^*} - E_B \quad (30)$$

Just as with EC, the removal of orbital electron is followed by filling the gap in the orbital by an electron from a higher orbital, resulting in the emission of a characteristic X-ray.

Auger electrons

In an atom in which an L -shell electron transitions to fill a K -shell vacancy will not always emit a photon, especially in low- Z elements. Instead, a transition may occur in which an L -shell electron is ejected from the atom, thereby leaving two L -shell vacancies. These ejected electrons are referred to as Auger electrons, which an electron emitted from an atom when the energy produced by the filling of a vacancy in an inner shell with an electron from an outer shell of higher energy is transferred to another electron, rather than emitted as a characteristic X-ray. Auger electrons are produced in decay of many radionuclides. They also are produced when an inner-shell vacancy is created by irradiation of an atom by photons or electrons. The energy of the Auger electron is given by the difference of the binding energy of the inner shell (E_{B_i}) and twice the binding energy of the outer shell (E_{B_o}):

$$E_{\text{Auger}} = E_{B_i} - 2E_{B_o} \quad (31)$$

If an electron transitions from the L_I level into a K -shell vacancy, it will release an energy equal to the difference in binding energies ($E_K - E_{L_I}$), which can be transferred to an L_{III} electron. Therefore, the energy of the ejected Auger electron is equal to $E_K - E_{L_I} - E_{L_{III}}$.

Activity

The number of nuclear transformations (decays of radionuclides) occurring in a given quantity of material per unit time referred to as the activity. The SI unit of activity is the becquerel (Bq), which was named after Henri Becquerel, who discovered radioactivity in 1896. The unit of 1 Bq is 1 decay per second. The traditional unit of activity was the curie (Ci) named after Marie and Pierre Curie who discovered radium in 1898. This unit (1 Ci) is defined as the activity of 1 g of $^{226}_{88}\text{Ra}$, which is defined as 3.7×10^{10} Bq.

The process of radioactive decay can be mathematically represented as a first-order linear differential equation:

$$\frac{dN}{dt} = -\lambda N(t) \quad (32)$$

where N is the number of radioactive atoms and λ is a constant called the decay constant. The decay constant is the probability that a radionuclide will decay per unit time. The reciprocal of a decay constant ($1/\lambda$) is the mean (average) lifetime of a radionuclide. If at $t = 0$, the initial number of radioactive atoms is N_0 , the number of radioactive nuclei remaining after time t can be obtained by solving Eq. (32):

$$N(t) = N_0 e^{-\lambda t} \quad (33)$$

This is called the "radioactive decay law." This behavior is plot as a function of time in Fig. 10.

The activity, A (Bq), of a sample is related to the number of radioactive atoms present in the sample by the decay constant:

$$A(t) = \lambda N(t) \quad (34)$$

In terms of activity, Eq. (33) can be re-written as:

$$A(t) = A_0 e^{-\lambda t} \quad (35)$$

where A_0 is the initial activity at time, $t = 0$.

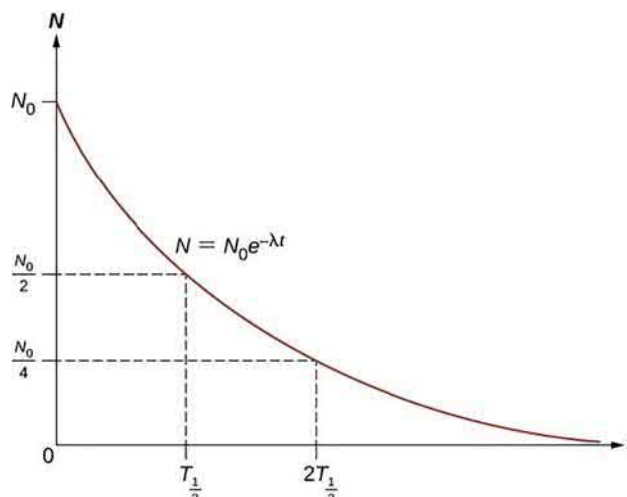


Fig. 10 Plot of radioactive decay law demonstrates the exponential decay of number of radioactive atoms to stable atoms; half-life is denoted as $T_{1/2}$ in this figure. Reproduced from Ling SJ, Sanny J and Moebs W (2019) *University Physics Volume 3*. OpenStax CNX. Available: <http://cnx.org/contents/af275420-6050-4707-995c-57b9cc13c358@12.2> (Accessed 2020).

Half-life

The period of time that a radioactive substance takes to decay to half of its initial amount (N_0) is called the half-life ($t_{1/2}$). After one half-life period has elapsed, 50% of the original number of atoms remain ($0.5 N_0$); after two half-life periods have elapsed, 25% of the original number of atoms remain ($0.25 N_0$); after three half-life periods have elapsed, 12.5% of the original number of atoms remain ($0.125 N_0$), etc.

The decay constant, λ , is related to the half-life, $t_{1/2}$, through the following relation:

$$t_{1/2} = \frac{\ln 2}{\lambda} \quad (36)$$

where $t_{1/2}$ is given in units of time (second, minutes, hours, days, years, etc.) and λ is given in units of inverse time (s^{-1} , h^{-1} , y^{-1} , etc.).

Measured half-lives can span from microseconds to billions of years. A list of radionuclides from natural, industrial, and medical sources/uses and their half-lives is summarized in **Table 2**.

Serial decay

Serial decay refers to a series of radioactive decay processes in which a parent radionuclide decays into one (or a successive series) of radionuclides before decaying to a stable isotope. For example, a serial decay case with a radioactive parent and a single intermediate radioactive daughter can be described as:



Table 2 Half-lives of commonly encountered natural, industrial, and medical radionuclides.

Source	Half-Life
<i>Natural radionuclides</i>	
Uranium-238	4.5 billion years
Thorium-232	14 billion years
Potassium-40	1.3 billion years
Radium-226	1600 years
<i>Industrial radionuclides</i>	
Cesium-137	30.17 years
Cobalt-60	5.271 years
Barium-133	10.5 years
Iridium-192	74 days
Americium-241	432.2 years
<i>Medical radioisotopes</i>	
Iodine-131	8.0 days
Thallium-201	3.0 days
Technetium-99m	6.0 h

where X_1 is a radioactive parent with decay constant λ_1 ; X_2 is a radioactive daughter with decay constant λ_2 ; and X_3 is a stable terminal daughter. To determine $N_1(t)$ of the parent radionuclide, X_1 , we may adhere to the solution to Eq. (32). However, the series of differential equations in Eq. (32) can be expanded to include the transformation of radioactive daughter X_2 to stable daughter X_3 .

$$\begin{aligned} N_1(t) &= N_1(0)e^{-\lambda_1 t} \\ \frac{dN_2}{dt} &= \text{Production} - \text{Decay} \\ \frac{dN_2}{dt} &= \lambda_1 N_1(t) - \lambda_2 N_2(t) \end{aligned} \quad (38)$$

where $N_2(t)$ is the number of radioactive atoms of the radioactive daughter, X_2 , at time t . For a general condition for a sample of X_1 with an initial number of radioactive atoms, $N_1(0)$, and for with an initial number of radionuclide X_2 of $N_2(0)$ yields the following solution to Eq. (38):

$$N_2(t) = N_1(0) \left[\frac{\lambda_1}{\lambda_2 - \lambda_1} \right] [e^{-\lambda_1 t} - e^{-\lambda_2 t}] + N_2(0)e^{-\lambda_2 t} \quad (39)$$

If there are no daughter nuclides initially present at $t = 0$ (i.e., $N_2(0) = 0$), such as from a fresh sample of just X_1 , then the second term in Eq. (39) crosses out.

If solving for the activity of X_2 by combining Eq. (34) and Eq. (39)

$$\begin{aligned} A_2(t) &= \lambda_2 N_2(t) \\ A_2(t) &= A_1(0) \left[\frac{\lambda_2}{\lambda_2 - \lambda_1} \right] [e^{-\lambda_1 t} - e^{-\lambda_2 t}] + A_2(0)e^{-\lambda_2 t} \end{aligned} \quad (40)$$

The general solution to serial decay for a chain of i radionuclides is given by the Bateman equations:

$$\frac{dN_i}{dt} = \lambda_{i-1}N_{i-1}(t) - \lambda_i N_i(t) \text{ for } (i = 2, n) \quad (41)$$

Radioactive equilibrium

Radioactive equilibrium for a decay chain occurs when each radionuclide decays at the same rate it is being produced. If the half-life of the daughter radionuclide is much longer than the parent, then equilibrium cannot occur.

There are three general cases for potential radioactive equilibrium.

Case 1: Secular equilibrium—The half-life of the parent is much larger than the daughter ($t_{1/2(1)} \gg t_{1/2(2)}$).

Case 2: Transient equilibrium—The half-life of the parent is approximately/greater than the half-life of the daughter ($t_{1/2(1)} \geq t_{1/2(2)}$).

Case 3: No equilibrium—The half-life of the parent is less than the daughter ($t_{1/2(1)} < t_{1/2(2)}$).

Case 1: Secular Equilibrium

Secular equilibrium is a case in which the half-life of the parent radionuclide is much larger than the half-life of the daughter. If $t_{1/2(1)} \gg t_{1/2(2)}$, then $\lambda_{1/2(1)} \ll \lambda_{1/2(2)}$. This means that $\frac{\lambda_2}{\lambda_2 - \lambda_1} \sim 1$ and $e^{-\lambda_1 t} \sim 1$ in Eq. (40) and simplifies to:

$$A_2(t) = A_1(0)[e^{-\lambda_1 t} - e^{-\lambda_2 t}] + A_2(0)e^{-\lambda_2 t} \quad (42)$$

After a period of ingrowth of $\sim 7t_{1/2(1)}$ and assuming at $A_2(0) = 0$, then secular equilibrium is reached where the condition $A_1(t) = A_2(t)$, i.e. $\lambda_1 N_1(t) = \lambda_2 N_2(t)$. Secular equilibrium is depicted in Fig. 11 (Oak Ridge Institute For Science Education, 2011).

A common example of secular equilibrium occurs during the double β^- decay of $^{90}_{38}\text{Sr}$, where $^{90}_{38}\text{Sr}$ is a product of nuclear fission from nuclear reactors, and it is found in the environment as a result of nuclear weapons tests (see section Fission).



In this decay chain, the parent $^{90}_{38}\text{Sr}$ has a half-life of 29.12 y, and the daughter $^{90}_{39}\text{Y}$ has a half-life of 64.0 h before decaying to $^{90}_{40}\text{Zr}$ (stable). As $t_{1/2(^{90}_{38}\text{Sr})} \gg t_{1/2(^{90}_{39}\text{Y})}$, the assumptions and simplifications of the serial decay equation apply, resulting in a case of secular equilibrium.

Case 2: Transient Equilibrium

Transient equilibrium is a case encountered in which the half-life of the parent is slightly greater than the half-life of the daughter. If $t_{1/2(1)} \geq t_{1/2(2)}$ then $\lambda_{1/2(1)} < \lambda_{1/2(2)}$. No simplification to the general solution is possible, but the behavior of $\frac{A_2(t)}{A_1(t)} = \frac{\lambda_2}{\lambda_2 - \lambda_1}$. This means that after $\sim 10t_{1/2(1)}$, and assuming $A_2(0) = 0$, then $e^{-\lambda_2 t}$ becomes negligible and A_1 and A_2 exist in a constant ratio to each other. Transient equilibrium differs from secular equilibrium in that significant decay of the parent takes place and the activity of the decay product is greater than the activity of the parent. Transient equilibrium is depicted in Fig. 12.

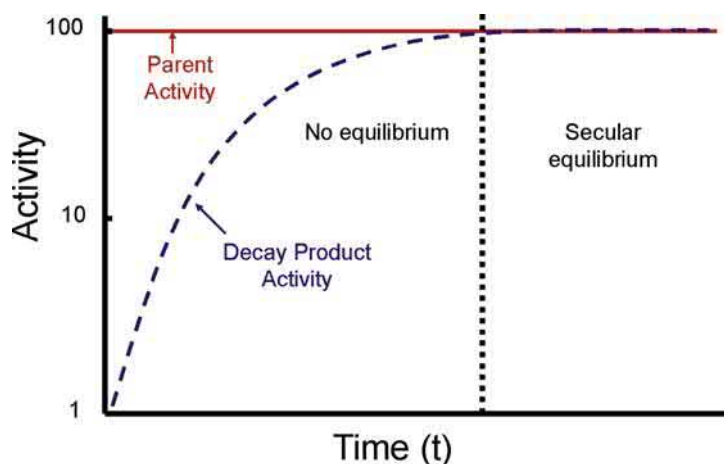


Fig. 11 Activity of parent and daughter in secular equilibrium. Equilibrium conditions occur where $A_1(t) = A_2(t)$ after a period of ingrowth of $\sim 7t_{1/2(1)}$. Reproduced from Oak Ridge Institute For Science Education (2011) *Radiological and Chemical Properties of Uranium*. US Nuclear Regulatory Commission.

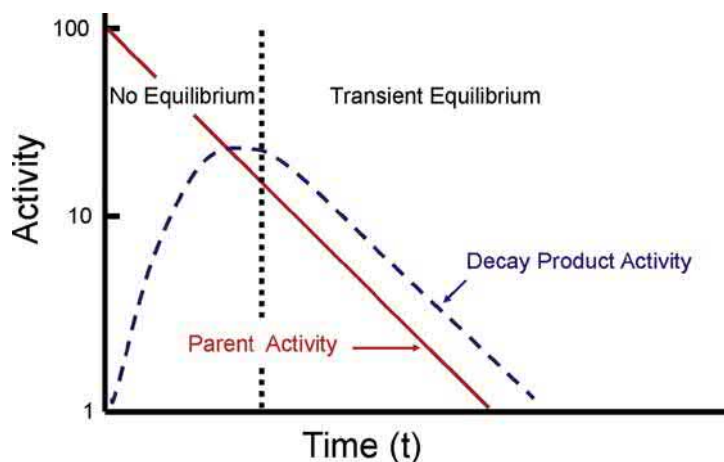


Fig. 12 Activity of parent and daughter in transient equilibrium. Equilibrium conditions occur $A_2(t)/A_1(t) = \text{constant ratio}$ after a period of ingrowth of $\sim 10t_{1/2(1)}$. Reproduced from Oak Ridge Institute For Science Education (2011) *Radiological and Chemical Properties of Uranium*. US Nuclear Regulatory Commission.

An example of transient equilibrium is the elution of the medical isotope, $^{99m}_{43}\text{Tc}$, where $^{99}_{42}\text{Mo}$ β^- decays into an excited metastable state $^{99m}_{43}\text{Tc}$, before emitting a 0.15 MeV gamma ray. $^{99m}_{43}\text{Tc}$ radiopharmaceuticals are used in medical diagnoses, including cardiac studies and bone scans.



In this decay chain, the parent $^{99}_{42}\text{Mo}$ has a half-life of 65.94 h, and the daughter $^{99m}_{43}\text{Tc}$ has a half-life of 6.02 h before decaying to $^{99}_{43}\text{Tc}$ (stable). As $t_{1/2}(^{99}_{42}\text{Mo}) \geq t_{1/2}(^{99m}_{43}\text{Tc})$, following an extended period of ingrowth, both $^{99}_{42}\text{Mo}$ and $^{99m}_{43}\text{Tc}$ reach transient equilibrium, and hold a constant ratio of their activities. This property makes it possible to elute $^{99m}_{43}\text{Tc}$ in hospitals for medical diagnostics onsite in hospitals through the use of "generators."

Case 3. No Equilibrium

In the case where the half-life of the parent is less than the daughter ($t_{1/2(1)} < t_{1/2(2)}$), then the parent rapidly decays while the daughter builds to a maximum before decaying away due to its own half-life. This is depicted in Fig. 13.

A case in which no equilibrium conditions exist is the alpha decay of $^{218}_{84}\text{Po}$, which exists in the $^{222}_{86}\text{Rn}$ decay chain originating from $^{238}_{92}\text{U}$.



In this decay chain $^{218}_{84}\text{Po}$ undergoes alpha decay with a half-life of 3 min to $^{214}_{82}\text{Pb}$, which β^- decays with a half-life of 27 min to $^{214}_{83}\text{Bi}$. Since $t_{1/2}(^{218}_{84}\text{Po}) < t_{1/2}(^{214}_{82}\text{Pb})$, no equilibrium conditions are met.

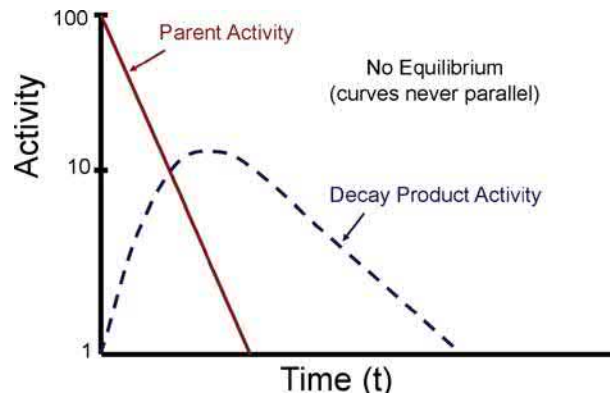


Fig. 13 Activity of parent and daughter where no equilibrium conditions are met. The parent rapidly decays as the daughter grows during the period of ingrowth, after which the daughter decays away at the rate governed by its own half-life, with no new production from the parent. Reproduced from Oak Ridge Institute For Science Education (2011) *Radiological and Chemical Properties of Uranium*. US Nuclear Regulatory Commission.

Photon interactions

Photons are electromagnetic waves having no charge or mass but exhibit both particle and wave-like behavior. Photons are emitted through radioactive decay to de-excite the atom through the emission of characteristic X-rays or the nucleus through the emission of gamma-rays. The distance the photon travels is dependent on the photon energy and on the material through which it is traversing, which may be estimated by probability of interaction per unit distance. Since photons are a form of indirectly ionizing radiation, interactions with an intermediary charged particle, i.e., electron, are the mechanism by which energy is deposited. The effect of a photon interacting in matter can yield three possible results per interaction: complete absorption of the photon; elastic scattering of the photon; or inelastic scattering of the photon. These behaviors are observed in the five major types of photon interactions.

Rayleigh (coherent) scattering

Rayleigh (or "coherent") scattering occurs when photons are scattered by very small particles (less than $\sim 1/10$ th of the wavelength of the photon). This phenomenon is of interest in some physics experiments, but not of much interest for health physics since no net energy is transferred to a charged particle, nor does any ionization or excitation occur to contribute to dose.

Photoelectric effect

The photoelectric effect was explained by Einstein (1905) as an absorption process in which an incident photon (gamma ray or X-ray) ejects an electron from an atom, with all the photon energy absorbed in ejecting the electron and imparting kinetic energy to it. A characteristic X-ray or an Auger electron then is produced when the vacated shell is filled by another orbital electron from a higher shell. Einstein was awarded the 1921 Nobel Prize in Physics "his discovery of the law of the photoelectric effect" (Nobel Foundation, 2020).

The relationship between the energy, frequency, and wavelength of electromagnetic radiation according to the equation derived by Max Planck (Planck, 1901) and harnessed by Albert Einstein to explain the photoelectric effect (Einstein, 1905) in Eq. (46):

$$E = h\nu = \frac{hc}{\lambda} \quad (46)$$

Photoelectric interactions occur when a photon with a given energy, E , interacts with a tightly bound atomic electron resulting in the absorption of the photon and the ejection of an electron, called the photoelectron. The maximum kinetic energy of the photoelectron, T , is equal to the incident photon energy less the energy required for the bound electron to escape:

$$T = h\nu - \phi \quad (47)$$

The energy required for the electron to escape from the surfaces called the work function, ϕ . This minimum energy must overcome attractive forces. Momentum is conserved by the photoelectron and the recoil atom.

The probability of photoelectric interaction occurring increases rapidly with the binding energy, with the greatest probability of an incident photon undergoing a photoelectric interaction if the photon has low energy, $h\nu$, and interacting in a high- Z material. The resulting vacancy left in the atom from the ejection of the photoelectron results in filling from an outer shell electron; the change of transition energy to fill the vacancy thus results in the emission of fluorescent radiation, i.e., a characteristic X-ray. Consequently, approximately 80% of photoelectric absorption processes take place in the K -shell of atomic electrons.

Compton effect

In 1922, Arthur Holly Compton was conducting scattering experiments using molybdenum K_{α} X-rays ($E = 17.4$ keV) directed at a graphite target, where the wavelength of the incident and scattered were measured (Compton, 1922). It was found that the energy of the scattered photon varied as a function of photon scattering angle following a scattering interaction with an electron. This discovery challenged the traditional notion that the incident and scattered photon wavelengths should remain unchanged. By demonstrating that light could not be explained as purely a wave phenomenon, the experimental results implied that the photon had particle-like behavior. This interaction in which an incident photon scatters from an orbital electron in an atom to produce a recoil electron, a residual ionized atom, and a photon of energy less than the energy of the incident photon was called the Compton effect. Compton was awarded the Nobel Prize in Physics in 1927 “for his discovery of the effect named after him” (Nobel Foundation, 2020).

Compton interactions are photon scattering interactions with an orbital electron, when the electron struck by the incoming photon is unbound and stationary. Although, the electrons are not “completely free,” so the binding energies have a very small relative impact on the energy transfer (Fig. 14).

The kinematics of Compton scattering conserve both energy and momentum before and after interaction. The energy of the recoil electron (rest mass $m_e c^2$) is given by T at an angle φ and energy of the scattered photon by $h\nu'$ at an angle θ with respect to the incident photon direction.

$$h\nu' = \frac{h\nu}{1 + \left(\frac{h\nu}{m_e c^2}\right)(1 - \cos\theta)} \quad (48)$$

where

$$T = h\nu - h\nu' \quad (49)$$

When the scattering angle of the photon (θ) is small, the scattered photon travels in the forward direction, imparting little energy to the electron, which moves off at nearly right angles to the direction of the incident photon. As the photon scattering angle increases from 0° to 180° (the maximum scattering angle), the recoil electron angle (φ) decreases from 90° to 0° .

At low photon energies, Compton scattering is not an efficient method of energy transfer, compared to higher photon energies. Let us compare the fraction of the maximum energy transferred (T) to the Compton electron (at angle of $\theta = 180^\circ, \varphi = 0^\circ$) from a 5.11 keV incident photon vs. a 5.11 MeV incident photon. If we use Eqs. (48) and (49) to determine the fraction of incident photon energy transferred to the Compton electron ($T/h\nu$), Table 3 demonstrates that Compton interactions for low energy photons are a poor mechanism of energy transfer to charged particles compared to higher energy photons.

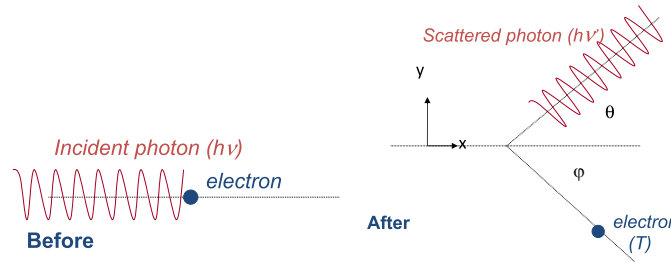


Fig. 14 Compton scattering kinematics of incident photon with a free electron at rest. Following the collision, the photon is scattered at an angle θ with an energy $h\nu'$ and the electron at with an energy T and angle φ .

Table 3 Comparison of fraction of energy transferred from low and high energy incident photons to a Compton electron through Compton scattering.

Photon energy, 5.11 keV	Photon energy, 5.11 MeV
$\alpha = \frac{5.11 \text{ keV}}{0.511 \text{ MeV}} = 0.010$	$\alpha = \frac{5.11 \text{ MeV}}{0.511 \text{ MeV}} = 10$
$E_{e(\max)} = 5.11 \text{ keV} \times \left(2 \times \frac{0.01}{1.02}\right)$	$E_{e(\max)} = 5.11 \text{ MeV} \times \left(2 \times \frac{10}{21}\right)$
$= 0.10 \text{ keV}$	$= 4.87 \text{ MeV}$
$h\nu'(\min) = 5.11 \text{ keV} \times \frac{1}{1.02}$	$h\nu'(\min) = 5.11 \text{ MeV} \times \frac{1}{21}$
$= 5.01 \text{ keV}$	$= 0.24 \text{ MeV}$
Energy transferred: 2%	Energy transferred: 95%

Pair production

Pair production is an absorption process in which an incident photon is annihilated in close proximity to an atomic nucleus, producing an electron and positron pair. This process occurs at incident photon energies greater than the rest-mass energy of an electron and positron combined ($h\nu \geq 2m_e c^2$). This phenomenon was observed in the cloud chamber experiments of Patrick Maynard Stuart Blackett, resulting in the 1948 Nobel Prize “for his development of the Wilson cloud chamber method, and his discoveries therewith in the fields of nuclear physics and cosmic radiation” (Nobel Foundation, 2020).

The threshold energy required for the observed reaction is twice the rest mass of an electron, resulting in the conversion of the photon into an electron-positron pair in the field of the atomic nucleus (Fig. 15).

When pair production occurs in the nuclear field, the nucleus has a negligible recoil kinetic energy. This results in the incident photon energy being converted into an electron and positron (masses $2m_e c^2$) with remaining kinetic energy distributed as T^+ and T^- between both the positron and electron, respectively. The kinetic energy distribution of the positron and electron can vary from 0 to $h\nu - 2m_e c^2$. At the end of the positron's flight path, the positron will slow and attract an electron to form an electron-positron complex called positronium, lasting 10^{-10} s before annihilating into two photons with energy $m_e c^2$ (0.511 MeV). Conservation of energy in the nuclear field from pair production becomes:

$$h\nu = 2m_e c^2 + T^+ + T^- \quad (50)$$

where the average kinetic energy ($\bar{T}$) imparted to each of the electron and positron is:

$$\bar{T} = \frac{h\nu - 2m_e c^2}{2} \quad (51)$$

Pair production can also occur in the atomic electron field (called “triplet” production, but with a lower probability and higher photon threshold minimum energy of ($h\nu \geq 4m_e c^2$)). In the electron field, the incident photon distributes its energy between the electron-positron pair, as well as the additional host electron. Conservation of energy in the electron field from pair production becomes:

$$h\nu = 2m_e c^2 + T^+ + T_1^- + T_2^- \quad (52)$$

where the average kinetic energy ($\bar{T}$) imparted to each of the three particles is:

$$\bar{T} = \frac{h\nu - 2m_e c^2}{3} \quad (53)$$

Photonuclear interactions

At very high photon energies, a photon can be absorbed in the nucleus, excite the nucleus, and liberate a nucleon (proton or neutron) through the process of photo-disintegration, which is known as a photonuclear interaction. Analogous to the photoelectric effect, the incident photon must have sufficient energy to overcome the threshold of the large binding energy of the nucleon, which, referring back to Fig. 3, is in the order of several MeV. The probability of photonuclear interactions, overall, is orders of magnitude lower than the combined probability of the photoelectric effect, Compton effect, and pair production combined.

Two interactions from photonuclear disintegration can occur: (γ, n) and (γ, p). The threshold energy for (γ, p) exceeds that of (γ, n) due to the required energy to overcome the Coulombic barrier a photon must overcome to escape the nucleus. From a health physics perspective, the (γ, p) events contribute to dose, though only $< 5\%$ due to the dominance of pair production. The (γ, n) interaction is of more practical importance from a health physics perspective, with neutrons leading to further challenges in radiation protection

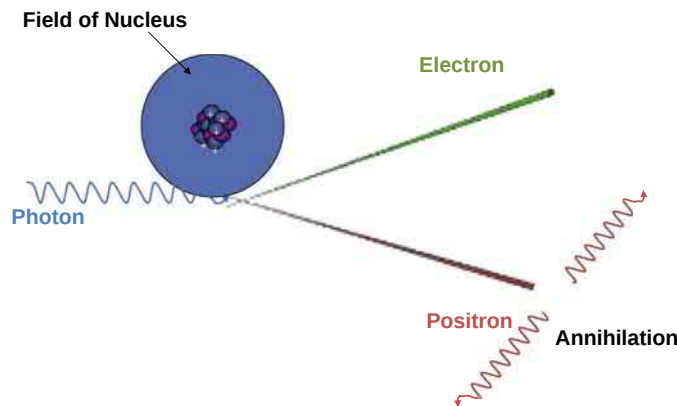


Fig. 15 Pair production interaction from an incident photon with energy $\geq 2m_e c^2$ interacting in the nuclear field, converting into an electron and a positron. The positron annihilates into two back-to back characteristic photons each with energy $m_e c^2$.

and more complex shielding design compared to just shielding considerations for γ -rays alone. Furthermore, (γ, n) become unwanted secondary effects in the use of high-energy radiotherapy beams.

Linear attenuation coefficient

Indirectly ionizing radiation (as photons or neutrons) have a high probability of traversing matter without interacting. As such, unlike charged particles, these uncharged particles are not characterized as having a range, but rather have a probability of interaction per unit path length. In the case of photons, these interactions are predominantly photoelectric effect, Compton scattering, or pair production. If it is assumed that a narrow beam of monoenergetic photons impinges perpendicularly on a flat plate of thickness, x (units of cm) and that each photon is completely absorbed in a single interaction without producing secondary radiation, the probability per unit path length that the photon will travel before interacting is defined by the linear attenuation coefficient, μ (units of cm^{-1}) (Fig. 16).

The probability that the photon will interact in an infinitesimal thickness dx becomes μdx . If N particles are incident upon the slab of thickness dx , the change (dN) in the number of photons impinging (N) due to absorption in the slab becomes:

$$dN = -\mu dx \quad (54)$$

To find the total change in the number of particles due to absorption in the slab of thickness x , we integrate over both sides of Eq. (54):

$$\int_{N_0}^{N_x} \frac{dN}{N} = - \int_0^x \mu dx \quad (55)$$

$$N(x) = N_0 e^{-\mu x} \quad (56)$$

where N_0 is the number of photons impinging on the slab and N_x is the number of photons transmitted through a slab of thickness, x . The number of transmitted photons is exponentially attenuated as a function of the thickness of the slab and the linear attenuation coefficient (which is defined for the photon energy). The number of interactions that occur are therefore equal to the difference of the number of incident and transmitted photons, $N_0 - N(x)$.

The average distance traveled between collisions is referred to as the mean free path, $1/\mu$ (units, cm). In health physics, the half-value layer (HVL) is the amount of material needed (x) to absorb 50% of the incident photons, where the tenth-value layer (TVL) is the amount of material needed to absorb 90% of the radiation i.e., reduce it to one-tenth of the original intensity.

Plural modes of absorption

If more than one absorption process is present, the total linear attenuation coefficient can be written as a sub of the constituent parts

$$\mu = \mu_1 + \mu_2 + \dots \quad (57)$$

where μ_1, μ_2 , etc., are called the partial linear attenuation coefficients for process 1, process 2, etc. For each constituent process, it is assumed that each event is totally absorbing and that it produces no secondary particles or interactions (e.g., scattering). Therefore, number of interactions due to a single process, for example μ_1 , alone is:

$$\Delta N = (N_0 - N(x)) \frac{\mu_1}{\mu} = N_0 (1 - e^{-\mu x}) \frac{\mu_1}{\mu}$$

where $\frac{\mu_1}{\mu}$ is the fraction of interactions that occur by process 1. The same analog can be made for μ_2 due to process 2, etc.

The linear attenuation coefficient for photons of a given energy in a specific material is composed of the individual contributions from the physical processes that can remove photons from the narrow beam, i.e., photoelectric effect, Compton scatter, and pair production (Rayleigh scattering and photonuclear interactions could be included, but due to the low probability of interaction, can be omitted for the purposes of this discussion).

$$\mu = \tau + \sigma + \kappa \quad (58)$$

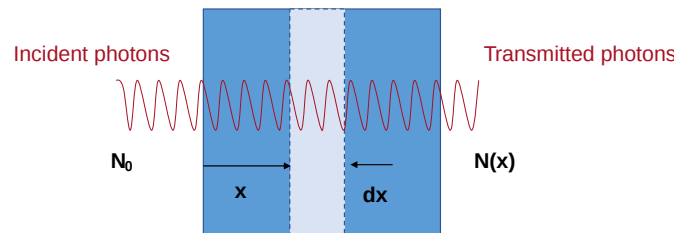


Fig. 16 A narrow monoenergetic parallel beam of photons (N_0) impinging on a slab of thickness x will exponentially attenuate and transmit $N(x)$ photons governed by $N(x) = N_0 e^{-\mu x}$, where μ is the linear attenuation coefficient defined as a function of absorber material and incident photon energy.

where μ is the total interaction probability, linear attenuation coefficient; τ is the photoelectric interaction probability; σ is the Compton interaction probability; and κ is the pair production interaction probability (all in units cm^{-1}).

Fig. 17 shows the relative importance of each of the three major photon interaction mechanisms as a function of absorber material with (Z) electrons per atom and photon energy ($h\nu$), and it outlines the region where each interaction dominates. Lines where $\tau = \sigma$ and $\sigma = \kappa$ mark where two types of interactions have equal probability. At low photon energies and low Z , the photoelectric effect dominates, where at higher energies, pair production dominates. In the low- Z region, Compton interactions dominate over a broad region of 20 keV to 30 MeV, and narrows as a function of increasing Z .

Mass attenuation coefficient

The attenuation coefficient can be normalized as a function of density, ρ (g/cm^3) to yield the quantity of the mass attenuation coefficient (μ/ρ) in units cm^2/g . Just as in Eq. (58), the mass attenuation coefficient is the sum of the photoelectric mass attenuation coefficient, Compton mass attenuation coefficient, and pair production mass attenuation coefficient.

$$\frac{\mu}{\rho} = \frac{\tau}{\rho} + \frac{\sigma}{\rho} + \frac{\kappa}{\rho} \quad (59)$$

Fig. 18 shows the plot of the total mass attenuation coefficient and the constituent interaction coefficients for photoelectric effect, coherent (Rayleigh) scattering, incoherent (Compton) scattering, and pair production in both the nuclear and electron fields (Berger et al., 2010). The sum of the pair production coefficients in the nuclear and electron fields yields the total pair production mass attenuation coefficient. As shown in Fig. 17, the probability of each interaction occurring varies as a function of photon energy and Z of the interacting absorber. The total and constituent mass attenuation coefficients are given for carbon, lead, and water in Fig. 18.

In Fig. 18c, a prominent K -edge of Pb is given at 88 keV. At this energy, the two K -shell electrons cannot participate in the photoelectric effect because binding energy is too large to overcome. Only the L -shell, M -shell, and higher shell electrons can participate in photoelectric interactions in Pb. Just above 88 eV, K -electrons can also participate. The magnitude of step function from 7.1 to $1.7 \text{ cm}^2/\text{g}$ indicates the importance of contributions of two- K -shell electrons in photoelectric cross-section, compared to other the 80 electrons in Pb atom.

Mass energy absorption and mass energy transfer coefficients

In health physics, we are interested in the energy absorbed in matter from a source of photons. The energy absorbed in a material (e.g., shield, tissue, etc.) is related to the linear attenuation coefficient. A photon interaction will, in general, result in transfer of energy to a short range particle (mostly electrons). Energy will be absorbed from photon absorption interactions, such as photoelectric effect and pair production, and transferred to electrons by Compton scattering interactions. The energy transferred to the electron may be absorbed within the material, or it may leave the region of interest. The energy transferred to charged particles is described by the photon energy transfer coefficient, and the fraction of the charged particle energy that results in local energy deposition (i.e., dose deposition) excluding radiative losses by charged particles (discussed in section, **Light Charged Particles**) is given by the photon energy absorption coefficient.

If we expand upon Eq. (59), the total mass energy transfer coefficient (μ_{tr}/ρ) for a photon of incident energy $h\nu$ is equal to the sum of the constituent transfer interactions.

$$\frac{\mu_{tr}}{\rho} = \frac{\tau_{tr}}{\rho} + \frac{\sigma_{tr}}{\rho} + \frac{\kappa_{tr}}{\rho} \quad (60)$$

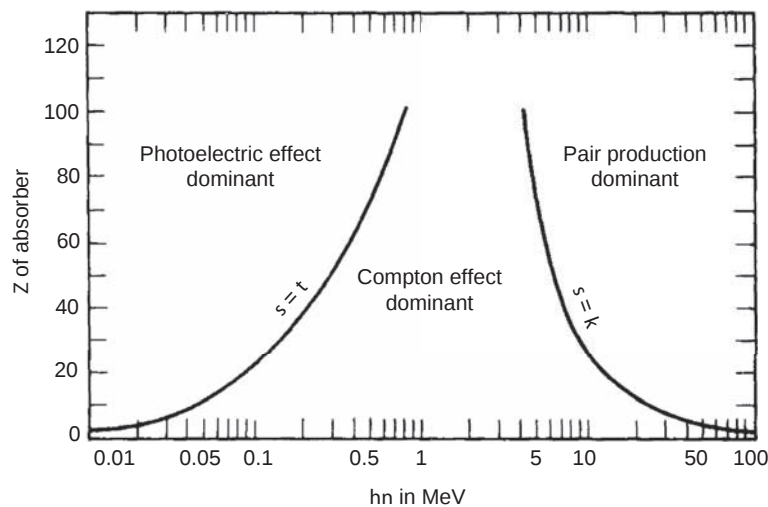


Fig. 17 Probability of photoelectric effect (τ), Compton scatter (σ), and pair production (κ) as a function of photon energy ($h\nu$) and Z of absorber. Lines where $\tau = \sigma$ and $\sigma = \kappa$ mark where two types of interactions have equal probability. Reproduced from Attix FH (2008). *Introduction to Radiological Physics and Radiation Dosimetry*. John Wiley & Sons.

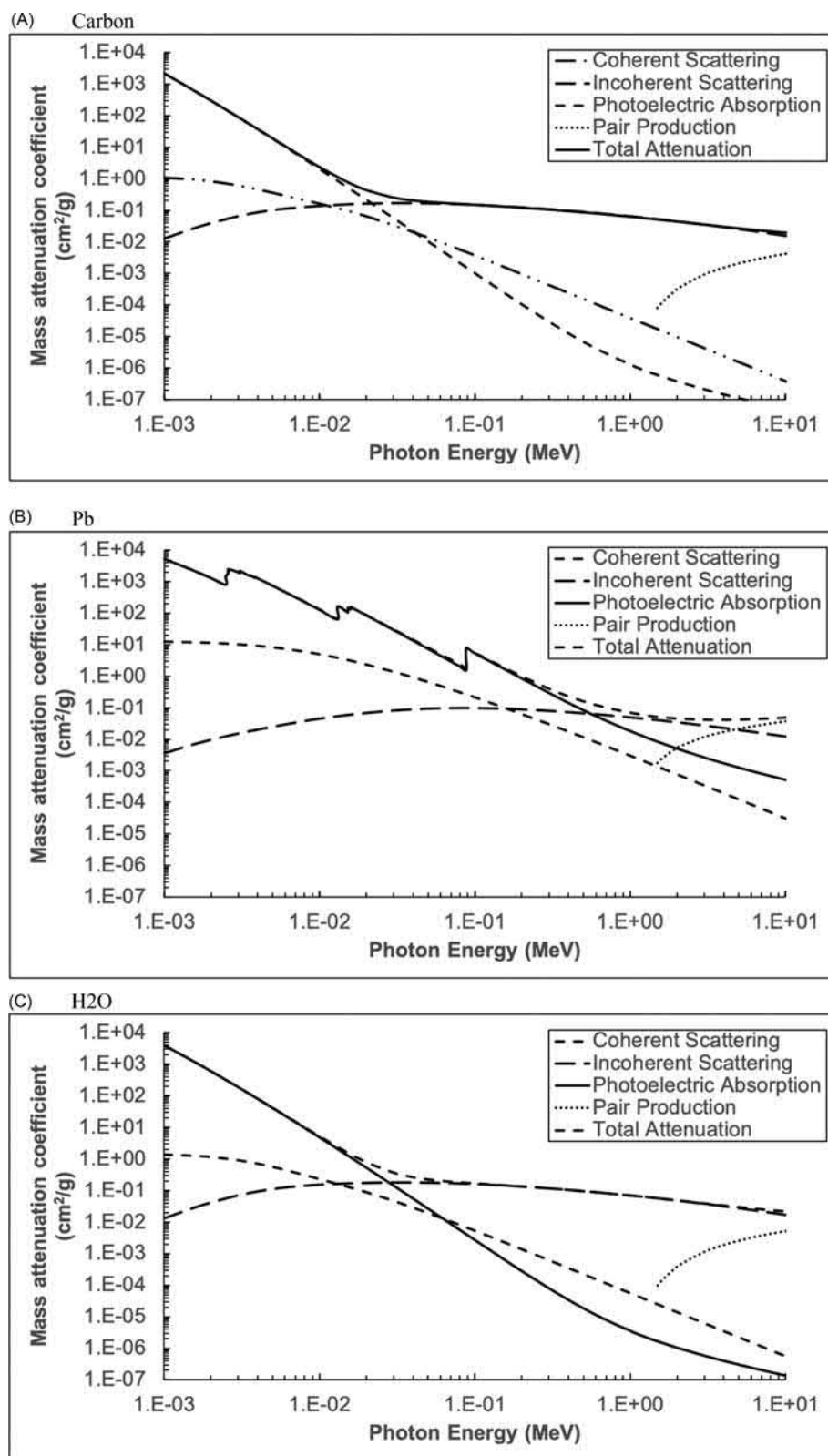


Fig. 18 Total mass attenuation coefficients (cm²/g) for (a) carbon ($Z = 6$); (b) water; and (c) lead ($Z = 82$). Total coefficient is broken down into coherent (Rayleigh) scattering; incoherent (Compton) scattering; photoelectric absorption; and pair production in nuclear (double) and electron (triplet) fields. Generated from Berger M, Hubbell J, Seltzer S, Chang J, Coursey J, Sukumar R, Zucker D and Olsen K (2010) *XCOM: Photon Cross Sections Database*, NIST Standard Reference Database 8 (XGAM). Available: <https://www.nist.gov/pml/xcom-photon-cross-sections-database> (Accessed 2020) at NIST.gov.

In each of the photoelectric, Compton, and pair production interactions, the transfer coefficient is the product of the total attenuation coefficient for the interaction multiplied by the fraction of the incident photon energy transferred to electrons (Turner, 2008), summarized in Table 4.

To define the mass energy absorption coefficient, $\frac{\mu_{en}}{\rho}$, we must first define g to represent the average fraction of the initial photon energy transferred to charged particles, that are subsequently emitted as radiative interactions, i.e., bremsstrahlung (discussed in section Light Charged Particles). Therefore, the mass energy absorption coefficient can be defined as:

$$\frac{\mu_{en}}{\rho} = \frac{\mu_{tr}}{\rho} (1 - g) \quad (61)$$

The mass absorption coefficient ($\frac{\mu_{ab}}{\rho}$) is the most important parameter for the calculation of dose. For high- Z absorbers, g is largest. For low- Z absorbers and high photon energies, $g \rightarrow 0$. Therefore, $\frac{\mu_{en}}{\rho} \sim \frac{\mu_{tr}}{\rho}$ for low- Z and low incident photon energies. This is summarized for our examples of C ($Z = 6$) and Pb ($Z = 82$) in Table 5 and retrievable at Hubbell and Seltzer (1996).

Neutrons

Scattering interactions

Interactions of neutrons with matter may occur by scattering with nuclei or by absorption. In the case of scattering, the neutron experiences a billiard ball-type collision, where the kinetic energy of the neutron and nucleus are the same before and after the collision, or it may interact with the internal energy of the nucleus and be emitted from the nucleus. Neutrons undergo either absorption or scattering reactions with nuclei. If a neutron is absorbed into the nucleus, it may decay by a variety of mechanisms. It may fission, eject one or more neutrons, undergo alpha particle decay, emit a photon, decay by ejecting a proton, etc. Scattering may be elastic or inelastic. Elastic scattering may be well-defined by two particle kinematics or it may be a resonance phenomenon. Inelastic scattering occurs when the neutron is absorbed and one is emitted with an isotropic angular distribution. This may or may not be a resonance phenomenon.

For the case of absorption, a compound nucleus is formed that may decay through a variety of channels, as illustrated by Eq. (62).



where

$$({}_Z^{A+1}X)^* \begin{cases} {}_0^1n + {}_Z^AX \text{ resonance elastic scattering} \\ {}_0^1n + ({}_Z^AX)^* \text{ inelastic scattering} \\ {}_Z^{A+1}X + \gamma \text{ radiative capture} \\ {}_Z^{A_1}X + {}_Z^{A_2}X + (2-3){}_0^1n \text{ fission} \end{cases}$$

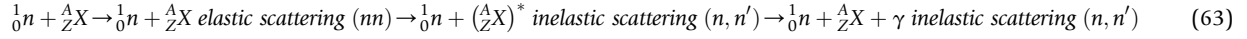
Table 4 Mass energy transfer coefficients defined for photoelectric, Compton scatter, and pair production interactions.

Mass attenuation coefficient (cm^2/g)	Fraction of incident photon energy transferred to charged particles	Mass energy transfer coefficient (cm^2/g)
Photoelectric ($\frac{\tau}{\rho}$)	$1 - \frac{\phi}{h\nu}$ from Eq. (47)	$\frac{\tau_{tr}}{\rho} = \left(\frac{\tau}{\rho}\right) \left(1 - \frac{\phi}{h\nu}\right)$
Compton ($\frac{\sigma}{\rho}$)	$\frac{T_{avg}}{h\nu}$ from Eq. (49)	$\frac{\sigma_{tr}}{\rho} = \left(\frac{\sigma}{\rho}\right) \left(\frac{T_{avg}}{h\nu}\right)$
Pair Production ($\frac{\kappa}{\rho}$)	$1 - \frac{2m_0c^2}{h\nu}$ from Eq. (51)	$\frac{\kappa_{tr}}{\rho} = \left(\frac{\kappa}{\rho}\right) \left(1 - \frac{2m_0c^2}{h\nu}\right)$

Table 5 Relationship of mass energy transfer coefficient and mass energy absorption coefficient for low- Z and high- Z materials as a function of incident photon energy.

Incident photon energy, $h\nu$ (MeV)	$(\mu_{tr} - \mu_{en})/\mu_{tr}$	
	$Z = 6$	$Z = 82$
0.1	0%	0%
1.0	0%	4.8%
10	3.5%	26%

Several examples of neutron-nucleus reactions are shown in Eq. (63).



with other reactions of interest including, $(n, 2n)$, (n, p) , (n, α) , and (n, γ) .

Nuclear reactions follow the laws of conservation:

- The total number of nucleons (total A) is conserved.
- The total charge (total Z) is conserved.
- The total energy (including changes in total mass) is conserved.
- The total momentum is conserved.

Cross sections are the measure of the probability that a specific process will occur when an incident radiation interacts with an orbital electron or electrons or with the nucleus of atoms in a target material and can be conceptually considered as the size (area) of the object (atom or nucleus) an incident photon or particle must hit for an interaction of interest to occur. The interaction rate between neutrons and nuclei is determined by microscopic (σ) and macroscopic (Σ) cross-section. Microscopic cross-sections have units of area (cm^2) and macroscopic cross-sections have units of inverse length (cm^{-1}). Microscopic cross-sections are fundamental nuclear data that are functions of energy and depend on the energy structure of the nucleus and are often defined in units of barns (10^{-24} cm^2). The reaction rate (R) is the product of the microscopic cross section with the intensity of the neutron source and the number of target atoms per unit area, given in Eq. (64):

$$R \left[\frac{\text{interactions}}{\text{cm}^2 \cdot \text{s}} \right] = \sigma \left[\text{cm}^2 \right] \cdot I \left[\frac{\#}{\text{cm}^2 \cdot \text{s}} \right] \cdot N_A \left[\frac{\#}{\text{cm}^2} \right] \quad (64)$$

Fig. 19 illustrates the concept of a microscopic cross-section as the interaction probability between a neutron of intensity I and an individual particle or nucleus in a material one layer of atoms thick in dx .

Macroscopic cross-sections are calculated from known compositions and microscopic cross-section. Eq. (65) defines microscopic cross-sections and Eq. (66) illustrates the total microscopic (absorption plus scattering) with a breakdown of components for scattering (elastic and inelastic) and of absorption.

$$\sigma = \frac{R/N_A [\text{number of reactions/nucleus/s}]}{I [\text{number of incident neutrons/cm}^2/\text{s}]} \quad (65)$$

The total cross section (σ_t) is the sum of absorption (σ_a) and scattering (σ_s) interactions. Scattering interactions can be further broken down as the sum of elastic and inelastic energy transfer, where elastic scattering behaves like a billiard-ball type collision and inelastic scattering results in compound nucleus formation, $\sigma_m = X(n, n')X$. The absorption cross section is the aggregate sum of fission, radiative capture, etc. from $(n, 2n)$, (n, p) , (n, α) , (n, γ) neutron interactions. The total cross section can therefore be written as:

$$\sigma_t = (\sigma_e + \sigma_{in}) + (\sigma_f + \sigma_\gamma + \sigma_{n\alpha} + \sigma_{np} + \sigma_{n,2n} + \dots) \quad (66)$$

Fission

When a compound nucleus of a fissionable isotope is formed, it may energetically disintegrate into two or more lower-mass elements (fission) or it may transmute to another element or isotope and emit radiation that corresponds to the transition that occurs. Fissionable isotopes are those that may fission when a compound isotope is formed with a neutron of some kinetic energy, whereas fissile isotopes are those that will fission after the formation of a compound nuclide of zero kinetic energy. Examples of practical fissile isotopes include ^{235}U , ^{239}Pu , and ^{233}U . Both ^{238}U and ^{232}Th are examples of useful fissionable isotopes that will fission if the kinetic energy of the neutron that forms the compound is about 1 MeV. The kinetic energy of the recoil fission products, neutrons, neutrinos, and other radiative products may be calculated from binding energy per nucleon versus mass data, as illustrated in Fig. 3. Fig. 20 depicts the fission cross-section for ^{235}U with a strong characteristic resonance structure (Pritychenko et al., 2006). When the kinetic energy of the neutron-nucleus system corresponds to an energy level in the compound nucleus to be formed then the effective size of the compound nucleus may increase by several orders of magnitude.

The energy distribution of neutrons emitted during the fission of ^{235}U is illustrated by Fig. 21 (US Department of Energy, 1993).

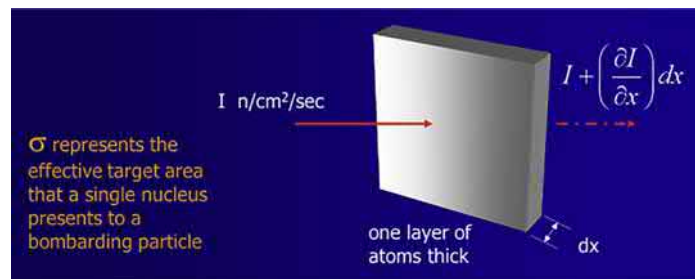


Fig. 19 Microscopic cross-section (σ) depicting the probability of a neutron interacting in an infinitesimal layer one atom thick.

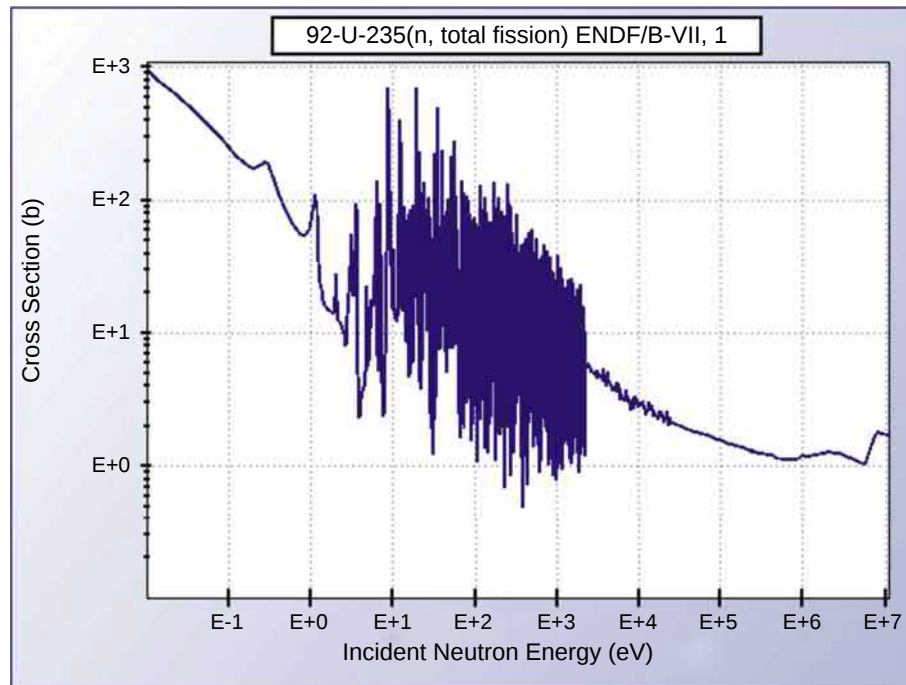


Fig. 20 Fission cross section of ^{235}U . Generated from Pritychenko B, Sonzogni A, Winchell D, Zerkov V, Arcilla R, Burrows T, Dunford C, Herman M, McLane V and Obložinský PJAONE (2006) Nuclear reaction and structure data services of the National Nuclear Data Center. *Annals of Nuclear Energy* 33: 390–399.

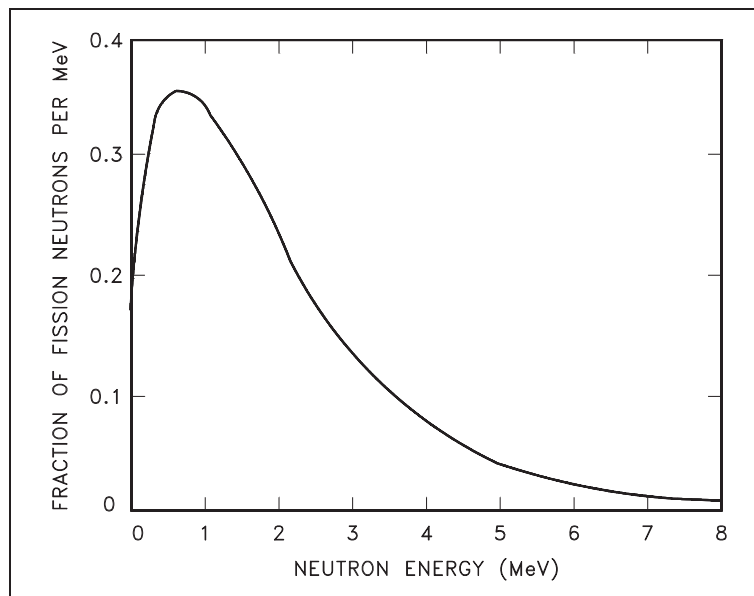


Fig. 21 Fission energy spectrum of neutrons emitted during fission of ^{235}U . The most probable energy is at ~ 1 MeV and the average energy is ~ 2 MeV. Reproduced from US Department of Energy (1993) *DOE Fundamentals Handbook-Nuclear Physics and Reactor Theory (DOE-HDBK-1019/1-93)*. US Department of Energy.

Neutron reactions in tissue

A number of neutron reactions are important in various aspects of health physics and radiation protection. This is governed by capture interactions with large neutron cross sections, often accompanied by a photon. The magnitude of the cross section, efficiency of energy transfer, and the abundance of hydrogen in soft tissue, neutron-proton (n,p) scattering is the dominant method

by which fast neutrons deliver dose to the tissue. Over 85% of “first collision” dose in soft tissue, comprising hydrogen, carbon, oxygen and nitrogen, originate from (n, p) scattering for neutrons < 10 MeV.

Slow and thermal neutrons interacting through capture interactions are dominated in the body by two key interactions. First, is the (n, γ) reaction in the body of $^1\text{H}(n, \gamma)^2\text{H}$, which releases a highly energetic gamma-ray with $Q = 2.22$ MeV. The energetic gamma will not be a localized dosimetric risk in a small tissue sample, but will rather pose a notable hazard over larger regions, such as that of the whole body. Comparatively, the second reaction of interest is the $^{14}\text{N}(n, p)^{14}\text{C}$ interaction, with a $Q = 0.626$ MeV. This interaction poses more of a localized risk, as the ^{14}C and proton deposit their energy locally at the site where the neutron is absorbed. Neutron capture by both ^1H and ^{14}N represent the predominant mechanism by which thermal neutrons deliver significant dose to soft tissue. Fast neutrons interact predominantly via elastic scattering in H, O, C, and N in soft tissue.

Charged particle interactions

Nuclei have diameters that range from about 1.6 to 15 fm (10^{-15} m), and the distance between them in solid materials is about an Angstrom (10^{-10} m); thus, this distance is about 100,000 times greater than the diameter of an alpha particle. The inter-nuclear space is populated with negatively charged electrons that govern the energy loss rate of charged particles to the medium through which it passes. This energy loss rate depends on charge and momentum of the charged particle and on the electron density of the medium through which it is passing. Two-particle kinematic models that utilize conservation of energy and momentum show that the energy transfer from a proton with an electron in a single collision is a fraction of a percent of the initial energy of the proton when its energy is less than 10 MeV (Turner, 2008).

Heavy charged particles

Positively charged particles transfer energy to electrons through several mechanisms. The dominate mechanism through electrostatic attraction of electrons that results in ionization of the nucleus and transfer of kinetic energy to electrons.

The rate of energy loss is referred to as stopping power ($-dE/dx$), and an example of this function for a proton is shown in Fig. 22 (Provision Health Care, 2020). You may notice that the energy loss rate is close to a linear function for about one-half of its total track length. When the range of the is about three-fourth of its way to the end of its track it forms a Bragg peak. The Bragg peak is the peak on a curve of the energy loss of ionizing radiation during a charged particle’s travel through matter. For protons, alpha particles, and other ions, the peak is quite pronounced and occurs close to the terminal point of the path of travel. The prominent feature of the Bragg peak makes proton therapy appealing due to its localized deposition of energy in target cancer cells with no exit dose to healthy tissue.

Niels Bohr (1885–1962) developed a formula for stopping power in 1913, without the benefit of quantum mechanics, that gave results within a few percent. Later, Bethe used relativistic quantum mechanics to obtain a more accurate formula for stopping power that depends on the following parameters:

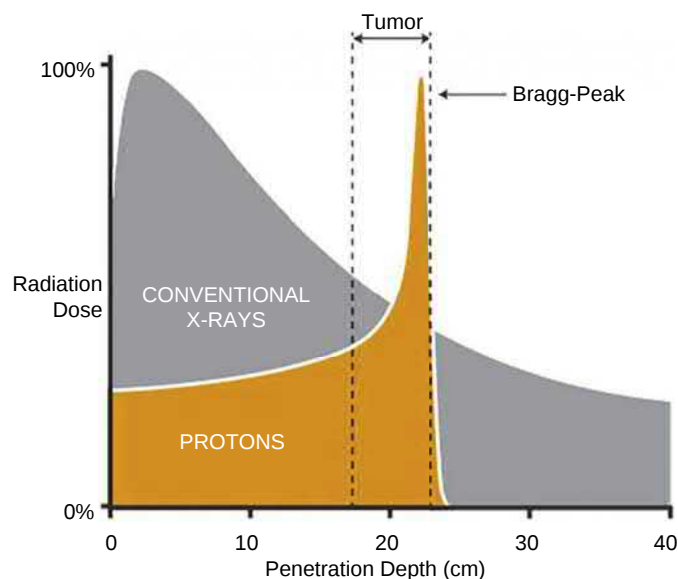


Fig. 22 Comparison of radiation dose with depth of penetration in tissue of X-rays and protons. The prominent feature of the Bragg peak makes proton therapy appealing due to its localized deposition of energy in target cancer cells with no exit dose to health tissue. Reproduced from Provision Health Care (2020) *What Is Proton Therapy?* Available: <https://provisionhealthcare.com/about-proton-therapy/advantages-of-proton/>

- Atomic number of the heavy charged particle;
- Magnitude of the electron charge;
- Number of electrons per unit volume in the medium;
- Electron rest mass;
- Speed of light in a vacuum;
- The speed of the particle relative to light; and
- Mean excitation energy of the medium.

Stopping power depends only on the charge and velocity of the heavy charged particle and on properties through which the particle is passing. Bethe derived the following relationship for slowing down of charged particles (MeV/cm), using relativistic quantum mechanics:

$$-\frac{dE}{dx} = \frac{4\pi k_0^2 z^2 e^4 n}{mc^2 \beta^2} \left[\ln \frac{2mc^2 \beta^2}{I(1 - \beta^2)} - \beta^2 \right] \quad (67)$$

where $k_0 = 8.99 \times 10^9 \text{ N m}^2 \text{ C}^{-2}$, (the Boltzmann constant); z = atomic number of the heavy particle; e = magnitude of the electron charge; n = number of electrons per unit volume in the medium; m = electron rest mass; c = speed of light in vacuum; $\beta = v/c$ = speed of the particle relative to c ; I = mean excitation energy of the medium.

Range and slowing down rate

Heavy positively charged particles lose energy primarily through Coulombic interactions with electrons in the medium through which they are traveling. Loss through collisions with relatively very light electrons may be neglected. Alpha particles undergo continuous slowing down and consequently well-defined relationships exist between range and energy. The range of a heavy charged particle is obtained by integrating the stopping power from zero to the initial energy of the particle.

$$R(T) = \int_0^T \left(-\frac{dE}{dx} \right)^{-1} dE. \quad (68)$$

From Fig. 23, the range of a 10 MeV proton in water (density of 1.0 g/cm^3) is about 0.1 cm and for a 10 MeV alpha it is about 0.01 cm.

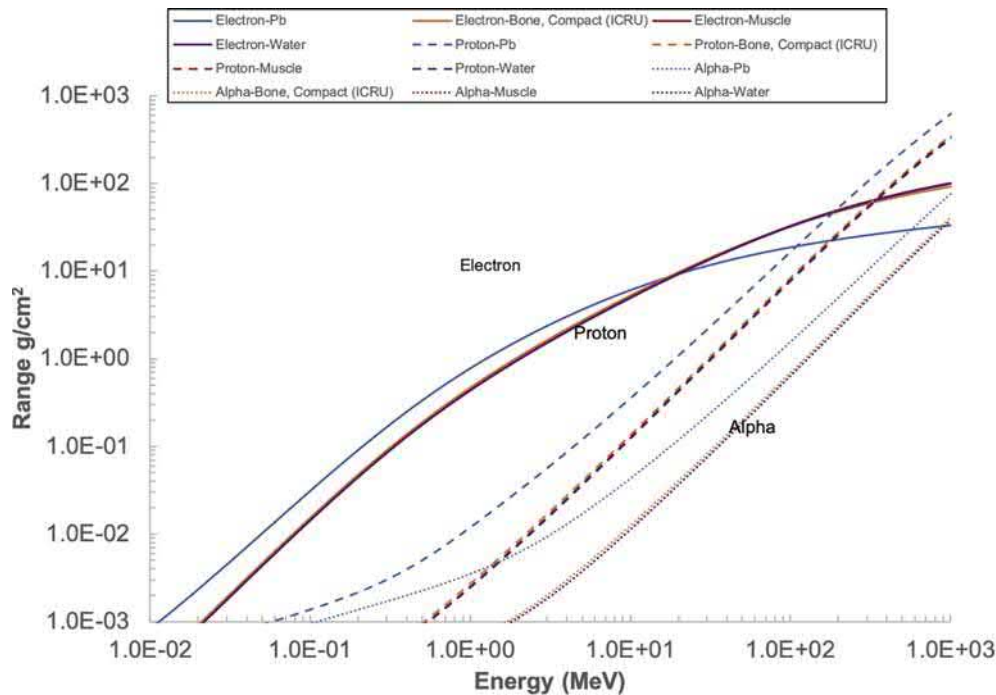


Fig. 23 Ranges of protons, alpha particles, and electrons in water, muscle, bone and lead, expressed in g cm^{-2} . Generated from <https://physics.nist.gov/PhysRefData/Star/Text/intro.html>.

Given the initial velocity, mass, and stopping power of a charged particle traveling through a given medium, it is relatively straightforward to calculate the time required to stop. For a proton of 1 MeV in water this time is 2.7×10^{-12} s (Turner, 2008), and the time for a 10 MeV proton it is 5×10^{-11} s. Chemical reactions of free radicals generated by proton or alpha particle irradiation are completed in less than a microsecond. This implies that diffusion times of free radicals is several orders of magnitude longer than the start to stop time of the irradiating particles.

The range of a 1 MeV alpha is about 8 μm (estimated from Fig. 23) and on the order of 100 μm for a 10 MeV alpha. Thus, an alpha of several MeV effectively creates a highly ionized cylindrical path over the distance of about five to 10 cell diameters. If about 2 MeV of energy is deposited over a volume of 100–1000 cells, about 50,000 ionizations could be generated. The spatial distribution would be very non-uniform since energy dissipates as a function of the cube of the radius; thus, an extremely rough estimate of the number of cells that would experience the contact of free radicals from this event could be on the order of 100–1000.

For the case of beta particles, that same amount of energy would be deposited over a much larger volume. As given in Fig. 22, the range of an electron is about 50 times longer than for an alpha particle; consequently, the corresponding volume would be about 100,000 times larger.

Light charged particles

The physics of slowing down electrons is different from that of positively charged heavy particles in several ways. One is that the mass of the irradiating beta particle (electron or positron) is the same as the electrons that fill the volume between nuclei (in a solid medium), and another is that the charges are the same. Another difference is that electrons lose energy by radiation and by collisions. Thus, the total stopping power for electrons is the sum of the collision stopping power and the radiative stopping power.

$$\left(-\frac{dE}{dx}\right)_{\text{tot}}^{\pm} = \left(-\frac{dE}{dx}\right)_{\text{col}}^{\pm} + \left(-\frac{dE}{dx}\right)_{\text{rad}}^{\pm} \quad (69)$$

The analytical expressions for these terms are complex and are not included here, but the interested reader may understand the details by consulting Attix (2008), Turner (2008), or Veinot (2019). In summary, the collisional component is due to “knock-on” collisions from incident electrons interacting with electrons in the target medium. The radiative component stems from the deceleration of the electron as it deposits its energy in the target medium, resulting in the release of photon energy known as bremsstrahlung. Bremsstrahlung (translated from German as “breaking radiation” is produced in a continuous energy spectrum by the deceleration of charged particles. Bremsstrahlung becomes a more pronounced feature of electron interactions in matter for both high-Z target materials, as well as high-energy electron sources. Bremsstrahlung is emitted as a characteristic spectrum of the source electron energy and target medium and becomes a radiation protection consideration with energetic beta/electron sources and particle accelerator shielding.

Beta particles do not follow a continuous slowing down model very well since electrons can lose large fractions of their energy in a single collision. However, the continuous slowing down model yields sufficiently accurate results that are very useful.

Restricted stopping power and linear energy transfer

When charged particles lose energy through Columbic interactions, they create ionizations and may transfer sufficient energy to an electron that travel much farther from the path of the charged particle than the length of the path of the charged particle that generated them. These electrons are referred to as delta rays, and they carry significant energy away from the path of the charged particle. One consequence of the delta-ray phenomena is that biological damage may occur sufficiently far from the location of the charged particle interaction that it should be considered a separate event. This phenomenon significantly complicates calculations of dose.

In order to account for delta rays and other phenomena that may result in energy of the primary particle being transported away from the site of initial interaction, the concepts of Restricted Stopping Power and Linear Energy Transfer (LET) have been found to be useful for dosimetric calculations. Stopping power is defined as the energy loss by a particle per unit path length of the particle. Restricted stopping power (Turner, 2008) is defined as the linear rate of energy loss due only to collisions in which the energy transfer does not exceed a specified value. It is obtained by integrating the weighted energy loss spectrum from a lower limit to the selected (or “restricted”) upper limit of integration. If the upper limit of integration is extended to infinity the result of this integration is the stopping power.

The concept of LET is defined slightly differently than restricted stopping power, but they may be numerically equivalent. LET was first defined in 1962 by the International Commission on Radiation Units and Measurements (ICRU) as the “average energy locally imparted”. The words “locally imparted” were not precisely defined so LET has not always been used with exactly the same meaning (Turner, 2008). In 1970, the ICRU defined LET_{Δ} as the restricted stopping power where the upper limit of integration is set to Δ (ICRU, 2011). Fig. 24 illustrates the Bragg peak associated an LET plot (Sgouros et al., 2010). It has essentially the same shape as the plot in Fig. 22 for stopping power.

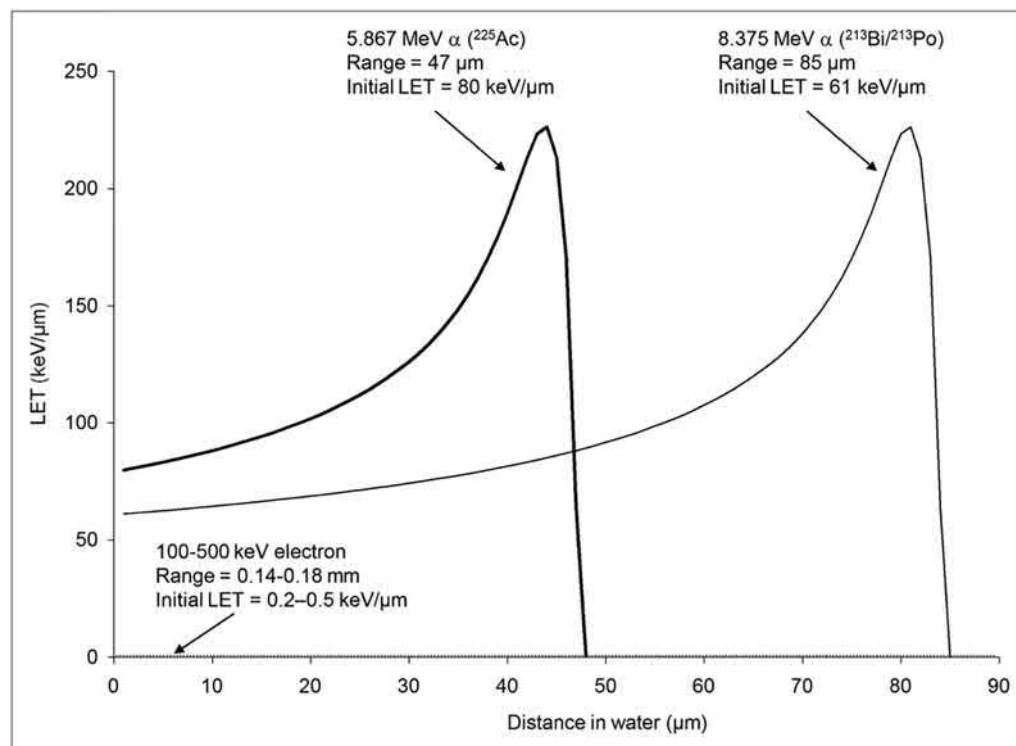


Fig. 24 LET as a function of penetration depth in water for alpha particles of two energies. Alpha particles with lower initial kinetic energy travel a shorter distance before depositing energy in their Bragg peaks, resulting in higher LET. At the bottom of the plot, electrons with initial kinetic energy of 100–500 keV are shown for comparison. Reproduced from Sgouros G, Roeske JC, Mcdevitt MR, Palm S, Allen BJ, Fisher DR, Brill AB, Song H, Howell RW and Akabani G (2010) MIRD Pamphlet No. 22 (abridged): Radiobiology and dosimetry of α -particle emitters for targeted radionuclide therapy. *Journal of Nuclear Medicine* 51: 311–328.

Conclusion

Interactions of radiation with matter in health physics and radiation protection all stem from fundamental physical interactions at the atomic and subatomic levels for target media of interest including tissue, shielding materials, and radiation detectors. By understanding the fundamental mechanisms of energy transfer via radioactive interactions yielding energetic particles (protons, alpha, beta, neutron, etc.) and waves (X-rays, gamma rays) relevant to health physics, we can use these fundamental quantities to measure radiation fields and correlate these quantities with regulatory standards of these interactions.

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Natural and Man-Made Sources of Radiation

Laurence Miller, A. Iulian Apostoaei, Michael Howard, and Vikram Singh, Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

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Naturally occurring radioactive isotopes

There is a general scientific consensus that all elements were created by the “Big Bang,” followed by subsequent transmutations that took place in Supernova and in other stellar events. Cosmic clouds of these isotopes presumably condensed into stars, planets, and other objects that are found in our planetary system. Given that the Earth is estimated to be about 4.5 billion years old, only radioactive isotopes that originated with the Big Bang, and have half-lives longer than about 100 million years, can be readily measured with current technology. These are referred to as primordial radionuclides. Radioactive isotopes with relatively long (millions of years) and short (days to thousands of years) half-lives are continually produced by decay of primordial isotopes and by transmutations associated with cosmic radiation. Radioactive isotopes that are continually produced by the interaction of cosmic radiation with the atmosphere and other materials are referred to as cosmogenic.

Naturally Occurring Radioactive Materials (NORM), such as uranium, thorium, radium, radon, carbon-14, beryllium-7, potassium-40, fission products (from spontaneous fission of uranium in NROM), etc. are encountered by humans every day (epa-2). These materials, in conjunction with cosmic radiation, result in doses to humans that range from about 1.5 to 3.5 mSv/year (150 to 350 mrem/year). Surprisingly, the following locations: Guarapari, Brazil; Kerala, India; Ramsar, Iran; and Yangjiang, China, have natural background radiation that are relative high (> 5000 mrem/year or 50 mSv/year) (Sohrabi, 1997, 1998). The activity of 1% uranium ore is 33,000 pCi/g, in soil it ranges from about 0.5 to 5 pCi/g, and the activity in barrels of oil radium concentration has been reported to be 400,000 pCi/g.

Primordial

Primordial and other terrestrial radionuclides contribute about 20 mrem/year from direct external radiation and about 220 mrem/year to lungs due to alpha particles emitted by the decay products of radon (Fig. 1). Because radon is a noble gas, it does not attach to the lungs when inhaled so direct dose contributions are negligible. The mechanism for exposure is through its decay products [primarily Po isotopes]. They attach to dust particles in the air which may be inhaled and remain embedded within the lung for a time that depends on their solubility. Emission of alpha particles by radon decay products, namely polonium, cause essentially all of the dose to lungs.

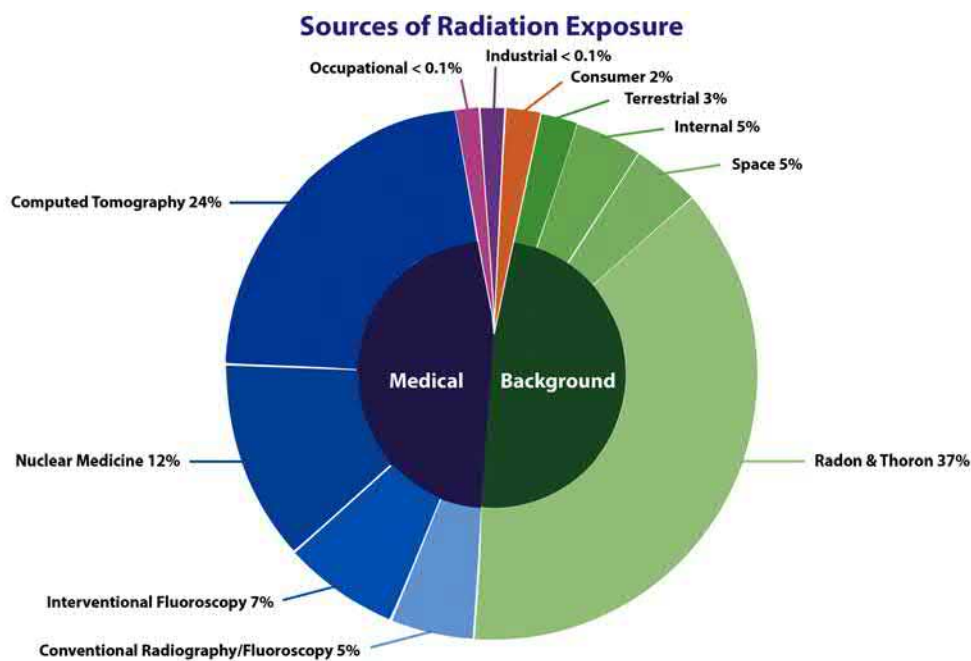
Illustrations of primordial isotope decay chains are shown in Fig. 2, and lists of decay modes are shown in Tables 1 through 3. Two of these decay chains, U-238 and Th-232, dominate the natural background source term, but the contribution of the U-235 decay chain (which originally started with Pu-239) is also relevant even though its natural abundance is only 0.7% of natural uranium. You may note from Fig. 2 that there are three Po isotopes in the U-238 decay chain, two in Th-232, and one in U-235. Table 1 lists two Po isotopes in the Pu-239/U-235 decay chain, but it has only a 1.78 ms half-life so it is neglected in Fig. 2.

The concentration of U-235 in uranium is only 0.7%, so its contribution to radiation exposure is small relative to that of U-238. Radioisotopes in ore that have not been disturbed for several half-lives of the longest-lived isotope all have the same activity (product of its number density times its decay constant). In particular, the radioactivity of U-238, U-235 and Th-232, decay chain products are effectively in equilibrium after several half-lives of the longest-lived daughter.

Cosmogenic

Interactions of cosmic rays with the atmosphere, earth, sea, etc., result in the production of several radionuclides that are readily measurable. Those of most significance include: H-3, Be-7, C-14, and Na-22. Others are produced by secondary neutrons and by high energy collisions with galactic radiation. These radionuclides contribute about 1 mrem/year to the annual equivalent dose. The reactions that produce these isotopes are as follows:

1. Tritium (H-3 or T) is produced primarily by the reactions: $^{14}\text{N}(\text{n},\text{T})^{12}\text{C}$ and $^{16}\text{O}(\text{n},\text{T})^{14}\text{N}$, and it has a half-life of 12.3 years.
2. The reaction $^{14}\text{N}(\text{n},\text{p})^{14}\text{C}$, accounts for most of the C-14 produced by cosmic radiation. C-14 has a half-life of 5730 years.
3. Be-7 is produced by high energy interactions with nitrogen and oxygen and has a half-life of 53.4 days.
4. Na-22 results from neutron interactions with argon and has a half-life of 2.6 years.



Average Annual Radiation Dose											
Sources	Radon & Thoron	Computed Tomography	Nuclear Medicine	Interventional Fluoroscopy	Space	Conventional Radiography/Fluoroscopy	Internal	Terrestrial	Consumer	Occupational	Industrial
Units											
mrem (United States)	228 mrem	147 mrem	77 mrem	43 mrem	33 mrem	33 mrem	29 mrem	21 mrem	13 mrem	0.5 mrem	0.3 mrem
mSv (International)	2.28 mSv	1.47 mSv	0.77 mSv	0.43 mSv	0.33 mSv	0.33 mSv	0.29 mSv	0.21 mSv	0.13 mSv	0.005 mSv	0.003 mSv

(Source: National Council on Radiation Protection & Measurements, Report No. 160)

Fig. 1 Sources of human exposure to radiation as compiled by the US Environmental Protection Agency.

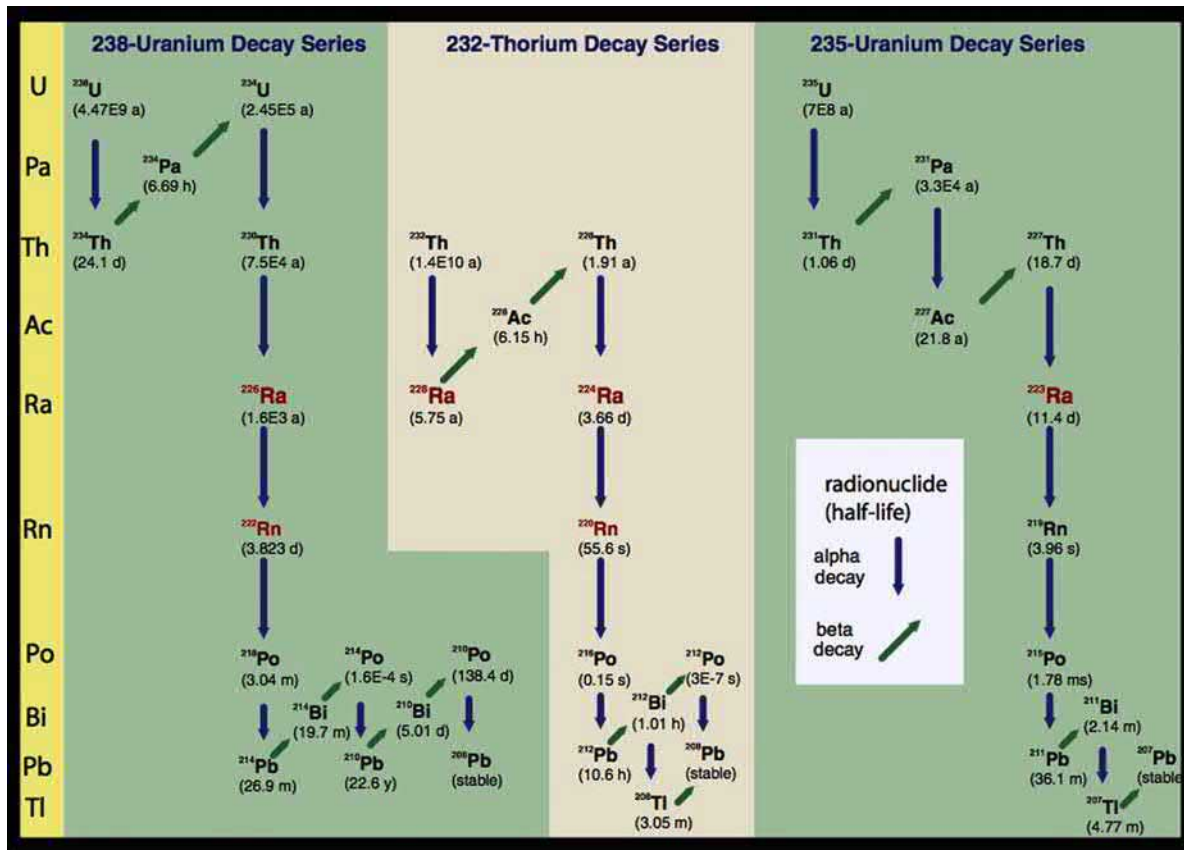


Fig. 2 Decay chains for U-238, Th-232, and U-235.

Table 1 Pu-239 & U-235 decay chain data.

Isotope	Half-life	Decay energies (MeV)		
		α	β	γ
^{239}Pu	2.41×10^4 years	5.16, 5.14, 5.11	—	0.05
^{235}U	7.04×10^8 years	4.40, 4.37	—	0.19, 0.14
^{231}Th	1.06 days	—	0.14, 0.31, 0.22	0.03, 0.08
^{231}Pa	3.28×10^4 years	5.01, 4.95, 5.03	—	—
^{227}Ac	21.8 years	4.95	0.05	—
^{227}Th	18.7 days	6.04, 5.98, 5.76	—	0.24
^{223}Fr	21.8 month	—	1.15	0.05, 0.08
^{223}Ra	11.4 days	5.72, 5.61	—	0.27, 0.15
^{219}Rn	3.96 s	6.82, 6.55	—	—
^{215}Po	1.78 ms	7.39	—	—
^{211}Pb	36.1 month	—	1.39	—
^{211}Bi	2.14 month	6.62, 6.28	0.60	0.35
^{211}Po	0.52 s	7.45	—	—
^{207}Tl	4.77 month	—	1.44	—
^{207}Pb	Stable	—	—	—

Technology enhanced naturally occurring radioactive material

The concentration of uranium in commonly encountered soil is about 3 ppm (2 pCi/g or 0.07 Bq/g). However, uranium ore may contain uranium from a fraction of a percent to over 10%. In seawater, uranium has a concentration of about three parts per billion (3 ppb) and generally about 1 ppb in crude oil.

Processes that concentrate NORM to produce Technology Enhanced Naturally Occurring Radioactive Material (TENORM), which are of interest in radiation protection include the following:

1. Mining
2. Hard Rock Metal Mining
3. Rare Earths Mining Wastes
4. Uranium Mining Wastes
5. Copper Mining and Production Wastes
6. Bauxite and Alumina Production Wastes
7. Energy production
8. Oil and Gas Production Wastes
9. Coal Combustion Residuals
10. Water treatment
11. Drinking Water Treatment Residuals
12. Wastewater Treatment Residuals
13. Consumer products
14. Fertilizer and Fertilizer Production Wastes
15. Cigarettes
16. Building Materials
17. Granite Countertops

Each of these processes concentrate NORM in different ways. For example, cigarettes use tobacco plants which accumulate uranium from soil, waste water treatment plants facilitate precipitation of radionuclides in sewage, NORM in crude oil plates out on pipeline and tanker car surfaces, and crushing of ore during milling releases decay products of uranium and thorium which can easily leach into the environment. The concentration of NORM and the pathways for human exposure vary for each of the items above. A detailed 300-page report ([epa-2402-R-08, 2008](#)) provides a thorough assessment of concentration of NORM during mining operations, as well as its health effects.

The Environmental Protection Agency (EPA) has overall regulatory responsibility for TENORM, and it is investigating health-related concerns in following ways:

1. Characterization of residual wastes,
2. Evaluation of potential exposures,
3. Identification of origin of materials,
4. Collaboration with industry, and
5. Collaboration with other regulatory agencies.

Reactor- and accelerator- produced isotopes

Radioisotopes find various uses in modern life, including medicine, industry, agriculture, and research. Although some of these isotopes may occur naturally through the processes described earlier, their concentrations are usually too dilute for use. Hence, they are produced artificially, primarily in research reactors and accelerators. The choice of using a reactor or accelerator in the production of isotopes for medical, industrial, or research use is due in large part to the type of radiation desired. For example, for use in Positron Emission Tomography, isotopes of interest must decay by emitting a positron. Thus, they need to be proton rich, and this is accomplished by irradiating the target material with high-energy protons using an accelerator to cause proton-induced nuclear reactions. On the other hand, if a beta emitter is preferred, isotopes that are neutron rich need to be produced since they decay by beta emission to reach the line of stability. These can be formed as fission products or through neutron irradiation of targets in a reactor. Isotopes that emit alpha particles generally have an atomic number greater than 80. They may be generated in reactors by transmutation of elements or in nature as decay products of NORM. In either case, they are obtained by chemical extraction from source material.

Reactor-produced isotopes

More than 80% of medical imaging for diagnosis of diseases such as cancer makes use of the isotope technetium-99m ($Tc-99m$), which is the decay product of the radioisotope molybdenum-99 ($Mo-99$) which is produced in research reactors.

$Mo-99$ makes up approximately 6.1% of the fission product yield from fission of $U-235$. It has a half-life of 66 h, which requires continuous production and delivery to hospitals. As a result, $Mo-99$ is produced in research reactors through neutron irradiation of high-enriched uranium targets. The targets are exposed to neutrons for several hours followed by chemical processing to isolate the $Mo-99$ produced. In a typical generator used in a medical setting, the $Mo-99$ is preloaded by adsorption onto the surface of aluminum oxide columns. When a saline solution is passed over the column, any produced $Tc-99m$ is preferentially washed away and thus extracted.

As of 2020, the global demand for Mo-99 is provided by the following research reactors:

1. BR-2 reactor at the SCK-CEN in Belgium
2. HFR at Petten in the Netherlands
3. LVR-15 reactor in the Czech Republic
4. MARIA reactor in Poland
5. OPAL reactor in Australia
6. SAFARI-1 reactor in South Africa

Additional information on several of these reactors is available in the following chapters of this encyclopedia:

01006. BR-2 (100 MW) by *Steven Van Dyck*,

01008. High Flux Isotope Reactor (HFIR) by *Tim Powers, David Chandler and Christopher Bryan*,

01014. MARIA (30 MW Poland) by *Michal Gryzinski, and*.

01015. OPAL (20 MW Australia) *David Vittorio*.

Radioisotopes such as I-131 and Xe-133 that are also used for some medical imaging are also important fission products and may be produced at the same facilities.

Some isotopes, such as Co-60, are produced through dedicated neutron irradiation in a reactor. Cobalt-60 (Co-60) is a strong gamma emitter used mainly for sterilization of medical devices and in industrial radiography. It is produced through neutron capture in nuclear reactors. Specially prepared targets of cobalt-59 can be introduced into a flux trap through a dedicated chamber, or in some cases dedicated rods of Co-59 are placed in certain fuel assemblies. The cobalt-59 absorbs a neutron and converts into Co-60 which is processed in hot cells and sealed in tungsten lined containers filled with an inert gas before being shipped off for use.

In the United States, the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) provides one of the world's highest steady-state thermal neutron fluxes (up to 3×10^{15} neutrons $\text{s}^{-1} \text{cm}^{-2}$). It is a light water-cooled reactor that uses high-enriched uranium as fuel. HFIR contains more than 40 target locations within the core and reflector. Products made at this facility include Ni-63, W-188, Re-186, Cf-252, and Ir-192.

A few additional examples of useful reactor-produced radionuclides include the following applications:

Ni-63: Used for detection of explosives in homeland defense initiatives and as a power source for remote instrumentation.

Ga-68: Used as a positron source to calibrate Positron Emission Tomography (PET) scanners that are widely used in medical research and cancer imaging.

W-188: This is the parent isotope of rhenium-188 used to prevent the re-closure (restenosis) of coronary arteries following heart surgery. Also used as a monoclonal antibody label and to relieve bone cancer.

Ir-192: Used in gamma radiography to image the structural integrity of aircraft, pipe welds, ships, bridges, and other structures.

Ac-225: Extracted from Th-229, which is extracted from a unique inventory of U-233 at ORNL. Both Ac-225, and its decay daughter Bi-213, are alpha emitters and have shown considerable promise in the treatment of cancer.

Pu-238: Owing to the high density of plutonium ($< 19 \text{ g/cm}^3$) and half-life of 87.7 years, the strong alpha-emitter Pu-238 is used as a source of heat and energy in radioisotope thermoelectric generator units deployed in rovers and satellites for space missions.

Cf-252: Used to treat cervical cancer. It also undergoes spontaneous fission and is a source of neutrons to start up reactors. Also used to gauge the moisture content of soil at road and building construction sites.

Accelerator produced isotopes

The efficacy of producing specific isotopes may favor use of a reactor or an accelerator. Generally a higher specific activity can be obtained by using charged particles (generated by accelerators) rather than neutrons (generated by reactors) since multiple isotopes of a particular element will be generated by neutron irradiation but not when irradiating with charged particles (usually protons). Other factors that may influence decisions for using neutrons versus protons include the following:

1. Chemical processing requirements following irradiation,
2. Half-life of the isotope of interest,
3. Transportation logistics,
4. Cost of production, and
5. Specific activity.

Discussion of the advantages or disadvantages using neutron, protons, or other charged particles change because of technological capabilities over time and due to regulatory considerations. Some useful accelerator-produced isotopes include the following:

Cd-109: Used to induce X-ray emission in coal and other materials such as metal alloys for quantitative and qualitative analysis. Characteristic X-rays (induced by photons from Cd-109) from sulfur, for example, provide information on sulfur content of coal and lead to cleaner burning coals.

Si-32: Used as a radioactive tracer to measure silicon uptake by oceanic phytoplankton and thereby monitor possible trends in global warming.

Na-22: Used medically in neurological research and commercially as a radiation source.

Sr-82: As the parent of rubidium-82, it is often sold as a generator. Rubidium is a chemical analog of potassium which makes it ideal for cardiac imaging.

Several high-energy accelerator facilities and their characteristics are identified below.

Los Alamos Neutron Science Center (LANSCE)

The Los Alamos Neutron Science Center (LANSCE) at LANL is a half-mile-long accelerator that delivers 800 MeV protons, at up to 1 mA of H⁺ and 100 microamperes of H⁻.

The unique characteristics of the LANSCE accelerator include a high-energy, high-beam current that allows production of higher quality radioisotopes as well as radioisotopes that cannot be produced in other facilities. The site's three major products are Ge-68, Sr-82, and Na-22.

Isotope production facility (IPF)

The isotope production facility (IPF) is a 100 MeV proton bombardment station located at LANSCE. The facility can produce high current beams for large scale production of radioisotopes as well as delicate low current beams for experiments. IPF also maintains a robust hot cell facility for processing and handling of targets used in radioisotope production. Thus, the isotope program at LANSCE produces radioisotopes that are in short supply utilizing the unique capabilities of the facility.

The program produces Sr-82 generators for critical cardiac imaging and Ge-68 used for diagnostic procedures. The IPF is also gearing up to increase the availability of Ac-225 for targeted alpha therapy. Other isotopes produced include high purity Na-22 for positron emission tomography, As-73 for environmental research, and ³²Si as a tracer for groundwater discharge.

Brookhaven Linac Isotope Producer (BLIP)

The Brookhaven Linac Isotope Producer (BLIP) at BNL utilizes the excess beam of a linear accelerator that injects 200 MeV protons into the 33 G electron-Volt Alternating Gradient Synchrotron. The BLIP facility produces such radioisotopes as strontium-82, germanium-68, Tc-95m, Cu-67, and Be-7.

Research, medical, and industrial applications

Radioactive materials often have sufficient activity that remote processing and handling is required to avoid excessive exposure to personnel. The Department of Energy (DOE), institutions, and other industries have developed handling, processing, and measurement facilities and capabilities that make stable and radioactive isotopes available for use.

Isotope Programs

Hot cell facilities: A hot cell is a closed work area in which radioactive materials may be manipulated without exposing the operator to gamma radiation. Some cells are dedicated to the production of a single radioisotope in order to minimize contamination. Other cells are used to process a wide range of isotopes while still others are used for storage and transfer functions. They are an integral part of radioactive isotope production and their care and maintenance are high priorities. Isotope Programs make use of hot cell facilities at Oak Ridge National Laboratory (ORNL), Idaho National Laboratory (INL), Los Alamos National Laboratory (LANL), and Brookhaven National Laboratory (BNL).

Byproducts and stockpiled materials: The program obtains isotopes from the processing of byproducts and stored nuclear materials from other DOE programs. For example, Pacific Northwest National Laboratory (PNNL) has produced radio-chemically pure Y-90 by processing Sr-90, a byproduct of nuclear fission reactions. Researchers throughout the United States are now using Y-90 in treating Hodgkin's disease and other types of cancers. PNNL has a large stock of partially purified Sr-90 and some which have been completely purified.

Highly fissile U-233 has been processed by DOE to extract Th-229, the parent radioisotope of Ac-225 and other alpha emitters that show promise as cancer therapeutics. Other stockpiled materials that could be processed for sale include Cs-137, Ac-227, U-232, C-241, Am-243, Cm-244, and a small amount of Ra-226.

Calibration, research and radiography

Effective radiation protection makes use of various instruments that detect and measure the radiation present in the environment. These devices come in different sizes, with some being portable. Regulations require that such radiation detection instruments must be routinely calibrated against known radiation sources. For example, all portable count rate meters (commonly known as "Geiger" counters) and exposure rate meters (commonly referred to as ion chambers) require annual calibration. Co-60 and Cs-137 are generally used for calibration given their relatively long half-lives (few years) and the high energy gamma rays that they produce. Disposal of such sources could pose a health risk. Uncontrolled disposal of Co-60 with scrap metal has been responsible for radioactivity found in various steel and iron-based products (NRC IN-83-16, 1983). In one instance, the company Petco had to recall

several models of steel pet food bowls in August 2012 after it was determined that they were emitting low levels of radiation. The source of the radiation was determined to be Co-60 that had contaminated the steel.

In another incident in Ukraine, a Cs-137 source used for detector calibration was lost in a quarry. Eventually, the gravel from the quarry was used in the construction of an apartment building. Several residents had been exposed to high levels of radiation which tragically led to six deaths.

Research activities in various fields make use of radioactive materials. For example, researchers in the field of radiopharmaceuticals use radioactive materials as “tracers” to develop new medicines and test their effectiveness. To do this, some researchers use specialized detectors called “tracers” that track how material travels through a person or animal. In a radioactive tracer, the researcher replaces some atoms of the medicinal compound with a radionuclide. The radiation from the isotope can be used to study the biokinetic chemical interactions that the medicine undergoes within the body. Positron emitters such as carbon-11, nitrogen-13, and fluorine-18 are generally used as tracers in radiopharmaceutical research. Agricultural researchers use similar type of tracers to track the movement of certain materials through plants. Tritium (H-3) and C-14 are commonly used to study metabolism.

An imaging technique called radiography makes use of X-rays, gamma rays, and other radiation to image the internals of an object. Common radiographical techniques include X-ray imaging and computed tomography (CT) scanning in medicine. In industrial application, radiography is used as a method of non-destructive examination to verify the integrity of the product. Commonly used sources include Co-60, Sr-90, K-85, Cs-137, Ir-192, and Yb-169.

Radioactive sources also find uses in hydraulic fracturing (fracking) to determine injection profile and location of created fractures. Tracers with different half-lives are employed in the different stages. According to the NRC, common tracers include Sc-46, Br-82, Sb-124, I-125, I-131, and Ir-192.

Nuclear fuel cycle

Natural uranium may be used as reactor fuel to sustain a chain reaction if it is combined with materials that have very low neutron absorption cross sections and effectively reduce neutron energies to thermal equilibrium with their environment. There is evidence that a sustained chain reaction occurred naturally in Africa about 2×10^9 years ago. Since the half-life of U-235 (7.04×10^8 years) is significantly shorter than that of U-238 (4.48×10^9 years), the natural abundance of U-235 has decreased from about 4% at that time to a value of 0.71% today. Most commercial vendors of nuclear reactors use U-235 as the fissile isotope for sustaining the chain reaction for Light-Water Reactors (LWRs), where the enrichment of U-235 is about 4%–5%. One exception is the CANDU (Canadian Deuterium-Uranium) reactor, which uses water with deuterium (heavy water) in place of light water (ordinary water) to moderate the neutron energy and natural uranium fuel. Other designs have used high purity graphite with natural uranium to obtain a sustained chain reaction.

Uranium is mined and converted into chemical and physical forms that are suitable for production of energy by fission, and these activities require an extensive industrial infrastructure. Variations in reactor types that use different types of fuel (and alternative methods for utilization of energy produced by fission reactions) result in many corresponding variations on the facilities and materials required to support particular methods for utilization of nuclear energy. Thousands of reports and technical papers have been written to evaluate configurations of infrastructure and reactor types that support the sustainable use and safe use of nuclear energy. An illustration of a generic fuel cycle is provided by Fig. 3, and approximate material flows are shown in Fig. 4.

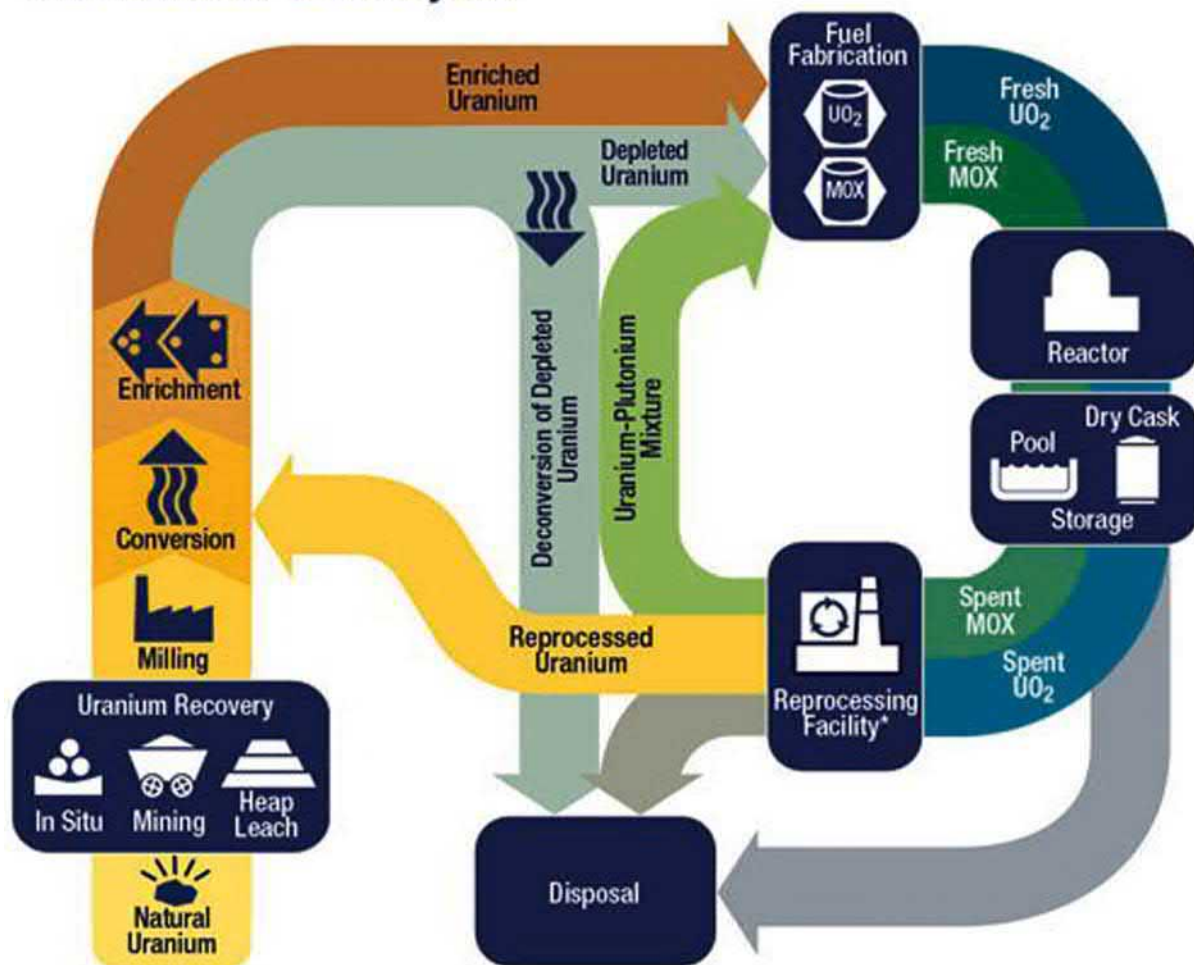
In order to produce electricity by fission of uranium or plutonium, several distinct processes are required. For the once-through fuel cycle practiced today in the United States, the following are required:

1. Mining and milling,
2. Chemical conversion,
3. Enrichment,
4. Fuel fabrication, and
5. Storage or disposal of used nuclear fuel.

For many LWRs, the initial enrichment of U-235 is 5% or less, and the remainder is about 95% U-238. During the 3–5 years of residence time in the reactor several percent of the U-238 is converted to Pu-239. Roughly one-third of the energy produced by the fuel is due to fission of Pu-239, and approximately one-half of the reactor power is being produced by Pu-239 near the end the fuel's residence time in the reactor. In order to utilize the energy potential of used nuclear fuel (UNF) more effectively than with the once-through fuel cycle, fuel reprocessing and recycling is required as is the practice in France. It is important to notice that each of these processes result in exposure to radiation that requires unique protocols for radiation protection. All facilities are operated to comply with established regulations; consequently, radiation exposures are held below levels that are of health-related concerns.

Each activity separately identified in Fig. 5 (Credits-1, n.d.) has the potential for releasing radioactivity of different isotopes and of different chemical composition to the environment. For example, surface and underground mining results in crushing uranium ore, and this, in turn, permits primordial radioactive isotopes to easily migrate. Potential releases from the most significant activities shown in Fig. 5 are described below.

The Nuclear Fuel Cycle



* Reprocessing of spent nuclear fuel, including mixed-oxide (MOX) fuel, is not practiced in the United States.
Note: The NRC has no regulatory role in mining uranium.

As of January 2019



Fig. 3 General illustration of the Nuclear Fuel Cycle.

Mining and milling

Mining of uranium may be accomplished by surface mining, pit mining, and by leaching. Although the radioisotopes in the ore are the same, exposure pathways are very different for each of these methods of mining. For the case of surface mining, radon, and its decay products should be of significantly less concern than for the case of pit mining when personnel are directly exposed to contaminated air. Underground mines, from about 1950 through 1970, were not well ventilated, and as a result, miners received sufficient dose from radon decay products that, based on probability of causation analyses, developed cancer from their work in uranium mines.

Note from **Tables 1–3** that there are many isotopes that are initially in secular equilibrium with U-238, U-235 or Th-232 when the ore is mined. The decay chains listed in these tables collectively produce seven polonium isotopes below Radon. They are alpha emitters and are of primary concern for exposure to radiation from mill tailings. These polonium isotopes, Po-211, 215, 210, 214, 218, 212, and 216, attach to dust particles in air, and about 30% (depending on particle size) of the intake is retained in the lungs after they are inhaled.

During early mining and milling activities (1960–80), about 30 separate mining locations were in operation, and mill tailings were accumulated in piles near the location. Radioactivity in the crushed rock was, at that time, considered to be as hazardous as spent fuel, and this led to the passage of the Uranium Mill Tailings Radiation Control Act (UMTRA) of 1978 (Public Law 95-604). As

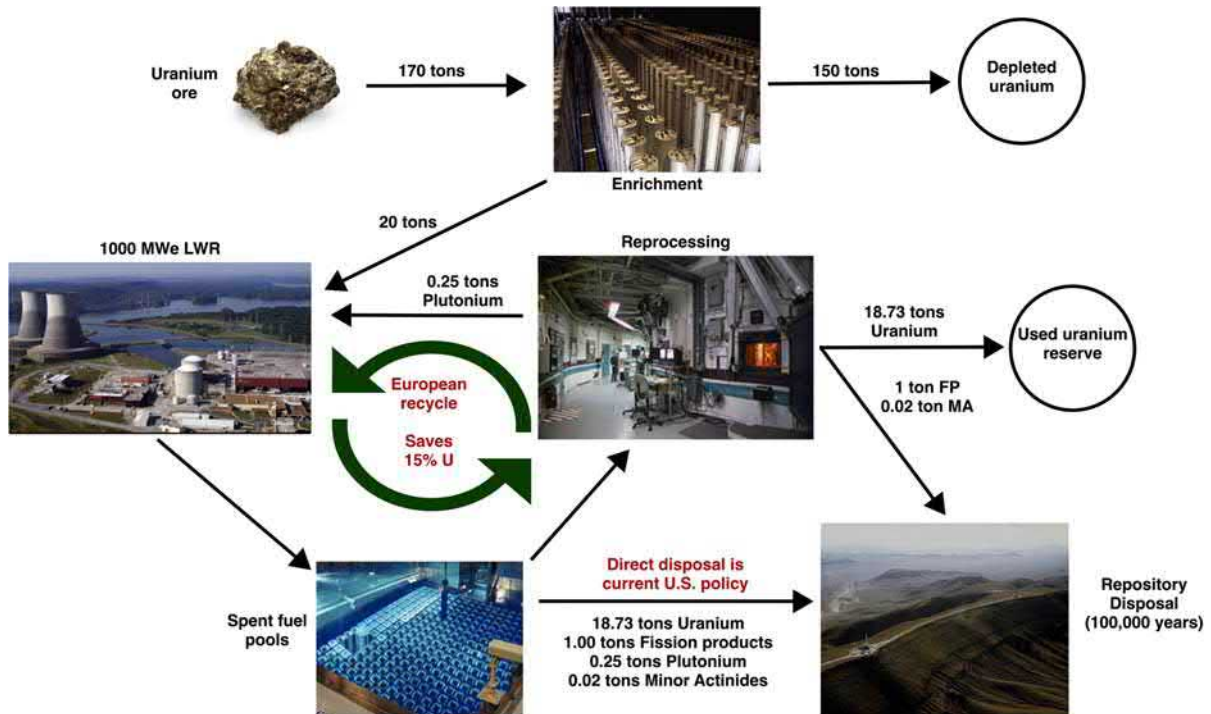


Fig. 4 General illustration of the nuclear fuel cycle with approximate masses.

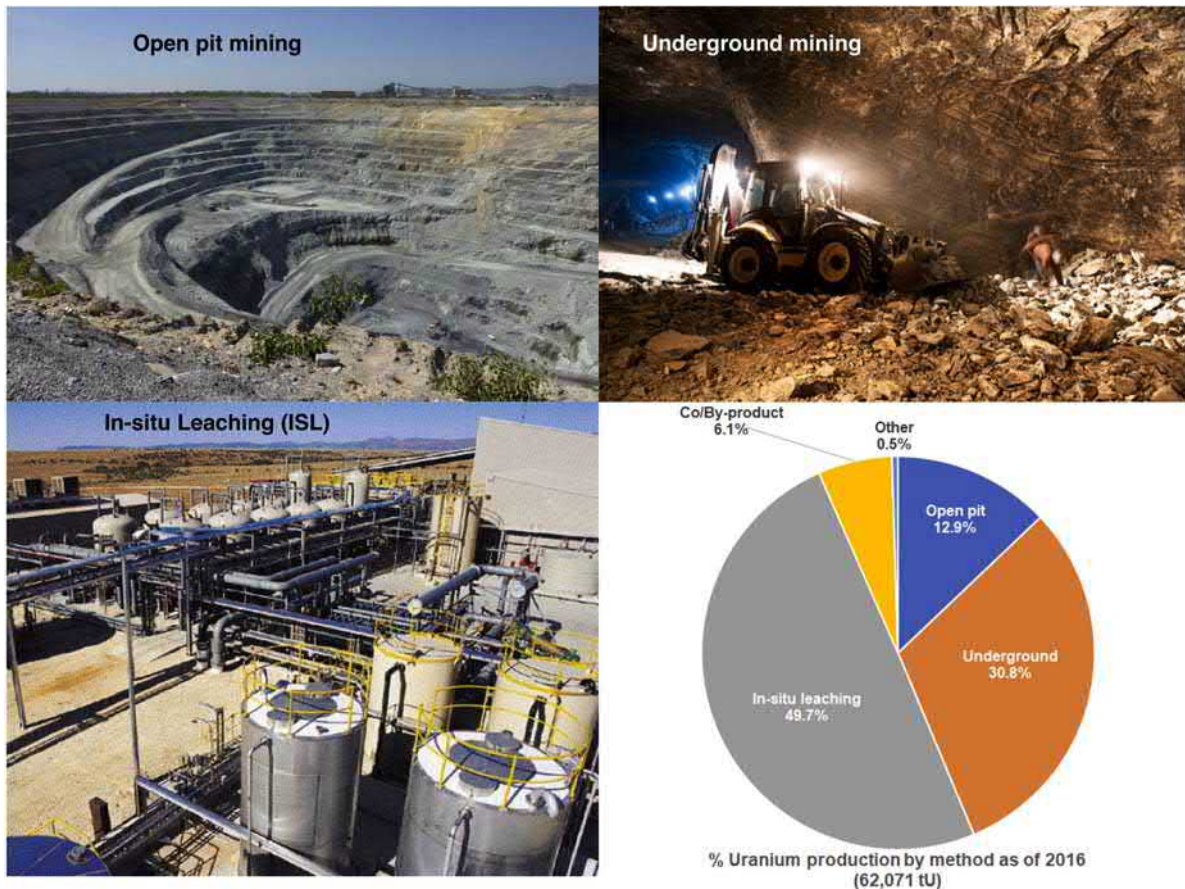


Fig. 5 Illustration of Uranium Mining Methods. Credits-1. (n.d.), Credits: Top Left—Ranger Uranium Mine, Rio Tinto subsidiary Energy Resources of Australia; Top Right—Cameco; Bottom Left - Beverley plant, Heathgate Resources Pvt. Ltd.; Bottom Right—OECD NEA Uranium 2018: Resources, Production, and Demand.

Table 2 Th-232 decay chain data.

Isotope	Half-life	Decay energies (MeV)		
		α	β	γ
²³² Th	1.40×10^{10} years	4.01, 3.95	—	—
²²⁸ Ra	5.76 years	—	0.04, 0.02	—
²²⁸ Ac	6.15 h	—	1.18, 2.09, 1.75	0.91, 0.97, 0.34
²²⁸ Th	1.91 years	5.42, 5.34	—	—
²²⁴ Ra	3.66 days	5.69	—	—
²²⁰ Rn	55.6 s	6.29	—	—
²¹⁶ Po	0.145 s	6.78	—	—
²¹² Pb	10.6 h	—	0.33, 0.57	0.24
²¹² Bi	1.01 h	6.05, 6.09	2.25	—
²¹² Po	298 ns	8.78	—	—
²⁰⁸ Tl	3.05 month	—	1.80, 1.28, 1.52	2.61, 0.58, 0.51, 0.86
²⁰⁸ Pb	Stable	—	—	—

Table 3 U-238 decay chain data.

Isotope	Half-life	Decay energies (MeV)		
		α	β	γ
²³⁸ U	4.49×10^9 years	4.20, 4.15	—	—
²³⁴ Th	24.1 days	—	0.19, 0.10	0.09
²³⁴ Pa	1.17 month	—	2.29	—
²³⁴ U	2.46×10^5 years	4.77, 4.72	—	—
²³⁰ Th	7.77×10^4 years	4.68, 4.62	—	—
²²⁶ Ra	1.60×10^3 years	4.78	—	0.19
²²² Rn	3.82 day	5.49	—	—
²¹⁸ Po	3.08 month	6.00	—	—
²¹⁴ Pb	26.8 month	—	0.66, 0.72	0.35, 0.30
²¹⁴ Bi	19.7 month	—	1.51, 1.00, 3.26	0.61, 1.76, 1.12
²¹⁴ Po	164 μ s	7.69	—	—
²¹⁰ Pb	22.3 years	—	0.02, 0.06	0.05
²¹⁰ Bi	5.01 days	—	1.16	—
²¹⁰ Po	138 days	5.30	—	—
²⁰⁶ Pb	Stable	—	—	—

a result, to date (2020), these locations are remediated as required by regulations and are not considered to present any significant radiological hazards to the public. Various methods for mining and milling are illustrated in Fig. 5. Radiation exposure results from inhalation and direct radiation of NORM.

Conversion

The chemical form of uranium after milling is primarily U_3O_8 and is often referred to as “yellow cake,” which is converted to uranium hexafluoride (UF_6) for enrichment. At this stage of the fuel cycle, the isotopes below U in the decay chain are essentially not present. Thus, uranium is the radioactive isotope of concern for radiological-related health issues during conversion processes.

Enrichment

Light Water Reactors (LWRs) require the concentration of the fissile isotope (U-235) in natural uranium to be increased well above its natural abundance (0.7%) in order for them to sustain chain reactions. Uranium is the radioactive isotope of concern for radiological-related health issues.

Changing the relative concentrations of a particular isotope in an element can be accomplished through the use of selective migration of one isotope versus another. Some of the basic physics differences that may be utilized to accomplish separation include the following:

- atomic absorption spectra,
- momentum,

- mass, and
- molecular size

The process of increasing the fraction of U-235 relative to U-238 in uranium (enrichment) results in radioactive source terms through the following pathways:

- loss of material that becomes airborne,
- loss of material that plates out on surfaces, and
- concentration of trace radioactive elements.

Fuel fabrication

Commercial fuel fabrication plants in operation in the USA are located at Richland, WA (Framatome), Columbia, SC (Westinghouse), and Wilmington, NC (General Electric (GNF-A)). Fuel fabrication plants that support the Department of Energy are located at Lynchburg, VA and at Erwin, TN. These facilities release naturally occurring radioactive material (primordial) in very small quantities; consequently, they pose minimal radiological hazards to the environment and to humans.

Reactors

There are about 400 nuclear fission power reactors in operation (CE 2020) that use water for coolant and for moderating the neutron spectrum to a near thermal equilibrium with the reactor moderator. These reactors are generally classified as Light Water Reactors (LWRs) and are further characterized as Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). Some other reactor designs of significant interest include:

- Heavy-water reactors
- gas for coolant and graphite for moderation,
- sodium cooled,
- molten salt reactors with fuel dissolved in the salt,
- molten salt cooled, and
- lead cooled.

The production of fission products per unit of energy production is within a few percent, regardless of the reactor type. In particular, they produce about one ton of fission products per giga-watt-year of energy production. The thermal spectrum, low enriched uranium fueled, reactors produce about 200 kg of Pu-239 (and other transuranic isotopes) and about 1000 kg of fission products per giga-watt-year of energy production. However, the mass of transuranic isotopes produced varies significantly, depending on the reactor neutron spectrum, fuel cycle type and enrichment of the fissile isotope. The transuranic isotopes typically have half-lives of thousands of years and therefore present the primary concern for radiation exposure many generations in the future.

During normal operation commercial LWRs release radioactivity in gaseous, liquid, and solid forms. The United States Nuclear Regulatory Commission (USNRC) is responsible for assuring compliance with protocols defined in the licensee's permission to operate these facilities. Measurements are required at the points of release, and dosimetry external to these plants is periodically monitored. Shipments of radioactive materials to offsite locations is regulated by the Department of Transportation, the Environmental Protection Agency, and by Waste Acceptance Criteria (WAC) established by the receiving organization.

During the past 50 years of LWR operation the activity of radioactive waste produced per unit of energy generated (excluding fission products and transmuted isotopes) has decreased by about a factor of 100 for Boiling Water Reactors (BWRs) and by about a factor of 10 for LWRs (Till and Grogan, 2008).

Used nuclear fuel

Irradiated nuclear fuel is stored inside the reactor containment building associated with the reactor and is moved to temporary storage when extra storage capacity is needed. Fig. 6 (Credits-2, n.d.) illustrates some options for handling, storing, and disposing of used nuclear fuel.

Reprocessing

When nuclear power plants were first being built in the 1960s and 1970s, there was a general consensus that used nuclear fuel from LWRs would be reprocessed and that the plutonium transmuted from uranium would be extracted and used to start sodium-cooled fast breeder reactors. One commercial reprocessing plant was built at West Valley, New York, and operated from 1966 to 1972, and remediation was completed in 2002 under the West Valley Demonstration Project (WVDP) with a cost of about \$5 billion. Another commercial reprocessing plant, the Barnwell Nuclear Fuel Plant (BNFP) was built in Barnwell, South Carolina, but it was not placed into operation due to regulatory changes after construction and due to changes in federal policy in 1977. Various studies have shown that the once-through nuclear fuel cycle for LWRs is more economical than one that involves reprocessing and mixed oxide (MOX) fuel

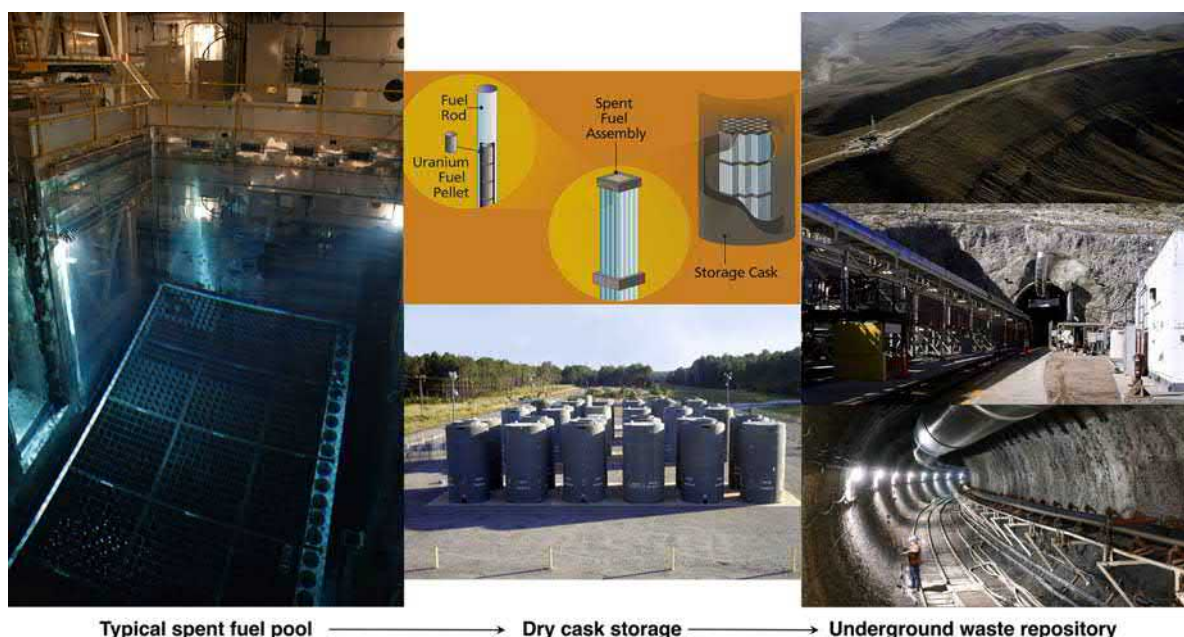


Fig. 6 Options for Handling and Storage of Used Nuclear Fuel. Credits-2. (n.d.) Credits: Left—Perry Nuclear Power Plant, FirstEnergy Nuclear Operating Co., Middle—U.S. Nuclear Regulatory Commission, Right—U.S. Department of Energy.

fabrication for use in either LWR or sodium-cooled reactors. In the long term, when transmuted actinides are considered, reuse of used nuclear fuel may be more economical. Nuclear fuel reprocessing plants are in operation in France and in the Russian Federation.

The Department of Energy has operated nuclear fuel reprocessing plants (for military applications) at the Idaho National Laboratory (INL), the Hanford Site and Savannah River Site. Acidic liquid waste from the first extraction columns (the High Level Waste (HLW) stream) at the Hanford Site and Savannah River National Laboratory (SRNL) is stored in stainless steel tanks, and the pH was subsequently adjusted with a caustic liquid. Thirty-five million gallons of HLW are stored in 43 tanks at the Savannah River Site and 56 million gallons are stored in 177 tanks at the Hanford Site. The HLW liquid at Idaho National Laboratory (INL) was converted to a solid by calcining it to be thermodynamically stable.

Some of the single-walled tanks at the Hanford Site have leaked and some detectable levels of radioactive material has migrated to the Columbia River. No significant leaks are noted to have been reported for the Savannah River site nor at INL. Results from a Radiological Assessment (Till and Grogan, 2008) for the Hanford Site show minimal health impacts to the nearby residence.

Reprocessing of used nuclear fuel (UNF) in the USA has resulted in large expenditures for remediation, and some publications claim that reprocessing is more costly than direct disposal of UNF. Tables 00702-1 and 00702-2 list international locations where reprocessing capabilities of both UO₂ and Mixed Oxide (MOX) exist (Tables 4 and 5). However, the only commercially significant location is in France at LaHague since other locations have either been shut down or have relatively small capacities. Fig. 7 is a photo of the storage facility for vitrified HLW at LaHague in France.

Disposal of radioactive materials

The method for disposing of radioactive materials generally depends on its potential for exposing the environment to radiation, which depends on considerations of radiotoxicity, mobility and half-life. The United States Nuclear Regulatory Commission (USNRC) classifies Low Level Radioactive Waste (LLRW) A, B, C and greater than Class C. This classification scheme is described in 10CFR61.55, and it is based on criteria, such as: toxicity, half-life, and mobility of specific and general radionuclides. Regulations for disposal and transport differ for each class. Materials classified as greater than Class C may not be disposed of by surface land burial.

To date, commercial generators of LLRW may ship to four locations, depending on its classification and on the location of origin. The four active, licensed low-level waste disposal facilities are located in Agreement States and are illustrated in Fig. 8 and listed below. Additional information about the facilities may be found at the Web sites maintained by the respective Agreement States.

- EnergySolutions Barnwell Operations, located in Barnwell, South Carolina: Currently, Barnwell accepts waste from the Atlantic compact states (Connecticut, New Jersey, and South Carolina). Barnwell is licensed by the State of South Carolina to dispose of Class A, B, and C waste.
- U.S. Ecology, located in Richland, Washington: Richland accepts waste from the Northwest and Rocky Mountain compacts. Richland is licensed by the State of Washington to dispose of Class A, B, and C waste.
- EnergySolutions Clive Operations, located in Clive, Utah: Clive accepts waste from all regions of the United States. Clive is licensed by the State of Utah for Class A waste only.

Table 4 Global spent fuel reprocessing capacity and future estimates.

<i>Estimates in metric tons of heavy metal per year</i>								
<i>Country</i>	<i>Fuel Type</i>	<i>2014</i>	<i>2015</i>	<i>2016</i>	<i>2020</i>	<i>2025</i>	<i>2030</i>	<i>2035</i>
OECD America								
United States	LWR	0	0	0	0	0	0	0
OECD Europe								
France	LWR	1700	1700	1700	1700	1700	1700	1700
United Kingdom ^a	Others	600	600	600	0	0	0	0
	Magnox	1500	1500	1500	0	0	0	0
OECD Pacific								
Japan	LWR	0	0	N/A	N/A	N/A	N/A	N/A
OECD								
Russia ^b	LWR	400	400	400	N/A	N/A	N/A	N/A
NEA total		4200	4200	3800	1700	1700	1700	1700

N/A Not available. The entries with bold fonts refer to regions.

^aOthers refers to Thermal Oxide Reprocessing Plant (THORP) in which both LWR and AGR fuels can be reprocessed. As of the end of 2015, the THORP facility was used primarily, although not exclusively, for reprocessing AGR fuels.

^bNuclear Energy Agency (NEA) estimate.

Source: Organization for Economic Co-operation and Development (oecd-nea.org).

Table 5 Current uranium-plutonium mixed oxide reprocessing facilities.

<i>Estimates in metric tons of heavy metal per year</i>			
<i>Country</i>	<i>Facility Name</i>	<i>Scale</i>	<i>Capacity</i>
OECD America			
Canada	Recycle Fuel Fabrication Laboratory (RFLL)	Laboratory	0
OECD Europe			
France	New AREVA MELOX	Commercial	195
OECD Pacific			
Japan	JAEA Tokai (PFDf-MOX)	Laboratory	0.3
	JAEA Tokai (PFFF-ATR)	Pilot Plant	10
	JAEA Tokai (PFPF-FBR)	Pilot Plant	10
OECD			
Russia ^a	Mayak—Paket	Pilot Plant	0.5
	Research Institute of Atomic Reactors (RIAR)	Pilot Plant	1
NEA total			216.8

The entries with bold fonts refer to regions.

^aLast updates to IAEA submitted in 2003.

**Fig. 7** Storage facility for vitrified high-level waste at the La Hague reprocessing plant in France. Credit: radioactivity-1 PHILIPPE LESAGE /AREVA.



© Vemaps.com

Fig. 8 Low Level Waste Acceptance Sites.

- Waste Control Specialists (WCS), LLC, located near Andrews, Texas: WCS accepts waste from the Texas Compact generators and outside generators with permission from the Compact. WCS is licensed by the State of Texas to dispose of Class A, B, and C waste.

The Low-Level Radioactive Waste Policy Act of 1985 made States responsible for managing the radioactive material generated under licenses that they generated. Since radioactive waste disposal sites in each state is not necessary, various states joined to form “compacts” so they could collectively disposal of radioactive waste at a single site.

Exposure from medical treatment and diagnostics

Radiation exposures in the medical field are of concern for patients, health care workers and members of the public. The field of medicine uses radiation in a number of different ways to diagnose and treat a variety of conditions. Diagnostic imaging which utilizes x-ray machines, Positron Emission Tomography (PET) scanners, Computed Tomography (CT) scanners as well as nuclear medicine imaging have a potential for significant doses to patients as well as incidental exposures to healthcare workers. The field of radiation oncology utilizes high energy linear accelerators to treat cancer. In addition, special treatment units such as gamma knife use gamma rays from Co-60 to treat brain tumors whereas the similar cyberknife uses an accelerator on a robotic arm to treat a variety of tumors including extracranial cancers. These highly specialized devices are effective at treating complex cancers but have a significant risk for patient mistreatments. Radioactive material is also used to treat other cancers. Treatment can either be LDR (Low Dose Rate) or HDR (High Dose Rate). The use of LDR therapy is most often associated with prostate cancer treatment. High Dose Rate therapy is becoming the standard of care in radiation therapy and can treat a wide array of cancers throughout the body. Additional information is available in the following chapters of this encyclopedia.

Introduction

Radiation exposures in the medical field are not limited to patients; healthcare workers as well as members of the public must be considered. Sources of exposure are most often related to diagnostic or therapeutic uses of X-rays. Although X-rays are the most common source of radiation, radioactive materials are also widely used in the field of medicine.

Patient exposures are necessary in order to obtain information to diagnosis many medical problems and to treat cancer, while at the same time, reducing dose to the patient is critical in order to minimize risks for secondary cancers. Dose is only one consideration. Image quality must be excellent in order to reduce the need for repeating studies, which increases doses, but high-quality images allow the practitioner to more accurately diagnose the patient. Thus, optimization of shielding may be utilized to reduce dose to workers as well as members of the public. Newer types of scanners that have entered the marketplace have driven the need for improved shielding.

The field of radiation oncology utilizes X-rays with energies above several MeV and thus requires significant attention to the configuration of the radiation dose. Patient exposures are comprised of the prescribed treatment dose as well as scattered radiation. Healthcare workers and members of the public generally have much less exposure due to excellent shielding of accelerator vaults, which must consider all areas near the linear accelerator.

Radiation therapy also utilizes radioactive material to treat a variety of cancers, known as brachytherapy. These treatments can be either LDR or HDR. Both techniques are very effective but take different approaches from dose rate to isotope. A highly specialized form of cancer treatment is gamma knife. This system utilizes small Co-60 sources to treat brain tumors.

Nuclear Medicine is a specialty area in diagnostic imaging and radiation therapy. For diagnostic studies, patients are injected with radioisotopes and monitored with a gamma camera. The gamma emission of the isotope is captured in the camera. Patients receive dose from the injection, while workers receive incidental dose from the patient and handling of the radioactive material. Another aspect of nuclear medicine involves therapeutic doses to kill cancer cells. Thyroid cancer is the most commonly treated with I-131. This procedure can be done on an outpatient basis, which creates potential issues with exposure to members of the public.

Diagnostic imaging

The largest source of radiation exposure in the medical field is X-ray imaging. Millions of people annually receive a variety of X-ray exams for various conditions. The medical field has developed complex devices and enhanced techniques in order to improve image quality. This improved image quality has led to a reduction in patient dose. Current day CT scanners will deliver a dose of approximately 20 mSv for an abdominal scan. This is a relatively low dose, but proper shielding is required in order to minimize dose to medical workers. For example, CT scanners used to operate as single slice scanners, i.e., a single row of detectors. Modern CT scanners, shown in Fig. 9, are now capable of having multiple rows of detectors and thus increasing their ability to collect additional data per scan. This data can provide dynamic imaging of organs as well reduce scanning times. These systems have an increase in scatter radiation that must be taken into account when shielding is placed and scanner is operated.

Diagnostic imaging is critical in vascular and cardiac procedures. Cardiac catheterization and various vascular studies require significant "beam on" time and a larger field in order to accurately locate and treat the area of arterial blockage. A dose of 8–10 mSv is common during a standard procedure cardiac catheterization. Higher doses are possible if the case is complex. There is a risk to the patient in such instances due to the high skin dose. Patients are advised of this prior to the procedure and must be



Fig. 9 CT scanner. Courtesy of Canon Medical Systems, USA.

monitored for skin erythema post procedure. Physicians and technologists are also at a higher risk due to scattered radiation. Proper use of As Low As Reasonable Achievable (ALARA) techniques is critical to reduce dose. A typical cardiac lab can be seen in Fig. 10.

PET scanners provide a unique challenge in radiation shielding. Newer PET systems frequently incorporate a multi slice CT scanner, shown below in Fig. 11. This adds a significant improvement in functionality in that patients can now obtain two scans during the same procedure. In addition, the scans are fused which lead to increased accuracy of results. Patients receiving a PET scan often receive an injection of F-18, with a half-life of approximately 110 min. The positron emission of 511 keV gammas provides a unique challenge in shielding. Additional lead shielding is required as compared to a standard CT installation. A hot lab is also required in order to store the F-18 isotope. Safety precautions need to be established in order to minimize dose to patients and staff. Healthcare workers are required to receive no more than 5 rem annually.



Fig. 10 Cath lab. Courtesy of GE Healthcare.



Fig. 11 PET scanner. Courtesy GE Healthcare.

Radiation therapy

Radiation therapy also uses X-rays, but at much higher energies. Diagnostic X-rays are generally in an energy range of 30–130 kV. Radiation therapy accelerators produce X-rays in the energy range of 2–23 MV. The radiation therapy treatment beam penetrates deep within tissue, while sparing the skin surface. The percent depth dose can be seen in Fig. 12. The maximum dose occurs below the skin surface. Electron therapy is also utilized for superficial tumors. Electron energies range from 4–20 MeV. The radiation beam is highly targeted in order to reduce dose to normal tissue. The patient inherently receives a large scatter dose depending on the type of treatment regimen that is prescribed.

The primary concern in radiation therapy is radiation exposure to workers as well as members of the public. Accelerator vault design is critical in order to reduce doses to within State and Federal limits. Types of treatments, workload, occupancy factor and use factor are just some of the considerations when calculating primary and secondary barrier thickness.

The use of radioactive materials (brachytherapy) to treat cancer is also common in radiation therapy. Low-dose rate (LDR) procedures are commonly used to treat prostate cancer. LDR therapy is defined as delivering a small amount of gamma radiation dose over an extended period of time. For prostate cancer, I-125 or Pd-103 seeds are inserted into the prostate gland. The dose delivery time can range from 170–600 days, with a prescribed dose to the prostate between 110–144 Gy, depending on the cancer and isotope. Radiation risk to patients and members of the public is very small. The seeds are permanent.

High dose rate (HDR) therapy uses Ir-192 to treat cancers throughout the body. This type of therapy delivers a high dose of gamma radiation in a short period of time. Patient treatment times last from 1 to 20 min, while utilizing a 10 Ci source of Ir-192. Catheters are placed within the patient for treatment delivery. The Ir-192 source travels to preset positions for a calculated amount of time. Due to the high dose rate and energy of the Ir-192 source, shielding is required. Many HDR treatments are performed in accelerator vaults. Some facilities have specialized rooms that are shielded specifically for the average gamma energy of 0.38 MeV. HDR therapy has a number of benefits including rapid delivery, better radiobiological effect on certain tumors and hypo fractionated treatment regimens. A typical HDR unit is shown in Fig. 13.

Due to Ir-192 having a 74 day half-life, source exchanges are generally done every 3–4 months. The required handling of incoming sources as well as shipping back the depleted sources, presents increased exposure risks for the medical physicist as well as the service engineer. State and Federal guidelines require strict controls and processes for these exchanges. Some State agencies will actually be onsite for exchanges to validate proper handling.

Radiation exposures with HDR treatments are potentially more impactful. Emergency situations during treatment delivery require detailed policies on how to handle their radioactive sources safely. Previous incidents have occurred that have resulted in patient deaths as well as severe radiation injuries to healthcare workers due to poorly implemented programs.

Gamma knife is a highly specialized treatment system for brain tumors. Unlike accelerators, a gamma knife (Fig. 14) uses radioactive material for treating patients. A total of 192 Co-60 sources are loaded into the system to deliver dose with submillimeter accuracy. Co is an ideal isotope for brain treatments due to its gamma energies of 1.17 and 1.33 MeV along with a 5.27 year half-life.

Exposure concerns are significant. Unlike accelerators where the machine can be “turned off,” the gamma knife has a constant exposure concern with the Co sources. Medical physicists and radiation therapists have the greatest risk for exposure. Source exchanges, which are done within 5 years require extra caution as sources are removed and new sources are inserted.

Nuclear medicine

Nuclear Medicine is subset of both diagnostic imaging and radiation therapy. Patients receive radioactive material injections to diagnose various conditions. The most common injection is Tc-99m (a metastable state of Tc-99). This radioactive isotope has a half-life of 6 h and does not remain in the body for an extended period of time as well. The gamma emission of Tc-99m is 140.5 keV, which is ideal to be detected using gamma cameras. A typical gamma camera is shown in Fig. 15. Patients need to exercise caution after

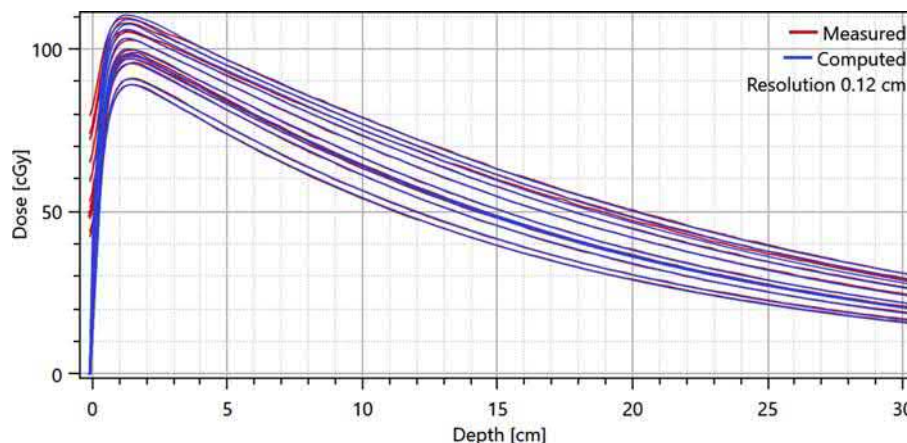


Fig. 12 Percent depth dose curves for various collimator settings.



Fig. 13 HDR Unit. Courtesy of Varian Medical.



Fig. 14 Gamma knife. Courtesy of Elekta.

leaving a nuclear medicine procedure since the Tc-99m is still in their system. Excretion of technetium is largely gastrointestinal and renal though the physiology is complex. Nuclear medicine technologists are at an increased risk for exposure due to the number of patients that are seen on a regular basis. ALARA practices are strictly enforced in medical facilities.

Nuclear medicine is primarily used as a diagnostic tool. However, it can include therapeutic treatments. The most common treatment is thyroid cancer. Patients are given a pill containing up to 300 mCi of I-131 where the dosage may be more or less for different patients, based on medical evaluation. The patient is able to leave the hospital and return home in isolation in most cases. This type of treatment presents significant risk to members of the public. The body fluid of patients is radioactive during the treatment. The majority of the I-131 is absorbed by the thyroid within 2 days. The isotope is primarily excreted through the urine. Home isolation is critical during the therapy in order to minimize public exposure. There have been documented cases where patients received the I-131 therapy and returned to work the same day. These instances pose significant radiation risk to members of the public.



Fig. 15 Gamma camera. Courtesy of GE Healthcare.

Nuclear weapons testing

The first test of a nuclear device, named Trinity, took place in New Mexico, United States in July 1945, as a culmination of the Manhattan Project and the U.S. World War II efforts (Simon et al., 2006, 2020). It was followed a few weeks later by the bombings of the cities of Hiroshima and Nagasaki, in Japan. Nuclear weapons testing continued for much of the 20th century, with atmospheric detonations being carried out mainly by the United States and former Soviet Union, followed by United Kingdom, France, and China. It is now understood that radioactive fallout debris from these nuclear detonations deposited across the entire Earth, being detected in soil, water, foods, even polar ice, and creating exposure conditions via inhalation, ingestion and external irradiation, for people of all ages, ethnicities and locations.

Historical background

The United States was the only country with operational nuclear weapons until 1949, when the former Soviet Union detonated their first device in Semipalatinsk (now part of Kazakhstan). The United Kingdom started testing shortly after, in 1953. Numerous tests were carried out until 1958, when they were suspended through a voluntary moratorium that lasted until September 1961. Extensive testing took place again in 1961 and 1962, but negotiations between the United States and the former Soviet Union led to the Limited Test Ban Treaty, signed on April 5, 1963, which prohibited all nuclear tests except if conducted underground, to ensure little or no releases of radioactivity. However, atmospheric nuclear tests carried out by France and China, who did not sign the Limited Test Ban Treaty, continued until 1980. India, Pakistan and North Korea started carrying out nuclear testing recently, but none of their tests involved atmospheric detonations (Bouville, 2020). A Comprehensive Test Ban Treaty, banning nuclear weapons test detonations in any environment was adopted by the United Nations General Assembly in 1996, but it is not in full force at this time because a few countries have not ratified the document. Nevertheless, constant efforts have been made by most governments to limit the number of nuclear weapons and the number of countries who possess them. From 1945 to 1980, over 500 nuclear devices were exploded in the atmosphere at 13 primary nuclear testing sites around the world, with an equivalent explosive power ranging from 1 kt to 50 megatons of TNT per device and a total power of 440 megatons (Simon et al., 2006; Bouville, 2020).

Characteristics of radioactive releases

Nuclear weapons generated about 150 radionuclides that were released into the environment, many having short half-lives. Out of these about 20 radionuclides are of radiological importance. These radionuclides can be divided into three types:

- Fission products and their decay products (^{131}I , ^{133}I , $^{132}\text{Te-I}$, ^{90}Sr , ^{137}Cs , $^{140}\text{Ba-La}$, ^{141}Ce , ^{103}Ru , ^{106}Ru , ^{89}Sr , ^{91}Y , $^{95}\text{Zr-Nb}$, ^{144}Ce , ^{125}Sb)

- Neutron activation products and actinides (^{54}Mn , ^{55}Fe , ^{14}C , ^{239}Np)
- Leftover fuel material (^{239}Pu , ^{240}Pu , ^{241}Pu , ^3H)

Weapons testing took place in remote locations, at least 100 km from populated areas (continental deserts, isolated islands in oceans), to minimize potential exposures. Depending on the yield and location of an atmospheric detonation (altitude and latitude) radionuclides were injected into the troposphere (the atmospheric layer closer to ground) or higher into the different layers of the stratosphere. Radionuclides in the stratosphere were diffused mainly over the latitude band in which they were injected, with some exchange between hemispheres, and they slowly deposited to the Earth's surface in a matter of months and years. Radionuclides in the troposphere were carried by prevailing winds and circled the Earth for 2–3 weeks within the same hemisphere, before being deposited on the ground. Deposition of radionuclides occurred by settling of particles (dry deposition) or by wet processes (rain, snow, fog). Fractions of radioactive fallout were deposited locally (50–500 km from ground zero) or regionally (several thousand kilometers downwind), with the remainder (global) fallout being dispersed throughout the world.

The activity of radionuclides released from nuclear weapons tests were much larger than the activities resulting from nuclear reactor accidents, such as the Chernobyl accident in 1986. For example, the activities of ^{131}I , ^{137}Cs , and ^{90}Sr released from all atmospheric tests were greater than releases from the Chernobyl accident by factors 400, 10, and 60, respectively (Bouville, 2020). Given that radionuclides lingered in the atmosphere anywhere from several weeks to several years, the activity of short-lived radionuclides decreased significantly before deposition. As such, less than 1% (5300 PBq) of the estimated activity of ^{131}I released globally (675,000 PBq; half-life = 8 days) was deposited onto the ground, while close to 100% of activities ^{137}Cs (900 PBq; half-life = 30 years), ^{90}Sr (600 PBq; half-life = 29 years), or ^{239}Pu (6.5 PBq; half-life = 24,000 years) were deposited. The long-lived radionuclides changed the global radiation environment on Earth and are now part of the radiation background.

All humans have been exposed to small amounts of radioactivity in fallout from nuclear weapons testing. A substantial number of workers and military personnel involved in preparation of tests, carrying out radiation measurements, and decontamination of test sites were exposed at levels higher than the average member of the public. Radiation doses from fallout are commonly calculated starting with activity concentrations of radionuclides deposited on the ground measured at different locations, augmented by information on expected composition of fallout at those locations (Simon et al., 2004). External gamma emitters in fallout produce a relatively uniform exposure of the entire body, but doses from inhaled or ingested radionuclides can differ significantly among organs. Largest doses from weapons fallouts were delivered to the thyroid gland from ingestion and inhalation of ^{131}I . Historical exposures of members of public around the world depended highly on distance from the nuclear test sites, testing conditions, climate conditions, and dietary and lifestyle habits, with annual effective doses being, in general, in the order of a fraction of a milligray, lower than doses from a single whole-body CT scan. However, significant exposures of some populations did occur, leading to thyroid doses of tens or hundreds of milligrays, for example, in people living in villages near the Semipalatinsk test site in the former Soviet Union (Land et al., 2008) or in children in Washington Co, Utah, or Lincoln Co., Nevada near the Nevada Test Site in the U.S. (Till et al., 1995), while children and adults on Rongelap and Ailinginae atolls in the Marshall Islands received doses in excess of several Grays to the thyroid gland (Simon et al., 2010). Effective doses from fallout can be small compared to effective doses from natural background of radiation, but exposures are non-uniform across the human body, with thyroid gland receiving much larger doses. As such, an additional 49,000 cases of thyroid cancer would be expected to occur among the US population from exposure to radioactive fallout from the atmospheric nuclear weapons tests that were conducted at the Nevada Test Site in the 1950s. In addition, there could be as many as 11,000 deaths from non-thyroid cancers related to fallout from all atmospheric tests that were conducted at all sites in the world, with leukemia making up 10% of this total (Bouville, 2020).

See Also: Global Market for Radiation Applications; Medicine: New Drug Developments—Radiolabeled Molecules in Drug Development; Medicine: Radionuclides Used in Nuclear Medicine; Medicine: Radiopharmaceuticals and Their Use in Nuclear Medicine; Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection; Radiation Sources.

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Source Terms for Nuclear Reactors Operations, Accidents, and Waste

Vikram Singh and Alexander M. Wheeler, Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, United States

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Introduction

An assessment of health effects from a radiological release requires information about the quantity and form of the release. Such a radiological assessment of health risks is interdisciplinary. This is because radioactive sources are transported through the environment in myriad ways, and their uptake in biological systems may lead to resurfacing in different forms. As an example, consider a radioactive source that is released in the form of a smoke plume. It may travel for a few miles with the wind, be deposited onto the soil via rainfall, some of it may end up in groundwater and some may be taken up by grass. Cattle grazing on this grass would ingest this material which will then show up in dairy products consumed by a group of people. It is quite clear that tracking the radioactive material through these systems would require knowledge of a variety of disparate subjects. So it comes as no surprise that the methods used today in radiological assessment evolved within individual sciences and merged gradually over the years. Such assessments may be made for a past release (retrospective), a present-day release, or predictive of a future release (prospective). An illustrative equation to describe the interdisciplinary nature of such an assessment is as follows (Till and Grogan, 2008):

$$Risk = (S \cdot T \cdot E \cdot D \cdot R)_{uvc} \quad (1)$$

where S is the source term; T is the environmental transport factors; E is the exposure factors; D is the conversion to dose; R is the conversion of dose to risk further, u is the uncertainty analysis; v is the validation from prior knowledge; c is the communication of results; p is the stakeholder participation.

In the above equation, the ultimate objective is to decipher the relationship between the “source term” and the potential effects on the environment and human health. In the case of a nuclear installation—be it a reactor, an accelerator, or a processing and handling facility—the types and amounts of radioactive material that are available for release into the environment is known as the *source term*. The nature of the source term will vary depending on the facility and the particulars of the process being considered. It is contingent on the mode of release and affected by parameters including the rate of decay of the nuclides, the magnitude and duration of release, and the physio-chemical characteristics of the radioactive contaminants.

In assessing health effects, it is important to distinguish the radionuclides present in the source as different radiation types since they interact differently with matter based on their physical characteristics. Similarly, the chemical properties of the radioactive material affects the modes of propagation that are accessible to it. This makes some sources relatively harmless while others extremely dangerous to living organisms.

Radioactive releases may be routine in operation or may arise from an abnormal event. In either case, an accurate estimation of the source term includes a description of the quantity and form of radioactive material released and of the location and timeline of the release that occurred. In determining the timeline, the rate of release is an important consideration as it may be intermittent or with a time varying rate. Furthermore, the radioactivity may be released directly into the atmosphere, into groundwater, or the soil. Thus, once the source term has been established for a facility, the migration paths through which the radioactivity disperses will be similar, although their relative importance may vary.

For these reasons, a major requirement in the safety analysis report (SAR) for licensing new nuclear facilities is establishing the source term for all credible accident scenarios and for routine operations. This allows the facility licensee to demonstrate to regulatory bodies that inherent features in the design of the facility contribute to the mitigation of accidents. Thus, licensing of new designs and their sites requires that these features be assessed with enough detail in order to meet scientific requirements for the design and regulations aimed at protecting the general public and alleviating concerns. Additionally, in the unfortunate event of an accident, the source term needs to be confirmed through direct observation of radiation release and transport, and future studies will use the data observed for their estimates.

This article is aimed at presenting readers from a broad variety of backgrounds with an overview of the approaches used in estimating source terms for nuclear reactors and for some regulations governing them. Characteristics of different reactor types are discussed as they apply to determining source terms. Normal operation and accident source terms are discussed. Computer codes used for such studies are mentioned along with examples of the types of source terms arising from different reactor types. Some examples of historical events of significance are discussed and their source terms provided. The goal of this article is to inform the reader about the following:

- (1) the nature of source terms from nuclear reactors,
- (2) the considerations that go into determining these source terms,
- (3) representative source terms from normal operation,
- (4) representative source terms from reactor accidents, and
- (5) reactor accidents, and the radiotoxicity of spent nuclear fuel.

Considerations for estimating source term

The primary objective of radiological risk assessment is to minimize exposure to the public. Thus, assessments are concerned with the potential for transporting emitted radioactivity through air, soil, or water. Such sources are expected to reach the public in trace quantities of parts per million (or billion). Hence, only those components of the source term that are capable of escaping containment in a form that can travel along environmental pathways or be carried by surface contamination are of significant interest. Therefore, when examining source terms, it is important to evaluate the probabilities of escape of certain radionuclides from the containment under both routine operation and accident scenarios. Components of the source term that are of little interest include the following:

- (1) inherently immobile (chemically stable solid),
- (2) rigidly fixed in position,
- (3) of such low activity (long half-lives) that they have negligible contribution to health risks
- (4) are so short-lived that they would decay away before migrating to a target population.

A complete description of the source term requires information about the radioactive material released, its chemical form, and when and where the release occurred. These factors must be described with sufficient accuracy and detail to carry out epidemiological studies on the probable health effects in the exposed public. However, this can prove difficult following a source release given the inherent uncertainties associated with reconstructing doses retrospectively. Hence, it is best practice to estimate the source term using different methods, with redundancy being preferable. There are, in general, three independent methods used in estimating the source term, not all of which may be applicable in every situation. The first is to carry out a scoping study using engineering estimates that provide a preliminary assessment of the magnitude of source release expected for various scenarios. Such a study generally utilizes sophisticated computer codes and may either be purely deterministic, or make use of probabilistic/risk-based assessments, or a combination of the two. The second makes use of historical data from source release events at similar facilities and the radioactive release observed in the environment. And the third data set involves direct sampling of the environment following release.

As shown in the illustrative Eq. (1), the source term is first and most important consideration of radiological risk assessment. Its estimation requires the most resources relative to the other parts of the equation. It is evident that predicting the transport pathways and behavior modes for all important radionuclides poses a significant problem. This is especially difficult for the most volatile components and noble gases. For example, a biologically important radionuclide such as iodine-131 is transported in the environment differently and poses differing degrees of risk depending on whether it was released in an organic or inorganic form (Eichholz, 1977). Here it is important to distinguish the “initial source term” from the “release source terms” as the two may vary substantially. For the latter, a reformulation of the initial source term is carried out with considerations of all likely internal pathways and environmental interactions that may modify the chemical or physical form of a radionuclide.

Historically speaking, an often-neglected aspect of determining source terms has been the inclusion of uncertainty figures. Providing a single value for the source term assigns an exaggerated sense of accuracy to the data that may harm public confidence. Instead, it is best practice to provide a range since, in reality, there may not be reliable data. Alternatively, an upper bound deterministic value may be presented to establish the significance of a particular source. In doing so, the doses and risks calculated are far greater than those expected to occur and thus provide a useful comparison for determining the relative impact of different sources. Finally, the source term should be derived with as many different and independent approaches as feasible. Such redundancy allows for cross-examination of the calculations, strengthens the conviction of the investigators, and increases public confidence in the credibility of the estimation.

If a defensible estimate of the source term cannot be established, credible dose and risk estimations cannot be made. This irreparably hurts public confidence in any assessment of the radiological risks emanating from a facility. This is especially important following an accident when considerable media attention is present and there is popular demand for a credible appraisal of the extent of damage and the risks to public health. Therefore, a meticulous and comprehensive evaluation of the source term is critical in radiological risk assessment and should be carried out prior to licensing, in regular intervals during operation, and following accidental radioactive release to the environment.

Modes of release

A range of human activities release radionuclides into the environment. In the case of nuclear power generation, radioactivity may be released both during routine operation of the facility or when accidents occur. Radioactivity released from routine operation is well below regulatory limits and poses no danger to the public.

The released radionuclides may be in the form of dust particles carried by the air, aerosolized particles, gas or vapor, or dissolved or suspended in water or other liquids. Routine releases of radionuclides during operation occur via discharge into air through filtration systems, and to water bodies via liquid discharge outflows. Spent fuel from the reactor that is either stored onsite and eventually sent to a geologic disposal facility also poses a risk, albeit minimal, of releasing radioactivity into the air or water, or to the soil and then to groundwater. In all cases, it is important to have information about the particle size and chemical form or solubility of the released activity to properly model environmental transport of the radionuclides.

For both routine and accidental release, understanding the accessible modes of release is crucial in determining the downstream consequences of the released radioactivity. The release may occur in several ways depending on the process or the accident that occurred. Radioactivity may be released into the air and carried by the wind to be deposited over a large area. Tritium formed from nuclear processes in the reactor may diffuse out of containment through discontinuities such as welds and joints. Some radionuclides may be released in the form of dust particles that can settle on various surfaces at different distances from the facility depending on the weather conditions. Radionuclides dissolved in water may be released into water bodies through underwater discharge. In the case of accidents such as a loss-of-coolant, fission products may vaporize from the fuel or small particles of the fuel can be dispersed.

The type of discharge into the environment may be continuous (such as tritium) or in batches (volatilized fission products from spent fuel discharge). Nevertheless, source term analysis requires a holistic assessment of all the release modes available for dispersion of radioactivity into the environment.

Postulated initiating events (PIE)

In assessing the source term for accidental release, the first step is to define postulated initiating events (PIE). The PIEs establish the scenarios to be analyzed to predict the amount and types of release. The range of PIEs must cover all credible accident scenarios that could influence the safety basis of the facility. The methodology applied in defining PIEs and their consequences predicts the risk and the safety margin associated with the operation of the facility, and must demonstrate that both are acceptable to the regulatory body. Such an analysis covers all accidents identified as design basis accidents (DBA). In addition, the safety analysis of the reactor must cover beyond-design basis accidents (BDBA) for emergency planning and accident management. While a BDBA is defined as an event whose frequency of occurrence is very low, such an event is still theoretically possible and its consequences may be quite significant. Some DBAs and BDBAs could lead to the release of radionuclides to the environment (the source term). Observed frequency of various initiating events at US nuclear reactors was reported by [Poloski and Marksberry \(1999\)](#) for the US Nuclear Regulatory Commission (NRC).

General features of LWRs, FBRs, GCRs, and MSRs important for source term estimation

Nuclear reactors produce power by converting the heat generated through nuclear fission into electricity by operating a power conversion system. The nuclear fuel is most commonly U-235, although Pu-239 and U-233 can also be used. While the Rankine Cycle, which uses high-pressure steam as the working fluid, is the most widely deployed power conversion system today, a variety of thermodynamic power conversion systems have been proposed.

Nuclear reactors are designed to sustain a fission chain reaction. They can be broadly classified depending on the average speed of the neutrons causing fission. That is, the neutron may either be moderated (thermal) or unmoderated (fast). Several types of reactor designs have been devised. The most commonly deployed reactor design in the world today is the Light-Water Reactor (LWR) which includes Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). In both designs the liquid water is pressurized to > 100 times atmospheric pressure and serves as both a neutron moderator and coolant for the fuel rods. Another type of thermal reactor that uses heavy water (D_2O) instead of regular water is called the Heavy Water-Moderated Reactor (HWR). An archetype of this design is the Canada Deuterium Uranium Reactor (CANDU) designed and deployed by Canada. Similar reactors are also operated in India. HWRs use similar fuel type as LWR although usually use natural uranium that is discharged at $\sim 1/5$ th the burnup of LWR fuel. Some reactors use pressurized gases such as helium or carbon dioxide as coolants and are called Gas-Cooled Reactors (GCR). These reactors were popular in the United Kingdom in the latter half of the 20th Century but have all been decommissioned. Some advanced designs of GCRs also exist that utilize pebble fuel. Unmoderated or fast reactor designs have the ability to produce more fuel than they consume (breeding). These reactors, called Fast Breeder Reactors (FBR), use liquid metals such as sodium and lead as coolants. Then there is a class of reactors that utilizes fluid fuel in the form of molten salts, called Molten-Salt Reactors (MSR). MSRs can be both moderated or unmoderated. The coolant of both fast reactors and MSRs is at near atmospheric pressure.

LWRs use cylindrical ceramic pellets of uranium oxide as fuel. The pellets are loaded into thin tubes of zirconium alloy which are in turn arranged into assemblies. These assemblies are placed in a grid in the reactor core. As the chain reaction proceeds and fuel

burnup increases, fission products build up within the fuel pellets. Some fission products, especially gases, coalesce into bubbles which can lead to swelling and eventually cracking of the fuel elements. The gases may diffuse out into the gap between the pellet and the zircaloy cladding. These diffused fission products have the highest probability of being released into the environment. They may leak out into the coolant through imperfections in the cladding. Unlike PWRs, in BWRs the water in the core is allowed to boil. The steam thus formed passes through a turbine and condenser and the water returns back to the reactor core. Consequently, radioactive contaminants in the coolant that are confined to the reactor building in a PWR travel through the power conversion system in a BWR and may be released into the environment with any steam release. Furthermore, radioactive noble gases are particularly mobile in a BWR and are released to the environment in greater quantities than PWRs during normal operation, albeit well below regulatory limits.

GCRs are similar to LWRs with the fuel in micro encapsulations within graphite. In the former case, the fuel is encased in graphite for moderation. The pebbles incorporate fuel in the form of tiny spherical particles encapsulated in silicon carbide that are embedded in a graphite matrix. Such TRISO fuel pebbles have been designed to be better at containing fission products than traditional zircaloy-clad fuel pellets. In general, reactors in which the coolant is under high pressure (LWR, CANDU, and GCR) possess potential energy for releasing radioactive material into the environment in case of a reactor accident. This is a major concern and hence these reactors are generally placed in large containment buildings made of steel-reinforced concrete.

In sodium-cooled FBRs, the fuel is typically metallic and bonded to stainless steel cladding using sodium. Similar processes as in LWRs lead to swelling and cracking in FBR fuel. Some fission products may also migrate and dissolve into the sodium bonding. If any fuel pins fail, the gaseous fission products will be released into the coolant sodium along with a fraction of the bonding sodium containing migrated fission products. The molten sodium offers excellent fission product retention and with the lack of pressure, there is little potential for dispersal. However, a major accident scenario in FBRs involves coolant leakage. Liquid sodium reacts violently with air and the moisture in it. Thus, the resulting fire can widely disperse any radioactive materials present in the sodium.

MSRs are a unique reactor design in which molten fluoride or chloride salts contain the dissolved fuel. As burnup proceeds, the fission products formed are retained in the system to various degrees depending on their chemical compatibility with the salt. Some fission products form stable compounds while others remain in their elemental form and “plate out” on various surfaces. Furthermore, volatile fission products simply depart from the liquid fuel and can be captured in a gas treatment system. Such a system possesses its own source term in the case an accident occurs. MSRs operate at just above atmospheric pressure and as a result have little potential for dispersing radioactive fission products as compared to a high-pressure system such as an LWR. Additionally, molten salts are generally non-reactive with air or water, although individual fission products entrained in the salt may react and be dispersed.

All nuclear reactors are designed with radioactive waste management systems to retain as much of the radioactive materials as possible such that the amount that is released into the environment can be described to be “as low as reasonably achievable” (ALARA) and “as low as reasonably practicable” (ALARP). As discussed in [Rempe and Williams \(2021\)](#), attainment of the ALARA objective is of paramount importance in both the designing and operation of nuclear reactors ([U.S. Nuclear Regulatory Commission, 2013](#)).

Important radionuclides in nuclear reactor source terms

Source term estimation for a nuclear reactor facility will include information about several radionuclides of interest. While radionuclides are generally categorized as natural and “man-made,” the differences are not always quite apparent. For example, nuclides such as ^3H and ^{14}C which are formed via nuclear interactions in an operating reactor are also formed through cosmic ray bombardment in the atmosphere ([Sinclair et al., 1987](#)). Although rare, uranium atoms undergo spontaneous fission in nature and the resulting radionuclides are present in trace quantities. Furthermore, geologic studies have found evidence of natural nuclear reactors such as the one in Oklo in the Central African nation of Gabon that existed 2 billion years ago, when the relative $^{235}\text{U}/^{238}\text{U}$ ratio on Earth was greater ([Cowan, 1976](#)).

The naturally occurring isotopes of uranium, ^{235}U and ^{238}U , are most used in nuclear facilities. While these isotopes have long half-lives, their short-lived decay products are important in risk assessment. When used in a nuclear fission reaction, for every fissile atom of ^{235}U consumed, two smaller “fragments” or atoms called fission products are produced. Since the fissile atoms are neutron rich, the fission products produced possess an excess of neutrons which makes them highly radioactive. They shed these excess neutrons, mainly via β^- decay, until the neutron-to-proton ratio is stabilized. Often, these decays are accompanied by gamma rays. Only those fission product combinations are produced that conserve both charge and mass. Hence, the fission product distribution differs with the fissile fuel. [Fig. 1](#) shows the typical fission product yield for ^{233}U , ^{235}U , and ^{239}Pu in a thermal reactor.

PWRs also utilize soluble boron for reactivity control. Boron has a high neutron capture cross-section and undergoes a variety of reactions including $^{10}\text{B}(n,2\alpha)^3\text{H}$ which leads to the contamination of coolant with tritium. Some structural materials may become radioactive through interaction with neutrons through a process called activation. Examples include $^{54}\text{Fe}(n,\gamma)^{55}\text{Fe}$, $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$, $^{58}\text{Ni}(n,p)^{58}\text{Co}$, and $^{56}\text{Fe}(n,t)^{54}\text{Mn}$. An activation product that is important from a radiological risk assessment perspective is ^{134}Cs that is formed by neutron capture of stable fission product ^{133}Cs . While other cesium isotopes are formed as fission products, ^{134}Cs is formed almost exclusively via neutron activation. Thus, the ratio of ^{134}Cs to other isotopes is an indicator of the extent of radioactive contamination from a reactor accident where ^{133}Cs is left in high neutron flux as opposed to nuclear weapons fallout where ^{133}Cs is not ([Perkins et al., 1990](#); [Chino et al., 2016](#)). In FBRs with sodium coolant, neutron capture reaction in ^{23}Na creates ^{24}Na as

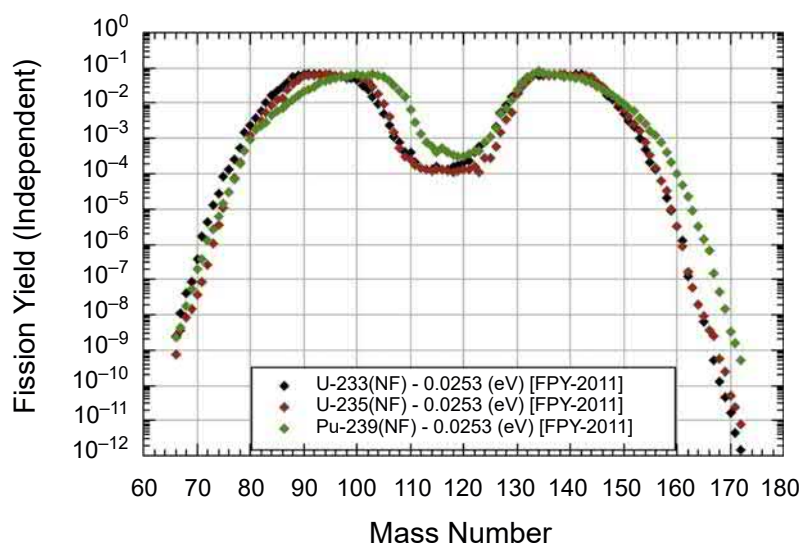


Fig. 1 Thermal neutron-induced fission yield for ^{233}U , ^{235}U , and ^{239}Pu per fission. Source: JAEA Nuclear Data Center.

an activation product which has a half-life of 15 h. These fission and activation products form the bulk of the source term in nuclear reactors. **Table 1** presents a list of important radionuclides produced during LWR operation.

Detailed methods for calculating activities of individual radionuclides resulting from fission product decay chains and neutron activation can be found in most textbooks on Nuclear Engineering (Glasstone, 1955; Lamarsh and Baratta, 2001). Similarly, several sources of information exist which present the decay schemes of radionuclides of interest in radiological risk assessment such as the Nuclear Data Services section of the IAEA. Several sophisticated computer codes such as ORIGEN have been created to perform isotopically detailed calculations of the buildup, decay, and release of radionuclides based on reactor operational history (Rearden and Jessee, 2018).

Source terms encountered during normal operation of LWRs

One could believe that nuclear reactors today are like old aging cars resulting in the misleading view that they pose a danger to public wellbeing. After all, a cross country trip in a 1969 Chevy is not very ideal. However, this comparison to old automobiles could not be farther from the truth. While cars remain mostly fixed in their design after exiting the assembly line, operating nuclear power plants undergo a series of improvements during their lifetime resulting from gathered operational data that leads to better operating practices, comprehensive maintenance programs, and several design upgrades. Moreover, there is a continuous improvement in the design of the nuclear fuel that is replaced every few years of reactor operation. The US NRC has implemented a regulatory approach that maximizes safe operation which has resulted in the US nuclear industry achieving one of the best industrial safety records of any other industry (Nuclear Energy Institute, 2020).

Since all power-producing reactors in the United States are presently LWRs, there is long-term data available for their operational source terms. Of the radionuclides listed in **Table 1**, only some are released into the environment during reactor operation, and during reactor spent fuel handling, and storage—all of which are essential to plant operation. Not all the radionuclides are equally important from a health risk perspective and their contribution to the source term varies with the reactor type. To show a representative snapshot, **Table 2** provides normalized amounts of radionuclides released from US power reactors for the year 1988. The amount released by any particular reactor will depend on the type of effluent treatment system employed, the power level, and a mix of other factors.

The radionuclide releases from PWRs and BWRs differ. While the absence of steam generators in a BWR simplifies piping and reactor layout, it also allows radionuclides from the reactor core that are dissolved or suspended in the coolant to travel through the power conversion system and become available for release into the environment. **Fig. 2** shows normalized release rates of noble gases from PWRs and BWRs in the United States. It is clear that BWRs, in general, release more noble gas effluents than PWRs. However, the overall trend shows a drastic decrease in the release rates over the last few decades; approximately 99.9% decrease for BWRs and even more so for PWRs over the considered time period. This is in keeping with the ALARA principle and also indicative of the continuous upgrades made to reactor plants. In the case of noble gas effluents, the primary contributors here are improved fuel pin integrity and use of advanced off-gas treatment systems.

For reductions in radioactive release for day-to-day operations, the design elements that provide greatest impact are improvements in fuel rod fabrication methods aimed at operational integrity, radionuclide filtering systems used in the primary and secondary water loops, and chemistry monitoring and control systems. With the solid fuel enclosed in zircaloy cladding, the

Table 1 Important radionuclides generated by nuclear power plant operation.

Radionuclide	Half-life	Radionuclide	Half-life
<i>Noble gas fission products</i>		<i>Major corrosion and activation products (continued)</i>	
^{83m} Kr	1.8 h	³ H ^a	12.4 years
^{85m} Kr	4.5 h	¹⁴ C	5730 years
⁸⁵ Kr	10.7 years	²⁴ Na	15 h
⁸⁷ Kr	76.3 m	^{110m} Ag	250 days
⁸⁸ Kr	2.8 h	¹³⁴ Cs	2.1 years
⁸⁹ Kr	3.2 m	<i>Some important fission products</i>	
¹³¹ Xe	11.9 days	⁸⁸ Rb	17.8 m
^{133m} Xe	2.2 days	⁸⁹ Rb	15.2 m
¹³³ Xe	5.2 days	⁸⁹ Sr	50.5 days
^{135m} Xe	15.3 m	⁹⁰ Sr	29.1 years
¹³⁵ Xe	9.1 h	⁹⁰ Y	64 h
¹³⁷ Xe	3.8 m	⁹¹ Sr	9.5 h
¹³⁸ Xe	14.2 m	⁹¹ Y	58.5 days
<i>Radioiodine and precursors</i>		⁹² Sr	2.7 h
¹²⁹ I	1.57 × 10 ⁷ years	⁹² Y	3.5 h
^{131m} Te	30 h	⁹⁵ Zr	64 days
¹³¹ I	8 days	⁹⁵ Nb	35.2 days
¹³² I	2.3 h	⁹⁷ Zr	16.9 h
¹³² Te	78.2 h	⁹⁷ Nb	72.1 m
¹³³ I	20.8 h	⁹⁹ Mo	66 h
¹³⁴ I	52.6 m	¹⁰³ Ru	39.3 days
¹³⁵ I	6.6 h	¹⁰⁶ Ru	368.2 days
<i>Major corrosion and activation products</i>		¹²⁴ Sb	60.2 days
⁵¹ Cr	27.7 days	¹²⁵ Sb	2.8 years
⁵⁴ Mn	312.5 days	¹³⁶ Cs	13.1 days
⁵⁵ Fe	2.7 years	¹³⁷ Cs	30 years
⁵⁶ Mn	2.6 h	¹³⁸ Cs	32.2 m
⁵⁷ Co	270.9 days	¹⁴⁰ Ba	12.7 days
⁵⁸ Co	70.8 days	¹⁴⁰ La	40.3 h
⁵⁹ Fe	44.5 days	¹⁴¹ Ce	30.5 days
⁶⁰ Co	5.3 years	¹⁴⁴ Ce	284.3 days
⁶⁵ Zn	243.9 days	¹⁴⁴ Pr	17.3 m

^aAlso formed via ternary fission.Source: Till JE and Grogan HA (2008) *Radiological Risk Assessment and Environmental Analysis*. 1st edn. Oxford University Press.**Table 2** Normalized release in liquid effluents from US reactors, 1988 (United Nations Scientific Committee on the Effects of Atomic Radiation, 1994).

Radionuclide	Normalized release (GBq per GW-year)	
	PWR	BWR
²⁴ Na	0.12	32
⁵¹ Cr	2.8	21
⁵⁴ Mn	1.9	5
⁵⁵ Fe	5.6	1.9
⁵⁸ Co	26.7	2.8
⁵⁹ Fe	0.22	1.7
⁶⁰ Co	7.6	11
⁶⁵ Zn	0.016	1.3
⁹⁵ Nb	0.56	0.016
^{110m} Ag	2.4	0.060
¹²⁴ Sb	0.85	0.018
¹²⁵ Sb	3.9	0.018
¹³¹ I	2.3	0.32
¹³³ I	0.75	0.047
¹³⁴ Cs	2.3	1.3
¹³⁷ Cs	4.6	2.9

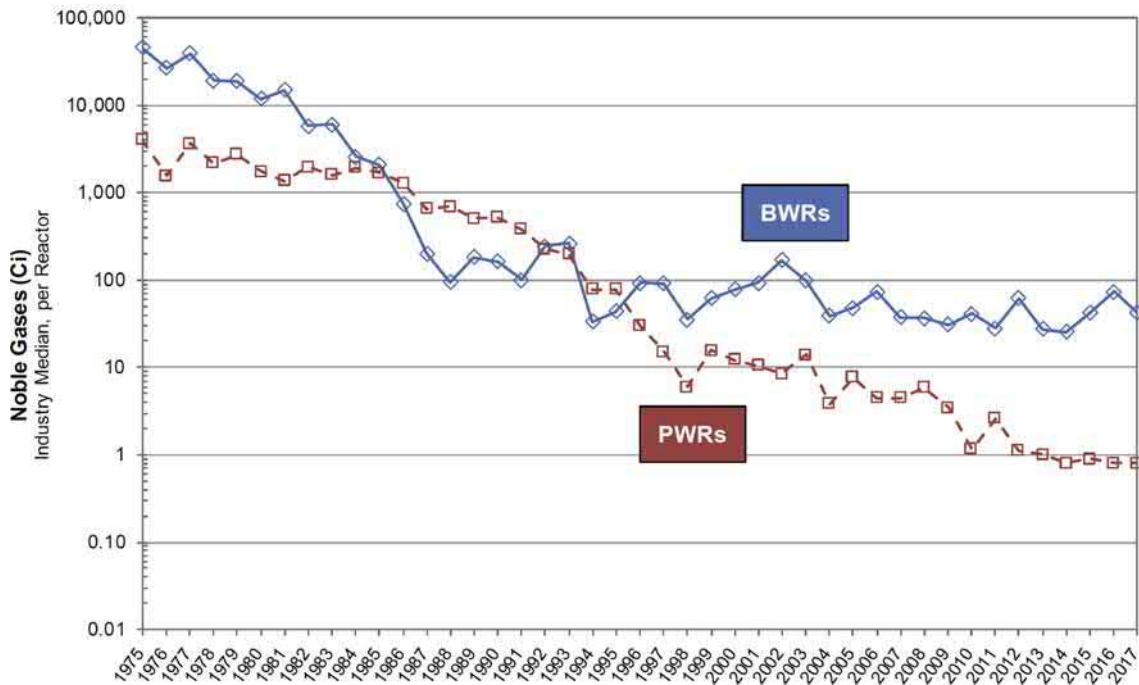


Fig. 2 Long-term trend in noble gas effluents from US reactors. Source: US NRC.

fuel pellets and the cladding themselves are barriers to fission product release. However, there is no such thing as a perfect manufacturing method and microfractures are present in the cladding. Furthermore, noble gases and tritium may diffuse through the cladding. But as Fig. 2 demonstrates, considerable improvements have been made leading to orders of magnitude decrease in noble gas effluent release. The NRC classifies radionuclides other than tritium, ^{14}C , and noble gases that are present in liquid effluents as mixed fission and activation products (MFAP). Fig. 3 shows a similar trend to Fig. 2 in the release of MFAPs from normal operation of PWRs and BWRs. Again, improved fuel integrity plays a part, but effective reductions have been made through the

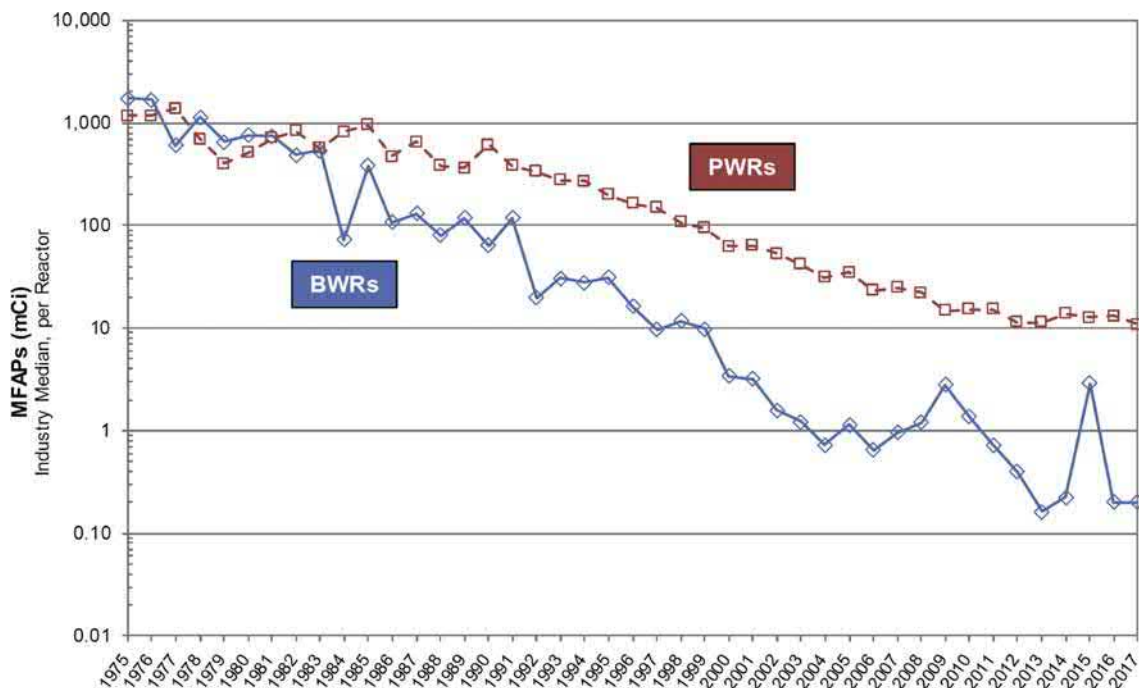


Fig. 3 Long-term trend in mixed fission and activation product effluents in liquid discharge from US reactors. Source: US NRC.

adoption of liquid effluent treatment systems including ion exchange resins and other filtration equipment (U.S. Nuclear Regulatory Commission, 2019).

It is clear from Fig. 3 that BWRs consistently release less MFAPs than PWRs. Here, the difference comes down to operation practices. BWR operators have adopted a zero liquid-release strategy whereby the reactor cooling water is reused with only tritium released as a gaseous effluent. This means that only the inventory of the MFAPs that is filtered out by cooling water treatment systems is released. Since PWRs use soluble boron in the primary coolant for management of excess reactivity, reuse of cooling water is not practical. Regardless, it is clear that there is a long-term trend showing reductions in the MFAPs released from routine operation of US reactors.

Monitoring of coolant consists of measurements of pH, any additives for corrosion control, and in case of PWRs, soluble boron concentrations. Regular records of measurements of radionuclide concentrations in the coolant provide benchmarks for source term analysis methods. The particulars of the activated corrosion products depend on the choice of materials. Some activation products have limited solubility in water and get deposited on various surfaces throughout the primary loop. Ever so often, these plated out activation products get dislodged and resuspended in the coolant loop leading to temporary spikes in the coolant radioactivity measurements. One example of such an activation product is ^{60}Co .

Radioactive gas treatment systems for BWRs implement delay lines while PWRs use gas holdup tanks that allow for the shortest-lived isotopes to significantly decay away. Longer term holdup times for BWRs are achieved through the use of activated charcoal beds where noble gases adsorb onto the surface of the charcoal. The details of volume, flow rates, holdup times, and overall effectiveness of such gaseous and liquid effluent treatment systems are important considerations when estimating the operational source terms from a given reactor plant. Several computer programs have been devised for making such calculations. The NRC code used for this purpose is called Gaseous and Liquid Effluent (GALE) with PWR and BWR-specific versions (Bangart et al., 1978, Chandrasekaran et al., 1985). The codes provide default inputs based on the ANSI/ANS-18.1-1999 standard for calculations when specifics are not available (American Nuclear Society, 1999). They also allow the user to adjust the parameters to match the specific performance characteristics of the reactor plant being analyzed for a more accurate description of the source release.

Radiation releases from LWR accidents

It is an unfortunate reality of the modern world that failures occur in engineered systems. Nuclear reactors, although shown to be one of the safest sources of energy (Wang, 2011), are not immune to accidents. Even the best designed systems cannot be fully immune to equipment failure. With human beings involved in design and operation, errors can occur which may lead to a scenario where radioisotopes are released to the environment. Some of these factors have a higher frequency of occurring than others. Unforeseen natural disasters can lead to a cascade of malfunctions leading to BDBAs. By their very nature, less frequent BDBAs are more difficult to characterize. In the case of DBAs, the damage extent can be characterized and planned for in advance. Calculations are made to estimate the magnitude of radiation releases and strategies, such as engineered safety features and operator guidance (Lutz and Williamson, 2021a,b,c) are implemented to minimize impact to the public and the environment.

There have been a relatively small number of accidents in the history of nuclear reactors that have led to significant radiation release. It should be noted that even though in total the cost of these accidents has been significant, the times where radiation posed a significant concern for public health are exceedingly rare. Furthermore, as emphasizes in Skillman and Rempe (2021), Sich (2021) and Mizokami and Rempe (2021), past events have increased our understanding of radiation release from an accident.

Estimating source release from an accident is more of a predictive science than a deterministic one. Even though the exact sequence of an accident is difficult to predict, the scope and risk from postulated events can be estimated with some uncertainty. With this in mind, regulatory bodies seek to ensure the safety of the public and to reduce any undue risk. As described in Bley and Malsch (2021a,b), US federal regulations that govern LWR are enumerated in Title 10 of the Code of Federal Regulations (CFR). In addition to the regulations, there are guidance documents for determining the consequences for DBAs as well as BDBAs. When requesting a license for any reactor, the designers must evaluate possible releases for selected PIEs, and then demonstrate that the facility design can adequately mitigate releases such that the regulatory limits are met. All analysis models and input assumptions must be supported with evidence to show that they reasonably capture the underlying physics of the radioactive release and accessible environmental transport pathways. Additional measures, such as backup safety systems, are required for defense-in-depth (Rempe and Williams, 2021). Part 100 of Title 10 CFR deals with reactor siting criteria and specifies requirements for the exclusion zone, low population zone and population center distance for a proposed plant location. Depending on the reactor site, the accessible pathways for environmental transport of radionuclides can be evaluated. Certain pathways are more accessible than others depending on the geology of the reactor site and the surrounding biosphere. Taking into account the available transport pathways, the exclusion zone is set such that any individual located at any point of the boundary would receive a maximum dose of 25 rem to the whole body (or 300 rem to the thyroid from exposure to iodine) 2 h post release.

As mentioned earlier, the fission and activation product inventory in nuclear reactors forms the bulk of the initial source term. The nature of the release source term which accounts for that fraction of the initial source term that volatilizes out of the fuel and is released to the biosphere depends largely on the specifics of the accident that occurs. Depending on the dose being calculated, certain radionuclides are more important than others due to their behavior in the environment as well as their propensity of being taken up by humans (EPA, 2017). ^{131}I is one such important radionuclide that is highly radioactive and is preferentially concentrated in the thyroid gland. ^{131}I can be combated by consuming potassium iodine that saturate the body with stable iodine and

reduce absorption of radioactive iodine. ^{90}Sr is an isotope of interest due its long biological half-life and preferential concentration in bones and teeth. Another important radioisotope for latent dose calculations is ^{137}Cs , which is an analogue to potassium in living cells, and can travel readily through the biosphere. These radioisotopes, in particular, pose significant health risks to the public and are of great interest when evaluating the release source term.

As discussed in [Sich \(2021\)](#), an International Nuclear (and radiological) Event Scale (INES) was developed by the International Atomic Energy Agency to rate the severity of nuclear accidents after the accident at Chernobyl Unit 4. There have been two INES level-7 events (highest level) in history: the event at Chernobyl Unit 4 (ChNPP4) and at Fukushima Daiichi (1F).

The Chernobyl accident was precipitated by human error coupled with a poorly designed reactor and an improper design of an experiment at low power. [Sich \(2021\)](#) provides details. It is widely considered the worst nuclear power plant accident and perhaps the worst possible accident. One of the design faults of the RBMK reactors deployed at Chernobyl is a positive void coefficient of reactivity. As reactor power and coolant temperature increased during a low-power experiment, the water coolant started to boil leading to an increase in reactivity and correspondingly increase in power. Eventually a runaway chain reaction developed leading to the explosion of the core. Nearly all power reactors are required to have a negative reactivity feedback, and a Chernobyl type reactor would not be licensed by any regulatory organization in other countries. The Chernobyl reactors also did not have a containment building which led to an unabated release of radioactivity into the environment. The exact amount is debated, but an estimated source term for the accident is given in [Table 3](#). The accident itself was kept secret until radioactive material had traveled through the atmosphere to other countries ([Sich, 2021](#); [United Nations, 2020](#)).

The disaster at the Fukushima Daiichi (1F) reactor plant, which included six BWRs ([Mizokami and Rempe, 2021](#)), was the result of a BDBA sequence resulting from a strong earthquake and a subsequent tsunami. With existing disruptions caused by the earthquake, the reactor plant was struck by an up to 15-m tsunami wave—well above that predicted by the plant design engineers. As a result, the backup diesel generators were rendered inoperable and the reactor cores of the three operating units at 1F were left without sufficient cooling water leading all three cores to their meltdown. At the greatly increased temperatures, the zircaloy cladding oxidized by reacting with the water/steam to form free hydrogen gas. The now oxidized cladding also released the contained volatile fission products. The over pressurization caused by the steam, hydrogen, and fission gas mixture forced operators to vent the gases which led to the accumulation of explosive concentrations of hydrogen in the upper part of the buildings in two of the operating units (1F1 and 1F3) and one building of a unit that was shutdown for refueling (1F4). Combustion of the accumulated hydrogen led to explosions that also dispersed volatile isotopes of iodine, cesium, and xenon into the atmosphere. [Table 4](#) shows

Table 3 Representative source term for the Chernobyl accident ([Metivier, 2002](#)).

Radionuclide	Activity released (PBq)
^{133}Xe	6500
^{131}I	~1760
^{134}Cs	~54
^{137}Cs	~85
^{132}Te	~1150
^{89}Sr	~115
^{90}Sr	~10
^{140}Ba	~240
^{95}Zr	196
^{99}Mo	> 168
^{103}Ru	> 168
^{106}Ru	> 73
^{141}Ce	196
^{144}Ce	~116
^{239}Np	~95
^{238}Pu	0.035
^{239}Pu	0.03
^{240}Pu	0.042
^{241}Pu	~6
^{242}Cm	~0.9

Table 4 Best estimate of atmospheric release source term for the Fukushima Daiichi nuclear accident ([International Atomic Energy Agency, 2015](#)).

Radionuclide	Activity released (PBq)
^{131}I	144 ± 36
^{133}Xe	502 ± 359
^{137}Cs	12.4 ± 2.5

an estimate atmospheric release source term from the Fukushima Daiichi nuclear power plant. Interestingly, there were several warnings issued by the IAEA and Japan's own regulatory bodies regarding the vulnerability of Japanese reactors to earthquakes and flooding from tsunamis (Bloomberg, 2012; Krolicki, 2011). However, the suggested measures were never implemented.

As described by Skillman and Rempe (2021), the Three Mile Island Unit 2 (TMI-2) accident is the only core melt accident in a commercial PWR in the United States where there was a release of radioactive material to the public. The accident resulted in a partial meltdown and release of relatively small quantities of radioisotopes above regulatory limits. It is widely agreed that there were no adverse health effects as a result of the accident. TMI is historically important because it showed that core melt accidents can occur in commercial PWRs and that the release mode and accident progression are important in estimating the release source term. As emphasized by Soffer et al. (1995), the quantity of radionuclides released were well below those predicted during licensing. Moreover, the initiating event at TMI-2 (a malfunctioning overpressure relief valve) had been previously observed at a different plant with the same reactor design (Drummond Ayres, 1979). The TMI accident led to many lessons learned (Skillman and Rempe, 2021), including the need for better information sharing among reactor operators. It also provided data for reducing uncertainties in source term release estimates.

Many computational tools are available to help reactor designers predict radiation release from a given reactor accident sequence. One such widely-used tool is MELCOR (MELCOR, 2017). It was developed by the NRC to predict accident progression and radiation releases. MELCOR can address a wide variety of accidents and provides reasonably accurate predictions of the radiation released from accident for which there is relevant prior knowledge. The chief drawback is that MELCOR is developed specifically for LWRs. The NRC is engaged in expanding MELCOR models for non-LWRs. However, data for validation of such models is limited. Likewise, developers wishing to license non-LWR reactor designs will have to use a different analysis tool which have not been rigorously benchmarked due to limited validation data. Perhaps, current codes could be expanded to become applicable to other types of reactor designs. Most upcoming advanced reactors (Stanculescu, 2021) are envisioned to be immune to some types of accidents encountered in LWRs. However, accidental radiation release is still possible, and the full scope of such accidents is not yet understood. There is a significant amount of research that is on-going to fully understand accidents in advanced reactors to limit their uncertainty. However, most, if not all, of the advanced reactor designs are expected to be safer than LWRs, but that is not to say that LWRs are not already very safe with a proven track record.

Used nuclear fuel

The fuel extracted from LWRs (and any other type of nuclear reactor) contains many of the fission products, and it generates a significant amount of decay heat. As a result, the fuel needs to be first stored in a way to dissipate heat. After this initial storage, the fuel can either be reprocessed and recycled or sent for long term storage. Each option has its own associated source term. In this section, the source terms from the storage of LWR spent nuclear fuel is explored.

Spent fuel from a reactor is intensely radioactive owing to the decay of fission and activation products and the accumulation of trans-uranium isotopes (actinides). These radioactive nuclei decay over time and eventually become stable. Fig. 4 depicts the normalized activity of spent nuclear fuel over time and highlights the important component isotopes that contribute to the radiotoxicity. In order to compare used nuclear fuel's activity or radiotoxicity with that of the uranium ore it comes from, the grade of ore and radioisotopes must be known. Therefore, these comparisons are not included in the article. Storing the spent fuel would ideally isolate these harmful radioisotopes from the environment. Regulatory requirements exist that limit the release amounts (NRC Title 10 CFR Part 72). To accomplish this isolation, several storage options have been developed. This section will look at the three largest: spent nuclear fuel pools, dry-cask storage, and geological repositories.

Spent fuel forms and waste streams for advanced reactors can diverge significantly, and the corresponding source terms will not be explored in this article.

Immediately following the removal of spent fuel, the shortest lived isotopes produce the most significant amount of heat. To properly disperse this heat, the spent fuel is stored on-site in pools of demineralized water. These spent nuclear fuel pools dissipate heat and provide radiation shielding to on-site workers. The water also consists of soluble boron to prevent nuclear criticality. When considering source terms from the spent fuel pools, the release rate is low but still present. There are multiple barriers against the release of radioactive material in a spent fuel pool. Most of the possible source material remains in the fuel inside the cladding. For high burnup (what would be typically seen in waste storage), the failure rate for cladding is about 1% (Ahn et al., 2012). From there, the majority of fission products that do escape remain in the water, but some like the gaseous fission products do escape over time. The failure rate can increase if the fuel assemblies are dropped during transportation (Kamas et al., 2006). Additionally, the cladding failure outside the pool can lead to oxidation (UO_2 to U_3O_8), which in turn, increases fission product escape from the stored fuel rods.

Following a cooling down period of 10 or more years, the spent fuel can be moved to dry cask storage, which are an increasingly used option for long-term spent fuel storage, the dry casks are stored on-site although they could be consolidated at a dedicated storage facility. In the dry cask made of steel and concrete, the spent fuel assemblies are put into a canister filled with inert gas before being placed into the cask. The steel and concrete surrounding the spent fuel provide radiation shielding. There has been no significant leak of radioisotopes observed from the spent fuel while in dry cask storage, and a leak does not seem likely in the foreseeable future. In the NRC report on the Holtec independent spent fuel storage facility, the conclusion was that the only significant source would be from gamma radiation that penetrate shielding (U.S. Nuclear Regulatory Commission, 2020).

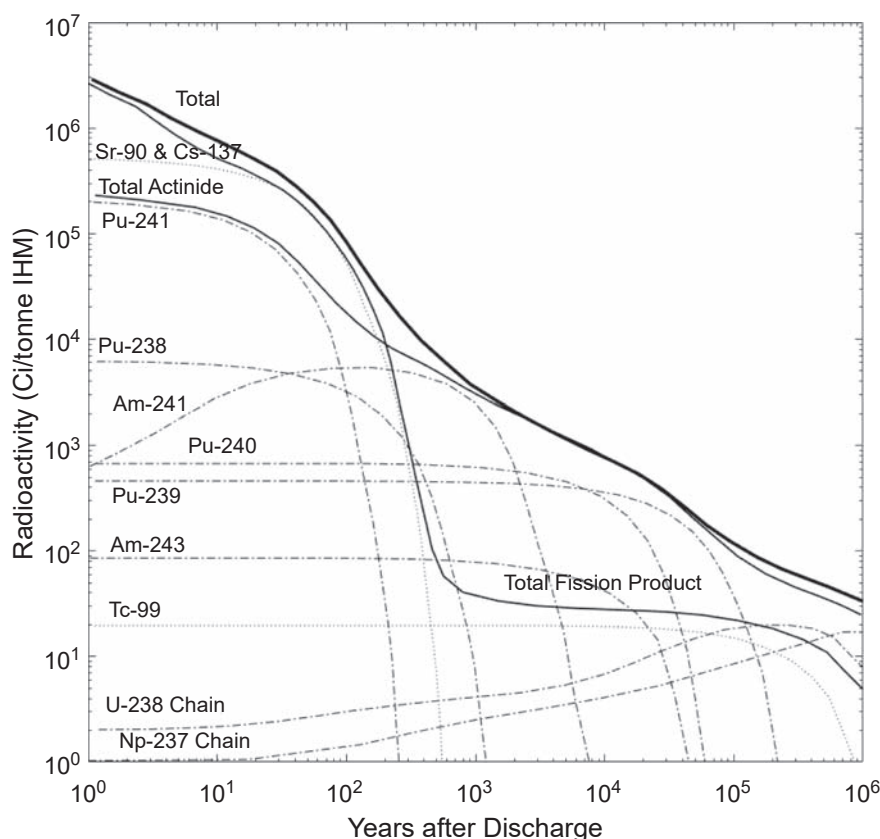


Fig. 4 The activity of radionuclides in spent nuclear fuel for a PWR with 4.5% enrichment and 50 MWd/kg. Source: digitized from Xu Z, Kazimi MS, and Driscoll MJ (2005) Impact of high burnup on PWR spent fuel characteristics. *Nuclear Science and Engineering* 151(3): 261–273.

Dry cask storage on the surface could be indefinite, but the current plans call for about 40 years of cask storage followed by longer-term geologic disposal. Such geologic disposal sites are designed with the purpose of creating extreme separation between the biosphere and the nuclear waste. The only major pathway that radioisotopes could take to travel back into the environment is through groundwater. A study of the KBS-3 canisters used at the Onkalo geological storage site in Finland found that there would be no significant radiological damage or corrosion of the canisters (Yang et al., 2019). Any breakage of the canisters would have to result from the slow movements of the Earth's crust. Such geological scale events are far longer than recorded human history, and if a site is chosen correctly, there would be little tectonic activity or groundwater penetration.

All told, each of these storage options has little chance of exposing the public to a significant dose of radiation. This is because the spent fuel has a predictable activity and the storage sites are fairly stable. The greatest risk of source release is from the transportation of the spent fuel since drops could lead to breakage and oxidation. However, there has been no significant release that has occurred during transportation of spent nuclear fuel (IAEA, 2018).

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External Dosimetry

Ken G. Veinot, Y-12 National Security Complex, Oak Ridge, TN, United States

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Introduction

The field of external dosimetry is focused on the determination of delivered doses to persons from radiation sources outside the body. This may include individual radioactive sources, man-made radiation (i.e., X-rays), radioactive contamination, cosmic rays, and other nuclear radiation sources. In general, the radiation is detected (measured) using detectors worn on or near the body and some interpretation is then made to convert the signal received by the detector to the dose received by various parts of the body. The system of external dosimetry is, in some ways, unchanged from when it was first conceived over a hundred years ago when nuclear radiation was first discovered. Technology has certainly advanced, but many of the assumptions, interpretations, and limitations remain. This article will introduce regulations and standards applicable to external dosimetry programs, discuss the philosophy of external dosimetry, review common available technologies, provide a review of dose coefficients used in the assessment of external dose, and develop a simple example of how a dosimeter may be used to determine doses delivered from a mixed radiation field and sources via a dosimeter algorithm.

Regulations, standards, and accrediting bodies

There are two primary regulatory drivers for personnel dosimetry in the United States—10 CFR Part 20 (US DOE, 1991) and 10 CFR Part 835 (US DOE, 2007). Part 20 applies to facilities regulated by the Nuclear Regulatory Commission, which includes commercial nuclear power sites, hospitals, universities, and commercial interests. Part 835 applies to Department of Energy sites. Each of these regulations places occupational dose limits on personnel—5 rem/year for the whole body, 15 rem/year to the lens of the eye, and 50 rem per year to the skin that includes the extremities (wrist, ankle, and finger). For both 10 CFR 20 and 10 CFR 835 the occupational dose limits are given in terms of the protection quantities; although, the quantities themselves are different between the two regulations. 10 CFR 20 follows the protocols of the International Commission of Radiological Protection (ICRP) Publication 26 including the use of effective dose equivalent while Part 835 was revised in 2007 to use ICRP Publication 60. So, there are inherent (and important) differences between the two. The differences between ICRP 26 and ICRP 60 are many, including the application of different methods for converting energy absorbed in tissue (dose) to a dose that takes into account different type of radiation and different types of tissue. These methods have changed over the past 120 years and so has terminology. Methods used in regulatory documents are not the same as those that are currently considered best practices among radiation protection professionals. For example, neutron quality factors recommended in ICRP 26 differ from those of ICRP 60. While both ICRP 26 and ICRP 60 use risk-based approaches for the protection quantities, there are differences in organ and tissue weighting factors, radiation weighting factors, and terminologies. As a result, individuals exposed to similar radiation fields under ICRP 26 and ICRP 60 protocols may be assigned slightly different doses; however, this is of no concern from a radiation risk perspective.

Applicable standards

ANSI N13.11

The American National Standards Institute (ANSI) published a standard, ANSI N13.11 (ANSI, 2009) that describes a procedure for testing the performance of whole-body dosimetry systems for monitoring personnel exposure to ionizing radiation. Performance testing is conducted under controlled conditions, and it includes irradiation with photons, beta particles, neutrons, and mixtures of these radiations. This standard provides the basis for whole-body dosimeter testing under the two primary dosimetry accreditation programs in the US, namely the Department of Energy Laboratory Accreditation Program (DOELAP) and National Voluntary Laboratory Accreditation Program (NVLAP) to be discussed in following sections. Under ANSI N13.11 a program may test under various categories, with each category representing radiation fields applicable to the site or program. Accident level doses are available for photon testing. For routine dose levels, the available categories include photons and photons mixtures (with energies ranging from 20 keV to 1.33 MeV), low and high-energy beta fields, photon and beta source mixtures. Photon fields include discrete energies as well as a variety of X-ray spectra. Neutron sources used for testing are unmoderated and D₂O-moderated ²⁵²Cf, and neutron and photon mixtures. ANSI N13.11 provides specific limits for mixtures categories. Testing results are evaluated using criteria described in the standard.

ANSI N13.32

While ANSI N13.11 applies to whole body dosimetry, a separate testing standard, ANSI N13.32 (ANSI, 2008), applies to extremity monitoring. The standard applies to dosimetry systems used to assess personal dose equivalent at a depth of 0.07 mm in International Commission on Radiation Units and Measurements (ICRU) tissue in extremities, specifically, to the performance of dosimeters worn on fingers and on wrists or ankles.

ANSI N13.32 testing categories include high and low-energy photons, high and low-energy beta particles (including ⁹⁰Sr, ⁸⁵Kr, and depleted uranium), and limited mixtures of these. At the present it does not include testing categories for neutrons.

Both N13.11 and N13.32 are strictly dosimeter performance test programs and do not contain guidance for the administration of dosimetry programs such as reporting, field calibrations and corrections, dosimeter placement on workers, or the use of multiple dosimetry. Requirements for some of these program aspects are provided in the accreditation programs DOELAP and NVLAP.

External dosimetry accreditation programs

There are two primary accreditation programs in the United States for dosimetry programs—the US Department of Energy Laboratory Accreditation Program (DOELAP) and the National Voluntary Laboratory Accreditation Program (NVLAP). These programs are briefly described in the following sections. Both accreditation programs follow the dosimeter performance testing protocols described in ANSI Standards N13.11 and N13.32.

National Voluntary Laboratory Accreditation Program (NVLAP)

The National Institute of Standards and Technology (NIST) Handbook 150-4 (Ho, 2019), NVLAP Ionizing Radiation Dosimetry, presents the technical requirements and guidance for the accreditation of laboratories under the NVLAP Ionizing Radiation Dosimetry Laboratory Accreditation Program. The laboratory accreditation program for Ionizing Radiation Dosimetry was established in 1984 in response to a request from the U.S. Nuclear Regulatory Commission (NRC). The purpose of the NVLAP Ionizing

Radiation Dosimetry Program is to recognize competent dosimetry processing laboratories and to improve the quality of personnel dosimetry by providing periodic evaluations of each laboratory, including an assessment of the laboratory's management system (Ho, 2019).

Accreditation is available to any laboratory that processes radiation dosimeters used to monitor individual exposure to ionizing radiation. A foreign-based laboratory may also be accredited by NVLAP if the laboratory meets the same requirements as domestic laboratories and pays any required additional fees associated with the on-site assessment (Ho, 2019).

Department of Energy Laboratory Accreditation Program (DOELAP)

The Department of Energy (DOE) implemented the DOE Laboratory Accreditation Program (DOELAP) for external dosimetry in 1986. The objectives of the DOELAP external dosimetry program is to assure the competency of dosimetry measurements through calibration inter-comparisons, performance testing, on-site assessments; and to encourage applied research in areas where there is a technology shortfall. DOE also expects the program to enhance cooperation and technical information exchange among its sites and facilities to provide a more standardized and uniform radiation dosimetry capability. DOE sites and facilities are expected to use standards and other technical guidance from the Department to ensure that the performance of personnel dosimetry programs are adequate to meet the requirements of 10CFR835 (US DOE, 2018).

DOELAP accreditation involves performance testing of dosimeters and the documentation of program elements important to the long-term quality assurance of a dosimetry program and its ability to accurately measure, record, and report occupational whole body and extremity dose (US DOE, 2018). On site assessments are conducted triennially by dosimetry program experts who are selected from participating organizations. These assessors undergo recurring qualification training conducted by the DOELAP administrators.

Quantities, calibrations, and phantoms

The foundation of a dosimetry program is its ability to accurately and reliably determine the dose delivered to a person. Since the dose is not measured in the person, a dosimeter is used as a surrogate and this dosimeter must be referenced against, and traceable to, a national standard. The standard of radiation dose is provided by a calibration facility that performs the dosimeter irradiations using methods described in the relevant ANSI standards. For testing purposes the methods of irradiating dosimeters, the sources and phantoms to be used are described in the relevant ANSI standards. Calibration facilities follow strict protocols to ensure the quality of exposures are acceptable. The two primary categories applicable to external dosimetry are whole body and extremity. The radiation protocols for each are briefly described below.

Whole body

For purposes of dosimeter calibrations to obtain operational quantities, the dosimeters must be placed on a phantom that provides a reasonable approximation to the backscatter properties of the part of the body on which the dosimeter is worn. The phantom used for whole body testing is a polymethyl-methacrylate (PMMA) rectangular slab with dimensions $30 \times 30 \times 15$ cm thick for photons and neutrons. Backscatter measurements (when a source particle scatters from material external to a detector back into a detector) are used to verify adequacy for photon irradiations. A $40 \times 40 \times 15$ cm phantom is sometimes used for neutron dosimeter testing. For beta irradiations the same area is used, and the thickness must be at least 5 cm. Specifications are provided in the standard for dosimeter mounting and for minimum irradiation distances. For photon and neutron testing, dosimeters are not mounted near the phantom edges to eliminate radiation scattering effects for these positions.

Whole body irradiations are performed in order to test a dosimetry system's ability to calculate reference doses. For each of these measurements, methods for converting the measured air kerma ($\text{Kinetic Energy Release per unit MAss}$) to the operational quantities of interest are provided in the ANSI standard. Thus, the irradiation facility measures the air kerma and from this measurement applies the conversion to arrive at the reference dose for the quantity of interest such as personal dose equivalent.

Extremity

There are two phantoms used in the extremity dose testing—the rod to simulate the ankle or wrist and the pillar to simulate the finger. The rod phantom is a solid, right circular PMMA cylinder with a diameter of 19 mm while the arm phantom is a solid, right-circular PMMA cylinder with a diameter of 73 mm. Both the rod and pillar phantoms are 300 mm in length. Testing categories include: high-dose photons, photons with energies between 20 keV and 1.33 MeV, high and low-energy betas, depleted-uranium slab, and mixed fields, where testing may be blind or non-blind. Many extremity dosimeters contain a single element so only a single correction factor is required to convert the signal on the dosimeter to dose. Since only a single factor is used, the reference field must be known (to the test participant) so the appropriate correction factor may be applied.

For both the whole body and extremity testing programs, it is the responsibility of the individual site to determine the categories to be tested. These should be based on the radiation fields encountered at the site.

Dose coefficients

Overview

Dose coefficients (sometimes referred to as dose conversion coefficients) relate the incident fluence (total path length of all particles that have passed through a unit volume) of particles to a corresponding dose quantity in tissue. The coefficients that convert fluence to dose vary as a function of incident particle energy, and they are computed using representative phantoms. The complexity of some of these phantoms has evolved as computing capabilities has increased. Consequently, the ability to more accurately calculate delivered doses to reference phantoms has dramatically improved. Early phantom models were commonly defined in simple geometric shapes such as spheres or cylinders. For operational quantities, these simple phantoms are still used. During the past several decades, phantoms designed to obtain protection quantities for individual organs and tissues have become more realistic with increasingly sophisticated mathematically described phantoms. More recently volumized-pixel (voxel) phantoms have been created based on high-resolution medical images of humans. The latest recommendations from the ICRP utilizes these phantoms for determining radiation protection quantities (ICRP, 2007, 2009, 2010). A full discussion on the development, calculation, and variabilities of these coefficients is too lengthy to be included here. A brief review of their history, calculation, and use is provided for the convenience of the reader.

Protection or operational quantities (doses of various types) are calculated based on physical specifications that describe the radiation field and on dose coefficients applicable to a particular radiation field and to a specified physical environment. For practical applications, conversion coefficients applicable to a field quantity (fluence or air kerma) are converted to the operational quantity based on methods appropriate for the reference phantoms (ICRP, 1996, 2007, 2009, 2010; ICRU, 1998).

Sets of dose coefficients were published by ICRU in Reports 43 (ICRU, 1988) and 47 (ICRU, 1992), and jointly with the ICRP in ICRU Report 57 (ICRU, 1998) and ICRP Publication 74 (ICRP, 1996). In Publication 103 (ICRP, 2007), the ICRP reviewed the protection quantities, including effective dose, originally introduced in ICRP Publication 60 (ICRP, 1991). Dose coefficients that transfer physical quantities to the revised protection quantities were published in ICRP Publication 116 (ICRP, 2010). These dose coefficients were computed for a wide range of particles (including those occurring only in cosmic radiation and at particle accelerators) and for an energy range reaching, in some cases, up to 200 GeV. The report identified energy ranges for photons and neutrons where the operational quantities introduced in ICRU Reports 39 (ICRU, 1985) and 51 (ICRU, 1993) deviated significantly from the protection quantities.

Operational quantities rely on phantoms for their definition and for the calculation of dose coefficients (i.e., converting field quantities to the personal dose equivalent). The phantoms and the methods of calculation introduce inconsistencies to the system of dosimetric quantities for radiation protection. Most recently, the ICRP and ICRU have modified the operational quantities to more realistically approximate the corresponding protection quantity. It was noted that the evaluation of personal dose equivalent at a single, defined depth does not reflect the geometric complexity of the human body with numerous organs implicitly present in the definition of the protection quantities. There are some discrepancies when using quality factors for operational quantities and radiation weighting factors for protection quantities. Also, the personal dose equivalent is defined at a depth, d , at a specific location in the body, but for the calculation of dose coefficients simple geometrical bodies like slabs, cylinders and rods are used. The lens of the eye and the local skin have been monitored with a dose equivalent quantity for deterministic effects, which depend, to our best knowledge, on absorbed dose. The most recent guidance is that absorbed dose without radiation quality modifiers should be used for these quantities. Historically dose coefficients for operational quantities have been calculated and published using the kerma approximation. This assumption does not consider the consequences of charged particles created in various reactions from uncharged radiations and can lead to overestimates of dose. Instead, kerma calculations were employed whereby all energy from reactions were assumed to be deposited at the location where the interaction occurred. This assumption neglects the importance of charged particle equilibrium (CPE). For photons in tissue, CPE is lost at about 3 MeV at a 1 cm depth, at about 700 keV for a 0.3 cm depth, and around 65 keV at a depth of 0.007 cm (Veinot and Hertel, 2011). For neutrons, similar discrepancies exist, predominantly at intermediate and high energies (Veinot et al., 2020).

Dosimeters

Thermoluminescent dosimeters

Thermoluminescent dosimeters (TLDs) are among the most widely used for personnel dosimetry and the most common thermoluminescent (TL) materials include LiF, CaF₂, Li₂B₄O₇, and CaSO₄. Each of these materials is mixed with small impurities called dopants. The dopant materials provide lattice sites in the crystalline structure that can be occupied by electrons. As will be discussed later, an advantage of LiF over the other materials is its near tissue equivalent absorption and scattering properties, resulting in a close response to the dose equivalent response in tissue for photons.

Trap formation

The scintillation (light emission) process depends on energy states that are determined based on a material's crystalline lattice structure. In insulators and semi-conductors only discrete energy states are available for electrons. Bound electrons are maintained at the lower energy band (called the valence band) while higher energy state electrons are found in the conduction band. Conduction band electrons can migrate through the material creating current. The region between the conduction and valence bands is called

the forbidden band and, in pure crystals, electrons are not allowed to exist in this region. The forbidden band separates the conduction and valence band by a distance (energy) called the band gap. If an electron in the valence band absorbs energy equal to or greater than the band gap energy, it can be elevated to the conduction band. Once this energy is expended it returns to the valence band and emits a photon in the process (Attix, 2008). Typically, this band gap energy is beyond the visible light range where the wavelength of light can be found from the photon energy (Eisberg and Resnick, 1985):

$$E_{\gamma} = h\nu \quad (1)$$

where E_{γ} is the photon energy, ν is the frequency, and h is Planck's constant.

Additional energy states are allowed when impurities (called dopants) are added to the material. Valence band electrons can be elevated to one of these activator energy states creating electron-hole pairs of energy less than the band gap energy. These electron-hole pairs may have recombination energies that are in the visible light range. The de-excitation of these trapped electrons has a low probability at room temperatures because the distance between the trapping centers and the conduction band is too large, and the energy created by incident radiation remains trapped. As more energy is deposited additional traps are formed within the TLD. The characteristics of the TLD are determined by the materials and dopants (activators) used to construct the dosimeter. A schematic showing the energy levels, conduction and valence bands, and band gap is shown in Fig. 1.

Glow curve emission

The probability that an electron will escape an activator trap site is governed by the Boltzmann equation:

$$P = Se^{\left(\frac{-E}{kT}\right)} \quad (2)$$

where S is a normalization constant, E is the trap depth (or the energy of the trap below the bottom of conduction band), T is the temperature, and k is the Boltzmann constant ($8.617 \times 10^{-5} \text{ eV K}^{-1}$) (Eisberg and Resnick, 1985; Knoll, 1989). As the temperature increases additional energy is available to the trapped electron and the probability of escape from the trap increases. Adding additional energy to the system via heating accelerates the process as deeper traps are emptied as the temperature increases. Most TLD materials have multiple activator energy states and simply heating to a single temperature may not empty all available traps. As the TLD is heated to higher temperatures, additional higher energy traps will empty and a characteristic glow curve for the material is produced. As the temperature is maintained the total probability of trap depletion increases (Knoll, 1989). For LiF TLDs doped with magnesium and titanium the energy band gap is approximately 10 eV (Horowitz, 1984) with the activator energy states having de-excitation energies between approximately 140 °C and 300 °C.

Most TLD readers will heat the material at a constant rate to some maximum temperature and maintain this temperature for a few seconds. For an equal absorbed dose in the TLD, the intensity of the glow curve (light output as a function of time) is dependent on the material characteristics for a given heating cycle. The energy released during a particular heating cycle depends on the availability and energy depth of available traps as well as the release rates of those traps. Integration of the glow curve over the heating time produces the total signal. Since the heating rate and maximum temperature control the release of trapped electrons it is important that this time-temperature profile be maintained and reproduced to ensure consistent results.

Energy transfer rate dependence

As ionizing radiation enters material, it interacts through various mechanisms with the material creating electron ionization events. These energy transfers may result in free electrons (and their associated positively charged atomic nuclei), kinetic recoil energy of nuclei, or the creation of new nuclei as in the case of the ${}^6\text{Li}(n,\alpha){}^3\text{He}$ reaction. The available traps formed in the band gap are energy dependent and traps are formed at various energy levels beneath the conduction band. Since the traps have varying energy states, deeper traps require more energy transfer to an electron-hole pair. As a result, the number of traps filled that produce electron-hole

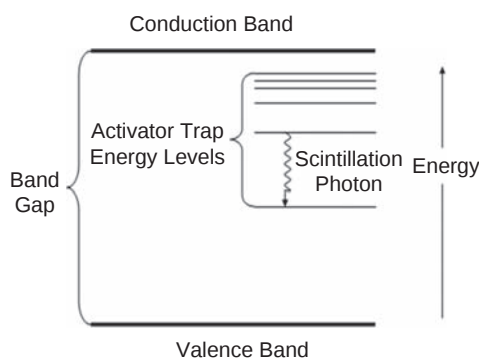


Fig. 1 Energy band structure of TL material. The band gap is shown as the energy difference between conduction and valence bands. The activator traps are shown as discrete energy states within the band gap. Scintillation photons are produced during de-excitation to lower energy states.

pairs depend not only on the quantity of incident radiation, but also on the ionization rate and average energy transfer per ionization event of the incoming radiation (Horowitz, 1984). Incident radiation that produces higher ionization energies allows deeper (higher energy) traps in the TLD to be created. Consequently, higher temperatures during heating are required in order to impart sufficient energy to release these traps.

Fade

Based on Eq. (2), it is apparent that filled traps in thermoluminescent material (doped with activators) are not permanent. In fact, over time some traps will release their energy and the electron-hole pairs recombine as a result of instability or thermal stimulation. For some materials, exposure to light can increase the signal loss. Each of these has the effect of reducing the net signal produced during a read and is referred to as fade. The fade properties of a TL material is determined by the TLD makeup including dopants, length of time since anneal, length of time since irradiation, and environmental factors. Since the activator energy traps have varying energy states, the probability of fade is energy dependent. More precisely, traps with lower energy states have a higher probability of recombination due to fade than deeper traps. For practical purposes, it is generally assumed that peaks with short half times (less than a few days) will have decayed to negligible values before the processing of personnel dosimeters is performed. This may not be the case in emergency situations where a dosimeter is retrieved and processed in a matter of hours. This overall impact of signal loss is important and should be quantified by performing fade studies for the particular dosimeter material. The fade is dependent not only on post-irradiation time, but also on the time between anneal and irradiation since the maximum number of available traps are found soon after anneal. Fade studies should consider both determinants. For acute irradiations, the fade can be described as a series of terms having the form:

$$F(t_{pre}, t_{post}) = C_1 e^{\left(\frac{-\ln(2)}{\lambda_1} t_{post}\right)} \left[C_2 + C_3 e^{\left(\frac{-\ln(2)}{\lambda_2} t_{pre}\right)} \right] + C_4 e^{\left(\frac{-\ln(2)}{\lambda_3} t_{post}\right)} \left[C_5 + C_6 e^{\left(\frac{-\ln(2)}{\lambda_4} t_{pre}\right)} \right] + \dots \quad (3)$$

where λ_n is the decay probability of trap n , t_{post} is the time between exposure and read, t_{pre} is the time between anneal and exposure, and C_n are constants determined by the material construction. The number of terms in Eq. (3) will also be determined by the material construction since each term is describing the decay half times of individual traps in the activator band. The fade of different dosimeters should be similar provided they are constructed of the same TL material and dopant (i.e., LiF doped with copper, for example). Different TL materials will have different trap structures and therefore different fade characteristics. In fact, dosimeters of the same TL material composition, but different dopants, may have different fade characteristics.

Background

Background signal accumulation will occur immediately following anneal. This signal on the TLD is a result of background radiation and is commonly assumed to increase at a consistent rate. Environmental factors such as heat, optical stimulation, tribological effects, and other sources can lead to additional signal (traps being filled) in TL material. In general, this effect increases with time after the TLD is annealed. To account for this in practice, an understanding of the accumulation of background signal is necessary.

Since the TLD is an integrating dosimeter, accumulated signal should increase with time. In reality, effects from fade and non-uniform exposures must be considered, and the exposure pattern would be expected to vary, depending on a number of variables. In order to arrive at a reasonable approximation of expected signal over time, background studies are performed. These studies generally involve the placement of dosimeters in various storage areas. They are not stored in process areas nor where they would receive occupational exposures from various sources. After storage the dosimeters are retrieved and read. Often a background accumulation rate is determined under the assumption that the signal is imparted to the dosimeter at a constant rate.

Glow curves

After irradiation thermoluminescent dosimeters are read by heating the TL material. The plot of light intensity as a function of temperature (the glow curve) is integrated to determine the relative dose delivered to the dosimeter material. The light emission produced by heating is detected using a photomultiplier with appropriate optical filters to ensure detection of thermoluminescence while rejecting spurious light from heating of the surroundings or the substrate. Signal processing electronics record glow curves and perform data analysis to produce computations of the dose delivered to the TLD.

Heating of the TLD can be accomplished using a resistance heater, flash bulbs, or gas. Thermocouples are typically used to verify that the correct heating rate is applied to the material since this will affect the emptying of trapping centers. An example of a glow curve produced by a LiF TLD exposed to a $^{90}\text{Sr}/^{90}\text{Y}$ source is shown in Fig. 2. For this example the TLD was pre-heated to 50 °C then the heating was increased at a rate of 25 degrees per second to a maximum of 300 °C. The 300 °C temperature was then maintained for 3.33 s. This glow curve is representative of a TLD that has been exposed to beta and/or photon radiations.

The relationship between trap formation and the type of radiation (such as photons, betas, alphas, and neutrons) impacts the nature of the glow curve. For example, photons transfer energy to material at a lower rate than do alpha particles or neutrons (that produce charged particle reaction products or recoil charged particles). This difference in the rate of energy transfer is often characterized by a term referred to as Linear Energy Transfer (LET). Photos and betas transfer energy at a much lower rate than alphas and neutrons and are referred to as low-LET particles. Alpha particles and neutrons are referred to as high-LET particles. LiF TLDs enriched in ^6Li that have been irradiated with neutrons produce alpha particles and tritons created in the $^6\text{Li}(n, \alpha)^3\text{He}$ reaction, which are

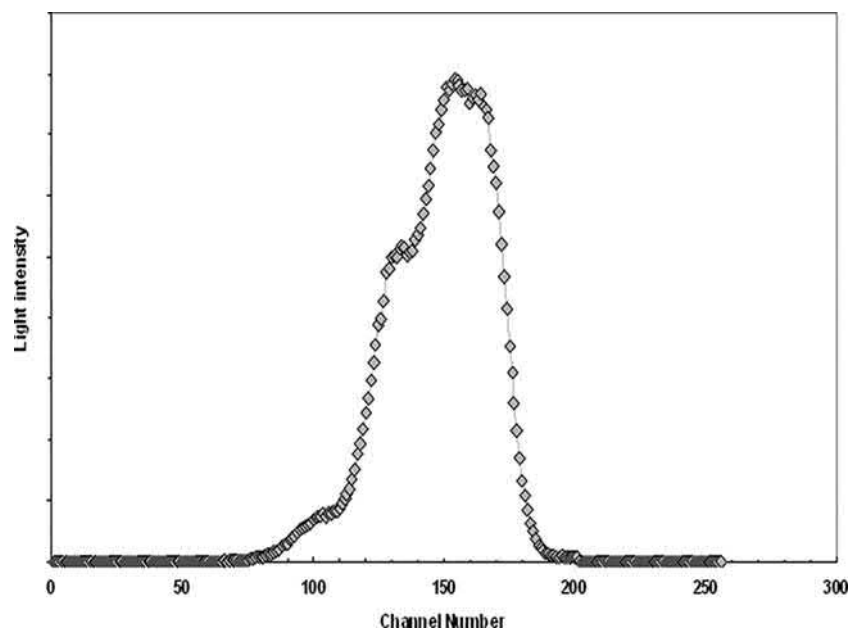


Fig. 2 A representative glow curve of LiF exposed to $^{90}\text{Sr}/^{90}\text{Y}$.

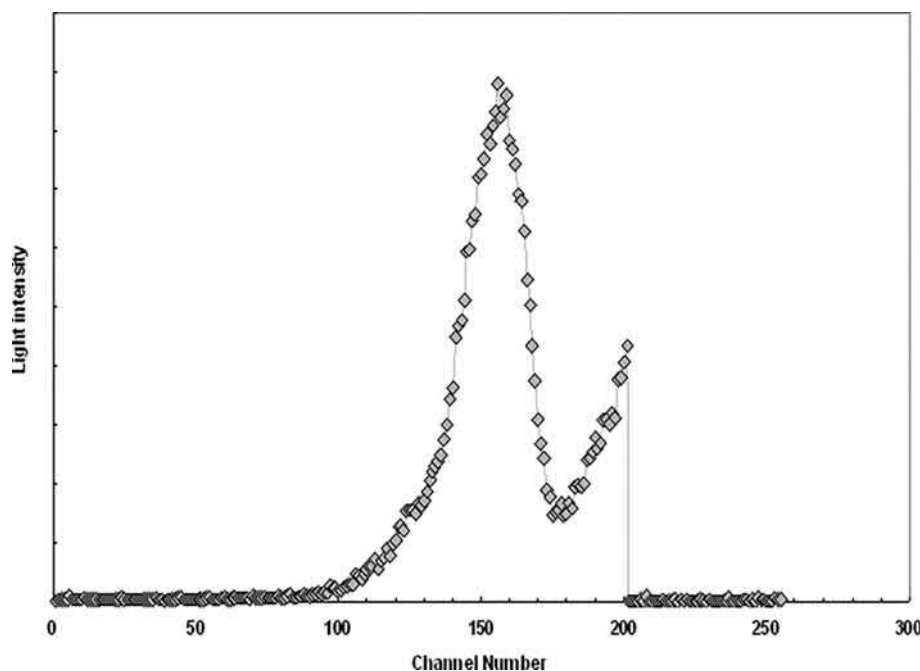


Fig. 3 Glow curve of LiF TLD enriched in ^6Li after irradiation by neutrons.

high-LET particles and lead to more efficient creation of deep traps compared to gamma or beta radiation. Glow curves of these elements indicate traps that are released at high temperatures as illustrated in [Fig. 3](#). For this readout the same temperature profile was used as for the glow curve shown in [Fig. 2](#).

Dosimeters will show some non-linearity of response as a function of dose. At high doses (above 1 Gy for LiF) the response becomes supralinear and must be adjusted to correct for this non-linearity.

Different TL materials exhibit varying responses to photons as a function of energy because of their material composition, trap structure, and interaction mechanism. At low energies LiF has a response that is nearly tissue-equivalent which is advantageous from a dosimetric perspective, while other materials such as CaSO_4 , Al_2O_3 , and CaF_2 over-respond to low energy photons. The differences in the responses of the various materials are due in part on the dependence of photon absorption upon the effective atomic number Z_{eff} and the intrinsic sensitivity of the recombination centers in the crystal structure. The effective Z of LiF is 8.2 while that of tissue is 7.4.

Neutron TLDs

The LiF dosimeters typically issued to personnel for beta/gamma monitoring are enriched in ^7Li , which has a very small response to neutrons. The ^6Li isotope, however, has a strong neutron response with a thermal cross section of 941 b. For personnel that work in neutron fields, the TLD dosimeter contains a ^6Li -enriched chip and a ^7Li -enriched chip. The photon response of the ^6LiF and ^7LiF are similar over a wide energy range (Veinot et al., 1998) so signal on the ^7LiF is subtracted from signal on the ^6LiF to leave neutron signal. The energy response of neutron-sensitive elements can be calculated and has been reported for common dosimeters (Veinot and Hertel, 2001, 2005).

You may notice in Fig. 3 that the tail at high channel numbers (where the 300 °C temperature is being maintained) the deeper traps within the TLD are liberated. The probability of trap release at a given temperature is constant as a function of time and this tail will continue to increase until a statistically significant number of traps have been liberated. At that point the tail will decrease since fewer and fewer traps are available. In this case the glow curve capture stopped at channel 200.

For ^6Li and ^{10}B -based TLDs the response of the TLD decreases as the neutron energy increases (since the interaction cross section has $1/v$ energy dependence) while the dose coefficients tend to increase as a function of neutron energy. This negative correlation between TLD response and dose equivalent must be accounted for when determining personnel neutron exposures from TLD results. To appropriately modify the TLD results for spectral dependence correction factors are applied. The spectral correction factors (SCFs) must be defined for various neutron spectra, although often the specific spectrum is not known. In practice SCFs are determined experimentally by first measuring the dose rate in a neutron field via multisphere detectors and unfolding the field spectra (Bramblett, Ewing, and Bonner, 1960; Sweezy et al., 1998) or through the use of single-sphere "Rem-balls" used for neutron surveys (Veinot, 2008). These dose rate measurements are compared to the measured responses of TLDs by exposing the dosimeters (mounted on a suitable phantom) to the same field.

Extremity dosimeters

Extremity dosimeters are worn when there is an indication that the largest portion of dose equivalent will be received in a particular area such as the fingers or other extremities. Many extremity dosimeters use thermoluminescent detectors that are mounted as part of a plastic ring or wrist band. The dosimetric characteristics of extremity dosimeters are similar to those of the TL detectors described above. For ring dosimeters a single element is often used. For wrist/ankle dosimetry multi-element dosimeters are sometimes used.

Optically stimulated luminescence dosimeters

In some ways optically stimulated luminescence (OSL) dosimeters are similar to TLDs. Both detector types are composed of doped crystalline materials, and both produce visible light with intensity proportional to the delivered dose. One of the most common commercially available materials used as an OSL dosimeter is aluminum oxide. OSL dosimeters must be enclosed in light-tight holders since they are sensitive to visible light. As opposed to TLDs that are heated to produce luminescence, OSL dosimeters are stimulated using light of a specific frequency. The laser light source is of a different frequency than the signal emitted by the OSL. The stimulation light has a wavelength of about 530 nm while the emitted light has a mean wavelength between 370 and 420 nm. Since the wavelength of stimulating light is somewhat near that of the emitted signal, optical filters are used to discriminate the light and only the emission signal is passed to the photomultiplier for amplification and processing. The energy deposited in the OSL by the laser stimulates trapped electrons that can then combine with trapped holes at lattice sites within the crystal known as F-centers (McDonald, 2019), and therefore produces a signal.

The effective atomic number (Z) for Al_2O_3 is about 10.2 so it over-responds (relative to tissue) to lower photon energies where the photoelectric cross section dominates the overall response. Metal filters can reduce this over-response and are often used in dosimeter holders. This over-response is evident as shown in Fig. 4 which compares the response of OSL and LiF TLDs to photons (Mobit et al., 2006).

OSL detectors are insensitive to neutrons, so neutron monitoring requires the use of other dosimeters such as ^6LiF or track etch detectors (to be covered in the next section). The use of radiators (which converts one type of radiation into another) placed on top of the sensitive OSL elements has been attempted to allow the OSL dosimeters to be used for neutron monitoring, but these tests showed little advantage over the use of a separate neutron-sensitive dosimeter (Passmore and Kerr, 2011).

An advantage of OSL dosimeter is the ability to stimulate only a small portion of the sensitive element so only the illuminated portion is processed while the remaining sensitive element retains its signal. This method is often used to "pre-screen" a dosimeter to determine if, for example, a high exposure occurred and a priority read is to be performed. The remaining sensitive element is unaffected and can be read at a later time with little or no signal loss. This ability can be particularly beneficial for dose investigations and dose reconstruction efforts. Of course, this requires the dosimeter be a one-time use since if it was re-issued to a worker an additional signal would be imparted on the dosimeter that would mask the original signal. Thus, a new dosimeter must be issued at each monitoring cycle which may be cost prohibitive if the pre-screening option is used.

Multiple measurements of OSL dosimeters, using pulsed OSL (POSL) stimulation or continuous wave stimulation (CW-OSL) have been reported. Up to 15 independent measurements have been demonstrated for the POSL technique and up to 10 for CW-OSL. At doses of 10 mSv there was a reported loss of signal of less than 3% (McKeever and Moscovitch, 2003).

Other advantageous characteristics of optical stimulation are the relatively high luminescence efficiency, stability, speed of readout, and no requirement for the application of heat to process the dosimeters. The high efficiency of OSL is partly due to the ability to read the dosimeters at room temperatures. Using POSL a very short read time on the order of a few hundred

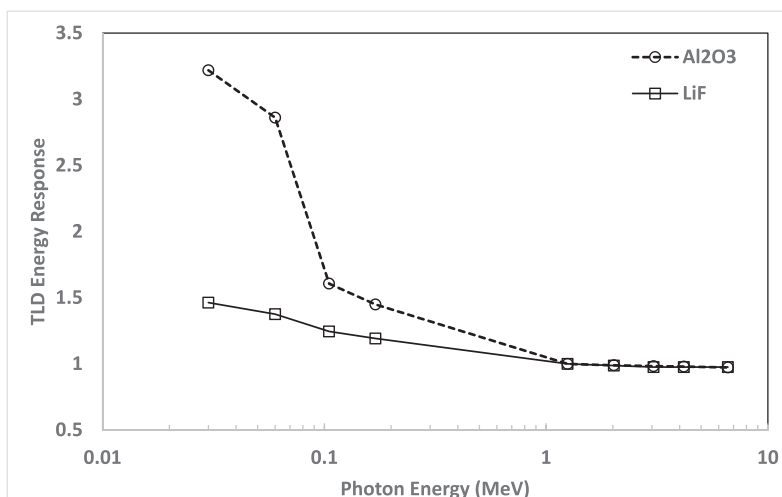


Fig. 4 Response of Al₂O₃ OSL as a function of photon energy compared to LiF TLD. Data from Mobit PN, Agyingi EO and Sandison GA (2006) Comparison of the energy-response factor of LiF and Al₂O₃ in radiotherapy beams. *Radiation Protection Dosimetry* 119(1–4): 497–499.

milliseconds is typical (McKeever and Moscovitch, 2003). With these short read times, a high throughput can be achieved by OSL processing centers.

Thermal quenching is the loss of luminescence efficiency as the temperature of the material is increased. Since TLDs must be heated to be read, the process of reading the dosimeters leads to a lack of sensitivity. In general, this effect is negligible for modern materials and doses of interest. OSL suffers from thermal quenching since the efficiency of luminescence emission from F-centers decreases as the temperature of the material is raised over the range from approximately 150 °C to 250 °C. Since OSL dosimeters are stimulated using light at room temperature (where luminescence efficiency is a maximum) this effect is minimized (McKeever and Moscovitch, 2003).

Track-Etch (CR-39) dosimeters

Neutron scattering is a primary energy deposition mechanism of fast neutrons. This is particularly true in low- Z materials, especially those with high hydrogen content, such as tissue. On average, the energy deposited by scattered neutrons in hydrogen is one-half the incident neutron energy (Veinot, 2019). Plastics and polycarbonates contain high levels of hydrogen in various chemical forms. One particularly useful from is allyl-diglycol polycarbonate which is marketed as CR-39. When fast neutrons undergo scatter reactions in CR-39 the molecules are damaged along the track of the recoiled particle. When exposed to caustic solvents, such as KOH and NaOH, the CR-39 is etched at a certain rate based on several parameters including the concentration of the etching compound and temperature. The damaged tracks, however, are etched at a faster rate than the undamaged CR-39 leading to the formation of pits in the plastic. These etch rates are referred to as the track etch rate, ν_t , and the bulk etch rate, ν_b . As the CR-39 is exposed to higher neutron fluence the pit formation increases. By counting the number of pits after etching the incident neutron fluence can be inferred. The etching process can be enhanced through the application of an electric field during etching.

Neutrons do not cause direct ionizations by themselves but by secondary particles resulting from scattering between them and target nuclei. Most tracks are produced by recoils of particles from scattering reactions with hydrogen, carbon and oxygen nuclei. Tracks can also be induced by charged particles created via (n, α) reactions (Milenkovic et al., 2011). Proton recoils are the most predominant and their energy strongly depends on the angle of collision following the laws of mechanics. The recoil protons are emitted in different directions and their latent tracks are oriented randomly in all directions. Some tracks will be etched from the point where the particle was created, while others will be etched from the point where the particle was stopped or from where it left the CR-39 detector. The first case is called direct etching, because etching was in the direction of particle motion and the second is called reverse or opposite etching (Milenkovic et al., 2014).

Not every recoil proton will be able to produce a readable track. The creation of visible tracks (those tracks that will be able to read after processing) is dependent on the incidence angle of the neutron relative to the detector surface. The track etch rate (ν_t) relative to the detector surface normal must exceed the overall bulk etch rate (ν_b) for a readable track to be produced. The critical angle, θ_c , is dependent, among other variables, on the etching process, chemical concentrations, electric field strength (for electrochemical etching), and the equipment used to read the tracks. The critical angle is generally defined as

$$\theta_c = \arcsin \frac{1}{\nu} \quad (4)$$

where $\nu = \nu_t / \nu_b$

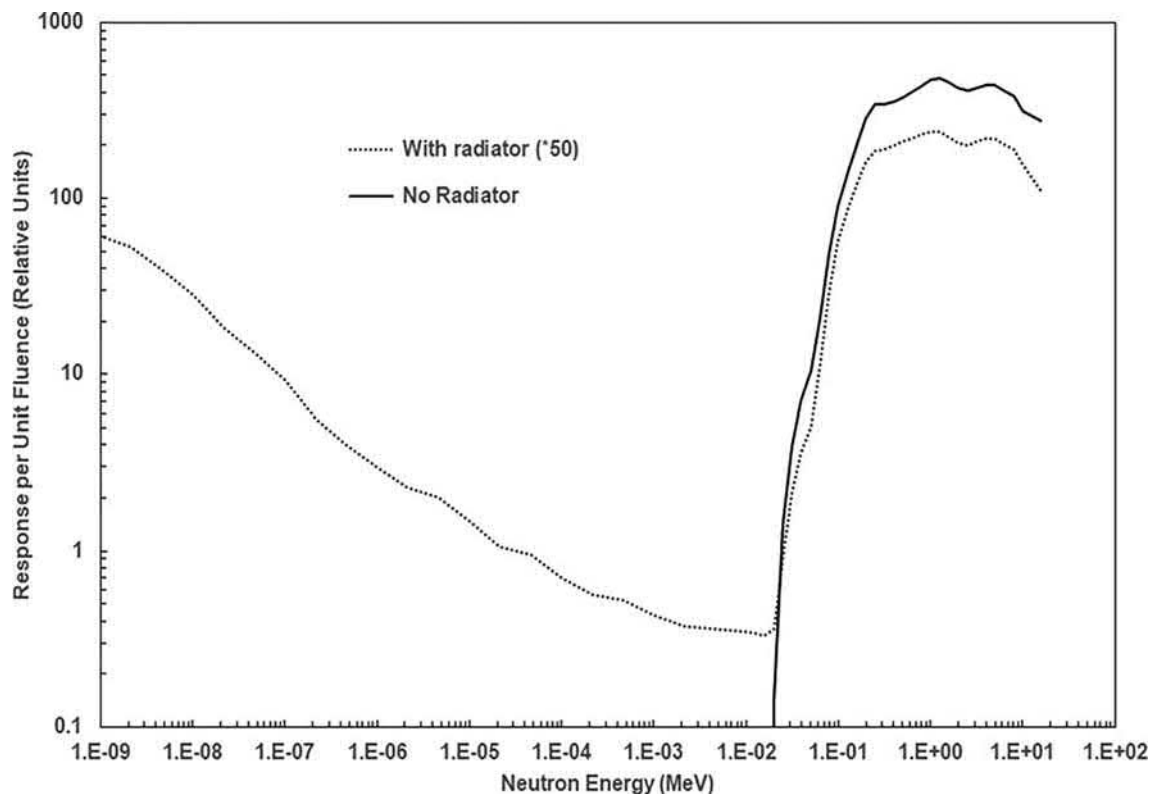


Fig. 5 Energy dependence of CR-39. The CR-39 response is bolded while the other curve includes a thermal neutron radiator. A scaling factor of 50 has been applied to the CR-39 with radiator detector response for plotting (IAEA, 2001).

The neutron energy response of CR-39 is negligible below a few 100 keV, depending on the etching mechanism, but it remains relatively flat above this value. To allow the detection of low-energy neutrons a radiator can be placed over the CR-39. This radiator is often a material that emits energetic alpha particles following absorption reactions with thermal neutrons. A representative energy response curve for CR-39 is shown in Fig. 5 along with the response for CR-39 that utilizes a radiator to enhance the low-energy neutron response. The energy dependence of the neutron cross section for the radiator is evident.

At relatively high doses the density of tracks makes the counting of individual pits impossible. At these doses the dose is instead determined by examination of light transmission which is reduced due to the high density of tracks.

System calibration

Modern personnel dosimeters often contain multiple sensitive elements. These elements are placed behind various filters in order to better determine the type of radiation incident on a dosimeter and of its energy. For mixed radiation fields (i.e., beta/gamma and/or neutron) the determination of the type of radiation is a critical step in the assessment of personnel dose. For a personnel dosimetry program there are a number of system calibrations that must be performed in order to correlate the signal present on an individual sensitive element of a dosimeter to the incident radiation type and energy. Finally, once the type and energy of radiation are resolved one must be able to determine the corresponding dose of interest (i.e., dose to the skin, lens of the eye, and/or dose to the whole body).

A dosimetry calibration program typically consists of the following elements: field (personnel) dosimeters, reference (calibration) dosimeters, quality control dosimeters, a dosimetry processing system (i.e., a reader), and standard dosimeter holders. Since there are variabilities associated with each of these, a method for normalizing each of these must be established. This normalization is the system calibration and, in fact, is comprised of multiple individual factors. Ancillary programs such as quality control, blind audit, and maintenance are important for program performance, but not directly relevant to this discussion.

Algorithm calibration

The dosimetry algorithm interprets dosimeter readings and attempts to calculate the dose received by personnel. Multi-element dosimeters are commonly used with each of the individual elements placed behind various filters located on the dosimeter holder. These filters serve to reduce the particle fluence incident on the sensitive elements. Some filters may, for example, discriminate

low-energy from high-energy photons or beta particles from photons. The algorithm is tuned to optimize performance in a variety of radiation fields that may be encountered at the site of use. This optimization is initially performed using a set of reference dosimeters. The reference dosimeter responses to the various fields is used to develop and assess the accuracy of the dosimetry algorithm. Before an algorithm can be used to monitor workers, personnel dosimeters must first be normalized to the reference dosimeter response. This normalization involves multiple steps which will be briefly discussed in the following sections. While the following discussion is specific to TLD and OSL dosimeters, the principles can be applied to other dosimetry systems such as CR-39.

Element calibration

Each of the dosimeter sensitive elements are manufactured using precise quality control measures. Regardless, some fluctuation in overall sensitivities among a sample population is expected. As part of initial and periodic inspection, each element's response is compared to that of a reference population such as a set of dedicated calibration dosimeters. The elemental response correction factors (element correction coefficients, ECC) are used to create a single population of cards with equivalent responses. These factors are applied following dosimeter processing but before input into the dosimeter algorithm.

Reader calibration

The dosimeter reader processes the elements resulting in light emission which is assumed to be proportional to the absorbed dose delivered to the dosimeter. Readers are equipped with photomultiplier tubes (PMT) that amplify the charge created when the emitted light interacts with photocathodes on each PMT. The signal is amplified and collected on charge collection plates. The total signal (Coulombs) from each element is then compared and scaled to reference responses. An NIST (National Institute of Standards and Technology) traceable source may be used to irradiate the calibration cards used for this reader sensitivity calibration. The resulting reader calibration factors are applied to processed dosimeters to calibrate to a reference response.

Local source calibration

Dosimeters used for reader calibration may not be irradiated while enclosed in holders. Instead, only the element responses to the local source are obtained. The dosimeter algorithm is typically calibrated to a reference radiation field such as ^{137}Cs . If a different local source is used, a correction must be applied to account for the differences in elemental responses relative to a NIST calibrated source. Since workers wear dosimeters enclosed in holders a relationship must be established between the response of the dosimeters exposed to the local source and the response of the dosimeter enclosed in the holder mounted on a suitable phantom. This is performed by comparing dosimeters irradiated using the local source with those irradiated by a suitable (i.e., NIST-traceable) source in holders on an appropriate reference phantom. This approach allows the algorithm to maintain traceability to a NIST reference source.

Algorithms

The dosimetry calibration process converts the signal on the sensitive elements (after correcting for competing effects and contributions to signal such as background, fade, etc.), to a corresponding dose quantity. The calculation of dose equivalent (based on readings obtained from individual elements within a multi-element dosimeter) is a complex process and requires evaluating the energy and type of radiation that gives rise to the signals. In addition, the detector elements must have a large enough sensitivity range to enable accurate determination of personal dose equivalent over a wide range of energies and intensities.

The simplest algorithm is one that performs a single correction to the amount of signal on a dosimeter to the delivered dose. Examples include a single-TLD extremity dosimeter worn on the finger or a CR-39 dosimeter worn on the body. In each case a conversion factor must be applied that converts dosimeter response to dose received by the wearer. Since there is only one element it is not possible to determine specifics on the radiation field such as radiation type or energy. For these single-element dosimeters the radiation field characteristics must be known a priori or assumed based on process knowledge.

For multi-element dosimeters as discussed earlier the algorithm attempts to resolve the type and energy of incident radiations. This process is generally accomplished in one of two ways. The two commonly used methods are discussed in the following sections.

Neural network

This type of algorithm is often referred to as a matching or "neural network" type that can also be described as a least-squares-type fitting routine that compares existing results from actual irradiation of dosimeters to those predicted by the algorithm for an unknown radiation field. The basic component of a neural network is a node, and the network typically consists of an input layer of nodes, an output layer, and additional hidden layers in between. The network is connected by links between the various nodes and these links carry specific weights. The functionality of the network lies in ability to modify the various weighting factors. These weighting factors are determined by allowing the network to vary input/output data pairs. The weights are continuously updated by a "learning" algorithm until the network is able to successfully associate the appropriate input with an appropriate output.

The inputs for a neural network dosimetry algorithm are the various signals on the individual dosimeter elements and the outputs are typically the dose equivalent quantities of interest. The sets of input and output data sets are generated using actual dosimeter data for a variety of single and mixed radiation fields. These fields may, for example, include the beta and photon fields

that are encountered at a facility. The mixtures would include beta + gamma fields, mixtures of photon fields, and possibly mixtures of beta fields. Including more types or exposures to the training set the network generally improves the algorithm performance.

The weighting coefficients are calculated by minimizing the difference between the desired output and the actual output of the network. This equation is linear, i.e., it can be expressed as a linear combination of the logarithmic functions and their powers. This linearity makes it possible to use a variety of multiple regression techniques (Perry et al., 1999).

Step-wise or branching algorithm

In a step-wise algorithm various element ratios are compared to one another and decisions are made based on these element ratios. For example, if two elements are compared (where one of the elements is placed behind a metal filter) the combination of these elements may be used to discriminate low-energy photons. If the ratio of the elements meets certain criteria the algorithm may determine that the field contains low-energy photons. The algorithm would then branch to a low-energy photon step. Each step is a process of acceptance or elimination based on Boolean operators. If the required parameters are met, the algorithm has identified the radiation field composition and moves to the dose determination phase. If the parameters are not met a new set of Boolean operators are selected and the process continues until the field is adequately characterized. This process is termed branching in reference to the various branches (or steps) of the algorithm. Each algorithm branch can have sub-categories to further refine the radiation field. Once a step decision is made the algorithm further analyzes the element ratios to determine an appropriate conversion factor to relate element readings to absorbed dose.

Neutron dosimeter algorithms

For neutron monitoring with TLDs, the dosimeter is enriched in a material with strong neutron sensitivity such as ^6Li and neutrons interact with these elements via the $^6\text{Li}(n, \alpha)^3\text{H}$ reaction. Since the absorption cross section of ^6Li is energy-dependent, the response of the TLD will vary with the spectral distribution of the neutron field to which it is exposed. This energy dependence will differ from that of the dosimetry quantity of interest, such as personal dose equivalent. The photon contribution to the ^6LiF element can be eliminated by subtracting the signal on the ^7LiF chip since the photon response of both is similar (Veinot et al., 1998).

The response of neutron TLDs (mounted on a phantom) include contributions from incident and albedo neutrons. These responses as a function of incident neutron energy can be calculated and have been reported (Veinot and Hertel, 2001, 2005) in the literature. For personnel neutron dosimetry a single correction may be applied to account for the spectral energy dependence.

In the case of CR-39 dosimetry a conversion between track density and delivered dose is used. If the CR-39 includes filters or radiators then additional conversion factors are applied. Some CR-39 dosimeters include radiators that enhance the low-energy response and a different conversion factor would be necessary for this element.

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Internal Dosimetry

Lisa M. Snapp^a and Keith F. Eckerman^{b,*}, ^aY-12 National Security Complex, Oak Ridge, TN, United States; and ^b Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Introduction

The field of internal dosimetry focuses on the techniques used to quantify the amount of radioactive material deposited in the human body. It does so by determining the conditions of exposure and by utilizing biokinetic models to evaluate individual biological monitoring data, ultimately resulting in a dose estimate. Various scientific disciplines are involved including physics, chemistry, mathematics, anatomy, physiology, and radiobiology. Estimation of internal dose is a complex process because the magnitude of intake is usually unknown and there is a high degree of biological variability governing the retention and transport processes within the body. There are also differences between radiation sources external to the body and those that are internal to the body. Time, distance, and shielding play key roles in affecting the dose incurred by radiation sources external to the body. By minimizing the time spent around an external source and maximizing the distance and shielding, the external dose can be reduced. However, with radiation sources that are internal to the body, the distance and shielding are fixed by the body and the dose is delivered over time as the radionuclides reside within the body. One way to mitigate potential internal exposures is to implement engineering and administrative controls, such as performing work in a glovebox or utilizing respiratory protection when working with the potential for airborne contamination. Since you cannot see or smell radioactive material that might be taken into the body, there is no immediate way to determine if an internal exposure has occurred without performing some type of monitoring. Therefore, internal dosimetry programs are necessary and important to confirm the adequacy of engineering and administrative controls and to demonstrate compliance with the occupational dose limits set forth in regulatory guidance used to ensure personnel safety.

Basic concepts of internal dosimetry

This section discusses and defines some of the fundamental concepts of internal dosimetry including specific absorbed fractions, source organs, target organs, specific effective energy, equivalent dose rate, and effective dose. To start with, we will define absorbed fraction as that fraction of energy absorbed by the target that was emitted from the source. To gain a visual representation of this concept, consider a source organ, S , from which a radiation of type i is emitted. If target organ, T , of mass M_T , absorbs a fraction of the energy emitted in nuclear decays of the radionuclide present in S , the specific absorbed fraction, $SAF_i(T \leftarrow S)$, is defined as follows:

$$SAF_i(T \leftarrow S) = \frac{1}{M_T} \frac{\text{Energy absorbed by target, } T}{\text{Energy emitted within source, } S} \quad (1)$$

Consequently, the generic equation for calculating the absorbed dose rate in the target organ, T , at time t post intake due to the activity, $A_S(t)$, in the source organ, S , is defined as:

$$D(T \leftarrow S) = A_S(t) \sum_i n_i E_i SAF_i(T \leftarrow S) \quad (2)$$

where $n_i E_i$ = mean energy emitted by radiation, i ; $A_S(t)$ = activity in the source organ; $SAF_i(T \leftarrow S)$ = specific absorbed fraction of radiation, i .

By grouping the factors related to emitted radiations and their contribution to absorbed energy, the coefficient $S(T \leftarrow S)$ can be defined as:

*Retired.

$$S(T \leftarrow S) = \sum_i n_i E_i \text{SAF}_i(T \leftarrow S) \quad (3)$$

By applying the appropriate quality factor for each type of radiation to the $S(T \leftarrow S)$ coefficient one addresses equivalent energy and the resultant coefficient is denoted as $S_W(T \leftarrow S)$. This coefficient previously was referred to as specific effective energy. $S_W(T \leftarrow S)$ is equivalent energy imparted per unit mass of the target organ, T, due to the disintegration of a specific radiation occurring in source organ, S, and is represented by the following equation:

$$S_W(T \leftarrow S) = \sum_i n_i E_i Q_i \text{SAF}_i(T \leftarrow S) \quad (4)$$

where $n_i E_i$ = mean energy emitted by radiation, i ; Q_i = quality factor for radiation, i ; $\text{SAF}_i(T \leftarrow S)$ = specific absorbed fraction of radiation, i .

Substituting into the equation for dose rate in a target organ, T, and summing over all sources, one obtains the first principles equation for the equivalent dose rate, $H_T(t)$:

$$H_T(t) = \sum_s A_s(t) S_W(T \leftarrow S) \quad (5)$$

To represent stochastic risk the dose quantities of equivalent dose, H_T , and effective dose, E, are used. Integrating the equivalent dose rate with respect to time commitment (typically taken to be 50 years for an adult) yields the equivalent dose in tissue T, after an intake at time t_0 , as follows:

$$H_T = \sum_s \int_{t_0}^{t_0+50} A_s(t) S_W(T \leftarrow S) dt \quad (6)$$

Equivalent dose and effective dose are expressed in terms of the sievert (Sv), or rem, which implies that the relative biological effects have been taken into account. According to the recommendations made by the International Commission on Radiological Protection (ICRP) in Publication 103, effective dose is the tissue-weighted sum of the equivalent doses in all specified tissues and organs of the human body and represents the stochastic health risk to the whole body (ICRP, 2007). Thus, effective dose is defined as follows:

$$E = \sum_T w_T H_T = \sum_T w_T \sum_R w_R D_{T,R} \quad (7)$$

where w_T = appropriate tissue weighting factor; H_T = equivalent dose to an organ or tissue; w_R = appropriate radiation weighting factor; $D_{T,R}$ = absorbed dose in tissue, T, by radiation, R.

Further, the protection quantities of effective dose and equivalent dose are defined using mathematical models such as the anatomical and physiological parameters of Reference Man detailed in ICRP Publications 23 and 89 (ICRP, 1975, 2002). Reference Man provides the foundation for the biokinetic models that describe how radioactive materials are transported and retained within the human body. Using these models, dose coefficients have been derived and tabulated in various compendiums, such as ICRP Publications 68 (ICRP, 1994b), 72 (ICRP, 1995), and 119 (ICRP, 2012), which can be utilized to calculate effective doses once an internal deposition has been ascertained.

Internal dosimetry parameters and models

Several important factors influence internal dose estimation and the impacts of these factors should be understood. First, when assessing internal dose, the dosimetrist should establish the exposure scenario. Second, the physicochemical properties of any internally deposited radioactive materials should be characterized to determine their impact on the movement of material throughout the body. Finally, the dosimetrist should have a basic understanding of the types of computational models involved and how they work together.

The exposure scenario determines the time involved and how the material entered the human body. As far as entry into the body, there are multiple pathways providing routes of entry. Inhalation and ingestion are routes of intake, and entrance into the bloodstream or absorption through the skin is referred to as an uptake. As for the time involved in the exposure scenario, there are two possibilities – acute exposure or chronic exposure. Chronic exposure occurs over an extended period and an acute exposure occurs all at once. In reality, chronic exposures can be quite complex often involving multiple acute exposures superimposed on an underlying continuous exposure. An acute exposure typically can be traced back to a specific event such as a fire involving airborne radioactive material. If a worker was in the area at the time of the fire and sustained an inhalation intake before evacuating the area, this would be considered an acute inhalation intake. On the other hand, a chronic exposure typically involves daily routine work with low-levels of airborne radioactive material. An example would be a worker that enters radiological areas on a daily basis and performs hands-on work with radioactive material. This type of worker has the potential to sustain small inhalation intakes each day that are continuous and build up over an extended period of time and this type of intake is called a chronic intake.

The radioactive material involved makes an impact on the overall internal dose estimate, so it is important to characterize the work area to identify and understand what might be available as potential source material. Some material properties that are necessary are the chemical form of the material, which determines its solubility, and the aerosol size (known as activity median aerodynamic diameter (AMAD)). These factors are important because they impact the deposition and clearance of material in the human respiratory tract. Initially, in ICRP Publication 2 (ICRP, 1959), the lung model was a two-compartment model consisting of the upper respiratory tract and the lower respiratory tract. ICRP Publication 30 (ICRP, 1979) adopted a multi-compartment lung model consisting of the nasopharyngeal (NP), tracheobronchial (TB), pulmonary (P), and lymphatic (L) regions which classified material according to clearance through the lung as shown in Fig. 1. The large arrows in the figure represent the deposition (D) into each region of the lung (30% in the NP region, 8% in the TB region, 25% in the P region, with the remainder being exhaled) and the smaller arrows indicate the fraction of the material, F, that leaves each compartment with an associated clearance half-time, T, specified in days. The three solubility classes considered with this lung model involve clearance times from the pulmonary region on the order of days, weeks, and years—Class D, W, and Y.

The updated lung model in ICRP Publication 66 (ICRP, 1994a) classifies material in terms of the transfer to the blood with the mechanical clearance being independent of chemical form. Thus, radioactive material clearance, or how it is being transferred to the blood, is now described as either fast, medium, or slow - Type F, M, or S. The ICRP Publication 66 lung model incorporates current knowledge of the morphological, physiological, and radiobiological characteristics of the human respiratory tract. Fig. 2 shows the compartments in which inhaled particles may be deposited. As shown in this figure, the lung is divided into the following compartments: anterior nasal passages (ET₁), naso-oropharynx/larynx or posterior nasal passages (ET₂), bronchi (BB), bronchioles (bb), and alveolar interstitium (AI). The anterior nasal passages account for the removal of material via extrinsic means such as nose blowing. The naso-oropharynx/larynx region accounts for material that is assumed to be swallowed, passed into the GI tract, and excreted in the feces. The bronchi, bronchioles, and alveolar interstitium make up the thoracic region. Depositions in the thoracic region are transported to the blood through absorption, to the GI tract by mechanical processes, and to the lymph nodes. Material that is absorbed in the blood is passed through the kidneys and bladder and ultimately excreted via the urine pathway.

In addition, as shown in Table 1, the aerosol size affects the lung deposition and clearance because finer particulates deposit deeper into the lung and clear more slowly than larger particulates deposited in the upper airways. The chemical form of the material determines its solubility and the ultimate uptake into the blood for biological processing through the body. More soluble materials (Types F and M) are rapidly cleared from the lung and preferentially excreted in the urine whereas insoluble materials (Type S) are more slowly cleared and excreted in the feces. For occupational exposures the default value recommended for the Activity Median Aerodynamic Diameter (AMAD) is 5 μm , which is considered to be more representative of workplace aerosols than the 1 μm default value adopted in ICRP Publication 30 (Dorrian and Bailey, 1995).

To derive internal dose estimates, the parameters of Table 1 are incorporated into computational models describing the movement of radioactive material throughout the human body. The intake models describe the movement of material through the lung and gastrointestinal tract. The systemic models describe how the material is absorbed into the bloodstream and its distribution of the uptake throughout the body. The excretion models describe how the material is processed and excreted from the body via the urinary and fecal pathways. Ultimately, these models work together as shown in Fig. 3, by incorporating the various material and exposure parameters to result in an overall dose consequence from internally deposited radionuclides.

Before being able to apply the readily available dose coefficients to convert intake into internal dose, the intake has to be estimated. These intake estimates can be accomplished either by using observed air concentrations or through bioassay measurements.

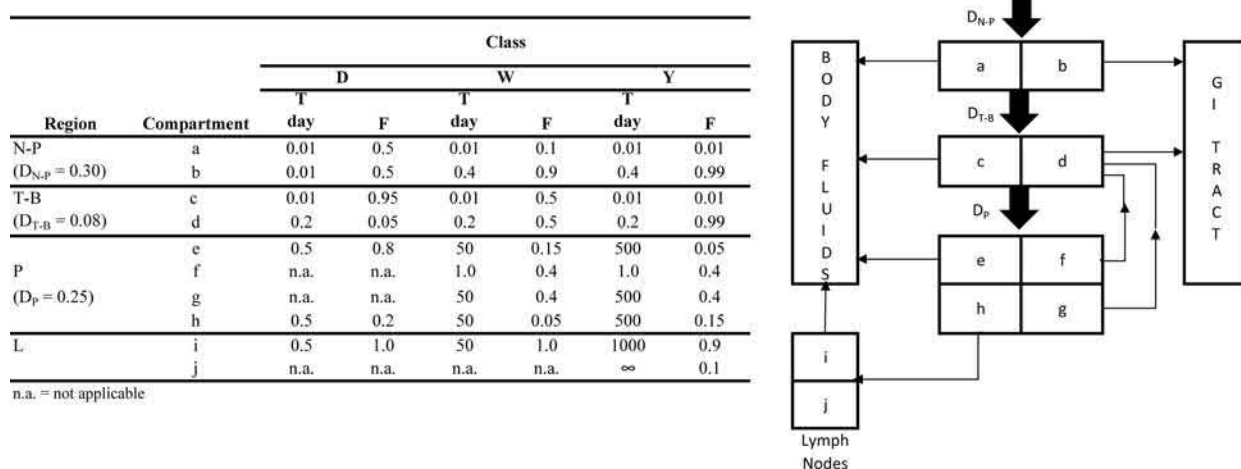


Fig. 1 Mathematical model used to describe clearance from the respiratory system. The values for the removal half-times, T, and compartmental fractions, F, are given in the tabular portion of the figure for each of the three classes of retained materials. The values given for D_{N-P} , D_{T-B} and D_P (left column) are the regional depositions for an aerosol with an AMAD of 1 μm . The schematic drawing identifies the various clearance pathways from compartments a-i in the four respiratory regions, N-P, T-B, P, and L. Reproduced with permission from Figure 5.2 in ICRP (1979) Limits for intakes of radionuclides by workers, ICRP Publication 30, Part 1. *Annals of the ICRP* 2(3/4). Oxford, UK: Pergamon Press.

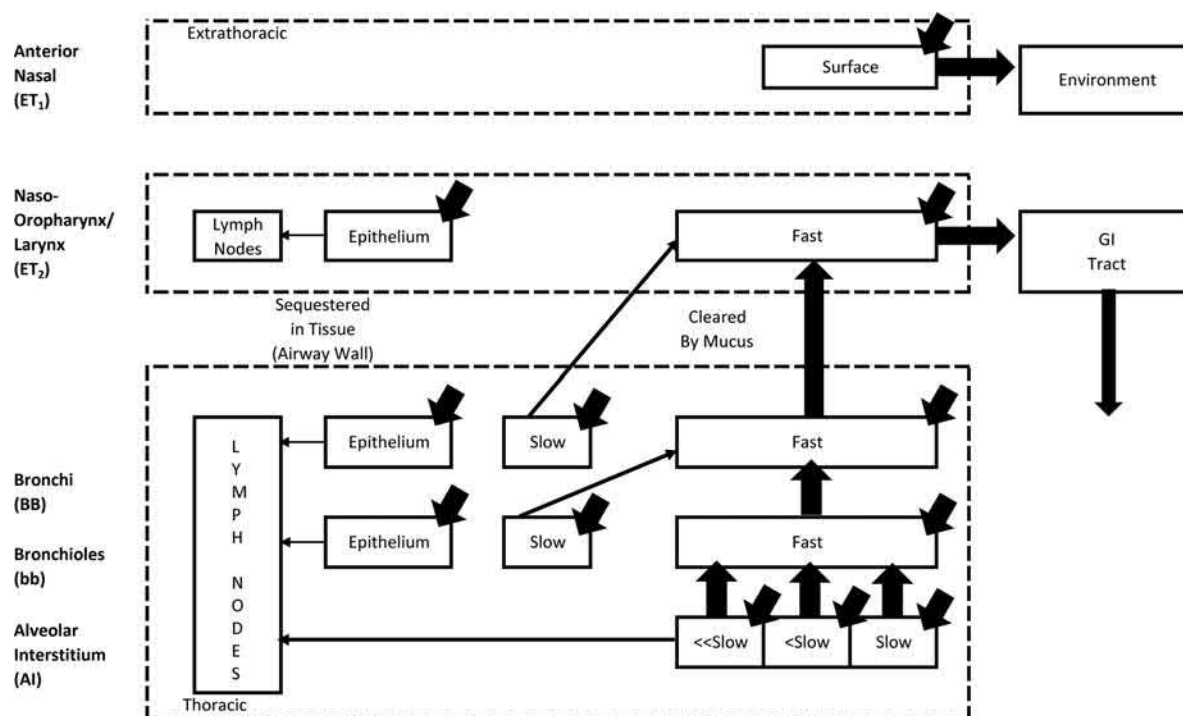


Fig. 2 Schematic showing the respiratory tract compartments in which material can be deposited in the ICRP Publication 66 lung model. Obtained from Figure 7 in ICRP (1994a) Human respiratory tract model for radiological protection, ICRP Publication 66. *Annals of the ICRP* 24 (1–3). Oxford, UK: Pergamon Press.

Table 1 The effect of particle size on regional deposition within the lung of inhaled aerosols.

Region	Aerosol AMAD	
	1 μm	5 μm
ET ₁	16.52%	33.85%
ET ₂	21.12%	39.51%
BB	1.24%	1.78%
bb	1.65%	1.10%
AI	10.66%	5.32%
Total	51.19%	81.96%

NOTE: Fractional deposition based on use of the recommended default values for describing the physical characteristics (e.g., particle density of 3 g/cm³, etc.) of the involved aerosol.

ICRP (1994b) Dose coefficients for intakes of radionuclides by workers – Replacement of ICRP Publication 61, ICRP Publication 68. *Annals of the ICRP* 24 (4). Oxford, UK: Pergamon Press.

Air concentrations can be measured using several types of air sampling devices such as low-volume air samplers, high-volume air samplers, continuous air monitors, and lapel air samplers. Refer to ANSI N13.56 for a more detailed discussion concerning the different types of air sampling devices (ANSI, 2019). Using the air concentrations measured by these devices, the intake activity can be determined by multiplying the air concentration by the individual's breathing rate and duration of the exposure. When using bioassay measurements to estimate the intake activity, there are two categories of bioassays to consider—direct bioassay (also called *in-vivo* bioassay) and indirect bioassay (also called *in-vitro* bioassay). As can be inferred by the name, direct bioassay techniques measure the activity present in the body, whereas indirect bioassay techniques infer the activity based on measurement of excreta. Whole body counts, lung counts, thyroid counts, and scanning bed counts are examples of direct bioassay techniques. Urinalysis and fecal analysis are examples of indirect bioassay techniques.

To illustrate the concepts of intake estimation and calculation of dose consequence, consider the following example:

A worker spends time in an area where Co-60 oxide is present. The physical half-life, $T_{1/2}$, for Co-60 is 5.271 years. At the end of the quarter, the worker submits their routine urine bioassay (70 days after a suspected exposure) and activity is detected in his sample. Additional urine sampling indicated the following excretion pattern:

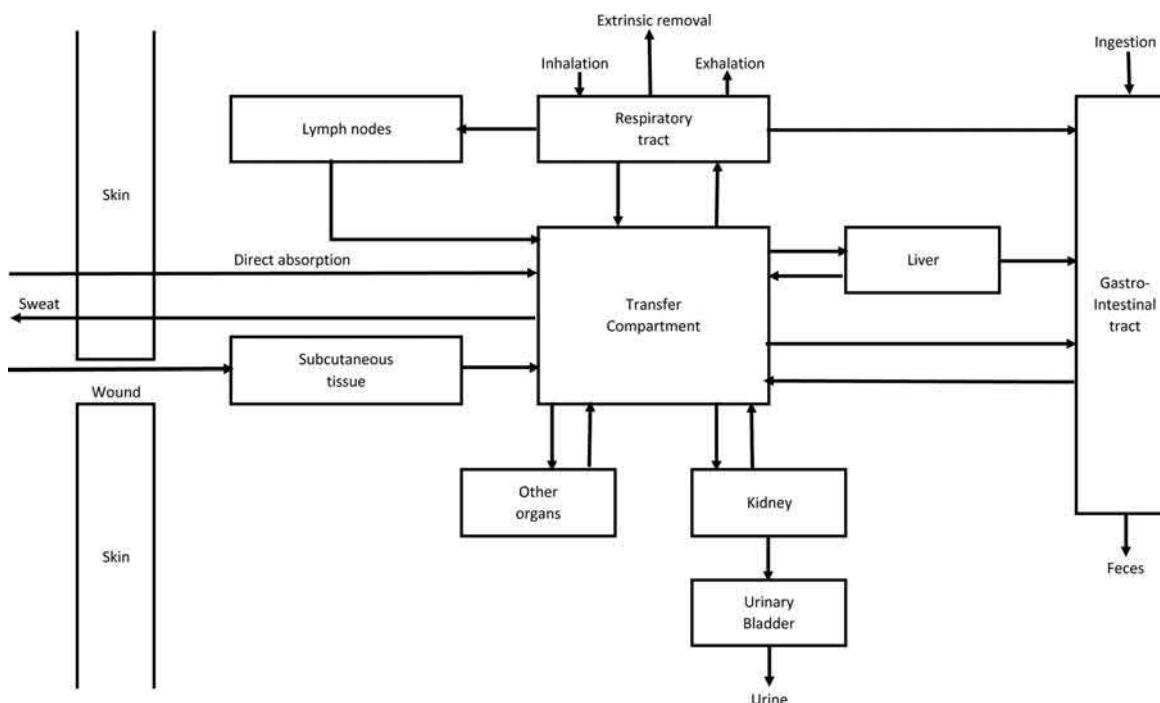


Fig. 3 Routes of intake, transfers, and excretion. Reproduced from Figure 1 in ICRP (1977) Recommendations of the International Commission on Radiological Protection, ICRP Publication 26. *Annals of the ICRP* 1(3). Oxford, UK: Pergamon Press.

Days after suspected exposure	24-h Urine activity (Bq)
70	60
100	50
130	45

In addition, the worker had a lung count performed on day 70 and the result was 0.25 MBq of Co-60.

Step 1: For this example, look up the intake retention fractions based on the system of models from ICRP Publication 68 for Type S Co-60 (oxide) tabulated by [Potter \(2002\)](#).

Time post suspected intake (days)	Fraction of initial intake In:	
	24-h Urine	Lung
70	2.06E-05	4.02E-02
100	1.50E-05	n/a
200	7.53E-06	n/a
130 ^a	1.16E-05	n/a

^aNOTE: A log-log interpolation was performed between 100 and 200 days to obtain the value at 130 days.

Step 2: Perform point estimates of the intake for each type of assay. If the estimates for each type are consistent then their average is the overall estimated intake. If they are not consistent, re-evaluate the assumptions (e.g., chemical form, route of intake, etc.) to determine if there may be a problem. A good rule of thumb when determining consistency is if the intake estimates are within a factor of two.

Point estimates of the intake are determined using the following equation:

$$\text{Intake} = \frac{R}{\text{IRF}(t) * e^{-\lambda t}} \quad (8)$$

where R = bioassay result; IRF(t) = expected fractional activity in the bioassay result at time t per unit intake, $\lambda = \frac{\ln(2)}{T_{1/2}} = 3.60\text{E-}04 \text{ d}^{-1}$.

Intake estimates based on urine results				
Time (days)	Urine results (Bq)	IRF(t)	$e^{\lambda t}$	Intake (MBq)
70	60	2.06E-05	0.975	2.99
100	50	1.50E-05	0.965	3.45
130	45	1.16E-05	0.954	4.07
Intake estimates based on lung count results				
Time (days)	Lung count results (MBq)	IRF(t)	$e^{\lambda t}$	Intake (MBq)
70	0.25	4.02E-02	0.975	6.36

Averaging the three estimated intakes based on urine sampling results in an intake estimate of 3.50 MBq. Combining the average intake estimate from the urine data with the intake estimate based on the lung count data yields an overall intake estimate of 4.94 MBq.

Step 3: Use one of the compendiums of dose coefficients to look up the inhalation dose coefficient for Type S, 5 μm AMAD, Co-60. Applying the dose coefficient of $1.7\text{E-}08 \text{ SvBq}^{-1}$ from ICRP Publication 68 results in an effective dose for the worker in this example of 0.084 Sv or 84 mSv.

Since there are multiple data points associated with the urine bioassay technique in this example, further refinement of the intake estimate can be achieved by applying the method of least squares to evaluate the data. Unweighted least squares fitting estimates the unknown parameters by minimizing the sum of the squared deviations between the data and the model as follows:

$$I = \frac{\sum_i R_i \text{IRF}(t_i)}{\sum_i [\text{IRF}(t_i)]^2} \quad (9)$$

Weighted least squares fitting can be used in the case where there is a high degree of variability between the precision of the data points to properly reflect the behavior of random errors in the data model as follows:

$$I = \frac{\sum_i R_i}{\sum_i \text{IRF}(t_i)} \quad (10)$$

Because there were multiple data points for the urine bioassay in the example, utilizing the intake equations for both the unweighted and weighted methods together with the data in the table below demonstrates the effect of least squares fitting when estimating the intake.

Time (days)	Urine result (Bq)	IRF(t)	$R^* \text{IRF}(t)$	$\text{IRF}(t)^2$
70	60	2.06E-05	0.00124	4.24E-10
100	50	1.50E-05	0.00075	2.25E-10
130	45	1.16E-05	0.00052	1.34E-10
Totals:	$\Sigma R = 155$	$\Sigma \text{IRF}(t) = 4.72\text{E-}05$	$\Sigma R^* \text{IRF}(t) = 0.00251$	$\Sigma [\text{IRF}(t)]^2 = 7.83\text{E-}10$

With the unweighted least squares method, using Eq. (9) and the values from the table above, the intake based on the urine data in our example is estimated to be 3.21 MBq. With the weighted least squares method, using Eq. (10) and the values from the table above, the intake is 3.28 MBq. Multiplying the estimated intake activity by the dose coefficient of $1.7\text{E-}08 \text{ SvBq}^{-1}$ (from Step 3 in the example) results in the following effective dose estimates for our worker: (a) for the weighted intake, the effective dose is 0.056 Sv, and (b) for the unweighted intake, the effective dose is 0.054 Sv. No matter what method was selected to estimate the intake in this example, the results were similar. The annual dose limit of 0.05 Sv was exceeded and further bioassays will be needed to refine and finalize the internal dose for this worker.

Operational internal dosimetry

What makes a good internal dosimetry program? The fundamental goals of a good program should be to monitor workers appropriately for internal depositions of radionuclides and to assess and record the resultant internal doses from these depositions. The first step in achieving these goals is to characterize the workplace to identify what radionuclides are present, and to determine their chemical forms and other characteristics, which might affect internal dose assessments, such as particle size and potential intake routes. Bioassay is integral to being able to determine the time and amount of internal radionuclide depositions. Therefore, it is important to use the right types of bioassay and to assign the best scheduling frequencies to achieve the operational goals of the

program. There are two main categories of bioassay—routine and special. Routine bioassay samples are needed to ensure that administrative and engineering controls are working as intended and they are scheduled at pre-determined frequencies. Special bioassay samples are scheduled at the direction of an internal dosimetrist and are needed to confirm an internal deposition has occurred or to provide follow-up information for an internal deposition that has already been confirmed.

When determining the types of bioassay to implement within an internal dosimetry program consideration should be given to the source radionuclides and the detection limit associated with each bioassay type. The type of bioassay chosen should be capable of detecting an intake at the suggested monitoring level for internal doses from applicable regulatory guidance. However, at the very least, the type of bioassay chosen has to provide evidence of compliance with the regulatory dose limits. As stated in ANSI N13.39, “Design of Internal Dosimetry Programs”, the critical point of administering a routine bioassay program is to find the balance between satisfying detection objectives and avoiding unnecessary monitoring (ANSI, 2011). At the Y-12 National Security Complex, one way we have accomplished this balance is to utilize Radiological Work Permits (RWPs) to convey the bioassay requirements from our workplace characterization efforts. RWPs are used to initiate routine bioassay scheduling instead of defaulting to a set frequency for monitoring independent of radiological work. The RWP logins are captured electronically and transferred to the scheduling system where scheduling is initiated for an individual only after they have signed in on an RWP requiring bioassay. The required bioassay appointment is then scheduled at a prescribed frequency (e.g., quarterly, monthly, or bi-monthly). The individual can then continue to use RWPs as often as necessary and they will not be scheduled for additional bioassay appointments until after submission of the initial required appointment. This type of scheduling can significantly reduce the number of unnecessary bioassays by only scheduling when triggered by RWP use.

The concept of minimum detectable dose (MDD) is used when determining the appropriate default frequency for scheduling routine bioassay measurements. Minimum detectable dose is an estimate of the internal dose that could go undetected between scheduled routine bioassay measurements. For a single acute inhalation intake, the greatest minimum detectable dose will occur when the most recent bioassay result equals the minimum detectable activity (MDA) of the analytical technique and the intake occurred just after the previous bioassay submittal. The minimum detectable dose for a single acute intake is calculated as follows:

$$MDD = \frac{MDA}{IRF(t)} \times DC \quad (11)$$

where MDD = minimum detectable dose; MDA = minimum detectable activity of the analytical technique; IRF(t) = intake retention fraction for the appropriate radionuclide at time, t, post intake; DC = dose coefficient (earlier referred to as the dose conversion factor).

For a chronic inhalation intake, the assumption is that the individual sustains a fraction of the overall intake each day over an extended period. Since the intake is continuous, the amount of radioactive material observed in a given compartment will increase with time until it reaches equilibrium with the intake rate. Similar to the acute intake scenario, the greatest minimum detectable dose occurs when each bioassay result throughout the exposure period equals the minimum detectable activity level for the analytical technique. For this case, the minimum detectable dose is given by the following:

$$MDD = \frac{A(t)}{IRF(t)} \times DC \quad (12)$$

where MDD = minimum detectable dose; A(t) = activity observed in a given bioassay at some time, t; IRF(t) = fraction of the total intake since exposure began until time, t; DC = dose coefficient (earlier referred to as the dose conversion factor).

As an example of the minimum detectable dose concept, consider the values shown in the table below for Americium urine bioassay. The intake profile is based on the absorption type of the material (Fast (Type F), Medium (Type M), or Slow (Type S)), the particle size of the material (typically 5 µm for occupational intakes), and the type of intake/uptake. The activity in the urine bioassay for this example is given in terms of disintegrations per minute (dpm) observed in a 24-h sample. The tabulated values are the expected minimum detectable internal doses per observed urinalysis result specified in units of $mrem \cdot dpm^{-1}$. If the analytical laboratory states that their minimum detectable activity for Am-241 urinalysis is 0.1 dpm, selecting a monthly frequency of participation would result in the following minimum detectable dose values: 172 mrem for an acute inhalation scenario, 42 mrem for a chronic inhalation scenario, and 258 mrem for an acute ingestion. In this example, to be able to detect a 100 mrem internal dose, either the MDA needs to be lower, the scheduling frequency needs to be shorter, or one should consider adding fecal sampling in addition to urinalysis to achieve the goals.

		Time post Intake (days)						
Intake profile ^a		7	14	30	60	90	180	365
Acute Inhalation (5 µm)	Type M	783	1110	1720	2410	2870	4120	6460
Chronic Inhalation (5 µm)	Type M	132	232	416	697	923	1460	2300
Acute Ingestion	f ₁ = 5E-04	515	957	2580	4900	5660	6880	8200

^aNOTE: For reference, the recommended absorption type (e.g., Type M) and gastro-intestinal absorption fractions (f₁), can be found in publications like ICRP Publication 78 (ICRP, 1997).

As you can see by the variability of the dose numbers in the table for minimum detectable dose, the default assumptions are important and can have great impact on the overall dose consequence. This is why the default assumptions should be documented for each area of the facility. Guidance on the selection of appropriate monitoring techniques and assumptions regarding the intake profiles for specified radionuclides are in publications like ICRP Publication 78 (ICRP, 1997).

In addition to determining the default exposure scenarios, reference levels should also be documented for the bioassay program. These reference levels are used to determine follow-up actions when the value of a quantity exceeds or is predicted to exceed the reference level (ICRP, 1988). Reference levels apply to both routine and special monitoring. According to ANSI N13.39, if there are choices for default assumptions to characterize the conditions applicable to a worker or set of workers, the default assumptions should be chosen to optimize the derived reference level (ANSI, 2011). There are different types of reference levels including the following: (a) the screening level which is the level below which a bioassay result does not need to be investigated and dose does not need to be assigned, (b) the verification level which is the level above which an effort should be made to confirm if an intake is real by invoking special bioassay protocols, (c) the investigation level which is the level requiring initiation of an investigation and special bioassay to help determine the conditions of exposure for the purposes of assessing dose instead of using default parameters, and (d) the medical referral level is the level at which medical staff at the facility should be notified. Fig. 4, obtained from ANSI N13.39, demonstrates how to use these reference levels.

According to the guidance provided in ANSI N13.39, workers should participate in a routine bioassay program if they have the potential to incur an intake at the investigation level specified in the applicable regulatory requirements or if respiratory protection is used to limit exposure potential in the workplace. When determining potential, one must consider both existing conditions and conditions that could occur due to a loss of control in the workplace. The success of any monitoring program depends on the workers and if they do not submit their scheduled *in-vitro* samples on time or show up for scheduled *in-vivo* counts then the accuracy of the program is impacted. Therefore, it is important to have administrative policies in place to ensure compliance with documented bioassay requirements. At the Y-12 National Security Complex, individuals are given a grace period of 2 weeks to submit requested bioassay samples which allows them flexibility in their schedule to collect the samples. The individual is sent a notice to remind them of the need for sample submission when their sample is 10 days past due. If the individual does not submit the requested sample by the 15th day past the scheduled appointment date, or if they do not show up for a scheduled lung count,

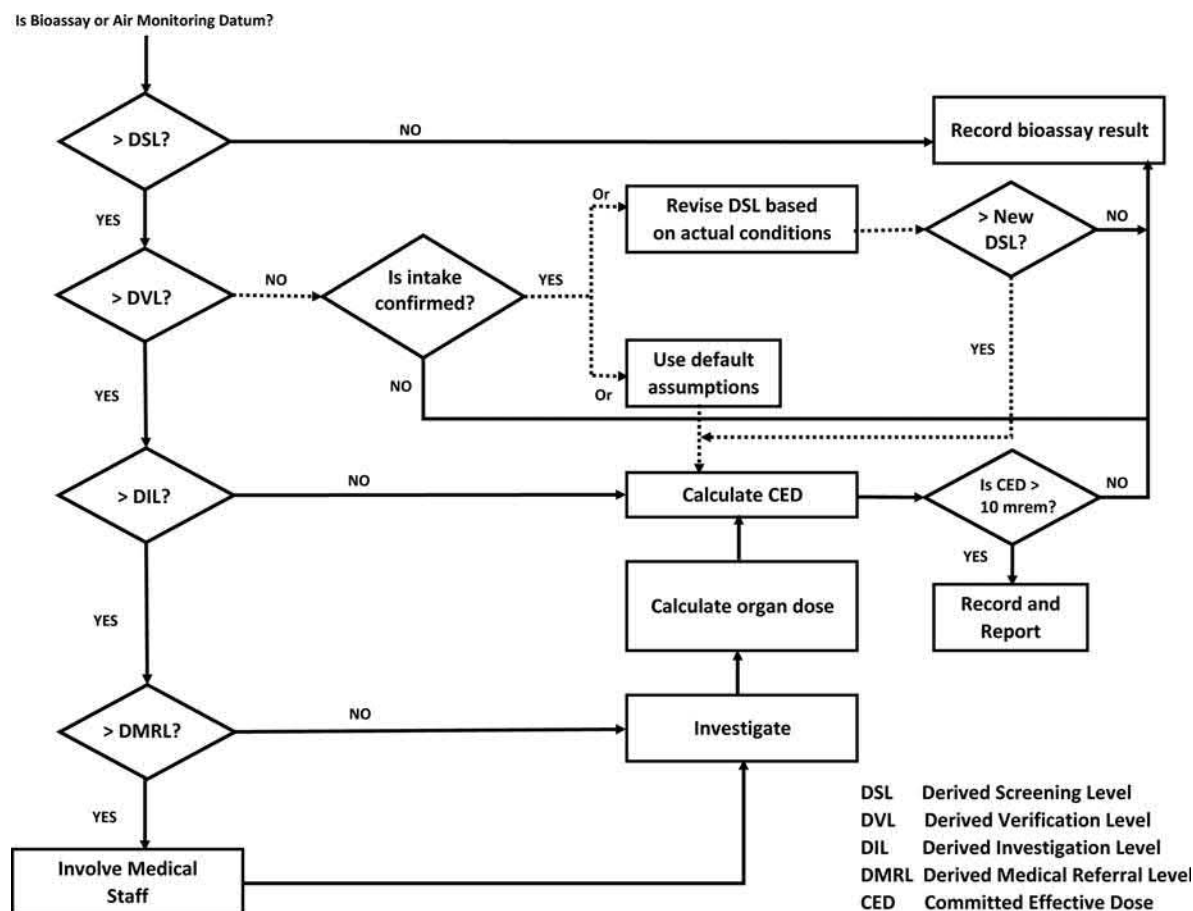


Fig. 4 Flowchart demonstrating the use of reference levels when evaluating bioassay results. Reproduced with kind permission from ANSI/HPS N13.39-2001(R2011), "Design of Internal Dosimetry Programs".

they are restricted effectively preventing them from performing work in radiological work areas governed by RWPs. This type of administrative policy encourages workers to submit their samples on time to avoid restriction.

Even with timely submission of requested bioassay samples, other factors can greatly influence an internal dose assessment. Therefore, all the assumptions utilized in a dose assessment need to be thoroughly documented. If an acceptable fit to the bioassay data cannot be obtained, then additional attempts can be made by changing assumptions. Examples of the assumptions to be changed include the following: (1) the exposure scenario, (2) the intake mode, (3) the solubility of the material, (4) the particle size of the material, and (5) the deposition and metabolic parameters in the biokinetic models. In addition, one could discount an outlying result or exclude interferences from prior intakes or non-occupational exposures.

As an example of how to characterize the work area and document modification of one of the parameters affecting internal dose estimation, consider the material solubility parameter and its effect on internal dose calculations at the Y-12 National Security Complex. A study conducted at Y-12, selected several radiation workers as representative of the site and scheduled them for special bioassay. Each worker submitted paired urine and fecal bioassay samples, and as instructed, collected the samples on or about the same date. The analysis of the data set provided an indicator of the solubility of the uranium involved with different operations at the site. From this study, Y-12 observed that the urine and fecal data were not consistent with older ICRP models (e.g., ICRP Publications 30 and 54). The older models indicated a fecal to urine ratio that was smaller than observed. The data from this study confirmed that the newer models of ICRP Publication 78 were consistent with the Y-12 observations as documented in ORNL/TM-114 (Eckerman and Kerr, 1999). Fig. 5 demonstrates this fact in the panel that compares Class Y and Type S urinary and fecal excretion.

Therefore, if urine and fecal data are available, the solubility mixture can be determined by calculating independent estimates of the intake rate based on urine and fecal samples using the newer ICRP models specified in ICRP Publication 60 and later. When evaluating the bioassay data submitted by individuals whose exposure potential is with insoluble uranium, one should expect that the urine and fecal data would produce similar estimates of intake. If the intake rates derived from the urine data, assuming Type S uranium, are significantly larger than those based on the fecal data, then the exposure may involve a mixture of Type M and Type S uranium. The calculations will then determine what fraction of the intake rate is attributable to Type M uranium using

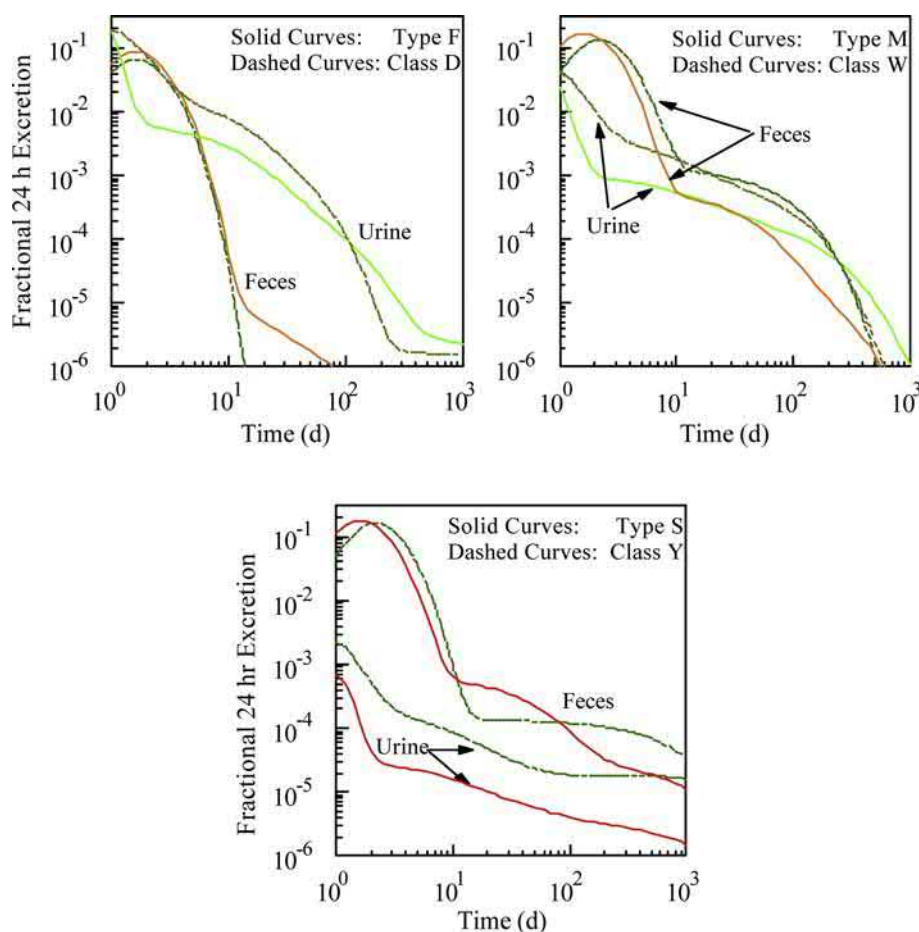


Fig. 5 Expected 24-h urinary and fecal excretion of uranium following an acute intake using the models of ICRP Publication 54 with an AMAD of $1\ \mu\text{m}$ and ICRP Publication 78 with an AMAD of $5\ \mu\text{m}$. Obtained from Figure 6 in ORNL/TM-114.

Eq. (13) below, such that the intake rate predicted by evaluating the urine data with the assumed solubility mixture, will agree reasonably well with the intake rate indicated by the fecal data alone.

$$f_M = \frac{\frac{1}{\langle I \rangle} \sum_i^{N_u} A_u(t_i) - \sum_i^{N_u} E_u^S(t_i)}{\sum_i^{N_u} E_u^M(t_i) - \sum_i^{N_u} E_u^S(t_i)} \quad (13)$$

where f_M = fraction of the intake rate attributable to Type M uranium; $\langle I \rangle$ = intake rate; N_u = number of urine samples; $A_u(t_i)$ = Activity observed in the i th urine sample at time t_i ; $E_u^S(t_i)$ = Estimated activity, assuming Type S, in the i th urine sample at time t_i ; $E_u^M(t_i)$ = Estimated activity, assuming Type M, in the i th urine sample at time t_i .

The conclusion of this study was that the method of determining the solubility based on the fraction of uranium intake was a more stable index of uranium exposures at the site than using the older methods. It also resulted in the development of a new internal dosimetry code, DOSE68, that implemented the ICRP Publication 60 recommendations at Y-12.

Software for internal dosimetry

The internal dosimetry computational framework has advanced as a result of input from the National Council on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), and the Medical Internal Radiation Dose (MIRD) committee of the Society of Nuclear Medicine. Currently, a common mathematical formulation has been achieved within the nuclear medicine and radiation protection communities (Loevinger and Berman, 1968, 1976). This status of the mathematical modeling of physical, anatomical, and biological processes has been achieved due to advanced computer hardware. Two early software packages noting the capability of micro or personal computers in internal dosimetry were Killough and Eckerman (1984) and Birchall and James (1989).

In 1977, the ICRP issued revised radiation protection guidance in Publication 26 (ICRP, 1977). The secondary quantities and dose coefficients based on this guidance were released in 1979 in ICRP Publication 30 (ICRP, 1979). The dosimetric data were computed at Oak Ridge National Laboratory on a mainframe computer (Watson and Ford, 1980). The set of coupled first-order differential equations describing the behavior of the radionuclide and its progeny within the body was solved numerically. Software initially developed for the MIRD committee was used to tabulate the nuclear decay data (Dillman et al., 1975; Dillman, 1980), and this data was later issued in ICRP Publication 38 (ICRP, 1983). Similarly, the anatomical model developed within the MIRD committee was used to derive the photon specific absorbed fraction to enable assessment of the absorbed dose in a tissue resulting from nuclear transformations occurring in other organs. With emerging micro and personnel computers, the IREM code was developed independent of the mainframe code (Killough et al., 1978). At this time, the ICRP did not address dose coefficients for members of the public.

In 1990, the ICRP revised its primary radiation protection guidance. It was noted that many radioactive compounds cleared from the lung in a manner different from the Days (D), Weeks (W), or Years (Y) classification of the Task Group Lung Model used in ICRP Publication 30 (Bates et al., 1966). Thus, ICRP Committee 2 updated the respiratory tract model and published the details in ICRP Publication 66 (ICRP, 1994a) with the uptake rate to blood (absorption) classified as F (fast), M (medium) and S (slow). The corresponding dose coefficients for occupational exposure were derived by an ICRP Task Group and appeared in ICRP Publication 68 (ICRP, 1994b).

The ICRP also began developing dose coefficients for members of the public in Publication 56 (ICRP, 1990) with a final summary in Publication 72 (ICRP, 1995). The need for such coefficients was highlighted by the Chernobyl accident in 1986. An international team of experts carried out the derivation of these coefficients using several independent computer codes. One member of the ICRP team employed an eigenanalysis solver (Fell et al., 2007). Another code used in these derivations was the DCAL computer code (Eckerman et al., 2008). Following completion of the ICRP tasks, the U.S. Environmental Protection Agency (EPA) freely distributed the DCAL code. The system of differential equations is solved within DCAL as discussed by Leggett et al. (1992) and Eckerman et al. (2008). The DCAL software was also used to derive the risk coefficients of Federal Guidance Report 13 (U.S. EPA, 1999).

Currently ICRP Committee 2 is deriving dose coefficients of occupational and environmental exposures that are consistent with the primary radiation protection guidance of ICRP Publication 103 (ICRP, 2007). This ongoing work has appeared in ICRP Publications 130, 134, 137, and 141. This work implements the human alimentary tract model of ICRP Publication 100 (ICRP, 2006), the anatomical model of ICRP Publication 110 (ICRP, 2009), the updated nuclear decay data of ICRP Publication 107 (ICRP, 2008), the specific absorbed fraction data of ICRP Publication 133 (ICRP, 2016) and updated biokinetic models in which the behavior of progeny is independent of the parent's biokinetics. Presumably, the updated software will be released following completion of these efforts.

Numerical databases of dose and bioassay coefficients have been assembled and are generally distributed with a "viewer" software. The DFACT software available from Oak Ridge National Laboratory (<https://www.ornl.gov/crpk/software>) contains the dose coefficients for the radionuclides considered in ICRP Publication 30 (ICRP, 1979). ICRP also distributed a viewer containing the occupational dose coefficients of ICRP Publication 68 (ICRP, 1994b) and environmental dose coefficients of ICRP Publication 72 (ICRP, 1995). These dose coefficients and occupational bioassay coefficients as well as other radiation protection matters are

included in the Radiological Toolbox distributed by the Nuclear Regulatory Commission on the RAMP website (<https://ramp.nrc.gov/>). ICRP is now offering a viewer of the dose and bioassay coefficients recently published (ICRP, 2015, 2017a,b). Table 2 below lists the lung dose coefficients for occupational exposures in ICRP guidance over the past four decades. The Publication 30 clearance rates from the lung are noted in terms of Days (D), Weeks (W), and Years (Y) and characterize both the mechanical and absorption processes. The Publications 68 and 137 clearance rates refer to absorption (Fast (F), Medium (M), and Slow (S)) and reflect the chemical form of the material. Note Publication 137 includes six lung dose coefficients with specific guidance (see table footnotes) regarding chemical form.

There are two widely-used commercial software packages for estimation of a worker's intake based on observed bioassay data—Integrated Modules for Bioassay Analyses (IMBA) (Birchall et al., 2003) and ICRP's Internal Dosimetry Expert Analysis (IDEA) System (Doerfel et al., 2007). Both packages, IMBA (currently updated and renamed TAURUS) and the IDEA System, are being updated to merging ICRP guidance (ICRP, 2015, 2017a,b). There are, of course, institutional-specific software codes established to access bioassay databases and derive dose records to estimate a worker's annual dose. The ICRP 30 based DOSEXPRT software used in the 1990s in the Oak Ridge area, and the newer ICRP 60 based DOSE68 software currently in use at ORNL and Y-12 are such examples.

Applicable regulations and guidance documents related to internal dosimetry

In the United States, federal regulations convey the radiation protection requirements that provide the basis for occupational internal dosimetry programs. There are two main groups of facilities in the United States—those that fall under the purview of the Department of Energy (DOE)/National Nuclear Security Administration (NNSA) and those under the purview of the Nuclear Regulatory Commission (NRC). For NRC sites, the radiation protection requirements are in Title 10, Part 20 of the Code of Federal Regulations (10CFR20, 2016). For DOE/NNSA sites, the requirements are in Title 10, Part 835 of the Code of Federal Regulations (10CFR835, 2007). There is a fundamental difference in the radiation protection requirements used by NRC sites and those used by DOE/NNSA sites. The difference lies in their technical bases. The requirements related to internal dosimetry in 10CFR835 are based on the recommendations from ICRP Publication 60 (ICRP, 1991) and those in 10CFR20 are based on the older recommendations from ICRP Publication 26 (ICRP, 1977).

To assist with the implementation of the regulatory requirements the NRC provides technical guidance in the form of published Regulatory Guides. These Regulatory Guides provide additional details to NRC licensees on acceptable methods to achieve implementation of the NRC regulations. For reference, below is a listing of the applicable Regulatory Guides that pertain to internal dosimetry:

Table 2 Uranium lung committed dose coefficients (Sv/Bq).

Nuclide	Class	ICRP publication 30 ^a		Type	ICRP publication 68 ^b		ICRP publication 137 ^c	
		1 μ m	5 μ m		1 μ m	5 μ m	1 μ m	5 μ m
U-234	D	3.2E-07	1.3E-07	F	3.4E-07	4.0E-07	2.9E-07	2.4E-07
	W	1.6E-05	5.1E-06	M	2.4E-05	1.6E-05	1.7E-05	9.3E-06
	Y	3.0E-04	9.6E-05	S	7.1E-05	4.1E-05	1.6E-04	8.4E-05
					Intermediate Type F/M ^d		3.6E-06	2.2E-06
					Intermediate Type M/S ^e		6.6E-05	3.3E-05
					Uranium aluminide ^f		3.6E-05	1.9E-05
U-235	D	2.9E-07	1.2E-07	F	3.2E-07	3.7E-07	2.6E-07	2.1E-07
	W	1.5E-05	4.8E-06	M	2.2E-05	1.4E-05	1.5E-05	8.5E-06
	Y	2.8E-04	9.0E-05	S	6.4E-05	3.6E-05	1.5E-04	7.7E-05
					Intermediate Type F/M ^d		3.3E-06	2.0E-06
					Intermediate Type M/S ^e		6.1E-05	3.2E-05
					Uranium aluminide ^f		2.2E-05	1.8E-05
U-238	D	2.8E-06	1.1E-06	F	3.0E-07	3.5E-07	2.4E-07	2.0E-07
	W	1.4E-05	4.5E-06	M	2.0E-05	1.3E-05	1.4E-05	8.0E-06
	Y	2.7E-04	8.6E-05	S	6.0E-05	3.4E-05	1.4E-04	7.4E-05
					Intermediate Type F/M ^d		3.1E-06	1.8E-06
					Intermediate Type M/S ^e		5.8E-05	3.0E-05
					Uranium aluminide ^f		3.1E-05	1.7E-05

^aQuality factor of 20 applied to alpha absorbed dose.

^bRadiation weighting factor of 20 applied to alpha absorbed dose.

^cType F - Uranium hexafluoride, uranyl tributyl-phosphate, $f_A = 0.02$. Type M - Uranyl acetylacetonate; depleted; vaporized metal, $f_A = 0.004$. Type S - $f_A = 2E-04$.

^dUranyl nitrate, uranium peroxide hydrate, uranium trioxide, $f_A = 0.016$.

^eUranium octoxide, uranium dioxide, $f_A = 6E-04$.

^f $f_A = 0.002$.

1. Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data"
2. Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program"
3. Regulatory Guide 8.11, "Applications of Bioassay for Uranium"
4. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure"
5. Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection"
6. Regulatory Guide 8.20, "Applications of Bioassay for Radioiodine"
7. Regulatory Guide 8.22, "Bioassay at Uranium Mills"
8. Regulatory Guide 8.25, "Air Sampling in the Workplace"
9. Regulatory Guide 8.26, "Applications of Bioassay for Fission and Activation Products".

As for the DOE and NNSA sites, additional guidance can be found in standards published by the American National Standards Institute (ANSI), which oversees standards and conformity assessment activities for the United States. For reference, a list of the standards related to internal dosimetry is included below:

1. ANSI N13.14, "Bioassay Programs for Tritium"
2. ANSI N13.22, "Bioassay Programs for Uranium"
3. ANSI N13.30, "Performance Criteria for Radiobioassay"
4. ANSI N13.39, "Design of Internal Dosimetry Programs"
5. ANSI N13.54, "Fetal Radiation Dose Calculations"
6. ANSI N13.56, "Sampling and Monitoring Releases of Airborne Radioactivity in the Workplace".

Future direction in internal dosimetry

The emerging ICRP internal dosimetry formulations explicitly address both males and females. However, the sex differences in dose coefficients are largely associated with anatomical differences and not reflected in the biokinetic models. Sex differences in the volume of air breathed may need to be considered when interpreting bioassay measurements. In addition, dose and bioassay coefficients are presented in ICRP publications for a number of updated chemical forms for each radionuclide instead of just the standard default formulations. As noted in [Table 2](#), ICRP Publication 137 now addresses six chemical forms of uranium and the lung clearance no longer needs to be defined in terms of the default Type F, M, and S parameters ([ICRP, 2017b](#)). Implementation of the new guidance in national regulations awaits completion of the ICRP series of publications. Continued advances in computer hardware will lead to further improvements in the internal dosimetry methodology. This includes advances in physiological-based biokinetic modeling, mesh-type reference computational phantoms, and implementation of more detailed bioassay procedures.

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Instrumentation and Practices for Radiological Assessments

Timothy J. Vitkus,^a David A. King,^a Nick A. Altic,^a and Wade P. Ivey^b, ^aORAU, Oak Ridge, TN, United States; and ^b Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Abbreviations

α	Alpha radiation
ADC	Analog to digital converter
β	Beta radiation
γ	Gamma radiation
Bq cm ⁻²	Becquerels per square centimeter
cpm	Counts per minute
cm ²	Square centimeter
CsI	Cesium iodide
CZT	Cadmium zinc telluride
D&D	Decontamination and decommissioning
dpm 100 cm ⁻²	Disintegration per minute per 100 square centimeters
FIDLER	Field instrument for the detection of low-energy radiation
GAB	Gross alpha and beta
GM	Gieger-Muller
GPS	Global positioning system
HPGe	High-purity germanium

ICPMS Inductively coupled plasma mass spectrometry
 hr Hour
 LaBr Lanthanum bromide
 keV Kiloelectron volt
 LBPC Low-background proportional counting
 LSC Liquid scintillation counting
 MCA Multi-channel analyzer
 mg cm⁻² Milligrams per square centimeter
 NaI(Tl) Sodium iodide (thallium)
 NDA Non-destructive assay
 NORM Naturally occurring radioactive materials
 PMT Photomultiplier tube
 QA/QC Quality assurance/quality control
 RAM Radioactive materials
 ROC Radionuclide of concern
 ROI Region of interest
 μR h⁻¹ Micro-Roentgen per hour
 ZnS(Ag) Zinc sulfide silver activated

Introduction to radiological survey instrumentation

Although humans are not able to see, hear, feel or smell radiation; we are able to detect the presence of radiation provided the appropriate instrumentation is available. A facility operating within the conditions of its regulatory license should know exactly where its radioactive materials (RAM) are located. Under operational conditions, such facilities have appropriate radiological survey instrumentation for general monitoring purposes that could include environmental monitoring; work place monitoring such as surveying a lab bench or hood, a process area, or routine required surveys of RAM storage areas; and personnel safety monitoring. However, situations occur during the facilities operational lifetime—such as spills, leaks, past disposal practices, etc.—that may have led to contamination requiring remediation. When conducting decommissioning and decontamination (D&D) activities for license termination, RAM is removed from the site and residual contamination is identified and remediated such that the concentrations of any remaining contamination are quantitatively shown to be below the regulatory limits by direct measurements made using appropriate radiation survey instrumentation, by the radiological analyses of samples, or both (Vitkus, 2021).

It is critical to choose the right tool to identify contamination and for collecting the data required to demonstrate that any contamination levels satisfy personal protection, operational, or regulatory limits. Different forms and types of radiation require different types of radiation survey instrumentation. The instrumentation to consider includes that used for surveys of land areas, structural surfaces, and the laboratory systems used for sample analysis. Laboratory analysis of samples is necessary to quantify residual concentrations of those radionuclides that are either difficult or undetectable using field instruments. Instrumentation for personnel monitoring, air monitoring, and similar radiological protection considerations are also available but are not included within the discussions that follow.

Radiological detection systems generally rely on measuring the energy deposited within the active volume of the detector. Some detectors use material that emit light (scintillate) when the radiation deposits energy within the active volume. Other detectors use gas that is ionized when interacting with radiation; the ions are collected and result in an electronic pulse (Han, 2021). In general there are two components of the detection systems that may be either separate or integrated into a single package. The detector captures the interaction energy and is coupled to an instrument that provides a high voltage to the detector and then records the interaction as a count. The instrument connected to the detector is likely either a ratemeter or ratemeter-scaler. Ratemeters provide an instantaneous reading—e.g., counts per minute (cpm), micro-Roentgens per hour (μR hr⁻¹)—based on the amount of radiation absorbed over a short time interval. An analog needle or a digital readout is observed that will continuously change as radiation emissions are random events and are observed as fluctuating measurements, most notable at lower, background count levels. The meter may also have a speaker to allow the surveyor to listen to the audible clicks. Contamination meters referred to as ratemeter-scalers have a digital scaler mode which allow for quantifying the number of counts over a defined time period, generally between one-half to 10 min.

The type of radiation—alpha, beta, or gamma radiation or a combination thereof—helps determine the detection system used. The selected detection system also depends on the surveyed medium, whether land area, structures, equipment or other media. As such, the physical and radiological characteristics of RAM contamination at a site determines which type(s) of detectors, discussed in the following sections, are used for surveys.

Land area (surface soil) survey instrumentation

Land area radiological investigations, referred to as radiological surveys, are performed to identify where soil contamination may be present and, depending upon the concentrations may require posting or remediation. The survey is performed to locate anomalous areas where elevated count rates are observed above the detector's background response range and signals that contamination may be present. Background radiation is ubiquitous and therefore the detection systems have a continuous background response; commonly stated in units of cpm. Identified anomalies where count rates exceed a nominal background are further investigated to determine if the elevated detector response is simply the result of natural background variability, other physical factors, or is due to radiological contamination within the soil volume.

Qualitative gamma radiation scanning: Hand-held detectors

The most efficient land area surface soil survey method for identifying anomalies is surface scanning for gamma radiation, provided of course that potential radioactive contaminants include one or more gamma emitters. Surface scanning involves moving the detector just above the soil surface either by a surveyor walking over the area in systematic or random transects—referred to as a gamma walkover survey. The surveyor listens to the audio response—each radiation detection event results in an audible click for those survey instruments with integrated speakers—and if elevated counts are heard, the surveyor will pause and investigate. This investigation may consist of concentrated scanning of a defined area or collecting a surface soil sample and then checking the subsurface soil for elevated counts and possibly following that up with more sampling. This is an important point requiring edification. Although the scanning has been introduced as a “surface” scan, it is important to be aware that the detected photons could have originated from buried contamination. Surface scanning might be better referred to as surface volume scanning, because the gamma photons, depending on their energy, are able to penetrate several centimeters of soil and therefore be detected at the surface.

Depending on the site, these surface scans will typically cover hundreds, or more likely, thousands of square meters. For very large sites, arrays of detectors are sometimes mounted to a vehicle or cart and driven over the surface. Drones are also being introduced to carry detectors over surfaces (Hackett, 2021). These mobile systems rely on assessing collected data post-survey, normally via computer programs, to identify anomalies, rather than real-time by an individual listening to meter audio.

Surface scanning land areas for alpha or beta radiation is rarely practical for a variety of reasons that are outside the scope of this discussion, including the fact that the alpha and beta particles are either completely shielded by, or unable to penetrate thin soil layers. Alpha or beta measurements on land area soil, if performed, are likely to be limited to static measurements, meaning collecting a count rate measurement while holding the detector stationary directly on the soil surface or perhaps the surface of a sample of the soil.

The workhorse detector used for soil surface gamma scanning is the sodium iodide (NaI) thallium activated (Tl) scintillation detector seen in Fig. 1.

NaI(Tl) scanning systems in general use only provide the observed counts without identifying information as to what radionuclides are present. These systems are considered qualitative only, and require additional investigation when count rate anomalies are encountered. The investigation is likely to involve sample collection with laboratory analysis to identify specific radionuclides present and the activity concentrations in units of picoCuries per gram or Becquerels per gram.

This type of gamma radiation detector is constructed with a NaI crystal into which trace amounts of thallium have been incorporated as an activator. NaI(Tl) is used because it is one of the most sensitive material to x-rays and gamma rays, although there are



Fig. 1 Surveyors performing gamma radiation walkover survey using 5.1 cm × 5.1 cm NaI(Tl) coupled to ratemeter-scaler and GPS/data logging equipment.

other inorganic crystal types such as lanthanum-bromide (LaBr) and cesium-iodide (CsI) that are used in special applications. When gamma energy is deposited, the crystal material will scintillate, i.e., emit a pulse of light. The light is converted into an electrical signal and then, in order for the signal to be registered as a count, must be amplified by an attached photomultiplier tube (PMT). This detection and amplification package, schematically illustrated in Fig. 2, is contained within a hermetically-sealed metal housing which may be either configured as an external probe connected to a ratemeter or contained within a stand-alone meter, both types are shown in Fig. 3.

An example stand-alone meter is a μ R meter which measures exposure rates. This instrument has a small NaI(Tl) detector within a protective case that includes the electronics that supply the power to the detector and an analog readout of the real-time measurements. The structure/surface survey instrumentation section below provides additional information on ratemeters and ratemeter-scalers.

NaI(Tl) crystals are manufactured in common sizes, typically cylinders of 2.5, 5.1, or 7.6 cm (1, 2, or 3 in.) in height and diameter, although much larger crystals are manufactured for special applications. The larger the crystal volume the greater the gamma photon interaction probability is that a gamma photon that enters the volume of the crystal will deposit all of its energy and be recorded as a count. Therefore, detector sensitivity increases with crystal size, including the background radiation, which means the large crystals will have relatively high background counts.

Not all gamma photons are the same, meaning they have varying energies, and the NaI detectors are energy dependent. At very low energies, the gamma may not be able to penetrate the housing and interact with the crystal. As energy increases, not only will the photon enter the crystal, but is likely to interact and deposit all of its energy, and be registered as a count. The sensitivity increases with increasing energy up to a point, when the higher energy photons may escape the crystal and sensitivity begins to decrease with increasing energy. Fig. 4 illustrates this concept for different sizes of gamma detectors.

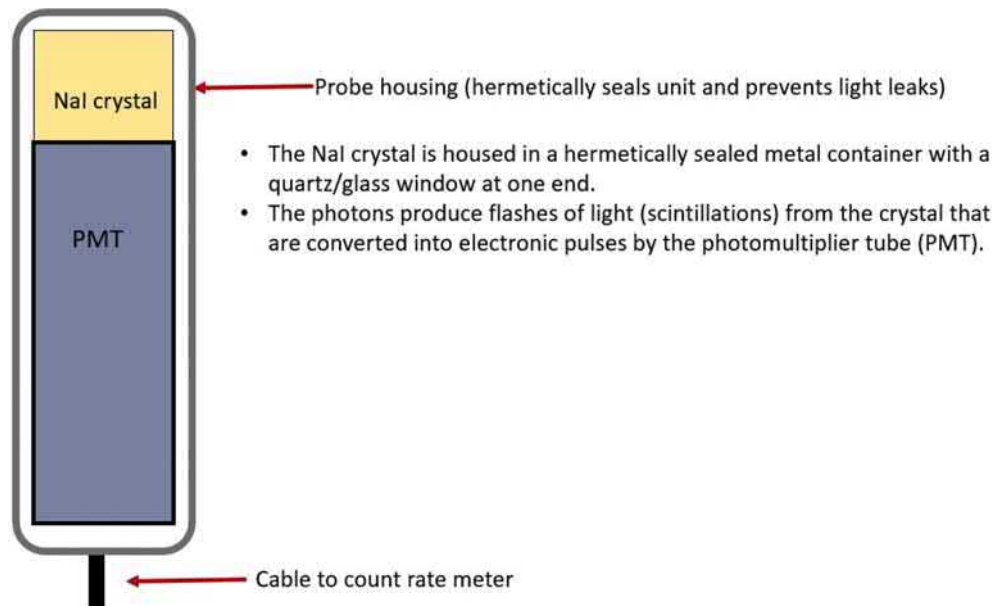


Fig. 2 NaI detector schematic.



Fig. 3 NaI with ratemeter-scaler (left) and μ R meter (right).

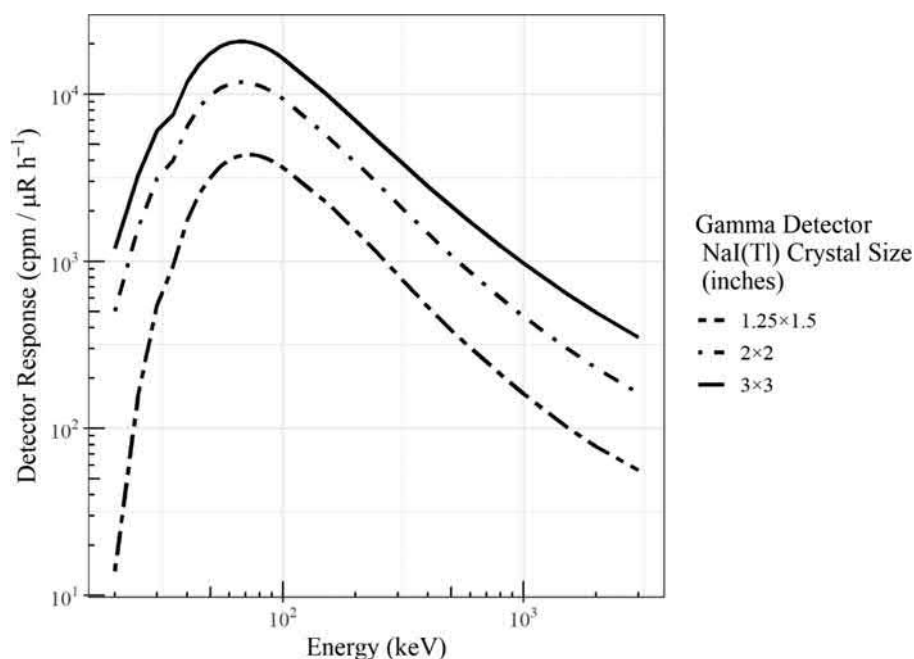


Fig. 4 Comparative energy response curves for NaI(Tl) detectors (unlabeled curves are computer modeling results).

Specialty and semi-quantitative gamma radiation detectors

Radiation detection applications can take advantage of the energy dependent response illustrated in Fig. 5. Detector modifications such as crystal geometry, additional shielding, or electronic discrimination can be used to focus detection on specific energy regions, referred to as regions of interest, or ROIs of the energy spectrum. Radionuclides emit characteristic gamma rays that are consistent in both energies (measured in kiloelectron volts, keV) and the abundance. Where abundance is, the fraction of decays that result in a characteristic photon. Therefore, specific radionuclides can be identified and quantified by knowing (1) the energies of the photons, (2) the abundance of the specific gamma emission, and (3) the gamma detector efficiency at a given energy. More on that later.

There are physical and logistical constraints to consider when selecting the right detector for the job. Detectors with large crystals, although more sensitive, are heavy to carry during a walkover survey, and often require a cart or other mode of transportation. Fig. 5 shows a custom-made NaI(Tl) gamma detection system. The case contains two 6-in. (15.2 cm) by 6-in. (15.2 cm) NaI(Tl) detectors that are coupled to a ratemeter. The case is attached to a cart due to the size and weight of the system.

Another specialty detector is the FIDLER. The FIDLER (field instrument for the detection of low-energy radiation) has a large-area but very thin crystal. This particular detector was made to be sensitive to the low-energy x-rays emitted by Pu-238 as well as other low-energy emissions from some other radionuclides. Because of the thin crystal, higher energy photons will pass through without interacting. However, crystal geometry is not all that is required as the typical metal housing would have blocked the low-energy photons that the FIDLER is designed to detect. Therefore, the design includes a very thin window that the low-energy photons are able to penetrate, as illustrated in Fig. 6.

Vehicle mounted plastic scintillators

An alternative to NaI(Tl) detectors are plastic scintillators made from organic scintillating molecules that are suspended in a solid polymer base. Relative to NaI(Tl), plastic scintillators are much less expensive and can be manufactured in most any size or shape. The plastic scintillators are commonly used for applications where very large detectors are required. The detectors are configured in an array and either mounted directly to vehicles or towed behind. They are also found in fixed systems, such as a portal monitor where vehicles drive between what is normally two banks of detectors, one on either side. These large detectors are quite sensitive and if configured appropriately, are able to identify a specific region of a tractor-trailer or other load that is exhibiting elevated direct radiation levels and may require investigation. The disadvantage of the plastic scintillators is that the photon energy resolution is much less than that of the NaI(Tl) making them generally unsuitable for spectroscopic applications.

Peripherals

Surprisingly, there has been very little change in detector technology over the decades since radiation measurement systems were developed. Most technology advances have been in the peripherals, primarily miniaturization of the electronics that are coupled



Fig. 5 Encased cart-mounted large volume NaI(Tl) detector array coupled to GPS system.

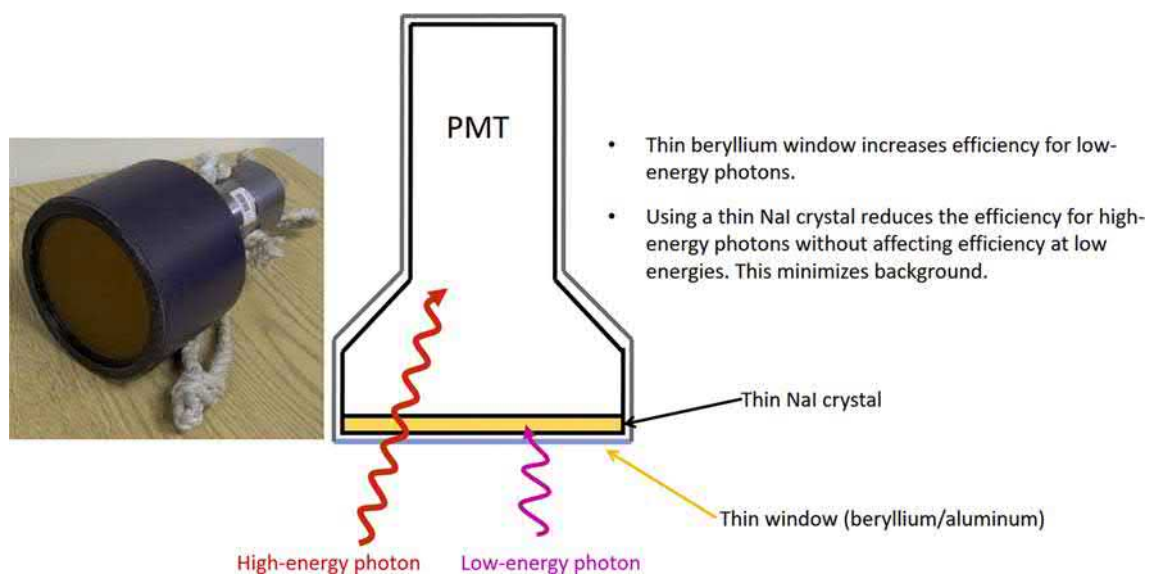


Fig. 6 FIDLER gamma radiation detector.

to the detectors and software development. Important advances are the global positioning systems (GPS) and data logging systems which enable exceptional radiological investigation documentation. Until these systems became common place, survey data over large land areas consisted of a hand-written notes and annotated hard-copy maps. The surveyor might note the general observed count rates and whether any anomalies were identified and investigated. Data loggers now electronically record count rate data and geographic reference position every 1 or 2 s. So rather than a hand-written notes, there can be thousands of individual data points available as survey documentation. These data are then managed using computer systems and specific software to illustrate the results of the survey. In most cases, the data are presented as color-coded dots superimposed on an aerial photograph of the site. With the color-coding, anomalies should be readily identifiable to show contamination or conversely, the illustrate the absence of contamination as evidence stakeholders in a final report showing that a site has been adequately decontaminated. Fig. 7 is an example image of the data collected from a land area with two buildings present. The blue, green, and yellow shades illustrate variable background gamma radiation levels that are due to the varying concentrations of naturally occurring radioactive material (NORM) in different soil types present. Count rates of suspected anomalies are identifiable in orange, red, and magenta. This color coding, noting that red and magenta indicate the highest count rates, suggests contamination surrounding the building and at isolated locations seen as small red areas in both the upper and lower left-hand side of the Fig. 7. Contamination? Such locations would be investigated, possibly sampled for laboratory analysis, and if found to be contamination, would be addressed. The analysis of samples collected showed there was no contamination in the soil around the building. Actually, the higher levels around the buildings were due to the higher NORM concentrations common to some types of brick. Samples collected from the isolated locations away from the buildings would also be necessary to determine if the anomalies were from NORM within rocks that were common to the area—or represented contamination.

Other peripherals can be added to the gamma radiation scanning or fixed position detectors and instrumentation that have capability to perform spectroscopic analysis in real-time, possibly avoiding or greatly reducing the need for sample collection and



Fig. 7 Gamma radiation walkover scan data.

analysis. Such systems, discussed later, might include electronics and software to take advantage of the signature gamma photons emitted by various radionuclides, including NORM, that resolve the gamma radiation scan data into ROIs for radionuclide identification.

Common structure/surface survey instrumentation

Thus far, discussions have focused on radiation detection instrumentation for investigations of land areas. Perhaps more relevant to the forthcoming discussion is the edification that the gamma radiation measurements observe photons emitted from the volume of the medium, not just the surface. This distinction is important because volumetric sources (e.g., soil or drummed materials) can emit enough gamma radiation to detect from a distance—the detector does not have to be in contact, but can be several centimeters, even meters from the source. When radiological contamination is spread over a thin layer (e.g., dust or spill residues) on a surface, gamma radiation measurement may not be ideal when the contamination levels are much lower than that seen in a volume. This is not to say that gamma scans of structural surfaces would not be performed, because they absolutely would in many cases. More importantly, the specific radionuclide contaminating the surface may not emit gamma radiation. Traditionally, surveys of structural or other surfaces are for alpha and/or beta radiation and the measurements are reported as alpha, beta, or total activity levels—in units of dpm 100 cm⁻² or Bq cm⁻² (disintegration per minute per 100 square centimeters or Becquerels per square centimeter)—for an individual or mixture of radionuclides. The detectors are held close to the surface because the air gap between the detector window and the surface may act as a sufficient shield for alpha and beta radiation. This is especially true for alpha radiation, which can travel only a few centimeters in air. Therefore, alpha and beta detectors are held about a centimeter or less from the source's surface.

Much like with land area surveys, there are many instrument choices, and sometimes the best instrument for the contaminant(s) in question is not obvious. **Table 1** describes common detectors used for structures/surfaces surveys; each has advantages and disadvantages, as will be discussed. The trick is finding the right detector(s) to address the project objectives and to ensure the detectors respond to the radiations of concern. Some variables to consider while selecting structures/surfaces survey instrumentation are:

- Surface area—if surveying large areas like walls and floors, the best selection may be the detector with the largest probe area (i.e., the physical surface area assessed by the detector). Conversely, tight spaces that cannot be physically reached by a larger detector, require the smallest detector area.
- The radiation type—does the contaminant emit alpha radiation, beta radiation, or alpha and beta radiation? Do the surveyors have to report different results for each radiation type which would require multiple instruments to ensure that all the necessary data are collected?
- Convenience and mobility—there are cases where a surveyor needs to work in difficult environments, such as working from a ladder or in a confined space, and cannot be bogged down with a lot of bulky equipment. In these cases, the surveyors may only want a single instrument that covers all bases, or a few instruments that are easy to maintain and use.
- Detection capability—if the most important survey objective is the ability to detect very low contamination levels, the options narrow quickly and the detector selection process is extremely important. The wrong selection can lead to poor data quality, false

Table 1 Comparison of common structural/surface survey instruments.

<i>Instrument</i>	<i>Relative α sensitivity</i>	<i>Relative β sensitivity</i>	<i>Probe Area^a (cm²)</i>	<i>Pros</i>	<i>Cons</i>
Geiger Mueller	Poor	Fair	15.5	Easy to use, small detector for collecting measurements in hard-to-reach spaces	Poor to fair detection capabilities; small surface area so small coverage
Alpha Scintillator	Better	N/A	100	Easy to use; convenient when specifically trying to measure α-only radiation	Works for α radiation only
Plastic Scintillator	Better	Better	100	Easy to use; responds to both α and β radiation	Cannot discriminate between α and β radiation; also responds to γ radiation which is problematic in high radiation areas
Dual Phoswich	Better	Poor	100	Easy to use; can switch between α and β measurements using single instrument	Convenience of having one instrument for α and β balanced by poor to fair detection capabilities
Gas Proportional	Best	Best	Hand-held: 126 Large-area floor monitor: 584	Best detection capabilities; responds to both α and β radiation	Requires P-10 gas and extra tubing; need different electronic setup for α and β measurements (might need multiple detectors)

^aLudlum Measurements, Inc. Model 44–9 Geiger Mueller; Model 43–93 dual phoswich; Model 43–92 alpha scintillator; Model 44–142 plastic scintillator; Model 43–68 gas proportional.

negative decisions—deciding that contamination is not present when in fact it is—or errors that lead to rework, misrepresentation of radiological conditions, and other negative consequences. Consider an example where personnel health and safety and limiting exposures is a primary objective. Misrepresenting the radiological conditions could lead to personnel being exposed to higher contamination levels than those planned for. Conversely, in the case of over-stating levels, unnecessary personnel protective equipment might be required that introduce complications from additional physical stress or the addition of costly engineering controls.

Before describing individual detectors, saying that detector will detect either alpha or beta radiation is a bit of a misnomer. It is more accurate to say that some instruments can detect either alpha, beta, or alpha-plus-beta radiation. Which ones depends on the specific detector and the detector configuration. For example, an alpha scintillator only measures alpha radiation (hence the name), while a GM detector responds to alpha (α), beta (β) and gamma (γ) radiation, although they are not very efficient for alpha and gamma. The surveyor cannot tell which radiation type resulted the GM's audible "clicks" without additional procedures to help discriminate radiation types. Some detectors like the gas proportional detector can be configured to discriminate radiation types—the gas proportional detector can be configured to respond to only alpha, only beta, or alpha-plus-beta radiation, as will be demonstrated.

Detectors are useless by themselves unless they are either integrated with an electronics package or connected via cable to a ratemeter-scaler which was introduced earlier when discussing land area detectors. The main purpose of the ratemeter-scaler is to: (1) provide the voltage necessary for the instrument to function (e.g., from batteries), (2) produce an audible response ("clicks") that suggest the presence or absence of contamination, and (3) display the detector's response so that surveyors are not solely reliant on the audible clicks. For example, approximately 900 V are required for a typical pancake GM to operate properly—this voltage is supplied by the batteries in the ratemeter-scaler. When the GM is used in a survey, the ratemeter-scaler will generate clicks through a microphone or headphone jack, where the rate of clicks (a.k.a., cpm) are proportional to contamination levels. A surveyor may be able to hear an increase in the count rate but will not be able to quantify that increase. Therefore, the ratemeter-scaler displays the count rate using digital and/or analog display. Consider this scenario:

A surveyor connects a GM to a ratemeter-scaler via a cable, turns the ratemeter-scaler on, and notes that the voltage is set at 900 V—perfect. The surveyor then begins a survey noting an ambient count rate of about 60 cpm—i.e. typical background for a GM detector—on the display and the intermittent random "click—click—click—click—click—click—" audible output. As the survey progresses, the surveyor hears an increase in the audible clicking rate, looks at the display, and sees that the rate has jumped to approximately 200 cpm. So, the ratemeter-scaler has provided the necessary voltage, the audible stimulus that caused the surveyor to take notice, and quantitative count rate from a potentially contaminated area.

All the detectors listed in [Table 1](#), with the exception of the dual phoswich detector, can be coupled to the same type of ratemeter-scaler. The discussions following [Table 1](#) emphasize the detector, not the ratemeter-scaler.

Geiger Mueller detector

Geiger Mueller detectors come in a variety of configurations including "pancake," end-window, and sidewall. Pancake GMs are among the most widely-used radiation detectors for contamination surveys, and are often coupled with alpha scintillators (discussed later) as "friskers"—scanning hands and feet or personal items before exiting a potentially contaminated area—for beta and alpha radiation measurements, respectively. [Fig. 8](#) illustrates an health physicist performing a contamination survey using a pancake GM detector. As shown in [Table 1](#), advantages of using GMs are that they are easy to use and the small surface area (15.5 cm²) allows for surveys in hard-to-reach places where larger (e.g., 100–126 cm²) instruments cannot fit. While advantageous for surveys of hard-to-reach areas, the small size can be a disadvantage when surveying large areas—small detector area means more time is required to scan the surface area. Finally, other detectors described later in this section are better for measuring beta radiation (the GM is rated as fair), and GMs are rated as poor for measuring alpha radiation. Therefore, surveyors may select a pancake GM for beta frisking, for performing routine measurements when a lightweight mobile instrument is needed, or for performing measurements in a hard-to-reach area; but different instruments are likely better for large area surveys (e.g., floors, walls, an entire room or rooms, etc.) or when more sensitivity is required.

Before evaluating the next class of detectors, other GM configurations should be considered: end-window and sidewall GM detectors. The end-window is functionally the same as a pancake, though the pancake is relatively flat with a handle for ease of use while the end-window is cylindrical shaped and does not have a handle. The side-wall GM includes a thin metal filter that reduces the number of low-energy gamma rays reaching the detector when in the "closed" position, so surveyors can collect measurements in the open and closed position to see what contributions are from beta and beta plus gamma, and are most often used for measuring exposure rates. It is also noteworthy that GMs detectors will over-respond to low energy gamma (e.g., <200 keV) without energy compensation. Energy compensated side-wall GMs are available sometimes called "hotdog" or "pickle" probes, and different designs are available to measure low exposure rates (i.e., $\mu\text{R hr}^{-1}$ range) to high radiation exposure rates (e.g., 1 R hr⁻¹).



Fig. 8 GM detector in use to survey lab work counter.

Zinc sulfide alpha scintillator

Zinc sulfide silver activated (ZnS(Ag)) scintillators are designed to measure only alpha radiation, which can either be limiting or handy, depending on objectives. If, for example, only alpha emitters, like thorium-230 or plutonium-239 are present, then all the surveyor needs is a ZnS(Ag). If the contaminant includes a beta component, and beta data are needed to meet project objective, then another instrument will be required. As mentioned in the previous discussion, ZnS(Ag) and pancake GMs are commonly paired to measure alpha and beta radiation, respectively, for frisking or performing routine surveys. There are other detectors more efficient than a ZnS(Ag) for measuring alpha radiation, meaning it is not the best for some situations, but the detector is easy to use, and sometimes it is important to know that the only response is due to alpha radiation.

Zinc Sulfides typically have a 50 to 100 cm² probe area and are suitable for large area scans, such as walls or large pieces of equipment. The sensitive part of the detector is covered by a thin aluminized Mylar[®] window. The Mylar[®] window must be thin enough to allow alpha particles through to the detector, but must also be effective at blocking out all visible light. These fragile windows must be tested for light leaks before each use, and should be protected while in storage or transport.

Plastic scintillator

Plastic scintillators include a thin layer of Mylar[®] over a thin plastic detector and are designed to measure beta radiation. However, plastic scintillators will also respond to alpha and gamma radiation, and cannot discriminate between radiation types. Therefore, the surveyor may need to know which contaminants are present and the radiation types emitted, to fully understand the nature of the response. Because plastic scintillators will respond to gamma radiation, and as noted earlier are sometimes used for land area gamma radiation surveys, it may be important to account for background radiation levels while performing surveys near, for example, drummed materials emitting non-trivial levels of gamma radiation. As with the other detectors discussed so far, surveyors may need another detector (e.g., a ZnS(Ag)) if it is important to quantify alpha and beta activity separately. Surveyors may also take advantage of detector's sensitivity to multiple radiation types. For example, contaminants like radium paint and thorium oxide emit alpha, beta, and gamma radiation, and plastic scintillators can improve detection capabilities over other instruments discussed here by responding to all these emissions. Fig. 9 illustrates a surveyor performing a surface scan using a hand-held scintillation detector. The technician in Fig. 10 is using a gas proportional detector to scan for radioactive contamination on the inside of a research reactor cooling water pump cover.

As shown in Table 1, plastic scintillators are ranked as "better" for alpha and beta detection, relative to the other detectors listed. It could be ranked as best overall for alpha radiation, except the specific gas proportional described later has a larger probe area



Fig. 9 Surveyors using a plastic alpha/beta scintillator.



Fig. 10 (A) Example of a hand-held gas proportional detector. (B) Technician scanning an object with gas proportional detector coupled to P-10 gas supply and ratemeter-scaler.

(126 cm^2) than the plastic scintillator (100 cm^2), giving the gas proportional certain practical advantages. The relatively large probe area and ease of use make plastic scintillators suitable for large area scan, such as walls or large pieces of equipment.

Dual phoswich alpha/beta detector

Imagine the versatility if a detector was available that combined both a $\text{ZnS}(\text{Ag})$ scintillator and a plastic scintillator, and then also included a toggle so that the surveyor could discriminate between the alpha, beta, and alpha-plus-beta response? Such a detector is available. A dual phosphor alpha/beta detector includes a Mylar[®] window, followed by a thin layer of $\text{ZnS}(\text{Ag})$ that is sandwiched to a plastic scintillator. A specialized ratemeter-scaler is required to be coupled to the detector. The ratemeter-scaler has a discriminator toggle switch that surveyors use to discern the radiation type or types of interest that are being counted. Beta particles are far less massive and overall much less energetic than alpha particles. In addition, the beta particles are able to penetrate further in a media, whether that media is air, a piece of paper, or, in this case the detector volume itself. The alpha particles will hit the $\text{ZnS}(\text{Ag})$ layer,

deposit all their energy and stop, while the beta particles will pass through the ZnS(Ag) layer and interact with the plastic scintillator portion of the detector. From an energy standpoint, think about the difference between dropping a pebble (beta) and a bowling ball (alpha) from a tower into a deep pool of water. The detector's electronics has two channels; one that can tell the difference between the "small ripple" and second that tunes into the "large splash" of energy that has been deposited. One switch setting is used to respond only to small pulses (beta); another switch setting is used to respond to only large pulses (alpha), and yet another switch setting is used to combine the pulses (alpha-plus-beta as well as some gamma). The big advantage is that surveyors only need one detector for measuring alpha and beta radiation—combined or individually—so one instrument checks several boxes:

- Relatively large (100-cm²) probe area—check
- Detects multiple radiation types with discrimination—check
- A convenient and mobile detector—check

There are a couple of disadvantages. The first is the need to acquire the specialized ratemeter-scaler and the second is a loss in detection sensitivity. Sensitivity is typically described in terms of efficiency, where efficiency is basically the fraction, in terms of percentage, of radioactive particles that are counted per the total number of radioactive particles that were emitted from the surface beneath the detector during a given unit of time. **Table 1** shows that the detector is rated as better for alpha and poor for beta efficiency, specifically low-energy beta particles. This means that in cases where it is important to find low levels of beta contamination, surveyors should probably leave the dual phoswich on the shelf. However, dual phoswich may fit the need if convenience and flexibility are desirable.

Gas proportional detector

The gas proportional detector includes a Mylar[®] window and requires a continuous flow of P-10 gas (10% methane, 90% argon) to operate. The gas is supplied from a tank connected via a tube. By adding the gas tube and tank to the cable from the ratemeter-scaler, the proportional detector is the least convenient to use of those considered here. **Fig. 10** illustrates an example hand-held gas proportional detector (A) and a technician using this detector to scan an object (B). So the obvious question is, why would a surveyor choose to use a gas proportional detector when other options are available? There are two reasons. First, gas proportional can be configured to discriminate between alpha, beta, and alpha-plus-beta radiation. Discrimination is accomplished by varying the Mylar[®] thickness and voltage, like so:

- Alpha-only response by using thinner Mylar[®] and a lower voltage setting.
 - o The Mylar[®] is thin enough for both alpha and beta particles to pass through (mass thickness of 0.4 or 0.8 mg per square centimeter [mg cm⁻²]), and
 - o The voltage is high enough to produce pulses from alpha radiation but is too low to generate a pulse from beta radiation (1100–1400 V).
- Alpha-plus-beta response by using thinner Mylar[®] and a higher voltage setting.
 - o The Mylar[®] is thin enough for both alpha and beta particles to pass through (0.4 or 0.8 mg cm⁻²), and the voltage is high enough to generate a pulse from both alpha and beta radiation (1600–1800 V).
- Beta-only response by using thicker Mylar[®] and a higher voltage setting.
 - o The Mylar[®] is too thick for alpha to pass through but still thin enough to let beta particles to pass through (3.2 mg cm⁻²), and
 - o The voltage is high enough to generate a pulse from beta radiation (1600–1800 V).

These detectors can be used with a dual channel ratemeter to distinguish between alpha and beta response. If paired with a single channel ratemeter, multiple instruments will have to be configured if results for different radiation types are required. For example, one detector can be set for alpha-only, and another for alpha-plus-beta or beta only.

But still, why would a surveyor go through the hassle of the gas tank and tubes, and multiple detectors for different radiation types? The answer is the second reason for selecting gas proportional over other options. As shown in **Table 1**, the gas proportional detector rates "best" for sensitivity to both alpha and beta radiation. This means that in cases where it is important to find low levels of contamination, surveyors may opt for, or even be required to use, gas proportional detectors. The largest probe area or 126 cm² (of those considered) also means the gas proportional is well suited for large area surveys. Speaking of large area detectors, gas proportional detectors also come in large-area varieties (e.g., 500 to over 800 cm²) which are commonly mounted on carts.

The gas proportional detector's high sensitivity to alpha and beta radiation, large probe area, and cart mobility make these detectors ideal for scanning large flat areas, like floors. Because a cart is used, the extra gas tubing and tank is less burdensome. Even when the relatively high sensitivity of a gas proportional is not required, surveyors may want the floor monitor to more easily cover large surfaces in a relatively short time. If elevated radiation levels are discovered using the floor monitor, a surveyor can always use one of the hand-held detectors to hone in on the specific "hot spot" location.

Specialized survey equipment

Specific scenarios may arise throughout a decommissioning process where the instrumentation discussed thus far is either inconvenient or inappropriate to address the problem. This section will outline some instrumentation and equipment used to address

specialized decommissioning problems. Most of the instrumentation presented in this section is commercially available; however, some problems will require novel instrument designs to be manufactured in-house.

Non-destructive assay

Non-destructive assay (NDA) refers to the assessment of a material's radiological properties without the use of destructive means and *ex situ* analysis at a laboratory. Collecting volumetric concrete cores and analyzing the samples at an analytical laboratory is an example of destructive assay. Here passive NDA techniques using gamma ray spectrometry common to decommissioning are outlined. Note that some of the instrumentation previously presented in Section "Common structure/surface survey instrumentation" may fall into the NDA category, however, for the purpose of this chapter, NDA refers to measurements that are intended to quantify radionuclides *in situ*.

High resolution In Situ gamma spectrometry

A lack of information related to the radionuclide source term is a problem for many radiological investigations or material evaluations such as site or waste characterizations. As such, for NDA measurements it is desirable for the detector to resolve neighboring gamma energies associated with different radionuclides. A high purity germanium (HPGe) detector is highly desirable for these applications. Fig. 12 illustrates the use of an HPGe detector for quantifying the radionuclide concentration in structural concrete. The detector setup in Fig. 11 is relatively unremarkable, other than the form of the cryostat and dewar, the components are similar to those used in an analytical laboratory. The key difference is how the efficiency calibration of the detector is performed. Clearly, it



Fig. 11 *In situ* gamma spectrometry measurement of a wall using a portable HPGe detector.

would be difficult to fabricate a standard that represents the measurement geometry presented in Fig. 12. A solution is to use computer software to simulate the detector efficiency under a variety of measurement geometries. From the computer estimated energy efficiency curve and the acquired measurement spectrum, a quantitative value, in terms of activity units, is determined. Often regulators view this equipment as a “black-box” technology. Therefore, it is incumbent on the user to demonstrate that the detector system is accurately assessing the estimated parameter. Quality assurance/quality control (QA/QC) programs typically involve performance testing and confirmatory measurements. For example, a confirmatory measurement may include collecting volumetric samples representative of the detector field-of-view. Volumetric samples may be required anyway to assess hard-to-detect radionuclides that are not detected by gamma spectrometry.

The HPGe detector system mentioned previously can be further specialized for more specific applications, such as automated waste monitoring systems. Commercially available solutions exist for monitoring waste packages or waste shipments. The primary drawback to HPGe detectors is that these detectors are operated at liquid nitrogen temperatures. Detectors can be outfitted with electromechanical coolers that eliminate the need for liquid nitrogen, however these detectors remain sensitive to mechanical shock and must be handled with care.

Hold-up measurements

Consider a decommissioning project that involves dismantling and demolishing a uranium enrichment facility. These facilities house miles of piping that retains a build-up of process material (referred to as product hold-up). When the building is in the normal configuration, this hold-up is in a safe geometry. Now, what happens then we remove the piping and place it in a large pile for disposal. The hold-up went from a distributed state to a fairly condensed pile that may be favorable for a criticality event (unwanted nuclear fission). In order to prevent a criticality event, the quantity of hold-up remaining in the piping must be verified. There are commercially available systems (Fig. 12) that utilize the Generalized Geometry Hold-up methodology, developed at the Los Alamos National Laboratory, to measure the product inside a pipe without dismantling it. The system consists of a specialized detector with either a NaI or LaBr crystal coupled to a portable multi-channel analyzer (MCA) and hand-held software controller to measure 186 keV uranium-235 decay gamma emission and then quantify the weight. As with the previously discussed HPGe-based *in situ* measurement system, performance demonstration and QA/QC programs are important to ensure the measurement system can identify holdup volumes.

Hand-held gamma spectrometers

There are instances where it may be beneficial to simply identify a gamma-emitting radionuclide rather than quantitatively determining its concentration, which greatly simplifies the problem. The detector and associated electronics for this task can be made in a compact form for ease of use in the field. Fig. 13, shows two different isotope identifiers. Crystals used for these applications are lower in resolution than HPGe, such as NaI, cadmium zinc telluride (CZT), and LaBr, which may give rise to limitations depending on the site’s source term and action level objectives. These instruments are fairly easy to use, and operate more along the lines of point-and-shoot, rather than carefully positioning the detector. Gamma energy spectrum acquisition time is on the order of a few minutes, whereas an *in situ* measurement with an HPGe detector may take 15 min or longer. The electronics package typically contains a complex algorithm to analyze the spectrum in order to compensate for the reduced resolution. The unit accesses the

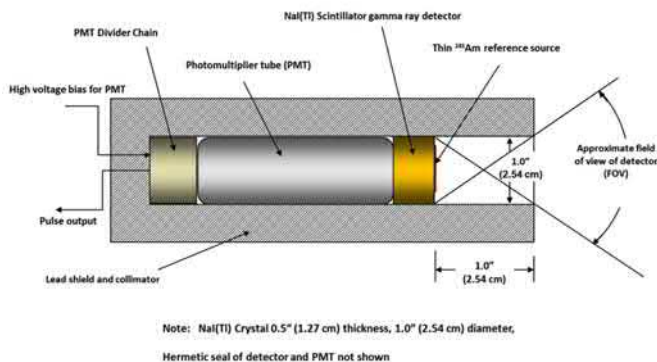


Fig. 12 Example of a measurement system used to quantify uranium-235 hold-up.



Fig. 13 Example of two different hand-held isotope identifiers with LaBr (larger unit) and CsZnTe (smaller unit) detector crystals.

built-in gamma energy library to tentatively identify and then display the radionuclide(s) present. It is helpful for the technician to exercise professional judgment when using this type of instrument.

Pipe detectors

Detectors fabricated for investigating piping generally consist of one or more of the detectors presented in Sections “**Land area (surface soil) survey instrumentation**” and “**Common structure/surface survey instrumentation**”, though packaged in a ruggedized housing. These detectors will collect either qualitative or quantitative measurements, depending on the calibration process. **Fig. 14** shows four different types of pipe monitors, the top two (A and B) are custom made GM based detectors, while the bottom two (C and D) are commercially available scintillation detectors. The type of detector selected for use in the field depends on a number of factors, including the radionuclide of concern and the condition of the pipe. Beta sensitive detectors, such as those in **Fig. 15B** and C, are prone to punctures from debris. A data logger may be coupled to the instrument to electronically record the count rate data. Survey of piping can be extremely challenging due to lack of access or other unknown parameters about the pipe. These detectors are required to be placed inside the piping or ductwork, then physically pulled through, or in some cases can be mounted to a robotic transport vehicle.

Calibration of the detector is an important factor to consider, and it can be challenging to calibrate the detector in a geometry that is representative of field conditions. If hot spots are a concern, then it may be appropriate to calibrate the detector with a small area source, such as button source. However, for typical decommissioning projects involving the survey of piping, the parameter of interest is related to the total residual activity contained within the pipe. Therefore, it may be necessary to calibrate the detector with a large area flexible source. The flexible source is placed inside of a pipe surrogate to simulate the geometry of the field measurements.

Gamma imaging systems

Remediation planning and implementation would be a lot easier if we could visually see the radiation. Human senses cannot detect ionizing radiation, we therefore need to use sophisticated detector systems to audibly or visually represent the radiation. Detectors in this visual classification are often informally referred to as gamma cameras. CZT is a typical detector crystal of choice due to room temperature operation, and better resolution than NaI. The gamma camera is placed in a stationary position in front of the item to be imaged, and a spectrum is acquired. Using complex computer algorithms and the acquired spectrum, the radiation intensity is overlaid on a digital image. **Fig. 15** provides an example of a resulting image. Because the CZT detectors and electronics typically have spectroscopic capability, different heat maps for each radionuclide of interest are generated. Pure beta emitters are not detected by these systems, so one must either know the relationship of the beta emitter to a gamma emitter, or use a different type of detector and measurement method (i.e. scanning with a plastic scintillator).

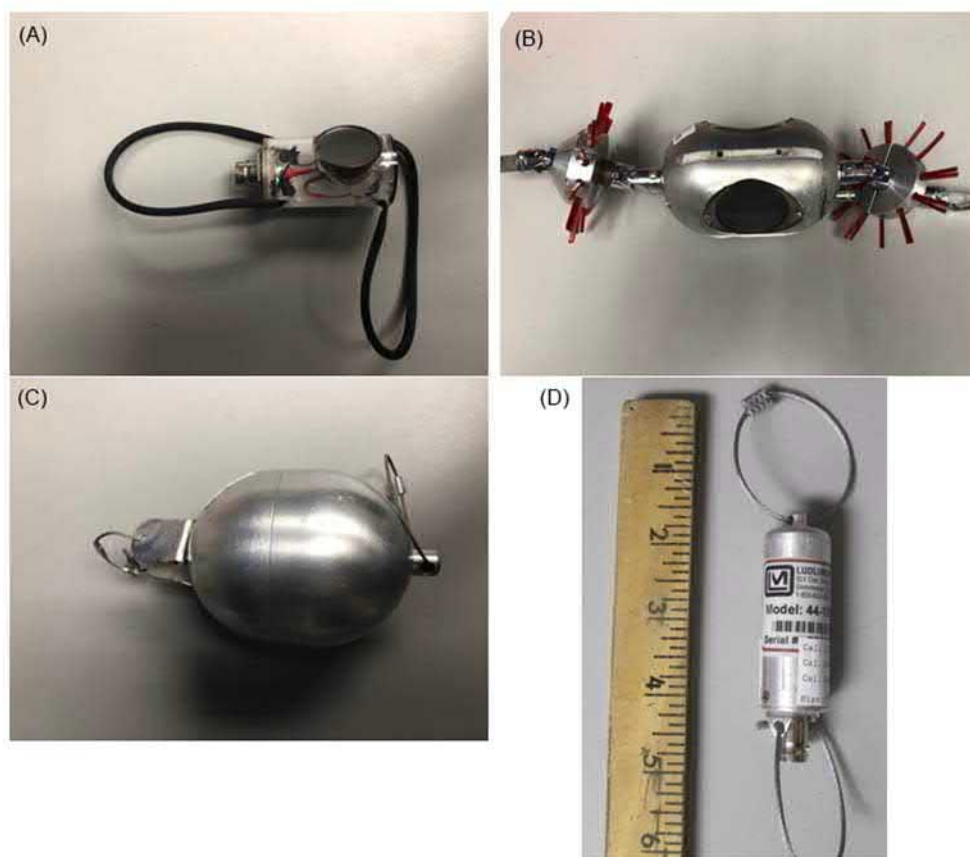


Fig. 14 Specialized commercial and custom-built detectors for assessing pipe internals: (A) small GM-detector, (B) 3 GM detectors in a ruggedized housing, (C) 5.1×5.1 cm NaI Ball detector, and (D) CsI detector.

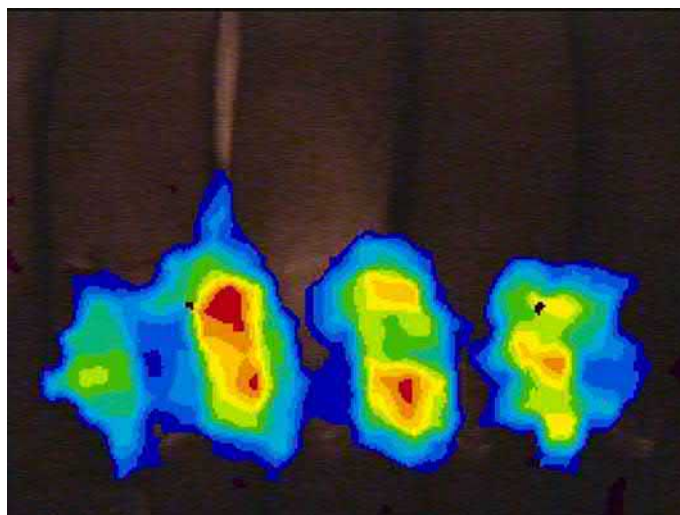


Fig. 15 Example gamma camera image of reactor piping (red represents the highest radiation levels).

Position-sensitive structural survey systems

Because building structures block satellite reception of GPS receivers used for outdoor gamma walkover surveys, alternative methods must be employed indoors if one wishes to generate position and radiation level maps similar to Fig. 7. A laser measurement system, similar to that used by highway surveyors, can be repurposed to map indoor radiological surveys. Fig. 16 shows



Fig. 16 Large-area gas proportional floor monitor coupled to position sensitive robotic total station.

a large-area gas proportional detector used to survey floors and with the right type of cart and adapter arms, the detector could be turned vertically to scan walls. The detector in the example includes a prism mounted on a pole. The prism allows for continuous line-of-sight laser tracking using the robotic total station seen mounted on the tripod, and with a data logger, the count rate and position data can be electronically recorded and mapped similar to that described for land area surveys. Fig. 17 shows a 3-D plot of structural scan data collected with such a system.

Additionally, there are automated scanning systems that remove the human aspect of scanning. A detector is attached to a track that is mechanically moved across the building surface. The obvious advantage of these systems is that the motion of the detector and distance to the surface remains constant as surveyor fatigue is eliminated. Disadvantages to these type of systems include a fairly complex setup process and limitations due the physical layout of some structures that interfere with (block) the laser tracking

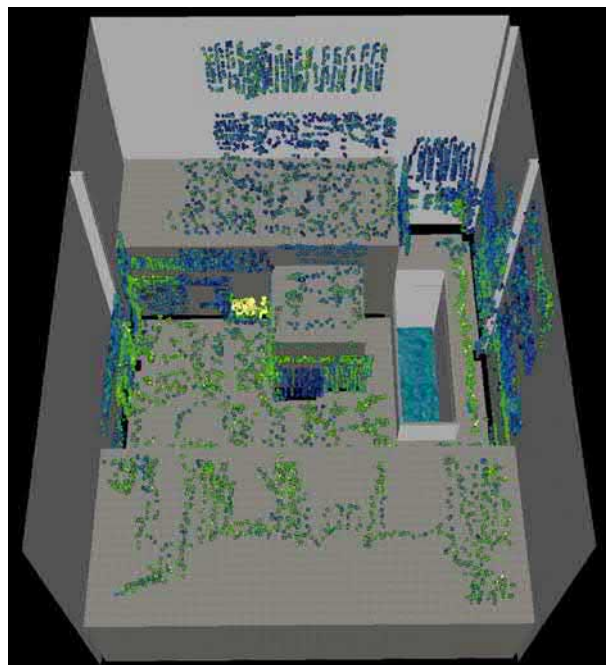


Fig. 17 3-D rendering of a research reactor bay alpha-plus-beta scan data.

resulting in the need to reposition the base station, which can be timely. Surfaces that are not smooth, because of remediation, may prevent the use of automated systems, unless system can continuously vary the surface to detector distance to account for uneven surfaces.

Laboratory analysis and instrumentation

Portable field detectors and associated hardware play a key role in routine radiological investigations of land areas and structures during decontamination and decommissioning activities. In fact, one may argue that these types of detectors and instruments are the dominant tool used for the detection and, under certain conditions, the quantitative determination, of radioactive material during clean-up activities. Although these detectors provide a large amount of information and data to the user, there are times in which more information and data about the material being scanned is needed. Laboratory analyses may be needed to obtain detailed sample information and data. Laboratory analyses optimize measurement conditions and has several advantages over field measurements. Some of these advantages include identification and quantification of specific radionuclides, individual radionuclides separations, better detection limits, and better quality control.

This section discusses some of the commonly used radiological sample analysis methods and the associated laboratory instrumentation. Sample analyses can be divided into two broad categories: non-destructive analyses and destructive analyses. The types of instrumentation used for sample counting is often used for both types of analyses.

Non-destructive analysis

Non-destructive analyses are types of sample analyses in which the sample is not processed in a way to change its physical or chemical form (other than drying the sample). In addition, the sample is not consumed during analyses. Two common non-destructive analyses in a radiochemistry laboratory include gamma spectrometry and gross alpha and beta (GAB).

Gamma spectrometry

Gamma spectrometry allows for both the identification and quantification of multiple radionuclides simultaneously. A large percentage of radionuclides—such as cesium-137, cobalt-60, and many others—that are often associated with decontamination and decommissioning sites can be analyzed and quantified by gamma spectrometry. In fact, only radionuclides that are pure alpha or pure beta emitters cannot be analyzed by gamma spectrometry.

Gamma rays are electromagnetic radiation produced in the nucleus of an atom. These rays are photons and have energy (50–3000 keV) but no mass or charge, and therefore, do not readily interact or transfer their energy in matter. When a gamma photon does interact and deposit energy, the interaction is via one of three principle means: the photoelectric effect, Compton scattering, or pair production. All three types of interactions occur when a gamma photon interacts with the detector material. However, it is the photoelectric effect interaction that creates the well resolved energy peak from the radionuclide.

Most current laboratory gamma spectrometry systems use solid-state semi-conductor detectors, namely HPGe, coupled with a high voltage power supply, pre-amplifier, amplifier, MCA, analogue to digital converter (ADC), shield, dewar with liquid nitrogen (the HPGe must be kept at liquid nitrogen temperatures), and computer software to deconvolute the spectra. Fig. 18 depicts a dewar and shield and Fig. 19 shows the inside of the shield with an HPGe detector.

Nearly any type of matrix may be analyzed by gamma spectrometry as long as there is an available calibration source (both for energy and efficiency) and counting geometry. Sample types include soil, water (ground, surface, drinking), air filters, smears, vegetation, construction materials, wastes, etc. Sample preparation for gamma counting is usually minimal and requires only homogenization of the sample and putting the sample in a calibrated counting container (geometry). However, for more difficult matrices like construction material, the sample may need to be acid-leached prior to counting. HPGe detectors have extremely good energy resolution (~ 1 – 2 keV) and are typically used for gamma emissions between 50 and 2000 keV (although there are detectors that can be used below 50 keV). Fig. 20 shows a spectrum of a soil sample with the 186 keV peak for Ra-226 in orange (spectrum is shown with a linear scale).

Advantages and disadvantages of gamma spectrometry are shown in Table 2 below.

Gross alpha and gross beta for smear samples

There are some circumstances in which the total beta activity or total alpha activity (or both) of a sample needs to be determined. This total activity is called the “gross” activity of either alpha or beta activities. Gross alpha and beta (GAB) measurements are often used as a screening tool to determine.

GAB analyses are often performed on smear samples at D&D sites to determine if removable activity is present on solid surfaces (walls, floors, equipment, etc.) and if the removable activities are above some defined personnel protective or other limit. The smears collected in the field are sent to the laboratory, up to 100 smears loaded onto the instrument, and counted sequentially. The count times are usually short, thus, results can be provided to the project manager within a few hours.

Other sample types may be counted for GAB, including soil, air filters, and water. Usually these analyses are done to screen a sample for radioactivity and determine when radionuclide-specific analyses are necessary. Fig. 21 depicts two low-background



Fig. 18 Lead shield and dewar with liquid nitrogen.



Fig. 19 Inside lead shield with HPGe detector.

proportional counters used for GAB analysis. Note the long, extended sample holders which can hold up to 100 smears for sequential counting.

Advantages and disadvantages of GAB smear analyses are shown in [Table 3](#) below.

Destructive analysis

Destructive analyses are types of sample analyses in which the sample is processed in a way to change its physical or chemical form. In addition, the sample, or a portion of the sample is typically consumed during analyses. Often, the sample undergoes some type of

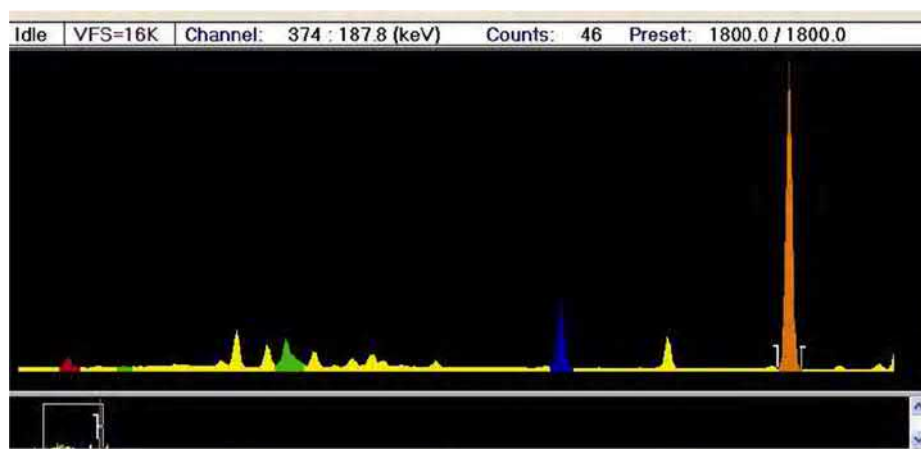


Fig. 20 Gamma spectrum showing the Ra-226 peak in orange.

Table 2 Advantages and disadvantages of gamma spectrometry using HPGe.

<i>Advantages</i>	<i>Disadvantages</i>
Non-destructive (typically)	Cannot analyze pure alpha or beta emitters (e.g. Pu-239, U-234, Sr-90, etc.)
Multi-isotope	Requires very good calibrations
Relatively fast analysis	Requires large and heavy shields
Several counting geometries (if calibrated)	Requires liquid nitrogen or mechanical chillers
Excellent energy resolution	Isotopes with similar energy must be interpreted by experienced workers
Good detection limits (with adequate count time)	
Many matrix types	
Excellent software available for data reduction	
Tolerates high radioactivity	
Good tool for sample radionuclide screening without quantification	
Relatively inexpensive analysis cost	

chemical separation to purify the radionuclide(s) of interest prior to sample counting. This section will provide a brief discussion on sample preparation for various sample matrices, along with the most common sample analyses performed after sample preparation—alpha spectrometry, liquid scintillation, and low-background proportional counting.

Sample preparation

Sample preparation is often the most overlooked and underappreciated aspect of sample analysis. The final analytical data generated from the sample analysis is only as good as the sample preparation prior to the analysis. Two major goals of sample preparation are to provide representative aliquots that are free from laboratory contamination and to avoid sample loss that will be used in the next steps of the laboratory protocol. It is also of utmost importance to understand the data quality objective for the project prior to commencing sample preparation as some preparation methods may not be suitable for the requested analyte.

For solid samples including soil, sediment, biota, animal tissue, filters, metal, concrete, etc.), the sample is typically dried to a constant mass at a low temperature. This is done because most radiological results for solid materials are reported on a dry weight basis (although some solid materials such as food-stuffs are often reported on an “as received” basis, or wet basis). After drying, the samples may also be dry ashed to remove unwanted organic material. Next, the samples are homogenized via grinding, blending, or other mechanical method. Matrices that cannot be homogenized—e.g. metal, asphalt, etc.—are often acid-leached to remove any radionuclides from the surface. The percentage of the activity removed is estimated by making a pre- and post-leaching measurement on the sample surface using a hand-held detector or by gamma spectrometry. The acid leach is then brought up to a known volume for use in the final concentration calculation.

Once homogenization and processing are complete, sub-samples (aliquots) are taken and subjected to the various requested specific analyses. The aliquots for destructive analyses are usually digested by some means—either complete dissolution or acid-leaching. Complete dissolution may be accomplished by a variety of fusion-type reactions at high temperature. The fusion ensures the analyte is taken into an aqueous media and allows the chemist to manipulate the element for the subsequent chemical separations.



Fig. 21 Low background proportional counters.

Table 3 Advantages and disadvantages of GAB for smears analysis.

<i>Advantages</i>	<i>Disadvantages</i>
Non-destructive (typically)	No isotopic speciation
Good screening tool	Fair to poor detection limits
Fast analysis	High uncertainties
Inexpensive	

Aqueous samples are often filtered and acidified prior to sample analysis. Again, one must know the data objectives prior to performing either. Acidification is performed to reduce the chance the analyte will absorb to the container walls. This is particularly a concern for the larger analytes such as plutonium and uranium.

Table 4 lists common sample preparation techniques for various matrices and **Fig. 22** shows a soil sample undergoing high-temperature fusion.

The following are important factors to consider as part of sample preparation:

- **Potential Sample Losses During Sample Preparation**

There are three main ways in which a portion of the sample or analyte may be lost during sample preparation. Loss mechanisms can be monitored and the sample results adjusted accordingly by the addition of carriers or tracers at the earliest point during the sample preparation. These added tracers and carriers allow one to calculate the amount of analyte loss.

First, drying samples in an oven or dry-ashing in a muffle furnace will often make the sample produce fine particulates or dust. These particles can easily be lost from air currents from temperature changes or from chemical hoods. Second, some radio-nuclides are volatile under certain conditions and care must be taken during preparation and analyses, such as drying samples

Table 4 Sample preparation by sample matrix.

<i>Sample type</i>	<i>Sample preparation</i>
Soil/sediment	Dry Homogenize Subsample Fusion Acid leach Microwave digestion
Vegetation	Dry (% moisture) Homogenize Subsample Fusion Acid leach Microwave digestion
Biota	Dry (% moisture) Homogenize Subsample Fusion Acid leach Microwave digestion
Air Filters	Fusion Acid leach Microwave digestion
Concrete/asphalt, construction materials	Acid leach
Water (ground, surface)	Acidify Filter

**Fig. 22** Fusion of soil sample.

below the analyte's boiling. In addition, reagents may be added to the sample to change the chemical form of the analyte to a less volatile form. Last, some elements may react and adsorb into the sample container or preparation container. Acidification of the glassware or liquid samples is sometimes required to prevent adsorption.

- Contamination Control

Because of the emphasis on sample data for decision making, cross contamination between samples must be avoided. That begins in the field when the sample is first collected. Clean sample implements are used to collect each sample, samples are individually containerized, and prevented from leaking. Once in the laboratory, sources of contamination could include the certified pure chemical reagents are used, glassware and equipment must be kept clean, and the potential for airborne contamination or transfer between samples eliminated. Simple techniques include working within hoods, covering samples, using disposable or thoroughly cleaning labware, and sample segregation. Laboratory blank analyses using all the reagents used for the samples can provide information regarding any cross contamination that may occur from the reagents or labware.

Alpha spectrometry

Several types of detectors, such as ionization chambers, proportional counters, solid-state silicon semiconductors, and scintillators (ZnS, plastic, liquid), can measure alpha particles. The most common type of detection system in a laboratory is the solid-state silicon semiconductors that have a high-voltage power supply, preamplifier, amplifier, pulse discriminator, scaler, and recording device. An ADC and MCA is also required for spectrometry systems. Since alpha particles do not travel far distances, even in air, the alpha detector and sample is placed under vacuum.

Alpha spectrometry is an excellent way to identify and quantify alpha-emitting nuclides. Common D&D encountered radionuclides such as americium-241, plutonium-238, plutonium-239/240, thorium-228, thorium-230, thorium-232, uranium-233/234, uranium-235/236, and uranium-238 as well as any radionuclide that emits alpha radiation may be detected on the system. Nearly any sample type may be analyzed by alpha spectrometry. The detection limits and spectral resolution are very good. However, the chemical separations needed for ideal counting conditions is time-consuming and requires experienced chemists. **Table 5** lists the advantages and disadvantages of alpha spectrometry. **Fig. 23** depicts a solid-state semiconductor alpha spectrometer system and **Fig. 24** shows a natural uranium spectrum from a soil sample.

Liquid scintillation counting

Liquid scintillation counting (LSC) is often the method of choice for measuring radionuclides that decay by low energy beta emission or for radionuclides that are pure beta emitters, i.e. no associated gamma ray. However, LSC is versatile enough that it can be used for higher energy beta and alpha emitters. LSC is an analytical option for all types of sample matrices including water (tritium), smears, and soils. Samples can be prepared with or without chemical separation and placed into a liquid scintillation “cocktail”.

Table 5 Advantages and disadvantages of alpha spectrometry.

<i>Advantages</i>	<i>Disadvantages</i>
Excellent resolution	Time consuming
Good stability (low energy drift)	Requires well trained staff
High efficiency	Requires good analyte separation
Good detection limits	Potential interferences if separation is not complete
Good uncertainties	



Fig. 23 Solid-state alpha spectrometry system.

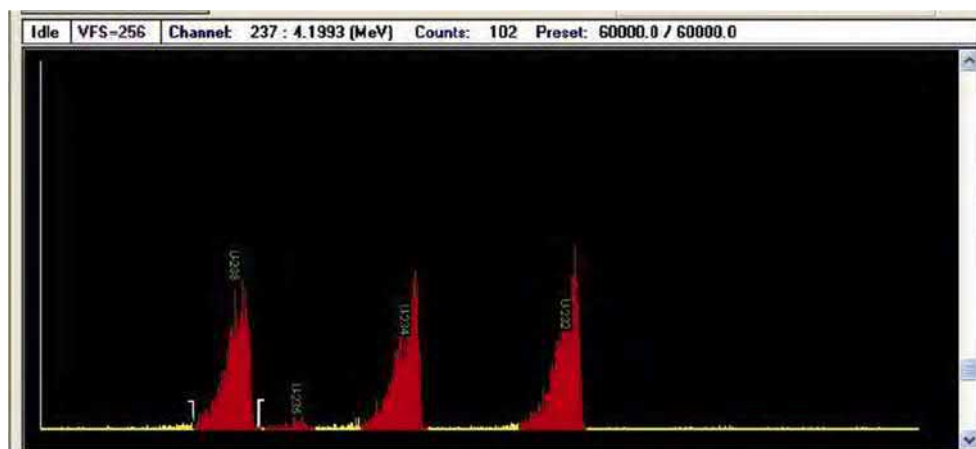


Fig. 24 Natural uranium spectrum in soil sample.

When the radioactive particle transfers its energy to the fluor in the cocktail, photons of light are produced. The amount of light produced is proportional to the activity in the sample.

LSC has very high measurement efficiencies because the sample is completely surrounded by the detection molecules of the cocktail. In fact, beta energies greater than 200 keV have essentially a 100% counting efficiency. However, this high efficiency is countered by the loss of resolution because beta emissions follow an energy continuum, two or more beta emitting radionuclides in a mixture generally cannot be distinguished without chemical separation. Therefore, LSC is often a screening method to quantify the “total activity” of a sample by examining the full 0–2000 keV energy range (this would include alpha and beta activity).

A picture of a typical liquid scintillation counter is depicted in [Fig. 25](#). Advantages and disadvantages of liquid scintillation counting are shown in [Table 6](#).

Low background proportional counting

As mentioned previously, low background proportional counting (LBPC) is a common method to count smears for gross alpha and beta activities. However, this method is also commonly used to measure the activities in medium to high energy beta emitters after



Fig. 25 Liquid scintillation counter.

Table 6 Advantages and disadvantages of liquid scintillation counting.

<i>Advantages</i>	<i>Disadvantages</i>
Excellent for low energy beta emitters	Cannot distinguish radionuclides in a mixture
Good for smear samples	Requires chemical separation for radionuclide-specific determination and quantification
Excellent efficiency	Potential interferences if separation is not complete
Low uncertainties and detection limits.	

Table 7 Advantages and disadvantages of low background proportional counting.

<i>Advantages</i>	<i>Disadvantages</i>
Good efficiencies for alpha and beta of medium to high energies	Time consuming
Good stability (low energy drift)	Requires well trained staff
Can count alpha and beta simultaneously	Requires chemical separation for radionuclide-specific determination and quantification
Low detection limits and uncertainties	Potential interferences if separation is not complete
Relatively simply instrumentation	

Table 8 Advantages and disadvantages of low background proportional counting.

<i>Advantages</i>	<i>Disadvantages</i>
Extremely sensitive	Relatively complex instrument
Good stability (low energy drift)	Requires well trained staff
Multiple analytes simultaneously	Requires good separation of analytes
Very low detection limits (parts per billion) and uncertainties	Potential interferences of isobars (e.g. U-232/Th-232)
Excellent	Potential matrix interferences
Relatively fast analysis after sample prep	

chemical separations. LBPC are typically calibrated such that alpha can be distinguished from beta radiation by controlling the applied voltage to the detector. The counting efficiency is good for both alpha and beta emission, but not as high as liquid scintillation. As with liquid scintillation, LBPC cannot distinguish energy differences between beta particles or between alpha particles. Thus, other than for GAB analysis, separation chemistry is required to isolate a single radionuclide prior to counting. Advantages and disadvantages of LBPC are listed in [Table 7](#).

Inductively coupled plasma mass spectrometry

Inductively Coupled Plasma Mass Spectrometry (ICPMS) is another destructive method used for the determination of long-lived radionuclides (those with half-lives > 10,000 years). Nearly any sample type after proper processing may be analyzed by ICPMS. Processed samples are introduced into the instrument and the sample is de-solvated, atomized and ionized in the plasma. The resulting ions are then separated by the mass spectrometer based on their mass to charge ratio. Thus, ICPMS does not measure the samples radioactivity, but it measures the mass of the sample. ICPMS is often used for the measurement of isotopic uranium, plutonium-239/240, and technetium-99. The method offers several advantages including very high sensitivities (low detection limits), multiple radionuclides can be measured nearly simultaneously, fast analysis once the sample is prepared, and can resolve elemental isotopes that alpha spectroscopy cannot distinguish (plutonium-239/240). However, the instrument cost is relatively high and requires a well-trained personnel to operate.

Advantages and disadvantages of ICPMS are listed in [Table 8](#) below.

See Also: Radiation Detectors.

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Emergency Response

Craig M. Marianno, Texas A&M University, College Station, TX, United States

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Introduction

Radiation protection in emergency response is challenging. In “normal” working conditions sources are well understood and doses are able to be kept well below regulatory limits. Conversely, in emergency response you may not immediately know your source term, amount or even the type of radiation involved. Additionally, there may be situations where responders or even the public may receive a recordable dose far higher than is normally permitted in an occupational setting. The goal in emergency response health physics is to make sure that immediate doses due to the situation will not result in non-stochastic effects (radiation sickness) and that any doses received will minimize the risk of long-term stochastic effects (cancer). These goals are applied to responders and public alike. The purpose of this chapter is to introduce emergency response radiation protection and how it is applied during a radiological event.

Protective actions

In order to minimize radiation doses during an event it is imperative that protective actions be taken. For the public, these actions would be established and implemented by the local, state or the federal government. Protective actions could include shelter-in-place, evacuation, administration of prophylactic drugs, relocation and food supply warnings/interdictions. All of these can cause major disruptions in the daily routines of the public and each must be carefully considered before implementation.

To help public officials during a radiological event the Environmental Protection Agency (EPA) created a document that provides protection action guides (PAGs) that are used to determine when protective actions are warranted. The *PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents* was revised in 2017 and is sometimes simply called EPA400 (EPA, 2017). This document defines a PAG as “the projected dose to an individual from a release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended.” In a radiological emergency, experts would take radiological data from the field to estimate future doses. Advisors to a public official (or decision maker) would then relate how these doses relate to PAGs. Based on the provided information and advice it is the decision maker that determines if (and what) protection actions should be implemented.

PAGs are not legally binding regulations and do not need to be strictly followed because other risks may exist. The key word in PAG is “guide,” because these values should only be considered when taking into account the unique environment of the radiological event and the risks involved when the PAG is executed. Implementing a PAG only reduces the health risk caused by the radiation, but there are other risks to be considered. For instance, during a radiological release it is projected that a large area may need to be evacuated. If everyone in the affected area simultaneously tries to evacuate, what are the risks of fatal accidents with so many cars suddenly on the road? If traffic jams are caused, is the public then being exposed to more radiation because they are stuck outdoors in the plume? Similarly, if a blanket order is given to evacuate what do you do for sensitive populations? Is the risk of death to hospitalized patients in critical condition greater because they are being moved or from the dose they may receive? Implementing a PAG is a complex judgment process because sometimes the radiation might not be the most critical factor to consider.

Phases of an emergency and how they relate to PAGs

Radiation protection considerations during an emergency are broken down into phases to aid in response planning. The beginning of the radiological incident is defined as the *Early Phase* and can last for hours to days. It is during this period that immediate decisions on protective actions could greatly reduce the doses due to inhalation or plume passage. Protective actions during the Early Phase would be based on predictions of radiological conditions or available measurements. They could even be based on the

Table 1 Emergency phases and their associated PAGs (EPA, 2017).

Phase	Protective action recommendation	PAG, Guideline, or planning guidance
Early Phase	Sheltering-in-place or evacuation of the public Supplementary administration of prophylactic drugs	PAG: 1 to 5 rem (10 to 50 mSv) projected dose over 4 days PAG: 5 rem (50 mSv) projected child thyroid dose from radioactive iodine
Intermediate Phase	Relocation of the public	PAG: ≥ 2 rem (20 mSv) projected dose in the first year; 0.5 rem (5 mSv) per year projected dose in the second or subsequent years
	Apply simple dose reduction techniques Food interdiction	Guideline: < 2 rem (20 mSv) projected dose in the first year PAG: 0.5 rem (5 mSv) per year projected whole body dose, or 5 rem (50 mSv) per year to any individual organ or tissue, whichever is limiting
	Drinking water	PAG: 0.1 rem (1 mSv) projected dose for 1 year, to the most sensitive populations (e.g. infants, children, pregnant women and nursing women); 0.5 rem (5 mSv) projected dose, for 1 year, to the general population
	Reentry	Guideline: Operational Guidelines (stay times and concentrations) for specific activities

potential of worsening conditions. In most cases the protective action during the Early Phase is evacuation or shelter-in-place and is triggered if the projected dose could reach 1–5 rem (10–50 mSv) over 4 days of exposure (Table 1). Sometimes default protective actions are established before an accident even happens to assure the public is protected as rapidly as possible. For instance, in some states for a nuclear power plant accident, evacuation and shelter-in-place orders would be initiated 360° within a 10-mile radius of the plant. If there is a potential of a radiological release due to facility conditions or other events, protective actions can be initiated before the event occurs.

The *Intermediate Phase* begins when the initiating event has been brought under control. Unlike the Early Phase, protective action decisions will primarily be based on environmental measurements that define the radiological environment. Actions taken during this phase could include population relocation, food embargos and implementing water restrictions. The different PAGs associated with this phase, listed in Table 1, are meant to protect that portion of the public who was not immediately affected by the release, but may receive an accumulated dose from deposited material. This phase could last from weeks to months and may overlap with the Early Phase.

The US Food and Drug Administration (FDA) has issued guidance based potential doses received from the ingestion of food (FDA, 1998). These are typically implemented during the Intermediate Phase. The FDA suggests *Intervention Levels* that are based on PAGs of 0.5 rem (5 mSv) Committed Effective Dose or 5 rem (50 mSv) to any specific organ. These Derived Intervention Levels (DILs) are established for 24 radionuclides and represent the concentration ($\mu\text{Ci kg}^{-1}$) of radioactive material in food that, in the absence of any intervention, will lead to an individual reaching a PAG (Table 2). These DILs are for food ingested during the first year after an accident and are based on the most sensitive population (usually an infant less than 3 months old) and target organ. This conservatism is meant to provide a large margin of safety to the exposed public. DILs for other radionuclides can be calculated and the resources to do so are listed later in this chapter.

The Late Phase focuses on clean-up and attempting to bring environmental doses back to pre-event levels. This phase can last months to years and ends only when all recovery actions have been completed. Since this phase focuses on long term clean-up, dose avoidance through the use of PAGs is not appropriate. Since this phase occurs after the actual emergency, it will not be discussed in this chapter.

Dose guidelines for emergency responders

EPA400 also provides guidance for emergency response personnel. In a radiological incident responders could be exposed to elevated levels of radioactivity. In most cases, when following proper precautions doses can be kept below the annual occupational dose limit of 5 rem (50 mSv). It can still take several hours or days to reach this limit even working in what would be considered a high radiation field in “normal” conditions. A responder may find themselves working in 100 mrem hr^{-1} (1 mSv hr^{-1}) field to save trapped victims. In a normal occupational setting this radiation field would be unacceptable, but in an emergency, it could be unavoidable. Even in this elevated dose rate area a single worker would have to continuously work 50 h to even reach the “normal” limit. There may be situations like responding to an improvised nuclear device (IND) or serve reactor accident, where radiation fields are extremely high, but operations must take place to save lives or make sure the situation does not worsen. Any dose received during a radiological emergency should still follow ALARA (As Low As Reasonably Achievable) concepts: the risk of the accumulated dose is justified by the benefit of the actions.

Table 2 List of DILs for the 24 specific nuclides established by the FDA (DOE, 2010).

Radionuclide Group	FDA DIL ^a (Bq/kg _{wet})	FDA DIL ^a (μCi/kg _{wet})
<i>Principal Nuclides</i>		
⁹⁰ Sr	160	4.3E-03
¹³¹ I	170	4.6E-03
¹³⁴ Cs + ¹³⁷ Cs	1200	3.2E-02
¹³⁴ Cs	930	2.5E-02
¹³⁷ Cs	1360	3.7E-02
²³⁸ Pu + ²³⁹ Pu + ²⁴¹ Am	2	5.4E-05
²³⁸ Pu	2.5	6.8E-05
²³⁹ Pu	2.2	5.9E-05
²⁴¹ Am	2	5.4E-05
¹⁰³ Ru + ¹⁰⁶ Ru	(¹⁰³ Ru/6800) + (¹⁰⁶ Ru/450) < 1	(¹⁰³ Ru/0.18) + (¹⁰⁶ Ru/1.2E-02) < 1
¹⁰³ Ru	6800	0.18
¹⁰⁶ Ru	450	1.2E-02
<i>Other Nuclides</i>		
⁸⁹ Sr	1400	3.8E-02
⁹¹ Y	1200	3.2E-02
⁹⁵ Zr	4000	0.11
⁹⁵ Nb	12,000	0.32
¹³² Te	4400	0.12
¹²⁹ I	56	1.5E-03
¹³³ I	7000	0.19
¹⁴⁰ Ba	6900	0.19
¹⁴¹ Ce	7200	0.19
¹⁴⁴ Ce	500	1.4E-02
²³⁷ Np	4	1.1E-04
²³⁹ Np	28,000	0.76
²⁴¹ Pu	120	3.2E-03
²⁴² Cm	19	5.1E-04
²⁴⁴ Cm	2	5.4E-05

^aA food sample is considered to exceed the DIL if it meets or exceeds the DIL for any individual nuclide. Analysis results are not summed across nuclides except the combinations specifically stated (i.e., ¹³⁴Cs + ¹³⁷Cs, ²³⁸Pu + ²³⁹Pu + ²⁴¹Am, and ¹⁰³Ru + ¹⁰⁶Ru).

Table 3 lists the suggested guidelines from EPA400. Like the public PAGs, these are only guides and not set limits. The National Council on Radiation Protection even suggests the concept of a “decision dose” at 50 rem (500 mSv) (NCRP, 2010). It is at this dose that an incident commander should decide to remove a responder from the field or let them remain to accomplish mission critical actions. All of the values listed in **Table 3**, and the decision dose, are below dose levels where radiation sickness occurs (**Table 4**) and at levels where the risk of the onset of cancer in later years is minimized. To make sure all responders know how to reduce their doses during operations they should be properly trained. At a minimum this training should include elements of the emergency response plan, operating procedures and procedures for notification and handling of emergency nuclear and radiological incidents (NCRP, 2005).

Table 3 Dose guidelines for emergency responders (EPA, 2017).

Guideline	Activity	Condition
5 rem (50 mSv)	All occupational exposures	All reasonably achievable actions have been taken to minimize dose
10 rem (100 mSv)	Protecting critical infrastructure necessary for public welfare (e.g. a power plant)	Exceeding 5 rem (50 mSv) in unavoidable and all appropriate actions have been taken to reduce dose Monitoring available to project or measure dose
25 rem (250 mSv) ^a	Lifesaving or protection of large populations	Exceeding 5 rem (50 mSv) in unavoidable and all appropriate actions have been taken to reduce dose Monitoring available to project or measure dose
> 25 rem (250 mSv)	Lifesaving or protection of large populations	Exceeding 5 rem (50 mSv) in unavoidable and all appropriate actions have been taken to reduce dose Monitoring available to project or measure dose

^aIn the case of a very large incident, such as an IND, incident commanders may need to consider raising the property and lifesaving emergency worker guidelines to prevent further loss of life and massive spread if destruction.

Table 4 Radiation levels where acute radiation sickness are witnessed (EPA, 2017).^a

Feature or Illness	Effects of whole body absorbed dose from external radiation or internal absorption, by dose range in rad (gray)				
	0–100 (0–1 Gy)	100–200 (1–2 Gy)	200–600 (2–6 Gy)	600–800 (6–8 Gy)	> 800 (> 8 Gy)
Nausea, Vomiting	None ^b	5–50%	50–100%	75–100%	90–100%
Time of Onset		3–6 h	2–4 h	1–2 h	< 1 h to minutes
Duration		< 24 h	< 24 h	< 48 h	< 48 h
Lymphocyte Count	Unaffected	Minimally Decreased	< 1000 at 24 h	< 500 at 24 h	Decreases within hours
Central Nervous System Function	No Impairment	No Impairment	Cognitive impairment for 6–20 h	Cognitive impairment for > 20 h	Rapid incapacitation
Mortality	None	Minimal	Low with aggressive therapy ^c	High	Very High: Significant neurological symptoms indicate lethal dose

^aPercentage of people receiving whole body doses within a few hours expected to experience acute health effects.

^bA small number of exposed individuals may experience symptoms such as nausea and vomiting at dose between 50 and 100 rad (0.5 and 1 Gy).

^cThe lethal dose with NO medical intervention to 50 percent of the population in 60 days is between 320 and 450 rad or 3.2–4.5 Gy.

From DoD (2003) *Medical Management of Radiological Casualties*, 2nd edn. Bethesda, MD: Armed Forces Radiobiology Research Institute. April 2003.

Tracking doses to first responders has been made easier with technology. Now many first responders carry alarming personal radiation dosimeters. These devices monitor both dose rate and accumulated dose. If preset levels are exceeded, the device would alarm warning the individual of the current radiation levels. These types of instruments only register gamma radiation because it is assumed that normal personal protective equipment (bunker gear, gloves, respirator, etc.) would protect the responder from other forms of radiation like alpha and beta particles. The UltraRadiac™-Plus is an example of such a device manufactured by Mirion Technologies. It is a rugged alarming radiation monitor specifically designed for first responders. It has a detection range between $1 \mu\text{rem hr}^{-1}$ and 200 rem hr^{-1} ($0.01 \mu\text{Sv hr}^{-1}$ – 2 Sv hr^{-1}) and an accumulated dose range of $0.1 \mu\text{rem}$ – 999 rem ($0.001 \mu\text{Sv}$ – 999 Sv) (Mirion Technologies, 2018). Devices such as these will warn responders well before any set limits for an event are surpassed.

Risk related to dose

Dose guidelines are based on the relative risk of the radiation dose. Risk levels have been established based on several studies including clinical studies, research on atomic bomb survivors and severe accidents involving radioactive material. Table 5 lists radiation effect risks as a function of absorbed dose. These risks are compared to the lifetime risk of fatal cancer without radiation exposure. Even without being exposed to radiation the lifetime risk of an individual dying from cancer is 24% (24 cases out of 100 people). Notice that even at the decision dose of 50 rem that the excess risk only increases by 4%. At the 10 rem guide (protection of property) the risk is less than 1%. In comparison, smoking can increase the risk of prostate cancer by 9–30% and risk of lung cancer by 15–30 times. The point is that while emergency dose guidelines may seem high compared to normal operations, the relative risk to the exposed individuals are still low.

There may be an age consideration when assigning duties to emergency responders. One point of limiting doses is to reduce the risk of cancer. Due to the latency period of cancer, younger responders face a larger risk of fatal cancer than their older counterparts.

Table 5 Values for the acute death, acute symptoms, and fatal cancer risk as a function of whole body dose equivalent. These values are compared to the lifetime risk of fatal cancer without radiation exposure (NCRP, 2005).

Short-term whole-body dose [rem (Sv)]	Acute death from radiation without medical treatment (%)	Acute death from radiation with medical treatment (%)	Acute symptoms (nausea and vomiting within 4 h) (%)	Lifetime risk of fatal cancer without radiation exposure (%)	Excess lifetime risk of fatal cancer due to short-term radiation exposure (%)
1 (0.01)	0	0	0	24	0.08
10 (0.1)	0	0	0	24	0.8
50 (0.5)	0	0	0	24	4
100 (1)	<5	0	5–30	24	8
150 (1.5)	<5	<5	40	24	12
200 (2)	5	<5	60	24	16
300 (3)	30–50	15–30	75	24	24
600 (6)	95–100	50	100	24	>40
1000 (10)	100	>90	100	24	>50

For individuals between 20 and 30 years old, they have a 0.9% risk of premature death when exposed to 25 rem (250 mSv), while 40–50 years olds have a 0.5% risk of premature death.¹ Therefore, using older responders in higher radiation fields could be justified based on the increased risk of the younger responders.

Estimating dose

Estimating the potential dose to the public and responders can be a complicated process. During a response these calculations are made using techniques described in the Federal Radiological Monitoring and Assessment Center (FRMAC) Assessment Manual (DOE, 2018). This three volume, multi-agency document was created from the input of 9 Federal agencies, reviewed by experts across the health physics community and forms the basis of how United States' response teams would calculate doses during an emergency. It utilizes the International Commission on Radiological Protection (ICRP) 60 methodology to estimate doses from 44 different radionuclides that would be common across different radiological emergencies. It also provides techniques for experts to determine doses from other nuclides not listed in the document. It is separated into four different sections:

- Public Protection
- Emergency Worker Protection
- Ingestion Pathway Analysis
- Supplemental Methods

To anticipate the dose assessment needs that would occur for different radiological emergency events, Volume 2 of the Assessment Manual contains default scenarios. This volume is meant to expedite response for specific events before field data becomes available and uses "expected" nuclides and dispersion methods. These pre-assessed events include:

- Nuclear power plant accident
- Nuclear weapon accident
- Aged fission product accident
- Nuclear fuel accident
- Radionuclide thermoelectric generator (RTG) accident
- Radiological dispersal device (RDD)
- A domestic nuclear explosion is under development

Each default scenario includes an event description, possible radionuclides, potential consequences, potential actions, required measurements for prediction tools and default dose rate and dose values known as Derived Response Levels (DRLs) that help predict areas where PAGs should be implemented.

In addition to the Assessment Manual, dose assessment professionals can use software to estimate exposure to the public and responders. In fact, the dose Assessment Manual serves as the scientific basis for the software package called TURBO-FRMAC®. Used by trained health physicist this software automates the calculations performed in the manual. It can rapidly calculate DRLs and estimate doses based on field data collected by response teams. While it does initially use default values based on generic scenarios, assessment scientists can modify inputs to accommodate incident-specific conditions.

How a response progresses

A radiological incident can take many forms. It can be as simple as a vehicle accident involving only local responders and a small localized spill to an IND affecting thousands of square miles. In many cases local responders can handle the situation, but there are cases where local responders and officials will be overwhelmed and need assistance from state and federal authorities. At each stage, the response can end and not require the full assistance of the federal government. It should be noted that no matter the size of the incident there are support mechanism within the federal government that will come to the aid of local and tribal authorities to assist in the response. This aid can take the form of advice over the phone to sending experts to the affected area. Below is a description of how a response can progress. It will include several teams that could respond and their functions.

The timeline described below is primarily from the perspective of the Department of Energy's (DOE) National Nuclear Security Administration (NNSA). Under the National Response Framework (NRF) Nuclear/Radiological Incident Annex (NRIA) the DOE is given the responsibility to coordinate federal radiological monitoring and assessment activities during many radiological emergencies (DHS, 2016). Other response functions and teams will most likely respond as well. Response teams can also vary based on the type of incident. If it is a deliberate event the Federal Bureau of Investigation (FBI) would coordinate the response and utilize some

¹The numerical estimate of cancer risk presented above (from Federal Guidance Report #13) was obtained by linear extrapolation using the nominal risk estimates based on data from human exposures at high doses and high dose rates. Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of cancer deaths. Studies of human populations exposed at low doses are inadequate to demonstrate the actual magnitude of risk at low doses (about 0.1 Sv or 10 rem and below). There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiological observation and the possibility of no risk cannot be excluded (EPA, 1999).

of its response assets. If it is an accident, other federal agencies may coordinate the response. However, it is still the DOE that would organize the radiological assessment.

Below is a generic timeline of a major radiological emergency (Fig. 1). It is only meant to show what assets will arrive to support the situation in a general time period. Depending on the location of the event, response teams may be closer and arrive much faster. In many cases the federal response will only be needed through home team support or with limited on-scene response. The federal response would be tailored to the situation and the requests from the local and state governments.

0–2 h: Initial stages of the incident where little is known. The only information that might be known is that detection equipment is showing the presence of radiation.

During the first few hours of a radiological emergency local authorities will have to coordinate the response. They will have to rely on their own first responders to deal with the situation in the field. It is imperative that the incident commander and/or local officials execute protective actions as soon as possible to minimize doses to the public. The local government can request assistance for the federal government. During the very early stages of the response this assistance would take the form of advice and predictive modeling so the local officials and responders can better address the situation.

- *Local Response:* Local first responders would take charge of the scene and attempt to bring the situation under control and deal with any victims. Depending on the complexity of the response, local fire and rescue may rely heavily on their own Hazardous Material (HAZMAT) units. Immediate decisions on public protection will be from the on-scene Incident Commander. Once established, the local Emergency Operations Center (EOC) and senior public official will evaluate the situation and decide if further protective actions are necessary

Federal Response

- *Consequence Management Home Team:* Experts that are available through video/teleconference or bridge lines to provide aid to local responders or officials. This aid can take the form of scientific support, data management, event visualization through Geographic Information Systems (GIS) and logistics. This asset would be available throughout the incident to help coordinate the response and aid field assets
- *National Atmospheric Release Advisory Center (NARAC):* This asset utilizes a complex atmospheric dispersion model to predict the release of the material. In the early stages it may only use a “smoke” model if the source term is unknown. In this way responders and local official will be able to see the direction and possible extent of any releases. As more is known about the incident and environmental measurements are accumulated, NARAC will refine its models to give a better representation of the situation. Working in conjunction with the CMHT, maps can be produced and provided to local officials allowing them a better idea of what protective actions should be in place and how to respond to the event (Fig. 2). This asset is used throughout the incident.
- *Remote Advisory Team (Remote A-Team):* Representatives of federal agencies available through the CMHT that may provide additional advice on the situation. Members of the A-Team includes representatives from the EPA, Department of Agriculture (USDA), Food and Drug Administration (FDA), the Center for Disease Control (CDC) and other federal agencies as needed.

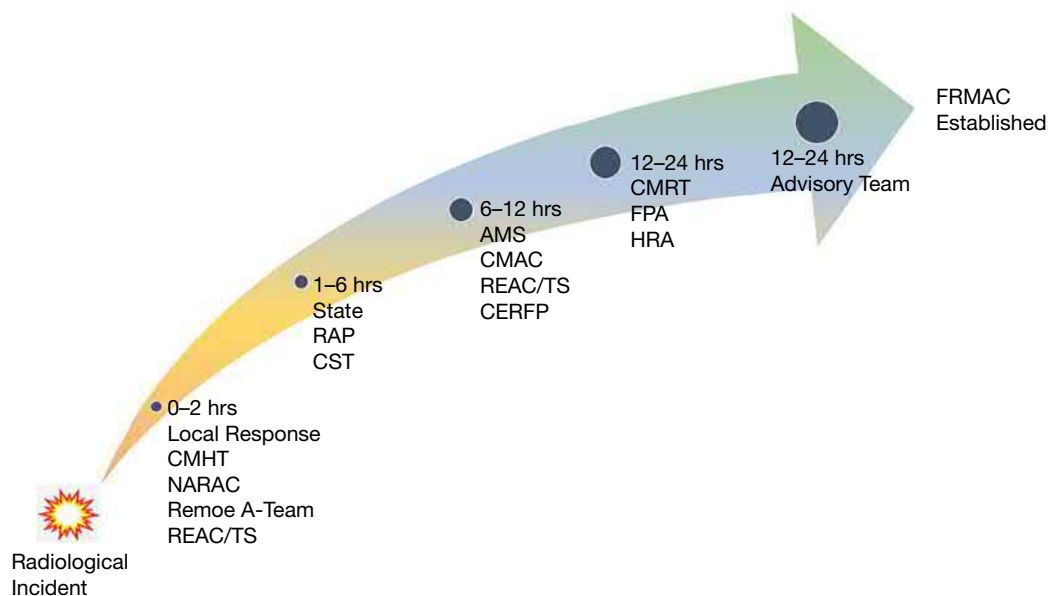


Fig. 1 Visual representations of the time of a radiological emergency response. Department of Energy (2019) Federal Radiological Monitoring and Assessment Center (FRMAC): Federal Outreach, DOE/NV/03624-0615. Las Vegas, NV: Department of Energy/NV.

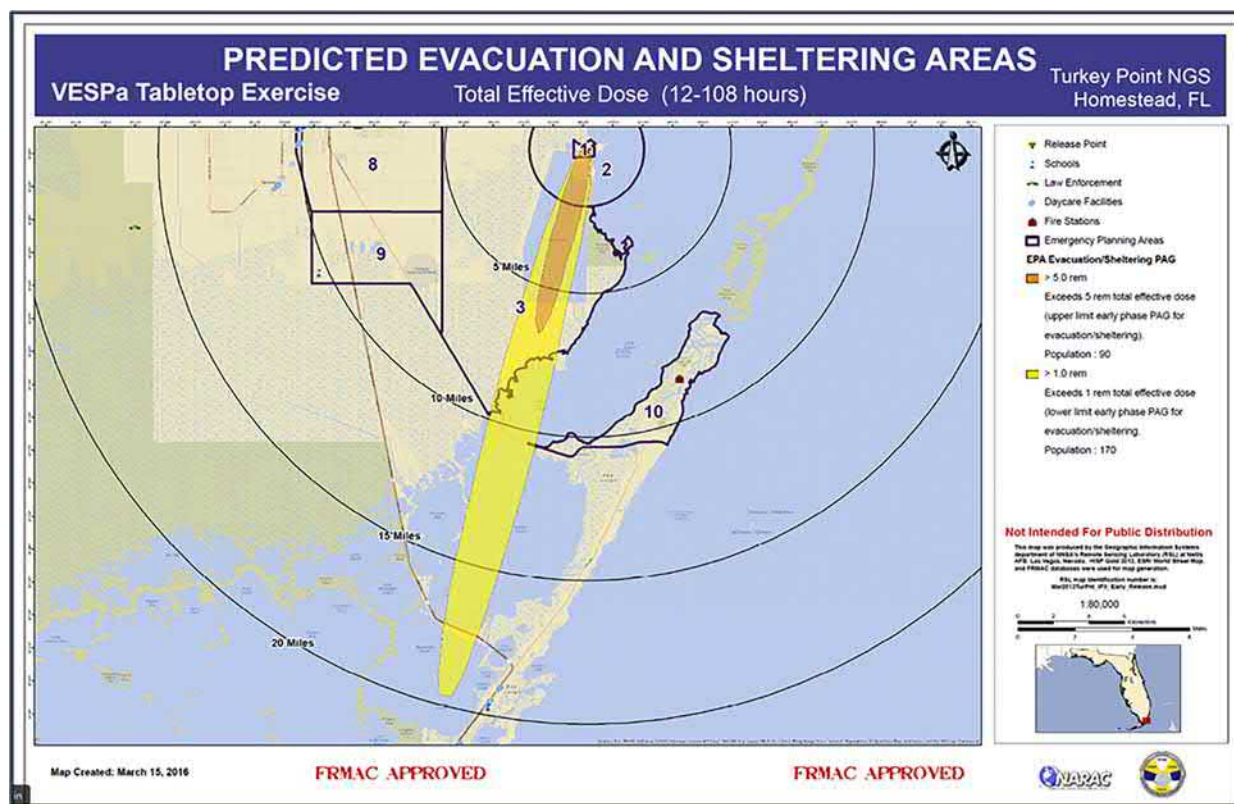


Fig. 2 Example map product produced by the CMHT with modeling data from NARAC. This map would be produced in the early phase of the incident and give decision makers an idea of areas where evacuation or sheltering may be warranted based on Federal guidelines. Department of Energy (2019) Federal Radiological Monitoring and Assessment Center (FRMAC): Federal Outreach, DOE/NV/03624-0615. Las Vegas, NV: Department of Energy/NV.

- **Radiation Emergency Assistance Center/Training Site (REAC/TS):** Experts in the areas of medical treatment and evaluation from situations involving radiation exposure. In the early stages of the incident they would be available for consultation with medical professional, first responders and local officials. They would be available throughout the response primarily in the form of remote support. However, they do have a small deployable team that could respond.

1–6 h: Still early stages of the incident. Little may be known about the source term or extent of the emergency.

- **State:** The State may have resources such as experts or monitoring teams that can help local responders and officials.
- **Civil Support Team (CST):** These are Department of Defense (DOD) assets that are associated with the state's National Guard and are trained in Chemical, Biological, Radiological/Nuclear (CBRN) response. All states have at least one CST. These teams can deploy and provide assistance to local responders and medical personnel. Because they are a regional asset, they would be one of the first teams to arrive. The CST is a governor controlled state capability.
- **Radiological Assistance Program (RAP):** This is a regional team sponsored by the DOE and deploys from various DOE sites throughout the United States. It is a regional asset and can be on the scene early in the radiological emergency. A RAP team can deploy with up to 8 members to conduct field monitoring and sampling measurements in addition to providing radiological advice to protect the health and safety of responders and the public.

Support for most events typically ends here. The event is localized and only minimal advice and support is needed. However, if the event is larger or more complex, additional federal assets will be on the way. The initial teams are working with incident command. They are taking radiological data with specialized instruments and transmitting that data back to the CMHT. While the source term might not be completely known, most likely at least some of the deposited isotopes have been identified through field-gamma spectroscopy. The presence of alpha and beta radiation would also be known, but the identification of those isotope may not be possible without laboratory analysis. This type of analysis could take days to weeks. All data being collected in the field is ultimately used by NARAC to continually refine dispersion models so that the release can be better defined (Figs. 3 and 4).

6–12 h: The initial event may or may not be under control. Initial PAGs are in-place.

- **Aerial Measuring System (AMS):** This DOE asset is comprised of fixed-wing aircraft and helicopters. They are stationed in two locations in the United States: Las Vegas, NV and Washington, DC. Depending on their proximity to the event it could take

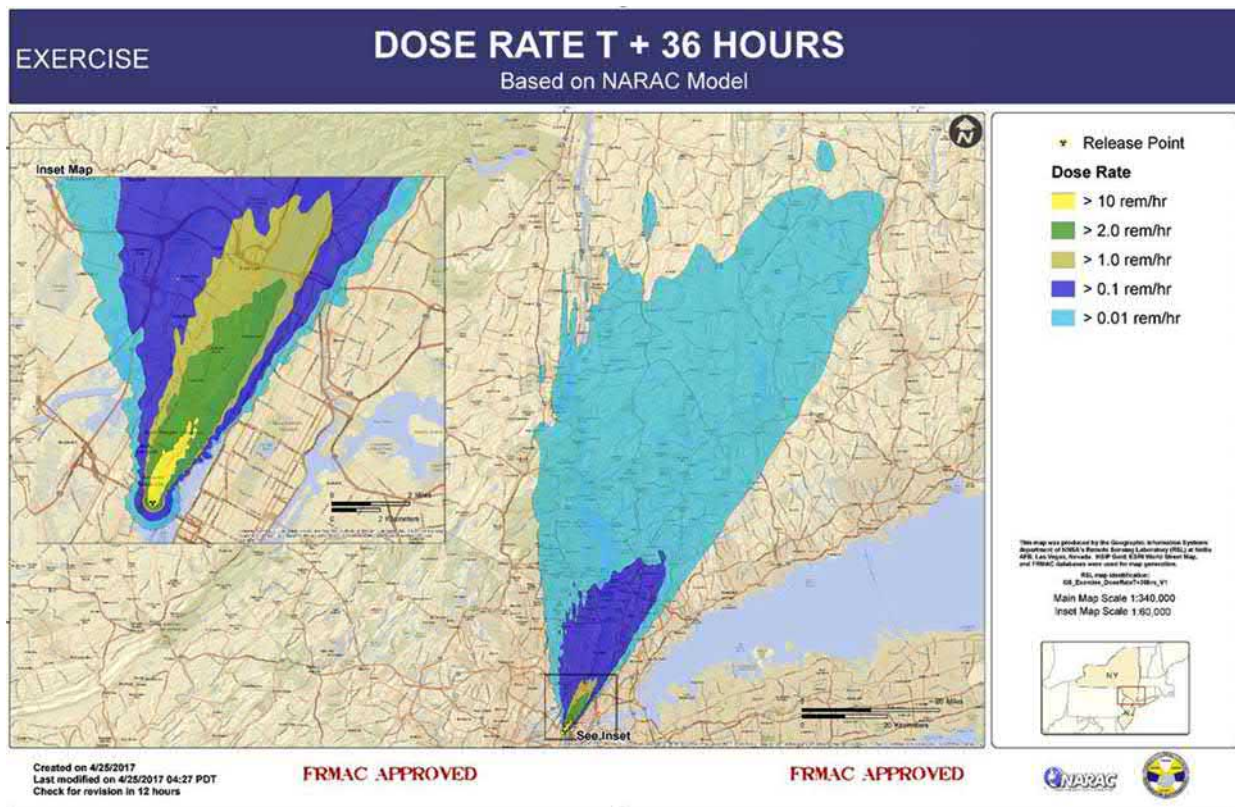


Fig. 3 NARAC can you use field measurements and data from other sources to refine dispersion models throughout an event. This map shows dose rate estimates 36 hours post-event and is a result of NARAC using more available data to better refine their prediction. Department of Energy (2019) Federal Radiological Monitoring and Assessment Center (FRMAC): Federal Outreach, DOE/NV/03624-0615. Las Vegas, NV: Department of Energy/NV.

several hours for them to arrive. The fixed-wing aircraft will arrive first because it is faster. AMS employs specialized airborne radiation detection systems and provide real-time, geo-located measurements of air and ground contamination. The advantage of aerial systems is that they can rapidly cover large areas and are not constrained by roadways. The fixed-wing has limited detection capability in that it must fly higher and faster than the helicopter and therefore gives a less defined visual indication of the ground contamination. The helicopter because it can fly lower and slower performs detailed surveys and can detect lower levels of radioactivity. **Fig. 4** is an AMS map produced during the Fukushima Daiichi disaster in 2011. The surveyed area covers urban and isolated areas out almost 50 miles from the plant. A color legend allows the viewer to know the exposure rate on the surface.

- **Consequence Management Advance Command (CMAC):** This is an advance team that helps facilitate the integration of a large team that will soon arrive called the Consequence Management Response Team (CMRT). The CMAC deploys directly to the local command centers established to manage the incident. The CMAC is comprised of the primary CMRT management team and tasked to establish the consequence management planning element in the site Incident Command Post (ICP) or designated incident management node where response planning is taking place.
- **REAC/TS:** If the field element of REAC/TS has been requested it is expected it would arrive during this period.
- **CBRN Enhanced Response Force Packages (CERFPs):** If the incident has caused mass damage and requires specialized victim recovery/rescues the CERFPs may be activated by the state's governor. Like the CST this team is part of a state's National Guard. CERFPs employ search and extraction, casualty decontamination, Fatality Search and Recovery Team (FSRT), and emergency medical triage and treatment capabilities to maximize the lifesaving response to a WMD attack or other CBRN incident.

12–24 h: The initiating event may still not be under control. By this time the gamma emitting nuclides should be known. AMS data may be available. If the event has stabilized the extent of the contamination may be known. Data has continually been fed back to NARAC and more refined prediction maps have been generated. Few monitoring teams are on the ground and a full characterization of the deposition has not occurred.

- **CMRT** – A large response team of 50 to 100 scientists, technicians, health/safety professionals and logistics support staff. They will come equipped with several pallets of equipment to support the response. The assembled team and equipment will include:
 - Assessment scientists to support data evaluation and visualization and support public and responder safety calculations

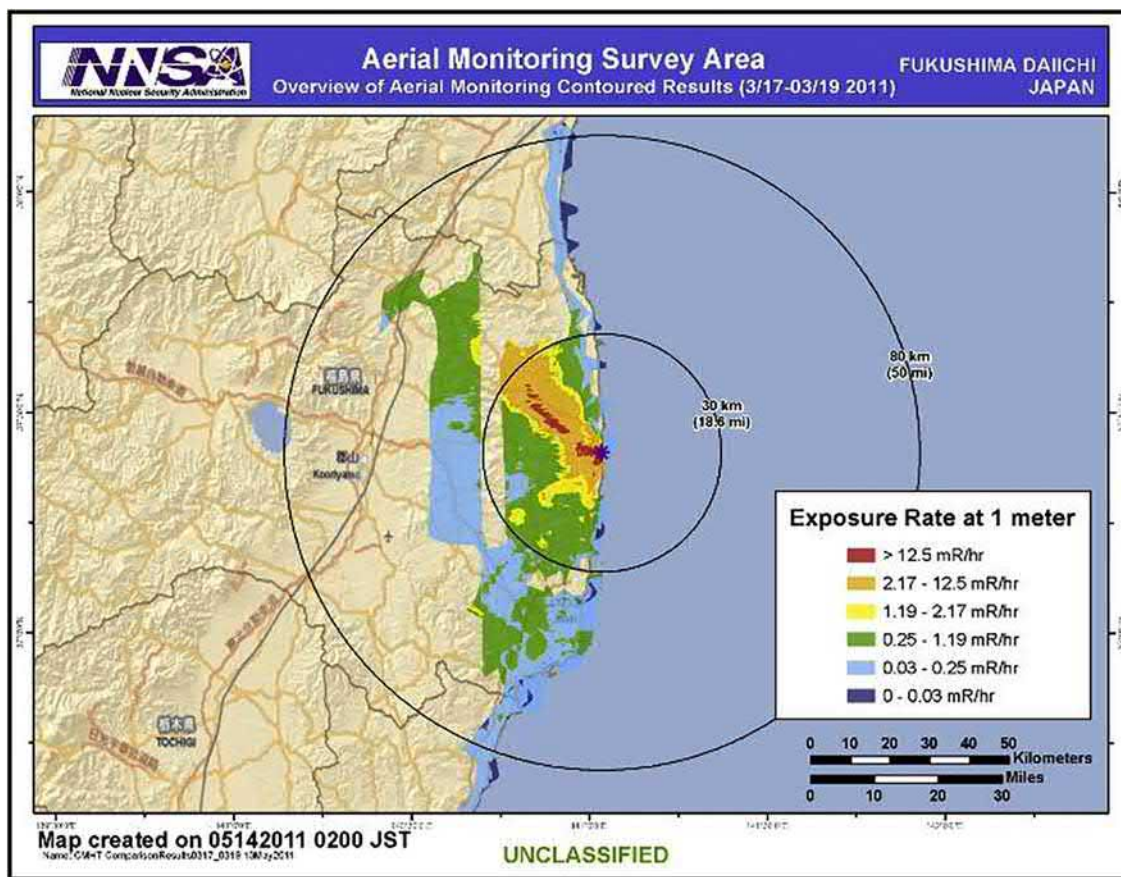


Fig. 4 Aerial measurement system data from the response to the Fukushima Daiichi nuclear power plant accident. Department of Energy (2019) Federal Radiological Monitoring and Assessment Center (FRMAC): Federal Outreach, DOE/NV/03624-0615. Las Vegas, NV: Department of Energy/NV.

- o Geographical Information Systems (GIS) equipment and personnel for creating map products to visualize data and potential extents
 - o Health and Safety specialists to support responders
 - o Field teams to assist with data and sample collection
 - o Contamination control to support field team activities
 - o Sample control personnel to catalog and manage sample collection and analysis
 - o Logistics support for FRMAC teams
- *Fly Away Laboratory (FAL)*: This is a mobile laboratory set-up that can be established in the field. It has limited sample analysis capability, but can conduct gamma spectroscopy and gross alpha/beta counting. It can have no radiochemistry capabilities, but will conduct limited sample screening and prioritization for fixed laboratories that are supporting the event.

24–48 h: Characterization of the deposited material is still taking place. Other agency response teams have arrived and are integrating into a unified response.

- *Federal Radiological Monitoring and Assessment Center (FRMAC)*: Sometime after 24 h and if multiple federal agencies respond to the incident a FRMAC will be formed. This allows the US government to have unified monitoring and assessment activities that efficiently work together to meet the objectives of the response set by incident command and the state. The FRMAC is organized by the DOE regardless of the agency that is managing the overall response. All federal teams that have responded or will respond to the incident will fold under the FRMAC so that the radiological assessment will be an organized endeavor.
- *A-Team*: Members of the A-team will be on site to directly support response to advise incident command and decision makers.

By this time the incident is potentially fully manned. Addition staffing and equipment may still be necessary, but the infrastructure is in-place to request and integrate any needed materials, equipment and personnel. Portions of Table 6 below are pulled directly from the NRIA and list other agencies and teams that may join the response as the emergency progresses.

Table 6 Additional assets, resources and teams that may join a radiological emergency response.

<i>Organization</i>	<i>Resource name</i>	<i>Description</i>
DHS (FEMA)	Radiological Operations Support Specialist Concept	Provides technical radiological/nuclear support to Incident Command at the state Emergency Operations Center level
DHS (FEMA)	RadResponder Network	A free Cloud-based radiological/nuclear data collection; management; and analysis tool for local, state, federal, tribal, and territorial governments. Assists with radiological/nuclear incident characterization and situational assessment. Provides a common framework to rapidly and accurately collect, aggregate and share radiological monitoring and sampling data; manage specialized equipment and personnel; and track radiological response teams during an emergency. The RadResponder Network can be accessed on computers and smart phone/tablet devices
DOE (NSA)	Nuclear Weapons Accident Response Group	Provides technical guidance and responds to U.S. nuclear weapons accidents. The team assists in assessing weapons damage and risk, and in developing and implementing procedures for safe weapon recovery, packaging, transportation, and disposal
DOJ (FBI)	Evidence Response Team Unit (ERTU): Hazardous Evidence Response Team (HERT)	Provides training, leadership, and subject matter expertise in hazardous evidence collection, as well as in the management and processing of forensic evidence in CBRN crime scenes. ERTU also provides coordination and oversight for operational response and activities of FBI field office HERTs
EPA	Airborne Spectral Photometric Environmental Collection Technology (ASPECT)	The ASPECT aircraft is managed by EPA's CBRN Consequence Management Advisory Team and provides remotely sensed chemical/radiological (gamma and neutron) data and imagery (situational awareness). It can identify, quantify, and map chemical plumes and ground-based radiation. It is also capable of collecting high-resolution digital photography and video products. Data products are transferred to ground base support within minutes of collection through satellite communications, while in flight
EPA	CBRN Consequence Management Advisory Team (CMAT)	Provides scientific and technical support for all phases of environmental response to a CBRN incident, including health and safety site characterization, environmental sampling and analysis, environmental monitoring, risk assessment building and structure decontamination, waste treatment environmental cleanup, and clearance; manages the EPA's ASPECT fixed-wing aircraft, which provides chemical/radiological data; deploys and operates ground-based characterization and mapping capability for radiological incidents (CMAT Asphalt)
EPA	Environmental Response Team	Provides scientific and technical expertise, including health and safety, environmental sampling, air monitoring, toxicology, risk assessment, waste treatment, contaminated water/scientific divers; and site decontamination and remediation; provides field-analytical and real-time air monitoring with the EPA mobile laboratories known as Trace Atmospheric Gas Analyzers
EPA	Radiation Task Force Leaders (RTFLs)	A sampling and monitoring force multiplier comprised of EPA Response Support Corps members based throughout EPA's Regions and Labs. The RTFLs are specially trained EPA personnel who will lead small teams of personnel in performance of tasks including field radiological measurements, contamination monitoring, soil sampling, air sampling, decontamination line setup and support, radiological control area support, and dose management support

(Continued)

Table 6 Additional assets, resources and teams that may join a radiological emergency response.—cont'd

<i>Organization</i>	<i>Resource name</i>	<i>Description</i>
EPA	RadNet	Monitors the nation's air, precipitation, and drinking water to track radiation in the environment. RadNet sample testing and monitoring results show the fluctuations in normal background levels of environmental radiation. The RadNet system will also detect higher than normal radiation levels during a radiological incident. During a radiological incident, officials use RadNet data to help make science-based decisions about protecting the public. Scientists use RadNet air monitoring data to help estimate the potential radiation dose to people
EPA	Radiological Emergency Response Team	Provides advice on protective measures to ensure public health and safety; assessments of dose and impact of release to public health and the environment; monitoring, sampling, laboratory analyses, and data assessments to assess and characterize environmental impact; and technical advice and assistance for containment, cleanup, restoration, and recovery
EPA	Radiological Environmental Assessment Equipment	Sample preparation trailers and mobile laboratories carry electrical generators and supplies for approximately 1 week; not applicable for other assets
HHS (CDC)	Strategic National Stockpile Agents for Nuclear/Radiological Incidents	National repository of antibiotics, chemical antidotes, antitoxins, life support medications, IV administration, airway maintenance supplies, and medical/surgical items. Nuclear/radiological-specific resources include chelating agents (Calcium and Zinc Diethylenetriamine pentaacetate), Prussian Blue, and Growth Factors/Cytokines for White Blood Cells
U.S. Department of Veteran's Affairs	Medical Emergency Radiological Response Team (MERRT)	The MERRT responds to radiological disasters that require medical assistance and/or radiological decontamination of victims. The MERRT provides medical assistance including direct patient treatment, assisting and training local health care providers in managing, handling, and treatment of radiation-exposed and contaminated casualties; assesses the impact on human health; and provides consultation and technical advice to local, state, and federal authorities

This information was taken directly from the NRIA (DHS, 2016).

Conclusion

Radiation protection in an emergency response is a complex undertaking. A radiation release can be limited to a localized event or result from a major release affecting thousands of square miles and millions of people. In all cases the goal of the response is the same for radiation protection: limit doses to the public and responders. Emergencies are broken into phases and each phase has specific goals in terms of dose control. In the early phase protective actions, based on EPA suggested guidelines, are implemented to primarily avoid doses that will cause short term effects such as radiation sickness. This is accomplished through sheltering-in-place or evacuation. After the early phase, protective actions are used to avoid doses that can lead to long-term effects like cancer. It is important to remember that when protective actions are limited you are only reducing the risk of radiation affects. Other risks that result from the actions may be greater and should be carefully considered before implementing any actions.

The federal government has issued tools and documentation to aid in the implementation of protective actions. One of the most useful documents that associates doses with protective actions is EPA400. The values listed in the document are only meant as guides to allow flexibility of implementing protective actions when considering the complexity and uniqueness of the specific radiological event. To assess and predict doses Turbo FRMAC was created so the experts can interpret radiological data and associate them to EPA 400 guidance. This chapter only presented a limit number of available resources and further research by the reader may yield other guidance and tools for areas of interest.

One of the most important aspects of radiation protection in emergency response is that you are not alone in the event. There are several state and federal resources that are only a phone call away and free to use. Most situations can be handled by local first responders and HAZMAT teams. However, situations may arise when advice or additional field support is needed. This aid can come in the form of advice over the phone to hundreds of technical experts responding on-site. This chapter listed a majority of these teams, but other teams might exist that would be available specific emergencies.

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Standards to Control Radiation Exposures of Workers and the Public

David C. Kocher, Oak Ridge Center for Risk Analysis, Inc., Oak Ridge, TN, United States

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Introduction

This chapter presents an overview of standards to protect workers and the public from the harmful effects of ionizing radiation. The following topics are discussed:

- Health effects of concern in radiation protection;
- Basic principles of radiation protection;
- Radiation protection standards recommended by the International Commission on Radiological Protection (ICRP) and by the National Council on Radiation Protection and Measurements (NCRP) in the United States (U.S.)¹;
- Current standards established by regulatory authorities in the U.S.

Discussions of ICRP and NCRP recommendations summarize their development in several iterations over the last six decades. Although each set of new recommendations superseded those developed previously, an historical perspective is important when current standards in the U.S. incorporate some elements of all recommendations from the late 1950s to the present.

Health effects from radiation exposure

Two types of health effects, called stochastic effects and tissue reactions, are of concern in radiation protection.

Stochastic effects are effects for which the probability of occurrence, but not their severity, is a function of dose, without threshold. These effects are random in nature and can result from misrepair of damage in a single cell nucleus. The most important stochastic effects from radiation exposure are solid cancers and leukemias. Heritable effects in descendants of exposed individuals also are regarded as stochastic.

Tissue reactions, also called nonstochastic or deterministic effects, are effects for which the severity varies with dose and a threshold usually exists. These effects occur when many cells in an organ or tissue are damaged, resulting in a loss of tissue

¹ICRP and NCRP are nongovernmental organizations of radiation protection professionals whose recommendations have been incorporated in laws and regulations in many countries.

function. Examples of tissue reactions from radiation exposure include cataract of the lens of the eye, reddening of the skin, depletion of white blood cells from irradiation of bone marrow, and impairment of fertility from irradiation of the gonads (testes and ovaries).

To protect against the two types of health effects, radiation protection standards have two basic objectives:

- [1] Limitation of the risk (probability) of stochastic effects;
- [2] Prevention of tissue reactions.

Principles of radiation protection

The system of radiation protection developed by ICRP and NCRP is based on three principles:

- [1] Justification of exposures;
- [2] Optimization of protection;
- [3] Limitation of dose to individuals.

The principles of justification and optimization apply to all situations in which radiation exposures can be controlled in some manner. The principle of dose limitation applies to planned (routine) exposures only, that is, to situations involving the deliberate introduction and operation of radiation sources.

Current statements of the basic principles (ICRP, 2007) are summarized as follows:

- **Justification:** Any decision that alters the radiation exposure situation—that is, any decision to introduce a new radiation source, reduce an existing exposure, or reduce the risk of potential exposure—should do more good than harm (positive net benefit to an individual or society).
This principle is important, for example, in ensuring that the benefit to patients from uses of radiation in medicine outweighs the associated health risks. It also is the basis for bans on frivolous uses of radioactive materials in jewelry and children's toys.
- **Optimization:** The likelihood of incurring exposures, the number of people exposed, and the magnitude of their individual doses should all be kept as low as reasonably achievable (ALARA), taking into account economic and societal factors.
When exposures are optimized, doses to individuals can vary greatly depending on the situation (e.g., doses to the public from routine emissions from nuclear power plants vs radon in homes).
- **Dose limitation:** The total dose to any individual in planned exposure situations other than medical exposure of patients should not exceed appropriate limits.
Dose limits are intended to ensure that doses and associated health risks to the most highly exposed individuals in planned exposure situations do not exceed levels regarded as acceptable.

ICRP and NCRP recommendations on radiation protection standards

This section discusses recommendations on radiation protection standards for workers and the public developed by ICRP and NCRP over the last six decades. Elements of recommendations from the late 1950s and their subsequent revisions are incorporated in many current standards established by regulatory authorities in the U.S. Recommendations prior to that time are discussed elsewhere (Taylor, 1971, 1979; Kocher, 1991).

ICRP publications 1 and 2; NCRP report nos. 22 and 39

Standards for workers

ICRP Publications 1 and 2 (ICRP, 1959, 1960) and NCRP Report No. 22 (NCRP, 1959) presented recommendations on maximum permissible dose to workers, which were intended to limit doses to the whole body or critical organ (usually the organ or tissue receiving the highest dose) from external exposure and from internal exposure due to inhalation and ingestion of radionuclides combined.²

Recommended maximum permissible doses to the whole body, gonads, blood-forming organs, and lens of the eye in workers included:

²In previous recommendations, ICRP and NCRP developed limits on external and internal exposure of workers separately. The recommendation that dose limits for the whole body or critical organ should apply to doses from external and internal exposure combined was not incorporated in radiation protection standards for workers and the public established by the U.S. Atomic Energy Commission in 1960 and was not adopted in the U.S. until standards to limit exposures of the public to radionuclides released in operations of uranium fuel-cycle facilities were established in 1977 (see Table 3).

- a cumulative (total) dose equivalent^{3,4} at age N of 50(N–18) mSv⁵;
- to the extent permitted by the limit on cumulative dose, a dose equivalent in any 13-week period (quarter) of 30 mSv.

The limit on cumulative dose implied a limit on average annual dose equivalent of 50 mSv. The greater importance of the limit on cumulative dose was based on the view that total dose was more important than dose rate in determining risks from radiation exposure, provided that dose rates were sufficiently low to prevent tissue reactions and only small exceedances of the implied limit on average annual dose occurred in any year (Taylor, 1971, 1979).

Maximum permissible doses given above were substantially lower than previous ICRP and NCRP recommendations (Kocher, 1991). Reductions in dose limits for workers were based on two considerations: (1) the assumed genetic hazard (heritable effects) from irradiation of the gonads and the large increase in the worker population in the nuclear weapons industry; and (2) a desire to reduce possible increases in leukemias, which were first observed in radiologists in the mid-1920s (Taylor, 1971, 1979).

ICRP and NCRP also recommended higher maximum permissible doses to other organs and tissues, which, along with the dose limits for the lens of the eye, were mainly intended to prevent tissue reactions.

In Report No. 39 (NCRP, 1971), NCRP retained the limit on cumulative dose equivalent of 50(N–18) mSv, but the limit of 30 mSv in any quarter was replaced by a limit on annual dose equivalent of 50 mSv. NCRP also recommended (1) higher limits on annual dose equivalent to the skin, forearms, hands, and other organs or tissues to prevent tissue reactions and (2) a limit on dose equivalent during the entire gestation period of pregnant women of 5 mSv to protect the embryo/fetus.

Standards for the public

Recommendations on radiation protection standards for the public were developed beginning in the mid-1950s, when ICRP recommended that dose limits for the public, excluding natural background and medical exposures, should be one-tenth of limits for workers (ICRP, 1955). This recommendation was based on the consideration that, unlike workers, the public receives no direct benefit associated with their exposures (Taylor, 1979).

Based on dose limits for workers in Publications 1 and 2 (ICRP, 1959, 1960), ICRP's recommended limit on annual dose equivalent from whole-body exposure of the public was 5 mSv. ICRP also recommended a limit on average dose equivalent to the gonads in large population groups—the so-called genetically significant dose to the population—of 50 mSv in 30 years. This recommendation, which addressed the concern about heritable effects from radiation exposure, implied a limit on average annual dose equivalent to the gonads of 1.7 mSv.

NCRP also recommended a limit on annual dose equivalent from whole-body exposure of the public of 5 mSv (NCRP, 1959). This dose limit was retained in Report No. 39 (NCRP, 1971), which also included a limit on average annual dose equivalent to the whole body and gonads in the population of the U.S. of 1.7 mSv.

ICRP publication 26; NCRP report no. 91

Recommendations in ICRP Publication 26 (ICRP, 1977) included two important developments. First, limits on dose equivalent to the whole body, gonads, and blood-forming organs were replaced by limits on effective dose equivalent, defined as a weighted average of dose equivalents to six organs or tissues and a remainder category comprised of the five other organs or tissues receiving the highest dose equivalents. By this change, doses to many organs or tissues could be accounted for in limiting risks of stochastic effects, especially when nonuniform irradiation of the whole body from internal exposure occurred.

The tissue weighting factor (w_T) for an organ or tissue used in calculating effective dose equivalents as a weighted average of dose equivalents was intended to represent the fractional contribution from exposure of that organ or tissue to the total risk of cancer mortality or severe heritable effects when the whole body was irradiated uniformly. By use of the effective dose equivalent, doses from external and internal exposure, whether from uniform or nonuniform irradiations of the whole body, could be combined on a common risk basis.

Second, ICRP no longer considered that the genetic hazard was the primary concern in radiation protection. By adopting a w_T for the gonads of 0.25, the risk of severe heritable effects from uniform whole-body irradiation was assumed to be one-third of the risk of cancer mortality. The reduction in the importance of the genetic hazard was based mainly on radiobiologic data in mice and data in offspring of Japanese atomic-bomb survivors.

ICRP also developed estimates of risk coefficients (risks per Sv) for stochastic effects. Risk coefficients for organs and tissues other than the gonads were based mainly on data on cancer mortality in Japanese atomic-bomb survivors. ICRP assumed a nominal risk coefficient for cancer mortality from uniform whole-body irradiation of 10^{-2} Sv^{-1} .

Standards for workers

Recommended dose limits for routine exposure of workers in ICRP Publication 26 (ICRP, 1977) included:

³Terms related to radiation dose, quantities used in radiation protection, and their units in this chapter are defined in ICRP and NCRP reports referenced in this chapter.

⁴Dose equivalent averaged over an organ or tissue is now called equivalent dose (ICRP, 1991, 2007).

⁵Prior to introduction of SI units in radiation protection, absorbed dose was given in rads (1 rad = 0.01 Gy) and dose equivalent was given in rems (1 rem = 0.01 Sv).

- an annual effective dose equivalent of 50 mSv to limit stochastic effects;
- annual dose equivalents of 0.3 Sv to the lens of the eye and 0.5 Sv to all other organs and tissues to prevent tissue reactions;
- once a pregnancy was diagnosed, an annual effective dose equivalent to the mother of 15 mSv to protect the embryo/fetus.

The limit on annual effective dose equivalent was consistent with the previous limit on cumulative dose equivalent to the whole body at age N of $50(N-18)$ mSv. However, ICRP no longer recommended a limit on cumulative dose.

For planned special exposures that occur infrequently, ICRP recommended limits on effective dose equivalent of 100 mSv in single event and 250 mSv over a working lifetime.

Based on data in other safe occupations, ICRP judged that risks to workers would be acceptable if the average annual mortality risk did not exceed 10^{-4} . At ICRP's nominal risk coefficient for cancer mortality of 10^{-2} Sv^{-1} , the limit on annual effective dose equivalent of 50 mSv corresponded to an annual mortality risk about an order of magnitude higher than the average risk judged to be acceptable. However, the annual limit of 50 mSv was justified on the grounds that application of the ALARA principle had resulted in average annual doses to workers of about 5 mSv, so the risk of mortality for an average worker would be acceptable.

Recommended dose limits for routine exposure of workers in NCRP Report No. 91 (NCRP, 1987) included:

- an annual effective dose equivalent of 50 mSv;
- a total effective dose equivalent at age N of $10 \times N$ mSv;
- annual dose equivalents of 0.15 Sv to the lens of the eye and 0.5 Sv to all other organs or tissues;
- a total dose equivalent to the embryo/fetus (excluding medical exposure) of 5 mSv, and a dose equivalent of 0.5 mSv in any month once a pregnancy was known.

The limit on total effective dose equivalent, which replaced the previous limit of $50(N-18)$ mSv, was intended to limit the lifetime risk of fatal stochastic effects to $< 10^{-2}$.

NCRP also recommended that (1) the effective dose equivalent from rare planned special exposures should not exceed 100 mSv in a single event and over a working lifetime and (2) only actions involving life-saving justify acute exposures > 100 mSv in emergency situations.

Standards for the public

Recommended dose limits for routine exposure of the public in ICRP Publication 26 (ICRP, 1977) included (1) an annual effective dose equivalent of 5 mSv and (2) an annual dose equivalent to any organ or tissue of 50 mSv. In developing the limit on annual effective dose equivalent, ICRP judged that the risk of stochastic effects from exposure of the public should not exceed other risks the public accepts in everyday life, that is, an annual risk of mortality in the range of 10^{-6} to 10^{-5} . At ICRP's nominal risk coefficient for cancer mortality of 10^{-2} Sv^{-1} , the limit on annual effective dose equivalent for the public should have been about 1 mSv. However, the annual limit of 5 mSv was adopted based on a judgment that average annual doses to the public normally should be less than 0.5 mSv.

In a later clarification, ICRP (1985) emphasized that:

- the principal dose limit for the public was an annual effective dose equivalent of 1 mSv;
- a subsidiary limit of 5 mSv could be applied in some years if the annual effective dose equivalent averaged over a lifetime did not exceed 1 mSv.

Recommended dose limits for routine exposure of the public in NCRP Report No. 91 (NCRP, 1987) included:

- annual effective dose equivalents of 1 mSv for continuous or frequent exposure and 5 mSv for infrequent exposure;
- an annual dose equivalent to the lens of the eye, skin, and extremities of 50 mSv.

NCRP also recommended a remedial action level for indoor radon in dwellings, which was assumed to correspond to an annual effective dose equivalent of 20 mSv and was expected to limit the annual risk of lung cancer to about 4×10^{-4} .

Based on the assumed reduction in the importance of severe heritable effects, ICRP and NCRP did not retain their previous recommendations on limiting the genetically significant dose to the whole population at about one-third of the dose limit for individuals.

ICRP publication 60; NCRP report no. 116

In ICRP Publication 60 (ICRP, 1991), the effective dose equivalent was replaced by the effective dose, which differed in two important ways. First, dose equivalents to six additional organs and tissues were included in the remainder category.

Second, tissue weighting factors (w_T) used in calculating effective doses were intended to represent the fractional contribution from exposure of each organ or tissue to the total detriment, rather than risk, from uniform whole-body irradiation. Total detriment was calculated from estimated risks of cancer mortality or severe heritable effects by accounting for (1) lethality fractions for cancers in each organ or tissue (i.e., a weight was assigned to nonfatal cancers relative to fatal cancers) and (2) the relative years of life lost due to a cancer death attributable to radiation and the relative years of impaired life due to nonfatal cancers and heritable effects.

In Publication 60, ICRP developed nominal probability (risk) coefficients for cancer mortality from uniform whole-body irradiation of $4 \times 10^{-2} \text{ Sv}^{-1}$ in workers and $5 \times 10^{-2} \text{ Sv}^{-1}$ in the whole population.

Standards for workers

Recommended dose limits for routine exposure of workers in ICRP Publication 60 (ICRP, 1991) included:

- an annual effective dose averaged over defined periods of 5 years of 20 mSv and an effective dose in any year of 50 mSv;
- annual equivalent doses of 0.15 Sv to the lens of the eye and 0.5 Sv to localized areas of the skin and the hands and feet;
- once a pregnancy was declared, an equivalent dose to a woman's abdomen from external exposure of 2 mSv for the remainder of pregnancy and a committed effective dose from intakes of radionuclides of about 1 mSv.

The limit on annual effective dose averaged over defined 5-year periods was based on a judgment that a lifetime effective dose to workers > 1 Sv (probability of attributable death $> 4 \times 10^{-2}$) would be unacceptable.

In emergency situations, ICRP recommended that the effective dose should not exceed 0.5 Sv and the equivalent dose to the skin should not exceed 5 Sv, except in life-saving actions.

Recommended dose limits for routine exposure of workers in NCRP Report No. 116 (NCRP, 1993) are summarized as follows:

- Limits on annual and total effective dose were the same as the previous limits on effective dose equivalent in Report No. 91.
- The previous limit on annual equivalent dose to the lens of the eye was retained, but the higher limit of 0.5 Sv applied to the skin and the hands and feet only.
- The previous monthly limit on equivalent dose to the embryo/fetus once a pregnancy was known was retained, but the limit on total equivalent dose was not.

NCRP also adopted ICRP's recommended dose limits in emergency situations given above.

Standards for the public

Recommended dose limits for routine exposure of the public in ICRP Publication 60 (ICRP, 1991) and NCRP Report No. 116 (NCRP, 1993) were essentially the same:

- annual effective doses of 1 mSv for continuous or frequent exposure and 5 mSv for infrequent exposure;
- annual equivalent doses of 15 mSv to the lens of the eye and 50 mSv to the skin.

NCRP also applied the dose limit for the skin to the hands and feet.

The principal limit on annual effective dose of 1 mSv recommended previously was retained even though the nominal probability coefficient for cancer mortality in the whole population was a factor of five higher than the previous assumption. This decision was based on two considerations. First, widespread use by regulatory authorities of restrictions on doses from specific sources at a fraction of the annual limit of 1 mSv had reduced the highest doses to the public to levels well below the limit. Second, an annual limit of 1 mSv was comparable to variations in unavoidable doses from natural background in different locations, which the public generally did not consider in deciding where to live.

NCRP retained its previous remedial action level for exposure to indoor radon. For radon in dwellings, ICRP recommended an action level when a remedial action was simple that was assumed to correspond to an annual effective dose of about 20 mSv.

ICRP publication 103; NCRP report no. 180

In its current recommendations in Publication 103 (ICRP, 2007) [see also ICRP (2006)], ICRP addressed three exposure situations: (1) planned, (2) emergency, and (3) existing. In addition to dose limits, which apply in planned (routine) exposure situations only, ICRP recommended so-called dose constraints in planned exposures to specific sources and reference levels in emergency and existing exposure situations, which should be used as restrictions (starting points) in applying the ALARA principle. Use of dose constraints in planned exposure situations had been recommended previously (ICRP, 1991).

ICRP retained use of the effective dose and radiation detriment in its current recommendations. However, two important changes were made. First, the tissue weighting factor (w_T) for the gonads was reduced from 0.20 (the value in Publication 60) to 0.08. This reduction in the assumed importance of severe heritable effects compared with cancer was based mainly on an absence of clear evidence of such effects in offspring of Japanese atomic-bomb survivors.

Second, total radiation detriment was calculated from estimated risks of cancer incidence, rather than mortality, in different organs and tissues. However, by accounting for lethality fractions for different cancers and the relative years of life lost due to a cancer death or severe heritable effect, ICRP still regards cancer mortality as the primary stochastic effect of concern in radiation protection.

Standards for workers

Recommendations on protection of workers in planned exposure situations in ICRP Publication 103 (ICRP, 2007) are summarized as follows:

- The previous limits on (1) annual effective dose of 20 mSv averaged over defined periods of 5 years and 50 mSv in any year and (2) annual equivalent dose to the lens of the eye (0.15 Sv) and to the skin and hands and feet (0.5 Sv) are retained.
- The equivalent dose to the embryo/fetus for the remainder of pregnancy in declared pregnant women should not exceed 1 mSv.
- Dose constraints for specific sources should be ≤ 20 mSv annual effective dose.

ICRP's recommendations for other exposure situations include:

- In emergency situations, no restrictions on life-saving actions by informed volunteers if the benefit to others outweighs the risk to a rescuer; in actions not involving life-saving, reference levels of 1 or 0.5 Sv effective dose in urgent rescue operations and ≤ 0.1 Sv effective dose in other rescue operations;
- For existing exposures to radon at work, a reference level < 10 mSv annual effective dose (radon concentration in air < 1500 Bq m⁻³).⁶

Recommendations on protection of workers in planned exposure situations in NCRP Report No. 180 (NCRP, 2018) are summarized as follows:

- The previous limits on annual effective dose of 50 mSv, total effective dose at age N of $10 \times N$ mSv, and monthly equivalent dose to the embryo/fetus once a pregnancy is declared of 0.5 mSv are retained.
- The previous limit on annual equivalent dose to the skin and extremities is retained but is expressed as a radiation-weighted absorbed dose of 0.5 Gy⁷; the limit for the lens of the eye is 50 mGy.
- A worker's estimated annual and total effective dose should include the contribution from exposure to radon if the concentration of radon in air in the workplace, after application of mitigation measures, remains above 300 Bq m⁻³.

NCRP's recommendations for emergency exposure situations include:

- to prevent acute life-threatening effects, an absorbed dose to the whole body from photons < 0.5 Gy;
- in all activities other than life-saving or prevention of catastrophic conditions, an effective dose for the duration of an emergency operation < 0.1 Sv.

Standards for the public

Recommendations on protection of the public in planned exposure situations in ICRP Publication 103 (ICRP, 2007) include:

- a limit on annual effective dose of 1 mSv, with a higher effective dose allowable in a single year in special circumstances if the average annual effective dose over 5 years does not exceed 1 mSv;
- limits on annual equivalent dose of 15 mSv to the lens of the eye and 50 mSv to the skin;
- dose constraints for specific sources ≤ 1 mSv annual effective dose (e.g., ≤ 0.3 mSv for radioactive waste disposal, ≤ 0.1 mSv for prolonged exposure to long-lived radionuclides);
- dose constraints for volunteers in biomedical research ranging from < 0.1 mSv effective dose if the benefit to society is minor to > 10 mSv if the benefit is substantial;
- a dose constraint for comforters and care-givers of patients undergoing diagnosis or treatment of 5 mSv effective dose per episode.

In most emergency exposure situations, ICRP recommends a reference level for the public of 20 to 100 mSv annual effective dose for all planned countermeasures combined.

ICRP's recommendations for existing exposures of the public include:

- for exposure to radon at home, a reference level < 10 mSv annual effective dose (radon concentration in air < 600 Bq m⁻³);
- for exposure to naturally occurring radioactive material (NORM), natural background radiation, and radioactive residues in human habitats, a reference level of 1 to 20 mSv annual effective dose.

Recommendations on protection of the public in planned exposure situations in NCRP Report No. 180 (NCRP, 2018) include:

- limits on annual effective dose of 1 mSv for continuous or frequent exposure and 5 mSv for infrequent exposure;
- limits on annual radiation-weighted absorbed dose of 15 mGy to the lens of the eye and 0.5 Gy to the skin and extremities;
- for volunteers in biomedical research, an effective dose ranging from < 0.1 mSv if the benefit to society is minor to > 0.1 Sv if the benefit to the individual is very substantial;
- for comforters and care-givers of patients undergoing diagnosis or treatment, the effective dose should not exceed 5 mSv per episode.

NCRP's recommendations for existing exposures of the public are summarized as follows:

- Concentrations of radon in air in dwellings should be < 300 Bq m⁻³ (expected to correspond to an annual effective dose of 1 to 20 mSv, depending on the exposure time).
- In exposure situations not previously subject to control, including exposure to elevated levels of NORM whether enhanced by human activities or not, the effective dose in the first year should not exceed 20 mSv, and the effective dose in later years should be reduced further by applying the ALARA principle.

⁶Prior to introduction of SI units in radiation protection, activity was given in curies (Ci) ($1 \text{ Ci} = 3.7 \times 10^{10} \text{ Bq}$).

⁷Weighting factors for densely ionizing radiations (e.g., neutrons, alpha particles, heavy ions) recommended in NCRP Report No. 180 are less than radiation weighting factors (w_R s) for stochastic effects used in calculating equivalent dose in an organ or tissue.

The latter recommendation also applies to exposure of the public to radioactive material introduced into the environment intentionally or by accident.

Current U.S. standards

Federal standards to control radiation exposures of workers and the public in the U.S. are established by the Nuclear Regulatory Commission (NRC), Department of Energy (DOE), and Environmental Protection Agency (EPA). NRC and DOE establish enforceable standards under authority of the Atomic Energy Act that apply to NRC licensees or to DOE and its contractor organizations only.⁸ EPA develops guidance to all federal agencies on radiation protection standards and establishes enforceable standards or guidance for a wide variety of specific sources of radiation in the environment under authority of the Atomic Energy Act and several other laws.

Not discussed here are (1) standards established by state authorities that apply to sources not regulated by a federal agency (e.g., exposure of workers to radiation emitted by x-ray machines at privately owned facilities) and (2) a standard used by the National Aeronautics and Space Administration to protect crew members during activities in space (NASA, 2014).

Radiation protection standards for workers

NRC's current radiation protection standards for workers in Title 10, Part 20, of the Code of Federal Regulations (10 CFR 20), first established in 1991, are based mainly on federal guidance developed by EPA (Office of the President, 1987), which incorporated recommendations in ICRP Publication 26 (ICRP, 1977). DOE's current radiation protection standards for workers in 10 CFR 835, first established in 1993, are based mainly on recommendations in ICRP Publication 60 (ICRP, 1991).

Radiation protection standards for workers in NRC and DOE regulations are summarized in Table 1. They differ from recent ICRP and NCRP recommendations by not including a limit on average annual dose over a defined period less than the dose limit in any year or a limit on total dose over a working lifetime.

Radiation protection standards for the public

Current radiation protection standards for the public in NRC and DOE regulations are summarized in Table 2. These standards apply to the total doses at a location from all sources that are under the control of the same organization. In NRC standards in 10 CFR 20 first established in 1991, limits on annual effective dose equivalent are based on recommendations in ICRP Publication 26 (ICRP, 1977). In DOE standards in Order 458.1 established in 2011, dose limits are based on recommendations in ICRP Publication 103 (ICRP, 2007).

Environmental radiation standards for specific sources

EPA, NRC, and DOE have established many enforceable standards or guidance to control exposures of the public to specific sources of radiation in the environment.⁹

Table 1 Radiation protection standards for workers established by NRC (10 CFR 20) and DOE (10 CFR 835).

Doses to workers shall be maintained as low as reasonably achievable (ALARA)
Dose limits for routine exposures ^a :
• annual effective dose equivalent or effective dose of 50 mSv
• annual dose equivalent or equivalent dose of 0.15 Sv to lens of the eye and 0.5 Sv to any other organ or tissue, including skin and extremities
• dose equivalent or equivalent dose to embryo/fetus from conception to birth in declared pregnant woman of 5 mSv
• weekly limit on intake of soluble uranium of 10 mg (NRC regulations only; consideration of chemical toxicity)
Dose limits for planned special exposures ^a :
• effective dose equivalent or effective dose of 50 mSv in any year and 250 mSv over a working lifetime ^b

^aNRC standards specify limits on effective dose equivalent or average dose equivalent in an organ or tissue (ICRP, 1977). DOE standards specify limits on effective dose or equivalent dose (ICRP, 1991).

^bAllowable doses in planned special exposures are in addition to allowable doses in routine exposures.

⁸NRC standards apply to exposures to (1) radioactive materials associated with operations at nuclear-fuel cycle facilities in the civilian sector and (2) certain accelerator-produced and naturally occurring radioactive materials. DOE standards apply to exposures associated with operations at any of its facilities, including operations related to the development and production of nuclear weapons.

⁹These standards and other standards and guidance, such as licensing criteria in NRC regulations, are discussed in more detail elsewhere (Kocher, 2008).

Table 2 Radiation protection standards for the public established by NRC (10 CFR 20) and DOE (Order 458.1).

Doses to the public shall be maintained as low as reasonably achievable (ALARA).

NRC's dose limits for routine exposures^a:

- annual effective dose equivalent of 1 mSv, excluding dose from permissible releases of radionuclides into sanitary sewerage
- annual effective dose equivalent up to 5 mSv with prior authorization
- dose equivalent in any unrestricted area from external sources of 0.02 mSv in any hour
- annual average concentrations of radionuclides in air and water in unrestricted areas based on annual effective dose equivalent of 0.5 mSv from inhalation or ingestion

DOE's dose limits for routine exposures^b:

- annual effective dose of 1 mSv, excluding dose from radon and decay products
- annual effective dose of 5 mSv with prior authorization, provided that average annual effective dose over any 5-year period does not exceed 1 mSv
- annual equivalent dose of 15 mSv to lens of the eye and 50 mSv to skin and extremities

^aNRC standards use effective dose equivalent and dose equivalent as defined in ICRP Publication 60 (ICRP, 1977).

^bDOE standards use effective dose and equivalent dose as defined in ICRP Publication 103 (ICRP, 2007).

Environmental radiation standards for specific sources established under the Atomic Energy Act are described in [Table 3](#). The various protection criteria represent exposure conditions judged to be reasonably achievable and, thus, are starting points for further application of the ALARA principle. Dose criteria in those standards are examples of dose constraints, which were used in environmental standards in the U.S. long before their use was recommended by ICRP (1991).

Environmental radiation standards and guidance established by EPA under laws other than the Atomic Energy Act are described in [Table 4](#). Except for standards for cleanup of contaminated sites under CERCLA, those standards and guidance were based on exposure conditions judged to be reasonably achievable using existing technology.

Standards for radioactivity in drinking water and cleanup of contaminated sites in [Table 4](#) differ in an important way from standards in [Table 3](#) (NCRP, 2004; Kocher, 2008). Standards established under the Atomic Energy Act are based on the principles of radiation protection described in the section "[Principles of radiation protection](#)." Those principles define a "top-down" approach to radiation protection, in which a limit on exposure that implicitly defines an upper bound on acceptable risk is prescribed and exposures are reduced below the prescribed limit by applying the ALARA principle.

The approach to regulation under the Safe Drinking Water Act and CERCLA is the opposite of the approach under the Atomic Energy Act. Those laws define a "bottom-up" approach to regulation, in which a risk goal is prescribed and an increase in risks above the goal is allowed based primarily on considerations of technical feasibility and cost.

Under the Safe Drinking Water Act, the goal for levels of all radionuclides in drinking water is zero, which cannot be achieved at any cost. Maximum contaminant levels (MCLs) in EPA regulations—the allowed relaxations above the goal—were based on existing levels in groundwater and surface waters and considerations of the costs and benefits to public health of reducing those levels.

The difference between the "top-down" and "bottom-up" approaches to regulation has been a particular source of confusion in decisions on cleanup of contaminated sites under CERCLA. Stakeholders in the public often assume that achieving a lifetime cancer risk in the range of 10^{-6} to 10^{-4} is a requirement, i.e., that any risk above that range is unacceptable. However, achieving a lifetime cancer risk of 10^{-4} or lower is required only if the cost is reasonable; i.e., the risk goal under CERCLA essentially defines an upper bound on negligible risk. Indeed, achieving a lifetime cancer risk of 10^{-4} has not been feasible in cleanup of many radioactively contaminated sites (NCRP, 2004).

Protective action guides for radiological incidents

In cooperation with other federal agencies, EPA (2017) developed guidance to state and local authorities on protective actions to be taken in response to accidents and other incidents in which radionuclides are released to the environment, including accidents at nuclear facilities, transportation accidents, and terrorist acts involving a radiological dispersal device or improvised nuclear device.

Protective action guides (PAGs) are defined for three time phases:

- Early phase: Time from hours to days after an event, when immediate decisions about responses (e.g., sheltering, evacuation, administration of stable iodine) are required;
- Intermediate phase: Time from weeks to months after an accident or incident is brought under control, when the need for additional protective actions (e.g., control of foodstuffs and drinking water, control of access to contaminated areas, relocation of populations) can be assessed based on environmental measurements;
- Recovery phase: Time from months to years, when actions can be taken to reduce levels of radionuclides in the environment and allow permanent residence at a site.

EPA's current PAGs are described in [Table 5](#). Dose criteria generally are consistent with protection standards for workers or the public in other exposure situations.

Table 3 Protection criteria in regulations for specific sources of radiation in the environment established by EPA, NRC, and DOE under authority of Atomic Energy Act.

<i>Regulation</i>	<i>Protection criteria</i>
Operations of uranium fuel-cycle facilities—EPA's 40 CFR 190 (1977)	Annual dose equivalent to whole body (0.25 mSv), thyroid (0.75 mSv), or any other organ (0.25 mSv) Releases of ^{85}Kr , ^{129}I , ^{239}Pu plus other alpha emitters
Management of uranium and thorium mill tailings—EPA's 40 CFR 192 (1983, 1995) ^{a,b}	Emission rate of radon to atmosphere or concentration of radon decay products in air (including background) in occupied or habitable buildings Concentrations of ^{226}Ra plus ^{228}Ra , gross alpha-particle activity excluding radon and uranium, ^{234}U plus ^{238}U in groundwater Concentration of ^{226}Ra in soil above background Gamma radiation level above background in occupied or habitable buildings Individual dose constraints (EPA's 40 CFR 190) Radioactivity in liquid discharges to surface waters (EPA's 40 CFR 440) Standards at uranium processing sites, except standards for radon and decay products, apply to thorium and decay products
Release or clearance of property containing residual radioactive material—DOE Order 458.1 (2011)	Dose constraints for each clearance—annual effective dose for real property (0.25 mSv) or personal property (0.01 mSv) Preapproved authorized limits for release of personal property [annual effective dose (0.01 mSv) and collective effective dose (0.1 person-Sv)] or real property (concentrations of ^{230}Th / ^{226}Ra , ^{232}Th / ^{228}Ra in soil) Authorized limits requiring DOE approval based on applicable dose constraint (radioactivity per unit surface area, mass, or volume)
Decontamination and decommissioning of licensed facilities (except uranium and thorium mill tailings sites)—NRC's 10 CFR 20, Subpart E (1997)	Annual effective dose equivalent to permit unrestricted use of contaminated land and structures (0.25 mSv) or license termination under restricted conditions (0.25, 1, or 5 mSv)
Management and disposal of spent fuel, high-level waste, and transuranic waste, except disposal at Yucca Mountain, Nevada, site—EPA's 40 CFR 191 (1985, 1993) ^{c,d}	Management and storage—Annual dose equivalent to whole body (0.25 mSv) or any organ (0.25 or 0.75 mSv) Disposal—For 10,000 years, (1) cumulative releases of radionuclides to accessible environment, (2) annual effective dose equivalent from releases from undisturbed disposal system (0.15 mSv), and (3) protection of groundwater (EPA's 40 CFR 141)
Management and disposal of spent fuel and high-level waste at Yucca Mountain, Nevada, site—EPA's 40 CFR 197 (2001) ^e	Management and storage—Annual effective dose equivalent (0.15 mSv) Disposal—(1) Annual effective dose equivalent from releases from undisturbed disposal system or inadvertent human intrusion up to 10,000 years (0.15 mSv) and beyond but within period of geologic stability (1 mSv) and (2) protection of groundwater for 10,000 years (EPA's 40 CFR 141)
Near-surface disposal of radioactive waste—NRC's 10 CFR 61 (1982, 1993)	Annual dose equivalent to whole body (0.25 mSv), thyroid (0.75 mSv), or any other organ (0.25 mSv) beyond site boundary Protection of inadvertent intruder—(1) Dose limits in radiation protection standards (10 CFR 20) and (2) concentrations of radionuclides in three classes of waste
Disposal of low-level waste—DOE Manual Chapter M 435.1–1 (1999)	For 1000 years— Annual effective dose equivalent from all release and exposure pathways, excluding radon and decay products (0.25 mSv) Annual effective dose equivalent from releases to air, excluding radon and decay products (0.1 mSv) Average release rate of radon or concentration of radon in air at facility boundary Concentrations of radionuclides based on effective dose equivalents to inadvertent intruder (1 mSv annually for chronic exposure, 5 mSv total for acute exposure)

^aApplicable standards depend on whether site is active or inactive (uranium processing sites only) and whether site is managed by DOE or by NRC or Agreement State licensee.

^bNRC standards for decontamination and decommissioning of licensed thorium mills and uranium recovery facilities in 10 CFR 40, Appendix A (1999), conform in most respects to EPA's 40 CFR 192.

^cStandards presently apply only to disposal of DOE's defense transuranic waste at Waste Isolation Pilot Plant (WIPP) in New Mexico.

^dNRC's licensing criteria for geologic repositories subject to EPA's 40 CFR 191 are established in 10 CFR 60 (1983, 1996). NRC does not license disposal at WIPP.

^eNRC's licensing criteria for disposal at Yucca Mountain site in 10 CFR 63 (2001) include standards in EPA's 40 CFR 197.

Summary and conclusions

The system of radiation protection developed by ICRP and NCRP is the basis for many current standards to control exposures of workers and the public in the U.S. The foundation of this system is the three principles of radiation protection: (1) justification of all exposures (positive net benefit); (2) optimization of protection (maintaining all exposures ALARA); and (3) limitation of dose to individuals in planned (routine) exposure situations.

Table 4 Protection criteria in regulations and guidance for specific sources of radiation in the environment established by EPA under authority of laws other than Atomic Energy Act.

<i>Regulation/applicable law</i>	<i>Protection criteria</i>
Radioactivity in community drinking water systems—40 CFR 141 (1976, 2000) (Safe Drinking Water Act) ^a	Maximum contaminant levels (MCLs) of radionuclides— Concentrations of alpha-emitting radionuclides (²²⁶ Ra plus ²²⁸ Ra, uranium, gross alpha-particle activity excluding radon and uranium) Concentrations of man-made beta/gamma-emitting radionuclides corresponding to annual dose equivalent to whole body or any organ (0.04 mSv), ^b except MCLs for ³ H and ⁹⁰ Sr
Liquid discharges from radium, uranium, or vanadium mines and mills—40 CFR 440 (1982) (Clean Water Act)	Concentrations of dissolved ²²⁶ Ra, total ²²⁶ Ra, uranium in daily effluents and average concentrations in daily effluents for 30 consecutive days
Cleanup of residual radioactive material other than uranium and thorium mill tailings—40 CFR 300 (1990) [Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)]	Applicable or relevant and appropriate requirements (ARARs) established under other federal and state environmental laws, including EPA's drinking water standards (40 CFR 141) for remediation of groundwater or surface waters Goal for upper bound of lifetime cancer risk from all substances and all exposure pathways combined (10^{-6} to 10^{-4}) ^c
Airborne emissions of radionuclides—40 CFR 61 (1989–1996) (Clean Air Act)	Annual effective dose equivalent from emissions of radon from uranium mines (0.1 mSv) Annual effective dose equivalent from emissions of radionuclides other than radon from DOE facilities other than those subject to EPA's 40 CFR 191 or 192 (0.1 mSv) Annual effective dose equivalent from (1) emissions of radionuclides, including iodine, from facilities other than NRC licensees and DOE sites (0.1 mSv) and (2) emissions of iodine (0.03 mSv) Annual emissions of ²¹⁰ Po from elemental phosphorus plants Emission rate of ²²² Rn from DOE facilities, inactive phosphogypsum stacks, and inactive uranium mill tailings piles except those licensed by NRC Concentration of ²²⁶ Ra in phosphogypsum used in agriculture Radon concentration in air—
Guidance on mitigation of indoor radon in homes and schools (EPA and DHHS, 1994) (Indoor Radon Abatement Act)	Mitigation should be undertaken at $> 150 \text{ Bq m}^{-3}$ (4 pCi L^{-1}) Mitigation should be considered at 70 to 150 Bq m^{-3} ($2 \text{ to } 4 \text{ pCi L}^{-1}$), especially when $< 70 \text{ Bq m}^{-3}$ is achievable

^aDOE Order 458.1 (2011) applies EPA's drinking water standards to releases from DOE sites that may impact public or private drinking water supplies.

^bRadionuclide concentrations corresponding to specified annual dose equivalent to whole body or any organ are calculated assuming daily consumption of drinking water of 2 L and committed dose equivalents per unit activity intake from NCRP Report No. 22 (NCRP, 1959), as amended in 1963.

^cEPA considers that goal of 10^{-4} is met at radioactively contaminated sites if annual effective dose equivalent does not exceed 0.15 mSv.

Table 5 Guidance on protective actions in response to radiation accidents and incidents developed by EPA and other federal agencies (EPA, 2017).

<i>Time phase</i>	<i>Protective action</i>	<i>Protective action guide (PAG)</i>
Early	Sheltering in place or evacuation of population	Projected effective dose over 4 days > 10 to 50 mSv
	Supplemental administration of potassium iodide (KI)	Projected equivalent dose to child's thyroid from exposure to radioactive iodine > 50 mSv
Intermediate	Limit exposure of emergency workers over entire response	Annual effective dose < 50 mSv; higher limit may apply under exceptional circumstances
	Relocation of population	Projected effective dose in first year > 20 mSv Projected effective dose in second and subsequent years > 5 mSv
	Apply simple dose reduction techniques	Projected effective dose in first year > 20 mSv
	Control of food supplies	Projected annual equivalent dose > 5 mSv to whole body or > 50 mSv to any organ
	Control of drinking water supplies	Projected effective dose in one year > 1 mSv to most sensitive populations (e.g., infants, children, pregnant and nursing women) or > 5 mSv to general population
Late ^a	Limit exposure of emergency workers over entire response	Annual effective dose < 50 mSv
	Reentry for access to relocation zone	Effective dose in 1 year < 5 mSv for temporary access
	Cleanup	Description of planning process
	Waste disposal	Description of planning process

^aIn previous guidance, EPA (1992) recommended that annual effective dose equivalent to the public after the first year should not exceed 5 mSv and total effective dose equivalent over 50 years, including first and second years, should not exceed 50 mSv (annual average of 1 mSv). For reactor accidents, EPA expected that objectives would be met if PAG for intermediate phase of 20 mSv in first year is met.

Standards to control radiation exposures of workers and the public have two basic objectives: (1) limitation of the risk (probability) of stochastic effects, principally solid cancers and leukemias, and (2) prevention of tissue reactions (nonstochastic effects) in all organs and tissues. In planned exposure situations, tissue reactions usually are prevented by complying with limits on effective dose equivalent or effective dose to individuals, which are intended to limit risks of stochastic effects. This is especially the case in planned exposures of the public.

As radiation protection practices evolved over many decades, application of the ALARA principle became the most important means of controlling exposures of workers and the public in any situation. For example, the average annual dose to workers at nuclear power plants typically is only a few percent of the limit of 50 mSv in NRC regulations, and the highest annual doses to the public due to releases of radionuclides in normal operations of nuclear power plants typically are < 1% of EPA standards in 40 CFR 190. Consequently, in normal operations of nuclear power plants, dose limits for workers and the public are largely irrelevant. More generally, most radiation exposures of workers and the public are controlled by use of dose constraints in planned exposure situations at a fraction of an applicable dose limit and other protection criteria in environmental standards for specific sources, which represent exposure conditions judged to be reasonably achievable, and further application of the ALARA principle.

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Risk Tradeoff Analysis

T.W. Hansen, Jr, Ameriphysics LLC, Knoxville, TN, United States

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Introduction

By now, readers of this section on radiation protection will know that ionizing radiation presents a human health risk, that the risk associated with most radiation exposures is cancer incidence, and that the chance of developing a radiation-induced cancer increases with increasing dose. If one presumes that any amount of radiation exposure, no matter how small, is associated with an increased chance of cancer, then the relationship between dose and cancer risk is described as conforming to a linear no-threshold (LNT) model.

If the LNT model is all one understands about radiation, or all one considers when establishing regulations or controls, then the approach for mitigating risk seems straightforward—prevent or prohibit any radiation exposure and then all excess cancers will be avoided. To a large extent, this is a simple and practicable protective approach, and maintaining exposures as low as reasonably achievable, or ALARA, is an important principle that guides virtually every radiation protection program in the world.

In the United States, federal policymaking requiring elements of ALARA can be traced back to at least 1970 when the predominant nuclear regulatory agency at the time, the Atomic Energy Commission, added an “as low as practical” provision to its regulations rather than publish absolute dose limits. The Energy Reorganization Act of 1974 marked the end of the Atomic Energy Commission and transferred certain regulatory functions to a new agency, the Nuclear Regulatory Commission (NRC), and in 1975, the NRC changed the terminology from “as low as practical” to “ALARA”. The ALARA requirement remains in effect today and obliges radioactive material licensees under NRC and Agreement State jurisdictions to demonstrate efforts to minimize both individual and collective doses to the lowest extent achievable.

The underlying promise of an ALARA-based program is that regardless of how low the current dose, reasonable efforts to make the dose even lower are valuable. The reward seems clear, as according to the linearity predicted by an LNT model, a reduction in dose translates directly to fewer cancers. As a result, considerable extra effort is often expended on actions to further reduce doses that are already below regulatory limits because such efforts are expected to improve health.

In terms of net health outcomes, however, there is not any direct evidence that the nation’s five-decade commitment to ALARA has benefited the public. Studies examining health outcomes in nuclear workers, in persons who reside near nuclear power stations, and in other regularly exposed cohorts do not produce results showing the linearity predicted by the LNT model. Conversely, there is ample evidence to suggest that policy actions such as ALARA that are singularly focused at reducing one risk often lead to important risk tradeoffs that create or exacerbate other health risks.

Take for example a worker who is routinely exposed to small amount of airborne radioactivity within regulatory limits for such an exposure. If one is interested in reducing the probability that the worker develops a radiation-induced cancer according to ALARA, additional controls should be taken to further reduce the worker’s dose if deemed reasonably achievable. A simple yet effective intervention where airborne radioactivity is concerned is to require the worker to wear respiratory protective equipment. Because the respirator prevents radioactivity from entering the body to a large extent, the worker dose and presumed risk of radiation-induced cancer incidence are reduced. However, wearing a respirator limits a person’s field of vision and makes breathing more difficult, and these and other factors resulting from respirator use introduce new health risks. The relative magnitude and potential severity of the new risks should be evaluated to determine the net impact of the intervention on the worker’s health. If the new risks outweigh the perceived benefits, the value of the intervention should be reconsidered.

As the previous example demonstrates, it is not always reasonable to make doses lower. However, without criteria explaining what is reasonable, practitioners who are reluctant to be found operating at odds with regulations that require ALARA will often take actions to make doses as low as possible. “As low as possible” does not have nearly the same meaning as ALARA. If we are truly interested in lowering doses as a means of protecting the public and improving health, then we should be as equally concerned with negative outcomes of such interventions.

The purpose of this article is not to criticize the practice of ALARA. Rather, it examines certain factors that should be understood whenever dose-limiting actions are planned to determine if such actions are reasonable. These factors include a factual understanding of radiation effects in humans, risk tradeoff, and moral disengagement theory.

Perceived versus actual effects of radiation

Ionizing radiation was discovered in 1895, and it was initially studied and used without any concerns or controls (Walker, 2000). Researchers often experimented on themselves and others in a manner leading to severe overexposures, and the degrading health of scientists and their subjects in the first few decades following radiation's discovery left little doubt that it was harmful. By the early 20th century, the public understood that large exposures presented a health hazard; nonetheless, work and research with radiation continued without a tremendous amount of public outcry.

Public perceptions changed in 1945 when the United States detonated two nuclear weapons over the Japanese cities of Hiroshima and Nagasaki. Although it is not known which deaths were due to radiation and which were due to the blast or heat, about 39% of the population of Hiroshima and 28% of the population of Nagasaki were killed (Nagataki, 2010). About half of the deaths occurred in the period spanning 20–29 days after the bombings, and nausea, vomiting, diarrhea, purpura, epilation, and other symptoms of acute radiation syndrome (i.e., radiation sickness) were observed within a few days to several weeks after the bombing (Nagataki, 2010). The shocking images returned from the aftermath and reported by the media generated considerable public concern about the harmful effects of radiation, and to a large extent, that concern persists to the present day.

Shortly after the bombings, radioactive fallout could be detected on the California coast which further exacerbated public concerns in the United States. These concerns resonated in a California town with an appetite for what interests the public, Hollywood, and blockbuster after blockbuster promising nuclear apocalypse soon followed. The effects of radiation were overwhelmingly embellished in such films, and audiences left theaters contemplating to what extent radiation could mutate insects and other creatures into monsters or cause other horrific abominations.

Fortunately, the range of possible radiation-induced biological effects differs greatly from the Hollywood exaggerations that continue to entertain the public. When such effects are described by researchers and practitioners, a popular way to categorize them is whether they are “early” (often called “acute”) or “late” (sometimes called “delayed”). When this manner of categorization is used, the category names refer to the length of time for the biological effect to manifest. Acute effects are usually observed in the days, weeks, or months following an exposure; whereas late effects may not be seen for years or decades.

Evidence that high levels of radiation present acute effects is observed in studies of atomic bomb survivors and nuclear accidents (NAS, 2006). The acute effects demonstrated in bomb survivors have already been described, namely, radiation sickness and death. The effects have been confirmed in other studies involving large acute exposures. For instance, 134 people demonstrated symptoms of acute radiation syndrome following the Chernobyl nuclear accident, and 28 died within 3 months (Nagataki, 2010). Although these are grave effects, it is important for readers to understand that the exposures leading to acute effects must be unusually large and received over a short period of time.

In addition to acute effects, late effects are also observed in A-bomb survivors. Although other late effects are possible, a 50% excess solid-cancer risk is estimated in survivors exposed to 1 Sv (Nagataki, 2010; NAS, 2006). A solid cancer is characterized by an abnormal growth of cells that leads to a solid “mass” or “tumor”, as opposed to liquid cancers such as lymphomas and leukemias that are present in body fluids. A sievert, or Sv, is a unit of biologically weighted absorbed dose that is equivalent to depositing 1 joule of energy per kilogram of tissue. This definition of a Sv is not particularly meaningful except to radiation professionals, so it is helpful to think about it as a unit of dose that may be translated to risk.

Because an understanding of radiation-induced cancer risk is so important to the current discussion, the risk-magnitude inferred by the sentence introducing the preceding paragraph will be reviewed. First, a whole-body dose of 1 Sv will not be obtained except in unique circumstances such as nuclear catastrophes. For example, if a standard chest X-ray procedure is presumed to give a dose of 0.0001 Sv, then 1 Sv is approximately the dose a person would receive if exposed to 10,000 chest X-rays. A 50% excess risk corresponds to a relative risk of 1.5, meaning a person exposed to this large dose would still only be 1.5 times as likely to develop a solid cancer as a person not exposed. This does not mean a person would die from the cancer, only that the probability of developing a cancer at some point in that person's lifetime is increased. Finally, radiation-induced cancer is not different from other cancers. For example, it is impossible to distinguish which brain cancers in A-bomb survivors are due to radiation and which are due to other factors.

In A-bomb survivors, higher and lower cancer risk is associated with exposures above and below 1 Sv, respectively, and the dose-response relationship appears to be linear down to 0.1 Sv (NAS, 2006). Research involving late effects in other cohorts, however, do not necessarily conform to predictions using data from atomic bomb survivors. Nonetheless, solid cancer (i.e., cancers other than leukemia and other hematopoietic cancers) is usually discussed in terms of being the most important late effect.

Below a dose of about 0.1 Sv, or approximately equal to what would be received from 1000 chest X-rays, the actual risk remains unknown. This is true despite examining health outcomes in hundreds of thousands of diverse human subjects over the better part of a century. Researches have studied late effects in A-bomb survivors; persons receiving medical radiation; nuclear industry workers; airline and aerospace employees; cleanup workers; doctors and dentists; people living near nuclear facilities; populations exposed to the Chernobyl accident, environmental releases of radiation, fallout, and natural background radiation; children of adults exposed to radiation; and other exposed cohorts (NAS, 2006; GAO, 2000). Thus, the fact that the risk remains unknown is not because

scientists lack data, but because so few (or any) excess cancers and other late effects are observed below 0.1 Sv that the risk is too low to determine. If we categorize doses above and below 0.1 Sv as “high” and “low” doses, respectively, then we simply do not have any statistical evidence of late effects for low doses unless we presume it is reasonable that the relationship for high doses applies to both dose ranges.

The fact that low doses of radiation presents little if any risk is evidenced by the everyday exposures that the public tolerates. Air travelers receive a higher dose of cosmic radiation than persons who travel by car because less cosmic radiation is shielded by the atmosphere. Living at a higher altitude has a similar effect, thus residents of Denver receive more dose than persons living near sea-level. A large number of medical and dental diagnostic procedures rely on radiation that patients accept. People choose professions that result in a bit more dose than the general public will receive, such as nuclear power plant workers, X-ray technicians, and pilots and flight attendants. If these exposures were regularly resulting in excess cancers, the public would be inclined to avoid them.

The takeaway from this concise primer on biological effects is that ionizing radiation is considerably less hazardous than the public perceives it to be. In the end, we find that the most likely effect is an increased probability of developing cancer, but even then it takes a large amount of radiation exposure to a large population to see a measurable increase in cases as low-dose radiation is not a very efficient cancer-causing agent.

Risk tradeoff and analysis

If low-dose radiation is understood to present little hazard, one might ask why it is regulated at all. Shortly after the NRC was established, the agency did contemplate whether very low levels of radiation exposure could be designated as “below regulatory concern” (BRC). Such levels of exposure were stated to be associated with public health risks so small or nonexistent that there did not appear to be value in expending regulatory resources. After more than a decade of debate, the NRC issued a policy statement in June 1990 declaring certain low levels of radiation BRC (Walker, 2000). The resulting public outcry was so enormous that Congress initiated legislation to block BRC, and 12 states and more than 100 local governments eventually took steps to prohibit implementation of the policy (Walker, 2000). Due to the overwhelming public and political response, the NRC declared a moratorium on implementation of the policy, and the moratorium was indefinitely extended in December of 1991 (Walker, 2000).

In part, what the NRC was trying to accomplish with its policy on BRC was to improve public health by preventing overregulation. Interventions, such as policies, regulations, and other actions aimed at mitigating a risk, are understood to often exacerbate or create other health risks (Graham and Wiener, 1995). The risk that is addressed by the intervention is called the target risk, and the risks that arise or are increased are called countervailing risks. The process wherein a target risk reduction is traded for countervailing risk is called risk tradeoff (Graham and Wiener, 1995). Where overregulation happens, risk tradeoff occurs without the benefit of a tangible reduction in the target risk, and the net result is overall poorer health outcomes than what are possible if the target risk were optimally regulated.

The fact that risk tradeoffs do occur is evidenced in contemporary research that examines the net benefit of radiation policymaking. For example, more than 1900 unnecessary deaths are attributed to mandatory evacuations that followed the 2011 disaster at the Fukushima Daiichi nuclear power plant (Hayakawa, 2016). The evacuations were conducted to avoid the same amount of radiation and risk one might receive from a common medical procedure, a dose of about 0.02–0.1 Sv, but the physical and mental stresses associated with evacuee living have caused considerable harm including death (Hayakawa, 2016). In another example, a 2004 study conducted by the US Department of Energy concluded that the agency had spent more than \$60 billion since 1989 on radioactive site cleanup without realizing a reduction in actual health risk because the work introduced a number of ecological, temporal, and human health tradeoffs (Burger et al., 2004).

Sources of risk tradeoff

The likelihood of risk tradeoff occurring increases when decision-making is conflicted by certain concerns that are deemed “sources” of risk tradeoff (Graham and Wiener, 1995). The first of these sources is called “bounded roles” and reflects the biases that arise when regulatory agencies are structurally separated into groups of specialists. Specialists assigned to radiation protection agencies or departments who are tasked with reducing radiation-induced cancers will naturally accept risk tradeoffs to achieve such reductions. Moreover, experts on radiation effects may possess little knowledge of risks or hazards outside their specialty. “Omitted voice” is a source of risk tradeoff when the voices of some impacted parties are heard but others are not, as decision makers are likely to consider what is heard as reflecting the desires of the majority when that might not be the case. “Heuristics” are cognitive tools that have evolved in humans over generations. These tools allow large amounts of data to be quickly sorted for use in decision-making, but they do not work flawlessly. Heuristics are responsible for a tendency to focus on recent events or what are perceived as immediate concerns and to ignore the consequences of one’s responding actions until they become issues later. People are reluctant to accept change, making “old technology bias” a source of risk tradeoff whenever something new or innovative is attempted, and “human behavioral responses” are sources of risk tradeoff when actions taken to reduce risk unintentionally lead to other risky behaviors.

When decision-making is impacted by one or more of the preceding source concerns, the benefits and harms of the outcome decisions affecting risk should be analyzed. Federal agencies have been required to accomplish some means of benefit-cost analysis in conjunction with their regulatory actions since 1981 when then President Ronald Reagan issued Executive Order 122291 (NRC,

2017). Presidents Bill Clinton, George W. Bush, and Barack Obama have signed similar Orders requiring agencies to select interventions that maximize net benefits when economic, environmental, public health and safety, equity, and other factors are considered. Notwithstanding such Orders, NRC maintains that as an independent agency, it is not statutorily required to conduct comprehensive regulatory analysis (NRC, 2017). Instead, it relies on an analysis tool called probabilistic risk assessment, or PRA, that exclusively examines the monetized population dose that is averted and other economic costs of a decision.

All of the possible sources of risk tradeoff are present when ALARA is practiced to limit dose. The practice itself is handed down in regulations borne from specialists in agencies with bounded roles. These specialists are likely to hear from entities interested in radiation's risks and not from voices interested in other risks. The term ALARA is meant to appeal to our heuristic nature as it seems reasonable to keep doses as low as possible, and ALARA has been practiced in some form or fashion since the onset of environmentalism, making it subject to old technology biases. Over the years, human behavioral responses are responsible for ALARA being practiced in a manner aimed at achieving lower and lower doses, even when such practices are known to exacerbate other risks. There is little doubt that ALARA-based interventions lead to risk tradeoffs, yet few of these interventions are analyzed in terms of their net benefit to society.

Risk tradeoff analysis

Risk tradeoff analysis, or RTA, is a comparative risk assessment technique that seeks to identify risk tradeoffs, weigh the comparative importance of target and countervailing risks, and explore opportunities to reduce risk (Graham and Wiener, 1995). Because it is concerned with optimizing health outcomes rather than the financial costs borne by taxpayers, regulated industry, and consumers, RTA is particularly valuable for evaluating the reasonableness of interventions where there is a reluctance to make tradeoffs between monetary costs and health (Graham and Wiener, 1995). As a mechanism for eliminating policy alternatives that are clearly not to society's benefit, RTA is superior to other popular frameworks for comparative risk assessment that rely heavily on cost-effectiveness, such as benefit-cost analysis (Hofstetter et al., 2002).

RTA is practiced by first identifying the goal of the risk-reducing intervention. To "maintain exposures ALARA" is not an explicit enough goal—the target risk and target population must be stated. Where radiation dose limitations are concerned, the target risk is almost exclusively cancer. Target populations are typically workers, members of the public, or both. Radiation protections are sometimes planned to protect hypothetical populations, and the distinction of whether the population is real or theoretical is important to RTA. For example, moving the fence at the boundary of a nuclear facility outward 10 meters could be planned to reduce dose and thus cancer risk to a person who builds and resides in a home just outside the boundary. If no one presently lives at the boundary and it is unknown (or unlikely) that anyone ever will, the protection is based on reducing risk to a theoretical receptor. Workers would be needed to implement the intervention, and these real persons would be subject to risks such as occupational injuries while lifting and moving the fence.

The next step in RTA is to identify any countervailing risk(s). This is not a task that should be taken lightly, because for RTA to be effective, the list of countervailing risks must be thorough. As an example, consider that the NRC applies a dose limit of 0.25 mSv to cleanup of former nuclear sites and that several states use lower limits. It would be easy to conclude that the countervailing risks presented by the states' levels are a bit of extra cleanup and cost, and that these risks do not outweigh the benefits of the dose reduction. A substantive look at the extra cleanup work, however, reveals that it is done by workers who are subject to risk from occupational injuries and illness. Cleanup work often involves degraded structures, heavy equipment and tool use, cranes and rigging, and a variety of other ergonomic and environmental hazards. Wastes that are generated during cleanup are transported for disposal, and extra waste means extra vehicles in transport and an increased risk of traffic accidents. Heavy equipment and the vehicles used to transport waste burn fossil fuels, and their exhausts contribute to health risks from air pollution. If the cleanup is conducted outside, ecosystem disruptions and resulting risks must be considered. Because risks can lead to negative or positive outcomes, a positive outcome that must also be considered is that workers benefit from the extra wages and health insurance they are afforded if the job last longer. All these risks aside, extra cleanup rarely if ever returns the benefit it promises, fewer radiation-induced cancers, because (1) the models used to design cleanup are almost always based on doses to worst-case theoretical receptors rather than to real people and (2) excess cancer incidence is already insignificant or nonexistent at a level of 0.25 Sv.

Once countervailing risks are identified, each is examined in terms of its typology. According to RTA, there are four qualitatively different types of risk tradeoffs: risk offset, risk transfer, risk substitution, and risk transformation (Graham and Wiener, 1995). The typology is determined by deciding if the countervailing risk is the same or different than the target risk and if it impacts the same or different population.

A "risk offset" occurs when the target risk and the countervailing risk are the same and affect the same population. In the case of radiation protection efforts, this would mean that a dose-limiting intervention is expected to cause a certain number of cancers in the same population (e.g., workers) that it was expected to benefit. Any cancer-risk increases offset anticipated reductions. Intrinsically, risk offsets are easier to analyze than other types of risk tradeoffs because the comparisons are said to be "apples to apples". Note the importance of understanding the reality of the impacted population, as preventing five cancers in theoretical workers is not the same as causing five cancers in the actual workforce.

"Risk transfer" happens when the target risk is transferred from one group to another. The risk is the same, so for dose-limiting controls this means cancer risk is reduced in one group and transferred to another. The circumstances surrounding extra cleanup to reduce public dose lead to risk transfer because workers receive extra dose to accomplish such cleanup. Thus, radiation-induced cancer risk is transferred from the public to workers.

Actions taken to prevent cancer through dose reduction can lead to other risks. A “risk substitution” occurs if the population benefiting from the target risk reduction experiences a different risk, and a “risk transformation” occurs when a new group is affected by a different risk than the intervention was meant to address. A worker whose dose is limited by a respirator is subject to risk substitutions from occupational injuries that the respirator could cause, and a public that benefits from cleanup experiences a risk transformation when the vehicles used to transport wastes degrade traffic conditions and air quality.

While knowing the typology of risk tradeoff is important, the true value of RTA is not realized until the benefit and detriment of a proposed intervention are compared in a meaningful manner. This is accomplished by weighing the risks according to magnitude of risk, size of population impacted, certainty of risk estimates, type of adverse outcome, distribution, and timing (Graham and Wiener, 1995).

The magnitude of a risk refers to its probability. Magnitude is a particularly useful comparison tool when the outcomes from the target and countervailing risks are identical, for example fatal cancer versus fatal cancer. A compelling advantage of probability comparisons is that successful decisions are evaluated based on their ability to achieve risk-superior outcomes rather than their ability to drive a target risk down to some arbitrary threshold that may not be significant.

Considering the size of the population exposed to the risk is important in order to avoid transferring risks onto larger populations. For example, if both the target and countervailing risks are 1 in 100,000 chance of lifetime attributable cancer risk, and 100,000 persons are exposed to the target risk whereas 1,000,000 people are exposed to the countervailing risk, a dose-limiting action resolving 1 target cancer could result in 10 unintended cancers.

Certainty of risk estimates is important as some outcomes are consistently and concisely defined while others are speculated or assumed. RTA acknowledges that the degree of scientific uncertainty may differ and recommends quantifying uncertainty with classical statistics when possible. When circumstances cannot be translated objectively it may be necessary to introduce technical judgments from experts. Symmetrical reporting is needed for certainty comparisons. For example, if worst-case estimates are reported for target risk, the same type of estimates should be reported for countervailing risks, otherwise, comparisons are not possible.

Comparing the types of adverse outcomes is possible when risks can be viewed in meaningful, common ways, but such comparisons are often complicated by a lack of precision. They normally involve value judgments rather than objective comparisons along a unidimensional scale. For example, can equal probabilities of death by cancer and death by heart disease be characterized as equivalent outcomes? Is death by cancer worse than living with decades of permanent disability or pain? These are the kinds of value judgments that must be made and qualified when comparing different types of adverse outcomes. Nonetheless, research does demonstrate that humans possess the ability to quantify their dislike for certain equally probable outcomes, and it is possible that as risk analysis measures mature, meaningful comparisons of different outcomes may be possible. A key qualitative difference that is valuable for comparisons is that some risks are felt to be controlled by the individual and voluntarily, while others seem outside of one’s control and involuntary. Risks that arouse fear and risks that explicitly affect children and disadvantaged members of society are also qualitatively different from other risks.

Distributional considerations assess how the risk is shared. Who incurs the risks, are they equally distributed among the population, and is one or more groups disproportionately affected? These and other ethical societal matters are examined when comparing target and countervailing risks along distributional factors. From a distributional viewpoint, risks are generally more tolerable when every person has an equal chance of experiencing the adverse outcome, regardless of any discernable characteristics such as race, age, or socioeconomic status. Political pressures may incentivize governments to trade risk from one group to another; thus, distributional considerations are particularly important when the countervailing risk impacts underrepresented human and nonhuman populations.

Timing of risks is concerned with the ethics of which group should bear the risks and when. This factor considers the time it takes the risk to pose an imminent threat to health. An outcome that presents after a protracted latency may be perceived as a smaller risk than an imminent one; for example, cancer may not express itself for years or decades, whereas trauma is proximately distressing. Timing also considers whether the risks impact current or future generations.

Recognizing and identifying the risk tradeoffs due to dose-limiting actions can be challenging, and the effort is complicated by ethical concerns. RTA cannot be accomplished exclusively with empirical means. Rather, these analyses are informed by a combination of objective information and personal judgment. In some instances, countervailing risks will be determined to exhibit inconsequential or tolerable risks, and in others, the countervailing risk may rival or outweigh the benefits expected from reducing dose. Equipped with knowledge of countervailing risks, a decision-maker is faced with weighing risk versus risk in a manner that best serves the public, with appropriate consideration given to the analysis of experts and what is ethical. In the end, it is hoped that decision makers find opportunities to resolve tradeoffs with risk superior mitigating actions, materials, and technologies. In some instances, this may mean accepting the risk reflected by a low dose of radiation and that further actions to reduce dose are not reasonable.

Moral disengagement theory

If the actual biological effects of low-dose radiation are known to be largely insignificant, and the phenomenon of risk tradeoff and its sources are similarly well understood, then it seems almost inconceivable that radiological control policies and practices would be designed in a manner resulting in other than optimal health outcomes. One would be naive to believe that decision-making practices, particularly decisions leading to policymaking, are based entirely on a factual understanding of risk. That is, while

knowledge certainly contributes to decision making, other important factors such as beliefs, feelings, and perceptions of how others will judge decisions are just as important—or more so—when it comes to decision outcomes. In the end, decision making is a behavior, and like other behaviors, it is influenced by a variety of social-cognitive elements and is happening at a level of consciousness that is not always concerned with factual evidence. Such influences are described and investigated with behavioral models and theories.

Of the tools available for analyzing social behavior related to public health decision making, Bandura's theory of moral disengagement is a popular technique for examining behaviors that lead to detrimental outcomes. The theory seems somewhat misnamed, as it does not stem from a belief that people are bad actors. Rather, the underlying assumption of the theory is that people have an innate desire to make decisions and to otherwise act according to an intrinsic moral code (Bandura, 2002). Because people know right from wrong and are interested in acting in a moral manner, their behavior is expressed via self-regulated processes that prevent violating personally-held moral standards (Bandura, 2002).

When one's actions are contrary to that code, a variety of similarly embedded cognitive tricks are used to justify the behavior. Such tricks allow behaviors leading to detriment to be interpreted as not at odds with one's moral character, including behaviors that could be deemed reprehensible such as cruelty and various forms of physical, social, and cognitive harm to others. We see evidence of moral disengagement in countless everyday situations: a person may overwhelm their children with gifts for the holidays but ignore people next-door who are hungry, a responsible health and safety manager may operate their own automobile at speeds above posted limits or cross the street without waiting for a crosswalk signal, a radiation safety professional may subject workers and the public to unnecessary non-radiological hazards to protect against low-dose radiation. When these behaviors are exhibited by people who seem fundamentally moral, there must be elements underlying the decision-making leading to such outcomes that can be examined and explained.

Moral disengagement theory is characterized by nine constructs that reflect justifying behaviors (Bandura, 2002). "Moral justification" refers to the tendency to consider oneself acting as a social or moral agent and in a way benefiting society. With moral justification, otherwise harmful conduct is accepted because it serves a moral purpose. Via "palliative comparison", harmful conduct is deemed a righteous retaliatory behavior against the outcome it is meant to contradict, prevent, or eliminate. The moral response to harmful conduct can be softened with language skills such as "euphemistic labeling" wherein humans and outcomes are described with sanitizing language such as "targets" and "collateral damage". Euphemistical labeling can also be used to make reprehensible conduct seem to be the work of nameless forces rather than people. A person may minimize their role as an agent in the harm that is caused with "displacement of responsibility", particularly if the decision or action can be attributed to a group. This tactic is used even when the harm comes from a group to which they belong or to claim that they are not responsible for the harm because they were not the one who authorized it. Similarly, "diffusion of responsibility" refers to the justifying behavior through which personal agency is softened by attributing responsibility to other actors as group decision-making provides an opportunity to shirk individual responsibility and blame immoral behavior on others. "Minimizing, ignoring, or misconstruing the consequences" is used to minimize, disregard, or distort the impact of one's actions, "dehumanization" is used to strip away human qualities from people, and "attribution of blame" allows people to view themselves as victims. Self-exoneration is realized in such instances because the harmful conduct is viewed as insignificant, as an act against groups that have not earned our concern, or as a protective response rather than a personal decision that would require moral control. Finally, people are not instantly transformed into cruel actors, and "transformative power of progressive moral disengagement" is the construct by which mildly harmful behaviors become tolerable and slowly progress to more harmful ones.

Just as each of the sources of risk tradeoff is reflected in ALARA practice, actions to limit dose are unmistakably complicated by each of moral disengagement theory's constructs. Because the public perceives radiation to pose such a terrible hazard, moral justification of actions to further limit dose is possible even when such hazards cause other harms. Cancer is considered a terrible outcome, so by palliative comparison, any actions against cancer are moral. The term ALARA was introduced into policymaking as a euphemism to soften the public response to prescriptive dose limits. Displacement and diffusion of responsibility are possible because radiation protection professionals are limiting doses because the rulemaking comes from others and because their department expects them to follow such rules. Radiation protections are known to create other hazards, but these hazards may be ignored or minimized. Regulations provide "occupational dose limits" and "dose limits for individual members of the public" and by separating and dehumanizing workers in this manner it becomes easier to conclude that the public deserves lower limits. Persons living near nuclear facilities may see themselves as victims and blame the operator if they become ill. ALARA will be implemented in an increasingly conservative manner over time as practitioners are praised or rewarded for their efforts, even when they know such efforts may be creating other risks, which demonstrates the transformative power of moral disengagement.

The practice of moral disengagement theory's compensating behaviors occurs instinctively and subtly as a matter of the human cognitive process that has developed over the course of evolution. That is, such behaviors are not constrained to only those persons who are intrinsically cruel. To maximally prevent risk tradeoffs, even the most experienced, knowledgeable, and well-intentioned regulators and practitioners should examine their radiation-protection decision making for evidence of influences leading to moral disengagement.

Conclusion

This article is not an exhaustive examination of the factors leading to risk tradeoff where radiation protections are concerned. Rather, it provides an introductory examination of underlying social, cognitive, and behavioral phenomena that influence decisions related to dose-limiting interventions.

Contrary to public perceptions, low-dose radiation presents little if any risk. Nonetheless, limiting doses with conservative measures such as ALARA pleases the public and provides assurance that excess radiation-induced biological effects are being maximally prevented. If these measures are practiced in a manner that prioritizes radiation protections over other health considerations, however, then such measures are expected to result in risk tradeoffs and possibly harms.

Except in instances where the dose reduction is achieved relatively easily, efforts to make low doses even lower may not be reasonable according to ALARA. That is, as soon as risk tradeoffs become obvious, the intervention is probably not benefiting the public. Regulatory dose limits are already substantively protective, and there is not any evidence of health improvements below these levels. Even the longest-tenured radiation protection professionals will not be convinced that practicing ALARA results in fewer cancers than would be observed by adhering to prescriptive dose limits. Conversely, excess radiation protections are known to contribute to worker injuries, ruin ecosystems, and result in transportation and disposal of vast amounts of relatively innocuous waste. These adverse outcomes are prevented by adhering to prescriptive dose limits without further lowering exposures because of pressure to demonstrate ALARA.

The challenge of making risk informed decisions that consider tradeoffs is that to do so means remaking the paradigm for radiation policy. Omitted voice, heuristics, bounded roles, old technology bias, human behavioral responses, and moral disengagement are pervasive features of the status quo, and it is hard to imagine radiation policymaking occurring without such complications. These complications reflect a systematic failure to see the “whole patient”, where the term “patient” implies the person, population, ecosystem, or combination thereof treated with the intervention. In part, what is needed to remake the policymaking paradigm is to better educate the public on the harmful effects of radiation and the consequences of overregulation. Only by teaming with the public are radiation protection professionals likely to realize optimal public health policy making that takes a whole patient approach where environmental hazards and risk prevention are concerned.

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Computer Codes for Determining Acceptable Levels of Contamination

John Arnish, Knoxville, TN, United States

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Introduction

When determining acceptable levels of residual levels of contamination in the environment, the question “How clean is clean?” while easy to ask, can be difficult to answer. We live in radioactive world where the food we eat, air we breath, water we drink and the ground we live on contains naturally occurring radioactive materials. This is compounded by the fact that radionuclides that emit ionizing radiation are carcinogens, yet at noted above some of these carcinogens occur naturally in our world. For example, potassium is a vital mineral in our diet however, 0.012% of all naturally occurring potassium contains radioactive potassium-40 (^{40}K). ^{40}K decays by emitting a beta particle (ionizing radiation) yet this radionuclide/carcinogen is present naturally in our environment and can be found in the heart healthy fruit, bananas. A potassium-rich diet can help lower blood pressure and people who eat plenty of potassium have a 27% lower risk of heart disease. So if we were to completely remove potassium from our daily lives because of the small amount of ^{40}K there would be an overall health detriment to the population. Therefore, we balance the minute risk of cancer associated with ingesting small amounts of ^{40}K to overall positive benefits of consuming potassium as a whole.

Obviously not all radioactive elements provide health benefits such as potassium, however, we often approach risk assessments in the same manner. Does the health risk of one action provide a benefit/detriment compared to the health risk of another action? As an example, lets say a site is contaminated with a carcinogen, in this example it does not matter if it is radioactive material or not. Obviously, if the cancer risk was such that there was a 100% chance a person would develop cancer while living on the site, there would be no argument that cleanup would be required. So cleanup to what level? The degree to which we must remediate the site goes back to our original question; “How clean is clean?” Our first instinct may be; “Let’s lets clean to zero.” However, that may be cost prohibitive and what is “zero”? Is “zero” our detection limit? What if new technology is developed and a new detector can now measure the carcinogen at a 100-times lower level than older model that was available during remediation activities? Does that mean the site would have undergo remediation yet again? Or, is there an “acceptable” cancer risk, we as a society are willing to accept such as; 1: 10,000, 1:100,000, 1:1 million, 1:10 million, 1:100 million?

In addition, are there competing risks, associated with remediation activities? For example, if we have to remove 1000 times more soil to go from 1:1million risk to 1:5 million risk, that means there are likely going to be additional shipments of soil from the site, which could lead to more fatal transportation accidents associated with these additional shipments, In addition there are risks associated with added exhaust emissions from the heavy equipment operations and offsite transportation activities of the excavated material. At what point do the remediation activities ultimately increase the fatal risk when compared to leaving the carcinogen in place at an “acceptable” level?

The establishment of an “acceptable” dose or risk level is beyond the scope of this chapter. This chapter will focus on the methodologies used to estimate acceptable residual concentration levels *given* the established dose or risk constraints. This chapter will describe the overall methodology used to estimate residual levels of contamination and review common computer programs available to estimate these residual concentration levels.

Radiation dose limits for remediation activities

Given that there is an established radiation dose or risk level involving remediation/decommissioning activities of a contaminated site, how does one implement these dose or risk constraints. For example, in the U.S. Code of Federal Regulation, 10 CFR 20 Subpart E states (NRC, 2018):

A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA).

In addition 10 CFR 20 Subpart E states: (NRC, 2018).

When calculating TEDE to the average member of the critical group the licensee shall determine the peak annual TEDE dose expected within the first 1000 years after decommissioning.

Therefore, in addition to complying with a dose-level at the time of decommissioning it is required, in the case of 10 CFR 20, that this dose-level must be met 1000 years after license termination. It should be further noted that in this case 10 CFR 20 is specific in that it states a "TEDE," which is the Total Effective Dose Equivalent. The TEDE is the dose from external radionuclides plus the "50 year committed effective dose equivalent." The 50 year committed effective dose equivalent is the dose accumulated by the body over 50 years given a single uptake of a radionuclide. For accountability purposes the accumulated dose is "delivered" within a year, even though the actual delivered dose over that time period may be far less.

Consider the notional example: The TEDE for radionuclide ^{123}A is 0.2 mSv. However, the dose may be delivered such that 0.1 mSv is delivered in the first year, 0.05 mSv is derived in the 2nd year, 0.03 mSv is delivered in the 3rd year and 0.01 mSv is delivered in the 4th year and the remainder (0.01 mSv) is delivered over the 5th year. Even though the 0.1 mSv was actually delivered to the person in the first year, the TEDE would be 0.2 mSv in this example.

Since the dose-requirement is in the form of a TEDE, or in the case for U.S. Department of Energy (DOE), DOE Order 458.1 the Total Effective Dose (TED) it is not possible to "measure" the dose during remediation activities (DOE, 2011). While there are plentiful radiation detectors that can measure external dose, there are no detectors available that can measure TEDE or TED. Even if such a detector was somehow magically created, you would need a time machine in order to ensure the TEDE or TED was not exceeded 1000 years post decommissioning in the case of 10 CFR 20 Subpart E.

Therefore, there is a disconnect between the regulatory requirement, which is generally given in terms of TEDE or TED, and the quantity of the contaminant in the environment, which for radionuclides is given in terms of activity or activity per unit mass of the medium that the radionuclide is in.

Converting dose to concentration

Environmental pathways and scenario development

To alleviate this disconnect, it is necessary to convert these dose-based decommissioning criteria to an activity-based criteria. These activity-based criteria are typically measured as a concentration in terms of activity and the mass of the media of interest. For example in soil, the concentration in SI units would be Bq/kg or pCi/g in conventional units.

To perform this conversion it is necessary to estimate the intake of the radionuclide by an individual from inhalation, ingestion as well as estimating the external dose a person would receive. While the external and inhalation pathways are somewhat straightforward, the ingestion pathway can get complicated. Typically the ingestion pathways consider:

- ingestion of plant foods grown on site,
- ingestion meat and poultry products,
- ingestion of milk or other dairy products,
- ingestion water from a nearby well or surface water body,
- ingestion of fish and other aquatic organisms from a surface water body,
- Ingestion of soil.

These additional sub-pathways can get even more complicated? What if the water was used to irrigate the plants, or feed the cattle? These pathways can be represented graphically as those shown in Fig. 1.

In each of these sub-pathways it is necessary to estimate the concentration of the radionuclide in each media. These estimates require the use of mathematical models described by many different variables or input parameters. For example, to estimate the amount of radionuclides taken up by the plant through the roots, one can use a simple plant to soil transfer coefficient although more complex models that take into account the metabolic processes by the plant could be developed. However, to model the amount of radionuclides in the water, more complex models are typically required. Some of these models take into consideration

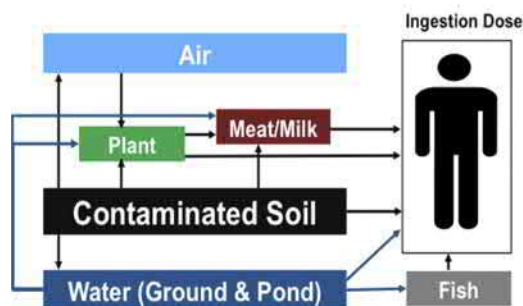


Fig. 1 Graphical representation of the ingestion pathway and a few of its sub-pathways. Yu C., Zielen AJ, Cheng J-J et al. (2001). *Users Manual for RESRASD Version 6*, ANL/EAD-4, Argonne, IL: Environmental Assessment Division, Argonne National Laboratory.

the chemical form of the radionuclide, the distance from the contaminated site to the well, the depth of the well, the total pore space of the soil, the distance to the aquifer, as well as other factors.

In addition to modeling the migration of the radionuclides through the environment, the dose to the individual is ultimately based on the behavior of the person that may be exposed to the contaminated site. It should be obvious that an individual who may live permanently at the site, grows and raises all of their food on the site, obtains their drinking and irrigation water on the site, would receive a larger dose than a person who may only visit the site one or two times a month for recreational activities (hiking, playing sports etc.). The person living on the contaminated area would spend more time on the area compared to a visitor, ingest a greater fraction of their drinking water while living on the site and perhaps consume a greater fraction of their fruits, vegetables, meat and milk products compared to someone living elsewhere.

Therefore, estimating the dose to an individual and thereby determining the ultimate residual contamination levels will be based upon the scenario analyzed and the site-specific attributes of the contaminate and surrounding media. In addition both 10 CFR 20 Subpart E and DOE Order 458.1 specify ALARA (NRC, 2018; DOE, 2011). This has typically been interpreted as, when possible one should perform additional remediation activities if those additional remediation activities do not incur substantial competing risks.

Determining the cleanup criteria

Once the scenario(s) have been selected and the data collected, analyzed and turned into input parameters, the mathematical models are then used to estimate the dose. The estimated dose a receptor receives is linearly related to the concentration of the radionuclide in the media of interest. For example if 3700 Bq/kg of ^{123}A in the soil resulted in a dose of 1 mSv at a particular point in time, then the dose associated with 1850 Bq/kg of ^{123}A would be $\frac{1}{2}$, or 0.5 mSv, at the same point in time given the same scenario and parameter inputs.

As a notional example lets say a dose assessment is conducted on a site once again containing our hypotheical radionuclide ^{123}A . Based on the scenario analyzed and the data collected the maximum annual dose to an individual was 0.2 mSv/yr for a unit concentration of 1 Bq/kg. Using a 0.25 mSv/yr dose constrain, the cleanup criteria would be:

$$\text{Cleanup Criteria} = \frac{0.25 \text{ mSv/yr}}{\frac{0.2 \text{ mSv/yr}}{1 \text{ Bq/kg}}} = 1.25 \frac{\text{Bq}}{\text{kg}} \quad (1)$$

Note that the annual dose cancels leaving only concentration, which can be often times easily measured. So to continue our example if a site was determined to have an average concentration of ^{123}A of 2 Bq/kg then remediation is required. However if a site had an average concentration of ^{123}A of 0.75 Bq/kg then remediation activities may not be required. As noted above the actual cleanup criteria will be based on ALARA as well as any presence of localized elevated radionuclide concentrations. The implementation of radionuclide cleanup criteria is beyond the scope of this chapter.

The general form for the cleanup criteria each radionuclide can be written as:

$$\text{Cleanup Criteria} = \frac{\text{Dose limit}}{\frac{\text{Maximum Dose}}{\text{Initial Concentration}}} \quad (2)$$

As described in the example the maximum dose is often used in the development of cleanup criteria. As noted in 10 CFR 20 Subpart E (NRC, 2018):

When calculating TEDE to the average member of the critical group the licensee shall determine the peak annual TEDE dose expected within the first 1000 years after decommissioning.

While perhaps non-intuitive, the maximum dose may not necessarily occur early on but may occur one hundred years or more into the future. These phenomena may result due to the time required for the radionuclides to transport to the groundwater. For

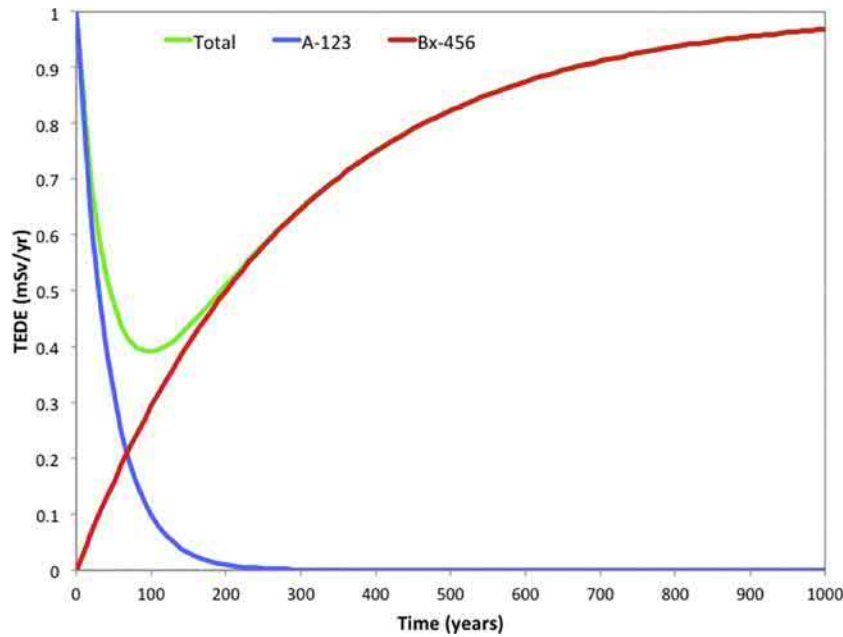


Fig. 2 Notional example of the dose associated with two radionuclides and the total dose over time.

radionuclides with short half-lives, the maximum dose typically occurs early on, however for longer-lived radionuclides the maximum dose may occur later in time.

Consider the notional example where a site is contaminated with our hypothetical radionuclides ^{123}A and ^{456}Bx . As shown in **Fig. 2** ^{123}A reaches its maximum dose at $T = 0$ years while ^{456}Bx has its maximum dose at $T = 1000$ years. In this case the cleanup criteria for ^{123}A would be based on its dose at 0 years while ^{456}Bx would be based on its dose at $T = 1000$ years. Note that in this case the total dose only reaches its maximum at the two extreme timeframes. If a site has many radionuclides present, the time for each radionuclide to reach its maximum dose may differ from each other. The implementation of cleanup criteria based on this situation where the time to the maximum dose is different for different radionuclides can become quite complex.

Assessment tools

Most radiological assessment models contain dozens if not hundreds of input parameters used to describe the transport of the radionuclides through the environment, the scenario used to describe human activities at the site post remediation such as a resident farmer or recreational use scenario, and parameters describing how the radionuclides are brought into and metabolized in the human body. While desirable it is not often possible to have site specific or even scenario specific information available for each input parameter. For input parameters where site-specific information is not readily available most dose assessment professionals will use “conservative” values so that any assessed radiological doses would not be underestimated. However, being too conservative or not collecting site specific data could lead to clean up criteria that is either unattainable or may result in competing risks where the actions of cleaning up a site to extremely low levels may result in greater risks from transportation-related activities.

Fortunately tools have been developed to assist in data collection activities and to determine the importance of the various input parameters. There are two classes of tools

1. deterministic analysis and
2. probabilistic analysis.

Deterministic sensitivity analysis is the simplistic of the two classes. In deterministic sensitivity analysis a parameter or several parameters are increased by a specific value compared to its input value. The same set of parameters are then decreased usually by the same factor and the model is run again. This method of sensitivity analysis is called the direct perturbation method. The dose associated with the upper, base, lower parameter values are compared. If the dose does not change significantly or the time of the peak dose does not change significantly the parameter is not considered sensitive and resources do not have to be spent collecting information on this parameter. However if there are significant changes in the dose then that parameter is considered sensitive and resources should be directed at collecting data on that parameter or perhaps that parameter should be set to a more “conservative” value. Note that this chapter does delve into what is considered “significant.” An example of deterministic sensitivity analysis using the direct perturbation method is shown in **Fig. 3**. Note that the dose changes by a factor of approximately two when the thickness of the contamination is changed by a factor of two.

A drawback of deterministic sensitivity analysis is that it is not possible to investigate the interrelationships among parameters. Some parameters are inherently coupled such as the total porosity of the soil and the soil density. As the total pore space of the

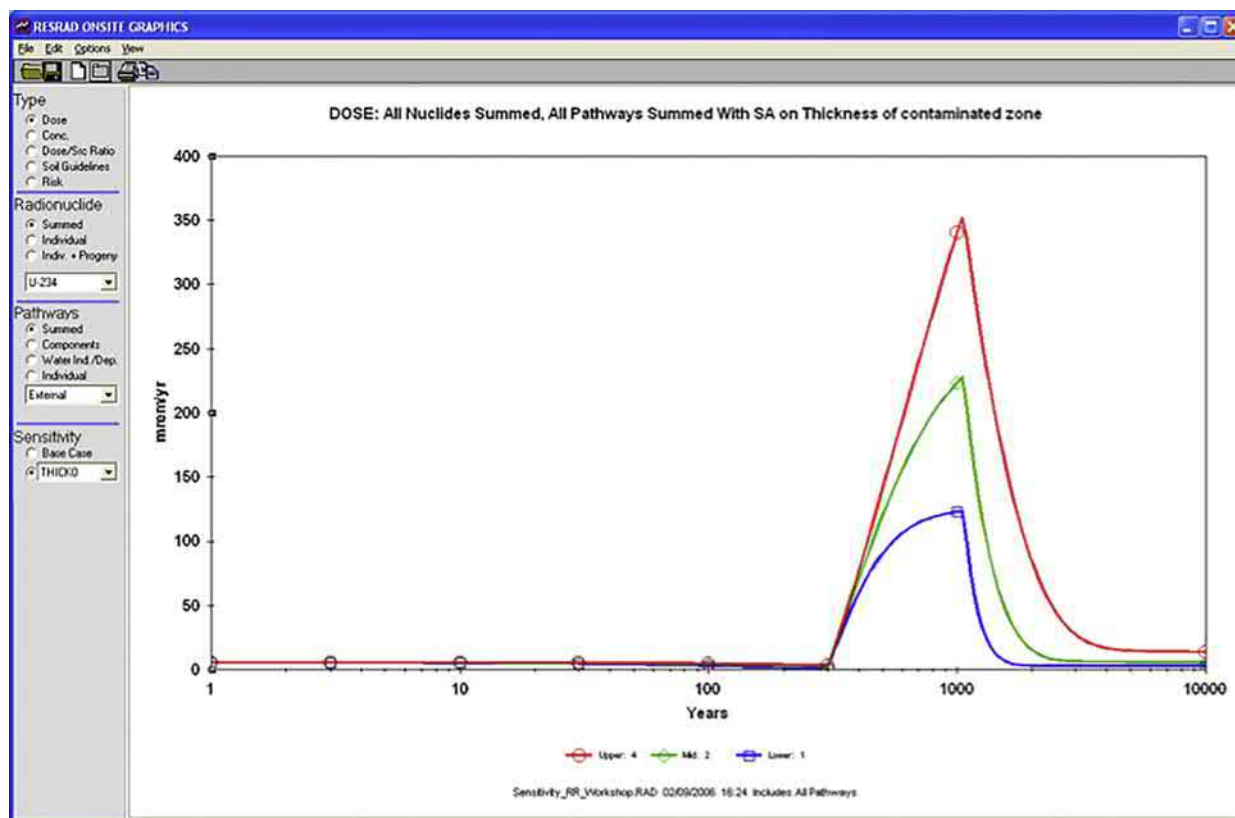


Fig. 3 RESRAD implementation of deterministic sensitivity analysis via the direct perturbation method. Yu C., Zielen AJ, Cheng J-J et al. (2001). *Users Manual for RESRAD Version 6*, ANL/EAD-4, Argonne, IL: Environmental Assessment Division, Argonne National Laboratory.

soil increases, the density of the soil would decrease. While one can perform direct perturbation on both of these parameters it is not possible to see the impact of increasing the total porosity while simultaneously decreasing the soil density. Probabilistic analysis is required to effectively investigate the sensitivity of parameters that are related to other parameters. However, interpreting the results of a probabilistic analysis can be challenging even for the most experienced dose assessment professionals.

In probabilistic analysis each parameter that undergoes analysis is assigned a probability distribution. Furthermore, relationships between the various parameters must be specified. In the total porosity vs density example a negative correlation would be provided, so that as the total porosity increases the density would decrease. The strength of the correlation is provided through the correlation coefficient. A positive correlation coefficient will pair parameter values such that as one parameter increases the other parameter will increase. A negative correlation coefficient will pair low parameter values of one parameter with high parameter values of another. A zero correlation means that the parameters are not correlated whatsoever.

One of the challenges is determining the statistical distribution used to characterize the parameters. While there are documents available that have data specific to certain computer codes, these may not represent the site you are modeling or the computer code you may be using. In addition, these statistical distributions must be numerically approximated so they can be used in mathematical models. The method used to approximate these distributions can result in the generation of hundreds of values thereby requiring hundreds of simulations. Therefore, depending on the complexity of the model and whether the model has been optimized for modern computers, the runtime for a complex probabilistic problem with hundreds of approximations for each input parameter can be excessive.

Another challenge is interpreting the output, of the computer generated of simulations when there are hundreds of results to review and analyze. While there are some statistical tools available, some of these advanced tools may not be intuitive to the casual user. Furthermore these statistical tools have difficulty in analyzing non-linear behaviors which can often occur when the dose is very sensitive to small changes in the parameter value in a small subset of its possible values.

Available computer codes

Computer programs are commercially available to assist in converting dose-limits to activity-based concentrations. The remainder of this chapter will discuss the more common computer programs available to users. The computer programs are used to develop site-specific activity concentrations based on site-specific data. Data collection and analysis is imperative to generate site-specific

cleanup criteria. For example using the wrong data for radionuclide solubility could lead to an over or underestimation of dose especially if the use of on-site water is considered in the scenario. It is up to the user to check the validity of the scenario and the data input into the program.

RESRAD family of computer codes

The RESRAD family of computer codes is developed by Argonne National Laboratory to analyze potential dose to humans and biota from residual radioactivity in the environment. (RESRAD, 2019) The computer codes use a pathways analysis methodology to estimate radiological dose and risk over three major exposure pathways: external pathway, inhalation pathway and the ingestion pathway. The estimated dose when compared to regulatory limits or guidelines can then be used to estimate cleanup criteria for the radionuclides that were analyzed. The RESRAD-ONSITE code was the first member of the RESRAD Family, and was simply known as RESRAD (RESidual RADioactivity) (RESRAD, 2019).

Since the development of the initial RESRAD computer codes, Argonne has developed and maintained RESRAD-OFFSITE, RESRAD-BUILD, RESRAD-RDD and RESRAD-BIOTA (Fig. 4). There are other RESRAD codes that have been developed but are not currently maintained, these include RESRAD-RECYCLE, RESRAD-BASELINE, RESRAD-CHEM and RESRAD-ECORISK.

RESRAD-ONSITE

As stated above, RESRAD-ONSITE or simply RESRAD was the initial member of the RESRAD Family of Computer codes (Yu et al., 2001). The code was initially developed in the late 1980's and has been widely used both in the United States as well as internationally. The code is the only code specifically mentioned in DOE ORDER 458.1 regarding the development of authorized limits (i.e. cleanup criteria) (DOE, 2011). The code, is a scenario driven, pathways analysis program that looks at nine exposure pathways (Fig. 5) that include

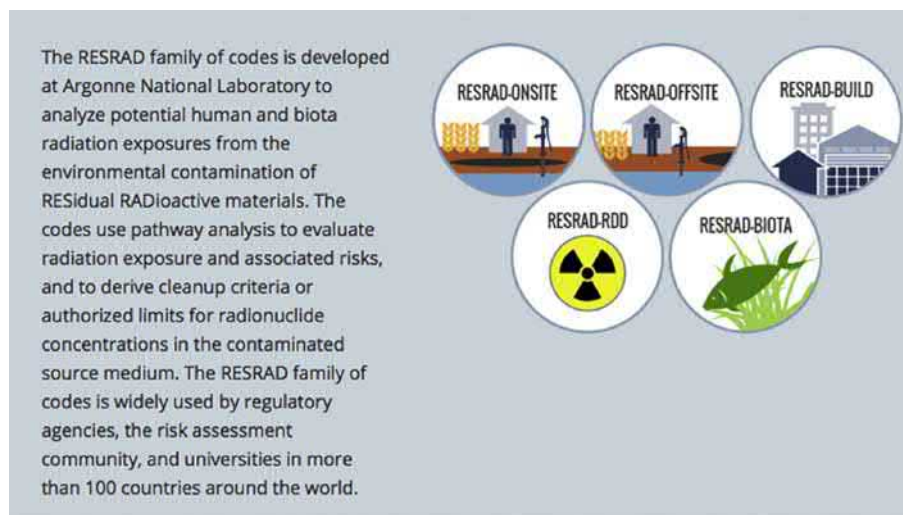


Fig. 4 Current members of the RESRAD Family of Computer Codes. RESRAD (2019) *RESRAD Family of Computer Codes*, <http://resrad.eva.anl.gov>.

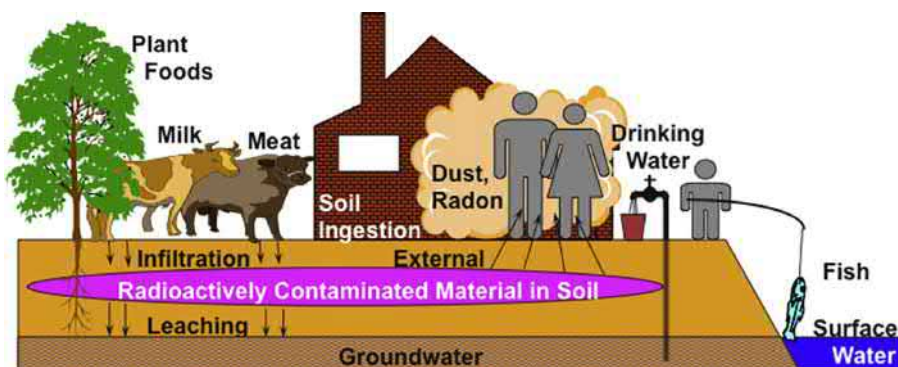


Fig. 5 Graphical Representation of the exposure pathways in RESRAD Yu C., Zielen AJ, Cheng J-J et al. (2001). *Users Manual for RESRAD Version 6*, ANL/EAD-4, Argonne, IL: Environmental Assessment Division, Argonne National Laboratory.

1. direct external radiation,
2. inhalation from airborne radionuclides,
3. incidental soil ingestion,
4. ingestion of plant foods grown in contaminated soil, and irrigated with contaminated water,
5. ingestion of meat produced by livestock fed with contaminated fodder and water,
6. ingestion of milk produced by livestock fed with contaminated fodder and water,
7. ingestion of drinking water from a well or pond adjacent to the contaminated area,
8. ingestion of aquatic foods from the pond, and
9. inhalation of radon emitted from contaminated soil.

The RESRAD code contains features that aid users in developing data sets including sensitivity analysis using the direct perturbation method and uncertainty analysis using standard “Monte Carlo” or Latin Hypercube sampling. The results of the RESRAD simulations are provided in both graphical and standard text outputs. As part of the textual output RESRAD will calculate the individual radionuclide cleanup criteria (soil guideline in the RESRAD summary report) based on a user-defined dose-constraint.

A major benefit of RESRAD is the imbedded features that allow users to perform sensitivity analysis and probabilistic analysis on the input parameters. While it is desirable to have site or scenario specific data associated with each parameter often this is not practiced. Therefore, as mentioned previously, sensitivity and probabilistic analysis is often used to prioritize data collection activities so that adequate resources can be provided on the most important parameters. Data analysis tools exist for both determinist sensitivity analysis and probabilistic analysis.

To help guide the user in developing site-specific parameter values, or to use as a starting point in their analysis, Argonne has an extensive library of supporting documents specific to RESRAD-ONSITE, such as the “Data Collection Handbook, “Development of the Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes (NUREG/CR-6697)” as well as a host of other documents.

RESRAD-OFFSITE

RESRAD-OFFSITE extends the capabilities of the RESRAD-ONSITE computer code (Yu et al., 2007). With RESRAD-OFFSITE the receptors no longer has to be co-located with the contaminated site. This expands radiological assessment capabilities such that the well or off-site water body is no longer directly adjacent to the contamination and could be several hundred meters to several kilometers away. Similarly, the areas where crops and livestock reside may be away from the contaminated area or be partially on the contaminated area (Fig. 6). A Gaussian plume model was incorporated into RESRAD-OFFSITE to calculate air concentrations for an off-site receptor and an accumulation model was developed to account for the buildup of radionuclides associated with off-site deposition, while irrigating with contaminated water.

Additional models above and beyond those implemented in RESRAD were developed for RESRAD-OFFSITE specifically those models that consider the accumulation of radionuclides in uncontaminated areas (Yu et al., 2007). However the situation is more complicated in that in addition to accumulating radionuclides in off-site areas, RESRAD-OFFSITE will also model the release those radionuclides based on the chemical properties of the radionuclides and soil type. For example, RESRAD-OFFSITE will model the accumulation of radionuclides to off-site growing areas through irrigation. Simultaneously, as those radionuclides are accumulating they are also being leached out of the surface soil using similar models as those found for the release of radionuclides in RESRAD (Fig. 7).

Similar to RESRAD-ONSITE the results of the RESRAD-OFFSITE simulations are in the form of both text and graphics display. However unlike RESRAD-ONSITE, the radionuclide cleanup criteria must be calculated outside of RESRAD-OFFSITE using Eq. (2). RESRAD-OFFSITE has the same capabilities for deterministic sensitivity analysis and enhanced probabilistic sensitivity analysis.

RESRAD-BUILD

RESRAD-BUILD is a computer code designed to estimate the dose to persons occupying buildings (Yu et al., 2003). Unlike standard pathway analysis tools that model the environmental transport of radionuclides from the soil to the food products, RESRAD-BUILD models the interior movement of radionuclides from room to room by a series of compartment models. The external dose is modeled based on the location of the individual relative to the radiological sources. Unlike RESRAD-ONSITE or RESRAD-OFFSITE there where only one source is modeled at a time, up to 10 sources and 10 receptors can be modeled in a single simulation. The sources include, point, area and volumetric sources. Sources can be co-located such that a point source can be collocated on top of an area source to model “hot spots” or an area source can be modeled on top of a volume source to represent surface contamination on top of volumetric contamination. Since the user can specify up to 10 receptors, the program is flexible such that the receptors can be a single individual located at various spots within a building to estimate an annual individual dose, or several receptors inside a building to estimate a collective dose.

The user must specify air exchange rates between the rooms and with the outside. The underlying assumption is that the outside air is not contaminated. RESRAD-BUILD calculates airflow based on the room height, room area and airflow between rooms. For example a two story house with a basement could be modeled by setting the airflow with the basement to the outside as $0 \text{ m}^3/\text{h}$ (Fig. 8).

The total dose to the receptor(s) is a linear combination of the radiation dose associated with each source. Since the dose to the receptor is based on the total amount of time present at that location, the units for dose are in mSv as opposed to mSv/yr. However if the user is modeling a situation where the receptor is to represent a person exposed for a year, the dose can be interpreted as mSv/yr.

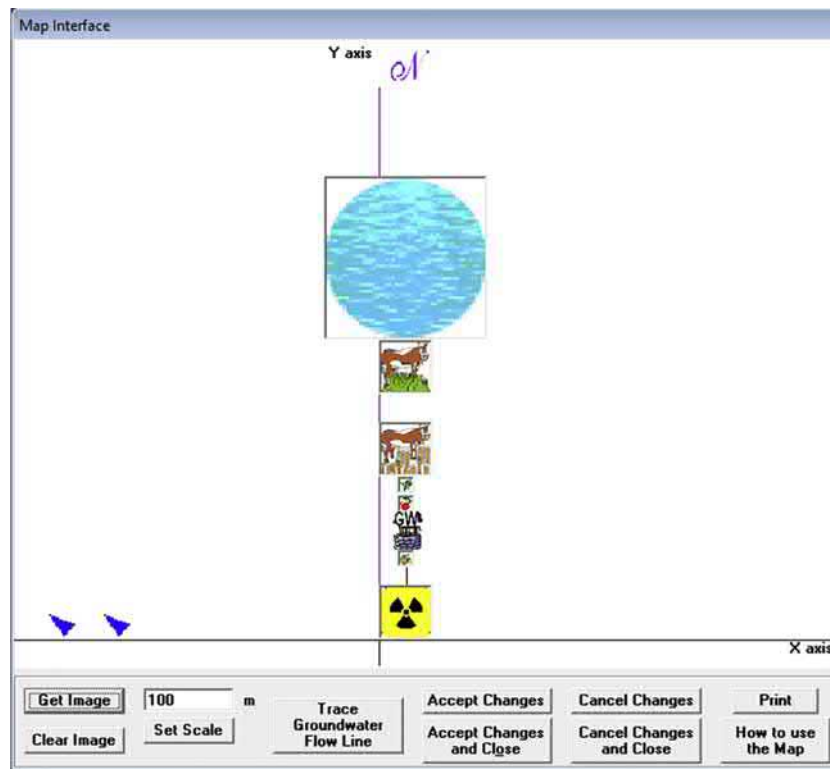


Fig. 6 Map interface for RESRAD-OFFSITE. The size of the icons scale with area, in this case the pond is further from the contamination area. All residential and growing areas are away from the contaminated zone and would rely on the accumulation models built into the program to calculate concentration and therefore dose. Yu C, Gnanapragasam E, Biwer BM, Kamboj S, et al. (2007) *User's Manual for RESRAD-OFFSITE Version 2*, ANL/EVS/TM/07-1, DOE/HS-0005, NUREG/CR-6937, Argonne, IL: Environmental Science Division, Argonne National Laboratory.

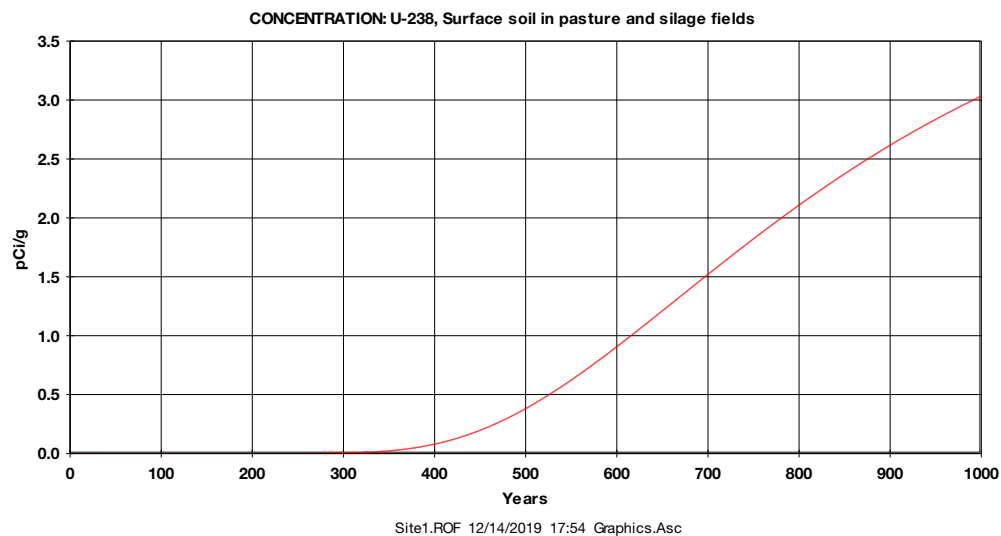


Fig. 7 Radionuclide concentration versus time for an off-site pasture due to irrigation. At time $T = 0$ to approximately $T = 300$ years there is no contamination present. After $T = 300$ years (the amount of time required for the radionuclides to reach the well in this instance) radionuclide concentration steadily increases. Yu C, Gnanapragasam E, Biwer BM, Kamboj S, et al. (2007) *User's Manual for RESRAD-OFFSITE Version 2*, ANL/EVS/TM/07-1, DOE/HS-0005, NUREG/CR-6937, Argonne, IL: Environmental Science Division, Argonne National Laboratory.

Since RESRAD-BUILD is flexible in its capability to model either individual or collective doses for multiple sources, the program does not calculate cleanup criteria like the RESRAD computer code. Therefore, cleanup criteria must be developed for each source/receptor combination using a Eq. (2). The units associated with the cleanup criteria are dependent on the type of source (Bq for a point source, Bq/m² for an area source and Bq/kg for a volume source).

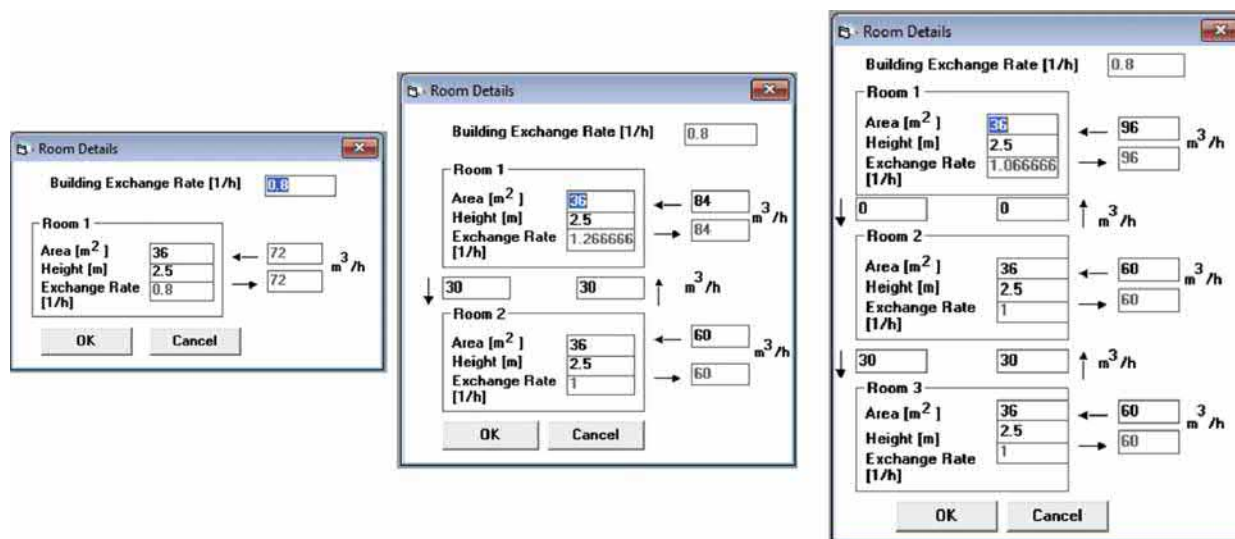


Fig. 8 One, Two, and Three room air models in RESRAD-BUILD. Yu C, LePoire DJ, Cheng J-J et al. (2003). *User's Manual for RESRAD-Build Version 3*, ANL/EAD/03-1, Argonne, IL: Environmental Assessment Division, Argonne National Laboratory.

DandD computer program

The DandD computer code is an environmental pathways analysis program developed by Sandia National Laboratories to implement the methodology found in NUREG/CR-5512 Volume 1 Residual Radioactive Contamination From Decommissioning, Technical Basis for Translating Contamination Level to Annual Total Effective Dose Equivalent (Kennedy and Strenge, 1992; McFadden et al., 2001). Specifically the methodology was developed for U.S. NRC licensees who needed to decontaminate lands and structures to acquire unrestricted use of their properties based on a 0.25 mSv/yr public dose limit. The program has two built in scenarios, the "Residential" and "Building Occupancy." "DandD Version 2.1 incorporates probabilistic analysis into the two scenarios (McFadden et al., 2001).

The "Residential" scenario models the a site with surface soil contaminated soil and considers the following pathways:

1. direct external radiation,
2. inhalation from airborne radionuclides,
3. incidental soil ingestion,
4. ingestion of plant foods grown in contaminated soil, and irrigated with contaminated water from a well,
5. ingestion of beef produced by livestock fed with contaminated forage, stored grain, stored hay and water from a well,
6. ingestion of milk produced by livestock fed with contaminated forage, stored grain, stored hay and water from a well,
7. ingestion of poultry fed with contaminated forage, stored grain, and water from a well,
8. ingestion of eggs where the poultry is fed with contaminated forage, stored grain, and water from a well, and
9. ingestion of drinking water from an on-site well.

The "Building Occupancy Scenario" models surface contamination within a building. The exposure pathways include

1. direct external radiation
2. inhalation from resuspended surface contamination
3. inadvertent ingestion of surface contamination
4. external exposure from submersion of dust particles
5. internal contamination from puncture wounds
6. dermal absorption and
7. inhalation of radon.

While DandD is a pathways analysis model, and are conceptually appear the same as those in RESRAD and RESRAD-OFFSITE, the models in the DandD computer program are less complex than those in RESRAD-OFFSITE, RESRAD and RESRAD-BUILD. These simplifications may lead to an overall larger dose to a receptor for similar exposure conditions. For example RESRAD and RESRAD-OFFSITE allow up to five distinct unsaturated layers below the contaminated zone whereas the DandD code only allows for a single unsaturated layer. In addition the user may specify that the contamination layer is buried below clean cover in RESRAD and RESRAD-OFFSITE whereas the DandD program only calculates surface soil contamination for the "Residential" scenario.

Regardless of the computer codes used for radiological assessments, it is important that users understand the limitations of the program and the inherent assumptions built into the models. For example while it would appear that RESRAD could be used to

directly model an engineered landfill, this is not the case due to the complexities involved in the various engineered barriers encounter in a disposal facility. In this instance RESRAD-OFFSITE may be more applicable due to the availability of multiple types of leaching models built into the program.

Other computer programs

The computer programs discussed previously assumed that soil was initially contaminated by some means. The manner by which the soil was contaminated is beyond the scope of the computer models. However, it may be necessary to perform radiological assessments for working facilities or to perform a dose reconstruction after an event such as an unexpected airborne release. We will discuss two programs CAP-88 PC and GENII, keep in mind that other site-specific models may be developed for large or high-profile public releases of radioactive material.

CAP-88 PC

The CAP-88 (Clean Air Act Assessment Package - 1988) computer model is a set of computer programs, databases and associated utility programs for estimating dose and risk. (EPA, 2019) CAP-88 is a regulatory compliance tool under the National Emissions Standard for Hazardous Air Pollutants (NESHAP). Dose and risk estimates from CAP-88 PC are applicable only to low-level chronic exposures, since the health effects and dosimetric data are based on low-level chronic intakes. CAP-88 PC cannot be used for either short-term or high-level radionuclide intakes.

CAP-88 uses a modified Gaussian plume model to estimate dispersion and ultimately air and ground concentrations from six sources (EPA, 2019). The sources can be modeled as elevated releases (stacks) or ground level releases such as piles of contaminated soil. Radiological risk assessments are performed for a circular grid of 16 sectors by setting distance and direction within a radius of 80 km (50 miles) around the facility. The pathways considered in CAP-88 include:

1. inhalation of airborne radionuclides,
2. inhalation from radionuclides attached to resuspended dust,
3. ingestion of leafy vegetables,
4. ingestion of milk,
5. ingestion of meat products,
6. external dose due to deposition,
7. external dose due to air submersion.

Note that there is no drinking water pathway or water pathways associated with irrigation. The dose and risk estimated using CAP-88 is based on annual radionuclide releases to the air and therefor the doses associated with ground water associated with radionuclide transport through the underlying soil would be considered a minor component when compared to the primary airborne release from a facility.

GENII

The GENII computer program is a suite of computer models integrated through the FRAMES software package (Napier et al., 2012). There are separate models for airborne and waterborne releases. The airborne release is similar to that of CAP88 however in addition to estimating doses and risks from routine normal operations, the GENII atmospheric transport model can estimate radiological dose and risk from acute releases from accidents. For the atmospheric transport model, GENII uses both a Gaussian plum and Lagrangian puff model depending if the release is either a chronic or an acute release.

The surface water module is included to represent transport through surface waters to the terrestrial environment. (Napier et al., 2012) The models include acute and chronic releases to non-tidal rivers and chronic releases to near shore lake environments. The mechanism of how the radionuclides are released to these media is beyond the scope of GENII. The user must specify release rates of the radionuclides to the air and water modules.

The atmospheric module and surface water modules are then coupled to the receptor intake module and health impact module to estimate dose and risk. Therefore depending on the problem set analyzed (airborne release, water release, or both) the pathways analyzed include (Napier et al., 2012):

1. External exposure
 - a. due to plume immersion
 - b. ground deposition
 - c. while swimming
 - d. exposure while boating
 - e. exposure at the shoreline
2. Inhalation
 - a. Contaminated air
 - b. Resuspended radionuclides on dust
 - c. Indoor air

3. Ingestion
 - a. Terrestrial farm product
 - b. Aquatic food
 - c. Drinking water
 - d. Inadvertent soil ingestion
 - e. Inadvertent water ingestion while swimming
 - f. Inadvertent ingestion while showering

Conclusion

Computer models are required to demonstrate compliance with applicable regulatory requirements for the exposure to members of the public from contaminated sites. Regulations are typically written in terms of dose or annual dose-rate to a member of the public. Radionuclide concentrations are typically measured prior to and during remediation activities, therefore a relationship between concentration and dose must be established. This is compounded by the fact that many times the regulations specify annual dose limits out into the future such as 1000 years.

There have been several computer programs written to estimate radiation dose to members of the public from contaminated land. The most prevalent is the RESRAD Family of Computer Codes. This suite of computer codes allows users to either estimate dose or clean up criteria associated with radiologically contaminated sites. Depending on the receptor location, whether on-site, off-site or within a building, there are individual computer programs available for each application. However, there are other codes available such as the DandD computer program, which is similar to RESRAD, as well as the CAP-88 and GENII computer programs. Unlike RESRAD and DandD, CAP-88 and GENII are typically used to estimate doses associated with chronic (CAP-88 and GENII) or acute releases (GENII) and are not typically used for environmental remediation activities.

It is up to the user to determine the applicable computer programs. Typically all radiological dose assessment programs estimate radiological dose and/or risk associated the main exposure pathways (external, inhalation and ingestion) however, the manner in which they estimate these doses and the underlying models and assumptions that go into the programs can be quite different.

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Characterization and Release of Contaminated Sites

Timothy J. Vitkus, Oak Ridge Associated Universities (ORAU), Oak Ridge, TN, United States

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Abbreviations

AA	Alternative action
ANSI	American National Standards Institute
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CSM	Conceptual site model
D&D	Decontamination and decommissioning
DCGL	Derived concentration guideline limit
DOD	Department of Defense
DOT	Department of Transportation
dpm/100 cm ²	Disintegration per minute per 100 square centimeters
DQA	Data quality assessment
DQO	Data quality objectives
EMC	Elevated measurement comparison
FSS	Final status survey
FUSRAP	Formerly Utilized Sites Remedial Action Program
HSA	Historical site assessment
H ₀	Null hypothesis
H _A	Alternative hypothesis
HTD	Hard-to-detect
IAEA	International Atomic Energy Agency
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MOU	Memorandum of understanding
NRC	U.S. Nuclear Regulatory Agency
pCi/g	picoCuries per gram
PNNL	Pacific Northwest National Laboratory
PSQ	Principal study question
RASS	Remedial action support survey
ROC	Radionuclide of concern
RSSI	Radiological survey and site investigation

SI International System of Units
 UCL Upper confidence limit
 VSP Visual Sample Plan
 WAC Waste acceptance criteria

Introduction

Sites involved in the nuclear fuel cycle and nuclear power generation, legacy sites from atomic weapons development and production, nuclear research in both private sector and national laboratories, specialized industrial products, and others are required to operate in accordance with radiological protection standards, licensing or other regulatory requirements in effect at the times of operation. When the facility concludes its mission, the cessation of activities under typical scenarios result in the facility seeking release from radiological and regulatory control where the desire may be to sell or otherwise transfer the property for future use without radiological restrictions. However, because of the historical radioactive material use, the default assumption is that these materials impacted (contaminated) the facilities themselves—i.e. structures, equipment, systems—as well as the external environment.

Therefore, prior to a site release being authorized, the status and regulatory requirements migrate from operations into decontamination and decommissioning (D&D). The ultimate objective of D&D is to either clean up residual radiological contamination or otherwise confidently prove that any residual contamination is less than the applicable criteria that must be met to release the site without radiological restrictions. When the criteria cannot be met, the site may either have a conditional release that includes some type of institutional controls or restrictions on future use, or will remain under regulatory control.

Prior to the cognizant regulatory authority granting the site release, the site must provide adequate documentation showing that the facility satisfies the release requirements. The release may be authorized in the form of license termination for those facilities licensed by the U.S. Nuclear Regulatory Agency (NRC) or an Agreement State. An Agreement State, per the NRC, is a state that assumes NRC regulatory authority under the Atomic Energy Act of 1954, as amended and provides a statutory basis under which NRC relinquishes to the States portions of its regulatory authority to license and regulate byproduct materials (radioisotopes); source materials (uranium and thorium); and certain quantities of special nuclear materials. In addition to State or Federally licensed private sector sites, U.S. Department of Defense (DOD) facilities in many cases also operated under NRC licensing while U.S. Department of Energy (DOE) sites are regulated within the Department itself. State agency stakeholders, although not the regulator of record, are also generally involved with DOE and DOD sites within their borders that are undergoing decommissioning. Furthermore, if the site is within the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), the U.S. Environmental Protection Agency (EPA) is also involved in oversight.

Historically, the guidelines to demonstrate that a facility could be released to the public or other uses were based on the sensitivity of the instrumentation available at the time to detect and quantify the radiation of concern. The NRC issued a new license termination rule in 1997 that revised their licensee's requirement for license termination from meeting a table of residual contamination values to a dose based limit of 25 millirem per year (0.25 milliSieverts). With this change, the D&D requirements and cleanup guidelines have since become more advanced, most notably in how the cleanup criteria are derived, where they are now derived using either default or site-specific inputs to satisfy the annual dose limits or other decommissioning criteria. The cleanup criteria are referred to as derived concentration guideline limits (DCGLs). The NRC published screening level DCGLs (NRC, 2006) for select radionuclides for allowable residual activity on building surfaces in units of disintegration per minute per 100 square centimeters (dpm/100 cm²) and for soil in picocuries per gram (pCi/g) (Note: SI equivalents can only be provided for a specific DCGL value.) The DCGLs—actually denoted as the DCGL_W—are the allowable average concentration of each contaminant within the survey unit. Because it is an average value, some individual samples may exceed the DCGL_W, which is acceptable in many cases, provided the average including hot spot contributions is less than the DCGL_W. Additional information regarding hot spot requirements are provided later in this chapter.

Agreement State dose limits are at least equivalent to NRC's or more restrictive, with some states essentially requiring no dose in excess of background. The DOE Order 458.1 (DOE, 2011) provides requirements for the development of site-specific authorized limits or application of pre-approved limits also in terms of dpm/100 cm² and pCi/g. The DOE limits differ from those of the NRC's and are based on DOE-specific requirements for radiological protection of the public and environment. The EPA requires their own risk based limits which results in different release criteria from those of the NRC. Because EPA may also have regulatory involvement with some NRC licensee's there can be conflicting requirements the site must meet. To minimize these conflicts, a memorandum of understanding (MOU) was established between the two organizations. Similar guidelines exist for other countries, where many countries have adopted the International Atomic Energy Agency's (IAEA) standards for clearance or exemption of materials (IAEA, 2005).

A major component of the decommissioning is the remediation of radioactive contamination to reduce contamination levels within structures equipment, and impacted environmental media until radiological measurements and samples confidently demonstrate that applicable cleanup criteria have been met. However, prior to commencing remediation, radiological investigations are performed to identify where contamination is present, the levels of contamination, and the media impacted.

The principal steps in the D&D radiological survey and site investigation (RSSI) life cycle of a contaminated site is best described by the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NRC, 2001). The investigation steps begin with the historical site assessment (HSA), scoping survey, and characterization survey to identify contaminated areas. Site remediation, when required, follows the characterization survey. The remedial actions will involve processes such as soil excavation, scabbling of concrete surfaces, or other methods. When decontamination is thought to be adequate, a remedial action support survey (RASS) is performed that involves a few qualitative or quantitative measurements. If the RASS investigation indicates successful decontamination, then the final status survey (FSS) is performed and documented. The FSS is formally planned and implemented to satisfy regulatory guidance. The results are documented in a report and serve as the primary evidence submitted to the regulatory authority demonstrating that the decommissioning requirements for license termination or site release have been met. The regulator then determines whether all conditions are satisfied and the site may be released. As part of the release decision, the regulator may elect to perform an independent confirmatory survey to provide additional assurance that results are accurate and reproducible. The RSSI process is illustrated in Table 1.

Further discussion of each investigation is provided below, and includes methods generally included in the design, implementation, and assessment of each.

Historical site assessment (HSA)

A site identified as potentially contaminated begins the RSSI process with an HSA. The HSA is not an actual survey, but rather is thorough records review to gather and review all the information relevant to the site's radiological status. Information reviewed might include the following:

- operational records and data
- interviews with current and previous employees
- aerial photographs
- licensing documents
- facility maintenance or remodeling records
- waste disposal records and material handling routes
- records of spills or effluent releases
- routine survey/environmental monitoring data

Table 1 Principal steps in the D&D RSSI lifecycle.

RSSI Phase	Description
Historical Site Assessment ↓	Document available information to determine if the site is impacted
Scoping Survey ↓	Describe limited confirmation on presence, magnitude, and location of contamination
Characterization Survey ↓	Define the nature and extent of contamination and support remediation planning
Remedial Action Support Survey ↓	Determine if the site is ready for final status surveys; provide inputs for final status planning
Final Status Survey ↓	Demonstrates compliance with the remedial action objectives
Confirmatory Survey	Quality assurance investigation performed by or on the behalf of the regulatory agency

The HSA goal is to paint a complete picture as possible in the form of a conceptual site model (CSM), identifying the different portions of the site that could be impacted by radioactive material from historical site operations and waste disposal practices. If possible at this early stage, the CSM includes preliminary contamination potential classifications for impacted areas. The definitions from greatest to least contamination potential are:

Class 1: potential to contain or at one time contained contamination in excess of the DCGL_W;

Class 2: potential to contain contamination but at concentrations less than the DCGL_W.

Class 3: potential to contain contamination, but at a small fraction of the DCGL_W.

Fig. 1 shows an aerial photograph of a site that was involved in storing radioactive material in warehouses. The second image is a graphic of the site plot plan showing how the land areas and structures were preliminarily classified following the HSA.

A primary reason for classification is to provide a mechanism for determining where future resources would be expended. Areas with the greatest contamination potential will receive the greatest effort. If there is insufficient information, all areas may have to be designated as Class 1 until adequate data are obtained. Other HSA objectives are to:

- major contaminants and relative ratios among multiple contaminants (for DCGL_{Ws})
- contaminants chemical/physical form (for DCGL_{Ws})
- time since operations and material use (short-lived radioactive material decayed and is no longer a potential contaminant or for progeny in-growth for chemically separated radionuclides such as uranium or thorium)
- identify potential contaminated media:
 - Surface soil
 - Subsurface soil
 - Building surfaces
 - Process equipment
 - Support equipment
 - Other media

An alternative to an impacted classification is when an area might be considered non-impacted. Non-impacted areas have no reasonable potential for contamination, and therefore do not require further investigation to support release. In reality, stakeholders are many times hesitant to accept the non-impacted status without at least some data to support the decision. Therefore, many D&D sites perform a limited scope survey.

It is expected at the completion of the HSA that there will knowledge gaps requiring further investigation. The conclusions reached and identified gaps serve as inputs to planning the next phase, the scoping survey.

Scoping survey

The scoping survey objectives are to validate, refine, or refute the preliminary site status from the HSA. Where the HSA was a paper-work exercise, the scoping survey requires field work across the site. The level of effort or the focus given to each area is based on a graded approach. The graded approach simply means that the most survey and sampling effort is expended toward those areas with the greatest potential for contamination, areas with the most unknowns or data gaps, or a combination thereof.

The scoping survey design consists of cursory radiological surveys, primarily qualitative scan surveys, and the collection of a limited number of judgmental measurements or samples, although under certain circumstances more rigorous, statistically-based scoping surveys may be planned. The data collected serve to confirm the impacted or non-impacted status, and in impacted areas, confirm types, levels, and variability of contaminants present and substantiate area classifications. Other scoping survey objectives might include hazard assessments or identifying background reference areas.

The instrumentation used for the scoping survey are selected based on the radiation emissions of the contaminants. For sites with gamma-emitting radionuclides of concern (ROC), gamma walkover scans of land areas and building surface scans are most commonly performed using sodium iodide (NaI) scintillator detectors. **Fig. 2** shows surveyors performing gamma walkover scans with NaI detectors coupled to global positioning system (GPS) equipment. The surveyor listens via the audio output for elevated radiation count rates. The GPS equipment continuously captures both the count rate and geo-referenced position data for spatial evaluation. The GPS and radiation detector systems mounted are also commonly mounted onto vehicles when surveys of large land areas are required.

Building surfaces are also scanned for alpha and/or beta radiation based upon the suspected ROCs. The main detector types for alpha scanning of surfaces, are gas-flow proportional counters and zinc sulfide (ZnS) scintillators. Gas-flow proportional, pancake Geiger-Mueller (GM), or plastic scintillator detection systems are used for beta scans.

As indicated, the land and building surface scans are intended to identify radiation anomalies. When suspect contamination is identified on a building surface, an alpha/beta surface activity measurement or sample for laboratory analysis is collected. Similarly, a soil sample is collected from land area anomalies. These samples and measurements that are based on scan results are examples of judgmental sampling. That is, based on experience, professional judgment, or preliminary results, the surveyor goes to areas expected to have the highest likelihood of contamination. Some random sampling may also occur during scoping.

How the graded approach is applied to determine scan coverage for land or building surfaces or an appropriate number of judgmental and/or random samples or measurements provides the opportunity to introduce the concept of data quality objectives

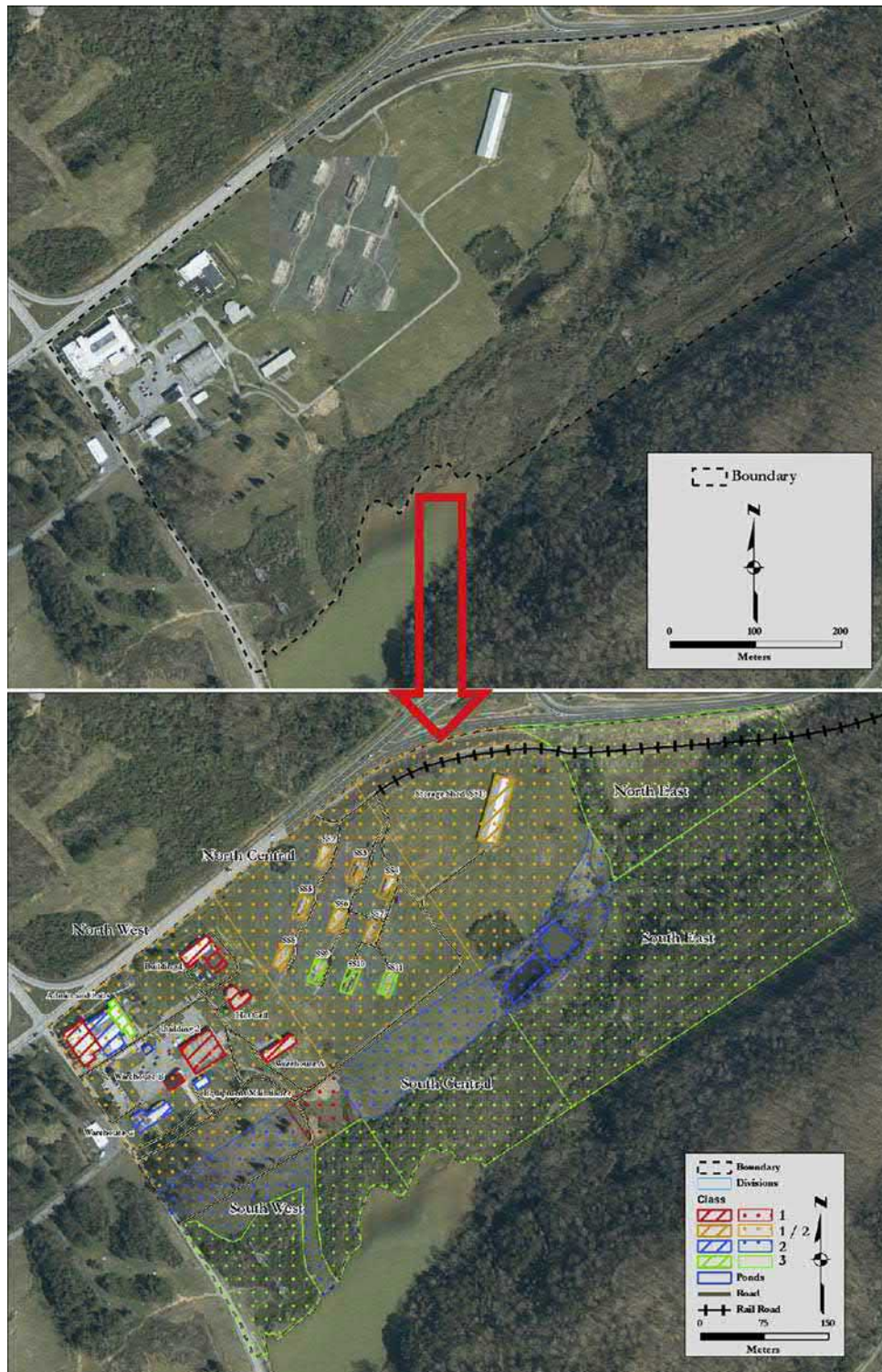


Fig. 1 Aerial site view and conceptual site model plot plan showing site classification.

(DQOs). DQOs are a critical component of the MARSSIM guidance for planning final status surveys. The application of DQOs throughout the RSSI process help ensure the right quantity and quality of data are collected for decision making. EPA provides additional information on DQOs in an excellent environmental data collection guidance document (EPA, 2006). The next section further explains DQOs.



Fig. 2 Surveyors performing gamma radiation walkover scans. From: <https://www.ornl.gov/environmental-assessments/index.html>.

This scoping survey discussion concludes by introducing the data quality assessment (DQA) phase. There may be multiple data objectives, dependent upon the DQOs. Again, one of the primary scoping survey objectives is to confirm the HSA's preliminary classification of areas, or otherwise modify/refine the CSM. The absence of contamination in collected data may be evidence that the area is indeed Class 3 or should be reclassified as Class 3. Under certain scenarios, when sufficient information is available for a Class 3, the scoping survey may be designed to also satisfy the FSS. In other words, skip ahead from the HSA directly to the FSS. This is only advisable when confident in the classification and instrumentation and procedures satisfy FSS requirements. Alternatively, contamination is confirmed or otherwise identified and the area is reclassified as Class 1 or 2.

Data quality objectives

The seven-step DQO process helps ensure that the type, quantity, and quality of the survey data used in decision-making are appropriate for its intended use. The DQO steps are:

- Step 1. State the problem.
- Step 2. Identify the decision.
- Step 3. Identify inputs to the decision.
- Step 4. Define the study boundaries.
- Step 5. Develop a decision rule.
- Step 6. Specify limits on decision errors.
- Step 7. Optimize the design for obtaining data.

Table 2 lists each step, what the step involves and provides a simplified approach using a scoping survey scenario for investigating a preliminary classification for a site area as Class 3.

Characterization surveys

Site characterization can be complex and because sites and therefore study objectives are unique, there is no single method upon which to base a default process. Because of this, DQO application is critical. The American National Standards Institute (ANSI) and the Health Physics Society developed an industry standard for DQO application to characterization. According to both MARSSIM and ANSI N13.59, the characterization survey is the most comprehensive of all survey types and generates the most data (ANSI, 2008; NRC, 2001).

The amount of data needed is because characterization survey data serve many uses beyond just defining the nature and extent of contamination. Some primary characterization data needs are shown in **Fig. 3**. In addition to determining location of contamination, total contaminated surface area or waste volumes, the data often are used in pathway analysis inputs and derived concentration guideline level (DCGL_W) development, remedial action planning support, and even determining the site's final status.

A summary-level discussion is provided here for the characterization purposes with the exception of pathway analysis and DCGL_W calculations addressed in other chapters.

Table 2 DQO steps, description, and examples.

<i>DQO step</i>	<i>DQO step description</i>	<i>Specific example</i>
Step 1. State the problem	Resources also are listed (i.e., the budget, schedule, and personnel) and the conceptual site model (CSM) is presented. However, the main objective of Step 1 is to provide a concise description of the problem	One portion of the entire site that was classified as Class 3 following the HSA Problem Statement: A scoping survey is necessary to validate that HSA that the area is Class 3 and contains contamination at no more than a small fraction ($<10\%$) of the $DCGL_W$
Step 2. Identify the decision	The goal of Step 2 is to define the principal study question (PSQ) that the project will attempt to resolve, identify alternative actions (AAs) that may result from answering the PSQ, and combine the PQS and AAs into a decision statement	PSQ: Are contamination levels less than 10% of the $DCGL_W$? AA1: Yes, maintain the Class 3 designation AA2: No, reclassify to Class 2 or Class 1 based on observed levels Decision Statement: The survey will be performed and the data used to determine the area classification
Step 3. Identify inputs to the decision	The purpose of Step 3 is to identify the information that is required to address each decision statement. Sources of information also are identified	Only HSA records, a screening level $DCGL_W$ are available as inputs Radiological measurements and analytical data for samples are to be generated
Step 4. Define the study boundaries	Step 4 presents spatial, temporal, and practical boundaries to the problem set	The spatial boundaries are the current subdivision of the Class 3 area The temporal boundary is the time period over which the survey must be completed and the data assessed
Step 5. Develop a decision rule	Step 5 provides the specific parameters that will be obtained and how those data are used in making the decision	Decision Rule: If all sample radionuclide concentrations are less than 10% of $DCGL_W$ maintain Class 3 designation, if concentrations are $>10\%$ and $<DCGL_W$ reclassify as Class 2, and if $>DCGL_W$ reclassify as Class 1
Step 6. Specify limits on decision errors	Step 6 presents the basis for saying "how good" the decisions need to be by presenting the null hypothesis and tolerable limits on decision errors	This could include a formal hypothesis statement. With hypothesis testing or any decision there is a chance of an incorrect decision. This example's assumed base condition is a Class 3 area, and evidence is required to overturn that condition and decide the area is Class 2 or 1. The Type I error, or false positive represents the decision confidence. 95% confidence means that there is a 5% chance probability of incorrectly concluding the example area is Class 2 or 1 when it should have remained a Class 3. The Type II error, or false negative, represents the probability of failing to conclude the area is Class 2 or 1 when it should be
Step 7. Optimize the design for obtaining data	Step 7 is used to evaluate Steps 1 through 6 and consider ways to optimize the project design. This step typically specifies the number, location, and type of information required for decisions	Decide investigation coverage, determine sample locations, specify instrumentation, procedures, laboratory analyses and other applicable methods

Define the nature and extent of contamination

The characterization survey fills HSA and scoping survey data gaps to describe the primary contamination. Without this knowledge, remediation planning may be inaccurate, leading to poor decisions, schedule delays, and rework. A proper characterization effort will:

- Identify all radiological contaminants, including any hard-to-detect (HTD) radionuclides mentioned (or not) in the HSA/scoping survey
- Estimate the statistical characteristics (range, average, etc.) of contaminants
- Establish relationships, if any, between contaminant concentrations
- Describe the physical form and distribution of the contamination, such as removable or fixed contamination on structural surfaces, distributed or small areas of elevated contamination in soil, etc. and locations
- Determine whether remediated material will meet the waste acceptance criteria of the selected disposal facility and also transportation requirements for the waste to meet Department of Transportation (DOT) requirements. The IAEA has adopted international regulations for the safe transportation of radioactive material. As a member state, the United States, through DOT

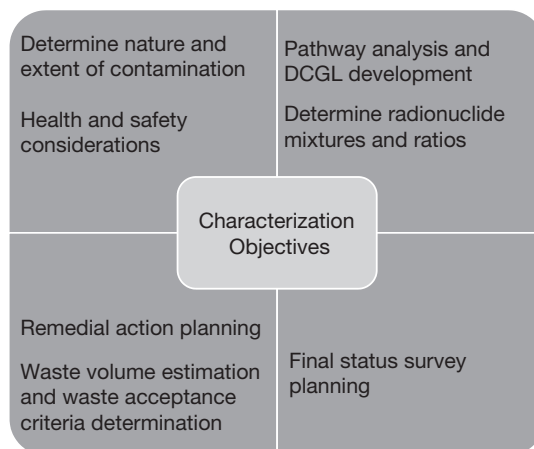


Fig. 3 Characterization objectives.

and NRC, has amended its domestic regulations to make them compatible with the IAEA standards.” Applicable Code of Federal Regulations are [10 CFR Part 71.5](#) Transportation of Licensed Material, (NRC citation of DOT regulations) and DOT requirements are specified in 49 CFR Parts 72 and 73.

From the above list of goals, the following are possible PSQs asked during Step 2 of the DQOs and should emphasize why the characterization plans significantly benefit by applying the DQOs:

- What is the physical form of contamination?
- What are the specific contaminants?
- What is the magnitude of contamination?
- What are the contaminant variabilities, averages, and ranges?
- Are contaminants mixed or isolated?
- What are the contaminated media?
- Are contaminants present in a mobile form?
- Is groundwater contaminated?
- Could groundwater be contaminated in the near future?
- Is contamination surficial or volumetric within structural surfaces?
- Is contamination limited to surface soil or has subsurface soil been impacted? What is the depth of the contaminated layer?
- Are there site-specific conditions—e.g. soil type/porosity, depth to groundwater, etc.—that may impact DCGL_W modeling?
- Are areas candidates for FSS without remediation?

Plan for remedial actions

Characterization results also support remedial action planning to:

- Assess the scope of proposed decommissioning actions
- Support evaluation of alternative decommissioning actions
- Ensure decommissioning worker and public safety
- Evaluate potential environmental releases during decommissioning and determine methods to minimize the occurrence and impact
- Determine the suitability of various survey instruments and analytical procedure and methods
- Estimate post-remedial action (residual) contaminant concentrations
- Determine the type, concentration, and extent of non-radiological contamination
 - Because remediation can remove both radiological and non-radiological contamination, it might not be necessary to assess both in a given area
 - Evaluate the potential for encountering mixed waste or accidentally generating mixed waste—mixed waste can be extremely expensive to disposition

Characterization plan preparation

The DQOs are incorporated into the characterization plan itself. There are a myriad of sampling design choices available when planning an environmental investigation. In principal, the choices expand out from two basic sampling tenets: non-probabilistic and probabilistic.

Non-probabilistic uses non-random techniques to select a sample or its location and includes judgmental (biased/targeted) sampling and convenience sampling.

- As discussed for scoping surveys, judgmental sampling techniques are based on some subjective condition using professional knowledge or experience.
- Convenience sampling—also referred to as opportunistic, haphazard, grab, or accidental sampling—is exactly as the name implies: a location is sampled because it is close by, easy to obtain, or is happened upon while performing other site investigation activities. These are not random samples.

Probabilistic sampling uses random techniques such that every possible subset (sample) has an equal selection probability to represent the population of interest. The sampling plans also account for the expected type of population distribution, normal (parametric) or skewed (non-parametric). Sample placement is either in an ordinary random, systematic, or stratified manner. The plans may also incorporate other techniques that include composite sampling where multiple increments are combined to form one sample for analysis, thereby saving analytical costs, or processes where an inexpensive field screening tool may be used to select samples for more rigorous analysis. All of the common methods are described in another EPA document titled *Guidance on Choosing a Sampling Design for Environmental Data Collection* (EPA, 2002). This guidance has been incorporated into the computer-based sample planning program, Visual Sample Plan (VSP), which is maintained by the DOE's Pacific Northwest National Laboratory (PNNL). The VSP software may be downloaded at no cost from the PNNL-VSP website (<https://vsp.pnnl.gov/>).

Several factors are considered in the judgmental [non-probabilistic] or random [probabilistic] sample planning decision, including level of effort to implement the sample plan, costs, and foremost, the data end-use. Random samples are necessary to optimize the precision, accuracy, and control bias and uncertainty of the measurements used for statistical decision-making and other data evaluations or study result validations. Judgmental samples remain valuable. Judgmental sample results are typically evaluated independently—for example, comparing soil concentrations at a hot spot to action levels or outlier tests—and are not pooled with random sample results for statistical calculations. They may also be useful for specific parameters such as establishing the ratios of radiological contaminants to one another, samples should be collected from locations that have elevated activity rather than from a group of random samples that may only represent background.

Table 3 provides some advantages and disadvantages of the two methods.

Ultimately, one or both of these sample types are collected during the site investigation process. The important thing is to know the data end use and to recognize the differences between a judgmental and random location. The caution is that some plans may not be appropriate for specific problems. In other words, a plan designed to assess the average concentration against an action threshold will not be appropriate for identifying small areas of unacceptable concentrations, also known as not-to-exceed thresholds. As such, the decision that will be made with the data during the DQA is a key factor in selecting the appropriate plan and again spotlights why the DQOs are important.

The type of plan will also determine the number of samples that are required to make a decision and control the decision errors. There are several factors that impact the required number of samples. The most significant sample size input is how close the estimate of the assumed base condition is to the decision action level. In statistical terms, the difference between the assumed base condition and decision threshold is referred to as the gray area. In those instances where the parameter clearly exceeds or is clearly less than the threshold, statistical tests are not necessary. However, when the parameter is within the gray area, things are too close to call. A narrow gray area coupled with a lot of variability in the parameter, i.e. a lot of uncertainty, then a large number of samples is required. Using the example average radionuclide concentration and DCGL_W action level scenario, the closer the average concentration is to the DCGL_W, and the less certainty there is the average estimate, the greater the number of samples required to make a good decision. **Fig. 4** illustrates this idea, as the mean approaches the decision change threshold, the increased uncertainty illustrated with the two curves shows how the uncertainty in the mean estimate extends well past the action level.

Table 3 Comparison of random versus judgmental sample plans.

<i>Random</i>	<i>Judgmental</i>
Advantages	
• Uncertainty calculations • Statistical analysis/inference • Decision error control	• Less expensive to plan and implement • Efficient when site is well known • Provides important information for specific data needs, such as contaminant ratios and maximums
Disadvantages	
• Random locations may be difficult to locate in large survey units • Requires more planning to optimize design	• Requires expert knowledge to acquire targeted information • Not reliable for estimating uncertainties • Depends on subjectivity to interpret study objectives

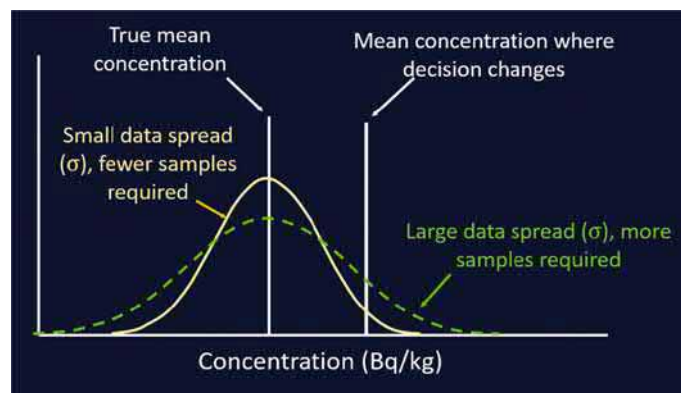


Fig. 4 Illustration of parameter population distribution: mean, variability, and action level.

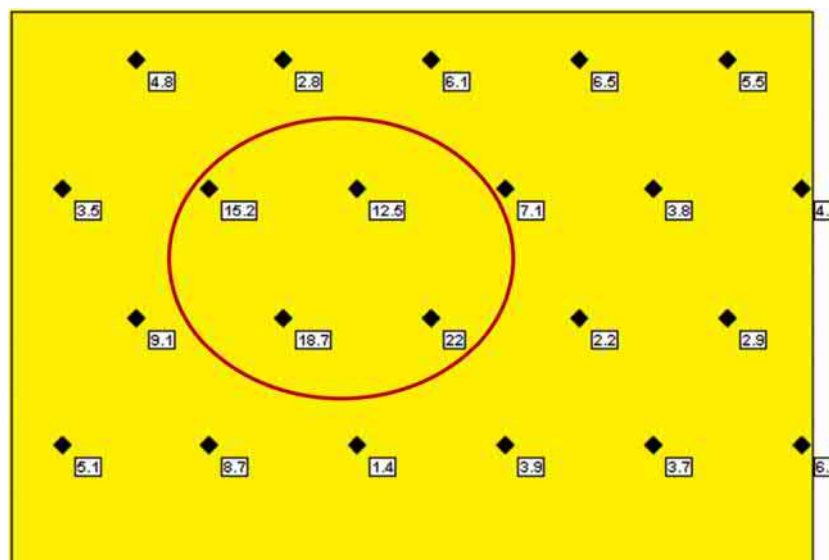


Fig. 5 Posting plot (created using Visual Sample Plan, PNNL, 2019).

In addition to characterization sampling to determine remediation requirements, oftentimes there are large portions of the site that will not require remediation. For these areas the characterization plan may be intended to evaluate spatial variability over Class 2 or 3 areas. Information on spatial variability may be extremely beneficial for demonstrating that ROC levels are less than $DCGL_W$ s and optimizing survey unit boundaries. A posting plot is one of the best methods to visualize spatial relationships in the concentrations. Fig. 5 shows an example plot. The values shown are the radionuclide concentrations in the samples collected. The red outline highlights the area with highest concentrations and that may require remediation if greater than the $DCGL_W$, or segregated as a separate FSS survey unit.

Other plans may be designed to estimate the average concentration of an area requiring remediation to ensure waste acceptance criteria (WAC) requirements are met and also identify areas via a systematic sampling method that may require special handling. Fig. 6 is an example VSP plan with sample locations shown. The example plan is a unique method known as ranked set sampling (RSS). It combines random sampling with professional judgment. The goal of this method is to reduce the number of samples required to obtain a as good an estimate of the mean obtained with simple random samples. Note that random sampling would require 27 samples while RSS requires 15 for the same confidence in the mean estimate. EPA (2002) provides more information on RSS.

There are other sample plans that may be developed with and without formal hypothesis tests or thresholds as part of the data assessment. These plans include the following:

- Tests of proportions
- Establish a baseline for trend analyses, a control chart is one example
- Estimate the mean and CI of the mean
- Identify unacceptable areas (e.g., identify presence/absence of hot spots of area $\geq X$ with $Y\%$ confidence)

The plan that is selected for characterization studies varies considerably for each unique site. In most all cases, multiple types of plan are implemented for different site areas, dependent upon the decisions that will be made.

Procedures

The characterization survey procedures depend upon the type of data required as determined during DQO development. Land areas are almost always gamma scanned to identify soil contamination, provided that there are gamma-emitting ROCs. Locations where radiological anomalies are identified or other suspect locations are sampled and the samples submitted to the laboratory for analysis by gamma spectroscopy or various radiochemistry methods. If subsurface contamination is suspected, borehole sampling is performed via various mechanical (drill rigs or direct push sampling) or manual methods using hand augers. The combination of scanning and sampling are used to bound the contamination extents. Fig. 7A–C shows gamma radiation scan data for a contaminated land area, the sample locations collected from the area, and the area after the soil was excavated for disposal. The sampling data were used to determine the excavation boundaries, calculate the total volume of soil requiring remediation, the concentrations needed for comparison to the WAC, and for generating shipping papers.

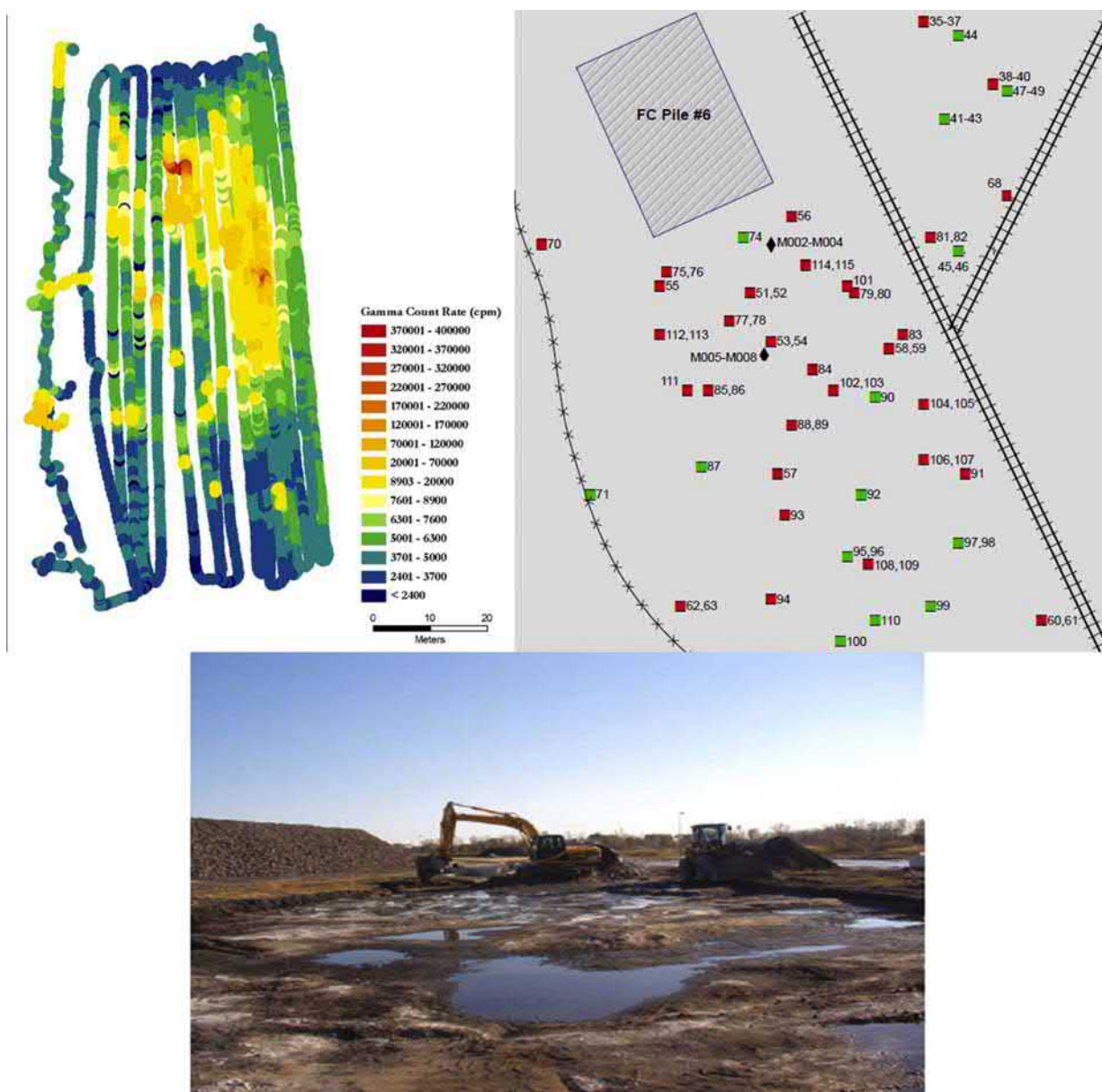


Fig. 7 (A–C) Gamma radiation scan results, soil sample locations and remediation of a land area.

Buildings undergo a similar process for determining the location and magnitude of contamination. Surfaces are scanned for alpha, beta, and/or gamma radiation followed by direct measurements of the total surface activity levels for comparison with the $DCGL_W$. These measurements are supplemented as determined during DQO development with samples. Samples may include smears or large-area wipe samples to assess the fraction of the total contamination that is removable (loose) and may be easily remediated versus the amount that is fixed to the surface and must be decontaminated via aggressive techniques. Physical samples are also necessary to assess the specific ROC mixture when more than one contaminant is present, or to determine if the contamination rather than being only surficial, has penetrated into the volume of the material. Volumetric contamination also occurs when materials of construction are radiologically activated due to exposure to a neutron field, common with nuclear reactor or accelerator facilities. Fig. 8A and B are images of a floor undergoing investigation. The first image is where a top layer of asphalt that covered the original, contaminated concrete floor, has been marked where elevated gamma radiation was measured, indicating contamination beneath the asphalt. The second image shows floor removal to sample the subfloor surface that became contaminated from liquid migrating through an expansion joint. The third and fourth images are a plot plan showing measurement and sampling locations—color-coded to indicate contaminated area and locations that were less than the $DCGL_W$ —and the same area after remediation.

Remedial action support survey (RASS)

Once the site has been characterized, the focus shifts to completing remediation and the next RSSI phase begins; the remedial action support survey (RASS). The RASS is not necessarily one discrete survey—it may be a combination of several data collection efforts to determine remediation effectiveness in real-time, and verifying characterization phase assumptions (e.g. depth of contamination). An example is performing surface contamination measurements immediately following concrete scabbling to verify that the thickness removed was adequate or in the case of soil, when excavation activities have reached a sufficient depth for contaminated soil removal. The RASS data are used to determine when an area is ready for the FSS. Consider that with remediation, the population distribution, median or mean, and skewness of contamination drastically changes. The average contaminant concentration in a remediated area should certainly go down below the levels seen during characterization. The variability in contaminant



Fig. 8 (A) Areas of elevated direct gamma radiation detected while scanning. (B) Exposed contaminated sub-floor strata, sample collected for laboratory analysis. (C) Characterization results map. (D) Floor slab removed, exposed contaminated sub-floor strata.

concentrations would also be expected to be reduced, i.e. fewer locations with higher concentrations relative to lower concentration areas. In other words, one would not want to use the characterization survey data as the descriptive statistics for an area that has been cleaned up. Why are averages and variability important? Because the mean estimate of the contamination level in the remediated area and the uncertainty of that mean are the critical statistical inputs to the decision (DQO Step 3) for the remediated area's next phase, the final status survey (FSS).

Final status surveys

The final status survey is the only survey that is a requirement in impacted portions of the site, and FSS design constitutes the bulk of the MARSSIM guidance. The reason it is required is because the FSS generates the main data to demonstrate compliance with the cleanup objectives, and a comprehensive survey report is submitted to the regulator to support the site release request.

Overview of MARSSIM

The *Multi-Agency Radiation Survey and Site Investigation Manual* (NUREG-1575, Rev.1) was a multi-agency effort involving the NRC, EPA, DOE, and DOD. Their purpose was to develop an industry standard for how the FSS is performed and to ensure a consistent, integrated yet flexible approach, is applied when conducting radiological surveys. From the licensee's perspective, an important point is that MARSSIM-based final status surveys generally require fewer samples and/or measurements than past guidance. Fewer samples required for laboratory analysis results in considerable cost savings.

MARSSIM encourages the use of non-parametric statistics rather than parametric statistics. The non-parametric statistics are appropriate because they do not require the underlying assumption of a normal distribution that the parametric statistics require. This is important because of the population skewness expected due to areas of residual contamination.

Counter to the advantages, there are also limitations. MARSSIM's guidance does not include the assessment of subsurface soil contamination, or volumetrically contaminated structural components (e.g., walls). These matters must be addressed on a case-by-case basis. Also, MARSSIM's methodology for addressing localized areas of high contamination, or "hot spots," and other discrete particles of radioactive material (e.g., fragments of depleted uranium) is not particularly robust. Technology has also advanced since MARSSIM's inception and therefore the guidance does not address the use of modern computer-based instrumentation systems, e.g., systems that can simultaneously locate (i.e., scan) and quantify contamination, such as the GPS systems coupled to scanning equipment shown previously.

While MARSSIM's flexibility can be considered an advantage, the large number of decisions that the licensee must make can complicate matters. Like Newtonian physics, for every action there is an equal and opposite reaction. A change in one parameter will have other consequences, either negative or positive that must be understood. For example, taking fewer samples will reduce statistical power and thereby increase the probability of a Type II error. When taking advantage of MARSSIM's flexibility, the licensee must understand its methodology to avoid compromising the survey quality. Again, the use of the DQO process is necessary to ensure that the survey data will be of the right quantity and quality for decision making.

Design and implementation of the FSS

The site is first divided into discrete areas called survey units. The prior investigations are likely to have involved much larger and possibly less defined decision units. For the FSS, it is the survey unit that serves as the fundamental unit of compliance; every survey unit must be shown to meet the release criterion for the site to be released. Quite a bit of thought is required when subdividing the site into survey units. Different survey units at a site can differ in their radionuclides of concern, size, scan coverage, type and number of measurements, etc. but a given survey unit can only have a single classification (Class 1, Class 2, or Class 3) and its size is linked to its class. As indicated in Fig. 9, the greater the potential for contamination, the smaller the survey unit to ensure a reasonable sample density based on contamination potential.

Also, MARSSIM provides guidance on maximum allowable survey unit sizes for each classification that are shown in Table 4:

A sampling plan is prepared for each survey unit. The default sample plans are based on two non-parametric statistical tests, the Sign test and Wilcoxon Ranked Sum (WRS) test. The Sign test is a single sample test and is used when the contaminant is not present in background or, the $DCGL_W$ is well above background and thus background does not need to be accounted for. The test compares the median/mean (because it is non-parametric the test is actually comparing the median which is not affected by outliers) concentration in the survey unit to the $DCGL_W$ in order to determine whether the median concentration is less than the $DCGL_W$. The WRS test is a two sample test in which the survey unit median is compared to a background reference area, to determine if the median concentration is less than the background median plus the $DCGL_W$.

Both tests apply a hypothesis test. The assumed base condition is referred to as the null hypothesis (H_0). In the case of the FSS, the base condition assumes the survey unit is contaminated above the $DCGL_W$. This is written as:

$$H_0 = \text{median}_{\text{survey unit}} \geq DCGL_W$$

The alternative hypothesis (H_A) is that the survey unit median concentration is less than the $DCGL_W$ or:

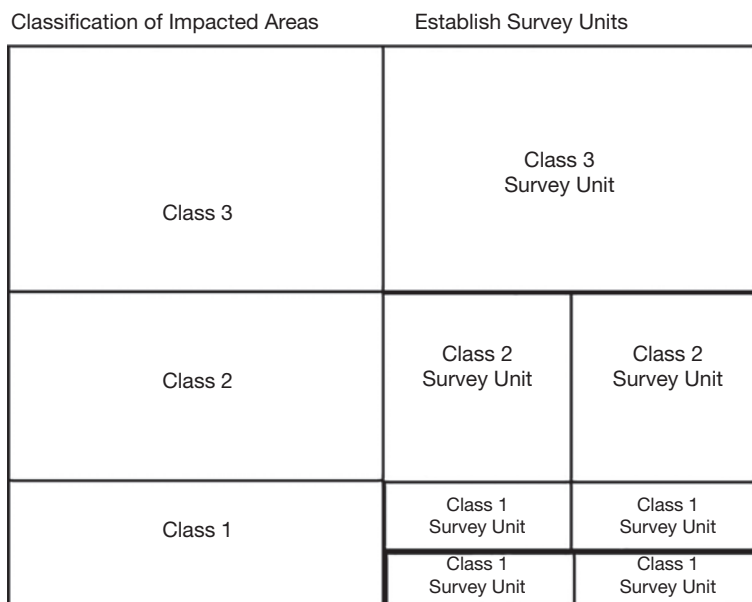


Fig. 9 Illustration of MARSSIM survey unit sizes.

Table 4 MARSSIM recommended maximum survey unit size.

<i>Survey Unit Classification</i>	<i>Land Areas</i>	<i>Buildings (floor area)</i>
Class 1	2000 m ²	100 m ²
Class 2	10,000 m ²	1000 m ²
Class 3	no limit	no limit

$$H_A = \text{median}_{\text{survey unit}} < \text{DCGL}_W$$

It takes overwhelming evidence to reject H_0 and accept H_A . This is analogous to the legal system, except with the opposite presumption of guilt; the survey unit is assumed guilty until proven innocent rather than innocent until proven guilty. This is a very important distinction. Whichever condition has the most severe consequence if incorrectly rejected is set as the base condition. The erroneous release of contaminated survey unit is more severe than incorrectly deciding not to release a clean survey unit, hence the reason for establishing H_0 as assumed contaminated.

For both tests, the inputs required to determine sample size are the Type I (referred to as the regulator's error under the MARSSIM paradigm) and Type II error which is the site's error of concern. The other inputs are the DCGL_W , the lower bound of the gray region (LBGR)—normally established at the estimated mean concentration—and the variability in the survey unit. Those three factors are combined into an equation known as the relative shift.

$$\frac{\text{DCGL} - \text{LBGR}}{\sigma}$$

Once the relative shift is calculated, a table in MARSSIM can be consulted for number of survey unit samples needed. **Table 5** reproduces Table 5.5 of MARSSIM, used when the Sign test is planned. To illustrate, suppose the Type I error is selected at 0.05, equivalent to 95% confidence, and the Type II error is established at 10% which is equal to 90% power. Then, for plugging into the relative shift equation: an example DCGL_W of 1.0 is used (any units will do for the example), the LBGR is 0.5 and sigma is also 0.5 (both are in the same units as the DCGL_W):

$$\frac{1.0 - 0.5}{0.5} = 1$$

Looking up the relative shift and decision errors in the table is simple as shown in **Table 5** where the column corresponding to the decision errors intersects with the relative shift value. The answer: 23 samples are required to satisfy the planning inputs.

Further emphasizing the concepts introduced with **Fig. 4**, the closer (larger) the LBGR is to the DCGL_W and the larger the variability (σ) is, the smaller the relative shift becomes. Small relative shift values equate to larger required sample sizes to reject H_0 . Suppose now that both the LBGR and sigma were much larger, we will say 0.75 (meaning the average contamination level is much closer to the DCGL_W and there is much greater variability in the population), and the resulting relative shift equaled 0.3; then 185

Table 5 Sample number for Sign test.

Δ/σ	$\alpha=0.01$					$\alpha=0.025$					$\alpha=0.05$					$\alpha=0.10$					$\alpha=0.25$				
	β					β					β					β					β				
	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25
0.1	4095	3476	2984	2463	1704	3476	2907	2459	1989	1313	2984	2459	2048	1620	1018	2463	1989	1620	1244	725	1704	1313	1018	725	345
0.2	1035	879	754	623	431	879	735	622	503	333	754	622	518	410	258	623	503	410	315	184	431	333	258	184	88
0.3	468	398	341	282	195	398	333	281	227	150	341	281	234	185	117	282	227	185	143	83	195	150	117	83	40
0.4	270	230	197	162	113	230	1921	162	131	87	197	162	136	107	68	162	131	107	82	48	113	87	68	48	23
0.5	178	152	130	107	75	152	126	107	87	58	130	107	89	71	45	107	87	71	54	33	75	58	45	33	16
0.6	129	110	94	77	54	110	92	77	63	42	94	77	65	52	33	77	63	52	40	23	54	42	33	23	11
0.7	99	83	72	59	41	83	70	59	48	33	72	59	50	40	26	59	48	40	30	18	41	33	26	18	9
0.8	80	68	58	48	34	68	57	48	39	26	58	48	40	32	21	48	39	32	24	15	34	26	21	15	8
0.9	66	57	48	40	28	57	47	40	33	22	48	40	34	27	17	40	33	27	21	12	28	22	17	12	6
1.0	57	48	41	34	24	48	40	34	28	18	41	34	29	23	15	34	28	23	18	11	24	18	15	11	5
1.1	50	42	36	30	21	42	35	30	24	17	36	30	26	21	14	30	24	21	16	10	21	17	14	10	5
1.2	45	38	33	27	20	38	32	27	22	15	33	27	23	18	12	27	22	18	15	9	20	15	12	9	5
1.3	41	35	30	26	17	35	29	24	21	14	30	24	21	17	11	26	21	17	14	8	17	14	11	8	4
1.4	38	33	28	23	16	33	27	23	18	12	28	23	20	16	10	23	18	16	12	8	16	12	10	8	4
1.5	35	30	27	22	15	30	26	22	17	12	27	22	18	15	10	22	17	15	11	8	15	12	10	8	4
1.6	34	29	24	21	15	29	24	21	17	11	24	21	17	14	9	21	17	14	11	6	15	11	9	6	4
1.7	33	28	24	20	14	28	23	20	16	11	24	20	17	14	9	20	16	14	10	6	14	11	9	6	4
1.8	32	27	23	20	14	27	22	20	16	11	23	20	16	12	9	20	16	12	10	6	14	11	9	6	4
1.9	30	26	22	18	14	26	22	18	15	10	22	18	16	12	9	18	15	12	10	6	14	10	9	6	4
2.0	29	26	22	18	12	26	21	18	15	10	22	18	15	12	8	18	15	12	10	6	12	10	8	6	3
2.5	28	23	21	17	12	23	20	17	14	10	21	17	15	11	8	17	14	11	9	5	12	10	8	5	3
3.0	27	23	20	17	12	23	20	17	14	9	20	17	14	11	8	17	14	11	9	5	12	9	8	5	3

Reproduced from Table 5.5 of NRC (2001) *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*. NUREG-1575, Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission.

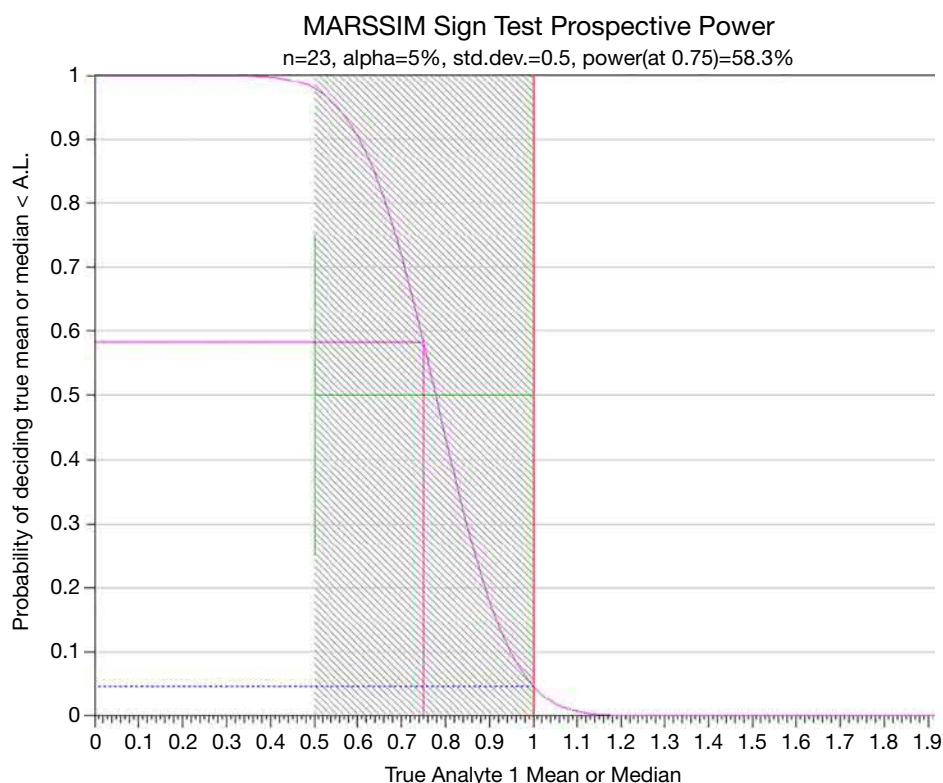


Fig. 10 Example power curve (created using Visual Sample Plan, PNNL, 2019).

samples would be necessary to satisfy the desired confidence level and power. That number of samples would likely be unrealistic due to cost or other factors. This is where the Newtonian physics metaphor comes into play. One alternative to reduce sample size would be to adjust, decrease, the LBGR. However that increases the risk of failing to release a clean survey unit. The power curves shown in Figs. 11 and 12 show this relationship.

Power curves

VSP includes an interactive graphing capability to examine the impact on sample sizes and decision error probability. An example power curve is shown in Fig. 10 for a one-sample Sign test.

If unfamiliar with how to read a power curve, it essentially shows what mean concentration was input for the sample planning design. The LBGR was 0.5; the action threshold of 1 was selected; and the assumed standard deviation was 0.5. Note that the aforementioned input parameters are specified in terms of unity. The Type I error is 5% (0.05 on the Y-axis), which the curve crosses at the derived concentration guideline level (DCGL_W) of 1, and the Type II error is 10% (0.10), which occurs at the LBGR. Power is defined as $1 - \beta$, so where does one look on the curve to find power or the probability of making a correct decision?

The circled area of Fig. 11 shows if the mean/median of the survey unit was equal to $\frac{1}{2}$ the DCGL_W what the probability would be of correctly deciding that the survey unit is less than the DCGL_W. The Type II error, ($1 - 0.90$ in this case) is incorrectly deciding that the survey unit stated concentration is greater than the DCGL_W. The curve should go through the selected power, 90% in this example, at the LBGR. However, the power is actually higher than 90%, approximately 98%. Why is that? The higher power results from VSP increasing the sample size of 19 by 20% to $n = 23$ in accordance with MARSSIM guidance. Sample sizes are increased to maintain planned power should some samples either not be collected or the data are not useable due to quality control issues.

The curve is useful for examining planning error impact on desired power. What if the 0.5 mean (LBGR) is an underestimate? Suppose the mean is closer to 0.75? In that case, find where 0.75 on the X-axis intersects the curve on the Y-axis—shown on Fig. 11.

Based on the planned 23 samples, the probability of making the correct decision if the actual mean is 0.75 is only 58.3%. The VSP inputs can be toggled to determine how many samples are required to achieve 90% power at a mean of 0.75. The answer is 71.

Conversely, suppose the planning team had overestimated the mean at 0.75 by using judgmental sample data and the actual mean was 0.5. With 71 samples, the power curve would show $>90\%$ at 0.75 and essentially 100% power at 0.5, but at what cost? 48 more samples than necessary.

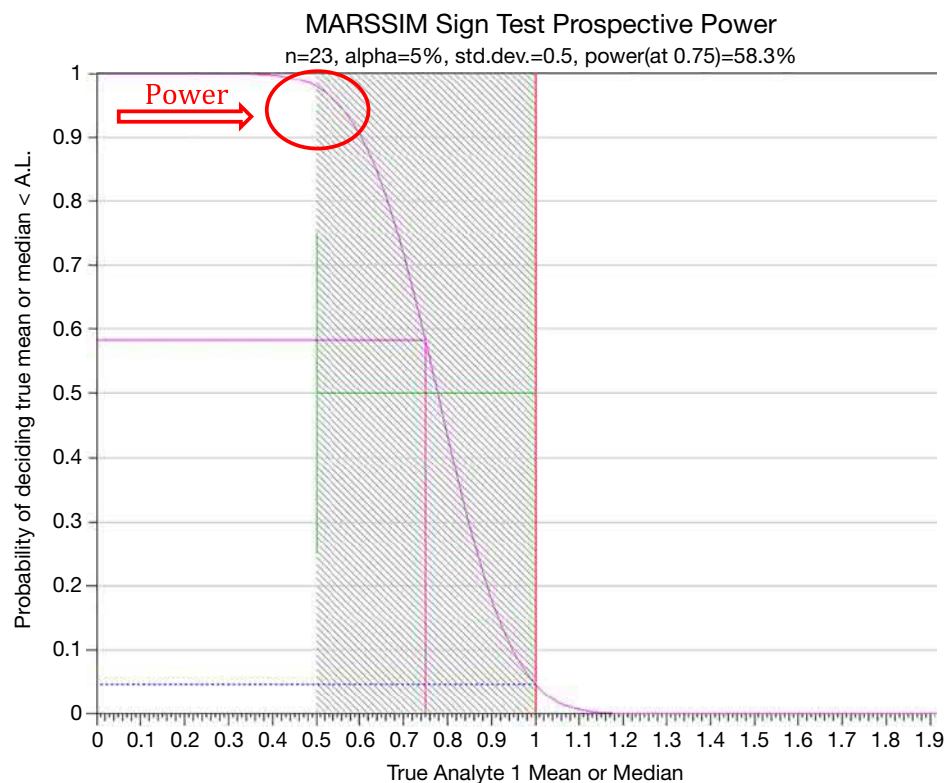


Fig. 11 Power illustration at a mean of 0.5 (created using Visual Sample Plan, PNNL, 2019).

Indistinguishable from background

Another alternative to the default MARSSIM methodology for determining whether site concentrations satisfy the DCGL_W is referred to as Indistinguishable from Background, or Scenario B. Scenario B is reserved for very specific situations where the ROC is also present in background, the DCGL_W is very low relative to background, and there is significant variability in background where it proves difficult for the statistical tests to distinguish contamination from what might only be higher background. The assumed Scenario B base condition is that the survey unit ROC concentrations are less than the DCGL_W . The null hypothesis is reversed, thereby easing the burden of proof that the site is clean as it will take overwhelming evidence to conclude the site is contaminated in excess of the DCGL_W . A complete description of the statistical methods used in Scenario B, in addition to the MARSSIM adopted statistical methods, may be found in NUREG-1505 (NRC, 1998).

FSS procedures

The FSS procedures are similar to the other surveys, consisting of radiological scans and direct measurements using appropriately sensitive alpha, beta, and gamma detectors and sampling. MARSSIM has specific requirements that must be met regarding scanning coverage and detection sensitivity.

Recall the discussion that MARSSIM introduced an integrated process. Scan coverage and required scan sensitivity demonstrate both the graded approach concept and how different procedures are integrated together to strengthen the confidence in the investigation. For the graded approach, the greatest effort is expended in areas with the highest residual contamination probability. In decreasing order of potential: Class 1, 2, and 3. In Class 1 areas, remediation is not necessarily precise. For instance while a trackhoe is excavating soil, all the contaminated soil may not be removed, or after loading into a truck material spills enroute to the waste handling area. Therefore, Class 1 survey units require 100% scanning coverage for the radiations of concern. Furthermore, the actual scan sensitivity, or minimum detectable concentration (MDC) must be demonstrated to be less than a required scan MDC. The applicable required scan MDC is a function on the allowable concentration in a hot spot of a predetermined size of concern. These allowable hot spot limits are referred to as DCGL_{EMC} ($\text{DCGL}_{\text{Elevated Measurement Comparison}}$). Smaller hot spots with higher concentrations can still satisfy the dose limit. However, for certain projects there are no allowances for hot spots that exceed the DCGL_W . In that case, the required scan MDC would be equivalent to the DCGL_W . Fig. 12A and B show the completed required scanning results as well as the planned sampling locations that resulted from the FSS DQO outputs.

Class 2 areas should not have any contamination in excess of the DCGL_W , and should not have required remediation. Had contamination above the DCGL_W been found either during an earlier survey or during the FSS, the survey unit would have required

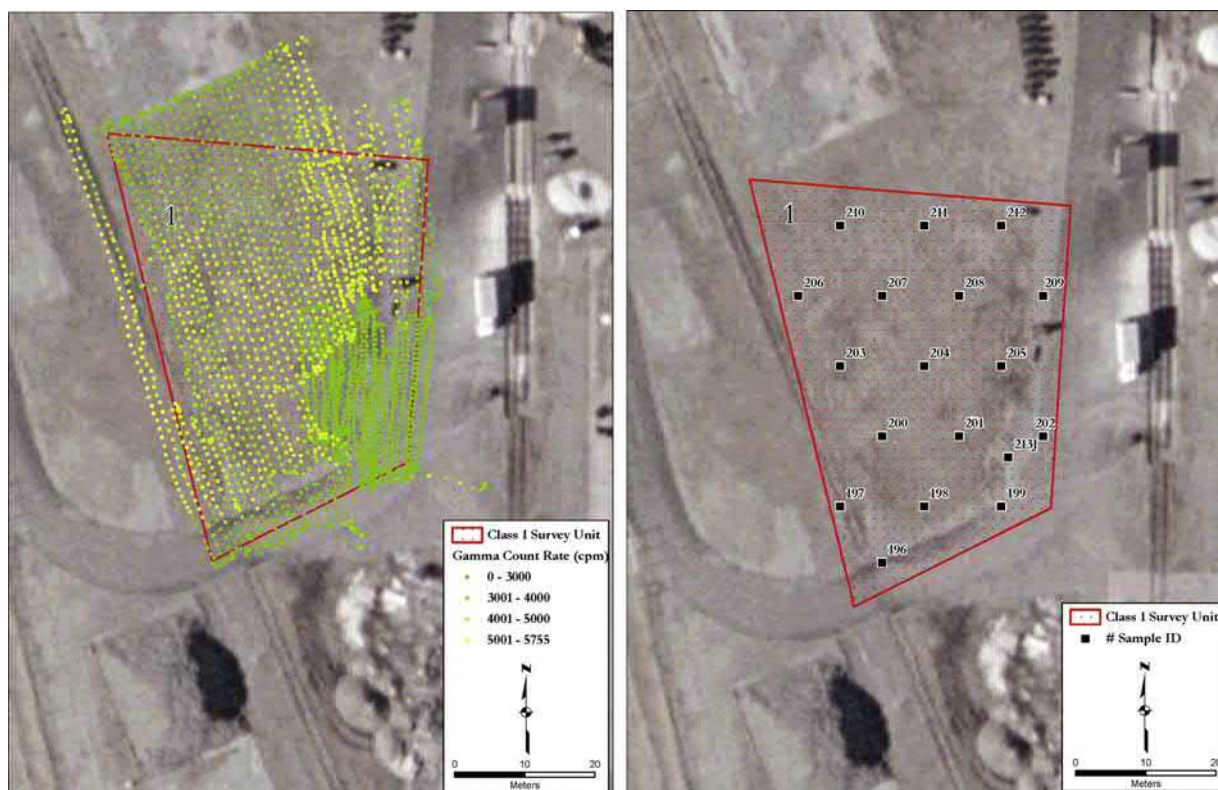


Fig. 12 (A and B) Final status survey scan and sampling maps for the remediated soil area seen in Fig. 7A–C.

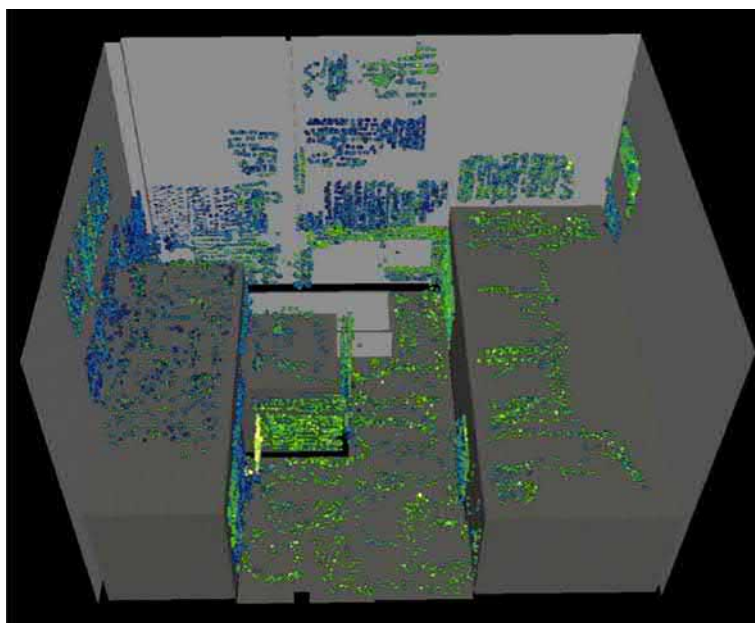


Fig. 13 3-D image of FSS alpha-plus-beta scan results.

reclassification to a Class 1. Class 2 FSS scan coverage requirements are 10 to 100%. Higher density scans occur in areas with the greatest probability of having become contaminated above the $DCGL_W$, such as where a Class 2 unit shares a boundary with Class 1 units or areas with the greatest potential for anthropogenic impact from radiological materials. Fig. 13 represents Class 2 scanning results. The data captured were from a former research reactor bay and spent fuel pool. The reactor bay was a Class 2 with approximately 25% coverage, the fuel pool was Class 1 and scanned 100%. The data show no residual contamination for the areas investigated.

Finally, because there should be very low potential for contamination in a properly classified Class 3 area, scan coverage requirements are minimal with judgmental scans recommended. That means scanning the areas that would be the most suspect locations to have possibly accumulated contamination.

Once scans are completed, judgmental samples are collected from suspect anomalies and random samples from each survey unit for laboratory analysis. The results are provided in units of the DCGL_Ws and the direct measurement results are converted to surface activity levels using appropriate detector efficiency and other conversion factors. Survey unit data are then assessed for compliance with the DCGL_Ws.

DQA

FSS data are compiled and evaluated. The scanning data are reviewed, plotted/mapped, and examined for anomalies. The judgmental data collected from anomalies are not included in the statistical assessment but are examined independently to ensure that the concentrations are below classification thresholds; a fraction of the DCGL_W for Class 3s and the Class 2 threshold is the DCGL_W. Hot spot samples that exceeded the DCGL_W from a Class 1 survey unit are assessed to ensure they were below the applicable DCGL_{EMC}.

The random sample evaluations include generating basic statistical parameters, graphing, or plotting of the data. The decision to perform the Sign or WRS test depends upon the initial assessment outcome. In many cases the initial review will show that either the survey unit will always pass the hypothesis test—the null hypothesis will always be rejected—while in other cases the survey unit concentrations will obviously fail the hypothesis test. Table 6 reproduces MARSSIM Table 8.2 hypothesis testing summary for when a survey unit would always fail, always pass, and when the statistical test is required to decide when the survey unit does or does not satisfy DCGL_W.

Refer to MARSSIM for examples of how the tests are performed.

Confirmatory/verification survey

The surveys presented thus far are performed by the site or their contractor, the regulatory authority also has a place in the RSSI phases. The regulator may conduct a verification survey, also referred to as a confirmatory survey, to collect independent data to substantiate and provide confidence in the reported FSS results. Verification activities that might be performed include performing independent scan surveys, measurements, and sampling. Data are tested for agreement with the site's results.

The RSSI process does not always proceed in this sequential flow. To a large extent, the radiation survey and site investigation process proceeds independently in different portions of the site. Additionally, all the surveys types discussed may not necessarily be performed in all areas. For instance, an area determined during characterization to not exceed the cleanup criteria is unlikely to be remediated, and therefore there would not be a need to perform a remedial action support survey. In one area of the site, only the FSS might be required. In other areas, the scoping, characterization, remedial action, and final status may all be necessary. There may be cases where one could go straight from the scoping survey to the FSS, such as an area that is clearly Class 3, or plan a scoping or characterization to also satisfy FSS requirements. After all, when the DQOs are planned correctly and appropriate survey methodologies implemented, a site may ensure that the quantity and quality of data collected are of the right type to satisfy multiple objectives.

Table 6 MARSSIM Table 8.2.

<i>For WRS test</i>	
<i>Survey result</i>	<i>Conclusion</i>
Difference between largest survey unit measurement and smallest reference area measurement is less than DCGL _W	Survey unit meets release criterion
Difference of survey unit average and reference area average is greater than DCGL _W	Survey unit does not meet release criterion
Difference between any survey unit measurement and any reference area measurement greater than DCGL _W and the difference of survey unit average and reference area average is less than DCGL _W	Conduct WRS test and elevated measurement comparison
For Sign test	
All measurements less than DCGL _W	Survey unit meets release criterion
Average greater than DCGL _W	Survey unit does not meet release criterion
Any measurement greater than DCGL _W and the average less than DCGL _W	Conduct Sign test and elevated measurement comparison

Reproduced from Table 8.2 of NRC (2001) *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*. NUREG-1575, Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission.

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Radiological Control Practices

T.W. Hansen, Jr, Ameriphsics LLC, Knoxville, TN, United States

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Introduction

This article is preceded by topics that are important to a discussion of radiation protection, but none is a protection, per se. That is, even the most thorough understanding of health effects, radiation detection systems, regulations, etc. will not protect people and the environment from the harmful effects of radiation unless that knowledge is used to design and implement appropriate radiological control practices. This article is an explanation of such practices.

The importance of planning

The National Institute for Occupational Safety and Health (NIOSH) has developed a “Hierarchy of Controls” that serves as a valuable planning tool for organizations seeking to reduce occupational injuries, illnesses, and fatalities. The hierarchy is presented in [Fig. 1 \(CDC, 2020\)](#). It is not explicitly a radiation protection; rather, it depicts an order of controls that can be implemented to curb exposures to any occupational hazard. Although the hierarchy was developed as a means of protecting workers, it reflects an approach that can be used to consider control solutions that also protect the public and the environment.

The premise underlying the hierarchy is that controls at the top of the figure are most effective and therefore more protective. Controls at the bottom are least effective and protective. Unfortunately, the controls at the top, eliminating the hazard or substituting it, may be impossible to implement, particularly at existing installations. For example, it is not feasible to eliminate or substitute radiation at a reactor site. Thus, the possible controls that remain are engineering controls, administrative controls, and personal protective equipment (PPE). This sequence of controls, starting with engineering controls and ending, as a last resort, with PPE, is preferred by virtually every safety organization in the world.

If the hierarchy is applied to radiation exposure in instances where the hazard cannot be eliminated or substituted, then ideally the hazard is controlled through facility and equipment design features such as ventilation, containment, shielding, and interlocks. Engineering controls of this nature are considered in the early planning stages of a facility, project, or process. When design features do not sufficiently mitigate radiation exposure, then administrative and procedural controls are planned. The takeaway then is that the most important factor in preventing harm from radiation must be planning. That is, exposures are kept low by incorporating radiological considerations well before the hazard presents itself rather than by reacting to whatever radiation arises from a given activity.

A special case for planning is needed when work such as maintenance, modification, or decommissioning is performed on or at a nuclear installation as these activities can temporarily or permanently suspend existing engineering or administrative controls ([DOE, 2008](#)). Moreover, these activities may create or exacerbate radiological hazards such that existing controls are not robust enough to adequately manage exposures. Consequently, an evaluation of the appropriateness of existing controls is warranted whenever non-routine work is performed.

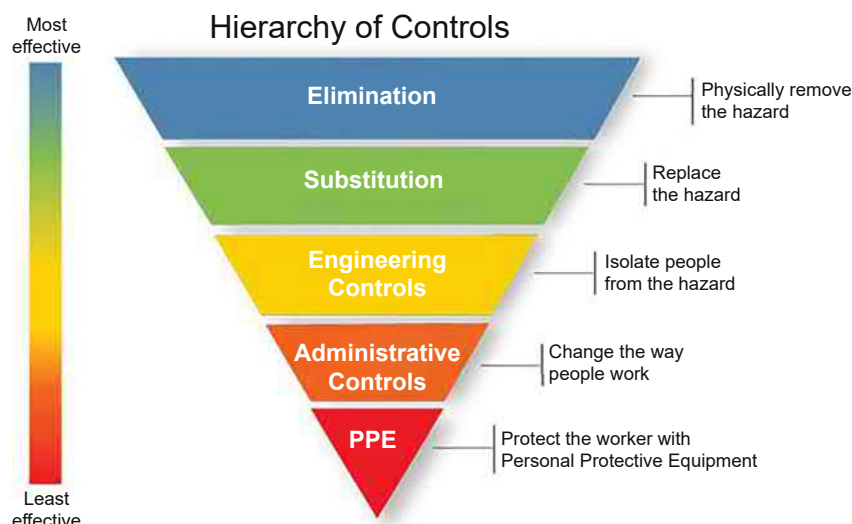


Fig. 1 NIOSH hierarchy of controls. The National Institute for Occupational Safety and Health, www.cdc.gov/niosh/topics/hierarchy.

It could be argued that controls for emergency situations such as nuclear catastrophes are special cases that rely more on a strong reactionary response than design. Nonetheless, many elements of emergency response can and should be planned to increase the effectiveness of the response. Who will respond? What training will responders need, and is there a way to get them high quality training before the emergency arises? Who will be in charge of the response, and who will communicate to the public? What procedures and equipment will be needed, and is it possible to secure these items in advance? Can the facility design be amended in a manner that either prevents the emergency from occurring or mitigates its severity? It is impossible to plan for every element of every emergency, but emergency responses are more efficient when planning is leveraged to the extent practical.

Engineering controls

Engineering controls work by removing the hazard at the source or isolating people from the hazard (CDC, 2020). The most common engineering controls are shielding, containments, ventilation, and interlocks.

Shielding is accomplished with a physical barrier that blocks radiation. The barrier works because as radiation interacts with the shielding material, it loses some or all its energy. The radiation is said to be attenuated. The effectiveness of shielding is material-specific, and different materials are better suited for some radiations than others. Dense metals such as lead and steel are generally understood to be effective shields against photon radiations including X- and gamma rays. These same materials are ineffective for neutrons and high energy betas, as the former simply bounce off in another direction without losing much energy and the latter are stopped so fast that new photon radiation is emitted which must now be shielded. Neutrons are best shielded with hydrogenous materials such as water and concrete, and betas are effectively shielded with plastic, but photons are not stopped nearly as well by any of these materials. Lead, steel, water, concrete, and plastic will all stop alphas, but so will a standard thickness of paper; thus, complex alpha shields are difficult to justify.

Shielding is characterized as being permanently installed or temporary. A disadvantage of temporary shielding is that it is generally the more mobile of the two types, and if it is inadvertently moved could result in less protection than anticipated. As the nomenclature suggests, temporary shielding is intended to be a used as a temporary control during infrequent or irregular activities, and sites should not rely on temporary shielding as a long-term fix for poor permanent designs that fail to control exposures during routine or recurring activities.

Containments are used when handling hazardous quantities of unsealed radioactive material. They rely on physical barriers to keep contamination from spreading to uncontrolled areas. Containments may be constructed from durable materials, as in the case of permanently installed concrete, steel, or gypsum walls, but temporary containments such as gloveboxes and a variety of flexible enclosures including tents, bags, and catch containments are also popular. In some cases, a series of containments are used. Radioactive material is used within a "primary" containment that provides the main protection, and a "secondary" containment surrounds the primary containment in case the main protection fails or is otherwise breached.

Ventilation removes air from a space and thus any entrained radioactive particles the air contains. In turn, those particles are not present to be inhaled or ingested or to settle out as dust and buildup to hazardous concentrations over time. Ventilation is often used in conjunction with containment, and a fume hood is an example of a control that relies on a combination of containment and ventilation. Ventilation is used with installed filtration that is suitable to the contaminants being exhausted. For example, High

Efficiency Particulate Air (HEPA) filtration is used to remove radioactive airborne particulates but is not effective for radioactive noble gasses.

Interlocks prevent areas from being entered when hazardous levels of radiation may be present. For example, an interlock could cause doors to remain locked when a radiation-producing machine is on but allow the doors to unlock when the machine is switched off. Similarly, an interlock could prevent a machine from starting while a door is open.

Administrative controls

Administrative controls change the way people work (CDC, 2020). The number of important administrative controls is vast, but critical examples include written operating procedures, access controls, surveys and monitoring, decontamination, and waste management. These controls are explained in the following paragraphs.

Procedures

Once a site is designed, planning for operations, maintenance, and modifications begins with development of site-specific operating and radiation safety procedures. These documents describe programmatic elements and serve as formal administrative controls that convey and promote consistent radiation safety practices. In a broad sense, the radiation protection procedures at a nuclear facility will cover topics including program and organizational structure; training and qualification; surveys and monitoring; exposure, contamination, and airborne radioactivity controls; waste management; receipt, storage, and control of radioactive material; emergency preparedness; and recordkeeping, but this list is not entirely inclusive. The procedure list will vary from facility to facility based on site-specific factors.

The first step in developing appropriate procedures is to identify the applicable regulatory requirements for radiation protection and associated standards and guidance documents. For example, the regulatory requirements for a US Department of Energy site differ from the requirements for a US Nuclear Regulatory Commission licensee, so the respective radiation protection programs and corresponding manuals and procedures will differ.

Procedures should not just restate, paraphrase, or cite regulatory requirements and related documents. Rather, they should convey site-specific policies, programs, and instructions that are necessary to ensure compliance with the related regulatory requirements. Because site-specific procedures are better suited to provide protection than a generic approach, each site should have its own radiation protection manual and procedures rather than adopting the procedures used by others.

Access controls

Access is controlled with a variety of tools. Functionally, the number or physical accesses to and from an area containing radioactive material is limited. The limitation can be accomplished by rule or procedure, but barriers are ideal. Robust structural walls provide the best barriers, but poly containments and ropes on stanchions are popular. Painted or taped lines may be used to bound areas where controls apply, but such designations are not as effective as physical barriers that must be moved to gain access. Fig. 2 shows how tape may be used to denote boundaries where control is needed.

As few openings as possible are allowed through the barrier for egress, with consideration given to work efficiency and emergency exit and response, as fewer access points means fewer travel routes for contaminants to be unknowingly spread to other areas and better control, tracking, and monitoring of personnel entering and exiting radiological areas.

Rope barriers are commonly constructed of yellow and magenta twisted polypropylene as this combination is generally recognized as designating an area that is controlled for radiological purposes. Although these ropes are often a sure sign that an area is



Fig. 2 Example taped boundary. Courtesy Ameriphsysics, LLC.

under radiological control, this is not always the case. Other color combinations and materials are also used, a rope's colors are subject to fade over time, and the color combination tells potential entrants nothing about the severity of the hazard within. Consequently, postings and signs are used to convey instructions that are important to entrants.

The most common postings use nomenclature that signifies the magnitude of the risk presented by entry. For example, facilities operated according to US Nuclear Regulatory Commission rules categorize their work areas according to the degree of control that is warranted. At such facilities, a "restricted area" is any area to which access is limited for the purpose of protecting individuals against undue risks from radiation and radioactive materials, a "controlled area" is an area outside of a restricted area which can be limited for any reason, and an "unrestricted area" is an area to which access is neither limited nor controlled (NRC, 1991). Restricted areas are further categorized a radiation areas, high radiation areas, or airborne radioactivity areas according to the radiological hazard they present. US Department of Energy uses slightly different nomenclature than what is described here and as follows, but it is similar in that areas are named according to a graded approach that considers dose or contamination potential.

According to US Nuclear Regulatory Commission rules for restricted areas, a "radiation area" is an accessible area in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.05 mSv in 1 h at 30 cm from the radiation source or from any surface that the radiation penetrates. A "high radiation area" is similar, except that an individual could receive a dose equivalent in excess of 1 mSv. A "very high radiation area" is an accessible area in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 5 Gy in 1 h at 1 m from a radiation source or 1 m from any surface that the radiation penetrates. Finally, an "airborne radioactivity area" means a room, enclosure, or area in which airborne radioactive materials exist in concentrations that approach or exceed certain constraints on such radioactivity.

Radiation, high radiation, very high radiation, and airborne radioactivity areas are required by US Nuclear Regulatory Commission regulations to be posted. The typical posting is comprised of a sign bearing a magenta, or purple, or black three-bladed standard radiation symbol on yellow background and the words, CAUTION, RADIATION AREA; CAUTION or DANGER, HIGH RADIATION AREA; GRAVE DANGER, VERY HIGH RADIATION AREA; or CAUTION or DANGER, AIRBORNE RADIOACTIVITY AREA, as appropriate. Rooms in which licensed material is used or stored are required to be posted with the radiation symbol and the words CAUTION or DANGER, RADIOACTIVE MATERIAL(S).

Fig. 3 shows a typical posting—in this instance a high radiation area. Persons who understand what a high radiation area is are warned of the hazard potential simply by reading the sign, however, anyone recognizing the familiar standard radiation symbol is warned that a radiological hazard is present. For this latter reason, it is common to see postings displaying the radiation symbol used even when not required, such as in controlled areas and restricted areas presenting a dose equivalent less than the defining criteria for radiation areas.

Postings are purposely displayed in conspicuous locations, and the sign bearing the posting is often used to convey additional information regarding controls. Statements such as "no eating, drinking, smoking, or wearing of cosmetics", "dosimetry required for entry", "not an entrance", "contact radiation protection office for entry", and "radiation work permit required for entry" are often used even though not required by regulation.

Although postings convey important access-limiting information to entrants, space limitations restrict such information to concise statements or conclusions that may be inferred by area nomenclature. A radiation work permit, or RWP, is an additional access control that conveys details regarding the requirements for entry. For example, suppose a posted radiation area will be entered by two groups of workers, and one will conduct routine site visual inspections while the other will open potentially contaminated liquid systems for maintenance. The first group may be required to wear dosimetry but could potentially traverse the area without extensive PPE, monitoring, or contamination control training. The second group may require respiratory protective equipment, waterproof coveralls, continuous air and contamination monitoring, and specialized training. As the task-specific controls are too lengthy to attach to a sign, a separate RWP is issued to each group of workers for entries into the same posted radiation area.



Fig. 3 Example posting bearing the radiation symbol. Courtesy Ameriphsysics, LLC.

In addition to being task-specific, RWPs are site-specific. That is, the format and content of such will vary from site to site. Nonetheless, certain common elements are typical of RWPs. First, the content, format, issuance, and usage of the RWP will be described in an administrative procedure. Regarding content, most RWPs will contain space to record expected radiological conditions, such as general area and maximum radiation, contamination, and airborne concentration levels; required PPE; and monitoring requirements. RWPs are approved and rescinded according to a formal process, and the signatures for such are usually located on the form. Finally, anyone entering an area is expected to be briefed on the requirements for entry, so RWPs will contain space for recording that a briefing was held and for entrants to acknowledge by signature that they have attended the brief and have read, understand, and will comply with the requirements for performing work according to the RWP. Fig. 4 shows an example RWP format.

It has already been mentioned that the number of accesses to and from restricted areas are limited, and another common administrative control is to establish an “access control point” at such locations. A control point provides a central location to maintain applicable RWPs, a log of entrants and the time entering and exiting the area, a station for storing, donning, and doffing PPE, and

RADIATION WORK PERMIT			
Project Number:		Location:	
RWP Number:		Task:	
Begin Date:	End Date:	Type:	
		☐ Routine RWP ☐ Job-Specific RWP	
Radiological Conditions			
<u>Radiation Levels</u>	Location	Dose Rate	Limiting Conditions
General Area			
Hot Spot			
<u>Contamination Levels</u>	Location	dpm/100 cm ²	dpm/100 cm ²
General Area			
Hot Spot			
<u>Airborne Activity</u>	Location	% of DAC	% of DAC
General Area			
Maximum			
Protective Equipment			
☐ Latex Gloves	☐ Lab Coat / Apron	☐ Dust Mask	
☐ Rubber Gloves	☐ Coveralls	☐ Half Face Respirator	
☐ Chemical Resistant Gloves	☐ Chemical Suit	☐ Full Face Respirator	
☐ Leather Gloves	☐ Hood / Cap	☐ PAPR	
☐ Booties	☐ Hard Hat	☐ SCBA	
☐ Shoe Covers	☐ Safety Glasses	☐ Other (Describe):	
☐ Steel Toed Shoes	☐ Hearing Protection		
Monitoring			
☐ TLD	☐ Area Air Sample	☐ Continuous HPT Coverage	
☐ SRD	☐ Lapel Air Sample	☐ Periodic HPT Coverage	
☐ Other (Describe):			
RWP Approval	Pre-Job Briefing Conducted	RWP Termination	
PM Signature / Date	PM Signature / Date	PM Signature / Date	

Fig. 4 Example RWP. Courtesy Ameriphsysics, LLC.

equipment for personal monitoring. Access control points are often assigned an attendant who verifies controls, particularly in cases where radiological conditions present a substantive risk.

Surveys and monitoring

Radiological surveys are conducted to comply with regulations, but also as a necessary control to evaluate the magnitude and extent of radiation levels, concentrations or quantities of residual radioactivity, and the potential radiological hazards (NRC, 1991). Radiation and contamination surveys are used to locate radiation sources and quantify dose rate, surface contamination, and atmospheric contamination, but the real benefit of surveys as a control is when they are used to identify changing conditions. That is, an area may be surveyed upon initial entry or before work begins, and then routinely thereafter to determine if operations are causing radiation or residual radioactivity levels to increase or decrease. Surveys are also valuable for identifying areas to be avoided and areas that may be cleaned to reduce exposure. Surveys at control points and other boundaries are valuable for verifying that contamination controls are working, and surveys of personnel and equipment that exit restricted ensure that contamination is not spread elsewhere.

When surveys are performed to verify the effectiveness of containment, contamination limits provide action levels at which an intervention such as decontamination is required to ensure continued control. Surface contamination limits designate acceptable levels of fixed or removable radioactivity, where fixed contamination is tightly adhered to a surface and removable contamination is loose and easily transferred or dispersed. When fixed and removable contamination are measured together, the nomenclature total contamination is used.

Just as surface contamination limits apply to surfaces, airborne contamination limits are used to signify when airborne concentrations are high enough to justify action. The action could be as simple as ventilating or cleaning the space, or as serious as vacating the area or donning respiratory protective equipment.

Personal monitoring is accomplished with survey meters as individuals exit restricted areas to ensure they are not contaminated, with dosimetry to track external dose, and with in vivo and in vitro bioassay to determine internal dose. The term “frisking” is used to describe the process wherein radiation detectors are passed slowly and as close as possible to a person’s body.

The equipment used for surveys and monitoring must be calibrated and in proper working order. Calibration is the checking of the values obtained by a measuring device against known values. In the case of radiation detectors, it is accomplished by measuring the emissions from calibration standards of known radioactivity. The frequency of calibration is a site-specific decision, but an interval not exceeding 1 year is typical. Calibrations for radiation detectors must take into account the energy of the radiation that will be measured. Some detectors are impacted by pressure and temperature, and these factors must be considered when using instruments in environments that differ from the calibration environment, such as when instruments are calibrated near sea level and used at higher elevations.

Calibrations are critical, but calibration by itself is not sufficient to ensure that radiation detection equipment is working properly. The equipment must also be checked regularly prior to use. Typical checks include examining physical integrity, checking batteries, and ensuring that values received when measuring background and a source of known activity are in expected ranges.

Decontamination

Operations may cause small amounts of residual radioactivity to build over time, or maintenance, leaks, or emergencies may result in significant short-term releases of radioactive material. Decontamination is the process wherein some sort of cleaning or remediation effort is conducted to return contamination to acceptable levels. Because decontamination concentrates contaminants onto towels, mops, filters, or other items used in the cleanup, containment of these items and control according to a waste management program are important.

Waste management

Radioactive waste can be collected, chemically and physically altered in form, packaged, and stored, but the radioactivity making the waste hazardous does not become inert via these processes. Since we cannot make radioactivity go away, we must wait for it to decay. For very long-lived radionuclides, this can take millions or billions of years, so it is important that management approaches are planned to protect present and future generations. Consequently, long-lived radioactive waste is usually transferred to specialty facilities that are designed to provide for long-term management. When containers are used to confine waste, the design of such packages must consider situations that might be encountered during transportation and disposal including traffic accidents, emergency braking, rain, etc.

Landfill burial is usually considered the practical endpoint of a waste management program. Engineering controls such as landfill liners and caps, and administrative controls such as site selection, permits, procedures, and monitoring are used to mitigate the hazard presented by the waste the landfills contain. In the end, however, burial is a specialized means of storage rather than a solution. That is, although radioactive wastes can be safely stored in these landfills for generations, the radioactivity still exists, and the public would be better served if the waste had not been generated at all. Consequently, waste minimization is a critical element of any management program as preventing a waste from being generated at its origin is the only completely hazard-mitigating control.

Exposure controls

When radioactivity presents an external hazard, radiation protection is accomplished with practices that control time, distance, and shielding (Cember and Johnson, 2009; Gollnick, 2011). Collectively, these practices are called “exposure controls”. In simple terms, exposure is minimized when time is also minimized and distance and shielding are maximized.

Time

The hazard presented by an exposure of a constant intensity is directly related to the duration of the exposure, meaning that if the time spent near a source presenting an external exposure is halved, the total exposure and corresponding risk are also halved. Similarly, if the time is doubled, the total exposure and risk are doubled. The manner in which exposure and time are related is easy to conceptualize, making practices that aim to minimize time practical exposure controls. Typical controls include prohibiting loitering near radioactive material and making sure workers understand and have practiced the work to be done (e.g., with mockups).

Distance

Imagine a baseball-sized mass of material that uniformly emanates gamma radiation in all directions. A person standing very near that source, just a few centimeters away, will be traversed by many of the gamma rays as they escape the surface, whereas few rays would point to and interact with a person standing across the room and away from the source. A simplification of the relationship between exposure and distance is that exposure is inversely proportional to the square of the distance of separation. Thus, if distance is doubled, then the exposure decreases to one-fourth of its original value. For example, if the previously described baseball-size source is delivering an effective dose 4 mSv per hour to a person working 1 m away, that person can reduce the dose rate of the field they are working in to 1 mSv per hour by doubling their distance—or working 2 m away from the source instead of one.

The extent to which this inverse-squared relationship is applicable depends on the geometry of the source relative to the distance at which the exposure occurs. Generally, the relationship works when the exposure distance is at least three times the largest dimension of the source and the source can be treated geometrically as a point source (Gollnick, 2011). Long-handled and remote-operated tools are a means of maximizing distance. Such tools allow workers to distance themselves from radioactive work surfaces that present a concern.

Shielding

Relying on the baseball-sized mass as an example again, a shielding material could be placed between a person and the source to shield the person from the gamma radiation emitted. If the gamma rays interact with the shielding in a manner in which all of their energy is lost, then they are prevented from interacting with and causing harm to the person. Similarly, if the gamma emissions are only partially blocked, then the person is exposed to the rays that penetrate the shielding.

Alphas and betas are charged particles and thus have predictable, finite ranges they can travel through shielding. This means that charged particles can be completely shielded. For radiations not characterized as charged particles, such as x-rays, gamma-rays, and neutrons, the amount of shielding needed to reduce an exposure is calculated with the concepts of half-value layer (HVL) and tenth-value layer (TVL). As the names suggest, each corresponds to the thickness of the layer needed to reduce the intensity to one-half or one-tenth of the original exposure, respectively. As an example, if a beam of gamma radiation is aimed at a shield twice as thick as the TVL, a beam 10% of the original intensity (one-tenth of 100%) will penetrate the first half of the shield, and a beam of 1% of the original intensity (one-tenth of 10%) will survive the entire shield thickness. Note that no matter how many HVLs or TVLs are used, mathematically the number of penetrating emissions is never reduced to zero. Nonetheless, a shielding thickness of 10 TVLs will reduce even deadly amounts of radiation to inconsequential levels, and five to seven TVLs are protective in all except the most extreme instances.

HVLs and TVLs vary numerically based on the composition of the shielding material and the type and energy of radiation involved. Once the value reflecting an HVL or TVL is known, however, that thickness works the same whether the shielding is placed close to or away from the source. From a design standpoint, this means that placing the shielding as close to the source as possible is often the best practical solution. As an example, consider the feasibility of building a two-inch thick lead box to contain a baseball-sized radioactive mass versus the complexity of surrounding the entirety of the same source by a room equipped with two-inches of lead on all its walls, floor, and ceiling when both designs accomplish the exactly the same amount of shielding and protection. On the other hand, if a room contains a thousand of such sources, shielding the room or using a combination of source and room shielding becomes reasonable.

Contamination controls

Unlike exposure controls that are concerned with external radiation, contamination controls are used to mitigate the internal hazard presented by radioactive material that is deposited on or within the body. In terms of possible harms, dose is dose, meaning the

effective dose from internally deposited radioactivity does not present an inherently different risk than the same amount of external dose. Whereas external dose quits accumulating as soon as the area containing the source is exited; however, internally deposited radioactivity will continue to deliver a dose as long as it is in the body, and in some instances this can be long after the exposed person vacates the area originally presenting the hazard.

Contamination controls are designed to prevent contaminants from entering the body through three pathways: inhalation, ingestion, and absorption through the skin (Cember and Johnson, 2009). This is accomplished by engineering and administrative controls that confine and enclose the source or PPE such as anti-contamination clothing and respirators that provide a barrier between the exposed person and the contamination (Cember and Johnson, 2009).

Confinement

Confining the source begins with designating areas for handling radioactivity, particularly unsealed sources. The buildings and rooms in which radioactive materials are used, including the ventilation, drains, and emergency systems they contain, are designed in a manner that leverages engineering controls to keep contamination contained during normal operations and certain emergencies. But confinement goes a step farther than designing and designating rooms; smaller areas within rooms such as hoods, certain bench tops, or other precise locations are further designated that can be equipped with contamination-limiting controls. For example, if the material presents a potential for emitting radioactive fumes, filtered ventilation may be situated near the release point rather than relying on the normally located room exhausts. Previously described containments and trays and pads that are used to cover benchtops are other sources of confinement that can be used within rooms.

When materials are used in the practice of confinement, such as in the cases of ventilation filters, portable containments, and pads, they are expected to become contaminated because of their use. Repair, replacement, cleaning, or maintenance involving such items presents a special case for containment as their mishandling may cause accumulated contamination to be released. Depending on the manner and concentration of the release, the resulting exposure may far exceed the dose that the confinement was designed to avoid.

Personal protective equipment

No containment system is 100% effective all the time, and even the most robust engineering controls are subject to failure. Consequently, PPE is worn when handling contaminated materials, when working in contaminated and airborne radioactivity areas, and as otherwise directed by the radiological control organization or as required by an RWP. Note that as described at the onset of this article, PPE is considered the least effective of the three types of control measures and thus is not a suitable substitute for proper planning that confines the hazard with engineering and administrative controls.

The term PPE is used to describe any equipment that is worn by an individual to prevent exposure to workplace hazards. In the usual sense where radioactivity is a concern, PPE is used to prevent contamination of a person's skin or clothing or inhalation or ingestion of radioactive material. That is, PPE is not usually worn as a radiation shield, except that lead aprons are sometimes used in the presence of X- and gamma-radiation and plastic eyewear is an effective shield against beta radiation to the eyes.

The simplest PPE ensemble that is protective against radioactive contamination consists of a water-repellent gown or coveralls, safety glasses or goggles, surgical-style gloves, and shoe covers. The coveralls may be equipped with a hood or used with cap, and the gloves and shoe covers are often taped to the coveralls. Protective clothing can be reusable or disposable, and the former is laundered at facilities specially designed to collect and manage radioactivity in wash water.

Respirators are a special type of PPE that control internal exposure by limiting the intake of radioactive material via inhalation or ingestion. Numerous respirator types operate via a variety of operating modes, and each combination is associated with a performance characteristic called protection factor. In order of increasing protection factor, respirators are generally categorized as being air purifying, atmosphere supplying, or self-contained breathing apparatus (SCBA). Because air purifying respirators rely on filtration, they are only effective where airborne contaminants present in a particulate form. Atmosphere-supplying respirators and SCBA are protective against particulates but are also effective when the contaminant is a gas or a vapor. Misuse of respirators or their mechanical failure during use could expose workers to contaminated atmospheres in a manner leading to dire consequences. Consequently, respirator use is accomplished according to robust respiratory protection programs that include controls such as training, maintenance, and inspection requirements.

Assigning respiratory protective equipment is not a control decision that is taken lightly. Respirators make work harder, and they make it difficult to see and breathe. As such, respirators are only assigned to persons who have been medically cleared and trained on the respirators they will be using. Because the respirator only works if it fits and seals properly, wearers are also required to receive and pass a fit test with the exact respirator type and size that they will be using.

Just as a containments become contaminated through their use, PPE is expected to become contaminated as it is used. Wearers must be careful not to contaminate themselves when they are removing such clothing, and collecting and managing the waste presented by used PPE is important.

Optimization

In the end, the best radiation protection programs are site-specific and rely on a combination of the engineering, administrative, and PPE controls described herein. The fact that such controls are planned and used, however, does not mean they are working in an optimal manner. Moreover, the implementation of controls aimed at mitigating radiological risks may cause or exacerbate other hazards as discussed in the article on risk tradeoff. Consequently, performance assessment is used to verify that controls are operating as expected and to seek opportunities for improvement. The Integrated Safety Management platform US Department of Energy uses to integrate safety into management and work practices at all levels is an example of an assessment process that invokes principles of continuous improvement.

Assessment is accomplished with formal surveillance programs that observe people and their environments with techniques aimed at determining the effectiveness of controls. Because a key objective of performance assessment is continuous improvement, surveillance programs are designed to regularly reexamine performance. Finally, such surveillances are expected to return findings and offer recommendations. That is, assessments that always show perfect performance should be considered evidence of an underperforming surveillance program rather than of perfectly optimized and executed radiological controls.

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Radiation Shielding Methods

Ronald Pevey, Nuclear Engineering Department, University of Tennessee, Knoxville, TN, United States

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Introduction

The world is now full of sources of radiation, both man-made and natural; many of them are purposefully created for medical, energy, national defense, and industrial uses. The field of radiation shielding involves the protection of people (and, to a lesser extent, equipment) from the danger posed by the deposited energy of this radiation.

The need for shielding arises because radiation energy that is produced in one place is deposited at distant points, with the distances involved proportional to distances at a human scale. Some radiation particles (e.g., low-energy alpha particles) deposit their energy so close to their point of creation that there is no need for shielding from an external source beyond a piece of paper to protect personnel (although ingestion and inhalation can cause damage); others (for example neutrinos) deposit their energy so far from their point of origin that it is futile to try to protect personnel from them.

The primary need for shielding on the Earth's surface arises for protection from uncharged particles, primarily neutrons and gamma rays (although some high-energy electrons need direct shielding as well). But the primary task for shield designers is concentrated on uncharged particles that have average traveling distances between interactions with material (referred to as *mean free paths*) in the range of centimeters to meters. The scale of these mean free paths make it both necessary and effective to create material shields for protection of nearby personnel.

The physical reason why neutral particles travel so much farther than charged particles is that the neutral particles have to directly impact the sparse nuclei in materials through which they pass in order to interact with them, whereas charged particles electrically interact with the electrons and protons in atoms near their path, in the process depositing their energy in a far shorter distance. Charged particles also collide with the atoms of the materials through which they pass, but this is a much less significant source of energy deposition.

So we will concentrate our attention on the methods that have proven themselves useful in predicting the damage caused by neutrons and photons. Shielding against high energy charged particles is discussed by (Heilbronn, 2021).

Basic principle of radiation shielding: Conversion of particle flux to dose rate

The actual damage to biological tissue is produced by the deposited energy of the short-range charged particles that are created when neutrons or photons interact with matter. These secondary particles are not as penetrating as the neutral particles that produced

them, but they cause far more damage to living structures because their energy is deposited in a concentrated manner, over a short distance (often just a single or a few biological cells).

Although the purpose of radiation transport calculations is to predict biological effects (or other physical properties), we get to that goal by the calculation of *particle flux* (denoted by the Greek symbol ϕ), which is defined as particle density (particles/unit volume) times velocity (distance travelled/unit time), resulting in a measure of the particle path length per unit volume per unit time; this results in flux units of particles/unit area/unit time, usually expressed in units of particles/cm²/sec.

This flux is then translated into a biological effect by multiplying by a factor that is generically referred to as a *response function*; particular response functions are deduced from physical or biological experimental data. This approach works both for determining rates (e.g., dose per unit time) and for the accumulation of effect over time (total dose), depending on whether the particle source rate is given in source rate (particles/unit time) or total particles released during a limit-time event (like a burst). In the latter case, the flux would not be a time rate, and is referred to as a *fluence* with units of total particles/unit area (usually particles/cm²).

The resulting dose rates are usually expressed in either units of sieverts (Sv) or the older unit of *rems*; there are 100 rems in a sievert. For photon dose, a sievert corresponds to the biological effect of a deposited energy of a *gray* (Gy), defined as the deposition of 1 J/kg of tissue. For neutrons, the conversion rate from gray to sievert is usually greater than one (referred to as a relative biological effectiveness) due to the greater biological damage caused by charged particles produced by neutron interactions with the tissue nuclei.

Basic tools of radiation protection: Time, distance, shielding

There are three primary defense mechanisms that protect people from radiation: time, distance, and shielding.

Time

The first of the defense mechanisms is time. In most cases (burst-type accidents being the primary exception), a neutron source or photon source consists of radioactive material that emits particles at a rate that decreases as the radioactive source material decays. This source rate produces a dose rate, e.g., in units of Sv/sec. Since our goal is the reduce dose (in Sv) received by personnel, the first defense from the radiation is to minimize the time spent in places with high dose rates.

In a typical shielding analysis, the analyst would produce a map of dose rates by position and this would be combined with occupancy information for both radiation workers and the public to determine expected total doses received by personnel.

Distance

The second defense mechanism is distance. In an idealized situation, a source that emits all of its particles from a single point, equally in all directions, at a constant rate of S particles/sec would—in the absence of any shielding—produce a dose rate that decreases as a factor of distance squared. This can be easily deduced from the fact that at a distance of r cm from the source point, the spreading particles would be evenly distributed on the surface of a hypothetical sphere of area $4\pi r^2$ (with the source point at the center of the sphere.)

For example, in the idealized situation where a million particles/second are being emitted from a given point source, the flux at a distance of 50 cm from the point would be the emitted particles spread out over the area of a sphere of radius 50 cm, resulting in a flux given by:

$$\phi = \frac{10^6 \text{ particles/second}}{4\pi(50 \text{ cm})^2} = 31.83 \text{ particles/cm}^2/\text{sec}$$

If the million particles per second being emitted are photons with an energy of 2 MeV (MeV), using the data presented later (in **Table 1**) gives us a dose rate of:

$$\begin{aligned} \text{Dose rate} \equiv D &= R\phi = (7.45 \times 10^{-12} \text{ Sv} \cdot \text{cm}^2) \times (31.83 \text{ photons/cm}^2/\text{sec}) \\ &= 2.37 \times 10^{-10} \text{ Sv/sec} \end{aligned}$$

At first glance this seems like a very small dose rate, until you realize that Sv is a relatively large dose unit and second is a very short period of time; in the United States, radiation dose limits are usually expressed in mrem/year. Since a rem is 0.01 Sv and the typical work year is 2000 hr, the dose rate would translate into:

$$\begin{aligned} \text{Dose rate} &\equiv 2.37 \times 10^{-10} \text{ Sv/sec} \times \frac{100 \text{ rem}}{1 \text{ Sv}} \times \frac{1000 \text{ mrem}}{1 \text{ rem}} \times \frac{3600 \text{ sec}}{1 \text{ hr}} \times \frac{2000 \text{ hr}}{1 \text{ year}} \\ &= 171 \text{ mrem/year} \end{aligned}$$

A person exposed to this dose rate for a full year would receive a bit more than half of the average annual dose rate in the U.S. from environmental radiation sources of 310 mrem/year (Wahl, 2010).

Table 1 Photon response functions and sample linear attenuation coefficients.

Energy (MeV)	Response function $R(E)$ (10^{-12} Sv-cm ²)	Linear attenuation coefficient, $\mu(E)$, cm ⁻¹		
		Concrete	Iron	Lead
0.01	0.0496	60.1	1334	1427
0.02	0.211	7.99	198	954
0.05	0.363	0.813	14.4	83.8
0.1	0.519	0.383	2.63	60.6
0.2	1.00	0.285	1.07	10.6
0.5	2.47	0.200	0.649	1.70
1	4.48	0.146	0.469	0.772
2	7.45	0.103	0.335	0.514
5	13.7	0.0666	0.248	0.484
10	23.0	0.0532	0.236	0.564

Data derived from Shultis and Faw (2000).

Shielding

The third defense mechanism is the placement of shielding material of some sort between the particle sources and the places that need to be protected. Predicting the effect of such shielding can be very complex because of the different physical consequences of radiation interacting with material. For example:

- The particle can be absorbed, thus providing loss of the particle;
- The particle can scatter off the material nuclei, changing direction and energy; and
- The particle can interact with an atom of the shield material and generate more particles of the same or different types.

The complex combination of space, direction, and energy effects of particle interaction with the shield material makes the determination of the resulting dose rate maps more complex than the simple consideration of source rate and distance. Of special concern are the dose rates from neutron sources, because allowance must be made for the photons that are produced by neutrons undergoing scattering and neutron absorption. This generally means that complicated computer codes need to be utilized to get accurate dose rate maps.

It is possible to use more approximate methods to check the reasonableness of the more complex computer analyses or even (for simple geometries) to get answers for which a safety factor can be utilized to get satisfactory results for design purposes.

Result: Dose rate map

A typical result of a shielding analysis is an area dose map, which gives a graphical representation of how the dose rate varies by space—at a given elevation (e.g., 100 cm above the floor for personnel working on the same level as the source). Careful attention to the dose maps will reveal the effects of distance, shielding, and the effect of particles reflecting off of the floor, ceiling, and walls.

Hand calculations of spatial dose rates

Hand calculations can be effective in determining the dose effect of shielding in simple arrangements and in estimating the dose rate in more complicated geometries as a check of the results from more accurate methods.

We shall focus on a simplified situation, often encountered in practice, of a shield of a known material and thickness from a small source, approximated by assuming the source particles are emitted from a single point.

Uncollided dose rate

The first approach is to determine the dose impact of the original particles emitted from the source, ignoring the contribution of particles that have scattered or have been produced from particle reactions. In effect it is what the dose rates would be if particles interacting with nuclei in the shields were always absorbed. This is the portion of the dose rate produced by original source particles that have passed through the shield without colliding with any of the atoms in the shield.

The uncollided dose rate, due to both distance attenuation and shielding attenuation of a source of energy E , can be very easily shown to follow the relation:

$$D_0(r, E) = \frac{S_p R(E)}{4\pi r^2} e^{-\mu(E)t}$$

where: S_p is the source intensity in particles/second of energy E ; $R(E)$ is the flux-to-dose conversion factor in $\text{Sv}\cdot\text{cm}^2$ for particles of energy E ; r is the distance between the source and the dose point in cm; t is the straight-line thickness of shielding material between the source and the dose point in cm; $\mu(E)$ is the linear attenuation coefficient for particle of energy E in the shielding material in cm^{-1} ; and $D_0(r, E)$ is the resulting dose rate due to the original uncollided source particles of energy E in Sv/second .

Typical values for $\mu(E)$ and $D_0(r, E)$ is provided in (Shultis and Faw, 2000) and (ENDF Data—T-2). A subset of these data is given in Table 1 for photons and Table 2 for neutrons. These data clearly illustrate how the interaction coefficient decreases with particle energy (i.e., higher energy particles are much more penetrating).

Buildup of scattered photons

Calculating the dose contribution of the particles that come directly from the source is straightforward because the dose reduction depends only on the straight-line path between the source and the dose point. Accounting for the contribution to the dose of source particles that are scattered from the shield nuclei or that are created from other particles emitted as a result of the original particle reactions (like the photons produced from neutron absorption and inelastic scatter) is more complicated.

For the case in which the intervening material is a single homogeneous material—often a good approximation of shielding situations—some rough accounting for the effects of scattered photons can be made through the use of *buildup factors*. These are tabulations (or function fits) of the ratio of the total dose to the uncollided dose as a function of initial particle energy and uncollided attenuation of the intervening shielding. A common way of accounting for this buildup is to use the two parameter Taylor formula, given by:

$$B(E_0, \mu(E_0)t) = A_1 e^{-\alpha_1 \mu(E_0)t} + A_2 e^{-\alpha_2 \mu(E_0)t}$$

where: $B(E_0, \mu(E_0)t)$ is the buildup factor; E_0 is the initial energy of the emitted photons; $\mu(E_0)t$ is the linear attenuation factor times straight-line shielding distance (i.e., the same factor used in the exponential term of the uncollided dose rate calculation); A_1, α_1, α_2 are source-energy-dependent parameters; $A_2 = 1 - A_1$ (to preserve the initial value $B(E_0, 0) = 1$).

Notice the difference between r (the total distance from the source to the desired dose point) and t (the portion of this distance corresponding to the shielding).

Typical values for the parameters of the equation are provided in (Shultis and Faw, 2000). A subset of these data is given in Table 3.

The resulting determination of the total dose is then found from the relation:

$$D_{\text{Total}}(r, E) = D_0(r, E)B(E_0, \mu(E_0)t)$$

Tenth value thicknesses for neutrons

For neutrons, the same equation for the uncollided dose contribution can be used (with appropriate values of the linear attenuation coefficients and flux-to-dose conversion factors), although accounting for the dose contribution of scattered neutrons is usually more difficult than for photons. One common technique for approximating neutron and photon dose rate reduction through shielding is to specify *tenth value thicknesses*—corresponding to the shield thickness of a given shielding material and particle source energy that would produce a total dose reduction of a factor of ten; these tenth value thicknesses can be determined either experimentally or through computer calculations. Table 4 gives the tenth value thicknesses for several materials for both neutrons and photons; these were determined from the most conservative (highest) asymptotic dose reduction factors over a range of particle energies from 0.01 to 10 MeV. Of particular note is how effective iron and lead are for shielding from photons. (The corresponding

Table 2 Neutron response function and sample linear attenuation coefficients.

Energy (MeV)	Response function $R(E)$ ($10^{-12} \text{ Sv}\cdot\text{cm}^2$)	Linear attenuation coefficient, $\mu(E)$, cm^{-1}		
		Concrete	Iron	Lead
2.5×10^{-8}	10.2	0.481	1.182	0.376
1×10^{-7}	10.2	0.415	1.066	0.373
1×10^{-6}	12.4	0.391	0.988	0.370
1×10^{-5}	12.4	0.388	0.962	0.370
1×10^{-4}	12.0	0.386	0.927	0.368
1×10^{-3}	10.2	0.383	0.735	0.362
1×10^{-2}	9.90	0.367	0.575	0.350
1×10^{-1}	60.0	0.294	0.310	0.308
1	370	0.488	0.208	0.158
5	430	0.130	0.298	0.243
10	410	0.120	0.256	0.167
14	580	0.124	0.219	0.176

Data derived from Shultis and Faw (2000) and ENDF/B-VIII.0.

Table 3 Sample of photon buildup factor equation parameters.

Energy (MeV)	Concrete			Lead		
	A_1	α_1	α_1	A_1	α_1	α_1
0.015	1.110	0.0000	0.1166	1.020	−0.0003	0.2000
0.02	1.192	−0.0010	0.2185	1.020	−0.0003	0.2000
0.05	3.846	−0.0241	0.2267	1.040	−0.0012	0.2500
0.1	43.630	−0.0461	0.0000	0.643	−0.4470	−0.6470
0.2	95.634	−0.0779	−0.0505	1.400	−0.0010	0.5550
0.5	68.606	−0.0685	−0.0378	2.266	−0.0106	0.1314
1	53.820	−0.0439	−0.0174	6.000	−0.0091	0.0529
2	34.930	−0.0262	0.0000	11.660	−0.0206	0.0149
5	8.788	−0.0271	0.0364	3.390	−0.0924	0.0000
10	5.590	−0.0259	0.0339	2.391	−0.2140	−0.0920

Data taken from Shultis and Faw (2000).

Table 4 Tenth value thickness for various materials.

Material	Tenth value thickness (cm)	
	Neutron	Photon
Concrete (regular)	28	52
Iron	63	11
Lead	81	5
Water	49	141
Polyethylene	53	167
Polyethylene with 5 wt% B ₄ C	40	142

half value thicknesses—the approximate shield thickness that reduces the dose rate by a factor of two—can be found by multiplying the tenth value thicknesses in the table by $\log_{10}(2)$, which is about 0.3010.)

It should be noted that the tenth value thicknesses are asymptotic values; that is, they are more useful for estimating the effect of adding shield material onto an existing shield than they are for determining the effect of the initial shielding layers. As will be shown in “Sample Problem #2” section (see Fig. 4), dose rate in the shield does not reduce proportional to the tenth value thickness until the shield has significant thickness.

Room reflection

The preceding results—both photon buildup factors and neutron/photon tenth value thicknesses—are based on the common situation in which the shield provides complete protection between the particle source and the detector point (i.e., no particles can scatter around corners or through holes in the shielding).

But for situations in which the particles can find other paths to the desired dose point—doors, equipment or ventilation ducts, serpentine shielding designs, etc.—a significant dose contribution can come from particles that find their way to a dose point through these indirect paths. This is especially true for neutrons, which tend to backscatter more than photons. In such situations, the hand calculations can give the analyst a rough idea of the expected dose rates, but more accurate computer calculations are required to deal with these more complex configurations.

Shield design strategies

Both neutrons and photons have the property that the lower the particle energy, the more readily it is absorbed. Therefore, in selecting shield materials, the general principle is to select shield materials that effectively reduce particle energy and then absorb the particle once its energy is reduced.

For high-energy photon sources, the selection of materials is simplified by the fact that the most effective materials for reducing the photon energy are the same materials that most effectively absorb photons: dense, high-Z metals like iron and lead. These materials are effective in reducing photon energy because of their high electron density and are effective in absorbing photons because of large photoelectric effect interaction coefficients (This will be illustrated in “Sample Problem #1” section.).

For high-energy neutron sources, the selection of shielding materials is complicated by a number of physical considerations:

1. The most effective materials for slowing down neutrons above about 2 MeV are materials that have a high inelastic scattering probability. These tend to be high-Z metals (like is best for photons). However, since inelastic scattering is a threshold reaction

- (i.e., that once a neutron energy gets below a minimum value, the reaction is no longer possible), this material should be the first shield material encountered, but only thick enough to be the first collision the neutron has with the shield.
- Below the inelastic scattering threshold energy, the most effective atoms for slowing down the neutrons shifts to the other end of the periodic table—the low-Z materials (hydrogen being the most effective).
 - After neutron energy has been reduced, the most effective material are those special nuclides (like boron and cadmium) that have abnormally high neutron absorption probabilities for slower neutrons.
 - Beyond these considerations for the neutrons, the shield designer has to deal with the fact that neutron inelastic scattering and neutron absorption reactions both release photons. Therefore, effective photon shielding material has to be built into the neutron shielding as well.

The result of this is that neutron shielding tends to be combinations of high-Z, low-Z, and special neutron-absorbing materials, either in layers or mixed together. Special mixtures of concrete with steel aggregate (and sometimes with boron carbide added) have been found effective for neutron shielding (This will be illustrated in “[Sample Problem #2](#)” section.).

Computer codes

As with other engineering calculations, hand calculations of radiation transport (and resulting dose effects) have been systematically replaced with computer calculations as computers have gotten faster and more common. And, as with other engineering calculations, the approximate tables and function fits appropriate for use in design and analysis have both been transferred to more efficient computer implementations and improved upon by development of computer codes that utilize more rigorous transport calculations.

Automation of hand calculations

The most straightforward application of computational power to radiation shielding analyses is to simply port the hand calculation methods (described earlier) to the computer. Although this can be done through entering the appropriate data and equations into a spreadsheet, much of the tedium of such a process has been automated into available computer software (most of it sold commercially).

Software (e.g., MicroShield) is available that automates the uncollided and buildup equations described earlier for point sources shielded by lead, concrete, or water, and that commonly extends the capability with:

- Built-in source geometries beyond the simple point source described above—line sources, area sources, volumetric sources.
- Built-in shielding materials (usually with the user having the capability to describe other shielding materials) complete with data libraries for the included materials.
- Data libraries themselves, that not only include the attenuation, buildup, and dose rate conversion factors discussed previously, but also automate the determination of the source rate (yields, energies, and geometries).
- The capability to calculate the decay of the source materials, including daughter radionuclide production and the resulting effects on particle sources and decay heat production.
- Extensive capability to list and plot resulting dose rates for the convenience of inclusion in engineering reports, as well as advanced capabilities (useful to shield designers) to determine sensitivities of the dose to time, source dimension, shielding thickness or distance.

Solution of the radiation transport equation

A more rigorous and accurate approach to determining the dose rate maps is to solve the underlying equation that governs radiation transport, the Boltzmann Transport Equation ([Lewis and Miller, 1993](#)). Originally developed by Boltzmann in the late 1800s to apply to the diffusion of gasses through the atmosphere, this equation was adapted for use in nuclear reactor design, radiation shielding, and criticality safety applications in the 20th century,

The equation is:

$$\hat{\Omega} \cdot \vec{r} \phi(\vec{r}, \hat{\Omega}, E) + \int_0^{\infty} \int_{4\pi} \Sigma_s(\vec{r}, \hat{\Omega}' \rightarrow \hat{\Omega}, E' \rightarrow E) \phi(\vec{r}, \hat{\Omega}', E') d\Omega' dE' + \chi(E) \int_0^{\infty} \nu \Sigma_f(\vec{r}, E') \phi(\vec{r}, E') dE'$$

where $\vec{r}, \hat{\Omega}, E$ are the position, direction, and energy of the particle; $\phi(\vec{r}, \hat{\Omega}, E)$ is the angular flux at the specified point, direction and energy in particles/unit area/unit solid angle/unit energy; $\phi(\vec{r}, E)$ is the scalar flux at the specified point and energy (found by integrating the angular flux over all directions) in particles/unit area/unit energy; $S(\vec{r}, \hat{\Omega}, E)$ is the particle source rate in particles/unit volume/unit solid angle/unit energy; $\Sigma_s(\vec{r}, \hat{\Omega}' \rightarrow \hat{\Omega}, E' \rightarrow E)$ is the coefficient (referred to as a *cross section*) to be multiplied by the flux at point $\vec{r}$ to determine the scattering from $(\hat{\Omega}', E')$ to $(\hat{\Omega}, E)$ in units of probability/unit distance travelled/unit solid angle/unit energy; $\nu \Sigma_f(\vec{r}, E')$ is the cross section used to determine the number of neutrons produced from fission by a neutron of energy E' at point $\vec{r}$; and $\chi(E)$ is the energy distribution of fission-born neutrons per unit energy.

In the general case, there is no closed-form solution for the flux in the above equation, so the solution to the equation must be approximated. There are two principal approaches for implementing this approximation that have proven useful: deterministic and stochastic.

Deterministic calculations: The discrete ordinates method

In deterministic solutions to the transport equation, each of the phases (space, energy, and direction) are attacked using traditional numerical methods. Interestingly, experience has shown that each of them are better approached with different numerical approaches.

For each of the numerical treatments, the general goal is to replace functions of continuous variables (each of which can be evaluated at an infinite number of values of the variable) to a finite set of values that deliver an “accurate enough” approximation to the underlying functions.

Energy: the multigroup method

The energy dependence of the flux is approximated by dividing the energy range into a finite number of *energy groups* with interaction probabilities in each of the groups that can accurately reproduce the reactions rates for the energies within the group; this is referred to as the *multigroup method*. For most applications, this determination of multigroup parameters is performed with a combination of (1) pre-calculated values to account for the smooth, problem-independent energy dependence of the interaction probabilities (cross sections) and (2) special techniques to account for the problem-dependent contribution due to the presence of energy resonances—spikes or dips in interaction probabilities—that occur at specific energies characteristic to the isotopes in the materials.

The determination of appropriate multigroup parameters is seldom performed by the analyst, but the multigroup cross sections for common isotopes are collected into multigroup libraries for general use and available to users from sources like the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory in Oak Ridge, Tennessee.

It is the analyst’s responsibility to choose a multigroup library appropriate for the problem under consideration from the set of data libraries available from RSICC; in general, the important considerations for this choice are that the isotopes needed to describe the materials of the problem are included in the library and that the number of energy groups is sufficient to deliver the accuracy required by the analyst.

Direction: Quadrature integration

The directional dependence of both neutron and photon fluxes is approximated using a quadrature integration technique referred to as the *discrete ordinate method*. In this method, the flux is calculated at a finite number of particular (“discrete”) directions (“ordinates”) and these are combined using predetermined weights to estimate the direction-integrated (scalar) flux that is multiplied by the dose response functions to obtain doses or dose rates.

It is the analyst’s responsibility to choose a quadrature appropriate for the problem under consideration (usually from a set of quadratures built in to the computer code); in general, the higher the number of directions used, the greater the accuracy obtained (at the cost of increased computer time).

Space: Traditional finite difference

The spatial dependence of the neutron and photon fluxes is approximated using traditional finite difference techniques that are carefully formulated to conserve particles—that is, to assure that neutrons and photons are not non-physically created or destroyed due to the numerical treatment.

This conservation is usually accomplished by integrating the spatial flux over regularly-shaped spatial cells that contain a single material and then relating the cell averaged flux (for a particular direction and energy group) to the surface fluxes on each boundary of the cell in order to conserve particles.

These particle balance relationships for each cell result in an underdetermined numerical situation (that is, more unknowns than equations), so approximate auxiliary equations must be defined and incorporated into the equations. The most basic of these auxiliary equations is the diamond difference relations, in which the average cell flux is set equal to the average of incoming and outgoing boundary fluxes in each coordinate direction. This (or a similarly geometrically based) set of auxiliary equations generally restricts the problem geometry to regular orthogonal cells, limiting the ability of deterministic methods to model complex geometries.

Once the numerical approximations are made in the spatial, energy, and directional domains, the result is a large and sparse matrix equation. The solution of this linear algebra problem gives the average flux for each of the cells, energy groups, and directions; these can be used to approximate the true flux and from this produce an estimate of dose rates throughout the modeled geometry.

Advantages and disadvantages

The principal advantages of the deterministic discrete ordinates approach are:

- If the problem can be usefully simplified to one or two dimensions (e.g., a single wall, a set of concentric spheres, or a set of cylinders sharing the same axis), the resulting matrix relationship can generally be solved fairly quickly on a modern computer, not only allowing for quick solutions, but also for efficient parameter optimization.

- The resulting fluxes are found throughout the calculational geometry, making it well suited for the production of spatial dose maps.
- The accuracy versus discretization will generally increase by a factor of $1/N$ or greater, where N is the number of unknowns to be determined (i.e., product of spatial, energy, and directional subdivisions).
- The method can be effectively applied to adjoint flux. The equation that is the mathematical adjoint to the Boltzmann equation. Its solution measures the particle “importance” or effectiveness for contributing to the parameter of interest, such as dose. The adjoint flux is often precalculated and used to reduce the variance of Monte Carlo calculations (discussed below).

The disadvantages of the methods are:

- For 3D problems, the method is generally no faster than the more accurate Monte Carlo method.
- The analysis generally requires approximation of the problem geometry.
- The required discretization reduces the resolution of the answer in space, energy, and direction.
- Flux and dose profiles from two or three dimensional problems show the problem of ray effects, in which the limitation of particles to travel in only particular directions tends to result in profiles that show “waves,” with peaks in the calculated directions and troughs between them.
- The large number of unknowns requires large computer memory resources.
- There is no indication of the accuracy of the solution on the simplified model, so the user must estimate this by some other means (e.g., repeat calculations with different discretization).

Stochastic calculations: The Monte Carlo method

The stochastic (or *Monte Carlo*) method of calculating radiation transport involves a completely different approach to the solution of the Boltzmann transport equation. Although the Monte Carlo algorithm can be derived directly from the transport equations, the resulting random walk can be understood as a *simulation* of the physics of particle interactions.

Simulation of a radiation particle history

The Monte Carlo technique follows the lives (or *histories*) of multiple particles—simulating as closely as possible the actual events that would occur in the life of a particle, with the likelihood and outcome of those events following the natural distributions as closely as they are known. As the particle histories progress, the computer code collects information of interest to the analyst—reaction rates, reactions outcome, dose contributions, etc.—for each particle history and presenting an average of these contributions as an estimate of what the total physical system would produce. These information collections are referred to as *tallies*.

In contrast to the deterministic method presented previously, no numerical approximation is called for—except for what might be present in our best estimate of the natural distributions. Therefore, the results have the potential to be as accurate as possible with our current information. In fact, the principal source of inaccuracy in a Monte Carlo calculation generally comes from the fact that only a finite number of particle histories can be simulated in a finite amount of time. But other than this limitation, the Monte Carlo method is the so-called “gold standard” for solving reactor, radiation shielding, and criticality safety problems.

Of particular contribution to this preference for Monte Carlo solutions is the fact that complicated three-dimensional geometries can be modeled with much greater accuracy than with deterministic methods and the fairly complicated energy dependence of interaction cross sections (particularly for neutrons) can be directly incorporated into the calculation—without the need for the multi-group energy treatment.

In general, a particle history consists of using appropriate probability distributions to simulate the following events:

1. The original location of the particle.
2. The original energy of the particle.
3. The original direction that a particle is traveling.
4. The distance to the first (next) collision in the material.
5. The type of reaction that occurred in the material at the collision site.
6. (If the type of reaction results in the scatter of the original particle or emission of a different particle) The type, energy, and direction of the new particle emerging from the collision.

At this point, the decision tree returns to Step 4 for the next collision. This loop continues until either the type of reaction in Step 5 is a capture (without re-emission of any particle) or the distance found in Step 4 results in the particle escaping the geometry being analyzed.

Unfortunately, there are two serious limitations of the Monte Carlo method as applied to radiation shielding problems.

The first of these is a basic shortcoming of all Monte Carlo solutions: a precision (or error reduction) proportional to the square root of the number of histories run. So, for example, if an analyst wants the error reduced by a factor of 10, they must run 100 times as many histories as have previously been run. Since each successive digit of accuracy is so expensive, Monte Carlo radiation shielding calculations tend to run to a precision of only 1–5%.

This disadvantage is partially compensated by the fact that the statistical processing of the dose rates allows the method to tell the analyst the standard deviation of each of the requested outputs; in contrast to the “hidden” errors of the deterministic methods, the analyst knows the approximate uncertainty in the produced outputs.

The second of the shortcomings is that radiation shielding (in contrast to reactor or criticality safety analysis) tends to involve passage of the radiation through thick shielding, for which it is expected that only a small fraction of source particles will successfully penetrate. Since Monte Carlo solutions depend on simulations, this results in very low fraction of simulated particles contributing to dose rates at heavily shielding points in the geometry, with resulting large uncertainties in the results (or maybe no results at all).

Variance reduction techniques for stochastic calculations

The second disadvantage—the low efficiency of simulated particles penetrating shielding—can be mitigated through the use of techniques generally referred to as *variance reduction* techniques. These techniques can be divided into two rough categories: (1) techniques that increase efficiency by replacing the natural probability distributions with other distributions that encourage particles to contribute to tallies of interest to the analyst and (2) techniques that serve to better utilize computer time by amplifying the number of particles in “important” regions of the problem and culling the number of simulated particles in “unimportant” regions of the geometry. It is the analyst’s responsibility to specify the techniques for the computer code to use, making Monte Carlo radiation analysis a bit of an art form.

Variance reduction techniques assign each computational particle a numerical *weight*, which begins as 1.00 and represents the fraction of a physical particle that the computational particle represents.

The first category of variance reduction techniques emerge from the root of all Monte Carlo, the Law of Large Numbers, which relates the value of an integral (corresponding to a Monte Carlo tally) to the result of a stochastic process:

$$\bar{f} = \int \pi(x)f(x)dx = \lim_{N \rightarrow \infty} \frac{1}{N} \sum_{n=1}^N f(\hat{x}_n)$$

where: $\bar{f}$ is the weighted average of $f(x)$; $\pi(x)$ is the weight function used as the probability density function for choosing samples of x . It must obey:

$$\pi(x) \geq 0 \quad \int \pi(x)dx = 1$$

N is the number of samples; $\hat{x}_n$ is the n^{th} sample of x .

Comparing the integral to the stochastic equivalent, it can be seen that the two functions in the integral serve two different purposes in the stochastic process: (1) the distribution $\pi(x)$ is used to choose the sample values x used in the tally and (2) the function $f(x)$ is used in the tally itself.

The roles of these two pieces can be moved around a bit without changing the expected answer; in particular, if another probability distribution $\tilde{\pi}(x)$ is used instead of $\pi(x)$, the answer can remain unchanged if a corresponding change is made to the scoring function:

$$\bar{f} = \int \tilde{\pi}(x) \left(\frac{\pi(x)f(x)}{\tilde{\pi}(x)} \right) dx = \lim_{N \rightarrow \infty} \frac{1}{N} \sum_{n=1}^N \left[\frac{\pi(\hat{x}_n)}{\tilde{\pi}(\hat{x}_n)} \right] f(\hat{x}_n)$$

After such a modified distribution is used for a decision in the life of the particle, the particle weight is multiplied by the factor $\pi(\hat{x}_n)/\tilde{\pi}(\hat{x}_n)$.

The main variance reduction techniques in the category are:

1. Absorption weighting, which raises the probability of scattering to 100%, keeping particle histories from terminating by absorption.
2. Exponential transform, which increases the length of path segments in a desired direction (presumably toward a detector or desired dose point) and decreases the length of path segments away from the desired direction.
3. Source importance weighting, for which the user replaces the natural source spatial, energy, or direction distributions with preferential distributions (e.g., source positions closer to the detector, source initial directions toward the detector, source initial energies that prefer higher energies that are more penetrating, etc.).

The second category of variance reduction do not change any of the distributions that govern particle history, but have the user specify the relative “importance” of regions of the problem (with importance generally increasing toward desired dose points) and the computer code focusing its effort on the important regions.

Subsequently when a particle crosses between regions of differing importances, the weight might change in the following way:

- If the particle moves into a more important region, the computational particle is split into multiple computational particles in order to more thoroughly sample the region; the weight of the original particle is divided among the new particles. For example, if a particle crosses into a region with twice the importance of the region it is leaving, the computer code would divide its weight between two computational particles, each with half of the presplit particle weight.
- If the particle moves into a less important region, multiple particles are statistically combined into one by the computer code letting the crossing particle survive with probability p (commonly the ratio of the lesser to greater importance). If the particle survives, its weight is increased by a factor of $1/p$; otherwise the particle history is terminated.

These strategies work to better utilize computer resources without changing the expected result, resulting in lower uncertainties with the same calculational time. The bottom line is that a higher fraction of computational particles penetrate the shield than physical particles would, but the particles arrive at tally points with fractional weights; this greatly decreases the uncertainty of the calculated results.

The current state of the art of Monte Carlo analysis is to use approximate (and faster) deterministic methods to solve the adjoint form of the Boltzmann equation to determine the cell importances of the problem regions, and then to subsequently use these importances to increase the efficiency of the Monte Carlo simulations.

Advantages and disadvantages

The principal advantages of the stochastic Monte Carlo approach are:

- The capability to have near-exact geometry and complete energy-dependent cross sections for the materials in the problem. This largely eliminates errors introduced by geometric modeling approximations.
- The given estimate of the uncertainty of the reported values of interest to the analyst.

The disadvantages of the methods are:

- For almost all problems, the Monte Carlo method is much slower than lower-dimensional deterministic solutions; as a practical result, the analyst is usually limited to 1–5% accuracy in the dose rates beyond thick shielding.
- The errors in the tallies decreases as the inverse of the square of the number of histories run (and therefore as the inverse of the square of computer time required), which is very slow numerical convergence compared to deterministic methods.
- Without efficient variance reduction techniques applied, particle history simulations tend to be very inefficient in calculating doses beyond thick shielding.

Availability of computer codes

In the United States, the primary source of computer codes that have been developed at National laboratories (or submitted for free distribution by developers) is the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory in Oak Ridge, Tennessee.

In addition, other (usually proprietary) computer codes—some of which have been developed to interface with RSICC-distributed codes—are available for purchase or licensing from commercial entities.

A list of useful websites for finding and understanding these computer codes is presented below.

Research areas

At the current time the development of computer techniques and the application of those techniques to shielding problems of interest is an active area of research in nuclear physics and engineering. Current research papers and abstracts are available in the various publications of the American Nuclear Society (ANS) from their website at www.ans.org.

Sample Problem #1: 2 MeV photon source

To illustrate the use of both the hand calculation methods and a common computer analysis tool, a 2 MeV photon source sample problem, using the geometry shown in [Fig. 1](#), is considered. For this exercise, the problem was modeled in MCNP and the dose rates at the three indicated points (equally distant from the photon point source) are compared to the results of more approximate hand calculations.

A source strength of 1.345×10^{10} photons/sec was chosen so that the unshielded hand-calculated dose rate (using [Table 1](#) data) at each of the points would be 100 mrem/hr. For this problem (and “[Sample Problem #2](#)” section), we assume that the goal of the shielding is to reduce the dose rate at the entrance to the “maze” (near the lower right of [Fig. 1](#)) to below 1 mrem/hr, so the shielding needs to reduce the dose rates by about a factor of 100. This is, of course, about two tenth value thicknesses.

The principal problem parameters are:

- The walls are 10 cm thick lead walls (density 11.35 g/cm³)—corresponding to two tenth value thicknesses from [Table 1](#) data.
- Not shown in the figure are the floor and ceiling for the room, which have the same 10 cm thickness of lead; inner height is 300 cm and the point source is at the mid-elevation at the indicated (x,y) position.
- The MCNP calculation was run for 500 min (which resulted in a 1.7847×10^9 histories) to generate the dose rate map shown in [Fig. 2](#); the dose rates at the points P1, P2, and P3 were extracted from the dose map produced by MCNP.

Note that all three points that were selected have the same line-of-sight distance from the source (170 cm) and have the same direct-line shielding thickness (10 cm), so one would calculate the same dose rate with the hand calculation equation for each point. The problem was designed to show how the dose rate might be affected by three different scattering environments:

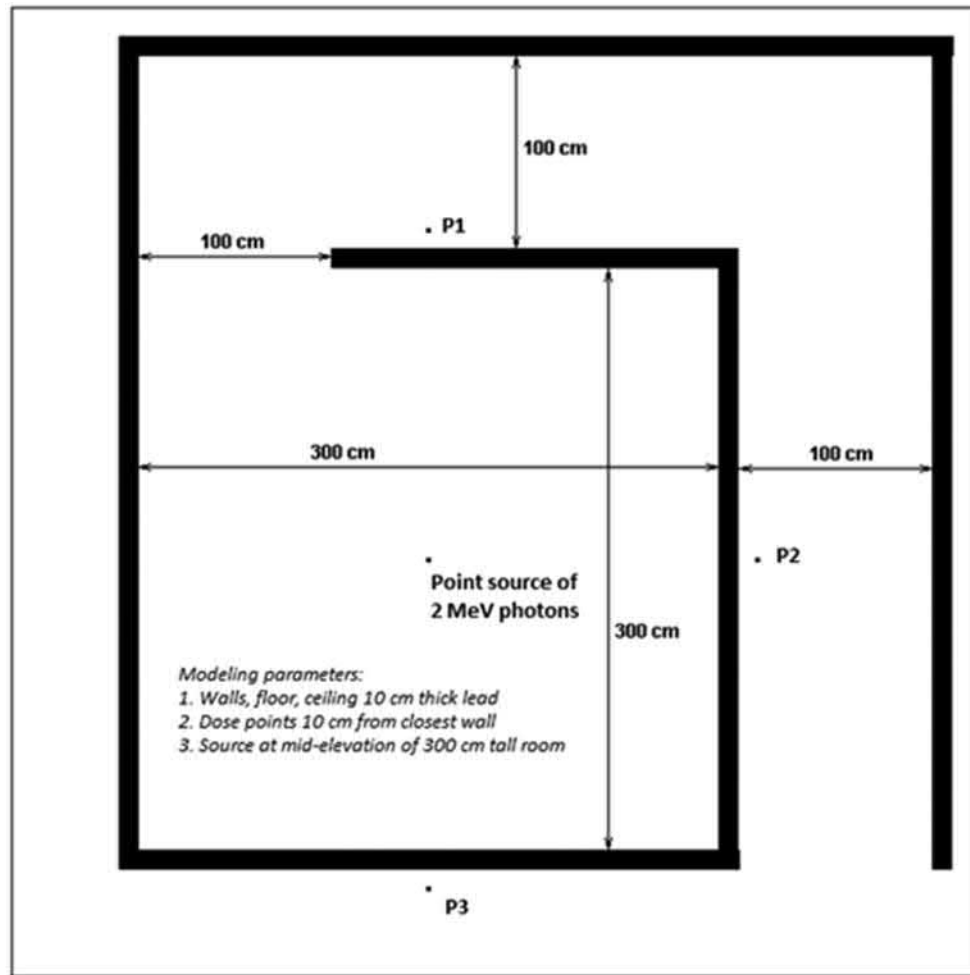


Fig. 1 Source, geometry, and materials of the sample problem.

- P1 at point (0,170,0)—with the source as the origin: There is a path by which the photons can reach the detector by bypassing the shield wall and there is a reflecting wall behind the point;
- P2 at point (170,0,0): There is no nearby bypass path, but there is a reflecting wall behind the point; and
- P3 at point (0, -170,0): There is neither a bypass path nor a reflecting wall behind the point.

The hand calculation parameters used (using data from [Tables 1 and 3](#)):

- $S_p = 3.7 \times 10^{10}$ photons/second
- $R(2 \text{ MeV}) = 7.45 \times 10^{-12} \text{ Sv-cm}^2$
- $r = 170 \text{ cm}$
- $t = 10 \text{ cm}$
- $\mu(2 \text{ MeV}) = 0.514 \text{ cm}^{-1}$
- $A_1(2 \text{ MeV}) = 11.66$
- $\alpha_1(2 \text{ MeV}) = -0.0206$
- $\alpha_2(2 \text{ MeV}) = 0.0149$

[Fig. 2](#) shows the resulting dose rate map; [Table 5](#) gives the results of both the hand calculation and the MCNP estimate of the dose at the three points, along with the percentage difference between the two methods. The hand calculation gives very good results for this problem.

Sample Problem #2: 14 MeV neutron source

The second sample problem is an analysis of a 14 MeV neutron source using the same basic “maze” geometry as in “[Sample Problem #1](#)” section. The source strength was again set to give us 100 mrem/hr unshielded dose rate at the three points; the

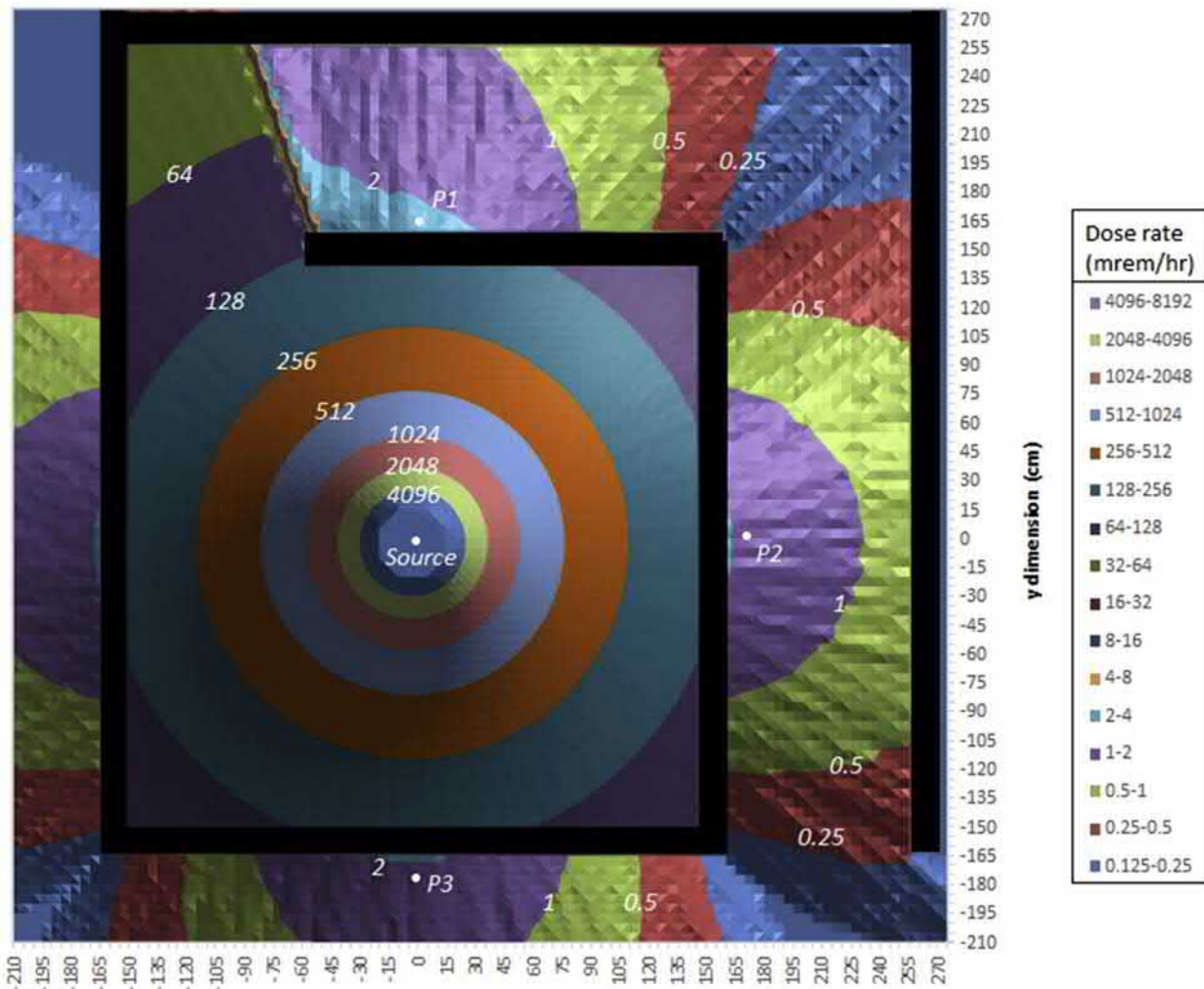


Fig. 2 Dose rate map from MCNP Calculation of the Sample Problem #1.

Table 5 Results of hand and MCNP calculations for Sample Problem #1.

Point	Hand calculation results			MCNP results	
	Uncollided dose rate (mrem/hr)	Buildup factor	Total dose rate (mrem/hr)	Total dose rate (mrem/hr)	MCNP/hand ratio
P1	0.581	3.089	1.795	2.162 (± 0.021)	1.205
P2				1.891 (± 0.019)	1.053
P3				1.895 (± 0.017)	1.056

resulting source strength is 1.739×10^8 neutron/second. The wall thickness was increased to 50 cm (chosen to be roughly two tenth value thicknesses of concrete for neutrons from Table 1); the same three dose points were used as in “Sample Problem #1” section.

Based on the more complicated considerations for neutron shielding discussed above, the shield material was designed considering a combination of regular concrete (contents shown in Table 6), iron, and boron carbide. A parametric study was performed to compare the expected shield dose reduction as a function of the volume fractions of each of these materials; a one-dimensional slab discrete ordinates calculation of 50 cm thickness with a 14 MeV perpendicular beam neutron source was used in this comparison, using the contents from (McConn et al., 2011) and the HILO2k shielding library from RSICC.

The results are shown in Fig. 3. From these results, it looks like a combination of iron and concrete is more effective than either of them alone; the results indicate that a dose reduction of about a factor of two can be obtained with iron contents properly chosen in the 40–70% range, which is (barely) credible as aggregate contents of a concrete mixture. Much less of a reduction is brought about by including the B_4C , but it might still be worth doing (depending on cost and the extra trouble that including it in the concrete “recipe” might cause for concrete cohesion and curing time).

Table 6 Isotopic contents of concrete mixture used (Material 89 from PNNL, 2011).

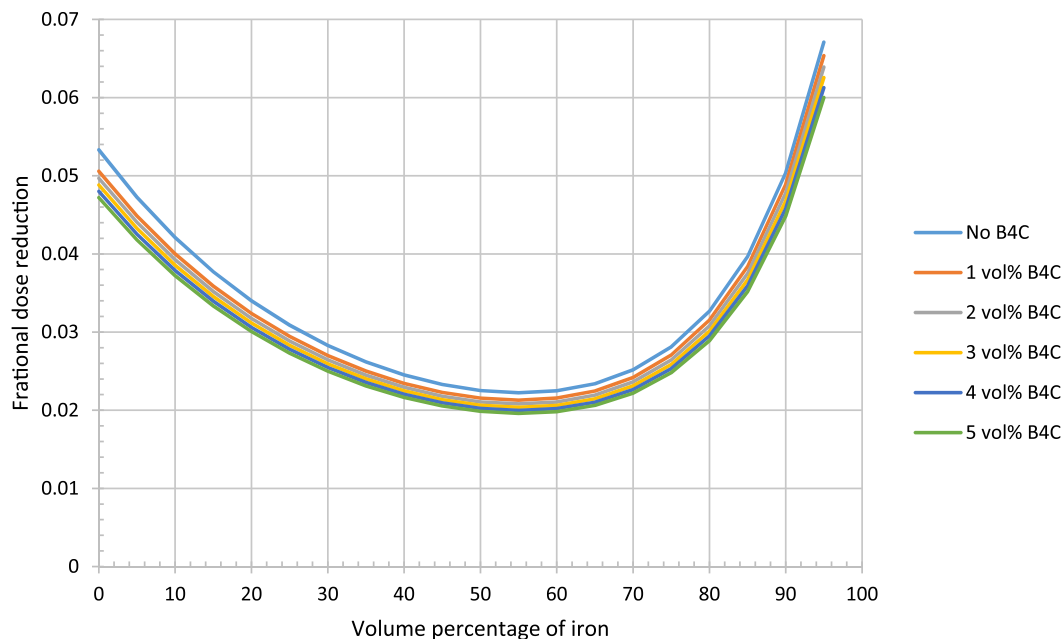
Isotope	Regular concrete (PNNL material #99) density = 2.35 g/cm ³		Concrete mixture (PNNL material #89) density = 4.50 g/cm ³	
	Mass fraction	Atom density (atoms/bn-cm)	Mass fraction	Atom density (atoms/bn-cm)
H	0.01	0.014039	0.008	0.021509
B	—	—	0.009	0.002256
O	0.532	0.047056	0.107	0.018124
Mg	0.029	0.001785	0.043	0.004794
Cl	0.034	0.001783	0.021	0.001605
Mn	0.337	0.016976	0.003	0.000148
Ca	0.044	0.001553	0.011	0.000744
Fe	0.014	0.000355	0.798	0.038724

For a simple comparison of a homogeneous shield versus a layered shield, a layered calculation was run on these same constituents: regular concrete, iron, and boron carbide. Since B₄C is a powder, two material layers were used: 20 cm of iron followed by 30 cm of 5 vol% borated concrete. (This preserves the total weight of the three material contents used in the “3 vol% B4C” curve at the 40 vol% iron point, which produced a dose fraction of 0.0225). The layered shield produces a dose fraction of 0.0167, a 33% dose reduction that might be worth the extra complexity of construction. We will assume, for this example, that the designers stay with the homogeneous shield design.

Rather than using the artificial mixture from this parametric study, a note of reality was added (to represent the real-life dialog that would have to happen between the shield designer and the suppliers of concrete) by restricting our choice of concrete mixture to those in [McConn et al., 2011](#) with a 50/50 or lower iron volume fraction in the mix. Looking at the available options from that document, the material 89 (“Concrete, M-1”) was chosen—as the one with volume fraction of steel closest to (without exceeding) 50%. The isotopic makeup of the material is given in [Table 6](#); since the atom densities of pure iron and boron in boron carbide are 0.084911 and 0.109858 atoms/barn-cm, respectively (Materials 158 and 40 from [McConn et al., 2011](#)), the iron volume fraction is ~45% and the boron carbide volume fraction is ~2%. (This is a very heavy concrete—almost twice the density of regular concrete.)

To gauge the expected effectiveness of this mixture, another one-dimensional discrete ordinate case was run for this material through a 50 cm slab, again with a 14 MeV perpendicular beam neutron source. The resulting fractional dose reduction as a function of depth in the shield is shown in [Fig. 4](#), along with an exponential least-squares fit (dotted line). Notice how thick the shield has to be before the asymptotic dose reduction starts to occur (due to scattered particles adding to the dose). Notice also the drop-off of the dose rate at the outer edge of the shield; this is because of the reduction in the number of backscattered particles. The estimated dose reduction factor through the shield is 0.0162, a bit better than the lowest point in [Fig. 3](#) (but not quite the factor of 100 we were looking for.)

Again, the MCNP calculation was run for 500 min like section “[Sample Problem #1](#)”, with about 8.5×10^7 histories, to generate the dose rate map shown in [Fig. 5](#). [Table 7](#) compares the results at the three points for both the MCNP calculation and the simplified

**Fig. 3** Results of 1D slab parametric study of concrete shielding contents.

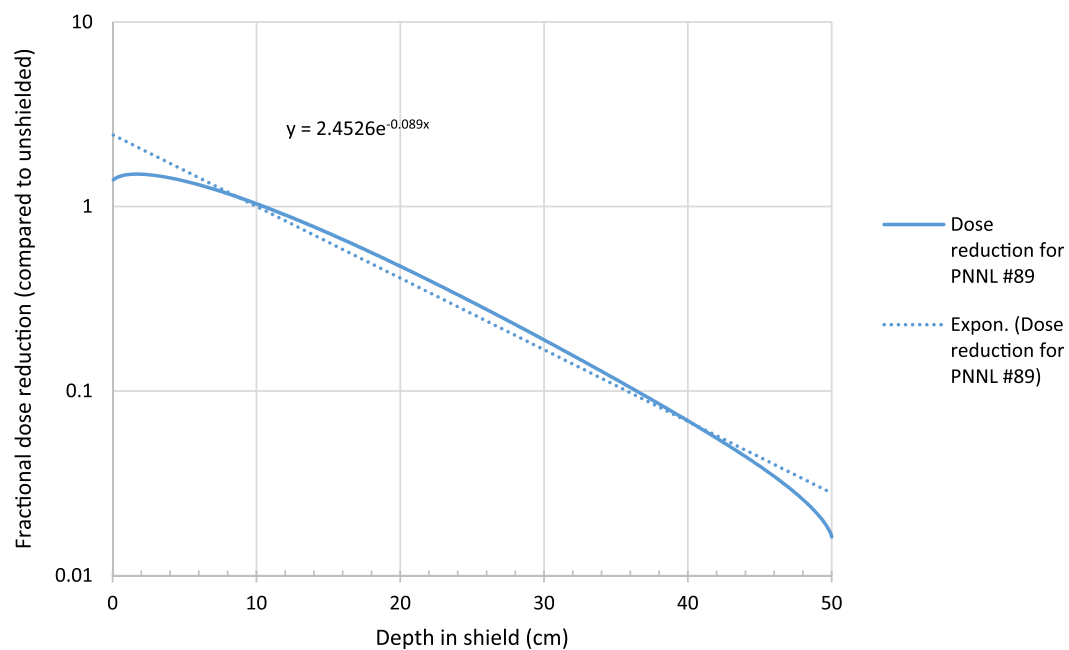


Fig. 4 Dose reduction factor for the concrete mixture as a function of depth in the 50 cm shield.

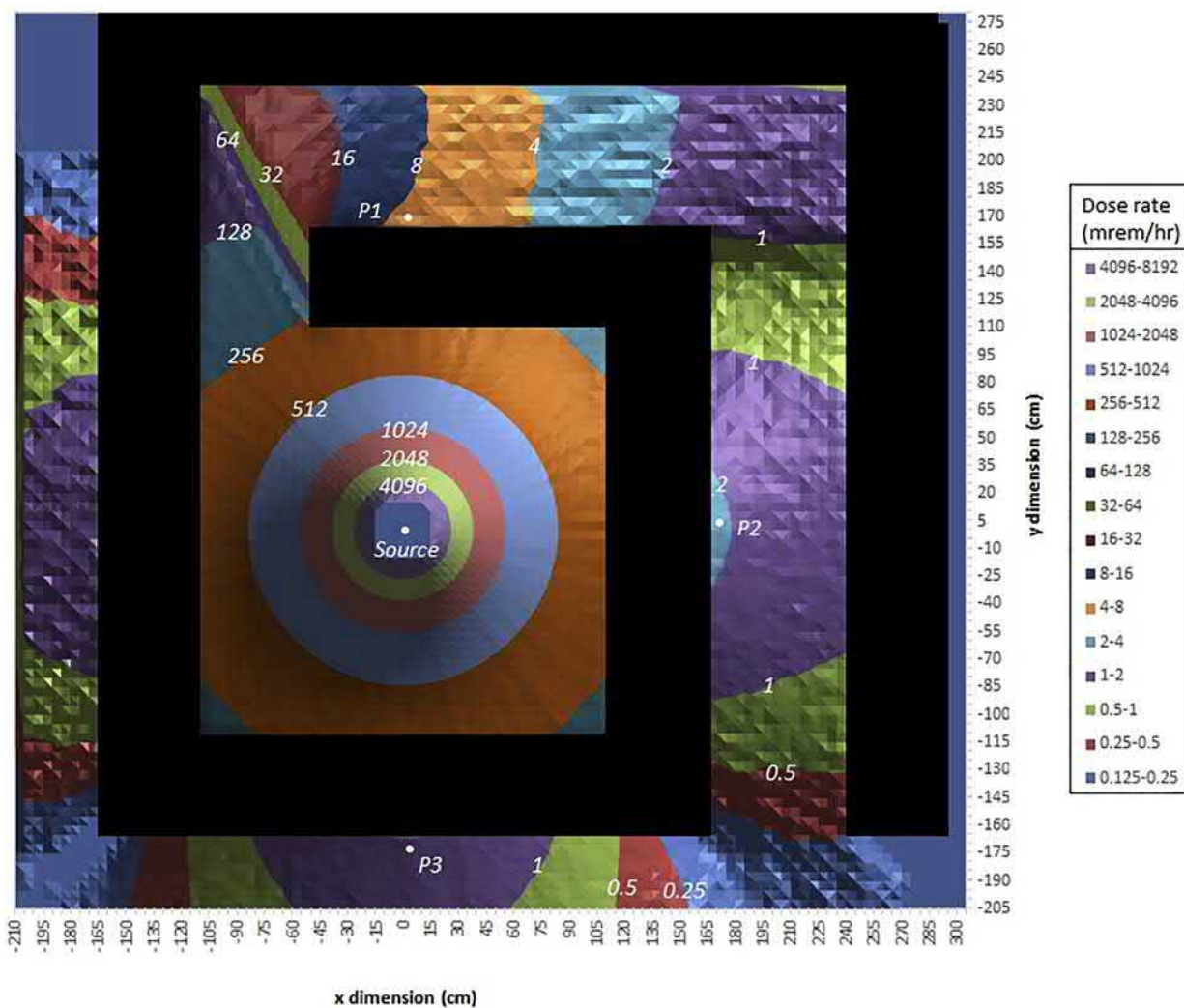


Fig. 5 Dose rate map from MCNP Calculation of the Sample Problem #2.

Table 7 Results of 1D Slab and MCNP calculations for Sample Problem #2.

Point	1D Slab calculation results			MCNP results	
	Unshielded dose rate (mrem/hr)	Expected dose reduction from concrete mixture	Total dose rate (mrem/hr)	Total dose rate (mrem/hr)	MCNP/1D slab ratio
P1	100	0.0162	1.62	7.193 (± 0.097)	4.44
P2				2.113 (± 0.013)	1.30
P3				1.898 (± 0.012)	1.17

estimate of the dose from combining the uncollided dose and the 1D estimate of the shielding reduction. The simplified calculation gave only moderately high (but still useful) estimates of the dose rates at points P1 and P2, although it severely underestimates the dose rate at point P1.

Conclusion

The ability to analyze dose effects of radiation sources has been well developed over the 125 years since the discovery of radioactivity. A range of analysis methods is available to today's analyst that allow for accurate assessment of radiation transport applications.

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- <https://rsicc.ornl.gov/>—Radiation Safety Information Computational Center.
- <https://mcnp.lanl.gov/>—A General Monte Carlo N-Particle (MCNP) Transport Code.
- <https://www.ornl.gov/scale>—SCALE.
- https://mcnp.lanl.gov/pdf_files/la-ur-03-1987.pdf—MCNP User Manual, Version 5.
- <https://www.radiationsoftware.com/microshield>—MicroShield—Radiation Software | Grove Software.
- https://www.pnnl.gov/main/publications/external/technical_reports/PNNL-15870rev1.pdf—Compendium of Material Composition Data for Radiation Transport Modeling.
- <http://www.icrp.org/>—International Commission on Radiological Protection.
- <https://ncrponline.org/>—National Council on Radiation Protection & Measurements.

Shielding Against Galactic and Solar Radiation in Space

Lawrence Heilbronn, Nuclear Engineering Department, University of Tennessee, Knoxville, TN, United States

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Glossary

ALARA As Low As Reasonably Achievable, a guiding principle of radiation protection.

A.U. Astronomical Unit. The average distance between the centers of the sun and Earth, equal to 1.496×10^{11} m.

Dose Energy absorbed per unit mass.

Equivalent dose Dose multiplied by the LET of the radiation delivering the dose.

eV Electron volt, a unit of energy equal to the energy of an electron after passing through a 1 volt potential difference.

"keV" = 10^3 eV, "MeV" = 10^6 eV, "GeV" = 10^9 eV, "TeV" = 10^{12} eV.

GCR Galactic cosmic rays. The space radiation environment originating outside the solar system.

Gy Gray, a unit of dose = 1 J/kg.

HZE High energy ($E > 10$ MeV/nucleon) heavy ion ($Z > 2$) nuclei.

HZETRN High Z and Energy TRaNsport code.

LEO Low Earth orbit. Orbits generally within the innermost radiation belt and below.

LET Linear energy transfer, the amount of energy lost by an ion per unit length.

OLTARIS On-Line Tool for the Assessment of Radiation In Space.

SEP Solar energetic particles. Energetic protons, electrons and heavy ions emitted from the sun.

SAA South Atlantic Anomaly, the region of the inner trapped belt that dips to lower altitudes.

Sv Sievert, and unit of equivalent dose = 1 J/kg.

Introduction

On August 7, 1912, Victor Hess flew an electroscope on a balloon flight up to 5300 m and found the ionization rate at that elevation was three times that of the rate on the ground. Based on that and data from earlier flights, he concluded that the source of radiation was not from the sun, but of cosmic origin, making his measurement the first to detect and identify galactic cosmic radiation. One hundred years later to the day on August 7, 2012, the Radiation Assessment Detector aboard the Mars Science Laboratory Curiosity Rover made the first measurements of cosmic rays on the surface of another planet. Over that span of 100 years, detector technology advanced to develop instruments that simultaneously detect and identify protons, heavy ions, electrons, gamma rays and neutrons, yielding detailed information on the primary radiation environment in space.

Measurements at high altitude and in space have shown that the space radiation environment is much different than the natural and man-made radiation environment encountered on Earth. Particle energies can exceed tens of TeV per nucleon, and ion species span every natural isotope as well as electrons, positrons, and gamma rays. As such, different shielding strategies than the ones typically used on Earth are required to minimize the risk to astronauts and electronics from exposure. The health effects from exposure to radiation and the current National Aeronautics and Space Administration (NASA) standards for space activities are discussed in other chapters, and as such those topics won't be discussed here other than to note that radiobiology research is continuing in order to better understand the damage and effects from the high-energy components of galactic cosmic rays.

Mitigating the risk from exposure to the radiation environment for facilities in space requires an understanding of the particle species and energies from all sources, and then using that information to develop appropriate shielding strategies. This article first

discusses the components of the primary radiation field in the solar system and then outlines the shielding methodologies employed to mitigate the effects from that radiation. The article is divided into the following sections:

- Galactic cosmic rays (GCR)
- Solar energetic particles (SEP)
- Trapped belt radiation
- Shielding methodologies and materials
- Integrated shielding concepts

Galactic cosmic rays (GCR)

The ionizing particles that make up Galactic cosmic rays (GCR) present a risk to crews and electronics, especially for long-term missions. GCR originate outside of solar system and are made up of fully stripped atomic nuclei (98%) and electrons (2%), with the nucleonic component consisting of nuclei ranging from protons through uranium. The majority of the nucleonic component is protons (~87%), with helium nuclei the next most abundant (~12%) (NCRP, 2000). The rest of the remaining 1%–2% nucleonic flux is composed of heavier nuclei. In the solar system, the energies of GCR range from tens of MeV per nucleon up to 10^{14} MeV/nucleon, with the majority of the flux between 100 MeV per nucleon and 10 GeV/nucleon. The differential flux (expressed in units of number of ions per steradian per square meter per MeV per nucleon per second) of each GCR ion species generally peaks between 300 and 600 MeV/nucleon.

Although the fluence (time and energy-integrated differential flux) of high-energy heavy ions (HZE) with $Z > 2$ is only 1%–2% of the total nucleonic fluence of GCR, their contribution to free space (unshielded) dose is significant. Dose from charged particles is a function of mass stopping power of the ion, given by:

$$D = \left(\frac{1}{\rho} \frac{dE}{dx} \right) \phi$$

where D = dose, ϕ = fluence in units of cm^{-2} , and the term in the parentheses is the mass stopping power. Stopping power is proportional to Z^2 of the ion, and as such the contribution to the free space dose from each GCR nuclei is also proportional to Z^2 of the ion. **Table 1** shows the fraction each GCR ion contributes to the overall nucleonic fluence and dose in free space. Although HZE make up a small fraction of the total fluence, they make up almost 40% of the free space dose.

Upon entering the heliosphere at a distance of around 100 Astronomical Units (A.U.), GCR particles are affected by the sun's variable magnetic field and the resulting changing conditions of the solar wind. As the GCR particles propagate through the solar wind medium, three mechanisms are responsible for modulation of the GCR spectrum (NCRP, 2000). The solar wind (primarily low energy protons with a velocity around 400 km/s) acts as scattering centers, leading to diffusion (scattering). A second mechanism is convection due to the outward motion of the solar wind, and the third mechanism is adiabatic cooling of GCR particles. Thus, depending on the magnetic field strength, associated solar wind strength, and the particle's energy, the shape and magnitude of the primordial GCR spectrum is altered inside the heliosphere, with the effect greatest for lower energy particles. Because the strength of the sun's magnetic field and solar wind depends on the 11-year solar cycle, the GCR spectrum inside the heliosphere is also a function of the solar cycle. During solar minimum (minimum solar activity and magnetic field strength), the GCR spectrum is at a maximum magnitude. At solar maximum, the GCR spectrum is at a minimum. The GCR spectrum also shows a clear dependence on the 22-year Hale solar cycle of alternating polarity of the magnetic field (Ross and Chaplin, 2019).

Due to the resulting cyclical time dependence of the local GCR spectrum, GCR shielding must take account of the modulation of the spectrum during the period of time in the solar cycle under consideration. Models have been developed that generate time-dependent GCR spectra which can then be used in transport model calculations. These models take the time-dependence into account using any of several indicators related to solar activity, including sun spot number, neutron monitor rates, X-ray flux measurements, and GCR flux measurements in spacecraft. Two of the more commonly used GCR models are the Badhwar-O'Neill 2014 model (BON14) (O'Neill et al., 2015) and the Matthiä model (Matthiä, 2013). Both models generally agree within 10% of each other and have been extensively validated against past GCR measurements. **Fig. 1** below shows GCR spectra for protons, He, C, and Fe ions as calculated by the BON14 model under solar minimum conditions.

Table 1 The fractional contributions to the total unshielded fluence and dose from selected nuclei of GCR.

GCR ion	Fraction of fluence	Fraction of dose
Proton	0.897	0.432
He	0.094	0.180
C	0.00256	0.0443
O	0.00242	0.0746
Si	0.000366	0.0345
Fe	0.000252	0.0821
All HZE	0.009	0.388

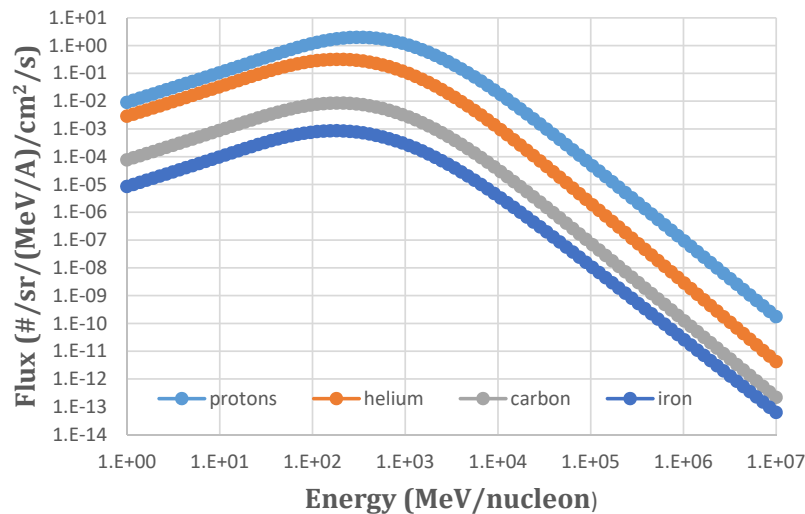


Fig. 1 Calculated GCR spectra for protons (light blue), He (orange), C (gray), and Fe (dark blue) ions. The calculations are from the Badhwar-O'Neill 2014 model, run under solar minimum conditions.

Solar energetic particles (SEP)

There is a continuous flux of protons and other ions emitted from the sun that make up the solar wind. However, the energies of these ions are very low, generally below 1 keV, and pose little to no risk from radiation effects. Much more damaging are the energetic ions emitted during solar events such as coronal mass ejections and solar flares.

The frequency of energetic solar particle events is correlated with the solar cycle, with the probability of an event the greatest around solar maximum. Particles emitted from the sun follow the solar magnetic field lines, and as such the intensity of the particle flux depends on one's location in space and time. Electrons, protons, and heavier ions are ejected during these events. Among the ejected nucleons, protons make up the vast majority of the flux, with energies up to 1 GeV and above. Once an event occurs on the sun, the particle flux increases over several hours or days to a maximum intensity, then experiences a decrease that can last several days. Events can have more than one burst, as well. Fig. 2 shows the proton flux at 1 A.U. as a function of time during the Halloween 2003 event. Three primary ejections produced elevated levels of proton flux lasting 2 weeks, with a maximum proton flux four orders of magnitude above levels during no solar activity.

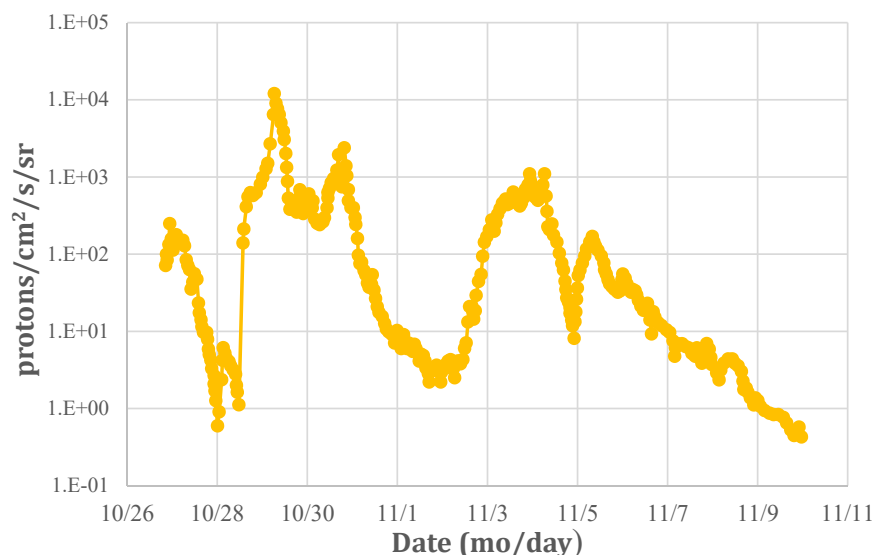


Fig. 2 The proton flux at 1 A.U. as a function of time during the October 2003 event. Data from Jiggins P, Chavy-Macdonald M-A, Santin G, Menicucci A, Evans H, and Hilgers A (2014) The magnitude and effects of extreme solar particle events. *Journal of Space Weather and Space Climate* 4. <https://doi.org/10.1051/swsc/2014017>.

Each SEP event produces a unique energy spectrum and intensity. The energy spectrum generally follows either a power law or exponential distribution that drops off with increasing energy. **Fig. 3** shows proton energy spectra at 1 A.U. integrated over the entire event from four historically significant events: February 1956, November 1960, August 1972, and October 1989 ([Mewaldt, 2006](#)).

In most cases, the majority of the proton SEP flux lies below 50–100 MeV, requiring modest amounts of shielding to stop such energies. For example, a 100-MeV proton stops in 3.7 cm of aluminum. However, if caught in an intense SEP event without shielding, such as excursions outside a spacecraft or habitat, severe biological consequences, including death, can occur. In addition, SEP can cause significant electronic effects in thinly shielded satellites, probes, and spacecraft. As such, the development of space weather prediction and early warning systems to provide opportunity to seek shelter or place electronics into safe mode are important components of protection from SEP.

A repository of measured particle spectra from SEP events since the late 1990s is kept at the California Institute of Technology through a website (details in the Relevant Websites section) and can be used to select particular SEP spectra for source terms in radiation transport codes. The OLTARIS code maintained by the NASA Langley Research Center ([Singleterry et al., 2010](#)) allows the user to select from a list of historical SEP events when calculating particle fluences and doses behind shielding in space.

Trapped belt radiation

Any planetary body with a strong enough magnetic field will capture lower energy GCR, solar wind and SEP ions, creating a flux of energetic ions located in specific locations along the geomagnetic field lines, called trapped belt radiation. Once trapped by the magnetic field, the particles will traverse back and forth between the poles along the magnetic field lines. At the poles, particles can enter and interact in a planet's atmosphere, which on Earth creates the Aurora Borealis/Australis. Compared with GCR, the flux of particles within these belts is much higher, but with lower average energies.

Trapped belt radiation—Earth

Earth has two primary radiation belts, commonly referred to as the Van Allen belts, and has a transient third outer belt that appears to be produced by coronal mass ejections and can exist up to several weeks. The inner two belts are permanent, although Earth's geomagnetic field lines are moved and distorted by the sun's magnetic activity, leading to temporal variations in the trapped radiation environment. The intensity of particle flux within the belts is a function of the latitude and radial distance to the field line. Computational tools have been developed that will calculate the energy-dependent proton and electron flux values for Earth's radiation belts. The Air Force Research Laboratory maintains one of the most commonly used codes by mission planners, the AE9/AP9/SPM model (see Relevant Website section for details and download) ([Ginet et al., 2013](#)).

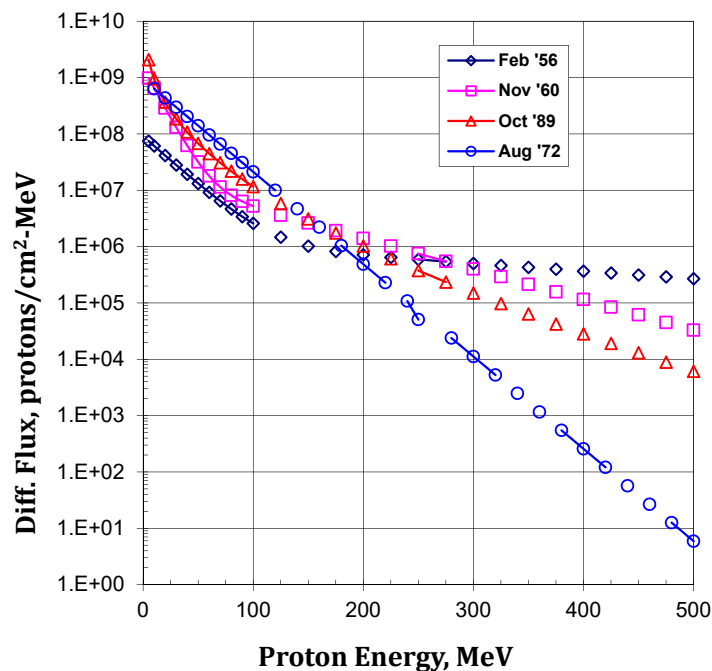


Fig. 3 Proton energy spectra in units of number per cm^2 per MeV as a function of proton energy in MeV. The February 1956, November 1960, August 1972, and October 1989 events are shown. Data generated from OLTARIS.

Earth's inner belt has an altitude that ranges between 600 and 8000 miles (950–12,800 km). It is primarily made up of protons with energies up to 400–500 MeV and electrons with energies primarily in the 100s of keV. There are electron energies that go up to several MeV, but at a greatly reduced flux. Most manned low Earth orbit (LEO) missions are below the inner belt. However, due to an offset of the Earth's magnetic dipole center and rotational center, there is a dip in the inner belt that extends down to an altitude of 120 miles (190 km) over the Atlantic coast of South America called the South Atlantic Anomaly (SAA). LEO missions with moderate orbital inclination will pass through the SAA several times per day and will encounter elevated levels of radiation during those passes, leading to an increase of a factor of 100 or greater relative to the dose rate from GCR in LEO, depending on the altitude during the time of the crossing. For reference, at an altitude of 400 km, the contributions to the dose rate from GCR and trapped belt radiation are about equal to each other, with a total dose rate of about 140 $\mu\text{Gy/d}$ behind shielding comparable to the amount found in the Russian space station MIR (between 20 and 30 g/cm^2 , on average) (Badhwar, 1997).

Fig. 4 shows an AP9 calculation of the proton flux experienced by the International Space Station (orbital inclination of 51.6 degrees, altitude of 470 km). There is a total flux of 2.9×10^6 protons per cm^2 per day, with approximately 75% of the flux below 100 MeV, and 93% of the flux below 200 MeV.

The primary concern for radiation protection in the inner belt is the protons because of their high energies. 100 MeV protons will penetrate 10 g/cm^2 (3.7 cm) of aluminum, for example. Inner belt electron energies are low enough that 1 mm of aluminum will stop over 95% of the flux (flux with energies 500 keV and below). In addition to radiation effects to humans and electronics, the charging of spacecraft and components from the capture of trapped belt radiation can also be a concern for missions that spend an appreciable amount of time in the inner belt.

Earth's outer Van Allen belt approximately extends from 19,000 to 40,000 km and consists mostly of electrons with energies between 10s of keV to 10 MeV. Protons are also present in the outer belt, with energies between 0.1 and 5 MeV. The radiation field within the outer Van Allen belt is of concern for many of the satellites orbiting the planet. GPS satellites orbit the Earth at an altitude around 20,000 km, and satellites in a geosynchronous orbit are at an altitude of approximately 36,000 km, placing both of these orbits within the outer belt. Many satellites are placed in eccentric orbits that carry them through the outer belt for periods of time.

Trapped belt radiation—Outer planets

All four gas giants in the solar system—Jupiter, Saturn, Uranus and Neptune—have strong magnetic fields and corresponding intense trapped belt radiation environments. Jupiter's magnetic dipole moment is 20,000 times greater than Earth's, creating trapped belts that extend out to 2,100,000 to 3,500,000 km from the center of the planet (Bolton et al., 2004). In addition to the capture of SEP, GCR, and solar wind, eruptions from the moon Io are also a source of trapped ions in the belts. Jupiter's moons are within the belts, and as such they sweep through the belts, removing some of the trapped radiation. There are three magnetosphere regions around the planet. The inner magnetosphere (inside of 350,000 km) is the source of the intense, high-energy electron, proton and heavier-ion trapped radiation around the planet. Measurements made by the Pioneer and Galileo probes indicated high-energy (> 5 MeV) electron fluxes of 10^6 to 10^7 electrons/ cm^2/s , and high-energy (> 80 MeV) proton fluxes up to values greater than 10^5 protons/ cm^2/s (Bolton et al., 2004). Pioneer 10 received an external dose of at least 5000 Gy from electrons and 10,000 Gy from protons. Inside the 3-mm of aluminum shielding, Pioneer 10 received at least 4500 Gy (Miller et al., 1976).

Based on existing measurements and physical modeling of the Jovian magnetosphere, two codes are primarily used to predict the radiation environment around Jupiter. The European Space Agency maintains the Space ENVironment Information System "SPENVIS" (spenvis.oma.be) tool that can be used to calculate the radiation environment anywhere in the solar system. The Jet Propulsion Laboratory maintains the Galileo Interim Radiation Electron model "GIRE", which will calculate electron and proton flux values (Garrett et al., 2012).

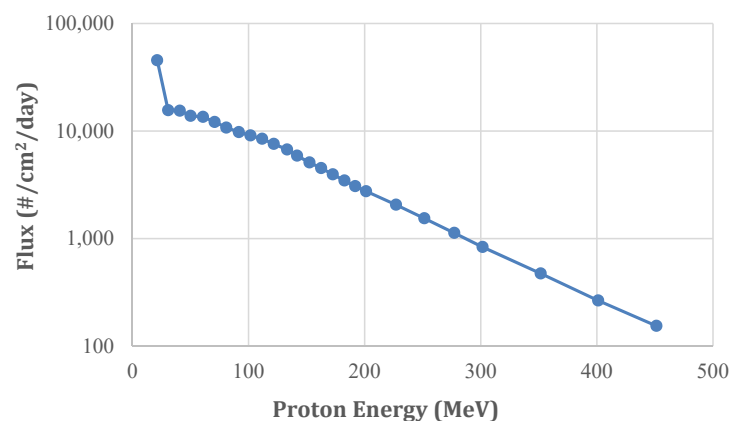


Fig. 4 Trapped belt proton flux at a 51.6-degree inclination, 470 km altitude, during solar maximum. Data generated from AP9 calculations.

Shielding methodologies and materials

As described in the previous sections, the space radiation environment is complex, consisting of particles over a wide range of energies. Adding to its complexity, the environment fluctuates based on its time during the solar cycle and the occasional appearances of solar particle events. In free space, the radiation environment consists of GCR and SEP. Near a planet with a significant magnetosphere, trapped belt radiation must be considered along with GCR and SEP. **Table 2** shows the primary particle species, energies, and current models used to compute the particle flux spectra from those three sources.

No facility on Earth is able to produce the entire space radiation environment, and testing of shielding materials in space is expensive and only relevant to the particular location and time of the test. As a result, transport model calculations are the primary tool used for the design of shielding used for a particular mission scenario. Both deterministic and stochastic (Monte Carlo) codes are used for space radiation transport, each with their own strengths. Deterministic codes are faster and are the engineering tool used to narrow the scope of shielding options. Monte Carlo codes are considered to be more accurate, but require more time to execute than deterministic codes. As such, Monte Carlo codes are usually used in the final design process rather than for trade studies.

The deterministic code HZETRN (Wilson et al., 1991) is the code used by NASA Langley Research Center for shielding design studies. The shielding design computational tool OLTARIS (Singletary et al., 2010) incorporates HZETRN and is widely used within the community. A number of Monte Carlo codes, such as Geant4 (Agostinelli et al., 2003), PHITS (Sato et al., 2018), MCNP6.2 (Goorley et al., 2012), FLUKA (Battistoni et al., 2015), and MARS (Mokhov et al., 2004), have developed physics models that handle the transport of the wide range of particles and energies that make up the space radiation environment. As particles transport through materials, they can slow down and stop in the material, slow down and exit the material with less energy, or breakup and create secondary radiation through nuclear and electromagnetic interactions. Both deterministic and stochastic codes incorporate stopping power formalisms for charged particles and nuclear and electromagnetic interactions for all particles.

Relevant measurements of radiation transport are needed for comparison with transport code predictions. Accelerator facilities are able to produce beams of relevant ion species and energies that can test the shielding properties of materials with respect to particular parts of the space radiation environment, and many measurements of dose, dose equivalent, and particle flux behind shielded spacecraft and probes have been made in space, providing a data base of experimental measurements that can be used to guide shielding design in space. In particular, these data are used for comparisons with radiation transport model calculations of those experiments in order to validate the codes and determine the uncertainties associated with their predictions of transport of the entire space radiation environment.

When comparing the shielding effectiveness of various materials against each other, they are compared at the same values of areal density, where areal density is equal to the thickness of the material times its density. One reason for this is the following: a material like carbon (graphite) has a wide range of densities. A 1-cm thick sample of graphite with a density of 2 g/cm^3 will have a different amount of dose reduction versus a 1-cm thick sample of graphite with a density of 1.5 g/cm^3 , even though they are the same material. If the thickness of the 2 g/cm^3 is reduced to 0.75 cm, such that both graphite samples have an areal density of 1.5 g/cm^2 , then the incoming radiation passes by the same number of carbon atoms and will experience the same amount of dose reduction. Another reason for comparing materials at the same values of areal density is that different shielding materials with the same areal densities have the same mass, a critical parameter in mission planning.

Transport code predictions and measurements have confirmed that shielding materials which contain hydrogen provide better results in terms of dose reduction due to two factors. One factor is that hydrogen provides the best stopping power per unit mass because of its charge to mass ratio of 1, whereas other elements have charge to mass ratios at or near 0.5. The other factor is that the hydrogen nucleus presents the smallest target for incoming ions, reducing the amount of breakup of heavy ions into more penetrating, smaller secondary fragments when compared to interactions with other target nuclei. **Table 3** shows experimental data on the percent reduction in dose from 1 GeV/nucleon Fe particles passing through elemental and composite targets (Zeitlin et al., 2006). The reduction in dose is normalized by the areal density of the targets used in the experiments (percent dose reduction per g/cm^2). Materials that contain hydrogen provide the largest dose reduction. Although aluminum is commonly used for spacecraft and satellites, it does not provide the dose reduction for GCR-like Fe ions that the composite materials provide. Results from these experiments and accompanying transport model calculations have led to polyethylene being chosen as the standard GCR shielding material against which all other materials are compared.

Using the Linear Energy Transfer (LET) values for all primary and secondary particles that penetrate the shield, a quality factor Q is applied to the dose from each ion to determine the dose equivalent behind shielding. Calculations of dose equivalent behind several thicknesses of aluminum and polyethylene shielding in a GCR environment were performed using several transport codes (Slaba et al., 2017). The calculations simulated an enclosed environment in space, where primary and secondary particles that

Table 2 Overview of source terms in the space radiation environment relevant to radiation effects to humans and electronics.

Source	Particle ion species	Energies	Source codes, catalogs
GCR	e^- , e^+ , p, He, $Z > 2$	up to 10s TeV/nucleon	BON14, Matthiä
SEP	e^- , protons, He	Up to 1 GeV/nucleon	See Relevant Websites section for a location of a catalog of events
Trapped belt	e^- , e^+ , p, He, $Z > 2$	Up to 1 GeV	AE9/AP9/SPM, SPENVIS, GIRE

Table 3 Percent dose reduction in 1 GeV/nucleon Fe ion passing through elemental and composite targets, normalized to unit areal density.

Shielding material	Percent dose reduction (per g/cm ²)
Carbon	3.86
Aluminum	2.34
Copper	1.36
Lead	0.40
Polyethylene	5.07
Lucite	5.04
Boron epoxy	4.0
LiF	3.8
Marsbar	3.1

Data taken from Zeitlin C, Guetersloh SB, and Heilbronn LH (2006) Measurements of materials shielding properties with 1GeV/nuc ⁵⁶Fe. *Nuclear Instruments and Methods in Physics Research Section B* 252: 308–318.

penetrate a shielding wall can then go through the enclosed space and interact in the back wall. In those calculations, the dose equivalent from all primary and secondary ions, including secondary neutrons, was determined for shielding up to 100 g/cm² thick. Fig. 5 (Fig. 13 in Ref. (Slaba et al., 2017)) shows that for aluminum shielding thickness beyond 20–30 g/cm², the dose equivalent actually increases with increasing aluminum thickness. For polyethylene shielding beyond 20–30 g/cm², there is only a slight decrease in dose equivalent.

Although the dose equivalent initially drops quickly as shielding is added, the calculations show an optimal shielding thickness of around 20–30 g/cm². In the case of aluminum, there is a detriment to going beyond that thickness. For polyethylene, there is little incentive to produce a slight decrease in dose equivalent given the cost to the mission to add more shielding mass. The reason for the increase for aluminum and the flattening of the curve for polyethylene is the buildup of light ions in the enclosure produced by particles that penetrate the front wall and interact in the back wall of the enclosed space. Going far beyond 100 g/cm², the dose equivalent curve will drop to lower values for any shielding material, but the cost to the mission becomes prohibitive.

Another feature of the results shown in Fig. 5 is that there is a minimum dose equivalent rate that can't be improved with additional shielding, which in turn has implications for either the duration of a manned mission or the determination of exposure limits for that mission. On a 3-year mission to Mars in an aluminum spacecraft and shelter, the total dose equivalent received by an astronaut during the mission is predicted to be between 1.5 and 2.1 Sv, using the calculated rates in Fig. 5. NASA's guidance on radiation exposure in space, found in NASA-STD-3001 (link to publication may be found in the Relevant Websites section), states "planned career exposure to ionizing radiation shall not exceed 3% Risk of Exposure-Induced Death (REID) for cancer mortality at a 95% confidence level to limit the cumulative effective dose (in units of Sievert) received by an astronaut throughout his or her

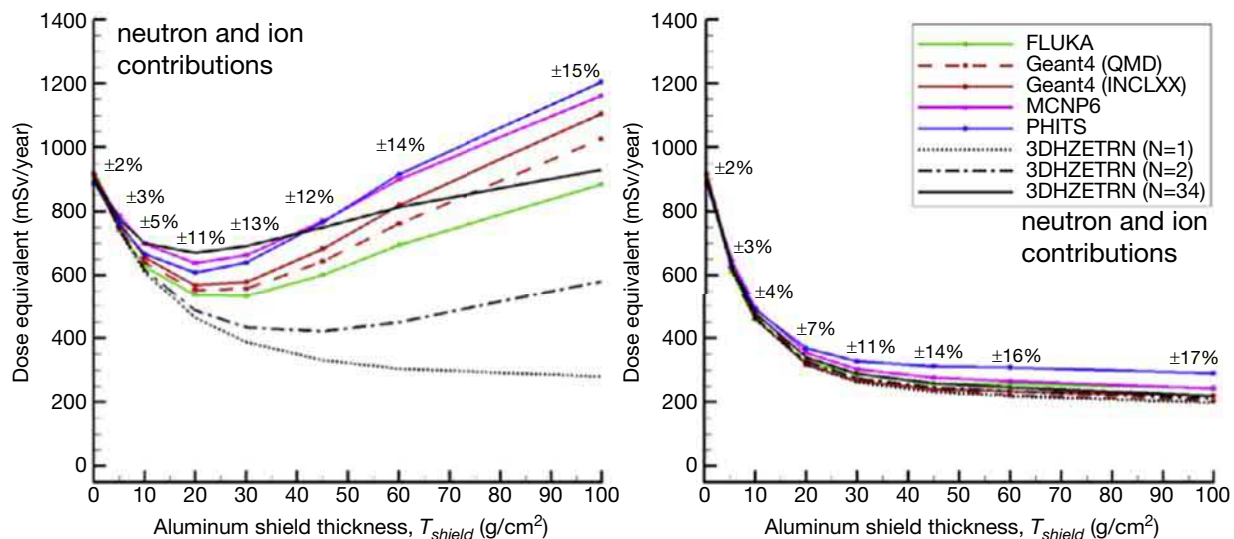


Fig. 5 Calculated dose equivalents behind various thicknesses of aluminum (left plot) and polyethylene (right plot) in a GCR environment, using several Monte Carlo transport codes and three different versions of the deterministic transport code HZETR. Figure courtesy Slaba TC, Bahadori AA, Reddell BD, Singleterry RC, Cloudsley MS, and Blattnig SR (2017) Optimal shielding thickness for galactic cosmic ray environments. *Life Sciences and Space Research* 12: 1–15, Fig. 13, page 9.

Table 4 Ranges of protons in aluminum and polyethylene.

Proton energy (MeV)	Aluminum $\rho = 2.7 \text{ g/cm}^3$	Polyethylene $\rho = 0.92 \text{ g/cm}^3$
10	0.171 g/cm ² (0.063 cm)	0.113 g/cm ² (0.123 cm)
20	0.575 g/cm ² (0.213 cm)	0.395 g/cm ² (0.429 cm)
30	1.180 g/cm ² (0.437 cm)	0.824 g/cm ² (0.896 cm)
50	2.928 g/cm ² (1.08 cm)	2.081 g/cm ² (2.26 cm)
100	10.01 g/cm ² (3.71 cm)	7.242 g/cm ² (7.87 cm)
200	33.31 g/cm ² (12.3 cm)	24.43 g/cm ² (26.6 cm)
500	148.9 g/cm ² (55.1 cm)	110.5 g/cm ² (120 cm)

career". The REID for an individual depends on sex and age, but it should be noted that current risk models show an exposure above 1.5 Sv exceeds the 3% risk of REID for most adults. Ultimately, exposure limits for a mission will be set by the guidance outlined in section 4.2.10.4 of NASA-STD-3001, which states "Exploration Class Mission radiation exposure limits shall be defined by NASA based on NASA-requested recommendations from the National Academy of Sciences, the Institute of Medicine, and the National Council on Radiation Protection (NCRP)".

As shielding thickness increases, the differences between the models' predictions becomes more apparent. The percentages next to each data point in Fig. 5 are calculated by taking half the difference between the extremes and then dividing by the average of the extremes, excluding the $N = 1$ and $N = 2$ versions of HZETRN.

For SEP shielding, most of the proton flux lies below 50–100 MeV, requiring a modest amount of shielding. For shielding of protons in the inner Van Allen belt, proton energies are also primarily below 100 MeV. Table 4 shows the amount of aluminum and polyethylene shielding required to stop protons of energies present in SEP and trapped inner belt spectra.

Although particle flux in a SEP event can be large enough to produce serious acute radiation effects, a thicker-shielded area within a spacecraft or habitat will provide adequate protection while waiting for the flux to decrease. During the initial buildup of particle flux in a SEP event, real-time monitors can alert personnel to the potential to seek shelter and place electronics into safe mode. The rise in flux can happen quickly, presenting an issue for personnel far from shelter, such as a lunar excursion located hours away from a habitat. If the transportation vehicle does not have a shielded shelter, shelter-in-place may be required, such as taking up a position along a crater wall or inside a lava tube.

Integrated shielding concepts

Shielding materials with high hydrogen content provide the best reduction in dose and dose equivalent, but many such materials, such as polyethylene, are poor choices for structural components. Research continues into the development of new, light-weight materials containing hydrogen and other elements with low atomic number that can be used for spacecraft and habitat construction, but to date aluminum remains the most used structural material in space. Innovative design, however, can utilize non-structural materials within a spacecraft or habitat to augment the shielding provided by structural materials that have less effective shielding properties.

Placement of water and food supplies along the walls of a spacecraft or habitat will take advantage of the hydrogen and low-Z elemental constituents of those materials to provide additional dose reduction. For example, on the ISS the walls of the sleeping quarters have been augmented with polyethylene blocks to provide additional shielding in a space where astronauts spend a sizable fraction of their time. The use of the transport model design tools discussed in the previous section will help guide mission planners in determining the best use of these non-structural, low-Z materials in the overall design.

In-situ resource utilization can reduce the cost of launching materials to missions on the Moon or Mars by using local materials for habitat construction. On the Moon, a relatively low-mass inflatable habitat can be sent to the moon, and once in place, lunar regolith (soil and rock) can be put on the structure as the primary shielding material. On Mars, one method of in-situ resource utilization is to combine the carbon dioxide atmosphere and water in the regolith to make a polymer binder that can then be used with the Martian regolith to make a solid structural material and radiation shield (see "Marsbar" in Table 3).

In general, the best solution for radiation shielding in LEO, free space, or surface operations is to use low-Z materials, especially materials with hydrogen content. Through materials development and integrated shielding designs, the principles of ALARA are applied to reduce dose while keeping mission costs affordable. As radiobiologists continue to determine the effects from the HZE and other high-energy components of space radiation, the risks of long term exposure to the low-dose, low-dose rate GCR and other components of the space radiation field will become better known. Better knowledge of the risks will in turn help mission engineers improve the state of the art in space radiation shielding.

"All HZE" represents all nuclei with heavier than He ($Z > 2$).

"Marsbar" is a target made of 85% simulated Martian regolith (rock and soil) and 15% polyethylene.

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Relevant Websites

- <https://three.jsc.nasa.gov/#section=main>—THREE: The Health Risks of Extraterrestrial Environments. Web-based articles by experts in the fields of space radiation biology and radiation protection.
- <https://srag.jsc.nasa.gov/>—SRAG: Space Radiation Analysis Group. NASA's scientists and engineers that measure and monitor crew doses.
- <https://spaceradiation.larc.nasa.gov/>—SRG: NASA Langley Research Center Space Radiation Group, NASA's scientists and engineers that develop shielding and risk models for space radiation protection.
- http://www.srl.caltech.edu/sampex/DataCenter/DATA/EventSpectra/index_ace.html—California Institute of Technology website with measured SEP data for events since the late 1990s.
- <https://www.vdl.af.mil/programs/ae9ap9/>—AP9/AE9/SPM model. Website maintained by the U.S. Air Force Research Laboratory with information and download of the AP9/AE9/SPM trapped belt radiation model.
- <https://standards.nasa.gov/standard/nasa/nasa-std-3001-vol-1>—NASA publication NASA-STD-3001. Guidance on NASA's relevant standards and limits for radiation exposure.

Health Effects and Radiation Biology

Trent L. Nichols^a and Thomas Byrne^{b,*}, ^a Nuclear Engineering Department, The University of Tennessee, Knoxville, TN, United States; and ^b Radiation Oncology Department, Cumberland Medical Center, Crossville, TN, United States

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Glossary

Activity The rate of decay or transformation of a radioactive element.

Apoptosis Cell death that is programmed into the genetics.

Atrophy The wasting of cells with some dysfunctional systems resulting in a gradual loss of homeostasis.

Bcr-Abl gene This gene codes for tyrosine kinase (a cell signal enzyme) that causes uncontrolled cell growth and is causes chronic myelogenous leukemia (CML).

Carcinoma A malignancy of the epithelial cells arising from the endoderm or ectoderm layers in the three embryonic layers.

Chromosome Two intertwined strands of complimentary DNA that contains many genes as well as sequences that are not known to code for anything.

c-Sis group A group of proteins that are responsible for glioblastomas, osteosarcomas, and melanomas.

Deoxyribonucleic acid The molecule comprised of two polynucleotide that are coiled around each other in a double helix that carry all genetic information in the ordering of the cross-linked base pairs.

DNA See deoxyribonucleic acid.

Epigenetics The alteration of gene expression resulting in phenotypic changes that is caused by the physiological environment that can be in response to environmental pressures.

Eukaryote Cells whose DNA is organized into chromosomes in a distinct nucleus that describes all organisms except eubacteria and archaeobacteria (see prokaryote).

Fluence The time integrated flux for particles in units of $[n/m^2]$.

Flux The flux, Φ , is the transfer of energy in terms of radiative energy or particles across a surface $\bar{A}$ given by $\Phi = \oint \bar{E} \cdot d\bar{A}$ which for electric flux has units $[N \cdot m^2/C]$.

Gene A portion of DNA on a chromosome whose sequence of base pairs is the code to synthesize a specific molecule, e.g. a protein or enzyme.

Glioblastoma The most common primary, i.e. arising in the brain itself, brain tumor that has a poor prognosis.

Homeostasis The condition where there is an equilibrium to maintain physiological processes for cells, tissues, or organisms.

Leukemia A cancer of the white blood cells with four common cell types: acute lymphocytic leukemia (ALL), acute myelocytic leukemia (AML), chronic lymphocytic leukemia (CLL), and chronic myelocytic leukemia (CML).

Melanoma A carcinoma of the melanocytes in the skin.

Myc gene A regulatory and proto-oncogene on chromosome 8 that is related to Burkitt's lymphoma and carcinoma of cervix, colon, breast, lung and stomach.

Necrosis Cell or tissue death.

*Retired.

Oncogene A gene that when under certain physiologic circumstances can cause the cell to become cancerous.

Osteosarcoma A primary, i.e. arising from the bone itself, bone cancer that is among the small number of cancers that originate in mesenchymal tissues.

Phenotype The set of observable characteristics of an organism from the interaction of the genetic makeup and the environment.

Philadelphia chromosome The specific activated oncogene due to translocations of chromosome 9 and 22 that results in the development of leukemia.

Prokaryote A single cell organism with an organized nucleus, nuclear membrane, or organelles that describes bacteria and cyanobacteria.

Proto-oncogene A gene that can undergo a mutation and transform into a oncogene that precipitates a cancerous transformation.

Ras protein The regulatory protein responsible for adenocarcinoma of the pancreas, colon, and thyroid as well as myeloid leukemia.

Sarcoma A malignancy of the embryonic middle layer called the mesenchymal layer that usually are less responsive to chemotherapy, radiotherapy, and immunotherapy than carcinomas.

Src-oncogenes The oncogenes are responsible for colorectal, breast, ovarian, lung, breast, and brain cancers.

Introduction

Radiation is the emission or transmission of energy through electromagnetic waves or particles that can interact with matter. Energy deposition in biologic tissue must be sufficient to break chemical bonds or to damage structures (cell membranes, internal cellular structures, chromosomes, etc.) to result in significant damage. For common chemical species found in biology, the majority of the chemical bond energies are of the order of $\sim 2\text{--}6$ electron volts (eV) which is the energy required for free radical formation. Photons are attenuated by Rayleigh and Compton scattering, photoelectric absorption, and electron-positron pair production so that penetration is measured by mass attenuation coefficients (see the National Institute of Standards and Technology [NIST] <https://www.nist.gov/pml/x-ray-mass-attenuation-coefficients>). Photons from radioactive elements are in the x-ray (124 eV to 124 keV) and gamma ray (a few keV to about 8 MeV) energies so that all are capable of bond breaks though lower energies will be near to the entry point of the radiation into the body. Radiation that does not break chemical bonds by direct damage or indirectly through other mechanisms such as free radicals will not have harmful effects. Radiation that breaks bonds is of biological significance and comprises the study of radiation biology. The bond breakage must occur in a biologically significant location that usually means in a cell nucleus. Breaking bonds and generating free radicals in interstitial spaces (outside of cells) cannot penetrate the cell membrane phospholipid bilayer to affect the nucleus. Thus, damage from all ionizing radiation is stochastic. Effects can be mild and local such a first degree sunburn to an unprotected patch of skin to more general and severe with third degree sunburn for prolonged sun exposure in a swim suit resulting in a large proportion of involved skin. In very severe cases such as exposure to the very high fluences of radiation due to an atomic bomb, thermonuclear bomb or a criticality accident, death can occur immediately or within days of the exposure. The book *"A Review of Criticality Accidents"* (McLaughlin et al., 2000) provides an excellent review of criticality accidents and can provide some understanding of the important factors for radiation exposure: distance, time, and shielding. Other important factors for radiation exposure include fluence, energy, and type of radiation.

It is estimated that humans are composed of 30–40 trillion cells. For a person to survive, they must maintain homeostasis, which is the ability of an organism to regulate physical and chemical processes to maintain an equilibrium that is specific for a type of cell. Cellular equilibrium means that chemicals must be within tolerances that are specific for the organism and cell type. Intravascular equilibrium are the normal ranges that patients receive when they have a routine chemistry panel. More specifically, human neural cells have a low intracellular sodium concentration but a high potassium that is exactly the opposite of the intravascular space (blood). So, for humans, intravascular (blood) homeostasis is maintaining the chemistry values on the typical panel but also refers to hormones, cell counts (white cells, red cells, and platelets), blood proteins, and other blood constituents that have normal ranges. So equilibrium conditions are dependent upon the organism and the cell type in the organism. When homeostasis is compromised, cells, and if extensive, the organism will cease to function normally. If the loss of homeostasis persists, cells (and the organism) will die when the excursions from normal are severe and persist long enough to create a cascade of systems that, in turn, become abnormal. Cell's and organisms, such as humans, have elaborate mechanisms to return the system to normal and maintain homeostasis. Thus, a small cut on an extremity will cause a localized loss of homeostasis, or drinking too much water can lower the level of sodium in the blood sufficiently to result in a generalized loss of homeostasis. If the cut is large and not addressed, the person can lose so much blood that there is a generalized loss of homeostasis which results in death after some time as the excursion from homeostasis becomes more pronounced. Likewise, a person with psychogenic water intoxication (a psychological condition in which the patient believes themselves to need to drink more water when they are actually overhydrated) will continue to drink too much water until the very low blood sodium will result in severe problems leading to death. So for radiation to cause serious immediate or near immediate (hours to days) problems, there must be a significant disruption in a person's homeostasis.

Sunburns are illustrative because they are common, caused by a familiar form of radiation, and a sunburn is a condition experienced by most people at one time or another. Sunburns can be mild with reddening of the skin or severe with third degree burns as well as be confined to a small area or generalized. Third degree sunburns covering a large portion of the body's surface area are life threatening conditions since the skin is so badly damaged that it cannot prevent large fluid losses and cannot protect from infections, i.e. cannot maintain homeostasis. The body loses homeostasis in widespread third degree burns. Sunburns also illustrate that in our environment, radiation is ubiquitous and that radiation in lower amounts is harmless. Low level light is used to read, do office and home activities, and enjoy the outdoors but a large dose of outdoor light can be harmful. Case in point, one can shine a flashlight at some point on the skin surface for a very long time with absolutely no ill effects because the low energy photons cannot damage molecular bonds or damage deoxyribonucleic acid (DNA). Whereas a relatively short time of sunshine on an area of skin can result in the radiation damage of a sunburn.

Sunburns also illustrate how the body responds to injuries affects what the extent and severity of the injury. When a sunburn occurs, part of the swelling is caused by direct cellular damage. The cellular damage releases cellular debris that triggers an immune (inflammatory) response with an accretion of white blood cells consisting of monocytes that mature into macrophages which are phagocyte (phagocytosis is the process of devouring and digest cells, bacteria, or viruses), lymphocytes, and neutrophils. These white cells and the cellular debris will cause capillaries to leak leading to localized edema (swelling) and nerves resulting in pain and tenderness. The body's response is to destroy bacteria and other pathogens and then to sweep the phagocytized pathogens out as quickly through the increased blood flow. Inflammatory responses result in normal, uninvolved cells to be damaged, so it is not ideal for all insults. The body reacts to all insults with inflammation designed to destroy pathogens even when the process does not include pathogens, i.e. a sunburn that is sterile though second- and third-degree burns can become infected. The body has a limited number of responses. Increased blood flow gets more nutrients and oxygen that helps the repair process but, in the case of third-degree burns, the increase blood flow exacerbates the fluid losses. So cellular damage leads to an inflammatory response that has both helpful and harmful effects.

Cells must also maintain homeostasis. If a cell loses its ability to maintain homeostasis due to damage from age or its local environment, it will die and release its cellular contents which attracts phagocytes to digest it. If enough cells are damaged or die in a local area, an inflammatory response will be triggered whereas a single cell may result in little inflammation. Direct damage by ionizing radiation on a cellular level is localized along the track because radiation (photons, ions, etc.) are small compared to the size of a cell and its organelles (specialized structures in a cell), but indirect effects, chemical bond breakage, from chemical species such as free radicals that have half-lives of only a few microseconds can cause damage, only approximately 30 Å from the initially generated free radicals from the track (Hutchinson, 1957). The effects are localized on the cellular level unless there is a large fluence of high energy radiation. In the case of sunburn, the damage occurs at the cellular level but since the ultraviolet photon fluence is very large, the affected area can be large too.

So, radiation in the form of electromagnetic waves (photons) and particles, both charged or uncharged, causes damage on the cellular level that can disrupt homeostasis. The damage can lead to cell death releasing cellular debris that can, if enough cells are damaged, trigger an inflammatory response. If this is widespread enough, then there can be a loss of homeostasis leading to illness and possibly death. Remember though that the body has many mechanisms to return to homeostasis.

Additionally, radiation can result in DNA damage breaking one or both strands of DNA. Cells have mechanisms to locate, identify defects, and repair damaged DNA. Single strand breaks are easily repaired but double strand breaks can be very difficult and may not be repairable (Frankenberg, 1984). It is estimated by Lodish et al. in *Molecular Biology of the Cell* (2004) that amazingly 10,000 to as many as 1 million molecular lesions per cell per day are successfully repaired from DNA damaged during normal metabolic function and environmental factors including radiation. This means that a single event is unlikely to result in a problem that leads to eventual cellular demise. Rather it requires many injuries in a short period of time or very significant damages such as a double strand DNA breaks. Should a defect in DNA not be detected and repaired properly, it may lead to a problem that will, in time, lead to cell death or a viable mutation. It can also lead to a defect that does not lead to cell death but some other dysfunction. Some of the mutations can lead to a cancer as is discussed below.

To have a clinically significant effect, energy transfer to cells by radiation must be sufficiently large to break chemical bonds and have a sufficient fluence that results in a dose that causes significant organelle damage, foils cellular repair mechanisms, or result in irreparable DNA damage. So that low fluence, low energy radiation has a lower probability of significant human morbidity and mortality. High energy radiation must typically have enough fluence to be statistically likely that the above damage occurs often enough to see clinically significant results. Clearly, being exposed to high doses (more than about 0.1 Gray, see glossary or below) of radiation in a very short time period such as the atomic bomb survivors in Hiroshima and Nagasaki and those exposed in criticality accidents where the dose was delivered on the order of a microsecond have significant morbidity (health issues). However, if the same dose is spread over many years, cells have ample time to repair DNA and other damage as compared to the enormous number of chromosomes damaged as well as identification and repair proteins that are damaged in a microsecond pulse of high energy radiation (see, for example, Rahu et al., 2013). High doses over short times (seconds) can overwhelm the ability of cells to identify and accurately repair the damage leading to severe injuries or death.

Radiation exposure data of workers in various fields who deal with radiation, who may be exposed to low doses (less than about 0.1 Sv) (see glossary or below) of radiation over relatively long periods of time, have been studied over the years to determine health effects. The Million Worker Study is a large cohort of workers in all aspects of radiation associated industries. The Million Worker Study has several sections submitted for publication and a few sections have been published. A brief discussion can be found in the conference proceedings listed (Sanders and Sanders, 2018). A summary table shows no statistically significant

risk for cancer though there are some possible relationships. The principal investigator on the Million Worker Study, J. D. Boice, Jr., notes (Boice, 2019) that

“the excess numbers of cancers associated with low-level radiation exposure, is so very small that it cannot be seen against the very high background occurrence of cancer in the population, i.e. a lifetime risk of incidence reaching up to about 38% (i.e. 1 in 3 persons will develop a cancer in their lifetime).”

It is important to understand that the 38% lifetime incidence of cancer is due to all causes such as from chemical irritants such as hepatocellular carcinoma due to carbon tetrachloride (a once common cleaning agent), environmental factors such as radon exposure, and genetic factors such as the presence of the BRCA 1 or 2 gene mutations (Breast CAncer susceptibility gene) that increases the risk of ovarian cancer by age 70 to 39% and 11% respectively while it increases the risk of breast cancer by age 80 to 72% and 69% respectively. These causes account for the lifetime risk of cancer. Consider lung cancer that has four common cell types: small cell carcinoma, large cell carcinoma, squamous cell carcinoma, and adenocarcinoma. Of those four cell types, only adenocarcinoma is not associated with cigarette smoking. Males smokers are 23 and females 13 times more likely than never smokers to develop lung cancer and account for 90% and 80% of the deaths from lung cancer (Finkelstein, 1996). So smoking is a clinically significant disease where a physician needs to know whether a patient has ever smoked and, if so, the pack years [average number of packs smoked per day times the number of years that the patient smoked] is an important historical fact since it is the dose of smoking. A heavy smoking history is 30 pack years. For those patients who have a history of heavy smoking, smoke now or have quit within the past 15 years, and are between 55 and 80 years old, yearly screening with low dose computed tomography scanning is recommended. Since so far there is no clear relationship between low dose radiation exposure and cancer (Boice, 2017), the same kind of history is not required (unless the patient is part of a study) and since there is no known associated lung cancer from low LET radiation, there can be no routine screening beyond the standards currently be employed as the standard of care.

Additionally, the standardized mortality ratios in the Mound Study of ^{210}Po workers (Boice et al., 2014) showed that the fewer workers had died than was expected as the “vital status was determined for 98.7% of the workers of which 3,681 had died compared with 4,073.9 expected (standardized mortality ratio [SMR] 0.90; 95% confidence interval [CI] 0.88–0.93)” and that “among the 4977 radiation workers, all cancers taken together (SMR 0.86; 95% CI 0.79–0.93), lung cancer (SMR 0.85; 95% CI 0.74–0.98), and other types of cancer were not significantly elevated” (Boice et al., 2014). These findings agree with previous, dated findings of no statistically significant relationship between cancer and low dose radiation exposure. This implies that there is not a clinically significant relationship between low dose (<0.1 Sv) radiation and cancer. In other words, there is no reason for a physician to inquire about radiation exposure history unless the exposure is to a high dose (0.1 Sv or higher) exposure because the relationship is insufficient. Asking questions in regard to exposure to heavy metals might be a pertinent question on certain circumstances because most heavy metals that are encountered in the nuclear industry are also chemical poisons whose harm is not from radiation but from chemistry.

Types of ionizing radiation

Important types of ionizing radiation in radiation biology include charged particles that possess enough energy to cause ionization such as negatively charged beta particles (electrons) and positively charged alpha particles (helium nuclei). Ionization is also caused by uncharged particles such as photons and neutrons that have enough energy to directly form ionized particles (such as free radicals) by photon interactions with electrons and by neutrons scattering with, primarily, protons. Nuclear transformations can result from gamma rays as in (γ, n) , (γ, p) , and $(\gamma, \text{fission})$ reactions or neutrons as in (n, α) and (n, p) reactions that will add to the total dose. [The notation (a,b) is a reaction with “a” entering the nucleus and “b” exiting the nucleus. The symbol before the parentheses is the initial atomic species, and the symbol after the parentheses is the final atomic species. For example, given the reaction $^{10}\text{B}(n, \alpha)^7\text{Li}$, ^{10}B is the target isotope and ^7Li is the product isotope. A neutron is absorbed by the ^{10}B and the excited nucleus decay by emitting an alpha particle, a photon, and ^7Li .]

Alpha particles are helium nuclei with two neutrons and two protons but stripped of its two electrons; thus, an alpha particle carries a charge of 2^+ (the two positively charged protons in the nucleus are no longer balanced by the two negatively charged electrons). In biological tissues, alpha particles can cause considerable damage on the cellular level because of the transfer of its kinetic energy to ionization of atoms and molecules over its range. For example, the approximately $10\ \mu\text{m}$ track of a 2 MeV alpha particle in tissue is essentially a straight line with energy deposited along the path but with a larger energy deposition at the end of the track in what is called the Bragg peak as shown in Fig. 1. The shape of the energy distribution resembles a teardrop with the majority of the energy deposited around the Bragg peak. Electrons are considerably lighter than alpha particles [the rest mass of an electron is $m_e = 9.109 \times 10^{-31}$ kg and the alpha $m_\alpha = 6.644 \times 10^{-27}$ kg so $m_\alpha/m_e = 7294$] so beta particles (electrons) scatter in random directions with other electrons associated with components of cells; thus, they have a path similar to a random walk. Energy is deposited all along the track of a heavy charged particle and for a short distance on either side of the track that is approximated by a teardrop.

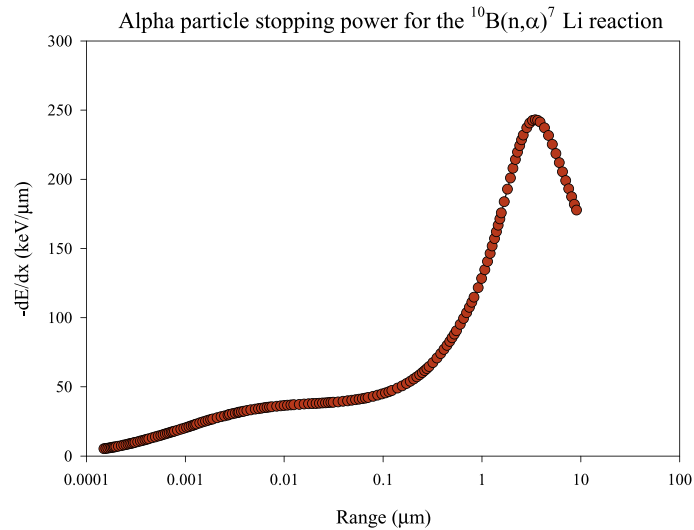


Fig. 1 The alpha particle stopping power for the $^{10}\text{B}(n, \alpha \gamma) ^7\text{Li}$ reaction showing the energy deposition per unit length as a function of the remaining range and the Bragg peak near the end of the track.

For ionizing radiation traversing matter, the mean energy lost, dE_l , in a path length, dl , is called the linear energy transfer (LET) and is a linear approximation of the actual energy deposited below a cutoff energy and its geometry. The definition of the International Commission on Radiation Protection (ICRP) publication 103 (2007) for LET is used where dE is the mean energy lost by charged particle collisions with electrons while traversing a distance dl is:

$$\text{LET} = \frac{dE}{dl}$$

LET differs from stopping powers used in physics as stopping powers include all radiation (i.e. neutrons and photons) as well as charged particles and stopping powers approximate the actual energy loss per unit distance traveled through the energy absorbing medium as shown below rather than a linear approximate.

$$\text{SP}(E) = -\frac{dE}{dx}$$

Stopping powers are not linear and represent the energy lost traversing matter but are more complex to calculate. For example, the stopping power graph for a 1470 keV alpha particle from the $^{10}\text{B}(n, \alpha \gamma) ^7\text{Li}$ in air is shown in **Fig. 1** where the Bragg peak is seen. Linear energy transfer (LET) is a simplification that works well in most circumstances whereas others benefit from a full stopping power that has the Bragg peak. A full discussion of this can be found in textbooks on microdosimetry such as (Rossi and Zaider, 1996).

The topic of stopping powers is important at the level of cellular doses that constitutes the field of microdosimetry. Interest in the theory of energetic ions interacting with matter and slowing down dates back to Marie Curie's work in 1898–99 (Curie, 1900). Niels Bohr was the first to provide a theory of stopping powers (Bohr, 1913; Bohr, 1915) following the experiments of Hans Geiger and Ernest Marsden (1909) with Ernest Rutherford's theoretical explanation (Rutherford, 1911). Bohr did not take into account quantum effects that were first addressed by Hans Bethe (1930, 1932) that was later refined by Felix Bloch (1933a,b) and then by Ugo Fano (1963). The history of determination of stopping powers is well reviewed by James Ziegler (1999). The major contribution to the stopping power for ions is due to the electromagnetic interaction between the target electrons and the positively charged ion. The stopping power, SP, may be written as:

$$\text{SP} = \frac{dE}{dx} = \frac{\kappa Z_2}{\beta^2} Z_1^2 [L_0(\beta) + Z_1 L_1(\beta) + Z_2^2 L_2(\beta) \dots]$$

where

$$\kappa = 4\pi r_0^2 m_e c^2,$$

$$Z_1 = \text{ion charge},$$

$$Z_2 = \text{target atom charge},$$

$$\beta = \frac{v}{c},$$

$$r_0 \equiv \frac{e^2}{m_e c^2},$$

m_e = electron mass,

c = speed of light,

v = ion velocity,

and

$$L(\beta) = [L_0(\beta) + Z_1 L_1(\beta) + Z_2^2 L_2(\beta) \dots]$$

where L_0 is the stopping number that contains corrections to the stopping power. The primary stopping number, L_0 , contains the largest correction and [Fano \(1963\)](#) expressed it theoretically by:

$$L_0 = \frac{1}{2} \ln \left(\frac{2m_e c^2 \beta^2 \Delta E_{max}}{1 - \beta^2} \right) - \beta^2 - \frac{C}{Z_2} \ln(I) - \frac{\delta}{2}$$

where C/Z_2 is the shell correction for the target atom, $\langle I \rangle$, is the mean ionization energy of the target atom, and $\delta/2$ is the density correction. The largest possible energy loss in a single collision with a free electron is given by:

$$\Delta E_{max} = \left(\frac{2m_e c^2 \beta^2}{1 - \beta^2} \right) \left(1 + \frac{2m_e}{M_i (1 - \beta^2)^{1/2}} + \left(\frac{m_e}{M_i} \right)^2 \right)^{-1}$$

where M_i is the mass of the ion. The shell correction term takes into account the assumption that the ion velocity is much larger than the target electron velocity. The calculation requires careful accounting of the ion with the electrons in the various electron shells in the target. The mean ionization corrects for the quantum mechanical energy levels that target electrons have available for transfer. Finally, the density term corrects for polarization effects in the target.

The L_1 term is called the Barkas correction after the work by [Barkas et al. \(1956\)](#) and the L_2 term is the Bloch term. These terms deal primarily with two experimental observations. The first is that two ions of the same mass and velocity in the same target have different ranges where the only difference is for different sign of the ion charge. The other effect is the corrections required for the magnitude of the charge. For example, according to the basic Bethe-Bloch equation, an ion with charge +2 should have four times the stopping of a similar particle of charge +1 but experimentally the stopping exceeds a factor of four.

Alpha particles are produced in some nuclear decays such as radium ([Evans, 1974](#)) and thorium. Radium and thorium have been responsible for bone cancers (osteosarcoma and Ewing's sarcoma) in radium dial painters ([Rowland, 1978](#)) since both radioisotopes are deposited in bone in humans. Radon gas and its decay products inhaled by uranium miners is associated with the development of lung cancers ([Archer, 1976](#)). In general, isotopes of uranium that decay into isotopes of thorium that decay into isotopes of radium that decay into isotopes of radon gas which is the source of the radon found in uranium mines. In the radium mines of Schneeberg, Germany and Joachimstal, Czechoslovakia over half of the miners that died, had deaths attributed to lung cancer that presented at an early age. These uranium alpha particles do not penetrate skin so are not a hazard externally, but inhaled ([Harley, 1972](#)) or ingested alpha emitters do great damage to living tissues. It is always important to consider all risk factors that modify risks for cancer.

Beta particles are electrons emitted from the nucleus when a neutron transforms into a proton and electron. Commonly used beta emitters that have applications in university and commercial research are ${}^3\text{H}$ (called tritium), ${}^{14}\text{C}$, and ${}^{32}\text{P}$. Beta emitters of significant energy in the human body include the naturally occurring background radioisotopes ${}^{40}\text{K}$ and ${}^{14}\text{C}$. Potassium-40 (${}^{40}\text{K}$) isotopes ([Hardy, 1978](#)) are present among all naturally occurring potassium atoms at 0.012% (120 ppm) and release beta particles with enough energy to cause a dose of about 0.18 mGy (milliGray where a Gray is the measure of absorbed dose described below) in a 70 kg man in 1 year. This is the most energetic of the internal background isotopes. Carbon-14 (${}^{14}\text{C}$) is produced in the atmosphere as cosmic ray neutrons hit nitrogen nuclei and the nitrogen absorbs the neutron forming ${}^{14}\text{C}$. The energy ${}^{14}\text{C}$ releases in a 70 kg man in 1 year is about 0.01 mGy. Potassium-40 and carbon-14 combined are the greatest internal natural sources, which excludes radon, of dose in humans ([Oakley, 1972](#)). Other radioactive elements add far lesser amounts to the internal radiation of humans, and sensitive equipment can measure these small amounts of natural background radiation.

Gamma rays and X-rays are energetic photons (electromagnetic waves) that have the same properties as all other electromagnetic energy, have short wavelengths, and travel at the speed of light. Since gamma rays and X-rays are energetic photons, they penetrate biological tissue to a very deep level. For example, a 500 keV photon has a penetration depth of 10 cm in a four component tissue sample where the mass attenuation coefficient μ/ρ is 9.593E-02 (g/cm²) and the range is defined to be where the ratio of the intensity to the initial intensity is 0.37 (<https://physics.nist.gov/PhysRefData/XrayMassCoef/tab4.html>). Gamma rays come from transitions within the nucleus whereas X-rays come from transitions of the electrons around the nucleus, but the two are identical if they have the same energy (the identical wavelengths). An important radioisotope used in the treatment of thyroid cancers is radioactive ${}^{131}\text{I}$ that most often (89% abundance) emits 606 keV β^- particles and a 364 keV gamma ray (81% abundance). The ${}^{131}\text{I}$ is provided to the patient with the knowledge that iodine is taken up exclusively by the thyroid in doses that can kill the cancer cells.

However, healthy thyroid cells also up-take iodine. In the past, nuclear weapons testing released both $^{125}_{53}\text{I}$ and $^{131}_{53}\text{I}$ with both isotopes having up-take exclusively in the thyroid gland (Beierwaltes, 1963), salivary gland, and gastric mucosa adding to the total dose to the population with increases in cancers observed (Carter, 1977).

Understanding the reactions and processes on the cellular level and modeling those is area of study called microdosimetry (see Rossi and Zaider, 1996). Microdosimetry begins to explore the damage that occurs to a cell as a function of the actual energy deposition in the cell. A cell dies if it cannot maintain homeostasis that results from radiation caused damage. Cells have been described as little bags of chemicals, which implies a loose collection of randomly interacting molecules that have no relationship to the actual highly organized eukaryotic human cells (eukaryote cells have an organized nucleus where the DNA is organized into chromosomes and includes all organisms except eubacteria and archaeobacteria).

Cells have substructures that are called organelles. Eukaryote cells have a double phospholipid cell membrane that can sustain injuries and heals itself by the physics of the double phospholipid structure. Only damage that removes a large portion of the cell membrane that allows the contents to leak out will kill the cell with the immediate loss of homeostasis. The cell membrane encloses a number of organelles as shown in Table 1.

The DNA in a chromosome will uncoil a section of DNA that will be read by messenger ribonucleic acid (mRNA) which at a ribosome works with ribosomal RNA (rRNA) and with transfer RNA (tRNA) to construct proteins. Mitochondria are believed to be of bacterial origin, have their own DNA, and use the electron transport chain to supply energy for cell functions. Red blood cells have no mitochondria whereas hepatic (liver) cells may have 2000 mitochondria. Whenever a section of DNA is unwound for reading by messenger DNA, it provides a larger probability of interactions with radiation at the microscopic level due to geometric effects.

The largest likelihood for DNA damage from ionizing radiation occurs when a cell is dividing. At that time, the nucleus disappears, all chromosomes are unwound, and all the DNA is copied. The basic ideas are straightforward, but the mechanisms are complex to assure that the copy is exact. So tissues that are rapidly dividing are the most vulnerable such as cells in the bone marrow producing red blood cells, white blood cells, and platelets that are rapidly dividing. DNA damage in the bone marrow can result in a number of diseases but in particular leukemias. Fortunately, leukemias can be much more effectively treated now than in the past as stem cell transplants are being improved.

Microdosimetry looks at energy deposition damage at the cellular level. Usually, the calculations focus primarily on the total dose delivered to the cell and the dose delivered to the nucleus. Thus, microdosimetry is based on stochastic interactions between ionizing radiation and a cell and its nucleus. Energy deposition in the nucleus can lead to irreparable DNA damage that cause no problems (a section of DNA that does not code for anything), or lead to loss of homeostasis.

Protection from external radiation sources may be reduced by limiting time of exposure, increasing distance from the source (dose decreases by an inverse square law), and shielding. The ICRP report 103 (2007) makes the point that human exposure to radioactive substances should always be minimized no matter the estimated risk. Even though low level (less than 0.1 Sv) radiation has little likelihood of causing irreparable damage that results in a malignancy or death from loss of homeostasis, it is good policy to reduce the danger from chemical toxicity as well as radiation from heavy metals associated with reactors as fuel and waste products.

The Curie and the Becquerel

Marie Curie discovered radium and polonium (Nobel Prize in Physics 1903 and Nobel Prize in Chemistry 1911) and her daughter Irene (Nobel Prize in Chemistry 1935) discovered artificial radioactivity (where an element is made radioactive by irradiation) both are thought to have died of leukemia due to their exposure to radioactive elements in their experiments. To honor the Curie's family as multiple winner of the Nobel Prize, the Système International (SI) unit of activity was originally the Curie. Activity is the rate of decay or transformation of a radioactive element.

Since 1975 the SI unit for activity is the Becquerel named for Antoine Becquerel (Nobel Prize in Physics 1903) who discovered radioactivity from uranium compounds in 1896. Perhaps the first record of recognized biological radiation effects was when Becquerel left a radium container in his vest pocket and experienced skin redness (erythema) with ulceration requiring weeks to heal.

These units of activity are defined as:

Table 1 Eukaryotic cell organelles.

Organelle	Function
Centrosome	Forms microtubules that helps to transport molecules throughout the cell and aids in cell cycle progression
Golgi apparatus	Protein modification and transport
Lysozyme	Protein degradation
Mitochondria	Energy production via the electron transport chain
Nucleolus	Structure in the nucleus for ribosome biogenesis
Nucleus	Contains DNA along with repair and replication proteins
Peroxisome	Lipid degradation
Rough endoplasmic reticulum	Protein synthesis (has ribosomes)
Smooth endoplasmic reticulum	Lipid synthesis and detoxification

1 Bq = 1 disintegration per second.

1 Ci = 3.7×10^{10} disintegrations per second.

The rate of decay or activity is an exponential function and can easily be shown to be:

$$A = A_0 e^{-\lambda t}$$

where A is the activity at some time, t , with a decay constant of λ and an initial activity of A_0 . The half-life is determined by the time required for the activity to decrease to $\frac{1}{2}$ of the initial activity, A_0 , or

$$A = \frac{A_0}{2}$$

so that

$$e^{-\lambda t_{1/2}} = \frac{1}{2}$$

or

$$\lambda t_{1/2} = \ln(2)$$

which gives this relation for the half-life

$$t_{1/2} = \frac{\ln(2)}{\lambda} \cong \frac{0.693}{\lambda}$$

Protective quantities and absorbed dose

The ICRP report 103 defined dosimetric quantities in terms of protective quantities to relate energy deposited in organs in the body to radiation risk (detriment). The fundamental physical quantity is the absorbed dose, D , that is measured in units of Gray (Gy) where 1 Gy is 1 joule per kilogram (1 J/kg). The rad is the traditional unit of absorbed dose that is 62.4×10^6 MeV per gram and $1 \text{ Gy} \equiv 100 \text{ rads}$ where an electron volt (eV) is the kinetic energy that a single electron acquires when accelerated in a vacuum from rest through an electric potential difference of 1 V or 1.6022×10^{19} J. The absorbed dose is

$$D = \frac{d\bar{E}}{dm}$$

where $d\bar{E}$ is the mean energy deposited into the mass, dm , by ionizing radiation.

When ionizing radiation interacts with human tissue, damage that results is dependent on the energy of the particle and on the cell types in the tissue. There are some general findings that relate to detrimental effects: (1) ingested or inhaled alpha particle emitters do more damage than beta particle emitters; (2) tissue damage increases as the distance over which the ion travels is slowed to release ionization; (3) tissue in humans and animal model experiments with high cell division rates display increased damage (e.g. blood forming cells leading to leukemia) because, during cell division, the otherwise tightly folded DNA that makes it a small target since unwinding it increases the likelihood of interaction with a particle or electromagnetic wave that is also generating free radicals that diffuse a short distance before acting upon a molecule. As a particle interacts with matter, it slows down gradually with energy being deposited along the path of the particle as it travels through the tissue. LET approximates the energy transfer as a straight line with uniform energy loss all along its path.

In the context of radiobiology, detriment refers to the total health effect to a radiation exposed group and its descendants that has principal components from stochastic quantities including such things as cancer, inheritable defects, and premature demise. Radiation is assigned a tissue weighting factor, w_T , that is unique for each tissue, type of radiation, and energy of the radiation. The effective dose is the energy deposition from all radiation in a given tissue and is given by:

$$E_D = \sum_T w_T \sum_R w_R D_{T,R} = \sum_T w_T H_T$$

where w_T is the tissue weighting factor, $w_R D_{T,R}$ or H_T is the equivalent dose in a tissue or organ where weighting factors are dimensionless, and doses are in Gray $\left(\frac{\text{J}}{\text{kg}}\right)$. The system from ICRP provides a firm foundation to determine the effect of a type of radiation dose to a tissue and the likely detriment that is likely to be suffered. Please see Table 2 for some examples of radiation weighting factors and refer to ICRP publication 103 for more details on doses.

DNA strand breaks—Single and double, chromosome aberrations

Deoxyribonucleic acid (DNA) is the genetic carrier for life, consists of two strands of polynucleotides connected to one another by base pairs, and is wound into chromosomes in the nucleus of cells. This wound structure is a complex of control elements (histone proteins), enzymes for reproduction of the double helix before cell division, and ribonucleic acid structures to replicate proteins.

Table 2 Radiation weighting factors for different types of radiation.

Type of radiation	Radiation weighting factor, w_R
Photons	1
Electrons and muons	1
Protons and charged pions	2
Alpha particles, fission fragments, heavy ions	20
Neutrons	A continuous function of neutron energy

From ICRP publication 103, 2007.

Radiation damage to the nucleus may result in single or double strand breaks that require repair. If the damage is in an area of DNA (Muller, 1927; Lea, 1955) that is crucial for life the cell will die. If the damage activates genes that lead to pre-cancerous cellular changes then time and additional cellular insults may lead to the development of carcinomas, sarcomas, or leukemias (more about this later in the article).

The radiation interaction with water may result in reactive hydroxyl radicals that proceed to damage DNA by breaking chemical bonds. The detection of radiation exposure in the nucleus of lymphocytes (a white blood cell) is observed as chromosome breaks (Littlefield, 1991). Specific nuclear fluorescence dyes display chromosome damage in the form of inversions, deletions, and rings. Anaphase bridges may easily be observed during cell division. Researchers have found chromosome damage in subjects' years after exposure, for example Japanese atomic bomb survivors have observable chromosome aberrations many years after exposure (Doida, 1965).

Cell survival curves, DNA repair

In the laboratory established animal and human cell lines may be irradiated and seeded to petri plates to observe cells forming colonies. Cell survival curves account for the plating efficiency or number of colonies grown divided by the number of cells plated with no radiation, this is done first. After irradiation, the surviving fraction is the number of colonies grown divided by the cells plated times the plating efficiency expressed as a percentage. The shape of survival curves for cells is plotted with dose on a linear scale and surviving fraction on a logarithmic scale (Fig. 2). At low LET radiation the curve starts out straight on the log-linear plot; at higher dose the curve bends as a function of dose. If cells are irradiated with high LET radiation of neutrons the curve is almost a straight line from the origin (Byrne, 1997).

Cell survival data for human cells under radiation treatment for cancer tend to follow a linear-quadratic relationship as seen in:

$$S = e^{-\alpha*d - \beta*d^2}$$

in which S is the fraction of cells that are surviving a dose, d, with α and β as constants of the linear and quadratic components of cell killing.

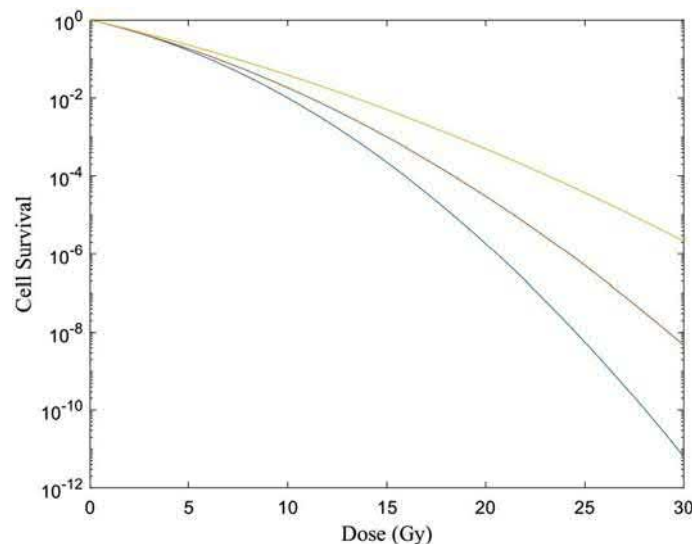


Fig. 2 Illustrative cell survival curves showing the linear quadratic nature of cell demise in radiation with different values of α and β taken from the literature.

Modifiers to radiation effects, oxygen effect and radioprotectors

Oxygen enhances the effects of X-rays, called the oxygen enhancement effect. Oxygen must be present at the time of exposure because, when irradiated with ionizing radiations, it produces free radicals that will make the damage permanent. In the absence of oxygen, the damage produced by X-rays can be repaired. This is important in the treatment of solid tumors due to the area of hypoxia in the center of the tumor mass. In standard photon radiation therapy, the patient is given a dose daily or 24 h between doses. This allows re-oxygenation of the hypoxic cells in the solid tumor. This is repeated 5 days in a row with the weekend off for recovery for a total of 6 weeks.

Radioprotective compounds have been studied and work by scavenging free radicals produced by ionizing radiation. US astronauts on lunar trips carried amifostine (Grdina, 2002) sold under the trade name Ethiol. This chemical, $\text{NH}_2(\text{CH}_2)_3\text{NHCH}_2\text{CH}_2\text{SPO}_3\text{H}_2$, is a phosphorothioate that is not reactive until the enzyme alkaline phosphatase removes the phosphorylated end of the molecule to allow scavenging of the free radicals. It is radioprotective for ionizing radiation.

Cancer

A brief aside concerning cancers that are also termed malignancies. Normal cells are able to determine when there are enough cells in a given tissue and cease cell division. Cancer cells have no such inhibition and continue with cellular division unabated. Often, there are visually obvious abnormalities in the DNA of the cancerous cells though all cancers have some sort of DNA abnormality. The unabated cancer growth causes a multitude of problems that, untreated, usually result in the death of the patient. Human malignancies are divided primarily into two groups where the most common cancers that are called carcinomas arise from the epithelial germ cell layer in the developing embryo. The other group of cancers are less common and arise from the mesenchymal germ cell layer in the developing embryo. As a generalization, sarcomas are more difficult to treat and often have poor prognoses. Some carcinomas are curable, though some have a uniformly fatal prognosis despite aggressive efforts to treat the disease.

In order for a cancer to spread by direct extension (growing larger) or through distant metastases (a very small nidus of cells break off the many tumor and travel via the cardiovascular or lymphatic system), a cell's DNA must have been altered, the alteration must not be fatal to the cell, must not prevent cell division, and must remove the inhibition to unbridled growth. Fortunately, most damage to DNA results in a fatal alteration or prevents cell division. When all conditions are present, the malignancy will grow. It is now understood that the immune system likely rids the body of many early malignancies and potential cancers are prevented by cellular repair mechanisms. Carcinomas and sarcomas that are discovered by physicians are usually causing significant problems.

How does DNA become damaged? One source of DNA damage occurs during cell division when a strand can be misread by the DNA polymerase, but the proof reading and error checking keep such events from being common. Another source of damage is from chemicals that alter or bind to the DNA. Ionizing radiation can form a chemical species called free radicals (a chemical species an unpaired valence electron or an open electron shell) that can chemically break bonds in DNA or, less often, the ionizing radiation can directly damage it. The energy loss along the track is not linear but rather teardrop shaped around the Bragg peak where the majority of energy is deposited and therefore the most damage is done. Free radicals from ionizing radiation tracks passing near or through the nucleus can cause a break in a single strand of DNA and less often a double strand break. Single strand breaks can be repaired accurately by a cell whereas double strand breaks cannot usually be correctly repaired. For a cellular DNA to be damaged beyond repair, it must experience a double strand break or many single strand breaks (that increases the possibility of the strands not being accurately repaired), there must be a very large number of ionizing particles producing many free radicals or rarely, a direct strike by a heavy particle, i.e. alpha particle.

Another important concept is that there must be enough energy to cause a chemical alteration to the DNA. That is the process of irradiation must result in multiple single DNA breaks or a double strand break that is difficult for the cellular repair mechanisms to successfully detect and repair the damage. There can be other damages that can cause the cell to lose homeostasis but only DNA damage will result in a malignancy. So, a large amount of energy in a short period of time such as a criticality accident or an nuclear detonation can cause immediate or short term loss of homeostasis. Such exposures and long term exposures to ionizing radiation can result in malignancies that lead to excess morbidity and mortality.

Oncogenes and cancer, radiation carcinogenesis, leukemia, and solid tumors

When cells are damaged, they normally progress to a cellular death (apoptosis). Cellular normal genes that are involved in grow and cell division may be mutated to proto-oncogenes and expressed in highly active states. With additional cellular mutations the proto-oncogenes may convert to oncogenes that proliferate and with other mutated genes form primary cancer cells.

Research in cell growth and differentiation has allowed for the discovery of dozens of proto-oncogenes that are usually involved in cell signal events and cell division. The Myc gene (a regulatory proto-oncogene) of Burkitt's lymphoma is a chromosomal translocation that moves a section of chromosome close to the enhancer area. The Philadelphia chromosome is the specific oncogene activation due to translocations of chromosome 9 and 22 resulting in the development of leukemia. The Bcr-Abl gene codes for tyrosine kinase (a cell signal) causing uncontrolled cell growth forming chronic myelogenous leukemia (CML).

The Src-family of oncogenes are responsible for colorectal, breast, ovarian, lung, breast, and brain cancers. The c-Sis group are responsible for glioblastomas (most common primary brain malignancy), osteosarcomas, and melanomas. Ras protein is a regulatory agent that is responsible for adenocarcinoma of the pancreas, colon, and thyroid as well as myeloid leukemia.

The ability of radiation to mutate DNA and cause sub-lethal damage provides for the activation of proto-oncogenes. The disruption of cell cycle regulation and the activation of receptors of cell proliferation allow for uncontrolled cell growth. Tumors that arise from embryological lining layers of ectoderm (skin and nervous system) or endoderm (digestive track) are called carcinomas. Tumors that arise from mesoderm, the embryological middle layer (bone and muscle), are called sarcomas. The white blood cells give rise to leukemia and lymphomas.

The Life Span Study of Japanese atomic bomb survivors is the primary source for estimating carcinogenic risk (Roberts, 1987). Leukemias have been linked to radiation exposure and show a dose response which is well described by a linear-quadratic dose response (Richardson et al., 2009). Solid tumors incident data are better fitted to a linear dose response. When comparing atomic bomb survivors to Chernobyl emergency workers both data sets for leukemia display a much higher risk at a given dose than other cancers (rapidly dividing white blood cells mutate to form leukemia). In both studies, excess risk has been observed as early as 2 to 5 years post exposure. Solid tumors of breast, lung (Yamamoto, 1987), and thyroid also display an association with radiation exposure, however the risk estimates are not as strong as for leukemia. A more recent study by Rahu et al. of Chernobyl cleanup workers from the three Baltic countries of Estonia, Latvia, and Lithuania found different results:

"the study of cancer risk in a cohort of Chernobyl cleanup workers from three Baltic countries revealed an elevated proportion of thyroid cancers in relation to the general male population; however, this finding could be attributable, at least in part, to more intensive medical examination, including thyroid screening. No excess of leukaemia cases or radiation-related cancers as a group was found."

So even in the extreme circumstances of trying to control the fires at Chernobyl, there has not been a statistically significant increase in cancers except for thyroid cancer. The only human organ that uses or stores iodine is the thyroid so any inhaled or ingested radioactive iodine would all be deposited in the thyroid. The lack of an association with other cancers is a significant finding.

Radiation effects on the embryo and fetus

Radiation to the developing embryo and fetus can cause multiple malformations. The dose, time of irradiation during gestation, and dose rate all have effects. In 1929–30s, medical professionals reported a relationship in the use of x-rays and microcephaly (a very small head with abnormal neurological function) in utero (Goldstein, 1929). The Russell's at Oak Ridge National Lab (Russell, 1965, 1966) reported on the timing of radiation during pre-implantation (0–9 days), organogenesis (10 days to 6 weeks) and the fetal period (6 weeks to full term). The types of congenital malformations are the same as observed in the general populations, however the number of malformations is greatly increased. The mental retardation and growth retardations are the main effects of low dose irradiation. A dose of 0.1 Gy (10 rad) to an embryo during gestation is considered a point at which a therapeutic abortion should be considered.

Radiation dose risk with mammography, X-ray and computed tomography

Everyone is exposed to natural background radiation on a daily basis. This radiation comes from cosmic rays, the earth's soil and rocks and even the food we eat. Radon is the largest source of background radiation that is commonly encountered. Diagnostic radiology (Bross, 1979) is the largest source of radiation exposure for almost all human populations. Dental (McKlveen, 1980) and medical X-rays account for a very small risk they pose for providing valuable clinical information. The risks are referred to as stochastic effects of radiation if that dose leads to carcinogenesis (Boice, 1977) or hereditary changes. When high dose radiation is used for the treatment of cancers, cures may be achieved. However, deterministic effects of radiation are also observed. These deterministic effects include transient erythema (skin redness), dry desquamation (skin dryness with possible cracking), moist desquamation (skin breakdown) where the return of basal skin cells cures this condition, and temporary epilation (hair loss). Late effects include skin atrophy (abnormally functioning and dying skin cells) and necrosis (dead skin cells) that may occur months or years after high radiation doses. Most patients exposed to high radiation treatment protocols are older and suffer life threatening cancers. The possibility of radiation induced cancers 10–15 years later is a real stochastic possibility (Brenner, 2000). Informed older patients willingly take the radiation risk of future cancers to enjoy many years of life now.

Cancer risk is expressed in terms of effective dose, the equivalent dose to organs multiplied by a weighting factor. The dose risk from mammography is reduced by employing mammography units with anodes of molybdenum or rhodium and filters for X-rays from molybdenum. The mammography must display a difference in the density of breast tissue. The radiation scatters in breast tissue from Compton scattering which degrades the contrast. The scatter is reduced with compression of the breast between a parallel paddle and the film screen. This process causes discomfort, but compression reduces tissue thickness and allows lower radiation

dose to be used. The fact that a mammographic exam may cause cancer is of concern. In radiation biology, it is assumed that the probability of induction of cancer in tissue with ionizing radiation is linearly proportional to the average tissue dose. A complete breast screening produces a mean glandular dose of 0.2–0.4 cGy. The risk of a fatal cancer from a mammographic exam over a lifetime of 20 mammograms is about 1 in 10,000. A Canadian study states that “it is seen that if the reduction in mortality owing to screening were 24%, 87 lives would be saved by screening in the 40–49-year age range, and 7.6 lives would be lost due to radiation-induced breast cancer, resulting in a benefit-to-risk ratio in lives of 11.4:1 (87 divided by 7.6) for our risk model” (Yaffe and Mainprize, 2011).

The computed tomography (CT) dose in a single CT slice tends to be in the range of 1 to 4 cGy at the skin and fall off in the middle of the head or chest. The whole body CT may contribute 10 cGy to the patient. CT scans involve higher effective dose due to the larger volume of tissue exposed. In small children, dose is adjusted to minimize the total effective dose. Typical values of effective dose for various medical X-rays are as follows in mSv; Body CT 10.6, cardiac angiography 7, upper GI 3, lumbar spine 2, screening mammograph 0.4, chest X-ray 0.025, dental X-ray 0.008.

Linear no threshold and hormesis theories

It is important to know the risks for serious disease due to a particular absorbed dose of ionizing radiation. Does a minimal absorbed dose exist below which there is no risk of a lasting deleterious effect? This question generates considerable debate. The linear, no threshold (LNT) theory is that the dose response curve is linear and predicts that there is no safe dose of ionizing radiation. The hormesis theory posits that low doses of radiation exposure actually induce protective systems that prevents cancers. At this time, there does not seem to be a definitive technique that can answer the question. It is important to examine the two theories in view of biology and physiology.

First, there is no debate that large doses of ionizing radiation, especially when delivered over short time periods, result in deleterious health effects. These theories deal with low dose exposure to ionizing radiation, often over many years. As noted above, the study of one million radiation workers has not detected a statistically significant relationship between low dose, low LET, radiation and cancer. For the Mound Workers in Ohio, not only was there no statistically significant relationship to cancer, but the expected total cancers was less than was expected for a matched cohort as noted above where the standardized mortality ratio (SMR) was found to be 0.86 and for lung cancer only the SMR was 0.85. So this cohort had a lower total lung and total cancer incidence than would be expected.

The concept for LNT is that a low absorbed dose can cause damage that results in significant deleterious effects. The radiation directly or through other processes such as free radicals must break DNA chemical bonds in the cell nucleus which is a stochastic process. The damage must result in a survivable mutation instead of leading to the more common apoptosis (cell death). The DNA damage must not be recognized and repaired by the several cellular systems designed to repair DNA errors. Such a sequence of events cannot be directly observed. That any absorbed dose of ionizing radiation can cause a cancer is a key concept of the LNT theory.

Hormesis contends that low dose radiation over long periods of time is of little danger and may actually be protective. This is based in cellular physiology where some systems can be induced when there is increased activity. As an example, consider alcohol dehydrogenase that is the first enzyme to begin to metabolize alcohol in hepatocytes (liver cells). When a person ingests alcohol on a regular basis, the hepatocytes will produce more alcohol dehydrogenase so that they can process more alcohol per unit time than someone who rarely or never drinks. Some studies have shown that there are inducible systems that provide radioprotection in cellular DNA repair (Dinant et al., 2007; Chatterjee and Walker, 2017) though other studies have shown that non-irradiated near neighbor cells are damaged even though the cell is not traversed by ionizing radiation. For a thorough review of radiation induced biological effects, see the paper by Shemetun and Pilinska (2019).

During all the normal activities of a cell, there will be errors and damage to DNA and other important molecules. As noted by Lodish et al. (2004), a typical human cell will address 10^4 to 10^6 such errors and damages per cell per day. Cells are very well equipped to detect and correct errors or damage to DNA. Also, some tumors cause an immune response that results in the elimination of diseased (including some cancerous cells) by phagocytes (cells that envelop and digest other cells). So for a low dose radiation to result in a cancer, it must cause damage DNA in a manner that it evades the multiple systems to detect and repair the damage. The damage must not result in apoptosis (cell death) due to a lethal change as most DNA alterations do. Then the cancerous cell must elude the immune system. The likelihood of such an event would be logically small and the lack of correlation between low absorbed dose radiation and cancer would support that logic.

Finally, it is most informative to consider an event in Taipei, Taiwan where construction of 180 buildings including schools, apartments (1700 units), and other buildings were constructed between 1982 and 1984 with recycled steel that was accidentally contaminated with ^{60}Co [$t_{1/2} = 5.3$ years] (Chen et al., 2007). The authors state:

“Approximately 10,000 people occupied these buildings and received an average radiation dose of 0.4 Sv, unknowingly, during a 9–20 year period. They did not suffer a higher incidence of cancer mortality, as the LNT theory would predict. On the contrary, the incidence of cancer deaths in this population was greatly reduced—to about 3 per cent of the incidence of spontaneous cancer death in the general Taiwan public. In addition, the incidence of congenital malformations was also reduced—to about 7% per cent of the incidence in the general public. These observations appear to be compatible with the radiation hormesis model.”

At this juncture, neither LNT nor hormesis have clear evidence to support it as the correct viewpoint. The above arguments support hormesis more than LNT but both have some logical issues and inconclusive research. It would seem that the fear and concern generated by the LNT theory is not warranted and that hormesis should be more highly considered. While it is reasonable to always strive to minimize radiation doses, it also makes sense to try to minimize the unfounded fears that many, especially the public, has of radiation.

Summary

Ionizing radiation can deposit energy into biological tissues that can damage systems that maintain homeostasis and the genetic information (DNA) that codes for all the components of the cell with the exception of mitochondria that has its own DNA. Damaged DNA can lead to carcinomas and sarcomas that are most probable in rapidly dividing tissues such as the bone marrow. There are clear relationships to cancers for large doses of ionizing radiation delivered over a short period of time such as seen in bomb survivors and victims of criticality accidents. Such exposures, though dangerous, are fortunately rare. The other extreme where people are exposed to low doses of radiation over very long periods of time have not conclusively been associated with an increased risk of cancers. So, a null hypothesis of “annual doses of low LET radiation below some low dose has no adverse effects on human health” cannot be rejected based on currently available data. Studies to demonstrate that there is not a dose below which it is safe has not been proved. At the present time, any excess cancers from low dose radiation delivered over a long period of time should have been extracted from the data so these doses are not clinically relevant risks. This effect is likely due to the ability of cells to repair errors in replicating DNA and other cellular tasks. Cells have many opportunities to make an error but are able to correctly repair most errors. Low dose radiation delivered over a long period of time provides cells time to repair damage whereas when a large dose is delivered over a microsecond there can be so much damage that the cell cannot complete all repairs accurately. Even when a cell becomes cancerous, the immune system may recognize it as a malignancy and destroy it (see, for example, [Davis and Bjorkman, 1988](#)). It is important to realize that not all cancers elicit an immune response and some can adapt to prevent immune attacks ([Ribas, 2015](#)).

The ICRP recommends limiting the absorbed dose for anyone working with radiation to the minimal amount required to carry out the job. Though a dose response for cancer has not been conclusively shown for low dose radiation received over a long period of time, the recommendation should be heeded to minimize any risk that might be present.

However, it is important to keep the risks in perspective with the benefits from the entire nuclear industry. Nuclear power is a clean source of electricity that can be safely operated for many years with no contribution to greenhouse gases that are associated with global warming. Radiation has truly revolutionized medicine with fast, safe diagnostic techniques to diagnose and treat disease. There have likewise been great advances made in therapeutic modalities using radiation. Imaging techniques allow greater national security from examining shipments entering a nation to screen for potentially dangerous items. As is gleaned in this encyclopedia, there are multitudinous ways that the nuclear industry make our lives better with from power generation to radiological imaging to radiation oncology to radiocarbon dating to improving our national security. There are risks associated with radiation and radioactive materials, but the risks of low dose radiation have not been shown to be clinically significant. Clearly, the benefits of the industry far outweigh the risks even for the workers involved in working with radiation.

It is important to continue to minimize exposures to keep doses as low as reasonably achievable that means maximizing distance, minimizing exposure time, and using shielding. We all know that the sun can cause severe burns, and can cause nausea, vomiting, myalgias, and other symptoms of radiation poisoning including death but it does not cause fear. Instead, we behave rationally and protect ourselves by limiting the time that we are exposed and using shielding (clothing, shade, or inside) though we cannot increase our distance from the sun. It is critical that we approach the entire nuclear industry in the same manner so that the marvelous diagnostic modalities and treatment modalities in medicine as well as the power to deploy them from nuclear power will continue to serve us all well.

See Also: Medicine: Radionuclides Used in Nuclear Medicine; Medicine: Radiopharmaceuticals and Their Use in Nuclear Medicine; Medicine: Therapeutic and Other Applications Using Sealed Radiation Sources.

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- The National Council on Radiation Protection and Measurement (NCRP) issues, reports, and recommendations (especially 103) that may be found on the internet with published copies available from NCRP Publications, Washington DC 20008.
- The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) issues reports and recommendations concerning radiation that can be obtained from UN headquarters in New York.

Estimation of Health Risks From Exposure to Ionizing Radiation

A. Iulian Apostoaei, Oak Ridge Center for Risk Analysis, Inc. (ORRISK, Inc.), Oak Ridge, TN, United States; and Nuclear Engineering Department, University of Tennessee, Knoxville, TN, United States

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Acronyms

BEIR NAS' Biological Effects of Ionizing Radiation committee
CDC U.S. Centers for Disease Control and Prevention
Ci Curie; non-SI unit of radioactivity representing 3.7×10^{10} disintegrations per second ($3.7 \times 10^{10} \text{ s}^{-1}$)
CT Computed Tomography
DDREF Dose and Dose Rate Effectiveness Factor
DOE U.S. Department of Energy
EAR Excess Absolute Risk
EEOICPA Energy Employees Occupational Illness Compensation Program Act
ERR Excess Relative Risk
Gy Gray; SI unit of absorbed dose from radiation; 1 Gy represents an energy of 1 J deposited by ionizing radiation in a mass of 1 kg.
ICRP International Commission on Radiological Protection
INWORKS International Nuclear Workers Study
IREP Interactive Radio-Epidemiological Program
LET Linear Energy Transfer
LSS Life Span Study of the Japanese atomic bomb survivors
mGy milli-Gray, one thousandth of a Gray (Gy)
mSv milli-Sievert, one thousandth of a Sievert (Sv)
NAS U.S. National Academies of Sciences
NCI U.S. National Cancer Institute
NIH U.S. National Institutes of Health
NCRP U.S. National Council on Radiation Protection and Measurements
NIOSH U.S. National Institute of Occupational Safety and Health
NRC U.S. National Research Council
pCi pico-Curie; 10^{-12} Ci (10^{-12} Ci)
SI International System of Units
Sv Sievert; SI-derived unit of equivalent dose defined as the absorbed dose (Gy) multiplied by a radiation weighting factor which accounts for biological effectiveness of different types of radiation.
UK United Kingdom

U.S. United States of America

UNSCEAR United Nations Scientific Committee on the Effects of Atomic Radiation

WHO World Health Organization

Introduction

Ionizing radiation has many beneficial applications, including uses in energy production, medicine, industry, security, agriculture and research (Waltar, 2021; Hor-Lacy, 2021). Electricity produced by nuclear reactors covers a large fraction of the energy needs of the world (e.g., 20% in the U.S). Medical procedures using radiation have saved countless lives through detection and treatment of medical conditions. Irradiation of foods or radiation sterilization of medical equipment is used to kill germs without affecting the substance that is being disinfected. Large societal benefits also come from the use of tritium in reflective signs, ^{241}Am in smoke detectors, or x-ray spectroscopy in determining the structure of substances, such as antibiotics.

Ionizing radiation has the potential of inducing health hazards if not properly used or contained. At moderate and low doses of radiation, health effects that occur from exposure to nuclear radiation are stochastic in nature. That is, the probability of occurrence, not the severity of disease, depends on radiation dose. Stochastic health effects may occur many years after exposure. Estimates of risk of stochastic health effects, defined as the probability of disease in a specified time interval, are crucial for understanding and controlling the health impact of radiation exposure.

Malignancies (solid and hematopoietic cancers) are the most important stochastic health effects of exposure to radiation. Emerging studies indicate that some non-malignant diseases, such as cardio-vascular diseases, may be regarded as stochastic, but data on possible associations with radiation are only now being collected. Also, there is no conclusive evidence of major hereditary effects in humans. Thus, this chapter focuses on available data and on methods for estimating the risk of cancer as a stochastic health effect.

Estimation of risks of stochastic health effects

Evaluations of health risks in a population exposed to radiation can be performed based on (1) radio-epidemiological studies, which follow up a population after exposure and observe the occurrence of any increases in late health effects, or (2) assessment methods to predict risk using on models derived from previous epidemiological studies.

Radio-epidemiology is a scientific discipline that uses direct observations to determine changes in rates of disease in human populations and their association with radiation exposure. *Risk assessments* rely on calculations using mathematical risk models. Both exposed individuals and decision makers are interested in the possible health outcomes of an exposure immediately following an exposure event. Epidemiological follow-up of a population takes many years, perhaps the entire lifetime of the exposed individuals, while assessments can be carried out rapidly to quantify expected risks. Assessments are also especially important when exposed populations are small, and epidemiological detection of an effect is hindered by statistical limitations.

Health effects in a population are quantified as incidence (or mortality) rates, defined as the total number of new onset disease events (or total number of deaths from a disease) during a given period of time, divided by the total number of persons. Common measures of risk are the *absolute risk*, representing the probability of occurrence of a health event in a population, and the *relative risk*, representing the incidence or mortality rates of disease in an exposed group relative to incidence or mortality rates of disease in a non-exposed group. An *excess absolute risk* (EAR) is the rate of disease in an exposed population minus the rate of disease in an unexposed population. An *excess relative risk* (ERR) is the rate of disease in an exposed population divided by the rate of disease in an unexposed population, minus 1.0. The EAR and ERR quantities are related mathematically through the rate of disease in an unexposed population, which represents the *baseline risk* (B). The name “excess” is used to indicate that EAR and ERR represent the portion of the total risk related to radiation exposure only. In an epidemiological study, risks are associated with the magnitude of exposure by means of a dose-response relationship that relates an excess risk to radiation dose and to other variables such as age or sex. In risk assessments, the total or excess risks in an exposed population of interest are estimated using dose responses obtained from epidemiology.

Radio-epidemiological studies

Epidemiological studies of incidence or mortality of cancer can be classified as *occupational*, *medical* or *environmental*, according to the type of exposed population. Table 1 summarizes epidemiological studies by type of exposed population. *Occupational* studies focus on individuals exposed in their workplace, and include nuclear workers, uranium and other miners, and radiologic technologists who work in the medical field. *Medical* studies focus on patients exposed to radiation during diagnostic or treatment procedures. *Environmental* studies include members of the public exposed to natural sources of radiation (including residential radon), releases from nuclear industrial facilities, fallout from nuclear weapons testing, and direct radiation from nuclear detonations, as in the case of the Japanese survivors of the atomic bombings in Hiroshima and Nagasaki, Japan. In addition, epidemiological studies

Table 1 Categories of epidemiological studies.

<i>Epidemiological studies by exposed population</i>		
<i>Occupational</i>	<i>Medical^a</i>	<i>Environmental</i>
Nuclear workers from Australia, Belgium, Canada, China, Finland, France, Germany, Hungary, Japan, Korea, Lithuania, Russia, Slovakia, Spain, Switzerland, UK, USA	General Imaging Diagnostic imaging In-Utero imaging Computed tomography (CT) imaging	Atomic bomb survivors Hiroshima, Nagasaki
Category of workers Radium dial painters Medical radiologists Uranium & other miners Uranium processing Plutonium processing Radiochemical plant Research reactors Nuclear power plant Chernobyl cleanup Flight crew	Benign disease Spondylitis (x-rays) Gynecological disorders Tinea Capitis Tonsils Thymus Skin Hemangioma Peptic ulcers Tuberculosis	Natural radiation Yangjiang, China Kerala, India Childhood Cancer, UK
	Malignant Disease Cervical cancer Breast cancer Lung cancer Hodgkin's Lymphoma Non-Hodgkin's Lymphoma Uterine cancer Testicular cancer Childhood cancer	Nuclear Facilities Techa River, Russia Hanford, WA, USA Chernobyl, Ukraine
	Systemic Diagnostic (¹³¹ I) Hyperthyroidism (¹³¹ I) Thyroid cancer (¹³¹ I) Thorostrast (²³² Th; ThO ₂) Bone cancer (²²⁴ Ra) Ankylosis Spondylitis (²²⁴ Ra)	Nuclear weapons fallout Nevada Test Site, USA Kazakhstan Test Site
		Residential Radon Austria, China, Czech Republic, Finland, France, Germany, Italy, Spain, Sweden, UK, USA

^aStudies are organized by medical purpose: general imaging, and diagnosis or treatment of benign or malignant disease based on external or internal (systemic) exposures.

may carry out analyses on combined (or pooled) data from multiple populations (e.g., multiple worker cohorts, multiple medical cohorts, combinations of worker or medical cohorts, including or excluding the cohort of Japanese atomic bomb survivors).

Radiation epidemiological studies are periodically evaluated by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), International Commission on Radiological Protection (ICRP), U.S. National Cancer Institute (NCI), U.S. National Council on Radiation Protection and Measurements (NCRP), and U.S. National Academies of Sciences' Biological Effects of Ionizing Radiation (BEIR) committee. Evaluations focus on the methodological quality of each study and are carried out with respect to (1) methods for estimating radiation doses; (2) methods of selection of study participants; (3) methods for measuring outcome; (4) sources of bias specific to study design; (5) methods to account for factors other than radiation that could affect the disease; (6) statistical methods; (7) study reporting; and (8) conflict of interest of authors.

Based on reports from these organizations:

- Risks of stochastic effects increase with increasing radiation dose.
- Risks of stochastic effects remain elevated throughout the lifespan of exposed individuals.
- For most cancers, risks of stochastic effects decrease with increasing age at time of exposure.
- Cancers frequently associated with radiation in epidemiological studies and with well-established risk estimates include *all cancers as a group, all solid tumors as a group, leukemia, thyroid, female breast, lung, colon, bladder, stomach, esophagus, brain and nervous system, and ovary.*
- Cancers occasionally detected and which have risk estimates often with negative lower limits are *kidney, liver, salivary glands, skin, rectum, uterus, prostate, bone, connective tissue, Non-Hodgkin's lymphoma, chronic lymphocytic leukemia, skin melanoma, and cancers of the pancreas, testis, and cervix.*

Recent radio-epidemiological studies of cancer are discussed below, while risk assessment methods and examples are discussed in the second part of this chapter.

Environmental exposures

Japanese atomic bomb survivors

The most thoroughly studied cohort of individuals for estimation of health effects of ionizing radiation is the Life Span Study (LSS) cohort, which includes survivors of the atomic bombs detonated in Hiroshima and Nagasaki, Japan. The LSS cohort includes more than 100,000 persons (> 80,000 exposed) which have been followed-up longer than 50 years, during which time more than 17,000 cancer cases and 10,000 cancer deaths were observed. Analyses of data from this cohort indicate that most cancer types can be associated with radiation exposure, and provide estimates of risks that are dependent on radiation dose, sex, age at exposure, and attained age or time since exposure. The cohort average dose to colon was about 120 mGy, with 79% of survivors receiving doses less than 100 mGy.

Analyses of data for most individual solid cancers and for all solid cancers as a group are available for cancer incidence (follow-up from 1958 to 1998; Preston et al., 2007) and cancer mortality (follow-up from 1950 to 2003; Ozasa et al., 2012). For hematopoietic cancers (leukemia, multiple myeloma and lymphoma) current risk estimates are provided by analyses of data in the LSS cohort with a follow-up from 1950 to 2001 (Hsu et al., 2013). The most current analysis of cancer incidence for all solid tumors combined includes an additional 11 years (follow-up from 1958 to 2009; Grant et al., 2017), with analyses of individual cancer sites being carried out at present time but not yet completed.

The LSS cohort represents a large population of both sexes and all ages which experienced a single acute exposure. This cohort has good dosimetry, a long-follow-up, large numbers of cases and good information on other factors affecting disease (e.g., smoking). Its possible weaknesses are a healthy survivor effect, referring to study subjects having to survive post-World War II living conditions, and the effect of better medical care than the general population in Japan received by study subjects in later years.

Fig. 1 compares estimates of sex-averaged excess relative risks (ERR) of solid cancer incidence and mortality at 1 Gy from analyses of LSS cohort data with follow-up periods of 1958–1998, 1958–2003, and 1958–2009, with ERRs obtained from selected

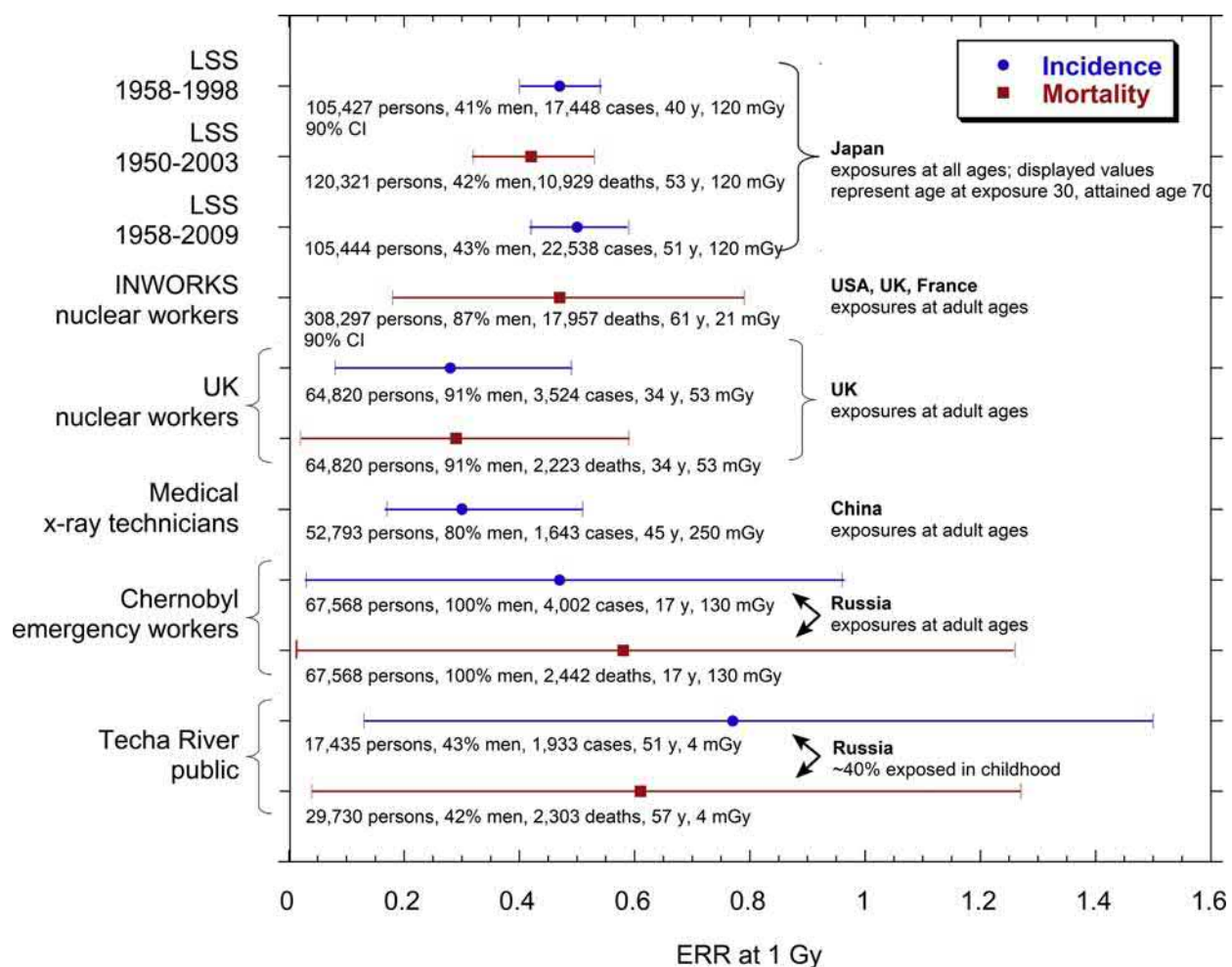


Fig. 1 Estimates of Excess Relative Risk (ERR) at 1 Gy for all solid tumors combined (Uncertainty ranges represent 95% confidence intervals, unless noted otherwise. Listed numbers of persons, percent men, numbers of cases or deaths and average doses are for entire cohort. The number of years represents the total follow-up duration.)

cohorts of workers and members of the public (discussed below). The ERRs at 1 Gy are consistent among studies, with central values ranging from 0.3 to 0.8 for cancer incidence and 0.3 to 0.6 for cancer mortality. The uncertainty ranges in estimated ERR at 1 Gy from studies of workers and members of the public are significantly larger than uncertainties in estimates from the LSS cohort, because of lower numbers of people and observed cases, shorter follow-up or lower doses.

Natural background of radiation

Epidemiological studies investigating associations between cancer and natural background radiation are categorized as studies of exposures to radon in homes and mines, and studies of exposures to gamma radiation.

Radon—Average levels of radon are about 0.4 pCi L^{-1} outdoors and 1.3 pCi L^{-1} indoors, with certain areas having indoor levels higher than 4 pCi L^{-1} and reaching 20 pCi L^{-1} or more. Lung cancer has been convincingly linked to residential radon exposures in epidemiological studies with reliable dose reconstruction, especially from analyses of data pooled from more than 10 cohorts from various countries. The risk estimates from these pooled analyses of residential exposures are compatible with the high-dose studies of radon-exposed underground miners.

High level natural radiation areas—Average dose rates of gamma radiation from terrestrial sources of naturally occurring radionuclides are normally less than 1 mGy y^{-1} . Areas of much higher natural background radiation can be found in Brazil, Iran, India, and China, where external gamma doses can vary from several times up to a hundred times normal levels, often exceeding 10 mGy y^{-1} . Radiation-related risks of cancer among residents of such areas were studied in populations living in Yangjiang, China (average of 2.1 mGy y^{-1}), and Kerala, India (levels varying from 1 to 45 mGy y^{-1}). Studies of these two populations did not reveal an increase in cancer associated with external gamma radiation, with negative ERRs per Gy reported for all solid cancer mortality in Yangjiang and incidence in Kerala.

Other studies of exposure to background radiation—A study of natural background radiation and childhood cancer was carried out in the United Kingdom using estimates of radiation exposures based on mother's residence at the child's birth from national databases. This study found a significant dose-response for leukemia in relation to red bone marrow dose from gamma radiation (mean dose 4 mSv ; range $0\text{--}31 \text{ mSv}$). Their estimate was consistent with the dose-response relationship from the LSS. There was a weak but not statistically significant relationship between radon exposure and leukemia, and no association between gamma or radon dose with any other childhood cancers. The UK study is currently being expanded and dose estimates refined.

Releases from nuclear facilities

Populations living near nuclear power plants—Several studies have been carried out worldwide (e.g., U.S., Spain, France, South Korea, Belgium, Canada, Slovenia, Slovakia, Ukraine; [Jabon et al., 1991](#); [Kim et al., 2016](#); [Desbiolles et al., 2017](#)) to detect increases in incidence or mortality cancer rates in populations living within 20 or 30 km from nuclear power plants. These studies focused on rather current day populations residing near relatively modern facilities which operated under normal conditions. In general, these studies reported either not significant or inconsistent increases in cancer rates, suggesting that health effects are non-existent or too small to be detected.

Chernobyl accident—The most important source of information on risks in the general population after the Chernobyl accident in 1986 comes from epidemiological studies of thyroid cancer in children internally exposed to ^{131}I , primarily from consumption of contaminated milk. Numerous scientific papers have been published and follow-up studies are still ongoing for cohorts from contaminated areas in Ukraine, Belarus, and Russia. Findings indicate that children and adolescents have elevated rates of thyroid cancer that can be associated to their thyroid dose from radioactive iodine, with higher risks at younger ages at exposure and with a suggestion that deficiency in stable iodine may increase the risk. Evidence on thyroid cancer risks in adults are less clear, as are increases in incidence and mortality from non-thyroid cancers and non-cancer diseases. Most recent studies come from a cohort of Ukrainian children with ^{131}I exposure estimated from measurements and interviews, who received several rounds of medical screenings for thyroid cancer. The linear dose-response for thyroid cancer is consistent with the risk from external acute exposure in the LSS. Similar results were reported in children from Belarus. Assessment of the effect of uncertainty in dose estimates did not significantly alter these risk estimates.

Techa river—The Mayak Production Association industrial complex, in Russia, was dedicated to the production of plutonium for the Soviet nuclear weapons program. In the early years of operation, from 1949 to 1956, large amounts of radionuclides were released into the Techa River, creating significant exposures to a population of about 30,000 residents of 41 rural villages along 250 km of this river. These residents are members of two cohorts, one for cancer mortality ([Schonfeld et al., 2013](#)) and one for cancer incidence ([Davis et al., 2015](#)), both including males and females of all ages. Members of the cohorts were exposed externally to gamma radiation from river sediments and flood-plain soil, and internally through ingestion of water and food products contaminated primarily with ^{89}Sr , ^{90}Sr , and ^{137}Cs , but also with ^{144}Ce , ^{95}Zr , and ^{95}Nb . Estimated cumulative doses to the stomach varied from 0 to 960 mGy , with about 90% of individuals receiving less than 100 mGy . Analyses of both incidence and mortality of solid cancers as group indicated statistically significant linear trends. The ERR per unit dose, shown in [Fig. 1](#), are compatible with those from other studies. An update of dosimetry including quantification of uncertainties in doses is ongoing, and revised dose-response analyses are expected.

Occupational exposures

Numerous cohorts of workers exposed to radiation in the nuclear, medical, and aviation industries have been established and followed up in recent decades. A common finding of studies of these cohorts is that, when compared with the general public, a working population has less chronic disease, longer survival, and may have less cancer and cardiovascular disease, depending on the many competing causes of cancer experienced by that population. Overall, a healthier working population would have lower all-cause incidence or mortality rates than the general population, with disease occurrence shifted to later in life. This healthy worker effect is measured by standardized incidence or mortality ratios defined as ratios of observed to expected illness-related cases or deaths in an occupational cohort, where the number of expected cases or deaths is determined from national rates adjusted to reflect the age and sex composition the cohort of workers. Selected epidemiological studies of nuclear, medical, and other radiation workers are described below.

Nuclear workers

The International Nuclear Workers Study (INWORKS; Richardson et al., 2015, 2018) provides an analysis of pooled mortality data of occupationally-exposed nuclear workers in UK, U.S., and France. The dose-response analyses indicate statistically-significant associations of mortality from leukemia and from all solid cancers combined with increasing dose. The ERR per Gy for all solid cancers, as shown in Fig. 1, is comparable to estimates from LSS and other worker studies. The average cumulative dose to the colon was 21 mGy among workers with a positive recorded dose and the distribution of doses was skewed (90th percentile, 53 mGy; maximum, 1300 mGy). Statistically significant and consistent estimates of ERR per Gy were observed when the cohort was incrementally restricted to workers having doses less than 200, 150 or 100 mGy, which indicates that the dose response is approximately linear.

INWORKS is based on a large cohort, with long follow-up and good dosimetry for external doses. However, workers had unaccounted or missed neutron and internal emitter doses which could potentially affect reported risks. Estimates of ERR per unit dose represent risks in adult male workers exposed to low-LET radiation, and they cannot be directly used to assess risks in cases of childhood exposures and are less relevant for estimating risks in adult women. Site-specific cancer mortality analyses in INWORKS have been carried out, but, except for leukemia, they produce imprecise risk estimates at this time. More reliable site-specific analyses are expected as the follow-up is extended and the cohort ages.

Medical workers

Stochastic health effects in medical radiation workers, such as carcinomas of the skin or leukemia have been reported shortly after x-ray diagnostic techniques were first used. With the development of radiation dose measurements, film badge monitoring, and personal and general radiation protection equipment (lead aprons, machine and building shielding), medical occupational doses decreased significantly, leading to a decline in observed leukemia, skin cancer, and female breast cancer in medical radiation workers. Radiologists, radiological technicians, and other medical workers with historical and modern day exposures are currently part of several epidemiological studies.

US Radiologic technologists—Radiation doses have been recently reconstructed for a cohort of US radiologic technologists of about 110,000 (mostly female) workers. Epidemiological studies to date found a significant dose-response relationship for breast cancer incidence in workers who started work before 1950, and elevated risks for all cancer mortality and squamous cell carcinoma incidence. However, no dose-response relationship for basal cell skin cancer was found in these workers, although skin doses were greater than 1 Gy in some individuals. Detailed dose-response analyses for individual cancer sites are ongoing, but statistical power is generally limited for sites other than breast cancer, due to low mean doses and relatively small number of cases.

Chinese medical X-ray workers—A study of solid cancer incidence in Chinese medical X-ray workers included individuals (80% males) who were employed between 1950 and 1980 at major hospitals in 24 provinces of China (Sun et al., 2016). Another group of physicians, who did not use x-ray equipment, was established as a control. The mean reconstructed cumulative badge dose was 250 mGy. A dose response was observed with a statistically-significant ERR per Gy at an attained age of 50 (Fig. 1).

Other studies of occupational exposures

Chernobyl emergency workers—Emergency workers from the Russian Federation who responded after the 1986 Chernobyl accident have been followed up from 1992 to 2009, to assess risks of solid cancer incidence and mortality (Kashcheev et al., 2015). Many of the cancers were diagnosed as a result of mandatory annual health examinations that started in 2003. The majority of doses (85%) were measured using individual dosimeters, while the remaining doses were estimated by different methods. Cumulative absorbed doses from external exposure to gamma radiation ranged from 0.1 to 1240 mGy (mean 132 mGy), with 46% of individuals receiving doses less than 100 mGy. Statistically significant linear dose responses were observed for both incidence and mortality, with estimates of ERR that are more uncertain but comparable with other studies (Fig. 1).

UK workers—Gillies and Haylock (2014) provided an updated analysis of cancer mortality and incidence in nuclear workers formerly employed by British Nuclear Fuels in the United Kingdom. The cumulative external gamma doses were less than 600 mSv with a cohort mean of 53 mSv. Statistically significant ERRs per unit dose, shown in Fig. 1, were found for both incidence and mortality. Estimates of ERRs for smoking-related and for non-smoking-related cancers were similar to the ERRs for all solid cancers combined, but with slightly larger uncertainty ranges.

Mayak workers—Workers at the Mayak Production Association experienced external exposures to gamma radiation, internal exposures due mostly to inhalation of plutonium and other alpha emitters, or a combination of both external and internal exposures. Lung, liver and bone are particularly affected by internal exposures to plutonium and other alpha emitters, and they have been studied separately. Studies of incidence and mortality of solid cancers other than lung, liver and bone (“non-LLB cancers”) focused on associations with external exposure to low-LET radiation. Workers received gamma doses as high as several Gy, with averages of 510 mGy in the incidence cohort and 350 mGy in the mortality cohort. Linear dose responses, with no evidence of non-linearity, were observed for non-LBB cancers for both incidence and mortality. Lower estimates were obtained when attempting to adjust for internal exposures to plutonium. The Mayak workers dosimetry system is being updated and revised dose–response analyses are expected.

Other worker studies—Many other cohorts of occupationally-exposed workers are being followed up in epidemiological studies. Compared with the studies discussed above, these cohorts are smaller and were followed-up for shorter periods of time. The following list describes such cohorts, for which ERR per unit low-LET dose for all solid tumors or all cancers excluding leukemia were estimated: (1) mortality in German uranium millers, (2) incidence among nuclear power plant workers in Korea, (3) mortality among nuclear power plant workers in Japan, (4) mortality in Canadian uranium processing workers, and (5) mortality in U.S. Rocketdyne workers. In all these studies, the estimates of the ERR were not statistically significant. Because these studies are rather small (i.e. small number of people and of cases) and have short follow-ups, they have low power to detect radiogenic risk. Results from these cohorts are expected to be more informative as workers age.

To increase statistical power, smaller worker studies can be pooled to create larger cohorts. A new, large-scale pooled study is now being developed with the aim of evaluating risk of cancer mortality from protracted doses less than 100 mGy by combining various groups of US workers, including nuclear weapons workers, nuclear power plant workers, industrial radiographers, medical workers, and “atomic veterans” (i.e., military personnel who witnessed atomic test blasts). This Million Worker Study is led by the NCRP (NCRP, 2018a; Boice et al., 2018). Current efforts are focused on epidemiological methods for obtaining complete and accurate follow-up for each cohort (e.g., confirming who is alive, and determining the cause of death) and on reconstruction of doses and their uncertainties. Worker studies often showed a healthy worker effect indicated by cancer incidence or mortality rates in the exposed group that were lower than those in a comparable general population; thus, analyses within the exposed group are required to assess a risk per unit dose, and to determine if risk is different at low doses than at high doses.

Medical exposures

Links between medical x-rays and cancer increases in patients have been observed since the 1950s, starting with an observation of a small excess of pediatric leukemia in the offspring of mothers exposed during pregnancy. Other studies showed increased breast cancer risks in women with tuberculosis who were monitored using fluoroscopy, in women with scoliosis who were evaluated with repeated x-rays, and increased thyroid cancer risks in patients treated in childhood for various medical conditions. Medical radiation exposure to the US population has increased by about 600% since 1980, with similar trends in other developed countries, leading to potential risks of future cancers in exposed patients. A significant part of the increase in population doses is due to the increase in the number of computed tomography scans (CT). This prompted the initiation of several studies of risk of cancer from CT scans, focused on exposures in childhood.

UK CT scan study—A retrospective cohort study in the UK included approximately 180,000 patients who were first examined with CT scans between 1985 and 2002 as children and young adults (ages <22 y) and who were followed up to 2008. Positive associations with radiation dose were observed for leukemia and brain cancer. The major concern about this and other CT scan studies is the potential bias in observed dose responses when the cohort includes patients that received CT scans due to pre-existing medical conditions. In this study observed risks were reduced, but remained statistically significant when patients who had an indication of cancer at the time of the scans and patients with previous unreported cancers were eliminated.

Australian CT scan study—An Australian CT scan cohort includes approximately 680,000 children who had CT scans between 1985 and 2005 at ages <20 y, and approximately 10 million children who were not exposed. During a period of follow-up from 1985 to 2007, a total of 60,674 cancers were recorded, out of which 3150 cancers were observed in the exposed children. Overall cancer incidence up to 2007 was 24% greater in exposed than in unexposed people after accounting for age, sex, and year of birth. The observed dose response was expressed as an increase in incidence rate ratio for each additional CT scan. Current results are difficult to compare with other results because this study used the number of scans as a surrogate for organ dose. It is expected that radiation doses will be reconstructed, which should facilitate comparisons.

Other CT scan studies—The UK cohort discussed above and cohorts from Belgium, Denmark, France, Germany, Netherlands, Norway, Spain, and Sweden are expected to be combined and analyzed together in a pooled European CT study. This pooled study aims to include approximately one million children, with results due in the next few years. A Canadian cohort study that will include exposed adults is also in progress.

Other studies of medical exposures—Radiographic and other diagnostic x-ray procedures continue to be a common source of medical radiation exposure. Doses from such procedures are typically one-tenth of doses of CT scans for the equivalent body region, suggesting that cancer risks from these procedures may too low to be detected. Studies of cancers from these procedures face a number of challenges, including low statistical power and exposure misclassification, which may impede detection of a dose-response.

Radiological health risk assessment

A radiological risk assessment is the process of estimating risks of adverse health effects in a population exposed to ionizing radiation, using risk models available from epidemiological studies. Risk assessments can be carried out quickly following an exposure event to quantify expected risks, for the purpose of informing exposed individuals and decision makers. Risk assessments can be used regardless of the size of exposed population but they are particularly useful when populations are too small for epidemiological detection of an effect, with the extreme case being a single exposed individual.

One can distinguish between *retrospective assessments*, in which risks are estimated for past or existing exposure situations (e.g., Chernobyl accident, Fukushima accident¹) and *prospective assessments*, in which risks are estimated for future, potential exposure situations (e.g., astronauts traveling to and spending time on Mars).

How is a radiological health risk assessment carried out?

A risk assessment process starts with dose-response relationships obtained from completed epidemiological studies, which are used to calculate the excess risk of one or multiple selected health effects (e.g., leukemia, thyroid cancer, brain cancer, or all cancers combined) using estimated radiation doses in the exposed population of interest. A dose response is often called a “radiation risk model” and is described by mathematical equations that express risk (ERR or EAR) as a function of radiation dose, and, if possible, of sex-, age-, and time-related variables which allow for an estimation of risk customized to an exposure situation of interest. The risk assessment process also accounts for the radiation type (e.g., photons, alpha) and exposure rate (e.g., acute, chronic, fractionated) experienced by the exposed population.

Although human health risks from exposure to radiation have been extensively studied, there are still important sources of uncertainty which need to be considered and evaluated when estimating radiation risk (UNSCEAR, 2015). The sources of uncertainties can be divided into two categories: (1) uncertainties arising from analyses related to the derivation of a risk model in a selected epidemiological study, and (2) uncertainties associated with the transfer or application of the selected risk model to a different population, exposed to different radiation types and under different exposure conditions than those of the population followed-up in the epidemiological study.

Uncertainty in derivation of a risk model in an epidemiological study is introduced by (1) statistical analyses related to curve fitting of the number of health outcomes as a function of dose and other variables, (2) uncertainties in dose estimates for members of the epidemiological cohort, (3) uncertainties related to the number of health outcomes (i.e., cancer ascertainment, selection of study participants), (4) uncertainties related to other factors that affect the disease outcome (e.g., smoking, alcohol consumption, exposure to harmful chemicals, exposures to radiation from other employment or from medical procedures). These uncertainties can be minimized by evaluating and selecting a good quality epidemiological study, which includes a large cohort, large number of cases, long follow-up, good cancer ascertainment, good information about other factors affecting disease, and reliable dosimetry (UNSCEAR, 2016). Methods to address the effect of uncertainties in radiation doses assigned to members of the epidemiological cohort have been developed during recent years, with the purpose of reducing potential biases and properly quantify the magnitude of uncertainties in observed dose responses.

To date, estimates of mortality and incidence of cancer in the LSS cohort of atomic bomb survivors in Japan have been the most comprehensive source of information on radiation-induced effects. Cohorts of nuclear and other workers have been recently established and are being followed. They rely on advanced methods for cancer ascertainment, dose estimation and adjustments for other factors that influence disease. Pooling data from high quality cohorts of workers (e.g., INWORKS study) will likely create combined cohorts larger than LSS, which should allow for more reliable risk estimates at low doses. These cohorts have the potential of providing risk models representative for chronic exposures to radiation. Similarly, large cohorts of patients exposed to radiation for diagnostic or treatment purposes, which include individuals of all ages and both sexes, are being organized and followed-up in studies expected to provide information on risk from fractionated exposures.

A key issue in risk assessment exercises is how to apply risk estimates from a selected epidemiological study (e.g., LSS) to an exposed population of interest (e.g., U.S. workers). This transfer of risk between populations is primarily an issue for cancer sites that have substantially different baseline rates in the two populations, such as stomach cancer (higher in Japan than in the U.S.) or female breast cancer (lower in Japan than in the U.S.). Two approaches can be used to describe the transfer between populations: a multiplicative and an additive risk-transfer model. In a multiplicative model, the excess risk in the exposed population of interest is assumed to be proportional to the baseline risk of cancer in that population, with the ERR observed by epidemiology being applied directly as a multiplier to the baseline risk. In an additive model, the excess risk in the population of interest is assumed to be independent of the baseline risk in that population, and the EAR from epidemiology is considered representative and is applied directly. Both models are plausible on biological grounds (NIH, 2003), but they can lead to different estimates of risk when baseline risks in the two populations differ greatly (e.g., stomach, liver, female breast cancer for Japan and U.S.). The estimates of risk based on the multiplicative and on the additive risk transfer models define an uncertainty range for the true, but unknown risk, with the best estimate often considered to be an average between the two estimates. Risk transfer can be further customized by race as baseline rates of skin (and other) cancers are different in Whites, Blacks, Hispanics, Asians, or Native Americans. Exposures to

¹Information about these accidents and their impact on safety, licensing, and decommissioning are provided in the Chapter “safety, licensing, and decommissioning” of this Encyclopedia.

a second carcinogen, such as smoking or chemo-therapy agents, may affect the risk from exposure to radiation. The total risk from radiation and from another carcinogen is also calculated using multiplicative or additive interaction models between the two agents.

In order to estimate radiation risks, it is often necessary to extrapolate data obtained at high doses and high dose rates to low doses (< 100 mGy) or low dose rates (< 0.1 mGy min^{-1}), representative, for example, of exposures to populations living in contaminated areas after nuclear accidents, or of occupationally-exposed individuals. To estimate risks at low doses or dose rates, a risk calculated using a model from a population exposed at high doses and high dose rates (e.g., the LSS cohort) is often divided by a “dose and dose-rate reduction factor” (DDREF). In the 1980s, DDREF values ranging from 2 to 10 were estimated by NCRP and UNSCEAR based on data from animal experiments. In the 1990s, a DDREF equal to 2 was adopted by the ICRP and UNSCEAR, while in the 2000s, a value of 1.5 (95% CI: 0.8, 2.7) was recommended by the National Academy of Sciences’ BEIR VII report based on analyses of both animal and human data. Current analyses comparing risk estimates from many epidemiological studies showed DDREF values both lower and higher than 1.0. In its assessment of health effects after the Fukushima accident, the World Health Organization (WHO, 2013) did not use of a DDREF, thus advancing the idea that dose responses observed in LSS cohort require no further adjustments when applied to a population exposed at low dose rates.

Dose responses observed in most epidemiological studies are linear in dose, with statistical tests indicating the absence of a threshold dose below which the risk would be zero. Such dose responses have been observed mostly in studies with average cohort doses in excess of 100 mGy, with several studies indicating statistically significant responses below 100 mGy. These observations led to the assumption of a linear non-threshold (LNT) dependence on dose. However, there are a few epidemiological studies that indicate downward (concave) or upward (convex) dose responses, and various interpretations of data suggest existence of a threshold or hormetic dose responses (i.e., risks at low doses lower than in unexposed). The shape of the dose-response curve to be used in estimating risks at doses below about 100 mGy has been strongly debated for some time and some organizations advise against estimating health risks to people from exposures that are near or less than the natural background levels because of statistical uncertainties (HPS, 2020). However, an expert committee that recently reviewed epidemiological studies on low dose radiation concluded that data support the continued use of the LNT model in radiation protection (NCRP, 2018b). The use of an LNT model implies the possibility of either overestimating or underestimating the true risks at low doses. In risk assessments, this issue can be addressed by quantifying the uncertainty in estimated risks at low doses using an uncertain DDREF or using multiple alternative risk models. Decisions based on such assessments need to be made using the full uncertainty range of the resulting risk estimates.

Dose-responses from the major epidemiological studies represent exposures to low-LET radiation. Only a few epidemiological studies are available for cases of exposures to high LET radiation, and they are limited to exposures to alpha particles (e.g., from inhalation of plutonium or uranium or their decay products) and to a small number of cancer sites (e.g., lung, liver, bone). In practice, it is often necessary to evaluate the radiological impact of a large variety of high-LET particles including alpha particles, protons, neutrons of various energies, or heavy ions. Evidence from limited human, and somewhat more abundant animal studies indicates that high-LET radiation is more effective than low-LET radiation in inducing cancer. In the system of radiological protection, absorbed doses from high-LET radiation are multiplied by quality factors or radiation weighting factors,² which are intended to represent the enhanced biological effectiveness of each type of radiation relative to low-LET radiation. Similarly, estimation of risks in cases exposures to high-LET radiation is carried out using multiplicative factors, with the difference that such factors are customized for the risk assessment of interest (e.g., astronauts exposed to heavy ions in space) and account for differences in effectiveness of inducing leukemia and solid cancers. These factors are based mostly on animal data and include estimates of uncertainty in the possible effectiveness of each radiation type (Kocher et al., 2005; Cucinotta et al., 2013).

Estimated risks of radiation-induced cancer obtained from epidemiologic studies are used in establishing standards to control radiation exposures of workers and the public. However, cancer risk assessments generally *inform* the development of standards by advisory groups and national regulatory authorities, usually they are not *definitive* in regard to establishing limits on acceptable exposures. That is, as described below, standards to control radiation exposures usually are not related solely or directly to limits on acceptable cancer risks.

- With the exception of a standard in the U.S. to protect astronauts (described later), standards developed by advisory groups and national authorities do not specify limits on acceptable cancer risk associated with radiation exposure. Rather, most standards specify limits on quantities other than cancer risk, such as effective dose, organ dose, or concentrations of radionuclides in a source of exposure (e.g., drinking water).³
- The effective dose, which is widely used in specifying limits on acceptable exposures, is not always a valid indicator of cancer risks. In cases of highly non-uniform exposures of the whole body, which can result from intakes of some radionuclides and are common in exposures of medical patients, estimates of cancer risks at the same effective dose can differ by as much as an order of magnitude. Consequently, ICRP has cautioned that estimates of effective dose should not be used to estimate risks from actual

²Most recent radiation weighting factors are recommended by the ICRP Publication 103 (ICRP 2007) and have values equal to 1.0 for photons, electrons and muons, 2.0 for protons and charged pions, 20 for alpha particles, fission fragments and heavy ions, and 2 to 20 for neutrons of various energies.

³Standards for cleanup of radioactively contaminated sites established by the U.S. Environmental Protection Agency specify a *goal* of limiting the lifetime cancer risk to members of the public to no more than about 10^{-4} (one chance in 10,000 of a radiation-induced cancer). However, the risk goal is not a limit, and estimated risks well above the goal are allowed based on such considerations as technical feasibility and cost. Meeting the risk goal generally has not been feasible at radioactively contaminated sites.

or anticipated exposures. Limits on organ dose or concentrations of radionuclides in a source of exposure also may not be valid indicators of cancer risks.

Risk assessments are an important addition to radiation protection quantities, such as effective dose, especially in considering approaches to risk management and risk communication. For example, given the trauma and possible injuries or deaths associated with evacuation of affected populations and concerns about whether it would be safe for those populations to return to their homes, it can be important to make decisions about evacuation and return based on reasonably rigorous assessments of cancer risks.

Example radiological risk assessment exercises and estimation tools

Assessments of radiological risk are often used directly to assess existing or potential situations involving exposure to radiation. Collections of risk models and methods to estimate risk are available from various national and international organizations, including UNSCEAR, ICRP, NCRP or U.S. National Academy of Sciences (NAS). This section presents selected examples of risk assessment methods and risk assessment applications.

General risk assessment tools—In the U.S., the NAS develops up-to-date and comprehensive risk estimates for cancer and other health effects from exposure to ionizing radiation, through the activities of the Biological Effects of Ionizing Radiation (BEIR) committees of the National Research Council. The BEIR VII report (NRC, 2006) is the seventh in a series of publications concerning radiation health effects, and it is among the first reports of its kind to include detailed estimates for cancer incidence in addition to cancer mortality. In general, BEIR VII supports previously reported risk estimates for cancer and leukemia, but it is based on new and more extensive epidemiological data from studies of Japanese survivors of the atomic bombs, including updated radiation doses to the survivors based on the 2002 LSS dosimetry system (DS02).

On average, assuming a sex and age distribution similar to that of the U.S. population, the BEIR VII lifetime risk models predict that approximately one individual in 100 persons would be expected to develop cancer (solid cancers⁴ or leukemia) from a dose of low LET radiation of 100 mGy. By comparison, approximately 42 of the 100 individuals would be expected to develop solid cancers or leukemia from other causes (Fig. 2). Lower doses would produce proportionally lower risks. Thus, it is expected that approximately one individual in 1000 would develop cancer from an exposure to 10 mGy. The excess lifetime risk of cancer mortality is predicted to be about one half of the risk of cancer incidence.

The BEIR VII committee developed ERR and EAR risk models for 11 cancer types (stomach, colon, liver, lung, breast, prostate, uterus, ovary, bladder, thyroid and leukemia) plus a remainder grouping of cancers, based on the cancer cases in the LSS cohort. Using these models, lifetime risk and their uncertainties in either males or females can be estimated for one or multiple cancer types.

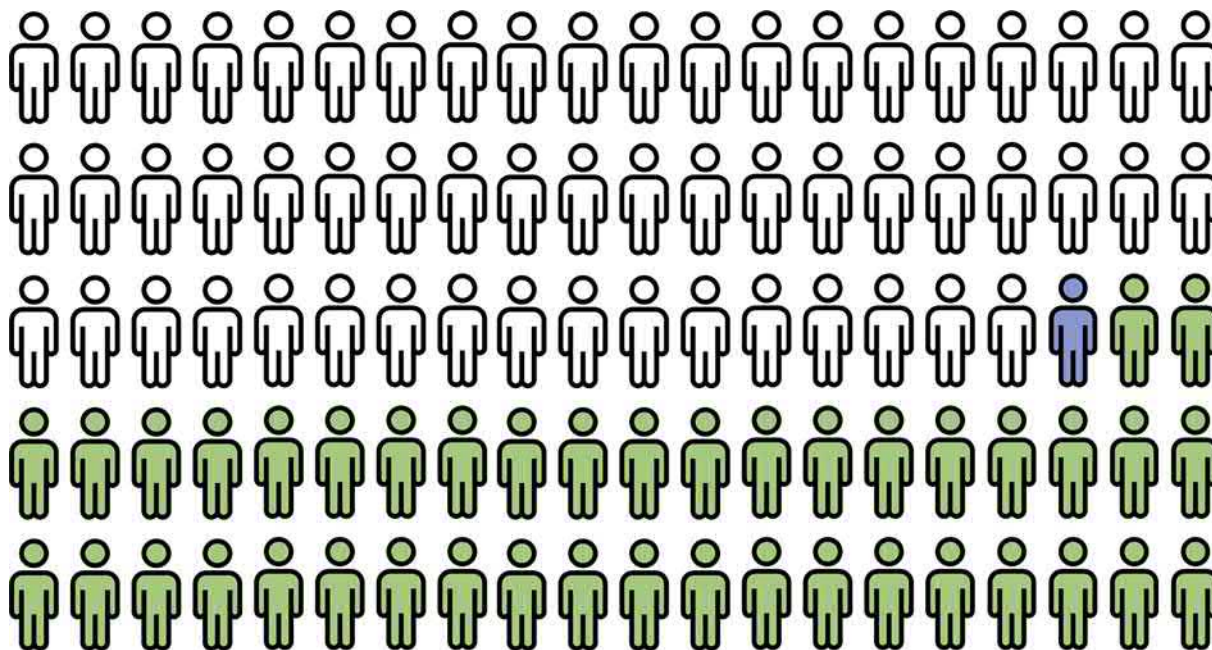


Fig. 2 Approximately 42 of 100 people will be diagnosed with cancer over their lifetime. Approximately one additional cancer in 100 people may result from a single exposure to 100 mGy of low-LET radiation (NRC, 2006).

⁴Excluding non-melanoma skin cancers

The largest lifetime site-specific risks in males are predicted for colon and lung cancers, followed by leukemia and bladder cancer. In females, the largest risks are estimated for breast and lung followed by colon, bladder and leukemia.

The U.S. National Cancer Institute (NCI) provided an extension of the analyses of radiogenic cancer incidence in the LSS cohort carried out by the BEIR VII committee, to include risk models for seven additional solid cancer sites (oral, esophagus, gallbladder, pancreas, rectum, kidney and brain), using the same epidemiological dataset and the same basic model formulation for the risk of solid cancer (Berrington de Gonzalez et al., 2012). The full set of risk models has been incorporated into an interactive web-based radiation risk assessment tool (RadRAT) which can be used to estimate the lifetime risk of radiation-related cancer with uncertainty intervals following a user-specified exposure history (<https://irep.nci.nih.gov/radtrat>). The calculator can be used to estimate lifetime cancer risk from single or multiple exposures with either non-uniform or uniform whole-body doses that are acute or chronic. It is most appropriate for low-LET radiation doses less than 1 Gy, and for individuals with life-expectancy and cancer rates similar to the general population in the U.S., although it can be adapted for other populations, by accounting for differences in baseline risks.

Compensation of radiation workers—An important application of radiological risk assessments is to provide the scientific basis for adjudication of claims for compensation for cancers that could have been caused by exposure to ionizing radiation. In 1985, the U.S. NIH developed specialized Radioepidemiological Tables to estimate a probability of cancer causation, defined as the likelihood that persons who have a radiation-related cancer and who received specific doses prior to the onset of such disease developed cancer as a result of these doses. The 1985 NIH Tables, which contained look-up tables of risk coefficients and appropriate equations, have been used by the Department of Veterans Affairs in a program to compensate military personnel who participated in the U.S. nuclear weapons testing program and later developed cancer.

The U.S. Energy Employees Occupational Illness Compensation Program Act (EEOICPA) of 2000 specified that the 1985 NIH Tables be updated and mandated compensation and medical benefits to eligible nuclear and energy workers (or certain survivors if the worker is deceased) for illnesses that could be related to exposure to radiation or other toxic substances while employed at U.S. DOE facilities. The process of updating the 1985 NIH Tables has led to the development an Interactive RadioEpidemiological Program (IREP), which is a Web-based computer code to estimate the probability that a given cancer in an individual was induced by past exposures to ionizing radiation. Accordingly, the National Institute for Occupational Safety and Health (NIOSH) adopted IREP for use in adjudicating claims under EEOICPA (U.S. DHHS, 2002). IREP can be accessed through NIOSH web-site (www.cdc.gov/niosh/ocas) or directly at www.niosh-irep.com/irep_niosh.

IREP calculates the probability of causation (PC) for a given cancer type as the risk from radiation divided by the total risk, defined as the risk from radiation plus the baseline risk in the absence of exposure. The probability of causation is expressed as a percentage between 0% and 100%. A probability of causation of 50% or greater means it is more likely than not that the cancer was induced by the past exposure to radiation. This occurs when estimated ERRs are greater or equal to 1.0 meaning that the total risk of cancer becomes double the baseline risk, due to radiation exposure. PCs are calculated based on the risk of cancer incidence estimated according to the age at time of exposure and the sex of the exposed person. IREP includes risk models for 33 cancer groupings and can accommodate estimation of PC for more than 90 individual cancer types, while accounting for up to 1000 prior distinct exposures to either electrons, photons, alpha particles, or neutrons of different energies. A defining characteristic of IREP is that it accounts for uncertainties in estimating cancer risks due to exposure to ionizing radiation and in evaluating causation of a given cancer in an individual. Part B of EEOICPA and its implementing regulations (U.S. DHHS, 2002) specify that eligible claims for compensation for cancer shall be granted when the upper 99% credibility limit of an uncertain PC is at least 50%.

Radiological risk assessment for astronauts—Another important application of radiological risk assessments is control of radiation exposures of astronauts, who are exposed to various sources of ionizing radiation throughout their career. Low-Earth-orbit missions aboard the International Space Station (ISS) and exploratory missions outside the Earth's magnetic field contribute to most of the expected increased risk of cancer in astronauts. Radiation exposures involve a mixture of low- and high-LET ions that are encountered in space. The most relevant fields are the galactic cosmic radiation, mostly energetic ions with atomic numbers 1 (H) to 28 (Ni), and secondary radiations (including neutrons) produced from nuclear interactions in shielding and the human body. In addition to cosmic sources of radiation present during space missions, occupational exposures of astronauts include low levels of atmospheric radiation during flight training on T-38 aircrafts and x-rays from medical imaging that are necessary for selection into the astronaut corps and flight certification.

NASA monitors the increased risk of cancer incidence and mortality over the lifetime of astronauts by recording their exposures and projecting the lifetime Risk of Exposure Induced Death (REID) and limiting it to less than 3% at the upper level of the 95% confidence interval. That career limit is called the Space Permissible Exposure Level. The projected increase in risk of cancer mortality limits the duration and number of missions that astronauts can fly in outer space. NASA performs both retrospective and prospective risk analyses, for past and future missions, respectively, using their specialized NASA Space Cancer Risk Models (NSCR-2012; Cucinotta et al., 2013), which were developed based on LSS data and adjusted to reflect the characteristics of the astronaut population and their exposures (non-smokers with expected survival longer than the average U.S. population who are exposed to space radiation).

Other radiological risk assessments—In early 1980, the U.S. Congress mandated scientific research to develop valid and credible assessments of exposure to ^{131}I that the American people received from radioactive fallout generated by world-wide atmospheric nuclear bomb tests, which resulted in estimates of radiation doses to the thyroid gland and estimates of lifetime risk of thyroid cancer produced by the U.S. NCI (cancer.gov/I-131; NCI, 1997; Simon et al., 2006). Radiation doses and risks of cancer from fallout of nuclear weapons tests have been estimated for individuals living near other test sites, for example the Semipalatinsk site in Kazakhstan or the Marshall Islands (Land et al., 2010).

Summary and conclusions

Ionizing radiation provides societal benefits, either through direct applications (e.g., power production, medical diagnostic or treatment) or through indirect uses (e.g., in agriculture, industry or security). Exposure to radiation is a well-studied carcinogenic risk factor in humans, compared with many other known or suspected carcinogens. Accurate estimates and proper interpretation and communication of radiological risks are important for operating a radiological protection system, and are necessary to maximize benefits from use of radiation by reducing potential adverse health effects. Estimation of radiation risks are carried out by direct observation of exposed populations in epidemiological studies or by risk assessments based on risk models derived from existing epidemiology. Such assessments are used to evaluate existing or future potential radiological exposure situations.

Numerous epidemiological studies of environmental, occupational, and medical exposures to radiation have been carried out to date. The life span study (LSS) of the survivors of atomic bomb detonations in Japan is considered the most reliable source of information on carcinogenic risks from exposure to radiation. Newer cohorts of radiation workers and of patients with radiation exposures from medical procedures are being developed, which are expected to provide information on radiological risks from chronic and fractionated exposures to radiation, with cumulated doses generally lower to those experienced by the members of the LSS cohort.

Risk assessments are powerful tools that can help in understanding the effects of radiation exposures. Proper communication of risks to interested parties is essential to avoid misinterpretations or exacerbations of preconceived and often incorrect views on effects of radiation. A common method for risk communication is a comparison of adverse health risks from radiation with baseline risks of cancer incidence or mortality, or with other personal or societal risks, for example, with risks from driving or smoking (Fig. 2).

Radiological risks estimates produced by scientists need to be weighted carefully by decision makers. For example, the decision to evacuate a populated area after a contamination with radioactivity may lead to adverse health effects and even deaths in elderly or hospitalized evacuees, to significant hardship for all evacuees, or to a social stigma for the affected area which is likely to have an economic impact. All these consequences may come as the price of avoiding a small radiation risk of cancer over the lifetime of the exposed population.

See Also: Relative Nuclear Energy Health Hazards and Popular Acceptance.

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Drones for Radiological Assessment[☆]

Alexandra C. Hackett, Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Glossary

Drone jamming Disrupting communications of a drone by interfering with the transmission of the communications signal.

Drone spoofing Remotely taking over a drone by deceiving the flight controller into using a false signal, most often a false GPS signal.

LiDAR (Light Detection And Ranging) A distance measuring method, similar to radar, using laser light reflecting off objects to measure the range between the sensor and objects.

Photogrammetry For 3D modeling and mapping, photogrammetry is a method of taking overlapping photos of an area and stitching the photos together to generate a larger model of the area being studied.

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Introduction and general information

Background

Unmanned aerial vehicles (UAVs) have been used by militaries for decades. In 1916–17, the Hewitt–Sperry Automatic Airplane became the first demonstrated unmanned aircraft. Research for the aircraft was sponsored by the US Navy, which was interested in developing aerial torpedoes. These remotely controlled and automatic aircraft became precursors to the modern-day guided missiles. The first unmanned aircrafts that resemble modern-day UAVs with reconnaissance capabilities were first developed during the Cold War. During this time, the United States, the Soviet Union, the United Kingdom, Canada, and Israel all made major advancements in reconnaissance drone technology (Valavanis and Vachtsevanos, 2015).

The significant success of military UAVs and the rapid expansion of reconnaissance missions has led to the fast advancement of UAV technologies. These technologies first pioneered by militaries are now finding their way into civilian life. UAVs have only been used for applications within the civil environment in the past decade. In the United States, the Federal Aviation Administration began approving commercial civilian UAVs in 2013. UAV use by civilians is still a relatively new concept. The basic technologies are generally well developed because of military use; however, the technologies need to be adapted to specific civilian functions. Civilian UAVs have become useful tools in a broad variety of applications, such as surveillance, search and rescue, and agriculture monitoring. Existing technologies can also be modified for nuclear safeguards and security related tasks. These powerful and versatile machines can provide additional information to nuclear workers without putting the workers at any additional health or safety risks.

Advantages of civilian unmanned systems

Unmanned vehicles have numerous advantages for nuclear safeguards and security. A primary advantage is that radiological assessment tools no longer have to be constrained by the physical limitations of a team of nuclear workers. Instruments can be attached to unmanned vehicles and be transported to areas a human would generally not be able to access. Unmanned platforms can be equipped with payloads that perform a great number of useful tasks. Between unmanned rovers and unmanned aircraft, a wide variety of platforms exist for carrying various types of payloads.

UAVs represent a large and diverse category of aircraft, and they have become useful tools for a broad variety of applications and consequently can have very different appearances and capabilities. Their weights can range from less than a kilogram to more than a tonne; their costs can range from a few hundred dollars to millions of dollars; and their flight radii can range from a few meters to thousands of kilometers. Image processing systems, mapping systems, and sample collection systems are just a few of the items that can be integrated into UAVs to provide additional capabilities.

UAV tasks are commonly characterized by four D's: dull, dirty, dangerous, and deep. A mission with one of these characteristics is better suited to be completed by an unmanned system than a manned system:

- Dull—UAVs are well suited for low-workload, low-intensity tasks, such as automated takeoff and landing. Such tasks can be simply automated, often only requiring human oversight rather than direct and continuous control.
- Dirty—UAVs are ideal for environments where contamination is known or suspected. For such tasks, humans can remain in a clean environment, while UAVs are deployed into polluted spaces.
- Dangerous—A UAV is well suited to travel to any area too dangerous for a human. Inexpensive UAVs may be used sacrificially in high-risk areas.
- Deep—Heavy, high-flying UAVs are suited for long-range tasks. UAVs can carry instruments to far away locations so humans do not have to traverse the distance (Valavanis and Vachtsevanos, 2015; Tice, 1991).

With so much diversity between aircraft design and payload options, UAVs can be custom-built for any specific aerial task. A UAV's size is the primary determinant of what tasks it is capable of performing in an environment. Larger UAVs have the power capacity and function similar to manned aircraft and are capable of travelling great distances, transporting large payloads, and gathering high volumes of data over hundreds of kilometers of airspace. However, large aircraft are limited by their high production and maintenance costs, and they are not well suited for gathering data because of maneuverability issues. They cannot be used to gather data that is easily skewed by significant vibration or draft. Also, larger UAVs cannot move discretely or explore small areas, like building interiors. Smaller UAVs are much more capable of moving safely through environments filled with people and expensive property. Small UAVs, such as the ones covered in this report, can maneuver around a nuclear facility while the pilot stays safely on the ground. UAVs also have robotic persistence and can complete repetitive tasks in an exact manner each time, reducing the variations occurring due to human fatigue and error.

Unmanned vehicles can be used to inspect nuclear sites and monitor nuclear activities. Instruments can be attached to unmanned platforms and be transported to areas nuclear workers, such as safeguards inspectors and reactor operators, would typically be unable to reach. The following sections provide an overview of UAV capabilities and challenges that are most relevant to safeguards practices. For example, small UAVs can transport lightweight detectors into crowded industrial environments, or large UAVs can transport cameras and sensors high into the air.

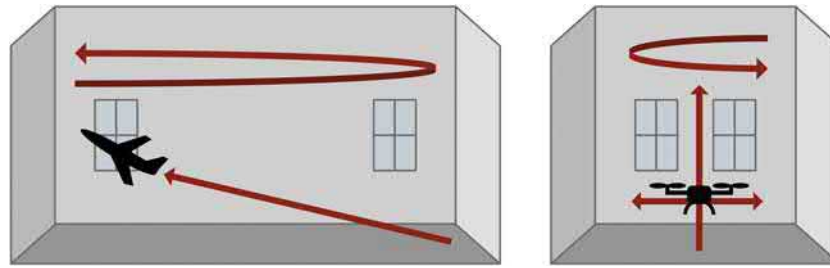


Fig. 1 Flight path visualization for fixed-wing and rotary-wing UAV.

Basic UAV design

UAV designs employ two general types of airframes: fixed-wing and rotary-wing. Fixed-wing and rotary-wing aircraft are equipped to perform different types of tasks. The efficiency and longer flight time of fixed-wing UAVs makes them well equipped for outdoor, long-haul tasks. Rotary vehicles are better suited for indoor use and short-duration tasks. **Fig. 1** shows the difficulties of indoor use of fixed-wing UAVs. The vertical takeoff of rotary-wing vehicles is much better suited for confined spaces.

The fixed-wing UAVs require a long, runway-like area for takeoff and makes wide turns to orbit a specific area, whereas rotary-wing UAVs take off vertically and can make sharp horizontal movements (Abid et al., 2014). If a rotary-wing and a fixed-wing aircraft were in rooms of equal height, the fixed-wing aircraft would need a much wider room to properly maneuver. However, if the aircraft have the same weight and energy capacity, the fixed-wing aircraft would be able to fly longer.

Rotary-wing UAVs

Rotary-wing aircraft vary greatly in appearance. The smallest UAVs are no more than an inch in diameter—these are often called micro-air vehicles. The largest rotary UAVs are similar in shape and size to manned helicopters. Rotary aircraft fly using rotary blades that spin around a single mast called a *rotor*. Rotary UAVs can have single or multiple rotors, and their names usually refer to the number of rotors onboard the aircraft. For example, a UAV with four rotors is called a *quadcopter*, and a UAV with six rotors is a *hexacopter*. Two primary advantages of rotary aircraft are obstacle avoidance and safe collision.

Rotary-wing UAVs are able to slow down and hover in a fixed position. This characteristic allows rotary-wing UAVs to approach objects as slowly as desired, which makes it ideal for aerial monitoring of stationary or slow-moving objects. Because rotary-wing aircraft are able to maintain slower speeds, they are better capable of avoiding objects, an important feature when flying close to sensitive nuclear structures and equipment. Rotary-wing UAVs can also have their propellers guarded by bumpers or fenders that prevent the propellers from coming in contact with objects located nearby. Fenders can be as simple as curved rods that extend past the radius of the propellers. These only protect the propellers from objects in a UAV's horizontal flight path; consequently, expanding the fenders to form a cage around the UAV is a more robust safety option.

Rotary-wing design also has the advantage of vertical takeoff/landing. Vertical landing capabilities are ideal for small spaces and uneven terrain. Rotary aircraft can be used to hover and spin at a fixed point to achieve 360 degree panoramic scans of the interior or exterior of a nuclear facility. They can also be used outdoors to hover and take images of rooftops or dry cask storage areas.

Fixed-wing UAVs

There are fewer moving parts with fixed-wing UAVs, making design and maintenance easier. Fixed-wing UAVs can travel greater distances than rotary-wing aircraft with the same maximum takeoff weight. Fixed wings are also less sensitive to dangerous weather conditions, and they are more energy efficient than rotary wings, allowing fixed-wing aircraft to travel greater distances and stay in the air longer. Also, it is easier to collect air samples with fixed-wing aircraft because fluctuation in air flow around the aircraft is reduced.

One of the disadvantages of fixed-wing UAVs is that they require large open areas for runway-style takeoff and recovery. A fixed-wing UAV cannot rise vertically, hover, or immediately reverse directionality the way a rotary-wing UAV can. If both a fixed-wing and a rotary-wing UAV fly over an area and see something on the ground that must be revisited, a rotary UAV can simply reverse its direction to return to the area, whereas the fixed-wing UAV must make a wide turn to circle back to the area of interest. The operator of a fixed-wing aircraft must be familiar with fixed-wing flight dynamics, so these aircraft are not as easily controlled by novice operators.

Vertical takeoff and landing UAVs

Vertical takeoff and landing UAVs are a hybrid of fixed- and rotary-wing UAVs. These aircraft are able to takeoff, land, and hover just like rotary-wing aircraft; however, they can typically fly with greater energy efficiency than standard rotary-wing aircraft because they

possess wings similar to fixed-wing aircraft, which allow them to glide. In the air, vertical takeoff and landing UAVs typically maneuver like standard fixed-wing aircraft, but they do not require large open areas for runway-style takeoff and recovery.

Internal components

An unmanned aircraft has four electronic subsystems: power, communications, sensors, and computer. Most UAVs feature a single board computer onboard, and computer hardware comes preinstalled on most commercially available UAVs (Valavanis and Vachtsevanos, 2015).

UAV power

Although smaller UAVs do not have enough space for gas-powered engines, large and mid-size UAVs have sufficient space to include a gas propulsion power system. Smaller UAVs are most often battery powered and can have either one battery system for the entire UAV or two separate battery systems—one to power the UAV and one to power the payload. Some UAV designs have a separate power supply for the payload, so the aircraft's power supply is not diminished by addition of a payload.

UAV communication

The communication link is used to transfer information such as fuel data, electrical data, location, speed, propulsion, flight control, sensors, and data from the payload. In general, the system status requirements do not dominate the link; sensors usually dominate the link. There are many design constraints when choosing the proper communication system for a UAV: maximum distance of signal, allowed frequencies, number of UAVs and other devices operating within the network (too many devices can create noise and overcrowd the bandwidth), and frequency and magnitude of control messages (the compression factor of images and the rate information is refreshed affect the bandwidth needed). The diagram in Fig. 2 shows how information from an unmanned aircraft is translated to the user.

UAV sensors

With unmanned aircraft, sensors are particularly important. UAVs do not have the convenience of an onboard pilot's situational awareness or "sight sense." Good sensors are required so that UAV pilots can properly see and react to their surroundings. Sensors are valuable for two purposes: (1) improving the ease of operation and (2) improving the pilot's situational awareness. If improving the ease of operation for the pilot is the primary goal, increasing the number of sensors can increase the UAV's autonomy. Improving the onboard capabilities will free up bandwidth for other types of communication, such as real-time video transmission. If improving the pilot's situational awareness is the primary goal, increasing the number of sensors can increase the amount of information communicated to ground control. Improving the onboard capabilities can free up bandwidth for sensors to communicate.

The main types of UAV sensors are radar, laser, infrared, electro-optical, acoustic, and motion detection. Sensor systems can be divided into two categories: passive and active. Passive sensors detect energy (such as light and heat) that is naturally available. Active sensors provide the energy source that they detect when reflected by objects in the environment.

Technology review

UAVs are ideally suited to tasks that may enhance nuclear safeguards and security practices. These UAV tasks include monitoring structural changes of facilities by detecting changes in three-dimensional (3D) scans of interiors; searching for possible contamination inside a nuclear facility with radiological mapping; taking aerial images of rooftops and dry cask storage areas; creating 3D maps of mines and facility exteriors; and collecting environmental samples in dangerous or difficult-to-access places.

Imaging of nuclear facilities

In nuclear safeguards, design information verification involves the comparison of visual photographs. Two-dimensional images from previous inspections are compared to current inspection images, but some aspects of a facility may be difficult for an inspector

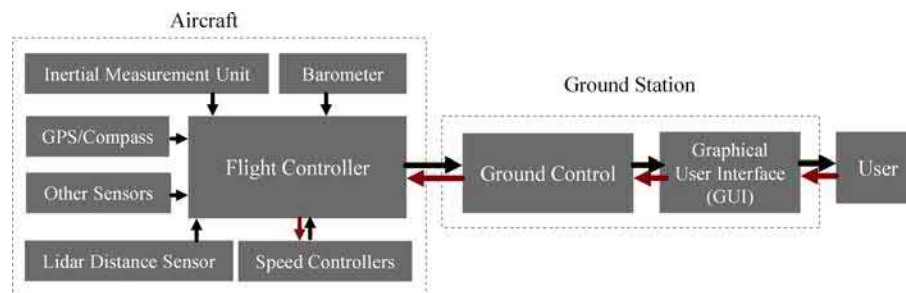


Fig. 2 Diagram of typical components found within the UAV communications structure.

to properly photograph. By sending UAVs into high-up, very tight or otherwise difficult-to-access places, inspectors can stay stationary and reduce chances of slips, falls, or overexertion. While inspectors control the aircraft from the ground floor, the UAV can provide inspectors a bird's-eye view of machines and structures. Furthermore, UAVs can fly around large objects to capture features at different angles. Some parts of a facility may be too far or small for inspectors to photograph with clear resolution. However small, collision-tolerant UAVs are capable of carrying cameras into crowded environments to take close-up images of the objects an inspector cannot reach.

3D mapping of facility interiors

Rotary-wing unmanned aircraft can be used for 3D mapping of facility interiors. Currently, nuclear facility operators and inspectors set up a 3D imaging system on the floor of a facility or carry a system throughout a facility (Juchmann, 2015; Nuclear Decommissioning Authority, 2014). However, a UAV can carry the system high into the air to gain an additional perspective of the facility. Additionally, a UAV can autonomously move around the facility to generate a complete 3D map while the inspector monitors the process, either in or outside the facility. The quality of a scan will depend upon the speed of the UAV and the amount of time the UAV is allotted to doing the scan.

LiDAR 3D mapping is already used in the aviation industry for automated aircraft inspection. This autonomous mapping can detect damage from hailstones or lightning, and even normal wear and tear. The same technique can be applied to pipes, walls, and machines in nuclear facilities. 3D scans generated during routine inspections of a nuclear facility could be used to track any changes made to the facility's structure. Spectral graph theory, a mathematical method for identifying and studying significant features of graphs, can be used to quantify the variance between 3D scans to help inspectors determine when two scans differ beyond the UAV device's standard amount of deviation. Although the 3D indoor imaging has already been proven useful for inspections, radiation contamination maps can be laid over 3D point clouds of nuclear facilities to provide greater awareness of contamination locations (Martin et al., 2016). In this technique, each point in the 3D map of the facility is given a value indicating the level of radiation present at each point. This technology was used to measure radiation levels in the damaged reactors of Japan's Fukushima Daiichi nuclear power plant.

3D mapping of facility exteriors

For nuclear safeguards, the International Atomic Energy Agency (IAEA) uses high resolution (< 1 m) satellite imagery (Johnson et al., 2015). With satellites, IAEA inspectors can verify additional protocol delegations and design information. However, some nations have resolution limits on commercial imagery from satellites. For example, the US Department of Commerce only relaxed restrictions on commercial satellite imagery in 2014. Previously, the Department of Commerce only allowed satellite companies to sell to customers imagery with 50 cm resolution, but now the department allows satellite companies to provide imagery with up to 25 cm resolution.

In a 2007 international safeguards project, aerial photographs were used to construct a map key (with parameters such as fences, buildings, and their relationships to each other) of a nuclear facility that was used to identify features in satellite images with resolutions between 0.61 and 1 m (Jasani et al., 2007). Aerial photographs are required to identify facility features that are difficult to identify using a satellite. Periodically collecting high definition aerial images of facilities helps inspectors track facility changes observed during the continuous satellite monitoring. Lightweight unmanned aircraft would be able to generate imagery for these map keys in a much less intrusive manner than manned aircraft.

Sample collection

Current nuclear safeguards processes involve swipe sampling at nuclear facilities. As part of the IAEA's standard processes, swipe sampling is performed regularly by swiping various surfaces of a nuclear facility. However, environmental sampling can be used for many nuclear activities beyond safeguards verification, including environmental monitoring and emergency response.

Environmental sampling is not a novel concept, but combining the process with unmanned vehicles for nuclear security and safeguards purposes is an emerging area of research. When using unmanned vehicles, environmental samples can be collected in areas with untraversable terrain, rooftops and platforms only reachable by ladders, areas with low lighting or view-obstructing smoke, or areas suspected of chemical or radiological contamination. Collected samples can be (1) transported back to the ground station for analyzation or (2) processed in situ by an analyzer onboard the aircraft. The diagram in Fig. 3 shows typical components found in simple sampling systems. In this example, particles are deposited in a collector to be analyzed at a later time. The

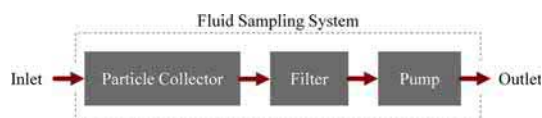


Fig. 3 Diagram of typical components in a mobile sample collection system.

portability of the air sampler, the analyzer, and the necessary pumps and tubing will depend on the types of particles needing to be detected and the predicted minimum detectable quantity. Unmanned vehicles can help nuclear workers collect samples in the air, soil, or water.

Air sampling

Lightweight and handheld air sampling technologies can be adapted to function onboard an unmanned vehicle. Typically, unmanned air sampling is conducted using a UAV, which enables sample collection to occur virtually anywhere in the air. Active and passive samplers can be used onboard UAVs. Active sampling involves using a pump to pull air through the collector. Passive samplers can collect samples from static air or take advantage of the motion of the aircraft to move air through the collector.

Air sampling can be used for environmental monitoring or emergency response. It is often impractical to set up a dense network of permanent sampling stations at a nuclear facility, so a mobile solution is often preferred. During emergencies, such as a radiological release, unmanned systems can be deployed to track and analyze plumes.

Unmanned aircraft are well suited for air sampling and possess many advantages over manned aircraft. During emergencies, such as a radiological release, unmanned aircraft can be quickly deployed. UAVs are typically smaller than manned aircraft and can be more easily transported to the site of the incident. Operating an unmanned aircraft typically requires less training than operating manned aircraft; therefore, training an entire incident response team to pilot unmanned aircrafts is significantly easier than piloting manned aircrafts. The costs of UAVs are significantly lower than manned aircraft, so an incident response team can purchase and deploy multiple unmanned aircraft for the cost of one manned aircraft.

Soil and water sampling

Soil and water sampling can also be conducted onboard unmanned vehicles. When samples need to be collected from areas deemed inaccessible to humans, unmanned aircraft and rovers can be used to transport sample collection equipment to the inaccessible environments. When collecting samples in the field, the robotic persistence of unmanned systems is advantageous when samples need to be collected from an exact location multiple times.

For soil sampling, unmanned vehicles can be equipped with payloads that collect particles from the surface of the soil. Simple collection techniques involve using a wipe, swab, or adhesive that makes contact with the soil surface. More advanced techniques involve using a heated probe to vaporize the surface of the soil and using a pump to pull the vaporized particles into the collection device. Water sampling techniques also involve using a pump to pull fluid into an onboard collection system.

Safety and security risks

Safety problems concerning unmanned systems

Unpredictable situations can cause unmanned systems to fail and create risks to the people and property around them. Unmanned vehicles can cause serious bodily harm. Even small UAVs can cause personal injuries by cutting people with their rotors or causing a blunt force trauma by crashing at high speeds. In terms of property damage, UAVs can crash into and break windows, dent pipes, or fall into and clog ducts. UAV retrieval is a potential safety issue as well. For example, a UAV could become stuck in an area where it is too dangerous for a human to retrieve it, such as an area with high levels of radiological contamination. Catastrophic failure causes the most apparent safety risks, yet any situation that prevents a UAV from completing its intended mission can result in a safety risk. Unmanned system mission failure can be caused by an assortment of circumstances including poor weather conditions, mechanical failure, operator error, or an attack.

Weather conditions

Some smaller UAVs are not built to withstand high winds or other strong weather conditions. Other UAVs are structurally capable of withstanding the weather but are designed and certified only for visual operation. For these aircraft, poor weather conditions may prevent the operator from keeping sight of the aircraft during the entire mission. Operating procedures should be adopted to ensure UAVs are not deployed in conditions that could create an unreasonable safety risk for people or property.

Mechanical failure

A UAV mechanical failure does not have to be catastrophic to cause a safety problem. Any failure that causes the UAV to abort a mission can pose a safety risk. For example, the failure of the GPS system while out of the operator's sight could cause the UAV to fly into a dangerous area. The likelihood of mechanical failure can be prevented by proper reliability and maintainability practices. Vital parts of the aircraft should be monitored for integrity. If components of a system are not considered highly reliable individually, the system's reliability can be improved by adding component redundancy. Although precautions such as these can make UAVs highly reliable, risk of failure cannot be completely eliminated.

Operator error

This category of UAV mission failure is virtually boundless. Good training will prevent ignorance and prevent causes of carelessness and uncontrollable conditions. Training programs reduce chances of operator error, but the risk of human error is always present

and unpredictable. Human error is the primary cause of 17% of unmanned mishaps; however, it is the cause of 85% of manned mishaps (Hart et al., 2013).

Operational accidents could be caused by ignorance, carelessness, or uncontrollable conditions. A poorly trained operator may be ignorant about proper flight rules or unaware they cannot fully control the UAV. Alternately, an operator may be knowledgeable but still choose to operate the vehicle in an unsafe manner. For example, an operator could intentionally ignore training and disobey guidelines. Finally, an operator may be working in an environment that makes it difficult to fully control the unmanned vehicle. Hectic environments and exhaustion can easily cause forgetfulness and poor judgment, leading to errors.

Attack

Many UAV attacks are premeditated by well-trained individuals and executed using high-tech equipment. Highly technical attacks could be physical in nature and be designed to down/destroy a UAV with a weapon, or the attack could be a form of cyber warfare. Hacking an unmanned vehicle's communication system and seizing control of the aircraft while it is flying, called *spoofing*, is a major safety concern. Once an attacker has control of the UAV, the UAV could be intentionally crashed or flown to a dangerous area. Attacks are not always carried out by people with specific intentions or strong motives. An attack could be done by fearful bystanders that are unaware of the UAV's nature or mission. Even wild animals, such as hawks, can attack UAVs (NBC, 2014).

Security problems concerning unmanned systems

United States Code 44 USC §3542 defines information security as "The means of protecting information and information systems from unauthorized access, use, disclosure, disruption, modification, or destruction in order to provide integrity, confidentiality and availability." NATO (North Atlantic Treaty Organization) Cooperative Cyber Defence Centre of Excellence describes UAVs as "highly exposed technical systems" that are "highly dependent on external input" (IAEA, 2014). UAV communications are difficult to secure because they require multiple wireless input channels to manage all external inputs. Information onboard a UAV can be compromised in many ways. For example, data confidentiality could be compromised by eavesdropping (loss of data confidentiality), jamming, and spoofing.

Small UAVs are less dangerous than manned aircraft. If a catastrophic failure occurs, a pilot stationed at ground control is much safer than a pilot onboard an aircraft. Unpredictable situations can cause UAVs to fail and create risks to the people and property around them. UAV mission failures include causes such as dangerous environments (e.g., poor weather, high levels of radiation), mechanical failure, and physical attacks on the aircraft and operator error, which are all similar to the primary mission failures of manned aircraft. However, there is one cause of failure that is more likely to occur with unmanned aircraft than manned aircraft: cyberattack.

UAV cyberattacks can be premeditated by well-trained individuals and completed using high-tech equipment. Eavesdropping and spoofing are the two main security problems. Hackers access the communication system and observe the information being transferred between the UAV and ground control. In a spoofing attack, cyberattackers seize control of the UAV in flight and guide it to a specific area or intentionally crash it. When examining a UAV's susceptibility to cyberattacks, a nuclear worker must consider a UAV's frequency spectrum, robustness of the "sense and avoid" software, data link issues for communication and surveillance, and separation minima.

Eavesdropping

Attackers can access the UAV's information by eavesdropping between the sensors and onboard computer or between the onboard computer and ground control. Environmental conditions such as landscape, weather conditions, and flying altitude can affect the integrity of a UAV's secured connection. Tactics such as encryption and tunneling protocol can reduce the risk of eavesdropping. Digital signatures and forgery detection also improve the integrity of data security. Encryption hides transmitted data, but forgery detection will prevent data from being manipulated or overwritten by a non-valid signature (ICAO, 2006).

Jamming and spoofing

In 2011, the US Department of Homeland Security funded a University of Texas research team, which successfully hacked a Department of Homeland Security drone using jamming and spoofing. The UAV's secure connection to ground control was first compromised by jamming, and then a spoofing technique was used to generate a counterfeit GPS signal to commandeer the UAV. The Department of Homeland Security was motivated to do this research after Iranians claimed to take down an American drone using spoofing. Iran claims it was able to jam communications between operators and the aircraft, take control of the UAV, and guide it to a new destination. The Iranians then landed and seized the UAV to inspect its components (Kerns et al., 2014).

A spoofing attack puts control of the UAV directly in the hands of the attacker. Once in the hands of the attacker, the UAV can be broken down, encryptions can be disabled, and the attacker has unlimited time to study and reverse engineer the UAV. Spoof detection and prevention tactics such as frequency unlock monitoring and jamming-to-noise monitoring are being studied, but spoofing attacks are still a threat. A 2014 paper published by the same Texas research team states, "To the best of our knowledge, all modern commercial civil GPS receivers, even those that provide high-integrity measurements, are vulnerable to civil GPS spoofing" (Federal Aviation Administration, 2015).

Conclusion

The unmanned technologies described in this section were selected based on commercial availability, reliability, and ease of use. The first technology described was two-dimensional imaging of confined spaces, which can be performed by collision-tolerant, rotary-wing UAVs that take visual and thermal images of nuclear facilities from difficult-to-access areas. Instead of having inspectors or reactor operators climb ladders to take images of tall equipment and ceiling pipes, UAVs can travel to these poorly accessible areas while the worker remains stationary.

The second technology introduced was 3D mapping of facility interiors. This task can be completed by autonomous, rotary-wing UAVs that have 3D mapping technology that can generate 3D point-cloud maps of facilities. The third technology, 3D imaging of facility exteriors, can be performed by rotary- and fixed-wing UAVs that are suited for outdoor use and also have 3D imaging technology. UAVs used for 3D mapping of facility exteriors can give inspectors better resolution than commercial satellites. In addition to these imaging technologies, rotary- and fixed-wing UAVs can carry air sampling systems in their payload, allowing inspectors to take an air sample virtually anywhere in the air.

Unmanned systems are technically capable of enhancing current nuclear safeguards and security practices. Although there are many advantages to using unmanned systems for safeguards purposes, there are some concerns that need to be addressed. Unmanned systems will need to overcome a series of institutional obstacles before they can be deployed in the field.

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An Illustrative Example of Microdosimetry—Boron Neutron Capture Therapy for Cancer

Trent L. Nichols, Department of Nuclear Engineering, Nuclear Engineering Building, Knoxville, TN, United States

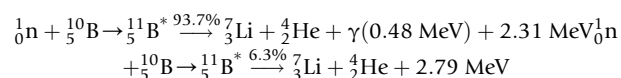
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A brief description of boron neutron capture therapy

This article provides an illustrative example of microdosimetry as it applies to boron neutron capture therapy (BNCT). BNCT is chosen as an example application of microdosimetry because it demonstrates calculation of doses dependent of the distribution of a particular molecule, in this case molecules that have ^{10}B as a component. Additionally, BNCT causes different doses to the cellular nuclei, the cellular cytoplasm, and to the interstitial space (volumes outside of cells but still within the organism). This example is based on the original work of BNCT researchers who attacked the most common primary brain tumor, glioblastoma multiforme. This is a devastating, deadly malignancy. Finally, this contribution is adapted from the doctoral dissertation and papers of the author (Kabalka et al., 2003; Nichols, 2008; Nichols et al., 2002, 2009). In the 1990s, there were BNCT clinical trials for glioblastoma multiforme conducted at Harvard Medical School that used the Massachusetts Institute of Technology's reactor and the Brookhaven National Laboratory's Brookhaven Medical Research Reactor. Dr. Nichols sent glioblastoma patients to both centers and managed them medically pre- and post-BNCT providing the most trial participants outside of the two centers themselves.

After James Chadwick's discovery of the neutron in 1932 (Chadwick, 1932), Maurice Goldhaber collaborated with Chadwick (Chadwick and Goldhaber, 1935), Taylor (Taylor and Goldhaber, 1935), and Burcham (Burcham and Goldhaber, 1936) to study slow-neutron bombardment of the stable isotopes of boron, lithium, and nitrogen. They found that the tracks from the disintegration of boron-10 were two short straight segments consistent with two heavy charged particles. The average range of the particles was found to be $\sim 8\text{ }\mu\text{m}$ in the photographic emulsion. Results obtained by these researchers would show that the actual reaction could be represented as two reaction channels:



where $^{11}\text{B}^*$ represents a metastable state that decays immediately by the two reaction channels as shown above and the probability of the reaction channel is above the arrow.

Boron neutron capture therapy (BNCT) was first proposed in 1936 by the German biophysicist, G.L. Locher, as a potential cancer therapy the same year that Taylor and Goldhaber described the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction (Locher, 1936). Locher had examined the theoretical biological effects of these particles and realized the therapeutic potential for cancer therapy. Part of the attractiveness of this idea is that ^{10}B has a large cross-section at 3840 b for low-energy neutrons ($\sim 0.025\text{ eV}$) relative to other isotopes found in human tissues. Additionally, ^{10}B is stable and even in large doses is non-toxic to humans.

Although much research was accomplished in the subsequent years, the first human trials did not occur until two decades later. Conger and Giles at Oak Ridge National Laboratory (Conger and Giles, 1950) demonstrated the deoxyribo-nucleic acid (DNA) damage from the (n,α) reaction in lily bulbs containing boron when exposed to a beam of thermal neutrons. This inspired neurosurgeon William Sweet and others at the Harvard University Medical School (HMS), Massachusetts General Hospital (MGH), the Massachusetts Institute of Technology (MIT), and the Brookhaven National Laboratory (BNL) (Sweet, 1997). The Brookhaven Medical Research Reactor (BMRR) was constructed expressly for BNCT and contained a full clinical facility for onsite hospital

care. An excellent review of the early work in BNCT at HMS/MIT and BNL may be found in an article written by Sweet (1997). As a neurosurgeon, Sweet wanted to treat the deadly, aggressive, and most common primary brain malignancy: glioblastoma multiforme.

Glioblastoma multiforme (GBM) is truly a horrific killer that recognizes no socio-economic factors, age group, race, or border (Wilson et al., 1991). It arises from the brain's supportive astrocytoma cells and is known as a high grade or Grade III (or IV in some grading systems) astrocytoma (Burger et al., 1988; Daumas-Duport et al., 1988; Escourolle and Poirier, 1978). The tumor is infiltrative in nature, which means that it extends by thin small streams of invasive cells analogous to the roots of a plant that infiltrate the soil. Glioblastomas are often quite large before symptoms occur. The presentation of a GBM might be a fall, seizure, or subtle change in speech (Wilson et al., 1991). Glioblastomas have some cells that undergo an amoeboid-like transition where the cells 'swim' through the brain to distant locations. The cells that undergo an 'amoeboid phase' have been called infiltrative tumor cells (ITC) (Daumas-Duport et al., 1987) and account for the failure of all attempts by all modalities (chemotherapy, surgery, radiation and combinations of these) to achieve long term survival. This 'amoeboid phase' is the reason that performing a hemispherectomy were uniformly unsuccessful (Daumas-Duport et al., 1983, 1982, 1987; Demuth and Berens, 2004; Kelly et al., 1987).

There are no known risk factors or genetic links to the development of GBM (Escourolle and Poirier, 1978). The risk of developing a GBM is proportional to age until the latter years when the risk grows faster than age (Wilson et al., 1991). In the 1950s the median life expectancy after the diagnosis of GBM was about 21 weeks. Nearly 70 years later, a 2017 review of therapies for GBM states a survival of only 8–15 months (Anjum et al., 2017). The course of the disease is progressive deterioration despite aggressive therapy so that the patient spends their most functional time going for treatments that makes them ill. The time following the patient's therapy is characterized by a low functional status prior to inevitable death. Thus, there has been little improvement in the past 70 years in the quality of life for the glioblastoma multiforme patient despite a modest increase in life expectancy. Most strategies begin with debulking surgery followed by radiation and/or chemotherapy. Traditional photon beam radiotherapy is about 40 treatments. BNCT delivers a larger total dose to the tumor and is only a single dose that usually has only very mild side effects (Barth et al., 2005; Busse et al., 2003; Coderre et al., 1998; Coderre and Morris, 1999; Diaz et al., 2000). BNCT provides a better quality of life than other radiotherapies but has not fulfilled its promise as described in the excellent review by Moss (2014). BNCT has been extended to other cancers in the past two decades with some most encouraging results in some such as melanoma (Suzuki, 2020).

Modelling the human cell

Cells in human tissues have irregular shapes and vary in size from a sperm cell of about $30 \mu\text{m}^3$ to nerves whose cell body lies in the spinal cord and innervate muscles in the toes. A good approximation of a 'typical' human cell is about $1000 \mu\text{m}^3$ which is a sphere of about 5 or 6 μm in diameter. It is better to approximate a cell as an ellipsoid as few cells are actual spheres. An ellipsoid may be defined by axes a , b , and c , as is common in nuclear physics (Jelley, 1990) for the atomic nucleus by an eccentricity, ϵ , so that:

$$a = b = \frac{r}{\sqrt{1 + \epsilon}}$$

and

$$c = r(1 + \epsilon)$$

where the volume, V , is given by:

$$V = \frac{4\pi}{3}abc = r^3$$

where r is the radius of a sphere of the same volume as the ellipsoid. Thus, when ϵ equals zero, the ellipsoid is a sphere with $a = b = c$. For $\epsilon > 1$, as ϵ increases, the ellipsoid becomes more elongated (football shaped) which is termed prolate while eccentricities in the range $-1 < \epsilon < 0$ lead to a flattened (pancake) shape which is termed oblate. Cells used in the calculations will be ellipsoids with the nucleus also being an ellipsoid that is entirely within the ellipsoid cell. All ellipsoids are defined as above with an eccentricity, but nuclei and cells do not have to have the same centroid. The nuclear boundary and cellular boundary can touch at a single point or circumferentially. Likewise, the ellipsoidal cells are arranged in a face centered cubic and can have a cell edge to edge separation from zero (touching) or as large as desired.

The cell modelling defines three regions. The first is the volume inside the ellipsoidal nuclei of the cells. Next is the volume inside the ellipsoidal cell minus the nuclear volume which defines the cytoplasm of the cell. Finally, the space outside of cells is called the interstitium and is the volume in a cuboid defined by the outer edges of the cells. As the volume of the cells grow and their edge-to-edge separation increases, the interstitial volume increases.

Other codes calculate the energy deposited in the nuclei, cytoplasm, and interstitium for user supplied concentrations of ^{10}B . The ^{10}B distribution in a volume is uniform though concentration is defined by the region where the boron has an interaction. For example, there could be 100 ppm in all nuclei, 50 ppm in all cytoplasm, and 5 ppm in the interstitium, but the distribution within each volume is uniform. The energy deposition is calculated for each region and then converted to doses to match clinically meaningful calculations.

Finally, for historical reasons, BNCT was originally studied in the most common primary brain malignancy, glioblastoma multi-forme (GBM). It is a very difficult malignancy to treat since it undergoes an amoeboid phase where tumor cells essentially ‘swim’ through brain. Consequently, isolated tumor cells surrounded by entirely normal tissue and the main tumor must be destroyed. It is these distant cells that are responsible for recurrence in traditional photon beam radiotherapy. BNCT holds promise if the tumor preferentially concentrates a boron-10 carrier molecule far more avidly than normal tissue. More recently, BNCT is being used with other malignancies with success (Suzuki, 2020). A good boron carrier though not ideal in the sense of very large preferential concentrations is the boronated essential amino—boronophenylalanine (BPA) that is entirely nontoxic and safe even in very high doses.

Characteristics of neutron beams for BNCT

Until recent years, accelerator sources for epithermal neutron beams were not able to deliver a sufficient neutron flux for BNCT in a time in which a patient could remain relatively motionless without sedation (a few tens of minutes). BNCT is based on the large absorption cross section of ^{10}B for thermal neutrons, i.e. neutrons with energies < 1 eV but with most ~ 0.025 eV (which is of the mean energy in a Maxwell–Boltzmann distribution of thermal neutrons) so it is necessary for neutron energies to be in the thermal range at the depth of the tumor in the human body. Many tumors of interest do not occur on the surface so techniques were developed to treat deep-seated tumors.

The technique used is to employ epithermal neutron beams as those developed at the MIT reactor (MITR) and the Brookhaven Medical Research Reactor (BMRR). These reactor facilities utilized filters (materials design to slow-down the source neutrons and optimize the spectrum of the neutron beam that impinges on the brain) to deliver neutrons with sufficiently high epi-thermal energy to penetrate several centimeters of tissue before slowing to thermal energies. Accelerator based neutron sources (often referred to AB-BNCT) are now capable of obtaining minimal acceptable epi-thermal neutron flux of $\geq 10^9$ n/cm²/s required for BNCT. Additionally, it is possible to adjust AB-BNCT systems so that the epi-thermal neutron energy spectrum will maximize thermal neutron flux at the tumor location in the body (Greenspan and Kani, 1998, 2001).

The distance traveled in tissue by an epithermal neutron may be estimated by assuming that each interaction is dominated by n-p scattering. For low energy n-p reactions, the scattering is predominately elastic which allows evaluation by the classically derived relation:

$$E_p = \frac{1}{2}E_n(1 - \cos\phi)$$

where ϕ is the center of mass scattering angle, E_n is the energy of the incident neutron, E_p is the energy of the recoil proton, and the mass of the proton and neutron have been taken to be equal, which is sufficiently accurate for this calculation. On average, the neutron losses $\frac{1}{2}$ of its kinetic energy in each collision with the proton in a hydrogen atom when E_n is $> \sim 0.05$ eV (at lower energies the neutron energy loss is smaller since the thermal motion of the target protons needs to be taken into account). Thus,

$$E_p = \frac{1}{2}E_n.$$

For an initial neutron energy of E_i that undergoes n collisions, the final average energy, E_f , is approximately:

$$E_i \left(\frac{1}{2}\right)^n = E_f$$

or solving for n :

$$n = \ln^{-1} \left(\frac{1}{2}\right) \ln \left(\frac{E_f}{E_i}\right)$$

For neutrons of initial energy $E_i = 10$ keV to be reduced to an energy of $E_f = 0.025$ eV, approximately 19 collisions are required. For comparison, approximately 5 collisions are required if the initial energy is 1 eV. The mean free path of a neutron—the average distance the neutron travels between collisions, λ , may be approximated from the macroscopic elastic cross section (product of the scattering cross-section with constituent $i - \sigma_i$, and the number densities, N_i , of the target constituent i) by:

$$\lambda = \left(\sum_i N_i \sigma_i \right)^{-1}.$$

Table 1 shows typical elemental abundance of light elements along with the corresponding number densities, elastic scattering cross sections, number density-elastic cross section products (macroscopic cross sections). By the above equation, the mean free path of a neutron (in tissue) with an energy of 0.025 eV is:

$$\lambda = 0.46 \text{ cm}$$

Using the same calculations, Table 2 contains the mean free path as a function of initial epithermal neutron energy. Using a typical value for the mean free path of 0.67 cm (for an epithermal neutron with an initial energy of 10 keV) neutrons will travel

Table 1 Typical tissue elemental abundance, number density, and elastic scattering cross section.

Element	Percent abundance	Atomic weight	Number density (cm^{-3})	σ (b)	$N\sigma$ (cm^{-1})
H	0.11	1.008	$6.51\text{E}+22$	30.39	1.977084
C	0.51	12.010	$2.53\text{E}+22$	4.746	0.120158
N	0.02	14.010	$8.51\text{E}+20$	12.19	0.010372
O	0.36	16.000	$1.34\text{E}+22$	3.97	0.053320
Na	0.001	22.990	$2.59\text{E}+19$	3.92	0.000102
P	0.001	30.970	$1.92\text{E}+19$	4.37	0.000084
S	0.001	32.070	$1.86\text{E}+19$	1.52	0.000028
Cl	0.001	35.450	$1.68\text{E}+19$	65.32	0.001099

Table 2 Mean free path, λ , for different epithermal neutron energies.

E_i (eV)	$N\sigma$ (cm^{-1})	λ (cm)
1	1.5131	0.66091
10	1.4939	0.66937
100	1.4919	0.67031
1000	1.4846	0.67356
10,000	1.4277	0.70043

~ 12 cm prior to reaching a thermal energy, while neutrons with an initial energies of 1 eV will travel ~ 3.4 cm before they are thermalized. Since the neutrons scatter in random directions, (in the center of mass coordinate system) the total distance traveled is not a straight line. The multiple scattering reactions result in neutrons having momenta vectors with random directions especially on the scale of microdosimetry calculations ($< \sim 150$ μm). Using phantoms at reactors involved in previous BNCT trials, the epithermal neutron flux was nearly totally thermalized at a depth in a phantom of ~ 10 cm (Coderre and Morris, 1999). Typical tumor depths are of a few centimeters so that an epithermal neutron beam will be thermalized close to their location. The important implication for microdosimetry is that, with the exception of superficial tumors, there is no preferred direction of the thermal neutrons a few cm into tissues making the probability of a $^{10}\text{B}(\text{n,p})^7\text{Li}$ reaction dependent only upon the distribution of ^{10}B in the region of interest.

Contributions to the microscopic dose

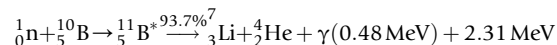
The goal of a neutron source is to produce a neutron beam with a relatively flat energy distribution in the epithermal range as discussed above. Reactor neutron sources will have some γ -ray contamination where the intensity and neutron energy spectrum are dependent on the details of the particular reactor and filters employed. These contaminants to the epithermal beam are not considered in this article because they are installation dependent and the application of BNCT has progressed to accelerator sources. Most AB-BNCT utilize the $^7\text{Li}(\text{p,n})^7\text{Be}$ or the $^9\text{Be}(\text{p,n})^9\text{B}$ reactions as a source of neutrons that have no γ -ray channels (Naito, 2018).

The major sources of the radiation dose from the neutron interactions with tissue are the reaction products from the ^{10}B reaction, i.e. the alpha particles, the lithium nuclei, and the γ -ray. Additional components in the radiation milieu are the proton produced in the nitrogen (n,p) reaction. Recoil protons produced when neutrons elastically scatter from hydrogen atoms attached to organic molecules can also contribute to the dose and will also be considered. Also, hydrogen has a non-negligible cross section for neutron capture that results in deuterium and a γ -ray. Since hydrogen is in all organic molecules, this contribution to the dose will also be considered. The cross sections for other reactions, or the number density, are sufficiently small that other reactions are not considered (Nigg, 1994; Nigg et al., 1991; Raaijmakers et al., 1997, 1995).

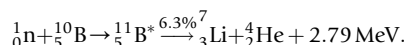
The dose from the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ and $^{10}\text{B}(\text{n},\alpha\gamma)^7\text{Li}$ reactions

Barth, Soloway, et al. report that if the boron concentration in the tumor is 20–40 $\mu\text{g/g}$ of tissue (10^9 ^{10}B atoms/cell) then 75–80% of the radiation dose arises from the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ and $^{10}\text{B}(\text{n},\alpha\gamma)^7\text{Li}$ reactions (Soloway et al., 1997). Other reactions such as the recoil protons in hydrogen scattered by the incident neutrons and the γ -ray from the ^{10}B reaction make up the majority of the remaining dose. Some components that are important to the dose delivered to the tumor and normal tissue are of a lesser significance in understanding of the microdosimetry due to their low magnitude or lack of significant microscopic spatial variation.

Begin with the dose from the ions produced in BNCT:



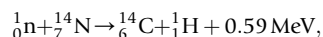
and



The heavy ions above are charged to the 2+ state for the alpha particle and 3+ state for the lithium nucleus. These highly charged ions have relatively short ranges in matter, but the high charge states interact with the electron clouds and atomic nuclei of surrounding molecules by Coulomb interactions resulting in rapidly slowing down that deposits all the energy from the disintegration of the metastable ${}^{11}\text{B}$ that decays immediately ($t_{1/2} \sim 10^{-12}$ s) (Wang, 2006). The ranges of the ions shown in Table 3 were generated by the online code SRIM2008 at www.srim.org. As noted above, the short ranges are beneficial because the ions travel on the order of a cell diameter depositing their entire energy so that the majority of the dose occurs wherever there are ${}^{10}\text{B}$ atoms. Thus, a cell may have the entire energy from the ${}^{10}\text{B}(\text{n},\alpha){}^7\text{Li}$ reaction (~ 2.34 MeV) deposited within its volume. The Coulomb interactions with the molecules along ion paths produce short-lived free radicals that significantly damage other molecules. Any ion paths through or near the cellular nucleus can result in major damage to the DNA from two mechanisms. The first is due to direct damage by the heavy doubly or triply charged ions resulting in double strand DNA breaks which the cell is less able to repair than single strand breaks. The second mechanism is from free radical formation, which also cause significant DNA damages. DNA damage leads to the cell no longer being able to synthesize critical molecules that leads to a loss of homeostasis that leads to cell death. Traditional photon beam radiation damages cells by free radical formation so that the combination of free radicals and direct damage by the heavy ions is much more effective at killing tumor cells than in the case with photon irradiation. Therefore, the efficacy of BNCT is dependent upon the local boron concentration (Barth and Soloway, 1997). Nearby normal cells or tumor cells in the G_0 (resting) phase of the cell cycle will not accumulate the boronated carrier molecule as avidly as tumor cells and will receive a smaller radiation dose than the active tumor cells where the boronated carrier molecule concentrates. The ${}^{10}\text{B}(\text{n},\alpha){}^7\text{Li}$ and ${}^{10}\text{B}(\text{n},\alpha\gamma){}^7\text{Li}$ reactions are responsible for about 80% of the total macroscopic dose and are responsible for most of the variability due to the variation in ${}^{10}\text{B}$ distribution (Barth and Soloway, 1997).

The ${}^{14}\text{N}(\text{n},\text{p}){}^{14}\text{C}$ reaction

Nitrogen is a common element in all human tissues and the ${}^{14}\text{N}(\text{n},\text{p}){}^{14}\text{C}$ reaction produces a significant portion to BNCT dose. The nitrogen capture reaction is:



which occurs in all tissues where there is a neutron flux. The proton carries the majority of the 0.59 MeV and deposits its kinetic energy in a range of $\sim 10.5\ \mu\text{m}$ in human tissue as determined by SRIM2008. The distribution of nitrogen is not entirely homogeneous for different cell types where the concentration is higher in protein rich cells, as opposed to lipid laden tissues, but the heterogeneity is small except for adipose tissue. Even adipose cells have cellular architecture with proteins so that the in vivo heterogeneity is not significant for most tissues and in particular for tumors that rarely have lipid laden cells (except for the rare liposarcoma). No studies describing the nitrogen distribution on a cellular or sub-cellular level have been found by my literature survey to date, and nitrogen is ubiquitous in all tissues. Thus, the nitrogen distribution is modeled as homogeneous. If the nitrogen distribution is homogeneous, the contribution to the total dose from the ${}^{14}\text{N}(\text{n},\text{p}){}^{14}\text{C}$ reaction will be small and additive. This may be shown by plotting the dose for spherical cells of radii $r_c = 4.0\ \mu\text{m}$ with nuclei of radii $r_n = 3.0\ \mu\text{m}$ that are plotted as a function of the edge-to-edge separation of the cells in a body centered cubic. The ${}^{10}\text{B}$ concentrations are modeled as 10 ppm, 50 ppm, and 50 ppm in the interstitium, cytoplasm, and nuclei respectively, which is consistent with experimentally determined values (Coderre et al., 1998; Coderre and Morris, 1999; Elowitz et al., 1998). Fig. 1 contains the doses from the ${}^{14}\text{N}(\text{n},\text{p}){}^{14}\text{C}$ contribution whereas Fig. 2 depicts doses without the proton dose. As the edge-to-edge separation between the cells increases, the dose to the nuclei and cytoplasm decreases which is because there is less dose from neighboring cells because the short range of the boron reaction products is less likely to go from the periphery of one cell to a neighbor. Likewise, the volume of the interstitium receiving a dose from a cellular source remains nearly constant (when cells are close together, some of the ion tracks will go from one cell to another, but the effect falls off rapidly as seen by the dose to the cell nucleus). As the cellular separation increases, the percentage of the interstitial volume receiving a dose originating inside of a cell decreases. These effects and the implications for the efficacy of treatment will be discussed further in subsequent sections.

Table 3 Range of BNCT associated ions in human tissue.

<i>Ion</i>	<i>Atomic number</i>	<i>Atomic mass</i>	<i>Initial energy (keV)</i>	<i>Range (μm)</i>
H	1	1.008	590	10.52
He γ	2	4.003	1470	7.39
He	2	4.003	1780	9.06
Li γ	3	7.016	840	3.93
Li	3	7.016	1010	4.38

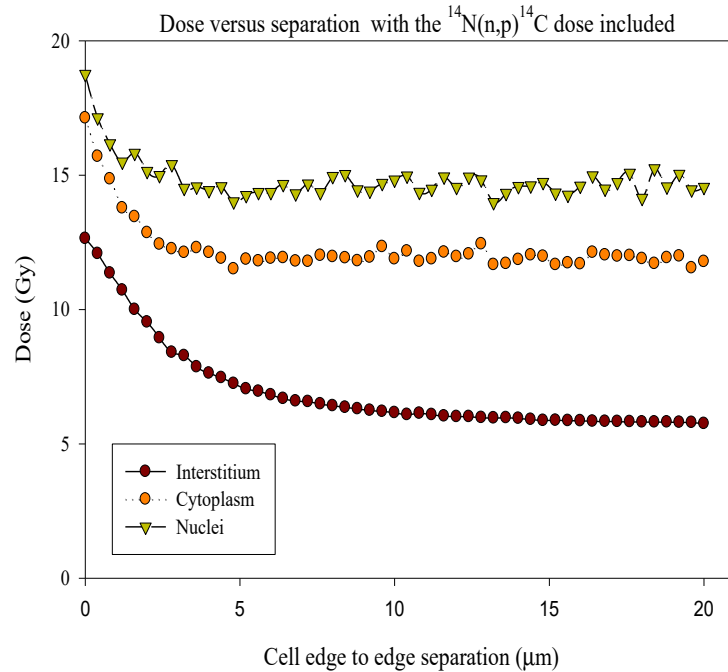


Fig. 1 Dose versus cell edge to edge separation for spherical cells of radius $r_c = 4.0 \mu\text{m}$, nuclei of radius $r_n = 3.0 \mu\text{m}$, and eccentricity of $\epsilon = 0$ for boron concentrations of 10 ppm in the interstitium, 50 ppm in the cytoplasm, and 50 ppm in the nuclei. Dose includes contributions from the boron (n,α) and nitrogen (n,p) reactions.

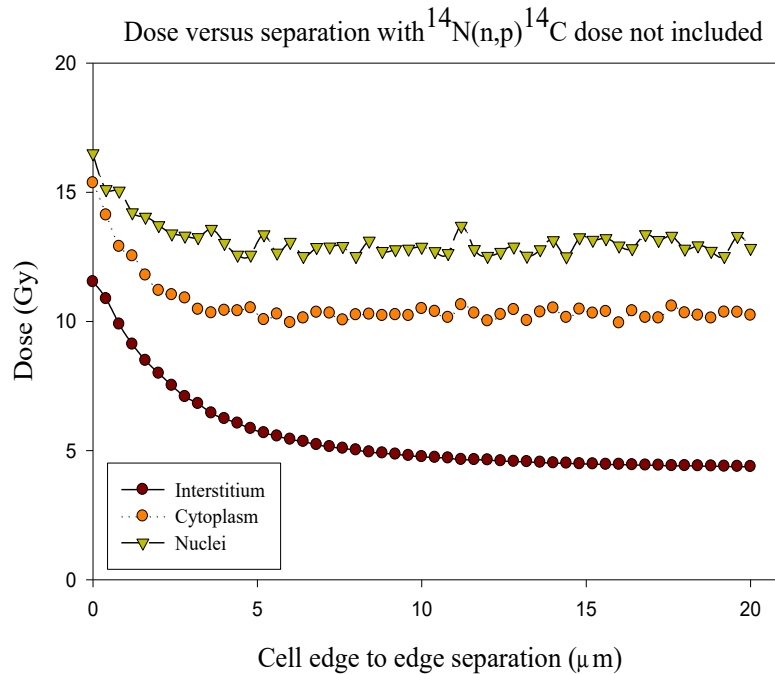


Fig. 2 Dose versus cell edge to edge separation for spherical cells of radius $r_c = 4.0 \mu\text{m}$, nuclei of radius $r_n = 3.0 \mu\text{m}$, and eccentricity of $\epsilon = 0$ for boron concentrations of 10 ppm in the interstitium, 50 ppm in the cytoplasm, and 50 ppm in the nuclei. The dose includes contributions only from the boron (n,α) reaction.

The destruction of the nucleus is generally considered the goal of radiation therapy. **Fig. 3** contains the nuclear dose (dose to the nucleus) as a function of cell edge to edge separation for both cases, i.e. with and without the dose from the $^{14}\text{N}(\text{n,p})^{14}\text{C}$ reaction. The nuclear dose curves (with and without the contribution for nitrogen) are seen to be the same shape with a mean difference of 1.68 Gy with good correlation between the two curves. The proton dose is seen to be $\sim 12\%$ higher in the asymptotic tail

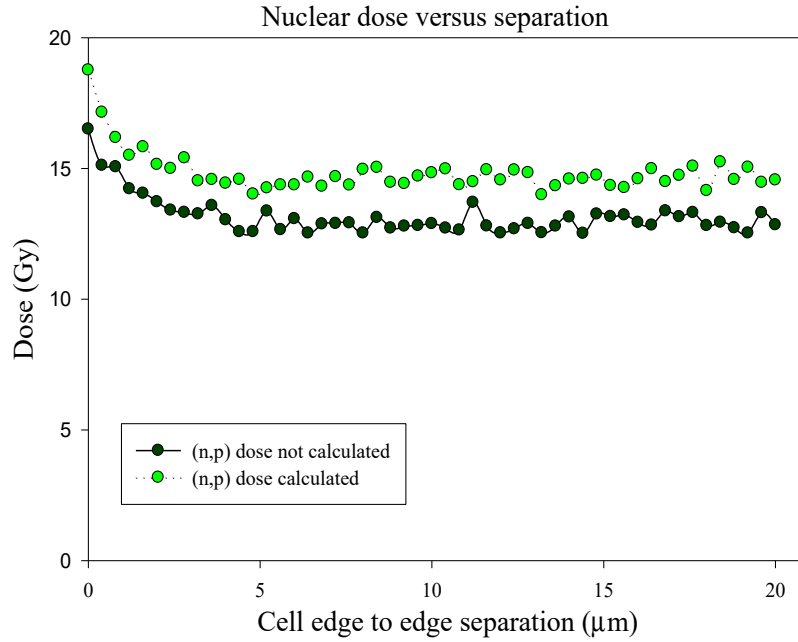


Fig. 3 Nuclear dose versus cell edge to edge separation for spherical cells of radius $r_c = 4.0 \mu\text{m}$ with nuclei of $r_n = 3.0 \mu\text{m}$ and eccentricity of $\varepsilon = 0$ for boron concentrations of 10 ppm in the interstitium, 50 ppm in the cytoplasm, and 50 ppm in the nuclei. The dose includes contributions from the boron (n, α) and nitrogen (n,p) reactions for the upper curve and only from the boron (n, α) reaction for the lower curve.

($\geq 10 \mu\text{m}$ separation) than the dose from the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ reaction alone. Since, in the current models, the nitrogen distribution is homogeneous, the $^{14}\text{N}(\text{n,p})^{14}\text{C}$ reaction will produce a constant additive dose of $\sim 2 \text{ Gy}$. Other authors in the field of BNCT microdosimetry either omit the dose from the $^{14}\text{N}(\text{n,p})^{14}\text{C}$ reaction or model it as a homogeneous distribution. In order to understand spatial variability, the dose from the $^{14}\text{N}(\text{n,p})^{14}\text{C}$ reaction will generally be omitted unless otherwise stated.

Gamma ray dose from the $^{10}\text{B}(\text{n},\alpha \gamma)^7\text{Li}$ reaction

In 93.7% of the ^{10}B interactions, a γ -ray is produced as is shown above in the equations defining the $^{10}\text{B}(\text{n,p})^7\text{Li}$ reactions. This is a 480 keV γ -ray that deposits energy along its track in the tissue with much of the energy being deposited outside of the body. An accepted ^{10}B concentration in the cytoplasm and the nuclei is $\sim 50 \text{ ppm}$ while the interstitial concentration is $\sim 10 \text{ ppm}$ (Coderre et al., 1998; Coderre and Morris, 1999; Elowitz et al., 1998). The interstitium occupies $\sim 15\%$ of the volume for the brain with cells and cytoplasm occupying the remaining volume (Haug, 1987; Moran et al., 1992) so that the number density (N) may be determined starting with the relative tissue composition:

$$N = 0.85(50 \text{ ppm}) + 0.15(10 \text{ ppm}) = 44 \text{ ppm} = 44 \frac{\text{mg}}{\text{kg}}$$

or

$$\begin{aligned} N &= \frac{44 \text{ mg } ^{10}\text{B}}{1 \text{ kg}} \left(\frac{1 \text{ g}}{1000 \text{ mg}} \right) \left(\frac{1 \text{ mole}}{10.0129 \text{ g } ^{10}\text{B}} \right) \left(\frac{6.0221367 \times 10^{23} \text{ atoms}}{1 \text{ mole}} \right) \\ &= 2.65 \times 10^{21} \frac{\text{atoms}}{\text{kg}}. \end{aligned}$$

The γ -ray energy from the ^{11}B metastable nuclear decay is:

$$E_\gamma = 0.48 \text{ MeV/interactions} = 7.69 \times 10^{-14} \text{ J/interactions}.$$

The total number of interactions, I, is given by:

$$I = N\phi\sigma$$

where the fluence, ϕ , for BNCT is a typically (Binns et al., 2005; Fukuda et al., 2003):

$$\phi = 4 \times 10^{12} \frac{\text{neutrons}}{\text{cm}^2}$$

so that the total number of interactions is:

$$I = 2.65 \times 10^{21} \frac{\text{atoms}}{\text{kg}} \left(4 \times 10^{12} \frac{\text{neutrons}}{\text{cm}^2} \right) (3837 \times 10^{-24} \text{cm}^2) = 4.093 \times 10^{13} \frac{\text{interactions}}{\text{kg}}.$$

Now consider an infinite medium, so that all of the γ -ray energy will be deposited in the medium. Then the total dose per kilogram will be given by

$$D_\gamma = E_\gamma I = 2.89 \frac{\text{J}}{\text{kg}} = 3.15 \text{ Gy}.$$

The mass energy-absorption coefficient for a 0.48 MeV γ -ray in human brain tissue may be determined from the Tables of X-Ray Mass Attenuation Coefficients and Mass Energy-Absorption Coefficients at the National Institute of Standards and Technology (NIST) online site [<https://www.nist.gov/pml/x-ray-mass-attenuation-coefficients>] and it is:

$$\mu_{en}/\rho = 3.276 \times 10^{-2} \frac{\text{cm}^2}{\text{g}}.$$

This may be applied to human brain tissue that is largely water and has a density, ρ :

$$\rho = 0.99 \frac{\text{g}}{\text{cm}^3}.$$

By combining these, the energy-absorption coefficient, μ_{en} , is:

$$\mu_{en} = 0.032 \text{ cm}^{-1}.$$

The mean chord length in a convex volume is approximately the mean distance likely to be traversed by the γ -ray as it passes through the cell. The Cauchy relation for the mean chord length in a convex volume provides such an approximation (Mazzolo, 2007; Mazzolo et al., 2003):

$$l = 4 * \frac{V}{A}$$

where V is the volume of the object and A is the area. So for a sphere of radius r , the mean chord length is given by the Cauchy Theorem:

$$l = \frac{4r}{3}.$$

In the present case of microdosimetry, the region of interest is often modeled by small cubes with sides of approximately 150 μm , which corresponds to a volume of $3.375 \times 10^6 \mu\text{m}^3 = 3.375 \times 10^{-6} \text{ cm}^3$. The cube can be approximated by a sphere of the same volume which would then have a radius, $r = 9.3 \times 10^{-3} \text{ cm}$. The mean chord length is then:

$$l = \frac{4 * 9.3 \times 10^{-3} \text{ cm}}{3} = 0.0124 \text{ cm}.$$

The dose to the sphere (with a radius of $9.3 \times 10^{-3} \text{ cm}$) may then be approximated by:

$$D = D_\gamma \left(1 - e^{-\left(\mu_{en} l / 2 \right)} \right) = 6.24 \times 10^{-4} \text{ Gy}.$$

Extrapolation to the whole brain is difficult because less is known about the distribution of boron for the entire brain. A typical glioblastoma may easily be 2–4 cm in diameter (some may be larger but would not likely be a candidate for clinical trials because killing so much tissue would result in a fatal cerebral edema). Consider a 4 cm diameter tumor that would have a volume of 33.51 cm^3 with a boron concentration of 44 ppm from above. The remainder of a typical 1400 cc brain (Lyden et al., 1994) would have a concentration of ~ 10 ppm. Let us first consider an infinite medium of uniform distribution with the number density of B defined by:

$$N = 44 \text{ ppm} \left(\frac{33.51}{1400} \right) + 10 \text{ ppm} \left(\frac{1400 - 33.51}{1400} \right) = 10.81 \text{ ppm} = 10.81 \frac{\text{mg}}{\text{kg}}.$$

So for an infinite medium:

$$I = N\phi\sigma = 9.24 \times 10^{12} \frac{\text{interactions}}{\text{kg}}$$

as before, the total γ ray dose is:

$$D_\gamma = E_\gamma I = 0.768 \frac{\text{J}}{\text{kg}} = 0.768 \text{ Gy}.$$

Approximating the brain by a sphere of volume $V = 1400$ cc, then the radius would be $r = 6.94$ cm with that mean chord length of:

$$l = 9.253 \text{ cm}$$

giving a tumor dose of:

$$D = D_\gamma \left(1 - e^{-\left(\frac{\mu_{en} l}{2}\right)} \right) = 0.106 \text{ Gy.}$$

Detailed analyses demonstrated that (Coderre and Morris, 1999) the microdosimetric or macrodosimetric (over the region of the tumor or brain) doses from the γ -ray from the $^{10}\text{B}(n, \alpha \gamma)^7\text{Li}$ reaction adds a small dose as compared to other doses that are on the order of 10–15 Gy in the microdosimetric region (at the cellular level) and can be ignored.

Dose from recoil protons

Noted above, the epithermal neutron beam is adjusted so that they are thermalized at the depth of the tumor which is usually a few centimeters into the brain for glioblastoma multiforme (GBM). Protons are bound as hydrogen atoms in organic molecules, most commonly to carbon or oxygen. Bond energies may be found in standard chemistry texts and have values of:

$$E_{H-C} = 413 \frac{\text{J}}{\text{mole}} = 4.28 \text{ eV}$$

and

$$E_{H-O} = 498 \frac{\text{J}}{\text{mole}} = 5.16 \text{ eV.}$$

Epithermal neutrons undergo scattering reactions with the protons. This elastic scattering has a rather flat cross section of ~ 20 b versus incident neutron energy over the energy range for epithermal to thermal neutrons as can be seen in plots in the ENDF data set at the National Nuclear Data Center [<https://www.nndc.bnl.gov/exfor/endf00.jsp>].

As shown above, for low energy elastic scattering reactions for energies $0 \leq E_p \leq E_n$, the average energy of the free recoil proton is $\sim$ half that of the incident neutron. The binding energy of the hydrogen to its parent organic molecule decreases the recoil proton kinetic energy in the following manner:

$$E'_p = E_p - E_B = \frac{1}{2}E_n = E_B$$

where E'_p is the net recoil proton energy and E_B is the binding energy that is equal to either E_{H-C} or E_{H-O} that defines a minimum average energy that the neutron must possess in order to break the bond and leave the proton with energy $E'_p \geq 0$ so that the neutron must possess an energy:

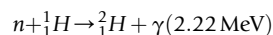
$$E_n \geq 2E_B$$

in order to break a free proton from the molecule. Thus, the neutrons must have an energy greater than ~ 10 eV to generate recoil protons. Neutrons with energies on the order of 0.025 eV will not have sufficient kinetic energy to break the common hydrogen bonds.

Neutrons that have not yet been thermalized can potentially break the chemical bonds as long as $E_n \geq 2E_B$. So, for the H-C bond, the neutron will have more than 8.56 eV of kinetic energy while the energy required to break a H-O bond with elastic scattering would have to be greater than 10.32 eV. If the neutron kinetic energy is equal to the bond energy, the bond will be broken but the proton will have no kinetic energy to dissipate and so provides no dose. Also, low energy neutrons, (a few eV or less), may excite molecular vibrational modes so that not every interaction of sufficient energy to break a bond will produce a recoil proton. The exceedingly large number of different molecular species found in humans with complex vibrational excitation patterns makes the estimation of this mode of energy dissipation beyond the scope of this work. Since this is looking at a depth in the tissue where nearly all neutrons are thermalized, there is no contribution to the microdosimetry from recoil protons because none are produced by thermal neutrons.

Dose from the $^1\text{H}(n, \gamma)^2\text{H}$ reaction

The cross section for thermal neutron capture by hydrogen is not small and hydrogen is found in abundance in all organic molecules. The thermal neutrons will be captured by the hydrogen nucleus which will emit a 2.22 MeV γ -ray in the following reaction:



The cross section for thermal neutron capture by a hydrogen nucleus may be found online at the National Nuclear Data Center and is:

$$\sigma = 0.332 \text{ b.}$$

where the γ -ray energy from the capture of the neutron by the proton is:

$$E_\gamma = 2.2 \text{ MeV/interactions} = 3.56 \times 10^{-13} \text{ J/interactions.}$$

The concentration of ^1H in human tissue is $\sim 10\%$ by weight so using the same calculations as above the number density is:

$$N = 5.97 \times 10^{25} \text{ H} \frac{\text{atoms}}{\text{kg tissue}}.$$

The total number of interactions is:

$$I = 7.93 \times 10^{13} \frac{\text{interactions}}{\text{kg}}.$$

As before, consider an infinite medium so that all of the γ -ray energy will be deposited in the medium then the γ -ray dose is:

$$D_\gamma = 28.22 \frac{\text{J}}{\text{kg}} = 28.22 \text{ Gy}$$

The mass energy-absorption coefficient for a 2.0 MeV γ -ray in human brain tissue may be determined from the Tables of X-Ray Mass Attenuation Coefficients and Mass Energy-Absorption Coefficients at the (NIST) online site:

$$\frac{\mu_{en}}{\rho} = 2.595 \times 10^{-2} \frac{\text{cm}^2}{\text{g}}$$

where the density for human brain tissue as above results in

$$\mu_{en} = 0.026 \text{ cm}^{-1}.$$

As before, for a typical human brain volume of 1400 cc the brain can be approximated by a sphere of volume of radius $r = 6.94 \text{ cm.}$ so that the mean chord length is:

$$l = 9.25 \text{ cm.}$$

The dose to the whole brain is then

$$D = D_\gamma \left(1 - e^{-\left(\frac{\mu_{en} l}{2}\right)} \right) = 0.11 D_\gamma = 3.16 \text{ Gy}$$

The volume over which a 2.2 MeV photon will deposit its energy is significantly larger than the brain volume. In microdosimetry, the region of interest is much smaller—approximately cubes of side $\sim 150 \mu\text{m}$ corresponding to a volume of $3.375 \times 10^6 \mu\text{m}^3$ which could be approximated by a sphere of the same volume with a radius of $r = 9.3 \times 10^{-3} \text{ cm.}$ Then mean chord length becomes

$$l = 0.0124 \text{ cm}$$

Which energy deposition by the 2.2 MeV photon in this volume will result in a dose to the region of interest of:

$$D = 4.94 \times 10^{-3} \text{ Gy.}$$

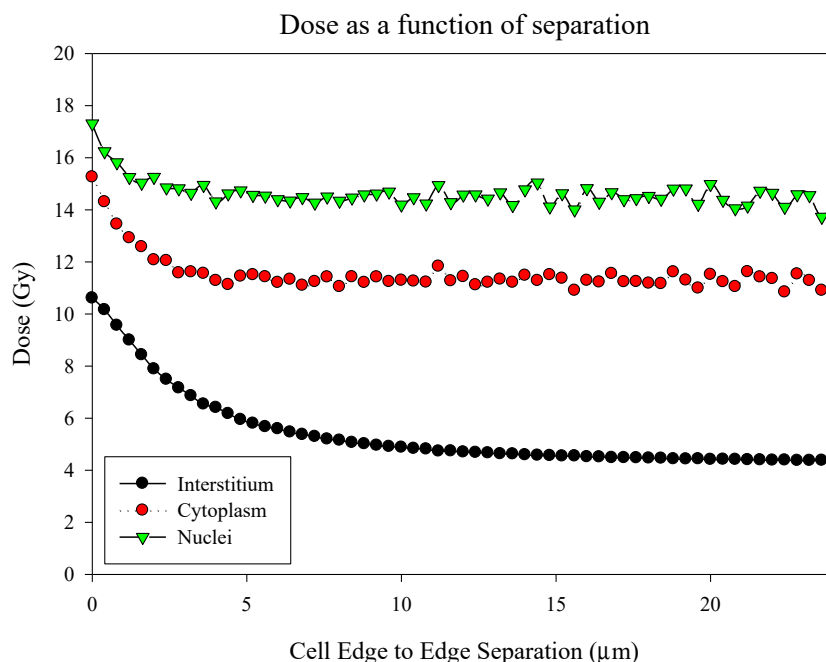
Macrodosimetric treatment doses to the normal brain dose are typically 2–3 Gy with a tumor dose of 30–61 Gy (Joensuu et al., 2003). The dose from the $^1\text{H}(n,\gamma)^2\text{H}$ reaction adds a negligible component to the total dose and will not be further considered. Likewise, the dose from the radiative capture will not be considered in this treatment.

Summary of the components to the radiation milieu

Several authors have reported that the $^{10}\text{B}(n,\alpha\gamma)^7\text{Li}$ reactions account for approximately 86% of the total dose to tumor cells (Barth et al., 2005; Sweet, 1997). Values listed in Table 4 were calculated for cells of radius $5.0 \mu\text{m}$, nuclear radius $4.0 \mu\text{m}$, eccentricity of $\epsilon = 0$, and a cell edge to cell edge separation of $0.0 \mu\text{m}$. The dose from recoil protons is dependent upon the details of the epithermal neutron beam that has flat neutron energy distribution in the energy range of interest. Elastic cross sections as a function of incident neutron energy are also flat in the epithermal energy range so that the elastic proton doses can be averaged as an approximation of the actual dose. These results are presented in Table 4 where there is good agreement with experimental determinations, especially since there are no contributions from neutron beam contaminants in these calculations. The table shows that the $^{10}\text{B}(n,p)^7\text{Li}$ and $^{14}\text{N}(n,p)^{14}\text{C}$ reactions account for over 97% of the dose. This is a crucial point for BNCT as a very large dose is delivered to a tumor with much smaller doses to surrounding normal tissue by assuming a preferential uptake of a ^{10}B containing carrier molecule by the tumor.

Table 4 Dose contribution from different radiation components.

Radiation component	Dose (Gy)	Percent of total (%)
$^{10}\text{B}(n,\alpha)^7\text{Li}$	43.12	89.8
$^{14}\text{N}(n,p)^{14}\text{C}$	3.79	7.9
γ from $^{10}\text{B}(n,\alpha)^7\text{Li}$	0.099	0.21
$^1\text{H}(n,\gamma)^2\text{H}$	0.0049	0.01
Elastic recoil protons	1.02	2.12
Totals	48.04	100.0

**Fig. 4** The dose for the three regions of interest for typical cell sizes of radius $r_c = 5.0 \mu\text{m}$ with spherical concentric nuclei of radius $r_n = 4.0 \mu\text{m}$ and boron concentrations that have been experimentally determined as 50 ppm in the cells and 10 ppm in the interstitium.

How the boron distribution affects the dose

As the boron concentration increases in a region, the number of neutron-boron interactions will increase proportionately. Since the dose is localized, the result scales linearly. Fig. 4 shows a typical case for spherical cells of radius $r_c = 5.0 \mu\text{m}$ with spherical concentric nuclei of radius $r_n = 4.0 \mu\text{m}$ where the dose is seen to be higher when the cells are near to one another as shown. In the asymptotic region where the cells are too far apart to affect each other, the nuclei and cytoplasm receive a larger dose than does the interstitium, where the concentration is one-fifth that of the cells. In order to better understand the role of the boron concentration, the dose may be broken into components. The components may be calculated by setting the boron concentration to zero in two of the three regions with the remaining region having the concentration as depicted in Fig. 5. By examining the nuclear dose only, it is seen that the sum of the three component curves produces the independently calculated upper curve that demonstrates the linearity of the dose in a given region as a function of the boron concentration. Of note is the small increase noted in the first portion of the nuclear dose when all of the boron is in the interstitium. The increase is due to the empty (of boron) cells being next to each other being replaced by boron containing interstitial space as the cells separate. A similar effect is seen in Fig. 6 where only the dose to the interstitium is shown. The three lower curves are once again seen to sum to the independently calculated upper curve. The interstitial dose (when all the boron is concentrated in the interstitial space) is of interest since it is seen to increase with separation. This is because when the cells are close together a larger proportion of the ions will deposit energy into a cell than into the interstitium. As the cells separate, a larger proportion of the ions will have their entire track in the interstitium because a greater portion of the track will be confined to the much larger interstitium.

Also of interest are cases where the ratio of cellular boron to interstitial fluid and to normal cells are higher than has been currently seen with boronophenylalanine (BPA), which is the most commonly used boron-10 carrier molecule. Early animal experiments have produced ratios of 10–20:1 which would mean 100–200 ppm in the cells with 10 ppm in the interstitial fluid and normal cells (Kabalka et al., 2004). The effect of these ratios is seen in Fig. 7 where the corresponding nuclear and interstitial curves

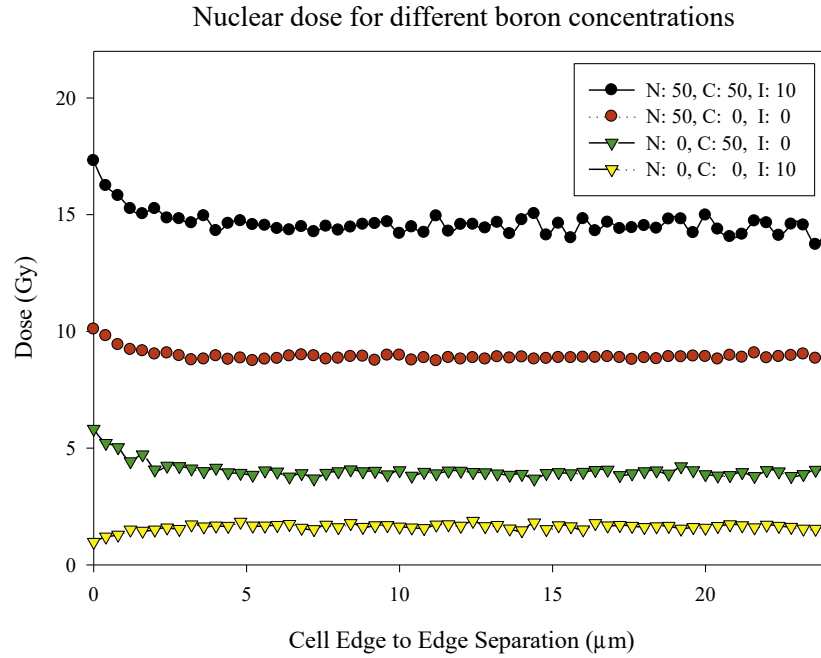


Fig. 5 The nuclear dose only for typical cell sizes of radius $r_c = 5.0 \mu\text{m}$ with spherical concentric nuclei of radius $r_n = 4.0 \mu\text{m}$ and boron concentrations that have been experimentally determined. The three lower curves are the produced by allowing only one of the three regions of interest to have boron with the remaining two regions to have a boron concentration of zero. The sum of the lower three curves is seen to produce the independently calculated upper curve.

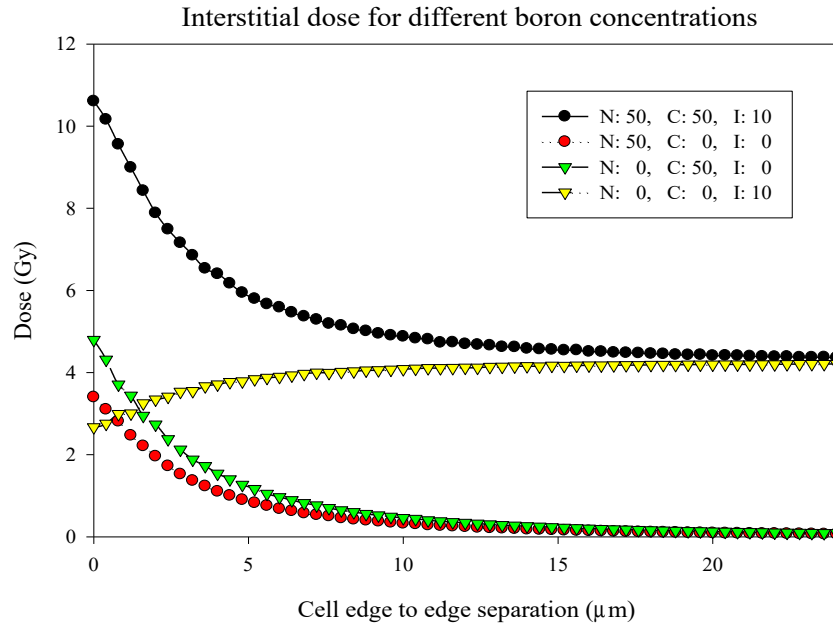


Fig. 6 The interstitial dose only for typical cell sizes of radius $r_c = 5.0 \mu\text{m}$ with spherical concentric nuclei of radius $r_n = 4.0 \mu\text{m}$ and boron concentrations that have been experimentally determined. The three lower curves were produced by allowing only one of the three regions of interest to have boron with the remaining two regions to have a boron concentration of zero. The sum of the lower three curves is seen to produce the independently calculated upper curve.

are seen to be twice the dose at 200 ppm in the cells compared to 100 ppm. The cellular doses are seen to be enormous and the scaling is again seen to be linear. These potentially obtainable doses are truly remarkable when it is remembered that the lethal free air *whole body dose* required for a 95% lethality is ~ 7 Gy. The most important aspect of Fig. 7 is that the nuclear dose is > 50 Gy even for widely separated cells for the 200:1 cell to interstitium boron concentrations. So that active (in the sense of avidly

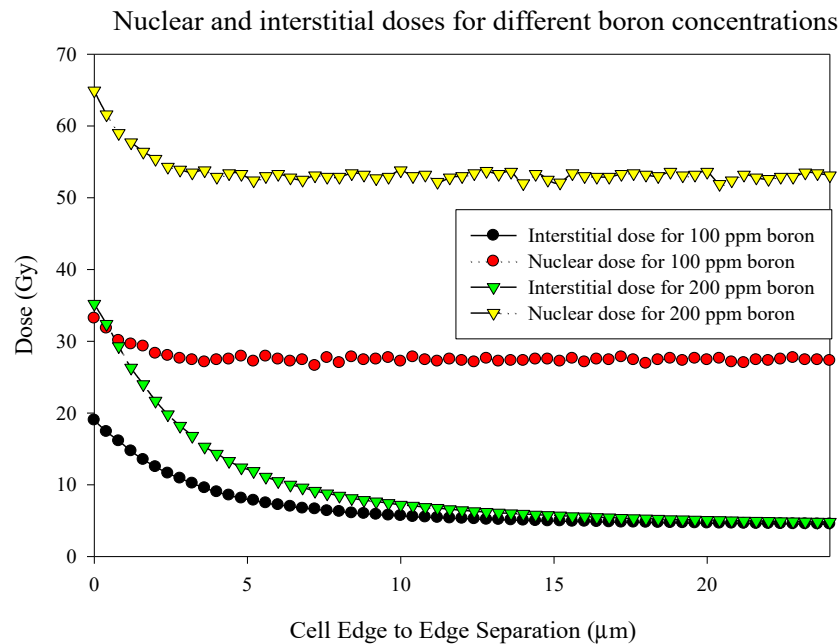


Fig. 7 The dose for the three regions of interest for typical cell sizes of radius $r_c = 5.0 \mu\text{m}$ with spherical concentric nuclei of radius $r_n = 4.0 \mu\text{m}$ and boron concentrations that are potentially achievable with new boron carrier molecules of 200 ppm in the cells and 100 ppm in the cells with 10 ppm in the interstitium in both cases.

accumulating the boron carrier molecule) tumor cells will be destroyed with doses as high as 65 Gy for nuclei in the body of the tumor. At the same time, the surrounding *tissue* will receive < 5 Gy, which is well below a dose that will kill brain cells. The normal tissue will be preserved with much more effective killing of tumor cells even if widely separated. Thus, as new boron carrier molecules are produced that increase the tumor to normal tissue ratio, the effectiveness of the modality will likewise increase.

The effects of cell geometry on the dose

The hypothesis of this dissertation is that cells closely packed together will provide a significant dose to neighboring cells as well as to the interstitial space which in the present model contains normal cells and cells in the G_0 (resting) phase. The model permits dose variance estimations to both the cells and the interstitium as the cells are separated. Observing the separation of the tumor cells corresponds to examining the edge of tumors. The cells will gradually thin out until they are more widely separated. This situation is especially true for infiltrative tumors such as GBM but less so for other tumors such as melanoma which tends to be more circumscribed with few cells separated from the main tumor body. In order to understand the actual *in vivo* microdosimetry, boron concentrations will be set as 10 ppm in the interstitium, 50 ppm in the cytoplasm, and 50 ppm in the nuclei, which is similar to the determinations of boron concentrations in actual patients as determined by Coderre et al. (1998) and Olsson et al. (1998), unless otherwise specified. The size and shape of normal cells is quite variable with cells varying from small (on the order of 3–4 μm in radius) to those up to radii of over 10 μm . Likewise, nuclei can vary from a radius of 1–2 μm up to 8–10 μm for some types of tumor cells. Cell shapes are also quite variable. Some are small, almost cuboidal, as in the Kupffer cells in the liver, others are spherical (especially in white blood cells such as lymphocytes and monocytes), and some are elongated such as myocytes (muscle cells) and neurons (diFiore, 1981; Junqueira and Carneiro, 1980; Reith and Ross, 1977). Spinal neurons may be more than a meter in length extending from the cell body in the upper lumbar spine to the foot. Tumor cells typically display more variability in size and shape than the tissues from which they arise especially in regard to the nuclei which are often larger and more eccentric in shapes (Curran, 1972; Escourolle and Poirier, 1978; Robbins and Cotran, 1979). Furthermore, new studies are demonstrating that living cells are constantly changing shape responding to the local biochemical and physical milieu. The programs used in this modelling were designed for ellipsoidal tumor cells. Changing the underlying geometry allows the modelling of the microdosimetry of different tumors. For example, closely packed spherical cells simulate melanoma whereas a very eccentric geometry with spherical nuclei far from the cell center better approximates neurons. Glioblastoma multiforme cells are somewhere in between these two geometries. To better understand the microdosimetry of BNCT, several cases will be considered.

Fig. 4 shows the effect of separating cells that initially are just touching and gradually increasing the edge-to-edge separation. The figure shows a 16% increase in the nuclear dose from the asymptotic 'tail' to the peak when the cells are just touching. This means that the nucleus of each cell receives an additional dose from neighboring cells which could be viewed as a near neighbor effect. Also, significant is the ratio of 2.46 of the maximum interstitial dose when the cells are just touching to the dose in the asymptotic tail of the interstitial dose. Thus, the interstitium receives a larger dose when neighboring cells are closer together, i.e., the dose from

^{10}B nuclei in the cells is more concentrated when the cells are closer to one another. As the cell radius increases, the effect on the nuclei diminishes if the nuclear radius does not increase in a like manner. This decrease in effect can be seen in Fig. 8 for spherical cells of radius $8.0\text{ }\mu\text{m}$ and a nuclear radius of $4.0\text{ }\mu\text{m}$. The dose to the nuclei is essentially flat whereas the dose to the interstitium shows the earlier noted effect. The explanation has to do with the short range of the ions. The maximum range of the alpha particle from the $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$ and $^{10}\text{B}(\text{n},\alpha\gamma)^7\text{Li}$ reactions (refer to Table 3) is $9.06\text{ }\mu\text{m}$ (6.3% of the time) and $7.39\text{ }\mu\text{m}$ (93.7% of the time) for a mean range of $\sim 7.5\text{ }\mu\text{m}$. Similarly, the mean range of the lithium nucleus is $\sim 4\text{ }\mu\text{m}$. When the cell radius is $8.0\text{ }\mu\text{m}$ and the nuclear radius is $4.0\text{ }\mu\text{m}$, a boron disintegration must be near to the cell edge along a line of centers for the ions to reach the nearest neighbor nucleus. When the cell edges are $1\text{ }\mu\text{m}$ apart, essentially no lithium ions reach the nearest neighbor nucleus and when the edges are $5\text{ }\mu\text{m}$ apart there is no contribution from the alpha particles. Fig. 8 contains a summary of the dose as a function of separation for cells when the cell radius is $8.0\text{ }\mu\text{m}$ and the nuclear radius is $4.0\text{ }\mu\text{m}$. The decrease in the nuclear dose is small decrease in the first $2\text{--}3\text{ }\mu\text{m}$. Thus, the geometry of the cells has an impact upon the nuclear doses. The interstitial dose is affected in a similar manner, with the magnitude of the effect (peak dose/asymptotic dose) is lower than that presented in Fig. 4. The ratio of peak dose/asymptotic dose for the smaller cells (Fig. 4) is ~ 2.3 versus the larger cells where it is ~ 1.9 (Fig. 8). This effect can be understood by first examining the packing fraction of a region defined for this case as the total volume inside the cells (ellipsoids) divided by the total volume of the region of interest. So consider a cube that encloses eight spheres of radius r so that the cube has side $l = 2 \cdot (2r) = 4r$ and the volume of the cube is:

$$V_C = l^3 = (4r)^3 = 64r^3.$$

The total volume enclosed by the spheres is eight times the volume of one of the spheres or:

$$V_s = 8 \left(\frac{4}{3} \pi r^3 \right)$$

The packing fraction, P_f , is then given by:

$$P_f = \frac{V_s}{V_c} = \frac{8 \left(\frac{4}{3} \pi r^3 \right)}{64r^3} = \frac{\pi}{6} = 0.524$$

which is a constant so that the variation seen is not due to the packing fraction for a given separation. Now consider the surface to volume ratio of a sphere that is:

$$R_{SV} = \frac{(4\pi r^2)}{\left(\frac{4}{3} \pi r^3 \right)} = \frac{3}{r}.$$

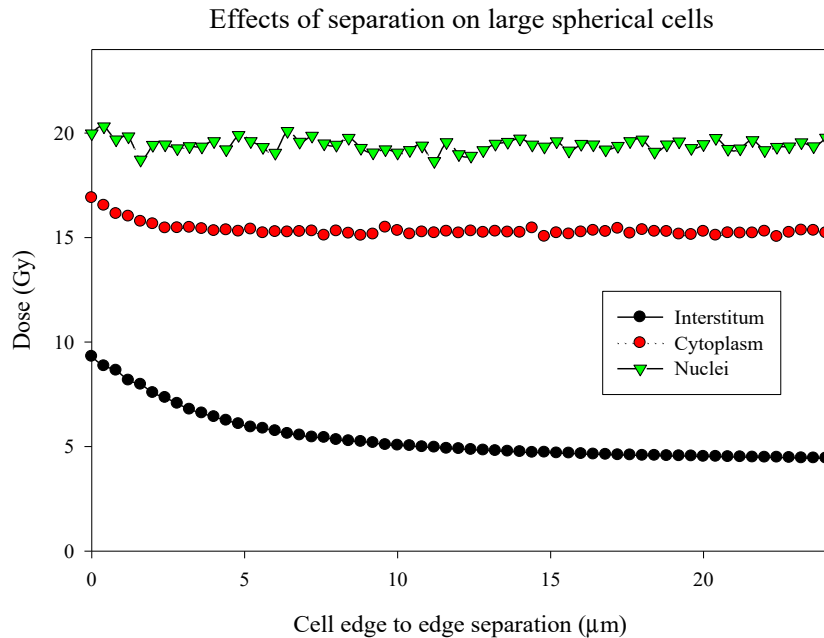


Fig. 8 Dose as a function of cell edge to edge separation for large spherical cells of radius $r_c = 8\text{ }\mu\text{m}$ with nuclei of $r_n = 4\text{ }\mu\text{m}$ for boron concentration of 10 ppm in the interstitium, 50 ppm in the cytoplasm, and 50 ppm in the nuclei demonstrating the increase in dose to neighboring cells and interstitium (at small separations). The effect is less than that observed with smaller cells.

Fig. 9 contains a summary of the average dose to the interstitium over 50 cases at 4 cell radii (5, 8, 11, and 14 μm) for a separation of 0.0 μm fit by non-linear regression to:

$$D = D_0 + \frac{a_0}{r}.$$

The fit is seen to be rather good, meaning that part of the explanation for the smaller dose has to do with the fact that the surface area is increasing less rapidly than the boron containing cell volume. However, the fit does not exactly fit $\frac{3}{r}$, which suggests that there are other factors that must be considered such as ions that pass from one cell to another. The basic observations are that near neighbors contribute to the dose of a cell and that, when cells are widely separated, the dose to the interstitium decreases. These findings have significant implications for the types of tumors that can be treated with BNCT.

The effects of the cell radius and cell separation are shown in Fig. 10 for the interstitial dose. The dose in the interstitium is highest when the cells are small and densely packed. As the cells increase in radius or the cells are farther apart, the dose decreases and, if both occur, the dose can decrease by a factor of 2. Thus, normal tissue will receive lower doses when the tumor cells are far apart and large. The corresponding nuclear dose for the same geometries is shown in Fig. 11. The data in this figure reveal that the nuclear dose decreases as the cells become more separated but increases as the cells increase in size. For large cells, there is little change as the cells separate for the concentric nuclei. Thus, large tumor cells display less near neighbor effects because of the distance to the nucleus. Since the idea of BNCT is to deliver a significantly larger dose to the tumor than is delivered to normal tissue, it is useful to plot the ratio of the total cellular dose to the interstitial dose, which represents normal tissue, as a function of separation and cell radius. The results are plotted in Fig. 12 where it is clear that normal cells far from the tumor will have a greater therapeutic benefit than those near or in the tumor. This once again demonstrates the effects of neighboring cells on the dose. In Fig. 13, the relative dose to the nucleus, as compared to the cytoplasm is plotted as a function of the cell radius for cellular boron concentrations of 50 ppm and 200 ppm while the interstitial boron concentration is 10 ppm for each case. The slight proportional decrease in dose to the nuclei is geometric for the increasing cytoplasmic volume in which an ion can travel without encountering the nucleus. As the nuclear volume goes from 51.2% of the cell volume for a nuclear radius of 4.0 μm and a cell radius of 5.0 μm to 4.8% when the cell radius increases to 11 μm , any given ion is more likely to deposit all of its' energy in the cytoplasm rather than in the nucleus.

The implications of these results combined with an improved boron carrier are obvious. Consider cells of radius 10 μm with concentric nuclei of radius 4 μm separated from one another by 20 μm and boron concentrations of 200 ppm in the cell and 10 ppm in the interstitium. In this situation, the boron reaction dose would be 5.82 Gy to the interstitium, 62.71 Gy to the cytoplasm, and 82.28 Gy to the nuclei. The interstitial dose is still reasonable while the nuclear dose is quite high with a nuclear to interstitial dose ratio > 15 . This would kill all cells that concentrated the boron carrier molecule with little damage to normal tissue in a single treatment. The patient could receive subsequent treatments to hopefully destroy any tumor cells far from the main tumor that were in a G_0 or resting state. The combination of boron distribution and tumor cell architecture has an enormous impact upon the dose to the tumor cells.

The eccentricity of an ellipsoid is a measure of how far the ellipsoid is from spherical. The volume, defined by a sphere of radius r , remains constant as the eccentricity varies. This definition of an ellipsoid also possesses rotational symmetry about

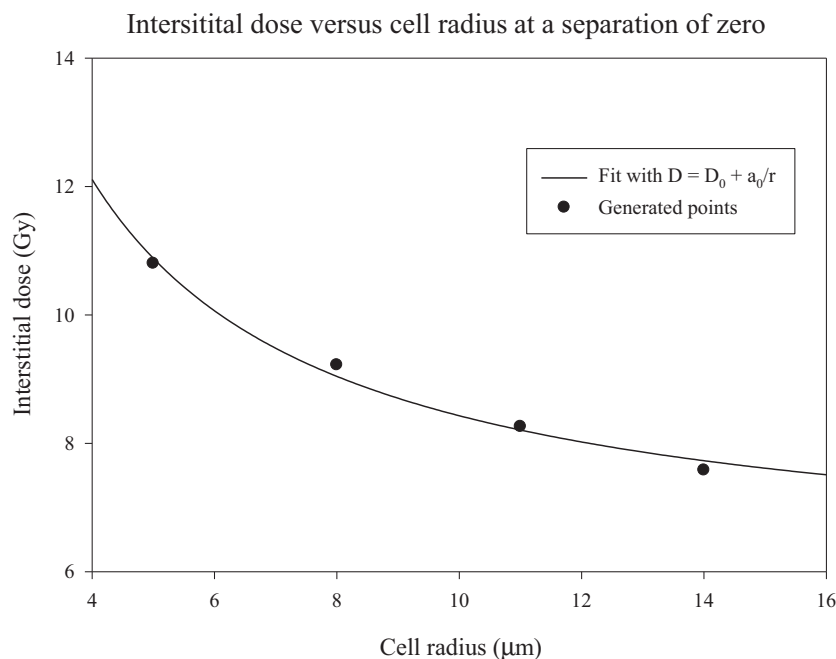


Fig. 9 Interstitial dose averaged over 50 cases with no separation at four cell radii. Fitting is performed non-linearly with the function $D = D_0 + a_0/r$ as derived from the ratio of the surface/volume for spheres. Nonlinear regression was used to fit the data with $D_0 = 5.97$ Gy and $a_0 = 24.52$ Gy $\cdot \mu\text{m}$.

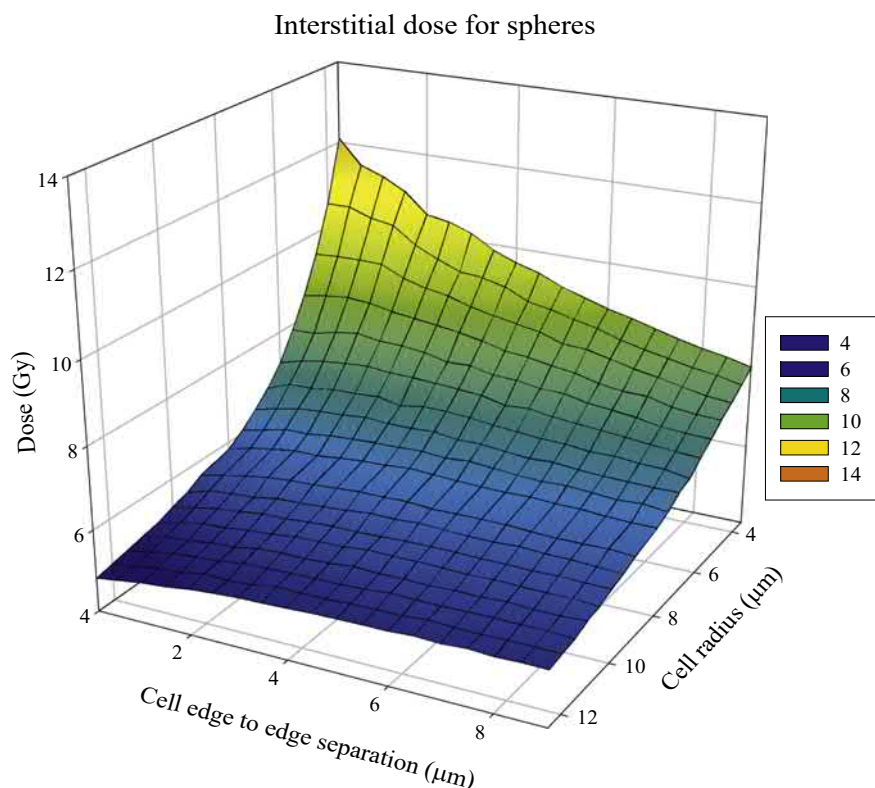


Fig. 10 The dose to the interstitium as a function of cell radius for a concentric nucleus of radius 3.0 μm and cell edge to edge separation for boron concentrations of 50 ppm in the cell (nucleus and cytoplasm) and 10 ppm in the interstitium.

the z axis. When ϵ equals zero, the ellipsoid is a sphere. As ϵ gets larger, the ellipsoid becomes more elongated or prolate, which is referred to as prolate as shown in Fig. 14. Eccentricities in the range $-1 < \epsilon < 0$ lead to a flattened (pancake) shape referred to as oblate. Another view of the ellipsoidal cells is seen in the two cross sections of an ellipsoidal cell shown in Fig. 15 with a radius of 10 μm and eccentricity of 2.5 encompassing a nucleus of radius 4 μm . To determine the effect of the eccentricity on dose, the ratio of the nuclear dose for an ellipsoid with a radius of 10 μm to the dose of an equal-volume sphere both with concentric nuclei is shown as a function of eccentricity and cell separation in Fig. 16. The plot is essentially flat indicating that the response is like that just described for large spherical cells. Ellipsoidal cells can be approximated by spherical cells especially for understanding effects rather than numerical values.

Finally, the nucleus can be allowed to be eccentrically centered. It can be randomly eccentrically centered in every cell, randomly eccentric in the first cell that is then applied to all subsequent cells, or a fixed direction or a fixed distance from the center specified with the other parameter randomly determined. The results from this are of interest because there is essentially no difference between the spherically concentric or randomly placed nuclei. The results are surprisingly similar. For example, the sum of the squares difference between spherical cells of radius 10 μm with concentric spherical nuclei of radius 4 μm and spherical cells of radius 10 μm with randomly placed spherical nuclei of radius 4 μm is 0.06. The data are very similar to those for concentric spherical nuclei and spherical cells and the same observations hold true.

Salient points

Boron distribution, cell geometry, and the relationship between cells are important factors in determining the total nuclear dose. The dose to the nucleus is the crucial dose that results in a loss of homeostasis. DNA damage can also lead to the inability for successful mitosis that also leads to cellular death or a halt to further replication (either case is successful cancer control). Lastly, damage to the DNA can lead to advanced apoptosis, i.e. the programmed cell death that is part of all cells is temporally advanced. Thus, the results are viewed primarily from the standpoint of the nuclear dose.

The interpretation of the results must take into account the interstitium which is the fluid filled space between cells in a tissue. The interstitium refers to the space between tumor cells or more specifically between cells loaded with boron. In this model, the interstitium could contain normal cells or tumor cells that do not avidly uptake the boron carrier. The dose to the interstitium in tumors such as malignant melanoma and adenocarcinoma of the colon represents the dose to the normal tissue so that low doses

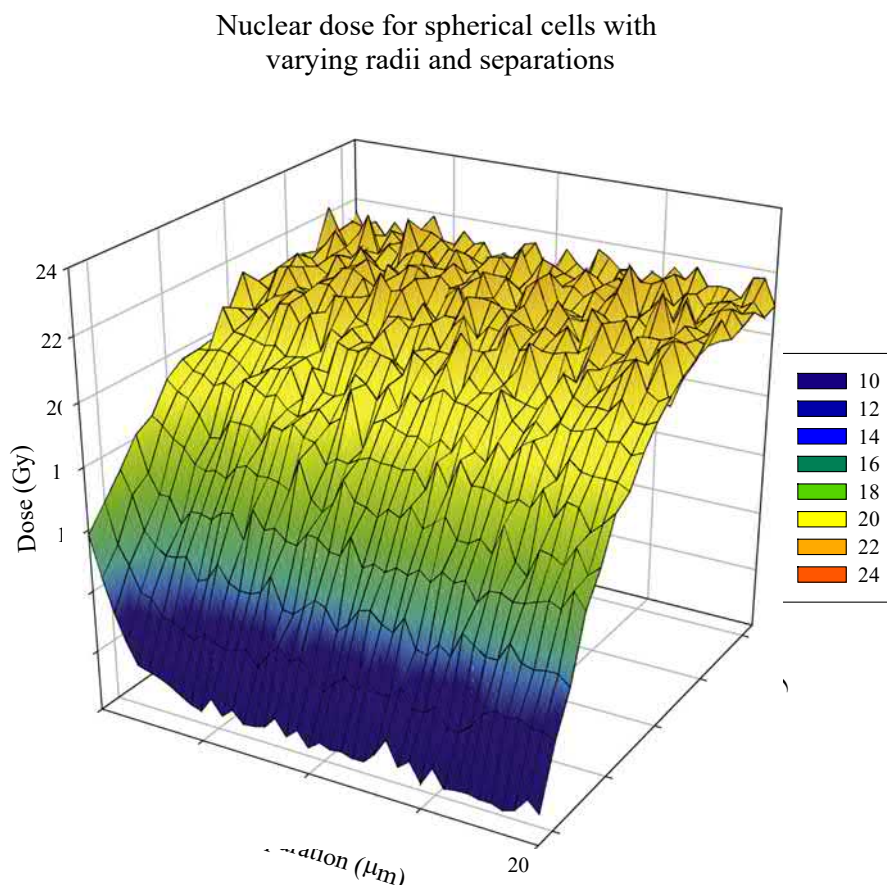


Fig. 11 The nuclear dose as a function of cell radius for a concentric nucleus of radius $3.0\ \mu\text{m}$ and cell edge to edge separation for boron concentrations of 50 ppm in the cell (nucleus and cytoplasm) and 10 ppm in the interstitium.

are important. In infiltrative tumors such as glioblastoma multiforme and angiosarcoma, the interstitium will contain both normal cells and tumor cells that did not avidly uptake the boron agent. This has significant clinical significance as will be discussed.

As noted above for Fig. 4, the dose to the interstitium is ~ 2.4 greater when the cells are touching than when widely separated. Thus, closely packing cells increases the nuclear dose that improves efficacy. The greater increase in dose to the interstitium can be interpreted to mean that cells without boron loading will receive a significantly larger dose when cells are close together but a small dose when they are apart. Furthermore, the ratio of the nuclear dose to interstitial dose ratio is greater when the cells are widely separated, with the tumor cells still receiving a dose greater than 15.5 Gy. Cell death is a statistical phenomenon and is expressed by a cell survival curve as illustrated in Fig. 17. The doses delivered by BNCT are adequate for cell killing especially when the dose to the cytoplasm is combined with the nuclear dose to determine the cellular dose. For example, referring again to Fig. 4, the cellular dose for no separation is ~ 32 Gy and for the asymptotic range is ~ 27 Gy. Both doses are clearly lethal.

The distribution of boron in the cell is also seen to be important with the greatest dose occurring when the boron is concentrated in the nuclei rather than the interstitium by a factor of ~ 2 as seen in Fig. 5. The boron in the interstitium provides a small nuclear dose and is the most important constituent of the dose to the interstitium itself. The greater the concentration of boron in the cell holding the interstitial boron concentration constant, the greater the nuclear dose, in a basically linear nature while the effect on the interstitial dose is minimal. This highlights the dependence of BNCT upon boron carrier molecules. If a carrier molecule can increase the absolute tumor cell boron concentration and increase the ratio of cell to interstitium boron, the efficacy of BNCT will greatly increase.

As cells increase in radius while keeping a constant nuclear radius, the nuclear dose will increase and the influence of nearby cells decreases. Also, the greatest ratio of cell to interstitial dose occurs for large separated cells. The influence of a non-concentric nucleus is relatively small for the geometries studied.

Summary

In summary, the nuclear dose for small cells is increased by nearby cells but shows little near neighbor effect for large or greatly separated cells. The greater the boron concentration, the greater the nuclear dose. If the cell to interstitial boron concentration

Ratio of cell dose to interstitial dose

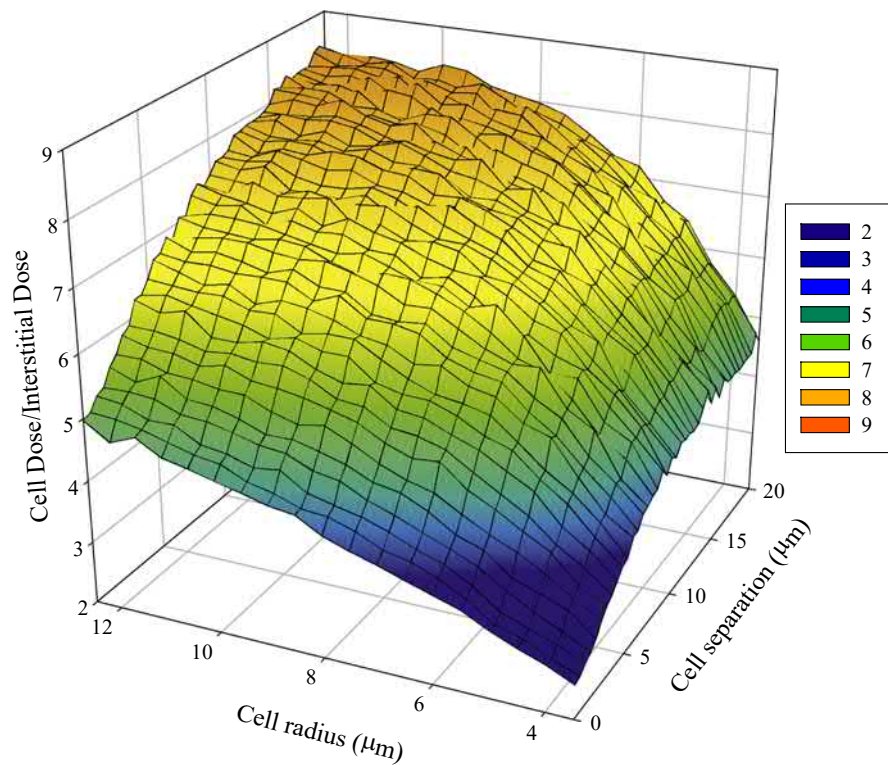


Fig. 12 The ratio of the dose to the cell to the dose to the interstitium as a function of cell radius and cell separation for boron concentrations of 50 ppm in the cell (nucleus and cytoplasm) and 10 ppm in the interstitium.

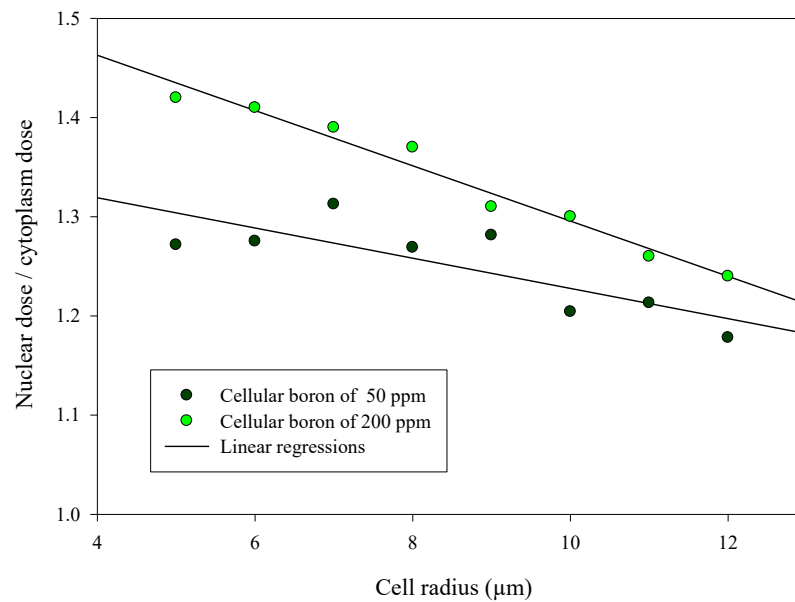
Nuclear to cytoplasmic dose ratio
as a function of cell radius

Fig. 13 The relationship of the nuclear to cytoplasmic dose ratio for two different cellular boron concentrations and for varying cell radii. The noisiness in the 50 ppm boron concentration data is due to the smaller number of boron reactions in total as well as the smaller proportion of the total number of reactions for the region of interest, i.e. 20:1 versus 5:1.

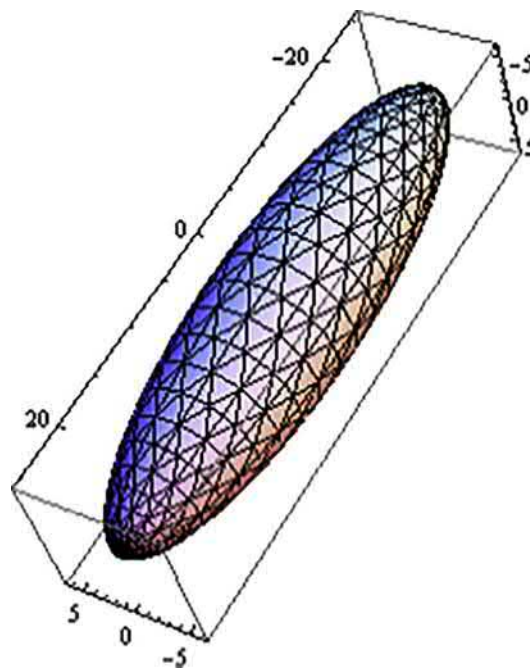


Fig. 14 A representative ellipsoid with radius $10\ \mu\text{m}$ and an eccentricity of 1.5.

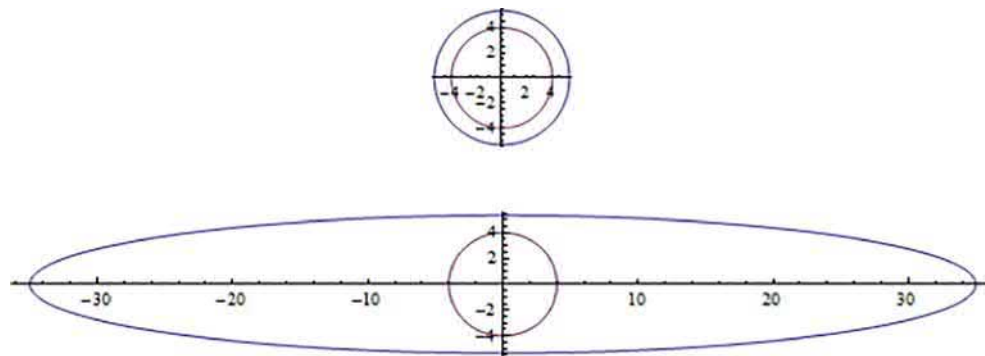


Fig. 15 Two cross sections through a cell with eccentricity of 2.5 and radius of $10\ \mu\text{m}$ and concentric spherical nucleus of radius $4\ \mu\text{m}$.

increases as the cellular boron increases, the interstitial dose changes little, but if the cellular concentration of boron is linearly increased, this should translate into much greater efficacy.

BNCT has been used to treat several malignancies including glioblastoma multiforme, malignant melanoma, head and neck tumors, undifferentiated thyroid carcinoma, and metastatic adenocarcinoma of the colon (Aihara et al., 2006; Ariyoshi et al., 2007; Behr et al., 1999; Dagrosa et al., 2007, 2002, 2003; Joensuu and Tenhunen, 1999; Kankaanranta et al., 2007; Kato et al., 2004; Kennedy et al., 2004; Kouri et al., 2004; Morris et al., 2005; Nano et al., 2004; Pettersson et al., 1992; Pisarev et al., 2007, 2006; Posner et al., 1993; Seppala et al., 2004; Viaggi et al., 2004). Of these, the most clinically studied tumor is GBM. The standard therapy for glioblastoma multiforme is neuro-surgical resection followed by 36 standard external photon beam radiation treatments and in some cases chemotherapy. This standard protocol provides a life expectancy of approximately 12 months from the time of diagnosis (Wilson et al., 1991). The BNCT trials substitute BNCT for conventional photon therapy and demonstrate a survivability of approximately 14 months (Busse et al., 2003, 1997; Diaz et al., 2000; Iwakura et al., 2002; Mishima, 1994). The clinical trials observed that the glioblastoma recurs at the edge of the tumor bed. Since the tumor is infiltrative, the edge of the tumor will be ragged with cells extending into the surrounding tissue. The microdosimetry of BNCT provides an answer. Consider cells in this periphery that did not accumulate a significant boron concentration (in G_0 phase or poor local availability of the boron carrier—usually BPA). This would correspond to a cell in the interstitium that is in the widely separated region. These cells will receive only a few Gray of dose ($\sim 5\ \text{Gy}$) unlike a cell in the interstitium in the tightly packed case with the dose being more than twice that of the periphery. These doses can go from possible to likely lethality. This model would indicate that infiltrative tumors cannot be successfully treated with BNCT with current boron carrier molecules. The only way that an infiltrative tumor can be successfully treated with BNCT is to find an agent that concentrates in all tumor cells avidly, even if the cell is in the G_0 phase, or to administer therapy over

Comparison of the dose to an ellipsoidal cell
and an equivolume spherical cell

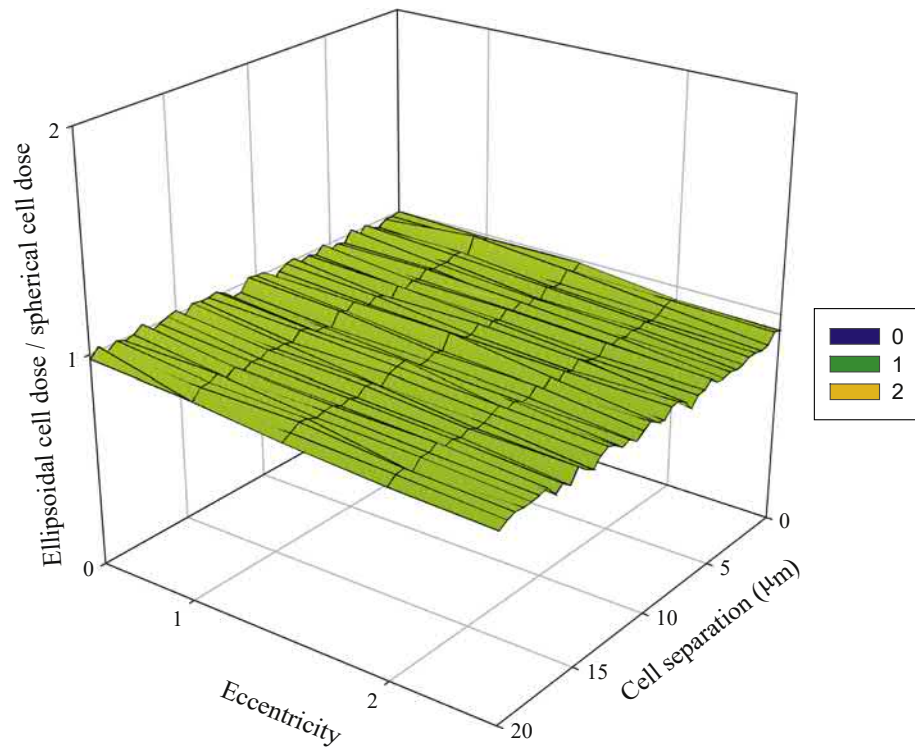


Fig. 16 A comparison of the dose to an ellipsoidal cell and an equivolume sphere both with concentric nuclei of radius $4.0\ \mu\text{m}$. The spherical cell and ellipsoidal cell have the same radius of $10\ \mu\text{m}$. For an eccentricity of ϵ , the three axes (a , b , and c) of the ellipsoid are defined as $c = r(1 + \epsilon)$ and $a = b = \frac{r}{\sqrt{1+\epsilon}}$.

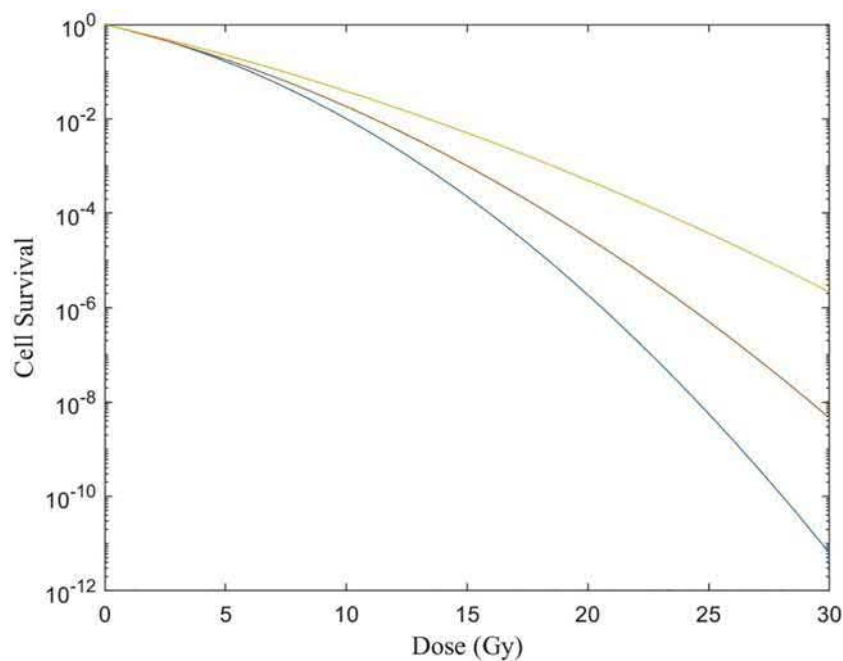


Fig. 17 Illustrative cell survival curves showing the linear quadratic nature of cell demise in radiation with different values of α and β taken from the literature.

several days. Though giving dose fractions might be successful, fractionating the treatment decreases the attractiveness of the therapy when compared to a single session treatment.

The microdosimetry of BNCT is dependent upon cellular geometry and on boron concentrations in the nucleus, cytoplasm, and interstitium. Clinical results correlate with microdosimetric predictions. Future therapies with radiation will have increasing attention paid to the microdosimetry as modalities become more focused and specific.

See Also: Medicine: New Drug Developments—Radiolabeled Molecules in Drug Development; Medicine: Radiopharmaceuticals and Their Use in Nuclear Medicine; Medicine: Therapeutic and Other Applications Using Sealed Radiation Sources.

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Radiation Protection: Glossary of Terms

Shaheen Azim Dewji,^a David C. Kocher^b, and Sherry Adadi^a, ^a Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States; and ^b Oak Ridge Center for Risk Analysis, Inc., Oak Ridge, TN, United States

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Glossary of Terms

Absorbed dose, D The fundamental radiation dose quantity given by

$$D = \frac{d\bar{\epsilon}}{dm}$$

where $d\bar{\epsilon}$ is the average energy imparted to matter of mass dm by ionizing radiation. The SI unit of absorbed dose is joule per kilogram (J kg^{-1}), and its special name is gray (Gy). For purposes of radiation protection and assessments of health risks from exposure to ionizing radiation, the quantity of interest is the average absorbed dose in a tissue or organ (T) given by

$$D_T = \frac{\epsilon_T}{m_T}$$

where ϵ_T is the total energy imparted by ionizing radiation in tissue or organ T and m_T is the mass of that tissue or organ.

Activity, A The number of nuclear transformations (decays of radionuclides) occurring in a given quantity of material per unit time. The SI unit of activity is per second (s^{-1}), and its special name is becquerel (Bq).

Alpha decay Process of radioactive decay in which the nucleus of a heavy element undergoes spontaneous emission of an alpha particle resulting in the nucleus of the decay product with two less protons and two less neutrons. Alpha particles are emitted with discrete energies that are characteristic of the emitting nuclei. See also "alpha particle."

Alpha particle Positively charged particle of electric charge $+2$, identical to the nucleus of a helium-4 atom consisting of two protons and two neutrons, emitted spontaneously from the nucleus in decay of some radionuclides. Alpha particles have low penetrating power and short ranges in tissue and air. Health risks from exposure to alpha particles occur mainly when radionuclides are inhaled, ingested, or absorbed through the skin. Higher-energy alpha particles also can irradiate radiosensitive cells in the basal layer of the skin when alpha-emitting radionuclides are deposited on, or located close to, most areas of the body surface.

Ambient dose equivalent, $H^*(10)$ The dose equivalent at a point in a radiation field that would be produced by the corresponding expanded and aligned field in the ICRU sphere at a depth of 10 mm on the radius vector opposing the direction of the aligned field. The SI unit of ambient dose equivalent is joule per kilogram (J kg^{-1}), and its special name is sievert (Sv). See also "directional dose equivalent," "personal dose equivalent," and "operational quantity."

As low as reasonably achievable (ALARA) See "optimization of protection."

Atomic mass number, A The number of protons and neutrons in the nucleus of an atom.

Atomic number, Z The number of protons in the nucleus of an atom. It determines the chemical properties of an element as well as its place in the periodic table.

Auger electron An electron, similar to an internal conversion electron, emitted from an atom when the energy produced by the filling of a vacancy in an inner shell with an electron from an outer shell of higher energy is transferred to another electron, rather than emitted as a characteristic X ray. Auger electrons are produced in decay of many radionuclides. They also are produced when an inner-shell vacancy is created by irradiation of an atom by photons or electrons.

Becquerel (Bq) The special name for the SI unit of activity: $1 \text{ Bq} = 1 \text{ s}^{-1}$.

Beta decay A form of radioactive decay in which a neutron in the nucleus of an atom is transformed into a proton, thereby increasing the atomic number by one unit, and an electron and antineutrino are emitted from the nucleus. Kinetic energies of emitted electrons have a continuous distribution from zero energy to a maximum value called the endpoint energy. See also, "beta particle," "electron capture," and "positron decay."

Beta particle Energetic electron or positron (positively charged electron) emitted spontaneously from the nucleus in the decay of some radionuclides. Beta particles are not highly penetrating, and higher-energy beta particles have a range in tissue of no more than a few cm.

Bethe stopping power The energy loss of an ion per unit distance traveled in a material, $\frac{d\epsilon}{dX}$. It is also known as the linear energy transfer.

Binding energy The energy required to remove a subatomic particle from an atom, whether removing an electron from the orbital shell of the atom, or a proton or neutron from the nucleus of the atom.

Bragg peak The peak on a curve of the energy loss of ionizing radiation during its travel through matter. For protons, alpha particles, and other ions, the peak is quite pronounced and occurs close to the terminal point of the path of travel. This property makes beams of these particles quite useful in treating relatively small tumors in cancer patients.

Bremsstrahlung Electromagnetic radiation (X rays) produced in a continuous energy spectrum by the deceleration of charged particles, such as when charged particles are decelerated in passing through matter or decelerated, either by a change in velocity or a change in direction, in a device that produces charged particle beams.

Collective effective dose or effective dose equivalent The sum of effective doses or effective dose equivalents to individuals in a defined group from a specified source within a specified time period, or the product of the average effective dose or effective dose equivalent to individuals in a defined group within a specified time period and the number of individuals in that group. The SI unit of collective effective dose or effective dose equivalent is joule per kilogram (J kg^{-1}), and its special name is person-sievert (person-Sv).

Committed effective dose or effective dose equivalent The effective dose or effective dose equivalent received over a specified time following an intake of radionuclides by inhalation, ingestion, or absorption through the skin. The commitment period usually is taken to be 50 years for adults and to age 70 for children.

Committed equivalent dose or average dose equivalent The equivalent dose or average dose equivalent received in a particular organ or tissue over a specified time following an intake of radionuclides.

Compton effect An attenuation effect in which an incident photon interacts with (scatters from) an orbital electron in an atom to produce a recoil electron, a residual ionized atom, and a photon of energy less than the energy of the incident photon.

Contamination Unwanted radioactive material deposited in or on the surface of structures, areas, objects, or people.

Coulomb (C) The SI unit of electric charge, defined as the quantity of electricity conveyed in 1 s by a current of 1 A. One coulomb is an electric charge of $1/e$, where e is the charge of an electron equal to $1.602 \times 10^{-19} \text{C}$.

Critical organ The tissue or organ that is most susceptible of radiation damage resulting from exposure conditions of interest, taking into account doses received by different tissues or organs under those conditions. The critical organ often is the tissue or organ receiving the highest dose, especially when internal exposure is the primary concern.

Cross section, σ A measure of the probability that a specific process will occur when an incident radiation interacts with an orbital electron or electrons or with the nucleus of atoms in a target material and given in SI units of m^2 . It can be thought of conceptually as the size (area) of the object (an atom or its nucleus) that an incident photon or particle must hit for the process of interest to occur. A commonly used unit of cross section is "barn" where $1 \text{ b} = 10^{-24} \text{ cm}^2$.

Curie (Ci) The conventional unit of activity prior to introduction of SI units, defined originally as the rate of radioactive decay (number of decays per second) in 1 g of radium and later as an activity of exactly $3.7 \times 10^{10} \text{ Bq}$.

Decay constant, λ The probability that a radionuclide will decay per unit time, equal to $\ln(2)/T_{1/2}$, where $\ln$ denotes a natural logarithm and $T_{1/2}$ is the radionuclide half-life. The reciprocal of a decay constant ($1/\lambda$) is the mean (average) lifetime of a radionuclide. The decay constant relates the activity of a radionuclide, A , in a given quantity of material to the number of atoms, N , of that radionuclide in the material: $A = \lambda N$.

Decay law A mathematical expression of the amount of activity, A , of a radioactive material remaining at time t , given by $A(t) = A_0 e^{-\lambda(t-t_0)}$, where A_0 is the initial activity at time t_0 .

Decay product A radionuclide produced in decay of another radionuclide.

Deterministic effect See "tissue reaction."

Directional dose equivalent, $H'(d, \Omega)$ The dose equivalent at a point in a radiation field that would be produced by the corresponding expanded field in the ICRU sphere at a depth, d , on a radius in a specified direction, Ω . The SI unit of directional dose equivalent is joule per kilogram (J kg^{-1}), and its special name is sievert (Sv). See also "ambient dose equivalent," "personal dose equivalent," and "operational quantity."

Dose (1) A general term often used to refer to exposure of a material to ionizing radiation. (2) Term which, when quantifying exposure of a material to ionizing radiation, means "absorbed dose" but, in exposure of persons or other living things, does not denote other dosimetric quantities that account for the biological effectiveness of different radiation types, including "dose equivalent," "equivalent dose," "effective dose," "effective dose equivalent," and "radiation-weighted absorbed dose."

Dose coefficient A term used to describe the quantity of dose per unit exposure or dose per unit intake of a radioactive substance.

Dose constraint A prospective and source-related restriction on the individual dose from a source, which provides a basic level of protection for the most highly exposed individuals from a source and serves as an upper bound on the allowable dose in optimization of protection for that source (application of the ALARA principle).

Dose equivalent, H The product of the absorbed dose, D , and the quality factor, Q , for a radiation type of interest at a point in tissue:

$$H = DQ$$

The SI unit of dose equivalent is joule per kilogram (J kg^{-1}), and its special name is sievert (Sv).

For purposes of radiation protection and assessments of health risks from exposure to ionizing radiation, the quantity of interest is the average dose equivalent in a tissue or organ. The average dose equivalent has been superseded by the equivalent dose.

Dose limit Value of the effective dose, effective dose equivalent, equivalent dose, or average dose equivalent to individuals in planned exposure situations that shall not be exceeded.

Effective dose, E The tissue-weighted sum of equivalent doses in specified tissues and organs, first defined in ICRP Publication 60 (ICRP, 1991) and revised in ICRP Publication 103 (ICRP, 2007), given by:

$$E = \sum_T w_T \sum_R w_R D_{T,R} \text{ or } E = \sum_T w_T H_T$$

where H_T or $w_R D_{T,R}$ is the sex-averaged equivalent dose in tissue or organ T, and w_T is the tissue weighting factor. The SI unit of effective dose is joule per kilogram (J kg^{-1}), and its special name is sievert (Sv).

Effective dose equivalent, H_E The tissue-weighted sum of average dose equivalents in all specified tissues and organs, first defined in ICRP Publication 26 (ICRP, 1977), given by:

$$H_E = \sum_T w_T H_T$$

where w_T is a weighting factor representing the ratio of the stochastic risk resulting from tissue or organ T to the total risk when the whole body is irradiated uniformly and H_T is the average dose equivalent in tissue or organ T. The SI unit of effective dose equivalent is joule per kilogram (J kg^{-1}), and its special name is sievert (Sv). The effective dose equivalent has been superseded by the effective dose.

Electron capture (EC) A form of beta decay in which an atomic electron is captured by the nucleus, thereby transforming a proton into a neutron, and the neutron and a neutrino are emitted from the nucleus. Electron capture always competes with positron (β^+) decay whenever the transition energy is greater than twice the rest-mass energy of an electron (1.022 MeV). See also "beta decay", "positron decay."

Electron volt (eV) The energy gained by an electron or other charged particle carrying unit electronic charge in accelerating through a potential difference of 1 V: $1 \text{ eV} = 1.602 \times 10^{-19} \text{ J}$.

Emergency exposure situation An unexpected situation that occurs during operation of a practice, requiring urgent action.

Environmental radiation standard A source-related restriction on the individual dose, radiation level above background, concentrations of radionuclides in the environment, or release rates or cumulative releases of radionuclides to the environment. See also "dose constraint."

Equivalent dose, H_T The dose in tissue or organ T given by:

$$H_T = \sum_R w_R D_{T,R}$$

where $D_{T,R}$ is the average absorbed dose from radiation R in tissue or organ T, and w_R is the radiation weighting factor. Since w_R is dimensionless, the SI unit of equivalent dose is the same as the SI unit of absorbed dose, J kg^{-1} , and its special name is sievert (Sv).

Existing exposure situation A situation that already exists when a decision on control has to be made, including natural background radiation and residues from past practices that were operated outside requirements for radiation protection.

Exposure (1) A general term often used to indicate an event in which ionizing radiation interacts with a person or other living thing. (2) Term which, in measurement of amounts of ionizing radiation, means the amount of ionic charge produced in a volume of air under specified conditions; see "Roentgen."

External exposure Exposure to ionizing radiation emitted by sources outside the body, including sources deposited on the body surface.

Fission A process in which the nucleus of an atom splits into several parts, including two lighter nuclei and several neutrons, with a release of energy. Fission can be induced by absorption of neutrons in the nucleus of some isotopes of heavy elements, and it occurs spontaneously in decays of several such isotopes.

Fusion A reaction in which two nuclei of low atomic number join together (fuse) to form a heavier nucleus with the release of energy equivalent to the difference between the masses of the two lighter nuclei and the mass of the fusion product.

Gamma ray Highly-penetrating electromagnetic radiation produced in de-excitation of atomic nuclei. See also "photon."

Gray (Gy) The special name for the SI unit of absorbed dose: $1 \text{ Gy} = 1 \text{ J kg}^{-1}$.

Half-life, $T_{1/2}$ The period of time that a radioactive substance takes to decay to half of its initial amount. Measured half-lives can span from microseconds to billions of years. This term also is referred to as the physical or radiological half-life.

Half-value layer (HVL) Thickness of a material required to reduce the intensity of incident radiation by one half.

ICRP (International Commission on Radiological Protection) An independent, charitable, and non-governmental organization created in 1928 to advance the science of radiation protection for the public benefit. The ICRP provides recommendations and guidance on protection against the risks associated with ionizing radiation from artificial sources used in medicine, general industry, and nuclear enterprises and from naturally occurring sources. ICRP recommendations have been incorporated in requirements for radiation protection established by regulatory authorities in many countries.

ICRU (International Commission on Radiation Units and Measurements) An organization formed in 1925 to establish international standards for radiation units and measurement. The ICRU's mission is to develop and promulgate internationally accepted recommendations on radiation related quantities and units, terminology, measurement procedures, and reference data for the safe and efficient application of ionizing radiation to medical diagnosis and therapy, radiation science and technology, and radiation protection of individuals and populations.

Internal conversion (IC) Process of radioactive decay in which a nucleus in an excited state interacts electromagnetically with one of the atom's orbital electrons, resulting in emission of the electron from the atom. Emission of internal conversion electrons always competes with emission of a photon (gamma ray) in de-excitation of excited states of nuclei.

Internal exposure Exposure to ionizing radiation emitted by sources inside the body resulting from intakes of radionuclides by inhalation, ingestion, or absorption through the skin.

Ionizing radiation Radiation—in the form of energetic particles or electromagnetic waves—that can remove electrons from atoms, thereby producing ions and an absorption of energy in a medium.

Isobar Atoms having the same mass number but different atomic number.

Isomeric transition (IT) A type of radioactive decay involving de-excitation of a metastable excited state of an atomic nucleus, called an isomer, to a lower-energy state of the same nucleus by emission of a gamma ray or internal conversion electron. An excited state is considered to be metastable if it exists for a measureable period of time.

Isotone Atoms having the same number of neutrons in their nuclei.

Isotope Atoms having the same number of protons but different numbers of neutrons in their nuclei.

Joule (J) The SI unit of work or energy, equal to the amount of energy exerted when a force of 1 N is applied over a displacement of 1 m. One joule is equivalent to 1 W of power radiated or dissipated for 1 s.

Justification The process of determining whether (1) a planned activity involving radiation is, overall, beneficial, i.e., whether the benefits to individuals and society from introducing or continuing the activity outweigh the harm, including radiation detriment, resulting from the activity; or (2) a proposed remedial action in an emergency or existing exposure situation is likely, overall, to be beneficial, i.e., whether the benefits to individuals and society, including the reduction in radiation detriment, from introducing or continuing the remedial action outweigh its cost and any harm or damage it causes.

Kerma, K The quotient of the sum of the initial kinetic energies of charged particles liberated by uncharged ionizing radiation (i.e., photons and neutrons), $d\epsilon_{tr}$, in matter of mass dm . The SI unit of kerma is joule per kilogram (J kg^{-1}), and its special name is gray (Gy).

$$K = \frac{d\epsilon_{tr}}{dm}$$

Kerma consists of contributions from radiative and collisional processes.

Linear attenuation coefficient, μ The quotient of the fraction of particles that undergo interactions $\left(\frac{dN}{N}\right)$ by the traversed distance in a medium, dl ; a quantity that characterizes the interaction probability of uncharged ionizing radiation (i.e., photons and neutrons) in a medium, as a function of the source radiation, energy, and the density of the medium. The SI unit of the linear attenuation coefficient is m^{-1} .

Linear energy transfer (L or LET) The radiation energy lost per unit path length through a material:

$$L = \frac{d\epsilon}{dl}$$

where $d\epsilon$ is the average energy lost by a charged particle owing to collisions with electrons in traversing a distance dl in matter. The SI unit of L is J m^{-1} , often given in $\text{keV } \mu\text{m}^{-1}$.

Mass attenuation coefficient The quotient of the fraction of particles that undergo interactions $\left(\frac{dN}{N}\right)$ by the traversed distance, dl , within a material having density, ρ , given by $\frac{\mu}{\rho} = \frac{dN}{N\rho dl}$. The mass attenuation coefficient has units $\text{m}^2 \text{kg}^{-1}$.

Mass energy absorption coefficient The product of the fraction of the energy of liberated charged particles lost in radiative processes (g) and the mass energy transfer coefficient μ_{tr}/ρ . The mass energy absorption coefficient, μ_{en}/ρ , has units $\text{m}^2 \text{kg}^{-1}$.

$$\frac{\mu_{en}}{\rho} = (1 - g) \frac{\mu_{tr}}{\rho}$$

Mass energy transfer coefficient The quotient of the fraction of incident radiant energy that is transferred to kinetic energy of charged particles dR_{tr}/R by the traversed distance, dl , within a material having density, ρ , given by $\frac{\mu_{tr}}{\rho} = \frac{dR_{tr}}{R\rho dl}$. The mass energy transfer coefficient, μ_{tr}/ρ , has units $\text{m}^2 \text{kg}^{-1}$.

Maximum permissible dose See "dose limit."

Medical exposure Exposure to ionizing radiation incurred by patients as part of their medical or dental diagnosis or treatment; by persons, other than those occupationally exposed, knowingly while voluntarily helping in the support and comfort of patients; and by volunteers in a program of biomedical research involving their exposure.

Natural background radiation Ionizing radiation that occurs naturally in the environment, including cosmic radiation; radiation emitted by naturally occurring radionuclides in rock, soil, air, and water, including radon except as a decay product of uranium, thorium, or radium in materials subject to requirements for radiation protection; radiation emitted by naturally occurring radionuclides in tissues of organisms; radiation emitted by man-made materials containing incidental amounts of naturally occurring radionuclides (e.g., building materials); and radiation emitted by global fallout from testing of nuclear explosive devices.

NCRP (National Council on Radiation Protection and Measurements) A non-profit corporation chartered by the U.S. Congress in 1964 to collect, analyze, develop, and disseminate in the public interest information and recommendations on (1) protection against radiation and (2) radiation measurements, quantities, and units, particularly those concerned with radiation protection.

Neutron An uncharged particle of mass slightly greater than the mass of a proton that is found in the nucleus of every atom except the isotope of hydrogen with atomic mass number of one.

Nominal risk coefficient Estimates of sex-averaged and age-at-exposure averaged lifetime risks of cancer per unit effective dose equivalent or effective dose in a representative population.

Non-stochastic effect See "tissue reaction."

NORM (naturally occurring radioactive material) Radioactive material containing no significant amounts of radionuclides other than naturally occurring radionuclides. Material in which concentrations of naturally occurring radionuclides have been changed by some process are included in NORM.

Occupational exposure All exposure to ionizing radiation incurred by workers in the course of their work, except for (1) excluded exposures and exposures from exempt activities involving radiation or exempt sources, (2) any medical exposure, and (3) the normal local background radiation.

Operational quantities Quantities used in practical applications for monitoring and investigating situations involving external exposure. They are defined for the purpose of measurement and assessment of dose equivalents in the body from external sources. No operational dose quantities have been defined for the purpose of measurement and assessment of dose equivalents from internal exposure. See also "ambient dose equivalent," "directional dose equivalent," "personal dose equivalent."

Optimization of protection (and safety) The process of determining what level of protection and safety makes exposures to ionizing radiation, and the probability and magnitude of potential exposures, as low as reasonably achievable (ALARA), economic and societal factors being taken into account.

Pair production An absorption process in which an incident photon (gamma ray or X ray) is annihilated in close proximity to an atomic nucleus, producing an electron and positron pair. This process occurs at incident photon energies greater than the rest-mass energy of an electron and positron combined (1.022 MeV).

Personal dose equivalent $H_p(d, \Omega)$ The dose equivalent in soft tissue (commonly interpreted as the "ICRU sphere") at an appropriate depth, d , below a specified point on the human body at an angle of incidence, Ω . The SI unit of personal dose equivalent is joule per kilogram (J kg^{-1}), and its special name is sievert (Sv). The specified point is usually given by the position where the individual's dosimeter is worn. See also "ambient dose equivalent," "directional dose equivalent," and "operational quantity."

Photoelectric effect An absorption process in which an incident photon (gamma ray or X ray) ejects an electron from an atom, with all the photon energy absorbed in ejecting the electron and imparting kinetic energy to it. A characteristic X ray or an Auger electron then is produced when the vacated shell is filled by another orbital electron from a higher shell.

Photon Quantum of electromagnetic radiation, having no charge or mass, that exhibits both particle and wave behavior, such as a gamma ray or X ray.

Photonuclear interaction A process in which a highly energetic photon penetrates the nucleus, resulting in the ejection of a particle, usually a neutron, proton, or alpha particle.

Planned exposure situations Exposure situations involving the planned (routine) operation of sources of ionizing radiation including decommissioning, disposal of radioactive waste, and rehabilitation of previously occupied land. Practices in operation are planned exposure situations.

Planned special exposures Planned exposures to ionizing radiation that occur infrequently, for which allowable doses are in addition to dose limits for routine exposures.

Positron decay A form of beta decay in which a proton in the nucleus of an atom is transformed into a neutron and positron, thereby decreasing the atomic number by one unit, and a positron and a neutrino are emitted from the nucleus. Positron decay occurs only when the transition energy is greater than twice the rest-mass energy of an electron (1.022 MeV). See also "beta decay," "electron capture."

Potential exposure Exposure to ionizing radiation that is not expected to occur with certainty but that may result from an accident at a source or an event or sequence of events of a probabilistic nature, including equipment failures and operating errors.

Principles of radiation protection A set of principles that apply equally to all controllable exposure situations, including: (1) the principle of justification, (2) the principle of optimization, and (3) the principle of application of limits on maximum doses in planned exposure situations.

Protection quantities Dose quantities developed for purposes of radiation protection that allow quantification of the extent of exposure of the human body to ionizing radiation from whole or partial body external irradiation and from intakes of radionuclides. Protection quantities include the average absorbed dose in an organ or tissue, equivalent dose (or average dose equivalent), and effective dose (or effective dose equivalent).

Protective action guides Guidelines on doses that would call for public safety measures, such as evacuation or staying indoors, to reduce or prevent radiation exposure during an emergency. See also "reference level."

Public exposure Exposure to ionizing radiation incurred by members of the public from radiation sources, excluding any occupational or medical exposure and the normal local natural background radiation.

Quality factor, Q A factor characterizing the biological effectiveness of a radiation based on the density of ionization along the tracks of charged particles in tissue. Q is a defined function of the unrestricted linear energy transfer (L or LET) of charged particles in water. The quality factor has been superseded by the radiation weighting factor (w_R) in defining the equivalent dose, but it is still used in calculating dose equivalent quantities (dose equivalents at a point) used in monitoring of external exposure.

Rad The special name for the traditional unit of absorbed dose used prior to introduction of SI units: $1 \text{ rad} = 0.01 \text{ Gy}$.

Radiation detriment A concept, first defined in ICRP Publication 60 (ICRP, 1991) and revised in ICRP Publication 103 (ICRP, 2007), used to quantify the harmful effects of radiation exposure in different parts of the body. Detriment is a defined function of several factors, including (1) incidence of radiation-related cancer or heritable effects, (2) the lethality of those conditions, (3) reduced quality of life, and (4) years of life lost.

Radiation-weighted absorbed dose A quantity analogous to the equivalent dose, in which the radiation weighting factor for high-LET radiations represents the biological effectiveness of those radiations, compared with low-LET radiations, in inducing tissue reactions. These weighting factors are less than values that represent the biological effectiveness of high-LET radiations in inducing stochastic effects.

Radiation-weighting factor, w_R A dimensionless factor, first used in ICRP Publication 60 (ICRP, 1991) and revised in ICRP Publication 103 (ICRP, 2007), by which the average absorbed dose to a tissue or organ from a radiation type of interest is multiplied (modified) to reflect the higher biological effectiveness of high-LET radiations, such as alpha particles, neutrons, and heavy ions, compared with low-LET radiations, such as photons and electrons, in inducing stochastic effects. It is used to derive the equivalent dose from the mean absorbed dose in an organ or tissue.

Radioactive decay See "radioactivity."

Radioactive material Material designated in national law or by a regulatory authority as being subject to regulatory control because of its radioactivity, often taking into account the total activity and activity concentration.

Radioactivity Property or characteristic of an unstable atomic nucleus to spontaneously transform with the emission of energy in the form of ionizing radiation, such as alpha particles, beta particles, gamma rays, and X-rays.

Radioisotope An unstable isotope of an element that decays spontaneously, emitting radiation. Approximately 5000 natural and artificial radioisotopes have been identified.

Reference level In emergency or existing exposure situations, the level of dose or risk, above which it is judged to be inappropriate to plan to allow exposures to occur, and below which optimization of protection should be implemented. The chosen value for a reference level will depend on the prevailing circumstances of the exposure under consideration.

Relative biological effectiveness (RBE) The ratio of a dose of a low-LET reference radiation to the dose of a radiation of interest that gives an identical biological effect. Values of RBE vary with the dose, dose rate, and biological endpoint. In radiation protection, the RBE for stochastic effects at low doses (RBE_M) is of particular interest.

Rem (roentgen equivalent man) The special name for the traditional unit of dose equivalent prior to introduction of SI units: $1 \text{ rem} = 0.01 \text{ Sv}$.

Remedial action level See "reference level."

Risk In the context of health effects associated with exposure to ionizing radiation, the probability that an outcome of interest, such as incidence of cancer, will occur during a specified time period, usually the rest of life, following an exposure.

Roentgen (R) The special name for the traditional unit of exposure to ionizing radiation, defined as the amount of gamma rays or X-rays required to produce one electrostatic unit of positive or negative ionic charge in 1 cm^3 of dry air at standard temperature and pressure. In SI units, $1 \text{ R} = 2.58 \times 10^{-4} \text{ coulomb per kilogram (C kg}^{-1}\text{)}$.

Secular equilibrium A condition involving a long-lived parent radionuclide and its radioactive decay product in which the half-life of the decay product is much shorter (e.g., about 100–1000 times shorter) than the half-life of the parent, and the activities of the parent and its decay product become constant and equal after a time comparable to several half-lives of the decay product. This condition occurs only when the parent is sufficiently long-lived that essentially no decay of the parent takes place during a period of about 10 half-lives of the decay product. See also "transient equilibrium."

Serial decay A series of radioactive decay processes in which a parent radionuclide decays into one (or a successive series) of radionuclides before decaying to a stable isotope.

SI units International system (Système International) of units that define the modern form of the metric system. Base SI units of interest in nuclear energy include: length—meter (m); time—second (s); temperature—kelvin (K); and mass—kilogram (kg). Other base SI units include: amount of a substance—mole (mol); electric current—ampere (A); and luminous intensity—candela (cd). SI units of other quantities, such as energy in joule (J), are defined in terms of base SI units.

Sievert (Sv) The special name for the SI unit of equivalent dose and effective dose: $1 \text{ Sv} = 1 \text{ J kg}^{-1}$.

Source (1) An entity for which radiation protection can be optimized as an integral whole, such as the X-ray equipment in a hospital, or releases of radioactive material from an installation. (2) Sources of radiation, such as radiation generators and sealed radioactive sources, and, more generally, the cause of exposure to radiation or radionuclides.

Stochastic effect Malignant disease (cancer) and heritable effects from exposure to radiation for which the probability of an effect occurring, but not its severity, is regarded as a function of dose without threshold.

Tenth-value layer (TVL) Thickness of a material required to reduce the intensity of incident radiation by a factor of ten.

Threshold dose for tissue reactions Dose estimated to result in only 1% incidence of tissue reactions.

Tissue reaction Radiation-induced injury in populations of cells, characterized by a threshold dose and an increase in the severity of the effect as the dose is increased further. Also called “deterministic effect” or “non-stochastic effect.”

Tissue-weighting factor, w_T A dimensionless factor, defined in ICRP Publication 60 (ICRP, 1991) and revised in ICRP Publication 103 (ICRP, 2007), by which the equivalent dose in tissue or organ T is weighted to represent the relative contribution of that tissue or organ to the total health (radiation) detriment resulting from uniform irradiation of the whole body. The value for each tissue or organ T is weighed such that:

$$\sum_T w_T = 1$$

In calculating the effective dose equivalent, as defined in ICRP Publication 26 (ICRP, 1977), w_T represents the relative contribution of organ or tissue T to the total risk of stochastic effects resulting from uniform irradiation of the whole body.

Transient equilibrium A condition involving a parent radionuclide and its radioactive decay product in which the half-life of the decay product is shorter than the half-life of the parent, and the ratio of the activities of the parent and its decay product, but not their activities, is constant after a time comparable to several half-lives of the decay product. Transient equilibrium differs from secular equilibrium in that (1) significant decay of the parent takes place and (2) the activity of the decay product is greater than the activity of the parent. See also “secular equilibrium.”

Worker Any person who is employed, whether full time, part time, or temporarily, by an employer, and who has recognized rights and duties in relation to occupational radiation protection. See “occupational exposure.”

X-ray (1) Electromagnetic radiation produced in de-excitation of bound atomic electrons, which produces X-rays of discrete energies referred to as characteristic X-rays, or (2) electromagnetic radiation produced in deceleration of charged particles, which produces a continuous energy spectrum of X-rays referred to as bremsstrahlung. See also “photon.”

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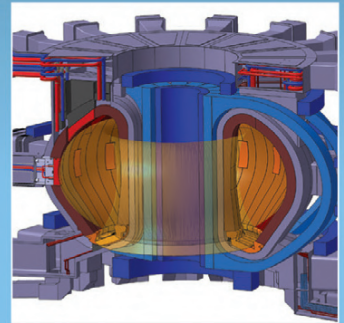
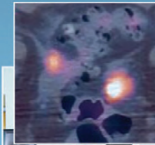
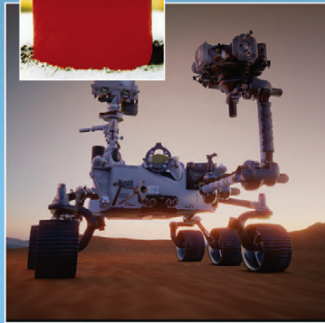
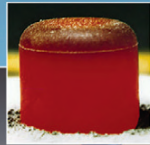
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Ehud Greenspan

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Volume Three



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Nuclear Fusion R&D: *David Campbell*

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ENCYCLOPEDIA OF NUCLEAR ENERGY

EDITOR-IN-CHIEF

Ehud Greenspan

*Retired Professor, Nuclear Engineering Department,
University of California at Berkeley*

VOLUME 3

**Non-Electric Applications of Terrestrial Nuclear Reactors.
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SECTION EDITORS

Shannon M. Bragg-Sitton

Idaho National Laboratory

David I. Poston

Los Alamos National Laboratory

David J. Campbell

Associate Laboratory Director, Idaho National Laboratory



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Radarweg 29, PO Box 211, 1000 AE Amsterdam, Netherlands
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COVER PAGE

Bottom: View of a nuclear power plant by the sea in Vandellos, Spain. It is a 1043 MWe Pressurized Water Reactor.
Top, from left to right (sample applications of nuclear energy):

- a submarine powered by nuclear energy
- an exploration vehicle/lab on MARS powered by a Plutonium-238 nuclear heat source (insert)
- a nuclear medical diagnostic machine for humans with a sample of PET-CT scan (insert)
- an artist vision of a future Tokamak-type fusion energy demonstration power reactor

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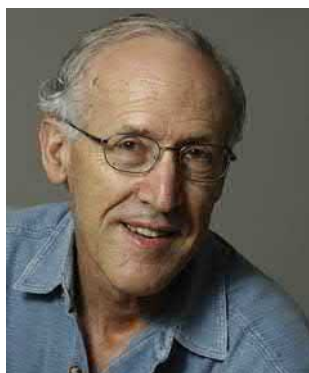
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Ehud Greenspan

University of California at Berkeley
gehud@nuc.berkeley.edu

Professor Greenspan (Ph.D., 1966, in Nuclear Science and Engineering from Cornell University, Ithaca, N.Y., USA) retired from the Department of Nuclear Engineering of the University of California at Berkeley (UCB) in 2014 but served as a Professor in the UCB Graduate School until 2019. He has nearly 60 years of nuclear energy related research experience starting in 1961 as an M.Sc. student at the Department of Nuclear Engineering of the Israel Institute of Technology. His research spans a wide range of nuclear engineering related disciplines including: theoretical and experimental reactor physics; computational methods development including development of new optimization methods for determining the minimum critical mass and for finding optimal design of radiation shields, fusion reactor blankets and medical installations; conception and analysis of advanced fusion fuel cycles and of fusion-fission hybrid reactors; as well as advanced fission reactors and nuclear fuel-cycle conception, design and analysis. The latter activity was the focus of his research during the last twenty-five years. The thrust of this

research is improving sustainability of nuclear energy along with improving economics, safety, and proliferation resistance. He has been the Principal Investigator on studies of dozens of advanced nuclear systems, including those corresponding to Generation-IV reactors and to Small Modular Reactors (SMRs). He was one of the early advocates of SMRs and led a multi-institutional international team that developed a preliminary conceptual design of the Encapsulated Nuclear Heat Source (ENHS)—an innovative nuclear battery-type SMR that is free of pumps, pipes and valves in the primary reactor system. The advanced reactor technologies addressed in his research include: heavy water reactors, pressurized and boiling water reactors, prismatic and pebble-bed liquid-salt cooled reactors, molten-salt reactors, sodium and lead-alloy cooled fast reactors, breed-and-burn reactors, heat-pipe reactors as well as fusion-fission hybrid reactors and accelerator driven nuclear reactors. His research is documented in over 400 refereed publications, over 200 non-refereed publications, 5 book chapters, and a number of patents.

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Introduction to the Encyclopedia of Nuclear Energy

Ehud Greenspan, Department of Nuclear Engineering, University of California, Berkeley, Berkeley, CA, United States

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Introduction

From the beginning of the Universe through to the present day, nuclear energy, nuclear reactions, and nuclear radiations have played major roles in the evolution and shaping of our universe and planet Earth. All the chemical elements we find on Earth were produced by nuclear processes in the Universe. Planet Earth is continuously exposed to nuclear radiation from primordial radioactive isotopes produced in cosmological processes that followed the Big Bang. A number of primordial radioactive isotopes are still among the Earth's constituents, including thorium ^{232}Th , the uranium isotopes ^{235}U and ^{238}U as well as potassium ^{40}K . The spontaneous radioactive decay of the nuclides of these isotopes releases nuclear energy that provides a major contribution to the heat balance of planet Earth. The internal heat source is responsible for keeping the outer core of the Earth in a molten magma state and played a major role in the geologic evolution of the Earth's continents and oceans. The primordial radiation and Earth's heat source may have also contributed to the creation as well as evolution of life on Earth. Since its birth, the Earth has been also constantly bombarded by galactic cosmic nuclear radiation, which primarily consists of protons (hydrogen nuclei) and alpha particles (helium nuclei). Hence, mankind as well as any organism on Earth, were and are being continuously exposed to background nuclear radiation throughout their evolution. The background radiation level as well as the radioactive heat source due to the decay of the primordial radioactive isotopes were significantly higher at the time of Earth formation and are slowly decreasing with time.

It is only within the last hundred and twenty years or so that mankind discovered the structure of the atom and its nucleus and learned how to harness the enormous energy potential contained within the nucleus as well as the different types of radiation nuclei and atoms can emit. Nuclear radiation was discovered in 1896, when Henry Becquerel and Marie Curie discovered that salts of uranium emitted radiation that could be detected with photographic plates. In the 125 years since this discovery, mankind deciphered the structure of the atom and its nucleus, learned how to produce artificial radioactive isotopes (Irène Joliot-Curie and Frédéric Joliot in 1934), discovered how to fission certain heavy isotopes (Otto Hahn, Fritz Strassmann, Lise Meitner and Robert Frisch, 1938/39) and learned how to establish a fission chain reaction (Leo Szilard, Enrico Fermi et al., 1942) and developed numerous applications of nuclear radiation and nuclear energy. The feasibility of releasing nuclear energy by fusing nuclei of light isotopes was postulated already in 1920 (Arthur Eddington) who speculated that such nuclear reactions provide the enormous energy released in the sun and stars. Although research of harnessing nuclear fusion for controlled generation of nuclear energy is being pursued since the late 1940s, mankind has not yet succeeded in producing sufficiently more energy from fusion reactions than the energy invested in inducing these reactions. Additional information on the history and fundamentals of harnessing nuclear energy can be found in (Norman, 2021).

Encyclopedia scope

The Encyclopedia of Nuclear Energy (The Encyclopedia) provides a comprehensive authoritative coverage of a wide range of topics related to the generation, utilization and safe handling of nuclear energy, nuclear radiation and associated technologies, for the benefit of mankind and planet Earth. It covers past developments, the present status, and research and development of advanced innovative technologies that may enable improvements in the generation and utilization of nuclear energy (including nuclear radiation). The Encyclopedia consists of 272 thematic articles written by experts and arranged in 13 Sections; each Section covers the subject area identified in Table 1.

The fundamental concept of this encyclopedia is to provide all this vast and very diversified information in convenient to read cross-referenced articles within a single publication. Each article provides an effective solid foundation of its topic and references for readers interested in deeper exploration of the subject matter.

Table 1 Encyclopedia sections, number of articles per section and the section editors.

Sec. #	Section title (# of articles)	Section editor	Section editor affiliation
1	Fundamentals and history of development of nuclear energy (17)	Eric B. Norman	Professor in the Graduate School, Nuclear Engineering Department, University of California at Berkeley
2	Commercial nuclear power reactors and their design (22)	Russell E. Stachowski	Chief Consulting Engineer, Global Nuclear Fuel
3	Advanced nuclear reactor concepts under R&D (28)	Alexander Stanculescu	Retired Generation Four International Forum Technical Director; Retired Idaho National Laboratory Division Director
4	Safety, regulation, and decommissioning of commercial nuclear power reactors (24)	Joy L. Rempe	Rempe and Associates, LLC
5	Fuel cycle front-end: from mining through utilization (17)	Douglas C. Crawford	Director, Transient Reactor Test Facility, Idaho National Laboratory.
		Colby B. Jensen	National Technical Lead for Fuel Safety Research, Idaho National Laboratory
6	Fuel Cycle back-end: from recycling to deep geological disposal (19)	Christophe Poinssot	Deputy General Director (CEO) and Scientific Director of the French Geological Survey (BRGM) ^a
7	Radiation protection (21)	Laurence Miller	Professor Emeritus, Department of Nuclear Engineering, the University of Tennessee
8	Non-electric applications of nuclear energy (15)	Shannon M. Bragg-Sitton	Lead for Integrated Energy Systems, Idaho National Laboratory ^b
9	Nuclear power for space (16)	David I. Poston	Chief Reactor Designer, Los Alamos National Laboratory
10	Research reactors (23)	Sean O'Kelly	Associate Laboratory Director, Idaho National Laboratory. Chair of IAEA Technical Working Group on Research Reactors
11	Medical, industrial and agricultural applications of nuclear technology (26)	Alan E. Waltar	Retired Professor and Head, Department of Nuclear Engineering, Texas A&M University, College Station
12	Nuclear fusion R&D (26)	David J. Campbell	Munich, Germany. Former Director for Science and Operations, ITER Organization
13	Social issues of nuclear energy (18)	Andrew C. Kadak	Kadak Associates, Inc. Former Professor of the Practice Massachusetts Institute of Technology

^aAlso: Prof. of Nuclear Chemistry at the National Institute of Nuclear Science and technology (INSTN).

^balso National Technical Director for the DOE-NE Integrated Energy Systems program.

Intended readership

The articles are written at a level that users, with a bachelor degree in engineering, scientific and medical disciplines, especially in nuclear engineering, will find understandable and helpful. The Encyclopedia is intended to be an invaluable resource for students and will make an important contribution to the preservation of the vast amount of knowhow developed since the dawn of the nuclear era in the nuclear energy related fields and to its transfer to the new generation.

The Encyclopedia will also be very useful to industry and academia researchers and practitioners in the many scientific and engineering fields related to the generation and use of nuclear energy, including nuclear radiation. It will enable researchers and practitioners to refresh their memory in their field of specialization or expand their knowhow on less familiar subject areas.

The Encyclopedia will also be useful to members of the public with education in engineering or sciences who are interested in learning about specific nuclear energy or nuclear radiation related topics without having to struggle with textbooks or journal articles.

On the need for this Encyclopedia

The publication of this Encyclopedia is very timely for a number of reasons, including the following:

- Nuclear energy can play a very important role in drastically reducing the amount of fossil fuels burned for generation of electricity and, hence, the amount of greenhouse gas (GHG; primarily CO₂) emissions while, at the same time, making more electricity available for developing countries and for supporting the expected growth in the world population. Moreover, nuclear energy can make an important contribution to drastically reducing the amount of GHG emission from the transportation, industrial, residential and agricultural sectors of the economy. Hence, nuclear energy can make a vital contribution to meet the goal set by the 2015 Paris Agreement on climate change and ratified by many nations—to limit global warming to well below 2, preferably to 1.5 degrees Celsius, compared to pre-industrial levels. This point is elaborated upon below in the section: “On the Importance of Nuclear Energy in Avoiding Climate Change Catastrophe”. The state of the art in nuclear energy generation is covered in (Stachowski, 2021; Crawford, 2021).

- The Generation IV (GEN-IV) International Forum (GIF) was created in 2001 as a co-operative international endeavor of research, development, demonstration and deployment of advanced fourth generation innovative nuclear reactors that, hopefully, will be available for commercial deployment by 2030. The objectives of the innovative GEN-IV nuclear energy systems include improved sustainability, economics, safety and reliability, as well as proliferation resistance and physical protection. This encyclopedia provides an update on the state of the GEN-IV R&D efforts and explains the benefits the innovative GEN-IV reactors may provide. The Encyclopedia also describes (Stanculescu, 2021) the advanced nuclear fuel cycles that offer highly improved resource utilization, greatly reduced nuclear waste, and reduced proliferation risks.
- Over the past decade or so, there has been a surge in the number of private companies, mostly supported by venture capital, that are developing advanced innovative Small Modular Reactors (SMRs); reactors designed to generate between 10 to 300 MW of electricity per unit (Stanculescu, 2021). The SMRs offer a new paradigm in the generation and delivery of nuclear power; they promise to improve all aspects of design, engineering, construction and operation of commercial nuclear power plants, including enhanced safety performance, better economic affordability, and reduced financial risk. The components (modules) of SMRs are to be of a standard design, shop fabricated in assembly lines subjected to high quality control, and then transported as complete modules to the power plant site for rapid installation. The SMRs also offer flexible power generation for a wider range of users and applications—central power stations can consist of multiple units installed at a rate that best matches the growth in power demand; alternatively, a single SMR unit can be installed in the vicinity of the power customer using a so-called distributed (localized) power grid. It is even being proposed that the hundreds of shuttered coal power plant sites could be repurposed for SMRs. This encyclopedia describes (Stanculescu, 2021) a sample of the SMRs under development and explains the incentives for their development and the benefits they offer. The Encyclopedia also describes the motivation for, and examples of, so called “microreactors”—reactors producing an electrical power of less than 10 MW. Such reactors are primarily intended for deployment in remote areas which are not connected to the grid and to which the transport of fossil fuel is very expensive, as well as to provide reliable power to sensitive installations even in case of failure of the central power grid.
- The very ambitious space explorations being planned by leading industrial countries cannot succeed without nuclear energy. Space Fission Power (SFP) systems offer the potential to enable truly ambitious exploration of outer space, and ultimately a permanent human presence in space. Most of the benefits of SFP is tied to the very high amount of energy that can be extracted per unit weight of fissile material, which can allow high powers and/or long lifetime at a relatively low mass. SFP systems also offer great flexibility in terms of being able to operate in diverse environments, including environments with no or limited solar radiation, and the ability to provide abundant electrical power and heat from kilowatts to megawatts. Applications of SFP include propulsion and/or power for spaceships, electricity and heat for exploration rovers for landing on Mars (other planets and moons) or for human colonies on the Moon or Mars. This encyclopedia describes (Poston, 2021; Bennett, 2021) the many benefits nuclear energy offers to space exploration and the many challenges that need to be overcome.
- Controlled thermonuclear fusion is the focus of major international research programs that aim to demonstrate the scientific and technological feasibility of large-scale generation of nuclear energy by the fusion of light atomic nuclei. The widely distributed fusion fuel resources together with the favorable anticipated safety and environmental characteristics of future fusion power plants provide a significant motivation for the continued development of the challenging science and technology towards the exploitation of fusion energy for the benefit of mankind. In addition to the description of the fundamental elements of the fusion process and of the principal approaches to the controlled and sustained generation of fusion energy being pursued, this Encyclopedia presents the state-of-the-art for fusion research and development and the progress towards commercial fusion energy generation (Campbell, 2021). The advantages and challenges of nuclear fusion as an energy source are summarized.
- This encyclopedia also describes the many ways nuclear radiation can be harnessed to provide enormous benefits in the fields of medicine, agriculture, modern industry, environmental protection, and public safety (Waltar, 2021). Described in the Encyclopedia are more than a dozen unique properties of radiation technology that allow for these benefits to be achieved. Even though enormous and widespread, the benefits to society from these technologies are largely unknown and unappreciated. The global markets for these technologies exceed USA \$500 billion annually and are growing rapidly.
- Also provided in this Encyclopedia is information on dozens of research reactors (O’Kelly, 2021) that are operating or under construction around the world and provide unique research capability for scientists pursuing advanced research in physics, chemistry, biology, material and other sciences. Another invaluable utilization of research reactors is the production of special radioactive isotopes for numerous applications in medical, industrial and agricultural applications, and scientific research. Some research reactors are also used for demonstrating the performance of evolutionary as well as advanced nuclear fuels and of advanced nuclear power reactor components.
- Public perception of nuclear energy varies from country to country. It tends to be most negative in wealthy democracies like Germany, Australia, and Japan. Contributing to the negative perception is a fear of reactor accidents, a fear of nuclear radiation, a fear of long-lived radioactive waste, and a fear of the nuclear technology misuse for weapons proliferation. The encyclopedia addresses these concerns and provides authoritative objective information that, hopefully, will contribute to dispelling the public misconceptions (Kadak, 2021).
- The Encyclopedia describes measures implemented to ensure the health and safety of the public from nuclear power (Rempe, 2021). These measures evolved as new lessons were learned from operating experience and unplanned events, in particular events with core damage and radionuclide release (including in Three Mile Island, Chernobyl, and Fukushima). Clearly, the

experience gained from the design, evaluation, operation, decommissioning, and regulation of nuclear power plants has led to significant safety improvements. Future advanced reactor designs are expected to incorporate innovative inherent safety features (Stanculescu, 2021).

- The encyclopedia provides a detailed description of the biological effects of nuclear radiation, of how we reliably measure radiation levels and ensure that they do not exceed the permissible levels (Miller, 2021). The Encyclopedia describes the theory and practice of radiation protection that provide the basis for regulatory requirements that are essential for the safe and beneficial use of the many advanced technologies that involve exposure to nuclear radiation. The Encyclopedia explains that everyone is exposed to natural background radiation on a daily basis; that radiation is routinely in use for diagnostics of health conditions, such as for dental and medical X-rays, and that this is the largest source of radiation exposure for almost all human population in developed countries and that the benefits from these practices far exceed their risk. The regulatory permissible level of nuclear radiation is harmless much like a moderate exposure to sunshine is harmless (and may even be beneficial, whereas over-exposure can cause sunburns). In other words, the regulatory limits of doses induced by exposure to nuclear radiation are already substantively protective, and there is no evidence of health improvements by reducing the doses below these limits. Insisting to reduce radiation exposure to as low as possible below the regulatory limits may result in harm.
- Also provided in this Encyclopedia is detailed information about the nuclear waste and how it can be disposed of safely (Poinssot, 2021; Kadak, 2021). Using advanced fuel cycles (Stanculescu, 2021; Poinssot, 2021) the amount and activity of the long-lived nuclear waste can be reduced to few percent of the amount that needs to be disposed of per unit of energy generated using the contemporary “once-through” fuel cycle.

On the importance of nuclear energy in avoiding climate change catastrophe

The world’s population is expected to grow from current ~6.7 billion people to over 9 billion by 2050. As recognized in the United Nations Resolution 70/1 “Transforming our world: the 2030 Agenda for Sustainable Development” (Stanculescu, 2021; United Nations, 2015), providing “universal access to affordable, reliable and sustainable energy” and in particular to reliable electrical energy, is central to human development, social stability and even world peace. Currently, about 2.8 billion people are still lacking modern energy services (International Energy Agency, 2017), and more than one billion do not have access to electricity. Not only the growth of worldwide population, but also the economic growth in developing countries will lead to a continuing increase in the demand for energy.

The tremendous increase in energy generation required for the benefit of the global population has to be accomplished while reducing the rate of GHG (primarily CO₂) emission. Already, “human activities are estimated to have caused approximately 1.0 °C of global warming above pre-industrial levels” according to the Intergovernmental Panel on Climate Change (IPCC, 2018). In order to limit the global temperature increase to around 1.5 °C, total CO₂ emissions need to be reduced to net zero by around 2050.

Germany, for example, had decided to achieve this goal by installing renewable energy generators made of solar photovoltaic panels and wind turbines. Germany also decided to close down its nuclear power plants. As a result, according to Goldstein et al. (2019), “Germany, which went all-in for renewables, has seen little reduction in carbon emissions, and, according to our calculations, at Germany’s rate of adding clean energy relative to gross domestic product, it would take the world more than a century to decarbonize, even if the country wasn’t also retiring nuclear plants early. A few lucky countries with abundant hydroelectricity, like Norway and New Zealand, have decarbonized their electric grids, but their success cannot be scaled up elsewhere; the world’s best hydro sites are already dammed.”

Goldstein et al. (2019) are also saying “we actually have proven models for rapid decarbonization with economic and energy growth: France and Sweden. They decarbonized their grids decades ago and now emit less than a tenth of the world average of carbon dioxide per kilowatt-hour. They remain among the world’s most pleasant places to live and enjoy much cheaper electricity than Germany to boot. They did this with nuclear power. And they did it fast, taking advantage of nuclear power’s intense concentration of energy per unit weight of fuel. France replaced almost all of its fossil-fueled electricity with nuclear power nationwide in just 15 years; Sweden, in about 20 years. In fact, most of the fastest additions of clean electricity historically are countries rolling out nuclear power.”

In the words of World Nuclear Association Director General Agneta Rising in 2019 (Bisconti, 2021): “The 445 nuclear reactors in 30 countries are the low-carbon backbone of electricity systems, operating in the background, day in day out, often out of sight and out of mind, capable of generating an immense amount of clean power. They are the silent giants upon which we rely today.” Table 2 lists the fraction of electricity generated during 2018 in selected countries and globally from energy sources that do not emit greenhouse gases (Pioro et al., 2021). According to Poinssot and Bourg (2021), nuclear energy is within the top three of the most environmental-friendly energy sources; in the same range as hydropower and wind-power. Measured by CO₂ emissions, nuclear energy is even the most environmentally friendly energy source.

It is being suggested (Conca, 2021) that recent research has revealed the potential for extracting practically unlimited supplies of uranium from the oceans at a cost that may come close to that of land mining. If successful, this may make fission nuclear energy practically as renewable as wind and solar (Conca, 2021).

It is not sufficient to decarbonize the electricity generation sector. In the U.S., for example, 33% of CO₂ emissions are attributed to energy generated to support the electricity sector but an additional 36% of emissions result from transportation and 19% result

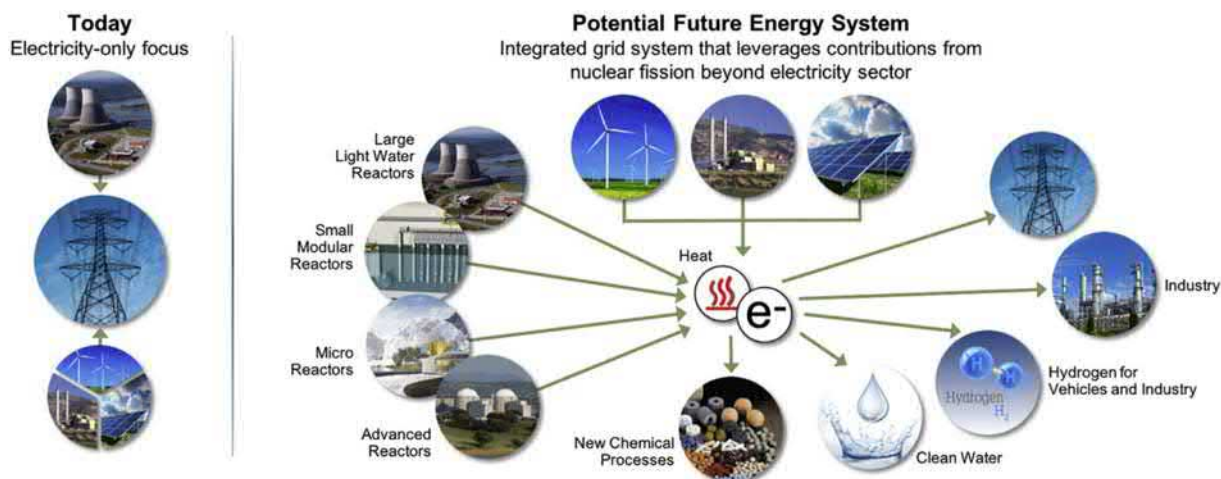


Fig. 2 Conceptualized coordination of energy generation sources and demand to maximize flexibility and economic performance while ensuring grid reliability and resilience.

Most energy experts around the world are advocating the development of a portfolio of low-carbon multi-energy technologies including, in addition to hydro-power plants (with very limited potential for expansion in capacity) and the currently favored “renewable” electricity generators, nuclear power plants. Because the power output of the renewable generators greatly fluctuates over the day and over the year, these generators need to be supplemented by effective energy storage devices that significantly add to the cost of energy.

Nuclear power plants, on the other hand, can generate power at a constant rate throughout the day and the year (excluding a shutdown period of ~1 month for refueling and maintenance once every 18–24 months), practically independent of weather conditions, while requiring a smaller land area per unit energy generated relative to the area required by renewable energy generators.

Moreover, nuclear energy can be generated in the form of high temperature heat (Stanculescu, 2021; Bragg-Sitton and Boardman, 2021) that is the most economical form of energy to store and to use as a backup for the intermittent nature of the renewable energy generators. Therefore, a suitable mix of nuclear and renewables is likely to be the optimal strategy for carbon-free energy generation.

Fig. 1 (Bragg-Sitton and Boardman, 2021) is a simplified illustration of the evolving perception of integrated energy systems using contemporary nuclear power reactors (primarily Light Water Reactors, LWR) in combination with solar and wind renewable energy generators to provide electricity and heat energy for transportation and many industries including non-grid energy users.

Fig. 2 (Bragg-Sitton et al., 2021) is a simplified conceptual perception of the proposed transition away from the use of contemporary large central nuclear power plants from their traditional electricity-only production mode to their exploitation for the provision of a variety of outputs. Future energy systems that seek to maximize energy utilization, generator profitability, and grid reliability and resilience must necessarily consider all available carbon-free energy sources, using each available resource in a way that maximizes its benefits while minimizing less-desirable attributes. The future energy systems should include advanced nuclear reactors including Gen-IV reactors, SMRs and even microreactors and should use distributed in addition to centralized electricity grids.

In summary, the need to meet the objective of zero net CO₂ emission by 2050 combined with the need to greatly increase the total global energy generation rate is an enormous challenge that can be achieved only by making use of all available clean energy generation technologies including nuclear in addition to solar and wind electricity generators. At the 2021 Earth Day climate summit, U.S. president Biden called (WSJ, 2021) for slashing U.S. greenhouse gas emissions by 2030 by 50%–52% compared with the baseline year of 2005, as he seeks to put America at the forefront of world-wide efforts to combat climate change. According to WSJ (2021), meeting this objective would require dramatically reshaping key sectors of the economy, from energy to transportation to agriculture.

Concluding remarks

This Encyclopedia of Nuclear Energy provides a very comprehensive, up-to-date and authoritative information related to nuclear energy, including nuclear radiation—from the discovery of the structure of the atom and its nucleus; through the discovery of three different ways to extract energy from nuclei (fission of heavy nuclei, fusion of light nuclei and radioactive decay); through the state-of-art of the nuclear industry for energy generation and for exploitation of several nuclear technologies for medical, industrial, agricultural and other applications; to the research and development of innovative nuclear technologies that are expected to further

increase mankind benefits from nuclear energy. The reader can find in the Encyclopedia a wide range of topics ranging from the contribution of nuclear energy to the evolution of the continents and oceans of planet Earth to the enabling of very ambitious space explorations including the provision of electricity and heat to human colonies on the Moon, Mars and beyond. The Encyclopedia explains why it is essential to greatly increase the nuclear energy generation rate in order to prevent a global warming catastrophe and, at the same time, provide sufficient energy to the growing world population and to the billions of people in underdeveloped countries.

It is hoped that this Encyclopedia will make an important contribution to transferring of the huge amount of technical knowledge accumulated in the field to the young generation, will inspire students to join research and development of innovative nuclear technologies and will contribute to better understanding of the nuclear technologies, their merits and safety by the general public and policy makers. The appreciation of the enormous benefits in the fields of steady reliable clean energy generation, medicine, agriculture, modern industry, environmental protection, and public safety that mankind enjoys every day and which is described in this Encyclopedia will, hopefully, contribute to much greater public acceptance and support of enhanced utilization of nuclear energy in the mix of clean energy technologies.

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CONTRIBUTORS TO VOLUME 3

David J Ampleford

*Sandia National Laboratories, Albuquerque, NM,
United States*

Anil Kumar Anand

*MICROTROL Sterilisation Services Pvt. Ltd., ATL
Corporate Park, Mumbai, India*

Tom Anklam

*Lawrence Livermore National Laboratory, Livermore,
CA, United States*

Jalal Askander

*Pacific Northwest National Laboratory, Richland, WA,
United States*

Stefano Atzeni

*Dipartimento SBAI, Università degli Studi di Roma "La
Sapienza", Roma, Italy*

Christian Bachmann

*Technical University of Denmark, Lyngby,
Denmark*

Adarsh Bafana

*Energy Systems Division, Argonne National Laboratory,
Lemont, IL, United States*

Gary L Bennett

*Consultant (Formerly with NASA and DOE), Boise, ID,
United States*

Frédéric Bienvenu

*Idaho National Laboratory, Idaho Falls, ID, United
States*

Richard D Boardman

*Idaho National Laboratory, Idaho Falls, ID, United
States*

Lorenzo Virgilio Boccacini

*Karlsruhe Institute of Technology (KIT),
Eggenstein-Leopoldshafen, Germany*

Shannon M Bragg-Sitton

*Idaho National Laboratory (INL), Idaho Falls, ID,
United States*

Jeffrey Brown

*Arizona Public Service (APS), Phoenix, AZ, United
States*

Haydn C Bryan

*Idaho National Laboratory, Idaho Falls, ID, United
States*

David J Campbell

Munich, Germany

E Michael Campbell

*Laboratory for Laser Energetics, University of Rochester,
Rochester, NY, United States*

Gustavo P Canal

*Institute of Physics, University of São Paulo, São Paulo,
Brazil*

JW Coenen

*Forschungszentrum Jülich GmbH, Institut für Energie-
und Klimaforschung, Jülich, Germany; and Department
of Engineering Physics, University of Wisconsin
Madison, Madison, WI, United States*

Valentina Corato

ENEA, C.R. Frascati, Rome, Italy

Christian Day

*Karlsruhe Institute of Technology (KIT), Eggenstein-
Leopoldshafen, Germany*

MR Deinert

*Associate Professor, Department of Mechanical
Engineering, Nuclear Science and Engineering,
Colorado School of Mines, Golden, CO, United States*

Brian D Dold

*Idaho National Laboratory, Idaho Falls, ID, United
States*

AJH Donné

EUROfusion, Garching, Germany; and Dutch Institute for Fundamental Energy Research, Eindhoven, The Netherlands

RJ Dumont

CEA, IRFM, Saint-Paul-lez-Durance, France

Mike Dunne

SLAC National Accelerator Laboratory, Stanford University, Menlo Park, CA, United States

I Duran

Institute of Plasma Physics of the Czech Academy of Sciences, Praha, Czech Republic

Amgad Elgowainy

Energy Systems Division, Argonne National Laboratory, Lemont, IL, United States

Aaron S Epiney

Idaho National Laboratory (INL), Idaho Falls, ID, United States

Gianfranco Federici

Fusion for Energy, Garching, Germany

SM Finnegan

Los Alamos National Laboratory, Los Alamos, NM, United States

Charles W Forsberg

Massachusetts Institute of Technology, Cambridge, MA, United States

Andrew W Foss

Idaho National Laboratory, Idaho Falls, ID, United States

Konor L Frick

Idaho National Laboratory (INL), Idaho Falls, ID, United States

Ricardo MO Galvão

Institute of Physics, University of São Paulo, São Paulo, Brazil

Yu Gasparyan

National Research Nuclear University MEPhI, Moscow, Russian Federation

MR Gilbert

United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, Oxfordshire, United Kingdom

Matthew R Gomez

Sandia National Laboratories, Albuquerque, NM, United States

NN Gorelenkov

Princeton Plasma Physics Laboratory, Princeton, NJ, United States

Cassandra Graham

Beyond NERVA, Dayton, OH, United States

Nils H Haneklaus

Danube University Krems, Krems an der Donau, Austria; and Freiberg University of Mining and Technology, Freiberg, Germany

J Stephen Herring

Center for Space Nuclear Research, Universities Space Research Association, LLC, Idaho Falls, ID, United States

Jamie Holladay

Pacific Northwest National Laboratory, Richland, WA, United States

E Hollmann

University of California—San Diego, La Jolla, CA, United States

Steven D Howe

Howe Industries LLC, Scottsdale, AZ, United States

A Ibarra

CIEMAT, Madrid, Spain

Ryota Imazawa

National Institutes for Quantum and Radiological Science and Technology (QST), Naka Fusion Research Institute, Ibaraki-ken, Japan

Takashi Inoue

National Institutes for Quantum and Radiological Science and Technology (QST), Naka Fusion Research Institute, Ibaraki-ken, Japan

Frank Jenko

Max Planck Institute for Plasma Physics, Garching, Germany

Jeromy Jenks

Pacific Northwest National Laboratory, Richland, WA, United States

Samuel Jimenez

CCFE Culham Science Centre, Abingdon, United Kingdom

L Todd Knighton

Idaho National Laboratory, Idaho Falls, ID, United States

Kenichi Kurihara

National Institutes for Quantum and Radiological Science and Technology (QST), Naka Fusion Research Institute, Ibaraki-ken, Japan

Lang L Lao
General Atomics, San Diego, CA, United States

Shuyun Li
Pacific Northwest National Laboratory, Richland, WA, United States

Ch Linsmeier
Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung, Jülich, Germany

A Litnovsky
Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung, Jülich, Germany; and National Research Nuclear University MEPhI, Moscow, Russian Federation

Xinyu Liu
Energy Systems Division, Argonne National Laboratory, Lemont, IL, United States

YQ Liu
General Atomics, San Diego, CA, United States

So Maruyama
ITER Organization, Saint-Paul-lez-Durance, France

Patrick R McClure
Los Alamos National Laboratory, Los Alamos, NM, United States

Michael G McKellar
University of Idaho, Idaho Falls, ID, United States

Wayne Meier
Lawrence Livermore National Laboratory, Livermore, CA, United States

J Menard
Princeton Plasma Physics Laboratory, Princeton, NJ, United States

Christopher Morrison
Astro Nuclear Engineer, USNC-Tech, Seattle, WA, United States

S Nogami
Department of Quantum Science & Energy Engineering, Graduate School of Engineering, Tohoku University, Sendai, Japan

Jose Manuel Perlado
Instituto Fusión Nuclear “Guillermo Velarde”, Universidad Politecnica Madrid, UPM, Spain

Steven Phillips
Pacific Northwest National Laboratory, Richland, WA, United States

David I Poston
Los Alamos National Laboratory, Los Alamos, NM, United States; and Space Nuclear Power Corporation, Los Alamos, NM, United States

Cristian Rabiti
Idaho National Laboratory (INL), Idaho Falls, ID, United States

Sean P Regan
Laboratory for Laser Energetics, University of Rochester, Rochester, NY, United States

Susana Reyes
SLAC National Accelerator Laboratory, Stanford University, Menlo Park, CA, United States

Gregory A Rochau
Sandia National Laboratories, Albuquerque, NM, United States

Javier Sanz Gozola
Instituto Fusión Nuclear “Guillermo Velarde”, Universidad Politecnica Madrid, UPM, Spain; and Departamento de Ingeniería Energética, Universidad Nacional de Educación a Distancia, UNED, Spain

Yanick Sarazin
CEA, IFRM, Saint-Paul-lez-Durance, France

AE Schweikert
Research Assistant Professor, Department of Mechanical Engineering, Nuclear Science and Engineering, Colorado School of Mines, Golden, CO, United States

Peter A Seidl
Lawrence Berkeley National Laboratory, Berkeley, CA, United States

SE Sharapov
CCFE, Culham Science Centre, Abingdon, United Kingdom

Daniel B Sinars
Sandia National Laboratories, Albuquerque, NM, United States

CH Skinner
Princeton Plasma Physics Laboratory, Princeton, NJ, United States

Stephen A Slutz
Sandia National Laboratories, Albuquerque, NM, United States

Lesley Snowden-Swan
Pacific Northwest National Laboratory, Richland, WA, United States

Hong Yong Sohn

*Department of Materials Science & Engineering,
University of Utah, Salt Lake City, UT, United States*

Pingping Sun

*Energy Systems Division, Argonne National Laboratory,
Lemont, IL, United States*

Sellathurai(Sam) Suppiah

*Canadian Nuclear Laboratories Ltd., Chalk River, ON,
Canada*

Mary Ann Sweeney

*Sandia National Laboratories, Albuquerque, NM,
United States*

Koji Takahashi

*National Institutes for Quantum and Radiological
Science and Technology (QST), Naka Fusion Research
Institute, Ibaraki-ken, Japan*

Neill P Taylor

*UK Atomic Energy Authority, Culham Science Centre,
Abingdon, United Kingdom*

Vladimir T Tikhonchuk

*CELIA, University of Bordeaux, CNRS, CEA, Talence,
France; and ELI-Beamlines, Institute of Physics, Czech
Academy of Sciences, Dolní Břežany, Czech Republic*

Hiroyuki Tobari

*National Institutes for Quantum and Radiological
Science and Technology (QST), Naka Fusion Research
Institute, Ibaraki-ken, Japan*

Michael Tobin

*Johns Hopkins University Applied Physics Laboratory,
Air and Missile Defense Systems, Laurel, MD, United
States*

Alan D Turnbull

General Atomics, San Diego, CA, United States

Susan S Voss

*Global Nuclear Network Analysis, LLC, Taos, NM,
United States*

Mickey Wade

*Oak Ridge National Laboratory, Oak Ridge, TN, United
States*

F Warmer

*Max-Planck Institute for Plasma Physics, Greifswald,
Germany*

Daniel S Wendt

*Idaho National Laboratory, Idaho Falls, ID, United
States*

Hiroshi Yamada

*Graduate School of Frontier Sciences, The University of
Tokyo, Kashiwa, Chiba, Japan*

Jeong-Ha You

*Max Planck Institute for Plasma Physics, Garching,
Germany*

S Zinkle

University of Tennessee, Knoxville, TN, United States

Hartmut Zohm

*Max-Planck-Institute for Plasma Physics, Garching,
Germany*

M Zuin

*Consorzio RFX (CNR, ENEA, INFN, Università di
Padova, Acciaierie Venete SpA), Padova, Italy*

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Introduction to Nonelectric Applications of Nuclear Energy

Shannon M. Bragg-Sitton and Richard D. Boardman, Idaho National Laboratory, Idaho Falls, ID, United States

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Glossary

CHP Combined heat and power.

GHG Greenhouse gas, i.e., gases in the Earth's atmosphere that trap heat, such as carbon dioxide, methane, nitrous oxide, and fluorinated gases.

HTGR High temperature gas reactor.

Hybrid production plant A combination of two power generation or industrial plants that share one or more energy sources or material inputs and produce variable amounts of energy products to optimize revenue or to meet other time varying objectives.

Integrated energy system (IES) An integrated energy system, including hybrid production plants, is connected by physical or cyber connections and operated in a manner that optimizes any or all of the following objectives: system flexibility and efficiency, revenue, resilience, security, resource stewardship, and environmental sustainability. May include two or more collocated or jointly operated subsystems of energy generation, energy storage, or other energy technologies.

LWR Light water reactor.

MSR Molten salt reactor.

Net demand Remaining demand that must be met by conventional dispatchable generation sources after variable generation is subtracted from the total electricity demand.

SFR Sodium fast reactor.

VHTR Very high temperature reactor (gas-cooled).

Nuclear power reactors provide highly reliable, dispatchable energy generation that can support a wide array of energy demands without emission of greenhouse gases. Based on 2018 data, nuclear power plants provide 18% of total energy generation globally and are the largest contributor to low-emissions electricity production in advanced economies ([International Energy Agency, 2019](#)). However, nuclear energy is poised to also provide low-emissions energy generation to support other energy demands beyond the electric grid.

Motivation

New solutions for energy generation, distribution, and utilization are emerging as the world demands a reduced environmental impact—namely, reduced CO₂ and greenhouse gas emissions—while retaining reliability, resilience, and affordability in meeting energy demands ([Arent et al., 2020](#)). These demands span the electricity, industrial, and transportation sectors, and goals for reduced emissions are being reflected in many regional, national, and organizational efforts ([Benahmed et al., 2020](#)). Meeting these goals requires leveraging all clean energy sources available, and it requires that we look to how these sources can be utilized in meeting more than just electricity demand.

In the United States, approximately 34% of CO₂ emissions are attributed to energy generated to support the electricity sector, but an additional 36% of emissions result from transportation and 19% result from industrial processes; the remainder of CO₂ emissions are attributed to residential and commercial use of natural gas and, to a small extent, petroleum resources ([Lawrence Livermore National Laboratory, 2018](#)). Many efforts to decarbonize focus on the electricity sector and look for opportunities to

electrify transportation and industrial processes. However, not all of these processes can or should be electrified. The heat demand across all energy sectors far exceeds electricity use—83% of energy is used in the form of heat, while only 17% is used in the form of electricity. In the industrial sector, as much as 88% of the energy used is heat ([Lawrence Livermore National Laboratory, 2019](#)). Hence, making significant progress toward emissions reduction must necessarily entail the increased utilization of non-emitting or low-emission thermal energy sources, such as nuclear energy, beyond the electricity sector.

Overview of nonelectric applications

Historically, nuclear power plants have primarily focused on the conversion of nuclear-generated thermal energy to electricity to support baseload demands, relying on other generators that can be rapidly ramped to meet peak electricity demand. Although some regions have long demanded flexible output from nuclear plants (e.g., France) due to the relatively large fraction of electricity demand met by nuclear energy, such a flexible operation of traditionally “baseload” plants is becoming more necessary and prevalent as more variable energy sources are added to the grid around the world ([Cox and Bragg-Sitton, 2020](#)).

In 2016, the US Department of Energy national laboratories for nuclear energy (i.e., Idaho National Laboratory) and renewable energy (i.e., National Renewable Energy Laboratory) partnered to complete a study to characterize how emissions could be reduced across the industrial sector via the use of non-emitting thermal energy generation sources, such as nuclear, concentrated solar, and geothermal ([McMillan et al., 2016](#)). Using data from the US Environmental Protection Agency Greenhouse Gas (GHG) Reporting Program, the most significant CO₂ emitters were identified across US industry, comprising 960 facilities across 14 industry sectors in 2014 and representing 25% of industrial GHG emissions for the United States in that year (5% of total US emissions). Typical processes in each of these target industries were defined, and material and energy flows were identified, in order to assess the potential to replace highly emitting resources with zero or low-emission resources. Overall, the study found that nuclear energy technologies, specifically small modular reactors, are expected to be well-matched to the scale of the demand of oil refineries, pulp and paper manufacturing, methanol, and fertilizer plants, among others. Heat recuperation and temperature augmentation were identified as important aspects of thermal energy management within these system designs, particularly for lower temperature thermal generators, and the incorporation of thermal energy storage may support matching clean energy delivery with thermal load schedules, particularly for intermittent or batch plant operations. The latter is also reflected in a recent study on the role of thermal storage in maximizing use of non-emitting energy sources across a variety of applications ([Forsberg and Bragg-Sitton, 2020](#)).

Nuclear energy generated via fission can be directed to meet a number of energy demands in a variety of configurations. These may include configurations such as:

- Electricity production only, via power conversion system (single-input, single-output system), to meet grid demand or to directly support electricity-driven processes
- Heat applications only, such that no power conversion system would be required (single-input, single-output system)
- Cogeneration, producing both heat and electricity (single-input, multioutput system)
- “Integrated” or “hybrid” energy systems, combining nuclear energy generation with other generators (multi-input, multioutput system).

The *Integrated Energy System: 2020 Roadmap* defines integrated energy systems (IES) as such ([Bragg-Sitton et al., 2020a,b,c](#)):

IES are cooperatively-controlled systems that dynamically apportion thermal and/or electrical energy to provide responsive generation to the power grid. They are composed of multiple subsystems, which may or may not be geographically co-located, including a thermal energy generation source (e.g., nuclear), a turbine that converts thermal energy to electricity, additional electricity generation source(s) (e.g., renewable generation, either directly integrated with the nuclear plant or present in the grid balancing area), and one or more industrial processes that utilize heat and/or electricity from the energy sources to produce a commodity-scale product.

The non-grid applications of nuclear energy described in this section may be implemented via one or more of these configuration options, as determined by the specific implementation scenario.

The role of energy storage

Energy storage (e.g., thermal, chemical, mechanical, hydro, electrical) may be coupled with a nuclear generator to support integration with various energy users ([Forsberg, 2021](#)). Incorporation of energy storage components may attenuate the dynamics imposed by one of the connected subsystems, or it may allow energy delivery to coupled energy users at a later time. Electrical storage options primarily focus on batteries, which are increasing in prevalence to support renewable generators. Thermal storage options include both liquid (e.g., molten salt) and solid (e.g., firebrick) forms. Stored heat may be used directly in the coupled industrial process, or it could be used to generate steam that will be fed to the steam turbine. Electrical energy may also be stored in the form of heat (via electric resistance heaters) for conversion back to electricity when it is needed to support thermally driven processes. Chemical energy storage may include the production of intermediate products, such as hydrogen, which can later be combusted or used for the production of electricity. Note that the specific need for and potential benefits of energy storage may differ for each combined heat and power (CHP) or IES architecture.

District heating

One application of thermal energy from a nuclear plant is the direct use of relatively low-temperature heat in district heating loops, as discussed in more detail by [Schweikert and Deinert \(2021\)](#). District heating describes the use of a network to distribute heat from a centralized location through insulated energy transport pathways to support residential and commercial heating requirements. Heat to support such a network is often obtained from a cogeneration plant via the combustion of fossil fuel or biomass, or heat may be derived from geothermal sources, heat pumps, or centralized solar heating. Nuclear-driven district heating systems often operate using waste heat rejected from the secondary side of the nuclear plant in the power conversion process. District heating is most often applied in climate zones having long, cold winter seasons, with heat being provided in the form of hot water or steam at low pressure and at temperatures below 200 °C ([International Atomic Energy Agency, 2019a](#)).

Most nuclear district heating applications are located in Eastern Europe and the Russian Federation, with China bringing a nuclear district heating plant online in 2020 ([World Nuclear News, 2020](#)). As summarized in ([International Atomic Energy Agency, 2019a](#)), 43 reactors around the world have supported district heating as of 2019, providing an average energy withdrawal of less than 5%. Hence, in all of these applications, the primary output has still been electrical power. Heat transport distance is generally limited due to inefficiencies over long distances, with steam transport often limited to ~5 km ([International Atomic Energy Agency, 2012](#)). The most recent nuclear district heating system to come online at China's Haiyang nuclear power plant uses steam from two AP1000 light-water reactor (LWR) reactors, extracted from the secondary side and directed to a multistage heat exchanger to transfer offsite, to heat water for municipal heating. Haiyang plans to heat up to 30 million square meters using the two AP1000 units, potentially providing heating for up to 200 million square meters of buildings and housing within a radius as large as 100 km with the addition of more nuclear units at the site ([International Atomic Energy Agency, 2019a](#)).

Coupling of industrial processes

When an industrial process is coupled within a nuclear cogeneration or integrated energy system, the process receives heat or power from the nuclear reactor(s), the turbine, and other integrated energy sources (e.g., renewable energy source) as needed, or as the power or heat is available. The connected energy user would produce commodity products that provide another income stream to the nuclear generator. Key products may include potable water, hydrogen, synfuels, and chemicals, among others, produced using the heat or electricity.

In a cogeneration or integrated system, when heat from the nuclear reactor is diverted to electricity production, the heat delivered to the industrial process can be reduced, or the heat necessary to operate the process can be derived from another source (e.g., natural-gas-fired unit or stored energy, such as thermal energy storage or stored hydrogen, if the aim is to reduce carbon emissions). When the nuclear-generated heat or electricity replaces fossil-based resources, such a solution can dramatically reduce environmental emissions associated with the industrial process.

Another option is for the nuclear generator to be deployed in a traditional CHP system that may not support the electricity grid as a primary user. In this case, the nuclear reactor would be owned by a given industry or complex of industries to provide a dedicated source of heat and power to the industrial processes. If necessary, the system may occasionally direct electricity to the grid under a contract to provide a reserve capacity when needed. This option is similar to the manner in which many CHP units operate today. The evolution of the public electricity grid to include more variable generation sources may require the more frequent support of grid peak demand, such that industrial CHP units may be contracted to provide spinning or non-spinning reserve capacity or even power modulation to help regulate the grid while also supporting industrial processes.

Most industrial processes demand a constant energy input to support constant operation to ensure economic profitability and optimal performance, although some processes could be designed to operate flexibly if sufficient economic incentives are offered. The ability to ramp many industrial processes is limited due to performance reduction, impacts on economic profitability, and wear or damage on the process equipment. These implications must be considered in process development and design of the coupled CHP or IES.

Recent modeling and simulation efforts have effectively demonstrated the value of flexible operation of LWR plants ([Boardman et al., 2019](#); [Epiney et al., 2019](#); [Frick et al., 2019](#); [Cox and Bragg-Sitton, 2020](#)) to improve economic performance while supporting the grid demand. Flexible LWR operation (i.e., flexible electrical output to the grid while maintaining constant reactor thermal power) allows these plants to support both the industrial and transportation sectors by providing low-cost, low-emissions energy. Potential nuclear plant connections to large US industries are shown in [Fig. 1](#), where the nuclear power plant is the primary source of energy for producing fuels, ammonia, steel, polymers, and hydrogen. In the illustrated system configuration, the production of hydrogen as a process intermediate provides a key energy currency that can effectively incorporate nuclear energy into existing or new industrial applications. There are several processes for producing hydrogen without the CO₂ emissions that accompany the predominant steam methane reforming process, as described by [Suppiah \(2021\)](#) and [Boardman et al. \(2021\)](#).

Introduction of advanced, higher temperature reactors to CHP or IES offers the potential to extract heat at higher temperature (e.g., 500–750 °C) to more efficiently support industrial processes, meeting both their thermal and electricity demands while maintaining environmental stewardship ([Sabharwall et al., 2012](#); [Futterer 2021](#); [Reitsma 2021](#)). Although most of the global research and development (R&D) for advanced reactors focuses on temperatures below 750 °C, note that some R&D associated with the development of very high-temperature reactors (VHTRs) still continues. VHTR research seeks to achieve operating temperatures as high as 900 °C, but these systems require additional development due to significant materials challenges ([International Atomic Energy Agency, 2019b](#)).

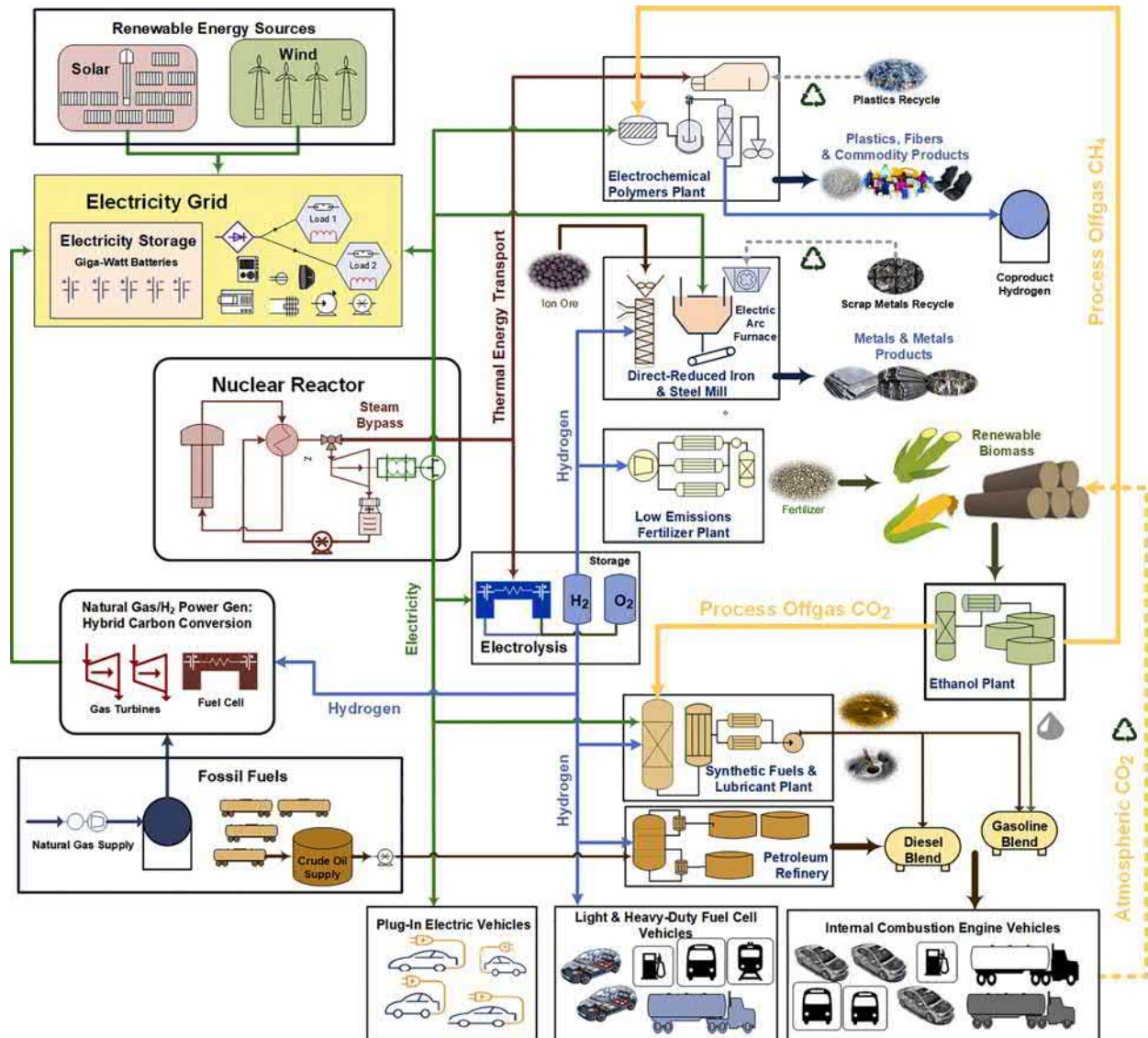


Fig. 1 Illustrative examples of the near-term integration of existing LWRs with various industry applications. From Bragg-Sitton SM, et al. (2020) *Integrated Energy Systems: 2020 Roadmap*. Idaho Falls: Idaho National Laboratory. <https://doi.org/10.2172/1670434>, used with permission.

High priority industrial applications

The *Integrated Energy Systems: 2020 Roadmap* issued by the US DOE Idaho National Laboratory (Bragg-Sitton et al., 2020a,b,c) breaks down energy-intensive manufacturing industry applications based on heat requirement and potential heat transfer media (Pelligrino et al., 2004), see Table 1. Specialized markets, such as pharmaceuticals, that require tight quality control and do not demand a large electrical or thermal input are not listed here.

The US manufacturing industry used about 25 exajoules (i.e., 25×10^{18} J) of delivered energy in 2014, of which approximately 20% is from electricity (with about one-third produced onsite for captive use), 40% from steam (all generated onsite), and 40% from fossil-fired combustion as a source of either direct heating, such as in a cement kiln, or indirect heating, such as in fired-heaters (Ruth et al., 2014). Combustion of fossil fuels currently provides over 90% of the primary energy required for these processes. Hydroelectrical dams that support the aluminum metal production industry and biomass refuse combustion in CHP plants are still the main source of non-fossil energy sources used by the industrial sector.

Alternately, a majority of the industrial steam and heat duty requirements to support the processes summarized in Table 1 could be directly derived from LWRs through temperature augmentation techniques or could be derived by coupling energy use processes with advanced, higher temperature reactors. Steam super-heating with a fossil fuel, chemical heat pumps, or other technologies could be used to amplify LWR steam temperatures to the necessary service temperatures of processes requiring heat in excess of 300 °C (the approximate temperature at the outlet of an LWR) with minimal GHG emission. Use of high-temperature reactors, especially gas- and molten-salt-cooled designs, would reduce the need to augment steam heating. Heat augmentation technologies

Table 1 Development needs for chemical manufacturing process heating and electrification with nuclear energy.

Type of heat duty	Process examples	Process temperature ^a (°C)	Nuclear reactor options	Heat transfer media	Process-related R&D needs
Feedstock Drying & Minerals Concentration	- Biomass - Lumber - Phosphate production - Food dehydration and cooking - Wood pulp production and paper/cardboard plants	50–175	All	Steam or hot gases; phase-change heat storage media	Heat transport networks and other thermal energy storage and delivery approaches
Sub-combustibility	- Biomass torrefaction^b - Thermal plastics forming and modeling	50–300	All		
Evaporation	Ethanol distillation	120–150	All		
Petroleum Distillation	Reboiler heating	550–600	MSR, VHTR	Superheated steam, hot gases, molten salts	Reboiler design; heat circulation systems
Biomass and Coal Pyrolysis	Indirect heating	500–650	MSR, VHTR	Hot gases, molten salts	Reactor heating concepts to support pyrolysis; heat circulation systems; heat augmentation
Combined Heat and Power	Brayton or Rankine power cycles	450–950	SFR, MSR, HTGR, VHTR	Steam or hot gases	Power cycles turbo-machinery
Hydrotreating or Hydrogenation	Fluid catalytic cracker	700–750	MSR, HTGR, VHTR	Hot gases	Reactor heat transfer design and testing
Steam Cracking	- Natural gas reforming - Olefin reforming	800–850	VHTR	Helium	Bottoming heat use (e.g., power generation); reforming process modifications
Oxidative Coupling	Benzene for styrene, etc.	800–850	VHTR	Helium	Bottoming heat use; e.g., power generation
Calcination	Lime and quick lime	800–850	VHTR	Helium	Bottoming heat use; e.g., power generation
Low-Temperature Electrochemical Process Heating	- Formic acid production - Low-temperature electrolysis	25–80	All	Hot oil, pressurized water	Low-cost heat transport manifold and heat recuperation
Intermediate Temperature Electrochemical Process Heating	- Alkane deprotonation - Aqueous CO₂ reduction - Proton-conduction solid-oxide electrolysis	500–600	MSR	Hot inert gases	Low-cost heat transport manifold and heat recuperation
High-Temperature Electrochemical Process Heating	High-Temperature Electrolysis	800–850	All	Ultra-supercritical steam; hot inert gases	

^aNote that low process temperatures can be supported by electric heating versus direct thermal integration.

^bTorrefaction is a thermal process to convert biomass into a coal-like material, which has better fuel characteristics than the original biomass.

From Bragg-Sitton SM, et al. (2020) *Integrated Energy Systems: 2020 Roadmap*. Idaho Falls: Idaho National Laboratory. <https://doi.org/10.2172/1670434>, used with permission.

that could enable utilization of lower temperature reactor technologies (e.g., LWRs) for high-temperature process applications may include resistive heating, high-temperature heat pumps (i.e., having a heat sink temperature > 90 °C, as defined in Herring, 2021), or chemical heat pumps. While a standard heat pump utilizes a vapor-compression refrigeration cycle to increase the temperature of the working fluid, chemical heat pumps utilize reversible chemical reactions for energy storage (endothermic reaction) and release (exothermic reaction) to increase temperature of the working fluid (Sabharwall et al., 2013).

Interface technologies

Integration of non-grid processes with a nuclear plant may be accomplished via direct thermal integration or electrical integration, each of which pose different technical and regulatory challenges.

Electrically integrated processes

Some non-grid energy use applications that require electrical energy input may benefit from power transactions behind the grid (e.g., direct electrical integration with the nuclear plant, outside of the nuclear island but behind the interconnection to the grid). For example, industrial energy users often include batch operations that can be coordinated to smooth the overall power use profile, thereby optimizing the operating efficiency of the nuclear plant. An industrial complex can be designed in a manner

that shares the capital costs across different subsystems, and the complex may also engage in power transactions with the grid to meet peak demands. These arrangements currently are very difficult to negotiate in a regulated electricity market, while they are more common in deregulated markets.

The electrical power system within a nuclear plant supplies power to all systems in the plant on both the primary and secondary side, including safety systems. Hence, the plant's electric power system design must be engineered with every possible contingency in mind. Direct integration with the industrial users will require modifications to a typical switchyard to account for power delivery to the coupled power user (e.g., hydrogen production, desalination) and other unit operations that divert more than 10% of the total capacity of a nuclear reactor in a hybrid system. Power system modeling tools should be employed to support the design of the power distribution and transmission systems to ensure that they retain realistic behavior and to reduce the risk of tripping the nuclear reactor due to interruptions in the power system. Additionally, the incorporation of behind-the-grid electricity users requires the introduction of a new order of communications with the grid operator, which needs to enable the operators to command the nuclear-industry systems functioning behind the grid to effectively use the system for grid services.

Thermally integrated processes

Heat exchangers provide a means to transfer heat to or from fluids of differing pressures, temperatures, and compositions. This is one of the key components needed to thermally connect IES subsystems. As described in Bragg-Sitton et al. (2016), heat exchangers must provide a pressure and chemical boundary and should demonstrate both a high performance efficiency and reliability under dynamic conditions. Advanced heat exchanger design and manufacturing capabilities, such as diffusion bonding or additive manufacturing, may be needed to meet these requirements in a satisfactory manner. Candidate advanced heat exchanger designs should be developed with the support of multiphysics computational modeling methods. It is expected that heat exchanger designs will vary depending on imposed requirements of the coupled subsystems.

Tightly coupled, thermally integrated systems may require a tertiary loop to deliver heat or power for industrial processes. Such a design is not standard to nuclear systems, but this integration approach can provide the isolation of the nonnuclear systems from the nuclear system, ensuring that the coupled process is isolated from potential radioactive contamination under all normal and off-normal operating scenarios. Within integrated energy parks, tertiary loops must be viable for all systems directly attached to it from both a hazards and an operational perspective. These requirements could demand the selection of multiple working fluids depending on the systems being integrated, potentially requiring a range of step-down fluids in the tertiary loops. For example, a gas turbine that operates at 600 °C will require a different tertiary loop fluid than a nuclear power plant that operates at 300 °C. The selected industrial application and its corresponding temperature requirement will also affect the working fluid selection for tertiary loops. Understanding potential fluid interactions between loops will be a vital part of the selection process as well.

Limited research is publicly available on the mass transport of thermal energy via these tertiary loops for medium temperature applications applicable to many IES configurations. Research is being conducted for various medium temperature intermediate loops through several research projects (Stoots et al., 2018; Frick et al., 2020). However, increased investment is needed to support research and development into the possible fluid interactions between loops, loop control systems, amount of holdup required, and siting relative to different thermal generators.

Topics covered in section 8

This section covers a suite of potential applications of nuclear energy to support non-grid energy demands. While the topics covered here are not intended to be comprehensive, they do cover many of the options currently being pursued by research organizations, reactor developers, and industry around the world.

Chapter Summary:

District Heating and Nuclear Power	A. Schweikert and M. Deinert
Process Heat for the Chemical Industry	R. Boardman, M. McKellar, B. Dold, A. Foss, H. Bryan
Nuclear Driven Water Desalination	A. Epiney, K. Frick, C. Rabiti, J. Brown, S. Bragg-Sitton
Coupling Heat Storage to Baseload Nuclear Reactors for Heat, Electricity, and Hydrogen	C. Forsberg
Marine Propulsion	A. Anand and J.S. Herring
Heat Pumps	J.S. Herring
Nuclear-supported Electrification of the Transportation Sector	A. Elgowainy
Thermochemical Hydrogen Production	S. Suppiah
High Temperature Steam Electrolysis	R. Boardman, D. Wendt, C. Rabiti
Synthetic Hydrocarbon Fuels from H ₂ and Captured CO ₂	J. Holladay, L. Snowden-Swan, S. Li, J. Askander, J. Jenks, S. Phillips, L.T. Knighton, D. Wendt
Industrial Processes for Grid Stabilization	F. Bienvenu
Ammonia Synthesis Using Nuclear-produced Hydrogen	R. Boardman
Hydrogen-based Flash Ironmaking Technology (HyFIT): A Novel Green Ironmaking Technology with Low Energy Consumption	H.Y. Sohn
Calcination	N. Haneklaus

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Relevant Websites

- <https://ies.inl.gov>—Integrated Energy Systems Program, U.S. Department of Energy, Office of Nuclear Energy.
- <https://lwrs.inl.gov>—Light Water Reactor Sustainability Program, U.S. Department of Energy, Office of Nuclear Energy.
- <https://nice-future.org/>—Nuclear Innovation: Clean Energy Future, an initiative of the Clean Energy Ministerial.

District Heating and Nuclear Power

A.E. Schweikert^a and M.R. Deinert^b, ^a Research Assistant Professor, Department of Mechanical Engineering, Nuclear Science and Engineering, Colorado School of Mines, Golden, CO, United States; and ^b Associate Professor, Department of Mechanical Engineering, Nuclear Science and Engineering, Colorado School of Mines, Golden, CO, United States

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Overview

District heating systems distribute thermal energy from a centralized source to residential and commercial buildings through a network of pipes to provide space heating and/or hot water. The piping networks convey steam, hot water, or other mixtures and can be located above or below ground. They are often insulated to reduce heat losses during transmission. Most modern systems (“third generation”) have piping networks underground for safety reasons and to reduce thermal losses during conveyance (Mazhar et al., 2018). District heating systems can vary in size and serve a small industrial complex or small residential neighborhood as well as larger, urban systems (Lund et al., 2014; Mazhar et al., 2018). These systems are best suited for locations with dense buildings where losses from longer piping networks can be minimized. Historically, district heating was developed in regions with harsh winter climates including Scandinavia, Eastern Europe and Russia. However, they are increasingly being upgraded and reconsidered globally due to the potential to generate secure, low-emissions heat products when powered by clean and renewable sources (Joint Institute for Strategic Energy Analysis (JISEA), 2020; Lund et al., 2014; Mazhar et al., 2018; Zhivov et al., 2006). In many regions, district heating systems are widely utilized and provide safe, efficient, low-cost heat (World Nuclear News, 2020; Zhivov et al., 2006).

The heat sources for district heating can be dedicated to the purpose, although waste heat from power generation is often used. Cogeneration facilities that are designed to combine heat and power (CHP) production for specific campuses or industrial complexes are an example. When waste heat is used, district heating has the added benefit of reducing carbon emissions that would come from burning conventional fuels to provide heating as well as for reducing thermal pollution from waste heat (discharge of heat to the environment) (Burkhard and Basler, 1976). For nuclear power, district heating can be one of several cogeneration products, alongside desalination, hydrogen production, industrial heat and other products designed to increase the thermal efficiency and revenue of the power plant. The benefits of cogenerating district heating with nuclear power include increased revenue from otherwise unused waste heat, reduced operational environmental impacts from cooling water discharge into lakes or rivers, and increased energy security (International Atomic Energy Agency, 2002; Joint Institute for Strategic Energy Analysis (JISEA), 2020). The most significant challenges related to district heating with nuclear power include the distance between generation source and end use and considerations about the reliability of supply from single-generator facilities (International Atomic Energy Agency, 2002). In this context, the potential for small and micro modular reactors (SMRs) may provide new avenues for clean, reliable and sustainable CHP solutions in many regions of the world (Joint Institute for Strategic Energy Analysis (JISEA), 2020; Värri and Syri, 2019).

History

The earliest implementation of a district heating system is attributed to Lyon, France, where geothermal sources were harnessed to provide hot water to homes (Mazhar et al., 2018). Modern district heating systems have evolved over approximately 150 years, with the earliest “first generation” systems developing in the 1880s in the United States and Europe. These systems conveyed high temperature (> 150 °C) steam through concrete duct transmission systems to dense urban centers, including New York City (United States) and Paris (France). High rates of heat loss and severe accident potential motivated further development (Mazhar et al., 2018). Second generation systems emerged in the 1930s and were characterized by lower (but still considerable) temperatures,

typically ≥ 100 °C, and the introduction of pressurized water as the heat carrier (Lund et al., 2014). Much of the development of district heating systems in the former Soviet Union consisted of second-generation technology, which suffered from high heat losses due to a lack of demand control (Lund et al., 2014).

Third generation systems, sometimes referred to as “Scandinavian” systems, emerged in the 1970s (Lund et al., 2014). Pressurized water was still used for thermal transport, but with temperatures ranging around, and often below, 100 °C. This allowed for less costly piping materials, increased insulation opportunities and potentially reduced risk from damages or leakages of pipes (Zhivov et al., 2006). Third generation systems also included innovations such as prefabricated, pre-insulated pipes made of steel and other materials and buried underground in trenches or tunnels (Mazhar et al., 2018). This included underground pipes insulated with polyurethane foam which could be used with lower temperature water transport (< 140 °C) to reduce thermal losses and lower maintenance requirements (Zhivov et al., 2006). Third generation technology is the current standard for new systems (including those in China, Korea, Europe, the United States and Canada) as well as the target when upgrading older systems (especially prevalent in Central and Eastern Europe and Russia). An overall trend of “generational” development is the lowering of transmission temperatures and upgrades in piping fabrication and materials to reduce costs and heat losses. A summary of generational developments and key attributes can be found in Table 1, summarized from Lund et al. (2014) and Mazhar et al. (2018).

System components

District heating systems can be broken into three main components: generation, transmission, and distribution, Fig. 1. The generation system is comprised of one or more heat producing facilities, with water as the primary heat transfer medium, although some systems use steam or other products. The benefits of using water (when compared to steam) include ease of control, reduced maintenance costs, increased safety, less prone to leakage, and longer piping life (Zhivov et al., 2006). The most common type of generation system is CHP, which can be implemented with conventional fuels but also served by municipal waste combustion, geothermal sources, solar thermal heat, or nuclear power (Mazhar et al., 2018). Attempts have been made to rank generation options for district heating based on economics, environment and energy use to identify the optimal system based on multi-factor considerations. CHP is generally seen as the most optimal choice. However, it is also noted that the optimal solution can vary if only one parameter (e.g., focus exclusively on only the environmental, economics, or energy use parameter) is considered above the others (Wei et al., 2010).

Table 1 Summary of district heating system evolution and key attributes.

	<i>1st Generation</i>	<i>2nd Generation</i>	<i>3rd Generation</i>	<i>4th Generation</i>
Time Period	1880–1930	1930–1980	1980–present	<i>future</i>
Primary region deployed	USA, Europe	Soviet Union	Scandinavia ^a	
Heat Carrier	Steam	Pressurized hot water	Pressurized hot water	Lower-temperature water
Temperature of heat carrier ^b	> 150 °C	100–120 °C	70–120 °C	40–60 °C
Pipes	In situ (concrete duct) steel pipes	In situ (concrete duct) steel pipes	Pre-insulated steel pipes, underground	Pre-insulated flexible pipes
Circulation systems	Steam pressure	Central pumps	Central pumps	Central and decentralized pumps
Substations/Heat Exchanger	None	Tube-and-shell	Heat exchangers with or without plates	<i>not yet determined</i>
Radiators	High-temperature radiators (> 90 °C), steam or water	High-temperature radiator (90 °C) using DH water	Medium-temperature (70 °C) radiators using DH water	Floor heating, lower-temperature (50 °C) radiators
Infrastructure Planning	Accounting for different DH infrastructures	Developing and expanding DH, focus on efficient use of CHP	Identifying and implementing infrastructure, largely fossil-based system	Identifying and implementing fossil-free energy system applications
Primary motivation	Comfort in buildings, reduced risk	Fuel savings, reduced costs	Security of supply, safety	Sustainable energy system, environmental impacts

^a“DH”, district heating; “CHP”, combined heat and power; “°C”, degrees Celsius.

^bLargely developed in Scandinavia, but deployed to upgrade older systems in Eastern Europe and former USSR regions, new systems in China, United States, Europe, Korea and Canada.

^cFrom Mazhar AR, Liu S and Shukla A (2018) A state of art review on the district heating systems. *Renewable and Sustainable Energy Reviews* 96, 420–439. <https://doi.org/10.1016/j.rser.2018.08.005>.

Modified from Lund H, Werner S, Wiltshire R, Svendsen S, Thorsen JE, Hvelplund F and Mathiesen BV (2014) 4th Generation District Heating (4GDH): Integrating smart thermal grids into future sustainable energy systems. *Energy* 68: 1–11. <https://doi.org/10.1016/j.energy.2014.02.089>.

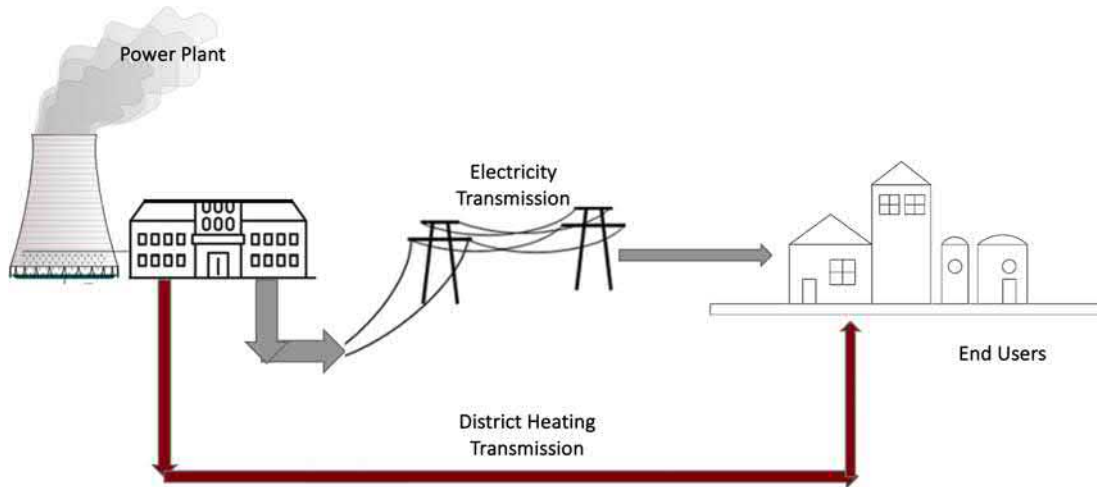


Fig. 1 District heating and electricity cogeneration example.

The transmission network in district heating systems consists of the main transmission lines that connect the generation source to the distribution networks that bring heat to the end users. Costs for transmission networks can be significant and vary widely by geography within (rural, urban) and between countries (Mazhar et al., 2018). Other factors such as pipe diameter, length, material, insulation thickness, and burial depth are also important, and network costs can vary by an order of magnitude or more as a result (Zhivov et al., 2006). Transmission networks are typically made of insulated steel buried in trenches or tunnels, with a lifetime of 20–50 years. They can operate at varying temperatures and pressures, with higher temperature and pressure pipes (16–25 bars) made of steel, copper or aluminum. Lower temperature and pressure pipes can be made of different materials including plastic, potentially lowering costs for construction and operations (Mazhar et al., 2018; Zhivov et al., 2006). Transmission networks also include pumps, control equipment and sensors. Heat losses during transmission are a major source of loss, ranging from 5–20% or higher, especially dependent on insulation of pipes and distance traveled between source and end use.

The distribution network is the third component of a district heating system and is the part of the system directly connected to end users, Fig. 2. There are two types of distribution systems: direct and indirect. Direct systems do not have a decoupling between the transmission line from the heat source and the distribution. This reduces losses and costs associated with heat exchangers. However, direct systems can result in higher costs for control and pressure, and leaks or contamination from within buildings can directly affect the entire district heating system (Mazhar et al., 2018). Indirect systems reduce these problems. With indirect systems, a heat exchanger at a substation is used to decouple fluid in the transmission from the distribution piping networks that take heat to end users (Mazhar et al., 2018). This allows the transmission and distribution systems to operate at different

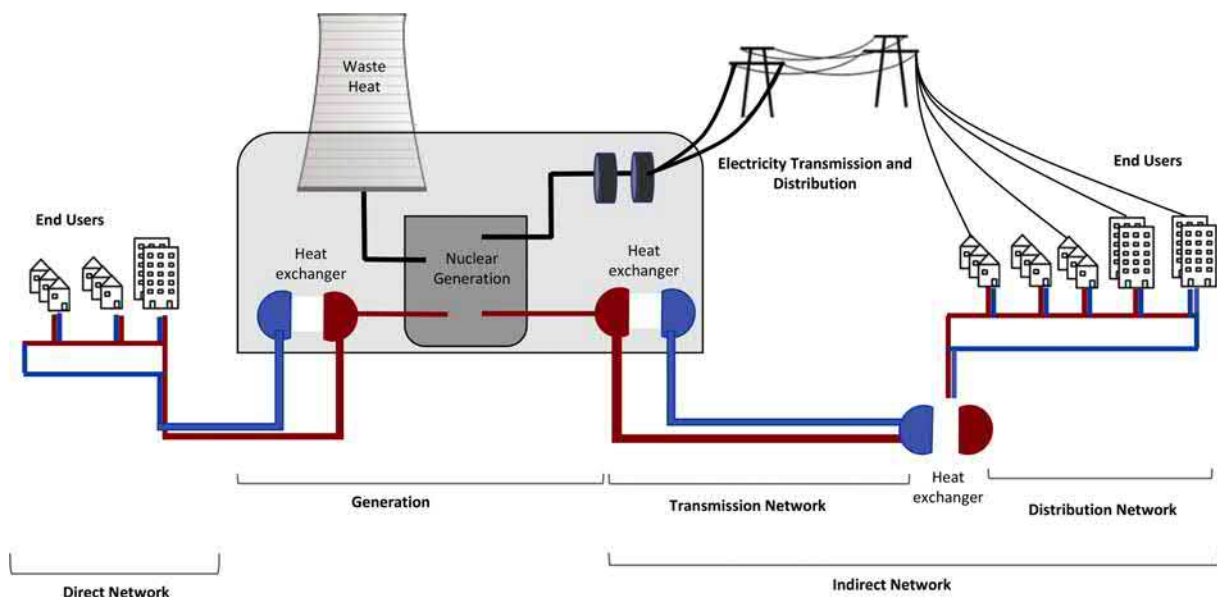


Fig. 2 Simple schematic of combined heat and power (CHP) system including direct and indirect distribution networks.

pressures and helps to isolate any potential sources of contamination. Heat exchangers are typically made of stainless steel, with the most common types being plate heat or shell-and-tube. Shell-and-tube systems allow for better heat transfer but are larger in physical size and less dynamic in load shifting. Heat meters account for the heat use at each end-use destination. This is a key area for emerging research and application of “integrated” and “smart” grids to reduce energy demand and consumption (Lund et al., 2014; Mazhar et al., 2018).

Nuclear power generation

Nuclear power can operate in cogeneration mode or a heat-only mode; all current nuclear power plant concepts are potentially applicable to cogeneration (International Atomic Energy Agency, 2002). Concerns about radioactivity being distributed through a district heating network are eliminated with thermal transfer that occurs in a secondary or tertiary exchange loop that fully decouples the reactor from the transmission and distribution systems (Nuclear Energy Agency, IAEA, 2013). Interest and application of nuclear generation for district heating purposes dates back several decades. Compared with other generation sources, nuclear power provides more energy security than some other fuel generation sources due to fuel density and long cycles of operation between refueling (Joint Institute for Strategic Energy Analysis (JISEA), 2020). This is especially relevant in climates where winter weather can cause disruption to rail or other transportation of fuels such as coal or oil (International Atomic Energy Agency, 2002). Nuclear generation also operates as a clean fuel source, reducing air pollution and greenhouse gas emissions (Joint Institute for Strategic Energy Analysis (JISEA), 2020; Margen, 1978). This was found to be true in Switzerland, which has operated several nuclear power plant-district heating systems, where further benefits include better economic returns when carbon emissions credits are considered and increased political acceptance of nuclear power (Nuclear Energy Agency, IAEA, 2013).

The use of waste heat from nuclear generation can increase the overall thermal efficiency of the plant and provide another revenue stream aside from electricity (Joint Institute for Strategic Energy Analysis (JISEA), 2020). A study of district heating in Helsinki showed that switching the generation source from coal and natural gas to a nearby nuclear power plant could reduce carbon dioxide emissions by up to 4 million tonnes annually. However, the study also showed that a major challenge would be the long distance piping network that would be required to connect the nuclear plant to the Helsinki heating system, presenting an economic constraint (Nuclear Energy Agency, IAEA, 2013).

There are some key challenges related to utilization of a nuclear power plant as a source for district heating. Existing reactors typically produce 500–1500 MW of electricity, plus approximately twice as much waste heat. This thermal output is likely far larger than demands for district heating within proximity of conventional nuclear power plants (International Atomic Energy Agency, 2002). The distance between nuclear power plants and the dense urban centers where residential and commercial district heating is most efficient can require considerable transmission network development (Nuclear Energy Agency, IAEA, 2013). This is noted as the major barrier for implementation in many countries even where district heating is already used for a high percentage of home heating needs (Werner, 2017). This is exemplified in a techno-economic analysis of switching two district heating networks in France to a nuclear generation source. The study found no technical hurdles for implementation. However, it noted that the major challenge to implementing a nuclear-fueled district heating system would require network modifications and a lengthy transmission network to connect generation and distribution components of the system. In some regions, the resulting costs (potentially exceeding 10 million Euros per kilometer) may limit the application of conventional nuclear facilities situated far away from the denser population centers (Jasserand and Devezeaux De Lavergne, 2016). Another challenge for nuclear generation in district heating is the refueling cycle of large plants. Outages may be problematic for generation stations with a single, large reactor and thus require backup generation units that add to the overall costs of operation (Jasserand and Devezeaux De Lavergne, 2016). However, the planned timing for outages could consider seasonal demand and potentially avoid this challenge. A final challenge associated with nuclear reactor operation, especially as heat-only sources, is the licensing and regulatory framework governing nuclear plant operations. In some countries, the regulatory framework for existing nuclear power does not account for heat-only and/or cogeneration options, requiring legislative changes and potential time and monetary costs to account for new operational flexibility (Värrö and Syri, 2019).

A 1987 technical document from the International Atomic Energy Agency (1987) collates experiences from existing district heating applications (nuclear and non-nuclear generation sources) from member states to identify key benefits and considerations for expanding the systems within different regions. Studies and/or operational experience from 10 nations identify the technical data for types of reactors, power ratings, coolant/moderator used, type of design and circuit configurations, temperatures and pressure and other key features in operation or under investigation.

Current state of district heating deployment globally

District heating systems account for approximately 6% of global heat consumption, with China and Russia constituting the majority of use (International Energy Agency, 2019). Renewables account for approximately 3–4% of district heating globally, with Germany, China, Austria and Sweden slightly above this value and Denmark, Russia and Finland closer to 20% of generation from renewable sources (International Energy Agency, 2019). In China and Russia, estimates are that over 200 million persons are

currently supplied by district heating systems (Mazhar et al., 2018). One estimate for the market size of district heating in 2019 was USD\$137 billion in Europe and USD\$173 billion globally (Fortune Business Insights, 2020). This is expected to increase in the latter half of the 2020 decade to over USD\$200 billion, with growth largely driven by desires to reduce emissions (Fortune Business Insights, 2020; Global Market Insights, 2020). The International Energy Agency has a collaborative effort on CHP and district heating and cooling systems, and provides periodic reports on progress in different countries relating to state of research, deployment and systems integration (International Energy Agency, 2021).

District heating is widely deployed in several countries. While not an exhaustive list, Denmark, Finland, Sweden, Germany, Poland, Russia and China have long historical experience with district heating and relatively high penetration of networks that serve heating needs. In Denmark, electricity production is split between small and large-scale CHP providers, with a net production of nearly 20,000 TWh in 2006. Of this, 285 decentralized plants are CHP producers, 16 are centralized (larger) CHP plants and an additional 130 centralized district-heat only plants (International Energy Agency, 2008a). Together, these meet nearly 50% of the heating demand for the country and are estimated (as of 2004) to have saved 8–11 million tonnes of carbon dioxide emissions annually (International Energy Agency, 2008a). The network length is estimated at 29,000 km (Mazhar et al., 2018). Copenhagen has an integrated network that has been serving 270,000 households for many decades, with a transmission network length of 54 km and integration with power systems, natural gas and waste management systems (Mazhar et al., 2018). All cities in Finland have operational district heating systems, serving just under 50% of collective heating requirements, with CHP generation accounting for around 80% (Värrä and Syri, 2019). The Finnish systems produce approximately 13.5 TWh annually from CHP plants, of which approximately 20% is from renewable generation sources (Mazhar et al., 2018). In 2017, Finland's nuclear power energy production was approximately 30% of the total energy produced, while renewables including wind, wood-based fuel and hydropower was just over 50% (International Atomic Energy Agency, 2019a). In Helsinki, more than 90% of the buildings are served through district heating networks, although most CHP plants are powered by higher emitting fossil fuels and a key consideration for reaching climate commitments is the reduction in generation source emissions (Värrä and Syri, 2019). In 2015, Sweden estimated that 55% of heating needs were covered by district heating. Commitments to move towards cleaner generation sources resulted in a reduction in fuel oil for generation to nearly zero use, compared with a 1970 high of almost 80% (Werner, 2017). An early example of district heating fueled by nuclear power was the Ågesta reactor operating south of Stockholm, which provided 10 MW electricity and 70–80 MW thermal heat for district heating use. However, its small size and early deployment led to unfavorable economic returns and it was shut down several decades ago (Märgen, 1978). Germany is the largest energy market in Western Europe and 29% of the final energy is consumed by households, of which almost 90% is used for heating and hot water generation (International Energy Agency, 2008b). District heating is estimated to serve approximately 14% of heating needs in Germany, with production exceeding 70 TWh in 2015 and an estimated network length of approximately 20,000 km (Mazhar et al., 2018). The district heating network in Poland serves approximately 40% of the population through nearly 500 operators, with the majority of generation through CHP. In Warsaw, over 80% of heating demand is supplied through district heating networks, with a production of 11 TWh annually and 1600 km of installed piping (Mazhar et al., 2018). Russia has the largest network globally and one of the longest operational histories for district heating. The estimated network length is 200,000 km carrying over 1700 TWh of heat annually serving 70–80% of the country's heating demand (Mazhar et al., 2018). The Russian network is largely supplied through CHP, primarily natural gas and coal, but has an operational history with nuclear power plant generation dating back many decades (International Atomic Energy Agency, 1987). Extensive operational history, experience and research in district heating, including nuclear power generation, can be found in several publications from the International Atomic Energy Agency (e.g., International Atomic Energy Agency, 1977, 1987, 2002).

Nuclear power and district heating

The current deployment of nuclear reactors for district heating is a small overall percentage of CHP generation, totaling just over 2.1 GWh of equivalent heat in 2018 (Joint Institute for Strategic Energy Analysis (JISEA), 2020). This was supplied by 58 reactors cogenerating or dedicated to district heating purposes (Joint Institute for Strategic Energy Analysis (JISEA), 2020). Considering that nuclear power currently supplies over 1/3 of the world's low carbon energy globally (Joint Institute for Strategic Energy Analysis (JISEA), 2020), interest in "flexible" nuclear power plant operations is increasing. This is evident in the launch of the *Flexible Nuclear Campaign* at the Clean Energy Ministerial's 10th annual meeting in 2019, where integration of cogeneration systems including district heating were highlighted (Joint Institute for Strategic Energy Analysis (JISEA), 2020). In China, the first commercial nuclear heating project came online in November 2020. The Haiyang power plant operates two AP1000 units where non-radioactive steam is extracted from the secondary circuit of the reactors and fed through an on-site heat exchange station, which feeds an off-site circuit. The latter is connected to a heat exchanger which sends heat to a local thermal company (World Nuclear News, 2019a). The nuclear power plant provides approximately 20 TWh of electricity annually and currently provides district heating serving over 700,000 sq. meters of housing, with plans to expand by the end of 2021 to serve up to 30 million sq. meters (World Nuclear News, 2019a, 2020). The operation of this facility is estimated to offset the use of 23,200 t of coal annually, reducing air pollution (estimated soot reduction is 222 t per year) and environmental greenhouse gas emissions (estimated reduction of carbon dioxide is 60,000 t per year) from the use of heating generated from nuclear power (World Nuclear News, 2020).

Considerations for current and future development

Climate and environmental aspects

Increasing concerns around air pollution and greenhouse gas emissions and commitments to clean and sustainable energy systems continue to grow. A recognition that a climate change will impact the energy demand for heating and cooling introduces additional uncertainty into future planning, although climate models can provide some insight into the magnitude and location of selected changes, [Fig. 3](#). Most countries have committed to rapidly decarbonizing their economies, including the energy sector ([United Nations Framework Convention on Climate Change, 2015](#)). A 2015 estimate showed that buildings in the European Union produced 634 million tonnes of non-emissions trading system (non-ETS) greenhouse gas emissions ([European Environment Agency, 2020](#)). Cogeneration systems have been specifically identified as one way to achieve with 10s of TWhs in thermal generation possible from CHP, including nuclear power, if the existing challenges including longer transmission distances could be addressed ([Jasserand and Devezeaux De Lavergne, 2016](#)).

Some evaluations of decarbonizing the district heating sector do not include consideration of nuclear power in future investments (e.g., [Lund et al. 2014](#)). However, increasing global recognition of the role nuclear power must play in “de-risking the pathways to significantly lower emissions” to create more clean, affordable and resilient technology portfolios is ongoing ([Jasserand and Devezeaux De Lavergne, 2016](#); [Joint Institute for Strategic Energy Analysis \(JISEA\), 2020](#); [Parsons, 2021](#)). One promising possibility for district heating with nuclear power generation systems is found in small and micro modular reactors systems (SMRs) ([Ingersoll, 2021](#); [Wallenius, 2021](#)). In Canada, the testing of district heating systems as a cogeneration option for SMRs is ongoing ([Joint Institute for Strategic Energy Analysis \(JISEA\), 2020](#)). Designs capable of district heating are currently available from multiple countries. The South Korea SMART reactor (System-integrated Modular Advanced Reactor) is a 330 MW (thermal) SMR that could be used for district heating. The reactor was certified by Korean Nuclear Safety and Security Commission in July 2012 ([International Atomic Energy Agency, 2019b](#)). In September of 2015, a pre-project engineering agreement was signed with the Kingdom of Saudi Arabia for the deployment of SMART in Saudi Arabia ([International Atomic Energy Agency, 2018](#)). The Akademik Lomonosov reactor, has been installed in Pevek, Russia and became operational at the end of 2019. It is comprised of two 35 MW (electric) reactors of the KLT-40S floating model ([International Atomic Energy Agency, 2021](#)), housed on a non-self-powered barge measuring 144 m long by 30 m wide ([World Nuclear News, 2019b](#)). In addition to electricity, the each of the Akademik Lomonosov reactors provide 150 MW (thermal) of heat, and include additional cogeneration capabilities of supplying fresh water in capacities up to 240,000 L each day ([Golay and Kadak, 2021](#)). The Akademik Lomonosov replaces two existing power plants, including a coal-fired plant, saving an estimated 50,000 tons of carbon dioxide emissions per year ([Boyd, 2019](#)). In the US, the NuScale reactor is in the final stages of licensing and is capable of providing electricity and thermal energy for district heating systems ([International Atomic Energy Agency, 2018](#)). The NuScale modules, 160 MW (thermal) each, are designed to operate in packs of up to 12 modules, allowing for flexible cogeneration options where some modules provide heating and others provide electricity ([NuScale Power, 2021](#)). Additionally, the US Nuclear Regulatory Commission granted NuScale’s system a site boundary emergency planning zone. This feature, which is expected to be a common feature of SMR, would greatly reduce the length of transmission networks required to convey heat between generation source and distribution end use, thus reducing system costs ([NuScale Power, 2020](#)).

Other aspects

The continued development and greening of the energy sector includes upgrading existing district heating systems and developing new, integrated systems. This is a multi-faceted, challenging and complex issue. As new systems are developed, an emphasis on systems that integrate heating and cooling are being developed as part of a more sustainable “fourth generation” of district heating often called “cold district heating” ([Lund et al., 2014](#)). In an integrated system, cold and hot water are moved between buildings depending on individual needs with heat pumps moving thermal energy into or out of a building. Such systems could allow for better year-round demand from CHP plants and build on expertise developed in district heating ([International Atomic Energy Agency, 2002](#); [Werner, 2017](#)). The consideration of cooling may be especially important as warmer regions of the world, where energy poverty and electrification are major focus areas of development efforts, continue to industrialize and where demand may increase significantly as electrification is increased. [Fig. 3A](#) and [B](#) show current global estimates for heating degree days (HDD) and cooling degree days (CDD), a measure of building energy use that calculates the difference between ambient and indoor air temperatures (e.g., [Moustris et al., 2015](#)). [Fig. 3C](#) and [D](#) show the estimated increase in HDD and CDD globally, using the 2050 decade and a median climate model ([NASA Center for Climate Simulation, 2019](#)) (Model represents the median model from the CMIP5 RCP4.5 ensemble, based on average heating and cooling degree days in Paris, France in the year 2050. Data from authors’ calculations.). These figures show that both cooling and heating have significant demand globally, although they vary spatially. Looking out to the 2050 decade, very little (if any) increase is expected in heating degree days globally for the climate scenario shown. By contrast, cooling degree days are expected to increase 25–50% throughout most of the tropics, a region with already underserved populations for energy demand in many locations.

A challenge to the deployment and upgrading of existing systems, especially for nuclear power, is the regulatory frameworks within different countries that govern their licensing, inspections and operations. For countries with existing nuclear infrastructure, the regulatory framework might not currently cover operations for heat production, an issue identified in France ([Jasserand and Devezeaux De Lavergne, 2016](#)). For countries without any current nuclear regulatory infrastructure, the need to develop this alongside the technology presents additional considerations.

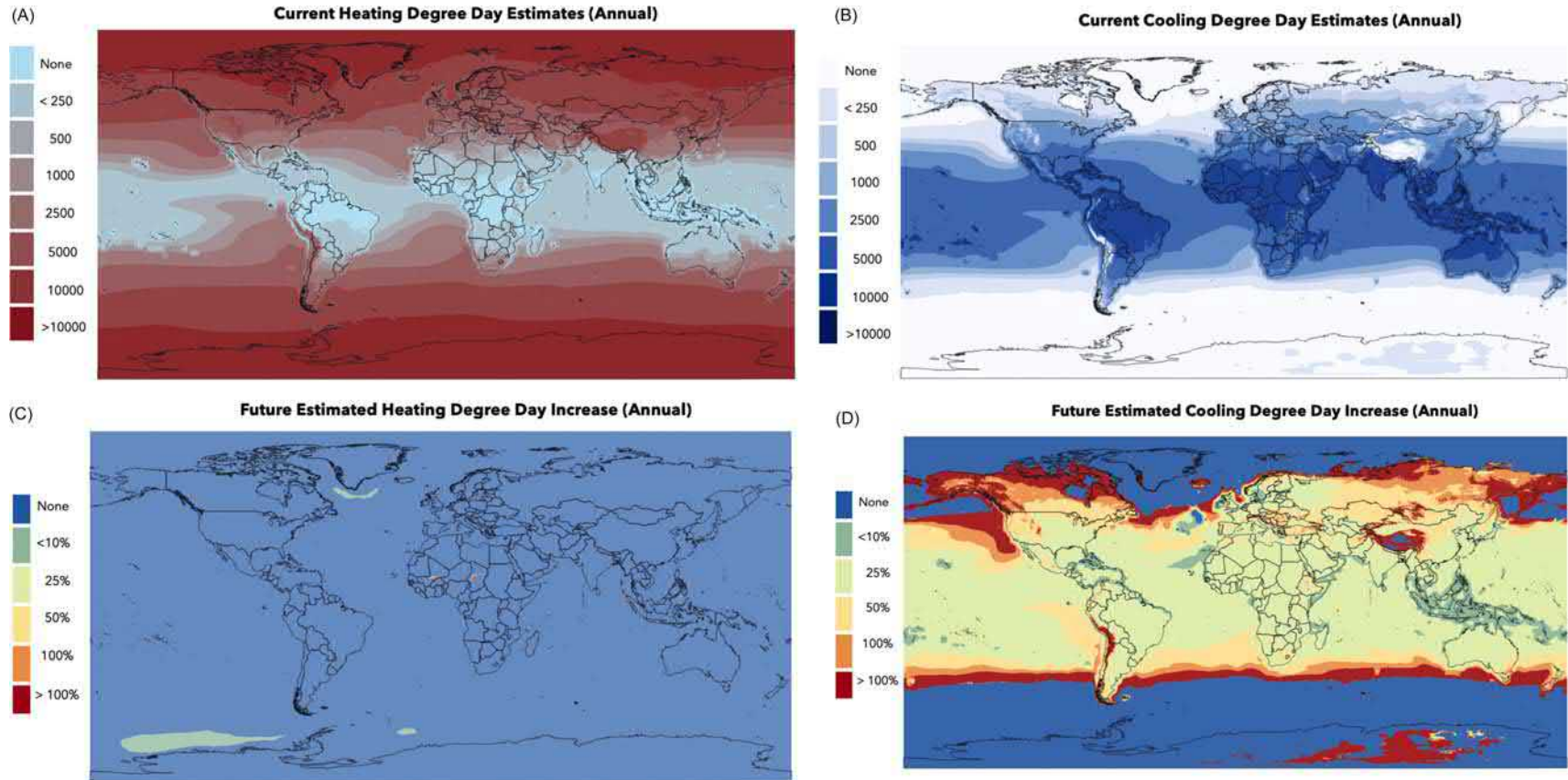


Fig. 3 Estimated annual heating and cooling degree days in the current climate (A and B) and estimated future changes (C and D). Future changes are in percentage relative to the current climate. Future estimates are based on a median climate scenario from the CMIP5 RCP4.5 data and calculated by the authors.

Finally, broader reasons outside pure economic and environmental impacts may incentivize the consideration of new or expanded district heating networks, especially those powered by clean fuels including nuclear power. Energy poverty is very high in some developing regions of the world and electrification investments are ongoing, and many of these regions are expected to continue to increase in demand, especially for cooling (see Fig. 3D) (Bhatia and Angelou, 2015). However, energy poverty also exists within more developed economies. A particularly high burden for many households is the resources required for fuel (Bednar and Reames, 2020). For example, in the United Kingdom, one estimate puts fuel poverty at 11% (Mazhar et al., 2018). In the US, 2015 estimates showed that 25 million households (about 20% of total in US) had to forgo food and medicine to pay energy bills (Bednar and Reames, 2020). These issues may be exacerbated by changing climates, with energy being an integral part of the complex nexus of poverty and vulnerable populations (Jessel et al., 2019).

Summary

District heating is widely used in many regions of the world and can be produced by heat-only generators or as a by-product of existing power plants through cogeneration operations. Cogeneration from nuclear power presents both challenges and opportunities, including the benefits from clean, reliable energy production as well as the costs associated with the transmission of heat from large, conventional facilities located tens or hundreds of kilometers away from dense urban centers where district heating is most widely used. However, there are instances where nuclear-powered district heating has been and is currently deployed, and as concerns around greenhouse gas emissions continue to take a priority, nuclear power generation is being considered in many new locations. Examples of this, specific to district heating, include the Haiyang Chinese power plant (very large) and the Akademik Lomonosov SMR in Russia (total design of 70 MW (electric)). In emerging fourth generation district heating designs, an integrated systems deployment including consideration of cooling needs, electricity and other operations can result in more optimized energy delivery and reduction in greenhouse gas emissions (Lund et al., 2014). The potential of SMR and MMR technology is particularly relevant in the planning of integrated systems. This is due to the smaller, more flexible units that include additional safety considerations, standardized designs, and siting parameters that allow for use nearer to populations and flexibility for deployment at varying sizes. These parameters make SMR technology more applicable in many areas that currently experience energy poverty and may play a role in both replacement and new power plant builds.

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Nuclear Driven Water Desalination

Aaron S. Epiney^a, Konor L. Frick^a, Cristian Rabiti^a, Jeffrey Brown^b, and Shannon M. Bragg-Sitton^a, ^a Idaho National Laboratory (INL), Idaho Falls, ID, United States; and ^b Arizona Public Service (APS), Phoenix, AZ, United States

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Abbreviations

AF Acre foot
APS Arizona Public Service
IES Integrated Energy System
INL Idaho National Laboratory
kWe-h Kilowatt hours electric
kWt-h Kilowatt hours thermal
LCOPW Levelized Cost of Potable Water
m³ Cubic meters
MED Multiple Effect Distillation
MSF Multistage Flash Desalination
MW Megawatt
MWh Megawatt hours
NPP Nuclear Power Plant
O&M Operations and Maintenance
PVGS Palo Verde Generating Station
RO Reverse Osmosis
VC Vacuum Compression Desalination
WRF Water Reclamation Facility
WWTP Waste Water Treatment Plant

Introduction

Sustained growth of renewable energy generation in the U.S. has imposed economic challenges on many large, traditionally baseload-generation plants. Many plants within the domestic nuclear fleet have been or will be economically challenged when these plants are required to operate at less than full capacity either seasonally or daily when sufficient renewable resources are available to support grid demand. As renewable output decreases, however, nuclear plants are critical to ensuring that grid demand is met. Several utilities and operators of these nuclear plants in the U.S. are investing in integrated energy systems (IES) that can supply dispatchable loads to integrated energy users to maintain full output during low demand times; supply to non-grid energy users can then be reduced when power is needed on the grid.

At the same time, availability of potable water is already a limiting factor in local population growth and wealth for several areas of the U.S. and worldwide. Several major reports from large international organizations (IAEA, 2007) identify the availability of fresh potable water as a major challenge for future growth.

The IES paradigm shift recasts those two challenges as an opportunity. An integrated regional desalination process provides for additional or dispatchable loads (up to 100 Megawatt (MW) electric) that may help avoid nuclear energy generation curtailment during the day while producing potable water. Later, Nuclear Power Plants (NPPs) can reduce output of the coupled desalination plant to facilitate greater power output to the grid. This integrated process provides valuable grid capacity along with large-scale production of potable water.

As a specific example for U.S. markets, the municipalities west of Phoenix, Arizona, are presently experiencing challenges to sustaining growth because of limited water resources. As groundwater and surface water resources are stressed, these municipalities are now focusing on other, lower quality brackish groundwater resources that are available and sustainable in large supply. Use of these resources is dependent on cost-effective desalination and management of the process concentrate or reject stream.

There are limited options for the management of the treatment process concentrate from a large-scale municipal water treatment facility. Deep well injection in the shallow Phoenix sub-basin is improbable, plus the injection process currently lacks the required authority in Arizona. Surface evaporation ponds are very expensive and require significant surface area for the large-scale municipal projects. These options are cost-prohibitive and provide an incentive to consider other, more cost-effective and innovative options. The Palo Verde Generating Station (PVGS) is located nearby and could benefit from the dispatchable load of a regional reverse osmosis (RO) water treatment facility, and can process the regional system concentrate. Selection of the most cost-effective technology is needed to optimize design and maximize economic value.

The Arizona Public Service (APS) utility, which serves as the operating owner of PVGS, which comprises three nuclear units, could benefit from the introduction of a large load-modulating resource such as a desalination plant. In particular, this would allow increased penetration of variable renewable energy, especially solar photovoltaic generators, while preserving the value of PVSG as a large baseload, carbon-free asset on the regional electric grid.

Desalination technologies overview

There are several processes that have been demonstrated for producing clean water from saline-rich water (either brackish or salt-water). Still, global experience with desalination plants coupled to nuclear reactors is dominated by three primary processes: two distillation-based technologies (multiple-effect distillation (MED) and multi-stage flash (MSF)) and one membrane-based technology, reverse osmosis (RO) (International Atomic Energy Agency, 2000) as shown in Table 1 (Al-Othman et al., 2019). In fact, according to a recent IAEA publication on nuclear cogeneration (International Atomic Energy Agency, 2017), the global use of nuclear energy for desalination applications has accumulated over 250 years of reactor operation, when tallied across all nuclear-desalination units. Other purification technologies like Vacuum Membrane Distillation, Vacuum Compression Desalination (VC), Forward Osmosis, and Electrocoagulation have potential long-term opportunities, but are not presently at scale relative to the process rates considered.

MED consists of multiple stages, each at descending pressure. Saline water in each stage is heated to vaporization. Vapor from each stage is condensed in the next successive stage, thereby giving up its heat to drive more evaporation. Seawater is then sprayed over these hot tubes to evaporate the water. The vapor created from condensing the previous effects vapor is then streamed to the next effect. To avoid mixing the boiler chemicals with the pure distillate, the distillate from the first effect does not join the main distillate stream. The brine is collected at the base of each effect, which is either circulated to the next effect or transported out of the system, as illustrated in Fig. 1 (Miller, 2003). The use of MED fell from favor in the 1950s due to scaling on the tubes which caused thermal degradation. However, in recent years MED has been making a resurgence as corrosion and scaling science have matured (Saadat et al., 2018).

Desalination using MSF is based on heating seawater in a brine heater to around 100 °C. The hot brine then enters a flash chamber, which is held at vacuum. Since the water entering the chamber is above the boiling temperature at vacuum, part of the water flashes to steam. This steam rises to condensing coils where it condenses into fresh water. A diagram of this process is shown in Fig. 2.

MSF is a proven technology for water desalination. It is a simple and extremely reliable process that requires no moving parts other than pumps. However, MSF is more energy intensive than current RO plants. MSF also requires a direct steam connection to

Table 1 Nuclear desalination plants across the world.

Country	NPP type	Desalination technology	Water capacity (m ³ /day)	Status
India	PHWR (Kalpakkam 1,2)	Hybrid MSF/RO	6300	Operational
Japan	PWR (Ohi 1,2)	MED	(2 × 1300)	Operational
		MSF	(1 × 1300)	
		RO	(2 × 1300)	Operational
	PWR (Ohi 3,4)	MSF	(2 × 1000)	Operational
	PWR (Ikata 1,2)	RO	(2 × 1000)	Operational
	PWR (Ikata 3,4)	MED	(1 × 1000)	Operational
	PWR (Ginkai 3,4)	RO	(1 × 1000)	
	PWR (Takahama 3,4)	MED/VC	(2 × 1000)	Operational
	BWR (Kashiwazaki)	MSF	(1 × 1000)	Not used
Pakistan	PHWR (KANUPP)	MED	1600	Operational
United States	PWR (Diablo Canyon 1,2)	2 Stage RO	2200	Operational

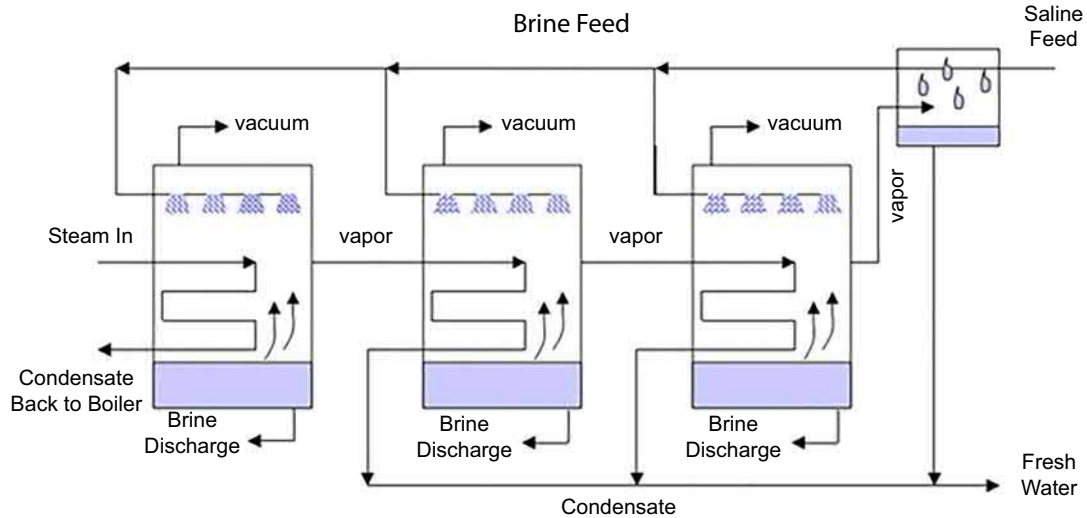


Fig. 1 Example of a Multiple Effect Distillation Unit. From Miller JE (2003) "Review of Water Resources and Desalination Technologies". Sandia National Labs Unlimited Release Report SAND2003-0800.

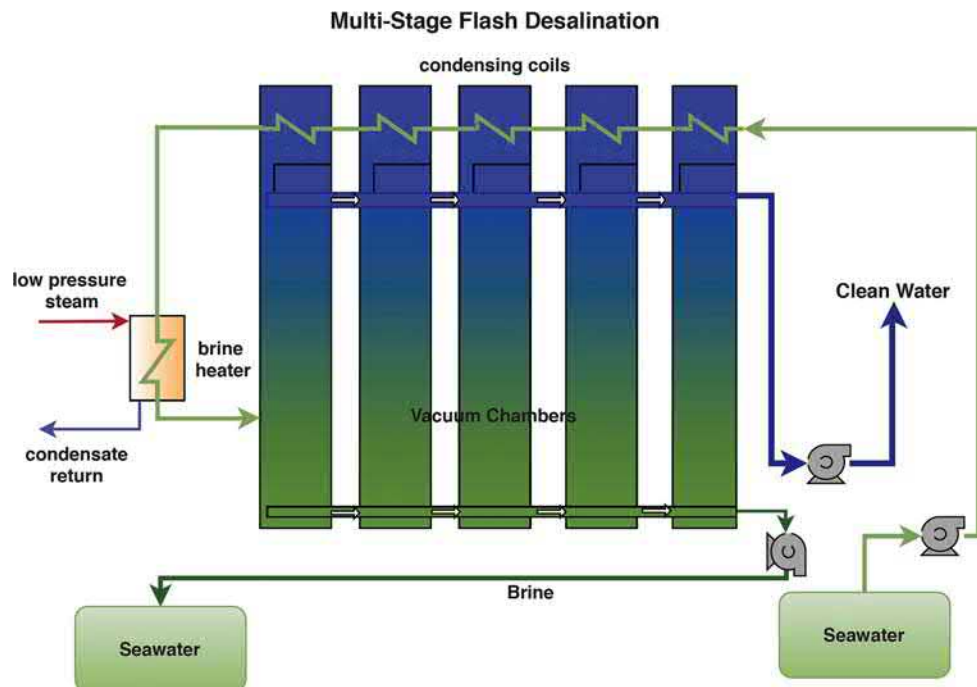


Fig. 2 Example of a multistage flash desalination plant.

the power plant, and typically cannot operate at less than full capacity due to its system design and to limit potential system instabilities (Darwish and Al-Najem, 2000). These challenges limit MSF's applicability to integrated energy systems where a non-uniform load would be sent to the desalination process as demand fluctuations occur on the grid.

The main competitor to multistage flash desalination is RO. The RO process works by passing water at high pressures through fine membranes that only allow water molecules to pass. A typical RO plant works in two stages. First is a pretreatment stage where chlorine and other chemical additives are used to remove biological organisms and control both the pH and hardness of the water, i.e., the amount of dissolved solids in the water such as magnesium, calcium and others. The water is then sent to the membrane filtration system where high pressure forces the water molecules through the membranes into an inner collection tube. A representative diagram of an RO system is shown in Fig. 3.

Reverse Osmosis has the benefit of consuming less energy than MSF. Also, RO does not need a direct steam connection to the power plant as it only requires electric energy to run pumps and has simple start/stop operating capability (Darwish and Al-Najem,

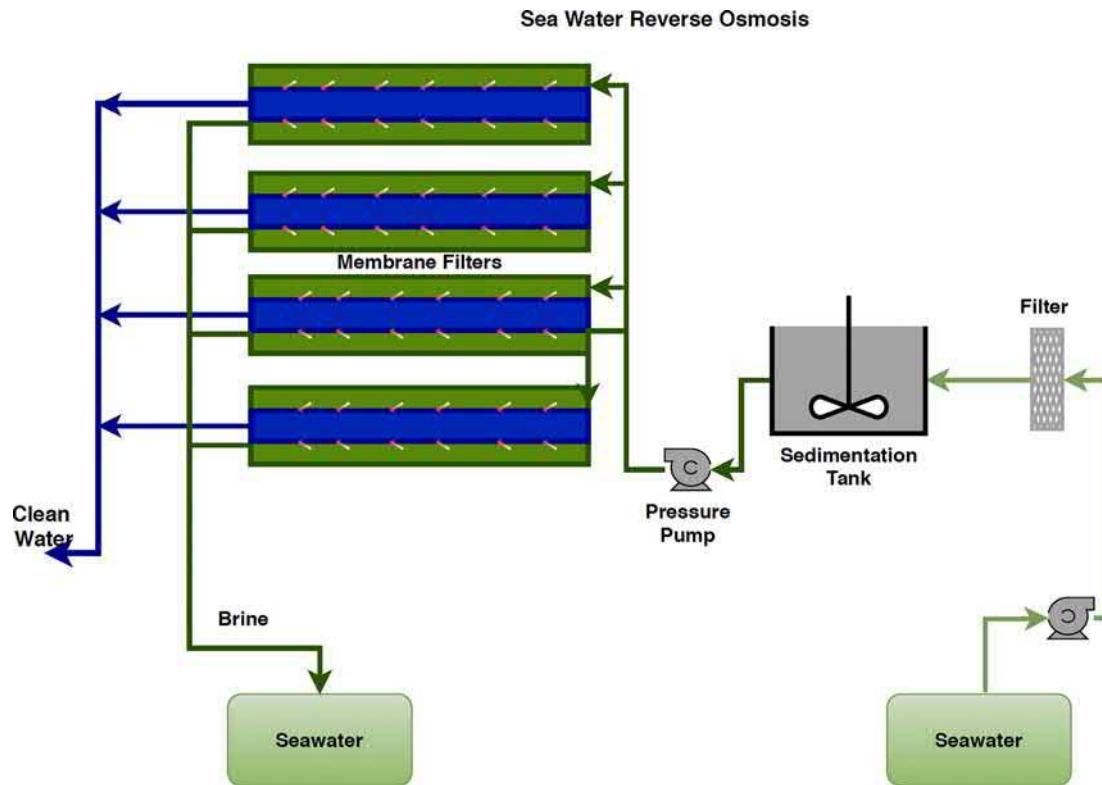


Fig. 3 Example of a Reverse Osmosis (RO) plant.

2000). Since RO is operated in modules, operation can be staged to take advantage of available excess energy instead of the continuous full operation of MSF. RO processes also consume less energy than MSF.

The choice of desalination method(s) is determined primarily by the characteristics of the source water and the water quality required by the end user. For example, RO technology typically has a lower capital cost but is less effective with feedwater that contains high levels of organic materials that can foul the membranes or that have high salinity levels and can only produce potable water without further treatment. The two thermal distillation processes are much more tolerant of “dirty” or “salty” feedwater and produce high-purity water.

Connection and control with NPPs

A key distinction in the two desalination methods, namely distillation and membrane-based processes, is how they couple with a power source. The RO plant has the most straightforward coupling since it operates using only electricity, which is needed to run the high-pressure pumps. Therefore, it is not essential to co-locate the desalination plant with the power plant so long as a grid connection is available. However, there may be an advantage for co-location of the power and RO desalination plant in terms of shared infrastructure and protection against grid disruption. Also, low grade steam or warm wastewater from the power plant can be used to preheat the saline feedwater of the RO plant to improve its clean water production efficiency, although the quality of the distillate may be adversely impacted.

Both MED and MSF plants require a thermal heat source, such as a steam line from the secondary side of the NPP. For NPPs this steam would typically be extracted either by the high-pressure or low-pressure tap on the turbine. Since energy quality requirements for MSF and MED are low, the heat could be extracted from a low-pressure turbine stage. For space-constrained power plants or steam cycles that preclude such integration, it is possible to take the higher quality steam from the steam generator through an existing bypass valve and use it for distillation.

Economics

Several factors are needed to understand the potential cost of potable water produced through a coupled nuclear-desalination process (Ingersoll et al., 2014): (1) energy requirements of the desalination processes both thermally and electrically, (2) the

Table 2 Specific energy consumption of desalination plants.

Process	Specific heat consumption (kWt-h/m ³)	Specific electricity consumption (kWe-h/m ³)
MSF	100	3
MED	50	2–3
RO	0	4.5

International Atomic Energy Agency (2007) *Economics of Nuclear Desalination: New Developments and Site Specific Studies. Final Results of a Coordinated Research Project 2002–2006.*

cost of that energy, and (3) the cost of the integration between the nuclear plant and the desalination process. Energy consumption for each of the main three desalination processes are summarized in [Table 2](#).

The cost of the energy required to produce the water is market specific, as it can be considered the maximum of the opportunity cost of the NPP not producing electricity to the grid or the operational and maintenance cost of the nuclear plant, as expressed below.

$$Energy_{Cost} = \max(\text{market}_{price}, O\&M_{NPP})$$

This expression holds because the plant operator always aims to maximize profit. If more profit can be created by selling to the grid, then the desalination plant will have to match this price to purchase the power. Alternatively, if the market price is below cost of operations and maintenance (O&M) of the plant, then the plant operator would decide to shut down the plant rather than sell to the market. Thus, the desalination plant would need to purchase power at the O&M cost. While there are scenarios in which this is not always the case (depending on market structure and ownership of the plants), it is sufficient to say that even with this simplified formulation it is difficult to specify a precise energy cost.

The cost structure is also impacted by the physical integration approach. For a generic off-the-shelf RO plant, integration is electrical only, thus requiring minimal expenditures even if the RO plant is integrated with the NPP behind the NPP connection to the grid (the RO plant could also be grid-connected). If efficiency needs to be boosted, feedwater pretreatment can be added if NPP waste heat is already heading to the condenser. Integration of MSF and MED is far more involved. These processes require a tap off the high-pressure, medium-pressure, or low-pressure turbine along with associated control schemes.

According to a 2007 IAEA report, economics of nuclear-desalination is site specific and is dependent on several factors and the economic assumptions used. However, the case studies have shown that, in general, nuclear desalination costs can vary from 0.5 to 0.94 \$/m³ for RO, from 0.6 to 0.96 \$/m³ for MED and from 1.18 to 1.48 \$/m³ for MSF plants (IAEA, 2007). Further studies have shown that RO technology, given the superior efficiency, is generally more cost effective than MSF or MED.

Providing regional potable water in Arizona

An example is presented here to illustrate how case dependent benefits and peculiarities—such as cooling water production—can play into the economics of nuclear driven desalination. A techno-economic study conducted in 2018–19 (Epiney et al., 2018, 2019) analyzed the economic viability of an Integrated Energy System (IES) that couples an RO water desalination facility with an NPP in the Southwest region of the U.S. The case study was conducted by Idaho National Laboratory (INL) in collaboration with APS, which is the operating owner of the PVGS NPP.

Palo Verde is the largest nuclear single site facility in the U.S., generating on average 32,000,000 Megawatt hours (MWh) electric annually and providing significant regional grid stability. However, sustained growth of renewable generation in this region has resulted in seasonal over-generation and consequential frequent low-power pricing. Palo Verde is also the only nuclear facility in the world not located on or near a natural body of water. Cooling water for the reactor steam cycle is derived from municipal wastewater of the Phoenix area (effluent). The wastewater is pre-treated in treatment plants near the municipalities and is conveyed to an additional onsite tertiary treatment facility at Palo Verde that further treats the water to plant chemistry standards. The cost of the municipal effluent escalates annually and represents an increasing fractional cost of generation.

Efforts to reduce cooling costs have identified alternative water sources to replace the increasingly expensive effluent. Brackish water from a regional aquifer is available in large supply and is sustainable. The quantity of brackish groundwater that could be used for plant cooling is limited because of high salinity and the potential of scaling in system components. The Palo Verde tertiary treatment facility can treat the increased hardness that would result from a blend of the groundwater with the effluent. However, several constituents such as chlorides are not affected by the treatment process and, in sufficient quantity, would result in levels that exceed the operational limits in the PVGS circulating water system and evaporative cooling towers. Consequently, supplemental treatment would be required to specifically address these constituents such that a greater amount of brackish groundwater could be used to reduce the demand on effluent.

Evaluations of supplemental treatment options suggest that a higher concentrated blend of effluent and brackish water could be processed while maintaining the Palo Verde water balance when only a fraction of the total tertiary treated water mixture undergoes the supplemental treatment. Moreover, the cooling water loop chemistry at PVGS allows for the supplemental treatment system to



Fig. 4 Phoenix west valley region including PVGS, the municipalities of Buckeye, Goodyear, Avondale, Tolleson and Phoenix downtown. Google Maps.

be sized such that it could also support concentrate management for a regional desalination plant. The desalination plant could process brackish water in the regions and provide significant value to the Phoenix west valley municipalities that are currently growth-challenged due to the limited availability of potable water (see Fig. 4).

The INL study investigated the economics of blending a fraction of the regional brackish water to offset a corresponding amount of effluent. Due to constraints that limit concentration of select water constituents in the Palo Verde cooling water, the economics continue to favor the use of effluent. Consequently, the INL assessment considered the integration of the regional RO facility and a supplemental treatment process at Palo Verde to manage the regional concentrate to provide an alternative water supply to the surrounding municipalities. The economic assessment considered the market value of the potable water and the benefit of concentrate management in this area. Options are limited and not economically viable considering that deep well injection is not authorized and surface evaporation ponds require large tracts of land and have high capital cost.

The investigated case considers building a regional RO desalination plant (designated RO2) near the regional waterlogged area where the brackish water reserves exist in the shallow aquifer. Concentrate from the regional RO (designated RO2) desalination plant could be transported in the pipeline that conveys the municipal effluent from the municipal wastewater treatment plant to Palo Verde. The supplemental treatment process onsite (designated RO1) would ensure that the Palo Verde water balance is maintained. This case is shown in Fig. 5.

The case study has three main degrees of freedom:

- The amount of brackish water sent into the regional RO (i.e., RO2 size).
- The percentage of Water Reclamation Facility (WRF) outlet water treated by RO1 (slip stream designated as α) (i.e., RO1 size).
- The amount of brackish water directly injected into the pipeline between Waste Water Treatment Plant (WWTP) and WRF.

Ranges for the degrees of freedom have been investigated to show impact on the case study economics as well as to find optimal (most profitable) combinations of RO sizes. Fig. 6 shows the Levelized Cost of Potable Water (LCOPW) (color map) as a function of RO1 and RO2 size for 0 m³/year (0 AF/year) and 1.9e7 m³/year (15,500 AF/year) of brackish water directly injected in the effluent water supply line.

In general, the study shows that there exist combinations of regional RO and PVGS RO sizes that satisfy total blowdown and water chemistry limits at PVGS (including regional RO concentrate treatment at PVGS either through increased blowdown or onsite RO treatment). However, possible combinations only exist if no additional brackish water is directly mixed into the tertiary system at PVGS. The lowest LCOPW (~ 0.5 \$/m³) can be achieved with a size of the regional RO of $\sim 1.4e7$ m³/year (11,000 AF/year) capacity, which leads to $\sim 8.63e6$ m³/year (7000 AF/year) of potable water while no onsite RO at PVGS is built.

The maximum-size regional RO that respects the physical boundary conditions can be achieved when no additional brackish water is directly mixed in the tertiary system at PVGS, and a PVGS onsite RO is built that treats $\sim 2.3\%$ of the effluent and regional RO concentrate blend. The corresponding regional RO size is $1.5e7$ m³/year (12,500 AF/year), which is only slightly bigger than the regional RO with the lowest LCOPW (within the physical boundaries). However, the LCOPW for that biggest regional RO is slightly less favorable compared to the case where no RO1 is built. A RO of that size consumes 6.75e4 MWh-e/year.

It is worth noting that for a given total blowdown limit (cooling tower and onsite RO1), the maximum salinity in the reservoir can be higher (i.e., a larger RO2 can be built) when the water is run through the cooling tower (with fewer cycles) and blown down to the evaporation ponds, compared to desalination with an onsite RO. This is because the reduction in blowdown from the circulating water system by having an onsite RO (lower salinity water in cooling loop) is lower compared to the concentrate produced from the onsite RO1. In other words, for a given total blowdown limit, the circulating water system blowdown is more efficient in

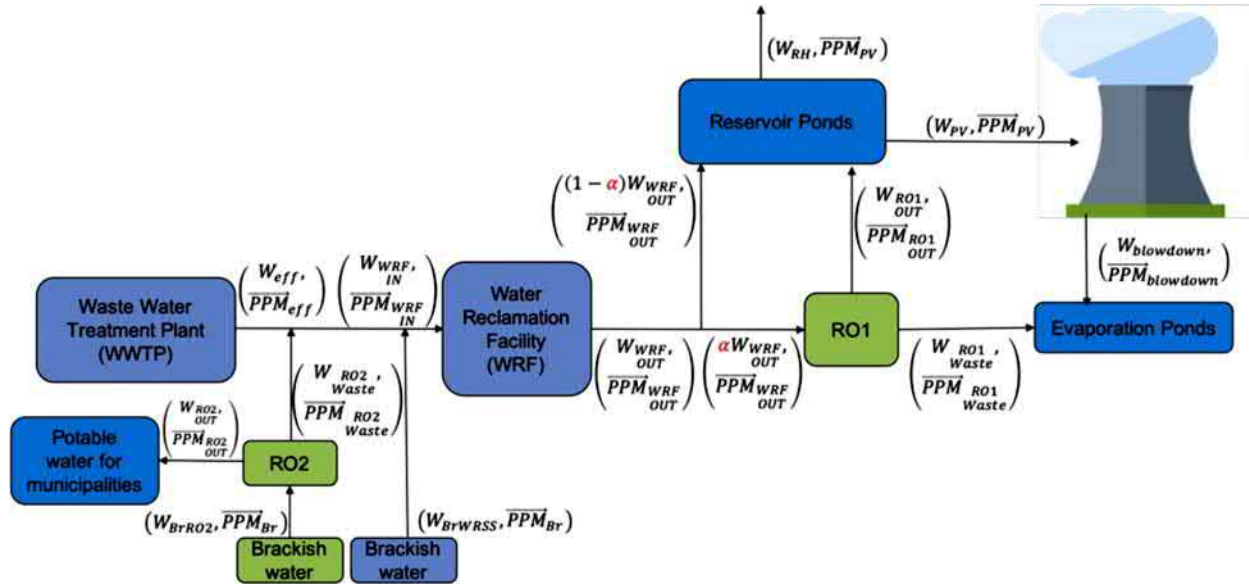


Fig. 5 Brackish water is pumped from the ground. Some of the brackish water is purified and sold, while the waste and some of the brackish water is blended with the effluent. The brackish water wells are relatively close to the WWTP and blending waste and brackish water with effluent allows to use the already existing pipeline infrastructure between the WWTP and WRF. This mix needs to be treated in an onsite RO added to the WRF. From Epiney A, Richards J, Hansen J, Talbot P, Burli P, Rabiti C and Bragg-Sitton S (2019) “Case Study: Integrated Nuclear-Driven Water Desalination—Providing Regional Potable Water in Arizona”. INL/EXT-19-55736. Idaho National Laboratory.

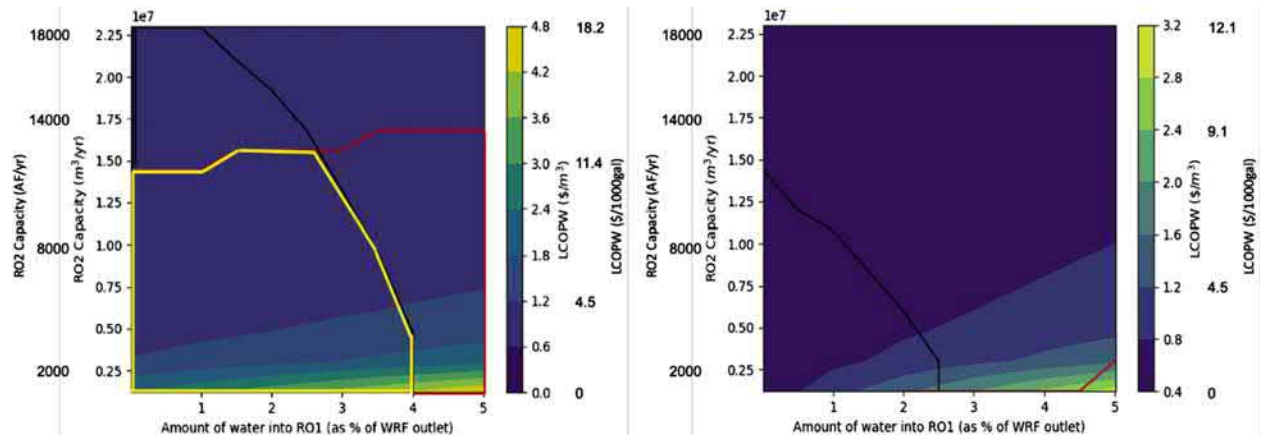


Fig. 6 LCOPW as a function of RO1 and RO2 capacity. The brackish water directly into the pipeline is 0 m³/year (0 AF/year) in the left figure and 1.9e7 m³/year (15,500 AF/year) in the right figure. The red box represents the region for which the chloride concentration in the reservoir ponds is below 450 ppm. The black box represents the region for which the total blowdown is under 4.38e6 m³/year (3550 AF/year). The yellow box shows the region for which both the salinity and the blowdown limits are respected. From Epiney A, Richards J, Hansen J, Talbot P, Burli P, Rabiti C and Bragg-Sitton S (2019) “Case Study: Integrated Nuclear-Driven Water Desalination—Providing Regional Potable Water in Arizona”. INL/EXT-19-55736. Idaho National Laboratory.

chloride reduction in the loop than the RO1. However, this conclusion only considers the blowdown limit and not the associated economics.

A developed water market model for the area suggests that bigger regional ROs (beyond 11,000 AF/year) will be even more profitable. However, to dispose of the concentrate of larger regional ROs at PVGS, additional evaporation ponds will have to be constructed. This cost was not considered in the study.

Conclusions

The integrated energy system paradigm successfully addresses a challenge that large, traditionally baseload-generation plants are currently facing. The increase of variable renewable energy generation creates high volatility that decreases the amount of the

demand that can be addressed with baseload suppliers. Within the IES concept, baseload generators are not anymore exposed to the volatility of the demand of the grid. The demand is internally stabilized by the presence of a variable load. The integrated baseload suppliers can maintain full output during low electricity demand times. Supply to non-grid energy users is reduced when power is needed on the grid. Using this integrated system approach, traditional base-load plants like NPPs can stay economically viable which is important not only to secure continuous electricity supply, but also enhances grid stability to which large rotating machines contribute considerably. The illustrative study for a regional RO integrated with PVGS shows that case dependent benefits and peculiarities such as electricity market structure and possible cooling water production can play favorably into the economics of nuclear driven desalination. This case shows how IES is not only about the introduction of variable, off the grid baseload, but it is a more holistic approach where all the feedstocks, waste, electricity co- and by-products are considered with respect to the enhancement of the financial and socioeconomic metrics of the project.

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Nuclear-Supported Electrification of the Transportation Sector

Xinyu Liu, Adarsh Bafana, Pingping Sun, and Amgad Elgowainy, Energy Systems Division, Argonne National Laboratory, Lemont, IL, United States

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Abbreviations

BEV	Battery Electric Vehicle
BWR	Boiling Water Reactor
CAP	Criteria Air Pollutants
CCS	Carbon Capture and Storage
FCEV	Fuel Cell Electric Vehicle
GREET	Greenhouse gases, Regulated Emissions, and Energy use in Technologies Model
GHG	Greenhouse Gases
G.H ₂	Gaseous Hydrogen
GJ	Gigajoule
GWP	Global Warming Potential
HTGR	High-Temperature Gas-Cooled Reactor
ICEV	Internal Combustion Engine Vehicle
km	Kilometer
L.H ₂	Liquid Hydrogen
LWR	Light Water Reactor
MWh	Megawatt hour
NG	Natural Gas
NO _x	Nitrogen Oxides
PM ₁₀	Particulate Matters with diameter less than 10 μm
PTW	Pump-to-Wheels
PWR	Pressurized Water Reactor
SMR	Steam Methane Reforming
WTP	Well-to-Pump
WTW	Well-to-Wheels

Introduction

The transportation sector contributed 28% to the total greenhouse gas (GHG) emissions in the U.S. in 2018. Of all the GHGs emitted by the transportation sector, 41.2% were emitted from passenger cars, 17.4% from light-duty trucks, with the remaining 23.2% from medium- and heavy-duty vehicles (EPA, 2020a). In addition, the transportation sector also contributes significantly to criteria air pollutant (CAP) emissions. According to the U.S. Environmental Protection Agency (EPA), in 2017, on-road

transportation contributed 29.5% to the national emission of nitrogen oxides (NO_x), which is a precursor of ground-level ozone formation and poses a serious public health risk in urban areas (EPA, 2017). Similarly, particulate matters (PM) are released from on-road vehicle operation. The PM are a mixture of solid particles and liquid droplets, having significant environmental and health hazard, e.g., causing lung and heart disease. Particularly, the PM with diameters less than 10 µm (PM₁₀) pose greater risks due to their ease of penetration to lungs (EPA, 2020b).

To reduce air emissions, electrification of the transportation sector with zero tailpipe emission vehicles is of increasing interest. Plug-in battery electric vehicles (BEVs) and hydrogen fuel cell electric vehicles (FCEVs) are two leading electrification technologies. Both BEV and FCEV have higher energy conversion efficiencies compared to internal combustion engine vehicles (ICEVs) powered by conventional gasoline. Lohse-Busch et al. conducted a laboratory test and demonstrated that a 2016 Toyota Mirai FCEV has an average powertrain efficiency of 62%, much higher than the 23% of ICEVs (Liu et al., 2020a; Lohse-Busch et al., 2020). A BEV powertrain is more efficient compared to FCEV, since BEV directly utilizes electricity from the battery to power the vehicle wheels, while FCEV needs to convert chemical energy in hydrogen to electricity in a fuel cell (Lee et al., 2018). According to www.fueleconomy.gov, a BEV has only a 15% to 20% energy loss (U.S. DOE, 2020). However, FCEV may be more cost effective compared to BEV, especially for medium- and heavy-duty vehicle applications where more onboard energy storage is needed.

Even though there are no direct tailpipe emissions from electric vehicles during on-road operation, the upstream processes to produce and distribute electricity and hydrogen might result in GHG and CAP emissions, depending on the production technology pathways and energy sources. Nuclear energy is of particular interest for both electricity and hydrogen production for BEV and FCEV applications, since it is a low-carbon energy source. Mining and purification of nuclear fuel, e.g., Uranium-235 (U-235), do consume fossil energy and emit fossil GHG, the contribution of which, however, is usually small. We conducted a full life cycle, or well-to-wheels (WTW) analysis to comprehensively assess the environmental performance of electric vehicles powered by nuclear-based electricity and hydrogen. The WTW emissions for BEV and FCEV are compared with those from conventional ICEVs powered by conventional gasoline.

The WTW analysis covers both well-to-pump (WTP) and pump-to-wheels (PTW) life cycle stages. For the case of BEV where the battery is recharged by nuclear-based electricity, the WTP life cycle includes stages such as uranium mining, enrichment, conversion, fabrication, and transportation to nuclear power plants (Wu et al., 2006). It also encompasses electricity generation via different nuclear reactor technologies, and transmission and distribution of electricity to the charging outlet. The PTW life cycle refers to the vehicle operation. Even though they have no direct evaporative and tailpipe emissions, BEV and FCEV produce minor PM emissions during on-road operation through brake and tire wear. Fig. 1 shows the system boundary of the WTW life cycle.

We utilized the Greenhouse gases, Regulated Emissions, and Energy use in Technologies (GREET[®]) model to conduct the WTW analysis (Wang et al., 2020a). The GREET model evaluates energy use and emissions for conventional transportation fuels (produced from oil refineries), conventional and emerging hydrogen production pathways (as will be discussed in the next sections), as well as regional and national electricity generation. The GREET model is updated annually to reflect both current industrial information and emerging technology status. For example, the emissions from electricity sector are updated annually based on reporting by Energy Information Administration (EIA) of the U.S. Department of Energy (DOE) (EIA, 2020).

The functional unit for WTW emissions is based on per vehicle kilometer (km) traveled. In this analysis, we focused only on the fuel cycle. The GHG emissions associated with the vehicle manufacturing cycle, including the manufacturing of engines, batteries and hydrogen fuel cells and carbon fiber tanks, has not been included, since their contribution to the total life cycle emissions (when amortized over the vehicle's lifetime driven kilometers) is small compared to the WTW emissions (Elgowainy et al., 2018; Liu et al., 2021). Similarly, the GHG emissions associated with the manufacturing of infrastructures, such as power plants and hydrogen production plants, has also not been considered in this analysis due to their even smaller impact (when amortized over total lifetime throughput of these plants). Several nuclear technologies to generate electricity are discussed below.

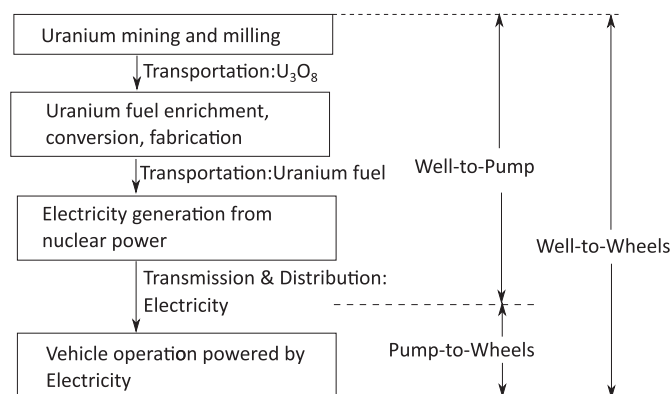


Fig. 1 The WTW system boundary of the BEV powered by nuclear electricity.

Overview of nuclear reactors

Nuclear fission is a process in which an atom splits into two or more smaller nuclei, while releasing energy in the form of heat. Two major types of nuclear reactor technologies are considered by the GREET model, namely, light water reactor (LWR) and future advanced high-temperature gas-cooled reactor (HTGR). According to the U.S. DOE Office of Nuclear Energy, all nuclear reactors operating commercially in the U.S. are LWRs (NUCLEAR 101, 2020), including both pressurized water reactors (PWR) and boiling water reactors (BWR); other reactor types are in commercial operation globally. More detailed description on these technologies can be found in other chapters of this encyclopedia, namely Brown (2021) for PWRs, Hamon (2021) for BWRs, and Fütterer (2021) for HTGRs. In contrast to LWRs, which uses ordinary water as both coolant and neutron moderator, HTGRs use a graphite moderator and helium as coolant (Wu et al., 2006; Breeze, 2019; Fütterer, 2021). Helium passes through the HTGR and carries away the heat released by the nuclear fission. The helium gas at high temperature flows through a heat exchanger to heat water for generating steam, which drives a steam turbine to provide shaft work to the power generator. Alternatively, the heated helium gas can directly drive a gas turbine to produce electricity, while the remaining heat can be used to produce steam for additional power generation or to export for other use. The HTGR has not yet been commercialized in the U.S., but it is included in this chapter for comparison with the LWR technologies in terms of life cycle emissions performance.

HTGRs require uranium fuels with a higher degree of enrichment compared to LWRs. HTGRs also have higher energy conversion efficiency since they can operate at high temperature (Wu et al., 2006; Breeze, 2019). From one gram of U-235, 31.3 GJ (8.7 MWh) of electricity can be generated using a HTGR, while a LWR may only generate 24.8 GJ (6.9 MWh) of electricity (Wu et al., 2006; Wang et al., 2020b). Details about the amount and share of energy sources required for the operation of nuclear reactors and the upstream production and transportation of uranium fuel are documented in Wu et al. (2006) and are already incorporated in the GREET model.

In the next section, we assess WTW GHG and CAP emissions of BEVs recharged with nuclear-based electricity, national and regional grid electricity mixes, and compare the results to those from conventional gasoline powered ICEVs.

WTW GHG emissions of BEV recharged by electricity from nuclear and other sources

In the current case, we assume that U.S. 2019 grid mix is used by all stationary processes in BEV's WTW analysis. The BEV operation stage can be powered by electricity from various sources, including nuclear power and national or regional grid electricity mix. The GREET model provides projections of future (i.e., 2035) electricity generation mix and emission factors (e.g., g CO₂ kWh⁻¹ electricity), based on EIA's Annual Energy Outlook (EIA, 2020). It also incorporates projected improvements in generation efficiencies based on literature data, which enables the estimation of the impacts of evolving technologies on WTW emissions. In this study, we consider two scenarios: the "current" scenario represents technology year of 2019 and the "future" scenario represents the technology year of 2035. For BEVs, we assume a battery charging efficiency of 85%.

Fig. 2 shows that for gasoline ICEV, over 80% of the WTW GHGs are emitted during the vehicle operation stage, i.e., when the gasoline is combusted in the engine. In contrast, BEVs do not produce GHG emissions during vehicle operation, with all their WTW GHG emissions occurring upstream, i.e., during the WTP stage. Fig. 2 also indicates that BEVs recharged by the current U.S. grid generation mix, which has a significant contribution from fossil power generation, has much lower WTW GHG emissions compared to gasoline ICEV. Clearly, the WTW GHG emissions of BEVs are highly dependent on the electricity generation fuel shares. BEVs

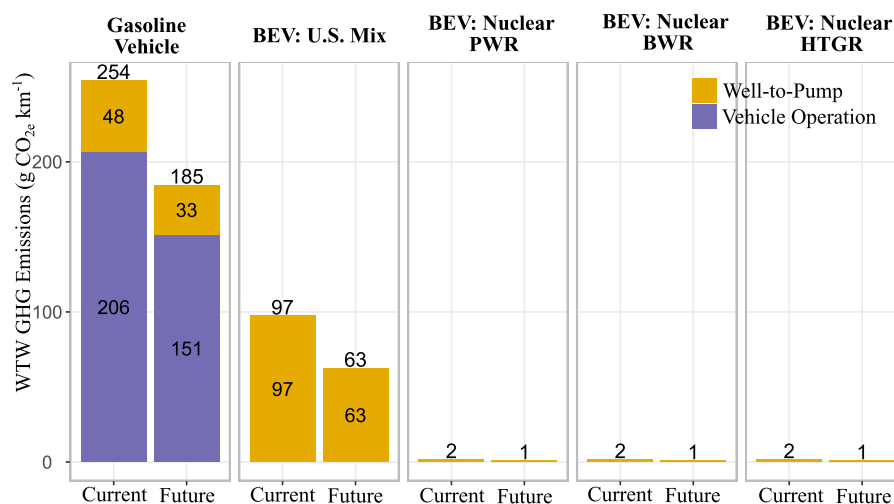


Fig. 2 WTW GHG emissions of gasoline ICEV and BEV recharged with electricity from various sources/technologies. The gram of CO₂ equivalent (CO_{2e}) is calculated using global warming potential (GWP) of CO₂, CH₄ and N₂O. The GWP of CO₂, CH₄ and N₂O is 1, 30, and 265, respectively.

recharged with nuclear-based electricity provide deep reductions in WTW GHG emissions, by 99% compared to gasoline vehicles and by 98% compared to BEVs powered by U.S. average grid generation mix. Among different nuclear technologies, PWR and BWR incur the same WTW emissions as provided by GREET 2020. The HTGR technology has slightly lower WTW GHG emissions compared to LWR technology due to its higher energy conversion efficiency, but the difference is negligible when the WTW results are calculated on a per-km basis.

In addition, Fig. 2 suggests that future U.S. average electricity generation mix for BEV recharging will provide deeper reduction in WTW GHG emissions compared to the current grid generation mix. Even though not 100% carbon-free, the nuclear technologies have the advantage of providing a steady electricity supply (with capacity factor of > 90%), when compared to solar photovoltaic (20–30% capacity factor) and wind power generation (30–40% capacity factor) (Liu et al., 2020b). We note that if all light-duty vehicles in the United States in 2019 (about 273 million vehicles) are electrified, the total annual electricity use for recharging these vehicles will be in the order of 1000 TWh. Similarly, if all medium- and heavy-duty vehicles in the United States in 2019 (about 12.7 million vehicles) are electrified, the total annual electricity use for recharging these vehicles will be in the order of 1200 TWh. To put this in context, the total annual generation capacity of nuclear power plants in the United States in 2019 was in the order of 850 TWh.

WTW CAP emissions of BEV recharged by electricity from nuclear and other sources

Fig. 3 shows that BEVs recharged by nuclear LWR technology emit only 4 mg NO_x per km traveled on a WTW basis, representing a 98% reduction compared to gasoline ICEVs that emit 159 mg NO_x per km. For gasoline ICEVs, approximately 47% (i.e., 75 mg km⁻¹) of the NO_x are emitted during the vehicle operation stage. This suggests that replacing gasoline ICEVs with BEVs in an urban driving setting can mitigate the public health concern caused by ground-level ozone formation, which utilizes NO_x as a precursor (Lee et al., 2018).

The emissions reduction potentials of BEVs recharged by nuclear-based electricity are significant for other CAPs as well. In terms of WTW PM₁₀ emissions (Fig. 4), BEVs recharged by nuclear LWR electricity emit 11 mg PM₁₀ per km traveled, representing over 50% reduction compared to gasoline ICEVs that emit 23 mg PM₁₀ per km. For both ICEVs and BEVs, the majority of the PM₁₀ emission occurs during the vehicle operation stage due to brake and tire wear. For ICEVs, the additional PM₁₀ emissions are emitted from tailpipe due to the combustion of gasoline in engine during vehicle operation. For BEV recharging using nuclear-based electricity, the contribution of the WTP life cycle to WTW PM₁₀ emission is negligible.

Impacts of regional electricity generation mix

The contiguous U.S. has eight major electricity regions according to North American Electric Reliability Corporation, and each region varies in shares of fuels use and electricity generation technologies. Fig. 5 shows the potential GHG emissions reduction benefits of BEVs powered by electricity generated from nuclear LWR technologies, compared to those powered by 2019 regional electricity generation mixes. The values in Fig. 5 indicate the ratio of WTW GHG emissions of BEVs powered by regional electricity generation mixes to those powered by electricity generated from nuclear LWR technologies. The higher value in a given region indicates a higher GHG emissions intensity associated with electricity generation mix in that region. Fig. 5 indicates that even if BEVs are recharged by the regional mix with the lowest carbon intensity, i.e., Northeast Power Coordinating Council or NPCC, they still emit 33 times more WTW GHGs compared to those recharged by nuclear power. This clearly shows the GHG reduction benefits of electrifying the transportation sector with nuclear-based electricity.

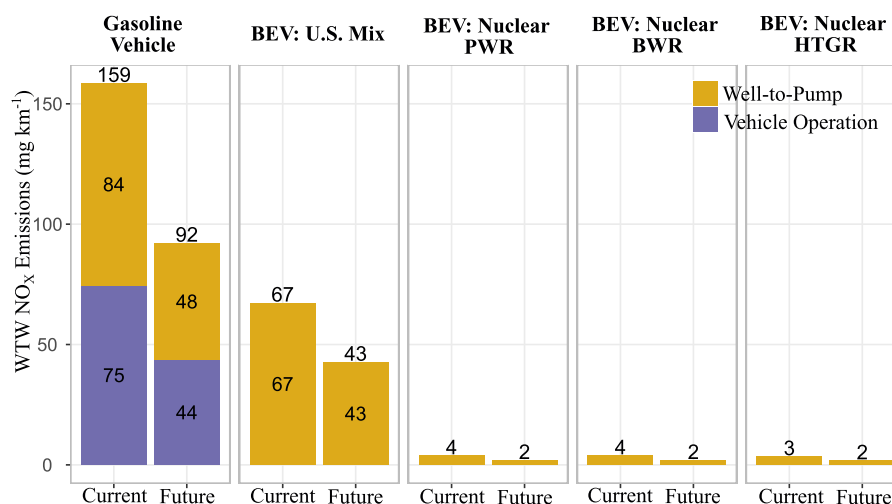


Fig. 3 WTW NO_x emissions of gasoline ICEV and BEV recharged with electricity from various sources/technologies.

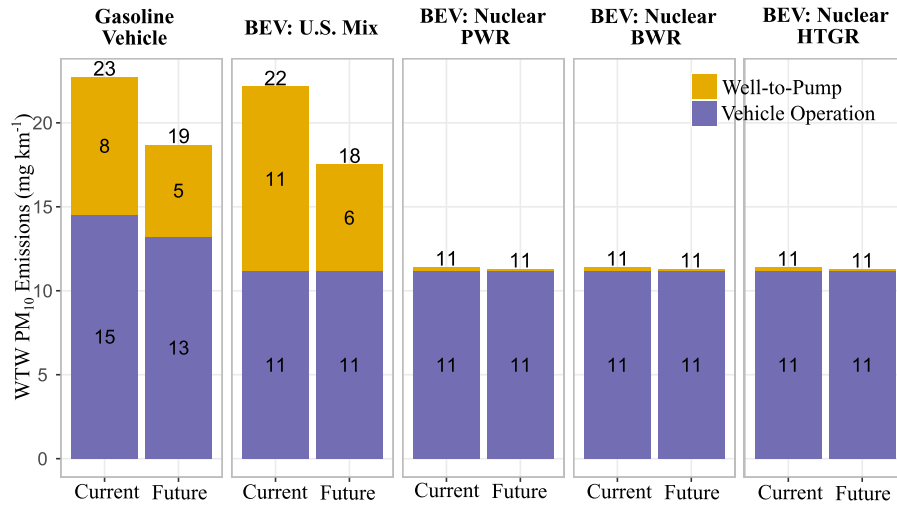


Fig. 4 WTW PM₁₀ emissions of gasoline ICEV and BEV recharged with electricity from various sources.

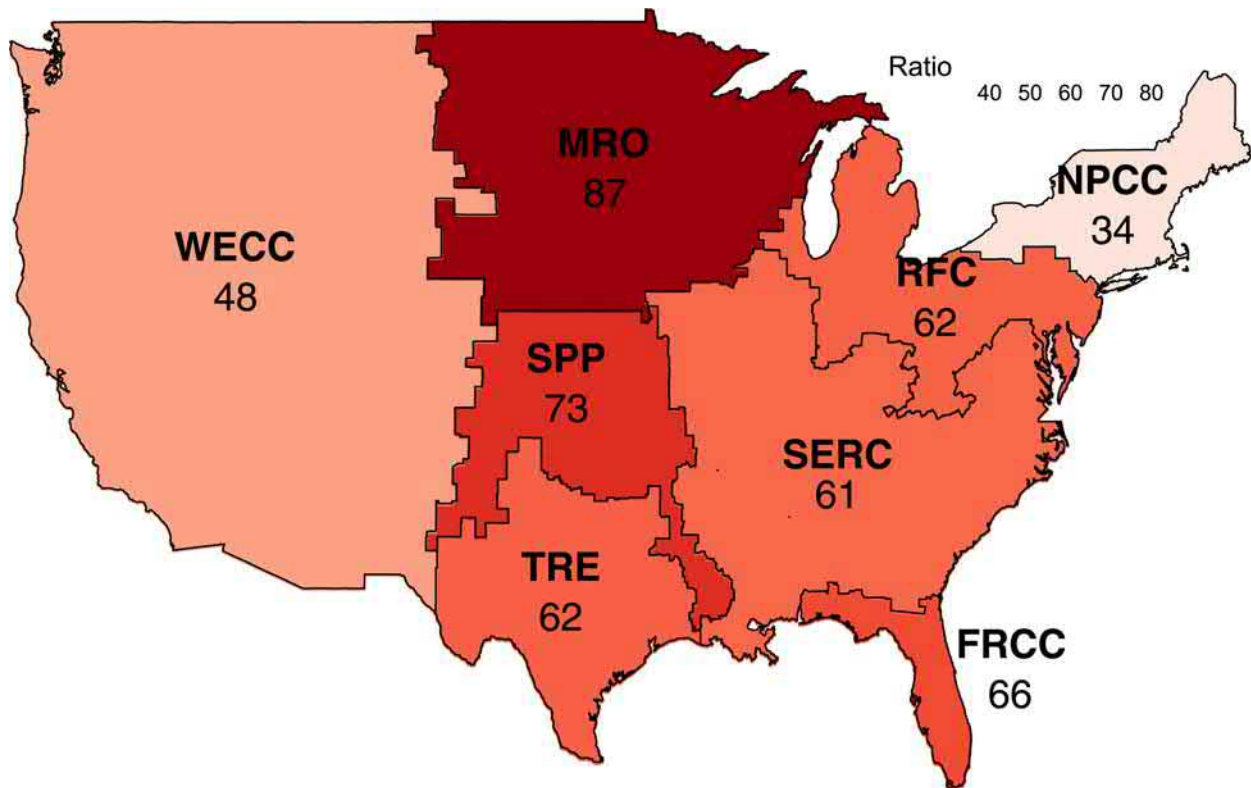


Fig. 5 The ratio of WTW GHG emissions of BEVs recharged by regional electricity generation mixes to those powered by electricity generated from nuclear LWR technologies. Regional electricity grids: Western Electricity Coordinating Council (WECC), Midwest Reliability Organization (MRO), Southwest Power Pool (SPP), Texas Reliability Entity (TRE), SERC Reliability Corporation (SERC), Florida Reliability Coordinating Council (FRCC), Reliability First Corporation (RFC), and Northeast Power Coordinating Council (NPCC).

FCEV powered with H₂ produced from various technology pathways

FCEVs using hydrogen (H₂) as fuel provide another clean alternative to petroleum gasoline and diesel in transportation applications by producing zero tailpipe emissions. H₂ can be produced from a diverse range of energy sources and technology options, such as natural gas (NG) steam methane reforming (SMR), biomass gasification, and water electrolysis using electricity generated from various sources.

H₂ production technologies using various electricity sources

Nuclear energy can be utilized for producing hydrogen, with near-zero emissions (Knighton et al., 2020), as described in other chapters within this section (Suppiah, 2021; Boardman, 2021). In a centralized plant, H₂ can be produced via water electrolysis, which uses nuclear-based electricity to split water molecules into H₂ and oxygen (O₂). Additionally, H₂ can be produced via high temperature electrolysis in a solid oxide electrolyzer cell (HTE SOEC) using nuclear energy.

Alternatively, H₂ can be produced from low temperature electrolysis in a proton exchange membrane electrolyzer (LTE PEM) using nuclear-based electricity at refueling stations. The H₂ production using nuclear energy has significant environmental benefits relative to the current industrial hydrogen production via NG SMR.

In NG SMR, methane reacts with steam at high temperature and pressure in the presence of a catalyst to produce H₂ and carbon monoxide. This is followed by the “water-gas shift reaction,” where the carbon monoxide and steam react in the presence of a catalyst to produce carbon dioxide and additional H₂. Hydrogen is also co-produced at chlor-alkali plants that use an electrolysis process for splitting sodium chloride solution to produce chlorine and caustic soda, as well as H₂ gas as a byproduct.

To quantify the environmental benefits of FCEVs fueled by H₂ produced via electrolysis using nuclear-based electricity, the WTW GHG and CAP emissions are evaluated and compared to those from conventional ICEVs and FCEVs fueled by H₂ produced from other sources.

Fig. 6 depicts the WTW system boundary of the FCEV powered by H₂ produced from NG SMR and water electrolysis using electricity generated from various technologies, including nuclear LWR and HTGR. As with the assessment of BEVs, the technology year 2019 was selected in the current scenario and the technology year 2035 as future scenario. The WTW environmental impacts of the hydrogen FCEVs were evaluated in terms of their GHG and CAP emissions (e.g., PM₁₀ and NO_x) and compared to those from conventional gasoline ICEVs.

The WTW GHG and CAP emissions of electrolysis pathways for H₂ production depend strongly on fuel shares and generation technologies used for electricity generation. Therefore, the WTW emissions for FCEVs using H₂ produced via water electrolysis are evaluated for the following electricity generation sources: (1) electricity from nuclear power plants (nuclear electricity, as detailed in previous sections), (2) the current average U.S. grid generation mix (2019 U.S. grid mix), (3) future U.S. grid mix (2035), (4) current California grid mix (2019 CA mix), (5) future CA mix (2035), and (6) wind/solar electricity.

The production of H₂ can be centralized for large scale production to benefit from economies of scale, however, the H₂ product needs to be distributed and delivered to refueling stations, which consumes additional energy and generates air emissions. In contrast, H₂ can also be produced at refueling stations to reduce the energy use and cost associated with H₂ delivery, but this approach requires additional capital investment and land area at refueling sites. The emissions associated with H₂ delivery from central production sites to refueling stations are discussed below.

Gaseous and liquid hydrogen delivery pathways

In addition to its production, the delivery of H₂ plays an importation role in the overall WTW air emissions. Today, H₂ is delivered from central production sites to refueling stations by trucks in compressed gas or liquid product form. For gaseous H₂ (G.H₂) delivery, H₂ may be compressed to a pressure between 200 and 500 bar and loaded onto gaseous tube-trailers for transportation and distribution. The tube-trailer at a refueling station supplies H₂ to a gaseous compressor, which compresses it to 950–1000 bar and stores it in a high-pressure buffer storage system for dispensing into vehicle tanks at 700 bar. The hydrogen is typically precooled to –40 °C to avoid the overheating of vehicle tanks during fast refueling. The hydrogen packaging, delivery and refueling processes contribute to the WTW emissions of FCEVs (Liu et al., 2020). H₂ may also be liquefied, typically using liquid nitrogen to

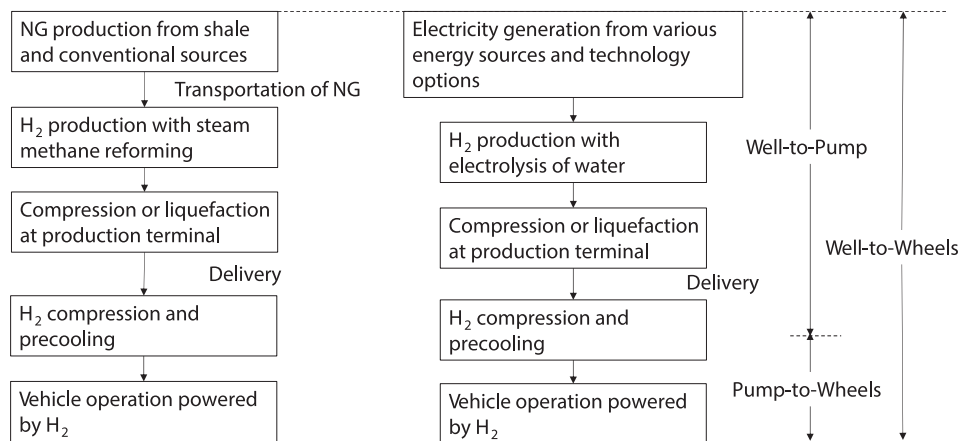


Fig. 6 The WTW system boundary of the FCEV powered by H₂ produced from NG SMR and water electrolysis using electricity generated from various technologies.

precool the H_2 from ambient temperature to near 80 K, followed by a series of compression and expansion processes to reach 20 K, the temperature necessary for H_2 liquefaction near ambient pressure (Liu et al., 2020). Liquid hydrogen ($L.H_2$) is then stored in large cryogenic storage tanks at an adjacent distribution terminal, and subsequently loaded into cryogenic-liquid tankers for delivery to refueling stations. In general, $L.H_2$ delivery is more cost-effective compared to compressed $G.H_2$ delivery in tube-trailers, especially for long transportation distances, owing to its higher payload (4 vs. 1 metric ton). However, the H_2 liquefaction process is energy-intensive, consuming between 11 to 15 kWh of electricity per kg of H_2 , depending on H_2 supply pressure and the energy efficiency of the heat exchange, compression and expansion processes during liquefaction. Thus, the electricity used for liquefaction is an important element in the $L.H_2$ WTW analysis.

WTW air emissions of FCEVs using H_2 from various production and delivery pathways

For FCEVs, the only emission from the tailpipe is water vapor, so the WTW GHG and CAP emissions are mainly determined by the emissions from the upstream H_2 production and delivery activities. Table 1 shows the WTW GHG and CAP emissions for FCEVs using H_2 from 12 different production and delivery pathways and compares the results to those from gasoline ICEV.

The FCEVs with the highest WTW GHG emissions are those utilizing H_2 from fossil-fuel dependent pathways, namely NG SMR $G.H_2$ and $L.H_2$, as well as electrolysis pathways using current U.S. and CA grid mix. The WTW analysis considers emissions from the following stages: (1) H_2 production and its upstream processes (e.g., for SMR technology, the upstream processes include the natural gas recovery, processing, and transmission to hydrogen plant), (2) H_2 transportation and distribution from production sites to refueling stations, and (3) H_2 compression and precooling at refueling stations (only for $G.H_2$ delivery). We assume that compression and precooling of gaseous hydrogen at refueling stations sources electricity from the U.S. grid, and thus result in upstream (indirect) emissions from fossil power plants within the grid.

As seen in Fig. 7, FCEV using either $G.H_2$ or $L.H_2$ produced via HTE SOEC, exclusively sourcing nuclear energy, leads to lower WTW GHG emissions compared to other pathways. In contrast, FCEV using H_2 produced via LTE PEM using current U.S. grid mix electricity has the highest WTW GHG emissions, since over 60% of the current U.S. grid mix relies on fossil fuel sources, i.e., coal and NG (Wang et al., 2020a, b). Currently, H_2 is mainly produced from NG via SMR process. The FCEVs sourcing $L.H_2$ and $G.H_2$

Table 1 WTW emissions of FCEVs using H_2 from various production and delivery pathways.

Pathways	Abbreviation	GHG ($g\ CO_{2e}\ km^{-1}$)	NO_x ($mg\ km^{-1}$)	PM_{10} ($mg\ km^{-1}$)
1. ICEVs Gasoline Vehicle	ICEV-Gasoline	254	159	23
2. Central Plant NG SMR ($G.H_2$)	CP NG SMR ($G.H_2$)	127	48	17
3. Central Plant NG SMR ($L.H_2$ US mix)	CP NG SMR ($L.H_2$ US mix)	169	79	22
4. Central Plant NG SMR ($L.H_2$ Nuclear Electricity for Liquefaction)	CP NG SMR ($L.H_2$ NE Liqf.)	105	37	15
5. Central NG SMR w CCS ($G.H_2$)	CP NG SMR w CCS ($G.H_2$)	37	51	18
6. Central Plant Byproduct H_2 from Chloro-Alkali ($G.H_2$)	CP Chloro-Alkali ($G.H_2$)	123	19	18
7. Central Plant HTE SOEC with Nuclear Energy ($G.H_2$)	CP HTE NE ($G.H_2$)	23	18	14
8. Central Plant HTE SOEC with Nuclear Energy ($L.H_2$)	CP HTE NE ($L.H_2$)	66	51	19
9. Refueling Station LTE Electrolysis with US mix ($G.H_2$)	RS LTE US mix ($G.H_2$)	250	172	39
10. Refueling Station LTE Electrolysis with future US mix 2035 ($G.H_2$)	RS LTE US mix 2035 ($G.H_2$)	133	91	25
11. Refueling Station LTE Electrolysis with CA grid mix current ($G.H_2$)	RS LTE CA mix ($G.H_2$)	120	110	29
12. Refueling Station LTE Electrolysis with future CA mix 2035 ($G.H_2$)	RS LTE CA mix 2035 ($G.H_2$)	41	26	13
13. Refueling Station LTE Electrolysis with Solar/Wind ($G.H_2$)	RS LTE Solar/Wind ($G.H_2$)	17	12	13

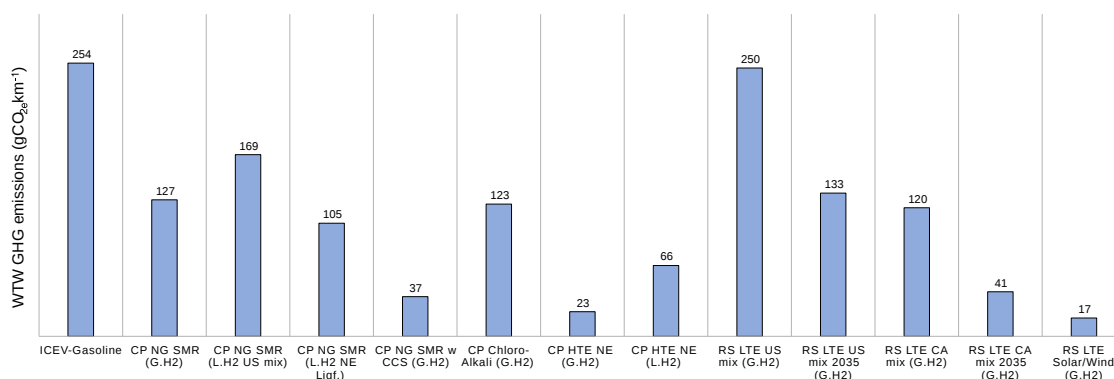


Fig. 7 WTW GHG emissions of FCEVs using H_2 from various production technologies, compared with gasoline ICEV.

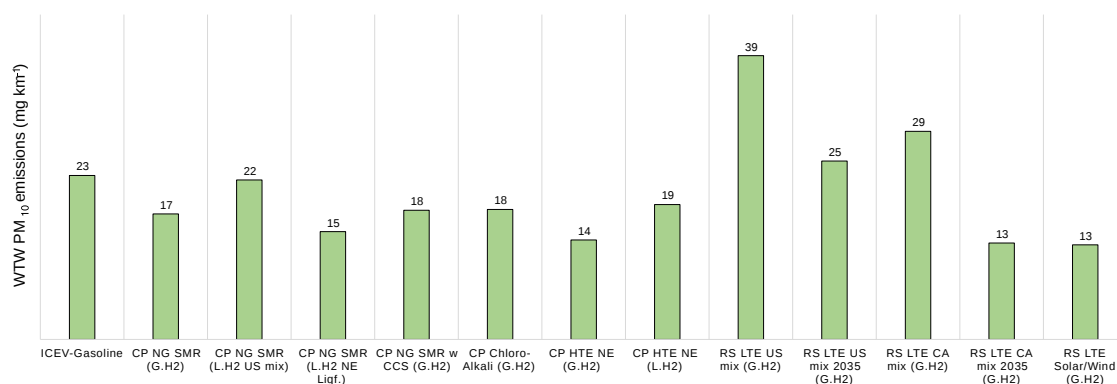


Fig. 8 WTW PM₁₀ emissions of FCEV using H₂ from various production technologies, compared with gasoline ICEV.

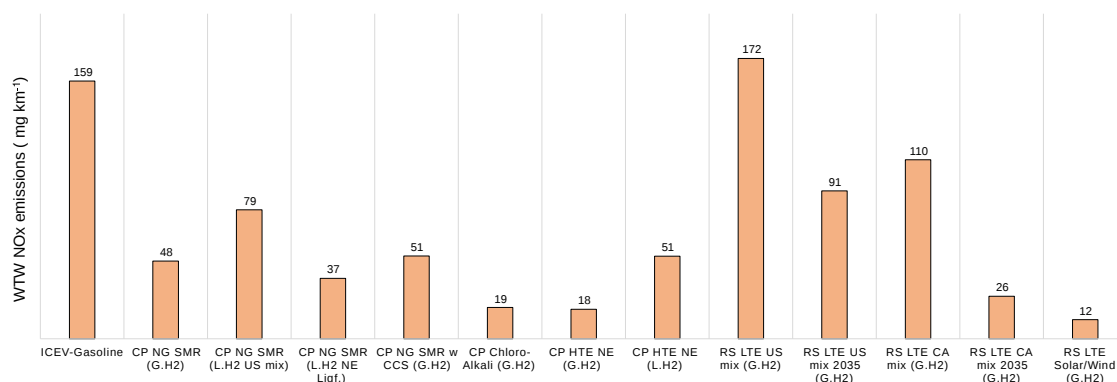


Fig. 9 WTW NO_x emissions of FCEVs using H₂ from various production technologies, compared with gasoline ICEV.

produced from the NG SMR pathway have WTW GHG emissions of 169 and 127 g CO_{2e} km⁻¹, respectively. If H₂ is liquefied via nuclear electricity, instead of U.S. mix, the WTW GHG emissions of FCEVs reduce from 169 to 105 g CO_{2e} km⁻¹. Evidently, the electricity source has a significant impact on the WTW GHG emissions of FCEVs when electricity is used for H₂ liquefaction and compression or when electrolysis sources electricity from the grid for hydrogen production. The NG SMR pathway with carbon capture and storage (CCS) has lower GHG emissions when compared to conventional NG SMR or water electrolysis with current or future U.S. and CA electricity grid mix.

The WTW GHG emissions of H₂ FCEVs are compared to gasoline ICEV, which has a WTW GHG emission of 254 g CO_{2e} km⁻¹, as shown in Fig. 7. The WTW GHG emission reduction potential of FCEVs relative to gasoline ICEVs depends strongly on the H₂ production and delivery technology pathway, and the carbon intensity of their energy sources. FCEVs powered by H₂ produced via water electrolysis using current U.S. grid mix has a WTW GHG emission of 250 g CO_{2e} km⁻¹, comparable to gasoline ICEV. In contrast, the FCEVs powered by H₂ produced via water electrolysis using nuclear and wind/solar electricity have the largest WTW GHG emissions reduction compared to gasoline ICEV.

Figs. 8 and 9 show the WTW PM₁₀ and NO_x emissions, respectively, for FCEVs using H₂ from various production and delivery pathways. As with WTW GHG emissions, FCEVs using H₂ produced from nuclear and wind/solar electricity-based water electrolysis produce the lowest WTW PM₁₀ and NO_x emissions. In general, all pathways that use a significant amount of grid electricity (e.g., for electrolysis and/or liquefaction) are associated with significant WTW NO_x and PM₁₀ emissions, consistent with share of fossil generation in the grid mix.

Conclusions

Using nuclear energy for either BEV (to supply electricity for battery recharging) or for FCEV (to produce H₂ for fuel cells) provides significant environmental benefits in terms of WTW air emissions reduction, particularly for GHG, PM₁₀ and NO_x. With power generation and transportation as the leading sectors for GHG emissions, nuclear energy can play an important role in the concerted effort to combat climate change via both near zero carbon electricity generation and near complete decarbonization of the transportation sector.

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Relevant Website

<https://greet.es.anl.gov/>—Detailed information of GREET model.

Heat Pumps Driven by Zero-Carbon Electricity

J. Stephen Herring, Center for Space Nuclear Research, Universities Space Research Association, LLC, Idaho Falls, ID, United States

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Glossary

COP Coefficient of performance, the ratio of the quantity of heat pumped from the cold temperature source to the hot temperature sink divided by the temperature difference between the source and the sink. The Carnot COP is the maximum COP and is: $COP_{Carnot} = \frac{T_H}{T_H - T_C}$ where T_H and T_C are the absolute temperatures in Kelvin (K) [Water freezes at 273 K and boils at 373 K when the pressure is 1 atm, 14.7 psi.]

HTHP High Temperature Heat Pump, defined as having a heat sink temperature $> 90^\circ\text{C}$.

Refrigerant (or Coolant) The fluid flowing through the heat pump system, generally as a liquid or a gas in various locations, that transfers heat from a heat source to a heat sink. In most of the instances described in this article the heat source is a lower temperature than the heat sink.

T_{Lift} Temperature difference between the sink and the source, $T_H - T_C$.

T_{Sink} For heat pumps in a heating mode, the output temperature, T_H For heat pumps in the cooling mode, the temperature at which heat is absorbed, T_C .

Introduction

Heat pumps are proof that, Joule for Joule, electricity is more valuable than heat from the point of a heat consumer. The best boilers have an efficiency around 90%, which means that they can turn 1 kWh of natural gas (heat content) into 0.9 kWh of useful heat. That same 1 kWh of heat from natural gas, burned in a turbine-generator, produces about 0.5 kWh of electricity. A heat pump can turn the 0.5 kWh of electricity into 2.5 kWh of useful heat (sometimes even more). This comparison, as well as several others included later in [Table 2](#), illustrates the importance of the Second Law of Thermodynamics in supplying heat to domestic and industrial processes, particularly at temperatures below 200°C ([Bienvenu, 2014](#)).

The need to severely constrain emissions of greenhouse gases in order to reduce climate change translates into a need to use carbon-free electricity for processes now heated by fossil fuels. Those processes include drying, cooking, pasteurization and other processes now commonly heated by the combustion of natural gas. A possible, though inefficient, heat source is resistance heating, such as the heating in an electric toaster. The thermal conversion efficiency (i.e., electrical energy divided by the thermal energy input) for most power plants, be they coal, natural gas, nuclear or wood heated, is generally between 30% and 60%. Therefore, the heat delivered to the product is only 30–60% of the energy of the fuel.

If the natural gas, coal or wood is used directly in the heating process, then in theory 100% of the energy in the fuel could be delivered to the process. In reality some heat is lost via the exhaust gases and to the ambient air surrounding the equipment.

But the temperatures in many industrial and commercial processes do not require the highest temperatures that can be attained via natural gas, coal or wood flames. A better solution in those cases is to use the high temperature flame to generate electricity, which in turn could operate a heat pump. Heat pumps can transfer heat from a low temperature, T_C , to a higher temperature, T_H , through the use of mechanical energy for compression. Today T_H is limited to about 140°C , though continuing research is raising that limit.

If the mechanical energy needed to drive the compressor is furnished by wind turbines, solar panels, hydro or nuclear, then the process heat can be delivered without the emission of greenhouse gases.

Theoretically heat pumps could deliver heat at very high temperatures, but the practical limitations of the working fluids (i.e., the “refrigerants”) limit the delivery temperatures to about 200°C . When the industrial processes require temperatures above 300°C , direct heat exchange with the reactor coolant, particularly helium from a Very High Temperature Reactor, is more advantageous.

Above about 1000 °C concentrating solar furnaces or the use of electricity via induction, arcs, microwaves or resistance heating is necessary.

It is often suggested that the heat of the reactor coolant could be used via a heat exchanger to supply the heat directly to the industrial processes. In those cases, tens or hundreds of megawatts could be supplied to a large facility such as a refinery, a chemical processing plant, a desalination plant or a hydrogen production facility using a thermochemical process (e.g., the sulfur-iodide or hybrid sulfur processes) (McMillan et al., 2016). Such industrial complexes would be large enough to host a dedicated reactor within their site. Serving several adjacent facilities might be feasible, though the long-distance transport of a high temperature gas, molten salt or liquid metal would require a substantial investment in the piping and pumping infrastructure. Furthermore, the possibility of radioactive contamination permeating through the heat exchanger and being carried in the intermediate coolant has to be carefully considered. In order to be economically feasible, the aggregate heat demands of the industrial complexes would have to be massive if the nuclear plant is dedicated to heat applications alone.

Heat pumps, especially High Temperature Heat Pumps (HTHPs) have the advantage of being able to serve demands in the kW to few MW range at locations remote from the reactor (or other generators, such as wind turbines or solar panels) with modest costs for added electrical distribution infrastructure. Heat pumps available today range in heating capacities from 20 kWth to 20 MW. In addition, concerns for the potential transport of radioactive contamination in the intermediate coolant are avoided through the use of electrical transmission.

Often, the low temperature (i.e., T_C) heat source can be waste heat that is typically vented to the atmosphere as exhaust steam, exhaust air from industrial processes and cooling of products. In fact, some installations use heat pumps to supply both air conditioning and domestic hot water, using the heat rejected by the air conditioner as the heat source for water heating. Some more elaborate installations in supermarkets supply hot water and building heat while cooling freezers and fresh product cabinets. These systems have to be flexible to accommodate changing seasons, product arrangements and customer traffic.

The largest market today for heat pumps is for building heating and air conditioning. In the heating mode the heat source is the outside air ($-15\text{ }^{\circ}\text{C}$ to $15\text{ }^{\circ}\text{C}$) or groundwater ($5\text{ }^{\circ}\text{C}$ – $20\text{ }^{\circ}\text{C}$) while the heat sink is the room temperature of the building or home ($20\text{ }^{\circ}\text{C}$ – $25\text{ }^{\circ}\text{C}$). In the cooling mode the heat source is the building interior ($20\text{ }^{\circ}\text{C}$ – $25\text{ }^{\circ}\text{C}$) while the heat sink can be summer outside air ($30\text{ }^{\circ}\text{C}$ – $45\text{ }^{\circ}\text{C}$) or groundwater which remains ($15\text{ }^{\circ}\text{C}$ – $25\text{ }^{\circ}\text{C}$). The global heat pump market surpassed USD 48 billion in 2019 and annual sales are anticipated to exceed 16 million units by 2026 (GMInsights, 2020).

The operation of a heat pump

The operation of a heat pump such as that depicted in Fig. 1 is shown for a two-phase coolant in Fig. 2. The pressure of the coolant is on the vertical axis and the specific volume (m^3/kg) is shown on the horizontal axis. The points 1 through 4 in Fig. 1 correspond to the conditions at points 1–4 on the p, V diagram in Fig. 2. Beginning at point 1, the mixture of liquid and vapor is boiled in the evaporator at a constant temperature, T_C , to a completely vaporous state at point 2, the top of the evaporator in Fig. 1. The vapor is then compressed to a higher pressure by the compressor and is slightly superheated (i.e., is above the boiling temperature for the hot temperature, T_H) at point 3. Because of the compression, the coolant condenses at a higher temperature, T_H , between points 3 and 4, and delivers useful heat to a process, such as those shown in Tables 1 and 2. The cycle is completed when the high-pressure liquid

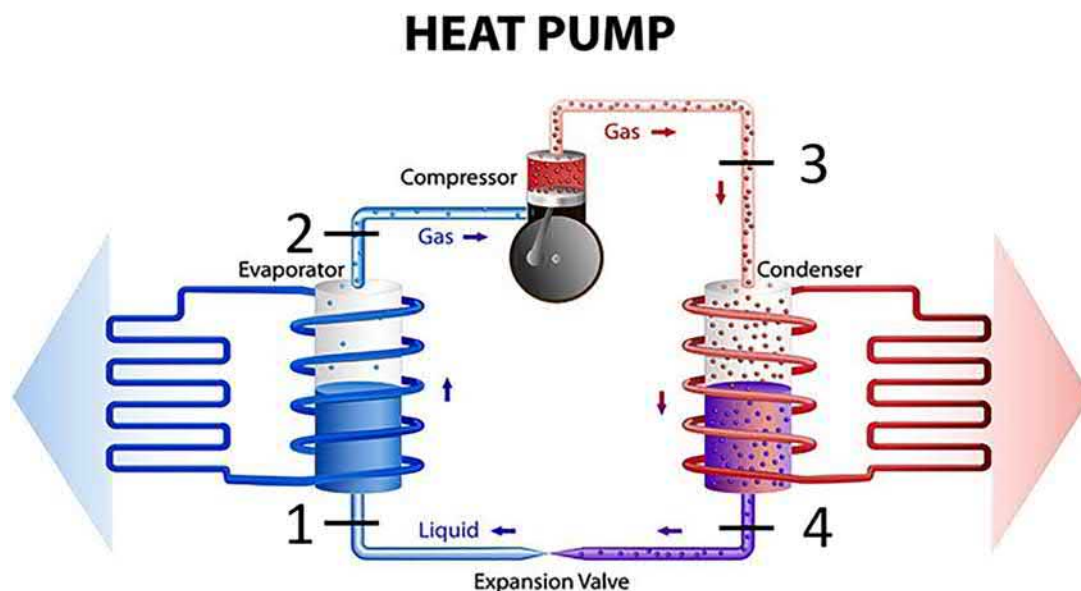


Fig. 1 Components of a simplified heat pump. <https://www.eec.org.au/for-energy-users/technology-2/heat-pumps>

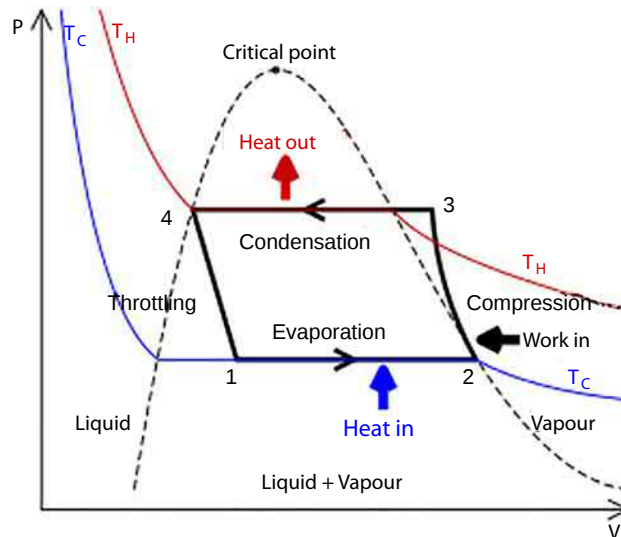


Fig. 2 Pressure—Volume diagram for a heat pump. File:Refrigeration PV diagram.svg

Table 1 US process heat demand for which HTHPs might be used.

2008 Process heat demand ($PJ_{th} = 10^{15} J_{th}$)				
Process	40 °C–80 °C	80 °C–120 °C	120 °C–160 °C	160 °C–200 °C
Paper and pulp			1050	180
Chemicals			420	630
Food industry	230	180	60	100
Primary metals				140
Other manufacturing	15	30	220	150

The total demand in these industries equals 3416 PJ_{th} , resulting in the emission of 170 million t of CO_2 if supplied by natural gas.

Adapted from Arpagaus C, Bless F, Uhlmann M, Schiffmann J and Bertsch S (2018) High temperature heat pumps: Market overview, state of the art, research status, refrigerants, and application potentials. In: *International Refrigeration and Air Conditioning Conference, Paper 1876*. <https://docs.lib.purdue.edu/iracc/1876> (Accessed on 23 January, 2021).

flows through an expansion (throttling) valve between points 4 and 1 of Fig. 1, exiting as a liquid-vapor mixture at the lower evaporator pressure. Heat does not flow effortlessly from a low temperature to a higher temperature; the mechanical energy exerted in the compressor pumps the heat.

The relationship between the amount of heat pumped up to the higher temperature to the mechanical work done in the compressor is known as the Coefficient of Performance, COP. For a perfect, isentropic (i.e., Carnot) cycle COP is given by

$$COP = \frac{T_H}{(T_H - T_C)}$$

where both T_H and T_C are absolute temperatures (i.e., the number of degrees above absolute zero).

Currently available high temperature heat pumps

Arpagaus et al. (2018) compared the COP values for about 35 industrial HTHP systems for a range of temperature lifts ($T_{lift} = T_H - T_C$) supplying heat at 140 °C, here labeled T_{sink} . As shown in Fig. 3, nearly all of the HTHPs had COP values between 30% and 60% of the Carnot COP, the theoretical maximum.

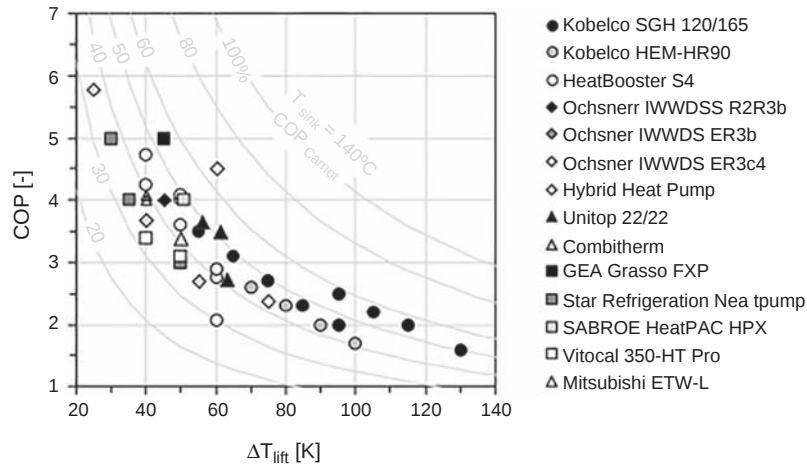
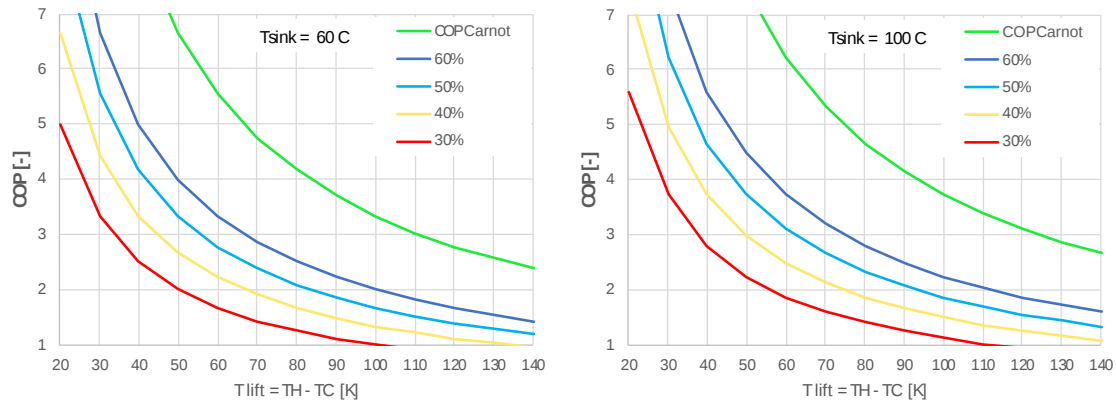
Comparisons of simple HTHP applications

In Fig. 4, the curves for the Carnot Coefficient of Performance, the theoretical maximum, and for various fractions of COP_{Carnot} from 30% to 60% are plotted. Based on the performance shown, we can estimate the performance of HTHPs in various situations.

Nine different configurations using an arbitrary amount of fuel (here, 1 GJ_{th}) are shown in Table 2. The amount of fuel used has no effect on the comparisons; the ratios remain the same. Example 1 is a natural gas burner in a food processing plant cooking

Table 2 Comparisons of the impact of HTHPs on energy efficiency and CO₂ emissions.

Example	Fuel	Customer	Technology	Generation efficiency	T_{cold} [°C]	T_{hot} [°C]	T_{lift} [K]	% of carnot	COP	Delivered heat [GJ _{th}]	CO ₂ per GJ _{th} delivered [kg]
1	NG (48 MJ/kg)	Food processing	Gas range	97%	n.a.	110	n.a.	n.a.	1.00	0.97	709
2	NG	Electric range	Combined cycle	60%	n.a.	110	n.a.	n.a.	1.00	0.60	1146
3	NG	Food processing	Steam turbine	38%	35	110	75	60%	3.06	1.16	590
4	NG	Food processing	Combined cycle	60%	35	110	75	60%	3.06	1.84	374
5	NG	Hot water	Steam turbine	38%	35	70	35	60%	5.88	2.23	308
6	NG	AC/hot water	Combined cycle	60%	20	70	50	60%	4.12	2.47	278
7	Nuclear	Food processing	Steam turbine	32%	35	110	75	60%	3.06	0.98	0
8	Nuclear	Cooking/ freezing	Steam turbine	32%	−10	80	90	60%	2.35	0.75	0
9	Nuclear	Distillation	VHTR	45%	60	160	100	60%	2.60	1.17	0

**Fig. 3** COP as a function of the temperature lift for various industrial HTHPs. Arpagaus C, Bless F, Uhlmann M, Schiffmann J and Bertsch S (2018) High temperature heat pumps: Market overview, state of the art, research status, refrigerants, and application potentials. In: *International Refrigeration and Air Conditioning Conference, Paper 1876*. <https://docs.lib.purdue.edu/iracc/1876> (Accessed on 23 January, 2021).**Fig. 4** Coefficients of performance for output temperatures of 60 °C and 100 °C.

a stew. In this case, assume that most of the heat (97%) from the flame is delivered to the water. Example 2 is an electric range using electricity from a natural gas fired combined cycle plant where the electrical generating efficiency is 60%, the highest efficiency presently achievable. Despite that high efficiency only 60% of the fuel's combustion energy is delivered to the boiling water.

Example 3 considers a food processing plant preparing the same stew considered in Example 1. that also requires boiling water. The electricity is generated in an older plant via a steam turbine with an efficiency of 38%. The food processing plant has a high temperature heat pump that draws heat from a waste heat source (e.g., hot air exiting the plant) at 35 °C and pumps that heat up to 110 °C. In this

case the COP is slightly above 3.0, 60% of the theoretical maximum (i.e., Carnot) efficiency. In this example the HTHP provides 1.16 times more heat to the boiling water than would be furnished by the natural gas. In Example 4 we see the effect of more efficient combined cycle turbines, allowing the HTHP to provide 1.84 times more heat to the stew than would be provided by the natural gas.

Process heat is needed at different temperatures to meet various needs. In Example 5, the HTHP draws heat from a low temperature source (e.g., exhaust air) to heat water to 70 °C for washing. Even though the electricity is derived from a less efficient plant, the HTHP can deliver 2.23 times the heat of the initial natural gas combustion because of the lower output temperature. Example 6 shows a combination of air conditioning and domestic hot water heating, drawing heat from the living or working spaces at 20 °C and delivering heat to the water at 70 °C. In Example 6 the HTHP provides a service at both ends of the cycle.

Examples 7–9 track the cycles powered by 1 GJ_{th} of heat from nuclear fuel. The nuclear plant is assumed to be a Light Water Reactor producing saturated steam and having a typical generating efficiency of 32%. The industrial process is that of Example 3, but the lower generating efficiency decreases the delivered heat, though it is still nearly equal to the fuel input.

In Example 8, we see another dual-purpose cycle, providing heat for a cooking step at 80 °C and drawing the heat from a freezer at –10 °C. Finally, Example 9 tracks the performance of a cycle driven by a Very High Temperature Reactor, operating at about 800 °C and cooled by helium gas. The HTHP delivers heat at 160 °C to a distillation process while drawing its heat from a waste source at 60 °C.

The nine relatively simple examples show the flexibility and effectiveness of cycles incorporating HTHPs, pumping heat from low temperature locations where it is not wanted or is going to waste and delivering heat at the higher temperatures needed.

Requirements for refrigerants/coolants

The more important component of a heat pump or refrigeration system is the refrigerant that flows from the evaporator through the compressor to the condenser thence through the throttling valve to complete the cycle. These fluids are generally known as refrigerants even when the purpose of the cycle is to pump heat to a higher temperature.

In order to have an efficient cycle and a reasonably sized condenser and evaporator, it is generally necessary for the refrigerant to be a two-phase mixture for both the condenser and evaporator pressures and temperatures. That requirement then implies that the output temperature cannot be higher than the critical temperature, T_{crit} , for a particular refrigerant. Recall from Fig. 2 that T_{crit} is at the top of the two-phase “dome” in which the cycle operates. The top of the dome also corresponds to the critical pressure. Operating at very high pressures in order to approach T_{crit} poses challenge for design of the compressor, piping, valves and gaskets. Table 3 lists the characteristics of refrigerants suitable for HTHP applications. (Z) and (E) indicate *cis*- and *trans*-isomers respectively.

The next set of refrigerant characteristics concern the consequences of releasing the fluids into the atmosphere. Early refrigeration cycles used ammonia (NH₃), but its accidental release resulted in injuries and death to those caught in the plume. Less toxic refrigerants, the chlorofluorocarbons (CFCs) and the hydrochlorofluorocarbons (HCFCs), were developed in the 1920s as both refrigerants (Freon) and as fire extinguishing agents (Halon).

Beginning in the 1960s, CFCs and HCFCs were found to be depleting the atmospheric ozone layer that protects the surface of the earth from ultraviolet light coming from the sun. The Montreal Protocol of 1987 phased out the use of chemicals known to be depleting the ozone.

Table 3 Refrigerants suitable for HTHP applications.

Type	Refrigerant	Chemical formula	T_{crit} [°C]	p_{crit} [MPa]	Ozone depletion potential [–]	Global warming potential Myhre et al. (2013) [–]	Safety group ASHRAE (2016)	Normal boiling point [°C]	Mole-cular weight [g/mol]
HFO (Hydrofluoro-olefin)	R1336mzz(Z)	CF ₃ CH=CHCF ₃ (Z)	171.3	2.9	0	2	A1	33.4	164.1
	R1234ze(Z)	CF ₃ CH=CHF(Z)	150.1	3.53	0	<1	A2L	9.8	114.0
HCFO (hydrochloro- fluoroolefin)	R1233zd(E)	CF ₃ CH=CHCl(E)	166.5	3.62	0.00034	1	A1	18.0	130.5
	R1224yd(Z)	CF ₃ CF=CHCl(E)	155.5	3.33	0.00012	<1	A1	14.0	148.5
HC (Hydro-carbons)	R601 (pentane)	CH ₃ CH ₂ CH ₂ CH ₂ CH ₃	196.6	3.37	0	5	A3	36.1	72.2
	R600 (butane)	CH ₃ CH ₂ CH ₂ CH ₃	152.0	3.80	0	4	A3	–0.5	58.1
CF6	Novec™ 649	CF ₃ CF ₂ C(O) CF(CF ₃) ₂	168.7	1.88	0	<1	n.a.	49.0	316.0
Natural	R718	H ₂ O (water)	373.9	22.06	0	0	A1	100.0	18.0
	R744	CO ₂ (carbon dioxide)	31.0	7.38	0	1	A1	–78.5	44.0

Adapted from Arpagaus C, Bless F, Uhlmann M, et al. (2018) High temperature heat pump using HFO and HCFO refrigerants—System design, simulation, and first experimental results. In: International Refrigeration and Air Conditioning Conference, Paper 2199, <https://docs.lib.purdue.edu/iracc/2199> (Accessed on 23 January, 2021).

The hydrofluoroolefins (HFOs) and hydrochlorofluoroolefins (HCFOs) were developed as the next generation of refrigerants, though the HCFOs have small Ozone Depletion Potential (ODP) values.

Some refrigerants are strong greenhouse gases, trapping infrared radiation within the atmosphere. The values of the Global Warming Potential (GWP) for various refrigerants are based on a 100-year time horizon setting $\text{CO}_2 = 1.0$ (Myhre et al., 2013). The toxicity of the refrigerants is according to the safety groups of ASHRAE (formed at the American Society of Heating, Refrigerating and Air-Conditioning Engineers). The groups are A1: (Lower Toxicity, No Flame Propagation), A2: (Lower toxicity, Lower Flammability) and A3: (Lower Toxicity, High Flammability).

There is continuing interest in H_2O and CO_2 as refrigerants because of the abundance and low toxicity/flammability. Water has the potential for reaching higher temperatures, albeit at very high pressures (Wang et al., 2010), and carbon dioxide has been demonstrated in an integrated supermarket system in Italy (Azzolin et al., 2018).

The economics of heat pump operation

Space and water heating represent about 80% of the final energy use in the European domestic sector; thus, heat pumps represent attractive methods for decarbonization (Barnes and Bhagavathy, 2020). However, the relatively high taxes on electricity compared with those on natural gas presently have the unintended effect of discouraging investment in heat pumps. Worldwide, only about 5% of residential heat demand was met using heat pumps in 2019, though in the United States more than 40% of new single-family dwellings and nearly 50% of new multi-family buildings utilize heat pumps (IEA, 2020). Sales of heat pump water heaters have tripled since 2010, primarily due to efforts in northern China to improve air quality through the replacement of coal-fired boilers.

For the decarbonization of large scale industrial processes, Zühlsdorf et al. (2019) considered reversed Brayton cycles and steam compression, allowing delivery temperatures of 280 °C. The analysts reviewed a wide range of processes including lumber and paper drying, petroleum refining, preheating, and digesting bauxite slurry for aluminum production and spray drying in the food industry. The various HTHP cycles were evaluated on the basis of capital costs and with a tax on CO_2 emissions of 50€/t. If the electricity is supplied from zero carbon sources, the HTHP systems are more economical than the present use of natural gas for direct heating.

On-going research projects

In order to make heat pumps and especially HTHPs relevant to both residential and industrial needs in a carbon-constrained world, research is being pursued in many countries. Korea, China, Japan, Switzerland and The Netherlands are leading the experimental research on HTHPs, with the main research goals being to improve heat pump efficiency, testing laboratory functional models from 1.8 to 12 kW. Some larger prototypes are capable of producing several hundred kW of heat. Laboratory systems primarily use piston compressors (Arpagaus et al., 2016, 2018).

Technical barriers to wider market penetration include a lack of high temperature refrigerants with low Global Warming Potentials and a lack of demonstration facilities. On the marketing side there is a lack of understanding of the HTHP technology among consultants, designers, producers and installers. Furthermore, there is a lack of knowledge about the integration of HTHPs in industrial processes. Finally, low fossil energy prices foster competing technologies, despite the carbon footprint.

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Coupling Heat Storage to Base-Load Nuclear Reactors for Variable Heat, Electricity and Hydrogen

Charles W. Forsberg, Massachusetts Institute of Technology, Cambridge, MA, United States

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Introduction

The modern world was made possible by low-cost, easy-to-store, easy-to-transport fossil fuels. Fossil fuels enabled billions of people to move to the middle class. The capital costs of fossil power plants, furnaces, and other power conversion systems are low relative to the cost of fossil fuels, and the cost of fossil fuel storage is low. These two characteristics make it economic to operate these systems at part load to economically provide variable electricity, mechanical work and heat to the customer, and they enable economic integration into an electricity grid that has medium quantities of high-capital-cost low-operating-cost nuclear, wind and solar power plants. Alternately, operating nuclear, wind and solar plants at half their capacity approximately doubles the cost of energy. Utilization of fossil fuel plants to support peak energy demands enables economic operation of nuclear and renewable technologies at full capacity while meeting the need for variable electricity.

The world is undergoing massive changes in energy markets in response to a long-term goal of a low-carbon economy with limited to no carbon dioxide emissions to the atmosphere. Reaching that goal requires implementation of carbon capture and sequestration (CCS) approaches or replacement of fossil fuels in providing variable electricity and heat to match customer demands. Recent studies (Sepulveda et al., 2018) show that (1) the lowest-cost low-carbon systems include some combination of dispatchable (nuclear) and non-dispatchable (wind and solar photovoltaic (PV)) energy sources and (2) the lack of very low-cost energy storage is a major factor in higher energy costs of low-carbon systems. To economically match production with demand requires very low-cost methods to store energy. There are three classes of storage technologies: (1) work (electricity) storage: batteries, hydro pumped storage, etc.; (2) heat storage; and (3) chemical storage (hydrogen, etc.). Electricity storage best couples to the electricity grid and electricity generating technologies such as wind and solar PV. Heat storage technologies best couple to heat generating technologies such as nuclear, concentrated solar power (CSP), fossil fuel plants with CCS, and geothermal. Hydrogen storage couples to different hydrogen production technologies.

Heat storage is much less expensive than electricity storage because low-cost materials (crushed rock, liquid salts, etc.) are used. However, to date, its deployment in large systems has been limited to CSP plants. Historically that situation has been because it is inexpensive to store fossil fuels and burn them to produce heat, obviating the need for large-scale heat storage. That situation is changing. First, the goal of a low-carbon energy system requires energy storage to replace the role of fossil fuel storage in providing variable energy. Second, the deployment of significant capacity of non-dispatchable wind and solar PV has created periods of low and high electricity prices. This creates economic incentives for nuclear reactors that produce heat to include heat storage to enable selling more electricity at times of high electricity prices (high demand) and less electricity at times of low electricity prices (low demand).

This chapter first describes energy systems that incorporate heat storage to enable variable electricity and heat production with the reactor operating at baseload—the most economic mode of operation. The three components of heat storage systems are then described: (1) transferring heat from the reactor to storage, (2) heat storage, and (3) conversion of stored heat to electricity. Last, the integration of hydrogen production and storage into nuclear systems with heat storage and electricity production is described. Low-carbon nuclear hydrogen production technologies require heat and electricity, creating large economic incentives to couple heat storage with hydrogen production.

Heat storage systems

The nuclear heat storage system is shown in Fig. 1 to address hourly to seasonal needs for variable energy. The same system design can be used for any heat generating technology. To minimize the cost of energy, the reactor operates at full capacity. When electricity demand is high, resulting in high prices, reactor heat is sent to the turbine to produce electricity. When demand is low, resulting in low electricity prices, a majority of the heat is diverted to heat storage. During peak demand that exceeds the baseload reactor electricity output, combined heat from the reactor and heat storage is sent to the turbine-generator for electricity production, supported by either oversizing the turbine generator or building a separate peaking turbine-generator for peak power output. At times of very low (or negative) electricity prices, grid electricity can be converted into stored heat using resistance heaters coupled to the heat storage system. Hence, the power plant system can both sell and buy electricity. If stored heat is insufficient to meet peak demand, combustion of natural gas or low-carbon biofuels and hydrogen can provide heat to support peak electricity production.

The system design shown in Fig. 1 enables economic larger-scale use of wind and PV by several mechanisms. The large-scale addition of wind and PV in countries such as Germany and Denmark as well as states such as California has resulted in increased electricity prices (Bushnell and Novan, 2018). Large amounts of wind and PV collapse the price of electricity at times of high output and low demand, thus collapsing revenue. Some type of subsidy is required to compensate for revenue collapse. Second, added generating capacity is needed for times of low wind and solar production. In grids having only wind and solar generation, batteries or pumped hydro provides backup to wind and solar. These storage systems can be depleted and thus must be backed up with a gas turbine or equivalent generating technology. The stacked cost of assured generating capacity (wind or PV + batteries + gas turbine) is what makes systems with large quantities of wind and solar PV expensive (Sepulveda et al., 2018). The nuclear heat storage system addresses both challenges.

- **Storage.** Low-cost storage provides a sink for excess wind and solar PV production and sets a minimum price for electricity. There are two factors that enable low cost storage. First, today the cost of heat storage per unit of electricity is a factor of three to four less than storing electricity and in the future may be more than an order of magnitude less than electricity (EPRI, 2017). Unlike electricity storage technologies, relatively little work has been done to develop large-scale heat storage systems. Second, the cost of converting electricity to stored heat is much lower than any other system. Deployment of large-scale wind or solar results in times where some wind and solar is shut down when production exceeds demand. This creates large economic incentives to find ways to store this excess energy. To sell electricity the power plant has switchgear, transformers, and grid connections. The same equipment is used to import electricity into the plant for conversion into heat at no added cost.
- **Generating capacity.** In the above system, if the generating capacity equals peak demand, one can assure meeting electricity demand—no stack of generating technologies is needed to provide assured capacity. A low-cost combustion burner (rather than a more expensive gas turbine) provides the assured peak generating capacity. The fuel would be natural gas or low-carbon hydrogen or biofuels.

The minimum-cost system depends upon (1) the capital and operating costs of nuclear, heat storage, wind, and solar, (2) the quality of local wind and solar resources, and (3) the heat and electricity demand on an hourly to seasonal basis. The optimum ratios of installed capacities of each generating technology depends upon local conditions. If a particular market has a high demand for air conditioning that matches solar output, the optimal system design will support higher PV capacity. If there is a large demand for industrial heat, this will favor installation of thermal generators, such as nuclear systems. Nuclear reactors produce heat. The laws of thermodynamics (Carnot cycle) implies several units of heat are required to make a unit of electricity. Consequently, nuclear reactors produce low cost heat and more expensive electricity. Electricity can be converted to heat using resistance heating resulting in

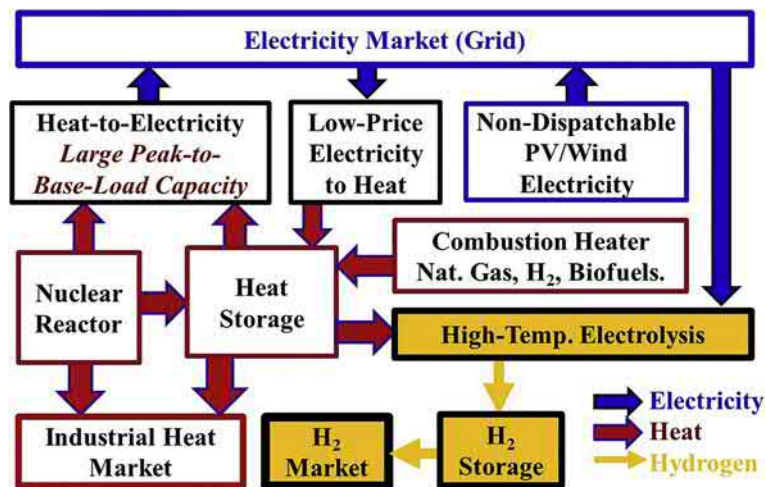


Fig. 1 Base-load nuclear, wind and solar with heat storage to provide variable heat, electricity and hydrogen.

one unit of electricity producing one unit of heat. Wind and solar PV produce electricity that results in expensive heat. If everything else is equal, the different conversion efficiencies of heat-to-electricity versus electricity-to-heat favor heat-generating technologies if the customer needs heat and electricity-generating technologies if the customer needs electricity. Today fossil systems are very similar worldwide. The relative amounts of nuclear, wind, solar and heat storage will vary with location.

The system in Fig. 1 can also be used for cogeneration of electricity and heat. Cogeneration has significant implications for the electricity grid because it directly links the industrial heat market to electricity markets. Some industrial processes can operate flexibly or vary heat demand, freeing up heat for electricity production when needed and consuming added heat at times of low electricity demand. Coupling the industrial sector with the electricity sector via storage adds another dimension to balancing energy production with demand.

As discussed in other chapters (see Suppiah, 2021; Boardman, 2021), energy can be stored in the form of hydrogen. One option is to store hydrogen underground in the same facilities used for seasonal natural gas storage; thus, enabling seasonal storage of energy in the form of hydrogen. As described later, there is a large and growing industrial demand for hydrogen, and it can also be used as a low-carbon replacement fuel for fossil fuels. The leading candidate for nuclear hydrogen production is high-temperature (steam) electrolysis (HTE) of water (Pinsky et al., 2020)—a process that requires heat and electricity (see Suppiah, 2021). Hydrogen production facilities are capital intensive, not just due to the production process but also the compressors, pipelines, and associated facilities—all with large economics of scale. Hence, it is uneconomic to operate such facilities at low capacity factors. This may require that plants producing hydrogen operate more than 80% of the time (Boardman et al., 2019). The HTE plant can be embedded into a system that includes nuclear and renewable generators and heat storage (Fig. 1). At times of low electricity prices, electricity from the grid, or electricity produced by the nuclear reactor, can be used for HTE while heat from the nuclear plant is directed to storage and the HTE plant for hydrogen production. At times of high electricity prices, heat from the reactor and heat storage produce peak electricity with no hydrogen production.

This nuclear heat-storage hydrogen system has three key characteristics. First, because large-scale hydrogen storage is inexpensive using natural-gas storage technologies, stopping hydrogen production does not disrupt hydrogen supply to the customer. Second, the system design allows low-value wind and solar electricity, when available, combined with heat from the nuclear reactor to produce higher-value hydrogen. Any excess heat from the nuclear plant is directed to lower-cost heat storage for later use. Capital-intensive nuclear, wind, solar, and hydrogen facilities are fully utilized. Third, it enables the nuclear plant to operate at full capacity, producing peak electricity 20% of the time and hydrogen for as much as 80% of the time. The capital cost of hydrogen production facilities is high; thus, economic operation of the hydrogen plant requires high capacity factors (Lucid Catalyst, 2020). This enables the nuclear plant to economically address seasonal peaks in electricity demand that total a hundred to a thousand hours per year.

There are many heat storage technologies that can be incorporated into the described system (Forsberg, 2019; Forsberg et al., 2019, 2020b). At present, all of the large-scale commercially-deployed heat storage technologies were developed for concentrated solar power (CSP) systems, but their characteristics make them suitable to nuclear integration as well. The largest CSP systems store heat in liquid nitrate salts where the temperature varies from 285 to 565 °C. In these systems (Fig. 2) cold nitrate salts enter the CSP system, are heated in tubes exposed to concentrated sunlight, and are sent to the hot-salt storage tank. Hot nitrate salt is sent to steam generators where water is converted into steam to drive the power cycle. The resulting cold salt is sent to the cold-salt storage tank, and ultimately back to the CSP system to be reheated. In large CSP systems, the heat storage capacity is several gigawatt hours (GWhs).

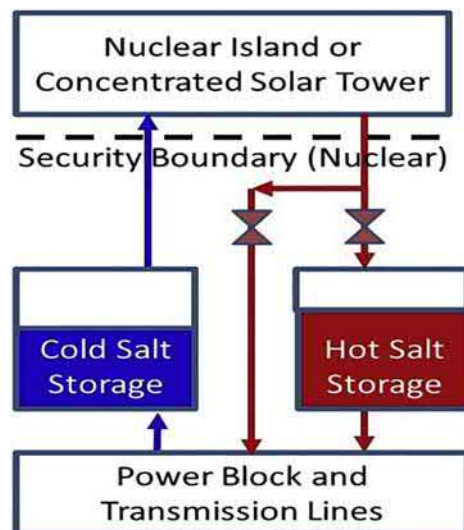


Fig. 2 System design for CSP and nuclear with heat storage.

Storage accomplishes several functions. Peak electricity output is separated from the heat generating system and sized to match market needs. The peak electricity output may be much larger than the maximum output of the heat generating technology. Second, incorporation of energy storage simplifies operations. On partly cloudy days, the power output of a CSP system varies depending upon whether the clouds are blocking the sun. Without heat storage the power output would go rapidly up and down with changing cloud cover. In the nuclear plant, the reactor can operate at steady state independent of variable demand for electricity or any electricity grid disturbances.

Nuclear reactor heat to heat storage and power cycles

In a typical storage system, such as a pumped-hydro-electricity facility, there are losses associated with pumping the water uphill and losses when the water goes through the turbine. As a consequence, the round-trip efficiency is 75–80%; that is, for every kWh of electricity going into the pumped-hydro system, 0.75–0.85 kWh of electricity is sent back to the grid. These losses may or may not exist for heat storage systems. In the above CSP example (Fig. 2), the nitrate salt is heated and sent directly to storage. There is no temperature loss across a heat exchanger. There are minimal losses in going from storage to the power cycle—the power cycle does not know if the nitrate salt came from the CSP system or heat storage. There are heat losses during storage through the tank insulation, but these are typically 1–2% per day. The round-trip efficiencies in such heat storage systems can be greater than 95%, which is significantly better than batteries or pumped hydro facilities.

The same strategy can be used for some but not all storage systems coupled to a nuclear reactor. Many reactors require intermediate heat transfer loops to separate the low-pressure reactor from the high-pressure power cycle. Most designs of salt-cooled reactors now propose (Forsberg et al., 2020a, b; Forsberg, 2020) to use a nitrate-salt intermediate loop that can support heat storage—the same technology used in CSP plants. Nitrate loops achieve three purposes: separate the high-pressure power cycle from the low-pressure reactor, heat storage, and trapping of tritium. Many of these reactors produce tritium that can escape via the power cycle. If tritium diffuses through the heat exchangers into the power cycle, the tritium is oxidized, converted to tritiated water, and removed from the nitrate off-gas as steam. Intermediate nitrate salt loops are also proposed for some sodium fast reactors and for advanced fusion reactors with salt blankets. In these systems (Forsberg et al., 2020b) heat storage completely decouples the reactor from the power system, having multiple implications:

- The peak power output is independent of the reactor output.
- Multiple turbines may be used, where one turbine is designed to operate most of the time and is designed for high heat-to-electricity efficiency and a second turbine is designed as a peaking turbine with somewhat lower efficiency but lower capital costs.
- The power cycle is separated from the reactor without the constraints imposed by coupling the reactor and turbine. The turbine can be built to non-nuclear commercial standards and be on automatic control by the grid operators because the reactor does not “see” any transients from the operation of the turbine.

Heat storage can also be incorporated into the power cycle rather than between the reactor and the power cycle. Westinghouse and others have been examining heat storage coupled to light water reactors (LWRs) with peak power obtained by oversizing the main turbine generator. In this system, the electricity output may vary from 30 to 130% of baseload output. At times of low electricity demand, high-temperature steam transfers its heat through a heat exchanger to a heat-transfer oil. These oils are stable to temperatures up to 400 °C at atmospheric pressure. Peak LWR temperatures are about 300 °C. Heat transfer oils have been used in the chemical industry for almost a century and are used in lower-temperature CSP systems. The hot oil transfers its heat to high-temperature concrete, crushed rock or another low-cost storage material. This design minimizes the inventory of the more expensive heat transfer oil. The process is reversed to send steam back to the power cycle. Round trip efficiencies up to 80% (Amuda and Field, 2020; Kluba and Field, 2019) are possible, implying that the electricity produced from heat in storage is 80% of the electricity that would have been produced sending steam directly to the turbine generator.

Because of temperature losses in the heat storage cycle, steam returning from storage is at a lower temperature than the steam sent to storage. In an LWR saturated steam cycle under normal operations, steam is sent through multiple turbines with up to one third of the steam extracted from the turbines at different temperatures to heat feedwater. Cold feedwater from the condenser is heated by multiple feedwater heaters before entering the steam generator in a pressurized water reactor or returned to the reactor in a boiling water reactor. Preheating cold feedwater improves steam cycle efficiency. Lower-temperature steam from storage matches the temperatures of steam extracted from the steam turbine and can be used in the feedwater heating system. Reducing steam extraction from the turbines for feedwater heating increases the power of the turbine, producing peak electricity.

Heat storage technologies

Many heat storage technologies are under development for nuclear and CSP systems, but only a few have been commercially deployed at scale in CSP systems. Two recent workshops (Forsberg, 2019; Forsberg et al., 2019) summarize the current status of different systems for nuclear applications. Heat can be stored as sensible heat, latent heat, and in chemical systems. All commercial large-scale systems use sensible heat because of its lower capital costs.

There is no single heat storage system that will optimally meet all needs. Different reactor types operate at different temperatures; thus, only some heat storage systems couple to specific reactors. Second, the optimum system depends upon the specific market. If a nuclear reactor is in a utility system with excess electricity produced during the day, the economics will favor systems with daily storage. If a nuclear reactor is in a system where there are large weekday/weekend variations in demand, another storage technology may be preferred that has a somewhat lower capital cost per unit of heat storage capacity but with a lower round-trip efficiency. If a heat storage system is used daily, the storage system earns revenue 365 days per year. If a heat storage system is used once a week, the storage system earns revenue 52 times per year. The economics requires that heat storage systems used fewer times per year have much lower capital costs. In practice, this results in using heat storage systems that may be less efficient for long-term storage if their capital costs are much less. Several leading candidates are described below.

Liquid salts

The primary heat storage materials used in high-temperature CSP systems are nitrate salts, with solar salt (60 wt% NaNO_3 –40 wt% KNO_3) being the most common salt. Sensible heat storage is typically obtained by varying temperatures from 290 to 565 °C. With control of gas compositions over the salt storage tanks and control of the salt chemistry, salt storage temperatures of 600 °C may be viable. These salts are chemically stable in air. Heat storage system capital costs in CSP systems are ~\$20/kWh of heat. The largest CSP deployed storage system sizes are measured in gigawatt-hours of capacity. Nitrate-salt intermediate-heat-transfer loops are currently proposed for Sodium Fast Reactors (SFRs), Fluoride-salt-cooled High-Temperature Reactors (FHRs) with solid fuel and clean salt, thermal-spectrum molten salt reactors (MSRs) with fuel dissolved in the salt, and fusion machines with salt blankets. Different companies either include heat storage in the baseline design or as an option: SFRs (TerraPower), FHRs (Kairos Power), and MSRs (Moltex, others). Nitrate salts can also be used to move heat to industrial customers at higher temperatures than heat transfer oils but has not been commercially deployed.

Work is underway (Mehos et al., 2017) to develop second generation salt systems that would allow CSP systems to operate at peak temperatures of ~750 °C with higher-temperature stored heat. The goal is to have a pilot plant by 2025. The proposed salt is a sodium, potassium, magnesium chloride eutectic with a melting point of 383 °C. This salt was chosen because of its extremely low cost combined with reasonable physical properties. Allowable peak operating temperatures could exceed 1000 °C. The very low cost is because of (1) the low cost of the raw materials and (2) a potassium magnesium salt that is used in the large-scale production of magnesium. Hence, there is an industrial supply chain. There are two challenges: proper control of chemistry to minimize corrosion and development of a low-cost reliable tank insulation system. In a nitrate salt storage tank, relatively low-cost steels can be used so the tank insulation is on the outside of the tank. At much higher temperatures the low-cost storage option is to put the insulation on the inside of the storage tank so low-cost steels operating at lower temperatures can be used in large tanks. Ceramic insulation is used on the inside of salt storage tanks in industry, but it is not available at the sizes required for heat storage. Chloride salt storage is proposed to be used with molten chloride fast reactors with reactor peak temperatures near 750 °C. The chloride storage salts would also couple to higher-temperature HTGRs.

Heat transfer oils

Lower-temperature CSP systems use heat transfer oils, such as Therminol-66[®]. These systems have operating temperatures below 400 °C—the upper temperature limit for these oils. These heat storage systems are compatible with existing LWRs with peak temperatures of ~300 °C. These oils are stable, well understood, low vapor pressure to minimize fire hazards and over 50 years of industrial experience. Their major drawback is their cost when used for heat storage; thus, the incentive to couple with a lower cost storage material such as concrete or crushed rock.

Crushed rock and concrete

The costs of liquid salt and heat-transfer oil heat storage systems can be reduced with the use of a lower-cost filler material. Both crushed rock and special high-temperature concretes are being considered. Concrete can be formed into specific shapes such as monoliths with narrow channels to minimize the inventory of heat transfer fluid. Crushed rock is the lowest-cost fill material but has a higher void volume.

The existing CSP systems with hot and cold tanks of nitrate salts or oils can be viewed as first-generation heat storage systems. The second generation systems under development fill the storage tanks with lower-cost solid materials. Westinghouse is examining a system for LWRs where steam is used to heat oil that, in turn, transfers its heat to concrete in prefabricated boxes filled with closely packed concrete plates with small cooling channels between the plates (Forsberg, 2019). This design minimizes the inventory of the more expensive heat transfer oil. At times of high electricity demand, the oil transfers heat back to the steam cycle as described above.

Korea (Amuda and Field, 2020; Kluba and Field, 2019) is examining a similar system for LWRs that uses crushed rock as the heat storage material. The design would include many tanks of crushed rock with heat-transfer oil only in tanks where heat is being transferred to the crushed rock or from the crushed rock. At any other time the different tanks would contain hot or cold crushed rock (without oil). This reduces the inventory of expensive heat-transfer oil to what is in the piping and tanks where one is adding or removing heat from the crushed rock. Most tanks at any one time have no oil in them—either hot or cold rock. Round-trip

efficiencies can approach 80%; that is, if a megawatt hour is generated without storage, 0.8 MW hours of electricity is generated from the stored heat. The Korean design proposes that the storage system be built as a large barge (60 m by 450 m) with multiple tanks having a total heat storage capacity of 20 GWh of electricity. The barge, the size of a supertanker, would be delivered to coastal nuclear power sites where it would be floated into a dry dock at the reactor site. Hot-oil heat transfer also allows easy coupling to industrial heat customers.

Germany (Odenthal et al., 2019) is experimentally examining nitrate salts in single tanks filled with crushed rock with lower-density hot salt on top of higher-density cold salt. This is a thermocline heat storage system. The crushed rock is both a heat storage material and helps minimize mixing between hot and cold liquid nitrate salt. The single tank reduces costs relative to the use of separate hot and cold salt tanks.

In third generation systems, where only limited work has been done (Forsberg, 2020), heat is stored in crushed rock in insulated and covered trenches up to 60 m wide, 20 m high and a kilometer long. A leak-tight roof over the trench provides for insulation and allows a controlled atmosphere. For systems coupled to LWRs using oil as a heat transfer medium, every 10 m of crushed rock provides about one GWh of heat storage assuming a 200 °C hot-to-cold temperature swing. Total system capacity could exceed 100 GWh of heat storage. When there is excess energy available, some of the steam from the reactor can be used to heat oil rather than being sent to the turbine-generator to produce electricity. The hot oil is sprayed over sections of crushed rock to heat the rock as it flows down to the oil pan under the crushed rock. The oil is collected and cycled back to the reactor to be reheated. Crushed rock is heated sequentially by section over time (Fig. 3). At times of high electricity demand, cold oil is sprayed on the hot rock, flows through the rock, collected by the oil pan, and used to convert water into steam. A nitrogen gas atmosphere is maintained above the crushed rock to prevent degradation of the heat-transfer oil. The steam is sent to a peaking steam turbine-generator to produce electricity. There is an equivalent parallel system for nitrate salts that operates at higher temperatures approaching 600 °C. The atmospheric composition is controlled to minimize nitrate salt degradation with time.

With the oil and salt systems, appropriate rocks must be chosen to avoid chemical reactions and to ensure appropriate mechanical integrity under temperature cycling conditions. Liquid clean-up systems will be needed to remove rock fines and other impurities that will build up over time. This system with oil or salt has the potential to be the lowest cost system (Forsberg et al., 2020b). Crushed rock is the lowest cost heat storage material available. Flowing the salt or oil over the rock rather than filling all the void spaces in the rock minimizes the inventory of more costly heat transfer fluid. The large size minimizes the surface to volume ratio and thus the cost of insulation and structural materials. Significant development work is required to elevate this system to commercial readiness.

Other researchers are also investigating crushed rock for GWh heat storage systems because it is the lowest-cost heat storage material. Siemens (Kosowatz, 2020) is developing a hot rock heat storage system where air is heated by electric resistance heaters at times of low electricity prices, and the hot air is blown through the crushed rock to heat it to 650 °C. At times of high electricity prices, cold air is blown through the hot rock to produce hot air for a steam boiler.

The U.S. Department of Energy goal for the capital cost of heat storage systems is \$15/kWh of heat. The commercial nitrate salt storage system costs are approaching \$20/kWh. The projected costs for second generation nitrate-salt crushed-rock heat storage systems are near \$10/kWh, whereas predicted capital costs for some of the third generation systems are only a few dollars per kWh (Forsberg, 2020) in sizes to 100 GWh to enable weekend/weekday storage. At present, commercial heat storage system costs per unit of electricity are a factor of three to four less than electricity storage technologies. Advanced heat storage technologies have the potential to be an order of magnitude lower cost than electric storage technologies in the future, reflecting lower cost materials of construction (i.e., crushed rock and thermal salts versus lithium, cobalt, carbon, or steel) and higher operating efficiency.

Cast iron with cladding

Sensible heat can be stored in tanks filled with solid tightly-packed hexagonal assemblies 10–20 m high made of cast iron, with a stainless steel or other cladding chosen for chemical compatibility to match the coolant—sodium, salt, lead or helium. The pumped coolant flows between the solid assemblies. It is compatible with any reactor coolant with the appropriate choice of cladding. The storage system can be located in the primary system or intermediate reactor heat transfer loop. It may be particularly attractive for SFRs with its low-pressure secondary loop because (1) iron is considerably lower cost than sodium and (2) the fire hazard associated with storing large volumes of sodium is eliminated. The geometry is similar to a SFR reactor core that has hexagonal fuel

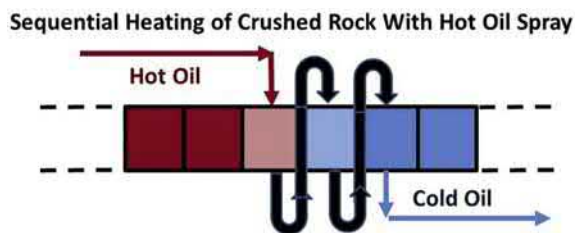


Fig. 3 Sequential heating crushed rock bed with hot oil.

assemblies; thus, the behavior of such geometries as a function of temperature is well understood. Cast iron is inexpensive, thus the key economic question is a manufacturing question: how far can one drive down the cost of large ingots clad with stainless steel when mass produced? Such storage systems have higher capital costs than crushed rock systems but have very small temperature losses from reactor through heat storage to the power cycle. Such systems are potentially attractive for daily heat storage but not for longer time periods.

There is ongoing work to develop advanced CSP systems that use sodium rather than nitrate salts. Sodium allows higher operating temperatures, has better heat transfer properties and does not thermally decompose if heated. The CSP sodium temperatures would be higher than in SFRs because there are no safety implications if sodium boils. Sodium systems enable the use of alternative power cycles including open-cycle gas turbines with (1) higher-temperature heat and (2) higher-performance lower-pressure-drop sodium-air heat exchangers. There is work necessary to develop gas turbines that would couple to SFRs and is directly applicable to sodium CSP systems (Forsberg, 2020). The potential to couple to lower-cost gas turbine power cycles creates significant incentives to develop such a storage system for sodium CSP plants that would be directly applicable to SFRs.

Geothermal heat storage

In a geothermal heat storage system thermal energy is stored by injecting hot water heated by steam from the reactor into an underground reservoir; energy is discharged by pumping hot pressurized water back to the surface for electricity production in a conventional geothermal plant. Only limited studies have been completed (Lee et al., 2010; Lee, 2011; Forsberg, 2012), but there is a large experience base for the key technologies. Steam and hot water injection is used to recover heavy oil. Many geothermal power plants exist around the world. This heat storage technology has different characteristics than the other heat storage options described, namely:

- *Gigawatt year capacity.* Geothermal storage can provide seasonal energy storage but can only be deployed as a large system because there is no way to insulate rock deep underground. In an ideal system to minimize heat losses, the heated rock zone would be spherical to minimize the surface (heat loss) to volume (heat capacity) ratio. Practical systems have underground geologies that are large rectangles. The minimum heated rock zone to reduce conductive heat losses has dimensions of hundreds of meters. The minimum capacity of heat storage system to avoid excessive heat losses is greater than 0.1 GW·year—far larger than any other heat storage system. The system size can be tens of gigawatt years. The round trip efficiency of these systems is lower than other options, but the incremental capital cost is extremely low.
- *Geology limits.* The right geology is required to implement geothermal storage. The geology must be reasonably permeable to water flow. There is also the requirement that the rock solubility in the fluid be reasonably low to prevent dissolution of the rock and excess dissolved solid contents in the fluid.

Steam power cycle options

There are several heat storage systems that are intrinsically coupled to the power cycle. In a steam accumulator (Forsberg, 2019) there are pressure vessels filled with cold water. At times of low electricity prices, steam is injected into the water, raising its temperature to the boiling point. For a LWR, the pressure would match the peak steam pressure. At times of high electricity prices, valves are opened, the water flashes to steam, and the steam is sent to turbines. Banks of pressure vessels are used to provide steam to turbines at different pressures. Steam accumulators have been used for a century for various applications; thus, the technology is well understood. This experience base includes the building and operating for several decades of a peaking steam accumulator in Berlin that was coupled to a coal plant but was shut down when the coal plant was shut down. At times of low electricity demand, the steam accumulator would be charged to produce peak power when required. Steam accumulators do have one unique property—the response time can be very short with fast increases in power levels. There have been limited recent studies to understand the economics in the context of capital costs and markets.

An additional option is to use cold water for peak power production. On hot days the power output of nuclear reactors decreases as cooling water temperatures increase and steam condenser temperatures increase. At times of low electricity prices, water can be cooled to near its freezing point and stored. At times of high prices, the cold water is used to lower steam condenser temperatures to increase turbine output. Heat storage costs can be very low by storing water in deep ponds where the cold water stays at the bottom of the pools. There are also options to put a floating insulator on top of the water. The system costs are driven by the cost of refrigeration. The economics of this system are strongly dependent upon how many hot days per year and thus the number of storage cycles per year (Forsberg et al., 2019).

Other options

There are several classes of thermochemical systems where heat is stored in chemical bonds. In hydride systems heat is used to decompose a hydride, producing hydrogen that is stored. When the reaction is reversed, the formation of hydride releases heat. In carbonate systems heat is used to decompose a carbonate, such as calcium carbonate, into calcium oxide and carbon dioxide. The carbon dioxide is stored. When the reaction is reversed to form the carbonate, heat is released. Finally, there are a set of chemical reactions that involve forming hydrates where steam is released when the hydrate is heated and heat is generated in the reverse direction. All of these systems are at much earlier stages of development.

Relationship between storing heat, work (electricity) and chemical (hydrogen) energy

The broad classes of energy storage systems are based on thermodynamics: heat, work, and Gibbs free energy. Their fundamental characteristics are reflected in the design of storage systems.

Heat storage couples to thermal generation technologies (e.g., nuclear, fusion, geothermal, fossil fuel with CSS) and, on the product side, with power conversion technologies to produce electricity and with heat users (industry, commercial, etc.). Heat storage is inexpensive because the materials of construction of heat storage systems have very low costs (rock, salt, etc.). Converting heat to work requires power cycles (steam, supercritical carbon dioxide, gas turbines, etc.). It takes time to transfer heat to the power cycle and for power cycle startup; thus, response times are slow—minutes to hours. The response times are faster if the power cycle is operating at part load. There are massive economics of scale that can be leveraged. The characteristics of heat storage suggest that it has the potential become the low-cost primary large-scale storage technology for storage periods from 1 h to 1 week at sizes measured from a few megawatt hours to a hundred gigawatt hours.

Electricity storage technologies couple to electric generating technologies (e.g., wind, solar PV) and the electricity grid. Work is a more valuable form of energy than heat because it takes several units of heat to produce one unit of work. Electricity storage systems (batteries, capacitors, etc.) are more expensive because their construction requires the use of more expensive materials in more refined engineered forms (batteries, flywheels, etc.). Most of these systems have fast response and thus are likely to become the preferred storage technologies if rapid response is required for short periods of time (frequency control, short peak times, etc.).

The two exceptions to these broad characterizations are hydroelectricity and nuclear geothermal heat storage. Each of these technologies is dependent upon geology; hence, their performance and cost are more dependent upon the specific site geology and topography than the technology. Nuclear geothermal systems require a permeable rock formation several hundred meters underground and rock formations with limited rock-water interactions. Hydroelectricity systems require impermeable rock near the dam site to avoid water flow around the dam site and/or dam failure. Topography is only one limitation on siting hydroelectric facilities of any type.

The third class of energy storage technologies is chemical storage, where the primary chemical form is hydrogen (IEA, 2019). There is a large existing demand for hydrogen because of its chemical characteristics. The U.S. consumes 10 million tons of hydrogen per year to produce liquid fuels, chemicals, and fertilizer. Recent studies (Ruth et al., 2020) have examined alternative hydrogen futures with high-end scenarios increasing hydrogen production by a factor of 10 with 17% of total energy production being used for hydrogen production. Hydrogen is a chemical reagent in the production of fertilizer, metals, and liquid fossil fuels where in most cases there is no substitute. It is likely to replace fossil fuels as a chemical reducing agent in the production of steel and other materials, reducing carbon dioxide emissions associated with steel manufacturing by as much as 90% (Millner et al., 2017). Other large future markets include production of high-quality biofuels, use in fuel cells for vehicle transport, and hydrogen combustion as a high-temperature heat source for industry. In each of these three future markets hydrogen is used as an energy source and there are competitive alternatives.

Today hydrogen is primarily produced from fossil fuels. Hydrogen is used on a large scale, moved long distances by pipeline in large volumes and stored at low costs in underground facilities. There are two primary hydrogen production options: reforming and water splitting technologies. Steam methane reforming (SMR) of fossil fuels (where inclusion of CCS reduces the carbon emissions) is the predominant method of hydrogen production today. Hydrogen is a chemically reduced form of methane (CH_4), whereas for water-splitting processes hydrogen is in its oxidized form—water (H_2O). In SMR natural gas and steam are converted to hydrogen and carbon dioxide, taking less energy than electrolytic processes. In a low-carbon world SMR will likely be economic in locations with low natural gas prices and good carbon sequestration sites, partly because the resultant carbon dioxide from the process is concentrated. Water electrolysis for hydrogen production is accomplished via low temperature electrolysis (LTE) of water using electricity, high-temperature electrolysis (HTE) of steam, or thermochemical hydrogen production from water with heat input (see Suppiah, 2021; Boardman, 2021). These processes are less technically mature than SMR. HTE has significant economic advantages (IEA, 2019; Pinsky et al., 2020) because (1) part of the energy input is in the form of steam that costs less than electricity, (2) no expensive catalyst is required and (3) it is more efficient. While one cannot predict technological futures, the expectation is that HTE will likely become the low-cost electrolytic route. There are large incentives to integrate heat storage and hydrogen. At times of low-electricity prices heat from the reactor goes to heat storage and HTE. That heat and electricity from the grid produces hydrogen. This is more efficient than converting excess grid electricity to stored heat and back to electricity where the round-trip efficiency is below 50%.

Hydrogen production facilities are capital intensive, not just due to the production process but also the compressors, pipelines, and associated facilities—all with large economies of scale. Nuclear-driven hydrogen production facilities show technical and economic potential in existing U.S. markets (Frick et al., 2019; Boardman et al., 2019), and U.S. utilities are currently working to demonstrate these technologies at existing LWR power plants.

Conclusions

Historically, fossil fuels provided variable electricity and heat to meet human needs. This was economic because fossil fuels are storable, the capital cost of power conversion equipment is relatively low, and most of the cost was in the fuel. Hence, it is economic to operate fossil power plants and boilers at part load. In a low-carbon economy energy storage is required to match energy production

with energy demand to minimize the energy production costs from high-capital-cost low-operating-cost nuclear, wind and solar. Operating these technologies at part load is expensive.

Nuclear reactors produce heat and thus couple to heat storage technologies. The cost of heat storage today is a factor of three or four less than storage of electricity. The cost of advanced heat storage systems may be an order of magnitude less than electricity storage systems; however, the development of these technologies lag electricity storage technologies. This is because markets for electricity storage have existed for a century, whereas interest in thermal storage is only more recent. The low cost of heat storage has the potential to provide nuclear energy with a large competitive advantage to provide variable heat and electricity.

Finally, very low-cost nuclear heat storage may be the enabling technology for the larger-scale use of wind and solar by storing excess energy at times of low prices to be used as heat and electricity when needed. A nuclear system with heat storage (Fig. 1) has the electrical interconnection to export peak electricity when needed. The same transmission lines, transformers and other equipment enable receiving electricity from the grid at times of low demand. The same peaking power units are used for heat from storage whether it came from the reactor or electricity purchased from the grid. The primary added cost is incremental expansion of the heat storage system—making the system a very low-cost option to beneficially use any excess electricity from wind and solar. The system design is as important as the design of the specific storage technology.

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Process Heat for Chemical Industry

Richard D Boardman^a, Michael G. McKellar^b, Brian D. Dold^a, Andrew W. Foss^a, and Haydn C. Bryan^a, ^a Idaho National Laboratory, Idaho Falls, ID, United States; and ^b University of Idaho, Idaho Falls, ID, United States

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Glossary

Brayton cycle A thermodynamic work cycle named after George Brayton that describes the workings a gas compression, gas heating and gas expansion turbine a gas cooling source, and gas compressor. Paired to a heat source. A once-through air Brayton cycle draws and discharges air from the ambient atmosphere. A closed Brayton cycle incorporated a low-temperature heat rejection source.

Exergy The maximum theoretical useful heat and work (pressure-volume, shaft, or electrical) obtainable as the system is brought into complete thermodynamic equilibrium with the thermodynamic environment while the system reacts with this environment only.

High temperature gas cooled reactor A nuclear reactor with a fuel core that is cooled by a gas, such as helium, nitrogen, or carbon dioxide (HGTR).

Light water nuclear reactor Nuclear reactor cooled by normal water (H₂O); either a pressurized water reactor (PWR) with steam generators that typically produces steam up to 330 °C and 17.4 MPa or a boiler water reactor (BWR) that typically produces steam up to 290 °C at 7.6 MPa.

Microreactor A nuclear reactor with a thermal energy output of ranging from one to 20 MW (MW) electrical energy generation.

Molten salt reactor A nuclear reactors with a fuel core that is cooled with a molten salt, such as eutectic mixtures of fluoride/lithium/beryllium (e.g., FLiBe) or sodium/potassium nitrate (e.g., NaNO₃-KNO₃) or potassium/fluoride/zirconium (e.g., KF-ZrF₄).

Next generation nuclear plant A concept for a project to provide nuclear energy to industrial processes; based on the high temperature gas-cooled reactor used for Next Generation Nuclear Plant (NGNP) concepts.

Small modular nuclear reactors A nuclear reactor with a thermal energy output of ranging from 20 to 300 MW (MW) electrical energy generation. Modules may be grouped to form a complex of reactors rivaling the output of traditional large-scale nuclear reactors (MSR).

Subcritical steam Steam at conditions below the critical point of 374 °C and 22.10 MPa.

Supercritical steam Steam at conditions above the critical point of 374 °C and 22.10 MPa.

Ultra-supercritical steam Steam at conditions which are typically greater than 600 °C (and ranging up to 760 °C) and in excess of 24 bar (and ranging up to 34 bar).

Introduction

The Section Introduction discusses opportunities and gaps for application of nuclear produced thermal energy for the process industries. The most common methods of heat deposition in the process industry are performed with fired heaters that provide hot exhaust for gas-to-gas and gas-to-liquid heating, chemical reactor vessel jackets, steam reforming processes, and high temperature processes such as glass, cement, and steel making processes. It is estimated that about 70% of the energy supplied to the process industries is derived by combustion of fuels, including process tail gasses that are conveniently burned to raise heat while avoiding pollutant emissions. It is therefore imperative that strategy intent on achieving deep decarbonization of industry look for a substitute for fossil fuel combustion. Moreover, it is important to note that unlike the transportation sector and electrical grid, energy use by

industry often involves direct conversion of primary energy sources to thermal and electrical energy at the point of consumption. The heterogeneity and variations in scale of industry and the complexity of modern industrial firms' global supply chains are among the sector's unique challenges to minimizing its greenhouse gas (GHG) emissions. Because nuclear energy is essentially GHG-emissions free, concentrated, high quality, and wholly dispatchable, it is a strong candidate for providing process heat for the chemical industries.

This section necessarily begins with a review of best fit industries for initial application of nuclear-supported process heating. McMillan et al. (2016) composed the most complete review to date on process heating that can be supported by solar, geothermal, and nuclear energy sources. The diversity of processes within the industry represents many potential applications for nuclear technologies. Many chemical processes have heat needs within the ranges generated by reactor designs.

Application of nuclear energy to industry is motivated by the same drivers as electrification of industrial manufacturing: reducing overall emissions from the industrial sector. However, the greatest thermodynamic efficiencies are ultimately attained by direct application of the thermal energy produced by nuclear reactors. This is borne out by understanding and applying a thermoeconomic analysis. Application of the 1st and 2nd Laws of classical thermodynamics leads to an assessment of the disposition of the exergy input into the system. Exergy is the maximum theoretical useful heat and work (pressure-volume, shaft, or electrical) obtainable as the system is brought into complete thermodynamic equilibrium with the thermodynamic environment while the system reacts with this environment only (Tatsaronis, 1993). This necessitates a systematic analysis of both the work-producing (e.g., electrical power) and work-adsorbing processes (e.g., a gas liquefaction column or a chemical product reactor) unit operations.

Industrial heat requirements

Fig. 1 shows the breakdown of emissions and the concentration and level of heat use by major industrial sectors. While the potash/soda ash/borate mining industry has relatively low total emissions and relatively few facilities over the reporting threshold, each plant consumes a large amount of thermal energy. Petroleum refineries account for the most emissions among considered industries, with many refineries in the U.S. consuming moderate levels of thermal energy.

Approximately one-half of the heat used by industry is provided by package boilers to raise steam, which is the predominate medium of heat transport and heat transfer. Space heating accounts for about 20% of the total fuel usage, most of which is for boilers in houses.

The majority of steam duties simply require subcritical steam. Some applications may utilize supercritical steam, but it is rare for any process to use ultra-supercritical steam, except in power generation, given the corrosivity of high temperature steam which necessitates expensive alloys. For very high temperature processes, either direct or indirect heating with combustion gasses is the most common practice. However, with the advent of advanced, high temperature reactors, it is desirable to substitute steam duties with non-corrosive hot gasses. Inert gasses such as He and CO₂, or even heated air, are some choices.

The common feature of all process industries is that they aim to convert raw materials into a new form by means of chemical and physical changes. Some of these "processes" are listed in Table 1. Most require external heat, some in very large quantities.

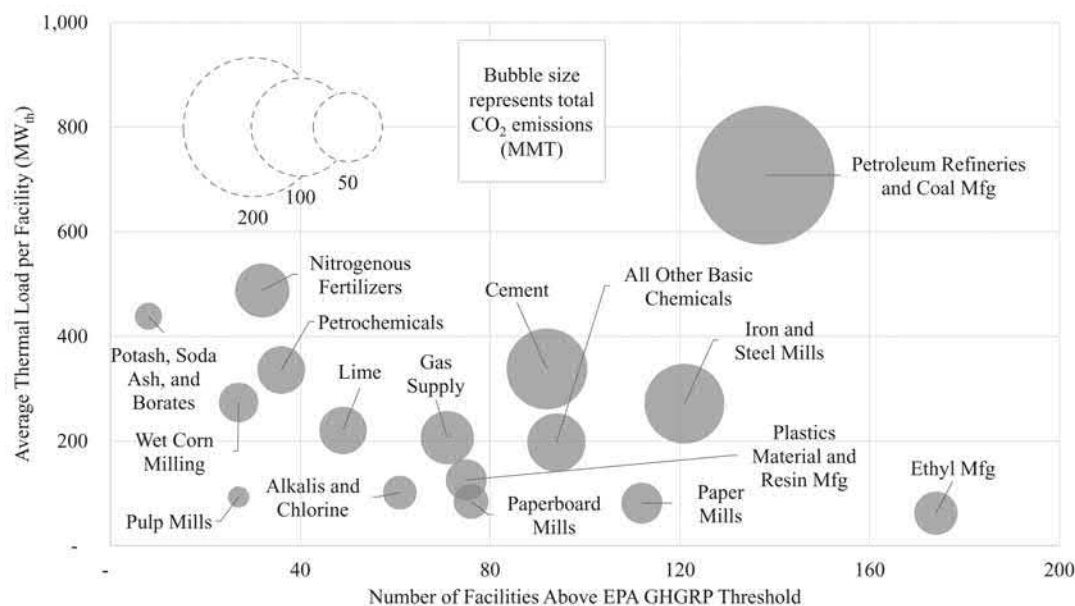


Fig. 1 Size and emissions of selected industries in the United States.

Table 1 Process industry examples of processes using heat.

Process industry	Examples of processes using heat
Petroleum refining	Feedstock heating, distillation, hydrocarbon cracking, hydrocarbon reforming (i.e., steam-methane reforming)
Chemicals (includes ammonia, fertilizers, pharmaceuticals, alcohols, detergents, resins, polymers, textiles, paints/pigments/dyes)	Endothermic chemical reactions, hydrogenation, pyrolysis, distillation, purification, evaporation, crystallization, polymerization, drying, prilling
Metals Manufacture (includes iron, steel, copper, and other non-ferrous materials)	Smelting of ores, refining, alloying, melting, casting, forging, rolling, annealing, soaking and heat treatment, sintering
Nonmetallic Minerals and Acids (includes phosphorous, soda ash or caustic, sulfurous, chlorine, alkalis/alkaline)	Slurry concentration and drying
Refractories (includes bricks, pottery, glass, cement, and other refractory materials)	Firing, kilning, drying, calcining, melting, forming (e.g., plate glass), annealing
Wood Product (includes lumber, paper, and paperboard)	Kiln drying, wood pulping and liquefaction, rolling, drying, coating, and forming
Food and Drink (includes dairy products, brewing, farinaceous products, meat and vegetables, soft drinks, and tobacco)	Milling, blending, forming as liquids, powders, or composite solids, purification, sterilization, pasteurization, fermentation, refrigeration
Sugars and artificial sweeteners	Glucose-to-fructose conversion, artificial sweeteners
Industrial gasses	Temperature-swing adsorption

Nuclear reactors for process heating

Nuclear heat is well-matched to process heating, and in some cases vaporizing chemical feedstock entering a reactor process. Nuclear heat can be directly substituted for steam that is used indirectly to dry, concentrate, or distill most aqueous solutions and to fractionate petroleum distillates. In some cases, nuclear heat can be used to break low-order chemical bonds such as biomass ligands or coal moieties. In summary, approximately 75% of all industrial heat duties require less than 700 °C, with about 50% of the duties being less than 300 °C.

Combining original graphs first presented by McMillan et al. (2016) and Massachusetts Institute of Technology (MIT, 2018), Fig. 2 visualizes the temperature ranges of process heat required in selected industries, while the Introduction tabulates and compares to the output temperature limits of different nuclear reactor designs. As the figure indicates, advanced nuclear reactors

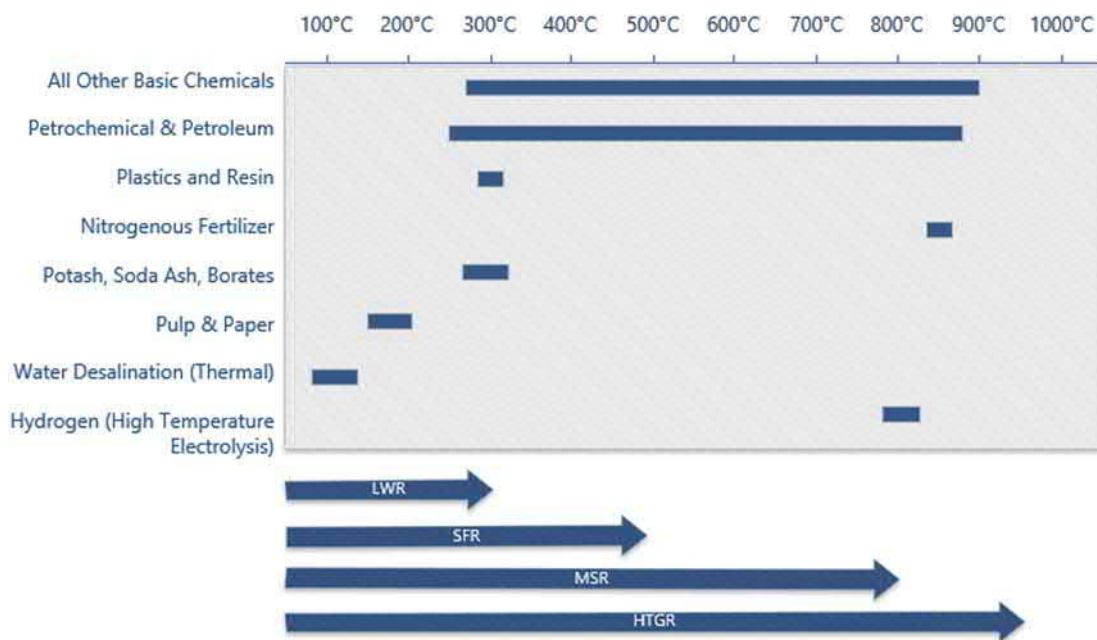


Fig. 2 Comparison of industrial process heat ranges to output temperature limits of nuclear reactors. Sources: McMillan C, Boardman R, McKellar M, Piyush S, Ruth M and Bragg-Sitton S (2016) Generation and use of thermal energy in the U.S. industrial sector and opportunities to reduce its carbon emissions. *Technical Report DE-AC36-08G028308*. Joint Institute for Strategic Energy Analysis, and Massachusetts Institute of Technology (2018) *The Future of Nuclear Energy in a Carbon-Constrained World: An Interdisciplinary MIT Study*. MIT Energy Initiative. <https://books.google.com/books?id=UJj-xAEACAAJ>.

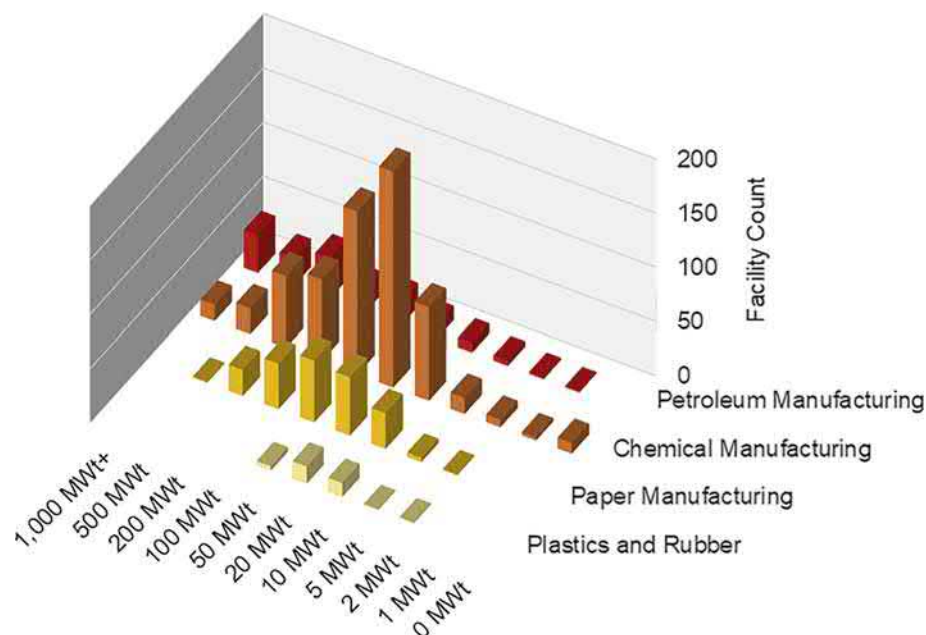


Fig. 3 Distribution of plant heat consumption for different industries. Source: Analysis of US EPA (2021) FLIGHT: Facility Level Information on GreenHouse gases Tool. U.S. Environmental Protection Agency. <https://ghgdata.epa.gov/ghgp/main.do>.

(ARs) produce higher-quality heat than light-water reactors (LWRs), and therefore have a greater potential to replace existing industrial heat sources. Combined with an efficient, high-temperature heat transport system, such as solar industrial process heat systems used with solar concentrators, sodium fast reactors (SFRs), molten-salt reactors (MSRs), and high temperature gas-cooled reactors (HTGRs) are expected to be capable of supplying process heat for most processes in these industries without the need for topping heat. Where necessary, low-carbon heat augmentation can be supplied electrically or with chemical heat pumps (Bragg-Sitton et al., 2020). Also, use of advanced nuclear reactors in industries or plants with processes requiring lower-quality heat provide the opportunity for bottoming processes, such as electricity production. In light of these findings, and based on temperature compatibility alone, none of the industries in Fig. 2 should be excluded from consideration for thermally coupled nuclear energy systems.

Naturally, a reactor's thermal output capacity must exceed the thermal energy consumption requirements of the industrial plant or plants that it serves. Fig. 3 shows the distribution of plant thermal energy consumption in four of the industries of interest. Plant size aligns well with the size of advanced reactors, which are expected to range from 1 to 20 MW_e for microreactors, 20 to 300 MW_e for small modular nuclear reactors (SMNRs), and from 300 to 1000 MW_e and higher for full-size reactors. A breakdown of advanced reactors under development is given in a fact sheet by the United States Department of Energy (2020).

Chemical process considerations (subheading)

The main challenge for nuclear reactors is replacing fired-heaters that provide on-demand heating of chemical processes. The chemical process reactors, in which chemicals are made in industry, vary in size from a few liters to several cubic meters and hold, at any one time, well over 100 t of materials. The design of the chemical reactor is determined by many factors, but of particular importance are the thermodynamics and kinetics of the chemical reactions being carried out.

In processes such as cement kilns, float glass making, and metals annealing, hydrocarbon-flames transfer heat by radiation to heat exchanger surfaces with the reactor vessel or process. Cement kilns and metals decarbonization require very high temperatures that cannot be indirectly transferred to the solid. Hydrogen can be burned to provide very high temperature gasses; however, hydrogen flames are virtually invisible and produce very little radiant heat to support indirect heating of a material.

The basic concepts for chemical process heating, evaporation, and reaction heating are illustrated in Fig. 4. When evaluating the use of nuclear heat sources for the chemical industry, it is important to understand the chemical processes and the equipment used for feed heat-up, phase-changes, and reaction enthalpy requirements. Heat recuperation must be optimized in a manner that optimizes the use of the nuclear heat source. Finally, heat augmentation may be applied, but only when the net effect uses the energy contributed by the nuclear heat source (for example, see Fig. 4F).

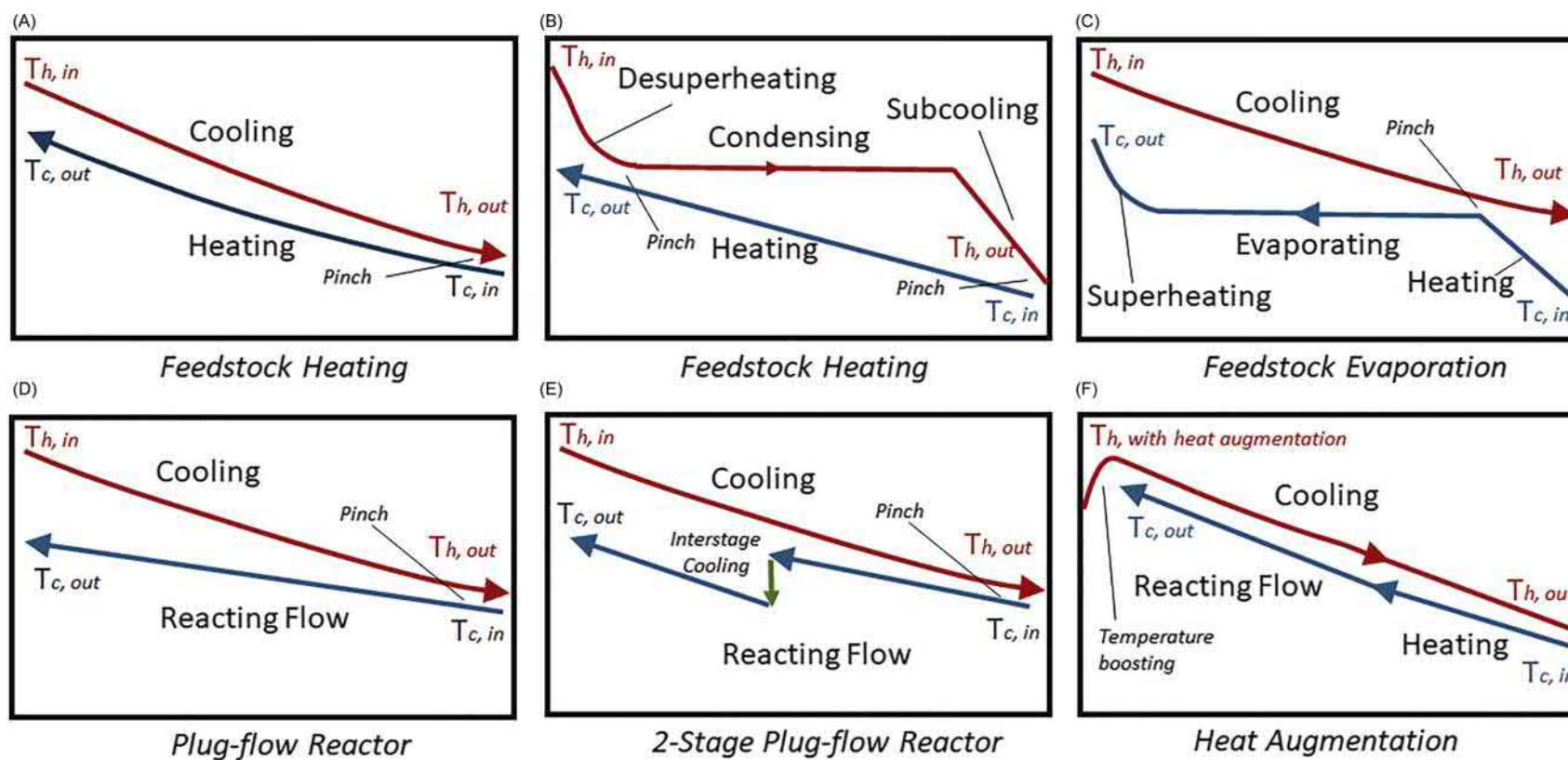


Fig. 4 Nuclear heat transport to chemical process heaters and chemical reactors.

Fluid mechanics considerations (subheading)

A number of integration issues must be considered when applying process heat applications to nuclear reactors. These considerations include:

- Reactor outlet temperature
- Reactor inlet temperature
- Fluid composition
- Pressure within coolant, heat transfer loop, and process heat application
- Tritium migration

Low temperature reactors—typically LWRs—can provide process heat to most industrial applications. Steam heating can be accomplished with topping heat from fossil fuels and/or electric resistance heating for higher temperature process heat applications. Conversely, high temperature heat produced by a high-temperature gas-cooled reactor (HTGR) or molten-salt reactor (MSR) may exceed the temperature limits of a given reaction process. In this case, a top-cycle use can be advantage, such a power generation unit. This is essentially the concept of a combined heat and power systems that converts supercritical pressure steam into power while dropping the pressure and temperature of the steam to the desired industry service quality. Similarly, a bottoming cycle may be necessary to lower the temperature of the heat transport media before its return to the nuclear heating loop.

In the case of the HTGR, the temperature difference of the gas temperature across the core optimally needs to be 350–450 °C. For most MSR concepts, the temperature difference is generally 50–150 °C, given the higher heat capacity of the salt versus a gas. In summary, the temperature of the thermal hydraulic heat delivery system must be carefully engineered. The choice of nuclear reactor depends on the industry heat application.

The melting temperature of liquid coolants, such as molten salts and sodium, and the heat transport system must be designed and operated to avoid freezing the salt. Depending on surrounding temperature, it may be necessary to insulate the pipes, pumps, and control valves of the delivery systems or to provide some form of electrical or jacket heating to re-liquify the frozen salt. Steam, carbon dioxide, and helium do not have such issues, although steam condensate traps and blow-out lines are needed at latitudes and altitudes where temperature may drop below the freezing point. For this reason, synthetic oils have become a preferred heat delivery media.

Heat transfer distance and the working fluid can both be limiting factors when considering industrial process heat input requirements. In the case of an application similar to the following artist's rendition (Fig. 5) created by the Next Generation Plant Alliance

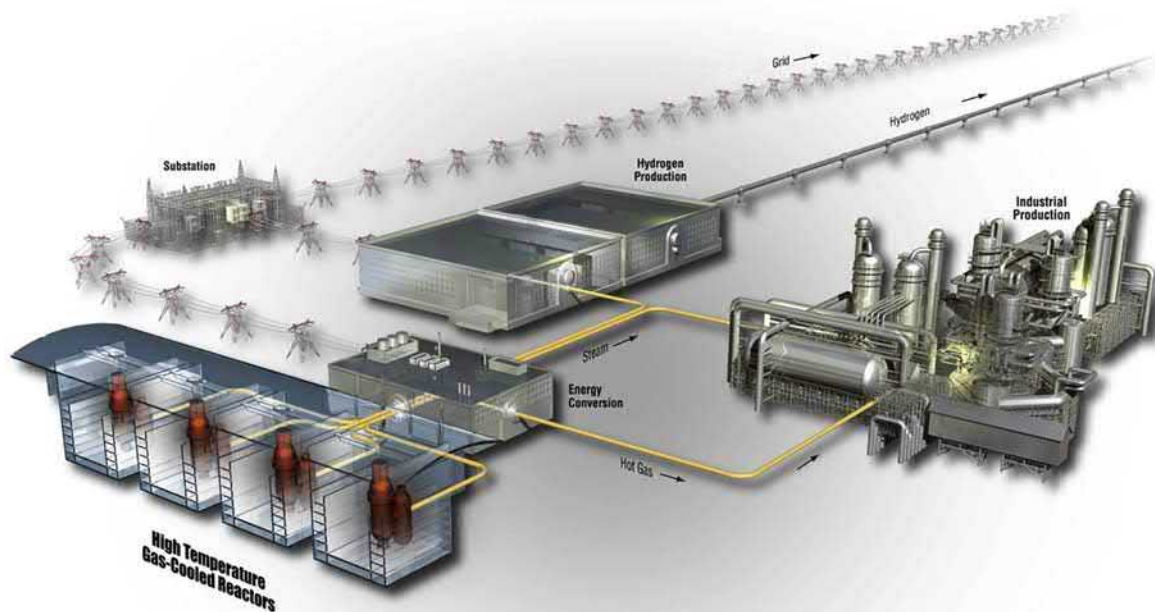


Fig. 5 Artist rendering of the High Temperature Gas-Cooled Reactor to industrial plants. Source: Next Generation Nuclear Plant Alliance. (Nelson L, Gandrik A, McKellar M, Patterson M, Robertson E, Wood R and Maio V (2011) Integration of high temperature gas-cooled reactors into industrial process applications. *Idaho National Laboratory External Report, INL/EXT-09-16942 rev.3, September 2011.*)

(Nelson et al., 2011), coupling between the nuclear reactor and one or more industrial processes may involve a heat transport distance less than 1 km. As this distance expands, the choice of heat transport media and thermal energy circulation loop become more important. The tradeoff between liquid pumping and gas compression costs may alone determine the choice of heat transfer media.

Long heat transport distances (subheading)

Engineering considerations for heat-transfer distance, such as capacity matching and working fluid temperature, are straight forward, but not necessarily easy. Steam/water and molten salts perform better than gasses for relatively long heat-transfer distances (up to 20 km), primarily because gasses such as helium require extremely high pumping power (McKellar, 2011). The high pumping power makes long-distance hot-gas heat transport applications impractical (Yoon and Sabharwall, 2014; McKellar, 2011).

Fig. 6A and B compares the transport of molten KFZrF_4 to helium using a circulation loop of a comparable size. For this comparison, the pipe is considered to be buried, although this may not be practical when considering thermal-expansion design requirements. The molten salt, having the highest volumetric heat capacity and lowest pumping energy, is capable of transporting 54.4 TJ/day or 629 MW_t (at a coefficient of performance of 4670 TJ/day- MW_e) versus 16.3 TJ/day or 188 MW_t (at a coefficient of performance of 4.85 TJ/day- MW_e) using helium in the same sized piping system. The coefficient of performance is the heat divided by the pumping or compression power used to transport the heat. In a practical engineering design, pipe material and corrosion, as well as pump versus compressor costs, are important considerations.

Process heat applications generally require a secondary heat transfer loop to isolate the primary core cooling loop from the process heat application. One purpose of this secondary loop is to reduce the migration of tritium from the reactor core to the process product. Additional heat transfer loops may be added to reduce tritium migration, but each loop degrades the exergy of the process heat temperature.

Temperature boosting (subheading)

Temperature boosting through heat pumping is one approach to use energy conversion processes to create high temperature heat from the relatively low temperature heat produced by LWRs. There are a variety of possibilities for these conversions; some of which are illustrated in Fig. 7A–E. Thermodynamic principles require a net input of work to raise the temperature of heat in this manner, with more work needed as the final temperature increases. If an LWR provides the work input, the net effect is to convert a given amount of its 300 °C heat into a somewhat smaller amount of high temperature heat. However, this high temperature heat comes from a CO_2 -free non-combustion source, providing environmental benefit relative to most current industrial practices. Table 2 compares other factors specific to each of these cycles. A variety of chemical looping possibilities have been developed. An exergoeconomic analysis is required to justify any application of these concepts.

Example: Use of nuclear heat for methanol production

Methanol is one of the predominant commodity chemicals in the United States and around the world. It is a feedstock to produce chemicals, such as acetic acid and formaldehyde, which in turn are used in products such as adhesives, foams, plywood subfloors, solvents, and windshield washer fluid. In recent years, the use of methanol in the production of olefins, or methanol-to-olefins, has grown rapidly (Alvarado, 2016). Methanol can also be used on its own as a vehicle fuel or blended directly into gasoline to produce a high-octane, efficient fuel with lower emissions than conventional gasoline.

There are at least two opportunities to replace natural gas combustion in methanol production: one is the heat needed for the primary reformer, and the second is the heat required for methanol purification. The former case is illustrated Fig. 8. In conventional steam reforming, methane is reformed in tubes within a furnace. Heat is provided by firing natural gas. Heat transfer from the flame to the tubes is primarily via thermal radiation.

When heat is available above 700 °C (i.e., high temperature molten-salt or gas-cooled reactors), it is possible to integrate nuclear heat directly into the natural gas reforming process, thus displacing combustion of natural gas as the primary source of reforming heat. A brief discussion of heat integration for the various scenarios is presented below.

Fig. 9 shows a simple block flow diagram indicating heat delivery to the primary reformer in a natural gas reforming plant (Wood and McKellar, 2013). In this example, the primary coolant for a fluoride/lithium/beryllium (FLiBe) high temperature reactor (FHR) is used to heat helium, which is then pumped into the primary natural gas reforming heating box. The hot helium replaces the hot gas produced by natural gas combustion, which in some reforming plants consumes as much as 30% of the natural gas input to the plant. The helium loop also provides separation of the primary reactor coolant from the chemical plant.

Bragg-Sitton et al. (2013) evaluated the technical and economic feasibility of producing methanol using a hybrid system in which the FLiBe salt is circulated in two parallel loops (Fig. 10): (1) a primary loop that exchanges heat with a helium heat transfer loop to deliver heat to a methanol plant in the manner described above, and (2) a secondary high temperature heat transfer loop

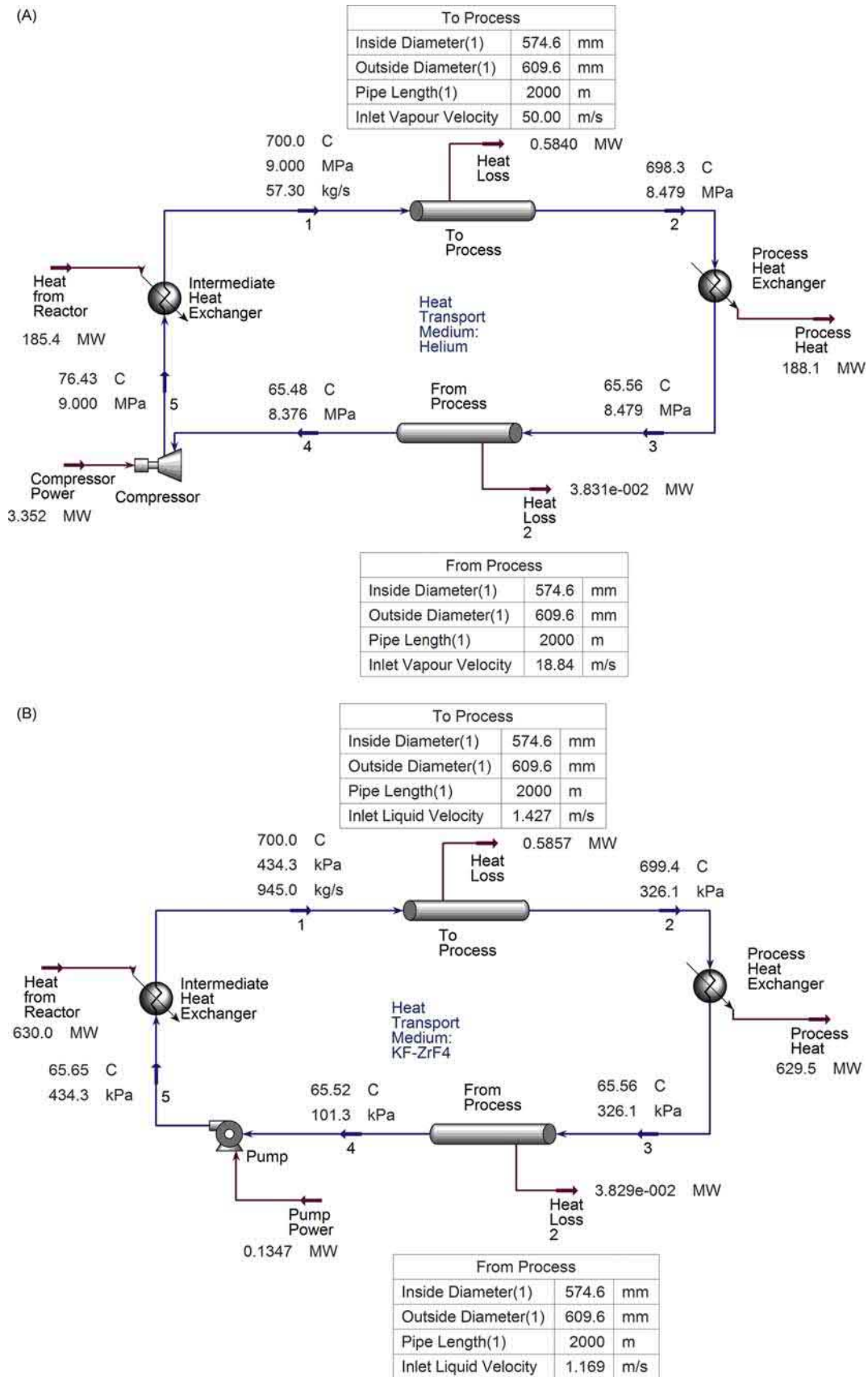


Fig. 6 Operating conditions and performance of high temperature reactor heat transfer loops (McMillan et al., 2016). (A) 2-km helium circulation loop, where the dimensions for pipe section 1 are tabulated, and (B) 2-km KFZrF₄ molten-salt circulation loop, where the dimensions for pipe section 1 are tabulated.

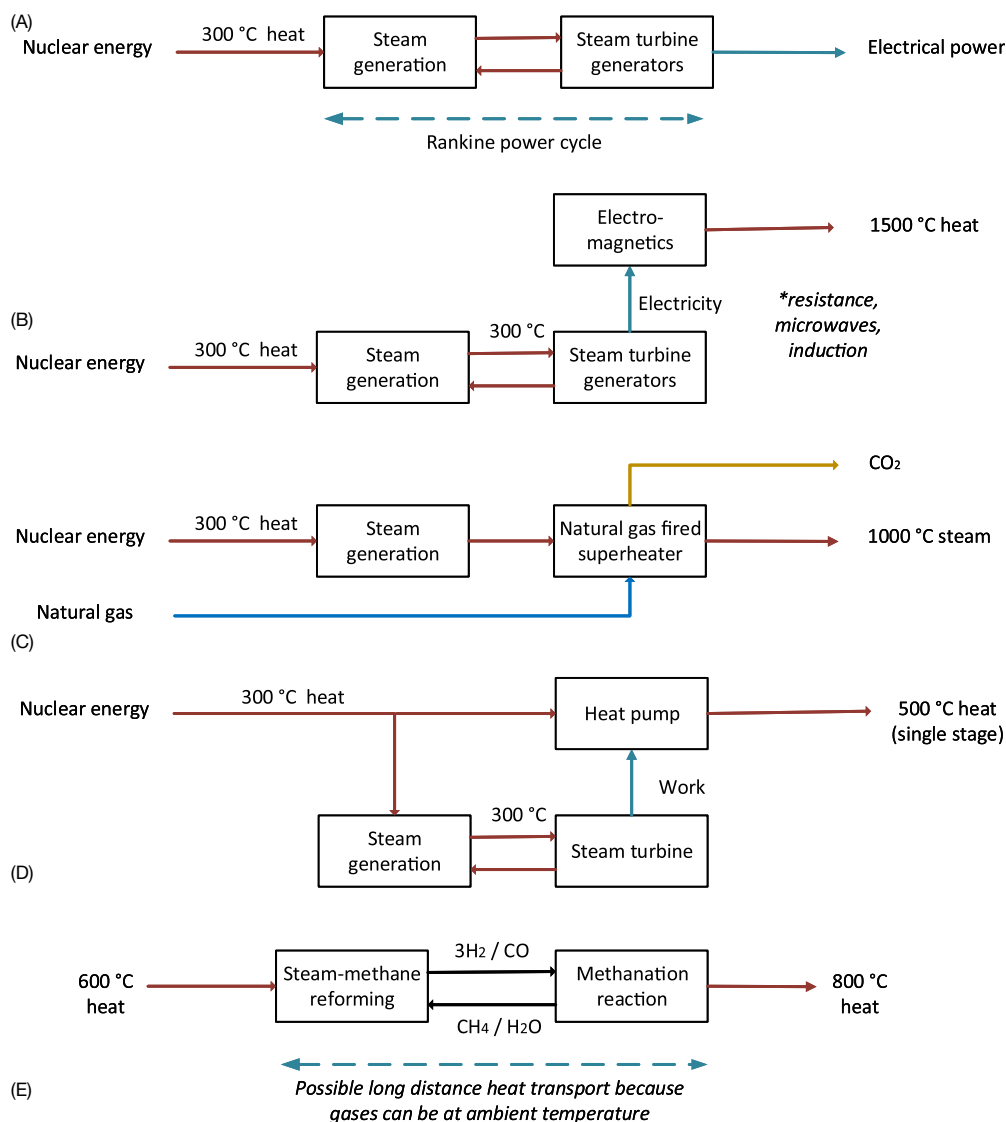


Fig. 7 Temperature boosting heat pump concepts. (A) Conventional power production, Use of electric power to generate heat, (B) fossil-fired topping cycle with nuclear heat, (C) split-duty heat pump, (D) looping chemical reaction.

Table 2 Comparison of energy conversion cycles for making high temperature heat.

Process	Delivery temp. range	Heat delivery efficiency	Advantages	Disadvantages	Comment
Rankine cycle	Makes power	32–34% as power	Well established technology	Makes power, not heat Fairly low efficiency	Reference process
Electric-based heating	To melting point of materials, 1500 °C	33% to power, 100% power to heat	Well established technologies Easy to integrate into applications	Fairly low efficiency No particular synergy with nuclear generation	Isothermal heat delivery
Fossil-fired topping	To melting point of materials, 1500 °C	~85% for furnace	Well established technology	Half or more of heat input comes from the furnace Associated CO ₂ emissions	Energy delivered as sensible heat over a temperature range
Split duty heat pump	ca. 200 °C increase per stage	~60% per stage, rest is reject heat	All energy comes from the nuclear reactor Adaptable for cogeneration Only a small amount of equipment sees high temperatures	High temperature compressors are not standard items (but turbines are)	Explore working fluids besides steam
Chemical reaction for heat delivery	ca. 800 °C, perhaps to 1000 °C	Estimated 60–70%	Can support long distance heat delivery by piping cold gas Storage of energy (as syngas mixture) is possible Could integrate with a HTGR	Requires intermediate temperature heat input for reforming (could potentially integrate with a heat pump)	Explore other endothermic chemistry, such as water splitting Heat can be delivered roughly isothermally

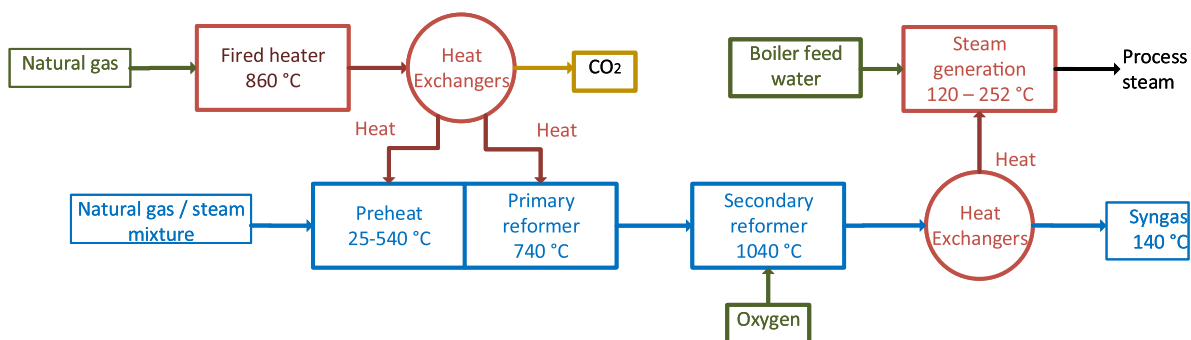


Fig. 8 Energy use in synthetic gasoline production for use in methanol synthesis.

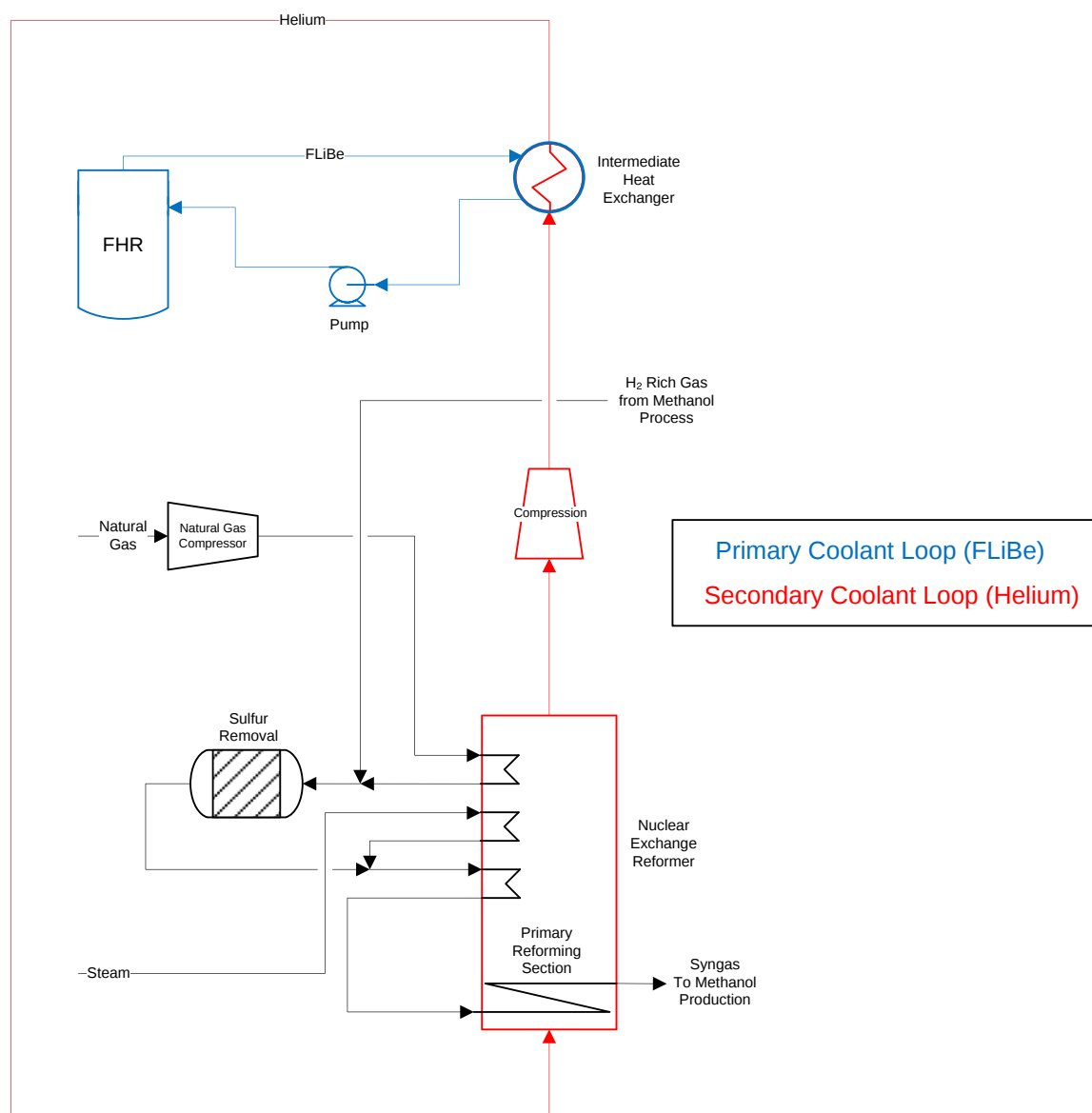


Fig. 9 Steam methane reforming with a fluoride-salt high temperature reactor (FHR).

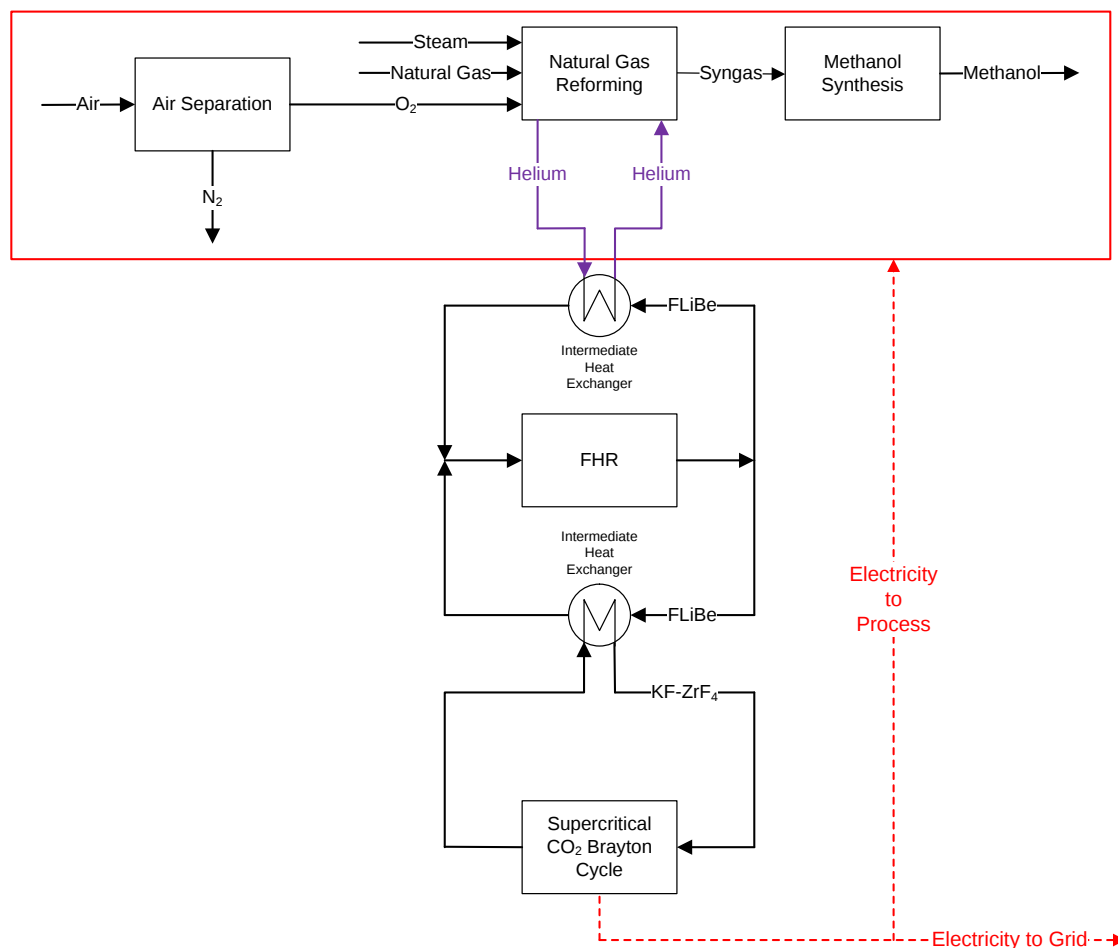


Fig. 10 Advanced hybrid FHR methanol configuration.

fluid that uses molten KF-ZrF_4 salt to transfer heat to a supercritical CO_2 Brayton Cycle power generation plant. The pressure of the helium loop was set at 9 MPa.

The methanol process generates some excess steam that is fed to a small Rankine cycle. The amount of power generated from this cycle, however, is insufficient to meet the entire electrical duty of the plant. The Brayton cycle therefore supplies power to the methanol plant to avoid the need for the methanol plant to import power from the grid. Excess power can be sent to the grid—an action that can be desired to support grid demand.

A discussion of the economics for the hybrid methanol and electrical power case is presented by Bragg-Sitton et al. (2013). For this case study, a dynamic model was developed to evaluate the profitability (measured in terms of the net present value of the project) as a function of variable energy apportionment between the power cycle and the chemical plant. The study developed a physics-based transient process model to simulate dynamic ramping of the natural gas reforming and power generation blocks in accordance with grid demands tied to wind power variation and grid supply demands. This type of hybrid system results in underutilization of the capital assets while optimizing profit by responding to the highest value market. For this case study, a positive profitability of this example hybrid system was achieved when wind penetration exceeds 20% of total system capacity. This was attributed to the high value of electricity when reserve capacity is required.

Additionally, at approximately this level of variable power generation capacity, traditionally baseload plants begin to be affected and are required to flex their operations. The benefits of an integrated plant, and similar hybrid systems, depends on the local grid markets, the chemical commodities markets, and the stochastics of variable power generation sources.

Conclusions

This section highlights the opportunity to use nuclear heat for a wide spectrum of industrial manufacturing processes. This could arguably reduce industry GHG emissions by 90% when eliminating combustion in steam boilers, combined heat and power systems, fired-heater, and high temperature processes. The application of nuclear heat may be as simple as supplementing or

replacing the steam duties of a plant to direct heat transfer to a chemical reaction process vessel. The cost of nuclear depends on many factors, but the impetus to reduce climate altering gasses like CO₂ provides motivation to the nuclear heat sources. There remains a significant amount of research to understand the optimum choice of nuclear reactor to use and the separation between tightly coupled systems. However, the studies that are being completed in this area, including the limited examples shown in the Chapter, are promising. Dynamic operation of integrated energy systems is especially of high interest because these systems can help firm the grid when the penetration of variable generation sources like wind and solar energy begins to affect traditional baseload plant operations.

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Industrial Processes for Grid Stabilization

Frédéric Bienvenu, Idaho National Laboratory, Idaho Falls, ID, United States

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Glossary

Conventional power production plants Electricity producing plants that include hydroelectric dams and all thermal plants, e.g. coal, fuel, gas and nuclear power plants.

EAF Electric Arc Furnace, a furnace in which metal is melted thanks to an electric arc.

EAS Electric Arc Steelmaking, the process of recycling steel from scrap steel in an EAF.

Electrolytic process Industrial process that uses, at least at one step, an electrolysis.

Flexibility The capacity, for a power plant or an electric load, to change its power production or consumption.

FT process Fischer-Tropsch process, a process that produces liquid hydrocarbons from syngas, which is a mixture of carbon monoxide and dihydrogen.

(Electricity) Grid stabilization The balancing of power flows in an electricity grid so that no line is congested and global electricity production equals global electricity consumption.

Load shedding The reduction of the electricity consumption of a load to help grid stabilization.

(Variable) Renewable Generation Power generation coming from variable and renewable sources such as wind and solar power productions.

RO Reverse Osmosis, most common process to produce pure water from seawater or brackish water.

Introduction

Industrial processes tend to be used more and more by electricity grid regulators for the wide range of services they can provide. In fact, grid stabilization is a synonym for regulating power flows in a possibly multi-state electricity grid, and industries, because of their significant electricity consumptions, could play a major role in power flows. In particular, industries can participate in balancing consumption and production of electricity at all times, or in relieving grid congestions when the maximum capacity of an electricity line is reached, simply via smart management of their electricity consumption.

Those services are especially required since some hazards affect grid stabilization on a daily basis, like unexpected technical constraints on power production units, or more commonly forecasting errors, whether they concern the consumption of electricity, or the production coming from renewable sources (e.g., wind and solar power production mostly). Those uncertainties have been historically compensated by automatic and manual modifications of the setpoint of conventional power production units, including nuclear power production units. However, the growing share of renewable generation in the power production mix not only increases the amplitude of the hazards due to forecasting errors, but it also decreases the amount of flexibility offered by conventional power production units. Besides, new uses of electricity, such as electric vehicles or load shedding, affect the electricity consumption forecasts: it is still unclear if these new uses will make consumption more or less predictable.

As a consequence, electricity consumption will have to provide the flexibility that is being replaced by renewable generation. In the USA, total industry energy consumption represents 10,700 TWh in 2018, which is 36% of the world industrial energy consumption of 30,000 TWh (IEA, 2019a). Electricity consumption represents 29% of this number in the USA, which is high, even though this share has been slowly decreasing since 2008, as can be seen in Fig. 1.

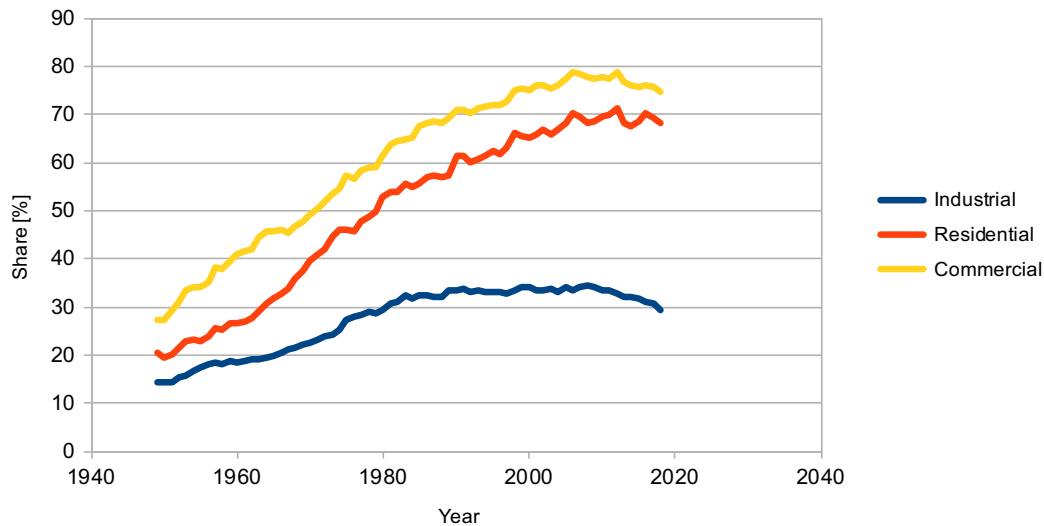


Fig. 1 Electricity share in energy consumption in the USA per sector. Data from IEA (2019a) *Data & Statistics—IEA*. [online] Available at: <https://www.iea.org/data-and-statistics> [Accessed 11 Dec. 2019].

Fig. 1 also highlights the fact that the electricity share in residential and commercial consumption is higher than in industrial consumption, for a similar amount of total energy consumption. However, residential and commercial consumptions are distributed on many sites (each individual home or shop being a site) and therefore constitutes a far less manageable load, with a less predictable impact on power flows. Managing electricity consumption is therefore much more economically and technically efficient if it is installed in an energy-intensive factory rather than in a common house. Additionally, it is regarded as more socially acceptable to control the electrical load of an industrial consumer than to control the electricity consumption of common citizens. In the first case, it affects the planning and the design of an industrial process whereas in the second case it may directly affect the comfort of citizens in their homes. Finally, the way industries are managed makes them more sensitive to economic incentives, and if electricity markets and regulations make investing in control devices profitable, industries are likely to adapt faster to those changes and therefore provide the corresponding services.

On the other hand, industrial processes are also subject to more technical constraints than residential and commercial consumers. Unplanned load shedding could result in loss of a full batch of goods for some industries, which can cost up to several million dollars for some products. As a consequence, and it is the object of this article, each industrial process needs to be analyzed to determine if its electricity consumption can be adapted according to the needs of the electrical grid, if it needs to be redesigned, or, finally, if it is simply not suitable for providing such services.

The answer may depend on several factors, in particular how fast the process' electricity consumption can be modified, and how long this modification can last. If no industrial process can provide services as fast as electrical inertia for grid stabilization, as it would require spinning machines connected directly to the grid and operating always at a multiple of 60 Hz, some may be able to, to some extent, adapt their consumption in a few seconds or a few minutes. For time scales shorter than 15 min, it is often considered that the regulation must be automatic and must be performed through a control device directly connected to the grid regulator's system. For time scales longer than 15 min, manual actions can be taken in coordination with the grid regulator, as long as it is technically feasible to manually change the consumption setpoint. For some processes, this last feature is not even necessary: smartly planning when to start or stop the process is already a form of load management. Other dimensions to the answer can also be important, such as the amplitude of the modification of the consumption, its duration, previous activations, etc.

The answer can also depend on economic incentives. The services that can be provided by a given industrial process are sometimes limited by the cost of the required investment. Besides, in some cases where the load has to be shed during long periods of time, or involving a significant amount of energy, the yield of a process, or the daily output of a factory, can be significantly decreased due to electricity grid needs, which means that industrial load management may also have variable costs.

Desalination

Industrial load management can simply be achieved if industries offer some kind of electricity storage. Electricity does not have to be stored directly as in battery storage, but for example in the form of the output of an industrial process. So far, the best kind of grid level electricity storage is performed with pumped-storage hydroelectricity (PSH) power plants, by transferring water between two reservoirs at different altitudes, upstream for downward needs, downstream for upward needs. An analogy can be made with desalination processes: seawater coming from the lower reservoir, and pure water from the upper. The desalination process consumes

electricity and/or thermal energy to turn seawater into pure water, which is equivalent to using electricity to pump water from a lower to an upper reservoir. The opposite effect, if theoretically possible, is less relevant to analyze since pure water is considered to be more needed than grid stabilization services. However, reducing the flow of pure water production compared to what was originally planned can be considered as an additional flow of pure water being turned into seawater while producing electricity. Desalination plants can, therefore, provide a form of electricity storage.

Besides those theoretical arguments, the potential of desalination industries for grid stabilization is huge. According to the International Desalination Association (Idadesal, 2019), there are more than 21,000 desalination plants worldwide, located in more than 150 countries. This corresponds to a total capacity of desalination of 142 Mm³/day, which more than 300 million people rely on for their daily needs. This number is very likely to increase (Jones et al., 2019) as 40% of the global population faces severe water scarcity conditions: this share could even increase to 60% by 2025. This trend is confirmed by the evolution of the global capacity of desalination plants which increased by more than 30% between 2010 and 2020 (Jones et al., 2019).

In terms of energy, the global energy consumption of desalination plants per year is around 350 TWh (Desware.net, 2019; Dach, 2008; Singh, 2015), including 200 TWh of electricity consumption, which is about the total electricity consumption of Spain, and 1% of the world electricity consumption.

However, desalination is not one industrial process, it is achieved through different methods, whose distribution can be seen in Fig. 2 (Jones et al., 2019).

The most common industrial processes to perform water desalination are Reverse Osmosis (RO - 69%), Multi-Stage Flash Distillation (MSF - 18%), Multi-Effect Distillation (MED - 7%), Nanofiltration (NF - 3%) and Electrodialysis (ED - 2%). If, in terms of global installed capacity, all processes are getting more and more widely used, only the share of RO has been increasing since 2000, where it was a bit less than 50% (Jones et al., 2019). This success of RO for water desalination can be explained by the small amount of energy it requires, only 3–5.5 kWh/m³, which is much lower than MSF 13.5–25.5 kWh/m³ (Desware.net, 2019), as well as for the numerous varieties of water it can purify. For example, although NF consumes less than half the electricity of RO, it does not remove all impurities from water. As a consequence, RO global yearly electricity consumption represents 65% of global desalination electricity consumption, which is 130 TWh. This paragraph will therefore focus on RO.

Before analyzing how it could be used for grid stabilization, a good understanding of the process is necessary. RO properly speaking needs a very pure brine input. Consequently, several stages of pre-filtrations are required. This is due to the very sensitive membrane that the process uses: many impurities have the potential to damage it. The problem of filtration is that the smaller the removed particulates are, the more sensitive the membrane used to filtrate them is. The process must therefore be progressive. Different types of filtration are available depending on the composition of the water. Once appropriately filtered, seawater is ready for RO. The process uses two water tanks separated by a special membrane; one contains seawater and the other relatively pure water. The membrane is the core of the process, the progress made in its design explains why this process has become so widely spread. Its function is to allow almost only water go through it. Usually, because of the difference of chemical potentials between the two liquids due to the differences in ionic concentrations, pure water tends to go to the seawater tank to dilute it and therefore equilibrate the potentials. This phenomenon results in a pressure against the membrane called the osmotic pressure. Between seawater and pure water, it is around 27 bar. An RO plant increases the pressure on the seawater side until the flow reverses, which occurs at a pressure above 27 bar (Lachish, 2002). In fact, pressure is usually around 40–82 bar in the seawater tank for matters of kinetics and percentage of recovery. Usually, up to 50% of the water is extracted from seawater with this method. Afterwards, a post-treatment is necessary because in some cases, water obtained by RO lacks essential minerals. Fig. 3 sums up the whole process.

The potential of RO for grid stabilization is huge, not only because it represents a growing and significant share of the electricity consumed worldwide, but also because its process can easily be adapted for load management.

On large time scales, storing water is indeed extremely easy, so, if the desalination system does not need to be used at 100% capacity at all times, planning when to desalinate water could significantly help the electrical grid. This should neither constitute

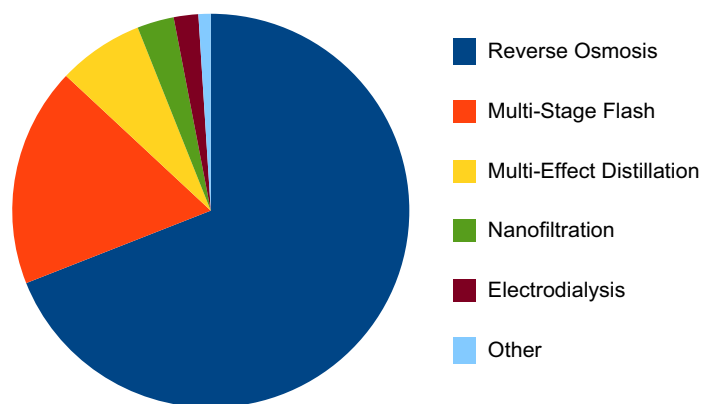


Fig. 2 Share of main desalination processes in the world. Data from Jones E, Qadir M, van Vliet M, Smakhtin V, and Kang S (2019) The state of desalination and brine production: A global outlook. *Science of the Total Environment*, 657: 1343–1356.

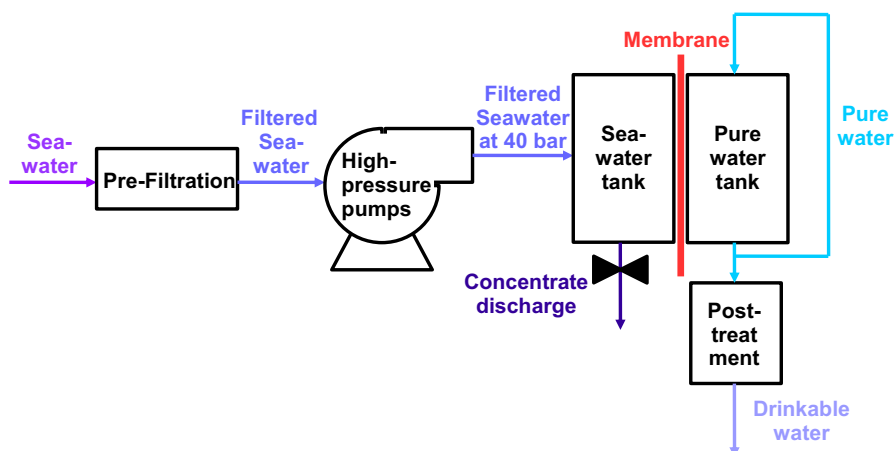


Fig. 3 Diagram of the reverse osmosis process to turn seawater into drinkable water.

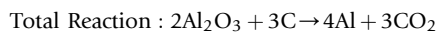
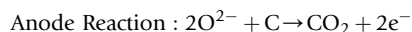
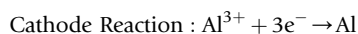
an economic loss insofar as the consumption of water would not change in total, nor would the cost of seawater. The only drawback is that this service may require desalination plants to have excess capacity, which is globally true at present (Idadesal, 2019). Another lower cost option could be to only schedule maintenance according to electricity grid needs.

On shorter time scales, and therefore for lower amount of electricity and water involved, other services can be provided by desalination plants. In fact, the major source of electricity consumption in the RO process comes from the high-pressure pumps for seawater. The pressure they impose is usually between 40 and 82 bars for kinetics and percentage of water recovery, but as long as the pressure remains above 27 bar, pure water would not counter-flow from the pure water tank to the seawater tank. This means that, if needed, the pressure produced by the pumps, and therefore their electricity consumption, can be lowered, resulting only in a lower amount of purified water per hour, partly compensated by a slower tiring of the very fragile membranes. Symmetrically, increasing the pressure above the nominal level of the desalination plant, and therefore consuming more electricity, may be more difficult and could damage the membrane unless an appropriate design is chosen for that purpose. Moreover, there does not seem to be any limit to the duration of the modification of the pressure setpoint.

Alumina reduction

World alumina production is 130,433 Mt. in 2018, of which about 53% is produced in China (World-aluminium.org, 2019). Alumina mostly comes from bauxite ore and is obtained through the Bayer refining process. About half of this alumina is then smelted into different grades of pure aluminum through the Hall-Heroult process, which corresponds to 64,336 Mt. of global aluminum production, 55% being produced in China. Aluminum production from alumina is a very energy intensive process since it consumes about 14–15 MWh of electricity per ton of aluminum. The global electricity consumed by aluminum smelting can therefore be estimated around 867 TWh, which is almost 4% of the world electricity consumption, or a quarter of the electricity consumed in the USA.

The Hall-Heroult process is an electrolytic reaction which produces carbon dioxide and aluminum from alumina and carbon through the following reactions:



These reactions take place in a heat-insulated rectangular steel shell with a carbon lining inside, as shown in Fig. 4. The cathode is a conductor going through the shell inside the carbon lining which protects it from the heat but conducts electricity. The anode is covered in carbon and immersed in an electrolyte bath, contained inside the shell. The anode is composed of Na_3AlF_6 , and additives such as AlF_3 and CaF_2 (Li_2CO_3 and MgF_2 are also used), which are used to control the operating temperature, solubilities, activities of ionic species, conductivity, viscosity, interfacial voltage bath density, vapor pressure, hardness of the crust, etc., as well as to increase current efficiency (Starke et al., 2009). AlF_3 is the most common additive. It lowers the operating temperature, the solubility of reduced aluminum, surface voltage, viscosity and density. However, it also decreases alumina solubility and electrical conductivity.

Alumina is fed and dissolved inside the tank through point feeders (its concentration must be kept around 2–8 wt%) to enable the best distribution of alumina in the bath. Alumina is also used to cover the carbon anodes and the frozen bath surface to reinforce

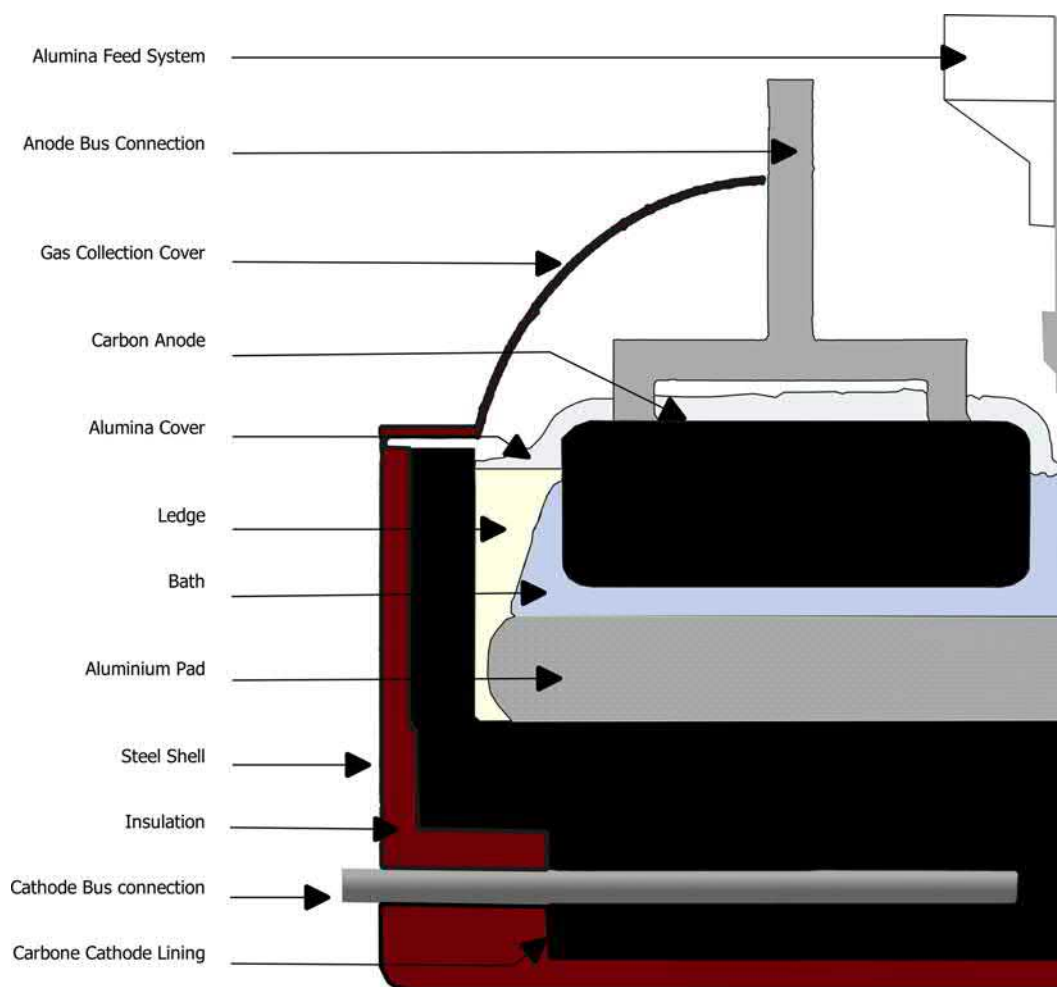


Fig. 4 Diagram of an electrolytic cell used for aluminum production and its shell. Reproduced from Bangs C (2011) *Rolling Out Aluminium to Meet IPCC 2050 - A Dynamic MFA Approach for Greenhouse Gas Emission Projections for the Aluminium Industry*. doi: 10.13140/RG.2.2.19000.75527.

thermal insulation and reduce air burning of the anode. Despite this very accurate feeding, the bath is polarized (gradient of concentration), which increases its resistance.

During the reaction, the carbon anode is slowly consumed (0.45 kg/kg of aluminum produced), creating bubbles (mostly CO_2 , but also CO or even C_2F_6 and CF_4 due to anode effects [local shortage of alumina which can create overvoltage up to 40 V]). Changing the anode when completely consumed is the first reason to stop the plant. The distance between the anode and the aluminum pad being created at the cathode has to be changed permanently given the wearing of the anode and the size of the aluminum pad. This distance is usually 4–5 cm; it is a compromise between reducing the resistance of the bath, keeping the aluminum pad and the anode apart, enabling bubbles to escape and allowing alumina ionic species to easily reach the charged surfaces.

The main restriction to load management in aluminum plants is that the temperature inside the tank must be kept within close values, the heat losses being compensated by Joule losses inside the shell. However, using thermal inertia and the huge quantity of heat stored in the electrolyte bath enables small variations of the electricity consumption of the plant. In fact, the temperature inside the tank must be kept between 950 and 980 °C (Choate and Green, 2003), whereas Na_3AlF_6 usual fusion temperature is 1011 °C, due to the additives in the bath that lower its fusion temperature. As a consequence, there is a wide range of operational temperatures that can be explored, using more or less additives.

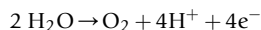
Operating at high temperature presents the advantage of opening a wider range of power consumption as well as alumina concentrations, but it is very energy-intensive and may destroy the frozen protective layer of electrolyte (which protects the carbon lining from corrosion). Operating at low temperature may enable the ledge to grow too much and alumina may precipitate, which has various consequences that can lead to the destruction of the cell.

Thermal inertia allows only “small” variations of power consumption, since the temperature range is still quite limited. However, the current going through this electrolytic bath is around 300–600 kA, the voltage of a cell is 4–4.6 V, and there are 150–288 cells in a single line. The total power consumption of a line is therefore between 200 and 800 MW (almost equivalent

to a large-scale nuclear power plant). Small variations actually corresponds to several MWs, which correlates to grid-scale regulation, although those MWs may not be available for long consecutive periods of time.

Current density is the first lever to perform load management in aluminum plants. If it is usually between 0.8 and 1.0 A/cm², it can be increased to accelerate the quantity of aluminum produced per cell. On the other hand, decreasing it improves current efficiency, and the process consumes less energy per unit of aluminum produced. However, the cost of the saved energy has to be balanced with the fact that a decreased current density increases capital and labor costs.

As for load management through voltage regulation, it is more complicated insofar as it is determined by the reaction itself and the number of cells. Several cells could be short-circuited for small periods of time, but it is economically very inefficient. What could be possible would be to activate a parallel set of anodes, without cathodes, so that the anode reaction becomes:



This anode reaction requires a much higher voltage and increases the acidity of the electrolytic bath, but it is mentioned in this article to highlight that other anode reactions could be considered if economic incentives for grid stabilization were created. In that case, investments may be extremely expensive.

Finally, planning could also be an important source of aluminum plants load regulation. For instance, the replacement of used carbon anodes could be conducted during peak hours of grid electricity consumption, or it could be focused on Spring and Autumn, where electricity consumption is reduced.

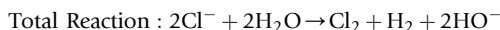
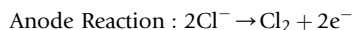
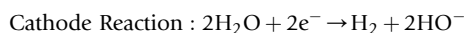
Considering their huge energy intensity, aluminum plants are already used in different parts of the world for grid stabilization, even if it is only done so far for small and very short term power consumption variations, like in Alcoa's Warrick operation plant (Choate and Green, 2003). This aluminum plant, which usually consumes at steady-state 500 MW, regularly supplies about 15 MW of fast and short-lasting load management for grid stabilization.

Chlorine production

The global production of chlorine reached 65 Mt. in 2018 (Lazonby, 2018), of which 11 Mt. was produced in the USA and 10 Mt. in Europe. Most of chlorine global production is obtained through the Chlor-Alkali process (Mansfiel et al., 2000), which can be performed with three different kind of cells. Historically, the mercury cell process has been the most predominant, but it has been slowly replaced by the diaphragm cell process and then more recently by the membrane cell process. In 2018 in Europe, the mercury cell process completely disappeared due to its large energy requirements, about 3.7 MWh/t of produced chlorine, and concerns of mercury pollution. This shift leaves the membrane process with a share of 85% and the diaphragm process with a share of 10%, with the remainder coming from other processes where chlorine is a by-product, such as sodium production through Down's process (Eurochlor, 2019; Corrosion-doctors.org, 2019). In the US, the situation is similar (Lenntech.fr, 2019). The energy consumed by the membrane cell process is about 2.5 MWh/t whereas the diaphragm cell process consumes around 2.9 MWh/t of chlorine (Chlis-tunoff, n.d.), which also explains why it is slowly being replaced by the membrane cell process.

This article therefore only focuses on the Chlor-Alkali process performed with membrane cells. The global production of chlorine, 65 Mt./year, represents 162.5 TWh of electrical energy (0.7% of the world electricity consumption); in Europe it represents 25 TWh, and in the USA, 27.5 TWh. Those figures, even if lower than for aluminum production, make chlorine production a good candidate for grid stabilization, because it may still be economically relevant for a given chlorine plant, and at the country scale, the order of magnitude of those numbers is sufficient for considering grid stabilization.

The Chlor-Alkali process with a membrane cell is, as aluminum smelting, an electrolytic process. The input is a constant flow of concentrated pure brine, which can quite easily be obtained from purified seawater so it only contains sodium and chloride ions, which are, in other words, dissolved common salt. This flow fills a tank in which the anode and the cathode are immersed, separated by a membrane, the input flow being on the anode side. The membrane, which only lets water and cations through, is the main difference with a diaphragm cell (which uses a diaphragm instead of a membrane), as the diaphragm only prevents counterflowing. With a voltage of around 3.1–3.3 V between the two electrodes, the following reactions occur:



Dichloride is a gas that is collected through an appropriately placed pipe above the anode side of the brine tank. On the other side of the tank, dihydrogen is collected with a similar system. There is also an output flow that compensates the input flow so the pressure in the tank does not increase. This flow is made of a concentrated solution of caustic soda, since sodium cations go through the membrane to electrically balance the tank, and hydroxide anions are created by the reaction on the cathode side. In the case of a membrane cell, it is already pure and therefore very valuable, whereas a diaphragm cell allows some chloride anions to reach the output flow. The Chlor-Alkali process consequently produces three valuable output products: dichloride gas, dihydrogen gas, and concentrated caustic soda.

The potential of chloride production for grid stabilization is yet to be discovered, since its lower energy intensity relative to aluminum production has driven it away from concrete experimentation. The first source of flexibility that chloride production could offer would simply consist in varying the cell current. Usually, the number of cells in a line, and therefore the total current in the plant, is optimized to obtain a good compromise between the amount of chloride produced per hour and the Joule losses. However, nothing seems to prevent the current per cell to increase or decrease by a few percent, provided that the input flow is also adapted. An increase in current, and therefore in Joule losses, would be partly compensated by faster and more important chloride production, and vice versa. This would result in a possibility for short-term load variations just as in aluminum production.

Moreover, chloride production could offer a better source of flexibility for grid stabilization by introducing a flow of pure oxygen on the cathode (Chlistunoff, n.d.). In fact, this would change the cathode reaction to $O_2 + 2H_2O + 4e^- \rightarrow 4H_2 + 2HO^-$, which has a potential of 1 V less than the original cathode reaction. As a consequence, this modification could result in a decrease of one third of the total energy consumed by the chloride production plant, without any duration limits. The drawback is that not only oxygen is consumed at the cathode, but dihydrogen is not produced, so the oxygen flow should only be triggered when providing grid stabilization flexibility is sufficiently valued so it becomes profitable to consume oxygen and not produce dihydrogen. In terms of grid stabilization, there is another flexibility source that could be employed: dihydrogen can easily be converted in electricity through a fuel cell, with good efficiency. Storing hydrogen can therefore be equivalent to storing electricity, as is described later in this article. Once again, electricity storage comes with very few technical constraints. Those two features underline that chloride production has a great potential for grid stabilization, despite the additional investments they may require.

Magnesium production

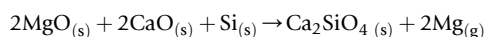
The global magnesium production has continued to decrease over the past few years (Bray, 2019). In 2018, it reached almost 1.1 Mt. (Bray, 2019; Essentialchemicalindustry.org, 2016a), with almost 90% of it being produced in China. The limited world production of magnesium means that if a new design is needed to achieve grid stabilization using magnesium production plants, the economies of scale would be limited, but that does not mean that grid stabilization could not be economically profitable for the owner of one particular plant, or technically profitable for a local electrical grid.

There exist two main industrial processes to produce magnesium: the Pidgeon process, which is a thermal reduction process mostly used in China with dolomite ore as input, and an electrolytic process used in the US with brine containing magnesium cations as input, such as at the Great Salt Lake facility (Tripp, 2009). To illustrate that there are actually few magnesium production plants around the world, it can be inferred from (Bray, 2019; Essentialchemicalindustry.org, 2016a; Tripp, 2009) that the Great Salt Lake plant alone produces almost 10% of all magnesium produced globally each year. The electrolytic process could be used for grid stabilization, as discussed for aluminum and chlorine production, if locally needed by the grid, with short-term variations of current. However, this article focuses on the most used process, the Pidgeon process, even if its inherent carbon dioxide emissions may not enable it to remain relevant for the long term.

The Pidgeon process takes dolomite ore as input: $MgCO_3 \cdot CaCO_3$. The first step is calcination, which consists of separating gaseous carbon dioxide from solid dolomite in a kiln at around 1300 °C. Only solid magnesium and calcium oxides are then left.



The second step consists of adding solid silicon to magnesium and calcium oxides to separate them, since, at around 13 Pa and 1200 °C (Rizley and Petersen, n.d.), the magnesium is produced as a gas whereas calcium combines with silicates:



Even though the partial pressure of magnesium is 1 atm at 1800 °C, removing the magnesium vapor produced with powerful electric pumps enables high yields, as well as very pure magnesium, up to 99.99%. Magnesium is then cooled down and forged into ingots.

Although the Pidgeon process itself does not consume electricity directly, electricity consumption is a non-negligible part of the process input. Since reactions occur respectively at 1300 °C and 1200 °C, it can be deduced that this temperature is not achieved with heat pumps, whose efficiencies would be too low at such temperatures. Consequently, the only significant electricity consumption in the process comes from the vacuum pumps that enable the second reaction to occur at 13 Pa.

The electricity consumption of those pumps can be adapted very easily, on short and long periods of time, with simple electronic components. At higher than nominal consumption, yields would increase, but this requires a stronger design for the pumps and the vacuum chamber. At lower than nominal consumption, yields would decrease (reaction could even stop at some point), but this requires no special design. The major drawback of this latter opportunity is that there would remain huge thermal losses since the process occurs at 1300 °C; the fuel needed to maintain such temperatures may create costs above the benefits obtained through grid stabilization. Consequently, only short-term load variations should be considered.

In conclusion, electricity consumption of magnesium production plants, whether it comes from the Pidgeon process's pumps or the electrolytic process, can only be adapted on short term horizons, as for aluminum production, because of thermal losses (the electrolytic process takes place at 1000–1200 °C). The rather significant carbon dioxide emissions of the Pidgeon process, however, make it a strange candidate to perform grid stabilization that is mostly needed because of the integration of renewables.

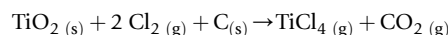
Titanium dioxide reduction

Titanium dioxide is mostly used pure, as a non-toxic white pigment. However, around 5% is transformed into titanium sponge (Seagle, n.d.), which is pure titanium manufactured in a sponge-like structure. The most common process to do so is the Kroll process, although other processes such as the Armstrong process exist, in this case to produce high-purity titanium for electronic applications, as in the American plant located in Salt Lake City, Utah (Bedinger, 2019).

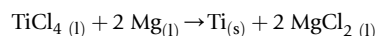
In 2018, 190 kt of titanium sponge were produced worldwide, mostly in China (37%), Japan (27%) and Russia (21%) (Bedinger, 2019). The electricity consumption of the Kroll process is roughly 15 MWh/t (Kosemura et al., 2002), which results in an annual global electricity consumption of 2.85 TWh. This last figure is extremely low compared to the electricity consumption of any industrialized country, but as discussed for magnesium production, there could exist local technical and economic incentives that could make grid stabilization through titanium production plants relevant.

The Kroll process is a three-step chemical process that makes titanium sponge out of titanium dioxide (which is already the result of a chemical process, as it is rarely found at sufficient purity in nature) (Essentialchemicalindustry.org, 2016b):

1. The first step is the carbo-chlorination of titanium dioxide, which takes place in a chlorinator at around 1100 °C. The reaction being very exothermic, cooling is needed to keep the temperature at its nominal value.



2. The second step consists of purifying titanium chloride. This step is performed mostly by simple distillation, even if additional chemical treatment is needed to remove contaminants such as vanadium oxychloride (VOCl_3), whose boiling temperature is too close to titanium chloride's. The result is 99.9% pure titanium chloride.
3. The final step is the exothermic reduction of titanium chloride into titanium sponge. This step is accomplished by adding liquid magnesium to liquid titanium chloride at a temperature of 1100 °C, which must be maintained by cooling. The atmosphere must be free of all nitrogen and oxygen since titanium sponge is very porous and reacts strongly with those gases. Nitrogen inclusions are the first cause of impurities in titanium sponges (Kosemura et al., 2002; Madehow.com, 2019). Consequently, an atmosphere of argon must be maintained during the whole reaction.



Even at 1100 °C, the last reaction is quite slow; this reduction step can take up to four days for an 8-ton batch (Kosemura et al., 2002).

In parallel to the Kroll process, magnesium chloride is recycled through electrolysis into magnesium and dichloride. This parallel step accounts for 60–70% of the electricity consumption of the Kroll process (Kosemura et al., 2002). The opportunity for grid stabilization with titanium production therefore first comes from magnesium production's grid stabilization opportunities, which are described in the previous section of this article.

As for the rest of the process, the electricity consumption is too diffuse to clearly identify the opportunities. Nevertheless, the Kroll process is the first batch process described in this article. Each batch taking a few days, the electricity consumption could be synchronized with periods of the week/month/year where other sources of consumptions on the grid are low, or simply where the services they can provide are most needed.

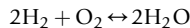
Titanium sponge is rarely directly used after the Kroll process: after being crushed and mechanically compacted, it is forged into ingots in a vacuum electric arc furnace. Electric arc furnaces are the subject of the final section of this article. However, titanium production highlights the strong links between all the industrial processes described in this article: titanium dioxide reduction involves magnesium dichloride electrolysis (which produces magnesium and chloride) as well as electric arc furnaces. In terms of grid stabilization, this opens a door for seasonal load variations. In fact, (Bedinger, 2019) underlines the fact that titanium production plants don't operate at full capacity all year long (capacity is 293 kt/year, which is 154% of annual production). As a consequence, titanium production could be performed during periods of the year where residential and commercial electricity consumptions are low. This would result in the fact that magnesium and chloride production and electric arc furnaces would also be more used in such times of year, reinforcing the load management service provided. In general, by planning the production of oversized industries over the year, a more constant electricity consumption could be achieved all year long, which is a valuable service for grid stabilization.

Hydrogen production

Hydrogen production has been thoroughly studied over the years, as it appears to offer an extremely valuable service that science has been struggling to achieve for decades: electricity storage at grid scale, and at all time scales. This trend has several sources:

- Hydrogen is the most abundant element in the universe (IEA, 2019b), and, in particular, it is present in water, which is not a scarce resource on Earth.

- It is possible to extract hydrogen from water with electrolysis, and it only produces oxygen as a byproduct.
- More importantly, it is possible to convert hydrogen back to electricity by combining it with oxygen, which is also an abundant resource on Earth, with only water as a byproduct. The complete reaction is as follows:

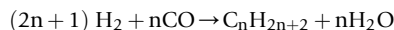


However, when looking at world hydrogen production, the situation is quite different. First, hydrogen production was already described with regard to chlorine production: the 65 Mt. of chlorine produced in 2018 only correspond to 1.8 Mt. of hydrogen, whereas the global production of hydrogen is above 70 Mt. (IEA, 2019b). In fact, 50% of hydrogen is produced by Steam Methane Reforming (SMR), 30% by other hydrocarbon reforming, and 17% by coal gasification, which only leaves less than 3% for electrolysis and other methods (Grandviewresearch.com, 2018).

This current situation comes primarily from the fact that electrolysis is not energy-efficient, and therefore not cost-efficient. Electrolysis is actually 3–10 times more costly than SMR (Raissi and Block, 2004), mostly because its energy requirement is around 4.9 kWh/Nm³ of hydrogen produced, whereas SMR can consume as low as 2 kWh/Nm³ of hydrogen produced. The other hydrocarbon-based processes have intermediate energy requirements. However, high-temperature electrolysis could be a great opportunity to produce hydrogen at lower costs, especially if the temperature is obtained using nuclear reactors or solar concentrators (O'Brien et al., 2010; Ayre, 2013).

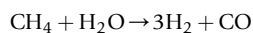
This efficiency issue is further challenged if the full cycle, converting electricity to hydrogen and then hydrogen back to electricity, is considered. Consequently, other end uses of hydrogen are considered so that hydrogen production may present a good opportunity for upward load variations. There are two main ways to do so:

- Hydrogen can be used as a reducing agent instead of carbon or carbon monoxide, for instance in the making of iron from iron oxides (Chatterjee, 2012). This possibility is all the more interesting as the process produces only water as a byproduct with hydrogen, instead of carbon dioxide. Besides, hydrogen is non-toxic gas, whereas carbon monoxide is extremely dangerous.
- Hydrogen, if combined with carbon monoxide, can produce liquid hydrocarbon fuels through the Fischer Tropsch (FT) process (Leibbrandt et al., 2013), which can be summed up by the following reaction:

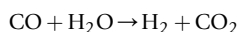


To produce the alkanes ($\text{C}_n\text{H}_{2n+2}$) typical to diesel fuels, n should be around 10. Even with catalysts, this requires the reaction to be performed at relatively low temperatures, and therefore to have low kinetics. The FT has many advantages: it is quite efficient, the hydrocarbons it produces can be stored easily, and it produces hydrocarbons which may be still be needed in a zero net emission system, for planes or trucks. Nevertheless, it requires carbon monoxide as input, and all the industrial processes that produce it burn fossil fuels. In fact, carbon monoxide can also be obtained from the electrolysis of carbon dioxide, which may be widely available as carbon capture devices spread, but this is still early research.

Since hydrogen production may prove to be useful on an industrial scale, it is possible to look into the potential for grid stabilization via the processes that make it. First, SMR is a thermal process with methane and heat as inputs. At a temperature of 700–920 °C and at a pressure of 20–30 bar (Sciencedirect.com, 2015), this process produces synthetic gas containing mostly hydrogen and carbon monoxide:

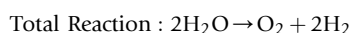
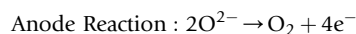


This reaction is endothermic. However, as the temperature falls to around 200 °C, another reaction occurs: the Water Gas Shift Reaction, which is exothermic.



Since almost no electricity is involved in SMR, it has no potential for grid stabilization.

As for the electrolytic process, this article highlights the potential of high-temperature electrolysis (Doctor et al., 2004), for it is the electrolytic process with the highest efficiency. It relies on the following reactions, which occur at around 800 °C:



As high temperature electrolysis is not yet an industrial process, only assumptions can be made on its potential for grid stabilization. First, it probably shares with the other electrolytic processes described in this article the potential for short-term load variations by varying the current of each electrolytic cell. However, this process could be valuable only if coupled with a solar thermal power plant or a nuclear plant since it could consume waste heat, or evacuate heat that is not needed to be converted to electricity for the grid. If a design enabling this were available, it would probably allow longer term production variations of the solar or nuclear power plant, as well as more flexible control of the output of the plant since it has more adaptable technical constraints (especially in the case of nuclear power plants). Grid stabilization through hydrogen production would then be achieved in a very indirect way, far from the perfect storage it seemed to be able to provide at first sight.

Electric arc steelmaking

Electric Arc Steelmaking (EAS) is performed in Electric Arc Furnaces (EAF) and produces pure steel ingots from scrap steel. This process represents 28% of the global production of steel (Worldsteel.org, 2019), the remaining 72% corresponding to blast furnaces in which iron ore is converted to steel ingots by burning oxygen and fossil fuels. Consequently, EAF represents the recycling of used steel parts, whereas blast furnaces are the primary production of steel from ore. Total world production of steel reached 1817 Mt. in 2018 (Worldsteel.org, 2019), including 509 Mt. through EAF. If the increasing global need for steel still limits the share of steel recycled with EAF, it is expected to grow, especially since producing 1 t of steel produces 1.82 t of carbon dioxide on average and EAFs are much less polluting than blast furnaces.

The average electricity of consumption of EAS is around 640 kWh/t of steel produced. This represents a global electricity consumption of 325 TWh, which is 1.5% of the world electricity consumption, or the electricity consumption of the United Kingdom. If all steel was produced through EAF, its electricity consumption would reach 1163 TWh, which is 5.3% of the world electricity consumption. This is the first reason why EAS should be considered on a grid stabilization perspective.

Moreover, typical EAFs consume between 10 and 100 MW (Boulet et al., 2003). If an EAS plant is composed of several EAFs, its electricity consumption can be comparable to the production of typical conventional power plants (coal or gas power plants for instance). As a consequence, EAS plants have the potential to provide huge services for the electricity grid, by consuming or not, since they can provide load variations corresponding to the size of a conventional power plant.

The process of EAS itself is quite simple. After loading a relevant mix of scrap steel in a hearth made of fireclay bricks, with a lining of magnesita-carbon resin bonded bricks, a water-cooled roof is placed on top to form a closed tank. Three carbon electrodes are inserted through the tank roof (for AC EAFs which are the most common type, but there also exist DC EAF where only one electrode is needed, the second one being the hearth itself, but this only works with small hearths) (Wondris et al., 2019). The scrap steel is then melted using electric arcs between the electrodes at temperatures around 1600–1800 °C. In fact, oxyfuel burners are also used to homogenize the temperature in the hearth by burning natural gas in pure oxygen, since hot spots appear behind electrodes. The voltage between the electrodes progressively increases as the electrodes go through the scrap steel mix, until a long stable electric arc is achieved, as Fig. 5 illustrates.

The melted mix obtained then requires a refining step. Oxygen is injected in the hearth, and it combines with the elements with which it has the best affinities, such as phosphorous, sulfur, silicium, magnesium and carbon, before it combines with iron itself. Those oxides form a slag in which carbon is injected and will combine in priority with iron, so it leaves the superficial slag to sink in the core of the melted mix. For impurities such as nickel or copper, they have to be removed by other chemical processes or simply by selecting well the steel scrap before loading the hearth.

Given the previous description of the process, the electricity consumption of EAF is quite complicated to handle for the grid: it is subject to huge and fast variations, and it can go from zero to several hundred MWs in the same day. The fact that the voltage can be controlled suggests an opportunity to provide grid stabilization services without new designs, but this voltage seems to be

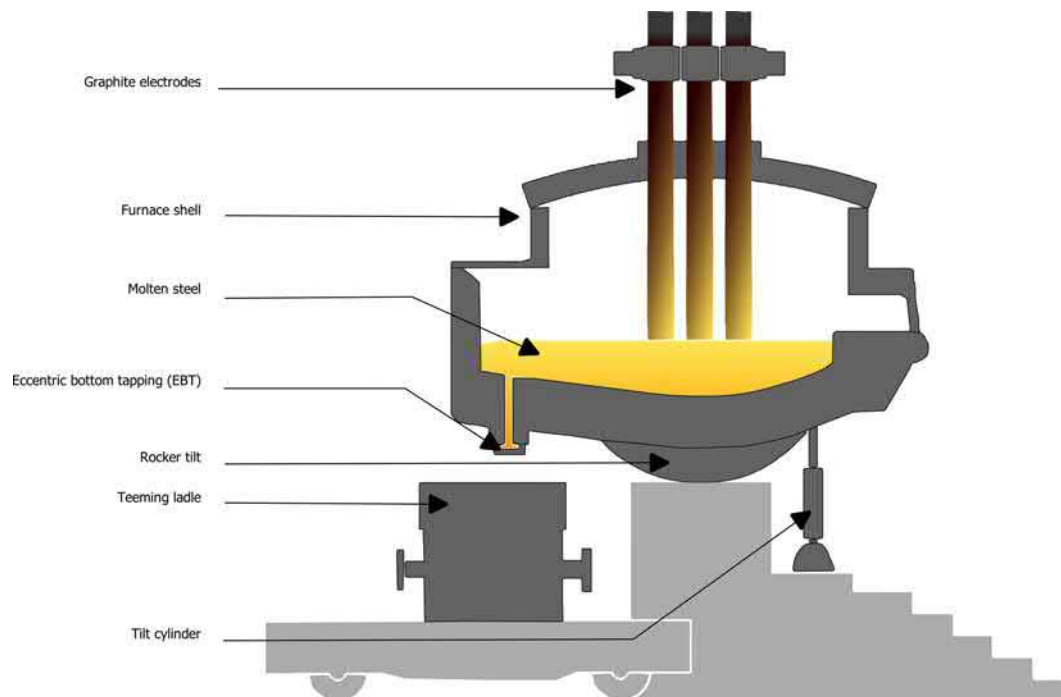


Fig. 5 Diagram of an electric arc furnace. Reproduced from Allied Mineral Products (2020) *Electric Arc Furnace*. [online] Available at: <https://alliedmineral.com/solutions/steel/electric-arc-furnace/> [Accessed 8 Jan. 2020].

determined more by the content of the hearth rather than by an economically-driven design choice which could be adapted according to the needs of the grid.

Consequently, the main service that EAF can provide consists in planning, whether it is at the scale of a day, month or year, when the batch is processed and the steel is stored. Although it may not appear as a very flexible service for the grid, given the growing share of EAF in steel manufacturing and the huge amounts of electrical power it consumes, especially locally, it could eventually have an influence on grid balance as important as maintenance shutdowns for nuclear power plants.

This feature of EAFs should not be underestimated as they are no longer only used for steel making, but also to recycle an increasing number of different kinds of metal. Plasma arc furnaces, which operate in similar ways, and can reach temperatures high enough to melt titanium are also emerging (Svirchuk, 2011). The order of magnitude of EAFs in electricity consumption may therefore be expected to increase with the development of metal recycling and carbon dioxide emissions' reductions.

Conclusion

Industries can provide many different kinds of services for grid stabilization. Those services can have two purposes: either to solve a locally constrained electricity grid, or balance consumption and production of electricity at grid scale, instead of turning on a thermal power plant or to absorb excess power from variable renewable sources. From the point of view of a given industrial plant, both drivers would consist of changing electricity consumption. However, those load variations can be essentially discriminated according to how fast they can happen, and how long they can last.

In terms of fast and short-duration services, the electricity consumption of several significant industrial processes can be adapted. For most electrolytic processes, the current per cell is the result of an economic optimization between Joule losses (thermic losses) and rate of production, and not a technical constraint. As a consequence, if an economic incentive is provided to offer short-term load variations, electrolytic industrial processes can adapt easily to such needs, as it is already the case in several aluminum production plants worldwide. In emergency cases, short-circuiting several cells can be considered, even if it quickly becomes economically inefficient. Nevertheless, those solutions cannot last long since it relies greatly on thermal inertia.

Another possibility for fast grid stabilization services can come from the use of pumps in several thermal or chemical processes; just like the current in electrolytic processes, the pressure induced by pumps can be the result of an economic optimization between increasing the rate of production induced by evacuating a produced gas, and the sizing and electricity consumption of the pumps. In that particular case, if no thermal inertia is involved, the service can be provided on longer time scales, as long as the sizing of the pumps is still profitable.

For fast load variations required to last indefinitely, there can still be solutions to automatically change the consumption of industrial processes. If voltage is a technical requirement in electrolytic processes, some reactions can be, wholly or partially, changed so that the voltage is lowered or increased in exchange of losing a valuable byproduct, such as hydrogen in chlorine production. This kind of service requires expensive new design, but it manages to combine fast and long-duration load variations.

For forecast grid congestions or imbalances that can be anticipated, industrial processes can usually be planned according to grid stabilization needs. Whether it concerns daily, weekly or seasonal planning, it can be extremely relevant for grid stabilization to accurately create economic incentives for industries to consume electricity when it is less problematic for the grid. This solution is all the more efficient as industries in many industrialized countries are oversized, and therefore do not need to produce at maximum capacity all-year long. Besides, in terms of delay of response, it should be possible to postpone batch processes with only a few hours (or less) notice. At this time scale, the forecast production of renewables can change significantly.

To conclude, this article only discusses variations in the electricity consumption of industrial processes, but their contribution to grid stabilization could go beyond those variations of said "active" power. Grid stabilization also relies on stable voltages, which is the result of local balances of "reactive" power. There exists very little literature on the subject; in particular, "reactive" power consumption and production of industrial processes are not easily available, but there could be equally interesting opportunities in that domain.

Acknowledgments

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Thermochemical Hydrogen Production

Sellathurai(Sam) Suppiah, Canadian Nuclear Laboratories Ltd., Chalk River, ON, Canada

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Introduction

The first and the lightest element in the periodic table is H_2 . Although it is the most abundant element in the universe, it is present mostly in combination with other elements and hence offers a challenge to its separation. Fig. 1 shows the global demand for H_2 since 1975 for the production of a range of consumables. In the graph “Other pure”, “DRI” and “Other mixed” are specific applications that require mostly pure H_2 , directly reduced iron steel production and demand for applications that use H_2 as part of a mixture of gases, respectively. The demand has been rising steadily and is expected to continue unhindered with the rising population in the developing parts of the world. Another critical area of growth in the future is in transportation—with the advent of efficient and economically priced fuel cells, the use of H_2 to fuel the transport sector will grow substantially in addition to the demand for other applications.

Currently, the majority of H_2 (>90%) is produced from fossil fuels: natural gas, coal, oil, biomass, etc. In the highly efficient steam methane reforming of natural gas, one of the major processes currently used for mass production of H_2 , natural gas is reacted with steam under pressure and high temperature to produce H_2 , CO and small amounts of CO_2 , and subsequently the CO is reacted with steam (water-gas shift reaction) producing H_2 as the main product and CO_2 as a by-product. The impurities are removed through subsequent processing by a suitable method, such as pressure swing adsorption to deliver pure H_2 .

There are two main consequences to the production of H_2 in large scale using the current fossil fuels-based processes—consumption of large quantities of energy and concomitant emission of greenhouse gases (CO_2 and fugitive CH_4). As shown in Table 1, for the conventional steam methane reforming, the required thermal energy is provided by natural gas combustion, which adds significantly to the emission of carbon dioxide (Table 1).

Although the CO_2 emission is ~ 8.0 t/t of H_2 produced (Table 1), in practice it can be much higher. Incorporation of a carbon sequestration step can reduce emissions. The other fossil fuel based H_2 production processes also represent similar if not greater levels of greenhouse gas (GHG) emissions.

Using the high heating value of methane (55.50 MJ/kg), the total minimum energy required can be estimated (based on 2.92 t of CH_4 per t of H_2 produced) as 162 GJ for the production of 1 million t of H_2 (407 GJ/1 MSCF of H_2).

If 95% of the approximate 70 million t of annual global production of H_2 is assumed to be from fossil fuels, then one would derive ~ 600 million t of CO_2 emission to the atmosphere, just from H_2 production. The energy consumed in the form of natural gas is about 11.3 million Terra Joules, an enormous amount. This simple analysis indicates the significance of GHG emissions and energy consumption that accompany the production of H_2 worldwide.

How can such an enormous amount of GHG emission and the use of fossil fuel-based energy for H_2 production be minimized or avoided? Currently, H_2 can be produced using conventional alkaline or PEM (Proton Exchange Membrane) electrolysis without the emission of GHG. Although these technologies are well developed and can produce H_2 at small and medium scale, they completely rely on electrical energy. Obviously, the electricity needs to be produced from non-fossil fuel to reduce the overall carbon footprint. Nuclear and renewable energy sources can be used to produce electricity—the former is a reliable and steady energy source, while the latter can vary quite drastically depending on geographical location, season and weather conditions.

A significant negative impact of H_2 production processes that rely solely on electrical energy is the overall low efficiency achieved in the conversion of the thermal energy to electrical energy using an energy conversion means such as Rankine Cycle, Brayton Cycle, etc. Unfortunately, a substantial loss (60–70%) in the utilization of the thermal energy occurs due to this conversion using conventional commercial nuclear power plants. So, H_2 production processes that rely on thermal or hybrid thermal/electric process have the potential to offer significantly better overall efficiency. Also, if such processes can operate at high temperatures, the efficiency can be further increased due to the thermodynamic advantage gained at high temperatures (Ebbesen et al., 2014). In consideration of this advantage of high temperature operation, a vast amount of research has been undertaken over several decades into the development of high temperature H_2 production processes. More than 100 processes were identified (Brown et al., 2002), and the development of the most promising ones ensued. These processes include thermochemical cycles, high-temperature electrolysis and

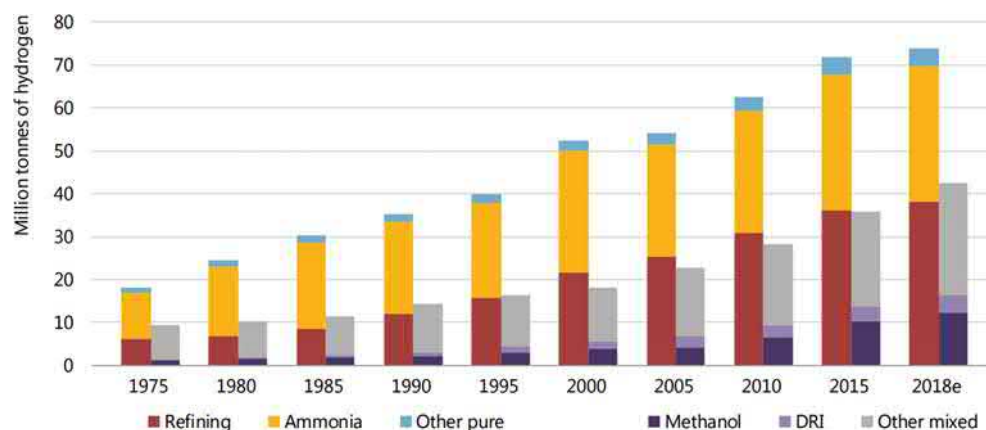


Fig. 1 Global demand for H₂ since 1975. The Future of Hydrogen—Seizing Today's (2019) [iea.org](https://www.iea.org/reports/the-future-of-hydrogen). Retrieved From The Future of Hydrogen—Seizing Today's Opportunities: [iea.org/reports/the-future-of-hydrogen](https://www.iea.org/reports/the-future-of-hydrogen).

Table 1 Production of H₂ from steam methane reforming.

Operation	Reaction	CO ₂ emission (metric tonnes/metric tonne of H ₂)	CH ₄ use (metric tonnes/metric tonne of H ₂)
Reforming (+ WGS)	CH ₄ + H ₂ O → CO + 3H ₂ CO + H ₂ O → CO ₂ + H ₂	5.50	2.00
Combustion for reforming	CH ₄ + O ₂ → CO ₂ + H ₂ O	1.46	0.54
Combustion for steam production	CH ₄ + O ₂ → CO ₂ + H ₂ O	1.00	0.36
Power production	CH ₄ + O ₂ → CO ₂ + H ₂ O	0.04	0.02
Total		8.00	2.92

hybrid thermochemical cycles. Since these processes require high temperature energy sources for their deployment, high temperature nuclear reactors and/or solar technologies are ideal for coupling. The nuclear reactor technologies under consideration in the Generation IV (GEN IV) International Forum ([Stanculescu, 2021](#)) are potential candidates for coupling with these high temperature H₂ production processes.

More recently, a significant level of interest in small modular reactor systems, substantially based on GEN IV concepts, have evolved for niche geographical deployment for a range of applications including H₂ production ([Reitsma, 2021](#)).

Thermochemical and hybrid thermochemical hydrogen production processes

An extensive search for the best candidate thermochemical cycle for development was carried out in the first phase of a U.S. DOE International Nuclear Energy Research Initiative (INERI) project reported by ([Brown et al., 2002](#)). Criteria such as efficiency, practicality, adaptability to a nuclear heat source were used in the screening process. As the outcome of the first phase of this study, the Sulfur-Iodine (S-I) Cycle was deemed the most promising. Consequently, the S-I Cycle was selected for further development in the latter phases of the project potentially due to its highest projected efficiency, the best-suitability for coupling to an advanced high temperature nuclear reactor ([Futterer, 2021](#)) and the greatest potential for further improvement.

Some of the other cycles, namely Hybrid Sulfur (HyS) Cycle, Westinghouse process, Hybrid Westinghouse process, Mark-13 process, UT-3 process and Copper-Chlorine (Cu-Cl) Cycle ([Williamson et al., 1982](#)) have also received significant consideration over the last few decades. To-date, out of all of the thermochemical processes considered for high temperature application, the S-I process has been the most researched globally and promoted through demonstration. The HyS and Cu-Cl Cycles have also been developed to various extents, with the latter receiving significant attention in Canada recently.

Sulfur-iodine cycle (S-I cycle)

Following the initial selection of this process as a potential candidate for development towards industrial application in the US, extensive developments ensued in the United States, Japanese Atomic Energy Agency (IAEA), the French Commissariat à l'Energie Atomique (CEA), the Korean Institute of Energy Research (KIER) and Korea Atomic Energy Research Institute (KAERI), and Institute of Nuclear and New Energy Technology (INET) at Tsinghua University (China). Although the US and France collaborated on the demonstration of the S-I Cycle through an INERI project ([Russ et al., 2012](#)), their interest in this process declined mostly due to

down selection of the high temperature steam electrolysis (HTSE) process for continuing focus. However, JAEA, KAERI/KIER and INNET continued their development of the S-I process and resolved many of the technical issues through extensive fundamental and engineering studies.

A pictorial view of the S-I Cycle is shown in Fig. 2 and a simplified schematic is provided in Fig. 3.

In the Bunsen reaction section, SO_2 and I_2 react with H_2O to produce H_2SO_4 and HI . The product solution is separated spontaneously into $\text{H}_2\text{SO}_4\text{-H}_2\text{O}$ solution and $\text{HI-I}_2\text{-H}_2\text{O}$ (HIx) solution. In the H_2SO_4 section, impurities of iodine compounds are removed from the $\text{H}_2\text{SO}_4\text{-H}_2\text{O}$ solution by reverse Bunsen reaction and vapor-liquid separation. Then, H_2SO_4 is concentrated by H_2O vaporization. The concentrated H_2SO_4 is decomposed in two steps; H_2SO_4 vaporized and decomposed into SO_3 and H_2O in the first step and the SO_3 is decomposed into SO_2 and O_2 in the second step. The SO_2 and O_2 in the product gas stream is returned to the Bunsen section, from where O_2 exits as the main potentially valuable by-product of the process. In the HI separation subsection, purification of the HIx is carried out first by removing minor sulfur compounds for subsequent separation of HI . If the HIx solution is distilled to remove HI without any concentration, an azeotropic mixture of $\text{HI-H}_2\text{O}$ is vaporized because the solution is hypo-azeotropic. Hence, a method for HI pre-concentration is critical to avoid excessive energy consumption in HI distillation. Since an energy-efficient method of pre-concentration of HI is required upstream of distillation, an important innovation in

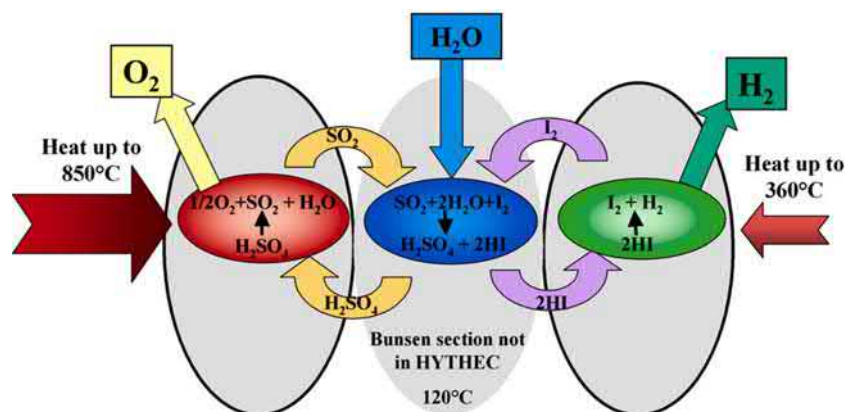


Fig. 2 A pictorial view of the S-I Cycle.

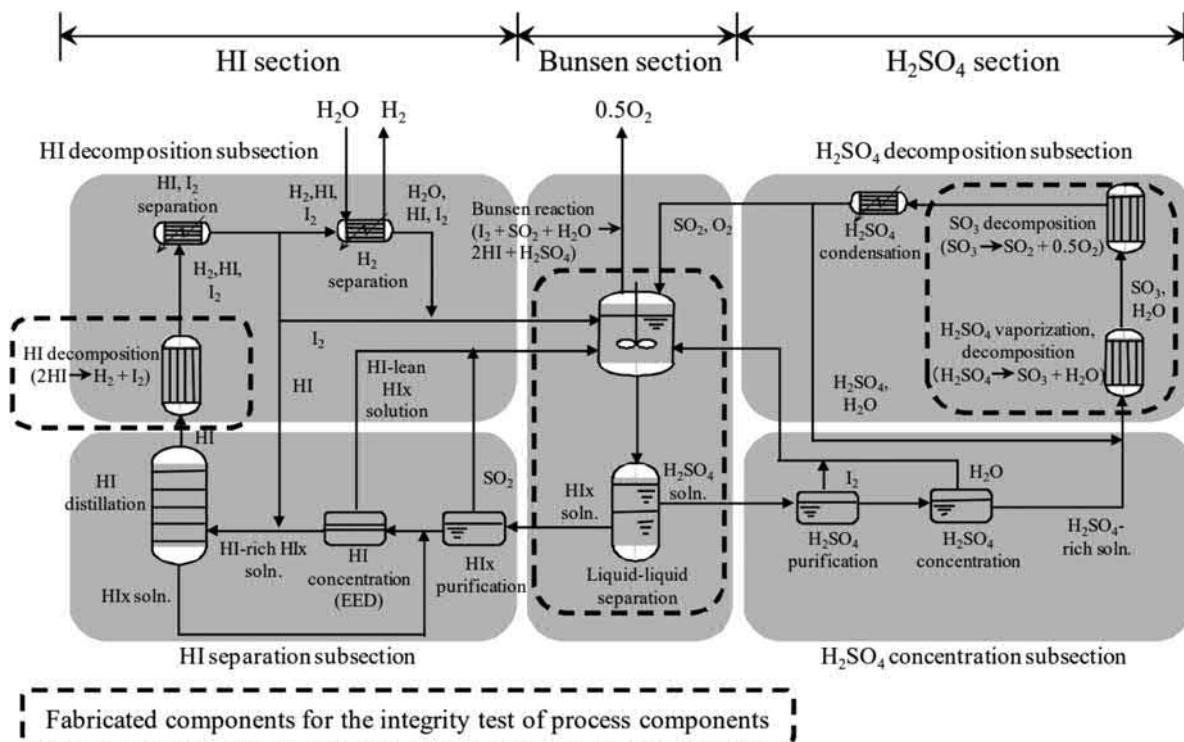


Fig. 3 A simplified schematic of the S-I Cycle (Kasahara et al., 2014).

the form of electro-electrodialysis (EED), first proposed by Onuki et al. (2000), was implemented by JAEA to concentrate HI to hyper-azeotropic. In the EED cell, HI in the solution is dissociated to proton (H^+) and iodide ion (I^-). Iodide ion (I^-) in the anolyte is electrochemically oxidized to I_2 at the anode. H^+ in the anolyte permeates through the cation exchange membrane to the catholyte. I_2 in the catholyte is electrochemically reduced to I^- at the cathode. Consequently, HI in the catholyte is concentrated to hyper-azeotropic to avoid excessive consumption of energy in the subsequent distillation step.

EED technology was further developed and demonstrated in the form of a stack of cells by INNET in 2014, at their demonstration of the S-I Cycle in the IS-100 facility designed for 60 NL h^{-1} H_2 production (Zhang et al., 2018). HI from the distillation column is catalytically decomposed at $350\text{--}450^\circ\text{C}$ into H_2 and I_2 . The product I_2 and remaining HI are separated from the product gas and returned to the Bunsen section and the HI separation subsection, respectively. Small amounts of HI and I_2 in the remaining gas is captured into water, which is fed from outside to compensate for the H_2O consumption as H_2 and O_2 products. Remaining H_2 is obtained as the product of S-I process.

Development of catalysts for HI decomposition is one of the most intensively investigated aspects of the S-I process. Mainly, Pt based monometallic and multi-metallic catalysts supported on a range of support materials have dominated as potential catalysts for this application. Based on these results, Pt-Ir/activated carbon was selected as the catalyst for HI decomposition in the closed-cycle S-I experiment in INNET's Integrated Laboratory System (ILS) (Zhang et al., 2018), with good activity and stability. However, verification of long-term stability of HI decomposition catalysts is still pending.

The decomposition of H_2SO_4 is a common reaction step of the sulfur-family of thermochemical cycles. This reaction is composed of two steps: thermal decomposition of H_2SO_4 to SO_3 and H_2O at approximately 450°C and decomposition of SO_3 to SO_2 and O_2 at $\sim 800^\circ\text{C}$. Since the latter reaction is kinetically slow and carried out at very high temperatures, there have been extensive investigations into catalyst performance and decomposer design. The general conclusion has been that Pt supported catalysts offer the best performance and acceptable stability followed by Fe_2O_3 -based transition metal catalysts. Any instability observed of the catalysts has been attributed to the formation of sulfate, agglomeration of catalyst particles, and the decreasing specific area during the process.

A Bayonet-type decomposer design Fig. 4 (Corgnale et al., 2020) integrating both decomposition steps has been adopted by most groups working in this area. This design enables effective management of heat between the two reactions efficiently. H_2SO_4 solution is fed at the bottom of the outer manifold of the decomposer with the solution rising through the space between the inner and outer tubes of SiC and is vaporized by heat from outside. The decomposition of H_2SO_4 into SO_3 and H_2O occurs in

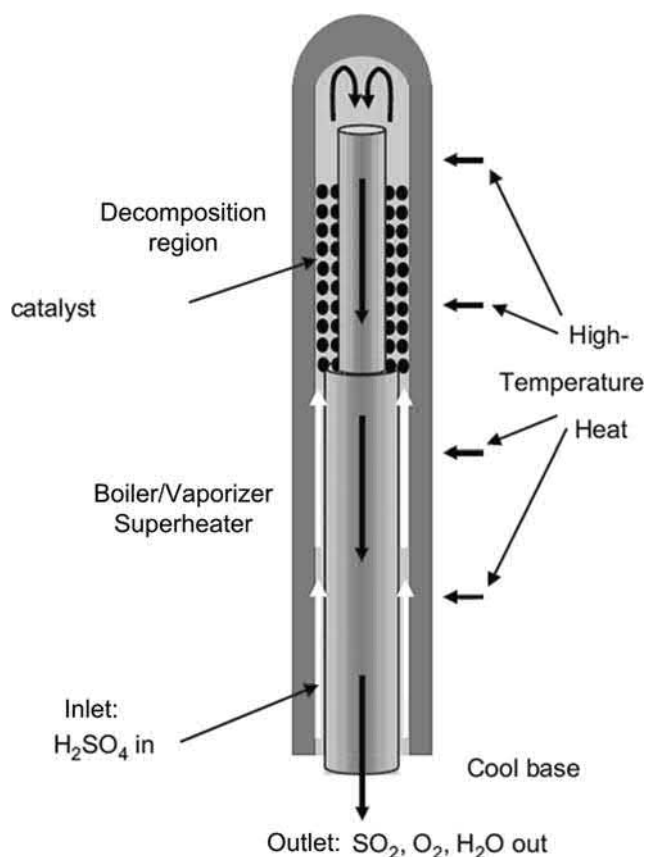


Fig. 4 Bayonet type H_2SO_4 Decomposer (Corgnale et al., 2020).

the lower temperature bottom section of the outer shell and the rising vapor is decomposed in the catalyst section with further heating provided externally as well as the product gases exiting downwards through the inner tube. The requirement for the very high temperature for the decomposition of SO_3 to O_2 and SO_2 necessitates a very high temperature heat source such as very high temperature gas cooled reactor or high temperature solar thermal technology.

Various efforts have been undertaken to develop effective simulation tools for designing and optimizing the S-I Cycle flowsheet and operational parameters, as well as for evaluating energy consumption. In general, Aspen Plus has been commonly used as the platform for this purpose with custom built thermodynamic data packages for phase separation, H₂ distillation, and EED.

Although the long-term continuous and stable operation of the closed-cycle S-I process still remains a challenge because of several engineering and technical difficulties, several closed-cycle loops have been built and operated for short periods over the last two decades. These include the Lab-Scale Integrated Experiment by the United States and France (Russ et al., 2012) designed for 100 L h^{-1} H_2 production at up to 20 bar pressure; Japan's integrity test facility designed for 200 L h^{-1} H_2 production and selection of engineered materials; China's INNET Integrated laboratory-scale S-I facility designed for 100 L h^{-1} H_2 production; and Korea's S-I integrated facility designed for high pressure operation and 50 L h^{-1} H_2 production (Shin et al., 2015). As part of the down-selection effort, Russ and Buckingham (2009) derived a thermal efficiency of $\sim 38\%$ for the S-I H_2 plant assuming a 40% electrical conversion efficiency for a Next Generation Nuclear Power (NGNP) plant coupled to the S-I plant.

Hybrid copper-chlorine cycle (Cu-Cl)

Following the initial identification of this cycle as a potential process for large-scale production of H_2 using moderate temperature nuclear reactor systems, Argonne National Laboratories, under the leadership of Michelle Lewis, examined several cycles based on the Cu-Cl chemistry (Lewis et al., 2003) and selected the most promising cycles for their research activities. The Cu-Cl thermochemical cycle is a hybrid cycle that uses both heat and electricity to carry out a series of chemical reactions and an electrochemical reaction with the net result being the splitting of water into H_2 and O_2 . The Cu-Cl process is of interest because the maximum required temperature is only 530°C , which means that the thermal requirement of this process can be fulfilled by moderate temperature next generation nuclear reactors such as Super Critical Water Reactors, sodium cooled fast reactor, molten salt reactors, and solar thermal technologies.

Over the last two decades, two variations of the Cu-Cl Cycle have undergone considerable research. In the cycle originally considered for development (Table 2) (Lewis et al., 2003), copper metal is produced electrochemically by the disproportionation of cuprous chloride (CuCl) (Table 2, Eq. 2) in an aqueous HCl electrolyte. Hydrogen gas is then produced by a chemical reaction that takes place between copper metal and gaseous HCl at a temperature of $425\text{--}450^\circ\text{C}$ (Table 2, Eq. 1). The cycle is completed with the steps shown in Eqs. (3)–(5) (Table 2). In the early days of the development of this 5-step Cycle, one of the main impediments to the progress was related to the electrolysis step; dendritic copper growth on the electrode and high voltage resulted in high power consumption. Research activities at AECL in Canada helped to overcome these issues by focusing on the single electrochemical step (Eq. 6, Table 3). It was recognized that the two reactions shown in Eqs. (1) and (2) (Table 2) could be combined/replaced by a single electrochemical reaction that generates H_2 gas directly as shown in Eq. (6) in Table 3. During the last decade, developments in Canada have mainly focused on the 4-step Cycle shown pictorially in Fig. 4 (Eqs. 4 and 5 in Table 2 and Eqs. 6 and 7 in Table 3). The feasibility of each step involved in the cycle has been verified (Naterer et al., 2017) by many groups working on this cycle. Some of the advantages of the Cu-Cl Cycle compared to the S-I Cycle are its moderate temperature requirement, absence of catalyst requirement for the chemical steps, and inexpensive and significantly lower hazardous nature of the chemicals involved. However, the main disadvantage is that there are solid chemicals involved in the cycle and they require special and possibly difficult handling considerations (Fig. 5).

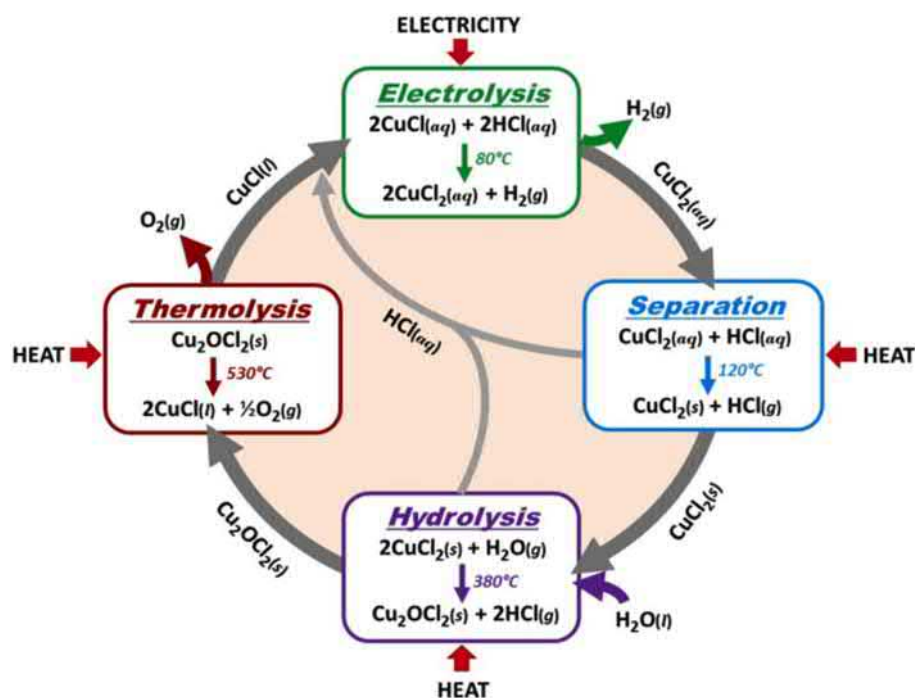
A collaborative project between AECL and the Ontario University (formerly known as University of Ontario Institute of Technology) in Canada produced significant progress within a short time frame towards the development of the cycle through extensive fundamental and engineering studies of all the steps involved in the cycle (Naterer et al., 2017). Naterer et al. (2014, 2015, 2017) have detailed the progress on each of the steps in a series of papers. Li et al. (2020) have detailed the development of the electrochemical step to enable H_2 production at a rate of 50 L h^{-1} at 0.4 A cm^{-2} in a 2.0 M CuCl solution at 80°C using a 100 cm^2 area cell. In addition, studies have led to progress in achieving a significant reduction in the noble metal present in the electrolyser, thus positively impacting the electrolyser CAPEX.

Table 2 Reactions involved in the Cu-Cl Cycle—original 5-step option.

Equation	Reaction stoichiometry	Temperature ($^\circ\text{C}$)	Remarks
1	$2\text{Cu(s)} + 2\text{HCl(g)} \rightarrow 2\text{CuCl(l)} + \text{H}_2\text{(g)}$	425–450	Thermal
2	$4\text{CuCl(a)} \rightarrow 2\text{CuCl}_2\text{(a)} + 2\text{Cu(s)}$	< 100	Electrochemical
3	$\text{CuCl}_2\text{(a)} \rightarrow \text{CuCl}_2\text{(s)}$	< 100	Drying
4	$2\text{CuCl}_2\text{(s)} + \text{H}_2\text{O(g)} \rightarrow \text{Cu}_2\text{OCl}_2\text{(s)} + 2\text{HCl(g)}$	350–400	Hydrolysis
5	$\text{Cu}_2\text{OCl}_2\text{(s)} \rightarrow 2\text{CuCl(l)} + \frac{1}{2}\text{O}_2\text{(g)}$	450–530	Thermal

Table 3 Electrochemical and separation/drying steps of the 4-step Cu-Cl cycle.

Equation	Reaction stoichiometry	Temperature (°C)	Remarks
6	$2\text{CuCl(aq)} + 2\text{HCl(aq)} \rightarrow \text{H}_2(\text{g}) + 2\text{CuCl}_2(\text{aq})$	< 100	Electrochemical
7	$\text{CuCl(a)} + \text{CuCl}_2(\text{aq}) + \text{HCl(aq)} \rightarrow \text{CuCl}_2(\text{s}) + \text{CuCl(s)} + \text{HCl(g)}$	< 100	Separation/drying

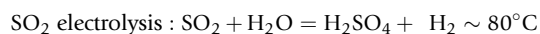
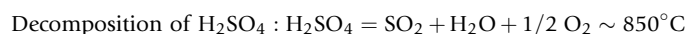
**Fig. 5** The 4-step Cu-Cl hybrid thermochemical cycle.

Out of all the steps involved, the hydrolysis step appears to be the most challenging with regard to achieving the performance required to integrate the four steps. Some of these issues are in feeding solid CuCl_2 to the hydrolysis reactor with steam, minimizing the steam to CuCl_2 ratio while maintaining high conversion to copper oxychloride (Cu_2OCl_2). On the other hand, development of the thermolysis step has proven to be less challenging. Significant progress has been reported by [Vega et al. \(2020\)](#) through proprietary developments on all the steps of the process to enable integration of the four steps of the process to produce $50 \text{ L h}^{-1} \text{ H}_2$ during 2021 ([Fig. 6](#)).

[Dincer and Balta \(2020\)](#) assessed the thermodynamic performance of the Cu-Cl Cycle and derived an efficiency of 40–50%, signifying the attractiveness of this moderate temperature cycle for nuclear based H_2 production.

Hybrid sulfur cycle (HyS)

The hybrid sulfur (HyS) Cycle ([Fig. 7](#)) uses both thermal and electrical energy to carry out the two steps, one chemical, the other electrochemical, to achieve the objective of splitting water into H_2 and O_2 .



The H_2SO_4 decomposition step is the same as that of the S-I process, requiring heat from a very high temperature source such as a gas-cooled reactor ([Futterer, 2021](#)) or concentrated solar. Since the H_2SO_4 decomposition is highly developed as part of the S-I process, the main focus of the HyS process development has been on the SO_2 de-polarized electrochemical step. In the 1970s Westinghouse Corporation developed a basic electrolyzer with two compartments separated by a membrane, which was followed by the use of PEM electrolyzer technologies. The HyS Cycle was under considerable development effort in the early 2000 with some critical work led by W. Summers at the Savannah River National Laboratory ([Summers, 2006](#)). Following the down selection of HTSE for

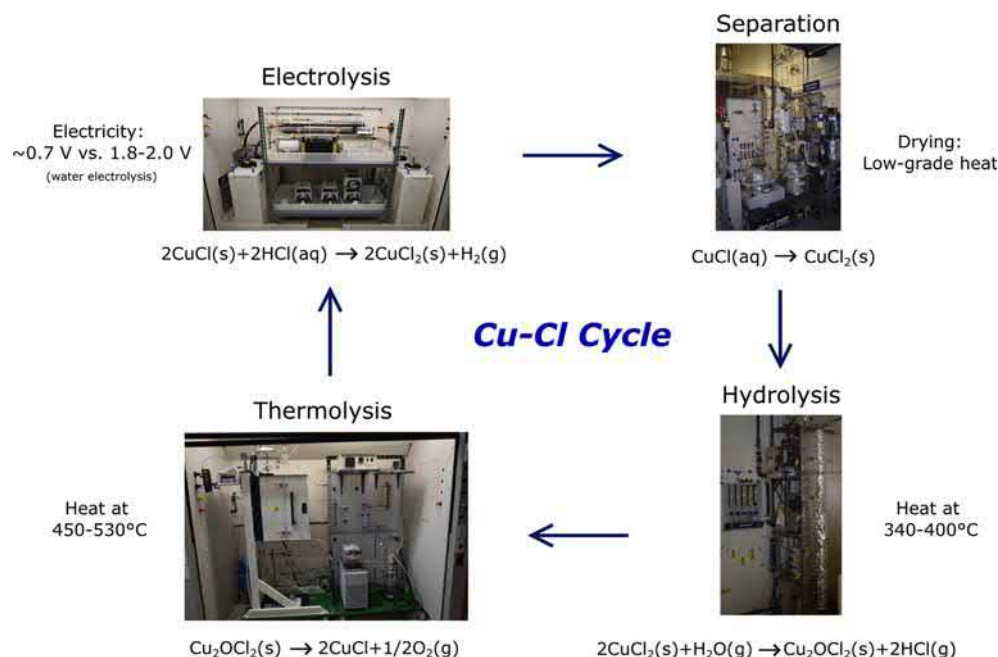


Fig. 6 Equipment representing the 4-step Cu-Cl cycle.

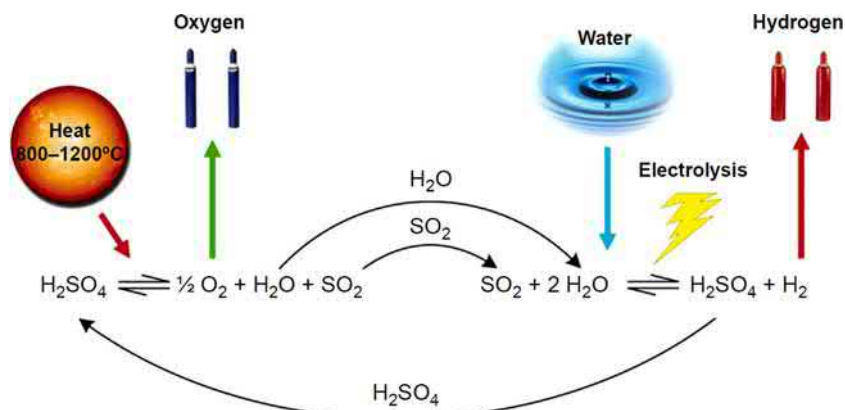


Fig. 7 A pictorial view of HyS process (Roeb et al., 2012).

advancing nuclear H₂ production in the United States, further development of HyS Cycle practically came to a halt in the United States. However, some research activities continued in Europe Roeb et al. (2012) and other parts of the world (Diaz et al., 2019) mostly as basic studies with no known large-scale demonstration planned towards industrialization of the technology.

The conceptual requirements considered by Summers (2006) for the development of the SO₂-depolarized electrolyzer were that the SO₂ is oxidized at the anode at a practical voltage in the range 0.5–0.6 V with high current density operation at 20 bar pressure and 100–120 °C with 50–60 wt% H₂SO₄ without SO₂ crossover. Although the conventional PEM electrolyzer technology was applied, SO₂ crossover issues needed to be resolved. Diaz et al. (2019) have conducted an extensive experimental investigation into the catalysts used on the anode, and reviewed the latest published research in this area to conclude that Pt-based catalysts are the most promising anode catalysts for the SO₂ depolarized electrolyzer even though the performance was still lower than the target of 0.5 A cm⁻² at 0.6 V required for economic application. More recently, Kruger et al. (2015) have reported their studies of various operating methods and electrolyte/electrode parameters for SO₂ electrolysis.

Energy requirements and economics

The cost of hydrogen produced from any process would depend on its CAPEX and OPEX. One of the major components of OPEX is the energy requirement for the process. A rudimentary comparisons of the energy requirement for each method is presented in

Table 4 Energy requirement, capital cost and H₂ production cost per kg.

Method	Thermal energy requirement (GJ/t H ₂)	Capital cost	H ₂ cost (US\$/kg)	Notes
PEM	645	US \$300 M for a 100 t per day H ₂ plant (US \$ 1332/kWe)	2.22 to 9.12	https://www.sciencedirect.com/topics/engineering/hydrogen-production-cost 2016 estimates
S-I	624	US \$2672 M for 400 t per day H ₂ plant	10.71	Allen et al. (2009)
HyS	394	US \$916 M for 360 t per day H ₂ plant	6.83	Allen et al. (2009)
HTSE	359	US \$509 M for 360 t per day H ₂ plant	6.04	Allen et al. (2009)
Cu-Cl	297	US \$131 M for 125 t per day H ₂ plant	4.00	Ferrandon et al. (2008)

Table 4 with the capital cost and the hydrogen cost of production. It should be noted that these values are extremely approximate considering the limited reliability of the assumptions and the low Technology Readiness Level (TRL) development of the advanced production methods. Even for the well-established low temperature PEM electrolyser technology, a review of the CAPEX data revealed very wide ranging values. As a result of the uncertainties in the values, no effort was made to harmonize the data, and the data presented in **Table 4** should be considered only illustrative. The energy requirement for PEM electrolysis is the most reliable value among all values shown. On the other hand, for the advanced production methods, although there has been significant advancements in the developments towards increased efficiency and cost reduction, there is no reliable data available due to technical challenges faced by the developers. It should be noted that the values presented in **Table 4** for the S-I Cycle do not reflect the recent progress that has been achieved in Japan and China on the development of this cycle. Similarly, the efforts in France (EU) and the United States on the development of HTSE technology is expected to positively affect values presented in **Table 4**. For the Cu-Cl Cycle, the developments in Canada is expected to improve on the values shown in **Table 4**. For the low temperature electrolyzer (PEM or alkaline), more reliable CAPEX cost may be obtained directly from manufacturers while the OPEX would depend on the local availability and price of electricity. The references presented in the table provide the assumptions and the validity of the estimates.

Conclusions

In summary, currently, the S-I and Cu-Cl Cycles are being developed with significant global interest for demonstration of integrated continuous operation. Japan and China are continuing with significant progress in the S-I Cycle while Canada is planning to demonstrate the integrated operation of the Cu-Cl Cycle in 2021. Economic application of these processes would ultimately depend on positive outcomes on the overall efficiency achieved (e.g., energy consumption), long-term endurance of structural materials employed for carrying out the chemical steps and the cost of H₂ produced from these processes. The development of all the advanced processes has progressed significantly since the data presented in **Table 4** was published. However, substantial additional progress is required to establish valid CAPEX and OPEX data on these processes before reliable hydrogen production costs can be estimated. However, the recent global activities clearly demonstrate the interest and potential for these processes to become viable methods for eliminating GHG emissions from the conventional hydrogen production processes.

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High Temperature Steam Electrolysis

Richard D. Boardman, Idaho National Laboratory, Idaho Falls, ID, United States

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Glossary

Hybrid production plant A combination of two power generation or industrial plants that share one or more energy sources or material inputs and produce variable amounts of energy products to optimize revenue or to meet other time varying objectives.

Integrated energy system An integrated energy system, including hybrid production plants, that is connected by physical or cyber connections and operating in a manner that optimizes any or all of the following objectives; system flexibility and efficiency, revenue, resilience, security, resource stewardship, and environmental sustainability.

Temperature boosting An action that raising the temperate of a given material.

Introduction

Hydrogen is currently used for a wide variety of applications: production of ammonia for fertilizers, hydrodesulfurization of crudes and natural gas, hydrocracking of petroleum, refining low-quality (heavy oil) resources, direct iron reduction, and power generation in a combustion device or fuel cell. Large-capacity hydrogen related energy storage concepts can help accommodate variable power generation sources (wind, solar, hydro) and seasonal periods when baseload power capacity is reduced (e.g., from nuclear and fossil sources) (Mogensen et al., 2019). As such, there has been widespread interest in the development of large-scale hydrogen production technologies. Among the various hydrogen production options, such as low-temperature electrolytic (alkaline and proton-exchange membrane electrolysis), high-temperature electrolytic (steam electrolysis) and high-temperature thermochemical (e.g., sulfur-iodine cycle, copper-chloride) processes, a high-temperature steam electrolysis (HTSE) plant is perceived as a promising flexible load resource (Kim et al., 2018). HTSE that utilizes the thermal and electrical power produced by nuclear reactors has several advantages over other options: non-fossil process, no production of greenhouse gas (GHGs), high electrolysis efficiency due to minimized voltage and current losses (which in turn decrease with increasing temperature), and a less severe corrosive condition under a high-temperature environment compared to thermochemical process (O'Brien et al., 2010; Bo et al., 2010). As a result, in the past decade, there has been a growing interest in research and development on HTSE stacks for producing hydrogen.

This chapter begins with a review of the basic materials, electrochemical physics, and transport phenomena of HTSE cells and stacks, heat management, and controls systems that enable HTSE to optimally exploit nuclear energy. This is followed with a review of recent technical and economic assessments of the value proposition for producing hydrogen using energy generated by nuclear power plants. Next, a description of the application of HTSE within nuclear integrated systems and operation as hybrid plant, and as reversible fuel cell/electrolysis cell units that can firm high penetration of variable renewable energy power generation sources is provided to bolster the role of nuclear energy. The potential benefits of co-electrolysis of steam and CO₂ is covered to provide an understanding of the emerging potential to extend nuclear energy into the transportation and steel making industries. Finally, an overview of commercial advancements and testing underway at institutions around the globe is provided.

Cell makeup and materials

A high temperature solid oxide cell comprises three main components: an electrolyte and two electrodes. The electrolyte is a ceramic membrane that can conduct ions and is sandwiched between two porous electrodes that can conduct. More than three decades have been dedicated to developing oxygen-ion (O^{2-}) conducting electrolyte cells which can be operated in fuel cell mode (SOFC) burning natural gas or hydrogen, or in electrolysis mode (SOEC), splitting steam when the electricity poles are reversed causing the oxygen ion to conduct in the opposite direction (see Fig. 1) [Oxygen-Conducting SOEC].

SOECs are operated in the range of 700–1000 °C and are further capable of splitting steam and CO_2 mixtures into synthesis gas (syngas; $CO + H_2$) (Stoots et al., 2009). Within the past decade, efforts have been intensified to develop a reversible proton-conducting (H^+) electrochemical cells that operate in the range of 400 to 600 °C and can be used to split steam (Fig. 2) (Ding et al., 2020).

A review summary of the state of cell development can be found in Hauch et al. (2020), who remark:

“SOEC technology has undergone tremendous development and improvements over the past 10–15 years. The initial electrochemical performance of state-of-the-art SOEC single cells has more than doubled, while long-term durability has been improved by a factor of ~ 100 . Similar improvements in performance and durability have been achieved on the stack level. Furthermore, SOEC technology is based on scalable production methods and abundant raw materials such as nickel, zirconia, and steel, not precious metals. Performance and durability improvements as well as increased scale-up efforts have led to a hundredfold gas production capacity increase within the past decade and to commissioning of the first industrially relevant SOEC plants. Over the next 2–3 years, plant size is expected to further increase by a factor of almost 20.”

The main factors of consideration in designing a cell are summarized by Minh (2016). Options include choice of:

- Ion-conducting ceramic electrolyte; either O-SOEC or H-SOEC
- Electrode materials; fuel side (either steam/ H_2 or CO_2/CO) and oxygen-evolution side (sometimes referred to as the air side electrode)
- Supporting layer; either by the electrode, electrolyte layer, or a metal-plate; the support must provide mechanic stability to the cell during fabrication, assembly of cells into stacks, and operation of the stacks when thermo-mechanical stresses are exerted
- Geometry; the most common designs are planar with gas and ion mass transport being orthogonal to the plane; cylindrical designs with radial gas and ion transport have also been proven successful
- Coatings that are applied between the electrodes and electrolytes to mitigate reactions and migration of elements between the ceramic layers
- Plates and flow passages for gas distributor and gas collection from the respective electrodes, and which serve to deposit externally supplied heat or remove excess heat that is generated above the thermal neutral point
- Connection of electrical connectors and charge distributors
- Sweep gas for the oxygen electrode, if any
- Hydrogen content added to the inlet steam gas to mitigate Ni oxidation
- Steam content left in the outlet hydrogen stream to avoid hydrogen embrittlement of the cell interconnects.

Fig. 3 illustrates a typical oxygen-conducting cell configuration buildup.

Oxygen-conducting electrolytes

The functional physics of the most common materials used in oxygen-conducting solid oxide cells were first described by Gazzari (2007). The main cell components have not changed significantly in 15 years. For an electrolyte-supported cell, yttria-stabilized zirconia (YSZ) is used. A composite of metallic Ni and YSZ is most often used for fuel side electrode supported cells. The performance of the electrolyte depends on how well it can conduct oxide oxygen ions. Yttria content of at least 8 mol% (connotated as 8YSZ) is needed to fully stabilize the electrolyte composition. Other electrolyte materials with higher ionic conductivity, such as scandia-stabilized zirconia (ScSZ) and various stoichiometries of lanthanum/strontium/gadolinium/magnesium oxides (LaSr-GaMgO), are also receiving significant attention.

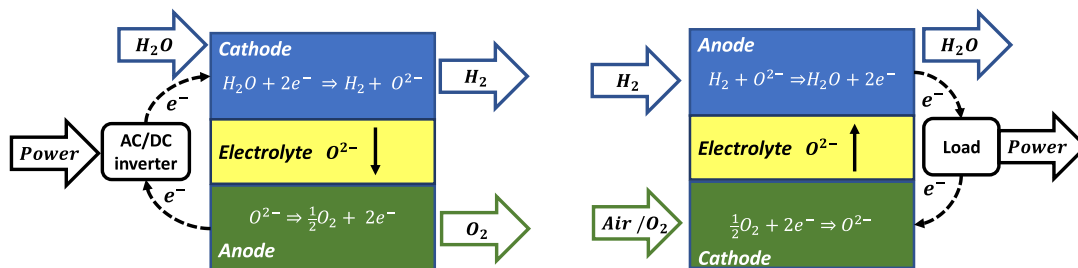


Fig. 1 Reversible solid oxide cell operation in electrolysis mode (A) and fuel cell mode (B).

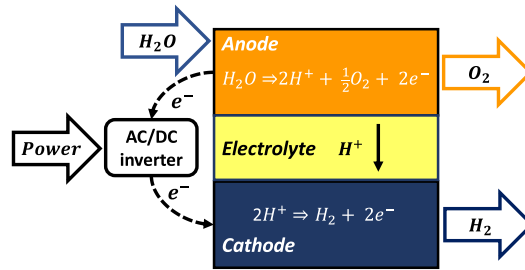


Fig. 2 Proton-conducting solid oxide cell operating in electrolysis mode.

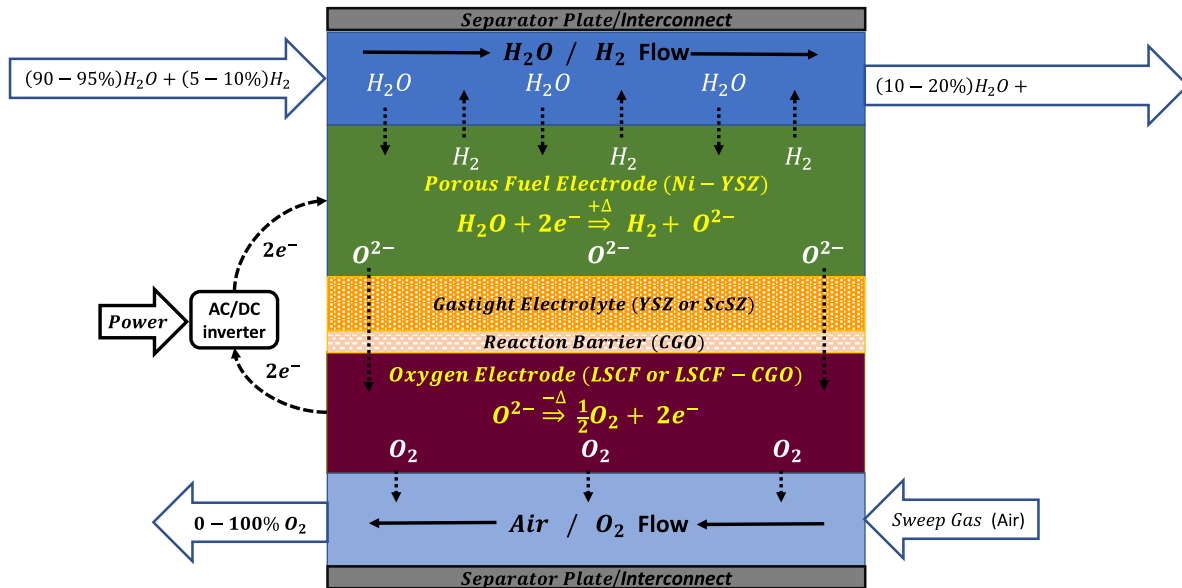


Fig. 3 Typical components in an electrode-supported oxygen ion conducting SOEC.

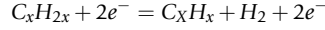
The compositions and stoichiometries of the oxygen electrode have evolved more over time and can vary according to specific developer cell design and manufacturing objectives and experiences. Strontium-doped lanthanum/manganese oxides (LSM), has evolved to mixed conductors- such as stoichiometries of lanthanum/strontium/cobaltite/ferrite (LSCF) and lanthanum/strontium/cobaltite (LSC) (Hjalmarsson et al., 2014). In order to prevent reactions and species migration (e.g., Y, Zr, La, Sr) between the oxygen electrode materials and YSZ electrolyte, thin layers (0.1- to 5 mm) of gadolinia-doped ceria (CGO) are commonly applied (Ploner, 2019).

Proton-conducting electrolytes

Proton-conducting electrolyte based cells (H-SOEC) represents a promising developing technology for steam splitting. Advantages include conversion at reduced temperatures which helps mitigate thermal cycling stresses on the cells and stacks leading to longer system durability and the possibility of using less expensive materials for the cell metallic interconnects and manifolds. However, the challenge for any H-SOEC is overcoming sluggish electrode kinetics at intermediate temperatures and materials corrosion under high-steam concentration which are noted by Lei et al. (2017) and Grimaud et al. (2012).

An important focus for H-SOEC is the development of a durable oxygen electrode, which bears the rigorous function of splitting the H_2O molecule with efficiencies comparable to fuel side electrodes of an O-SOEC. Currently, both the water oxidation reaction and the oxygen reduction reaction must occur in the triple phase boundary where ion, electron, and gas meet. This requires a mixed ion/electronic-conducting electrode material (referred to as a triple conducting oxide, TCO), and it is further desirable to introduce proton conduction to extend the triple-phase boundary from the electrode bulk to the electrolyte/electrode interface. The work by Dong and Ding at Idaho National Laboratory (Ding et al., 2020, 2018) leads the U.S. Department of Energy efforts to develop H-SOEC materials. Recent research has focused on the creation of a perovskite TCO electrode with a stoichiometry of $\text{PrNi}_{0.5}\text{Co}_{0.5}\text{O}_{3-\delta}$ (PNC) which exhibits relatively high water oxidation and oxygen reduction activity, and excellent durability and thermal cycling capability at reduced temperature range (400–600 °C). In addition to water splitting capability, the potential to

use H-SOECs to abstract hydrogen from ethane and other alkanes which produces hydrogen and valuable ethylene, and higher alkenes that are feedstocks for polymer production, as shown in the equation below, has been proven.



SOEC operating conditions

The ionic conductivity of ceramics is highly dependent on the ceramic temperature and the thickness of the electrolyte. High operating temperatures (700–850 °C) are required to obtain sufficient overall conductivity in the oxygen-conducting electrolyte cells, while the target range for proton-conducting cells is lower (400–600°). The thinner the electrolyte, the higher its ion conductivity and the lower the cell's ohmic resistance and loss of power to joule heating. In an electrolyte-supported cell, mechanical strength is provided if the electrolyte thickness is large (150–250 μm), which leads to relatively high ionic resistance. Therefore, if the mechanical strength can be provided by the steam/H₂ electrode, the electrolyte thickness can be reduced by a factor of 10 or so. The total polarization loss of an operating cell is governed by three dominant factors: activation (or charge transfer) polarization (η_{act}), concentration (or diffusion) polarization which includes chemical reaction polarization (η_{conc}), and ohmic resistance polarization (η_{ohm}). Under the same operating conditions of temperature and current density, a SOFC and a SOEC are likely to have the same ohmic and activation overpotentials. However, the concentration overpotential values are different for the SOEC and SOFC because the gas transport mechanisms through the electrodes are different in two cases.

Fundamental thermodynamics

A comprehensive treatise of the thermodynamics of SOEC is given by O'Brien (2008). In order to electrolytically split H₂O or CO₂ molecules, a voltage must be applied across the cell that is greater than the open-cell potential. The standard-state open-cell potential, V^o , is given by:

$$V^o = \frac{\Delta G_R^o}{jF}$$

where j is the number of electrons transferred per molecule of hydrogen produced. For the steam-hydrogen system, $j=2$, O²⁻ ions are transported through and O-SOEC electrolyte (or $j=1$ for H⁺ ions transported through H-SOEC). The standard-state open-cell potential applies to the case in which pure reactants and products are separated and at one standard atmosphere pressure. In most practical HTE systems, the incoming steam is mixed with some hydrogen and possibly some inert gas. For O-SOEC, hydrogen is required in order to maintain reducing conditions on the nickel cermet electrode. Also, it is not desirable to run the electrolyzer to 100% steam utilization, because localized steam starvation will occur, severely degrading performance. Therefore, the outlet stream will include both steam and hydrogen. Residual steam can be removed from the product by condensation. On the oxygen-evolution side of the O-SOEC, air is often used as a sweep gas, so the oxygen partial pressure is only about 21% of the operating pressure. In addition, the electrolysis system can operate at elevated pressure.

To account for the range of gas compositions and pressures that occur in a real system, the open-cell (or Nernst) potential, V_N , can be obtained from the Nernst equation, which can be written in terms of the cell temperature, T , and pressure, P , universal gas constant, R_u , gas mole fractions, γ_i , for of H₂O, H₂, and O₂, and standard pressure, of P_{std} .

$$V_N = V^o - \frac{R_u T}{iF} \ln \left[\left(\frac{\gamma_{H_2O}}{\gamma_{H_2} \gamma_{O_2}^{1/2}} \right) \left(\frac{P}{P_{std}} \right)^{-1/2} \right]$$

The performance parameter that quantifies the ohmic losses associated with the operation of solid-oxide electrolysis cells is the area-specific resistance (ASR), expressed as:

$$ASR = \frac{E - E_{ref}}{i}, \text{ where } i = \frac{I}{A_{cell}},$$

where E is the applied potential and E_{ref} is the reference or open cell potential, I is the current, A_{cell} is the cell surface area, and i is the current density (A cm⁻²).

Operation of an electrolysis stack is fundamentally different than operation in the fuel-cell mode. From the standpoint of heat transfer, operation in the fuel-cell mode typically necessitates the use of significant excess air flow in order to prevent overheating of the stack. The potential for overheating arises from two sources: (1) the exothermic nature of the hydrogen oxidation reaction, and (2) ohmic heating associated with the electrolyte ionic resistance and other loss mechanisms. Conversely, in the electrolysis mode, the steam reduction reaction is endothermic. Therefore, depending on the operating voltage, the net heat generation in the stack may be negative, zero, or positive.

The ohmic heat flux, q_{Ohm}'' (W cm⁻²), is given by:

$$q_{Ohm}'' = i^2 ASR = i(V_{op} - V_N)$$

where V_{op} is the actual operating voltage of the cell.

The reaction heat flux can be expressed in terms of the Gibbs Energy of Reaction, ΔG_e (J mol^{-1}), and Enthalpy of Reaction, ΔH_R (J mol^{-1}):

$$q_R = i(\Delta G_e - \Delta H_R)$$

The operating point where the ohmic heating equals the endothermic heat of reaction is referred to as the thermal neutral point. This phenomenon is illustrated in Fig. 4 which shows the respective internal heat sink/source fluxes in a planar solid-oxide stack associated with the electrochemical reaction and the ohmic heating.

Operation at or near the thermal-neutral voltage simplifies thermal management of the stack since no significant excess gas flow is required and component thermal stresses are minimized. In fact, in the electrolysis mode, since oxygen is being produced, there is also no theoretical need for air flow to support the reaction at all. In a large-scale electrolysis plant, the pure oxygen produced by the process could be saved as a valuable commodity. However, there are several good reasons to consider the use of a sweep gas on the oxygen side. Materials considerations aside, a sweep gas reduces the average mole fraction of oxygen, γ_{O_2} , thereby reducing the open-cell and operating potentials, resulting in higher electrolysis efficiencies.

As shown presently, operation below the thermal neutral voltage requires heat addition to maintain an isothermal state in the cell. This heat may be added by an external heat source, such as a nuclear reactor.

Synergy with nuclear energy

Thermodynamic synergy between nuclear energy and high temperature steam electrolysis can lead to efficiency gains. The combination of a high-efficiency power cycle and the direct utilization of nuclear process heat can result in overall thermal-to-hydrogen conversion efficiencies of 50% or higher. Electricity conversion efficiencies of over 100% are easily attainable when steam and additional thermal energy are supplied from the nuclear process.

To illustrate the concept of increasing efficiency with direct thermal deposition to an SOEC, O'Brien (2008) plotted the heat flux necessary to maintain an isothermal condition in a representative cell as a function of operating voltage and area specific resistance (ASR, in Ohm cm^2) (Fig. 5). As the operating voltage is raised above the thermal neutral voltage (V_N), heat must be extracted. This can occur naturally as cell degradation increases ASR over time, necessitating increased operating voltage to maintain a constant current and hence a constant hydrogen production rate. Conversely, when the operating voltage is dropped below the thermal neutral point, then heat input is required to maintain the operating temperature. This heat can be conveniently provided by superheating the steam input into the cell, as well as a sweep gas for the oxygen electrode. Although there are many factors to take into consideration, including steam utilization, increased corrosion, and the kinetics of cell operation, cell operating parameters can be optimized through integrated materials selection, stack design, and integrated systems engineering.

The plot in Fig. 6 compares the operating voltages of the leading methods for hydrogen electrolysis by water at operating current around 1.5 A/cm (compare Graves et al., 2011). The research goal for advanced alkaline electrolysis (AE) and proton-exchange

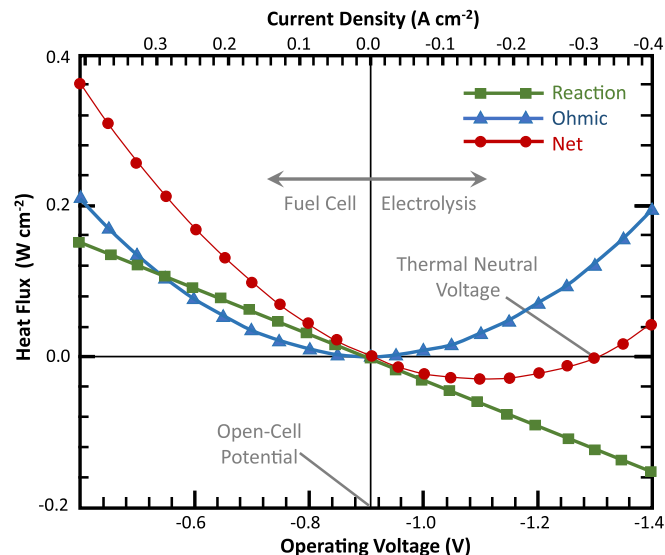


Fig. 4 Thermal contributions in electrolysis and fuel cell modes of operation. Data Source: O'Brien JE (2008) Thermodynamic considerations for thermal water splitting processes and high temperature electrolysis. *Proceedings of the 2008 International Mechanical Engineering Congress and Exposition*, IMECE2008—6880. <https://doi.org/10.1115/IMECE2008-68880>; calculated for ASR of 1.25 Ohm cm^2 and cell temperature of 1200 K, Y_{H_2} of 0.1 and 0.95 at cell inlet and outlet respectively.

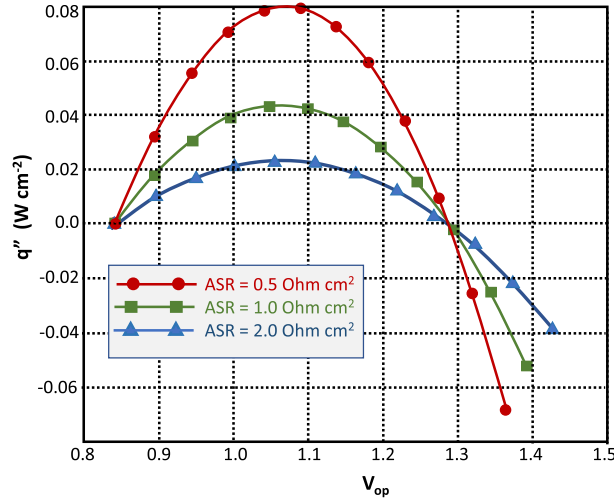


Fig. 5 Heat flux required for isothermal operation of SOEC cell. Source: O'Brien JE (2008) Thermodynamic considerations for thermal water splitting processes and high temperature electrolysis. *Proceedings of the 2008 International Mechanical Engineering Congress and Exposition, IMECE2008—6880*.

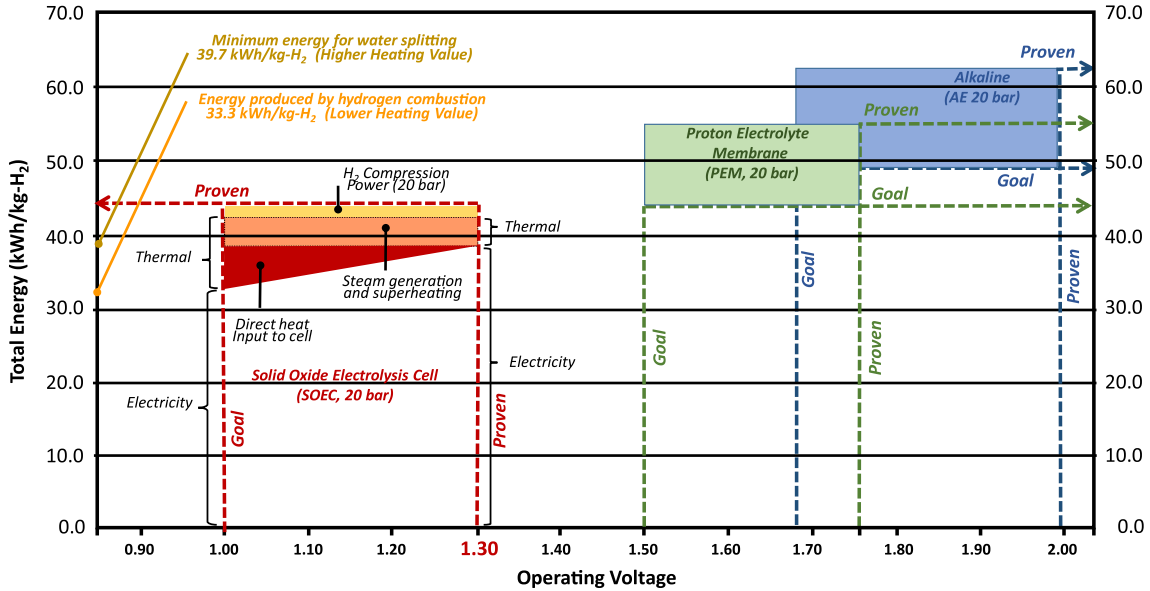


Fig. 6 Comparison of electrolysis cell operating voltages to produce one kilogram of hydrogen at 20 bar pressure [SOEC indicates breakdown of kW h^{-1} thermal vs. kW h^{-1} electrical].

membrane (PEM) is to decrease the operating voltage with advanced membranes for efficient hydrogen separation and reduced thermal energy dissipation. The energy for AE and PEM is currently 100% electrical power.

PEM and AE are capable of operating at higher pressure, while an SOEC stack assembly would either require a pressure containment vessel or post process gas compression. The power for hydrogen multi-stage compression is added to the SOEC electrolysis production featured in Fig. 6 to establish parity with the AE and PEM technologies that can already conveniently split pressurized water.

For high temperature steam electrolysis (HTSE) the energy required is split between thermal and electrical power, where thermal power can be provided in four stages, as shown in Fig. 7; (1) steam generation, (2) steam superheating (which can be provided by recuperation from the hot effluent H_2 and O_2 product streams), (3) feed gas temperature boosting and (4) direct heat deposition in the cell to maintain an isothermal operating condition when operating below the thermal neutral point. Waste heat (5) can be recovered for boiler feed water heating. Direct cell or stack heating complicates the stack design; hence, the likely best engineering solution is to raise the inlet gas temperatures with the nuclear reactor source. In case the temperature of the nuclear reactor heat transport fluid is less than the HTSE cell operative temperature, then electrical power can be used (6) to provide the topping heat needed. When an exogenous source of heat is unavailable, electrical heating can be used to generate the steam needed for HTSE; however, the thermodynamic efficiency is greatly diminished.

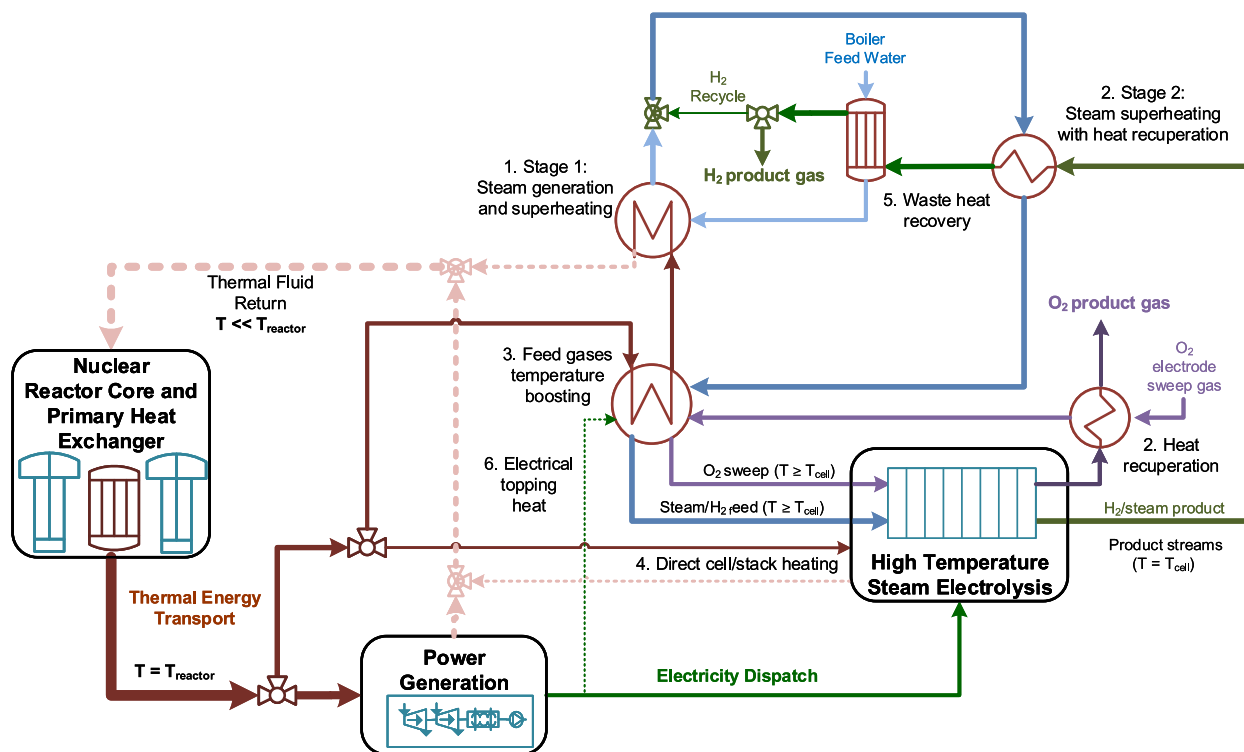


Fig. 7 Coupling of nuclear energy with high temperature steam electrolysis.

This comparison confirms the advantages of higher thermal dynamic efficiencies of high temperature steam electrolysis; ranging from 25–30% greater energy efficiency overall. In order to achieve higher SOEC efficiencies, thermal energy that is delivered above the operating temperature of the cell must be provided. The Very High Temperature Gas-Cooled reactor is capable of delivering heat in the range of 750 °C (current design) to 950 °C (future design). With the advent of proton-conducting SOEC (P-SOEC), the operating temperature will be reduced to ≤ 600 °C. At this operating temperature, the heat delivered from the emerging class of molten salt and liquid-metal cooled reactors can be coupled with P-SOEC to realize the benefits of water splitting with a primary heat source.

Economics of high temperature steam electrolysis

In recent years, Idaho National Laboratory has led efforts to evaluate the costs of low temperature and high temperature steam electrolysis with the express interest of coupling nuclear power plants to hydrogen production. The traditional method of reforming natural gas to produce hydrogen (i.e., steam/methane reforming; or SMR) currently enjoys the benefit of historically low prices due to the surplus of natural gas production in the United States, Russia, and locations in Africa and the Middle East. Similarly, the cost of producing H_2 by electrolysis is a strong function of the cost of electricity, which has dropped with the advent of low-cost solar panels and wind towers along with a buildup of new gas turbine power plants. These factors have motivated nuclear plant owners and the United States Department of Energy (U.S. DOE, 2020a, b), cooperating together, to consider new markets for operating nuclear plants, including especially water-splitting electrolysis options (Frick et al., 2019; Boardman et al., 2019a, b).

The economic proforma presented herein is for a nominally 900 MW electrical power nuclear plant coupled to a greenfield hydrogen plant that is separated 1 km from the nuclear plant. The levelized cost of hydrogen production (LCOH) for a discounted cash analysis was computed when invoking the assumptions listed in Table 1. Wholesale prices of electricity ranging from 25 to 50 \$ $MW h^{-1}$ were evaluated. This brackets the cost of energy production by existing Generation II light-water reactors in the United States, Europe, and elsewhere. Steam purchased from the nuclear facility (also referred to as “nuclear process heat”) was priced at the cost of electricity times the 32.3% thermal efficiency expected for the power conversion system at a typical single-unit nuclear power plant. It effectively equates the cost of heat to the cost of electricity that could otherwise be produced and sold using this heat.

Direct capital costs, as bare-module costs not-including the cost of stacks, for the HTSE process model were estimated using the preliminary plant design developed by Dominion Engineering (Krull et al., 2013). For some equipment items, the Aspen Process Economic Analyzer (Aspentech, 2019) was used to estimate the capital costs. Finally, it is necessary to estimate the cost of stack assemblies that are expendable and must be replaced when the cell degradation reaches a point of inefficient operations. Based on early indications of stack testing using a 25 kWe test stand capability (O’Brien et al., 2020), cell degradation rates of less than

Table 1 Summary of financial parameter used for a discounted cash flow calculation of LCOH.

Description	Value	Comments
Oxygen selling price	0.03 \$ kg ⁻¹	HSTE production of pure oxygen
Nominal IRR	12%	Internal Rate of Return on equity investment
Electricity price (\$/MWh-e)	20, 25, 30, 40, 50	
Thermal-energy (steam) cost (\$/MWh-t)	Varies	Multiple of an electricity price and the thermal-to-electrical conversion efficiency of 32.3%
Capacity market payment (\$/MWh-day)	132.4	
Debt to equity ratio	80% debt, 20% equity	80%/20% debt-to-equity ratio assumes strong government support for clean hydrogen production will apply to the first several electrolysis plants producing clean hydrogen
Debt interest rate	5%	Debt is backed with governments bonds
Debt period	20 years	
Annual inflation rate	1.9%	Based on the average inflation for the past 20 years
Overall tax rate	24.8%	20% federal; 6% state (current average rates applicable in the United States)
Capital depreciation schedule	Standard Modified accelerated cost recovery system (MACRS) depreciation method	Depreciation method, with a property class of 15 years
Decommissioning	10% of total depreciable capital	
Salvage value	10% of TCI	Total Capital Investment
Working capital	10% of the annual change in total operating costs	
Plant construction period	1 year; HTSE	HTSE plants are based on modular units that are factory assembled, reducing the time needed to build out the plant
Start-up year	2025	
Start-up time	1 year	Operating costs and revenues during start-up are 75% and 50% of the total values, respectively
Plant life and analysis period	20 years	Excluding construction time
HTSE Operating Capacity Factor	0.92	Assume nuclear plant and hydrogen plants will undergo coordinated outages/maintenance

From Boardman RD, Kim JS, Hancock S, Hu H, Frick K, Wendt D, Rabiti C, Bragg-Sitton SM, Elgowainy A, Weber R, and Holladay J (2019) *Evaluation of Non-Electric Market Options for a Light Water Reactor in the Midwest*. INL/EXT 19-55090.

0.5% per 1000 h of operations can be expected. For the current cost calculations, it was assumed that cells would be replaced when performance drops to around 80% of initial operating output. Hydrogen production rates and associated steam utilization factors were taken from estimates identified by O'Brien (2008).

A critical parameter for the current cost estimation is SOEC stack manufacturing costs. Currently, no manufacturing plants have been constructed; therefore, the cost of stacks at full commercial capacity were estimated from SOFC stack manufacturing experiences and technology providers. The cost of stacks is currently being debated. For the HTE plant, the baseline SOEC stack cost considered was \$50 kW⁻¹ (direct current [DC] power input to SOEC). This stack cost is based on input from Krull et al. (2013) and is assumed possible when high-volume manufacturing is achieved; namely for a scale of at least 500 MW of annual electrolysis stack volume (Boardman et al., 2019b). Notwithstanding, the cost of hydrogen production is not strongly correlated to stack manufacturing costs; rather it is more important the stacks are durable and remain effective for at least approximately 40,000 h of operation. Here stack costs of \$75 kW⁻¹ (+50% baseline) and \$35 kW⁻¹ (−30% baseline) were considered to reflect an expected spread in the cost of SOEC.

The SOEC plant matching the 900 MW nuclear plant is 578 t d⁻¹ (or 534 t d⁻¹ based on a 92.4% OCF). The divided power duties are 846 MW-electrical and 192 MW-thermal, respectively. Such a world-class scale would provide H₂ to multiple industries and is defined as a “large-scale” (or high-volume) production case in this analysis. Preliminary safety risks pertinent to locating a large hydrogen plant with appurtenant hydrogen storage within 1 km of a nuclear plant are being considered (Vedros et al., 2020).

Fig. 8 plots the levelized cost of hydrogen (LCOH) versus the cost of HTSE stack costs. The cost of producing hydrogen with an equivalent scale steam methane reforming (StMR) plant is also given for three projected natural gas price scenarios corresponding to industrial natural gas costs in the United States in 2019 (U.S. Energy Information Agency, 2019). The sale of oxygen results in a 0.25 \$ kg⁻¹ reduction in the minimum manufacturing price of hydrogen. Credits for the benefits of avoiding carbon dioxide emissions and other greenhouse gases versus hydrogen production by steam methane reforming can be significant. The evaluation by Idaho National Laboratory shows that each kilogram of hydrogen produced by electrolysis supported with nuclear energy merits 0.01 \$ kg⁻¹ for a CO₂ tax of 1 \$ t⁻¹ CO₂. Therefore, a tax of 100 \$ t⁻¹ CO₂ would increase the value of nuclear-sources hydrogen by 1 \$ kg⁻¹ H₂. This provides a compelling argument for hydrogen produced with clean power.

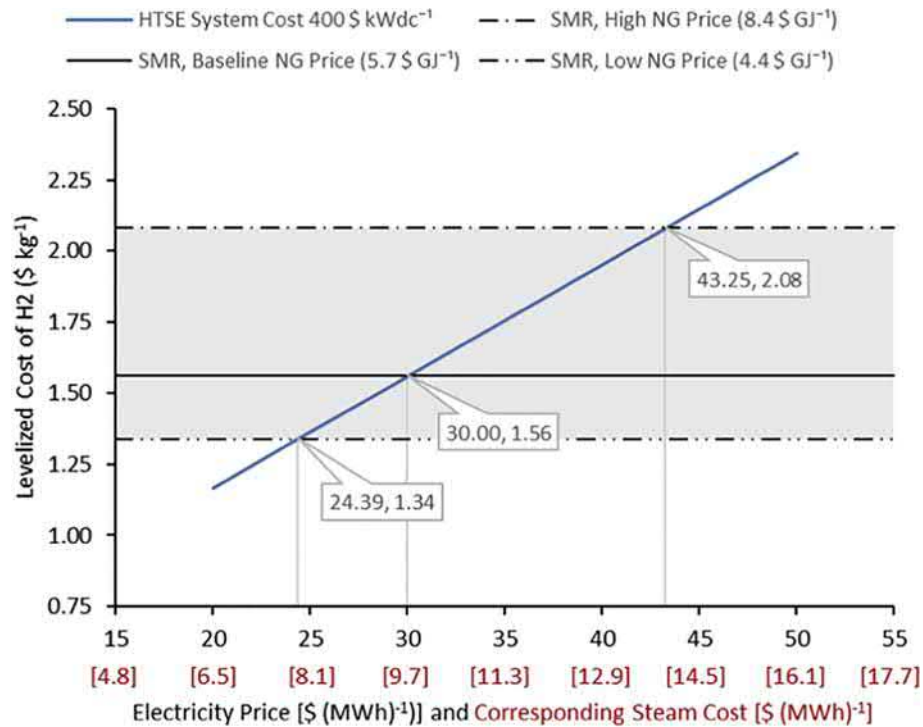


Fig. 8 Calculated cost of hydrogen production with high temperature steam electrolysis and steam methane reforming. (Note: the cost of HTSE hydrogen was reduced $0.25 \text{ \$ kg}^{-1}$ with the sale of oxygen.) Source: Boardman RD, Kim JS, Hancock S, Hu H, Frick K, Wendt D, Rabiti C, Bragg-Sittion SM, Elgowainy A, Weber R, and Holladay J (2019a) *Evaluation of Non-Electric Market Options for a Light Water Reactor in the Midwest*. INL/EXT 1955090.

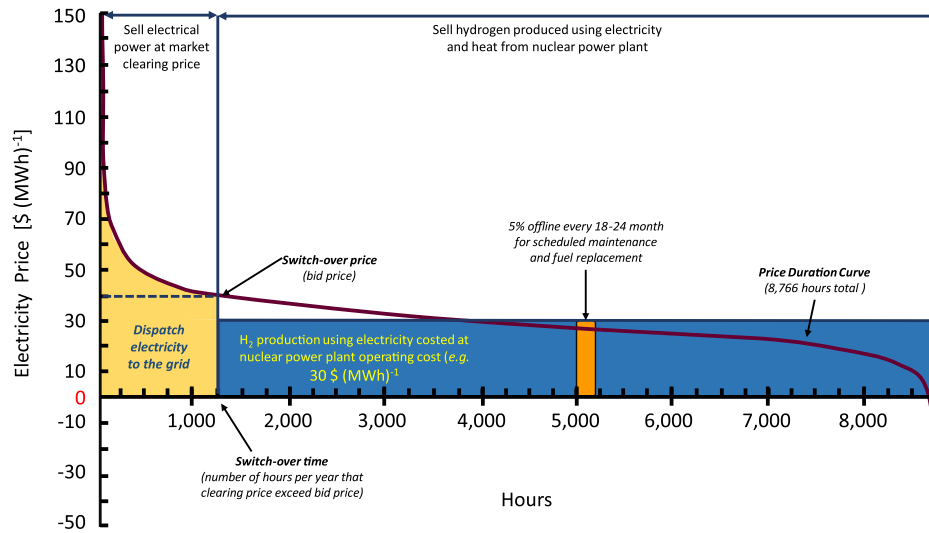


Fig. 9 Hybrid nuclear power/hydrogen plant concept for the price-dependent electricity market. Source: Concept explained by Boardman RD, Wendt D, Bates S, Elgowainy A, Hawkins T, Juden P, Meeks N, Otgonbaatar U (2019b) *Region-Specific Merchant Hydrogen Market Assessment and Techno-Economic Assessment of Electrolytic Hydrogen Generation*, INL/EXT-19-55247.

Dynamic operations of HTSE

Hybrid plant operations

For purposes of terminology, a hybrid plant is defined as a plant that dispatches one or more energy outputs to produce at least two unique products in a manner that optimizes at least one objective such as (1) revenue generation, (2) pollutant emissions reduction, or (3) grid reliability and/or resiliency and/or stability. The marginal price of electricity (P_{marginal}) rises when the demand

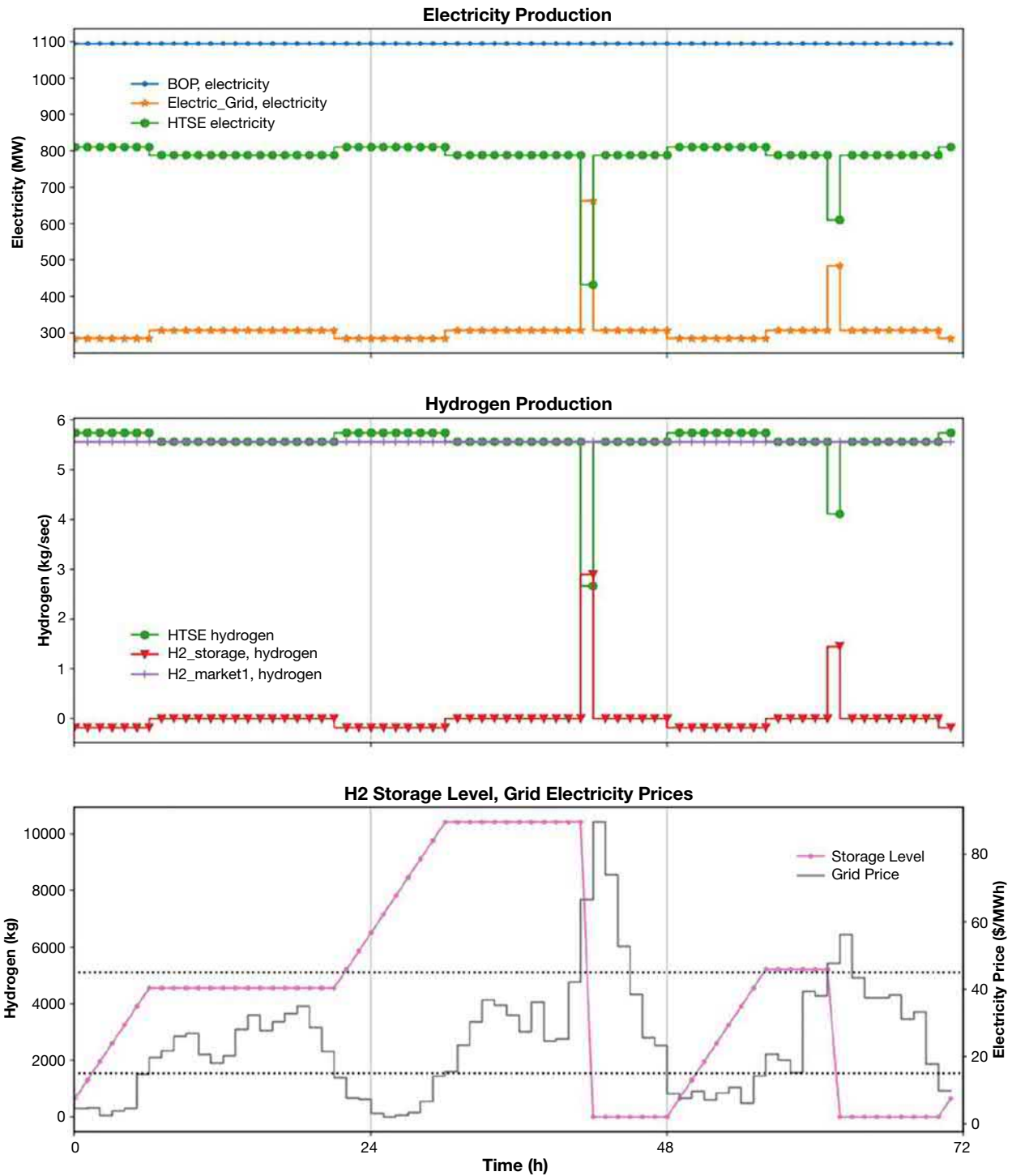


Fig. 10 Example of hybrid plant operation dynamics. Source: Frick K, Talbot P, Wendt D, Boardman RD, Rabiti C, Bragg-Sitton SM, Levie D, Frew D, Ruth M, Elgowainy A, and Hawkins T (2019) *Evaluation of Hydrogen Production Feasibility for a Light Water Reactor*. INL/EXT-19-55395.

exceeds supply. This is projected to happen with increasing frequency as variable renewable supply continues to rise. This presents a unique market case for nuclear hydrogen production which takes advantage of market conditions to optimize revenue generation. Fig. 9 illustrates how electricity produced at the marginal electricity selling price could be sold to the grid or used to generate hydrogen. In this case, the systems may be viewed as a “price taker” so far as it will schedule ahead for the power market anytime the marginal electricity price exceeds some price point (referred to herein as the Switch-over Price). When the marginal price of electricity is high, the electrolysis plant is rapidly turned down while dispatching electricity to the grid. Conversely, when the price of

electricity is low, then the plant will ramp up hydrogen production. Naturally, in order to supply hydrogen to a dedicated industrial user, hydrogen storage will be necessary.

Hybrid production of hydrogen leads to an opportunity to deploy a reversible SOFC/SOEC system in which the hydrogen produced below the cutoff-price is stored to produce electricity when the price of electricity rises above the cutoff price.

To analyze the economic viability of hybrid operations, particularly given the uncertainty surrounding load demand, electricity prices, and the availability of variable power generation resources, a stochastic technoeconomic analysis package (Holistic Energy Optimization Network [HERON]) has been developed for use with a risk analysis framework- Risk Analysis Virtual Environment (RAVEN). An example of a predicted 72 h dispatch of power from a nuclear power plant is reproduced here in Fig. 9 for a deregulated market in the United States where the objective is maximum revenue for the plant (Frick et al., 2019; Talbot et al., 2020). In this case, two resources are tracked-electricity and hydrogen- for five components: (1) an electricity source (balance of plant; BOP), (2) a hydrogen production facility (HTSE) that consumes electricity to make hydrogen, (3) an electricity market (Electric Grid) that consumes electricity, (4) a hydrogen storage unit (H2_storage), and (5) a constant hydrogen consumer (H2_market1). The graphs show the operation of each unit in terms of electricity (top), hydrogen (middle), and storage as well as price of electricity (bottom). When the marginal price of electricity is low hydrogen is produced and stored. When the price of electricity near the cutoff price, hydrogen is taken from storage to supply the hydrogen consumer so that all electricity produced can be sold at the grid. The performance over the three-day period illustrates how a general dispatch optimization can optimize the profit for a given hybrid nuclear power plant (Fig. 10).

Physics-based HTSE modeling

Dynamic optimization of a hybrid nuclear/hydrogen plant requires a realistic understanding of the flexibility for modulating the HTSE plant as much as dispatching the thermal and electrical power from the nuclear plant. A physics-based model, consistent with the heat management process flows shown in Fig. 7 was developed by Kim et al. (2018). This model was incorporated into HERON to constrain solutions to the practical operating ramp rates and limits of the plant. The entire system was dynamically modeled with the Modelica modeling language using a commercially available Modelica-based modeling and simulation environment (Fritzon, 2014; Dassault Systèmes, 2017).

The case study and control schemes developed by Kim et al. (2018), showed a coupled nuclear/HTSE plant can respond quickly (on the order of hundredths of a second) with fast ramping capability, settle adequately fast and maintain the required change for a sufficient duration in response to rapid, large load changes of grid generation demands. The dynamic response characteristics of the coupled nuclear and HTSE plants are able to provide various ancillary grid services to manage the variability and uncertainty introduced with high penetration of renewables while supporting large-scale, carbon-free hydrogen production.

Conclusion

High temperature steam electrolysis (HTSE) is a promising method for high-efficiency hydrogen production. HTSE cells currently being developed operate in the range of 750 to 900 °C, but can also be coupled with lower temperature heating sources with minimal loss of efficiency through heat recovery. Hybrid plant operation of a nuclear/HTSE plant can support grid capacity demands by dispatching electricity to the grid when demand exceeds variable renewable generation sources. Hydrogen is produced using heat and electricity that is not immediately needed to meet grid demand. This hydrogen can be sent to an industrial user, or burned in a reversible solid oxide fuel cell (SOFC) and solid oxide electrolysis cell (SOEC), i.e., SOFC/SOEC plant to provide additional electricity back to the grid to meet peak demands.

Efforts to develop robust materials and efficient stacks and systems are achieving the performance standards and durations needed for commercial operations. When high volume manufacturing rates are realized, the capital costs will reach levels that can make electrolysis systems competitive with the incumbent hydrogen carbon reforming processes.

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Synthetic Hydrocarbon Fuels From H₂ and Captured CO₂

Lesley Snowden-Swan^a, Shuyun Li^a, Jeromy Jenks^a, Steven Phillips^a, Jalal Askander^a, Jamie Holladay^a, L. Todd Knighton^b, and Daniel S. Wendt^b, ^a Pacific Northwest National Laboratory, Richland, WA, United States; and ^b Idaho National Laboratory, Idaho Falls, ID, United States

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Abbreviations

ACCE	Aspen capital cost estimator
ASR	Area-specific resistance
BETO	Bioenergy Technology Office
BPD	Barrels per day
CAPEX	Capital expenses
CCC	Cryogenic carbon capture
DME	Dimethyl ether
DOE	Department of energy
EPA	Environmental Protection Agency
ER	Electrochemical reduction
FA	Formic acid
FCI	Fixed capital investment
FT	Fischer-Tropsch
GHG	Greenhouse gas
GREET	Greenhouse Gases, Regulated Emissions, and Energy Use in Transportation model
INL	Idaho National Laboratory
LCFS	Low-carbon fuel standard
LWR	Light water reactor
LWRS	Light Water Reactor Sustainability program
MeOH	Methanol
MFSP	Minimum fuel selling price
MTG	Methanol can also be converted to hydrocarbons via the methanol-to-gasoline
MTO	Methanol-to-olefins
MTP	Methanol to propylene
NG	Natural gas
NGCC	Natural gas combined cycle
NPP	Nuclear power plants
OPEX	Operating expenses
PNNL	Pacific Northwest National Laboratory

REC Renewable energy credits
 RFS Renewable fuel standard
 RIN Renewable identification number
 RSTOIC Stoichiometric reactor in AspenTech software
 RWGS Reverse water-gas-shift
 SOEC Solid oxide electrolyzer cell
 SOFC Solid oxide fuel cell
 TEA Techno-economic analysis
 TRL Technology readiness level
 UOP Honeywell UOP, formally known as Universal Oil Products

Introduction

Nuclear energy is increasingly being recognized as a valuable low-carbon, low-emissions energy source that can help achieve clean energy targets being set by states, commissions, and utilities in the United States. Currently, nuclear power provides about one-fifth of the country's electricity. Nuclear power plants (NPPs) further provide the grid with all-weather season-long baseload capacity that is important to grid reliability and resiliency. Light water reactor (LWR) NPPs in the United States, like other sources of electricity generation, are facing increasing cost pressure on the electricity grid due to historically low-priced natural gas (NG) and the rapid expansion of solar and wind energy. Solar and wind energy provide spikes on the grid during periods of high production, but there will be a continued opportunity for baseload generators, such as NPPs, to provide electricity to the grid when solar and wind energy installations are producing little output. During times of grid overgeneration the NPP energy can be diverted to create other value-added chemicals and fuels. Therefore, the U.S. Department of Energy (DOE) Light Water Reactor Sustainability (LWRS) Program is addressing flexible plant operations that can diversify the revenue of NPPs.

Previous reports have evaluated opportunities to couple LWRs with hydrogen production (Frick et al., 2019; Hu et al., 2019; Knighton, 2020). This article analyzes several synthetic fuel (synfuel) production pathways that could be coupled with LWRs to provide alternative options to utility companies for using nuclear energy to create value added products during periods of overgeneration of electricity to the grid. A conceptual integrated plant would consist of a hybrid LWR delivering power and heat to a synthetic fuel and/or chemical facility. The synfuels plant would employ co-electrolysis to convert CO₂ and water into syngas (synthesis gas, a mixture of H₂ and CO). The CO₂ ideally would come from a source that is in close proximity to the LWR and one in which the CO₂ is currently being released to the atmosphere, to take advantage of possible clean energy credits. Sources such as an ethanol plant release CO₂ in high concentration which makes the CO₂ separation and utilization more cost effective. In this article an ethanol plant located between 50 and 100 miles from the LWR is assumed to be the CO₂ source. CO₂ sources in close proximity to LWRs, such as a natural gas combined cycle (NGCC) plant, and state-of-the-art carbon capture technology are also possible options. The syngas would then be converted to synthetic fuels via the most economical processes. Choices for the conversion of syngas to synfuels include Fischer-Tropsch (FT), methanol-to-fuels, and ethanol-to-fuels, among others.

Co-electrolysis via a solid oxide electrolyzing cell (SOEC) is assumed in the techno-economic analysis (TEA) detailed herein to take advantage of its high efficiency as compared with a standard polymer electrolyte membrane (PEM) electrolyzer. Although higher value synchems offer compelling investment potential, the focus of the TEA is on production of synfuel blendstocks compatible with existing liquid transportation fuels. The methanol (MeOH) route was selected because methanol is a common and flexible intermediate that can either serve as a base chemical or be converted to hydrocarbon fuel. The plant scale assumed is commensurate with the typical scale used for a cellulosic biorefinery (2000 dry t/day biomass feed) to provide a consistent comparison with renewable fuel from biomass. This is equivalent to a syngas flow of 141 t/day H₂ and 973 t/day CO. Detailed process models and TEA were developed for the CO₂ to synfuel pathway incorporating the use of the LWR heat and electricity as an energy source. Equipment CAPEX and OPEX are detailed in the economic modeling, including reactor and other unit operation costs, and feedstock and product valuations. Sensitivity analysis around key process and economic assumptions is also presented.

Routes for CO₂ to fuels or chemicals

There are many ways to make fuels or chemicals from CO₂. Fig. 1 provides a flowchart of some of the general pathways for producing CO₂-derived fuels integrated with an LWR. Steam and power from the LWR are provided for CO/H₂ (syngas) production in addition to any demand required by the chemical plant. Syngas can be produced from CO₂ via co-electrolysis or the reverse water-gas-shift (RWGS) reaction. Co-electrolysis is preferred over having separate electrolyzers for CO₂ and water as it has been found to be more efficient (Fu et al., 2010). Syngas is used to produce a wide range of fuels and chemicals, including but not limited to synthetic NG, dimethyl ether, methanol, ethanol, and hydrocarbon fuel blendstocks. Compounds such as methanol and ethanol can also be produced directly with co-electrolysis. However, it is thermodynamically unfavorable compared to making syngas, as the

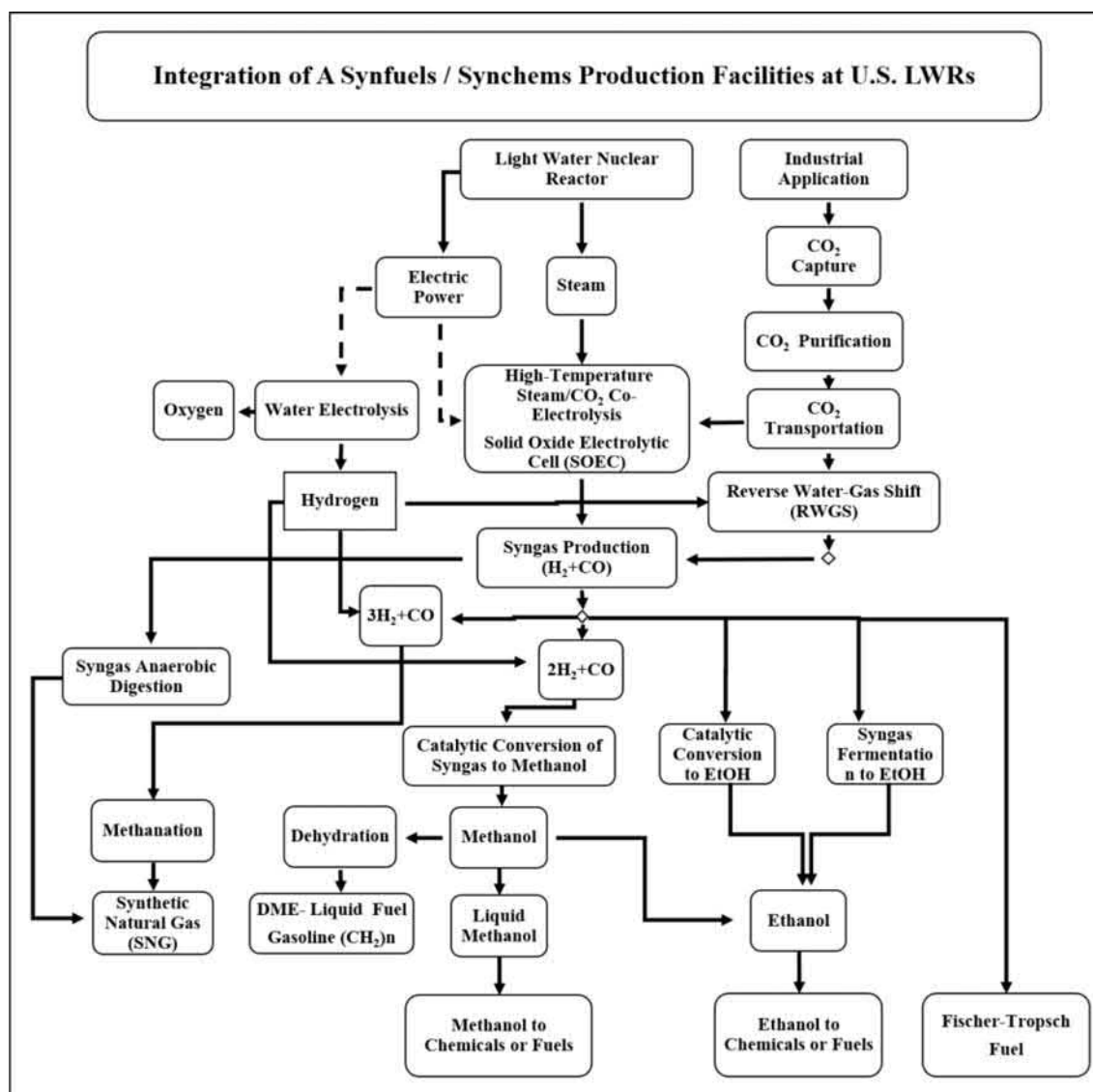


Fig. 1 Overview of various synfuel/synchem pathways integrated with an LWR.

number of electrons needing fixed is 6 for methanol and 12 for ethanol, versus 2 for CO. **Table 1** lists the numbers of electrons required for CO and other chemicals.

Fuels

Syngas can be converted to oxygenated fuels, such as ethanol or methanol, or converted further into hydrocarbon fuels. Ethanol can be produced from syngas through biological or thermochemical means, as shown in **Fig. 2**. Fermentation has been indicated by

Table 1 Number of electrons transferred for converting CO₂ to syngas and chemicals.

Compound	Electrons transferred
Syngas	2e ⁻
Formate	2e ⁻
Methanol	6e ⁻
Methane	8e ⁻
Ethanol	12e ⁻
Octanol	48e ⁻

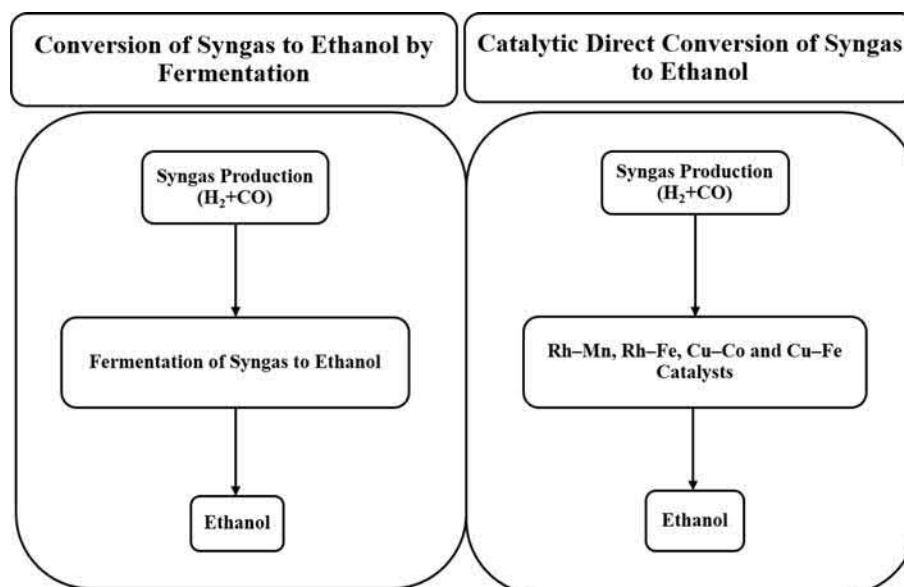


Fig. 2 Fermentation and catalysis of syngas to ethanol.

many researchers as an efficient and cost-effective method for conversion of syngas and several companies have now made the process commercial. For example, LanzaTech now commercially produces ethanol via syngas fermentation using its proprietary Clostridial biocatalyst. Direct thermo-catalytic conversion of syngas to ethanol is also possible using several different catalysts, as shown in Fig. 3.

Fig. 3 details three alternative pathways for syngas conversion to ethanol that are in earlier stages of development than direct conversion shown in Fig. 2. Fig. 3A shows a direct conversion of syngas to ethanol through dimethyl ether (DME) as a key intermediate, the catalyst $ZnAl_2O_4/H\text{-MOR } ZnAl_2O_4$ produces ethanol with a selectivity of 52% (Zhou et al., 2018). There are some

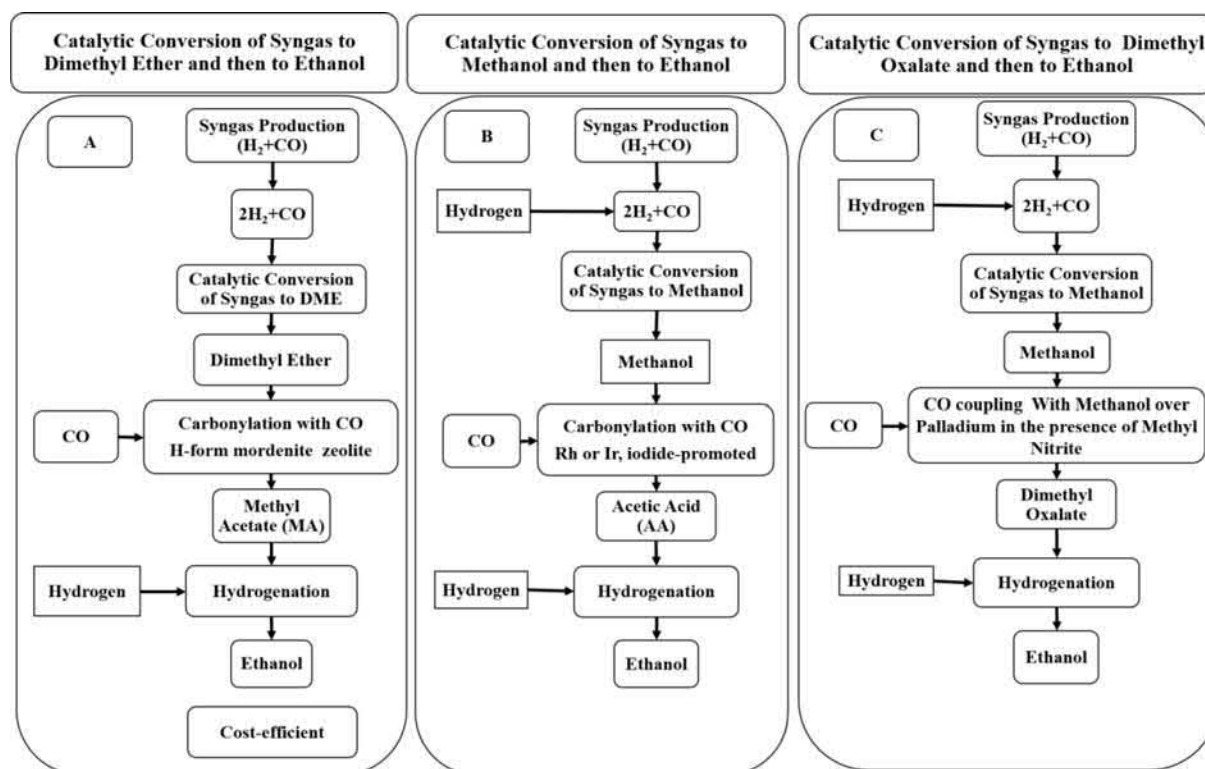


Fig. 3 Syngas conversion to ethanol via catalysis.

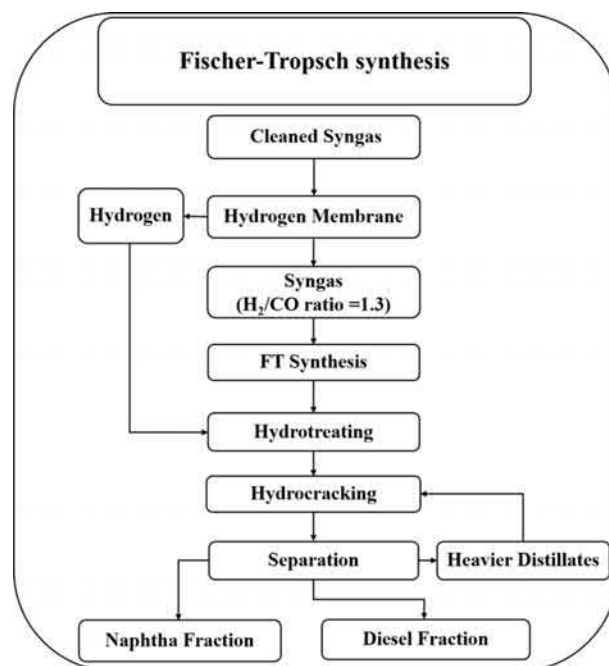


Fig. 4 Syngas to hydrocarbon fuel via Fischer-Tropsch synthesis.

other fuel alternatives, such as DME, which is a clean-burning, non-toxic, potentially renewable fuel that can be produced from methanol. The path of methanol-to-ethanol production is not necessarily the cheapest. Some other tracks with inexpensive catalysts may be worth evaluating, such as methyl acetate. The two-step conversion of methanol to ethanol via the methyl-acetate process is at the stage of pre-commercialization. Fig. 3B details industrial methanol carbonylation to produce acetic acid that is performed either over the Ir-based (Cativa process) or Rh-based (Monsanto process) catalysts (Lu et al., 2016). Fig. 3C shows the methanol conversion process through dimethyl oxalate. In this process ethylene glycol and ethanol can be produced, making it a versatile method for producing essential chemicals (Yue et al., 2014).

Methanol can also be converted to hydrocarbons via the methanol-to-gasoline (MTG) process or the MTO process, both of which were originally introduced by the Mobil Oil Corporation. MTG is carried out over a HZSM-5 catalyst with high selectivity and little side-products, producing a hydrocarbon mixture of narrow compositional range (Gogate, 2019). The MTO process was developed by essentially controlling process conditions to interrupt the MTG methanol to hydrocarbons reaction and has been commercialized by Honeywell UOP (UOP) and others.

Syngas can also be converted directly into hydrocarbon fuel using the established FT process, as shown in Fig. 4. The FT process is the oldest coal-to-liquids technology, invented in the 1920s and used by the Germans during World War II to provide needed liquid hydrocarbon fuels. Several FT-based commercial plants are operating today, including Sasol's Sasolburg coal-to-liquids plant (South Africa). Sasol started developing designs for a gas-to-liquids plant in Lake Charles, Louisiana, but the project was cancelled in 2017 due to the collapse of oil prices which decreased the differential between the NG feedstock and the value of the fuel products (Grisggs, 2017).

Chemicals

Many pathways exist for converting CO₂ to chemicals via thermochemical or electrochemical processes followed by catalysis. CO₂ reduction reactions can yield various valuable multi-carbon compounds including ethylene, acrylic acid, propylene, and C1 chemicals and polymers (Alper and Yuksel Orhan, 2017). As of 2017, 130 Mt. of CO₂ is used annually to generate urea, salicylic acid, polycarbonates, and cyclic carbonates (Koempel and Liebner, 2007). Fig. 5 shows two pathways for converting methanol to propylene or ethylene. The MTO reaction detailed in Fig. 5B is one of the most critical reactions in C1 chemistry, which provides a chance for producing basic petrochemicals such as ethylene and propylene (Eng et al., 1998). The methanol-to-propylene (MTP) process shown in Fig. 5A produces propylene from methanol (Koempel and Liebner, 2007). Significant differences exist between the MTO and MTP processes in regard to reactor design and productivity (Barger, 2002). MTO uses a fluidized-bed reactor, where heat can be removed quickly, and catalysts can be easily regenerated. MTP uses a fixed bed reactor where heat removal is problematic but overcome by using multiple catalyst beds. Fixed bed reactors are easier to scale up compared to fluidized beds, are cheaper and have better product selectivity. However, MTO can use crude methanol whereas MTP must use pure methanol, thus adding to the overall cost for MTP (Jasper and El-Halwagi, 2015). As is evident, the chemical derivatives of ethylene and propylene are numerous and have a variety of industrial applications. Polymerization processes are not detailed here.

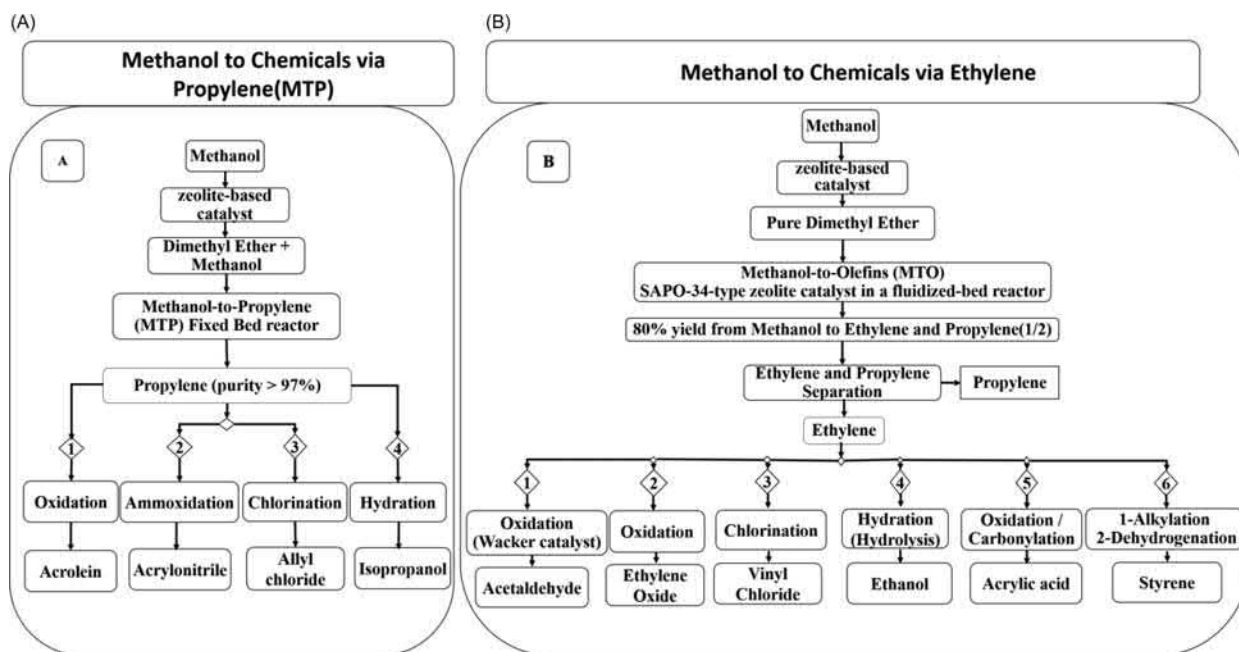


Fig. 5 Methanol conversion to chemicals via methanol-to-propylene (A) and methanol-to-olefins (B).

Ethanol is an essential source of many chemical compounds, including para-xylene. Ethylene is the intermediate compound in the process of producing para-xylene, either directly (Fig. 5B), or indirectly (Fig. 6A). In a direct path, ethylene is used in more than one step of a complex process to produce para-xylene (Zhang et al., 2015). In the indirect method, DMF is the primary compound for the production of para-xylene, and ethylene plays a role in forming double bonds (Lyons et al., 2012).

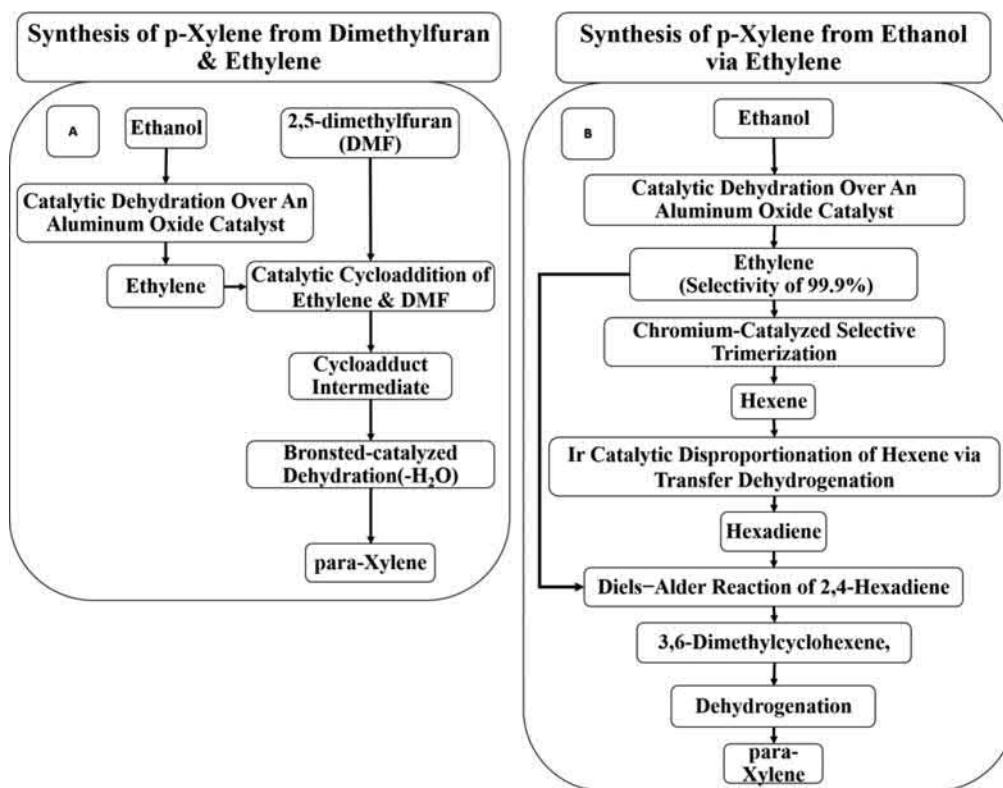


Fig. 6 Ethanol conversion to para-xylene.

As a final note on chemical conversion, formic acid (FA) is a critical commodity used in agricultural, pharmaceutical, food, textile, and other chemical markets. The global demand for FA is expected to be 820,000 metric tons in 2021 (Sesto, 2020). FA can be produced via electrochemical reduction (ER) or homogenous catalysis of CO₂ and H₂. Processing requirements range from 25 to 400 MJ/kg of FA produced (Rumayor et al., 2018). This would require less than one and up to three LWRs dedicated solely to the production of FA to meet global demand.

Selected hydrocarbon fuel route for TEA

As the focus of this study is synthetic transportation fuels, conversion pathways converting CO₂ to fungible hydrocarbon fuel were selected for detailed modeling and TEA. Production of syngas via co-electrolysis or RWGS result in the same overall chemical equation (Eq. 1); therefore, similar energy consumption. Co-electrolysis via SOEC was selected. As discussed, the possible range of syngas-to-fuel pathways is very extensive. From this extensive list, the possible range of technologies was down-selected to a methanol-based pathway, as methanol is a versatile chemical that can be further converted into fuels, and a wide range of chemicals and products.



The pathways selected for detailed analysis are shown in Fig. 7. The methanol pathway is based on established technology for methanol generation from syngas and production of olefins from methanol using the UOP's commercialized MTO process (Vora et al., 1998). Olefin oligomerization and hydrogenation technology are based on a PNNL-patented process (Lilga et al., 2016). The methanol pathway is considered to be at a high technology readiness level (TRL), as the syngas-to-methanol technology (using coal or NG) has been in use for decades and there are several industrial installations of the MTO process based on coal gasification, one being in China and one in Belgium (Gogate, 2019).

Techno-economic analysis methodology

The approach to developing conversion process techno-economics is similar to that employed in previous analyzes conducted for the DOE's Bioenergy Technologies Office (BETO) (Dutta et al., 2015; Jones et al., 2013; Jones and Zhu, 2009). Process flow diagrams and models are developed based on experimental research by PNNL, INL, and others, along with information from the literature and commercial vendors for mature and similar technologies. To assure consistency across all conversion pathways, BETO developed a set of economic assumptions that are used for bioenergy TEAs (DOE, 2016), which are also adapted for this work. An important aspect of these assumptions is that they reflect an "nth-plant" design, as described below.

Definition of Nth plant

A standard reference basis common to the conceptual design reports, known as the "nth" plant design, is used. These assumptions do not account for additional costs that would normally be incurred for a first-of-a-kind plant, including special financing, equipment redundancies, large contingencies, and longer startup times necessary for the first few plants. For nth-plant designs, it is assumed that the costs reflect a future time when the technology is mature, and several plants have already been built and are operating. The

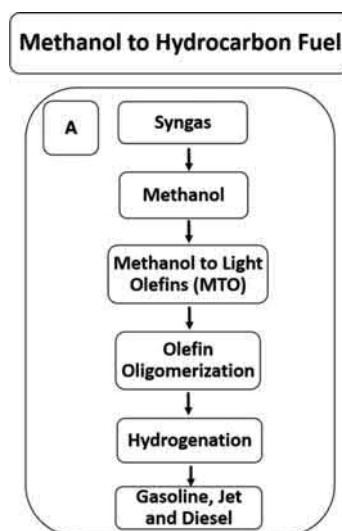


Fig. 7 Selected syngas intermediate chemicals and pathways for fuel production through methanol.

Table 2 Nth-plant assumptions.

Assumption description	Assumed value
Internal rate of return (IRR)	10%
Plant financing debt/equity	60%/40% of total capital investment (TCI)
Plant life	30 Years
Income tax rate	21%
Interest rate for debt financing	8.0% Annually
Term for debt financing	10 Years
Working capital cost	5.0% of fixed capital investment (excluding land)
Depreciation schedule	7-Years MACRS ^a schedule
Construction period	3 Years (8% 1st year, 60% 2nd year, 32% 3rd year)
Plant salvage value	No value
Startup time	6 Months
Revenue and costs during startup	Revenue = 50% of normal Variable costs = 75% of normal Fixed costs = 100% of normal
On-stream factor	90% (7920 Operating hours per year)

^aModified accelerated cost recovery system.

specific assumptions are shown in **Table 2**. Note that tax incentives and other credits that may be applicable (e.g., credits under the Renewable Fuel Standard or cellulosic biofuels bonus depreciation) but are excluded from the analysis to represent plant economics independent of any government subsidies.

General cost estimation basis

All costs in this report are on a 2019 constant dollar basis. This is the current reference year that BETO uses to facilitate comparison of various conversion technologies. Capital costs are estimated from a variety of resources. The heat and material balances generated by the simulation software (AspenTech ASPEN-Plus; CHEMCAD v.7) are used to size the major pieces of equipment. Aspen Capital Cost Estimator (ACCE), information from published literature and vendors quotes are used to cost individual pieces of equipment. The original cost reflects the year of the cost quote or estimate, and the scale of the equipment. All capital costs are adjusted to an annualized 2019 basis using the Chemical Engineering magazine's published indices:

$$\text{Cost in 2019 \$} = \text{equipment cost in quote year} \times \left(\frac{2019 \text{ index} = 541.7}{\text{quote cost year index}} \right) \quad (2)$$

The scale is adjusted to match the appropriate scaling term (heat exchanger area for example) by using the following expression:

$$\text{Scaled equipment cost} = \text{cost at original scale} \times \left(\frac{\text{scale up capacity}}{\text{original capacity}} \right)^n \quad (3)$$

where n is the scale factor, typically, 0.6–0.7.

After equipment is scaled and adjusted to the common cost year, factors are applied to calculate the total capital investment. Individual installation factors calculated by ACCE are multiplied by equipment costs, unless installed costs are already available from vendors. The total direct cost is the sum of all the installed equipment costs, plus the costs for buildings, additional piping, and site development. Indirect costs are estimated as 60% of the total installed costs. Factors for the calculation of these additional direct and indirect costs are listed in **Table 3**. The sum of the direct and indirect costs is the fixed capital investment (FCI). The total capital investment is the fixed capital plus working capital and land costs.

Operating costs are estimated by using the results from the ASPEN-Plus heat and material balances and applying raw material and utility prices (given in individual pathway TEAs in following sections). Labor requirements and rates for the modeled synfuels plant are consistent with past TEAs performed for biomass-gasification-based fuel plants and are listed in **Table 4**. Note that labor needs associated with running the front-end SOEC portion of the plant as compared to a gasifier may be lower; therefore, these costs are likely conservative.

With the capital and operating costs, the MFSP is determined using a discounted cash flow rate of return analysis. The MFSP is the plant gate selling price of the fuel product that makes the net present value of the project equal to zero given the financial factors assumed (see **Table 2**).

Co-electrolysis of CO₂ and water to syngas

SOECs offer a unique method for converting CO₂ and steam into syngas. Co-electrolysis is preferred over separately electrolyzing steam and CO₂ because of reduced cell resistance (area-specific resistance, or ASR) and lower conversion of CO to C ([Stoots, 2010](#)).

Table 3 Cost factors for direct and indirect costs.

<i>Direct costs</i>	
<i>Item</i>	<i>% of Total installed cost (TIC)</i>
Buildings	4.0%
Site development	10.0%
Additional piping	4.5%
Total direct costs (TDC)	18.5%
Indirect costs	
Item	% of TDC
Prorated expenses	10%
Home office and construction fees	20%
Field expenses	10%
Project contingency	10%
Startup and permits	10%
Total indirect costs	60%
Working capital	5% of FCI
Land	HTL: 6 acres @ \$15,000/acre Upgrading: 6% of total purchased equipment cost

Table 4 Labor costs for modeled synfuels plants.

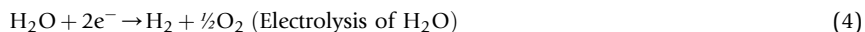
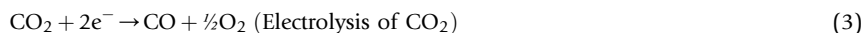
<i>Fixed operating costs</i>	<i>\$/Year</i>	<i>No. workers</i>	<i>Base year \$/h</i>	<i>\$/year Total in 2011\$</i>	<i>\$/year per Worker in 2019\$</i>	<i>\$/year Total in 2019\$</i>	<i>\$/h in 2019\$</i>
Plant manager	161,362	1	70.67	161,362	170,761	170,761	82.10
Plant engineer	76,839	1	33.65	76,839	81,315	81,315	39.09
Maintenance supervisor	62,569	1	27.40	62,569	66,214	66,214	31.83
Lab manager	61,471	1	26.92	61,471	65,052	65,052	31.27
Shift supervisor	52,690	5	23.08	263,450	55,759	278,795	26.81
Lab technician	43,908	3	19.23	131,724	46,466	139,397	22.34
Maintenance tech	43,908	16	19.23	702,528	46,466	743,449	22.34
Shift operators	43,908	27	23.08	1,185,516	46,466	1,254,570	22.34
Yard employees	30,736	12	13.46	368,832	32,526	390,316	15.64
Clerks and secretaries	39,517	3	17.31	118,551	41,819	125,456	20.11

Fig. 8 details the process inclusive of the SOEC stack, heat exchangers, compressor, pump, and separators. In this process CO₂ and water enter the stack in vapor phase. Power supplied to the stack yield syngas and oxygen. Oxygen flows from the cathode to the anode and is removed by a sweep gas, typically air. Excess CO₂ and steam are separated from the stack output and reintroduced to the stack inlet.

SOEC design

Co-electrolysis of CO₂ and H₂O have been studied for the better part of a decade as method of producing syngas using the same technology as has been used over several decades for solid oxide fuel cell (SOFC)s. Rather than producing power, SOECs consume electrical energy (and sometimes heat) to convert CO₂ and H₂O in syngas.

Solid oxide electrolysis makes use of the same material and technology as solid oxide fuel cells. A single cell in an SOEC stack is comprised of an anode, cathode, and solid electrolyte. The cathode is typically constructed of nickel and yttria stabilized zirconium and the anode is a mixture of lanthanum, strontium, and manganese oxide. When power is supplied to the SOEC, steam and CO₂ are converted to H₂, CO, and O₂. This occurs via three reactions, namely the reverse water-gas shift and the co-electrolysis reactions:



A model was initially developed in ASPEN-Plus V10 to represent product stream compositions, and heat and power requirements for an SOEC stack. Inlet mole and conversion fractions were specified to produce the required ratio of H₂/CO of 2 in the syngas for the downstream methanol synthesis step. The model was developed to match results from experimentally validated SOEC models from the literature (O'Brien et al., 2009; Redissi and Bouallou, 2013) for approximately the same syngas ratios.

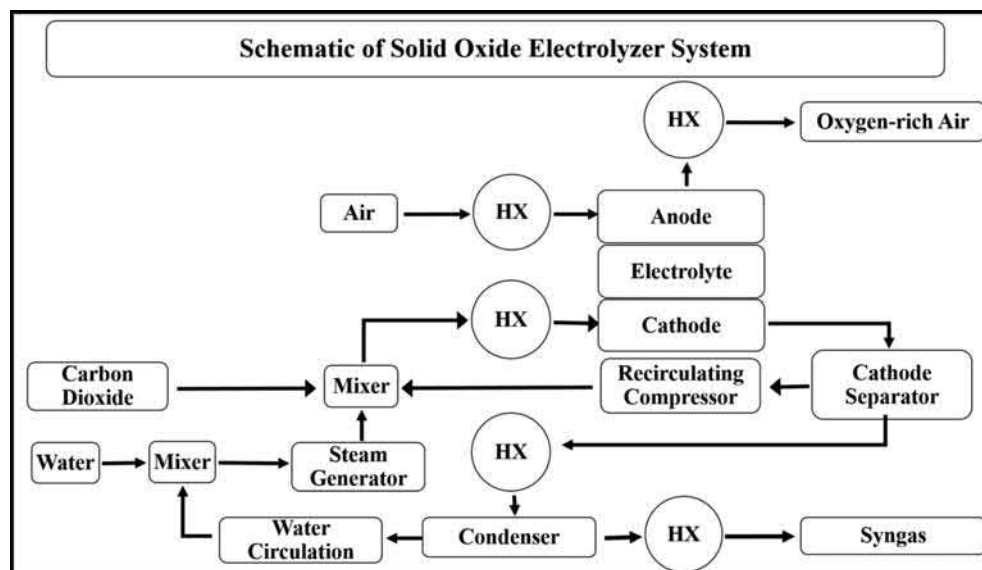


Fig. 8 Schematic of SOEC.

The model introduces CO₂/H₂O/H₂ at a molar ratio of 45:45:10 and a temperature of 300 °C to the stack. The mixture enters the stack whereupon RWGS occurs modeled as an equilibrium reactor labeled as LRWGS. A topping heater brings the mixture to the adiabatic stack operating temperature, in this case 800 °C. The stack is operated at a pressure of 0.1 Mpa, thermal neutral voltage of 1.34 V, and optimal current density and ASR of 0.4 A/cm² and 1.5 Ohm cm², respectively (O'Brien et al., 2009). Electrolysis occurs at thermo-neutral voltage in the stack modeled by a stoichiometric reactor (RSTOIC) using a fractional conversion from electrolysis reactions at 0.95 and 0.05 for H₂O and CO₂, respectively. RWGS again occurs at 800 °C. Although the Redissi model was validated experimentally, it is not representative of how a typical SOEC plant would operate. Nevertheless, the Redissi model served as a valuable first step in validating a working model for this study. Missing from the Redissi model was heat recuperation and recirculation of steam and CO₂ streams. Additionally, hydrogen is not a primary input to the stack as it is in the aforementioned model. As shown in Fig. 9, the model was updated to include heat recuperation (RHX1 and RHX2), greatly reducing the power input to the topping heater. The remaining electrical heaters were replaced with steam heat exchangers available from the LWR at 260–300 °C to super-heat the incoming gas streams (WX1 and WX2). Approximately 30% of the hydrogen was consumed in the RWGS; this was modeled by recirculating a portion of the hydrogen product to the inlet of the stack. The final syngas ratio was 2:1.

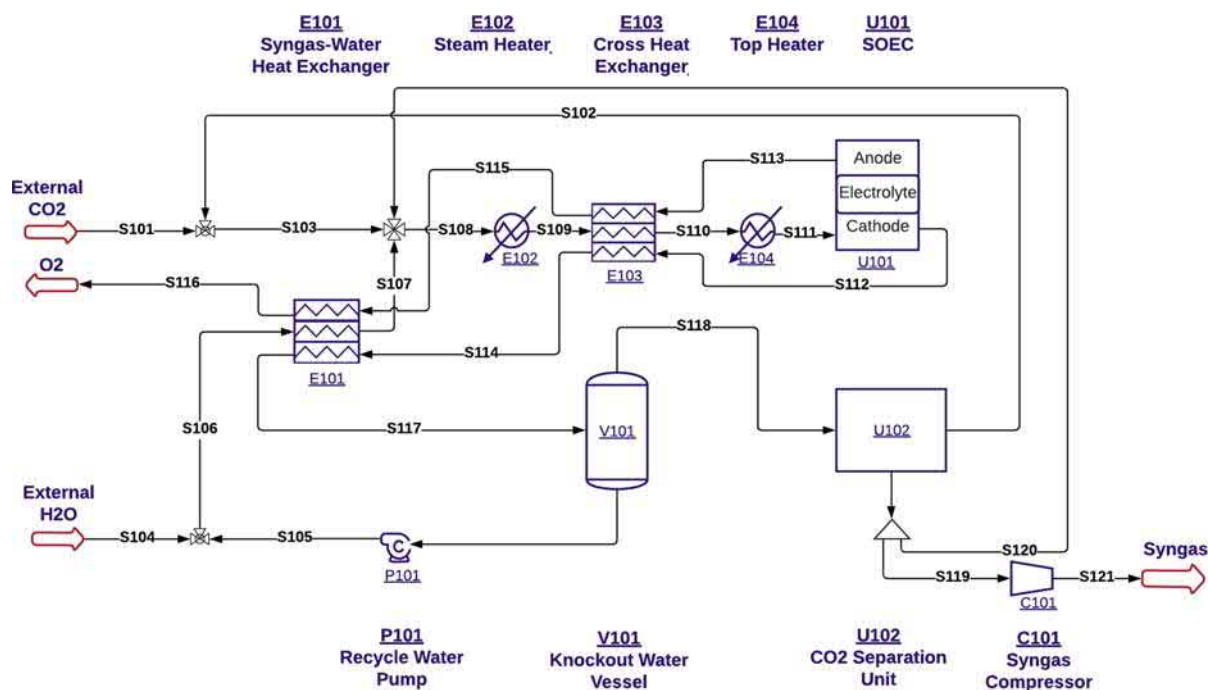


Fig. 9 ASPEN-Plus model with recirculation and heat recuperation.

SOEC cost estimation

Two different methods were used to estimate the cost of an SOEC system, one based on an extensive cost analysis of SOFCs, and the other based on an area-specific cost analysis of SOECs. The SOFC method only required the total electrical power required for the stack, whereas the other method required the number of cells required as calculated from Faradays Law. A detailed cost analysis was conducted for manufacturing, installation and operation of an SOFC for a nominal power output of 270-kW sized for ground-based distribution generation (Weimar et al., 2013). It found that electricity costs for a mass manufactured SOFC could be \$0.07/kWh based on a standard approach to manufacturing cells. A detailed study was conducted to understand the various steps required for manufacturing the units that included materials, equipment, and labor. Volumes were projected at 10,000 units/year. In addition, a sputtering approach for processing the units was considered and found to increase the performance of the stack and reduce capital costs by 33%. The cost study was based on 400-cm² cell area; however, the result was specific to power generation. Given the mole flow of monatomic oxygen from the cathode to the anode as given by the model, the number of cells can be determined from Faradays Law:

$$\Delta N_{\text{O}} = \frac{I_e}{2F}; \text{ where } I_e = iA_{\text{cell}}N_{\text{cells}}$$

Where the cell area is 250 cm² and the current density is 0.4 A/cm².

Using the sputtering method, the power-specific manufacturing and installation costs were found to be \$65/kW and \$182/kW, respectively. These costs are specific to the stack and housing costs and were scaled to provide an estimated installed cost for the SOEC in the modeled MTO-based fuel plant (278 MWe, see Section “Syngas to fuels via methanol-to-olefins process”) at \$68.8 M.

Table 5 shows a comparison of other SOEC cost estimates from the literature compared to Weimar’s method for the MTO-based model (Section “Syngas to fuels via methanol-to-olefins process”). Giglio et al. (2015) estimated the area-specific costs for a stack and its enclosure to be \$1500/m² of SOEC active area (i.e., the cell area). Using an assumed a cell size of around 225 cm² (O’Brien et al., 2009) to be acceptable given the available manufacturing techniques in 2009, and along with the current density of 0.4 A/cm² and monatomic oxygen flow across the cell from the model, the total cell area required is found to be 52,986 m². The total cost from this approach comes to \$79.5 M, for a difference of roughly 15% between the two approaches. Estimates from Buttler et al. (2015) and Anghilante et al. (2018) are \$93 M and \$61 M, respectively. The costs selected for this study (Weimar et al., 2013) is within the range of other literature values shown.

Syngas to fuels via methanol-to-olefins process

Design and modeling

The overall block flow diagram for the modeled distillate fuel production process through the MTO route is shown in Fig. 10. Syngas is first generated via co-electrolysis, as described in Section “SOEC design”. Raw syngas from the SOEC section is compressed to 420 psia. Entrained water is separated in knock-out pots prior to and between stages of compression and recycled back to the SOEC. Amine-based acid gas removal is then used to separate CO₂ from the syngas for recycle back to the SOEC. Removal of CO₂ also serves to reduce its concentration in the syngas feed to the downstream processes, thereby reducing equipment size and capital. Saturated steam at 374 °F is used to regenerate the amine solvent in a mass ratio of 2:1 for steam-to-CO₂ removed (Tan et al., 2016). Single-pass conversion of CO₂ is 29% and overall carbon efficiency to CO including recycle is 98.7%.

Syngas is then compressed to 925 psia, mixed with unreacted syngas, heated to 440 °F, and fed to the methanol reactor. Methanol synthesis occurs over a copper/zinc oxide/alumina catalyst. Process conditions for the methanol reactor are given in Table 6. Heat from the exothermic synthesis reaction is removed from the reactor via steam generation (Tan et al., 2016). Steam is used for downstream reboiler duties and for a portion of the steam needed in the CO₂ removal process. Single-pass conversion of CO is 34% and overall conversion with syngas recycle is 94%. Overall carbon efficiency of CO (and the low levels of CO₂ in syngas) to methanol is 93.2%.

Methanol is then preheated and fed to the MTO fluidized-bed reactor where ethylene, propylene, 1-butene, and 1-pentene are produced (Vora et al., 1998). Table 7 lists the assumed process conditions and conversion efficiencies. Some lights gases and coke are formed by side reactions; the coke is burned off periodically. The effluent mixture from the MTO reactor is cross exchanged to pre-heat the inlet gas and then quenched in a direct-contact, circulating-water spray tower to separate the non-condensable gases

Table 5 SOEC stack cost estimates from literature.

Source	Basis	Installed cost (stacks and enclosure)	MTO model (Section “Syngas to fuels via methanol-to-olefins process”) SOEC plant cost (2019\$; Scale 278 MWe)
Weimar 2013	Fuel cell power	\$247/kW	\$68.8 M
Giglio 2015	SOEC area	\$1500/m ²	\$86.7 M
Buttler 2015	SOEC area	\$1755/m ²	\$101.5 MM
Anghilante 2018	SOEC power	\$219.5/kW	\$61.5 M

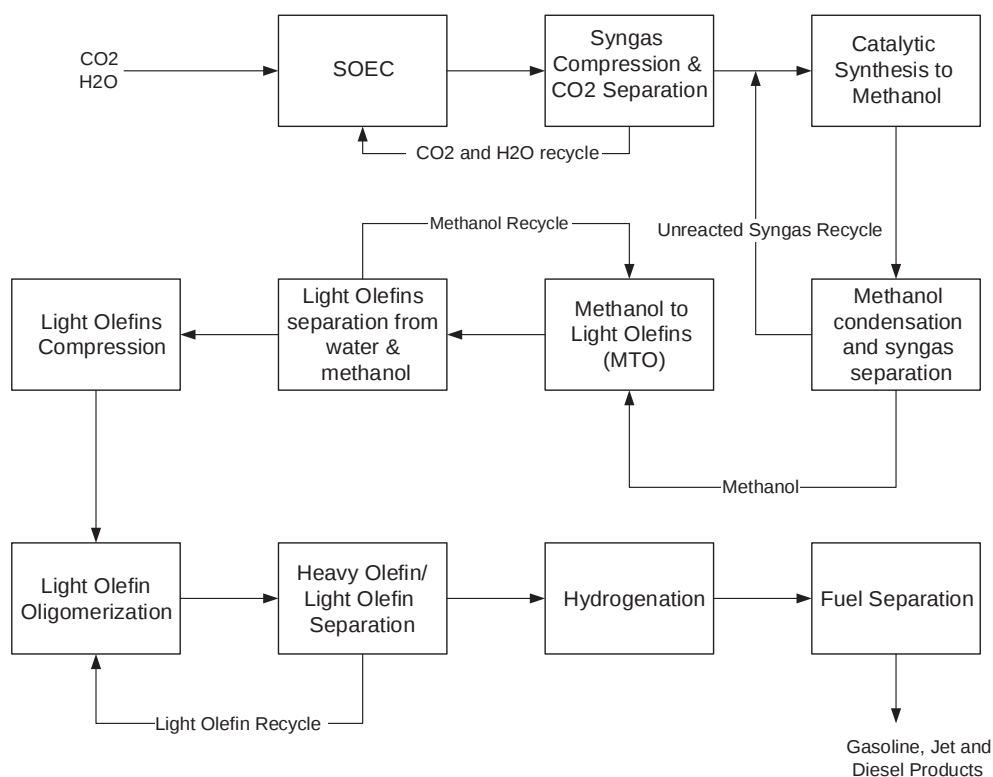


Fig. 10 MTO process diagram.

Table 6 Process conditions for methanol synthesis reactor (exothermic).

Assumption	(Jones and Zhu, 2009; Phillips et al., 2011)
Temperature, F	440
Pressure, psia	920
H ₂ : CO ratio	2.0
CO ₂ concentration (mol%)	4%
Single-pass CO conversion	34%
Overall CO conversion	94%

Table 7 Process conditions for methanol-to-olefins (exothermic).

Assumption	Reference: Vora (1998) and Gelbein (2003)
Temperature, F	814
Pressure, psia	19.7
WHSV	1
Single-pass methanol conversion	99%
Overall methanol carbon efficiency to olefins	93.4%

from water generated by methanol dehydration. Residual methanol is stripped from the quench column wastewater and recycled to the MTO reactor. The overall conversion of methanol is 99.98% with an MTO carbon efficiency of 93.4%.

The olefin-rich gas mixture goes through various separation steps and catalytic reactors to produce longer-chained hydrocarbons in the diesel-boiling range in an oligomerization step (Lilga et al., 2016). Oligomerization reactor conditions and conversions are listed in Table 8. The separation steps include a lean-oil scrubber used to recover olefins and remove methane and other paraffin by-products to prevent their accumulation in recycle streams used to increase the product yield.

Lighter olefins are recycled to increase their carbon number to the desired range of 9–16. Two catalysts are used to produce dimers and trimers of the reactive olefins to increase the carbon number. In the first oligomerization reactor, operated at 570 °F

Table 8 Process conditions for oligomerization of olefins (exothermic).

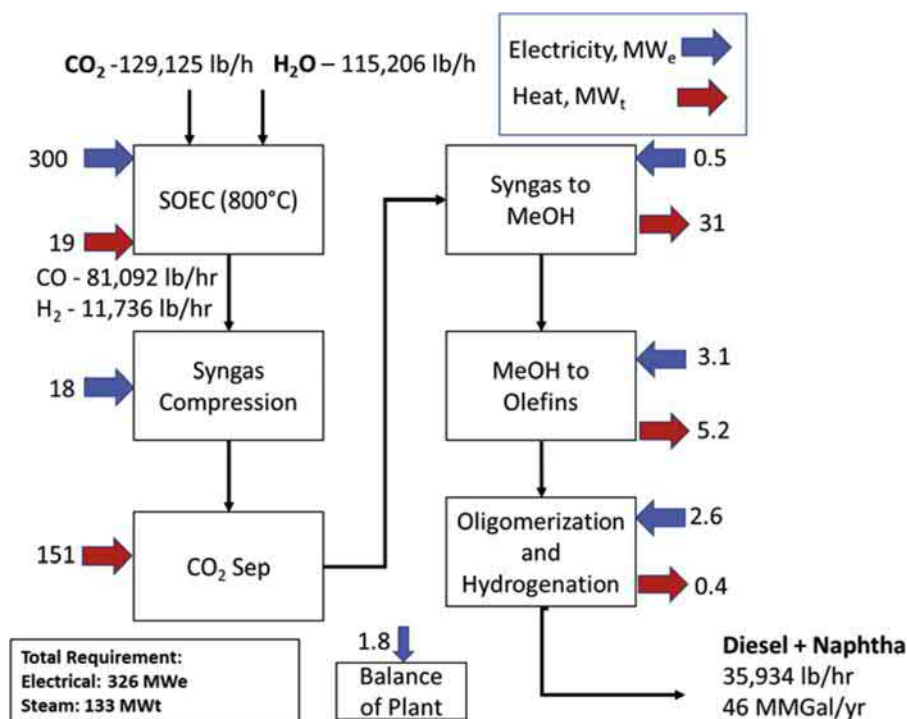
Assumption	Dutta et al. (2015) and Tan et al. (2016)
Temperature, F (first stage)	320
Pressure, psia (first stage)	302
Temperature, F (second stage)	288
Pressure, psia (second stage)	140
Stage 1 C ₂ –C ₅ olefins C eff. to C ₆ + olefins	48%
Stage 2 C ₂ –C ₅ olefins C eff. to C ₆ + olefins	49%
Overall C ₂ –C ₅ olefins C eff. to C ₆ + olefins	98.6%

and 302 psia, an H-Beta Zeolite catalyst is used to produce dimers and trimers of smaller (C₂–C₄) olefins to C₆+ olefins. The overall oligomerization reaction is exothermic; therefore, no heat is needed to drive the reaction. The assumed net conversions of ethylene, propylene and butene are 75%, 82%, and 27%, respectively. The second reactor, operated at 326 °F and 140 psia, uses an Amberlyst catalyst to increase the yield of C₉–C₁₈ compounds. The net conversion assumed for C₄, C₅, C₆, C₇, C₈, and C₉ olefins to their dimers are 40%, 36%, 100%, 26%, 23%, and 3%, respectively.

Lighter olefins (in the C₄–C₉ range) are recovered and recycled as part of the lean-oil scrubber feed, and the heavier olefins are hydrogenated using separately purchased hydrogen over a Pd-on-alumina hydrogenation catalyst operating at 750 °F and 130 psia. The H₂ partial pressure is maintained at about 70% of total pressure to minimize coking and to completely hydrogenate all double bonds in the feed. The excess hydrogen is separated from the diesel product after condensation and knock-out pots. It is recycled to the reactor operating pressure using a pressure booster. The hydrogenated product is distilled into gasoline (9%) and diesel (91%) blend stocks in a final distillation column. Overall carbon efficiency of olefins to fuel product is 98.5%.

Performance and economic results

Fig. 11 shows the resulting power and heat inputs and outputs for the modeled MTO process. About 92% of the power demand is for the co-electrolysis unit, with the remainder consumed for gas compressors and pumps throughout the plant. Steam is used for heating of the feed CO₂ and water to the SOEC system, for recovery of the amine solvent in the CO₂ recovery process, and for column reboilers and heating of various process streams. About 84% of the steam requirement for the integrated plant is for amine solvent recovery (shown as CO₂ Sep block below). Heat from the methanol reactor (syngas to MeOH block below) is used to produce a portion of the steam requirement, with the balance supplied by saturated steam (536 °F) from the LWR. The net power

**Fig. 11** Electrical and thermal inputs/outputs for the synfuels via MTO process using nuclear heat.

and heat requirements for the fuel plant are 326 MWe and 133 MWt. Electrical and thermal energy usage for syngas production for the modeled SOEC portion of the plant is 3.2 MWe/lb. syngas and 19.4 MWt/lb. syngas, respectively, for a H₂:CO ratio of 2.1. Note that this is only the energy usage for producing raw syngas and does not include steam used to recover amine solvent for separation of CO₂. All reactors downstream of syngas production and CO₂ separation are exothermic; therefore, no steam/heat is needed to drive these steps. For the syngas-to-methanol step, steam is generated on the shell side of the methanol reactor, capturing 31 MW of thermal energy. For the methanol-to-olefins step, 8.5 MWt of steam is generated from heat contained in the flue gas from the MTO reactor catalyst regenerator and 3.3-MWt steam is used to heat the feed stream and for the methanol recycle recovery column reboiler. For the oligomerization step, the heat generated is relatively low quality (<450 °F) and is not included in the overall thermal heat generation; however, it can be used for boiler water preheating. Distillation towers for separation of oligomers use 5.3 MWt of steam in reboilers. For the hydrotreating step, 6.3 MWt of steam is generated from the hot reactor effluent and 0.6 MWt of steam is used for reboiler heat in the diesel/naphtha fractionation column.

Table 9 lists the raw material, waste disposal and utility costs assumed for estimating variable operating costs for the fuel production plant. [Boardman et al. \(2019\)](#) estimated the levelized cost (including capital and operating expenses) of CO₂ delivered to a hybrid LWR-methanol plant scenario including capture from an ethanol plant, compression, storage, and transportation via pipeline to the nuclear power plant. They determined a range of \$14.6 to 38.3/t for a 530 t/day capacity methanol plant. The mean of this range (\$33.3/t) was used in this study and sensitivity analysis was conducted around this assumption. For comparison, the breakeven CO₂ sales price estimated for capture from fossil energy plants ranges from \$44/t to \$119/t ([James et al., 2019](#)). A range of \$0 to 120/t CO₂ cost was investigated in the sensitivity analysis. Although clean O₂ could potentially be a co-product of the process, to be conservative, no O₂ credit is included in the analysis.

Table 10 gives the major performance and economic results for the synfuel plant. Also presented for comparison is the biomass case, where gasification of woody feedstock is used on the front end instead of co-electrolysis. Woody feedstock cost is assumed to be \$63.23/dry ton ([Hartley et al., 2018](#)). Feedstock CO₂ and water for the SOEC are 1549 t/day and 331,369 gal/day, respectively. For perspective, this is about 1.75 times the daily CO₂ produced from a typical corn ethanol plant (100 million gallon/year) and about half the capacity of an Olympic-sized swimming pool, respectively. Fuel generation is about 2900 barrel/day (BPD) of diesel and 295 BPD of naphtha (motor gasoline blendstock). The U.S. demand for diesel and gasoline in 2019 was 3.3 million BPD and 9.3 million BPD, respectively ([EIA, 2020a](#)). Energy efficiency for the co-electrolysis process is similar to the gasification case, but overall carbon efficiency is much higher due to the high selectivity of the electrolysis reactions. Feedstock cost is higher for the biomass case due to the lower carbon efficiency as compared to the co-electrolysis case. Capital costs for the syngas generation are higher for the co-electrolysis case due to the assumed cost of the SOEC stacks. However, syngas cleanup and methanol production costs are lower due to a cleaner and lower total flow of gas (fewer light ends and CO₂ than from gasification) through the compression system and reactor than results from biomass gasification. The fuel production cost breakdown is given on the bottom half of **Table 10** and illustrated in **Fig. 12**. The MFSP for the SOEC case is \$4.45/gal, with electricity and steam cost making up about half of the total production cost. Optimistic and conservative cases using the lower and upper bounds respectively for estimated electricity and steam price from an LWR (\$0.02/kWh with \$2.49/1000 lb1000lb and \$0.04/kWh with \$4.70/1000 lb1000lb steam, respectively) are included for comparison. Most of the steam requirement is for removal of CO₂ from the syngas (recovery of amine solvent). In the biomass gasification case, extra steam is generated from heat recovered from the char combustor, which supplies all heat needs for the plant as well as onsite generation of power. In the SOEC case, steam must be supplied from the LWR.

Table 9 Variable operating costs for the MTO to fuels model TEA.

<i>Variable operating costs</i>	<i>2019 Price</i>	<i>Unit</i>	<i>References</i>
<i>Raw materials</i>			
CO ₂	33.3	\$/t	James et al. (2019)
Methanol synthesis catalyst	12.03	\$/lb	SRI PEP 2007 Yearbook
MTO catalyst	37.14	\$/lb	Gelbein (2003)
1st stage oligomerization catalyst	11.30	\$/lb	Dutta et al. (2015)
2nd stage oligomerization catalyst	18.10	\$/lb	Tan et al. (2016)
Hydrogenation catalyst	59.0	\$/lb	Gerber (1999)
Amine makeup	1.70	\$/lb	Phillips et al. (2007)
Boiler chemicals	3.26	\$/lb	Phillips et al. (2007)
Cooling tower chemicals	1.95	\$/lb	Phillips et al. (2007)
Hydrogen gas (for hydrotreating of oligomers)	0.95	\$/lb	2020 PEP Yearbook
<i>Waste disposal</i>			
Wastewater treatment	3.20	\$/100 ft ³	Phillips et al. (2007)
<i>Utilities</i>			
Cooling tower makeup	236.5	¢/1000 gal	Phillips et al. (2007)
Process water to SOEC	327.0	¢/1000 gal	Redissi and Bouallou (2013)
Boiler feed water makeup	236.5	¢/1000 gal	Phillips et al. (2007)
Process steam	359	¢/1000 lb. steam	Boardman et al. (2019)
Electricity	3.0	¢/kwh	Boardman et al. (2019)

Table 10 Economic results for syngas to fuels process (all costs in 2019 \$).

Flowrates	Co-electrolysis		Biomass gasification	
CO ₂ feed, lb./h (t/day)	129,125 (1549)		Biomass: 183,718 (2205)	
H ₂ O feed, lb./h (t/day)	115,206 (1381)		N/A	
Diesel blendstock, lb./h (BPD)	32,808 (2899)		32,835 (2903)	
Gasoline blendstock, lb./h (BPD)	3126 (295)		3129 (296)	
Diesel blendstock, mmBtu/h (MW)	623.4 (182.7)		624.1 (183.0)	
Gasoline blendstock, mmBtu/h (MW)	59.8 (17.5)		59.9 (17.6)	
Carbon efficiency (C in synfuel/C in feed)	86.1%		32.4%	
Energy efficiency (fuel)/(power + steam + H ₂)	41.5%		43.0% (including input biomass)	
Capital costs, \$ million				
Installed costs				
Syngas generation	87.4		51.3	
Syngas compression and cleanup	53.0		71.8	
Methanol production	30.6		49.0	
Hydrocarbon fuel production	60.1		58.8	
Steam cycle/power gen	5.4		34.1	
Balance of plant	4.8		7.7	
Total installed capital cost	241.3		272.7	
Indirect costs	125.2		140.9	
Fixed capital investment	400.7		451.900	
Total capital investment (TCI)	422.3		475.2	
Operating costs and production cost breakdown				
	\$ million/year	\$/gal fuel blendstock	\$ million/year	\$/gal fuel blendstock
Variable operating cost				
Feedstock	15.5	0.34	45.8	0.99
Hydrogen (for hydrotreating oligomers)	4.0	0.09	3.6	0.08
Catalyst and chemicals	3.9	0.08	4.8	0.10
Waste disposal	0.3	0.01	6.0	0.13
Electricity	77.1	1.67	0.8	0.02
Steam	22.3	0.48		
Fixed costs	21.9	0.47	23.7	0.51
Capital depreciation	20.0	0.43	22.6	0.49
Average income tax	5.0	0.11	5.6	0.12
Average return on investment	35.4	0.77	38.6	0.83
MFSP, \$/gal fuel (91% diesel, 9% gasoline)	4.45		3.28	
MFSP, \$/GGE fuel (91% diesel, 9% gasoline)	4.42		3.26	
MFSP, \$/gal diesel blendstock	4.38		3.23	
MFSP, \$/gal gasoline blendstock	4.21		3.11	

Economic studies from the literature for synthetic fuels production yield a wide price range. This is to be expected with the highly variable processes and technical and economic assumptions that are possible. One of the most established technologies for converting syngas to hydrocarbon fuel is via the FT route. Several pilot and demonstration tests using co-electrolysis-based syngas with FT synthesis from syngas to fuels were conducted between 2014 and 2022 in Europe (Dieterich et al., 2020). Several groups have conducted TEAs for FT fuels via electrolysis of CO₂. Li et al. (2016) reported a range of \$3.80 to 9.20/gal with a range of well-to-gate energy efficiency of 41–65%. Becker et al. (2012) found a range of \$4.4 to 15/GGE (gasoline-gallon equivalent) for electricity price range of \$0.02 to 0.14/kWh and plant capacity range of 90–40% and reported an overall efficiency of 51% (LHV). In a study by Fu et al. (2010), production cost ranged from \$2.50 to 6.79/gal with an electricity price of \$22–88 MWh. Cost results from this analysis lie within the general cost range of FT fuels found in the literature.

It is evident that the MFSP for the low-carbon fuel from this pathway fuel is higher than current market petroleum fuel prices (Fig. 13). However, possible carbon credits through government incentives or mandates are not considered in these results and could be substantial if put into place in the future. For example, credits like those granted under the Environmental Protection Agency (EPA) Renewable Fuel Standard (RFS) for biofuels may be a good first approximation for future incentives associated with CO₂-based fuels. State programs such as California's low-carbon fuel standard (LCFS) could bring additional value incentives. Renewable identification number (RIN) credits for advanced biofuel, defined as fuel with associated lifecycle greenhouse gas (GHG) emissions that is 60% lower than the petroleum baseline (e.g., gasoline and diesel), was valued in the range of \$1.01 to 2.74 per gallon over the 2018–19 time period (Fig. 14, (EPA, 2020)). Lifecycle GHG emissions for the MTO-based fuel from co-electrolysis are estimated at 13.8 g CO₂-e/MJ as shown in Table 11. This represents an 85% reduction in GHGs compared to petroleum diesel (91.8 g CO₂-e/MJ, GREET, 2019). Note that this is a relatively high-level estimate that includes emissions associated with feedstock production/preparation (compression of CO₂ feed at the ethanol plant for CO₂ and woody biomass collection and preparation for

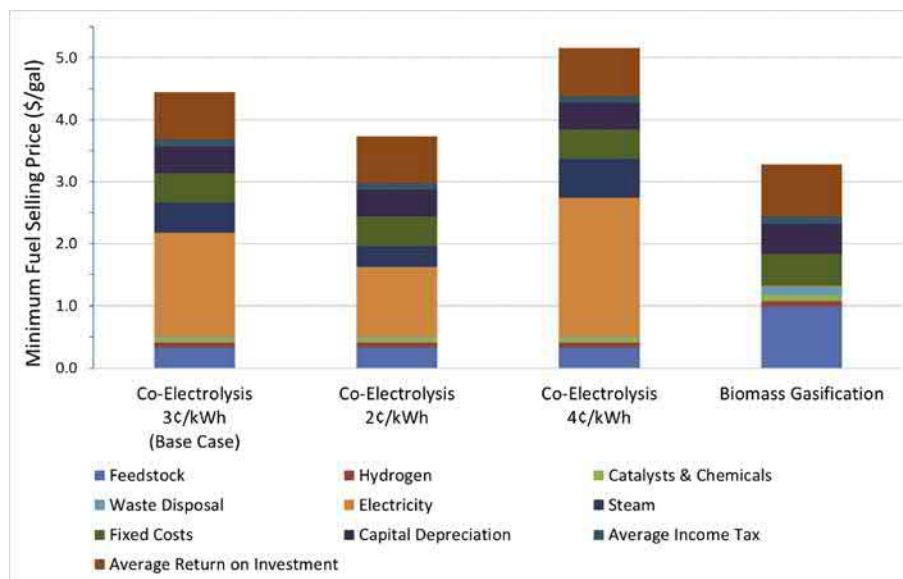


Fig. 12 Fuel production cost breakdown for renewable fuel blendstock via co-electrolysis and biomass gasification and the MTO route for the conversion of the syngas to synfuels.

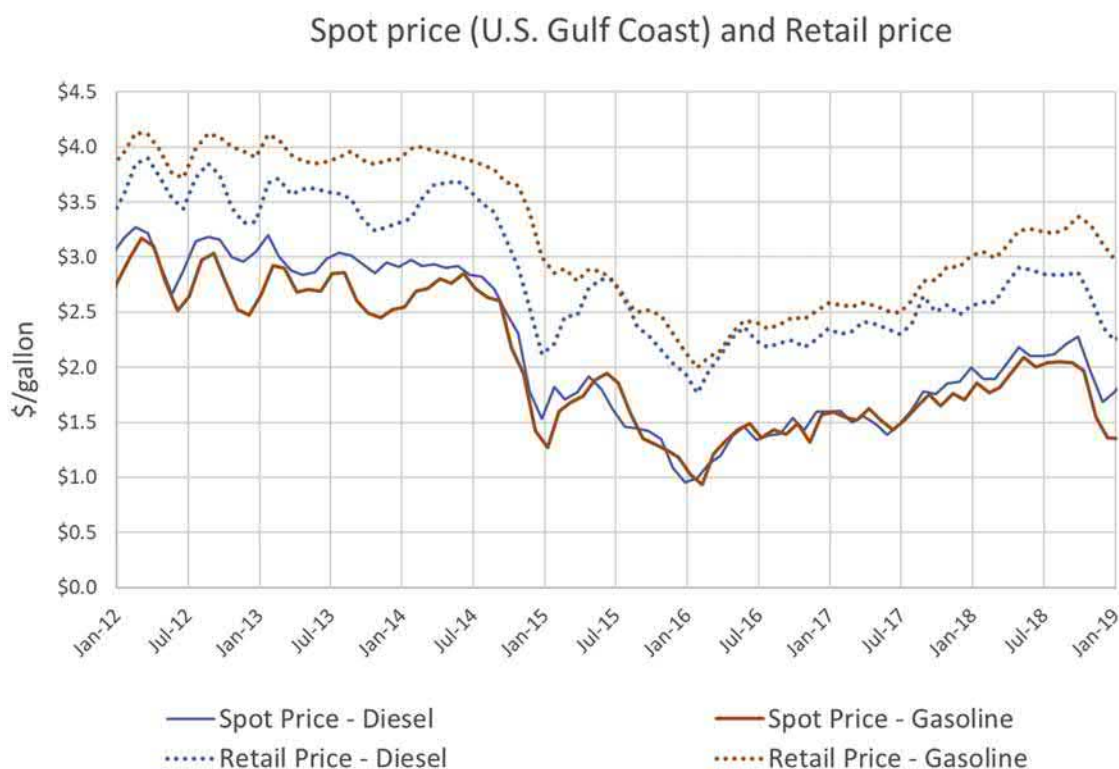


Fig. 13 Petroleum fuel price history (EIA, 2020b).

the gasification case), electricity (nuclear power for the co-electrolysis case and grid power for the biomass case) and hydrogen used for hydrotreating olefins into final fuel blendstock.

Sensitivity analysis (Fig. 15) investigating variable CO₂ cost, CO₂ credits, and electricity price shows that with optimal CO₂ and electricity prices and inclusion of carbon credits through incentives or mandates could make this process more cost competitive with petroleum fuels. With a hypothetical carbon tax of \$100/t CO₂ the MFSP is reduced to ~\$3.75/gal. An RFS credit would further aid in competitiveness of fuels produced via this route. Carbon sequestered into fuel minus CO₂ emitted in process off gas combustion and embodied GHGs for H₂ requirement for hydrotreating of the final fuel was estimated. The carbon credit reduces MFSP by

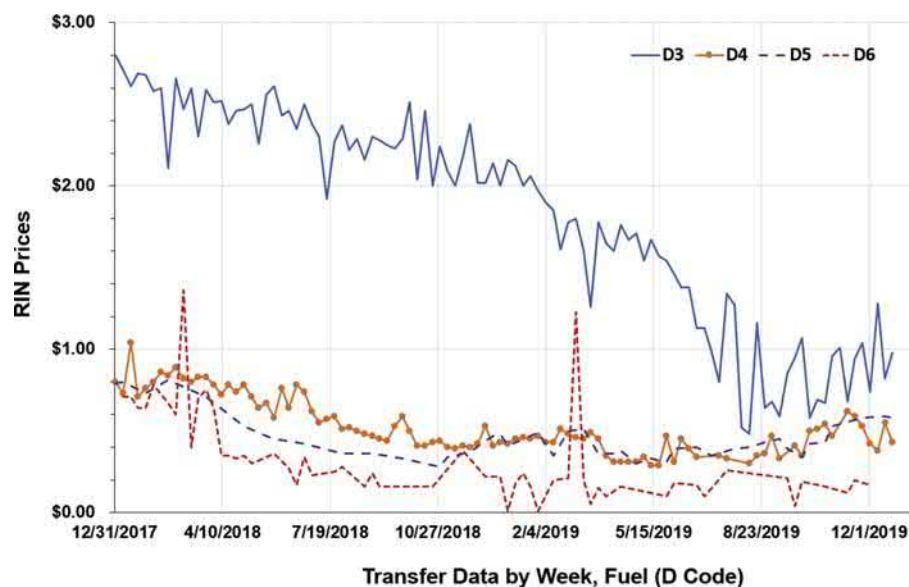


Fig. 14 Price for renewable fuel credits during 2018–19 (EPA, 2020).

Table 11 Greenhouse gas emissions calculation for fuel from the MTO-based pathway.

	<i>Co-electrolysis</i>	<i>GHG factor and ref</i>	<i>Biomass gasification</i>	<i>Ref for GHG emission factor</i>
Feedstock	6.2 (CO ₂ compression)	157.1 kWh/t feed (Boardman et al., 2019); 142 kg CO ₂ -e/mmBtu grid mix (GREET, 2019)	13.9 (50/50 forest residue/clean pine)	109 kg CO ₂ -e/dry t (Hartley et al., 2019)
Electricity	3.6 (Nuclear)	2.4 kg CO ₂ -e/mmBtu (GREET, 2019)	0.7 (Grid mix)	142 kg CO ₂ -e/mmBtu (GREET, 2019)
Hydrogen for hydrotreating	3.9	105 kg CO ₂ -e/mmBtu (GREET, 2019)	3.9	105 kg CO ₂ -e/mmBtu (GREET, 2019)
Total ^a	13.8		18.5	
Reduction from petroleum diesel	85%		80%	

^aDoes not include contribution of chemicals and catalysts consumption.

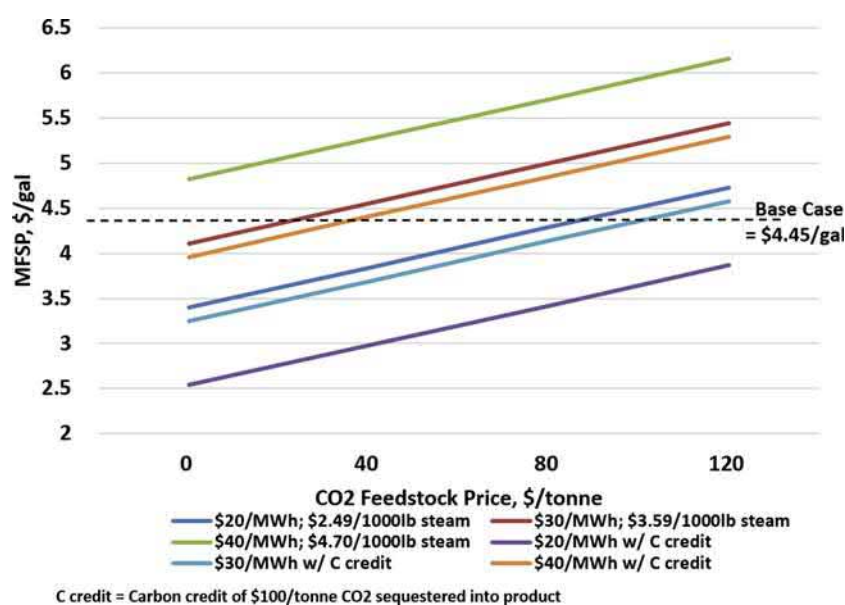


Fig. 15 Sensitivity of MFSP for MTO fuel to CO₂ and electricity price and considering the potential impact of a carbon credit.

approximately \$0.86/gal, somewhat less than the average 2018/2019 D3 advanced biofuel RIN price of \$1.88. This analysis indicates that fuel from this pathway could be more competitive with petroleum prices with a combination of lower electricity price, lower CO₂ price, and qualifying carbon credits. If a D3 RIN-type price could be applied, competitiveness could be further improved.

Also, there are innovative cryogenic carbon capture (CCC) processes that could have significant impact on the viability of an LWR/synfuels plant using methanol as intermediate. Further, the refrigerant used in the CCC process could be produced using LWR energy. The synergies of the LWR with the CCC process and techno-economic modeling of the CCC process will be explored in detail in future studies.

Conclusions and future directions

This article gives a general overview of the many possible pathways to produce sustainable hydrocarbon fuels and chemicals from CO₂ and H₂ and presented detailed process model and TEA for one example of power-to-fuels using steam and electricity from an LWR. Using the base case assumptions of \$33.3/t CO₂ cost and \$30/MWh electricity cost, the modeled MFSP for the methanol-based route to fuel is estimated to be \$4.45/gal for a fuel production plant of 3194 BPD capacity. Production costs for the co-electrolysis routes using these above assumptions leads to a cost about 40% higher than using wood gasification for syngas production, primarily due to electricity usage and steam usage for co-electrolysis and CO₂ separation. The assumptions used in the base case are conservative and do not account for improving technologies in the areas of CCC. Sensitivity analysis shows that a combination of lower electricity and steam price (\$20/MWh and \$2.49/1000 lb. steam), low CO₂ price (\$15–50/t), and inclusion of carbon credit of a \$100/t can make the process more competitive with petroleum fuels. GHG analysis indicates that the fuels have 85% less GHG emissions than petroleum fuels (diesel) and could conceivably qualify for the highest RIN credit if approved by the EPA. This, along with additional state incentives could bring an even higher potential credit for the fuel.

A recent report studied the framework of options that can incentivize nuclear power plant operations by providing credits for the low-carbon grid power and non-electric products that may be produced from nuclear power, similar to REC. Some conclusions from this report are highlighted below as they apply to synthetic fuel produced by coupling with low-carbon nuclear energy.

Other reports completed by the DOE LWRS program have highlighted the vast and diverse markets for non-electric products that can be produced using nuclear energy (Knighton, 2020; Hu et al., 2019; Frick et al., 2019). The current chapter has supplemented these studies by providing a first look into two possible pathways for producing synthetic transportation fuels by coupling with low-carbon nuclear energy.

Retiring nuclear plants and not valuing this low-carbon energy with the commensurate credits given to renewable energy may lead to drastic increases in carbon emissions (from substitute baseload plants such as NGCC) at a time when decreases in carbon emissions are being sought, which would be contrary to the goals of decarbonization.

Important points to consider related to possible future low-carbon/zero emissions credit legislation:

- The retention of nuclear power generation is critical to achieving federal and states' decarbonization goals across multiple energy sectors.
- Producing hydrogen and other products such as synthetic fuels from nuclear power, especially at low demand periods, increases the capacity utilization factor of NPPs, which can improve the economics of their operation.
- The contribution of nuclear power to zero-carbon power markets can be extended further to serve other energy sectors such as transportation, as well as building and industrial heat demand, thus contributing to the goals of decarbonization across multiple energy sectors.

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Ammonia Synthesis

Richard D. Boardman, Idaho National Laboratory, Idaho Falls, ID, United States

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Glossary

Ammonium nitrate A chemical compound with the chemical formula NH_4NO_3 . It is a white crystalline solid consisting of ions of ammonium and nitrate. It is highly soluble in water and hygroscopic as a solid, although it does not form hydrates. It is predominantly used in agriculture as a high-nitrogen fertilizer. Global production was estimated at 21.6 million t in 2017.

Haber-Bosch process An artificial nitrogen fixation process and is the main industrial procedure for the production of ammonia today. It is named after its inventors, the German chemists Fritz Haber and Carl Bosch, who developed it in the first decade of the 20th century.

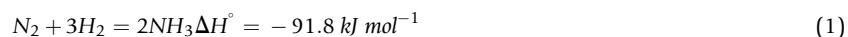
Urea A common form of fertilizer produced by combining ammonia with carbon dioxide which can be formulated into a solid or solution for agriculture application.

Introduction

Ammonia is the basic building block of the world nitrogen industry; consumption of ammonia for nitrogen fertilizer accounts for more than 93% of the world ammonia market. Nitrogen fertilizers are the most widely used fertilizers in the world, accounting for close to 73% of all fertilizers. With the exception of China, where much of the ammonia is produced from coal gasification, most of the world's ammonia is produced from natural gas via steam reforming. From 2011 to 2020, world apparent consumption of ammonia increased at about 2% per year (Food and Agriculture Organization of the United Nations (FAO), 2019; ICIS, 2017). Ammonia capacity is increasing primarily in areas where the availability and cost of natural gas are lower, in particular the United States and the Middle East. The global ammonia supply/demand balance is expected to move toward a surplus, as future capacity additions will continue to outpace consumption, leading to declining utilization rates.

The major method for commercial production of anhydrous ammonia is a modified Haber-Bosch process from natural gas via steam reforming (Pattabathula and Richardson, 2016).

The process converts atmospheric nitrogen (N_2) to ammonia (NH_3) by a reaction with hydrogen (H_2) using a metal catalyst under high temperatures and pressures:



This process was first demonstrated in 1909 and was commercially developed in 1913 by BASF in Germany with a capacity of 30 Mt. d^{-1} . During the past 60 years, ammonia process technology has improved drastically. Plant layouts evolved from multi-train designs, often with different numbers of trains in the front end and synthesis loop, to single-train designs. Synthesis gas preparation in the front end of the plant increased from atmospheric pressure to 30–50 bar_g pressure. Capacities increased from 100 t d^{-1} to as much as 3300 t d^{-1} in a single train. Energy efficiencies have improved as well—from consumption above 60 GJ t^{-1} of ammonia in coal-based plants to 40–50 GJ t^{-1} in the first natural gas based plants to 30–40 GJ t^{-1} in single-train plants. Modern plants have added heat recovery by steam production at pressures as high as 125 bar_g in both the syngas preparation section and the synthesis loop.

Nuclear energy can be used to support ammonia production and its associated fertilizer products in both a direct and indirect direct manner. Although ammonia synthesis is exothermic and gives up some useful heat, hydrogen production by natural gas reforming at the start of the process train is highly endothermic and requires extra, high temperature heat. This is provided by burning about 25–35% of the natural gas feed. Even after heat integration, the best plants still emit a significant amount of CO_2 . With some simple modifications to the natural gas reformers, nuclear-generated thermal energy can replace all of the natural gas that is combusted.

Alternatively, hydrogen production by electrolysis or water or steam and separation of nitrogen from air can be performed separately and stored for use by the Haber-Bosch ammonia synthesis reactor. Hydrogen, nitrogen, and ammonia production need not be synchronously operated and can be powered by a nuclear plant. This form of indirect coupling with a nuclear plant has several advantages, including total elimination of GHG emissions. It also provides flexibility in allowing the nuclear power plant to provide peak power to the electricity grid when the revenue generation by participating in the capacity market is greater than the cost of hydrogen and nitrogen storage. In this manner the Haber-Bosch process can be operated at steady state.

This section summarizes a comprehensive analysis completed by Idaho National Laboratory under the United States Department of Energy (US DOE) Next Generation Nuclear Plant Program (Wood, 2010). In this study, the possibility of integrating a High Temperature Gas-Cooled Reactor (HTGR) with conventional steam methane reforming and coal-gasification ammonia production plants were evaluated. The principles developed in the research activity are widely applicable to other advanced nuclear reactor designs. At a minimum, hydrogen that is produced by coupling a nuclear plant to a water-splitting electrolysis plant can have a large impact on ammonia and ammonia-derivative products and fertilizers, such as urea, nitric acid, and ammonium nitrate.

Conventional ammonia production plants

Producing ammonia from natural gas and coal are fundamentally different. Natural gas is hydrogen rich with a carbon-to-hydrogen ratio of 1:4. Coal is hydrogen limited with a carbon-to-hydrogen ratio of approximately 1:0.8. Coal requires significant capital to gasify and to clean up the resulting syngas of CO and H₂. CO is then used to produce additional H₂ via the water-gas shift reaction:



Excess heat is generated in the coal-based case due to carbon rejection (as CO₂).

Fig. 1 shows a representative block flow diagram for a conventional natural gas-to-ammonia plant and associated products. The overall process includes unit operations for natural gas reforming, syngas cleaning and conditioning, CO₂ compression, ammonia synthesis, nitric acid synthesis, ammonium nitrate synthesis, urea synthesis, power production, cooling towers, and water treatment.

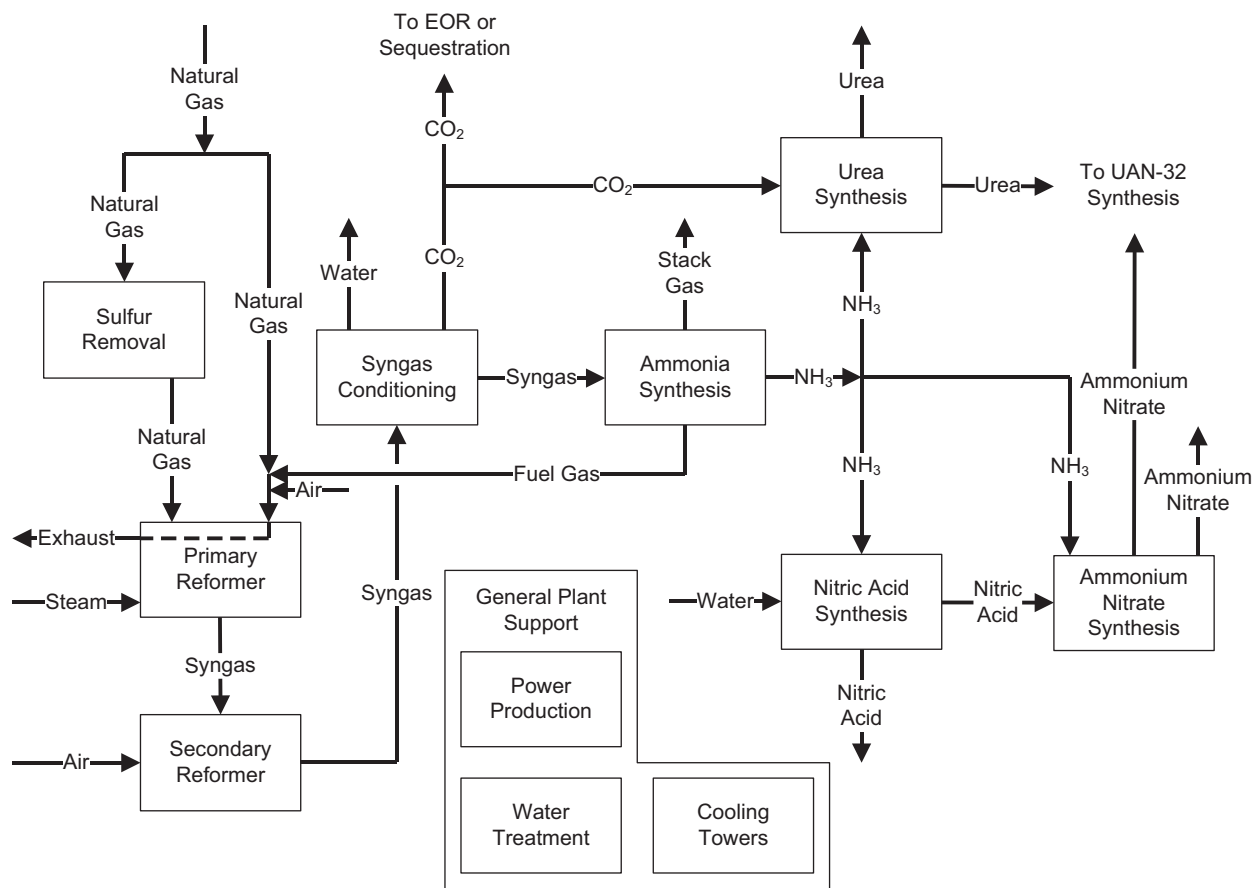


Fig. 1 Block flow diagram for the conventional natural gas-to-ammonia process. Next Generation Nuclear Plant Wood RA (2010) *Nuclear-Integrated Ammonia Production Analysis*. Technical Evaluation Study-666. Project 23843.

Syngas feeding the ammonia synthesis unit is adjusted to achieve the H_2/N_2 molar ratio 3.0. Incoming feed gas is compressed to 3000 psi. Unreacted recycle gas is mixed with the fresh feed gas and preheated by cross exchanging with hot reactor effluent gases. Equilibrium conversion is assumed in the ammonia converters by the following simple molecular balance of Eq. (1).

Two-step reforming consisting of primary steam reforming followed by secondary autothermal reforming was selected for syngas generation. Air is used as the oxidant in the autothermal reforming step, as this provides nitrogen to the process for downstream ammonia synthesis (Eggerman, 2010). By carefully controlling process parameters, such as the steam-to-carbon inlet molar ratio, primary reformer temperature, amount of preheat to the secondary reformer, and secondary reformer temperature, a syngas containing the appropriate H_2/N_2 ratio for ammonia synthesis can be produced. Additionally, if the steam-to-carbon ratio is set high enough, additional water will not be required prior to downstream shift conversion. For this case, key parameters were set as follows:

Preformer steam-to-carbon ratio	3.30
Primary reformer exit temperature	790 °C
Autothermal reformer outlet temperature	950 °C

Effluent from the first ammonia converter is cooled by cross exchange with the reactor influent, followed by cooling in a steam generator. Additional steam is generated from the hot syngas downstream of the second and third ammonia conversion stages. Final cooling of the third stage effluent gas is accomplished using cooling water and recuperation with the cool recycle gas stream. Ammonia product is recovered in an ammonia separator. Effluent gas from this separator is further cooled using refrigeration. Additional ammonia is recovered in a second separator downstream of the refrigeration unit. Effluent gas from the second separator is recycled to the ammonia converters. Before entering the ammonia converters, the recycle gas is recompressed using a boost compressor and mixed with fresh syngas.

A mono-pressure process is used for nitric acid synthesis due to capital cost advantages over a dual-pressure process. Cold ammonia is first preheated prior to mixing with compressed air. The ammonia-air mixture is reacted in an ammonia converter to produce N_2 , NO, and H_2O . Due to the exothermic nature of the reactions, an exit temperature of 925 °C is reached. Heat recuperation and steam generation are implemented downstream of the converter. As the gas cools, additional reaction with oxygen takes place to produce NO_2 , N_2O_4 , and finally HNO_3 . Aqueous nitric acid is separated from the product stream while gaseous products are routed to an absorber to increase the yield of acid. A bleacher is included to further enhance reaction of NO with oxygen to improve acid yield. Unreacted gas exiting the absorber is routed to a selective catalytic reduction unit for NO_x abatement prior to release to the atmosphere. This tailgas is also run through an expander, which significantly reduces the net electrical load required for air compression.

Ammonium nitrate is produced by combining ammonia with nitric acid after preheating the feed to a neutralizer where the following reaction occurs (Speight, 2002):



Feeds to the process are controlled to maintain a pH of 1.5 in the neutralizer. The neutralizer products are then separated under slight negative pressure to remove excess water from the ammonium nitrate solution. Ninety-five percent of the solution is recycled back to the neutralizer, while 5% is withdrawn and further concentrated in a two-stage evaporator. Following evaporation, the concentrated ammonium nitrate solution is solidified using a prill tower. Process recycle and heat recuperation minimize energy input requirements.

Urea is produced by reacting ammonia with CO_2 to form ammonium carbamate, which simultaneously dehydrates to urea to form liquid product (Mavrovic et al., 2000):



It is convenient to separate the CO_2 produced by the combustion of natural gas or the coal gasifier outgas. Liquid urea product is concentrated using evaporation under vacuum conditions. The concentrated urea solution is then solidified using a fluidized bed granulator.

Fig. 2 shows the block flow diagram for a conventional coal-to-ammonia plant with production of derivatives. Major differences relative to a gas-to-ammonia plant are the air separation, coal milling and drying, coal gasification, syngas cleaning and conditioning, sulfur recovery, and CO_2 purification/compression unit operations. These units are not insignificant. The capital cost and maintenance of coal (or biomass) gasifier, and syngas cleanup is higher than a steam methane reforming process unit. Sulfur (mostly in the form of H_2S) must be separated from the syngas stream and recovered in a Claus sulfur recovery plant (Breckenridge et al., 2000; Burr and Lyddon, 2008). Oxygen is produced via a standard cryogenic Linde-type air separation unit (ASU) that utilizes two distillation columns and extensive heat exchange in a cold box. The oxygen product is used for gasification. An O_2 purity of 99.5% is specified, although a 95% O_2 purity plant would also be suitable for this application. Oxygen demand for gasification drives the size of the ASU; hence, in addition to the primary nitrogen demand for ammonia synthesis, excess nitrogen is also available for other plant uses, such as coal drying and transport.

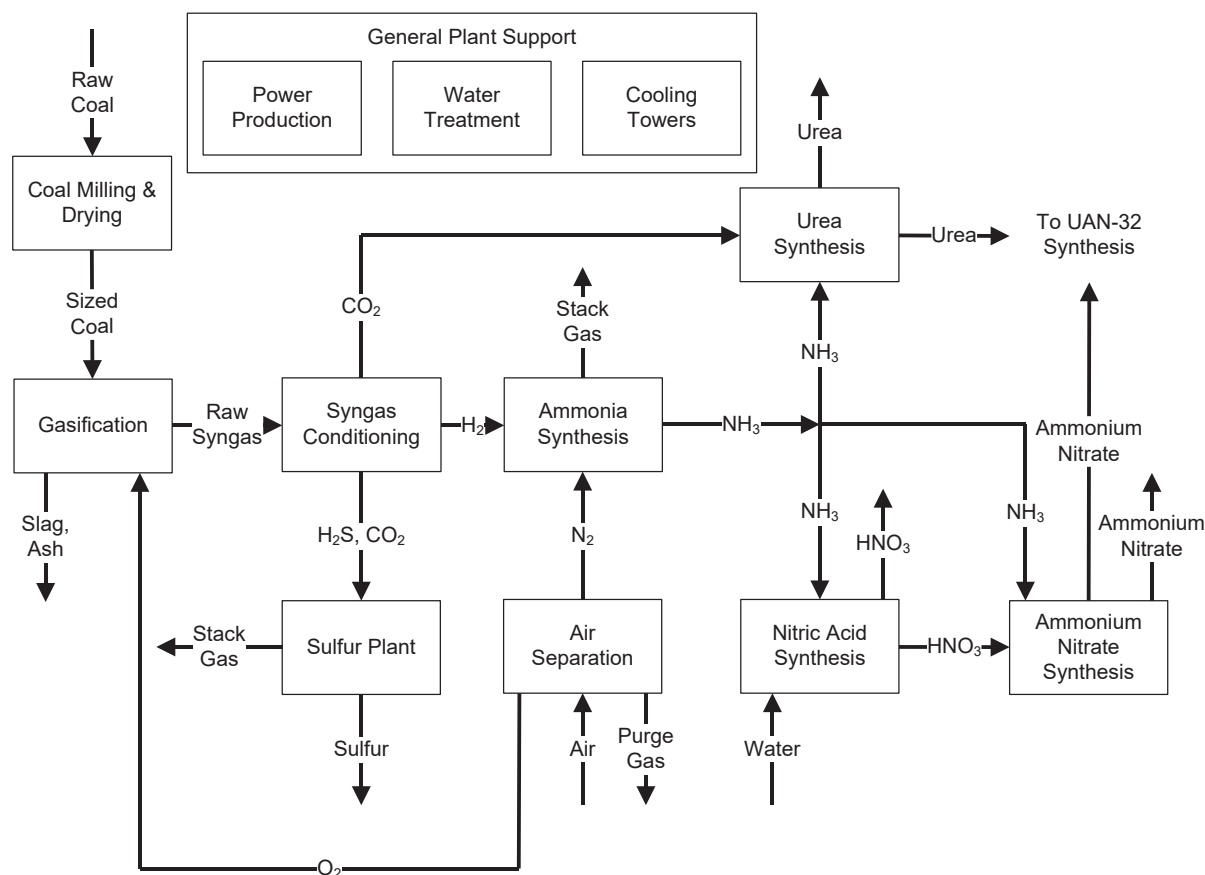


Fig. 2 Block flow diagram for the conventional coal-to-ammonia process. Next Generation Nuclear Plant Wood RA (2010) *Nuclear-Integrated Ammonia Production Analysis*. Technical Evaluation Study-666. Project 23843.

Nuclear-integration with ammonia plants

Fig. 3 shows the block flow diagram for the nuclear-integrated natural gas-to-ammonia case. The proposed process includes the same unit operations as the conventional natural gas process except nuclear heat is used in the primary reformer rather than burning natural gas. Nuclear heat is also used to preheat all streams entering the primary reformer. Although nuclear heat could be used by the urea synthesis and water treatment plants, sufficient low-temperature heat is produced elsewhere in the plant to meet these requirements. Because higher temperature heat is desired for steam reforming of natural gas, this configuration assumes the use of an intermediate heat exchanger (IHX) to provide 700 °C helium gas as the heat transfer fluid to the reformer. This allows the conventional tubular reforming reactors to be adapted without significant modification to the reactor. Additionally, an electrically assisted compander in the nitric acid block allows pressure exergy to be conserved.

Fig. 4 shows the block flow diagram for the nuclear-integrated coal-to-ammonia case. As this diagram shows, a need for high-temperature heat (> 300 °C) integration was not identified. Additionally, modeling results for the conventional coal-to-ammonia case indicate that heat for coal drying, water treatment, and urea synthesis can be supplied by existing unit operations in the conventional configuration. Hence, electricity is the best integration option for this flow sheet. Because light water reactors could supply this need, there is no driving reason to integrate this flow sheet with that for advanced nuclear reactors.

Fossil fuel-free ammonia plants

The advent of water-splitting electrolysis is recognized an approach to producing ammonia without the production of syngas and hydrogen from fossil-fuel sources. It only becomes necessary to produce a stream of nitrogen to produce the ammonia. This can be accomplished with an air separation unit (ASU) equivalent in operation, but smaller than the ASU featured in the coal gasification process. Alternatively, nitrogen at sufficient purity can be generated by burning hydrogen and condensing the vapor byproduct, but this option was proven to be significantly more energy intensive and costly than obtaining nitrogen with an ASU. In this section, high temperature steam electrolysis (HTSE) is considered favorable to low temperature electrolysis given the greater thermodynamic synergy with nuclear reactors.

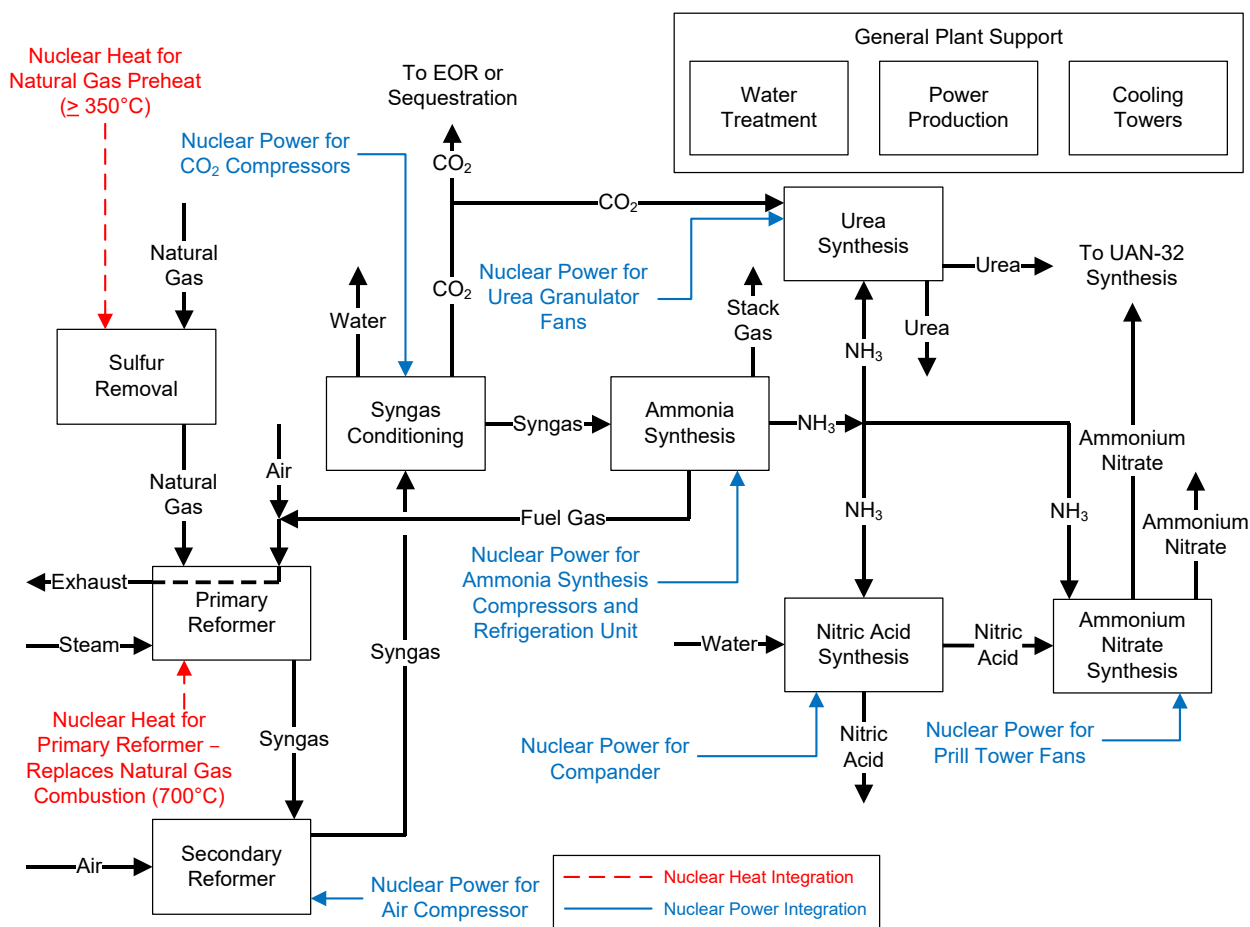


Fig. 3 Block flow diagram for the nuclear-integrated natural gas-to-ammonia process. Source: Next Generation Nuclear Plant Wood RA (2010) *Nuclear-Integrated Ammonia Production Analysis*. Technical Evaluation Study-666. Project 23843.

Fig. 5 shows the block flow diagram for a high temperature steam electrolysis (HTSE) hydrogen-to-ammonia case. Hydrogen for ammonia synthesis is produced by HTSE. Nuclear HTGRs supply heat and power for the electrolysis units as well as power for the ASU. An opportunity was identified to use heat from the natural gas burner as topping heat for the electrolysis unit. As HTGR technology matures and reactor outlet temperatures increase, the nuclear reactors may be able to supply electrolysis topping heat. However, because of the upper limit of 700 °C deliverable heat from the currently proven class of HTGRs, topping heat from the burner to the electrolyzers is an attractive means of increasing electrolyzer efficiency and leads to a small penalty on the overall carbon footprint.

Urea production requires CO₂ as a feed to the process. For this case, a CO₂ supply is not readily available and must be produced to support urea manufacture; hence, natural gas is burned with oxygen provided from the ASU to produce the required CO₂. If a source of CO₂ is available from a carbon capture plant, or a biological ethanol fermentation plant, then the life-cycle carbon footprint will be lower.

Process modeling results

Based on the results for the conventional coal-to-ammonia case, a need for high-temperature (> 300 °C) heat integration was not identified for the coal-fed case. In addition, because light water reactors can supply power in this scenario, a strong need for integration with advanced nuclear reactors is not apparent. Hence, modeling of a nuclear-integrated coal-to-ammonia case was not pursued by Wood (2010). Other gasification technologies (hydrogasification) could be considered, and these technologies may be more complementary for high temperature nuclear reactors.

Analysis of the conventional natural-gas-to-ammonia case indicates a strong potential heat integration opportunity for a HTGR. In the conventional case, 22.7% of the natural gas feed to the process is burned to provide heat in the primary reformer. The operating temperature in the primary reformer is 790 °C, which is somewhat higher than the assumed 700 °C temperature that can be supplied by the current generation HTGR. Analysis seems to indicate, however, that primary reforming can be accomplished in two

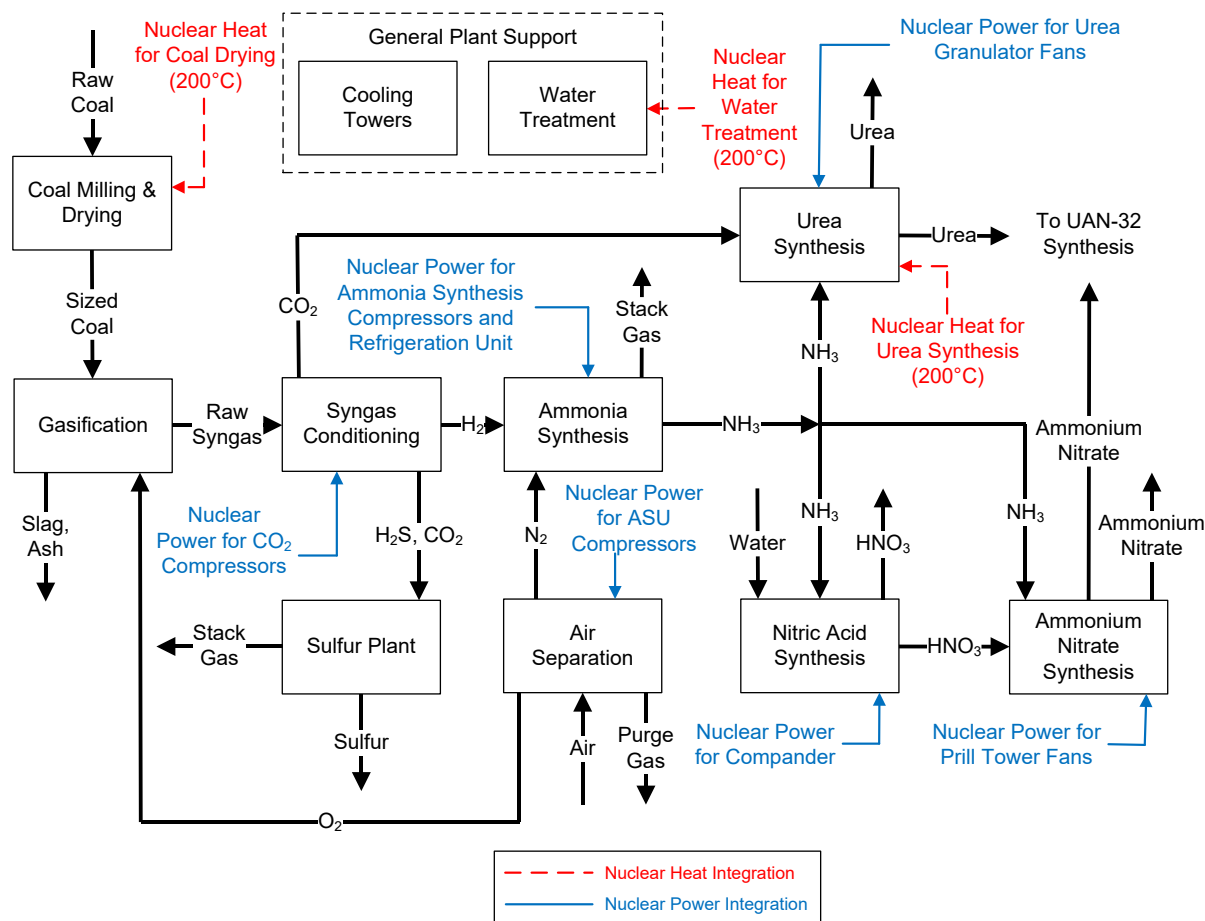


Fig. 4 Block flow diagram for the nuclear-integrated coal-to-ammonia process. Next Generation Nuclear Plant Wood RA (2010) *Nuclear-Integrated Ammonia Production Analysis*. Technical Evaluation Study-666. Project 23843.

stages—the first using nuclear heat at 690 °C, followed by a second stage reformer fueled entirely by purge gas that operates at a somewhat higher temperature.

Fig. 6 graphically presents a high-level material and energy balance summary for each case (Nelson, 2010). Results for the nuclear-integrated natural gas-to-ammonia case look promising. One 450 MW_t HTGR would be required to support this configuration. The split of heat to power required from the nuclear plant is 42%/58%. By substituting nuclear heat for natural gas combustion in the primary reformer, natural gas consumption can be reduced by 18%. In addition, CO₂ emissions can be reduced by 70%.

Although not shown previously, a case for producing nitrogen by burning hydrogen to deplete the oxygen in air was also evaluated by Wood (2010). The block flow diagram for this case was not presented because the cost of capital was significantly higher. This is reflected in the number of HTGR units required to support such an option.

Results for the HTSE hydrogen/ASU case look promising. CO₂ emissions are cut to very low levels, and the need to sequester carbon is eliminated. In addition, natural gas consumption is reduced to only the amount required to supply carbon for urea production. This option requires roughly four 600 MW_t HTGRs to fulfill the total electrical and heat duties required. This leaves surplus electricity for sale to the grid; alternately, the size of the hydrogen plant can be scaled up to match the energy output of all four HTGE units.

Economic comparison

The economic viability of the ammonia processes was assessed by Wood (2010) using standard economic evaluation methods. The economics were evaluated for the conventional and nuclear-integrated options described in the previous sections. The total capital investment based on the total equipment costs, annual revenues, and annual manufacturing costs were first calculated for the cases. For purposes of comparing the various options, the cost of producing urea is provided relative to the price of natural gas and the cost of the capital and operating costs of the plant.

The cost of one 600 MW_t HTGR in 2010 was estimated to be 2000 \$ kW⁻¹ installed (Nelson, 2007). Projections for an nth-of-a-kind target were estimated to drop to \$1400 \$ kW⁻¹ (Wood, 2010). Over the past decade, several reactor developers have

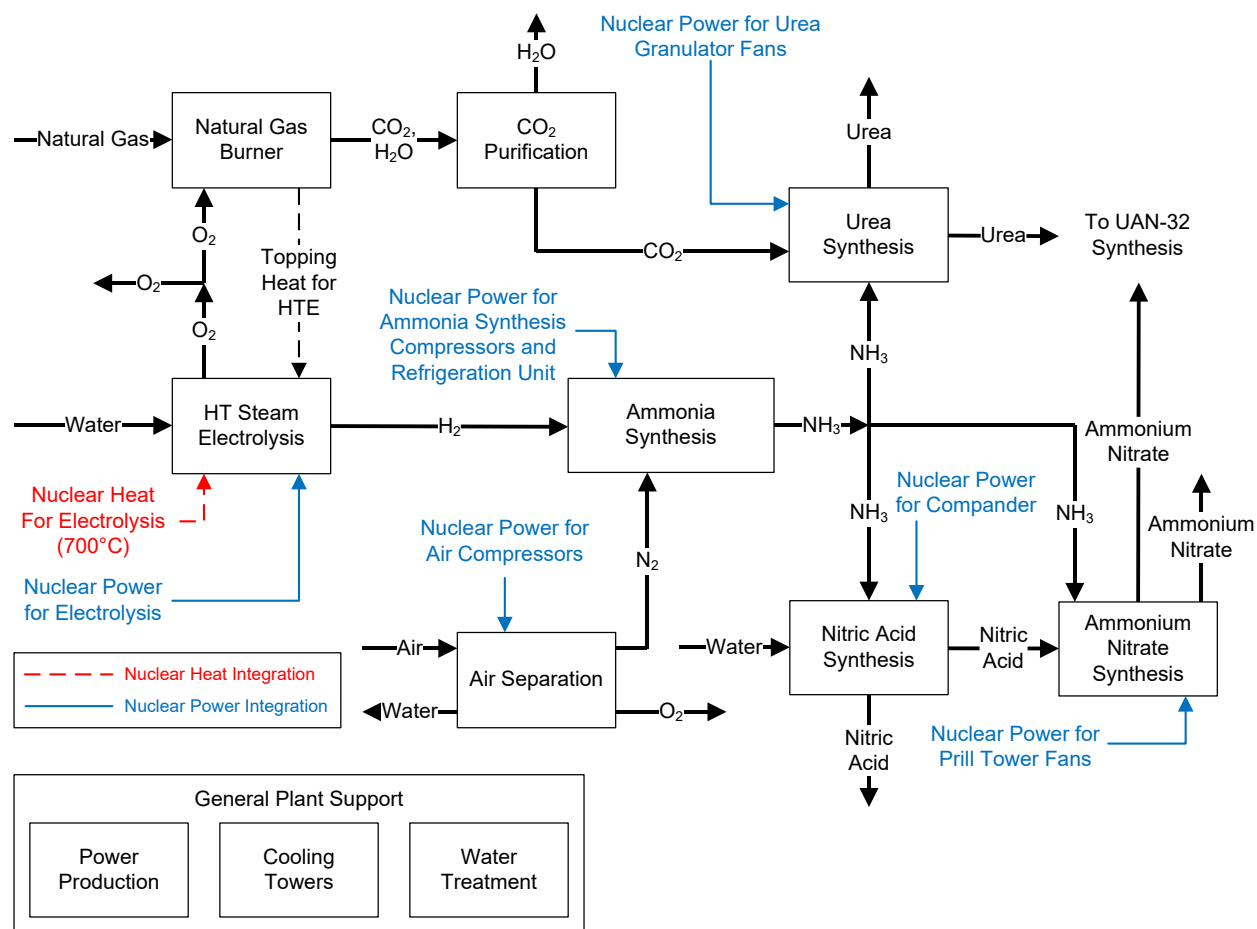


Fig. 5 Block flow diagram for the HTSE hydrogen-to-ammonia process. Next Generation Nuclear Plant Wood RA (2010) *Nuclear-Integrated Ammonia Production Analysis*. Technical Evaluation Study-666. Project 23843.

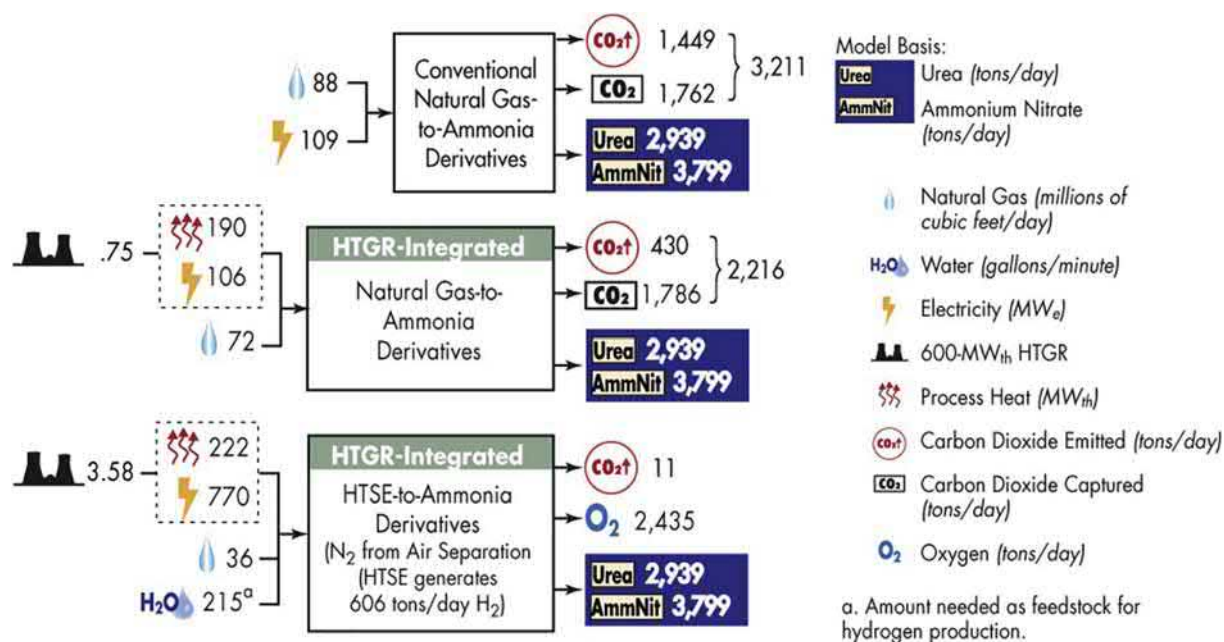


Fig. 6 Ammonia modeling case material balance summary. Next Generation Nuclear Plant Nelson L (2010) *Integration of High-Temperature Gas-Cooled Reactors into Industrial Process Applications*. INL/EXT-09-16942.

Table 1 Urea production economic results summary for a 12% IRR.

Option	Urea price (\$ t ⁻¹)					
	No CO ₂ tax			50 \$ t ⁻¹ CO ₂ tax	100 \$ t ⁻¹ CO ₂ tax	200 \$ t ⁻¹ CO ₂ tax
	Low Nat. gas	Mid. Nat. gas	High Nat. gas	Low Nat. gas	Low Nat. gas	Low Nat. gas
Conventional gas-to-ammonia	244	272	348	274	304	363
HTGR integrated, gas-to-ammonia, 2000 (\$ kW ⁻¹) HTGR	263	286	348	293	304	345
HTGR integrated, gas-to-ammonia, 1400 (\$ kW ⁻¹) HTGR	249	272	335	270	291	332
HTGR integrated, gas-to-ammonia, 700 (\$ kW ⁻¹) HTGR	236	259	321	257	277	318
HTSE integrated, gas-to-ammonia, N ₂ from ASU, 2000 (\$ kW ⁻¹) HTGR	493	497	505	497	500	511
HTSE integrated, gas-to-ammonia, N ₂ from ASU, 1400 (\$ kW ⁻¹) HTGR	429	433	441	432	438	447
HTSE integrated, gas-to-ammonia, N ₂ from ASU, 700 (\$ kW ⁻¹) HTGR	365	369	377	368	374	377
Urea price range in DTN (2020)	225–275			255–305	285–335	345–395

Green shading indicates nuclear assisted options that are closely competitive with conventional urea plants and recent market prices.

proposed concepts for advanced reactors, including new designs for the HTGR. Molten-salt reactors that operate in a comparable temperature regime are also being developed by private industry. Hence, for comparison, three levels of capital costs associated with the attendant nuclear plant are included here; 2000, 1400, and 700 (\$ kW⁻¹) installed capacity.

The costs of urea production based on a detailed discounted cash flow analysis are tabulated in **Table 1** for the following assumptions:

- Debt to equity ratios equal to 80%/20%, with a discount rate on the debt of 8%, and an internal rate of return on investment (IRR) of 12% on the equity.
- 15-year debt payoff and project life of 30 years.
- Combined (effective) tax rate of 38.9%.
- Inflation rate of 2.5% for the life of the project.
- Operating on-line capacity of 92%.

Natural gas prices beginning at the start of the project (prior to inflation) were varied to account for the large fluctuations seen in the market. The average cost of industrial market prices of natural gas for the life of the project are low (4.7 \$ GJ⁻¹), (mid-term 6.50 \$ GJ⁻¹), and high (12.00 \$ GJ⁻¹).

Based on the results for the economic evaluation of Wood (2010), integration of nuclear heat into a gas-to-ammonia process appears to become competitive once the cost of the HTGR reaches the target of \$1400 (\$ kW⁻¹) installed thermal capital cost. Although the HTGR/HTSE integrated flowsheets offer even greater benefits in terms of reduced natural gas consumption and carbon emissions, this case only is only shown to be competitive under the following conditions: (1) the capital costs of the nuclear reactor are reduced below 700 (\$ kW⁻¹) installed thermal capital costs, (2) the average price of natural gas prices are relatively high (viz., > 12.00 \$ GJ⁻¹) and (3) credits for avoiding CO₂ emissions are high (here above 200 \$ t⁻¹).

Conclusions

Ammonia production, including its attendant derivative fertilizers (urea and ammonium nitrate), is one of the largest industries worldwide. This Chapter Section shows nuclear energy integration with ammonia production is technically feasible and economically viable. High temperature reactor heat can be substituted for the heat used to reform natural gas. When combined with the direct use of electricity for the compression duties and pressure exergy conservation with application of companders, GHG emissions of a conventional process can be reduced to near zero. For the assumptions invoked in reference study, indirect integration of hydrogen and nitrogen with a Haber-Bosch process is becomes competitive when the cost of the nuclear reactor falls to around 700 \$ kW⁻¹ and there is a carbon emissions offset of around 200 \$ t⁻¹.

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Hydrogen-Based Flash Ironmaking Technology (HyFIT): A Novel Green Ironmaking Technology With Low Energy Consumption

Hong Yong Sohn, Department of Materials Science & Engineering, University of Utah, Salt Lake City, UT, United States

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Glossary

Computational fluid dynamics (CFD) Pertaining to computer-based, quantitative description of a flow system.

Concentrate The product of ore dressing operations in which unwanted minerals are largely removed from the ores collected from mining operations before further treatment to obtain products. Iron ore concentrate typically contains 0.02–0.06 mm iron oxide particles.

Flash reaction Reaction of fine solid particles with a gas in a particle-gas flow stream, usually at a high temperature.

Residence time The average amount of time that a portion of material spends in a reactor while flowing through it.

Introduction

Ironmaking is an energy-intensive process and is thus one of the largest emitters of carbon dioxide. The conventional ironmaking technology is asymptotically approaching its practical efficiency limits. This means that even if we invest in the most modern facilities, only marginal energy and emissions reductions would be achieved. Therefore, the steel industry must develop a new technology to produce iron. An ideal process would replace the blast furnace and coke oven to reduce energy requirements and greenhouse gas emissions.

This article describes the development of a novel Flash Ironmaking Technology, conceived by (Sohn, 2007) that is based on the gaseous reduction of iron oxide concentrates in a flash reactor. It takes advantage of the large specific area of fine particles of iron ore concentrate. The development of the novel process also contains two of the most critical components for the future of metal production in general and steelmaking in particular, i.e. CO₂ emissions and energy saving. The steel industry contributes around 6–7% of the total anthropogenic emissions of CO₂ (Tacke and Steffen, 2004). The technology described here has the potential to significantly reduce energy consumption by 57% and lower carbon dioxide emissions in the ironmaking step down to a negligible level of just 4% of the level from the blast furnace process, when hydrogen is used as the fuel/reductant and that hydrogen is made by non-emitting processes, such as nuclear-driven electrolysis.

More than 90% of iron is currently produced by the blast furnace process, while the balance is produced by the direct reduction processes (IISI, 2007). The blast furnace process requires the feed in the form of sinters or pellets and coke made from high-grade coking coal. These two steps consume much energy and are problematic from the environmental perspective.

Previously developed processes for the gaseous reduction of iron oxide can be grouped into two broad types: shaft furnaces (Midrex and Energiron; Lockwood Greene Technologies, 2000) and fluidized-bed reactors (FINMET and the earlier FIOR (Brent et al., 1999), CIRCORED (Husain et al., 1999), and SPIREX (Macauley, 1997)). Most of these processes, however, are not sufficiently intensive to replace the blast furnace because they cannot be operated at high temperatures due to the sticking and fusion of particles. The shaft furnace processes, which currently are the dominant types for alternate ironmaking, require pelletization of iron oxide concentrate accompanied by additional cost and environmental problems, and also frequently suffer from pellet disintegration and sticking problems.

Therefore, a new technology named the Flash Ironmaking Technology (FIT) has been developed at the University of Utah for producing iron directly from fine concentrates by a flash smelting process. This is accomplished by adopting a reductant gas such as natural gas, hydrogen, syngas, or a combination thereof, while bypassing pelletization/sintering and cokemaking required

in typical ironmaking processes (Sohn, 2007; Sohn and Choi, 2010; Sohn et al., 2009a,b). Unlike other gas-based alternative ironmaking processes based on shaft furnaces or fluidized-bed reactors, the FIT will not suffer from the problems that plague other processes when operated at high temperatures, mainly from the sticking and fusion of particles. Low-temperature disintegration problems encountered in the processes using pellets are also eliminated. The availability of large quantities of iron ore concentrate produced in the United States and elsewhere will also be a backbone for this technology. Up to 70% of the US iron production is based on taconite concentrate (50–55 million tons/year) (USGS, 2007) with particle sizes of $< 100 \mu\text{m}$.

Below, we will present the description of the novel ironmaking process and the results of the operation of a pilot plant together with work on flow sheet development, process simulation, and economic feasibility analysis.

Description of the flash ironmaking technology (FIT)

A schematic diagram of the Flash Ironmaking process is given in Fig. 1. Hydrogen or natural gas is partially oxidized with tonnage oxygen through a burner on top of the flash reactor to generate a reducing gas stream at 1600–1900 K. Iron ore concentrate is injected near the burner and reduced in the reactor. Although this figure shows the operation of the process in which the reduced iron is collected as a molten bath for direct steelmaking in the unit, iron may also be collected as solid product to be used in steelmaking furnaces (Sohn, 2007).

For annual production of 20 million tons of iron, which amounts to about 40% of the current U.S. production rate, the FIT would require 2 million tons/year of hydrogen, making steel industry a large potential consumer of hydrogen.

The FIT will remove many of the shortcomings suffered by other alternative processes. Specifically, it will be operated with (a) direct use of iron oxide concentrate without the need for pelletization or sintering; (b) no cokemaking required; (c) high temperature conditions with no pellet sticking or disintegration problems; (d) low refractory problems; and (e) ease of feeding the raw materials.

The development of the FIT started with the verification of sufficient kinetic feasibility considering the fact that there are only a few seconds of residence time available in a typical flash reactor. The results of the kinetic feasibility work have been reported

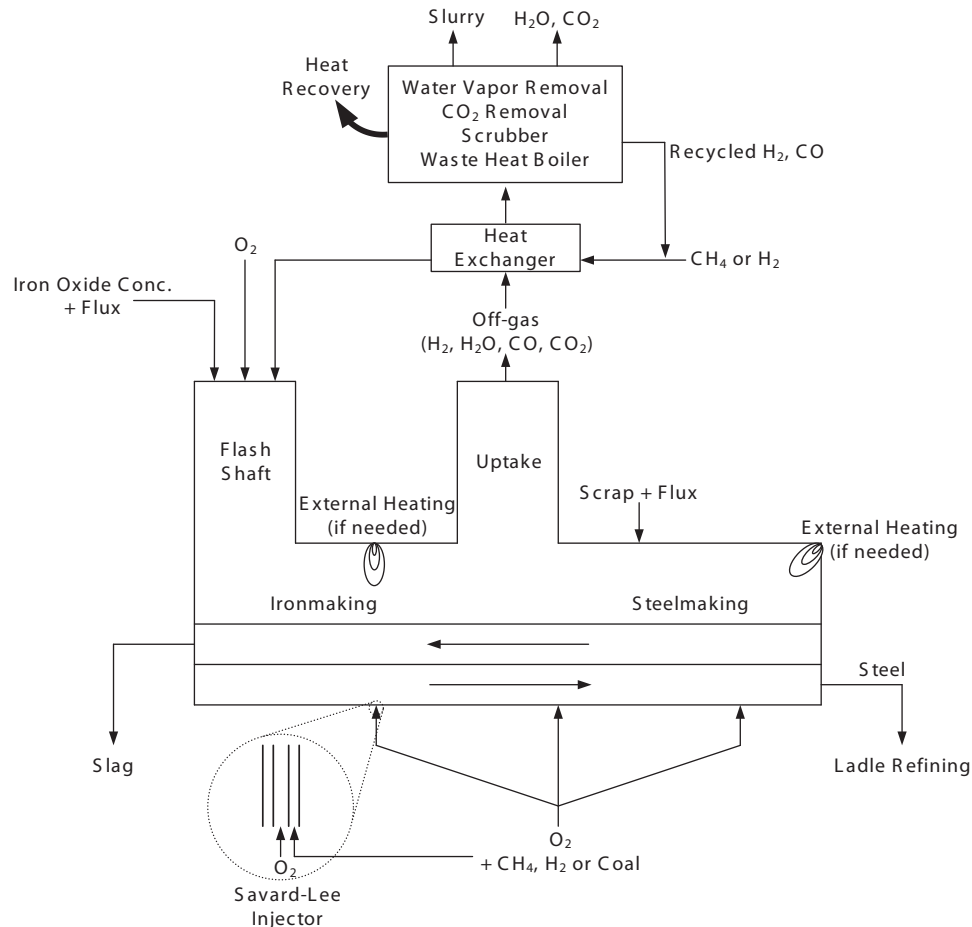


Fig. 1 A sketch of a direct steelmaking process based on the Flash Ironmaking Technology (FIT).

in detail and the reader is referred to the literature (Chen et al., 2015; Elzohiery et al., 2017, 2021; Fan et al., 2016a,b, 2017, 2021; Wang and Sohn, 2015). Further, a laboratory flash reactor was operated in preparation of the construction and operation of a pilot-scale reactor, as described elsewhere (Elzohiery et al., 2020; Fan et al., 2016b; Sohn et al., 2009a), and a comprehensive computational fluid dynamics (CFD) model of the flash reactor operation was carried out (Elzohiery et al., 2020; Fan et al., 2016b). Upon the establishment of the kinetic feasibility and based on the success with the laboratory flash reactor as well as the CFD simulator development, a pilot-scale reactor test program was carried out, as described below.

Operation of a pilot-plant scale flash reactor

Here, we will describe a Pilot Flash Reactor operated at 1200–1600 °C with a concentrate feeding rate of 1–7 kg/h, shown in Fig. 2. This reactor was the first flash ironmaking reactor where the heat and reductant are produced by partial oxidation of hydrogen or natural gas with industrial oxygen.

Facility

The Pilot Flash Reactor consisted of a reactor vessel, a vessel roof, burners, a quench tank, off-gas piping, a flare stack, an off-gas analyzer, a gas valve train, a water cooling system, gas leak detectors, a concentrate feeding system, and human machine interface. Fig. 3 shows the main components of the reactor body.

The Human Machine Interface (HMI) consisted of the main programmable logic controller (PLC) and a computer. The main PLC was connected to all the different parts of the system and to the computer where the operator could monitor all the different parts and run the reactor. The programming in the main PLC was responsible for all the safety and emergency steps.

During the preheating stage, natural gas was burned with industrial oxygen to preheat the reactor at a ramp rate of 90–95 °C/h. The operating temperature range was 1200–1550 °C. During the actual operation in which iron ore concentrate was fed, natural gas and oxygen were fed at a pre-determined ratio through the main burner to start the partial combustion, which provide the heat needed to maintain the process temperature, and generate reducing gases H_2 and CO. Although the laboratory flash reactor tests indicated that operation with hydrogen is very similar to that with natural gas (Elzohiery et al., 2020; Fan et al., 2016b; Sohn et al., 2009a), the pilot reactor was operated only with natural gas for ready availability and low costs. Fig. 4 shows a screen shot that displayed and controlled all the parameters of the reactor operation. Further details of the operations of this pilot plant facility is available elsewhere (Elzohiery, 2018).



Fig. 2 The pilot plant with a Flash Reactor installed at the University of Utah.

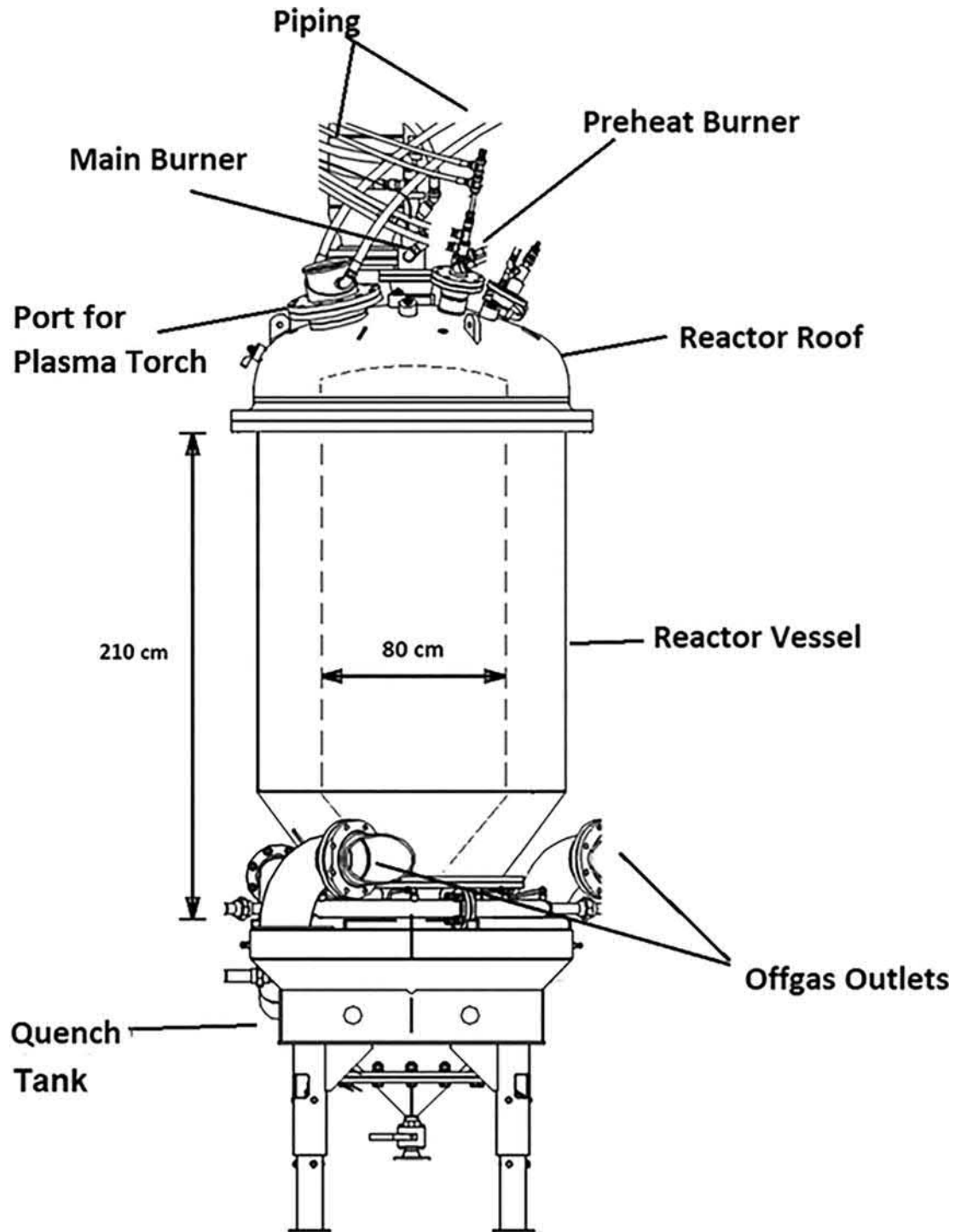


Fig. 3 Schematic diagram of the Pilot Flash Reactor.

Results from pilot flash reactor runs

The results of the Pilot Flash Reactor runs are expected to help in designing the industrial reactor, including the identification of the difficulties and how to improve the various aspects of operation. This reactor was simulated by CFD to optimize the operating conditions to achieve a high reduction degree at the optimum conditions and reactor dimensions to be used in an industrial reactor (Abdelghany et al., 2019). Table 1 shows the results of the runs performed in the reactor using industrially produced, as-received magnetite concentrate with a particle size range less than 90 μm .

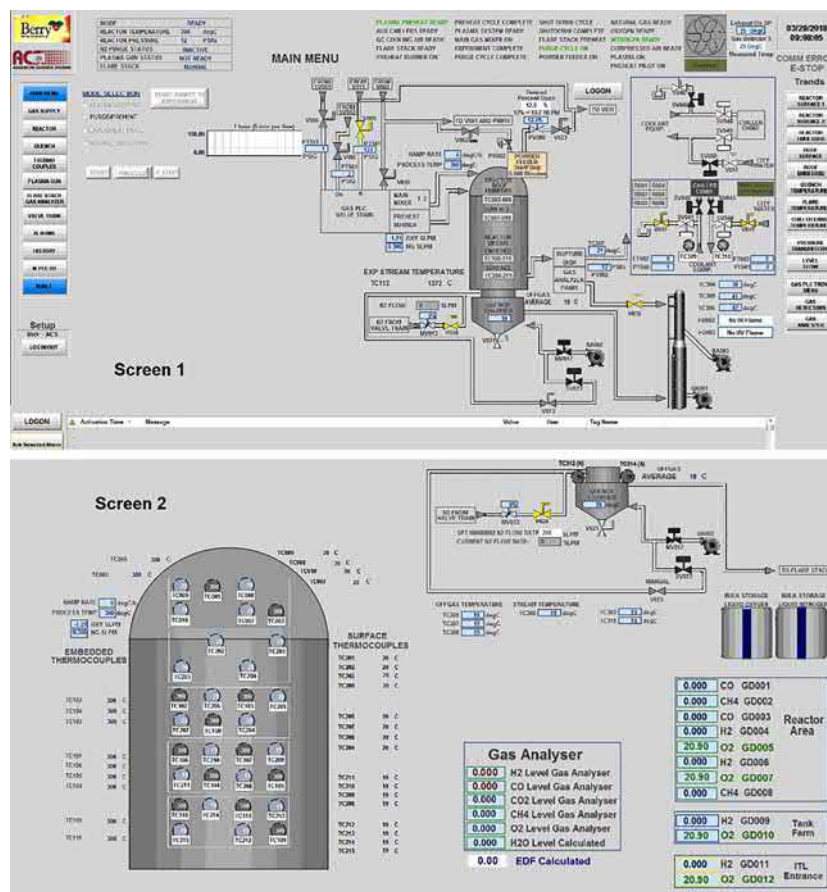


Fig. 4 The controlling screen for the HMI for the pilot flash reactor.

Different experimental runs were made in this reactor to yield a wide range of reduction degrees deliberately at less than complete reduction to better examine the effects of the operating conditions and validate the CFD model under different conditions. The experiments in the pilot plant were performed at different temperatures and reducing powers of the gas with the aim of obtaining sufficient data to design the industrial flash reactor. In order to represent the data obtained from the reactor, the nominal residence time for the particles was calculated based on the total volumetric flow rate of the gases at the experiment temperature and the total volume of the reactor vessel which was 1.01 m^3 . When the velocities of the gases and the particles at the injection points are taken into consideration, the actual residence time would be shorter.

The results showed good reproducibility within $\pm 5\%$ of the average reduction degree. This represents a very high degree of reproducibility, considering the complexity of the operation and design of this large unit.

Flow sheet development, process simulation, and economic feasibility

Based on the potential advantages discussed above and the results of the process feasibility studies, detailed flow sheets for different versions of the new process depending on the fuel type and economic analysis have been developed by Pinegar et al. (2011, 2012, 2013a,b).

Energy requirement for the standard case operation compared with that of the average blast furnace process, assuming that hydrogen and coking coal were the starting raw materials, indicated that the proposed process would consume 57–60% less energy. However, it is noted that the energy for hydrogen or coal production must be added for a comprehensive comparison of the energy requirement. An economic feasibility analysis was also performed (Pinegar et al., 2011, 2013b).

The flow sheet development and process simulation as well as economic analysis for the process based on natural gas is available elsewhere (Pinegar et al., 2012, 2013a,b). Here, those analyses for the process using hydrogen (HyFIT) will be summarized (Pinegar et al., 2011).

Table 1 The results of runs performed in the Pilot Flash Reactor.

Inner wall temperature (°C)	Magnetite concentrate feeding rate (kg/h)	Gas flow rate (SLPM) ^a		H ₂ excess driving force (EDF)	Nominal residence time (s)	Reduction degree (RD) (%)
		Main burner				
		NG (SLPM)	O ₂ (SLPM)			
1200–1130	5.0	404	321	0.76	12.5	65
1290–1220	1.8	410	293	0.84	12.0	79
1290–1210	2.9	410	293	0.96	12.0	82
1290–1230	2.5	358	270	1.00	13.3	83
1290–1240	3.5	512	327	1.07	10.2	76
1330–1230	4.7	330	200	1.36	15.3	89
1330–1230	4.5	330	200	1.44	15.3	87
1330–1230	5.2	500	290	3.00	10.6	80
1330–1230	4.3	500	290	3.00	10.6	82
1355–1260	5.5	235	190	0.03	18.3	7
1350–1300	4.0	255	209	0.15	17.0	49
1350–1270	4.5	275	212	0.20	16.2	31
1340–1280	5.0	280	209	0.21	16.2	37
1350–1290	4.6	280	230	0.50	15.6	80
1400–1300	6.3	300	240	0.82	14.4	88
1400–1300	5.0	330	200	1.51	14.6	100
1415–1350	4.5	220	191	0.07	18.0	18
1410–1360	4.0	240	195	0.33	17.1	32
1410–1330	5.0	295	221	0.50	14.7	66
1410–1330	6.0	300	210	0.70	14.9	74
1410–1320	5.0	300	210	0.82	14.9	82

^aThe flow rates of NG and O₂ in the pilot burner were 9.6 and 37.6 SLPM, respectively. The flow rate of N₂ in the powder feeder was 10.7 SLPM.

Material and energy balances

Tables 2 and 3 summarize the material and energy balances for one-step and two-step processes operated with the ironmaking reactor temperature of 1500 °C as the standard case. Hydrogen preheating temperature was set to 900 °C for the one-step process, and 750 °C for the two-step process. Material and energy flows for both processes at the standard case with all the stream values and conditions can be found elsewhere (Pinegar et al., 2011).

In this work, hydrogen and coking coal were chosen to be the starting raw material of fuel/reductant reagent for the energy requirement comparison. However, as for the comparison of energy consumption, it is noted that a more comprehensive evaluation of the energy requirement of the proposed process and the blast furnace is complicated and depends on the choice of starting raw materials. Depending on the starting material, the energy required for raw material extraction should also be included.

When hydrogen is used as the starting fuel/reductant material, energy requirements are 5.48 and 5.03 GJ/ton of iron, respectively, for the one-step and two-step processes (HyFIT). These energy requirements were obtained using the higher heating value of hydrogen used as the fuel and reductant. Compared with the average blast furnace process assuming that coking coal is the starting material, the proposed processes consume 57% and 60% less energy, respectively.

In the proposed process, flux calcination was assumed to be performed separately before the flux was fed to the reactor so that the reactor off-gas would not be diluted by carbon oxide gases.

The energy requirement was examined at different operating conditions. When the ironmaking reactor operating temperature rises to 1600 °C, energy requirements based on the higher heating value of hydrogen become 5.86 and 5.32 GJ/ton of iron for the one-step and two-step ironmaking processes, respectively.

Although the energy requirement of the proposed process HyFIT based on purchased hydrogen was estimated to be much smaller than the average blast furnace process, even with a higher operating temperature and heat loss, a comprehensive comparison of energy requirement including the energy consumption for raw material extraction such as coal mining, natural gas pumping, is very complicated due to diversity of operation and processes and thus beyond the scope of this work.

Note on “Energy Requirement”: It must be kept in mind when comparing the “energy intensity values” or “energy requirements” to clearly understand the energy items included in these values (Sohn and Olivas-Martinez, 2014). There are currently different approaches for selecting energy items to include in the overall “energy requirements” when an input fuel also serves as a reactant.

For a process involving heat generation from fuel combustion, the choice of which endothermic reactions to select to compute the energy requirement can cause confusion. The essential question is whether the heating value of a reactant that is also burned to generate process heat should be added as an item in the “energy requirement.” Some investigators include this item (Fruehan et al., 2000; Remus et al., 2013; Stubbles, 2000) and others do not (Burgo, 1999; Pinegar et al., 2011, 2012, 2013a). The presented “energy

Table 2 Material balance for commercial-scale one-step and two-step ironmaking processes (HyFIT) with the operating temperature of 1500 °C and the hydrogen preheating temperature of 900 °C in the one-step process and 750 °C in the two-step process, compared with that for the average blast furnace process.

		<i>One-step process</i>	<i>Two-step process</i>	<i>BF process (Choi, 2010)</i>
Input (tons/ton of iron)	Hydrogen	0.10 ^a	0.08 ^a	
	Oxygen	0.43	0.30	0.68
	Iron ore	1.46	1.46	1.43
		(magnetite)	(magnetite)	(hematite)
	Flux	0.06	0.06	0.29
	Nitrogen			2.21
	Coke			0.46
	<i>Total-ironmaking</i>	<i>2.05</i>	<i>1.90</i>	<i>5.07</i>
	Cooling water for scrubber	17.9	15.6	
Output (tons/ton of iron)	Iron	1.0	1.0	1.05 (with 4.5% C)
	Slag	0.14	0.14	0.21
	Condensed water vapor	0.71	0.63	
	Total discharged gas	0.2	0.13	3.81
	Carbon dioxide			1.60 ^b
	Nitrogen			2.21
	<i>Total-ironmaking</i>	<i>2.05</i>	<i>1.90</i>	<i>5.07</i>
	Cooling water in slurry	17.9	15.6	22.3 ^c
	CO ₂ from CaCO ₃ and MgCO ₃ calcination	0.04	0.04	

^aThis number does not include the carbon dioxide emission from ore/coke preparation.

^bThe actual amount may be slightly higher if heat loss from the hydrogen recycling step is considered.

^cWater requirement for US Steel's Fairfield No. 8 Blast Furnace (production rate: 5500 tons/day) (United States Steel Corporation, 1985; Camlic and Goodman, 1997) including water for top gear, stack plates, bosh channels, tuyere jacket, tuyeres and tuyere coolers, hearth and subhearth, and stove valves, as the reference years of water usage is older than that of the production rate, this number may be somewhat larger than the number for a modern plant.

requirement" varies with the selected approach. The choice is arbitrary, but one should indicate explicitly which approach is used in showing the energy calculations, especially in comparing different processes.

The net differences in the energy requirements between technologies are not strongly affected by the selection of calculation approaches if the same method is used for different technologies, but the consumption values for individual processes themselves depend on them. Thus, whether the heat of combustion of a reactant that can also be burned is included in the energy requirement should be clearly indicated. It will make it much more definitive to indicate the amounts for "process energy" that contains only the heating value of the fuel and "reductant energy" ("feedstock energy" in petrochemicals production) that includes the heating value of the substance serving as a reactant.

For detailed description of these different calculation methods and discussion on the subject, the reader is referred to (Sohn and Olivas-Martinez, 2014). An additional caution that follows from this consideration is that the comparison of different ironmaking processes for their energy requirements should only be done with the values obtained using the same method by the same investigators. In other words, the energy requirement value obtained by one investigator for a certain ironmaking process should not be compared with a value obtained by another investigator for a different ironmaking process without carefully checking the bases of the calculations.

The energy values presented above in this article are based on the "process energy," i.e. they do not include the heat of combustion of the reductants that react with iron oxide.

Summary

The use of hydrogen in the novel ironmaking process would greatly reduce carbon dioxide emissions and energy consumption as well. Conversely, the steel industry is a huge potential consumer of hydrogen from the viewpoint of a developing hydrogen economy. Based on the tremendous scale of demand for steel, it appears that the steel industry and the development of hydrogen production would be a great match, especially when hydrogen is produced from carbon-free sources of energy such as nuclear or solar energy.

This article describes the development of a novel ironmaking technology based on flash reaction, which is expected to require up to 60% less energy in the ironmaking step with little carbon dioxide emissions when operated using hydrogen (HyFIT). Process flow sheets for the commercial-scale novel suspension ironmaking process based on purchased hydrogen were constructed and simulated. The proposed process was modeled in two configurations: one-step and two-step processes in which the number of iron ore reduction reactor is different. Simulation was performed under various operating conditions to examine the effect of operating conditions on the fresh hydrogen amount which would affect the economic feasibility greatly due to expensive hydrogen price.

Table 3 Energy balance for commercial-scale one-step and two-step ironmaking processes (HyFIT) with the operating temperature of 1500 °C and the hydrogen preheating temperature of 900 °C in the one-step process and 750 °C in the two-step process compared with that for the average blast furnace process.

		One-step process	Two-step process	BF process ^a
INPUT (GJ/ton of iron)	Fuel combustion	8.28 ^b	6.02 ^b	8.33
	Heat recovery	−2.80	−0.99	−1.32 ^c
	Sub-total	5.48	5.03	7.01
	Energy input for ore/coke preparation			5.68
	Total-ironmaking ^d	5.48	5.03	12.7
OUTPUT (GJ/ton of iron)	Energy input for CaCO ₃ and MgCO ₃ calcination	0.26 ^e	0.26 ^e	
	Reduction	0.91	0.91	2.09
	Slag making			0.16
	Sensible heat of iron	1.27	1.27	1.35
		(1500 °C) ^f	(1500 °C) ^f	(1600 °C)
	Sensible heat of slag	0.24	0.24	0.47
		(1500 °C) ^f	(1500 °C) ^f	(1600 °C)
	Slurry (H ₂ O(l))	1.93	1.69	
	Sensible heat of off-gas			0.26
				(90 °C)
	Removed water vapor	0.01	0.006	
	CaCO ₃ decomposition			0.33
	Heat loss in the reactor	0.78	0.78	2.6
	Heat loss in the heat exchanger	0.34 ^g	0.14 ^g	0.07 ^c (heat loss in heat recovery)
	Sub-total	5.48	5.03	7.01
	Pelletizing			3.01 (Stubbles, 2000)
	Sintering			0.65 (Stubbles, 2000)
	Cokemaking			2.02 (Stubbles, 2000)
	Total-ironmaking ^d	5.48	5.03	12.7
	CaCO ₃ and MgCO ₃ decomposition at 1000 °C	0.26 ^e	0.26 ^e	

^aEnergy balance was calculated by METSIM based on the material balance.

^bFuel combustion energy generated in the process was calculated by subtracting the energy used for iron ore reduction from the balance of heats of formation of all output components from the entire system and input components into it. The higher heating value of natural gas was used for this calculation. If the lower heating value is used, the total energy requirements for the one-step and two-step processes will decrease, respectively, to 3.74 and 3.49 GJ/ton of iron.

^cHeat loss in the heat recovery step was estimated to be 5% of the energy difference between the hot off-gas (calculated to be 426 °C) and cooled reactor off-gas (90 °C), and the rest was estimated to be the recovered energy.

^dHydrogen and coking coal are considered as the starting raw materials. This energy requirement comparison does not include electrical power required for the plant operation.

^eThe energy requirement for CaCO₃ and MgCO₃ calcination to generate CaO and MgO for the flux was calculated by METSIM. Calcination was assumed to be done at 1000 °C separately from the ironmaking process.

^fIf the temperature of iron and slag is replaced by 1600 °C (same as the blast furnace process), the total energy for the one-step and two-step processes increases to 5.86 and 5.32 GJ/ton of iron, respectively, by the increased sensible heat of iron and slag.

^gHeat losses in a heat exchanger were assumed to be 5% of the total sensible heat of input streams.

Energy requirement for the standard case operation compared with that of the average blast furnace process, assuming that hydrogen and coking coal were the starting raw materials, indicated that the proposed process HyFIT would require 57% and 60% less energy, respectively, for the one-step and two-step processes. However, it is noted that the energy for hydrogen or coal production must be added for a comprehensive comparison of the energy requirement.

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Calcination

Nils H. Haneklaus, Danube University Krems, Krems an der Donau, Austria; and Freiberg University of Mining and Technology, Freiberg, Germany

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Introduction

Research on solar and nuclear high-temperature ($> 600\text{ }^{\circ}\text{C}$) process heat applications is largely concerned with the production of hydrogen (Angulo et al., 2012; Barlev et al., 2011; Forsberg, 2020; Kuhr, 2008; Nowotny et al., 2005). Green hydrogen production can be considered a key process, since the hydrogen can again be converted into synthetic fuels that can be used without having to change a large part of the present fossil fuel dominated energy infrastructure. Using high-temperature solar and nuclear heat for mineral processes such as calcination may also be attractive though, as this practice has the potential to mitigate fuel combustion altogether and can dramatically increase energy efficiencies if compared to using synthetic fuel from green hydrogen. Besides, mineral products just like hydrogen can be produced with excess power from solar or nuclear power plants.

A significant mineral process that could profit from solar and nuclear process heat is calcination. Mineral calcination that is for instance essential for cement production, accounts for some 5% of all anthropogenic carbon dioxide (CO_2) emissions per year (Worrel et al., 2001). It is an energy intensive service that is notoriously challenging to provide without adding CO_2 to the atmosphere (Davis et al., 2018). More than half of the CO_2 released during mineral calcination results from burning fossil fuels for heat generation, while the other half is a product of the calcination reaction itself. The calcination reaction is provided below for limestone (Latin: calcinare = to burn lime) that is calcinated for cement production.



It is worth noting that the calcium oxide (CaO) is not stable and cement hydration products reabsorb CO_2 from the atmosphere over time in a process called carbonation. Xi et al. (2016) estimate that carbonating cement materials reabsorb a bit more than 20% of the initial direct CO_2 emissions from calcination and heat generation that amounted to some 2.5% of global anthropogenic CO_2 emissions in 2013. If it would be possible to further eliminate direct CO_2 emissions during calcination, carbonating cement materials would not only reabsorb one fifth of their CO_2 emissions but could act as a substantial global CO_2 sink.

Direct solar calcination

Concentrated solar power plants can reach the high temperatures ($550\text{--}1150\text{ }^{\circ}\text{C}$) required for the calcination of lime. At the World Fair of 1904 in St. Louis (MO, United States) Gomes Himalaya presented a prototype of an invention he patented 3 years earlier (Gomes Himalaya, 1901) that concentrated solar radiation for industrial use. He melted metals in front of spectators and received several awards for his invention (Rosa, 2019) that is depicted in Fig. 1.

Several research groups are actively working on solar calcination using concentrated solar radiation just as Gomes Himalaya proposed. Flamant et al. (1980) presented several solar reactor designs suitable for solar decarbonization of lime (CaCO_3). More recently, Costa Oliveira et al. (2019) conducted experiments proving that solar calcination is indeed possible despite the low solar absorptance of the mineral feed that make it challenging to realize large mineral throughputs. Costa Oliveira et al. (2019) envision a rotary kiln furnace similar to the ones used for fossil fuel powered calcination but heated by the power of the sun using concentrated solar radiation only. Such solar rotary kiln designs have been reviewed for different industrial processes by Alonso et al. (2017). Particularly comprehensive experiments were conducted by Meier et al. (2004, 2005a,b, 2006) with an indirectly heated multitube rotary kiln calcination furnace. Besides technical feasibility studies an economic evaluation (Meier et al., 2005b) was conducted that indicate that solar calcination, as a result of the relatively low material throughput, but the high quality of the final product, could supply a niche market such as pharmaceuticals that require relatively low quantities of high-quality lime.

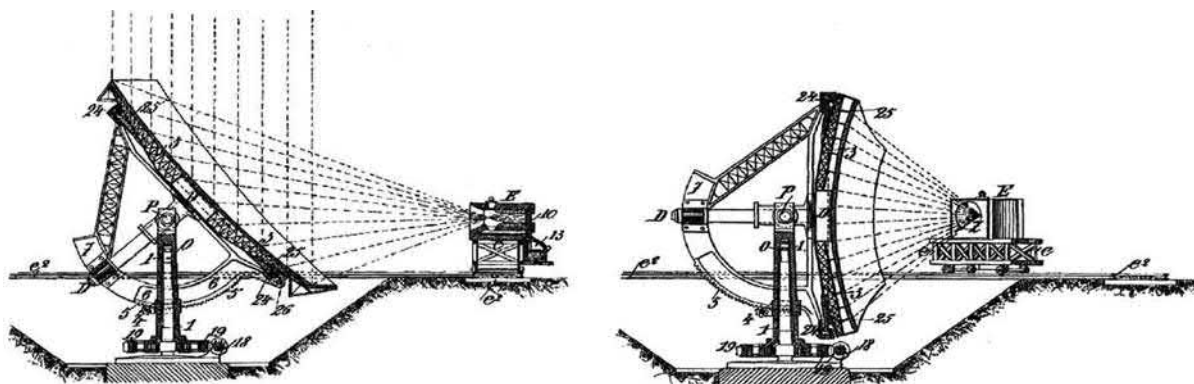


Fig. 1 Apparatus for making industrial use of solar energy registered by Gomes Himalaya (1901).

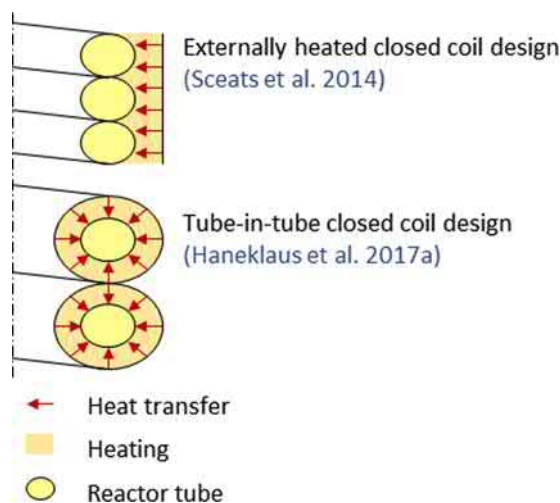


Fig. 2 Simplified cross section of the externally heated closed helical coil design (top) and the tube-in-tube closed helical coil design (bottom). Haneklaus N, Zheng Y and Allelein HJ (2017d) Stop smoking—Tube-in-tube helical system for flameless calcination of minerals. *Processes* 5(4): 67.

Indirect flameless calcination

Traditionally, the fuel combustion process and the calcination reaction take place together to enhance heat transfer in a calcination furnace. For a significant reduction of CO₂ emissions radically new systems will have to be developed and tested that might challenge this approach. Haneklaus et al. (2017a,b) coined the term “flameless” for calcination reactions by other means than fossil fuel combustion. Fossil fuel powered (mostly natural gas) systems using indirect heat transfer, i.e., separate fuel combustion and calcination are already occupying a niche market supplying higher quality calcination products such as the high-quality lime needed for pharmaceuticals. Sceats et al. (2014) envisioned a system for indirect calcination of minerals using a helical tube reactor. Sceats et al. (2014) proposed to choose the pitch of the helical tube in a way that it is equal to the outer diameter of the reactor tube, so that a closed coil is formed that can be externally heated. Fuel combustion, electrical heating or a heat transfer fluid (HTF) were foreseen in the externally heated closed coil design. Based on the closed coil tube design Haneklaus et al. (2017a) proposed using a tube-in-tube helical design for the calcination of minerals. Using a tube-in-tube reactor design may be beneficial in a way that the area for heat input from the HTF can be doubled as depicted in Fig. 2.

According to He et al. (2020) the 3rd generation of concentrated solar power plants that is currently being developed will use molten salts, He, supercritical CO₂ and particles as heat transfer fluids that will allow operating temperatures above 700 °C. These temperatures are still lower than the about 1000 °C used in the majority of lime calcination operations at the moment. Increasing the volume to surface ratio by using fine powder and inducing steam can reduce calcination temperatures. Sceats et al. (2014) conducted feasibility tests with batches of 0.2–2.0 kg finely ground (125 μm) magnesite ore (97% MgCO₃). It could be shown that the residence time and the required calcination temperature can be reduced to a few seconds and <500 °C if the material is finely ground and superheated steam also at <500 °C is added.

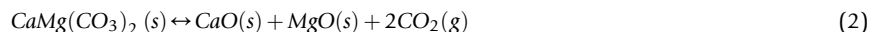
The calcination designs of both Sceats et al. (2014) and Haneklaus et al. (2017a) are designed to work with fine powder and use superheated steam to further reduce processing temperatures. A cross section of the whole tube-in-tube flameless calcination system

proposed by and Haneklaus et al. (2017a) is provided in Fig. 3 (left) and a schematic overview of the system with HTF in- and outflow is presented on the right of Fig. 3.

The heat transfers from the outer tube (blue, in cross section) to the inner reactor tube (yellow, in cross section). The finely ground mineral feed is mixed with steam to further reduce the required calcination temperature and time in the reactor tube.

With a market volume of a bit more than USD 200 billion and a production of over 5 billion tonnes the cement industry is the largest calcination industry in the world. The relatively high calcination temperatures might not make it the first calcination industry that could profit from flameless calcination, however. Phosphate rock often contain large amounts of calcite (CaCO_3) and dolomite ($\text{CaMg}(\text{CO}_3)_2$) that need to be removed prior to the production of mineral fertilizers. Calcination of phosphate ores to remove carbonates is expensive as a result of the high costs for energy. Phosphate rock calcination is practiced for approximately 10% of phosphate rock, mostly to improve final product quality by removing minor amounts of carbonates and organic matter (Ruan et al., 2019).

For fine powder of dolomite ($\text{CaMg}(\text{CO}_3)_2$), calcination temperatures with superheated steam range from 500 to 650 °C and usually take place in a single endothermic reaction described in Eq. (2) (Olszak-Humienik and Jablonski, 2015).



The required calcination temperatures for dolomite can be comfortably reached using inexpensive solar salt ($0.6 \text{ NaNO}_3 + 0.4 \text{ KNO}_3$) that is already in use in a number of concentrated solar power plants today. Interestingly the global phosphate rock reserves reported by the US Geological Survey (USGS) (2020) by country are largely located in regions with excellent to good solar radiation as indicated in Fig. 4.

Besides concentrated solar power, high temperature reactors are another greenhouse gas lean energy source that are capable of heating a HTF (steam, molten salt, etc.) to temperatures relevant for flameless calcination. Among the various high temperature reactors capable of reaching the required high temperatures (> 600 °C) for flameless calcination of phosphate rock (Haneklaus et al., 2016), high temperature gas-cooled reactors (HTGRs) are presently the most mature technology. HTGRs such as the HTR-PM demonstration plant that is currently under construction in Shandong province in China, have been under development since the 1960s (Reitsma et al., 2018). Fig. 5 provides an overview of past and present HTGR projects by country.

Despite the maturity of HTGRs and the demonstration plant presently erected in Shandong Province, China (Zhang et al., 2016) it will still take considerable efforts until HTGRs or any other group of high temperature reactors can be employed for other services than electricity production. The very high temperatures will further present new material challenges that need to be overcome (Sabharwall et al., 2013).

Keeping in mind that calcination accounts for some 5% global anthropogenic CO_2 emissions (Worrel et al., 2001) taking on these challenges is justified. Fig. 6 provides a brief overview of the plant layout for flameless calcination with the proposed tube-in-tube helical reactor and concentrated solar power (top) as well as high temperature reactors (bottom) as power sources.

The technical feasibility of flameless calcination with concentrated solar power and high temperature reactors has yet to be proven. The work by Sceats et al. (2014) for Calix Ltd. from Australia resulted, among other things, in the Leilac 1 and 2 ("Low Emissions Intensity Lime and Cement") projects (Hills et al., 2017; Hodgeson et al., 2018) that received nearly EUR 30 million in funding from the EU to build a prototype calcination furnace that promises to greatly reduce CO_2 emissions. The projects show the tremendous interest of the European mineral processing industry as well as the EU cleaner mineral processing technologies.

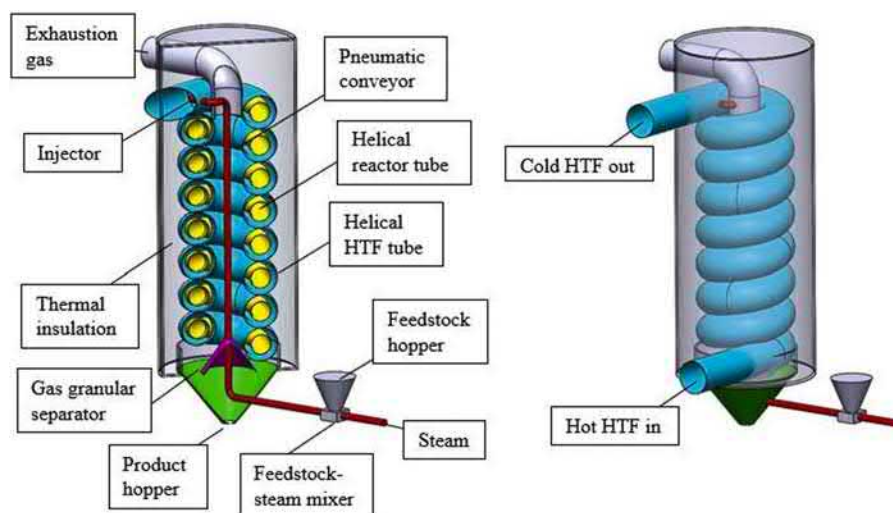


Fig. 3 Overview of the tube-in-tube helical system for flameless calcination proposed by Haneklaus et al. (2017a). Haneklaus N, Zheng Y and Allelein HJ (2017d) Stop smoking—Tube-in-tube helical system for flameless calcination of minerals. *Processes* 5(4): 67.

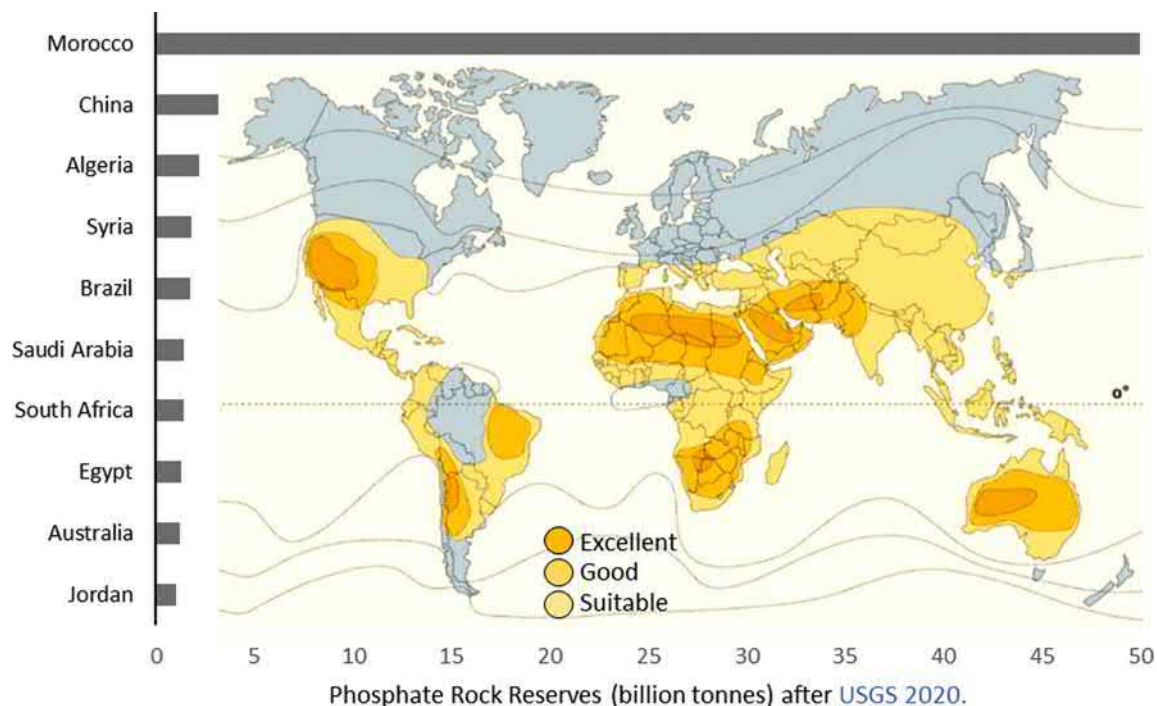


Fig. 4 Global solar radiation potential for concentrated solar power plants and 10 countries with the largest phosphate rock reserves.



Fig. 5 Past and present high temperature gas-cooled reactor (HTGR) projects by country. Reitsma F, Woods P, Fairclough M, et al. (2018) On the sustainability and progress of energy neutral mineral processing. *Sustainability* 10(1): 235.

Based on the estimated heat requirements ($3.6 \text{ MW}_{\text{th}}$ in total; $2.6 \text{ MW}_{\text{th}}$ for 1.45 kg/s calcite that needs to be heated to $600\text{--}750^\circ\text{C}$ and $1.0 \text{ MW}_{\text{th}}$ for $0.05\text{--}0.5 \text{ kg/s}$ superheated steam at $0.02\text{--}0.3 \text{ MPa}$) of the flameless calcination using the tube-in-tube helical system, an adequate cogeneration layout for power supply with a HTGR was designed by BrivaTech Consulting as shown in Fig. 7.

The brief analysis showed that a HTGR with $180 \text{ MW}_{\text{th}}$ would be enough to provide high temperature process heat and superheated steam for up to 46 calcination units that could process approximately 5762 t mineral feed (calcite/dolomite or other) per day. In the analysis, electricity is only generated for the cogeneration system itself. Different configurations with fewer calcination units and more electricity production could of course be considered.

Average calcination furnaces process 1 t calcite (CaO_3) per day and emit between 1.1 and 1.6 t CO_2 per tonne lime product (EuLA, 2014). In the European Union, approximately 68% of the emitted CO_2 emissions result from the calcination reaction itself with another 30% being a result of fuel related emissions and 2% are indirect CO_2 emission associated with electricity production. Fossil-based solid fuels (51%), natural gas (34%), waste (8%), oil (5%) and biomass (2%) are used to generate the required heat (EuLA, 2014). Flameless calcination could reduce direct CO_2 emissions in mineral calcination by 32% and if exhaustion gases are captured as indicated in Fig. 3, CO_2 emissions could be largely eliminated during mineral calcination.

Economics of flameless calcination

Besides the technical feasibility, Haneklaus et al. (2017b) conducted an economic assessment of flameless phosphate rock calcination with concentrated solar power and high temperature reactors. The assessment compared heating costs (USD per tonne calcine) and net present values (NPVs) at different heat selling prices (USD 15, 30 and 45 per MWh_{th}). Cost estimates were based on literature values of the costs for electricity production of the GemaSolar concentrated solar power plant in Spain (Burgaleta et al., 2012)

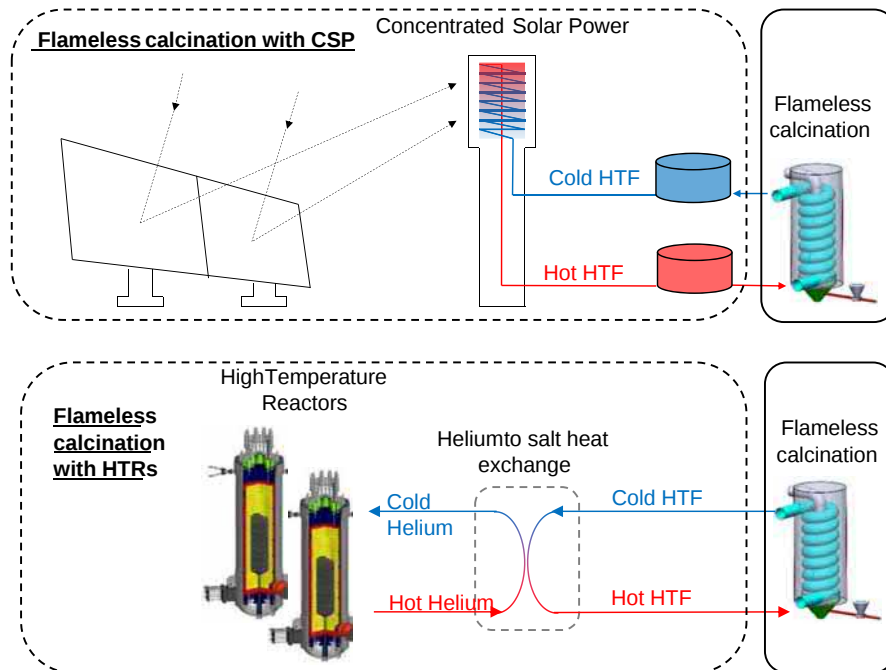


Fig. 6 Proposed flameless calcination plant layout with concentrated solar power (top) and high temperature reactors (bottom).

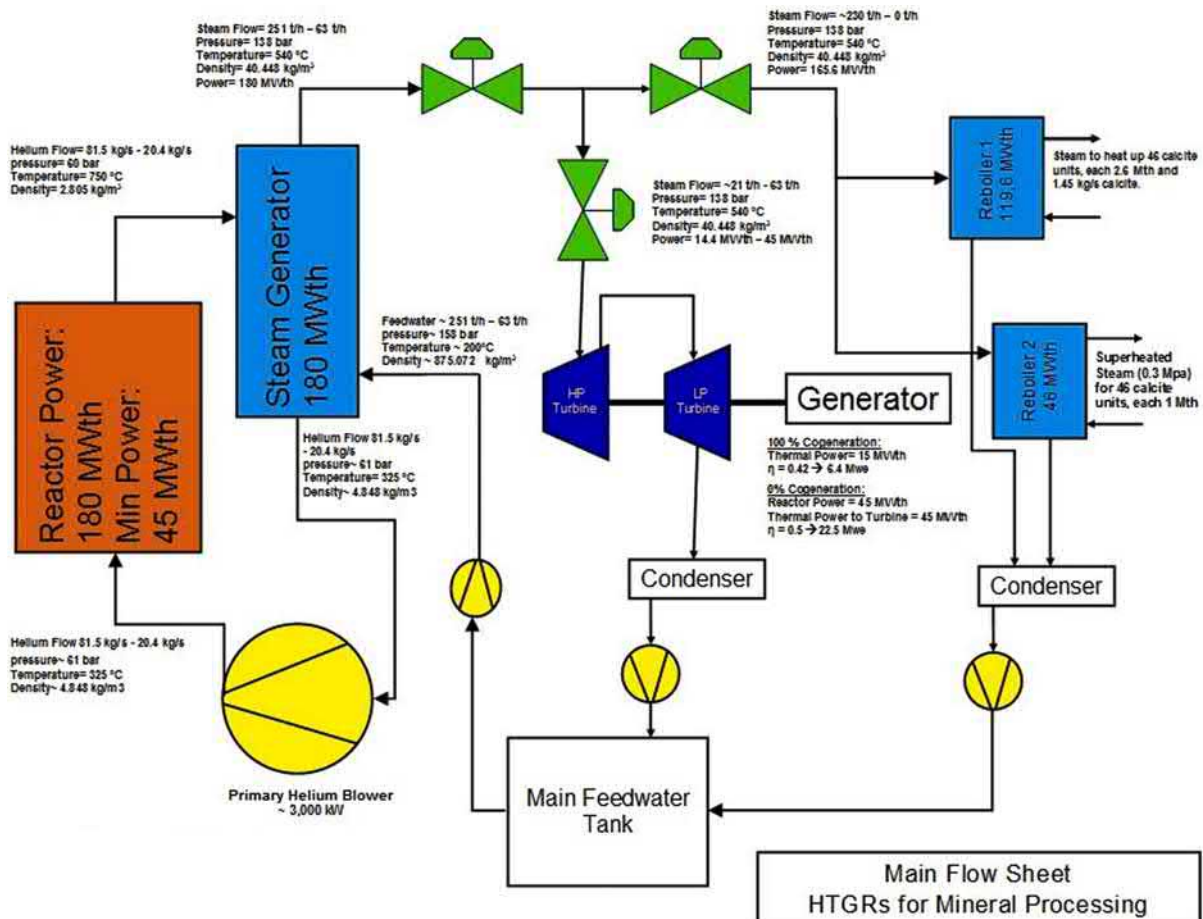


Fig. 7 High temperature gas-cooled reactor (HTGR) cogeneration system for flameless phosphate rock calcination. Source: BriVaTech.

(capital costs of USD 11,445/kW_e and O&M costs of USD 38.15/MWh_e for the 1st of a kind) and the HTR-PM (High-Temperature Gas-Cooled Reactor Pebble-Bed Module) currently under construction in China (Zhang et al., 2016) (capital costs of USD 5000/kW_e, O&M costs of USD 14.54/MWh_e and fuel costs of USD 9.60 MWh_e for the 1st of a kind).

The economic assessment revealed that natural gas-fired kilns calcine phosphate rock at lower costs than the flameless system with concentrated solar power/high temperature reactors. This is no surprise given the current costs for concentrated solar power plants and high temperature reactors. The study did further show that flameless calcination could become cost competitive if potential future cost reductions for concentrated solar power plants (capital costs of USD 5723/kW_e and O&M costs of USD 19.08/MWh_e for the nth of a kind) and high temperature reactors (capital costs of USD 2500/kW_e, O&M costs of USD 14.54/MWh_e and fuel costs of USD 9.60 MWh_e for the nth of a kind) as they are foreseen by the International Renewable Energy Agency (IRENA) and the reactor developers can be realized, low-interest rates of 5–10% and relatively high natural gas prices above USD 7.5–10 per mmBTU are present. Besides, the study considers a carbon tax of USD 30/t CO₂ emitted. Even at these benevolent conditions a flameless calcination system will always be more complicated than much simpler gas-fired kilns that are around for centuries and potential consumers interested in flameless calcination will need to consider if the additional risks can be justified (Haneklaus et al., 2017b).

Energy neutral mineral processing

It is well-known that phosphate rock can contain natural uranium at elevated concentrations of 70–200 mg/kg (Haneklaus et al., 2017c,d). This accompanying uranium can be recovered and used as raw material to produce nuclear reactor fuel. To illustrate the significant concentrations of natural uranium in phosphate rock, Haneklaus et al. (2015a,b) and Haneklaus and Schnug (2016) proposed energy neutral mineral processing of phosphate rock to mineral fertilizers in a way that uranium is recovered during phosphate rock processing and after enrichment and fuel manufacturing used as nuclear reactor fuel in high temperature reactors that provide electricity and process heat for mineral processing with unconventional uranium recovery. If the amount of recovered uranium is sufficiently high to operate the nuclear reactor the process can be labelled energy neutral as illustrated in Fig. 8.

Since the introduction of energy neutral mineral processing the quantities of unconventional uranium that could be recovered during phosphate rock processing in Argentina (López et al., 2019), China (Shang et al., 2021; Ye et al., 2019), the European Union (Tulsidas et al., 2019) and the United States (Kim et al., 2016; Steiner et al., 2020) have been evaluated. Recovering uranium from calcinated phosphate rock can be more economic since a cleaner phosphoric acid is produced after calcination compared with phosphoric acid produced after flotation of phosphate rock. Most importantly flameless calcination and uranium recovery are independent processes that can be further developed in parallel.

Conclusions and outlook

Realistically, present day high temperature reactor technology can support high temperature process heat applications in the range of 700–800 °C. Such temperatures may not be sufficient to calcinate lime for cement production that decomposes at temperatures of 500–1150 °C and is usually performed at approximately 1000 °C. It could, however, be shown that steam can be used to reduce the calcination temperature and it is further well-known that dolomites and other minerals can be decomposed at temperatures below 800 °C. Of particular interest may be the calcination of selected phosphate rock that can occur at temperatures well below

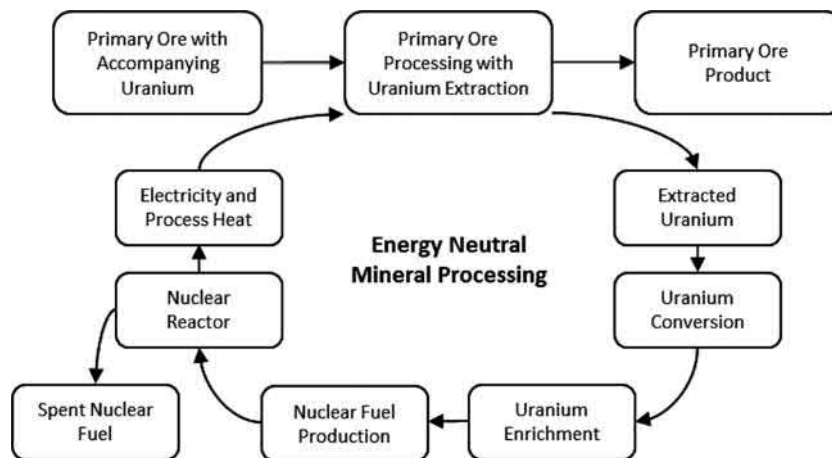


Fig. 8 Schematic overview of energy neutral mineral processing. Reitsma F, Woods P, Fairclough M, et al. (2018) On the sustainability and progress of energy neutral mineral processing. *Sustainability* 10(1): 235.

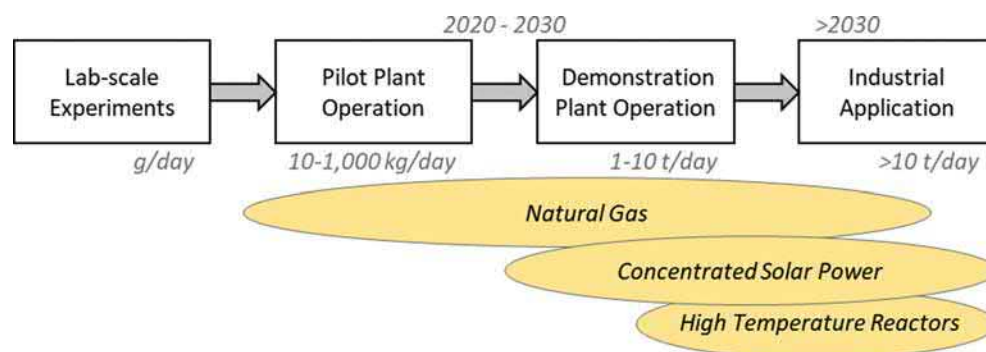


Fig. 9 Path forward for flameless calcination.

800 °C. Ironically most phosphate rock reserves are located in regions with excellent to good solar radiation for concentrated solar power plants such as northern Africa (more than 75% of all phosphate rock reserves are located in Morocco). **Fig. 9** provides an outlook on the next steps of making flameless calcination (with or without uranium recovery) a reality.

The first lab-scale experiments that can support flameless calcination were conducted in Tunisia recently (Abbes et al., 2020) and there is generally enough data to estimate the required calcination temperatures of phosphate rock from different deposits. As a next step, flameless calcination, most likely with natural gas heating a HTF, will need to be performed on pilot plant- and subsequently demonstration plant scale before this technology can be used for industrial applications that are then powered by concentrated solar power and high temperature reactors.

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Marine Propulsion

Anil Kumar Anand^a and J. Stephen Herring^b, ^a MICROTRON Sterilisation Services Pvt. Ltd., ATL Corporate Park, Mumbai, India; and ^b Center for Space Nuclear Research, Idaho Falls, ID, United States

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Glossary

AEC Atomic Energy Commission US government agency, successor to the Manhattan Project and succeeded by the Energy Research and Development Agency (ERDA) and by of the Nuclear Regulatory Commission (NRC).

AIP Air Independent Propulsion, not needing air for propulsion.

knot Unit of nautical speed, one nautical mile per hour = 1.15 statute miles per hour = 0.51444 m/s = 1.852 km/h.

MIT Massachusetts Institute of Technology, university in Cambridge, Massachusetts, United States.

MWe/MW_{shaft} Megawatts electric/Megawatts shaft, measures of the output of usable work from a heat engine, such as a steam turbine or a diesel engine.

MW_{th} Megawatts thermal a measure of the thermal input to a heat engine from a nuclear reactor, boiler or diesel fuel combustion.

NPT Treaty on the Non-Proliferation of Nuclear Weapons initially signed in 1968 and came into force in 1970.

Q244 French submarine reactor project.

SSBN Ballistic Missile Nuclear-powered Submarine (US Navy designation, SS = SubSurface).

SSGN Guided Missile Nuclear-powered Submarine (US Navy designation)

SSN Nuclear-powered Attack Submarine

t Metric ton, 1000 kg.

TEU Unit of measure for container ship capacity, 20-foot Equivalent Units, a standard container 20' long, 8'6" high and 8' wide (6.096 m × 2.59 m × 2.438 m). This unit continues to be used although most containers are now 40 ft in length and amount to 2.0 TEU. Other containers are 45, 48 or 53 ft in length and some are 9.5 ft. in height. All containers have standardized maximum weight limits. While actual loaded weights of course vary, the average gross weight is 14 t/TEU.

Abbreviations

MTCR Missile Technology Control Regime

SLBM Submarine-launched ballistic missile

The need and the beginning

Submarine warfare during world war II

Germany used submarines to devastating effect in [World War II](#). During the first few years of the War, their U-boat fleet scored unprecedented success. These diesel electric boats had to surface periodically to charge their batteries and met the same fate. In 1943, hoping to hide their U-boats from the increasingly devastating air patrols, Germany perfected an idea that had been floating around for some time—the use of a breathing tube to allow running on diesel power just below the surface, thus also keeping batteries fully charged. They dubbed it the “snorkel” but the innovation also presented several problems to their users. A U-boat with a snorkel raised was limited to a speed of 6 knots to avoid damage to the tube and its sound-detection gear was useless with the diesel engine in operation. The increasingly sophisticated detection equipment deployed by the Allies led to severe losses and German designers began to fully realize the potential of a truly submerged boat. In response to the mounting U-boat losses, in 1943, a submarine research center was created by the Germans at Blankenburg in the Harz Mountains with the aim of producing an operational high-speed U-boat capable of prolonged submergence.

Air independent propulsion systems

Air-independent propulsion (AIP) is any [marine propulsion](#) technology that allows a non-nuclear [submarine](#) to operate for longer periods in submerged condition. Air Independent Propulsion (AIP) technology became the holy grail of submarine design ever since German engineer Hellmuth Walter first began experimenting with his high purity hydrogen peroxide powered engine in the early 1930s. A Closed-cycle Diesel Engine in a submarine can be operated conventionally on the surface, but can also be provided with [oxidant](#), usually stored as [liquid oxygen](#), when submerged. Closed-cycle steam turbine is the propulsion system with heat generated by [ethanol](#) and oxygen stored at a pressure of 60 [atmospheres](#). A number of countries are working on different AIP systems; fuel cells and lithium batteries are most predominant. Clearly, no AIP technology currently resolves all submarine challenges.

Nuclear propulsion

After the discovery of nuclear fission in 1939 ([Michel, 2000](#)), the original interest of the US Government in harnessing the power of the atom was not to produce an atomic bomb but to devise nuclear propulsion for ships. Before the outbreak of the Second World War, US Naval Research Laboratory was engaged in studying the idea of building a propulsion plant for ships. To build a light weight, small nuclear plant for ships, the Laboratory developed the first method of separating the essential isotope of Uranium 235 from natural uranium. This was the thermal diffusion method which did not prove to be as practical as the gaseous diffusion later developed at Oak Ridge.

Once the decision was made to develop an atomic bomb, the initial idea of the propulsion plant was shelved and the bomb project was given priority. After the war, when the subject of nuclear propulsion was rekindled, there were those who believed that atomic reactors with large amounts of steel-shielding and concrete could never be carried aboard ships. They had in mind the very large reactors at Hanford, built for producing plutonium for bombs during the war. Others thought that humans could not live and work in close proximity to radiation sources. Fortunately, there were those who thought that these problems could be overcome; the credit must go to them for the success of atomic submarines and even the first commercial nuclear-powered ship “Savannah,” which was commissioned a few years after the first nuclear propelled submarine “Nautilus.”

Development in different countries

United States

Toward the end of 1945, the idea of an atomic powered submarine was fascinating to Captain Hyman Rickover who led the program and was later promoted to Admiral. Since the atomic reactor did not require oxygen, for the first time the Navy could have a true submarine, which could stay underwater indefinitely without contact with the surface. The final limits would be for replacing equipment that wears out, the fatigue limit of the hull, crew morale and the amount of food it could carry. Advantages were tremendous; it could sneak up to an enemy surface ship at full speed, destroy and escape while submerged.

Submarines

Captain Rickover concluded that the atomic powered submarine was the first order of business for the United States Navy. If a reactor could be built to fit in a small submarine, it was sure to be fitted in any larger ship. He could convince the authorities that the submarine project, like the hydrogen bomb, would become a key to the nation's defense; civilian nuclear power program would become easier once the nuclear propulsion was developed. Captain Rickover had by then concluded that the reactor would have to be thermal/slow neutron type and use water as a coolant. At this stage the Westinghouse Electric Corporation was brought on board; Westinghouse was required to set up a laboratory with requisite scientific personnel to carry out experiments on a heat exchanger system for an atomic power reactor similar to the General Electric program, but the difference was that Westinghouse would be working with pressurized water as the coolant instead of low-pressure sodium being used by General Electric for the core coolant. Conducting experiments was not enough; they had to work on the most difficult task, the reactor itself. This was a step in the right direction; two of the then largest and most capable firms were competing with each other toward this end. Equally important was the shielding, which would protect personnel from the extraordinary amounts of radioactivity generated within the reactor fuel. A series of experiments were performed in research reactors to test the performance of various shielding materials. There was a challenge to metallurgists to find a metal that could withstand high stress while operating at high temperatures and also did not absorb neutrons. This metal was a strange new metal called zirconium and large-scale production had to be started. In addition, the project also required substantial quantities of beryllium (Be) and later hafnium (Hf); Be slows-down fast neutrons and Hf absorbs neutrons. Both materials can withstand high temperatures. Captain Rickover became the "Chief" of the naval reactor branch of Atomic Energy (Anand et al., 2016).

The contract between the Atomic Energy Commission and Westinghouse was formalized; AEC agreed to provide funds for the construction of an atomic power laboratory—the Bettis Laboratory, in conjunction with the scientists of their Argonne National Laboratory. The Bettis Lab was built by Westinghouse at a site outside Pittsburgh where physicists and engineers assembled to design and build the thermal water-cooled submarine reactor. With the backing from the Navy and the Atomic Energy Commission, the educational staff at the Massachusetts Institute of Technology arranged a special 1-year course in nuclear engineering.

The design of the "Reactor Core" consisting of fuel elements, control and shut off rod mechanisms housed in the reactor pressure vessel is probably the most difficult task for the reactor engineers. Evolving a practical design of fuel elements proved to be an especially difficult task. Not until March 1950 did Argonne and Bettis decide that it would be feasible to assemble a fuel element consisting of uranium zirconium alloy clad with zirconium.

The biggest problem was to fit the bulky atomic reactor inside the hull; some felt that the reactor should be partly outside the hull away from the ship like an aircraft engine on the wings and others argued that the reactor should be inside the hull in the designated "reactor compartment" thus containing all the radioactivity inside the compartment. Finally, Westinghouse solved the problem of accommodating the reactor in the hull. When a new type of propulsion plant is built for a ship, aircraft or any other vehicle, it is mandatory to first build an experimental version on land to ensure it performs properly. Captain Rickover insisted that this land-based plant be built inside the submarine hull on land. If Westinghouse painstakingly built a land-based plant, checked it thoroughly, and then redesigned it to fit in the submarine, the two stages would take years to complete. The Captain's idea was to build two plants simultaneously; one, a land-based prototype, and the second would go into the submarine. The schedule would be worked out in such a way that the sea-going reactor would lag slightly behind the land-based reactor so that any changes and improvements could be incorporated into the sea-going version expeditiously.

Nautilus and seawolf

Captain Rickover convinced the Electric Boat Company to be a sub-contractor to the Westinghouse Electric Corporation in designing and building the desert bound, land-based prototype submarine and its machinery. GE reoriented its program with emphasis on a fast reactor for the submarine and the Electric Boat Company readily agreed to assist GE in the ship building aspects of the project as they were doing for Westinghouse.

To avoid confusion, labels were put on both the projects; Westinghouse project was named STR short for "submarine thermal reactor" and GE project as SIR for "submarine intermediate reactor." The land-based reactor and the sea going version of Westinghouse STR, were named Mark One and Mark Two, respectively, while the GE's SIR were named Mark A and Mark B for the land based and sea going version, respectively. A typical flow diagram of a propulsion reactor is shown in Fig. 1.

By 1951, Mark One was under construction near Arco, Idaho at the National Reactor Testing Station (NRTS), and Mark Two was about to be started at Bettis Atomic Power Laboratory by Westinghouse. Electric Boat Company started the plans for construction of the submarine itself. The keel of the "Nautilus" was laid on June 14, 1952 in Groton, Connecticut. In May 1953, the land-based prototype was started in the NRTS. It performed much better than expectations and started "nonstop" long runs. The work on

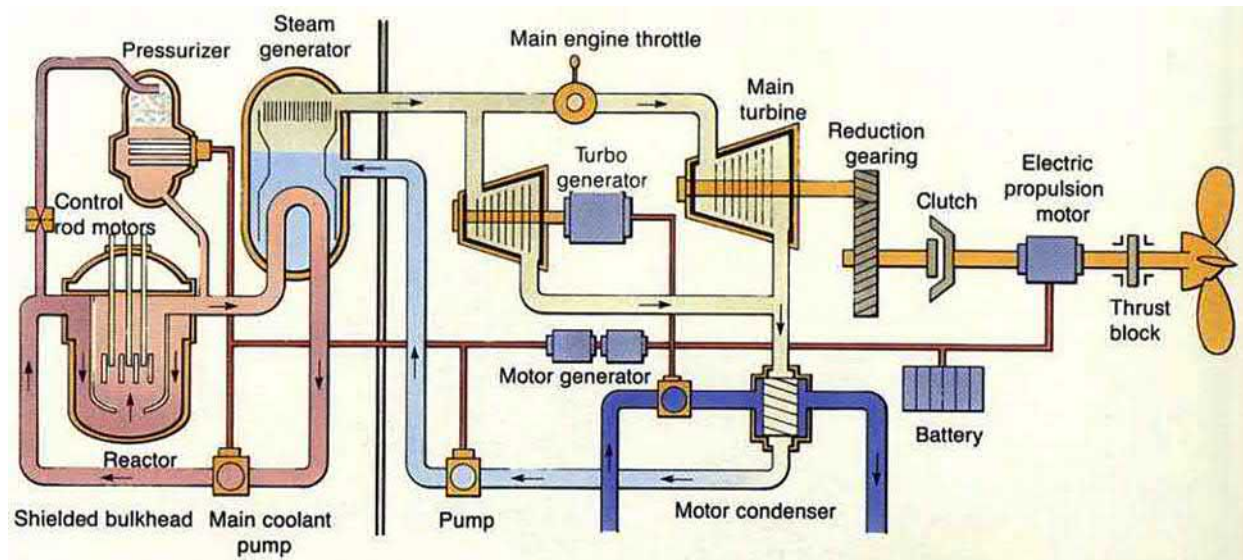


Fig. 1 Typical flow diagram of a propulsion Reactor. https://en.wikipedia.org/wiki/File:Diagram_Submarine_Reactor_&_Steam_Plant.jpg This file is licensed under the Creative Commons Attribution-Share Alike 4.0 International license.

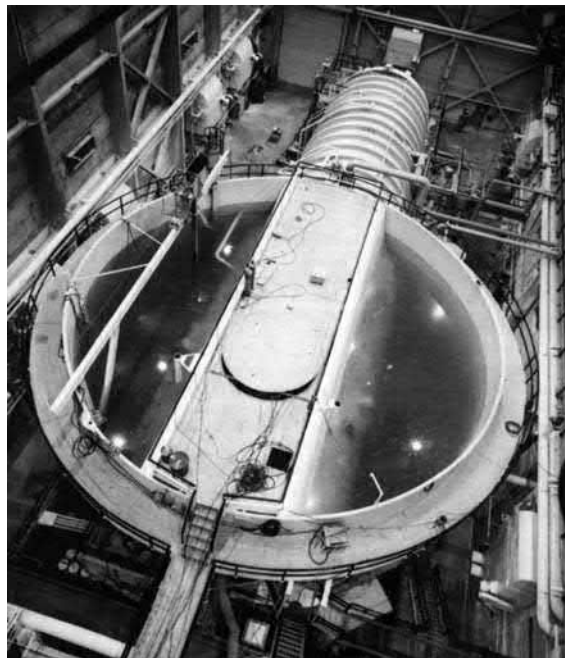


Fig. 2 Mark one, the land based plant. https://en.wikipedia.org/wiki/File:Nautilus_core.jpg, <https://en.wikipedia.org/wiki/File:Nautiluscore.jpg>. This is a work of the U.S. federal government, it is in the public domain in the United States. Public Domain Mark 1.0.

"Nautilus" progressed rapidly. The submarine was launched on January 21, 1954. On January 3, 1955 the Mark Two reached full power. On January 17, 1955, as the Nautilus slipped down the Thames, a signalman on the submarine blinked to the escort tug "Skylark" "UNDER WAY ON NUCLEAR POWER." **Fig. 2** shows the Mark one, the land-based plant, and **Fig. 3** shows the launching of Nautilus.

The GE's Mark A and Mark B propulsion plants were also commissioned soon thereafter. The Submarine Intermediate Reactor (SIR) nuclear plant designed by General Electric's Knolls Atomic Power Laboratory was designated S1G and Seawolf's plant as S2G. The Seawolf's keel was laid on 7 September 1953 by the Electric Boat division of General Dynamics Corporation in Groton, Connecticut. Seawolf was launched on 21 July 1955 and commissioned on 30 March 1957; its propulsion system was more technologically advanced. It was carrying a liquid sodium cooled and more powerful **epithermal** reactor and a superheated steam power plant. The Nautilus carried the alternative light water reactor and saturated steam plant. In Seawolf, the size of the machinery in the



Fig. 3 Nautilus launched. https://en.wikipedia.org/wiki/File:Nautilus_core.jpg, <https://en.wikipedia.org/wiki/File:Nautiluscore.jpg>. This is a work of the U.S. federal government, it is in the public domain in the United States. Public Domain Mark 1.0.

engineering spaces was reduced by about 40% relative to that of Nautilus. However, the USS Seawolf was plagued by super heater problems, with the result that the USS Nautilus delivered far superior performance. This, and the risks posed by liquid sodium in the event of an accident at sea, led Admiral Rickover to select the PWR as the standard U.S. naval reactor type. The S2G was removed from USS Seawolf and replaced by the [S2Wa reactor](#), using components from the spare S2W that was part of the USS Nautilus program. All subsequent U.S. naval reactors have been PWRs.

Skipjack class

After Nautilus came the first of the Skipjack class of attack submarines. Skipjack's keel was laid down on 29 May 1956 by the Electric Boat Division of the General Dynamics Corporation in [Groton, Connecticut](#). Skipjack was launched on 26 May 1958 and commissioned on 15 April 1959. The new class introduced the teardrop hull and the S5W reactor to U.S. nuclear submarines. The Skipjacks were the fastest U.S. nuclear submarines until the [Los Angeles-class submarines](#), the first of which entered service in 1974.

Later classes of submarines

Next came the George Washington class Ballistic Missile submarines; the first was launched in June 1959 and commissioned in December 1959 (The US also later added some SSGNs, carrying guided missiles.). The largest submarines in the US Navy, USS Ohio class, are the first Trident ballistic missile submarines, commissioned beginning in 1981. The latest three submarines of this class are USS South Dakota (SSN 790, commissioned in February 2019), USS Delaware (SSN-791, commissioned in April 2020) and USS Vermont (SSN-792, commissioned in April 2020).

Aircraft carriers

USS Enterprise was the first [nuclear-powered](#) aircraft carrier and its keel was laid in 1958. It was launched in September 1960 and commissioned in November 1961. She was deactivated on 1 December 2012, and officially decommissioned on 3 February 2017. Enterprise is also the only aircraft carrier to house more than two nuclear reactors, having an eight-reactor propulsion design.

The Nimitz class is a class of 10 nuclear reactor-powered aircraft carriers in service with the United States Navy. The lead ship of the class was commissioned in May 1975, and the tenth and last of the class was commissioned in January 2009.

Ford-Class is a class of aircraft carriers being built to replace USS Enterprise and eventually the United States Navy's existing Nimitz-class carriers.

Soviet Union/Russian Federation

The important questions for the West in the early 1950s were about the capability of Soviet Union. Was the Soviet Union technically capable of producing an atomic powered submarine? If so, had the Soviet Navy the background and experience and would they

make good use of it? Was the Soviet Union capable of waging effective anti-submarine warfare against atomic powered submarines of the West such as the “Nautilus?” The first question was partially answered a few months after the launch of “Nautilus” when it was announced that the Soviets had built and successfully operated the world’s first peace time atomic power reactor in Obninsk. The significance of this announcement was the fact that, after successfully developing the atomic powered reactor, the Soviets had crossed the most difficult hurdle in building an atomic powered submarine.

Submarines

The Soviet Union soon followed the U.S. in developing nuclear-powered submarines in the 1950s. Stimulated by the U.S. development of the Nautilus, Soviet work on nuclear propulsion reactors began in the early 1950s at the Institute of Physics and Power Engineering in Obninsk, under Anatoliy P. Alexandrov, later to become head of the Kurchatov Institute. In 1956, the first Soviet propulsion reactor designed by his team began operational tests. Meanwhile, a design team under Vladimir N. Peregudov worked on the vessel that would house the reactor. After overcoming many obstacles, including steam generation problems, radiation leaks, and other difficulties, the first nuclear submarine, Leninsky Komosomol, entered service in the Soviet Navy in 1958. The submarine was built in Molotovsk, launched on 9 August 1957 after being commissioned in July 1958. It used a water cooled reactor (Anand et al., 2016).

During the construction of the Leninsky Komosomol, the first November-class submarine, the Soviet Union was also building liquid-metal cooled submarine propulsion systems. The liquid metal used by the Soviets/Russians was lead-bismuth eutectic (rather than the sodium tried by the US Naval Program). This technology was developed at the Institute of Physics and Power Engineering (IPPE) in Obninsk and used in two submarine classes: Project 645, a class in itself, and the Alfa class. Using liquid metal coolant was considered to have several advantages. It is more compact than PWRs since it needs no moderator. No heavy pressure vessel is needed; it operates at higher temperatures and, therefore, has a higher thermal efficiency than water cooled reactors. Additionally, lead-bismuth is not chemically reactive with water as is sodium. Refueling is faster, since the core is removed in a single operation, but was more frequent than in the US submarines. However, there are disadvantages: the melting point of the coolant is above room temperature, so the primary system must be kept heated at all times for the coolant to remain liquid. If not, the coolant will solidify and cooling will be interrupted. The liquid metal coolant will gradually oxidize and the oxides have to be removed regularly to avoid blockage of the coolant flow through the core. The LMC reactor was first used in 1962 in a special version of a November-class submarine (Project 645, K-27), which used two RM-1 reactors with capacity of 73 MWth each. At the height of the Cold War, approximately 5–10 nuclear submarines were being commissioned from each of the four Soviet submarine yards. From the late 1950s through the end of 1997, the Soviet Union, and later Russia, built a total of 245 nuclear submarines, more than all other nations combined. They even built submarines with a titanium hull, which came as a big surprise to the US and other nations. Soviet designated Project 661 was the Soviet Navy’s nuclear-powered cruise missile submarine and the only submarine of that design. It was the world’s fastest submarine and the first submarine ever constructed with a titanium hull. The boat is best known in the West by its NATO reporting name Papa class.

The Yankee-class nuclear submarines were the first of Soviet ballistic missile submarines (SSBN) to have thermonuclear fire-power comparable with that of their American and British Polaris submarine counterparts. The Yankee class was quieter in the ocean than were their Hotel-class predecessors, and had better streamlining that improved their underwater performance. These boats were all armed with six submarine-launched ballistic missiles (SLBM) with multiple nuclear warheads as nuclear deterrents during the Cold War. The new Yasen Class is projected to replace the Russian Navy’s current Soviet-era nuclear attack submarines. Fig. 4 shows a Typical Block Type propulsion system used in Soviet/Russian submarines and ice breakers.

Icebreakers

At the same time, the Soviet Union started work on “nuclear propulsion” for ice breakers. A nuclear-powered icebreaker is a nuclear-powered ship built for use in iced waters. Nuclear-powered icebreakers are much more powerful (35–60 MW_{shaft}) than their diesel powered counterparts (< 25 MW_{shaft}). Although nuclear propulsion is expensive to install and maintain, very heavy fuel demands, the need to clear ice for their fuel tankers, and limitations on range can make diesel vessels less practical and economical, overall, for ice-breaking duties. During winters, ice along the Northern Sea Route varies in thickness from 1.2 to 2.0 m and in central parts of the Arctic Ocean is 2.5 m thick on average. Nuclear-powered icebreakers can force through this ice at speeds up to 10 knots and in ice-free waters the maximum speed of the nuclear-powered icebreakers is as much as 21 knots. All eight of the Russian nuclear icebreakers have steam turbine—electric drive trains (AKER, 2016).

The first nuclear powered ice breaker was named “Lenin.” At its launch in 1957, the icebreaker NS Lenin was both the world’s first nuclear-powered surface ship and the first nuclear-powered civilian vessel. The Lenin was put into regular operation in 1959. Nuclear-powered icebreakers have only been constructed by the USSR (and later, Russia), primarily to aid shipping along the Northern Sea Routes in the frozen Arctic waterways north of Siberia. In all, 10 civilian nuclear-powered vessels have been built in the USSR and Russia. Nine of these are icebreakers, and one (Sevmorput) is a container ship with an icebreaking bow. The icebreakers have also been used for a number of scientific expeditions in the Arctic. Since 1989 nuclear-powered icebreakers have also been used for carrying tourists to the North Pole. TASS also reported in July 2019 that Russia hopes to build its first nuclear-powered aircraft carrier.

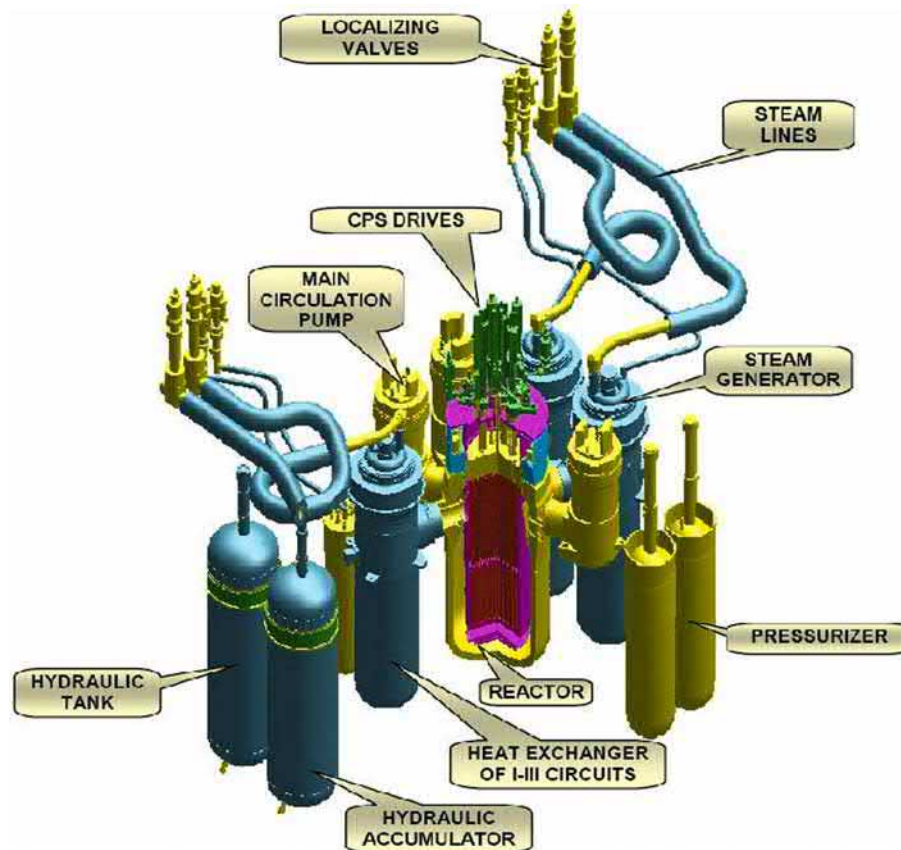


Fig. 4 Typical Block Type propulsion reactor used in Soviet/Russian submarines and ice breakers. <https://aris.iaea.org/PDF/KLT-40S.pdf>. This image is taken as reference only from IAEA technical document on KLT-40S.

United Kingdom

The Royal Navy had been researching designs for nuclear propulsion plants since 1946, but this work was suspended indefinitely in October 1952. The UK's Naval Nuclear Propulsion Program, NNPP, is closely tied to the US. The US supplied the UK with its first naval nuclear reactor and has supplied fissile material for core fabrication. US assistance to the UK NNPP has been part of a wider program of military nuclear cooperation that dates back to the 1940s Manhattan project. Cooperation abruptly ended in 1946 when the US Congress passed the Atomic Energy Act that severely limited the transfer of restricted nuclear information and materials to any other state, causing a major rift with its wartime ally in London. Negotiations to reestablish exchange of military atomic defense information, materials and technology began after an independent UK nuclear weapons program tested its first atom bomb in 1952 and hydrogen bomb in 1957; the United States agreed to supply the UK with one complete submarine nuclear reactor plant. Surprisingly, every country made a nuclear weapon before making nuclear reactors for propulsion. This enabled the UK to deploy its first nuclear-powered submarine, HMS Dreadnought, several years earlier than originally envisaged. It was launched in 1960 and commissioned into service in 1963. On 3 March 1971, Dreadnought became the first British submarine to surface at the North Pole. In 1973 it took part in the Royal Navy's first annual Group Deployment, when a group of warships and auxiliaries would undertake a long deployment to maintain fighting efficiency and "show the flag" around the world. The Royal Navy Submarine Service operates seven fleet submarines (SSNs), of the Trafalgar and Astute classes, and four ballistic missile submarines (SSBN), of the Vanguard class. The UK's first indigenous plant and core based on the S5W Westinghouse design was built by Rolls Royce and deployed on the attack submarine HMS Valiant in 1966. The current PWR2 core design, "Core H," has been designed to last the lifetime of the submarine (25–30 years), thereby eliminating costly mid-life reactor refueling. Core H fuels the new Astute-class SSNs and the four Vanguard-class SSBNs were fitted with the core during their long overhaul and refueling refits between 2002 and 2012 (Ritchie, 2015).

The UK has also been dependent upon the US for special nuclear materials (highly enriched uranium and plutonium) and tritium gas for its nuclear warhead. A total force of seven astute class fleet submarines is planned. As of April 2016, the first three boats are in commission and in service, while boats four to six are in various stages of construction. HMS Ambush, the Royal Navy's newest nuclear attack submarine, is one of the most sophisticated and powerful vessels of its type ever built.

France

The Q 244 project that started in 1955 was the first French attempt to build a nuclear propulsion reactor (Marie-Josée, 1996, Fribourg). The technology would be what was known best and could realistically be achieved with what was available: natural uranium and heavy water. This would require a submarine larger than the largest developed at the time. The project was abandoned in 1958 and France turned to the US for help (GOLDSCHMIDT, 1962; CEA, 2021; Histoire, 2021; LeFebvre, 2002). Of their wish list, the French delegation only obtained 20 t of enriched uranium, in spite of strong opposition from the US's Admiral Rickover. When asked during a Congress hearing if there was any chance that France would succeed in developing a propulsion reactor, the Admiral replied "not the slightest," from which it was concluded that the supply agreement could be signed. One of the conditions was that the enriched uranium would only be used for a research reactor on the land.

Submarines

The first nuclear propulsion reactor was built in the new nuclear research center of Cadarache, 70 km north of Marseille, destined to be the new test bench of the CEA, Commissariat à l'Énergie Atomique, for various advanced nuclear technologies. Since the agreement with the Americans excluded any technology transfer, the small Groupe de Propulsion Nucléaire (GPN) started from scratch and miniaturized the components of the reactor. In September 1959, the GPN received requirements from the Navy: a concise one-page document accompanied by a request to receive the preliminary project document 2 months later to build a submarine on land. The Prototype on Land (prototype a terre, PAT) used pressurized water and the enriched uranium from the United States which would allow a more compact reactor (André, 2021).

The first uranium enrichment plant started in Pierrelatte. France was now independent for its supply of enriched uranium. The PAT was the model for the first six SSBNs. The first was *Redoutable*, laid down in 1964, launched in 1967 and commissioned in 1971. The sixth, *Inflexible*, was laid down in 1980, launched in 1982 and commissioned in 1985.

The Navy being satisfied with the first SSBNs in operation, the time had come to build attack submarines; the construction of a new prototype on land started in Cadarache in 1970. A major innovation was introduced: the steam generator was placed directly above the core of the reactor. The primary block was very compact, resulting in a massive weight reduction from 700 to 400 t. Up to a significant power, the coolant was naturally circulating and thus completely silent. At higher speed the reactor was still silent, as only low-power pumps were required. Based on the experience of this land-based plant, six *Rubies class* were commissioned, the first in 1983 and the sixth in 1993. Unfortunately, the noise levels turned out to be higher than expected and a silencing program needed to be incorporated for the fifth, *Amethyste*, and the sixth, *Perle*. The first four boats were also modified to the same standard.

In the early 1980s the French Navy planned to replace the first generation of SSBN by boats that would be much more silent but also replace the ageing aircraft carriers with a nuclear aircraft carrier. K15 PWR concept was suggested; that would be almost the same for both applications and would draw on the previous successes. Starting in 1984, a near total overhaul of the CAP (Chaudier Avancée Prototype) was undertaken in Cadarache. The construction of *Le Triomphant* started in June 1989, launched in 1994 and entered active service in 1997. The submarine displaces 14,335 t, its speed is approximately 25 knots and maximum depth is 400 m (ARCEA GASN, 2010; Delaeter, 1999). Most of all, it produces approximately one thousandth of the detectable noise as compared to *Le Redoutable class* boats and is 10 times more sensitive in detecting other submarines. *Le Téméraire*, *Le Vigilant* and *Le Terrible* followed *Le Triomphant*. These are the most expensive submarines. The Barracuda program to replace the *Rubis class*, was started in 2006, the first of its class, *Suffren* was laid down in 2007, was launched in July 2019 and expected to be commissioned in 2020 (Marine, 2013).

Aircraft carrier

The aircraft carrier *Charles de Gaulle* was commissioned in 2001 and is equipped with two K15 reactors and four diesel-electric engines developing 40,230 HP each, driving two propeller shafts. With a speed of 27 knots, the *de Gaulle* fields the powerful naval variant of the Dassault Rafale and can also house a variety of other specialized aircraft and helicopters as needed (Jean, 2006).

China

The first Chinese nuclear-powered submarine, Han class, was laid down in 1967 but not completed until 1974. China became the first Asian country and fifth nation in the world after the United States, USSR, United Kingdom and France to build a nuclear propelled submarine. Throughout the 1950s and early 1960s the Soviet Union provided assistance to the People's Liberation Army Navy (PLAN), in the form of naval advisers and export of equipment and technology. A new class of submarine has been in development since the 1980s, when the PLAN first sought a replacement for the Han-class. Type 093 Shang class are believed to have some Russian influence.

The news item in CHINADAILY in April 2016 says that China is planning to build about 20 floating reactor platforms to meet the demand of maritime atomic propulsion. There have been rumors and speculation that China would also switch to nuclear-powered carriers (China Daily, 2016).

India

Although paper studies were ongoing for a number of years, actual work on the land-based prototype started in 1984, to be built in the Atomic Energy complex at Kalpakkam in the state of Tamil Nadu (Anand et al., 2016).

Discussions were held between India and the USSR about possible cooperation and technical support for the nuclear propulsion program. After extensive negotiations, the agreement was concluded in April 1982, conforming to the Nuclear Proliferation Treaty (NPT) and Missile Technology Control Regime (MTCR). The issue of detail design assistance on the reactor was omitted. For gaining operation and maintenance experience, the Soviet Union would lease a nuclear propelled submarine to the Indian Navy before induction of its own nuclear propelled submarine. Physical access to the submarine and its propulsion plant to the designers would be permitted to familiarize them with the layout and space constraints. A new management structure was evolved to design and build a land-based prototype propulsion plant and to design and build future nuclear submarines which included the Navy, DRDO (Defense Research Development Organization), and DAE (Department of Atomic Energy) personnel, similar to the one chosen by the US in late 1940s. The land-based reactor became critical on November 11, 2003 (Anand et al., 2016).

The submarine Arihant was launched in 2009 and underwent 7 years of testing and sea trials before being commissioned in August 2016. On November 4, 2018, the Indian Prime Minister Shri Narendra Modi announced that the Arihant, which means “Slayer of Enemies” in Sanskrit, the Indian Navy’s first domestically-built nuclear-powered submarine, completed Arihant’s first deterrence patrol. Thus, India became the sixth nation to build nuclear powered submarines.

India has already launched a second Arihant-class submarine, the Arihant which is expected to be commissioned soon. Two more of the same class are being built/planned. The Indian Navy is planning to build six nuclear attack submarines, as reported by the press in December 2019.

Brazil

Brazil’s nuclear submarine is being built with French assistance (DCNS, 2010), except for the nuclear reactor and associated uranium fuel-cycle technology, which is being developed by the Brazilian Navy (Acordo, 2008a,b; BBC, 2006; Jobim, 2007; PRO-SUB, 2010). According to open sources, the Brazilian Nuclear Submarine Álvaro Alberto (SN 10), which would be the first of six planned nuclear submarines, would run on a 48–50 MWth reactor. These submarines are a modified version of the Scorpene class diesel-powered submarine (Scorpene 2008, 2010; CORRÊA, 2010; Hecht, 2011; Sarkozy, 2009). While the first four are conventional subs, the fifth will place Brazil in the club of only six nations with nuclear-powered submarines. The program has been delayed by budget cuts and corruption scandals; the launch date of the first submarines has been pushed back until 2029 (Lorient, 2010; de Eliézer Rizzo, 2005).

Present day characteristics

Longer Core life

For the present-day propulsion plants, the effort is to design cores to last the life span of the plant and avoid periodic refueling. One of the main reactor core design challenges is to develop an efficient reactivity control system. Generally, traditional PWRs use a combination of three methods of reactivity control: moveable control rods, burnable neutron absorbers, and soluble neutron absorbers. Earlier, excess reactivity during start-up, due to higher enrichment, was controlled by a soluble poison like Boron in the coolant and/or by using Gadolinium oxide powder mixed with the fuel. This could increase the interval between refueling; the fission gas pressure inside fuel elements was one of the limitations to longer cycles, in addition to corrosion of the cladding. Neutronic investigation of alternative and composite burnable poisons is ongoing, such that the cores can last the lifespan of the plant. In addition to increasing the thickness of the cladding, new material like ZrB₂ is being investigated; the burnable poisons being investigated are Pu 240 or Am 241, mixed homogeneously with the fuel. Only the Russians use lead bismuth alloy as coolant in the fast reactors deployed in some of the submarines.

Quiet operation

The effectiveness of the present-day submarine depends upon its ability to remain undetected for long periods of time while it searches, tracks, or attacks from under the sea. This medium of concealment, however, is advantageous to the submarine only so long as it is not detected or deprived of its ability to detect. Before a submarine can be attacked, it must be detected, and its subsequent positions determined within the requirements of the available weapons system. Detection and position fixing can take place in two ways; there may either be some radiation or reflection of energy from the submarine to the searcher, or else the submarine may disturb one of the natural, static, spatial fields, such as the earth’s magnetic field, and betray its presence. Acoustic energy, while lacking the propagation speed of electromagnetic waves, is capable of being transmitted through sea to distances that are operationally significant and is thus the physical phenomenon used for antisubmarine warfare, underwater communications, and underwater navigation. The optimum use of sound requires a thorough understanding of its limitations so that these effects can be minimized. For example, sea water is not uniform in pressure, temperature, or salinity and all these characteristics have important effects on sound propagation through the sea. The requirement for predicting these effects on sonar performance has become a necessity.

In the nuclear propulsion plant, the cooling water pumps and steam turbines are the main source of noise; these two are absent in diesel electric propulsion, thus diesel electric boats are relatively silent. All effort is focused now to reduce noise levels. Up to a significant power, the coolant was naturally circulating and thus totally silent. At higher speed the reactor is still very quiet, as only low-power pumps are required and the steam generator block filters the noise. The evolution of detection methods implied further reducing acoustic signatures to much lower levels.

Increased electrification of on board services

According to the US Navy's Electric Ships Office (ESO), established in 2007 to develop and provide new types of power capabilities; integrated power and energy systems (IPES) will play an increasingly important role in tomorrow's fleets. Warship designers until now have used hydraulics, pressurized air, and steam to move large masses, such as aircraft catapults, aircraft elevators, and ship propulsion systems, yet new advances in high-power electronic devices may lead to all-electric power aboard surface vessels, even the most demanding systems, such as propulsion and catapults aboard aircraft carriers.

The electromagnetic aircraft launch system (EMALS)

This is an aircraft launching system developed by General Atomics for the United States Navy, to replace the steam catapults. The system employs a linear induction motor rather than the conventional steam piston. EMALS was first installed on the United States Navy's Gerald R. Ford-class aircraft carrier. Its main advantage is that it accelerates aircraft more smoothly with less stress on airframes. Compared to steam catapults, EMALS also weighs less, is expected to be economical, low on maintenance, and can launch both heavy and light aircraft, contrasted to a steam piston-driven system. It also reduces the carrier's requirement of fresh water, thus reducing the demand for energy-intensive desalination. During an aircraft launch, the induction motor requires a large surge of electric power that exceeds what the ship's own continuous power source can provide. As of 1994, the EMALS energy-storage system design accommodates this by drawing power from the ship during its 45-s recharge period and storing the energy kinetically using the rotors of four disk alternators; the system then releases that energy in 2–3 s. In 2015, the first full speed ship-board tests were conducted. In 2017, this catapult launch trial was conducted on the USS Gerald Ford.

AZIPOD® (Azimuthing Podded Drive) propulsion system

This is the electrically driven propulsion with an AC motor incorporated in a streamlined azimuthing pod unit directly driving a fixed-pitch propeller. The motor is controlled by a frequency converter, which produces full nominal torque in either direction over the entire speed range, including standstill. Each Azipod propulsion system is individually designed and optimized to achieve maximum performance. The name "Azipod" is a registered trademark of ASEA Brown Boveri (AZIPOD, 2021) Figs. 5 and 6 show the AZIPOD®.

This is a gearless steerable propulsion system where the electric drive motor is in a submerged pod outside the ship hull. The propeller is a fixed pitch type where the propeller speed is controlled. This design does not need rudders, long shaft lines, or stern transversal thrusters. Direct seawater cooling makes it possible to reduce the motor diameter; a compact motor and pod dimensions lead to a slimmer body and better hydrodynamics.

If the propellers are electrically driven, the motors can be designed for the propeller rotational speed and power can be transferred from one train to another. In addition, since the rotating mass connected to the propellers is considerably smaller, electric drive is less vulnerable to ice or other debris in the water.

Nuclear propulsion of commercial vessels

Worldwide, only four nuclear-power merchant ships have been built since the 1950s and only one is still in operation. These four ships, the *NS ("Nuclear Ship") Savannah* (US), the *Otto Hahn* (Germany), the *Mutsu* (Japan) and the *Sevmorput* (USSR/Russia), were built as demonstration projects and to showcase each nation's technical prowess. All four were considerably smaller than the 100,000–300,000 t vessels now in common use transporting bulk iron ore, coal and grain, as well as intermodal containers.

NS Savannah

The first nuclear-power merchant ship was the *NS Savannah*, designed during the late 1950s and launched July 21, 1959 at Camden, New Jersey (Savannah, 2020). The ship was a product of the US Atoms for Peace Program and included accommodations for 60 passengers as well as interior furnishings to display American design and household technology e.g., a dance floor, swimming pool, futuristic sculptures and a (water-cooled) Raytheon Radarange® microwave oven. The ship was a joint project of the US Atomic Energy Commission, Maritime Administration and Department of Commerce.

With a length of 181.7 m, a beam of 23.8 m and a maximum displacement of 19,800 t (cargo capacity 14,040 t), the ship's 15.1 MW_{shaft} PWR power plant could cruise at 39 km/h (21 knots) with a maximum speed of 44 km/h (24 knots). The range of the *NS Savannah* was 560,000 km on a single core load of 32 fuel elements.

The *NS Savannah*'s Babcock and Wilcox reactor used fuel with an average enrichment of 4.4% U-235 in 32 fuel elements forming a core 1.6 m in diameter and 1.7 m high and generating 74 MW_{th}. The reactor vessel was 5.2 m high, and the containment vessel



Fig. 5 Azimuthing Podded Drive (side view). https://en.wikipedia.org/wiki/File:Mackinaw_WLBB-30_Azipod_thruster.jpg, https://en.wikipedia.org/wiki/File:First_Azipod.jpg. This is a work of the U.S. federal government, it is in the public domain in the United States. Public Domain Mark 1.0.



Fig. 6 Azimuthing Podded Drive (Front View). https://en.wikipedia.org/wiki/File:Mackinaw_WLBB-30_Azipod_thruster.jpg, https://en.wikipedia.org/wiki/File:First_Azipod.jpg. This is a work of the U.S. federal government, it is in the public domain in the United States. Public Domain Mark 1.0.

was 15 m high and 5 m in diameter and designed to withstand the 1.28 MPa pressure of a blow down from a loss of primary coolant.

After launching in 1959, the Savannah required another 30 months to complete installation and fueling of the reactor. Savannah began its maiden voyage August 20, 1962, first calling at Savannah, Georgia, its eponymous home port, and proceeding through the Panama Canal to Hawaii and ports on the West Coast. In 1964 Savannah visited Gulf and East Coast ports before crossing the Atlantic to call at European ports.

Passenger service was discontinued in 1965 and the ship was converted for all cargo use. At the same time the reactor was refueled after 3 years and 560,000 km of travel. Four of the 32 fuel elements were replaced and the other 28 were rearranged ("shuffled") to improve fuel usage. During its active use Savannah traveled 720,000 km (equivalent to 18 voyages around the earth), visited 32 domestic ports and 45 foreign ports, and was toured by 1.4 million people as part of the Atoms for Peace program. At the end of 1971, Savannah was deactivated and began a career as a museum ship, a role it performed at a pier in Baltimore harbor until

September [Savannah \(2019\)](#). The fuel was removed and the primary and secondary coolants, as well as activated internal components of the reactor, were removed. The reactor vessel itself remains in place.

The NS Savannah was a public relations and technical success, though there were operational problems which required the construction of a special service barge for on-site handling of contaminated coolant and the removal of radioactive fuel and core internals.

But Savannah could not compete economically as a cargo vessel. The stylish bow and placement of the superstructure and reactor amidships made the handling of cargo labor-intensive. The crew of 124 specially trained members was expensive. While the fuel and the reactor system performed well, at the time of inactivation in 1971 the price of oil was about \$3 per barrel. Later calculations indicated that an oil price of \$12 per barrel (a price that was doubled within the decade ([Macrotrends, 2020](#); [Inflation, 2008](#))) would have enabled Savannah to operate at a profit.

On September 19, 2019, Savannah was towed from Baltimore to Philadelphia where the U.S. Maritime Administration will keep the ship in “protective storage” for the near future, while planning for pre-decommissioning ([Chesapeake Bay Magazine, 2019](#)). The decommissioning must be completed by December 2031. NS Savannah is a National Historic Landmark ([Fig. 7](#)).

Otto Hahn

The Otto Hahn, named for the German discoverer of nuclear fission, was a trade and research vessel, whose keel was laid August 31, 1963 in Kiel and began service in October 1968. Its primary purpose was to test the feasibility nuclear marine propulsion. Having a length of 172 m, a beam of 23.4 m and a displacement of 17,141 t (standard), Hahn was slightly smaller but heavier than Savannah ([Hahn, 2009](#)). As a research vessel, the Otto Hahn had accommodations for 36 scientists, as well as a conference room and two laboratories.

Her 38 MW_{th} reactor and the steam turbine produced 8.09 MW_{shaft} for its single propeller at a speed of 17 knots. The PWR required fuel enriched to between 3.5% and 6% and contained fuel rods 830 mm long and 10.89 mm in diameter. The 1.7 t of fuel was arranged in 12 elements containing a total of 2810 rods. The lifetime of the core was 900 days at full power, resulting in an average burnup of 23 MW_{th}-d/kg. The primary coolant pressure was 8.3 MPa and outlet temperature was 300 °C. The three helical tube steam generators were inside the reactor vessel and had counter current primary and secondary flows, allowing for a superheat of 33 °C and 3 MPa at the steam outlet.

Otto Hahn was configured to carry passengers and ore. In 1972, after sailing 463,000 km in 4 years of operation, the reactor was refueled.



Fig. 7 NS Savannah. https://upload.wikimedia.org/wikipedia/commons/thumb/d/d7/NS_Savannah_exterior_MD2.jpg/2048px-NS_Savannah_exterior_MD2.jpg, this file is licensed under the Creative Commons Attribution-Share Alike 4.0 International license.

A major difficulty with continued operation of the Otto Hahn was the denial of permission to use the Panama and Suez canals and to visit many ports on a continuing basis. In 1979, after traveling under nuclear power 1.2 million km (equivalent to 30 times around the earth), the ship was deactivated and the reactor and steam turbine propulsion plant were replaced with a conventional Diesel engine.

In 1983, Otto Hahn was recommissioned and leased as the container ship Trophy. The ship served in commercial service under a series of different names until 2009 when it was scrapped at Alang, India.

Mutsu

Although the construction of the Mutsu was contemporaneous with the Savannah and Otto Hahn, the Mutsu had a delayed and controversial history (Mutsu, 2020). Nevertheless, the ship, minus its nuclear reactor, is still in service under another name today.

The Mutsu was ordered in November 1967, laid down in November 1968 and launched June 11, 1969. The reactor was completed on August 25, 1972 and fuel was loaded the following September. Mutsu's length is 130 m, beam 19 m and gross tonnage is 8240 t. The maximum thermal output of the reactor was 36 MW_{th}, producing 7.5 MW_{shaft} for propulsion at 32 km/h.

The original plan was to conduct initial nuclear tests at the pier in Ominato but protests from local fishermen forced a change in the testing site in the Pacific Ocean to 800 km east of Cape Shiriya. At the beginning of these tests on September 1, 1974, when the reactor was at 1.4% of full power, a neutron and gamma shielding deficiency was detected. The cause was streaming of radiation through gaps in the shielding. Westinghouse, in a review of the design, had warned of such a possibility, but the design was not changed. Although there was no significant radiation exposure, Mutsu was blocked from returning to port for 50 days. Later analysis of the situation pointed to the lack of coordination between the two contractors on the project and to conceptual mistakes in the shielding calculations.

From late 1974 until February 1991, Mutsu underwent a series of port changes, modifications and overhauls. Its original design objective was to complete 82,000 km under nuclear power. With that goal accomplished in 1992 Mutsu was decommissioned.

The reactor was removed from the Mutsu in 1995 and the vessel was decontaminated, then rebuilt as the ocean observation vessel RV Mirai, which remains in service today (Mirai, 2020).

Sevmorput

The most recent, and only currently-operating, nuclear commercial vessel is the Sevmorput (from Northern Sea Route—Severn Morskoy Put), whose design was begun by the Ministry of the Merchant Marine and the Ministry of Shipbuilding Industry of the Soviet Union in 1978. The keel was laid in 1982, launching was in 1986 and delivery was in 1988 (Sevmorput, 2020). The Sevmorput was designed to serve shallow ports in the Russian Arctic as an icebreaking LASH (lighter aboard ship) carrier. As such it had the capability of transport 74 shallow-draft barges ("lighters") weighing 300 t each and to on/off-board those lighters near the entrances to shallow ports along the Arctic coast. If it is not transporting lighters Sevmorput can carry 20- and 40-ft cargo containers with a total capacity of 1328 twenty-foot equivalent units (TEU). Its maximum displacement is 61,880 t and its maximum sum of cargo, fresh water and provisions is 33,980 t at maximum draft and 26,840 t when operating at reduced draft for icebreaking. The ship has a length of 260.3 m, a beam of 32.2 m and a maximum draft of 11.8 m (in summer). Therefore, it is about twice the length and three times the displacement of the Savannah.

Sevmorput's reactor is a KLT-40, having a thermal output of 135 MW, and using 150.7 kg of highly enriched uranium in the form of a uranium-zirconium alloy. The maximum power to the single propeller is 29.42 MW_{shaft} at a maximum speed of 20.8 knots (38.5 km/h). Unlike Russian nuclear icebreakers which have nuclear-turbo-electric drive trains, Sevmorput has a mechanically coupled steam turbine driving a single 6.7-m diameter propeller. It can also maintain a speed of 3.7 km/h in 1-m thick ice. The reactor has only been refueled twice since beginning operations in 1989.

Initially Sevmorput was not permitted to enter several of the Siberian ports which it was designed to serve because of popular protests and fears following the Chernobyl accident in 1986. The ship was mainly used on the Murmansk-Dudinka route and also made several voyages to Vietnam in the 1990s. In early 2020 it was reportedly preparing for a supply trip to the Russian research stations in Antarctica.

Future applications of nuclear propulsion

As described in the preceding section, four nuclear-powered cargo ships have been built since the late 1950s by four countries. They were designed as demonstration projects to gain experience in civilian marine propulsion using nuclear power. Generally, the reactors and power trains performed satisfactorily, though all spent more time in repair yards than comparable conventional cargo ships. The major economic issues were due the need for a larger number of highly trained crew members, the lack of public acceptance in some prospective ports and the limited number of locations where repairs and refueling can be conducted.

Very large container carriers

Conventional commercial shipping has also evolved. During the six decades since the *Savannah* was launched, large container ships have taken over almost all non-bulk cargo shipping. Very large low-speed diesel engines have been designed to operate at the propeller RPM, eliminating the need for large reduction gearing. Until recently, sulfur and carbon dioxide emissions from ocean-going vessels have not been an issue.

Among the largest of these recent ships are the Maersk Triple-E class, operated by the Danish firm A. P. Møller-Maersk. At the time the first of the class were delivered in 2013 they were the largest container ships in the world, capable of carrying 18,340 twenty-foot equivalent units (TEU). The 31 Triple-E ships were delivered in two generations, the first 20 between July 2013 and June 2015 and the second generation between April 2017 and January 2019. Their length is 399 m, beam 59 m and draft 16 m, with a maximum displacement of 251,000 t, more than 10 times the maximum displacement of the *Savannah*.

The two main diesel engines on Triple-E vessels produce 31 MW_{shaft} each at 73 RPM and are each coupled directly to a 9.8 m diameter four-bladed propeller. Designed cruising speed is 19 knots, slower than the 23–26 knots for other large container ships. Although these ships are capable of 25 knots, fuel consumption is reduced by about 42% at 19 knots. Cruising at 19 knots typically lengthens voyage times by 2–6 days.

High speed container ships

Beginning with the MV *Boston* in 2006, Maersk built seven “B-class” container ships, with capacities of 4196 TEU for high-speed service between China and the United States. These ships were capable of 68 km/h top speed and cruised at 54 km/h. At cruising-speed the engine power was about 60 MW_{shaft}. The B-class ships were designed to serve a market for cargo shipping faster than the standard container ships at 37 km/h but less expensive than airfreight. Because their fuel consumption was 300 t/day the B-Class ships proved to be uneconomic compared with the larger Maersk vessels in their E-Class and Triple E Class. The downturn in the world economy in 2008–10 meant that the vessels were never used on their intended route.

Very large bulk carriers

Large ships up 200,000–300,000 t carry commodities such as crude oil, coal, metal ores and grain on long oceanic routes. Because the commodities are not as time sensitive as container freight and to reduce fuel consumption, these bulk carriers cruise at 13.5–15 knots. Power requirements are about 15–20 MW_{shaft}, depending on displacement, propeller diameter and engine speed (MAN, 2018).

Atmospheric emissions of shipping

World seaborne trade in 2018 was 11 billion t (UNCTAD, 2018) and emitted 940 million t of carbon dioxide through the use of fossil fuels for propulsion. By comparison, in 2018 worldwide commercial aviation operations emitted 918 million t of carbon dioxide, 2.4% of all CO₂ emissions worldwide (Environmental Impact Aviation, 2020). Commercial aviation is currently under pressure to reduce their greenhouse gas emissions. Seaborne trade may soon face the same pressure.

Conclusion

The four commercial demonstration projects, *Savannah*, *Otto Hahn*, *Mutsu* and *Sevmorput*, showed that nuclear marine propulsion could be technically feasible although staffing and economic issues remain. During the last 50 years, the sizes of typical cargo vessels have increased by at least a factor of 10 and the propulsion power needed has increased to the 30–80 MW_{shaft} range.

During this time of low petroleum prices and little enthusiasm for reducing greenhouse gas emissions, rapid adoption of alternative power sources, particularly of nuclear power, is unlikely. Marine shipping is a very competitive industry and recent ships, such as the Triple-E Class, are finely tuned to the optimum operating point. Nevertheless, over the next 50 years, when the four demonstration ships will approach their centennial, worldwide trade will continue to grow, particularly as the economies of South Asia, Africa and South America develop. Seaborne trade will continue to be the most economical method for crossing oceans. If the many scientific models for global climate change are even approximately correct, there will be increased pressure to reduce Green-House Gas (GHG) emissions with the least impact on the world economy.

The power requirements for large container ships, bulk ore and grain carriers, and high-speed container ships are now in the same general range (30–80 MWe) as some of the small modular reactors (SMR) now being designed (Stanculescu, 2021; Ingersoll, 2021; Slaybaugh et al., 2021). Except for a return to sailing ships, there are no other options beyond small reactors for eliminating the GHG emissions of seaborne shipping. The operation and maintenance costs of SMRs could be dramatically reduced by using artificial intelligence based predictive maintenance and advanced control systems. SMRs can be designed to have inherent safety that will hopefully contribute to public acceptance. Nevertheless, careful design will be necessary in avoiding the public acceptance and economic problems faced by the early demonstration projects.

See Also: Floating Nuclear Plants to Improve Economics, Safety, Siting, and Proliferation Resistance; Floating Nuclear Power Plants With Small Modular Reactors.

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Introduction to Space Nuclear Power and Propulsion

Gary L. Bennett*, Boise, ID, United States

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Abbreviations

AEC U.S. Atomic Energy Commission
DOE U.S. Department of Energy
NASA National Aeronautics and Space Administration
NEP Nuclear Electric Propulsion
NERVA Nuclear Engine for Rocket Vehicle Applications
NTP Nuclear Thermal Propulsion
NTR Nuclear Thermal Rocket
RTG Radioisotope Thermoelectric Generator
SNAP Systems for Nuclear Auxiliary Power

Introduction

Two of the critical subsystems required for any space mission are power and propulsion. Power enables the space system to operate; for example, sensors to function, computers to provide control, and communication equipment to transmit information to and from the space system. Propulsion enables the space system to reach its destination and to orient itself as needed to meet mission objectives (for example, moving into orbit, pointing sensors at the desired targets and pointing antennae for communication back to Earth).

Because of the mass, size and cost constraints on payloads launched from Earth it is very important to have the lowest mass and smallest payloads that can be reasonably built. That, in turn, means getting the highest performance out of every subsystem on the payload (or space system). Nuclear power and nuclear propulsion provide such high performance for a number of proposed missions beyond Earth orbit. The following sections will provide an introduction to nuclear power and nuclear propulsion for space systems. Subsequent chapters in Section 9 will provide additional information as shown in the list below.

Chapter	Title
1	Introduction to Space Nuclear Power and Propulsion
2	Radioisotope Power: Benefits and Challenges
3	Radioisotope Power: Historical Review
4	Radioisotope Power: Technology Options
5	Space Fission Power: Benefits and Challenges
6	Space Fission Power: Historical Review

(Continued)

*NASA Retired.

Chapter	Title
7	Space Fission Power: Technology Options - Reactor Core
8	Space Fission Power: Technology Options - Heat Transport
9	Space Fission Power: Technology Options – Radiation Shielding
10	Space Fission Power: Kilopower and KRUSTY
11	Nuclear Thermal Propulsion: Benefits and Challenges
12	The History of Nuclear Thermal Rocket Development
13	Nuclear Thermal Propulsion: Technology Options
14	Nuclear Electric Propulsion for Rapid Transportation within the Solar System
15	Advanced Propulsion: Pulsed, Fusion, Antimatter and Breakthrough Physics
16	Space Nuclear Power Safety and Safeguards

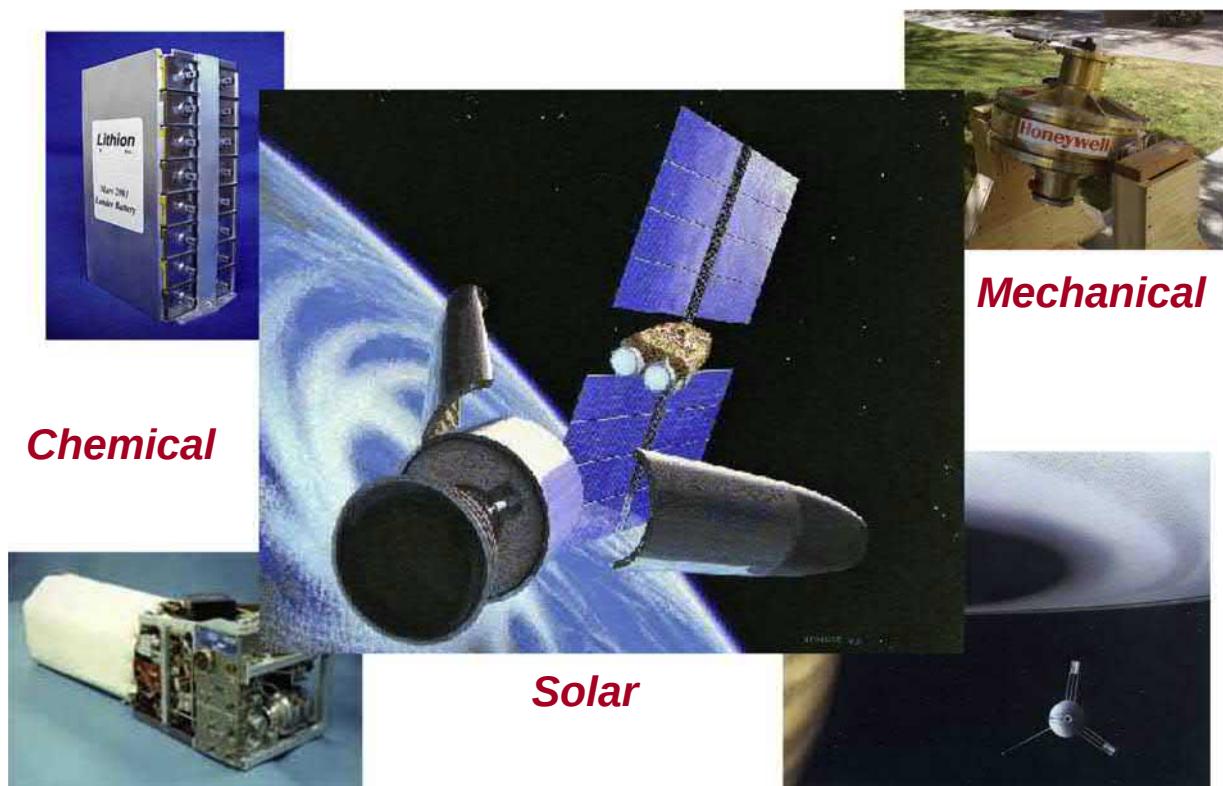
Power sources

Fig. 1 provides some examples of the different ways of providing electrical power to a space system. Fig. 2 illustrates the various technologies in a diagrammatic form.

As illustrated in Fig. 2, most power sources depend upon some source of heat or, in the case of solar power, the “heat” comes in the form of light from the Sun.

Subsequent chapters in Section 9.0 will provide additional information on these conversion systems (e.g., Stirling convertors aka “linear oscillators” and improved radioisotope power source designs).

Types of Spacecraft Power



Sources: <<http://powerweb.grc.nasa.gov/pvseel/>>; <<http://space-power.grc.nasa.gov/ppo/projects/flywheel/img/honeywell-facets-module.jpg>>
<www.grc.nasa.gov/WWW/RT1999/5000/5420manzo.html>; <www.utcfuelcells.com/space/spaceshuttle.shtm>; NASA SP-446

Fig. 1 Types of Spacecraft Power Sources. Source: NASA.

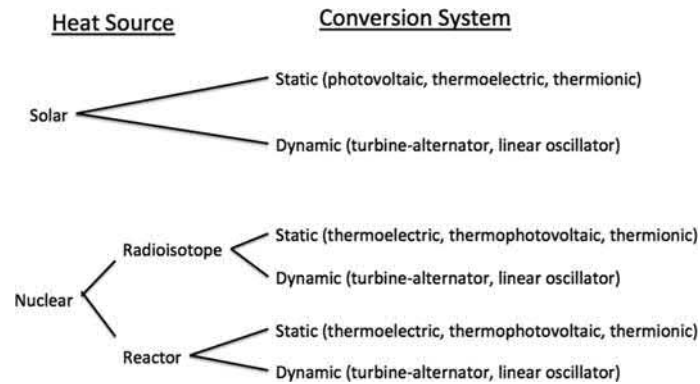


Fig. 2 Power source options for spacecraft. Source: NASA.

Historically, the conversion system of choice for space nuclear power systems (both radioisotope and reactor) has been thermoelectric, which is based on an effect discovered in 1821 by Thomas Johann Seebeck that an electric voltage is produced when two dissimilar metals (“legs”) are joined in a closed circuit and the two junctions are kept at different temperatures. Typically in these thermoelectric conversion systems, the material in the thermoelectric element is a semiconductor with one leg being “positive” and the other being “negative.”

Most space systems are “power hungry,” that is, they can use all of the power they can get. But space systems are also constrained by size (e.g., what will fit into the launch vehicle) and mass (e.g., how much the launch vehicle can carry). Nuclear power sources offer the advantage of providing lots of power in compact packages. This can be seen by considering that solar power or chemical power typically involves the release of electron volts (eV) of energy to create an electrical current whereas nuclear power sources release millions of electron volts of energy.

Fig. 3, which is drawn from U.S. Department of Energy and NASA studies, illustrates the qualitative regimes of space power sources in terms of power required and duration of the mission. In **Fig. 3**, the term “Nuclear Fission” refers to the use of the fissioning of a material (such as uranium-235) to produce heat that can then be converted into usable electric power. The term “Solar” refers to the use of solar cells on arrays to convert sunlight into usable electric power. The practicality of solar power is highly dependent on the application. The “Solar at Jupiter” region indicates the power level where solar power might be used under special orbital conditions (e.g., avoiding the Jovian radiation belts); noting that the power level would substantially smaller when further out in the solar system (note that the level of degradation will depend on radiation effects). The dashed lines on **Fig. 3** represent the power levels where solar power might be the most practical choice at other locations. The dashed line for “Moon or Mars surface” includes the effect of eclipses, dust storms, etc., but will vary by specific location and/or application. The term “Chemical” refers to the use of batteries or fuel cells to produce power (note that chemical systems have limited lifetimes; eventually they need to be recharged” using a power source such as nuclear or solar).

Both nuclear and solar power sources have been used on the same mission (for example, the early Transit Navy navigational satellites). And chemical energy (i.e., batteries) has been used to provide peak power on some nuclear and solar missions (such as the Viking Mars Landers).

In summary, nuclear power sources offer superior performance in those long-duration regimes where there is little or no sunlight (such as outer-planet missions) or where there are significant environmental threats (such as the radiation belts around Jupiter). (Chemical systems lack the lifetime to be the primary power source but they can supplement nuclear or solar power sources where more power is needed for short periods of time.) The long-duration performance can be seen with the RTGs of the two Voyager spacecraft which have been providing electrical power for over 43 years (cf. [Bennett, 2021](#)).

Nuclear power sources

As shown in **Fig. 2**, there are several ways in which heat energy can be converted into usable electrical energy. The heat from a radioisotope heat source or a nuclear reactor heat source may be converted into electrical energy by means of (1) dynamic conversion such as turbine-alternators (as is often done in terrestrial power plants) or linear alternators; or (2) by static conversion such as with thermoelectric elements, thermionic converters or thermophotovoltaic elements. To date, all U.S. space nuclear power sources that have been flown have used thermoelectric conversion.

Radioisotope power sources

Traditionally, radioisotope power sources (or, sometimes termed radioisotope power systems) have used static conversion (e.g., thermoelectric conversion) or (in ground experiments) dynamic conversion. Both types will be discussed in the next sections.

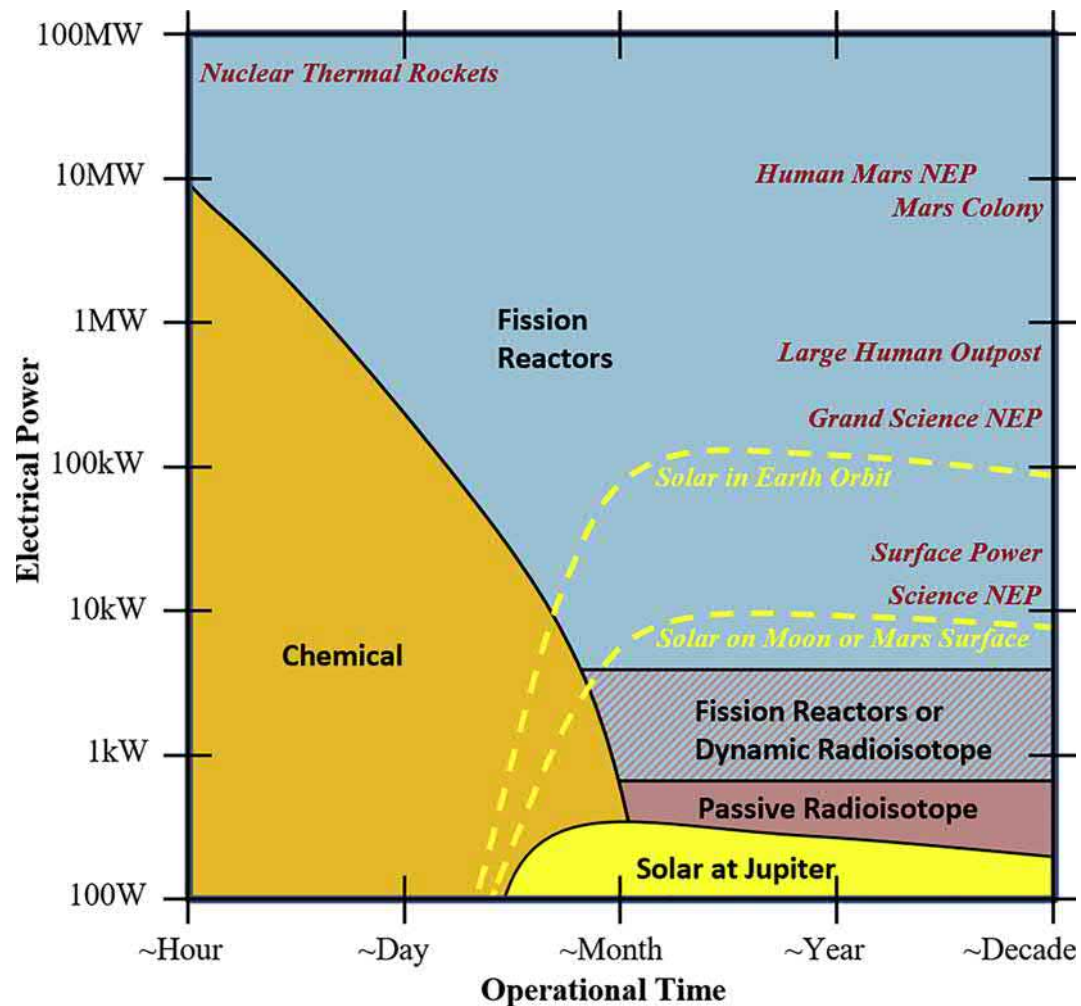


Fig. 3 Qualitative regimes of space power applicability. Source: U.S. DOE and NASA studies and David Poston studies.

Radioisotope thermoelectric generators (RTGs) (Bennett, 1995; Buden, 2011a; Corliss and Harvey, 1964)

Thermoelectric conversion is illustrated diagrammatically in Fig. 4 for a radioisotope thermoelectric generator (RTG). An RTG basically consists of two major components: the heat source (which contains the radioisotope fuel) and the converter (which converts the heat from the heat source into usable electrical power).

As shown in Fig. 4, Step 1, the radioisotope fuel decays spontaneously emitting particles that produce heat when the particles are absorbed in the materials of the RTG. All of the U.S. RTGs flown to date have used the radioisotope plutonium-238 (Pu-238 or ^{238}Pu) as the fuel. Plutonium-238 decays mostly by emitting alpha particles (essentially the nuclei of helium atoms). The alpha particles are absorbed in the heat source producing heat that can elevate the temperature over a range up to about 1300 K as needed by the converter. Fig. 5 shows a plutonium-238 oxide fuel pellet that has been self-heated through the use of an insulator causing it to glow in a darkened room. (Without such self-heating, the pellet would have a dull gray appearance.)

In Step 2, the heat is converted directly into electrical power by being absorbed in the thermoelectric elements. The electrical power is then sent to the spacecraft for management and distribution (Step 3).

While obviously an RTG must operate reliably or it won't be used, the first objective in designing and using an RTG is to ensure that it is safe, that is, that the fuel does not come in contact with humans. As noted earlier, size and mass are also critical—the smaller and lighter the better for space systems where every kilogram launched into space costs.

Thermoelectric conversion is based on the “thermoelectric effect” which was discovered in 1821 by Thomas Johann Seebeck. The idea of using the thermoelectric effect to produce electrical power from the heat of radioisotopes was developed by John Birden and Ken Jordan who were posthumously inducted in 2013 into the U.S. National Inventors Hall of Fame.

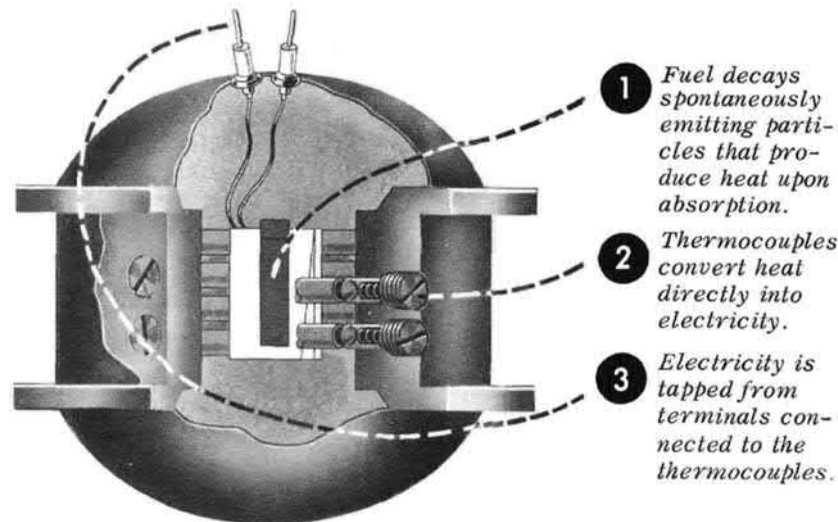


Fig. 4 How a radioisotope thermoelectric generator (RTG) works. Source: U.S. Atomic Energy Commission.

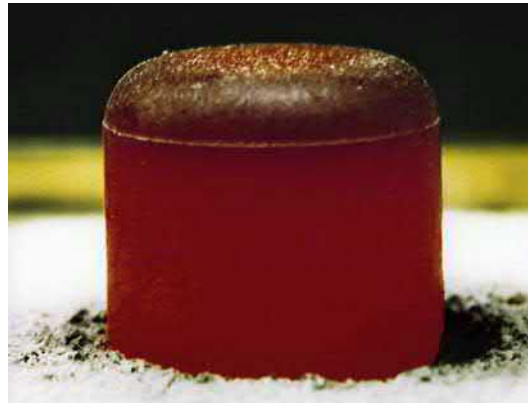


Fig. 5 Plutonium-238 oxide fuel pellet glowing after having self-heated. Source: Los Alamos National Laboratory.

Radioisotope dynamic power systems

The first U.S. radioisotope power source, termed SNAP-1 (for Systems for Nuclear Auxiliary Power One), used dynamic conversion. The objective of SNAP-1 (shown in Fig. 6) was to provide 500 watts of electrical power (We) for a 60-day mission.

The dynamic power conversion systems that have been studied (e.g., Brayton, Rankine, Stirling) operate through a changing magnetic field which induces a current in a conductor. The magnetic field can be changed by rotating a magnet (e.g., Brayton and Rankine) or by causing it to oscillate (e.g., Stirling). The induced current then becomes the source of electrical power. (The induced electromotive force associated with the electric current is a result of Faraday's Law.) Efficiencies in the order of 20–30% or more are possible (compare to the 5–10% efficiencies of thermoelectric conversion).

In the SNAP-1 concept, a heat source fueled with cerium-144 was to heat a mercury fluid, vaporizing it in the boiler and sending it through a miniature turbine-alternator system to produce 500 We of electrical power at an thermal-to-electric efficiency of about 9.4% which compared favorably to the (then) 5% efficiency of RTGs. (The program did not advance to the point of actually using cerium-244 which has a half-life of about 285 days and decays primarily by emitting beta particles (electrons)). The technical term for the conversion cycle is Rankine named after William Johnson Macquorn Rankine who studied the thermodynamic properties of steam. While the missions in the late 1950s did not require the high power of SNAP-1, a larger version of the mercury Rankine conversion system was used in the SNAP-2 space nuclear reactor.

In the 1970s, the Rankine technology was resurrected in the US for radioisotope power sources such as the Kilowatt Isotope Power System (KIPS), shown diagrammatically in Fig. 7. Instead of mercury, KIPS employed Dowtherm A (an organic eutectic mixture of biphenyl and biphenyl ether) as the working fluid to produce about 1400 We at a system efficiency of about 16.6%. The advantage of higher efficiency is that less of the expensive Pu-238 fuel is needed to produce the same electrical power and it may enable a lower mass radioisotope power source.

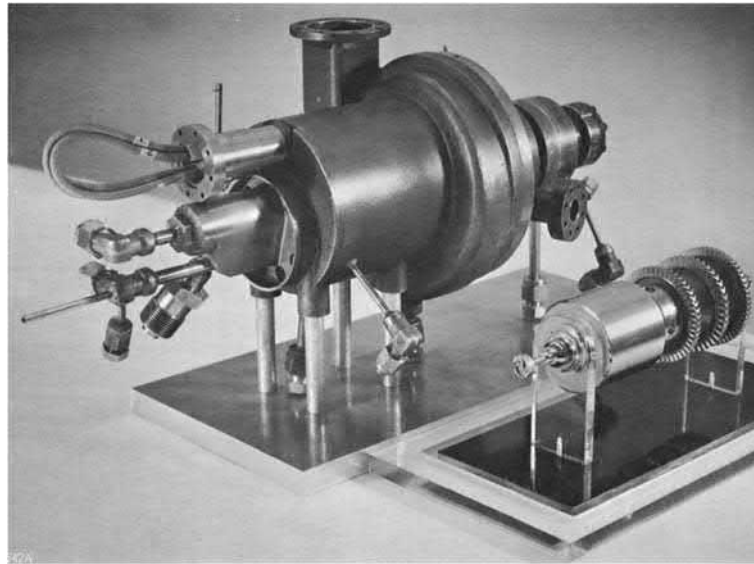


Fig. 6 SNAP-1 turbomachinery package with shaft assembly shown separately. Source: TRW and USAEC.

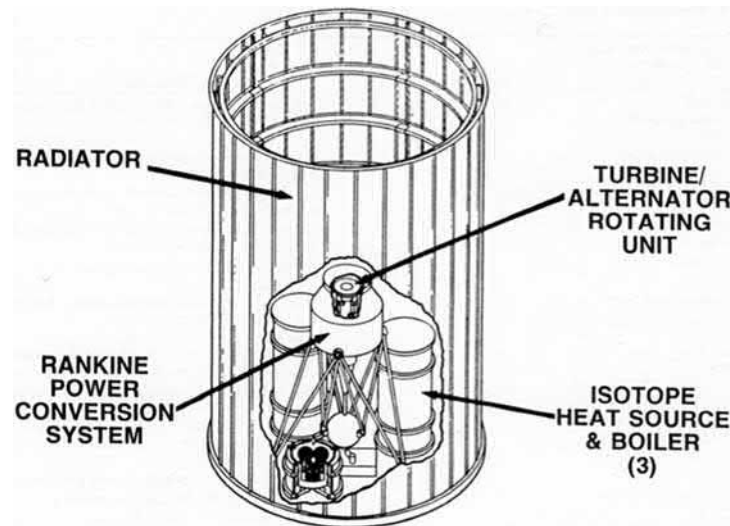


Fig. 7 Diagram of the Kilowatt Isotope Power System (KIPS) which used a Rankine cycle power conversion system. Source: U.S. Department of Energy.

Another dynamic conversion system that has been studied for both radioisotope and nuclear reactors is the Brayton system (named for George Brayton) which uses an all-gas working fluid with no liquid-to-vapor phase change. In tests run in the late 1970s under the Brayton Isotope Power System program a 25% thermal-to-electric conversion efficiency was demonstrated.

A third dynamic conversion option is the Stirling conversion cycle (named for Robert Stirling) which employs a linear oscillator. NASA has studied the Stirling conversion system for both radioisotope and reactor power sources.

Nuclear reactor power sources (Buden, 1994, 2011a; Voss, 1984)

For those missions requiring large amounts of power (10s of kilowatts and higher) and/or which will operate in a hostile environment (e.g., limited or no sunlight, high radiation fields, etc.), a nuclear reactor is a very competitive option, if not the only viable option. (In the same situation but for lower powers a radioisotope power source is competitive.) The overall concept of a nuclear reactor power source (shown diagrammatically in Fig. 8) is similar to that of radioisotope power sources, i.e., the reactor produces heat which is transmitted to a conversion system that converts the heat (often called thermal power) into usable electrical power. Unused or “waste” heat is rejected often by means of fins that allow the waste heat to be radiated to space. However, whereas the

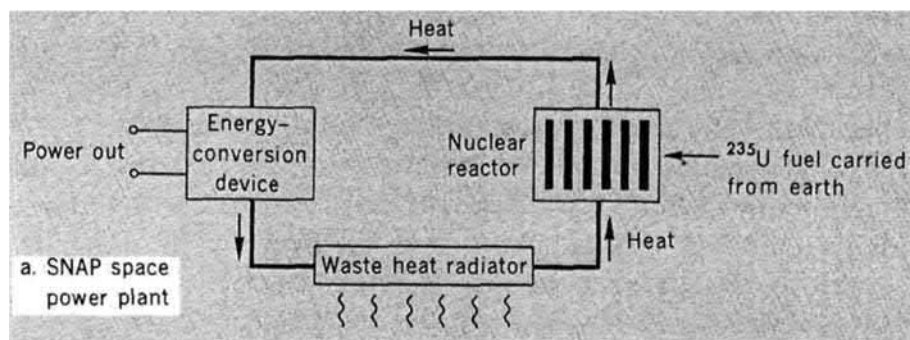


Fig. 8 Diagram of a generic nuclear reactor power source showing the flow of heat from the nuclear reactor to the energy conversion device. Any heat not used to produce electricity is radiated away to space via the waste heat radiator. Source: U.S. Atomic Energy Commission.

radioisotope power source produces heat from the radioactive decay of the radioisotope fuel, the nuclear reactor produces heat from the fissioning (“splitting”) of the reactor “fuel” (typically uranium-235 which is also written as U-235 or ^{235}U).

While a radioisotope heat source produces a source of thermal power (which decays at a known rate) without human intervention, a nuclear reactor can produce a variable source of thermal power through the reactor control system. Under the SNAP reactor program, studies (either analytical or experimental) were undertaken on several conversion cycles: thermoelectric (e.g., SNAP-10 and SNAP-10A), Rankine (e.g., SNAP-2, SNAP-4 and SNAP-8), Brayton (e.g., SNAP-8, Advanced Hydride Reactors and the Advanced Liquid-Metal-Cooled Reactor) and thermionic (the in-core thermionic reactor). (Staub 1973; Voss 1984) Thermoelectric conversion was the choice for the U.S. SP-100 space nuclear reactor power system program. More recent studies have focused on using a Stirling conversion cycle. The reader is referred to Chapters 5 through 10 for more information on the reactor concepts and their conversion systems.

The U.S. has only launched one space nuclear reactor, the SNAP-10A reactor launched on 3 April 1965 to become the world’s first nuclear reactor space power system. The objective of the reactor flight test was to produce 500 We under automatic control in space. The reactor initially produced more than 600 We. After 43 days of successful operation, the reactor was shutdown as the result of a high voltage failure sequence in the electrical system of the Agena spacecraft. (A ground test twin operated successfully for over a year.) Fig. 8 shows a technician standing next to a SNAP-10A reactor (not operating with nuclear power!).

Fig. 9 illustrates the various elements of the SNAP-10A reactor and Fig. 10 is an artist’s concept of the SNAP-10A reactor with its Agena upper stage in orbit (Fig. 11).

The overall diagram of the SNAP-10A space reactor power system is illustrated in Fig. 12.

In the 1980s and early 1990s, the U.S. worked on the SP-100 space nuclear reactor power system illustrated as a flight module in Fig. 13.

More recently, NASA and the U.S. Department of Energy (DOE) have been studying a kilowatt-class nuclear reactor termed “Kilopower” which could be used to power a lunar base or a Mars base using Stirling conversion (see illustration in Fig. 14).

Separately, the former Soviet Union flew over 30 nuclear reactors, mostly to power radar ocean surveillance satellites.

Nuclear propulsion

Humanity depends on propulsion systems to send its scientific and human missions where they want to go. Like space power systems, it is extremely important that the propulsion system have high performance and low mass. Nuclear propulsion offers just such benefits thanks to the inherently high energy content of nuclear fission (the splitting of atomic nuclei). In this section two types of nuclear propulsion will be considered: nuclear thermal propulsion (NTP) and nuclear electric propulsion (NEP).

One of the key attributes of any propulsion system is its specific impulse, that is, the amount of impulse (measured in units of force/thrust multiplied by time and divided by the mass of the thruster). In the metric system, the units are meters per second (m/s). One can think of specific impulse as analogous to gas mileage in an automobile—the higher the specific impulse the better.

Nuclear thermal propulsion (NTP) (Bennett et al., 1994; Buden, 2011b; Bussard and DeLauer, 1965; Dewar, 2004)

Chemical rockets, which have launched most of humanity’s payloads to date, operate by creating a hot gas through a chemical reaction and then expelling that hot gas out of the nozzle of the rocket engine. This usually means carrying two chemical propellants: the fuel which “burns” and the oxidizer that allows the fuel to “burn”. The nuclear thermal rocket replaces the two chemical propellants with one propellant that is heated by a nuclear reactor as illustrated in Fig. 15.

In principle, a nuclear rocket can produce a specific impulse twice that of a chemical propulsion system (and maybe more than that). This translates into a greatly reduced mass that must be carried into low-Earth orbit (LEO) for human missions to the Moon and Mars. Beginning in the mid 1950s through the 1960s, the U.S. tested some 20 reactors as part of its Rover/NERVA program (see Chapters 11–12). (NERVA is an acronym for Nuclear Engine for Rocket Vehicle Applications.) Fig. 16 shows a comparison of the NERVA reactor with a human.



Fig. 9 SNAP-10A mockup and technician. Source: Atomics International.

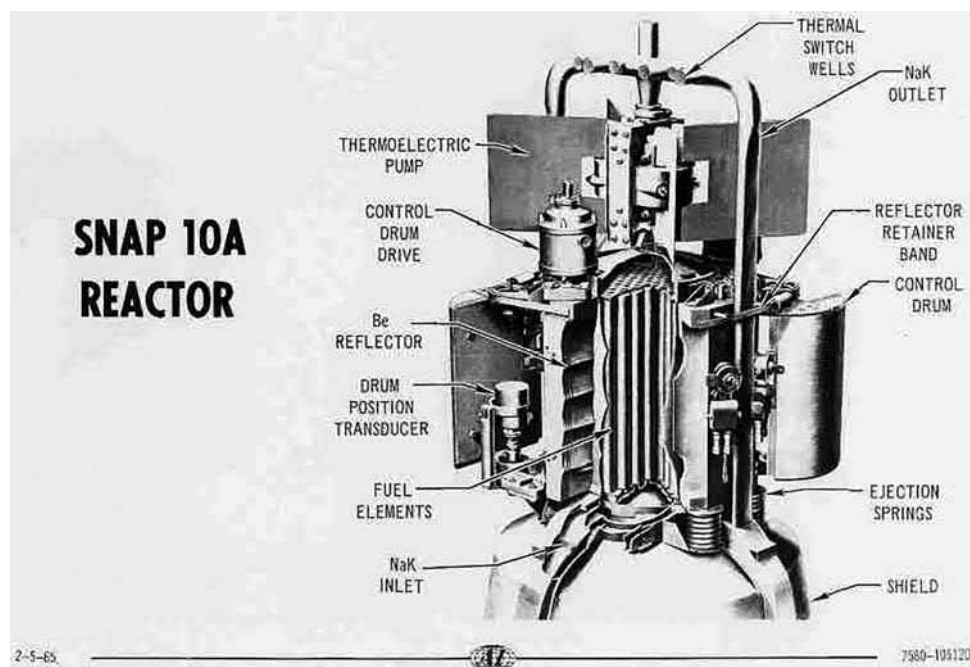


Fig. 10 Cutaway of the SNAP-10 nuclear reactor showing the uranium-235 fuel elements and the thermoelectric pump which moved the sodium-potassium (NaK) reactor coolant to the thermoelectric modules. Source: Atomics International and AEC.

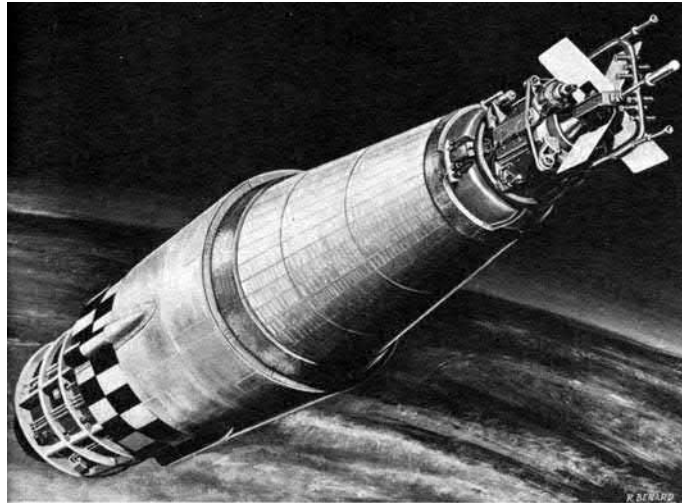


Fig. 11 Artist's conception of the SNAP-10 nuclear reactor space power system which was launched on 3 April 1965. This was the world's first operation of a nuclear reactor in space. The reactor is the assembly at the right end of the space vehicle. The thermoelectric converters are in the cone-shaped structure next to the reactor. Source: AEC.

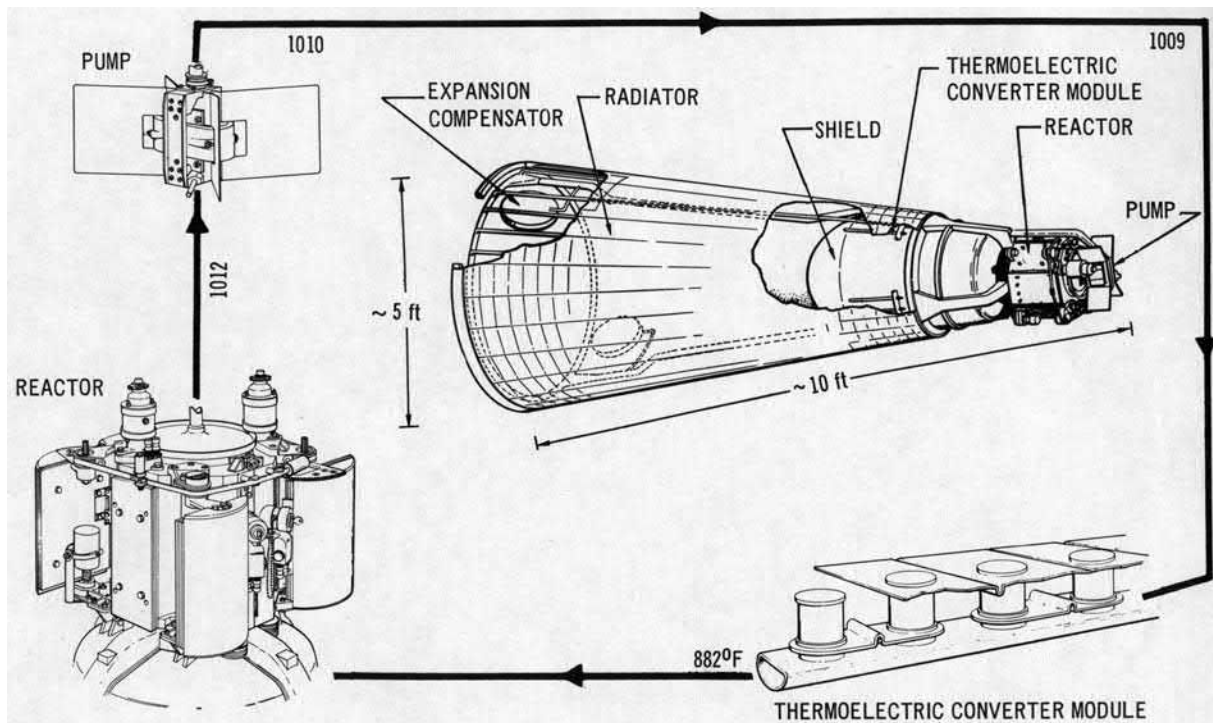


Fig. 12 Diagram of the SNAP-10A space nuclear reactor power system showing the reactor, the pump, the thermoelectric converter module and the radiator. Source: Atomics International.

Fig. 17 is an artist's illustration of a nuclear thermal rocket orbiting over Mars.

NASA, in cooperation with the U.S. Department of Energy, is continuing to study nuclear thermal propulsion for human missions to Mars and the Moon (see Chapters 13 and 14).

Nuclear electric propulsion (NEP)

Nuclear electric propulsion (NEP) is another way of utilizing the high performance of nuclear reactors to move spacecraft. Instead of heating a propellant (such as hydrogen), the nuclear reactor would provide electrical power to electric thrusters which would

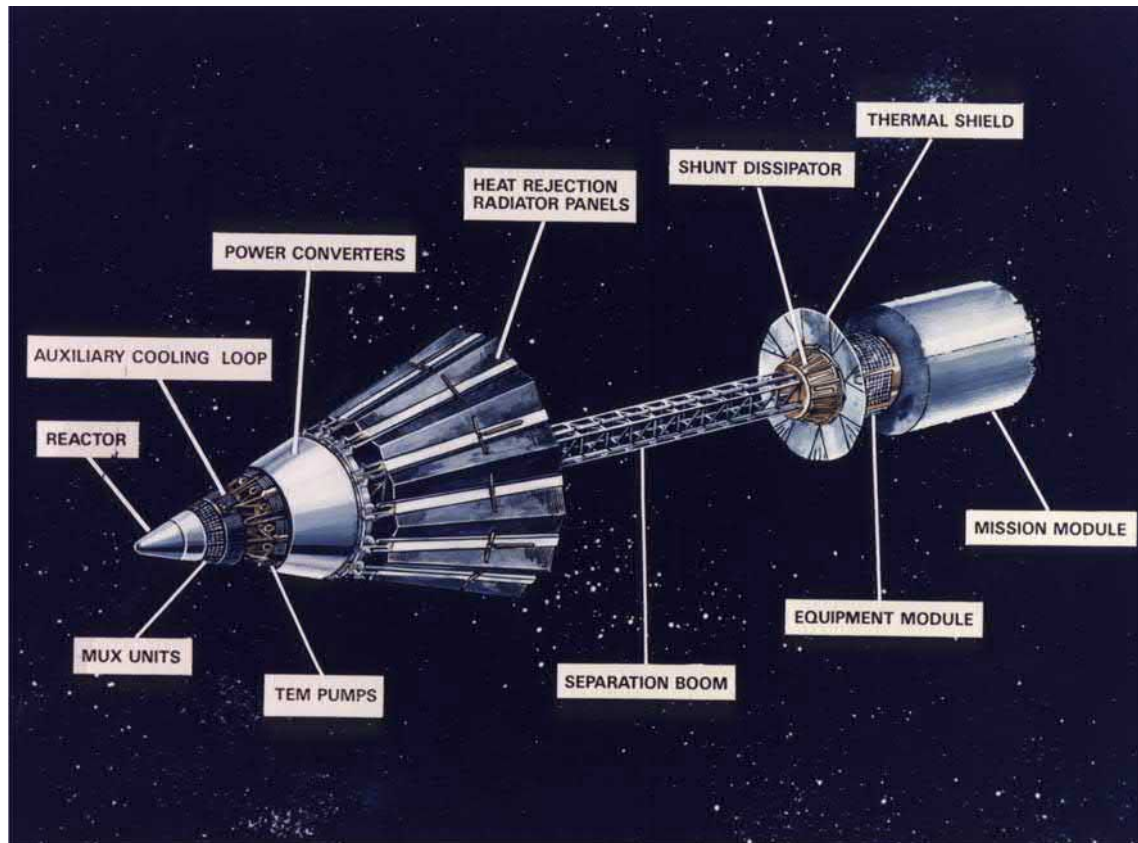


Fig. 13 SP-100 space reactor power system module. The nuclear reactor is shown at the left near the thermoelectric power converters. The mission module is shown at the right and attached to the reactor by means of a separation boom which helps reduce the nuclear radiation. MUX, Multiplexer; TEM, Thermoelectric ElectroMagnetic. Source: U.S. Department of Energy.

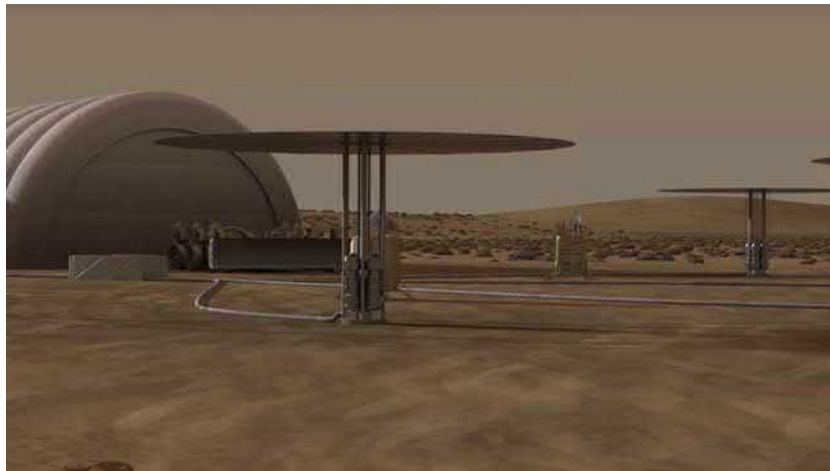


Fig. 14 Artist's concept of Kilopower nuclear reactors on Mars. The nuclear reactor and the Stirling conversion subsystem are shown at the base while the radiator is shown as a flat disk at the top. Source: NASA.

provide the propulsive force. In short, this is basically combining the space nuclear reactor technology discussed earlier with electric propulsion (see Chapter 14). A conceptual schematic of an NEP vehicle is shown in Fig. 18.

Studies of NEP have considered a range of electric thrusters such as ion engines, Hall-effect thrusters and magnetoplasmadynamic (MPD) thrusters. In a sense, NEP has already been flown—the SNAP-10A reactor flight test described earlier included an ion thruster that was operated briefly. (Chapter 15 provides more information on nuclear electric propulsion.)

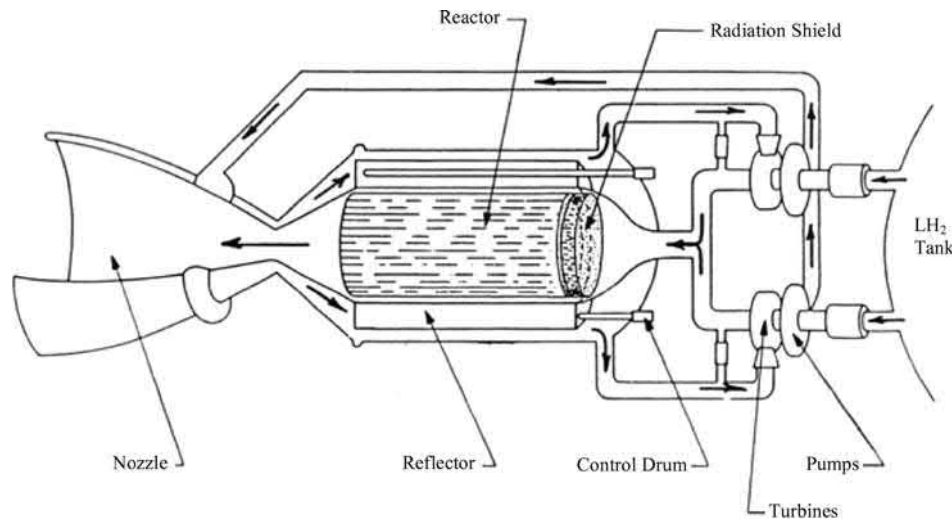


Fig. 15 Diagram of a nuclear thermal rocket (NTR). Liquid hydrogen is pumped from the tank on the right into the nuclear reactor where it is heated and then expelled through the nozzle. Since a nuclear rocket only needs one propellant (unlike chemical propulsion systems which need two) the mass of the ejected hydrogen propellant is less and the velocity of the hydrogen is much higher. Source: NASA and U.S. Atomic Energy Commission.



Fig. 16 NERVA mockup. (NERVA stands for Nuclear Engine for Rocket Vehicle Applications.) Source: Aerojet Nuclear Systems Company.

In theory, larger versions of NEP vehicles with much more power could be designed, large enough to permit cargo and human missions to Mars. Because of the low thrust of electric propulsion thrusters, the NEP vehicle would have to spiral away from Earth (no direct transfer such as with nuclear rockets or chemical rockets) but once the NEP vehicle has reached escape velocity in principle the transit time to Mars can be approximately equivalent to the higher thrust options. The advantage of NEP is that the electric thrusters have very high specific impulses, perhaps 10–20 times that of chemical rockets; hence, NEP offers a very fuel efficient method of transportation.

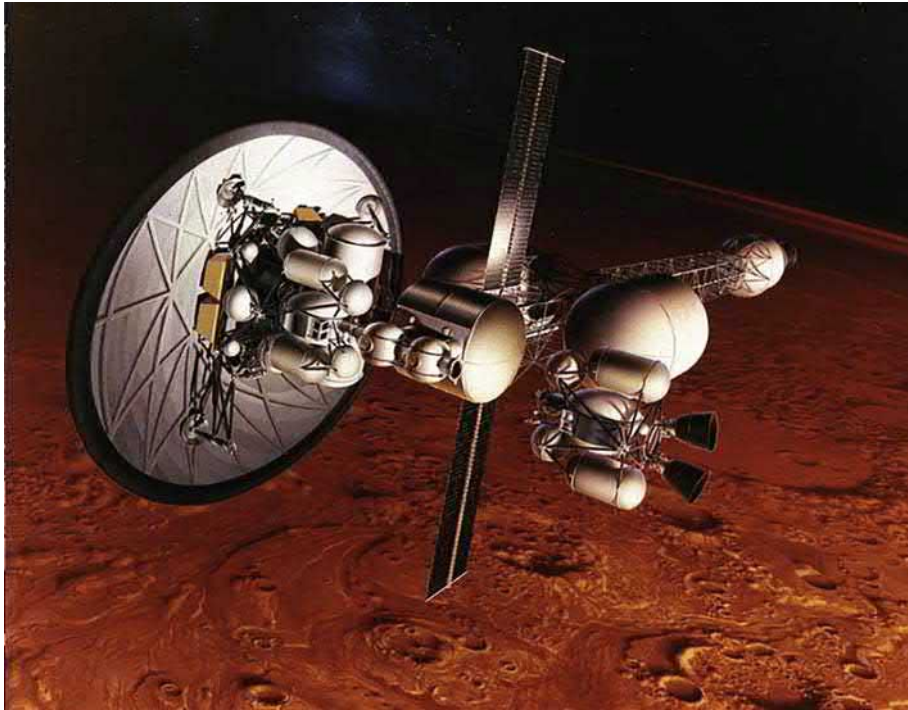


Fig. 17 Artist's illustration of a nuclear rocket over Mars. The nuclear rocket is at the far right. The aeroshell for the crew descent vehicle is at the far left. Note the inclusion of solar panels to provide power for the human spacecraft. Source: NASA/Marshall Space Flight Center.

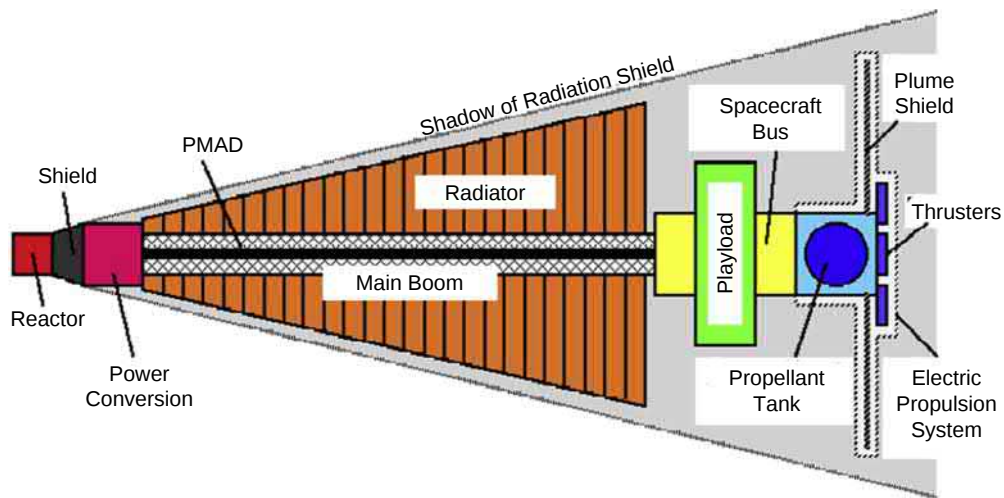


Fig. 18 Conceptual Schematic of a Nuclear Electric Propulsion (NEP) Vehicle. The nuclear reactor is shown on the left next to the power conversion unit. The payload and propellant are shown on the right. Source: JPL.

Fig. 19 shows an artist's concept of a nuclear electric propulsion spacecraft going to Mars.

Concluding remarks

Both nuclear power and nuclear propulsion have strong technical bases thanks to years of study and experiment. Both technologies address the critical need for energy-rich power and propulsion.



Fig. 19 Nuclear electric propulsion vehicle concept for a Mars mission. Three megawatt-class nuclear reactors are mounted on three booms. The ion thruster plume is shown in blue. Source: NASA/Glenn Research Center.

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Radioisotope Power: Benefits and Challenges

Gary L. Bennett*, Boise, ID, United States

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Abbreviations

RHU Radioisotope heater unit

RTG Radioisotope thermoelectric generator

USAEC U.S. Atomic Energy Commission (1947–74)

USDOE U.S. Department of Energy (1977-present)

Introduction

Radioisotope power systems provide electrical power or thermal power by using the natural radioactive decay of certain isotopes. These isotopes are unstable in the sense that they emit radiation; hence they are termed “radioisotopes” (Previtali, 2021). Typically the radiations emitted may be alpha particles (nuclei of helium atoms), beta particles (electrons) or gamma rays (high energy packets of energy). In certain cases (such as spontaneous fission), neutrons may be emitted.

The different radiations emitted vary in their capability to penetrate materials. For example, alpha particles are easily stopped such as with sheets of paper. Beta particles can penetrate more material and gamma rays are typically the most penetrating of the three. Ideally, for a radioisotope power source, the radiation emitted should be contained within the power source itself and therefore gamma-emitting radioisotopes are not generally used for space RTGs. Table 1 lists some radioisotope fuels.

Mission and power designers in the U.S. have historically selected plutonium-238 as their preferred radioisotope. More recently, European researchers have been investigating americium-241.

An important criterion in selecting a radioisotope for power is what is termed its half-life, that is, the time it takes for the radioisotope to decay to half of its original amount of radioactivity. For long-lived space missions (such as the two Voyager spacecraft which have operated for more than 43 years), the longer the half-life the better.

The long half-life of plutonium-238 has enabled the two Voyager spacecraft to complete their planetary missions (first spacecraft to visit Jupiter, Saturn, Uranus and Neptune) and to enter interstellar space (on 25 August 2012 for Voyager 1 and on 05 November 2018 for Voyager 2). With the benefit of long-lived radioisotope power, the Voyager spacecraft have become humanity’s envoys to the stars. Fig. 1 is an artist’s concept of Voyager 1 in interstellar space.

Fig. 2 illustrates diagrammatically the radioactive decay of the radioisotope plutonium-238.

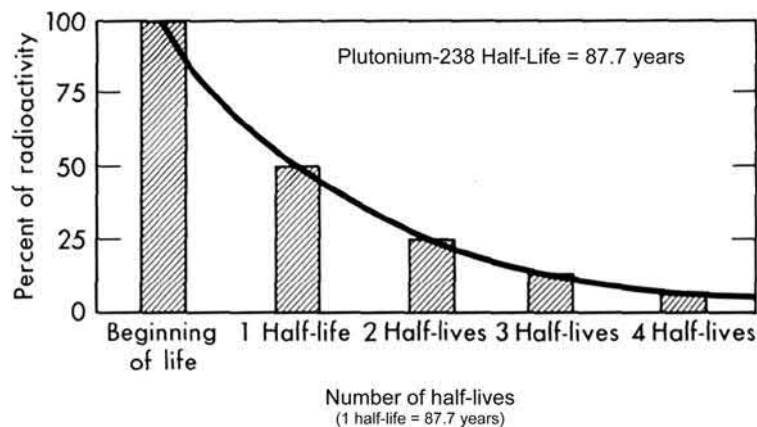
When the emitted radiation is absorbed in the material of the radioisotope power source, heat is released. That heat can be used to power a conversion system that converts heat to electrical power analogous to the way terrestrial power plants use a source of heat (e.g., coal, gas, nuclear) and a thermal-to-electric conversion system to produce electrical power. Fig. 3 illustrates the thermal power by showing a ~ 63-watt plutonium-238 oxide fuel pellet that has been allowed to self-heat by insulating it to drive the temperature up to hundreds of degrees kelvin then removing the insulation while viewing the pellet in a darkened room.

For the types of radioisotope power sources that have been studied and/or used in space missions, there are two principal types of thermal-to-electric power conversion: static (“direct”) and dynamic. In the static conversion systems there are no moving parts, i.e., the thermal power (heat) is directly converted into electrical power. For dynamic conversion systems, moving parts such as turbine-alternators or linear alternators are employed to produce the electrical power (similar to what happens in terrestrial power plants).

*NASA retired.

Table 1 Some candidate radioisotope fuels.

Radioisotope fuel	Half-life (years)	Initial power density (100% basis) (watts/g)	Major radiations
Americium-241	432.2	~0.1	Alpha
Cobalt-60	5.24	17.8	Gamma, beta
Strontium-90	28.6	0.933	Beta, gamma, bremsstrahlung
Polonium-210	0.38	144	Alpha
Plutonium-238	87.6	0.56	Alpha
Curium-244	18.1	13	Alpha

**Fig. 1** Voyager 1 spacecraft entering interstellar space. Courtesy of Artist's conception, NASA/JPL.**Fig. 2** Diagram of the radioactive decay of plutonium-238 which has a half-life of approximately 87.7 years. Courtesy of U.S. Atomic Energy Commission.

Radioisotope thermoelectric generators (RTGs) (Buden, 2011; Cataldo and Bennett, 2011; Corliss and Harvey, 1964; Corliss and Mead, 1971)

With one exception, all the nuclear-powered U.S. spacecraft have used radioisotope thermoelectric generators (RTGs). As shown in Fig. 4, an RTG consists of two basic elements: (1) a radioisotope heat source (which, as its name suggests, produces heat from the absorption of the radiation emitted by the radioisotope) and (2) the converter which directly converts the heat into usable electrical power.

In all the U.S. RTG-powered spacecraft flown to date, the radioisotope used to produce the heat is plutonium-238, which has a half-life of 87.7 years. This is sufficiently long for long-duration space missions such as the two Voyager spacecraft which were launched in 1977 and are still operating after having made flybys of Jupiter, Saturn, Uranus and Neptune. The primary radiations emitted by plutonium-238 are alpha particles which are largely absorbed in the heat source itself thereby producing the heat.

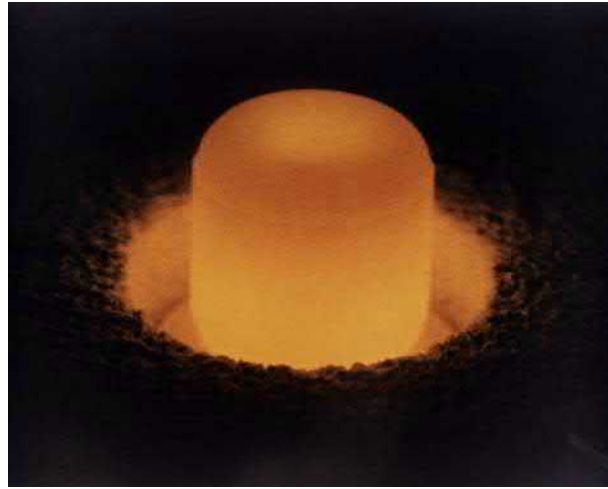


Fig. 3 Self-heating of the fuel pellet for the General-Purpose Heat Source. Radioisotope Thermoelectric Generator (GPHS-RTG). The GPHS-RTG contains 72 of these fuel pellets. (The pellet is ~ 27.5 mm in diameter and 27.6 mm in length. The mass of a pellet is ~ 151 g.)

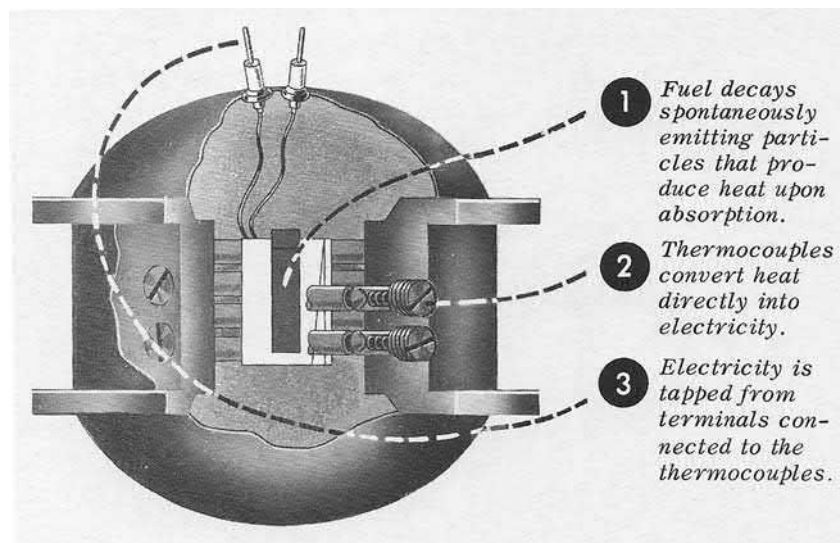


Fig. 4 Diagram of a radioisotope thermoelectric generator (RTG). Courtesy of U.S. Atomic Energy Commission.

Production of plutonium-238 begins at the U.S. Department of Energy's Idaho National Laboratory (INL), which stores the existing inventory of neptunium-237 feedstock and ships it as needed to the U.S. Department of Energy's Oak Ridge National Laboratory (ORNL). Engineers mix the neptunium oxide with aluminum and press the mixture into high-density pellets. They use the High Flux Isotope Reactor, a DOE Office of Science User Facility at ORNL, to irradiate the pellets, creating neptunium-238, which quickly decays and becomes plutonium-238. The plutonium radioisotope fuel is on the order of 84% plutonium-238 with the rest of the plutonium mostly plutonium-239.

The irradiated pellets are then dissolved and ORNL staff use a chemical process to separate the plutonium from remaining neptunium. The plutonium product is converted to an oxide and shipped to DOE's Los Alamos National Laboratory, where the material will be stored until needed for a mission. Remaining neptunium is recycled into new targets to produce more plutonium-238.

In an RTG, the heat from the decay of the radioisotope is transmitted to the converter by conduction (direct contact) or radiation (similar to the heat one feels from a fireplace). The converter in an RTG consists of many thermoelectric elements which operate on principles similar to the thermocouples that have been used to measure temperature. The thermoelectric element consists of a "hot shoe" which directly absorbs the heat and a "cold shoe" which radiates any unused heat into space (see Fig. 5). The principle is derived from that discovered by Thomas Johann Seebeck more than 200 years ago: if two dissimilar metals are connected at two junctions, one hotter than the other, there will be an electrical voltage (hence a current) produced. Efficiencies of thermoelectric conversion have historically been less than 10%, that is, less than 10% of the thermal power ("heat") is converted into usable electrical power (Bennett, 1989).

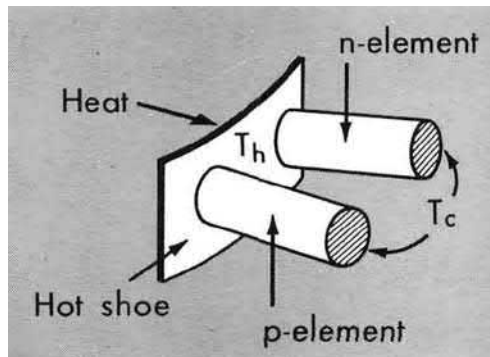


Fig. 5 Diagram of a thermoelectric element. Courtesy of U.S. Atomic Energy Commission.

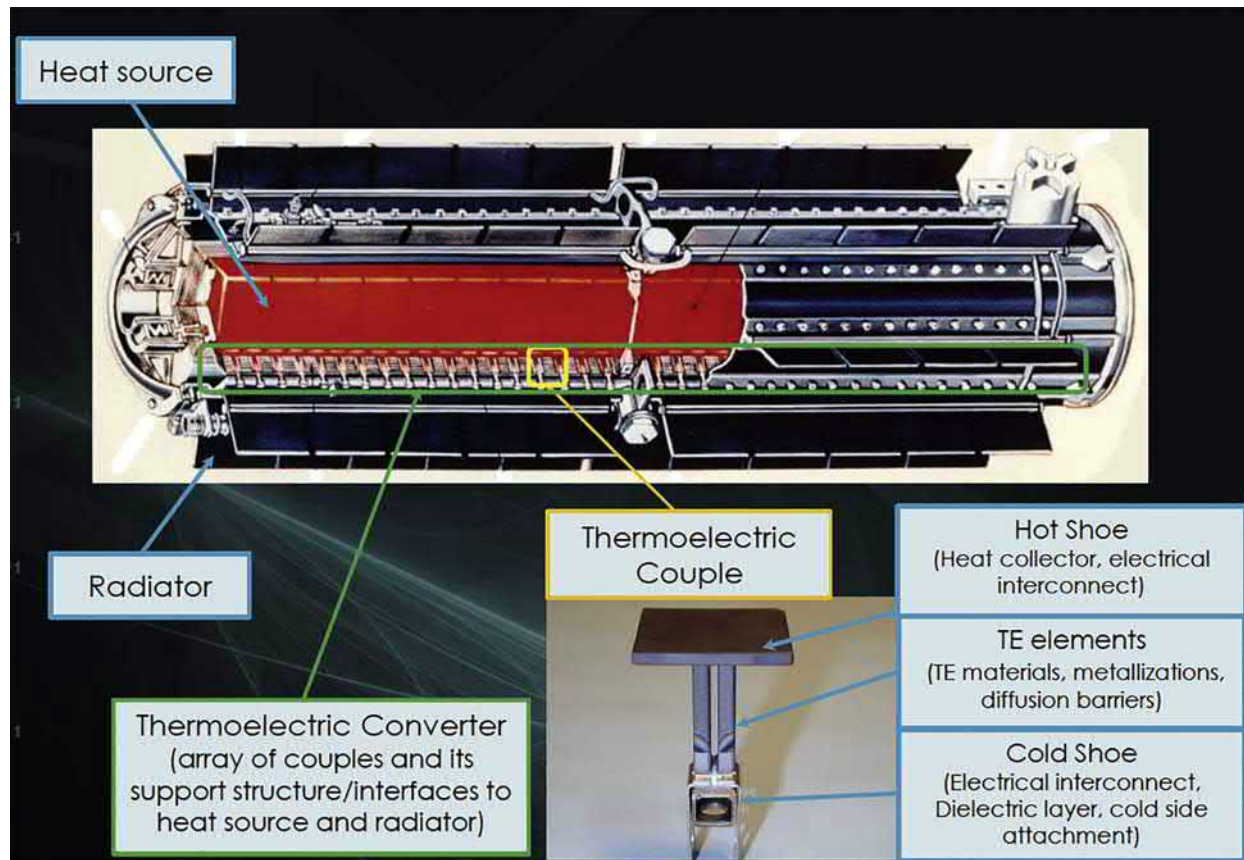


Fig. 6 Cutaway of the General-Purpose Heat Source Radioisotope Thermoelectric Generator (GPHS-RTG). The GPHS-RTG is capable of producing 300 We at beginning of life. The GPHS-RTG has an overall diameter of 0.422 m and a length of 1.14 m with a mass of about 55.9 kg. Courtesy of DOE/JPL.

Typical thermoelectric materials used in U.S. RTGs have been lead-telluride and silicon-germanium. The silicon-germanium RTGs have had a power decay (mostly the result of the decay of the plutonium-238 heat source) of about 1.6% per year. The lead-telluride RTGs have had a decay about twice that.

Fig. 6 illustrates how all of these parts come together in the General-Purpose Heat Source Radioisotope Thermoelectric Generator (GPHS-RTG) which powered the Galileo orbiter of Jupiter, the Ulysses solar polar orbiter, the Cassini orbiter of Saturn and now the New Horizons spacecraft that has traveled to Pluto and beyond.

Table 2 lists the space nuclear power sources that have been flown by the U.S. With one exception (SNAP-10A) they have all been radioisotope power sources. The former Soviet Union has flown at least 35 space reactors and at least 4 radioisotope power sources. European researchers have written about radioisotope power sources fueled with americium-241 and space reactors.

Table 2 Uses of space nuclear power by the United States.

(Spacecraft/year launched/type of nuclear power source^a/beginning-of-mission power)

Note: SNAP is an acronym for systems for nuclear auxiliary power
MHW-RTG = Multi-hundred watt radioisotope thermoelectric generator
GPHS-RTG = General-purpose heat source radioisotope thermoelectric generator
MMRTG = Multi-mission radioisotope thermoelectric generator

Transit navy navigational satellites

- Transits 4A and 4B (1961) SNAP-3B (2.7 We)
- Transits 5BN-1 and 5BN-2 (1963) SNAP-9A (> 25 We)
- Transit TRIAD (1972) Transit-RTG (35 We)

SNAPSHOT space reactor experiment

- SNAP-10A nuclear reactor (1965) (~500 We)

Nimbus-3 meteorological satellite

- SNAP-19B RTGs (1969) (2 @ 28 We each)

Apollo lunar surface experiments packages

- Apollos 12 (1969), 14 (1971), 15 (1971), 16 (1972), 17 (1972) SNAP-27 (> 70 We each)

Lincoln experimental satellites (communications)

- LES 8 and LES 9 (1976) MHW-RTG (2/spacecraft @ ~154 We each)

Interplanetary missions

- Pioneer 10 (1972) and Pioneer 11 (1973) SNAP-19 (4/spacecraft @ ~40 We each)
- Viking Mars Landers 1 and 2 (1975) SNAP-19 (2/Lander @ ~42 We each)
- Voyager 1 and Voyager 2 (1977) MHW-RTG (3/spacecraft @ ~158 We each)
- Galileo (1989) GPHS-RTG (2 @ ~288 We each) (>3-year delay)
- Ulysses (1990) GPHS-RTG (289 We) (>4-year delay)
- Cassini (1997) GPHS-RTG (3 @ ~295 We each)
- New Horizons (2006) GPHS-RTG (1 @ 245.7 We) (most of fuel >21 years old)
- Curiosity Mars Science Laboratory (2011) MMRTG (1 @ ~110 We)
- Perseverance Rover (2020) MMRTG (1 @ ~110 We)

^aThe odd-numbered SNAP power sources used radioisotope heating while the even-numbered SNAP power sources used nuclear reactors to produce the heat.

Radioisotope heater units

Another use for radioisotopes on spacecraft is the production of heat to keep sensitive instruments warm without introducing the potential for electrical interference from an electric heater. Such sources of heat are termed radioisotope heater units (RHUs). An advantage of an RHU is that it doesn't take critical power away from the spacecraft power source to produce the heat.

Fig. 7 shows the parts of a one-watt thermal RHU termed the "Light-Weight Radioisotope Heater Unit (LWRHU)" which has been flown on such missions as the Galileo mission to orbit Jupiter, the Cassini mission to orbit Saturn and on various Mars rovers. The LWRHU produces 1 watt of thermal power ("heat").

Benefits of radioisotope power

Radioisotope power sources are attractive for use inspace under a number of conditions:

Lifetime. Radioisotope power sources are a principal alternative for spacecraft which must operate for a long period of time. (As noted earlier, the two Voyager spacecraft have successfully operated for over 43 years).

Environment. Radioisotope power sources are less vulnerable to external radiation (e.g., Van Allen radiation belts) and to other potentially hostile environments (e.g., meteoroids, Martian dust storms, extreme temperatures (such as experienced on the lunar surface during the lunar day)). (When Pioneer 10 flew past Jupiter it was exposed to a radiation dose on the order of 10^5 rads from electrons and 10^6 rads from protons. The RTGs operated perfectly throughout the jovian encounter).

Self-Sufficiency. Radioisotope power sources make the spacecraft more independent. In this regard, radioisotope power sources have an advantage in being able to operate on the launch pad for full-up systems checkouts prior to launch.

Operational. Radioisotope thermoelectric generators have achieved extremely high reliability. They provide a compact, good power-to-mass (specific power) source of electrical power. The comparatively small exposed area of radioisotope power sources minimizes the radar cross section, attitude control, and structural considerations. Radioisotope heater units can be used to supply heat directly to sensitive spacecraft components without electromagnetic interference.



Fig. 7 Light-weight radioisotope heater unit (LWRHU). Courtesy of DOE/NASA.

Challenges of radioisotope power

In the U.S., the principal challenge facing users of radioisotope power sources is obtaining sufficient quantities of the radioisotope plutonium-238 to provide the necessary thermal power ("heat"). In the late 1980s, the U.S. ceased its main production of plutonium-238 which has caused the U.S. Department of Energy (DOE) to develop alternative sources of plutonium-238.

Related to availability is the issue of cost. In the early days of the radioisotope power program, the users (e.g., NASA or the U.S. Department of Defense) were charged appropriately considering the charter of the U.S. Atomic Energy Commission (predecessor to DOE) to encourage research and use of nuclear power.

The limited supply of plutonium-238 coupled with the cost has caused renewed attention on developing more efficient conversion systems such as "dynamic" conversion (Stirling linear alternators and Brayton and Rankine turbine-alternators). Dynamic conversion systems offer the potential of efficiencies in the range of 20–30% (instead of the 5–10% of static systems) which means a great reduction in the amount of plutonium-238 that must be used.

Some researchers have investigated alternatives to plutonium-238 (such as americium-241 and polonium-210) but these radioisotopes also require a nuclear production infrastructure.

See Also: Origins of Nuclear Energy; Radioactive Decay.

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Radioisotope Power: Historical Review

Gary L. Bennett, Consultant (Formerly with NASA and DOE), Boise, ID, United States

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Abbreviations

AEC	US Atomic Energy Commission (1947–1974)
ALSEP	Apollo Lunar Surface Experiments Package
ARC	Ames Research Center
BOM	Beginning of Mission
DOE	US Department of Energy
EASEP	Early Apollo Experiments Package
ESA	European Space Agency
GPHS	General-Purpose Heat Source
GSFC	Goddard Space Flight Center
JHU/APL	Johns Hopkins University Applied Physics Laboratory
JPL	Jet Propulsion Laboratory
MHW-RTG	Multi-Hundred Watt Radioisotope Thermoelectric Generator
MMRTG	Multi-Mission Radioisotope Thermoelectric Generator
NASA	National Aeronautics and Space Administration
RTG	Radioisotope Thermoelectric Generator
SNAP	Systems for Nuclear Auxiliary Power
SwRI	Southwest Research Institute
We	Watts of electrical power
Wt	Watts of thermal power

Introduction

Electrical power is essential for the operation of a spacecraft. When that spacecraft must travel far from the Sun or operate for longer periods in hostile environments (such as the long lunar night or the dust storms of Mars) then nuclear power becomes enabling. In response to the need for long-lived, Sun-independent power, the US Atomic Energy Commission (AEC) initiated studies in the 1950s of both space nuclear reactors and space radioisotope power sources. These early concepts were designated by a SNAP number where SNAP stands for Systems for Nuclear Auxiliary Power. Even-numbered SNAP power sources were reactors and odd-numbered SNAP power sources were radioisotope power sources.

K. C. Jordan and J. H. Birden, two researchers working at what was then the AEC's Mound Laboratory, built the first radioisotope thermoelectric generator in 1954 by coupling the thermal power of a radioisotope to thermoelectric elements ("thermocouples"). What Jordan and Birden achieved in 1954 defines the two basic elements of an RTG: a radioisotope heat source and a thermoelectric converter. Fig. 1 shows these two basic elements and their ancillary parts for the current RTG known as the Multi-Mission RTG (MMRTG).

All of the US RTGs flown in space have used the radioisotope plutonium-238 to provide the heat. With a half-life of 87.7 years, plutonium-238 is an excellent choice to power long-lived spacecraft. The chemical and metallurgical form of the plutonium-238 has evolved over the years. For current uses, the plutonium-238 is in the form of a high-temperature ceramic oxide. (The production of plutonium-238 is discussed in Bennett, 2021.)

An RTG basically consists of two components: the heat source (which contains the plutonium-238 and associated containment structures) and the converter. The converter in an RTG consists of an arrangement of thermoelectric couples or elements composed of two legs: a positive-type leg ("p-leg") and a negative-type leg ("n-leg"). In many of the RTGs, the same material was used for both the p-leg and the n-leg, the difference coming from the use of different dopants to produce the p-legs and the n-legs. Like the heat source, the chemical and metallurgical form of the thermoelectric elements has evolved over the years from telluride-based elements in the early SNAPs to an alloy of silicon-germanium. The efficiency with which the thermal power ("heat") is converted into electrical power has varied from around 5% to almost 7%. With the advent of the MMRTG for Mars surface missions, the design reverted to telluride-based thermoelectric elements. Research continues to develop higher efficiency thermoelectric elements as well as dynamic conversion systems such as Brayton, Rankine and Stirling.

Table 1 lists the RTGs flown by the United States. In addition to the 42 US RTGs flown, the United States also flew the first space nuclear reactor (SNAP-10A) in 1965. The former Soviet Union reportedly has flown RTGs on two Earth-orbiting spacecraft (Cosmos 84 and Cosmos 90) and on two lunar rovers (Lunokhod-1 and Lunokhod-2), mostly using polonium-210 as the radioisotope. For the aborted Mars 96 mission, the lander stations carried RTGs and RHUs fueled with plutonium-238. The major effort by the former Soviet Union was reportedly on space nuclear reactors with about 31 (and perhaps 33) BUK thermoelectric reactors used for marine radar observations. Two experimental thermionic reactors (designated "TOPAZ" not to be confused with the misnamed "TOPAZ-II" Russian reactor tested in the United States) were flown on Earth-orbiting satellites Cosmos 1818 and Cosmos 1867. European researchers have studied the use of RTGs, notably powered by americium-241. Chinese researchers have developed RTGs fueled with plutonium-238 as demonstrated with their Chang'e 4 lunar mission.

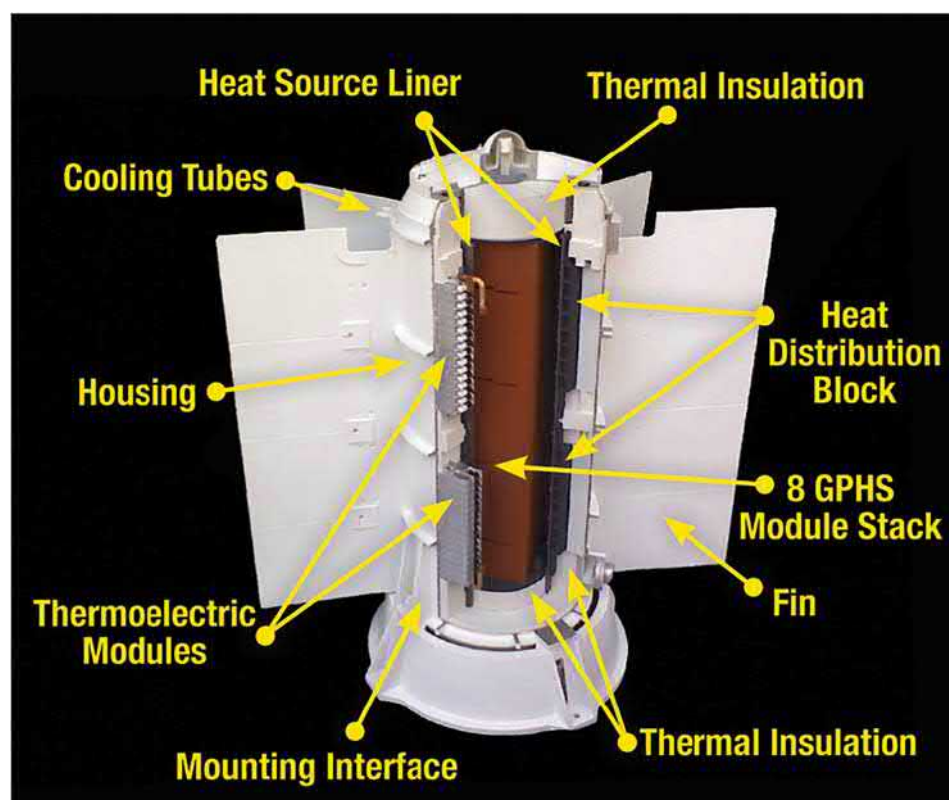


Fig. 1 Multi-Mission Radioisotope Thermoelectric Generator (MMRTG). The two basic elements of an RTG are the radioisotope heat source and the thermoelectric converter. The heat source is illustrated in red and labeled "8 GPHS module stack" where "GPHS" stands for General-Purpose Heat Source. Source: DOE/NASA.

Table 1 Uses of radioisotope thermoelectric generators (RTGs) by the United States (spacecraft/year launched/type of RTG/beginning-of-mission power).

Transit Navy Navigational Satellites
• Transits 4A and 4B (1961) SNAP-3B (2.7 We)
• Transits 5BN-1 and 5BN-2 (1963) SNAP-9A (≥ 25 We)
• Transit TRIAD (1972) Transit-RTG (35 We)
Nimbus-3 Meteorological Satellite
• SNAP-19B RTGs (1969) (2 @ 28 We each)
Apollo Lunar Surface Experiments Packages
• Apollo 12 (1969), 14 (1971), 15 (1971), 16 (1972), 17 (1972)
• SNAP-27 (> 70 We each)
Lincoln Experimental Satellites (Communications)
• LES-8 and LES-9 (1976) MHW-RTG (2/spacecraft @ ~ 154 We each)
Interplanetary Missions
• Pioneer 10 (1972) and Pioneer 11 (1973) SNAP-19 (4/spacecraft @ ~ 40 We each)
• Viking Mars Landers 1 and 2 (1975) SNAP-19 (2/Lander @ ~ 42 We each)
• Voyager 1 and Voyager 2 (1977) MHW-RTG (3/spacecraft @ ~ 158 We each)
• Galileo (1989) GPHS-RTG (2 @ ~ 288 We each) (> 3 -year delay in launch)
• Ulysses (1990) GPHS-RTG (289 We) (> 4 -year delay in launch)
• Cassini (1997) GPHS-RTG (3 @ ~ 295 We each)
• New Horizons (2006) GPHS-RTG (245.7 We) (most of fuel had > 21 years decay)
• Curiosity Mars Science Laboratory (2011) MMRTG (~ 115 We) (> 2 -year delay)

Transit Navy navigational satellites

Researchers at the Johns Hopkins University Applied Physics Laboratory (JHU/APL) developed the technique of using the Doppler shifting of telemetry signals from orbiting spacecraft for navigation. Beginning with the Transit-4A and Transit-4B satellites in 1961, JHU/APL included RTGs on board six navy navigational satellites to provide auxiliary power to supplement the solar power. The prime contractor for all but one of these RTGs was The Martin Company (Nuclear Division), which is now part of Teledyne Energy Systems. The US Atomic Energy Commission (now part of DOE) provided these RTGs to JHU/APL for the US Navy. (Of the six RTG-powered Transit satellites, one, Transit-5BN-3, failed to achieve orbit on April 21, 1964.)

Fig. 2 is a photograph of a technician holding a SNAP-3B RTG of the type used on the Transit-4A and Transit-4B navigational satellites.

Fig. 3 is a cutaway of the SNAP-3B RTG system which also illustrates how an RTG works. The SNAP-3B RTG provided about 2.7 We of electrical power from 52.5 Wt of thermal power in an envelope 12.1-cm in diameter by 14-cm high and a mass of 2.1 kg. The thermal-to-electric conversion was achieved by 27 spring-loaded, series-connected pairs of lead telluride thermoelectric elements with an efficiency slightly over 5%.

1. The radioisotope fuel decays spontaneously emitting alpha particles that produce heat upon absorption
2. Thermoelectric elements (thermocouples) convert heat directly to electrical power
3. Electrical power is tapped from terminals connected to the thermoelectric elements (the outer shell acts as a radiator to remove waste heat).

Fig. 4 illustrates how the SNAP-3B RTG was installed on a Transit-4 navigational satellite. Overall, both SNAP-3B RTGs were extremely successful and more than met their five-year lifetime requirement.



Fig. 2 SNAP-3B RTG. Source: US Atomic Energy Commission.

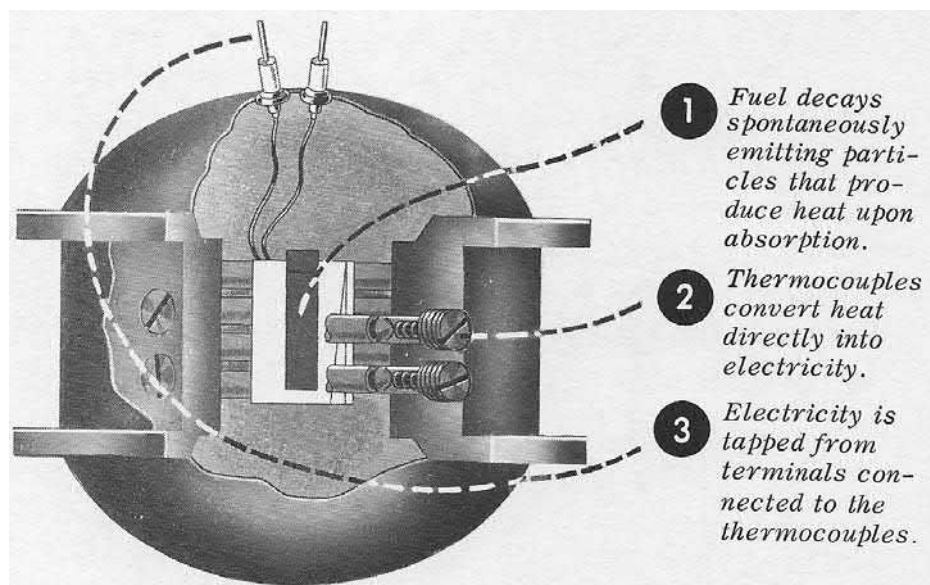


Fig. 3 Cutaway view of a generic radioisotope thermoelectric generator (RTG) (based on the SNAP-3B design) and diagram of how an RTG functions. Source: US Atomic Energy Commission.



Fig. 4 Paul J. Dick installing a SNAP-3B RTG on a Transit Navy navigational satellite (1961). Source: US Atomic Energy Commission.

For the follow-on Transit 5-BN series of navigational satellites, a higher-powered RTG was needed. A new RTG designated SNAP-9A was designed and fabricated to produce 25 We at a nominal 6 V for 5 years in space after 1 year of storage on Earth.

Fig. 5 is a photograph of a SNAP-9A RTG which had 70 pairs of series-connected lead-telluride-based thermoelectric couples assembled in 35 modules of two couples each. The couples were spring-pressed against the heat source block which produced about 525 Wt. Some of the waste heat from the SNAP-9A RTGs was used to maintain electronic instruments within the Transit 5 BN satellites at a temperature near 293 K and the rest of the heat was radiated into space.

The Transit 5BN-1 and Transit 5 BN-2 satellites were the first totally nuclear powered satellites. They gave the US Navy a power source resistant to radiation that enabled US Navy ships to navigate anywhere in the world.

For the JHU/APL TRIAD satellite, a different RTG, designated the TRANSIT RTG, was employed. **Fig. 6** is a cutaway of the TRANSIT RTG showing the heat source which radiated to the lead-telluride-based thermoelectric couples. The TRANSIT RTG was designed to produce 30 We after 5 years. While an electronic failure on the TRIAD satellite precluded obtaining RTG power data, the continued operation of the satellite over the decade or more after the 1972 launch showed that the TRANSIT RTG was providing the required power.

Nimbus-3 meteorological satellite

Under the overall management of NASA's Goddard Space Flight Center (GSFC), the Nimbus-3 meteorological satellite, which was launched on April 14, 1969, was NASA's first use of nuclear power on a spacecraft. (An earlier launch designated Nimbus-B1 on May 18, 1968 had to be aborted and the RTGs recovered.) For the Nimbus-3 mission, the AEC provided an evolved power source termed SNAP-19 which grew out of the SNAP-9A development program. **Fig. 7** is a photograph of the two Nimbus-3 SNAP-19 RTGs.

The two SNAP-19 RTGs enabled the Nimbus-3 satellite to operate in a full-time duty cycle. This opened the way for NASA to use RTGs in future missions.

Early Apollo Lunar Surface Experiments Package (EASEP)

For the Apollo 11 mission (the first human landing on the Moon) the astronauts assembled a lunar experiments package powered by solar cells and kept warm with two 15-Wt radioisotope heater units (RHUs). An RHU is basically a container of a radioisotope



Fig. 5 Photograph of a SNAP-9A RTG of the type used on the Transit-5BN satellites. The SNAP-9A RTG was designed to produce 25 We in an envelope of 26.7-cm high by 50.8-cm across the fins and a mass of about 12.3 kg. Source: US Atomic Energy Commission.

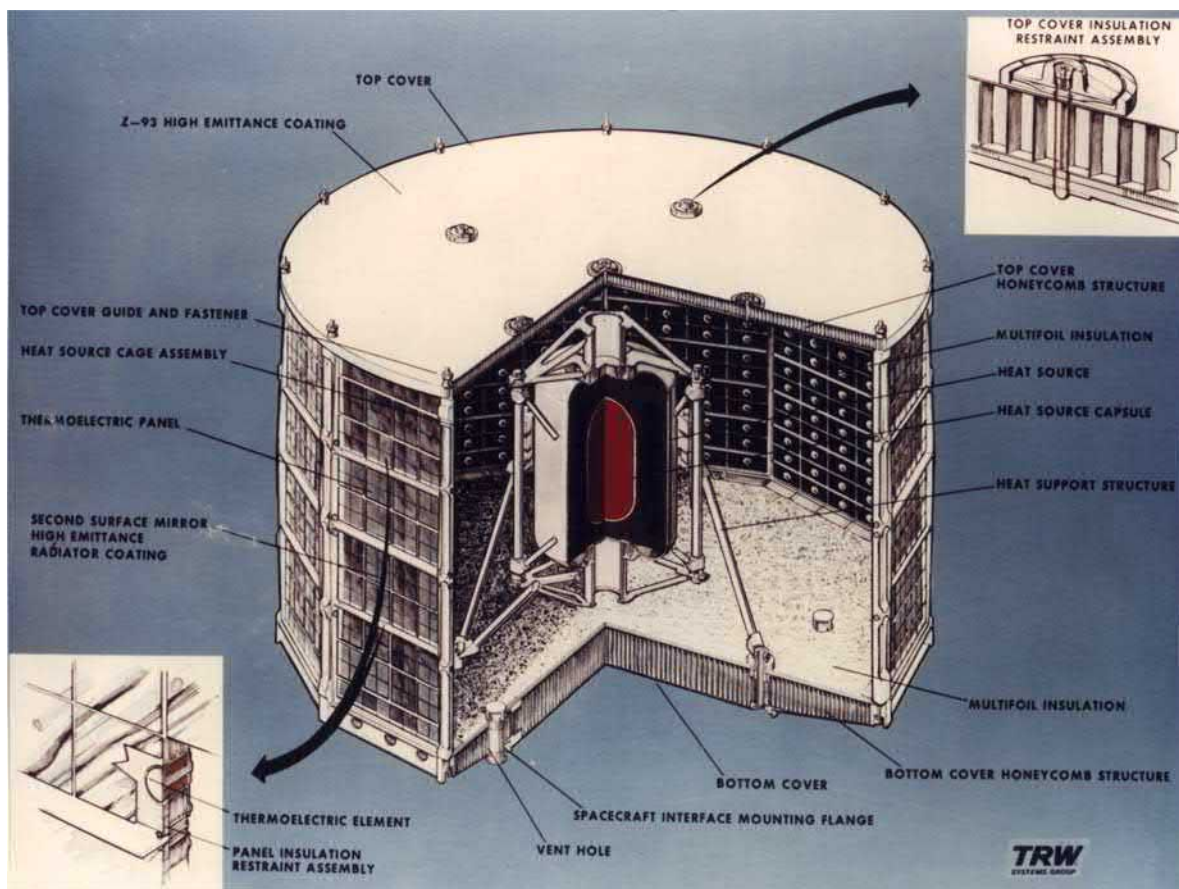


Fig. 6 Cutaway of the TRANSIT RTG. The TRANSIT RTG was designed to provide 30 We after 5 years. The mass of the TRANSIT RTG was about 13.6 kg and the distance across the flats was about 61 cm. Source: TRW under contract to the AEC.

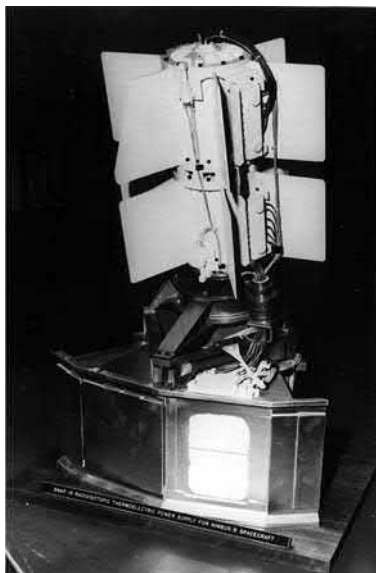


Fig. 7 Photograph of models of the two Nimbus-3 SNAP-19 RTGs shown in the mounted configuration for the Nimbus-3 satellite. The two RTGs produced 56.4 We at launch. The RTGs were 26.7 cm high and 53.8 cm across the heat-radiating fins. The overall mass was 16.3 kg. Source: US Department of Energy.

(plutonium-238 in the United States) that provides heat from the natural decay of the radioisotope. Fig. 8 is a diagram of the most recent US RHU, known as the Light-Weight Radioisotope Heater Unit (LWRHU) which has been used on the Galileo, Cassini and various Mars missions (see Table 2). Table 2 lists the NASA missions enabled by radioisotope heater units.

Apollo Lunar Surface Experiments Packages (ALSEP)

The long (14-Earth-day) lunar nights and lunar dust led to the use of RTGs to power the Apollo Lunar Surface Experiments Packages (ALSEPs) installed on the Moon by the crews of Apollo 12, 14, 15, 16, and 17. Fig. 8 is a cutaway of the SNAP-27 RTG. The design took advantage of the presence of astronauts who would insert the heat source into the converter on the Moon (Fig. 9). (As noted earlier, two radioisotope heater units were used on the Apollo 11 Early Apollo Experiments Package, EASEP.)

On November 19, 1969, Apollo 12 astronauts Charles Conrad, Jr. and Alan L. Bean assembled the first nuclear-powered ALSEP. With the success of the Apollo 12 SNAP-27 RTG, four more SNAP-27-RTG-powered ALSEPs (Apollo missions 14-17) were installed on the Moon. Fig. 10 shows astronaut Alan Bean removing the heat source from its container prior to insertion into the converter (shown in the foreground). (A sixth SNAP-27 RTG safely reentered the Earth's atmosphere after the abort of the Apollo 13 mission.)

All five SNAP-27 RTGs more than met their prelaunch requirements thereby enabling the ALSEPs to gather long-term scientific data on the internal structure and composition of the Moon.



Fig. 8 Diagram of light-weight radioisotope heater unit (LWRHU).

Table 2 NASA missions enabled by radioisotope heater units.

Apollo 11 EASEP Lunar Radioisotope Heater Unit—two 15-Wt RHUs	
Pioneer 10 & 11	12 RHUs each
Voyager 1 & 2	9 RHUs each
Galileo	120 LWRHUs (103 on Orbiter; 17 on Probe)
Mars Pathfinder Sojourner Rover	3 LWRHUs
Cassini	117 LWRHUs (82 on Orbiter; 35 on Huygens Titan probe)
Mars Exploration Rovers (Spirit & Opportunity)	8 LWRHUs each

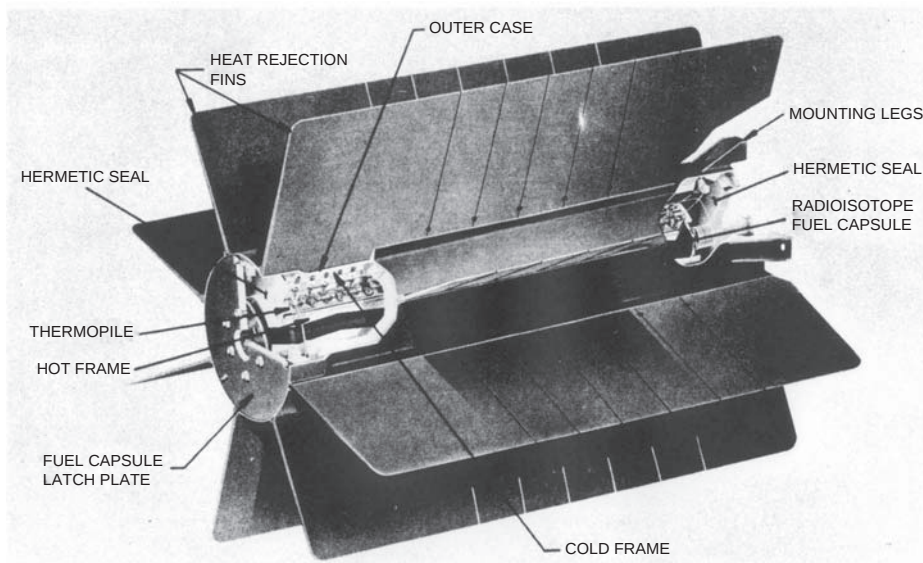


Fig. 9 Cutaway of the SNAP-27 RTG. The SNAP-27 RTGs were to provide at least 63.5 We 1 year after lunar emplacement. (For Apollo 17, the requirement was 69 We 2 years after emplacement.) The converter was 46 cm high and 40 cm in diameter (including the fins). Source: US Department of Energy.



Fig. 10 Astronaut Alan Bean removing the SNAP-27 heat source from the Apollo 12 Lunar Module on November 19, 1969. The SNAP-27 converter is shown in front of Alan Bean. Source: NASA.

Lincoln experimental satellites 8 and 9

The next RTG developed more than doubled the power of the SNAP-27 RTGs and more than quadrupled the power of the SNAP-19 RTGs. Developed by General Electric (now part of Lockheed-Martin) and termed the “Multi-Hundred Watt Radioisotope Thermoelectric Generator (MHW-RTG),” these RTGs powered the MIT Lincoln Laboratory (MIT/LL) satellites designated LES-8 and LES-9 (LES stands for Lincoln Experimental Satellite). The LES-8/9 satellites, which were launched together on 14 March 1976, were to test advanced technologies for strategic communications. Each LES carried two MHW-RTGs, producing about 154 We at 26.5 V each at beginning of mission by means of 312 thermoelectric couples using a new alloy based on silicon-germanium. Fig. 11 is a cutaway of the MHW-RTG.

The successful use of MHW-RTGs on the two US Air Force satellites LES-8/9 helped pave the way for their use on NASA’s Voyager 1 and Voyager 2 spacecraft (which were built and managed by JPL). For LES-8/9, the MHW-RTGs more than met their five-year life-time requirement providing reliable, continuous power even during the daily 70-min eclipses of the Sun by Earth. Fig. 12 illustrates the LES-8/9 concept.

Interplanetary missions

Radioisotope power was enabling for the next set of NASA missions—flights to the outer planets. Simply put, in the cold, dark regions of the outer Solar System or the cold, dusty deserts of Mars, there was no other long-term option. Some of the greatest robotic explorers of all time were powered by RTGs: Pioneer, Viking, Voyager, Galileo, Ulysses, Cassini, New Horizons and Curiosity Mars Science Laboratory.

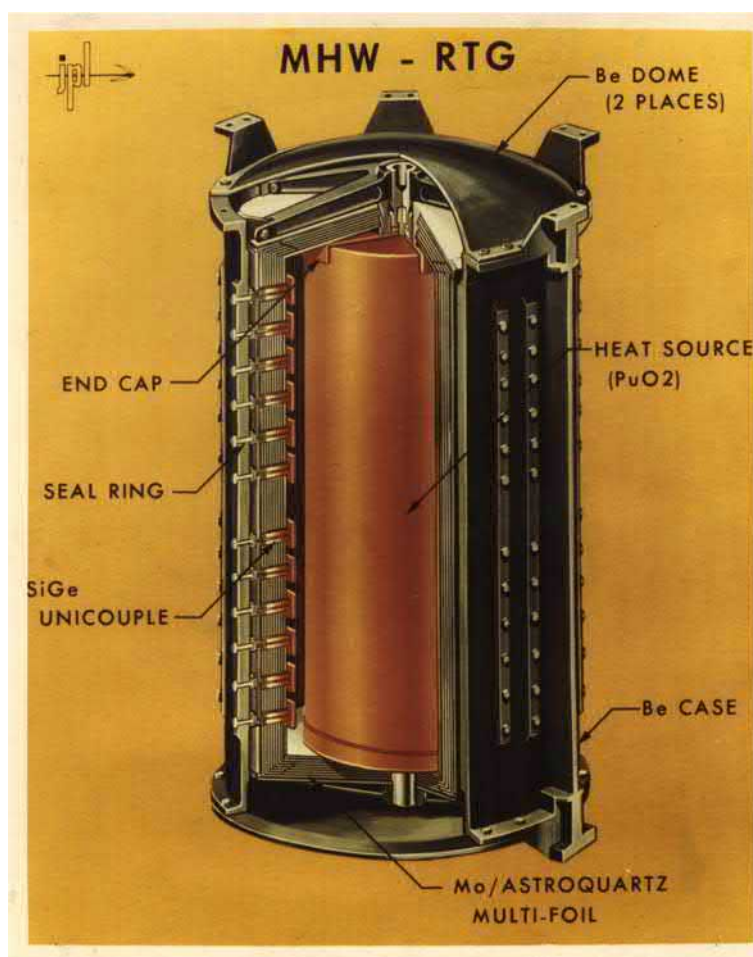


Fig. 11 Cutaway of the Multi-Hundred Watt Radioisotope Thermoelectric Generator (MHW-RTG). The MHW-RTG produced about 154 We per RTG at beginning of mission by means of 312 silicon-germanium alloy thermoelectric couples. The mass for the LES-8/9 MHW-RTGs was 39.69 kg within an envelope 58.31-cm long by 39.73-cm in diameter. Source: US Department of Energy.

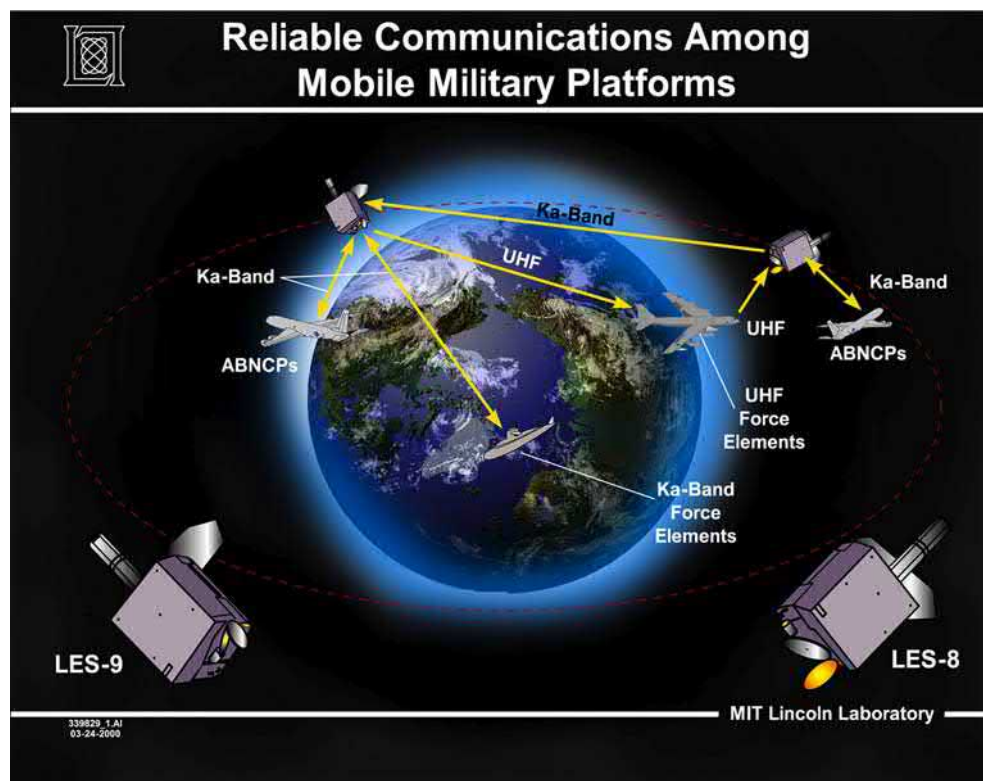


Fig. 12 Lincoln Experimental Satellites 8 and 9 providing reliable communications among mobile military platforms. The two MHW-RTGs on each spacecraft are mounted on the ends of the spacecraft. Source: MIT Lincoln Laboratory.

Pioneer 10 and Pioneer 11

For the Pioneer 10 and Pioneer 11 missions to Jupiter (which were managed by NASA's Ames Research Center), the SNAP-19 RTG was improved by Teledyne raising the beginning of mission power from 28.2 We per RTG on Nimbus to 40.3 We per RTG. The power conversion efficiency was also raised from 4.5% for the Nimbus-3 SNAP-19 RTGs to 6.2% for the Pioneer SNAP-19 RTGs. Related to this, the specific power was raised from 2.1 We/kg (Nimbus RTGs) to 3.0 We/kg (Pioneer RTGs). **Fig. 13** shows a cutaway of the Pioneer SNAP-19 RTG.

Both Pioneer 10 (launched on March 2, 1972) and Pioneer 11 (launched on April 5, 1973) successfully navigated the asteroid belt and became the first spacecraft to fly by Jupiter. Pioneer 10 paved the way for exploration of the outer Solar System—"for Voyager to tour the outer planets, for Ulysses to break out of the ecliptic, for Galileo to investigate Jupiter and its satellites, and for Cassini to go to Saturn and probe Titan" (NASA Fact Sheet on "The Pioneer Missions"). Thanks to the excellent performance of the two spacecraft and their RTGs power sources, NASA directed Pioneer 11 to fly by Saturn, making it the first spacecraft to encounter the ringed world. Pioneer 10's radiation measurements at Jupiter were crucial in designing the Voyager and Galileo spacecraft. Both spacecraft continued to operate decades after launch, becoming humanity's first interstellar messengers (see illustration in **Fig. 14**).

Viking Lander 1 and Viking Lander 2

To meet the requirements of being encapsulated for the voyage to Mars and to operate on the surface of Mars, the basic SNAP-19 RTG design was modified to include a dome reservoir of gas. Each Viking Lander carried two SNAP-19 RTGs which were to produce a minimum of 35 We during the "primary mission" (90 days on the surface of Mars). The Viking SNAP-19 RTGs more than met their 90-day requirement and operated for years after their 1976 landings thereby providing valuable scientific data on the surface of Mars. (It was estimated that the SNAP-19 RTGs on Viking Lander 1 were capable of providing sufficient power for operation until 1994—18 years beyond the mission requirement.) Using a model, **Fig. 15** shows the location of the two Viking SNAP-19 RTGs on a Viking Lander. (Overall management of the Viking Lander program resided at NASA's Langley Research Center.)

Voyager 1 and Voyager 2

The Multi-Hundred Watt RTG design originally flown in 1976 on the LES-8/9 satellites was modified for the Voyager 1 and Voyager 2 missions principally by reinforcing the converter case and removing an iridium canister. The net effect was to reduce the mass of

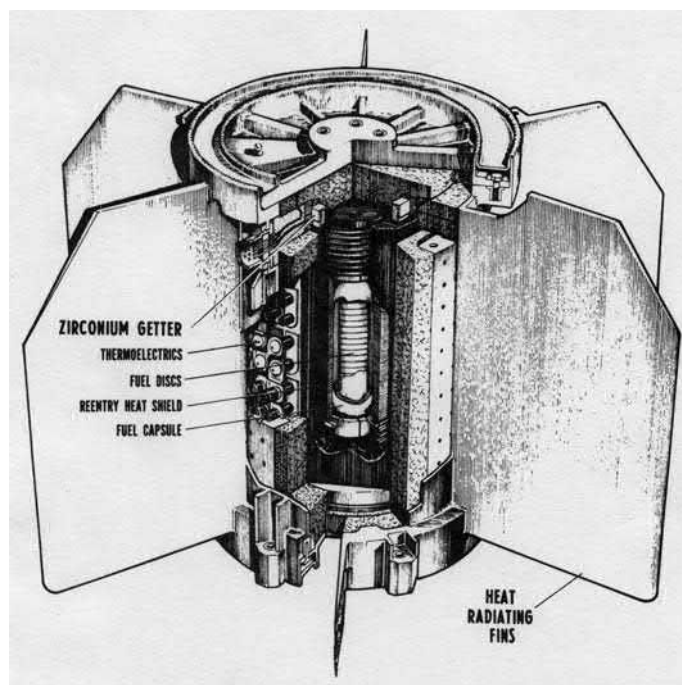


Fig. 13 Cutaway of the Pioneer SNAP-19 RTG. Beginning of mission power was 40.3 We per RTG. The height was 28.2 cm and the fin span was 50.8 cm. The average mass was 13.6 kg per RTG. Source: US Department of Energy.

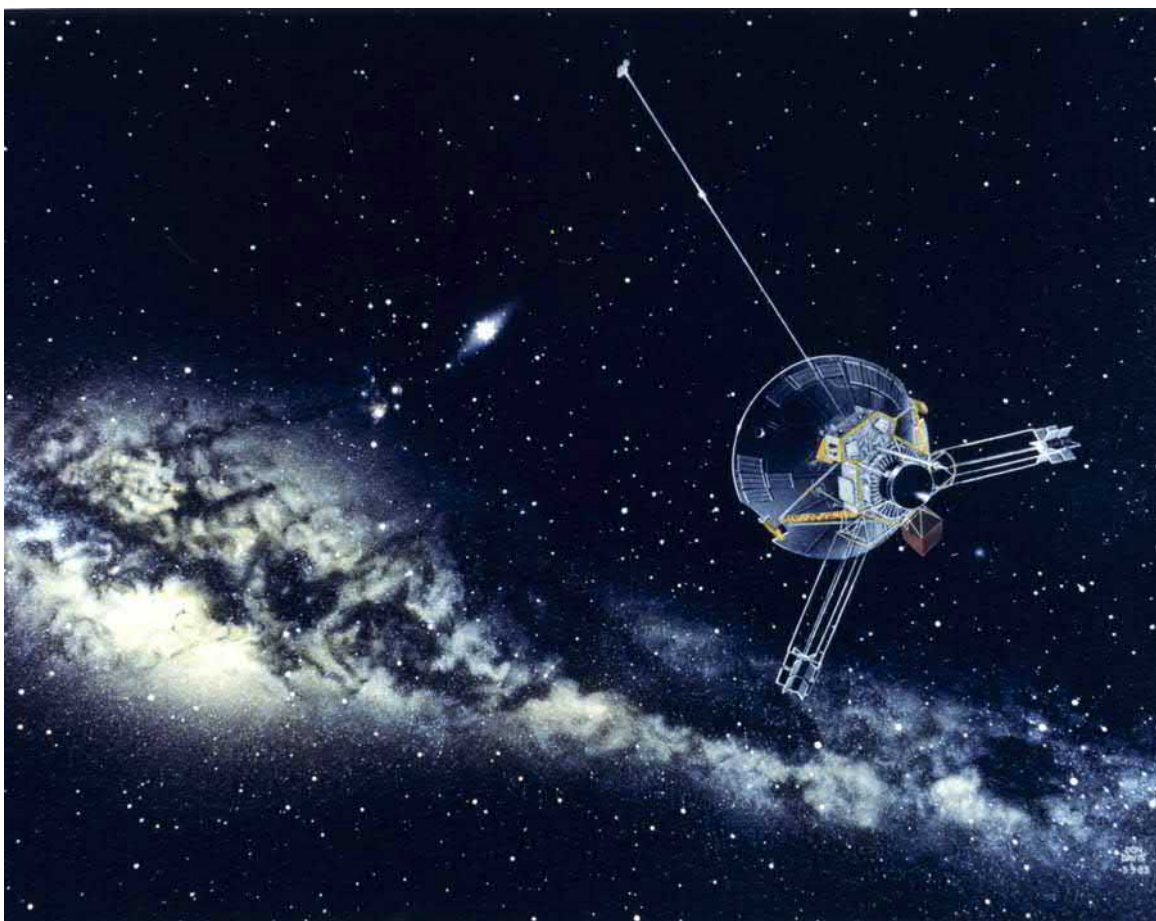


Fig. 14 Artist's conception of Pioneer spacecraft entering interstellar space. Each Pioneer spacecraft carried four SNAP-19 RTGs, two each mounted on booms. The high-gain antenna is 2.7 m in diameter and the spacecraft mass is 270 kg. Source: NASA.



Fig. 15 Location of the Viking SNAP-19 RTGs on the Viking Lander. Each Viking SNAP-19 RTG produced an average beginning of mission power of 42.7 We. The mass of the Viking SNAP-19 RTG was 15.2 kg. The height of the Viking SNAP-19 RTG (with dome) was 40.4 cm and the diameter across the fins was 58.7 cm. Source: US Department of Energy and NASA.

the Voyager MHW-RTGs by about 2 kg. Whereas each Lincoln Experimental Satellite carried two MHW-RTGs, each Voyager spacecraft carried three MHW-RTGs, each producing an average of 158 We at about 30 V at beginning of mission.

The original objective of the Voyager 1 and Voyager 2 missions was to fly past Jupiter and Saturn. The successful accomplishment of this objective and the outstanding power performance of the MHW-RTGs allowed NASA to send Voyager 2 on to Uranus and Neptune. Both Voyagers are still operating now in interstellar space over 42 years after their 1977 launches (see Fig. 16).



Fig. 16 Artist's conception of Voyager 1 in interstellar space. The three MHW-RTGs are shown on the boom below the spacecraft. The Voyager antenna is 3.7 m in diameter and the spacecraft mass is 825 kg. Source: NASA/JPL.

Galileo

For NASA's Galileo Jupiter orbiter spacecraft and for the European Space Agency's Ulysses solar polar orbiter, a new, more powerful RTG was designed and produced: the General-Purpose Heat Source Radioisotope Thermoelectric Generator (GPHS-RTG). Versions of the GPHS-RTG would go on to be used on the Cassini Saturn orbiter and the New Horizons mission to Pluto and beyond. With new plutonium-238 fuel at the time of fueling, the GPHS-RTG was capable of providing at least 300 We making the GPHS-RTG the most powerful RTG ever flown.

As illustrated in Fig. 17, the GPHS-RTG is composed to two main elements: the general-purpose heat source (GPHS) assembly and the converter. The GPHS assembly consists of 18 modules, each containing four encapsulated fuel pellets ("fueled clads") that combined (using new plutonium-238 fuel) produced about 245 Wt per module for a total of about 4410 Wt which the converter with its 572 silicon-germanium alloy thermoelectric elements converted into at least 285 We at beginning-of-mission (BOM). (At the time of fueling, the GPHS-RTG was capable of producing over 300 We.)

Fig. 18 is an artist's illustration of the Galileo spacecraft in orbit about Jupiter. The Galileo mission consisted of an Orbiter (powered by two GPHS-RTGs) and a Probe which entered the Jovian atmosphere. In addition to the GPHS-RTGs on the Galileo Orbiter, both the Orbiter and the Probe carried radioisotope heater units for thermal management. The Galileo spacecraft was launched on October 18, 1989 and, after various planetary flybys transiting some 3.7 billion kilometers, reached Jupiter on December 7, 1995. The Galileo spacecraft was the first RTG-powered spacecraft to be launched on Space Shuttle.

Galileo was the first spacecraft to orbit an outer planet; it orbited Jupiter for almost 8 years, making close passes by all its major moons. Measurements made by Galileo allowed scientists to determine that Jupiter's icy moon Europa probably has a subsurface ocean with more water than the total amount found on Earth. Magnetic data indicated that the moons Ganymede and Callisto may also have a liquid saltwater layer. The indications of water on Europa, Ganymede and Callisto are of keen scientific interest because water is thought to be one of the precursors to the origin of life. As a result of the Galileo mission, NASA is developing a mission to study Europa in more detail even as NASA's Juno mission is providing close-up data on Jupiter.

At beginning of mission the two Galileo GPHS-RTGs produce a combined 577 We. The power was about 2% less than it could have been because there had been a three-year delay following the Space Shuttle Challenger accident.

On September 21, 2003, after 14 years in space and 8 years orbiting Jupiter and running out of propellant for spacecraft control, Galileo was sent into the atmosphere of Jupiter so that it would not accidentally contaminate satellites such as Europa.

Ulysses

Launched on October 6, 1990 from Space Shuttle Discovery, the Ulysses spacecraft (shown in an artist's conception in Fig. 19) was a joint European Space Agency (ESA)-NASA-designed five-year mission to study the never-before-examined north and south poles

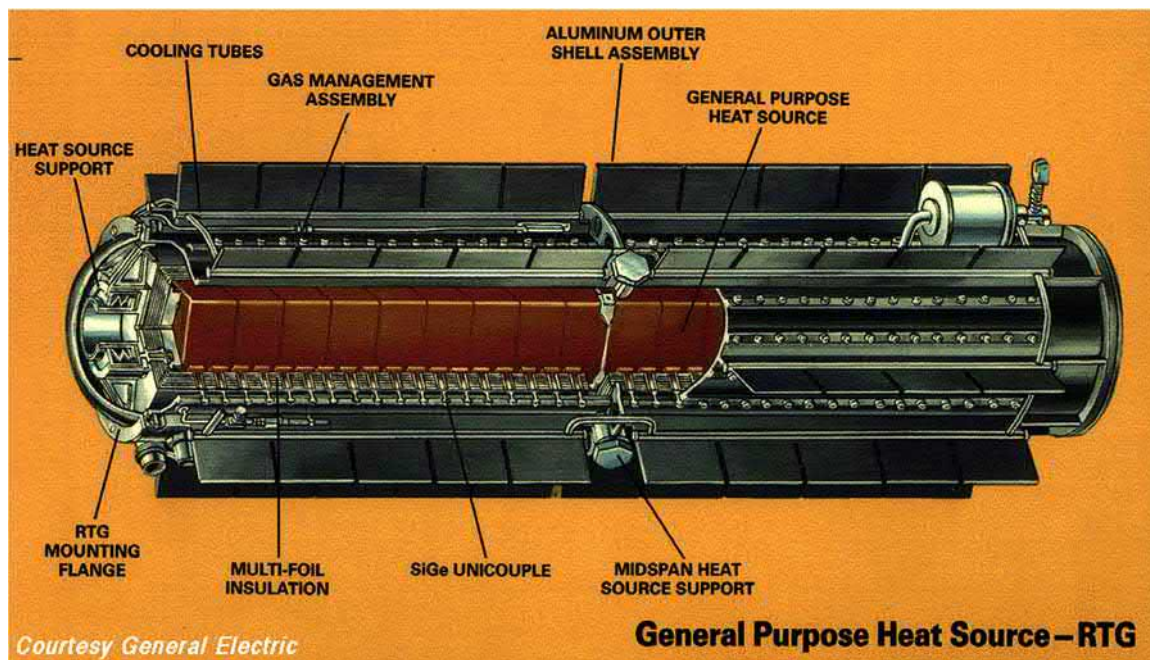


Fig. 17 General-Purpose Heat Source Radioisotope Thermoelectric Generator. The GPHS-RTG was capable of producing at least 300 We at time of fueling. The GPHS-RTG overall diameter was 0.422 m with a length of 1.14 m and a mass of about 55.9 kg (Galileo/Ulysses). Source: GE (under DOE contract) and DOE/NASA.



Fig. 18 Artist's concept of the Galileo Orbiter at Jupiter receiving signals from the Galileo Probe which has entered the Jovian atmosphere. Galileo carried two GPHS-RTGs mounted on booms. The magnetometer boom extends 11 m to one side; the mass of the Orbiter was 2223 kg. Source: NASA/JPL.

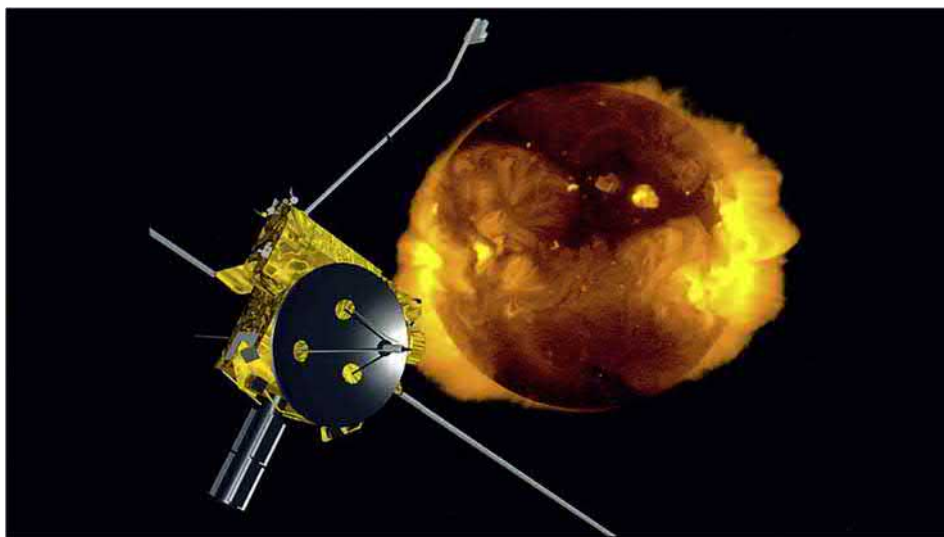


Fig. 19 Artist's illustration of Ulysses and the Sun. Ulysses was powered by a GPHS-RTG (shown mounted on the lower side). The Ulysses spacecraft carried nine scientific instruments and had a mass of 371 kg. Source: ESA/NASA/JPL.

of the Sun. To do that, Ulysses made an unprecedented gravity assist maneuver at Jupiter to hurl itself out of the ecliptic plane and into its solar polar orbit. Since sunlight at Jupiter is about 25 times less intense than at Earth, Ulysses carried a GPHS-RTG for power. The beginning of mission power was 289 We (it would have been higher but there was a four-year launch delay following the 1986 Space Shuttle Challenger accident). Thanks to the long-lived GPHS-RTG, Ulysses was able to operate 13 years longer than the 5-year goal.

Cassini

The Cassini Saturn orbiter with ESA's Huygens probe was launched on October 15, 1997. The Cassini spacecraft was powered by three GPHS-RTGs. Additionally, as with Galileo, Light-Weight Radioisotope Heater Units (LWRHUs) provided thermal power to Cassini and to the Huygens probe that was to land on Titan, Saturn's largest satellite (Fig. 20).

Launched by a Titan/Centaur rocket, Cassini arrived at Saturn on July 1, 2004 after two gravitational assists at Venus, one at Earth and one at Jupiter. The Huygens Probe was released on December 24, 2004 and descended to Titan on January 14, 2005.

The three GPHS-RTGs produced a total of 885 We at beginning of mission and 633 We at end of mission.

Cassini provided a wealth of data on Saturn and its satellites finding evidence that Saturn's moon Enceladus very likely has an ocean under its surface ice and, with Huygens, that Saturn's largest moon, Titan, has intriguing pre-biotic chemistry.

After two extensions of the original 4-year mission, Cassini was sent into the atmosphere of Saturn on September 15, 2017 to avoid accidentally contaminating the satellites (of particular concern was Enceladus which, like Europa, may have a large subsurface water ocean).

New Horizons

THE 478-kg New Horizons spacecraft, built by the Johns Hopkins University's Applied Physics Laboratory, was launched on January 19, 2006 to begin a 9.5-year voyage to Pluto and its satellites. New Horizons is powered by a single GPHS-RTG which produced about 245.7 We at beginning of mission (more power would have been possible had DOE not been forced to use some "old" fuel).

New Horizons successfully flew past Pluto on July 14, 2015 sending back very valuable scientific information on what was once termed the ninth planet. New Horizons has gone on to fly past the Kuiper Belt object (KBO) Arrokoth and is now making ground-breaking measurements in the Kuiper Belt. Fig. 21 is an illustration of New Horizons.

Curiosity Mars Science Laboratory

For NASA's Curiosity Mars Science Laboratory rover, a new RTG, termed the Multi-Mission Radioisotope Thermoelectric Generator (MMRTG), was developed. The MMRTG is designed to operate on planetary surfaces or in deep space. Like the GPHS-RTG, the



Fig. 20 Artist's illustration of the Cassini spacecraft just after release of the European Space Agency (ESA) Huygens Probe to land on Titan. Two of the three GPHS-RTGs can be seen in the upper left of the figure. Cassini was 6.7 m high by 4 m wide with a mass at launch of 5712 kg (including propellant, Huygens Probe, adapter, etc.) Source: NASA/JPL.



Fig. 21 Artist's conception of the New Horizons spacecraft at Pluto. Pluto's largest satellite, Charon, is in the background. The New Horizons GPHS-RTG is shown on the left side of the spacecraft. New Horizon has a triangular-shaped body almost 0.76 m thick. Source: JHUAPL/SwRI.

MMRTG uses the modular General-Purpose Heat Source but its thermoelectric technology harkens back to the Pioneer and Viking SNAP-19 RTGs. The MMRTG is designed to produce at least 110 We at beginning of mission.

Fig. 22 illustrates the components of the MMRTG and shows the location on the Curiosity rover.

Curiosity was launched on November 26, 2011 and landed in the Gale crater on Mars on August 6, 2012. Since then Curiosity has moved across Gale crater and is climbing Aeolis Mons (Mount Sharp). Curiosity has made a number of exciting discoveries including evidence that Gale crater was likely the floor of an ancient stream or pond (**Fig. 23**).

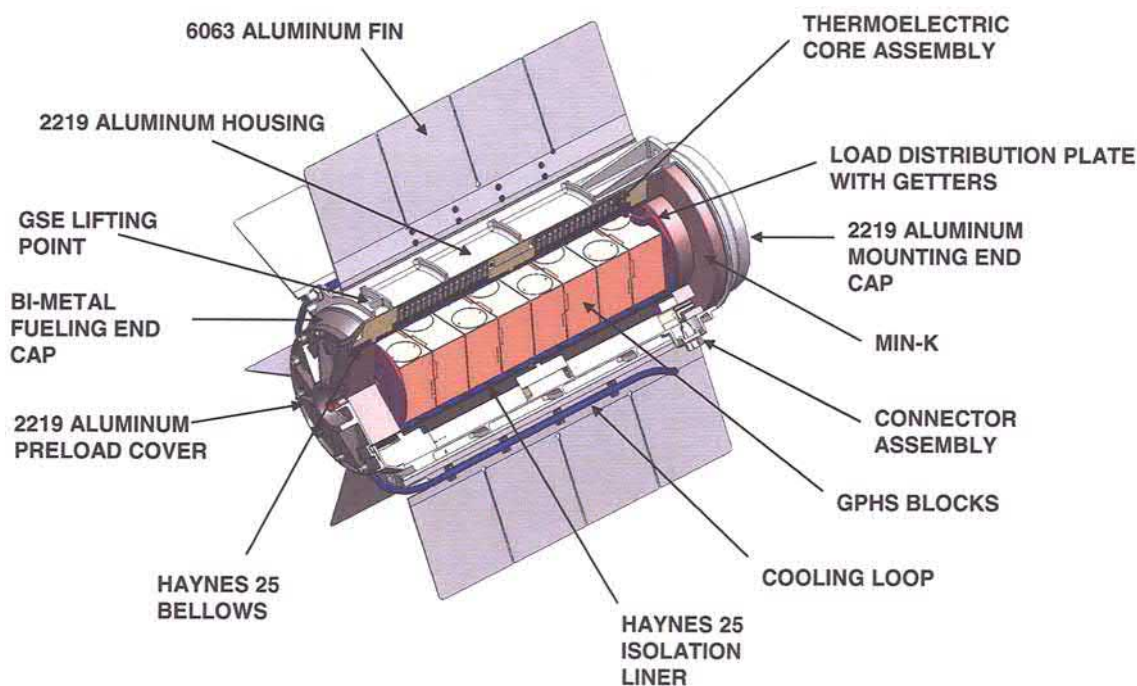


Fig. 22 Cutaway of the Multi-Mission Radioisotope Thermoelectric Generator (MMRTG). The MMRTG produces power from the heat emanating from the eight GPHS modules shown in red. Source: DOE.



Fig. 23 Viewing showing the MMRTG (in white fins) at the aft end of the Curiosity Mars Science Laboratory. Source: NASA/JPL.

Conclusions

Radioisotope power sources, both RTGs and RHUs, have been enhancing and/or enabling challenging space missions for over 58 years. Since 1961, the United States has successfully launched 42 RTGs on 24 space missions along with hundreds of RHUs. The RTGs have powered the Apollo Lunar Surface Experiments Packages; the Pioneer flybys of Jupiter and Saturn; the Viking Mars Landers; the Voyager flybys of Jupiter, Saturn, Uranus, and Neptune; the Galileo orbital exploration of Jupiter; the Ulysses solar polar explorer; the Cassini orbital exploration of Saturn; the New Horizons mission to Pluto and the Kuiper Belt and the Curiosity Mars Science Laboratory.

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Radioisotope Power: Technology Options

Gary L. Bennett*, Boise, ID, United States

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Abbreviations

ASC Advanced Stirling Convertor
ASRG Advanced Stirling Radioisotope Generator
BIPS Brayton Isotope Power System
BOM Beginning of Mission
DIPS Dynamic Isotope Power System
DOE U.S. Department of Energy
eMMRTG Enhanced Multi-Mission Radioisotope Thermoelectric Generator
EOM End of Mission
GPHS-RTG General-Purpose Heat Source Radioisotope Thermoelectric Generator
JPL Jet Propulsion Laboratory
KIPS Kilowatt Isotope Power System
MHW-RTG Multi-Hundred Watt Radioisotope Thermoelectric Generator
MMRTG Multi-Mission Radioisotope Thermoelectric Generator
NASA National Aeronautics and Space Administration
PCU Power Conversion Unit
PSD Planetary Science Division (NASA)
RPS Radioisotope Power Source (or System)
RTG Radioisotope Thermoelectric Generator
USAF U.S. Air Force
We Watts electric ("electrical power")
Wt Watts thermal ("thermal power")

Introduction

Radioisotope power sources/systems (RPSs) have been enabling for some of the most challenging space missions such as the Pioneer 10/11 flights to Jupiter, Saturn and beyond; the Viking Mars Landers 1/2; the Voyager 1/2 flights to Jupiter, Saturn, Uranus, Neptune and beyond; the Galileo Jovian orbiter, the Ulysses solar polar orbiter; the Cassini Saturn orbiter; the New Horizons flight to Pluto and beyond; and the Curiosity and Perseverance Marsrovers. All of these missions have been powered by radioisotope thermoelectric generators (RTGs), that is, radioisotope power sources that produce electrical power by means of the thermoelectric effect. A number of these missions also employed radioisotope heater units (RHUs) to provide thermal power ("heat") to sensitive components on the spacecraft. **Fig. 1** provides a visual summary of these U.S. RPS missions. **Table 1** summarizes the RTGs flown by the U.S. (Bennett et al., 1984, 2006).

The former Soviet Union reportedly has flown RTGs on two Earth-orbiting spacecraft (Cosmos 84 and Cosmos 90) and on two lunar rovers (Lunokhod-1 and Lunokhod-2), mostly using polonium-210 as the radioisotope. For the aborted Mars 96 mission, the

*NASA retired.

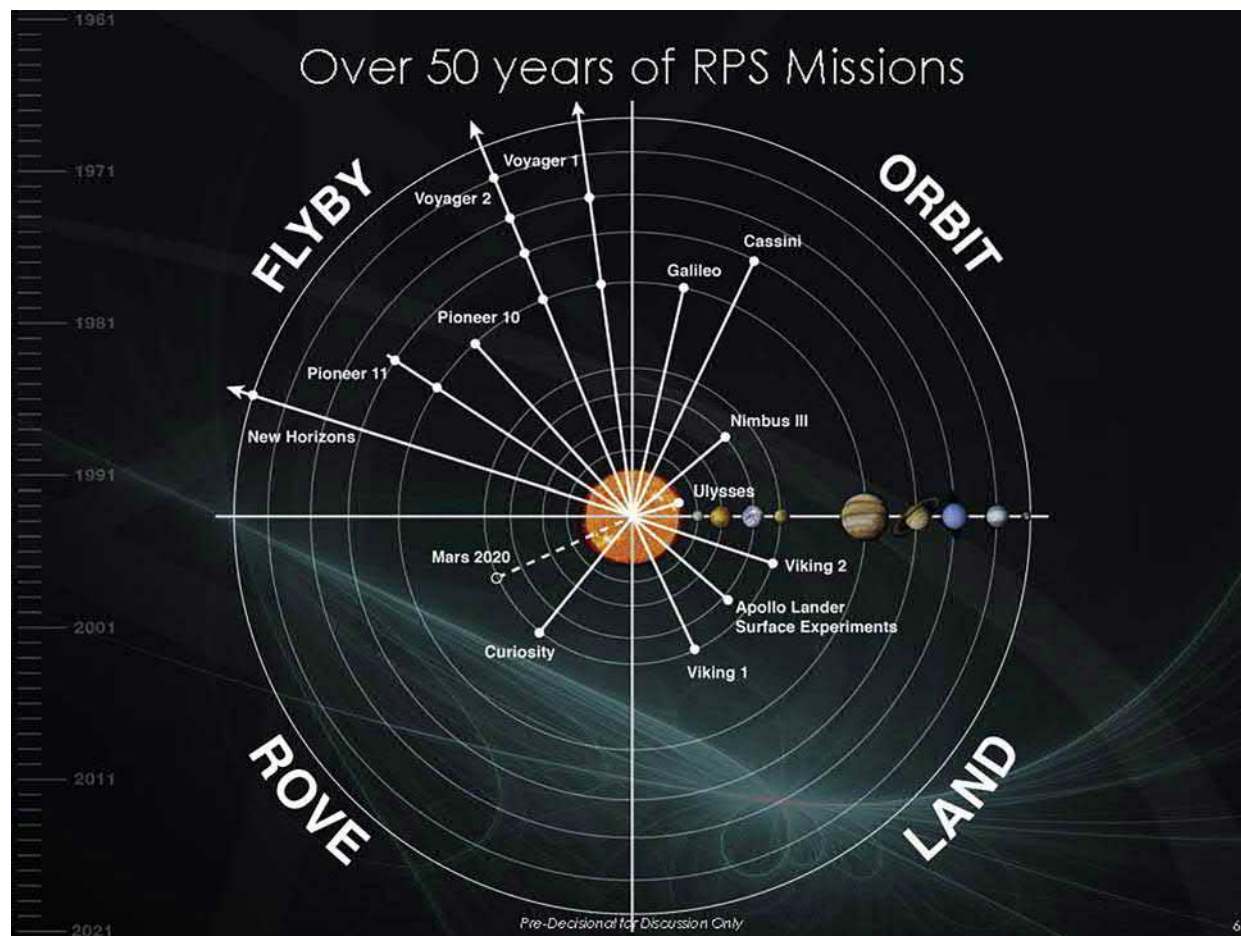


Fig. 1 Summary of NASA Radioisotope Power Source (RPS) Missions. (Not shown: Sojourner, Spirit and Opportunity Mars rovers which carried radioisotope heater units (RHUs) for heat.) Source: NASA.

Table 1 Uses of Radioisotope Thermoelectric Generators (RTGs) by the United States (Spacecraft/Year Launched/Type of RTG/Beginning-of-Mission Power) (Bennett et al., 1984, 2006).

Transit Navy Navigational Satellites

- Transits 4A and 4B (1961) SNAP-3B (2.7 We)
- Transits 5BN-1 and 5BN-2 (1963) SNAP-9A (≥ 25 We)
- Transit TRIAD (1972) Transit-RTG (35 We)

Nimbus-3 Meteorological Satellite

- SNAP-19B RTGs (1969) (2 @ 28 We each)

Apollo Lunar Surface Experiments Packages

- Apollos 12 (1969), 14 (1971), 15 (1971), 16 (1972), 17 (1972)
- SNAP-27 (> 70 We each)

Lincoln Experimental Satellites (Communications)

- LES-8 and LES-9 (1976) MHW-RTG (2/spacecraft @ ~ 154 We each)

Interplanetary Missions

- Pioneer 10 (1972) and Pioneer 11 (1973) SNAP-19 (4/spacecraft @ ~ 40 We each)
- Viking Mars Landers 1 and 2 (1975) SNAP-19 (2/Lander @ ~ 42 We each)
- Voyager 1 and Voyager 2 (1977) MHW-RTG (3/spacecraft @ ~ 158 We each)
- Galileo (1989) GPHS-RTG (2 @ ~ 288 We each) (> 3 -year delay in launch)
- Ulysses (1990) GPHS-RTG (289 We) (> 4 -year delay in launch)
- Cassini (1997) GPHS-RTG (3 @ ~ 295 We each)
- New Horizons (2006) GPHS-RTG (245.7 We) (most of fuel had > 21 years decay)
- Curiosity Mars Science Laboratory (2011) MMRTG (~ 115 We) (> 2 -year delay)
- Perseverance rover (2020) MMRTG (~ 110 We)

lander stations carried RTGs and RHUs fueled with plutonium-238. European researchers have studied the use of RTGs, notably powered by americium-241. Chinese researchers have developed RTGs fueled with plutonium-238 as demonstrated with their Chang'e 4 lunar mission.

Looking to the future, there are technology options available or that could be developed to increase the efficiency of converting thermal power to electrical power. A higher efficiency RPS would allow obtaining the desired electrical power with less plutonium-238, a radioisotope that is in limited supply (although more is being produced). Similarly, a higher efficiency RPS could provide the desired amount of power in a smaller package. Thus, from the beginning of the RPS program, there have been studies and experiments to develop and qualify higher efficiency conversion systems such as improved thermoelectric materials and dynamic conversion systems. Some of the principal conversion systems will be discussed in the next sections.

Mission requirements

For future RTGs, in 2016, NASA's RPS Program conducted a study to determine possible options for a next-generation of RTGs that could be produced in the late 2020s and would be useful throughout the following 20 years. The objective of the study was "... to determine the characteristics of the next RTG that would "best" fulfill NASA's Planetary Science Division (PSD) mission needs." This study was limited to systems that convert heat to electricity using thermoelectric couples. "Best" was defined as a confluence of the following factors (Woerner 2017):

- An RTG that would be useful across the Solar System
- An RTG that maximizes the types of missions: flyby, orbiter, lander, rover, boats, submersibles, balloons
- An RTG that has reasonable development risks and timeline
- An RTG value (importance, worth and usefulness) returned to PSD for investment, as compared with retaining existing baseline systems

A thermoelectric converter is what is termed a "heat engine" because it depends on heat flow and temperature differences for its operation. To increase efficiency (amount of thermal power converted to electrical power): raise the temperature on the "hot side" (end of the thermoelectric element closest to the radioisotope heat source) and/or lower the temperature on the "cold side" (the end of the thermoelectric element that conducts the unused or "waste" heat outside the RTG). The temperature difference can be likened to the pressure difference that pushes fluids through pipes. Similarly, as will be seen in the sections on dynamic conversion systems, temperature difference is also a critical feature in achieving higher efficiency.

In assessing thermoelectric materials there are three critical properties: the Seebeck coefficient, the electrical resistivity, and the thermal conductivity. The Seebeck coefficient is the voltage produced by each degree of temperature difference; it is a property of the material used. An "ideal" thermoelectric material would have a high Seebeck coefficient, low electrical resistivity (so that very little of the generated electric current is turned into heat), and low thermal conductivity (so that the temperature difference across the thermoelectric elements is increased to provide the "push" for the electrical current).

Currently, since 1976, as shown in Fig. 2, the U.S. has three extant flight RTGs that have been used in space missions.

The Multi-Hundred Watt RTG (MHW-RTG) was designed for two missions: the U.S. Air Force (USAF) Lincoln Experimental Satellites 8 and 9 (LES-8/9) and NASA's two Voyager interplanetary/interstellar spacecraft (Voyagers 1 and 2). (The MHW-RTG was also considered for the Galileo Jovian orbiter.) The MHW-RTG is capable of producing ~158 We at beginning of mission (BOM) with a mass of less than 38 kg. The General-Purpose Heat Source RTG (GPHS-RTG) is capable of producing 300 We at BOM with a mass of less than 56 kg. By virtue of the design requirements (outer-planet missions), the MHW-RTG and the GPHS-RTG operate in a vacuum (although a cover gas could be applied if needed). The Multi-Mission RTG (MMRTG) is a sealed system that is designed to operate in space or in planetary atmospheres. The MMRTG can produce >110 We at BOM with a mass of ~45 kg.

As illustrated in Fig. 3, from the RPS mission studies three classes of missions were identified for RPS mission concepts. End-of-mission (EOM) powers ranged from <150 to >600 We (Woerner, 2017).

Improved thermoelectric materials

Both for space applications and Earth-based commercial applications, there has been a focus on improving the efficiency of thermoelectric materials. This research generally takes the form of developing high-temperature materials with a high Seebeck coefficient, a low thermal conductivity and a high electrical conductivity. Unfortunately, materials with a low thermal conductivity also tend to have a low electrical conductivity. Thus, a tradeoff or compromise must be made.

To improve the efficiency of the MMRTG, researchers at NASA's Jet Propulsion Laboratory investigated skutterudites, a cobalt arsenide mineral named after the city of Skotterud, Norway. Skutterudites offer the potential for lowering the thermal conductivity. The inclusion of skutterudites-based thermoelectric elements was projected to increase the efficiency of the "enhanced" MMRTG (eMMRTG), allowing the production of ~145 We versus the 110 We of the MMRTG.

Considering a target operating temperature range of about 1275 K (hot side) and 475 K (cold side), various other materials such as zintl (named for the German chemist Eduard Zintl), lanthanum telluride (La₃-xTe₄) as well as segmented, cantilevered and multicouples have been or are being considered. Segmented thermoelectric elements (elements composed of more than one material)

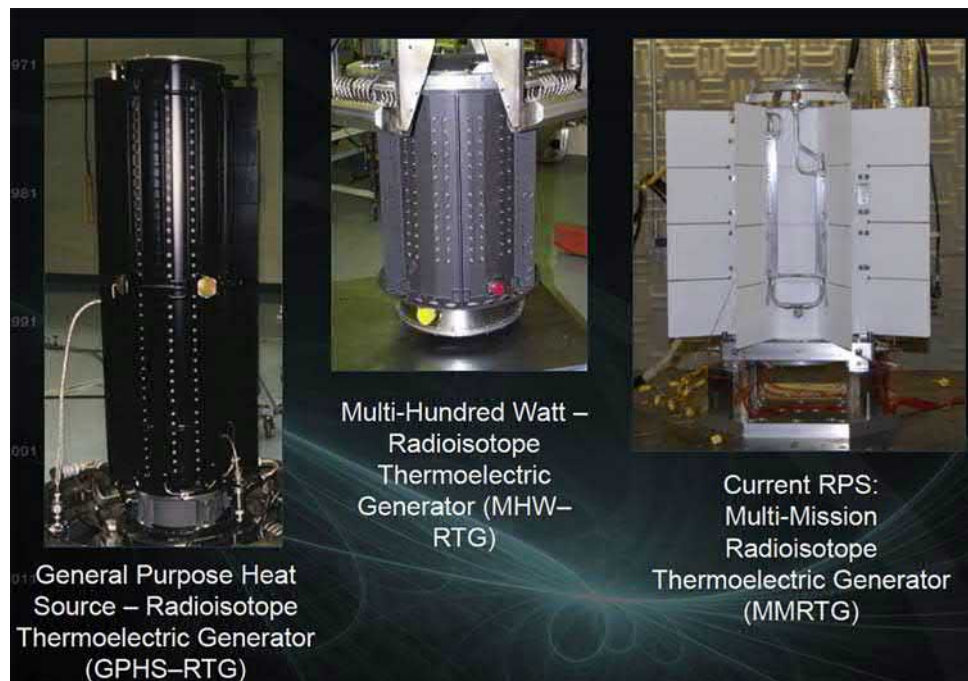


Fig. 2 Flight Radioisotope Power Sources for Current Missions. (NOTE: The GPHS-RTG is currently powering the New Horizons mission that flew past Pluto. The MHW-RTG is currently powering the two Voyager spacecraft which have left the Solar System. The MMRTG is currently powering the Curiosity Mars Science Laboratory and will power the “Mars 2020 is now named ‘Perseverance’.”)

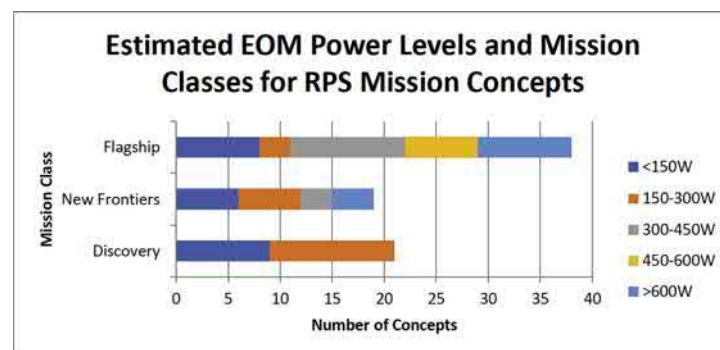


Fig. 3 The RPS mission studies, sorted by mission class and end-of-mission (EOM) power (Woerner, 2017).

allow the optimization of thermoelectric properties over the temperature range of the thermoelectric element. (Zintl phases are a group of high-melting intermetallic compounds that are poor conductors.)

In addition to developing improved materials design considerations such as having a modular RTG that would allow adaptability to different missions are another avenue of research. Fig. 4 shows conceptually how such a segmented modular RTG might be designed so that in the example shown powers ranging from 50 to 500 W could be produced by “stacking” more RTG modules (Zakrajsek et al., 2017).

In addition to developing new thermoelectric materials and new RTG designs, there are options to improve existing RTG concepts using technologies and processes that have been developed since those RTGs were designed. For example, there are several aspects of the GPHS-RTG that can be improved to provide more power per kilogram (Vining and Bennett, 2010). Also, there are proposals to cascade the thermoelectric elements, that is, in effect, build a second thermoelectric converter “outside” the one closest to the heat source so that some of the waste heat from that first converter can be converted into electrical power by the second converter.

Dynamic conversion systems

Most of the electrical power generated on Earth comes from dynamic machinery such as turbine-alternators which use the heat produced by fossil fuels or nuclear reactions to pump a fluid (e.g., water) to turn an alternator. The process depends upon

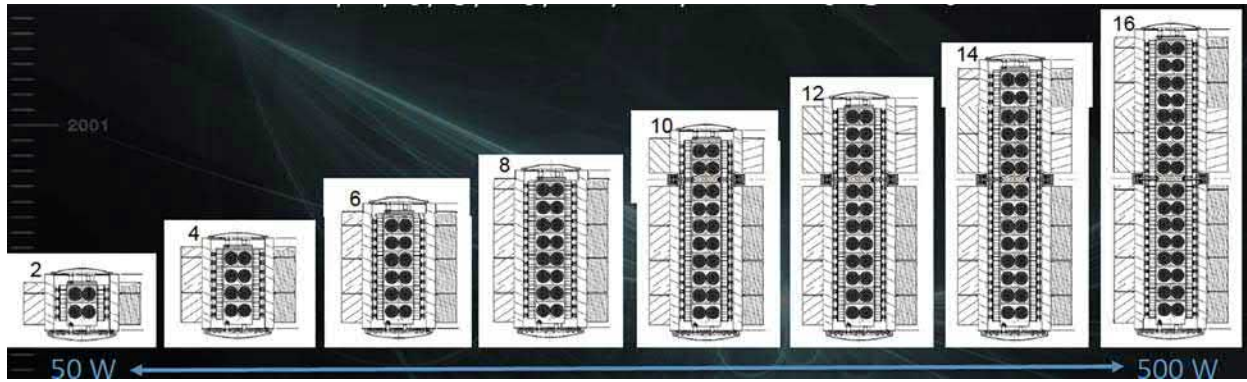


Fig. 4 Segmented Modular Radioisotope Thermoelectric Generator Concept. The Segmented Modular RTG would be designed to allow flexibility in power production by using a modular approach to the fabrication of the RTG. Source: NASA.

a fundamental feature of electromagnetism, namely, that a changing magnetic field will produce an electric current, a phenomenon termed Faraday's law of induction (named for Michael Faraday who discovered electromagnetic induction in 1831). In general, dynamic conversion systems are two-to-three (or more) times as efficient as static conversion systems such as thermoelectrics. Thus, dynamic conversion offers the potential to produce the same amount of power as an RTG but with much lower amount of the radioisotope fuel or higher power for the same amount of fuel as the RTG. Moreover, dynamic systems are scalable (on Earth they are in the multi-megawatt range) and they can be operated at lower temperatures than some RTGs, hence, making the use of more conventional materials easier. Three types of dynamic conversion will be considered in the following sections: Brayton, Rankine and Stirling.

Closed Brayton cycle

The Brayton cycle (named after the American engineer George Brayton) is a thermodynamic cycle that can be used to produce power such as in a jet engine or a power plant. In the closed Brayton cycle (CBC), an inert gas working fluid (e.g., a mixture of helium and xenon) is heated and re-circulated through a compressor and turbine coupled to a rotary alternator. **Fig. 5** is a composite diagram showing the features of the closed Brayton cycle and the Rankine cycle (which has some similar features). As a Brayton radioisotope power source, the working fluid (gas) is heated by the radioisotope heat source then expanded through the turbine which cools the gas while rotating the alternator. The gas then is pressurized by the compressor. Waste heat is rejected to space by means of the radiator. The rotating alternator provides alternating-current (AC) power to the spacecraft. Cycle efficiency is improved by using a recuperator to pre-heat the gas before it returns to the heat source. (In the Rankine system that function is carried out by a similar subsystem called the "regenerator".) (Mason and Schreiber, 2007).

NASA began work on the Brayton system in the 1960s, taking the program through kilowatt-class Brayton power conversion systems as well as to lower powers in the hundreds of watts range. NASA selected Brayton converters for its planned 200 kWe

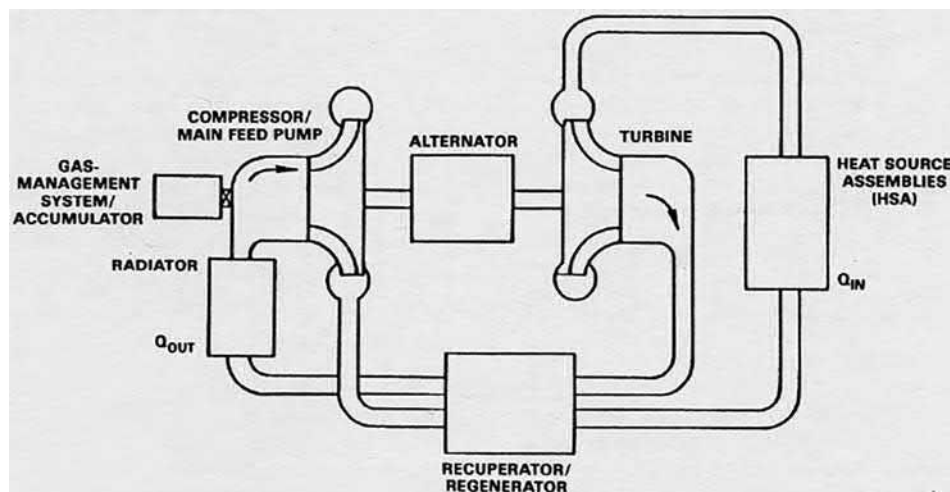


Fig. 5 Generic diagram of a Radioisotope Brayton/Rankine Power conversion system. Source: DOE.

reactor-powered Jupiter Icy Moons Orbiter. DOE also funded development in the 1970s and later in the 1980s, of Brayton technology (Brayton Isotope Power Systems, BIPS) through its Dynamic Isotope Power System (DIPS) program (AiResearch Manufacturing Company of Arizona, 1978). Fig. 6 shows the Mini-Brayton Rotating Unit (Mini-BRU) which was fabricated by AiResearch and which was capable of producing 500–2100 We at an efficiency up to 30% (compared to the less than 10% efficiency of thermoelectric converters). Thousands of hours of successful operation were achieved on the Mini-BRU (Mason and Schreiber, 2007).

Studies in the late 1980s and early 1990s by Rocketdyne led to the design of a 2.5 kWe Brayton power module illustrated in Fig. 7. Efficiencies in the range of 20% were calculated. Versions of this concept could be used to power the first lunar outpost as well as high-power spacecraft (Rocketdyne, 1992).

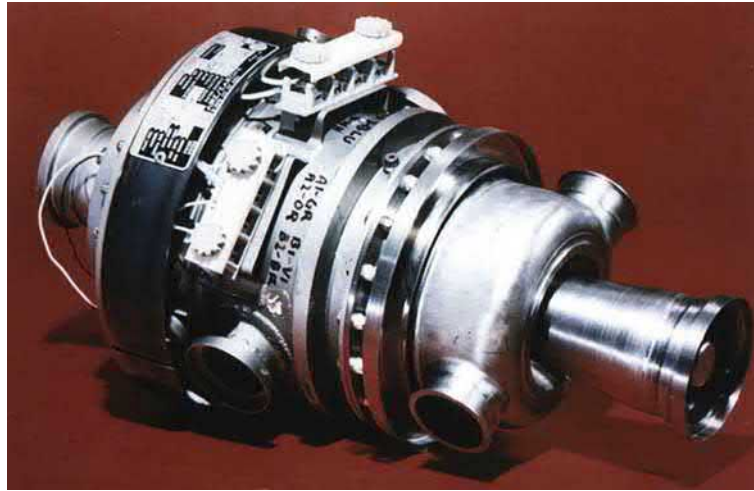


Fig. 6 Mini-Brayton Rotating Unit (Mini-BRU). The Mini-BRU was capable of producing up to 2100 We with a mass of 17 kg. (NOTE: Other components not shown.) Source: NASA.

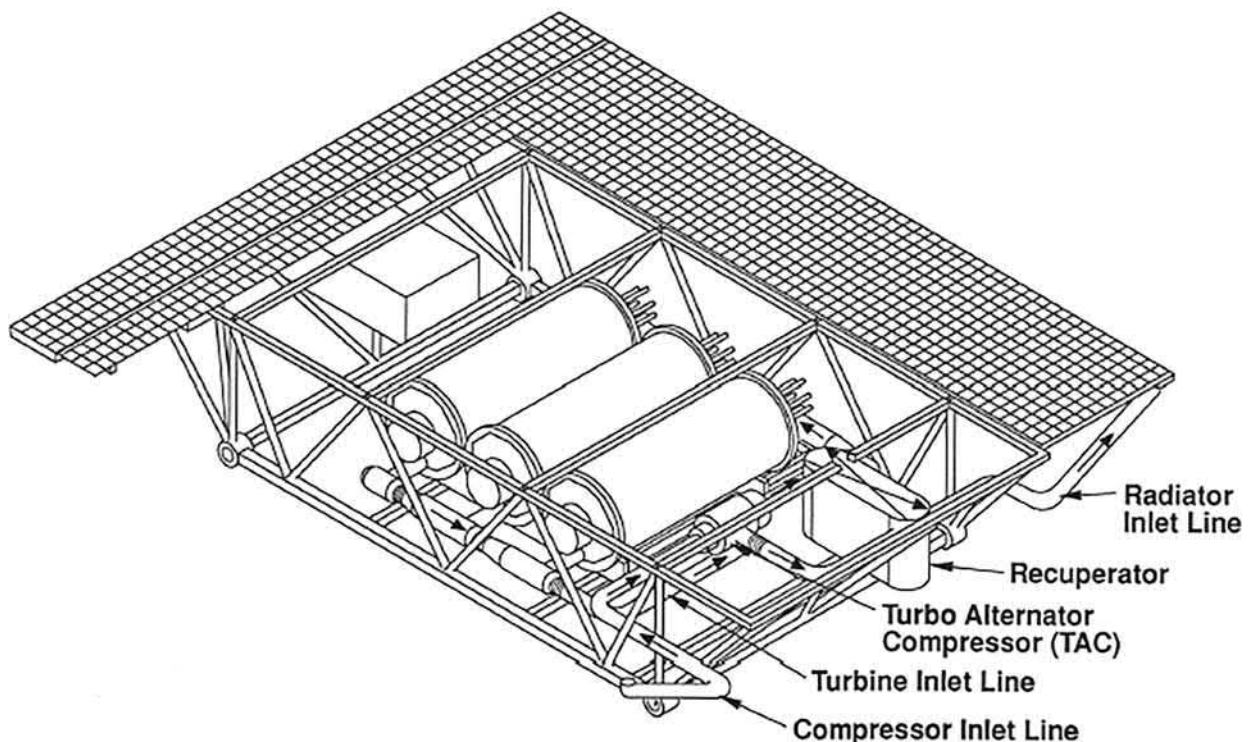


Fig. 7 Components of the proposed 2.5 kWe Modular DIPS Power Conversion Unit (PCU). Overall dimensions for the 2.5 kWe module are $2.44 \times 3.55 \times 0.5$ m (Rocketdyne, 1992).

More recently, Creare, Inc., using its micro-turbomachinery technology and working with Rocketdyne has developed a small Brayton unit that fits nicely into the lower power (hundreds of watts or less) regime often filled with RTGs. Fig. 8 shows the conceptual design (Breedlove et al., 2019).

Rankine cycle

Like the Brayton cycle, the Rankine cycle (named for Scottish professor William John Macquorn Rankine) converts heat into mechanical work and/or electrical power. The Rankine cycle accomplishes this by heating the working fluid until there is a phase change (e.g., from liquid to vapor). Many terrestrial power plants operate by heating water until it becomes steam; in essence, a Rankine cycle. The first U.S. RPS operated with the Rankine cycle to produce electrical power (see Fig. 9). Fig. 5 illustrates the overall Rankine cycle in a radioisotope power source.

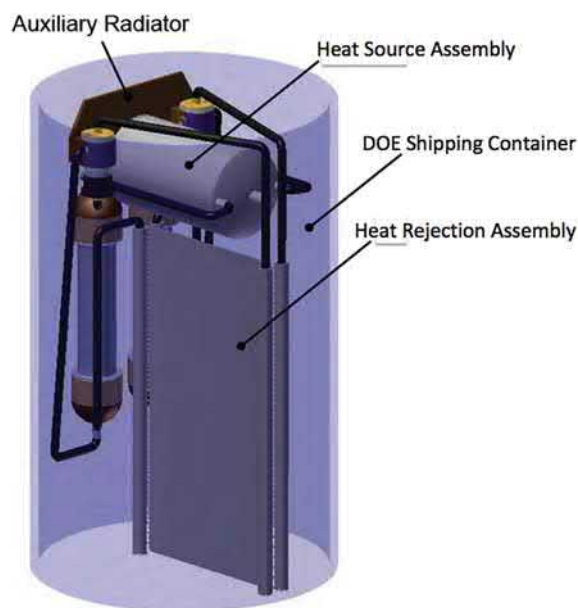


Fig. 8 Turbo-Brayton Converter concept. Predicted performance: 355 We at an efficiency of 26.3% and a specific power of 21.6 We/kg. Source: Creare (2018) *Turbo-Brayton Converter for Radioisotope Power Systems, RPS Community Review*.

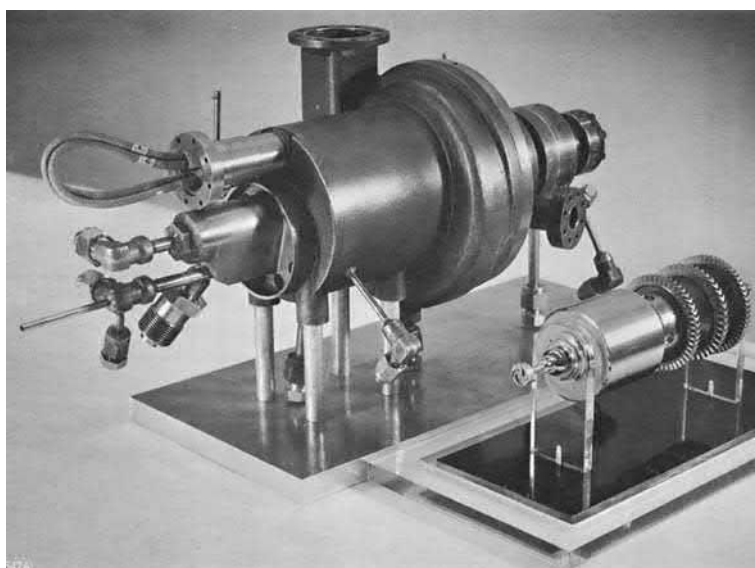


Fig. 9 SNAP-1 turbomachinery package with shaft assembly shown separately. Source: Thompson Ramo Wooldridge, Inc. for U.S. Atomic Energy Commission.

SNAP-1 was designed to produce ~ 500 We of electrical power by using a planned radioisotope heat source to boil liquid mercury at a temperature of ~ 977 K to drive a miniature turbine-alternator which would then be pumped back to the boiler to complete the cycle. The overall efficiency was just under 10% in a mass of 5.9 kg (without the heat source). Given that this was the first Rankine RPS, the results were very encouraging. And given that terrestrial Rankine systems have efficiencies in the range of 30% or more, the Rankine system offers a way to achieve higher efficiencies with less radioisotope.

Mercury Rankine converters were baselined for the higher-powered U.S. SNAP-2 (3 kWe) and SNAP-8 (10–50 kWe) space nuclear reactor programs.

For the Rankine part of DOE's Dynamic Isotope Power System (DIPS) program, a lower temperature approach using an organic working fluid, Dowtherm A, was employed. The turbine inlet temperature was ~ 640 K and the overall system DC efficiency was over 16% which was over twice that of thermoelectric converters. Fig. 10 shows the organic Rankine DIPS and an illustration of how the unit could be assembled.

Stirling cycle

A Stirling engine operates by the cyclic compression and expansion of a gas. A free-piston Stirling engine (FPSE) produces electrical power by the linear motion of the piston (instead of the rotary motion of a Brayton or Rankine system) (Sundstrand Corporation, 1982). Over the years Stirling engines have been tested in everything from automobiles to an experiment converting the heat of a nuclear reactor to electricity (such as might be needed at a lunar base).

The Stirling engine is named for Robert Stirling who developed one in 1816. One of the key advantages of the Stirling cycle is that it is scalable from the lower powers (hundreds of watts or less) associated with RTGs to the higher powers (kilowatts to megawatts) for larger applications.

In the early 2000s, NASA pursued the development of an Advanced Stirling Radioisotope Generator (ASRG) that would provide ~ 140 We at beginning-of-mission and a specific power of 4.7 We/kg with efficiencies of 20% or more. For comparison, the ASRG could have produced almost as much power as the MMRTG but with only two GPHS modules versus the eight GPHS modules in the MMRTG. Electrically heated ground test units were built and accumulated tens of thousands of hours of operation. Fig. 11 is a cutaway diagram of the ASRG concept and Fig. 12 shows the components of the Advanced Stirling Converter (ASC) which is the "heart" of the ASRG. Fig. 13 shows the engineering test unit of the Advanced Stirling Radioisotope Generator.

As illustrated in Fig. 12, inside the ASRG, the oscillating piston and its displacer move in synchronization inside a closed, helium-filled cylinder. The displacer forces the helium gas to move back and forth between the "hot side" (heated by a radioisotope heat source module) and the "cold side" (which radiates to space). This back-and-forth movement drives the magnetized piston through the coil of wire more than 100 times per second to produce an alternating current of electricity.

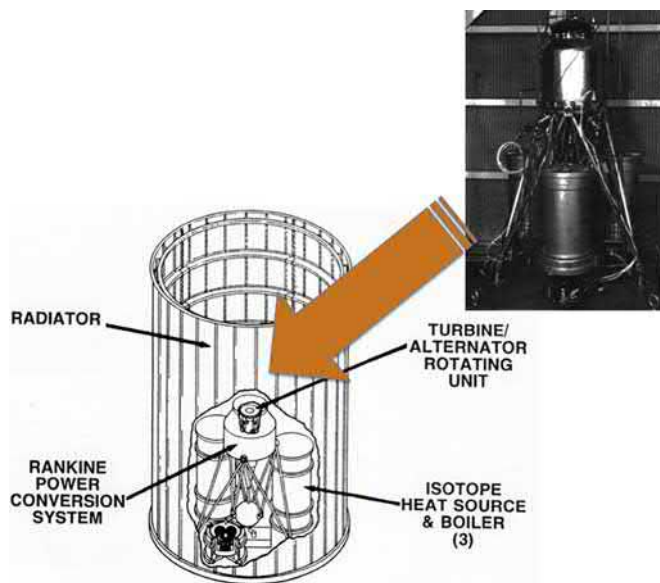


Fig. 10 Organic Rankine Dynamic Isotope Power System. Source: Sundstrand.

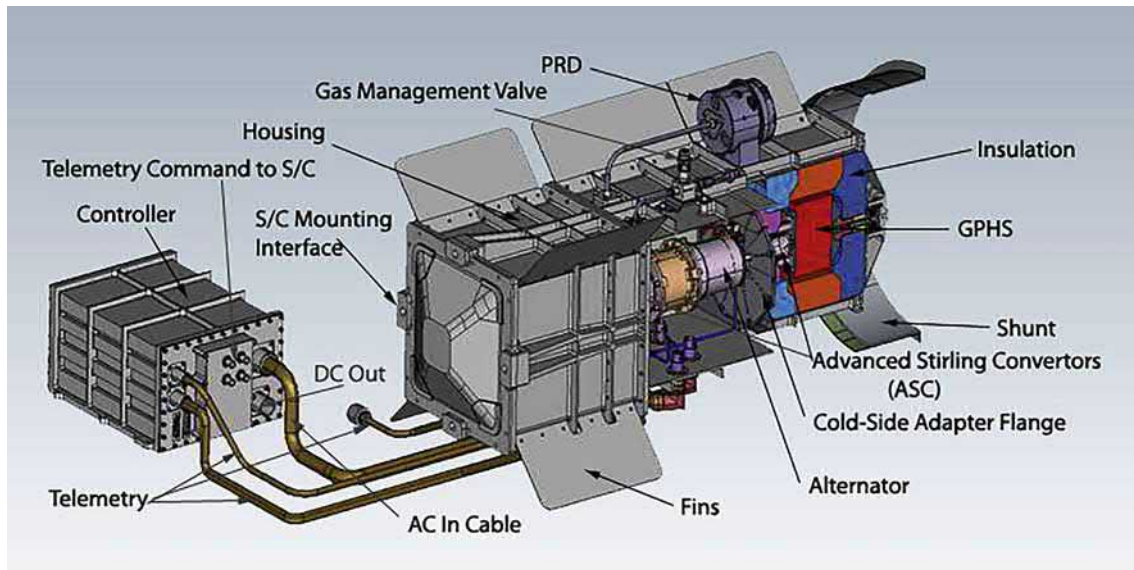


Fig. 11 Cutaway of the Advanced Stirling Radioisotope Generator (ASRG). Source: NASA.

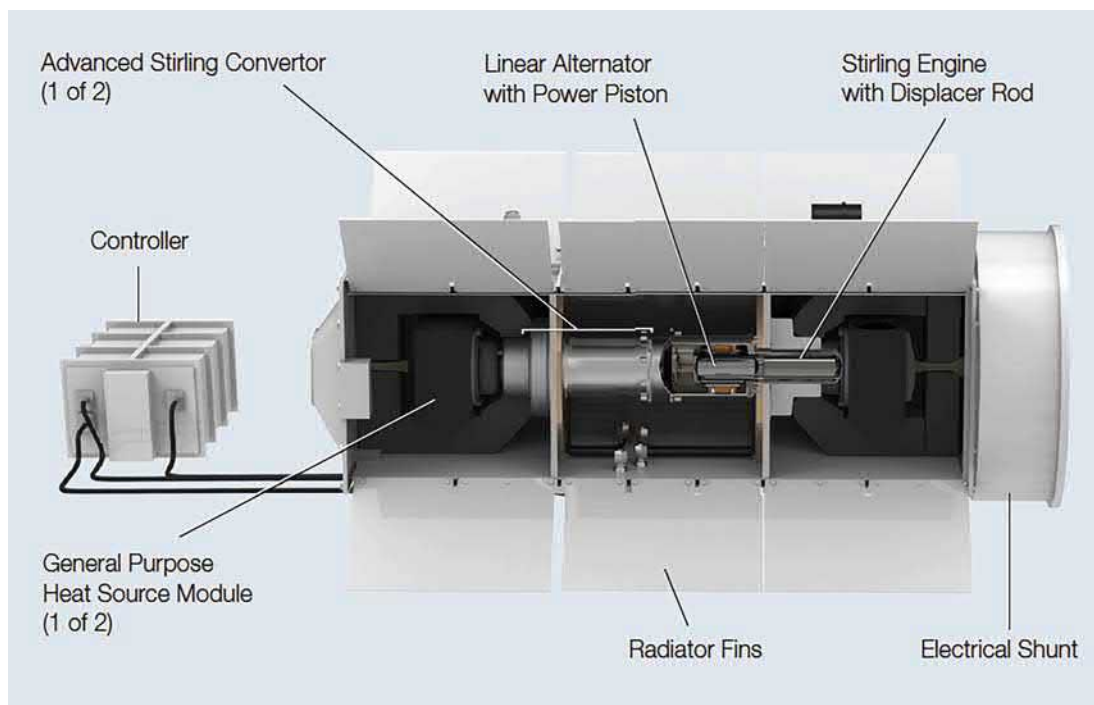


Fig. 12 Cutaway of the Advanced Stirling Converter. Source: NASA.

Concluding remarks

A range of static and dynamic conversion systems is available or can be made available for radioisotope power sources/systems. These conversion systems will allow RPS to achieve higher powers for the same amount of radioisotope fuel or to use less fuel to achieve a given power. Furthermore, dynamic conversion systems lend themselves to growth to kilowatts and megawatts of power for the advanced exploration missions planned for the future.



Fig. 13 Engineering unit of the Advanced Stirling Radioisotope Generator (ASRG). Source: NASA.

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Space Fission Power: Benefits and Challenges

David I. Poston, Space Nuclear Power Corporation, Los Alamos, NM, United States

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Introduction

The use of fission reactors for power in space has been envisioned since the dawn of the nuclear era. During that time, a handful of SFP systems have been launched into space; one by the United States in 1965 (SNAP-10A) and several by the USSR (ending in the 1980s). Since then, several programs to develop space reactor systems have been pursued, but none have resulted in success. This encyclopedia chapter discusses the benefits and challenges of SFP systems. The following chapter in the encyclopedia investigates the historical efforts to develop SFP. The subsequent chapters then address the technology options that might some day be used to establish fission power in space.

The need for a robust power source in space

On Earth, life and humanity has relied on energy from the sun (plants, animals, organic fuels, heat, etc.), but in space you're essentially on your own. Power is the single most important element to the survival of both spacecraft and humans in space, and an abundance of power is essential to providing safe and reliable missions—without power there is no existence. For human exploration, the need for abundant power provides a different paradigm that we've come to know on Earth. A reliable supply of sufficient power is essential to the life of an astronaut, and much more important to their well-being than radiation risk from a reactor.

Power can be binned into three areas when discussing space exploration: electricity, propulsion, and heat.

Electricity is the most valuable asset that space power systems can provide. Almost all terrestrial technologies have evolved within the paradigm of available electricity, and with enough electricity almost anything can be accomplished. It is hard to imagine any spacecraft or human outpost in which electricity will not be the lifeblood of the mission. Nomads and Pioneers could live off the land to explore and settle the Earth; space explorers can potentially do the same thing sans one component: electricity. Some things can be done more efficiently without our favorite energy "middleman," but there is almost nothing we can't do if we have abundant electrical power. For near-term scientific missions, increased electricity allows more capable instruments, increased instrument duty cycles, onboard scientific analysis, higher data-rate communications, and smaller antennas. For human missions, abundant electricity enables science and exploration, but more importantly it is the foundation of human safety and life support.

Space Propulsion is the other required source of space power. The first order physics of space propulsion is very simple—throw something off the back of your spaceship and you will go faster (Newton’s third law of motion). In this respect, the brute force approach to space propulsion is to launch as much propellant as you can into Earth orbit, and if launch costs can be made affordable this is a great option for relatively low delta-V missions (like a conjunction class mission to Mars). If launch costs are high, and/or you need to get somewhere fast, then the key to making propulsion effective is to gain as much thrust as you can for every unit of mass that you eject (Specific Impulse or I_{sp}). On top of this, the thrust to weight of your propulsion system and spacecraft can be very important. Ultimately, a power source can provide a propulsion system in three ways: the byproducts of the power source are used directly as the propellant (direct propulsion), the power source provides heat directly to the propellant (thermal propulsion), or the power source creates electricity that is used to accelerate a heated/ionized propellant (electric propulsion). In all cases, abundant energy/power is needed to enable effective human space propulsion (with due concern given to the effort required to place those energy sources and propellants into Earth orbit).

Heat is needed to keep things warm and to drive chemical processes—these functions can be completed via electricity, but they are done more efficiently without it (because heat generally creates the electricity in the first place). In space, thermal heating is needed for systems ranging from the smallest spacecraft to large human outposts. Where adequate sunlight is not available, NASA has routinely used the Radioisotope Heater Unit (RHU) to keep components warm in space, while waste heat from a large thermodynamic power system could keep entire spacecraft or human outposts warm at essentially no “cost” to the infrastructure. Process heat is important to create practical and sustainable human outposts elsewhere in the solar system. Many technologies have been envisioned, and in some cases developed to transform in situ materials to usable consumables (air, water, rocket fuel) and/or construction materials (ceramics and metals). One of the most important and fundamental use of process heat may be to melt ice. Again, if a large-scale power system is used to create electricity, the process should reject enough “free” heat to meet many of these needs.

In the very long term, we would need to use in situ resources exclusively to power a sustained space-faring existence. These in situ resources could simply be the materials to fuel our previously developed technologies, or the in situ resource might provide a unique energy alternative (using site specific chemical or geothermic resources). In all cases, an Earth-based energy technology that can provide abundant power is needed to (1) get us to our destination and (2) establish a semi-autonomous outpost/colony which can eventually evolve to a self-sufficient location.

Advantages of fission over other power sources

Abundant power is needed to significantly explore and expand into space. A robust power source is needed that can provide high energy and power density while being independent of location, environment, and application. The primary energy alternatives that are available in the near-term are discussed below.

Solar power

Using the sun as an energy source in space is often the preferred option, but is limited in power density and is location specific; i.e., the sun can become a poor energy source if you’re ever in the sun’s shadow or in a dusty/cloudy environment, or as you move deeper into the solar system. Even on Earth, the sun is not practical as a source of baseload electricity. Solar power on the Moon is significantly hampered by the 28 day cycle (i.e., 14 Earth-days of darkness). The technology to efficiently and reliably store energy in the deep cold on the lunar night presents substantial challenges, and would likely be substantially more complex and heavier than the solar power system itself. On Mars, the diminished sunlight decreases the value of solar power, but it is the dust, and more importantly dust storms that make solar power unattractive. The potential of a protracted dust storm will require more baseload power and a longer term storage system, presenting the same challenges as a lunar system (in addition to the decreased solar insolation during the day). Finally, the lower power density of sunlight beyond the asteroid belt would make the required size of a solar powered system excessive for a high power mission.

Chemical power

Since the dawn of the human race, our favorite energy alternative to the sun has been to burn things (primarily wood until the industrial revolution). On Earth we are surrounded by materials laden in hydrogen and carbon, fortuitously surrounded by a gas that contains oxygen (air). When we move about, we generally worry about our fuel supply but take the oxidizer for granted, which is the heavier constituent. In space, we have to bring both fuel and oxidizer with us, or have a means of generating in situ (which requires power). Space does not change fundamental chemistry (except for gravitational effects on combustion), but using chemical power in space is extremely inefficient because of all the power needed to transport or create constituents in situ. In addition, storing these constituents in the extreme temperatures of space presents problems in logistics and reliability.

Energy storage

Energy-storage systems (e.g., batteries) have a role for almost any space mission, but generally only in the very early stages. The advantage of batteries is that we can use low-cost Earth-based technologies to generate the energy, and then extract it from the

storage technology to utilize it in space. There is considerable ongoing research in fuel-cells and alternative means of storing energy, which could be much more mass efficient than traditional batteries; however, there is no chance that a chemical-energy storage system would ever be practical to power ambitious space exploration. Energy storage is of course essential to many space power architectures, in particular solar power, but not as the primary power source, where the energy is launched from Earth. Some nuclear sources, such as anti-matter and even ^{238}Pu might fit within the definition of an energy storage system, but they are better classified as nuclear power sources.

Nuclear power

Nuclear power can be used to provide electricity, propulsion and/or heat in space. The major benefit of nuclear technology is that power can be provided in a robust form that is independent of location or application. There is a great deal of experience in space radioisotope power systems (Bennett, 2021), and limited experience in space fission power systems. On Earth, the physics of fission are well understood, and engineered systems are well established. Other nuclear power types like fusion, anti-matter, or options like triggered isotopes, are not well developed for either terrestrial or space application.

“Nuclear power in space” encompasses a wide range of sources, technologies and applications. The physical sources of nuclear energy that can be utilized are radioactive decay/transition, fission, fusion, and antimatter. The technologies to harness these forms of energy are almost limitless, but the list of practical near-term technologies is rather small. Various combinations of nuclear power sources, technologies, and applications can be used to enable our exploration and expansion into space. In the end, the attractiveness of nuclear energy in space is the ability to provide robust, long-lived, high power density (kW/kg) systems at any location.

The current forms of nuclear power that we know and can reproduce on some level are radioactive decay, fission, fusion and antimatter. In the table below $\text{LH}_2\text{-LOx}$ (liquid-hydrogen-liquid-oxygen) combustion is what is used in high performance chemical rockets (e.g., the space shuttle main engine), Pu-238 decay is used in radioisotope power sources (Bennett, 2021), $\text{D-}^3\text{He}$ fusion is the most likely space fusion power source, because it will be easier to utilize the energy (due to lack of high energy neutrons) and the potential ^3He inventory on the Moon, and antimatter represents the annihilation of a proton with an anti-proton (Howe, 2021).

Energy source	Energy density	Max. thermal power density
$\text{LH}_2\text{-LOx}$ combustion	13 MJ/kg	Limited only by engineering
Pu-238 decay	2,100,000 MJ/kg	0.54 kW/kg
U-235 fission	82,000,000 MJ/kg	Limited only by engineering
D-He3 fusion	354,000,000 MJ/kg	Limited only by engineering
Antimatter	90,000,000,000 MJ/kg	Limited only by engineering

The two key points of the above table is that (1) nuclear power sources offer enormously higher energy densities than chemical systems, and (2) that power density is limited for ^{238}Pu . Fission, fusion and antimatter can all provide energy and power densities beyond what we could feasibly utilize in the foreseeable future. However, discussions of fusion and anti-matter are essentially moot for decades or perhaps centuries to come, because even if we engineer the technologies required to make these power sources practical, we will not have the technologies to engineer the high power density systems needed for space application. Restated, fission has the ability to provide energy densities so high that it will take many generations of technology advancement (materials and fabrication) until we could take advantage of a higher energy density power source (even if that source itself was off the shelf). Therefore, since fission technology is established and well-understood, it is the obvious technology to focus on for space nuclear technology development.

It is possible that once we establish large scale settlements on Mars or elsewhere, that we could shift away from nuclear power to an in-situ source (geothermal power, solar power with an in situ panels and energy storage mechanisms, wind power where there is an atmosphere, or perhaps nuclear if we could mine thorium, uranium, D, He-3, etc.). Until then, we need to have a technology that can enable the energy intensive missions and operations that could get us to that end goal, and that power source is fission power.

Difference between space and terrestrial reactors

Space fission systems are significantly different from terrestrial fission systems, due both to the requirements and operating environment.

Things that make space reactor design more difficult than on Earth:

- Waste heat generated by a space fission system must be rejected via thermal radiation (except for a few surface power options), whereas terrestrial systems typically transfer waste heat to water or air. This difference results in a strong performance incentive to operate space fission systems at a higher temperature than their terrestrial counterparts.
- Most space reactor applications require the system to operate autonomously for the life of the mission with no maintenance, whereas terrestrial systems typically can be maintained/repared as needed. As space infrastructure grows over the decades, the opportunity for maintenance will grow, but the first generations of space fission systems will be mostly on their own.

- Space fission systems must be capable of: (1) being packaged into a payload shroud, (2) withstanding launch loads, (3) remaining safe during launch accidents, (4) self-deployment, (5) operating in zero or reduced gravity, and (6) surviving potential micrometeorite impacts.
- Low-mass is very important to space fission systems, and the performance of a space fission system often is measured in terms of its total mass (kilograms) divided by the kilowatts of power it produces (kilograms per kilowatt electric). This parameter (specific mass) is of little or no importance to terrestrial fission systems.
- Shielding requirements combined with the desire for low mass present issues unique to space reactors. On Earth many effective options are available, including plentiful supplies of dirt and water, heavy but effective structural materials like concrete and steel, and the potential to create significant distance/separation. In space, separation is limited, mass effective materials are important, and issues like thermal management and structural integrity (launch, landing, and operation) can be significant engineering issues.

Things that make space reactor design easier than on Earth:

- Space fission power systems will generally operate at power levels three to four orders of magnitude lower than traditional terrestrial nuclear power plants. This difference enables numerous design and control simplifications to be made, and allows long-lived reactors without refueling. First generation fission power systems might provide 1–30 kWe, and ultimately grow to 1–10 MWe power systems. Beyond 10 MWe the technology would probably no longer be considered a “space reactor”; i.e., it would probably look more like a terrestrial power plant. This is because (1) the need for such a high power indicates that a very robust infrastructure would already be in place, which could provide significant materials, excavation, assembly capability to construct a power plant and (2) the mass and size of the power system would eventually be bigger than a practical launch vehicle could bring from Earth.
- Terrestrial reactor design is focused on being able to meet rigorous safety requirements as demonstrated by consequence-based (dose-to-public) analyses of an extensive series of design-basis accidents (DBAs). Because space fission systems will not operate until they reach a high orbit, consequence-based accident analysis is mostly limited to launch and reentry accidents starting from a cold shutdown condition. As human presence increases, more attention will have to be made towards operational safety, but contaminations and environments issues should still be of less consequence.

These (and other) differences can result in different fuels, materials, and coolants being chosen for space reactors than are typically chosen for terrestrial systems. However, the infrastructure, experience, and trained personnel available from terrestrial use of fission technology are still of great use in the development of space systems.

Applications/benefits of space fission power

A key benefit of space fission power is its versatility, such that it can provide robust power for a wide variety of space applications.

Space power

Technically, any fission power source used off the Earth might be considered space power, but in this article “space power” refers to systems that provide electricity to spacecraft in the vacuum of outer space (in or beyond Earth orbit). Every reactor ever launched into space, either by the United States or USSR, has been in this class. The attractiveness of the space reactor versus other sources depends primarily on two parameters: power level and the availability of sunlight. Power levels $\ll$ 1 kWe are best provided by radioisotope systems, assuming that the supply/utilization of the radioisotope is not prohibitively expensive. Solar power is generally the best option inside of the asteroid belt, as long as the duration of eclipse is small and orientation can ensure that some of the solar panels always have a view of the sun. Therefore, the best use of space reactors is for missions where a power approximately equal or greater than 1 kWe is required, and the availability of sunlight is diminished by distance, eclipse, or attitude.

High power missions ($\sim$ 1 kWe or greater) to the outer solar-system are the most clear-cut applications for space reactors. Solar intensity drops by the square of the distance from the sun. Solar intensity is shown at various locations in the solar system in the table below.

	<i>Distance from sun (1 AU $\sim$ 1.5 e + 8 km)</i>	<i>Relative solar intensity</i>
Earth	1.0 AU	1.0000
Mars	1.5 AU	0.4328
Vesta	2.4 AU	0.1736
Ceres	2.8 AU	0.1276
Jupiter	5.2 AU	0.0370
Saturn	9.6 AU	0.0109
Uranus	19.2 AU	0.0027
Neptune	30.1 AU	0.0011
Pluto	39.5 AU	0.0006

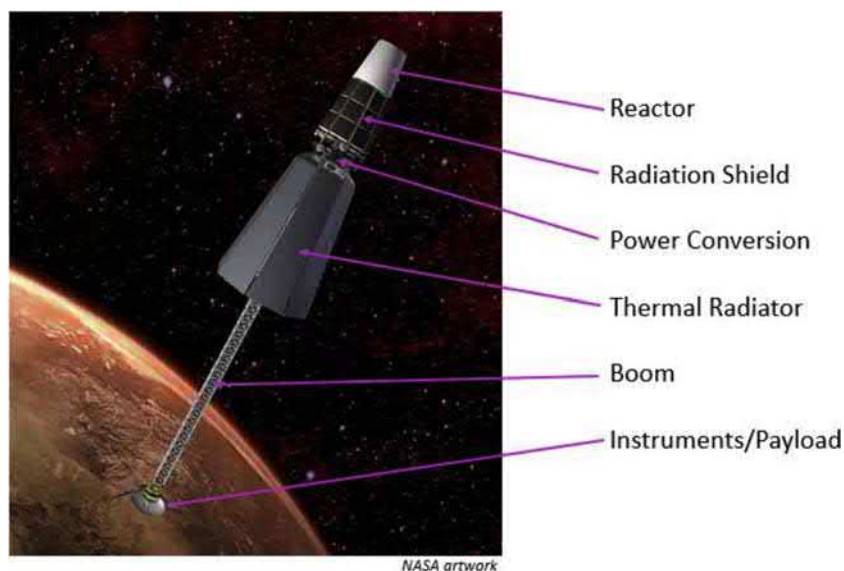
Beyond the asteroid belt, solar power becomes very difficult: <4% of the solar intensity at Jupiter versus the Earth. Amazingly, solar technology has progressed enough to make the low-power (500 We) Juno mission (Bolton, 2017) to Jupiter practical. Thus at Jupiter, solar power competes well with radioisotope power, but at powers > 1 kWe fission should have an advantage. At Saturn and beyond it is hard to envision solar power becoming practical, so the trade will again be between radioisotope and fission.

Inside of the asteroid belt, with a clear view of the sun, the only time fission might make sense is at very high power levels ($\gg 10$ kWe). The reason for this is that the radiator of the fission power system will generally be much smaller than a solar panel array (for the same power). In this case the reactor might have an advantage in deployability and spacecraft maneuverability. The International Space Station (ISS) has a solar power array that provides a cycle-averaged output of ~ 100 kWe (> 200 kWe in direct sunlight) (Gibb, 2018); however, this array is massive and cumbersome. Another advantage of the relatively small radiator might be the ability to avoid detection, if desired. Solar panels reflect considerable sunlight and to work they have to face the sun; the ISS is brighter than Venus under the right conditions. A radiator does not have to face the sun, and generally will be designed not to in order to achieve maximum efficiency. The reactor is very hot, but it will be surrounded as much as possible by insulation to prevent thermal losses, such that the outer surfaces will be cool (and of very small profile). Nuclear radiation from the reactor might be detectable to spacecraft within 10s or even 100 s of kilometers by a highly specialized detector, but would be virtually impossible from 1000s of kilometers or from the surface of the Earth. The detectable distance would depend greatly on the reactor power and what fraction of radiation leaks from the reactor.

Currently, there are not many space missions under consideration that might require > 1 kWe in space (other than propulsion, which is discussed later). This is largely a chicken-and-the-egg dilemma—no one will design/develop a high power instruments or spacecraft because high power levels are not currently available, while no one will design/develop high-power power systems because no one is asking for them. While NASA science missions have been extremely successful since the 1960s, there has been no fundamental change in where we can go and how much power we have when we get there. We could increase the benefits of science missions by orders of magnitude with a robust, low-mass source of abundant power. The high cost and limited availability of ^{238}Pu also increases the future need for space fission power.

One of the key drivers for high power space fission systems might be communications. As more expansion into space occurs, around the Moon, Mars and beyond, there will be a need for a more bandwidth, which eventually might benefit from the high powers of a space reactor. For ambitious outer solar system missions, a high power source might be needed to signal strength. For all type of science missions, more power might be useful to provide more ability for interrogation, by offering high power/sensitivity sensors, probing signals (for subsurface mapping), ablative lasers, etc. Most importantly, there will likely be applications for high power that have not been considered yet. Many space scientists have focused their careers on trying to develop instruments that require 10s of Watts, instead of trying to envision instruments that use 100 s of Watts to kilowatts. This is partially because higher power instruments are generally heavier, but the recent reduction of Earth-to-orbit launch costs and the increased lift capability should alleviate that concern. Overall, it is likely that once abundant power is available, that people will find innovative ways to use it.

A typical space reactor configuration is shown in the figure below.



Surface power

The other major application of space fission power systems is surface power; i.e., power on the surface of any planet (other than Earth), moon, asteroid, or comet. Surface reactors could be the most practical near-term application of space fission systems, because of the difficulties of using solar power on the Moon and Mars. Fourteen days of darkness on the moon would require

substantial power storage if a solar-power system is used, including the ability to provide thermal heat. In addition, all of the concentrated and easier-to-process water of the Moon is in permanently shadowed regions; it might be easier to land a reactor there rather than try to deploy cables to a sunlit region. Mars presents a rather different challenge than the Moon. The length of a day (sol) is only slightly longer than an Earth day, so energy storage can be more practical. However, Mars is further from the sun (~ 1.52 AU). The solar intensity reaching Mars is $\sim 43\%$ that of the Moon (or Earth orbit). Mars also has a thin atmosphere which attenuates sunlight. The nominal optical depth ranges between 0.5 and 1.0 (depending on elevation, season and latitude), such that between 40% and 60% of sunlight reaches the surface. The biggest challenge is the potential for long-duration (months) dust storms, which can cause the direct sunlight to be reduced to $\ll 10\%$ of nominal, although scattered/diffuse light lessens the impact of the dust storm on solar panel output. This dust-storm issue requires several times more solar panel area/mass that would be required otherwise, and makes solar power more massive and complex than a fission power system. Surface power might also be of great interest on many of the moons in the outer solar system: e.g., Europa, Titan, Enceladus, Triton, etc.

Simple surface power systems could be the workhorse of the near-term human exploration infrastructure on the Moon, Mars, or other potential destinations (e.g., Titan, large asteroid). Some potential surface power electrical loads include habitats, in situ resource utilization plants, rechargeable rovers, construction equipment, and science experiments. The power output of a single workhorse surface system might be in the range of 10–40 kWe, with a lifetime of ~ 10 –20 years. In addition, the relatively low-power and “high” mass allowance of a surface reactor might make it the easiest space fission system to develop. For a human surface exploration mission, fission power has so many advantages over other power alternatives that the mass requirements will not be as stringent. This allows “mundane” reactor technologies to be used (i.e., stainless-steel, UMo fuel) at very benign power/flux levels, which makes development simple. Plus, as part of a “heavy” mission architecture there should be less programmatic pressure to meet mass targets, or worse, potentially decreasing mass requirements during development—a contributor to the downfall of the SP-100 program (Truscello, 1992).

The uses of electricity on the Moon, Mars, asteroids, etc. are vast. Nearly all technologies will be developed to run on electrical power, or off of another technology that uses electrical power (i.e., a pneumatic device would still use an electrically-driven compressor). There are several areas where abundant electricity are likely needed.

In situ resource utilization (ISRU): The ability to process materials that are available on-site (in situ) at your destination is essential to establishing a practical long-term presence anywhere in space. ISRU generally requires lots of power, and in most cases a lot of electricity. The most enabling aspect of ISRU might be rocket propellant production for return propulsion, assuming that a one-way existence is not preferred by most early space explorers and settlers. The rocket equation makes Mars missions extremely inefficient if you have to carry and land propellant on Mars in order to launch yourself back off of Mars. The ability to create oxygen from the abundant CO₂ supply may be the difference between practical and prohibitively expensive round trip travel to Mars (Zubrin, 1991). If a supply of hydrogen is available then both methane and oxygen can be made from CO₂ via the Sabatier reaction (Zubrin et al., 2013). The ideal situation would be to gain hydrogen from water/ice (which also would produce more oxygen), but early missions might bring hydrogen from Earth, since it is low mass relative to the oxygen. As the Moon, Mars, or asteroids become more industrialized, then power could be used to mine and process all of the materials required to sustain an infrastructure. The rapid advent of additive manufacturing, or 3D printing (Betzler, 2021) will make the effective utilization of in situ materials even more practical. Ultimately, if our capabilities in space become vast, then mining remote resources in space and returning them to Earth would be practical as well—there are several startup companies already hoping to do this, but it is probably many decades from potential profitability.

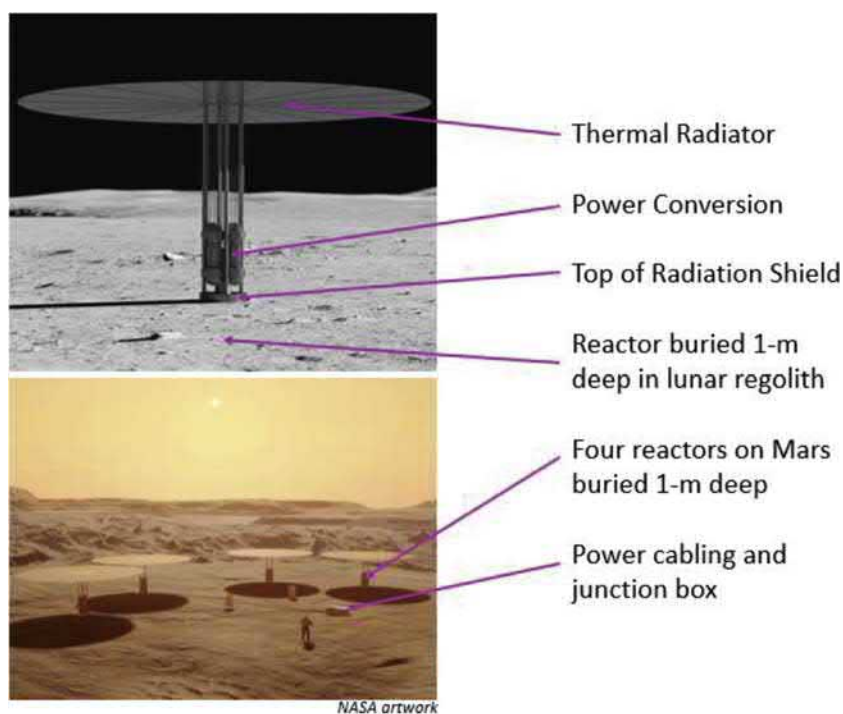
Life support: The first aspect of life support is tied to ISRU; i.e. the ability to create breathable air, drinkable water, and eventually an environment to produce food. The other aspect of life support is that most systems that maintain life support will be run by computers/electricity. Heating may not be electrical; it might be provided by “waste” heat from the reactor, or if the habitat is very well insulated (e.g., buried) then heat from the humans, computers, and appliances might be enough to keep the habitat warm by themselves. Overall, it is unlikely that it will be practical to develop a space life-support architecture that can operate without electricity, whereas this has always been very simple on Earth (although our lives become increasingly dependent on electricity every year).

Surface Transportation: Logistics dictate that electrically-powered vehicles will be the standard of transportation on the Moon or Mars. The rapid emergence of practical electrically-powered cars on the Earth makes this even more certain. Small fission reactors could create an ideal network of charging stations for mobility. A reactor would have a strong advantage over solar power in this application to allow transit on the Moon day-or-night, or to the “winter” side of Mars. For larger settlements, an electrically powered shuttle or trains would make sense: e.g., Martian Rails from the game Terraforming Mars (Fryxellius, 2017). In all cases, once a colony grows to a substantial size, the high power requirements for transportation will strongly favor a reactor.

Infrastructure and Habitat: Just as on Earth, everyday life on another planet/moon will use electricity for almost everything: lighting, construction, communication, science, exploration, refrigeration, computers, displays, entertainment, drilling, etc. The average household on Earth uses 2–3 kWe averaged over the course of the day, but can be much higher during peak demand. A remote isolated outpost would probably require more energy consumption. Considering all of the uses of electricity in this and prior paragraphs, it is reasonable to expect that the power system should be sized to produce 5–10 kWe per person on the Mars.

As human presence in space increases, then the economical paradigm would probably shift from “bring your power source with you” to “buy your electricity from an extra-terrestrial utility company.” This will happen similarly as it did on Earth, although there are still remote communities and mining operations that find it more economical to bring in their own power supply, instead of trying to connect to the grid.

A typical application of a surface power system and grid is shown in the figure below. The power systems shown are 10-kWe Kilopower reactors (McClure et al., 2020).



Power for nuclear electric propulsion (NEP)

Nuclear electric propulsion (NEP) utilizes space fission power to provide electricity for an electrically-powered thruster; the reactor for NEP is technically a space reactor, but is often considered in a separate class. In reality there would be very few differences between a reactor designed to power an electric thruster as opposed to a traditional spacecraft bus. In most cases the electricity would be used for both spacecraft power and propulsion, although the power conditioning (voltage, frequency) might be different.

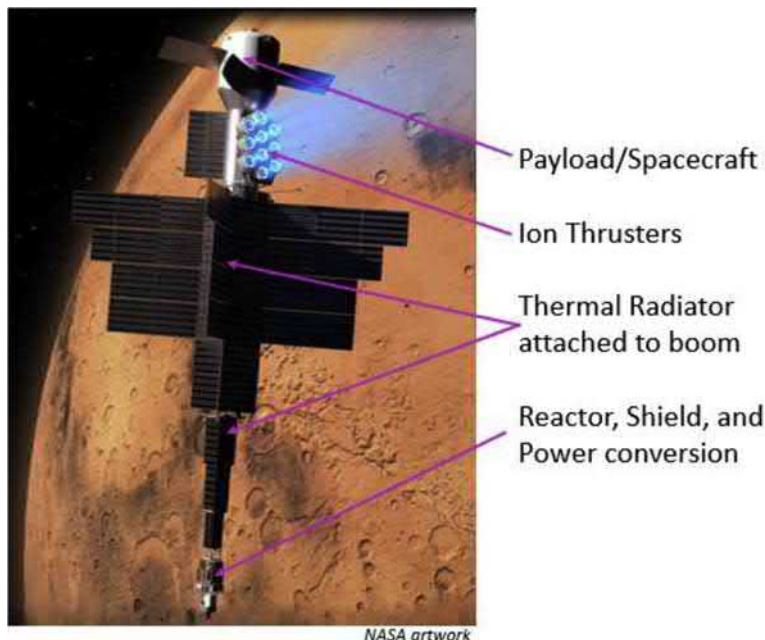
The specific mass (kg/kWe) for NEP systems is generally much more important than for other space reactor applications, because of how system dry mass factors into the rocket equation. NEP reactors usually trend towards higher power levels because this generally lowers the specific mass, and because the thrust of electric propulsion systems tend to be low. Any bit of additional thrust (power) can be helpful in scenarios where trip time is limiting or the mission involves escape from a gravity well.

The best near term use for NEP is science missions to the outer planets. A recent study <ref-Casani> found that a 10 kWe NEP systems can enable numerous missions, including a tour that orbits (and deploys landers on) the Saturn moons Titan and Enceladus or a spacecraft that orbits Neptune and then its moon Triton and deploys a lander. The study also evaluates enabling missions to orbit Centaurs, Uranus, Pluto, and beyond.

Specialized NEP systems might also be of use in Earth orbit or cislunar space (any region between Earth and the orbit of the Moon) for missions to perform spacecraft maintenance, clean space debris/junk, or alter, maintain, or boost the orbits of satellites. An NEP system might also be used to help deflect an asteroid destined to hit the Earth, either to deliver an explosive charge or provide tugging force itself. An attractive option might be a gravitational tug, where the spacecraft constantly thrusts away from the asteroid at a force equate to its gravitational pull. The use of NEP for planetary defense will of course depend on the amount of advanced warning, and the power/thrust capability to adjust the orbit of asteroid/comets enough to miss the Earth. A high power, high specific impulse NEP system might also be needed for economical mining and transport raw materials from asteroids.

A high performance NEP reactor (> 1 MWe, < 10 kg/kWe) offers more potential benefit to long-term human space exploration than any other space reactor application. Whereas Nuclear Thermal Rockets (NTRs) offer a factor of 2 performance benefit over chemical rocket technology, a higher performance NEP system can offer orders of magnitude better performance (ISP)—a truly enabling and paradigm shifting technology. Advances are continuously being made in all of the non-nuclear technologies needed for an NEP system, most importantly power conversion, heat rejection, and thrusters. Significant advances are being made in ion thrusters and plasma thrusters, such that the deployment of high power EP systems appear possible within a decade or two. Conversely, a reactor for this kind of NEP system is essentially no closer to deployment than it was 50 years ago, just as NTRs are further from being flight-ready than they were 50 years ago. The difficulty in a high-performance NEP reactor is not achieving high power, but delivering power with a very low mass system, which requires very high temperature operation due to heat rejection requirements. It might require several reactor generations (technology evolution in steps) to reach performance levels for human

propulsion missions. This evolution can start with simpler space science or perhaps cargo missions, which will have simpler requirements but still provide useful performance for their application. The use of fission reactors for NEP is discussed in more detail in another chapter in this encyclopedia ([Morrison, 2021](#)).



Power for process heat

The use of thermal power for process heat is another potential application for space fission power; i.e., for heating, fuel production, ice melting, etc. In many applications, process heat might normally be provided by the waste heat from an electrical reactor power system, so technically the reactor would not be designed solely for process heat. However, there may be instances where thermal power is the primary requirement of a reactor; perhaps melting tough ice, producing chemicals via thermal processes, creating steam from Martian soil, etc., but even in those cases some electrical power conversion might be incorporated into the system design for bimodal use.

Challenges of space fission power

The need for low mass (high temperature)

To be useful or economical, a space fission system must have low enough mass to be launched into space, and in many cases transported to and/or landed at its final destination. The key distinction is low power system mass, as opposed to low reactor mass. In almost all cases, this requires a high temperature reactor, because most space environments do not have a thick atmosphere or bodies of water to provide an efficient heat sink. The efficiency of most power conversion technologies is rooted in the Carnot equation, or the ratio of the system hot-side (reactor) to cold-side (radiator) temperatures. If thermal radiation is used for heat rejection, the cold side usually needs to be ~ 400 K or higher (depending on sunlight, surroundings, etc.) to keep radiator size/mass reasonably small. This requires a reactor temperature of ~ 900 K to obtain reasonable thermodynamic efficiency, when taking into account various temperature gradients, losses from the core to the radiator, etc. If efficiency is too low, the need for heat rejection becomes a vicious circle; as efficiency drops, reactor power and radiator size increases, and the larger temperature gradients cause a further drop in efficiency, and so on.

A practical entry point for space reactor temperature is ~ 900 K or higher; this essentially precludes the use of water-cooled reactors, which is unfortunate because water-cooled reactor technology is highly developed. However, even if temperature wasn't a problem, water-cooled reactors would be troublesome for deployment in the cold temperatures of space (freeze/thaw issues) and for long-duration operation without maintenance due to chemistry concerns. Fortunately there are many existing reactor (fuel, clad, structure) that can provide power at ~ 900 K or higher—these are discussed in another chapter in this encyclopedia ([Poston, 2021a](#)). There are also several technologies to efficiently transfer reactor heat to the power conversion system ([Poston, 2021b](#)).

The need for low mass also requires materials to be mass effective. All other things equal, a space reactor design prefers: (1) Fuel-forms with higher uranium loading ($\text{U-metal} > \text{UN} > \text{UO}_2$) and pure fuel forms as opposed to particle or composite fuels

(e.g., TRISO, solid composites, etc.). (2) Structural materials with the highest strength to mass ratio in the desired temperature and irradiation environment. (3) Core materials (fuel, structure, neutron reflector, etc.) with low neutron capture other than fission. (4) Shielding materials with ability to scatter/capture neutrons with the lowest mass at modest temperatures.

The need for low mass sometimes dictates the use of materials that might be expensive or difficult to work with, for example beryllium as the neutron reflector or lithium hydride as the neutron shield (Poston, 2021c). Of course there are always exceptions to any of the above, and there is no single best material for space reactors in general (except perhaps beryllium). The combination of mass, power, temperature, stress, lifetime, cost, fabricability, availability, etc. must all be considered to determine the optimal material, but mass is usually one of the most important considerations.

The need for autonomy

Most space reactor applications will require the ability to operate reliably without any remote control or maintenance, especially for initial reactor generations before a substantial space infrastructure is established. Autonomous system control of some type is essential, because of the time delay between any space application and the Earth. Certain commands for startup and state changes could be given remotely, as long as a time delay is acceptable: seconds to the Moon, minutes to Mars, and hours to the outer planets.

The need for autonomous control places a strong emphasis on the ability to reliably instrument a reactor power system, and interpret and process the diagnostic signals. Solutions to these issues have been developed for reactors on Earth, but there are many issues unique to a space reactor. It is possible that some diagnostic instruments used on Earth might be too heavy, be launch sensitive, require too much power, have issues with extreme cold, or have trouble being packed close the reactor (irradiation and internal heating). On Earth, there is often the ability to replace faulty instruments, or interrogate instruments and determine if they are reading accurately, and perhaps recalibrate them. Also, terrestrial reactors generally have more room to include/attach diagnostics, more space to route cables, plentiful power, considerable floor space and shielding options for instrumentation racks, and no concern for mass. This allows terrestrial reactor designers to preclude most potential diagnostic issues by over-instrumenting the system; i.e., by providing enough diversity and redundancy to ensure the ability to get a reliable signal. If there are enough signals, especially if they are from diverse instruments, it is relatively easy to determine which readings are solid, which are reasonable, and which are bogus. In space, there will be diminished ability to any of the above.

The difficulty of ensuring accurate and dependable diagnostics is not the only instrumentation and control issue for a space reactor. The signals need to be interpreted and processed, and then commands must be given to control features, which must work properly upon command. The control computer must withstand radiation and have a thermal path to a heat sink to cool the components. One of the biggest concerns might be single event upset (SEP) caused by reactor radiation (mostly neutrons). Radiation hardened (rad hard) electronics exist for use in space, but they are generally designed without consideration of neutrons (e.g., they are hardened for solar radiation, galactic cosmic rays, Van Allen belts, etc.). The tolerance of many “rad hard” electronic components to neutron irradiation is unclear. The mitigation of potential electronic failure modes is redundancy, but this adds mass, power, cooling, and complexity concerns—the difficult question for any space reactors will be how much risk to take versus the “cost” of instrumentation and control redundancy.

The need for reliability

Many missions will only be practical with the use of only one reactor, thus the reactor power system as a whole represents a single-point failure. If a space reactor cannot be effectively designed with inherent redundancy, then components must be designed to be extremely reliable. The best path to high component reliability is to use proven technologies and design with large performance margins, but this often competes with the desire for low mass. Simplicity is another important factor that can reduce uncertainty and improve reliability. One of the biggest components that affects reactor reliability is the selection of cooling technology. A heat pipe reactor can offer redundancy, but might be more difficult to fabricate than a pumped fluid system, which is likely to have single point failures.

For human missions that rely on a fission power system, reliability is tantamount to safety. A loss of power in space or on Mars is likely a much bigger risk to astronauts than reactor radiation. This places reactors in a different paradigm, where more emphasis is placed on evaluating design basis accidents that impact system integrity, rather than radiation release. Another approach for human missions is to deploy several smaller reactors; i.e., if a mission requires 40 kWe, then deploy five 10-kWe reactors. This approach may be preferable, especially if it is simpler and cheaper to develop a 10-kWe reactor. This approach also provides more mission flexibility.

The need for long lifetime

For many applications, the best metric for the utility of a space reactor is probably energy (Joules), rather power (Watts), whether it be J/kg or J/\$. The costs of fabrication, launch, and deployment make it inefficient to frequently launch replacement reactors. This is true to potential surface outposts, long term orbital, cislunar missions, and high delta-V NEP missions (an optimal delta-V is best achieved with high J/kg system, depending on many factors). In these cases, lifetime is just as important as power. In addition, many deep space missions require long lifetime regardless, due to long travel times.

The desired lifetime of most space reactors might be measured in decades rather than years. This is no different from terrestrial power plants, except that there is no chance for refueling or replacing various life-limited parts. The desire for longer lifetime can exacerbate almost every engineering issue in the reactor: burnup reactivity swing (i.e., the deviation from criticality caused by fuel burnup), fuel performance (burnup related swelling or cracking), fission gas release, erosion, corrosion, mass-transfer/diffusion, creep strain, and irradiation damage (especially loss of ductility with dpa—displacements per atom).

The possible need for nuclear-powered system testing

Perhaps the biggest challenge to deploying a space fission power system is the potential need for a ground nuclear-powered test (GNT). Testing of new nuclear systems has become almost prohibitively cumbersome and expensive—this is surely part of the reason there has not been a successful new reactor deployed in the US for many decades. In addition, the testing of systems that operate in space presents other unique challenges. The combination of these two, nuclear and space, puts space reactors in an unfortunate class of their own. For a traditionally engineered system, the development approach can allow significant uncertainty in system performance, and a vigorous test program can be used to flush out performance issues and ultimately optimize performance (even for terrestrial reactors, as long as there is a chance to perform exploratory low-power reactor operations, and make subsequent changes, before extended full-power operation). This allows the design and development process to focus on what most view as traditional engineering: materials, procurement, fabrication, component development, and assembly. Alternatively, due to the cost and risk of testing, a space fission system must be designed so that a system “test” is in effect an actual “demonstration,” with as few uncertainties and unknowns as practically possible.

The primary purpose of a GNT would be to test the steady and dynamic performance of the integrated reactor power system. In most cases, other risks can be addressed in less expensive ways; e.g., nuclear lifetime risks are better tested in-pile in a test reactor, power conversion system risks are better addressed with electrically-heated testing, etc. The need for a GNT depends mostly on three primary factors. The first of these factors is how uncertain the performance of the integrated power system is. A GNT, and perhaps several GNTs are likely needed if a concept has complex and uncertain reactor dynamics; i.e. how the reactor power system will operate (e.g., startup, nominal operating states, transients, failure response, stability, nuclear burnup effects, etc.). Another measure of performance uncertainty is how similar the concept is to others that have successfully operated. The second factor that determines the need of a GNT is how hard it is to mimic coupled nuclear-thermal system performance with electrical testing. Heat-pipe-cooled reactors can often use resistance heaters to represent fuel pins, without disrupting the thermal-structural performance of the rest of the system, whereas high-pressure gas-cooled reactors may have great difficulty allowing penetrations (or remote coupling) for heaters without disrupting flow. The third factor that determines the need of a GNT is whether a fast-acting reactivity control system might be needed, which could only be adequately tested when coupled to a prototypic reactor.

Ultimately, the need for a GNT is a cost-benefit program decision that needs to be considered from the start of the program; i.e., can a concept be designed that meets the criteria in the prior paragraph; e.g., simple and predictable dynamics, electrically-heated system testing, and no need for a fast-acting reactivity control system. If these criteria can be met, then a GNT may not be desirable. If none of the criteria are met, the program might have to plan for numerous design, build, and test cycles to get the system to work adequately. Unfortunately, the more dependent a concept is on a GNT to resolve dynamics and control issues, the harder it will be to get regulatory approval for the GNT in the first place. A regulator will be much more likely to approve a test for a system with predictable and simple reactor performance, than one that is uncertain and complex. Another negative aspect of a GNT is that the test will not occur until near the end of the entire development program, and it is generally unwise to defer the majority of program risk until the end of a project.

Launch safety

As discussed previously, space reactor development is unique to terrestrial reactor development because the consequences of operational reactor accidents are diminished. However, space reactor deployment may become encumbered by potential safety issues that might result from launch accidents. All fission systems that use uranium fuel are minimally radioactive prior to operation. There is negligible radiological risk to the public if there was a launch explosion (or other accident) that dispersed all of the fuel material into the air (the radiation risk to on-site personnel would also be negligible) (McClure, 2021). The key to launch safety is ensuring that the reactor does not accidentally “turn on” (go critical and produce radioactivity) and subsequently provide a harmful radiation dose to anyone. Fission reactors are designed to achieve criticality with intentional/coordinated movements of control elements (to insert reactivity). Mechanisms must be engineered to ensure that the elements do not move inadvertently during any credible scenario; this is standard practice for every type of nuclear reactor, but space reactors have the additional issue of potential high speed impact and/or landing in the ocean. To simplify launch safety, most space reactor concepts are designed to preclude the possibility of inadvertent criticality during launch accidents, including water immersion.

Lack of progress and experience

Another significant challenge to deploying space reactors is not technical; it is the result of the poor track record of prior programs. There has not been a successful launch, or test of a space reactor system in the US since SNAP-10A in 1965 (Voss, 2021). The development of space fission power systems is indeed unique from any other advanced technology, and testing is extremely difficult, but

there is no reason why a simple, useful SFP system cannot be developed and deployed quickly (<5 years) and affordably (<<\$500M). Unfortunately, past programs in the US have gone too far in trying to develop advanced, higher performance reactors, and ultimately have failed.

Today, there is essentially no hands-on experience in the US for designing, building, and testing new reactors. There are a lot of capabilities to run codes and create conceptual designs, but limited ability to translate an idea into an actual engineering system. There is also no infrastructure for testing new reactors, or enough existing test reactors to obtain all of the needed fuel/material data for a novel reactor. Programs currently exist within DOE to address all of these issues, as there have been repeatedly in the past, but until a new reactor is successfully deployed it is unlikely that any significant capability or infrastructure will be maintained. The poor track record of prior space reactor programs is also making it increasingly difficult to convince potential users and funding agencies that a real program can be completed affordably.

Every year, advances in other technologies raise the bar (in terms of performance) for a useful first-generation space reactor. Other space power technologies (e.g., solar cells, energy storage, chemical rockets/cost) have been evolving to increasingly better performance over the past 50 years, while space reactor technology has been stagnant. Fortunately the entry point for fission still allows a rather simple reactor, but the more time that passes without a successful first-step, the more likely it becomes that the vast performance potential of space fission systems may never be realized.

The need for SFP technology evolution

It is clear from the “benefits” section above that SFP is needed to enable ambitious exploration and settlement of outer space. Power levels from kilowatts to megawatts can be provided at low mass and high reliability in almost any environment. It is clear from the “challenges” section that truly enabling SFP systems are very difficult engineer and deploy. Unfortunately, these challenges have continuously become greater over the years, as infrastructure and experience have been lost, nuclear testing more difficult and expensive, and other technologies have continued to evolve to better performance.

The key to breaking the cycle of SFP program failures over the past 50 years may be to start with a simple low power system; a system with perhaps some useful capability, but most importantly low development cost and risk. This was the impetus for the Kilopower reactor program (Poston, 2021d). Previous programs had tended to emphasize enabling performance over development risk, and all of them failed because they did not make sufficient progress within the touted cost and schedule. This issue was recognized decades ago by Admiral Hyman Rickover, in a letter he penned in 1953 concerning “academic versus practical reactors” (Rickover, 1973), and more recently in a white-paper concerning “real versus paper” reactors (Poston, 2020). The ironic consequence of this cycle of failure is that if any previous decision-maker had decided to go the simpler path in the past, we could already be on our 2nd or 3rd generation of reactors, which would probably have even much better performance than what they are shooting for now. Too many decision-makers are concerned about what they consider “dead-end” technologies; i.e., there is a specific aspect of an achievable, real reactor that will not get them to where they ultimately want to go. Another argument that was made in past programs was “if we’re going to invest in a big development program, we’d better get the technology we want.” This is a flawed argument because “big” is a strong function performance requirements. A high-performance system might have two or three times more development cost/risk than one with modestly lower performance. The real dead-end is never making enough progress to learn and build anything. Any technology that leads to a working reactor power system creates forward progress towards any and all highly advanced systems.

Summary

Space fission power is an enabling technology that will be required for humankind’s exploration and expansion into space. Solar and radioisotope power systems are very well suited for many potential uses, but many applications have been identified which will require fission power. The challenges of developing SFP are substantial, and have prevented the use of fission power in space over the past 50 years. This encyclopedia chapter identifies the challenges, and the promising benefits of FSP. The following chapters discuss what has been done previously, and what the options are to eventually unlock the vast potential of fission power in space.

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Space Fission Power: Historical Review

Susan S. Voss, Global Nuclear Network Analysis, LLC, Taos, NM, United States

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Glossary

ACT Auxiliary Cooling and Thaw (loop)
AEC Atomic Energy Commission (prior to DOE)
AI Atomics International
ALMCR Advanced Liquid Metal Cooled Reactor
Bouk Russian meaning Beechtree, also seen as Buk
CDBMB Central Design Bureau of Machine Building, St Petersburg, Russia
DAF Device Assembly Facility, NNSA
DEW Directed Energy Weapon
DNA Defense Nuclear Agency (now DTRA Defense Threat Reduction Agency)
DOD Department of Defense
DOE Department of Energy
DUFF Demonstration Using Flattop Fissions
EM Electromagnetic (pump)
FPS Fission Power Systems
FS-4 Flight System (SNAP-10A)
FSP Fission Surface Power
GE General Electric
GRC Glen Research Center, NASA
HEU Highly Enriched Uranium
Hg Mercury (Rankine Power Conversion System)
JCAE Joint Committee on Atomic Energy (Congressional Committee)
JIMO Jupiter Icy Moons Orbiter
JPL Jet Propulsion Laboratory
K Potassium
KEW Kinetic Energy Weapons
KIAE Kurchatov Institute of Atomic Energy, Moscow, Russia
KRUSTY Kilopower Reactor Using Stirling Technology
kWe Kilowatt-Electric
LANL Los Alamos National Laboratory
LASL Los Alamos Scientific Laboratory
LEO Low Earth Orbit

LEU Low Enriched Uranium <20% Uranium-235
 LOCA Loss of Coolant Accident
 MMW Multimegawatt
 MOA Memorandum of Agreement
 MPRE Medium Power Reactor Experiment
 MWe Megawatt-Electric
 NaK Sodium-Potassium (coolant)
 NASA National Aeronautics and Space Administration
 NAT Nuclear Assembly Test
 Nb-1Zr Niobium-1% Zirconium
 NEP Nuclear Electric Propulsion
 NERVA Nuclear Engine for Rocket Vehicle Application
 NNSS Nevada National Security Site, NNSA
 P&W Pratt and Whitney
 PCS Power Conversion System
 RTG Radioisotope Thermoelectric Generator
 SDI Strategic Defense Initiative
 SDIO Strategic Defense Initiative Organization
 SNAP Systems Nuclear Auxiliary Power
 SNAPSHOT Space System Abbreviated Development Plan for Nuclear Auxiliary Power Orbital Test
 SNL Sandia National Laboratory
 SP-100 Space Power 100 kWe
 SPAR Space Power Advanced Reactor
 SPR Advanced Space Nuclear Program
 SPUR Space Power Unit Reactor
 TDU Technology Demonstration Unit
 TE Thermoelectric (power conversion system)
 TEM Thermoelectric Electromagnetic (pump)
 TFE Thermionic Fuel Element
 TSET Thermionic System Test
 UC Uranium-Carbide (fuel)
 UMo Uranium-Molybdenum (fuel)
 UN Uranium-Nitride (fuel)
 UNM University of New Mexico
 UO₂ Uranium-dioxide (fuel)
 USAF United States Air Force
 USSR Union of Soviet Socialist Republics
 UZrH Uranium-Zirconium-Hydride (fuel)
 We Watt-Electric
 ZrH Zirconium-Hydride

Introduction

The importance of high energy, compact power systems for space was recognized in the early 1950s as the United States (US) began to consider developing satellites for reconnaissance missions. Fission power systems (FPS) were considered a primary candidate to power the early satellite missions. The US space reactor program began in the mid-1950s with the initiation of the Systems Nuclear Auxiliary Power (SNAP) program along with other nuclear development efforts such as naval reactors, Army Nuclear Power Program, Airplane Nuclear Program and nuclear rocket program—Rover/Nuclear Engine for Rocket Vehicle Application (NERVA).

Within the United States there have been multiple programmatic attempts to develop, manufacture, test and demonstrate in space a viable FPS, but the United States has only flown a single demonstration mission in 1965. Conversely, the Union of Soviet Socialist Republics (USSR) developed small compact FPSs that were successfully flown on over 30 reconnaissance missions and two advanced demonstration missions. Russia has since reported new efforts to develop higher power reactor systems (Zak, 2020).

While the US FPS program has not led to a viable space power system, the radioisotope power systems (RPS) has become a mainstay for National Aeronautics and Space Administration (NASA) missions. RPS are highly reliable, long-life low power systems (0.1–0.8 kWe). The perennial question is: why has the RPS program been so successful while after multiple attempts the FPS program has been stalled?

In this paper I will provide a short overview of the FPS programs followed by a technical summary of each program and their specific goals.

SNAP 1955–73

In 1951, the US Air Force (USAF) and Rand Corporation, requested the Atomic Energy Commission (AEC), the predecessor to the US Department of Energy (DOE), to study small reactors that could be used as power sources for early satellite systems (Perry, 1961). Based upon favorable results, the Rand Corporation recommended a small 2 kWe nuclear reactor as the baseline power for their proposed reconnaissance satellite program in their landmark study “Project Feedback” published March 1, 1954 (Lipp, 1954). In 1955 the USAF and AEC issued a formal request for proposals leading to the uranium-zirconium-hydride (UZrH) fueled design by Atomics International (AI). In 1957, the AEC took over control of the SNAP reactor program.

In 1958 through 1959, the USAF formally requested AI produce a SNAP system with thermoelectric power conversion (TE PCS) for a 0.5–3 kWe reconnaissance satellite mission. This led to the development of two systems: the 0.5 kWe SNAP 10A designed with a SNAP 2 reactor core at lower power with sodium-potassium (NaK) coolant, an electromagnetic (EM) pump with TE PCS and the 3 kWe SNAP 2 with a dynamic mercury (Hg) Rankine PCS. The SNAP 2 reactor mockup was completed in October 1957, the experimental reactor test began power operation in 1959 and the developmental reactor test began in 1961 (Jarrett, 1973). In 1960, the AEC and the AF jointly initiated the Space System Abbreviated Development Plan for Nuclear Auxiliary Power Orbital Test (SNAPSHOT) with the plan to complete four flights (JCAE, 1963). In 1963–64, the USAF mission was reoriented resulting in the cancellation of the SNAP 2 reactor development and the proposed SNAPSHOT flight tests. Development of the Hg Rankine PCS was funded through October 1966.

The Congressional Joint Committee on Atomic Energy (JCAE) recommended that the AEC fund the planned SNAP 10A flight as a technology demonstration mission. The SNAP 10A Flight System-3 (FS-3) was launched in April 1965 from Vandenberg AF base, CA and placed into a 500 nautical mile orbit. The system operated successfully for 43 days. The reactor permanently shut down due to what is believed to be a failure in the voltage regulator on the spacecraft (Berg, 1966). The ground test of the SNAP 10A flight system (FS-4) operated successfully for 417 days before the test was completed.

In 1959, NASA had a requirement for a 30 kWe and above space nuclear power system for nuclear electric propulsion (NEP) and interplanetary communication. NASA and the AEC established a joint effort to develop a higher power SNAP 8 FPS using the SNAP 2 technology with NaK coolant and an Hg Rankine PCS capable of providing from 30 to 60 kWe for up to 1 year of operation in space. Two SNAP 8 reactor tests were completed: the experimental reactor (S8ER) and the developmental reactor (S8DR) with integrated flight configuration. The S8DR testing began in June 1968 and shutdown prematurely in December 1969 due to an increase in coolant activity indicating a rupture in the fuel cladding. A large number of the fuel element cladding had cracks but the overall system performed well (AI, 1973).

The SNAP 50/Space Power Unit Reactor (SPUR) concept was developed by Pratt and Whitney (P&W), Canel, CT with a focus on higher power operation in the range of 300–1000 kWe for USAF or NASA NEP missions. The SNAP-50 was designed with uranium-nitride (UN) or uranium carbide (UC) fuel, refractory metal clad, lithium coolant and a potassium (K) Rankine PCS (P&W, 1962). The fuel and nonnuclear components were extensively tested under this program from 1961 through 1965 the component and subcomponent testing had been completed. P&W requested \$500 M for a ground test of a prototype and \$1B for a flight qualified system, and several billion more for a manned space flight in 1965 (AWST, 1965). In 1965, the program was transferred to the Lewis Research Lab (now NASA Glen Research Center) and converted to a long-term technology development program under the Advanced Liquid Metal Cooled Reactor (ALMCR) and Fast Power Reactor Program (Wahlquist and Voss, 1989). The SNAP 50 program was cancelled in 1968 with a total estimated cost of \$99 M 1968 dollars (Carwile, 1973).

Additional design and research efforts were focused on additional reactor technologies including the Advanced ZrH Reactor, Medium Power Reactor Experiment (MPRE), the 710 Gas Cooled Reactor, the Advanced Space Nuclear Power Program (SPR) and Thermionic reactor programs (Dix and Voss, 1984). There was a strong Aerospace Safety program that provided support to the SNAPSHOT program and SNAP 8 system design. Table 1 is a listing of each of the SNAP systems and their operating parameters.

As can be seen in Table 1 the program power goals increased over time from 0.5 to 300 kWe requiring higher temperature operation, more advanced materials, and higher PCS efficiencies. While the SNAP programs were successful in meeting many of the proposed goals it was not selected by USAF or NASA for specific missions leading to the program cancellation in 1973. The decision to develop different technologies and a range of power levels contributed to the programs cancellation. The total estimated costs through 1973 were \$730 M for all FPS technologies (Carwile, 1973) plus additional costs for operating, facility and equipment for a total of \$840 M (Dix and Voss, 1984) in 1973 dollars equivalent to around \$5B in 2020 dollars.

Interim studies and designs 1973–82

From the end of the SNAP program in 1973 through the early 1980s, research continued on space reactor designs and technologies. The government continued to grapple with identifying firm needs and requirements from NASA and USAF to identify clear path

Table 1 SNAP reactor system designs (Jarrett 1973, Staub 1973, Voss 1994, Asquith 1968, SNAP-50 Space Powerplant, 1962).

	<i>SNAP 10A</i>	<i>SNAP 2</i>	<i>SNAP 8</i>	<i>SNAP 50 SPUR</i>
	1959–1965	1959–1966	1959–1969	1962–1965
Power (kWe)	0.5	3–5	35 to 50	300–1000
Thermal power	30	55	600	2500
Conversion efficiency (%)	1.6%	9%	8%	14%
Primary coolant	NaK	NaK	NaK	Li
Reactor outlet temperature (F)	1000	1200	1300	2000
Conversion	GeSi TE	Hg Rankine	Hg Rankine	K Rankine
Lifetime	10,000 h	10,000 h	12,000 h	10,000 h
Agency developing system	AEC ^a	AEC ^a	AEC ^a /NASA	AEC ^a /USAF

^aAEC changed to DOE and split off non-DOE regulatory function to the NRC.

forward for space nuclear power (Dix and Voss, 1984). This led to the initiation of a small technology development program at Los Alamos Scientific Laboratory (now LANL) entitled Space Power Advanced Reactor (SPAR) to investigate a heat pipe reactor with TE PCS capable of producing 10–100 kWe (Buden and Angelo Sr, 1981). The SPAR system was designed to meet Department of Defense (DOD) requirements identified in 1977 including a 7–10 year lifetime in geosynchronous orbit and 3–5 years in low earth orbit with no single point failure. The DOD missions included space-based radar, surveillance, communications, orbital transfer vehicles using NEP and jammers (Boudreau and Buden, 1982).

SP-100, thermionic fuel element (TFE) and MMW 1982–94

In the early 1980s the decision was made to establish a joint DOE, DOD and NASA program to develop a 100 kWe space power system (SP-100). NASA selected the SP-100 for outer planetary missions using NEP and for planetary power. A Memorandum of Agreement (MOU) was established by the three agencies in November 1985 (Gebicke, 1992). DOE had the program lead and the Jet Propulsion Laboratory (JPL) had the project lead. The program was divided into three phases: 1983–85 Phase I technology and safety evaluation; FY1986 Phase II system design, technology development and testing, and ground test preparation, and Phase III a flight experiment (US Department of Energy (USDOE), 1988).

The SP-100 baseline design requirements were to provide 100 kWe for 7 years of full power operation over a 10 year lifetime with high reliability. It was designed to be launched within the space shuttle cargo bay (Manvi and Fujita, 1986). The original specific mass goal was 30 kg/kWe. Technology development goal was to ensure technology for lower power systems in the 10–100 kWe range up to the megawatt range by changing to a higher efficiency converter such as a Rankine system (DeMuth, 2003). A simpler set of requirements for an initial demonstration mission would have helped reduce the overall costs, size and system complexity.

A highly enriched uranium (HEU) UN fueled, refractory metal system with lithium coolant and TE PCS was selected as the primary system design. General Electric (GE) was chosen to be the primary system contractor and Hanford was selected for the nuclear assembly test (NAT). Throughout the program, from 1983 through 1994, a significant amount of work was completed on the system and components including: manufacturing and testing of 90 experimental fuel pins and the fabrication of 55,000 fuel pellets for the planned ground test (Matthews 1993; Matthews et al., 1994), mechanical and environmental testing of the control and safety drives (DeMuth, 2003), Nb-1%Zr and advanced Nb-1%Zr alloy, PWC-11, fabrication and qualification for the fuel clad and primary loop refractory material (Truscello and Rutger, 1992), advancement of thermoelectric conversion technology through design and testing (DeMuth, 2003), a gas separator concept, and design concepts an instrumentation multiplexer (MUX), thermoelectric electromagnetic (TEM) pump for startup and restart.

A thorough nuclear safety program was established for the SP-100 program with initial design guidelines that under accident conditions the reactor would remain subcritical, remain intact, and maintain physical integrity if there was a loss of coolant accident (LOCA) in space. With the demise of the DOD Strategic Defense Initiative program, the LEO mission was eliminated but the reentry shield remained in the baseline design as a precautionary measure to ensure the reactor would reenter intact and remain subcritical upon impact (Timme et al., 1993) and the LOCA loop was incorporated into the Auxiliary Cooling and Thaw (ACT) loop for startup and restart capability (DeMuth, 2003). Engineering mockup's of the reactor core were made at the Zero Power Physics Reactor at Argonne National Laboratory to validate the computer analysis. Fig. 1 illustrates the baseline SP-100 system as deployed in space.

The program was cancelled in the mid-1990s as the estimated costs to complete Phase II ground tests or to alternatively pursue a lower power flight demonstration were considered to be too high. Phase I technology review and down-select, and technology development cost approximately \$51 M in 1992 dollars (Gebicke, 1992). The actual funding for Phase II of \$438 M for 1986–1993 was significantly less than the amount agreed upon within the tri-agency MOA of \$768 M (Gebicke, 1992) resulting in an overall increase in the cost of the program and schedule extension.

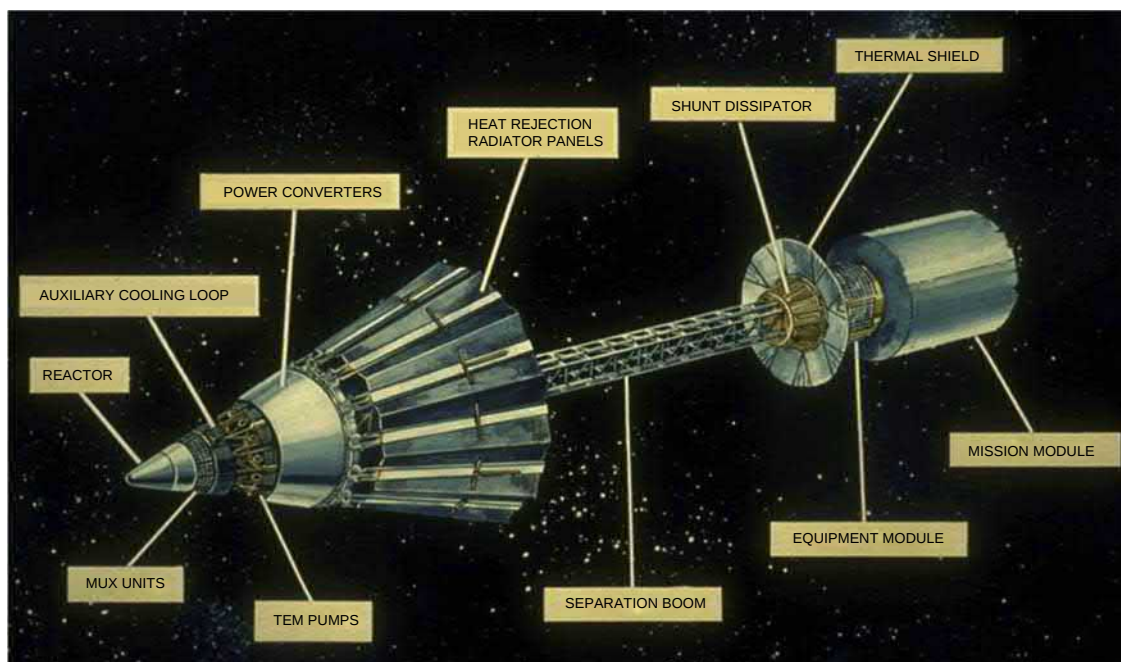


Fig. 1 Baseline SP-100 system view in space.

Thermionic fuel element (TFE) development

In addition to the baseline SP-100 program, there was a joint DOE/DOD thermionics development program to demonstrate the technology readiness of a Thermionic Fuel Element (TFE) that could be used in an in-core reactor capable of producing 0.5–5 MWe at full power for 7 years (General Atomics (GA), 1994). The performance requirements for the TFE's included 9.3% efficiency, 4.1 a/o burnup for a 7 year lifetime using HEU uranium-dioxide (UO_2) (General Atomics (GA), 1994). A baseline design capable of producing 2 MWe was used for the sizing and operational requirements of the TFE's. A number of TFE with different materials and designs were tested in the Training, Research, Isotopes, General Atomics (TRIGA) test reactor providing operational data (General Atomics (GA), 1994). The program was later reoriented to the updated USAF power requirements in the 5–40 kWe range.

A total of \$37 M in 1992 dollars was spent on the TFE program (Gebicke, 1992) with the estimated cost to space qualify the system in 1992 was set at about \$451 M by the design contractor although DOE refuted the low cost estimate (Gebicke, 1992).

SDIO–MMW 1985–90

The need for megawatt-class power systems was recognized in 1984 with the initiation of the SDI program. Proposed SDI applications identified high power needs for directed energy weapons (DEW) and kinetic energy weapons (KEW) (US National Research Council (USNRC), 1989). Three modes of power were identified to support SDI mission needs (US National Research Council (USNRC), 1989):

1. Housekeeping Mode of kWe's, tens of kWe to hundreds of kWe continuous power for up to 10 years,
2. Alert Mode 100 kWe to up to about 10 MWe not easily defined but possibly up to a year or more, and
3. Burst Mode 10s to 100s of MWe and beyond for a few 100–1000 s.

In 1989, the MMW program received six reactor concepts and selected three for further evaluation during Phase II. The program was cancelled in 1990 prior to the start of Phase II, as the SDI program shifted priorities, due in part to the dissolution of the Soviet Union. In addition to the Phase I concept designs there was also a small amount of effort on fuel development (Buden, 1993).

Summary SP-100, TFE and MMW programs

The end of the MMW program in 1990, and the SP-100 and TFE programs in the mid-1990s illustrates the same issues that plagued the UZrH SNAP, SNAP-50/SPUR and thermionic reactors during the 1950s through the early 1970s and that is a lack of a clear mission, a continual change in the mission power level, and a diffusion of funding across multiple technologies and programs. In the end, the lack of focus, consistent funding and increased costs resulted in cancellation of all the programs. A clear US focus on a low power system could help end this pattern.

US/Russian Topaz II space reactor program 1991–95

The Topaz II program formally began in 1991 as a joint US and Russian team under the SDIO leadership with US participation including UNM, SNL, LANL, and the USAF. The program was initiated in 1989 when a team of Russian's attended the January 1989 Space Nuclear Power Conference in Albuquerque, NM and there was interest by a commercial firm to create a cooperate partnership (Nikitin et al., 2000; Dabrowski, 2013). The Topaz II space reactor was developed under the Soviet rule by the Central Design Bureau of Machine Building (CDBMB), St Petersburg and Kurchatov Institute of Atomic Energy (KIAE), Moscow beginning in 1969 (Nikitin et al., 2000). The system was designed to produce 6 kWe for 3 years using single-cell in-core thermionic converters (Voss, 1994). In 1989, there were multiple Topaz II units available for ground and flight testing. The Topaz II system had been extensively tested by the Russians including 12 thermo-physical tests, 4 dynamic tests, 4 static tests, 2 shock and vibration tests, 10 zero power tests and 6 full-up nuclear ground tests, but unlike its Soviet counterpart the Topaz I, it had not been flight tested (Voss and Rodriguez 1994).

The joint program consisted of three main parts: the Thermionic System Test (TSET) of a Topaz system using electrical heaters in place of nuclear fuel (Wold, 1992), the Topaz II flight experiment (Cameron et al., 1994), and thermionic converter test. Russian equipment was shipped to the United States in two separate deliveries: the first arrived in April 1992 with the complete reactor test stand and two ground reactor systems in an American military cargo plane to Albuquerque, NM and the second delivery of 4 Topaz II reactors in March 1994 via a commercial flight (Dabrowski, 2013; Nikitin et al., 2000). The United States had planned to purchase the already fabricated UO_2 fuel from the Russians for the flight demonstration. Estimated costs to complete the flight demonstration, excluding the cost of the launch vehicle was \$150M: \$80M for the satellite and \$70M for the power system (Gallup, 1992).

In all aspects of the Topaz II program, the United States and Russian participants worked closely together to achieve the technical goals. It was an unprecedented cooperative program that faced many cultural differences and yet was able to achieve many successes in testing and flight preparations. In October 1995 the program was transferred to the Defense Nuclear Agency (DNA), (now DTRA) as the SDIO program began to close down and the US government sold the Topaz II reactors back to the US contractor and returned to the Russian nuclear institutes as specified in the original contract. The program was terminated on March 19, 1997 and total program costs exceeded \$100M in 1997 dollars (US General Accounting Office (USGAO), 1997).

USSR space reactor program 1960–present

The USSR began development of FPS in the 1960s with the Bouk (Beechtree) FPS (Kukharkin, 2012). The Soviet Union had been using space nuclear power in space since 1970 with 32 launches of the Bouk 2.5 kWe reactor into LEU from 1970 through 1988 and two launches in 1987 of the more advance Topaz I thermionic system capable of producing 5–7 kWe for up to a year (Zakirov, 2011; Bennett, 1989). The Bouk and Topaz I were developed by the Red Star Institute and the Central Design Bureau for Machine Building (CDBMB). The reentry of Cosmos 954, a Bouk reactor, on January 24, 1978 resulted in the spread of radioactive material over a broad swath of Canadian territory resulting in a unique international legal situation between countries (Galloway, 1979). As a result of the Cosmos 954 accident, the Russian's spent several years redesigning the system to ensure the reactor would be boosted to a higher orbit at the end of life or eject the core upon uncontrolled reentry to ensure fuel burnup and dispersal in the upper atmosphere. Unfortunately the reactor systems launched after the redesign ejected their cores in higher orbit resulting in a significant increase in space debris.

Currently the Russian's are working on advanced FPS programs including a 150 kWe and 1.5 MWe Brayton system (Zak, 2020, Gabaraev et al., 2007).

Jupiter icy moons orbiter (JIMO)/Prometheus 2003–05

The Jupiter Icy Moons Orbiter (JIMO)/Prometheus project was a NASA program initiated in March 2003 and ended in October 2005. The initial program goals were to develop a nuclear power system capable of producing 200 kWe for 15–20 years in space with a proposed flight test in 2015 (Lehman et al., 2006). To meet the rather aggressive schedule the team chose a relatively low temperature reactor technology to meet the lifetime goals with high reliability. The program was initially managed by DOE but was transferred in March 2004 to Naval Reactors prime contractors (Ashcroft and Eshelman, 2006). The JIMO/Prometheus program initiated a comprehensive technology review and by the program end they had selected a conceptual baseline design based on a high temperature gas-cooled reactor with either UO_2 or UN fuel directly coupled to four Brayton engines with helium/xenon coolant (Ashcroft and Eshelman, 2006). An artist rendition of the lithium metal cooled reactor subsystem considered under the Prometheus project is shown in Fig. 2.

As the program continued, the costs estimates to meet the program goals increased significantly and lower power and shorter missions were considered. The program costs from 2003 through 2005 were \$463M in 2005 dollars (NASA, 2005) and the total estimated costs for a flight qualified systems was \$4.165B in 2005. Much of the high costs were driven by the long-life requirement, and the desire to complete a multi-year nuclear-powered test.

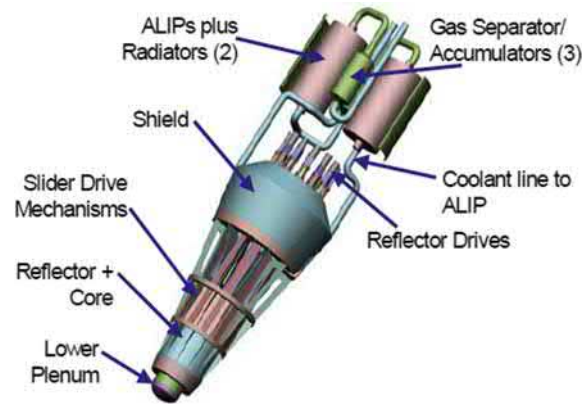


Fig. 2 Liquid metal cooled reactor subsystem design considered in the JIMO/Prometheus program (National Aeronautics and Space Administration (NASA), 2005).

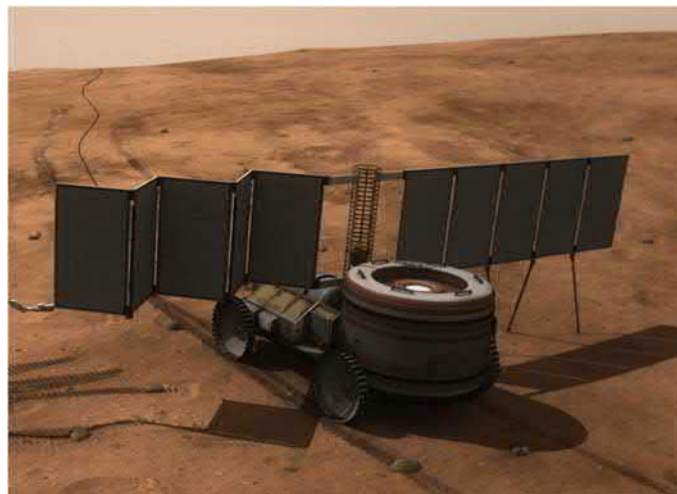


Fig. 3 Mars FSP concept (Mason and Poston, 2010).

Fission surface power (FSP) 2006–07

The Affordable Fission Surface Power (FSP) was initiated by NASA in partnership with DOE in 2006 as the Prometheus program and JIMO mission was phased out (Mason and Poston, 2010). The project focused on developing FSP concepts operating from 10 to 100 kWe to meet lunar surface power needs, although the same technology could be used for Mars. The FSP was to produce 40 kWe with a low-risk approach to maintain costs to a reasonable level (Mason and Poston, 2010) by opting for “low risk, operational robustness, and affordability” (Mason et al., 2013). The reference design was a NaK cooled, UO_2 fueled fast spectrum reactor with Stirling power conversion and titanium-water heat pipe radiator (Mason et al., 2008; Mason and Poston, 2010). System operating temperature was 900 K (Mason et al., 2013). The system as it would be deployed on Mars is shown in Fig. 3.

A Technology Demonstration Unit (TDU), a nonnuclear system test with an external heat source simulating the reactor subsystem was tested in the 2014–15 timeframe in a vacuum chamber at NASA’s GRC. Component and subsystem tests were conducted prior to the TDU including the annular linear induction pumps, core simulator, and subscale and full scale Stirling engines (Briggs et al., 2016). The system simulated a single loop of the proposed 40 kWe FSP design with a primary pumped loop NaK loop transferring power from the reactor simulator to a single 12 kWe Stirling power converter and rejected heat to two pumped water loops (Briggs et al., 2016). The TDU as built produced 9.6 kWe at 18.4% efficient (Briggs et al., 2016); most of the system performed well, although the pump efficiency was very poor and was operated on facility power.

One of the impressive aspects of the FSP program, was their emphasis on early hardware testing and results. If continued, the program could have gone to an early flight demonstration and operation on the Mars surface. The 2010 FSP study estimated that the costs to develop, flight-qualify, and deliver a 40 kWe system to the lunar surface by 2020 would cost approximately \$1.4B with follow-on unit costs of \$215 M in 2010 dollars (Mason and Poston, 2010). If the FSP program had continued, it would provide a step-wise advancement of space nuclear technology from a low temperature system to a higher temperature superalloy or refractory metal system with UN fuel and dynamic power conversion thereby providing power in the 100 to 1 MWe and above range.

Kilopower 2013–18

In the early 2010s, the NASA Space Missions Directorate identified the need for a low mass 1–10 kWe space nuclear power system with a 10 year lifetime that would bridge the gap between RTGs and the FSP to enable deep space missions ([Mason et al., 2013](#)). A series of designs were evaluated leading to the down select of a solid cast uranium-molybdenum (UMo) fuel, with heat pipes on the outside of the core for lower power levels that directly transferred the heat to thermoelectric or Stirling engine power conversion system. The excess heat was transferred via water cooled heat pipes to the radiator system. The reactor was designed with HEU to maintain a small core and shield necessary for deep space missions. An artist rendition of a Kilopower system on the lunar surface is shown in [Fig. 4](#).

There were two primary reactor demonstration tests at the Nevada National Security Site (NNSS) Device Assembly Facility (DAF) facility in the Kilopower program: Demonstration Using Flattop Fissions (DUFF) test in 2012 ([Poston et al., 2013](#)) and Kilopower Reactor Using Stirling Technology (KRUSTY) 1 kWe demonstration ([McClure et al., 2020](#)). In 2016, a depleted uranium system was tested at GRC as part of the DUFF test series to develop the in-core heat pipe bonding and test the operation with electric heaters before completing the nuclear test at DAF in 2017. The KRUSTY test was done with a prototypical nuclear core and external heat pipes integrated to a Stirling system to demonstrate the system performance, load following capability and safety response. The KRUSTY system was designed, built and tested for \$18M in 2017 dollars, a fraction of the costs of other programs by choosing existing facilities and technologies, and maintaining a small focused team. After the completion of the KRUSTY test, the program goal was to design the flight unit and complete the first flight for an estimated \$150–\$250 M ([Palac, 2015](#)).

Later mission studies demonstrated that a 10 kWe Kilopower system could be used to provide power for the initial Mars mission, and a Kilopower systems was more advantageous than an FSP for a number of technical reasons ([Rucker, 2015](#)). Therefore, while Kilopower provided a low cost system for a 10 year lifetime at 1–10 kWe, it was not selected by NASA for continued development after the 2018 testing was completed.

Follow-on space nuclear: Fission surface power and megawatt NEP 2019–present

In 2019, the decision was made to transfer primary responsibility from NASA to DOE for a 10 kWe Fission Surface Power (FSP) system for lunar operations by 2027 with extensible operation on the Martian surface ([DOE/NE 2020](#)). System designs using high assay low enriched uranium (HALEU) rather than highly enriched uranium (HEU) are being evaluated as directed by Congress. The project is anticipated to kick off a design effort in early 2021. In addition, NASA and DOE are evaluating a high power reactor 1 MWe and above as part of a Nuclear Electric Propulsion (NEP) system.

Conclusions US space reactor program

The US space reactor program for power and NEP from 1960 to the present resembles more of a random walk rather than a clear and established technology program. Projects have been initiated to achieve power levels of hundreds of watts, kilowatts, tens of

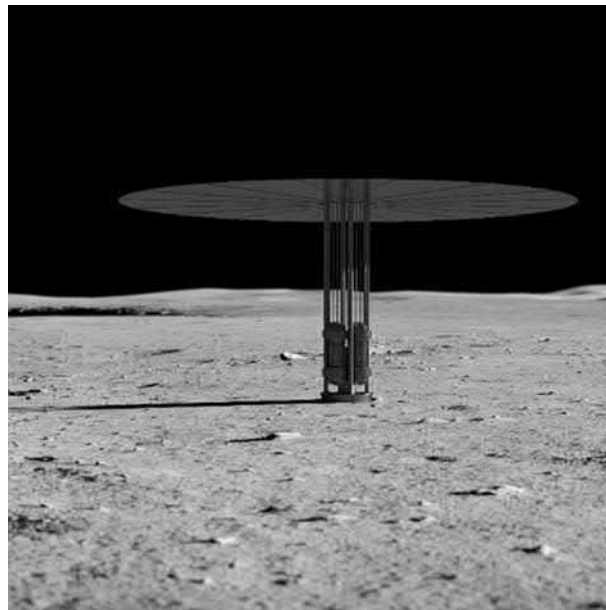


Fig. 4 Artist's concept of Kilopower fission power system on the lunar surface ([National Aeronautics and Space Administration \(NASA\) News Release, 2018](#)).

kilowatts, hundreds of kilowatts and up to several mega-watts yet not a single program has moved forward into a viable space power alternative. Past studies and testing have included low and high temperature systems, moderated and unmoderated reactors, UO_2 to UN, stainless steel to refractory metals, static to dynamic power conversion, in-core and out of core power conversion, and many other variables. The biggest issue has been just selecting a simple first-generation technology and moving forward. As demonstrated in the Kilopower program, a low power 1 kWe system is approximately 10 times higher power than the current RTG for space power and would provide spacecraft designers a higher power option for deep space missions. Yet rather than building upon successes, the US program continues to stop and restart on a continual basis. Progress in this critical field will not occur until a step-wise approach is embraced similar to the RTG program.

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Space Fission Power: Technology Options – Reactor Core

David I. Poston, Space Nuclear Power Corporation, Los Alamos, NM, United States

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Abbreviations

Cermet Ceramic-Metallic
CTE Coefficient of Thermal Expansion
HALEU High-Assay Low Enriched Uranium (typically 19.75% ^{235}U)
HEU Highly Enriched Uranium ($> 20\%$ ^{235}U , typically 93% ^{235}U)
LEU Low Enriched Uranium ($< 20\%$ ^{235}U)
mT Metric Ton (1000 kg)
RTC Reactivity Temperature Coefficient
SFP Space Fission Power
Slider Sliding Neutron Reflector
SS Stainless-Steel

Introduction

Space Fission Power (SFP) systems can enable ambitious exploration and human expansion throughout the solar system and beyond. Previous chapters in this encyclopedia discuss benefits and challenges of SFP (Poston, 2021a), and the history of past SFP programs (Voss, 2021). This chapter discusses several of the reactor technology options that could be used to overcome the challenges of space reactors. In this encyclopedia, Space Fission Power (SFP) systems, or space reactors, use the following definition:

SFP System (a.k.a. space reactor) = fission power system that can be launched from Earth.

The reason for this definition is to preclude discussion of future power plants that might operate in space, after industrial scale in-situ resource utilization and fabrication are established, or complex assembly capabilities might exist in Earth orbit. A space reactor is considered a mostly completed power system than must be light/small enough to be launched into space, and in some cases

transported and landed as well. As space architectures evolve the ability to use in-situ materials for shielding, heat rejection, or other components will gradually broaden this definition, but eventually the system will look more like a traditional power plant.

SFP Plant = fission system that requires significant construction/assembly after launch.

The practical upper power limit of a space reactor might be ~ 10 MWe; assuming a launchable mass limit of 100 metric tons (mT) and an optimal (near-term) power system specific mass of 10 kg/kWe. These are certainly not strict limits, rather they are round numbers meant to approximate practical limits for at least the next several decades. Anything of greater power is difficult to speculate about, because it will depend on what path that advanced rockets and space reactors follow to evolve to greater capability.

Technology options for space reactors

There are a wide range of technologies that can be considered for Space Fission Power (SFP) systems. The following section attempts to describe technology options independently of mission or application, which is not ideal because the selection of one technology often leads to a preference of another, and vice-versa. However, the alternative approach, i.e. describing classes of reactors and the combined suite of technologies that might best work for them, may be worse because black-and-white boundaries cannot be defined to include one technology and exclude another. The selection of reactor coolant and cooling technology is equally important to space reactor design, and is discussed in another chapter in this encyclopedia ([Poston, 2021b](#)).

Nuclear fuels

Choice of fissile material

The space reactor design process generally starts with the selection of fissile material; however, in almost every case this selection requires almost no thought; i.e. the best fissile material to use is ^{235}U . Uranium-235 is the fissile material loaded into almost every power plant and test reactor operating today. It is still relatively abundant and it is only slightly radioactive, which makes it economical for all reactor applications. Plutonium-239 is a possible fissile isotope that could significantly reduce mass of some systems, but would be more expensive to process, fabricate, transport, and assemble in a space reactor, create unique core chemistry and thermal concerns, and create significant operational safety and launch safety concerns. Uranium-233 can also reduce mass; it also has radiation concerns (from ^{232}U contamination) and the supply is very limited, although it might be easier to utilize in a space reactor than ^{239}Pu . Another issue with both ^{239}Pu or ^{233}U is a small delayed neutron fraction, less than half the value of ^{235}U . A smaller delayed neutron fraction introduces both operational and safety concerns. Americium-242 is another isotope that can offer low mass reactor cores, but is probably the less practical than ^{239}Pu or ^{233}U . Note that speculation about “better” fissile isotopes is only relevant for low power systems that could significantly benefit from lower critical mass. Lastly, thorium-based cores (i.e. ^{233}U breeder reactors) would be too heavy to be practical for SFP system, but could make sense for future SFP plants, especially if sufficient Th is found on a planet, moon or asteroid.

Choice of uranium enrichment

^{235}U is the obvious fissile material choice for early generation space reactors, which then leads to the question of enrichment. From a performance perspective the choice is very simple; highly enriched uranium (HEU), with enrichments $> 90\%$, will offer lower mass in almost every application. First order, the advantage of using HEU comes down to how “criticality limited” the “simplest” reactor option is, versus being limited by other design issues like heat transfer, swelling, strain, complexity, irradiation damage, mass transfer, etc. A criticality limited reactor generally has much more fuel than it otherwise would need to complete its mission, other than it requires “excessive” fuel to sustain a fission chain reaction (i.e. go “critical”). The distinction of “simplest” above is important because almost any reactor can move away from being criticality limited by adding enough moderation ([Poston, 2020](#)).

The negatives of HEU are almost all associated with proliferation concerns ([McClure, 2021](#); [Lal and Locke, 2020](#)); i.e. potential use of HEU for nuclear weapons. The US has regulations in place that allow the usage of HEU. These regulations restrict operations and require adequate security in order to prevent proliferation. This results in higher costs for certain aspects of development and deployment of HEU systems vs low enriched (LEU) systems. The definition of LEU within the regulatory framework is 20%, such that any fuel $< 20\%$ ^{235}U is LEU and any fuel $> 20\%$ is HEU. Clearly there is a huge difference in the attractiveness of 21% enrichment (enr.) and 93% enr., but they are effectively fall under the same requirements for protection. Therefore, most space reactors are designed to either 93% enr. (existing weapons grade material) or 19.75% enr. The specification of 19.75% is intended to give enough margin that no batch of processed fuel will end up at $> 20\%$ and thus violate the regulations. This enrichment level of 19.75% is commonly referred to as High Assay Low Enriched Uranium (HALEU), to differentiate it from the 4 to 5% enr. LEU used in commercial power plants (note that commercial power plants are not criticality limited – which allows for the low enrichment).

Ultimately, the performance benefit of HEU must be compared to disadvantages. In addition to the higher cost of operations, another disadvantage of HEU could be political. Even if the laws allow HEU to be used when properly protected, there might be resistance to using HEU depending on the policy stance of the government. The use of HEU might be considered a bad precedent

because other countries might try to establish a weapons program under the guise of establishing a space reactor program. The final decision to approve the use of HEU for launch in the US must come from the Office of the President, which adds programmatic risk. Another disadvantage of HEU could be availability and/or cost of enrichment. Currently this is not a factor, because there is no significant availability of HALEU, nor the ability to enrich it (it is actually created by downblending HEU). Commercialization of space reactors will probably be more difficult using HEU than LEU, and it is hoped that commercial production of HALEU will be available in the relatively near future.

Overall, it is hard to make a blanket statement about the pros and cons of HEU. The mass and/or simplicity benefit can be very substantial for systems that can enable a paradigm shift in space exploration, while for other applications the benefit of HEU will not outweigh the advantages of HALEU. The final decision will have to be made by the end-user or customer, based on their own cost/benefit assessment.

Uranium metal fuels

Metal based fuel, typically uranium alloyed with molybdenum or zirconium, can be very attractive for space reactor applications. The majority of reactors flown in space have used metal fuel (i.e. the Buk reactors of the USSR), and metal fuel has also been used in many terrestrial reactors. Metal fuels have many strong advantages over other fuels: ease of fabrication, high uranium density, and high thermal conductivity, but they do not perform well above 800–900 °C (depending on structural requirements) or with high nuclear burnup (> 1 a/o that is, fissioning of > 1% of initially loaded uranium atoms).

Metal fuels can be used to higher burnups, in fact they are intended to go to very high burnups ($\gg 10$ a/o) in advanced terrestrial reactors. However, the difficulty of ensuring fuel-pin lifetime would require significant testing. In addition, the high level of swelling with burnup reduces the uranium density to a point where it starts to lose its neutronic/mass advantage over other fuels, and the control system is required to accommodate the large reactivity swing caused by burnup.

<i>Metal fuel advantages for SFP</i>	<i>Metal fuel disadvantages for SFP</i>
- Very high uranium density - Very good thermal conductivity - Good testing history and database - Easy to manufacture with existing fabrication infrastructure - Good compatibility with liquid metals	- Relatively low peak temperature - Very high fuel swelling potential - Very high fission gas release potential - Poor chemical compatibility with steels and nickel-based alloys - Solid form phase-change potential

Metal-fuels (UMo or UZr) are almost always the best choice for space reactors that do not require fuel temperatures over ~ 900 °C, and are low enough power and lifetime to keep fuel burnup < 1 a/o.

Uranium oxide

UO₂ is probably the most robust and well understood fuel option for any nuclear reactor, as it is used in most commercial power plants today. Uranium oxide is often a euphemism for uranium dioxide (UO₂), although other oxide forms do exist (U₂O₅, UO₃, U₃O₈, UO₄). The sintering process to fabricate high density UO₂ pellets is well understood, and well as its neutronic and chemical behavior.

<i>UO₂ advantages for SFP</i>	<i>UO₂ disadvantages for SFP</i>
- Excellent testing history and database - Existing manufacturing infrastructure - Very high temperature capability - Relatively low swelling and fission gas release - Good stability and chemical compatibility with most structural materials	- Relatively low uranium density - Very low thermal conductivity - Poor compatibility with liquid metals - Brittle, low thermal shock resistance

UO₂ is the lowest-risk option for high power, high temperature, and/or long lifetime space reactors, and is often the best choice for any reactor that is not strongly criticality limited.

Uranium nitride

UN is a fuel that is perfectly suited for most space reactor applications, especially because of high uranium density and high thermal conductivity. Unfortunately, there is not a lot of existing infrastructure and data that pertains to UN; largely because UN has poor compatibility with water (and thus the vast majority of reactor experience). Uranium nitride was the baseline fuel for the SP-100

program, and ~200 kg of high-density pellets were successfully fabricated, via cold-press and sinter (Matthews et al., 1998). Irradiation data is limited, but this data plus corresponding analysis indicate relatively low swelling and fission gas release up to 6 a/o burnup. The melting point of UN is very high, but decomposition into free uranium and nitrogen gas has been shown to become significant between 1600 and 2000 K. Close correlations between performance (e.g., low swelling and fission gas release) and chemical impurities in UN requires careful fabrication pathways, adding to cost and complexity to beyond bench-scale facilities.

<i>UN advantages</i>	<i>UN disadvantages</i>
- High uranium density - Good thermal conductivity - Relatively high temperature capability - Relatively good chemical compatibility with most SFP structural materials and coolants. - Low fuel swelling and fission gas release	- Limited test and operation experience - Relatively difficult to manufacture - Limited fabrication capability - Need for nitrogen overpressure at high temperatures.

UN generally offers the best performance for reactors that require high fuel temperature ($\sim 900\text{ }^{\circ}\text{C} < T_{\text{fuel}} < \sim 1300\text{ }^{\circ}\text{C}$) and/or require relatively high burnup ($\sim 1\text{ a/o} < \text{burnup} < \sim 6\text{ a/o}$), and is the best choice for these reactors if a program is willing to take a slight risk (only risk until UN is successfully utilized).

Uranium carbide

Uranium carbide, UC or UC₂, were investigated heavily in the 60s and 70s, primarily as a fast reactor fuel, but did not find an attractive space reactor niche. The biggest problem with UC is swelling, and potential for runaway chemical reactions or swelling under certain conditions, while almost all advantages of UC are also realized by the better performing UN.

<i>Uranium carbide advantages</i>	<i>Uranium carbide disadvantages</i>
- Good uranium density - Good thermal conductivity - Good compatibility with liquid metals - High temperature capability	- Very high fuel swelling potential - Potential runaway reactions - Limited existing manufacture capability and experience

Uranium carbide does currently not appear to have a space reactor niche, unless a major fabrication and testing effort can establish that it is cost effective and can perform robustly.

Uranium-zirconium hydride

UZrH was the fuel for the only space reactor ever launched by the US – SNAP 10-A. The advantages of UZrH are due to the moderate effect of the hydrogen, which allows the potential for low mass reactors. UZrH is also a well understood fuel with a lot of experience and infrastructure (primarily in TRIGA type research reactors, Reese, 2021). It is usually not an attractive option for space reactors because relatively low temperature is required to prevent hydrogen loss, and it has rather complex swelling behavior.

<i>UZrH advantages</i>	<i>UZrH disadvantages</i>
- Excellent neutronics for low mass - Good testing history and database - Existing manufacturing infrastructure - High reactivity feedback mitigates accidental reactivity insertion scenarios.	- Low peak temperature - Limited lifetime (due to issues below) - Hydrogen loss can reduce reactivity - Fuel swelling rate can be high and is very sensitive to temperature

UZrH may have a niche for space reactors, but only applications with very short lifetimes ($< \sim 2$ years) and operate at relatively low fuel temperatures ($< \sim 750\text{ }^{\circ}\text{C}$).

Composite fuels

Composite fuels, or fuels that are mixed/integrated with other materials, represent a wide range of possible fuel technologies. Particle-based composites disperse small kernels of fuel within in a matrix of non-fuel materials. High temperature composites could use almost any fuel form (UO₂, UN, UC, UCO, etc.) and be dispersed into a wide variety of matrix materials (graphite,

SiC, BeO, etc.). The fuel particles are usually clad/coated with layers of graphite, ZrC, SiC, etc. The number and thickness of layers depends on the intended application. In traditional TRISO fuel (Helmreich, 2021), a SiC layer prevents fission gas release into the matrix, internal graphite layers isolate the SiC from the fuel, and an outer graphite layer protects the SiC during handling and fabrication. Composite fuels are not necessarily particle based. Some composite fuels disperse the fuel material in vein-like structures, while others attempt to create a uniform dispersion of fuel material, in what is referred to as a solid-solution.

<i>Composite fuel advantages</i>	<i>Composite fuel disadvantages</i>
- Very high temperatures - Very high burnup - High fission gas retention	- Low uranium density - Difficult to fabricate - Generally limited data and experience

Composite fuels are generally unfavorable for space reactors considered in this chapter (e.g. < 10 MWe). The low uranium density requires high mass or the complexity of heavy moderation. In addition, the benefit of fission gas retention is not as important for space reactors as for reactors on Earth.

Cermet fuels

If a ceramic fuel (UO₂, UN, etc.) is placed in a metal matrix it is referred to as cermet fuel (although it is still technically a composite fuel). In a particle-based cermet fuel, the fuel kernels do not necessarily use any coating layers, which allows the fuel loading to be higher than TRISO-type fuels. Cermet typically implies that a refractory metal like Mo or W is used as the matrix, but a stainless steel (SS) matrix could be included in the cermet family. If W is used as the matrix, then the reactor design will usually require HEU to have acceptable mass, whereas the mass can be acceptable with HALEU with an SS matrix, or perhaps Mo. The primary advantage of cermets is a high temperature limit and the ability to uniformly distribute heat due to high conductivity. The disadvantages are generally high mass and difficult fabrication.

<i>Cermet fuel advantages</i>	<i>Composite fuel disadvantages</i>
- Potential for very high temperatures - Potential for high power density core	- Relatively low uranium density - Difficult to fabricate - Limited data and experience

Cermet fuels are also generally unfavorable for the definition of space reactor used in this chapter (e.g. < 10 MWe). Their potential benefit is for very advanced high-temperature, high power reactors, which hopefully will be utilized several space reactor generations into the future.

Aqueous/salt fuels

It is very unlikely that a molten salt or aqueous fuel form will even be attractive for a space reactor, at least until a planet or Moon has very high power demands and substantial infrastructure. These types of reactors tend to be large in size and may require an online chemistry control system, plus they are gravity dependent. Once a large colony is established, then this type of reactor could have an advantage because of the ability to get very deep burns of nuclear fuel (i.e. extract more energy from limited fissile resources).

Cladding/structural materials

The choice of structural material is as important as the fuel selection, and in some ways more important. In this chapter, “structural” is meant to include fuel cladding, coolant boundaries, and core support. Cladding is the structural material that specifically contains the fuel. Coolant boundaries might be a pressure vessel, flow baffles, plenum, or heat pipe wall. Core support provides the internal structure to hold the entire core together.

In many reactors, the cladding, coolant boundaries, and core support are all the same materials; this is generally preferable, provided that the same material can provide all of the necessary properties. Other reactors can benefit by the use more than one structural material type, as long as they are not too dissimilar. Dissimilar materials often present problems due to concerns with mass transfer, differential thermal expansion, uneven wear, and fabrication concerns (e.g. the inability to weld, or perhaps 3D print). Most importantly, the fuel and cladding cannot generally be selected independently – the two must be chosen as what is referred to as a “fuel system.”

Like all space reactor choices, the selection of structural materials is made to provide the best performance (power, mass, reliability, surety) at the lowest practical cost and development risk. This decision is driven by a combination of factors, most of which can be included in one of the categories listed below.

Mechanical properties: A structural material must have adequate strength to prevent excessive strain or failure when presented with the anticipated primary and secondary stresses. Any particular reactor application might be limited by a combination of yield, ultimate, and/or creep strength. Another important mechanical property in most reactors is ductility, because the large changes in temperature generally create large thermal stresses that need to be relieved by thermal strain. Brittleness can be an important consideration for launch and landing loads. In some concepts, fatigue resistance can be important if vibrations are significant. Other concepts might prefer hardness and toughness to prevent erosion, although these parameters can make the material hard to fabricate. Ultimately, all mechanical properties must be accounted for (with margin) to create a practical engineering design. The mechanical properties at elevated temperatures are often the most important discriminator in terms of performance, because of how core temperature impacts system efficiency and mass.

Thermophysical properties: The most important thermophysical property of structural material is the coefficient of thermal expansion (CTE). The CTE usually plays a direct role in reactivity feedback (the change in reactor neutron multiplication with temperature), especially in compact, fast spectrum reactors. In many cases it can be the dominant reactivity temperature coefficient (RTC); e.g. in reactors where the structural material holds fuel pins/rods in place, either via a grid plate or a core monolithic-block. The structural expansion decreases reactivity by allowing more neutron leakage. It is generally favorable for the CTE to dominate reactivity feedback, because it is relatively constant with temperature, has less uncertainty than most cross section temperature effects, and can be accurately measured with repeated non-nuclear testing. The relation of the structural CTE to the fuel CTE is also extremely important for thermo-mechanical performance, because it affects gap sizes and can cause substantial stresses if mismatched. The importance of thermal conductivity depends on the specific reactor; high conductivity is very beneficial in monolithic-block reactors, but far less important in pin-type reactors. Specific heat can be important for transient behavior, but until a detailed design is completed it is hard to predict if a higher or lower specific heat is desirable.

Materials compatibility: The structural material must be compatible with the fuel, as mentioned previously with reference to the fuel system. At the same time the material must be compatible with the coolant, and in some cases the structure must create a mass transfer barrier between the fuel and coolant. One potential mitigating possibility is the use of a coating or liner. The coating needs to be compatible with both of the materials in question, provide a mass transfer barrier, and allow the reliable deposition/place-ment of the coating/liner. The use of coatings is becoming an increasingly attractive option as our ability to reliably apply thin coatings of materials improves. The problem with coatings is the ability to ensure long-life, and thermal-bonding if required, which can require length testing and potentially nuclear testing (some coatings can be damaged by modest levels of neutron fluence). For surface reactor applications the structural material also needs to be compatible with the environment; likely not a problem on the Moon, probably manageable on Mars, but a huge issue on the surface of Venus.

Neutronic properties: The ability for a reactor to go critical (sustain a fission chain-reaction) is the most fundamental requirement of any reactor. The neutron absorption cross sections (probabilities), and to a smaller extent scatter cross sections of the structure are a major consideration, especially in a moderated reactor. This gives unmoderated reactors an advantage because they have a larger suite of potential structural materials available. If possible, the ideal neutronic material would absorb minimal neutrons during nominal operation, and excess neutrons during off-nominal conditions (e.g. elevated temperatures, flooded accidents, etc.). The importance of neutron absorption goes well beyond the magnitude of the cross sections. The temperature dependence of the cross sections, and their effect and/or dependence on neutron energy spectrum plays a major role in the stability and controllability of the reactor.

Irradiation tolerance: Irradiation (exposure to reactor radiation) can affect both mechanical and thermophysical properties. In most cases damage is caused by neutrons. Fast, or high energy neutrons can damage the lattice structure and/or grain boundaries of a material and significantly change properties, often causing a slight increase in strength but a large decrease in ductility. Irradiation effects are often tied to the neutronic properties, e.g. a material for which neutron absorption can create an alpha particle (helium) can result in swelling and/or helium embrittlement. The level of irradiation damage is often a strong function of material temperature. In some cases high temperature exacerbates issues (e.g. swelling) while in other cases high temperature can mitigate issues (e.g. retention of ductility via annealing). The susceptibility to irradiation damage can be also strong function of how the material was fabricated and processed; i.e. how they affect lattice structure (phase), grain sizes, voids, defects, etc.

Material performance database: Most available materials have an existing database for mechanical and physical properties, but there are two aspects of space reactors that make this issue very problematic. The first issue that poses a problem is temperature. High temperature is needed for space reactors to achieve high power conversion efficiency (due to the difficulty of rejecting heat in space), more so than for most other existing engineering applications. Space reactors can also potentially experience very cold temperatures, well-below what most engineered system would experience. As a result, there is often limited or no performance data for candidate materials in the desired temperature regime (< 250 K and/or $\gg 1000$ K). The even more problematic issue is the irradiation tolerance of a candidate material. It is extremely hard to predict a specific material's sensitivity to irradiation. In addition, many materials have shown unique irradiation sensitivities which could never have been predicted without testing. A lack of required irradiation data for any specific material can present substantial cost and development risk for a concept. The lack of test reactor availability also makes this issue a major schedule driver. Until more test reactor capability becomes available, it is unlikely that a material that requires irradiation data would be selected for a near-term space reactor application. Either a material with

existing irradiation data will be used, or the neutron flux/fluence will have to be so low that the majority of materials experts would concur that it poses little damage potential.

Fabricability and infrastructure: The ability to physically produce the reactor is a non-negotiable requirement. The selection of structural material often comes down to the ability to successfully fabricate and assemble the core within acceptable time, cost, and risk. This generally gives common materials like stainless-steel (SS) an advantage over more advanced or novel structural materials. Just as important as the ability to make parts, is the ability to join them; a strong material is generally useless if it can't be welded or fabricated in a way to provide a hermetic seal. The emergence of additive manufacturing (3D printing) is a potential game-changer in this area (Betzler, 2021), because it could allow the manufacturing of more complicated and robust core components in the future, potentially even with refractory metals or even reactor fuel. The other consideration is the available infrastructure to reliability and consistently produce components, not just for the completed reactor, but very early on for the prototyping and testing that will be required. The level of existing infrastructure can have a major impact on cost and schedule of developing and delivering a reactor.

Mass density: For a space reactor, density is clearly one of the most important physical properties (in order to allow low mass), but it is listed last because it is rarely the basis for structural material selection. The selection is based on what material can best provide all of the above (strength, good thermal-neutronic characteristics, materials compatibility, infrastructure, etc.) at the lowest density. The density of the structural materials usually is not a major discriminator in system mass, as compared to the allowable temperatures, stresses, and surety that the material can provide. Only if the designer is lucky enough to have multiple candidates that equally meet all requirements, will density influence the selection.

Candidate structural materials

There are many possible structural materials. Traditionally, metals have been the material of choice for reactors (for many reasons), but advances in manufacturing of carbon-based composites and ceramics like SiC are making them increasingly attractive. All of the desired attributes listed above can quickly limit the list of structural candidates to only a handful of material types. Zirconium-based cladding is by far the most common structural material used in nuclear reactors, because it absorbs less neutrons than SS at thermal neutron energies. The neutronic advantage of Zr is rather small in most space reactors (because the neutron spectrum is less-thermalized), such that SS is preferred due to its higher ductility, ease of fabrication, and lower chemical activity at high temperatures (not an issue in space, but better for testing and potential use on Mars). The material choice for space reactors thus usually starts with SS, and then considers more advanced materials when only a higher temperature capability can meet requirements.

The table below gives a high-level, very-generalized view of structural metals from the design perspective of a space reactor. This table should only be used to perhaps decide what materials to further investigate for a trade study, because (1) the ratings have many caveats that would change for one concept versus another (e.g. with respect to fuel compatibility/experience), (2) advances in materials technology and testing might change the ratings (especially for SiC), (3) the ratings were developed by a reactor designer, not materials experts.

	SS	Ni-Alloy	Nb/Zr	Mo44Re	TZM(Mo)	W	SiC
Reasonable Temperature Limit	1000 K	1150 K	1400 K	1550 K	1600 K	1800 K	???
Fabricability	++++	+++	++	+	-	—	-
Experience /infrastructure	++++	+++	+	-	-	-	-
Materials database (incl. nuclear)	++++	++	+++	-	+	-	—
Cost	++++	+++	+	—	-	-	-
Availability (stock and vendor)	++++	+++	+	-	+	-	-
Cold ductility (launch/shipping loads)	++++	+++	++	+	-	-	—
Oxidation resistance	++++	+++	—	-	-	-	+++
Heat pipe performance, experience	++++	+++	+	+	++	-	—
Na and/or K compatibility	+++	++	+	++	+++	+	-
Density	++	+	++	-	+	-	++++
Thermal conductivity	—	-	-	-	++	+++	+
Neutronic performance (nominal)	+++	+	+	-	+	—	++++
Neutronics (water immersion)	—	-	-	+++	++	+++	—

Warning: do not rely on this table for design or technology decisions, use it only to perhaps guide what materials to investigate for a trade study. (1) The ratings have many caveats that would change for one concept versus another (e.g. with respect to fuel compatibility/experience). (2) advances in materials technology and testing might change the ratings (especially for SiC). (3) The ratings were developed by a reactor designer, not materials experts.

As mentioned above, do not rely on this table for any design or technology decisions (for the three reasons stated above), use it only to perhaps guide what materials to investigate for a trade study.

In the above table, the relative number of “+” (good) and “-” (poor) are meant as relative comparisons to the other materials. Neutronic characteristics are split into two categories: nominal is an indication of neutron absorption in the operating spectrum and

water immersion is how the change in absorptions might help mitigate accidental flooded criticality. The columns can be considered to represent a variety of materials, as listed below.

- SS = Stainless Steel: SS316 (hands-down the best combination of experience, data, and infrastructure), SS304 (usually lower mass if simpler requirements allow it), HT9 (only if extremely high fluence expected), oxide dispersion-strengthened steels (may be promising, but little irradiation data).
- Ni-Alloy = Nickel-Based Alloys: IN718 (probably the best combination of experience, data, and infrastructure), HastelloyXorN (frequently considered for reactors), Haynes-230 (great strength and infrastructure, limited irradiation data).
- Nb1Zr: Nb with 1w/o Zr is perhaps the easiest refractory metal to work with, and proposed for past space reactor programs. However, the potential for oxidation is a major problem during fabrication and testing (e.g. the need to test in a very strong vacuum).
- Mo44Re: Provide a nice combination of high temperature and manufacturability, but Re only neutronically practical in fast reactors.
- TZM(Mo): TZM is an alloy of mostly molybdenum (with small amounts of Ti, Zr, C), which provides higher strength and in some cases better ductility. TZM/Mo provide great strength and relatively good neutronics, but both TZM and Mo require special care to manage ductility concerns.
- W: Tungsten, or W alloys like WMo or WRe, offer great strength at high temperature; however fabricability, cost, and ductility all make it very difficult to design a reactor with, but can ultimately provide the temperature needed for future generations of low mass space power systems.
- SiC: Silicon carbide based materials, as a reactor structure, are the hardest material to rank on this chart, and the ratings should be taken even less literally than the others. Various forms of SiC (e.g. silicon carbide ceramic matrix composite) are gaining increasing attention and research because of its excellent potential performance. Perhaps in a few years a better comparison can be made. Other options in this class could include ZrC or carbon-carbon composites.

Core structural configuration

In addition to the choice of structural materials, there are various options that concern the general geometry and configuration of the core.

Grid-Plate: A grid-plate core uses a structural plate(s) to position and secure the entire lattice of fuel pins; likely cylindrical or hexagonal in shape, but not necessarily. This type of configuration is limited to relatively small reactor cores, partly because of spatial power variations that can occur in a large reactor (e.g. the need to change pin spacing or orificing of coolant flow rate in different regions of the core or enable to shuffle fuel.). The advantages of a grid-plate structure are the simplicity of construction and assembly, and the prior experience in designing and constructing pin-type reactors. The disadvantages of a pin-type structure are the potential for pin vibrations, geometric distortions (e.g. bowing), grid plate wear, relatively high void fraction (increases the potential for flooded criticality), and potential reactivity effects of relative fuel movements (including the potential for compaction of pins closer together via launch accident).

Monolith-block: A monolithic-block reactor core is effectively one solid piece; it could be fabricated as one piece, or pieces that are joined together to create the thermal-mechanical performance of a solid-block. In very low power reactors, the monolith can be the fuel itself, as for the Kilopower concept (Poston, 2021c). In most cases the monolith will be constructed from the structural material, and fuel, coolant, and potentially moderator are placed into holes within the monolith. In general, a monolithic-block concept provides the most robust design, which allows good prediction of fuel movements (reactivity feedback) and a solid conduction path between adjacent regions of the core (which are useful for decay heat removal and if there are failures in individual heat removal paths). The monolith block also has an advantage in launch accident safety because of the inability to compact the fuel pins closer together. The biggest drawbacks of using a monolithic-block are the potential difficulty of fabricating the block, and subsequently completing any required thermal or structural bonding requirements for the fuel, and potentially heat pipes and/or moderator (including numerous hermetic welds). The continued development of metallic 3D printing might make this option even more attractive.

Modular options: A modular option brings together several smaller assemblies to create the final core geometry; these assemblies can be either collections of fuel pins or smaller block-type sections. A modular-block design differs from a monolithic-block in that the core pieces are not physically bonded to each other. A modular-pin design uses fuel bundles or assemblies with smaller grid structures, instead of attaching all fuel pins to one large grid structure. The vast majority of terrestrial reactors use a modular collection of fuel assemblies, because their large size makes a monolithic or singular core impractical. The advantage of a modular option is almost entirely in fabrication; i.e. several smaller assemblies are easier to create than a single large one. A modular option also allows easier in-pile testing of components, and lessens the impact of an error during fabrication (e.g. it is less costly when damage occurs to one module as opposed to the entire core). The disadvantages of a modular approach is that it introduces additional complexity in many areas. An additional structural support subsystem is required – essentially a grid-plate structure for the modules. A modular option also adds potential negatives and uncertainties in performance. Movements of modules with respect to each other can create unwanted gaps or concentrated stress points. These movements create additional reactivity feedback mechanisms and potentially create unique flooded criticality configurations (e.g. if they become dislodged). A modular approach can also increase mass due to the need for additional gap tolerances and/or structural material.

Neutron reflection

Neutron reflectors have a very important role in space reactors. A good neutron reflector can:

- Allow the reactor to achieve criticality at lower fuel and system mass.
- Flatten the fission power distribution to decrease peak temperature and burnup.
- Provide shielding of the core radiation to help reduce shield mass.
- Reduce the potential power spike of criticality accidents.
- Provide enough neutronic worth to allow simple external control.
- Allow high temperature operation and good heat transfer when needed.
- Allow a fail-safe option to preclude launch accident criticality.

Fortunately, one natural material exists that does all of the above very well – beryllium. Beryllium has low mass density, high neutron scatter, low neutron absorption, has high thermal conductivity, and can tolerate relatively high temperatures. No other material comes close to the benefits of beryllium; in fact space reactors might not be very attractive without it.

The only significant negative of beryllium is the potential health issue of berylliosis. Only a small fraction of the population is sensitive to beryllium, but those that are can develop a very serious chronic lung condition. Modern fabrication facilities incorporate many precautionary measures and have greatly limited exposure to workers, but this is an issue that must be considered for operational safety and potential accidental exposures, including launch accidents. The other negative aspect of beryllium is that it is relatively expensive (partly because of the health measures that must be followed in order to use it), but this expense is not likely to be a major fraction of a space reactor budget (whereas on Earth, the health and cost issues might outweigh the benefits of beryllium). There are many other potential materials for the reflector (silica, alumina, magnesia, boron-carbide (enriched in ^{11}B), diamond, etc.), but it is highly unlikely that any of these would be preferable to beryllium (unless a reactor could ultimately use in-situ reflector materials at its final destination).

There are two forms of beryllium that can be used as a neutron reflector: beryllium metal (Be) and beryllium oxide (BeO). The metal form is easier to fabricate and is a much better structural material (less brittle). The ceramic form, BeO, can potentially operate at higher temperatures (depending on structural requirements) and is create more neutron reflection. Oxygen packs tightly within the BeO lattice such that the material has a higher number density than Be, and oxygen has even high scatter cross section than beryllium. A relative comparison of Be and BeO is listed below.

<i>Advantages of Be</i>	<i>Advantages of BeO</i>
- Lower mass density	- Higher macroscopic scattering
- Easier fabrication	- More chemically inert
- Better structural material	- Higher temperature potential
- Higher conductivity at high temperatures	- Higher conductivity at low temperatures
- Less potential for cracking, swelling	- Better shielding effect
- More neutron moderation	- Less neutron moderation

The last row may appear to be a contradiction (i.e. “more” or “less” can both be an advantage), but there are some reactor concepts where an influx of moderated neutrons is beneficial, while in others it is a problem. Beryllium metal provides increased moderation because of lower atomic mass, and also because neutrons that return to the fuel incur more net scattering events (neutrons travel further out into the lower density reflector, thus have a smaller solid angle to return directly to fuel).

On a mass basis the two options, Be and BeO, are often rather similar to each other. The BeO reflector is thinner, but the lower density of the Be balances the mass (depending on many factors). The lowest mass option can depend on the reactor’s preference for moderated neutrons, depending on the level of isotopes that absorb neutrons in the fuel and/or structure. The biggest discriminator between Be and BeO is sometimes the reflector effect on power peaking. The goal of the designer is generally to have the same fuel power density on the edges of the reactor as in the center. If a Be reflector makes the core edge power density too high, then BeO may be preferred, and vice-versa. The goal of a space reactor design is to minimize system mass, so the shielding effect of the reflector also plays a major role. Structural mass must also be considered, which sometimes gives Be an advantage because it can better offer its own structural support.

Thermal management is another major factor that affects the choice of reflector material. The conductivity of both Be and BeO are very good, but vary differently with temperature; the preferential conductivity profile will depend on the differences between nominal and decay power heat transfer. The potential for BeO to crack is a thermal management risk, while the higher possible temperature of BeO (depending on other factors) can create a thermal management advantage.

The final discriminator between Be and BeO is irradiation damage potential. Both materials can undergo significant swelling under the right conditions. Beryllium metal is generally more resistant to irradiation swelling than BeO. There is relatively little swelling in Be at fast neutron fluences $< 1\text{e}21 \text{ n}\cdot\text{cm}^{-2}$, regardless of temperature. If fast neutron fluence exceeds $1\text{e}21 \text{ n}\cdot\text{cm}^{-2}$ the Be temperature should generally be kept below $\sim 800 \text{ K}$, to prevent swelling and He embrittlement. Beryllium oxide exhibits the opposite temperature dependent behavior; swelling is much greater at lower temperatures than at higher temperatures (due

to an annealing effect). At temperatures >1000 K swelling is rather small up to $1e21$ n/cm² and even higher. At temperatures <500 K swelling is very high at fluences as low as $2e20$ n-cm⁻². Another possible negative of BeO is that most of the irradiation data is based on older grades and manufacturing techniques; “modern” BeO might exhibit different behavior and/or require more testing.

Practical space reactor designs are most likely to be cylindrical, which generally simplifies fabrication, control, heat removal, etc. To be mass effective, these cylindrical designs will usually utilize a radial reflector (around the outer circumference of the core) and axial reflectors (at top and bottom). In general, the axial reflector region will operate at high temperatures and also be in contact with other high temperature materials (fuel, spacer, structure, etc.). Therefore, BeO is usually the best choice for the axial reflectors, given the high temperature capability and chemical inertness. In addition, BeO is usually good choice for the axial reflector because it can play a better role in shielding. Sometimes, a design will use a thicker axial reflector on one end versus the other, due to shielding requirements, or to tailor the axial power distribution if desired. The radial reflector region is usually much larger (more mass sensitive) and more difficult to thermally manage. The higher scattering of BeO can be advantageous because it can allow smaller thickness, and this reduction in shield radius can reduce system mass. Although, if the reflector can be kept cool and power peaking on the core edge is low, then Be can have an advantage.

All of the above issues make the choice between Be or BeO rather complex, and frequently the optimal selection is not which material provides the lowest mass based solely on a criticality basis. A real engineering design is usually required to make a proper selection.

Neutron moderation

The reactor neutron spectrum is one of the fundamental design choices for any fission reactor. Many times the decision is presented at the highest level as fast versus thermal neutron spectrum. In reality, the decision is whether to use any special means to modify the spectrum, which can result in an energy spectrum anywhere between fast and thermal; although, in a compact space reactor it is difficult to achieve a truly thermal spectrum. For the purpose of this discussion “unmoderated” will be used interchangeably with “fast,” even though an unmoderated TRISO reactor may not be considered fast. A “moderated” system refers to a reactor that uses specific low-Z materials to slow down (moderate) neutrons to lower energies than would otherwise be seen in an unmoderated reactor with the same technologies. Reactors that use UZrH or carbon-based fuel, and/or use water cooling are inherently moderated, whereas externally-moderated systems will position moderator separate (heterogeneously) from the fuel.

The two most attractive candidates for external moderators are generally ZrH and YH, because they can provide high hydrogen density at relatively high temperatures; ZrH can potentially operate up to 950 K, while YH can perhaps go as high as 1100 K. These limits are general and should not be used as design limits for any particular reactor; they depend greatly on the desired hydrogen ratio, the effectiveness of coatings/liners to prevent hydrogen loss, the ability to stay within a temperature range to discourage swelling, and enough margin to prevent overheating due to uncertainties and potential transient conditions. Other hydrogenous materials like LiH (enriched in ⁷Li), water, polypropylene, etc. are generally unattractive because they would require significant cooling within the reactor. Beryllium and graphite can potentially be used, but they are relatively ineffective in the small geometries required for space reactors.

The following paragraphs discuss the pros and cons of fast versus moderated systems for space reactors.

Mass: At first glance, a moderated system appears to have a significant mass advantage over a fast system. This is because the fuel inventory can be reduced substantially (assuming the same enrichment is used) and the fuel accounts for a significant fraction of space reactor mass. The actual mass savings would depend on how well the moderating material (most likely hydrogen) could be placed in the core geometry; if the reactor could be made more compact, the use of a moderator could also save significantly on reflector and shield mass (although moderated reactors are often larger in size, not smaller). A moderated system has a lower neutron flux (because the fission cross section is higher) and a smaller neutron-leakage fraction, both of which can help reduce neutron shield mass. Alternately, the gamma shielding mass of a moderated system can be higher because it generally has less dense material in the core than a fast system. As mentioned earlier, the most important contributor to system mass is temperature, so even though the reactor and shield might be physically lighter for a moderated system, it may not provide a lighter spacecraft if the temperature is lower. Another factor is that a moderated system is likely to have more uncertainty and development risk than a fast system, which increases the probability of upward mass creep during development.

Thermal performance: In a very small reactor, only hydrogen can be an effective moderator (hydrogen slows down neutrons far more efficiently than any other atom because its mass is essentially the same as a neutron, which maximizes momentum exchange due to conservation of energy). The primary difficulty of using hydrogen or a hydrogenous material is desire for high reactor temperature. There is no established technology that can hold hydrogen at temperatures >1100 K for a long period of time. The other option is to attempt to cool the moderating material, which adds a new level of complexity to the reactor system and would add substantial risk in cost, schedule, and reliability. Adding a second cooling mechanism, whether pumped or heat-pipe driven, and integrating it into the system would be very difficult. Another factor that could limit the potential for a moderator to save mass is if the reactor were heat-transfer limited. If the same fuel enrichment were used (which would be the case if mass reduction was the goal), then the power density would be higher in the moderated reactor, which may not be acceptable in a heat-transfer-limited system.

Nuclear performance: There are considerable neutronic differences between fast and moderated reactors. In a moderated system, the neutrons undergo many collisions before absorption, which adds significantly to the neutron lifetime, especially as its velocity becomes slower. The difference in neutron generation time between the fast (microseconds) and moderated systems (milliseconds) is great, but the overall effect of this difference is minor. Neutron lifetime could impact system stability and levels of overshoot and undershoot of powers and temperatures, but these differences will generally be small. The only condition where short neutron lifetime could present a problem is if a scenario arose where prompt supercriticality was achieved. Moderated systems could have an advantage in prompt insertion accident, but unmoderated systems generally require much less excess reactivity, so they might have a lower probability of experiencing a prompt criticality. Moderated systems also have much larger temperature reactivity feedback coefficients. This is considered to be a major advantage in a terrestrial power plant because the large negative feedback is effective in preventing overheating in many accident scenarios. Preventing core damage during overheating is not a major safety concern for space reactors, so this advantage is diminished.

Conversely, large feedback coefficients create a significant problem during start-up and shutdown because considerably more excess reactivity is needed to sustain criticality from the start-up temperature and operating temperature. Moderated reactors also usually require more excess reactivity for reactivity loss due to fuel burnup. The need to add significant reactivity during startup increases the chance of large power swings occurring while trying to bring the reactor up to temperature. One factor that could mitigate the need for excess burnup reactivity is the potential to use burnable poisons to reduce burnup reactivity loss; although the moderated system is starting at a disadvantage in this respect if it uses less fuel for the same power level. In addition to the potential complication of higher-temperature feedback, a moderated system has additional feedback mechanisms (moderator temperature and potentially others) and the behavior of the temperature feedback is more complex. A fast space reactor is simpler because it receives almost all of its temperature feedback from physical core expansion, which generally provides simple, monotonic feedback. Also, there is more of a spatial dependence of reactivity feedback in a moderated system because of the shorter neutron mean free path. This dependence not only makes reactor behavior more complex but also makes simple point-kinetics calculations and simple testing techniques less valid.

Lifetime: The impact of moderation on system lifetime is dependent on many factors. If system lifetime is limited by fast fluence, then the moderated system has an advantage because fast flux will be lower; although this is application and technology dependent, and fluence does not appear to be a life-limiting factor for most potential space reactors. If system lifetime is limited by fuel burnup, then a moderated system will lose the potential mass advantage that is gained by using less fuel. To have a competitive mass, a moderated system must have considerably less ^{235}U inventory than a fast reactor; therefore the moderated system loses a larger fraction of its fissile material with burnup. Also, a moderated system will be much more sensitive to the build-up neutron absorbing fission products. Perhaps the biggest lifetime issue might be moderator lifetime itself (possibly hydrogen loss), which would depend on the technology selected for moderation and the means to ensure it stays cool. All of the effects require more fuel to overcome, but also increase the uncertainty in achieving the required “nuclear” lifetime, versus a very predictable fast reactor.

Complexity: The complexity of a moderated system is much greater than for a fast system at almost every level. Some of these factors have been discussed previously, such as the potential need to develop and integrate a moderating cooling system and the fact that the neutronic behavior of a moderated system is more complex. Any added complexity propagates through the entire program, through design, development, fabrication, analysis, testing, and system integration, thus significantly adding to program development risk.

Safety: All space reactors must be safe; the discriminator in safety is how much effort and difficulty there is in ensuring that the system is safe. As mentioned previously, the only significant nuclear safety concern is inadvertent criticality. The fast system should have two advantages in this area. First, the fast system requires less excess criticality because of the temperature defect (change in reactivity from ambient to operating temperature). The reduced excess criticality results in a smaller reactivity change between nominal and flooded conditions, which is an easier condition to meet. Second, given the nature of the material neutron cross sections, a faster spectrum can take better advantage of intrinsic spectral safety. Core materials can be used that have very low cross sections in the fast-spectrum region but moderately large cross sections in the epithermal region. If a thermal system could be designed to be fully or overly moderated, it would have an advantage, but this is probably impossible for this size of reactor. Another advantage of fast space reactors is the ability for the radial reflector to have very large reactivity worth; this simplifies operational safety because it is generally much harder to find any scenario where the fuel would go critical during nominal or accidental operations.

Safeguards: Safeguards is essentially the only area where a moderated system has a clear advantage. It can use a lower quantity of ^{235}U —either less HEU or potentially a lower-enriched fuel. A lower inventory of HEU is probably not a significant safeguards advantage because safeguards would have to be treated similarly (there is no possibility that the volume of HEU could be reduced to the point where protection requirements became less stringent). The only potential advantage of a moderator is that it offers the potential to provide a lower mass system the uses of HALEU, but there is a difference between low mass on paper versus an actual built reactor, which was discussed earlier.

Cost and development risk: The only significant cost advantage of a moderated system would be fuel security and handling costs, if it could use HALEU where a fast system could not. However, almost all of the other factors mentioned favor the fast system as being lower cost and easier to develop.

The table below gives a high level look at the relative merits of moderated and fast systems for space reactors. Unless safeguards becomes the predominant issue or a low-mass spacecraft application can be found at lower reactor temperatures, the fast system

appears to be the best choice. Although, this is a highly generalized view – it is hard to make a blanket statement about the use of moderations without considering the specifics of an application.

	<i>Moderated</i>	<i>Fast</i>
Fuel mass	+++	
Reactor mass (including shield)	++	
Power system mass	+	
Thermal performance/simplicity		+++
Neutronic performance/simplicity		+++
Lifetime		+++
Testing effort		++
Safety effort		++
Safeguards effort	++	
Cost and development		+++

Reactivity control

Reactivity control for most reactors is achieved by physically introducing/removing a neutron poison material into/from the core; although many other methods are possible: e.g. creating internal voids, poisoning the coolant, introducing moderator, etc. The compact size of space reactors allows for the option to control the reactor externally versus internally. Internal control is what is typically used for terrestrial reactors, i.e. control rods or elements move within the fueled region. External control utilizes elements that move outside of the fueled region; this method of control used for some small test reactors and all successfully deployed space reactors.

Internal control

For most space reactors, internal control (control rods or blades) is generally not desirable unless sufficient reactivity worth cannot be obtained with external control. This is because internal control elements provide more difficult system integration, including penetrations through the reactor vessel (if applicable), disruptions in the core structure/lattice, additional in-core structure, and thermal management concerns. Thermal management can be a major issue, because the poison will have a higher power density when inside the core and the heat removal path from the poison may be difficult and/or uncertain. The additional heating and irradiation damage also increases the possibility of bowing or distortion, which increases the possibility of a stuck element.

In cases where even more reactivity worth is required (or lower mass is needed), the internal rod can contain a length of poison “followed” by a length of material that serves as a reflector, moderator and/or fissile material (as opposed to a void). The use of a “follower” can increase reactivity worth significantly (thus lower mass), but exacerbates all of integration/heating issues. The power deposition in the follower will make it run hotter than the fuel; such that a fission follower is very problematic (unless active cooling of the follower can be easily and reliably provided). In addition, the length of the rod is twice as long (which makes alignment more difficult) and requires storage length on both sides of the reactor. The use of a follower often looks very good on paper, but it should not be selected unless a full engineering design can justify it.

Like all of the discussions in this encyclopedia, the above statements are generalized. There may be cases where internal control not only provides low mass, but also provides the simplest option. The use of internal control can be ideal for very small, low-power reactors without a pressure boundary, e.g. Kilopower.

External control

Poison Control Drums: The most common option for external control is the use of rotating cylindrical control drums. One side of a control drum contains a material with strong neutron reflection (e.g. Be or BeO). The other side contains a material with strong neutron absorption (e.g. B₄C, Hf, etc.). Reactivity is increased by rotating the poison material away from the core, and vice-versa. General nomenclature defines control “drums in” as the shutdown condition, in which the poison is closest to the core. Control “drums out” defines the maximum reactivity condition, in which the poison is furthest from the core.

Leakage Control Drums: Another form of control drum uses void, as opposed to poison as the method to decrease reactivity. A segment of the cylinder is “removed,” delineated by a circular chord. When the voided portion of the cylinder is closer to the core there is more neutron leakage (lower reactivity), and vice-versa. In a high L/D reactor, the least reactivity position might be when the drums are partially rotated, to give the core a direct view of space. In low L/D reactors, the least reactivity position might be when the drums are fully rotated such that the voided area is all closest to the core.

Sliding/Moving Reflectors: Another option for external control is to physically move the neutron reflecting material away from the core. Reflector segments (sometimes referred to as “sliders”) can be translated or slid axially along the core perimeter, exposing part

of the core to a voided leakage gap. This technique usually provides more reactivity worth than any other external control option, because of the high radial reflector worth of compact reactors. Another similar option is to pivot reflector segments outward like flower petals or an umbrella; this option removes the system integration issue (if the segments fan outward but stay within the shield cone angle), but does not provide nearly as much reactivity worth as sliders.

Poison Shutter Control Elements: External reactivity control can also be achieved by a thin annulus (or other shape) of poison material that slides like a shutter between the core and reflector. Usually, this method does not provide enough reactivity worth to satisfy all neutronic requirements, unless the returning flux from the reflector is highly moderated (which is generally not an attractive design option). The only practical use of the poison shutter might be as a launch safety mechanism, because the poison will be much more effective in a flooded reactor state than the nominal state.

The options above represent the four combinations of the following two choices: absorption vs leakage and rotation vs translation. Each choice comes with specific disadvantages. Some of the relative disadvantages of absorption vs leakage are listed below.

<i>Absorption/poison control disadvantages</i>	<i>Leakage/void control disadvantages</i>
- Generally less reactivity worth, higher mass - Additional material/component required - Heat removal path needed from poison - Burnup of poison isotopes diminishes worth	- Radiation leakage is major shielding issue unless a pure shadow shielded application - Gaps create a thermal short circuit from core, causing major variability in power balance - Reactor criticality affected by surroundings

The disadvantages of using a poison are all rather small issues, except for heat removal and perhaps burnup in a very high power/flux reactor. The radiation issue caused by leakage control is essentially a showstopper for any application that requires radial shielding, or any surface application. The power balance issue is also a significant engineering issue, typically much more problematic than any of the issues listed with using a poison.

Some of the relative disadvantages of rotation vs translation are listed below.

<i>Rotating drum control disadvantages</i>	<i>Translating control disadvantages</i>
- Generally less reactivity worth, higher mass - Azimuthal power variations with rotation	- Potential for very large axial power peaking - The need for space to withdraw elements creates system integration and shielding issues - Long non-cylindrical elements with asymmetric heating may be more prone to bowing or distortion - Potentially more issues with alignment, bearings, gravity and power requirement - Perhaps more likely to move into fully reactive position with accidental impact - Change in center-of-gravity with movement

The only significant disadvantage of using rotation is mass. The azimuthal power peaking of rotation is generally less of a concern than the axial power peaking of translation, plus as seen above translation introduces several potential issues.

The issues discussed above usually lead to the selection of poison control drums for reactivity control, unless mass is of the highest priority.

Control redundancy and diversity

In terrestrial reactors, control redundancy and diversity are generally required to ensure safe shutdown (although this is not always an absolute requirement). Redundancy implies that at least 2 independent options for shutdown exist and diversity implies that at least two different types of shutdown mechanisms exist.

For a reactor that operates in space, the ability to shutdown will not present a significant safety issue, so the need for redundancy and/or diversity is mostly a reliability issue; unless astronauts are nearby, but even then, the reliability of their power source is probably more important to their well-being than radiation release. Regardless, if control drums are used it is likely that only 1 or 2 of them would have to operate to shut down the reactor. The only area where redundancy and/or diversity might be important to space reactor safety is to prevent the inadvertent withdrawal of control elements, which could be provided through redundant interlocks or features that prevent motion without a controlled initiation. Some space reactor control reliability issues are: whether to incorporate reactivity margin for the reactor to operate if 1 or 2 drums are stuck in the shutdown (drum in) position, whether to gang control elements together or keep them independent, whether to include a redundant control motor for each drum/gang, etc.

Summary

There are a wide variety of technology choices for space reactors, although in some cases the required performance (e.g. temperature, lifetime, mass) reduce the number of practical choices to a relatively small number. The fuel and structural material choice must be

made in tandem, and also consider the selection of coolant (which is discussed in the next chapter of this encyclopedia). All technology choices must attempt to balance system performance (e.g. mass, power, lifetime, reliability, etc.) with the cost and risk of successful deployment (e.g., design, development, procurement, integration, testing, safety, etc.).

See Also: Fuel Rod, Fuel Assembly, and Reactivity Control Assembly Design; Heat Transfer Systems; Instrumentation and Control Systems of Nuclear Power Plants; Physics of the Fission Chain Reaction in Thermal and Fast Reactors; Principles of Design and Operation of Nuclear Power Reactors; Transient Analysis Methods.

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Space Fission Power: Technology Options—Heat Transport

David I. Poston, Space Nuclear Power Corporation, Los Alamos, NM, United States

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Introduction

Space fission power systems (space reactors) can enable truly ambitious space exploration and a permanent human presence beyond Earth orbit. Earlier articles in this encyclopedia discuss the benefits and challenges of space reactors (Poston, 2021a) and the history of space reactor development and deployment (Voss, 2021). Other articles present the technology options for space reactor core technologies (Poston, 2021b) and shielding (Poston, 2021c), while this article discusses the options for core cooling technology. The discussion and comparison of these technologies separately is not preferable, because for any specific space reactor application the selection of each technology cannot be decoupled (i.e. they are dependent on each other). However, at the highest level this article can provide a general comparison of various cooling technologies.

Power conversion assumptions

The only way to adequately compare reactor cooling technologies is when they are coupled with power conversion systems (PCS). For this comparison a power in the range of hundreds of kilowatts thermal is assumed. At this thermal power level, there are two likely near-term candidates for power conversion—Stirling and Brayton. Thermoelectric (TE) power conversion is certainly viable, but its relatively low efficiency usually makes it unattractive at high power levels, because the reactor must produce (and the radiator must reject) substantially more thermal power. A TE PCS might still be attractive at low powers ($< \sim 10$ kWe) if high reliability is valued over mass, provided that a sufficient quantity of qualified thermoelectrics is available. There are numerous other power conversion options that could be considered, but are generally considered either lower performance and/or less mature for operation in space: e.g., Rankine, thermionic, AMTEC, thermophotovoltaic (TPV), magnetohydrodynamic (MHD), etc.

The comparison between Stirling and Brayton is not straightforward. In many cases Stirling engines are a better choice than Brayton systems, because of generally higher efficiency, and perhaps more options for redundancy (due to smaller unit sizes). Stirling converters also may be more attractive for surface applications, because gravity can be used to make coupling to the Stirling engines more simple and reliable. However, Brayton systems have the experience of a large commercial terrestrial infrastructure for

turbo-machinery. Also, the mass of Brayton systems tends to scale much better at higher power than Stirling engines. A very rough, simplified rule of thumb might be that Stirlings are preferred below 50 kWe and Braytons are preferred above 100 kWe. Clearly, this rule of thumb depends greatly on the specific application, and also on technology development; e.g., there is current research involving low power Braytons and high power Stirlings, either of which could change the balance.

As noted above, the comparison provided is generally directed at a power level of 100 s of kWe, which leans the discussion towards Brayton systems and allows a direct-gas-cooled reactor to be considered more favorably. Wherever possible, statements are caveated to indicate how technologies and trades might be impacted by power level and/or PCS technology.

More information about dynamic power conversion, specifically Stirling versus Brayton, can be found in (Mason, 2013). A discussion of heat transport and power conversion for the Kilopower and KRUSTY reactors can be found in Gibson et al. (2020). There is also a very good discussion of potential heat transport and power conversion technologies in Angelo and Buden (1985).

Core cooling options

The three most commonly considered means of cooling a space reactor are:

- Gas cooled,
- Heat-pipe cooled, and
- Pumped-liquid-metal cooled.

There are many other options, including natural convection, liquid fuel, conduction-only, water coolant, organic coolant, internal boiling, etc., but these are usually found to have either poor performance and/or require significant technology development.

Each of the three common power removal technologies are introduced in following sections. In each section, a representative schematic of the reactor and core are shown. Each of the concepts shown in Figs. 1–3 are preliminary ~500-kWt conceptual designs for the Jupiter Icy Moons Orbiter (JIMO) program (Poston et al., 2004). These reactors were all designed to use uranium nitride fuel and were intended to power a 100 kWe Brayton PCS for 12 years.

Gas-cooled space reactors

A gas-cooled (GC) reactor is the simplest choice for core cooling—the same working fluid for the Brayton system can also be the coolant gas for the reactor. The basic flow layout would have the working fluid emerging from heat rejection system (HRS), being compressed by a compressor using the shaft power from the turbine, flowing through the reactor and heated, performing work as it expands and passes through the turbine and then back to the HRS for cooling. This system will have very simple plumbing, with a single compressor/turbine (although a multi-staged turbine could be considered), and one or more simple heat exchangers (a recuperator is almost always desirable to improve efficiency).

The core flow channels will generally be larger than for a liquid metal system, because the gas will have a lower heat transfer coefficient and lower heat capacity. The gas will need to be at a high pressure (to enable heat transfer) which can increase the stresses and structural mass. The gas boundary will also represent a single point of failure. In summary the system is very simple, but that simplicity comes with some increased weight and potentially lower reliability. Some of these concerns can be reduced if

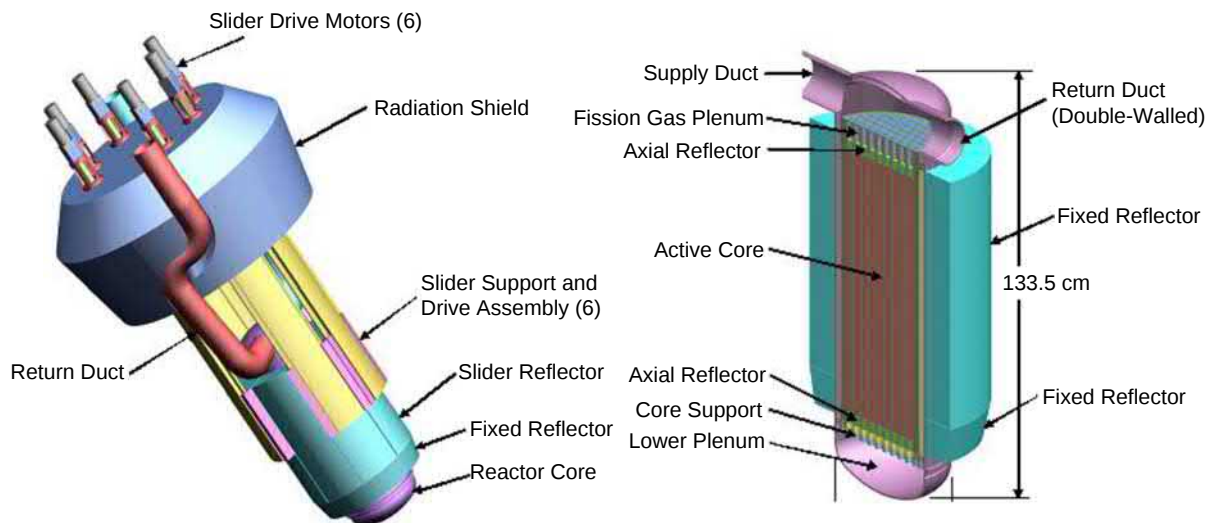


Fig. 1 Gas-cooled JIMO reactor concept.

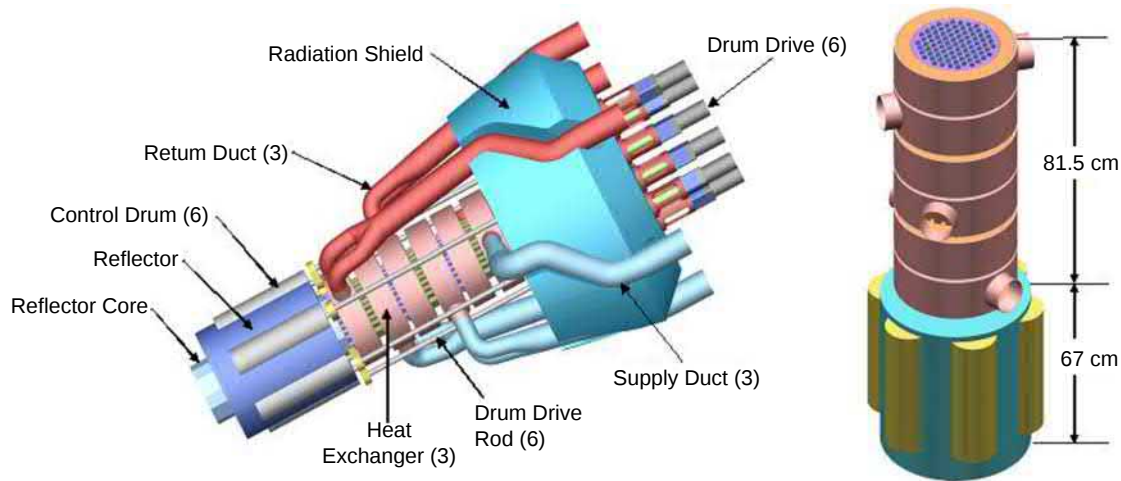


Fig. 2 Heat-pipe-cooled JIMO reactor concept.

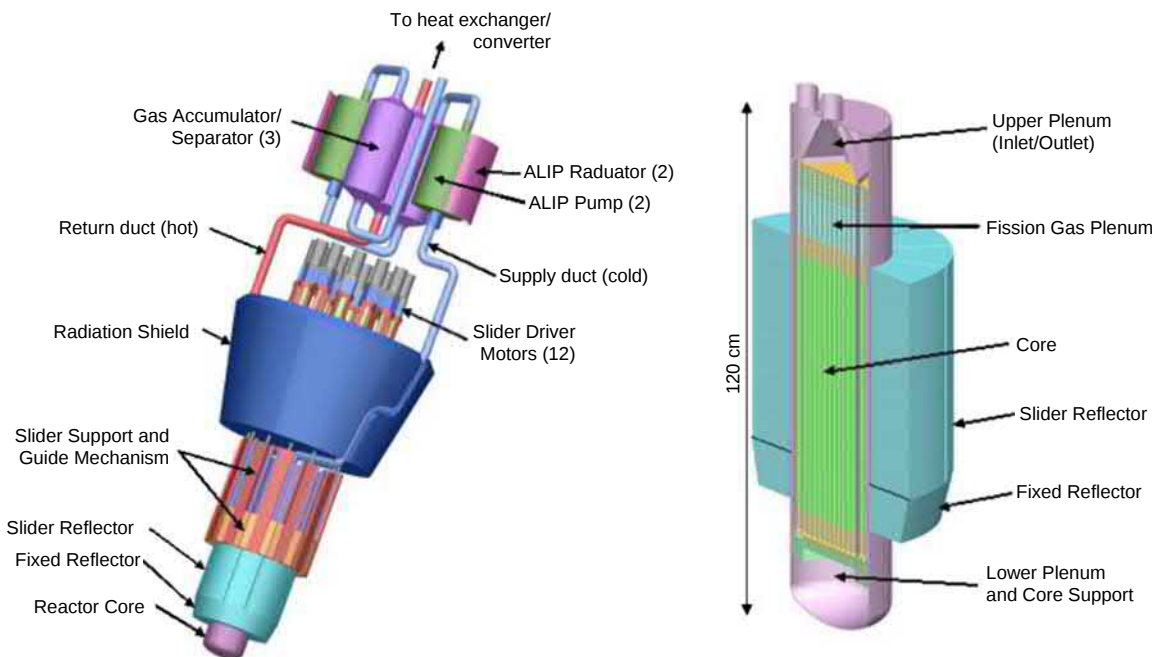


Fig. 3 Liquid-metal-cooled JIMO reactor concept.

a super-critical CO_2 reactor/Brayton can be developed, but in the near-term coolants like He, HeXe, HeKr, or standard CO_2 appear more likely.

A gas-cooled space reactor concept has never been developed, but it was selected as the baseline technology for the JIMO program shortly before its cancellation (Donald, 2005). The GC concept was selected as the baseline (for a higher power 200-kWe system than the ones shown) mostly because it would be quicker to build and test, with the backup being the heat pipe reactor (which was preferable for reliability reasons, but a HP reactor had never been build and tested at the time). A more recent study of gas-cooled Mars surface reactors can be found in Poston and McClure (2019).

Heat-pipe-cooled space reactors

Heat-pipe-cooled (HP) reactors have approximately tens to potentially thousands of heat pipes bonded to the reactor core, which transfer heat out of the core using passive physics. In a heat pipe, power enters the evaporator region to boil the working fluid (which is a liquid metal in the high-temperature heat pipes used for space reactors). The resulting pressure build-up causes the vapor to travel (limited by the speed of sound) to the colder end (condenser) of the heat pipe where it condenses and transfers heat. The

condensed liquid is then brought back to the evaporator of the heat pipe via capillary forces. The capillary force is generally produced by a wick of metal screen, but can also be provided by grooves or slots. Gravity can aid or deter fluid flow depending on orientation—a heat pipe that uses only gravity to drive flow is generally classified to as a thermosiphon. This method of heat transfer (boiling liquid metal) is so efficient that the heat pipe can remain close to isothermal (constant temperature) from the evaporator (hot end) to the condenser (cold end), even if several kilowatts are being transferred over the length of the pipe.

Heat pipes used as primary heat transport will have a heat exchanger (HX) on the cold end of the heat pipe to transfer the heat to the power conversion system; although this HX can be as simple as a conductive or radiative interface. For a Brayton system the HX will transfer heat to the working fluid, which is the gas that will be circulated to the Brayton system. The plumbing for the working fluid might look similar to a gas-cooled reactor except that the gas is heated in the heat pipe/gas heat exchanger instead of directly inside the core.

An HP reactor core is typically smaller than the gas core, since boiling liquid metal in a heat pipe moves more heat much more effectively. Heat pipes also have the feature of being able to incorporate several independent HXs attached to their condensers, thus allowing for more redundancy in the system and potentially increased reliability. In Fig. 2, heat pipes exit vertically from the “top” of the reactor, and run straight through three independent gas HXs.

Heat-pipe-cooled space reactors have been considered since the invention of the heat pipe (Grover et al., 1964). Early designs were proposed by Koenig and Ranken (1977) and Houts et al. (1995), and more recently the Kilopower reactor, which was successfully tested in Poston et al. (2020).

Pumped-liquid-metal-cooled space reactors

A pumped-liquid-metal-cooled (LM) space reactor generally uses a low-density, low-melting point fluid such as sodium, potassium, sodium-potassium eutectic (NaK), or lithium. Space LM reactors require a mechanical or electromagnetic pump to circulate the liquid metal through the core to an HX and then back to the core. Natural circulation could be considered on the Moon or Mars, but it is extremely unlikely that natural convection could provide enough power to justify the mass of the reactor. The HX heats the working fluid gas used by the Brayton system; thus, much like the heat-pipe-cooled space reactor, the heat exchanger is the heating portion of the Brayton circuit that produces power. An LM reactor can have multiple heat exchangers to increase the reliability of the system, and can be easily integrated with Stirling, TE, or other power conversion technologies. Fig. 3 shows a lithium-cooled reactor concept that uses annular linear induction pumps (ALIPs) to circulate flow.

Liquid-metal-cooled reactors often produce the smallest reactor core, since liquid metals have very high heat transfer coefficients and have high specific heats. The biggest downsides are the various system components (pump, accumulator, gas separators, etc.) that may be required, and that the coolant must be thawed prior to reactor startup (or launched warm and kept warm). So, a thaw/heater system is needed to warm the coolant in the heat-exchanger, pump and core prior to operation.

Liquid metal cooled space reactors have been built and flown by the United States and the USSR, and are the most tested of all space reactor concepts. Most notable are the US SNAP reactor (Voss, 1984) and the Soviet BUK and TOPAZ reactors (El Genk, 2009). Another substantial US LM space reactor program was SP-100 (Truscello, 1992), but it did not result in a test or flight.

Other cooling options for a space reactor

Several other cooling methods have been postulated; some of them are briefly described below for completeness, but will not be included in the comparison.

A conduction only cooled core was investigated in the very early space reactor projects. It is a simple and elegant solution, but it is limited a few 100 W in power. Unless a super heat-conducting material can be found, the applications for this technology are limited.

Molten-salt cooled cores have been postulated as an alternative to pumped liquid metal. In general, a molten salt cooled reactor has many of the same positives and the same negatives as a liquid metal cooled reactor. But molten salt doesn't provide as good of heat transfer coefficients as liquid metal and this makes them slightly larger than a comparable liquid metal cooled system. A molten-salt reactor also may have unique chemistry concerns, although not necessarily better or worse than liquid metal. Being comparable to liquid metal, but slightly larger and heavier has meant that no substantial effort has been put into molten-salt cooled space reactors.

Water-cooled space reactors have been postulated, but the number of potential issues is large. The system would need to be at extremely high pressure at the temperatures needed for the reactor core. There are also chemistry issues that are manageable on Earth, but might be a major engineering problem for a space reactor. The large volumetric change with freeze/thaw is also a potential problem. Given these issues, no serious work has been done on water-cooled space reactors.

Another cooling technology that has been explored is a boiling-liquid-metal-cooled reactor. This concept has excellent heat transfer and therefore has a very good power-to-mass ratio. But the reactor dynamics is unpredictable without a large amount of nuclear test data. The impact of gravity on performance is substantial, and the testing of such a system is considered challenging because of the safety issues with boiling liquid metals. A boiling-liquid-metal cooled system has been examined analytically, but no real testing has ever been pursued.

A comparison of cooling technologies

The key to any comparative study is to provide a level, and well-defined “playing field.” This requires consistency in three main areas: (1) system requirements and attributes, (2) design approach and assumptions, (3) design methodology and tools. The best way to achieve a level playing field is for the same team of engineers to design the “competing” concepts in parallel, because this is the only way to ensure consistency in each of these three areas. It is also very difficult, and sometimes ill-advised, to try to compare reactor concepts with high-level comparisons. The best way to discriminate between concepts is to generate detailed designs and development plans for each concept. The value of the comparison depends on the level of thought and depth in the concept design and development plans.

In the discussion below, the disadvantages of the concepts are dwelled on more than the advantages. In cases where one concept has an advantage in a specific area and the other two have a disadvantage, an effort is made to avoid redundancy as much as practical. Things that are felt to be relatively equal obstacles for all concepts are not mentioned here (e.g., fuel fabrication, Be and/or BeO reflectors, etc.).

The order of listing is not necessarily in order of perceived magnitude of advantage or disadvantage, but because the reasoning flows better in many cases depending on the order of the listing.

Gas-cooled (GC) system Pros and Cons

The gas-cooled system considered here is a “direct-drive” concept—the core is in essence the heat exchanger that provides the delta-T (change in temperature) to the Brayton system. In general, the key advantages of this concept are its lack of an intermediate HX and other components, and relatively straightforward fabrication. The disadvantages are associated with the single-point failure loop which encompasses several unknown issues, and less ability to adapt core design (as opposed to an HX design) if Brayton performance is not exactly as expected while being developed.

Gas cooled advantages

Few components: The GC concept has few reactor components and subsystems. This can simplify design, procurement, and integration, and reduce technical risk. In addition, simple GC concepts could be constructed with relatively little need for technology development.

Fabrication: From a purely reactor standpoint, the GC reactor is probably the quickest to produce. This could be a significant advantage provided that the other components desired for testing (fuel, control, facilities, etc.) can also be developed/produced on the same time scale (i.e., there is less advantage to having the “reactor” ready early if other factors dictate the overall schedule).

Gas cooled disadvantages

System reliability: The entire flow system encompasses several single point failures that are dependent on the integrity of both the core and each of the Brayton components (leaks or blockages caused by launch loads, stress/creep, particulate release, mass transfer, erosion). Unfortunately, many of these issues might not occur until after several years of operation, and a nuclear-powered lifetime test that might flush out these issue may be prohibitively expensive. Brayton power conversion systems (PCS) are likely to be less reliable than the reactor, so most designs incorporate redundant power conversion systems. The problem with a GC concept is that a leak anywhere in any of the PCS can cause system failure. This system-wide single-point failure arises because the only practical GC concept appears to be one in which each PCS unit shares the same gas with the core; i.e., the use of an intermediate gas HX and/or redundant gas-flow through the core does not provide an acceptable specific power (power per unit mass of the entire system). The best option would be if reliable high-pressure, high-temperature, long-life, fast-acting, remote-control hermetic valves could be used to seal off a failed PCS unit, but that will be very difficult as well.

There is some logic for the GC reactor to have no Brayton redundancy, because if a certain percentage of Brayton failures are caused by leaks (or could subsequently cause leaks or significant particulate release), then the system may become less reliable when redundancy is added. There are also several additional pipes branches and welds associated with redundancy, all of which add to the potential list of single-point failures. Also, the GC system would require valves (in this case non-hermetic) to prevent flow through non-operating loops—these will decrease reliability if they fail to deploy when called upon or deploy inadvertently and close off a working unit (so redundancy could hurt in this regard too). The bottom line is that the GC reactor only makes sense if the user/designer is confident that a highly reliable Brayton PCS can be developed and used in space.

Sensitivity to Brayton development: The GC reactor concept offers little flexibility to change working fluid conditions (pressure, composition, temperatures) or relax reliability requirements as Brayton development proceeds. Integrated system performance can be surprisingly sensitive to component pressure drops. Problems with Brayton development could arise late in program relative to reactor development with little chance for the reactor to accommodate a change, whereas the other concepts would only require HX redesign or worst case could potentially utilize an alternate power conversion technology.

Impact of safety requirements: The simplest option for space reactor launch safety is referred to as intrinsic safety. An intrinsically safe reactor requires no specific safety mechanisms (e.g., safety rods) to remain subcritical if it becomes immersed in water. This sometimes requires that the core contains spectral-shift absorbers; i.e., materials that will absorb proportionally more neutrons in a flooded neutron energy spectrum as compared to the nominal neutron spectrum. Regardless, it is extremely hard to achieve intrinsic safety in a core with large internal coolant area (i.e., a void that can become flooded with water). The GC concept normally

requires the highest void fraction of the concepts considered, even if a high-pressure, high He-ratio coolant can be used. If intrinsic safety is not possible, it may be necessary to incorporate internal safety elements into the design, which is difficult because of the GC concept's pressure vessel, and may significantly impact reliability. This issue compounded by the GC concept's sensitivity to Brayton parameters, which could significantly change the required core coolant area as development progresses.

Decay power removal: The GC concept has several disadvantages when it comes to decay heat removal. It will be very difficult to pull heat from the core with a loss of flow. The path of heat from the internal core to the radial edge is impeded by the gas gaps (unless a monolithic core is used) and the core is generally of larger diameter and length than in the other options. It is also unlikely there would be enough buoyancy force on Moon/Mars to drive significant natural convection. The radiological safety consequences of core damage are lesser in space, but the loss of the reactor power might be life threatening to an astronaut (more so than radiation release). If a ground nuclear test is desired, then safety and approvals may be more difficult, and/or complicated by the need for additional emergency cooling capability.

Impact of fuel pin failure: Catastrophic failure of a single fuel pin, where portions of the fuel pin clad and fuel pellets are introduced into the gas-flow could be a mission ending event.

Leaks from a fuel pin have the potential to transport potential contaminants and radioactivity from the core to the power conversion system. In addition to introducing potential materials issues in the Brayton loops, pin failure could also introduce dose issues in the Brayton or nearby components.

Mass: The mass of a GC concept is usually significantly heavier than the LM and HP; unless Brayton's could be developed to use extremely higher pressure/lower molecular weight gas (HeXe is typically assumed) or perhaps super-critical CO₂. This mass disadvantage becomes even more significant for surface systems, because some form of 4pi shielding is generally required, and the GC reactor usually has the largest surface area to shield, very large pipes to route, and may require sliding reflectors (or internal control) for reactivity control as opposed to drums, which adds shield mass.

Gas cooled unknowns

Testability: Non-nuclear testing is made more difficult by the impact of electrical leads through the plenum (on flow distribution, potential leaks, and/or contamination of the working fluid in the test article). Also, heater/instrumentation failures or variation of test configurations will require breaching of the pressure boundary, which increases chance of contamination and/or schedule delays. Lack of inherent decay-power removal could require an additional safety system to be incorporated into a potential ground test (although for lower powers it may not be an issue). Overall, the testability of the GC concept appears relatively good, but it is possible one of the above issues, or an unknown will make testing more difficult than the other concepts. Also, the tightly integrated response of the core and Brayton system might require additional nuclear-powered dynamic testing.

Heat-pipe-cooled (HP) reactor Pros and Cons

The HP system considered here places Brayton heat exchangers directly adjacent to the reactor, and pipes route the Brayton gas past the shield. An alternative is to place the HXs one far side of the shield, where the HPs travel through the shield to an HP-to-gas HX (or to a potassium boiler to transfer heat if Stirling engines were to be used). The core is comprised of a monolithic block that contains the HPs and fuel pins, and allows good heat transfer in the case of failed HP(s). The key advantages of the HP system are in its reliability, testability, and flexibility in safety. The disadvantages are the potential impact of failed HPs and the complexity of fabrication and assembly.

HP advantages

Redundancy: The HP reactor has inherent core redundancy (the level of redundancy depends on the impact of failed HPs on performance). In addition, the HP reactor can be configured to provide excellent redundancy in the power conversion system (e.g., totally independent and redundant Brayton systems).

Design Flexibility: For a Brayton system, the HP reactor is far more flexible to Brayton system gas pressure and delta-T than the GC concept, which allows flexibility in the coupled reactor and Brayton system development. The HP reactor is also far more flexible than a GC reactor with respect to Stirling or TE power conversion (especially for surface reactor applications), but not quite as much as the LM system, which will be discussed later.

Uniform core temperature: A successfully operating heat pipe is essentially isothermal, whereas other concepts are colder at the coolant inlet and warmer at the coolant outlet. An axial temperature gradient can introduce thermal-structural issues. There are also some material concerns (e.g., loss of ductility with irradiation) that can be temperature dependent, such that the colder end of the core might experience a problem that would not occur if the core was isothermally warm.

Transient response and control: When heat pipes are operating within their limits, they act essentially as an infinite thermal conductors (relative to the other heat transfer mechanisms in the system). This simplifies the system response (and prediction/design of response) of the reactor power system to transient events. More importantly, there is no control action required to change the reactor power (i.e., no command to change flow rate through the core), because the reactor (if the physics are sufficiently simple) can load follow the power draw without any reactor control response.

Testability: The HP reactor can be designed with unobstructed access to the bottom of the core; thus, resistance heaters can be easily placed within fuel tubes to realistically simulate fission power deposition. This ease of testing can allow rapid build and test iterations during development, and thorough testing of the integrated flight-unit before launch. The HP reactor also allows

core instrumentation to be implemented without breaching a pressure vessel. This is not only a testing advantage, but also an advantage in monitoring startup and benchmarking initial operation in space. The HP reactor also offers the best decay power removal capability which could simplify ground test safety concerns. This may allow the ground testing of a more prototypic unit, by not requiring the integration of an additional heat removal system in the ground test article. The figure below shows the electrical testing configuration of the SAFE-30 reactor (Van Duyn et al., 2002) (Fig. 4).

Safety: The HP reactor has a very low core void fraction (even if all HPs are breached). This gives the HP reactor an advantage in mitigating water immersion scenarios. The HP reactor also provides the most flexibility if very stringent safety requirements are imposed, because the open access to the core bottom allows the use of removable neutron absorbers to further suppress reactivity in flooded conditions. The compact core geometry also allows the use of control drums, which could present a safety advantage. Finally, the HP reactor offers the best option for transporting the fuel separate from the remainder of the system (because the coolant loop need not be breached), which offers an option to simplify transport concerns.

Heat pipe disadvantages

Core fabrication and assembly: One of the biggest early design choices for a HP reactor is whether to use a modular or monolithic design. The modular design has several fabrication and assembly advantages, but a monolithic design provides a significant increase in reliability and performance—by providing good thermal bonding (heat transfer path) in the case of a failed HP. The Kilopower concept provides the simplest monolithic option (McClure et al., 2020), because the core monolith is actually the fuel. Higher power cores will require a structural monolith (perhaps stainless-steel), with fuel rods placed within it. The biggest issue with monolithic HP reactor fabrication and assembly is that several things have to go right in order to produce an acceptable end product. One possible sequence is to: (1) manufacture the core monolithic block, (2) bond heat pipe tubes to the monolith, (3) fill the HPs individually, (4) test the heat pipes with resistance-heating (integrally and individually) as much as possible with core instrumentation, (5) integrate the upper shield and HX assembly and attach (potentially braze) the HX onto the HPs, inspect and verify HX coupling, and (6) insert the fuel and weld on end-caps. There are other options that can be envisioned if problems arise in any of these steps, but no matter what option is chosen, great care will be needed for a successful product. The biggest risk is the cost and schedule risk of potentially running several “cores” through the procedure until a successful core is completed. The ability to fix errors may be the most important thing to focus on during development, in addition to a rigorous QA/QC plan. There are potential methods to drain, clean, and fill a HP if necessary, or perhaps un-braze and re-braze a HX, but it is unclear how well fabrication risks can be mitigated. If constructing a monolithic core is problematic, then there are also ways that the core could be made in segments and then bonded together in a final step. Or perhaps a modular approach could provide acceptable performance (perhaps with a liquid metal bond or fat braze). Fabrication and assembly issues are a function of the number of heat pipes, which appear manageable at powers $\ll 1$ MWt, and very difficult at powers $\gg 5$ MWt.

Potential for failed HP: A failed heat pipe is essentially a blocked flow channel for a typical reactor, but a failed heat pipe is probably more likely than a blocked flow channel in most reactors. HPs can be developed to have relatively low failure probabilities ($< 1\%$ appears easily achievable for most HPs using state-of-the-art technology), but given the large number in the core there is a good chance of one or more failing over a decade or more, so the system should be designed to tolerate failed HPs (depending on risk tolerance). If a HP fails, the power generated in fuel next to the failed HP must be transferred to adjacent HPs. This introduces four potential problems: (1) increased temperature and throughput of adjacent HPs, (2) increased temperature of the fuel pins near the failed HP, (3) increased temperature and thermal stresses in the monolith, and (4) increased stress in the heat exchanger. Each of these problems can have significant consequences, except perhaps the fuel pin temperature because the impact of isolated failed fuel

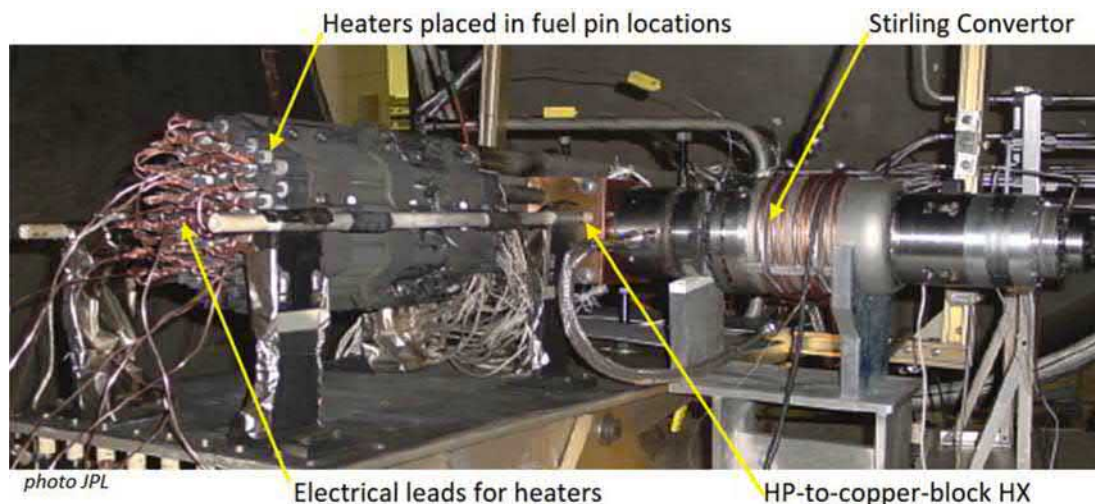


Fig. 4 Electrically-heated testing of SAFE-30 reactor coupled to Stirling convertor.

pins may not be too significant for the HP concept. The consequences of #3 and #4 can mostly be solved by designing a core with more, smaller components (which adds complexity), and/or more robust materials. All of these issues can be addressed via resistance-heated testing of a prototypic geometry.

Potential for HP failure propagation/cascade: Failed HPs increase the temperature and throughput of adjacent HPs, which in turn increases the failure probability of an adjacent HP (the increase in failure probability is mostly a function of design margin). This offers the potential for a failure cascade if there is not adequate design margin. The monolithic block design has the benefit of allowing heat to spread further away from the failed area, so that considerable heat can flow past the nearest adjacent pipes; regardless, design margins still must be well understood and tested to preclude the occurrence of a cascade. The key HP design margins are for power throughput (typically capillary and boiling limits) and corrosion. In most HP reactors the capillary limit is the closest margin, but HP failures may drive the adjacent HPs closer to the boiling or corrosion limits than the capillary limit.

HX/boiler thermal bonding: Brayton HXs will likely be slid over the heat pipes from the top, but enhanced thermal bonding may be needed to achieve high system efficiency (to minimize the delta-T from the heat pipes to the Brayton cycle working fluid). It may be difficult to promote and maintain adequate heat transfer. Candidate braze materials exist that should perform adequately, but this performance (including the effect of irradiation) will require demonstration. The braze will probably need retain about 10% contact with the HX sleeve/thimble and the HP wall to provide good thermal performance, and this contact must be distributed around most of the HP-HX surface area (e.g., it cannot be all located on the first few cm of the HP). The most significant stresses on the braze could occur because of a failed HP. A failed heat pipe will cool and axially shrink relative to the other HPs, and potentially fail the braze; although it may be of little consequence if the braze fails around a failed HP (in fact this may be good in reducing HX stress in a failed HP condition). If a braze becomes a significant issue, there are other options for thermal bonding, such as a gas or LM gap (which requires a hermetic vessel to be placed around the HP/HX assemblies, except for perhaps on Mars because of the atmosphere), thermal radiation (aided by high emissivity surfaces), or perhaps an integrated HP/HX assembly printed via additive manufacturing.

Heat pipe unknowns

Upper power limit: The manufacturing and assembly issues of the HP reactor make it difficult to project the upper power level at which a heat pipe reactor can be practical. Studies have indicated that a practical upper limit simple stainless-steel HP reactor might be $\sim 1\text{--}2$ MWe (Poston and Kapernick, 2012), but this is largely speculation. Advances in 3D printing might allow the fabrication of monolithic reactors with thousands of heat pipes, including integration with an HX, but there would still be other issues to resolve. If a modular, instead of monolithic core is selected, the power may not be limited by fabrication, but instead by diminished ability to prevent core overheat due to failed heat pipes. Second or 3rd generation heat pipe reactors could go to significantly higher powers ($\sim 5\text{--}10$ MWe?) if a refractory metal or carbon-based material (SiC?) could be used to increase operating temperature to $\gg 800$ K.

Pumped-liquid-metal (LM) Pros and Cons

The liquid metal system considered here is a fast-spectrum, pumped LM system. In general, the advantages of the liquid metal system are in performance (mass and flexibility), and the disadvantages are in development risk and reliability.

Liquid metal advantages

Scalability to very high power: The core power can be increased significantly in an LM reactor without a major increase in coolant flow channel area (depending on pump effectiveness, fuel performance, etc.). This can start to create a significant mass advantage at powers $\gg 1$ MWt, while at lower powers HP and LM concepts are often of similar mass, with GC concepts are generally heavier at all power levels.

Flexibility. Flowing liquid metal provides an excellent medium for transferring heat to most power conversion systems. The importance of this advantage depends on the development of future power conversion systems, and would be huge if low-mass, high-efficiency magnetohydrodynamic (MHD) energy conversion can be developed.

Liquid metal disadvantages

Freeze/thaw: Freeze/thaw is a complex issue that cannot be fully tested until a complete prototypic flow loop is completed. An adequate system for high-purity liquid-metal loading, freeze, thaw, and surface wetting must be developed and its performance verified. Specific issues associated with loading and freeze/thaw include the following: (1) High purity must be ensured. Flushing the system with high-purity fluid theoretically enables a high-purity final fill and provides the potential for maintaining high purity throughout the mission. However, the system must be designed to allow complete draining between certain operations, and all potential sources of contamination must be identified/eliminated. (2): The desired prelaunch location for the frozen coolant must be determined, the coolant must be loaded into that location, and the mass and location of the loaded coolant must be determined and verified to be within acceptable limits. (3): The system must be designed to keep stresses associated with initial coolant thaw within acceptable limits. Methods for measuring the stresses/strains associated with thaw must be devised and implemented for development and qualification testing. Methods for verifying that no unacceptable internal damage has occurred must also be devised/implemented. (4): Gravitational effects during development and qualification testing of the freeze/thaw system must be determined and compensated for. (5): The system should also be designed to accommodate anomalies in flight-system start-up, which could include partial melting/refreezing/re-melting of the coolant.

Testing difficulty: Realistic flow system tests are not possible until all components (core, pump, accumulator, etc.) are completed—delays in any one component could seriously set back a program. Test apparatus will be relatively complicated, and even when successfully completed, may have many glitches and be difficult to debug. It will be difficult to use resistance heaters to simulate realistic power deposition. The effort required to make any changes or corrections will be significant. When the coolant loop is breached to make repairs and adjustments, it will be time consuming and could introduce impurities. Each test requires unit to be unfrozen (and refrozen to desired inventory location).

Pump development and/or recapture: Liquid metal pumps have been successfully deployed for various applications. One of the potential pump technologies for space reactors is the Annular Linear Induction Pump (ALIP), an example of which is shown below. Unfortunately, the technology status of pumps for relatively low-power space application is in question—especially as a function of required performance (efficiency, reliability and mass). A decrease in pump efficiency (i.e., increase in required electrical power) propagates through the entire system with a substantial mass impact (plus the increase in thermal power increases “nuclear” issues, e.g., burnup, swelling, materials damage). The other issue of a pumped system is the additional complexity of providing and distributing power to the pumps (Fig. 5).

Reactor Instrumentation and Control: The additional components of the LM concept requires more diagnostics (both in number and unique technologies). This subsequently requires more collection and processing of signals, and more control commands to be given. The development and reliability of instrumentation and control will likely be more expensive and complex than for the other reactors, and require more integrated system testing.

Heat exchanger development and integration: The HX design and fabrication for the LM system has the potential to be simpler than for the HP system, however there are other problems with the HX specific to the LM design. The HX must be designed to allow wetting, draining, freezing, and thawing—the potentially small passages of the HX may introduce problems in this area. Also, in the case of the LM concepts each HX may represent a potential single-point failure.

Development risk (cost and schedule): There are a large number of technical unknowns with the LM concept mentioned individually below. These in combination with the difficulty of testing can cause significant increases in program cost and schedule, and more importantly it significantly decreases the probability of successful completion.

Liquid metal unknowns

Reliability and lifetime: While the single-point failure susceptibility may not be as great as for the GC concept (because of isolation from the Brayton loops and the difference in holding relatively low-pressure liquid-metal as compared to high-pressure gas), the complexity of the LM system makes it very vulnerable to reliability and lifetime issues. The difficulty, cost, and schedule of testing exacerbates reliability and lifetime issues to a potentially show-stopping level.

Gas-separator development: The need for a gas separator depends on the choice of coolant, and the magnitude/spectrum of the core neutron flux. Lithium and potassium generate gases via neutron absorption, while sodium does not. The operation of a gas-separator in zero-g that does not significantly impede flow is difficult, as well as hard to test in a gravity environment. The separator must also be designed to preclude the possibility the gas being reintroduced to the fluid due to a failure or off-nominal condition.

Coolant activation: This is another issue that depends on the coolant and flux magnitude/spectrum, but the bad actors are reversed when compared to the gas separator. Sodium (and thus NaK) can present a significant shielding/dose issue, while potassium and lithium are much less of a concern. Coolant activation is only a significant issue if the coolant is routed close to other radiation sensitive components, or if an astronaut needs to approach a shutdown reactor within a day or two (for repairs on the reactor or other infrastructure). The HP reactor mostly avoids this issue due to its low inventory, and all of this coolant is internal to the power system.

Integrated performance: There are enough coolant loop components (pump, thaw heaters, accumulator, and potentially gas-separator and/or chemistry modules), that even if components are successfully developed, it may be difficult to assemble them and operate them as planned as an integrated unit. The components must work together nominally, be able to tolerate off-normal performance in other components, and work together with the freeze-thaw strategy. The photo below shows technology demonstration unit for the NASA pumped-NaK fission surface power system (Briggs et al., 2016) (Fig. 6).

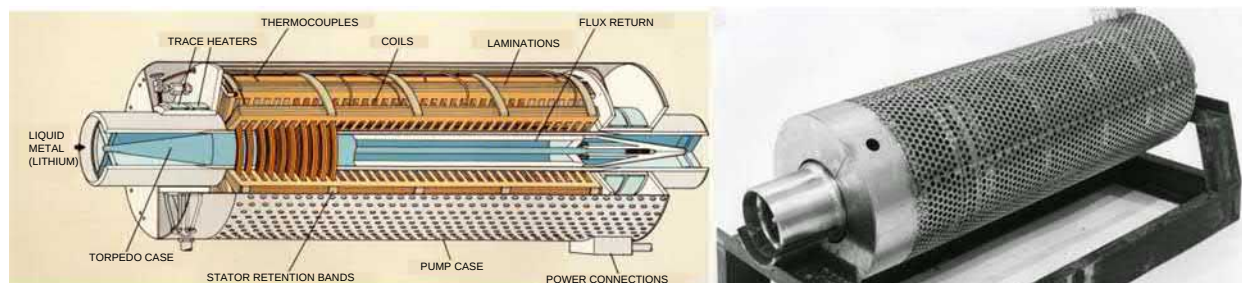


Fig. 5 Drawing and photo of a lithium Annular Linear Induction Pump (ALIP) (Adkins, 1984).

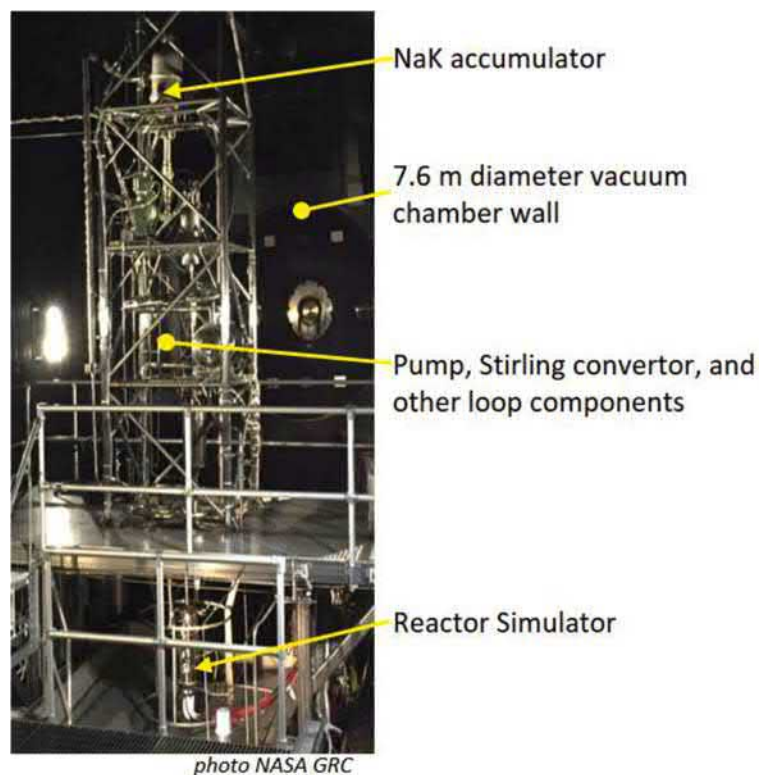


Fig. 6 Photo of NASA fission surface power technology demonstration unit.

Design constraints of each option

The design issues of each cooling technology are discussed below. Many of them overlap with the disadvantages discussed above, but are presented in the context of how they need to be considered by a reactor designer.

Gas cooled design issues

Reactor design flexibility: In the GC concept, Brayton development is constrained by core development and vice-versa, because both systems have to work with the same coolant flow rate, pressure drop, and delta-T. During the development process, if the Brayton system is found to be more sensitive to pressure drop than anticipated, the core design may not have enough margin to provide lower pressure drop, while it would be much easier to redesign an HX for an LM or HP system. Therefore a GC concept should be designed with extra thermal-hydraulic margin.

Reactivity control options: The GC concept can also be constrained in the selection of control options. External control can be difficult for a GC concept because it usually requires a larger diameter core (to provide the required flow and heat transfer area) and the potential need for thick cylindrical vessel, downcomer, baffle, and/or stagnant gap between the fuel and the reflector (depending on the design option). This may require a GC design to use sliding reflectors instead of drums, or perhaps internal control. Both of these options create development and performance difficulties, including problems with thermal management and shielding (Poston, 2021a,b,c).

Piping: The physical layout of gas piping can create integration and shielding issues. A single pipe to handle all of the gas flow can be many inches in diameter (depending on pressure and flow rate). This size of pipe introduces welding curvature issues at interfaces and cannot generally be packed into the system design without significantly increasing radiation streaming and/or shield mass.

Reactor temperature gradient: The thermal stresses of the internal axial temperature gradient must be accommodated. Also, the fuel pin is irradiated over the entire gas-coolant temperature range. Some materials issues, especially those involving radiation damage (loss of ductility), are significantly temperature dependent. This may not be an issue, but the higher the power/flux and the greater the temperature range of operation (delta-T), the greater chance that a negative materials issue could arise.

Launch loads: Launch loads will present issues for all concepts, but they may have the biggest impact on mission reliability on the GC concept (because of the large number of locations that contain potential single point failures). Launch loads, particularly random and lateral forces, will not be characterized until very late in the program, and even then, there will still be an uncertainty band and a random component to them. Launch loads on the core internals may be an issue, but the bigger issue might be the potential for a local stress at a pipe joint that damages a weak or even solid hermetic seal, creating a gas leak.

Erosion and flow induced vibration: The GC concept must consider flow-induced vibration of the fuel pins (unless flow through a monolithic-block reactor is used). Erosion could also be more problematic for the GC concept because of the complexity of flow paths and the need to flow the entire Brayton gas volume through the core (as opposed to being separated into lower-power HXs for the other Brayton concepts).

Transient response: Tight coupling and increased sensitivity to the Brayton operating state could result in dynamic issues (although it may also provide possible advantages). The lack of thermal inertia between the Brayton fluid and the fuel-pin could increase the chances of overheat scenarios (the LM concept has the Li loop, and the HP concept has the HX wall, the HP, and the monolith between the gas and the fuel). A GC concept might have more need for a nuclear-powered test to study and verify system dynamics.

Heat pipe design issues

HX/boiler fabrication: For a Brayton system, the HP-to-gas HX fabrication might require several welds and potentially difficult machining, although advances in 3D printing may eliminate this risk. There are many design options for the HX, but an annular flow geometry may provide the best combination of performance and fabricability. In an annular flow HX, the Brayton gas flows in an annulus along the length of the HPs, with inlet and outlet plenums on opposite ends. The figure below shows the SAFE-100 annular flow HX integrated with an electrically-heated HP reactor operating at 700 °C in a vacuum environment (Steeve, 2005) (Fig. 7).

A cross-flow heat exchanger offers some mechanical advantages, but will likely require more surface area, and may have more difficulty removing power from HPs that surround a failed HP. For a Stirling system, the potassium boiler should present no significant fabrication issues (except for the thermal bonding of the boiler to the HPs and Stirling engines as mentioned above). The key factor for the boiler will be thermal-hydraulic testing (including the effects of superheating), which will have to be correlated to the gravity environment of the Moon and/or Mars.

HX and shield integration: HX integration (the physical sliding of the shield and HX onto the HPs) is a design/fab issue for the HP concept. The design needs to allow for significant tolerances between the HPs and the HX holes, and the HP straightness tolerance must be tight as well. The tolerance of HP position in the x-y dimensions is not as crucial, because an “imprint” of the actual geometry could be taken and fed into the fabrication process if x-y tolerance was found to be an issue. The difficulty of integration depends on the number of HPs and HXs, and could become very difficult for high power (> 1 MWt) cores with several Brayton loops. Physical integration was not found to be a problem with the SAFE-100 test article shown above, but this HX only incorporates 19 HPs.

Operating temperature flexibility: The HP reactor has several factors that must be balanced to determine the optimal core operating temperature (i.e., the heat pipe temperature). All reactors need to balance the desire for high temperature to the PCS while maintaining sufficiently low temperature to the core materials. The heat pipe reactor has an additional constraint, in that the heat pipes must be warm enough to have sufficient power throughput limits. Also, the in-core x-y temperature gradients caused by failed heat pipes are usually greater than the delta-T in the other concepts. In most cases, at low powers $\ll 1$ MWt, there is sufficient margin to accommodate all issues. At higher powers, the larger core temperature gradient, more delta-T between the heat pipes and PCS, and higher heat pipe throughput requirements can make the window tight to accommodate. This in conjunction with the fabrication issues discussed earlier probably make the HP concept unattractive at powers $\gg 5$ MWt (until perhaps refractory heat pipe reactors could be developed).

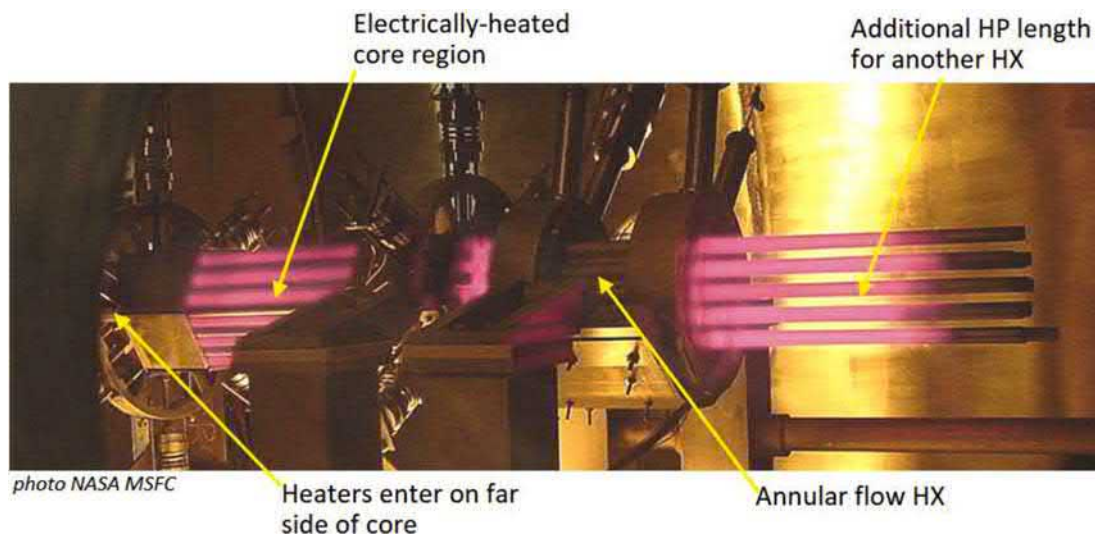


Fig. 7 Electrically-heated testing of the SAFE-100 reactor coupled to annular flow gas HX.

Liquid metal design issues

Redundancy impact: The level of PCS redundancy required for an LM system can impact design significantly, and have a surprisingly negatively impact on mass. Consider the use of four 50%-power independent PCS units. (1) If the HX are in series (and you assume two are working and the other two have failed or are in standby), the furthest downstream working HX will have coolant at a temperature lower (by about half the core delta-T) than the upstream HX. Ideally, you would design the downstream HX to be larger (to achieve the same PCS temperature), but the middle HXs may have to operate as either upstream or downstream (depending on which other HX/PCS is working), so there is no way to tailor their design. This leads to two issues, (a) all HX need to be designed to the worst case (lowest liquid-metal temperature—thus highest mass, highest gas delta-P) condition, and (b) the gas temperature and/or thermal power will be different in each working PCS loop, so the PCSs will operate at different state points. (2) If the HX are in parallel, “bypass” flow in the non-working HXs creates a significant problem. If 2 PCS loops are being used as part of a 4–50% PCS system (and LM flow continues through all), the delta-T in the working HXs will have to be ~2-times the core delta-T; this lowers the exit temperature of the working fluid from the HX, and thus lowers efficiency. Also, the size, mass, pressure drop of the HX has to be designed to meet this colder LM temperature condition, which will be much larger than for a HX designed for the nominal case. Therefore, if a parallel configuration is used, it is likely that valves will be incorporated to close-off non-working HX/PCS branches. This also requires additional hardware and development to instrument and control the valves, and more importantly the system will rely on the reliability of those valves. In both cases (series and parallel) the closer the coolant temperature is to the desired hot PCS temperature, and the higher the core delta-T, the worse this issue gets. This issue is unique to the LM concept because the HP concept has a very small delta-T along the heat pipe, while the GC concept would never use redundant systems in series, and the parallel issue was already discussed with respect to reliability.

Core support and flow configuration: The fuel pin/core support structure must be designed to prevent flow-induced vibration and/or clumping depending on flow direction. Most LM concepts use either a 1-pass or 2-pass flow configuration. A 2-pass flow configuration has down-flow in one section of the core and up-flow in the other. This allows a compact reactor design, but significantly increases core pressure drop (thus pump mass/power) due to the decreased flow-area of each pass, and could also create internal core thermal stresses. Many designs use a downcomer (an annular region surrounding the core) to feed coolant to the inlet of a 1-pass core. The generally reduces pressure drop, but adds reactor mass because it separates the fuel from the radial neutron reflector. Other designs will use feed pipes outside of the pressure vessel to feed coolant to the inlet of the core; this mitigates the mass penalty of a downcomer, but complicates system integration and creates additional regions where stresses and potentially leaks could occur.

Control system “knobs”: The LM system has an additional knob to use in reactor control (i.e., pump power/flow rate) than the other systems. This adds complexity to the instrumentation and control system, and can create scenarios where the flow control system conflicts with the PCS control system; i.e., where a change on one control variable causes a change in the other, such that both systems have to actively respond to each other. This effect complicates control algorithms significantly, and introduces additional reliability concerns. Note that there may be some operating scenarios where the flexibility provided by an extra control knob could be beneficial, but for a system that has to operate remotely and reliably for years simplicity is probably preferred.

Pumping power requirements: The need to provide electrical power to the pump introduces several issues (unless the pump is powered internally by its own thermoelectrics, which is a separate development issue). This power system development becomes very sensitive to the pump development, most importantly in terms of pump efficiency (if the pump doesn’t meet the desired efficiency the entire system can quickly become unattractive). In comparison, the GC concept will generally have a much higher thermal pumping power, but this is effectively/efficiently incorporated into the Brayton turbomachinery, while the HP concept uses passive physics to “pump” the working fluid. The LM need for pumping power also requires the PCS to create more electrical power than the other systems. This not only makes the PCS and heat rejection system larger, it also requires a higher power reactor. A higher power reactor can exacerbate “nuclear” issues (burnup, swelling, materials damage) depending on the design, and increases the shield mass (which may also be required to cover more radiator space). There is also the additional complexity of distributing and controlling power to the pump(s).

Conclusions

There are three classes of reactor that appear to be best suited for near-term space reactor applications: gas-cooled (GC), heat-pipe-cooled (HP), and pumped-liquid-metal-cooled (LM). It is not possible to make a blanket statement regarding the superiority or inferiority of these options, unless very specific performance criteria are provided: e.g., power, lifetime, environment, mass, reliability, etc. However, some generalities generally hold true. GC concepts are only well-suited for relatively high-power space applications ($\gg 100$ kWe) and are only practical with Brayton power conversion (in particular, high-reliability Brayton systems). HP concepts are the most attractive option at lower powers (< 100 kWe) and in cases where high system reliability is emphasized. LM concepts are often the most attractive when low mass is desired at relatively high power levels ($\gg 100$ kWe). Unfortunately, the number of caveats to the above generalities, and the ultimate details of every mission application, make it unwise to make design decisions based on the above.

In some ways it is unwise to speculate on what future high-power, high-performance space reactors might look like. A lot will depend on how each technology progresses in the future. GC concepts would gain an advantage with progress in space Brayton technology, and if methods to prevent and potentially repair gas leaks were developed. HP concepts will gain an advantage if

progress in 3D printing continues, and if another HP reactor concept like KRUSTY (McClure et al., 2020) is developed and tested. LM would gain an advantage if progress is made in LM pumps and a new LM reactor is successfully deployed on Earth.

Most important may be how “reactor” technology progresses; i.e., the design, built, and operation of actual reactors, instead of progress on specific reactor technologies. Reactor technology can be considered as the knowledge and infrastructure to create actual working reactors, which is somewhat independent of the specific reactor components. This is the “technology” that is most severely lacking in the present day (2020). The best way to advance reactor technology is to start with simple and affordable systems that have a decent shot (>50%?) of being actually being successfully developed, as opposed to the numerous failed space reactor programs of the past, which have pursued high-performance reactor programs with low-probability of success (Voss, 2021). For this reason the simpler, low-power HP reactor concept may be the best place to start, especially since these low-power HP concepts can provide near-term benefits for surface power on the Moon/Mars and for science missions (Poston, 2021a,b). These early generation HP power systems could then evolve to even more enabling high-power HP, GC and/or LM systems.

See Also: Heat Transfer Systems; Transient Analysis Methods.

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Space Fission Power: Technology Options—Radiation Shielding

David I. Poston, Space Nuclear Power Corporation, Los Alamos, NM, United States

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Acronyms

ALARA	As low as reasonably achievable
B ₄ C	Boron carbide
DPA	Displacements per atom
EHA	Extra-habitat activities
GCR	Galactic cosmic radiation
LiH	Lithium hydride
NaK	Sodium potassium eutectic
NEP	Nuclear electric propulsion
NVT	Neutron velocity time
PMAD	Power management and distribution
SEU	Single event upset
SFP	Space fission power

Introduction

The primary function of reactor shielding is to prevent the transport of reactor neutrons and gammas to sensitive locations. For space reactors, radiation shielding is an integral system design concern, as opposed to most terrestrial reactors, where shielding can to some extent be left as an afterthought. This is because mass, volume, and separation are all valuable commodities in space, while they are rather easy to obtain on Earth (except for submarines in some respects). Space reactors also don't have the benefit of regular maintenance to replace components that might be sensitive to radiation. The difficulty of cooling

or thermal management in space also can make shield design difficult. The combination of these and several other system issues (launch loads, differential expansion, radiation tolerance, etc.) requires that shield design be an integral part of the space reactor design process.

First order, the biggest complication of shielding is mass; i.e., if mass were not an issue, shielding would not too difficult. It is the desire for low mass that leads to the selection of materials and geometries that make engineering difficult; although this is often a necessary complication, or shield mass could end up dominating the entire power system mass.

Radiation shielding criteria and requirements

Radiation shielding should never be considered a fixed high-level mission requirement. Shielding requirements should be considered and constructed as a means to best complete a mission: this starts with determining what the goals of a mission are. Shielding considerations might help decide whether a certain type of measurement or capability is practical, whether a function is performed by one instrument or another, how long a mission should last, where human activities could take place, etc.—all of these factors must be weighed against shield mass and complexity. Even once mission requirements are set, there will be many trade-offs that determine how much physical shielding is needed; e.g., how hard should one try to find or develop rad-hard (radiation hardened) components, how much separation or boom length (separating reactor from payload) can be utilized, how much redundancy should be considered for radiation sensitive components, are there in-situ materials at our destination that could help with shielding, etc. In an ideal engineering world, every space fission power (SFP) mission/system would be designed and developed in this manner. In a practical engineering world, a successful space reactor will probably need to be flexible enough to provide shielding for many applications, such that shielding modules and/or features could be added or removed depending on the mission (as long as this does not complicate system structural, thermal, and neutronic performance).

As discussed above, “radiation shielding requirements” should be part of the evolution of requirements from mission, to system, to function, to component. The only true requirements are component radiation requirements, which might ultimately be translated to regional radiation requirements once a configuration is fixed. Note that a “component” can be a transistor, a magnet, an epoxy, a telescope, a fuel tank, or even a human being. Each component will have its own radiation limits or requirements, in fact the shield itself will often have its own internal radiation requirements. Once a mission shielding solution is determined, the only true requirement for reactor shielding might be how much the fluxes/doses need to be reduced; often measured in orders of magnitude or decades.

In general, there are two sources of reactor radiation that need to be shielded: neutrons and gammas. The neutrons of concern are almost entirely from fission (prompt and delayed), with a small potential number from $(n,2n)$ reactions and an occasional photoneutron. Gammas of concern are from fission, radioactive decay products, and neutron capture, in every region of the reactor (core, reflector, vessel, and the shield itself). In this encyclopedia the terms photon and gamma are used interchangeably, because lower energy photons do not generally travel far enough though their source material, nor carry enough energy to be of significant shielding concern (although in some cases hard X-rays, between 5 and 50 keV, can make a noticeable contribution). The highest gamma energy of significant quantity from a reactor is ~ 20 MeV, which is a long way from the highest class of photon, referred to as very high energy (VHE) gammas (> 100 GeV). Charged particles, notably beta, alpha, and fission products do not escape from the reactor at a high enough rate to present a shielding concern; although most of the actual damage from high energy photons is caused by the electrons it dislodges (thus the term ionizing radiation). Neutrons are not directly ionizing, but significant ionization does result from most neutron interactions. For a space reactor, high energy charged particles and VHEs can be of concern due to galactic cosmic radiation (GCR), solar flares, and within the radiation in the belts of Earth, Jupiter, etc. For missions in a high background radiation environment, e.g., Jupiter, the shielding solution must balance reactor and background effects.

The negative effects of neutron and gamma radiation are usually binned into two categories: cumulative damage effects and single event effects (SEEs). As the name implies, cumulative effects depend on how much radiation dose a component receives over time. The performance of the component (processor, memory, insulator, etc.) might degrade gradually with time until it is partially- or non-functional. Single effects are generally caused by the direct interaction of a single high energy particle with a component, where the resulting flood of electrons causes a failure. A common SEE is a single event upset (SEU), in which a change of state (e.g., flipping of a bit) occurs in a component. Damage from an SEU is usually not permanent to the component, but the temporary failure of the component could cause a system to fail (e.g., cause a faulty result, lock up a computer, etc.). The net effect of an SEU usually depends on the robustness and/or redundancy of an architecture, and the importance of the specific electronic feature that is affected. Other SEEs can have destructive effects, like single event burnout (SEB), single event latchup (SEL), etc. Since a SEE is a random 1-time event, the probability is linear with the radiation flux or dose rate. All of the damage phenomena tend to increase significantly with particle incident energy, and some phenomena have threshold values below which radiation will not cause a problem. Over the years, SEEs have become an increasingly significant issue as the feature size of electronics have gotten smaller, progressing from the mm range to the nm range. As the size gets smaller, the critical charge (i.e., number of electrons/ions that can cause a fault) continues to decrease.

Neutron dose limits: The cumulative damage from neutrons depends on many factors. In most materials, the most significant damage is displacement damage, in which a neutron knocks an atom of its nominal lattice position. This can change the electrical

properties of a semiconductor, the strength and ductility of a metal, the swelling of a ceramic, the magnetic properties of a magnet, etc. The potential for damage is usually defined as a function of displacements-per-atom (dpa). For most materials, dpa is a function of neutron fluence (cumulative number of neutrons crossing a unit area); in general, most dose limits are expressed as a function of neutron fluence. The displacement damage potential is a strong function of neutron energy, and in many cases the damage is expressed as a function of “1 MeV equivalent neutrons” or “> 100 keV neutrons.” Since fluence is equal to flux accumulation over time (n/cm^2), and flux equals number density, n , times velocity, v , then fluence is sometimes referred as neutron-density velocity time (nvt). In most cases when nvt is used with respect to irradiation damage, it implies fast fluence, such that nvt implies the fluence of 1 MeV equivalent neutrons.

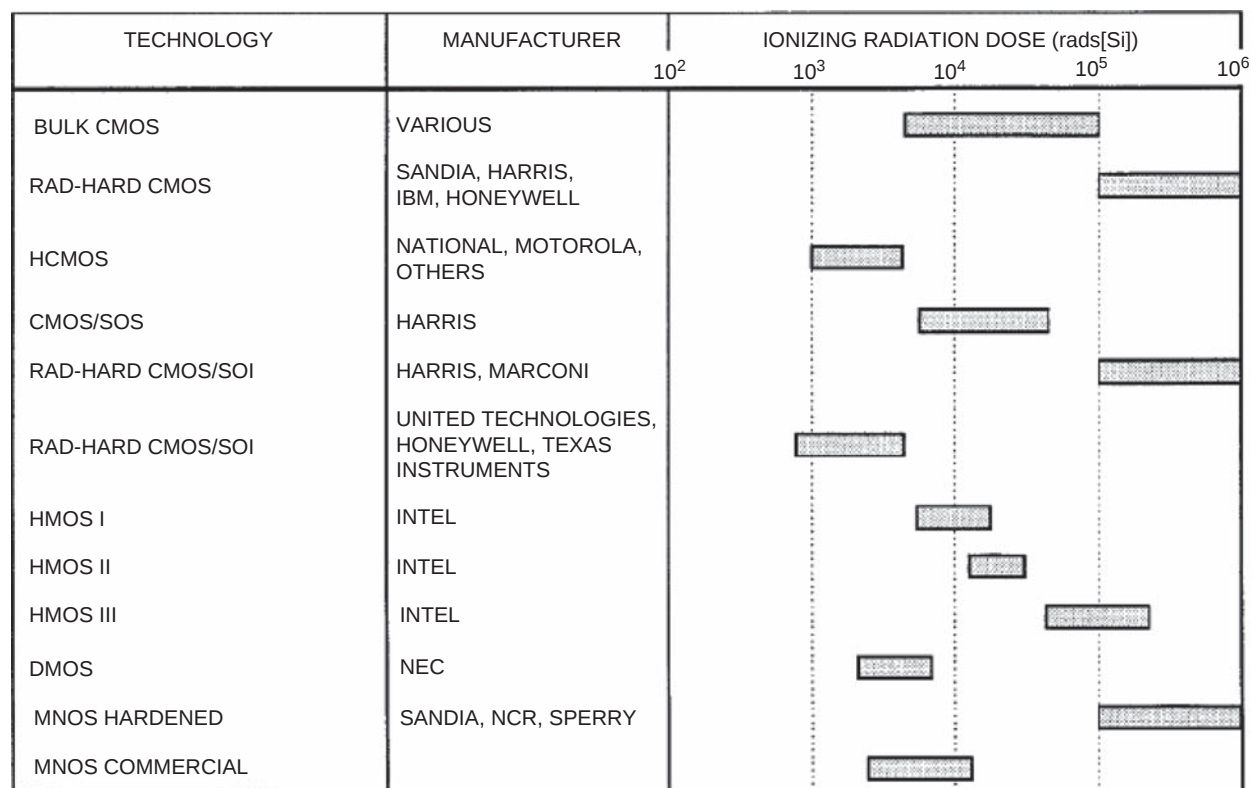
The damage tolerance of components range from $1e10$ nvt (bipolar silicon electronic components) to $> 1e23$ nvt (structural steels). Most off-the-shelf computers might tolerate between $1e10$ and $1e11$ nvt, but standard rad hard components should allow space reactor computers up to $\sim 1e12$ nvt without excessive cost, mass or power budgets. Tolerance up to $1e13$ nvt should be possible, but may be expensive, heavy, and/or power intensive. Another concern is the damage neutron fluence can have on reactor components like bearings, magnets, epoxies, coatings, etc. Magnetic integrity is of great importance for most space reactors because of the motors that move control elements and the alternators that produce electricity. Commonly used magnets might incur significant loss of performance (remanence) at fluences in the range of $1e14$ – $1e15$ nvt. Bearings for the control elements might be effected, either by changes in surface roughness and/or greases/oils (if used). Organic components, perhaps resins or epoxies, can also be intolerant to neutron radiation. Note that all the numbers listed above are general estimates, and can depend significantly on temperature or the details of the neutron spectrum.

The ionizing effect of neutron collisions also provides some damage, but generally on a smaller scale than gammas, because only a fraction of the neutron energy might lead to ionization, while nearly 100% of the gamma energy is ionizing. The other significant source of neutron irradiation damage potential is caused by neutron induced transmutations (the change of one isotope into another, or into a pair of products). The most common issue is the production of a gas after a neutron absorption, and this gas can lead to swelling, loss of ductility, cracking, etc. Many potential space reactor materials have the potential to create gases: Li, Be, B, C, Ni. Gas production is sometimes a serious issue within the shield itself, either $^3H + He$ from 6Li or He from ^{10}B . Neutrons can also dissociate the H from LiH, which is a flux issue rather than a fluence issue. In the case of LiH, potential radiation issues sometimes lead to the requirement to “shield the shield,” i.e., place a more robust shielding material, like Be, on the leading edge of the LiH to reduce its dose. Nickel-containing alloys are known to undergo helium embrittlement under very high fluences, but that is an in-core issue, not a shielding-related issue.

All of the above discussion is meant to give a flavor of the type of issues that might require neutron shielding. Any space reactor system design will require a detailed evaluation on a component-by-component basis to determine the appropriate radiation limits. In many cases this will require in-pile irradiation testing. Unfortunately there is very little neutron irradiation data for non-nuclear materials, especially for space rated electronics. This is because the space environment has very few neutrons, so relatively few customers or vendors have taken time to evaluate neutron tolerance (as compared to photons and charged particles).

Gamma dose limits: The cumulative dose damage from gammas is much easier to characterize than for neutrons; except for VHE gammas, which as mentioned previously are not a reactor shielding concern. The ionizing damage from gammas, and X-rays, is essentially proportional to the total energy deposited by a photon. Damage thresholds for most materials and components are expressed as a function of Rad ($1e-5$ J/g), or sometimes Gray (J/kg). Every material is different in how it interacts with photons. Energy deposition from radiation to matter is defined by Kerma (kinetic energy released per unit mass); every material has a different set of Kerma conversion factors. The difference in Kerma among materials is not very large for fission gammas (averaging few MeV), but can vary significantly at other energies depending on atomic number (i.e., number of electrons per atom). Most often dose limits are expressed in Rad(Si), because Si is the most common material of interest; even components with little or no Si might have their dose limits expressed as Rad(Si) by convention.

The bulk of gamma dose is ionizing dose, which causes gradual degradation of material performance. The damage threshold for even the most sensitive parts is generally ~ 10 kRad, thus accumulated gamma doses < 10 kRad are not usually of concern. Sensitive, non-rad-hard electronics might start to lose functionality at a dose of ~ 25 kRad, although there is a chance that limit could drop even further as the feature size of modern electronics continues to drop. However, there are many COTS (commercially off-the-shelf) electronics available that allow > 100 kRad, and up to 1 MRad. The figure below displays the total ionizing dose threshold (i.e., where damage can begin to occur) for various components. More information and the descriptions of components can be found in the reference from which the figure was obtained ([Messenger and Ash, 1992](#)).



In general, the use of rad hard components will save shield mass, but add cost, diminish capability, require more power, and/or require more cooling. The “good news” about component gamma/ionizing dose is that the limits are much easier to quantify than for neutron dose, both because of existing data, and the relative ease of obtaining new data via a gamma source, versus a neutron source or reactor.

Dose rate limits

The neutron and gamma dose limits discussed above are cumulative, and effectively independent of dose rate. There are other cases where damage or failures are a function of dose rate: this is true both for electronics and human beings. One way that dose rate can impact component degradation/failure is if the materials have a self-repair or annealing mechanism. For example, some metals will anneal out displacement damage at a sufficiently high temperature. If the rate of self-repair (often a function of temperature) is similar to the rate of damage caused by a specific dose, then the component can survive to a higher total dose, and perhaps indefinitely with respect to that specific dose.

The most significant impact of dose rate might be the effect of SEEs on electronic components and functionality. Some SEEs can cause permanent physical damage, but others cause temporary disruptions which may or may not cause a system failure depending on the rate. SEUs are stochastic and generally a linear function of flux or dose rate. As such, the same number of total SEUs might be expected for a flux of $1\text{e}3 \text{ n cm}^{-2} \text{ s}^{-1}$ for 1000 s as compared to $1\text{e}6 \text{ n cm}^{-2} \text{ s}^{-1}$ for 1 s; the difference between these two scenarios might be how the logic circuitry handles upsets. Most circuits have some ability to tolerate individual bit failures, using error detection and correction logic. The problem with high dose rate is that several bit failures might occur over a short period of time, which could overwhelm the components/circuitry and cause a system failure. This failure might cause a shutdown and reboot, or could permanently lock the system.

Human dose limits

The effects of radiation on humans is discussed in other chapters in this encyclopedia (Dewji and Miller, 2021; Kocher, 2021). There are two primary health risks to humans from radiation.

Radiation sickness: this occurs after a very high dose ($> 100 \text{ Rem}$) over a short period of time. Fatality is associated with dose $> 500 \text{ Rem}$ over short period of time. This can only occur if someone was standing close to an insufficiently shielded reactor during powered operation.

Latent cancer: this is harder to quantify and is rather controversial. Most US policies recognize the highly conservative Linear-No-Threshold (LNT) model, which assumes that any dose (no matter how small) incrementally increases one's chance of later developing cancer. This risk is considered a cumulative effect, and is not a function of dose rate.

There are other potential risks including eye damage, reproductive issues, etc., but the two above generally define human dose limits, and the latter risks are used on a case-by-case basis if they are pertinent.

In the nuclear industry, the ALARA principle (As Low As Reasonably Achievable) is generally the human dose standard. A limit of 5 Rem/year for a nuclear worker was established as a reasonable balance between productivity and risk, although there is no significant evidence that 5 Rem/year provides any risk at all. In space, a 5 Rem/year limit would preclude stays on the space station, so NASA has used ALARA to balance productivity and risk with a much higher dose guideline of 50 Rem/year. NASA career limits are based on age and sex, from 100 Rem for a 25 year-old female to 400 Rem for a 55 year-old male. These dose limits are based upon a maximum 3% lifetime excess risk of cancer mortality, which is a relatively small risk compared to the numerous risks faced by an astronaut, especially on a potential Mars mission. The total natural dose from a Mars mission is expected to be ~120–150 Rem over 3 years, which is acceptable, but even if a mission scenario exceeds the current NASA limits, the dose allowance could be increased based on a benefit/risk basis.

If a reactor is used on a human mission, whether for propulsion or power, the dose from that reactor should be weighed against the improvements it provides to mission capability. In some cases a reactor can also provide enhanced safety; e.g., if a Mars surface reactor provides more reliable life-support than a solar array (perhaps due to dust storm concerns), or a nuclear electric propulsion (NEP) system helps reduce travel time.

The reactor radiation dose limit should depend on benefit/risk, but at a minimum the reactor limit should be 5 Rem/year if it significantly enhances the mission. 5 Rem/year is the level accepted for personnel on Earth, and more-so has been proven to be safe (health statistics of nuclear workers receiving 5 Rem/year or less have not shown any negative effects). If the reactor is truly enabling or provides increased astronaut safety, then an acceptable additional dose from the reactor might be 10s of Rem/year.

There are two primary points:

- 1) There are no hard dose thresholds or limits that should apply across the board for all astronaut missions. NASA currently has a 50 Rem/year guideline, but going higher or lower will not significantly change the integrated mission risk for an astronaut that leaves Earth orbit (because numerous morbid mission risks will likely far exceed the additional risk of developing cancer later in life).
- 2) Radiation dose should not be considered a zero-sum game. If a reactor provides a dose of 5 or 10 Rem/year, the decision to use the reactor should be independent of the expected natural radiation dose for the mission. The increase in radiation risk caused by the reactor is the same regardless, i.e., if the reactor provides 5 Rem/year, there is no difference in added risk going from 10 to 15 Rem/year or from 50 to 55 Rem/year. The decision to use the reactor should be on its own merits, i.e., does the risk from the reactor radiation outweigh the benefits and increased capabilities provided by the reactor.

Shielding material options

Any material can generally be used to shield neutron and gamma radiation, but the selection of space reactor shielding is limited by several factors: including the desire for low mass, the difficulty of thermal management, the lack of gravity, the ability to survive launch/landing loads, etc.

Neutron shield materials

The primary function of neutron shielding is to scatter neutrons. The scatter of a neutron is a win-win from a shielding perspective. The change of direction caused by a scatter will usually decrease the probability of the neutron reaching the shielded area. The slowing down (decrease in energy) caused by a scatter will almost always reduce the damage caused by the neutron if it reaches the shielded area. Ironically, the ultimate absorption of the neutron is not a neutron shielding issue, but rather a gamma shielding issue, if the capture results in one or more gammas.

Most elements/isotopes are similar in terms of how they interact with fast neutrons, with heavier elements having a slightly higher interaction probability (cross section); however, the best neutron shields are low atomic mass elements because they are more effective at scattering neutrons per unit mass, plus they slow down (decrease energy of) neutrons more per scatter. The latter effect generally makes hydrogen the preferred element for neutron scatter, if a hydrogenous material can handle the physical, thermal, and irradiation requirements of the intended application.

Lithium hydride

Lithium hydride is often the optimal space reactor shield, because it is a very low mass-density hydrogenous material. ${}^6\text{Li}$ is also a strong neutron absorber, which is desirable in a hydrogenous material to reduce capture gammas. Hydrogen neutron capture produces a 2.2 MeV gamma, which then creates a shielding issue in itself. ${}^6\text{Li}$ produces minimal capture gammas, and it preferentially captures neutrons over hydrogen. Naturally enriched Li contains 7.5% ${}^6\text{Li}$, and the fraction of captures within the Li will

depend on the neutron energy spectrum. The fraction can be increased by raising the ${}^6\text{Li}$ enrichment level; this not only reduces H capture gammas but also decreases mass because ${}^6\text{Li}$ is 6/7 the mass of ${}^7\text{Li}$ (which represents a significant shield mass savings).

Unfortunately there are several engineering issues associated with the use of LiH. The first complication results from the benefit described above; i.e., instead of producing capture gammas, ${}^6\text{Li}$ produces He and tritium gas when it captures a neutron. This causes the LiH to swell under irradiation, with significant swelling occurring at an ionizing dose of ~ 500 Mrad (Wang et al., 2016). However, swelling does not occur if the temperature is high enough to allow the gases to diffuse out of the material. Testing indicates that significant irradiation swelling does not occur at $> \sim 600$ K, although this will be dependent on can internal pressure and how the LiH is manufactured.

The other significant issue involved with LiH is potential H loss. Hydrogen dissociation pressure is very low at ~ 600 K, but is rather high above 800 K. In addition, H moves through SS and potential coatings much easier as temperature increases. Therefore, hydrogen loss from LiH generally starts to become significant at 800 K, becomes problematic about 900 K, and likely a showstopper at 1000 K; although with coatings or thick walls the loss rate might be mitigated with a properly engineered clad/containment.

In low dose regions ($< \sim 500$ Mrad), LiH can be used at low temperature without much concern. At high doses, LiH should only be used if its temperature can be maintained between ~ 600 K and ~ 800 K. This is a very difficult engineering challenge, especially because LiH has poor thermal conductivity (and potential for cracks). Power deposition is sharply higher in the regions near the core, and the inability to spread and reject the heat can be problematic. If LiH is used in a high dose region (resulting in high internal power deposition) then a design may need conductive plates, rods, wires mesh, etc. interspersed with the LiH to help remove heat. Another factor that complicates design is that LiH has a thermal expansion coefficient twice that of SS, which is the most likely can material. Also, there is relatively little experience and infrastructure for manufacturing LiH. Despite all of these challenges, the mass benefit of LiH is substantial enough to warrant its selection in many cases.

Boron carbide

Boron carbide (B_4C) is a well-established nuclear material; it has been utilized extensively as a neutron absorber (for control and shielding) in almost every reactor ever operated. The biggest advantages of B_4C are that (1) it is a high temperature material, (2) material performance is well characterized in extreme temperature and fluence regimes, and (3) there is an existing infrastructure to process and fabricate B_4C , either as solid components or as a powder. Boron, specifically ${}^{10}\text{B}$ is a nice neutron shielding material because it emits a rather low energy gamma (478 keV) when it absorbs a neutron, as compared to the 2.2 MeV hydrogen capture gamma. The other nice attribute of ${}^{10}\text{B}$ is that it is 9% lighter than ${}^{11}\text{B}$, so highly enriched boron is preferable, unless cost or availability dictates otherwise.

While B_4C does not have the upper temperature limit of LiH, it does have the potential to swell due to He production associated with ${}^{10}\text{B}$ neutron capture. One option to reduce swelling is to use low theoretical density B_4C parts. Another option is to use a fine powder, such that most of the He could escape the material and thus not cause swelling. A shield where B_4C powder is placed in a can (perhaps made of SS) is probably the easiest and lower cost option to fabricate. State-of-the-art techniques might allow packing fractions from 75% to 85%. The biggest drawback of a powder shield is very low thermal conductivity. The concern is not that the B_4C itself will get too hot, nor the SS can (except for extreme examples); the issue may be the difficulty of transporting power through the B_4C shield. Some reactor concepts require power to escape through the shield for thermal balance or decay heat removal reasons. Another risk of a powder shield is the potential for the material to sinter and then create localized stresses in the can during thermal cycles.

The primary drawback of B_4C is that it results in a significantly heavier neutron shield than LiH. The overall mass penalty of using B_4C largely depends on how it helps reduce gamma shield mass; B_4C not only shields gammas better than LiH, but also produces much lower energy capture gammas. However, in most cases this gamma shielding benefit does not come close to offsetting the mass penalty of B_4C as a neutron shield. A B_4C shield is generally only desirable if the potential for high temperature or high dose makes the use of LiH too high risk (the risk of losing hydrogen from the material), or the system design can benefit from the better structural properties of B_4C . In some cases, B_4C is best used as a leading-edge shield to reduce dose and power deposition to acceptable levels for a subsequent layer of hydrogenous material. It is unlikely that the lower cost or shorter procurement time of B_4C would be a major advantage for a flight reactor program.

Water

Water is potentially an attractive neutron shield material for surface power missions, especially if in-situ water can be used in future architectures. Water is not a good shielding option for zero-gravity (zero-g) space missions because the absence of gravity creates uncertainty and variability as to where voids may exist, plus there's no driving force (buoyancy) to convect heat within the shield (the thermal conductivity of water is very poor). It might be possible to use surface tension and capillary forces to mitigate these issues, but the benefits of water shielding would likely not justify the complications for shielding in zero-g.

Water is very effective at shielding neutrons, similar to LiH. The major advantages of using water shielding are design simplicity and cost. Prototyping and testing can proceed immediately and at low cost, which translates to rapid, inexpensive development, and low-cost flight units. There is a wealth of fabrication and operating experience with SS or Ti and water, and a significant irradiation database. A water shield is also very easy to test. Radiation transport can be tested with portable sources or in research reactors, although the nuclear cross sections of water are very well understood and in a gravity field there is no opportunity for streaming paths (dependent on the design). Thermal-structural performance can be tested inexpensively with electric heating to verify heat transfer, natural convection, and structural integrity (although the difference in gravity has to be accounted for). Also, the ease

of testing allows several potential configurations to be tested, which can be used to better optimize the shield and power system design. One other advantage of a water shield is the potential to use borated water to cut down hydrogen capture gammas, which can provide a modest reduction in shield mass if there are no significant chemistry or pressure issues. The shielding effect of ^{10}B in a water shield is investigated in Poston.

One of the limiting issues in using a water shield is temperature. The vapor pressure of water rises dramatically as it heats up to $\sim 400\text{ K}$, and at temperatures $> 450\text{ K}$ the thickness of the pressure vessel could make the shield mass excessive. In space, the low thermal conductivity of water would make it very difficult to keep the temperature low, even with small power deposition and/or throughput. In a gravitational environment, natural convection should be able to keep the ΔT through the shield reasonably small, depending on shield geometry and where the heat is added and rejected. Natural convection could also provide a significant system-wide thermal management benefit in configurations that require heat removal from the core/reflector assembly.

Another concern with a water shield is the impact of radiolysis (radiation induced breakdown that can lead to corrosion). Water chemistry is a major issue in all terrestrial water-cooled reactors. A water shield would probably not make sense for a space/surface system if an active chemistry system were required. Fortunately, the expected temperature and irradiation of a surface reactor water shield should be lower than the coolant in most terrestrial reactors. Titanium might be preferred as the shield vessel/can, because it is less chemically active with water (and radiolytic components) than SS. Freezing of the shield may also be a concern, but it is possible that water in a mostly wide-open vessel will tolerate freeze-thaw without any special attention (depending on how many penetrations there are through the shield, the overall geometry, the space available for expansion, and the number of thermal cycles). If freeze-thaw is found to be a design issue, it may not be too difficult to keep the water from freezing, especially after initial operation if there is some decay power being transferred to the shield. Expansion volume will also have to be available for the heated water. Depending on the peak design-basis temperature, it could be that the expansion volume required for freezing is the same as the expansion volume required for heating. Another issue with a water shield is the potential for leaks. If it is determined that a leak is a significant mission reliability issue, then different approaches (e.g., double wall or multi-compartment) would need to be investigated. There is a good chance that water will be used for system heat rejection and/or elsewhere in the system architecture, so the technical issues associated with water will need to be addressed regardless of the shield material.

Beryllium

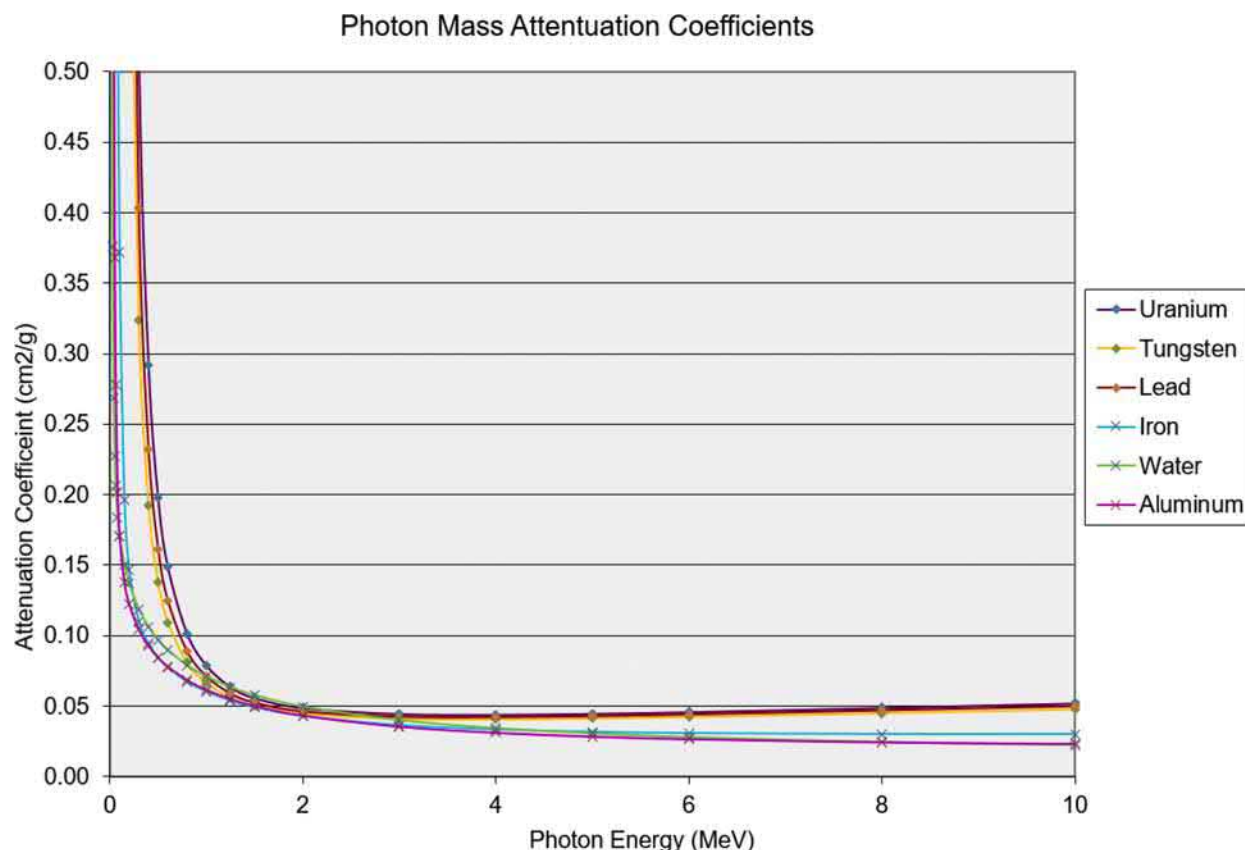
The high scattering effectiveness and low density of Be make it a decent neutron shield, although not nearly as effective as LiH or water, and generally not as effective as B_4C . However, Be has a niche for space reactor shielding, particularly as the “leading edge” of a high power reactor shield. The high temperature capability and thermal conductivity of Be allow it to tolerate and transport power deposited in a very high flux/dose regions. The high strength of Be can also allow it to serve as the structural foundation for the entire shield. A small additional benefit is that the Be will reflect more neutrons back to the core and therefore increase nominal reactivity; although this might be a disadvantage for a particular modular shield, because it might cause reactivity to be significantly different for one mission application versus another.

Other neutron shield materials

There are many potential hydrogenous materials that could be considered for neutron shielding, but they are generally less desirable. Polyethylene (Poly) is a commonly used as a neutron shield in nuclear facilities, but it is not well-suited for most space reactor applications due to thermal constraints; i.e., a long-life operational temperature limit of $\sim 350\text{ K}$, plus poor thermal conductivity, irradiation tolerance, and strength. Other plastics, rubbers, or hydrocarbons have similar issues. Conversely, there are higher temperature hydride materials, e.g., ZrH_x or YH_x , which could achieve higher temperatures than LiH, but they are much less effective at shielding neutrons on a mass basis. They could be considered over LiH if extra gamma shielding and/or a slightly higher temperature were desirable; although these compounds, particularly YH, are not desirable from a capture gamma perspective. In situ regolith (dirt and rock) certainly has great potential on a surface mission, and this is discussed in depth later.

Gamma shield materials

Gamma neutron shielding is usually considered to be a high-Z material such as tungsten, lead, or depleted uranium, but any material can be effective. The majority of reactor gamma dose is often in an energy range where all materials have a similar mass attenuation coefficient; fission gamma dose is highest in the 1–5 MeV range. Shielding is mostly determined by the density-times-thickness ($\text{g}\cdot\text{cm}^{-2}$) of any material between the reactor and the shielded location. As a result, any “free” material (spacecraft components, structure, regolith, etc.) that can be placed between the reactor and the shielded location can be of great benefit. In some cases the advantage of high-Z materials is mostly that they can provide the more mass in a compact volume. However, there are other scenarios where a high-Z material provides a significant benefit. Any shield that contains significant ^{10}B will have a large amount of 478 keV gammas; high-Z materials have a much higher attenuation coefficient at this energy. High-Z materials are also more effective at shielding very high energy neutrons, $> 5\text{ MeV}$, but there are relatively few fission neutrons of that high energy; ironically most of the very high energy gammas come from capture in high-Z materials. The figure below shows the mass attenuation coefficient for potential shielding materials.



In many cases, capture gammas represent more of a shielding issue than fission gammas, because neutrons tend to penetrate further into the shield, and then create a high energy gamma near the back (downstream) end of the shield. In shielding terminology the ratio of dose from capture gammas to fission radiation is called the “buildup factor.” As discussed previously, it is desirable to have a neutron shield with strong neutron absorption, preferably an absorption that does not produce a high energy gamma. High neutron absorption is also important because it prevents capture gammas from being produced in the gamma shielding. If the neutron shield does not contain a strong neutron absorber, it is possible to place a small layer of $1/v$ absorber in front of (upstream) of the gammas shielding. A small layer (~ 1 mm) of B_4C can be very effective in reducing high energy capture gammas. Another option would be to use a layer of borated steel, which could also serve as a structural component and a heat removal path.

Shield design considerations

Shield design is not simply a neutronic exercise, nor should it be performed without being integrated with reactor and system design. The shield is often the structural foundation of the reactor power system, as well as the structural interface to the spacecraft. In those cases the shield should be a very robust component, in terms of surviving temperatures, stresses, fluences, etc. Thermal management can be a significant design issue, and will often drive the material selection as noted above. Thermal expansion must be considered with respect to all connections to the core, control drive motors and shafts, power conversion, etc. There also must be penetrations or slots for instrumentation cabling and coolant flow (or heat pipes). All of this must be done while creating minimal streaming paths for radiation through the shield. Finally, the shield must be practical from a fabrication, test, and cost perspective.

The relative level and balance of neutron and gamma shielding is highly dependent on mission and component requirements, the surrounding environment (e.g., space or surface), power level, spatial separation, etc. Some reactors require more neutron shield mass, while others might require more gamma shield mass. As mentioned previously, no material is distinctly a neutron or gamma shield. These terms are used to define the primary function; i.e., a neutron shield’s primary purpose is to shield neutrons and vice-versa.

As a rule of thumb, neutron flux needs to be reduced 2 decades (orders of magnitude) more than gamma flux from the reactor to a shielded area; this is component dependent, but is generally true for most materials as well as humans. This difference also depends on reactor spectrum; a fast spectrum reactor will emit a relatively higher ratio of fast neutrons to gammas, while a moderated reactor will emit a relatively higher ratio of gammas to fast neutrons (due to the differences in mass density and overall fast neutron flux).

Another generality is that it requires significantly more mass to reduce gamma dose by a decade (or any amount) than it takes to reduce fast neutron flux by the same amount. Very rough rules of thumb can be applied to common neutron and gamma shielding materials: ~ 4 cm of LiH/H₂O can reduce fast neutron flux by a factor of 2, while ~ 1 cm of W/U can reduce gamma dose by a factor of 2. One centimeter of W or U is $\sim 5 \times$ heavier than 4 cm of LiH or H₂O, so in this example gamma shielding would require $\sim 5 \times$ more mass than neutron shielding.

The combined effect of the generalities above is that reactors with very low shielding requirements (perhaps only to reduce neutrons by 3 decades and gammas by 1 decade) require almost no gamma shielding mass; in fact the neutron shield might provide enough gamma shielding in itself. Whereas a reactor with very high shielding requirements (perhaps to reduce neutrons by 8 decades and gammas 6 decades) will not only be thick and heavy, but most of the shield mass will be gamma mass.

As mentioned previously, no material is purely a neutron or gamma shield, which leads to a few other rules of thumb. LiH/H₂O provide $1 \times$ gamma shielding for every $5 \times$ of neutron shielding, and W/U provide $1 \times$ neutron shielding for every $5 \times$ of gamma shielding. Neutron shield materials like B₄C and Be are actually more like hybrid materials, because they can provide almost as much gamma shielding as neutron shielding. Of course, all of the rules of thumb provided above are highly generalized, and are dependent on every other aspect of shield effectiveness: including the level of streaming and scattering, the reactor spectrum, material density and composition, where this particular thickness is utilized within the shield, etc.

Relative shield material placement

There are several factors that determine the optimal placing of various neutron and gamma shielding materials relative to each other. Buildup factor, or the production of capture gammas, is often the most important of these design concerns. In this case, it is desirable to place most of the neutron shielding in front (upstream) of gamma shielding. However, neutron shielding is relatively thick, and the more that is used upstream the further the gamma shield moves back in the geometry. This creates a mass concern, because any realistic shielding geometry will have a radial component, such that size increases with further separation from the reactor. The two predominant examples of radial mass increase are moving backward in a shield cone or moving outward in a cylindrical or spherical surface shield component. Another factor that guides relative placement are the dose requirements of the shield materials. If LiH is used, it is often beneficial to place another material in front of the LiH to prevent swelling and high power deposition; this material could be a gamma shielding, or a most robust neutron shield such as Be or B₄C. Another placement consideration is heat removal and structure, which often leads to a layered shielding approach. Each shield layer might have ~ 10 cm of neutron shielding, followed by a ~ 1 cm of gamma shielding. This helps reduce capture gammas while also removing internal power generation, plus the downstream layers can sometime effectively shield the capture gammas generated upstream. All of this is application dependent, and must take into account the hybrid effect of each material (i.e., that all materials shield both gammas and neutrons to some extent).

Spot shielding

The term “shield” generally refers to the bulk shielding close to the reactor, which serves to reduce the radiation environment over a large volume. In some applications, there will be a component that is much more sensitive to radiation than anything nearby. In this case a small shield, or spot shield can be placed directly in front, or upstream of this component. Spot shielding works very well for gammas because they do not scatter very effectively; most dose comes the source point where the gamma was generated. The vast majority of gammas will originate in the reactor or shield (either by fission or capture), so they will all travel from a similar upstream direction. In addition, gammas can be shielded with thin layers of material (cm), thus gamma spot shielding can fit in potentially tight geometries in the payload region.

Spot shielding is rarely effective for neutrons. Neutron flux or dose can come from a scatter with any component nearby, thus a single directional shield does not work as well. As a result, the shielding in front a component needs to be a bit wider or slightly wrap around the component (as opposed to shielding solely in the primary upstream direction). This not only adds mass, but it introduces the potential for counterproductive scattering back into the spot-shielded component or any component nearby. Neutron spot shielding is also difficult due to the need for thick shielding (~ 4 cm for a factor of two vs. ~ 1 cm for gammas). It can become heavier with increasing diameter (if there is a radial component to the spot shield), and a thicker shield further increases the potential for counterproductive scattering.

In-space reactor shielding

Shielding in space is relatively straightforward because of the ability to shadow shield the reactor radiation. There is no natural path for radiation to scatter from the surrounding air, ground, building, etc., therefore anything behind the shield (or in its shadow) is protected. The shadow is generally in a conical shape, where the smaller diameter of the cone is near the reactor and the cone expands in area beyond the shield. The shield cone angle is determined by the diameter of the payload/electronics region and the axial distance from the reactor to this region. The axial separation has a major effect on shield mass for two reasons. First, the greater the separation the smaller the cone angle, thus the smaller the shield diameter. Far more important is the decrease in radiation intensity with increased separation. Beyond a few meters of separation, the drop in radiation intensity approaches the

$1/r^2$ effect of distance from a point source. Note that the $1/r^2$ effect only applies to radiation which originated from the reactor, scattered neutrons or capture gammas from the shield will require even further separation to approach the $1/r^2$ effect.

Most space reactor concepts utilize a boom to separate the reactor from the payload (or any sensitive components). From a shielding perspective, the longer the boom the better, but there are many limiting factors. If a non-deployable boom is used, the separation distance is limited by the launch vehicle shroud, which is rarely optimal because it requires a very high mass shield. A deployable boom introduces several complications, including the unfurling of cables and potentially coolant lines that extend along the length of a spacecraft. The longer the boom, the more difficult it is to package everything in a small volume and successfully deploy it. The other concern is structural, because the moment of inertia increases with the axial distance squared; thus as length increases the boom mass per unit length will increase and spacecraft maneuverability will be hampered. Many historical concepts have considered deployable booms in the ~ 10 – 20 m range, but recent technology improvements might allow $\gg 20$ m booms, which could allow very low shield masses. Boom length is one of many variables to be optimized as part of an integrated shielding solution. Some other variables, as discussed previously, are mission length, component rad-hardness, power level, axial reflector length, and material selection.

In a shadow shield configuration, the primary purpose of neutron shielding is to scatter neutrons to a new trajectory that will not hit the shielded region: referred to as a “productive” scatter. Unfortunately, after a productive scatter has occurred, there is always a chance that a “counterproductive” scatter will occur; i.e., a subsequent scatter that retargets the neutron so that it will hit the shielded region: this is referred to as a “recoup-scatter.” The net effect of these two scatters is diminished to only the decrease in neutron energy. Recoup-scatter can be important in optimizing the shield, in particular around the perimeter on the far (downstream) end of the shield. Material in the back corner of a shield cone can cause enough recoup scatter to substantially offset any productive scatters that might occur, thus the mass of shielding in that area may not be justified. In some cases a portion of the shield geometry might actually increase dose, e.g., recoup-scatter dose outweighs the effect of productive scatter, so clearly mass should not be placed there. The potential for recoup scatter also makes segmented shielding less attractive; i.e., placing a void or gap between two or more neutron shields.

Another integrated design consideration is to optimize the component layout in front and behind the shield, to place as much mass between the reactor and sensitive components as possible. Concepts with an intermediate heat exchanger (HX) can easily gain a shielding benefit via HX placement. Brayton and Stirling components can be used to shield downstream components, but they also need to be positioned to minimize their own internal dose. First order, the goal is to use one component to create small shadow shield for others downstream. This works well for gammas, because it is not very effective for neutron shielding because of multi-directional and recoup-scatter. Also, if there are voids or gaps in a component or subsystem it might leave a streaming path for a hot spot in a downstream component; although any material will generally reduce the average flux/dose downstream.

Penetrations through the shield for the coolant pipes, control shafts, cables, etc. can be a major design issue due to radiation streaming. In many cases the gaps and insulation around the penetrating components are more of a streaming issue than the components themselves; depending on whether the pipes are empty (e.g., filled with gas/vapor instead of liquid metal), and how much insulation and fabrication/expansion tolerance (gaps) is needed between then components and shield. The latter is an important design consideration that balances shielding, mechanical design, thermal design, and potentially reliability. Features that create direct/straight streaming paths are of the greatest concern, so it is usually desirable to use staggers, kinks, bends, spirals, etc. to prevent a straight line-of-sight for radiation. However, any non-linear design feature will almost always add cost and complexity to any component, as well as system integration and assembly issues. Shield construction and integration will usually be much simpler with straight penetrations, and coolant tubes and heat pipes themselves will be simpler (although in some cases a bend in pipes is desirable to create a spring effect to accommodate differential thermal expansion). The magnitude of a streaming depends largely on the solid angle (path area divided by path length) and what the upstream end of the path “sees” (i.e., does the streaming path emerge from the core, does it view fuel, reflector, vessel, or space?). The significance of streaming depends substantially on how many orders of magnitude the shield reduces radiation. If the shield reduces gammas by only 2 decades, then a modest amount of streaming is of minimal consequence (except for the small region in the line-of-sight of the streaming path). If the shield reduces gammas by 6 decades, then even a small level of streaming can dominate dose globally and locally.

Any shadow shielded system will have to evaluate the pros and cons of placing some components outside of the shadowed area: e.g., radiator, coolant pipe, satellite dish, probe, robotic arm, etc. It may be easier to engineer a system with coolant pipes outside the perimeter of the shield, but that would have to be weighed against the added dose from radiation scatter from the pipes. There are many options of where to position and how the potentially coil pipes around the shield, or in grooves on the outside of the shield. The biggest potential design decision with respect to external scatter is the position and geometry of the radiator panels. If a large area of radiator is positioned outside of the shield cone angle, it is possible for the neutron dose scattered from the radiator to be higher than the dose coming through/from the shadow shield. The significance of external neutron scatter also depends greatly on how many decades of shielding is utilized.

Surface reactor shielding

Surface reactor shielding is quite different from a space power mission, where a shadow shield can be used. This is because scattering from the regolith and surrounding spacecraft components usually require some shielding around the entire reactor, commonly referred to as “4pi” shielding. A 4pi shield does not imply the same shield thickness in every direction, rather the optimal amount

in every direction to achieve a combined set of shielding goals. The shield will generally be thicker in the direction of components, electronics, and human areas. The ground or regolith is shielded to balance the neutron scatter than reaches all of those regions. If the shield design is non-cylindrical, e.g., thicker in the azimuthal direction of the human outpost, then it is sometimes referred to as a “shaped shield.” One of the biggest challenges of surface reactor design is balancing all requirements, which can include:

- Cumulative dose to power conversion and other reactor components.
- Power deposition to reactor components, spacecraft, and surroundings.
- Dose rate and cumulative dose to all electronics and sensitive components.
- Dose rate to infrastructure elements: e.g. rover charging, propellant production.
- Dose rate to human outpost.
- Activation dose to humans for near-reactor operations.

The need for 4pi shielding can make surface reactor shields very heavy, and they can dominate the system mass. Fortunately there are many potential options for using in situ resources as shielding materials, which can provide substantial mass savings. Regolith could be used in several ways: berming, sandbagging, digging/burying, using natural topography (e.g., craters or ridges), or any combination of these. In-situ resources could also be used to fill permanent structures or cans as a fixed part of the reactor structure (e.g., sand could be scooped or vacuumed), and in the best scenarios in situ water could fill shield tanks or perhaps concrete could be made. All shielding options that require in situ resources will depend on significant robotic ability (or existing robotic/human infrastructure) to successfully complete the mission. Many of these options could add significantly to mission risk, but the mass savings potential is so large that in situ options always warrant serious consideration. Any option that can pre-deploy and verify in situ shielding prior to human mission departure would have a significant advantage.

Regolith as shielding

Regolith is a blend of soil and rocky debris that covers the surface of the moon, Mars, and most asteroids or planetary bodies. On the Moon and Mars regolith is predominantly in the form of metallic-oxides. The composition of regolith, primarily oxygen and silicon, makes it a very effective at scattering neutrons, which makes shadow shielding impossible if regolith is nearby. If regolith is within a few meters of the reactor, e.g., the ground, then regolith scatter can dominate and otherwise well-shielded system; thus the need for 4pi shielding to reduce scatter from the ground. Regolith composition varies between the Moon and Mars, more so in different regions on the respective surfaces. The table below shows various regolith compositions ([Heiken et al., 1991a](#); [De Angelis et al., 2004](#)).

Regolith Mass Fractions				Regolith Atom Fractions			
	Lunar Mare	Lunar Highlands	Mars		Lunar Mare	Lunar Highlands	Mars
SiO ₂	45.34	44.65	49.16	O	60.57	60.94	62.65
Al ₂ O ₃	15.13	27.09	8.38	Si	16.93	16.12	17.96
FeO/Fe ₂ O ₃	13.47	5.06	18.44	Al	6.65	11.53	3.61
CaO	11.19	15.58	6.26	Ca	4.47	6.03	2.45
MgO	9.68	5.66	7.82	Mg	5.40	3.05	4.26
TiO ₂	3.62	0.54	1.23	Fe	4.20	1.53	5.07
Na ₂ O	0.50	0.46	2.35	Na	0.36	0.32	1.66
Cr ₂ O ₃	0.31	0.33		Ti	1.02	0.15	0.34
K ₂ O	0.27	0.17	0.34	Cr	0.09	0.09	
P ₂ O ₅	0.25	0.11		Mn	0.06	0.09	
MnO	0.18	0.30		K	0.13	0.08	0.16
SO ₃	0.05	0.07	5.47	S	0.04	0.05	1.50
Cl			0.56	P	0.08	0.03	
				Cl			0.35

The tables above show that the regolith on Mars and two different locations on the Moon are rather similar. From a neutronic perspective, the biggest difference is that Mars has more neutron absorbing material; thus it will normally not scatter back quite as

many neutrons, with a corresponding small increase in capture gammas. The average particle theoretical density is ~ 3.1 g/cc on Mars, ~ 3.2 g/cc in the Lunar Highlands, and ~ 3.4 g/cc in the Lunar Mare. The bulk material density will be much lower than these values depending on the packing fraction of the particles in the regolith. A good initial for the actual, undisturbed density of regolith (65% packing of 95% TD particles) might be ~ 2 g/cc. In most cases, the density very close to the surface will be slightly lower than the average; the moon has a surprisingly low-density layer on top. The natural packing fraction (density) will be location dependent, but in many cases the density will be affected by human activity; e.g., when containers are filled, holes are refilled, or the ground is compacted.

Overall, there is not too much difference from a shielding perspective between the regolith on the Moon and Mars, but there are some environmental differences. The atmosphere on Mars creates an interesting, and sometime negative scattering effect. Topography can play a major role in the shielding solution, and in some cases the spherical curvature of the bodies can also play a role. These impact of environmental differences is discussed further in the sections below.

Buried reactors

Burying the reactor within the regolith is generally the lowest mass shielding solution for any application that requires a permanent human presence nearby. The mission advantages of a buried reactor are substantial, including the potential to place the outpost and equipment < 100 m from the power system and to approach the reactor for maintenance/repair with minimal radiation risk. Both of these advantages depend on the hole depth. Any depth that covers part or all of the core is very valuable, and 1 or 2 m of depth is all that is needed to provide most of the advantages. If the reactor is buried several meters deep, then it may be possible to allow human activity above the reactor even while operating at full power.

Another small advantage of a buried reactor is that it can be left in place after operation, either after its nominal operation or a potential failure. Although, this advantage is rather small because it is unlikely that any above-surface reactor would have to be moved/buried as part of decommissioning. The dose surrounding a space reactor will become insignificant months after shut-down, so the only risk is release of radioactive material into the environment. Even then, the potential radiation release will be orders of magnitude lower than that from terrestrial power plants (due to lower power), so a potential exclusion after a release zone might be tens of meters instead of tens of kilometers. More importantly, environmental contamination will be much less of a negative on the Moon or Mars because no one will be breathing the air; contamination of various regions would have to be considered, but will likely be negligible. Therefore, there should be minimal risk of leaving an above-surface reactor in place indefinitely.

The difficulty of buried reactors is that they require considerable infrastructure. The capability is required to dig a hole, remove, transport, and position the reactor in the hole, potentially refill the hole, and if possible compact the refilled regolith. These actions need to be performed reliably while also not damaging the power system. Expediency might also be an issue if the operation requires installation during a lunar day or a fixed amount of battery power; although if the infrastructure is available to bury the reactor another power system will likely already be in place.

The key to successfully burying a reactor will be to dig in a region with sufficiently deep loose regolith; i.e., where the depth above the bedrock (or whatever hard surface exists) is greater than the intended hole. A study by Wilcox et al. (2005) investigated several regions, and noted that lunar regolith depth varied between 2 cm and 40 m, with significant local variability. Heiken et al. (1991b) states that the regolith is generally about 4–5 m thick in mare regions and 10–15 m in highland areas, which provides plenty of margin for digging a hole. Also, more information on lunar and Mars regolith composition will certainly be obtained in the relatively near term. Hopefully, before a buried reactor is deployed, there will be enough infrastructure to scout out appropriate sites before digging begins.

There are several possibilities for the geometry of the hole in which the reactor is buried (e.g., trench, slot, hole, etc.). Each technique has pros and cons from an architecture and energy perspective, but from a shielding it is preferable to leave as regolith next to the reactor undisturbed. Refilled regolith will have significantly lower density, depending on how much effort is used to try and compact it, which will allow more neutrons to leak to the surface and potentially create dose to any region. Each method of creating a hole will have a different profile of undisturbed regolith and refill density, and more importantly potential hole depth for a specific region and application. In many respects, auguring a hole might be a preferable option, because it leaves surrounding regolith undisturbed, creates a small berm with the expelled regolith, and might be the easiest tool to use (assuming enough power can be provided). It is also possible to consider burying a reactor in a horizontal configuration, if a trench is easy to dig and refill, but this generally adds significant complication to system design.

There is one other significant engineering issue of a buried reactor—thermal balance. Regolith is a very poor heat conductor, both the rocks/particles themselves and more importantly the gaps between them. The conductivity is even lower on the moon because of the lack of atmosphere to facilitate heat transfer. This is a problem because if environmental heat removal is unavailable, then the system design cannot take advantage of the environment to reject secondary heat (from reflector, shield, components, and potentially decay heat). In this case secondary heat would have to be rejected either by the primary heat removal system or an additional dedicated one. If the primary coolant system is used, then all of the ex-core components must run hotter than the reactor coolant. If a dedicated system (sometime referred to as a cavity cooling system) is used, then the peripheral components can run cooler, but an entirely new subsystem must be engineered, assembled, and installed.

Ironically, the biggest thermal concern can be the temperature of the regolith itself. The idea of a buried reactor is to minimize shield mass, which means placing regolith as close as practical to the reactor. In some cases this can cause the power deposition in

the regolith to be so high that it cannot be conducted out (either back to the reactor cooling system or to the surface). The temperature limit of the regolith will depend on many factors, but a reasonable assumption might be ~ 1000 K, or a temperature close to what the reflector and shield would operate. This design scenario would probably call for a high temperature reactor shield (B_4C or Be) as opposed to LiH or H_2O . When the regolith gets hot, it may be that the regolith will sinter, and thus increase the conductivity to a level where overheating would not occur. Although it would be hard to verify if and when sintering might occur, plus the sintered material might crack, which would once again reduce conductivity and perhaps leave radiation streaming paths to the surface. Also, very hot regolith might off-gas materials that could attack the reactor system. If the regolith gets too hot it will lose structural and/or shielding integrity and ultimately melt. In cases where the power/flux is high enough to overheat regolith, there are two options. The first is to include additional shielding in the reactor design, i.e., use reactor shield material to shield the regolith. The other option is to include even another heat removal system; i.e., one that is placed in the regolith to transport power from the regolith to the surface. The most difficult place to keep regolith cool might be the lower radial region surrounding the core, or perhaps below the reactor, depending on local power density and heat transfer paths.

Shielding the axial region above the reactor in a buried configuration involves a lot of design trades. It might be desirable to save additional mass by inserting regolith into the gaps between the reactor and power conversion system, but this may introduce significant thermal and shielding uncertainties. If the hole is deep enough, it might be desirable to leave an open cavity in part of this axial region above the core, to minimize launched mass and/or simplify thermal management, and use a “plug” shield at the top of the hole. Finally, if a horizontal configuration is considered, as opposed to vertical (assuming the reactor is roughly cylindrical), the problem of thermal management becomes even more difficult.

Above-surface reactors

Early surface reactor missions may not have the ability to bury a reactor. Initial reactor missions will likely be needed to provide power for the building of the required infrastructure, or perhaps early missions will be purely robotic missions or technology demonstrations. The brute-force solution is to launch every kilogram of shielding from Earth, which is practical for a small robotic mission, but undesirable for a human mission. However, there are still plenty of ways that an above-surface reactor can utilize regolith and topography to substantially reduce system mass.

From a reactor perspective, the easiest way to provide in-situ shielding is to use topography. The reactor could be landed in a crater or on the other side of a ridge from the outpost location. This method is becoming increasingly attractive as the technology improves for precision spacecraft landing, as well as the ability to create detailed mapping of potential reactor and outpost sites. The drawback of this approach include the potential difficulty to route/connect the electrical cabling, and scatter from surrounding ridges/features/air back to the spacecraft or to the outpost (i.e., scatters that circumvent the primary topographic shielding).

If a natural topographic feature is not available, it is possible to create a manmade shield, or berm, between the reactor and outpost. A berm can be constructed by a bulldozer and/or backhoe, by stacking of locally made concrete, sandbags, or containers filled with regolith/sand, or simply rock-by-rock by a robot or human. This wide variety of options offers great flexibility, and methods can be utilized to best integrate with the existing and proposed infrastructure, as well as the mission architecture.

One of the biggest variables for an above-surface is how high the final reactor position is off the ground. From a shielding perspective the best configuration is to place the reactor on the ground. Shielding mass can be reduced by the elimination of the bottom shield. In addition, berm construction can be simplified by lowering the required height, or even better if topography solution can be enabled. The curvature of the Moon is so high that distance can help reduce line-of-site, but the distance would have to be relatively large. Great benefit could be provided if the reactor could be placed several km from the outpost. At 3.7 km of separation the curvature would cause a line-of-site deficit of ~ 4 m, which would allow a person standing at the outpost to be shielded from direct radiation up to a height of ~ 2 m on the power system. On Mars, the distance to achieve ~ 4 m of deficit would be ~ 5.2 km. Of course these curvature calculations assume a perfectly flat surface (actually a perfect sphere). Ultimately topography will dominate line-of-site and curvature might provide an additional small benefit, especially because several km of separation will be a difficult task on early missions. Ultimately, if advanced architectures could allow a separation of 10 km then curvature might dominate: the Moon would provide ~ 29 m of curvature deficit and Mars would provide ~ 15 m enough to allow even tall structures to remain shielded.

Unfortunately, placing the reactor on the surface will require the capability to lower the reactor from the reactor deck to the ground, or the infrastructure to remove the reactor and place it elsewhere. If the reactor cannot be lowered to ground level, there are many factors to consider. Constructing the berm, or using topography may become increasing difficult as the required height increases. If the reactor remains at a height of several meters or more, then the preferred option might be to try to install regolith around the reactor in place. Pre-formed or deployed containers could be filled by human and by robotic means (e.g., conveyor belt, vacuum hose, robotic arm, shovel/bucket, etc.). Alternatively sand bags or pre-formed in-site concrete could be stacked. These methods will be highly dependent on location, infrastructure, and lander design. One potential consequence of placing regolith close to the reactor is thermal balance; generally not as difficult as for a buried concept, but still a significant engineering design issue. The height of the reactor off the ground has several other design impacts, in addition to line-of-site. The closer the reactor is to the ground the more power deposition in the regolith, and warmer regolith is generally undesirable from a heat rejections perspective (although warmer regolith might help keep certain components from freezing overnight). The biggest effect of reactor height is on dose to the power conversion system and anything on the lander. If the reactor is very close to the ground (~ 1 m or less), there is some benefit because much of the ground scatter will return directly to the lower shielding. If the reactor is very high off

the ground ~ 4 m or more then the ground scatter paths become long enough that dose from the ground to the spacecraft is significantly reduced. As a result, there is generally a worst case height from a shielding perspective, which must be balanced with all other considerations. The inclusion of all spacecraft components (e.g., rockets and tanks) into the shielding solution is also important.

Skyshine on Mars

Skyshine is a shielding term that represents radiation that is scattered or created in the sky/air that eventually reaches a shielded area. On Earth, the atmosphere is relatively thick and provides enough shielding such that Skyshine is only a potential issue within $\ll 1$ km of a radiation source. The mean free path of a fast neutron in the low-density Mars atmosphere is ~ 1 km, and the bulk of the atmosphere is composed of two very effective neutron scattering elements—C and O. Therefore, any escaping neutron from the reactor can reach any location within a km or two from the reactor via only 1 or 2 scatters. For some bermed/topographic reactor shielding solutions on Mars, Skyshine can actually dominate the dose delivered to the shielded area. This can especially be the case if a reactor is placed horizontally (or anything other than vertical) on the surface. If Skyshine is significant, then additional 4pi shielding would be needed, mostly for regions facing the sky, although significant dose might also come from neutrons scattered off of the ground and then the sky. As distance increases beyond several km, then the shielding effect of the Mars atmosphere can provide more benefit than the negative provided by Skyshine.

Outpost separation

As indicated above, an important trade for an above surface reactor is the separation from a potential human outpost. The bulleted list at the start of this section shows that the dose to humans at the outpost is only one of several shielding concerns. In most cases, a good amount of 4pi shielding will be needed to meet other dose requirements, so the design question is: how much additional mass and/or construction work (burying, moving, berming, etc.) will be needed to accommodate humans at the outpost. Separation is a very useful shielding mechanism due to the $1/r^2$ effect. A typical design trade might consider the following: (1) separate 3 km without any additional shield mass, (2) separate 1 km with extra mass for shaped shielding, (3) separate 500 m with regolith installed on the lander, (4) separate 250 m with an on-surface configuration and a berm (or topographic feature), (5) separate 100 m with 1-m deep buried reactor, or (6) zero separation with a 4-m deep buried reactor. There are dozens of possibilities, and each will be mission dependent.

In most cases, a separation of a few km would be ideal except for one factor—the need to carry electricity from the reactor to the outpost. Power beaming will not be practical for surface reactors for decades, perhaps ever; thus a traditional power cable will be needed. The further the distance, the greater the concern of losses due to electrical resistance. There are two options to mitigate losses, (1) heavier cables with more conductive area or (2) using transformers for increasing the transmitted voltage (like what is done on Earth). Both of these options can make the mass penalty of a $\gg 1$ km separation greater than the shield mass savings. This issue adds another layer of complexity in determining the appropriate reactor separation for a specific mission.

The table below summarizes one example of a shield architecture trade based on the use of four 10 kWe Kilopower reactors on the surface of Mars (Poston et al., 2016). In column four, PMAD is the power management and distribution system, which is mostly electrical cable mass.

	Total Shield (kg)	Ave. Unit Shield (kg)	Instrument Distance (m)	Outpost Distance (m)	Low-Volt PMAD (kg)	Electric Power (We)	Total Mass (kg)	Specific Power (We/kg)
Left on Lander “as is”	2424	606	160	2000	1865	33.5	7989	4.2
Placed on Surface	2022	506	140	2000	1865	33.5	7587	4.4
Bermed/Topographic	1952	488	~ 50	~ 500	700	38.0	6352	6.0
Buried 75-cm-deep hole	1600	400	~ 25	~ 200	600	38.5	5900	6.5
Buried 150-cm-deep hole	1248	312	10	80	500	38.8	5448	7.1

The table above is a conceptual un-optimized study, but gives the flavor for the types of surface shielding trades that can be considered (note that the bermed/topographic case assumes the reactor is on the ground). This still only represents a small fraction of options, as it doesn't include the option of using regolith to surround the reactor on the lander, or use voltage transformers, include curvature effects, consider other materials, use regolith for axial reactor shielding, etc. Most importantly this chart does not spell out all of the individual design complexities and risks associated with each option; nothing should ever be decided solely based on numbers on a spreadsheet.

Radiation exclusion zones

Any time a reactor is used in the presence of humans there will be regions where an astronaut should limit time spent. Every one of these regions should be designated on an ALARA basis. The outpost or living quarters should be separated and/or shielded to lower the reactor dose to acceptable human levels; not necessarily as low as the 5 Rem/year used on Earth, but something that balances the benefits of the reactor (a capable and reliable power supply) versus the risk. Even if a restrictive 5 Rem/year is designated for the reactor, a large fraction of the reactor radiation can probably be shielded by the habitat, by the proper placement of any possible material between the reactor and regions where the astronauts sleep, eat, meet, etc. The difference between reactor radiation and GCR is that it can be shielded by any thickness of any material. If the dose in the habitat could be reduced by an order of magnitude (the equivalent of few inches of Al or regolith), then the reactor dose rate in the outpost region could be allowed to be closer to 25 Rem/year (similar to the GCR dose on the Moon or Mars), but only provide ~ 5 Rem/year to the astronaut assuming they spend 20 h/week performing EHA (extra-habitat activities) on the outpost grounds. Frequent outdoor work locations could perhaps be set up on the downstream or far side of the habitat (or on the far side of a ridge or manmade berm, etc.) to reduce EHA dose rate as well. Activities in the region surrounding the reactor (inside of the outpost distance) should be limited based on their importance. If a region of interest or infrastructure is half way between the outpost and the reactor, then the dose rate will be $\sim 4\times$ higher, but not of substantial risk for operations that might take a few of days of work. If an operation requires humans very close to the reactor, then it might limit time to an hour or two, or the reactor power could be significantly reduced to allow more time. If work is required on the power system itself (e.g., leak repair, component replacement) then the reactor would preferably be shut down and allowed to cool (radioactively) for a while; although if the power from the reactor is required to sustain life, then any potential dose would be acceptable to try and fix it. If shaped shielding is used to reduce reactor dose in the direction of the outpost, then an exclusion regions/rules would be azimuthally dependent.

Activation dose

The dose surrounding a reactor is highly dependent on its operational status. Prior to operation the radiation field from the reactor is negligible, and no shielding of the reactor of any kind is needed. During powered operation the reactor requires considerable shielding, unless humans are stationed many hundreds of meters from the reactor. After reactor shutdown, the dose environment immediately drops by at least 1 order of magnitude, and then drops off quickly from there (depending on the materials in the reactor, shielding and surroundings). In addition to decay radiation coming from the reactor core, there are also many potential activated materials that could provide dose to humans. This includes the nearby regolith, spacecraft, and reactor materials.

The activation dose depends primary on two factors: how long the reactor has operated and how long it has been shut down. The table below provides an example of dose rate surrounding a 40-kWe NaK-cooled reactor that is placed into a 1.5 m deep hole on the Moon (Poston et al., 2010) (technically not buried, because no regolith is backfilled on the top). The dose rate shown is to a person standing on the ground directly adjacent to the power conversion system (1 m from the core radial center), perhaps performing repairs.

Dose Rate After Shutdown (rem/hr) - 1 meter from Reactor C/L						
	Reactor Full Power Operating Time					
Time After Shutdown	1 Minute	1 Hour	1 Day	1 Month	1 Year	8 Years
100 sec. (1.7 min)	1.2E-01 <i>Rego 65%</i>	6.2E-01 <i>Rego 54%</i>	8.8E-01 <i>Rego 63%</i>	3.7E+00 <i>NaK 79%</i>	3.7E+00 <i>NaK 79%</i>	3.7E+00 <i>NaK 79%</i>
1e3 sec. (17 min)	5.4E-03 <i>NaK 40%</i>	2.2E-01 <i>NaK 59%</i>	4.5E-01 <i>Rego 59%</i>	3.2E+00 <i>NaK 89%</i>	3.2E+00 <i>NaK 89%</i>	3.2E+00 <i>NaK 89%</i>
1e4 sec. (~3 hours)	2.6E-03 <i>NaK 76%</i>	1.5E-01 <i>NaK 78%</i>	2.9E-01 <i>Rego 54%</i>	2.8E+00 <i>NaK 92%</i>	2.8E+00 <i>NaK 92%</i>	2.8E+00 <i>NaK 92%</i>
1e5 sec. (~1 day)	6.4E-04 <i>NaK 97%</i>	3.8E-02 <i>NaK 97%</i>	5.3E-02 <i>NaK 69%</i>	8.3E-01 <i>NaK 96%</i>	8.4E-01 <i>NaK 96%</i>	8.4E-01 <i>NaK 96%</i>
1e6 sec. (~2 weeks)	2.3E-07 <i>Rego 67%</i>	4.2E-06 <i>Fuel 70%</i>	9.1E-05 <i>Fuel 80%</i>	1.5E-03 <i>Fuel 72%</i>	2.7E-03 <i>Fuel 47%</i>	4.4E-03 <i>In718 39%</i>
1e7 sec. (~4 months)	1.6E-07 <i>Rego 89%</i>	3.5E-07 <i>Rego 73%</i>	4.6E-06 <i>Rego 66%</i>	1.2E-04 <i>Rego 59%</i>	6.8E-04 <i>In718 46%</i>	2.2E-03 <i>In718 74%</i>
1e8 sec. (~3 years)	1.6E-07 <i>Rego 89%</i>	1.9E-07 <i>Rego 74%</i>	8.1E-07 <i>In718 77%</i>	2.4E-05 <i>In718 76%</i>	2.7E-04 <i>In718 78%</i>	1.3E-03 <i>In718 87%</i>
> 10 rem/hr <i>leaving only</i>	> 1 rem/hr <i>mission critical ops?</i>	> 100 mrem/hr <i>maintenance ops?</i>	> 10 mrem/hr <i>> lunar background</i>	< 10 mrem/hr <i>< lunar background</i>	< 1 mrem/hr <i>negligible</i>	< 0.1 mrem/hr <i>Earth background</i>

The table above shows that the dose after shutdown can come from several sources, and the dominant source can change with operating and shutdown time. The text in italics shows the primary contributor and the fraction of dose it contributes. One day after shutdown the dose from the NaK coolant in the piping provides most of the dose, regardless of operating time. If the reactor has operated 8 years, then the Ni in the In718 (the Inconel 718 used in the Stirling engines) dominates dose after the reactor has been shutdown more than 2 weeks. Ironically the fuel itself only dominates dose in a few scenarios, weeks after shutdown.

The key point of the table above is that it is not unreasonable to assume maintenance could be performed on the system, except for the core itself (which in most cases should be highly reliable). Any above ground component (Stirlings, motors, piping, radiator, etc.) can be safely approached and repaired after perhaps a 1 day cooldown period, and in an emergency situation repairs could be performed almost immediately after shutdown without acute radiological consequences. However, this is a very specific case and all reactors and applications will be different: e.g., an above ground reactor would likely increase regolith/fuel dose, or many reactors will not use NaK and/or Inconel, etc.

Summary

Shield design and integration for space reactors is rarely a simple task. The shield's primary purpose is to reduce the dose from neutrons and gammas to specific regions; however, the modeling and calculation of radiation transport is usually the simplest part of shield design. The low mass and volume constraints for a space reactor can create significant thermal balance and structural issues. High temperature and low mass can lead to the use of materials with little experience/infrastructure or confidence in lifetime performance. The requirements for shielding are highly dependent on the mission goals, architecture, and environment. Shielding for in-space reactors presents considerably different issues than for surface reactors. There are many options for using in situ materials for surface reactor shielding, with the ideal case being able to use topography (ridges, craters, etc.) to perform the bulk of the radiation shielding. Regardless of the approach, a reactor can be used on a human exploration mission without adding considerable risk or limitations to the astronauts. Furthermore, the robust power-rich capability of fission reactors will likely make astronauts safer and more flexible/capable than any other power alternative.

See Also: Radiation Shielding Methods; Shielding Against Galactic and Solar Radiation in Space.

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Space Fission Power: Kilopower and KRUSTY

David I. Poston, Space Nuclear Power Corporation, Los Alamos, NM, United States

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Abbreviations

B ₄ C	Boron Carbide
BeO	Beryllium Oxide
DAF	Device Assembly Facility
DUFF	Demonstration Using Flattop Fissions
GRC	Glenn Research Center
HEU	Highly Enriched Uranium
ISRU	In-Situ Resource Utilization
KRUSTY	Kilowatt Reactor Using Stirling Technology
LANL	Los Alamos National Laboratory
MSFC	Marshall Space Flight Center
NCERC	Nuclear Criticality Experiments Research Center
NEP	Nuclear Electric Propulsion
NNSA	National Nuclear Security Administration
NNSS	Nevada National Security Site

RPS Radioisotope Power System
 We Watt electric
 Wt Watt thermal

Attributes of a Kilopower reactor

Fig. 1 shows the basic layout of a 1-kWe in-space Kilopower system. The basic Kilopower reactor components are the fuel, heat pipes, control, reflector, and shield. The Kilopower system includes the reactor plus the Stirling power conversion system and the radiator (plus other sub-components). The 1-kWe reactor core (fuel) is only 11 cm in diameter and 25-cm tall. Eight Haynes230 heat pipes are placed in slots on the outside of the fuel, and are held in place by Haynes230 clamping rings via an interference fit. The heat pipes, which nominally operate at 800 °C, use sodium as the working fluid and include a nickel mesh wick in the evaporator region. The heat pipes transport heat to the hot-end of eight Stirling convertors, which are cooled on the cold-end by a titanium/water radiator. The in-space power system design is very similar to a surface reactor system (e.g. for Moon or Mars) except for the layout of the shielding and the radiator. Higher power systems will have larger Stirling engines, increased radiator area, more heat pipes, etc., with only a modest growth in core size.

Each Kilopower component, and how they integrate together, is simplified by low power; e.g., at low powers (<100 kWt) thermal management and irradiation damage of components do not complicate system design. The simplicity of the system also leads to high reliability. The reactor is essentially solid-state, with the control rod being the only moving part. Actually, at low powers (~10 kWt) the burnup effect on reactivity is so small that long lifetime (10+ years) could be achieved without any control movement after startup; higher power systems would require occasional (monthly or annual) control movement to maintain reactor temperature. At all power levels, Kilopower systems can survive worst case transients (e.g. loss of heat removal by Power Conversion System (PCS)) without any control action. The lack of need for real-time reactor control greatly simplifies system control. The Stirling controller can independently control the reactor/system power level, without potential interference and interactions caused by a separate control feature associated with the reactor. The “reactor” control system only needs to move the control rod at startup, and whenever a boost in reactor temperature is desired—this could possibly be done remotely when deemed necessary by a ground engineer, and thus not require any automated control software.

The Kilopower reactor concept is one of the simplest space power reactor concepts ever proposed. Perhaps the most important “simplicity” of the system is in neutron kinetics and system dynamics. The kinetics of a compact, fast reactor are dominated by one factor—changes in material density/geometry (changes in neutron interaction rates, i.e., cross sections, have small effects). The Kilopower solid core eliminates potential movements of fuel rods/pieces relative to others, and the surrounding geometry is fixed (except for small potential relative movements due to thermal expansion), thus the only major reactivity effects are changes in neutron leakage due to material expansion. This makes the startup and operational system dynamics easy to predict/verify.

Another system attribute that leads to high reliability is inherent redundancy in heat transport. Each heat pipe is an independent, highly reliable mechanism. In all proposed Kilopower systems, full power can be delivered even with several heat pipes or Stirling engines failed. If three heat pipes fail that are directly adjacent to each other, then power level may need to be reduced to avoid exceeding the fuel temperature limit (assuming that heat pipe failures can be diagnosed). The baseline



Fig. 1 1-kWe Kilopower system layout.

Kilopower power conversion approach is to attach a single Stirling engine to a single heat pipe—referred to as the 1-for-1 approach. This was the configuration used in KRUSTY, which provides the simplest suite of technologies, the simplest system dynamics, and the highest efficiency (i.e., smallest temperature drop). The 1-for-1 configuration requires a large number of small engines, which may or may not be optimal from a cost and development perspective. One negative of this approach is that if a Stirling engine fails, it effectively fails a heat pipe in the core, whereas an intermediate heat transfer mechanism would eliminate this problem. However, the 1-for-1 approach provides a reliable diagnostic of heat pipe failure, which will allow mitigation of worst case failure patterns if they indeed occur (noting that the probability of heat pipe failure will likely be much lower than that of a Stirling converter).

Kilopower reactors should also be very reliable with respect to launch and landing loads. A solid block of fuel eliminates potential fuel-pin, grid-plate movements. Heat pipes should also be less fragile than the alternative—coolant piping to and from the reactor, including connections to other loop components; plus, the piping and connections will likely provide a single point failure. The Kilopower project has started to evaluate launch loads and the system appears robust.

Finally, the compact reactor size allows for a simple approach to launch safety, transport, and nuclear and non-nuclear system testing. Kilopower reactors are essentially non-radioactive prior to operation. The only condition that can create a significant nuclear hazard is an inadvertent movement of the control rod that causes criticality. There is no conceivable launch or transport accident (including water, wet-sand, etc.) that can cause criticality unless the rod is removed. More so, the system will only go critical if the rod is removed and the radial neutron reflector is geometrically intact. It is unlikely that any impact strong enough to remove the rod will not also remove, or least significantly crack/deform the radial reflector, and system design can help ensure this. Therefore, the only significant nuclear safety engineering required for Kilopower is to ensure the rod does not move unless it is properly commanded to do so (a feature that any reactor must have by definition).

The mass of a Kilopower system is highly dependent on the application (power level, lifetime, environment), the risk tolerance of the customer (e.g. level of redundancy and margin), and can change substantially due to shielding requirements (in-space or on-surface, separation distances, rad hardness, etc.). The table below shows a range of masses for Kilopower systems designed to operate 10 years at full power with a typical shield mass. Space reactor shielding is discussed in-depth in [Poston \(2021\)](#).

<i>Kilopower system</i>	<i>Mass</i>	<i>Specific power</i>
1-kWe HEU	400 kg	2.5 We/kg
10-kWe HEU	1400 kg	7.1 We/kg
10-kWe HALEU	2200 kg	4.6 We/kg
30-kWe HALEU	3500 kg	8.6 We/kg

The mass estimates could be higher or lower based on the aforementioned factors (especially shielding), but the uncertainty in mass for a specific application should be relatively low due to the simplicity of Kilopower, and comparison to the successful Kilopower Reactor Using Stirling Technology (KRUSTY) test. KRUSTY was tested with HEU (93% enr. U), but in some cases HALEU (19.75% enr.) might be preferred for non-technical reasons. Kilopower reactor core lifetime is limited by fuel swelling criteria, which is a function of fuel burnup (fraction of uranium atoms fissioned). The HALEU systems have substantially more uranium mass, and therefore much lower burnup at a given power level. This is why the HALEU system can be designed to much higher power levels than the HEU system. Note also that technical risk increases slightly with higher power for all concepts.

Origins of the Kilopower concept

Early push for a simple concept

By the early 2000s there was a realization that previous space reactor programs had reached for advanced concepts that could not be readily built without significant investment in both research and facilities ([Poston, 2002](#)). These programs typically called for:

- Designs using materials for which little or no data existed requiring extensive research and development;
- Designs with uncharacterized physics that required research to verify;
- Designs with advanced fuels that required investment in fuel research and testing, new production facilities or alternatively new production lines in existing facilities; and
- Designs requiring significant ground testing of the reactor in new test facilities to prove out the advanced concept.

The programs had schedules of a decade or more and cost estimates for a completed space reactor in the billions. Given the time and cost for the programs, Congress did not have the patience to fund them until completion (an example is presented in the GAO report on the SP-100 program ([General Accounting Office, 1992](#)). After approximately a dozen failed space reactor projects this situation led to many space reactor enthusiasts to call for a simpler concept that could be built in a reasonable amount of time and for a reasonable cost.

2010 Fission study for gas giant decadal survey

In 2010 as part of the Planetary Science Decadal Survey for gas giant planet missions, a simple small fission power system for space was evaluated. The study was documented in the 2011 report “Small Fission Power System for NASA Planetary Science Missions (Mason et al., 2011).” The report outlined a concept for a simple low power reactor. The reactor was assumed to have a small metal-fueled core, about the size of a coffee can, and was cooled by liquid metal heat pipes. Two version of the reactor were proposed, one with thermal electric power conversion and one with Stirling engine power conversion. The reactor thermal power was set at 15 kWt, such that the reactor using thermal electric power conversion would produce 1 kWt of electricity and a version using Stirling engine power conversion would produce 3 kWt of electricity. The goal was to produce a reactor that could be built and tested easily and cost effectively. As the report states:

At this scale, numerous test facilities exist within the NASA community to conduct thermal- vacuum performance and mechanical vibration testing. The reactor fuel can be fabricated in existing production facilities, and the reactor neutronics can be verified in existing nuclear criticality test facilities within the DOE complex.

The report concluded that this low risk reactor concept could be built and be readied for flight in 10 years or less. The only criticism of the concept was that a heat pipe reactor had never been built and tested. This criticism would be addressed by the eventual DUFF and KRUSTY tests.

The reactor was not given a name until after the completion of the report, but the NASA program lead named the reactor concept “Kilopower” and the name would stick for the rest of the program.

Beginnings of the DUFF test

In the winter of 2011 LANL staff began to explore how to build and test a Kilopower reactor concept. Based on the design from the 2011 study, the thermal power was lowered to 4 kWt from 15 kWt. The Stirling engine version was embraced over the thermal electric version. By going with the Stirling engine version, the 4 kWt reactor would produce ~1 kWe. This smaller version was deemed even easier to build and test.

Conversations were then started with the LANL critical experiments facility staff located at the Nevada National Security Site. A prototype reactor would need to go to peak temperatures of 800 °C. However, the critical experiments facility regulatory limit on excess reactivity limited reactor temperature to 300 °C. Also, a prototype reactor was more expensive than could be done for the limited budget the LANL team thought could be secured through internal R&D funding. A compromise was proposed to use an existing critical experiment reactor, called Flattop, to run at 300 °C and power a small low temperature Stirling engine using a water heat pipe. The thought was that by demonstrating a heat pipe reactor, a major criticism of the 2010 report could be addressed and be used a foundation to launch a prototypic test.

LANL secured internal R&D funding for this demonstration in early 2012. A meeting was held in Nevada with NASA, LANL and the Nevada National Security Site contractor. Agreements were reached for each organization to fund their part of the demonstration test. The test was set for late summer/early fall of 2012. This early test was named Demonstration Using Flattop Fissions or DUFF. DUFF will be described in more detail later in this chapter.

After the test the Kilopower reactor concept was developed further into the baseline 1 kWe (4 kWt) version for deep space power and a 10 kWe (~40 kWt) version for nuclear electric propulsion and surface power. This concept was documented in early 2013 by LANL and NASA team (Poston et al., 2013).

The birth of KRUSTY test

After the DUFF test was completed in 2012, NASA interest in the Kilopower concept grew rapidly. A white paper called “Why Kilopower” (McClure et al., 2012) was distributed in early 2013 to elicit support for the prototype test.

The Y-12 production facility in Oak Ridge, TN was queried about making the fuel for a Kilopower prototype. Y-12 assured the team that the fuel would be easy to make since it was similar to existing components already built by the facility.

In 2014, the NASA Mars campaign, run out of Johnson Space Flight Center, baselined the Kilopower concept for potential Mars Missions. The Mars campaign need 40 kWt of electricity for the baseline Mars missions. Five 10-kWe reactors (four plus a spare) could perform this mission. A preliminary Mars concept was developed, and requirements for this system were defined in partnership with NASA’s Human Spaceflight Architecture Team (Rucker, 2015).

In early 2014, with support from the Mars campaign, NASA management began to consider funding a prototype of Kilopower. By the summer of 2014, the National Nuclear Security Administration (NNSA) stepped up to be the Department of Energy partner and help perform the experiment. The NNSA managed not only LANL, but the critical experiments at the Nevada National Security Site and the Y-12 production site that would make the fuel. This support became even greater when the Criticality Safety Program in the NNSA agreed to partially fund the project since it could obtain valuable nuclear data (cross-section measurements of BeO) as part of the test. NASA kicked off the Kilopower project in October of 2014 with the goal of completing a nuclear reactor demonstration by the end of 2017. This reactor demonstration was called the Kilowatt Reactor Using Stirling Technology or KRUSTY. KRUSTY will be described in more detail later in this chapter.

The DUFF test

As was described in the 2010 study, a heat pipe cooled reactor had never previously been tested. The Demonstration Using Flattop Fission (DUFF) test was devised as a proof-of-concept experiment for the aforementioned small space reactor.

The test was conducted in September of 2012 and was successful on many levels, but most importantly it produced electricity using nuclear fission, heat pipes, and Stirling engines. DUFF demonstrated the first Stirling engine to be powered with fission energy, the first use of a heat pipe to transport heat from a reactor to a power conversion system, and the first nuclear-powered test of a potential space reactor technology in over 40 years. One of the biggest benefits of the testing was that it provided valuable experimental data. The data demonstrated that the Kilopower concept was viable and the physics well characterized. The data also provided the ability to benchmark criticality and dynamic reactor modeling tools used to design the KRUSTY test. A complete description of the DUFF test is presented by [Poston et al. \(2014\)](#).

DUFF experimental setup

The experiment to model the reactor consists of three main components combined into a single experimental setup. The three components of the experiment are:

- The critical experiment assembly “Flattop”,
- A water-based heat pipe, and
- A Stirling Engine power conversion system.

The three pieces of the experiment were coupled together to form a complete nuclear system that could go critical, develop fission heat, transfer the heat via the heat pipe to the Stirling engine, and the Stirling engine producing electricity.

The Flattop Critical Experiment Assembly was designed to provide benchmark neutronic measurements in a spherical geometry. Flattop includes a spherical core of highly enriched uranium (HEU), a natural uranium reflector, a control system to bring the system critical. A distinguishing feature of Flattop that made the assembly ideal for this experiment was that the HEU spherical core and the natural uranium reflector contain a 1.283 cm (0.505 inch) hole through the reflector and the core. This allows for a 1.270 cm (0.5 inch) heat pipe to be placed into the core with no modification to the existing critical assembly.

The water heat pipe that connected Flattop and the Stirling engine converters utilizes a stainless-steel wall and a sintered-nickel wick and was approximately 114.3 cm (45 inches) in length.

The set up used an electric chiller to provide cold-end temperatures below room temperature, and “off the shelf” Stirling engines that could operate at these temperatures. Two of these engines were used and connected to each other and the heat pipe by means of a copper block. They were specifically designed for, and tested at, low temperatures. The engines were designed to operate at -30°C on the cold end and up to 400°C on the hot end.

The experiment was performed at the Nevada Nuclear Security Site (NNSS) Device Assembly Facility (DAF) which houses the National Criticality Experiments Research Center (NCERC). The complete experimental setup is presented in [Fig. 2](#).

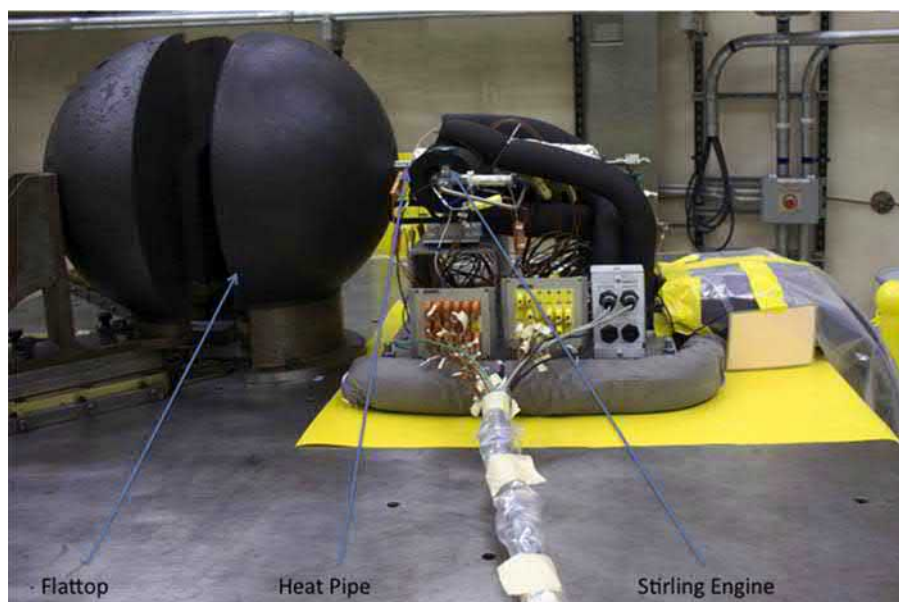


Fig. 2 DUFF, Demonstration using flattop fissions, experimental setup.

DUFF experimental results

The DUFF experiment was performed twice, once on September 13, 2012 and the second time on September 18, 2012. The experiments were very similar except that the first experiment used a rapid ramp up of reactivity, while the second experiment used a slower ramp up. Each is described in detail below.

The timeline of events for the rapid reactivity ramp up experiment are shown graphically in Fig. 3. The test results showed the expected negative temperature feedback behavior of the reactor physics. Fast reactors in this size range are controlled by thermal expansion and subsequent negative reactivity feedback. The simplest representation of this phenomenon is in the fuel. If the fuel heats up then the atoms expand further apart, it is easier for more neutrons to leak out of the fuel and avoid a fission reaction. This leakage results in lower reactivity (lower neutron multiplication), and thus decreases the slope of power production. If the system is operating at steady state, the slope of power will go from zero to negative, thus system power decreases. Therefore, during nominal operation, if more heat is extracted by the power conversion system, then thermal feedback will raise reactor power, or vice-versa. In this manner, the reactor is a load following system component; i.e. the reactor thermal power automatically matches itself to the thermal power removed by the Stirling engines and/or passive heat losses.

The timeline of events for the slow reactivity ramp up experiment are shown graphically in Fig. 4. This test was a repeat of the September 13, 2012 test but with a different rate of reactivity insertion. The test again behaved as expected with reactor power changing to accommodate changes in the power conversion system. This result validated the expected changes given the negative temperature feedback of the reactor.

Summary of the DUFF test

The DUFF experiment demonstrated that a heat-pipe-cooled fast reactor with Stirling Engine power conversion might be a practical reactor concept for space reactor applications. Heat-pipe-cooled reactors have been postulated in the past and tested using separate effects testing (i.e. electrically heated), but this was the first heat pipe reactor using nuclear fission as a heat source. The test demonstrated that a heat pipe cooled fast reactor with Stirling engine power conversion has well characterized physics that can be easily modeled.

Despite its success, DUFF was not a prototypic heat pipe cooled space reactor. Only a small fraction of reactor power was removed by the heat pipe, the rest was passively lost. In addition, the temperature was very low and the heat pipe materials were different. The most important attribute of DUFF was that it showed that nuclear testing could still be done affordably, provided that it is made simple enough. The test laid the groundwork for how to construct a more prototypic Kilopower test that could be completed affordably, and provided credibility to the engineering team and proposed plan. Therefore, the key value of DUFF was that it generated the interest and confidence to pursue a full reactor demonstration, the KRUSTY test.

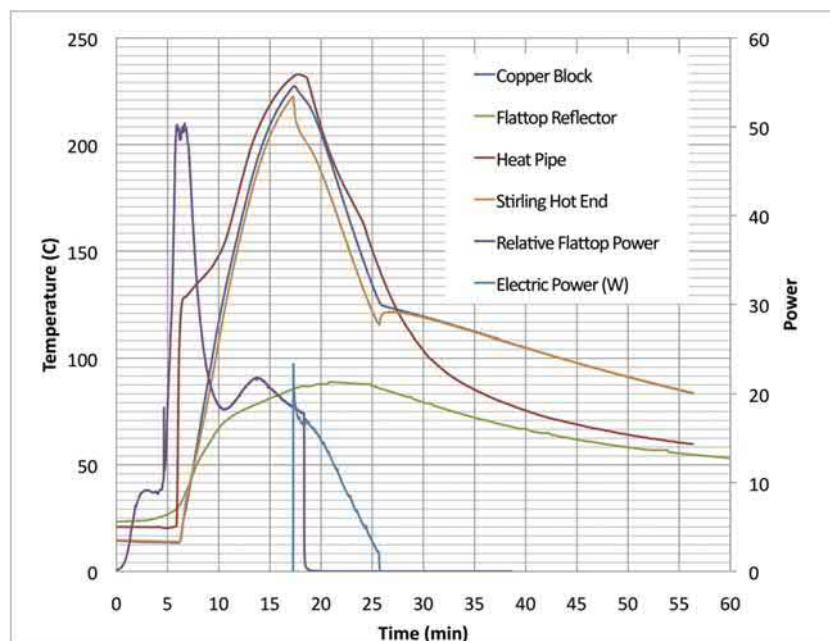


Fig. 3 DUFF, September 13, Fast reactivity ramp-up test results.

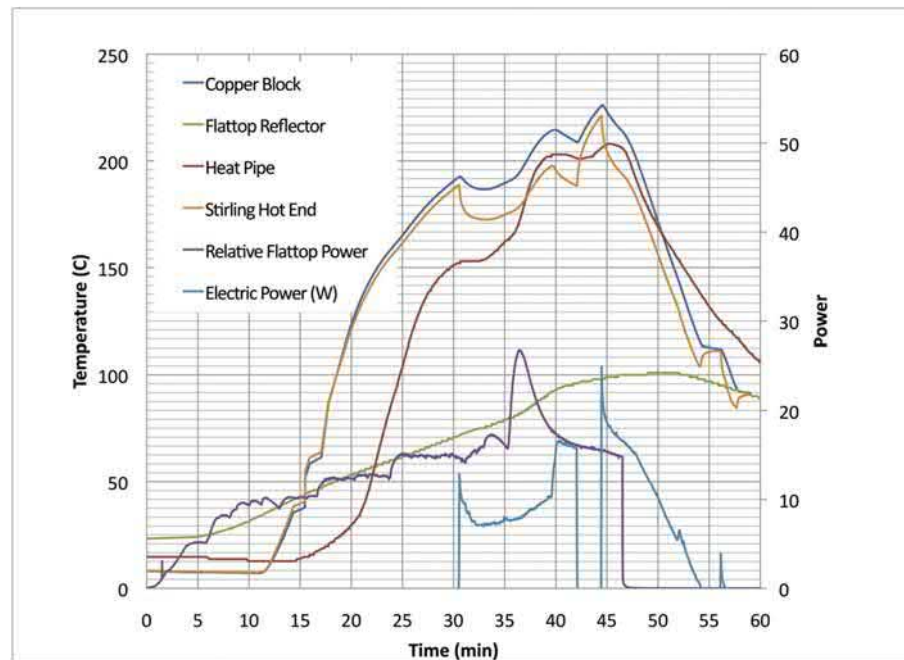


Fig. 4 DUFF, September 18, Slow reactivity ramp-up test results.

The KRUSTY test

After the success of the DUFF test, NASA and NNSA agreed to fund and manage the KRUSTY test or Kilowatt Reactor Using Stirling Technology test. KRUSTY was designed to be a full test of the reactor in a relevant vacuum environment that would demonstrate the key physics of the Kilopower reactor concept. Namely;

- Startup of the reactor with frozen heat pipes;
- Steady state performance of the reactor;
- Load following of the reactor to changes in the power conversion (Stirling engines) draw;
- Fault tolerance to heat pipe failure;
- Verification of temperature change for reactivity changes;
- Ability to withstand a complete loss of active cooling.

The test is described in the next sections. The test is described fully in a series of papers in a special issue of Nuclear Technology (Poston et al., 2020a,b,c; Gibson et al., 2020; Sanchez et al., 2020; Grove et al., 2020).

KRUSTY development plan

The KRUSTY test consisted of three major phases:

1. Electrically heated mechanical prototype testing using a stainless-steel core
2. Electrically heated testing of the system using a depleted uranium core, and
3. Nuclear testing, which was further broken down into three phases.

The major phases of the KRUSTY test are shown in Fig. 5.

NASA's Glenn Research Center (GRC) led the effort to design the test and built the power conversion and heat rejection portions of the system. NASA's Marshall Space Flight Center (MSFC) developed the electrical heat source for the non-nuclear testing and contributed significantly to KRUSTY mechanical design. Los Alamos National Laboratory led the design of the reactor and performed nuclear testing. The Y-12 production facility fabricated the highly enriched uranium (HEU) reactor core. The Nevada Nuclear Security Site Mission Support & Test Services supported nuclear testing at NCERC at the Device Assembly Facility.

KRUSTY thermal prototype development and test

In 2015, the first step to developing a KRUSTY technology demonstration was to validate the form, fit, and function of the power system, and the transfer of heat from the reactor core to the Haynes 230 heat pipes filled with sodium working fluid. Stainless steel was used to prototype the reactor core as an inexpensive option that also has thermal properties somewhat

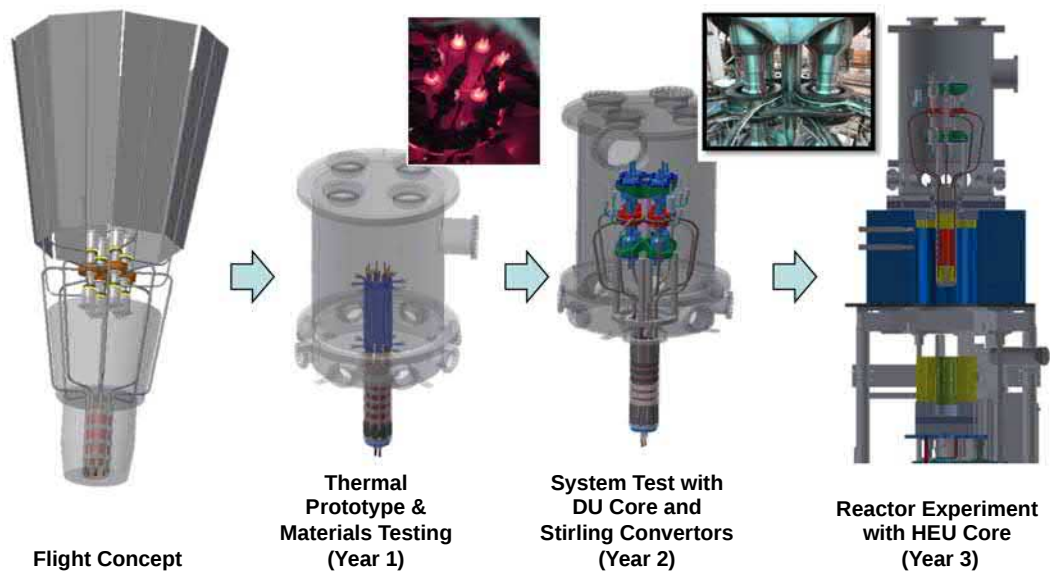


Fig. 5 KRUSTY development phases.

similar to uranium. An electrical heater was developed at NASA MSFC to heat the core in the central hole that, in the flight configuration, would accommodate a boron carbide poison rod. A vacuum chamber with gas-cooled walls was constructed to accommodate this assembly. Several methods of clamping the heat pipes into grooves cut in the circumference of the core were tested. As a result of this testing, clamping of the heat pipes to the core by encircling the heat pipes with heated stainless-steel rings and allowing them to cool and contract around the assembly provided the best thermal contact between core and heat pipes. This result permitted the project to proceed with the next phase of testing, using non-fissioning depleted uranium (DU) and Stirling engine power conversion.

KRUSTY depleted uranium (DU) test

An electrically heated depleted uranium reactor configuration was tested at GRC during the next phase of the test. This series of testing allowed for evaluation of alternative configurations of the Stirling power conversion interface with heat pipes, and was used to determine the configuration to be used for the nuclear test in Nevada. The testing also served as a dry run of the instrumentation, control, and testing conditions anticipated for the final KRUSTY nuclear test at NCERC.

The DU core alloyed with molybdenum 8% by weight core sections were successfully cast by the Y-12 National Security Complex. While the casting produced core sections that met the specifications for the KRUSTY experiment, the mold geometry and curing furnace temperature profile resulted in small voids in the final casted sections, mostly near the upper surfaces. Changes to the mold geometry and furnace temperature profiles were made, that when implemented for the HEU core section casting, produced excellent results.

The 1-kWe Kilopower reference reactor includes eight Stirling engines for electrical power generation. For the KRUSTY test, two Stirling engines designed for nominally 80 W_e output each were available from the NASA Radioisotope Generator program. For the KRUSTY experiment, it was determined that six nitrogen gas-cooled thermal simulators, controlled to approximate the thermal load of a real Stirling engines, would be substituted for the six remaining Stirling engines at a much lower cost.

Two configuration options of the KRUSTY experiment were assembled and tested in vacuum with the electrically heated DU cores. These configurations differed in the interface between the Stirling engines/simulators and the heat pipes. In the first configuration, the Stirling engines/simulators were grouped in pairs, a traditional configuration for Stirling engines that dynamically balances the linear motions of their pistons to eliminate vibrations in the adjacent spacecraft or planetary surface mounting structure. For this first configuration, the Stirling engine/simulator hot ends were connected to a metal plate that collected heat from two heat pipes. For the second configuration, a second set of heat pipes were built with flared ends that interfaced directly with individual Stirling/simulator acceptor plates. This single engine array configuration requires an electrically operated mass balancer to cancel the vibrations from the unpaired engine. The test of these two configurations allowed the determination of which configuration was simpler and more robust, and which had the lowest weight. The increased efficiency of the second configuration's improved heat pipe/engine interface more than compensated for the mass and power consumptions of the mass balancers, and the single engine array configuration of KRUSTY was selected for the Nevada test at NCERC. This configuration is shown in **Fig. 6**.

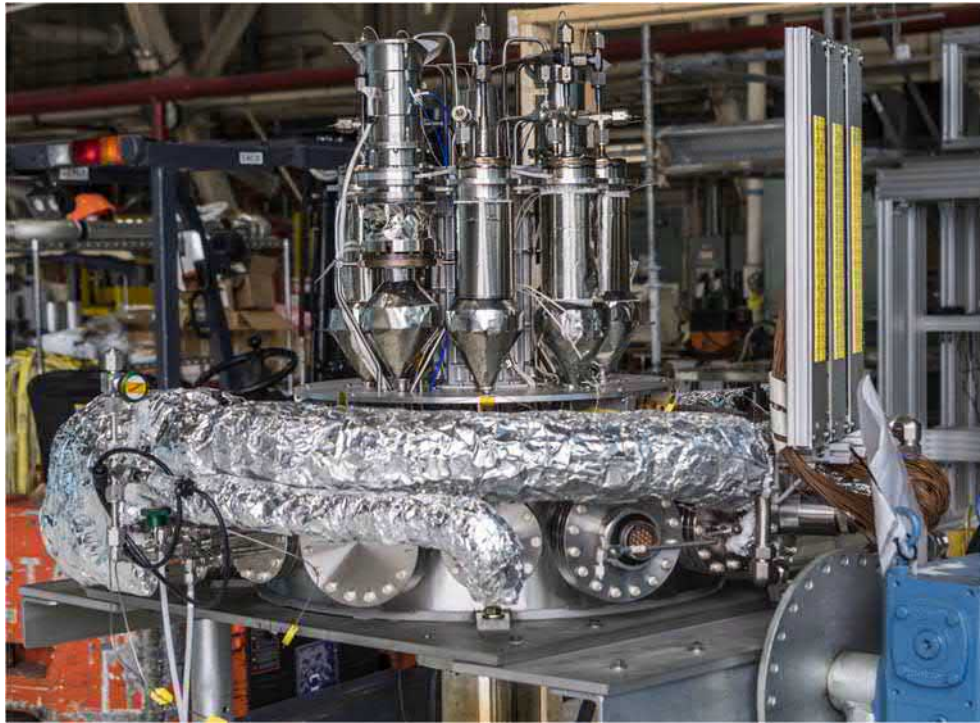


Fig. 6 Stirling engines and simulators attached to vacuum chamber bottom.

KRUSTY preparation for nuclear test

The Y-12 National Security Complex cast the KRUSTY HEU core sections during the summer of 2017. The non-nuclear systems were all thoroughly tested and checked out using an electrically heated core simulator earlier in the year, followed by assembly dress rehearsals in a mocked-up NNSS test facility environment located at the GRC aircraft hangar 6. The KRUSTY test hardware was then shipped from GRC and arrived at the NNSS in October 2017. It was inspected before transportation to the Device Assembly Facility (DAF).

KRUSTY nuclear testing

KRUSTY nuclear testing consisted of four phases:

- Reactor Component Critical Testing;
- Cold Critical Testing of the Reactor;
- Warm Critical Testing of the Reactor;
- High-Temperature, Full-Power System Testing.

The test phases were of sufficient length to permit thorough, incremental understanding of the reactivity of the KRUSTY core, and the response of the KRUSTY system to the reactivity changes.

In each of these phases the reactor was made critical on the Comet machine by raising the BeO reflector around the reactor core and inside the shield. This method was used because it takes advantage of the safety qualified lift table on the Comet machine. To have used the control rod in the central hole in the core would have required qualifying a new control mechanism that would have required funding and time not available to the project. **Fig. 7** shows the configuration of the KRUSTY experiment on the Comet platform for various states of reactivity.

The reflector position corresponds to a calculated level of reactivity. During critical experiments, the magnitude of reactivity is determined by the slope of the time-increase in neutron flux (power), and then the reflector is lowered. The measured reactivity is compared to predictions. This process is repeated with increasing levels of reactivity sufficient to confirm the understanding of the experiment performance against predictions.

It is important to note that the difference between moving the BeO radial reflector and a B₄C internal rod (flight) is very small. The raising of the BeO increases reactivity by decreasing neutron leakage, while withdrawing the B₄C rod increases reactivity by decreasing neutron absorption. This difference will cause second order effects on power distribution and feedback, but as far as the neutron population (i.e. power) is concerned, there is very little difference between these two reactivity mechanisms, despite their different geometric locations. This is because KRUSTY is a very good example of a point-kinetic reactor, which occurs when

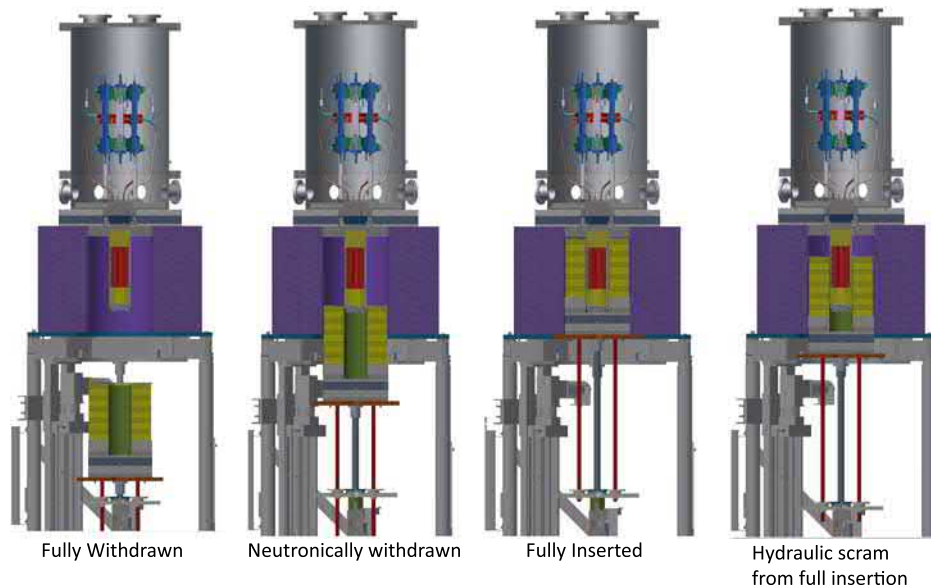


Fig. 7 KRUSTY assembly on the Comet platform at different stages of reactivity.

the neutron mean-free-path is a significant fraction of the core geometry. In such a system, all regions of the reactor communicate neutronically very well with each other and regional feedbacks effects are negligible (except for how they affect integral reactivity). Therefore, the reactor will respond the same whether reactivity is inserted externally or internally.

Component critical testing

The first-phase testing was the characterization of the reactivity of the KRUSTY reactor components, including the core, reflector, and shielding. The reactor core and shielding were installed onto the top platform of the Comet experiment platform. The reflector was assembled on an elevating lift table, below the core. Numerous component configurations were tested, including changing the material of some components. Iterative tests were performed to determine the reactivity worth of the beryllium oxide reflector and central boron carbide rod. The core first went critical on November 16, 2017. The results of the component critical testing were used to evaluate the analytical model used by LANL to predict reactivity of the KRUSTY experiment. The agreement between the analytical model and the results were outstanding, with only minor variations.

The core was then removed from the platform and installed into the KRUSTY heat pipe and Stirling engine/simulator assembly, and this assembly was installed on top of the Comet platform, to complete the readiness of the experiment for the subsequent critical test phases.

Cold critical testing

With the core, heat pipes, and Stirling engine/simulator assembly integrated, the same process of iterative reactivity increases was repeated. Cold critical testing began on January 12, 2018. This phase of testing continued until the end of February 2018. The results of the cold critical testing were used to again update analytical models and to prepare the system for the warm critical experiments.

Warm critical testing

Warm critical testing began on March 7, 2018. The reflector was stacked BeO with less than 80 cents of total excess reactivity in preparation of the warm critical experiments. The warm critical experiments were used to determine the temperature feedback of the reactor assembly for three progressively hotter experiments. The experiments had the following excess reactivity insertions:

- 15-cent insertion on March 7;
- 30-cent insertion (15-cent followed by small insertions until ~ 30 cents reached) on March 8;
- 60-cent insertion (15-cent followed by small insertions until ~ 60 cents reached) on March 14.

The information obtained from the warm critical runs was used to determine the appropriate reflector configuration (the necessary amount of reactivity excess) for the high-temperature run and was used to validate the accuracy of the analytical models in predicting the behavior of the KRUSTY core during the high-temperature run. The 60-cent insertion test was modeled before the test to predict temperature behavior in the core. This prediction was then compared to actual data after the experiment. The DOE regulator required close agreement between the model and the data prior to proceeding to the high-temperature testing. The peak temperature of the outer surface edge of the core at the axial midplane was used for comparison. The predicted peak temperature was 446°C , the actual was 447°C . The DOE approved the high temperature test approximately 2 days after the end of the 60-cent insertion test. The



Fig. 8 Fully assembled KRUSTY test showing reflector (BeO) in white on platen. Shield block and vacuum chamber above.

BeO reflector was stacked with enough BeO to provide approximately \$2.00 in excess reactivity. **Fig. 8** presents a picture of the KRUSTY experiment on the Comet machine with the lower platen fully loaded with BeO.

28-Hour high-temperature, full-power test

The final full-power (~ 4 kWt), core temperature at 800°C , 28-h experiment was begun on March 20 and ended on March 21, 2018. The final phase of testing consisted of a simulated space mission profile for a Kilopower system. Start-up of the fission power system was accomplished by increasing reactivity gradually over $1\frac{1}{2}$ h until the core reached 800°C ; temperature was measured by 18 thermocouples placed on the outer edge of the fuel. The Stirling engines were started at approximately 1.2 h. Steady state full-power operation was held for approximately $5\frac{1}{2}$ h to validate normal operation. Over the remaining 21 h a set of transient experiments were performed on the core to demonstrate the self-regulating characteristics of the Kilopower reactor. These transients included the following:

- Demonstrating load following characteristics by increasing the stroke on the Stirling engines to increase power draw on the reactor core. The stroke was then decreased to lower the power draw on the reactor core.
- Simulating the failure of one and two Stirling engines.
- Adjusting the reactivity to demonstrate the core temperature response.
- The turning off of all active heat removal by the power conversion system.

Fig. 9 shows a condensed version of the results over the entire 28 h of the nuclear system test. The time scale on **Fig. 9** and all subsequent plots is in hours, relative to planned test start time “T-0” of 10 am Pacific Daylight Time (PDT) reprinted from [Poston et al. \(2020c\)](#). The figure displays three types of measurements. The majority of the colored curves plotted depict fuel thermocouple data. The figure contains a red line that represents fission power, although the actual measurement is a current (in amperes (amps)) reading from a neutron detector, and fission power was later normalized to amps. A dashed black line on the figure depicts the relative platen (lift table) position. KRUSTY’s reactivity is changed by axial movement of the BeO radial reflector on the COMET platen. The value displayed was normalized so it could be cleanly co-plotted along with other variables on a common scale.

Startup and steady-state operation

At $T = 0.63$ the core temperature reached 500°C and the heat pipes began to “operate” (i.e. at temperatures less than 500°C there was not enough sodium vaporization within the heat pipes to transfer significant power). After the heat pipes “turned on,” the core gained access to the thermal inertia outside of the core. At $T = 0.67$, the core temperatures started to significantly diverge. The temperature near the bottom of the core resumed a heat-up rate close the adiabatic value ($\sim 30^\circ\text{C}/\text{min}$), while the temperatures

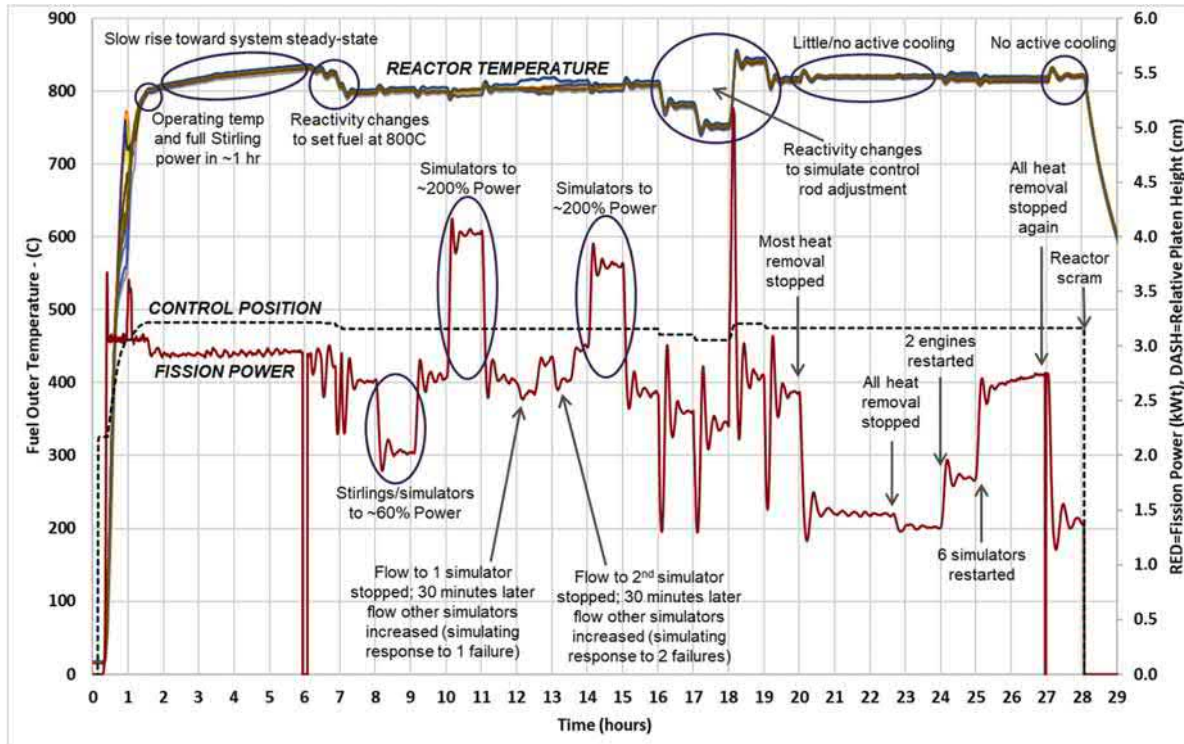


Fig. 9 Power and temperature data from the 28-h KRUSTY nuclear system test.

near the top were cooled very well. This is a clear indication that the heat pipe was struggling to remove power from the bottom of the core. At $T = 0.92$, after the condenser temperature had finally caught up to the rest of the heat pipes, the reactor power starts to uniformly heat up the core and entire heat pipe, and move the heat pipe away from the temperature limited regime. At $T = 1.14$ the two Stirling convertors were turned on as planned, when the temperature of the hot end of the Stirling engines reached greater than 650°C , which is the nominal operating point of the convertors. At $T = 1.7$ the system had reached quasi-steady state; i.e. a state where the core and Stirlings were in sync with each other, and only external or second order factors could significantly alter system power and temperature.

Load following transients

The cornerstone of the KRUSTY transient testing was to assess the thermal load-following capability of the Kilopower reactors. At $T = 8.0$ h the power draw from the Stirlings was reduced. The reduction in heat removal caused a rise in average core temperature, thus a decrease in reactivity (primarily via thermal expansion causing more neutron leakage) and thus a drop in power. The gradual drop in power subsequently caused core temperature to fall back toward the reactor “thermostat” set-point of $\sim 800^{\circ}\text{C}$.

At $T = 10.02$ another positive step change in power draw was initiated. The Stirling engine simulator flow rates were increased to approximately double their power draw. The increase in simulator flow caused the fission power to change from 2.75 to 4.05 kW, a rise of 1.3 kW. As with all of the load-following transients, the fuel temperatures responded very quickly to the increased power removal; the increased heat removal inside the simulator was fully evidenced by a drop in the fuel temperature.

Fault tolerant transients

One of the most attractive features of Kilopower systems is reliability, including the ability to avoid single-point failures and in many cases tolerate numerous failures. The “fault tolerance” transients were conducted to verify the ability of KRUSTY to deliver nominal power in the case of failed Stirling modules. A Stirling module (also referred to as a string) includes the heat pipe, the Stirling, the heat rejection, and the interfaces that connect them; these components are in series, so a failure of any of them fails the string. In KRUSTY, the flow through the simulators was cut-off to fail the string. The heat is redistributed to the other “strings” such that the reactor continues to produce full power. Fig. 9 show the fault-tolerance transients occurring from $T = 12$ to $T = 14$.

Reactivity control transients

The “reactivity control” transients were conducted to measure the clean reactor response to system reactivity changes. Fig. 9 show the response to the reactivity control transients, which occurred from $T = 16$ to $T = 20$. The platen containing the BeO reflector was gradually raised to its highest position during the testing. This subsequently took the KRUSTY reactor to

its highest temperature of 840 °C (840 °C was the measured edge temperature, the internal temperature is estimated to reach ~880 °C). The rate of insertion was kept slow enough to prevent a very high-power spike, with the goal of peaking at 5 kW thermal power output.

Loss of heat removal

The final transients were conducted to determine the response of KRUSTY to a complete loss of active heat removal and abrupt restart of the convertors and simulators. Fig. 9 plots these transients, which occurred from $T = 20$ to $T = 28$, ending with the reactor scram. The most important of these tests was the removal of active cooling. During this transient, the core temperature quickly increased, decreasing reactivity, causing the power to drop shortly thereafter. During the transient, the average core temperature reading rose about 16 °C showing overall safety of the reactor design. These experiments show that reactor could not have a core melt accident caused by total failure of the heat removal system (i.e. the Stirling convertors).

Final assessment of the KRUSTY experiment

The KRUSTY experiment demonstrated that the Kilopower reactor concept would perform as designed. The desired performance parameters versus the outcome of the experiment are shown in Table 1.

Kilopower applications overview

Reliable, long-life power systems are among the basic requirements of any space exploration mission. NASA has relied upon plutonium-based radioisotope power systems (RPS) (Bennett, 2021) for decades for deep space missions like Pioneer, Voyager, Galileo, and Cassini, where solar power was not a feasible option. RPS systems provide reliable and lightweight power for missions in the 100-W_e to 1-kW_e range. Kilopower was developed to address NASA's needs in the 1- to 10-kW_e range, potentially up to ~30-kW_e with the same technology. For science missions to the outer planets, Kilopower could enable future flagship science missions and exploration precursor missions that may not otherwise be possible. For human missions, Kilopower could provide surface power system modularity, mobility, and reliability for Lunar and Mars surface missions.

Example Kilopower missions

Kilopower was designed for numerous missions both for planetary surface power and for deep space missions. Examples of both types of missions are presented below. The detail presented varies by mission. The analysis for these missions was done by different NASA centers with varying objectives. As such, consistent mission data was not available.

Example planetary surface missions

Mars

The first mission discussed for Kilopower was as a power supply for human missions to Mars. The Human Spaceflight Architecture team has assumed four 10-kW_e power systems for the 40-kW_e first human Mars surface mission requirement, with a possible fifth unit for added reliability.

Moon

The other planetary surface mission that has been proposed for Kilopower is a technology demonstration mission for the lunar surface. A 10-kW_e reactor has been proposed for deployment in the mid-2020s. The reactor would be part of a lander demonstration. Potential missions could be as either a power source for rovers or a power source for an In-Situ Resource Utilization (ISRU) demonstration unit.

Table 1 KRUSTY experiment outcomes.

<i>Event scenario</i>	<i>Metric</i>	<i>Experiment</i>	<i>Status</i>
Reactor startup	<3 h to 800 °C	1.5 h to 800 °C	Exceeds
Steady state performance	4 kWt at 800 °C	>4 kWt at >800 °C	Exceeds
Total loss of active cooling	<50 °C transient	<15 °C transient	Exceeds
Converter efficiency	>25%	>30%	Exceeds
Converter operation	Start, stop, hold, restart	Start, stop, hold, restart	Meets
System electric power turndown ratio (operational range)	>2:1 (half power)	>16:1	Exceeds

Example deep space missions

The National Academy of Sciences 2013–22 Decadal Survey (National Research Council, 2011) identified many RPS-powered mission concepts that could be performed using a 1-kWe or 10-kWe reactor. A reactor-powered Nuclear Electric Propulsion (NEP) system for these missions would support an increased science payload mass, increased communications rate, and shorter flight times. With a lesser increase of science payload mass, an NEP system could also support a greater than 40-year mission life-time. A 10-kWe NEP system can provide the power for multi-year outer solar system robotic missions, such as a Neptune+Triton Orbiter, Chiron Orbiter, Titan+Enceladus Orbiter, Pluto Orbiter and Kuiper Belt Object Orbiter (Casani, 2019).

Conclusion

Kilopower is designed to provide from 1 to 30 kWe with a simple and robust technology, and can ultimately evolve up to 5 MWe with a few manageable stepwise technology changes. The Kilopower technology development and demonstration project has stepped through a build-and-test sequence of phases that completed the demonstration of a flight-like system operated at prototypic conditions in a relevant environment (Technology Readiness Level (TRL) 5) for space science and human exploration power needs. The innovations of monolithic reactor fuel and heat pipe heat transfer have been applied and tested through all challenges. The nuclear steady-state and transient test of this space reactor is the first such experiment performed in the United States since the 1960s. This accomplishment establishes the readiness of nuclear-tested fission power system technology for affordable, lightweight application to human and robotic exploration wherever it is needed.

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Nuclear Thermal Propulsion: Benefits and Challenges

David I. Poston, Los Alamos National Laboratory, Los Alamos, NM, United States

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Introduction

Nuclear Thermal Propulsion (NTP) describes a class of technologies that use nuclear power to thermally heat a rocket propellant. Traditional rockets utilize a chemical reaction to heat the rocket propellant, whereas NTP uses the heat from nuclear reactions. An NTP concept is usually referred to as a Nuclear Thermal Rocket (NTR), although both terms are often used interchangeably. In general, and in this encyclopedia, the term NTP is associated with fission reactors that heat a hydrogen propellant. Fusion, anti-matter, pulsed-nuclear, or other nuclear powered concepts might still be technically labeled as NTP, but they are usually referred to under the category of advanced nuclear propulsion ([Howe, 2021](#)). Advanced concepts may offer performance advantages (on paper), but NTP has the distinct advantage of having established nuclear physics and to some extent proven technology.

Nuclear thermal rockets have been dreamed of since the discovery of fission in 1938. The dream of NTP came close to reality during the Rover and NERVA programs, which occurred in the US from 1955 to 1973. Substantial progress was made as 20 different reactors were designed, built, and tested over that period; unfortunately, none of them reached the flight-development stage, and no NTP reactors have been built since then. Several smaller scale NTP programs in the US and Russia have been conducted over the past 50 years. A detailed look at the history of NTP is presented in the next chapter in this encyclopedia ([Graham, 2021](#)).

Benefits of NTP

The benefit of NTP is rooted in the rocket equation.

$$\Delta v = v_e * \ln \left(\frac{m_{initial}}{m_{final}} \right) = I_{sp} * g_0 * \ln \left(\frac{m_{wet}}{m_{dry}} \right)$$

Specific impulse (I_{sp}) is a measure of how much thrust occurs per mass of propellant (with a unit of seconds), which is directly proportional to the rocket propellant exit velocity (v_e). Delta- v (Δv) is a measure of useful acceleration. Therefore, first order rocket performance is a linear function of I_{sp} ; i.e., double the I_{sp} = double the Δv . However, the benefit of higher I_{sp} can be much greater depending on the proposed mission; the more ambitious the mission (higher delta- v) the greater the benefit of higher I_{sp} . This is because of the logarithmic mass ratio in the rocket equation of initial (wet) mass vs. final (dry) mass: wet mass (m_{wet}) is mass of spacecraft plus propellant, dry mass (m_{dry}) is the spacecraft without propellant. For ambitious missions, a lower I_{sp} will require exponentially more propellant (wet mass) simply to push the additional propellant needed, to the point where either the launched mass is nearly all propellant, or the mission cannot be performed at all. In these cases, a substantial increase in I_{sp} can be provide more than a factor of two benefit.

As a heated gas is exhausted through a nozzle, the I_{sp} (exit velocity) is a function of the gas temperature, molecular weight, and ratio of specific heats ((specific heat at constant pressure)/(specific heat at constant volume) usually referred to as gamma). The ratio

of specific heats is relatively similar for all gasses, so in general I_{sp} increases with the square root of temperature and decreases with the square-root of molecular weight. Hydrogen (H_2) is the only gas with a molecular weight low enough to provide performance better than a traditional chemical rocket; thus it is the only propellant worth considering for NTP (unless perhaps a planet/moon has a large quantity of available Helium). The I_{sp} vs. temperature for H_2 propellant is shown in Fig. 1.

The I_{sp} of a hydrogen-oxygen chemical rocket is ~ 450 s, so Fig. 1 can be used to show that a H_2 chamber temperature (mixed-mean reactor outlet temperature) needs to be ~ 800 K to equal the best chemical rocket. Since NTP is an unproven technology then it would at least have to achieve at least 500 s to even be worth a second look. However, an NTR will likely have a much heavier dry mass than a chemical rocket, so the I_{sp} of an NTR may need to be > 600 s to be useful, and perhaps > 700 s to be cost effective on a per mission basis (considering that at NTP unit will likely cost more than a chemical unit). Then, given the high cost/difficulty of development and testing, a target of > 800 s is generally the low end of where NTP is considered worth pursuing. Of course, these targets are all generalizations, and could change by up to 100 s either direction depending on the intended application, the development cost, and the unit cost.

The highest gas outlet temperature of any fuel element during the Rover/NERVA program was 2550 K (Koenig, 1986), which would imply an I_{sp} of 850 s; although this was at rather low power density and was not an integrated “rocket” test. The XE’ test was closest to a potential flight-prototypic rocket test of any Rover/NERVA test, and had an fuel exit temperature of > 2400 K; this would imply an I_{sp} of 810 s, but the actual overall rated I_{sp} was only 710 s (Finseth, 1991). In order to keep things simple, they did not attempt to implement some design features that would have improved system I_{sp} closer to 800 s.

Most NTP studies since the Rover/NERVA program have targeted an I_{sp} range between 850 and 950 s, because there is almost no experience with fuel materials that can survive > 3000 K temperatures (although there are several proposed fuels that could). Plus, engineering realities dictate that the gas outlet temperature of at least 100–200 K below the maximum fuel temperature. Highly advanced concepts have been proposed with liquid or gas fuel (with corresponding cooled walls) to increase I_{sp} well above 1000 s. Gas or plasma core rockets have been theorized with I_{sp} as high as 3000 s, but these concepts are a long way from reality. For the remainder of this article, the discussion will generally pertain to an NTR with an I_{sp} of ~ 900 s, which provides a factor of two improvement over hydrogen-oxygen chemical propulsion.

Human Mars missions are the kind of ambitious missions for which NTP can be very attractive. For one class of mission, conjunction class (where at launch, Mars is on opposite side of sun from Earth), the primary benefit of NTP is a reduced number of earth-to-orbit launches required to ultimately get the payload to Mars. Given the large payload of a human Mars mission, up to a factor of two fewer launches could provide a great cost savings. Also, less in-orbit assembly and fueling could make the mission architecture simpler and thus even less expensive. The net benefit of NTP for a conjunction class mission depends greatly on the cost of heavy-payload earth-to-orbit launches. Currently (in 2020) there is a major effort underway to reduce heavy-lift launch costs.

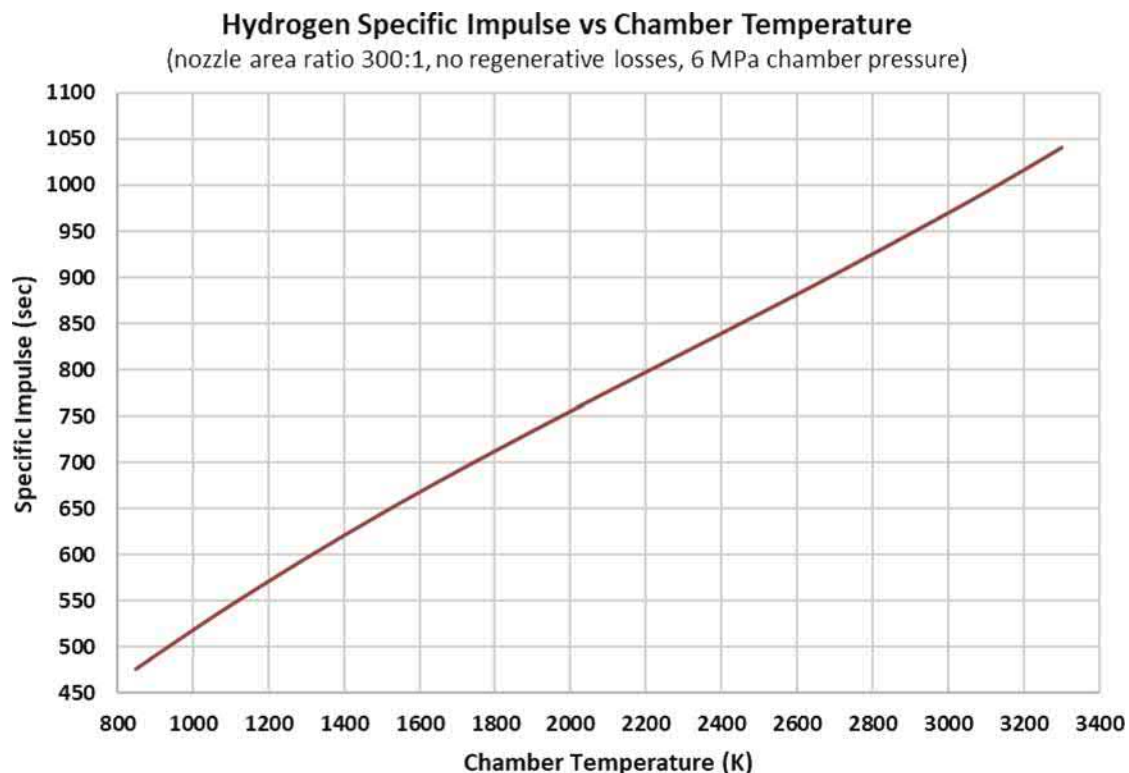


Fig. 1 Rocket specific impulse versus chamber ($\sim$ reactor outlet) temperature.

SpaceX has made the most progress to date, and if SpaceX, Blue Origin, NASA, etc. succeed in substantially reducing heavy-lift launch costs, then the value of NTP for conjunction class missions will be significantly diminished. If that occurs, then NTP might only be worthwhile if a human-rated (i.e., very high reliability) system can be developed at relatively low cost (perhaps for billions of dollars as opposed to the perceived cost of 10s of billions). Fortunately, NASA has had a significant program underway since 2011 to determine the “affordability” of NTP, and it’s ramping up to an even larger effort in 2020. Thus, by the mid-2020s we should have a good answer concerning both issues: the cost of heavy-lift and the feasibility of affordable NTP.

The other class of human Mars mission proposed for NTP is an opposition-class mission (where at launch, Mars is on same side of sun from Earth). Opposition-class mission are shorter; the time spend on Mars is only ~ 1 month vs. ~ 18 months for a conjunction class mission. This shorter time may be attractive for early exploration missions to minimize time at risk for the astronauts, at the expense of potential exploration advantages of a longer stay. However, for an opposition-class mission the return trip from Mars is much more difficult and of longer duration. For chemical rockets, the time in space is almost a year longer, which eliminates most of the time advantage of the opposition-class mission (depending on the astronaut risk in space and/or on the surface). NTP however, has the potential for a much quicker return trip, which is probably the biggest advantage of NTP for exploration in the near future.

For an opposition class mission, the value of NTP depends on how much astronaut risk is reduced by shorter trip times. Risks to astronauts can generally be binned into four areas: zero-gravity risks, radiation risks, away-from-home risks, and acute-mission risks (caused by system failures). Zero-gravity effects are becoming well known. The most pronounced effects of zero-g are loss of muscle and bone mass, which can be partially mitigated by exercise with specialized equipment, and consequences of the redistribution of fluids in the body (e.g., vision impairment, cardiovascular issues, etc.), which can be partially managed by constrictive clothing. There are many other potential zero-g issues that could provide significant health consequences, which will learn about as human beings spend more and more time in space. Fortunately, astronauts have already accumulated substantial time in zero-g (technically micro-gravity, but essentially the same) over the past decades, which has not only shed light on the risks, but in many cases has led to mitigations of to zero-g risks. To date, no health issues have been identified that would preclude a human mission to mars, or effect a crew in a way that would likely jeopardize a mission. The biggest issue is to allow astronauts time to recover and adapt, so they can perform their duties once they reach Mars.

The most emphasized risk for a human Mars mission usually radiation, even though there is no compelling evidence that dose received from the astronauts would create significant health issues. It is expected that a Mars astronaut would receive on the order of 1 Sv during a Mars mission (unless a significant solar storm occurs and an adequate storm shelter has not been provided). This dose rate is well below the threshold of any acute/deterministic radiation effects; although astronauts have developed cataracts at a higher rate than the general public, which could be due to radiation, zero-g, or a combination of factors. The only other potential risk from radiation is genetic damages that might lead to cancer. The highly conservative (and greatly disputed) linear-no-threshold model indicates that 1 Sv of accumulated dose creates a $\sim 10\%$ chance of developing cancer. Even if true, this has almost no chance of occurring during the mission (5–15 year latency for leukemia and 10–60 year latency for solid tumors), and most likely much later in the astronaut’s life—if they are lucky enough to survive the mission and celebrate life as a hero to old age. More-so, cancer is becoming less and less fatal (33% survival in 1975, 25% survival in 2010, etc.), and by the time we eventually send an astronaut to Mars mortality rates may be very low. In any case, these radiation risks are insignificant compared to the risks taken by any significant explorer or pioneer, including NASA’s previous test pilots and astronauts.

Away-from-home risks, or more properly labeled away-from-civilization risks, have existed for every exploration campaign in human history. Once the astronauts leave earth they no longer have access to “standard” medical care; therefore, a condition that might be easily treatable on Earth may end up fatal on a spacecraft or on the Mars surface. There is also concern about the psychological effects of years of isolation in close quarters with a handful of crewmates. Several experiments have been conducted to study human behavior under simulated Mars mission conditions (Binsted et al., 2010); although there is no way to simulate the certainty of having zero chance to set foot on Earth for months/years. The reduced trip time offered by NTP would definitely be beneficial in this regard, although there is no clear evidence that any of the above issues will provide a major risk to a human Mars mission.

All of the above risks (zero-g, radiation, and away-from-home) appear to be very small when compared to acute-mission risks that can be caused by system failures. The “simple” act of launching into Earth orbit on a relatively new heavy-launch vehicle will likely be more risky than the combined effect of all of the above risks. A highly utilized spacecraft, the Space Shuttle, performing a relatively simple mission killed $> 1\%$ of astronauts. On a Mission to Mars astronauts can die acutely via: launch, Mars trajectory burns, Mars capture, landing, ascent, docking, Earth-return burns, Earth arrival and landing (and perhaps a Venus flyby or Mars aero-capture). That’s at least eight distinct risk points that each would be extremely hard to get below a 1% failure rate, and realistically may average 2% each (and maybe even higher if a very new technology is used like NTP). Then at any time during the mission there are a slew of risks, most notably loss of power supply, life support, vessel/space suit integrity, fire, etc. Not to mention risks due to human error by crew members or ground personnel. The sum of risks for a human Mars mission are daunting, but there will surely be no shortage of volunteers willing to take that risk for the benefit of us all, let alone the chance at fame and fortune.

The increased safety of a shorter trip time has to be weighed against all risks, and the cost of developing a useful NTP system. This comparison (for an opposition class mission) is less straightforward than it is for conjunction class missions, where the cost of developing NTP can be weighed against the increased launches and space integration required for a chemical mission. Noting that the cost of developing an NTR is a very strong function of requirements, most notably I_{sp} and thrust-to-weight, and potential policy issues concerning highly-enriched uranium and/or launch accident safety.

Another way that NTP can be used to increase safety is by providing increased ability for mission abort. For any given mission, a higher I_{sp} reduces the mass penalty to carry propellant margin to allow unplanned burns to get home quickly if an emergency occurs. Some NTP missions may even provide enhanced mission abort options with a minimal mass penalty (Joyner, 2017). Another potential advantage is that higher I_{sp} can enable potentially safer higher delta-V missions to Mars, e.g., by avoiding gravity flybys or aero-capture. This would increase propellant mass versus the “nominal” mission, but higher delta-v option might not even be possible for a chemical mission. In general, these benefits would need to be traded against the mass benefit for a conjunction-class mission or the nominal return travel time for an opposition class mission. The value of this flexibility is highly dependent on mission trajectories; e.g., some missions could more easily benefit from a mission abort capability than others.

The benefits of NTP also have to be weighed against other technologies than chemical propulsion; this includes the advanced technologies mentioned earlier, but most notably electric propulsion. Electric propulsion allows much higher I_{sp} than NTP, perhaps up to 10,000 s as compared to 1000 s. Electric propulsion is used regularly in space, but is generally very low thrust, so at the moment not very attractive to human missions. Solar electric propulsion is promising near the Earth, but becomes less and less attractive further out towards Mars and beyond. For a human Mars mission, Nuclear Electric Propulsion (NEP) is the other potential candidate besides NTP and chemical propulsion. A discussion of Nuclear Electric Propulsion is provided in another chapter in this encyclopedia (Morrison, 2021).

Ultimately, if NTP can be successfully developed, there are many other potential benefits beyond a human Mars mission. In general, NTP will allow missions that are twice as ambitious as chemical propulsion, i.e., twice the delta-v. Higher performance systems could potentially carry humans to the moons of Jupiter, and perhaps beyond. Low mass, low thrust system could benefit deep space science and maneuverability in sis-lunar space. For grand-scale missions, NTP could be used in tandem with higher I_{sp} nuclear electric propulsion (NEP) system, to provide high thrust when needed and/or escape gravity wells. Bimodal NTP systems could offer the benefit of providing both thrust and electric power. The potential benefits on NTP are great, but like all engineered systems, the desire for higher performance (higher I_{sp} , lower mass, greater flexibility) comes at the cost of more difficult development and technical risk.

Challenges of NTP

The benefits of NTP are impressive, so ultimately the attractiveness of NTP depends on the cost and effort of developing and utilizing a reliable system. In some ways, NTP has certain development advantages: e.g., short reactor lifetime and minimal radiological consequences of an operational failure in space. In other ways, nuclear thermal rockets are far more challenging than any other reactor applications: e.g., high temperature, high power density, and the use of a volatile coolant (H_2). Overall, the technical challenges are daunting, but the various successes of the Rover/NERVA program provide some confidence that success could be achieved; although this may depend greatly on how similar any specific concept might look to a Rover/NERVA system, and whether we have the will to pursue and complete a program on the vast scale of Rover/NERVA.

Technology challenges

Reactor fuel

The nuclear fuel for an NTR is a significant technology challenge. The fuel temperature required for a “useful” NTR is probably 2600–3000 K, which is significantly higher than any other reactor operated in the past 50 years (LWR power plant fuel has a peak internal temperature of ~ 2000 K, and the fuel structure/cladding operates at < 1000 K). In addition, the fuel power density required for acceptable rocket performance is generally 5000–10,000 W/cc, which is an order of magnitude higher than other power reactors (LWRs are on the range of 100 W/cc, advanced test reactors up to ~ 500 W/cc). The fuel, and associated coatings or internal structures, must also be resistant to erosion and corrosion to hydrogen flow at high velocities and high temperatures.

The raw fuel material itself is not necessarily the challenge, as there are several options that have melting points > 3000 K (UO_2 , UN, UC, and many more). The challenge is being able to fabricate the fuel (and internal structure) in a way that not only accommodates heat removal, but can survive extreme combination of stress, temperature and potentially vibration. The Rover program made a lot of progress in developing a carbon-based NTR fuel, the Pewee reactor fuel operated at a peak temperature of 2750 K (Finseth, 1991), but after 20 nuclear reactor tests there were still issues with erosion, corrosion and cracking. Many present day concepts propose using different fuels, with less heritage but perhaps better capability. Once a fuel is produced, then environmental testing will have to be performed to see if it can survive the proposed NTR conditions. A discussion of historical and proposed fuel types can be found in the following chapters in this encyclopedia Graham (2020) and Poston (2021).

Reactor structure

Reactor structure for an NTR is unique when compared to any other class of reactor. The biggest challenge is that during operation the reactor is very hot on one end (likely 2500–3000 K) and very cold on the other (perhaps 200 K), and there are other states (during startup and shutdown) where the entire reactor might be very hot or very cold. The various possible thermal states create a complex combination of expansion stresses, at temperatures where material properties might be structurally challenged (e.g., lack of strength at very high temperature or lack of ductility at very low temperatures). Hydrogen also has the potential to embrittle the structural materials, in addition cause potential corrosion/erosion.

In addition to steady-state challenges, the structure must also be able to survive the potential thermal shock caused by the injection of extremely cold coolant into a warm reactor, and transient stresses caused by rapid heat up. The structure must also be able to survive launch loads/vibrations of a heavy-lift rocket, and there may be structural requirements to prevent criticality that could be caused by an accidental impact. Finally, the structure must allow for the routing and distribution of high-pressure hydrogen flow to various flow paths throughout the system (in some cases two or three different flow streams in different directions), and all of the associated stresses.

The Rover/NERVA tests had numerous structural failures during early testing. Koenig (1986) lists 16 “significant structural anomalies” that occurred over the 6 design iterations of the NERVA NRX test series, including cracking, flaring, rubbing, breaking of various components. They were able to learn and resolve issues with continued design and testing, but any new NTR concept may likely require several ground test iterations as well.

Heat transfer and power balance

One of the design goals of an NTR is to warm the H_2 propellant as much as possible, i.e., maximize the mixed-mean outlet temperature. In a fission reactor, some regions of the fuel produce higher power than others—the ratio of peak power to average power is referred to as a peaking factor. A typical NTR heats the H_2 coolant by ~ 2500 K from core inlet to outlet; thus, if the nominal flow rate was used to cool a fuel region with 10% more power (i.e., a peaking factor of 1.10), it would have an outlet temperature 250 K higher than the average flow. To prevent this from happening each flow channel (or at a minimum each fuel assembly) must be orificed to match the flow rate to the power. This can be mitigated at the nominal design conditions (depending on the uncertainty in power deposition and pressure drop), but this can be complicated and diminish performance in cases where reactor peaking factors change with burn-time and control element position.

Nuclear power deposition also occurs outside of the fuel, as neutron and photons interact with other materials. Given the high power of most NTR concepts (> 100 MW), other reactor components and regions must also be actively cooled by the H_2 . Some of the regions that require adequate cooling are the neutron reflector, shield, vessel, and nozzle. In some cases too much cooling could be as detrimental as too little cooling, due to the need to balance reactivity feedback (Deviation from the state of a critical core—a core that sustains a fission chain reaction.), as well as provide the desired coolant temperature to the turbopump. Therefore, the power deposition and flow balance to all components must be well matched and have minimal uncertainty.

An NTR will generally operate as close to material limits as possible to maximize I_{sp} . It is important that heat transfer paths are robust and predictable, because small changes in thermal resistance can cause big changes in temperature gradient (due to the extremely high power density). If dissimilar materials or coatings are used, it is important to ensure that their interfaces will maintain the expected thermal resistance throughout all operating regimes. The effect of system strain/deformation on thermal resistance and heat transfer must also be considered, for all possible thermal/mechanical states of the system.

The most difficult heat transfer/balance challenges occur if the reactor core contains sub-cooled regions (i.e., regions that are cooled to be at lower temperature than the primary fuel and coolant), which might contain structure and/or moderator. In these cases an insulating material must be used to prevent heat from “leaking” from the fuel to the colder nearby region. Heat leakage not only decreases I_{sp} , but also requires more coolant flow through the sub-cooled region (which increases pressure drop and/or requires more flow area). The insulating material must be effective at very high temperature, and also meet neutronic requirements, by allowing a compact geometry and not absorbing too many neutrons. The Rover/NERVA program spent considerable time developing high temperature insulation, with fairly good success, but they also acknowledged that it would be very hard to predict performance in a system application without actually testing it within the system (Wagner, 1992). One of the biggest contributors to uncertainty they identified was hydrogen infiltration into the insulation, which could change conductance by up to an order of magnitude. Unfortunately, it is extremely hard to guarantee that H_2 would be kept out of a region at $\gg 2000$ K, and also hard to ensure that it would be fully saturated with H_2 (while also precluding convection); therefore, there will be very large uncertainty in insulation performance simply based on the unknown effect of H_2 on thermal conductance alone. It is important that the thermal performance of the insulator is well-understood, because the thermal resistance has a major effect on balancing the flow, temperature, and reactivity of the sub-cooled region (as mentioned previously, too little insulation will overheat the region and too much insulation will starve the pump from power).

Decay power removal creates another potential problem involving power balance and the use of a subcooled moderator. A concept with a moderator must keep that moderator cool after shutdown, assuming that the NTR is planned to be used for more than one burn. This creates two issues. First, the reactor must be actively cooled for a longer period on time; i.e., until the decay power decreases enough to match the lower passive heat rejection that occurs at the lower core temperature. Second, and more importantly, the cooling efficiency (J/kg) of the post-burn cooling flow will be substantially lower, because it will have a lower outlet temperature. Any NTP concept will require a certain amount of “wasted” propellant to remove decay power—not totally wasted, but the I_{sp} will be lower and the thrust profile will be unpredictable and sporadic. The problem created by an internal moderator (or any low temperature core region) is the existence of the thermal path from the fuel through the insulation. At nominal high temperature, the flow through the insulation might be 5–10% of the nominal power. At decay power levels ($\sim 1\%$), the fuel will quickly lose its high temperature via heat transfer through the insulation, and settle at a temperature slightly above the moderator/block/tie-tube temperature; thus the outlet coolant can’t be heated to a temperature significantly higher than the moderator. In a typical mission scenario, the total removed Joules of decay energy might be $\sim 2\%$ of the Joules of the entire active burn. If the use of a moderator requires the core temperature to be four times lower for decay mode versus nominal mode, then the amount of decay-mode propellant (“wasted”) would be 8% ($4 \times 2\%$) of the nominal flow; severely reducing

effective I_{sp} . An unmoderated NTR has the benefit of allowing the core to remain at or near the nominal max operating temperature. Although in that case, the issue might be keeping the vessel or structure cool, which could also decrease the efficiency of the decay power flow (but likely by a much smaller amount).

Individually, each of the heat transfer issues are difficult engineering challenges, occurring at extreme temperatures and power densities by most standards. What makes them even more difficult is the combination of uncertainties and reactivity feedback effects, and the difficulty of testing heat transfer without a full integrated test (because nuclear power is the only way to produce high enough power deposition in the fuel and other components).

Reactor instrumentation and control

Instrumentation and control is a challenge for almost any nuclear reactor, and NTP adds another layer of complexity and uncertainty. Establishing the reliability and accuracy of diagnostics in nuclear reactors is difficult, but 50+ years of experience has led to adequate solutions for terrestrial reactors (often via lots of redundancy). An NTR will have radiation fluxes and temperatures that far exceed any existing reactor, and thus there will be uncertainty regarding the readings of any instrumentation. In-core temperature measurements may not be possible at all, which would present a major problem for any system that has complex reactivity feedback and sub-cooled regions (i.e., more than one significant temperature reactivity feedback mechanism). The desire for low-mass may also limit the number of diagnostic instruments, possibly preventing diverse and redundant readings. In addition, the diagnostics must operate in the vacuum and extreme temperature of space, survive launch vibrations, and operate during the vibrations created when the NTR is thrusting.

Most NTRs propose to use rotating control drums to change reactivity, where one side of the drum contains neutron reflector and the other contains neutron absorber. The high power deposition in the drums/reflector combined with very cold coolant has the potential to create warping that might cause the drums to bind. Thrust vibrations may also effect clearance and rotation, as well as the thermal cycles between high temperature operation and the extreme cold of space between restarts. The rotation mechanisms must be robust to launch loads and prevent inadvertent rotation during potential accidents, and the motors must survive the temperature and radiation environments. Also, in some NTR concepts, active, rapid response control requires high speed drum movements.

The final challenge is developing a system to properly and reliably command the control mechanisms based on the diagnostic information. The biggest difficulty might be deciding which instruments to “believe,” because experience shows that all readings are not normally in agreement (some signals may be faulty while other signals experience drift). Usually when a reactor starts up, there is a long period of preliminary operations to flush out the system and learn which signals are bogus and what the remaining signals are actually inferring. An NTR will not have much opportunity, if any, to do this, because hydrogen must be flowing to diagnose the system and there is a strong desire not to “waste” propellant before going to a fully operational state. Then, after the input signals are processed, a computer must decide what actions to take and convey signals to the drum motors. Adequate shielding will need to be provided to the computer (likely more than 1 computer for redundancy) to avoid a single-event upsets caused by radiation.

Turbopump power and control

A high-thrust, high- I_{sp} NTR requires high volumetric flow rates and high pressure drops, because flow area is limited by nuclear criticality and high velocity is needed to enhance coolant heat transfer. This combination of high flow and pressure drop requires considerable pumping power. A turbopump is generally the only practical pumping option, except for extremely low-thrust systems where an electric pump, chemically driven pump, tank pressure, or other options might work. The pump hardware should not present a major technology challenge, because hydrogen turbopumps have been successfully developed and used for decades (including the space shuttle main engine). Radiation damage or power deposition could be an issue for seals and bearings, but this is not expected to be a major problem.

Power for the turbopump (to heat the turbine feed propellant) must come from the reactor: directly or indirectly, except for startup where an external power source could be provided or tank pressure can be sufficient. Some of the power can be provided by the regenerative cooling of the nozzle, plenum wall, reflector, shield, etc. For a very low pressure drop design this regenerative power might be adequate to power the turbopump (depending on the flow rate and the flow resistance though other components), but criticality requirements usually drive NTR designs to minimize flow area, and the resulting pressure drop is high enough to require substantial pumping power. The vast majority of NTR concepts require an additional source of heat to power the pump turbine.

During the NERVA program, there was only one system test that used a turbopump, the XE' test (Aerojet, 1969), where a hot-bleed cycle was used to power the pump. A hot-bleed cycle removes hot H_2 from the nozzle region to provide power for the turbine, and it is mixed with cold flow to make the turbine inlet temperature manageable. The hot bleed cycle was used for XE' because it was deemed easier than a topping cycle (where the turbine feed flow passes through separate channels in the core); although, pulling $> 2000\text{ K } H_2$ from a flow stream is certainly not “easy.” The major drawback of the hot-bleed cycle, combined with an open-cycle turbopump (where the turbine exhaust is ejected instead of recycled), is a reduction in I_{sp} . NERVA documentation states that they settled for the simpler, lower performance to complete XE', and would pursue a closed topping cycle in the future. A topping cycle is difficult because special sub-cooled flow paths must be integrated into the reactor and plenum design. These cold flow paths must be maintained next to extremely hot components (hard to insulate, potential for thermal stresses), they add regions of relatively high-density H_2 in the core (complicates reactivity), and plumbing must be integrated to provide for the inlet/outlet of this flow (difficult to engineer and never demonstrated).

Perhaps the biggest challenge for an NTR turbopump will be control. Most standard turbopumps, from cars to rockets, have a relatively direct and predictable response to increased flow—increased flow causes more combustion and a proportional increase in power that returns to the turbine. A subsequent increase in passive losses and reduced combustion efficiency dampen the system to a stable settling point. The dynamics that drive an NTR turbopump will be totally different, and depend greatly on the reactor design. The immediate response in an NTR may be similar to a classic turbopump, in that increased flow will cause more heat removal from the reactor, thus feeding a proportional increase in power back to the turbine. The difference is how the power and temperature of the reactor change due to this increased pressure/flow, and how that will feedback to the turbine inlet power. The response of the reactor to a flow change is not proportional or linear, it is exponential. For most concepts, increased cooling of the reactor will increase reactivity, which will cause an exponential increase in power (or vice-versa), until the power of the reactor causes a temperature change to counteract that reactivity. For a neutronically simple reactor (one with simple reactivity coefficients, predictable heat transfer, and minimal neutronic worth of the H_2 gas), the feedback to turbine inlet power should be manageable; in fact it was essentially proven by the XE' test. The reactor for that NERVA test was indeed neutronically simple. Also, as mentioned previously XE' used hot-bleed cycle, plus the turbopump was open-cycle (turbine inlet flow was exhausted, rather than returned to the system flow); both of these features simplify dynamics even further. Alternatively, a system with large and/or uncertain feedback coefficients, uncertain heat transfer, and/or high H_2 reactivity worth, will have a highly unique and unpredictable response to the turbine. In that case turbopump control might be very complicated.

Cryogenic hydrogen storage

Liquid hydrogen boils at very low temperature, and it is usually stored at 20 K. It is very hard to keep anything with a view of the sun that cold in space, despite great progress over the years in developing insulation and reflective surfaces. The difficulty of storing Liquid Hydrogen (LH_2) is best evidenced by the fact that no on-orbit or deep-space satellite/rocket uses liquid hydrogen as a fuel, despite the substantial I_{sp} advantage of hydrogen. If in-space cryogenic storage was not a difficult engineering challenge, then in-space rockets would use H_2 instead of storable or hypergolic propellants.

For NTP, the refrigeration technology must be very reliable, long-lived and meet mass requirements; other than that, the technology should not be uniquely challenging because of experience with hydrogen storage on Earth and in-space cryogenic cooling. One additional complication is that nuclear radiation from the reactor will heat the hydrogen in the tank. In theory, this could be beneficial to keep the tank pressurized as the H_2 is discharged; however, there will be a good deal of uncertainty in the actual amount of power deposition (depending on the path and various shielding between the reactor and the tank). It might be better to provide shielding to ensure the H_2 does not overheat, as opposed to planning on a pressurization effect (although shielding requires mass).

Programmatic challenges

Nuclear launch safety

The possibility of an accident during launch or reentry from orbit provides a unique challenge for NTRs and all space reactors. From a technical perspective, the risks of radiation to the public are negligible (McClure, 2020), but public and political biases specific to “nuclear” could present a significant hurdle.

All fission systems that use uranium fuel are minimally radioactive prior to operation. There is negligible radiological risk to the public if there was a launch explosion (or other accident) that dispersed all of the fuel material into the air (the radiation risk to on-site personnel would also be negligible). The key to launch safety is ensuring that the reactor does not accidentally “turn on” (go critical and produce radioactivity) and subsequently provide a harmful radiation dose to anyone. Fission reactors are designed to critical with intentional/coordinated movements of control elements (to insert reactivity). Mechanisms must be engineered to ensure that the elements do not move inadvertently during any credible scenario; this is standard practice for every type of nuclear reactor, but space reactors have the additional issue of potential high speed impact and/or landing in the ocean.

Preventing inadvertent criticality is part of any space reactor design process. Some space reactor concepts can easily be designed to be inherently subcritical in all scenarios, including high speed impact and water flooding. This can be achieved by (1) minimizing the core volume that can be flooded with water, (2) keeping the core geometry very compact, (3) adding additional movable neutron absorbing safety mechanisms, and/or (4) incorporating materials which preferentially absorb neutrons during flooded/moderated conditions (commonly referred to as spectral shift poisons). The challenge for NTP is that a high flow area is needed for adequate heat transfer (a problem for option 1), the power is generally high enough to require a relatively large core volume (a problem for option 2), and the incorporation of moving mechanisms to prevent water immersion criticality is extremely difficult (a problem for option 3). If an NTR uses a moderated neutron spectrum, then an attractive system cannot be practically designed with spectral shift poisoning (a problem for option 4). The only NTR concept that can easily prevent inadvertent criticality during water immersion (by using spectral shift poisons like tungsten or rhenium) is an unmoderated reactor. Unfortunately this concept would likely have to use Highly Enriched Uranium (HEU) to have a practical mass, so in this case this safety challenge would have to be weighed against safeguards challenges.

Even if a concept does go critical (produce radioactivity) during launch/reentry accidents, the radiation risk to the public should still be extremely low. On land, it is more likely that a person would be harmed by the shrapnel of the reentering reactor and spacecraft rather than any possible radiation. During water immersion, water provides an excellent shield to radiation, such that a person would have to be within a few meters of the reactor to receive a harmful dose. Given the size of the ocean, the chance of anyone

being so close is miniscule; the only danger might be to personnel, if they are sent to retrieve the reactor. The launch approval process in the US allows for a probabilistic approach that could include these factors, so from a technical perspective launch safety should not be a major challenge.

The real challenge will be overcoming public and political bias to radiation and nuclear reactors. In this respect, an NTR concept that inherently prevents inadvertent criticality would be preferable than one that doesn't. The alternative is to try and convince decision makers that it's ok if the reactor goes critical if it falls in the ocean. It may be easy to show that it is almost certain that nobody will be harmed, but they still might not be willing to take the political risk of a nuclear "accident".

Nuclear ground testing

The biggest programmatic challenge of NTP would probably be performing a full-power nuclear ground test, because of the combination of high cost and the difficulty of gaining approvals. A facility that could adequately test an NTR would likely cost billions of dollars. The Rover/NERVA cost >\$10B in today's dollars, and a good fraction of that was spent on building and testing. More so, they had far less regulation, paperwork, and oversight than we have today, and had much more expertise and infrastructure.

The manufacturing and testing infrastructure that existed at the end of the NERVA program makes any nuclear development capability that exists today look miniscule. There were hundreds of design, manufacturing, fabrication, testing and system experts working on NERVA in the 1970s. Not only are most of these people gone, there has been relatively little knowledge transfer since this was before the era of digital record keeping, and use of the technology has not been continuous. Back then, nuclear development was more art and elbow-grease, as compared to science and CPUs; they built, tested and tweaked things and used their ingenuity to make things work.

The ability to get government approval for system testing is a major risk to an NTP program (whether the regulator is the NRC or DOE). NTRs have very small temperature margins and extremely high power densities, which are off-the-charts relative to any other successfully deployed reactor (certainly far beyond the scope of any potential regulator). The adiabatic heat-up rate for a typical 25-klb (klb = 1000 pounds of force = 4448 Newtons) thrust NTR might be ~1000 K/s. This is 50 times higher than a typical PWR, and higher still than a typical BWR. The implication is that if cooling is lost during powered operation, an NTR core would melt within seconds. Decay power is also a significant concern; 1 h after shutdown the adiabatic heat-up rate can still be >1 K/s. Another concern is that the testing also involves high volume of combustible hydrogen. All of these concerns are then amplified because the physics and dynamics that underly operation are very uncertain and unknown. There is a chance that an NTR testing program wouldn't even make it past the NEPA (National Environmental Policy Act) process. The worst case would be getting preliminary approvals, completing the facility, and then getting denied the approval to actually run the test. It would be important to avoid a situation similar to the Yucca Mountain Nuclear Waste Repository, which is sitting empty after billions of dollars of investment. The ability to get test approvals must be evaluated before any NTR concept is seriously pursued.

Another programmatic challenge that makes an NTR ground testing campaign difficult is time. Even a highly-optimistic, success-oriented program plan would require more than a decade. Given the realities of government funding, a program that takes over 5 years is risky and a program that takes over 10 years is essentially a non-starter (even the incredibly ambitious Apollo program went from conception to landing a man on the moon in 9 years).

All of the above difficulties are generally acknowledged within the current space reactor community, so attention has turned to possible flight-testing of NTRs, which would eliminate this programmatic challenge altogether. The flight-test option has become more attractive as the cost of earth-to-orbit launch has been decreasing. The caveat for flight-testing is that the NTR should have minimal system performance challenges (described below). A flight test would have limited ability to resolve diagnostic issues, and a limited propellant supply to resolve performance issues. Thus, the NTR concept chosen should have very simple and predictable performance. At a minimum, the very first system flown as part of a flight-test sequence would need to be very simple (Poston et al., 2019).

Nuclear safeguards

Every NTR that has been built, or gotten past the conceptual design stage, has used highly-enriched uranium (HEU) as the fuel. This is because HEU offer a lower mass, and a much simpler and potentially safer NTR system. Recent studies are considering low enriched uranium (LEU); specifically high-assay low-enriched uranium or HALEU, which is ~20% enriched in ^{235}U . This has the potential to save cost due to reduced security requirements and a simpler recovery effort if the system fails to reach orbit. The use of LEU also fits within a general US policy to avoid the use of HEU for civilian purposes whenever possible. The goal of these studies is to determine if a useful NTR can be successfully developed using HALEU, which would then provide an alternative to a simpler and higher performance HEU system. If it is ultimately determined that HEU is strongly preferred for an NTR project, then the challenge will be to overcome any policy objections to the use of HEU. Current laws and regulations do indeed allow for the use of HEU for space reactors (the recent NASA/NSSA KRUSTY space reactor test (McClure et al., 2020) actually used HEU, including transport across the country). Also, several plans and procedures exist to provide robust security from fabrication to launch, which is why policy decisions are the only significant obstacle.

System performance challenges

There are several technology and programmatic challenges described above, but the greatest challenge to NTP development might be achieving the desired performance of the integrated system. The unique combination of technology challenges coupled with the

difficulty of nuclear system testing makes it extremely difficult to predict, test, and correct performance issues. System performance describes both steady-state and transient operation, the potential for instabilities, and the mass and flexibility of the system: e.g., restart capability, electric power generation, off-nominal operation, etc.

System performance challenges depend greatly on the specific NTR concept. Performance risk depends on the number, magnitude, and uncertainty of reactivity feedback coefficients, the number of flow passes through the core, the uncertainty of heat transfer paths, the uncertainty in turbopump outlet conditions, etc. Risk is also a function of uncertainty in instrumentation and control components, and how much system performance depends on proper I&C function. Integrated simplicity and lack of reliance on fast acting control are both very desirable in this regard. The “challenge” may actually be to find a combination of performance requirements that allows a simple NTR concept to be pursued.

There are a large number of uncertainties in neutronic, thermal, and mechanical performance, and in most cases they compound each other. For a first-of-a-kind engineering system, it is probably better to have 1 or 2 large independent uncertainties, then several smaller uncertainties that can exacerbate each other. An NTR concept that has the simplest combination of uncertainties among all areas is the most likely to avoid performance challenges. Also, as NTR development proceeds, several changes to components and design will occur as issues are resolved, and it is very likely that the performance of individual components will diverge for the original design point; the result might be a system that does not operate as intended, or operate adequately at all.

The key parameters that dictate system operation are reactivity feedback, component-to-component heat transfer, and the inlet flow conditions (i.e., variability in H_2 temperature and pressure, which is largely dictated by turbopump behavior). Most of the challenges involving heat transfer and the turbopump were discussed earlier. From a nuclear perspective, an NTR will be in a different operational paradigm than almost all prior reactor experience. The combination of materials is not only unique, but the potential temperature range of components is also outside normal experience (both at the cold end $\ll 300$ K and the hot end $\gg 2000$ K). There will be a significant uncertainty in nuclear data, which to some extent can be resolved via zero power criticals testing, but not over the full operating range, and probably not with the combined effect of high pressure hydrogen. The level of uncertainty in nuclear data will depend greatly on the concept that is pursued, in particular its neutron spectrum. The reactor isotopics and spectrum also effect the ability to restart an NTR within in a relatively short time (< 1 day), depending on how sensitive reactivity is to the buildup of the fission product ^{135}Xe .

Dynamic performance can become even more unpredictable because the uncertainties also effect the time constants at which components react to change. Stable reactors will nominally overshoot and undershoot steady-state conditions, and the magnitude of these oscillations depends on numerous factors. An advanced instrumentation and control system can operate in real time to mitigate overshoot, but this kind of development would require ground testing (i.e., an essentially unlimited supply of H_2 to slowly startup and investigate how the reactor responds). The variability and uncertainty of feedback, heat transfer, turbopump output, and time constants will also increase the chance for instabilities throughout the operating regime. Inherent instabilities can be controlled, but an NTR that requires real-time reliable signals and actions to keep the system under control has a huge safety and development disadvantage versus a system that operates passively in the nominal operating regime (i.e., standard reactor practice). During the Rover/NERVA testing they discovered operating regimes where the systems were unstable, and then they were able to avoid those regimes in subsequent tests.

Present day modeling capabilities are more advanced than in 1970, but we are still nowhere close to adequately modeling the dynamic performance of a complex NTR. Our capability to perform transient modeling on any first-of-a-kind reactor is extremely limited, despite major efforts in multi-physics modeling over the years. Transient calculations can be reasonably estimated for simple systems, like KRUSTY, because of simple feedback, relatively simple heat transfer, and the valid application of point kinetics (Martin, 2021). An NTR is potentially harder to model than any type of reactor, because of the importance that the turbopump plays in system dynamics, the modeling of compressible flow, and the need for very small computational time steps (due to the high flow velocity).

Experience during the NERVA program has the potential to help diminish system performance challenges, but this depends on how similar an NTP concept is to the tested NERVA rockets. Some of features that made NERVA rockets simple were: the lack of neutron moderator, the lack of multi-pass flow regions in core, the use of an open-cycle turbopump, the use of a hot-bleed cycle to power the turbine, relatively low temperature coolant outlet (2270 K), relatively low coolant reactivity worth, the presence of only one dominant reactivity feedback mechanism (i.e., fuel temperature), and the lack of uncertainty in most parameters due to numerous prior test iterations. The best option for future NTP programs might be to use the same approach as NERVA by simplifying system performance as much as possible; this could be accomplished by using an unmoderated HEU reactor. An unmoderated HEU cermet (ceramic-metallic) fueled reactor should provide the simplest system performance, while also providing the best option for launch safety (Poston, 2018).

Summary

Nuclear thermal propulsion is a high-reward, high-risk technology; i.e., the benefits are great as well as the challenges. The greatest near term benefit is for human travel to Mars, and future human and robotic missions could benefit as well. The greatest challenge is a large number of unproven technologies combined with a limited ability to perform full system testing. For a useful NTP system to be deployed, it will likely require evolution from much simpler space reactors, or a sustained program with a Rover/NERVA level of effort.

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The History of Nuclear Thermal Rocket Development

Cassandra Graham, Beyond NERVA, Dayton, OH, United States

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Introduction

The idea of using nuclear power to propel a spacecraft is a concept almost as old as either nuclear power or rocket design. Radium-powered interplanetary propulsion was proposed in 1913 by Robert Esnault-Pelterie ([Esnault-Pierre, 1913](#)); others were also investigating the use of nuclear power for spaceflight as well. However, it would be the Manhattan Project that would enable the first major work on nuclear thermal rockets (NTRs).

The United States was the first country to put significant resources behind developing NTR into a functional technology rather than a promising concept, starting as part of the Nuclear Energy for the Propulsion of Aircraft (NEPA) program in 1946 ([Schreiber, 1956](#)). This broad research program would be separated into different, more focused programs, with NTR development becoming Project Rover in 1955; the primary facility for NTR development became Los Alamos Scientific Laboratory in Los Alamos, NM starting at that time. Project Rover's development program became more focused under the Nuclear Engine for Rocket Vehicle Application (NERVA) program in 1961 ([Robbins and Finger, 1991](#)). NASA, the AEC, and civilian contractors worked to develop a flight-ready NTR to support post-Apollo crewed missions to the Moon and Mars. Sadly, both Rover and NERVA would be canceled in 1974, concurrent with the elimination of many parts of NASA's plans for further crewed spaceflight ([Portree, 2001](#)).

Starting in 1955, the Soviet Union would devote significant resources to a nuclear thermal rocket program of their own, and would continue to work in the field until the fall of the USSR ([Wetch et al., 1991](#)). Their efforts focused on more promising (but less well-understood) fuel element materials, and used an easier-to-test geometry in their design, so development followed a slower and more modular the American effort, with one of the earliest designs being consistently developed for the majority of the life of the program.

With very little funding for any astronuclear research in the US after Apollo (with one notable exception), and the financial crisis with the fall of the USSR, NTR development was largely relegated to individual design studies, concept development for both improvements on the tested models of NTR as well as more advanced designs, and materials development for various reactor components. Advances in computational capability (both in terms of simulation design and computational power) have led to many proposals for designs, but these have been limited by both the lack of a specific mission to justify the high expense of NTR development and the lack of a specific mission requirement for such a system.

Early US NTR efforts

The first published example of the nuclear thermal rocket was in 1946, when Robert Serber of Douglas Aircraft published "The Use of Atomic Power for Rockets" ([Serber, 1946](#)). Other publications followed in short order, but the highly secret nature of all nuclear technology at the time meant that it was not unusual for almost every study to start from first principles ([Schreiber, 1956](#)). In the earliest days of nuclear rocket research, a huge number of designs were proposed, most of which had a wide variety of variants in how they would be executed. Many of these designs are quite advanced by the standards of the nuclear rockets currently being proposed for active research, but have been either starved of funding or forgotten, sometimes for decades. One thing of note is

the wide variety of proposals, from reasonably conventional high temperature gas cooled reactors of various types, to reactors designed to vaporize themselves during operation, the use of nuclear explosives for propulsion, early nuclear electric propulsion proposals, and many other design architectures. This was due to the lack of communication from the various groups, and the requirement to begin from first principles.

One early source of funding was the Nuclear Energy for the Propulsion of Aircraft (NEPA) Project, which looked at a variety of propulsion methods—including pure rocket propulsion. Nuclear thermal designs were often based on a first stage for an upgraded, nuclear powered V-2 type rocket (being tested at the time at White Sands Missile Range in New Mexico, US), rather than for propelling interplanetary missions.

Within a reasonably short period of time, many American design programs for both aircraft and thermal rocket research settled on the use of a high temperature graphite matrix for the fuel element. This was considered the most mature form of fuel available, and able to give the surest results. Refractory metal fuel elements also showed promise, but the lack of manufacturing knowledge with regards to these metals meant additional uncertainty about their developmental timeframe. Because of this, most programs focused on the more easily manufactured graphite fuel elements, and in particular how to protect them from the highly corrosive hot hydrogen environment they would be exposed to during operation. Fuel element geometry ranged from arrays of hollow pipes to plates to various basic shapes with holes for the propellant drilled through them. This variety of fuel element designs would remain a hallmark of nuclear thermal rocket designs to the present day.

This design decision also led to another major focus of many NTR designs: the fuel clad. Due to the highly corrosive nature of most of the propellants used, the fuel would need to be protected by a coating of some high-temperature material able to resist hydrogen (the most efficient propellant available). The design, fabrication, and testing of clad materials would remain a major focus of work on most NTR programs (Schreiber, 1956).

The beginning of the space race: NASA and Rover

The best known NTR development program was Project Rover, which started in 1955 as a joint United States Air Force (USAF) and Atomic Energy Commission (AEC) program. The USAF portion of the program was transferred to the National Aeronautics and Space Administration (NASA) when that organization was created in 1958. Over the next 19 years, Rover would be the most extensive (and intensive) NTR testing program in history, and would result in system claimed as flight-ready by the time of its cancellation in 1974 (Robbins and Finger, 1991).

Early in the program, there were two primary designs, using different fuel element geometries, fuel element matrices, and flow paths.

Kiwi type reactors featured a uranium oxide particle fueled graphite plate, rod, or prism, usually with holes drilled through the axial length, to form the fuel element. Hydrogen propellant flowing through the reactor from end to end provided both cooling and neutron moderation. The number of fuel element geometries available was wide, and the same basic reactor geometry could be used with minor modifications for a number of other fuel matrix materials, such as refractory metal or carbide fuel elements. The uranium oxide could also be changed to uranium carbide to increase exhaust temperature (Schreiber, 1956).

Dumbo type reactors used corrugated sheets of uranium oxide embedded into a beryllium metal matrix to increase heat conduction over what graphite offered. Stacks of these fuel sheets would create individual flow paths, and the reactor's power and physical size could be adjusted by varying either the number of fuel stacks or the length of the fuel stack. Another distinct difference was the radial flow through the heat exchanging portions of the reactor core to allow for more even heating of the reactor across its entire major axis, and utilized laminar flow throughout the flow path. This offered the benefit of being able to test a small portion of the reactor in easier-to-conduct tests and have high confidence in the results for a full system. In addition, the reactor contained colder zones along the entire axial length, allowing for the more even distribution of lower-temperature materials than the Kiwi-type reactor (Knight et al., 1957).

Both reactors would be investigated at the beginning of Rover, with preference for the Kiwi-type reactors. These were better understood from a manufacturing point of view, and leveraged the already-existing research into fuel element fabrication and core design. The Dumbo design offered better performance, potentially easier nuclear and flow path testing, and greater modular flexibility. However, the lack of experience with the “metal” fuel, as it was known then (this fuel element type, which uses a ceramic fissile component in a metal matrix, is usually called CERMET today), was a concern. Additionally, there were significant computational challenges in ensuring that the flow through the reactor system would indeed remain laminar throughout the flow path. In 1957, the Dumbo design was defunded, and all research efforts would go into the Kiwi-type design and its successors (Kirk, 1992).

The testing of the future NTRs of Project Rover went through several steps. The first was gross core geometry configuration: a test stand called Honeycomb was used to easily replicate the general characteristics of a reactor core using square blocks of materials of several types (fissile fuel, moderator, reflector, neutron poison, and others) to create a sub-critical “sketch” of the reactor core. The core configuration was generally split into two halves, which would be brought together to test zero-power criticality, or the bare minimum requirements to test whether a core would be critical, the basics of reactor dynamics within the proposed core configuration, and other core characteristics.

Once the Honeycomb mockup demonstrated that a design was worth pursuing, the next step was a rough “zepo,” or zero power, mockup of the core. Because this was an experimental criticality test-bed, which wouldn't be put under any strenuous testing, the

machining was not as high quality as the following reactors—only good enough to verify that the initial Honeycomb tests were correct.

A finer-machined mockup, known as “ZEPO-fine,” was then constructed, to verify that the configuration was adequate for the manufacture of the final test reactor, as well as allowing the fabrication personnel to have a better idea of potential manufacturing challenges.

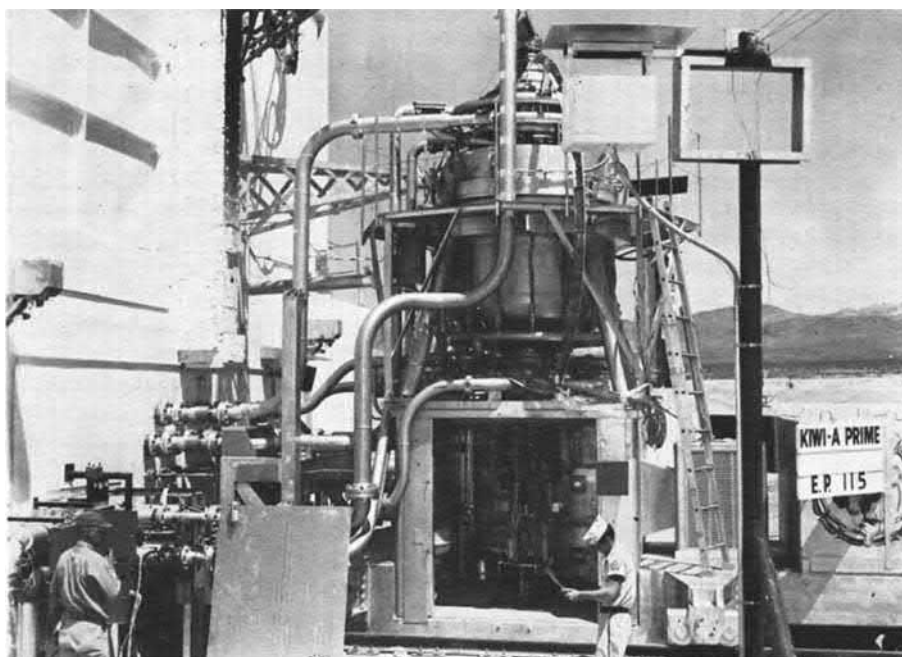
Finally, the NTR that would be hot-fire tested went through a series of criticality, zero power, and cold flow testing to verify that the configuration behaved as predicted, and to assess the reactivity worth of the individual components. This also served to verify the hydrodynamic properties of the reactor, and to calibrate any mechanical erosion issues from propellant flow, mechanical motion of the individual components, and other possible issues with the design (Paxton, 1981).

The reactor would then be transported on a custom-built rail car (which also served as a test stand) to the Nuclear Rocket Development Station at Jackass Flats, NV, where facilities for testing and post-test evaluation were constructed. There, the reactor would be integrated to the NRDS facilities, undergo a final set of checks, and then undergo fission-powered testing. Depending on the reactor, anywhere from 1 to 50 reactor tests would be carried out, at a variety of power levels, temperatures, power and propellant transient conditions, and durations.

After the completion of hot-fire testing, a hot cell laboratory at the NRDS, the Engine Maintenance and Disassembly facility, would be used to disassemble and examine the reactor. This would include: fuel element mechanical, radiological, and chemical analysis; control system mechanical analysis; nozzle damage analysis and a host of other tests (Finseth, 1991).

Hot fire testing: The experimental reactors

Kiwi



Kiwi A Prime pre-hot fire integration at the Nuclear Rocket Development Station. Image U.S. Department of Energy public domain.

The first (or A) generation of Kiwi reactors were experimental designs, designed for basic materials testing and component design experience rather than attempting to build a functional rocket. Because the reactor wouldn't be used as a flight model, and due to the rarity of fissile material at the beginning of the program, a design was decided that was strongly heterogeneous. The first Kiwi reactors used a low pressure heavy water (D₂O) “island” in the center of the core as a strong moderator, which allowed for lower fissile fuel content. These reactors also utilized gaseous hydrogen as a coolant, since cryogenic hydrogen technologies were still undergoing experimental development and not available at the test facility for hot-fire testing of these engines. Many of the tests focused on the basics of reactor control dynamics, fuel element materials and manufacturing techniques, and other broad questions about NTR technologies.

A total of three reactors in this series (KIWI A, A', A3) were hot-fire tested from July 1, 1959 to October 19, 1960. These early tests had many issues, from fuel element erosion due to hot hydrogen interactions, to broken fuel elements due to vibration and thermal stresses, to other reactor components being damaged either thermally, chemically, or mechanically during these relatively short

tests. New fuel element geometry and clad technologies (both materials and manufacturing methods) were continuously applied, and much of the information that was gained during the early program helped significantly inform the following reactor designs.



Kiwi B4 A upon arrival at the Nuclear Rocket Development Station. Image US Department of Energy public domain.

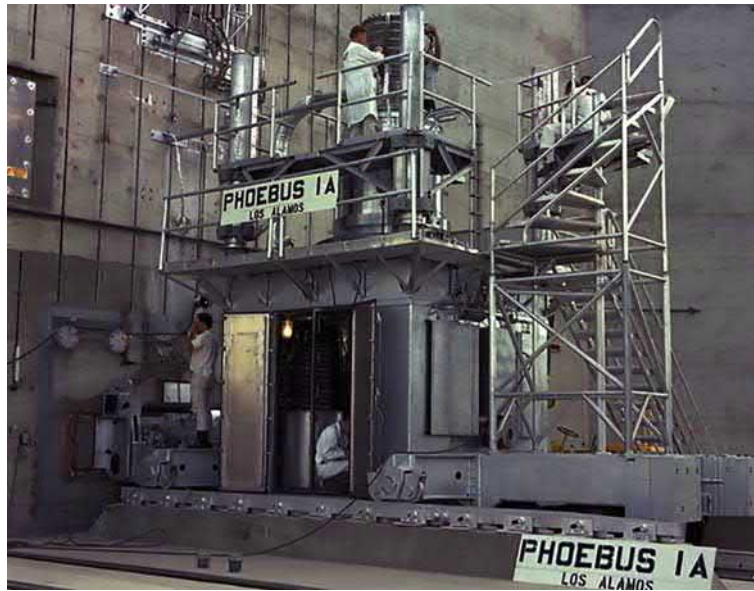
The second series of reactors, Kiwi B, ran from December 7, 1961 to September 10, 1964, and involved hot-fire testing of five different reactors (KIWI B1A, B1B, B4A, B4D, and B4E). While the number of design changes from the "A" series reactors was significant, the largest change in reactor core geometry was the elimination of the D_2O island in the center of the core. In addition, with the exception of the B1A reactor, all Kiwi B reactors used liquid H_2 , rather than the gaseous form, to match the thermal, density, and moderation environment that a flight system would experience (the B1A and B1B reactors were virtually identical except in the use of LH_2 instead of GH_2 coolant).

The B4A was the first reactor to use the hexagonal prism fuel form common to many NTR designs, replacing earlier cylindrical fuel elements held in graphite mounting structures. While the specific geometry of this type of fuel element would continue to change over the course of Project Rover and into the present day, this general fuel form would remain a mainstay of many American NTR designs.

The final reactors in the series (B4D and B4E) was notable in a significant way: was the first hot fire test to show significantly reduced fuel element damage when compared to other reactor tests. Up until this point, the majority of tests were shut down early due to fuel element failure, with mass being ejected from the reactor core during operation in many tests (seen as bright flashes in the exhaust stream). Careful redesign and cold-flow testing following the B4A test, as well as redesigning the thermal and mechanical design of the core components, managed to eliminate this persistent problem. The B4E was also the first widespread use of uranium carbide as the fissile component of the fuel elements, rather than the oxides that had been used up to this point, increasing the thermal stability of the fissile fuel and therefore the theoretical operational temperature of the reactors (Finseth, 1991).

The final test in the Kiwi series of reactors was highly unusual: the Transient Nuclear Test (KIWI-TNT) was a special flight safety test (based on the B4E reactor) to determine sudden power transient behavior of an NTR should it experience a chemical booster explosion or failure to orbit resulting in the immersion in water of an NTR, as well as testing potential in-space decommissioning possibilities using the reactor's power to force a rapid disassembly of the system. In order to meet the requirements of a runaway chain reaction, the reactor's control systems were heavily modified to increase reactivity insertion rates, fission poisons used in the core to balance the neutron spectrum in the core were removed, and a host of other modifications to the core neutronics to eliminate all safety margins were made. The reactor was then forced to undergo rapid deconstruction through thermal overload: they forced the reactor to blow itself up. It's important to note that this was not something that could occur in a standard Kiwi-derived reactor (Fenstermacher et al., 1965). This test was the only one of its type ever conducted, and provided valuable data to mission planners and engineers,—at the cost of significant environmental impacts in the region (Fultyn, 1968).

Phoebus: The transition to development



Phoebus 1A undergoing installation at hot fire test stand. Image US Department of Energy public domain.

Phoebus was a short program, only involving three hot-fired reactors, all based on the Kiwi B4E reactor. This program, while technically experimental, had many similarities to a development reactor program, since they were all based on the same reactor, with the same end goal: develop the Kiwi technology to a point that it could undergo developmental, rather than experimental, experimentation.

The main differences between Kiwi and Phoebus were in fuel element design, specific impulse improvements, power density of the reactor, and power level of the reactor. The end goal of the program was a 5000 MWt reactor, using improved fuel element design and enlarging the coolant channel diameter.

The Phoebus 1A, the first reactor, was damaged during the hot fire test on June 25, 1965, because the test stand ran out of hydrogen, permanently damaging the system. This cut the planned testing of this reactor short, and necessitated the next test of the 1B (as well as an emergency hydrogen dewar backup at the test facility). This reactor was similar to the 1A, except it achieved a higher power level (1500 MWt rather than 1050 MWt), and ran for 46 min on February 23, 1967.

The Phoebus 2A was a much more powerful reactor, and underwent four test runs from June 8, 1968 to July 18, 1968. It achieved a final power level of 3500 MWt in its final test, making it the most powerful NTR to ever be tested in a hot fire stand, and twice the thermal power output of any high-temperature gas cooled reactor used for electricity generation on Earth. Despite this huge power output, this was meant to be an intermediate power reactor on the development path to the ultimate goal of 5000 MWt, and incorporated many improvements in reactor design in order to achieve these higher power densities. Perhaps the second most noteworthy part of the Phoebus 2A's design was the incorporation of a new concept in NTRs: about 10% of the hydrogen was diverted through the fuel element support structures instead of being fed into the fuel element coolant channels, increasing the hydrogen content of the reactor and allowing for the use of more thermally limited support structures (Finseth, 1991). This is the tie rod, which was used for core structural support for all follow on reactor tests. Later, a more advanced structural element called a tie-tube was considered, which proposed returning the hydrogen flow back through the core to improve efficiency, but this was never tested in a reactor. Tie tubes are now commonly proposed in many NTR designs that use hexagonal fuel elements, and have undergone some lab-scale technology development (Benensky et al., 2015).

Pewee and nuclear furnace: Economical fuel element development reactors



Pewee being delivered via the NRDS locomotive to the test stand. Image US Department of Energy public domain.

By this time, the cost of constructing full-sized experimental reactors in inflexible configurations was becoming a significant burden on the program, and so the experimental testing of NTRs turned to smaller, more adaptable designs. The first of these was Pewee, a small (500 MWt) NTR design, which while similar to the Phoebus designs placed a moderating sheath of zirconium hydride around the tie rods to allow a far smaller core size. This reactor was indeed far less expensive than its predecessors, and achieved impressive power density and specific impulse (except this was not designed for, since this reactor was not meant to be a flight model).

The Pewee was tested three times from November 15 to December 4, 1968, and continued to improve fuel element design, both in improvements to fuel clad and coolant flow volume in the elements themselves. It met all of the experimental requirements for the program, and was considered for potential further use as a test platform, but this did not come to pass (Finseth, 1991).

The legacy of Pewee lives on, however. Most current NTR designs are approximately the same thrust level as Pewee could have achieved had it been flight optimized—in fact, many NASA papers use the term “Pewee-class” for these smaller NTR designs, which are usually used in proposals as a cluster of three engines.

The other reactor used for advanced fuel element development was the Nuclear Furnace, a reusable, cost-effective method of testing multiple types of fuel element at once. Unlike all of the experimental reactor designs up to this point, the Nuclear Furnace test was not configured as a rocket engine. Rather, it was designed as a more traditional experimental reactor (albeit based on the Pewee pressure vessel), allowing for replicating the neutron flux, thermal and chemical environment that a fuel element would experience in an NTR, and allowed for more configurations of fuel elements to be tested at the same time. It was also designed to be easily reusable and inexpensive to test, incorporate components from previous programs, and to test an exhaust filter for the hot hydrogen gas to scrub out radionuclides and other contaminants.

This reactor was engineered so that major components of the system would be reused, with only the central core structure (which held the fuel elements) and the effluent filters would need to be replaced. It also was designed with far fewer fuel elements—rather than the hundreds or thousands of fuel elements used in the previous tests, the Nuclear Furnace only had 47 advanced carbon composite fuel cells (UC-ZrC)C spread through a carbon matrix, and two fuel cells each containing 7 smaller, pure (U,Zr)C carbide fuel elements. This in turn required more moderation of the reactor to run with so little fuel, so water was used as a moderator in addition to the hydrogen coolant, with a beryllium reflector and control drums completing the neutronic features in the reactor (Finseth, 1991).

After being heated in the reactor, the hot hydrogen was passed through an Effluent Cleanup System (ECS). The ECS used a system of wire meshes, dehydrators, coolers, and a charcoal trap filter to capture the radionuclides released during the test, and operated very successfully during the test. This was the first implementation of a design concept for testing that would be required in all post-Rover NTR hot fire test programs, however subsequent test concepts have required much larger and more complex ECSs to meet the much higher hydrogen throughput that a full sized NTR under full power for extended operations would require.

The reactor behaved as expected, and the disassembly of the reactor went smoothly, indicating that the reactor concept would in fact be reasonably easy to reuse. The filters also showed very predictable behavior, meaning that if longer test durations were needed the modifications to the filter would be simple to design and implement (Kirk et al., 1973).

There was only ever a single test series of 10 hot firings using the Nuclear Furnace, NF-1, on June 29 to July 27, 1972, which also marked the end of hot fire testing in Project Rover.

NERVA: Building a flight-ready NTR



NERVA A3 undergoing integration at Kiva C at Los Alamos Scientific Laboratory before being delivered to NRDS. Image US Department of Energy public domain.

Once the basic technologies involved in an NTR were demonstrated to both NASA and the AEC, a program was begun to develop a practical, flight-worthy NTR for use in advanced Apollo missions and beyond. This program was named, Nuclear Engines for Rocket Vehicle Application, better known by its acronym: NERVA. While still a component for the overall Project Rover, this was a distinct program with its own goals and technological development based on the successful Kiwi B4 reactor design. The reactors would be informed by the ongoing Phoebus, Pewee, and Nuclear Furnace experiments, but would use better-developed, less experimental versions of the technologies that were still being advanced at a breakneck pace in the larger Rover program. Begun in 1956, this program would continue until the cancellation of US NTR development in 1974 ([Robbins and Finger, 1991](#)).

The first set of NERVA tests were the NRX, or Nuclear Reactor-Experimental, series. All reactors were a development of the Kiwi-B4, although the flow of coolant through the reactor vessel and reflector was changed to increase cooling of the reactor components. Other changes included modification to the nozzle design and fuel elements (both ongoing design evolutions during the NRX program), improved mechanical strength of the core, and increasing power density in the core.

NRX-A1 was used for cold flow and manufacturing testing, and was not hot fired at the NRDS. NRX-A2 was tested on 24 September 1964, being the first reactor to operate at full power (for a total of 6 min). This test was followed by the A3, which was tested three times from April 23 to May 28, 1965.

Following the successful A2 test, the first breadboard test of an NTR was conducted (i.e. using flight components, but not in a flight configuration). This was the NRX-EST test series (consisting of 11 firings), and it occurred between February 3 and March 25, 1966. This was also the first time that an NTR used its inherent core reactivity to facilitate reactor startup (a process known as bootstrapping). With these milestones achieved, the NRX-EST demonstrated that an operational NTR based on the Kiwi B4 could be achievable in the near term, and depended less on the technology than on mission requirements from NASA.

The next reactor to be hot fire tested was the NRX A5, which underwent two tests from June 8 to 23, 1966. After a few false starts, the testing showed that many of the development issues with the fuel elements themselves had been largely solved, although some challenges remained for the design of the fuel elements. This was the last of the NRX series of reactors, and ended the hot fire testing of these NTRs with a resounding success. The fuel elements of this reactor would be used in the XE-Prime flight system test, despite being superseded with more advanced elements in both Pewee and Nuclear Furnace testing, because of their successful behavior in testing, and their relative technological maturity (Finseth, 1991).

XE-prime: Testing the flight configuration



NERVA XE-Prime being delivered to NRDS in vertical orientation. Image US Department of Energy public domain.

XE-Prime was the final major test of NERVA, and as such was designed to mimic the flight configuration of an NTR as closely as possible. By this point, the development of the NERVA NTR made sense to divide into two parts: fuel element development and flight configuration verification.

While fuel element development could be continued using test stands that only mimicked the environment of an NTR reactor core, the flight configuration verification tests had to be as close to prototypical as possible. This meant a number of things had to

occur: all of the components of the engine system had to be in the same relative location as they would be on the flight engine, and the test needed to be conducted with the nozzle facing down, rather than the inverted configuration that all testing had been conducted in up to this point. A nearly year-long test program was carried out under these conditions at NRDS: XE-Prime. While the majority of the flight components of XE-Prime were the final design choices, the fuel was identical to the Kiwi A5 reactor, a less advanced but still adequate nuclear fuel. This means that fuel element performance on the XE-Prime test would not be as effective as the final flight system would be in terms of fuel lifetime, erosion, and potential cracking, but were sufficient for this test (Finseth, 1991).

The XE-Prime reactor was first fired on December 4, 1968, and the final firing was on September 11, 1969. A total of 24 hot fire tests, grouped into 10 experimental programs, were conducted during this time. Many of these experiments focused on start-up and shut-down procedures for an NTR, as well as reactor control system characterization, instrumentation requirements for the flight system, test stand validation, and finally post-test remote handling equipment and procedures for the post-test examination of the test article. Despite some difficulties with starting the engine under particular reactor conditions, as well as instrumentation and accidental scram (emergency shut-off) events, the testing program completed its objectives.

With these successful tests complete, focus would shift back to fuel element development as part of the Pewee and Nuclear Furnace programs in Rover. However, another flight configuration NTR was never hot fire tested in the US after 1969.

Funding, politics, and the end of NERVA

Project Rover and NERVA were always viewed as being used for the next steps after the Apollo missions: establishing a lunar base or crewed missions to Mars. While various organizations, such as NASA and the AEC, supported Project Rover, often the impetus for funding the program came from the political arena, with US Senators for the various locations involved in the program ensuring that funding would be made available and the program would not be canceled. This was especially true during the early days of the program, when fuel element failure was not only routine but expected. However, the combination of it being advanced technology, and highly politically dependent for its survival regardless of its technical merits, meant that the NTR program in the US had the potential to be rapidly canceled if political considerations changed—which in 1972 they would.

At its peak in 1965, Rover had a budget of \$181.1 M USD, but by 1967 had shrunk to \$140.3 M USD. In August 1967, Congress removed all advanced planning, a Mars probe program, and all funding for NERVA, but funding was able to be restored for the program for \$127.2 M USD, with the NRX A6 performing the first 60 min test at the end of the year. However, this was only a temporary reprieve, as the US involvement in the conflict in Vietnam became a larger burden (both in money and lives) leading to significant budget cuts, the administration changed political parties, and new priorities, namely what would become the Space Shuttle and International Space Station in the coming decades, all made NTR technology irrelevant for the direction the US was looking to with its space program. In fiscal year 1972, the combined funding from both the AEC and NASA for NERVA was only \$30 M USD (as opposed to the \$110 M requested), despite Congress offering \$81 M USD in funding for the program. The 1973 budget completely eliminated NERVA funding, and 1974s eliminated all nuclear rocket research funding for all programs (Portree, 2001).

This cancellation would lead the US NTR efforts into an extended period of low activity. While design work on advanced systems, both with different fuel types and different core geometries, as well as materials development, would be carried out, these programs were small, not well funded, and resulted in a relatively small amount of experimentation and testing. Despite some of these designs achieving a level of theoretical development where experimentation would be necessary to further the design, this would not change in any significant way until the Strategic Defense Initiative in the early 1980s.

The legacy of NERVA

Project Rover and NERVA remain the most well-known and extensive testing programs of an NTR system ever conducted. The number of engines, and number of hot fire tests those engines underwent, remains unsurpassed by any other program. By the time of the program's cancellation the NERVA-XE engine was considered flight-ready, the only US NTR system to reach that level of development (Robbins and Finger, 1991). The results of these testing programs are still used in many computational models for NTR design, and the program became synonymous with NTRs in general in the US.

NERVA-derived NTR designs have been proposed since its cancellation, most notably the Small Nuclear Rocket Engine (SNRE). This design uses an advanced version of the Pewee engine, with carbide fissile fuel distributed within a carbon-carbon fuel matrix (Borowski et al., 2015), and is a derivative of the NERVA Derived Reactor (Clark et al., 1992). Several variations of this design, including the Criticality Limited NTR (an even smaller NTR, designed for interplanetary probe missions and flight-proof-of-concept testing), the Lunar Oxygen Augmented NTR (LANTR) hybrid nuclear-chemical engine, and various sizes and power outputs of the original SNRE concept, have been proposed since (Borowski et al., 2017). However, none of these systems have been built or tested, and non-nuclear component testing has been financially limited. Most modern proposals have moved away from the NERVA-legacy graphite/C-C composite fuel matrix in favor of either a ceramic-metal composite (CERMET) or carbide fuel form.

Project Timberwind: The rebirth of US NTR development

The next major NTR program in the US was Project Timberwind, a uranium carbide fueled pebble bed NTR. Initially a part of the Strategic Defense Initiative, this program began in 1987, and was transferred to the US Air Force's Space Nuclear Thermal Propulsion

program in 1991. Development of both the reactor itself and the testing facilities required for hot fire testing continued until its cancelation in 1994.

As with all pebble bed fueled, gas cooled reactors, the increased surface area of the fuel increases the thermal transfer capacity from the fuel to the propellant. The design used two perforated cylinders, called "frits," to contain the fuel particles, which measured 0.5 mm in diameter. Being fueled by a solid solution of carbides, the maximum fuel temperature was increased, and therefore the exhaust temperature (and specific impulse) were increased accordingly. Additionally, due to the high thermal transfer rate, the reactor could theoretically produce more power per unit volume without undergoing thermal failure, leading to a more compact and lightweight design. This allowed for a significant improvement of thrust-to-weight ratio, allowing for greater flexibility in mission application.

Criticality, cold flow, and other testing of the Timberwind fuel elements were conducted at Sandia National Laboratory, Brookhaven National Laboratory, the US Air Force Phillips Laboratory, and other facilities, laying the foundation for additional component and systems tests.

Preliminary design reviews for the engine and upper-stage vehicle were conducted from 1987 to 1989, focusing on the NTR's use as part of the Boost Phase Intercept vehicle concept. With the transfer from SDIO to the US Air Force, this was de-emphasized in favor of a more general mission profile for multiple USAF space missions (Haslett, 1995).

The primary challenge of this program was designing and funding testing facilities for engine hot fire testing. Directly venting the propellant into the atmosphere, as was done for the majority of Project Rover, was no longer an acceptable method of hot fire testing, so an alternative needed to be found. Initially, the design of the testing facility focused on capturing all of the heated hydrogen propellant in a large system of tanks, but this concept was both expensive to construct and severely limited the testing duration available due to volumetric constraints. At the end of the program, underground testing of sub-scale nuclear systems which would be able to be directly applied to a flight test were proposed, in order to reduce test facility costs (Allen et al., 1993).

Project Timberwind was canceled in 1994 with the end of the Space Exploration Initiative, with cost being a primary driver for the decision. According to the Project Timberwind Final Report, the program cost would be approximately \$1 billion USD (1995 value), with potential flight testing to be conducted in the year 2000. This long lead time, the cancelation of the SEI, and the ballooning costs of testing the system led to the decision to cancel the program (Haslett, 1995).

Soviet NTR history

The Soviet Union started focusing on the development of nuclear thermal rockets starting in 1955. Multiple design bureaus across the country worked on the design and fabrication of the initial nuclear rocket engines (NRE), the preferred term for nuclear thermal rockets in Russia. This included the Kurchatov Institute of Atomic Energy (KIAE), the Research Institute for Thermal Properties (NIITP), the Institute of Physical Energy, Moscow (IEE), and Scientific Industrial Association, "LUCH," in Podolsk (Wetch et al., 1991).

The initial (and sustained) focus of this program was on a heterogeneous reactor design, unlike the more homogeneous core that was focused on in the US. These used active regions containing the fissile fuel and propellant flow at high temperatures, and passive, moderating regions kept at a lower temperature between the active fuel bundles. This offers the advantage of allowing for a more thermal neutron spectrum in the reactor, similar to the way the early Kiwi reactors in the US used a low pressure D₂O region. However, by increasing the fissile fuel density of the active region, the total power density of this type of reactor was higher than a homogeneous reactor of the same power level. By using zirconium hydride as a moderator, and separating the heated propellant flow from the thermally sensitive ZrH, it was possible to design an NRE that had good power density, excellent specific impulse and thrust, and was able to be tested using even a single fuel bundle in one of two research reactors that were built for NRE testing at the Baikal Research Reactor Complex in Kazakhstan.

Additionally, rather than using the graphite matrix fuel elements used in the US designs, the Soviet engineers decided to go immediately to a carbide fueled NTR/NRE. This allowed them to target a fuel temperature of 3000 K immediately, close to 500 K higher than the equivalent US system at the same time. This was an option that was considered in the earliest days of the US NTR program, but was rejected for the initial stages of Project Rover due to the unfamiliarity of the established industry with carbides as used in fissile fuel (in later NERVA fuel elements, and some experimental fuel elements tested during the Nuclear Furnace test, carbides were actively employed, but this was so late in the program that it had little overall effect on the technological development of Rover/NERVA systems as a whole). The Soviets, on the other hand, decided that the most promising technology was the one worth pursuing, even if it led to a longer development time. They developed a ternary carbide fuel form, made up of UC-ZrC-NbC and UC-ZrC-C. This allows for a maximum operating temperature of 3200 K, increasing the specific impulse of the NRE significantly (Zakirov and Pavshook, 2007).

The hot fire testing program for the NRE started in 1970, and 30 firings would be conducted by the time the program ended in 1989 as a result of political and financial challenges in the USSR. A total of 330 fuel bundles were successfully tested in 24 successful hot fire tests. However, due to the modular nature of the reactor, the independence of the fuel element bundles for propellant flow, heating, and other characteristics, neither a bread-board (i.e. constructed out of flight components, but not in flight configuration) or flight configuration hot fire test occurred for any Soviet or Russian NRE.

These advantages to the USSR's design made it a robust, flexible, scalable system, allowing for both smaller and larger NREs by varying the number of fuel bundles and the amount of moderation in each design. This design is apparently scalable from ~20 to ~980 kN of thrust with a specific impulse of around 950 s. The most common designs are a small, ~20 kN NRE, which would be

able to propel an interplanetary probe and provide an initial mission for the propulsion system; a larger, ~ 35 kN engine called the RD-0410 for crewed missions; and finally a very large, ~ 390 kN NRE, the RD-0411, which would be used for large crewed missions. Each of these designs is based on the same fuel element bundles, moderator blocks, and control systems, with minimal modifications to account for reactor dynamics changes due to reactor scaling (Zakirov and Pavshook, 2007).

In addition, the USSR and Russians began research on a bimodal NRE, the Nuclear Power and Propulsion System (NPPS), which uses a closed coolant loop to produce energy either in addition to, or instead of, producing thrust. This system, based on the RD-0410, was tested during system development using H_2 , and plans were made for a He-Xe mixture gas generator for long term power generation (Wetch et al., 1991).

There are no currently active programs for a thermal NRE in Russia. However, the technology developed during the 1950s–1980s remains attractive to mission planners, and could be a strong candidate for modern designs (Zakirov and Pavshook, 2007).

Advanced nuclear thermal rockets

While the systems discussed so far have been tested to a greater or lesser extent, and are considered achievable in the relatively near term, there have been designs for more advanced nuclear thermal rockets since nearly the beginning of NTR development (Cooper, 1965). The thermal limitations of solid fuel forms were understood at the time, and the idea of using fuel that was not in a solid state offered many possibilities for increasing temperature and power density. *For more information on these types of NTRs, see the Advanced Propulsion chapter.*

Discussions of using liquid fuel rather than solid fuel were had in the US during the initial design phase of Project Rover, but were not pursued at the time (Schreiber, 1956). At least one concept, the Liquid Annular Reactor System (LARS), proposed by Dr. James Powell in 1991, used a uranium metal fuel element designed to partially liquefy along the propellant flow. The fuel would be retained using centrifugal force by spinning the fuel elements individually at a high rotational velocity (Powell et al., 1991). However, this general form of NTR has had very little development, either theoretical or experimental, even when compared to the other advanced NTR concepts.

The use of uranium hexafluoride as a fuel, contained within a cylinder and compressed by a piston to achieve the necessary density to achieve criticality, was discussed by Dr. Robert Bussard in his 1953 paper “Nuclear Energy for Rocket Propulsion” (Bussard, 1953). Work on this concept was conducted by the United Aircraft Corporation, under contract to NASA, from the mid-1960s until 1972 (McLafferty, 1969; Rogers, 1972). The primary focus of this program was to develop a vortex-stabilized fuel element able to be tested in the Nuclear Furnace (see above), but that program was canceled before testing could occur. On the other side of the world, Soviet designers had plans for an NRE fueled using uranium gas, and operated an experimental reactor from 1957 to 1970, although this was not configured as an NTR (Dmitrievskii et al., 1972).

In the US, open cycle gas (or plasma) core NTR design centered on NASA’s Lewis (now Glenn) Research Center (LRC) outside Cleveland, OH, starting in 1953 (Ragsdale, 1963). Experiments concerning all aspects of a gas core NTR were conducted from early 1960 through the mid-1970s at LRC, including hydrodynamics, neutronics, and thermal management (Jarvis et al., 1976). Due to the incredibly high temperatures (up to 10,000 degrees K), the lack of physical containment of the fissile fuel itself, and other requirements, this type of nuclear reactor is very difficult to build for any function (other uses are high powered fission-boosted lasers or actinide transmutation to reduce waste from nuclear power plants (NASA Jarvis et al., 1976)), but the challenges of an NTR are exceptionally difficult. Advances in computer power, thermo- and hydrodynamics, and computer modeling led to a resurgence in theoretical design calculations on plasma phase NTRs in the early 1990s (Poston and Kammash, 1994). However, due to the high cost and regulatory burden of conducting fission-powered tests of experimental reactor designs in the US, fission-powered testing of such a reactor has been incredibly rare, with only two test runs of bench-top experimental apparatus being conducted since the middle of the 1970s. Occasional theoretical work on this challenging concept continues to be done on an infrequent basis to the present day (Beveridge, 2016), but there have been no major programs of study on this type of system since.

The Soviet Union investigated the use of uranium microparticles in a similar system. Work on this concept began in 1954, by NPO Energomash and the Keldysh Center in Moscow. Various tests of neutronics, hydrodynamics, thermodynamics, materials, and other design challenges have continued to the present day, although the project has not had a high priority since 1975 (Koroteev and Son, 2007).

Due to the significant theoretical, computational, materials, and engineering challenges of developing these advanced NTR concepts, very little testing has been conducted, and much of the work conducted since the 1970s has been conducted either through graduate and PhD student studies, small-scale efforts within organizations, and other similarly scaled programs. They remain an attractive option to leverage the advantages of NTR propulsion, but as during Project Rover their challenges in design and testing prevent large-scale testing and development programs.

Conclusion

Nuclear thermal rockets as an engineering solution to the challenges of spaceflight have been around since the beginning of the nuclear age. Their development as a real-world system began with the space race in both the US and USSR, and both countries made significant progress towards flight-ready systems until their programs were canceled.

The challenges of NTR development, though, are huge. Not only do these systems require the nuclear reactor to be operating at as close to its operating limits as possible, but they also necessitate the use of certain types of materials and subsystems that at the time of the initial development push were only experimentally validated, not applied in the real world. New types of fuel elements, coolant/propellant systems, and other key components had to be designed and tested almost from scratch, with all the challenges of new complex engineering systems needing to be addressed.

NTR development has also been hindered by the lack of testing in the field, with fewer than 1000 hot-fire tests conducted in total, and none in the US since 1972. This is in contrast to chemical rocket engines, which are tested hundreds or thousands of times before they are used for a mission. Fully capturing the heated propellant, along with the fission products and any other material that was dislodged from the reactor during operation, is a significant engineering challenge for even a sub-scale system, such as the Nuclear Furnace, much less a fully integrated, full-size NTR.

The costs of developing an NTR are incredibly high. Not only does the reactor itself require specialized high-temperature materials for many of its components, but the power density of these systems is far higher than most other types of nuclear reactors as well. Finally, the testing equipment for an NTR, from initial component testing through full-scale hot firing is costly, complex, and must be incredibly robust in order to ensure that fission products and other radioactive materials aren't released into the environment (this was not the case in the initial Rover testing, but since then has become an absolute necessity).

More advanced NTR concepts, such as non-solid-phase fuel elements, pebble bed designs, and others, offer even greater benefits in terms of specific impulse, thrust-to-weight ratio, and power density, and as such have undergone their own programs of study and development. The challenges of thermodynamics, hydrodynamics, reactor stability, and other key characteristics may have kept them from becoming as well developed as the solid core NTRs, but programs in both the US and USSR have addressed many of the basic physics questions—although significant hurdles remain, and the development of these systems is still far longer term than a solid core NTR would be.

Despite these challenges, NTR systems offer unique capabilities for interplanetary missions, and as such have been an interesting mission planning tool for many mission concepts for the Moon, Mars and beyond. They remain an attractive option for deep space missions, and will remain so for the foreseeable future. As humanity continues to expand into the solar system, the benefits of NTR propulsion will continue to grow, and additional development will advance the NTR to flight-ready status, culminating in their much-delayed application in real world missions.

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Nuclear Thermal Propulsion: Technology Options

David I. Poston, Los Alamos National Laboratory, Los Alamos, NM, United States

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Introduction

The dream of Nuclear Thermal Propulsion (NTP) has been pursued off-and-on since the initiation of the Project Rover in 1955. Previous chapters in this encyclopedia discuss the challenges to successful NTP development (Poston, 2021a) and the history of past NTP programs (Graham, 2021). The Rover/NERVA programs made significant progress in developing a useful NTP technology, but unfortunately could not advance the technology to flight. After Rover/NERVA was ended in 1973, there have been several NTP programs, but none have even progressed to the building of a prototypic system, nor a system test. The lack of progress in NTP is not surprising given the large set of unique challenges discussed in the previous chapters. Especially since there has been almost no success in developing any new reactor technology over the past 50 years, even for systems much simpler than NTP. This article focuses on the potential technologies and testing approaches that might serve as options to mitigate these challenges, and perhaps ultimately result in the realization of NTP.

Note that NTP (Nuclear Thermal Propulsion) and NTR (Nuclear Thermal Rocket) can often be used interchangeably. The distinction in this article is that NTP typically encompasses the field/technology of nuclear thermal propulsion, and NTR refers to an individual nuclear thermal rocket.

Reactor configuration options

A “typical” NTR is referred to as a solid-core nuclear rocket. In a solid-core NTR, a propellant is heated as it flows through and cools a nuclear fission reactor, and is expanded through a nozzle to create thrust. There are numerous potential materials and geometries to create a solid-core NTR, and they are discussed below. All of the historical testing, and virtually all of the historical NTP research has involved solid-core NTRs.

In a solid-core NTR, the propellant outlet temperature (which is the primary metric of efficiency) is limited by the maximum material temperatures in the reactor core (generally between ~ 2500 and 3000 K). The attractiveness of NTP depends greatly on the propellant outlet temperature, which translates directly to specific impulse or I_{sp} (a measure of how effectively propellant is used ([Poston, 2021a](#))). This has lead others to consider higher performance concepts that could achieve substantially higher temperatures. The next level of performance (temperature) could be obtained by allowing the use of liquid fuel, although there would still need to be structural material that would limit the maximum temperature, perhaps in theory up to ~ 3500 K. An even higher temperature option is to operate the fuel as a gas/plasma. Various schemes have been developed to contain a very-high-temperature uranium plasma and prevent contact with reactor wall. In all these gas-core concepts hydrogen flows between the reactor vessel and the plasma. Containment options include a transparent silica barrier (a.k.a. nuclear-light-bulb), magnetic fields, hydrodynamic flows or vortices, or even the rocket acceleration itself. Most gas-core NTR concepts have theoretical propellant outlet temperatures of ~ 5000 – 6000 K, but there is no specific lower or upper limit. There are also many other nuclear rocket types but they would not fall under the NTR umbrella: e.g., fusion, antimatter, direct use of fission fragments, and even the detonation of small nuclear explosives. Further discussion of advanced propulsion concepts is provided in [Howe \(2021\)](#).

There is also the potential to create a reactor that creates both thermal-thrust and electrical power. This configuration is referred to as a bimodal NTR. Adding the requirement for longer-life electricity production on top of the already enormous difficulties of developing NTR, might make any pursuit of bimodal NTR concept premature. Another “hybrid” NTR approach utilizes chemical combustion to enhance the thrust of an NTR by combusting the exhaust with oxygen ([Borowski, 1994](#)). The liquid-oxygen-augmented NTR propulsion approach can give an NTR the ability to throttle to higher thrust (at the expense of I_{sp}) at times when it could benefit the mission profile.

Technology options for solid-core NTRs

The remainder of this article focuses on NTP technology options for “traditional” solid-core NTRs; although some of the technologies might be applicable to other NTP configurations or advanced reactors.

Fuel options

Rover (carbon-based) fuels

The fuels developed during the Rover program, and also tested as part of the NERVA program, are the only NTP fuels ever tested under prototypic conditions, and thus remain one of the options that must be considered for NTP. The term “Rover fuel” encompasses a large variety of different graphite-based or carbide fuels. All of the Rover fuel elements were prismatic (elongated hexagonal-shaped) elements that could be arranged in a triangular lattice pitch in the reactor core. These elements typically had 19 holes for hydrogen coolant (propellant) flow. [Fig. 1](#) shows a photo of a typical fuel element geometry.

The early years of the Rover program used UO_2 fuel particles embedded in a graphite (where the graphite is referred to as the matrix material). Later, the rather poor performance of the UO_2 fuel lead to a switch to UC_2 . The UC_2 particles were then coated with pyrolytic graphite (PyC) because UC_2 is very reactive with air/oxygen. The hydrogen flow channels were coated with NbC to prevent erosion due to graphite-hydrogen reactions. All of the NERVA rocket tests used this UC_2 fuel coated in PyC. As Rover/NERVA testing progressed they found that ZrC coating worked better as a liner than NbC and it became the baseline. Molybdenum was also tested as a liner material, but there was not enough data to draw a clear conclusion ([Finseth, 1991](#)).

Despite numerous tests, there was still a problem with fuel corrosion and fission-product release with the UC_2 /PyC fuel, even when lined with ZrC. This issue was believed to be caused by cracking in the ZrC, due to the difference in thermal expansion between graphite and ZrC, as well as significant vibrations of the fuel elements. Towards the end of the Rover program work began on new fuel forms that could be more robust in the high-temperature hydrogen environment. There was considerable success with a composite fuel where the graphite matrix existed among a network (veins) of U-Zr carbides ([Lyon, 1973](#)). Forms of graphite were used that matched the higher thermal expansion, and the resulting fuel was much more resistant to cracking or damage. Composite fuels were tested from 5 to 16 w/o (weight percent) uranium; the lower the U w/o the higher the potential operating fuel

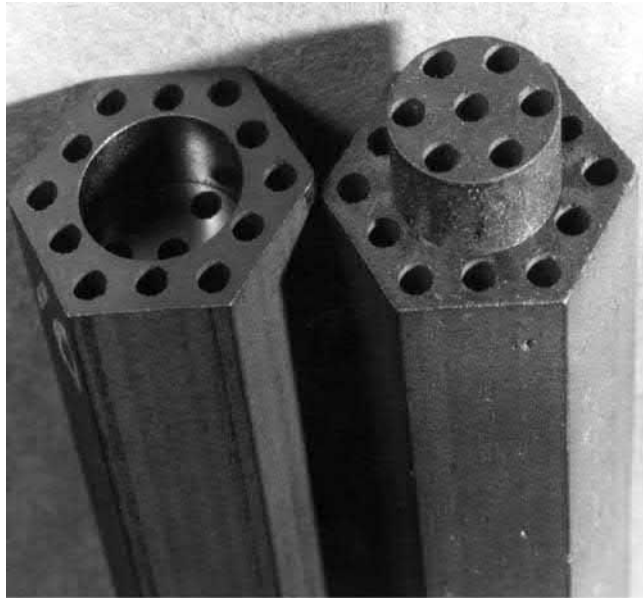


Fig. 1 Photo of a Rover prismatic composite fuel element (Bhattacharyya, 2001).

temperature and vice versa (ranging from ~ 2500 to ~ 2800 K). **Fig. 2** shows a side-by-side depiction of the coated particle and composite fuels.

The final fuels tested in the Rover program were binary carbide fuels consisting of U and Zr carbides, where the fuel is mixed as a solid-solution, or effectively a homogeneous mix of different carbide fuel forms. These fuels have the capability to prevent significant carbon loss, as well as any cracking that would expose the graphite to the hydrogen, in addition they have the potential to operate at > 3000 K in NTRs. Unfortunately, this bi-carbide fuel form only received limited research and testing prior to the program's cancellation. **Fig. 3** shows the projected endurance of the Rover fuels as a function of the propellant exit temperature.

Since the completion of the Rover/NERVA programs there has been continued research on various type of carbide fuels (Pranda, 2005), and interest appears to be currently intensifying in 2020. There is also growing interest in other Rover-like fuels that might be of interest to NTRs and other reactor applications. There is ongoing research into other cercer fuels, which place a ceramic fuel (UO_2 , UN, UC) in a ceramic matrix (e.g., SiC, MgO). Technically Rover fuel is also a cercer (because graphite is considered a ceramic), but cercer fuel generally refers to the use of any ceramic material other than graphite. There is also ongoing research in to TRISO fuels (Helmreich, 2021), which is similar to the Rover coated-particle fuels except that there are three coatings surrounding the fuel kernels (referenced by the "TRI" in TRISO); these coatings make the fuel more robust to thermal expansion, fission product release, and damage during processing. TRISO fuel has great potential for reactors on Earth, mostly because of how they effectively preclude

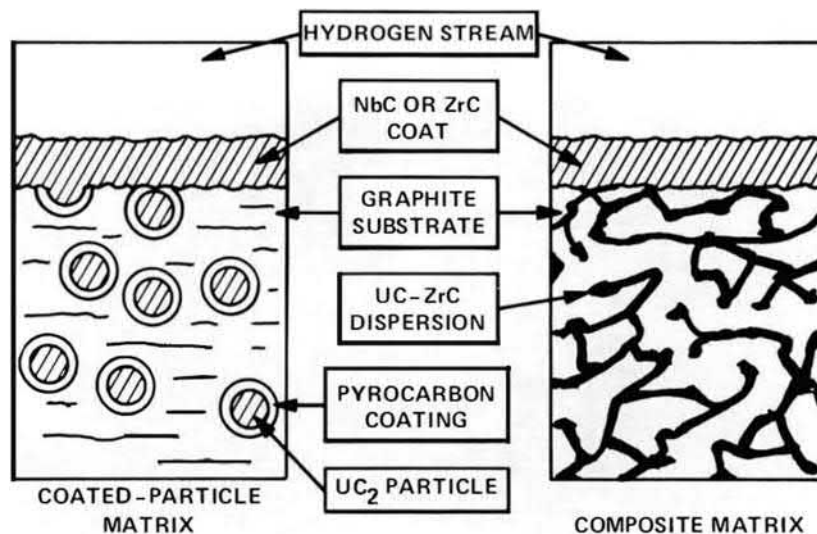


Fig. 2 Schematic depictions of Rover coated particle and composite fuels (Koenig, 1986).

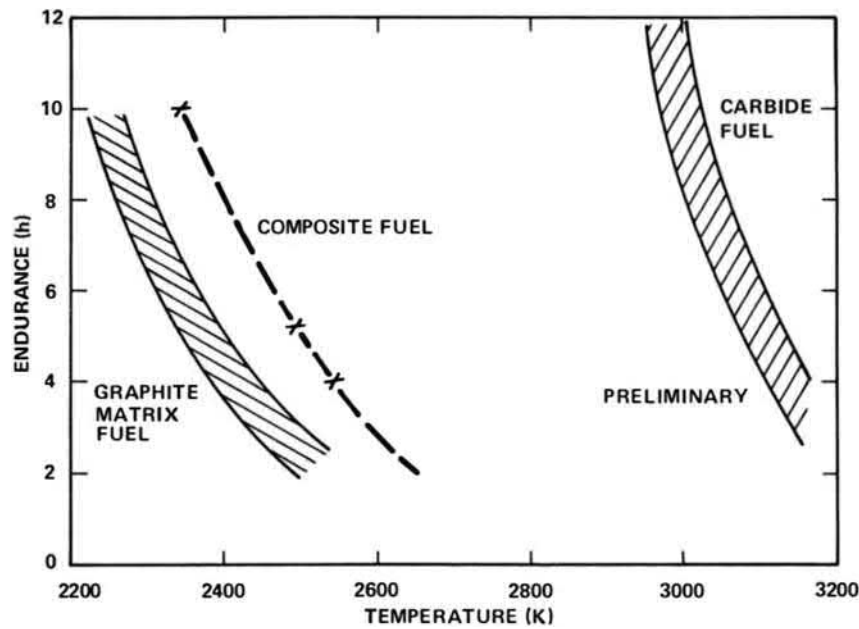


Fig. 3 The projected endurance of Rover fuels versus hydrogen exit temperature (Koenig, 1986).

the release of radioactive fission products, but TRISO appears unattractive for NTRs and other space reactors due to low uranium density (i.e., poor neutronics).

Cermet fuels

Cermet (ceramic-metallic) fuels utilize ceramic fissile particles embedded in a metallic matrix. The fuel fraction in a cermet fuel might range from 50 to 70 v/o (volume percent); a high fuel percent is desirable for better neutronics/mass, but higher fuel fraction leads to poorer fuel structural properties at high temperature. The metal matrix of cermet fuel provides strength at very high temperatures, as well as the strength to withstand significant loads that could occur under launch and re-entry accident conditions. The excellent thermal conductivity of the metal results in a low temperature difference between the fuel and the coolant gas, thus allowing high H_2 exit temperatures.

Most of the cermet fuel researched and tested in the US has been UO_2 fuel particles coated in W, then embedded in a W matrix. Typical elements have been shaped in the same prismatic form as the Rover fuel; i.e., hexagonal blocks with holes for the passage of gas coolant/propellant. The perimeter of the hex and the coolant channels were clad/lined with W-Re. This liner was intended to minimize chemical interactions between the hydrogen and the fuel. In addition, the various layers between the fuel and the H_2 propellant were meant to provide multiple barriers to fission product release during operation; this may be required to gain safety and public acceptance for both the ground testing and space flight phases.

Previously UO_2 cermet development programs have found that a small fraction of oxide stabilizer is needed to promote the chemical stability of the UO_2 : Gd_2O_3 was the most often used, which even a stronger neutron absorber than W. Cermet fuels could also use Mo, or a mixture of Mo-W, to facilitate easier fabrication and better neutronics (lower mass), but W should ultimately provide the best high-temperature performance. In addition, other ceramic fuels could be used to provide better neutronics than UO_2 , most notably UN, but it may have a lower temperature limit as well than UO_2 (due to nitrogen dissociation from the fuel at high temperatures). One possible approach to reduce mass is to use a Mo/UN cermet in the cooler inlet region and a W/ UO_2 cermet at the high temperature outlet region, although this would complicate development, plus the resulting power deposition shift to Mo/UN region diminishes the overall benefit.

The cermet fueled reactors are characterized by a hard/fast neutron energy spectrum under operating conditions. The presence of large amounts of tungsten and rhenium in the core make the core inherently safe under submersion conditions, because of the increased absorptions of neutrons in the thermalized spectrum of the submerged core. The fast spectrum is also preferable from a system dynamics and operational perspective (Poston, 2021a). These characteristics, plus the robustness of the fuel material as discussed above, appear to make the cermet fuel the best option for a typical NTP application from a performance perspective.

There are three primary difficulties associated with the use of cermet fuel. The first is the lack of reactor test experience, especially when compared to the wealth of experience for prior Rover fuels, although this disadvantage may disappear if proposed NTR concepts do not use fuels similar to those that were demonstrated (even then, it may be difficult to correctly recapture the technology). The second disadvantage is that cermet-fueled reactors may only be attractive if the fuel uses highly enriched uranium (HEU), due to the neutronic absorption of the refractory metals (although it is debatable if an attractive cermet reactor could be



Fig. 4 Various W/UO₂ fuel samples (Bhattacharyya, 2001).

successfully developed without HEU as well—all of the Rover/NERVA testing used HEU). The third hurdle to the use of cermet fuel is fabrication, which is especially difficult if the matrix material is tungsten. Fig. 4 shows some W/UO₂ cermet fuel samples; this type of fuel was fabricated by Argonne National Laboratory and the General Electric Company in the 1960s. More recent cermet fuel fabrication efforts have been conducted by Hickman (2012) and O'Brien and Jerred (2013). A historical review of cermet fuel development is provided in Stewart (2015).

Alternative fuel geometries

The aforementioned prismatic fuel form has been the most studied shape for fuel assemblies, and the only form tested in a NTR, but other options have been investigated that have potential advantages.

Wire fuel elements

Wire fuel is potentially attractive because it might be easier to fabricate than prismatic hexes, it provides excellent heat transfer to the propellant (via high surface area and low conduction thickness), and potentially a lower mass core. A study was conducted in the mid-60s by Atomics International of a compact, high performance nuclear rocket engine that employed tungsten wire fuel elements (Arnold, 1965). The wire was fabricated by filling a braided tungsten wire tube with 100 μm UN particles, vapor depositing tungsten onto the tube, and then mechanically forming and strengthening the wire (e.g., swaging, drawing, twisting, etc.). The core was to be constructed of layers of wire wound over alternate layers of spacer wires, which formed an annular lattice. The Soviet Union also pursued what they referred to a twisted-ribbon fuel, which was a carbide based fuel (Borowski, 1993) and is shown in Fig. 5. Although the theoretical performance of wire fuel is very good, the biggest drawback is ensuring the required flow distribution to adequately cool all sections of the fuel. Even a small mismatch between power and flow can be catastrophic in an NTR, due to the high power densities and low margins required to gain adequate performance. A wired-fuel concept might require numerous ground nuclear tests to determine whether power-flow mismatch can be avoided in all operational regimes.

Particle fuel elements

Particle fuel elements are designed with annular beds of small diameter ($< 1\text{ mm}$) fuel particles contained between two coaxial porous cylinders (frits); the frits effectively replace the matrix for containment of the fuel. The most significant efforts for a particle-bed NTR were conducted by Brookhaven National Laboratory (Powell, 1988) and later by the US Air Force in the SNTP program (Walton, 1993). In these concepts the propellant flows radially through the particle-fuel element from the cool outer frit to hot inner frit, and then the propellant exits axially towards the nozzle. The reactor core consists of particle fuel elements arranged in a hexagonal pattern, potentially surrounded by moderating material. The advantage of this design is generally easy fuel fabrication and high surface area per unit volume. Similar to a wire-fueled concept, the disadvantage is ensuring that flow is well distributed through the bed of particles. There is also the potential of particles clogging the holes

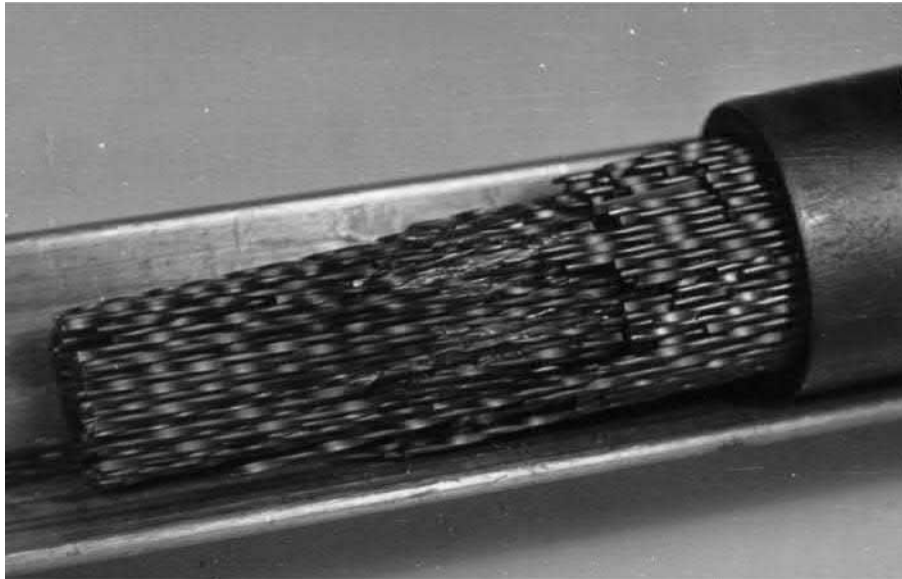


Fig. 5 Soviet twisted ribbon fuel (Bhattacharyya, 2001).

within the frits, and the difficulty of ensuring the structural integrity of the frits themselves. Therefore, this concept is also likely to require several system ground tests to determine if it is practical.

Pebble bed reactors

Pebble-bed reactors place the fuel in larger (several mm to cm) diameter round packages (pebbles). The fuel is typically comprised of small particles placed in a matrix of graphite or ZrC; thus it is a certer fuel in a different form than the prismatic hexes. The pebble is typically clad in a relatively thick layer of ZrC. Pebble bed NTRs have the advantage of larger and potentially more predictable flow gaps (as compared to particle fuel element concepts); however heat transfer to the H_2 is diminished by the larger thermal conduction path, potential thermal resistance between components, and a longer path for the hydrogen (which requires a larger pressure drop). The biggest problem with pebble bed reactors might be engineering and qualifying the core structure, which is discussed later. The use of pebble bed reactors for nuclear thermal propulsion is discussed in El-Genk et al. (1994).

Enrichment options

The safeguarding of special nuclear material represents a significant development and deployment risk to the HEU systems. The costs for security and potential infrastructure are substantial, and the use of HEU also involves political and bureaucratic risk. While HEU will present significant increases in cost and risk, it must be noted that an HEU system has recently been built and tested by NASA (i.e., KRUSTY (Poston, 2021b)); this shows that even in today's environment HEU fuel can be successfully fabricated, shipped across US and tested. The total amount of ^{235}U required might also be a discriminator. At present, the US has a large supply of HEU that could be used, but civilian applications will generally not be given high priority for its use, and there is also the risk that it will be down-blended for political reasons. However, if the need (and will) exists, additional uranium can be enriched with modest technical cost as compared to NTP development. Most concepts consider using either HEU (93% enr.) or HALEU (19.75% enr.), because the costs and policy issues generally remain high until the enrichment falls below 20%.

The most obvious advantage of using HEU is mass. For any given core type of geometry, HEU will offer a much lower mass option. The less obvious advantage of using HEU is simplicity. In many cases, LEU cannot provide a low-enough mass without the use of neutron moderation. The use of a neutron moderator adds several layers of complexity. The moderator technology must be developed and integrated into the core. The moderator also needs to operate at a much cooler temperature than the fuel, thus requiring an entirely separate cooling path to be plumbed into the system, and a very high effectiveness insulator to be developed/integrated that operates at extreme temperature in a potential H_2 environment. Perhaps the biggest risk of adding a moderator is that it greatly increases the complexity of system dynamics and operations, due to the addition of large, competing reactivity coefficients, and the potential for high reactivity worth of the moderator coolant (which is colder and higher pressure/density than other H_2 in the core). Overall, the use of LEU provides either a significant mass penalty or adds significant development risk. Given that NTP development is exceedingly difficult, and low-mass is highly desirable, it may be very unlikely that a useful LEU NTR could ever be developed.

Propellant options

Hydrogen is the choice for almost every NTR application. The key is the low molecular weight of H_2 , which allows the atoms to be accelerated faster per unit energy than any other atom or molecule. The key to efficient rocket propulsion is the exit velocity of the gas, and given the same rocket conditions (e.g., chamber temperature and pressure), H_2 will provide substantially more thrust per propellant mass expelled (specific impulse).

Unfortunately, the use of hydrogen comes with many complications that make it much harder to deal with than other propellants: One of the issues with H_2 is that it penetrates into and through most materials rather easily, more-so at elevated temperature. Hydrogen embrittlement is a concern for structural materials, and leakage through pipes, tanks, and other boundaries must be mitigated. Probably the biggest issue with using H_2 propellant is that it has to be stored cryogenically. The tank thickness to store a sufficient amount of gaseous H_2 at pressure, and without diffusion loss, would generally create an unacceptable mass penalty. Cryogenic storage of fuels is difficult, especially for the long anticipated trip times of human Mars missions, and even then a liquid H_2 still requires a rather large storage volume (due to low density). Hydrogen is also very erosive/corrosive, which presented a major problem during the Rover/NERVA programs. Lastly, hydrogen is a highly volatile substance, and can create significant safety issues associated with storage and transfer.

Carbon dioxide (CO_2) could provide an option for a return flight from a Mars mission. The propellant could be drawn and stored directly from the atmosphere, although the electrical power required to compress the very low-density atmosphere into a high-pressure tank would be substantial. Also, the CO_2 would dissociate at high temperature, into carbon monoxide and oxygen, and then to just carbon and oxygen. This would introduce many potential materials issues within the reactor that would have to be mitigated. Methane (CH_4) would be a possibility for the mission to Titan. Methane dissociates into carbon and hydrogen, which will provide materials issues, but it does have the advantage of storing more efficiently than CO_2 or H_2 . However, if methane is available, it might be more efficient to try and use it as a conventional rocket combustion propellant than as an NTR propellant, provided that an oxidizer could also be provided. Ammonia (NH_3) is another alternative that could be attractive. It has a low molecular weight and dissociates into nitrogen and hydrogen, which is likely better from a materials and specific impulse perspective. Also, ammonia is commonly used in space today, so it comes with a lot of technology heritage. If supplies of ammonia can be found on asteroids or outer planets, it might be the “in situ” propellant of choice, or it could also be made by a chemical plant if the infrastructure existed.

Structural options

The ability to keep the core and reactor intact provides a unique challenge to NTP development; because of extreme temperatures, temperature gradients, pressures, flow velocities, and vibrations (during launch and operation). Regardless of the structural approach, it might be expected that numerous structural failures would occur during initial system testing, as is often the case during traditional chemical rocket deployment. Structural problems were clearly a significant problem during the Rover/NERVA program. The NRX test series of the NERVA program had 16 significant structural failures and anomalies that occurred over the 6 ground test design iterations, including cracking, flaring, rubbing, breaking of various components (Koenig, 1986). Fortunately, the repeated test and fail cycles did ultimately lead to a successful demonstration of core structure, via tie rods and tie tubes.

Tie-rods

The majority of the Rover tests and all of the NERVA tests used what were referred to as support-rods or tie-rods. Tie-rods were used to support the prismatic fuel elements, which were unable to provide mechanical support to themselves. The tie-rod used a metal tube (typically Inconel) to support an array of six surrounding fuel elements as shown in Fig. 6. In order to maintain strength and integrity, the tie-rods were required to be cooled by diverting some of the H_2 flow along the length of the rod. This approach worked very well, but there were a few difficulties. First, if there was a sudden shutdown and loss of flow, the heat transfer from the fuel sometimes overheated the tie-rod and caused a failure. This problem is unavoidable to some extent if the core support must operate at a lower temperature than the fuel. The second issue of using a tie-rod is in rocket performance; the low-temperature flow that cools the tie-rod is ejected into the rocket plenum, and therefore decreases the bulk coolant temperature, and thus thrust and specific impulse. The final Rover test, Pewee, placed a sleeve of ZrH moderator around the tie-rod to moderate the reactor, which allowed a smaller core, and reduced fuel and core mass (Koenig, 1986).

Tie-tubes

A tie-tube is a modification of a tie-rod which eliminates the performance penalty of ejecting the low-temperature flow into the rocket plenum. Instead of a simple 1-pass flow geometry, a 2-pass flow geometry is used, where flow proceeds down the tube from the inlet, reverses in a plenum, and proceeds back up to the inlet region where it is remixed with the flow to proceed through the fuel. This allows all of the flow to be heated to high-temperature by the fuel, although it significantly complicates flow plumbing and requires the turbopump to create higher pressures to create the tie-tube flow. The only Rover/NERVA test that used tie-tubes was the Phoebus 2A test (Fig. 7), and the tubes reportedly worked well (Koenig, 1986). Subsequent tie-tube concepts have been proposed to incorporate a moderating material inside of the tie-tube, but that was never tested (that option will be discussed below).

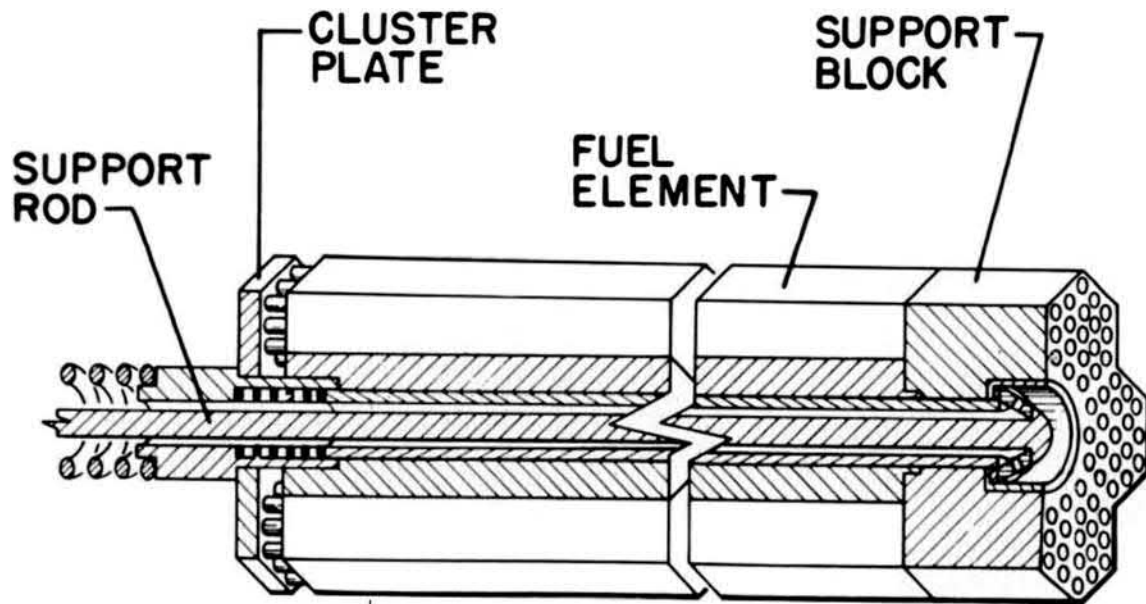


Fig. 6 Rover fuel assembly using tie (support) rod (Koenig, 1986).

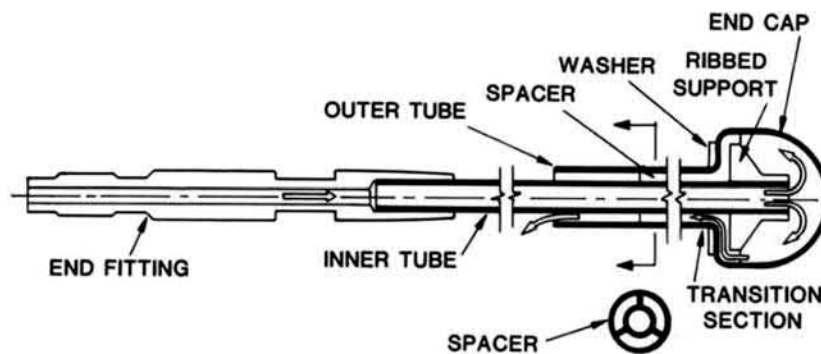


Fig. 7 Tie-tubes used for the Phoebus 2A Rover test.

High temperature structure

The strategy that was successfully deployed by the Rover/NERVA program was to employ a sub-cooled reactor structure (i.e., the structural material was maintained at a lower temperature than the primary coolant outlet temperature). The drawback of this method is the diversion of flow, and the necessary plumbing to cool the structure, and the potential failure of the structure due to loss of flow (which led to damage in some tests). Other NTP concepts have considered using high temperature core support, which eliminates the aforementioned problems, but introduces other substantial engineering challenges. The biggest challenge is the ability for the structure to perform adequately at the coolant outlet temperature (typically 2500–3000 K); this is made even more difficult by the difficulty of testing the proposed structure without a prototypic nuclear-powered ground test. Also, the best candidate structural material at these temperatures might be tungsten, but it is difficult to fabricate, is very brittle, and is a strong neutron absorber.

Cermet-fueled reactors offer the opportunity to use high temperature structure because the cermet fuel, and potential cladding, may have the ability to support themselves. The fuel elements can be supported at the cold end similar to tie-tubes, and potentially other high-temperature structure surrounding or at the bottom of the core. Some previous cermet NTR concepts have progressed to the point of creating a structural design and it does appear feasible (Argonne National Laboratory, 1966); Fig. 8 shows the fuel element design for the Argonne concept.

The other application that could use a high temperature structure for core support would be a pebble-bed reactor, which would use frits to hold the entire core together, as opposed to a particle fuel element where the frits are part of smaller fuel elements. A typical pebble-bed NTR is shown in Fig. 9, and a discussion of the thermal-hydraulic issues presented by a pebble-bed NTR can be found in (Morley and El-Genk, 1994). The frit must contain the entire core at the H_2 outlet temperature, survive the intense thermal stress/strain of a high power NTR, and also allow enough flow area to prevent excessive pressure drop. The engineering and development of this core structure presents an enormous engineering challenge, and would almost certainly require repeated nuclear-powered ground testing to succeed.

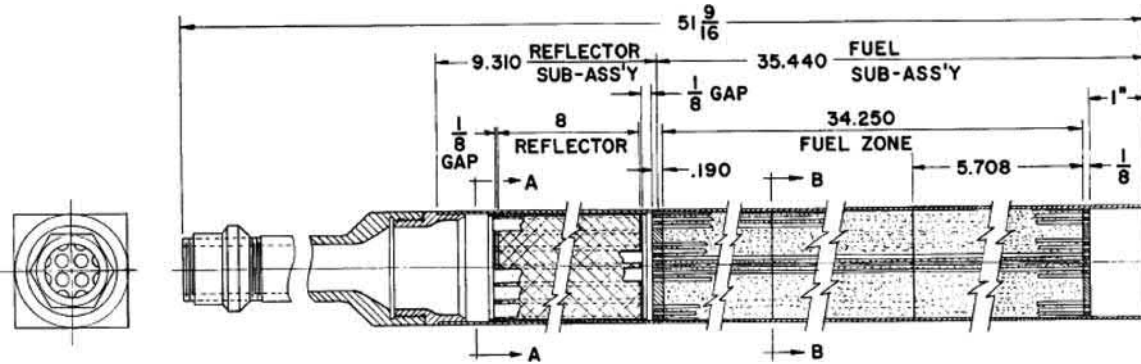


Fig. 8 Design drawing of cermet NTR fuel element (dimensions in inches).

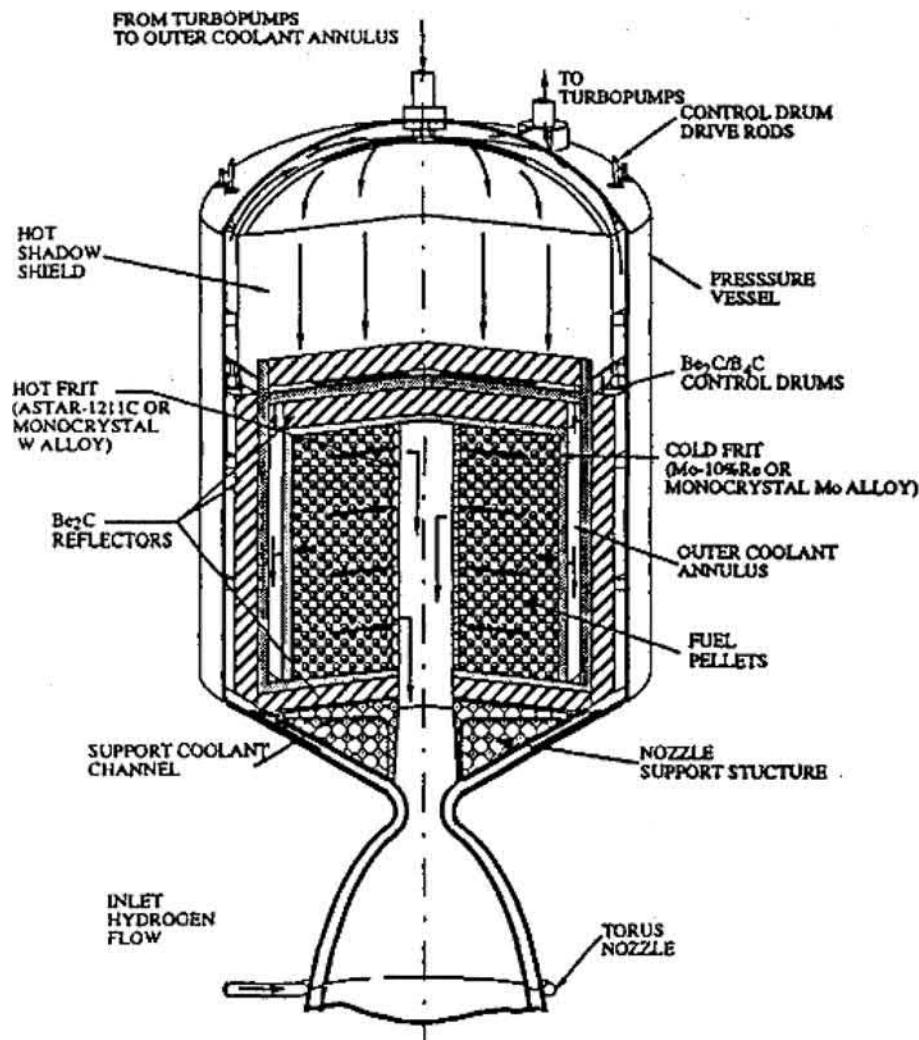


Fig. 9 Drawing of a pebble bed NTR (El-Genk et al., 1994).

Moderator options

One of the primary performance metrics of NTP is mass, and moderation is generally the most effective option to reduce mass (all other design parameters being equal). Carbon-based fuels offer some moderation, as well as the H₂ propellant itself, but increased moderation can be obtained by interspersing additional moderating material throughout the core. On Earth, large high-temperature reactors often utilize graphite as additional moderator, but the small size required for NTP does not allow additional graphite to be

effective. In general, hydrogen is the only effective moderator, because it thermalizes neutrons much more efficiently per collision than any other element (because it is effectively the same mass as a neutron).

Moderator material

The most attractive candidate material for external moderation is likely ZrH. Zirconium-hydride can operate at relatively high temperatures (~ 900 K) and provide a high hydrogen ratio, generally ranging from 1.6 to 2.0H/Zr ratio. The biggest technology risk associated with ZrH is hydrogen loss, which could reduce reactivity and potentially prevent criticality. The higher the hydrogen ratio, the lower the operating temperature needs to be to prevent hydrogen loss. In most applications, cladding, liners, or coatings are used to prevent significant hydrogen loss.

An NTR has the potential to cool ZrH to very low temperatures (< 400 K) such that hydrogen loss would not even be an issue. Also, the short operating times of an NTR (hours) can allow a relatively high loss rate. Unfortunately, the hydrogen loss rate increases exponentially with temperature, such that if the ZrH reaches > 1000 K the H can be lost in a matter of minutes (depending on hydrogen ratio and the types/thickness of coating used). In addition, it may not be possible to keep the ZrH as cold as desired because of the need to remove power deposited in the ZrH, and more importantly the spillage heat from adjacent fuel elements. "Spillage" represents the heat flow from the very-hot fuel to the very-cold moderator coolant. In most scenarios the predicted power from spillage is much higher than the power deposited in the ZrH, because it is difficult to fit sufficient insulation between the fuel and the moderator coolant channel. This added power makes it difficult to limit the moderator coolant temperature rise from inlet to outlet. Also, the uncertainty in insulation performance under prototypic conditions will make it very hard to predict spillage, so that considerable design margin will be needed to prevent ZrH exceeding its upper temperature limit (again, depending on the hydrogen ratio and types/thickness of the coatings used). The extreme temperature swings in an NTR, specifically during startup, will also create significant opportunity for the coating to crack. The overheating failures that occurred with the Rover/NERVA tie-rods due to loss-of-flow will be even harder to prevent, given that ZrH will require lower temperatures than an Inconel tie-rod. All of these issues make the use of ZrH risky, because of the additional constraints it imposes on NTR design. The Pewee test took the simple approach by diverting sufficient excess H_2 flow to ensure ZrH cooling, but this approach will not yield sufficiently attractive NTR performance.

Another potential moderator is YH, which does not provide as much neutronic benefit as ZrH, but has a higher temperature limit, which could provide more margin for the aforementioned thermal design challenges. Conversely, LiH (enriched in ^7Li) would provide even better neutronic benefit than ZrH, but has an even lower temperature limit. Note that "neutronic benefit" refers to lower fuel mass, not necessarily easier development, because added moderation creates other significant complications, and will be discussed below. Other hydrogenous materials like water, polypropylene, etc. are generally unattractive because they require even colder temperatures. Graphite could potentially be used, but as mentioned above it is relatively ineffective in the small geometries required for space reactors. Beryllium is a possibility, even though it is considered more of a reflector material (high scatter) than a moderator material (large atomic mass difference versus hydrogen), but given its very low neutronic absorption it can work fairly well as a moderator if the region is thick enough. Some potential NTR configurations (e.g., the use of a moderator block) might be able to provide enough physical space to effectively use Be or BeO as a moderator. There is also the potential to use lower-temperature fuels in low temperature regions to boost reactivity and thus reduce mass; however, this is likely to amplify the aforementioned power removal issue such that it would not be practical.

Moderated tie-tubes

A moderated tie-tube is an idea to place moderator in the potential empty space in a Rover/NERVA tie tube, in order to reduce the required fuel mass. Although moderated tie-tubes were never built or tested, the idea is based on the successful heritage of moderated tie-rods and unmoderated tie-tubes. In the Pewee test, a small annulus of ZrH moderator was placed around the tie-rods, but was still cooled by single-pass flow diverted from the primary flow. In the tie-tubes of Phoebus 2A, the hydrogen flow moved down through the central rod/tube, and was returned to the core inlet via a counter-flow annular tube. The moderated tie-tube would fill the region between the down-flow tube and counter-flow annulus with a relatively low temperature moderator (likely ZrH) to aid in neutronics. Fig. 10 shows the proposed tie-tube design for the Small Nuclear Rocket Engine (SNRE) concept (Durham, 1972), which was created shortly after the end of Rover/NERVA testing.

The difficulties of using a moderated tie-tube are mostly associated with thermal management. The power deposited in the moderator has to be removed, as well as the power "spillage" (leakage heat transfer) from the fuel assemblies to the tie-tube. A layer of ZrC insulation is intended to prevent spillage, but it will be hard to verify the magnitude of heat transfer. Also, the density of hydrogen in the tie-tube is much higher than in the core (it is at higher pressure and colder temperature than the thrust fuel elements), which creates the potential for significant reactivity changes caused by changes in H_2 flow conditions.

Moderator block/monolith

A monolithic structure can also provide a low-temperature location for a moderator, or the monolith could be the moderator itself, if ZrH, Be, or BeO could provide reliable structural support. Alternatively, a stainless-steel block could hold ZrH pieces, or a very

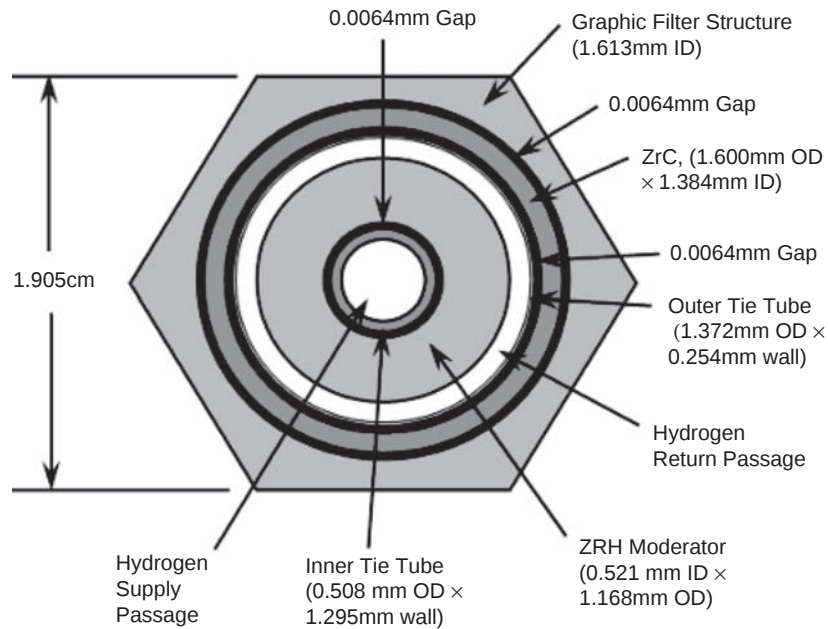


Fig. 10 NERVA derived moderated tie-tube (Fittje and Buehrle, 2006).

high performance system could use a beryllium block to hold ZrH pieces. In all cases the block would need to be cooled by a low-temperature flow stream separate from the fuel.

The Soviet NTR program reportedly pursued the use of a ZrH block, in which carbide-based fuels were interspersed; however there is very little public technical information about this program, so it's difficult to know if they made any significant progress or perform a nuclear-powered system test of this concept. However, the use of a ZrH block with axial holes that contained fuel elements was ground tested in the TOPAZ-II reactor program (Voss, 1993). The configuration was different than for an NTR, in that the block and fuel (thermionic fuel elements) used the same cooling stream and the block had significantly different structural requirements, but the test effectively demonstrated the viability of a moderator block system. Recent efforts by NASA are also considering a moderator block, either using BeO or ZrH (Houts, 2018). As for the tie tubes, the biggest issue of a moderator block might be predicting the heat spillage from the fuel to the moderator, which must be known rather precisely to provide enough power to the turbopump while not overheating the moderator. The primary advantages of using a moderator block instead of moderated tie-tubes is the ability to include more moderator (which allows lower mass) and the reduction of local power-peaking factors (due to a more symmetrical moderating effect). The disadvantages of a moderator block will be ensuring the structural function of the block and the plumbing of the coolant flow, and the potentially larger reactivity feedback caused by increased moderation.

No moderator

Apart from mass, the best option for any NTP concept is to not use a moderator (other than inherent carbon that might be part of the fuel or structure). The use of a moderator creates many technical risks: (1) an additional material/technology is used in the reactor, which must be developed and qualified for NTR conditions. (2) The core geometry must allow the inclusion of the moderator and allow mechanical integration. (3) The nominal neutron spectrum of a moderated reactor does not generally allow the use of intrinsic launch safety. (4) The smaller neutronic worth of the radial reflector of a moderated reactor can make ground, transport, and launch safety harder to address. (5) Thermal management of the moderator becomes an additional design constraint, plus other design limits/requirements that the moderator might impose. (6) The low maximum temperature of the moderator increases the potential for irrecoverable core damage during reduced flow scenarios. (7) A separate coolant flow stream must be provided to cool the moderator. (8) A moderated reactor is generally much more sensitive to impurities or as-built material specs, as well as unknown or uncertain reactivity effects. (9) The reactivity of a metal-hydride-moderated system can be very sensitive to the hydrogen/metal ratio maintained by the hydride material, with detrimental operational and dynamic effects if hydrogen deviates a few percent from the design value, or hydrogen is lost or redistributed during operation. (10) Moderation will almost always substantially increase the magnitude of reactivity feedback coefficients, which makes operation and transient response more uncertain. (11) Moderation will almost always move the reactor neutron spectrum into regimes where nuclear data is more uncertain (e.g., epithermal versus fast), plus the added neutronic uncertainty of the moderator in the first place. (12) The moderator temperature can be very uncertain because of less certain power density distribution across the core, potential heat transfer from hotter regions, and a less certain heat rejection path, in particular, if a high level of insulation is required. (13) A moderated reactor often

result in more than one strong reactivity feedback component, which can conflict with each other depending on power and temperature, create various possible zero-reactivity state points, and create instabilities. (14) The higher magnitude and uncertainty of feedback (as well as more uncertain thermal performance) of a moderated system will tend to require more reliance on robust instrumentation and a flexible, fast-acting autonomous control system. (15) The combination of #7 through #14 makes performance and system dynamics extremely hard to predict, and in general make the moderated option more reliant on nuclear-powered ground testing than a similar, higher-mass system with no moderation (or a smaller amount of moderation).

Point #15 can easily become a showstopper without a ground test infrastructure to perform design, build, and test iterations. In general, it would be preferable to avoid the use of a moderator if possible, and even then, perhaps not until the 2nd or 3rd generation of NTP systems, where the development risk might be reduced to a level which could justify the pursuit of lower mass.

Despite the above generalities, it is unwise to make a blanket statement about the pros and cons of moderation. The same is true when selecting fuel form, or any other NTP design/technology option, because every application is unique. The comparisons and trades need to be made for a specific set of assumptions and requirements. An example trade study can be found in [Poston \(2018a,b\)](#).

Turbopump power options

High-thrust, high-Isp NTRs require high volumetric flow rates and high pressure drops; because flow area is limited by criticality and high velocity is needed to reduce film temperature drop (temperature drop between the cooled surface and the coolant). This combination of high flow and pressure drop requires considerable pumping power. A turbopump ([Chen, 2006](#)) is usually considered the best pumping option, except for extremely low-thrust systems where an electric pump, tank pressure, or other options might work.

For very low-thrust NTRs (< 44.5 kN (10 klbf) of thrust) the pumping power can sometimes be gained via what might be considered a traditional expander cycle. At thrusts $\gg 44.5$ kN the only practical way to get this pumping power is to utilize power from the reactor core, potentially by a hot-bleed or topping cycle.

Expander cycle

The expander cycle is often used in chemical rockets. In this cycle, the cold propellant is used to cool the chamber, nozzle, and other components, and the enthalpy gained by the fluid is then used to drive the turbopump. An NTR will have a considerable amount of regenerative heat from the nozzle because of the extremely high propellant temperature. Also, a good fraction (several percent) of the reactor power is deposited outside of the core, primarily in the radial reflector and shield regions. The cooling of these regions can also contribute to the regenerative heat that powers the turbopump, provided that these components are plumbed upstream of the turbine (note that this heat is not technically regenerative, but it best describes it in the context of standard rocket terminology).

For low thrust options (< 44.5 kN) it may be possible to gain enough turbine power from these regenerative sources, although startup will be complicated by the differing nature of these two power sources. The regenerative power from the nozzle is proportional to the system temperature (driven by the temperature gradient from the hot propellant), while reflector/shield power is proportional to core power (power deposition from neutron and gamma flux). Therefore, the availability of these power sources will depend on the power/temperature profile of the startup.

Hot-bleed cycle

A hot-bleed cycle extracts hot gas from the thrust chamber (core exit plenum) or nozzle (upstream of the nozzle constriction); this was the technique used for NERVA testing, and as such has an existing NTR heritage. A hot-bleed cycle introduces technology issues and can significantly hurt performance. The bleed inlet/piping, where the very-hot gas from the thrust chamber is extracted, has to be transpiration-cooled (injection of cold gas through the wall), unless extremely high temperature piping/turbine can be used; this reduces the bulk fluid temperature via mixing. After powering the turbine, the relatively low-temperature hydrogen must be discharged at a low specific impulse. In addition to the design complexities, this can lower ISP by > 50 s ([Koenig, 1986](#)), a huge penalty for any propulsion system. The NERVA documentation recognized that the hot-bleed cycle was not preferable for performance, but indicates that hot-bleed was selected because it was deemed simpler than attempting to extract turbine power from the core (e.g., topping or driver cycles, which are described next).

Topping cycle

In a chemical topping cycle, a pre-burner is used to provide additional power to the flow prior to entering the turbine. A nuclear topping cycle instead routes coolant flow through the reactor core to gain enthalpy prior to reaching the turbine. The topping flow stream can utilize the entire reactor flow rate or a fraction it, and can be in series or parallel with the expander (regenerative) flow stream.

Tie-tube flow

Tie-tubes have been the proposed option to provide turbopump power in most NTP concepts that have been generated since the completion of Rover/NERVA testing. Instead of exhausting the tie-rod flow into the chamber, which will reduce the bulk temperature and I_{sp} , the flow is returned to provide enthalpy to the turbine. In a moderated system, there is significant power deposition in the moderator as it slows down neutrons. The bigger source of power to the tie-tube flow is the unwanted heat spillage from the fuel to the tie-tube; unwanted because it hurts rocket performance. The spillage effectively causes the peak fuel power density to occur at higher coolant temperature, which reduces the rated power and/or I_{sp} . The heat spillage also presents significant operational and performance difficulties, although in some cases the extra spillage power is beneficial to reach the desired turbine power.

Core block/monolith flow

Some NTR concepts propose the use of a monolithic block structure to support the fuel and provide a cool environment for a moderator. In this case, the topping flow could be gained by cooling the block. The direction of the flow would depend on the balance of structural, material, and thermal issues. Flow in the same direction as the thrust flow might help reduce unwanted heat transfer from the fuel to the block, ease the insulation burden in the chamber region, and mitigate thermal expansion stresses. Flow counter to the thrust flow direction might simplify the plumbing/routing of the flow stream, by routing the nozzle regeneration cooling directly to the bottom of the core. In either case, providing a flow plenum/piping to the extremely hot chamber region might present significant engineering challenges.

Driver fuel flow

The tie-tube and core-block flow options above are “benefits” of a moderated NTR concept, in that the cooling is required regardless, so it might as well be put to good use. Alternatively, an unmoderated NTR has the advantage of not requiring a highly-subcooled core region, but is left with the challenge of cooling a fueled region to power the turbine.

If an unmoderated NTR requires more pump power than can be provided by regenerative cooling, it will have to designate certain fuel regions or elements to power or “drive” the turbine (Poston, 2014). Driver fuel elements could be designed to heat the topping flow to the optimal temperature for the turbine. The lower outlet temperature of the driver coolant flow (versus the thrust flow, i.e., the flow from fuel elements that exhaust into the thrust chamber) would allow fewer, larger flow channels than a thrust element (while still adequately cooling the fuel). A smaller number of larger flow channels improves neutronics (lower coating fraction) and more importantly can allow a relatively low pressure drop.

Three in-core options for driver fuel are discussed below. It might be possible to place driver fuel outside of the core, for example as part of the radial reflector, but that might introduce unwanted issues with power balance, insulation, flow routing, and system dynamics.

Perimeter driver elements

The most attractive position for driver fuel elements might be on the core perimeter. The elements could be designed for 2-pass flow (inner down-flow and outer counter-flow like tie-tubes) or flow could return in the slat/structural region outside of the core. Perimeter driver fuel allows relatively simple plenum design and more flexibility for insulation. The biggest advantage of perimeter driver elements is that they essentially eliminate a big potential reactor design issue—power peaking in the perimeter fuel. The low coolant temperature can allow for the potentially large temperature gradients that might be created by edge power peaking (i.e., the localized peaking around the perimeter of the core due to the return of moderated neutrons from the reflector).

Internal driver elements

These elements would be 2-pass elements very similar in function to moderated tie-tubes, but use fuel instead of moderator and could operate at a higher temperature. They would use a plenum structure similar to that proposed for tie-tubes and potentially provide core structural support if the thrust elements pose structural difficulties. The difficulties of internal driver fuel elements might be differential thermal expansion (from the thrust elements) and the potential for too much heat transfer (spillage) from the thrust elements to the driver elements without significant insulation.

Inlet driver elements

These elements would be short elements placed at the core inlet. The best option might be a 3-path element, which would allow passage of the primary core coolant through a large central hole (low pressure drop and heat transfer), with internal flow channels cooling the fuel, and driver flow returning around the element exterior. Inlet driver elements would have the ability to take advantage of the entire core area (lower pressure drop) and would not have differential axial expansion concerns. The biggest challenge of inlet driver elements would likely be the complex plumbing; also, the inlet region has a relatively low flux/power, so more fuel volume would have to be allocated to driver fuel.

Testing options

The ability to ultimately perform a successful rocket test is the largest obstacle facing NTR development. The performance conditions of an NTR (e.g., temperature, power density, coolant volatility) are far more challenging than any reactor ever successfully deployed in human history. Fuel development, structural performance, safeguards, and launch safety are all major obstacles, but the ability to create the infrastructure (money, facilities, people) to test, get the approvals to test, and design/engineer a system that actually works is the biggest hurdle.

The need for testing is highly dependent on the NTR concept selection. It is important to design the system to be as simple as possible in terms of system dynamics and controllability, so that testing can be closer to a demonstration than an actual test. The option for in-space testing (i.e., flight testing) may be very attractive for NTRs; although if this option is pursued, then it is even more important to avoid operational complexity and uncertainty; i.e., it needs to be even closer to a flight demonstration than a test. Another approach is to construct a facility that guarantees containment for all credible core dispersion scenarios, but qualifying this facility might be harder than qualifying the NTR itself.

Ground test facility

During the Rover/NERVA programs 19 different NTR reactors/engines were tested at the Nevada Test Site (NTS), now designated as the Nevada National Security Site (NNSS). These tests were conducted in open air as shown in [Fig. 11](#). There were no reported radiological health issues associated with the testing, and even after serious reactor accidents they were able to successfully cleanup the test areas for subsequent testing. New regulations appear to make this type of testing impossible, such that the exhaust from an NTR would not be allowed to directly enter the atmosphere. This issue was starting to be addressed by the Rover/NERVA program in the early 1970s, when they demonstrated a scrubbing system that prevented the majority of radioactive particulates and noble gasses from being released into the open air.

Several options for scrubbing and condensing NTP exhaust have been proposed. The basic approach used by Rover/NERVA is described in [Gerrish \(2014\)](#):

The system cooled the hot hydrogen exhaust from the engine with a water spray before entering a particle filter. The exhaust then passed through a series of heat exchangers and water separators to help remove water from the exhaust and further reduce the exhaust temperatures. The exhaust was next prepared for the charcoal trap by passing through a dryer and effluent cooler to bring exhaust temperatures close to liquid nitrogen. At those low temperatures, most of the noble gasses (e.g., Xe and Kr made from fission products) get captured in the charcoal trap. The filtered hydrogen is finally passed through a flare stack and released to the air.



Fig. 11 NTR testing at the Nevada Test Site.

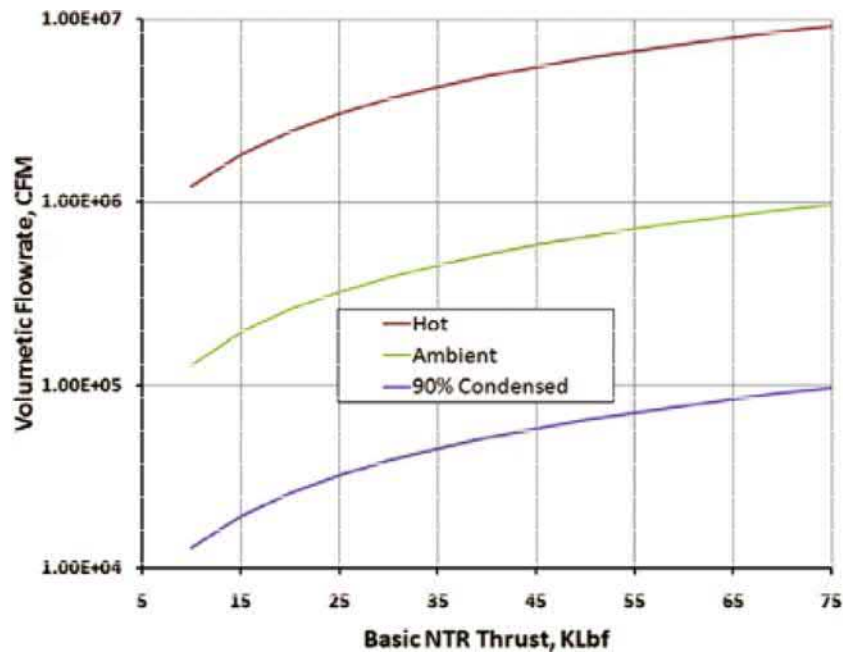


Fig. 12 Volumetric flowrate of gas generated by an NTR (1 cfm = 28.3 L per minute, 1 klbf = 4.45 kN).

The sheer volume of gas produced per minute by an NTR makes scrubbing difficult; volumetric flow rates are shown in [Fig. 12](#) ([Bulman, 2011](#)). Also, because an NTR can melt within seconds if coolant flow is lost, the scrubbing system might have to be able to filter the entire core inventory as well.

Bore-hole or underground testing

It is unclear if a facility could ever be built and qualified to capture NTR effluents and potential core dispersion within a realistic cost and schedule. This has led people to consider testing alternatives that use holes in the ground (either tunnels or boreholes) to capture and filter exhaust. The most attractive location might be at the NNSS, the same location as the Rover/NERVA testing. This is also the same location as the proposed spent nuclear fuel facility at Yucca Mountain, which is very remote and the prevailing rock (Alluvium) is well suited to inhibit the spread of actinides and fission products. [Fig. 13](#) shows a schematic of the SAFE

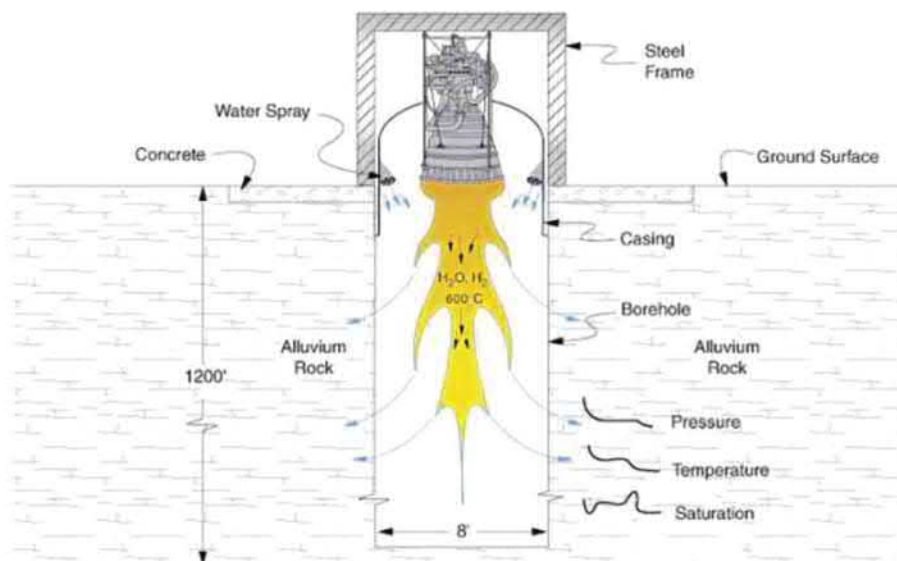


Fig. 13 Schematic of the SAFE borehole testing concept.

(Subsurface Active Filtration of Exhaust) borehole testing concept (Howe et al., 2001). Another option is to test in tunnels deep within the ground, many of which also exist at the NNSS (Borowski, 2016).

Approvals for testing

For a US NTP effort, the ability to get government approval to test is a major risk (whether via NRC or DOE). NTRs have very small temperature margins and extremely high power densities, which are far beyond any other successfully deployed reactor (certainly far beyond the scope of any potential regulator). Average adiabatic heat-up rates range from ~ 500 to ~ 2000 K/s depending on the concept. These values are 25–100 times higher than a typical PWR, and higher still than a typical BWR. The implication is that if cooling is lost during powered operation, each of the proposed NTR cores would melt within seconds. Decay power is also a significant concern; 1 h after shutdown the adiabatic heat-up rate can still be > 1 K/s. NTR safety concerns, compounded by a high volume of combustible hydrogen, are well beyond any reactor safety approval that has been attempted in decades. During the recent DUFF (Demonstration Using Flattop Fissions) and KRUSTY (Kilopower Reactor Using Stirling TechnologY) reactor projects (McClure et al., 2020), DOE regulators were focused on the confidence, or lack thereof, in system dynamics and the potential to melt fuel. This will create a circular dilemma for NTR systems with complex, unknown reactor dynamics and control issues; i.e., these issues can only become known via nuclear system testing, while without in-hand solutions to these issues, there may be no ability to get approval for and successfully execute the tests. In light of this, the best, and perhaps only hope of getting regulator approval for testing is to make the NTR as predictable, simple, and safe as it practically can be.

Space flight test or demonstration

Given the difficulties of ground testing of NTRs, the other option is to develop an NTR via flight testing/demonstrations. A possible flight development path would likely be composed of several increasingly difficult flight demonstrations. The initial demonstrations would require systems with very simple performance and dynamic behavior, because of the limitations of diagnostics to both operate the system and subsequently learn from failures, as well as a limited supply of onboard H_2 (which will not be sufficient to spend hours starting up and exploring the operating space as was done during Rover/NERVA tests). Systems that have strong/complex reactivity feedback, high hydrogen reactivity worth, low tolerance to heat transfer uncertainties, etc. might never be successful via flight development, or at a minimum require numerous flight tests/demonstrations. Alternatively, if a NTR could be pursued that has simple dynamics, (e.g., an HEU cermet-fueled system), then success might be achieved with only a few evolutionary steps.

The most important feature of a flight demonstration (FD) system is that the performance is well-understood and predictable. For a first-of-a-kind flight demonstration simplicity of operation is imperative because of uncertainties in thermal and neutronic behavior, and most importantly because of limited system diagnostics in space. Establishing the reliability and accuracy of diagnostics is extremely difficult in nuclear reactors, and an NTR will have radiation fluxes and temperatures that far exceed those in any existing reactor. Therefore there will be great uncertainty regarding the readings of any instrumentation. In-core temperature measurements may not be possible at all, which would present a major problem for any system that has complex reactivity feedback, including large moderator reactivity feedback coefficients.

For a first flight demonstration it will be nearly impossible to design and develop an advanced automated control system, i.e., one that can determine which instruments to “believe,” what can be actually inferred from their readings, and then decide what actions to take. The use of an NTR with simple and predictable operation is needed if for no other reason but to learn how well diagnostic information can be measured and processed. A system that requires complex control (real time control of reactivity and pump/valves) is not only likely to fail for lack of poor diagnostics, but more-so will not allow a determination of which diagnostics are performing adequately in the first place (because performance would be largely unknown). In that case, very little would be learned from the first or any flight demonstration—which is why a system with significant uncertainties and/or complex dynamics should be developed via ground testing, or evolve gradually from simpler systems.

The US Defense Advanced Research Projects Agency (DARPA) is actively pursuing an NTP flight demonstration program (Greiner, 2020). NASA also recently investigated the potential for a very simple flight system, referred to as Flight Demonstration One (FD1) (Poston et al., 2019), but it was not pursued further.

Ground development or flight development?

As indicated above, the development of NTRs, via ground or flight provides a major challenge; which presents a difficult choice for present and future NTP programs.

Ground testing is clearly the most likely approach to succeed technically, because of the complexity of NTRs. The path to success would probably require numerous design, build, test, iterations to succeed, as evidenced by the need for 19 different ground test campaigns during the Rover/NERVA program, after which they gradually learned how to design, test, and operate the system (and it likely would have taken many more to achieve success). The ground development path would require a massive investment in testing infrastructure and government support at the highest levels to sustain the program. Then, even if the aforementioned can be obtained and sustained over multiple executive administrations, there is still a risk of not getting ultimate approvals to test, or getting into a “Yucca Mountain” paradigm (i.e., not getting final permission to use the facility after it is ready to start being used).

Flight testing is clearly the lower cost of the two options, since a fully-qualified NTR test facility would likely cost $> \$10\text{B}$; based on a comparison to the cost of other DOE facilities that have had much simpler requirements. Borehole or underground testing would be initially lower cost, but the overall cost would depend on how many separate tests were needed. However, as discussed above, it will be very difficult to learn and evolve a technology as difficult as NTP in space. Perhaps the only option would be to test/demonstrate very simple concepts initially, to develop techniques for testing in space, as well as develop NTP technologies.

There is no simple answer to this testing dilemma. Many years after the completion of Rover/NERVA, some of the most experienced and respected NTR engineers weighed in on the side of ground testing, as evidenced by the quoted text below (Bohl, 1990):

The 17 years of experience in the ROVER/NERVA program clearly demonstrated that invaluable results are always obtained from testing these systems—either in the form of verification of successful designs, or by identifying problem areas. Furthermore, such results are not mutually exclusive, as components of marginal design can be “smoked out” while others perform without a hitch in the same test. Even without need for further concept development, it would be imprudent to launch a flight engine on an interplanetary mission with no prior attempt to demonstrate that the integral system could satisfy all of its performance requirements—and, that no component performs marginally. The amount of instrumentation required for the latter confirmation may be prohibitive on a flight vehicle.

Launch safety options

Launch safety for a space reactor is unique when compared to all other space technologies. Even if fully dispersed, an un-operated uranium-fueled reactor does not pose a realistic radiological threat to personnel or the public. The only credible nuclear risk is inadvertent criticality of the reactor. Preventing criticality during water immersion is very difficult for NTRs, due to their relatively large diameter (less radial reflector/drum worth), large coolant area, and difficulty of integrating internal safety rod(s).

Some cermet reactor concepts can use the preferred method for launch safety; i.e., they can be inherently designed to remain subcritical upon water immersion (i.e., there is no “safety” system). The use of this option depends on the fraction of W, Re and/or Mo, the radial reflector reactivity worth, and the core flow area (which can be reduced at the expense of higher pressure drop). In most cermet concepts, the W and Re in the fuel and liners serve as spectral poisons; i.e., they absorb far more neutrons in the moderated neutron spectrum during water immersion than they do in the nominal fast spectrum state.

To achieve subcriticality during water immersion a cermet and/or moderated concepts might utilize neutron-poison wires (gadolinium, borated material, etc.) within the flow holes of the centermost fuel elements. These wires would extend through the nozzle and be removed in a controlled fashion after the system had launched and the probability of reentry is adequately low. The “perfect” bundle size (tight-packed triangular pitch) would have to be substantially less than the nozzle area because tangling will be inevitable. Also, the greater the tug angle (i.e., the angle at which the wire is pulled relative to its axial direction in the core), the greater the chance for loads that could jam/break wires and/or damage the fuel-coating, orifice, or nozzle surfaces.

The engineering of a potential wire-safety system (with perhaps 1000–4000 wires) presents great technical challenge: inserting the wires into the coolant holes, bundling them neatly into the pull mechanism, the design and integration of the mechanism with the nozzle (where attached, how held structurally, how ejected). The greatest difficulty might be designing and proving that the mechanisms (and individual wires) will stay in place during impact while also allowing reliable withdrawal and ejection after launch. If a single wire is not fully withdrawn (breaks off or gets stuck), it will almost certainly result in mission failure unless the reactor is designed with excessive thermal-hydraulic margin. A concept that requires hex and individual flow channel orificing could also be impacted, by not allowing any orificing at the core exit. The testing of the wire system will have to be extensive—enduring launch and impact loads, as well as simulating wire withdrawal in micro-gravity environment (thus effecting how the wires might tangle).

There is some heritage for a wire-safety system. During the Rover program a system was used to place wires in several holes during transport, although, apparently it only used wires that could be drawn straight out (not at an angle) and the system was not designed for launch loads or possible impact loads. There are other possible techniques to prevent inadvertent criticality, but they do not appear promising. Explosive dispersal could be used but will be tough to prove and extremely harder to “sell” (and may not be preferred for safeguards for an HEU system). Methods could be developed to prevent water ingress in the core (removable containment, fast acting foam, etc.), but these types of system seem impractical.

An NTP concept could also attempt to use safety rods which could be withdrawn into the region above the core, into the shield space and potentially beyond. Integration would present the major problem in this case; the biggest consideration would be whether the rods are inside the pressure boundary or not.

The final option for launch accident safety is to allow criticality in water immersion, and prove that it does not represent a consequential risk to the public. This is an acceptable option and almost certainly is technically credible. The difficulty will come in how concrete the technical argument must be. The fully-flooded k_{eff} of some NTP systems can be > 1.1 ; if this level of neutron multiplication was actually achieved the results would be catastrophic. Fortunately, there is no credible possibility that any NTP system could approach these levels, in fact it may even be unlikely that any sort of prompt reactivity insertion would occur (depending on the rate of water ingress). Regardless, the hypothetical prompt reactivity insertion accidents will certainly be questioned and have to be evaluated. There is also another programmatic risk of this approach; i.e., in addition to having to satisfy review boards that it is ok if the reactor goes critical during water immersion. A decision-maker might be wary of allowing a system to launch that would go critical if it fell in the ocean, because of the way it would likely be treated in the media; even if it posed no real danger.

Summary

NTP provides the most substantial challenges of any fission reactor ever conceived, given the extreme conditions presented by temperature, power-density, rapid startup, corrosive coolant, etc., plus the need to operate remotely and reliably in space. If not for the amazing accomplishments of the Rover/NERVA program the idea of NTP might appear far fetched. Fortunately, the successes of Rover/NERVA, although far from achieving flight, gives hope that a useful NTR could be developed. The key will be to find the simplest path to getting a first generation NTR in space. Of the technologies presented in this article, the HEU-cermet NTR might have the best chance for success. Unfortunately simple often competes with other priorities. For example, an HEU NTR would be dramatically simpler than an LEU NTR, but policy concerns might prevent the use of HEU. Similarly, an 700 s I_{sp} NTR would be dramatically simpler, lower-cost, and more likely to succeed than a 900 s I_{sp} NTR, but some potential customers would not want to pay for what would be considered the first step in the evolution of NTP. Even then, a realistic 700 s I_{sp} NTR program might take more time and funding to succeed than most present-day customers would be willing to sustain. The advantages of NTP are compelling, but will likely only be realized by taking relatively simple incremental steps.

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Nuclear Electric Propulsion for Rapid Transportation Within the Solar System

Christopher Morrison, Astro Nuclear Engineer, USNC-Tech, Seattle, WA, United States

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Introduction

Liquid and solid rockets have dominated the space propulsion field since before the space race of the 1960s. However, in the last decade electric space propulsion systems have been developed that can revolutionize the way we travel through the solar system. These propulsion systems are 10–100 times more efficient in their use of propellant than traditional rocket engines. These electric propulsion systems are well developed and have sailed across the solar system traveling to distant locations such as asteroids and the moon (Levchenko et al., 2018).

While these electric propulsion systems are incredibly efficient with their propellant, these systems require electrical power to operate. So far, missions have used solar panels providing kilowatts of power. However, if the power source were able to provide the engines with megawatts of power without significantly increasing the mass of the system, then rapid transportation around the solar system would be possible.

A nuclear fission reactor is one type of system that could feasibly provide megawatts of power while maintaining a relatively low mass. Nuclear fission systems have been used before in spacecraft to provide electrical power to spacecraft components, but they have not been used to power a propulsion system (Buden, 2011a). Over the last two decades electric propulsion has now demonstrated a clear application for megawatt-scale nuclear electric power systems. NASA's *Technology Roadmap for Space Power and Energy Storage* states this about space fission technology:

Fission provides ‘game-changing’ solutions for powering advanced NASA missions. Game-changing attributes of space fission systems include virtually unlimited fuel energy density, the ability to operate independent of solar proximity or orientation, and the ability to design for operation in extremely hostile environments (e.g., high dust, high radiation, or high temperature). Fission can enable a power-rich environment anywhere in the solar system. (Lyons et al., 2010, p. 12)

Why is electric propulsion any better than any other propulsion technology? On Earth electric propulsion is useless because it needs a vacuum to work. But once in the vacuum of space, it can revolutionize the way we travel around in our solar system neighborhood. To sum it up in one thought: electric propulsion is the most fuel-efficient form of propulsion that changes the paradigm from a capsule strapped on top of a skyscraper of fuel to a comfortable spacecraft filled with tools and supplies with a small propellant tank.

The importance of “ ΔV ”

Sometimes space is thought of as a place that lacks gravity. However, in reality all objects in space are bound by gravity. Satellites orbit the Earth because they are stuck in Earth's gravity well. The Earth orbits the sun for the reason.

ΔV can be thought of as the final speed of a rocket as it travels through space. Traditionally it is measured in km/s. A rocket starting at rest will have a final velocity equal to its ΔV if there are no other forces acting on the rocket. Gravity is a force that is acting on the rocket. If a rocket wants to travel from one point to another, the rocket must overcome the difference in gravitational energy between the two points. This difference is overcome with ΔV (also known as delta-v).

When traveling in a car on Earth, the trip is measured in miles. In space, a trip is measured in ΔV . A mission to Mars, the Moon, or another star all have different ΔV requirements. Just like a car needs more gas to overcome air resistance at high speeds, a rocket requires more ΔV to travel to its destination very quickly. ΔV by itself isn't enough to describe everything just like looking at a car's odometer can't tell you how fast the car went, but ΔV is a place to start.

The rocket equation

In the field of space propulsion there is one fundamental equation that applies to all forms of propulsion that use propellant. It is called the rocket equation. The rocket equation determines how ΔV for a spacecraft.

$$\Delta V = V_{\text{exit}} \ln \left(\frac{m_{\text{spacecraft}} + m_{\text{propellant}}}{m_{\text{spacecraft}}} \right) \quad \text{The Rocket Equation} \quad (1)$$

Where ΔV is the total change in velocity, V_{exit} is the exit velocity of the propellant from the rocket nozzle. Looking at this equation, there are three ways to increase the ΔV . One way is to increase the ΔV is to add more propellant. Another way is to decrease the mass of the spacecraft. Neither of these options are attractive.

Liquid and solid rockets typically have a propellant mass fraction of around 90%. For example, the Space Shuttle's final mass on orbit was 6.5% of the initial launch pad mass meaning the 93.5% of the shuttle was propellant and staging. Increasing the mass fraction helps achieve high ΔV , but greatly reduces the payload that can be delivered.

The third and best way to increase the ΔV is to increase the exit velocity of the propellant. The increase in ΔV is proportional to the increase in exit velocity. The table below describes the exit velocity values for generic liquid, solid, and electric engines.

The exit velocity for electric engines is one to two orders of magnitude higher than the exit velocity of the solid or liquid propellants. This means that electric engines are approximately 1–100 times more efficient with their fuel than traditional liquid or solid engines. In other words, an electric powered rocket could complete a mission with 1–100 times more ΔV as compared to a similarly sized chemical rocket. This gain for electric propulsion is reduced by the “Orbith Effect” (which will be talked about in a later section of the document), but still leaves electric propulsion as a propellant efficient form of propulsion.

Nuclear thermal rockets are like chemical rockets except instead of using combustion to increase the exit velocity of the rocket, they flow their propellant through a reactor core. Nuclear thermal rockets have been ground demonstrated in the 1960 and 1970s NERVA/ROVER program that achieve double the exit velocity of liquid rockets make them twice as fuel efficient as liquid systems (Buden, 2011c). A few more advanced nuclear rockets technologies using liquid core, gas core, fission fragment and or nuclear light-bulb technologies can achieve exit velocities levels in the range of 10–15 km/s.

There are several types of electric engines, but their basic function is to take an electric power source and excite a propellant to very high speeds and temperature. Most electric engines (also known as thrusters) use electromagnetic fields to contain and accelerate the high temperature propellant allowing the propellant to achieve extreme temperatures without melting the structure of the rocket. Electric propulsion systems are varied and deserve more discussion, but in general to achieve exit velocities close to the speed of light and practical systems have.

A commonly used measure of exit velocity is called the specific impulse or I_{sp} . Specific impulse is proportional to the exit velocity and is measured in seconds.

$$I_{\text{sp}} = \frac{V_{\text{exit}}}{g_{\text{sea level}}} \quad \text{Isp Relationship} \quad (2)$$

Where $g_{\text{sea level}}$ is equal to Earth's acceleration due to gravity at sea level which is equal to a constant 9.81 m/s^2 . For example, the I_{sp} of a 3 km/s exit velocity is approximately 300 s. I_{sp} is a standard unit used in industry to describe the performance of rocket engines.

The propulsion systems in Table 1 are systems that could be used in inter-solar travel. Traditionally rockets delivering a payload in the Solar System use their bottom stages to get into orbit around Earth and then use their upper stage to propel a rocket stage into the Solar System. Table 2 details the upper stages of several chemical launch rather than the lower stages for this reason.

Table 1 Exit velocity of different types of propulsion.

	<i>Solid</i>	<i>Liquid</i>	<i>Nuclear thermal</i>	<i>Electric^a</i>
Exit velocity (km/s)	2–3	2.5–4.5	8–15	10–>300

^aSeveral types of electric thrusters span this range, see Table 9.

Table 2 I_{sp} of various rockets for LEO to mars travel.

<i>System</i>	<i>Type</i>	I_{sp} (s)
SLS Block II EDS	Liquid (O ₂ /H ₂)	448
Falcon Heavy 2nd Stage	Liquid (RP1/O ₂)	342
Delta III/IV CDSS	Liquid (O ₂ /H ₂)	462
Atlas V second stage XX1	Liquid (O ₂ /H ₂)	451
Ariane 5 ECA	Liquid (O ₂ /H ₂)	446
NERVA	Nuclear thermal	900
Hall Thruster (Smart 1)	Electric (ES)	1000–1600
ITR (Hayabusa)	Electric (ES)	3000
NEXIS (JIMO)	Electric (ES)	6000–7500
NEXT (Dawn DS1)	Electric (ES)	3100
VASIMR (VX-200)	Electric (EM)	2000–30,000

ES: Electrostatic; EM: Electromagnetic.

Liquid engines have an upper limit around 450 s. A few advanced liquid technologies using fluorine and beryllium can achieve an I_{sp} in the 500 s range. Solid core nuclear thermal engines could achieve an I_{sp} as high as 1000 s. Electric engines shown have all been lab tested. Many of them have been flow in deep space missions. Electric propulsion has a wide range of I_{sp} from 100 s to approximately 30,000 s or even higher depending on the design.

Staging, gravity loss, and drag

The rocket equation can express the theoretical ΔV of a rocket. However, knowing the ΔV of a rocket alone cannot give us all the information required to plan a trip into the Solar System. The solar system can be thought of as a “interplanetary superhighway” system. Sometimes shortcuts can be taken reducing the required ΔV just like a car could take a shortcut over a toll road. Some types of propulsion systems can’t take certain routes on the highway similar to how semi-trucks aren’t allowed on certain bridges. Different propulsion systems will require different ΔV . It’s important to understand the differences in order to make a valid comparison between electric propulsion systems to other types of propulsion.

Staging

Rockets typically are split into stages that separate the propellant into tanks that can be dropped before the next stage is ignited. This improves the ΔV of the rocket by around 20–35% depending on several factors including the payload mass, fuel mass, and fuel tank mass. Rockets rarely use more than three stages because each additional stage offers less benefit. Chemical propulsion benefits most from staging because the fuel tanks are very large. Electric propulsion can utilize stages, but the propellant needs are greatly reduced, and the benefit of staging is reduced. However, a single stage electric propulsion system is easier to reuse as there are less “pieces” to logistically track. A single stage propulsion system is like a car which has a gas tank that can be refilled.

Planetary positions and gravity assists

The planets are constantly moving around the Sun and missions only become practical when planets are aligned. Sometime this occurs on a regular basis such as when Mars and Earth are lined up every 2.1 years. Sometimes much rarer events occur such as Voyager II’s use of the gravity from Jupiter, Saturn, Uranus and Neptune to ultimately escape from the Solar System. Gravity assists fly the spacecraft closely in front or behind a planet to use the gravity to accelerate or decelerate the spacecraft. Based on the alignment of the planets, it can take long periods of time to traverse the circuitous routes between planets. The Voyager probes used a rare planetary configuration that which enabled the probes to visit multiple planets, picking up excess velocity along the way to escape the Solar System faster than any other human made object (Fig. 1).

Atmospheric drag

Atmospheric drag increases the ΔV required if a spacecraft is attempting to leave a planet, but it can drastically decrease the ΔV required for a mission if the spacecraft is attempting to slow down to either orbit or land on a planet. On Mars the atmosphere is thick enough that a technique known as aerobraking can be used. Instead of using the rocket’s ΔV to slow down to enter orbit,

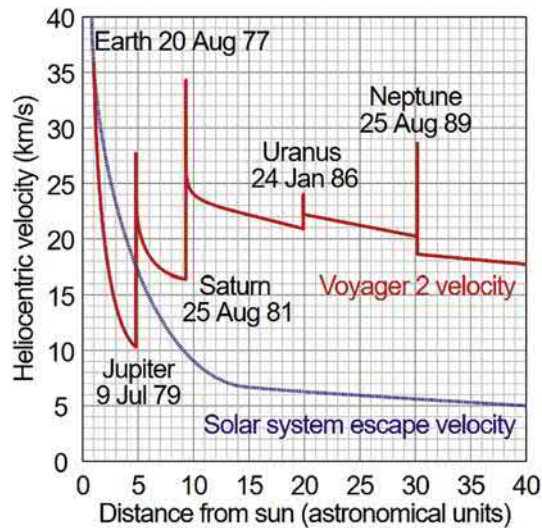


Fig. 1 Voyager II gravity assists (Doody and Matousek, 2020).

the rocket can be flown close enough to Mars to allow the atmospheric drag to help slow the spacecraft and “aerobreak” reducing much of its excess speed. The ability of a rocket to use aerobraking is dependent upon its ability to withstand high temperatures and dynamic pressure from the atmosphere. All forms of propulsion should be able to utilize aerobraking if designed to be structurally capable and in some cases a spacecraft could deliver a smaller payload to a planetary body by detaching the payload and utilizing aerobraking.

Orbith effect/gravity loss—High acceleration vs. low acceleration propulsion

When a spacecraft is traveling through space there are locations where high thrust/acceleration is more efficient than other locations; typically, in locations in an orbit such as the apogee where the velocity of the spacecraft is greatest. Chemical and nuclear thermal rockets can achieve high acceleration for short periods of time. They can minimize their gravity loss by quickly utilizing their propellant in those locations. Electric propulsion provides low thrust but for very long periods of time. Electric propulsion typically requires 20–100% more ΔV compared to the high acceleration propulsion depending on the type of maneuver (for example by using perigee kick maneuvers can reduce the loss significantly but requires more time). Roughly for electric propulsion there is a 20–50% loss in effective ΔV .

Average velocity matching—When transit time matters

While ΔV determines if a mission can be accomplished, it doesn’t specify the amount of time it takes to complete a mission. A first estimate of time matching can be made using a kinematics that estimates the average velocity. If a propulsion system is high thrust typically it can be assumed that it can instantaneously increase and decrease the velocity. For an object wanting to travel a certain distance, the average velocity is equal to the total ΔV as it takes little to no time to accelerate to full speed. For a low thrust, continuously accelerating mission, the spacecraft requires time to accelerate to the maximum speed. The average velocity in this case is one half the total ΔV . Electric propulsion missions that are time constrained require around twice the ΔV of similar missions using high thrust propulsion to effectively match the average velocity. This is particularly challenging in deep gravity wells such as in near Earth orbit where the spacecraft doesn’t travel very far while taking many days to spiral out of the gravity well. For missions to distant locations, for example to the outer planets, it is less of an issue as the spacecraft will build up relative velocity as the gravity well of the sun is much smaller. Some electric propulsion engines can vary their I_{sp} . In the case of variable I_{sp} and constant power, the I_{sp} is reduced in gravity wells which will increase the thrust to accelerate the spacecraft at the beginning of a mission leaving earth and then the I_{sp} is raised and thrust lowered as the spacecraft reaches a greater relative speed. Another method to reduce the time penalty is to look at hybrid high thrust (chemical or nuclear thermal) and low thrust/electric propulsion missions. This strategy can be very effective in certain scenarios.

General propulsion system comparison

Considering all the effects mentioned above, it is difficult to directly compare the ΔV of high thrust and low thrust propulsion types. However, a general approximation can be made that can allow for a very broad comparison of different propulsion systems taking into account the effects above.

Assume that three spacecraft were developed, one chemical, one nuclear thermal, and another electric. Each one has the same total mass and propellant mass. Armed with general knowledge from the preceding sections, a first-look comparison can be made between the three different propulsion systems.

Looking at [Table 3](#) a general comparison of chemical nuclear and electric propulsion fuel efficiency can be made. Electric propulsion can be less fuel efficient compared to chemical and nuclear thermal at relatively low I_{sp} . However, higher I_{sp} is clearly capable of greater propellant efficiency and superior performance of electric propulsion. [Table 3](#) is a general first-order comparison of different propulsion system. Electric propulsion systems can require up to four times more ΔV to achieve a similar level of performance to high thrust propulsion. Despite this disadvantage, electric propulsion can achieve an I_{sp} well over four times that of high thrust propulsion systems to overcome this loss. Electric propulsion systems with on the order of 3000 s of I_{sp} and above start to yield an advantage over high thrust propulsion.

Space environment and humans—Higher ΔV and mass requirements

The conjunction mission to Mars is one of the lowest and repeatable ΔV maneuvers to get to Mars. The NASA DRA document ([Chang-Díaz et al., 2013](#)) describes the conjunction mission as a 2.1 year long mission with 380 days spent traveling through space and 539 days on Mars. For astronauts in space for 2.1 years in space is taxing.

Spending 2.1 years in space is hazardous to astronaut's health for several reasons. First because of the microgravity environment bone and muscle atrophy is very serious. The psychological impact is also not trivial. The native space environment is moderately radioactive. Very high energy protons from outside the solar system easily penetrate aluminum spacecraft shielding. Solar events such as flares pose the risk of causing radiation sickness in astronauts and damaging spacecraft. The radiation dose to astronauts on a 380-day round trip to Mars is around 55 rem ([Chang-Díaz et al., 2013](#)). That is approximately 10 times the maximum yearly occupational dose for radiation workers and 100 times the average yearly dose for Americans. Two ways to reduce the exposure to radiation is to travel through deep space quickly or to travel with more radiation shielding. Traveling more quickly requires significantly more ΔV , and more shielding typically requiring more propellant and larger spacecraft.

Electric propulsion and the importance of power and thrust

Electric engines have been well developed over the last decade and are much more efficient with their fuel as compared to the traditional chemical rocket. However, they have one major drawback they require a power source.

The rocket equation describes the amount of ΔV (total change in velocity) a rocket can achieve. How long does it take to achieve the ΔV ? The time required is inversely proportional to the power source. Rapid transportation is not possible without a low mass, robust power source. Eq. (3) describes the relationship between power and time.

$$t = \frac{m_{propellant} V_{exit}^2}{2P_{jet}} \quad \text{Power Relation} \quad (3)$$

P_{jet} is the jet power or the power delivered to the propellant, t is the time to achieve ΔV , and $m_{propellant}$ is the total mass of the propellant on board the spacecraft. Note that the jet power is less than the power generated from the power source because the power generated by the power source cannot all be used in the jet because of inefficiency such as heat dissipation. Eq. (4) below describes the relationship between jet power and the power source.

$$P_{jet} = \eta_{engine} \quad \text{Jet Power and Power} \quad (4)$$

Where η_{engine} is the efficiency of the propulsion system in transferring energy from the power source ($P_{power\ source}$) into the propellant jet in the spacecraft direction of travel (P_{jet}). Thrust is defined as:

$$T = mV_e = m I_{sp} g_0 \quad \text{Thrust Relationship} \quad (5)$$

Where T is thrust, m is the mass flow rate per unit time, and V_e is the exit velocity. Jet power is defined as:

$$P_{jet} = \frac{mV_e^2}{2} \quad \text{Jet Power Relationship} \quad (6)$$

Assuming a constant power source, we can increase I_{sp} at the expense of thrust. [Fig. 2](#) below describes the relationship between I_{sp} and thrust for a constant 1 MW of power.

Table 3 Comparable fuel efficiency.

	<i>Chemical</i>	<i>Nuclear thermal</i>	<i>Electric</i>
Exit velocity	4.5 km/s	10 km/s	10–300 km/s
Staging effect	135%	135%	100%
Orbeth effect	100%	100%	50–80%
Average velocity match	100%	100%	50–100%
Comparative propellant efficiency	6.0 km/s (600 s)	13.5 km/s (1350 s)	2.5–240 km/s (250–24,000 s)

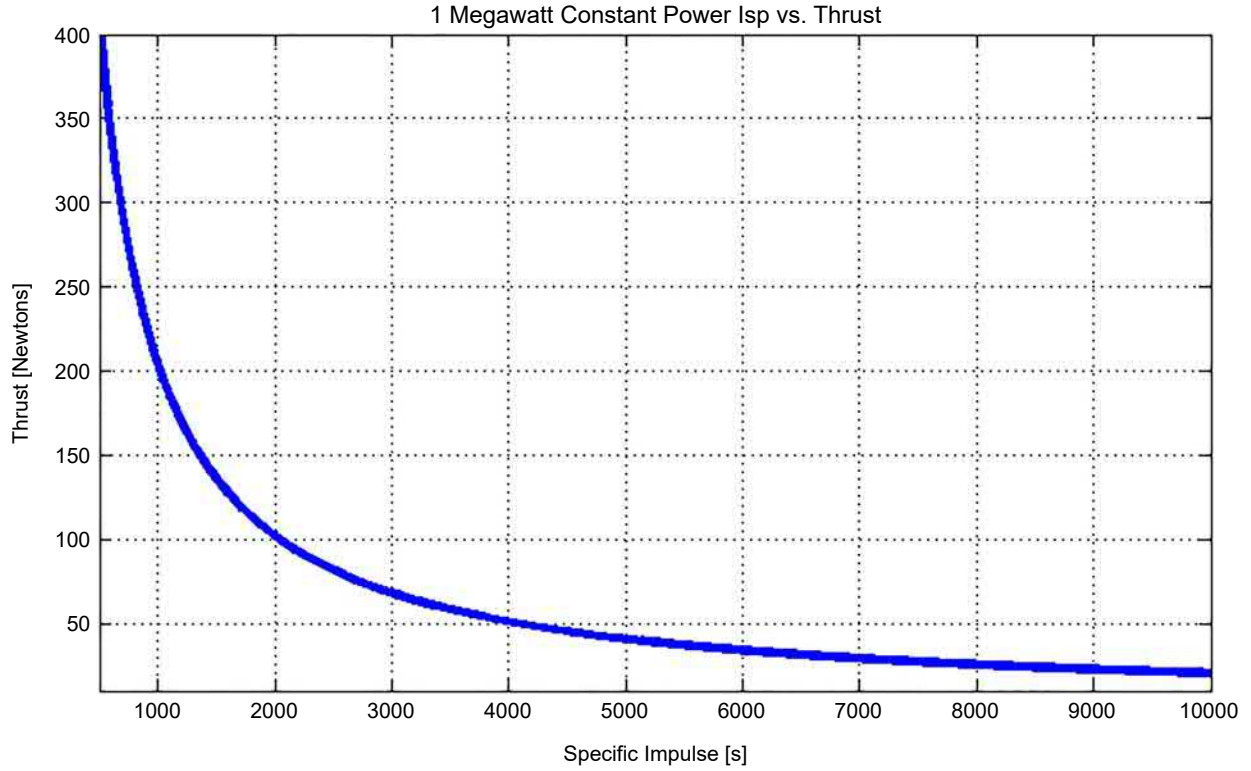


Fig. 2 Thrust vs. specific impulse for a constant 1 MW power source.

Electric systems are revolutionary because they can reach very high specific impulses; however, to reach those specific impulses, a cost is paid in lower thrust. To develop a spacecraft that can transport cargo, a high Isp is needed. To transport cargo rapidly, a high thrust is needed. Therefore, to achieve rapid transportation, a strong power source is needed to complement the electric propulsion system.

The importance of specific power and specific mass

Higher I_{sp} allows spacecraft to transport more cargo and higher thrust allows for spacecraft to get to the destination faster. The available power limits the amount of thrust and I_{sp} that can be obtained. So, it is logical to think that maximizing power is the most important aspect for spacecraft. However, there is another piece of the puzzle, specific power. Specific power is a measure of the spacecraft's power per unit mass. The specific power of the spacecraft is defined as the total power of the spacecraft divided by the mass of the spacecraft without the propellant.

$$\bar{P}_{sc} = \frac{P_{jet}}{m_{sc}} \quad \text{Specific Power} \quad (7)$$

Where $\bar{P}_{sc}$ is the specific power in W/kg, P_{jet} is the power delivered to the propellant directional jet, and m_{sc} is the mass of the spacecraft without propellant. Let's reanalyze the situation by writing the rocket equation in terms of specific power (Table 4):

$$\Delta V = \sqrt{2E} \ln \left(1 + \frac{\bar{P}_{sc} m_p}{P_{jet}} \right) \quad \text{Electric Power Rocket Equation} \quad (8)$$

Where P_{jet} is the directional jet power of the spacecraft in Watts, $\bar{P}_{sc}$ is the specific power in W/kg, m_p is the propellant mass in kg, and E is the energy density imparted per unit mass on the propellant measured in J/kg. Looking at the average acceleration of the spacecraft in terms of the specific power (Table 5):

$$\bar{a} = \frac{\sqrt{2} P_{jet} \ln \left(1 + \bar{P}_{sc} m_p / P_{jet} \right)}{\sqrt{E} m_p} \quad \text{Average Acceleration} \quad (9)$$

Where $\bar{a}$ is the average acceleration of the spacecraft while it is achieving its ΔV . These two equations (Eqs. 8 and 9) are the fundamental equations that govern space travel. ΔV tells what missions can be accomplished, and $\bar{a}$ tells how fast those missions can be

Table 4 Specific power rocket equation derivation.

$\Delta V = V_e \ln \left(\frac{m_o}{m_f} \right)$	The rocket equation
$\Delta V = V_e \ln \left(\frac{m_{sc} + m_p}{m_{sc}} \right)$	Initial mass of the rocket is composed of the spacecraft mass and the propellant mass.
$\Delta V = \sqrt{2P_{jet} \frac{m}{\dot{m}}} \ln \left(\frac{m_{sc} + m_p}{m_{sc}} \right)$	$m_o = m_{sc} + m_p$ The exit velocity can be expressed as a function of power and mass flow rate.
$\Delta V = \sqrt{2P_{jet} \frac{m}{\dot{m}}} \ln \left(\frac{P_{jet}/\bar{P}_{sc} + m_p}{P_{jet}/\bar{P}_{sc}} \right)$	$V_e = \sqrt{2P_{sc} \frac{m}{\dot{m}}}$ Using the Eq. (7) $m_{sc} = P_{jet}/\bar{P}_{sc}$
$\Delta V = \sqrt{2P_{jet} \frac{m}{\dot{m}}} \ln \left(\frac{P_{jet} + \bar{P}_{sc} m_p}{\bar{P}_{sc}} \right)$	Multiply the specific power to the numerator.
$\Delta V = \sqrt{\frac{2P_{jet}}{E}} \ln \left(\frac{P_{jet} + \bar{P}_{sc} m_p}{P_{jet}} \right)$	The mass flow rate is proportional to power and inversely proportional to the energy imparted per unit mass. For example, twice the power means twice the propellant used. $m = P_{jet}/E$

Table 5 Power acceleration derivation.

$\bar{a} = \frac{\Delta V}{\Delta t}$	Average acceleration can be defined as change in velocity divided by change in time
$\bar{a} = \frac{\sqrt{2E} \ln \left(1 + \bar{P}_{sc} m_p / P_{jet} \right)}{m_p / m}$	Change in velocity was taken from Eq. (8) and change in time can be found by $\Delta V = m_p / m$
$\bar{a} = \frac{\sqrt{2E} \ln \left(1 + \bar{P}_{sc} m_p / P_{jet} \right)}{E m_p / P_{jet}}$	$m = P_{jet}/E$

Table 6 Specific power rocket equation dependencies.

Variable	$\bar{a}$		ΔV	
	Effect on $\bar{a}$	Dependence	Effect on ΔV	Dependence
E	Propellant energy density	↓	↑	$\sqrt{E}$
m_p	Propellant mass	↓	↑	$\ln \left(1 + \frac{\bar{P}_{sc} m_p}{P_{jet}} \right)$
P_{jet}	Power	↑ asymptotic	↓	$\ln \left(1 + \frac{\bar{P}_{sc} m_p}{P_{jet}} \right)$
$\bar{P}_{sc}$	Specific power	↑	↑	$\ln \left(1 + \frac{\bar{P}_{sc} m_p}{P_{jet}} \right)$

accomplished. Both equations are based on the same four independent variables. Table 6 below describes the relationship of these two equations against their independent variables.

Specific power is the dominant factor in electric propulsion. The three variables have mixed results on ΔV and $\bar{a}$. Increasing the propellant mass increases ΔV , but decreases $\bar{a}$. Similarly, E has the same problem. Increasing power has an asymptotic effect on ΔV . Specific power is the only variable that has a positive effect on both the ΔV and $\bar{a}$. Therefore, *specific power is the most important variable to maximize in electric propulsion spacecraft*. The other three variables can be balanced based against the requirements for the ΔV and time frame of the specific mission, but in general the performance of an electric propulsion system is limited by the specific power.

To define potential power sources, a better description of the spacecraft is needed. So far, a limited description of a spacecraft has been given that includes propellant (m_p) and everything else (m_{sc}). The masses can be further broken down in Fig. 3.

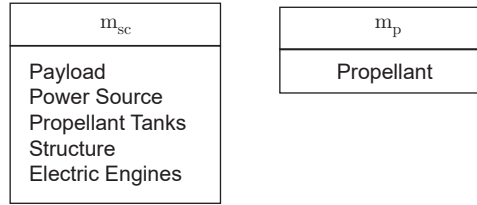


Fig. 3 Spacecraft mass breakdown.

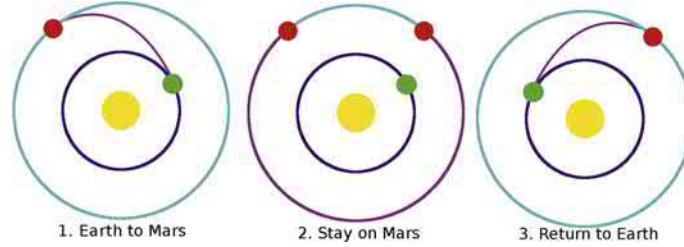


Fig. 4 Conjunction class missions.

$$\alpha_{sc} = \frac{m_{sc}}{P_{jet}} = \frac{1}{P_{sc}} = \frac{m_{payload} + m_{power\ source} + m_{tanks} + m_{structure} + m_{engine}}{P_{jet}} \quad \text{Spacecraft Specific Mass} \quad (10)$$

Where α_{sc} is the specific mass and is measured in kg/W. Smaller values of α_{sc} are better. The specific mass can be defined as a sum of each of the component specific jet masses.

$$\begin{aligned} \alpha_{sc} &= \frac{1}{P_{jet}} \sum m_{component} = \sum \alpha_{component_{jet}} \\ &= \alpha_{payload_{jet}} + \alpha_{tanks_{jet}} + \dots \quad \text{Spacecraft Jet Specific Mass Expressed as Sum of Component Specific Mass} \end{aligned} \quad (11)$$

The specific mass of the spacecraft is mission dependent. For example, the payload can change from mission to mission and it is not useful to compare the performance of the propulsion system with two missions with different payloads. To better compare propulsion systems, it is useful to look only at the power source and engine specific mass.

$$\alpha_{PPS_{jet}} = \frac{m_{power\ source} + m_{engine}}{P_{jet}} = \alpha_{power\ source_{jet}} + \alpha_{engine_{jet}} \quad \text{Preliminary Payload, Power, and Propulsion System Specific Power} \quad (12)$$

The power and propulsion system are the key components of an electric propulsion system by which performance can be compared. Maximizing the propulsion system's specific power is the best approach to increasing the specific power of any spacecraft which utilizes electric propulsion. A fundamental question comes into play, what is a competitive specific power? Is there a way to find out what a competitive specific mass is regarding other propulsion technologies?

Conjunction class round trips to Mars

There are many different approaches to getting to Mars. Typically, the lowest ΔV approach which is regularly available is known as a conjunction class mission. A conjunction class mission regularly occurs approximately every 2 years when Earth and Mars line up. **Fig. 4** graphically depicts conjunction class mission.

Earth is green, Mars is red, the Sun is yellow, and the path of the rocket is in purple. The conjunction class mission begins with a transfer from Earth to Mars. The rocket must stay on Mars until the planets again line up for a transfer. The overall mission takes over 2 years due to the timeframe of the planets aligning. NASA and other organizations have put a lot of research into characterizing the conjunction class mission to Mars. **Fig. 5** below summarizes the results from NASA's conjunction class mission analysis in the Human Exploration of Mars Design Reference Architecture 5.0 over the 2030–2046 timeline.

All missions begin in 440 km circular low Earth orbit (LEO). There are three different maneuvers. First is the trans-Mars insertion (TMI) which takes around 200 days to propel the rocket from Earth toward Mars. Second is the Mars orbital insertion (MOI) that puts the rocket in orbit around Mars (250 km perigee and 33,793 km apogee). The rocket stays in orbit while the crew spends over 500 days on Mars. Finally, the trans-Earth injection (TEI) propels the rocket home from Mars (**Table 7**). The NASA DRA 5.0 contains analysis for both a chemical and nuclear thermal mission to Mars using the ΔV shown in **Fig. 5** for the near-term worst-case year 2037. The results of the nuclear thermal and chemical design process are shown in **Fig. 7**.

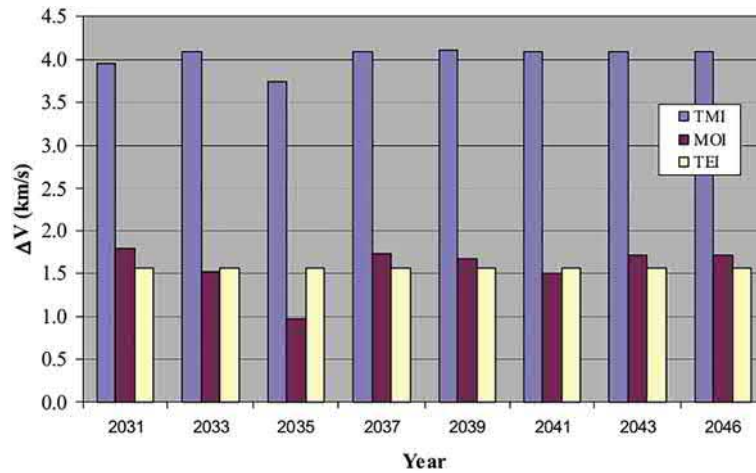


Fig. 5 High thrust conjunction class mission ΔV from NASA DRA v5.0 July 2009 Pg. 23 (Drake, 2009).

Table 7 NASA Mars design reference architecture 5.0-2037 Mars mission 374 transit days.

	Chemical	Nuclear thermal
Initial mass low Earth orbit	486.0	356.4
Stages	4	1 (plus one drop tank)
Payload (metric tons)	51.9 (10.7%)	62.8 (17.6%)
Fuel mass (metric tons)	347.6 Liquid H_2/O_2 (71.5%)	202.7 Liquid H_2 (56.9%)

What about electric propulsion? What would an electric propulsion spacecraft look like that has similar capabilities to the NASA Design Reference Architecture? A study was published by Ilin in 2013 that analyzed similar missions using nuclear electric propulsion (Chang-Díaz et al., 2013). The mission utilized the same starting location, initial mass, and payload assumptions as the NASA DRA 5.0 missions looking at nuclear thermal propulsion and can also be compared to the chemical propulsion DRA 5.0 mission as Fig. 6 gives the results from Ilin's work.

Ilin's work looked at several different $\alpha_{PPS_{jet}}$. This figure was made by using the data from his research which used the VASIMR engine and back calculating the specific mass of the power and propulsion system and verified in correspondences with the author including rerunning the calculations for 2037 rather than 2035. On the y-axis the in-flight days to and from Mars is shown. A $\alpha_{PPS_{jet}}$ of approximately 20 kg/kW_{jet} was able to reach the 374 days transit time for the conjunction class mission. This corresponds to a jet power of around 5.6 MW. The payload for the mission was 60 metric tons and the system started in low Earth orbit with an initial mass of 356 metric tons. By matching the payload and the number of days to and from Mars for the nuclear thermal propulsion the nuclear thermal and electric propulsion systems are objectively equivalent. Table 8 below summarizes the nuclear electric mission by Ilin and the DRA missions for chemical and nuclear thermal propulsion.

In the table you can see that in general the three missions are matched well in terms of the days to Mars and for the nuclear thermal and nuclear electric missions the payload is approximately equal. The specific impulse for the three systems varies significantly between all three with the electric propulsion having an average specific impulse approximately an order of magnitude higher. The ΔV for the missions is the same for chemical and nuclear thermal propulsion but almost four times higher for the electric propulsion mission. Some of this is due to gravity losses, however most of the extra ΔV required for electric propulsion is required due to excess ΔV for average velocity matching.

Fig. 7 brightly depicts the differences between the three propulsion systems. The percent of the rocket required as propellant is the important distinction. In the case of chemical, 71.5% is required as propellant shown as the red portion of the top bar. Nuclear thermal does much better with 59.6%, however electric propulsion is reducing the propellant to 46.6% of the rocket required as propellant. The green portion of the bar represents the payload. The black portion is the mass of the rocket that is neither propellant not payload. For the four-stage chemical propulsion system the black would represent the mass of the fuel tanks and the rocket engines and turbomachinery associated with the stages. The black portion of the nuclear thermal rocket would also represent the fuel tanks and engine. For the electric propulsion system, the mass of the tanks and power and propulsion system is explicitly shown. The electric propulsion's reduced need for propellant looks great, however there is one caveat. Unlike the other two propulsion systems it requires an electric power source.

What exactly is the "power source?" Chemical propulsion systems do not need a power source because the propellant provides the power for the rocket. Nuclear thermal rockets require a reactor as a power source to heat the propellant. The benefit of nuclear thermal propulsion is that the reactor power source can be provided by thermal energy. Electric propulsion on the other hand

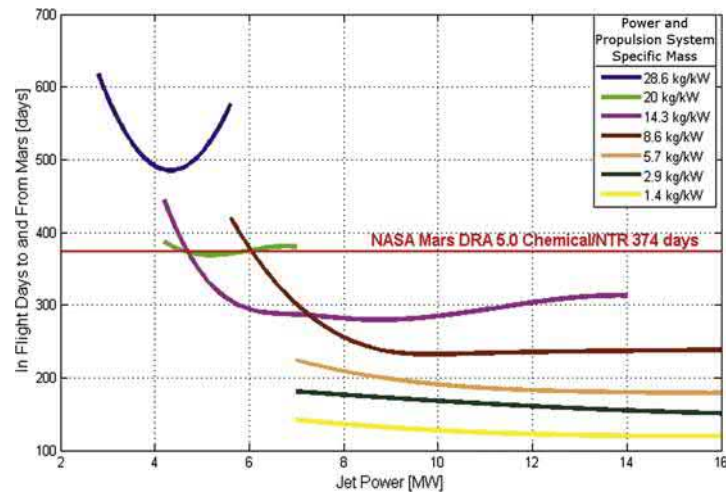


Fig. 6 Jet specific power, power, and flight time for a 2037 flight to Mars using VASIMR with 70% electric power to jet power efficiency, and a 62 Metric Ton Payload (Chang-Díaz et al., 2013).

Table 8 Summary of NASA DRA 5.0 chemical and nuclear thermal propulsion missions.

	<i>Chemical</i>	<i>Nuclear thermal</i>	<i>Nuclear electric</i>
In flight days	374	374	370
Mission type	Conjunction	Conjunction	Conjunction
Initial mass low earth orbit	486.0	356.4	365.4
Payload (metric tons)	51.9	62.8	62.0
Specific impulse (s)	450	950	Average 4644
Year	2037	2037	2037
ΔV (km/s)	7.25	7.25	30.2
Tankage mass fraction	10%	10%	10%
Stages/tanks	4 stage	Single stage with jettison tanks	Single stage with jettison tanks
Propellant	H ₂ /O ₂	H ₂	Argon
Starting orbit	LEO	LEO	LEO

requires a power conversion system to turn the thermal energy into electrical power. Power conversion systems require many components including a heat rejection system. For fission systems, the power conversion system is typically much heavier than the reactor itself. A common criticism of electric propulsion is that while the propellant requirements are clearly reduced the mass of the power source required to run the electric propulsion has taken the place of the propellant. This is a valid criticism. However, the counter argument in favor of electric propulsion is this: nuclear thermal and chemical propulsion have hit their physics limits. Both of these technologies are bound by the laws of physics to have exceedingly high propellant requirements; later generation of chemical and nuclear thermal rocket will not be significantly improved. Chemical rockets are limited by the amount of energy attainable from combustion, and nuclear thermal rockets are limited by the melting point of the rocket components.

Electric propulsion is challenging due to the engineering of a strong robust power source. However, the power source is an engineering problem that has not come close to approaching a physics limit. Higher performing electric propulsion systems can reduce the propellant required by a much greater degree and later generation of power systems can greatly outperform previous generations.

The power and propulsion system

It is useful to define the electric engine efficiency:

$$\eta_{engine} = \frac{P_{jet}}{P_{power\ source}} \quad \text{Electric Engine Efficiency} \quad (13)$$

Where η_{engine} is the efficiency of the engine in transferring the energy from the power source ($P_{power\ source}$) into the propellant jet (P_{jet}). Thruster efficiency has multiple factors that affect the efficiency including the nozzle efficiency, power density, propellant density, propellant type, input voltages, input amperages, and many other factors which are thruster specific. Eq. (14) now expands from Eq. (12).

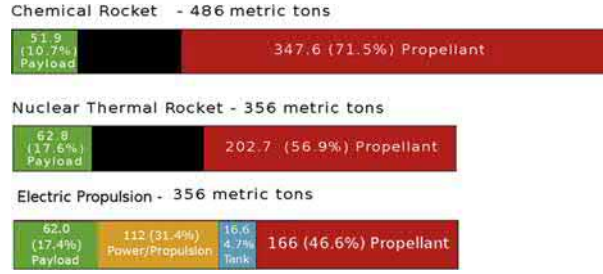


Fig. 7 Comparison of NASA DRA 5.0 chemical and nuclear thermal and Ilin's electric propulsion round trip.

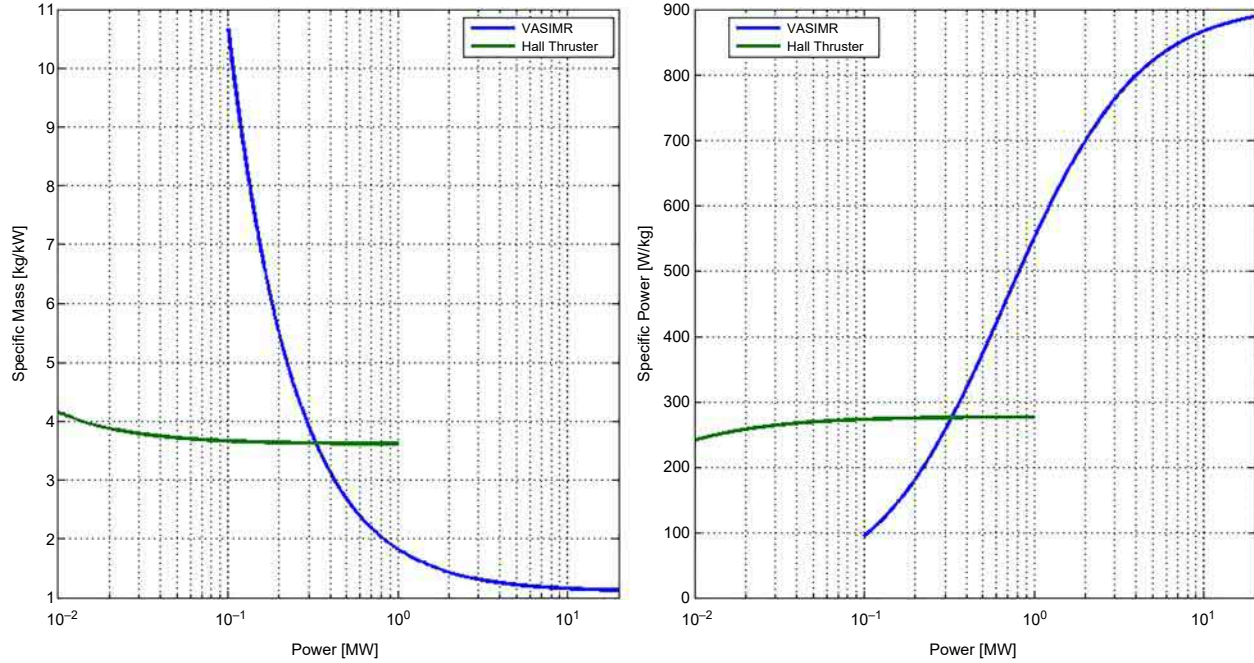


Fig. 8 Specific mass and specific power scaling for hall thrusters and the VASIMR engine (Chang-Díaz et al., 2013; Hofer and Randolph, 2013).

$$\alpha_{PPS_e} = \eta_{engine} \alpha_{PPS_{jet}} = \alpha_{power\ source_e} + \alpha_{engine_e} \left[\frac{kg}{kW_e} \right] \text{ Power and Propulsion System Electric Power Specific Mass} \quad (14)$$

Electric propulsion engines have established efficiencies (η_{engine}) and specific masses (α_{engine}). Fig. 8 shows the specific mass estimates for Hall thrusters and the VASIMR engines as a function of power.

Table 9 shows the efficiency, specific power, and specific mass for several electric engines cited in different studies.

The required power source specific mass can now be calculated by assuming a specific thruster.

$$\alpha_{power\ source} = \eta_{engine} \alpha_{PPS_{jet}} - \alpha_{engine} \text{ Power Source Specific Mass} \quad (15)$$

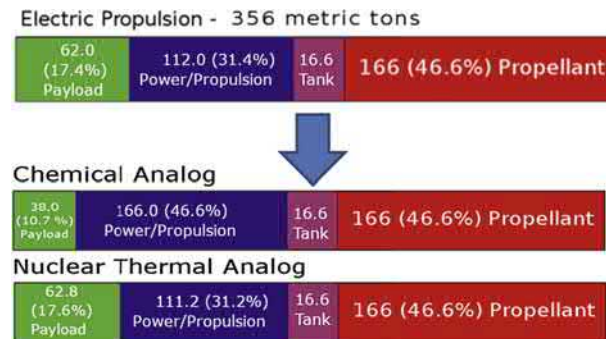
Ilin's work found that an electric propulsion system comparable to chemical and NTP (for a human Mars mission) would have a total power of 5.6 MW of jet power and a power and propulsion specific mass of 20 kg/kW. By using this data in combination with the propulsion system characteristics, we can derive a power source requirement.

The chemical propulsion mission had a different initial mass in low Earth orbit, as well as a different payload mass fraction (10.7% as opposed to 17.4%). In orbital mechanics as well as with the rocket equation, a mission with the same mass fractions and acceleration profiles will produce the same trajectory. Therefore, the initial mass isn't a major concern as long as the payload fractions match. We can therefore make the findings of this analysis applicable to a comparison to both nuclear thermal and chemical systems.

Assuming a thruster specific mass of the of 1.2 kg/kW_{in} and efficiency is approximately 70% using the VASIMR engine data, we can derive the specific mass of the power system that can achieve the $\alpha_{PPS_{jet}}$ that can achieve parity with the respective propulsion systems in Fig. 9. The chemical payload was modified to 38 metric tons to maintain 10.7% payload fraction. This yields a nuclear thermal equivalent $\alpha_{PPS_{jet}}$ of 19.9 kg/kW_e and a chemical $\alpha_{PPS_{jet}}$ of 29.6 kg/kW_e. Table 10 calculates the equivalent power source specific masses.

Table 9 Thruster efficiencies (Chang-Díaz et al., 2013; Hofer and Randolph, 2013).

Thruster	Efficiency (%)	I_{sp} (s)
Hall thruster (Smart 1)	60–75	1000–1600
NEXIS (JIMO)	70–78	6000–7500
NEXT (Dawn)	60–70	3100
VASIMR	60–72	2000–5000

**Fig. 9** Chemical and nuclear thermal analog electric propulsion missions.

A competitive first generation nuclear electric propulsion system should have a performance that is similar or better than other near-term and state-of-the-art systems. A competitive performance would justify the research and development placed into such a technology. Electric propulsion systems are under development that can fulfill the efficiency and specific mass requirements. The power source however is the key area where research and development are needed. The author recommends that a first-generation nuclear electric propulsion system target a specific mass of 10 kg/kW_e. Such a system would clearly outperform state of the art chemical system and would be undoubtedly be competitive likely to outperform nuclear thermal propulsion systems in many scenarios even accounting for some margin.

Historical power source specific mass requirements

In the previous section power source requirements were derived for competitive crewed missions to Mars and used as a general guideline for a competitive electric propulsion system. While the crewed human mission scenarios are a good baseline, it is very challenging to compare performance in a single number. For some missions where time is a major constraint or where ΔV requirements are very low, high thrust propulsion will be very competitive. Aerobreaking, gravity assists and in-situ propellant production are all additional wildcards that can greatly change which propulsion method is more optimal. Table 11 below details different studies that summarize the power and propulsion requirements for different missions.

For example, an unmanned mission to Mars or a mission to an asteroid would most likely have a lower threshold for power and specific power than more difficult manned mission to Mars due to different time constraints. Many sources have cited the specific mass 0.5–1 kg/kW_e as a holy grail of electric propulsion and power systems and 5–20 kg/kW_e as empowering manned missions to Mars.

Fast way to get to Mars one-year round-trip

A one-year round trip to Mars is attractive proposition from the prospective of human safety. What would make a trip possible? One way to travel to Mars and return to Earth in 1 year is with the orbital maneuver described in Fig. 10.

The one-year Mars trip begins in a similar way to the 2-year conjunction class mission though the spacecraft leaves earlier traveling from Earth to Mars. Upon arrival Mars is visited for about a month before returning to Earth using a much longer transit leg that travels slightly inside Earth's orbit around the Sun. Analysis was done by Ilin in 2013 to assess the chemical and nuclear thermal rocket ΔV requirements for a 1 year mission to Mars starting in the year 2033 (Chang-Díaz et al., 2013). His result was that a 1-year mission to Mars required 18 km/s of ΔV for high thrust propulsion systems. Ilin also calculated the ΔV required for an electric propulsion system with 8 MW power and an initial mass of 356 metric tons at 28 km/s. Unlike the previous analysis these missions are assumed to start at the L1 Earth Moon Lagrange point. Fig. 11 illustrates the size of the different propulsion systems to achieve a one-year mission. ****

The graph on the left represents the 18 km/s ΔV for both chemical and nuclear thermal systems based on the rocket equation. The graph on the right represents the 28 km/s ΔV for the electric propulsion. On the x-axis is the I_{sp} and on the y-axis is the percent of the rocket that is not propellant.

Table 10 Power source requirement derivations for conjunction class missions.*Nuclear thermal equivalent power source*

$$\alpha_{\text{power source}} = 0.70 \left[\frac{\text{kW}_{\text{jet}}}{\text{kW}_e} \right] \times 19.857 \left[\frac{\text{kg}}{\text{kW}_{\text{jet}}} \right] - 1.2 \left[\frac{\text{kg}}{\text{kW}_e} \right]$$

$$\alpha_{\text{power source}} = 12.7 \text{ kg/kW}_e$$

Chemical power source equivalent

$$\alpha_{\text{power source}} = 0.70 \left[\frac{\text{kW}_{\text{jet}}}{\text{kW}_e} \right] \times 29.64 \left[\frac{\text{kg}}{\text{kW}_{\text{jet}}} \right] - 1.2 \left[\frac{\text{kg}}{\text{kW}_e} \right]$$

$$\alpha_{\text{power source}} = 19.5 \text{ kg/kW}_e$$

Table 11 Summary of several relevant electric propulsion use cases and specific mass.

Mission study	$\alpha_{\text{power source}_e}$ (kg/kW _e)	Descriptions
Brown and Coates (1966)	16.9	Early suggested human mission to Mars
Jahn (1968)	5.0	Suggested point at which electric propulsion is clearly advantageous over nuclear thermal propulsion for manned missions
Clark et al. (1994)	9.2	Human mission to Mars
McGuire et al. (2006)	7.1	Human Mission to Martian Moons
McGuire et al. (2006)	8.1	Cargo Mission to Martian Moons
Ilin et al. (2011)	0.5	39 Day to Mars
Mercer et al. (2011)	26.8	Near Earth Asteroid
Manzanek (2013)	20	Competitive specific mass vs. NTP for opposition style Mars missions
Gilland et al. (2011)	17.0	Cargo missions to Mars
Gilland et al. (2011)	14.0	Cargo mission to Mars
Gilland et al. (2011)	12.9	Cargo missions to Mars
Chang-Diaz et al. (2013)	10.0	One Year Mars Round Trips
Chang-Diaz et al. (2013)	14	Suggested point at which electric propulsion can match nuclear thermal performance for number of days in transit
DRM 2002, Drake (2009)	6.7, 5	590 day and 550 day opposition round trip to Mars
Clark et al. (1994)	11.1	960 day conjunction class round trip to Mars

Assuming an I_{sp} of 450 s and using the rocket equation, a chemical rocket would require over 98% of its initial mass to be propellant. This is not feasible. The mass of the propellant tanks alone would be more than 2% allowing no room for a payload. Using chemical propulsion to achieve 18 km/s of ΔV cannot be done unless an “in situ” propellant from Mars (propellant mined on Mars) could be utilized to refuel the rocket around Mars.

For a nuclear thermal system with an I_{sp} of 1000 s a mission could be achievable. With 16% of the rocket remaining after propellant is loaded perhaps half or 8% could be used for the payload. Assuming that the payload is 50 metric tons then an approximate nuclear thermal rocket would be around 1000 metric tons. The propellant requirements for the electric system are heavily affected by the I_{sp} . Going to a higher I_{sp} can greatly reduce the propellant requirements and increase the payload percentage.

The point here is that electric propulsion opens door to incredible missions that could not be otherwise accomplished with existing technology. Electric propulsion could enable manned access to Jupiter, Saturn, and beyond. Chemical simply cannot enable these missions. Even with in situ propellant generation chemical would be very hard. Nuclear thermal perhaps could be adapted to advanced missions across the Solar System but with smaller capability to transport. There is a tipping point at specific masses of 10 kg/kW_e and below that open entirely new kinds of missions that revolutionize space transportation.

Electric propulsion technologies open the door to the Solar System. Electric propulsion comes the closest to being a real spaceship. A craft that can travel from point a to point b in the Solar System without dropping parts of the spacecraft behind for staging. Electric propulsion does not require elaborate infrastructure to put in place. A single electric propulsion craft could be self-sufficient for a trip to and from Earth without requiring “gas stations” along the way that require handling of cryogenic propellants mined from a celestial body. Electric propulsion possesses enough excess ΔV that there is significant operational flexibility. If a crew sets off to Mars and a crew member gets very sick, the spacecraft has the flexibility to reroute to earth.

Electric propulsion can utilize many types of propellant. Typically, Argon, Xenon, or Krypton gas is used currently, however water, Iodine, and even iron can be used as a propellant as any atom can be ionized and accelerated with an electromagnetic field.

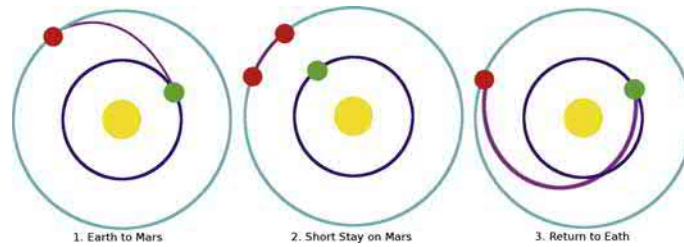


Fig. 10 One year Mars trip.

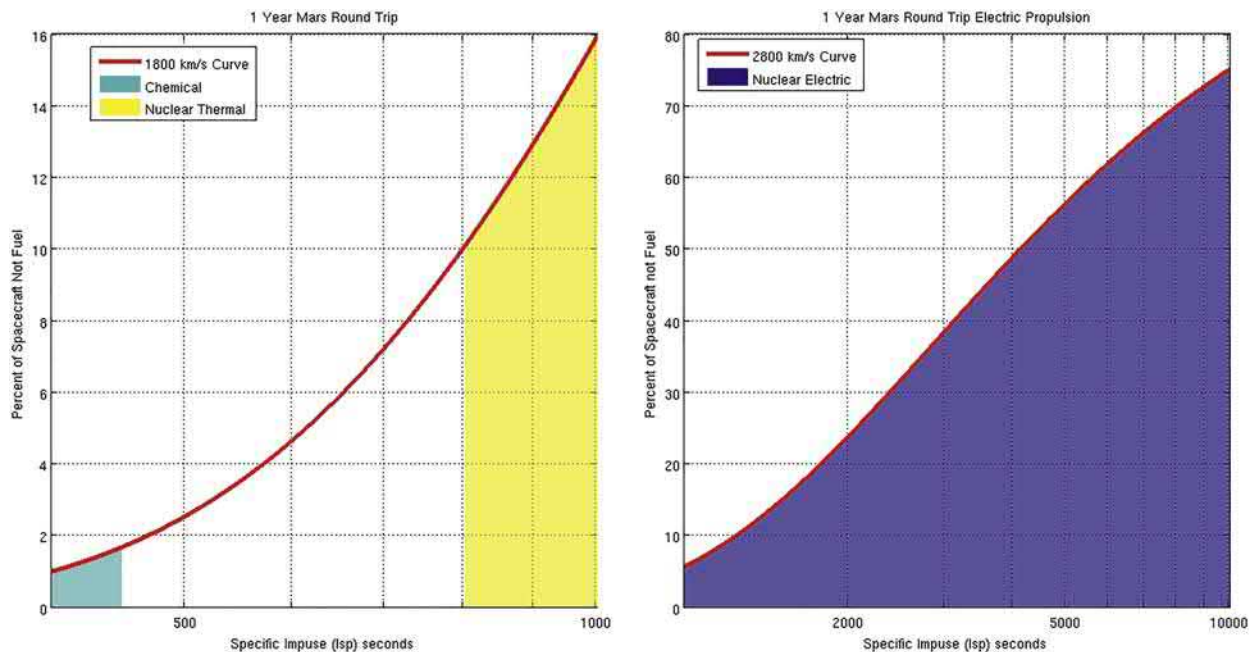


Fig. 11 Isp vs. percent of rocket that is not propellant for both 18 km/s and 28 km/s ΔV .

Therefore, any celestial body could be used to gather in-situ electric propulsion propellant without significant need for mining and extraction; making it much easier to resupply an electric propulsion spacecraft.

The case for electric propulsion

Electric propulsion systems cannot be used within Earth's atmosphere because they need a vacuum to operate. However, once delivered into the vacuum of space, electric propulsion is much better suited to the environment when compared to traditional chemical power systems. Dr. Chang Diaz, the developer of the VASIMR electric engine, likens it to shifting in a car. "You start in first gear, and then you shift to second and third, and so on."

First gear is the chemical propulsion system that delivers the spacecraft to Earth orbit. Chemical rockets are a necessity for quickly allowing a rocket to gain speed and height without crashing the rocket into the ground. Chemical rockets can provide the high thrust to accomplish an orbit. However, the first-gear chemical rocket is not suitable for higher speed, more fuel-efficient travel.

Second gear is an electric propulsion system. While in orbit around the Earth, the spacecraft does not need the high thrust that a chemical engine provides. There is no immediate danger of crashing back into the Earth, so electric engines can use fuel more slowly and efficiently. Similar to the difference between drivers who floor their cars so they won't get good gas mileage and drivers who slowly accelerate their vehicles, electric propulsion gets 10 or more times more ΔV out of its fuel than chemical rockets even when considering the Orbeth losses.

Traditional rockets are expendable. A large rocket begins on the launch pad and comes back as a small capsule. The remainder of the rocket is sometimes recovered and reused, but typically the stages are dropped in the ocean, burn up in the atmosphere, or are dropped in the middle of space. The reason for this is because rockets gain more ΔV if they drop their used, empty fuel tanks before using a new fuel tank.

Electric engines change this staging paradigm. Because electric engines use much less fuel than chemical engines, dropping stages is not as advantageous. This means that electric propulsion rockets making a round trip back to Earth will return in the same state

they left except their propellant tanks will be empty. If the components on the electric propulsion vehicle are built with longevity in mind, the same electric propulsion spacecraft could be used for many trips. Instead of thinking about electric propulsion as a one-shot rocket, electric propulsion can be thought as a car that would just have to replenish its fuel tank to go on the next journey.

Moore's Law is the idea that a technology can improve quickly. For example, computer transistors per square inch of Silicon over the last several decades has followed a trend of doubling every several months. The benefit of investing in a technology that follows Moore's Law is profound after a few generations of doubling.

Liquid and solid rockets have dominated the space propulsion field since before the space race of the 1960s, yet modern chemical rockets perform only slightly better than the first prototypes. The process of combustion is the limiting factor in increasing Isp.

The chemical energy from the combustion provides the propellant with the energy to achieve the exit velocity. There is no clear path to increase the energy derived from combustion. Small gain can be made with different fuel flow cycles, nozzles, higher temperature materials, and fuel additives, but there are few avenues to exponential growth of performance.

Nuclear thermal rockets are limited by the material properties of the rocket engine. Their ISP can be increased by increasing the temperature of the nuclear core that heats the propellant. However, the core's temperature cannot exceed its melting point. Small gains can be made with different ceramic and high-temperature refractory metals; however, there is no foreseeable path for exponential growth.

Electric propulsion has made rapid advances since its beginnings and each generation of electric propulsion has greatly improved upon the last. Electric systems were first used for attitude control and station keeping in the 1970s. Since that time, have greatly increased their Isp and their electrical efficiency. Electric systems are constantly improving. Electric engines are fundamentally based on electric and magnetic fields. Recently inventions (such as high-field strength, lightweight ceramic magnets, and the increasing capacity and temperature ranges of superconducting magnets) have greatly increased the Isp capabilities of electric propulsion, and it appears that they will continue to do so over the coming decades.

There are many types of electric propulsion systems such as hall thrusters, ion engines, arc jets and magneto-plasma-dynamic thrusters (MPD), to name only a few. There are electric thruster designs which span the range of I_{sp} from 500–50,000 s or more with efficiencies ranging from 30–75% efficient. Mature and flown electric thrusters exist for watt to multiple kilowatt levels. At the time of this writing, the nested Hall thruster and VASIMR engine are working toward flight ready systems for power levels at 100 kW (Shark et al., 2019; Anon, 2020). These systems are capable of human missions to Mars. This document will not focus on describing the state of electric propulsion beyond this general summary, but many well written sources exist (Szabo, 2019; Brophy et al., 2011). While there are challenges moving forward, electric propulsion is clearly on a path toward success.

Electric propulsion systems come in three main varieties: electrostatic, electromagnetic, and electrothermal. Electrothermal propulsion is not of much interest here as it uses electricity to heat a propellant directly using an electrode or electric arc. This type of heating cannot achieve an I_{sp} of much over 1000 s. Electrostatic and electromagnetic propulsion utilizes electric and magnetic fields to isolate and heat gas to extreme temperatures. Electrostatic use only electric fields accelerating charged ions of propellant over high voltages to accelerate and heat the ion to a high I_{sp} . The main types of electrostatic thrusters include the gridded ion thruster and the Hall thruster. These types of thrusters require high voltage direct current from a power conditioning system.

Electromagnetic thrusters use a combination of magnetic and electric fields which are often referred to together as the Lorentz force. There are many types of electromagnetic thrusters. Some of the more well-known thrusters include the VASIMR engine and several different kinds of electrode and electrodeless Lorentz thrusters as well as pulsed magneto plasma dynamic engines. These thrusters typically require superconducting magnets and are typically powered by alternating current. Magnets keep charged propellant particles trapped and the propellant is excited often using radio waves or inductive heating. Generally electrostatic thrusters are more mature and simpler technologies, but electromagnetic thrusters can reach higher power densities.

The power source

In the proceeding sections, a specific power of 10 kg/kW_e was established. What kinds of technology are available to achieve the specific power numbers? Historically the only sources of power in space are fission and solar. There are other concepts such as beamed power and fusion that have potential; however, these concepts are removed from consideration because they are not mature technologies or would require a significant buildup of infrastructure over the course of decades.

The case for solar power

Solar powered electric propulsion has been effectively used in several missions to the asteroid belt and within Earth's neighborhood even recently extending to Jupiter. The first use of solar power for beyond Earth orbit electric propulsion was on NASA's 1998–2001 Deep Space 1 spacecraft that passed by an asteroid and a comet. Its 2.5 kW solar arrays powered an NSTAR ion engine. This was followed by Japan's 2003–2010 Hayabusa spacecraft that completed a sample return mission from an asteroid. The European Space Agency also sent a 1.2 kW solar electric spacecraft spiraled out from Earth's orbit to the moon over the course of 3 years. The Dawn spacecraft launched in 2007, visiting the dwarf planet Ceres and the large asteroid Vesta. Solar electric's flight heritage demonstrates that it is a candidate for reliable solar system transportation.

A basic analysis of a solar array's specific power can be found by using the following equation:

$$\alpha_{\text{solar power source}} = \frac{S}{I\zeta} \quad \text{Solar Power Source Specific Power} \quad (16)$$

Where S is the mass per unit area of the solar array including all of the energy components associated with processing the power generated; ζ is the efficiency of the solar cell or solar concentrator in processing that power into electricity; and I is the intensity of sunlight upon the solar panel. Overall power of the solar array is also important to minimize mission times. Solar power is dependent upon the sun. The following graph depicts the solar power density as a function of the distance from the sun. Fig. 12 depicts the solar power available as a function of distance from the sun.

Earth (at 1 AU from the Sun) receives 1366 W/m^2 of solar power. Mars (1.52 AU) receives 43% of that of Earth, meaning that a solar panel array would require over twice the area in orbit of Mars as it would in orbit of Earth. Jupiter has approximately 1/20th the solar energy density of Earth. In addition to distance, temperature is a key parameter that determines efficiency. Temperatures exceeding 100 degrees C for Silicon cells disable them as the thermal energy of the electrons exceeds the bandgap energy. Other semiconductors such as GaAs can achieve higher temperatures but lose efficiency as temperatures exceed $70\text{--}100^\circ\text{C}$. Low temperatures drop efficiency as well in solar cells and -20°C is usually considered to cause low intensity low temperature (LILT) losses. For this reason, solar cells may need to be cooled (or heated), especially if using mirrors to concentrate sunlight or when coming closer to the sun. It is for this reason that solar cells will not be able to greatly exceed the 1300 W/m^2 intensity of sunlight seen near earth without active cooling systems. Solar thermal technologies (as opposed to solar photovoltaic) do not have these constraints and use mirrors to focus sunlight on a heat engine producing electricity. Solar cell efficiencies have been steadily increasing over the last few decades. As multi-junction cells have matured, efficiencies for space solar power can easily obtain close to 40% efficiency.

Solar cells suffer from degradation in the space environment. Generally, that degradation is caused by solar particles. Typically, high efficiency solar arrays degrade 3–6% each year depending on the orbit and particle flux. During a solar event such as a flare they can degrade quickly. The Hayabusa electric propulsion spacecraft experienced a moderate solar flare on its voyage to the asteroid Itokawa. The solar flare damaged the solar cells reducing the cell efficiency by approximately 50%. Photovoltaic cells for spacecraft are designed to be higher efficiency than terrestrial and often have radiation hardening features which reduce degradation such as coatings and redundancy. However, the cost and mass of the systems are affected. Typical space rated solar panels are 100 times more expensive per watt than ground-based systems not counting the launch costs.

Space deployed solar power systems have been steadily increasing as shown in Fig. 13.

Note that the power shown is in the power of the spacecraft's solar panel array at 1 AU. BOL stands for beginning of life as solar panels slowly degrade over time. The International Space Station generates 240 kW of electricity, and it is the highest power, space-based solar panel array.

New launch vehicles are being developed and the cost of delivering mass to orbit has dropped dramatically from \$20,000 per kg to under \$2000 and is may drop under \$1000 per kilogram in the 2020s; a promising development to achieve megawatt scale solar arrays.

Mass per solar panel area depends both on solar panel technology and lightweight structures that can be developed to keep the panels connected to the spacecraft. Having motors and actuators that can direct and point the solar panels to keep the correct orientation to the Sun is also a necessity as well as the technology that manages the solar panels output. Cooling and heating systems may be required. Radiation hardening increases mass as cover glass, thick coatings, and redundancy may be required. The power processing unit (PPU) which processes the input voltage and amperages must also be included in the mass budget as well. Rigid panel solar cells using triple junction cells have been used extensively in space and have specific masses in the range of $15\text{--}40 \text{ kg/kW}_e$ at 1 AU. Flexible fold out arrays can achieve specific masses in the 10 kg/kW_e at 1 AU. Thin film solar arrays are very promising as they are extremely thin

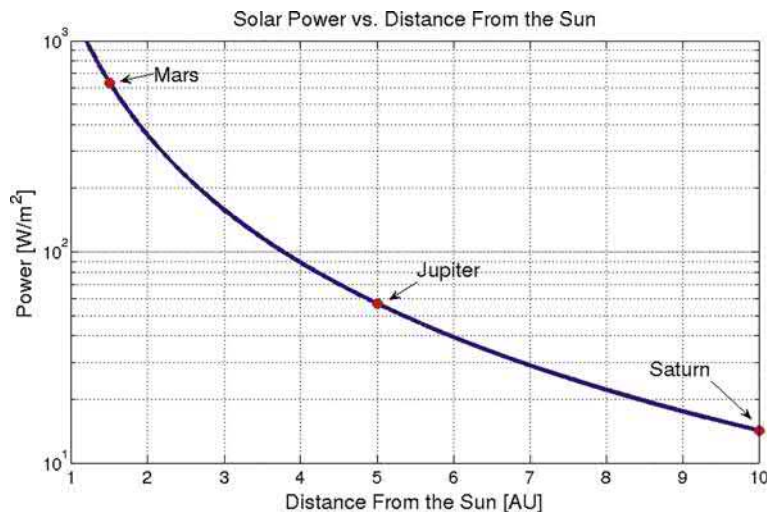


Fig. 12 Solar power density vs. distance from the Sun.

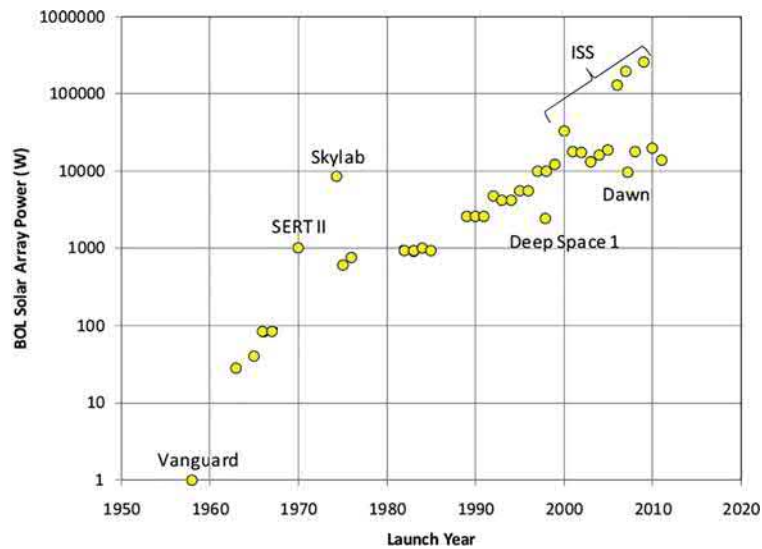


Fig. 13 Space deployed solar power vs. time (Brophy et al., 2011).

Table 12 Energy density.

Fuel	Specific energy (MJ/kg)
Antimatter	90,000,000,000
D-T Fusion	350,000,000
U-235 Fission	82,000,000
Hydrogen-oxygen	15
Gasoline-oxygen	8
Lithium ion battery	1
Lead acid battery	0.2

sheets and can yield specific masses of as little as 0.5–1 kg/kW_e at 1 AU. However, thin film technologies have significant challenges ahead to reach these performance levels and are much more susceptible to space radiation (Beauchamp et al., 2015).

Solar power systems have already shown they can achieve specific masses of 15 kg/kW_e in missions such as NASA's Juno. Near-term technology demonstration such as SpiderFAB, Megaflex, and Rosa and will lead to 10 kg/kW_e (Szabo, 2019; Beauchamp et al., 2015). Solar has and will continue to be a key power source for electric propulsion. Advances in thin film cell technology will continue for the foreseeable future. Challenges based on large scale engineering, long-term reliability, and operational environments can be solved in the long-term and there is a strong industrial push from space based solar power entrepreneurs. The main long-term challenge for solar that remains unsolvable is distance from the sun. Missions beyond Mars will be challenging. Solar may be constrained to the inner solar system. But, within the inner solar system solar electric propulsion will have a future role to play.

The case for a nuclear power source

One kilogram of Uranium-235 has 82 million times the energy of a lithium ion battery (Table 12).

Fusion and antimatter technologies hold promise for the future and both have more energy potential. But fission is a straightforward technology that is ready and capable of providing reliable Solar System transportation and achieving the specific power requirement.

Fission power is rugged. It can operate regardless of proximity or orientation to the Sun and is largely unaffected by the environment. Missions to Jupiter Saturn and other deep space planets would only be possible with nuclear fission technology. Fission would be unaffected by the shadows of planets, or the intense radiation fields from the Van Allen belts or Jupiter's strong magnetic field. Solar flares would pose little threat and micrometeoroid impacts would be mitigated both from the compact size of the spacecraft and the ability to carry more shielding material.

Fission technology also allows scientists and future astronauts to have access to an abundance of energy that enables a new wave of energy intensive activities. For example, access to fission power on the surface of Mars would allow for large scale in-situ operations. Mars has all the ingredients to produce oxygen, water, chemicals, methane and other materials needed for settlement. With

a fission power source in situ manufacturing becomes possible. Fission power systems produce heat that must in general be converted into electricity, however for many of the in-situ processes heat can be used directly. Another chapter in this encyclopedia discusses the benefits and challenges of space fission power (Poston, 2021a).

Space nuclear power flight heritage

Nuclear power's flight heritage is strong. Nuclear Radioisotope Thermal Generators (RTGs) have been used in 44 US deep-space missions, including the Apollo, Cassini, Viking, Mars Science Lab, and Voyager missions. RTG's are passive nuclear power sources that operate on radioactive decay and can reasonably provide around 1 kW of electric power even to the fringes of deep space (Buden, 2011b; Bennett, 2021).

Nuclear thermal rockets have also been developed. From 1959–72 over 20 major ground tests were performed demonstrating that the technology was close to flight-ready (Graham, 2021). Nuclear thermal rockets use the heat energy of the reactor to increase the escape velocity of the fuel (Buden, 2011c; Poston, 2021b).

Fission reactors have been used on over 30 Earth-orbiting satellites, including the SNAP10A, the Russian BUK, and Cosmos radar systems (Voss, 2021). Fission systems are active nuclear power sources that have potential for very high specific power.

Existing and well-studied fission power systems

To date, 33 nuclear fission systems have been launched into space. The first reactor unit was the SNAP10A launched in 1965 as a technology demonstration unit. The last flight reactor was launched in 1988 by Russia. These 33 systems generated on the order of a kilowatt and were used for onboard power rather than for propulsion (Table 13).

Before the advent of electric propulsion, there was little need for medium- and high-power systems because there was no use for them. The Medium Power Reactor Experiment (150 kW) and SPUR program (300 kW) were both space focused ground test reactors from the 1960s; however, they were cancelled because of "lack of mission."

In the mid-to-late 1980s, as electric propulsion began to show signs of maturity, the SP-100 program was started to develop a 100-kW power system. The design was heavily studied and was moving toward flight testing; however, the fall of the Soviet Union, lack of mission, and other factors resulted in its cancellation in 1993.

In 1999 NASA's Deep Space One spacecraft (a solar-powered mission) demonstrated the first use of electric propulsion in beyond Earth orbit applications. This mission clearly demonstrated electric propulsion as viable and helped to spur NASA to again pursue nuclear electric propulsion.

In 2003 NASA Director Sean O'Keefe approved the Prometheus mission to three of Jupiter's moons using nuclear electric propulsion. The Prometheus mission was also known as the Jupiter Icy Moons Orbiter (JIMO) (Buden, 2011a). Prometheus was a 200-kW electric nuclear electric system powered by NEXIS ion engines and Hall thrusters. Work was completed on Phase A of the project, meaning that there were no significant issues with Prometheus, and it was ready to move on to the final design. In 2005 the Administration lowered the priority of the Prometheus mission in favor of the Constellation mission and returning to the Moon. This change in priorities, coupled with a change in the NASA director, the Colombia Shuttle accident, and the \$17 billion estimated Prometheus price tag, resulted in the program's cancellation in 2005. Prometheus represents the first, and thus far, only well hashed out attempt at nuclear electric power for propulsion.

In 2018 NASA completed the nuclear testing of a 1-kW electric Kilopower space reactor prototype (Poston, 2021c), however the project has not proceeded to a flight program. The Jet Propulsion Laboratory has identified several potential NEP missions that could be enabled by Kilopower systems (Casani et al., 2020).

Previously purposed fission power systems

Prometheus is the last well-studied nuclear electric design that could have proceeded as a flight project. However, several designs have been put forth that demonstrate the reasonable capability of nuclear technologies as shown in Fig. 14.

Table 13 Space fission history (Buden, 2011a,b,c; Prost, 2020; Voss, 2021).

System	Launches	Ground reactor tests	Fuel	Thermodynamic cycle	Power	Reactor specific power (W/kg)	Reactor specific mass (kg/kW)
Snap10A (Launched 1965)	1	4	UZrH	Thermoelectric	500 W	1.2	830
RORSAT (1967–1980s)	31	3	UO ₂	Thermoelectric	1.7 kW	2.2	450
TOPAZ (1987–1988)	2	4	UO ₂	Thermionic	5 kW	5	200
SP-100 (Cancelled 1993)	0	0	UN	Thermoelectric	100 kW	19	53
Prometheus (Cancelled 2005)	0	0	UO ₂	Brayton	200 kW	30	33
Kilopower (2015–2018)	0	1	UMo	Stirling	1 kW	3.5	400

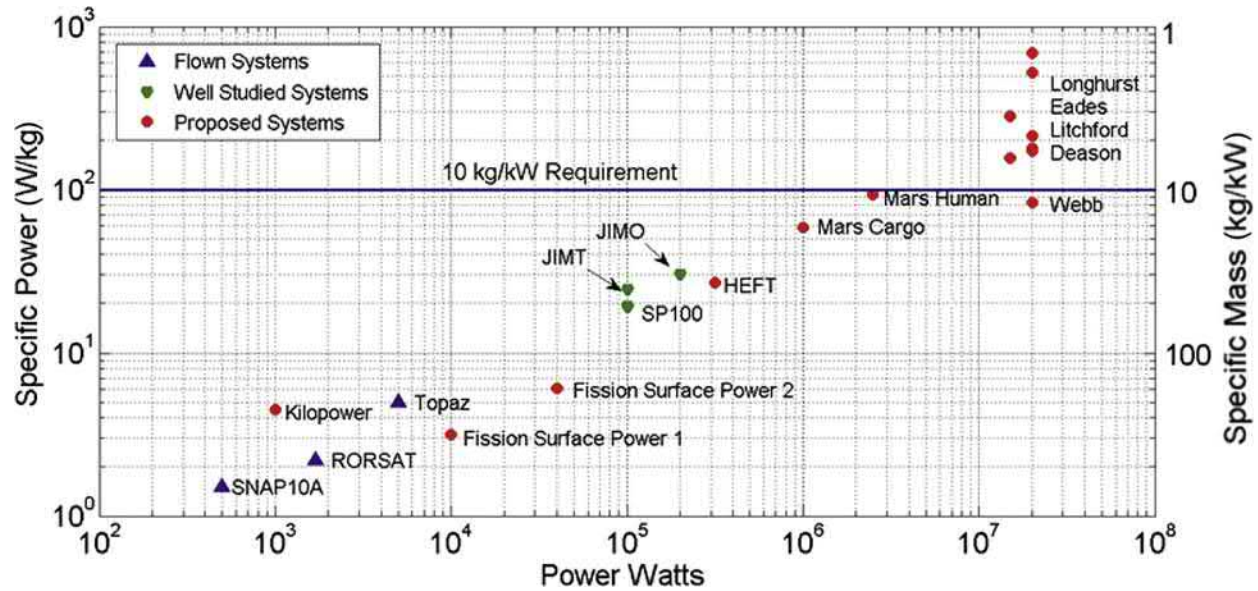


Fig. 14 Reactor power scaling (Morrison, 2014).

The blue “flown systems” identify the power systems that have been launched, the SNAP10A, TOPAZ, and BUK on board power systems previously mentioned. The green “studied systems” are systems that have been designed but not built. SP-100 and Prometheus were both heavily studied systems which had identified potential solutions to all known technical obstacles. The red “proposed systems” all fall under the 15–20 MW range. It can be clearly seen that as power increases, the specific mass of the fission power system decreases.

Brayton case study

While there are multiple technologies that can achieve a low specific mass, Brayton cycle technology seems to have the strongest case for near-term development and it is chosen as focal point for achieving a near-term 10 kg/kW_e power system. For the remainder of this section, we will analyze a case study evaluating the design of a Brayton nuclear electric propulsion system that was originally published by Morrison in 2020 (Morrison, 2020). Traditionally closed-cycle Brayton nuclear systems have been limited by temperature to minimize specific mass. There are six recent advancements in the last decade that have been enabling for low specific mass Brayton systems.

1. *Reactor fuels*: Particle fuels capable of operation at temperatures in excess of 1500 K with high burn up capability have reached a high state of maturity (Zinkle et al., 2014).
2. *High temperature turbines*: Turbine reliability has been improved and options for film cooling or ceramic blades have enabled extremely high temperature operation (Torralba, 2016).
3. *Carbon-carbon composite radiators*: Bare carbon fiber radiators with low areal specific mass, flexible construction, and high temperature heat rejection, have been tested and show promise for near-term development (Tombouliau, 2014).
4. *Compact Power Modulation Technology*: Gallium Nitride and Silicon Carbide semiconductor technology have increased efficiency and decreased mass required for power modulation (Anon, 2020).
5. *Reduced Launch Costs*: Specific mass scales very favorably to higher total power. Higher total power requires larger total mass vehicles. Recently, launch prices have reduced by approximately a factor of 10 from the early 2000s and stand to decrease even more into the 2020s based on maturing commercial technologies (Jones, 2018).
6. *Low Enrichment Feasibility*: As the government has been the main customer for space nuclear power system, highly enriched uranium (HEU) has been the focus of nuclear power systems. However, commercialization requires low enriched Uranium (LEU). Low enrichment has recently been shown feasible in the nuclear thermal rocket and this analysis will show that reactors utilizing low enriched Uranium systems can still achieve 10 kg/kW (Venneri and Kim, 2017).

A system model was developed in (Morrison, 2020) for a closed-cycle, direct, multi-stage, recuperated, Brayton cycle and was used to complete a design scoping study to evaluate specific mass. Fig. 15 goes in depth into the model and each of its components.

Each component in the model was modeled either as a design, a trend or correlation back up by literature, or for the power modulation and distribution system (PMAD) a point estimate based on a survey of estimates from other research. The specific mass of the power system a_{sys} is the system mass m_{sys} divided by system electric power P_{sys} and can be defined as the sum of each component’s specific system mass $a_{\text{component}}$.

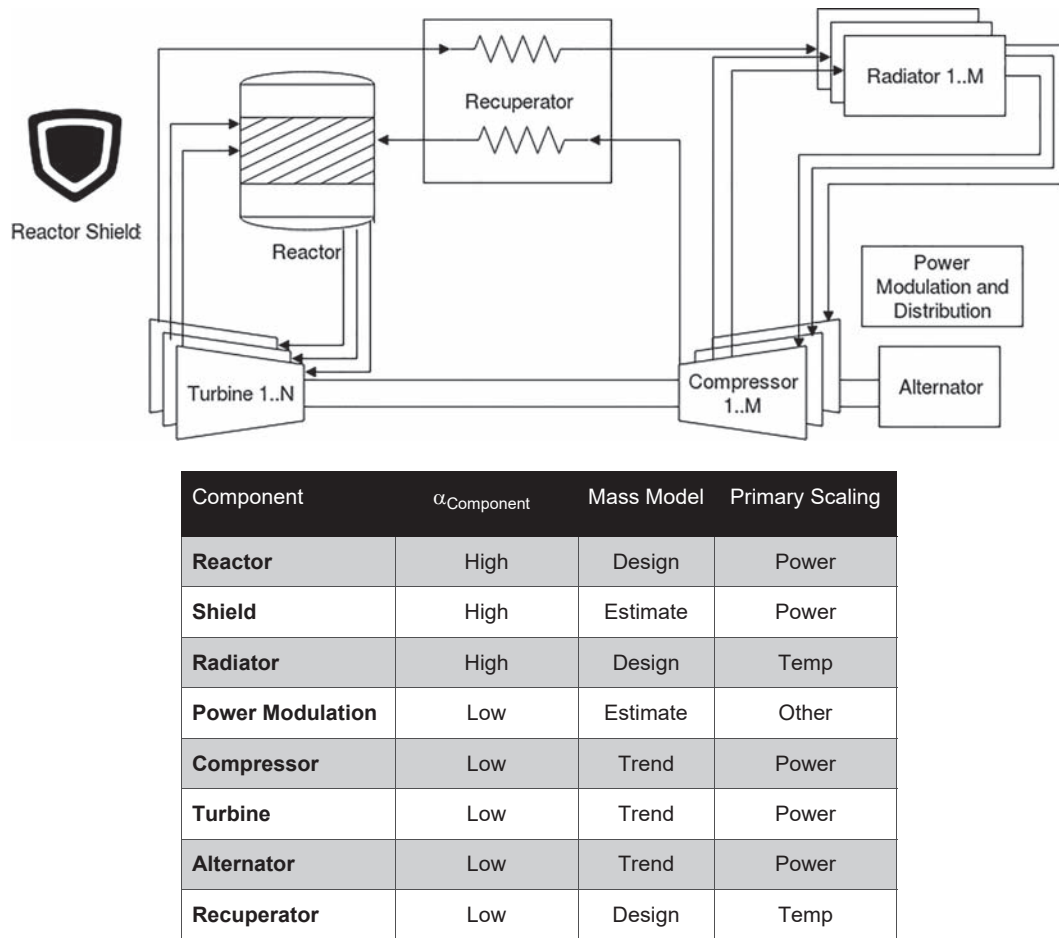


Fig. 15 Brayton power system model with multiple turbine and compressor stages (Morrison, 2020).

$$\alpha_{\text{power source}} = \frac{m_{\text{sys}}}{P_{\text{sys}}} = \sum \frac{m_{\text{component}}}{P_{\text{sys}}} = \sum \alpha_{\text{component}} \quad \text{Specific Mass Summation} \quad (17)$$

Each component in the system has a specific mass that is primarily a function of temperature or power. $\alpha_{\text{component}}(P_{\text{sys}})$ or $\alpha_{\text{component}}(T_{\text{max}})$. For example, a radiator's mass will scale linearly with heat rejection power; twice the heat rejection requirements will require twice the radiator area for the same temperature. However, radiator area scales with temperature to the fourth power; doubling the temperature will reduce the size of the radiator by a factor of 16. Increasing the temperature will reduce the component specific mass but increasing the rejection power of the component will not reduce specific mass. Another example is a radiation shield. A reactor shield scales semi-logarithmically with power; higher reactor power requires more shielding. A reactor shield however has no direct relationship to temperature. The temperature or power scaling of each of the components in the Brayton system are given in Fig. 1. Because of the intrinsic coupling of power and temperature with regards to specific power, any design must be evaluated with regards to both parameters.

$$\alpha_{\text{system}}(P_{\text{sys}}, T_{\text{hot}}) \approx \sum \alpha_{\text{component}}(P_{\text{sys}}) + \sum \alpha_{\text{component}}(T_{\text{max}}) \quad \text{Temperature and Power Specific Mass Scaling of Components} \quad (18)$$

Combining the power scaling components and the temperature scaling components yields the combined system specific mass scaling with power and temperature. Fig. 16 shows the power and temperature scaling combined into a contour plot.

For a fixed power, a lower specific mass can be attained by increasing temperatures, however it is a diminishing gain as higher power asymptotes to a minimum specific mass. For a fixed-power system it is vice versa. Higher performance cannot be achieved with higher temperature technology. For a given temperature there is an optimum power domain to minimize specific mass that exists in the "knee of the curve" in the contour plot. Low temperature technologies will be competitive at low powers and high temperature technologies will be more effective at higher temperatures. The operational temperatures for stainless steel and Inconel 718 are shown in Fig. 16 to gain a better understanding of temperature on materials requirements. Steel based materials would achieve acceptable performance up to a few hundred kilowatts. Inconel would be for

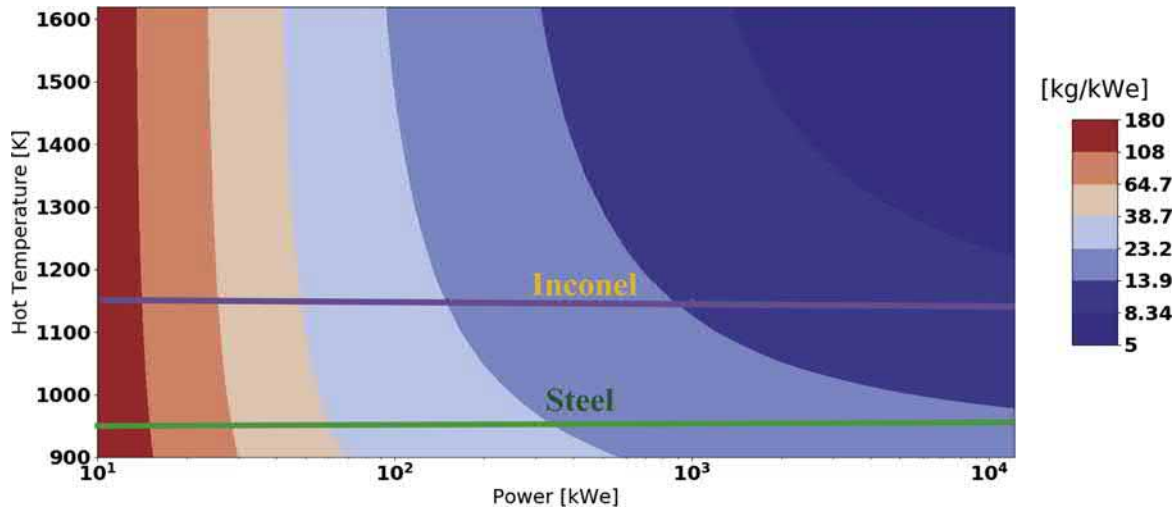


Fig. 16 Scaling specific mass with power and temperature (Morrison, 2020).

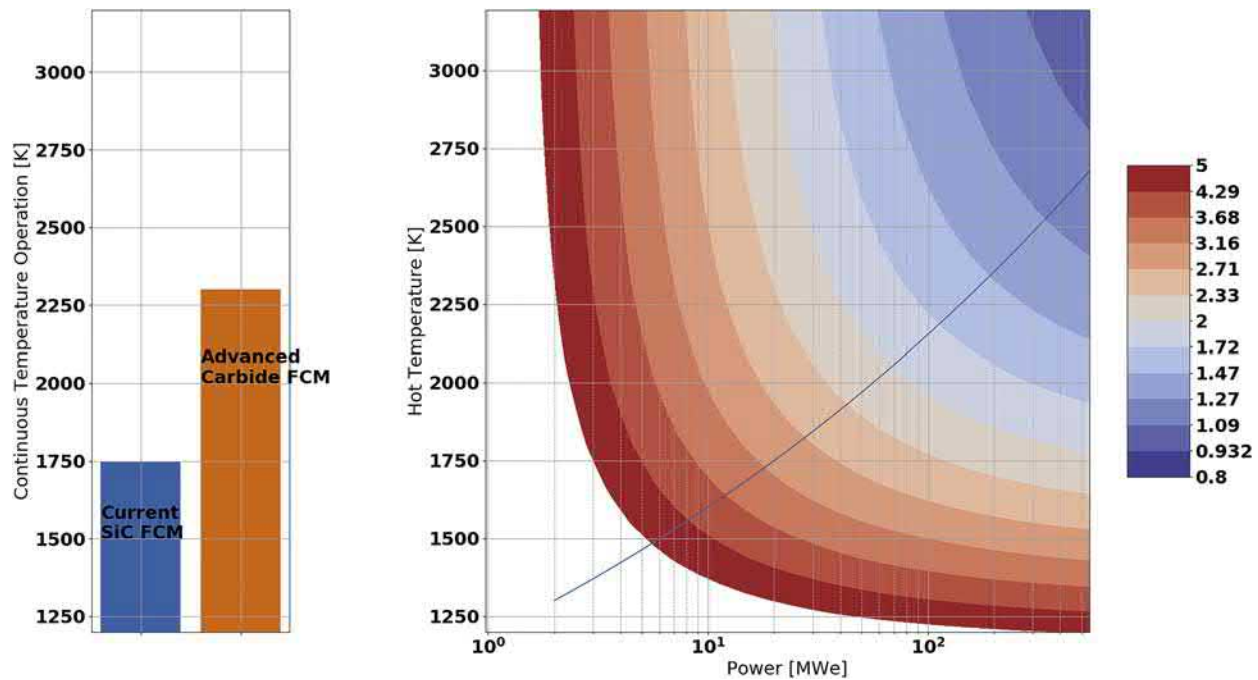


Fig. 17 Scaling of specific power at higher temperature and power. Bar graph of feasible nuclear fuel technology capable of enabling the high temperatures (Morrison, 2020).

a few megawatts. Higher power would be optimized by higher temperatures and would likely require ceramic technology, especially for the turbine and reactor.

Based on the results of the Brayton study, a Brayton power system with a power in the range of 5 MW_e and a hot cycle temperature around 1400–1500 K would be able to achieve a specific mass on the order of 10 kg/kW_e. Such a power system would have a total mass around 50 tons and be launchable on a Falcon Heavy or New Glen class rocket.

High temperature materials are key for ultra-low specific mass. The extreme temperatures shown in Fig. 17 can be achieved by using the advanced carbides (Torralba, 2016).

For advanced nuclear systems ceramics will be an important technology for both the reactor and power conversion systems. Currently terrestrial turbomachinery is working on ceramic blades that can tolerate temperature well over 1800 K but typically require significant maintenance, use film cooling, and have much greater pressure ratios. These more advanced systems could achieve a specific mass below 5 kg/kW_e and potentially as low as 1 kg/kW_e. These more advanced systems could enable fast travel to nearly any destination in the solar system.

Conclusion

Nuclear electric propulsion can be developed in the near future to achieve a 10 kg/kW_e specific mass target. These systems can be iteratively improved in later generations. Increasing power and temperature capability can produce more capable systems expand human presence throughout the solar system and enable a rapid, sustainable travel with efficiency that eliminates the need for significant infrastructure required by chemical or nuclear thermal propulsion. Nuclear electric systems will be able to connect all the planets in the Solar System and facilitate large scale human and cargo transportation. The power systems developed for electric propulsion are synergistic and can be adapted for surface power and support outposts, industrial production, settlements, food production, and much more. Even if chemical propulsion is a predominant propulsion technology, nuclear power sources will be key for exploration and power production. I encourage the bright minds reading this to further develop the systems needed to make nuclear electric propulsion a reality.

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Advanced Propulsion: Pulsed, Fusion, Antimatter and Breakthrough Physics

Steven D. Howe, Howe Industries LLC, Scottsdale, AZ, United States

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Introduction

The distances between destinations in space are significant. The galaxy and the solar system are mostly empty space. Although travel on the Earth's surface is measured in miles, travel between planets is measured in billions of miles. Typically, distances to targets within the solar system are more often discussed in astronomical units (AU), i.e. the distance from the Earth to the Sun. Since 2012, 1 AU has been defined as exactly 149,597,870,700 m, or about 150 million kilometers or 93,225,000 miles. For examples, Saturn is 9.6 AU from the Sun and Pluto ranges between 30 and 49 AU. However, the nearest star, Proxima Centauri b, is 270,000 AU from our Sun, 24.8 trillion miles. Travel between stars is measured in light years (LY), the distance light travels in 1 year or 5,868,000,000,000 miles. To travel these distances in reasonable time will require significantly higher velocities than have been achieved by human missions so far.

Historically, in-space propulsion systems have relied on one simple principle—expel mass in one direction to push your ship in the opposite direction. The higher the speed of the ejected mass means that less mass has to be ejected for a given change in momentum of the ship. Typically, the speed of the ejecta is related to temperature of the gas. Chemical systems use combustion to provide the high temperature. Nuclear rockets use fission heating. Electric thrusters use electro-magnetic fields to accelerate charged gases. The higher the exhaust velocity, the higher delta-V, or change in velocity, can be acquired by the ship.

Most advanced propulsion concepts simply involve methods to increase the velocity of the expelled material in order to reduce the amount of the exhaust mass. If the energy for the propulsion is contained on-board the ship, then very high specific energy, joules per kilogram, sources are needed. The first part of this article will examine concepts which have been proposed that have very high specific energy and produce very high specific impulse. The concepts include fusion (fission-fusion, antimatter driven fusion, and fusion only), and antimatter (direct and sails).

Several alternative ideas have been proposed in which the energy for the mission is not on-board the ship. These concepts have the advantage of not having to have high specific energy and can be massive. However, the concepts must transport the energy over the distance to the ship to provide thrust at the ship. The second part will examine concepts where the energy source is roughly stationary (on Earth or in orbit) and beams the power to the ship via particles, laser, or microwave.

Over the years, the entire idea of momentum transfer in normal space has been called into question. Must a ship progress from two points in space through every point in between? Quantum mechanics indicates that particles can be entangled at a distance for

example. Is it possible that space can be altered or “short circuited” to provide rapid transport between locations? Some concepts have been proposed that invoke trans-dimensional rifts (e.g. the Stargate), mass to energy conversion and transport (e.g. teleportation), space warp (e.g. the Alcubierre effect), reactionless drive (e.g. EM drive). All of these concepts rely on physics which has not been proven. The third part of this article will provide brief summaries of the issues involved in these concepts.

Part 1: On board energy sources

The Voyager II spacecraft launched in 1977 has just reached the 115 AU distance in 2019. The New Horizons spacecraft, launched in 2005, had the highest average transit speed of any spacecraft and took 10 years to travel the 35 AU to Pluto. The distances traveled are modest on the scale of interstellar travel. The velocities obtained are wholly insufficient for regular transport throughout the solar system.

Traveling to destinations in space inherently requires high velocities due to the tremendous distances that have to be traversed. High velocities translate to very high kinetic energy per kilogram of ship mass. That energy has to come from somewhere. If it is contained on board the ship, then the ship at the beginning of the mission must have very high potential energy per kilogram of ship mass to be converted into kinetic energy. Thus, energy sources with high specific energy are needed for advanced concepts.

While fission power and solar power may allow missions between the planets inside the solar system, more energetic reactions will be required to go to the outer planets or beyond. To get a spaceship up to the very high speeds requires energy, or rather high specific energy, i.e. energy per unit mass of the ship.

Table 1 shows some example destinations, their distance from the Earth, and the average velocity necessary to reach them in the designated time. The classical kinetic energy per kg of ship mass is also shown.

In order to reach the specific kinetic energies in **Table 1**, the potential energy of the reactions must have roughly the same magnitude. **Table 2** shows a comparison of specific potential energy for various possible energy sources. Typically, chemical reactions have 10s of MJ kg⁻¹ with the maximum of 142 MJ kg⁻¹ for liquid hydrogen and liquid oxygen, LH2/LOX. Fission based systems have around 70,000,000 MJ kg⁻¹, i.e. over 500,000 times the energy content per unit mass of chemical combustion. Fusion of a proton and Boron-11 (pB11) is even higher at 750,000,000 MJ kg⁻¹.

Comparing the specific energies between the tables shows that chemical combustion is wholly insufficient for transport to the outer solar system. Fission systems with realistic burn-up fractions have just sufficient potential energy to reach beyond the solar system but not much farther. For missions well beyond the solar system or to the stars will require fusion or antimatter.

Most propulsion systems execute impulsive burns, i.e. the thrust is generated for short interval and then the ship coasts until reaching the destination. During the burn, for a fixed exhaust velocity of the propellant and a constant power output of the system, a ship will increasingly accelerate as the propellant is expelled and the mass of the ship decreases. The final velocity achieved by impulsive propulsion systems is clearly summarized in the well-known rocket equation:

$$M_{\text{final}}/M_{\text{initial}} = \exp(-\Delta V/V_{\text{exhaust}})$$

where M_{final} is the final mass of the ship after all propellant is used, M_{initial} is the initial mass of the ship at the start of the mission and equals dry mass plus propellant mass, ΔV is the change in velocity, i.e. delta-V, required, and V_{exhaust} is the exhaust velocity of the propulsion system. The exhaust velocity is usually expressed as specific impulse, i.e. I_{sp} , which equals the exhaust velocity divided by gee, i.e. 9.8 m s⁻². ΔV is determined by the desired mission time. V_{exhaust} is determined by the technology being used.

Table 1 Velocities and kinetic energies required for missions to distant destinations.

Mission	average velocity (km s ⁻¹)	specific energy (J kg ⁻¹)
Kuiper Belt (250 AU in 10 years)	117	6.8e09
Oort Cloud (10,000 AU in 40 years)	1200	7.2e11
Proxima Centauri (270,000 AU in 40 years)	30,000	4.5e14

Table 2 Specific energy contained in selected reactions.

Reaction	Specific energy (J kg ⁻¹)
Chemical combustion (LOX + LH2)	1.4e08
Fission (100% reacting) (realistic burnup—5%)	7.1e13
	3.5e12
Fusion (100% reacting)	7.5e14
Antimatter	9e16

For example, in order to send the probe to Alpha Centauri within a 40 year interval, the average velocity must be greater than 11% the speed of light, c , or roughly $3.3 \times 10^7 \text{ m s}^{-1}$. This is the mission total ΔV . For an impulsive burn mission where the ΔV is acquired quickly relative to the total mission time, one can derive a formula for a minimum total-energy requirement. The minimum propulsive energy required to complete the mission is when the exhaust velocity of the ship is 60% of the ΔV , i.e. $V_{\text{exhaust}} = 0.6 \Delta V$. From the rocket equation, this means that the mass fraction, $M_{\text{final}}/M_{\text{initial}}$, is around 0.2. In this case, then, V_{exhaust} needs to equal $1.98 \times 10^7 \text{ m s}^{-1}$. For a hydrogen atom, this is an energy of 2 MeV. Clearly, at these energies, the exhaust is a high temperature plasma.

Producing MeV scale plasmas with sufficient mass flow for appreciable thrust is the crux of the problem. Such energies are usually relegated to particle accelerators with mass flows of micro-grams per second. Potentially, plasmas created by fusion or antimatter can reach these levels. However, these energies are well above any material vaporization temperature so that use of physical nozzles is unrealistic. Directing the energetic mass into thrust becomes a major problem.

Another major problem for advanced propulsion concepts is power. To cover large distances, the ship must have sufficient thrust to get up to speed. The thrust of the ship, in N, is equal to the mass flow, in kilogram per second times the exhaust velocity, in meter per second, but the power required to provide that thrust is proportional to the square of the exhaust velocity.

High Isp systems, therefore, require very high power. For example, a system with the exhaust velocity of $1.98 \times 10^7 \text{ m s}^{-1}$ and a mass flow of 1 g s^{-1} , has a thrust of 19,800 N and a steady state power level of 196 GW. One of the fundamental problems with on-board energy sources is the huge power levels involved—power levels well outside current operational experience. Consequently, pulsed power systems are often considered which allow very high power levels while producing the same overall energy to the ship.

Conversion to propulsion

As stated earlier, high Isp entails the handling of high temperature plasmas. The temperatures of these plasmas are well above the melting point of any material known. Consequently, the plasmas are often generated in confining magnetic fields to keep the hot materials from the walls. To produce thrust, however, the plasma must push against the ship and be ejected from the confining volume. Several studies have examined the formation of pulsed plasma jets using specially configured magnetic fields. The basis for these concepts is that a fully ionized, neutral plasma will compress magnetic field lines and be reflected away from high field density.

Typically, these studies have relied on large electromagnets (EM) to create the field. The mass of the EM magnets can be reduced by using superconducting (SC) materials to flow the electrical current instead of copper coils. However, superconducting systems require very low temperature and would consume coolant at a fixed rate. Also, SC materials are subject to quenching, i.e. reversion to non-superconducting state, under sufficient irradiation. At velocities needed for interstellar transport, the diffuse material between solar systems could impact the ship at sufficient energies to cause quenching. In addition, for the interstellar mission, the power supplies to drive electromagnets may be too heavy.

Pulsed nozzles

The high power requirements needed for deep space missions does not necessarily require steady state operations. Pulsed operations offer the ability to create very high power with fractional duty cycles. Between pulses, the system has the chance to remove heat so that the average temperature of components is kept lower.

Often, magnetic fields are assumed to be configured to “catch” the expanding plasma and reflect it rearward of the ship to provide the thrust. However, for plasma with sufficient temperature, the photonic emission becomes an issue. The demands on the surfaces that are in line-of-sight to the energy release become difficult. Opaque materials being exposed to pulses of high energy illumination will ablate. This causes loss of material and erodes the surface irregularly. Alternatively, such surfaces can be coated with a thick layer of transparent materials such as sapphire or fused silica which have photon mean free paths of several centimeters and cooled with gas flowing in channels through the material. This allows the surfaces to survive the intense illumination of pulsed systems.

To summarize, on board energy sources suffer from (1) limitations of specific energy of the reactants, (2) need for very high power and very high temperatures of the exhaust, (3) need to keep energetic exhaust from material boundaries, and (4) waste heat removal. Concepts have been proposed that may meet these constraints for a variety of missions (Frisbee, 2003). For interstellar missions, though, only two energy sources meet these criteria for on board propulsion- fusion and antimatter.

Fusion

Fusion occurs between light elements such as hydrogen and helium nuclei at moderate energies due to the relatively low coulomb repulsion between the nuclei. Typically, the atoms must have temperatures in the several 10s of keV to have sufficient impact velocities to overcome the coulomb barrier and merge the two nuclei. However, reaching these temperatures is difficult and containing the high temperature atoms long enough to allow fusion is problematic. To date, no system has operated where more energy has been produced than was required to make the fusion occur, i.e. breakeven.

Because of the high temperatures needed for fusion, the research over the past 60 years has tended to focus on using deuterium-tritium (DT) fuels. The DT reaction cross section peaks around 50 keV, i.e. 550 million degrees Kelvin, at $5 \times 10^{-28} \text{ m}^2$ which is the lowest temperature of the many possible fusion reactions (Santarius et al., 2005).

However, DT fusion produces a 14 MeV neutron in each event. Neutrons require significant shielding and are, essentially, an energy loss mechanism. In short, only around 3.5 MeV per reaction is available for conversion to power or propulsion. All fusion reactions except pB11 and $\text{He}^3 + \text{He}^3$ create neutrons. Consequently, because of the low He^3 cross section, the goal for many researchers has been to find a method to make pB11 fusion a possibility. As the reaction products for pB11 fusion are three, charged, alpha particles, very little shielding is required, a compact system can be built, and the system should be very affordable. The problem is that p-B11 fusion has a peak cross section around 600 keV, 6.6×10^{-9} kelvin, of $1.2 \times 10^{-28} \text{ m}^2$ which is very difficult to achieve in a dense plasma. Thus, the pursuit of this fusion system has languished.

Current fusion systems

Two types of fusion containment have been pursued—magnetic fusion (MF) and inertial confinement fusion (ICF). Magnetic confinement tries to trap the hot plasma in a magnetic bottle and hold it for long periods. Inertial confinement impacts beams of photons or ions onto a target and compresses the material, raising the temperature to fusion conditions. Unfortunately both methods rely on the plasmas reaching thermal equilibrium.

Because of the low density associated with fusion systems, the physical systems are often very large. The International Thermonuclear Experimental Reactor (ITER) is a large, expensive, international effort that hopes to reach break-even within the next decade (Rebut, 1995). The National Ignition Facility (NIF) at the Lawrence Livermore National Laboratory in the US (Miller et al., 2004) is also a large, expensive facility that produces 196 high power laser beams to focus onto a small target to reach the required temperatures.

Fusion propulsion concepts

Several concepts (Santarius and Logan, 1998) have been investigated that use different configurations of magnetic fields to contain the hot plasma long enough for fusion to occur. The primary issue is that net energy must be produced to make the concept effective. Otherwise, if the energy produced is less than the energy required to produce the reactions, the fusion system is simply a complex electric thruster which is more massive than other thruster concepts. Examples of magnetic fusion propulsion concepts that have been investigated are (1) the Field Reversed Configuration (FRC) (Slough, 2001) which uses rotating magnetic fields to create an spherical region in which a torus of plasma is held, and (2) the Dense Plasma Focus (DPF) (Thomas et al., 2005) which uses a large electrical current passing through a jet of gas to ionize and then compress the gas.

An example of Inertial Confinement Fusion (ICF) (Orth, 1998) propulsion is the Vehicle for Interplanetary Space Transport Applications (VISTA) study which uses high power lasers to compress a deuterium-tritium target. VISTA entailed a large ship with a mass of 6000 t to contain the power supplies for the inefficient lasers to produce 17.5 GW of fusion power.

Pulsed fission-fusion—ORION

In the 1960s, Freeman Dyson and others proposed a concept dubbed ORION (Dyson, 1965). The concept would utilize the very high specific power of a nuclear warhead to propel a space ship at high speed. The basic idea required a very heavy “pusher plate” to intercept the explosion coupled to huge shock absorbers to spread out the momentum pulse to the ship. The primary design trades involved (1) the yield of the device and (2) the stand-off distance from the pusher plate. If the device is detonated far enough from the plate to reduce ablation, then the fraction of the explosion that is subtended is small. If the explosion is close so a larger fraction of the blast is accepted, the plate ablates and the Isp is reduced. A major hurdle for the concept was the need for around 10,000 shots to move the massive ship. Placing this many small nuclear devices in orbit appeared problematic from an international agreement perspective. Although the concept was impractical from a variety of political reasons, the basic idea of using pulses to propel a ship is sound.

Pulsed fission-fusion—Puff

The ORION concept offered the possibility of very high power, high Isp, and high thrust. This combination is very hard to produce and very few concepts offer the performance. Unfortunately, the requirement to use compact nuclear devices precluded the development of the concept. An alternative method to produce the explosion, however, may allow ORION type performance with smaller explosions.

The Pulsed Fission Fusion (PUFF) project is a NASA Innovative and Advanced Concepts (NIAC) Phase II that is designing an advanced in-space propulsion concept (Adams et al., 2014). The concept utilizes a very high electrical current (10s of megamps) surrounding a metallic target comprised of fissile material. The current creates an intense magnetic field which compresses the fissile material to reach critical mass. A pulse of neutrons injected into the target will initiate a fission chain reaction which will exponentially grow over a period of 100 s of nanoseconds. The fissile material will reach temperatures in the keV range which will initiate fusion in a central core. The fusion reactions will produce 14 MeV neutrons which will boost the fission process to higher rates. The result is a pulse of plasma with very high temperature producing an impulse of very high power for propulsion with high specific impulse (Isp).

Generation of the high current needed to initiate the pulse requires massive electrical storage systems. The system mass could dominate the ship mass and, ultimately, the effectiveness of the concept. Thus, designing a target that requires the smallest current possible while achieving criticality is a major objective. However, lowering the driving current will push the mass and size of the target higher which will dramatically increase the cost of operating the PUFF system. Thus, an optimum target design may exist that keeps the ship mass low, the driving current low, and the cost of the targets mass low.

Antimatter

Antimatter has the highest specific energy, joules per kilogram, of any material or process known to mankind. Antimatter is the quantum mechanical mirror image of normal matter, i.e. the mass of the antiproton is identical to the mass of the proton but the charge and spin are reversed. Antimatter exists in nature and can be produced in larger quantities via particle accelerators.

Anti-electrons, or positrons, are produced in nature via radioactive decay of isotopes such as sodium-22. They can also be produced by accelerating electrons up to high speed and impinging the beam onto a metal target. While large quantities of positrons can be produced for relatively modest energy expenditure, they cannot be stored with high density because of the space-charge repulsion, i.e. they repel each other so limited amounts can be stored in a given volume.

Antiprotons were first postulated by Dirac in the 1920s (Pais et al., 2005) and discovered in 1955 by Chamberlain et al. (1955) who won the Nobel Prize for the discovery. Antiprotons are not made in our natural Earth environment (but they are seen in Galactic Cosmic Rays) but must be produced in particle accelerators. Both CERN in Europe and the Fermi National Accelerator Laboratory, FNAL or Fermilab, have the ability to produce antiprotons. Fermilab discontinued production a few years ago but the antiproton production line is still intact and can be reinstated. CERN still produces antiprotons at an intensity 1000 times less than the Fermilab rate and decelerates the particles down to low energies (Maury et al., 2014) that can be stored in Penning Traps or used to make antihydrogen. The CERN low energy beam could be used to demonstrate the momentum transfer from fissioning uranium induced by antiprotons for the antimatter sail concept (see below).

Antihydrogen is the combination of an antiproton and a positron to make a hydrogen atom. This is a neutral particle so does not suffer from the space charge repulsion. This allows for the possibility of storing large amounts of antimatter in very small volumes. Currently, antiprotons have been stored in a Penning Trap and combined with positrons to make antihydrogen in the ATHENA experiment at CERN (Rotondi et al., 2005). Conceptually, these neutral antihydrogen atoms can be confined in an Ioffe Trap (Gabrielse et al., 2007) that possesses a high magnetic field gradient. If confined long enough, the neutral atoms should combine to form diatomic antihydrogen molecules. The molecules, if confined in an ultracold cavity will condense to form nano-particles of antihydrogen ice. Conceptually, if the ice particles are exposed to a few positrons, the positrons will annihilate a few electrons leaving the ice particle slightly charged. The charged microparticles can then be manipulated using electric and magnetic fields. Thus, at least conceptually, antiprotons may be able to be stored for decades.

Fermilab produced roughly 1 ng of antiprotons per year at its peak (Thompson, 1994). However, this number is misleading. Fermilab's mission was to capture antiprotons and accelerate them up to 99% the speed of light in order to collide with normal protons to generate exotic particles such as hyperons or the Higgs Boson. To do this, they "accepted" those antiprotons produced in a very narrow energy band so they could be guided by electric and magnetic fields and reaccelerated. Roughly 1 particle in 1 million was accepted. Thus, efficient antiproton accumulation was not the goal.

Consequently, with modest modifications to the existing line, the amount of antiprotons that can be captured at Fermilab could be increased by a factor of 1000–10,000 to raise production to the micro-gram per year level. This will allow the "capture methodology" to be developed that can be used in future high production accelerators.

The next level of production would be to build a new accelerator with a higher beam current. This would allow higher production levels. For example, the LANSCE facility at the Los Alamos National Laboratory (Garnett, 2011) can run a beam current of 1 milli-amp, i.e. 6×10^{15} protons per second. However, it only has an energy of 800 million electron volts (800 MeV). Efficient antiproton production needs an energy of around 50 billion electron volts (50 GeV). A new machine that could produce 1–10 mA at 50 GeV could produce around 0.06 g per year. If the accumulator technology or the beam current could be raised by a factor of 15, then a gram per year might be possible. The accumulator technology can be developed in the near term using the Fermilab facility. The beam current technology may be possible using the superconducting magnet systems developed for the Large Hadron Collider (LHC) in Europe.

Thus, conceptually, a facility that could produce 1 g year^{-1} of antiprotons may be achievable within a 10–20 year period. In parallel, though, the storage of antihydrogen needs to be developed to utilize the material produced. In addition, such a facility will need to be located on a very remote site.

Conversion to propulsion

Antiprotons react with protons and neutrons in normal matter in a process called annihilation. Both protons and antiprotons are composed of quarks. When the two particles annihilate, these quarks are merged and recombine into charged and neutral pions (pions), particles composed of two quarks, with average energies of 250 MeV. The neutral pions decay in roughly a femtosecond (10^{-15} s) to two energetic gamma rays with an average energy of 200 MeV. The main issue then for converting antimatter annihilation into thrust is coupling the very energetic annihilation products into normal matter. The ranges of the pions and gamma rays is very large in matter, i.e. they do not deposit their energy in small volumes. Thus, heating materials requires large numbers of annihilations.

Several concepts have been proposed to convert antimatter annihilation into thrust. These involved (1) depositing the reaction products into a surrounding block of tungsten which would be heated to 3000 K and perform as a thermal rocket with an Isp of 1000 s, (2) surrounding the annihilation region with a solenoidal magnetic field containing a gas which would be heated to plasma temperatures and expelled with an Isp of several 1000 s, and (3) expelling the charged pions directly using a magnetic field with an

Isp of several million seconds but very little thrust. All of these concepts either produced lower Isp or very low thrust. They also entailed heavy massive systems.

Interstellar mission

The goal of sending a human-built probe to the nearest star, Proxima Centauri, would be the most ambitious effort ever attempted. The difficulty of this goal exceeds the development of the atomic bomb, construction of the pyramids, or the Apollo mission to the moon. This is a very hard problem. The Proxima Centauri system is 4.2 light years away, i.e. 24.8 trillion miles. In order to get there in a reasonable time frame, e.g. an individual's career of 40 years, the average speed of the ship must be 0.11 times the speed of light. At this speed, every kilogram of mass of the ship will have 544 trillion joules of kinetic energy which is equivalent to the aircraft carrier Nimitz traveling at 7500 MPH or Mach 11.

Previous studies have assumed very large ships to contain the power supplies, magnets, and systems necessary for antimatter storage. Thus, the amount of antimatter required to acquire the 0.11 c velocity was tremendous (Schmidt et al., 2000). According to the study, many kilograms of antimatter would be needed. This amount of material could not be produced by any means known as yet, would cost more than the US GDP for decades to make, and would represent an extreme hazard for explosion. In short, an intractable approach.

In order to reduce the amount of antimatter required for a deep space mission, two approaches have been investigated—antimatter induced fusion and the antimatter sail. The idea of inducing fusion is attractive in that it uses the antimatter to “enable” a fusion process with net gain. It also produces more energy from fusion than from the antimatter annihilation. Thus, a small amount of antimatter can “light” the fusion reactions. However, large magnets, power supplies, and complex structures are still required. An alternative approach is to drastically reduce the mass of the ship and the complexity of the propulsion system. The antimatter sail concept relies on the use of antiprotons to induce fission in a thin sail. The fission products exiting the surface have an Isp of 1.6×10^6 s, sufficient for an interstellar mission. The main issue for this concept is very high density storage of the antimatter and the ability to handle/manipulate it.

Antimatter induced fusion

Two approaches have looked at using antimatter to initiate fusion. The Antimatter Initiated Microfusion (AIMstar) effort at Penn State University sought to use inertial confinement methods to implode a pellet with fusion fuel using antiproton beams (Lewis et al., 1990, 1999). The pellets were compressed prior to antiproton injection using intense lithium ion beams. Consequently, significant structure was required for the ion beams and power sources. The micro-explosion was induced behind the ship and caught by a magnetic field which expelled the plasma. Alternatively, the Magnetically Insulated Inertial Confinement Fusion (MICF) concept (Kammash, 1998), also relied on laser imploding a spherical target but incorporated the creation of intense magnetic fields in the implosion to reduce energy loss by the plasma, i.e. insulate the plasma so higher temperatures could be reached. After implosion, a small amount, roughly a nanogram, of antimatter was injected into the compressed plasma to heat the plasma and produce significantly more fusion energy.

Antimatter sail

In the NASA Institute of Advanced Concepts (NIAC) study in 2003 (Howe and Jackson, 2005) Hbar Technologies showed that an antimatter driven sail could achieve the velocity required for deep space missions. The basis of the concept was that low energy antiprotons have two unique features: (1) they induce fission in uranium with 100% efficiency and (2) they induce fission on the surface of a uranium sheet allowing the fission products to escape at high speed, i.e. 1.38×10^7 m s⁻¹ or over 1 million seconds of Isp. The NIAC showed that an antimatter driven sail could accomplish the Kuiper Belt mission, i.e. 250 AU in 10 years, for 30 mg of antimatter and the Proxima Centauri b mission in 40 years with 17 g of antimatter. While this amount of antimatter may be possible in the next few decades, creation of the accelerator facility to produce this much antimatter would be expensive and take a decade or more.

Recent work has examined the concept of harvesting antimatter in the solar system (Jackson, 2006). Antimatter is present in Galactic Cosmic Rays entering the solar system. The concept looked at using an orbiting ship to collect the material. Such a concept could reduce the cost of antimatter production and, perhaps, enable propulsion concepts to be implemented.

Part 2: Beamed propulsion

The requirement to carry the source of energy on board the ship puts a huge constraint on the options available. If the energy source can be displaced from the ship, then lighter, less complex ships may be possible. Several concepts have been proposed where the energy source is roughly stationary (on Earth or in Earth orbit) and beams the power to the ship via particles, laser, or microwave. The basic advantage of these concepts is simple—the power generation systems can be massive. The primary issue with all of these concepts is the ability to maintain the energy deposition into the ship as it accelerates away and the distance increases. Thus, the targeting of the beams over large distances and the focusing of the beams onto the ship become the big obstacles. In order to circumvent these issues, very high power must be delivered to the ship so that it can accelerate up to mission speed within a short distance before the beams grow larger than the receiver.

For photonic beam transport, the rate the beam spreads as it propagates, i.e. the divergence, becomes a major issue. The beam divergence (beam spot radius to distance ratio) is proportional to the wavelength of the photons. Typically, beam divergence for visible wavelength lasers is a 0.5 mrad. Thus, for a 10 m^2 sail intercepting the beam in space, i.e. radius is 1.78 m, the beam spot starts to exceed the sail diameter at 3600 m distance. Thrust starts to decrease at this point. Consequently, photonic beamed energy concepts tend to rely on phased arrays of multiple transmitters. By spreading out the transmitter over a larger area and focusing them at a common point in the far field, the beam can be put on target over longer distances and greater speeds achieved. The power per unit area at the ship decreases rapidly with distance. Thus, power levels must be high and coupling the high power into thrust is critical. The divergence for microwave beams is large but the efficiency of generation, i.e. the power density, is good. The divergence for optical frequency photons is lower but the efficiency of generation is low so the power plants to generate sufficient power density at the ship are more massive.

For particle beams such as protons, the divergence of the beam is a function of the “thermal” temperature of the particles. Particle beams are made using electromagnetic accelerators. Facilities on Earth such as Fermilab, CERN and the LHC have demonstrated that bunches of protons can be accelerated up to very high speeds, very close to c . However, the particles in the bunch have a thermal temperature independent of the speed of the bunch. The bunch will also expand due to Coulombic repulsion once the bunch leaves the accelerator.

During the Strategic Defense Initiative (SDI) (Drell et al., 1985) in the 1980s, substantial research was performed at the Los Alamos National Laboratory (LANL) to produce intense neutral particle beams to destroy nuclear warheads coming in from the Soviet Union. These concepts used accelerators in orbit to produce high intensity beams of protons that were neutralized, i.e. turned into neutral hydrogen atoms, at the exit of the accelerator via passage through thin films. This technique would produce a beam of particles that would not expand due to Coulombic repulsion.

While in the accelerator environment, the particle bunch can be “cooled” using electron cooling (Nagaitsev et al., 2006) or stochastic cooling (van der Meer, 1985). This allows the bunch to hold together over greater distances. The material required to intercept and stop the beam of high energy particles cannot be a thin foil as in photonic schemes but must be of sufficient thickness, usually several millimeters. Thus, the mass of the system gets big if large diameters are needed. The typical divergence for a neutral particle beam is essentially the ratio of the particle bunch thermal temperature to the bunch velocity. Thus, for temperatures below 1 eV, the divergence will be well below 1×10^{-6} . Therefore, at a distance of 1 million kilometer, the beam spot would be 1 km in diameter. The primary issues for particle beam propulsion are (1) coupling the incident particle bunch into thrust after impact with the ship and (2) targeting the beam to impact the ship over large distances.

Beamed energy propulsion concepts

Microwaves

Several researchers have examined the use of microwave beams to power the spacecraft. One concept claims high DC to DC efficiency of conversion, 54%, has been experimentally measured (Brown and Eves, 1992) and has shown applicability to missions in the near Earth space.

Another significant effort has looked at millimeter wave technology instead of using microwaves with greater than 10 GHz frequency. The longer wavelength allows for transmission through the atmosphere. In addition, roughly CW operation with mega-joule pulses are claimed to be available currently (Benford and Dickinson, 1995).

High Power Microwave (HPM) sails have also been examined for deep space missions (Benford, 2008). Experiments have demonstrated that stable flight can be achieved using conical sails. Proposed missions include mapping the solar system, exploring the Kuiper Belt, and eventually an interstellar flyby.

Laser beam propulsion

High powered lasers have been proposed for decades to push sails up to high speeds (Forward, 1989). Early concepts relied on extremely large sails with large optical lenses in space to keep the beam focused onto the sail over large distances. This was primarily due to the limited laser power level and the need for longer acceleration times. As power levels have increased, higher power designs have occurred (Kare, 2001).

In 2016, Yuri Milner formed the Breakthrough Foundation and created the Starshot project (Atwater et al., 2018). The goal of the project was to utilize a phased array of very high power lasers, roughly 1 million, on the Earth’s surface to project an intense laser beam to a sail with an area of 10 m^2 and accelerate it up to 20% the speed of light. The mission was to reach Proxima Centauri b in 20 years. The total laser power would be around 70 GW and operate continuously for 1000 s. The sail would experience roughly a 6000 gee acceleration. The payload would be 1 g and the sail would have a mass of around 1 g. Thus, materials issues are a dominant factor in the concept.

The major issues associated with this project are (1) the laser power level, (2) the thermal balance of the sail, (3) the reflected beam destination, and (4) mission flyby time. Current continuous lasers have demonstrated a power level (unclassified) of around 30 kW. The lasers for Starshot need to be up to 1 MW. The thermal balance stems from the amount of the laser power absorbed by the sail and the uniformity of the laser beam spot, i.e. hot spots will evaporate the ultra-thin material. The reflected issue is that 70 GW of laser beam are hitting the sail and “reflected” back. The sail is initially close to Earth. The potential environmental impact of the reflected beam onto the Earth or the passage of the beam through the atmosphere must be closely evaluated. Finally, at 20% c ,

the ship will sail through the Proxima system in a few hours limiting the amount of data that could be collected and transmitted back to Earth.

Particle beams

Particle beam propulsion offers the advantage of large momentum transfer in short time due to the mass of the beam. If the beam is composed of charged particles, then a magnetic field can be employed to reflect the beam so the sail material can be thinner. However, the range of the beam is reduced so accelerations must be higher and targeting the beam through the Earth's and Solar magnetic fields will be difficult. If the beam is composed of neutral particles, the range of acceleration is larger but the sail material must be more massive. Several studies have examined the feasibility of using particle beams for an interstellar mission ([Landis, 2001, 2004](#), [Gilster, 2005](#)).

Particle beams have an advantage of producing higher momentum transfer more efficiently than laser beams so the ground based power may be less. However, the coupling to thrust and the shorter range make the concept difficult.

Part 3: New physics

Over the years, some concepts have been proposed that invoke trans-dimensional rifts (e.g. the Stargate), mass to energy conversion and transport (e.g. teleportation), space warp (e.g. the Alcubierre effect), reactionless drive (e.g. EM drive). All of these concepts rely on physics which has not been proven. The third part of this article will provide brief summaries of the issues involved in these concepts.

Alcubierre drive

In 1994, Miguel Alcubierre published an article ([Alcubierre, 1994](#)) detailing a mathematical description of what has been termed "the warp drive" ([White and Davis, 2006](#)). In the article, he showed that if a region of negative energy can be created in space, then an asymmetry can be created where space is compressed ahead of the spacecraft. The result is that extremely fast transit is possible through space. The concept is similar in description to that posited in the TV show *Star Trek* as the basis for their warp drive. To date, no experimental evidence for the Alcubierre effect has been demonstrated.

EM drive

In 2001, Roger Shawyer proposed a concept that used contained microwave field to impart momentum to the ship via coupling to the quantum vacuum ([Shawyer, 2006](#)). Several researchers have reported results of experiments to prove the concept ([White et al., 2016](#); [Brady et al., 2014](#); [Tajmar, 2018](#)). To date, no definitive results have been obtained clearly verifying the concept. However, thrusts in the micro-newton level and thrust to power ratios of 1.2 mN kW^{-1} are claimed.

Transdimensional gates

A common factor among all advanced propulsion concepts is the assumption that the ship must travel through normal space, i.e. it will occupy every point between its origin and its destination. Consequently, all concepts must increase the speed of the ship to some fraction of the speed of light if missions of reasonable duration are to be achieved. This may cause a problem.

Our solar system is surrounded by the Oort Cloud. The Oort Cloud is a spherical shell of primordial material. Objects in the Cloud can range from asteroid sizes to specks of dust. A ship accelerating up to a fraction of c will be traversing the Oort Cloud at high speed. If every solar system has an Oort Cloud equivalent, then the ship will also be traversing their cloud at high speed.

From the reference frame of the ship, a hailstorm of grit may impact the ship during the transition through the cloud. A grain of sand with a mass of one gram traveling at a speed of $0.1 c$ will have a kinetic energy of $4.5 \times 10^{11} \text{ J}$ or 0.1 kt of TNT equivalent. Thus, impacting small grit during the journey could be catastrophic.

Shielding against such an impact will be very difficult. Using a material shield will be massive. Since the grains may be electrically neutral, a magnetic field will be of no use. If a laser is placed and shoots forward to ionize the grit so a magnetic field could deflect it, the mass of the ship will skyrocket to accommodate the power supply. Thus, interstellar transport through normal space may be precluded.

One solution popular among the science fiction community for decades is the transdimensional transport, i.e. the Stargate/wormhole/teleporter. This concept postulates that two points in normal space can be connected through a "subspace". The common illustration is that normal space is a piece of paper which is folded. By jumping across the gap, fast transit is possible between two points without traveling through every point in between. In 1935, Einstein and Rosen postulated the idea of a "bridge" in space to connect two points. This has later been described as a wormhole through which travel between points could be executed.

In order for transdimensional transport to become reality, significant new physics must be demonstrated. But such a system may be the only method to safely travel between the stars.

See Also: Nuclear Fusion; Origins of Nuclear Energy.

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Space Nuclear Power Safety and Safeguards

Patrick R. McClure, Los Alamos National Laboratory, Los Alamos, NM, United States

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Glossary

AEA Atomic Energy Act
CFR Code of Federal Regulations
Curie Describes the rate at which radioactive material emits radiation; 1 Curie = 3.7×10^{10} disintegrations per second
DOE Department of Energy
DOT Department of Transportation
EIS Environmental Impact Statement
EA Environmental Assessment
HEU Highly Enriched Uranium, enrichments greater than 20% ^{235}U
LEU Low Enriched Uranium, enrichments less than 20% ^{235}U
NASA National Aeronautics and Space Administration
NRC Nuclear Reactor Commission
OSTP Office of Science and Technology Policy
RTG Radioisotope Thermoelectric Generator
Rem Roentgen equivalent man: a CGS unit of effective dose; it measures the health effect on the human body of low levels of ionizing radiation
SAR Safety Analysis Report
SARP Safety Analysis Report for Packaging
SER Safety Evaluation Report
SNAP Space Nuclear Auxiliary Power

Introduction—Regulations for building and launching a space nuclear reactor

Historically, space reactors have been in the domain of government institutions, both in the U.S. and in Russia, and to date, they are the only entities to build and launch space reactors. It is now possible for a commercial company to both build and launch a space reactor. This section covers the regulations for building and launching a space reactor.

The Atomic Energy Act of 1954

The Atomic Energy Act (USC, 1954, AEA) is the fundamental U.S. law for both the civilian and the military uses of nuclear materials. As such, it serves as the basis for other regulations governing the possession, processing, and disposal of nuclear materials and the

operation of nuclear facilities in the United States. The AEA defines the Nuclear Regulatory Commission's (NRC) role in commercial nuclear power and non-government research and defines the Department of Energy's (DOE) role in military uses and government research. (Note, the jurisdiction of the NRC and the DOE is not always clear, given that commercial companies can use DOE facilities.)

The development of a space reactor, including the possession, the processing, and transport of nuclear material, along with the operation of a reactor are governed by the AEA. The agency with jurisdiction will be discussed later and is largely dependent on sites used for reactor development and testing. Recent changes to the Presidential directive site the Department of Transportation (DOT) as the approval authority for the launch of a commercial space reactor.

Launching nuclear material into space is not explicitly covered by the AEA. However, since no entity can possess, process, or transport nuclear material without government approval it is a near certainty that the act applies to the launching of nuclear material by a private company into space.

Designing, building, and testing

Any ground facility for the manufacture of fuel, reactor construction, or reactor testing must be licensed. Jurisdiction for licensing is largely dependent on who owns the site. If the facility were located on a DOE site then DOE would be the regulator and license the facility. If the facility were located on private or non-DOE public property (such as a university) then the NRC would be the regulator and license the facility.

DOE facilities are licensed under 10CFR830 (DOE, 2011). This would cover facilities such as fuel development and manufacture (example: the Y-12 plant in Oak Ridge, TN) or reactor testing (example: the Nevada National Security Site.) If Highly Enriched Uranium (HEU) is used as the fuel, then security requirements will dictate that manufacturing and testing be done at a DOE facility.

Commercial fuel cycle facilities are licensed by the NRC under 10CFR70 (NRC, 2003). A reactor test facility would be licensed by the NRC under 10CFR50 (NRC, 2006). The type of license under 10CFR50 depends largely on the thermal power of the reactor tested. Almost all conceivable space reactor concepts would fit into the research reactor category of 10CFR50. This would lower the regulatory burden and cost considerably. A Low Enriched Uranium (LEU) fueled space reactor could be developed by a private company at an NRC licensed facility.

Any license provided by the federal government will also have to include an environmental analysis under the National Environmental Policy Act (USC, 1970, NEPA). This would take the form of an Environmental Impact Statement (EIS) or Environmental Assessment (EA). Licensing and NEPA activities are lengthy (years) and can be very expensive (10's of millions of dollars). Whether or not a new EIS or EA is required would be a function of the site and its existing NEPA documentation. It is possible the existing NEPA for a site could cover the manufacturing and testing of a space reactor.

Security for the manufacturing and testing phase

All nuclear materials above a de minimis threshold require some degree of security. This can range from a few guards and a simple building and building security plan to a very complex security plan with a building requiring special security devices and specially trained guards.

The NRC and DOE have regulations for the amount of security required based on the type and amount of nuclear material. Currently, the NRC has rules that apply to facilities that handle only LEU and small quantities of other isotopes. DOE has facilities that handle both HEU and/or LEU.

The DOE determines nuclear material security by form, enrichment, isotope type, and amount. For an HEU space reactor that is enriched to 93% ^{235}U using an oxide fuel form, fuel amounts greater than 6 kgs would be Security Category 1, the highest level of security. Almost all space reactor designs using HEU will be Security Category 1. For an LEU space reactor that is enriched to less than 20% ^{235}U the Security Category would be less than Security Category 4, the lowest level of security.

Security Category 1 facilities typically spend 10's of millions of dollars a year on security in the DOE. Less than Security Category 4 facilities would have minimal security costs. Historically, commercial companies such as General Atomic and Atomic International have used HEU, but concerns over the use of HEU are much higher today than in the 1960s and 1970s. Because of the policy concerns for HEU proliferation and the cost issues associated with HEU, it is anticipated that an HEU space reactor would require DOE to maintain a chain of custody for the material from manufacture up to the reactor reaching earth orbit or beyond.

Transportation

Transportation of nuclear material is governed by the Department of Transportation (DOT) under Title 49 of CFR – Hazardous Material Regulation (HMR). Radioactive Material is specifically covered in 49 CFR 173 (DOT, 2010). The DOT coordinates with the NRC for the movement of commercial nuclear material and the DOE for the movement of federal nuclear material covered in 10 CFR 71 (DOT, 2011). The NRC certifies shipping packages for the commercial industry, while the DOE certifies packages for DOE.

The transport of nuclear material depends on the amount and the isotope. Small amounts of material are deemed type A and greater amounts are deemed type B. If the isotope is fissile it includes the letter F. LEU has no threshold between type A and B given its low health effects, but it is fissile material. Thus, the transport of fresh Uranium fuel would be type AF.

The type of material determines the requirements for shipping. The requirements must be demonstrated to be met as part of the development of a Safety Analysis Report for Packaging (SARP) on the specific container. A container is certified by either the DOE or NRC using the SARP process depending on the owner of the material. Certification of a container is very expensive and time-consuming.

There are currently no containers for shipping a complete space reactor. A new space reactor would require a new SARP and a new container. However, there are existing containers for shipping fresh fuel if the reactor is shipped in unassembled sub-assemblies.

Another complicating factor for the transportation of nuclear fuel is security. If the nuclear material is HEU or another type of weapons-grade material (an example is plutonium), then high-security requirements would be required. Note that currently, only the DOE is shipping highly secure nuclear material. There currently is no mechanism for a private company to ship highly secure nuclear material, although an agreement with the DOE is possible. Shipping highly secure nuclear material could be extremely expensive and as such is probably not an option for a commercial space reactor company.

Reactor ground testing

A new space fission system may require nuclear ground testing. The testing can be short term (days to months) or long term (years). Short term testing is typically used to determine the performance of the system to steady-state conditions and to a variety of transient situations. Long term testing is required if pre-flight reliability determination is required.

Nuclear ground testing can be very expensive given the regulatory hurdles and waste disposal issues. The cost can vary widely, depending on the site, security issues, presence of existing infrastructure, and length of the test. The need for testing must be carefully weighed against the benefits prior to going to space. The cost of nuclear ground testing may exceed the cost of performance testing in space. This decision will be very specific to the application and the type of space reactor that is desired.

Two primary regulators can be involved depending on where testing is done. If testing is done at a Department of Energy (DOE) Site, then DOE will be the regulator. If a non-DOE site is used for testing, then the regulator will be the Nuclear Regulatory Commission (NRC). There will be small advantages to either regulator, but in general, the requirements for testing will be very similar. The DOE sites will be governed by 10CFR830. The NRC sites will be governed by 10CFR50. The cost of producing a safety analysis report (SAR) for both regulators can be several million dollars. The cost of managing a test to the requirements of the SAR will also involve costs in the millions of dollars.

The use of highly enriched uranium (HEU) will necessitate that the space reactor is tested at a DOE site. HEU requires a high level of security that only exists in the DOE. The DOE security costs at facilities that perform nuclear testing of HEU are typically in the tens of millions per year. How much of the security costs for testing an HEU system that is borne by the company will have to be negotiated with the DOE. If LEU is used, then security costs will be minimal. Many universities, national laboratories, or private entities may have adequate facilities for testing LEU systems.

Existing infrastructure that does not need to be modified is ideal to keep costs low. Modifying existing infrastructure can be expensive since most necessary modifications are typically required by the SAR and come with nuclear safety grade quality assurance requirements. Building new infrastructure is very expensive because all of the construction will be required to be done under very strict nuclear safety quality assurance requirements.

Finally, if the test is short-term, then the cost of the test is lower. This may also mean that the reactor core (the fuel) has undergone very few fissions and might not be considered waste. If the test is long term, the cost is higher and the company may have to pay disposal cost for the reactor core.

Launch safety and safeguards

The approval for launching a nuclear system into space was updated on August 20th, 2019 by the Trump Administration. The presidential memorandum, *Presidential Memorandum on Launch of Spacecraft Containing Space Nuclear Systems* (NSPM-20, 2019), presents the new process and requirements for launching nuclear systems (both reactor and radioisotope) into space. It also for the first time covers commercial as well as government agency launches.

Historically, this approval process began in the Kennedy administration in 1961. McGeorge Bundy authored a memo to the National Aeronautics and Space Council (NSAM 50, 1961) requiring presidential approval (phrased as any announcements) for launching nuclear power into space, with emphasis on the Space Nuclear Auxiliary Power (SNAP) program. NSAM 50 was revised by the Johnson Administration in April 1965 about the time of SNAP-10A launch (the launch of the only U.S. space reactor). NSAM 50 was replaced by presidential directive NSC-25 during the Carter Administration and was signed by Zbigniew Brzezinski. NSC-25 (1977) formalized the requirement for government agencies to obtain Presidential approval through the Office of Science and Technology Policy (OSTP) for launch. NSC-25 covered the launch of a nuclear system by any agency with multiple agencies included in the approval process depending on the mission. NSC-25 was used to launch multiple radioisotope missions, such as Cassini and New Horizons missions. The new NSPM-20 replaces NSC-25 and will be the process going forward.

Launch approval process

NSPM-20 has three tiers for approval of a launch. The first tier is for small radioisotope heater units with low radioactive inventories (paraphrase of the actual requirements). The second tier is for large radioisotope units that produce electricity and have much larger radioactive inventories than Tier 1. Tier 2 also includes all LEU fueled reactors because of the potential for criticality accidents during launch. Reactors, in general, have very low radioactive inventories at launch (even lower than tier 1 radioisotope systems), but criticality accidents could impact people local to the accident, so are in Tier 2. Tiers 1 and 2 are approved by the head of the sponsoring agency (for example the NASA administrator). For commercial launches, the Secretary of Transportation (or designee) is the approval authority.

The final tier, Tier 3 is for HEU fueled reactors and any nuclear system that exceeds the risk limits explained later. Tier 3 launches require presidential approval (although this approval will be delegated to the head of OSTP.) HEU reactors are in this tier for security reasons, not safety reasons, since HEU has proliferation concerns. This is a change from the previous processes for launching nuclear systems where all launches required presidential approval.

Safety analysis

For the launch of all nuclear systems, a Safety Analysis Report (SAR) must be prepared. The SAR analyzes the hazardous/radioactive material, the potential accidents that could impact this material, the accident likelihoods, and accident consequences. The main goal of the SAR is to establish the systems, design features, or reactor physics that make the system safe. This goal is accomplished by presenting the risk of launch in terms of probabilities and consequence to the public in terms of dose as measured in rem¹ and then by demonstrating what features of the nuclear system keep risk low. This means that the SAR is a risk-based approach to safety. The risk limits from NSPM-20 are:

- 25 millirem de minimis, below which no probability criteria apply;
- A probability of 1 in 100 for accidents that have a dose of 25 millirem to 5 rem;
- A probability of 1 in 10,000 for accidents that have a dose of 5 rem to 25 rem;
- A probability of 1 in 100,000 for accidents greater than 25 rem.

If accidents with doses are above 25 rem and have a probability of greater than 1 in 100,000, then the launch approval would default to Tier 3 and require presidential approval. For accidents above the other limits, then normal approval by the agency head applies. Additional guidelines for non-terrestrial applications (example: planetary surface) may apply but have to date not been specified.

For space fission reactors, the new process and criteria present a practical and straightforward process to navigate. In general, space fission reactors will have in the 10's of curies² in initial radioactive inventory at launch with the radioactive inventory bounded by approximately 100 curies (assuming the reactor has never undergone fission). Doses at 1 km for extremely bounding conditions will be on the order of 1 millirem. Given the new 25 millirem de minimis, reactor launch accidents due to fires and explosions will be in the low millirem range. A bounding accident should be sufficient to establish that the accident dose is below that value.

The focus of the SAR for space reactors will then be the potential for the reactor accidentally going critical during a launch accident. Examples of such an accident would be the reactor falling into water where water adds neutron moderation causing criticality or the reactor being crushed on impact into an unfavorable geometry causing criticality. For these accidents, the goal will be to show that the probability of the event is below the criteria in NSPM-20. Criticality accidents (with insertion on the order of a few dollars of excess reactivity³) will in general have excursions on the order of 1.E17 to 1.E18 fissions. This result will produce doses on the order of a few hundred millirem and will therefore have to be shown to have a probability of less than 1 in 100. This is not hard to achieve with the proper design of the reactor, nor hard to prove with simple experiments and calculations. The reactor designer can add features that aid in the prevention of criticality accidents and include such choices as high length to diameter ratios for the reactor core⁴ or added poisons in the reactor core.⁵ These design choices will be specific to the reactor design.

The re-entry of an RTG or space reactor can have a serious impact on the Earth's population and environment. An RTG or space reactor may burn up on re-entry, however, the few historic accidental re-entry events (such as Cosmos 954, see [Heaps, 1978](#)) also indicate that the system could break up and scatter radioactivity across the Earth.

The operations of an RTG and space reactor where human life is involved will require analysis that evaluates the impact of accidents and power source failure on astronaut health.

¹Rem = Roentgen Equivalent Man; a measure of a dose of nuclear radiation. 1 rem = 0.01 Sievert (SI unit for equivalent dose).

²1 Curie = 3.7×10^{10} integration per second (of radioactive material).

³A reactor core having 1 dollar of excess reactivity is "prompt critical"; it does not need delayed neutrons to maintain criticality. With several dollars excess reactivity the chain-reaction will grow up exponentially very fast until the immense heat generated will disassemble the core.

⁴Heating of a high length-to-diameter ratio core will result in significant negative reactivity insertion.

⁵Presence of burnable poisons in a fast spectrum core can eliminate criticality upon immersion in water.

Review process

The review of the SAR is done by the Interagency Nuclear Safety Review Board (INSRB). The INSRB consists of representatives from the Departments of State, Defense, Energy, and Transportation, the Environmental Protection Agency, NASA, and, as appropriate, the Nuclear Regulatory Commission.

The INSRB is tasked with evaluating the quality of the safety analysis and will identify any significant gaps in analysis. The INSRB can recommend areas for additional analysis where it identifies gaps. It is anticipated that the INSRB will use national laboratories to perform some amount of confirmatory analysis in support of the INSRB. The INSRB will write a Safety Evaluation Report (SER) that recommends approval or disapproval to the appropriate approval authority.

Launching a reactor type that has been launched previously

The new launch safety process includes the ability to compare the launch of a previous system to the launch of the same or similar system. The process is similar to the process used by the NRC and DOE for the current licensing of the reactor and nuclear facilities.

For example, for commercial reactors, a design license can be issued that covers the reactor design, but in the context of a generic location or site. Specific differences related to the reactor site would then be addressed after the reactor is sold to a utility and a site chosen. Only issues with the new site that are not “bounded” by the original analysis must be addressed.

For launch safety, the “bounding” insults would be analyzed based upon the expected launch vehicles that would be available. A generic mission would be chosen along with possible mission-specific issues for analysis. This bounding analysis would attempt to examine the worst-case issues for the space nuclear system. Subsequent launches would only deal with new issues that arise for a specific mission. For example, an earth fly-by may have not been addressed in the original worst-case SAR. A mission is now proposed that would allow the reactor to operate after launch to provide electric propulsion while en-route to the outer solar system. The launch requires a fly-by of Earth after the reactor has started up. Since this scenario was not analyzed in the original SAR, the launch approval would require this scenario to be analyzed prior to approval for this launch.

Treaty issues

There are treaty issues to be considered. The United Nations Committee on the Peaceful Uses of Outer Space (UN, 1992) produced a document that outlined the principles relevant to the use of nuclear power sources in outer space. The document is not a treaty, but the U.S. has treated it like a treaty obligation. The requirements are nothing too onerous, but two issues must be addressed. The document requires the use of highly enriched uranium. It is assumed that this was done because that was the fuel used by the U.S. and Russia at the time. However, there is no reason why LEU would not be an acceptable replacement for HEU with respect to this agreement. In the event of an accident, the country of origin must notify other countries. This was a result of a Russian space reactor re-entering the Earth atmosphere and crashing into northern Canada (Heaps, 1978). This was a serious situation for the Russians and it is recommended that low-Earth orbit missions be evaluated carefully by any commercial space nuclear company.

Summary

A commercial company building and launching space nuclear reactors is now part of the U.S. regulatory structure and as such a private company can design, build, ground test, transport, and launch a space reactor. Regulation of the ground activities would depend on where the manufacturing testing and construction occurred and by the amount of enrichment of the fuel. Launching a reactor by a commercial company is now possible given the new Presidential directive, NSPM-20. This directive provides the approval authority for launch (the head of the sponsoring agency for government missions and the head of DOT for commercial missions.) New criteria have been established for launch safety that favors launching space reactors.

See Also: Non-Proliferation Aspects of Nuclear Energy; Safety Evaluations; Safety of Nuclear Reactors; Safety, Regulation, and Decommissioning of Commercial Nuclear Power Reactors—Introduction.

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Nuclear Fusion – Introduction and Overview

David J. Campbell*, Munich, Germany

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Glossary

Aspect ratio The ratio R/a , where R is the torus major radius and a the (horizontal) minor radius.

Beta β : Ratio of the volume averaged plasma thermal pressure, $\langle p \rangle$, to the pressure exerted by the toroidal magnetic field at the plasma center, $\beta = 2\mu_0 p B_t^2 / 0$. β is typically quoted in percent (Wade, 2021; Lao et al., 2021).

Bremsstrahlung 'Braking radiation': radiation emitted from a plasma via the interaction (deceleration) of free electrons with ions.

Burning plasma Plasma in which heating is dominated by fusion products of thermonuclear reactions, i.e. α -particles in deuterium-tritium plasmas.

Energy confinement time $\tau_E = W_{th}/P_{loss}$: ratio of the total thermal energy of the plasma, W_{th} , to the total power lost by all mechanisms (transport, radiation etc.), P_{loss} ; under steady-state conditions P_{loss} is equal to the total plasma heating power, i.e. the sum of the α -particle heating power resulting from DT fusion reactions, P_α , and the external heating power injected into the plasma, P_{heat} (Wade, 2021).

FPP Fusion power plant.

Hohlraum 'Hollow cavity': a small metallic cavity which holds the DT capsule in inertial confinement laser-indirect-drive experiments; laser energy is injected into the cavity generating X-rays from the walls which compress and heat the capsule to a density and temperature that enable the fusion of its constituents.

ICF Inertial confinement fusion.

ITER Major tokamak-based facility designed to produce 500 MW of fusion power using deuterium-tritium plasmas; ITER, which means 'the way' in Latin, is an international project under construction in southern France.

keV kiloelectronvolt: temperatures, T , in thermonuclear plasmas are normally quoted in keV ($T(\text{keV}) = 10^{-3} k_B T(\text{K})/e$, with k_B the Boltzmann constant and e the electron charge): $1 \text{ keV} = 1.1605 \times 10^7 \text{ K}$.

MCF Magnetic confinement fusion.

MHD Magnetohydrodynamic (relating to stability behavior of a magnetically confined plasma).

Q_{elec} Electrical gain: $P_{e,gross}/P_{e,circ}$ – ratio of the gross electrical output power of a power plant to the recirculating electrical power required to maintain the plant operation.

Q_{th} Fusion gain: P_{DT}/P_{heat} – ratio of the DT (thermal) fusion output power of a plasma to the input heating power from external sources.

RAFM Reduced-activation ferritic-martensitic: a family of steels in which the alloying elements are chosen to lower the susceptibility to radiation damage and to reduce the long-term activation under neutron irradiation.

Reactivity As applied to thermonuclear plasmas, the product of the reaction cross-section, $\sigma(v)$, for a given reaction (e.g., deuterium-tritium) and the relative velocities, v , of the reacting particles, averaged over the velocity distribution functions of the particle populations (for thermal plasmas, the velocity distribution function is taken to be a Maxwellian), written $\langle \sigma v \rangle$.

TBR Tritium breeding ratio: ratio of the number of tritium atoms bred in the breeding blanket per atom of tritium burned in fusion reactions within the plasma.

*Retired.

Introduction

Thermonuclear fusion research has developed, since its origins in the middle years of the 20th century, into a wide-ranging international program aimed at establishing the feasibility of generating energy for the benefit of mankind from the controlled fusion of light atoms. At its heart is the study of plasmas, i.e. fully ionized gases, at temperatures exceeding those at the center of the sun. However, as the challenges of controlling plasmas under the conditions required for net energy production have emerged and have been understood, if not fully resolved, fusion studies have drawn extensively on ideas, experimental techniques and technology from many other areas of science and engineering. The focus of thermonuclear fusion research remains the achievement of net energy production and the exploitation of fusion as a large-scale energy resource, but the research activities encompass a dynamic and stimulating range of interactions with diverse areas of science and technology.

As introduced by Morse (2021), the idea that fusion reactions produce the energy powering the sun was first proposed by Eddington in 1920, though the quantitative details were not worked out until the 1930s by Bethe and others. The source of energy in the sun's core, where the plasma has a temperature of 1.5×10^7 K and a density of 1.6×10^5 kg m⁻³ (corresponding to $\sim 10^{32}$ protons m⁻³), is the so-called proton-proton cycle, which converts protium (i.e. the lightest hydrogen isotope) into ⁴He. However, reproducing this reaction cycle at the parameters achievable in laboratory plasmas is simply not practical and the focus of contemporary fusion research is the deuterium-tritium (DT) reaction, which is the fusion reaction with the largest cross-section at the temperatures likely to be achieved in laboratory experiments (several 10^8 K). As discussed later, deuterium is relatively abundant in the earth's oceans, but tritium is radioactive, with a relatively short half-life of 12.3 years, and so must be produced within a fusion reactor by the reaction of neutrons with lithium contained within the reactor core structure. The most likely fuel cycle for a fusion reactor at the heart of a power plant is therefore referred to as the deuterium-tritium-lithium cycle, and it is this approach to producing fusion energy which will be the focus of subsequent chapters in this Section.

Although the study of hot, magnetically confined plasmas was initiated in the late 1930s (see, e.g., Braams and Stott, 2002), systematic studies toward the feasibility of producing energy by thermonuclear fusion reactions were not initiated until the late 1940s, firstly by figures such as Thomson, Thonemann and Ware in the UK, followed soon after by Spitzer, Tuck, Post and their collaborators in the US, and Tamm, Sakharov, Artsimovich and collaborators in the USSR. These activities were almost immediately enveloped in the curtain of secrecy under which most early nuclear research was conducted. The recognition of the complexities involved in magnetically confining plasmas with parameters approaching those required for the production of a net energy gain from fusion reactions led to a gradual emergence of the research into the public domain, and this process was essentially completed at the 1958 Conference on the Peaceful Uses of Atomic Energy, held in Geneva under the auspices of the United Nations (United Nations, 1958), at which the leading research communities presented the status of research on magnetically confined 'thermonuclear' plasmas. Since that time, the variety of magnetic confinement concepts has shrunk significantly, but remarkable progress has been made in approaching the conditions for achieving a net energy gain in DT plasmas, particularly in the tokamak (see Galvão and Canal, 2021). Key concepts related to an alternative approach to the controlled production of fusion energy, inertial confinement, emerged from the development of thermonuclear weapons. However, the practical study of inertially confined plasmas in the laboratory only became viable following the invention of the laser in 1960. Although much inertial confinement fusion research remained classified until the 1990s, the field has made remarkable progress toward demonstrating a net energy gain from laser-compressed DT capsules. The status of current studies in experiments, theory and technology across these major areas of contemporary fusion research and the approach to the achievement of large-scale (multi-MW) release of fusion energy are reviewed in the following chapters of this Section.

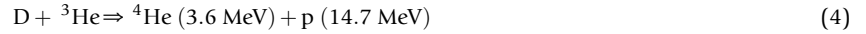
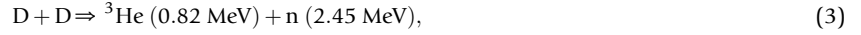
This chapter provides a brief overview of the major subjects which are discussed in subsequent chapters of *Nuclear Fusion R&D* Section of this Encyclopedia. The Section "**Energy production from the fusion process**" of this chapter introduces the deuterium-tritium-lithium fuel cycle and discusses the perceived advantages of large-scale energy production from fusion which have provided the motivation for the research in this area. The basic parameters allowing a simple characterization of the fusion performance of high temperature plasmas are discussed in the Section "**Measures of plasma fusion performance.**" The Section "**Current status of fusion R&D**" outlines the essential principles associated with the magnetic and inertial confinement approaches to plasma confinement and fusion energy production and the present status of R&D in these major fields. An overview of the topics reviewed in *Nuclear Fusion R&D* is presented in the Section "**Overview of nuclear fusion R&D,**" while the Section "**Conclusions**" provides some indications of the future directions of fusion energy research toward the construction of fusion power plants, which is the ultimate goal of the R&D activities in this field.

Energy production from the fusion process

The behavior of the binding energy curve at low atomic number provides numerous opportunities for producing energy from fusion reactions if a sufficiently high plasma temperature is attained, though this does not guarantee that a net energy gain can be obtained from any given configuration. The proton-proton cycle, which is the source of the sun's energy, is based on the weak interaction and its reaction rate is much too slow for earth-bound applications. Analysis of reaction rates between isotopes of hydrogen and helium have revealed several possible processes that could power a terrestrial fusion reactor. Of these, the deuterium-tritium reaction is the most favored:



This reaction consumes only 0.4% of the mass of the reactants in producing its energy output. In a reactor plasma, the 3.5 MeV α -particle (${}^4\text{He}$) interacts with the thermal plasma particles and provides the heat necessary to maintain the high temperatures required to sustain a high reaction rate, while the 14.1 MeV neutron escapes from the plasma and deposits its energy in the surrounding 'blanket' structure, producing heat for electricity production and, as discussed below, replacing tritium burned in the plasma by reactions with lithium contained within the blanket. The most immediate competitors for application to fusion energy production are those referred to in Fig. 1:



As can be seen in the figure, the DT reaction has the highest reactivity by almost two orders of magnitude at temperatures of $\sim 10 \text{ keV}$ (i.e. $\sim 10^8 \text{ K}$) that are likely to be practical in laboratory-based plasmas under conditions conducive to net energy production (as discussed in greater detail by Wade, 2021). A further reaction discussed by Wade, but again demanding much more challenging plasma conditions for net energy production than the DT reaction, is the proton-boron reaction,



which has attracted attention because the fusion products consist of 3 α -particles and the reaction is therefore almost entirely aneutronic – some neutrons with energies $< 1 \text{ MeV}$ are produced by side reactions – implying a very substantial reduction in the activation of the reactor structure and elimination of the need to produce and handle radioactive tritium. Further discussion of the issues related to the exploitation of so-called 'advanced fuels' for the production of fusion energy can be found in Stott (2005) and Dawson (1981).

The availability of tritium is potentially a major issue for the DT approach to fusion energy production, since its half-life of 12.3 years means that there are essentially no resources in nature, although operation of heavy-water moderated/cooled fission

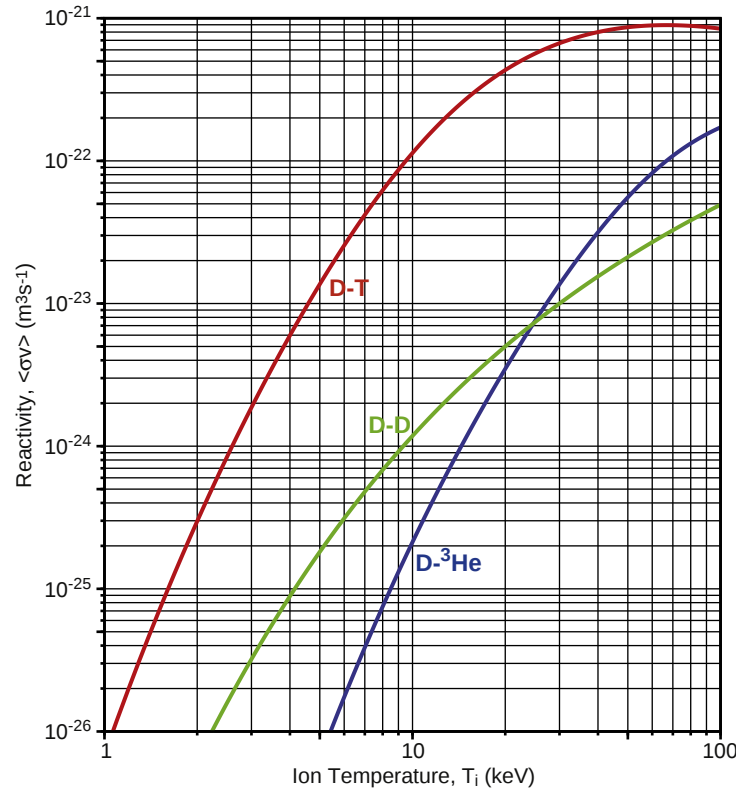
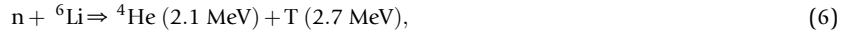


Fig. 1 The reactivity, $\langle \sigma v \rangle$, averaged over a Maxwellian velocity distribution as a function of ion temperature, T_i , for the D-T, D-D and D- ${}^3\text{He}$ fusion reactions. Note that the reactivity for the D-D reaction is the sum of the reactivities for the reactions in Eqs. (2) and (3), which are essentially equal. Curves are derived from the parametric fits in Bosch and Hale (1992). Based on data published by Bosch H-S and Hale GM (1992) Improved formulas for fusion cross-sections and thermal reactivities. *Nuclear Fusion* 32: 611–631, <https://doi.org/10.1088/0029-5515/32/4/107>.

reactors over the past 50 years or so has generated a stockpile of several tens of kg, perhaps as much as 50 kg (see, e.g., Kovari et al., 2018; Pearson et al., 2018). Since a power plant would consume 55.8 kg of tritium per GW_{th} yr of fusion power production, it is clear that an alternative source of tritium is required. This source is provided by the reaction of neutrons escaping from the plasma with lithium which is stored either as a ceramic or in a liquid form in the so-called ‘breeding blanket’ surrounding the plasma (Bocaccini and Day, 2021). There are two possible reactions with lithium isotopes:



It turns out that, over the relevant range in neutron energy, only the first of these reactions makes a significant contribution to tritium breeding. Therefore, it is foreseen that the lithium content of the breeding blanket would be enriched from the 7.4% content in natural lithium to values of up to 90%. Eqs. (1), (6) and, to a lesser extent, (7) therefore constitute the deuterium-tritium-lithium cycle which is currently expected to provide the most rapid access to the demonstration of significant fusion power gain and to form the basis of fusion energy production in the first generations of fusion power plants. The cycle is illustrated schematically in Fig. 2.

Deuterium and lithium are therefore the essential fuel resources for large-scale exploitation of fusion energy. Fortunately, both are abundant on earth: the natural abundance of deuterium in hydrogen is 1 part in 6700, so that the water in the world’s oceans holds $\sim 4 \times 10^{16}$ kg of deuterium. Assuming that a 1 GWe fusion power plant must produce ~ 3 GW_{th} of fusion power, the annual deuterium consumption would be ~ 110 kg. Operating a fleet of 1000 large (1 GWe) fusion power plants for a billion years would thus consume less than 1% of this deuterium reserve.

The 1 GWe power plant would also need ~ 5000 kg of natural lithium (taking credit only for the 7.4% content of ${}^6\text{Li}$ to breed the ~ 170 kg of tritium fuel required). Currently demonstrated and economically recoverable reserves of lithium are estimated to be $\sim 1.3 \times 10^{10}$ kg in ores and deposits, with the total land-based resource perhaps amounting to $\sim 3.9 \times 10^{10}$ kg (Bradley et al., 2017). Taking only the smaller value, the demonstrated lithium reserves would supply the fleet of fusion power plants for ~ 2600 years. Of course, there is competition for lithium use from a variety of other applications, including the increasing use of battery-powered electric cars, and the figure of 5×10^6 kg of natural lithium required each year to fuel a fleet of 1000 fusion power plants amounts to $\sim 10\%$ of the annual lithium consumption in 2019 (U.S. Geological Survey, 2020). This increasing consumption is also driving further exploration and an increase in identified reserves. Moreover, the use of enriched ${}^6\text{Li}$ in fusion breeder blankets implies that much of the natural lithium within the quoted figure of 5000 kg would produce ‘depleted’ ${}^7\text{Li}$ which could be made available for other applications. It should also be noted that the world’s oceans constitute the ultimate lithium resource, since each liter of seawater contains $\sim 2 \times 10^{-7}$ kg of lithium, yielding a total resource of $\sim 2.8 \times 10^{14}$ kg. If cost-effective means of extraction of lithium from seawater are developed in response to the increasing lithium demand, the timescale over which the D-T-Li cycle could be exploited for the production of fusion energy would extend into millions of years.

Implementation of a large-scale program of fusion energy production would depend not only on the technical feasibility of attaining significant fusion gain, as discussed later, but also on achieving public acceptability, primarily by a convincing demonstration of the safety and environmental aspects of fusion energy. As discussed in greater detail by Taylor (2020), Campbell (2021), and Dunne et al. (2021), analysis of the safety performance of fusion power plant operation, in terms of the risk of radiation exposure to personnel, public and the environment, has confirmed important safety characteristics, in particular, that potential off-normal

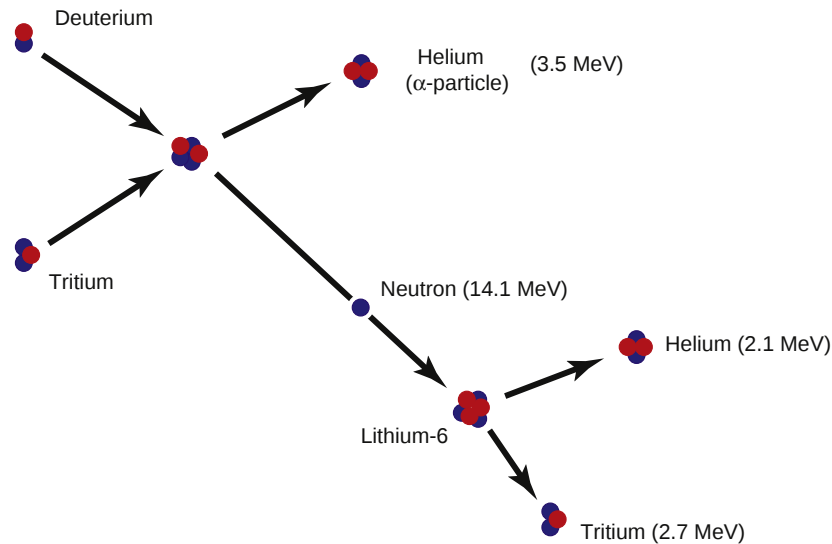


Fig. 2 A schematic illustration of the deuterium-tritium-lithium cycle foreseen in fusion reactors to both produce fusion energy (the DT reaction) and breed replacement tritium (the n - ${}^6\text{Li}$ reaction).

events within the plant would not require evacuation of public adjacent to the plant. This was also confirmed by the licensing process under French nuclear regulations for the ITER facility currently under construction in Provence (Taylor et al., 2013).

While the favorable characteristics of fusion energy as an essentially zero-carbon source of energy are recognized, e.g. (Tokimatsu et al., 2003; Yamazaki et al., 2011), a key aspect in the assessment of the environmental impact of fusion is the avoidance of a requirement for long-term storage of radioactive waste resulting either from the fusion reaction products or from irradiation of the reactor structure. Since the products of the DT reaction are helium and neutrons, there is no need for waste storage associated with either product. Irradiation of the reactor structure by neutrons will produce transmutation reactions and radioactive tritium will diffuse into the plasma facing components, and possibly also into elements of the tritium breeding and fuel processing systems. Techniques exist for detritiation of materials following plant decommissioning and a major R&D activity is underway to qualify so-called low-activation materials, e.g. reduced-activation ferritic-martensitic (RAFM) steels (see, e.g., Litnovsky et al., 2021; Stork and Zinkle, 2017 and related papers), which should allow the recycling of structural materials into further fusion power plants approximately 100 years after the decommissioning of the originating power plant.

Measures of plasma fusion performance

While the sun relies on the intense pressure produced by its gravitational force to maintain the conditions for fusion reactions to be sustained within its core, alternative approaches must be found to achieve the production of significant fusion energy in terrestrial laboratories. As is evident from Fig. 1, temperatures of 10–20 keV must be reached to produce significant reactivity, even for DT plasmas, implying that a means must be developed of heating and confining a high temperature plasma for sufficiently long to allow an adequate fraction of the plasma ions to react and produce fusion energy. Two basic methods have been developed to a high level of scientific and technical sophistication: *magnetic confinement* (MCF) and *inertial confinement* (ICF). Since, at these temperatures (in fact, above a temperature of ~ 10 eV), a hydrogen plasma is fully ionized, the use of a strong magnetic field to confine the charged ions and electrons of the plasma, and to minimize the interactions with the surrounding walls of the container, via the Lorentz force is clearly one possibility worth exploring, and so an extensive research program exploiting this approach has been implemented internationally during the past 70 years. Alternatively, if one can deliver an intense pulse of energy to a small capsule of fusion fuel sufficiently rapidly that the capsule is compressed, and the fuel heats and reacts before the energy delivered to the particles causes the resultant plasma to blow apart, it is also possible to obtain a significant yield of fusion energy. This approach relies on the *inertia* of the particles within the capsule to confine the plasma for sufficiently long periods – hence the term inertial confinement. This method has only become viable since the development of sufficiently high-power lasers to deliver intense pulses of energy to a DT capsule (although alternative ‘driver’ technologies are also now available, as discussed later). Nevertheless, the exploration of this approach to fusion energy production has also advanced rapidly in recent decades.

Although both approaches to the confinement of thermonuclear plasmas are conceptually appealing, a key question is the identification of the conditions under which sufficient reactions occur, not only to compensate for the energy used to generate the fusion reactions, but also to produce a sufficient gain in fusion energy so that a practical amount of excess energy is released which can be converted to electricity and distributed on the grid. The detailed considerations related to this question are discussed by Wade (2021), in relation to MCF, and by Atzeni (2021) and Tikhonchuk (2021) in relation to ICF. Lawson (1957) developed an initial response to this question, resulting in a condition, the Lawson criterion, based on the plasma density, n , and the energy confinement time, τ_E , under the assumption that the plasma ion temperature is ~ 10 keV. This criterion is typically quoted in the form,

$$n\tau_E > 1 \times 10^{20} \text{ m}^{-3}\text{s}. \quad (8)$$

This inequality, however, was derived under certain specific assumptions, including conditions for the conversion of thermal power to electrical power and, in essence, it defines a necessary, but not sufficient, condition for the achievement of an approximate balance between input heating power to the plasma and the fusion energy output, a condition generally referred to as ‘breakeven.’

A more general condition which is now widely applied in analyzing fusion energy production in magnetically confined plasmas is referred to as the fusion triple product, $n_i T_i \tau_E$, where n_i is the plasma ion density and T_i the plasma ion temperature. The value of this condition in assessing the approach to net fusion energy production is that in the temperature range of particular interest for the DT reaction, 10–20 keV, the reactivity $\langle \sigma v \rangle$, where σ is the (velocity-dependent) reaction cross-section, v the relative velocity of the reacting particles and $\langle \sigma v \rangle$ represents the average over the reacting particle velocity distributions, can be expressed, to within 10%, as, e.g., (Wesson, 2011),

$$\langle \sigma v \rangle = 1.1 \times 10^{-24} T_i^2 \text{ m}^3\text{s}^{-1} \text{ (with } T_i \text{ in keV)} \quad (9)$$

The criterion on the fusion triple product for *ignition*, the conditions under which the α -particle power produced by fusion reactions provides the power required to maintain the hot plasma (i.e. with no external heating), can then be expressed as,

$$n_i T_i \tau_E > 3 \times 10^{21} \text{ m}^{-3}\text{keVs}. \quad (10)$$

Thus, for example, this condition can be satisfied with $n_i = 10^{20} \text{ m}^{-3}$, $T_i = 10$ keV and $\tau_E = 3$ s, very similar parameters around which the ITER experiment, discussed later, has been designed, but with the aim of achieving $Q_{th} \geq 10$, rather than ignition ($Q_{th} \rightarrow \infty$). It should be emphasized that Eq. (10) has been derived under very specific assumptions, e.g., that there are no

impurities (i.e. ions other than D and T) diluting the plasma, so that the equation establishes a minimum criterion, under essentially ideal conditions, for reaching ignition. It should also be noted that the equation implicitly assumes uniform densities and temperatures across the plasma. However, in any realistic magnetically confined plasma the density and temperature profiles are peaked in the plasma center with a radial distribution, falling to low values at the plasma edge. The fusion triple product reported in the literature is then often expressed in terms of the central values of ion density and temperature. Thus, for example, if parabolic plasma profiles are assumed and central plasma parameters are inserted in Eq. (10), the numerical coefficient in the criterion rises from 3 to 5.

Inertial confinement involves very different plasma parameters from those typical of magnetically confined plasmas, since the reliance on plasma inertia implies a very short confinement time, which must be balanced by the production of much higher densities, assuming that the plasma temperatures are roughly similar in the two cases. Assuming that the compressed capsule has a radius of R_c , a characteristic value for the ‘inertial’ confinement time, τ , is given by the ‘matter confinement time,’ i.e. the time taken for the capsule to expand rapidly as a result of the internal pressure of the plasma,

$$\tau \approx R_c / c_s, \quad (11)$$

where c_s is the sound speed in the plasma. An approximate condition for the achievement of ignition in inertially confined plasmas can then be derived in terms of the ‘areal density,’

$$\rho R_c > 1 \text{ g.cm}^{-2}, \quad (12)$$

where ρ is the mass density of the compressed capsule. Note that the condition is expressed here in cgs units, as is conventionally the case in the inertial confinement analysis. A detailed discussion, with a more precise expression of the ignition conditions for inertially confined plasmas, is given by Atzeni (2021). For comparison with the magnetic confinement parameters given previously, this condition would be satisfied for $T_i \sim 5 \text{ keV}$, $n_i \sim 5 \times 10^{25} \text{ cm}^{-3}$ and $R_c \sim 10^{-2} \text{ cm}$. These conditions are being approached in experiments at the National Ignition Facility (NIF) discussed below.

Both MCF and ICF devices should ultimately evolve into fusion power plants producing electricity on a continuous basis. Whereas MCF devices are designed to confine plasmas under long-pulse, and eventually continuous, conditions with a steady production of fusion power, ICF devices would produce a stream of pulses of fusion energy, one associated with each ‘micro-explosion.’ The measure of fusion performance commonly used in each research field to characterize current experiments therefore differs somewhat. In MCF, the fusion gain, Q_{th} , is a measure of the ratio of total fusion output power to external heating power injected to maintain the plasma operating conditions, and this approaches infinity at ignition. A further measure of fusion gain, often referred to as ignition margin, is the parameter, C_i ,

$$C_i = P_\alpha / P_{loss} = n_i T_i \tau_E / 3 \times 10^{21}. \quad (13)$$

P_α is the α -particle power deposited in the plasma by fusion reactions, while P_{loss} is the power lost from the plasma by all means (transport processes, radiation etc.). In steady-state at ignition, these two factors balance and $C_i = 1$, illustrating the utility of the fusion triple product as a measure of fusion plasma performance, even in experiments where only hydrogen or deuterium is used. Since $P_\alpha = P_{DT}/5$, where P_{DT} is the total fusion power produced by DT reactions, it is straightforward to show that,

$$C_i = Q_{th} / (Q_{th} + 5), \quad (14)$$

so that Eqs. (13) and (14) provide a straightforward conversion between measured plasma parameters and potential fusion gain in DT plasmas. Since experimental studies in ICF currently consist of investigations of individual micro-explosions, the relevant gain factor, G , is usually defined by,

$$G = E_{DT} / E_{driver}, \quad (15)$$

where E_{DT} is the total fusion energy produced by the capsule compression and E_{driver} the laser (or other driver) energy used to compress the capsule. A quantitative comparison of the fusion performance measures in MCF and ICF has been given by Betti et al. (2010).

Current status of fusion R&D

Although the simplest approach to magnetic confinement uses the longitudinal field inside a long straight solenoid, the rapid transport of plasma particles along the field lines leads to unsustainable losses from the ends of the configuration, even when the ends are ‘plugged’ by magnetic mirrors, i.e. regions of increasing magnetic field intensity. This approach has therefore almost entirely been abandoned and, since the 1950s, research has predominantly focused on toroidal devices. As explained in greater detail by Galvão and Canal (2021), within a toroidal configuration, the confining magnetic fields must have a helical structure to provide effective confinement of the plasma particles, and this helical structure is produced by several different techniques. In tokamak plasmas (Zohm, 2021), which have, to date, produced the highest fusion gain, the helical fields are produced by a strong *toroidal* field applied by a dedicated coil system combined with a weaker *poloidal* field produced by an inductively-driven current flowing toroidally in the plasma. The stellarator (Yamada, 2021) has the merit that the helical fields can be produced entirely by external (to the vacuum vessel) coils, which may be continuously wound around the torus, or may be modular, analogous to, but of considerably more

complicated shape than, the toroidal magnetic field coils of the tokamak. A third toroidal configuration that has been extensively studied, but is less favored in contemporary research, is the reversed field pinch, RFP (Zuin, 2021), in which the plasma is confined by an externally applied weak toroidal field combined with a stronger poloidal field produced by an inductively driven current and enhanced by an internal dynamo effect. Typical magnetic field strengths in these devices vary over the range 1–8 Tesla (T).

As understanding of the behavior of magnetically confined plasmas developed, increasingly large and ambitious devices were constructed, since it became clear that plasma fusion performance, in particular the energy confinement time, was strongly coupled to the plasma size. The confirmation in 1969 that temperatures of ~ 1 keV had been attained in the tokamak T-3 at the Kurchatov Institute in Moscow (Peacock et al., 1969) focused attention on this configuration and the range of devices expanded significantly during the 1970s, leading to the construction of a trio of large tokamaks, the Tokamak Fusion Test Reactor (TFTR) in Princeton, USA, the Joint European Torus (JET) at Culham, UK and JT-60 (later modified to JT-60 U) at Naka, Japan. In parallel with the increase in scale of the devices, significant experimental and theoretical developments allowed improvements in the control of plasma stability and purity, which, combined with the scaling of energy confinement time with torus major radius, produced substantial improvements in the values of fusion triple product – by approximately 5 orders of magnitude between 1965 and 1995 (see, e.g., Wade, 2021). All of these facilities conducted experiments in either hydrogen or deuterium (and occasionally in helium) to minimize the activation of the device and allow for continued ‘hands on’ maintenance and upgrades.

Progress in the values of $n_i T_i \tau_E$ achieved by 1990 led to the preparation of the first tokamak experiments using DT plasmas. These were conducted in JET and TFTR, which had been designed to handle the complications of operating with DT fuel, between 1991 and 1997. The experiments produced a maximum fusion power of 10.7 MW in TFTR (Hawryluk et al., 1998) and 16.1 MW in JET (Keilhacker et al., 1999), the latter experiment reaching (transiently) a value of Q_{th} of (0.95 ± 0.17) . A comparison of the fusion output power obtained in the TFTR and JET experiments is illustrated in Fig. 3. The level of (equivalent) fusion performance was extended further in deuterium plasmas in JT-60 U, where a peak value of $n_i(0)T_i(0)\tau_E = 1.5 \times 10^{21} \text{ m}^{-3} \text{ keVs}$ was attained, with a central value of ion temperature of ~ 45 keV (Ishida et al., 1997). An extensive program of tokamak experiments continues to refine the understanding of fusion plasma behavior, with further DT experiments in JET planned in 2021, but much of the emphasis in the tokamak program has moved to the preparations for studies of burning plasmas with $Q_{th} \sim 10$ in ITER, which is due to begin operation in the late 2020s (ITER, 2002; Donn   et al., 2021). In parallel with the increases in fusion performance demonstrated by the largest scale tokamaks, experimental facilities exploiting superconducting magnet technology have been exploring plasma behavior in long-pulse operation out to several hundred seconds, for example, Tore Supra/WEST in France (Jacquinot, 2005; Loarer et al., 2020), EAST in China (Wan, 2007) and KSTAR in Korea (Kwon et al., 2011). An overview of progress in the tokamak experimental program is presented by Zohm (2021), while wide-ranging reviews of the progress in tokamak physics which provides the scientific basis for the ITER design can be found in ITER (1999) and ITPA (2007).

The stellarator concept was developed in the 1950s, but the breakthrough represented by the observation of high plasma temperatures in T-3 caused much of the research focus to shift from the stellarator (and also the RFP) to the tokamak. Continued conceptual developments that generated helical configurations with the potential for achieving an equivalent level of confinement to that

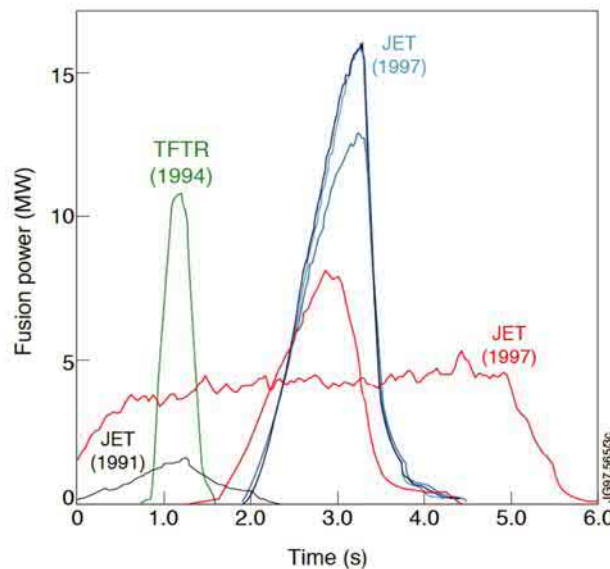


Fig. 3 Overview of the principal results from deuterium-tritium experiments in JET and TFTR during the 1990s. The highest JET fusion power production obtained in 1991 used only 11% T in D (JET Team, 1992), while all other experiments used approximately 50:50 mixtures of DT. The TFTR results obtained in 1994 are taken from Hawryluk et al. (1998) and the JET results in 1997 from Gibson et al. (1998). The JET quasi-stationary plasma with approximately 5 s of fusion power output produced the highest fusion energy of any pulse to date, 22 MJ. Reproduced courtesy of EUROfusion,    EURATOM 2021.

obtained in tokamaks revitalized stellarator studies, and construction of two medium-scale devices was launched in the 1990s. The Large Helical Device (LHD), in Toki, Japan, was brought into operation in 1998 (Iiyoshi et al., 1999), while Wendelstein 7-X, in Greifswald, Germany, began operation in 2015 (Bosch et al., 2017). These devices, which also have fully superconducting magnet systems, are confirming the expectations developed from theoretical analysis. They are achieving levels of performance similar to that of equivalent scale tokamaks while also developing the basis for steady-state operation of magnetically confined plasmas. Further details of advances made in stellarator plasma performance are discussed by Yamada (2021). Progress in increasing the plasma performance in the RFP configuration has been limited by the dominance of large-scale MHD instabilities responsible for the plasma dynamo that forms the field reversed configuration. Experiments such as RFX/RFX-mod/RFX-mod2 (Marrelli et al., 2019) are developing techniques to control these instabilities and contributing significantly to the deeper understanding of instability control in magnetically confined plasmas more generally, as discussed at greater length by Zuin (2021).

The possibility of using intense laser beams to compress a small target capsule of DT fuel to high pressures was recognized soon after the invention of the laser in 1960, but due to the classified nature of much of the relevant R&D, progress in the development of this concept was obscure until the partial declassification of some US research in 1972. Nuckolls et al. (1972) first published the idea of using laser compression of a capsule of DT to reach thermonuclear conditions and produce fusion energy. Although much of the research remained classified until 1994, the declassification of US research in that year made evident the significant progress that had been achieved in all key technology areas and in the physics understanding of the compression process, as well as in identifying the remaining challenges to be resolved to produce a net fusion energy gain. The leading concept in ICF remains the laser-driven approach (Regan, 2021) and two alternative approaches have been developed for compression of the DT capsule, referred to as laser-direct-drive (LDD), in which the laser beams illuminate the capsule directly, and laser-indirect-drive (LID), in which the laser beams are injected into a metallic cavity (or ‘hohlraum’) containing the capsule, generating X-rays from the wall of the cavity that compress the capsule.

The key concept in the inertial confinement approach to fusion is to produce a central hot-spot in the compressed DT fuel which initiates the plasma burn, resulting in the generation of α -particles. The burn then propagates through the compressed fuel due to the α -particle heating of the cold outer layers of fuel. Additional advanced ignition schemes, such as ‘fast ignition’ and ‘shock ignition,’ have also been developed, e.g. (Atzeni, 2021). The LDD and LID concepts across have been extensively studied in major facilities across the international research program in ICF (Finnegan, 2021) and two large-scale multi-beam facilities, the National Ignition Facility (NIF) at Livermore in the US, and the Laser Megajoule (LMJ) near Bordeaux in France, are in operation with laser parameters capable of compressing DT targets to ignition conditions (laser pulse energies of 1.8 MJ and 1.4 MJ, respectively, with pulse widths of several nanoseconds). Gradual progress has been reported from NIF LID experiments which have addressed the elimination of residual asymmetries in the compression and subtleties associated with the detailed pulse shape driving the compression. In experiments with 1.5 MJ laser pulses, areal densities, ρR_c , of 0.3 g cm^{-2} have been achieved, producing a total fusion yield of 54 kJ and ~ 10 kJ of α -particle energy deposited in the hot spot (Le Pape et al., 2018). As illustrated in Fig. 4, these experiments are approaching the critical point at which the deposited α -particle energy would balance bremsstrahlung and conduction losses.

In parallel with the progress in laser-driven ICF, alternative driver approaches using heavy ion beams or X-rays produced by high power Z-pinchs have been developed. As discussed by Seidl (2021), recent research on the former concept has focused on the development of the technology required to support the acceleration of ion beams with the parameters required to investigate high energy density plasmas, leading ultimately to the possibility of heavy ion beam driven compression of a DT capsule. The third major strand in ICF research, involving high power pulsed Z-pinchs to generate compression of fine tungsten wire arrays, thereby producing an intense burst of X-rays that subsequently compress a DT capsule, has also received considerable attention (Gomez et al., 2021). A particularly promising development of the Z-pinch concept is referred to as magnetized liner inertial fusion (MagLIF), proposed in the late 2000s (Slutz et al., 2010). This technique involves the application of a strong (10–30 T) axial magnetic field to a small (5 mm diameter) cylinder which confines DT fuel heated to a temperature of several hundred eV by a multi-kJ laser pulse. On application of the Z-pinch current pulse to the cylinder, it implodes, compressing both the DT fuel and the magnetic flux, so that multi-kilotesla magnetic fields can be produced, contributing to the confinement of the hot plasma produced by the compression. The most recent experiments using deuterium fuel have produced plasma ion temperatures of 3.1 keV and DD neutron yields that would correspond to a DT fusion energy yield of 2 kJ (Gomez et al., 2020). While significant challenges remain in applying this approach to the generation of fusion energy, these results are promising and motivate the further development of this concept.

Overview of Nuclear Fusion R&D

Subsequent chapters in the *Nuclear Fusion R&D* Section are organized to provide an overview of the current status of research in both the MCF and ICF programs, to identify many of the key facilities through which the potential for terrestrial energy production from the fusion process is being explored, to present the progress made in understanding the physics determining the behavior of high temperature thermonuclear plasmas, to highlight the major technological developments supporting the progress in fusion plasma performance, to summarize the physics and technology issues at the forefront of research, to look forward to the devices which will advance research in the coming decades toward the demonstration of significant fusion gain and substantial levels of fusion power production, and to outline the concepts that have been developed to date for fusion power plants supplying baseload electricity to the grid once the major outstanding scientific and technological issues are resolved.

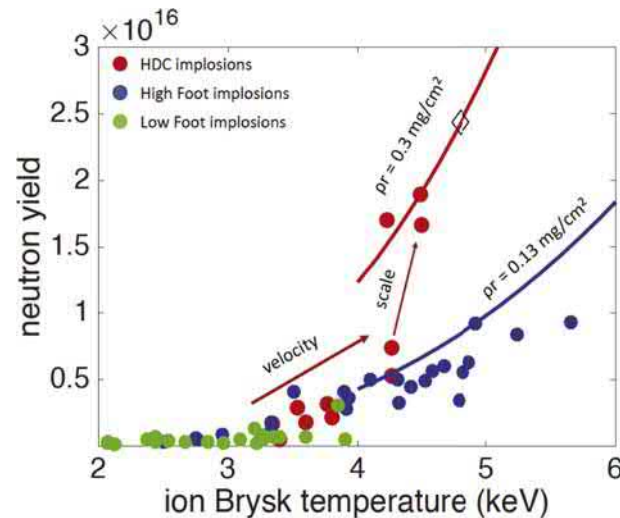


Fig. 4 Overview of the progress in laser indirect drive experiments at NIF: total DT neutron yield as a function of ion temperature, red dots are doped HDC implosions, blue dots are High foot implosions, green dots are Low foot implosions. The neutron yield is plotted against the lowest burn averaged DT ion temperature measured by neutron time-of-flight (NTOF) detectors (Brysk temperature). HDC refers to the high density carbon ablator material that encapsulates the DT fuel and the terms High Foot/Low Foot refer to the shape of the laser pulse used to drive the compression. The black open diamond identifies the point where the energy deposited by α -particles would equal bremsstrahlung and conduction losses. Solid curves are yield extrapolations with temperature using a constant value of areal density, ρr , and adiabat. Reprinted with permission from Le Pape S, Berzak Hopkins LF, Divol L et al. (2018) Fusion energy output greater than the kinetic energy of an imploding shell at the National Ignition Facility. *Physical Review Letters* 120: 245003, <https://doi.org/10.1103/PhysRevLett.120.245003>, © 2021 American Physical Society.

The first part of the Section reviews progress and outstanding issues in MCF. Wade (2021) discusses the basic concepts underlying the realization of net fusion power release in magnetic confinement experiments, and Galvão and Canal (2021) discuss the range of magnetic configurations which are being studied in this area. Key theoretical concepts in the physics determining the behavior of toroidal magnetically confined plasmas are discussed in the following series of chapters (Sarazin, 2021; Jenko, 2021; Lao et al., 2021; Gorelenkov and Sharapov, 2021; Dumont, 2021), while the subsequent chapters review major experimental advances made in the research programs of devices exploiting the three principal magnetic field configurations: tokamaks (Zohm, 2021), stellarators (Yamada, 2021) and RFPs (Zuin, 2021). Considerations influencing the choice of technologies for the fusion core of reactor-scale devices, such as ITER, DEMO and fusion power plants, are discussed by Federici et al. (2020), while the technologies related to plasma ‘auxiliary’ systems such as heating and current drive, fueling, measurement systems and plasma control, are reviewed by Inoue et al. (2021). Critical issues in the development of materials capable of surviving in the demanding environment of a fusion reactor are presented by Litnovsky et al. (2021), and the R&D issues which are being addressed to resolve the major challenges in developing a reactor blanket to breed tritium and generate high-grade heat for electricity generation in fusion reactors are detailed by Boccaccini and Day (2021). Experimental facilities designed to take the MCF program forward to large-scale fusion energy production and to prepare the scientific and technological basis for fusion power plants are discussed by Donné et al. (2021), and design concepts for fusion power plants based on the major magnetic configurations are reviewed by Campbell (2021).

The second part of the Section is devoted to a review of progress in the ICF research program and the key challenges being addressed en route to establishing the basis for a power plant based on inertial confinement. Atzeni (2021) develops the basic concepts characterizing fusion performance in ICF compression experiments, outlining the direct- and indirect-drive concepts, while Tikhonchuk (2021) discusses the physics processes determining the evolution of the capsule compression in greater detail. Key experimental progress made using the principal driver concepts are summarized by Regan (2021), who reviews results from laser-driven experiments, by Seidl (2021), who discusses the development of the heavy ion beam driven approach, and by Gomez et al. (2021), who present advances in the Z-pinch driven approach, discussing both the conventional Z-pinch concept for capsule compression and the more recently developed magnetized liner inertial fusion (MagLIF) concept, which blends elements of magnetic and inertial confinement and which shows considerable potential for realizing significant fusion gain. The major facilities which form the core of the experimental program driving forward the ICF approach to fusion energy production are reviewed by Finnegan (2021), the key technologies, and associated challenges, necessary for the eventual implementation of ICF power plants are presented by Campbell et al. (2021), and a range of design concepts for ICF power plants, together with the issues that must be resolved to establish the necessary scientific and technological basis for their realization, are discussed by Dunne et al. (2021).

If the scientific and technological challenges that are currently the subject of the wide-ranging research programs in the magnetic and inertial confinement approaches to fusion energy can be successfully resolved in the coming decades, the benign safety and environmental characteristics of fusion power plants are expected to provide a strong basis for public acceptance of the technology and its large-scale implementation, assuming, of course, that fusion electricity is economically competitive in the energy market prevailing at the time at which the technology is mature. Taylor (2021) reviews the safety and environmental arguments in favor

of fusion energy and summarizes the experience in nuclear licensing gained through the regulatory procedure applied to the ITER facility in France.

Conclusions

The study of controlled thermonuclear fusion has been underway since the late 1940s and, as presented at greater length in the subsequent chapters of this Section, the intervening decades have yielded significant progress in understanding the behavior of plasmas under the conditions required for fusion energy production, in advancing the theoretical description of the processes determining the behavior of such plasmas, in developing the sophisticated technology required to produce and sustain plasmas at the temperatures and densities appropriate to achieving significant fusion gain, and in demonstrating the production of energy from DT reactions. Important challenges remain to be resolved to establish the scientific and technological basis for the next major step following the ITER/NIF generation of experiments, the construction and operation of a DEMO-class device capable of electricity generation using fusion power. However, the extensive understanding developed to date and the experimental capabilities inherent in the ITER and NIF facilities give confidence that, in parallel with the ongoing wide-ranging science and technology studies being conducted across the international research programs, a practical path exists toward the realization of a future generation of fusion power plants.

Complementing the major scientific and technological thrusts in the fusion research programs, the preparation of the socio-economic basis to support the case for public acceptance of fusion energy is underway. Collaboration with industry is also growing to ensure that an adequate industrial base for the construction of fusion power plants will be in place when required. These are essential measures, but they are complicated by the likely timescale for the implementation of fusion power plants – fusion road-maps developed to guide the future planning of the development program suggest that it is likely to be several decades before this key step is possible (see, e.g., [Donné et al., 2021](#)), though contributions by privately-funded R&D activities complementing the major national and international programs could help accelerate this timescale. Of course, as the technology for fusion electricity production matures, it must also be economically competitive, and this issue will certainly influence the timescale for its implementation. Nevertheless, the long-term and widely distributed fuel resources that are available for fusion energy production, together with the favorable safety and environmental characteristics of fusion energy, provide a significant motivation for the continued development of the science and technology toward the exploitation of fusion energy for the benefit of mankind.

See Also: Nuclear Fusion.

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Magnetic Confinement Fusion—Principles

Mickey Wade, Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Glossary

Aneutronic Lacking the generation of a neutron (typically high energy) as a by-product of the fusion process.

Aurora Borealis, Aurora Australis Spectacular light shows at the North and South Pole resulting from the interaction of high energy particles with the atmosphere in those regions. Sometimes referred to as the Northern or Southern Lights.

Ballooning Instability Rapid displacement of the plasma due to the local plasma pressure exceeding the restoring strength of the magnetic field line, akin to local stretching of a balloon by air pressure until it ruptures.

Bremsstrahlung radiation Radiation due to the acceleration of charged particles due to their Coulomb interaction with other charged particles.

Coulomb barrier Because the Coulomb force increases strongly as the bodies move closer together, an effective barrier is formed between bodies of opposite charge.

Coulomb force A force between charged bodies that acts to repel bodies of similar charge and attract bodies of opposite charge and increases in strength as the bodies come closer together.

Debye sphere The volume over which local electrostatic potential perturbations due to individual or clustered charged particles can be felt. Outside this volume, these perturbations are screened out by the plasma due to the mobility of the electrons and ions. In fusion plasmas, the Debye sphere is typically very small ($\sim 10^{-12} \text{ m}^3$).

Drift waves Small-scale plasma waves generated by local fluctuations in the plasma, akin to small-amplitude waves in a stream generated by eddies in the water flow.

Ignition Condition in which sufficient fusion power is generated that the system is fully self-sustaining, only requiring the injection of additional fuel. This is akin to the point at which fire in a wood-burning fireplace requires no further heat source to continue burning.

Impurity line radiation Radiation resulting from the de-excitation of electrons from higher to lower energy states within the atom.

Inertial confinement Confinement approach that relies on the inertia of imploding material to compress and maintain the fuel sufficiently hot enough for long enough for fusion to occur.

Kink Instability Rapid displacement of the plasma due to local twisting of the plasma column such that the magnetic field tension can no longer resist the twisting, akin to a rope that is twisted beyond its ability to maintain its stretched-out state (i.e., it starts knotting).

Magnetic confinement Confinement approach that relies on strong magnetic fields to confine and maintain the fuel sufficiently hot enough for long enough for fusion to occur.

Magnetohydrodynamic (MHD) A fluid description of the response and dynamics of plasmas in electromagnetic fields.

Maxwellian distribution A statistical description of the probability that a particle will have a particular energy in a system in thermodynamic equilibrium at a given temperature.

Neoclassical transport The theory of how particle orbits impact collisional transport, taking into account both the basic gyromotion of the particle around the magnetic field line and the motion due to large-scale changes in the magnetic field strength.

Nuclear force A very strong, but very localized, force that holds together the subatomic particles (neutrons and protons) of the nucleus.

Plasma The fourth state of matter in which the constituents are primarily ions and electrons.

Superconductivity The property of zero electrical resistance in some substances, typically achieved at very low absolute temperatures.

Synchrotron radiation Radiation emitted when charged particles are accelerated radially due to the Larmor force ($F = v \times B$).

Turbulence Physical phenomenon associated with rapid changes in the background conditions that have the potential of being highly chaotic (i.e., not smoothly or predictably varying in time or space).

Introduction

The fusion process is readily apparent in nature, serving as the source of energy for billions of stars, and continues to liberate copious amounts of energy from matter throughout the universe. Yet, the conditions for fusion reactions to occur in any significant number are extreme and therefore quite difficult to obtain and sustain in the laboratory (Campbell, 2021). The most fundamental constraint on the fusion reaction occurring is the requirement that the relative energy between the reacting ions be extremely high. To occur, fusion requires two ions to approach each other close enough that the (attractive) nuclear force overcomes the (repulsive) Coulomb force. This simplistic constraint would imply that the particles must approach each other with very high energies (~ 400 keV which is in excess of 3 billion degrees Kelvin). Detailed theory of the fusion process has revealed several physical mechanisms that act to reduce this requirement. Most importantly, ions have the ability to tunnel through the Coulomb barrier at lower energies, enabling fusion to occur at much more modest energies (< 5 keV), though the reaction rate at these energies is quite small. Furthermore, particles in a gas that is in thermal equilibrium have a range of velocities (referred to as a Maxwellian distribution), including high-energy particles that react more readily, leading to non-negligible fusion reactivity at quite modest temperatures (~ 1 keV).

For a variety of reasons, light (i.e., low atomic number) nuclei have the largest fusion reaction rates. Due to the Coulomb barrier discussed above, the charge of the nuclei inhibits the ability of the reactants to closely approach each other, leading to a much higher fusion probability of lower-charge ions than their higher Z brethren. In addition, the ability of an ion to tunnel through the Coulomb barrier decreases rapidly with increasing atomic number and mass, thereby favoring the lightest nuclei. Nuclear resonances affecting the formation of the fusion-produced ion also play a role, leading to large differences between similar isotopes. For example, the deuterium-tritium (D-T) reaction has a broad resonance at modest energies that increases the fusion reaction rate dramatically relative to its D-D or T-T counterparts.

For all these reasons, there are only a few reactions of interest for producing self-sustained fusion on earth. The suitability of any particular reaction is dependent on its intrinsic reactivity $\langle \sigma v \rangle_{fus}$, the total energy released in the fusion process E_{fus} , and the total energy released that can be utilized to self-heat the plasma, E_{fus}^{+-} . Table 1 provides information on each of these quantities along with some other parameters whose importance will become apparent in the subsequent discussion.

Self-sustainment, Lawson criterion, and triple product

The basic requirement for any fusion system to be self-sustaining was first developed by Lawson (1957) in which he asserted that self-sustainment can only occur if the heating power provided by fusion reactions exceeds the loss power (i.e., $P_{fus}^{heat} > P_{loss}$). The fusion power available for further plasma heating per unit volume is given by:

$$P_{fus}^{heat} = \frac{n_{i,1} n_{i,2}}{(1 + \delta_{12})} \langle \sigma v \rangle_{fus} E_{fus}^{+-} \quad (1)$$

Table 1 Physical quantities and fusion metrics for several fusion reactions of interest (data from Bosch and Hale, 1992 and Nevins and Swain, 2002).

		E_{fus}	E_{fus}^{+-}	ξ_{m0}	$\langle \sigma v \rangle_{fus, max}$	$T_{(\sigma v)_{max}}$	$n_e T_e \tau_{min}$	$T_e T_i \tau_{min}$
	%	(MeV)			($10^{-22} \text{ m}^{-3}/\text{s}$)	(keV)	($10^{21} \text{ m}^{-3} \bullet \text{keV} \bullet \text{s}$)	(keV)
D + D $\rightarrow$ T + H	0.5	4.0	4.0	4	~ 3.0	> 1000	1000	15
He ³ + n	0.5	3.3	0.8					
D + T $\rightarrow$ He ⁴ + n	1	17.6	3.5	8	10	65	3.0	13.6
D + He ³ $\rightarrow$ He ⁴ + p	1	18.3	18.3	15	2.8	250	48	58
H + B ¹¹ $\rightarrow$ 3He ⁴	1	8.7	8.7	32	4.6	550	1000	123

where $n_{i,1}$ and $n_{i,2}$ are the density of reacting ions, $\delta_{12} = 1$ if the reacting ions are identical (0 otherwise). Note that Eq. (1) assumes that the non-charged particles (primarily neutrons) readily escape the plasma and therefore do not further heat the plasma and hence do not contribute to the power balance.

The loss power per unit volume can be written simply as the energy loss rate from the confining volume:

$$P_{\text{loss}} = \frac{3}{2} \left[n_e T_e + (n_{i,1} + n_{i,2} + \sum n_{Z,k}) T_i \right] / \tau \quad (2)$$

where n_e is the electron density, T_e and T_i are the electron and ion temperature, respectively, $n_{Z,k}$ is the impurity density, and τ is characteristic time constant for energy loss (through all processes). Impurities consists of any non-reactant particles including any of the charged particle products of the fusion reaction itself and impurities eroded from the chamber walls or intentionally introduced for power exhaust control purposes. The dominant loss processes in fusion plasmas of interest are radiative losses (e.g., bremsstrahlung radiation, impurity line radiation) (Jensen et al., 1977), and transport losses. Synchrotron radiation can become an important, additional loss mechanism in cases with very high magnetic fields.

Further constraints are imposed by charge neutrality such that:

$$n_e = n_{i,1} Z_1 + n_{i,2} Z_2 + \sum n_{Z,k} Z_k \quad (3)$$

Here, Z_k is the charge of the respective ions and impurities. Applying this constraint, observing that P_{fus} is maximized when $n_{i,1} = n_e / 2Z_1$, $n_{i,2} = n_e / 2Z_2$ and $n_{Z,k} = 0$, and combining Eqs. (1) and (2) into the self-sustainment requirement $P_{\text{fus}}^{\text{heat}} > P_{\text{loss}}$ yields the so-called Lawson criterion:

$$n_e \tau > \frac{\frac{3}{2} \xi_m T_i}{\langle \sigma v \rangle_{\text{fus}} E_{\text{fus}}^{+, -}} \quad (4)$$

Here, ξ_m is a reaction-specific multiplier. For $n_{Z,k} = 0$, ξ_m is given by:

$$\xi_{m0} = [4f_T Z_1 Z_2 + 2(Z_1 + Z_2)] / (1 + \delta_{12}) \quad (5)$$

where $f_T = T_e / T_i$. Values of ξ_{m0} for various reactions are given in Table 1. For a pure, D-T plasma (i.e., $f_{Z,k} = 1$, $Z_1 = Z_2 = 1$ and $\delta_{n1} = n_2 = 0$) with equilibrated electron and ion temperatures (i.e., $T_e = T_i \rightarrow f_T = 1$), one finds that $\xi_{m0} = 8$ and:

$$n_e \tau|_{D-T} > 12 T_i / \langle \sigma v \rangle_{\text{fus}} E_{\text{fus}}^{+, -} \quad (6)$$

Because $\langle \sigma v \rangle_{\text{fus}}$ is solely dependent on temperature, the right-hand side of Eq. (6) is solely a function of T_i . The temperature dependence of $\langle \sigma v \rangle_{\text{fus}}$ therefore is critically important in establishing the minimum $n_e \tau$. In the intermediate temperature range, $\langle \sigma v \rangle_{\text{fus}}$ scales roughly as T_i^2 ; at lower T_i this dependence is even stronger. These together would imply it is favorable to achieve as high a T_i as possible. However, at higher T_i , $\langle \sigma v \rangle_{\text{fus}}$ scales much more weakly with T_i , and for some reactions that benefit from nuclear resonances at intermediate T_i , $\langle \sigma v \rangle_{\text{fus}}$ actually begins to decrease with increasing T_i . This translates into the minimum $n_e \tau$ occurring at an intermediate T_i for most reactions (e.g. at ~ 14 keV for the D-T reaction).

As will be discussed later, in many cases the maximum attainable pressure in magnetic confinement systems is limited by fundamental properties associated with magnetohydrodynamic (MHD) stability. In such cases, it is more appropriate to rearrange Eq. (4) to form the so-called triple product:

$$n_e T_i \tau > \frac{\frac{3}{2} \xi_m T_i^2}{\langle \sigma v \rangle_{\text{fus}} E_{\text{fus}}^{+, -}} \quad (7)$$

Note that for the D-T reaction with $f_T = 1$, Eq. (7) recovers the typical triple product formula found in the literature of $n_e T_i \tau|_{D-T} > 12 T_i^2 / \langle \sigma v \rangle_{\text{fus}} E_{\text{fus}}^{+, -}$. As an aside, the fusion reactivity for D-T can be approximated in the temperature range of interest as $\langle \sigma v \rangle_{\text{fus}, DT} \approx 1.1 \times 10^{-24} T_i^2$.

Assuming $T_e = T_i$, the fusion triple product given by Eq. (7) for the reactions identified above are shown in Fig. 1a. As anticipated from the discussion on fusion reactivity, the reaction with the most favorable (i.e., lowest) $n_e T_i \tau$ is the D-T reaction with the minimum value $n_e T_i \tau \approx 3 \times 10^{21} \text{ m}^{-3} \cdot \text{keV} \cdot \text{s}$ occurring at $T_i \approx 14$ keV. The minimum $n_e T_i \tau$ for the D-He³ and p-B¹¹ reactions are orders of magnitude higher and occur at significantly higher temperatures. Therefore, while these reactions offer the potential advantage of being aneutronic (eliminating issues related to the damage caused by high energy neutrons), achieving the conditions necessary for a sustained fusion reaction are significantly more challenging.

Impressive progress was made in achieving the necessary $n_e T_i \tau$ in tokamak configurations in the late 1990s, as is depicted in Fig. 2. From the mid-1960s to the mid-1990s, the achieved $n_e T_i \tau$ doubled approximately every 1.8 years, slightly faster than the rate of progress described by Moore's law for semiconductor development (number of transistors on a chip doubled every 2 years). This impressive rate of progress was achieved through the deployment of new facilities with larger plasma size, higher magnetic field, and increased heating power. The next step in this evolution is ITER, a multi-national facility specifically designed to demonstrate sustained operation with $n_e T_i \tau$ values that are needed for future fusion reactors.

It's important to realize that Eq. (5) assumes no fuel dilution due to impurities (i.e., $n_{Z,k} = 0$). If impurities are included in the evaluation of the required $n_e T_i \tau$ (i.e., assuming $n_{Z,k} \neq 0$), the resulting contamination rapidly poses serious challenges to achieving the necessary conditions due to both dilution and increased radiation losses. Even small levels of impurities can cause significant

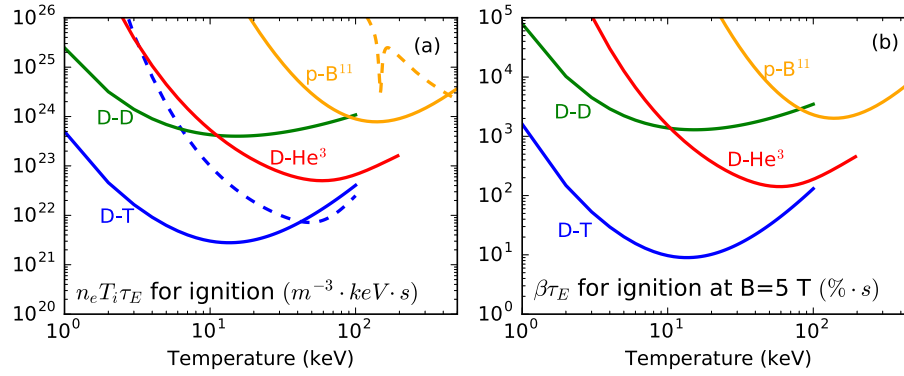


Fig. 1 (a) $n_e T_i \tau$ and (b) $\beta \tau$ for fusion reactions of interest assuming $n_{i,1} = n_e/2Z_1$, $n_{i,2} = n_e/2Z_2$ and $n_{Z,k} = 0$. Dashed lines represent $n_e T_i \tau$ substituting $\sigma_{fus} \nu$ for $\langle \sigma v \rangle_{fus}$ in Eq. (7). $\beta = p(B^2/2\mu_0)$ is the ratio of the kinetic pressure to the applied magnetic pressure.

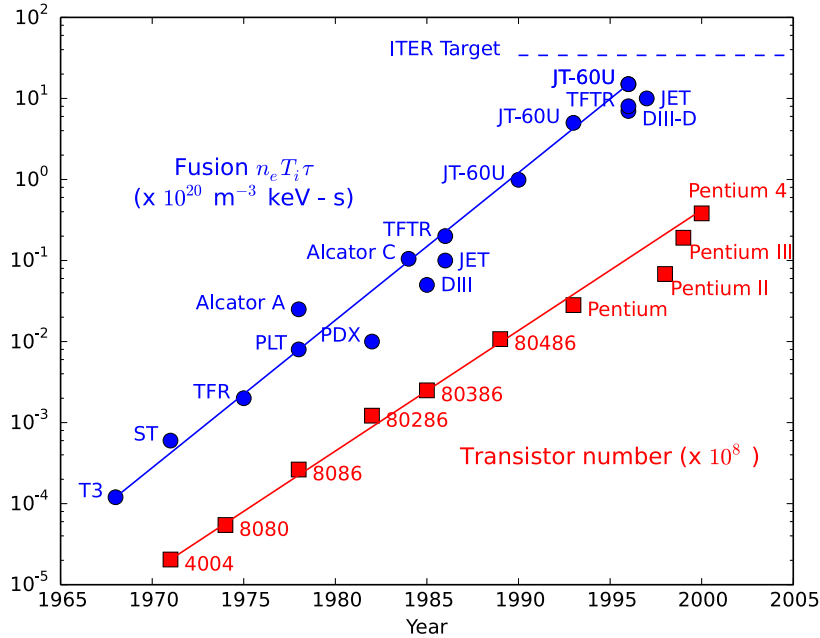


Fig. 2 Rate of progress in achieved triple product $n_e T_i \tau$ in tokamak facilities exceeded that of semiconductor development in the late 1990s.

increases in ξ_m through dilutive effects, and by extension, the minimum $n_e \tau$ needed for self-sustainment. An example of the impact on ξ_m is shown in Fig. 3 for helium (a typical fusion product), argon (used for power exhaust control), and tungsten (chamber wall material). For the D-T reaction, a 50% increase in ξ_m from its base level ($\xi_m = 12$ vs $\xi_{m0} = 8$) occurs at $f_{\text{Helium}} = 10\%$, $f_{\text{Argon}} = 1.3\%$, and $f_{\text{Tungsten}} = 0.4\%$. This situation is further exacerbated by the fact that higher Z impurities cause substantial increases in both the bremsstrahlung and line radiation, thereby reducing the effective confinement time τ and making it more difficult to achieve the necessary $n_e T_i \tau$. Hence, the practical limit on impurity levels is much lower than those listed above.

It should be stressed that Eqs. (4) and (7) represent fundamental requirements for self-sustainment of any fusion system. Because of the highly non-linear nature of these parameters and their large variation (many orders of magnitude), approaches that target operating conditions that are off the optimal temperature must deliver significant performance improvements to enable sustained fusion operation. In this respect, while D-T fusion systems with temperatures less than 10 keV are possible, it seems likely that the practical limit for realistic thermonuclear fusion systems will be with $T_i > 10$ keV.

A possible approach to reduce these requirements is to produce an energy distribution of particles that maximizes the fusion reactivity. This is typically referred to as a non-Maxwellian energy distribution in which particles are driven to a particular energy range and not allowed to thermalize within the system. As noted in the previous section, a modest increase in reactivity is obtained. However, one finds that if one simply substitutes $\sigma_{fus} \nu$ (monoenergetic) for $\langle \sigma v \rangle_{fus}$ in Eq. (7) that the minimum value of $n_e T_i \tau$ (where T_i now represents the energy of the population) actually increases and moves to higher energy, with the minimum occurring near the peak in σ_{fus} . This is shown for the D-T and p-B¹¹ reactions as the dashed curves in Fig. 1a. Hence, while counter-intuitive at first thought, the Maxwellian distribution actually improves access to self-sustained fusion conditions due to the non-negligible fusion cross section at lower energies.

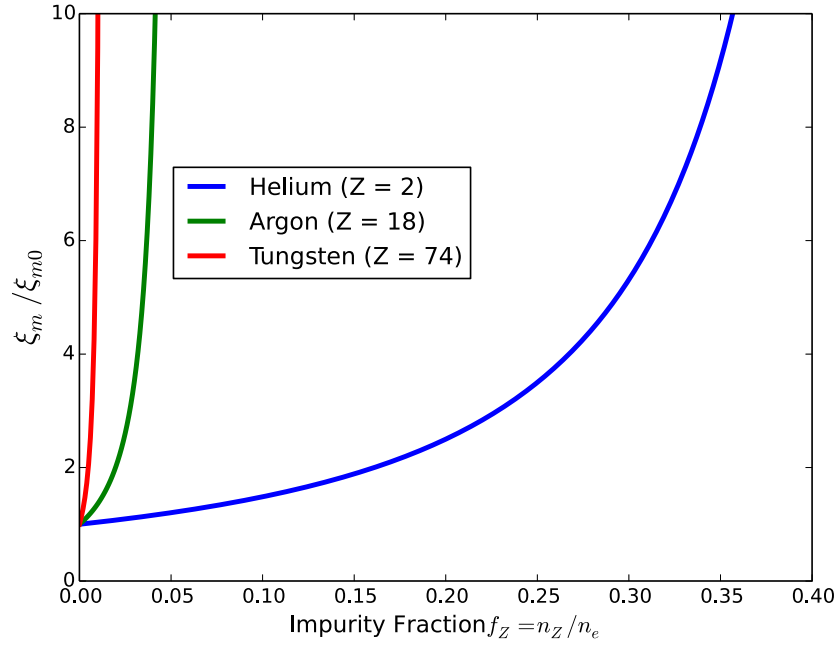


Fig. 3 Impact of increasing the impurity fraction for helium ($Z = 2$), argon ($Z = 18$), and tungsten ($Z = 74$) on the reactivity-specific multiplier ξ_m relative to its value with $f_Z = 0$ (DT reaction).

Practical fusion energy

Note that the above discussion assumes that the heating provided by the fusion process alone is sufficient to maintain the required plasma conditions (i.e., $P_{\text{heat}} = P_{\text{fus}}$). This condition is known as “ignition” and will be self-sustaining as long as sufficient fuel is supplied. The requirement on $n_e T_i \tau$ can be relaxed somewhat by applying external heating to supplement the fusion heating (i.e., $P_{\text{heat}} = P_{\text{fus}}^{\text{fus}} + P_{\text{heat}}^{\text{ext}}$). While the use of external heating will reduce the overall efficiency of the fusion system, external heating can be beneficial (and possibly required) in delivering important control capabilities (e.g., burn control, current drive).

Defining the factor $\gamma_{\text{fus}} = E_{\text{fus}}/E_{\text{fus}}^{+,-}$, and representing the external heating as a fraction of the fusion power $P_{\text{heat}}^{\text{ext}} = P_{\text{fus}}/Q_{\text{fus}}$, the triple product requirement can be rewritten as:

$$n_e T_i \tau > \frac{\frac{3}{2} \epsilon_m T_i^2}{\langle \sigma v \rangle_{\text{fus}} E_{\text{fus}}^{+,-}} \left(\frac{Q_{\text{fus}}}{Q_{\text{fus}} + \gamma_{\text{fus}}} \right) \quad (8)$$

The parameter Q_{fus} is generally referred to as the fusion gain since it represents the fusion power produced for a given input power. Note that $Q_{\text{fus}} = \infty$ (or what is commonly referred to as the ignition condition) recovers Eq. (4). As Q_{fus} is reduced and becomes comparable in magnitude to γ_{fus} , the $n_e T_i \tau$ required for sustainment is reduced. For example, for the D-T reaction with $\gamma_{\text{fus}} = 5$, $Q_{\text{fus}} = 10$ results in a 33% reduction in the $n_e T_i \tau$ requirement while $Q_{\text{fus}} = 5$ yields a 50% reduction.

While Eq. (8) suggests that lowering Q_{fus} is advantageous in lowering the $n_e T_i \tau$ requirement, one must be aware that there are practical limits on the minimum value of Q_{fus} if the goal is to produce net electricity. Assuming that the system is able to recover all the thermal power generated by the fusion process, the net electricity $P_{\text{electricity}}$ produced by the system can be represented as:

$$P_{\text{electricity}} = P_{\text{fus}} \eta_{\text{th}} M_{\text{bl}} - \frac{P_{\text{fus}}}{\eta_{\text{plug} \rightarrow \text{plasma}} Q_{\text{fus}}} - P_{\text{BOP}} \quad (9)$$

Here, η_{th} is the thermal efficiency for converting fusion-produced heat to electricity, M_{bl} is a power multiplier due to exothermic reactions as the neutrons are captured in an external blanket, $\eta_{\text{plug} \rightarrow \text{plasma}}$ is the plug-to-plasma efficiency for delivering external heating to the plasma, and P_{BOP} is the power required to operate the balance of plant (i.e., cooling systems, cryogenic systems, etc.). An example of the impact of Q_{fus} on the attainable net electricity for a fusion system producing 1 GW of thermal fusion power (i.e., $P_{\text{fus}} = 1$ GW) is shown in Fig. 4 for various assumptions of η_{th} and $\eta_{\text{plug} \rightarrow \text{plasma}}$ with $M_{\text{bl}} = 1.2$ and $P_{\text{BOP}} = 50$ MW. It is evident that $P_{\text{electricity}}$ drops precipitously for $Q_{\text{fus}} < 15$ in all cases and that there is no significant gain in $P_{\text{electricity}}$ for $Q_{\text{fus}} > 40$ in any of the cases.

Assuming $15 < Q_{\text{fus}} < 40$ is desirable for practical fusion energy, recognizing that the minimum $n_e T_i \tau$ occurs at 14 keV, and using the approximate expression for $\langle \sigma v \rangle_{\text{fus,DT}}$ introduced earlier, practical fusion energy requires achieving the criterion:

$$1.7 \lesssim n_e \tau|_{D-T} (\cdot 10^{20} \text{ m}^{-3} \cdot \text{s}) \lesssim 2.0 \text{ at } T_i = 14 \text{ keV} \quad (10)$$

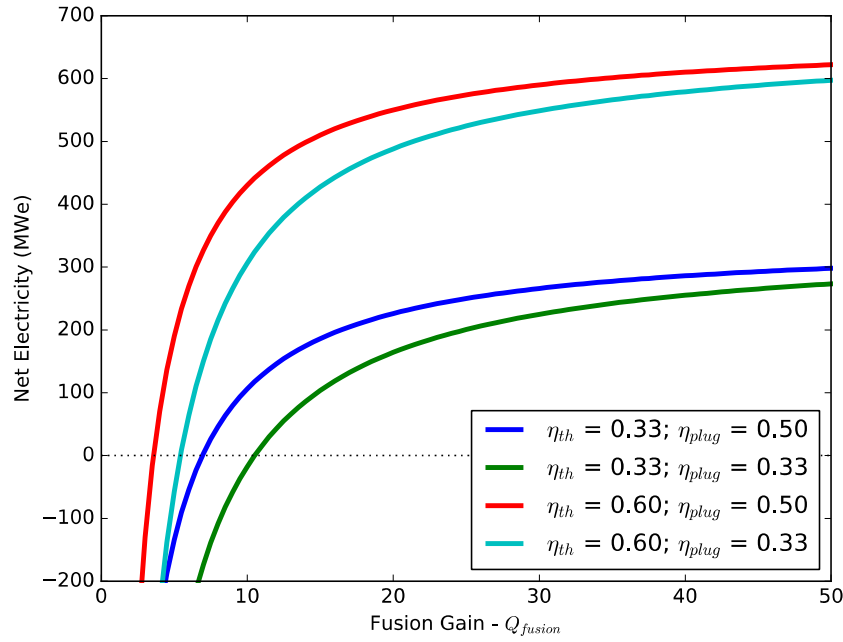


Fig. 4 Anticipated net electricity production for a 1 GW thermal fusion facility for various assumptions of thermal efficiency η_{th} and plug-to-plasma efficiency $\eta_{plug \rightarrow plasma}$ and $M_{bl} = 1.2$ and $P_{BoP} = 50$ MW.

As is evident from Fig. 1a, slightly off-optimum temperatures will result in a very rapid increase in this criterion.

Different confinement approaches seek to achieve the criterion in Eq. (10) via different approaches, employing a wide range of possible density values. For example, inertial confinement uses high-energy lasers to compress a small pellet to densities approaching 10^{31} m^{-3} , thereby requiring only picosecond-scale confinement times. In contrast, magnetic confinement uses strong magnetic fields to confine the plasma to achieve densities in the range of 10^{20} m^{-3} , leading to confinement requirements on the order of seconds. Other approaches lie in between these extremes.

It should be noted that the equations above assume thermal equilibrium conditions (i.e., systems with Maxwellian distributions of the reacting ions) with equal ion and electron temperatures (i.e., $T_e = T_i$). Note also that the condition in Eq. (10) can, in principle, be relaxed if techniques are employed that drive strongly non-Maxwellian distributions in the reacting ions or utilize systems with $T_i \gg T_e$. While such techniques can significantly increase the average fusion reactivity and thereby reduce the $n_e T_i \tau$ requirement, experience to date has shown that the external power required to produce such distributions is typically quite high, resulting in low Q_{fus} values that are impractical for fusion electricity production.

Magnetic confinement

As noted above, there are multiple potential approaches to achieving practical fusion energy. The most advanced in terms of achieving the necessary conditions is magnetic confinement, which takes advantage of the features of the very hot gas as it approaches fusion conditions. At temperatures relevant to thermonuclear fusion ($> 10 \text{ keV}$), most of the atoms in the system will be ionized to some extent, with the low-charge (i.e., low-Z) atoms fully stripped. As an example, the ionization potential of hydrogen and helium is 13.6 and 24.6 eV, respectively. The ion/electron bath that results is called a plasma, the fourth state of matter. In many respects, the plasma behaves similarly to a gas, diffusing along gradients of density and temperature or through turbulent eddies generated by irregularities in the plasma characteristics.

Deuterium atoms at 10 keV have an average velocity of nearly 700,000 m per second (over 1,500,000 miles per hour). Because of this mobility, effective confinement times without any confining mechanisms are quite short, leading to very low values of the plasma confinement τ and a much more severe requirement on the necessary plasma pressure $n_e T_i$. As importantly, any contact of the very hot plasma with the walls of the confinement chamber will cause rapid cooling of the plasma, thereby eliminating access to fusion-like temperatures. Hence, standard confinement methods as might be used for high pressure gases are not possible.

Fortunately, nature has provided a solution to this challenge in the form of the Lorentz force that acts on charged particles moving in a magnetic field. This force is given by (Jackson, 1975):

$$\vec{F}_{+/-} = \vec{v} \times \vec{B} \quad (11)$$

where the $\times$ operator indicates the cross product. As depicted in Fig. 5, this force causes the charged particle to accelerate in a direction both perpendicular to its own motion $\vec{v}$ and that of the magnetic field $\vec{B}$ it is passing through.

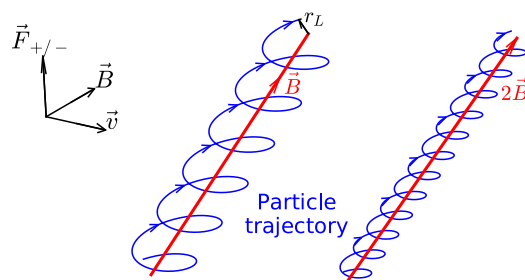


Fig. 5 Trajectory (blue) of a charged particle in the presence of a magnetic field $\vec{B}$ (red) due to the Lorentz force. As the magnetic field is doubled (right hand figure), the Larmor radius r_L decreases by a factor of two.

Because the force acts perpendicularly to the magnetic field line, the velocity in the direction of $\vec{B}$ is unchanged (i.e., $v_{\parallel}$ is unchanged). In the perpendicular direction, $\vec{F}_{+/-}$ is a centripetal (i.e., acting radially inward) force, producing a circular motion of the particle around the field line. The net effect is a spiral-like motion with the charged particle's trajectory being tied closely to that of the trajectory of the magnetic field line. Such a trajectory is shown in Fig. 5. The radial extent of the particle motion is inversely proportional to the magnetic field strength ($r_L = mv_{\perp}/ZeB$); hence, the stronger the magnetic field the more closely the particle's trajectory is tied to that of the magnetic field line.

This effect is the reason why the Aurora Borealis and Aurora Australis are only evident in polar regions as high-energy electrons ejected by the sun encounter the magnetic field lines generated by the Earth and are directed along those field lines to the polar regions. There, they collide with the Earth's atmosphere, generating fascinating light spectacles.

Magnetic fusion confinement systems attempt to generate sufficient confinement by taking advantage of this feature by devising magnetic field configurations that direct particles away from the walls of the confinement chamber. These configurations are typically produced by strong electromagnetic coils located outside the confinement chamber, capable of producing magnetic field strengths in the several Tesla range. For reference, the Earth's magnetic field is $< 10^{-4}$ T. Two types of magnetic confinement systems have been developed and tested—closed and open. The basics of each concept are described next.

Toroidal (closed) confinement

Closed systems attempt to close the magnetic field lines on themselves, typically using a doughnut-like (i.e., toroidal) geometry—though other geometries have been conceived and tested. In its simplest form conceptually, such a system is configured such that the magnetic field lines travel around the doughnut (i.e., torus) the long way, directing particle trajectories in a similar manner. Practically, several physics considerations impose a requirement that the magnetic field lines must also wrap around the torus the short way, creating a magnetic field structure that wraps the short way around the torus (known as the poloidal direction) as it traverses the full circumference of the torus (known as the toroidal direction).

The “pitch” of the field line as it makes this transit impacts a number of physics effects. As with a rope, increasing the twist (i.e., higher pitch) of the magnetic field lines increases the strength of the magnetic field bundle, enabling the confinement of plasmas with higher pressure. Accordingly, the variation of the magnetic pitch is a primary distinguishing feature of the range of closed magnetic confinement approaches (Galvao, 2020). Some approaches have low pitch throughout, others have low pitch in the center of the plasma rising towards the edge, others have high pitch in the center decreasing towards the edge, while others have flat pitch profiles.

Many, many toroidal confinement concepts have been evaluated throughout the history of magnetic fusion. The most advanced of these is the tokamak, originally developed in Russia and shown subsequently to have very good plasma confinement properties. The tokamak magnetic field structure is generated by a set of toroidal field magnets that create a strong toroidal field and a toroidal plasma current that generates the poloidal magnetic field. The toroidal field is typically much larger than the poloidal magnetic field with a pitch profile that increases from the core to the edge. The spherical tokamak is a variation of the standard tokamak with similar physics features.

The next most advanced configuration is the stellarator with a magnetic configuration that is defined by a set of 3D magnetic coils surrounding the device. There are many different forms of stellarator geometries but they are all primarily determined by the choice of magnetic coils (i.e., very little additional magnetic field is provided by internal plasma currents). This variability leads to many different magnetic pitch profiles, each having benefits as well as disadvantages.

Other toroidal confinement systems include:

- The spheromak has a very similar magnetic geometry as a tokamak but arrives at this state through dynamic formation of the plasma that drive toroidal and poloidal currents to form the magnetic field structure. This eliminates the need for toroidal magnetic field coils.
- The reversed field pinch is similar to tokamak magnetic geometry but with a much smaller toroidal field. As with the spheromak, these fields are primarily generated by currents flowing in the plasma. The driven toroidal plasma current is of sufficient strength that the toroidal magnetic field reverses direction, leading to a significantly different pitch profile from tokamaks or spheromaks.

Linear (Open) confinement systems

Open magnetic confinement systems take advantage of an additional feature of charged particle motion in magnetic fields. Because the Lorentz force is a centripetal force (i.e., the force only works in the radial direction), the angular momentum ($L = mv_{\perp}R$) is conserved. Using the expression for the Larmor radius (i.e., $r_L = mv_{\perp}/ZeB$), the perpendicular kinetic energy can be expressed as $E_{\perp} = mv_{\perp}^2/2 = \mu B$ where $\mu = ZeL/2m$. Since the kinetic energy ($KE = mv_{\parallel}^2/2 + mv_{\perp}^2/2$) of the particle is also conserved, the velocity along the field line can then be written as (Post, 2011):

$$v_{\parallel}^2(s) = \frac{2}{m}[KE - \mu B(s)] \quad (12)$$

Here, s represents the distance along a field line. Hence, as a particle moves into a region of stronger magnetic field, the velocity in the direction of the field line $v_{\parallel}$ must decrease and if the magnetic field becomes large enough, the particle will be reflected (i.e., $v_{\parallel}$ changes sign).

Taking advantage of this feature, linear confinement systems rely on this “mirror” effect to confine plasmas in configurations in which the magnetic field lines do not close on themselves. As one might expect from Eq. (12), not all particles are confined. The exact set of confined particles is given by $v_{\perp}/v > \sqrt{1/\rho_{\text{mirror}}}$. Here $\rho_{\text{mirror}} = B_{\text{max}}/B_{\text{min}}$ is the mirror ratio and B_{max} and B_{min} are the maximum and minimum magnetic field strength, respectively. Hence, by increasing ρ_{mirror} , one can increase the fraction of particles that are confined. An interesting feature of linear systems is that this confined fraction does not change as the plasma becomes hotter, in principle allowing operation up to fusion-relevant temperatures.

Similar to their toroidal confinement counterparts, many types of linear systems have been conceptualized and studied. The most notable of these are:

- Mirror devices that utilize strong magnetic field coils at either end of a cylinder to confine the plasma. Because of free-streaming losses out the ends of the cylinder, some variations use complicated magnetic field structures in these regions to improve overall confinement.
- Field-reverse configurations (FRCs) address the end-loss issue by dynamically generating internal currents in the plasma that results in regions of closed field lines, essentially producing a toroidal plasma within a linear system.
- Magnetic cusps use strong magnetic field coils to generate magnetic wells to confine the electrons and rely on electrostatic confinement of the ions within these wells.

To date, none of these have delivered performance on par with tokamaks or stellarators.

Stability, confinement, and fusion performance

The maximum pressure that can be sustained in a magnetic confinement system is dependent on the applied magnetic field with the ratio of the kinetic pressure to the magnetic field pressure:

$$\beta = \frac{\text{Kinetic pressure}}{\text{Magnetic pressure}} = \frac{n_e T_e + n_{i,1} T_i + n_{i,2} T_i}{B^2/2\mu_0} \quad (13)$$

serving as a critical parameter in the stability of the system. The attainable β depends on details of the particular magnetic configuration and can range from a few % in some confinement systems to near unity in others.

Revisiting the expression for sustaining a fusion plasma, one finds that the self-sustainment expression can be written as:

$$\beta \tau (\% \cdot s) > \frac{3\mu_0 T_i^2 \varepsilon_m^2 (1 + \delta_{12})}{4 \langle \sigma v \rangle_{fus} E_{fus}^{+-} B^2 Z_1 Z_2} \left(\frac{Q_{fus}}{Q_{fus} + \gamma_{fus}} \right) * 100 \quad (14)$$

Immediately, one can see that increasing the magnetic field strength is very beneficial in reducing the $\beta \tau$ requirement. Eq. (14) also offers other possibilities for reducing the required $\beta \tau$ through operation with $T_i \gg T_e$ (i.e., $f_T \ll 1$) or increasing $\langle \sigma v \rangle_{fus}$ through tailoring of the ion distribution (i.e., driving ions to high energies so that they are more likely to fuse).

It should be realized that β is constrained by the magnetic configuration in consideration. Although the β constraint does not introduce severe demands on systems utilizing the D-T reaction, it does introduce challenges for other fusion reactions. As shown in Fig. 1b, the minimum $\beta \tau$ for the D-T reaction is approximately 10%-s at 5 T, decreasing to 2.5%-s at 10 T. The situation is considerably different for the other fusion reactions, with the minimum $\beta \tau$ being above 100%-s for D-He³ and 1000%-s for p-B¹¹ at 5 T. For these reactions, systems that are limited to $\beta < 5\%$ (or even $\beta < 10\%$) will have stringent (and possibly impractical) demands on the confinement requirement.

Looking more specifically at the D-T reaction ($Z_1 = Z_2 = 1$, $\gamma_{fus} = 5$, $E_{fus}^{+-} = 3.5 \text{ MeV}$), assuming $T_i = T_e$, and using the expression introduced above for $\langle \sigma v \rangle_{fus}$ one finds:

$$\beta \tau (\% \cdot s)|_{DT} \gtrsim \frac{250}{B^2} \left(\frac{Q_{fus}}{Q_{fus} + 5} \right) \quad (15)$$

This equation is very useful in defining the challenge for fusion energy production from the plasma performance perspective as the right-hand side contains the output requirements (Q_{fus}) and technological constraints (B) of the system while the left-hand side only involves plasma performance metrics (β , τ). Several tradeoffs are evident. Lower fusion gain requirements or higher magnetic field can reduce the physics performance requirements. In contrast, higher physics performance could reduce the demands on the necessary magnetic field and enable higher values of fusion gain.

The various confinement systems outlined above attempt to exploit their intrinsic advantages to satisfy this requirement. For example, tokamaks take advantage of intrinsically higher confinement than other approaches (but suffer somewhat due to the minimum heating power set by the need drive the plasma current by external sources). Stellarators take advantage of a reduced need for external current drive power, thereby increasing the attainable Q_{fus} , reversed field pinch take advantage of a reduced requirement for magnetic field strength B and access to high β , etc. It is also important to note that all of the parameters in Eq. (15) have practical limits (e.g., β cannot exceed unity, τ cannot be arbitrarily high, B is limited by magnet technology, and Q_{fus} must be high enough to guarantee net electricity output).

In fact, the available combinations of β , τ , B , and Q_{fus} is somewhat limited. Because any improvement in the controllable stability (β) and confinement (τ) will open up the attainable operating space, we will take a look at these in the next few sections, followed by a brief section on the attainable magnetic field.

Plasma stability

As noted above, the plasma β is a critical parameter in determining the sustainable plasma pressure of a magnetic confinement system. This arises from the combination of the lowest order momentum balance equation:

$$\nabla p = J \times B \quad (16)$$

and Ampere's Law:

$$\mu_0 J = \nabla \times B \quad (17)$$

which upon combining and some arithmetic becomes:

$$\nabla(p + B^2/2\mu_0) \equiv \nabla(B^2/2\mu_0(1 + \beta)) = \frac{1}{\mu_0}(B \cdot \nabla)B \quad (18)$$

Magnetic confinement approaches (in which the magnetic field is utilized to confine the plasma and its associated pressure) have $B^2/2\mu_0 > p$ (or equivalently $\beta < 1$). More typically, $\beta \ll 1$. In Eq. (18), ∇p represents the kinetic pressure gradient, $\nabla B^2/2\mu_0$ represents the magnetic pressure gradient, and the RHS is the magnetic field tension force that acts to straighten the magnetic field lines (akin to the force on an arrow when a bow is stretched).

Note that even at zero pressure (or equivalently $\beta = 0$), there can be a magnetic field tension force depending on the choice of magnetic configuration. As ∇p is increased, the tension force increases due to the plasma pushing on the magnetic field lines. If the magnetic configuration can handle this increase, the system is stable. However, if ∇p increases beyond a point in which the system can supply additional magnetic tension, the magnetic field lines can rupture, causing rapid loss of plasma confinement. This is akin to pulling the string on a bow too far such that it breaks.

The maximum β a system can achieve therefore measures the relative amount of additional tension due to kinetic pressure that a system can sustain before loss of stability. Note that because $\beta \propto 1/B^2$, systems with higher field that are otherwise identical can sustain much larger increases in kinetic pressure.

As noted above, there are three main types of instability limit that govern the envelope of attainable β in magnetic confinement systems. These include the equilibrium limit, the "ballooning" instability limit, and the "kink" instability limit. The equilibrium limit is fundamental to a confinement system and determines whether or not the magnetic configuration itself is fundamentally stable or not. Typically, the equilibrium limit is not overly sensitive to β but this stability requirement must always be met at all values of β .

The ballooning instability limit occurs when ∇p increases to a point that the total pressure gradient locally exceeds the restoring force of the magnetic field lines. The name comes from the finger-like structures that form during the instability with features similar to the elongations formed in a balloon when it is squeezed. Ballooning instabilities are typically small in spatial scale, and therefore do not cause a complete loss of the confined plasma. However, they do reduce confinement (sometimes significantly), thereby limiting both the attainable β and τ . More details on this type of instability can be found in (Lao, 2021).

The kink instability limit, at its most fundamental level, occurs when a plasma column is distorted in such a way that magnetic pressure increases on one side of the column and decreases on the other, further amplifying the distortion. In essence, the tension on the RHS of Eq. (18) increases on one side of the column (becoming more stable) but decreases on the other side of the column (becoming more unstable), leading to an instability. This is similar to what occurs when twisting a rubber band to the point and suddenly knots start to occur due to the inability of the rubber bands to maintain the original structure. These instabilities are generally fairly large in spatial scale and therefore pose a severe risk to overall plasma operation. As such, this instability limit

generally serves as the primary limit on the attainable/sustainable β in many toroidal systems. For some devices, this limit is very low (e.g., $\beta < 1\%$ in linear devices), in others modest (e.g., $\beta < 10\%$ in tokamaks/stellarators), and others very high (e.g., $\beta \sim 100\%$ in reversed field pinches). More details on this type of instability can be found in (Lao, 2021).

Plasma transport and confinement

As noted in the introduction to this section, collision-less ions and electrons are “confined” to magnetic field lines through the Lorentz force, resulting in orbital motion with a characteristic gyro-frequency $\Omega = ZeB/m$ and radius $r_L = mv_{\perp}/ZeB$ with any transport off these field lines occurring only due to second order effects. These second order effects can have non-trivial consequences. For example, in a purely toroidal magnetic field configuration (i.e., no pitch of the field lines), particles will inevitably drift upward/downward depending on their charge. This effect is one of the primary reasons pitch of the field lines is required in any toroidal confinement device.

Collisions are another source of transport. Collisions can potentially knock particles off their collisionless orbits. While still generally confined on another field line and associated orbit, these collisions lead to effective transport.

Bohm transport

The degree of this effect can be estimated using the general formula for diffusion $D = \Delta x^2 / \Delta t$ where Δx is the characteristic step size due to collisions and Δt is the characteristic time between collisions. If one assumes that the step size is the distance a thermal particle would travel during a single particle orbit, $\Delta x = v_{th}/\Omega$ where $v_{th} \propto (T/m)^{1/2}$ is the thermal velocity and that the characteristic time step is given by the inverse gyro-frequency $\Delta t = 1/\Omega$, one arrives at the so-called Bohm diffusion coefficient with the sensitivities:

$$D_{Bohm} \propto \frac{T}{B} \quad (19)$$

Generally, one finds that projections with D_{Bohm} are very pessimistic for achieving sufficient conditions for fusion production due to the strong temperature dependence. Fortunately, experimentally observed transport rates in toroidal confinement systems are much lower than those predicted by D_{Bohm} . This overestimate is due to the fact that the collision time is much larger than the time for a particle to complete a single orbit (i.e., $\Delta t_{coll} \gg 1/\Omega$).

Classical transport

The “classical” diffusion rate can be estimated by assuming that the characteristic time step is the inverse collision rate $\Delta t = 1/\nu_{coll}$. Here, $\nu_{coll} \propto n/T^{3/2}$ is the collision frequency. These simple assumptions lead to a “classical” diffusion coefficient with the sensitivities:

$$D_{cl} \propto \frac{n}{B^2 T^{1/2}} \quad (20)$$

The inverse dependences in B and T here provided researchers with early hope on the prospects for fusion energy. However, experiments soon demonstrated that D_{cl} significantly underestimated the observed transport. This is primarily due to the fact that the assumed spatial excursions in this model only include first order orbit effects (i.e., motion around the field line) and neglect important second order orbit effects as well as collective transport due to turbulence and other anomalous effects. These effects are explored next.

“Neoclassical” transport

As noted previously, all toroidal confinement systems have the feature that a magnetic field line has a trajectory that travels both poloidally and toroidally. A further standard feature of toroidal confinement systems is that the toroidal magnetic field strength varies as $B = 1/R$. Hence, a particle will experience a range of magnetic field strengths as it follows the field line. Due to the trapping effect introduced earlier, this can lead to certain particles being reflected (i.e., $v_{\parallel}$ changing sign). When this occurs, the “trapped” particle is forced to make an excursion off that field line, thereby increasing the characteristic step size Δx . Consideration of this new type of orbit requires detailed calculations of the orbit classes and evaluation of the anticipated excursions of those orbit classes in the time between collisions. These effects are called neoclassical transport. The parametric dependence of D_{neo} derived within neoclassical transport analysis is a fairly complex set of formulae dependent on the magnetic topology and collisionality of the particular system involved. Although these calculations typically predict $D_{neo} > D_{cl}$ and hence resolve some of differences with experimental observations, detailed comparisons with neoclassical theory have generally shown that D_{neo} also underestimates observed transport rates, though in some cases consistency has been found.

Turbulent transport

Another effect that increases the effective transport step Δx is turbulent-driven transport. As in any fluid, the flow of energy and particles through the system (and the associated gradients), if large enough, has the potential to produce turbulence. This turbulence can drive transport through correlated fluctuations of the plasma density, temperature, electrostatic potential, and magnetic fields. For most cases to date, the primary turbulence mechanism is due to electrostatic fluctuations and is referred to as drift-wave turbulence. Mathematically, drift waves are small-scale-length solutions to the dispersion relation for waves propagating in a magnetic confinement system. These waves introduce period variations in the electrostatic potential (or equivalently the electric field E) and an associated periodic variation in the drift velocity $v_{dr} = E \times B/B^2$, leading to circulating, small-scale turbulent cells in the plasma.

Turbulent modes take many different forms. Some are driven by the density gradient, others by the temperature gradient. Some primarily cause particle transport, others energy transport. Some primarily affect electron transport, others ion transport, and still others both. Some act on very small spatial scales, others on extended spatial scales. Therefore, the net effect on overall transport is difficult to parameterize easily.

This is further complicated by the fact that these turbulent structures can be sheared apart by a variety of dynamics. If the background plasma is moving at different speeds on one side of the structure relative to the other side, the turbulence can be sheared apart, thereby reducing the net transport. A similar effect can happen if the field line pitch varies across the turbulent eddy. A further complication is that the turbulence itself can generate flows that act to tear apart the turbulence. Hence, predicting the level of transport due to turbulence, characterized by D_{turb} , requires sophisticated numerical models that capture the generation of the turbulence, the generation of flows by the turbulence, and finally taking all this into consideration, the resultant magnitude of the turbulence-driven transport. These models predict transport rates, D_{turb} , that are, in general, significantly above D_{neo} but much smaller than D_{Bohm} . Many experiments have shown consistency between the predicted transport rates and those observed as well as consistency with measurements of the underlying turbulence levels. A much more comprehensive overview of the turbulent-driven transport can be found in (Jenko, 2021).

Ultimately, the effective transport is a combination of both collisional and turbulent transport. Because these effects tend to occur on different spatial scales, the overall transport rate can be written as $D_{tot} = D_{neo} + D_{turb}$. In most cases, $D_{turb} \gg D_{neo}$; however, in cases where turbulence is suppressed through the means described above, D_{neo} provides a floor on the overall transport rate.

Overall confinement

Because of the complications described in the previous section, the overall confinement τ_E is very situation specific and can vary over a large range as key parameters are adjusted. For this reason, the fusion community has developed scalings based on experimental observations that seek to capture the major dependences of confinement on the underlying parameters. This is typically done by performing a regression analysis of an experimental data base on an identified set of engineering parameters. These engineering parameters are easily quantified and chosen to be representative of the major features of any system being considered or designed. A typical list might include toroidal field B , plasma current I , density n , heating power P , device size R , plasma shaping parameters for elongation κ and inverse aspect ratio $\epsilon = a/R$, and isotope mass M . A regression analysis of these parameters would take the form (Yushmanov et al., 1990):

$$\tau_E = C B^{\alpha_B} I^{\alpha_I} n^{\alpha_n} P^{\alpha_P} R^{\alpha_R} \kappa^{\alpha_\kappa} \epsilon^{\alpha_\epsilon} M^{\alpha_M} \quad (21)$$

As we will see, this list is somewhat specific to tokamak confinement and new parameters may be needed to fully parameterize confinement in other types of systems.

A regression analysis of the form given in Eq. (21) requires a sufficient dataset with representative variations in all the parameters shown. In general, data from multiple devices is required to provide a sufficient basis for reasonably accurate projections of confinement using this approach. To date, only two types of magnetic confinement systems have such a range of devices: tokamaks and stellarators. To illustrate this approach to projecting confinement, we will use the tokamak and in particular, the confinement scalings that have been developed to inform the design of ITER.

In addition to a sufficient range, the data must also be well conditioned to be representative of the types of operating conditions expected in ITER. Even with such conditioning, the database for tokamaks is quite extensive in the parameters of Eq. (21) as shown in Table 2. Applying the regression analysis of Eq. (21), one finds a reasonably good fit over the full data range as is shown in Fig. 6.

Table 2 Range of data in tokamak H-mode database and scaling exponents for the ITER H98y2 (ITER Physics Expert Group on Confinement and Transport, 1999) and DS03 scalings.

	B (T)	I (MA)	n ($10^{20} m^{-3}$)	P (MW)	R (m)	κ	ϵ	M
Minimum in database	1.05	0.14	1.21	0.18	1.29	0.93	0.16	1.0
Maximum in database	3.72	5.13	11.7	18.2	2.93	2.07	0.41	2.0
α_x (H98y2)	0.15	0.93	0.41	−0.69	1.97	0.78	0.58	0.19
α_x (DS03)	0.30	0.75	0.32	−0.47	2.09	0.88	0.84	0.0

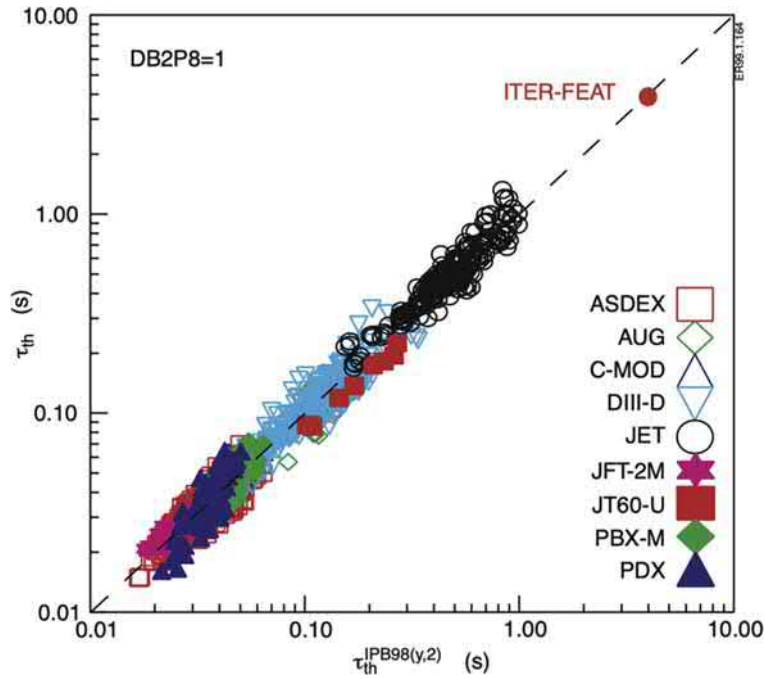


Fig. 6 Comparison of the experimentally measured thermal confinement time in tokamaks with the empirical scaling ITER H98y2 (Wagner, 2013).

The exponents in Eq. (21) for this fit are shown in Table 2 and project to $\tau_E = 3.2$ s for the design values of ITER. Combined with ITER design values of $\beta = 2\%$, $B = 5.3$ T, and $Q_{fus} = 10$, one can see that the criteria for self-sustained fusion operation given by Eq. (15) is satisfied.

A separate method to develop a confinement scaling is to utilize a wind-tunnel approach that first develops scalings of how confinement varies with key non-dimensional physics variables (as opposed to engineering variables). These parameterizations are then utilized to construct an overall confinement scaling that captures the dependence on engineering variables. In principle, there are many dimensionless physics parameters. The challenge is to identify those parameters that represent key aspects of the underlying physics phenomena.

From the discussion above on transport effects, the key physics effects are related to the characteristic orbit size and how often particles collide. Another key factor in the overall system description is any change in tension the field line experiences as particles move through or cause fluctuations within the system. A final consideration is the relative role of Coulomb interactions on the particle dynamics. Identifying non-dimensional physics variables that represent these physics effects then allows one to construct physics-based scalings in smaller devices that can be used reliably for projections to larger systems. This is akin to the testing of new automobile and plane designs in wind tunnels where care has been taken to ensure the non-dimensional variables in those laboratory tests are identical to those experienced in real-time situations.

The first scientist to do this for toroidal magnetic systems was Boris Kadomtsev (Kadomtsev, 1975) who identified four plasma physics parameters:

- the ratio of the gyroradius to the device size (ρ^*), which provides quantification of the characteristic orbit size relative to the size of the confining system;
- the collision frequency relative to the orbit bounce frequency (ν^*), which reflects the relative impact of collisions in modifying orbit trajectories;
- the plasma β [defined in Eq. (13)], which characterizes the relative tension that the magnetic fields within the system are under and therefore an indication of the relative impact of kinetic vs electromagnetic effects;
- and the total number of particles in a Debye sphere (N_D), which quantifies the relative impact of single-particle Coulomb interactions vs collective effects.

Combining with other non-dimensional parameters that are the ratios of like quantities such as the inverse aspect ratio $\epsilon = a/R$, the safety factor $q = eB_T/B_p$, and m_e/m_i , one can arrive at a set of parameters that describe any toroidal confinement plasma system. This can even be extended further by including additional parameters like the ratio of the plasma height to width (κ), separate ion and electron temperatures T_e/T_i , more than one ion charge state Z_{eff} , and plasma rotation (Mach number).

Further work by Connor and Taylor (Connor and Taylor, 1977) on scale invariance of the equations that govern plasma dynamics showed that N_D plays a negligible role in the scale invariance in the governing equations due to its vast difference in magnitude relative to the other parameters. In other words, an enormous change in N_D would be required to have an impact on the underlying physics that is comparable to the impact caused by small changes in the other variables.

This leaves ρ^* , β , and ν^* as the primary non-dimensional physics variables. This work also confirmed that ρ^* serves as a good proxy for electrostatic effects on transport, β a proxy for magnetic effects, and ν^* a proxy for collisional effects. The popularity of using these variables to describe plasma confinement is largely due to the transparent relationship between the underlying physical phenomena and these parameters.

On this basis, one can carry out experiments to determine how transport scales with each of these variables. This is typically done by varying one of the variables *while* holding all the other non-dimensional parameters approximately constant through appropriate choices in the underlying engineering parameters such as magnetic field strength, input power, and plasma density. While studies on different devices (or combination of devices) have revealed a range of variations, a generally accepted set of scalings are: $\Omega\tau_E \propto \rho^*{}^{-3}$, $\Omega\tau_E \propto \beta^0$, and $\Omega\tau_E \propto \nu^*{}^{-0.3}$ (Petty, 2008). Note that $\Omega\tau_E$, where Ω is the gyrofrequency introduced earlier, is used here as a “normalized,” non-dimensional confinement time. Its use results from applying the principles of scale invariance introduced above to the governing equations for plasma transport.

The above set of variations with ρ^* , β , and ν^* is broadly consistent with the theoretical expectations that the effect of small-spatial-scale electrostatic turbulence will diminish as the device size increases relative to the gyroradius (strong dependence on ρ^*), the limited role of electromagnetic fluctuations (very small dependence on β), and the modest favorable impact of reduced collisionality (modest dependence on ν^*). Because of the independence of these variations, one can then form an overall normalized confinement of the form:

$$\Omega\tau_E \propto \rho_*^{-3} \beta^0 \nu_*^{-0.3} F(\dots) \quad (22)$$

where $F(\dots)$ represents the variation due to other dimensionless parameters.

Because ρ^* , β , and ν^* (and the other parameters in F above) are themselves combinations of the engineering variables of Eq. (21), one can then recast Eq. (22) in the form of Eq. (21) to reveal the dependence on the engineering variables. Such an exercise yields the DSO3 exponents shown in Table 2. Note that the dependence on the engineering parameters are somewhat similar but have important differences (especially with regard to the input power), potentially leading to different projections than the fully empirical scaling. As an example, the projection for ITER design values using the DSO3 scaling is $\tau_E = 4.7$ s, much more favorable than the empirical scaling shown in Fig. 6.

Magnet technology

As noted in the discussion of Eq. (15), the ability to increase the magnetic field can be utilized to reduce the requirements on stability (β) and confinement (τ). This has been well recognized in fusion research since its onset and has served as a major driver in the worldwide development of large-scale, superconducting magnet technology. Substantial qualitative and quantitative progress has been achieved in magnet technology for fusion over the past 25 years, systematically addressing a range of limitations and increasing the operating regime over which long-pulse, high-field operation can be obtained. These limitations take on different forms depending on the magnet technology but generally fall into three categories: internal heating of the coil; current density limits; and mechanical stress limits. For superconducting magnets, the internal heating limitation is replaced by limitations on the maximum operating temperature and magnetic field for superconducting operation.

Fusion reactor applications typically require toroidal magnetic fields of > 5 T and demand long-pulse operation of the magnet. The magnetic field requirement translates into significant current density within the magnet and the potential for significant internal heating of the magnet due to its intrinsic resistance to current flow. This resistive process (known as Joule heating or Ohmic heating) scales as $P_{Ohmic} \propto I^2 R \propto J^2 A^2 R$ where R is the resistance of the conductor, J is the current density in the conductor, and A is the area of the conductor. This heat generation is problematic as more electrical power is required to produce the same current in the coil and significant cooling systems are required to dissipate the heat sufficiently to protect the coil from thermal damage. For these reasons, even the use of low-resistance conductors such as copper at very high current (or equivalently current densities) is precluded for steady-state operation of fusion magnets.

This realization led to the early adoption by fusion system designers of superconducting magnet technology. Superconductivity is the phenomena that occurs when a material is cooled below a critical temperature at which its resistance suddenly disappears. Superconductivity was first observed in 1911 by Heike Kamerlingh Onnes (Onnes, 1912), for which he won a Nobel Prize in physics a few years later. Early attempts to develop magnets based on this phenomenon were unsuccessful, and it wasn't until the mid-1950s that electromagnets (using niobium) were successfully produced; however, these magnets still had serious limitations in current density and in the magnetic field at which high current density could be maintained. In the early 1960s, a team from Bell Labs developed and demonstrated simultaneous operation at high current density (10^5 A/cm²) and high magnetic field (8.8 T) of a Nb₃Sn conductor (Kunzler et al., 1961). Since this demonstration, various alloys of niobium have been successfully developed, eventually leading to the adaptation of Nb₃Sn and NbTi conductors in the superconducting magnets for ITER and other planned fusion systems.

Although the niobium-based superconducting magnets have opened up a large range of potential fusion applications (Table 3), they have two limitations: the magnet must be maintained below a critical temperature T_c (typically near 4.2 K implying significant refrigeration cost) and the maximum operating magnetic field in which high current density is possible is limited to < 20 T (thereby limiting the attainable magnetic field for confinement purposes). Another Nobel Prize discovery in the

Table 3 Properties of potential superconducting magnet materials for fusion applications.

Superconducting material	Critical temperature T_c (K)	Critical magnetic field (T) at 4.2 K	Critical current density at 15 T (4.2 K)
NbTi	10	15	0
Nb ₃ Sn (ITER)	18	24	500
Nb ₃ Sn (Internal Sn)	18	24	1000
Bi ₂ Sr ₂ CaCu ₂ O ₈ (Bi 2212)	96	>40	1500
Bi ₂ Sr ₂ CaCu ₂ O ₁₀ (Bi 2223)	108	>40	350
YBa ₂ Cu ₃ O ₇ (YBCO/REBCO)	92	>40	3500

mid-1980s (Bednorz and Muller, 1986) that a certain set of ceramics (known as copper oxides or cuprates) exhibit a superconductivity transition at much higher temperatures has opened up the operating space for superconducting magnets even further. The most advanced of these high temperature superconducting (HTS) materials can operate with $T_c \sim 110$ K and at magnetic fields exceeding 40 T (theory suggests much higher fields are possible). The critical temperature, magnetic field, and current density for possible superconducting materials is provided in Table 1.

As noted previously, the ITER magnet design (Mitchell et al., 2008) utilizes NbTi for modest field applications (poloidal field coils with maximum fields less than 6 T) and Nb₃Sn for the high-field magnets (toroidal field coils and the central solenoid with maximum fields less than 13 T). For typical fusion systems, these maximum magnetic fields translate into 5–7 T fields in the plasma. New designs are emerging within the fusion community worldwide that take advantage of HTS magnets (Sorbom et al., 2015). These designs typically have maximum magnetic fields at the coils around 23 T, translating to ~ 12 T in the plasma. At present, the manufacturability of large-scale NbTi and Nb₃Sn magnets has been demonstrated by the ITER project with the production of sufficient lengths of NbTi and Nb₃Sn strand demonstrated by multiple companies. Several private firms are now pursuing such a demonstration of REBCO-based magnets.

While the focus of this development has been on taking advantage of the higher critical magnetic field of these materials, there are significant benefits derived from being able to operate HTS magnets at higher temperature and higher current density. Higher temperature operation will reduce the refrigeration system operating costs and also enables the possibility of jointed coils, which could significantly improve the maintainability of the fusion system. Higher current density operation potentially translates to a smaller magnet. However, there are a few considerations that could limit this. Firstly, higher field operation will inevitably need more mechanical structure to react against the electromagnetic forces. Because these forces increase as B^2 , significant amounts of support structure must be added as the field is increased. Secondly, the system must always be designed to withstand “quenching” (loss of superconductivity) of the magnet. This effect typically limits the overall current density during the quench phase, resulting in the need to provide additional conducting material in the design to keep the magnet from overheating during the quench event. For these reasons, the full capability of REBCO-based magnets may not be realizable in toroidal confinement systems, and a full systems analysis is required to assess the available operating space.

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Magnetic Confinement Fusion Concepts/Configurations

Ricardo M.O. Galvão and Gustavo P. Canal, Institute of Physics, University of São Paulo, São Paulo, Brazil

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Magnetic confinement configurations

As discussed in the previous two chapters, one of the most basic requirements for the development of fusion reactors is to confine plasmas with temperatures over 100 million degrees kelvin. This condition requires minimizing the contact between the charged particles and surrounding material walls of the vacuum chamber, inside which the plasma is generated. Since the dawn of fusion research, it became clear that an obvious way to perform this task was to use properly shaped magnetic fields to trap the charged particles. Several magnetic field configurations have been proposed for this purpose and, in this chapter, the main ones that have been developed, and are currently employed in magnetic field fusion research, are discussed.

Magnetic confinement basics

Three physics processes play a basic role regarding equilibrium, stability, and energy transport in magnetically confined plasmas: (i) particle drifts, (ii) macroscopic stability, and (iii) anomalous (i.e., non-collisional) particle and energy transport.

The drift effect can be easily understood by examining the motion of charged particles, let us say of species α , mass m_α and charge q_α , in a uniform magnetic field $\mathbf{B}$ and in the presence of a constant force $\mathbf{F}_\alpha$ (see Chapter 2 in Bittencourt, 2004). Considering first the case without an external force, the motion perpendicular to the magnetic field is restrained to circular orbits, with radius of gyration given by $\rho_\alpha = v_{\alpha\perp}/\omega_{c\alpha}$. Here, $v_{\alpha\perp}$ is the perpendicular component of the particle's velocity and $\omega_{c\alpha} = q_\alpha|\mathbf{B}|/m_\alpha$ is its cyclotron frequency. In the direction parallel to the magnetic field, however, the particle motion is unconstrained, and the parallel component of its velocity, $v_{\alpha\parallel}$, can have any arbitrary value. The resulting orbit becomes a helix winding around a magnetic field line and the instantaneous center of cyclotron motion is called the orbit's guiding center. Considering now the effect of a constant force over the gyromotion perpendicular to the magnetic field, the particle is accelerated when it moves in the direction of $\mathbf{F}_\alpha$ and decelerated when it moves against it. Therefore, it has a larger value of the perpendicular velocity $v_{\alpha\perp}$ at the end of the speed up phase and a smaller one in the end of the slow down phase. This makes the gyroradius ρ_α larger at the end of the former phase and smaller at the end of the latter, so that the orbits do not close circularly anymore in the plane perpendicular to the magnetic field. They become cycloids that displace sideways perpendicular to both $\mathbf{F}_\alpha$ and $\mathbf{B}$, corresponding to a motion of the guiding center with a drift velocity given by

$$\mathbf{v}_{D\alpha} = \frac{1}{q_\alpha} \frac{\mathbf{F}_\alpha \times \mathbf{B}}{B^2} \quad (1)$$

Since electrons and ions gyrate in opposite directions around a magnetic field line, they also drift in opposite directions, unless the external force is due to an electric field $\mathbf{E}$. In this particular case, since electrons are accelerated while moving against $\mathbf{E}$, the phases of increasing and decreasing perpendicular velocity reverse, causing them to drift in the same direction as the ions. This can be seen by the cancellation of charge in Eq. (1) for the particular case in which $\mathbf{F}_\alpha = q_\alpha\mathbf{E}$.

The equilibrium of magnetically confined plasmas is basically described by the balance between plasma pressure gradient and Lorentz forces, i.e., $\nabla p = \mathbf{J} \times \mathbf{B}$, where $\mathbf{J}$ is the current density. The macroscopic stability, however, is somewhat more difficult to explain in simple terms. From the equilibrium equation, it can be predicted that instability sources are mainly associated to pressure gradients and current density perturbations, to the curvature of magnetic field lines, and to spatial anisotropies. Despite this difficulty, some simple models can be explored to elucidate the basic physical mechanisms. One of such a simple models is that of a cylindrical plasma column of sharp boundary. In this model, the plasma is permeated by an uniform and constant longitudinal magnetic field $\mathbf{B} = B_z\hat{\mathbf{e}}_z$, produced by external coils, it has a constant plasma pressure p_0 and a surface current density $\mathbf{K} = K_a\hat{\mathbf{e}}_\phi$ at the sharp boundary (see Chapter 13 in Bittencourt, 2004). The magnetic field outside of the plasma column generated by the

plasma current is given by $B_p = B_\theta(r)\hat{e}_\theta = aB_a/r\hat{e}_\theta$, where $B_a = B_\theta(a) = \mu_0 K_a$ and a is the radius of the plasma column. The plasma equilibrium equation, integrated over the sharp plasma boundary, imposes that

$$p_0 = \frac{B_a^2}{2\mu_0} \quad (2)$$

Let us now assume that the sharp plasma boundary is perturbed as indicated in Fig. 1. Considering the plasma as an incompressible, ideally conducting fluid, one has that, in the regions where the column radius shrinks, the magnetic pressure due to the field generated by the plasma current increases and it decreases in the regions where the column inflates. Since the plasma pressure is constant, pressure imbalance favors the perturbation to keep further growing. However, the longitudinal field frozen in the plasma also exerts a magnetic pressure and reacts to compression and expansion. The variation of the magnetic pressure due to the longitudinal field associated with the perturbation can be calculated by considering the constancy of the magnetic flux $\phi_m = \pi r^2 B_z$ in the plasma column. Imposing $d\phi_m = 0$, the change in the corresponding internal magnetic pressure becomes $dp_{\text{int}} = -(2B_z^2/\mu_0 r) dr$.

The variation in the external magnetic pressure, i.e., the pressure associated to the magnetic field produced by the plasma current, is readily obtained, $dp_{\text{ext}} = -(B_a^2/\mu_0 r) dr$. For stability, it is necessary that $dp_{\text{int}} > dp_{\text{ext}}$, or

$$B_z^2 > \frac{B_a^2}{2} \Rightarrow \beta \equiv \frac{p_0}{B_z^2/2\mu_0} < 2 \quad (3)$$

In this last expression, we have used the equilibrium condition and introduced the parameter β , which is the ratio between plasma and magnetic pressures. This parameter appears in the stability condition of almost all macroscopic instabilities in magnetically confined plasmas and, for actual configurations, it is usually limited to values much below unity. The current flowing in the plasma column can also drive another type of instability called a kink instability, which is associated with helical perturbations of the plasma column. These perturbations require a more sophisticated model to be properly described and, in its simplest version, the stability condition is given by (see Chapter 6 of Myamoto, 2016).

$$\left| \frac{B_a}{B_z} \right| < \frac{2\pi a}{L} \quad (4)$$

where L is the length of the plasma column. This condition is called the Kruskal-Shafranov criterion, after M.D. Kruskal and V.D. Shafranov, who independently derived it in 1958. We see that, in this simple model, the stability condition is not directly associated with the value of β , but with the pitch of the magnetic field lines. In toroidal configurations, L represents the periodic length of the plasma column, i.e., $L = 2\pi R_0$, where R_0 corresponds to an equivalent plasma major radius, and the above criterion is written in terms of the edge safety factor, q_a , where

$$q_a = \frac{aB_z}{R_0 B_a} > 1 \quad (5)$$

For a given longitudinal stabilizing field B_z , this criterion imposes a maximum plasma current I_p that can be stably driven in the plasma column.

To avoid this type of current-driven instability, it is possible to conceive magnetic configurations in which the required confining field is entirely produced by currents flowing in external coils. In fact, this is the basis of the stellarator concept, which will be discussed in “Stellarators” section. Since in a confined plasma the pressure usually decreases outwards, one basic question is whether pressure perturbations can be unstable in this case. The instabilities that are most prone to be excited are those that distort the least the magnetic field lines, avoiding the back reaction due to magnetic tension along field lines. One such instability is the so-called interchange mode, where an adiabatic radial exchange of flux tubes occurs. A flux tube is a tube formed by field lines, i.e., a tube with a constant magnetic flux through its cross section dS , so that its volume is given by $V = \int d\mathcal{L} dS = d\phi \int d\mathcal{L}/B$, where $d\psi = BdS$, is the magnetic flux and the integral is carried out along the entire field line. Let us then suppose that a perturbation interchanges radially two flux tubes of equal flux but with pressures and volumes (p_I, V_I) and (p_{II}, V_{II}) . Because the plasma is a highly conductive fluid, it is “frozen” to the magnetic field lines and moves along with the flux tubes. Since these have equal flux, the magnetic energy remains unchanged in the process. Then, only the plasma energy $W_p = pV/(\gamma - 1)$, where γ is the adiabatic index, can change; a perturbation is unstable if $\delta W_p < 0$, because, in this case, the system evolves to a lower energy state. In adiabatic processes, $pV^\gamma = \text{const}$ and the new pressure in volume V_{II} will be $p_{II} = p_I V_I^\gamma / V_{II}^\gamma$ and in volume V_I it will be $p_I = p_{II} V_{II}^\gamma / V_I^\gamma$. Assuming an incremental variation, i.e., $V_I = V$, $V_{II} = V + \delta V$, $p_I = p$, $p_{II} = p + \delta p$, it is straightforward to show that

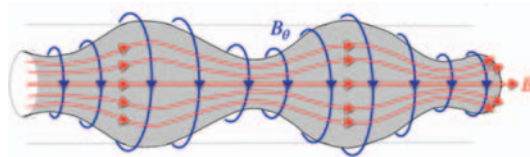


Fig. 1 Sketch of boundary perturbation on a cylindrical plasma column permeated by a constant and uniform magnetic field.

$$\delta W_p = \delta p \delta V + \gamma p \frac{\delta V^2}{V} \quad (6)$$

Since the second term on the right-hand side is always positive, it turns out that a sufficient condition for stability is $\delta p \delta V > 0$. Since in confined plasmas $\delta p < 0$, this condition implies that $\delta V = d\phi \delta(\int d\mathcal{L}/B) < 0$, that is, the magnitude of the magnetic field inside the flux tube has to increase. Magnetic configurations that satisfy this condition have field lines that are convex with respect to the region where the plasma is confined and are called magnetic wells. In closed magnetic configurations, a magnetic well can be achieved only in an average sense, as will be discussed later. Macroscopic instabilities, either pressure gradient- or current-driven, are usually referred to as MHD instabilities, because they can be well described by a magnetohydrodynamic model of the plasma (Goedbloed and Poedts, 2004).

The particle and energy transport in magnetically confined plasma is a process that still cannot be accurately described from first-principle models. Considering the standard random-walk collisional transport model, the classical diffusion coefficient should be $D = \rho_\alpha^2 / \tau_\alpha$, where τ_α is the characteristic collision time of particle species α . This result comes from the assumption that, after a collision, the characteristic step size across the magnetic field is of the order of ρ_α . This classical diffusion coefficient can be written in terms of practical units used in fusion research as

$$D = 2.0 \times 10^{-3} \frac{n}{B^2 \sqrt{T}} \left(\frac{\text{m}^2}{\text{s}} \right) \quad (7)$$

where B is the magnetic field, expressed in Tesla, n is the plasma density, expressed in units of 10^{20}m^{-3} , and T is the plasma temperature, expressed in keV. This is a quite optimistic result, as it indicates that the energy confinement should increase with the square of the magnetic field strength and with the square root of the plasma temperature. Considering typical parameters of current large fusion devices ($T = 10 \text{ keV}$, $n = 0.5 \times 10^{20} \text{m}^{-3}$ and $B = 3 \text{ T}$), one obtains $D = 3 \times 10^{-5} \text{m}^2/\text{s}$. Typical experimental values are not only five orders of magnitude larger than this, i.e., $D \approx 1 \text{m}^2/\text{s}$, but they also show a much less favorable scaling, following variations of the so-called Bohm empirical diffusion coefficient, $D_B \approx k_B T / 16eB$ (MKS units). The ratio between the two diffusion coefficients is $D_B/D \approx \omega_{ce} \tau / 16$, so that the experimental results approach the values predicted by the Bohm diffusion coefficient as the magnetic field and plasma temperature increase, making the plasma less collisional. The main reasons for this observed anomalous diffusion have been attributed to plasma micro-scale turbulence and the convective transport associated with it. Thus, in the comparison of different confinement schemes, it is important to take into consideration the level of anomalous transport. This is usually gauged by estimating their energy confinement time, defined as

$$\tau_E = \frac{\frac{3}{2} n k_B T}{P_{in}} \quad (8)$$

where P_{in} is the input power necessary to balance energy transport and radiation, and, therefore, to maintain the plasma at temperature T .

In what follows, the performance of different magnetic confinement schemes will be expressed not only in terms of the maximum values of their temperature and density, but also by the values of the parameters β , I_p , and τ_E , when applicable.

Linear devices

The basic advantage of linear devices is their geometric simplicity, making it somewhat easier, from an engineering perspective, the design of the components required for fusion reactors, such as magnetic coil systems, plasma control and heating systems, neutron blanket, etc. However, the simplest magnetic configuration one could envisage, a straight cylinder, such as that produced by a very long solenoid, would not work because the charged particles would leave by streaming out of its ends. A basic scheme to overcome these end losses is to increase the magnitude of the field at the ends of the cylinder, so that the charged particles feel a reflective force as they approach them. This effect occurs because the circular motion of a charged particle in a magnetic field produces a magnetic moment that opposes the external field. So, the particle movement towards the ends of such a straight configuration, with increasing field strength, can be seen as the effect of two magnets of identical poles facing each other.

Magnetic mirror

The first magnetic mirror systems were independently proposed in the middle 1950s by G.I. Budker, in the Soviet Union, and R.F. Post, in the United States (Beklemishev et al., 2012). The basic magnetic configuration is simply that produced by two coaxial coils, separated by some distance, as shown in Fig. 2A. Neglecting collisions, two main constants of motion are adiabatically conserved as the charged particles move between the two mirror throats, spiraling around the magnetic field lines, the particle's total energy, $E = mv^2/2$, and magnetic moment, $\mu_m = mv_\perp^2/2B$ (see Chapter 3 of Bittencourt, 2004). Denoting the values of the physical constants at the center of the mirror by the index "0", i.e., v_0 , $v_{0\perp}$ and B_0 , and the maximum value of the magnetic field at the mirror throat by B_m , it follows from the conservation of the adiabatic constants of motion that the condition

$$\sin(\theta) = \frac{v_{0\perp}}{v_0} > \sqrt{\frac{B_0}{B_m}} \quad (9)$$

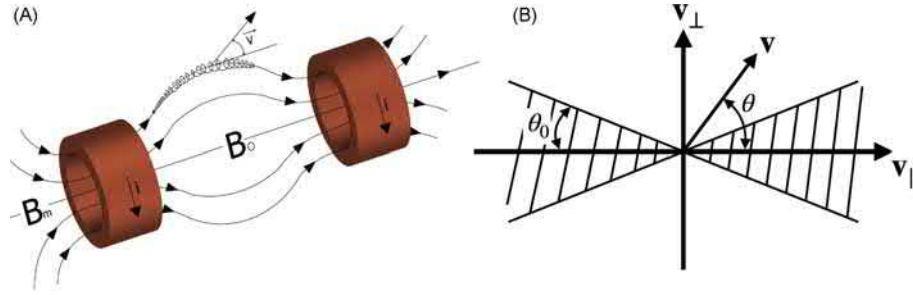


Fig. 2 (A) Basic magnetic mirror configuration and (B) loss-cone in space velocity. (A) Reproduced from Fig. 1 of Beklemishev AD, et al. (2012) *Magnetic mirrors: History, results, and future prospects. Problems of Atomic Science and Technology* 6: 8–12. <https://vant.kipt.kharkov.ua/TABFRAME2.html>, and (B) reproduced from Fig. 2.5 of Myamoto K (2016) *Plasma Physics for Controlled Fusion*. Springer.

must be satisfied for particles to be confined. This angle establishes a loss-cone in velocity space, as pictured in Fig. 2B. Particles that enter the loss-cone are lost, and, as a consequence, the particle distribution function in a mirror device is non-Maxwellian. Since Coulomb collisions between particles of the same species lead to chaotic diffusion in velocity space, trapped ions eventually fall in the loss-cone, so that the basic particle confinement time, τ_p , in mirror devices is proportional to the ion-ion collision time (Sivukhin, 1966), τ_{ii} ,

$$\tau_p = \tau_{ii} \ln \left| \frac{B_m}{B_0} \right| \quad (10)$$

For a deuterium plasma with $T = 10 \text{ keV}$ and $n = 1 \times 10^{20} \text{ m}^{-3}$, one obtains $n\tau_p = 8 \times 10^{17} \ln |B_m/B_0| \text{ m}^{-3} \cdot \text{s}$, i.e., a value well below that required for a fusion reactor (see Chapter 12.02). The non-Maxwellian distribution function also provides a source of free energy to drive velocity-space instabilities (Guest and Dory, 1965), whose non-linear evolution increases substantially the anomalous energy transport from the plasma.

It was theoretically predicted by Rosenbluth and Longmire that a symmetric mirror configuration should be MHD unstable, at all values of β (Rosenbluth and Longmire, 1957), to the flux tube interchange instability described in “Magnetic confinement basics” section. Although this instability was not seen in the first mirror experiments, it was identified by M.S. Ioffe in the PR-2 device, in 1961. He subsequently proposed a technique for avoiding this instability by adding a quadrupole magnetic field to the symmetric mirror field (Gott et al., 1962). The combination of the two fields produces a magnetic well configuration, in which $|B|$ increases in all directions. Other “minimum-B” configurations, in particular the “baseball-shaped coils,” were later proposed, and MHD stable plasmas were confined in mirror devices with $\beta > 1$ (Turner et al., 1979).

Despite the good MHD results, the single mirror minimum-B plasma column is highly asymmetric, and the experiments carried out in different devices have show rather poor energy confinement, that is, enhanced anomalous transport. These results, and the need to reduce end losses, lead to the development of the “tandem-mirror” concept, which became the main line of mirror research. The configuration is basically long solenoidal mirror with minimum-B cells at both ends. These not only play the role of “MHD anchors,” but also provide a localized volume where highly energetic neutral particles can be injected and the electrons strongly RF-heated. With this scheme, the ambipolar field in these “end-plugs” can be properly shaped to minimize particle losses (Beklemishev et al., 2012). However, the transition from the axisymmetric field of the central cell to the quadrupole anchors requires the addition of extra quadrupole fields, which increase the local asymmetry, spoiling the MHD stability and energy confinement of the device. The basic scheme of a tandem-mirror configuration is sketched in Fig. 3A (Fowler et al., 2016). A drawing of GAMMA-10, the world largest tandem mirror device, is shown in Fig. 3B (Imai et al., 2011). It has a more sophisticated design, including an axisymmetric mirror cell after each MHD anchor. The central coil length is $L = 5.6 \text{ m}$, with central magnetic field $B_0 = 0.4 \text{ T}$ and mirror ratio $B_m/B_0 = 5.2$. The plug/barrier cell has a length $L = 2.5 \text{ m}$, with magnetic-field at the midplane $B_0 = 0.5 \text{ T}$, and mirror ratio $B_m/B_0 = 6.2$.

In GAMMA-10, the ions are heated by ion-cyclotron resonance heating, ICRH, and the electrons by electron cyclotron resonance heating, ECRH. It has reached large ion temperatures ($T_i \approx 10 \text{ keV}$) and good values for the ambipolar plugging potential ($V \approx 500 \text{ V}$), reducing end-losses by a factor of 100. However, some other parameters remained modest: $T_e, \text{max} \approx 300 \text{ eV}$, $n_{\text{max}} = 1 \times 10^{19} \text{ m}^{-3}$, $\beta \approx 0.05$ and $\tau_E \approx 10 \text{ ms}$. Recently, the main scientific objective of GAMMA-10 was changed to divertor studies under similar condition of actual fusion devices. With additional RF wave heating, the ion flux at the end-mirror exit was increased to $3.3 \times 10^{23} \text{ m}^{-2}$, delivering a maximum heat flux of 21 MW/m^2 , which is of the order of that expect on the divertor plates of ITER (Nakashima et al., 2016).

An alternative scheme to the tandem mirror, keeping the attractive geometry of axisymmetric mirrors, was proposed in 1979 at the Budker Institute of Nuclear Physics (Mirnov and Ryutov, 1988). The fundamental rationale of the proposal was to have a central cell long enough, and with sufficiently high plasma pressure, to make the ion-ion mean-free-path smaller than the length L of the cell. In this case, there is enough time for collisional velocity-space isotropization, before the ions in the loss-cone escape the confinement region. The distribution function becomes isotropic and Maxwellian and the particles escape with velocities of the order of the ion-acoustic speed, $v_s = \sqrt{\gamma k_B T / m_i}$, where γ is the ratio of specific heats and m_i the effective ion mass. The

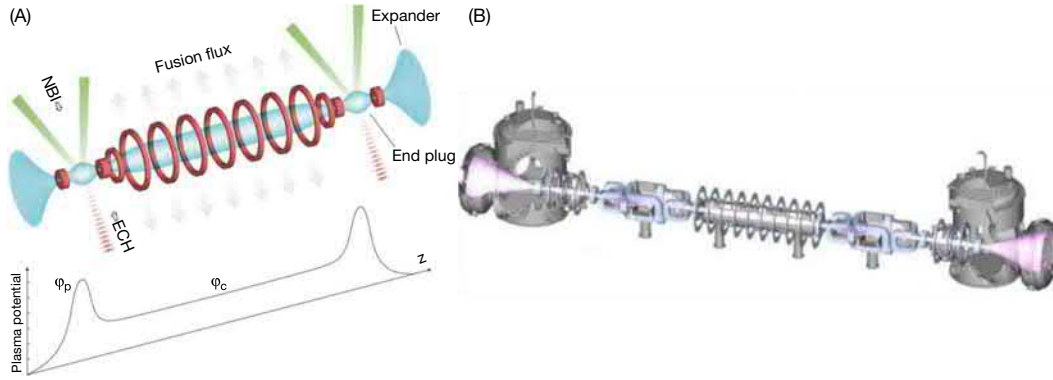


Fig. 3 (A) Sketch of a tandem-mirror magnetic confinement configuration and (B) drawing of the GAMMA-10 tandem mirror. The electrons inside the end-plugs are strongly heated by electron cyclotron waves (ECH) and the ions are fueled and heated by neutral-beam injection (NBI). The resulting ambipolar potential acts to decrease particle end loss. (A) Reproduced from Fig. 1 of Fowler TK, Moir RW and Simonen TC (2016) Prospects for an axisymmetric advanced fuel tandem mirror. *AIP Conference Proceedings* 1771(1): 080003. <https://aip.scitation.org/doi/abs/10.1063/1.4964242>, and (B) reproduced from Fig. 1 of Nakashima Y, Imai T, Sakamoto M, Katanuma I, Kariya T, Yoshikawa M, Ezumi N, Minami R, Hirata M, Kohagura J, Numakura T, Ikezoe R, Ichimura K, Wang X and Ichimura M (2016) Overview of recent progress and future in GAMMA 10/PDX project. *AIP Conference Proceedings* 1771(1): 020002. <https://aip.scitation.org/doi/abs/10.1063/1.4964155>.

resulting particle confinement time becomes $\tau_p = (B_m/B_0)L/v_s$, and, because of the dependence on the ratio between length and thermal speed, this configuration was dubbed Gas Dynamic Trap (GDT). Quite good results were obtained in the GDT experiments at the Budker Institute, with high mirror ratio, $B_m/B_0 = 35$; plasma density $n \approx 7 \times 10^{18} \text{m}^{-3}$, electron temperature $T_e \approx 650 \text{eV}$, energy confinement time $\tau_E \approx 0.5 \text{ ms}$, and $\beta \approx 0.6$ (Bagryansky et al., 2015).

Although a fusion reactor based upon the GDT concept would have to be quite long, $L \approx 1 \text{ km}$, it was recognized that a much shorter device, with modest electron temperature, $T_e \approx 1 \text{ keV}$, would be suitable for a deuterium-tritium neutron source, a facility needed for material testing in the fusion program (Ivanov and Prikhodko, 2013). This stimulated further investigation of this scheme, evolving to the concept of the gas-dynamic multiple-mirror trap (Postupaev et al., 2016). The GOL-NB multiple mirror was then built in the Budker Institute of Nuclear Physics and the results of the testing of its baseline systems were recently reported (Postupaev et al., 2019). The plasma is heated in a central cell of length $L = 2.5 \text{ m}$ by neutral beams of 0.75 MW of power at 25 kV injection energy. The plasma filling gun was successfully tested, providing plasmas that lasted for 2 s with $n \approx 8 \times 10^{19} \text{m}^{-3}$. The results in GDT and GOL-NB have inspired new possibilities for a mirror fusion facility, not only as a neutron source, but also as an advanced fuel reactor (Fowler et al., 2016).

Toroidal devices

The most straightforward closed magnetic confinement configuration is a toroidal solenoid, as sketched in Fig. 4A. The stabilizing magnetic field is produced by external coils placed around a toroidal chamber. Assuming that the number N of coils is large enough and that they are closely packed, so that their discreteness can be neglected in first approximation, the magnetic field can be easily calculated from Ampere's law,

$$B = \frac{\mu_0 N I}{2\pi R} \hat{e}_\phi \quad (11)$$

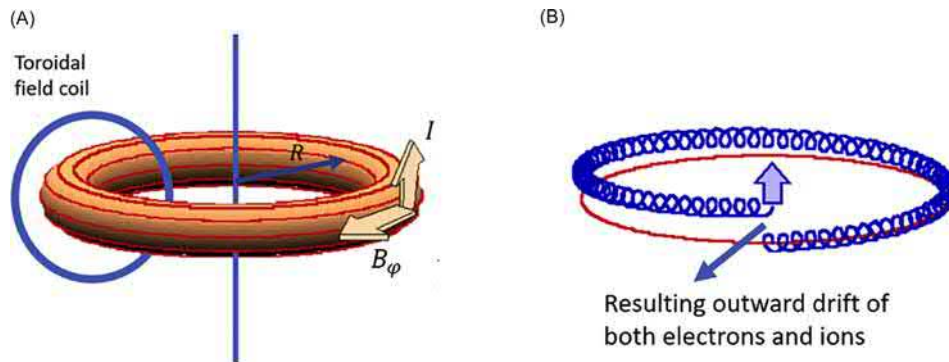


Fig. 4 (A) Magnetized toroidal plasma column and (B) particle drift orbits.

where N is the number of coils around the toroidal chamber, I is the current in each coil, R is the radial distance from the vertical axis, and $\hat{e}_\phi$ is the unit vector in the toroidal direction. The field lines are concentric circles centered at the vertical axis.

Unfortunately, the curvature of the field lines and the gradient of the magnetic field, which increase towards the vertical axis, introduce drift effects on the orbits of the particles that limit their confinement in this simple configuration. The two effects can be modeled as constant forces along the guiding center trajectory. Incorporating these into the expression for the drift velocity, it turns out that the ions and electrons drift in opposite directions, as sketched in Fig. 4B. This charge separation gives rise to an ambipolar electric field in the vertical direction. The drift due to this field crossed with the magnetic field causes both types of particles to displace outwards, and leave the confinement region.

This problem was independently identified by Enrico Fermi and Lev Artsimovich, and two rather clever schemes were conceived to overcome it. Both relied on adding a poloidal field, to transform the simple circular toroidal field lines into helical lines winding around a toroidal axis, as shown in Fig. 5A. After completing many toroidal transits, a helical line fully covers a surface, which is therefore called a magnetic or flux surface. Consider, for simplicity, the two magnetic surfaces A and B shown in Fig. 5B and that a charge departs from position 1, at the magnetic surface A, moving along a field line. Due to the curvature and ∇B drifts, its guiding center will displace upwards, towards the magnetic surface B. It eventually reaches the inside of the toroidal magnetic surface at position 2. As its guiding center goes on moving along the magnetic field, it keeps drifting upwards, so that it returns to the magnetic surface A at position 3, which corresponds to position 1 displaced toroidally, as indicated in Fig. 5A. Therefore, the displacements from the original magnetic surface A are somewhat compensated in one pitch of the helical field line.

It turns out, however, that due to the collective behavior of particle populations, as expressed by the pressure and current, plasma confinement is not entirely prescribed by the orbits of individual particles. The equilibrium in a magnetic configuration has to locally satisfy the pressure balance equation, already presented, $\nabla p = \mathbf{J} \times \mathbf{B}$, where $\mathbf{J}$ and $\mathbf{B}$ are, respectively, the self-consistent current density and magnetic field in the plasma, which are related through Ampère's law $\nabla \times \mathbf{B} = \mu_0 \mathbf{J}$. It follows from the equilibrium equation that $\mathbf{B} \cdot \nabla p = 0$, so that the constant-pressure surfaces have to coincide with the magnetic surfaces, in which the magnetic field lines lie. Therefore, to maintain a confined plasma, the magnetic surfaces must be closed and nested, enclosing a central field line called the magnetic axis.

In three-dimensional geometry, the existence of nested magnetic surfaces requires some underlying continuous spatial symmetry (Grad and Rubin, 1958; Grad, 1967). This is the case of magnetic configurations that preserve axisymmetry, i.e., that are independent of the toroidal angle ϕ . To see this, consider the combination of two simple magnetic field configurations presented in standard textbooks. The first, Fig. 6A, is the field produced by a vertical straight current I_1 . The resulting field $\mathbf{B}_1$ is in the toroidal direction $\hat{e}_\phi$ and the magnetic field lines are just closed circles centered at the vertical axis. The second configuration is the field produced by

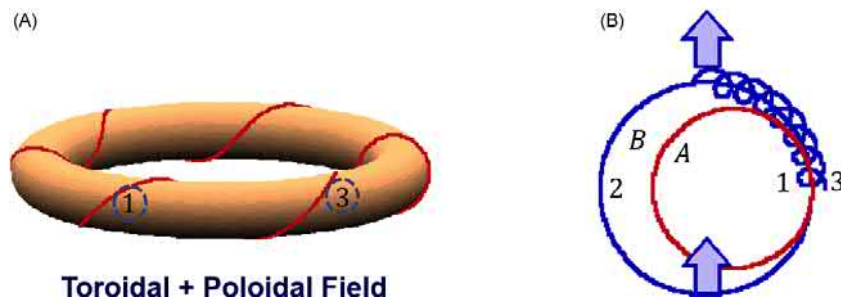


Fig. 5 (A) Toroidal plasma column with helical magnetic field forming magnetic surfaces. (B) Illustration of the displacement of a charged particle between magnetic surfaces as its guiding center moves along a field line.

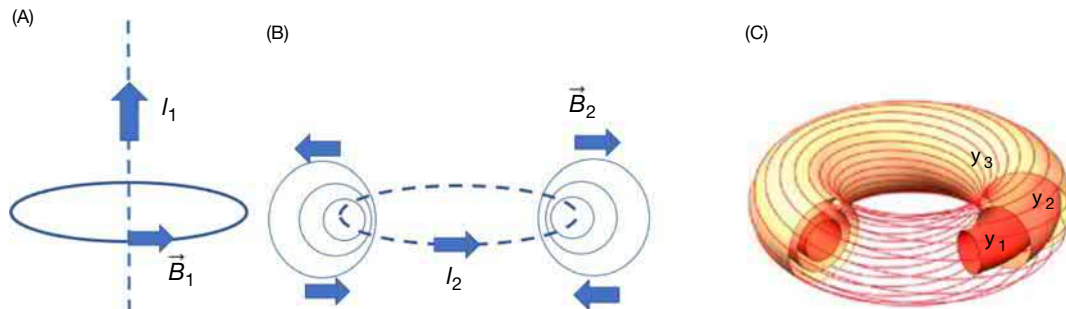


Fig. 6 (A) Field produced by a straight vertical current. (B) Field produced by a circular current loop. (C) Resulting helical field produced by the two currents. (C) Reproduced from Fig. 1 of Fasoli A, et al. (2016) Computational challenges in magnetic-confinement fusion physics. *Nature Physics* 12: 411–423. <https://doi.org/10.1038/nphys3744>.

a circular current loop I_2 . The field calculation, in this case, although somewhat more involved, is outlined in advanced textbooks. The expression for the field is given in terms of elliptic integrals, and the field B_2 lies on vertical planes perpendicular to the current loop, and these are denoted poloidal planes. The field lines are circles displaced outwards with respect to the current loop, as sketched in Fig. 6B. Considering the two currents in conjunction, the resultant magnetic field, $B_1 + B_2$, will then form helical field lines on toroidal surfaces, as sketched in Fig. 6C. This system with two current sources has axisymmetry and the flux surfaces are nested, as indicated by the surfaces with labels ψ_1 , ψ_2 and ψ_3 in Fig. 6C (Fasoli et al., 2016).

This scheme to create nested magnetic surfaces is the basis of the tokamak concept. It obviously requires a toroidal current flowing inside the plasma column (modeled by current I_2 in the example). However, it is possible to create nested magnetic surfaces using only currents flowing in coils outside the plasma column, and this is the stellarator concept. Both the tokamak and the stellarator configurations will be discussed in detail in the next subsections. One important parameter of the magnetic surfaces is the degree of “twistiness,” or helicity, of the field lines embedded within it. This is quantified by a parameter called rotational transform ι , defined so that the average angular advance of a field line, each time it crosses the same poloidal cross-section of the magnetic surface, after one toroidal turn, is given by $\Delta\theta \approx 2\pi\iota$. Considering that, in general, the field lines transit ergodically over the magnetic surfaces, a more exact definition of the rotational transform is (Morozov and Soloviev, 1966).

$$\iota = \lim_{m \rightarrow \infty} \frac{\sum_{k=1}^m (\Delta\theta)_k}{2\pi m} \quad (12)$$

where $(\Delta\theta)_k$ is the poloidal angular displacement of a field line after its k th toroidal transit. On rational magnetic surfaces, it turns out that $\iota = n/m$, i.e., the field lines close on themselves after m toroidal and n poloidal transits, where m and n are integers.

Tokamaks

The tokamak magnetic configuration was originally conceptualized by two Nobel laureate soviet physicists, Igor Tamm and Andrei Sakharov, in the beginning of the 1950s (Kadomtsev, 1988). The T-1 device, Fig. 7A, is considered to be the first tokamak to operate. It was developed under the leadership of the Ukrainian physicist Natan Yavlinsky, at the Kurchatov Institute, Moscow (Smirnov, 2009; Azizov, 2012). Although its electron temperature was rather low, due to strong line radiation from impurity ions present in the plasma, it has, for the first time, given experimental evidence of stable plasma confinement satisfying the Kruskal-Shafranov condition.

The program on development of tokamaks progressed firmly and successfully in the Kurchatov Institute during the 1960s under the leadership of L. Artsimovich, progressing not only experimentally, but also establishing the theoretical physics basis of plasma confinement in these devices. This continuous effort led to the outstanding results obtained in the T-3 tokamak, Fig. 7B, reported by Artsimovich in the Third International Fusion Conference held in Novosibirsk, Soviet Union, in 1968 (Artsimovich et al., 1968). Plasma discharges stable to kink modes with $q_a \geq 6$ were produced with plasma density $n \approx 3 \times 10^{19} \text{ m}^{-3}$, electron temperature $T_e \approx 1 \text{ keV}$, ion temperature $T_i \approx 350 \text{ eV}$, and energy confinement time $\tau_E \approx 3 \text{ ms}$, i.e., about 10 times better than that predicted by the Bohm anomalous diffusion coefficient. These plasma discharges exhibited the first evidence of thermonuclear neutron production in a confined plasma. However, these results, which were much better than those obtained in any other fusion device then under investigation, were received with some skepticism by scientists from other national laboratories. The main reason for the doubts was that the plasma temperature in T-3 was indirectly estimated through the diamagnetic effect of the plasma, that is, the decrease of the magnetic field as its pressure increases (see Chapter 2 of Hutchinson, 2005), and this method is known to be

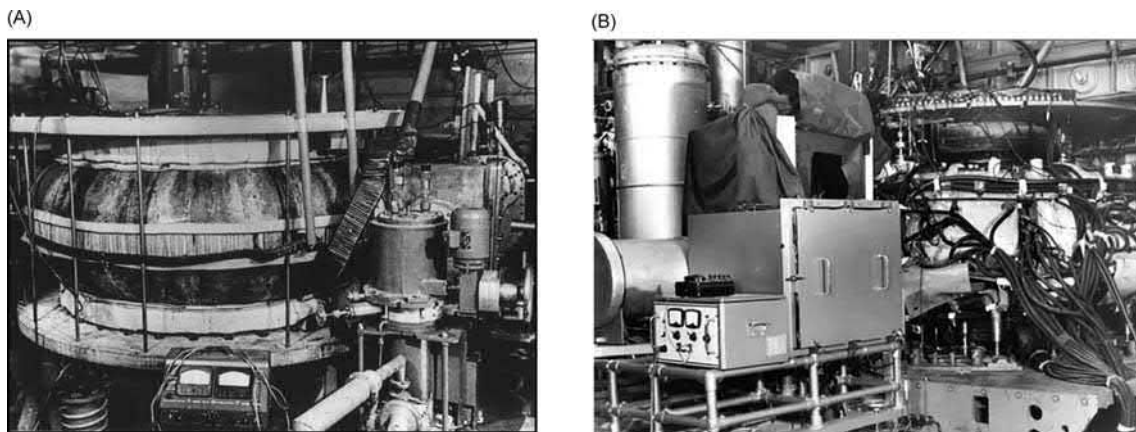


Fig. 7 (A) The T-1 and (B) T-3 tokamaks in the Kurchatov Institute, Moscow. The former started operating approximately in 1958 and the latter in 1962. (A) Reproduced from Fig. 1 of Smirnov VP (2009) Tokamak foundation in USSR/Russia 1950–1990. *Nuclear Fusion* 50(1): 014003. <https://doi.org/10.1088/0029-5515/50/1/014003>, and (B) reproduced from Fig. 2 of Azizov EA (2012) Tokamaks: 60 years later: Commemoration of the 90th anniversary of the birth of andrei dmitrievich sakharov (scientific session of the physical sciences division, russian academy of sciences, 25 may 2011). *Physics-Uspekhi* 55(2): 169–216. <https://doi.org/10.3367/ufne.0182.201202f.0181>.

inaccurate if runaway electrons (non-thermal highly energetic electrons) are present in the plasma. Artsimovich then invited Bas Pease, head of the Culham Laboratory, United Kingdom, to send a team of specialists to Kurchatov, to measure the electron temperature in T-3 using a new diagnostic system they had developed, based on the Thomson scattering effect (see Chapter 7 of [Hutchinson, 2005](#)). The invitation was accepted and the measurements fully confirmed the results reported by Artsimovich ([Peacock et al., 1969](#)). Since the first devices, around 200 tokamaks have been built in the World ([Wordpress, n.d.](#)).

The main features of a tokamak can be seen in [Fig. 8](#). The toroidal magnetic field is produced by external coils, TFC, outside the vacuum chamber. It is relevant to point out that, instead of circular coils, modern tokamaks use D-shaped coils, as shown in the figure. Due to the $1/R$ dependence of the toroidal field, this shape keeps the coil in a pure tension state, minimizing the bending moment in its plane. The poloidal field is mainly produced by the current flowing in the plasma, I_p , and by external poloidal field coils, PFC. These coils produce a vertical field, to balance the hoop expansion force of the plasma column, and radial fields to shape its cross section and feedback control its vertical position. The flux surfaces are nested around a planar magnetic axis, whose radius is denoted by R_0 , and the half-width of the outer most surface, which is the plasma boundary, is denoted by a . The ratio between these two quantities is called the aspect ratio, $A = R_0/a$. The current is induced in the plasma by varying the magnetic flux in the central solenoid, CS. Therefore, the plasma column plays the role of a one-loop secondary coil of a transform whose primary coil is the central solenoid. In the conventional tokamak design, this so-called Ohmic heating transformer (because I_p Ohmically heats the plasma) has an iron yoke. However, modern tokamaks use only an air-core CS, in series with extra PFCs to minimize stray fields in the plasma volume. This allows the design of tokamaks with small A , which improves the MHD stability of the plasma and reduces the current in the TFCs ([Peng and Strickler, 1986](#)). The necessity of inducing the plasma current is the main drawback of the tokamak concept because it makes it intrinsically a non-steady-state device. However, techniques to overcome this difficulty, at least partially, have been developed, in particular, current drive by electromagnetic waves ([Fisch, 1987](#)).

Due to axisymmetry in tokamaks, the MHD equilibrium equation, $\nabla p = \mathbf{J} \times \mathbf{B}$, can be transformed into a second-order, non-linear, elliptic partial differential equation for the flux function, $\psi = RA_\phi$, where A_ϕ is the toroidal component of the vector potential,

$$\frac{\partial^2 \psi}{\partial R^2} - \frac{1}{R} \frac{\partial \psi}{\partial R} + \frac{\partial^2 \psi}{\partial Z^2} = -\mu_0 R \left[R \frac{dp}{d\psi} + \frac{F}{\mu_0 R} \frac{dF}{d\psi} \right]. \quad (13)$$

This equation was independently obtained by Harold Grad and Vitaly Shafranov and, therefore, is named the Grad-Shafranov (GS) equation ([Mukhovatov and Shafranov, 1971](#)). The solution to this equation provides a 2-D map of the magnetic flux surfaces, such as that shown in [Fig. 9](#). It must be solved with either Dirichlet or Neumann boundary conditions specified along the boundary of a closed domain and, in some rather simple cases, this can be carried out analytically. However, in practical cases it is solved numerically, using robust schemes that have been developed over the years ([Jardin, 2010](#)). The source terms of the GS equation are two free functions: plasma pressure, $p(\psi)$, and poloidal current function, $F(\psi)$. While the pressure profile can be reasonably well specified, based on experimental data or transport modeling, specifying F is not straightforward. This difficulty arises from F often being nearly constant across the plasma and, therefore, $dF/d\psi$ is entirely dependent on small deviations of F away from constancy.

In axisymmetric confinement configurations, it is usual to specify the helicity of the magnetic field lines in each magnetic flux surface by the inverse of the rotational transform, through the parameter $q = 1/i$. This parameter is called the safety factor because its profile across the plasma plays a fundamental role on MHD stability. Rational surfaces correspond to $q = m/n$ and instabilities with this helicity can develop in their neighborhood, where the restoring force associated with the plasma displacement is reduced ([Goedbloed and Poedts, 2004](#)).

The plasma is confined within the flux surfaces and the last closed flux surface (LCFS) defines the shape of the plasma cross-section. The region outside the LCFS, denoted the scrape-off layer (SOL), is normally filled with a partially ionized

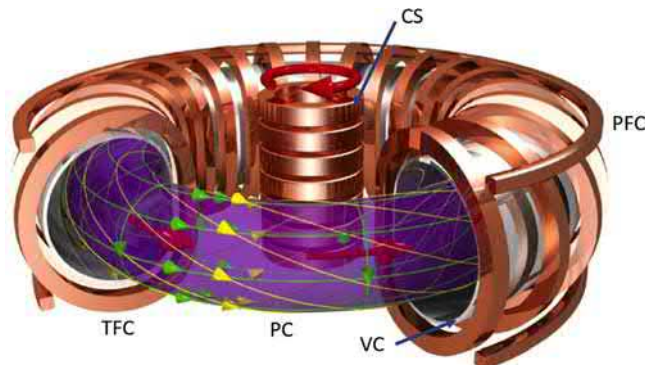


Fig. 8 (A) Sketch of a tokamak. Symbols are as follows: PFC, poloidal field coil; PC, plasma column; CS, central solenoid; and VC, vacuum chamber. Reproduced from Wischmeier M (2016) The tokamak principle-magnetic fusion. Presented at the Joint ICTP-IAEA, College on Advanced Plasma Physics, Trieste, Italy.

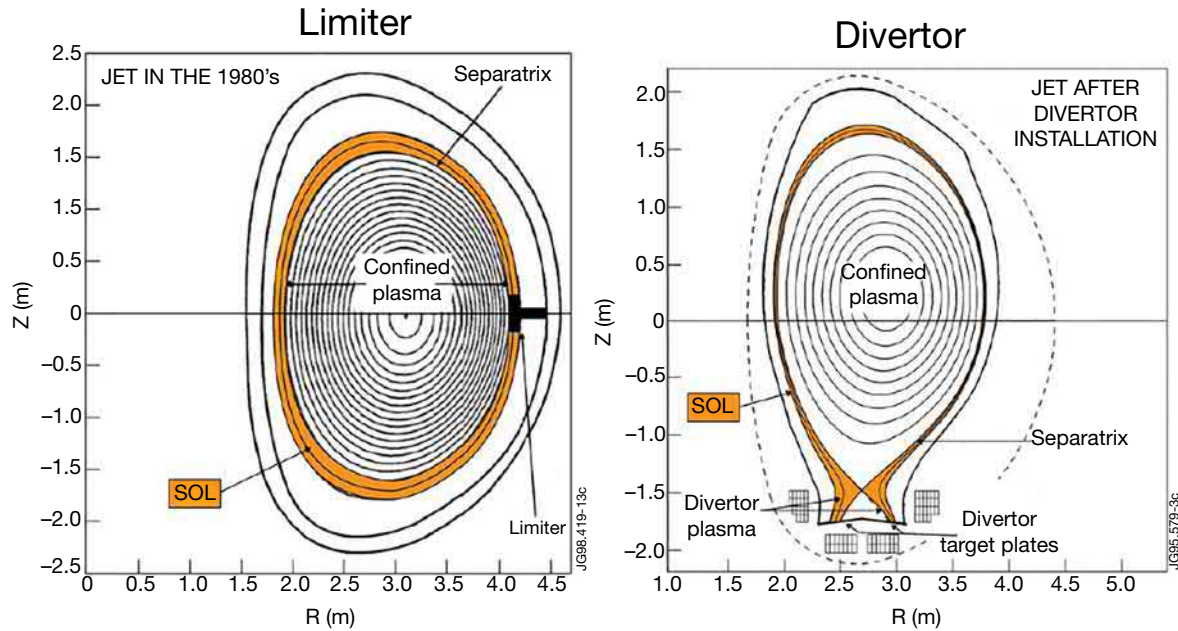


Fig. 9 Cross-section of the plasma column in the JET tokamak before and after the introduction of a magnetic divertor. Reproduced from Wischmeier M (2016) *The tokamak principle-magnetic fusion. Presented at the Joint ICTP-IAEA, College on Advanced Plasma Physics, Trieste, Italy.*

low-temperature plasma. The closed flux surfaces are not concentric with the magnetic axis because, due to toroidicity and pressure gradient, they displace outwards; the largest displacement being that of the magnetic axis with respect to the geometric center of the LCFS. This is called the Shafranov shift and is proportional to the plasma β (Mukhovatov and Shafranov, 1971). In the first tokamaks, the plasma cross-section was circular, and the last closed flux surface was established by contact of the plasma column with a material limiter, normally a carbon block. Later, it was realized that the stability of tokamaks to MHD modes is substantially improved if the plasma cross section is elongated in the vertical direction, assuming a D-shape (Freidberg, 1982).

In high-temperature and high-density plasmas, a large amount of power in energetic particles and radiation is transported across the magnetic surfaces and impinges upon the plasma-facing parts of the tokamak vacuum chamber. When material limiters are employed, this causes large erosion and the generation of impurities with a range of atomic numbers, depending on the material composition of the limiter and vacuum vessel walls. These impurities drift back into the plasma leading to increased line radiation, diluting the fuel and spoiling the fusion performance. To separate this source of impurities from the main plasma, a concept called magnetic divertor was suggested by Spitzer (1958). It is based on the introduction of a stagnation X-point in a flux surface external to the plasma column, so that the field lines become open in the SOL. This surface is dubbed the separatrix. The value of the safety factor diverges as one approaches the separatrix because, at the X-point, $\nabla\psi = 0$ and the magnetic field becomes purely toroidal, causing the rotational transform to vanish and the safety factor to diverge. Because of this, for gauging the stability of diverted tokamaks to kink modes, it has been established to replace the value of the safety factor at the LCFS by $q_{95} = q(\psi_N = 0.95)$ where ψ_N is the normalized flux function ($\psi_N = 0$ at the magnetic axis and $\psi_N = 1$ at the separatrix). In the SOL beyond the separatrix, energetic particles and heat are directed to divertor plates that can be designed to handle high heat loads and sputtering, the divertor plates.

After 1970, more advanced tokamaks were designed with non-circular plasma cross-section and a magnetic divertor. As an example, the largest tokamak in operation, the Joint European Torus (JET) (Euro-Fusion, n.d.), was initially designed with non-circular cross-section only and later modified to introduce a magnetic divertor. The change in the flux surfaces due to this modification can be seen in Fig. 9 (Wischmeier, 2016). Despite the reduction in plasma size and maximum plasma current, its performance with the magnetic divertor increased substantially, in particular regarding energy confinement. The introduction of a magnetic divertor not only allowed cleaner discharges, but led to a much better understanding of the relevance of edge-plasma phenomena for the performance of tokamak plasmas and also to the experimental discovery of an improved mode of operation - the high confinement mode (H-mode) (Wagner et al., 1982), in which the energy confinement time doubles with respect to that in the standard so-called low-confinement mode (L-mode). Although the transition from L- to H-mode still defies a self-consistent theoretical explanation, the H-mode has been reproduced in all major magnetic confinement devices, including stellarators and tokamaks, and with other plasma heating methods, such as RF heating (Gong et al., 2019).

Plasmas operated in the H-mode feature a steep pressure profile in the edge region, usually called the pedestal region, which leads to increased plasma density and temperature in the plasma core region. Therefore, the H-mode is characterized by an External Transport Barrier (ETB) close to the plasma edge. Besides the H-mode, other improved confinement regimes have also been observed, such as the Internal Transport Barrier (ITB). This typically occurs when the q -profile does not increase monotonically from the magnetic axis to the plasma boundary, presenting a minimum in an intermediate flux surface. This is called a

reverse-shear configuration, because the differential twist of the field lines change direction at this surface. For comparison, the plasma temperature profile in the different confinement regimes is illustrated in Fig. 10 (Zohm, 2015). Because the ITB is produced only transiently, the H-mode is seen as the most promising operational regime for an economically attractive fusion power plant. However, the H-mode is not a quiescent plasma regime due to the ubiquitous presence of repetitive instabilities known as edge localized modes (ELMs), which are driven by MHD instabilities termed peeling-ballooning (P-B) modes, triggered by the increased plasma pressure gradient and/or parallel current density in the plasma edge (Snyder et al., 2002).

A major achievement of tokamaks, not yet matched by any other magnetic confinement scheme, was the production of significant amounts of fusion power by the Tokamak Fusion Test Reactor (TFTR) and JET, operating with deuterium-tritium fueled plasmas. These two important tokamaks are shown in Fig. 11. While TFTR produced about 10.7 MW of fusion power (Bell, 2016a), JET retains the world record of peak fusion power produced, which is 16.1 MW. The promising results obtained in the last three decades in tokamaks led to the proposal of a prototype reactor, the ITER, that is being built in Saint-Paul-lès-Durance, France, by a consortium of countries, China, the European Union (including Switzerland), India, Japan, Russia, South Korea, and the United States. ITER will become the largest tokamak in the world and, among other goals, it was designed to operate in the H-mode, with $I_p = 15$ MA and $q_{95} = 3$, and to produce about 500 MW of fusion power, which will correspond to a D-T fusion power gain $Q_{DT} \geq 10$.

Conventional tokamaks usually have aspect ratio $A = R_0/a \approx 3 - 4$. However, during the 1980s, it was conjectured by two researchers from the Oak Ridge National Laboratory (ORNL), Ben Carreras and Tim Hender, that tokamaks of low-A, also termed spherical tokamaks (STs), could be more stable against MHD instabilities and should have a number of substantial practical

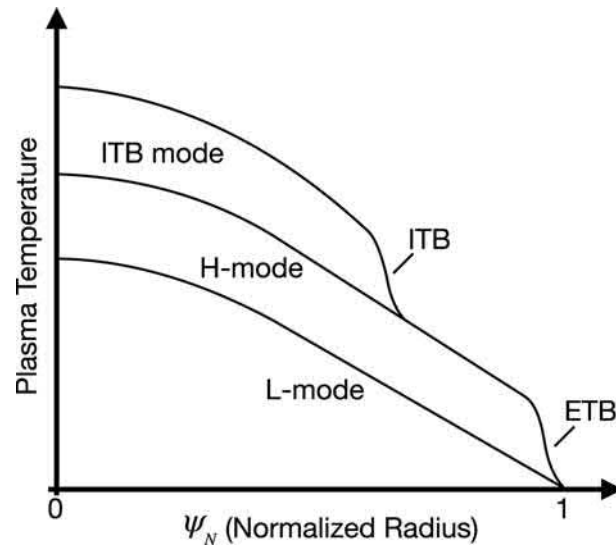


Fig. 10 Plasma temperature profiles in different confinement regimes, showing the internal (ITB) and external (ETB) transport barriers. Reproduced from Fig. 7.1 of Zohm H (2015) *Magnetohydrodynamic Stability of Tokamaks*. Wiley-VCH.

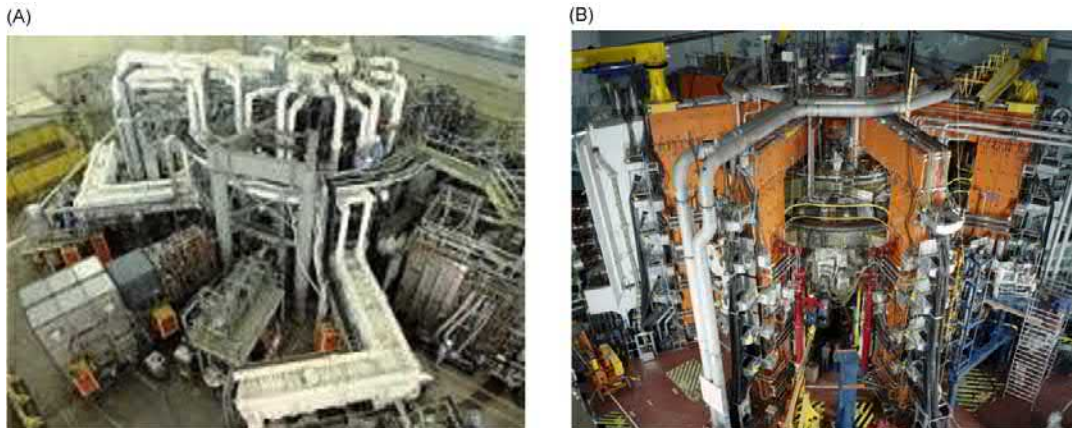


Fig. 11 (A) The TFTR, and (B) JET tokamaks. (A) Reproduced from Bell MG (2016a) *Chapter 6—The Tokamak Fusion Test Reactor*, pp. 119–166. Woodhead Publishing. <http://www.sciencedirect.com/science/article/pii/B9780081003152000064>, (B) reproduced from Euro-Fusion (n.d.) *The Joint European Torus*. <https://www.euro-fusion.org/devices/jet/>.

advantages over conventional aspect ratio devices (Peng and Strickler, 1986). In STs, the poloidal field in the outboard region is comparable with and larger than the toroidal field, while the fields in the inboard region are comparable. In addition, the toroidal circumference of the plasma in the outboard region is comparable with the poloidal circumference, but it is substantially shorter than the poloidal circumference in the inboard region. As shown in Fig. 12A, these two effects lead to highly pitched field lines in the outboard region and moderately pitched field lines in the inboard region, in comparison with those in conventional aspect ratio plasmas. As a consequence, most of the field line length is located in the inboard region, also called good magnetic field curvature region (good in terms of MHD stability), while only a small fraction of it is located in the outboard, also called bad magnetic field curvature. Theoretical studies indeed showed that not only the classic kink instability should be strongly suppressed by this improved average magnetic field curvature, but also ballooning modes, which are radially localized instabilities that develop predominantly perpendicular to the confining magnetic field, but whose perturbation has a finite parallel component, and that is driven by the plasma pressure gradient in regions of unfavorable magnetic field curvature. Due to its higher average pitched angles, and therefore higher values of safety factor, STs allow stable operation at higher plasma current and β . These properties, among others, suggested that low-A machines would not only be less expensive to build, but have better performance as well. The first ST to be built was the Small Tight Aspect Ratio Tokamak (START). It featured a minimum aspect ratio $A = 1.2$ and major radius $R_0 = 32.5$ cm. It was constructed at the Culham Centre for Fusion Energy and started to operate in January 1991. In its earliest operations, START plasmas showed improved plasma stability, as predicted by Peng and Strickler, and the maximum β -values, achieved with ohmic heating alone, were as high as those obtained on the DIII-D tokamak with NBI (~ 0.12) (Strait, 1994; Sykes, 1997). When neutral beam heating was added, the maximum values of β were about three times higher than those obtained in any conventional tokamak (Akers et al., 2002). In view of its improved performance and lower cost, with respect to conventional aspect ratio tokamaks, STs have attracted considerable interest within the fusion scientific community and a number of STs since been constructed, including MAST, NSTX, Pegasus, Globus-M, among others (see Fig. 12B for an image of a plasma in the MAST tokamak). However, despite the improvements in performance observed in STs, their development has been one generation behind that of conventional aspect ratio machines and their tightness makes it difficult to properly shield their central column against radiation damage in a fusion reactor. However, with the advent of high temperature superconductors, new technological routes have appeared and STs, as well as compact, high-field tokamaks, are again attracting great attention (Whyte, 2019).

Stellarators

In the stellarator concept, the helical magnetic field is entirely produced by coils placed external to the plasma column. Therefore, one of its advantages, with respect to tokamaks, is that it does not require a current to be induced in the plasma, so that it is intrinsically steady state. One scheme to accomplish this was originally proposed by Lyman Spitzer, in a meeting at the American Atomic Energy Commission in 1951. The basic idea was that the helicity of the field lines could be achieved by twisting the torus, around which the circular coils that produce the magnetic field are placed, into a figure-eight configuration, as sketched in Fig. 13A taken from a review on fusion research by Braams and Stott (2002). This scheme, dubbed the “Figure-Eight Stellarator,” was explained by Spitzer in a seminal paper published in 1958 (Spitzer, 1958). Some devices based upon this concept were actually built at the Princeton Plasma Physics Laboratory, but their first results were somewhat disappointing.

One of the drawbacks of the Figure-Eight stellarator concept is that the twisted magnetic axis, around which the helical field lines wind, lacks the underlying symmetry required for the existence of closed and nested magnetic surfaces. Therefore, any small perturbation driven by error fields can break up the closed magnetic surfaces, creating the so-called magnetic islands and stochastic regions, similar to the effect predicted by the KAM theorem in Hamiltonian Particle Dynamics (Boozer, 2005). Later, Spitzer conceived a simpler configuration, in which a planar toroidal magnetic field is produced by circular coils, as in a standard toroidal solenoid, and the helical field lines are produced by $\mathcal{L}$ helical dipole windings wound around the toroidal vacuum chamber, with toroidal periodicity number n , as sketched in Fig. 13B for $\mathcal{L} = 2$ and $n = 5$. This turned out to become the classical stellarator

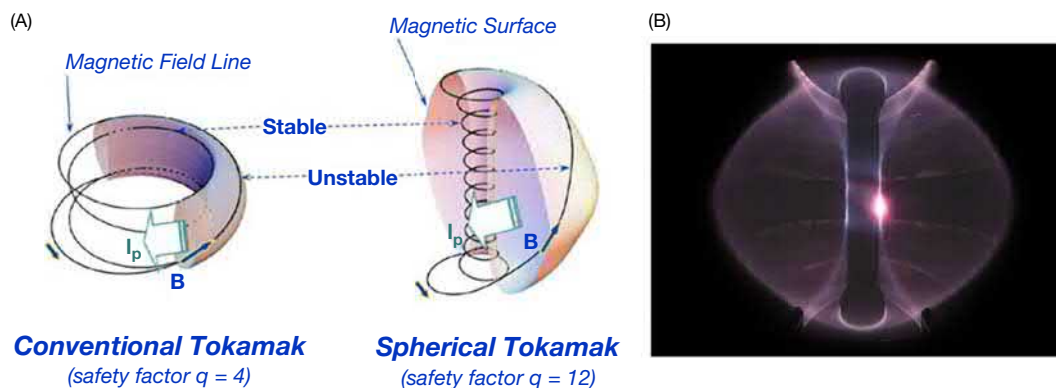


Fig. 12 (A) Schematic drawing showing a field line in the plasma edge region for two configurations: conventional aspect ratio and low aspect ratio. (B) Visible light image of a plasma discharge in the MAST spherical tokamak.

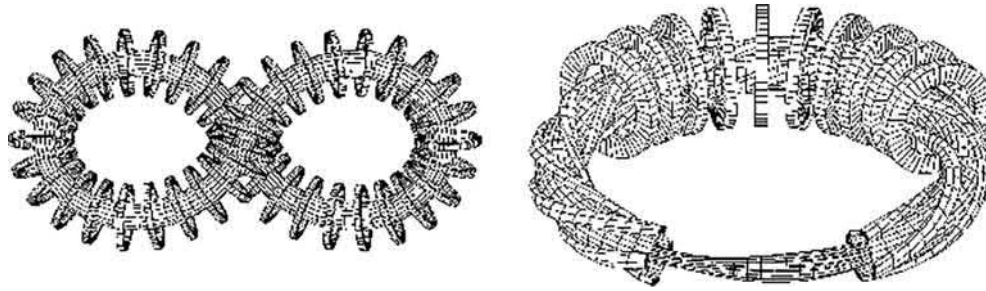


Fig. 13 (A) Sketch of the Figure-Eight stellarator and (B) the classical stellarator concept. Reproduced from Braams CM and Stott PE (2002) *Nuclear Fusion—Half a Century of Magnetic Confinement Fusion Research*. Bristol, UK: IOP Publishing.

concept, which was adopted in some later devices. The following explanation on how the nested magnetic surfaces are produced in this configuration has been extracted from the excellent introductory article on stellarators by Boozer (1998).

Let us start with the toroidal configuration shown in Fig. 6C, with closed magnetic surfaces. Most of the field lines do not close on themselves after any number of turns in the toroidal direction and continue to wind helically around the torus densely covering a magnetic surface. Those that close on themselves after an integer numbers of turns in both directions are called rational field lines. Let us consider a field line that closes on itself after one turn in the toroidal and poloidal directions, and let us now introduce a third current I_3 flowing in a helical coil coincident with it. The field produced by this current will disturb the field on the magnetic surfaces, breaking up the original field lines that approach the helical coil and splitting the magnetic surfaces to make a magnetic island, as shown in Fig. 14. The field lines inside this island form nested helical magnetic surfaces with no current inside.

The rotational transform defined in Eq. (12) plays a quite relevant role on the stability of plasmas confined in stellarators and it is a basic parameter to be specified in their design. It may appear somewhat surprising that the rotational transform of the field lines can be obtained inside a volume without currents, even close to the magnetic axis. The reader interested in looking deeper into this question is kindly directed to a review paper on stellarators by Helander (2014, 2018).

Another relevant quantity in the design of stellarators is the magnetic well, which is important for the stability of pressure-driven modes, as explained in “Magnetic confinement basics” section. In a closed magnetic configuration, topologically equivalent to a torus, it is not possible to have the plasma volume always facing convex field lines; in some regions the field lines will be convex and in others concave. The former are called good curvature regions and the latter bad curvature regions. Therefore, a magnetic well can be achieved only in an average sense, i.e., the average magnetic field strength on the flux surfaces increases with the enclosed volume, and it is defined as a dimensionless parameter on each flux surface,

$$\hat{f} = \frac{V}{\langle B^2 \rangle} \frac{d\langle B^2 \rangle}{dV} \quad (14)$$

The importance of a vacuum magnetic well depends on the profile of the rotational transform and, as the value of β increases, a magnetic well is self-generated by the plasma due to the Shafranov shift, becoming dominant over the vacuum one. Nevertheless, a vacuum magnetic well is realized in most stellarator designs.

The calculation of plasma equilibrium in three-dimensional stellarator configurations is not a trivial task. Because, in general, it is not possible to assure the existence of nested magnetic surfaces from the outset, the equilibrium MHD equations cannot be expressed in terms of an equation for a flux function ψ , as in the tokamak case. The usual procedure is then to assume their existence, at least approximately, over a large fraction of the plasma volume and to use a variational principle to minimize the total plasma energy, $W_p = \int (p/(\gamma - 1) + B^2/2\mu_0) dV$ over a toroidal domain. This is the method implemented in the widely used VMEC code (Hirshman et al., 1986).

Advanced equilibrium codes were not available at the dawn of stellarator research and several approximate methods were conceived to solve the equilibrium problem in terms of differential equations. One of the most successful schemes was developed

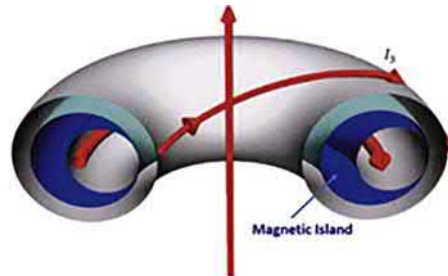


Fig. 14 Sketch of magnetic island with closed nested magnetic surfaces produced by the current configurations. Reproduced from Boozer AH (1998) What is a stellarator? *Physics of Plasmas*, 5(5):1647–1655. <https://doi.org/10.1063/1.872833>.

by Greene and Johnson using the so-called stellarator ordering (Greene and Johnson, 1961; Johnson and Greene, 1961). In this ordering, an assumption is made of a slender plasma column with planar magnetic axis, that is, with inverse aspect ratio $\epsilon = a/R_0 < 1$, where a and R_0 are the average minor and major radii of the plasma column. It is furthermore assumed that $\beta \sim \epsilon$ and $B_{\text{hel}}/B_{\text{tor}} \sim \sqrt{\epsilon}$, where B_{hel} is the characteristic magnitude of field generated by the helical windings and B_{tor} is the characteristic magnitude of the field produced by the toroidal solenoid. The periodic variation in the toroidal direction of the equilibrium quantities are averaged in the calculations.

Using this ordering, it is possible to qualitatively discuss one rather important issue in the MHD equilibrium of stellarators, i.e., the effect of the Pfirsch-Schlüter current (Hirsch et al., 2008). Its basic physics can be simply understood by calculating the expression for the component of the plasma current density perpendicular to the magnetic field, $J_{\perp}$. Taking the cross product $\mathbf{B} \times \nabla p$, and using the equilibrium equation, it follows that $\mathbf{J} = J_{\parallel} \mathbf{B}/B + \mathbf{J}_{\perp}$, where $\mathbf{J}_{\perp} = (\mathbf{B} \times \nabla p)/B^2$. The condition $\nabla \cdot \mathbf{J} = 0$ gives an equation for the component of the current density parallel to the field lines, $J_{\parallel}$,

$$\nabla \cdot \mathbf{J}_{\parallel} = \nabla \cdot \left(J_{\parallel} \frac{\mathbf{B}}{B} \right) = -\nabla \cdot \mathbf{J}_{\perp} = 2J_{\perp} \cdot \nabla \ln |B| \quad (15)$$

This current is called the Pfirsch-Schlüter current because they were the first to discuss its effect on stellarator equilibrium (Pfirsch and Schlüter, 1962). The Pfirsch-Schlüter current is essentially the current which flows along helical field lines to cancel the vertical separation of charges in a toroidal field. From this latter equation, it follows that the Pfirsch-Schlüter current is determined by the variation of the field strength on the magnetic surfaces. Doing the calculation in the stellarator ordering, one obtains $J_{\parallel}/J_{\perp} \approx 2 \cos(\theta)/\iota$ where $\iota = 1/q$ and θ is the poloidal angle (Miyamoto, 1978). The dipole structure of $J_{\parallel}$ gives rise to a vertical field that tends to displace the magnetic surfaces against each other. This is equivalent to the Shafranov shift Δ observed in tokamaks, but while in these devices it can be somewhat compensated by properly adjusting the equilibrium field and plasma current profile, in stellarators it can seriously affect the vacuum magnetic field configuration. Still within the stellarator ordering, and assuming that the cross-section of the plasma column is elongated by a factor $\kappa = b/a$, the expression for the shift is given by

$$\Delta = \frac{R}{\iota^2} \frac{\beta}{f(\kappa)}, \quad \text{where } f(\kappa) = \frac{\kappa + 1}{\sqrt{\kappa}} \quad (16)$$

Requiring $\Delta \leq a/2$ imposes a maximum limit on the value of β , which might be too low for a fusion reactor, if the configuration is not optimized. The Pfirsch-Schlüter current may also introduce a variation in the ι -profile, increasing its central value and decreasing its boundary value, with respect to the designed vacuum configuration. This is undesirable, because it can introduce low-order rational surfaces where localized instabilities and magnetic islands can develop, due to error fields. Therefore, optimized stellarator design aims at reducing $J_{\parallel}/J_{\perp}$ by decreasing the variation of $|B|$ on magnetic surfaces. There are four types of stellarator schemes being explored: classical stellarators (Fig. 15A), torsatrons/heliotrons (Fig. 15B) (Bell, 2016b), heliacs (Fig. 15C) (Fusionwiki, n.d.), and modular stellarators (Fig. 15D).

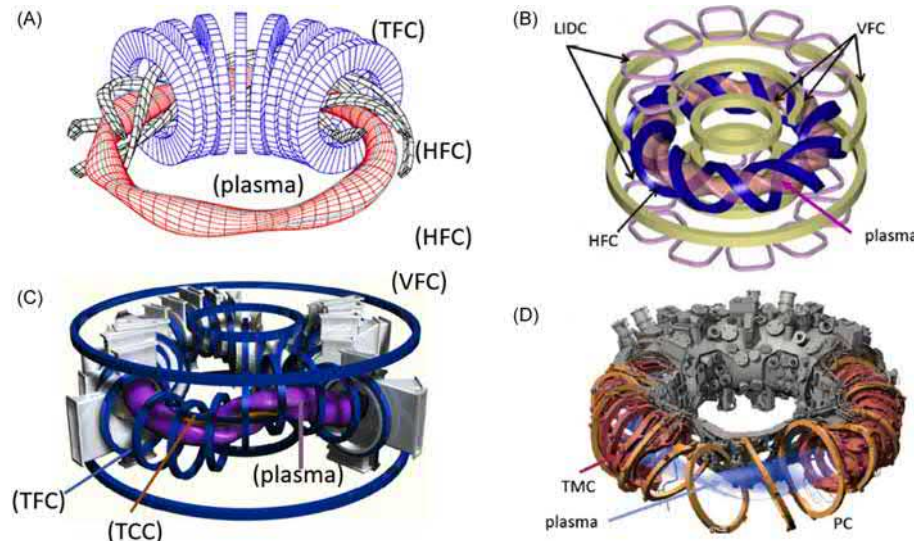


Fig. 15 Sketch of current stellarator concepts: (A) classical stellarator, (B) torsatron/heliotron, (C) heliac, and (D) modular stellarator. Symbols are as follows: HFC, helical field coil; LIDC, local island divertor coil; TCC, toroidal current coil; TFC, toroidal field coil; TMC, twisted modular coil; VFC, vertical field coil; PC-planar coil. (A) Reproduced from Hirsch M, et al. (2008) Major results from the stellarator Wendelstein 7-AS. *Plasma Physics and Controlled Fusion* 50(5): 053001. <https://doi.org/10.1088/0741-3335/50/5/053001>, (B) reproduced from Bell MG (2016b) *Chapter 15—The Tokamak Fusion Test Reactor*, pages 119–166. <http://www.sciencedirect.com/science/article/pii/B9780081003152000064>, (C) reproduced from Fusionwiki (n.d.) *The TJ-II Helicac*. <http://fusionwiki.ciemat.es/wiki/TJ-II>, (D) reproduced from Klinger T, Andreeva T and Bozhnikov S, et al. (2019) Overview of first Wendelstein 7-X high-performance operation. *Nuclear Fusion* 59(11): 112004. <https://doi.org/10.1088/1741-4326/ab03a7>.

Classical stellarators

In the classical stellarator scheme, the helical field is produced by a set of pairs of helical coils carrying currents in opposite directions, so that they produce no net toroidal field. The toroidal field is produced by a set of toroidal field coils, as in tokamaks. The number of dipole helical coils is denoted by $\mathcal{L}$ and the number of periods by n . The magnetic field configuration of Wendelstein 7-A, W7-A, a classical stellarator with $\mathcal{L} = 2$ and $n = 5$, is sketched in Fig. 15A. Other devices with different parameters have also been constructed, such as CLEO ($\mathcal{L} = 3$ and $n = 7$), at the Culham Laboratory United Kingdom, and L - M ($\mathcal{L} = 2$ and $n = 14$), at the General Physics Institute, Moscow, Russian Federation. Usually, devices with $\mathcal{L} \leq 2$ and low value of n have an almost uniform ι -profile, meaning that the shear of the field lines between the magnetic surfaces is small. However, configurations with high shear have also attracted interest, do to its effect on the stabilization of drift instabilities, which are driven by density and temperature gradients. That is the case of L-2 M. W7-A was the largest built classical stellarator ($R_0 = 2$ m, $a = 0.1$ m, $B_0 \leq 3.4$ T) and has produced quite outstanding results. Indeed, before W7-A, practically all stellarators operated with a small plasma current, to reproduce somewhat the better performance of tokamaks. The first demonstration of classical stellarator operation without a net toroidal current was demonstrated in W7-A for the first time (Grieger et al., 1985, 1986).

Torsatrons/heliotrons

In the torsatron/heliotron scheme, the magnetic surfaces are produced by helical windings with unidirectional currents (Gourdon, 1968; Uo et al., 1971). Therefore, the magnetic field has an $\mathcal{L}$ -fold symmetry and an average vertical field is also created. Depending on the winding law, the magnitude of this field can be large, so that additional compensating vertical field coils may be required. Usually, torsatrons do not have this extra coil system whereas heliotrons do have. The scheme shown in Fig. 15B corresponds to a heliotron (LHD).

The torsatrons were more extensively investigated in the National Science Center, at Kharkov, Ukraine, in a series of devices dubbed Uragan. The last of them, Uragan-3M, has the following parameters, $R_0 = 1$ m, $a = 0.12$ m, $B_0 \leq 1$ T, $\mathcal{L} = 3$, $n = 9$, $\iota(a) = 0.3$. The experimental results of this device are still modest, with high electron temperature obtained only in small central region of the plasma column, $r \approx 3$ cm, in which a system of nested magnetic surfaces exist. Different options are currently being investigated to enlarge this region, including forcing an inward shift of the magnetic axis (Moiseenko et al., 2019).

The heliotron concept was developed in Japan and intensively investigated in the National Institute for Fusion Science, where the world largest heliotron, Large Helical Device (LHD) was put into operation in 1998, Fig. 15B. This was the first helical device with superconducting coils. Its main parameters are $R_0 = 3.5$ m, $a = 0.6$ m, $B_0 \leq 3$ T, $\mathcal{L} = 2$, $n = 10$, $\iota(a) = 0.4 \rightarrow 1.4$. The variation of the rotation transform, from the magnetic axis to the plasma boundary implies a relatively large shear of the magnetic field lines, in distinction with the low-shear option of classical stellarators. LHD has also a set of local islands divertor coils (LIDC in Fig. 15B), which allow for the production of localized magnetic divertors, for particle and energy exhaust, as in advanced tokamaks.

Particularly good results were obtained in LHD, reaching ion and electron temperatures, $T_i \approx 5.6$ keV and $T_e \approx 10$ keV, plasma density $n \approx 2 \times 10^{19} \text{ m}^{-3}$, $\beta \approx 0.05$ and energy confinement time, $\tau_E \approx 0.2$ s. In 2017, LHD started to operate with fully deuterium plasmas and the isotope dependence of the plasma performance, in particular regarding energy confinement, is being carefully investigated as a basic information for the design of fusion reactors based on this concept (Tanaka et al., 2019).

Heliacs

The heliac concept is based upon a clever arrangement of the field coils that substantially simplify their assembly and maintenance and allows additional flexibility for adjusting the value of the rotational transform ι . Circular coils are arranged helically around a central circular coil. The plasma column snakes around the latter coil, so that the magnetic axis itself has a helical structure, facilitating the production of the rotational transform (Boozer et al., 1982). The major devices built on the basis of this concept are H-1, at Canberra, Australia (Blackwell et al., 2006) and TJ-II, at CIEMAT, Madrid, Spain (Sanchez et al., 2007).

In TJ-II, Fig. 15C, there are 32 toroidal coils, two extra helical coils wound around the central conductor, and vertical field coils added for better plasma position control. The extra helical coils provide improved flexibility to control the rotational transform profile (Harris et al., 1985). The main parameters of the device are $R_0 = 1.5$ m, $a = 0.2$ m, $B_0 \leq 1$ T, $n = 4$, $\iota(a) = 0.9 \rightarrow 2.5$. In electron-cyclotron heating discharges, plasmas with $T_i \approx 100$ eV and $T_e \approx 1$ keV, and $n \approx 1 \times 10^{19} \text{ m}^{-3}$ were regularly produced. Although these parameters seem somewhat modest, many fusion relevant basic physics topics were thoroughly investigated in TJ-II, in particular, regarding sheared flows and turbulence with long-distance correlation at the plasma edge (Pedrosa et al., 2008).

Regarding the suitability for a fusion reactor, a major challenge with the heliac concept is that the highly interlinked coil system may not be acceptable, due to the necessity of introducing the blanket and radiation shielding between the plasma column and the coils.

Modular stellarators

The modular stellarator scheme was originally proposed by Wobig and Rehker (1972). The idea behind this concept was to reverse the systematic approach normally adopted in conventional stellarator design, which involved starting with the design of the coil system to achieve the desired requirements of the vacuum magnetic configuration and then determining the characteristics of the plasma in MHD equilibrium within this configuration. Instead, the approach used in stellarator optimization is to firstly determine the shape of a closed flux surface for the plasma boundary, which produces an internal configuration with the desired equilibrium properties, such as rotational transform, average magnetic well, average value of β , closed magnetic surfaces and small error

fields, minimization of Pfirsch-Schlüter current, improved confinement of fast ions, etc. Then, calculate the surface current distribution on this surface and the vacuum field in its neighborhood. Finally, find a set of modular field coils that can reproduce it. Naturally, there is no guarantee of the existence of a finite number of isolated coils that can reproduce the required field on the plasma surface and the solution of this complex inverse problem became possible only after the advent of advanced supercomputers (Grieger and Milch, 1993).

The practicability of the modular scheme was tested in the W7-AS device with great success. Stable plasmas were obtained in quasi-steady state at values of $\beta \approx 0.034$, electron temperature $T_e \approx 6.8$ keV at average density $n \approx 2 \times 10^{19} \text{ m}^{-3}$, central ion temperature $T_i \approx 1.7$ keV and energy confinement time $\tau_E \approx 60$ ms (Hirsch et al., 2008). These outstanding results set the ground for the development of W7-X, which is the largest stellarator currently in operation (Gasparotto et al., 2005). The W7-X stellarator was developed at the Max-Planck-Institut für Plasmaphysik, in Greifswald, Germany, and produced its first plasma discharge in 2015 (Wolf et al., 2017). Its main parameters are $R_0 = 5.5$ m, $a = 0.55$ m, $B_0 \leq 2.5$ T, with mild shear, i.e., the rotational transform varying from $\iota(0) = 0.8 \rightarrow \iota(a) = 1.0$. Fifty twisted modular superconducting coils are used to produce the helical field, in five magnetic field modules, setting the periodicity in the toroidal direction, Fig. 15D. Since the field produced by these coils has fixed rotational transform, 20 planar and tilted superconducting coils are assembled over the modular coils, to allow the radial adjustment of the plasma column and modification of the rotational transform, so that many magnetic configurations can be realized. A special feature of W7-X is the availability of localized magnetic island divertors, a concept tested in W7-AS, for controlled power and particle exhaust. The results of W7-X in the first phase of operation with these divertors have been recently reported (Wolf et al., 2019). High density and high performance plasmas were produced, with the following parameters $n \approx 8 \times 10^{19} \text{ m}^{-3}$, $T_e \approx T_i \approx 3.4$ keV, $\beta \approx 0.05$, $\tau_E \approx 0.235$ s. In its peak performance, W7-X achieved a fusion parameter $n\tau_E T_i \approx 6.4 \times 10^{19} \text{ m}^{-3} \cdot \text{keV} \cdot \text{s}$, setting the current record for stellarators (Klinger, 2018; Klinger et al., 2019).

Quasisymmetric configurations

Recently, there has been renewed and intense activity related to the development of “quasisymmetric stellarators” (Helander, 2014). In a quasisymmetric configuration, the variation of $|B|$ along the field is the same for all field lines on a magnetic surface. In such a configuration, the radial drift of particles trapped in local minima of $|B|$ would be reduced, improving confinement, and, as already mentioned, the Pfirsch-Schlüter current would be minimized. Hence, if non-axisymmetric toroidal magnetic fields with quasisymmetry can be produced with technically viable coil systems, these configurations offer the possibility of developing stellarators with stable and efficient high-temperature plasma confinement.

Reversed-field pinch

As mentioned in “Magnetic confinement basics” section, one major instability in current-carrying plasmas is the kink instability, which produces helical distortions of the plasma column. A particular class of these modes are called external kink modes because the rational surface in which the helicity of the equilibrium field lines matches that of the perturbation lies outside the plasma column (Lao, 2020). The most worrisome of them is the $m = 1$, $n = 1$ mode, which corresponds to a helical perturbation that closes on itself after one turn in both poloidal and toroidal directions. Because of this instability, tokamaks normally operate with $q_a > 2$, to avoid the most dangerous modes. Since the poloidal field at the edge of the plasma column is proportional to the plasma current, $B_a \propto I_p$, it follows from Eq. (5) that the value of q_a decreases as the plasma current increases. What would happen then, if the value of the toroidal magnetic field, $B_T = B_\phi \hat{e}_\phi$, was held fixed and the plasma current rapidly increased, so that the $q_a = 1$ threshold was crossed from above? In a standard tokamak, this would trigger a violent instability that would disrupt the plasma column. In the Reversed-Field-Pinch, RFP, configuration, this evolution is prevented by the presence of a highly conductive shell closely fitting the vacuum chamber, which acts as a toroidal flux conserver and imposes that the perpendicular magnetic perturbation associated with the mode vanishes at the boundary, at least for a time of the order of the resistive skin time. A large turbulent state then sets in at the plasma edge, and through a nonlinear process called the dynamo effect, associated with a coherent combination of magnetic field and fluid velocity perturbations, an average poloidal current is driven inside the plasma boundary, reversing the edge toroidal magnetic field.

The first experiments on the pinch effect were initiated over 80 years ago (Bennett, 1934). The first pinches were highly unstable but, after the theoretical work on the kink mode, the concept of toroidal stabilized pinch evolved, with the introduction of the stabilizing toroidal magnetic field. With this scheme, grossly stable toroidal plasma columns were produced, but still with many residual instabilities and poor particle and energy confinement (Bodin and Newton, 1980). Further theoretical work indicated that, when current and pressure gradients were considered, many other unstable modes could be excited and, to avoid them, the equilibrium magnetic field lines had to be highly sheared (Goedbloed and Poedts, 2004). In a pedestrian explanation, this condition can be understood by considering the tension of the magnetic field lines embedded in a conducting fluid. When a fluid perturbation distorts the field lines, there is a restoring magnetic force due to the magnetic field tension. If the equilibrium field lines are shearless, the restoring force is minimized, and the perturbation can protrude across them without much resistance. However, if the equilibrium magnetic field lines are highly sheared, the perturbation, which is excited on a rational surface, encounters an increasing restoring force as it moves away from it and can be avoided if the shear of the field lines is sufficiently high. Actually, experimental data from the Zeta experiment indicated that, in some cases, and after an initial highly turbulent phase, the plasma spontaneously evolved to a short quiescent phase with quite high shear and reversal of the toroidal magnetic field at the plasma boundary, a RFP configuration (Bodin and Newton, 1980). A theoretical model to explain the conditions under which the self-reversal occurs, and the quiescent phase is achieved, was put forward by J.B. Taylor in a seminal publication (Taylor, 1974).

During the turbulent phase, resistive instabilities tear the field lines apart and reconnect them, changing the magnetic topology. This effect and microturbulence forces the plasma to relax towards an equilibrium state. The basic argument of Taylor was that, if the dissipation is weak, so that the plasma is not too far from a perfectly conducting fluid, and in the presence of an external conducting shell, the final equilibrium state can be derived by minimizing the total plasma energy, W_p , subjected to the constraint that the magnetic field helicity K over the total plasma volume remains invariant, that is,

$$K = \int \mathbf{A} \cdot \mathbf{B} \, dV = \text{const} \quad (17)$$

where $\mathbf{A}$ is the vector potential and the integral is over the entire plasma volume. The result of the minimization procedure is that the relaxed equilibria must be in a force-free state, satisfying the equilibrium equation $\nabla \times \mathbf{B} = \mu \mathbf{B}$ where μ is a constant with the same value on all field lines. Solving this equation in the cylindrical approximation for the toroidal plasma column, i.e., $A = R_0/a > 1$, $B_\phi = B_z(r)$; $B_\theta = B_\theta(r)$, and $B_r = 0$, one obtains that the equilibrium magnetic field components are given by Taylor and Newton (2015).

$$B_z = B_0 J_0(\mu r) \quad \text{and} \quad B_\theta = B_0 J_1(\mu r) \quad (18)$$

where $J_n(\mu r)$ are Bessel functions. It follows from this solution that the condition for spontaneous reversal of the toroidal field is $\mu a \geq 2.4$. These theoretical profiles are well reproduced by the experimental ones, as shown in Fig. 16A (Bodin, 1990; Marrelli et al., 2020).

Unfortunately, in the first RFPs, the minimum energy state achieved after the turbulent phase did not have good energy confinement, over most of the plasma cross section, and the values of relevant plasma parameters, such temperature and density, remained rather low as compared to the results obtained in tokamaks. The main reason for this is associated with the profile of the safety factor, which, in the cylindrical approximation, can be written as $q = r B_z / (R_0 B_\theta)$. Therefore, for the Taylor relaxed state, its value at the magnetic axis is $q_0 = 2/(\mu R_0) = 2a/(2.4 R_0)$ in the condition of field reversal. For values of the aspect ratio of standard RFPs, $R_0/a \approx 4$, as in RFX-mod, one obtains $q_0 \approx 0.2$, and the q -profile decreases monotonically to zero at the plasma boundary, as depicted in Fig. 16B. Consequently, inside the plasma, the rational surfaces $q = m/n$ occur for $m = 1$ and monotonically increasing large values of the toroidal mode number n , so that they tend to get densely crammed towards the plasma boundary. Due to the finite resistivity of the plasma, an instability called tearing mode develops at these internal rational surfaces (Lao, 2021), breaking up magnetic field lines and reconnecting them to form helical magnetic islands, as the one sketched in Fig. 14. A spectrum of these modes, with different values of toroidal mode numbers, then emerges, with nonlinear saturated amplitudes $\tilde{b}/B_\theta = 0.01$. It turns out that the associated fluid velocity perturbations, $\tilde{\mathbf{u}}$, have a non-zero correlation with the magnetic field perturbations, giving rise to a mean field electric field $\langle \tilde{\mathbf{E}} \rangle = -\langle \tilde{\mathbf{u}} \times \mathbf{b} \rangle$, which drives the poloidal current responsible for reversing the toroidal field. This regime is called a Multiple Helicity (MH) state. Unfortunately, the magnetic islands tend to overlap close to the plasma boundary in this regime, leading to a large region of stochastic magnetic field lines, spoiling confinement and increasing transport.

For many years, this undesirable feature was considered intrinsic to the dynamo self-organization process, which sustains the field reversal in quasi-steady state. A change of paradigm occurred as result of advanced numerical simulations and substantial improvement on systems to actively control the magnetic field perturbations at the plasma boundary. From these efforts, the possibility of a chaos-free configuration emerged, where the dynamo effect is provided by nonlinear evolution of the innermost $m = 1$ resistive mode that dominates over the others. Quasi-Single Helicity (QSH) states were first obtained in different RFPs, about 20 years ago (Escande et al., 2000). An example of a transient QSH is shown in Fig. 17. The formation of a predominant magnetic island is quite evident in the tomographic image of soft X-ray emission. The MH state is characterized by the poloidally symmetric emissivity in a central region, where enhanced energy and particle transport are indeed compatible with strong magnetic chaos. The QSH state, on the other hand, is characterized by a hot $m = 1$, $n = 8$ island and the spectrum predominance of this mode is evident from measurements made with pick-up coils outside the plasma.

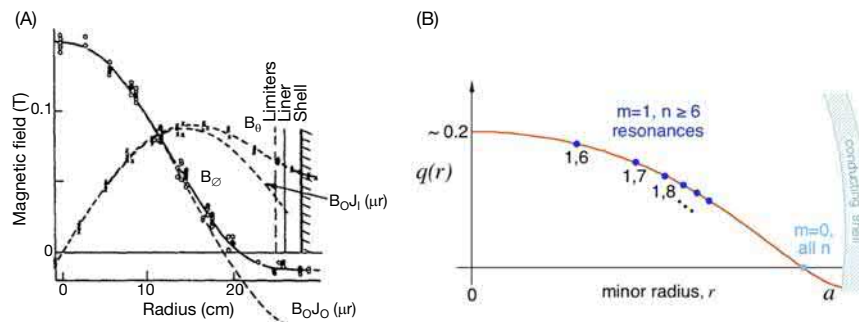


Fig. 16 (A) Experimental magnetic field profiles measured in the quiescent phase of HBTX-1A discharges and theoretical predictions based on Taylor's model. (B) Typical q -profile in an RFP discharge. (A) Reproduced from Fig. 1A of Bodin HAB (1990) The reversed field pinch. *Nuclear Fusion* 30(9):1717–1737. <https://doi.org/10.1088/0029-5515/30/9/005>, and (B) was reproduced from Fig. 2.3 of Marrelli L, Martin P, Puiatti M, Sarff J, Chapman B, et al. (2020) The Reversed Field Pinch. <https://hal.archives-ouvertes.fr/hal-02505414>.

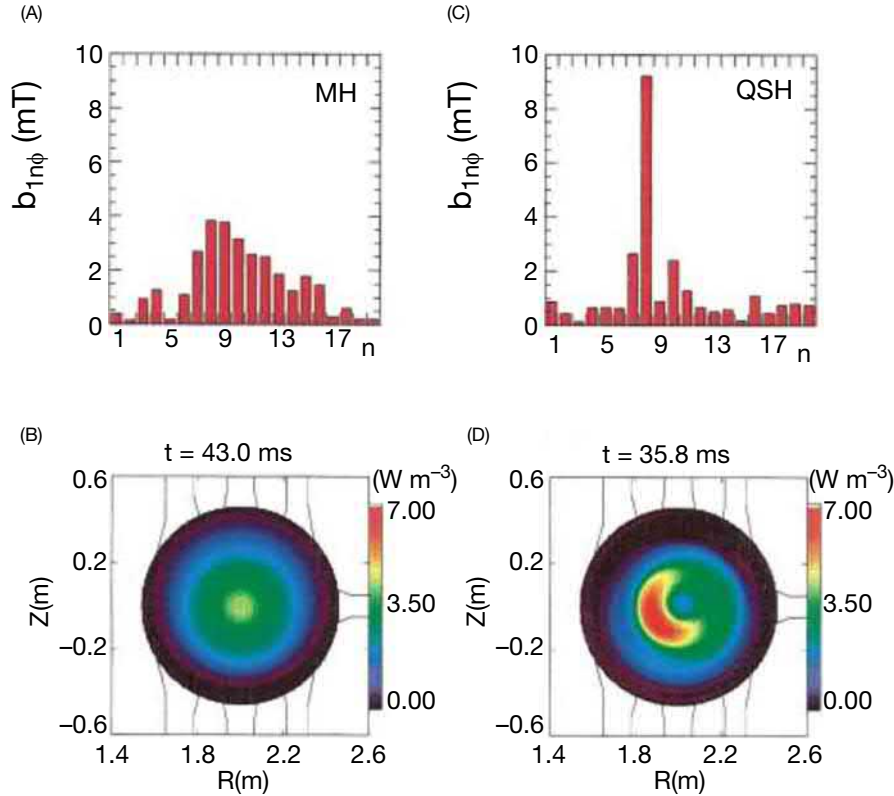


Fig. 17 Comparison between MH and QSH states in RFX discharges. (A) Spectrum of $m = 1$ modes and (B) corresponding tomographic SXR images during a MH state (RFX pulse No. 8818, $t = 43$ ms). R is the radial coordinate measured from the torus major axis. (C and D) The same quantities during a spontaneous QSH state (No. 11407, $t = 35.8$ ms) with dominant mode $m = 1$, $n = 8$. Reproduced from Fig. 1 of Escande DF, Martin P, Ortolani S, Buffa A, Franz P, Marrelli L, Martinez E, Spizzo G, Cappello S, Murari A, Pasqualotto R and Zanca P (2000) Quasi-single-helicity reversed-field-pinch plasmas. *Physical Review Letters* 85: 1662–1665. <https://link.aps.org/doi/10.1103/PhysRevLett.85.1662>.

A further improvement in producing chaos-free configurations was achieved in the RFX-mod device, by employing a quite advanced feedback system to control the radial component of the magnetic field perturbations at the plasma boundary and operating at high plasma currents $I_p \geq 1$ MA (Lorenzini et al., 2009). According to the authors, as the current is raised, the QSH state becomes purer and, above a threshold of about 4% in the dominant mode normalized amplitude, the separatrix X-point of its magnetic island merges with the main magnetic axis, and the two disappear. The former island O-point becomes the only magnetic axis, and a helical plasma column is formed, with a helical magnetic axis. This new regime, dubbed Single-Helical-Axis (SHAx) state, besides being an outstanding advancement in fusion research, certainly can be considered a paradigmatic example of self-organization in a complex physical system. In the SHAx state, the anomalous transport is substantially reduced, in comparison with the MH state, and reasonable values of the plasma parameters are obtained, as shown in Fig. 18.

The achievement of the QSH and SHAx states has opened new perspectives for the viability of the RFP magnetic confinement configuration for a fusion reactor, somewhat bridging the tokamak and stellarator concepts. Furthermore, the development of successful active control of radial magnetic perturbations at the plasma boundary made it less demanding the requirements for the external conducting shell, what simplifies practical reactor design. The RFP devices currently in operation and their main parameters are shown in Fig. 19.

Compact tori

From an engineering perspective, it would be quite attractive to have a fusion device without toroidal field coils interlocking with the plasma column, so that the plasma would be contained in a simply connected vacuum vessel. Such configurations with toroidal and poloidal magnetic fields of the same order, as in RFPs, are called spheromaks. The configurations without a toroidal field are called Field-Reversed Configurations (FRC). Sketches of the two configurations and associated fields are shown in Fig. 20 (Woodruff et al., 2010).

The spheromak resembles a compact RFP and indeed the plasma is normally formed in a relaxation process, preserving magnetic helicity, towards a Taylor equilibrium state, where the plasma current is nearly parallel to the magnetic field lines. Therefore, it is expected to be globally stable to MHD modes, although error fields and magnetic perturbations can lead to field line stochastization

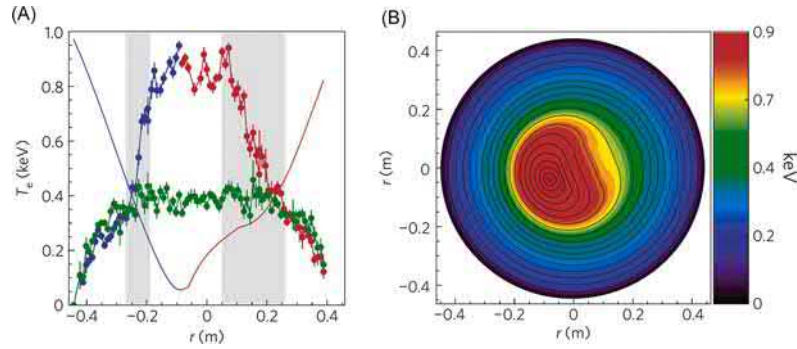


Fig. 18 (A) Typical electron temperature profile measured in $I_p \approx 1.5$ MA discharges, when the plasma is in a SHaX state. A high-temperature ($T_e = 0.8 - 0.9$ keV) region is present in the core plasma and extends on both sides of the geometric axis. The profile is asymmetric with respect to the geometric axis; however, the gradients in the strong gradient regions (shaded) are different on the two sides. Red and blue refer to the two opposite sides with respect to the helical magnetic axis. A typical profile in the MH state is shown in green for reference. (B) Reconstruction of the full 2D map of the temperature on a poloidal plane, obtained from the dependence of on the geometric coordinates. Reproduced from Fig. 3 of Lorenzini R, Martines E and Piovesan P, et al. (2009) Self-organized helical equilibria as a new paradigm for ohmically heated fusion plasmas. *Nature Physics* 5(8): 570–574. <https://doi.org/10.1038/nphys1308>.

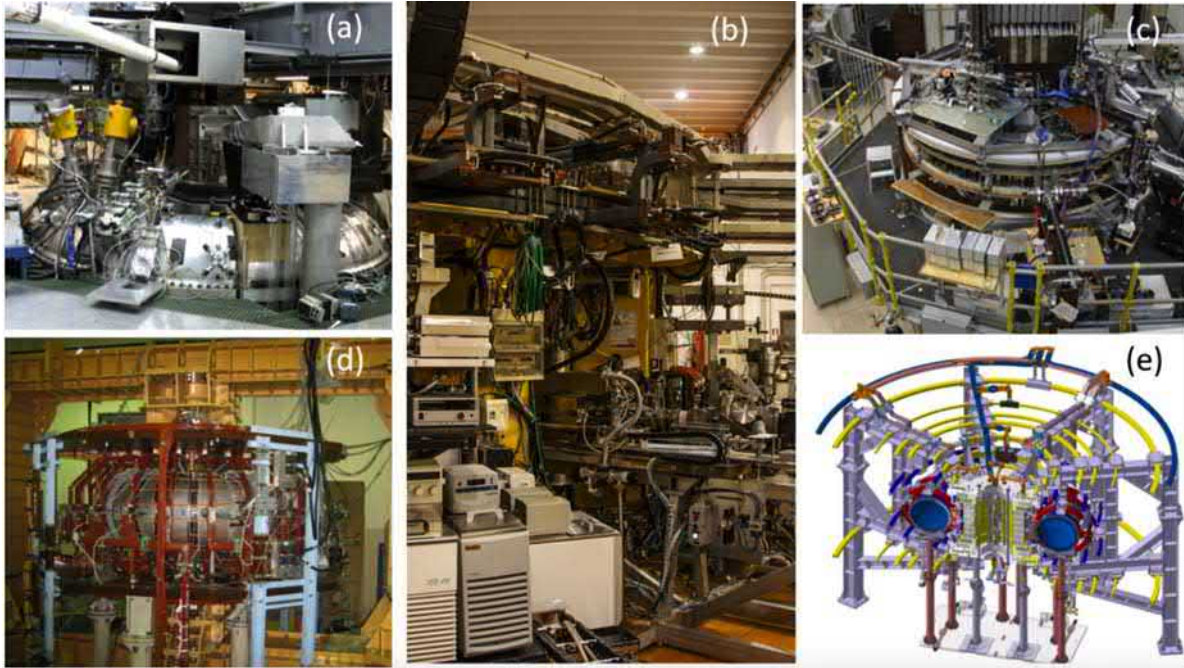


Fig. 19 Main RFP devices currently in operation. The parameters are given in following units: $A = R_0(m)/a(m)$, $I_{p, max}(MA)$, $k_B T_e(keV)$, $n_e(10^{19} m^{-3})$. (A) MST: $A = 1.5/0.5$, $I_{p, max} = 0.6$, $k_B T_e = 2$, $n_e = 1 - 3$. (B) RFX-mod: $A = 2/0.46$, $I_{p, max} = 2$, $k_B T_e = 1.2$, $n_e = 1 - 8$. (C) EXTRAP-T2R: $A = 1.24/0.18$, $I_{p, max} = 0.3$, $k_B T_e = 0.3$, $n_e = 0.5 - 2$. (D) RELAX: $A = 0.5/0.25$, $I_{p, max} = 0.12$, $k_B T_e = 0.3$, $n_e = 0.2 - 3$. (E) KTX: $A = 1.4/0.4$, $I_{p, max} = 0.5$, $k_B T_e = 0.3$, $n_e = 1 - 10$.

and loss of good confinement. The FRC, on the other hand, has a weak or no toroidal field, and the cross-section of the plasma column is bounded by a separatrix, with two hyperbolic stagnation points, outside which the field lines are open, and with an elliptic stagnation point in the confinement region. Contrary to the spheromak, this configuration is highly unstable to MHD instabilities and must be kinetically stabilized by highly energetic ions.

Research on spheromaks, although not in the front line of fusion research, has attracted much attention in recent years as its basic physics is akin to astrophysical processes, such as coronal mass ejection in the inner heliosphere (Singh et al., 2020). Stabilized FRCs are still being pursued as a viable fusion scheme because they can be translated along the axis of a linear configuration, merged, and adiabatically compressed (Binderbauer et al., 2010). Therefore, they are quite suitable for pulsed, high density approaches to fusion.

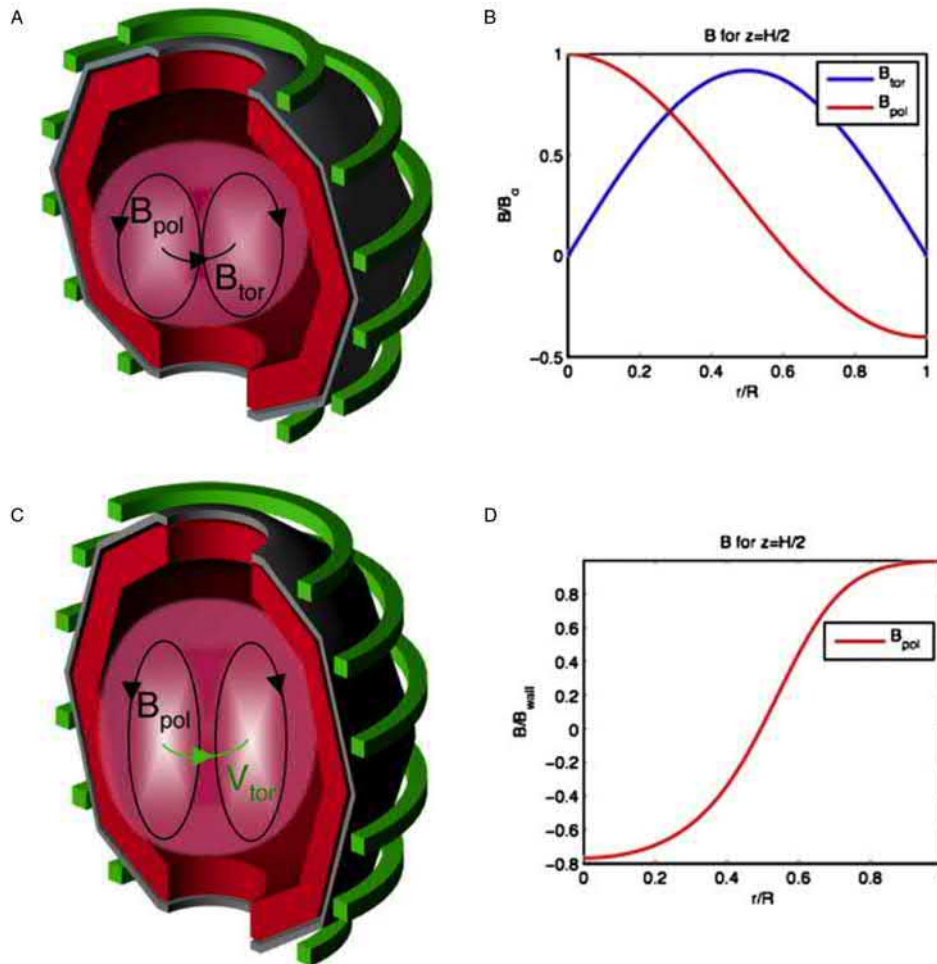


Fig. 20 Sketches of the magnetic configurations and the associated magnetic fields of (A and B) spheromaks and (C and D) FRCs. Reproduced from Fig. 5 of Woodruff S, Brown M, Hooper EB, Milroy R and Schaffer M (2010) Why compact tori for fusion? *Journal of Fusion Energy* 29(5):447–453. <https://doi.org/10.1007/s10894-010-9303-1>.

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Magnetic Confinement Fusion—Plasma Theory: Kinetic and Fluid Analysis

Yanick Sarazin, CEA, IRFM, Saint-Paul-lez-Durance, France

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Glossary

Langmuir waves Langmuir waves (also called plasma waves) are longitudinal electrostatic waves that propagate in a plasma because of variations in plasma electron density. Their frequency is close to the electron plasma frequency ω_{pe} . When accounting for electron temperature, their wavelength and frequency obey the Bohm-Gross dispersion relation, leading to wavelengths of order of the Debye length λ_{De} .

Adiabatic theory Adiabatic theory deals with systems characterized by variables which evolve slowly as compared to others. Perturbation theory can then be used to derive so-called adiabatic invariants, i.e. variables which remain approximately constant when changes occur slowly. In controlled fusion plasmas, electrostatic turbulence develops on time scales much longer than the ion (hence electron) cyclotron motion. The associated adiabatic invariant is the magnetic moment $\mu_s = m_s v_{cs}^2 / 2B$ with v_{cs} the velocity associated to the cyclotron motion.

Frenet frame The Frenet frame is an orthogonal basis associated to any continuous and differentiable curve in the 3-dimensional Euclidian space. Its three basis vectors are the tangent **T**, the normal **N** and the binormal **B** (Fig. 5). **T** points in the direction of motion. **N** relates to the derivative of **T** with respect to the curvilinear coordinate s , i.e. the arclength parameter of the curve. **B** is the cross product of **T** and **N**. The Frenet-Serret formulas state that $d\mathbf{T}/ds = \mathbf{N}/R_C$, $d\mathbf{N}/ds = -\mathbf{T}/R_C + \mathbf{B}/R_T$, $d\mathbf{B}/ds = -\mathbf{N}/R_T$, with R_C and R_T the curvature and torsion radii, respectively.

Improved confinement regimes Improved confinement regimes refer to plasma scenarios where the energy confinement time is enhanced as compared to the naturally turbulent plasma state, as a result of plasma self-organization. They are associated with the development of weaker turbulent cross-field transport—usually of density, temperature or both—on a radially localized region. These regions are characterized by transport barriers, i.e. large gradients of density and/or temperature. Several physical parameters have been identified as key to bifurcate towards such regimes, such as a sheared radial electric field, the twist of helical field lines (the safety factor) or the ion to electron temperature ratio.

Lagrangian (or material) derivative The Lagrangian (or material) derivative deals with systems which are subject to a space-and-time-dependent macroscopic velocity field $\mathbf{u}(\mathbf{x}, t)$. It is defined as the rate of change with time of some physical quantity (denoted Q hereafter, e.g. density, velocity, pressure, etc.) in the co-moving frame of the flow, i.e. when moving along with the fluid. It reads as follows: $dQ/dt = \partial Q/\partial t + \mathbf{u} \cdot \nabla Q$. Conversely, the Eulerian time derivative, which reduces to $\partial Q/\partial t$, is taken in the laboratory frame.

Fokker-Planck operator A Fokker-Planck operator refers to the partial differential equation which obeys the probability density function of a particle velocity or position under the influence of drag and random forces, as in Brownian motion. In hot and dilute plasmas, it describes the general form taken by the Coulomb collision operator, made of both convection and diffusion terms in the velocity phase space.

Grad-Shafranov equation The Grad-Shafranov equation derives from the static magneto-hydrodynamical (MHD) equilibrium of axi-symmetric tokamak plasmas, balancing the pressure gradient with the Lorentz force. It relates a generalized Laplacian of the poloidal magnetic flux—which labels magnetic flux surfaces—to the pressure gradient and to the stream function of the plasma current and its derivative.

Chapman-Enskog theory The Chapman-Enskog theory provides an asymptotic closure scheme for neutral gas dominated by collisions. It proceeds by expanding the distribution function, order by order, in the small parameter defined by the ratio of the collisional mean free path over the variation length scale of macroscopic quantities.

Markovian processes Markovian processes characterize the time evolution of systems in which the transition probability from one configuration in the phase space to another only depends on these two configurations themselves, and not of the past history of the system. Brownian motion—which characterizes the random motion of particles suspended in a fluid resulting from their collision with the fast-moving molecules in the fluid—is an example of such a process.

Onsager symmetry Onsager symmetry, or Onsager reciprocal relation, states that, in systems at local thermodynamical equilibrium, certain pairs of flows and forces are related by the same coefficients. When thermodynamical forces of the system are properly identified—such that the entropy production rate is the sum of the scalar products of these forces with their associated flows—the transport matrix which relates flows to forces is then symmetrical.

Introduction

This chapter introduces the basic characteristics of hot magnetized plasmas encountered in controlled fusion devices. Section “**Introduction**” reviews the typical length and time scales for ITER plasmas as an example, highlighting the existence of several small dimensionless parameters. Section “**Characteristic length and time scales in fusion plasmas**” explains why particle trajectories are well confined in the magnetic configuration of tokamaks. Especially, fast and slow motions are distinguished within the frame of the adiabatic theory, the latter leading to secular drifts transverse to the magnetic field lines. Two classes of particles are also identified, passing and trapped. Section “**Particle orbits in tokamak geometry**” details the two main descriptions of these hot plasmas, namely kinetic and fluid. Both can be simplified in the framework of the adiabatic theory, leading to the gyrokinetic theory and fluid drifts, respectively. The magneto-hydrodynamical description is introduced as a subset of the fluid one. The last section, Section “**From (gyro-)kinetic to fluid and MHD descriptions**,” addresses Coulomb binary collisions and their effects. In particular, the main results of neoclassical theory—dealing with the enhancement of collisional effects due to particle trajectory—are reviewed. Some open prospects regarding non-axisymmetrical configurations closes the section.

Characteristic length and time scales in fusion plasmas

The various processes taking place in fusion plasmas develop at different length scales which span a large number of orders of magnitudes (Fig. 1). It is illuminating to split these characteristic scales in two domains, whether they are larger or smaller than the Debye length, λ_D . λ_D is the characteristic distance at which the electric charge q_s of a given particle of species “s” can be considered screened by the charges of its neighbors. It scales like $\lambda_D = (\epsilon_0 T_s / q_s^2 n_s)^{1/2}$, with ϵ_0 the permeability of free space and T_s and n_s the temperature (in this chapter, temperature refers to thermal energy, i.e. the temperature in Kelvin times the Boltzmann constant k_B). It is expressed in Joules in SI, or in electron-volt, with the conversion $1 \text{ eV} \sim 1.602 \cdot 10^{-19} \text{ J}$ and particle density of the “s” species. The Debye length ranges from typically $9 \cdot 10^{-5} \text{ m}$ in the core of an ITER deuterium plasma ($n \sim 10^{20} \text{ m}^{-3}$, $T \sim 15 \text{ keV}$) down to about $3 \cdot 10^{-5} \text{ m}$ at its periphery, at the last closed magnetic surface ($n \sim 2 \cdot 10^{19} \text{ m}^{-3}$, $T \sim 300 \text{ eV}$).

Another important property of this scale can be estimated from the Maxwell-Gauss equation, which relates the Laplacian of the electric potential ϕ to the charge density: $(\lambda_D/L)^2 (e\phi/T_0) = (\sum_i Z_i n_i - n_e)/n_0$, with L the characteristic gradient length scale of ϕ , and n_0 , T_0 the relevant density and temperature. The subscripts “i” and “e” stand for ions and electrons, and Z the atomic number. Fusion plasmas are kinetic, in the sense that their kinetic energy much exceeds the potential one, so that $e\phi/T_0 < 1$. Then it readily appears that the left hand side vanishes at scales $L \gg \lambda_D$, so that plasmas can be considered quasi-neutral at these large scales. Notice

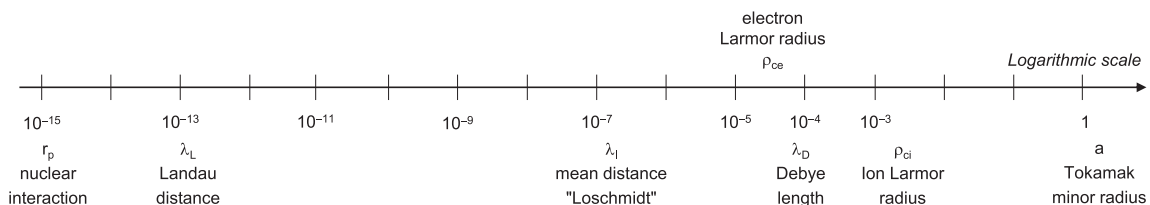


Fig. 1 Typical length scales in controlled fusion plasmas.

however that this does not preclude the development of any electric potential. It simply states that such a field exhibits large scale variations only.

At distances smaller than λ_{Ds} , particles experience mutual interactions via the Coulomb force, leading to binary collisions. Conversely, at distances larger than λ_{Ds} , particles mainly interact via the mean electromagnetic fields. Hence, λ_{Ds} marks the transition between singular and collective effects. A Debye sphere of radius λ_{Ds} contains approximately $3 \cdot 10^8$ particles in the core of ITER plasmas. Langmuir waves can develop at scales commensurable to λ_{Ds} to restore quasi-neutrality. Their frequency, set by electron inertia, is close to the electron plasma frequency: $\omega_{pe} = v_{Te}/\lambda_{De}$, with v_{Te} the electron thermal speed $v_{Te} = (T_e/m_e)^{1/2}$.

The shortest relevant distance is the one reached by two similar particles interacting frontally. This so-called Landau distance, λ_{Ls} , is reached when their entire kinetic energy is converted into potential energy, namely for $\lambda_{Ls} = q_s^2/4\pi\epsilon_0 T_s$. It is of the order of 10^{-13} m for both thermal electrons and hydrogenous ions in the core of ITER plasmas. It happens to be much larger than the acting distance of the weak nuclear force, of order of 10^{-15} m. The nuclear force ensures the nucleons—protons and neutrons—remain bounded within the nucleus. As a matter of fact, similarly to what happens in stars—although involving a different fusion reaction in this case, namely proton-proton—deuterium-tritium fusion reactions are essentially governed by quantum physics and the tunneling effect in tokamak plasmas. This shortest distance of interaction is much smaller than the average distance between two particles, sometimes called the Loschmidt distance $\lambda_{\mathcal{L}s} = n_s^{-1/3}$, of the order of 2×10^{-7} m in the ITER plasma core. The very large ratio $\lambda_{\mathcal{L}s}/\lambda_{Ls} \gg 1$ between the inter-particle distance and the minimal approach distance highlights that the vast majority of Coulomb binary collisions are grazing, i.e. leading to small angle deflections. This justifies a statistical approach of elastic collisions. The collision frequency—defined as the inverse time for a particle to have its velocity vector deflected by about 90 degrees—typically scales like $\nu_{coll,s} \sim n_s T_s^{-3/2}$. It is of the order of 10^4 s^{-1} for electron-electron and electron-ion collisions. The typical mean free path of both thermal ions and electrons in the core of ITER is of the order of 5 km, i.e. about 100 turns around the symmetry axis along the magnetic field lines—lines tangent to the magnetic field $\mathbf{B}$. As far as inelastic collisions, giving birth to fusion reactions, are concerned, these occur at a much lower frequency.

Controlled fusion plasmas are magnetized, so that each charged particle experiences a cyclotron motion at leading order (cf. next section). This is actually the primary essence of particle confinement. The cyclotron radius ρ_{cs} of their helical motion around magnetic field lines scales as the inverse of B : $\rho_{cs} = m_s v_{cs}/q_s B$, with v_{cs} the velocity transverse to $\mathbf{B}$. It is of the order of 3 mm for thermal deuterium ions and of 0.05 mm for thermal electrons in ITER core plasmas, i.e. in the ratio of the mass ratio square root. These lengths transverse to the magnetic field are much smaller than the minor radius of the configuration, which can be as large as a few meters: $\rho^* = \rho_{cs}/a \ll 1$. The same hierarchy holds for fusion-born alpha particles of 3.56 MeV, which have a gyro-radius of about 3.6 cm. The gyro-frequency scales like $\omega_{cs} = q_s B/m_s$. It is of order $40 \cdot 10^6 \text{ rad s}^{-1}$ for deuterium and $150 \cdot 10^9 \text{ rad s}^{-1}$ for electrons (i.e. in the range of ω_{pe}), so that the time scale associated with the cyclotron motion is much smaller than any other relevant characteristic time in controlled fusion plasmas, including electrostatic turbulence, which usually develops at frequencies in the range of a few to a few hundreds of kilohertz. This property is the backbone of the adiabatic theory developed in the next section.

Some other length and time scales are encountered when magnetic fluctuations enter into play. These can propagate along magnetic field lines at the Alfvén speed, v_A , which is the ion thermal velocity divided by the square root of the beta parameter, the ratio of the plasma pressure to the magnetic energy $\beta = nT/(B^2/2\mu_0)$, with μ_0 the permittivity of free space: $v_A = v_{Ti}/\beta^{1/2}$. It is of the order of the electron thermal velocity. In this case, small scale structures may develop in the plane transverse to $\mathbf{B}$, of the order of the collisionless electron skin depth, $d_e = c/\omega_{pe}$, i.e. in between the electron and the ion gyro-radii.

From this rapid overview, it appears that hot magnetized plasmas experience collective effects at spatial and time scales much smaller than those of the system, which correspond to a few meters and a few seconds—the latter referring to the energy confinement time. This hierarchy has long been advocated to address those processes on the basis of scale separation. Growing evidence shows, however, that plasma self-organization develops at intermediate—or *meso*—scales which fill the gap, hence justifying an equal footing treatment of small and large scales.

Particle orbits in tokamak geometry

The adiabatic framework of magnetized plasmas

In controlled fusion devices like tokamaks and stellarators, the cyclotron motion exhibits time and spatial scales much smaller than those characterizing the guiding magnetic field. They are typically in the range $1/(\omega_{ci}\tau_R) \sim 10^{-8} \ll 1$ and $\epsilon = \rho_{ci}/R \sim 10^{-3} \ll 1$, with τ_R the resistive diffusion time of the plasma current, as an estimate of the B-field evolution time, and R the major radius of the toroidal device. The same holds for the time evolution of the electric field, although its time evolution can be much shorter than τ_R when accounting for turbulence dynamics: $\omega_{turb}/\omega_{ci} \sim 10^{-3} \ll 1$. In this limit, plasma particles are magnetized, so that their motion can be decomposed into a fast gyromotion around the magnetic field lines, $\mathbf{v}_c = \boldsymbol{\omega}_{cs} \times \boldsymbol{\rho}_{cs}$, plus the motion of the gyro-center, $\mathbf{v}_{Gs}$.

Transverse drifts and parallel acceleration

Discriminating the secular drift of the particles across the magnetic field lines from their fast gyromotion is performed in the framework of the adiabatic theory. There, the electric and magnetic fields are decomposed into their average over the gyro-motion, $\langle \mathbf{B} \rangle$ and

$\langle \mathbf{E} \rangle$, plus perturbations, $|\delta \mathbf{B}| \ll B$ and δE . Decomposing the velocity in the same way, $\mathbf{v}_s = \langle \mathbf{v}_s \rangle + \delta \mathbf{v}_s$, Newton's law can then be split into average and fluctuating parts. At leading order in the small parameters mentioned above, the fluctuating part leads to the cyclotron motion governed by the averaged magnetic field, $\langle \mathbf{B} \rangle$. When magnetic turbulence is negligible, $\langle \mathbf{B} \rangle$ can be approximated to its value at the position of the gyro-center: $\langle \mathbf{B} \rangle \approx \mathbf{B}(\mathbf{x}_G) = \mathbf{B}_G$. The averaged part of Newton's law involves, in particular, the non-vanishing quadratic term $\langle \delta \mathbf{v}_s \times \delta \mathbf{B} \rangle$ coming from the Lorentz force. As already stated, $\delta \mathbf{v}_s = (q_s \mathbf{B}_G / m_s) \times \boldsymbol{\rho}_{cs}$, and $\delta \mathbf{B}$ is simply obtained by Taylor expanding the magnetic field at leading order from the gyro-center to the particle position: $\delta \mathbf{B} = (\boldsymbol{\rho}_{cs} \cdot \nabla) \mathbf{B}_G$. The nonlinear term can then be evaluated in terms of the averaged quantities only, by noticing that solely $\boldsymbol{\rho}_{cs}$ depends on the gyro-phase. It leads to $\langle \delta \mathbf{v}_s \times \delta \mathbf{B} \rangle \approx -(\mu_s / q_s) \nabla B_G$, where $\mu_s = m_s v_{cs}^2 / 2 B_G$ is the magnetic moment, which will be shown to be an adiabatic invariant in the following subsection. The resulting dynamical equation governing the average velocity $\langle \mathbf{v}_s \rangle$, i.e. that of the gyro-center $\langle \mathbf{v}_s \rangle = \mathbf{v}_{Gs}$, then reads as follows:

$$m_s d\mathbf{v}_{Gs}/dt = q_s (\langle \mathbf{E} \rangle + \mathbf{v}_{Gs} \times \mathbf{B}_G) - \mu_s \nabla B_G \quad (1)$$

It readily appears that the gyro-center experiences a novel force in addition to the Coulomb and Lorentz ones, due to the weak inhomogeneity of the magnetic field at the scale of the cyclotron radius. One can proceed further by decomposing the velocity in terms of parallel and transverse components: $\mathbf{v}_{Gs} = \mathbf{v}_{G//s} + \mathbf{v}_{G\perp s}$. Injecting this decomposition in Eq. (1) allows one to retrieve the expressions of the transverse drifts on the one hand, and of the parallel acceleration on the other hand.

Projecting Eq. (1) on the transverse plane, i.e. in practice subtracting its parallel component, leads to 2 terms on the left hand side: $(d\mathbf{v}_{G\perp s}/dt)_\perp$ on the one hand, and $(\mathbf{v}_{G//s} d\mathbf{b}/dt)_\perp$ on the other hand. The first term can be neglected at leading order; indeed, it is much smaller than the Lorentz term on the right hand side. The second term can be approximated by $(v_{G//s}^2 db/ds)$, with "s" the curvilinear abscissa along the magnetic field lines and $\mathbf{b} = \mathbf{B}_G/B_G$ the unit vector along the field lines. The latter term can be evaluated using the Frenet frame, which tells that $d\mathbf{b}/ds = \mathbf{N}/R$, with $\mathbf{N}$ the unit vector normal to the field lines, and R the curvature radius of these lines—which is nothing but the major radius of the torus. Then, cross-multiplying the transverse projection of Eq. (1) provides the expression of the transverse drifts at leading order (cf. Fig. 2A):

$$\mathbf{v}_{G\perp s} = (\langle \mathbf{E} \rangle \times \mathbf{B}_G) / B_G^2 + (1/q_s B_G^2) \mathbf{B}_G \times [\mu_s \nabla B_G / B_G + m_s v_{G//s}^2 \mathbf{N} / R] \quad (2)$$

The first component is the electric drift $\mathbf{v}_E$. It does weakly depend on the species, due to the fact that $\langle \mathbf{E} \rangle$ is averaged over a cyclotron radius. Therefore, it carries a limited amount of current in the transverse direction. The resulting current is mostly carried by the ions, for they have a larger gyro-radius. This drift leads to particle convection. It is also responsible for turbulent transport of particles, momentum and heat when the electric field develops a broad spectrum of turbulent fluctuations. The two last components constitute the magnetic drifts $\mathbf{v}_{ds}$: the grad-B and curvature drift, respectively. The latter results from the centrifugal force associated with the parallel motion along curved field lines. It can be recast differently by noticing that $\mathbf{N}/R = \mathbf{b} \cdot \nabla \mathbf{b} = (\nabla \times \mathbf{B}_G) \times \mathbf{b} / B_G + (\nabla_\perp B_G) / B_G$. Then the last two terms of Eq. (2) can be recast as follows: $\mathbf{v}_{ds} = (m_s v_{G//s}^2 + \mu_s B_G) \mathbf{B}_G \times \nabla B_G / (q_s B_G^3) + m_s v_{G//s}^2 (\nabla \times \mathbf{B}_G)_\perp / (q_s B_G^2)$. The last term of this expression, proportional to the transverse plasma current—hence to the pressure gradient through the magneto-hydrodynamical equilibrium—is small in low-beta plasmas. In turn, $\mathbf{v}_{ds}$ is mainly vertical in tokamak plasmas ($\mathbf{B}_G$ is mainly toroidal and ∇B_G is horizontal, pointing toward the symmetry axis). It results from the small deformation of the cyclotron radius in the presence of an inhomogeneous magnetic field, so that the cyclotron orbit is not exactly a close loop in

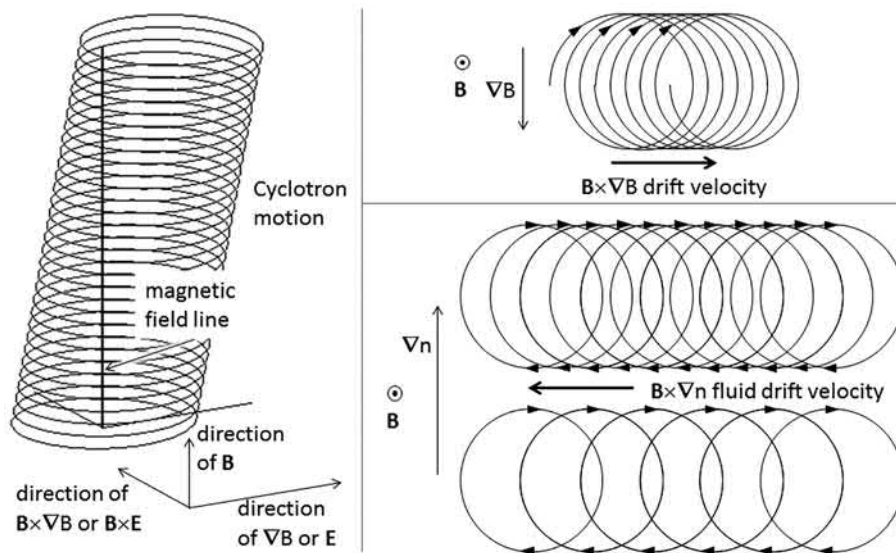


Fig. 2 (A) Charged particle motion in a strong and inhomogeneous magnetic field ($\rho_{ci}/R \sim 10^{-2}$ here). (B) The $\mathbf{B} \times \nabla B$ drift originates from the not perfectly circular cyclotron orbit. (C) Sketch illustrating the origin of the diamagnetic fluid drift velocity (case of constant temperature).

the transverse plane (cf. Fig. 2B). Ions drift in one direction, electrons in the other, leading to vertical charge separation. This current carried by the gyro-centers is only part of the total transverse plasma current. Indeed, it is to be added to the magnetization current carried by the particles themselves in their cyclotron motion. This point will be made clear when comparing kinetic and fluid approaches (cf. Section “Particle orbits in tokamak geometry”).

Getting the parallel acceleration requires in particular to derive the parallel projection of the time derivative of the transverse drifts. After some lengthy yet straightforward calculation, one finally obtains:

$$m_s dv_{G//s}/dt = \left[\mathbf{b} + \left(m_s v_{G//s} / q_s B_G \right)^2 (\nabla \times \mathbf{B}_G)_{\perp} \right] (q_s \langle \mathbf{E} \rangle - \mu_s \nabla B_G) + \left(m_s v_{G//s} / B_G^3 \right) (\mathbf{B}_G \times \nabla B_G) \langle \mathbf{E} \rangle \quad (3)$$

As discussed previously, the term proportional to the transverse current $(\nabla \times \mathbf{B}_G)_{\perp}$ is smaller than the other ones in the ratio β . All the other terms are a priori of the same order of magnitude. The parallel forces contained in the first terms on the right hand side can lead to particle trapping (cf. section “Passing and trapped particles”) due to either the inhomogeneity of the magnetic field or of the electric field along the field lines. The last term results from a synergy between the electric field and the B-field inhomogeneity.

In summary, the velocity of a particle immersed in a strong and weakly varying magnetic field—with respect to the cyclotron motion—can be decomposed in three components, made up of the cyclotron motion and the parallel and transverse components of the gyro-center motion: $\mathbf{v}_s = \mathbf{v}_{cs} + \mathbf{v}_{G//s} + \mathbf{v}_{G\perp s}$.

The three motion invariants

At equilibrium—as will be shown hereafter, equilibrium means here axisymmetry and time-independent scalar and vector electro-magnetic potentials—particle motion in tokamak plasmas is characterized by three motion invariants, i.e. scalar quantities which are conserved along individual trajectories. As a consequence, the motion is said to be integrable. Moreover, the existence of three periodic directions (the two angles of the torus plus the cyclotron phase) ensures that the motion is confined in a finite volume. This is the essence of particle confinement in tokamaks.

The energy ϵ_s . Let us perform the scalar product of Newton’s law ($m_s d\mathbf{v}/dt = q_s (\mathbf{E} + \mathbf{v} \times \mathbf{B})$) with the velocity $\mathbf{v}$. The Lorentz force, transverse to $\mathbf{v}$, disappears from the right hand side. If the scalar ϕ and vector $\mathbf{A}$ potentials do not explicitly depend on time, then the right hand side reads as follows: $\mathbf{v} \cdot \mathbf{E} = -\mathbf{v} \cdot \nabla \phi = -d\phi/dt$, with d/dt the Lagrangian derivative taken along the trajectory. Then, it readily follows that $d\epsilon_s/dt = 0$, with $\epsilon_s = m_s v^2/2 + q_s \phi$ the energy of the particle.

The magnetic momentum μ_s . It can be shown that, within the frame of the adiabatic theory, one can construct a magnetic momentum which gets conserved at all orders of the small parameter ϵ . We propose here a simplified demonstration at order one in ϵ . Let us consider the time evolution of the transverse energy $d(\mu_s B_G)/dt$. From the invariance of the energy, this term is straightforwardly related to the parallel acceleration and to the time derivative of the electric potential: $d(\mu_s B_G)/dt = -m_s v_{G//s} dv_{G//s}/dt - q_s d\phi/dt$. Using Eq. (3), one can show after some algebra that $d(\mu_s B_G)/dt = \mu_s dB_G/dt$. The conclusion is that $d\mu_s/dt = 0$, so that $\mu_s = m_s v_{cs}^2/2B_G$ is an adiabatic invariant of motion.

The toroidal kinetic momentum $P_{\phi s}$. At first approximation—especially when ignoring the finite number of toroidal field coils—and in the absence of turbulence, tokamak plasmas can be considered axisymmetric. All quantities are thus independent of the toroidal angle ϕ . The Noether theorem tells us that such a symmetry is associated to an invariant. This directly derives from the Lagrange equations, stating that $d[\partial L/\partial(q_i/dt)]/dt - \partial L/\partial q_i = 0$, with L the Lagrangian and q_i the generalized coordinates. The invariant associated to axisymmetry $\partial L/\partial \phi = 0$ is the toroidal kinetic momentum P_{ϕ} , which then reads $P_{\phi} = \partial L/\partial(d\phi/dt)$. Using the expression of the Lagrangian of a charged particle in an electromagnetic field $L_s = m_s v_s^2/2 - q_s \phi + q_s \mathbf{v} \cdot \mathbf{A}$ and noticing that $d\phi/dt = v_{\phi s}/R$, one readily finds $P_{\phi s} = m_s R v_{\phi s} + q_s R A_{\phi}$. The latter term can be related to the poloidal magnetic flux ψ , which is a label of the flux surfaces. In the end, one obtains $P_{\phi s} = m_s R v_{\phi s} + q_s \psi$. One consequence is that particle trajectories are not contained within magnetic flux surfaces—which would have been the case if ψ would have been the invariant. Yet, their distance from these surfaces remains small, scaling like the cyclotron radius, among others.

In summary, particle motion is characterized by three invariants in tokamaks at equilibrium: the energy ϵ_s provided the scalar and vector potentials do not depend explicitly on time, the magnetic momentum μ_s within the adiabatic framework, and the toroidal kinetic momentum $P_{\phi s}$ in the axisymmetric limit. Because they break the equilibrium conditions, turbulence and collisions break these invariances, hence leading to the breaking of confinement through cross-field transport.

Passing and trapped particles

The parallel acceleration Eq. (3) is primarily governed by the B-field inhomogeneity $\nabla_{//} B_G$. B_G decreasing like $1/R$, those particles with insufficient parallel energy will loop back before reaching the innermost poloidal angle, $\theta = \pi$, on the magnetic surface where the magnetic field is maximum, $B_{\max}$. These are called ‘trapped’ particles—the others being ‘passing.’ Their main characteristics can be simply derived from the motion invariants.

Let us consider a simple circular and concentric magnetic equilibrium where $B(r, \theta) = B_0 R_0/R$, with B_0 and R_0 the magnetic field and major radius on the outer mid-plane, at $\theta = 0$, and $R = R_0(1 + \epsilon \cos \theta)$. Here, $\epsilon = r/R_0$ the inverse of the aspect ratio will be considered as a small parameter, $\epsilon \ll 1$ (limit of large aspect ratio $R/a \gg 1$). Let us also neglect the potential energy, which is much smaller than the kinetic energy. The parallel velocity of the gyro-center then reads: $v_{G//s}^2 = (2/m_s)(\epsilon_s - \mu_s B)$. It appears that $v_{G//s}$ vanishes if ϵ_s is smaller than $\mu_s B_{\max}$. Performing Taylor expansion in the small parameter ϵ , this inequality can be recast as

follows, giving the trapping condition in terms of the ratio of the parallel and transverse velocities at $\theta = 0$: $(v_{G//s}/v_{cs})_{\theta=0} < (2\epsilon)^{1/2}$. In the same limit and assuming an isotropic Maxwellian distribution function, the fraction f_t of trapped particles is found to be $f_t = 2(2\epsilon)^{1/2}/\pi$.

Trapped particles move back and forth in between their two bouncing points—located at the same distance from the symmetry axis—on the same magnetic surface. Their trajectory has a non-vanishing width in the transverse plane, such that it looks like a banana. This width, δ_{bs} , can be estimated on the equatorial plane by invoking the invariance of $P_{\phi s}$. Indeed, writing that $P_{\phi s}$ has the same value on the inner and outer sides of the banana trajectory at $\theta = 0$, and performing Taylor expansion in the limit $\delta_{bs}/r \ll 1$, one gets: $\delta_{bs} \approx 2m_s R v_{\phi s} / (q_s d\psi/dr)$. The radial derivative of the poloidal magnetic flux depends on the poloidal magnetic field, so that $d\psi/dr = -RB_\theta \approx -rB_\phi/q$ with $q = rB_\phi/RB_\theta$ the safety factor in the cylindrical and large aspect ratio approximation. In the end, the so-called banana width is of the order of $\delta_{bs} \approx 2q\rho_{cs}/\epsilon^{1/2}$, where the toroidal velocity has been approximated by $v_{\phi s} \approx v_{G//s} = (2\epsilon)^{1/2}v_{cs}$, which holds for marginally trapped particles. Performing the same analysis for passing particles would show that their trajectory departs from their reference magnetic surface by the characteristic distance $q\rho_{cs}$ (cf. Fig. 3A).

Using the same relationship between parallel and transverse velocities provides a way to quantify the bouncing frequency ω_{bs} . It is given by the ratio of the parallel velocity over the parallel length in between the two turning points. For thermal particles, one then obtains: $\omega_{bs} \approx \epsilon^{1/2}v_{Ts}/2\pi qR$. Notice that ω_{bs} is much smaller than the cyclotron frequency, in the ratio $\omega_{bs}/\omega_{cs} \approx \epsilon^{1/2}\rho_{cs}/qR \ll 1$.

It turns out that trapped particles also undergo an even slower precession in the toroidal direction. This results from the fact that the magnitude of the parallel velocity is not exactly the same in the inboard and outboard side of the banana orbit due to the inhomogeneity of the magnetic field: $v_{G//s}^2 = (2/m_s)(\epsilon_s - \mu_s B)$, where we recall that ϵ_s and μ_s are motion invariants. This imbalance of the parallel velocity $\Delta v_{G//s} \approx (\mu_s \delta_{bs}/m_s v_{G//s}) dB/dr$ then results in the precession frequency $\omega_{ds} \approx \Delta v_{G//s}/R$ which then reads: $\omega_{ds}/\omega_{cs} \approx (q\rho_{cs}^2/rR) \ll 1$ (cf. Fig. 3B). One can further notice the hierarchy: $\omega_{ds} \ll \omega_{bs} \ll \omega_{cs}$.

Characterizing these various frequencies is an important issue since they can resonate with plasma waves and allow for the resonant transfer of energy between certain classes of particles and waves, possibly leading to instabilities and deleterious transport across the magnetic surfaces. Alternatively, they can be used to transfer energy from injected waves to particles, hence providing a way to heat up the plasma.

From (gyro-)kinetic to fluid and MHD descriptions

One can basically distinguish three levels of description of hot plasmas. In each case, the problem consists in calculating the self-consistent plasma response to electric and magnetic fields. These fields derive from particle charge and current densities via Maxwell's relations.

The most demanding description in terms of numerical resources (memory and computational time), yet the simplest in terms of equations, relies on Newton's law. One follows the motion of each individual particle under the action of the Coulomb and Lorentz forces. This approach is however intractable given the huge number of particles to be considered (of the order of the Avogadro number per species in ITER plasmas). The *kinetic* approach is the second one; it evolves the distribution function—quantifying the average number of particles per elementary volume in the phase space—of each species in the six-dimensional (6D) phase space (3 in position, 3 in velocity). The number of degrees of freedom reduces to six times the number of species. The third description is the *fluid* one, derived from the kinetic one by integrating over the velocity space. In this case, the number of fluid moments is

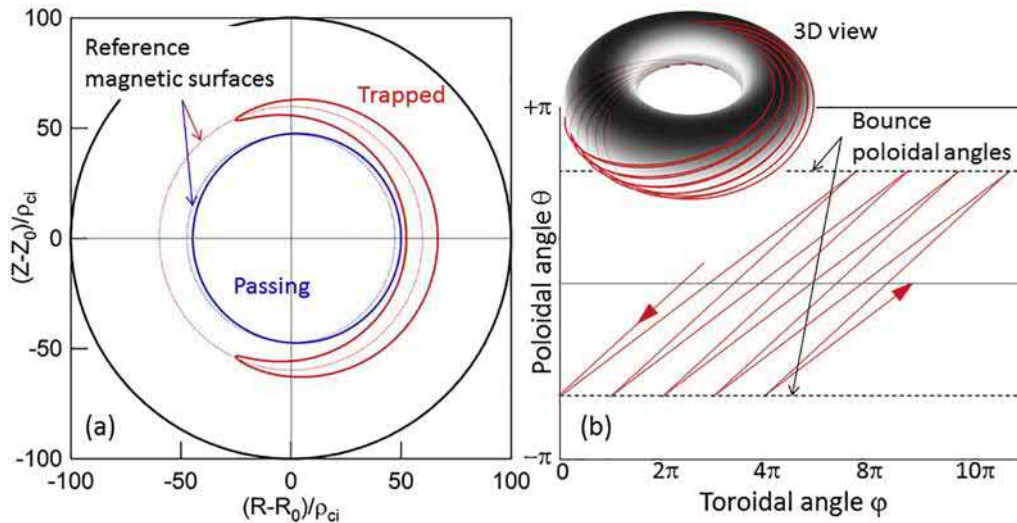


Fig. 3 (A) Projection on the poloidal cross-section of trapped and passing orbits. (B) Sketch of a trapped trajectory in the (θ, ϕ) plane illustrating the toroidal precession (inset: 3-dimensional view; courtesy D. Zarzoso /R. Varennes, GCT code).

a priori infinite, hence requiring a closure assumption. The two latter descriptions and their refinements are detailed in the following sections.

Vlasov and gyrokinetic equations

The one-particle distribution function $f_s(\mathbf{x}, \mathbf{v}, t)$ of a given species “ s ” measures the density of these particles at time t in an elementary volume of the six-dimensional phase-space centered on the $\mathbf{x}$ position and the $\mathbf{v}$ velocity.

In hot magnetized plasmas such as those encountered in the core of controlled fusion devices, binary Coulomb collisions play a minor role—although not negligible—the collision frequency being much smaller than most of other relevant frequencies (cf. Section “Introduction”). In turn, Vlasov proposed an equation in 1945 where the time evolution of f_s does not depend on binary collisions, the interaction between particles being mediated by the mean electric $\mathbf{E}$ and magnetic $\mathbf{B}$ fields. It states that f_s is constant along the trajectories, i.e. along the flow motion in the 6D phase space which obeys the following equations: $d\mathbf{x}/dt = \mathbf{v}$ and $m_s d\mathbf{v}/dt = q_s(\mathbf{E} + \mathbf{v} \times \mathbf{B})$. Its non-relativistic expression reads as follows:

$$\partial_t f_s + \mathbf{v} \cdot \nabla_{\mathbf{x}} f_s + (q_s/m_s) (\mathbf{E} + \mathbf{v} \times \mathbf{B}) \cdot \nabla_{\mathbf{v}} f_s = 0 \quad (4)$$

It can be derived by performing relevant truncations of the so-called BBGKY hierarchy of equations (after the name of its co-authors, Bogoliubov–Born–Green–Kirkwood–Yvon). There, the time evolution of the $(M-1)$ -particle distribution function can be related to that of the M -particle $f_{M,s}(\mathbf{x}_1, \dots, \mathbf{x}_M, \mathbf{v}_1, \dots, \mathbf{v}_M, t)$, with N the total number of particles and $1 \leq M \leq N$. The procedure relies on truncation of the hierarchical equations, obtained from the partial integration of Liouville’s equation, which is obeyed by the N -particle distribution function f_N .

Although weak, collisions often need to be accounted for, inter alia because they ensure the relaxation towards the Maxwellian distribution function. In this case, the right hand side of Eq. (4) is replaced by a Coulomb collision operator $\sum_{s'} C(f_s, f_{s'})$, where the sum is taken over all species (denoted by s') including the species s which is the subject of Eq. (4). It accounts for particle-particle interactions which take place at typical distances smaller than the Debye length. This intrinsically nonlinear operator can be shown to take the general form of a Fokker-Planck operator. It is often linearized, the distribution functions then being assumed to remain close to a Maxwellian. Several linearized expressions exist in the literature.

In the adiabatic limit which is relevant at least in the core of controlled fusion plasmas, the 6D phase space $(\mathbf{x}, v_{G//s}, \mu_s, \varphi_c)$ can be reduced to a 4D phase space $(\mathbf{x}_{Gs}, v_{G//s})$ plus the adiabatic invariant μ_s , basically by averaging out over the fast gyro-phase φ_c . The resulting *gyrokinetic equation* then involves the 5D distribution function f_{Gs} of the gyrocenters. Its expression looks similar to the Vlasov equation, with the particle flow replaced by that of the gyrocenters at order 1 in ϵ : $d\mathbf{x}_{Gs}/dt = v_{G//s} \mathbf{b} + \mathbf{v}_{G \perp s}$, $d\mathbf{v}_{G \perp s}/dt = 0$ and $dv_{G//s}/dt$ given by Eq. (3) in Section “Characteristic length and time scales in fusion plasmas.” Some refinements have however to be accounted for—which take the form of additional terms in the Jacobian of the transformation—to preserve the necessary Hamiltonian structure of the equation, so that it finally reads:

$$\partial_t f_{Gs} + B_{||}^{*-1} \nabla_{\mathbf{x}} \cdot [B_{||}^* (d\mathbf{x}_{Gs}/dt) f_{Gs}] + B_{||}^{*-1} \partial/\partial v_{G//s} [B_{||}^* (dv_{G//s}/dt) f_{Gs}] = 0 \quad (5)$$

with

$$B_{||}^* (d\mathbf{x}_{Gs}/dt) = v_{G//s} \mathbf{B}^* + \mathbf{b} \times [(\mu_s/q_s) \nabla \mathbf{B} + \nabla \langle \phi \rangle]; B_{||}^* (dv_{G//s}/dt) = -\mathbf{B}^* \cdot [\mu_s \nabla \mathbf{B} + q_s \nabla \langle \phi \rangle] / m_s.$$

and $\mathbf{B}^* = \mathbf{B} + (m_s/q_s) v_{G//s} \nabla \times \mathbf{b}$, $B_{||}^* = \mathbf{B}^* \cdot \mathbf{b}$. Here, the electrostatic limit has been considered, which assumes that the vector potential is constant in time. It is relevant in the plasma core where most of the turbulent transport is due to fluctuations of the electric potential. It appears that the above equations of motion derived from the phase space reduction in the Hamiltonian formalism are identical to those derived from Newton’s law (cf. Section “Characteristic length and time scales in fusion plasmas”) in the limit $(m_s v_{G//s} / q_s B^2) \mu_{0j//} \rightarrow 0$, which is equivalent to approximating $B_{||}^*$ by B . In the core of tokamak plasmas, $(m_s v_{G//s} / q_s B^2) \mu_{0j//}$ is indeed very small, of the order of $(a/qR) \rho^* \sim 10^{-3}$ for thermal ions in ITER.

Self-consistently solving the gyrokinetic Eq. (5) requires the knowledge of the electric potential ϕ . The closure is ensured by Maxwell’s equations, which however involve the *particle* distribution functions, while the gyrokinetic framework only treats the *gyro-center* distribution functions. The next section addresses this important issue.

Maxwell’s equations for fusion plasmas

The self-consistent treatment of plasma dynamics within the kinetic framework requires solving Maxwell’s equations, which link the electric, $\mathbf{E}$, and magnetic, $\mathbf{B}$, fields to the charge and current densities of particles. Maxwell-Gauss equation $\nabla^2 \phi = -\sum_s q_s n_s / \epsilon_0$ can be safely replaced by the quasi-neutrality condition $\sum_s q_s n_s = 0$ provided the electric potential, ϕ , develops on typical scales larger than the Debye length. This is actually the case for ion turbulence where the characteristic correlation length reaches a few ion gyro-radii. The approximation may however become marginal in cases where electron turbulence exhibits structures of a few electron gyro-radii. In controlled fusion devices, even when considering turbulence, it is often sufficient to replace $\mathbf{B}$ by the equilibrium guiding field, provided by the Grad-Shafranov equation (alternative formulation of the MHD—magneto-hydrodynamical—equilibrium). Conversely, MHD instabilities perturb the equilibrium magnetic field itself. In this case, Maxwell-Ampère equation, $\nabla \times \mathbf{B} = \mu_0 \sum_s \mathbf{j}_s$

(the displacement current $\partial_t E/c^2$ is negligible), provides the adequate response. Here, density and current simply relate to the particle distribution function through integrals over the velocity space $n_s = \int f_s d^3v$ and $j_s = q_s \int v f_s d^3v$.

The gyro-kinetic framework requires additional developments. Indeed, the challenge consists here in calculating n_s and j_s from the distribution function of the gyro-centers f_{Gs} , which is the only known kinetic quantity in this case. As a matter of fact, gyro-center f_{Gs} and particle f_s distribution functions are not equal a priori. Yet, the infinitesimal canonical transform from the particle phase space position $(\mathbf{x}, \mathbf{v})$ to the gyro-center one $(\mathbf{x}_G, \mathbf{v}_G)$ —permitted by the smallness of the ε parameter of the adiabatic theory—allows one to express f_{Gs} as a function of f_s . The transform makes use of the constraint, tied to the adiabatic limit, that the Hamiltonian of the gyro-centers should not depend on the fast gyro-phase φ_c . At first order in ε , the relationship reads as follows in the electrostatic case:

$$f_s(\mathbf{x}, \mathbf{v}, t) = f_{Gs}(\mathbf{x}_G, \mathbf{v}_G, t) + (q_s/B) [\phi(\mathbf{x}, t) - \langle \phi \rangle(\mathbf{x}_G, \mathbf{v}_G, t)] \partial f_{eq,s}(\mathbf{x}_G, \mathbf{v}_G, t) / \partial \mu_s \quad (6)$$

Notice that the gyro-average potential does not only depend on space but also on velocity coordinates, as expected since the gyro-radius ρ_s depends on the adiabatic invariant μ_s : $\rho_s = (2\mu_s/q_s\omega_{cs})^{1/2}$. At this point, closing the gyro-kinetic system in the long wavelength and electrostatic regime then simply requires solving the quasi-neutrality equation with n_s related to f_{Gs} in virtue of Eq. (6). In this case, further assuming $(k_\perp \rho_s)^2 \ll 1$ —usual although not always legitimate assumption—each density reads as follows:

$$n_s(\mathbf{x}, t) = \int J_s \cdot f_{Gs}(\mathbf{x}_G, \mathbf{v}_G, t) d^3v + \nabla_\perp \cdot [(m_s n_{eq,s} / q_s B^2) \nabla_\perp \phi(\mathbf{x}, t)]$$

The first term corresponds to the gyro-center density, with J_s the gyro-average operator. The second term corresponds to the polarization density, $n_{pol,s}$, which is negligible for electrons due to the proportionality to mass. A similar analysis can be performed in the electromagnetic case to compute the current density. In this case, in the same limit, the Maxwell-Ampère equation can be shown to take the following form:

$$-\nabla_\perp^2 A_{//}(\mathbf{x}, t) = \mu_0 \sum_s q_s \int [u_{G//s} J_s \cdot f_{Gs}(\mathbf{x}_G, \mathbf{v}_G, t) - (q_s/m_s) f_{eq,s}(\mathbf{x}, \mathbf{v}, t) J_s \cdot A_{//}(\mathbf{x}_G, \mathbf{v}_G, t) - v_{G//s} J_s \cdot f_{eq,s}(\mathbf{x}_G, \mathbf{v}_G, t)] d^3v$$

with $A_{//}$ the parallel component of the vector potential and $u_{G//s} = v_{G//s} + (q_s/m_s) A_{//}$ the canonical parallel velocity. In this formulation, the two first terms in the right hand side are very large as compared to the left hand side and almost cancel each other. This property is known as the “magnetic cancellation” issue. Computation strategies have been developed to deal with this difficulty.

Fluid and MHD equations

Fluid quantities are *moments* of the distribution function f_s , obtained by integrating f_s over the velocity space once weighted by k -order tensors of the velocity. The first three moments are the density $n_s(\mathbf{x}, t)$, the velocity $\mathbf{u}_s(\mathbf{x}, t)$ and the pressure tensor $P_s(\mathbf{x}, t)$. One also introduces the total heat flux $\mathbf{Q}_s(\mathbf{x}, t)$. They are defined as:

$$n_s = \int f_s d^3v; n_s \mathbf{u}_s = \int \mathbf{v} f_s d^3v; P_{s,ij} = \int m_s (v_i - u_i)(v_j - u_j) f_s d^3v; \mathbf{Q}_s = \int (m_s |\mathbf{v}|^2 / 2) \mathbf{v} f_s d^3v$$

The pressure tensor is often decomposed into a scalar pressure $p_s = \text{Tr}(P_s)/3$ (with Tr the trace of the tensor) plus a stress tensor π_s , such that $P_s = p_s I + \pi_s$ with I the unit tensor of rank 2 (other decompositions are sometimes useful, especially the one proposed by Chew-Goldberger-Low which accounts for pressure anisotropy: $P_{s,ij} = p_\perp s I + (p_{//s} - p_\perp s) b_i b_j + \Pi_{s,ij}$). Their time evolution is obtained by integrating the Vlasov Eq. (4)—including a collision operator in the right hand side if required—over the velocity space weighted by the corresponding k -order tensors. The three first equations lead to particle conservation, momentum balance and heat transport. In the case of short collisional mean free path, they are known as the Braginskii equations:

$$\partial_t n_s + \nabla \cdot (n_s \mathbf{u}_s) = 0 \quad (7)$$

$$n_s m_s (\partial_t \mathbf{u}_s + \mathbf{u}_s \cdot \nabla \mathbf{u}_s) + \nabla \cdot P_s = n_s q_s (\mathbf{E} + \mathbf{u}_s \times \mathbf{B}) + \mathbf{R}_s \quad (8)$$

$$\partial_t (3p_s/2 + m_s n_s |\mathbf{u}_s|^2 / 2) + \nabla \cdot \mathbf{Q}_s = q_s n_s \mathbf{u}_s \cdot \mathbf{E} + [\Delta \varepsilon]_{\text{coll.}}$$

where $\mathbf{R}_s$ is the drag force due to collisional interactions between species, and $[\Delta \varepsilon]_{\text{coll.}}$ Accounts for collisional energy transfer between species. At leading order, $\mathbf{R}_s$ reduces to the friction force $\mathbf{R}_s = -\sum_{s'} n_s m_s v_{ss'} (\mathbf{u}_s - \mathbf{u}_{s'})$. The conservation of the total energy (last equation), made of pressure and kinetic energy, can be rewritten by using mass and momentum conservations. Further neglecting the stress tensor $\pi_s \approx 0$ then leads to the following equation governing the dynamics of the pressure:

$$3/2 (\partial_t + \mathbf{u}_s \cdot \nabla) p_s + 5/2 p_s \nabla \cdot \mathbf{u}_s + \nabla \cdot \mathbf{q}_s = [\Delta \varepsilon]_{\text{coll.}} - \mathbf{u}_s \cdot \mathbf{R}_s \quad (9)$$

Here, $\mathbf{q}_s$ stands for the conductive heat flux: $\mathbf{q}_s = \int (m_s |\mathbf{v} - \mathbf{u}_s|^2 / 2) (\mathbf{v} - \mathbf{u}_s) f_s d^3v$.

The magneto-hydrodynamical (MHD) framework represents a peculiar sub-branch of the fluid approach, considering a one-fluid description of the plasma. It is particularly suited to study global equilibrium of magnetic configurations, and to explore the possible existence of magnetic instabilities. To this end, the mean flow, $\mathbf{U}$, is introduced: $\rho \mathbf{U} = \sum_s n_s m_s \mathbf{u}_s$, with $\rho = \sum_s n_s m_s$ the mass density (dominated by the ions). Similarly, $\mathbf{J} = \sum_s q_s n_s \mathbf{u}_s$ is the total current density. In the same spirit, the one-fluid pressure tensor $\bar{P}$ is defined as follows: $\sum_s n_s m_s \mathbf{u}_{s,i} \mathbf{u}_{s,j} = \rho \mathbf{U}_i \mathbf{U}_j + \sum_s n_s m_s (\mathbf{u}_{s,i} - \mathbf{U}_i)(\mathbf{u}_{s,j} - \mathbf{U}_j) + P_{ij}$. In the case of magnetized fusion

plasmas, $\bar{P}$ is diagonal and reduces to $P_{ij} = P\delta_{ij}$ in the isotropic case, with δ_{ij} the Kronecker symbol. In this framework, neglecting sub-dominant terms and considering the collisionless limit, continuity and momentum balance equations reduce to:

$$\partial_t \rho + \nabla \cdot (\rho \mathbf{U}) = 0 \quad (10)$$

$$\rho \, d\mathbf{U}/dt = \mathbf{J} \times \mathbf{B} - \nabla P \quad (11)$$

with $d/dt = (\partial_t + \mathbf{U} \cdot \nabla)$ the Lagrangian derivative. From Eq. (11), MHD equilibrium states that the Lorentz force balances the pressure gradient, and that the pressure is constant on magnetic surfaces. In the adiabatic limit, energy conservation takes the form of a state equation:

$$d(P/\rho^\Gamma)/dt = 0 \quad (12)$$

with $\Gamma = (N + 2)/N$ the adiabatic index (N = number of degrees of freedom), equal to 5/3 in the 3D case. The Ohm's law closes the system, complemented by Maxwell's equations. It derives from the dynamics of the current density and reads as follows in its simplest form:

$$\mathbf{E} + \mathbf{U} \times \mathbf{B} = \eta \mathbf{J} \quad (13)$$

where the Spitzer parallel resistivity η depends on the electron-ion collision frequency ν_{ei} : $\eta = m_e \nu_{ei} / (L_{11} n_e e^2)$, with L_{11} a correction coefficient of a few units accounting for electron-electron collisions ($L_{11} \sim 2$ for singly charged plasmas). Ideal MHD corresponds to the special case $\eta = 0$, while finite η values correspond to resistive MHD. The latter term introduces critical changes, especially allowing for changes in the magnetic topology leading to reconnection and magnetic islands. Several refinements with respect to these basic equations are considered in the literature, covering the broad spectrum of MHD physics, see (Lao, 2021).

Let us now address another important issue regarding the fluid approach. In the procedure leading to fluid equations, because Eq. (4) explicitly involves the velocity $\mathbf{v}$, it appears that any moment $M^{(k)}$ of order k is related to the $(k + 1)$ moment $M^{(k+1)}$, hence leading to an a priori infinite hierarchy of equations. A closure assumption is then required, consisting in replacing the dynamical equation of $M^{(k+1)}$ by some prescribed relation G between $M^{(k+1)}$ and the lower order moments: $M^{(k+1)} = G(M^{(1)}, \dots, M^{(k)})$. In regimes where collision frequencies are commensurable to, or even larger than, all other relevant frequencies, the distribution function can be assumed to remain close to a Maxwellian, which constitutes the kernel of collision operators. In this case, following the Chapman-Enskog procedure, which consists of a perturbative treatment at small Knudsen number (defined as the ratio of the collision mean free path to the macroscopic length scale of the system), Braginskii has derived in 1965 a set of closed fluid equations for collisional magnetized plasmas. Conversely, plasmas in the core of fusion devices are weakly collisional, the Knudsen number being much larger than unity. There, Braginskii equations are no longer valid. Actually, there is no universal closure in this case. In 1990, Hammett and Perkins (Hammett and Perkins, 1990) have proposed a procedure to account for kinetic effects in the closure. The idea consists in matching the kinetic and fluid linear response functions n_k/ϕ_k (with n_k and ϕ_k the Fourier components of density and electric potential fluctuations, respectively) in the kinetic limit, i.e. for phase velocities ω/k much smaller than the thermal velocity, $\omega/kv_T \ll 1$. The general result is that the Fourier component of the parallel heat flux scales like the sign of the parallel wave vector $k_{\parallel}$ times the temperature T_k . This corresponds to a non-local closure in real space, the local flux depending on distant values of the temperature. This procedure has been further extended and applied to several fluid models, exhibiting some success particularly in the astrophysical context. Yet, several limitations have to be mentioned: it remains linear, hence questioning its validity in the nonlinear regime; it is model dependent, and can lead to closures which may turn out to be complicated to implement numerically; last but not least, it does not prevent over damping of large scale flows, which efficiently contribute to turbulence saturation in tokamak plasmas.

Fluid drifts

The fluid equations derived in the previous section do not make use of the adiabatic limit which holds for hot and magnetized fusion plasmas. In this adiabatic framework, it will be shown that the components of the fluid velocity transverse to the magnetic field can be computed explicitly order by order with respect to the small parameter $\varepsilon = \omega/\omega_{cs} \approx \rho_{cs}/R \ll 1$, with ω the largest relevant frequency of the considered problem: $\mathbf{u}_{s\perp} = \mathbf{u}_{s\perp}^{(1)} + \mathbf{u}_{s\perp}^{(2)} + \dots$, with the notation $|\mathbf{u}_{s\perp}^{(j)}|/v_{Ts} \sim O(\varepsilon^j)$.

Let us consider the transverse components of the momentum conservation equation, Eq. (8). The leading order terms are the gradient of the scalar pressure and the Coulomb and Lorentz forces. Indeed, the Lagrangian derivative term is of higher order, scaling like ε times the Lorentz force, as a consequence of the adiabatic hypothesis. Also, the anisotropic and off-diagonal components of the pressure tensor contribute to a small correction of the scalar pressure, of order ε . Finally, the friction force can be neglected in these weakly collisional plasmas. Then, cross-multiplying this three-terms balance equation by $\mathbf{B}$ provides the expression of the leading order term of the transverse velocity:

$$\mathbf{u}_{s\perp}^{(1)} = (\mathbf{E} \times \mathbf{B})/B^2 + (\mathbf{B} \times \nabla p_s)/q_s n_s B^2 \quad (14)$$

The first term is the electric drift $\mathbf{u}_E$, which has also been encountered when deriving the transverse drifts of the gyro-center. Its gyro-fluid expression consists in replacing the electric field $\mathbf{E}$ by its gyro-average over a thermal gyro-radius $\langle \mathbf{E} \rangle_T$. In this case, the electric drift depends—weakly—on the species. The fluctuations of this velocity drift govern turbulent transport across the magnetic surfaces. The second term corresponds to the diamagnetic drift, often denoted $\mathbf{u}_s^*$. It is the same order of magnitude, but opposite in

direction, for electrons and ions. As such, it carries a non-vanishing transverse current, actually the one which ensures global MHD equilibrium through the Lorentz force. It results from the effective imbalance of particle gyro-motions in one direction and the other in the presence of a finite pressure gradient (cf. Fig. 2C). Interestingly, note that the divergence of the electric drift—in the electrostatic limit where $\mathbf{E} = -\nabla\phi$ —and of the diamagnetic current are vanishing in the case of an homogeneous magnetic field. Conversely, they scale like $\nabla \times (\mathbf{B}/B^2)$ in the inhomogeneous case, which is small because of the large gradient length R of the guiding magnetic field.

Deriving the second order fluid drift $\mathbf{u}_s^{(2)}$ is more complex. Indeed, the next order expression of the transverse momentum conservation involves the following terms, assuming parallel advection can be neglected: $[\partial_t + \mathbf{u}_s^{(1)} \cdot \nabla] \mathbf{u}_s^{(1)} + \nabla \cdot \pi_{s\perp} = q_s n_s \mathbf{u}_s^{(2)} \times \mathbf{B}$. In particular, it requires the evaluation of the divergence of the anisotropic and off-diagonal terms of the pressure tensor—the stress tensor π_s —which derives from kinetic effects. It can be shown, at leading order, that the Lagrangian derivative of the diamagnetic flow $[\partial_t + \mathbf{u}_s^{(1)} \cdot \nabla] \mathbf{u}_s^*$ balances the divergence of the stress tensor $\nabla \cdot \pi_{s\perp}$ up to a gradient term $\nabla_\perp \chi_s$. Here, the scalar χ_s turns out to be smaller than the pressure in the ratio $\chi_s/p_s \sim \omega/\omega_{cs} \ll 1$, so that it is often ignored. Then again, cross-multiplying by $\mathbf{B}$ yields the expression of $\mathbf{u}_s^{(2)}$, called the extended polarization drift $\mathbf{u}_{\text{pol},s}$. In the relevant electrostatic limit, it reads as follows

$$\mathbf{u}_{\text{pol},s} = - (m_s/q_s B^2) [\partial_t + (\mathbf{u}_E + \mathbf{u}_s^*) \cdot \nabla] \nabla_\perp \phi \quad (15)$$

It can be verified that it is $\epsilon \sim \omega/\omega_{cs}$ smaller than the electric drift, as expected. Because it is proportional to mass, it is negligible for electrons. The divergence of the polarization current is often commensurable to that of the diamagnetic current, so that both have to be accounted for in the charge conservation equation.

Let us now address the important issue of the consistency between kinetic and fluid descriptions within the adiabatic framework. We shall here restrict ourselves to the first order of the development in ϵ . At this order, particle and fluid drifts exhibit the same electric drift, but depart from each other regarding the other component. The issue then reduces to comparing the transverse currents originating from the particle motion on the one hand and from the fluid description on the other hand. The fluid one simply derives from the diamagnetic drift: $\mathbf{j}_s^{(\text{fluid})} = (\mathbf{B} \times \nabla p_s)/B^2$. Regarding the kinetic current, it is made of two contributions: the one arising from the transverse drift of the gyro-centers $\mathbf{j}_{ds} = \int q_s \mathbf{v}_{ds} f_{Ms} d^3v$, and the magnetization current due to the cyclotron motion $\mathbf{j}_{s,\text{magn}} = \nabla \times \mathbf{M}_s$, with $\mathbf{M}_s = -(\int \mu_s f_{Ms} d^3v) \mathbf{b}$ the magnetization. The integrals are performed over the entire velocity space, the volume element being $d^3v = (2\pi B/m_s) dv_{G//s} d\mu_s$. Consistently with the fluid analysis which we have considered here, we take an isotropic and centered Maxwellian distribution function for the particles (the calculation can be easily extended to the anisotropic case characterized by different parallel and perpendicular temperatures), $f_{Ms} = n_s/(2\pi v_{Ts}/m_s)^{3/2} \exp[-(m_s v_{G//s}^2/2 + \mu_s B)/2T_s]$. This yields: $\mathbf{j}_{ds} = 2p_s \mathbf{B} \times \nabla B/B^3 + p_s (\nabla \times \mathbf{B})_\perp/B^2$ and $\mathbf{j}_{s,\text{magn}} = (\mathbf{B} \times \nabla p_s)/B^2 - 2p_s \mathbf{B} \times \nabla B/B^3 - p_s (\nabla \times \mathbf{B})_\perp/B^2$, so that one effectively recovers $\mathbf{j}_s^{(\text{fluid})} = \mathbf{j}_s^{(\text{kinetic})} = \mathbf{j}_{ds} + \mathbf{j}_{s,\text{magn}}$, as one should. Also, consistently with the physical picture of the diamagnetic current, one finds that it is entirely contained within the cyclotron motion. Finally, notice that the magnetization current is divergenceless, so that it does not contribute to the charge conservation equation.

Collisions and neoclassical theory

Coulomb collisions are rare in the core of fusion plasmas, the collisional mean free path reaching typically one hundred toroidal turns along the particle trajectory. The resulting cross-field transport is therefore weak, several orders of magnitude less than turbulent transport in most confinement regimes in tokamaks. Yet, collision physics cannot be ignored. On the most fundamental ground, collisions enforce the Boltzmann H-theorem and relaxation of distribution functions towards Maxwellians. In addition, in fusion devices, collisional transport can become commensurable to turbulent transport in improved confinement regimes or in non-axisymmetric configurations such as stellarators. It may even become dominant for the transport of heavy impurities from the edge—where they are emitted due to plasma-wall interactions—to the core—where they degrade the performance through radiation and fuel dilution. Also, they often govern the level of large scale plasma flows, which play a critical role in turbulence saturation. Finally, they are responsible for the inward particle pinch and toroidal plasma current, both essential to fusion performance.

Expression and order of magnitude of collision frequencies

Coulomb interaction is long distance, so that particles interact with each other continuously. At distances larger than the Debye length, these interactions are mediated by the mean electromagnetic field, governed by Maxwell's equations. At smaller distances, they are treated as binary collision processes. The mean distance between particles, λ_D , being several orders of magnitude larger than the minimal distance of interaction, λ_L , most of the collisions are grazing. Their large number within a Debye sphere—the number of particles being of the order of one billion—then allows for a statistical treatment. The Coulomb collision frequency, $\nu_{ss'}$ of particle species s colliding on particle species s' is defined as the inverse of the time needed for the velocity vector of particles s to be deflected by about 90 degrees. It results from a Markovian process involving a large number of interactions. We need first to characterize one single interaction so as to derive the expression of $\nu_{ss'}$.

Let s and s' be two interacting charged particles, with s' assumed at rest. Let b be impact parameter, the distance between s' and the direction of the trajectory of particle s with velocity $\mathbf{v}_s$ at $t \rightarrow -\infty$. As derived by Rutherford, at $t \rightarrow +\infty$, the trajectory of s is

deflected by the angle θ such that $\tan(\theta/2) = \lambda_L/2b \ll 1$, where we recall that $\lambda_L = q_s q_{s'}/2\pi\epsilon_0 m_s v_s^2$. θ also measures the relative variation of the velocity projected on the trajectory at $t \rightarrow -\infty$: $\theta \approx \delta v_s/v_s$.

In the case where s interacts with a lot of particles s' , the average deflection $\langle \delta v_s^2 \rangle$ per time τ then reads as follows: $\langle \delta v_s^2 \rangle / \tau v_s^2 = \int \theta^2 n_{s'} dV / \tau$, with $\int n_{s'} dV$ the total number of interacting s' particles during the time interval τ . The volume element is then $dV = 2\pi b db v_s \tau$. The integral diverges in both limits, $b \rightarrow 0$ and $b \rightarrow +\infty$. It is typically to be performed between $b_{\min} = \lambda_L$ —limit of grazing collisions—and $b_{\max} = \lambda_D$ —limit of charge screening and interaction through the mean fields. Further imposing $\langle \delta v_s^2 \rangle / v_s^2 = 1$ and accounting for the fact that s' particles are not at rest a priori provides the expression of $v_{ss'}$:

$$v_{ss'} = \text{Ln}(\Lambda) (\sqrt{(2\pi)/3}) q_s^2 q_{s'}^2 n_{s'} (1/m_s + 1/m_{s'}) / 4(\pi\epsilon_0)^2 m_s (v_{Ts}^2 + v_{Ts'}^2)^{3/2} \quad (16)$$

With $\text{Ln}(\Lambda) = \text{Ln}(\lambda_D/\lambda_L) \approx 20$ defined as the Coulomb logarithm. Notice that $v_{ss'}$ scales typically like $n/T^{3/2}$. Also, it is all the larger since mass is small. Denoting $K = \text{Ln}(\Lambda) (\sqrt{(2\pi)/3}) e^4 n_e / 4(\pi\epsilon_0)^2$, the expressions of typical collision frequencies between electrons and ions of charge Z read as follows:

$$v_{ee} = K/(2m_e)^{1/2} T_e^{3/2}; \quad v_{ei} = Z\sqrt{2} v_{ee}; \quad v_{ii} = Z^3 (m_e/m_i)^{1/2} v_{ee}; \quad v_{ie} = Z\sqrt{2} (m_e/m_i) v_{ee}$$

Here, for the sake of simplicity, quasi-neutrality $Zn_i = n_e$ has been used and equal ion and electron temperatures have been assumed. In the core of ITER plasmas, v_{ei} is expected to be of order 10^4 s^{-1} . For deuterium ions, $(m_e/m_i)^{1/2} \approx 1/60$, so that v_{ie} is negligible by comparison. Finally notice that $m_e v_{ei} = m_i v_{ie}$ so that the friction force exerted by ions on electrons is exactly opposed to that exerted by electrons on ions, consistently with the action-reaction principle.

Basics of neoclassical transport

Collisions generate heat transport across magnetic surfaces in controlled fusion devices. Consistently with the random walk estimate of Brownian motion, they lead to diffusive transport with coefficient $D_{\text{neo}} = \langle \Delta r^2 \rangle / \tau$, where Δr is the random displacement and τ the characteristic time between collisions. Neoclassical theory refers to the theory of Coulomb binary collisions in the toroidal magnetic geometry of controlled fusion devices, where their effects are amplified by particle trajectories which explore the whole configuration. The precise derivation of neoclassical transport coefficients in the various collisional regimes is cumbersome. Yet, the critical dependencies can be captured through rather simple arguments. The key parameter is the collisionality ν^* , ratio of the effective detrapping frequency, ν_{eff} , to the bounce frequency, ω_b , for thermal particles: $\nu^* = \nu_{\text{eff}}/\omega_b = e^{-3/2} v_c qR/v_T$, with v_c the collision frequency. Here, the expression of ν_{eff} derives from the following analysis. Consistently with their Markovian character, elastic collisions lead to diffusion in velocity space with coefficient $D_v = \langle \Delta v^2 \rangle \nu_c \approx v_c v_T^2$. According to the section “**Passing and trapped particles**”, trapped particles become passing when the ratio $v_{\parallel}/v_{\perp}$ becomes larger than $(2\epsilon)^{1/2}$. The evolution of their parallel velocity is governed by $\langle \Delta v_{\parallel}^2 \rangle = D_v t$. Hence, $1/\nu_{\text{eff}}$ corresponds to the time required for $\langle \Delta v_{\parallel}^2 \rangle$ to be of the order of $2\epsilon v_{\perp}^2$. Considering thermal particles $v_{\perp} = v_T$, this condition then yields $\nu_{\text{eff}} \approx v_c/2\epsilon$.

In the low collisionality (banana) regime $\nu^* \ll 1$, trapped particles bounce back and forth several times before being detrapped by collisions. They bring the main contribution to $D_{\text{neo}} = D_b$. Their characteristic random displacement is $\Delta r = \delta_b$, and the characteristic banana lifetime is $\tau = 1/\nu_{\text{eff}}$. Then, accounting for the trapped particle fraction, f_t , one finally obtains: $D_b = f_t \delta_b^2 \nu_{\text{eff}} = q^2 \rho_c^2 v_c / \epsilon^{3/2}$, which is $q^2/\epsilon^{3/2}$ times larger than the classical diffusion coefficient in straight field line configurations.

In the intermediate (plateau) regime $\nu^* \approx 1$, most of thermal particles are passing—trapped particles being detrapped before exploring their full orbit—and collisionless. The characteristic displacement is then $\Delta r = q\rho_c$ and the characteristic time $\tau = qR/v_T$, so that $D_p = q\rho_c^2 v_T/R$. This does not depend on the collision frequency, so that D_{neo} exhibits a plateau when plotted with respect to ν^* . Also notice that it is equal to D_b at the banana-plateau boundary, $\nu^* = 1$.

The high collisionality regime is named after its two main discoverers, Pfirsch and Schlüter. There, for $\nu^* \gg 1/\epsilon^{3/2}$, particles experience collisions before completing the full connection length path along the magnetic field. Their parallel motion is diffusive with the characteristic diffusion coefficient $D_{\parallel} = v_T^2/\nu_c$. This determines the characteristic time scale over the connection length: $\tau_{\parallel} = q^2 R^2/D_{\parallel}$. The characteristic displacement transverse to the magnetic surfaces is still governed by the magnetic drifts—curvature and $\text{grad}(B)$ —which scale like $v_d \approx v_T \rho_c/R$, so that $\Delta r = v_d \tau_{\parallel}$. In the end, the Pfirsch-Schlüter diffusion coefficient reads as follows: $D_{\text{PS}} = q^2 \rho_c^2 \nu_c$. Again, both diffusion coefficients D_p and D_{PS} smoothly match at the plateau-Pfirsch-Schlüter boundary at $\nu^* = 1/\epsilon^{3/2}$.

Given the expected profiles of density, temperature and safety factor in ITER plasmas, electrons and main ion species are expected to be in the banana regime ($\nu^* \ll 1$) in most of the confined region—except in the very core where the trapped fraction drops to zero. Conversely, many impurities will be in the plateau or even the Pfirsch-Schlüter regime, the latter being expected in the edge region for heavy impurities.

Neoclassical theory within the fluid framework

The theory of neoclassical transport requires a kinetic treatment which is beyond the scope of the present introduction. Yet, the main results can be recovered within the fluid framework provided neoclassical effects are implemented via a parallel force which will be detailed below (Garbet, 2020). It is stressed that this calculation provides an illustration of the physical processes at work, but should not be used for quantitative estimates. All calculations in this section are performed in the limit of large aspect ratio

axisymmetric tokamaks, $\epsilon \ll 1$, and the cylindrical approximation, $q = \epsilon B_\phi / B_\theta \approx \epsilon B / B_\theta$. Also, all quantities are assumed to be independent of the toroidal angle. The non-axisymmetric case will be addressed in the next section.

Let us consider the momentum balance equation at equilibrium $d/dt = 0$ (Eq. (8), Section “The three motion invariants”): $\nabla p_s + \nabla : \bar{\pi}_s = n_s q_s (\mathbf{E} + \mathbf{u}_s \times \mathbf{B}) + \mathbf{R}_s$, with $\bar{\pi}_s$ the stress tensor and $\mathbf{R}_s$ the friction force. Neoclassical kinetic results provide the explicit expression of the parallel projection of $\nabla : \bar{\pi}_s$, which defines the opposite of the neoclassical parallel force $F_{//s}$:

$$F_{//s} = -\mathbf{b} \cdot (\nabla : \bar{\pi}_s) = -n_s m_s \mu_s (u_{\theta s} - k_s \nabla_r T_s / q_s B) \quad (17)$$

Where the coefficients μ_s and k_s , calculated within the kinetic framework, depend on the collisionality regime. Conversely, axisymmetry imposes $\nabla \phi \cdot (\nabla : \bar{\pi}_s) = 0$. Then, the radial, parallel and (parallel minus toroidal) projections of the equilibrium momentum balance yield the following set of equations, assuming the pressure is constant on magnetic surfaces (as imposed by the MHD equilibrium Eq. (11)):

$$\nabla_r p_s / n_s q_s = E_r + u_{\theta s} B - u_{//s} B_\theta \quad (18)$$

$$F_{//s} + n_s q_s E_{\text{ind}} - \sum_{s'} n_s m_s v_{ss'} (u_{//s} - u_{//s'}) = 0 \quad (19)$$

$$F_{//s} - q_s \Gamma_{rs} B_\theta = 0 \quad (20)$$

Here, $\Gamma_{rs} = n_s u_{rs}$ is the radial particle flux and E_{ind} the inductive field used to generate the poloidal magnetic field B_θ . Consistently with the large aspect ratio limit, the toroidal and parallel velocities have been taken equal.

The friction term in Eq. (19) is usually dominant for ion species, so that their parallel/toroidal velocities are approximately the same. We will assume it is the case in the following. Notice however that this approximation may break down when density and temperature gradients are large.

Summing Eq. (19) over all species shows that $\sum_s F_{//s} = 0$ (as a result of quasi-neutrality $\sum_s n_s q_s = 0$ and the action-reaction principle $\sum_{ss'} n_s m_s v_{ss'} (u_{//s} - u_{//s'}) = 0$: collisions do not inject net momentum), so that, summing now Eq. (20), neoclassical transport appears to be automatically ambipolar: $\sum_s q_s \Gamma_{rs} = 0$. This means that the radial electric field cannot be calculated from these equations. Breaking the axisymmetry hypothesis would alleviate the degeneracy.

Eq. (20) shows that poloidal velocities need to be calculated to get an expression of particle fluxes. The key step is provided by the relation $\sum_s F_{//s} = 0$. To further carry on the calculations, we will consider the case where several species are present, namely electrons, deuterium ions and an impurity of charge Z . In the banana regime, $\mu_e = 1.46 v_{eD} (0.53 + Z_{\text{eff}}) q / \epsilon^{1/2}$ and $k_e = (0.62 + 1.5 Z_{\text{eff}}) / (0.53 + Z_{\text{eff}})$, with $Z_{\text{eff}} = \sum_i n_i Z_i^2 / n_e$ the effective charge number. The coefficients for deuterium are deduced from those for electrons by replacing Z_{eff} by the dilution parameter $\alpha = n_Z Z^2 / n_e$ and v_{eD} by $v_{DD} \sqrt{2}$. In the case of a dilute impurity $\alpha \ll 1$, one finds $\mu_D = 1.10 v_{DD} q / \epsilon^{1/2}$ and $k_D \approx 1.17$. In the plateau regime, the coefficients are $\mu_s = 1.25 \epsilon v_{Ts} / R_0$ and $k_s = -0.5$. It appears that $F_{//e}$ is negligible as compared to $F_{//D}$, in the typical ratio $(m_e / m_D)^{1/2}$. Also, $F_{//Z}$ can be neglected if the impurity concentration is low ($\alpha \ll 1$) or if it is in the Pfirsch-Schlüter regime where μ_Z is very low. In these most relevant cases, the sum reduces to $F_{//D} = 0$, i.e.:

$$u_{\theta D} = k_D \nabla_r T_D / e B \quad (21)$$

The ion poloidal rotation is low due to friction on trapped particles which do not rotate poloidally by definition. In the frame of these approximations, Eq. (20) cannot be used to calculate Γ_{rD} . Instead, the ambipolarity constraint is used once Γ_{re} and Γ_{rZ} have been calculated.

Subtracting Eq. (18) for deuterium and impurity eliminates E_r and yields $u_{\theta Z}$:

$$u_{\theta Z} = (T_D / e B) [\nabla_r n_Z / Z n_Z - \nabla_r n_D / n_D + (k_D - 1 + 1/Z) \nabla_r T_D / T_D]$$

Here, equal temperatures and parallel velocities have been assumed. Note that, for large charge number, the impurity poloidal velocity can have the opposite sign to the deuterium poloidal velocity. The impurity flux then simply results from Eq. (20) and the above expression for $u_{\theta Z}$, valid in banana and plateau regimes:

$$\Gamma_{rZ} = -n_Z D_Z [\nabla_r n_Z / n_Z - Z \nabla_r n_D / n_D - H_Z \nabla_r T_D / T_D]$$

With $D_Z = (q / \epsilon) \mu_Z m_Z T_D / (Z e B)^2$ the impurity diffusion coefficient and $H_Z = [1 - k_D + (k_Z - 1) / Z]$ the screening factor. The first term in the parenthesis accounts for diffusion, the second one leads to impurity peaking and the last is thermo-diffusion. It can lead to thermal screening—i.e. the expulsion of impurities from the core—if H_Z is negative ($\nabla_r T_D < 0$ due to central heating). In the limit of large Z , $H_Z = 3/2$ if deuterium is in the plateau regime while H_Z is marginally negative (close to zero when accounting for finite ϵ corrections) if it is in the banana regime. If deuterium and impurity ions are in the same collisionality regime, then their diffusion coefficients are the same: $D_Z / D_D \sim (m_Z / Z^2 m_D) v_{ZD} / v_{DD} \sim 1$ since $v_{ZD} / v_{DD} \sim Z^2 m_D / m_Z$ (v_{ZD} dominates over v_{ZZ} at low impurity concentration). However, since $v_Z^* / v_D^* \sim Z^2 (m_D / m_Z)^{1/2}$, high Z impurities are often in the plateau or Pfirsch-Schlüter regime even if deuterium is in the banana regime. If the impurity is in the plateau regime, one then finds $D_Z / D_D \sim 1 / v_Z^* \ll 1$. This is why turbulence is often the dominant transport channel for impurity diffusion. For impurities in the Pfirsch-Schlüter regime, another expression should be used for their radial flux:

$$\Gamma_{iZ}^{(PS)} = -n_Z D_Z^{(PS)} \left[\nabla_r n_Z / n_Z - Z \nabla_r n_D / n_D - \left(Z H_Z^{(PS)} / K_Z \right) \nabla_r T_D / T_D \right]$$

With the new expressions: $D_Z^{(PS)} = v_{ZD} m_Z T_D (q / ZeB)^2$, $H_Z^{(PS)} = -0.5 + (0.29 + 0.68\alpha) / (0.59 + \alpha + 1.34/g^2)$, $K_Z = 1 - 0.52\alpha / (0.59 + \alpha + 1.34/g^2)$ and $g = v_D * \epsilon^{3/2} / \sqrt{2}$. If the main ion is in the banana or plateau regime, then $g \ll 1$ and $K_Z \approx 1$ and $H_Z^{(PS)} \approx -1/2$ so that impurities are screened in the Pfirsch-Schlüter regime.

Subtracting Eq. (18) for deuterium and electrons relates $u_{\theta e}$ to $u_{\theta D}$ and the parallel current $j_{||} = -ne(u_{||e} - u_{||D})$. The latter can be eliminated by using the Ohm's law Eq. (19) for electrons, hence leading to the following approximate expression (with the assumption $(\epsilon/q)(\mu_e/v_{eD}) \ll 1$ valid for $\epsilon \ll 1$):

$$u_{\theta e} = -(1/eB) [(T_D + T_e) \nabla_r n_e / n_e + \nabla_r T_e / T_e + (1 - k_D) \nabla_r T_D / T_D] - (\epsilon/q) e E_{ind} / (m_e v_{eD})$$

The electron radial flux and parallel current then read as follows in the banana regime:

$$\Gamma_{re} = -n_e D_e [(1 + T_D / T_e) \nabla_r n_e / n_e + (1 - k_e) \nabla_r T_e / T_e + (1 - k_D) \nabla_r T_D / T_e] - (C_e \epsilon^{1/2} / B_0) n_e E_{ind}$$

$$j_{||} = - (C_e \epsilon^{1/2} / B_0) [(T_D + T_e) \nabla_r n_e + (1 - k_e) n_e \nabla_r T_e + (1 - k_D) n_e \nabla_r T_D] + \sigma_{neo} E_{ind}$$

With $D_e = C_e v_{eD} m_e T_e (q/eB)^2 / \epsilon^{3/2}$, $C_e = 1.46(0.53 + Z_{eff})$ and $\sigma_{neo} = n_e e^2 / m_e v_{eD}$ the neoclassical conductivity. The term proportional to the inductive field in Γ_{re} is known as the Ware pinch.

Bootstrap current and neoclassical resistivity

The so-called 'bootstrap current' is the term proportional to density and temperature gradients (to the pressure gradient only in the limit of unit aspect ratio $\epsilon = 1$) in the expression for $j_{||}$. It is the Onsager symmetric of the Ware pinch. It arises from the fact that, when electrons get detrapped, they carry a finite amount of parallel momentum due to the difference of density and temperature on the inner and outer sides of their banana orbit. The bootstrap current plays a critical role in tokamaks, in particular in the perspective of a steady-state reactor. Indeed, in regions with non-vanishing density and temperature gradients, it can substantially contribute to the total current, hence decisively contributing to the magnetic equilibrium of the configuration. As such, it constitutes one of the most important outputs of neoclassical physics.

The bootstrap current adds to the inductive current, the last term in the expression of $j_{||}$. It is to be noted that neoclassical effects lead to an increase of the resistivity, η_{neo} , as compared to the Spitzer formula (cf. Eq. 13), which is valid for a straight magnetic field. Again, it results from the fact that trapped electrons do not carry any current. A more careful derivation of the neoclassical conductivity $\sigma_{neo} = 1/\eta_{neo}$ then yields $\sigma_{neo} \sim \sigma(1 - \epsilon^{1/2})^2$ with $\sigma = 1/\eta$ the Spitzer conductivity. Due to the increase of collision frequency towards the plasma edge, which gets lighter and cooler, resistivity also increases at the edge. This effect, combined with the large pressure gradient at the edge, leads to a bootstrap current profile which peaks away from the magnetic axis, as evident in Fig. 4.

Some considerations on neoclassical transport in non-axisymmetric systems

In tokamak magnetic configurations, axisymmetry (toroidal invariance) provides the third invariant—the toroidal canonical momentum P_ϕ —which ensures particle confinement. Helical configurations, such as stellarators, lack such symmetry and hence any third invariant—the same holds in tokamaks if the ripple due to the discreteness of the toroidal field coils is large. De facto, particle trajectories may then explore a large volume, and even hit the wall, unless the magnetic field is optimized to prevent this. Neoclassical transport in stellarators turns out to be much larger than that in tokamaks in the low collisionality, large mean free path case which prevails in the core of hot fusion plasmas (cf. for instance the review articles (Helander, 2014; Boozer, 2015) on non-axisymmetric magnetic confinement). The present short section cannot do justice to the vast and rich literature on this important and difficult topic. It mainly aims at highlighting major results and routes towards configuration optimization.

In helical configurations, the magnetic field strength may typically exhibit the following structure:

$$B \approx B_0 [1 - \epsilon_1 \cos\theta - \epsilon_2 \cos(m\theta - n\phi)]$$

Where the ϵ_1 term is due to toroidicity while the ϵ_2 term accounts for the dominant helical structure, with (m, n) integer constants. In the case of tokamak ripple, $m = 0$ and n is the number of toroidal coils. Along field lines, B then exhibits the usual sinusoidal shape plus corrugations which, if large enough—and this is usually the case in stellarators—can lead to local trapping of the particles and superbanana orbits. If superbanana orbits remain collisionless, then the transport coefficient can be estimated the same way as for the banana regime in tokamaks, with the banana width replaced by the typical radial width δ_{sb} of the superbanana orbits and ν_{eff} being the effective scattering frequency of particle detrapping in these local wells: $D = \delta_{sb}^2 \nu_{eff}$. In this situation, neoclassical transport can become very large, much larger than banana transport in tokamaks. The stringent assumption of collisionless superbanana orbits is however hardly fulfilled. Conversely, if the orbits are interrupted by collisions, δ_{sb} should be replaced by the radial excursion of the order (v_{ds}/v_{eff}) , with $v_{ds} \sim T_s/(q_s B R)$ the essentially vertical magnetic drift. It readily appears that, in this case, $D \sim (v_{ds}/v_{eff})^2 \nu_{eff} = v_{ds}^2 / \nu_{eff}$ is inversely proportional to collisionality. This so-called $1/\nu$ regime, which does not exist in tokamaks, is orders of magnitudes larger than the banana regime. Also, the corresponding

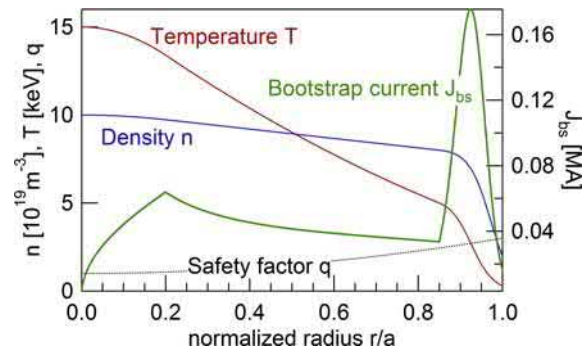


Fig. 4 Typical expected profiles of density, temperature, safety factor and bootstrap current in ITER plasmas.

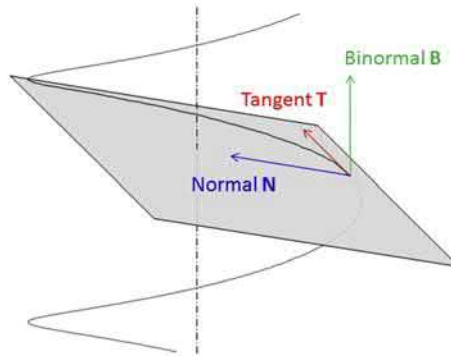


Fig. 5 Frenet frame attached to a 3-dimensional curve.

heat flux exhibits a very unfavorable scaling with temperature, like $T^{9/2}$, as compared to $T^{1/2}$ in the low collisionality regime of tokamak plasmas.

Considerable effort has been invested to develop clever optimized configurations aiming at reduced neoclassical transport in non-axisymmetric devices. The interested reader can refer to the pioneering successful efforts in this direction (Nührenberg and Zille, 1988; Grieger et al., 1992) and to more recent developments (Mynick, 2006; Gates et al., 2017). Quasi-helical or quasi-symmetric configurations target magnetic field strength exhibiting a single helicity state in appropriate coordinates. It can be shown to result in a new invariant which replaces P_ϕ . Such a state cannot be perfectly achieved in toroidal configurations, hence *quasi*. An alternative consists in making the magnetic field quasi-omnigenous, i.e. such that the net radial excursion due to the magnetic drift $\mathbf{v}_{ds}$ almost vanishes when bounce averaged over particle trajectories. Numerical simulations have already shown that orders of magnitudes could be gained regarding neoclassical transport in optimized configurations. Experiments are ongoing, most notably in the Wendelstein7-X stellarator (Yamada, 2021).

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Magnetic Confinement Fusion—Plasma Theory: Transport and Confinement

Frank Jenko, Max Planck Institute for Plasma Physics, Garching, Germany

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Introduction

Obviously, the goal of magnetic confinement is to devise a magnetic field structure that is able to retain a hot fusion plasma as well as possible. Most notably, the product of the central plasma pressure and the energy confinement time has to exceed the limit provided by the Lawson criterion to achieve a burning (i.e., self-heated) plasma, as explained elsewhere in this Encyclopedia.

In the early days of fusion research, many different concepts have been tested, but two emerged as the most promising candidates, the tokamak and the stellarator. Both of these types of devices create magnetic field lines which form nested toroidal surfaces, and one might expect that such a setup is perfect for confinement, given that the charged plasma particles follow those field lines and should thus remain on those surfaces.

Upon closer inspection, one notices that this consideration is not quite correct. Due to the fact that magnetic field lines in such toroidal devices are strongly inhomogeneous, varying spatially in both strength and direction, the plasma particles temporarily drift away from their flux surfaces, and collisions then lead to radial displacements of their drift centers, inducing a loss of heat and particles. Such transport processes are called “neoclassical,” and the corresponding physics basis has been described elsewhere in this Encyclopedia.

While quantitative estimates suggest that neoclassical losses are tolerable in practice, there exists yet another transport process which is much more restrictive. As it turns out, heat and particle transport across magnetic fields (at least in tokamak plasmas) is usually dominated by turbulence-driven processes, typically exceeding the predictions of neoclassical transport by about one order of magnitude or more. This is not surprising, given that turbulence is known to be an excellent agent of spatial transport in a wide range of physical systems. Consequently, it has become one of the central goals of fusion theory to help understand, predict, and control turbulent transport in fusion devices.

In the following, we will first describe the character, origin, and consequences of turbulent fluctuations which are observed in fusion experiments. Then we will discuss the fundamental nature of plasma microturbulence, including its relation to turbulence in fluids and astrophysical systems. Subsequently, we will provide a brief introduction to gyrokinetics, which has become the default physical model for theoretical and computational studies of turbulence in fusion plasmas. Finally, we will give two prime examples of combined experimental and theoretical efforts to control turbulent transport. We will close with some conclusions and an outlook.

Small-scale fluctuations in fusion plasmas

Both tokamaks and stellarators usually exhibit nested toroidal flux surfaces, spanned by magnetic field lines, in their core region (the situation is more complex in the edge region and will not be discussed here much further). Due to the fact that particle and heat

transport processes parallel to the field lines are very fast compared to their perpendicular counterparts in this region of the plasma, the average densities and temperatures of each particle species tends to be constant on each such surface. However, one observes small fluctuations, of the order of a percent or less. At the same time, there are also exist small fluctuations of other plasma quantities like the average parallel velocity or the parallel heat flux of each plasma species as well as of the self-generated electric and magnetic fields. These fluctuations are a result of plasma turbulence caused by the enormous spatial variations in the plasma density and temperature across flux surfaces, and the induced turbulent transport in the radial direction turns out to determine the energy confinement time—one of the key figures of merit of any magnetic confinement fusion device—as well as the confinement of plasma particles and impurities.

Characterization of the observed fluctuations

Via careful experimental studies, it became possible to characterize the observed fluctuations. Perpendicular to the background magnetic field, they tend to have a typical extent of several millimeters to a few centimeters, while in the parallel direction, they can be stretched out over tens of meters. This very anisotropic structure is due to the fact that plasma particles can move more or less freely along magnetic field lines, but drift only very slowly across the field lines.

The parallel extent is related to the so-called connection length, $L_c = 2\pi qR$, which is the length of a given magnetic field line while it completes a full poloidal turn. Here, R is the major radius of the toroidal device—and q is the safety factor associated with a given flux surface, i.e., the number of toroidal turns per poloidal turn for a field line on that surface. For an experiment like ASDEX Upgrade at Garching, Germany, we have $R \sim 1.65$ m and $q \sim 2$, which yields $L_c \sim 20$ m.

Meanwhile, the perpendicular extent of the fluctuations corresponds to several ion gyroradii. Noting that the gyroradius of thermal ions is given by $\rho_i = v_{ti}/\Omega_i$ where $v_{ti} = (k_B T_i/m_i)$ is the ion thermal velocity and $\Omega_i = (q_i B)/(m_i c)$ is the ion gyrofrequency, we have $\rho_i [\text{mm}] \sim 3.2 \sqrt{T_i [\text{keV}]/B [\text{T}]}$ for protons. (Here, k_B is Boltzmann's constant, T_i is the ion temperature, m_i is the ion mass, q_i is the ion charge, B is the magnetic field strength, and c is the speed of light.) In the case of ASDEX Upgrade, for $T_i [\text{keV}] = 4$ and $B [\text{T}] = 2.5$, one obtains a thermal proton gyroradius of about 3.6 mm. Measurements suggest that the perpendicular correlation lengths of the fluctuations are in the range of up to about $10 \rho_i$, i.e., a few cm. We are thus dealing with a parallel/perpendicular aspect ratio of the fluctuations of roughly 1000 in this example. The fluctuations are organized in thin, stretched out tubes which are aligned with the background magnetic field.

So how about the time domain? The frequencies of the fluctuations tend to span a fairly wide range around an average value which correlates with the characteristic frequency v_{ti}/R . For a device like ASDEX Upgrade, typical frequencies are of the order of a few 100 kHz. Moreover, time traces of fluctuation measurements do not exhibit any particular regularity (like periodicity). This kind of behavior suggests that the system is in a turbulent state, exhibiting spatio-temporal chaos.

What causes these fluctuations?

A plasma confined in a toroidal magnetic field is subject to a large number of waves and instabilities. The latter are often driven by the unavoidable density and temperature variations, and they are categorized in terms of macroinstabilities and microinstabilities, depending on their perpendicular scale lengths. Macroinstabilities exhibit variations on the largest scales of the system both parallel and perpendicular to the background magnetic field. They are typically described in the context of magnetohydrodynamic theory, and further details are provided elsewhere in this Encyclopedia. Microinstabilities, on the other hand, have spatio-temporal scales which align with those of the experimentally observed fluctuations.

Typical examples include ion temperature gradient modes and trapped electron modes, both of which are covered in a number of existing textbooks like [Weiland \(1999\)](#) and [Horton \(2017\)](#). Any accurate description of these microinstabilities is to be based on a (gyro-)kinetic treatment, as will be discussed in more detail below. They have been suggested as possible sources of the observed fluctuations for many years, and more recently, theory and simulation has advanced to a point where quantitative predictions can be made and then compared to experimental measurements. This has been successful in numerous cases, establishing the notion that turbulent transport driven by microinstabilities is a key factor in setting the energy confinement time of a fusion device. A snapshot from a numerical simulation with the gyrokinetic turbulence code GENE, described in [Jenko et al. \(2000\)](#) and [Görler et al. \(2011\)](#), is shown in [Fig. 1](#).

Turbulent transport: Bohm vs. gyro-Bohm

As in others physical systems like fluids or gasses, turbulence in plasmas is an excellent agent for transporting particles, momentum, and energy from one spatial domain to another. It is one of the key challenges for fusion research to understand, predict, and control turbulent transport.

A first question that arises in this context is: How is it possible that small-amplitude fluctuations (of the order of a percent or less) can have such a dramatic effect? A simple estimate of the turbulent ion heat diffusivity χ_i arising from the microinstabilities mentioned above can be obtained via the analogy to a random walk. Taking the size of a step Δx and the time elapsed between successive steps Δt to be given, respectively, by the perpendicular correlation length of the turbulence of several ion gyroradii and the inverse characteristic frequency of R/v_{ti} , the resulting turbulent heat diffusivity, $(\Delta x)^2/\Delta t$, can be estimated as



Fig. 1 Snapshot from a numerical simulation of microturbulence in the ASDEX Upgrade tokamak with the gyrokinetic code GENE. As described in the main text, the fluctuations of plasma density and temperature form thin, stretched out structures which are aligned with the background magnetic field. In the core plasma, the fluctuation amplitude is on the order of a percent.

$$\chi_i \sim \chi_g \equiv \rho_i^2 v_{ti} / R$$

Using the numbers quoted above for ASDEX Upgrade, this corresponds to a turbulent diffusivity of a few m^2/s , in line with what can be inferred from experimental measurements.

A second fundamental question is: How does this turbulent transport scale with system size? In other words, should we expect larger fusion devices (say, future power plants) to behave better or worse than present-day experiments in this regard? Well, for a given particle species, we obtain the scaling $\chi_i \propto T_i^{3/2} R^{-1} B^{-2}$. Given that all of three of these parameters will increase in ITER w.r.t. ASDEX Upgrade only by a moderate factor (of about 2–3.5), χ_i will roughly remain the same, namely in the range of a few m^2/s , provided that the employed scaling formula is correct. It is based on the assumption that the perpendicular correlation lengths of the turbulent fluctuations are related to ρ_i and do not increase with the system size. This idea has been questioned from time to time, however, and it was suggested instead that the perpendicular correlation lengths may in fact scale as $(\rho_i R)^{1/2}$, implying that radial variations of the density and temperature profiles (related to R) are a key ingredient in determining this quantity. This would lead to

$$\chi_i \sim \chi_B \equiv \rho_i v_{ti}$$

which is significantly larger than the estimate provided above. Given that χ_b is equivalent to the diffusion coefficient conjectured by David Bohm back in 1949 (up to a constant prefactor), it is called “Bohm diffusion.” And since χ_g involves an extra factor of ρ_i/R (which can be as small as $\sim 10^{-3}$ in ITER), it is termed “gyro-Bohm diffusion.” Obviously, if the Bohm scenario applied, this would be bad news for fusion research. Numerical simulations and experimental investigations suggest, however, that the majority of experimental situations is well described by the gyro-Bohm formula, in particular in larger devices.

On the nature of plasma turbulence

The most widely known example of a turbulent system is turbulence in a conventional fluid, as described by the well known Navier-Stokes equations. Plasma turbulence, on the other hand, deviates substantially from this scenario. In the following, we will first briefly describe the nature of turbulence in fluids and then highlight similarities and differences observed in plasma turbulence.

Turbulence in fluids

In conventional fluids, turbulent flows consist of eddies (or vortices) of a wide range of sizes which interact with each other non-linearly. Large eddies, typically driven by an external force, decay into somewhat smaller ones, which in turn decay into somewhat smaller ones etc. until viscous forces come into play and dissipate the fluctuations at small scales. This means that energy enters the system at the large drive scales and leaves at the small viscous scales. The scale range in between, characterized by the conservative transfer of energy in the absence of drive or dissipation, is called inertial range. In certain laboratory and natural systems, this inertial range can span many orders of magnitude. An excellent introduction to fluid turbulence can be found in [Davidson \(2015\)](#).

As it turns out, turbulence represents an interesting example of the coexistence of chaos and order. While the detailed dynamics in space and time depends very sensitively on all kinds of factors like the initial and boundary conditions or tiny perturbations of the system, the *averaged* properties of the turbulent flows in a saturated, quasi-steady state exhibit the existence of strongly reduced sensitivity and robust order in a *statistical* sense. It is considered a key challenge of turbulence research to predict these properties directly

from the Navier-Stokes equations. While a lot of progress has been made along these lines since the pioneering work of Kolmogoroff, Heisenberg, and others in the 1940s, a comprehensive theory of fluid turbulence remains elusive at this point.

Turbulence in astrophysical plasmas

Astrophysical systems are usually hot and/or dilute. This implies they are in a plasma state. Moreover, their dynamics tend to be strongly driven, inducing turbulence. The physics of turbulent plasmas is therefore a key aspect of many branches of astrophysical research. However, the basic nature of turbulence in astrophysical plasmas deviates from that in Navier-Stokes systems in fundamental ways. At the heart of this difference lies the fact that these plasmas—in contrast to conventional fluids—are characterized by the existence of waves. Therefore, turbulence has to be understood in terms of a large sets of waves (not eddies) interacting with each other.

Given the competition of linear and nonlinear dynamics in plasma turbulence, the question arises to which degree the properties of waves are retained or destroyed in a turbulent environment. If the influence of the nonlinear dynamics would be small, a system could be thought of as a collection of weakly interacting linear waves in the context of (the well-established) *wave turbulence theory*. If, on the other hand, the nonlinear dynamics would be dominant, the linear wave properties would become more or less irrelevant, and the system would resemble a turbulent fluid. As was first pointed out in Goldreich and Sridhar (1995), astrophysical plasmas tend to live exactly in the middle between these two extremes. The linear wave properties are perturbed substantially, but not wiped out completely. This feature is generally referred to as *critical balance*. In the turbulence state, many amplitude and phase relations between pairs of fluctuating quantities as prescribed by the linear dynamics are retained to some degree in fully developed turbulence. Meanwhile, conventional wave turbulence theory is not directly applicable. Instead, a new theory of strong wave turbulence needs to be developed.

Microturbulence in fusion plasmas

Interestingly, many other physical systems—from rotating fluids to Bose-Einstein condensates—share many of these features, as is discussed, e.g., in Nazarenko (2011). This actually includes microturbulence in fusion plasmas. A turbulent fusion plasma consists of a large collection of microinstabilities which interact with each other through nonlinear processes. This happens in a way that allows various linear wave properties to survive in the turbulent environment. The microinstabilities, of which there exist a number of different types, grow exponentially in time until their amplitudes reach a level at which linear and nonlinear terms in the underlying equations are in rough balance. At this point, nonlinear saturation takes place, and the system enters a quasi-steady state whose statistical properties, like turbulent transport coefficients, are of primary interest.

An example for this kind of dynamics is shown in Fig. 2. Here, the frequencies ω of the relevant microinstabilities (in this case, trapped electron modes and electron temperature gradient modes, both moving in the electron diamagnetic drift direction as indicated by the negative sign) are shown as a function of the perpendicular wavenumber k_y , with the data taken from a GENE simulation. For normalization, the quantities c_s and ρ_s are introduced; these correspond to v_{ti} and ρ_i but are evaluated using the electron temperature instead of the ion temperature. For linear modes, there exists a unique $\omega(k_y)$ relationship which is shown in red. In a nonlinear simulation, analyzing time traces of turbulent fluctuations associated with a particular value of k_y , one obtains a frequency distribution function whose center $\bar{\omega}$ and width $\Delta\omega$ are indicated in blue. It is important to note that $\Delta\omega \ll \bar{\omega}$ and $\bar{\omega} \sim \omega$. This finding is highly non-trivial and suggests that the linear modes interact nonlinearly in a significant way, while retaining their basic nature.

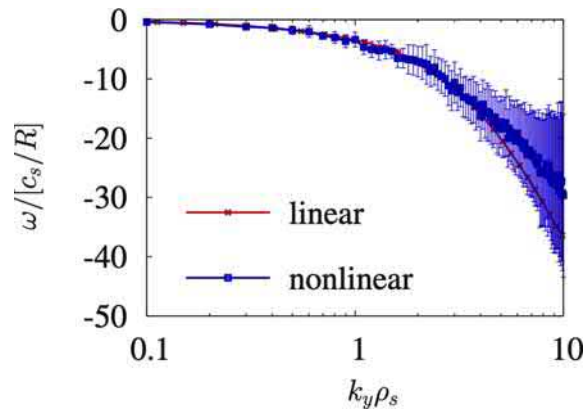


Fig. 2 Linear and nonlinear frequencies ω of the relevant microinstabilities as a function of the perpendicular wavenumber k_y , with the data taken from a GENE simulation; see Görler and Jenko (2008). These results suggest that the linear modes interact nonlinearly in a significant way, while retaining their basic nature, providing the basis for the development of reduced models of turbulent transport. Further details are provided in the main text.

The same conclusions can be drawn from inspection of other linear mode properties, like cross phases or amplitude ratios between pairs of fluctuating quantities, as is demonstrated, e.g., in [Dannert and Jenko \(2005\)](#). This provides the basis for the development of reduced models of turbulent transport which exploit these “quasilinear” properties and are very popular for providing (initial) interpretations of experimental results. In this context, it is assumed that the quasi-stationary turbulent state can be described by a superposition of linear waves, whose amplitudes are given by suitable simplified models that attempt to capture the essence of various nonlinear saturation mechanisms. Further details are provided, e.g., in [Horton \(2017\)](#).

Theoretical description of turbulent transport in fusion plasmas

A quantitative description of microturbulence in fusion plasmas needs to take into account a range of kinetic effects (like Landau damping and Finite Larmor Radius effects) which are beyond the scope of (standard) fluid theory or magnetohydrodynamics. In principle, one could solve the Vlasov equation (i.e., the collisionless Boltzmann equation) for the distribution function in six-dimensional position-velocity space for each particle species, calculating the self-consistent electric and magnetic fields with the help of Maxwell’s equations. Such a brute-force approach would imply that one needs to resolve many phenomena at extremely small spatio-temporal scales (like so-called Langmuir waves) which are irrelevant to the problem at hand. Fortunately, it was realized in the fusion theory community around the 1980s that a much more suitable plasma description exists, namely gyrokinetic theory.

Gyrokinetic theory of strongly magnetized plasmas

In fusion plasmas, the electrically charged electrons and ions move on helical trajectories along magnetic field lines. The characteristic frequencies of this gyromotion exceed the typical frequencies of the turbulent fluctuations by a substantial amount. Given this time scale separation, it is therefore natural to try to remove the fast dynamics of gyration from the equations analytically, considering the centers of charged rings instead of the particles themselves. Exploiting, in addition, various experimentally observed characteristics of the turbulent fluctuations (small amplitude, spatial anisotropy etc.), one can reduce the fully kinetic model to a gyrokinetic one, with Maxwell’s equations being replaced by a gyrokinetic version thereof.

A visual representation of this procedure is provided in [Fig. 3](#). A charged particle embedded in a strong magnetic field exhibits a helical trajectory, and the particle position $\mathbf{x}$ (open circle) can be expressed in terms of the guiding center position $\mathbf{X}$ (filled circle) according to

$$\mathbf{x} = \mathbf{X} + \mathbf{r}$$

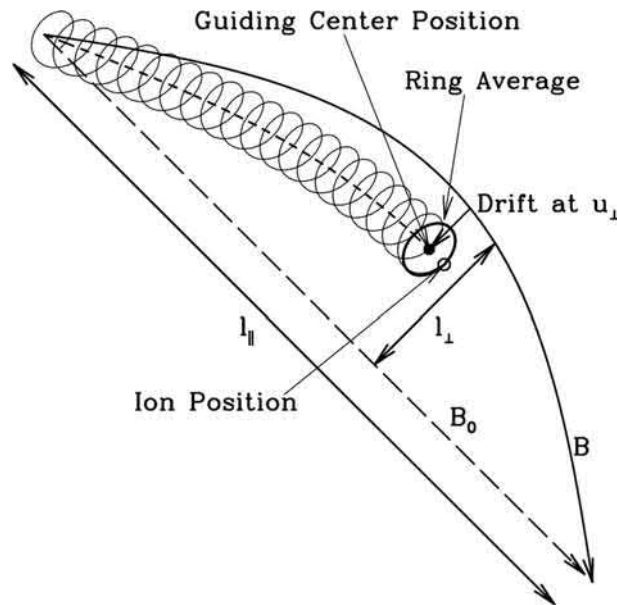


Fig. 3 Visual representation of the basic idea behind gyrokinetics. A charged particle embedded in a strong magnetic field exhibits a helical trajectory about the time-varying guiding center position. The motion of the particle is replaced by the motion of the guiding center, with the forces acting on the particle being averaged along respective rings. Further details can be found in the main text. Taken from Howes GG, Cowley SC, Dorland W, Hammett GW, Quataert E, and Schekochihin AA (2006) Astrophysical gyrokinetics: Basic equations and linear theory. *The Astrophysical Journal* 651: 590.

with

$$\mathbf{r} \equiv \frac{\mathbf{b} \times \mathbf{v}}{\Omega}$$

where $\mathbf{b}$ is the unit vector along the magnetic field, $\mathbf{v}$ is the particle's velocity, and $\Omega = (qB)/(mc)$ is the particle's gyrofrequency, all evaluated at the current particle position. The guiding center position tends to follow the unperturbed magnetic field B_0 (long-dashed line), but deviates from it due to perpendicular drifts (the sum of the curvature, ∇B , and $E \times B$ drifts) and magnetic field perturbations B (solid line). Given that the gyromotion is very fast in comparison with all other time scales of interest, it is averaged out. In this context, the motion of the particle is replaced by the motion of the guiding center, and the forces acting on it are averaged along respective rings. Here, the characteristic perpendicular and parallel length scales of the plasma fluctuations are denoted by $l_{\parallel}$ and $l_{\perp}$, respectively. As discussed before, $l_{\parallel} \gg l_{\perp}$.

Derivation of the gyrokinetic equations

While it may be natural to think that gyrokinetic theory is best derived from kinetic theory on the level of equations of motion, it turns out that there is a much better approach. In the context of classical mechanics, it is shown that a system's dynamics can be described with the help of a so-called Lagrangian. This mathematical object allows you to readily derive the equations of motion by evaluating the Euler-Lagrange equations. More information is provided in textbooks on Classical Mechanics like [Goldstein \(2013\)](#). So the challenge becomes how to derive a gyrokinetic Lagrangian from the kinetic one. The description of all explicit steps that are required to achieve this task is clearly beyond the scope of this text. The interested reader may want to consider exploring the overview paper by [Brizard and Hahm \(2007\)](#) and some of the literature cited therein. In the following, we will simply write down the Lagrangian of a kinetic system, its gyrokinetic counterpart, and the resulting gyrokinetic equations of motion.

At this point, one should recall that the equations of motion for a charged particle in prescribed electric and magnetic fields,

$$\dot{\mathbf{x}} = \mathbf{v}, \quad \dot{\mathbf{v}} = \frac{q}{m} \left(\mathbf{E} + \frac{\mathbf{v}}{c} \times \mathbf{B} \right)$$

can be derived from the Lagrangian

$$\mathcal{L} = \left(\frac{q}{c} \mathbf{A} + m\mathbf{v} \right) \cdot \dot{\mathbf{x}} - \left(\frac{m}{2} v^2 + q\phi \right)$$

via the Euler-Lagrange equations,

$$\frac{d}{dt} \frac{\partial \mathcal{L}}{\partial \dot{q}_i} - \frac{\partial \mathcal{L}}{\partial q_i} = 0$$

with $\mathbf{q} = (\mathbf{x}, \mathbf{v})$ as a six-component vector. Here, the vector and scalar potentials, $\mathbf{A}$ and ϕ , as well as the electric and magnetic fields,

$$\mathbf{E} = -\nabla\phi - \partial_t \mathbf{A}/c, \quad \mathbf{B} = \nabla \times \mathbf{A}$$

are generally dependent on $\mathbf{x}$ and t .

Neglecting—for the moment—fluctuations and transitioning from particle coordinates $(\mathbf{x}, \mathbf{v})$ to guiding center coordinates $(\mathbf{X}, v_{\parallel}, \mu, \varphi)$, where $\mathbf{X}$ is the guiding center position, $v_{\parallel}$ is the component of $\mathbf{v}$ parallel to the local magnetic field, $\mu \equiv (mv_{\perp}^2)/(2B)$ is the magnetic moment, and φ is the gyrophase, one obtains a new Lagrangian, formulated now with respect to the guiding center coordinates,

$$\mathcal{L} = \left(\frac{q}{c} \mathbf{A} + mv_{\parallel} \mathbf{b} \right) \cdot \dot{\mathbf{X}} + \frac{\mu B}{\Omega} \dot{\varphi} - \left(\frac{m}{2} v_{\parallel}^2 + \mu B + q\phi \right)$$

where $\mathbf{A}$ and ϕ (as well as B and Ω) depend on the guiding center position $\mathbf{X}$, not on the particle position $\mathbf{x}$. The resulting Euler-Lagrange equations read

$$\dot{\mathbf{X}} = v_{\parallel} \mathbf{b} + (B/B^*) (\mathbf{v}_E + \mathbf{v}_{\nabla B} + \mathbf{v}_c)$$

$$\dot{v}_{\parallel} = (q\mathbf{E} - \mu \nabla B) \cdot \dot{\mathbf{X}} / (mv_{\parallel}), \quad \dot{\mu} = 0, \quad \dot{\varphi} = \Omega$$

where

$$\mathbf{v}_E \equiv \frac{c}{B^2} \mathbf{E} \times \mathbf{B}, \quad \mathbf{v}_{\nabla B} \equiv \frac{\mu}{m\Omega} \mathbf{b} \times \nabla B, \quad \mathbf{v}_c \equiv \frac{v_{\parallel}^2}{\Omega} \mathbf{b} \times (\mathbf{b} \cdot \nabla) \mathbf{b}$$

are, respectively, the $E \times B$ drift, ∇B drift, and curvature drift, and

$$B^*/B = \mathbf{b} + (v_{\parallel}/\Omega) \nabla \times \mathbf{b}, \quad B^* = \mathbf{b} \cdot \mathbf{B}^*$$

In the next step, small fluctuations are taken into account, assuming that they satisfy the experimentally observed characteristics of $\omega \ll \Omega$ and $k_{\parallel} \ll k_{\perp}$. This leads to additional terms in the Lagrangian,

$$\mathcal{L}_1 = \left(\frac{q}{c} \bar{\mathbf{A}}_{\parallel 1} \mathbf{b}_0 \right) \cdot \dot{\mathbf{X}} - \left(\mu \bar{B}_{\parallel 1} + q \bar{\phi}_1 \right)$$

and associated terms in the equations of motion. In particular, three new drifts are introduced,

$$\mathbf{v}_{m1} \equiv v_{\parallel} \left(\bar{\mathbf{B}}_{\perp 1} / B_0 \right), \quad \mathbf{v}_{E1} = \frac{c}{B^2} \bar{\mathbf{E}}_1 \times \mathbf{B}, \quad \mathbf{v}_{B1} = \frac{\mu}{m\Omega} \mathbf{b} \times \nabla \bar{B}_{\parallel 1}$$

with $\bar{\mathbf{E}}_1 = -\nabla \bar{\phi}_1 - \mathbf{b} \partial_t \bar{A}_{\parallel 1} / c$ and $\bar{\mathbf{B}}_{\perp 1} = \nabla \times \left(\bar{A}_{\parallel 1} \mathbf{b} \right)$. Here, $\mathbf{v}_{m1}$ describes the motion perpendicular to the background magnetic field due to particles following fluctuating field lines, while $\mathbf{v}_{E1}$ and $\mathbf{v}_{B1}$ describe, respectively, contributions to the $E \times B$ drift and ∇B drift due to fluctuations in the electrostatic potential and parallel magnetic field. The overbar designates fluctuating quantities which are gyroaveraged appropriately.

The corresponding gyrokinetic Vlasov equation describing the time evolution of the distribution function F in five-dimensional gyrocenter space $\mathbf{Z} = (\mathbf{X}, v_{\parallel}, \mu)$, with the gyrophase dependence removed, can be written in advection form as

$$\left(\frac{\partial}{\partial t} + \dot{\mathbf{Z}} \cdot \frac{\partial}{\partial \mathbf{Z}} \right) F(\mathbf{Z}, t) = 0$$

It is the first cornerstone of gyrokinetics. The second one is the gyrokinetic version of Maxwell's equations, which is needed to calculate the self-consistent fluctuating electromagnetic fields. For further details on this topic and on gyrokinetic theory in general, the interested reader is referred to [Brizard and Hahm \(2007\)](#). The goal of the present introduction was merely to provide a flavor of the kind of physics and mathematics underlying state-of-the-art computations of turbulent transport in fusion devices.

Computational gyrokinetics

The complete system of gyrokinetic equations, including a modified version of Maxwell's equations, constitutes a set of nonlinear partial differential equations for a distribution function that depends on the three spatial coordinates, two velocity space coordinates, and time. As can be expected, it is impossible to provide exact analytic solutions to these equations for realistic cases. Instead, one has to resort to numerical solutions. Several gyrokinetic simulation codes have been developed over the years, and most of them fall into one of the following two categories.

- Particle-in-cell (PIC) codes adopt a Lagrangian view of the phase space dynamics, exploiting the fact that the value of the distribution function of a given particle species does not change along phase space trajectories. This can be expressed mathematically in terms of $df/dt = 0$, i.e., the total time derivative of the distribution function vanishes. One thus distributes a very large number of so-called marker particles across phase space and numerically computes their trajectories. To obtain the self-consistent electric and magnetic fields at each time step, the resulting charge and current densities are calculated on a fixed spatial grid, and the field equations are solved on that grid. This unique combination of numerical techniques is what gives this approach its name. The greatest advantage of the PIC approach may be that it is intuitively accessible, while its most significant drawback is the existence of numerical noise due to the use of a Monte-Carlo-type sampling approach. The accuracy of a given simulation increases only as $N^{-1/2}$ where N is the number of particles used in the simulation.
- Continuum codes, on the other hand, are based on a Eulerian view of the phase space dynamics, often adopting numerical methods originally developed in the context of Computational Fluid Dynamics (CFD). Here, the basic idea is to define small volumes in phase space and monitor how many particles enter and leave each of these volumes per time unit. In practice, combinations of finite difference, finite volume, and spectral methods often tend to be employed. In terms of time stepping, explicit methods tends to dominate, but the use of combinations of explicit and implicit methods has also been explored. Continuum codes are less straightforward to develop, often requiring highly original combinations of numerical methods, but they can readily be adapted and optimized for a large range of situations and do not suffer from numerical noise. In most cases, higher order methods are employed, ensuring fast convergence with increasing resolution.

Both types of codes have been used successfully to study fundamental aspects of turbulence in fusion plasmas and to interpret specific experimental results, and an overview of key activities in this area up to a certain point is provided in [Garbet et al. \(2010\)](#). Due to their complementary strengths, they are sometimes used in tandem to explore certain physics questions. Running realistic cases with high numerical resolution requires significant computer resources. For many years, gyrokinetic simulation has therefore been at the forefront of High Performance Computing (HPC). A lot of effort is put into the parallelization and optimization of gyrokinetic codes on the world's largest and most powerful supercomputers. This is the prerequisite for exploring a number of key physics issues in magnetic confinement fusion research.

Towards validated and predictive gyrokinetic simulations

Over the past two decades or so, gyrokinetic simulations have become increasingly comprehensive and realistic, allowing for direct comparisons between simulation results and experimental observations. In a large number of dedicated validation studies, the ability of these tools to provide a quantitative description of plasma microturbulence has been tested successfully. Both the

transport levels and various detailed properties of turbulence fluctuations in experiments have been recovered in gyrokinetic simulations. While, up to this point, the goal of such investigations has been to help interpret existing experiments, in particular with respect to the observed turbulent transport levels in the core region of fusion plasmas, it is becoming increasingly clear that we are about to enter a new phase in this kind of endeavor.

Emerging new focus areas include the interaction of plasma turbulence with other physical processes like MHD instabilities or energetic particles, a quantitative description of turbulence in the core of stellarators, and the treatment of the boundary region of fusion devices. Moreover, there is a growing expectation within the scientific community for these kinds of simulations to become truly predictive. This will be needed, in particular, to optimize the design and operation of future fusion experiments and power plants. The guidance provided by gyrokinetic simulations will be invaluable in this context, allowing for significant savings of development time and financial resources on the experimental side, thus accelerating fusion research. Obviously, such an ambitious research program calls for the inclusion of state-of-the-art techniques from applied mathematics and computer science. The main goal is to quantify the physics processes driving turbulence and to control the turbulent transport in various ways, to maximize the energy confinement time for any given device, avoid impurity accumulation, and help provide solutions to the problem of heat and particle exhaust.

Controlling turbulent transport

Turbulent fluctuations and the associated transport across flux surfaces have been observed in every fusion experiment that has been built and operated in the last few decades. While collisional (“neoclassical”) transport is known to also be significant under certain circumstances, including the core regions of almost all stellarators, it is *turbulent* transport that usually sets the energy confinement time of a given device, and thus a key performance indicator. Interestingly, several kinds of improved confinement regimes have been identified experimentally. And gyrokinetic simulations are being applied to gain a deeper understanding of the underlying physical processes. In the following, we will provide an overview of some recent and ongoing activities along these lines.

Improved confinement regimes: The L-H transition

As is explained elsewhere in this Encyclopedia, there exist several improved confinement regimes—with respect to the so-called low confinement mode or L-mode, which provides a baseline level of confinement in most magnetic confinement devices—that have been discovered experimentally. The most prominent example is the so-called high confinement mode or H-mode. The phenomenon of the L-H transition was first observed in the early 1980s in the ASDEX tokamak at Garching, Germany, see [Wagner \(1984\)](#), and it has been confirmed in countless other devices since then. The H-mode is the default scenario for high-performance operation in virtually all existing and planned tokamaks, including ITER. Therefore, the development of a detailed understanding of the L-H transition is of crucial importance to the fusion community, and reviews are provided in [Connor and Wilson \(2000\)](#), [Wagner \(2018\)](#), and [Bourdelle \(2020\)](#).

From the experimental side, one observes that the plasma exhibits a sudden transition to a new state which is characterized by a transport barrier—i.e., a narrow radial region of only several cm with strongly reduced turbulent heat and particle transport, leading to steep temperature and density profiles—near the plasma boundary, just inside the last closed flux surface. An example

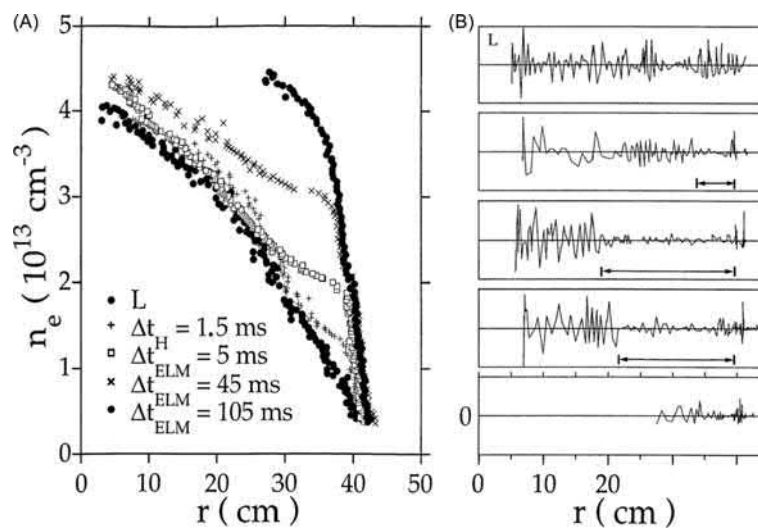


Fig. 4 L-H transition dynamics as observed in the ASDEX tokamak: (A) development of the edge density profile during the transition; (B) radial variation of edge density fluctuations for the same five time points. Taken from Wagner F, et al. (1991) *Plasma Physics and Controlled Nuclear Fusion Research*, vol. 1, p. 277. Vienna: IAEA.

from the ASDEX tokamak is shown in Fig. 4. Moreover, one finds that the transition occurs once the heating power exceeds a certain threshold. The latter can be described by an empirical formula reported in Martin (2008),

$$P_{th} = 0.049 \bar{n}_e^{0.72} B_t^{0.80} S^{0.94}$$

where P_{th} is the power threshold in MW, $\bar{n}_e$ is the line-averaged density in 10^{20} m^{-3} , B_t is the toroidal magnetic field in T, and S is the plasma surface area in m^2 . For ITER, with $B_t = 5.3 \text{ T}$ and $S = 678 \text{ m}^2$, one obtains $P_{th} = 52 \text{ MW}$ if one assumes $\bar{n}_e = 0.5 \cdot 10^{20} \text{ m}^{-3}$, with the 95% confidence interval extending all the way from 28 to 96 MW. For smaller or larger values of $\bar{n}_e$, all of these numbers have to be rescaled by a factor of $(\bar{n}_e/0.5)^{0.72}$, of course. Given that, at the beginning of its operation, ITER will have 73 MW of

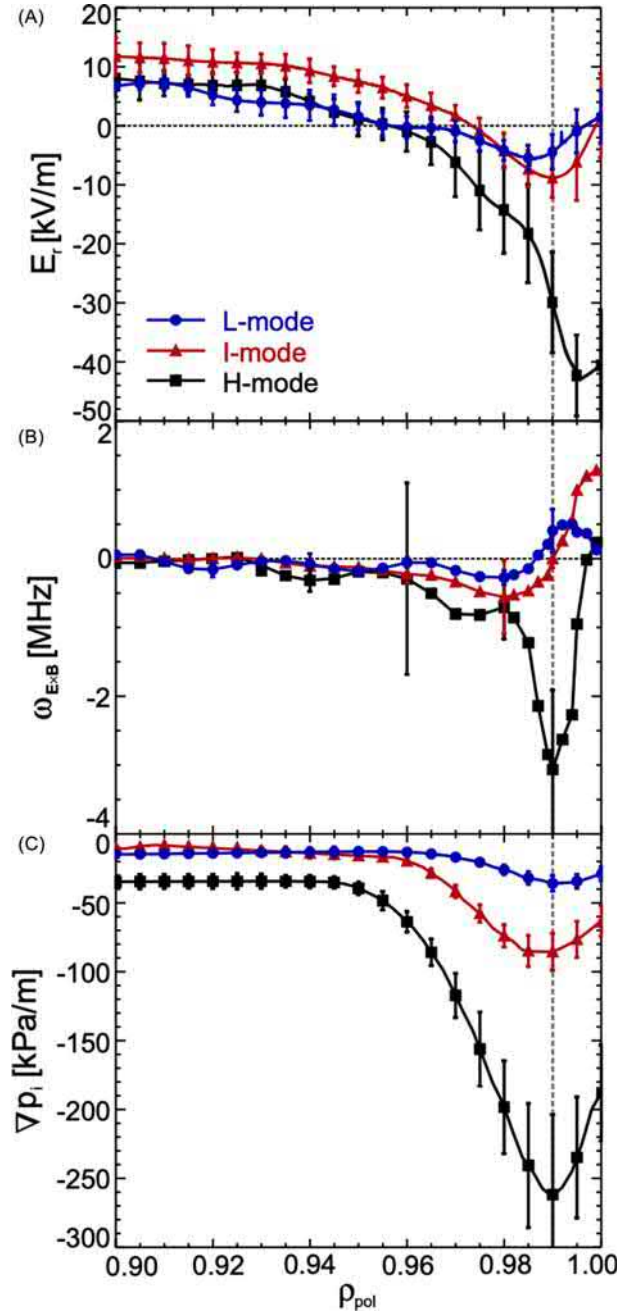


Fig. 5 Edge radial electric fields E_r —along with the resulting $E \times B$ shearing rates $\omega_{E \times B}$ and ion pressure gradients ∇p_i —measured in different confinement regimes of the ASDEX Upgrade tokamak, as a function of the radial coordinate ρ_{pol} . In L-mode, the shearing rates are roughly comparable to the typical frequencies of the turbulent fluctuations, while in H-mode, they exceed them by about an order of magnitude, suggesting the existence of a pronounced suppression effect. (I-mode is another improved confinement regime which is not discussed here.) Taken from Viezzer E., et al. (2013) High-accuracy characterization of the edge radial electric field at ASDEX Upgrade. *Nuclear Fusion* 53: 053005.

additional heating power, H-mode access at higher density might become marginal. This highlights the necessity for the development of a predictive computational capability for the L-H transition.

One of the key findings in the context of L-H transition studies is the fact that a large radial electric field is created near the transition from open to closed magnetic field lines at the separatrix, as shown in Fig. 5, which is taken from Viezzer et al. (2013). Such an electric field leads to $E \times B$ flows within flux surfaces which vary rapidly as a function of the flux surface label, resulting in strong shear flows which can be characterized by the so-called $E \times B$ shearing rate $\omega_{E \times B}$, i.e., the radial derivative of the $E \times B$ drift velocity associated with a given radial electric field. As it turns out, these sheared flows can be driven either by the turbulence itself or by a number of other mechanisms (like collision-induced effects in the presence of steep plasma profiles or highly energetic ions in the near-edge region which leave the plasma). They distort and destroy the turbulent structures associated with the underlying driving microinstabilities, thus suppressing the turbulence. As a consequence, the temperature and density profiles increase rapidly in this radial region, forming a pedestal-type structure.

The turbulence suppression mechanism is illustrated in Fig. 6, taken from Itoh et al. (1999), using a simplified geometry. Here, the magnetic field points in the z direction, while x and y represent, respectively, the radial and poloidal directions. Assuming that the poloidal component of the $E \times B$ velocity can be described as $V_y = S_v x$, we consider the time evolution of a turbulent structure which is circular at $t = 0$. At a later point in time, the circle of radius L will get distorted into an ellipse, with its top being displaced w.r.t. its bottom by $LS_v t$. Given that the area of the structure is preserved, one can easily calculate that $L_\perp = L(1 + S_v^2 t^2)^{-1/2}$. This means that for $t \gg S_v^{-1}$, the perpendicular wavenumber increases linearly with time, triggering linear and nonlinear damping mechanisms which reduce the turbulence level. The temperature and density profiles steepen accordingly, similar to what happens when one puts an extended insulating layer around an object (like a building or a container).

Given that one finds experimentally that the ratio of the core and pedestal temperatures tends to be roughly constant in many relevant regimes, the core profiles will adapt to these changes in the boundary conditions. This can be explained theoretically by the fact that one of the most important microinstabilities driving the turbulence, the ion temperature gradient mode, is only destabilized once a threshold in $\nabla T_i/T_i$ is exceeded. (Similar consideration also apply to other types of microinstabilities, like trapped electron modes.) The pressure in the plasma center can thus be increased substantially, providing better conditions for a fusion plasma to satisfy the Lawson criterion for a burning plasma.

While individual aspects of the formation of transport barriers have been identified and studied theoretically and computationally in isolation, a comprehensive and self-consistent description of an L-H transition remains elusive. Apparently, the type of gyrokinetic models being applied to this problem as of today are not yet sufficiently comprehensive. There is a general consensus in the community that beyond the well-established capabilities of core codes—which include collisions, magnetic field fluctuations, and cross-scale coupling effects—the following issues have to be addressed in any case:

- A simulation domain covering the entire plasma boundary, involving both the outer core region and the scrape-off layer (SOL), dealing with the complexities introduced by the existence of a magnetic separatrix (last closed flux surface) and open magnetic

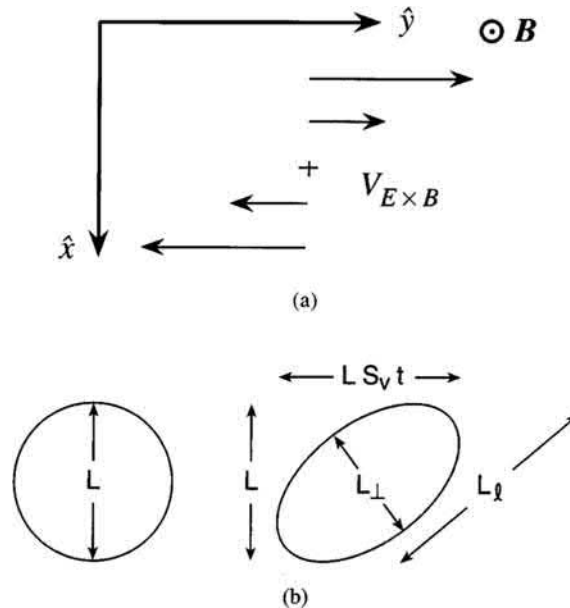


Fig. 6 Effect of sheared $E \times B$ flows—resulting from a radially dependent radial electric field—on turbulent structures, using a simplified geometry. Here, the magnetic field points in the z direction, while x and y represent, respectively, the radial and poloidal directions. Further details are provided in the main text. Taken from Itoh K, Itoh SI and Fukuyama A (1999) *Transport and Structural Formation in Plasmas*. Institute of Physics Publishing.

field lines. This implies that the standard methods for adapting the coordinate system to the background magnetic field as used in the core are not applicable here.

- A self-consistent treatment of the plasma background and fluctuations, based on the fact that fluctuation levels in the SOL can be as large as several 10%. This implies that codes optimized for the core region cannot be applied, and that new codes have to be developed, based on adequate numerical schemes.
- The inclusion of additional particle species, some of which are not ionized at all or only partially ionized. This is due to the fact that the plasma in the boundary region is relatively cold and close to the wall, which can act as a source of particles. This also implies that one needs to deal with the serious computational challenges associated with the treatment of plasma-wall interaction effects.

Such tools are currently under development, raising the expectation that a self-consistent description of the L-H transition may gradually come within reach. Various synergies with existing fluid boundary codes can be exploited in this context.

Meanwhile, existing gyrokinetic codes are being used to investigate specific aspects of the L-H transition. This includes, in particular, the nature of the turbulent transport in the edge of an established L-mode and H-mode discharge, various mechanisms to create shear flows in the plasma boundary, and effects of neutrals and impurities on the edge turbulence. Based on these building blocks, it is planned to gradually develop a comprehensive picture of the physics of this intriguing phenomenon. Once validated against existing experiments, the goal would be to apply such simulation capabilities to predict the performance of ITER and DEMO.

We note in passing that there also exists a range of other improved confinement regimes which is being studied. Moreover, transport barriers also occur in the core region, where they are called internal transport barriers (as opposed to the edge transport barriers discussed above). Empirically, their formation also depends on the occurrence of favorable local plasma conditions, e.g., in terms of magnetic shear or flow shear.

Transport-by-design in stellarators

A fascinating new research opportunity in the context of turbulent transport optimization is provided by the advent of optimized stellarators. Whereas the cross-field transport of particles and heat in the core region (but not the edge region) of conventional stellarators tends to be dominated by collisional (i.e., neoclassical) processes, a new situation presents itself to the fusion community in the case of optimized stellarators like Wendelstein 7-X (W7-X). Here, the magnetic geometry has been optimized in a way so that collisional transport processes are minimized. As recent experimental transport studies in W7-X have revealed, turbulence becomes the dominant transport mechanism under these circumstances even in the core region. Given that the magnetic field structure is much more complex than that in tokamaks, the question arises if this feature can be exploited to control the turbulence. First theoretical and computational studies have confirmed the suspicion, paving the way towards a realization of the notion of transport-by-design in stellarators.

As has been discussed above, turbulence in fusion plasmas is driven by various kinds of microinstabilities whose properties are determined by the structure of the background magnetic field and the density and temperature profiles of the plasma, and which tend to persist to some degree in the saturated quasi-steady state, as discussed above. Given that the magnetic field of a non-axisymmetric stellarator has many more degrees of freedom than an axisymmetric tokamak, this feature can be exploited to influence the properties of turbulent transport in significant ways, as has been shown in gyrokinetic simulations (see Fig. 7).

Since any kind of optimization calls for a very large number of model evaluations, an approach involving high-resolution gyrokinetic simulations directly is not viable, given that each single simulation typically requires significant resources on large-scale supercomputers. A more realistic approach is to develop an approximate theoretical framework which allows to identify

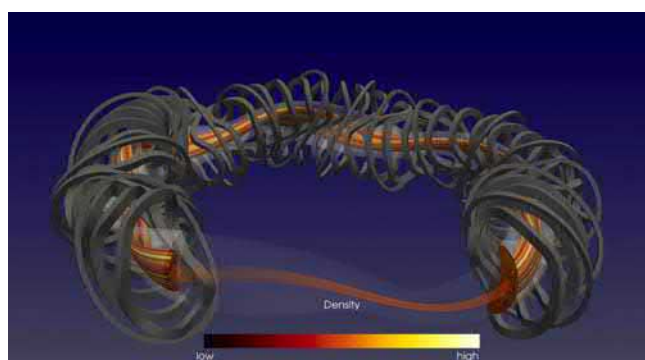


Fig. 7 Snapshot from a numerical simulation of microturbulence in the Wendelstein 7-X stellarator with the gyrokinetic code GENE-3D; see [Maurer et al. \(2020\)](#). Such simulations can be used to systematically investigate the effects of various aspects of the magnetic geometry on the turbulent transport, supporting the notion of transport-by-design.

important parameter dependencies, carry out a limited number of gyrokinetic simulations, and evaluate them carefully in the context of this framework. On this basis, reduced transport models can be constructed and then applied within suitable optimization procedures. Such approaches are currently being developed and will be used to aid the design of future stellarator devices.

Conclusions and outlook

Given the high temperatures and low densities of magnetically confined fusion plasmas, the collision frequencies in these systems tend to be extremely low, calling for a kinetic approach instead of a fluid one. Fortunately, many small spatio-temporal scales which are irrelevant to the problem at hand can be eliminated analytically from the fundamental equations. The resulting model is called gyrokinetics, based on the idea of averaging out the fast gyromotion of the charged particles, thus reducing the number of phase space dimensions from six to five by removing the gyrophase dependence. The resulting set of nonlinear partial differential equations requires a numerical treatment, and two main types of codes have been developed, PIC codes and continuum codes, which are used in a complementary fashion.

Gyrokinetic simulation has come a long way since its inception in the 1980s. Following a phase of pioneering studies using very simplified physical settings, there was a significant increase regarding the level of realism since about the year 2000. Over the last decade or so, a series of careful validation studies has been carried out successfully, demonstrating the validity and power of gyrokinetic simulation. The logical next step is to gradually transition from interpretive to predictive studies, providing guidance for the design of next-generation fusion experiments and first power plants. This ambitious plan will be aided by the emergence of exascale supercomputers in the early 2020s, allowing scientists to carry out computations involving more than 10^{18} floating point operations per second.

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Magnetic-Confinement Fusion—Plasma Theory: Tokamak Magnetohydrodynamic Equilibrium and Stability

Lang L. Lao, Y.Q. Liu, and Alan D. Turnbull, General Atomics, San Diego, CA, United States

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Abbreviations

Alfvén velocity Alfvén wave traveling velocity
Beta The ratio of plasma stored energy to confining magnetic energy
DIII-D Doublet III dee-shaped tokamak
Disruption Rapid termination of a plasma discharge
EFIT Equilibrium reconstruction and fitting code
ELM Edge localized mode
Flow shear A measure of radial variation of plasma angular flow across magnetic surfaces

H-mode High confinement mode

ITER International thermonuclear experimental reactor, an international project constructing tokamak designed to study fusion plasma production in deuterium-tritium plasmas

Magnetic shear A measure of radial variation of safety factor across magnetic surfaces

Magnetic surface A surface formed by magnetic-field lines

MHD Magnetohydrodynamics

Mode rational surface A magnetic surface where the safety factor has a rational value

RWM Resistive wall mode

Safety factor The number of toroidal transits for each poloidal transit when following a magnetic-field line

Sawtooth A periodic relaxation in the tokamak plasma central region

Tokamak A donut shaped toroidal confinement device

VDE Vertical displacement event

Introduction

In magnetic-fusion confinement devices such as tokamaks, to remain confined and produce fusion power, the plasma must be maintained in a force-balance equilibrium state and stable to perturbations around this state. To keep the plasma away from the surrounding vacuum vessel, the expanding hot-plasma pressure force, and the centrifugal force if the plasma is rotating, must be balanced by a counter magnetic force driven by electrical currents flowing in the external coils. As schematically illustrated in Fig. 1A and B, a stationary ball sitting in the bottom of a valley represents a stable equilibrium state, whereas the one sitting on the top of a hill represents an unstable equilibrium state.

Magnetohydrodynamics (MHD) (Bateman, 1978; Freidberg, 1982) provides a useful model to describe and evaluate this macroscopic plasma equilibrium and stability state by considering the plasma as a conducting fluid confined in a restricted region of space by the electromagnetic force from an externally imposed magnetic field. The magnetic field can be conveniently visualized and represented using magnetic field lines (Boozer, 2005). In an axisymmetric toroidal tokamak device, the magnetic field lines form a set of nested magnetic surfaces in a donut shape. The geometric and topological properties of the magnetic surfaces such as the aspect ratio, elongation, triangularity and squareness play important roles in determining the stability and performance of tokamak plasmas, and can be controlled using a set of external poloidal-field coils.

An important MHD application is to design an optimal set of external poloidal magnetic-field coils to produce a target plasma equilibrium shape that is stable to instabilities. An essential magnetic-fusion research element is to develop the physics basis to support such activities. This requires design and performance of experiments to test the equilibrium and stability properties of the plasma, development and validation of MHD equilibrium and stability physics models to explain the experimental observations, and development of diagnostics and computational tools to accurately measure and efficiently reconstruct the plasma state to interpret the measurements and facilitate the validation.

In an ideal perfectly conducting plasma, the plasma is tied to the magnetic surfaces. In the absence of collisions among ions and electrons and without large drift motions due to magnetic-field gradient and curvature in the magnetic-field lines, the magnetic surfaces act as a container to keep the plasma away from the surrounding vacuum-vessel wall and plasma facing components (PFCs). The hot plasma pressure and current flowing along the magnetic field lines can act as free-energy sources to drive the plasma away from the desired equilibrium and into various unstable states depending on the particular plasma operating configurations and conditions. The tension and shear in the magnetic-field lines can act as restoring forces to stabilize and keep the plasma in a stable equilibrium state. An excessive amount of plasma stored energy relative to the confining toroidal magnetic field energy

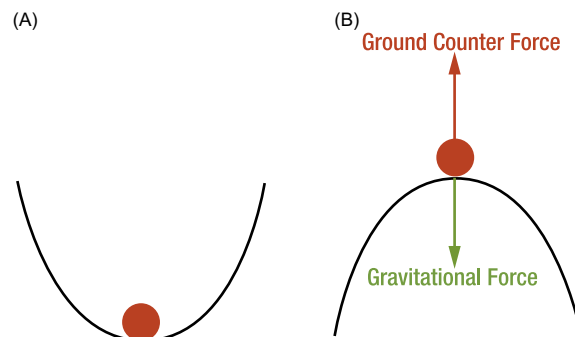


Fig. 1 (A) A stable equilibrium state, (B) an unstable equilibrium state.

or a large amount of flowing plasma current relative to the confining toroidal magnetic-field coil current can drive MHD instabilities that can potentially damage and shorten the lifetime of the confining device PFCs and in-vessel structures.

Two useful physics parameters to describe these important plasma stability properties are the safety factor q and the plasma toroidal beta β_T . For an axisymmetric toroidal device such as a tokamak, these are defined as

$$q(\psi) = \frac{F(\psi)}{2\pi} \oint_{\psi} \frac{dl_p}{B_p} \frac{1}{R^2} \quad (1)$$

$$\beta_T = \frac{\int_0^{\Omega_B} P d\Omega}{\Omega_B B_{T0}^2 / 2\mu_0} \quad (2)$$

Here, B_T and B_p are the toroidal and poloidal component of the magnetic field $\mathbb{B}$ from the electric currents flowing within the plasma plus those flowing in the external toroidal and poloidal magnetic-field coils. $F(\psi) = 2\pi R B_T / \mu_0$ is the poloidal current stream function, R is the plasma major radius, ψ is the poloidal flux per radian of the toroidal angle, and B_{T0} is the vacuum toroidal magnetic field at the geometric center of the plasma boundary surface. The line integration in Eq. (1) is along a poloidal cross section of a magnetic flux surface labeled by ψ , $\Omega(\psi)$ is the volume enclosed by the magnetic surface ψ , and Ω_B is the plasma volume. The safety factor q describes the average pitch of the magnetic field lines along a magnetic surface weighted by the geometric properties of the surface. q represents the number of turns a magnetic field line has to traverse toroidally before completing a poloidal turn.

Additionally, depending on the plasma transport conditions, strong pressure or current gradients can develop locally in a small region near a transport barrier that can act as free-energy source to drive the plasma into an unstable state (Lao, 2000). In the tokamak high confinement mode (H-mode) regime (Wagner, 2007; Burrell et al., 1987), a large edge pedestal pressure gradient can develop leading to edge localized modes (ELMs) (Leonard, 2014). Large ELMs can shorten the life time of the divertor wall.

MHD is also applied to develop robust techniques to mitigate and control MHD instabilities (Igochine, 2015). At high β_T or low q , a small perturbation can lead to unstable plasma motion that strongly distorts the magnetic surfaces allowing the plasma to quickly release its thermal and magnetic energy in a very short time scale, which can potentially damage the PFCs and in-vessel structures. Additionally, at rational q surfaces the magnetic field lines close on themselves rather than spanning the entire surface. A magnetic perturbation that has toroidal and poloidal mode numbers in sync with the pitch of the magnetic field lines can then interact resonantly with the magnetic field and strongly distort the magnetic surfaces, thus allowing the heat flux to quickly transport and escape to the surrounding vacuum-vessel wall. Additionally, together with the plasma resistivity and toroidal asymmetry, magnetic islands and stochastic regions can form around the rational q surfaces within the plasma that can tear open the magnetic surfaces and allow particle and heat flux to also redistribute and escape to the vessel wall. Impurities sputtered from the wall can then enter and accumulate in the plasma and can lead to a radiation induced thermal collapse and rapid decay of the plasma current. The shrinkage of the current channel in the plasma column can make it susceptible to vertical displacement events (VDEs) and then disruption, which can potentially damage the device PFCs and in-vessel structures due to the large attached halo current induced in the surrounding wall (Boozer, 2012; Lao and Jensen, 1991; Lehnen et al., 2015; Clauser et al., 2019).

In this article, the physics principles of toroidal tokamak equilibrium and stability are first discussed, including the derivation of the MHD equations from the Boltzmann kinetic equation. This is then followed by a section on toroidal equilibrium including a description of equilibrium reconstruction and the inverse representation. MHD stability including linear stability and the energy principle, the effects of plasma resistivity, energetic particles, and plasma toroidal flow is then discussed. A discussion of major MHD instabilities including axisymmetric modes, internal kinks (IKs), resistive-wall modes (RWMs), edge localized modes (ELMs), tearing modes (TMs), and toroidal Alfvén eigenmodes (TAEs) then follows. The plasma response to external 3D magnetic fields and numerical tools to compute MHD instabilities are also described. Lastly, a short discussion of methods and techniques to mitigate and control plasma instabilities including disruptions is given.

Physics principles of toroidal equilibrium and stability

MHD model and equations

MHD is the most basic plasma model, incorporating most large-scale phenomena, and including all major instabilities. The model is formally derived from the fundamental Bogoliubov–Born–Green–Kirkwood–Yvon (BBGKY) (Bogoliubov, 1946) equations describing an N -body system of N_s charged particles of various species in an electromagnetic field $\mathbb{E}$ and $\mathbb{B}$, coupled with Maxwell's equations for the fields. Each individual particle with (vector) position $\mathbb{x}_n$ and velocity $\mathbb{v}_n$ is acted on by the sum of the Lorentz forces from all other particles of the same or other species

$$m_n \mathbb{x}_n = m_n \mathbb{v}_n = \sum_{k \neq n}^{N_s} q_k (\mathbb{E} + \mathbb{v}_k \times \mathbb{B}) \quad (3)$$

Note that the $\mathbb{x}_k, \mathbb{v}_k$ refer to the particles of various species s ; the species label is suppressed from here on to keep the notation simple. These equations are intractable since $N_s \sim 10^{20}$. A more tractable form is obtained by replacing the full particle distribution

function for N_s particles of various species, $F_s(\mathbf{x}_1, \mathbf{v}_1, \mathbf{x}_2, \mathbf{v}_2, \dots, \mathbf{x}_{N_s}, \mathbf{v}_{N_s})$, by a sequence of partial distribution functions, $f_s^{(n)}$, $n = 1, 2, \dots, N_s - 1$, obtained by integrating over the eliminated variables, so that,

$$f_s^{(n-1)}(\mathbf{x}_1, \mathbf{v}_1, \mathbf{x}_2, \mathbf{v}_2, \dots, \mathbf{x}_n, \mathbf{v}_n) \equiv \int \int f_s^{(n)}(\mathbf{x}_1, \mathbf{v}_1, \mathbf{x}_2, \mathbf{v}_2, \dots, \mathbf{x}_n, \mathbf{v}_n, \mathbf{x}_{n+1}, \mathbf{v}_{n+1}) d^3 \mathbf{x}_{n+1} d^3 \mathbf{v}_{n+1} \quad (4)$$

with $f_s^{(N_s-1)}(\mathbf{x}_1, \mathbf{v}_1, \mathbf{x}_2, \mathbf{v}_2, \dots, \mathbf{x}_{N_s}, \mathbf{v}_{N_s}) \equiv F_s(\mathbf{x}_1, \mathbf{v}_1, \mathbf{x}_2, \mathbf{v}_2, \dots, \mathbf{x}_{N_s}, \mathbf{v}_{N_s})$.

These are the so-called BBGKY equations. These contain similar information as $F_s(\mathbf{x}_1, \mathbf{v}_1, \mathbf{x}_2, \mathbf{v}_2, \dots, \mathbf{x}_{N_s}, \mathbf{v}_{N_s})$, namely the position and velocity of every particle of species s . The final so-called single particle distribution function is then,

$$f_s(\mathbf{x}, \mathbf{v}) \equiv f_s^{(0)}(\mathbf{x}_1, \mathbf{v}_1) = \int \int \int \dots F_s(\mathbf{x}_1, \mathbf{v}_1, \mathbf{x}_2, \mathbf{v}_2, \dots, \mathbf{x}_{N_s}, \mathbf{v}_{N_s}) \prod_{k=2}^{N_s} (d^3 \mathbf{x}_k d^3 \mathbf{v}_k) \quad (5)$$

Combining this with the equations of motion for the particles then yields the Boltzmann Equation for $f_s(\mathbf{x}, \mathbf{v})$ (Colonna, 2016):

$$\frac{\partial f_s}{\partial t} + \mathbf{v} \cdot \nabla_{\mathbf{x}} f_s + q_s/m_s [\mathbf{E}(\mathbf{x}, t) + \mathbf{v} \times \mathbf{B}(\mathbf{x}, t)] \cdot \nabla_{\mathbf{v}} f_s = C_s(f_a, f_\beta, f_\gamma, \dots, f_s, \dots, f_\zeta) \quad (6)$$

Here, $\nabla_{\mathbf{x}} f_s$ means $\nabla_{\mathbf{x}} f_s \equiv \left(\frac{\partial f_s}{\partial X} \quad \frac{\partial f_s}{\partial Y} \quad \frac{\partial f_s}{\partial Z} \right)^T$, and $\nabla_{\mathbf{v}} f_s \equiv \left(\frac{\partial f_s}{\partial V_x} \quad \frac{\partial f_s}{\partial V_y} \quad \frac{\partial f_s}{\partial V_z} \right)^T$ (the T represents the transpose). This coupled set of equations for various particle species ($\alpha, \beta, \dots, s, \dots, \zeta$) describes the evolution of the probability of finding a particle of species s at any given point $(\mathbf{x}, \mathbf{v})$ in phase space, at any given time due to electrodynamic and thermodynamic forces. The final term on the right side formally represents the effects of collisions between species, including those of the same species. For a collisionless plasma, $C_s = 0$, and the equation is called the Vlasov Equation (Colonna, 2016). Combined with Maxwell's Equations for the fields, the equations give a complete self-consistent account of the system of charged particles.

The MHD equations are obtained by taking velocity moments of the Boltzmann equations, for electrons and each ion species. For the velocity $\mathbf{v} \equiv (V_s^X \quad V_s^Y \quad V_s^Z)^T$, the velocity moments are defined by,

$$\langle V_s^\alpha V_s^\beta \dots V_s^\zeta \rangle_s \equiv \frac{\int V_s^\alpha V_s^\beta \dots V_s^\zeta f_s(\mathbf{x}, \mathbf{v}) d^3 \mathbf{v}}{\int f_s(\mathbf{x}, \mathbf{v}) d^3 \mathbf{v}} \quad (7)$$

for each of the coordinates, $\alpha, \beta, \dots, \zeta \in \{X, Y, Z\}$. We will use the notation V_s^α (with the s subscript) when designating the velocity moments. Without the subscript they refer to dummy integration variables. The first velocity moment (zeroth order) is just the number density defined as:

$$n_s(\mathbf{x}) = \langle n_s \rangle = \int f_s(\mathbf{x}, \mathbf{v}) d^3 \mathbf{v} \quad (8)$$

The second is the species single-fluid velocity, with three different moments corresponding to the three coordinate directions:

$$V_s^\alpha(\mathbf{x}) = \langle V_s^\alpha \rangle = \frac{\int V_s^\alpha f_s(\mathbf{x}, \mathbf{v}) d^3 \mathbf{v}}{\int f_s(\mathbf{x}, \mathbf{v}) d^3 \mathbf{v}} \quad (9)$$

The third describes the energy for each species defined as:

$$\frac{1}{2} n_s(\mathbf{x}) T_s^{\alpha, \beta}(\mathbf{x}) = \int \frac{1}{2} m_s (V_s^\alpha - V_s^\alpha(\mathbf{x})) (V_s^\beta - V_s^\beta(\mathbf{x})) f_s(\mathbf{x}, \mathbf{v}) d^3 \mathbf{v} \quad (10)$$

This is a symmetric tensor and the temperature for each species is then just found from

$$T_s(\mathbf{x}) = [T_s^{x,x}(\mathbf{x}) + T_s^{y,y}(\mathbf{x}) + T_s^{z,z}(\mathbf{x})]/3 \quad (11)$$

$$T_s^{\alpha, \alpha}(\mathbf{x}) = \langle m_s (V_s^\alpha - V_s^\alpha(\mathbf{x}))^2 \rangle_s \quad (12)$$

A pressure tensor for each species can also be defined. This is a symmetric tensor, $\mathbb{P}$, with components

$$P_s^{\alpha, \beta}(\mathbf{x}) \equiv n_s(\mathbf{x}) T_s^{\alpha, \beta}(\mathbf{x}) = \langle m_s (V_s^\alpha - V_s^\alpha(\mathbf{x})) (V_s^\beta - V_s^\beta(\mathbf{x})) \rangle_s \int f_s(\mathbf{x}, \mathbf{v}) d^3 \mathbf{v} \quad (13)$$

The tensor $\mathbb{T}$ and the temperature $T_s(\mathbf{x})$ are in units of energy so $T_s(\mathbf{x})$ is really k times the temperature in Kelvins, where k is the Boltzmann's constant.

The full set of moment equations coupled with Maxwell's equations provides a complete formal description of the system. The first two moments provide a set of equations that essentially describe mass and momentum conservation for each species. For a plasma of just two species, ions and electron, one can recombine these first moment equations into a single-fluid form by replacing each pair of species equations by their weighted sum and difference to eliminate $\mathbf{v}_i$ and $\mathbf{v}_e$ by the fluid velocity $\mathbf{v}$, which is the common velocity of the two species,

$$\mathbf{v} \equiv (m_i n_i \mathbf{v}_i + m_e n_e \mathbf{v}_e) / (m_i n_i + m_e n_e) \quad (14)$$

and the velocity difference represented by the current density,

$$\mathbf{j} = (q_i n_i \mathbf{v}_i + q_e n_e \mathbf{v}_e) \quad (15)$$

The total mass density is,

$$\rho_m = (m_i n_i + m_e n_e) \quad (16)$$

and the total (tensor) pressure is,

$$\mathbb{P} = \mathbb{P}_i + \mathbb{P}_e \quad (17)$$

Then, the zeroth-order moment becomes the particle conservation equation;

$$\frac{\partial \rho_m}{\partial t} + \nabla \cdot (\rho_m \mathbf{v}) = 0 \quad (18)$$

familiar from conventional fluid theory. The first order moment equations become a simple force balance momentum conservation equation, which can be written as the vector equation,

$$\rho_m \left(\frac{\partial \mathbf{v}}{\partial t} + (\mathbf{v} \cdot \nabla) \mathbf{v} \right) + \nabla \cdot \mathbb{P} - (\mathbf{j} \times \mathbb{B}) = 0 \quad (19)$$

The first two terms on the left represent the inertia. The other two terms are the pressure force and the electromagnetic force. In equilibrium, with $\mathbf{v} = 0$, the pressure and electromagnetic forces are balanced. While at this point the model is two-fluid, these equations are in a single-fluid form.

In general, higher velocity moments from higher order equations are required in each new equation, and at second order, the situation becomes significantly more complicated. The individual electron and ion terms cannot be completely eliminated without additional approximations. At second order, one obtains two equations. One is effectively a generalized Ohm's Law

$$\mathbb{E} + \mathbf{v} \times \mathbb{B} = \eta \mathbf{j} + \mathbb{R} \quad (20)$$

$$\mathbb{R} = \left(\frac{1}{q_e n_e} \right) \nabla \cdot \mathbb{P}_i + \left(\frac{\Delta n_e}{n_e} \right) \mathbb{E} + \left(\frac{m_i}{q_i} \right) \left(\frac{\partial \mathbf{v}}{\partial t} + (\mathbf{v} \cdot \nabla) \mathbf{v} \right) + \left(\frac{m_e}{q_e} \right) \left(\frac{\partial (\mathbf{v}_i - \mathbf{v}_e)}{\partial t} + (\mathbf{v} \cdot \nabla) (\mathbf{v}_i - \mathbf{v}_e) \right) \quad (21)$$

$\mathbb{R}$ contains the remaining unbalanced two-fluid terms, including the Hall effect. Most are small and can normally be neglected. The other is a tensor equation for the evolution of the pressure tensor, $\mathbb{P}$ given below.

Truncating the separate ion and electron equations at this order, then one obtains the full set of MHD equations:

$$\frac{\partial \rho_m}{\partial t} + \nabla \cdot (\rho_m \mathbf{v}) = 0 \quad (22)$$

$$\frac{\partial \sigma}{\partial t} + \nabla \cdot \mathbf{j} = 0 \quad (23)$$

$$\frac{\partial \mathbb{B}}{\partial t} + \nabla \times (\mathbb{E}) = 0 \quad (24)$$

$$-\frac{1}{c^2} \frac{\partial \mathbb{E}}{\partial t} + \nabla \times (\mathbb{B}) = \mu_0 \mathbf{j} \quad (25)$$

$$\nabla \cdot \mathbb{B} = 0 \quad (26)$$

$$\nabla \cdot \mathbb{E} = \varepsilon_0^{-1} \sigma \quad (27)$$

$$\rho_m \left(\frac{\partial \mathbf{v}}{\partial t} + (\mathbf{v} \cdot \nabla) \mathbf{v} \right) = \sigma \mathbb{E} + \mathbf{j} \times \mathbb{B} - \nabla \cdot \mathbb{P} \quad (28)$$

$$\mathbb{E} + \mathbf{v} \times \mathbb{B} = \eta \mathbf{j} + \mathbb{R} \quad (29)$$

$$\frac{\partial}{\partial t} \mathbb{P} + (\mathbf{v} \cdot \nabla) \mathbb{P} = \nabla \cdot \mathbb{Q} + \mathbb{S} \quad (30)$$

with $\mathbb{Q}$ representing a (generalized) stress-energy tensor flux and $\mathbb{S}$ a generalized dissipation. In this general form, MHD is a full two-fluid theory but it is not closed. The generalized stress-energy tensor and dissipation in Eq. (30) are obtained formally from the higher order truncated terms, but most closures involve specifying them independently instead, along with several simplifying assumptions for $\mathbb{R}$.

Various closure schemes are possible and these lead to different versions of MHD. The most important approximation is the Darwin Approximation;

$$\frac{1}{c^2} \frac{\partial \mathbb{E}}{\partial t} \ll \nabla \times \mathbb{B} \sim \mu_0 \mathbf{j} \quad (31)$$

which essentially eliminates propagating purely electromagnetic waves from the system. This is almost always a good approximation and is consistent with the ignoring of relativistic dynamics. Of the various simplified MHD models, the Darwin approximation is the most valid and is almost always assumed. With the Darwin approximation, the remaining equations are usually referred to as the “Two-Fluid MHD Equations”, or sometimes, with an appropriate closure, as the Extended MHD Equations.

The remaining common assumptions such as no resistivity $\eta = 0$, isotropic pressure $p_\perp = p_\parallel$, or $T_e = T_i$ are variously valid under different circumstances. These are usually based on approximations from fluid dynamics and thermodynamics. For example, relations between $\mathbb{P}$ and the mass density for each species, or an assumption for off-diagonal pressure (stress) terms in $\mathbb{P}$ are often used. Separate leftover electron terms of the full two-fluid model (Hall and ∇p_e terms) are commonly dropped as small, leaving simpler single-fluid equations. These are usually referred to as “the MHD Equations” or the “Single-Fluid MHD” equations. Assuming a scalar pressure, one obtains a much simpler system with relatively simple closure options. An alternative is the Chew-Goldberger-Low CGL model (Hunana et al., 2019) which allows for pressure anisotropy $P_\perp \neq P_\parallel$. Finally, if the resistivity is also ignored, $\eta \mathbf{j} \ll \mathbf{v} \times \mathbb{B} \sim \mathbb{E}$, then $\mathbb{E} + \mathbf{v} \times \mathbb{B} \sim 0$ and

$$\frac{\partial \mathbb{B}}{\partial t} = \nabla \times (\mathbf{v} \times \mathbb{B}) \quad (32)$$

implies that the fluid moves with the fields; there is no slippage between them. This condition is called the frozen flux theorem and the resulting MHD model is commonly referred to as ‘ideal MHD’. In addition, since the equations are physically intuitive, representing various conservation laws, ad-hoc phenomenological terms can be included as closures to model effects that have been eliminated, for example, fast-particle drive effects, or neoclassical effects.

Toroidal equilibrium

The shape of the equilibrium magnetic surfaces plays an important role in determining its stability and performance. The ideal MHD momentum balance Eq. (19) provides a comprehensive physics constraint to compute and optimize the required external poloidal magnetic-field coil set and its currents necessary to produce a target plasma boundary shape (defined as the largest closed magnetic surface enclosed by the surrounding limiter and vacuum vessel), and the diversion of the external magnetic-field lines to direct the particle and heat flux to the divertor collecting plate.

At equilibrium and in the absence of plasma flow and pressure anisotropy, the MHD equilibrium Eq. (19) and Ampere’s law Eq. (25) become

$$\nabla P = \mathbf{j} \times \mathbb{B} \quad (33)$$

$$\nabla \times \mathbb{B} = \mu_0 \mathbf{j} \quad (34)$$

It follows from Eqs. (33) and (34) that the confining magnetic force is in the direction perpendicular to the magnetic field lines. At equilibrium, the pressure must be constant on a magnetic flux surface $P = P(\psi)$ and the plasma current density $\mathbf{j}$ flows along the surface.

In a toroidal axisymmetric device such as tokamak, a set of 2D nested magnetic surfaces is formed using a toroidal-field and a poloidal-field coil set. A dedicated Ohmic-coil set is also sometimes employed to drive the toroidal current flowing within the plasma to ease control as in the DIII-D (Luxon, 2002) and ASDEX Upgrade (Gruber and ASDEX Upgrade Project Group, 1986) tokamaks, rather than combining the functionality into a single set of poloidal-field coils for both shaping and driving Ohmic current as in more recent superconducting long-pulse tokamak devices such as EAST (Wan et al., 2006), KSTAR (Lee et al., 1999), JT-60SA (Kamada et al., 2011), and ITER (<https://www.iter.org/>) (ITER Physics Basis Editors, 1999a). In the presence of toroidal asymmetry, magnetic islands and stochastic regions can form, degrading the plasma confinement and making it more susceptible to MHD instabilities.

Axisymmetric 2D Grad-Shafranov equilibrium

In an axisymmetric toroidal device such as a tokamak, Eqs. (33) and (34) can be combined to yield a 2D elliptic equilibrium equation, the Grad-Shafranov (GS) equation (Grad and Rubin, 1956; Shafranov, 1958), by considering force balance in the direction perpendicular to the magnetic field $\mathbb{B}$ and using a cylindrical (R, φ, Z) coordinate system centered at the symmetric axis and dropping the toroidal φ -dependent terms

$$\Delta^* \psi = -\mu_0 R j_\varphi(R, \psi) \quad (35)$$

$$j_\varphi = RP'(\psi) + \frac{\mu_0 FF'(\psi)}{4\pi^2 R} \quad (36)$$

Here, the operator $\Delta^* = R^2 \nabla \cdot (\nabla/R^2)$. The magnetic field can be conveniently represented as

$$\mathbb{B} = \frac{\mu_0 F(\psi)}{2\pi} \nabla \varphi + \nabla \psi \times \nabla \varphi \quad (37)$$

The first term in Eq. (37) represents the toroidal component of $\mathbb{B}$, whereas the second term represents the poloidal component.

The GS equilibrium, Eq. (35), is based on the cylindrical (R, φ, Z) coordinate system. An alternative coordinate system is the inverse magnetic-flux coordinate system (ρ, θ, φ) using a flux-surface label $\rho(\psi)$ and a poloidal angle θ as independent variables (Boozer, 2005). The GS Eq. (35) can then be transformed to become the inverse GS equation for $R(\rho, \theta)$ or $Z(\rho, \theta)$ (Lao et al., 1981). Since the pressure $P(\psi)$ is constant on a flux surface, the inverse magnetic-flux coordinate system (ρ, θ, φ) provides a particularly convenient coordinate system to study plasma equilibrium, transport, and stability physics. In the case of a diverted plasma, the use of the magnetic-flux coordinate system is restricted to the nested magnetic-surface region within the plasma to avoid the singularity at the separatrix surface that appears in the Jacobian for transformation from the (R, φ, Z) to the (ρ, θ, φ) coordinate system.

Eq. (35) represents a differential form of the GS equation. It acts as a constraint linking the derivatives of ψ to the current sources. Given two stream functions describing the toroidal current density source such as $P(\psi)$ and $F(\psi)$ as shown in Eq. (36) and appropriate boundary conditions, Eq. (35) can be solved for the poloidal flux function ψ . Two other forms of the GS Eq. (35) useful for finding equilibrium solutions are the integral and the variational Lagrangian forms (Grad and Rubin, 1956; Lao et al., 1981, 1985a, 2005).

$$\psi(\mathbf{x}) = \sum_j G_\psi(\mathbf{x}, \mathbf{x}_{ej}) I_{ej} + \int_{\Omega_B} G_\psi(\mathbf{x}, \mathbf{x}') j_\varphi[\mathbf{x}', \psi(\mathbf{x}')] dR' dZ' \quad (38)$$

$$W = \int_{\Omega_B} \left(\frac{B_p^2}{2\mu_0} - \frac{B_T^2}{2\mu_0} - P \right) d\Omega \quad (39)$$

Here $G_\psi(\mathbf{x}, \mathbf{x}')$ is the Green induction function relating $\psi(\mathbf{x})$ to the current source at $\mathbf{x}'$. The variational Lagrangian form Eq. (39) provides a systematic means to transform the solution $\psi(R, Z)$ of the 2D GS equilibrium Eq. (35) into a series of 1D moment equations for the Fourier amplitudes of $R(\rho, \theta)$ or $Z(\rho, \theta)$ (Lao et al., 1981). The 2D variational moment approach has been successfully generalized to provide a robust and efficient method to find equilibrium solutions in 3D toroidal geometry with nested magnetic surfaces (Bhattacharjee et al., 1983; Hirshman and Whitson, 1983; Lao et al., 1985c).

Analytical solutions exist when the current source Eq. (36) has simple forms (Solov'ev, 1968; Srinivasan et al., 2010) or when the magnetic surface has simple circular geometry (Lao et al., 1981). In general, the GS Eq. (35) or its integral or variation form Eqs. (38) or (39) must be solved numerically (Johnson et al., 1979; Takeda and Tokuda, 1991). Many tools are available to numerically search and compute the equilibrium solutions given two stream functions describing the toroidal current density source and appropriate boundary conditions (Lao, 1984; Lao et al., 1985a, 2005; Haney et al., 1995; Ivanov et al., 2009). One class of applications is to find a solution that best matches a specified target plasma boundary given the two stream functions, a set of external poloidal-field coils, and a surrounding limiter. An example of a DIII-D equilibrium computed with the EFIT code (Lao et al., 1985a, 2005) is given in Fig. 2. Another class of applications is to find a solution that best matches a specified target plasma boundary but with the

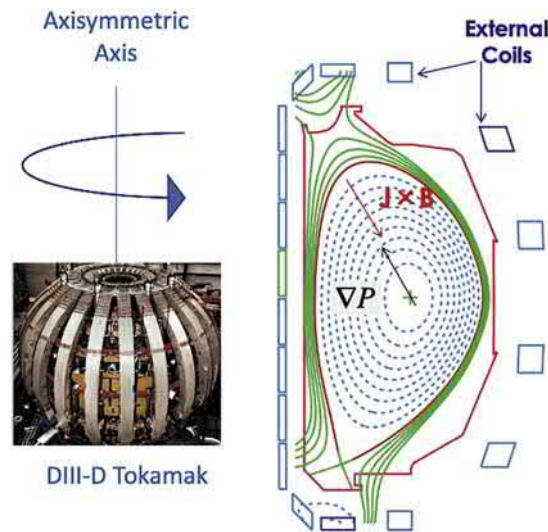


Fig. 2 Poloidal cross-section of a DIII-D tokamak lower single-null equilibrium computed using the EFIT equilibrium code. Also shown is a photo of the DIII-D tokamak with its toroidal- and poloidal-field coils.

two stream functions self-consistently computed based on the plasma transport properties (Grad and Hogan, 1970; Hirshman and Jardin, 1979; Meneghini et al., 2016).

Equilibrium reconstruction

Reconstruction of experimental MHD equilibria is fundamental to tokamak research and operation and is an important part of fusion data analysis and plasma control. Equilibrium reconstruction provides essential magnetic geometry and current and pressure profile information necessary to support tokamak operation and data analysis, and has contributed to several major physics discoveries, such as the experimental validation of theoretically predicted β stability limits (Strait et al., 1994) and the negative central-shear operating regime (Levinton et al., 1995; Strait et al., 1995).

A particularly important application of the GS Eq. (35) is to reconstruct the experimental plasma equilibrium from various available measurements such as external magnetic and internal current and kinetic profile diagnostics (Lao et al., 1985a, 1990, 2005). The amount of plasma information that can be reconstructed increases with the availability of the measurements. External magnetic measurements alone can only yield the plasma boundary shape and global plasma pressure and current profile parameters, such as poloidal β , $\beta_P = \int_0^{\Omega_B} P d\Omega / (\Omega_B B_{PA}^2 / 2\mu_0)$, and internal inductance of the plasma current density profile $\mathcal{L}_i = \int_0^{\Omega_B} B_P^2 d\Omega / (\Omega_B B_{PA}^2)$, and $B_{PA} = \oint \psi_B B_P d\ell_p / \oint \psi_B d\ell_p$ is the average poloidal magnetic field along the plasma boundary surface (Lao et al., 1985b; Braams, 1991). For critical applications such as analysis and control of MHD instabilities that require detailed pressure and current profile information, accurate full equilibrium reconstruction with kinetic and internal current profiles in addition to external magnetic measurements is required, as well as high spatial-resolution and tightly converged equilibrium solutions (Lao et al., 1990; Ren et al., 2011). Plasma control also imposes a stringent requirement on the equilibrium reconstruction computational speed. Various efficient numerical algorithms and high-performance computation technology have been developed and employed to reconstruct tokamak experimental equilibria in real time to provide the information necessary for plasma control (Ferron et al., 1998; Moret et al., 2015; Ramp et al., 2017; Huang et al., 2020).

MHD stability

Introduction and overview

In principle, the full nonlinear (Extended) MHD model should be reliable under the conditions that the plasma be quasi-neutral and that the length and time scales be long: specifically that the length scales be of the order of the plasma minor radius, and much larger than microscopic scales like the Larmor radius, and that the time scales are much slower than Alfvénic times.

Experimental diagnostic capabilities have been developed to the point where detailed predictions from MHD theory can be productively tested. As discussed in the previous Equilibrium Reconstruction Section, the key to this development has been the progress in reconstructing equilibria obeying MHD force balance and consistent with experimentally measured kinetic profiles (Lao et al., 1990). The linear MHD stability predictions using high quality discharge full equilibrium reconstructions have been thoroughly tested against observations for the principal limiting phenomena and MHD generally predicts ideal current and pressure limits well (Turnbull et al., 2002; Turnbull et al., 2005). In particular, global pressure driven instabilities are predicted to be unstable when the plasma β_T exceeds a value typically of a few percent. β_T measures the amount of thermal energy the plasma can hold for a given magnetic field strength.

A series of numerical ideal MHD stability calculations in the mid 1980s discovered that the limiting β_T obeys a scaling known as the Troyon limit, given originally as

$$\beta_T \leq \beta_{TROY} \equiv C_T (\mu_0 I_P) / (a B_T) \quad (40)$$

where $C_T = 2.8$ if I_P is in Amperes and B_T is in Tesla (Troyon et al., 1984). This was immediately confirmed by experiments on a number of tokamaks (Stambaugh et al., 1984) and is commonly referred to as the Troyon beta limit. (This has been a point of confusion; the difference between C_T and β_N^{max} has erroneously been claimed to be a discrepancy.)

$$\beta_T \leq \beta_N^{max} I_P(\text{MA}) / (a B_T) \quad (41)$$

with $\beta_N^{max} = 3.5$. In subsequent years, the Troyon limit scaling has been modified to account for the major profile and shaping effects. The most recent version replaces the constant $\beta_N^{max} = 3.5$ by

$$\beta_N^{max} = k_\beta \mathcal{L}_i \quad (42)$$

where $\mathcal{L}_i$ is the internal inductance of the current density profile and k_β varies from 2.5 for conventional elongated, single null, low-triangularity plasmas to $k_\beta = 4$ for high-triangularity double-null plasmas. In spherical tokamaks, the value can be much higher (Sabbagh et al., 2006). The $\mathcal{L}_i$ dependence is consistent with results from an analytical ideal ballooning β -limit study (Lao et al., 1992), and a numerical study of the $n = 1$ ideal kink β -limit (Howl et al., 1992).

Ideal linear MHD stability and energy principle

In the ideal case, the equations have special properties that lead to efficient numerical calculation schemes, the most important of which is the ideal MHD energy principle for linear stability against small departures from equilibrium. Assuming an ideal equilibrium without flows $\mathbf{v}_0 = 0$ and a scalar pressure, the dynamic quantities can be linearized in the small perturbation: $P(\mathbf{x}, t) = P_0(\mathbf{x}) + P_1(\mathbf{x}, t)$; $\mathbf{j}(\mathbf{x}, t) = \mathbf{j}_0(\mathbf{x}) + \mathbf{j}_1(\mathbf{x}, t)$; and $\mathbf{B}(\mathbf{x}, t) = \mathbf{B}_0(\mathbf{x}) + \mathbf{B}_1(\mathbf{x}, t)$, with each perturbed quantity assumed much smaller

than the equilibrium term, yielding an equation for the equilibrium and an equation describing the linear first-order force balance

$$\nabla P_0 = \mathbf{j}_0 \times \mathbb{B}_0 = \left(\frac{1}{\mu_0} \right) (\nabla \times \mathbb{B}_0) \times \mathbb{B}_0 \quad (43)$$

$$\rho_0 \left(\frac{\partial \mathbf{v}_1}{\partial t} \right) + \nabla P_1 = \mathbf{j}_1 \times \mathbb{B}_0 + \mathbf{j}_0 \times \mathbb{B}_1 \quad (44)$$

From here, in order to simplify the subsequent notation, we write $\mathbb{B} = \mathbb{B}_0$, and $\delta \mathbb{B}_p = \mathbb{B}_1$ and similarly for $\mathbf{j} = \mathbf{j}_0$ and $\delta \mathbf{j}_p = \mathbf{j}_1$. The ideal MHD frozen flux relation becomes a relation between the perturbed field $\delta \mathbb{B}_p$ and fluid displacement ξ ;

$$\delta \mathbb{B}_p = \nabla \times (\xi \times \mathbb{B}) \quad (45)$$

$\xi \times \mathbb{B}$ is the perturbed vector potential. After some algebra, one finds an equation of motion for the small displacement $\xi(\mathbf{x}, t)$ and perturbed velocity $\mathbf{v}_1(\mathbf{x}, t) \equiv \xi(\mathbf{x}, t)$;

$$\rho_0 \ddot{\xi}(\mathbf{x}, t) = \mathcal{F}(\xi(\mathbf{x}, t)) \quad (46)$$

with $\mathcal{F}(\xi)$ a linear operator on ξ : $\mathcal{F}(\xi) \equiv \mathbb{F}(\mathbf{x}) \cdot \xi$;

$$\mathcal{F}(\xi) \equiv \mathbb{F}(\mathbf{x}) \cdot \xi \equiv \left(\frac{1}{\mu_0} \right) [((\nabla \times \delta \mathbb{B}_p) \times \mathbb{B}) + ((\nabla \times \mathbb{B}) \times \delta \mathbb{B}_p)] + \nabla(\xi \cdot \nabla P + \Gamma P \nabla \cdot \xi) \quad (47)$$

$\mathbb{F}(\mathbf{x})$ is known as the (linearized) force operator and Γ is the adiabatic index. The solutions to $\xi(\mathbf{x}, t)$ must be temporally exponential, $\xi(\mathbf{x}, t) \equiv \xi(\mathbf{x}) e^{i\omega t}$, leading to an eigenvalue equation for the spatial variation of the displacement, $\xi(\mathbf{x})$ and its eigenvalue ω^2 ;

$$\rho_0^{-1} \mathbb{F}(\mathbf{x}) \cdot \xi(\mathbf{x}) = -\omega^2 \xi(\mathbf{x}) \quad (48)$$

Note that $\xi^\dagger \cdot \mathbb{F} \cdot \xi = \mathcal{F}(\xi) \cdot \xi^\dagger$ represents the work done by the infinitesimal force $\mathcal{F}(\xi)$ against the infinitesimal displacement ξ .

Boundary conditions need to be applied by specifying the displacement of the plasma boundary. For a plasma surrounded by a vacuum region and possibly a conducting wall outside, this is not explicitly known. Boundary conditions can be set, however, by solving simultaneously for $\delta \mathbb{B}_p$ and $\xi(\mathbf{x})$ and the ideal frozen flux relation linking them, and solving for the perturbed vacuum field, $\delta \mathbb{B}_v$, in the annulus between the plasma and the wall.

$$\nabla \times \delta \mathbb{B}_v = \nabla \cdot \mathbb{B}_v = 0 \quad (49)$$

which is most easily solved in terms of a magnetic potential, $\delta \mathbb{B}_v = \nabla \Phi$;

$$\nabla^2 \Phi = 0 \quad (50)$$

An appropriate magnetic boundary condition for a conducting wall, or for a wall at infinity can then be applied;

$$\hat{\mathbf{m}} \cdot \nabla \Phi|_w = 0 \quad (51)$$

The perturbed fields must then be matched across the plasma boundary. This condition is complicated since they must be matched across the perturbed boundary;

$$\hat{\mathbf{m}} \cdot \delta \mathbb{B}_p|_{pv} = \hat{\mathbf{m}} \cdot \nabla \Phi|_{pv} \quad (52)$$

The matrix $\mathbb{F}(\mathbf{x})$ is self-adjoint (or Hermitian); $\xi^\dagger \cdot \mathbb{F} \cdot \xi = \xi \cdot \mathbb{F} \cdot \xi^\dagger$. The Hermitian property provides a number of computational advantages (Bernstein et al., 1957). Notably, the eigenvalues $-\omega^2$ are purely real. Thus, the solutions are either purely growing or damped, or are purely oscillatory. The Hermitian property also means one can conveniently reformulate the dynamic problem as a variational problem. Pre-multiplying the equation of motion in the plasma by the adjoint, $\xi^\dagger$, and integrating over all space, one obtains a variational principle (Bernstein et al., 1957). Application of physically relevant boundary conditions in the variational formulation is non-trivial. This is solved by deriving an extended form of the energy principle in which conditions across the plasma vacuum interface, Eq. (52) above, is automatically satisfied by the variational solutions. Formulated as a statement of conservation of energy, the extended form of the energy principle is;

$$\delta W - \omega^2 \delta K = 0 \quad (53)$$

$$\delta W(\xi, \xi^\dagger, \delta \mathbb{B}_v, \delta \mathbb{B}_v^\dagger) \equiv \delta W_p(\xi, \xi^\dagger) + \delta W_v(\delta \mathbb{B}_v, \delta \mathbb{B}_v^\dagger) \quad (54)$$

$$\delta K(\xi, \xi^\dagger) \equiv \frac{1}{2} \int_p \rho_0(\mathbf{x}) (\xi^\dagger \cdot \xi) d^3 \mathbf{x} \quad (55)$$

$$\delta W_p(\xi, \xi^\dagger) \equiv \frac{1}{2\mu_0} \int_p (\xi^\dagger \cdot \mathbb{F}(\mathbf{x}) \cdot \xi) d^3 \mathbf{x} \quad (56)$$

$$\delta W_v(\delta \mathbb{B}_v, \delta \mathbb{B}_v^\dagger) \equiv \frac{1}{2\mu_0} \int_p (\delta \mathbb{B}_v^\dagger \cdot \delta \mathbb{B}_v) d^3\mathbb{x} \quad (57)$$

δW_v is the vacuum energy contribution from perturbing the plasma boundary. The integrals δK and δW_p are over the plasma volume, and δW_v is over the annular vacuum region. δW represents the total potential energy and δK is the kinetic energy of the perturbing motion. Then δW_p is the plasma potential energy. The energy principle states that if a physically permissible displacement ξ can be found such that $\delta W(\xi, \xi^\dagger) > 0$, then the plasma is unstable. The energy principle can therefore be used in a trial function approach. Finding a physically permissible trial function, ξ_0 , for which $\delta W(\xi_0, \xi_0^\dagger) > 0$, thus guarantees instability. Note, however, that the trial function is not necessarily the instability that would appear. The variational principle only implies that this or another more unstable mode will be destabilized.

Substituting for $\mathbb{F}(\mathbb{x})$, the Extended MHD Principle potential energy becomes after some algebra:

$$\delta W_p = \frac{1}{2} \iiint \left[\frac{\delta \mathbb{B}_p^2}{\mu_0} - \xi_\perp^* \cdot (\hat{\mathbb{j}} \times \delta \mathbb{B}_p) + (\xi_\perp \cdot \nabla P) \nabla \cdot \xi_\perp^* + \Gamma P |\nabla \cdot \xi|^2 \right] d^3\mathbb{x} \quad (58)$$

The integral in δW_p is over the plasma and the condition applied is simply that $\delta \mathbb{B}_p = \delta \mathbb{B}_v$ at the plasma vacuum interface. The terms in δW_p can be interpreted physically. The first term in δW_p represents the energy required to bend the field lines. It is always stabilizing—it takes energy input to create this; it is the energy in the perturbed field. The second term, $\xi_\perp^* \cdot (\hat{\mathbb{j}} \times \delta \mathbb{B}_p)$ is the destabilization from the current density. The third term is the pressure drive through $\xi_\perp \cdot \nabla P$. The final term represents the energy associated with plasma compression. δW_p can be rewritten in the alternative form;

$$\delta W_p = \frac{1}{2} \iiint \left[\delta \mathbb{B}_\perp^2 / \mu_0 + \left(\frac{\mathbb{B}^2}{\mu_0} \right) |\nabla \cdot \xi_\perp + 2\xi_\perp \cdot \kappa|^2 - \left| \frac{\hat{\mathbb{j}}_\parallel}{\mathbb{B}} \right| (\xi_\perp^* \times \mathbb{B}) \cdot \delta \mathbb{B}_\perp - 2(\xi_\perp \cdot \nabla P)(\xi_\perp^* \cdot \kappa) + \Gamma P |\nabla \cdot \xi|^2 \right] d^3\mathbb{x} \quad (59)$$

This shows more clearly the physical meaning of the terms. The first represents the energy required to bend field lines. The second corresponds to the energy involved in compression of the field. The third and fourth terms are respectively the destabilization from current driven modes, proportional to $|\hat{\mathbb{j}}_\parallel / \mathbb{B}|$ and ∇P , respectively. The last term, again is the energy from the fluid compression.

The MHD model works extremely well in most situations. In particular, the plasma cross section shape is easily included, at least in numerical calculations as boundary conditions, thus fully capturing the most basic zeroth order effect. MHD is now recognized as an indispensable guide to any design efforts.

Pressure effects

Plasma equilibrium pressure is mostly destabilizing for MHD modes. Associated with the destabilization is the free energy contained in the pressure gradient. This is evident from the ideal MHD energy principle, where the term $-(\xi_\perp \cdot \nabla P)(\xi_\perp^* \cdot \kappa)$ from Eq. (59), integrated over the plasma volume, can be negative and thus provides destabilization. This is the drive term for the ballooning and interchange instabilities. The interchange instability involves a Rayleigh-Taylor interchange of two neighboring field lines. The ballooning mode is then a local interchange of a section of the field line in a sheared field; essentially the field line under higher pressure ‘balloons’ through the lower pressure field lines where the local shear is weak. The pressure gradient triggers MHD instabilities when it exceeds certain critical values. Below are important examples. The pressure driven ideal external kink mode (EK) becomes unstable when the normalized beta exceeds the Troyon limit (Troyon et al., 1984). Being a global instability, the EK mode is not sensitive to the local pressure gradient, but is still driven by the “global” pressure gradient which is often measured by the so-called pressure peaking factor. The ideal IK mode becomes unstable when the plasma pressure (often measured in poloidal beta value β_p) exceeds the Bussac limit (Bussac et al., 1975). The infernal mode becomes unstable when β_p exceeds a critical value linearly scaling with Δq , which measures proximity of the global or local q_{min} value to a rational number. The neoclassical tearing mode (NTM), which is generally a non-linear MHD mode, requires a seed island of finite size which scales with β_p . Besides these direct drives, the equilibrium plasma pressure gradient also induces bootstrap current which in turn can drive MHD instabilities. Bootstrap current is a toroidal current driven by non-ideal effects due to the temperature and density gradients. It arises from trapped particle effects.

The plasma equilibrium pressure gradient can also be stabilizing for certain MHD modes. The most prominent example is the second ballooning stability regime, which arises from the large Shafranov-shift induced field-line compressing at the low-field side of the plasma, where the ballooning mode is typically located due to unfavorable magnetic curvature. In a tokamak plasma, the interchange index D_R , which scales with the local equilibrium pressure gradient at the mode rational surface, appears also in the tearing mode stability criterion (Glasser et al., 1975). This can be negative, as a result of the average curvature effect, resulting in stabilization of the tearing mode.

Current effects

Plasma equilibrium current is often a major driving force for MHD instabilities in a tokamak plasma. Being a large quantity in a tokamak, the plasma current can easily provide free energy if the current density profile is not well optimized. (We note that large plasma current offers good energy confinement in tokamak plasmas.). In terms of the MHD energy principle, the current-drive term is associated with the plasma volume integral of $-\frac{1}{2} |\hat{\mathbb{j}}_\parallel / \mathbb{B}| (\xi_\perp^* \times \mathbb{B}) \cdot \delta \mathbb{B}_\perp$ term from Eq. (59), i.e., the primary drive is due to the equilibrium parallel current $j_\parallel$. In MHD theory, the current drive is often manifested in the value of safety factor q , which is roughly

inversely proportional to the toroidal current integrated over a plasma volume enclosed by the given magnetic flux surface, as shown in Eq. (1). Therefore, many current driven MHD instabilities are controlled by the safety factor q . Below is a list of important MHD modes that are at least partly driven by the plasma current.

The most important current driven MHD mode is the $n = 1$ ideal EK, which becomes unstable when the edge safety factor drops below 2 in a tokamak plasma. The edge safety factor is measured in terms of the edge q_s for a limiter plasma, and is often measured in terms of q_{95} (q at normalized $\psi = 0.95$) for a divertor plasma. This results in the ideal EK instability which usually leads to plasma disruption. It is difficult to overcome this stability limit but partial success has been achieved in recent experiments via active control of this instability (Hanson et al., 2014; Piovesan et al., 2014). A less severe, but often observed current driven instability is the IK mode, which can be unstable when the core safety factor q drops below 1. This is often manifested as the sawtooth instability in experiments. At higher values of edge q close to a low order rational number, the edge localized kink-peeling mode can be unstable. Similarly, if the safety factor profile is non-monotonic along the plasma minor radius and q_{min} (the minimum q value) is close to a low order rational number, an unstable infernal mode can occur. (More precisely, a finite pressure gradient is also required to drive the instability. The infernal mode can therefore be regarded as a MHD instability driven by both plasma current and pressure.) Finally, the plasma current density profile, alone or in combination with the plasma pressure, determines the tearing-mode index Δ' (Glasser et al., 1975) from the ideal region, which directly affects the tearing mode instability. The plasma current density profile is also directly related to the magnetic shear. A strong (positive) magnetic shear is generally stabilizing for most of MHD modes. Negative magnetic shear which occurs in plasmas with non-monotonic q -profiles can stabilize the tearing mode.

Resistivity effects

Many ideal MHD instabilities have resistive counterparts. Typically, the most interesting situation is that the ideal branch is stable within a certain parameter range, but taking into account the finite plasma resistivity qualitatively changes the picture by rendering the mode unstable, resulting in a resistive instability. Prominent examples are the tearing mode, the resistive interchange, the resistive internal kink, and the resistive external kink mode. Of course, the plasma resistivity also modifies (mostly enhances) the growth rate of an unstable ideal counterpart and the degree depends on the corresponding ideal instability. For instance, the plasma resistivity typically has very minor effect on an ideal external kink mode which grows at the Alfvénic time scale.

Within the MHD theory, the key physics associated with the plasma resistivity is that the ideal frozen-flux constraint is not valid. The plasma displacement and the magnetic field perturbation can evolve separately. As a consequence, the magnetic topology is allowed to change near a mode rational surface, resulting in magnetic islands, thus creating additional freedom for the plasma motion and for the perturbation to grow. This is the fundamental mechanism for the destabilization of the resistive instability when the ideal counterpart is stable.

On the other hand, there are also cases where the plasma resistivity plays a stabilizing role by dissipating free MHD energy via the resistive layer. One example is the fishbone instability, which is stabilized by finite plasma resistivity (Biglari and Chen, 1986; Wu et al., 2018). Another example is the resistive-plasma resistive-wall mode (RPRWM), where the global ideal MHD instability RWM couples to the tearing mode localized near a rational surface. The favorable average curvature stabilization of the tearing component within the resistive layer helps to stabilize the RPRWM (He et al., 2014). This mechanism applies when the coupling between the ideal and the resistive components is strong.

The TM is probably the most studied resistive instability (Glasser et al., 1975; Hegna and Callen, 1994). Both linear and non-linear theories are well developed. Several approaches have been established to study resistive instabilities, including the asymptotic matching, typically employed in analytic theory but also be useful in numerical codes.

Effects of plasma toroidal flow

Equilibrium toroidal plasma flow, as well as flow shear can affect MHD instabilities. (Plasma poloidal flow can in principle also affect MHD instabilities such as the RWM, but this is often not a significant concern, because poloidal flow is often slow in a tokamak plasma due to neoclassical damping.) Parallel plasma flow (along magnetic field lines) typically does not have significant effect on the MHD instability (Xia et al., 2019). The stabilization mechanism varies depending on the type of MHD mode. First, for the effect purely from the flow amplitude, imagine a flow that is uniform along the plasma minor radius. For many localized MHD modes, this uniform flow merely introduces a Doppler shift to the mode frequency (a change of reference from the laboratory frame to the rotating plasma frame), without modifying the mode growth rate. This may, however, be different for more global (in terms of plasma displacement) MHD modes. Two important examples are the RWM and the IK modes. The RWM does not rotate with the plasma and therefore is subject to continuum wave damping due to plasma flow. The IK is subject to a gyroscopic stabilization when the Mach number reaches a value comparable to the inverse aspect ratio.

On the other hand, flow shear typically destabilizes macroscopic MHD modes, unlike microscopic instabilities for which flow shear often plays a stabilizing role. The fundamental destabilization physics originates from the Kelvin-Helmholtz mechanism, where two adjacent fluid elements flowing at different velocities develop an unstable motion that eventually leads to turbulent flow. Flow shear destabilization has been found for many MHD modes, such as the RWM and the IK.

On the other hand, flow shear may help to stabilize certain MHD modes. One example is the observed 3/2 NTM island reduction from plasma flow in DIII-D experiments (La Haye and Buttery, 2009). Reduction of the tearing instability index by plasma flow shear was proposed as an explanation, and experimentally, the mode is observed to grow when the rotation is reduced (Politzer et al., 2008; Solomon et al., 2013). Differential flow also helps to stabilize the double tearing mode (Dewar and Persson, 1993), by decoupling the two rational surfaces.

Energetic-particle effects

Energetic particles (EPs), although a minority particle species in fusion plasmas in terms of particle number density, often play important roles in MHD instabilities. There are two main reasons for this. (i) EPs have much higher energy than the background particle species, so that the EP pressure, which is the product of the particle density and temperature, can sometimes contribute a significant fraction to the total plasma equilibrium pressure. As an example, fusion born alphas contribute more than 20% to the total pressure in an ITER advanced plasma scenario (Liu, 2010). This modification of equilibrium pressure due to EPs changes the MHD stability behavior of pressure driven modes. (ii) EPs can also directly interact with MHD perturbations and thus modifying the instability and even triggering new instabilities.

There are two key physics mechanisms involved in the direct interaction between EPs and MHD modes. One is the wave-particle Landau resonance. MHD modes can often be treated as (stable or unstable) waves. Wave-particle Landau resonances are used in many contexts in fusion devices. An important application is plasma heating by launching radial waves with frequency matching the cyclotron frequency of thermal ions or electrons.) When certain frequencies associated with the EP motion match the frequency of the MHD perturbation, free energy transfer between EPs and the MHD mode then ensues. The direction of the energy transfer, which determines whether the MHD mode is stabilized or destabilized, depends on many factors. The other physics mechanism is associated with trapped EPs, whose bounce motion forms a banana orbit that rotates along the toroidal angle of a tokamak. This toroidal motion of the banana orbit tends to conserve the vertical flux enclosed by the center line of the banana (Northrop, 1963). If the toroidal rotation frequency of the banana orbit is faster than the perturbation frequency, the mode can be stabilized.

Examples of EP stabilization of MHD modes include the IK (Porcelli, 1991, 1996; Graves, 2004), the RWM (Chapman et al., 2009; Liu, 2010), and the TM and NTM (Hegna and Bhattacharjee, 1989; Cai et al., 2011). TM/NTM stabilization by EPs may involve either direct interaction (Hegna and Bhattacharjee, 1989) or another, indirect way where EPs modify the external tearing index (Liu et al., 2012). Examples of EP triggering of MHD instabilities include the fishbone (Chen et al., 1984; Coppi and Porcelli, 1986), e-fishbone (Wong et al., 2000), Alfvén eigenmodes (Fu and Van Dam, 1989), and energetic particle modes (Chen, 1994).

Principal MHD instabilities

Overview

MHD instabilities are generally driven by a combination of electromagnetic and thermodynamic forces. The basic equilibrium relation Eq. (33) $\nabla P = \mathbf{j} \times \mathbf{B}$ describes the balance of these forces. Normally, instabilities are categorized as being either “current-driven” or “pressure-driven” depending on what is considered to be the major drive. For example, at zero pressure, the instabilities are naturally considered as “current-driven.” Onset criteria can be obtained in this limit. With pressure, instabilities that arise when the current-driven mode onset criteria are not satisfied are then usually considered as pressure-driven. This categorization works quite well in most cases since there are features in the mode structure that correspond to the distinction.

Within this categorization, a number of other labels are applied to various MHD instabilities. Early in the history of MHD, the first instabilities studied were called kink, flute, and sausage modes. The label “kink” or “external kink” is largely synonymous with “current-driven kink.” These are driven by a current density gradient or jump at the edge. Flute modes are an alternative term for pressure-driven interchange modes. The sausage instability corresponds to pressure-driven modes with poloidal mode number $m = 0$. Peeling modes are a subcategory of the external kink, corresponding to toroidal mode numbers $n > 1$.

Ballooning modes are pressure-driven. Normally the label is applied to the high- n version but pressure-driven modes at intermediate and low n all have the characteristic “ballooning structure” with a large coherent motion on the outboard side as field lines interchange locally there. Interchange modes are simply ballooning modes in the low shear limit where the interchange is global along the field line. Infernal modes, similarly refer to localized instabilities destabilized by a large pressure gradient in a locally low shear region. Bear in mind that these labels are really only valid in certain limits; most instabilities in real plasmas are a mixture of different features. The peeling-ballooning modes understood to be responsible for ELMs are a prime example of instabilities that defy characterization as either current driven or pressure driven (Snyder et al., 2002). Both the pressure and current-density gradients are important drivers. The IK, responsible for the sawtooth phenomenon in tokamaks, is another example. In a torus, the ideal IK is stable at zero pressure (Bussac et al., 1975), but has a very low β limit. The β limit also vanishes if the wall is removed (Turnbull and Troyon, 1989).

Axisymmetric modes are the ideal $n = 0$ instabilities. These are essentially current-driven in the sense that they are largely independent of the pressure and depend only on the gross features of the current profile. For a circular cross section, this mode is stable. In an elongated tokamak, the mode is a vertical shift ($m = \pm 1$).

For non-ideal modes, resistivity is an important driver of instabilities. Resistivity allows rational closed field lines to break and reconnect with a different topology. Two different modes can form corresponding to two different parities. An EK can result when two adjacent flux surfaces interchange position by breaking and reconnecting. This is the resistive version of the ideal kink mode. Alternatively, the closed rational surface can split with the two parts moving apart and opening up a magnetic island. The tearing apart of the surface provides the name “tearing mode.” This mode has no counterpart in ideal MHD. Fig. 3 shows how various MHD instabilities qualitatively look like, in terms of the radial distortion of the plasma caused by the instabilities.

Wesson identified the regions unstable to current driven modes in a model cylindrical plasma in the space of q_0 and q_s/q_0 (Wesson, 1978). Here, q_0 and q_s are the safety factor on axis and at the edge. These are the most important parameters determining current-driven instabilities. Fig. 4 is a corresponding “Wesson diagram” replotted in terms of q_0 and q_s for a generic strong Dee shape in

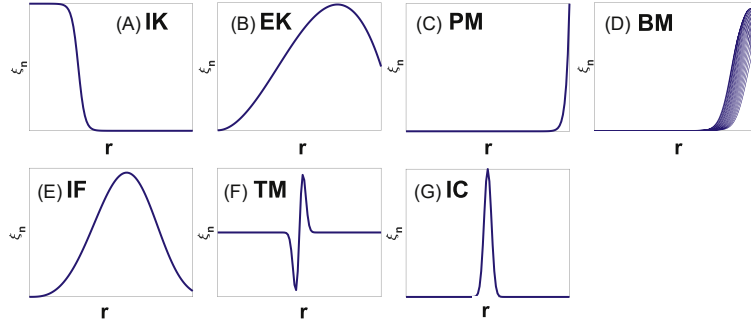


Fig. 3 Sketch of eigenmode structure, in terms of radial plasma displacement, for typical MHD instabilities: (A) internal kink (IK), (B) external kink (EK), (C) peeling mode (PM), (D) ballooning mode (BM), (E) infernal mode (IF), (F) tearing mode (TM), (G) interchange mode (IC).

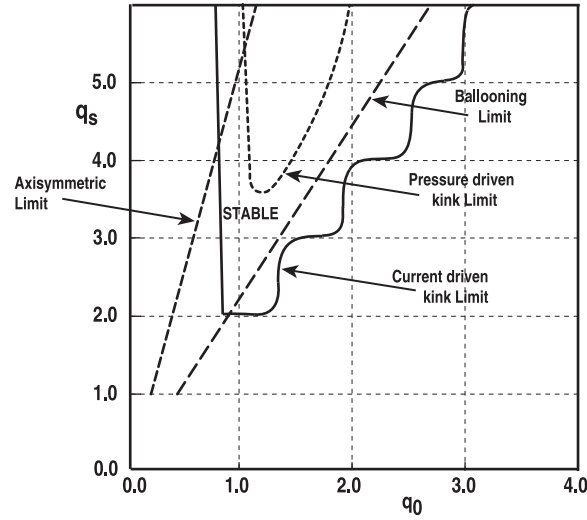


Fig. 4 Regions in parameter space of different modes in the q_0 , q_s parameter space. Here, q_0 and q_s are the safety factor on axis and at the edge.

toroidal geometry with a monotonic current density profile and zero β . The boundary follows a generic stair-step pattern with steps at or near rational values of either q_0 or q_s . Changes in the current profile shape for fixed q_0 and q_s change the steepness of the staircase. However, there is no stable range in particular for $q_s < 2$ for any current profile. Pressure tends to smear the boundaries. Superimposed is the axisymmetric stability boundary and the general form of the stability limit against pressure driven kink modes for a fixed, finite β_p . The stable region is in the center. The slope of the axisymmetric boundary varies with $\mathcal{L}_i$. With increasing β_p , the pressure driven kink boundary dips to lower q_s . A generic ballooning stability boundary is also shown; again, the slope in this diagram varies with $\mathcal{L}_i$, also with β_p .

Axisymmetric modes

The axisymmetric instability is a vertical displacement of the plasma cross section in a vertical magnetic field. The plasma consists of a toroidal current in a vertical magnetic field. One can imagine a rigid elliptical cross section like an egg being squeezed by the vertical field. The shape of the egg diverts the lateral squeezing forces into a net vertical force and the egg slips rigidly in the field. Stability is determined then by the shape of the egg relative to the vertical field. For a plasma with an elliptical cross section, the plasma slips easily. Image currents in the wall generated by Lenz's Law, however, provide a stabilizing reaction force by opposing the fields generated by the vertical motion.

For a circular cross section plasma, the current loop is stable to a vertical shift. Essentially, for the plasma to slip up or down from its equilibrium position, it has to compress the external vertical field, requiring a source of energy. However, an elongated elliptical plasma can slip vertically with less distortion of the vertical field. For some elongation, the plasma becomes unstable to this motion and disrupts. For a large aspect ratio ellipse, the eigenmode is the pure axisymmetric vertical shift characterized by toroidal mode number $n = 0$ and an equal mix of poloidal mode numbers $m = \pm 1$ with constant amplitude as a function of radius;

$$\xi(r, \theta, \phi) = \xi_0(e^{i\theta} + e^{-i\theta}) \sim \xi_0 \cos\theta \quad (60)$$

In a torus there is some deviation from this due to the $1/R$ dependence of the fields. Additional $m = \pm 3$ components appear in a Dee shape and a stronger radial profile dependence of the $m = 1$ shift plus higher m components appear with higher order shaping.

In actual experiments, the instability is stabilized by a nearby surrounding conducting wall in which image currents are induced to oppose the magnetic field changes, preventing the perturbed flux from penetrating the wall (Lenz's Law). With finite resistivity, the image currents decay in a characteristic $1/R$ time and requires active feedback to fully stabilize it. (L and R are the wall inductance and resistivity, respectively.) The axisymmetric instability is well understood. Calculations of the ideal eigenfunctions have been shown to match the measured boundary displacements. In most cases, the observed open loop resistive wall growth rate can easily be fitted by simple plasma models assuming a rigid vertical displacement. This is generally sufficient for the feedback algorithms.

Internal kink, sawtooth, and fishbone

The IK mode, with toroidal mode number of $n = 1$ and poloidal mode number $m = 1$ was predicted by ideal MHD theory to be unstable early in fusion research. In a toroidal plasma, the $n = 1$ internal kink mode, as a linear eigenmode, has many poloidal harmonics. The $m = 1$ harmonic is dominant. However, some of the aspects associated with this mode, in particular the non-linear aspects and the interaction between this mode and energetic particles, are still active research areas. For recent research on the internal kink and sawteeth, see for example (Jardin et al., 2020).

From the MHD viewpoint, the primary driving mechanism for the IK instability is the free energy associated with the plasma current. More precisely, a critical condition for the mode instability is that the safety factor on the magnetic axis, q_0 , should be below 1. This condition is, however, not sufficient for an unstable IK. In a tokamak plasma, the IK becomes unstable only if the plasma pressure exceeds certain critical value. The mode growth rate γ , within the MHD framework, is in fact proportional to $(1 - q_0)(\beta_{pcr}^2 - \beta_p^2)$, where β_{pcr} is the critical value typically below 0.4. This is the well-known Bussac criterion (Bussac et al., 1975) for the IK instability. This criterion shows that the IK is also partly driven by the plasma pressure.

Because the mode growth rate is proportional to $(1 - q_0)$ (provided that the Bussac pressure limit is exceeded), the IK instability strength depends critically on the proximity of the on-axis safety factor to unity. In typical tokamak discharges, the on-axis safety factor is not far from 1, resulting in an IK that is not far from marginal stability. A weakly unstable IK is subject to a range of non-ideal effects beyond the ideal single-fluid MHD description. Important effects include the plasma diamagnetic stabilization, the plasma resistivity, and the drift kinetic effects associated with plasma particles. The diamagnetic stabilization of the IK is often referred to in the literature as the finite Larmor radius (FLR) effect, since it originates from the FLR of thermal ions. On top of these non-ideal effects, the plasma equilibrium toroidal flow also affects the mode stability. In what follows, we briefly discuss each of these effects.

A weakly unstable IK is often subject to strong diamagnetic stabilization in a tokamak plasma (Ara et al., 1978). This is because the typical mode growth rate in this case is comparable to the thermal ion diamagnetic frequency. The dispersion relation for the mode (complex) frequency ω satisfies $[\omega(\omega_i^* - \omega)]^{0.5} = -\delta W$, where ω_i^* is the thermal ion diamagnetic frequency and δW is the normalized perturbed potential energy. δW here is normalized by the plasma inertia. As usual, $\delta W < 0$ indicates free potential energy that drives instability. In the absence of the diamagnetic stabilization, and for a purely growing IK mode, the mode growth rate scales linearly with the perturbed potential energy.

Plasma toroidal flow can be either stabilizing or destabilizing to the IK. The fundamental physics mechanism for flow stabilization of the IK is the so-called gyroscopic stabilization (Wahlberg and Bondeson, 2000), where the perturbation, which has a rigid structure, is stabilized by fast rotation. The required rotation speed for full stabilization of the mode has to be very large (a large fraction of sound speed). The flow shear, on the other hand, can destabilize the IK essentially due to the Kelvin-Helmholtz type of mechanism. Associated with toroidal flow are various inertial forces (centrifugal and Coriolis forces), which can be either stabilizing or destabilizing depending on the equilibrium profiles (Wu et al., 2019).

Non-linear evolution of the IK often manifests itself as the sawtooth phenomenon, which is observed in experiments. There are different theoretical models for sawteeth. The first and probably the most well-known model is the Kadomtsev model (Kadomtsev, 1975), where full magnetic reconnection was assumed within the $q = 1$ surface during the non-linear evolution.

Lastly, drift kinetic effects, in particular those associated with toroidal precession of trapped EPs, can have significant influence on the IK stability and the sawteeth behavior. Drift kinetic effects from thermal particles can also stabilize the IK (Hu et al., 2006), although the effect is often not dramatic. These effects are well summarized by the Porcelli model (Porcelli et al., 1996), where three different criteria were proposed to judge the IK instability and the subsequent onset of sawteeth. These Porcelli criteria have been successfully applied to interpret many experimental results in tokamak plasmas.

Drift kinetic effects of EPs not only stabilize the IK, but can also drive a new type of instability called the fishbone mode (FB). The word "fishbone" originates from the characteristic magnetic signals observed in experiments, which is actually a result of a non-linear interaction between the instability and EPs (Chen et al., 1984; Porcelli, 1991; Porcelli et al., 1996). The fundamental physics here is the wave-particle Landau resonance, where an otherwise stable IK absorbs free energy from EPs, and thus becomes unstable. Since the mode is driven unstable by EPs, the mode frequency matches that of the characteristic frequency of the drift motion of EPs. The most common case is again the toroidal precession of trapped EPs. Another drive is associated with diamagnetic rotation of EPs. Plasma resistivity often stabilizes the FB (Biglari and Chen, 1986; Wu et al., 2018)—an effect opposite to that for the IK. For the FB, the resistive layer dissipates the free energy.

EK/Resistive-wall modes (RWMs)

There are two drive mechanisms for the external kink instability, the plasma pressure and the plasma current. The purely current driven EK mode has been studied since early on in fusion research (Shafranov, 1970; Wesson, 1978). A key milestone in realizing the importance of pressure effects was the discovery of the Troyon β limit (Troyon et al., 1984), showing that the maximal achievable normalized plasma pressure, before the discharge disrupts, scales in proportion to the normalized plasma current, as shown in Eq. (41).

Plasma pressure is the most typical drive in present and future tokamak devices. This mechanism resembles the ballooning drive. In fact, the eigenmode associated with the pressure driven EK is typically localized at the outboard low-field side of the torus, similar to the ballooning instability. This type of EK is therefore also often called kink-ballooning instability. An important distinction with the ballooning instability, however, is the toroidal mode number associated with the perturbation. The EK is the long wavelength global MHD instability along the toroidal (and poloidal) angle, with a typical toroidal mode number of $n = 1 - 3$, while the ballooning instability has typical toroidal mode numbers greater than 10.

For the EK driven unstable in a low-pressure plasma by the plasma current, when the nq_s value (in a limiter plasma) is just below an integer m , an instability results with toroidal mode number n , and predominantly poloidal mode number m , which causes a large plasma displacement near the plasma edge. For $n > 1$ and $m > 2$, this instability is often called a kink-peeling mode. When the plasma current is sufficiently high in a limiter plasma that q_s is below 2 (Wesson, 1978), this always triggers a strongly unstable $n = 1$ EK which leads to plasma disruption if not suppressed.

In a plasma with a divertor configuration, the edge safety factor is large (mathematically infinite at the separatrix). The associated large magnetic shear near the plasma edge can stabilize the resulting mode (Webster and Gimblett, 2009). Despite the magnetic shear stabilization, a disruptive instability is still often observed in divertor experiments when the safety factor at the 95% flux surface, q_{95} , approaches 2.0. This puzzle was recently resolved by invoking the effect from a steep edge plasma resistivity profile (Turnbull et al., 2016).

Independent of the drive mechanism, it has been found that the presence of an ideal conducting wall, located sufficiently close to the plasma, can stabilize the ideal EK instability up to a certain limit. In reality, the wall almost always has finite conductivity. This allows partial escape of the perturbed radial magnetic field, on a time scale comparable to the eddy current decay time in the resistive wall. As a result, the EK again becomes unstable in the q_s or q_{95} parameter windows as described above, but now on a much longer time scale. This slowly growing instability is called the RWM (Chu and Okabayashi, 2010). Note that the resistivity here refers to that of the resistivity wall, not the plasma. The resistive-plasma resistive-wall mode is another interesting topic that has been under extensive studies in recent years (Betti, 1998; Finn, 2006).

The eigenmode structure of the RWM still remains global, similar to that the EK (Subtle differences in the mode structure for the resonant harmonics near the mode rational surfaces appear for the RWM in a toroidal plasma, which often do not play a significant role.). The mode growth rate is, however, significantly reduced as mentioned. This has several significant implications. (i) The presence of this residual instability means that the RWM can potentially still cause a serious disruption of the plasma. (ii) The much lower growth rate, or in general much lower (complex) mode frequency, implies other physics beyond ideal MHD may play important roles. (iii) The much longer time scale of the RWM growth, in milliseconds or longer, also makes it practical to design an active control system to actively stabilize the mode.

There exist several passive stabilization mechanisms for the RWM utilizing additional physics beyond ideal MHD. The first important effect is continuum resonance damping of the mode in a toroidally rotating plasma. Unlike many other MHD modes, the RWM, even in the linear regime, does not rotate with the plasma. A tokamak plasma typically rotates at a frequency of several percent of the Alfvén frequency along the toroidal direction. There are many reasons for finite toroidal rotation, e.g., due to toroidal momentum source associated with tangential neutral-beam injection, the presence of intrinsic toroidal torque. The RWM, on the other hand, has a frequency that is of the same order of the much smaller inverse wall time; the RWM appears to be essentially “locked” to the resistive wall. The mode thus rotates in the plasma frame, opening the possibility of Landau resonances between the mode and various continuous waves (which rotate together with the plasma), such as the shear Alfvén waves and sound waves (more precisely slow magneto-acoustic waves). Since these continuum waves are stable, they can tap free energy from the RWM through resonances, thus stabilizing the mode. Such a mechanism was discovered numerically in a seminal work by Bondeson and Ward (1994) and later analytically confirmed Betti and Freiberg (1995).

Numerical modeling finds that the above stabilization mechanism often requires a plasma toroidal rotation of several percent of the Alfvén frequency (Bondeson and Ward, 1994; Chu et al., 1995; La Haye et al., 2004), in order to fully suppress the RWM in a tokamak plasma. Later experiments (Reimerdes et al., 2007), where the plasma toroidal flow was intentionally kept slow by balanced neutral-beam injection, the RWM instability did not occur even when the plasma pressure exceeded the no-wall Troyon limit, contradicting theory predictions. The puzzle is resolved by evoking additional physics, namely drift kinetic theory (Hu and Betti, 2004), where Landau resonances between the mode and the drift motion of plasma thermal particle species become important. Indeed, because the toroidal precessional drift frequency of trapped thermal particles (both ions and electrons) is typically very low (well below the thermal particle diamagnetic drift frequency), a strong resonance occurs if the RWM also has a small frequency in the plasma frame, which is the case when the plasma toroidal rotation is slow. Since then, precessional drift-kinetic stabilization of the RWM has been confirmed by extensive numerical modeling work (Liu et al., 2008; Chapman et al., 2009; Berkery et al., 2010). The MHD-kinetic hybrid description of the RWM instability still remains an active research area.

Resistive interchange/TM/NTM/locked modes

The resistive interchange (RI) mode and the TM are the two most important spatially localized resistive instabilities. There are many interchange instabilities in fusion plasmas. For instance, the so-called sausage instability in a Z-pinch can also be viewed as an interchange instability based on the fundamental drive mechanism. Those with finite toroidal mode number and poloidal mode number $m > 1$ are most important. Compared to the TM, the interchange mode has opposite parity. More specifically, if we define the mode parity by the associated radial plasma displacement, the interchange mode has even parity (i.e., symmetric about the mode rational surface) and the TM has odd parity (i.e., anti-symmetric about the mode rational surface).

Both instabilities are driven by finite plasma resistivity (assuming that the ideal counterparts are stable), by allowing a change in the magnetic topology near the rational surface. According to the Mercier criterion (Mercier, 1960), the ideal interchange is unstable if a quantity $D_I = D_R + \frac{1}{2}$ is positive. D_R is proportional to the equilibrium pressure gradient at the mode rational surface. However, the resistive interchange is unstable if D_R is positive. An equilibrium with zero or negative D_R , on the other hand, can be unstable to the TM. Since the latter typically holds for a tokamak plasma, the RI mode is usually not as critical as the TM. (Even if a resistive interchange is unstable in a tokamak plasma, it can be easily stabilized by plasma rotation.) The following will therefore focus on the TM and its neoclassical counterpart, the NTM.

The linear TM instability is driven unstable by the plasma current profile. Because the mode displacement is strongly localized near a rational surface, the instability is often studied via a matching procedure, where the ideal MHD equations at marginal stability are solved and the solution is matched to that of the resistive MHD equations within a narrow layer around the rational surface. Typically, the perturbed radial field (or magnetic flux) is used to perform the matching procedure. The logarithmic derivative of the outer ideal solution experiences a jump across the mode rational surface. This jump, denoted as Δ' , characterizes the free energy associated with the equilibrium current density gradient that drives the TM unstable. The outer solutions are matched to the non-ideal solution obtained in a narrow inner layer around the rational surface to produce a continuous solution. In a pressureless plasma, the matching condition yields a simple relation between the mode growth rate γ and Δ' , $A\gamma^{5/4}\eta^{-3/4} = \Delta'$, with η being the plasma resistivity and $A > 0$ a geometric factor. This dispersion relation implies that the TM is unstable whenever the tearing index Δ' is positive, and that the mode growth rate scales as $\gamma \sim \eta^{3/5}$.

In an equilibrium with finite pressure (more precisely with finite negative D_R), an important effect, namely the so-called favorable average curvature effect, becomes important and modifies the TM dispersion relation to a form $A\gamma^{5/4}\eta^{-3/4}(1 - ED_R\gamma^{-3/2}\eta^{1/2}) = \Delta'$, where $E > 0$ is now another geometric factor (Glasser et al., 1975). This new physical effect, coming from the resistive layer solution, qualitatively changes the TM stability criterion. Then, Δ' has to be larger than a critical positive number, in order to ensure the TM instability. In other words, this average curvature effect plays a stabilizing role (thus “favorable”) to the TM instability. This is one of the key physics elements associated with the TM in a toroidal plasma. Later studies found additional interesting layer physics that modify the TM dispersion relation, e.g., a large island correction that restores the $\gamma \sim \eta^{3/5}$ scaling at high resistivity (Militello et al., 2004), and a cancelation effect on the favorable curvature stabilization due to anisotropic thermal transport (Lutjens et al., 2001; Connor et al., 2015).

Because of the strong radial localization of the instability, non-linear effects are important for the TM. A critical consequence of a developing TM instability is the change of the magnetic topology near the mode rational surface, where a chain of magnetic islands with helical structure forms. Essentially all the non-linear effects are related to the local change of the plasma current in the presence of magnetic islands. The first non-linear effect was identified by Furth et al. (1963), and leads to an algebraically (instead of exponentially) growing magnetic island. Later studies identified an important role played by the neoclassical bootstrap current (Qu and Callen, 1985; Fitzpatrick, 1995; Wilson et al., 1996), that modifies the TM stability and results in the neoclassical tearing mode (NTM). The bootstrap current is stabilizing to the TM. If, however, a portion of the bootstrap current is missing in the presence of 3D magnetic islands, the stabilizing role is reduced and the NTM can be triggered. Note that there can be different ways to change the bootstrap current, e.g., due to particle flattening of the plasma pressure profile inside the magnetic islands (Fitzpatrick, 1995) or due to induced polarization current (Wilson et al., 1996). All these effects involve the finite island size and are thus intrinsically non-linear, requiring a threshold in the island size. (There have been discussions on triggerless/seedless NTM which presumably originate from a linearly unstable TM due to large and positive Δ').

The time evolution of the NTM islands is well described by the quasi-linear Modified Rutherford Equation (MRE) (Sauter et al., 1997; Turnbull et al., 2002), which includes these effects, but excludes nonlinear coupling to other toroidal mode components with different n . The other non-linear effect is the interplay between a growing island and the plasma toroidal rotation. This is important for a full understanding of locked modes (LMs), which correspond to the locking of an NTM to the wall. This is, in fact one of the major causes of plasma disruptions in experiments (besides VDE and EK discussed before). Mode locking, as well as NTM control, will be discussed in the later section on MHD stability control.

Toroidal Alfvén eigenmodes

The Hermitian nature of ideal MHD implies that the squared frequency, ω^2 , which is the eigenvalue, is always real. Stable modes correspond to those with non-negative ω^2 . In the cylindrical approximation, it can be shown that the ideal MHD spectrum consists of a possible set of discrete or isolated unstable modes and a continuum of irregular modes on the stable side arising where the coefficient of the second order term in the eigenvalue equation vanishes. These continuum modes are highly localized at the point where the coefficient vanishes and actually consist of two overlapping continua corresponding to sound waves and Alfvén waves,

respectively. In the ideal limit, they have infinite energy but are regularized by non-ideal effects. On the unstable side, the eigenmodes correspond to the array of physical ideal MHD instabilities discussed above. There can be a finite (or zero) number of unstable modes or a countably infinite number. In the latter case, generally corresponding to violation of the local Mercier criterion, the eigenvalues have an accumulation point at marginal stability. The stable spectrum is shown in Fig. 5. In ideal MHD the stable continuum modes are purely oscillatory and have no damping or drive, lying on the imaginary axis in ω space. In reality, they tend to be strongly damped by various non-ideal effects, most notably ion Landau damping.

This situation persists in toroidal geometry but with an important modification. In the cylinder, there is a separate continuum for each poloidal mode number m . In the torus, the equations for different m are coupled and near the crossing points, degenerate perturbation theory applies and the curve crossings are replaced by the reconnected curves shown, leaving gaps in the frequency. This is shown in Fig. 5A. Within the gaps, global regular eigenmodes appear in the Alfvén continuum at isolated frequencies. These consist of coupled m and $m + 1$ components with a typical example shown in Fig. 5B. Again, in ideal MHD these are purely oscillatory modes with no damping. However, the non-ideal damping of the irregular continuum modes is inversely proportional to the radial wavelength and is greatly weakened for this global mode. In particular, when the gaps overlap across a wide radius, there is no radius at which this global mode interacts with or couples to a localized continuum mode and the damping becomes small. Instead, these modes are relatively easily driven unstable. These are known as the toroidal Alfvén modes (TAEs).

The frequencies of typical TAE and other similar higher frequency global modes are generally sufficiently high that the damping rates are much smaller; $\omega_{TAE} \gg \gamma_{damp} \sim \gamma_{drive}$. In that case, the ideal estimate of the real frequency is generally sufficiently accurate and the drive and damping can be computed using the computed ideal mode. The non-ideal effects simply add an additional

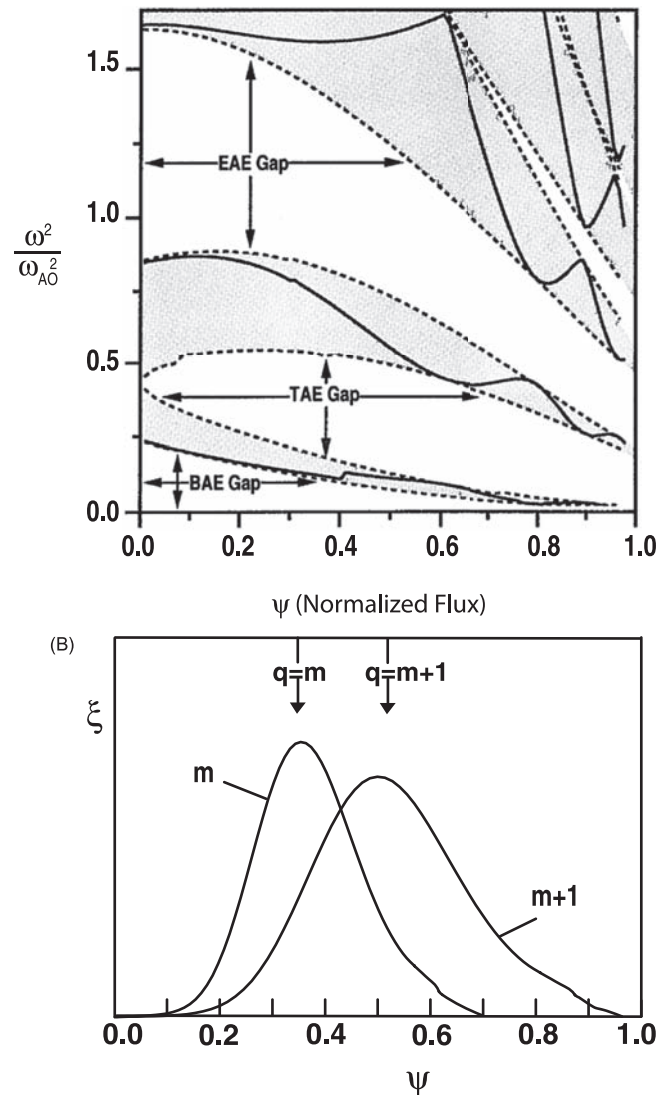


Fig. 5 (A) The continuous spectrum formed from frequencies ω^2 , at which the coefficient vanishes against radius for different m and showing the TAE, BAE, and EAE gaps. (B) TAE mode structure consisting of coupled m and $m + 1$ components.

imaginary part to $\omega_{TAE} \rightarrow \omega_{TAE} + i(\gamma_{drive} - \gamma_{damp})$, corresponding to a growth or damping rate. For most situations this is a valid assumption. In addition to the weakened ion Landau damping, several other non-ideal mechanisms dampen the TAE and other stable global modes. The most important of these is the so-called “continuum damping,” which results when an open gap does not extend all the way across the radius. In that case, the TAE at one broad location couples directly to a continuum mode (irregular in ideal MHD), inheriting its large inherent ion Landau damping. Several kinetic effects provide additional damping sources but more important, result in a second global mode in the gap at higher frequency. This is known as the Kinetic Toroidal Alfvén Eigenmode (KTAE) and has no MHD counterpart. The instability drive of most interest is from fast non-thermal ions via an inverse ion Landau process. Such ions are typically present in current experiments as a result of external ion heating of the plasma by either neutral-beam injection or RF waves. In future experiments, the concern is that energetic alpha particles from the fusion reaction would provide the drive.

In the case of radio frequency (RF) waves, the waves themselves can be designed to resonate at Alfvén-like frequencies and used to probe the plasma. When the external RF wave resonates with a stable MHD plasma wave as the frequency is varied, the width of the resonance peak provides a measure of the inverse of the damping rate of the mode. This technique, called MHD Spectroscopy, has been used to study the stable Alfvén continuum spectrum as well as low frequency marginally stabilized MHD kink modes.

In addition to toroidicity, other geometric effects can couple the poloidal harmonics and break open new gaps in the continuum. The most important of these are ellipticity and triangularity, with their corresponding global Ellipticity Alfvén Eigenmode (EAE) and Triangularity Alfvén Eigenmode. With finite β , the Alfvén continuum lifts off the marginal point $\omega^2 = 0$ leaving another gap below with width proportional to β_p . The gap is filled with sound wave continua. With respect to the discrete frequencies of these modes and their gaps, the effect of finite β appears to raise them by an amount roughly proportional to β_p . However, relative to the gap, the global mode moves slightly down in frequency with increasing β_p (Turnbull et al., 1993).

This lower gap is known as the Beta-induced Alfvén Eigenmode (BAE) gap (Turnbull et al., 1993). Early experiments searching for the TAE mode also found global Alfvén-like modes within the BAE gap. Numerical searching subsequently found global largely single harmonic modes sitting in this gap. Theory is somewhat complicated for these modes since the usual assumptions of small coupling to the sound waves are not really valid, although the numerical calculations included it. The key physics in this mode is compressibility. The gap and mode are associated with the energy required by a global Alfvén wave to compress as it moves in the curved torus. This is in contrast to a sound wave that propagates as an actual compressional wave. Ignoring compressibility to eliminate the sound waves also eliminates the BAE physics.

Several other Alfvén eigenmodes were discovered in experiments that revealed so-called “chirping modes,” in which the observed frequency “chirped up” until the mode disappeared or would be converted to a TAE mode. Ideal stable modes associated with these were subsequently found in continuum gaps that resulted from these particular discharges having an off-axis reversal in the q profile. Hence, they were called Reverse-shear Alfvén Eigenmodes (RSAEs) (Kramer and Fu, 2006). The gap was determined at any time by q_{min} which varied in time and explained the chirping behavior. Intermittent behavior sometimes results from a periodic expulsion of the fast ions driving the mode unstable, thereby suddenly stabilizing it, and a reforming of the previous beam distribution and plasma conditions.

Edge localized modes (ELMs)

Edge localized modes (ELMs) are instabilities driven by the large edge pedestal pressure gradient and its associated bootstrap current in tokamak H-mode plasmas. ELMs are generally divided into different types such as Type I-IV and “grassy” (Leonard, 2014). Not all these types of ELMs are driven by macroscopic MHD modes. For instance, the type-III ELM is believed to be driven by micro-tearing mode (Snipes et al., 1998). The focus in the following discussion is on Type-I ELMs which have macroscopic MHD origin. More importantly, Type-I ELMs have been experimentally demonstrated to lead to unacceptable material erosion in future reactor-scale tokamaks by inducing thermal and particle fluxes onto the plasma facing components.

Type-I ELMs can be explained by a model for edge localized modes as predominantly ideal instabilities with low to intermediate toroidal mode number. This idea was around for some years (Turnbull et al., 1986; Strait, 1994), and was proposed in later years on the basis of ideal MHD stability calculations for DIII-D discharges using a model for the pressure driven bootstrap current included in the equilibrium reconstruction and DIII-D ELM experiments (Osborne et al., 1999; Ferron et al., 2000; Lao et al., 2001; Snyder et al., 2002). This current produces a large current density peak near the edge of the plasma sufficient to destabilize low to intermediate n instabilities driven partly by that current density gradient (peeling mode component) and partly by the edge pressure gradient (ballooning mode component). The peeling-ballooning model has since been incorporated into a model known as the EPED model that combines the MHD peeling-ballooning stability limit with other non-MHD limits to predict the actual ELM onset (Snyder et al., 2009, 2011). This was made possible by the development of the ELITE (Edge Localized Instabilities in Tokamak Experiments) edge stability code to efficiently compute intermediate to high n modes (Wilson et al., 2002; Snyder et al., 2002).

The basic MHD instabilities behind Type I ELMs are known as peeling-ballooning modes. The external kink mode, when becoming unstable at relatively high edge safety factor (more precisely at high nq_{95}), tends to localize near the plasma edge with a large kink component driven by the strong bootstrap current gradient, as well as a peeling component at the separatrix, and a low to intermediate n toroidal mode number. This is called kink-peeling mode or simply peeling mode. The other component of the Type I ELM is the ballooning mode, which is typically a short wavelength perturbation along both toroidal and poloidal angle, and is geometrically localized near the plasma outboard edge at the low field side. The ballooning mode is described as local instability here in a relative sense. Sometimes, the mode with finite (but large n) is referred to as non-local, in comparison to the infinite- n ballooning mode

which is localized on magnetic flux surfaces. The ballooning instability is driven by the plasma pressure, more precisely the large pressure gradient near the plasma edge. Since large pressure gradients occur in H-mode plasmas with large edge pedestals, the latter is the primary free energy source driving ballooning instability in a tokamak plasma. Note that with further increase of the plasma pressure, the ballooning mode can enter into a second stability regime (Greene and Chance, 1981). This is primarily because higher pressure increases the Shafranov shift and compresses the equilibrium magnetic field lines in the low field side of a tokamak plasma, which is stabilizing for MHD instabilities. Because the ballooning mode is a high- n perturbation, the full MHD equations can be significantly reduced in order to efficiently but approximately describe this instability, by dropping higher order terms in $1/n$. A critical analytic development is discovery of the so-called ballooning representation (Connor et al., 1978), which allows a more natural (in physics sense) representation of the eigenmode structure and substantially facilitates theoretical (and often numerical as well) analysis of this mode. The extension of the ballooning formalism to higher order in $1/n$ enabled development of the ELITE code (Wilson et al., 2002; Snyder et al., 2002).

The ballooning mode is also subject to many physics effects beyond the single-fluid ideal MHD description (Connor et al., 1988). The most important one among these effects is the diamagnetic stabilization of the mode (Rogers and Drake, 1999; Hastie et al., 2000; Snyder et al., 2011). Based on understanding of both the ideal and non-ideal physics, a theoretical model—the EPED model (Snyder et al., 2009, 2011)—has been developed to predict the plasma pedestal conditions for onset of Type I ELMs. The key idea of the model is the realization that the pedestal structure—both the height and width—can be constrained by two modes in a tokamak H-mode plasma. One is the peeling-ballooning constraint and the other constraint is due to a more localized, transport type of instability such as the kinetic ballooning mode. These two modes constrain the pedestal height and width following two different scaling curves, and their intersection indicates what is realizable in experiments. The EPED model has been successfully applied to a range of tokamak experiments (Snyder et al., 2011). In addition, the model predicted the existence of, and the access route to, a new high performance plasma regime called super-H mode that was subsequently realized in experiments (Solomon et al., 2014, 2016; Snyder et al., 2015, 2019; Hughes et al., 2018; Knolker et al., 2020).

The EPED model reproduces the triggering mechanism of type I ELMs well (Snyder et al., 2011). However, the ELMs observed experimentally represent a highly non-linear stage of the instability. For instance, the filamentary structure that is observed in experiments inside the plasma separatrix is no longer a linear ballooning instability. This is an ongoing area of theoretical research.

Finally, there are ELM-stable regimes in tokamak plasmas. These ELM-stable regimes are favorable for tokamak operations due to their benign feature and thus minimizing bursts of heat and particle fluxes reaching the plasma facing components. The L-mode is ELM-stable but is not particularly attractive in terms of energy confinement. In the context of high confinement regimes, the QH-mode (Burrell et al., 2001, 2016) and I-mode (Whyte et al., 2010) regimes are two promising candidates without ELMs. The former often has a saturated MHD instability (the so-called Edge Harmonic Oscillation EHO) that helps to provide necessary transport in the pedestal region to avoid ELMs. Recent studies also identified a QH-regime with wide pedestal, where no EHOs are needed to provide transport (Burrell et al., 2016). The latter does not require an edge pedestal in the plasma density, only pedestal in plasma temperature, which helps avoid ELM triggering but meanwhile maintains good particle and energy confinement. Furthermore, application of 3D magnetic perturbations near the plasma edge also helps to suppress ELMs and produces ELM-stable regimes (Evans et al., 2004). Of particular recent interest are plasmas with negative triangularity shape (Medvedev et al., 2015; Ren et al., 2016; Austin et al., 2019; Kikuchi et al., 2019), which can also provide an ELM-stable high confinement regime. Negative triangularity equilibria generally do not have favorable average magnetic curvature, resulting in strong ballooning instabilities that suppress the formation of an edge pedestal and hence in many cases the discharges remain in L-mode, with no ELMs, even at high power.

Stable plasma response to external 3D fields

Since plasmas consist of unbound free charges, the charges individually respond and the plasma collectively responds to applied external fields. The predominant immediate response is electromagnetic and the MHD formalism captures this well. In essence, the plasma responds by setting up Alfvén waves that propagate through the system and re-establish local force balance everywhere forming a new 3D equilibrium state:

$$\nabla \cdot (\mathbb{P} + \delta\mathbb{P}) = \mu_0 (\nabla \times (\mathbb{B} + \delta\mathbb{B})) \times (\mathbb{B} + \delta\mathbb{B}) \quad (61)$$

This state generally has different transport properties from the original unperturbed state. For example, topological changes, particularly the presence of open field lines, can modify transport dramatically. Hence, following the MHD response, there is a longer term, transport response.

The response of the plasma is important for understanding several related phenomena. Early in tokamak fusion research, the phenomenon of locked modes plagued progress. Depending on the density, a rotating tearing mode would arise and as it grew, slow the plasma, amplifying the growth further, until the plasma locked to an error field fixed with respect to the vessel wall and usually subsequently disrupted. In studying resistive wall modes, near the stability limit it was observed that non-axisymmetric error fields were strongly amplified (Wang et al., 2015). Furthermore, in experiments designed to suppress ELMs, suppression was found but not by the mechanism expected. The mechanism involves the plasma response but is still not fully explained. One robust indicator of ELM controllability appears to be the edge-peeling response of the plasma to the applied 3D fields (Liu et al., 2016), valid for both mitigation (Liu, 2011; Ryan et al., 2015; Li et al., 2016) and suppression (Paz-Soldan et al., 2015; Yang et al., 2016).

When an external field is applied, the most obvious response on a particle level is that, just as in any conductor, currents flow in the plasma to oppose the changes by excluding the field from penetrating the plasma. Thus, the first response is for skin currents to

flow in the edge. These currents are singular on closed rational field lines that resonate with a harmonic of the perturbed normal field: when $q = m/n$, for a field harmonic $\delta B_{n, m}(r)e^{i(m\theta - nq\phi)}$, with a delta function radial dependence of the current density. However, finite resistivity in the plasma implies these currents decay and, unless regenerated, the field slowly penetrates. Effective regeneration of the currents can occur from plasma rotation. In pioneering work, [Fitzpatrick and Hender \(1991\)](#), [Fitzpatrick \(1993, 1995\)](#) demonstrated that non-ideal effects result in a complex nonlinear interaction between the rotation and the perturbing 3D field where the field slowly removes and dissipates rotational energy as it penetrates, slowing the rotation further, and enhancing the penetration. The result is a bifurcation in the rotation, with two dynamically stable states, one at high rotation with little penetration and the other at low rotation, essentially locked in the laboratory frame, with a fully penetrated field. In between is an inaccessible, dynamically unstable state. The net result of this is that by applying an external 3D field, the plasma rotation subsequently slows and locks to the wall. The plasma response at first expels the external field but subsequently incorporates it in a locked stationary state. The response is a nonlinear dynamic response with suppression occurring first, followed by a slow penetration depending on the resistivity and the plasma rotation.

The second major component of the plasma response is generally an amplification. This was first noted in numerical calculations of the linear MHD response where the expected suppression of specific field harmonics of the normal field component was observed at the respective rational surfaces but became finite again inside. This is attributed to coupling of the poloidal harmonics: non-resonant harmonics penetrate beyond the rational surface and drive the suppressed component back up, amplifying it.

These two parts of the MHD response are, in fact, inseparable. The total linearized response can be expressed as an expansion in the eigenfunctions of the linear eigenmodes of the ideal MHD operator $\partial^2 \xi / \partial t^2 - \mathcal{L}\xi = 0$, so that $\xi = \sum_{k=1}^{\infty} a_k \xi_k$, where $\mathcal{L}\xi_k = \lambda_k \xi_k = -\omega_k^2 \xi_k$, with λ_k being the eigenvalue and ω_k the frequency. The eigenmodes ξ_k inherently include the suppression of the resonant field harmonics at the rational surfaces and amplification in their structure. For ideal MHD, the eigenmodes of $\mathcal{L}$ form a complete orthonormal basis, so any possible ideal response can be expressed completely in the expansion. The non-ideal response, however, can have contributions not in the basis corresponding to tearing solutions.

By applying an external RF field with different frequencies, the ideal MHD eigenmodes can be probed by measuring the response. For this case, by substituting the expansion for the response ξ in terms of the ξ_k into the dynamic equation of motion for a driven perturbation, $\partial^2 \xi / \partial t^2 - \mathcal{L}\xi = A_0 e^{i\omega_0 t}$, one finds

$$\xi(r, t) = \sum_{k=1}^{\infty} [a_k / (\omega_k^2 - \omega_0^2)] \xi_k(r) \quad (62a)$$

$$a_k \equiv \int \xi(r, t) A_0 e^{i\omega_0 t} d^3 r \quad (62b)$$

The response has a resonant denominator proportional to $\omega_k^2 - \omega_0^2$ that vanishes whenever the driving frequency ω_0 coincides with a characteristic frequency of the ideal system. By measuring an aspect of the response as the frequency is varied, a series of peaks appears at these eigenfrequencies. It can also be shown that the width in frequency of the peaks is inversely proportional to the (non-ideal) damping rate. This procedure, dubbed “MHD spectroscopy” ([Reimerdes et al., 2004](#)) has been used in experiments to probe both the stable Alfvén spectrum, identifying TAE and other Alfvén eigenmodes, as well as marginally stabilized kink modes. The technique seems likely to become more prominent in the future.

The transport response follows from the changes in topology, edge conditions, and profiles due to the initial MHD response. With no topological changes, the local transport coefficients can be different as a result of changes in local profiles. When islands form, the transport is completely changed, not only from the island itself, but also because island formation necessitates formation of an associated local chaotic region. In the chaotic regions, parallel transport becomes important and competitive with perpendicular transport. Finally, when the field lines open and leave the plasma region, the global transport becomes a complex mixture of parallel and perpendicular transport, with various chaotic structures playing important roles.

There are several characteristic transport effects that are typically observed. Foremost of these is the observed rotation drag discussed above. The rotation screens the applied fields from penetrating the plasma, essentially by regenerating the screening currents formed at rational surfaces. However, the fields themselves provide a back effect by slowing the rotation. As the rotation slows, the fields are able to penetrate more easily and finally the rotation collapses to a new bifurcated locked state with fully penetrated field and decayed singular currents. The second most important observed effect is the “pump-out” effect. The root cause of this is the formation of open field lines connecting the edge plasma layers to the exterior vacuum and material structures. Along with this pump-out effect is a generally observed reduction in core impurities.

Numerical tools

Many numerical tools have been developed for modeling MHD instabilities in fusion devices, and these tools can be classified in many different ways. We will pursue two ways of classifications here for a list of well-known MHD codes, one is based on whether the code solves linear or non-linear MHD equations, and the other is based on which physics are included into the codes. We emphasize that what we provide below is not a full list of MHD codes that have been developed and used by the MHD community.

Many codes have been developed during the last 40 years, that are capable of solving linear MHD equations in full toroidal geometry, and are well validated against experiments. The linear stability problem can always be solved as an eigenvalue problem.

These codes are therefore mostly written as eigenvalue solvers, including PEST-II (Grimm et al., 1983), ERATO (Gruber et al., 1981), GATO (Bernard et al., 1981), NOVA (Cheng and Chance, 1987), MARS (Bondeson et al., 1992), DCON (Glasser, 1997), KINX (Degtyarev et al., 1997), MISHKA (Mikhailovskii et al., 1997), MARG2D (Tokuda and Watanabe, 1999), MARS-F (Liu and Bondeson, 2000), ELITE (Wilson et al., 2002; Snyder et al., 2007), AEGIS (Zheng and Kotschenreuther, 2006), and MINERVA (Aiba et al., 2009). Some of the codes were also developed into new versions with significant inclusion of drift kinetic effects, such as NOVA-K (Cheng, 1992), MARS-K (Liu et al., 2008), AEGIS-K (Zheng et al., 2010). The non-linear MHD codes include NIMROD (Glasser et al., 1999; Sovinec et al., 2004), M3D (Park et al., 1999), M3D-C1 (Ferraro and Jardin, 2009), JOEKE (Huysmans and Czarny, 2007; Pamela et al., 2020), BOUT (Xu and Cohen, 1998), BOUT++ (Dudson et al., 2009). The MARS-Q code actually solves the quasi-linear MHD equations by including the nonlinear interaction of the mode and the rotation (Liu et al., 2013).

In terms of physics, several early codes solve ideal, single-fluid MHD equations, such as ERATO, GATO, NOVA, and DCON. These are all linear codes. Because the ideal MHD energy principle has the important Hermitian property, most of these ideal MHD codes are based on the ideal MHD energy principle. Exceptions are the AEGIS code, which is based on the shooting method, and the NOVA code. There are also linear MHD codes that solve resistive MHD equations. These are typically based on the normal mode eigenvalue approach, such as the MARS code. Codes that solve the extended MHD equations (including two-fluid) are usually non-linear. These include M3D-C1, NIMROD, JOEKE, and BOUT++. Codes that solve the hybrid MHD-kinetic equations include NOVA-K, M3D-K, MARS-K, AEGIS-K, and NIMROD (a version with kinetic treatment of energetic particles).

Principles of control for MHD instabilities

Fusion gain can be expressed in terms of the fusion power P_F and input power P_I as $Q = P_F/P_I \sim \tau_E \beta^* B^2 \sim \tau_E p_f \beta_T B^2$, where τ_E is the energy confinement time and the pressure profile peaking factor p_f is defined as $p_f^2 \equiv \langle p^2 \rangle / \langle p \rangle^2$, and $\beta^* \equiv p_f \beta_T$. Thus, Q can be increased by increasing β_T and keeping all other factors constant. But, from the Troyon limit equation, Eqs. (41) and (42), β_T is limited by stability. Hence, one needs to optimize $\beta_T \lesssim \beta_T^{crit} = k_\beta \mathcal{L}_i (I_p / a B_T)$ against all the important instabilities. A useful view can be reached by then rewriting the maximum fusion gain as (Lazarus et al., 1997)

$$Q = [R_0^2 B_T^2] \times \left[p_f k_\beta \mathcal{L}_i (S^2 / \kappa) (H^2 / q_s^2) \right] \quad (63)$$

Here, H is a confinement enhancement factor relative to H-mode ($1 \leq H \leq 2$), and q_s is the boundary q , or for a divertor plasma, q_{95} . $R_0 = (R_{max} + R_{min})/2$ is the average of the maximum and minimum of the major radius of the plasma boundary. The plasma shaping enters here directly in terms of the shaping factor $S = q_s (I_p / a B_T)$ and the elongation κ . The current profile dependence appears directly through the internal inductance parameter $\mathcal{L}_i$.

The first factor enclosed in square brackets is a technological factor and a reactor has a direct cost roughly proportional to this factor. The rest contains quantities directly related to plasma physics. All but the confinement factor H are stability related and can be increased by optimization with respect to pressure peaking, cross section shape, and safety factor. The improvement in stability against pressure driven modes from cross section shaping is embodied largely in the factor S^2 / κ . Q can be increased by simply increasing this factor, thereby increasing $I_p / (a B_T)$ and hence β_T . This reflects the fact that $n > 0$ stability is improved by increasing S . However, S^2 / κ is limited by axisymmetric stability.

Optimization from profiles is embodied in the factors $p_f k_\beta \mathcal{L}_i$ and $1/q_s^2$. The pressure profile peaking factor p_f is limited by $n > 0$ stability. There is also a synergistic dependence of the β_T limit on cross section and profiles. For high S , the limiting stable k_β has an inverse dependence on p_f and the product is generally optimized by low p_f , whereas at low S , the dependence of k_β on p_f is weaker. The current profile factor $\mathcal{L}_i$ is limited by $n = 0$ stability. Stability to current driven modes provides a limit on how low q_s can be.

Advanced Tokamak operation

By optimizing against the second factor in the expression for Q above, essentially maximizing the β_T limit, β_T^{crit} , a smaller, more compact tokamak with reduced major radius R_0 and B_T can match the performance of larger, high field machines. Ultimately, this requires the stabilization of instabilities. The Advanced Tokamak (AT) concept follows this route. Control of the profiles pushes the discharge into a state with higher stability limits and maintains it there. Therefore, most AT approaches require some level of active feedback.

Control requires three key elements, namely sensors to detect deviations in the system state from the desired values, actuators to reposition the system toward the desired state, and control algorithms to convert the sensor signals to actuator commands. In the fusion plasma context, the sensors are diagnostic techniques for measuring the required aspects of the plasma state. The actuators are the means of affecting the plasma, for example, external heating, current drive and fueling systems, as well as external fields used to reconfigure the equilibrium or induce a particular plasma response. The control algorithms are essentially mathematical prescriptions, usually based on simplified (reduced) models for determining the actuator commands from diagnostic measurements.

In an AT, control of the equilibrium profiles is an essential element. This includes startup of the discharge, a steady state period, and a final shutdown. In the startup and shutdown phases, the major issue in practice is maintaining MHD stability of the plasma. In the steady state phase, the major issue is to optimize the profiles for high performance, while still avoiding MHD unstable states. Evolutionary paths to this state need to be stable at each point. Active feedback control of incipient instabilities is also essential. Active

feedback stabilization of the axisymmetric (vertical) instability is routine. Typically, the plasma position (for example the current centroid) or a number of points on the plasma boundary, the isoflux control points, are found and, if displaced from its target position or target boundary points, additional axisymmetric fields are used to reposition it (Ferron et al., 1998; Huang et al., 2020) by programming the currents in the external poloidal-field coils. Active stabilization of the non-axisymmetric RWM is similar in principal but requires a more complex set of sensors and applied field distribution (Liu et al., 2000, 2004). In practice, as discussed, sufficient plasma rotation passively stabilizes the RWM but the marginal RWM amplifies error fields which then slow the rotation, leading to destabilization of the RWM. In a reactor, active feedback stabilization of the RWM is likely required. Control of intermittent instabilities such as sawteeth and ELMs typically take the approach of either mitigating their effects or avoiding them through control of the equilibrium profiles. The other major control issue is the need to avoid and mitigate disruptions, which will be discussed in the next section.

MHD theory provides the basis for defining the algorithms used to relate diagnostic sensor measurements to actuator controls. Typically, the algorithms use simplified models but these are often derived from full MHD predictions and are generally tested against them.

Control of MHD instabilities

Sawteeth result in a transfer of the core energy inside the $q = 1$ surface to the outside. They are normally relatively benign but if the $q = 1$ radius is large and the core pressure high, an abnormally large amount of energy can be lost. This can happen in cases where the sawtooth period is very long, the so-called giant sawteeth, allowing a large amount of energy to build up in the core, followed by a large and fast crash that disturbs a large part of the cross section. These need to be avoided. Note that, like ELMs, sawteeth can be beneficial by periodically expelling impurities from the fusing core. Thus, it is advantageous in many cases to control the sawtooth rate, keeping it short. Control options that have been considered involve controlling the radius of the $q = 1$ surface. Control using RF waves, particularly Electron and Ion Cyclotron Current Drive (ECCD and ICCD) has been considered to modify the current and pressure profiles around the $q = 1$ surface and tried. Another option is to run in sawtooth free regimes. This simply means controlling the profiles so that q remains well above one, avoiding completely the 1/1 internal mode (ideal or resistive). This is the solution envisaged in the Advanced Tokamak concept.

TMs born rotating with frequency near local plasma rotation and saturate at some amplitude. But these islands slow the plasma rotation (Fitzpatrick and Hender, 1991) and the plasma and mode lock. At that point, the mode generally grows rapidly as wall stabilization is lost, and disrupts the plasma. The mode typically locks in a fixed phase relationship with a pre-existing error field. Threshold scalings have been obtained from experiments and can be used to avoid low and high-density locked modes. The thresholds also depend on the pre-existing error field. These modes can be partially controlled by maintaining the plasma rotation against the natural slowing that results from the incipient instability. This entails feedback on the momentum input to the plasma.

Control of NTMs has been demonstrated (La Haye et al., 2002) using ECCD to replace the reduced bootstrap current that results from the reduced pressure gradient inside the islands. A control scheme called “search and suppress” has been developed whereby the island is detected and ECCD is directed using mirrors to locally be deposited in the island center. An alternative strategy under development is to modify the local current density using ECCD, essentially changing the local Δ' . This can also be modified by applying additional helical fields. For tearing modes that are destabilized by proximity to the ideal MHD β limit (Brennan et al., 2002), the modes are linearly unstable and Δ' provides the major drive and this is the most promising strategy.

Control of the RWM follows the same strategy as the routine stabilization of the axisymmetric mode. For that case, the technique has been in use routinely since the 1970s and refined since the 1990s (Lazarus et al., 1990; Ferron et al., 1998; Liu et al., 2000; Strait, 2015). The ideal instability is slowed by a resistive wall from a growth time of the order of a plasma Alfvén time $\tau_A \sim 10^{-6}$ sec to the wall L/R time—the time scale for stabilizing image currents to decay. The fields are then sensed and growing fields suppressed by applying an opposing field. In the non-axisymmetric case, the major complication is the need for more sensors and actuators.

ELM control falls into four major categories usually referred to as ELM-stable, ELM-free, ELM mitigation, or ELM suppression. The first refers to plasma configurations such as QH mode or I-mode in which there are no ELMs. Control in this case is mostly concerned with maintaining the plasma operating conditions so that it stays in QH or I-mode. The second refers to plasma configurations for which the peeling-ballooning instability is suppressed for a long period. Typically, however, these are not stationary and an ELM finally appears that is larger than usual, similar to the situation with giant sawteeth. Control in this case appears to require control of the equilibrium profiles to avoid the final instability. ELM mitigation means changing conditions to replace the large Type-I ELMs by smaller, more frequent ELMs, either small Type I or Type II or Type III. Suppression, in contrast, refers to removal by full stabilization of the Type-I ELMs. ELM-free regimes however, suffer from the problem that, without ELMs, impurities and density build up in the core. ELMs do have a beneficial effect of removing core impurities, or helium ash in the case of a reactor. Mitigation solves this problem. Control of the ELM frequency in this case can be done using ELM pacing, with triggering by pellets (Lang et al., 2004a; Baylor et al., 2013), vertical kicks (Lang et al., 2004b), applied non-axisymmetric fields (Canik et al., 2010), or applied modulated non-axisymmetric fields (Solomon et al., 2012). On the other hand, ELM suppression regimes do not result in an impurity buildup. These are obtained for certain plasma parameter regimes by applying non-axisymmetric fields (Evans et al., 2004).

Disruption physics, prediction, prevention, and mitigation

A critical issue in tokamak experiments is the occurrence of plasma disruptions, where the discharges abruptly terminate with many undesirable consequences (ITER Physics Basis Editors, 1999b; Hender et al., 2007). The causes of disruption are complicated and are not always fully understood. These can be grossly divided into MHD and non-MHD causes. We will mainly discuss the MHD causes here. Many non-MHD causes are often directly or indirectly related to MHD events, e.g., the presence of MHD precursors, the presence of large magnetic islands in the radiative cooling model that explains the plasma density limit induced disruption (Gates and Delgado-Aparicio, 2012).

As described before, there are several macroscopic MHD instabilities that can cause plasma disruption. One is the external kink mode including the resistive wall mode. The current-driven external kink becomes very dangerous when the edge safety factor (q_s for limiter plasmas and q_{95} for divertor plasmas) approaches the value of 2. The pressure-driven external kink can cause disruption when the Troyon limit is exceeded. The other important MHD cause is mode locking, which typically occurs when the 2/1 tearing mode becomes unstable and large magnetic islands are created which (at least locally) break the plasma toroidal rotation.

A typical plasma disruption has two characteristic time scales, related to the thermal quench (TQ) and current quench (CQ). The first process, where the plasma thermal energy is rapidly lost, typically lasts a couple of milliseconds. This is a very fast process, likely involving many non-linear MHD events. The TQ is followed by the much longer CQ phase, where the plasma current resistively decays and eventually vanishes (together with the magnetic energy stored in the plasma). The time scale of the CQ depends on tokamak devices and in particular on the poloidal cross-section area (ITER Physics Basis Editors, 1999b; Hender et al., 2007).

There are two important consequences associated with plasma disruption. One is the electromagnetic force acting on the conducting structures surrounding the plasma, typically the vacuum vessel and the supporting structures. This force can be large during disruption and can potentially damage the PFCs and in-vessel structures. The other critical consequence is the generation of high-energy electrons with speed close to the light speed. These so-called runaway electrons can potentially melt the PFCs, and thus must be controlled in future reactor scale devices such as ITER.

Disruption prediction and avoidance

As mentioned before, since disruptions may have different origins, predicting them is generally challenging. If a disruption is caused by MHD event, there are often magnetic precursors that can be employed, but the reliability is always an issue. Since the late 1990s, methods based on artificial intelligence (AI) have been used for the purpose of disruption prediction in tokamak experiments (Wroblewski et al., 1997), with significant progress being made during recent years (Kates-Harbeck et al., 2019; Zhu et al., 2021). This approach requires a large operation database to train the AI algorithm (e.g., the neural network), before it can be used for predicting future experiments. Proper selection of the equilibrium input data, besides the AI architecture and the training algorithm, is critical for the success of the AI based approaches. For predicting specific classes of disruption, model-based approaches are also highly valuable. Examples are the DECAF model (Berkey et al., 2017) based on various MHD events, and the recent initiative of the predict-first approach (Lyons et al., 2018).

Disruption avoidance is a passive way of operating the plasma discharge. Based on a-priori knowledge, certain types of disruptions, in particular those involving MHD instabilities, are predictable. This knowledge can be used to constrain the discharge parameters to avoid entering into the dangerous range. For instance, the plasma pressure can be limited to be below the Troyon limit (often following the $\mathcal{L}_i$ scaling in Eq. (42)) during the operation. The plasma current and toroidal field can be designed to avoid the edge safety factor reaching below 2. By various methods, the toroidal rotation speed can be maintained at a certain level to avoid locked modes. With both disruption prediction and avoidance, it is important to realize that accurate and fast equilibrium reconstruction is a critical element.

Disruption mitigation

As discussed, development of effective strategies to mitigate and control tokamak disruption is critical to reduce and avoid potential damage to the PFCs and in-vessel structures. All mitigation methods rely on rapid injection of high charge number Z impurities into the plasma to radiate away most of the plasma thermal energy. MHD provides a useful model to develop and test techniques to mitigate and control MHD instabilities.

A leading candidate for a disruption mitigation system (DMS) is the shattered pellet injection (SPI) approach in which a stream of cryogenic cooled pellets is injected into a bended tube and shattered into small fragments before entering into the plasma to allow higher assimilation and more rapid delivery of injected impurities than a gas injection system (Shiraki et al., 2016; Baylor et al., 2019). Another promising technique is the dispersive shell-pellet injection (DSPI) method in which a low- Z hollow shell filled with a dispersive payload is injected into the plasma to allow a deep penetration of impurities into the plasma core and a more effective inside-out thermal quench (Izzo and Parks, 2017; Hollmann et al., 2019).

An accurate disruption mitigation simulation requires integration of a global 3D MHD code with a local pellet ablation code (Kim et al., 2019; Bosviel et al., 2021). Progress has been made to model disruption mitigation by impurity injection and interpretation of SPI and DSPI experiments with the NIMROD (Kim et al., 2019; Izzo et al., 2020), M3D-C1 (Lyons et al., 2019; Ferraro et al., 2019) and JOEREK (Hu et al., 2018; Pamela et al., 2020) 3D MHD codes using simplified reduced pellet ablation models. NIMROD predictions of DIII-D SPI and DSPI results are consistent with DIII-D experimental observations. An animation of a NIMROD simulation of DIII-D dual-injector SPI simulation is illustrated in Fig. 6 (Kim et al., 2020). However, significant

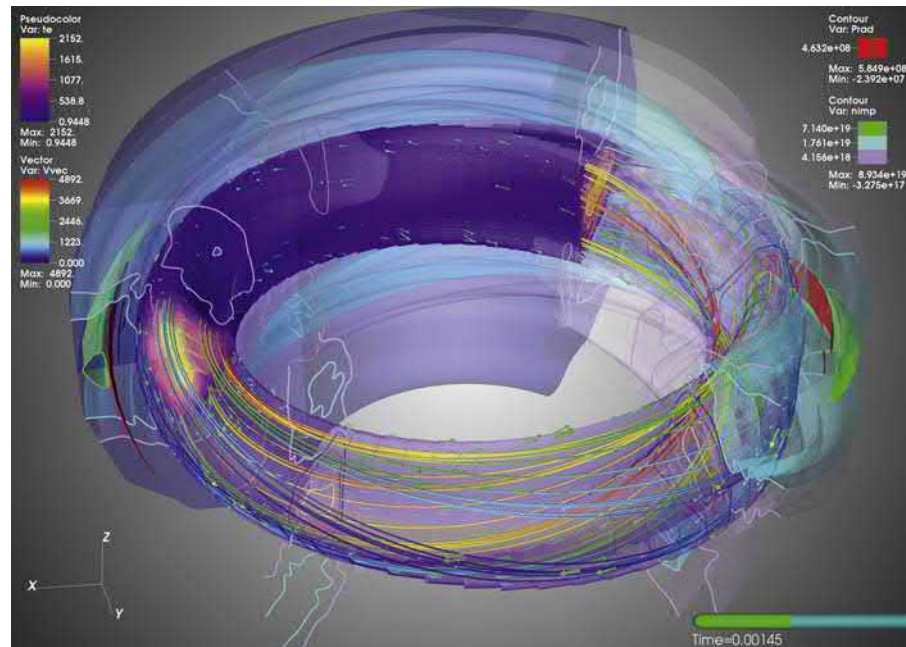


Fig. 6 NIMROD animation of a DIII-D dual-injector SPI simulation showing magnetic field lines and contours of plasma radiation and injected impurity concentration. Courtesy of Kim CC, et al. (2020) *Simulations and Validation of Disruption Mitigation and Projections to ITER's Disruption Mitigation System*. Bulletin of the American Physical Society C007.00015. <https://meetings.aps.org/Meeting/DPP20/Session/C007.15>.

challenges remain to develop a robust and effective disruption mitigation technique and their simulations and validations that can address all the disruption issues, particularly regarding the generation and mitigation of runaway electrons (Boozer, 2017; Lehnert et al., 2020; Sweeney et al., 2020).

Summary

MHD provides a useful model to describe the crucial plasma macroscopic equilibrium and stability behaviors in toroidal tokamak devices. The MHD equations provide a set of comprehensive physics constraints to understand and interpret tokamak macroscopic instabilities. Principal MHD instabilities include the internal kink modes, sawtooth, fishbone, external kink, resistive wall mode (RWM), resistive interchange, tearing and neoclassical tearing modes (NTMs), locked modes, toroidal Alfvén eigenmodes (TAEs), and edge localized modes (ELMs). Predictions from MHD theory are consistent with many observed features of these instabilities. Fast-growing MHD instabilities can lead to an abrupt plasma disruption and termination that can potentially damage the device PFCs and in-vessel structures. An important MHD application is to develop robust techniques to mitigate and control instabilities. These include control of NTMs and RWMs, mitigation and suppression of ELMs, and disruption avoidance and mitigation.

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Magnetic Confinement Fusion—Plasma Theory: Energetic Particle Physics

N.N. Gorelenkov^a and S.E. Sharapov^b, ^a Princeton Plasma Physics Laboratory, Princeton, NJ, United States; and ^b CCFE, Culham Science Centre, Abingdon, United Kingdom

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Glossary

Advanced Tokamak scenario Has been developed for tokamak continuous operation in a fusion power plant and is characterized by reversed magnetic shear, or q-profile radial dependence, in the central plasma region.

Boozer coordinates A set of magnetic coordinates often used in fusion simulations to simulate EP drift motion ([Boozer, 2004](#)). Their use is beneficial for the Hamiltonian formulation of charged ion orbit drift motion.

Cyclotron instabilities Instabilities which have a frequency comparable with the characteristic frequency of the cyclotron gyro rotation in the magnetic field of the plasma.

Electron Coulomb logarithm It is a factor by which the small-angle electron-plasma collisions are more effective than large-angle collisions. For many plasmas of interest it takes on values between 5 and 10. (For convenient formulas, see pages 34 and 35 of the [NRL Plasma formulary](#).)

Geodesic Acoustic Modes Low frequency oscillations strongly interacting with thermal ions (see [Jenko, 2020](#)).

Krook collision operator A simplified way to include the relaxation of a plasma ion distribution function in the Boltzmann kinetic equation.

Reversed shear plasma Plasma characterized by $\frac{dq(r)}{dr} < 0$ in the central region of the plasma.

Nomenclature

β_N Normalized plasma pressure, $= \% \beta / (I/aB_\phi)$

λ_e Electron Coulomb logarithm

BAE Beta-induced Alfvén eigenmode

CAE Compressional Alfvén eigenmode

EAE Ellipticity-induced Alfvén eigenmode

GAE Global Alfvén eigenmode
 GAM Geodesic acoustic modes
 ICRH Ion cyclotron radiofrequency heating (see [Dumont, 2021](#))
 LHD Large helical device (see [Yamada, 2021](#))
 NPA Neutral particle analyzer
 RSAE Reversed shear Alfvén eigenmode
 TAE Toroidicity-induced Alfvén eigenmode
 WDM Whole device modeling (see [Jenko, 2021](#))

Introduction

We consider energetic particles as a superthermal ion population which is required to sustain the burning conditions in the presence of inevitable energy losses from the plasma. As such EP includes (1) fusion products, primarily alphas particles, from deuterium-tritium fusion reaction, (2) energetic beam ions such as planned 1 MeV deuterium ions in ITER reactor, and (3) H-minority ions accelerated by ion cyclotron radiofrequency heating (ICRH).

This article is written with the aim of discussing what might be considered traditional aspects of EP physics, including Alfvénic instabilities, by developing the underlying physics understanding of these topics. We start this article with the description of a single particle confinement in section “**Single particle confinement**.” Then we present the basic elements of wave-particle interaction including the Alfvénic eigenmodes and low frequency, Alfvén acoustic modes in section “**Wave particle interactions**,” which covers the linear theory of Alfvénic eigenmodes. It gives the EP physics a strong base for applications and projections to burning plasmas. Nonlinear elements of EP physics, covered in section “**Nonlinear evolution of EP-driven modes and energetic particle transport**,” are currently investigated by many groups in the world and considerable progress has been made. Their development is becoming critical for making quantitative predictions, as discussed in section “**Implications for burning plasmas**.”

For further reading we recommend the following review papers on several aspects of EP physics which are covered in great details in topical review articles ([Heidbrink, 2008](#); [Cheng, 1992](#); [Kolesnichenko et al., 2011](#); [Toi et al., 2011](#); [Breizman and Sharapov, 2011](#); [Lauber, 2013](#)) as well as in comprehensive reviews ([Heidbrink and Sadler, 1994](#); [Gorelenkov et al., 2014](#); [Chen and Zonca, 2016](#)). These are natural extensions of earlier reviews ([Mikhailovskii, 1975](#); [Kolesnichenko, 1980](#)).

We consider EP physics to be a key element of the research toward making fusion a feasible component of sustainable energy sources. For the purpose of this encyclopedia, the article follows main elements of a recent review ([Gorelenkov et al., 2014](#)) which is recommended for further reading and references.

Single particle confinement

Drift motion. EP moment conservation in axisymmetric and in 3D plasmas

An energetic ion in the presence of a strong magnetic field, B , moves along the field line with parallel speed, $v_{\parallel}$, while rotating in the perpendicular direction around the center of the gyro-orbit with Larmor radius, $\rho_f = v_{\perp}/\omega_c$, where $v_{\perp}$ is the perpendicular speed. At the same time, the ion slowly drifts in the direction perpendicular to vector, B , with the drift velocity, $v_d = \frac{B}{B\omega_c} \times [(2\varepsilon_{\parallel}\kappa + (\varepsilon_{\perp}/B)\nabla B)]$, where $\omega_c = eB/mc$ is the cyclotron frequency, $\varepsilon_{\parallel}$ and $\varepsilon_{\perp}$ are the parallel and perpendicular component of particle energy variable given below at Eq. (1), e and m are the ion charge and the mass, and c is the speed of the light.

On a longer timescale EPs slow down by interacting with thermal ions and electrons with the characteristic slowing down time determined primarily by electrons $\tau_{se}[\text{sec}] \approx 0.01(10/\lambda_e)(m_i/m_p z_i^2) \frac{(T_e[\text{eV}])^{3/2}}{n_e} [10^{20} \text{ cm}^{-3}]$, where fast ion is denoted as a specie i , λ_e is the Coulomb logarithm of electrons, T_e and n_e are the temperature and the density of thermal electrons. The EP distribution function is formed as described below in section “**Slowing down distribution function of fast ions in quiescent plasmas**.”

The EP drift motion is described by three constants of motion: energy, $\mathcal{E}$, adiabatic moment, μ , and canonical angular momentum, P_{φ} :

$$\mathcal{E} = v^2/2; \mu = \varepsilon_{\perp}/B; P_{\varphi} = v_{\varphi} c R m / e - \psi_p, \quad (1)$$

where ψ_p is the poloidal flux function of the magnetic surface, subscript φ refers to the toroidal angle component.

The conservation of μ and P_{φ} is well-established in tokamak geometry.

In spherical tokamaks (or STs) μ and P_{φ} are conserved, although, as given by Eq. (1), they oscillate in time ([Belova et al., 2003](#)). For test particle simulations, it is often sufficient to know that there are expressions for μ and P_{φ} that are conserved, but the explicit expressions often are not required. More accurate expressions are required for the problems where the phase information of fast ion drift motion is needed, such as for cyclotron instabilities.

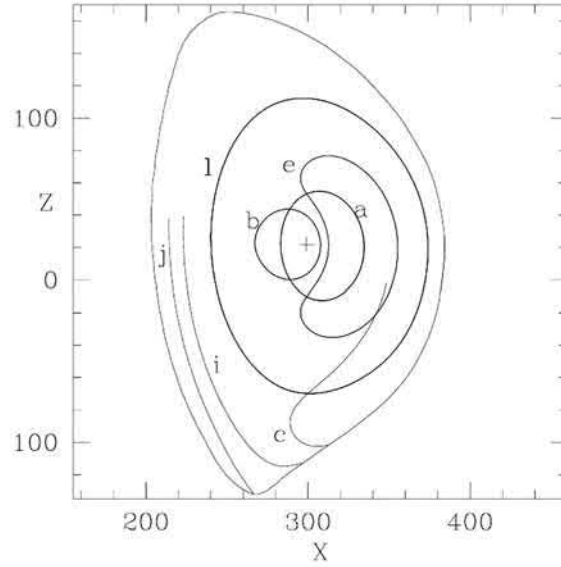


Fig. 1 Energetic proton or alpha particle guiding center trajectories in JET tokamak at energy 9.54 MeV and magnetic field 3.5 T. Different types of orbits shown in the figure are characteristic for the drift motion of the charged ions including thermal ions. They have different drift frequencies which are responsible for different wave-particle interactions.

The toroidal angular momentum as describe by Eq. (1) is, strictly speaking, not an adiabatic invariant in 3D configurations due to the lack of toroidal symmetry in the magnetic field (see also Galvão and Canal, 2020; Yamada, 2021). Nevertheless, an adiabatic invariant exists for a wide class of three dimensional geometries such as stellarator/helical devices where the rotational transform per toroidal field period is small, $(1/2\pi)/N \ll 1$, and the strength of the field does not change much due to toroidicity as compared to the variation along the field line due to the helical ripple, i.e., $\varepsilon_\phi(1/2\pi)/(\varepsilon_h N) \ll 1$ (Cary et al., 1988). Here we denote the rotational transform (or field line pitch) as $1/2\pi$, which defines the number of poloidal transits per single toroidal transit of a field line on a toroidal flux surface, N is the toroidal field period number, ε_ϕ and ε_h are the quantities which represent the relative magnetic field variation due to toroidicity and relative magnetic field ripple due to the helical coils, respectively (Cary et al., 1988).

In such 3D devices, two classes of particles exist. Particles of one class are trapped in local magnetic field ripple wells and others are untrapped. The former, trapped, particles have a bounce invariant or a modified second adiabatic invariant J^* and the latter (passing) also have an invariant. For passing particles with small pitch angle the invariant becomes equivalent to the toroidal angular momentum P_ϕ . The J^* conservation was successfully applied to the study of particle orbits in the ATF device that satisfied both requirements: $(1/2\pi)/N \ll 1$ and $\varepsilon_\phi(1/2\pi)/(\varepsilon_h N) \ll 1$ (Rome, 1995). In the LHD configuration with large $N = 10$, the toroidal angular momentum is conserved for passing EPs. The requirements defined above are not satisfied for quasi-axisymmetric helical devices with small $N = 2$ or 3. We note that the above requirements are satisfied even near the plasma edge and therefore the P_ϕ is approximately conserved from the plasma core to the edge.

Fig. 1 presents several guiding center orbits in the JET tokamak (White, 2006). The considered case corresponds to the plasma equilibrium with a major radius of 299 cm and EP gyroradius 9 cm, i.e., a fast proton or alpha particle having an energy of 9.54 MeV. Several types of orbits are shown: a,b,l are passing particle orbits, e,c are trapped particle orbits whereas c is the orbit of a lost particle. A more detailed discussion of particle orbits can be found in Sarazin (2021). Characteristic EP parameters in various tokamaks and in ITER are given in Table 1.

Ripple and other axisymmetry breaking effects on EP confinement

The confining magnetic field in conventional tokamaks is created by the finite number of the toroidal field coils (Galvão and Canal, 2020) in such a way that the equilibrium magnetic field is changing along the field line, i.e., in other words, it is modulated by poloidal and toroidal angles, ϑ and φ respectively, $B = B_0[1 - (r/R) \cos \vartheta + \delta \cos N\varphi]$, where $\delta = (B_{\max} - B_{\min})/(B_{\max} + B_{\min})$ is typically on the order of 1% on the low field side of the tokamak. It becomes negligibly small at the magnetic axis.

At that level the ripple losses are small, resulting in a few percent of α -particle losses in ITER (Fasoli et al., 2007). One of the consequences of the equilibrium rippled magnetic field is the dependence of its absolute value along the toroidal direction. The rate at which the trapped particle drifts out of the plasma is fast, $v_d = R\omega_c(\rho_f/R)^2$, which results from the direct drift velocity in the tokamak equilibrium field.

There is a critical value for the ripple magnitude above which an EP ion becomes unconfined. For the case of low β , cylindrical equilibrium and at a bounce point of the EP trajectory, $\vartheta_b = \pi/2$:

Table 1 Energetic ion parameters for various tokamak heating systems.

Parameter Machine	$P_t(0)$ [MW/m ³]	$n_t(0)/n_e(0)$ [%]	$\beta_t(0)$ [%]	$\langle\beta_t\rangle$ [%]	Max $ R \nabla\beta_t $	$V_t/V_A(0)$
NBI/TFTR ^a	3	13	0.9	0.4	0.04	0.35
ICRH/JET ^b	1–3	1–10	1–3	0.5	≈ 0.1	≈ 1 –2
α -particles TFTR ^a	0.3	0.3	0.26	0.03	0.02	1.6
α -particles JET ^c	0.12	0.44	0.7	0.12	0.035	1.6
α -particles ITER ^d	0.52	0.68	1.14	0.17	0.074	1.73
NBI/ITER ^e	0.04	0.23	0.14	0.054	0.02	1.3

^aParameters achieved in TFTR DT discharge #76770 with 40 MW of 100 keV NBI, $B_T = 5$ T.

^bTypical parameters achieved in JET deuterium plasmas with up to 15 MW of ICRH ³He minority heating with ³He tail temperature $\langle E_t \rangle \approx 1$ MeV.

^cAlpha-particle parameters in JET record high fusion power (16.1 MW) H-mode DT discharge #42976 with 22 MW of NBI and 3.1 MW of ICRH, $B_T = 3.6$ T.

^dThe anticipated alpha-particle parameters of ITER with 15 MA current. The simulations carried out using the ASTRA code (Data courtesy of A. Polevoi and S. Pinches, ITER Organization, 2020).

^eParameters expected for D0 beam with energy 1 MeV in the ITER scenario with 15 MA current. The simulations carried out using the ASTRA code (Data courtesy of A. Polevoi and S. Pinches, ITER Organization, 2020).

$$\delta > \delta_{cr} = \left(\frac{\varepsilon}{N\pi q} \right)^{3/2} \frac{1}{2\rho_f q'} \quad (2)$$

The magnitude of a magnetic field ripple, given by Eq. (2), is known as the Goldston-White-Boozer (GWB) critical ripple magnitude, which is often used for a rapid analysis of the ripple strength. Experimentally the ripple diffusion boundaries (see below) and/or the fraction of the losses of charged particles are measured in order to understand the EP classical confinement in the present day devices and to make projection for burning plasmas (Heidbrink and Sadler, 1994; Zweben et al., 2000). An important point in the above measurements is that the particle transport is expected to be *diffusive* in the ripple domain, i.e., the trapped ion motion is described by the diffusion term in the drift kinetic equation (Gorelenkov et al., 1997). The diffusion of the trapped particles was inferred by the PCX diagnostic in TFTR (Medley et al., 1996).

Slowing down distribution function of fast ions in quiescent plasmas

The energetic ion distribution function (DF) is the key integral characterization of all the effects on those particles from various physics phenomena. It accumulates the effects that the EPs encounter into a dependence on energy, pitch angle and canonical momentum, provided these effects influence the EPs within one slowing-down time. Examples of such effects include classical non-collective as well as collective effects due to waves and instabilities.

There are direct measurements of the EP distribution function available, e.g., by using an experimental technique in which energetic ions are neutralized by gaining electrons from a lithium pellet injected into the plasma and immediately leave the plasma as energetic neutrals, being detected at the plasma edge by a Neutral Particle Analyzer (NPA) (Medley et al., 1996).

Examples of such measurements are shown in Fig. 2 for fusion products, alpha-particles, in TFTR DT experiments (Gorelenkov et al., 1997). This illustrates the evolution of the alpha-particle energy distribution function during the alpha slowing down phase from near the birth time of the alpha-particles. One can see that the DF slows from the alpha-particle source near 3.52 MeV, computed at 0.12 s after the start of additional heating, down to the distribution function taken at 1.2 s.

To compute the alpha-particle DF, consider the relaxation of the classically confined EP distribution function given the known source of DT fusion alphas. The drift kinetic equation for the fast ion distribution function (DF) f , leaving only the source and the drag terms, is:

$$\frac{1}{\tau_{se} v^2} \frac{\partial}{\partial v} (v^3 + v_*^3) f - S_f(v) = 0, \quad (3)$$

where v_*^3 is the characteristic ion speed when the slowing down on electrons is equal to the slowing down on ions.

One can solve this equation for a delta-function source in velocity source, $S_f(v) \sim \delta(v - v_{j0})$:

$$F_{se} \approx \frac{3}{2^5 \pi^2} \frac{B^2 \beta_f}{E_f (v^3 + v_*^3)}, \quad (4)$$

which has the slowing down distribution function dependence, where the speed at birth, v_{j0} , corresponds to a birth energy, E_f .

Eq. (4) is a Green function of the inhomogeneous first order differential equation (3), the solution of which can be written through the integral for any source function such as the Gaussian in velocity, $S_f(v) = S_{j0} \int \exp[-(v' - v_{j0})^2 / \Delta_v^2] \delta(v' - v) dv'$. Substituting this and the Green function (4) into the equation (3) we obtain an expression which is the same as (4) if $v < v_{j0}$ and an integral of the EP source function near the birth velocity v_{j0} , which is a complementary error function, $\text{erfc}(x) = 1 - \text{erf}(x)$:

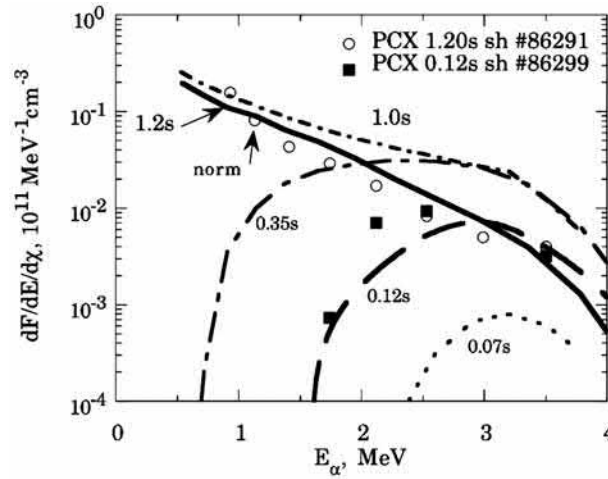


Fig. 2 Evolution of DT fusion alpha energy DF spectra computed using the FPPT code (curves, (Gorelenkov et al., 1997)) and its comparison with measured by the lithium Pellet Charge eXchange (PCX) diagnostic (Medley et al., 1996) alpha spectra. The measurements were performed at two time, 0.12 and 1.2 s after Li pellet injection as indicated.

$$F = F_{se} \operatorname{erfc} \left[(v - v_{f0}) / \sqrt{4\nu} \right] / 2. \quad (5)$$

This has a smooth behavior near the EP birth velocity, which can be applied to either α -particles or neutral beam injected ions near the beam injection velocity.

Wave particle interactions

Fusion-born alpha-particles, as well as the auxiliary heating ions such as the beam ions, in next-step burning plasma experiments will have longitudinal velocities comparable to the Alfvén speed $v_A = B / \sqrt{4\pi m_i n_{pl}}$, where n_{pl} is the plasma density, but well decoupled from both electron and ion thermal velocities of the bulk plasma, $v_{Ti} \ll v_A \sim v_\alpha \ll v_{Te}$. Under these conditions, shear Alfvén perturbations (see section “Collective effects” for more details) could resonate effectively with the fast ion population and release free energy associated with, e.g., the alpha-particle pressure gradient. At the same time, the resonant interaction of shear Alfvén waves with the thermal plasma species is exponentially small, and other damping mechanisms become dominant, such as radiative and continuum damping. In the case of Toroidal Alfvén Eigenmodes (TAEs having their eigenfrequencies in the gaps of the Alfvén continuum (see illustrations in Fig. 4 of such gaps)), the continuum damping of TAEs becomes weak, and the drive from alpha-particles could overcome the other damping mechanisms, leading to TAE excitation. In addition to shear Alfvén waves, the EP population could also excite compressional Alfvén waves, geodesic acoustic modes, and ion sound acoustic waves. In present day experiments, observed oscillations (such as TAEs) can be often characterized by their gradual frequency evolution, which occurs on an equilibrium time scale (Gorelenkov et al., 2014). Different nonlinear scenarios are considered in the next section “Nonlinear evolution of EP-driven modes and energetic particle transport.”

The interaction of EPs with waves and eigenmodes, particularly those having an “Alfvénic” character, is considered to be potentially a significant limitation of the tokamak-reactor performance (Fasoli et al., 2007). We should note that in most practically relevant cases wave-particle interactions (WPI) occur between the resonant beam ions or fusion products and the eigenmodes, or standing solutions, of 3 (generally coupled) fundamental oscillations. Those oscillations include sound (or acoustic), shear and compressional (or magnetosonic) Alfvén waves.

Interaction between waves and particles include an interesting, but, speculative, idea, worth mentioning here, to utilize energetic alpha particles through the so-called alpha-channeling effect (Fisch and Rax, 1992; Herrmann and Fisch, 1997). Alpha channeling diverts the energy from energetic alpha particles to plasma waves in order to avoid that energy simply heating electrons. Rather, that energy can be used to directly heat the fuel ions or can be used to amplify plasma waves that drive electric currents. The alpha channeling could result in a more economical deuterium–tritium tokamak reactor. Simulations and experiments on TFTR support certain separate building blocks that taken together might produce the desired effect (Fisch, 2000).

Collective effects

MHD and kinetic descriptions

Linear and nonlinear analyses of MHD modes have proven to be important for design and operation of fusion devices (Wesson, 1997; see also Lao et al., 2021). The ideal MHD model is based on the set of equations obtained from velocity moments of the

kinetic equations for thermal ions and electrons. These have the form of MHD equations, i.e., the plasma looks like a fluid that can carry the current and can be augmented by infinitely large conductivity, i.e., with vanishing electric field along the magnetic field line, $E_{\parallel} = 0$.

The quadratic form is important to understand the power balance in the wave-particle interactions and in MHD stability in general (Lao et al., 2021). It can be obtained by integrating the equations of motion of plasma particles

$$\delta W + \delta K = 0; \delta K = -\frac{\omega^2}{2} \int \rho (\vec{\xi})^2 dx^3; \delta W = -\frac{1}{2} \int \vec{\xi} \cdot \vec{F}(\vec{\xi}) dx^3. \quad (6)$$

Here δW and δK are the plasma potential and kinetic energies. ρ is the plasma mass density, $\vec{\xi}$ is the plasma displacement vector, $\vec{F}$ is the force acting on the plasma element.

This quadratic form is a useful constraint imposed on plasma oscillations to characterize the stability properties of the system within certain limits by providing the tools to find the real frequency and its imaginary part:

$$\omega_r^2 = 2\delta W / \int \rho \vec{\xi}^2 dx^3, \quad \omega_i / \omega_r = -\Im \delta W / 2\delta K. \quad (7)$$

The real part of δW is found using the force balance equation and can be written in the form

$$\delta W = -\frac{1}{2} \int \left[\frac{1}{c} \vec{\xi}_{\perp}^* \cdot \delta(\vec{J} \times \vec{B}) - \vec{\xi}_{\perp}^* \cdot \nabla \delta p_c \right] d^3x, \quad (8)$$

where the plasma displacement includes both incompressible Alfvén and compressible acoustic parts. Various forms of the potential energy of oscillations exist in the literature (Chen et al., 1984; Bussac et al., 1975; Wesson, 1997).

The kinetic EP contribution, δW_f , relies on the derivation of the EP perturbed distribution function, which is obtained by solving the drift kinetic equation and represents the most general way to find the growth rate in the linear perturbative theory of tokamaks (Porcelli et al., 1994; Gorelenkov et al., 1999). The expressions for δW_f can be written using the constants of motion (COM) variables. Then the EP contribution follows:

$$\delta W_f = -\frac{4m_f^2 \pi^2 c \omega}{e_f} \int dP_{\phi} d\mu d\mathcal{E} \tau_b \sum_{mm'l'} \mathcal{E}^2 F'_{\mathcal{E}} \frac{G_{ml}^* G_{ml} (1 - \omega_*/\omega)}{\omega - n\bar{\omega}_D - l\omega_b}, \quad (9)$$

where the drive is coming from the radial gradient of the EP distribution, $\omega_* = n(F_{p_{\phi}}/F'_{\mathcal{E}}) \sim n d\beta_f/dr$, $G_{ml} \sim v_d^* E$ represents the wave-particle interaction matrixes, and the toroidal mode number is explicitly introduced in the resonant denominator to underline its role in the resonance condition. The denominator in this expression determines the resonances by specifying the poloidal harmonic number and the bounce frequency. The coefficients, G_{ml} , are the harmonics of the EP—wave interactions, the scalar product $\vec{v} \cdot \vec{E}_{\perp}$, gyro averaged over time. Eqs. (9), (7) are standard for the perturbative theory to compute the growth and damping rates of the Alfvénic modes due to the resonances with ions or electrons (Mikhailovskii, 1975).

A knowledge of the resonances in phase space is important to understand whether the excitation of certain AEs may or may not lead to effective particle transport.

Alfvén eigenmodes, TAEs and other global Alfvénic modes

The Alfvén eigenmodes driven by energetic particles are often observed in tokamak plasmas. They belong to primarily one of three fundamental MHD branches, which is the shear Alfvén branch. In MHD theory, in general one can cast the plasma oscillations into a single equation for the plasma displacement vector, $\vec{\xi}$:

$$\frac{\partial^2 \vec{\xi}}{\partial t^2} = v_A^2 \nabla (\nabla \cdot \vec{\xi}) + v_A^2 \nabla_{\perp} (\nabla \cdot \vec{\xi}_{\perp}) + v_A^2 \frac{\partial^2 \vec{\xi}_{\perp}}{\partial z^2}. \quad (10)$$

The RHS of this equation contains three types of waves. One of them is the shear Alfvén oscillations, whose dispersion follows from this equation if the parallel component of the displacement to the magnetic field line and $\nabla \cdot \vec{\xi}_{\perp}$ are vanishing. Then the Alfvénic branch displacement has the form:

$$\frac{\partial^2 \vec{\xi}_{\perp}}{\partial t^2} = v_A^2 \frac{\partial^2 \vec{\xi}_{\perp}}{\partial z^2} \text{ or } \omega^2 = \omega_A^2 = v_A^2 k_{\parallel}^2. \quad (11)$$

Any solution of this equation written in a cylinder can be represented as a superposition of waves propagating along the z axis with the Alfvén velocity v_A . The parallel component of the wavevector for the axisymmetric plasmas in tokamaks is $k_{\parallel} = (m - nq)/qR$. Here m and n are the poloidal and toroidal mode numbers respectively.

In a toroidal plasma, poloidal harmonics are coupled due to the poloidal variation of the equilibrium magnetic field. The coupling is thus between the m and $m + 1$ harmonics when the so-called TAE couplet emerges. The couplet of two harmonics can be a stand-alone eigenmode solution. The coupling occurs at a point where two continua intersect, i.e., $k_{\parallel m} = -k_{\parallel m+1}$ and thus the value of the safety factor at which TAEs develop is given by $q_{TAE} = \frac{m+1/2}{n}$. When TAEs are formed, the Alfvénic m^{th}

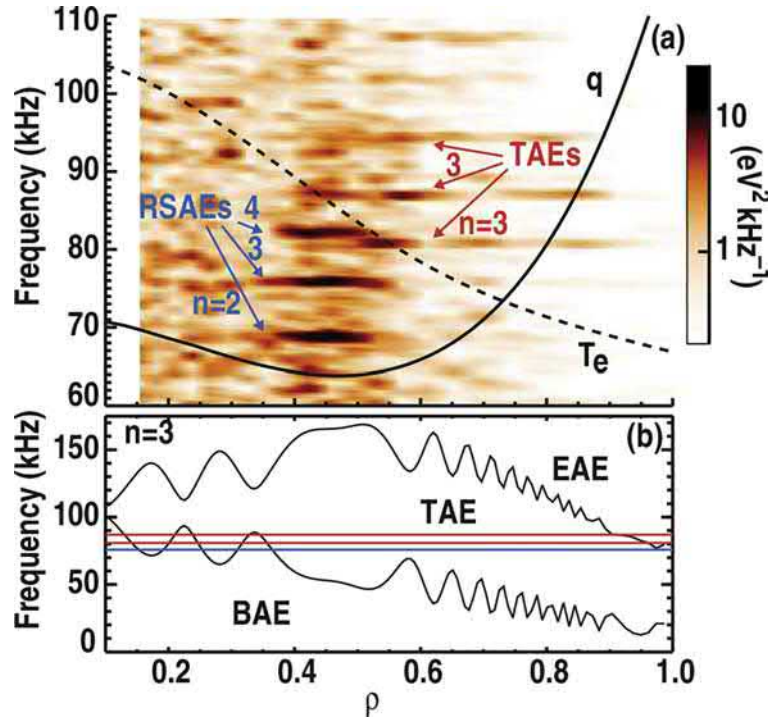


Fig. 3 (A) Experimental observations of Alfvénic eigenmode excitation in DIII-D by neutral beam-injected ions in a reversed shear plasma. Safety factor profile is indicated by «q» and electron temperature profile by «T_e». Measurements of a temperature fluctuation spectrum using the electron cyclotron emission (ECE) diagnostic is shown by the colored contours. Its numerical analysis using NOVA ideal MHD code indicates the presence of several reversed shear Alfvén eigenmodes (RSAE) and toroidicity-induced Alfvén eigenmodes (TAEs) corresponding to different toroidal mode numbers. (B) shows the radial variation of the predicted frequency of various classes of Alfvén eigenmodes: the TAEs, the ellipticity-induced Alfvén eigenmodes (EAE) and the beta-induced Alfvén eigenmodes (BAE) (Van Zeeland et al., 2006).

harmonic interacts with its neighbor, the $(m + 1)^{th}$ harmonic. The TAE location and its radial extent can be measured in experiments and compared with simulations in Fig. 3B.

The frequency of the couplet, $\omega_{TAE} = v_A/2q_{TAE}R$, is often considered as the nominal TAE frequency. If the magnetic shear is not too small the couplets are combined into a global TAE solution to be found numerically (Cheng and Chance, 1986).

Another, more complex expression for TAE frequency was derived analytically in the approximation of low inverse aspect ratio and low- β (Cheng, 1997):

$$\omega_{TAE}^2 = \left(\frac{v_A}{2qR}\right)^2 \left[1 + \frac{3}{2}\left(\frac{r_0}{R} + \Delta_{Sh}\right)^2 + 4\frac{\Delta_{Sh}}{R} + \frac{r_0^2}{R^2} - 2\frac{r_0\Delta_{Sh}}{R} + 2g^2\right], \quad (12)$$

where g^2 is the correction to the vacuum toroidal magnetic field from plasma equilibrium, r_0 is the minor radius of the mode location, and Δ_{Sh} is the Shafranov shift of the magnetic axis. The dispersion relation given by Eq. (12) provides the plasma parametric functional dependencies on the plasma equilibrium parameters, which are important in plasma diagnostics and in the stability studies of TAEs.

A single harmonic eigenmode, RSAE, was also measured, as shown in Fig. 3A (Van Zeeland et al., 2006). It is located at the maximum point of the Alfvénic m^{th} harmonic cocontinuum, i.e., at a point when its frequency is close to the continuum frequency. RSAEs, together with TAEs, are considered as the one of the most important eigenmodes limiting the energetic ion confinement. RSAEs are observed with their frequencies evolving on an equilibrium time scale, the frequency sweeping being characterized by a 30 – 100 ms timescale in present day devices, but with the timescale also depending on the toroidal mode numbers.

Characteristic AEs in 3D plasmas

The development of stellarators with sophisticated 3D geometries, allowing improved plasma confinement, has been a significant area of progress in recent magnetic confinement research (Mynick, 2006, see also Yamada, 2021).

The equations for AEs in stellarators have coefficients in 3D configurations which depend on both the toroidal and poloidal angles, leading to continuum gap formation through the mode coupling due to poloidal and toroidal variations of the equilibrium magnetic field. TAEs, in particular, have been found in stellarators experimentally and explained theoretically (Nakajima, 1996; Yakovenko et al., 2007).

The toroidal mode coupling is not important for TAEs/EAEs and GAEs/RSAEs in helical/stellarator devices such as LHD (Large Helical Device) with large toroidal period number, N , of the magnetic configuration (Fujiwara et al., 1999). A class of AEs for which toroidal mode coupling is essential exists only in 3D plasmas. In toroidal mode coupling, Fourier modes having the toroidal mode number, $n = n_1$, can couple with the Fourier modes having different, $n = n_2$, provided the condition $n_1 \pm n_2 = kN$ is satisfied, where $k = 0, 1, 2, \dots$. These modes consist of the mode family of $n = n_1$. The number of the mode families is finite, i.e., $1 + N/2$ (Schwab, 1993). Coupling among all modes belonging to the mode family of $n = n_1$ takes place and generates the helicity-induced Alfvén eigenmode (HAE) gaps (Nakajima et al., 1992; Kolesnichenko et al., 2001; Spong et al., 2003). The gap frequency can be evaluated analogously to the TAE frequency, by intersection of adjacent Alfvén continua as, $k_{\parallel m, n} = -k_{\parallel m+\nu, n+\nu N}$. Here, the magnetic field strength B is decomposed in a Fourier expansion with Boozer coordinates (Boozer, 2004) (ψ, θ, ϕ) as, $B = B_0 \left(1 + \sum_{\mu, \nu} \epsilon_B^{\mu, \nu}(\psi) \cos(\mu\theta - \nu N\phi) \right)$. The gap frequency is expressed as follows: $f^{\mu, \nu} = (\nu q^* N - \mu) \left(\frac{v_A}{4\pi R q} \right)$ and $q^* = \frac{2m+\mu}{2n+\nu N}$. This

expression corresponds to the gap frequency of the TAE with $\mu = 1, \nu = 0$, and the EAE with $\mu = 2, \nu = 0$. Typical stellarator configurations have helical coils of $l = 2$ polarity, i.e., where the coils correspond to the dominant helical component $\cos(2\theta - N\phi)$. The HAE is called HAE₂₁ here. As seen from the equation, the HAE frequency is approximately a factor of N larger than the TAE frequency. The HAE frequency is close to the TAE frequency in stellarators with medium and low N ($\approx 2 - 5$), and the HAEs may have noticeable impacts on EP transport, while the HAE frequency in high N stellarators, such as LHD ($N = 10$), is much higher than the TAE frequency. The coherent magnetic fluctuations in high beta LHD plasmas at lower toroidal field ($B_0 = 0.6$ T) were consistently interpreted as HAEs. The observed mode frequency lies above the lower bound of the HAE gap (Yamamoto et al., 2003).

In LHD plasmas with $N \gg 1$, TAEs, EAEs, GAEs and reversed shear Alfvén eigenmodes (RSAE) are essentially the same as in tokamak plasmas.

Kinetic TAEs

It was pointed out by Mett and Mahajan (1992) that a shear Alfvén wave may have significant coupling to kinetic Alfvén wave (KAW) in the vicinity of a TAE-gap where the radial structure of MHD-type shear Alfvén wave is peaked and the Larmor radius parameter, $k_{\perp}^2 \rho_i^2$, is not negligibly small. In particular, a TAE with frequency at the bottom of the gap couples to two outgoing KAWs with poloidal mode numbers m and $m + 1$, which propagate in radius and thus radiate away the energy of the TAE. This effect is called “radiative damping” of the TAE, which could be quite significant in hot plasmas, plasmas with high T_e/T_i , and for TAE mode structures with high radial gradients of plasma pressure.

At the top of the TAE-gap, KAWs with poloidal harmonics m and $m + 1$ propagate in radius toward each other. Two important effects result from this. Firstly, the outgoing energy flux resulting from the interference, is relatively small. Therefore, the radiative damping of a mode with frequency at the top of TAE-gap is much smaller than that of a mode at the bottom of TAE-gap. Secondly, waves with frequency above the TAE-gap, due to the finite Larmor radius of the first order in $k_{\perp}^2 \rho_i^2$, form a discrete spectrum of Kinetic Toroidal Alfvén Eigenmodes (KTAE), instead of the Alfvén continuum. In contrast to KAW eigenmodes considered in (Rosenbluth and Rutherford, 1975) and (Connor et al., 1983), the toroidicity effect that couples two poloidal harmonics is essential for KTAE existence (Candy and Rosenbluth, 1994; Breizman and Sharapov, 1995). To describe KTAE structure satisfactorily, the coupled system of MHD equations for TAE needs extra terms derived from the finite (i.e., non-zero) Larmor radius (FLR) of the particles.

The radial structure of the KTAE is similar to that of the TAE, apart from a narrow region at the TAE-crossing point of $q(r_{TAE}) = \frac{m+1}{n}$, where the mode width of KTAE is $\Delta r \sim \frac{\rho_i}{\epsilon^{1/2}}$. In comparison with TAEs, KTAEs have much lower radiative damping rate, but, due to their higher frequency, KTAEs experience somewhat lower drive by energetic particles. Also due to their anti-ballooning mode structure (with maximum amplitude at the high field side of the torus) KTAE can hardly be excited by trapped energetic ions produced with, e.g., ICRH, since the trapped ions are localized at the low field side of the torus.

KTAEs were searched for, and found experimentally on JET (Fasoli et al., 1996). Many KTAE modes arise above the TAE frequency, as was confirmed with TAE-antenna active diagnostics on the JET device, but their excitation by ICRH-accelerated trapped ions was rare. However, KTAEs can contribute to transport of passing α -particles and beam-injected ions similarly to TAEs.

Low frequency Alfvénic modes: BAEs, BAAEs

Below the characteristic TAE and RSAE frequencies, further gaps exist in the Alfvénic continuum and additional AE modes are found to be associated with these gaps which are products of the coupling of the Alfvénic and acoustic plasma oscillations. The Alfvén polarization branch is pushed up in frequency by finite beta effects by the value of the GAM frequency. (GAM or geodesic acoustic mode is another fundamental plasma oscillation required to be present in plasma simulations. They are important to correctly compute AE frequencies as well as the plasma turbulence.) A schematic of the low frequency gap structure and how it is formed by the finite plasma pressure effects from the Alfvénic continuum in a cylindrical approximation to toroidal geometry, are shown in the Fig. 4.

One can see that below the GAM frequency there is another, beta induced Alfvén-acoustic eigenmode (BAAE) gap, which is necessary to introduce in greater detail. The center of the BAAE gap can be estimated by comparing the sideband acoustic continuum, $\omega = v_A/q$, and the so-called “modified” Alfvén branch, $\omega = k_{\parallel} v_A / \sqrt{1 + 2q^2}$, which exists in the vicinity of the rational surface. Those uncoupled branches are shown in Fig. 4A.

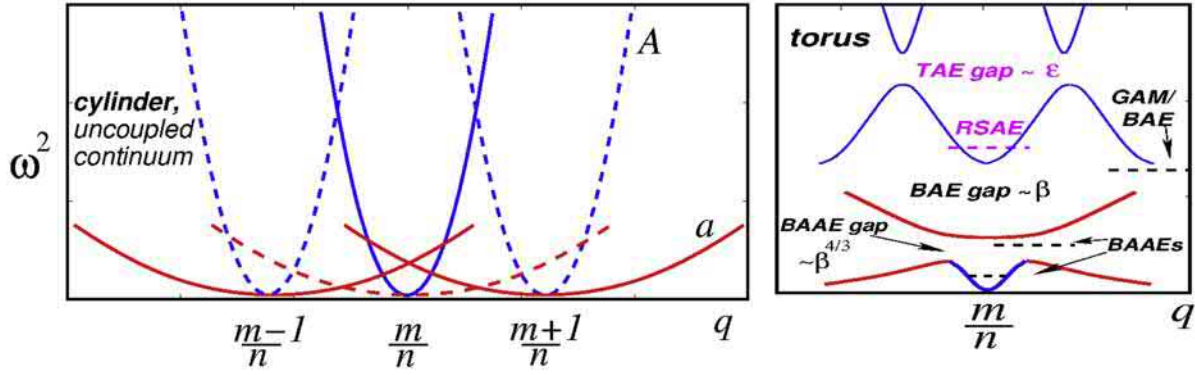


Fig. 4 Low frequency Alfvén-acoustic continua depicted for the cylinder equilibrium (left) and for the torus (right). Curves A (blue color) refer to the Alfvén branches whereas curves a correspond to the acoustic branch. Vertical axis is the frequency squared and horizontal axis is the radial variable, chosen here as safety factor, q , so that m/n is the location of the rational surface.

Thus, the center of the BAAE gap is at the frequency

$$\omega_{BAAE} = \frac{\gamma\beta_p}{2q^2} \frac{v_A/R_0}{1 + \sqrt{\gamma\beta_p/2}\sqrt{1+2q^2}}, \quad (13)$$

and the width of the BAAE gap is $\Delta\omega = (\gamma\beta_p/q)\sqrt{1+2q^2}$.

Moreover, in [Gorelenkov et al. \(2007a\)](#) and [Gorelenkov et al. \(2007b\)](#) these gaps were numerically shown to contain eigenmodes, beta induced Alfvén-acoustic eigenmodes (BAAEs). The eigenmodes are localized near the accumulation points (extrema continuum points) of the gaps. Experimentally they are shown to agree with the ideal MHD code NOVA simulations ([Gorelenkov et al., 2007a](#)).

A particularly surprising experimental result relates to BAAE observations, since they are the product of the coupling to the acoustic branch. Thus, BAAEs are subject to strong thermal ion Landau damping. It is then expected that these modes should be excited in a region of parameter space where the electrons have significantly higher temperature than the ions, $T_i \ll T_e$. Nevertheless, BAAEs were excited and identified in many toroidal devices such as NSTX, JET ([Gorelenkov et al., 2007a,b](#)), DIII-D, HL-2A ([Yi et al., 2012](#)) and also reported from AUG ([Curran et al., 2012](#)) when electrons had a comparable temperature to the ions. The Alfvén acoustic gaps are much more complex ([Kolesnichenko et al., 2011](#)) in helical devices such as stellarators and these phenomena are at an initial stage of experimental studies ([Bertram et al., 2012](#)).

Because the BAAE frequency is associated with the safety factor profile, it can be used to infer the q value at the location of BAAE excitation. This important technique is discussed in section “**Implications for plasma diagnostics, MHD spectroscopy.**”

Alfvén-acoustic continuum and the sea of modes

The discovery of the low frequency Alfvén acoustic modes enriched the understanding of these low frequency, potentially deleterious, instabilities ([Gorelenkov et al., 2009](#)). In addition to the aforementioned TAEs, there are many other eigenmodes present in the plasma which can be associated with the Alfvénic waves, i.e., those which have phase velocity equal to the Alfvén speed. As an example of the continuum formed by two fundamental oscillations, the Alfvénic and the acoustic, the instability diagram computed for an ITER plasma is illustrated in [Fig. 5](#) ([Gorelenkov et al., 2008](#)). It includes several gaps where different eigenmodes were found theoretically and experimentally in current devices, including the beta induced Alfvén-acoustic eigenmode, beta induced Alfvén eigenmode, toroidicity induced Alfvén eigenmode, and ellipticity induced Alfvén eigenmode gaps. The figure also shows schematically the location of higher frequency compressional Alfvén eigenmodes (CAEs) and of the global Alfvén eigenmodes (GAEs).

Implications for plasma diagnostics, MHD spectroscopy

The phenomena which energetic particle induce, such as collective instabilities, reveal important information about the plasma. A knowledge of the properties of the resonant particles and the properties of the underlying modes is instrumental in the development of relatively straightforward techniques for measuring plasma parameters. Consider just the frequency of the modes as an example. The detection of Alfvénic instabilities driven by fast ions was proposed as a plasma diagnostic technique by [Fasoli et al. \(2002\)](#). Instability frequencies and mode structures are well understood via linear perturbative theories and can therefore serve as the basis for so-called MHD spectroscopy. This method allows information about the macroscopic plasma properties and the energetic particles to be extracted.

MHD spectroscopy was first proposed as an *active* measurement technique, involving the use of the stable shear Alfvén waves excited by antennas or internal coils ([Goedbloed et al., 1993](#)). It was then extended to include Alfvén eigenmodes excited by EPs in a plasma ([Sharapov et al., 2001](#); [Fasoli et al., 2002](#)). Further extension to Alfvén acoustic modes (BAAEs) was developed

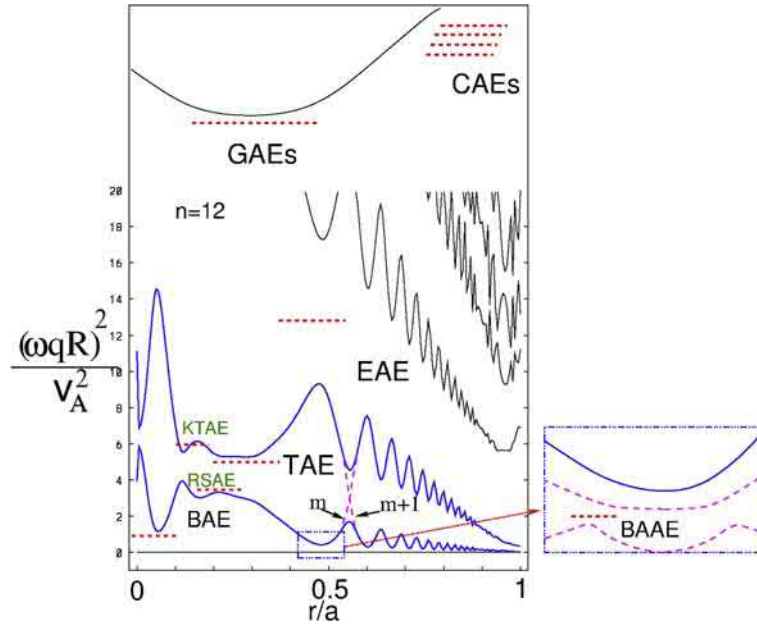


Fig. 5 Prediction of ITER Alfvén-acoustic continuum, plotted as a function of normalized minor radius, r/a , obtained with the help of the ideal MHD code NOVA at $n = 12$. Several of the lowest frequency gaps are plotted, corresponding to BAE, TAE, EAE and BAAE modes. The continua are upshifted by the coupling of the Alfvénic and acoustic waves as explained above.

by Gorelenkov et al. (2009). If both *active* and *passive* spectroscopies are combined, EP driven modes can help to provide a powerful technique for the accurate determination of plasma equilibrium information in toroidal plasmas.

As discussed by Fasoli et al. (2002) collective modes must satisfy certain conditions to be suitable for application to MHD spectroscopy. Firstly, their dispersion relation should depend on macroscopic plasma properties which are distinct from the EP instability drive. The modes should also persist without suffering strong damping, i.e., the eigenmode phase velocity should not be close to the electron and ion thermal speeds. A third requirement is that the frequency range of the modes should be far from the frequency range of background micro-turbulence. Finally, the mode amplitude should not be so high that the modes significantly affect the EP confinement and bulk plasma properties. Alfvén eigenmode instabilities typically satisfy these requirements.

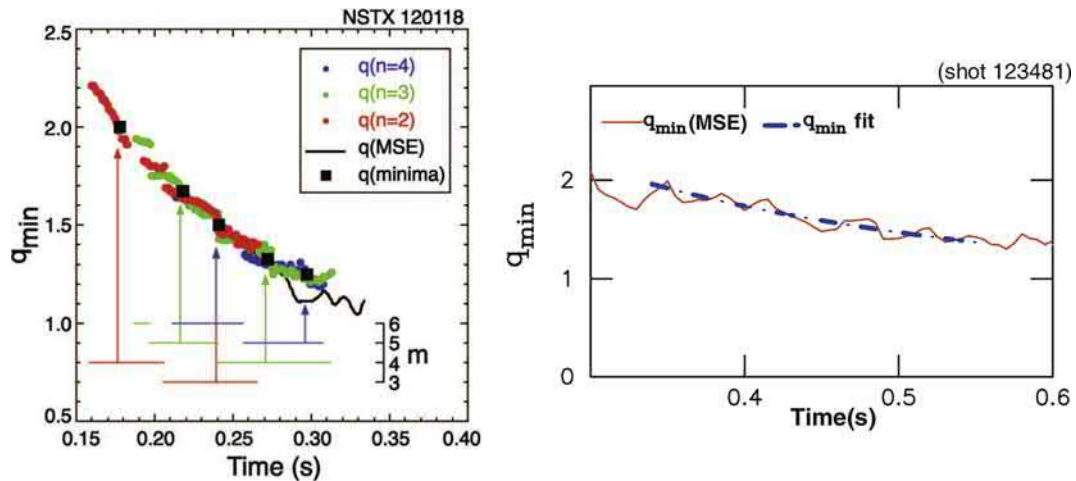


Fig. 6 Application of the MHD spectroscopy technique using observations of two distinct types of Alfvén eigenmode: left figure shows the technique applied to NSTX, where observations of RSAEs with different toroidal mode numbers, n , were used to derive the time evolution of the value of q_{min} , the minimum value of the safety factor in the plasma core. Also shown are two further experimentally measured values of the time evolution of q_{min} which are in good agreement with that derived from MHD spectroscopy: the equilibrium reconstruction values of q_{min} (solid squares) and MSE (motional Stark effect diagnostic) inferred values (solid line). The right figure represents a similar exercise using observations of BAAE modes in the same device, where q_{min} fit shows the time evolution of q_{min} derived from BAAE observations and q_{min} (MSE) that derived from the motional Stark effect diagnostic.

This spectroscopic method was applied to the NSTX experiment by Fredrickson et al. (2007), from which the illustration shown in Fig. 6 (left) is adopted. The physics underlying the interpretation of the results is that the GAM frequency is just below the shear Alfvén continuum frequency if corrected for the toroidal plasma rotation. To remove the frequency shift due to the pressure gradient, the safety factor should be close to 1. This helps to explain the good agreement of the time evolution of the q_{min} plots derived from the MHD spectroscopy with those determined by MSE measurements. The knowledge of q_{min} and its time evolution is often critical for designing and understanding plasma scenarios

A similar technique was applied to BAAEs with an up-swept frequency satisfying the modified Alfvénic branch dispersion, $\omega = k_{\parallel} v_A / \sqrt{1 + 2q^2}$ (Gorelenkov et al., 2009). Results of q_{min} reconstruction from such an application are shown in Fig. 6 (right). The advantage of the latest technique is that there is no frequency shift due to the GAM or the pressure gradient.

Nonlinear evolution of EP-driven modes and energetic particle transport

To understand how AEs behave on long timescales characteristic of EP slowing-down and plasma confinement and what could be the results of the AE interaction with energetic particles, i.e., for making projections for future burning experiments, the nonlinear phase of the AE instabilities must be investigated. For shear Alfvén waves excited by EPs, the main effect to be considered is the wave-particle nonlinearity affecting the slope of the EP distribution and the AE amplitude and its phase. On a long timescale, the coupled AE/EP dynamics is determined mostly by the competition between the electric field of the AE, which tends to flatten the EP distribution function, and collisional effects restoring the unstable EP distribution. The latter could be due to Coulomb collisions or other mechanisms such as interaction with ICRH waves.

The nonlinear phase of a single EP-driven mode observed experimentally may exhibit several scenarios, which could be categorized in terms of the mode amplitude and the mode frequency. Namely, the nonlinear temporal evolution of the mode amplitude could be either steady-state, bursting (a rapid increase of the amplitude followed by a decrease and a longer quiet period), or having either periodic or chaotic time modulation. In the case of a steady-state mode amplitude, the amplitude saturation is related to formation of a plateau in the EP distribution near the resonance. The case of a bursting AE amplitude is associated with generation of long-living nonlinear fluctuations, holes and clumps, in the EP distribution. The holes and clumps propagate in the phase space and have frequencies which sweep significantly from an initial value (by as much as 50% in some circumstances). The cases of periodic and chaotic amplitude modulation correspond to an EP distribution function oscillating around the plateau, but with amplitude insufficient for triggering a self-sustained nonlinear hole or clump. Theory of nonlinear mode evolution (Berk et al., 1996) rather well describes the qualitative nonlinear physics in so-called “near-threshold” approach of a single mode with very few key parameters involved (see section “Nonlinear models”).

In the case of multiple EP-driven modes, the experimental data is more complex in both linear and nonlinear phases. Often, out of several possibly unstable modes excited by the same EP population, only one type of mode becomes dominant, e.g., RSAEs dominate over TAEs in the Advanced Tokamak scenarios. Sometimes EP-driven modes are excited over a wide frequency range in an “avalanche” event such as observed on NSTX (Fredrickson et al., 2013; White et al., 2020). On a long timescale and with many EP-driven AEs, a plateau in EP distribution formed for one AE is not a plateau for other AEs, and so a “global” plateau is formed over much broader phase space when a steady-state AE scenario takes place. The bursting amplitude AE scenario can hardly be typical of the multiple AE case since the frequency sweeping range is much larger than the AE frequency separation.

Arguably the studies of the nonlinear effects are not as advanced as their linear counterparts. However, they are critical for building credible predictive models for EP relaxation in both velocity and real space.

Nonlinear models

The research in this area capturing the essence of the underlying physics in a theoretically tractable manner is highlighted below and includes significant progress achieved by several nonlinear models that describe rather well the observed evolution of energetic particle driven modes in experiments.

In burning plasmas conditions, such as in ITER, the characteristic time scales for building up the EP component has to be gradual and much longer than the Alfvénic growth time, $1/\gamma_L$, where γ_L is the linear growth rate. Under such conditions the nonlinear processes associated with relaxation of the EP distribution due to AEs require near-threshold consideration. A series of works on this topic emerged from the nonlinear kinetic treatment of Alfvénic instabilities using the 1D bump-on-tail problem, which became known as the Berk-Breizman (often abbreviated to BB) model (Berk et al., 1996). It includes the collision term (representing particle annihilation and injection processes) and wave damping.

An important parameter of the BB model is the ratio of the effective collision frequency to the linear growth rate $\hat{\nu} \equiv \nu_{eff}/\gamma$, where the growth rate γ is determined by the difference between the linear growth rate and the background dissipation γ_d : $\gamma = \gamma_L - \gamma_d$. $\hat{\nu}$ restores the distribution function of the resonant particles in the velocity space near the resonances. A cubic nonlinear non-local equation was derived (Berk et al., 1996) that depends upon the single control parameter $\hat{\nu}$:

$$\frac{dA}{d\tau} = A(\tau) - \frac{1}{2} \int_0^{\tau/2} dz z^2 A(\tau - z) \int_0^{\tau-2z} dx e^{-\hat{\nu}(2z+x)} A(\tau - z - x) A(\tau - 2z - x), \quad (14)$$

where A is the amplitude of the mode and $\tau = (\gamma_L - \gamma_d)t$ is the normalized time. The scattering frequency, $\hat{\nu}$, accounts for the competition between the mode itself and the relaxation processes which recovers the initial distribution. A more sophisticated cubic equation was subsequently derived which includes the velocity slowing down and has different integrand than Eq. (14) (Lilley et al., 2009).

Solutions of this equation at different values of $\hat{\nu}$ are summarized in Fig. 7 (Berk et al., 1996). At low collisionalities, $\hat{\nu} < 4.4$, the solution pulsates, with more complex time oscillations becoming irregular at lower scattering. The solutions show a well-defined sequence of transitions as $\hat{\nu}$ is decreased from periodic, through multiply periodic, to chaotic behavior. At very low collisionality an explosive behavior emerges, i.e., when the assumption in the derivations breaks down and when the amplitude becomes large enough. It is possible to consider the saturated steady-state solution of that system as a fixed point of the system dynamics.

The cubic equation of nonlinear mode behavior, Eq. (14), breaks down if the particle response needs to be considered. Such a consideration was published as a conjecture that structures are formed in the fast ion distribution function (holes and clumps) corresponding to a deficit or the surplus of fast ions (Berk et al., 1997). In later simulations these structures were confirmed with the help of a fully nonlinear Vlasov-Maxwell code (Lilley et al., 2010). Holes and clumps support long-lived nonlinear BGK waves

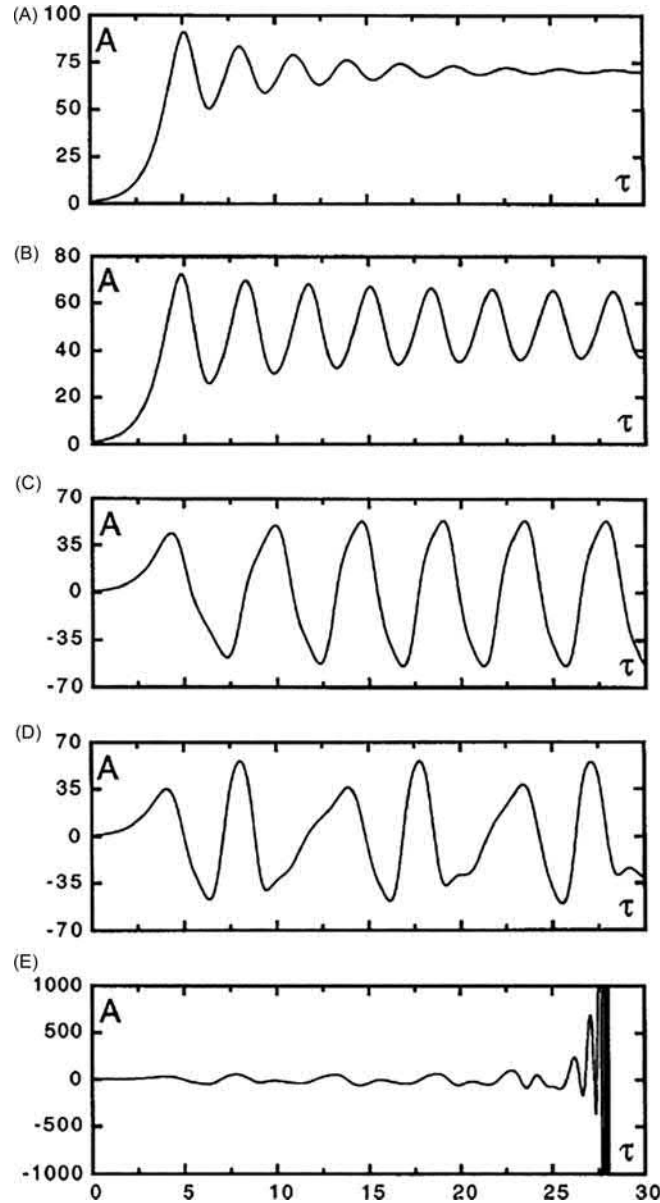


Fig. 7 Solutions of the cubic equation Eq. (14) for the nonlinear amplitude evolution of a model fast ion driven, but marginally unstable modes at $A(0) = 1$ and various values of $\hat{\nu}$: (A) $\hat{\nu} = 5.0$, (B) $\hat{\nu} = 4.3$, (C) $\hat{\nu} = 3.0$, (D) $\hat{\nu} = 2.5$, and (E) $\hat{\nu} = 2.4$. Note that $\hat{\nu} \equiv \nu_{eff}/\gamma$, where ν_{eff} is the effective collision frequency of the energetic particles and γ is the linear growth rate of the mode. The significance of the different plots is discussed in the text.

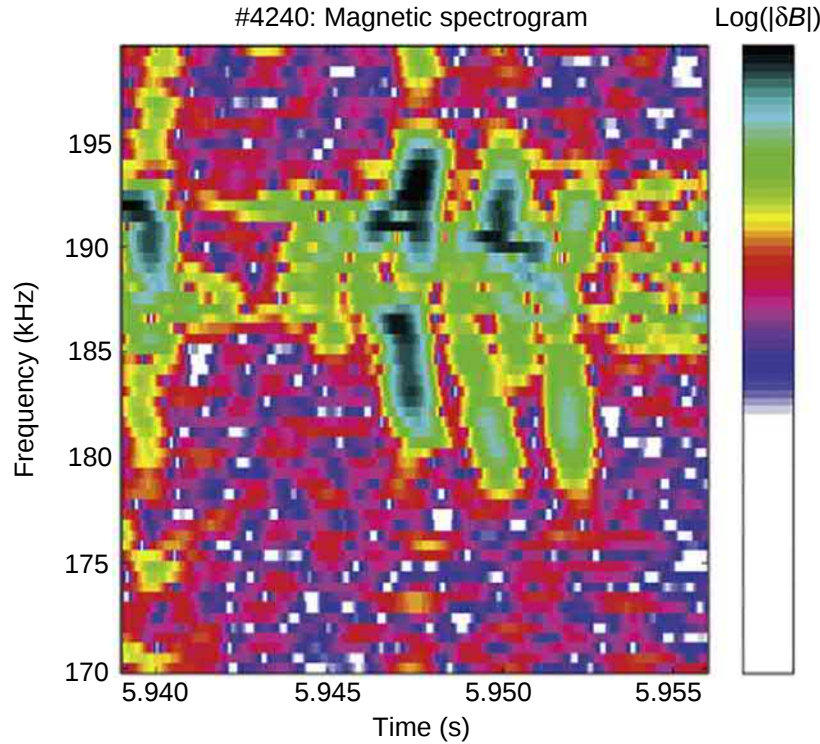


Fig. 8 Frequency chirping of an unstable mode during a shear optimized JET tokamak plasma discharge using deuterium-tritium fuel. The characteristic frequency change is $\sim 5\%$ over ~ 1 ms. The spectrum is measured by the magnetic field probes a.k.a. Mirnov coils.

(Bernstein et al., 1957) where the background dissipation is balanced by the chirping (i.e., frequency sweeping) behavior. The predicted frequency chirping was seen in JET DT operation, as show in Fig. 8 (Pinches et al., 2004).

Related observations from the JT-60U tokamak (Shinohara et al., 2001) were obtained from experiments with high energy neutral beams injected into plasma (0.36 MeV) in which the velocity of the EP component was near the Alfvén speed, $v_j/v_A \sim 1$. Several instabilities differing in time evolution and frequency were observed. A specific instability was identified as an abrupt large amplitude event (ALE), and this appeared at the end of the fast frequency sweeping (FS as referred to by the authors) instability. FS had a characteristic frequency chirp over the range 10 – 20 kHz with a duration of several milliseconds and was often characterized by a drop in neutron emission, as well as by losses of fast ions from the plasma.

All but one explosive nonlinear scenarios of wave-particle interaction predicted by the Berk-Breizman model were validated experimentally in JET tokamak plasmas. In specific JET experiments, TAEs were excited by energetic hydrogen-minority tail ions accelerated by ICRH. The unstable distribution of energetic ions in this case was replenished by the RF-induced diffusion of the energetic ions, which exceeded the diffusion due to Coulomb collisions by an order of magnitude. Fig. 9 top shows how the gradually increasing ICRH power and associated net growth rate of the TAEs caused the modes to go through the sequence of nonlinear saturated scenarios displayed in Fig. 9 bottom. These scenarios replicate the states shown in Fig. 7.

Firstly, a steady-state nonlinearly saturated amplitude of each mode that corresponds to Fig. 7A is seen in the spectrogram as a single line. This is followed by the nonlinear scenario with a periodic modulation of TAE amplitude (Fig. 7B) that gives a “pitchfork splitting” effect in the spectrogram with several parallel sidebands separated by the frequency of the modulation. Subsequently, the amplitude modulation becomes deep enough for the mode amplitude to cycle close to zero, thus corresponding to Fig. 7C. Finally, when the modulation becomes comparable to the amplitude itself, the chaotic scenario of the mode emerges with non-periodic amplitude variations (see Fig. 7D) and the phase flips near zero amplitude. This chaotic scenario is seen in the spectrogram at late times as a noisy band of frequency width similar to that of the pitchfork-split TAE.

An explosive nonlinear scenario corresponding to Fig. 7E is rather difficult to achieve with ICRH-driven TAEs, which are dominated by the RF-induced diffusion at the resonance due to strong, RF-driven, pitch angle scattering of the EPs. Such a scenario can be produced more easily for TAEs driven by NBI-produced energetic ions, where replenishment of the EP distribution function around the resonance condition is dominated by slowing down from the EP injection energy and Coulomb scattering is much weaker.

Reduced analytic and quasi-linear models

Due to the larger size and cost of future burning plasma experiments it is increasingly important to develop reduced models capable of predicting reliable scenarios for AE excitation and evolution in both present day and future experiments. Several models exist with such capabilities, for both single-mode and multi-mode cases. They include simple critical gradient models (Ghantous

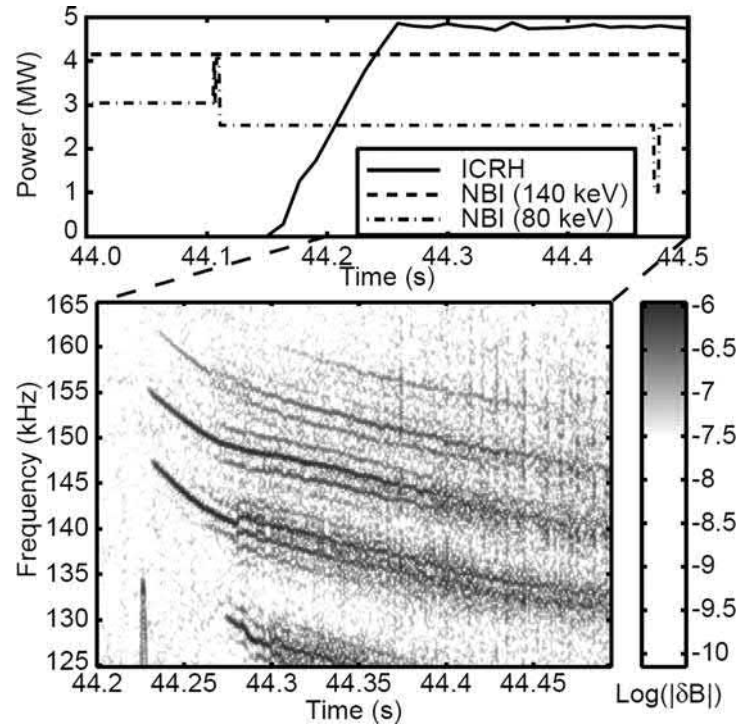


Fig. 9 Top: the wave-forms of ICRH and NBI auxiliary heating power in JET discharge #49447. Bottom: Amplitude spectrogram of magnetic fluctuations showing the various non-linear scenarios of TAE evolution. The TAEs have toroidal mode numbers $n = 3$ to 6.

et al., 2012; Bass and Waltz, 2020), which describe the relaxation of the EP plasma component without resolving the velocity distribution function. More sophisticated models include the quasi-linear (QL) simulations which incorporate the phase velocity diffusion by resolving the wave-particle resonances, their overlaps and particle dynamics near the resonances (Ghantous et al., 2014; Gorelenkov et al., 2019).

The reduced models are developed with the goal of being predictive for efficient WDM simulations. Depending on the efficiency of the models they can be used for different purposes within the simulations, yielding less (simple models without resolution of details of the distribution function) or more (more sophisticated, quasi-linear models) reliable predictions. Because of their demands, WDM require significant investment in terms of verification and validation. Nevertheless, the benefits for fusion research are substantial, since the required computational resources are significantly lower and yet the models are accurate enough to predict such demanding characteristics as the induced current drive.

Reduced analytic models have been developed recently. One particular noteworthy result emerged from simplification of the cubic equation (14) in the limit when the resonant particle dynamics can be described by the dominant pitch angle scattering only. This allowed derivation of the AE amplitude in a form which is relatively easy to evaluate:

$$A(t) = A(0) \frac{e^t}{\sqrt{1 - gA^2(0)(1 - e^{2t})}}; g \equiv \int d\Gamma H \frac{\Gamma(1/3)}{6\nu_{\text{eff}}^4} \left(\frac{3}{2}\right)^{1/3}, \quad (15)$$

where g is the resonance-averaged effective pitch angle scattering contribution and H is the phase space weighting (Duarte et al., 2019).

It is also noteworthy that the predictions of the AE amplitude can be performed analytically, which is extremely useful verification for numerical simulations and reduces simulation requirements.

Semi-analytic 1D models for bump-on-tail drive

Important insights into the nonlinear physics can be derived from simple models in which, by reducing the dimensionality of the problem, it is possible to focus on the underlying physics. Several 1D models (Vann et al., 2003; Lilley et al., 2010, and Lesur et al., 2009) were developed with the aim of capturing the important features of the wave-particle resonant interaction.

Vann et al. (2003) proposed a Vlasov (1968) approach to finding the EP distribution function in order to numerically characterize the nonlinear behavior in a marginally unstable dissipative system by accurately computing the wave particle interaction near each resonance in velocity space. A somewhat simpler approach by Lilley et al. (2010) addressed single resonance dynamics and elucidated the relative roles played by various collision effects, such as pitch angle scattering and velocity slowing down,

Krook-like relaxation and phase-space diffusion. The model was capable of including several additional important elements, such as the slowing down process (drag), which is destabilizing for energetic particle modes. The model also generated asymmetries in the evolution of the frequency chirps. Another group led by Lesur and Idomura (2012) independently developed a 1D electrostatic code and defined a qualitative definition for each of the chirping regimes with drag and diffusion. These nonlinear 1D models show that mode behavior close to marginal stability is well characterized both theoretically (Berk et al., 1996) and experimentally using simple parameters and can even provide opportunities to extract additional information on the particle phase space distribution from the measured instability spectral features (Fasoli et al., 1998).

Models for initial value simulations

There are two widely used codes which are unique in treating the wave-particle phenomena found in experiments. These codes, which treat the influence of the energetic particles perturbatively upon the linear eigenfunctions of the system, are ORBIT (White and Chance, 1984) and HAGIS (Pinches et al., 1998). These models have been used successfully to describe the fully nonlinear interaction of fast ions with the linear waves in the plasmas and have even shown how it is possible to recover the internal amplitude of a mode from observations of its frequency chirps (Pinches et al., 1998). ORBIT was able to successfully simulate AE chirping in the NSTX device and to predict the AE avalanche onset by slightly increasing (by 10%) the growth rate of the unstable modes from the values computed prior to the avalanche.

Going beyond the linear MHD eigenmode structures, as required for lower-frequency BAE modes, necessitates formulation of a model based on a nonlinear MHD approach. This is the method adopted in the MEGA code (Todo and Sato, 1998), which is a kinetic-MHD hybrid model that has been used to simulate the interaction of alpha particles with various Alfvén eigenmodes. The plasma is divided into two species: the background plasma and the energetic particles. The behavior of the energetic ions is coupled to the MHD equations through the current that they carry. A linearized version of the code also exists to facilitate understanding of the nonlinear behavior observed.

Another code that can capture the non-perturbative effects of energetic particles is the HMGC code (Briguglio et al., 1995). This is a hybrid MHD-particle simulation code that couples the sets of reduced MHD equations for the fluctuating fields with the gyrokinetic equations of motion for the fast ions. At each time step the particles move in the 3D electromagnetic field before the field is updated using a fluid-equation solver.

The above approaches use the kinetic treatment for the energetic particle behavior. However, it is also possible to take a gyro fluid-based approach, as done by the TAEFL code (Spong et al., 2012). This allows the study of TAEs destabilized by fast ions while still capturing the mode damping due to neighboring shear Alfvén continua. Recent work (Spong et al., 2012) has shown good qualitative agreement between this approach, experimental observations in a DIII-D discharge and the results of two gyrokinetic codes, GTC (Lin et al., 1998) and GYRO (Candy and Waltz, 2003), both for the mode evolution (frequency sweeping) and mode structure.

Implications for burning plasmas

One of the main uncertainties in predicting with confidence a burning plasma performance is a possible excitation of multiple Alfvénic instabilities (Fasoli et al., 2007; Gorelenkov et al., 2014). If the amplitudes of these instabilities are not benign, a transport of fast ions exceeding the neoclassical values can arise giving a radial re-distribution of the fast ions, affecting the power deposition profile and endangering the integrity of the first wall.

The fast ions in ITER will consist of fusion-born alpha-particles, ICRH-accelerated ions, and 1 MeV deuterium beam ions. In addition to the alpha-particles, the D-beam ions injected tangentially into ITER plasmas, will also have a destabilizing effect on TAEs. Due to the anisotropic distribution function of the beam, the beam drive of AEs can be comparable to the alpha-particle drive (Gorelenkov et al., 2005), implying that the beam-driven Alfvénic instabilities should be carefully assessed as an addition to alpha-driven instabilities, in particular by following the plasma simulation along the entire transport path that uses NBI toward achievement of the $Q = 10$ operation point.

The relaxation of the fast ion distribution caused by the modes depends on whether the wave-particle resonances overlap and can induce a stochastic transport of the fast ion orbits across a significant radial distance. The stochasticity condition depends primarily on the position of wave-particle resonances in the phase space, on the number of unstable modes, their toroidal mode numbers and the maximum amplitudes they reach. It also depends on whether the instabilities are in stationary regimes, where their frequencies and amplitudes are not changing much in time, or they are in the frequency chirping and amplitude bursting regime associated with the structures in the fast ion phase space (White et al., 2020) as described in section “Nonlinear evolution of EP-driven modes and energetic particle transport.”

For the linear AE instability the dominant dependence on the mode number comes from the finite orbit width effect (Fu and Cheng, 1992; Breizman and Sharapov, 1995), which can be cast into the condition that the radial width of the mode eigenstructure is comparable to the radial width of the EP drift orbit for maximizing the mode growth rate,

$$k_{\perp} \rho_f \sim 1, \quad (16)$$

where ρ_f is the Larmor radius of the fast (beam or alpha) ion.

This condition is important, especially for AE stability analysis in ITER. It allows an approximate evaluation of the most unstable AE mode number and the range of unstable modes.

As the linear AE theory is well developed, an important further step is the validation and verification of various linear and nonlinear TAE stability codes. An international benchmarking exercise has therefore been performed over several years under the auspices of the Energetic Particles (EP) Topical Group of the International Tokamak Physics Activity (ITPA, <http://www.iter.org/org/team/fst/itpa>). The overall aim of this exercise was, for each numerical simulation code, to verify the code's implementation and to validate its predictive capability for ITER.

Here we present a table of TFTR and JET energetic particle parameters achieved to date together with the predicted EP parameters in ITER. The most critical parameters for energetic ion studies are presented, such as their local gradient values $|R\nabla\beta_f|$, ratio to the Alfvén speed, EP density and pressure, which are important for evaluating the AE stability in the referenced tokamaks.

AE linear growth and damping rates

A comprehensive description of Alfvénic instabilities excited by energetic particles requires a development of numerical tools accurately computing, for a given equilibrium, the spectrum of Alfvén eigenmodes (AEs) and the energetic particle orbits. Historically, different approaches were used for describing both AEs and the orbits, and the numerical tools developed were validated against experimental data from the range of experimental devices. In order to assess with confidence the strong and weak sides of the codes developed, an international benchmarking exercise was implemented under the auspices of the Energetic Particles Topical Group of the International Tokamak Physics Activity (ITPA, <http://www.iter.org/org/team/fst/itpa>) to verify various implementations and to validate the codes' behavior for ITER.

The code benchmarking exercise centered around the application of each code to a specific, well-documented JET plasma (Borba et al., 2010; Testa et al., 2010) in which the damping of global TAEs was measured, or the codes were run for the analytic case (Könies et al., 2018). The models employed included the perturbative kinetic extension to MHD codes, CASTOR-K (Huysmans et al., 1993) and NOVA-K (Gorelenkov et al., 1999), warm dielectric tensor model code LEMan (Popovich et al., 2006), gyro fluid code TAEFL (Spong et al., 1994) and linear gyrokinetic code LIGKA (Lauber and Günter, 2008). Most importantly, the damping rate of a specific TAE mode was measured with the active TAE-antenna diagnostics, so that this activity validated the codes against experimental data.

The codes used found two TAE modes with $n = 3$ and radially extended mode structures. One of these, a Global TAE mode, was identified as a likely candidate corresponding to the measured TAE damping rate, because it had the eigenfrequency $f = 179 - 181$ kHz which did not cross the Alfvén continuum. The computed TAE structures allowed the interaction with the continuum at the edge and near the center. Computed damping rates gave a good agreement with the experiment for the $n = 3$ TAE for the codes using different approaches, see Fig. 10. Analysis shows that the computed rates consist of primarily two dominant mechanisms: the continuum and the radiative damping effects.

More recently, results have been obtained from a further significant code cross-verification focusing on the growth rate calculations by many codes, once again in the linear regimes. These codes use somewhat different and independent approaches to compute the fast ion growth rates and included CAS3D-K (Könies et al., 2008), GYGLES (Mishchenko et al., 2009), TAEFL (Spong et al., 1994), AE3D-K (Spong et al., 2010), NOVA-K (Gorelenkov et al., 1999), HMGC (Briguglio et al., 1995), MEGA (Todo and Sato, 1998),

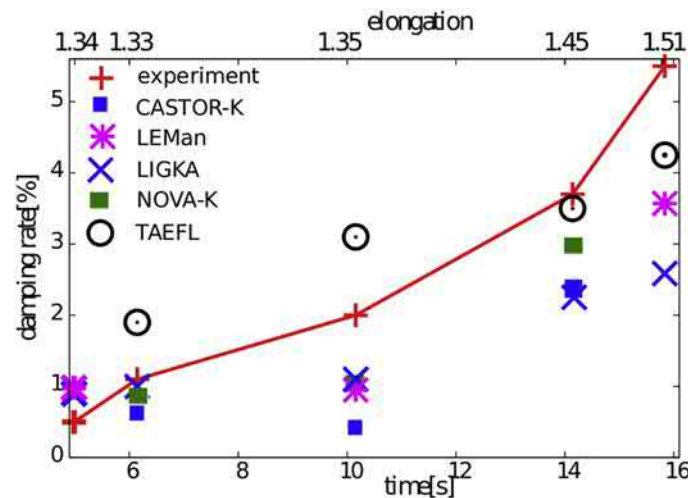


Fig. 10 The figure shows damping rates for TAE modes calculated by the computational codes indicated as a function of time and the comparison with the JET experimentally inferred damping rate.

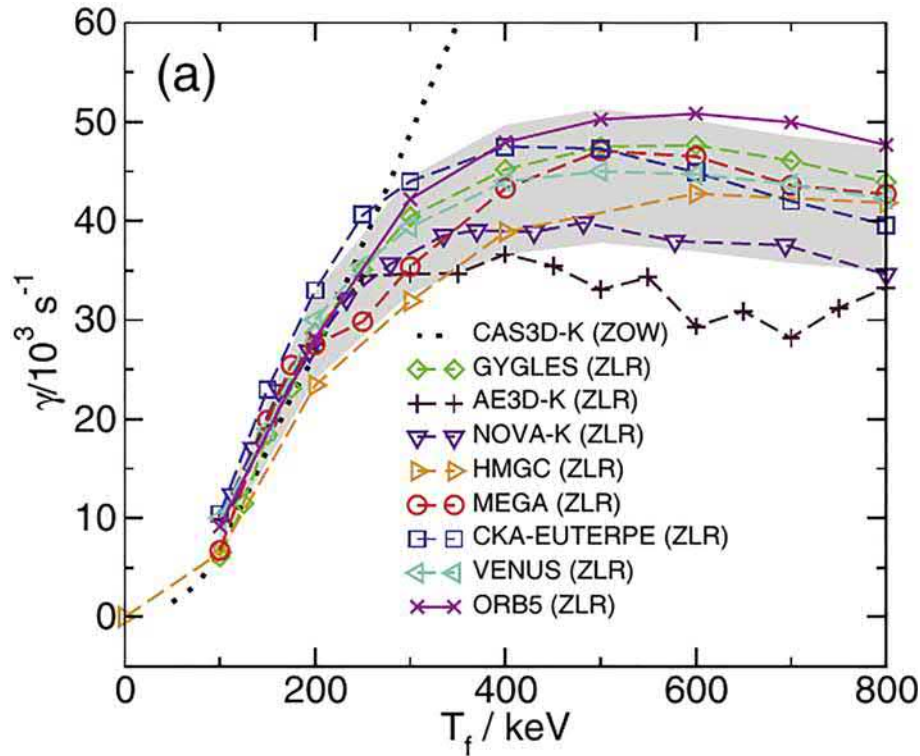


Fig. 11 Predictions for the $n = 6$ TAE growth rate, γ , as a function of the energetic particle ‘temperature’ (assuming a Maxwellian distribution for the energetic particles), T_i , derived from a range of codes using an analytic formulation of the plasma profiles (the comparison was performed within the framework of an ITPA benchmarking activity).

CKA-EUTERPE (Kleiber, 2006), VENUS (Cooper et al., 2011). Fig. 11 represents the growth rates resulting from these nine codes whose models range from the kinetic MHD/hybrid to gyro-kinetic and gyro-fluid approaches.

To simplify the comparison among the codes, analytic plasma profiles were chosen with a large aspect ratio $R/a = 10/1$ and a q -profile that allowed the growth of only one TAE couplet (m and $m + 1$ poloidal harmonics). Within a reasonable error range, indicated as a shaded region in Fig. 11, the code predictions are in agreement. They all reflect the existence of a plateau in the growth rate dependence as a function of the fast ion temperature, which is an important result underpinning burning plasma projections. For this analysis, the EP distribution function was taken to be a Maxwellian to allow ease of comparison among the code calculations. The growth rates shown include the finite orbit width effect but neglect the finite Larmor radius effect (i.e., zero Larmor radius is assumed). The agreement among the codes is satisfactory in terms of the growth rate vs ion temperature, within $\sim 15\%$ for energies up to the plateau point at ~ 400 keV.

Alfvén wave excitation in ITER baseline scenario

Predictive studies of both the linear and nonlinear behavior of Alfvénic instabilities in ITER plasma scenarios are a key priority for exploitation of the various numerical codes applied in the benchmarking exercise described in section “Alfvén wave excitation in ITER baseline scenario.” An important requirement for such predictions is the credibility in the representation of the dominant damping mechanisms and accurate calculations of the growth rates. The damping rates should include most of the known mechanisms, namely, electron and ion Landau damping, collisional electron damping and radiative/continuum damping. The fusion alpha-particle drive should include both the finite orbit width (FOW) and finite Larmor radius (FLR) effects, which are important to correctly represent the range of unstable AE growth rates and how they saturate at the maximum value, as defined by Eq. (16). The dominant drive of AE instabilities is generated by alpha-particles resulting from DT reactions, but other ions could also contribute, as pointed out in the introduction to this sub-section.

An important example of possible alpha-particle driven TAEs (Fasoli et al., 2007) is the case of an ITER baseline scenario with 15 MA current at the $Q = 10$ operational point. The main plasma parameters were predicted with the ASTRA transport code (Polevoi et al., 2002). In this scenario, the alpha-particle density is moderately high, with a normalized alpha-particle pressure on the plasma axis, $\beta_\alpha(0) \approx 1.2\%$, and with the maximum radial gradient at about $r/a \leq 0.4$ (Pinches et al., 2015). The predicted plasma profiles for this scenario are illustrated in Fig. 12.

This plasma has sawteeth, which make the safety factor profile $q(r)$ very flat in the central region of $r/a \leq 0.5$, with $q(0) \leq 1$. For flat $q(r)$ -profiles, TAE-gaps in the Alfvén continuum localized at

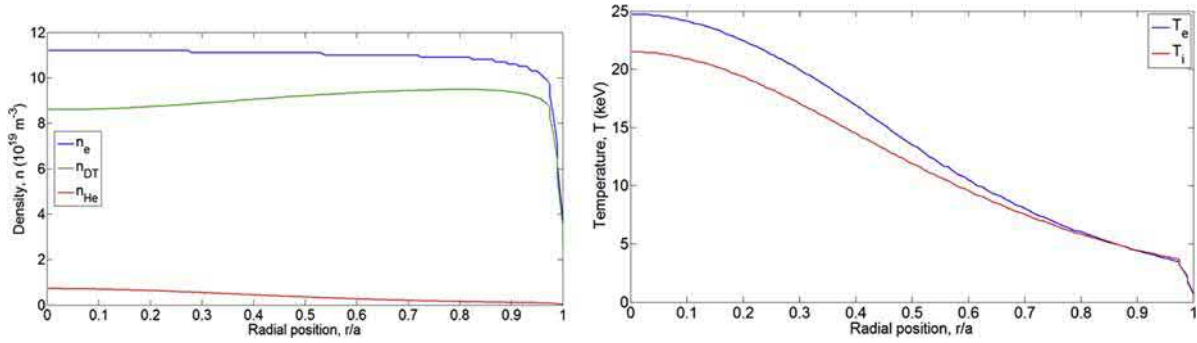


Fig. 12 Plasma profiles predicted with the ASTRA transport code (Polevoi et al., 2002) for ITER baseline scenario with current $I_p = 15\text{MA}$, $R = 621 \text{ cm}$, $a = 200 \text{ cm}$, and $B_T = 5.3 \text{ T}$ (Pinches et al., 2015). Left: density of electrons, D-T ion mixture (50:50), and He ash. Right: temperature of thermal ions and electrons.

$$q_{TAE} = \frac{m + 1/2}{n}, \quad (17)$$

become sparse in the core region of $r/a \leq 0.4$, where the maximum radial gradient of alpha-particles occurs. The existence of a TAE-gap is a necessary condition for the existence of a TAE-mode. Hence, the very low density of the TAE-gaps in the central plasma region indicates that this region has very few TAE-modes. Moreover, in this central region, the very few number of TAEs also have quite narrow radial mode width. If excited, such modes can only flatten the alpha-particle profile over a limited region corresponding to the radial mode width. In the area of larger magnetic shear, $r/a \geq 0.5$, the density of TAE-gaps increases significantly, so that the distance between the gaps becomes less than the width of the gaps. Global TAEs are formed in this case which are capable of transporting resonant alpha-particles over a significant radial distance. However, these TAEs are localized in the region, i.e., close to the plasma edge, with a smaller population of alpha-particles. The global character of such TAEs is therefore counter-balanced by a lower alpha-particle drive and a lower alpha-particle density.

Numerical modeling of TAEs

In order to assess both global linear stability of TAEs and nonlinear TAE evolution, all TAEs (there are 129 eigenmodes) with toroidal mode numbers $1 \leq n \leq 35$ were computed with the spectral MHD code MISHKA (Mikhailovskii et al., 1997). Among these modes, core-localized TAEs of both even and odd parity (i.e., with the same or opposite signs of the poloidal harmonics m and $m + 1$) were found, and a significant number of global TAEs were also identified. The linear alpha-particle drive was computed with the HAGIS code (Pinches et al., 1998) for every individual TAE, as shown in Fig. 13 (left). The damping of every TAE was then assessed with the CASTOR-K code (Borba et al., 2002), with the dominant damping being the thermal ion Landau resonance. The non-linear evolution of all TAEs with damping effects included, and interacting with the same alpha-particle population, was then computed with the HAGIS code. 2D contours of the saturated TAE amplitude are shown in Fig. 13 (right). It was found that TAEs reach amplitudes approaching $\delta B_r/B \approx 3 \times 10^{-4}$, with the core localized TAEs being the largest (Fitzgerald et al., 2016).

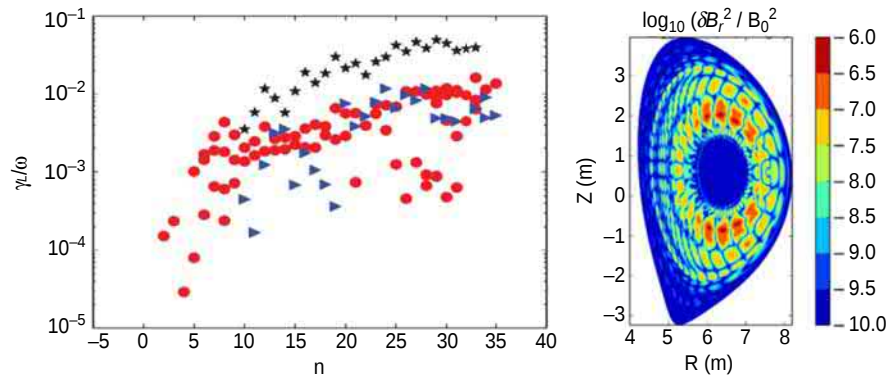


Fig. 13 Left: Linear growth rates for TAE modes in ITER computed by the HAGIS code (Pinches et al., 1998) by fitting to the first 20 wave periods in the absence of damping or coupling. The even parity core localized modes (black star) have higher growth rates than the odd parity core (blue triangle) and high shear TAEs (red circle). Right: 2D plot of computed saturated TAEs in the ITER baseline scenario (Fasoli et al., 2007). Here, non-linear saturation was computed of 129 TAEs with $n = 1-35$, including alpha particle drive, thermal plasma Landau damping, and radiative damping.

Prediction of a similarly low TAE saturation amplitude, i.e., $\delta B_r/B \sim 10^{-4}$, was independently obtained by (Schneller et al., 2016), where it was found that inclusion of Landau damping was essential to predicting the low saturation amplitude and the resultant small alpha-particle transport. Otherwise, nonlinear cascade effects were significantly exaggerated in the simulations.

At the low saturation levels derived from the simulations, radial transport of alpha-particles is limited to several centimeters near the location of the core localized modes. It should be emphasized, however, that the limited alpha-particle transport due to TAEs is partially due to the separation between core and edge TAEs, and that the analysis may be not so optimistic for plasmas with higher magnetic shear. Finally, the super-Alfvénic neutral beams and the potential error bar in damping predictions imply that some caution should be exercised in ruling out global alpha-particle redistribution in this scenario. Nevertheless, further simulations using the HAGIS code showed that global stochasticity effects, which produce large radial transport, only occur if the TAE amplitudes in Fig. 13 (right) increase by a factor of ~ 50 over the computed saturation levels (Fitzgerald et al., 2016). With such a significant enhancement factor required for global stochasticity, one could expect with confidence that TAEs will not cause a significant alpha-particle transport in the specific case of the ITER Q = 10 baseline scenario.

Both simulations acknowledge that the effect of Coulomb scattering plays a rather small role in AE driven relaxation of fast ions.

Acknowledgments

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Magnetic Confinement Fusion—Plasma Theory: Heating and Current Drive

R.J. Dumont, CEA, IRFM, Saint-Paul-lez-Durance, France

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Glossary

Alfvén velocity The group and phase velocities associated to Alfvén waves are of the order of the Alfvén velocity, $v_a \equiv B_0 / (\mu_0 n_i m_i)^{1/2}$, with B_0 the confining magnetic field, μ_0 the vacuum permeability and n_i (resp. m_i) the ion density (resp. mass). This ratio reflects the antagonistic effects of the magnetic field line tension on the one hand, and of the ion inertia on the other hand.

Alfvén waves A kind of magnetohydrodynamic wave characterized by a periodic oscillation of the plasma ion drift motion associated to a perturbation of the magnetic field. It is characterized by a frequency much lower than the ion cyclotron frequency. Alfvén waves play an important role in magnetic fusion plasmas because they can interact efficiently with energetic ions and, in some situations, induce a radial displacement of these particles (Gorelenkov and Sharapov, 2021).

Boltzmann equation A kinetic equation describing statistical properties of a thermodynamic system out-of-equilibrium. In practice, the generic form of the collision term appearing in this equation prevents it from being directly applied to fully ionized plasmas in which long-range Coulomb collisions dominate.

Bootstrap current A contribution to the toroidal plasma current resulting from the fact that the plasma pressure is maximal near the plasma center and decreases towards the edge. Because of this non-uniformity, the friction between trapped electrons and passing electrons results in a net transfer of parallel momentum from the former to the latter, inducing a finite parallel current (Helander and Sigmar, 2005). In tokamaks, the (beneficial) bootstrap contribution to the total current can be quite significant (typically $\sim 50\%$ in ITER, depending on the regime of operation).

Coulomb logarithm This quantity appears in the theory of Coulomb collisions taking place between the charged particles comprising a plasma. Denoting Λ the approximate number of particles contained in the Debye sphere, its logarithm, $\ln(\Lambda)$, is much larger than unity in weakly coupled plasmas, in which collective effects dominate (Goldston and Rutherford, 1995). Fusion plasmas are examples of weakly coupled plasmas and it is found that $\ln(\Lambda)$ ranges from ~ 17 (current fusion experiments) to ~ 18 (fusion reactors).

Fokker-Planck equation An equation deduced from the Boltzmann equation assuming that long-range collisions dominate, as is appropriate in hot magnetic fusion plasmas. The ensuing accumulation of multiple small angle deviations for a given particle results in a diffusion equation for the distribution function in velocity space.

keV In magnetic fusion plasmas, it is customary to express energies in terms of electronvolt (eV), kilo-electronvolt (keV) or mega-electronvolt (MeV). 1 eV is approximately equivalent to 1.602176×10^{-19} J. By extension, this unit is also employed for plasma temperatures: 1 eV is approximately equivalent to 1.160450×10^4 Kelvin.

Quasilinear equation A kinetic equation describing the secular (i.e. long duration) effect of the interactions taking place between an electromagnetic wave and the plasma particles. It is obtained by averaging the Vlasov equation over a time much longer than the wave period, and over a space region with a large extension compared to the wavelength.

Trapped/passing particles In a magnetic confinement device, the confining magnetic field B_0 is non-uniform. The velocity vector, $\mathbf{v}$, of a given particle of mass, m , can be written in terms of parallel ($v_{\parallel}$) and perpendicular ($v_{\perp}$) components with respect to the magnetic field direction. Magnetic confinement imposes that the particle kinetic energy, $E = m(v_{\parallel}^2 + v_{\perp}^2)/2$, and its magnetic moment, $\mu = mv_{\perp}^2/(2B_0)$, be conserved to lowest order. By manipulation of these two equations, it can be demonstrated that particles can be classified into two main categories: some are characterized by trajectories with $v_{\parallel} \neq 0$ at any point, whereas others periodically see their parallel velocity cancel. The former are passing particles, and perform complete toroidal revolutions. The latter, on the other hand, travel back and forth in the toroidal direction and are called trapped particles. In particular, trapped electrons do not carry any toroidal current.

Vlasov equation A kinetic equation obtained by neglecting inter-particle correlations in the kinetic description of the plasma, i.e. assuming that all particle interactions are mediated by a self-consistent average electromagnetic field satisfying Maxwell's equations. Since collisions are neglected in this process, this equation is applicable to timescales much smaller than the typical collision time. It is sometimes referred to as the collisionless Boltzmann equation.

WKB (Wentzel-Kramers-Brillouin) approximation A special case of multiple scale analysis. In the context of waves propagating in plasmas, it is relevant when the plasma varies on a space scale much larger than the wavelength. In this case, at a given position x , the electromagnetic field can be written in the form of $E(x) = E_0(x)\exp(iS(x))$, where $E_0(x)$ is assumed to vary slowly compared to the phase $S(x)$. It is applicable outside of wave cut-offs or resonances (Swanson, 2003).

Introduction

The principle of heating and current drive of magnetic confinement fusion plasmas consists of transferring energy produced by a power source to the plasma species. Whereas heating results in an increase of their temperature, more sophisticated schemes, among which current drive, are used as a means to control the plasma and ensure its state remains optimal with respect to fusion power production. In present day devices, the electrical grid typically supplies this power. On the other hand, the burning plasmas contained in future reactors will essentially be self-sufficient in terms of power: one fifth of the energy produced by the D-T reactions comes in the form of kinetic energy of ^4He (alpha) particles, which is much larger than the average energy of any other species present in the plasma. Upon collisional slowing-down on the background species, the alpha particles will yield their energy to the plasma. A fraction of the fusion power produced in-situ will also be recirculated to feed external generators used mainly for plasma control. These considerations apply to the two main classes of reactors envisaged today, i.e. tokamaks (Zohm, 2021) and stellarators (Yamada, 2021). In modern fusion installations, there exist two main classes of external power sources: (1) radio-frequency systems, (2) neutral beam injection systems.

Every radiofrequency (RF)-based heating and current drive method relies on an external power source on the one hand and an adapted antenna located near the plasma edge on the other hand. How this power travels between the source and the antenna depends on the frequency range. In an optical-like description of the process, in all cases, an electromagnetic wave eventually leaves the antenna and propagates to the plasma core where it is absorbed by the plasma species by two main non-collisional mechanisms: Landau damping and cyclotron damping. These mechanisms result in the direct damping of the wave power by one or several thermal species that will be heated as a result, and/or the creation of energetic particle populations (i.e. particles with typical energies much larger than the thermal energy characterizing the background plasma species) which eventually heat the plasma by collisional relaxation.

Neutral beam injection (NBI) consists of producing high-energy ions in an external accelerator and neutralizing them at the end of the acceleration process. These neutral particles, being immune to the effects of the magnetic trap in which the thermal species are confined, can penetrate deep in the plasma. In this process, the vast majority of the energetic particles are re-ionized. Since the re-ionization occurs when the injected particle energy is much larger than that of the thermal species, the result is a population of energetic ions that eventually slow down through collisions with the background particles, to which they transfer their energy.

Finally, fusion products are usually born at very high energies, the most prominent being the alpha particles from D-T reactions born around 3.5 MeV. The relaxation process is essentially similar to the one taking place for NBI ions, i.e. results in a transfer of the fusion product energy to the background electrons and/or fuel ions and the subsequent heating of the plasma. A plasma can be qualified as “burning” when this self-heating process provides a source sufficient for the fusion reactions to be maintained with little or no input required from auxiliary power sources.

Current drive is essential in tokamaks, which require the presence of a poloidal magnetic field produced by the toroidal plasma current (Wesson, 2011). This current can be induced in the plasma by the central solenoid, part of the poloidal circuit (ohmic current)—an inherently non-stationary process—or self-generated by the plasma pressure gradient (*bootstrap current*). The ohmic and bootstrap currents generally need to be supplemented by additional sources to achieve stationary plasma discharges. Furthermore, current profile control is known to be beneficial in terms of plasma stability and performance (Jenko, 2020), but requires

a level of flexibility only achievable with external current drive sources. Some NBI systems are flexible enough for part of the energy to be injected parallel to the magnetic field, which has the effect of driving toroidal current, albeit with a limited flexibility in terms of current profile control. RF wave systems have also been applied for non-inductive current drive for several decades, and are planned to be used for this application in next-step devices. In stellarators, the magnetic field, including its poloidal components, is entirely generated by the coil system, resulting in more modest requirements in terms of external current drive. Nevertheless, the capability to drive non-inductive current is still desirable to compensate for the self-generated bootstrap current.

A global summary of the process of heating and current drive is shown in Fig. 1.

Kinetic theory elements

Heating and current drive results in modifications of the velocity distribution function, f_s , of a given plasma species, denoted s . From this quantity, it is possible to construct the so-called fluid moments (Sarazin, 2021), which include the density, $n_s = \int d^3v f_s$, the current, $j_s = q_s \int d^3v v f_s$, and the temperature, $T_s = \int d^3v m v^2 f_s / 2$. To lowest order, plasma heating and current drive from external sources results in modifications of these three fluid quantities. Clearly, the knowledge of f_s is therefore the first and most important element of the description of the physics processes involved. Furthermore, the RF wave heating and current drive processes involve wave-particle resonances which are dependent on the particle velocities on an individual basis (as opposed to the fluid velocity, which is a macroscopically averaged quantity). As a result, the kinetic theory is the appropriate framework in which these phenomena must be described.

Fokker-Planck equation

The *Fokker-Planck equation* is the classical tool employed to describe heating and current drive processes (Brambilla, 1998). It is derived from the more generic kinetic *Boltzmann equation* assuming that Coulomb collisions result in multiple small angle deviations for a given particle, a valid assumption in the context of magnetically confined fusion plasmas. Formally, this equation takes the form of an evolution equation for the distribution function, f_s , in velocity space under the effects of collisions, particle source and particle losses:

$$df_s/dt = C(f_s) + S_s - L_s. \quad (1)$$

S_s is the particle source (relevant in the case of NBI and/or fusion-born alphas, in which new particles are permanently injected in the system) and L_s is a loss term describing the particles leaving the system by various mechanisms (orbit losses, charge exchange ...), while $C(f_s)$ represents the effect of collisions, and can be written in the form:

$$C(f_s) = -\nabla_v \cdot (\langle \Delta v \rangle / \tau_c f_s) + 1/2 \cdot \nabla_v \nabla_v \cdot (\langle \Delta v \Delta v \rangle / \tau_c f_s) \quad (2)$$

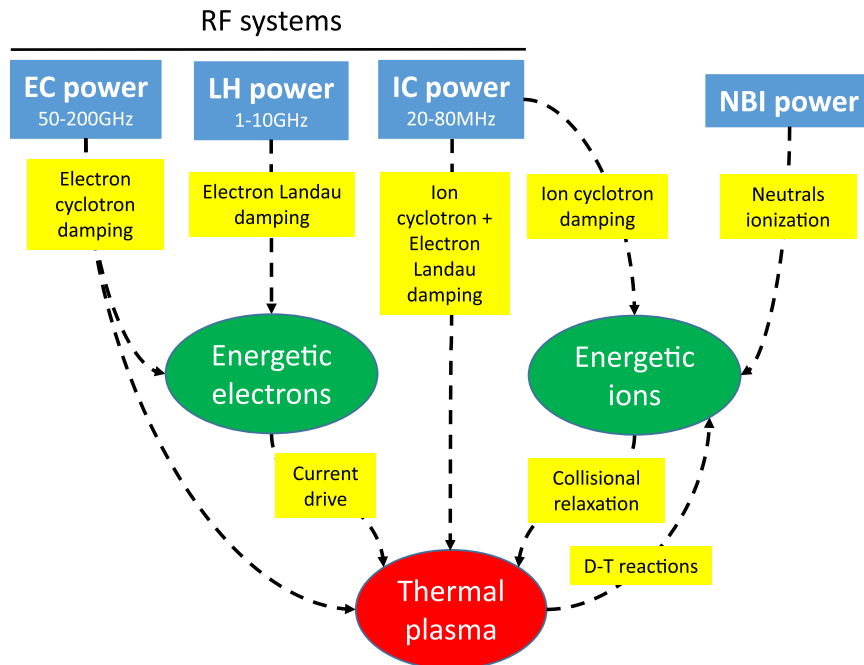


Fig. 1 Summary of the main heating and current drive sources in modern magnetically confined fusion plasmas.

τ_c is a collision time. ∇_v is a differential operator with respect to components of the velocity, $\mathbf{v}$. In Eq. (2), the first term represents a convection process, whereas the second term represents a random walk. The Fokker-Planck equation therefore describes a convection-diffusion process taking place in velocity space. The quantity $F \equiv \langle \Delta \mathbf{v} \rangle / \tau_c$ is called the coefficient of dynamic friction, whereas $D \equiv \langle \Delta \mathbf{v} \Delta \mathbf{v} \rangle / \tau_c$ is the diffusion tensor.

In its generic form—consistent with the assumptions made to obtain the Fokker-Planck equation, F and D involve all species present in the plasma, making C a non-linear operator which can be quite complex to evaluate. It becomes much more tractable when assuming that the background species, i.e. all species but the heated one(s) have Maxwellian velocity distributions, an assumption often legitimate in the context of heating and current drive in magnetic fusion plasmas.

Collisional processes

As described previously, the Fokker-Planck equation (1) describes the collisional processes governing the relaxation of any energetic particle population, and hence the energy transfers taking place between this population and the background plasma species (Goldston and Rutherford, 1995). This equation can be solved efficiently on any modern computer using various numerical methods, typically in a Monte Carlo form, or using finite differences/elements. However, much insight into the relaxation process itself can be gained by examining the relaxation of a single test ion on the background plasma species, which are assumed fixed. From the coefficients $\langle \Delta \mathbf{v} \rangle$ and $\langle \Delta \mathbf{v} \Delta \mathbf{v} \rangle$ derived to construct the collisional term (Eq. 2), trajectories in velocity space of an ion with charge number, Z_f , and mass number, A_f , injected (or born) at a given velocity, V_f , and undergoing the influence of Coulomb collisions can be computed.

In the range of energies relevant to energetic ions (NBI or fusion-born), V_f always satisfies $v_{th,i} \ll V_f \ll v_{th,e}$, with $v_{th,i}^2 \equiv 2k_B T_i / m_i$ and $v_{th,e}^2 \equiv 2k_B T_e / m_e$ the (squared) thermal velocities of background ions (of mass m_i) and electrons (of mass m_e). T_e and T_i are the corresponding temperatures and k_B is the Boltzmann constant. Using this assumption, the dominant collisional processes are found to be (1) a decrease of the test ion velocity resulting from the collisional friction against the background species; (2) a scattering of the test ion pitch-angle (relative to the magnetic field), i.e. a modification of the direction of its velocity with respect to its initial velocity.

A closer examination of the magnitude of these processes allows three important quantities to be identified: (1) the Spitzer slowing-down time (in seconds), $\tau_{se} \equiv 6.32 \times 10^{14} (A_f / Z_f^2) (T_e^{3/2} / n_e) \ln(\Lambda)$, with $\ln(\Lambda)$ the *Coulomb logarithm* and T_e (resp. n_e) the background electron temperature in eV (resp. density in m^{-3}); (2) the critical energy, $E_c \equiv 14.8 (Z_i^2 / A_i) A_f T_e$, with A_i and Z_i the background ion mass and charge number; (3) the thermalization time, $\tau_{th} \equiv \tau_{se} \ln(1 + (E_f / E_c)^{3/2}) / 3$. The Spitzer time describes the rate of change of the test particle velocity caused by the friction on background electrons, i.e. $dv_f/dt = v_f / \tau_{se}$ (i.e. assuming only electrons are involved in the slowing-down process). The critical velocity corresponds to the limit below which pitch-angle scattering and drag by background ions exceeds the scattering and drag caused by electrons. When the test ion reaches E_c , both the background electrons and ions contribute equally to its slowing down. However, the fact that the masses of the test and background ions are not very different also induces significant pitch-angle scattering by the latter. The corresponding decrease in the test particle energy does not result in background plasma heating, but rather in an increased spread in the energetic population velocities. Finally, the thermalization time corresponds to the delay for the test ion to reach the thermal velocities. It should be noted that τ_{th} and τ_{se} can be quite different, as illustrated by the data shown in Table 1.

Quasilinear description of RF heating

The first step towards describing plasma heating by RF waves is to solve the wave equation for the wave electric field, $\mathbf{E}$. The wave magnetic field, $\mathbf{B}$, can always be deduced from $\mathbf{E}$ using Maxwell's equations. Assuming a time-harmonic field varying at an angular frequency, ω (imposed by the generator), i.e. $\mathbf{E} \propto \exp(-i\omega t)$, with t the time, the wave equation is a straightforward consequence of Maxwell's equations, and is written here in the symbolic form:

$$\nabla \times \nabla \times \mathbf{E} - \omega^2 / c^2 (\mathbf{E} + i \mathbf{J}_p / \omega \epsilon_0) = i \omega \mu_0 \mathbf{J}_{ant}. \quad (3)$$

In this expression, ∇ is a differential operator (in real space), μ_0 is the vacuum permeability, ϵ_0 the vacuum permittivity, and c is the speed of light. $\mathbf{J}_{ant}$ represents the boundary condition imposed by the antenna. Despite its apparent simplicity, this equation is quite complicated because the response plasma current, $\mathbf{J}_p = \mathbf{J}_p(\mathbf{E})$, has a complex, temporally and spatially non-local, dependence

Table 1 Main parameters relevant to the collisional relaxation of NBI ions in JET D, JET DT, ITER D, and ITER DT plasmas.

Background plasma/test particle	Initial energy (keV)	Critical energy (keV)	Spitzer time (s)	Thermalization time (s)	Fraction of power to ions/electrons
JET D/D NBI	100	187	1.45	0.16	0.87/0.13
JET DT/DT NBI	100	206	1.81	0.18	0.89/0.11
ITER D/D NBI	1000	466	2.64	1.25	0.53/0.47
ITER DT/D NBI	1000	413	2.64	1.34	0.50/0.50

on the wave electromagnetic field. A first simplification occurs assuming that this dependence is linear, a hypothesis generally valid in the context of plasma heating and current drive, where the electromagnetic field is a perturbation for the magnetically confined plasma.

Another source of difficulty related to Eq. (3) is the fact that J_p depends itself on the distribution function of the various plasma species, including the species which directly absorb the injected power. In other words, it is often necessary to solve the wave equation and an equation to describe the modification of the distribution function in a self-consistent fashion. This two-step approach relies on the fact that the timescales related to the wave propagation and damping are small compared to the timescales related to the secular modifications of the distribution function and Coulomb collisions, which correspond to the macroscopic modifications of the plasma caused by the wave power.

Including RF wave effects on the distribution function requires additional assumptions with respect to those made to derive the Fokker-Planck equation (Eq. 1). The *quasilinear equation* is obtained from the *Vlasov equation* assuming that the non-linear effect of the wave on the distribution function is the product of interactions taking place between the wave field, as deduced from the linearized wave equation (Eq. 3) on the one hand, and of the linear plasma response on the other hand, hence the name quasilinear. This procedure requires an averaging over space (resp. time) scales larger than the wavelength, λ (resp. wave period τ), to eliminate all terms varying on the small space scale, λ , and fast time scale, τ . This quasilinear equation—customarily also referred to as Fokker-Planck equation—therefore applies to F_s , the time and space-averaged version of f_s . Following this procedure, it can be written in the form:

$$dF_s/dt = C(F_s) + Q(F_s) + S_s - L_s. \quad (4)$$

Q is the quasilinear operator, and represents the effect of the waves interacting with the particles of species s on the corresponding distribution function, F_s . In the framework of the quasilinear theory, this term also takes the form of diffusion/friction term involving a quasilinear diffusion tensor, D_{ql} :

$$Q(F_s) = \nabla_v \cdot (D_{ql} \cdot \nabla_v F_s) \quad (5)$$

Depending on the level of sophistication of the description used, Q can have a rather complicated form. Important features of the quasilinear operator can nevertheless be isolated by writing any of its components in the symbolic form:

$$D_{ql} = A \sum_p \left| d^{(p)}(E) \right|^2 \delta(\omega - p\Omega_{cs} - k_{||}v_{||}) \quad (6)$$

In this expression, A is a constant. $d^{(p)}(E)$ is a differential operator acting on the electromagnetic field and $k_{||}$ is the projection of the wave vector in the direction parallel to the magnetic field lines. These two elements are essentially representative of the wave field. $\Omega_{cs} = q_s B_0 / m_s$ is the (algebraic) cyclotron frequency, i.e. the frequency of the cyclotron motion of the considered particles (with charge, q_s , and mass, m_s) around the magnetic field line. Ω_{cs} and $v_{||}$, the particle velocity along the field line, are characteristics of the interacting particles trajectories. p , the harmonic number, is an integer which can have either sign, or be 0. $\delta(\omega - p\Omega_{cs} - k_{||}v_{||})$ is non-zero only when the particles are resonant with the wave, i.e. when the wave-particle resonance condition, $\omega = p\Omega_{cs} + k_{||}v_{||}$, is satisfied.

RF wave propagation and damping

In RF heating processes, generators located at some distance from the plasma confinement device generate oscillating currents. Depending on the frequency range, the corresponding power is transported to an antenna located inside the vacuum vessel, usually close to the plasma edge (i.e. within several cm). The wave coupling is the process by which the power emanating from the radiating structure in the form of a vacuum electromagnetic wave is transferred to the plasma species. This electromagnetic wave can excite plasma waves of various types: (1) Electromagnetic waves are waves for which the propagation does not rely on the plasma species. (2) Electrostatic waves are characterized by an oscillating electric field in the direction of propagation. This electric field induces an oscillating plasma current and associated space charge, which itself is responsible for a response electric field further away in the plasma. (3) Magnetohydrodynamic (MHD) waves are characterized by an oscillating magnetic field which induces a plasma current, itself inducing a magnetic field further away in the plasma. It should be emphasized that such a separation is an idealized view: actual plasma waves generally involve several of these processes at the same time (Stix, 1992). As a result, the separation between the reactive part of the wave power, reversibly exchanged between the electromagnetic field and the plasma species and therefore linked to wave propagation, and the resistive part of the wave power, irreversibly transferred to the plasma species and therefore responsible for plasma heating, is in practice a delicate task (Swanson, 2003).

Wave propagation

There exist several methods to solve the wave equation, Eq. (3). When the wavelength λ remains much smaller than the typical space scales characterizing the plasma equilibrium quantities, L_B , the *WKB approximation* is applicable, allowing Eq. (3) to be solved by ray-tracing or beam-tracing (quasi-optical methods). On the other hand, regimes in which λ becomes comparable to, or larger than,

L_B , or in which the wave undergoes reflection or resonance phenomena during its propagation make it necessary to solve Eq. (3) directly, by resorting to a full-wave approach. Although more intensive numerically, modern computers allow this type of calculation to be performed in a routine fashion, including for higher frequency waves.

However, in all cases, considerable insight can be gained by analyzing the dispersion relation corresponding to the wave equation. This relation is obtained by assuming that the plasma is stationary and homogeneous. Whereas the former assumption is relatively safe in the context of plasma heating by RF waves, the homogeneity hypothesis is often questionable. Despite this limitation, the dispersion relation is always very useful to analyze wave propagation in plasmas. Writing the local electric field as a monochromatic plane wave characterized by wavevector, $\mathbf{k}$, i.e. $\mathbf{E}(\mathbf{r}, t) \propto \exp(i(\mathbf{k} \cdot \mathbf{r} - \omega t))$, the dispersion relation takes the generic form, $D(\omega, \mathbf{k}) = 0$, and therefore expresses an implicit relation between the wave frequency, ω , and wavevector, $\mathbf{k}$, or equivalently the refractive index, $N \equiv kc/\omega$. Usually, multiple solutions to $D(\omega, N) = 0$ at a given point in the medium are possible, indicating the potential presence of several modes, each one characterized by a different dependence of N on the plasma parameters (density, magnetic field ...). Once a given solution (or mode) is chosen, the dispersion relation can be used in the wave equation to deduce the wave polarization, i.e. the ratio between the various components of the wave electromagnetic field. Among these components, the parallel, $E_{\parallel}$, left-handed, E_+ , and right-handed, E_- , turn out to be of particular importance because they physically correspond to a wavefield oscillating in a direction parallel to the magnetic field ($E_{\parallel}$), and rotating in the same direction as ions (E_+) or electrons (E_-).

In magnetically confined plasmas, the projection of N along the magnetic field, $N_{\parallel}$, is essentially determined by the antenna geometry, and usually does not vary much as the wave propagates. The perpendicular component, $N_{\perp}$, on the other hand, largely depends on the plasma parameters. It is instructive to investigate wave propagation in terms of $N_{\perp}$ in the context of the cold plasma theory, in which all species are assumed to have zero temperature. In such plasmas, the dispersion relation can be written as $D(\omega, N_{\perp}) = A(x)N_{\perp}^4 + B(x)N_{\perp}^2 + C(x)$, where the coefficients A , B and C are function of the local plasma parameters (Stix, 1992), parametrized here by the variable x . The consequence is that only two modes can exist at a given point, x , in a cold plasma. Furthermore, there may exist points verifying $N_{\perp}^2(x) = 0$, called cut-offs. In this situation, the wave can only propagate up to the cut-off where it is reflected. Points such that $N_{\perp}^2(x) \rightarrow \infty$ are called resonances.

The inclusion of finite temperature effects in the model regularizes the singular behavior observed at cold resonances. Wave absorption, a phenomenon absent in the cold plasma theory, takes place and results in an irreversible power transfer through the resonance mechanism involving the wave field and the particle motion. Retaining thermal effects also has the consequence of allowing more than two modes to propagate in the plasma. In particular, it is possible to encounter situations for which two modes characterized by $N_{\perp 1}$ and $N_{\perp 2}$ propagating independently reach a point such as $N_{\perp 1} = N_{\perp 2}$. In this case, part of the wave power in a given mode can be partially transferred to another mode, in a process called mode-conversion.

In magnetic fusion plasmas, these three processes (i) wave cut-off, (ii) resonant wave absorption and (iii) mode conversion, are quite prevalent and therefore need to be included in any description of wave propagation and absorption (Stix, 1992).

Wave absorption

As discussed previously (see the “Quasilinear description of RF heating” section), wave-particle resonance occurs when parameters characterizing a wave (ω , $k_{\parallel}$) and parameters characterizing a given plasma species are such that a resonance condition is fulfilled. $k_{\perp}$, on the other hand, is determined by the dispersion relation. Typically, such a resonance occurs when the wave frequency matches the local gyration motion of electrons or ions, $\omega = \Omega_{cs} + k_{\parallel}v_{\parallel}$ (cyclotron resonance), one of its harmonics, $\omega = p\Omega_{cs} + k_{\parallel}v_{\parallel}$ (harmonic cyclotron resonance, with $p > 1$) or the parallel particle motion, $\omega = k_{\parallel}v_{\parallel}$ (Cerenkov resonance). A synchronization process takes place between the wave-field and the particle motion. However, this process is reversible by nature. The transformation of reactive power to resistive power through the wave-particle resonance requires one or several mechanisms responsible for decorrelating the field oscillation from the plasma particle response. These processes are often referred to under the generic term of phase-mixing mechanisms. One such decorrelation process is the Coulomb collisions themselves. By their stochastic nature, collisions are clearly able to destroy any wave/particle correlations. However, experimental evidence pointed to the possibility of wave damping even in collisionless plasmas, triggering an intense effort to interpret these results and identify non-collisional damping mechanisms.

Landau damping

Non-collisional damping has been the center of vigorous debates. Several mathematical derivations have been put forward to describe this phenomenon. However, L. Landau is to be credited for the unambiguous explanation of this effect in 1946 (Landau, 1946; Ryutov, 1999). The mathematical demonstration is rather subtle and involves causality as a key element. The underlying physics idea can be understood by considering a particle with initial parallel velocity $v_{\parallel 0}$ in the presence of an electromagnetic field characterized by phase velocity, $v_{\phi} = \omega/k_{\parallel}$. A resonance occurs when the Cerenkov resonance condition, $\omega = k_{\parallel}v_{\parallel 0}$, is fulfilled. More precisely, during the process, particles with velocities near the wave phase velocity feel the wave effect, and see their motion modified in the synchronization process with the field. It can then be shown that in this case, the particle is either accelerated or decelerated with equal probability. As such, no net acceleration is imparted by the wave power. However, when considering an assembly of particles, the slower particles, i.e. those with $v_{\parallel 0} < \omega/k_{\parallel}$, which are accelerated, tend to feel the wave resonance with more intensity. On the other hand, those decelerated tend to resonate even less with the wave. Statistically, therefore, wave power will be employed to accelerate slower particles. The symmetrical situation involves faster particles, i.e. those with $v_{\parallel}$

$0 > \omega/k_{\parallel}$. In contrast with the previous case, the decelerated particles will become more resonant than the accelerated ones. As a result, kinetic energy from the particles will tend to amplify the wave. However, fusion plasmas are typically characterized by Maxwellian distributions, which have a decreasing dependence with parallel velocity, $v_{\parallel}$. The consequence is that slower particles are more numerous than faster particles, so that the net result of the two processes described previously is an absorption of the wave by the particles. This explains that γ , the wave damping rate, which characterizes the field attenuation with time, i.e. $E \propto \exp(-\gamma t)$, depends linearly on the slope of the distribution function considered at the wave-particle resonance, i.e. $\gamma \propto df_s/dv|_{v_{\parallel}=\omega/k_{\parallel}}$. The result of this interaction on the distribution function of the interacting species is described by Eq. (4), and takes the form of a local flattening in velocity space around the resonance.

Cyclotron damping

Cyclotron damping is related to the process of Landau damping. The resonance condition, however, involves the particle gyromotion, and can be written as $\omega = p\Omega_{cs} + k_{\parallel}v_{\parallel}$, with p a non-zero integer. The cyclotron resonance occurs when the wave frequency matches the cyclotron frequency Ω_{cs} ($p = 1$), i.e. the frequency characterizing the particle gyro-motion or one of its harmonics ($p > 1$). The last term, $k_{\parallel}v_{\parallel}$, is the Doppler shift correction caused by the parallel motion of the particles along the field lines. Although it bears some similarities with Landau damping, the dependence of the cyclotron frequency with the confinement magnetic field magnitude, B_0 , induces essential differences in the resonance phenomenon. By construction, B_0 has a spatial dependence, proportional to $1/R$ (the distance to the torus revolution axis) in tokamaks—but more complex in stellarators. The result is that the resonance only occurs in a spatially localized region. The particles having periodical motion, the resonance occurs at multiple times as the particle travels through the plasma, which introduces another possible decorrelation mechanism in the system. The other fundamental difference is that the cyclotron motion is, by nature, perpendicular to the magnetic field. The resulting velocity-space diffusion occurs mainly in the perpendicular direction, which tends to increase the perpendicular particle velocities. The heated distribution functions in velocity space can therefore exhibit pronounced anisotropies, with a perpendicular energy content often significantly exceeding the parallel energy content.

Wave-induced current drive

The process of RF-induced current drive (CD) is inherently linked to electron Landau damping. A wave characterized by angular frequency, ω , and parallel wavevector, $k_{\parallel}$, can resonate with electrons having a parallel velocity fulfilling the Cerenkov resonance condition, $v_{\parallel} = \omega/k_{\parallel}$. The consequent absorption of an energy quantum, $\Delta E = \hbar\omega$, by an electron results in an increase of its parallel momentum, $\Delta p_{\parallel} = \hbar k_{\parallel} = k_{\parallel}\Delta E/\omega$. Therefore, an injected wave characterized by an asymmetric spectrum in terms of $k_{\parallel}$ will impart momentum to the electrons, thereby inducing a net parallel current.

As illustrative as this physics picture may be, it must be stressed that the mechanism of momentum transfer from the wave to the plasma described previously is not the dominant one in RF current drive applications. Indeed, it turns out that even waves imparting only perpendicular momentum to the electrons can drive current in a relatively efficient fashion. This results from the fact that wave-induced current drive relies for the most part on the variation of the plasma collisionality with the energy of resonant particles. Therefore, creating a population of electrons characterized by an excess of perpendicular energy can be exploited for current drive purpose, as long as the distribution function is asymmetric for the parallel velocity. This process is known as the Fisch-Boozer mechanism (Fisch, 1987).

A figure of merit of any current drive method is the amount of current driven, Δj , for a given amount of power expended, p . If we consider an electron receiving parallel momentum, $\Delta p_{\parallel}$, from the wave, the corresponding current increase is $\Delta j = -e\Delta p_{\parallel}/m_e$, with e the absolute value of the electron charge and m_e the electron mass. Collisions, on the other hand, tend to oppose this momentum increase, a process which can be characterized by the counteracting force, $F = -v_e\Delta p_{\parallel}$, with v_e an effective collision frequency. By identifying the wave power spent to the work variation required to maintain the current drive process, we obtain the elemental current drive efficiency:

$$\Delta j/p = e/m_e v_e v_{\parallel}. \quad (7)$$

In order to maximize the efficiency, it would therefore seem natural to target thermal electrons, characterized by relatively low values of $v_{\parallel}$. Early current drive schemes, using *Alfvén waves*, were based on this idea. However, a large fraction of thermal electrons is trapped in the magnetic mirrors constitutive of the magnetic confinement structure, and therefore do not complete full revolutions in the toroidal direction. Consequently, they are unable to carry any parallel current and the resulting efficiency is too low to be of any interest. By contrast, in the range of velocities characterizing superthermal (or energetic) electrons, the effective collision frequency, v_e , varies as $(1/v_{\parallel})^3$, so that by virtue of Eq. (7), the efficiency increases with $v_{\parallel}^2$. Finally, superthermal electrons also suffer less from trapping effects than thermal electrons. As a result, modern RF-based current drive schemes, i.e. those exploiting lower hybrid and electron cyclotron waves, are based on energetic electrons.

Finally, it should be noted that whereas Eq. (7) is interesting from a physics standpoint, it is hardly adequate to characterize the global efficiency of a given CD system, i.e. to obtain an estimate of the level of driven current corresponding to the power available on a given fusion device. For this purpose, it is worthwhile realizing that the absorbed power (in W) scales as $P_{abs} \sim 2\pi R \pi a_0^2 p_{abs}$, whereas the total current (in A) scales as $I_{CD} \sim \pi a_0^2 j$, with R the major radius, a_0 the minor radius, p_{abs} the power density (in

W m^{-3}) and j the current density (in A m^{-2}). Accordingly, a useful quantity is the normalized current drive efficiency, a global figure of merit valid for any CD scheme and defined as $\eta_{\text{CD}} \equiv n_e R I_{\text{CD}} / P_{\text{abs}}$, with n_e the appropriately averaged electron density.

Plasma heating and current drive by energetic ions

Regardless of the heating method employed, collisions often play an essential role since they are responsible for the transfer of power between the energetic particles and the thermal plasma species. Indeed, one of the most prevalent processes in modern heating and current drive applications involves the collisional relaxation of energetic ions, injected by the NBI system, born from fusion reactions or generated by RF acceleration of initially thermal species. The transfer of energy occurring between these energetic ions and the background plasma results from the various collisional processes at play. Their respective influence depends on the initial energies of the injected particles, E_f , on the Spitzer slowing-down time, τ_{se} , and on the critical velocity, E_c . It must be pointed out that energetic ions (NBI-driven, RF-driven or fusion products) can trigger instabilities, potentially inducing a radial redistribution of these ions. The impact of these processes on heating, current drive efficiency and fusion performance, as well as control schemes aimed at minimizing them (including using RF waves), is currently an active topic of experimental and numerical studies to prepare for the operation of next-step devices (Gorelenkov and Sharapov, 2021).

Neutral beam heating and current drive

Neutral beam injection is an auxiliary heating method which involves the production of high-energy ions in an accelerator, and their neutralization as they are ejected towards the confined plasma. These neutral particles are immune to the effects of the magnetic field in which the plasma is confined, and can thus penetrate to the plasma core. They are progressively re-ionized as they travel through the hot plasma. Since they are injected at an energy much larger (typically in the range 50–150 keV in ongoing devices, and 1 MeV in ITER) than the typical confined plasma particle energies, they slow down through collisions with these particles. Adapted numerical codes (Schneider et al., 2011) taking into account the beam and plasma parameters are required to compute the neutral beam and corresponding power deposition. The NBI system installed in JET, as well as an illustration of a typical power deposition profile in the plasma is shown in Fig. 2.

Consistently with the collisional relaxation process discussed in the “Kinetic theory elements” section, the result of plasma heating by NBI ions depends on the background plasma and injected ion parameters. As an illustration, NBI ions with typical injection energies of $E_f \sim 100$ keV are used in JET and result in dominant thermal ion heating (see Table 1). In ITER, on the other hand, the plasma size and density make it necessary to resort to much higher beam energies to deposit the beam power in the plasma core. Negative NBI ions with $E_f \sim 1$ MeV have been adopted to fulfill this requirement, and tend to create a balanced heating on thermal electrons and ions, with little dependence on whether the background plasma consists of D ions only, or of mixed D-T fuel.

Depending on the NBI system geometry, in some situations, it is possible to inject energetic ions with a significant component of the velocity direction parallel to the confinement magnetic field, i.e. able to impart a significant amount of parallel momentum to the plasma. As a result, a finite level of toroidal current is driven, a scheme referred to as Neutral Beam Current Drive (NBCD). Assuming beam ions with mass, m_f , and density, n_f , the parallel momentum injected in the system is $m_f n_f \langle v_{\parallel} \rangle$, the average being performed on the (injected) energetic NBI ion distribution. Because of collisional relaxation, the beam ions will transfer this momentum to the background species through friction. Assuming, for simplicity a situation where $E_f \gg E_c$, this relaxation essentially takes place on electrons. The subsequent force on electrons in the parallel direction tends to increase their parallel momentum. In a stationary regime, the friction with thermal ions compensates this variation of parallel electron momentum. Taking into

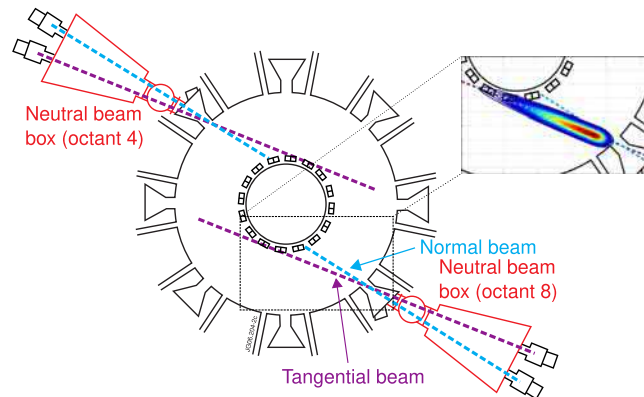


Fig. 2 NBI heating in the JET tokamak. This device is equipped with two separate NBI injectors located in octants 4 and 8, each consisting of eight beams arranged in a tangential bank and a normal bank (Source: EUROfusion). In the inset is shown an illustration of the power deposition profile in the plasma corresponding to the tangential bank of the injector located in octant 8.

account the respective relaxation times of these processes and plasma quasi-neutrality, it can be shown that the resulting driven current takes the form $J_{\text{NBCD}} = J_f(1 - Z_f/Z_i)$, with $J_f = Z_f n_f e \langle v_{\parallel f} \rangle$ the current carried by the beam ions, and Z_f (resp. Z_i) the charge number of beam (resp. thermal) ions. In many situations, Z_f and Z_i are comparable, which induces a large reduction in the NBI driven current because of the friction exerted by the background species on the energetic ions. A more accurate description of the process shows that the current drive efficiency predicted is generally larger than this analytical estimate. A first correction consists of taking into account trapped electron effects, which, in the case of NBCD, reduce the amount of electron current resulting from the cancellation of the beam current, and therefore entail an increase of the net efficiency (Fisch, 1987). The largest efficiency obtained to date is $\eta_{\text{CD}} = 0.155 \times 10^{20} \text{ A W}^{-1} \text{ m}^{-2}$ using 360 keV negative ion NBI in the JT-60U tokamak (Inoue et al., 2021). Predictions indicate that efficiencies of the order $\eta_{\text{CD}} \sim 0.2 - 0.4 \times 10^{20} \text{ A W}^{-1} \text{ m}^{-2}$ should be achievable in ITER (Gormezano et al., 2007).

Finally, it must be pointed out that NBI also plays an essential role in driving toroidal rotation of the plasma, which is known to have a favorable effect on confinement (Jenko, 2021) and MHD stability (Lao et al., 2021) in ongoing devices. Predictions indicate that this effect will be less significant in reactor-scale devices, however. A larger plasma volume and density translates into a larger moment of inertia, whereas the higher beam ion energy results in a reduction of the injected torque.

Burning plasmas

Burning plasmas consist of a mix of D-T fuel, which produces alphas born at 3.5 MeV able to heat the background species in a self-sufficient fashion (or nearly so). So far, only the tokamaks TFTR (USA) and JET (European Union) have operated with mixed D-T fuels. In these devices, the alpha power production remained small compared to auxiliary heating, making the effect of self-heating by alphas difficult to isolate. ITER is the first magnetic fusion device expected to contain a genuine burning plasma. In Table 2 are compared the situations in JET and in ITER with respect to plasma heating by the alphas. In the two cases, alphas predominantly heat the background electrons. However, the larger density (and thus collisionality) in ITER high-performance scenarios results in a more balanced heating partition between the electrons and the fuel ions, which is favorable for plasma fusion performance.

Plasma heating and current drive by radiofrequency waves

Ion cyclotron (IC) range of frequency

In the ion cyclotron range of frequencies (ICRF, 20–80 MHz), the antenna consists of a set of metallic straps enclosed in a box located at the edge of the magnetized medium and is capable of exciting a wave propagating in the plasma. In this frequency range, the system plasma-antenna must be considered as a global electrical circuit. This has the practical consequence that the antenna needs to be adapted to the plasma, by ensuring that their respective impedances match. In general, coupling ICRF power to the plasma is a delicate problem and a considerable effort is being devoted to this topic.

Examination of the dispersion relation in this range of frequencies reveals the existence of two waves, the so-called slow wave and fast magnetosonic wave. The slow wave, characterized by $N_{\perp}^2 \sim -\omega_{pe}^2/\omega_{ci}^2$ is evanescent with a small wavelength ($\lambda < 1 \text{ mm}$), which means that it is useless for plasma heating since it cannot propagate away from the antenna vicinity. Here, $\omega_{ps}^2 = n_s q_s^2/m_s \epsilon_0$ is the squared plasma frequency and $\omega_{ci} = Z_i e B_0/m_i$ is the cyclotron frequency of the main ions. It should be noted that the slow wave plays an important role in the problem of wave coupling, and in the development of performance limiting electrical sheaths, which can occur in some situations. The fast wave dispersion relation for a cold plasma is given by $N_{\perp}^2 \sim (N_{\parallel}^2 - R)(N_{\parallel}^2 - L)/(S - N_{\parallel}^2)$ with $R \sim -(c/v_a)^2 \omega_{ci}/(\omega - \omega_{ci})$, $L \sim (c/v_a)^2 \omega_{ci}/(\omega + \omega_{ci})$ and $S \sim -\omega_{pi}^2/(\omega^2 - \omega_{ci}^2)$, v_a being the Alfvén velocity. It is clear from these expressions that the fast wave exhibits two cut-offs: the right cut-off, $N_{\parallel}^2 = R$, the left cut-off, $N_{\parallel}^2 = L$; and a resonance called the Alfvén (or hybrid) resonance, $N_{\parallel}^2 = S$. The presence of the right cut-off implies that the fast wave does not propagate below a finite density, $n_e \sim 1 \times 10^{18} \text{ m}^{-3}$, which corresponds to a location significantly inside the plasma scrape-off layer. In other words, like the slow wave, the fast wave is evanescent near the plasma edge. However, with a space decay typically of the order of 10 cm, a significant fraction of the power can tunnel through the evanescence layer to reach the right cut-off, and propagate onward to the plasma core. The left cut-off, on the other hand, is located well inside the plasma core. Once coupled to the plasma, the fast wave is able to propagate between these two cut-offs, where it remains trapped until dissipation mechanisms damp the power.

The dominant damping mechanism of the fast magnetosonic wave is cyclotron damping by ions fulfilling the resonance condition, $\omega = p\omega_{ci} + k_{\parallel}v_{\parallel f}$. The corresponding generic form of the quasilinear diffusion coefficient (see Eq. 6) is

Table 2 Collisional plasma heating resulting from D-T alphas in JET and in ITER.

Background plasma/test particle	Initial energy (keV)	Critical energy (keV)	Spitzer time (s)	Thermalization time (s)	Fraction of power to ions/to electrons
JET DT/alpha	3500	330	0.73	0.86	0.17/0.83
ITER DT/alpha	3500	826	1.32	1.00	0.35/0.65

$$D_{ql} = A \sum_p |E_+ J_{p-1}(k_\perp \rho_i) + E_- J_{p+1}(k_\perp \rho_i)|^2 \delta(\omega - p\omega_{ci} - k_\parallel v_{\parallel i}), \quad (8)$$

with E_+ (resp. E_-) the left-handed (resp. right-handed) component of the electric field. J_p is the Bessel function of order p and $\rho_i = v_\perp / \omega_{ci}$ the (heated) ion Larmor radius. This expression shows that to lowest order, damping is determined by the left-handed component of the electric field, rotating in the ion direction. Furthermore, for small arguments, $J_p(x) \sim x^p$. In other words, fundamental heating ($p = 1$) is in principle very efficient since D_{ql} is at lowest order independent of $k_\perp \rho_i$. From a wave propagation standpoint, the situation described previously would seem to be ideal, since the cyclotron layer $\omega = \omega_{ci}$, around which the fundamental damping occurs, is always located between the two cut-offs. However, it turns out that the dispersion relation also dictates that the ratio of the left to right polarized components of the wave field be such as $E_+/E_- = (N_{\parallel i}^2 - R)/(N_{\parallel i}^2 - L)$, which goes to zero as $\omega \rightarrow \omega_{ci}$. The result is that fundamental damping of the fast wave is a weak phenomenon, forbidding its applicability in a “pure” plasma, i.e. a plasma including a single ion species.

This problem can be overcome by using a harmonic of the cyclotron resonance, i.e. $p = 2$ or more. This scheme, named harmonic heating, does not suffer from the same limitation as fundamental damping and can be used even in pure plasmas. However, Eq. (8) shows that the corresponding term is now proportional to $k_\perp \rho_i$, which depends itself on the pressure of the heated ion. To be efficient, second harmonic heating thus requires an already significant temperature of the heated ions. Nevertheless, second harmonic IC resonance heating (ICRH) has been successfully employed in various tokamaks as a heating method, and second harmonic tritium heating is a scenario which is planned to be used in ITER (Gormezano et al., 2007).

A possibility to exploit fundamental heating despite the limitation exposed previously is to introduce a small fraction (typically 2–8%) of a different ion in the plasma in addition to the majority ion and to perform fundamental cyclotron heating of this minority ion, hence the denomination: minority heating regime. In this case, it is possible to show that the left cut-off, $N_{\parallel i}^2 = L$, and the hybrid resonance, $N_{\parallel i}^2 = S$, are located very close to each other. The advantage is that the field polarization is now mainly imposed by the majority ion, and therefore E_+ does not cancel at the fundamental minority resonance. The second advantage is that the fundamental cyclotron layer, $\omega = \omega_{ci}$, is very close to the cut-off/hybrid resonance couple. This results in a locally large electric field magnitude and associated stored reactive power. The outcome is efficient heating of the minority species, and this scheme has been routinely used in magnetic fusion devices for decades. The most widespread scenario is hydrogen heating in deuterium plasmas, with a typical hydrogen concentration of 3–5%. Since the power per absorbing ion is quite large, minority heating schemes tend to create very energetic ion populations and one relies on collisional relaxation on thermal electrons and/or ions for plasma heating. In ITER, fundamental heating of helium-3 (^3He) ions is currently one of the envisaged schemes to heat DT plasmas with ICRF waves. It is even envisaged to initiate the heating phase by injecting a small quantity of ^3He ions in the plasma. Minority damping, which is to lowest order independent of the heated ion temperature, will dominate and increase the ion temperature. Once the tritium temperature is sufficient and as the ^3He concentration naturally decreases, second harmonic tritium damping (located at the same position, since $\omega = \omega_{c3\text{He}} = 2\omega_{cT}$) is expected to take over, thus allowing smaller quantities of the expensive ^3He gas to be used. Fig. 3 illustrates the principle of ICRF heating in a tokamak such as JET, as well as an actual calculation of the left-handed

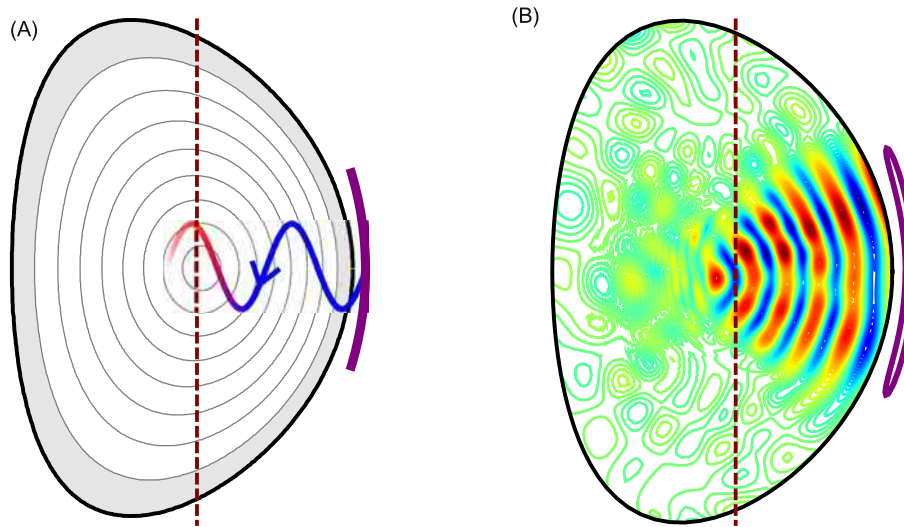


Fig. 3 Depiction of IC wave heating in a tokamak plasma consisting of 95% deuterium + 5% hydrogen. (a) General principle illustrated via a schematic wave propagating from the antenna across the plasma magnetic field surfaces (gray lines). The red color corresponds to locations where significant wave damping occurs; (b) Full-wave calculation of the left-handed electric field (shown in color contours). The vertical dashed lines show the approximate location of the fundamental ion cyclotron resonance layer. Efficient wave absorption by the minority ions results in a low amplitude residual electric on the left of the resonance.

electric field in a typical hydrogen minority heating scenario performed with a full wave code adapted to the description of waves in the Alfvén and IC range of frequencies (Dumont, 2009).

More recently, the so-called 3-ion scheme has been proposed and tested in several tokamaks (Kazakov et al., 2017). This involves the targeted heating of a third ion species characterized by charge number, Z , and mass number, A , in a plasma with two ion species (Z_1, A_1, Z_2, A_2). It can be shown that when the condition $Z_1/A_1 < Z/A < Z_2/A_2$ is satisfied, very efficient damping by the third species can be obtained, as long as it is present in very small concentrations (typically 0.1–1%). This scheme has been proposed to generate energetic ions in stellarators in order to experimentally document their confinement, and to provide efficient ICRF heating during phases of ITER operation at reduced magnetic field. It is also envisaged to use this scheme for ITER D-T plasmas, exploiting the natural presence of the beryllium (Be) impurity in the plasma. In a D-T mixture, Be satisfies $Z_T/A_T = 1/3 < Z_{Be}/A_{Be} = 4/9 < Z_D/A_D = 1/2$. The advantage is that the heavy Be ions tend to heat the fuel ions by collisions, rather than the electrons, which is beneficial in terms of fusion power production. It must be pointed out that, owing to its relatively novel character, this scheme is not as well established as the more traditional minority or harmonic heating schemes. Nevertheless, its flexibility and the fact that it proposes to exploit impurities naturally present in the plasma make it an appealing tool for future magnetic fusion devices.

A comparison of thermal plasma heating by typical RF-accelerated ions is shown in Table 3.

By increasing the minority concentration to ~10–15%, the fundamental cyclotron layer, $\omega = \omega_{ci}$, on the one hand and the cut-off/resonance structure on the other hand move further apart. In this case, it is possible to have a substantial part of the fast wave power tunnel through the evanescence layer and undergo a mode conversion towards the pressure driven Ion Bernstein Wave (IBW). In this mode conversion regime, one relies on the damping of the small wavelength IBW by the thermal electrons to obtain a localized electron heating source. It should be pointed out that despite good results obtained in several tokamaks, this scheme remains difficult to implement in practice, and requires delicate tuning of the magnetic field, minority concentration and antenna toroidal phase spectrum.

The reader should be aware that the physics of IC waves is very rich and many topics related to their use in fusion plasmas have been omitted. Among these, Fast Wave Electron Heating (FWEH), which has been successful in some past experiments, relies on the damping of the fast wave by the thermal electrons. FWEH is achieved by excluding all relevant cyclotron resonance layers by a careful choice of frequency/magnetic field combination. Using an asymmetric toroidal phase spectrum, furthermore, allows Fast Wave Current Drive (FWCD) to be achieved. FWEH is not envisaged in ITER, because the exclusion of all relevant cyclotron layers does not appear to be achievable in such a large device. FWCD, on the other hand, can be interesting since a moderate level of central non-inductive current can be obtained as a by-product of the IC heating wave damping on the electrons, simply by modifying the antenna current distribution and thus the toroidal phase spectrum. Other possibilities include MHCD (Minority Heating Current Drive), which relies on finite orbit width effects to drive a current related to the energetic ions; MCCD (Mode Conversion Current Drive) obtained by exciting a mode-converted wave with an asymmetric spectrum; HHFW (High Harmonic Fast Wave heating) relying on high frequency fast waves for localized electron heating/current drive, using traveling wave antennas to optimize wave coupling and toroidal directivity. To this day, these schemes are not considered as workhorse ICRH scenarios for ITER.

Lower hybrid (LH) range of frequency

Between 1 and 10 GHz, lower hybrid (LH) waves can be used for non-inductive current drive in tokamaks. In this frequency range, $\omega_{ci} \ll \omega \sim \omega_{LH} < \omega_{ce}$, with ω_{LH} the LH resonance frequency $\omega_{LH} \sim \omega_{pi}/(1 + \omega_{pe}^2/\omega_{ce}^2)^{1/2}$. The cold dispersion relation reveals two waves: the fast wave already discussed for IC waves, is useless for plasma heating because the evanescence region between the antenna and the R-cutoff is too wide compared to the wavelength in this range of frequencies; the slow wave, on the other hand, is characterized by the approximate dispersion relation $N_{\perp}^2 \sim N_{\perp}^2(m_p/m_e)\omega_{LH}^2/(\omega - \omega_{LH}^2)$. This expression shows that for given a magnetic field and density, the LH wave propagates at a finite angle, $\theta = \theta(\omega)$, with respect to the magnetic field direction. This kind of configuration is known as a resonance cone wave (Swanson, 2003). A further examination of the dispersion relation shows that the electric field polarization is essentially along the wave vector. The LH wave is therefore an electrostatic wave and does not have any magnetic field component associated to it.

Originally, the idea was to target ion heating by damping of the wave near the LH resonance, hence the name of this scheme. Indeed, as discussed in the “Wave propagation” section, this cold resonance is characterized by $k_{\perp} \rightarrow \infty$, so that efficient heating of ions at relatively low velocities, $v_{\perp} \sim \omega/k_{\perp}$ (i.e. thermal) by perpendicular Landau damping was anticipated. However, experimental evidence pointed to several phenomena preventing this damping mechanism from taking place. For instance, the large values of $N_{\perp}$ attained when approaching this resonance result in the parametric decay of the wave, a non-linear effect responsible

Table 3 Collisional plasma heating resulting from RF-accelerated ions in JET and in ITER.

Background plasma/test particle	Initial energy (keV)	Critical energy (keV)	Spitzer time (s)	Thermalization time (s)	Fraction of power to ions/to electrons
JET D/H	2000	93	0.73	1.11	0.09/0.91
JET D/He3	700	280	0.54	0.29	0.49/0.51
ITER DT/He3	700	619	0.99	0.26	0.72/0.28
ITER DT/Be	700	1858	0.74	0.05	0.92/0.08

for a degraded wave penetration in the plasma. Fortunately, it turns out that efficient wave damping by electrons satisfying the Cerenkov resonance, $v_{\parallel} \sim \omega/k_{\parallel}$, is also achievable in this range of frequencies, provided a slow wave with $N_{\parallel}^2 > 1$ can be excited from an external antenna. Like the fast wave, the slow wave is subject to a low-density cut-off at $\omega \sim \omega_{pe}$, but the corresponding value is of the order 10^{17} m^{-3} in typical conditions. As a result, the evanescence region is quite thin compared to the wavelength in vacuum, and it is possible to design efficient slow wave antennas with the required toroidal periodicity.

Once the slow wave has entered the plasma, it propagates towards the plasma center, with a corresponding increase of $N_{\perp}$. In this range of frequencies, however, if $N_{\parallel}$ is smaller than a given $N_{\parallel}^*$ which depends on density and magnetic field, a confluence between the fast and slow waves takes place in the plasma. Unless damping takes place before the slow wave reaches this confluence point, the converted fast wave propagates back towards the antenna. For all practical purposes, therefore, $N_{\parallel} > N_{\parallel}^*$ places a restriction on useful parallel refractive indices, densities and magnetic fields required for the slow wave to reach the plasma core (Brambilla, 1998), and is known as the Stix-Golant accessibility condition.

Typical values of the parallel refractive indices achieved with standard LH antennas are $N_{\parallel, \text{ant}} \sim 2$. For a plasma characterized by a core electron temperature, $T_e = 10 \text{ keV}$, the range of velocities of the resonant electrons is therefore found to be $v_{\parallel}/v_{th,e} = c/(N_{\parallel}v_{th,e}) \sim 2.5$. For a Maxwellian distribution function, the corresponding number of electrons is very small. LH wave damping therefore relies on the changes of $N_{\parallel}$ as it propagates. Indeed, it can be shown that various effects, among which are toroidicity and spectral broadening by density fluctuations, cause $N_{\parallel}$ to vary during the propagation (Peysson et al., 2017). At some point, $N_{\parallel}$ becomes sufficiently large for Landau damping by electrons with parallel velocities, $v_{\parallel} \sim v_{th,e}$, to take place. The regime in which the rays representative of the LH wave have to travel a long distance around the torus before damping is known as the multi-pass regime, and characterizes most past and current experiments using LH waves. If, on the other hand, T_e and/or $N_{\parallel}$ are sufficiently large from the onset, the wave can be absorbed on its first pass towards the plasma core, a situation referred to as the single-pass regime, which is what would happen in a next-step tokamak containing a very hot plasma. Fig. 4 illustrates the difference between these regimes in two discharges conducted in the Tore Supra tokamak, as computed with a Fokker-Planck code associated to a ray-tracing code (Decker et al., 2014).

In all cases, absorption initially occurs at large values of $N_{\parallel}$, and involves electrons close to the thermal velocity. An energetic electron tail is progressively created starting from these thermal electrons, eventually extending up to superthermal electrons at energies up to a few hundred keV. The quasilinear diffusion induced by the wave is in the parallel velocity direction. As a result, the electron distribution function features a flat region in terms of $v_{\parallel}$, known as the quasilinear plateau. Using an asymmetric $N_{\parallel}$ spectrum therefore results in an asymmetric electron distribution function with respect to $v_{\parallel}$, with a substantial superthermal population. Electrons diffusing in the parallel direction do not experience trapping effects as much as electrons diffusing in the perpendicular direction. Furthermore, in addition to creating an asymmetric resistivity, the LH directly transfers parallel momentum to the electrons. These advantages explain the large LH current drive (LHCD) efficiency achieved in several devices, up to $\eta_{\text{LHCD}} = 0.34 \times 10^{20} \text{ A W}^{-1} \text{ m}^{-2}$ in the large tokamaks JET and JT-60U. Projections for ITER predict that $\eta_{\text{LHCD}} \sim 0.24 \times 10^{20} \text{ A W}^{-1} \text{ m}^{-2}$ could be achieved, with a deposition profile peaked significantly off-axis (normalized radius ~ 0.7) because of the high density and temperature (Gomezano et al., 2007). Despite these results, at this point, uncertainties related to the LH wave penetration and current drive efficiency in dense plasmas, as well as parasitic wave damping by the fusion alphas in a D-T plasma, make their future applicability in reactors uncertain.

Electron cyclotron (EC) range of frequency

Electron cyclotron (EC) waves are typically injected in the frequency range 50–200 GHz, which coincides with the fundamental cyclotron resonance of electrons in most modern magnetic fusion devices. Lower field operation opens the possibility to use

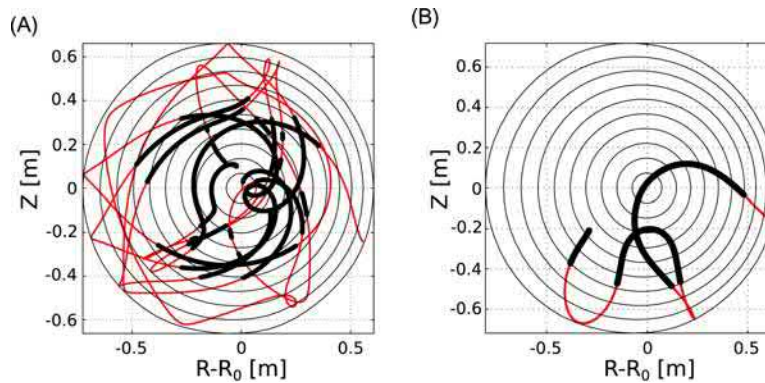


Fig. 4 LH wave current drive in the Tore Supra tokamak. Ray trajectories are shown projected in the poloidal cross-section of the tokamak, but also propagate toroidally around the plasma. (a) In a relatively dense and cold plasma (volume averaged density $\langle n_e \rangle \sim 3.8 \times 10^{19} \text{ m}^{-3}$, central temperature $T_{e0} \sim 2.4 \text{ keV}$), the LH wave is in the multi-pass regime and the rays propagate on large distances before damping takes place. (b) In a less dense and hotter plasma, on the other hand ($\langle n_e \rangle \sim 1.3 \times 10^{19} \text{ m}^{-3}$, $T_{e0} \sim 5.7 \text{ keV}$), damping occurs relatively early during the propagation and the single-pass regime is approached.

harmonic schemes. In this frequency range, $\omega_{ci} \ll \omega \sim \omega_{ce} \sim \omega_{pe}$. The wave is exclusively damped by resonant electrons. For these frequencies/wavelengths, the propagation is quasi-optical and is quite adequately described by ray-tracing or beam-tracing codes. This has the considerable practical advantage of allowing a coupling of the wave to the plasma using an antenna located at some distance, generally consisting of a set of steerable mirrors to vary the deposition location, and possibly to use these waves for current drive by injecting them obliquely with respect to the magnetic field.

The dispersion relation reveals two distinct waves, called the ordinary and extraordinary modes. The ordinary (O) mode has the simple dispersion relation, $N_{\perp}^2 = 1 - \omega_{pe}^2/\omega^2$, showing that it can propagate to the plasma core, at least up to the plasma cut-off, $\omega = \omega_{pe}$, which corresponds in practice to central densities larger than 10^{20} m^{-3} . Fundamental absorption of the ordinary mode is therefore a viable scheme for most tokamaks, including ITER. The extraordinary (X) mode has the somewhat more complex dispersion relation, $N_{\perp}^2 = (\omega^2 - \omega_R^2)(\omega^2 - \omega_L^2)/(\omega^2(\omega^2 - \omega_{UH}^2))$, with ω_R and ω_L the right and left cut-off frequencies, and ω_{UH} the upper hybrid frequency. The right cut-off prevents the X-mode from reaching the fundamental cyclotron resonance, $\omega = \omega_{ce}$, from the low field side of the plasma. Although conceptually appealing, the solution of installing an antenna on the high field side does not appear to be realistic in the context of a reactor. Alternatively, one can resort to the second harmonic resonance $\omega = 2\omega_{ce}$. The drawback of this method is that the right cut-off, while not preventing the resonance at $\omega = 2\omega_{ce}$ from being reached, imposes a cut-off density which is only 50% of the cut-off density of the O-mode for a given frequency and therefore requires operating a relatively higher frequency. The two schemes envisaged for EC resonance heating (ECRH) and current drive (ECCD) in modern magnetic fusion devices are therefore fundamental ordinary mode heating (O-1) or second harmonic extraordinary mode heating (X-2), and sometimes even third harmonic X-mode heating (X-3).

The fundamental resonance condition for EC waves is given by $\omega = \omega_{ce} + k_{\parallel}v_{\parallel}$, where $\omega_{ce} = eB_0/\gamma m_e$ is the electron cyclotron frequency and γm_e is the electron mass corrected for relativistic effects, which turn out to play a major role in the resonance condition. It can be shown that the resonance curve is an ellipse in velocity or momentum space, $(p_{\perp}, p_{\parallel})$, with $\mathbf{p} = \gamma m_e \mathbf{v}$ the electron momentum. Writing $\omega = \omega_{ce}(R_{\text{res}})$, with R_{res} the resonant position in terms of major radius R , shows that the energy of resonant electrons is given by:

$$E = m_e c^2 \left(R_{\text{res}}/R - 1 + N_{\parallel} p_{\parallel} / m_e c \right) \quad (9)$$

For a given position, R , the resonance parameters ω/ω_{ce} and $N_{\parallel}$ thus locate the interaction in the $(p_{\perp}, p_{\parallel})$ plane, which results in a coupling between the wave damping properties in physical space and the location of distribution function modifications induced by the wave momentum space. The quasilinear diffusion that is induced occurs mainly in the perpendicular direction near the resonance ellipse.

Studying wave damping more carefully requires the calculation of the full hot relativistic dielectric tensor (Brambilla, 1998). Some insight into the damping process can nevertheless be gained by realizing that for EC waves, the electric field-dependent coefficient in the quasilinear diffusion coefficient (Eq. 6) is, to lowest order, proportional to $|d^{(p)}(\mathbf{E})| \sim |v_{\perp} E_{\perp} J_{p-1}(k_{\perp} \rho_e)|^2$, with $E_{\perp}$ the right-handed component of the electric field, J_{p-1} the Bessel function of order $p-1$ and ρ_e the electron Larmor radius. For small arguments, $J_{p-1}(k_{\perp} \rho_e) \sim (k_{\perp} \rho_e)^{p-1}$, showing that fundamental damping does not depend directly on the electron temperature, whereas second harmonic damping is a finite temperature effect. This would seem to indicate that O-1 damping is more efficient than X-2 damping, but, as for IC waves, the electric field polarization near $\omega = \omega_{ce}$ is such that $E_{\perp}$ is relatively small (0 in the cold-plasma approximation), thereby decreasing the absorption strength. This is confirmed by comparing the respective optical depths for the O-1 and X-2 modes, i.e. the ratio between the power crossing the resonance layer and the incident power, which shows that X-2 damping is typically 5–10 times more efficient than O-1 damping. It should be pointed out that despite this difference, this damping is very efficient in both cases, so that the power deposition profiles are essentially always narrow.

Furthermore, as explained previously, EC antennas are composed of steerable mirrors (Farina et al., 2014). As a result, it is straightforward to inject the wave with a certain angle with respect to the magnetic field to (1) locate the damping at a precise position in radius, and (2) drive a finite toroidal current. This is illustrated in Fig. 5. As discussed in the “RF wave propagation and damping” section, even though no parallel momentum is imparted by an EC wave to the plasma since the quasilinear diffusion is in the perpendicular direction, the Fisch-Boozer mechanism still makes EC waves suitable for current drive applications. Successful ECCD in a tokamak requires that the damping take place before the fundamental resonance, so that, by virtue of Eq. (9), it occurs on significantly energetic electrons. In theory, the achievable CD efficiency is only a factor of $3/4$ lower than that attained with LH waves (the difference being due to the absence of any parallel momentum input). However, owing to trapped electron effects and to the relatively modest energies of resonant electrons, ECCD is significantly less efficient than LHCD, except at very high temperatures. The record ECCD efficiency to date, $\eta_{\text{ECCD}} = 0.042 \times 10^{20} \text{ A W}^{-1} \text{ m}^{-2}$, was obtained in JT-60U. In ITER, the EC launcher geometry and plasma parameters should allow $\eta_{\text{ECCD}} \sim 0.2 \times 10^{20} \text{ A W}^{-1} \text{ m}^{-2}$ to be achieved (Gormezano et al., 2007). The good localization of the deposition and the excellent flexibility offered by the injector arguably compensate for this lower global efficiency—the driven current density being locally quite large at a position chosen by an operator or by an automated control system. For this reason, EC waves have a wide range of applications, including current profile control, MHD control, and plasma start-up assist (required for stellarators, useful in large tokamaks to enlarge the domain of parameters for plasma breakdown), and their exploitation should increase further as the technological development of efficient and powerful EC generators advances.

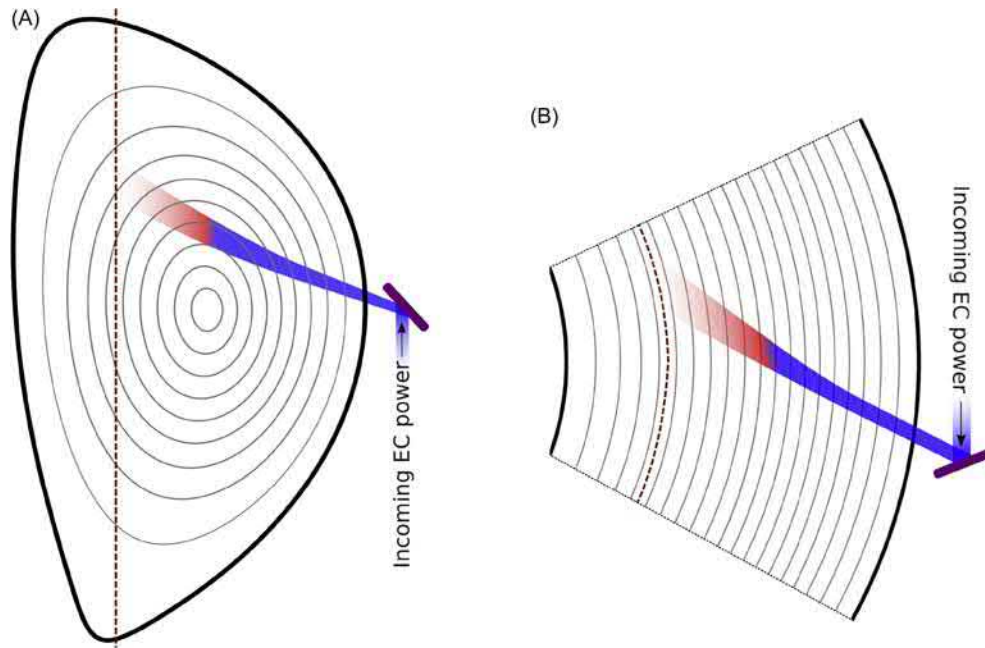


Fig. 5 Schematic of EC wave heating and current drive in a tokamak such as ITER. (a) Poloidal cut; (b) Top view. The beam is shown in blue, with red coloring at locations where significant absorption occurs. The (fictitious) EC steering mirror is shown in magenta, and is used in this representation to inject the wave with a toroidal angle with respect to the magnetic field direction, a situation typical of ECCD. The red dashed curve shows the cold EC resonance.

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Magnetic Confinement Fusion—Experimental Physics: Tokamak

Hartmut Zohm, Max-Planck-Institute for Plasma Physics, Garching, Germany

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Introduction

As discussed in (Galvao, 2020), there are a number of configurations for confining a fusion plasma by a magnetic field. Presently, the highest values of the triple product $nT\tau_E$, the figure of merit for achieving a self-heated ('burning') plasma introduced in (Wade, 2020), have been reached in the tokamak configuration. Here, n is the particle number density, T the temperature and τ_E the energy confinement time. In the following, we will express T in eV, where $T(\text{eV}) = k_B T(\text{K})/q_e$, with k_B the Boltzmann constant and q_e the electron charge. As introduced in (Galvao, 2021), the tokamak is an axisymmetric toroidal configuration in which the confining helical magnetic field is produced by a combination of the toroidal component, B_ν , generated by external coils, and the poloidal component, B_{pol} , generated by a strong toroidal plasma current induced by a central transformer coil. This Chapter reviews the status of experimental tokamak research, and the achievements made towards realizing a burning plasma as the core of a future fusion power plant.

For an attractive tokamak fusion power plant, ultimately two high-level goals have to be achieved:

- Producing enough fusion power P_{fus} to become an economically attractive plant. This condition can be characterized by the ratio of fusion power to auxiliary heating power P_{AUX} , $Q = P_{fus}/P_{AUX}$. Assuming that P_{AUX} is the dominant internal electricity consumer, and inserting typical efficiencies one arrives at values of the order of order $Q > 50$ to achieve acceptable recirculating electrical power below 10% of the generated power (Zohm, 2019).
- Achieving a pulse length that allows for high duty cycle, approaching steady-state operation. Since the sustainment of plasma current by the transformer is limited by the flux swing of the primary coil, this can only be achieved by relying heavily on non-inductive current driven by auxiliary systems (Dumont, 2021) or making use of the intrinsic (thermoelectric) bootstrap current introduced in (Sarazin, 2021).

To achieve these goals, one has to obtain the necessary $nT\tau_E$ for long enough pulse length τ_{pulse} . Two obvious areas of tokamak physics related to this are the energy confinement that sets the value of τ_E , and magnetohydrodynamic (MHD) stability that limits the achievable value of the kinetic pressure $p = nT$, or when ratioed against the magnetic field pressure, $B^2/(2\mu_0)$, the value of the normalized plasma pressure $\beta = 2\mu_0 < p > / B^2$. The experimental progress in these two areas is described in Sections "Energy confinement" and "Stability," respectively. Another important area, not explicitly appearing in the triple product, is the exhaust of power and particles: a substantial heat flux from the plasma impinges on the first wall, and in Section "Exhaust" we discuss the envisaged solution to this problem in a tokamak, including exhaust of the fusion 'ash,' i.e. the He particles after they have heated the plasma by collisional slowing-down.

Having presented the different physics elements, we discuss in Section "Tokamak scenarios" how to integrate these into what is called a 'tokamak operation scenario,' i.e. a recipe to operate a tokamak in a way that optimizes Q and/or τ_{pulse} . This optimization mainly relies on tailoring the radial profiles of pressure and toroidal current density, partly by sophisticated feedback control, but also relying on nonlinear self-organization. It will be shown that the strategies to optimize the two parameters are different, so that an ultimate tokamak scenario satisfying both requirements will have to rely on a trade-off between the two.

Energy confinement

Already the very first tokamak experiments in Russia showed, compared to other experiments at that time, impressive confinement qualities, with central electron temperatures reaching 1 keV. With more tokamak facilities being built also in the USA and Europe, the confinement time could be scaled with the device parameters, and a first analysis yielded a proportionality of τ_E with electron density n_e and poloidal cross section a^2 (a being the minor plasma radius). However, with purely 'ohmic' heating, i.e. the Joule heating due to the plasma current I_p driven by the transformer loop voltage, the range of achievable temperatures was very limited since the plasma resistivity drops as $T^{-3/2}$, i.e. the heating becomes ineffective at higher temperatures. This problem was overcome

with the application of auxiliary heating power P_{AUX} (Dumont, 2021), and the decoupling of I_p and P_{AUX} allowed to establish the basic dependencies of energy confinement in tokamaks in the form

$$\tau_{E,scal}[s] = c \times I_p^{\alpha_I} a^{\alpha_a} R_0^{\alpha_R} P_{tot}^{\alpha_P} B_t^{\alpha_B} n_e^{\alpha_n} K^{\alpha_K} [MA, m, m, MW, T, 10^{20} m^{-3}] \quad (1)$$

with c a fitting constant, R_0 the plasma major radius, P_{tot} the total heating power and κ the elongation of the poloidal plasma cross section. Exponents are shown in Table 1 for the L-mode (Yushmanov et al., 1990) and H-mode (ITER Physics Basis, 1999) confinement regimes introduced in (Jenko, 2021). The H-mode scaling is often used to normalize the experimentally determined confinement time $\tau_{E,exp}$ through the H-factor $H = \tau_{E,exp}/\tau_{E,scal}$.

While the exact values of the exponents differ between the two scalings, one can conclude that both show a nearly linear scaling with plasma current, while the dependence on toroidal field is much weaker. In addition, τ_E is roughly proportional to the plasma cross-section (a and R_0 are coupled via the aspect ratio which does not vary too much in the data sets). Most importantly, though, τ_E decreases strongly with heating power, meaning that plasma energy will only increase with $P^{(1-\alpha_P)}$. This ‘power degradation,’ which only emerged when experiments with additional heating could be conducted, proved to be a major obstacle for progress towards reactor relevant $nT\tau_E$ values, which are in the range of $10^{20} m^{-3}$, 10 keV and 5 s.

Comparing to collisional (‘neoclassical’) transport theory, neither absolute values nor the scaling with field and power are consistent, which is a clear indication that turbulent processes dominate the transport properties perpendicular to the magnetic field. Hence, studies of turbulent transport have been a major topic on tokamak experiments worldwide in the last two decades. An important fundamental finding was that of ‘profile stiffness,’ i.e. an invariance of the shape of the temperature profiles under large variation of external control parameters. The left plot in Fig. 1 shows this for the electron temperature on the ASDEX Upgrade tokamak: when plotted on a logarithmic scale, the profiles are just shifted, i.e. multiplied by a constant factor (Suttrop et al., 1997b).

Together with theoretical considerations, this has been connected to a fundamental property of the underlying turbulence, namely a threshold behavior of the heat conductivity with respect to the temperature gradient. This could be verified in several tokamaks worldwide using sophisticated heating scans, and an example is shown in the right part of Fig. 1, where the strongly nonlinear dependence of the heat flux q on the temperature gradient length is evident. This universal behavior has also been confirmed by measuring the radial propagation of heat waves generated by modulated localized heating of the plasma (Ryter et al., 2005). Similar findings hold for the ion temperature profiles (Mantica et al., 2009), while the density profiles have been found to exhibit a larger variation. This is due to the existence of an inward ‘pinch’ that leads to a peaking of density profiles even though the particle source is usually located in the periphery (Angioni et al., 2003).

Table 1 Exponents of the confinement scalings for L- and H-mode regimes according to Eq. (1).

	C	α_I	α_a	α_R	α_P	α_B	α_n	α_K
L-mode	0.048	0.85	0.3	1.2	−0.5	0.2	0.1	0.5
H-mode	0.0562	0.93	0.58	1.39	−0.69	0.15	0.4	0.78

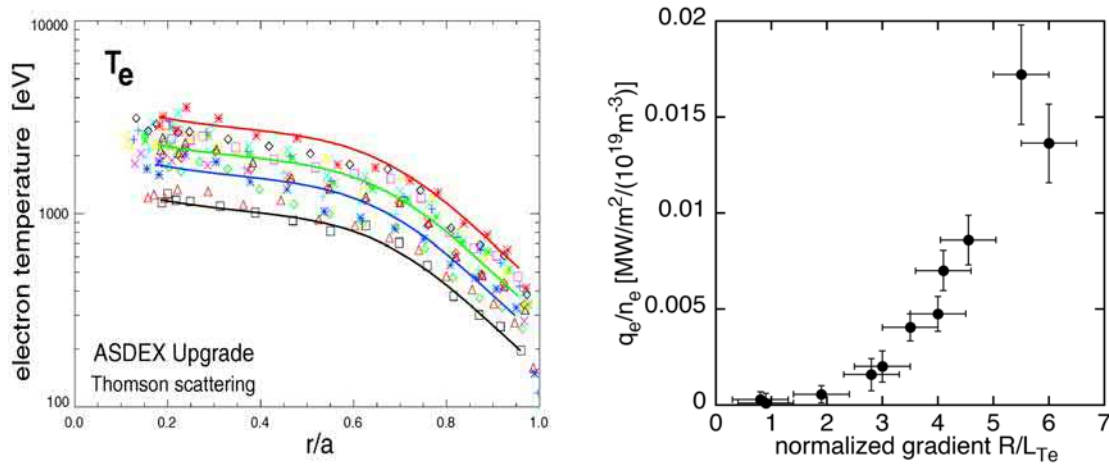


Fig. 1 Logarithmic plot of electron temperature profiles showing self-similarity (left). This self-similarity, also dubbed ‘profile stiffness,’ is due to the nonlinear threshold dependence of the electron heat flux per particle q_e/n_e on the normalized temperature gradient R/L_{Te} where $L_{Te} = T_e/\nabla T_e$ shown on the right. From (Left) Suttrop W et al. (1997b). Identification of plasma-edge-related operational regime boundaries and the effect of edge instability on confinement in ASDEX Upgrade. *Plasma Physics and Controlled Fusion* 39: 2051, Fig. 8, © IOP Publishing, Reproduced with permission. All rights reserved. (Right) Ryter F et al. (2005) Experimental study of trapped-electron-mode properties in tokamaks: threshold and stabilization by collisions. *Physical Review Letters* 95: 085001, Fig. 1.

While the stiff nature of the temperature profiles would indicate that profiles always have a canonical shape, it has been found in many tokamaks that there can be deviations due to a change in turbulence behavior. Such a modification of turbulence can e.g. be induced by a variation of the ratio of electron to ion temperature or by interaction of turbulence with suprathermal plasma particles. The most prominent effect, however, is the reduction or even suppression of turbulence by sheared rotation that can strongly decrease the radial correlation length of turbulent eddies. It was discovered in the ASDEX tokamak (Wagner et al., 1982) that in the tokamak edge, above a certain heating power threshold, a strong increase of the temperature and density profiles can occur, and an ‘edge pedestal’ develops, leading to an increase in temperature and density over the whole plasma radius. In this H(High Confinement) mode, the energy confinement time is roughly doubled with respect to the L(Low Confinement) mode conditions without a pedestal. The reason for the strong reduction of transport in this ‘edge transport barrier’ region was found to be a narrow layer of sheared rotation (Burrell et al., 1990). Subsequently, transport barriers could also be created in the plasma core, dubbed ‘internal transport barrier’ (ITB), and also their occurrence could be linked to a region of strongly sheared rotation, although here, also the magnetic shear plays a role (Levinton et al., 1995). Fig. 2 shows kinetic profiles of a discharge in the DIII-D tokamak exhibiting simultaneously transport barriers in the plasma edge and the core, together with the profile of the radial electric field E_r . The latter is connected to the plasma rotation and has been shown to be the key quantity for turbulence suppression (see also Jenko, 2020).

While the experimental characterization of the H-mode edge transport barrier has progressed significantly over the years, a quantitative theory for the heating power threshold to access H-mode, P_{LH} , has not yet been established. The threshold power is hence usually described in a scaling derived from a multi-device database (Martin et al., 2008), similar to that shown for energy confinement in Eq. (1).

Stability

The theory of MHD stability of magnetically confined fusion plasmas has been outlined in (Lao, 2020). According to theory, the fastest growing instabilities are those of ideal MHD, where the perturbations conserve the topology of the plasma and the growth can be on the so-called Alfvén time scale. This time scale is given by the inertia of the plasma as it moves with the magnetic field, and, due to the small mass density of a fusion plasma, is quite fast in tokamaks, on the order of 10s of μ s. Experimentally, the predicted stability limits are in good agreement with the theoretical prediction, especially if the presence of electrically conducting structures, which can slow down the growth due to eddy currents induced in them, is accounted for in the analysis. If a significant population of suprathermal particles is present in the plasma, these can also affect MHD stability via kinetic effects.

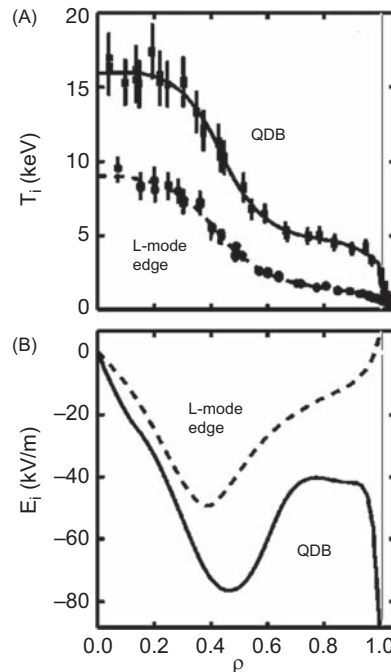


Fig. 2 Occurrence of transport barriers in the DIII-D tokamak: an internal transport barrier develops in the ion temperature profile T_i (left) around normalized radius $\rho = 0.4$ for both cases shown, in the ‘QDB’ (Quiescent Double Barrier) H-mode, also an edge transport barrier can be seen close to $\rho = 1.0$. The right graph shows that the barriers coincide with regions of strong shear in the radial electric field E_r . From Burrell KH et al. (2001) Quiescent double barrier high-confinement mode plasmas in the DIII-D tokamak. *Physics of Plasmas* 8: 2153, Fig. 10A and C.

For purely current driven modes, a kinking of the plasma columns (so-called ideal kink mode) resonant on a major rational surface (i.e. $m/n = q$, with m and n the poloidal and toroidal mode number and q the safety factor, as defined in [Lao, 2021](#)) close to the plasma edge is observed for significant edge current gradients. For tokamak operation, this limits the current ramp rate since, due to current diffusion, a fast ramp rate will create a large edge current, but in practice, current ramps are slow enough to avoid this problem. More important is the limit on the total current, expressed by the edge safety factor $q_{edge} \sim 1/I_p$. While theory predicts unconditional instability below $q_{edge} = 1$, the occurrence of a ($m = 2$, $n = 1$) kink mode practically already limits the operation to $q_{edge} > 2$ unless active control of the kink mode by external coils counteracting the kink field is applied ([Zanca et al., 2012](#)).

The situation becomes more complex when the destabilization by the kinetic pressure gradient is added. Increasing the averaged normalized plasma pressure β will ultimately also lead to the onset of a kink instability. Experimentally, the ideal β -limit in tokamaks largely follows the so-called Troyon scaling ([Troyon et al., 1984](#))

$$\beta_{max} [\%] = const. I_p / (aB) \text{ [MA, m, T]} \quad (2)$$

with the exact value of the constant depending on details of the current and pressure profiles as well as the shape of the poloidal cross-section. As a general rule, a broader current profile will be less stable, which has been taken into account by including the internal inductance of the plasma column, $\mathcal{L}_i$, into the scaling. Normalizing β to the Troyon scaling, one finds as a rule of thumb for the maximum ‘normalized beta,’ $\beta_N = \beta / (I_p / (aB))$ the expression $\beta_{N,max} \sim 4 \mathcal{L}_i$. In [Fig. 3](#), data from the DIII-D tokamak are shown versus the two scalings, indicating that the Troyon limit is a very good overall description of the ideal β -limit, especially if the $\mathcal{L}_i$ correction is taken into account ([Taylor et al., 1994](#)).

Since the maximum achievable β is directly linked to the maximum fusion power ($P_{fus} \sim \beta^2 B^4 R^3$ in the optimum temperature range), it is important to optimize its value. Experiments have clearly demonstrated that, in line with theoretical predictions, the shape of the poloidal cross-section can significantly increase the β -limit ($\beta_{N,max} = 2.8$ for a circular cross section, while the values shown in [Fig. 3](#), left, exceed this value due to shaping). Also, the presence of a large fraction of suprathermal ions has been shown to have a significant effect on stability. Finally, adding a conducting wall close to the plasma boundary counteracts the growth of an instability on the timescale for the penetration of the perturbation field through the wall. For close coupling, this can dominate the instability growth and the ideal kink turns into a resistive wall mode (RWM, [Bondeson and Ward, 1994](#)) that grows on the wall resistive penetration time scale. Experimentally, this allows the growth to be slowed down to a value that makes it accessible for direct feedback control by magnetic coils that counteract the growing perturbation ([Sabbagh et al., 2002](#)).

While the Troyon limit is a limit to the global (volume averaged) β , MHD instabilities can also be driven locally by steep gradients. The most prominent example is the Edge Localized Mode (ELM, [Zohm, 1995](#)), an instability driven by the large gradients of pressure and current in the H-mode edge transport barrier introduced in the previous section. Here, the current gradient is mainly due to the bootstrap current driven by the pressure gradient (discussed in Section “Tokamak scenarios”). ELMs manifest themselves as a relaxation instability of the plasma edge, periodically removing the gradients by ejection of particles and energy during the instability, followed by a phase in which the gradients recover until the next ELM becomes unstable. While the associated ELM energy loss decreases the energy confinement time, the ejection of particles provides an effective control of the particle inventory, and without ELMs, H-mode plasmas tend to accumulate impurities up to an intolerable level. However, extrapolating the divertor heat loads due to ELM events to fusion reactors shows that ELMs will have to be mitigated or fully suppressed. This is possible by

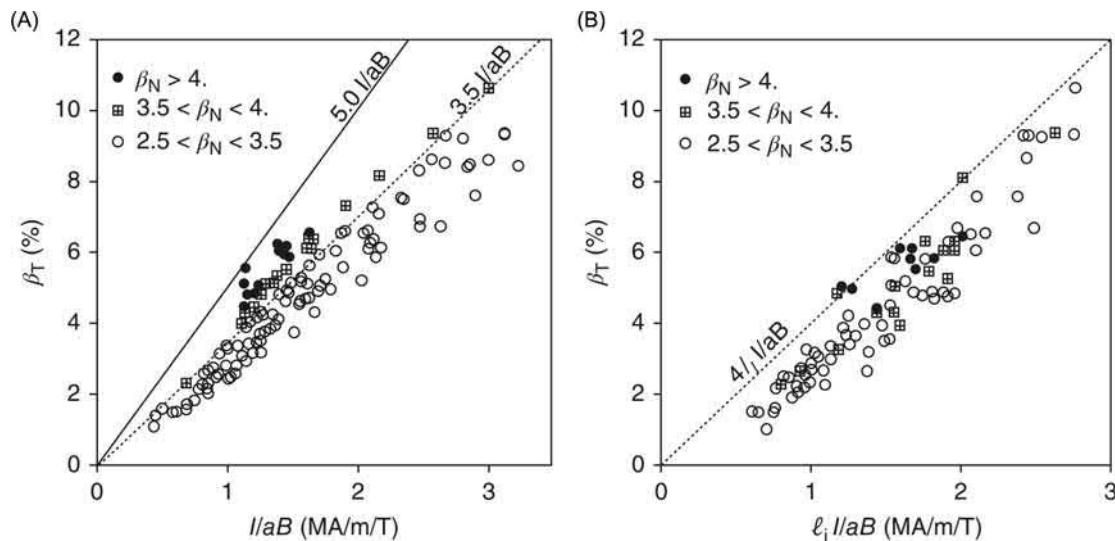


Fig. 3 Experimentally obtained values of toroidal beta β_T in selected DIII-D discharges versus the Troyon scaling factor I/aB (left). An improvement of the ordering of points is achieved if the internal inductance $\mathcal{L}_i$ is explicitly accounted for in the scaling (right). From Taylor TS et al. (1994) *Experimental achievement of toroidal beta beyond that predicted by ‘Troyon’ scaling*. General Atomics Lab Report, GA-A21821, Fig. 3.

applying 3-D resonant magnetic perturbations (Evans et al., 2008) to the plasma edge or by invoking other, more benign, transport mechanisms that keep the pedestal gradients below the ELM onset level. One candidate for such an operational regime is the Quiescent H-mode regime (Burrell et al., 2001), in which ELMs are replaced by a saturated MHD mode that distorts the plasma edge and leads to additional transport. This is an active field of research, and optimization of the plasma edge is still ongoing at the time of writing of this article.

The instabilities discussed so far are all described by ideal MHD, but another class of MHD instabilities affecting tokamak operation exists, namely that of ‘resistive MHD,’ i.e. finite electrical conductivity giving rise to reconnection of magnetic field lines and change of magnetic topology. Such ‘tearing modes’ manifest themselves in tokamak discharges in the form of magnetic islands that locally flatten the kinetic profiles, leading to a loss of plasma stored energy. For Neoclassical Tearing Modes (NTMs), the main drive is the local loss of bootstrap current due to the flattened pressure profile within the island, which reinforces the magnetic perturbation, increasing the island width. NTMs hence constitute a resistive β -limit that is often observed in experiments at β_N values well below the ideal limit (Sauter et al., 1997).

Even at very low β , tearing modes can still be driven unstable if the current density gradients are too strong. Most prominently, this occurs at the ‘density limit’: with excessive fueling of the plasma, the edge region can cool down rapidly due to a thermal instability such that the current is expelled from this region (remember that the electrical conductivity is proportional to $T^{3/2}$ Sarazin, 2020), which triggers current driven tearing modes. In the presence of multiple tearing modes, the magnetic field lines can become stochastic, leading to a rapid global loss of plasma energy (‘thermal quench’). During this event, the temperature falls to such low values across the whole plasma that the plasma current can no longer be sustained by the transformer and decays resistively on the timescale of ~ 10 ms in present day experiments. The main tearing modes observed during such a ‘disruption’ are of low order, i.e. ($m = 2, n = 1$) and ($m = 3, n = 1$) (Suttrop et al., 1997a). While no generally accepted first-principles description of the tokamak density limit exists, a multi-device study showed that it can be described by the empirical ‘Greenwald limit’ (Greenwald et al., 1988)

$$n_{e,max} [10^{20} \text{ m}^{-3}] = I_p / (\pi a^2) [\text{MA}, \text{m}] \quad (3)$$

While (3) was originally derived for the average density, it is meanwhile established it this is a limitation to the achievable edge density, so that in discharges with peaked density profiles, the central density can easily exceed the value given by (3).

Disruptions pose a threat to the integrity of device components that can experience large pulsed heat loads and $j \times B$ forces due to the current density j induced in components by the changing magnetic flux (Wesson et al., 1989). An example is shown in Fig. 4. Disruptions can not only be triggered by the density limit, but also by excessive radiation loss due to impurities entering the plasma (e.g. wall fragments) or even by NTMs induced by the resistive β -limit. Avoiding disruptions, or mitigating their consequences, is of crucial importance for any future reactor-grade tokamak. For example, magnetic islands can be removed by locally driving currents in the islands, usually by Electron Cyclotron Current Drive (ECCD, Zohm et al., 1999), which can drive current in a very localized radial region (Dumont, 2021). Heat loads and forces are effectively reduced by injecting large amounts of fuel gas or impurities that dissipate the available free energy isotropically through electromagnetic radiation. A related problem is the occurrence of ‘runaway’ electrons during the current quench, i.e. a population of electrons that acquires such high energy that collisional drag can no longer balance the acceleration in the induced toroidal electric field (which is large due to the changing plasma current during a disruption). These can pose an additional threat to future devices and mitigation or avoidance techniques are actively studied.

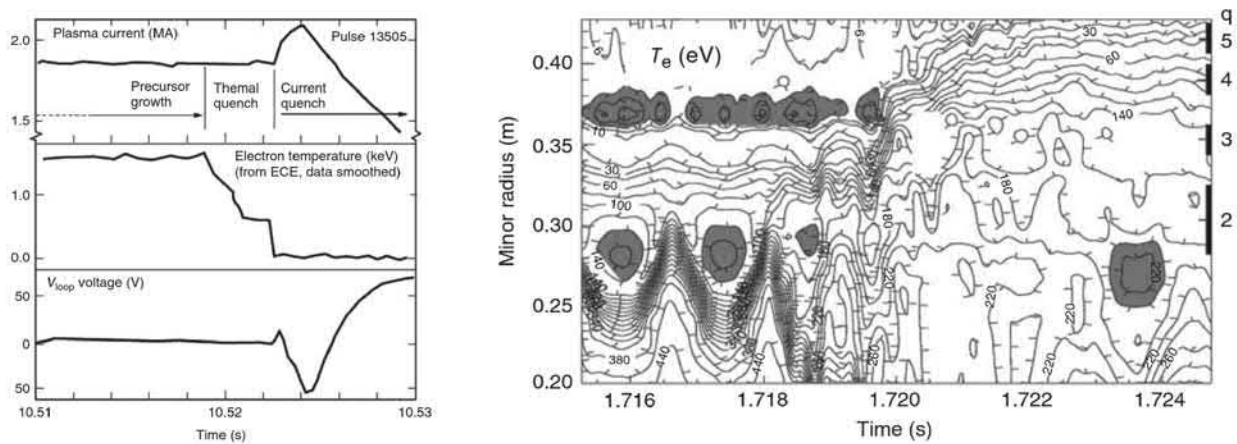


Fig. 4 JET density limit disruption (left): after the thermal quench, in which the electron temperature falls drastically, the plasma current cannot be sustained any longer in spite of the rising loop voltage. Example of a thermal quench through interacting magnetic islands during a disruption in ASDEX Upgrade, visualized by lines of equal temperature (right). Due to the toroidal rotation of the plasma, the toroidal angle is equivalent to time for stationary conditions up to 1.72 s, when the disruption happens. From (Left) Wesson JA et al. (1989) Disruptions in JET. *Nuclear Fusion* 29: 641, Fig. 29. (Right) Suttrop W et al. (1997a) Tearing Mode formation and radiative edge cooling prior to density limit disruptions in ASDEX Upgrade. *Nuclear Fusion* 37: 119, Fig. 4.

Three other types of MHD instabilities are common in tokamak discharges:

- The ‘sawtooth’ instability (Von Goeler and Sauthoff, 1974), which is a periodic resistive relaxation of the plasma core due to (1,1) kink instability in a region with $q < 1$ in the plasma center, mainly driven by the current density gradient. The main effect of sawteeth is to limit the central current density and to periodically redistribute energy and particles across the $q = 1$ surface.
- The ‘error-field locked mode’ (La Haye et al., 1992) which is a magnetic island occurring preferentially at low plasma density due to the plasma response to an externally applied helical magnetic perturbation. The perturbation can also be due to imperfections of the field coil system (even small perturbations are resonantly amplified) and since it is usually an ($m = 2, n = 1$) island, may lead to a disruption.
- ‘Fast particle modes’ (Gorelenkov and Sharapov, 2021) which are (often ideal MHD stable) plasma modes excited by a population of suprathermal particles, which in turn can be redistributed or expelled by the modes. These modes are especially important for future burning plasma experiments where fast particles from the fusion reaction are expected to be the main plasma heating mechanism.

Exhaust

While the preceding sections have focused on the performance of the plasma core, it is equally important for a future reactor to exhaust the heat and particles which are inevitably lost from the confined plasma due to radial transport. While the first tokamaks used material limiters that defined the last closed flux surface, it is now established to use a divertor configuration in which the last closed flux surface is defined magnetically by a separatrix and the field lines on the surrounding surfaces, the Scrape-Off-Layer (SOL), end on the divertor target plates, as shown in Fig. 5.

First experiments in divertor geometry showed that due to the avoidance of direct wall contact, the impurity concentration in the plasma could be kept lower than by using a limiter. Another important finding was that a substantial difference in neutral gas pressure between the main chamber and the divertor region can be maintained, easing the pumping of fuel and impurities from the latter region. For example, it was shown that the core He content could be effectively diminished by pumping He from the divertor in DIII-D, with the residence time of He in the system being of order 5–10 times the energy confinement time (Wade et al., 1995), which meets the requirements for He ash removal from future reactors (Reiter et al., 1990).

Concerning heat exhaust, the divertor offers the possibility to decouple the plasma temperature of the last closed flux surface from that at the target plate, allowing a hot plasma edge while the divertor plasma has a substantially lower temperature, which strongly decreases the sputtering of target plate material by the impinging plasma particles. This is possible since the ‘connection length’ between the main chamber and the target plate along magnetic field lines can be substantial, of order $2\pi qR$, i.e. tens of meters in present day experiments, and the temperature falls in the direction of the divertor target due to the finite parallel heat conductivity. Since the pressure must be constant along field lines, the density increases accordingly. Experimentally, different divertor operating regimes are identified, usually as the plasma density is increased:

- ‘Sheath limited’: at low density, there is hardly a drop in temperature along field lines and the target plasma temperature is given by the capability of the sheath at the target to transmit energy.
- ‘High recycling’: at higher density, the target temperature can drop substantially into the 5 eV range. The ion flux to the target increases substantially due to recycling neutrals.
- ‘Detached’: at even higher density, the plasma recombines before reaching the target plate, and the drop in plasma pressure is balanced by the neutral pressure in front of the plate.

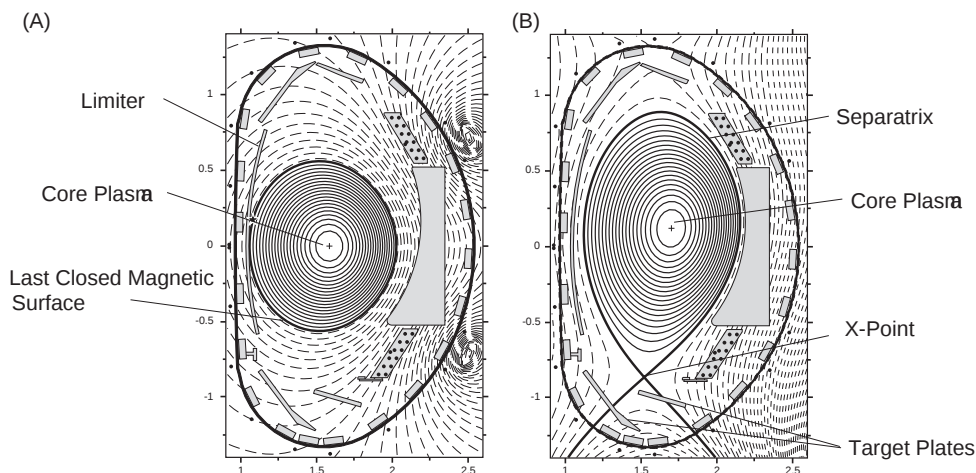


Fig. 5 Limiter (left) and separatrix (right) geometry to define the last closed flux surface (shown in red).

These regimes are illustrated in Fig. 6, which shows SOL pressure and temperature profiles in the main chamber midplane of the Alcator C-mod tokamak, compared to those at the divertor target plate. Note that the pressure is multiplied by a factor of 2 to account for the dynamic pressure of the plasma flow that reaches the plate at sound speed.

It can also be seen in Fig. 6 that the perpendicular width of the part of the SOL in which the power is conducted can be very narrow, of the order of a few mm in present day experiments. A multi-device scaling for attached divertor regimes shows that this width scales inversely with the poloidal magnetic field, and hence is not expected to increase with device size (Eich et al., 2011). For future reactor grade plasmas, the unmitigated power load on the divertor target plates thus extrapolates to values well beyond the steady state component limit (which is roughly 10 MW/m² using present-day technology). It will hence be crucial to achieve detachment in these devices, at least in the radial part of the SOL where the heat flux would exceed this limit.

Depending on the magnitude of the power flux into the SOL, the detached state may not be achievable at the tokamak's operating density. In this case, it has been shown that adding 'seed' impurities into the divertor plasma that increase the radiative losses by line radiation can ease the access to detachment. Likewise, impurity seeding in the main plasma has been applied to reduce the power flux across the separatrix. Fig. 7 shows an example where a simultaneous control of divertor and main chamber radiation has been feedback controlled by injecting two different impurity species (N for the divertor and Ar for the main chamber).

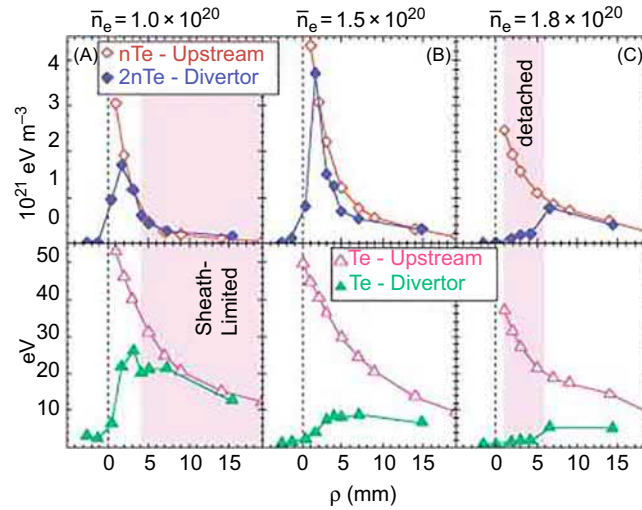


Fig. 6 Comparison of main chamber midplane ('upstream') and divertor temperature T_e and pressure nT_e profiles in the Scrape-Off-Layer of the Alcator C-Mod tokamak. In the sheath limited regime (left), temperature and pressure profiles are similar at both locations, while in the high recycling regime (middle plots), the temperature is reduced at constant pressure. In the detached regime (right), also the pressure is reduced substantially (La Bombard et al., 1995). From La Bombard B et al. (1995) Scaling and transport analysis of divertor conditions on the Alcator C-Mod tokamak. *Physics of Plasmas* 2: 2242, Fig. 5.

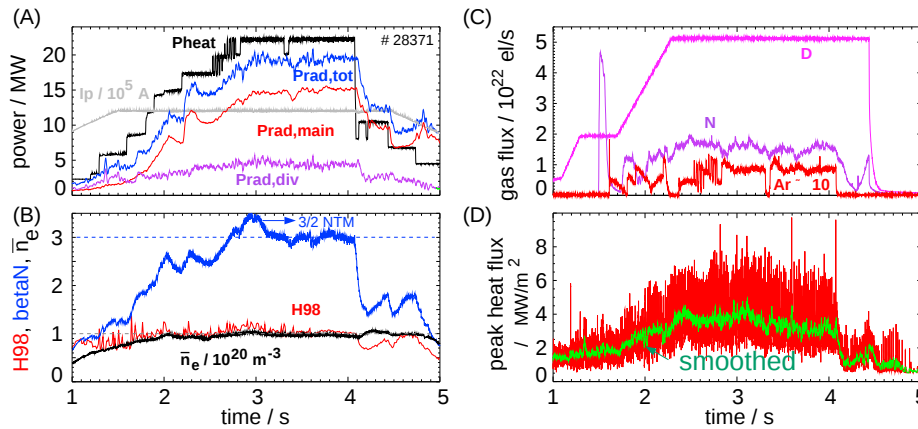


Fig. 7 Double feedback control of divertor and main chamber radiation (labelled $P_{\text{rad,div}}$ and $P_{\text{rad,main}}$ in the upper left panel) using N and Ar seeding, respectively (gas fluxes shown in the upper right panel) in ASDEX Upgrade (Kallenbach et al., 2012). Despite the high heating power of ~ 20 MW, the peak heat flux on the divertor target is only 4 MW/m² (lower right panel) and plasma performance in terms of normalized confinement H and pressure β_N is quite high (lower left panel). From Kallenbach A et al. (2012) Optimized tokamak power exhaust with double radiative feedback in ASDEX Upgrade. *Nuclear Fusion* 52: 122003, Fig. 4.

While these methods work well in order to control the heat flux on the target plates, they are limited by the maximum impurity fraction in the plasma that still allows stationary burn of the fusion plasma (Reiter et al., 1990). In addition, the impurity species have to be selected carefully such that the main radiation losses come from the plasma periphery. Usually, noble gases are preferred since they do not undergo chemical reactions with fuel or wall material, which could lead to trapping of impurities or even of the Tritium fuel.

Another topic related to exhaust is the choice of wall material. Due to erosion, some wall material always contaminates the plasma, leading to additional radiation losses and fuel dilution. For low- Z materials, where Z is the charge number of the nucleus, the allowable concentration in a reactor would be of the order of few percent, while for high- Z , the concentration must be kept of the order of some 10^{-5} . For this reason, tokamaks preferably operated with C wall in the 1990s. Extrapolating to a reactor, it became clear that the level of erosion and the trapping of fuel in hydrocarbons would be unacceptable, even for very low divertor plasma temperatures, since chemical erosion is still present with C first wall elements. For this reason, tokamak operation with all-metal wall (Mo in the case of Alcator C-Mod, W for ASDEX Upgrade and W-Be for JET) has been studied in detail. It is generally found that the H-mode operational space is restricted with respect to operation with a C-wall: in order to avoid excessive influx of wall material, plasmas have to be operated with higher fueling gas flux in order to keep the ELM frequency high enough for flushing impurities from the edge. Such a high fueling rate decreases H-mode confinement quality, which is now understood as an effect of reduced edge pedestal stability (Dunne et al., 2017). This has to be taken into account when extrapolating to future reactors that are supposed to operate with metal wall.

Tokamak scenarios

While the previous sections have described experimental progress in the areas of core confinement, stability and exhaust, we now discuss how to integrate these elements into a coherent plasma operation scenario ('tokamak scenario'). In order to achieve a burning fusion plasma, the triple product $nT\tau_E$ has to attain a certain value given by the Lawson criterion (Wade, 2021). Since this criterion has a minimum at $T \sim 20$ keV, it mainly means operating at high density and long energy confinement time. According to (1), increasing the confinement time can most prominently be achieved by increasing plasma current and device size, with the plasma current, for given device size, being limited by the low- q kink limit. Increasing the plasma current will also raise the achievable density according to (3). In addition, the product nT has to be kept below the pressure limit described by (3), which also increases with plasma current. Plasma scenarios that aim at optimizing the value of $nT\tau_E$ are hence usually run at the highest field the device can produce, with plasma current as high as possible, just avoiding the low- q limit. This recipe was used successfully in the D-T experiments in TFTR (Hawryluk, 1998) and JET (Keilhacker et al., 1999), which produced record values of fusion power of up to 16.1 MW in JET, at a power amplification of $Q = 0.67$. This enabled proof-of-principle experiments demonstrating plasma heating by the fusion-born α -particles (Thomas et al., 1998). Fig. 8 shows these results. In TFTR D-T experiments, Alfvénic Eigenmodes, driven by the α -particles were also clearly identified.

In addition to the D-T experiments in TFTR and JET, performance was also maximized at low q in deuterium discharges in JT-60U, leading to a record triple product of $nT\tau_E = 1.53 \times 10^{21}$ keV s m $^{-3}$, i.e. roughly a factor of 3 short of the numbers needed for a commercial fusion power plant mentioned in Section "Stability." The JT-60U tokamak also holds the record for the highest ion temperature achieved at $T_i \sim 45$ keV (Ishida et al., 1997).

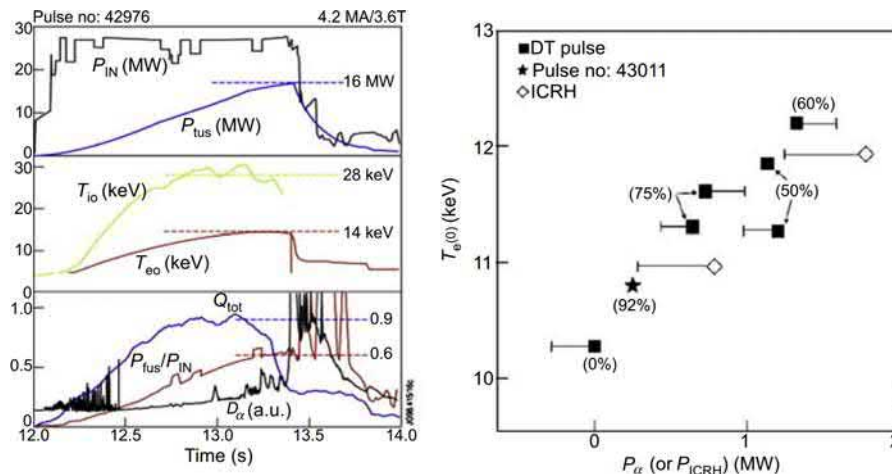


Fig. 8 Record fusion power ($P_{fus} = 16.1$ MW) discharge in JET (left) and proof of α -heating by varying the isotope mix (right, Tritium-percentage given in brackets). The highest central electron temperature $T_e(0)$ is indeed achieved with a roughly equal percentage of deuterium and tritium. From (Left): Keilhacker M et al. (1999) High fusion performance from deuterium-tritium plasmas in JET. *Nuclear Fusion* 39: 209, Fig 3. (Right) Thomas PR et al. (1998) Observation of alpha heating in JET DT plasmas. *Physical Review Letters* 80: 5548, Fig. 3.

The experiments described above were optimized in order to transiently achieve record values (note that the high performance phase of the record D-T discharge shown in Fig. 8 was terminated by an MHD instability). For future fusion reactors, it is essential that the fusion power can be generated quasi-continuously, i.e. the pulse length τ_{pulse} should be maximized as well. For tokamaks, this is a non-trivial task since the plasma current is usually maintained against the (small but finite) electrical resistivity of the plasma by slowly ramping down the current in the central solenoid coil until the flux swing of the coil is exhausted. In order to prolong the pulse length, the current hence has to be driven non-inductively. Two main methods exist for this:

- Current drive by external heating systems: as described in (Dumont, 2021), both neutral beam and RF heating methods have the possibility to generate toroidal plasma current.
- Current generated by the bootstrap effect: as described in (Sarazin, 2021), a radial pressure gradient in a toroidal confinement system drives a toroidal current by a thermoelectric effect.

An extrapolation of the current drive efficiency of external systems shows that in reactor-grade tokamaks, it will not be possible to drive the whole plasma current externally with reasonable power consumption. Experiments have hence focused on maximizing the fraction of bootstrap current f_{bs} . Since the bootstrap current is proportional to $\nabla p/B_{pol}$, the bootstrap fraction can be expressed as

$$f_{bs} = \frac{I_{bs}}{I_p} = c_{bs} \sqrt{\frac{a}{R}} \beta_{pol} \propto q \sqrt{\frac{R}{a}} \beta_N \quad (4)$$

with c_{bs} being a constant whose value depends on the shape of the current and pressure profiles, where the definition of β_{pol} is the same as given above for the toroidal beta, β_t , but using the poloidal magnetic field as normalization. This dependence has been verified experimentally in a number of tokamaks (Kikuchi and Azumi, 1995). While this means that, similar to optimization of the triple product, high β operation is favorable, the bootstrap fraction is maximized by increasing q , i.e. decreasing the plasma current, opposite to the strategy for optimizing the triple product. Hence, for long pulse or steady state (i.e. 100% non-inductive current drive) tokamaks, there needs to be a compromise between fusion performance and pulse length. These scenarios, known as ‘advanced tokamak scenario’ aim at optimizing the current and pressure profiles such that c_{bs} is maximized within the MHD stability limits. Here, a critical problem is the alignment of the bootstrap current profile with the current profile needed to produce the desired q -profile. Fig. 9 shows an example from the JT-60 U tokamak in which the majority (80%) of the current is driven by the bootstrap current for several seconds (Ide et al., 2005).

The profiles of q and T_i shown in the right panel of Fig. 9 demonstrate the optimization strategy described above: not only is q high at the edge, meaning a relatively small plasma current, but also the q -profile shows a non-monotonic behavior (so-called reversed shear). This increases the bootstrap fraction with respect to a typical q -profile in an ohmic discharge, which would be decreasing monotonously from the edge to the center since the ohmic current density is highest there. The T_i profile exhibits an ITB (see also subsection on transport) at $\rho = 0.5$, and the large pressure gradient there creates a local maximum in the bootstrap current density, supporting the q -profile reversal.

While this route to fully non-inductive tokamak plasmas has been demonstrated in principle on several devices (including a fully self-organized discharge in TCV where the plasma current was no longer feedback controlled Sauter et al., 2001), it remains difficult so far to maintain MHD stable profiles over a longer time and to establish a confinement quality sufficient for reaching a high triple product even at lower current. In an alternative approach, the so-called ‘hybrid’ scenario, one therefore aims at operating at higher plasma current, albeit still with central q -profiles that are elevated with respect to standard ohmic profiles, but not reversed, leading to increased MHD stability limits. Due to the lower bootstrap fraction, these scenarios, as e.g. established in ASDEX Upgrade (Sips et al., 2002) and DIII-D (Luce et al., 2003), might not extrapolate to true steady state, but could offer at least greatly enhanced τ_{pulse} .

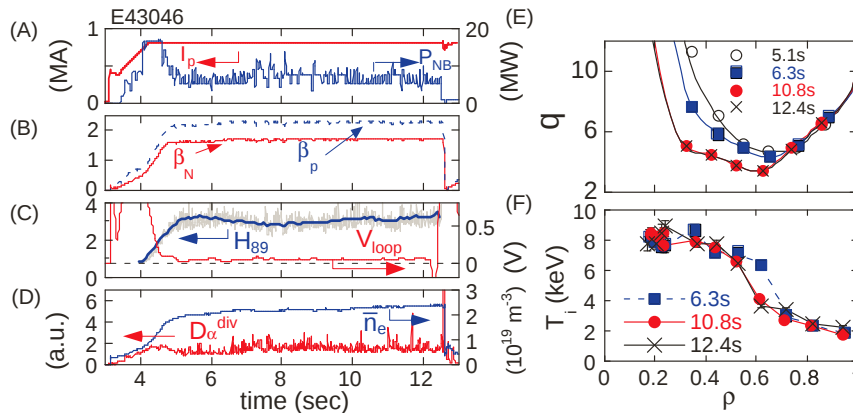


Fig. 9 Non-inductive discharge in JT-60U: time traces of plasma beta (feedback controlled by NBI heating), confinement quality H_{89} and loop voltage V_{loop} (left). The right panel shows the profiles of safety factor q and ion temperature T_i , where an internal transport barrier can be recognized at normalized radius $\rho \sim 0.5$. From Ide S et al. (2005) Overview of JT-60U progress towards steady state advanced tokamak. *Nuclear Fusion* 45: S05, Fig. 8.

It is important to mention that long pulse or steady state tokamak discharges also pose significant technological challenges in terms of steady state removal of heat and particles as well as reliability of the individual systems needed for tokamak operation. This is mainly the domain of tokamak devices with a superconducting coil set, allowing the study of long pulse physics and technology. In the Tore Supra tokamak, more than a Gigajoule of heating energy was supplied in a 6 min long discharge (Van Houtte et al., 2004), where steady state was achieved by driving the current externally with Lower Hybrid waves in a low density L-mode plasma. More recently, very long H-mode discharges were demonstrated in KSTAR (70 s Oh et al., 2018) and EAST (100 s Gong et al., 2019).

Finally, it is worth pointing out that the development of stable integrated tokamak operation scenarios relies strongly on feedback control of the plasma parameters (Lister et al., 1999). While the first tokamak experiments relied purely on feedforward optimization of the discharge evolution, a major breakthrough was obtained by active control of plasma current and position inside the vacuum vessel using pick-up coils and flux loops as sensors and the poloidal magnetic field coils as actuators in the 1970s. This was extended to magnetic feedback control of the shape of the plasma cross-section in the 1980s, and the magnetic configuration is routinely controlled in all tokamaks worldwide since then. However, in future fusion reactors, it will be necessary to also control the kinetic quantities plasma temperature and density, and to some extent also their radial profiles, in order to guarantee a stable operational point with reliable constant fusion power output. Since the 2000s, more and more sophisticated control loops have hence been introduced for kinetic control, and examples can be found in the beta control shown in Fig. 9 or the radiation control shown in Fig. 7. These rely on additional sensors for the controlled quantities, and hence have profited substantially from the development of sophisticated plasma diagnostics, which are also used for physics studies. Typical actuators are the heating and fueling systems described in this chapter as well as in (Dumont, 2021).

The integrated control of tokamak scenarios has hence become a MIMO (Multiple Input–Multiple Output) problem, which is also due to the nonlinear relation between the different control quantities, and thus requires sophisticated control strategies beyond classical linear control schemes. For future fusion reactors, one has to take into account that available sensors will be limited due to the harsh environment involving a significant neutron flux. Concerning actuators, the superconducting poloidal field coils with large stored magnetic energy imply a relatively slow reaction, and the power delivered by the heating and current drive systems will have to be small compared to the intrinsic α -heating power in an efficient reactor. On the other hand, a key aspect which will be different from the present experimental devices is that a fusion reactor will have one well-defined operational point and control can be optimized around it.

Summary

Tokamak experiments have made major progress in achieving plasma parameters close to those required for a fusion reactor, and, together with theory, the fundamental understanding of the underlying fusion plasma physics has advanced to a state that has allowed to design and construct the ITER device, which is set to demonstrate generation of fusion power at 10 times the value of power injected into the plasma.

Table 2 gives an overview of significant past and current tokamak devices around the world, including also the ITER device parameters.

Table 2 Parameters of significant past and current tokamak devices, including ITER.

Device name	Operation period	Country	Major radius [m]	Minor radius [m]	Toroidal field [T]	Plasma current [MA]	Divertor configuration	Additional heating [MW]			
								NBI	ICRH	LHRH	ECRH
Alcator-A	1973–1979	USA	0,54	0,1	10	0,31	lim		0,1	0,1	
Alcator-C	1979–1987	USA	0,64	0,17	12	0,9	lim			4	
Alcator-C MOD	1993–2016	USA	0,67	0,22	9	2	SN, DN		8	3	
ASDEX	1980–1990	DE	1,54	0,4	3	0,52	SN, DN	4,5	3	2	
ASDEX Upgrade	1991–	DE	1,65	0,5	3	1,6	SN, DN	20	8		8
COMPASS-D	1989–	UK/CZ	0,56	0,21	2,1	0,28	SN			0,6	2
DIII-D	1986–	USA	1,67	0,67	2,1	2,5	SN, DN	20	6		6
DITE	1975–1989	UK	1,17	0,26	2,7	0,26	BD	2,4			
EAST	2006–	CN	1,85	0,45	3,5	0,8	SN, DN		6	4	
FT	1975–1989	IT	0,83	0,2	10	0,8	lim			1	
FT-U	1990–	IT	0,93	0,3	8	1,6	lim		1	3	1,6
HT-7	1993	CN	1,22	0,3	3	0,4	lim		2	2	
ISX-B	1978–1984	USA	0,93	0,27	1,8	0,24	lim	2,5			2
ITER	2025–	F	6,2	2	5,2	15	SN	33	20		20
JET	1983–	UK	3	1,25	3,5	7	SN	24	32	12	
JFT-2M	1983–2004	Jp	0,9	0,25	1,8	0,23	SN	1,5	1	0,3	0,2
JT-60U	1991–2014	Jp	3,4	1,1	4,2	5	SN	30	6	15	3
JT-60SA	2020–	Jp	3	1,2	2,25	5,5	SN	34			7

Table 2 Parameters of significant past and current tokamak devices, including ITER.—cont'd

Device name	Operation period	Country	Major radius [m]	Minor radius [m]	Toroidal field [T]	Plasma current [MA]	Divertor configuration	Additional heating [MW]			
								NBI	ICRH	LHRH	ECRH
KSTAR	2008–	KO	1,8	0,5	3,5	0,8	SN, DN	1	0,5		0,35
MAST	2000–	UK	0,85	0,65	0,52	1,5	SN, DN	5			1,4
NSTX	1999–	USA	0,85	0,68	0,55	1,5	SN, DN	7	6		
PDX	1979–1983	USA	1,4	0,45	2,5	0,6	DN	7			
Petula	1974–1976	F	0,72	0,16	2,7	0,16	lim			0,5	
PLT	1975–1986	USA	1,3	0,4	3,5	0,72	lim	3	5	1	
Pulsator	1973–1979	DE	0,7	0,12	2,7	0,093	lim				
RTP	1989–1998	NL	0,72	0,16	2,5	0,16	lim				0,9
ST	1970–1974	USA	1,09	0,14	4,4	0,13	lim				
START	1991–1998	UK	0,32	0,27	0,31	0,31	SN, DN	1			0,2
TCV	1992–	SUI	0,89	0,25	1,5	0,17	SN, DN				4,5
TEXT	1981–1995	USA	1	0,27	2,8	0,34	lim				0,6
TEXTOR	1983–2013	DE	1,75	0,46	3	0,6	lim	4	4		1
TFR	1973–1984	F	0,98	0,2	6	0,41	lim	0,7	1,5		0,6
TFTR	1982–1997	USA	2,4	0,8	5	2,2(3)	lim	40	16		
Tore Supra	1988–2010	F	2,42	0,72	3,8	1,4	lim	1,7	12	10	1
Tosca	1974–1987	UK	0,3	0,09	0,5	0,02	lim				0,2
Tuman III	1978–	RF	0,55	0,15	3	0,2	lim				
T-3	1962–1971	RF	1	0,12	2,5	0,06	lim				
T-10	1975–	RF	1,5	0,37	2,5	0,35	lim				2
WEST	2016–	F	2,5	0,5	3,7	1	SN, DN		9	7	0,6

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Magnetic Confinement Fusion—Experimental Physics: Stellarators

Hiroshi Yamada, Graduate School of Frontier Sciences, The University of Tokyo, Kashiwa, Chiba, Japan

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Glossary

Ballooning mode One of the MHD instabilities driven by pressure gradients. The ballooning mode is localized on the outboard side of the torus where the magnetic field is weak and characterized by short wavelengths.

Bootstrap current In the presence of a pressure gradient in a toroidal plasma, the characteristic motion of charged particles trapped in a magnetic mirror due to toroidicity can produce a net toroidal current. This spontaneous non-inductive current is referred to as a bootstrap current. In stellarators, particles trapped in a helical magnetic field ripple play a complementary role to particles trapped in a toroidal ripple.

Electron/ion root In an axisymmetric system like a tokamak, fluxes of charged particles are intrinsically ambipolar due to collisional momentum conservation. Therefore, particle fluxes do not depend on the ambipolar electric field in tokamaks. In contrast, particle fluxes explicitly depend on the ambipolar electric field in stellarators due to asymmetry. The condition of charge neutrality requires a balance between electron flux and ion flux. A specific ambipolar radial electric field is given as a solution, i.e. a root, of this balance equation. Since this balance equation is usually non-linear with regards to the radial electric field, there can be multiple roots of the radial electric field. When the electron flux is predominant, the resultant positive electric field retards the electron flux, which consequently balances with the ion flux. This situation is called an electron root. The opposite case, which gives negative electric field, is called an ion root.

Helical ripple The magnetic field strength changes along the magnetic field lines in a toroidal confinement configuration. Primary modulation is due to the excursion of the magnetic field line between the inboard side with strong toroidal magnetic field and the outboard side with weak toroidal magnetic field of the torus. In addition, the magnetic field in stellarators is modulated by higher order harmonics due to helical coils or modular coils. This modulation is called a “helical ripple.” The change of magnetic field strength along the magnetic field line produces a “mirror trap” of charged particles. The characteristic motion of these trapped particles greatly influences transport in plasmas. Therefore, transport in stellarators is much more complicated due to a helical ripple than in tokamaks.

Interchange mode One of the MHD instabilities driven by the pressure gradient like a ballooning mode. In contrast to a ballooning mode, an interchange mode has global structure with a long wavelength. Stability against the interchange mode is violated, in particular, by magnetic hill.

Kink mode One of the MHD instabilities driven by the gradient of plasma current density. The flux tube undergoes a deformation without breaking (“tearing”) and reconnection of the magnetic field lines.

Local island divertor The last closed surface of the magnetic field is defined by the separatrix and the magnetic field lines change from a closed topology to an open topology across the separatrix. This open topology can be configured into

a “divertor” configuration. A local island divertor uses the separatrix of a magnetic “island” intentionally generated in the plasma edge by external application of resonant magnetic perturbations. A “scraper” structure is inserted into the magnetic island to provide a more enclosed region and enable divertor pumping of the resultant neutral particles.

Magnetic island The confining magnetic field of the plasma ideally forms a set of concentric nested magnetic cages like a matryoshka doll. When breaking and reconnection of magnetic field lines takes place, a magnetic cage (so-called flux tube) isolated from the rest of space can be formed. Since this topology looks like a crescent-shaped island on the poloidal cross-section, it is called a “magnetic island.”

Magnetic well/magnetic hill The plasma confined by the magnetic field prefers to expand towards a region of weaker magnetic field. When the averaged magnetic field along the magnetic field line becomes stronger in the outer confinement region than in the inner confinement region, any perturbed motion of the plasma is repelled inward and instabilities cannot develop. This situation is referred to as a “magnetic well.” In the opposite case of a “magnetic hill,” instabilities easily develop. This concept of magnetic well is an important element, together with magnetic shear, when MHD stability is discussed.

Pfirsch-Schlüter current Plasmas are diamagnetic. Since the toroidal field is predominant in toroidal confinement systems such as tokamaks and stellarators, diamagnetic currents primarily flow in the poloidal direction to reduce the toroidal field. In order to fulfill the condition that the divergence of the current is equal to zero in toroidal geometry, additional currents are driven along the magnetic field lines. This secondary current is called the Pfirsch-Schlüter current. This current deforms the magnetic flux surfaces.

Quasi-symmetry The stellarator magnetic field is intrinsically 3D and non-axisymmetric. Since the violation of toroidal symmetry degrades the confinement capability due to the drift of the particle orbits, recovery of symmetry is an essential strategy in stellarator optimization. The idea of quasi-symmetry is the establishment of a non-axisymmetric magnetic configuration where the magnetic field strength approximately depends on only one angular coordinate along the magnetic field lines. Depending on the selection of the primary angular coordinate, there are quasi-axisymmetry, quasi-helical and quasi-poloidal concepts. Quasi-isodynamics, or omnigenity, is a comprehensive approach to minimizing the drift of particle orbits.

Shafranov shift Pfirsch-Schlüter currents generated by the plasma kinetic pressure deform the magnetic flux surfaces. The primary component of the magnetic field induced by these currents is a vertical field and, therefore, the plasma column is pushed outward. This outward shift of the central axis of the plasma is called the Shafranov shift.

Tearing mode One of the MHD instabilities driven by the gradient of plasma current. Unlike a kink mode, the resistivity of the plasma plays an essential role in a tearing mode. In ideal MHD, i.e. in zero resistivity plasmas, a tearing mode does not occur. Reconnection of magnetic field lines due to resistivity produces a magnetic island and the growth of the magnetic island tears additional nested magnetic flux surfaces.

Zonal flow As a consequence of self-organization of short-wavelength turbulence in toroidal plasmas, macroscopic plasma flows in the poloidal direction are often seen. These flows are localized radially and form a multilayered structure. Inspired by the analogy to meridional flows on the earth these flows are designated zonal flows. Since zonal flows are usually sheared flows in the radial direction, they shred convective eddies of turbulence and regulate turbulent transport.

Introduction

Stellarators share the basic concept with tokamaks such that plasma is confined magnetically in a topological torus. It was pointed out in the very early stage of fusion research that a torus with simple circular magnetic field, i.e. with magnetic field lines oriented in the purely toroidal direction, is not sufficient to confine the plasma and the configuration has to have helically twisted magnetic field lines in order to avoid escape of the plasma from the magnetic bottle due to intrinsic particle drifts. Stellarators generate these twisted magnetic field lines by currents in coils outside the plasma while tokamaks generate those by currents flowing in the plasma. Eventually, stellarators are intrinsically non-axisymmetric while tokamaks are symmetric along the torus. Although the idea of stellarators was conceived by Lyman Spitzer in 1951 prior to the invention of the tokamak (Spitzer Jr., 1958a), development of this concept has fallen behind that of the tokamak in terms of the achieved performance approaching fusion conditions. This tardy progress is due to the complexity of its intrinsic 3D feature which makes theoretical modeling (Wakatanai, 1998; Boozer, 1998), as well as construction of experiment devices, difficult. However, remarkable progress in experiments and theory is revealing the stellarator’s innate advantages, such as steady-state operation (Miyamoto, 1978; Johnson, 1982; Carreras et al., 1988; Lyon et al., 1990; Lyon, 1990). As shown in Fig. 1, many experimental projects are underway and planned across the world. In particular, the present two major flagship devices; Large Helical Device (LHD in short, Toki, Japan) (Iiyoshi et al., 1990; Fujiwara et al., 1996; Motojima et al., 1999; Special Issue, 2010) and Wendelstein 7-X (W7-X in short, Greifswald, Germany) (Nührenberg et al., 1995; Wanner et al., 2001; Pedersen et al., 2017; Klincker et al., 2019) have demonstrated good performance, which is even better in some aspects than major tokamak experiments. These favorable experimental achievements, in conjunction with advances in extremely-large-scale simulation, are accelerating further optimization of the magnetic configuration and development of a stellarator fusion reactor concept.

The stellarator is a promising concept for a fusion power plant even with respect to the tokamak. The distinguishing property that there is no requirement for high currents in the plasma leads to a great advantage of stellarators in comparison to tokamaks. A

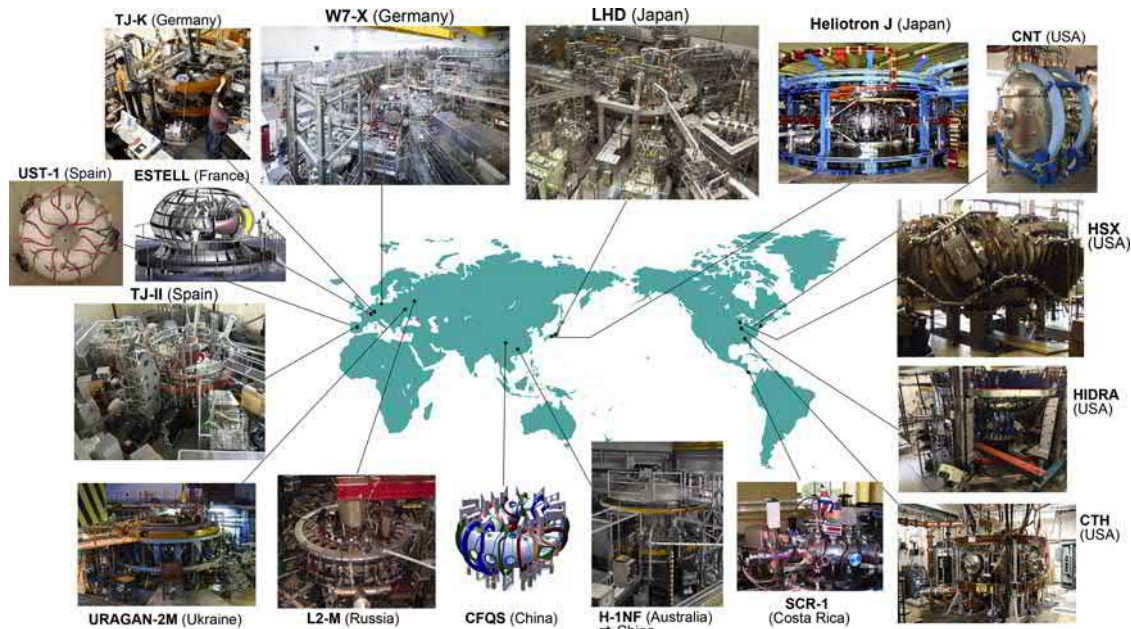


Fig. 1 Overview of the world's principal stellarator experiments. LHD: courtesy of National Institute for Fusion Science (NIFS). Heliotron J: courtesy of Institute of Advanced Energy, Kyoto University. CNT: <http://www.iccworkshops.org/icc2007/uploads/48/icctalk.pdf>; HSX: <http://www.psl.wisc.edu/projects/large/hsx/>; HIDRA: <https://cpmi.illinois.edu/2016/04/26/hidra-hybrid-illinois-device-for-research-and-applications/>; CTH: https://en.wikipedia.org/wiki/Compact_Toroidal_Hybrid; SCR-1: courtesy of Costa Rica Institute of Technology; H-1NF: <https://en.wikipedia.org/wiki/H-1NF>; CFQS: courtesy of National Institute for Fusion Science (NIFS); L2-M: courtesy of General Physics Institute; Uragan-2M: https://ipp.kipt.kharkov.ua/u2m/u2m_en.html; TJ-II: <http://fusionsites.ciemat.es/jvelasco/exploitation-of-tj-ii/>; ESTELL: <https://pdfs.semanticscholar.org/c0b1/06425dcbf7e77514b5531710de00255929f0.pdf>; UST-1: <http://www.fusionvic.org/>; TJ-K: <https://www.igvp.uni-stuttgart.de/en/research/plasma-dynamics-and-diagnostics/experiments/TJ-K/>; W7-X: courtesy of Max-Planck-Institut für Plasmaphysik (IPP).

particularly important feature is the freedom from disruptive instabilities driven by plasma currents, which secures stable operation, and the absence of a requirement for current drive reduces the circulating power in a fusion power plant. While the stellarator achieved performance in terms of the “fusion triple product” (Wade, 2021) lags that of the tokamak, the importance of its complementary and alternative role to the tokamak (Stroth, 1998; Helander et al., 2012) is emphasized due to its attractiveness in both the Japanese (MEXT, 2018) and European (EUROfusion, 2018) strategies for fusion energy development. In conjunction with the ongoing progress of experiment and theory in stellarators, reactor design studies are also underway (Beidler et al., 2002; Najmabadi et al., 2008; Sagara et al., 2010) in order to exploit the potential of the stellarator fusion reactor.

Basic concept of stellarators

The basic concept of nested flux surfaces formed by twisted magnetic field lines in a topological torus is common to stellarators and tokamaks. The essential difference is that there is no net toroidal current in a stellarator plasma. Torsion of the magnetic field lines is given only by external coils in stellarators. Magnetic configuration in stellarators have wide variety according to conceptual ideas on how to twist magnetic field lines and to form nested flux surfaces. The twist of the magnetic field lines is referred to as the “rotational transform,” ι (Galvão and Canal, 2021). The rotational transform normalized by 2π , ι , is the reciprocal of the safety factor, q , which frequently appears in tokamak physics. Stellarators are defined in a broad sense as a concept of toroidal magnetic confinement by external coils and there are many diversified concepts derived from this essential principle, such as advanced stellarator, bumpy torus, classical stellarator, heliac, heliotron and torsatron in alphabetic order. The profile of the safety factor, which is the reciprocal of the rotational transform, also has a specific form depending on the concept, as shown in Fig. 2.

Fig. 3 shows the configuration of coils, the main body and the visualized magnetic field of two leading devices: Large Helical Device (LHD) heliotron and Wendelstein 7-X (W7-X) advanced stellarator. Unlike in tokamaks, nested flux surfaces exist in vacuum and therefore they are observable by detecting fluorescence light and excited low-pressure gas with electron beams (Morisaki et al., 2010; Otte et al., 2016).

The consequent outstanding advantage of stellarators is steady-state operation. High net toroidal currents in tokamaks (in the range of several MA) drive magnetohydrodynamic (MHD) instabilities (Lao et al., 2020) and these can cause abrupt termination of the plasma discharge, referred to as a “disruption.” Stellarators are free from disruptions. Even in stellarators, plasmas collapse when they approach operational limits such as those involving pressure and density. However, this collapse is benign since the magnetic bottle to confine the plasma exists in vacuum to the first order. Also sustainment of these currents in tokamaks requires auxiliary

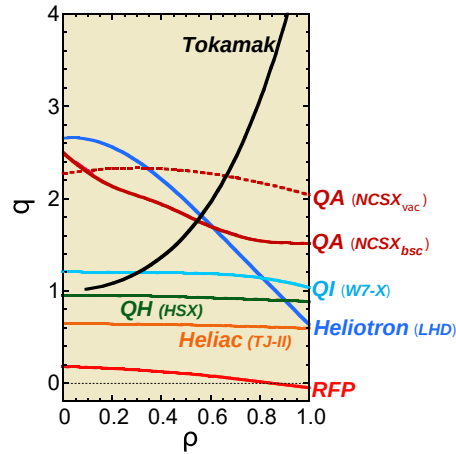


Fig. 2 Profiles of the safety factor for various stellarators. QA relies on “bootstrap current”, which increases the rotational transform (decreases the safety factor), as seen in the change from NCSX_{vac} to NCSX_{bsc}.

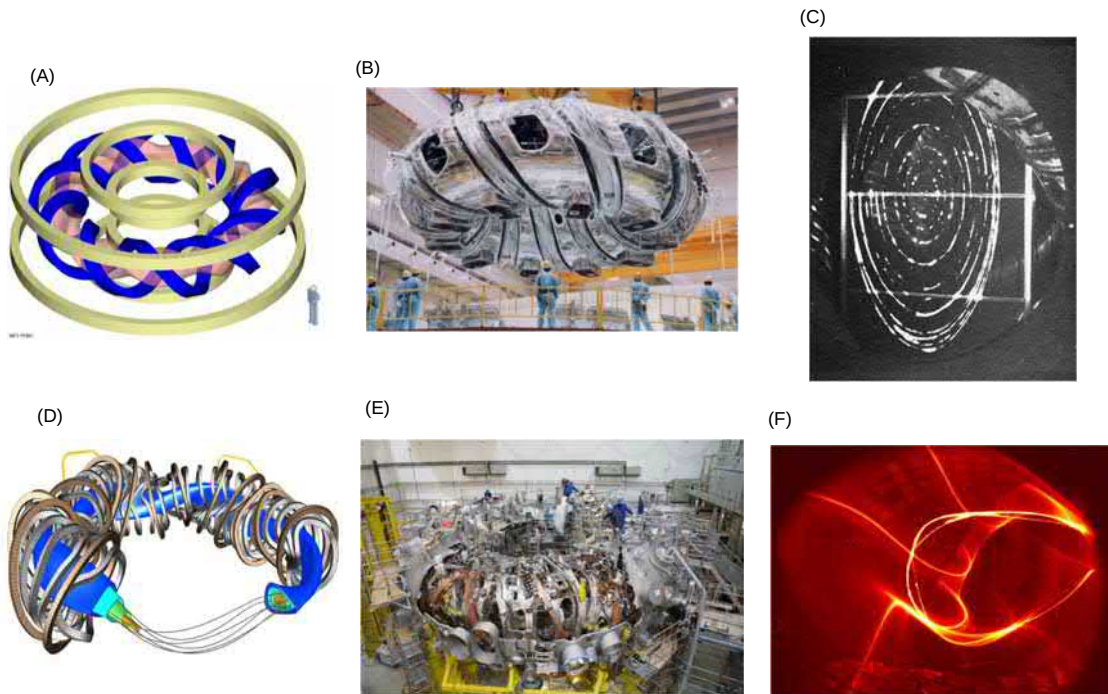


Fig. 3 Configuration of coils, (a) and (d), the main bodies, (b) and (e), and visualized magnetic field surfaces in poloidal cross-section, (c) and (f) for LHD (a)–(c) and W7-X (d)–(f). Courtesy of NIFS and IPP.

input power, since currents cannot be driven inductively in steady-state. Since stellarators do not need this current drive, the recirculating power in a power plant is expected to be reduced compared with tokamaks.

It should be noted that the above mentioned advantages are produced by the three-dimensionality of stellarators. Physics tells us that symmetry corresponds to a conservation law. Translational symmetry gives rise to momentum conservation and rotational symmetry settles angular momentum conservation. Consequently, particle motion is governed by these conservation laws. This fundamental physics is seen in toroidal plasmas as well. Magnetic field strength in tokamaks does not change in toroidal angle measured on a circle about the major axis of the torus. This means tokamaks have axisymmetry (referred to as toroidal symmetry), which leads to a certain conservation law (Sarazin, 2021). In contrast, stellarators should have three-dimensionality and do not have any such symmetry. Fig. 4 compares distributions of magnetic field strength on the flux surface in a variety of stellarators and ITER. In general, the magnetic field is stronger inboard than outboard due to toroidal effects for all cases. While the case of ITER (Fig. 4a) shows toroidal symmetry, modulation specific to the various concepts is added to the cases of stellarators. The loss of toroidal symmetry gives rise to drawbacks, such as significant deviation of particle orbits from magnetic flux surfaces. Straight stellarators have helical symmetry and therefore early stellarators had large aspect ratio, i.e. R/a , the ratio of major radius of torus, R ,

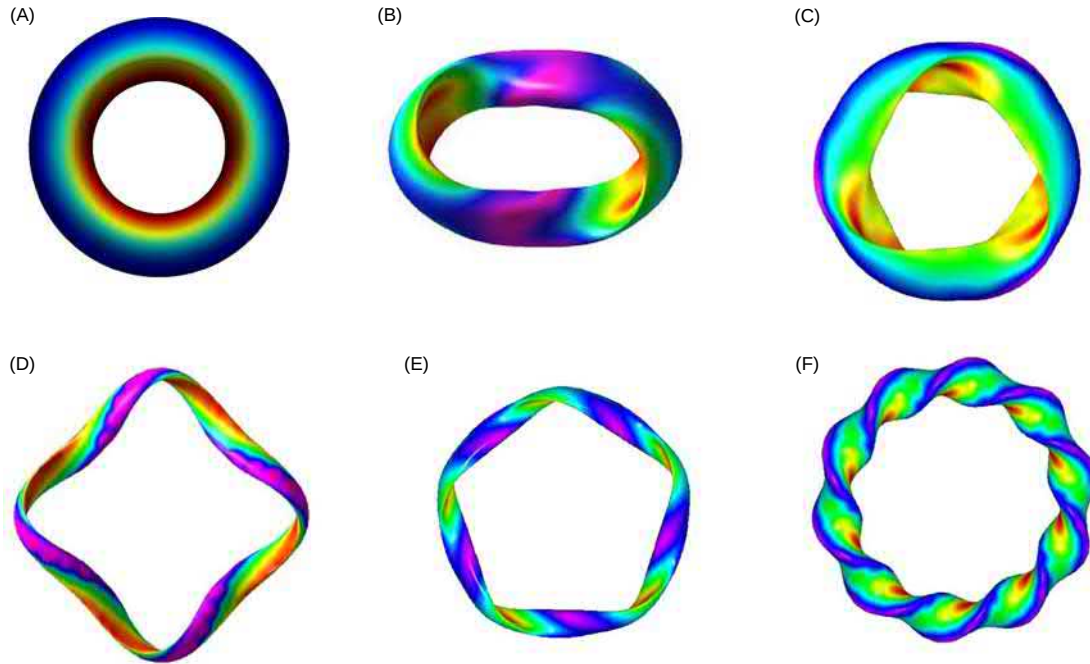


Fig. 4 Distribution of magnetic field strength on the flux surface in (a) ITER, (b) QP, (c) NCSX (QA), (d) HSX (QH), (e) W7-X (QI), (f) LHD. Magnetic field strength increases from blue to red. Courtesy of Dr. Y. Suzuki.

and minor radius of cross-section of the torus, a . Although particle confinement is improved in large aspect ratio, a device becomes very large and only low pressure plasmas can retain force-balance equilibrium. An innovative approach to address the resolution of these drawbacks is recovery of symmetry (Dewar and Hudson, 1998). Although this recovery is not complete, different approaches to optimization have been proposed in a variety of “quasi-symmetry” concepts such as quasi-axisymmetry (QA) (Reiman et al., 1999; Neilson et al., 2000), quasi-helical-symmetry (QH) (Nührenberg and Zille, 1998; Boozer, 1995), quasi-polooidal-symmetry (QP) (Spong et al., 2001) and quasi-isodynamics (QI) (Gori et al., 1997; Nührenberg, 2010).

Natural history and zoology of stellarators

The original idea of stellarators was proposed by Lyman Spitzer Jr. (1914–97) in 1951 (see Fig. 5). He is also known to be a father of Hubble Space Telescope. The stellarator project, coded as Project Matterhorn S, was conducted behind the classified curtain. The project was declassified in 1958 and a milestone paper on stellarators was published in the same year (Spitzer Jr., 1958a). In this article, Spitzer implemented the basic idea of rotational transform and nested flux surfaces by two methods. One involves a twisted circle with toroidal field coils like a “figure-8” (Spitzer Jr., 1958b). It is also surprising that Spitzer already considered a pumping concept, introducing the “divertor”, and heating schemes in his very first paper (Spitzer Jr., 1958a). This figure-8 concept was abandoned shortly afterwards. The second concept described in the paper was a combination of toroidal field coils and pairs of helical coils with currents directed oppositely in adjacent helical coils. Here the number of poles of the magnetic field created by pairs of helical coils is denoted by L . The magnetic configuration of the stellarator is characterized by the toroidal field period M together with L . The classical stellarator, employing pairs of helical coils ($L = 2$ or 3) with currents in opposite directions in the adjacent coils together with toroidal field coils, was pursued in United States, USSR, Germany, United Kingdom, Japan and France (Braams and Stott, 2002). While the American initiative (Young, 1974) was converted to a tokamak at the end of the 1960s, the study of classical stellarators has progressed significantly, in particular in Germany, as the Wendelstein line (Grieger et al., 1985) (WII-A, WII-B, W7-A, W7-AS (Hirsh et al., 2008) and W7-X (Klinger et al., 2019)). Wendelstein, a mountain in Germany (not as high as the Matterhorn), means ‘a helical stone’ in German. Conceptual development led by theory has established the concept of an advanced stellarator (Wobig, 1999) employing modular coils (Chodura et al., 1981).

It is pointed out that another concept employing a pair of helical coils ($L = 2$) with currents in the same direction was envisioned by Koji Uo (1925–92) in 1958 (Uo, 1961). Unlike stellarators, this configuration named “heliotron” produces both helical and toroidal magnetic fields by a pair of helical coils without toroidal field coils. His conception originates from consideration of the magnetic field in a twisted power cable. Since then, this idea has been developed in the heliotron line (Uo, 1971) such as Heliotron C, Heliotron D, Heliotron DM, and Heliotron E (Uo, 1985; Wakatani and Sudo, 1996), leading to the present LHD project, which started operation in 1998 (Iiyoshi et al., 1999) (see Fig. 6). The same configuration with $L = 2$ or 3 helical coils with the same current direction was proposed independently as the torsatron (Gourdon et al., 1969). ATF (Murakami et al., 1991a) in USA, CHS



Fig. 5 Prof. L. Spitzer Jr. and his figure-8 stellarator. Courtesy of Princeton Plasma Physics Laboratory (PPPL).



Fig. 6 Inside view of plasma vacuum vessel of LHD, viewing in both toroidal directions simultaneously. Superconducting coils are arranged behind the helical ridges. The closed divertor system is seen in the groove between ridges. Courtesy of National Institute for Fusion Science (NIFS).

(Matsuoka et al., 1989) in Japan, and URAGAN-2M (Pavlichenko, 1993) in Ukraine in this category produced valuable scientific results around the 1990s.

Heliac originally stands for helical axis stellarator. The basic idea comes from nested helical “magnetic islands” developing at the rational surfaces in a current carrying plasma such as a tokamak (Furth et al., 1966). The combination of an $L = 1$ helical coil and tokamak-like toroidal field coils with centers following a helical line around the torus, as illustrated in Fig. 7, generates twisted kidney-shaped magnetic flux surfaces (Boozer et al., 1983; Harris et al., 1985). This helical configuration is highlighted by a deep “magnetic well,” which is favorable for high β operation. The H-1 heliac (Hamberger et al., 1990) was studied in Australia and has been exported to China. The present leading heliac is TJ-II (Alejaldre et al., 1990, 1999) in Spain.

The development of the natural history of stellarators is schematically shown in Fig. 7. The list of stellarator experimental devices is given in Table 1. Device size and magnetic field strength have grown together with elaborated efforts to reduce unfavorable magnetic error fields which degrade the quality of plasma confinement. Based upon scientific achievements, the line of classical stellarators has evolved into the modular coil concept, which facilitates the implementation of advanced ideas. The present flagship device of the “advanced” stellarator is W7-X, which started operation in 2015 (Pedersen et al., 2017). Optimization to restore symmetry has also been accelerated owing to progress of theory and computational capability in these years. A helically symmetric stellarator, HSX (Anderson et al., 1995, 2006), has been working in the United States and the project of a quasi-axisymmetric stellarator (NCSX) (Zarnstorff et al., 2001) was also launched in the United States, but this project was cancelled in 2008. However, a small scale device based on quasi-axisymmetry, CFQS (Xu et al., 2018), is being built in China in collaboration with Japan, and an optimization design study of a potential project, QUASAR, is progressing in the United States (Gates et al., 2017). Heliotron

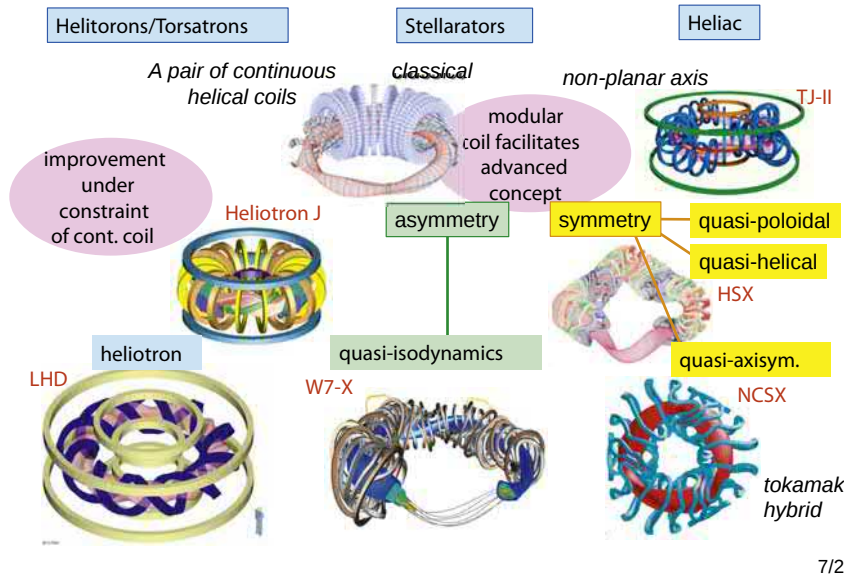


Fig. 7 Schematic development of stellarators illustrating the various concepts described in the text. Materials are due to courtesy of CIEMAT, IPP, Kyoto Univ., NIFS, PPPL and Wisconsin Univ.

J (Wakatani et al., 2000; Obiki et al., 2001) in Japan is a member of a family of stellarators with quasi-poloidal symmetry. LHD and W7-X are pursuing the mission of demonstrating reactor-relevant stellarator plasmas. At present, the potential to provide the basis for a fusion reactor is being explored by the leading three concepts: heliotron (LHD), advanced stellarator (W7-X) and quasi-axisymmetric stellarator.

Major findings and achievements in stellarators

Demonstration of net-current free plasmas

In principle, the stellarator magnetic field configuration produced purely by external coils has the capability to confine high-temperature plasmas. However, until the 1970s, even stellarators required ohmic heating by a plasma current, since there were no other effective schemes to initiate and heat plasmas. Therefore, the early stellarator experiments relied on the generation of net plasma currents, just as in tokamaks. From the late 1970s, auxiliary heating technology (Dumont, 2020), such as Electron Cyclotron Resonance Heating (ECRH) as well as Neutral Beam Injection (NBI), has been applied in stellarator experiments. Two innovative achievements of net-current free or current-less plasmas were reported during the 1980s.

The CLEO stellarator, an $L = 3$ classical stellarator, produced current-less plasma by injecting microwaves at 17.5 GHz without ohmic heating for the first time (Atkinson et al., 1981). Electron energy confinement time was similar to that in ohmically heated plasmas. In W7-A, an $L = 2$ classical stellarator, the plasma current was reduced to zero during NBI heating while keeping the rotational transform unchanged by increasing the current in the helical coils (Bartlett et al., 1981). This current-less operation was successful. MHD instabilities (Lao et al., 2021) such as “tearing modes” were suppressed and energy confinement was improved compared with ohmically heated plasma.

Initiation and heating by ECRH were demonstrated in Heliotron E (Iiyoshi et al., 1982) soon after CLEO, and subsequently this scheme has been established as a standard mode of operation of stellarators. It has also been proved that the advantage of scale in LHD has enabled plasma initiation even by collisions between neutral gas and NBI-particles (Kaneko et al., 1999). This method is useful, since the operational magnetic field is not constrained by the resonance condition required for ECRH.

With regard to plasma current, it should be noted that intrinsic “bootstrap currents” (Sarazin, 2021) driven by the plasma pressure gradient are generated in stellarator plasmas. There are three principal means to control or suppress bootstrap currents among the three leading stellarator concepts noted above. Since quasi-isodynamic stellarators, such as W7-X and Heliotron J, are characterized by weak magnetic shear, bootstrap currents change the rotational transform profile and often generate magnetic islands. In particular, deformation of the plasma boundary defined by chains of magnetic islands must be avoided in order to ensure the divertor function (Geiger et al., 2010). Therefore, minimization of bootstrap currents is incorporated in the optimization of magnetic geometry. In contrast, the heliotron configuration, such as LHD, is characterized by strong magnetic shear, and the magnetic configuration, which incorporates the divertor magnetic structure, is robust against bootstrap currents. The third concept is the “quasi-axisymmetric” stellarator with low aspect ratio. Since its rotational transform in vacuum is small, a contribution from bootstrap currents is intrinsically necessary to optimize this magnetic configuration. Therefore, the QA stellarator can be referred to as a tokamak-stellarator hybrid.

Table 1 List of stellarators. Here L is the number of poles of the magnetic field created by pairs of helical coils and M is the toroidal field period.

Device name	Country	Institute	Period	Concept	R (m)	a (m)	B (T)	L	M
Model C stellarator	USA	PPPL	1961–1969	Class.Stell.	1.0	0.05–0.075	3.5	2/3	8
L1	USSR	LPI	1963–1971	Class.Stell.	0.6	0.05	1	2	14
Sirius	USSR	KIPT	1964–1975	Class.Stell.	0.6	0.026	1.6	3	–
Wendelstein 1-A	Germany	MPIPP-Garching	1965	Class.Stell.	0.35	0.02	1	3	–
Wendelstein 2-A	Germany	MPIPP-Garching	1967	Class.Stell.	0.5	0.05	0.6	2	5
TWIST	UK	Culham Laboratory	1967	Class.Stell.	0.32	0.045	0.3	3	4
CLASP	UK	Culham Laboratory	1967	Class.Stell.	0.3	0.056	0.1	3	8
TOR-1	USSR	LPI	1967–1973	Class.Stell.	0.63	0.036	2.5	2	16
Wendelstein 1-B	Germany	MPIPP-Garching	1968	Class.Stell.	0.35	0.02	1	2	–
Proto-Cleo	USA/UK	Culham Laboratory Wisconsin Univ.	1968	Class.Stell. Torsatron	0.4	0.039	0.3	3	7/13
Uragan	Ukraine	KIPT	1968–1976	Torsatron	1.1	0.067	1	3	10
Satrun-1	Ukraine	KIPT	1970	Torsatron	0.36	0.063	0.7	3	8
JIPP-Ia	Japan	IPP-Nagoya Univ.	1970–1973	Class.Stell.	0.5	0.034	0.4	3	8
Wendelstein 2-B	Germany	MPIPP-Garching	1971	Class.Stell.	0.5	0.05	1.25	2	5
Heliotron D	Japan	Kyoto Univ.	1971	Heliotron	1.09	0.1	0.3	2	25
Vint-20	Ukraine	KIPT	1972	Torsatron	0.32	0.075	2	1	13
TORSO	UK	Culham Laboratory	1972	Class.Stell.	0.4	0.05	2	3	12
HIDRA/WEGA	USA	Univ. Illinois	1972–	Class.Stell.	0.72	0.19	0.5	2	5
	Germany	MPIPP-Greifswald							
	France	CEA-Grenoble							
JIPP-Ib	Japan	IPP-Nagoya Univ.	1973–1976	Class.Stell.	0.5	0.07	0.4	2/3	4/8
CLEO	UK	Culham Laboratory	1974	Class.Stell.	0.9	0.125	2	3	7
Wendelstein 7-A	Germany	MPIPP-Garching	1975	Class.Stell.	2.0	0.1	3.5	2	5
L2	USSR	LPI	1975–1992	Class.Stell.	1.0	0.115	1.3	2	14
	Russia	GPI							
Heliotron DM	Japan	Kyoto Univ.	1976	Heliotron	0.45	0.04	1	2	21
Uragan 2	USSR	KIPT	1976–1981	Torsatron	1.1	0.067	2	3	18
JIPPT-II	Japan	IPP-Nagoya Univ.	1976–1981	Class.Stell.	0.91	0.17	3	2	4
Heliotron E	Japan	Kyoto Univ.	1980–1997	Heliotron	2.2	0.2	2	2	19
Heliotron DR	Japan	Kyoto Univ.	1981	Heliotron	0.9	0.07	0.6	2	15
Uragan 3	USSR	KIPT	1982–1988	Torsatron	1.0	0.135	2.5	3	9
Auburn Torsatron	USA	Auburn Univ.	1983–1990	Torsatron	0.58	0.14	0.2	2	10
IMS	USA	Wisconsin Univ.	1983–1999	Modular Stell.	0.4	0.04	0.3	3	7
Shiela	Australia	Australian National Univ.	1984	Heliac	0.2	0.03	0.3	1	3
TU-Heliac	Japan	Tohoku Univ.	1988–2017	Heliac	0.48	0.06	0.3	1	4
ATF	USA	ORNL	1988–1995	Torsatron	2.0	0.27	2	2	12
Wendelstein 7-AS	Germany	MPIPP-Garching	1988–2002	Adv.Stell.	2.0	0.18	3	–	5
CHS	Japan	IPP-Nagoya Univ.	1988–2006	Torsatron	1.0	0.2	1.5	2	8
		NIFS							
Urgan-3M	USSR	KIPT	1988–	Torsatron	1.0	0.12	1.3	3	9
	Ukraine								
CAT	USA	Auburn Univ.	1990–2000	Torsatron	0.52	0.11	0.1	1/2	5
H-1NF	Australia	Australian National Univ.	1992–	Heliac	1.0	0.2	0.5	1	3
	China	Univ. South China							
L2-M	Russia	GPI	1993–	Class.Stell.	1.0	0.115	1.5	2	14
TJ-K/TJ-IU	Spain	CIEMAT	1994–	Torsatron	0.6	0.1	0.5	1	6
	Germany	Kiel Univ./Stuttgart Univ.							
LHD	Japan	NIFS	1998–	Heliotron	3.7	0.63	2.85	2	10
HSX	USA	Wisconsin Univ.	1999–	QH	1.2	0.15	1.25	–	4
TJ-II	Spain	CIEMAT	1999–	Heliac	1.5	0.22	1	1	4
Heliotron J	Japan	Kyoto Univ.	2000–	QP/QI	1.2	0.2	1.5	1	4
CNT	USA	Columbia Univ.	2004–	Two interlinked coils	0.3	0.10	0.2	1	1
CTH	USA	Auburn Univ.	2005–	Torsatron	0.75	0.29	0.7	2	5
Uragan-2M	Ukraine	KIPT	2006–	Torsatron	1.7	0.22	2.4	2	4
Wendelstein 7-X	Germany	MPIPP-Greifswald	2015–	Adv.Stell./QI	5.5	0.5	2.5	–	5
SCR-1	Costa Rica	ITCR	2016–	Modular Stell.	0.25	0.04	0.044	–	2

PPPL: Princeton Plasma Physics Laboratory; LPI: Lebedev Physical Institute; KIPT: Kharkov Institute of Physics and Technology; MPIPP: Max-Planck-Institut für Plasmaphysik; CEA: Commissariat à l'énergie atomique et aux énergies alternatives; IPP-Nagoya Univ.: Institute of Plasma Physics, Nagoya University; NIFS: National Institute for Fusion Science; GPI: General Physics Institute; ORNL: Oak Ridge National Laboratory; CIEMAT: Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas; ITCR: Instituto Tecnológico de Costa Rica. Class.: Classical; Stell.: Stellarator; Adv.: Advanced; QH: Quasi-Helical symmetry; QP: Quasi-Poloidal symmetry; QI: Quasi-Isodynamicity.

Performance of energy confinement time and its empirical scaling

Needless to say, energy confinement time is a primary parameter for demonstrating the performance of plasma magnetic confinement. Comparative studies of multiple experiments, extending beyond the plasma parameter scans possible in a single device have been developed, as is the case for tokamaks. A pioneering work compiling published data from Heliotron E, W7-A, L2 and Heliotron DR was reported in 1990 (Sudo et al., 1990), and this showed favorable density dependence, which is not seen in tokamaks, and a power degradation (i.e. a reduction in energy confinement time with increasing heating power), as seen in tokamaks. A more comprehensive analysis was conducted by an international collaboration on the International Stellarator Database. This confinement database was compiled by collecting data from medium sized stellarators: ATF, CHS, Heliotron E, W7-A and W7-AS. Careful statistical regression analysis showed an offset between the heliotron/torsatron line, i.e. with a strong positive magnetic shear and a limited magnetic well, and the Wendelstein concept, having very low shear but a magnetic well. The dependence on other operational parameters was, however, very similar for the two concepts. Averaging data across the offset, the International Stellarator Scaling 95 (ISS95) (Stroth et al., 1996) was derived as,

$$\tau_E^{ISS95} = 0.079 a^{2.21} R^{0.65} P^{-0.59} \bar{n}_e^{0.51} B^{0.83} \iota_{2/3}^{0.4},$$

where a , R , P , $\bar{n}_e$, B , and $\iota_{2/3}$ are minor radius in m, major radius in m, absorbed heating power in MW, line averaged density in 10^{19} m^{-3} , magnetic field in T and rotational transform at the two thirds minor radius. The ISS95 scaling was subsequently upgraded to the ISS04 scaling based on an extended database of more than 3000 data points which included data of the large-scale plasmas in LHD and the different magnetic configurations of TJ-II and Heliotron J (see Fig. 8) (Yamada et al., 2005). While a very similar dependence on the key operational parameters has been reconfirmed in the later scaling, a renormalization factor has been introduced for each subgroup to quantify offsets between different magnetic configurations, allowing the energy confinement time to be expressed in a unified scaling law:

$$\tau_E^{ISS04} = 0.134 a^{2.28} R^{0.64} P^{-0.61} \bar{n}_e^{0.54} B^{0.84} \iota_{2/3}^{0.41}.$$

Both ISS95 and ISS04 satisfy dimensional constraints derived from underlying physics consideration and exhibit a gyro-Bohm nature (Jenko, 2021), with $\tau_E^{ISS04} \Omega_i \propto \rho_*^{-2.79}$, where ρ_* and Ω_i are the ion gyro radius normalized by plasma minor radius and the ion cyclotron frequency, respectively. It should be emphasized that these scaling expressions also describe tokamak data surprisingly well by translating rotational transform in stellarators to equivalent plasma current (Dinklage et al., 2007). The study of ISS04 has also suggested that effective “helical ripple” has significant correlation with the renormalization factors noted above and that transport is determined by turbulent transport rather than neoclassical (i.e. collisional) transport (Yamada et al., 2005).

With regard to the isotopic mass effect on the energy confinement time, detailed comparison of hydrogen and deuterium plasmas in LHD has shown no significant dependence on the atomic mass of ions, M (Yamada et al., 2019):

$$\tau_E^{scl} = 0.072 M^{0.00} B^{0.84} \bar{n}_e^{0.76} P_{abs}^{-0.87},$$

which has verified provisional observations in the mid-size stellarators (ATF, W7-AS and Heliotron E) (Stroth et al., 1995; Yamada et al., 2004). The isotope effect, i.e. the observed dependence of the energy confinement time on the fuel isotopic mass, is a long-standing unresolved mystery in tokamaks and this study would contribute to a comprehensive physical understanding. Here it should be noted that a clear mass dependence appears together with the persistence of the gyro-Bohm nature when the scaling expression is rephrased in non-dimensional parameters:

$$\tau_{E,th}^{scl} \Omega_i \propto M^{0.99} \rho_*^{-2.98} \nu_*^{0.19} \beta^{-0.30},$$

where ν_* and β are the electron-ion collision frequency (Sarazin, 2021) normalized by the bounce frequency of the banana orbit (of trapped particles) and the plasma kinetic pressure normalized by the pressure of the magnetic field, respectively.

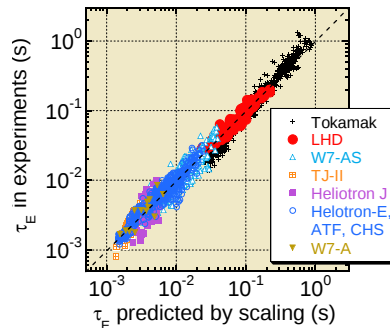


Fig. 8 Comparison of experimentally measured values of the energy confinement time in stellarator experiments with predictions derived from the ISS04 scaling, which is discussed in the text. A range of tokamak experimental data of H-mode plasmas (Kardaun, 1999) in which the plasma current has been translated into a rotational transform, is also shown for comparison.

H-mode

The H-mode regime of plasma confinement, highlighted by significantly improved confinement, has been studied very intensively in tokamaks since its first discovery on the ASDEX tokamak (Wagner et al., 1982). Since that time, it has been confirmed that the H-mode develops in a variety of stellarators over wide parameter ranges (Wagner et al., 2006). The H-mode transition has been reported from W7-AS (Erckmann et al., 1993), CHS (Toi et al., 1993), LHD (Toi et al., 2005), Heliotron J (Sano et al., 2005), H-1 (Shats and Rudakov, 1997) and TJ-II (Alejandre, 2004). The Tohoku heliac has also induced H-mode by inserting an electric biasing probe (Kitajima et al., 2004). This means that the development of the H-mode is a ubiquitous and fundamental phenomenon among toroidal plasmas. The transition to, and development of, the H-mode is closely linked with the radial electric field and its shear in the edge (Jenko, 2021). The pressure gradient plays a leading role in determining the radial electric field structure in tokamaks, although additional physics mechanisms, while still controversial, remain under investigation (Tynan et al., 2016). In addition to these mechanisms, an intrinsic radial electric field exists due to the ambipolar condition in stellarators (see discussion in the section *Identification of transport properties*).

The first identification of H-mode in a stellarator was performed in plasmas with low magnetic shear on W7-AS (Erckmann et al., 1993). The H-mode transition, accompanied by a characteristic drop of the H_α light emission and an increase of density, was observed in plasmas using an island divertor configuration and operating above a certain threshold density ($4\text{--}5 \times 10^{19} \text{ m}^{-3}$) which were heated by electron cyclotron waves. A steep pressure gradient developed in the plasma edge which correlated with formation of a deep negative electric field inside the magnetic separatrix. Characteristic edge localized MHD modes (ELMs), which expel energy, have been observed, although quiescent H-modes without ELMs have also been found in a narrow operational space above a power dependent critical density. In contrast to the situation in tokamaks, a critical level of heating power, or “power threshold”, was not observed. This may imply a contribution to the H-mode transition of a pre-existing radial electric field due to neoclassical ambipolarity in stellarator plasmas. At higher density, the H-mode in W7-AS developed into another characteristic regime referred to as a high-density H-mode (HDH) (McCormick et al., 2002). The HDH mode exhibits an improvement of energy confinement time by a factor of 2 beyond the prediction of the ISS95 scaling and simultaneously exhibits a significant drop of impurity confinement time. The significant consequence of this behavior is that plasmas in the HDH mode purify themselves (i.e. reduce the impurity concentration) and allow steady-state operation close to the density limit. Observations in Heliotron J share significant commonality with those in W7-AS. The H-mode transition occurs in an isolated narrow range of rotational transforms in the vicinity of rational values in both Heliotron J and W7-AS, which is interpreted as implying that plasma poloidal viscosity plays an important role in the H-mode transition (Wobig and Kisslinger, 2000). The dependence of the transition on the fuel isotope mass, in particular the increased ease of transition in deuterium plasmas compared to hydrogen plasmas, which is clear in tokamaks, was not been identified in Heliotron J and W7-AS.

In the heliotron line, with high magnetic shear, the H-mode transition was driven by increased rotational transform produced by supplemental plasma current in CHS (Toi et al., 1993). While the observed density threshold scaled as in tokamaks, the power threshold was higher by a factor of 2, which was interpreted as being due to the limiter configuration used in these experiments, as opposed to the divertor configuration commonly used for H-mode experiments in tokamaks. H-mode transitions have also been observed in LHD and are most pronounced in high β plasmas at low magnetic field (0.5–0.75 T) (Toi et al., 2005). The improvement of confinement in CHS and LHD resulting from the H-mode transition is limited to several 10s %, compared to a factor of 2 typically observed in tokamaks. Since the edge region is associated with a “magnetic hill” in the heliotron, an “interchange mode” appears immediately on development of a significant pressure gradient, limiting the improvement in confinement.

High beta

Achievement of high beta (β), where β is the kinetic pressure normalized by the pressure of the magnetic field, is a critical mission towards demonstrating an attractive fusion reactor, not only in tokamaks but also in stellarators. While high β is normally favorable, or even essential, for an efficient and economical reactor, it has another important impact on stellarators. Since the confining magnetic field is produced by external coils, these coils should be located as close as possible to the plasma in a stellarator in order to ensure a satisfactory magnetic field configuration. However, the space for the blanket, with about 1 m of thickness, must be accommodated. Then, if slim magnets are to be incorporated, implying low magnetic field and high β value, the reactor size could be reduced.

Since stellarator plasmas are current-less, MHD instabilities which prevent high β operation are driven by the pressure gradient and not by the current density gradient. Therefore, stellarators are free from “kink modes” and “tearing modes,” and the primary concerns are “interchange modes” and “ballooning modes.” Basically, the concept of quasi-symmetry in stellarators involves a shear-less magnetic field profile and MHD stability is secured by a magnetic well. In contrast, the heliotron configuration, which has a radially limited magnetic well, stabilizes MHD instabilities by strong magnetic shear.

Experimental achievement of high β has progressed together with stellarator concept development and advances in heating technology. The first notable result was the achievement of β of 2% in NBI heated plasmas in Heliotron E (Motojima et al., 1985). Subsequently, CHS recorded a slightly higher β of 2.1% (Okamura et al., 1995) and W7-AS extended the range further to β of 3.4% (Weller et al., 2003). The present record is a β value of 5% in LHD (Sakakibara et al., 2008). This reactor-relevant high β in LHD has been sustained for longer than 100 energy confinement times, as shown in Fig. 9. It has been shown throughout these experimental observations that the MHD instability limit to pressure in stellarators is benign (Yamada et al., 2010). MHD-driven

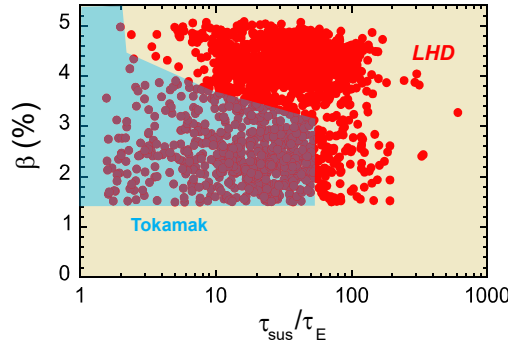


Fig. 9 Achievement and sustainment of high- β plasmas in LHD compared with β values achieved in a representative range of tokamak experiments. The abscissa is the duration of sustainment normalized by the energy confinement time.

disruptions, which terminate the plasma in tokamaks, have not been observed. In general, MHD instabilities are not harmful in stellarators and only degrade confinement. As a result, the β limit is soft and determined by energy confinement and heating power. Although linear MHD theory has been successful in predicting the β limit in tokamaks, stellarator observations have stimulated the development of more sophisticated models to explain the stabilizing mechanism allowing sustained operation at high β . Recent non-linear simulations including kinetic effects of trapped particles (Sato and Todo, 2019) and parallel thermal conductivity (Miura and Nakajima, 2010) are beginning to resolve this issue.

High density

High density operation is favorable for achieving high fusion power, as seen in the density dependence in the scaling of energy confinement time. In particular, high collisionality due to high density mitigates heat losses via neoclassical transport enhanced by a helical ripple in stellarators. High density is also advantageous in mitigating the divertor heat load. The so-called Greenwald density is a well-established criterion for assessment of the density limit in tokamaks (Greenwald et al., 2002), where disruptions often determine the highest achievable density. In contrast to tokamaks, a radiative collapse limits the achievable density in stellarators. Moreover, while steady-state operation of tokamaks benefits from operation at low density and high temperature, in order to ensure high current drive efficiency (Dumont, 2020), stellarators do not have this kind of constraint on operational density.

The most well-established expression of the density limit in stellarators is given by the Sudo scaling (Sudo et al., 1990)

$$n_e^{\text{Sudo}} = 0.25 P^{0.5} B^{0.5} a^{-1} R^{-0.5}$$

where n_e^{Sudo} is in units of 10^{20} m^{-3} . This scaling is based upon observations in medium-sized experiments, such as Heliotron E, W7-A, L2 and Heliotron DR. The background to this expression has been explained by a theory formulating the power balance between heating power and impurity radiation loss (Itoh et al., 2001; Zanca et al., 2019).

Later experiments in W7-AS (Giannone et al., 2000) and LHD (Peterson et al., 2001) showed that the accessible density is consistent with the picture of the Sudo scaling, but observed a significant improvement beyond its prediction. Since the radiative collapse is triggered by edge radiation, the edge density and local concentration of light impurities are considered to be essential parameters (Miyazawa et al., 2008).

Particularly noteworthy features of high density operation in stellarators include the high density H-mode (HDH) in W7-AS, noted previously, and the super-dense-core (SDC) in LHD (Ohyabu et al., 2006). The HDH in W7-AS has provided access to stationary high density operation at densities of up to $4 \times 10^{20} \text{ m}^{-3}$ at 2.5 T (McCormick et al., 2002). The SDC in LHD was discovered when using intensive core fueling by cryogenic pellet injection into plasmas with efficient pumping via a “local island divertor” (LID) (Morisaki et al., 2005). A steep density gradient develops in the core region and forms an extremely high density core. This operation was extended to the helical divertor configuration. When the magnetic configuration forms a thick layer with ergodic field lines in the plasma boundary, the SDC can be formed without active pumping. Under these conditions, the central density has reached $1.2 \times 10^{21} \text{ m}^{-3}$ at 2.5 T (Yamada et al., 2007). The SDC is accompanied by an improvement in energy confinement and the central pressure reaches 1 atm. It should be noted that in neither the HDH in W7-AS nor the SDC in LHD was significant impurity accumulation observed, which is an important result in relation to the potential application of these regimes in a stellarator reactor. Finally, when the rotational transform is replaced by plasma current, which is the key parameter of the Greenwald density in tokamaks, the operational high density regime of LHD and W7-AS far exceeds the Greenwald density (see Fig. 10).

Steady-state operation

Since stellarators can form the confining magnetic field using only external coils, steady-state operation, a key aspect of reactor operation, is intrinsically possible. On the other hand, demonstration of steady-state operation requires a sufficient engineering capability of the device. The key elements are steady-state operation of heating systems and a compatible heat removal capability, in

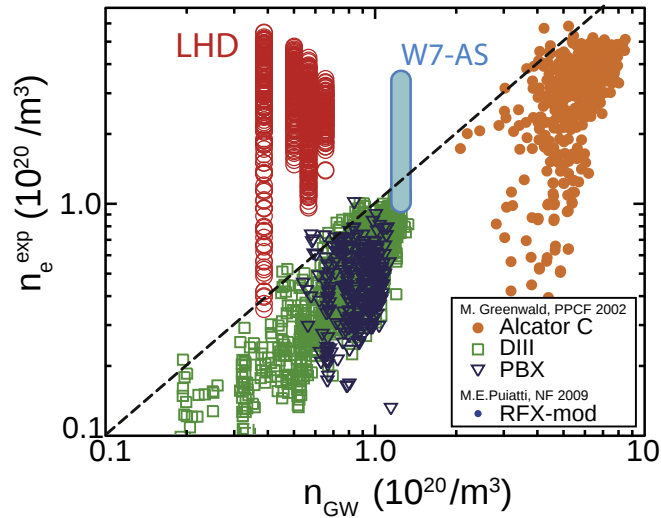


Fig. 10 Comparisons of the achieved plasma density in stellarator and tokamak experiments with the Greenwald density, $n_{GW} = I_p / \pi a^2$, with n_{GW} the predicted Greenwald density in 10^{20} m^{-3} , I_p the plasma current in MA and a the plasma minor radius in m. Data of the SDC in LHD and the HDH in W7-AS are plotted by translating the rotational transform to an equivalent plasma current. Data from the tokamak experiments Alcator-C, DIII and PBX (Greenwald et al., 2002), and the reversed field pinch experiment RFX-mod (Puiatti et al., 2009) are shown for comparison.

particular, from divertor plates. For example, although the ATF device employed normal Cu conductor for the helical coils, it was possible to operate in steady-state at a magnetic field at 0.5 T. ECRH operating at 28 GHz with a power level of 70 kW produced a current-less plasma with a pulse length greater than 1 h (4667 s) (Jernigan et al., 1995). Plasma parameters were modest (a density of several $\times 10^{18} \text{ m}^{-3}$ and a temperature of around 100 eV). The steady-state operation regime has been greatly extended in LHD by exploiting the superconducting coils. The typical steady-state discharge sustained by 0.26 MW of ECRH and 0.94 MW of ion cyclotron radiofrequency heating (ICRF), i.e. 1.2 MW total heating power, at 2.75 T for 2859 s exhibited plasma parameters with a density of $1.2 \times 10^{19} \text{ m}^{-3}$ and a temperature of 2 keV. The total injected heating energy to the plasma reached 3.4 GJ (Kasahara et al., 2014). While plasma parameters, as well as the temperature of the divertor plates, were held constant, a temporal change of recycling fuel gas from the plasma facing surfaces was observed in conjunction with the evolution of co-deposition of divertor material (carbon) and fuel gas on the wall (Motojima et al., 2015). The plasma pulse was terminated as a result of flaking of material from the wall or divertor, leading to penetration of impurities into the plasma (Shoji et al., 2016).

Figure 14 in Wolf et al. (2019) summarizes the performance of LHD and W7-X in terms of the achieved fusion triple product as a function of plasma pulse length and compares the results of several leading tokamak experiments. The W7-X stellarator foresees the production of stationary plasmas lasting for 30 min. With 10 MW of heating and an actively cooled divertor (Hathiramani et al., 2018).

Control of detached plasma

Limiting the divertor heat load in a fusion reactor to the level which can be handled by practical engineering solutions (Federici et al., 2021) is one of most critical issues and, while experimental progress has been made in this respect, there remains a significant discrepancy between the requirements in a fusion reactor and experimental achievements. Moreover, considerable further work is required to develop a fully reliable predictive theory. In this regard, stable detachment of the plasma in front of the divertor plate is a prerequisite to mitigate divertor heat load, regardless of whether a stellarator or tokamak forms the reactor core.

The HDH regime in W7-AS has been investigated intensively with regard to its compatibility with detachment. While energy confinement improved almost linearly until the plasma density reached the threshold of partial detachment, it decreased as the density increased further due to the encroachment of radiation into the core region following the onset of partial detachment. The ratio of radiation to heating power reached values of up to 90% at partial detachment. Considerable neutral compression in the divertor region was observed simultaneously, which is a precondition for efficient active pumping of exhaust gases, including helium. Further increase of density towards full detachment caused the penetration of the radiation layer into the core and eventually led to a radiative collapse. While favorable detachment behavior was observed in the HDH regime, stable detachment was obtained in a specific geometry surrounded by a sufficiently large magnetic island to reduce the penetration of neutral gases and impurities into the core plasma, i.e. to improve neutral and impurity screening efficiency (Feng et al., 2005).

Detachment often suffers from a thermal instability which causes radiative collapse of the plasma. In LHD, it has been found that application of resonant magnetic perturbations (RMP) promotes local enhancement of radiation and consequently facilitates the access to stable detachment. The RMP with $m/n = 1/1$ poloidal and toroidal mode numbers generates a magnetic island in an edge ergodic layer (Kobayashi et al., 2019). Although the energy confinement decreases due to reduction of the effective plasma radius by the generated magnetic island, the pressure profile becomes peaked during detachment. This suggests that the core confinement is

not degraded even with enhanced radiation and reduced divertor flux. Impurity penetration is prevented by screening through friction with the downstream particle flux. The observations in LHD have a great deal in common with the HDH regime.

Stable detachment with radiation reaching the level of the absorbed heating power in the core plasma has been achieved in W7-X (Zhang et al., 2019). Experiments in W7-X exhibit stable sustainment of a detached plasma for 28 s at $1.1 \times 10^{20} \text{ m}^{-3}$ with 5 MW of ECRH. The radiation fraction is at a level of 80–90% and the duration is limited by the capacity of the uncooled divertor used in these experiments. Following the upgrade to an actively cooled divertor and to an increased heating power of 10 MW, these favorable results will be investigated further in the extended parameter regime.

Identification of transport properties

In relation to plasma transport and confinement in the stellarator configuration, the major concern arises from the influence of neoclassical transport enhanced by the intrinsic “helical ripple” of the magnetic field. According to neoclassical theory, the confinement capability of the stellarator is significantly degraded by the enhancement of neoclassical transport as the collisionality decreases, i.e., as the temperature increases. The neoclassical theory of plasma transport in stellarator plasmas is well established and based on a rigorous formulation. On the other hand, competitive transport processes produced by turbulence often mask the neoclassical aspects of plasma transport.

Optimization of the magnetic field to reduce neoclassical transport has been driven by theory (Shaing and Hokin, 1983; Lotz and Nührenberg, 1988; Mynick, 2006). Each approach has been validated in experiments and has been successful to a significant extent. Nowadays, it is well accepted that, when neoclassical transport is reduced, heat transport in stellarators is governed by short-wave turbulence, as in tokamaks. For example, in LHD, the inward shift of the magnetic axis, i.e. to smaller major radius, has a favorable influence derived from the common physics of quasi-isodynamics. While, in the case with the inward shifted magnetic axis, the heat transport and the energy confinement time are not degraded even in the reactor-relevant low collisionality regime, cases with the outward shifted magnetic axis exhibit confinement degradation with decreasing collisionality, which implies enhancement of the neoclassical transport (Yamada et al., 2001a). Another important observation in this comparison is that improvement of confinement by the inward shift persists in the high collisionality regime, where helical ripple transport is suppressed. Since turbulent transport plays an essential role in the high collisionality regime, it is concluded that the inward shift of the magnetic axis suppresses turbulent transport. Indeed, a more comprehensive study of the international stellarator database indicated that the normalization factor relative to the average scaling, which depends on magnetic configuration, has a correlation with the effective helical ripple, and suggested that neoclassical optimization could suppress turbulent transport (Yamada et al., 2005).

A comparative study of dimensionally similar plasmas has been established in ATF (Murakami et al., 1993), W7-AS (Stroth et al., 1993) and LHD (Yamada et al., 2001b) in order to clarify the characteristics of heat transport, in particular, its dependence on normalized gyro radius, ρ^* . All analyses have shown a common gyro-Bohm nature, that is the heat diffusivity, χ , scales with $\rho^* \chi_B$, where $\chi_B = 16T/(eB)$ is the Bohm diffusivity, and T and e are the temperature and elementary charge respectively. Recent experiments in LHD made a detailed comparison of dimensionally identical plasmas with hydrogen and deuterium fuel and showed that $\chi \propto M\rho^* \chi_B$, with M the isotopic mass of the plasma ions (Yamada et al., 2019). The identified mass dependence is consistent with the characteristics of the energy confinement time, but the underlying physics determining this dependence remains an open issue.

It should be noted that, since the magnetic field in a stellarator is not axisymmetric, a radial electric field is generated by the ambipolar condition to balance electron and ion particle fluxes (Hastings et al., 1985), while in an axisymmetric system, such as a tokamak, particle fluxes are intrinsically ambipolar. The radial electric field generated in the low collisionality regime in stellarators has significant potential to suppress neoclassical transport. Many experiments have shown that a radial electric field is generated by the balance of neoclassical fluxes, even in cases where turbulence dominates heat and particle transport (Yokoyama et al., 2007). This implies that turbulent transport is ambipolar. It should be noted that there exist multiple roots in the neoclassical ambipolar condition. In particular, the “electron root,” which is developed when the flux of electrons trapped in the helical ripple dominates over the ion flux, produces a strong positive radial electric field. The resultant, greatly reduced, neoclassical transport allows very high electron temperatures to be achieved. This mode is referred to as Core Electron Root Confinement (CERC). In addition, radial variation between the electron and the ion root results in a strong electric field shear, which produces sheared poloidal flows of plasma, and this may lead to suppression of turbulence (Jenko, 2021).

Particle transport has not been analyzed in as much detail as heat transport and is less understood. Nonetheless, a pronounced feature that has commonly been observed in stellarator plasmas is a hollow radial density profile. Unlike sources of heat, the particle source is usually localized in the edge and then a simple diffusive analysis predicts a flat density profile. While the so-called anomalous inward pinch, which has been proposed to explain the peaked density profile observed in tokamak plasmas, is not usually observed in stellarator plasmas (Takenaga et al., 2008), the density profile tends to be hollow due to an outward convection driven by the temperature gradient, which is documented in the framework of neoclassical transport (Yamaguchi et al., 2007). This trend is expected to become more pronounced in large stellarators with low collisionality (Maassberg et al., 1999).

Other important physical processes driven by neoclassical transport are the bootstrap current and parallel viscosity. Both are due to transport parallel to the magnetic field, in contrast to heat and particle transport. With regard to the bootstrap current, good quantitative agreement with the predictions of neoclassical theory has been confirmed in ATF (Murakami et al., 1991b), CHS (Yamada et al., 1994) and Heliotron J (Motojima et al., 2007). In addition, reversal of the bootstrap current by changing the plasma elongation was observed in ATF and CHS, in agreement with theoretical predictions relating to the situation where the major contribution to the bootstrap current is switched from trapped particles in the depressed magnetic field on the outboard side to those in the

helical ripple. Damping of toroidal plasma flow owing to parallel viscosity was investigated by changing the magnetic field modulation in CHS (Ida et al., 1991). It was found that the toroidal velocity profile was dominated by the perpendicular viscosity, i.e., perpendicular momentum transport, when the modulation was weak. As the magnetic toroidal modulation increased, the toroidal velocity became dominated by the parallel viscosity. This observed tendency is consistent with neoclassical theory and has shown quantitative agreement with theory to within a factor of 3.

In summary, heat and particle transport in stellarator plasmas is dominated by turbulence and the radial electric field determined by the neoclassical ambipolar condition has a noteworthy impact on the turbulence. Moreover, neoclassical effects are pronounced in parallel transport processes associated with trapped particles, as manifested in phenomena such as the bootstrap current and parallel viscosity. A diagram to summarize the linkages among transport processes will be given in the next section, including a consideration of 3-D effects.

3D physics

Stellarators are intrinsically 3D magnetic configurations. Nowadays, tokamaks also require a 3D physics analysis to deal, for example, with physics processes related to the use of resonant magnetic perturbation to suppress/mitigate ELMs, with the damping of flows due to neoclassical toroidal viscosity and with plasma flows impeding magnetic island formation. Transport in the divertor and scrape-off layer is inevitably 3D, even in a tokamak. There is commonality in these physics issues between stellarators and tokamaks. Related examples have already been discussed in previous sections.

The theoretical description of a 3D MHD equilibrium has been confirmed in experiments on CHS (Yamada et al., 1992) and W7-AS (Weller et al., 2003). Beginning with the “Shafranov shift” due to plasma pressure, the deformation of flux surfaces, including the generation of magnetic islands and ergodic layers, is well described by 3D MHD equilibrium numerical codes such as VMEC (Hirshman, 1986) and HINT (Suzuki et al., 2006, 2013). In addition, spontaneous stochasticization of the magnetic field has been confirmed by heat pulse propagation and flow damping in a detailed experiment in LHD (Ida et al., 2017).

As describe in the previous section, the radial electric field generated through neoclassical transport in 3D affects turbulence. Moreover, the geometrical effect of the effective helical ripple and the isotope effect have been identified in experiments, and advanced gyro-kinetic simulations are beginning to clarify the responsible physics mechanism through the characteristics of “zonal flows” (Jenko, 2021) in 3D (Watanabe et al., 2008; Nakata et al., 2017). Here it should be emphasized that the first experimental identification of a zonal flow was carried out in CHS (Fujisawa et al., 2004). The linkage of various physical mechanisms involved in the determination of transport in toroidal plasmas is summarized in Fig. 11 (Yamada, 2009). Radial profiles of temperature and density are determined by turbulence self-consistently with gradients of temperature and density, shear of the radial electric field and zonal flows in 2D. In contrast, or complementarily, the radial electric field due to 3D neoclassical transport affects the shear of the radial electric field, the zonal flows via collisional damping and neoclassical transport itself.

Status of current leading projects

The device parameters of LHD and W7-X, the two largest stellarator experiments, are summarized in Table 2.

Large helical device (LHD)

The Large Helical Device (LHD), shown in Fig. 2a–c and located in the National Institute for Fusion Science (NIFS, situated in Toki-city, Japan), is based on a heliotron concept which employs a pair of continuous helical coils (Iiyoshi et al., 1990; Special Issue, 2010). The NIFS is an inter-university research institute and the LHD is open to research collaborators. The detailed device

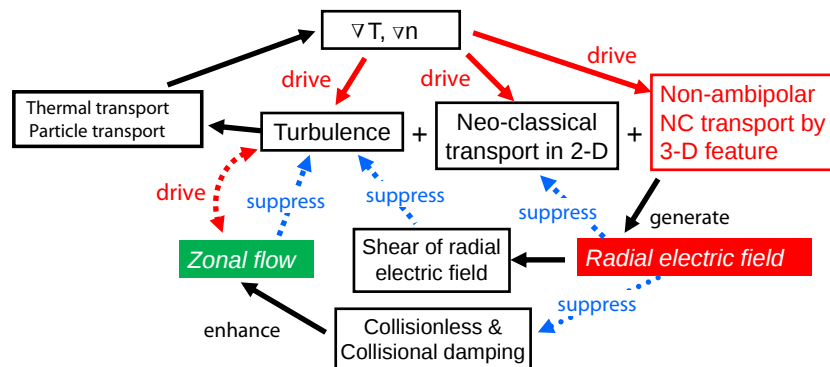


Fig. 11 Schematic inter-linkage of plasma transport processes in 3-D. ‘NC’ refers to neoclassical transport theory. The various contributing physics processes shown in the diagram are discussed in the text.

Table 2 Specifications and typical plasma parameters of LHD and W7-X.

	<i>LHD</i>	<i>W7-X</i>
Major radius (m)	3.7	5.5
Minor radius (m)	0.63	0.55
Plasma volume (m ³)	29	30
Magnetic field (T)	2.85	2.5
Rotational transform		
Center	0.35	0.8
Edge	1.5	1.2
Heating power		
NBI (MW)	34	2
ECH (MW)	5	7
ICH (MW)	3	—
Achieved parameters		
Central electron temperature (keV)	20	8
Central ion temperature (keV)	10	3.5
Central electron density (10 ²⁰ m ⁻³)	12	1.4
Fusion triple product (s · keV · 10 ¹⁹ m ⁻³)	5.2	6.8
Beta (%)	5.1 (0.425 T)	0.55 (2.5 T)
Pulse length	48 min (1.2 MW)	30 s (5 MW)

specification and achieved plasma performance of LHD are compared with those of W7-X in [Table 2](#). LHD has three up-down pairs of poloidal coils in addition to a pair of helical coils. All these coils are superconducting (NbTi) with the capability for fully steady-state operation ([Motojima, 1994](#)). Adjustment of the currents of the poloidal coils enables changes in the radial position and shape of the plasma cross-section. Optimization of the magnetic configuration has been carried out with the constraints set by the continuous helical windings, which accommodate a unique pronounced feature in the form of a built-in divertor configuration ([Ohyabu et al., 1994](#)). Outside the closed flux surfaces, field lines are guided by a pair of helical coils through the ergodic layer to the built-in double null divertor. The vacuum vessel wall is stainless steel, as shown in [Fig. 6](#). LHD also accommodates 10 pairs of perturbation coils (with normal conductors) to compensate error fields and to generate resonant magnetic islands intentionally ([Ohyabu et al., 1995](#)).

Since its initial operation in 1998 ([Iiyoshi et al., 1999](#); [Motojima et al., 2000](#)), LHD has operated without serious problems. The number of cool-down and warm-up cycles, and the number of excitations of the superconducting coils had reached 21 and 1890 respectively by 2020. A variety of sophisticated diagnostics have been installed in conjunction with enhancement of the heating capability. The plasma volume is about 30 m³, which is much greater than preceding stellarators, and LHD was the first stellarator comparable in scale to large tokamaks. About 40 MW of total heating power is delivered by electron and ion cyclotron resonance heating and by two types of neutral beam injection, tangential beams with negative ion sources (180 keV injection energy) and perpendicular beams with positive ion sources (80 keV injection energy) ([Takeiri et al., 2006](#)). Divertor operation has been also developed. The open divertor configuration in the inboard region, where the particle flux is concentrated, has been rearranged into a closed divertor configuration ([Morisaki, 2013](#)) with cryogenic pumps and getter pumps. The replacement of the carbon divertor target plates with tungsten target plates is planned and a provisional partial test has started. Deuterium experiments started in 2017, which facilitated a significant increase in the power of neutral beam injection ([Osakabe et al., 2018](#)) and provided the opportunity to study the influence of ion isotope effects on plasma physics processes ([Yamada et al., 2019](#); [Ida et al., 2020](#)). In particular, a landmark discovery revealed that the so-called inward shifted magnetic configuration suppressed both neoclassical and turbulent transport while maintaining MHD stability ([Yamada et al., 2001a](#)). This finding is shared by physics under optimization by quasi-isodynamics. Further important experimental results in LHD included the production of a reactor-relevant value of central ion temperature of 10 keV ([Takahashi et al., 2018](#)), the achievement of central electron density of $1.2 \times 10^{21} \text{ m}^{-3}$ ([Yamada et al., 2007](#)), production of high plasma pressure, with a β value of 5% ([Sakakibara et al., 2008](#)), and establishment of 50 min operation of a plasma with a temperature of 2 keV using 1.2 MW of heating power (3.4 GJ of injected energy) ([Kasahara et al., 2014](#)).

Wendelstein 7-X (W7-X)

Wendelstein 7-X (W7-X), shown in [Fig. 2d–f](#) and located in the Max-Planck-Institut für Plasmaphysik (MPP, situated in Greifswald, Germany), is an optimized advanced stellarator ([Nührenberg, 1995](#)) based upon a successful proof-of-principle demonstration in the preceding device W7-AS ([Grieger et al., 1985](#)). Deviation of confined charge particles from the magnetic flux surfaces are minimized, as are the “Pfirsch-Schlüter” ([Galvão and Canal, 2021](#)) and bootstrap currents which deform magnetic flux surfaces. The concept of an advanced “quasi-isodynamics” stellarator has been driven by theory ([Nührenberg and Zille, 1998](#)). This is a unique

approach in which an optimized magnetic configuration is given first and then external coils are designed to realize the required magnetic configuration. Therefore, a combination of non-planar modular coils is a prerequisite to satisfy the design constraint.

W7-X is also a superconducting (NbTi) device with a steady-state capability (Sborchia et al., 2006). The magnetic configuration of W7-X is folded into five periods around the torus and each period has five pairs of different-type modular coils. Together with 20 planar coils which control the rotational transform, W7-X has 70 superconducting modular coils in total (Wanner et al., 2001). The magnetic field is fine-tuned by two kinds of control coils (with normal conductors) located outside (Rummel et al., 2014) and inside of the vacuum vessel (Füllenbach et al., 2020). These correction coils are used to correct error fields and tailor the magnetic field at the plasma boundary. The plasma volume is 30 m^3 , which is the same as in LHD. The modular coil system forms closed flux surfaces bounded by island divertors. A pair of top and bottom divertors, where heat and particle fluxes are concentrated, are arranged within each period. Construction was launched in 2002 and plasma experiments started in 2015 after overcoming significant manufacturing difficulties in the complex modular coils (Risse et al., 2016). The first operation phase (OP1.1) was dedicated to basic commissioning of the facility with the limiter configuration under limited heating power and pulse length (Klinger et al., 2017). A divertor without active cooling was then installed and the subsequent operation phase (OP1.2) was executed in 2017–18 (Bosch et al., 2020). Although the heating power and total injected energy were limited (6 MW and 200 MJ respectively), operational performance was extended considerably and basic features of the quasi-isodynamics optimization were confirmed (Dinklage et al., 2018). The *fusion triple product* reached $6.5 \times 10^{19} \text{ m}^{-3} \text{ keVs}$, which is the record value achieved in stellarators (Klinger et al., 2019). Stable high density operation above the cut-off density of the 2nd harmonic electron cyclotron resonance heating in X-mode has been demonstrated by use of the second harmonic O-mode (Laqua et al., 2019). A further important achievement is the demonstration of stable sustainment of detached plasmas for 30 s, which strengthens the strategy towards high-power steady state operation.

Installation of actively cooled divertor targets (Hathiramani et al., 2018) following OP1.2 will enable 30-min operation with heating power of 10 MW. Operation phase OP2, which will make use of these full device capabilities, is planned in 2021.

Medium sized stellarators

While the poloidal shaping of tokamak plasmas is generally converging to a certain range in plasma elongation and triangularity, stellarator plasmas still exhibit considerable diversity. The extension of the plasma parameters towards the reactor-relevant regime is being advanced by experiments in LHD (heliotron) and W7-X (advanced stellarator), while distinctive concepts are being explored in medium-sized stellarators experiments.

TJ-II (CIEMAT, Spain)

TJ-II is a flexible “*heliac*” which has been in operation at Spain’s National Fusion Laboratory in CIEMAT (Madrid, Spain) since 1997 (Alejandre et al., 1999). The device was designed through a collaboration of CIEMAT, Oak Ridge National Laboratory (ORNL, USA) and Max-Planck-Institut für Plasmaphysik (MPIPP, Germany) (Harris et al., 1985). It has two central coils which are interlinked with 32 toroidal coils (see Fig. 7). One of the central coils is a simple circular coil and the other is a helical coil, which winds helically around the circular central coil. Investigations of plasma behavior focus, in particular, on the characteristics of turbulence (Hidalgo et al., 2004; Estrada et al., 2016) by making full use of the flexibility of the magnetic configuration and the intensive diagnostics for measuring plasma flows and fluctuations (Melnikov et al., 2017).

HSX (Wisconsin University, USA)

HSX stands for Helically Symmetric eXperiment and the device started operation in 1999 (Anderson et al., 2002; Gerhardt et al., 2002). HSX has 48 twisted modular coils which generate the confining magnetic field (see Fig. 7), applying the concept of “*quasi-helical symmetry*” (Anderson et al., 1995). Toroidal effects are minimized and the magnetic field configuration is described by a uniform toroidal field and a modulated helical field with a single helicity. Therefore, HSX possesses common features with axisymmetric tokamaks in the direction of symmetry, and the comparisons made of plasma physics properties with those observed in tokamaks attract considerable physics interest. The magnetic configuration is characterized by a large aspect ratio of 10 and a toroidal field period of 4. The major goal of HSX is the demonstration of the advantage of helical symmetry, in particular, in relation to neoclassical transport and to turbulence driven by magnetic field curvature.

Heliotron J (Kyoto University, Japan)

Heliotron J at Kyoto University (Uji, Japan) belongs to an extended family of devices exhibiting “*quasi-poloidally symmetry*.” A magnetic configuration satisfying isodynamics with a magnetic well is formed by tailoring the helicity and bumpiness of the magnetic field. Unlike HSX and W7-X, which share common features of quasi-symmetry, Heliotron J employs a combination of a continuous $L = 1$ helical coil and toroidal field coils (see Fig. 7), rather than modular coils (Wakutani et al., 2000). The toroidal field period is 4 and there are four straight sections and four curved sections. Experiments started in 1999 (Obiki et al., 2001). The project aims at the demonstration of the improvement of confinement capability by isodynamics. Confinement improvement due to achievement of the H-mode (Sano et al., 2005) and generation of an internal transport barrier (Kenmochi et al., 2020) has been confirmed in the experiment.

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Magnetic Confinement Fusion—Experimental Physics: Reversed Field Pinches

M. Zuin, Consorzio RFX (CNR, ENEA, INFN, Università di Padova, Acciaierie Venete SpA), Padova, Italy

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Introduction

The physics of the Reversed Field Pinch, previously introduced in [Galvao and Canal \(2021\)](#), has achieved in the last years a significant progress from both the experimental and the theoretical point of view. Experiments have been performed in various devices of different sizes. Among these, the largest is RFX-mod in Padova (Italy), which obtained impressive improvements of the performance in high current operation with the spontaneous occurrence of helical equilibria characterized by magnetic order. In MST in Madison (USA), the active control of plasma current profile showed to be effective in inducing tokamak-like confinement properties.

The experiments have been accompanied and inspired by intense fully 3D nonlinear visco-resistive MHD simulations, mainly focused on the investigation of the dynamics of the RFP's magnetic self-organization processes. Current profile control, successfully tested experimentally, was inspired by MHD modeling. The bifurcation toward the single-helicity regime, at first discovered in high viscosity simulations, emerged in experiments. Subsequently, the experimental MHD dynamics were reproduced with extremely good agreement when a helical boundary was included in the simulations.

A positive trend of confinement with plasma current characterizes RFP performances, as presented in [Fig. 1](#), where the central electron temperature is plotted as a function of the plasma current for the RFX-mod database.

A significant upgrade of the RFX-mod device is presently underway with the aim of confirming such scaling.

The RFP can operate at large current density (no physical current limit exists), which implies that high particle density regimes can be explored (the RFP obeys an empirical density law similar to that of tokamaks). The RFP plasmas, being dominated by self-organization processes, are almost disruption free and the ideal beta limit is very large (around 40%).

Interesting properties of the energetic ion confinement have been highlighted in RFP in the last years. Fast ion confinement is almost classical and largely better than the thermal particle confinement, even in the presence of a stochastic magnetic field. Values of energetic ion beta in the core above that expected for alpha particle in a reactor have been achieved.

The magnetic self-organization aspects of the RFP, such as the magnetic dynamo and tearing mode activity, as well as the ion heating and acceleration observed to occur during magnetic reconnection phenomena, are interesting not only from a magnetic confinement point of view but also for their natural connections to astrophysics.

The chapter is organized as follows: in section [The RFP devices](#) the characteristics of the main RFP devices are introduced; section [The RFP equilibrium](#) presents the basic features of the RFP equilibrium; section [The RFP MHD dynamics](#) is dedicated to the dynamics of RFP plasmas, the magnetic reconnection processes, and on the magnetic topology changes induced by tearing modes activity with particular emphasis on the spontaneous bifurcation towards a helical equilibrium; section [Particle transport and confinement](#) introduces the basic physics of particle and impurity transport in the RFP; section [MHD control in the RFP](#) is dedicated to a description of the various approaches for RWM control and of its relevance in determining the confinement properties of the configuration and of the results of real-time tearing mode control; in section [Energetic ion confinement and stability](#) the interesting confinement properties of fast particles are summarized; section [Inductive current drive and profile control](#) is devoted to the performance improvement obtained thanks to inductive control of the current profile; section [Plasma-wall interactions issues](#) describes the main plasma-wall interaction issues; in section [Density and limit](#) the operative limits in terms of plasma density and beta are discussed along with the small-scale stability properties of RFP plasmas; in section [Neoclassical bootstrap current at high \$\beta\$](#) a brief estimation of the expected bootstrap current fraction is given.

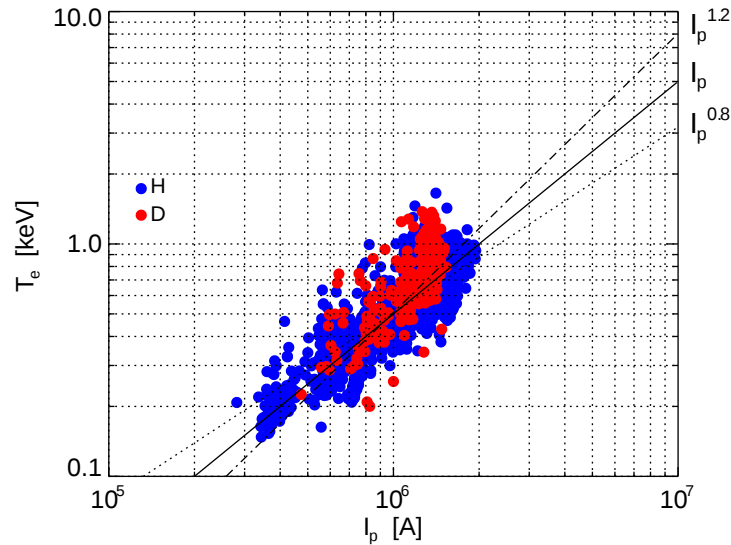


Fig. 1 Central electron temperature as a function of plasma current in RFX-mod. Red points refer to deuterium plasmas. Adapted from Piovan R, Bettini P, Bustreo C, et al. (2018) A continuously pulsed reversed field pinch core for an ohmically heated hybrid reactor. *Fusion Engineering and Design* 136: 1489–1493. <https://doi.org/10.1016/j.fusengdes.2018.05.040>

Table 1 Main RFP devices and their characteristics.

	RELAX	EXTRAP-T2R	TPE-RX	MST	KTX phase1(phase2) (projected)	RFX-mod
R/a (m)	0.5/0.25	1.24/0.18	1.72/0.45	1.5/0.5	1.4/0.4	2/0.459
Shell time τ_w (ms)	5	12	300	1100	40	100
Max I_p (MA)	0.12	0.3	0.45	0.6	0.5 (1)	2
Pulse length (ms)	3.5	100	80	100	–	500
Max T_e (keV)	0.3	0.3	1	2.0	0.3 (0.8)	1.5
n_e range 10^{19} m^{-3}	0.2–3	0.5–2	0.5–1	1–4	1–10	1–8

The RFP devices

Several RFP devices, listed in **Table 1**, have been in operation since the 1990s. RFX-mod (formerly RFX, [Gnesotto et al., 1995](#)), at Consorzio RFX in Padova, with $R = 2 \text{ m}$ and $a = 0.459 \text{ m}$, being R and a the major and minor radius respectively, reached the target of a maximum plasma current up to 2 MA. It is equipped with a thin copper shell with $\tau_w = 100 \text{ ms}$ ($\tau_w = \mu_0 a \delta \sigma$, where δ is the wall thickness and σ is the wall conductivity) and a system of actively controlled saddle coils for thin shell operations ([Sonato et al., 2003](#)). An upgrade of the RFX-mod machine (dubbed RFX-mod2) is presently being implemented ([Marrelli et al., 2019b](#); [Peruzzo et al., 2017, 2019](#); [Zuin et al., 2017](#)) with relevant magnetic front-end modifications. Its main aims are the reduction of the amplitude of tearing modes and of the associated magnetic chaos and the mitigation of the plasma wall interaction, locally enhanced by wall-locking.

The MST device, operating at the University of Wisconsin Madison, United States ([Dexter et al., 1991](#)), ($R = 1.5 \text{ m}$, $a = 0.50 \text{ m}$) can reach up to 0.6 MA plasma current. A 5-cm-thick aluminum shell serves as both the vacuum chamber and the equilibrium field magnet. Shell minor radius is 0.52 m with graphite plasma limiters. A 1 MW, 25 keV, 20 ms neutral beam injector is installed and used to study fast ion confinement and stability.

EXTRAP-T2R ([Brunsell et al., 2001](#)), at the Royal Institute of Technology, Stockholm, Sweden, ($R = 1.24 \text{ m}$, $a = 0.18 \text{ m}$, maximum plasma current = 0.3 MA) has a copper shell with shell penetration time of 12 ms. Its experimental activity is mainly focused on the development of advanced feedback tools for the control of MHD instabilities.

The TPE-RX device ([Yagi et al., 1999a,b](#)) RFP ($R = 1.72/a = 0.45 \text{ m}$), with a multiple layer shell made of a close fitting thin one (1 mm copper) ($b/a = 1.08$) and a thick one (5 cm aluminum) was operated until 2007 ([Koguchi et al., 2009](#)).

RELAX is a low aspect ratio RFP ([Masamune et al., 2007](#)) ($R = 0.51 \text{ m}$, $a = 0.25 \text{ m}$, maximum plasma current = 0.12 MA) operated at the Kyoto Institute of Technology, Japan and is mainly devoted to the study of Single Helicity plasmas, MHD stability and the generation of bootstrap current at a low aspect ratio ([Masamune et al., 2009](#)).

The KTX device ([Liu et al., 2014, 2017](#); [Zheng et al., 2012](#)) in operation at the University of Science and Technology in Hefei, China, ($R = 1.4 \text{ m}$, $a = 0.4 \text{ m}$) has a magnetic boundary similar to that of the RFX-mod and is planned to produce a plasma current up to 1 MA, with an active MHD control system. The KTX device aims to investigate the fluctuation induced transport in high temperature RFP plasmas by means of newly developed diagnostics tools ([Liu et al., 2019](#)).

The RFP equilibrium

The RFP equilibrium is a paramagnetic configuration: the plasma current profile arranges in a way to ensure equilibrium and generate most of the toroidal flux. Conversion of poloidal into toroidal flux occurs through a dynamo process proportional to the plasma current. The main parameters to describe a RFP equilibrium are the *pinch parameter* Θ :

$$\Theta = \frac{B_\theta(a)}{\langle B_\phi \rangle}$$

and the *reversal parameter* F :

$$F = \frac{B_\phi(a)}{\langle B_\phi \rangle}$$

negative for a RFP plasma, $\langle B_\phi \rangle$ being the average of the toroidal magnetic field over a poloidal cross section, a the plasma minor radius.

F and Θ are experimentally found to follow some trajectory in the (F, Θ) plane, i.e. are not independent quantities. The following family of functions

$$\mu(r) = \mu_0 \left(1 - \frac{r}{a}\right)^\alpha$$

gives a good cylindrical approximation of the axisymmetric part of the parallel current profile measured in experiments (see e.g. in Fig. 1b of Bodin, 1990).

The plasma current has its maximum at the magnetic axis and decreases to zero at the edge (Antoni et al., 1986; Ji et al., 1995).

The toroidal and poloidal components of the magnetic field are of comparable amplitude in a RFP, as shown in Fig. 2, where the radial profiles of the normalized magnetic fields are compared to those of a tokamak configuration. In contrast to the tokamak, magnetic field lines in a RFP are almost poloidal and the toroidal current is pushed to values above the Kruskal-Shafranov threshold.

The cylindrical safety factor $q(r) = rB_\phi/RB_\theta$ turns out to be a decreasing function of the radius, with $q(0) \approx a/2R$ and $q(a) \leq 0$, as shown in Fig. 3. The toroidal field at the plasma edge is in the opposite direction with respect to the core, hence the name of the configuration. With such a $q(r)$ profile, many resonant surfaces exist for $m = 1$ tearing modes with $n \geq 2R/a$, where m and n are the poloidal and toroidal mode numbers of the usual Fourier decomposition, along with those for $m = 0$ modes at the $q = 0$ surface.

In this equilibrium, helical MHD instabilities (resistive kinks) are triggered and nonlinearly saturate at finite amplitudes with the help of the strong radial decrease of the toroidal field at the edge. These modes, also named dynamo modes, produce helical perturbations of the plasma current, which, in a naïve analogy with a current-carrying wire in a cylindrical flux conserver (Escande et al., 2000a), induces a solenoidal effect, which increases the magnetic field and flux inside the kinked wire. In the presence of a flux conserver a decrease of the magnetic field and flux outside is also imposed, which is at the origin of the magnetic field reversal.

The dynamo process underlying the RFP configuration is thus a mere consequence of magnetic relaxation, as revealed by nonlinear visco-resistive MHD simulations (Bonfiglio et al., 2005). The helical distortions requires a modulation of the current density along current lines and hence the existence a modulated electric field (electrostatic in the case of a stationary saturated perturbation), which drives an $\mathbf{E} \times \mathbf{B}$ motion whose non axis-symmetric part is the dynamo velocity field.

It is worth noting that the helical deformation due to the saturated kink makes RFP plasmas similar to stellarator ones, but also that, while in a stellarator the magnetic field is almost completely defined by externally driven currents, in a RFP the magnetic field is

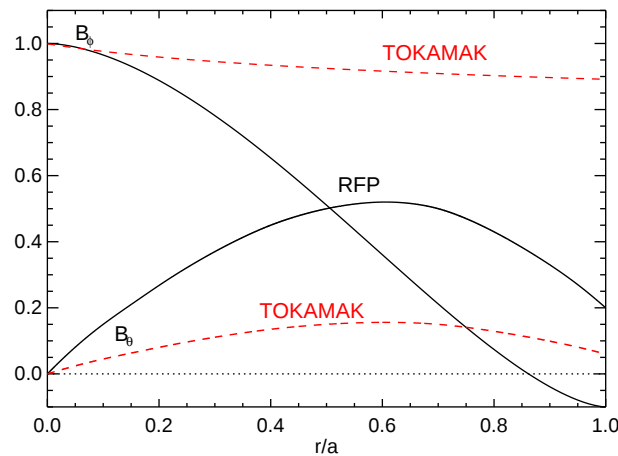


Fig. 2 Radial profiles of the Toroidal (B_ϕ) and poloidal (B_θ) magnetic fields (normalized to the on-axis toroidal field value) for a Reversed Field Pinch and a Tokamak. Taken from Ortolani S, and Schnack DD (1993) *Magnetohydrodynamics of Plasma Relaxation*. Singapore: World Scientific.

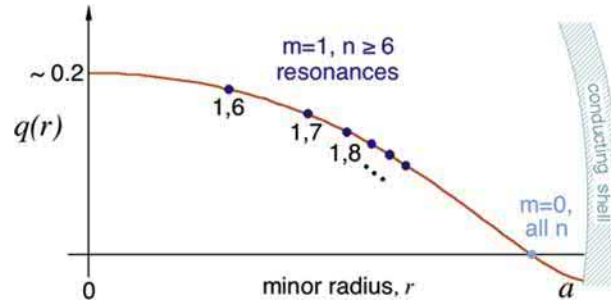


Fig. 3 Axisymmetric safety factor profile $q(r) = rB_\phi / RB_\theta$ for a typical RFP with an aspect ratio $A = 3$. The location of the $m = 1$ and the $m = 0$ resonant surfaces are indicated.

mostly produced by currents flowing inside the plasma. As a consequence, the RFP, corresponding to a full MHD relaxation of the magnetic field, can be considered as resilient to disruptions.

The RFP MHD dynamics

In the setting-up phase of an RFP plasma discharge, an initially flat q -profile evolves, exhibiting a series of local minima at the edge, which provide a source for kink instability (Robinson, 1971). As the plasma current is raised, various resonances enter the plasma region and the associated current-driven $m = 1$ modes with increasing toroidal mode number n occur, producing the poloidal current which amplifies the toroidal flux (Caramana et al., 1983). Subsequently, the toroidal flux in the flat-top phase is observed to be generated either continuously or during discrete events, associated with global rearrangements of the magnetic configuration through magnetic reconnection phenomena. Visco-resistive MHD simulations have offered an insight into the dynamics of such events (Ho and Craddock, 1991; Cappello and Biskamp, 1996a,b; King et al., 2012; Sauppe and Sovinec, 2016).

The magnetic equilibrium and the magnetic boundary determine the frequency and the amplitude of such events (Hattori et al., 1991; Watt, 1983). Large bursts of modes with $m = 1$ and $m = 0$ and low toroidal mode numbers (Verhage et al., 1978; Almagri et al., 1992; Hedin, 1998; Hokin et al., 1991; Lorenzini et al., 2008a,b; Momo et al., 2020) are signs of reconnection processes, associated with global relaxation of the current profile (Brower et al., 2002). This phenomenon is toroidally localized, being induced by nonlinear mode phase alignment (Howell et al., 1987). The generation of localized current sheets parallel to the edge magnetic field has been directly measured in RFX-mod (Zuin et al., 2009) and in MST (Crocker et al., 2003; Piovesan et al., 2008). An example of the dynamics of a reconnection process is shown in Fig. 4.

An important feature of this figure is in panel (e), where the ion heating phenomenon occurring as a consequence of the reconnection process is clearly visible. A large fraction of the magnetic energy released during reconnection events is indeed converted into ion thermal energy, (Scime et al., 1992; Hörling et al., 1996; Gangadhara et al., 2007, 2008). Ion heating in MST plasmas reveals that it is charge and mass dependent (Fiksel et al., 2009; Kumar et al., 2013) and anisotropic in space with $T_{i\perp} > T_{i\parallel}$ (Magee et al., 2011). Moreover, an energetic ion tail forms spontaneously with a power-law distribution. This has the effect of significantly enhancing the neutron rate from fusion reactions in deuterium plasmas in MST (Magee et al., 2011), as also confirmed in RFX-mod plasmas (Zuin et al., 2015).

In the relaxation phase an outward propagating heat pulse is detected, indication of the presence of stochasticity due to the multiple $m = 1$ modes (Frassinetti et al., 2005b).

During discrete reconnection events, rapid and spontaneous changes of the radial profile of the toroidal plasma rotation occur, sign of enhanced radial transport of toroidal momentum, about two orders of magnitude more rapid than what would be expected from classical viscosity. This effect has been interpreted as due to nonlinear interactions of multiple tearing modes producing internal torques that redistribute momentum along the plasma radius (Hansen et al., 2000).

This nonlinear coupling of tearing modes is also responsible for the toroidally localized deformation of the plasma column, named Slinky or Locked Mode in RFP literature (Bolzonella and Terranova, 2002; Nordlund and Mazur, 1994; Sarff et al., 1993). The phenomenon of phase-locking has been observed in many RFP experiments: OHTE (Tamano et al., 1987), TPE-1RM15 (Hattori et al., 1991), Reversatron II (Greene and Robertson, 1993a), MST (Almagri et al., 1992; Hokin et al., 1991), TPE-1RM20 (Brunsell et al., 1993), TPE-RX (Yagi et al., 1999), EXTRAP-T2R (Malmberg et al., 2000), RFX (Buffa et al., 1994; Zanca et al., 2001) and RFX-mod (Martini et al., 2007; Ortolani and team, 2006). The Locked Mode can rotate in the laboratory frame or *wall-lock*, depending on the properties of magnetic boundary, on modes amplitude level and on the plasma current level.

Wall locking is responsible for enhanced localized plasma wall interaction, which can induce overheating of the first wall (Valisa et al., 1997; Zanca et al., 2007a,b), anomalously large impurity influxes and radiation (Marrelli et al., 1999). Also global quantities, such as the loop voltage and fuel recycling are affected by wall locking. Techniques to mitigate these adverse wall locking effects have been successfully exploited in RFP plasmas, as will be described later on.

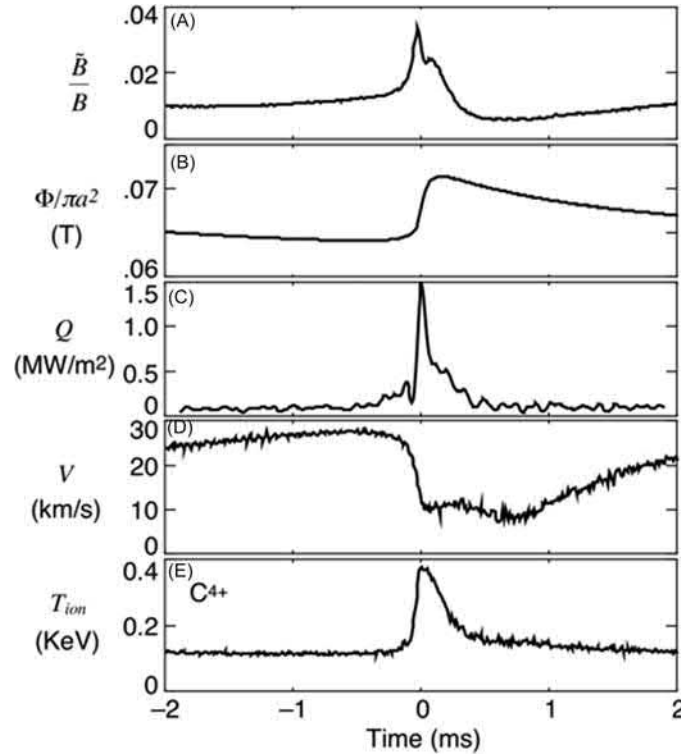


Fig. 4 Typical time traces of a global reconnection process, obtained by ensemble average over a large number of similar events recorded on MST plasmas. (a) relative amplitude of magnetic fluctuations; (b) toroidal magnetic flux density; (c) heat flux; (d) plasma flow and (e) ion temperature.

Depending on the experimental conditions, the core resonant dynamo modes spectra show up in the RFP in different regimes. The first, dubbed Multiple Helicity (MH) state, is characterized by the presence of many $m = 1$ Fourier components of the magnetic field at the edge of the plasma with comparable amplitude. In the second, dubbed Quasi Single Helicity (QSH) state, one single $m = 1$ Fourier component grows and becomes dominant over all the other modes, named “secondary” modes, whose amplitude is reduced with respect to that in MH.

In the MH state the overlap of the magnetic islands causes a broad region of stochastic magnetic field, where the temperature profile is observed to be flat and energy diffusion is dominated by transport in a stochastic field (Rechester and Rosenbluth, 1978).

Transient magnetic QSH states have been experimentally detected to last for several energy confinement times from 1993 in all large RFP devices (Brunsell et al., 1993; Nordlund and Mazur, 1994; Hirano et al., 1997; Sarff et al., 1997; Martini et al., 1999a; Martin, 1999). The emergence within the magnetic spectrum of a dominant mode, typically the most internally resonant one, is associated with the observed formation of a “bean”-like hot structure corresponding to the magnetic island with the same helicity of the dominant mode (Martin et al., 2000, 2002; Escande et al., 2000a; Piovesan et al., 2004a).

These states were named Double Axis (DAX) states (Puiatti et al., 2009a) as two magnetic axes are present within the plasma: the unperturbed axis-symmetric one and the axis associated with the island O-point (Lorenzini et al., 2008b). QSH states were observed to become purer and longer lasting as the plasma current was raised (in RFX-mod typically above 0.6 MA) and found to be favored by a magnetic equilibrium characterized by a slight reversal of the toroidal magnetic field at the edge (Ortolani and team, 2006; Paccagnella et al., 2006). Such a tendency was confirmed by the analysis in various RFP devices: in MST (Marrelli et al., 2002; Piovesan et al., 2004a), in TPE-RX (Piovesan et al., 2004b), in EXTRAP T2R (Frassinetti et al., 2007), and in RELAX (Ikezoe et al., 2012).

Transitions from MH to QSH can be either spontaneous or induced by external dynamic actions such as the oscillation of the edge toroidal magnetic field (OPCD technique), as demonstrated in RFX-mod (Terranova et al., 2007), or by applying a small positive pulse in the initially weakly reversed edge magnetic field (Hirano et al., 2006). The results from the RELAX device, where QSH phases lasting above 30% of the flat-top duration (Oki et al., 2012, 2013), suggest that low aspect ratio is a favorable condition for the establishment of QSH states.

The plasma current regimes above 1.2 MA explored in RFX-mod finally exhibited the predicted bifurcation towards a kink-like magnetic topology (Escande et al., 2000b), as the amplitude of the dominant mode increases and its island X-point collides with the original unperturbed axis of the configuration and separatrix expulsion occurs. This process is shown in Fig. 5a–c for a pure Single Helicity case with $m = 1$ and $n = -7$ periodicity (please note that negative toroidal mode numbers are conventionally used for modes resonant inside the reversal surface in RFX-mod) and with a fixed level of secondary modes Fig. 5d–f.

As a result of this process, a single axis remains in the plasma corresponding to the former island O-point. This new magnetic topology has been dubbed SHAX (Single Helical Axis) state, associated with a helical safety factor profile with a maximum located in the vicinity of the former separatrix.

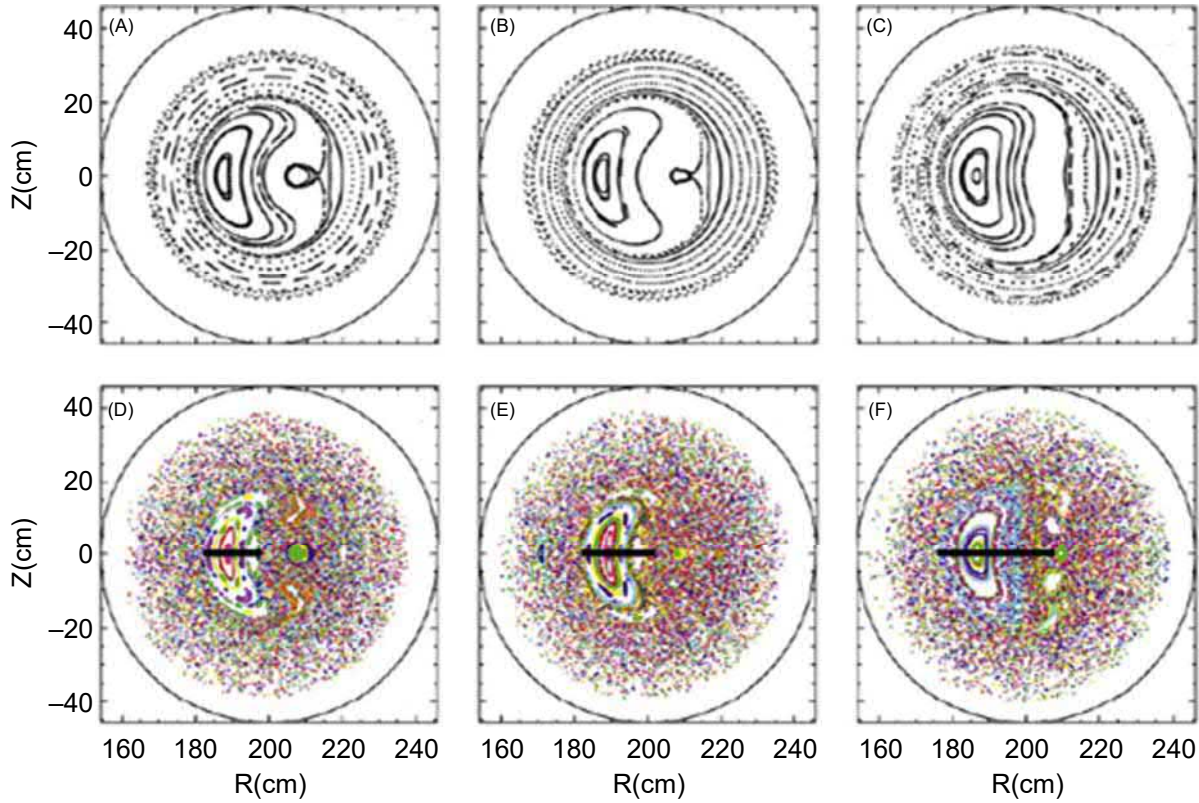


Fig. 5 (a)–(c) Poloidal Poincaré plots of the magnetic field at increasing amplitude of the $n = -7$ mode (3%, 4% and 5%, respectively) in the SH case and (d)–(f) with a fixed level of secondary modes. The black thick lines indicate the radial extent of the good-confinement region, based on test particle simulations. In (f) sticky magnetic field lines around the helical flux surfaces are visible (note in particular the blue and pink dots). Adapted from Piovesan P, Zuin M, Alfier A, et al. (2009) Magnetic order and confinement improvement in high-current regimes of RFX-mod with MHD feedback control. *Nuclear Fusion* 49: 085036. <https://doi.org/10.1088/0029-5515/49/8/085036>

The typical time traces of a RFX-mod discharge exhibiting a long lasting transition to SHAx state is presented in Fig. 6.

The experimental behavior of these SHAx states has been particularly well reproduced by 3D MHD modeling from the Specyl code, once a small $m = 1$, $n = -7$ boundary condition with an amplitude similar to the experimental one (about 1.5% of the central magnetic field) was included in the simulations at Lundquist numbers $S = 10^7$, consistent with the experimental one (Bonfiglio et al., 2013) (see Fig. 7).

The magnetic topology change in the QSH-SHAx state is associated with a modification of the kinetic profiles. In Fig. 8 the electron temperature profiles measured by the Thomson scattering diagnostic in MH, QSH-DAx and QSH-SHAx at three time instants in the same RFX-mod discharge are shown (Marrelli et al., 2021).

The small localized hot island, typical of the DAx state is replaced by a large plateau at an electron temperature typically above 1 keV, encompassed by large gradients establishing an electron internal transport barrier (eITB), normally radially localized in the vicinity of the maximum of the safety factor profile (Puiatti et al., 2011, 2009a; Gobbin et al., 2011a,b). As a consequence, the thermal content of the plasma increases, a sign of a self-organized transition to an improved confinement regime. An almost two-fold improvement of the energy confinement time is estimated (Piovesan, 2009), as shown in Fig. 9, where this is shown as a function of the residual secondary modes amplitude, which decreases as the amplitude of the dominant mode increases.

This result has been interpreted, by means of magnetic reconstruction, as due to magnetic chaos healing. SHAx states being typically more resilient to chaos than DAx states (Escande et al., 2000b; Cappello et al., 2012; Veranda et al., 2013; Terranova et al., 2010; Gobbin et al., 2011a; Gobbin et al., 2011b).

A lower threshold of about 3–4% on the normalized amplitude of the dominant mode with respect to the edge magnetic field for the occurrence of the transition QSH-DAx/QSH-SHAx has been identified, in agreement with theoretical predictions (Escande et al., 2000c). The amplitude of secondary modes is also a relevant parameter for the appearance of wide helical structures. In particular, the lowest amplitudes of the two main secondary modes: i.e. the $m = 1$, $n = -8$ and $m = 1$, $n = -9$ in RFX-mod are associated with the widest thermal structures.

As for QSH, SHAx states are routinely obtained both spontaneously, as mentioned at plasma current above 1.2 MA (Lorenzini et al., 2009a; Bonomo et al., 2009), and triggered by a dynamic poloidal current drive, by means of the OPCD technique (Lorenzini et al., 2008b, 2009b).

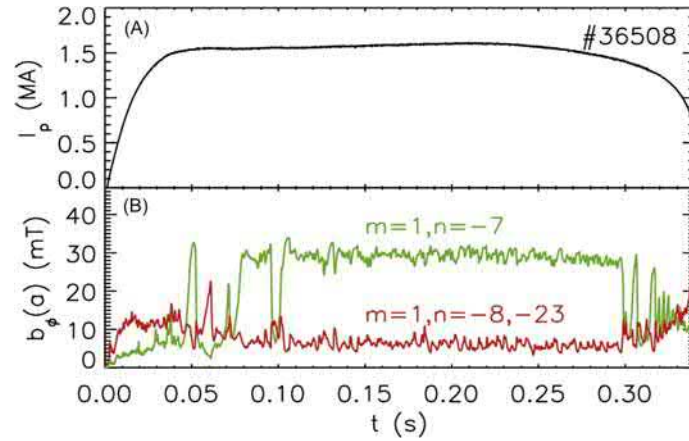


Fig. 6 Time traces of an RFX-mod discharge, exhibiting spontaneous long-lasting QSH: (top) plasma current; (bottom) amplitude of dominant $m = 1, n = -7$ tearing mode (red) and averaged amplitude of secondary modes. From Lorenzini R, Agostini M, Auriemma F, Carraro L, et al. (2015) The isotope effect in the RFX-mod experiment. *Nuclear Fusion* 55: 043012. <https://doi.org/10.1088/0029-5515/55/4/043012>.

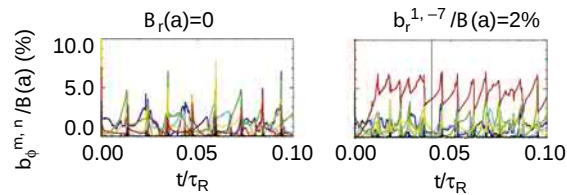


Fig. 7 Simulated time evolution of the $m = 1$ tearing modes by the Specyl code; (left) with ideal boundary conditions; (right) with 2% perturbation. Adapted from Bonfiglio D, Veranda M, Cappello S, Escande DF, and Chacón L (2013) Experimental-like helical self-organization in reversed-field pinch modeling. *Physical Review Letters* 111. <https://doi.org/10.1103/PhysRevLett.111.085002>

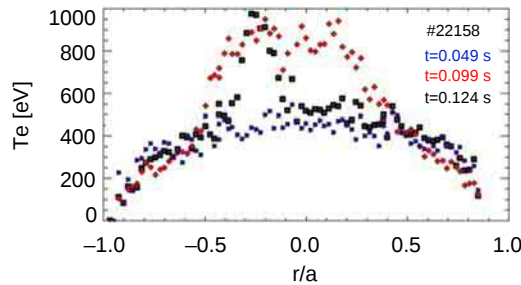


Fig. 8 Electron temperature profiles as a function of normalized minor radius, r/a , in RFX-mod Quasi Single Helicity (black squares QSH-DAX, red diamonds QSH-SHAX states) and MH (blue squares) plasmas. From Marrelli L, Martin P, Puiatti ME, Sarff JS, Chapman BE, Drake JR, Escande DF, and Masamune S (2021) The reversed field pinch. *Nuclear Fusion* 61: 023001.

In the MST plasmas QSH-SHAX states appear at lower plasma current levels (around 0.5 MA) than in RFX-mod but at similar Lundquist number S , with the dominant $m = 1, n = 5$ mode, which can reach amplitudes of up to 8% of the axisymmetric field (Auriemma et al., 2011; Sarff et al., 2013; Bergerson et al., 2011).

Back-transitions from QSH-SHAX to MH states occur as a consequence of the magnetic reconnection processes associated with global rearrangements previously described.

In various RFPs, the application of a stationary or slowly rotating magnetic perturbation by means of external radial field coils proved to be effective in increasing the duration of QSH phases, both with a feedforward approach, as in MST (Evans et al., 2004; Munaretto et al., 2016, 2015), or with a feedback controlled one, as in EXTRAP-T2R and in RFX-mod (Marrelli et al., 2007). In the latter experiment, the application of a $m = 1, n = -7$ field above a given threshold in plasmas with shallow toroidal field reversal strongly improves the helical state persistence (Piovesan et al., 2011) even at higher densities ($n/n_G \approx 0.5$) than in the spontaneous case (Piovesan et al., 2013), without strong effects on the loop voltage (Gobbin et al., 2015).

The role of microturbulence as a driving mechanism of transport in the RFP has been assessed by means of gyrokinetic numerical simulations and analytical analyses. It has been proven that, due to the short field connection length in the low- q cylindrical equilibrium and the related Landau damping, the Ion Temperature Gradient (ITG) mode threshold is larger by a factor R/a than that in

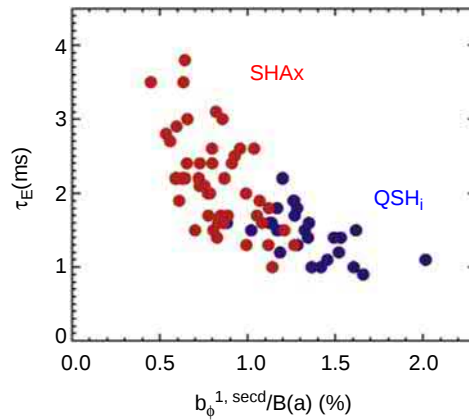


Fig. 9 Energy confinement time as a function of the secondary mode amplitude in QSH_i (blue) and SHAx states (red). Adapted from Piovesan P, Zuin M, Alfier A, et al. (2009) Magnetic order and confinement improvement in high-current regimes of RFX-mod with MHD feedback control. *Nuclear Fusion* 49: 085036. <https://doi.org/10.1088/0029-5515/49/8/085036>

the Tokamak (Guo, 2008; Predebon et al., 2010a; Sattin et al., 2010). However, this threshold decreases in helical states (Predebon and Xanthopoulos, 2015) so that ITG turbulence cannot be excluded to enhance total heat transport in this condition.

In the presence of the internal transport barriers forming during the QSH-SHAx states in RFX-mod, dedicated gyrokinetic simulations showed that microtearing (MT) modes become linearly unstable. These modes have actually been experimentally observed in RFX-mod helical states, by means of highly resolved in-vessel edge probes, with spectral properties in agreement with the theoretical expectations, as shown in Fig. 10, and with amplitude well correlated with the electron temperature gradient (Zuin et al., 2013).

The occurrence of collisionless microtearing instabilities has been also studied in RFX-mod (Predebon and Sattin, 2013) and MST (Carmody et al., 2015), indicating that the high magnetic drifts in the RFP could favor such microinstabilities.

A positive isotopic effect on the confinement of SHAx states has been found by comparing hydrogen and deuterium plasmas in RFX-mod. In high plasma current regimes, characterized by the occurrence of SHAx states, the energy and particle confinement times scale as $m_i^{0.3}$ and as $m_i^{0.45}$, respectively, with a reduction of particle influx. The confinement enhancement is mainly attributed to the mitigation of transport at the plasma edge (Lorenzini et al., 2015). A 0.2 keV improvement of the electron temperature in a large part of the plasma volume has been observed in deuterium plasmas. Calculation of particle orbits confirms that the loss time in QSH regimes is about 30% higher in deuterium than in hydrogen for both spontaneous and induced helical states (Gobbin et al., 2015).

As predicted by MHD numerical simulations (Veranda et al., 2013), non internally resonant QSH $n = 6$ states (Cappello et al., 2012) can also be excited in RFX-mod, with the aim of improving the region with well conserved magnetic surfaces and reducing the magnetic field lines transport (Veranda et al., 2017, 2020). Preliminary encouraging experimental results have been obtained in RFX-mod.

Particle transport and confinement

In RFX particle diffusion coefficients of the order of $20 \text{ m}^2/\text{s}$ in the core and $5 \text{ m}^2/\text{s}$ in the edge plasma had been estimated on the basis of the observation of hollow density profiles, which imply an outward particle pinch. In RFX-mod particle diffusivity scaling with the magnetic field fluctuation in MH and QSH states (Auriemma et al., 2015) has been found to be weaker than that predicted by the Rechester-Rosenbluth theory, which suggests that, despite the magnetic field is chaotic, hidden confining structures are anyway present within the plasma (Spizzo et al., 2009). A significant improvement of particle confinement in QSH with respect to MH has been experimentally observed in TPE-RX (Frassinetti et al., 2006) and in RFX-mod, where, thanks to pellet injection in the hot region of a SHAx state, an enhancement of up a factor 3 has been estimated for the particle confinement time (Terranova et al., 2010). These results are in agreement with dedicated simulations (Predebon et al., 2004).

Single particle trajectory studies indicate that in RFX-mod plasmas neoclassical diffusion coefficient is one order of magnitude larger than the classical one (Gobbin et al., 2007, 2009a,b). Moreover, the population of superbanana particles is rather low in the presence of observed helical perturbations, as the distortion decreases when moving towards the plasma edge (Gobbin et al., 2010). As a result, differently from what occurs in unoptimized stellarators, transport is proportional to the collisional frequency.

Impurity transport studies have been performed computationally (Gobbin et al., 2007) and experimentally in both MST and RFX-mod (Barbui et al., 2015, 2014; Carraro et al., 2000; Gobbin et al., 2017) and an outwardly directed pinch velocity has been estimated. Central accumulation does not occur in QSH states in RFX-mod (Carraro et al., 2009), with larger impurity diffusivity in the core than at the edge (Menmuir et al., 2010).

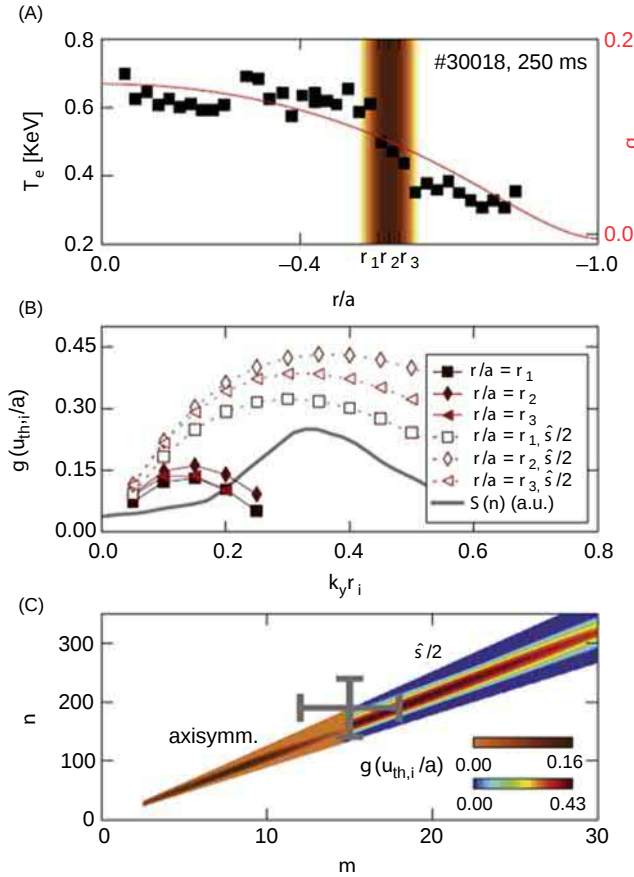


Fig. 10 Comparison of experimental measurements and theoretical expectations for microtearing modes in RFX-mod QSH-SHax states. For the shaded area of frame (a) the growth rates of the most unstable modes (all with microtearing parity) as a function of the wave number $k_y r_i$ are in (b), solid symbols; open symbols represent the same data set with q profile artificially flattened; the gray line is the experimental (toroidal mode number) magnetic spectrum; in (c) the gyrokinetic estimation of the microtearing spectra are exploded on the (m, n) plane along with the experimental estimate of the measured wave numbers at the edge (gray error bars). From Zuin M, Spagnolo S, Predebon I, Sattin F, et al. (2013) Experimental observation of microtearing modes in a toroidal fusion plasma. *Physical Review Letters* 110. <https://doi.org/10.1103/PhysRevLett.110.055002>

MHD control in the RFP

RWM control

In the past, a serious issue for the performance and the future of the RFP configuration was represented by resistive wall mode (RWM) instabilities, which strongly limited the pulse length. Early RFP experiments in the 1980s were equipped with resistive walls, characterized by wall time constants shorter than the discharge pulse length in order to study RWM dynamics. These devices include OHTE at General Atomics (Goforth et al., 1986; La Haye et al., 1988), the Reversatron at the University of Colorado (Greene and Robertson, 1993b), the HBTX-1C experiment at Culham (Alper et al., 1989) and REPUTE-I (Asakura et al., 1986a,b).

In HBTX-1C, first clear observations were obtained of non-resonant ideal kinks (RWMs) with growth rates about a factor of three times the wall time constant, in agreement with theoretical studies. Such results were later confirmed clearly by the EXTRAP T2R experiment (Brunsell et al., 2003); the observed growth time scale of the RWMs was comparable to the wall penetration time.

The first attempt at an active feedback experiment of the $(m = 1, n = 2)$ mode was reported in HBTX-1C (Alper, 1990) by means of two helically wound coils located outside the outer shell of the vessel. The active feedback of a single RWM was successful but the experiment also proved that multiple mode control would have been necessary in order to improve RFP plasma performances.

The requirements for active RWM control were determined by theoretical studies taking into account the role played by shell resistivity and distance of the boundary (Ho and Prager, 1988) and later on also that of rotational stabilization (Jiang et al., 1995; Guo et al., 1999) and nonlinear effects (Schnack and Ortolani, 1990) with the DEBS code, where also tearing modes were considered.

The action of a virtual system of helical active coils for multiple RWM control on Fourier harmonics was introduced with a feedback scheme named Mode Control, with different gains for each single mode (Paccagnella et al., 2002b); a mostly linear behavior of RWMs was also described, as confirmed by the experimental findings in EXTRAP-T2R (Brunsell et al., 2003), and RFX-mod (Bolzonella et al., 2008).

As an alternative to helical coils, a concept for RWM feedback stabilization, named the *intelligent shell* (IS), based on a set of saddle coils surrounding the plasma, was proposed by Bishop (Bishop, 1989). A set of corresponding sensors is used to feedback control the current in order to mimic a perfectly conducting shell. The limits of the IS scheme in the simultaneous control of multiple modes are due to finite number of actuators and subsequent generation of toroidal harmonics n together with toroidal sidebands (Fitzpatrick and Yu, 1999). These limits were confirmed experimentally in initial active feedback operations on EXTRAP-T2R (equipped with a matrix of 4×32 saddle coils and corresponding saddle loop sensors) (Brunsell et al., 2004).

By extending control theory techniques developed for tokamaks (Bondeson et al., 2002), a more realistic geometrical description of the coils and of the sensors was studied in (Paccagnella et al., 2002a) with a different control scheme.

In such a scheme (dubbed *Mode Control*), the feedback variables $b^{m,n}$ are the Fourier coefficients of the flux loop sensor signals b^j , while the coefficients of the Fourier decomposition of the currents into the control coils are the current harmonics $I^{m,n}$. An inverse Fourier transform of the spectrum of these $I^{m,n}$ current harmonics is applied to determine the current in each active coil (Drake et al., 2005), with a different gain for each single mode (Bolzonella et al., 2008; Martini et al., 2007).

In order to deal with the effect of sidebands aliasing, a more advanced feedback control scheme, dubbed *Clean Mode Control* (Zanca et al., 2007a,b), was developed.

Complete stabilization of the unstable RWM spectrum was achieved in EXTRAP-T2R (Brunsell et al., 2005; Drake et al., 2008) with an increase of pulse length, limited by the power supplies, of up to 90 ms (of the order of $10 \tau_w$) as shown in Fig. 11.

The same control algorithms developed for the EXTRAP-T2R device were exploited in the RFX-mod experiment, equipped with a thin copper shell (with $\tau_w = 100$ ms) and the most advanced system of coils (made of a matrix of 4×48 arrays of equally spaced coils) fully covering the plasma.

The discharge duration was prolonged by the feedback action from 150 ms to more than 300 ms, with a relevant loop voltage reduction (Paccagnella et al., 2006).

The flexibility of the RFX-mod device allows also operation as a circular (and shaped) ohmic tokamak with a toroidal field of 0.5 T. Due to the specific RFX-mod passive boundary, when $q(a) < 2$ the $m = 2, n = 1$ external ideal current-driven kink converts into a Resistive Wall Mode. The Clean Mode Control feedback algorithm was efficient in keeping the $m = 2, n = 1$ at a negligible level, despite the fact that the available number of actuators along the poloidal direction, i.e. 4, does not satisfy the stabilization criteria for such mode (Zanca et al., 2012a,b).

After the demonstration of successful multiple RWMs control, advanced issues on the topic have been dealt with in RFP devices and theoretically. Among these, it is worth mentioning the *Resonant Field Amplification*, of particular relevance as the effects of error fields, which are seed of RWM dynamics, can be significantly different during plasma operation and in vacuum conditions (Volpe et al., 2013). In particular, the phase and the amplitude of error fields from gaps in the shell, leakage flux from the iron core or lack of symmetry in toroidal coil positioning was addressed by means of actively induced both static and rotating harmonics. Resonant field amplification effects were clearly identified in good agreement with the theoretical expectations in RFX-mod.

A limited external saddle coil coverage of the plasma boundary was simulated experimentally in RFX-mod and EXTRAP T2R, exploiting the sophisticated control algorithm software and numerically with a flight simulator with interesting results (Baruzzo et al., 2012; Marchiori et al., 2012; Olofsson et al., 2012).

An analysis of the influence on the performance of the 3D passive conducting structures surrounding the plasma using the finite element electromagnetic code Cariddi and a 2D model for RFP plasma stability (MARS-F) (Liu et al., 2000; Portone et al., 2008) has been undertaken at RFX-mod (Villone et al., 2008). A full dynamic flight simulator, including the features of the CarMa (Liu et al., 2000) code and a model of the RFX-mod control system (Marchiori et al., 2012), was developed to simulate experimental conditions.

Tearing modes control

One of the main issues for high performance RFPs is the mitigation of wall and phase locking of tearing modes (TM). Wall locking of $m = 1$ modes in the RFP has been modeled in (Fitzpatrick et al., 1999; Chapman et al., 2004), in order to explain the differences

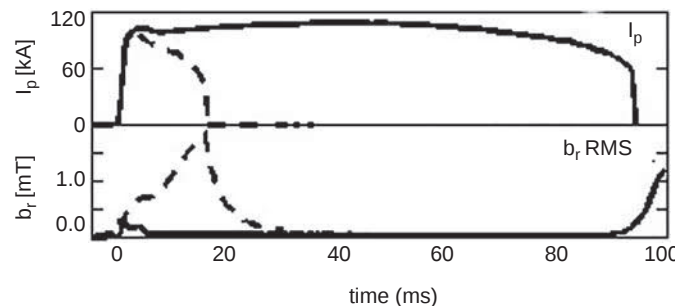


Fig. 11 Time traces of EXTRAP T2R plasma pulses without (dashed line) and with active control (solid line): (a) discharge current; (b) total radial field amplitude at the sensor coil array. From Drake JR, Bolzonella T, Olofsson KEJ, Baruzzo M, et al. (2008) Reversed-field pinch contributions to resistive wall mode physics and control, in: *Proceedings of 22nd International Conference on Fusion Energy*. IAEA, p. EX P9-7.

exhibited by MST and RFX in terms of the current threshold for this phenomenon, by balancing the electromagnetic torque, related to eddy currents on the first resistive conducting wall surrounding the plasma, and a viscous torque, which should tend to restore natural rotation of a mode which is not-wall locked mode.

The significantly higher resistivity of the first RFX boundary (defined by an Inconel 625 alloy vacuum vessel) (Gnesotto et al., 1995) relative to that of MST was at the origin of a significantly lower mode amplitude threshold for wall locking.

Shell gaps (necessary to allow the penetration of the vertical field) (Werley et al., 1985) or static misalignment of external coils are error fields sources which play relevant roles in the wall locking phenomenon (Fitzpatrick, 1999), as extensively confirmed in experiments using the active control systems for various RFP devices (Fitzpatrick and Zanca, 2002; Hansen et al., 1998; Frassinetti et al., 2015; Fridström et al., 2018).

In addition to wall locking, phase alignment among different TMs occurs in RFPs, (Fitzpatrick, 1999) due to the electromagnetic torques (Fitzpatrick, 1999; Fitzpatrick and Zanca, 2002; Guo and Chu, 2004; Hegna, 1996) from three-wave nonlinear coupling among $(1,n)$, $(1,n+1)$ and $(0,1)$ modes.

In RFX-mod, a significant mitigation of the effects of wall locked tearing modes was achieved thanks to the active feedback by saddle coils. A relevant role in determining the success of the control technique was the identification of the pollution in real mode amplitude estimation caused by coil sidebands.

Once clean harmonics had been obtained by a de-aliasing technique (Zanca et al., 2007a,b), the implementation in real time of the Clean Mode Control algorithm (Marrelli et al., 2007) allowed RFX-mod to reach the design current of 2 MA.

Clean Mode Control is capable of reducing the edge amplitude of each wall-locked tearing mode down to a minimum value, by properly optimizing the associated proportional gain (Zanca et al., 2007a,b), with a rotation of increasing frequency being generated by any further increase in gain.

In optimized very low plasma current (< 150 kA) experiments in RFX-mod, spontaneous tearing modes rotation was observed (Innocente et al., 2014), as shown in Fig. 12, and the typical hysteresis found in other RFPs on the locking-unlocking current threshold was found to be removed by the action of the Clean Mode Control algorithm.

Tearing mode rotation under Clean Mode Control was initially described by a stationary model (Zanca et al., 2007a,b) based on the same equations for single TM wall locking (Fitzpatrick and Zanca, 2002) and more recently by a time dependent approach including nonlinear phase coupling among multiple $m = 1$ tearing modes. In this code, dubbed RFXLocking (Zanca, 2009), mode edge amplitudes and phases are evolved under the action of the feedback coils, while the mode amplitudes at the resonant surface are estimated from the experimental measurements at the plasma edge. By also implementing in the code the multiple-shell structure of RFX-mod with a model-based approach (Piron et al., 2010), the optimal gain set was obtained in order to drive the edge radial magnetic field amplitude of each tearing mode to the lowest possible value. Indeed, the RFXLocking code shows that the edge radial field, because of a physical limit coming from the plasma-shell proximity, cannot be reduced below a minimum value.

Based on the interplay between tearing modes and passive conducting boundaries in an RFP, an upgrade of the RFX-mod machine assembly has been proposed, designed and is presently under implementation. The new version of the device, dubbed RFX-mod2 (Zuin et al., 2017; (Marrelli et al., 2019b; Peruzzo et al., 2017, 2019) will aim to explore RFP plasmas with a reduced saturated level of tearing modes, and the associated magnetic chaos, thanks to an improved plasma-shell proximity. This will be obtained in RFX-mod2 by removing the stainless steel vacuum vessel so that the copper shell will become the first conducting surface for the plasma, without modification of the existing magnetic coil system.

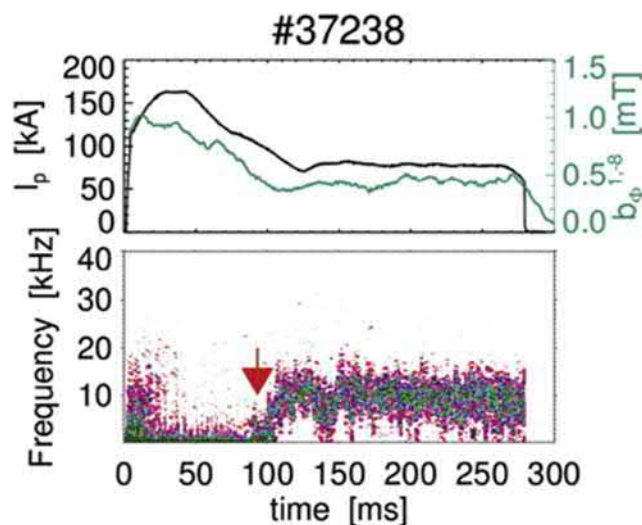


Fig. 12 Time traces for a low plasma current discharge in RFX-mod. (a) Plasma current (black line) and $m = 1$, $n = -8$ mode toroidal field harmonic amplitude (green line); (b) spectrogram of the mode frequency (the light green line is the estimated time behaviour). Red arrow indicates the time instant at which mode unlocking occurs. From Zuin M, Dal Bello S, Marrelli L, Puiatti ME, et al. (2017). Overview of the RFX-mod fusion science activity. *Nuclear Fusion* 57: 102012. <https://doi.org/10.1088/1741-4326/aa61cc>

The 3D nonlinear MHD Specyl code, including realistic boundary conditions, envisages a 30% reduction of the energy of unstable modes (Marrelli et al., 2019a), and model simulations by the RFXlocking code indicate a threefold reduction of the last closed magnetic surface deformation with respect to that of RFX-mod (Zuin et al., 2017), see Fig. 13.

Energetic ion confinement and stability

Thanks to the exploitation of its 1 MW NBI, interesting properties of fast ion behavior have been characterized in the MST device. Tangential injection is performed of 25 keV (40 A), H or D neutrals with pulse duration ≤ 20 ms; the “beam blip” technique is exploited for the estimation of the fast ions confinement, by measuring the decay in d-d fusion neutron emission (Fiksel et al., 2005). A neutral particle analyzer is devoted to the analysis of the dynamics of D and H energy distribution functions, and their role in determining energetic particle driven instabilities (Eilerman et al., 2012).

The energetic ion confinement time turns out to be an order of magnitude larger than that of thermal ions and consistent with classical slowing down (~ 20 ms) (Fiksel et al., 2005), (Bonofiglio et al., 2019).

This result is due to the drift of fast ions around the magnetic field lines, which implies that they do not experience magnetic field stochasticity, even in MH regimes. The safety factor in the phase space associated with fast particle motion turns out to be larger than the magnetic safety factor, so that it is not characterized by the resonance of the dominant core mode, while secondary mode islands superposition is reduced along with the associated stochasticity (Fiksel et al., 2005).

During NBI injection, both energetic particle modes (EPM) (Koliner et al., 2012) and magnetic-island-induced Alfvén eigenmodes (MIAE) (Anderson et al., 2013; Cook et al., 2016) are observed to be triggered (Lin et al., 2013). Measurements of density and magnetic field fluctuations by means of an interferometer-polarimeter show that energetic particle driven modes are nonlinearly coupled and induce a flattening of the fast ion profile, which has been modeled with the NUBEAM module in the TRANSP code (Budny et al., 1992; Waksman et al., 2012). The onset of EPM activity occurs for a critical gradient in the fast ion beta (Capecchi et al., 2019) and induces fast ions redistribution towards larger values of the minor radius where they are subject to stochastic transport (Bonofiglio et al., 2019; Capecchi et al., 2019). It is interesting to add that normalized gyro-radius of the 25 keV fast ions in MST ($B < 0.5$ T) is close to that of alpha particles in a reactor plasma with $B \sim 5$ T and that the central $\beta_{fi} \sim 8\%$ produced in the MST plasmas is comparable to β_a expected in a reactor one.

It is, moreover, worth mentioning that Alfvén eigenmodes have been also detected in ohmically heated RFP plasmas (i.e. without any external source of fast particles) in the RFX-mod discharges. These modes have been interpreted as global Alfvén modes (GAE) occurring around a minimum of the shear-Alfvén continuum at the plasma edge, with an amplitude raising during global reconnection phenomena (Spagnolo et al., 2011).

A stabilizing effect of the NBI particle population on core resonant tearing mode has been described (Anderson et al., 2013; Cai et al., 2015), which makes the MH to QSH states transition less frequent. On the other hand, NBI injection during QSH states produces, because of the helical distortion, a fast ion population distributed over a larger region in the core, which makes the stochastic transport region more effective around midradius. However, classical fast ion confinement is recovered if secondary modes are reduced (Bonofiglio et al., 2019).

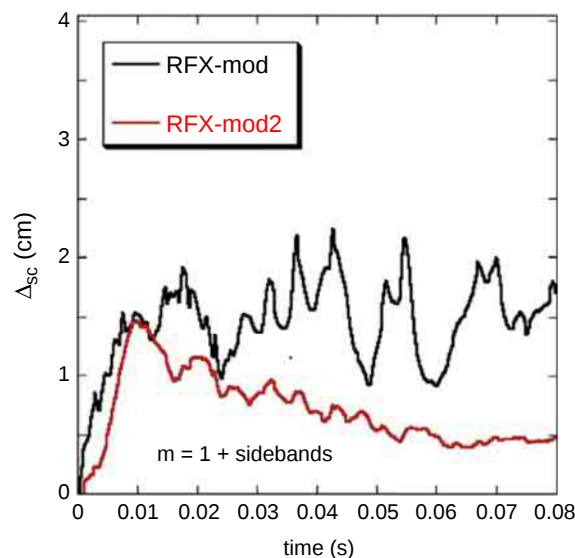


Fig. 13 Last closed magnetic flux surface deformation, with (black, for the RFX-mod device) and without the stainless steel vacuum vessel (red for RFX-mod2). From Zuin M, Dal Bello S, Marrelli L, Puiatti ME, et al. (2017) Overview of the RFX-mod fusion science activity. *Nuclear Fusion* 57: 102012. <https://doi.org/10.1088/1741-4326/aa61cc>

Inductive current drive and profile control

Tearing instabilities, whose action in helically deforming the current pattern sustains the RFP configuration, results from an electric field, which peaks towards the plasma axis. This stimulated the idea that adding current drive in the outer plasma region would allow the possibility that a self-consistent current profile would develop with no need of tearing action. Various techniques have been proposed for current profile control: current induction by means of external field coils, insertable polarized electrodes, and edge-localized Radio Frequency (RF) current drive.

Among these, the most successful has proven to be the inductive approach, as it has been demonstrated to lead to strong reduction of tearing activity and of the associated magnetic stochasticity. Consequently, the energy confinement of the RFP plasmas largely improves under these conditions.

The possibility to significantly reduce (or even suppress) tearing fluctuations with an active modification of the current profile is confirmed by MHD computations (Ho, 1991; Sovinec, 1995; Sovinec and Prager, 1999; Prager, 1999). An example of the results obtained with the DEBS code is shown in Fig. 14, where the profiles of the current parallel to B are presented in both a standard (steady state toroidal induction) RFP plasma (blue curve) and with the action of dynamic active current profile control (red curve). A dramatic change in the core-resonant tearing modes amplitude occurs, which drives a relevant stochasticity reduction, as clearly visible in the Poincaré plots in Fig. 14b and c.

Inductive profile control has been approached with a range of combinations of temporal variations of the toroidal and poloidal magnetic fields. The Pulsed Poloidal (or Parallel) Current Drive (PPCD) technique entails a fast ramp down of the toroidal magnetic field, while the Self-Similar Current Decay (SSCD) involves a ramp down of both fields. The Oscillating Poloidal Current Drive (OPCD) relies on oscillation of the toroidal magnetic field, and the Oscillating Field Current Drive (OFCD) incorporates an oscillation of both fields. The two latter techniques can be applied almost continuously, while the former two only transiently.

Among these, the PPCD is the most experimentally developed. It was first studied on MST (Sarff et al., 1994) and, as mentioned, by modelling with the DEBS code (Sovinec, 1995; Svidzinski and Prager, 2007; Reynolds et al., 2008).

An example of the behavior of a MST plasma with PPCD is given in Fig. 15, where the time traces are shown of the poloidal and toroidal components of the electric and magnetic fields at the edge, $E_\theta(a)$, $E_\phi(a)$, $B_\theta(a)$ and $B_\phi(a)$ respectively, along with that of the parallel electric field $E_{||}(a) = E \cdot B/B^2$. In order to obtain an $m = 1$ and $m = 0$ reduction, along with the non-coincidence with spontaneous reconnection processes (Chapman et al., 2002), $E_{||} \geq 0$ is required, which is obtained initially by a B_ϕ ramp down. The action is prolonged by an induced E_ϕ reversal. The technique drives the reversal equilibrium parameter, F , to largely negative values (around -2). The best plasma performances with the PPCD technique have been obtained at low density with good conditioning of the plasma-facing components and good control of the magnetic boundary (Sarff et al., 1995; Chapman et al., 2002; Frassinetti et al., 2003, 2004, 2005c).

During PPCD the μ profile ($\mu = -\mu_0 a J_{||}/B$) is more peaked in the center of the plasma, which has also been observed in MHD modeling of PPCD plasmas in RFX with the SpeCyl code (Puiatti et al., 2003) and decreases at the edge as experimentally measured in MST (Chapman et al., 2000), so that the electric field in the parallel Ohm's law is approximately equal to the resistive current terms $\eta_{neo} J_{||}$ over the whole plasma (i.e. the plasma is indeed almost dynamo free) (Anderson et al., 2004, 2005).

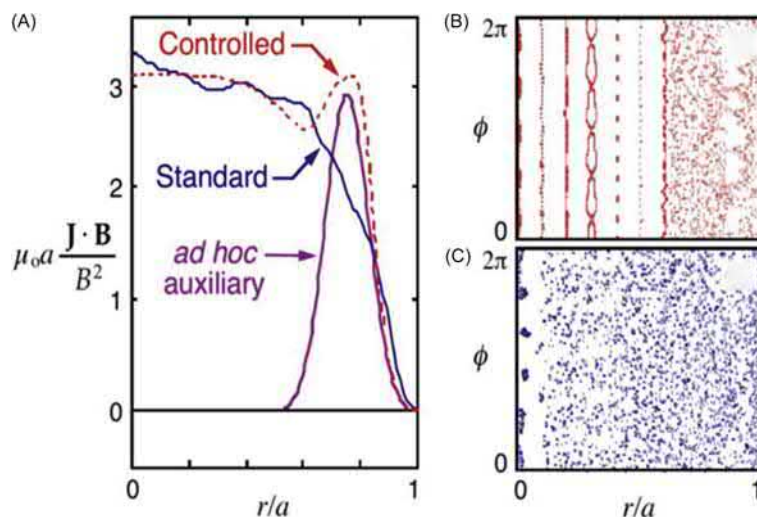


Fig. 14 Results of MHD simulations with the DEBS code: (a) comparison of the $J_{||}/B$ radial profile in the standard steady toroidal induction case (blue) and controlled case (red). Magenta curve is an auxiliary ad-hoc parallel electron force. Poincaré plot of magnetic field line in controlled (b) and standard (c) conditions. Adapted from Prager SC (1999) *Dynamo and anomalous transport in the reversed field pinch. Plasma Physics and Controlled Fusion* 41: A129. <https://doi.org/10.1088/0741-3335/41/3A/008>

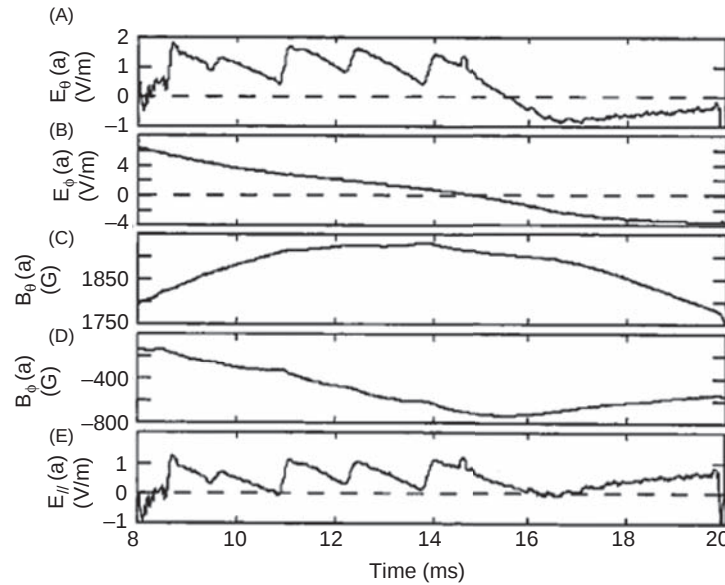


Fig. 15 Time evolution of a MST 500 kA plasma with PPCD: (a) poloidal and (b) toroidal electric fields, (c) poloidal and (d) toroidal magnetic fields and (e) parallel electric field. PPCD starts at 8.5 ms. From Chapman BE, Almagri AF, Anderson JK, Biewer TM, et al. (2002) High confinement plasmas in the Madison Symmetric Torus reversed-field pinch. *Physics of Plasmas* (1994-present) 9: 2061–2068. <https://doi.org/10.1063/1.1456930>

As a consequence, a relevant broadband reduction of higher- n modes is induced (Sarff et al., 2003a,b) as measured in MST both at the edge and in the core (Ding et al., 2003; Terry et al., 2004).

Higher-frequency magnetic turbulence is also affected, becoming less intermittent (Marrelli et al., 2005; Frassinetti et al., 2007b).

The reduction in core-resonant tearing modes has tremendous consequences on the magnetic topology, and two distinct (i.e. not overlapping) islands associated with the two innermost resonant modes have been clearly detected by tomographic X-ray measurements (Franz et al., 2004, 2006), a clear sign of decreased stochasticity.

A more ordered magnetic topology is confirmed by the detection of runaway electrons in the core of MST plasmas (O’Connell et al., 2003), normally prevented in RFP plasmas with stochastic magnetic field.

The reduced stochasticity with PPCD leads to a global improvement of the plasma performance in various RFPs (Martini et al., 1999b; Chapman et al., 2001; Yagi et al., 2002; Cecconello et al., 2003).

This improvement includes a better confinement of energetic electron and ion particles, of thermal electron particle, larger values of the electron density in terms of Greenwald fraction, record values of electron and ion temperature for a RFP. These are accompanied by a reduction of the loop voltage necessary to sustain the discharge (and hence of the Ohmic heating power) and beta values exceeding the Mercier criterion along with classical confinement of impurity ions.

Global particle confinement time improves eight-fold in MST (Lanier et al., 2000, 2001), while the particle diffusion rate drops by about a factor of two in RFX plasmas (Puiatti et al., 2003). In TPE-RX a fivefold improvement in the global particle confinement time (Koguchi et al., 2006) is estimated when pellets are injected during PPCD. This technique allows access to regimes with improved confinement at higher density in PPCD plasmas (Lorenzini et al., 2002; Puiatti et al., 2003; Koguchi et al., 2006), even above the Greenwald density limit (Wyman et al., 2008, 2009; Chapman et al., 2009; Caspary, 2014; Sarff et al., 2015).

An improvement of impurity ions confinement also occurs with PPCD; in RFX the core region of lower diffusivity is expanded from the outer 10% of the plasma in steady condition to 20% with PPCD (Carraro et al., 2002), while in EXTRAP T2R, diffusion decrease by about a factor of two in the core was estimated (Kuldkepp et al., 2006). In MST, impurity transport decreases to the classical value (Kumar et al., 2012a,b; Barbui et al., 2014). Impurity ions drifts are not dominated by neoclassical effects and the ions experience an outward convection, so that the impurity density profile and the Z_{eff} profile (Galante et al., 2015) become hollow.

A significant increase in the central electron temperature is typically found (Sarff et al., 1997; Stoneking et al., 1998; Bartiromo et al., 1999; Chapman et al., 2001; Cravotta et al., 2003; Yagi et al., 2003), with central $T_e \sim 2$ keV in MST. On the other hand, only a small effect is observed on ion temperature dynamics (Chapman et al., 2002) during PPCD in low density plasmas, but, interestingly, the ion heating coming from magnetic energy conversion during global reconnection processes is trapped in the plasma if such events are triggered shortly before PPCD starts, with a ten-fold increase of the ion energy confinement time.

When PPCD is applied in higher density plasmas in combination with pellet injection, an ion temperature increase is measured in MST (Wyman et al., 2008) as a consequence of the increased ion-electron collision rate and the larger input Ohmic power required for the sustainment of the discharge.

The energy confinement time doubles in RFX and EXTRAP T2R (Martini et al., 1999b; Cecconello et al., 2003), while in TPE-RX a five-fold increase was estimated (Yagi et al., 2003; Frassinetti et al., 2007a) and a ten-fold increase in MST, which exhibits τ_E up to 12 ms (Chapman et al., 2009, 2010). The MST values are not far from tokamak-like (Sarff et al., 2003a,b; Chapman et al., 2010), as

can be deduced by Fig. 16 where data from MST standard and PPCD discharges are compared to the tokamak reference with the same plasma size, magnetic field strength, and heating power as predicted by the IPB98(y,2) ELMY H-mode tokamak confinement scaling (ITER Physics Expert Group on Confin., 1999).

Obviously, the tokamak scaling does not apply to the RFP case, but it is interesting to observe that the best RFP energy confinement is comparable to that expected for a tokamak plasma with the same size and magnetic field.

The improved confinement and the reduction of fluctuations due to tearing modes in PPCD plasmas is accompanied by a relevant modification of the properties of the high frequency (> 10 kHz) component of the fluctuation spectra. In particular, density-gradient-driven trapped electron modes (TEM) have been identified in MST at $k_{\perp} \rho_i \approx 0.1 - 0.2$, where $k_{\perp}$ is the perpendicular wavenumbers and ρ_i the ion Larmor-radius, propagating in the electron diamagnetic drift (Duff et al., 2018). These modes are destabilized due to a density profile overpassing a critical gradient threshold $R/L_n \approx 20$, where L_n is the local density gradient scale length, $L_n \equiv n_e(r)/|\nabla n_e|$, as shown in Fig. 17. The results are in agreement with the predictions of linear gyrokinetic modeling by the GENE code (Carmody et al., 2015; Terry et al., 2015).

As mentioned, in addition to PPCD, various inductive current profile control techniques have been proposed and studied experimentally on RFP devices.

The Self-Similar Current Decay (SSCD) (Watterson and Gimblett, 1983; Caramana and Nebel, 1988) entails a self-similar temporal ramp-down of both B_{θ} and B_{ϕ} to produce an inductive electric field due to the magnetic flux decay which sustains the current with $E_{\parallel} = \eta J_{\parallel}$ (no dynamo) with a MHD stable current profile (Nebel et al., 2002). SSCD discharges in RFX-mod exhibit magnetic fluctuations drop, a 50% increase of the poloidal beta, ($\beta_p = 2\mu_0 \langle p \rangle / B_{\theta}^2 a$), and of the global energy confinement time (Zanca, 2007). First, non-optimized, attempts on MST suggest a three-fold increase of the energy confinement time. The SSCD technique is planned to be more deeply and extensively investigated in the RFX-mod2 device.

The quasi-steady Oscillating Poloidal Current Drive (OPCD), which entails oscillation of the toroidal magnetic field, was tested in both RFX (Bartirromo et al., 1999; Bolzonella et al., 2001; Carraro et al., 2002) and RFX-mod (Terranova et al., 2007). In OPCD plasmas during the PPCD-like phase (i.e. phase of decreasing reversal parameter, F), robust transitions to Quasi Single Helicity States occurred, as already mentioned. Helical electron temperature structures emerged, with global energy confinement time increasing by about 60% (Terranova et al., 2007), which is, anyway, lower than the one estimated during the spontaneous SHAx states.

Among the inductive current drive techniques, the Oscillating Field Current Drive (OFCD), which entails oscillation of both B_{θ} and B_{ϕ} fields, is the one with interesting implications for a RFP based fusion reactor as it opens the possibility of a steady-state sustainment scenario. In particular, in OFCD, oscillatory toroidal and poloidal loop voltages are applied, and the nonlinear self-organization process operating in RFP plasmas generates a dc current. In principle, the plasma current can be indefinitely sustained by zero time-average loop voltages, without the need for a gradual increase in the net magnetizing flux applied.

Nonlinear visco-resistive MHD modeling of RFP plasmas gives the theoretical basis for OFCD operation, passing through the principle of a relaxed state, where the magnetic energy is minimized with the conservation of the total magnetic helicity, K , constraint.

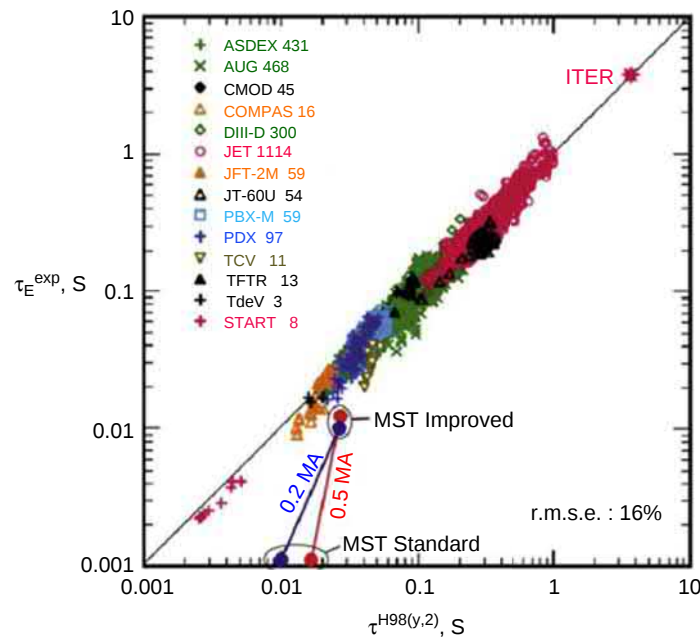


Fig. 16 MST confinement time in standard and improved (PPCD) conditions compared to the IPB98(y,2) ELMY-H mode empirical scaling for tokamak. From Chapman BE, Almagri AF, Anderson JK, et al. (2010) Generation and confinement of hot ions and electrons in a reversed-field pinch plasma. *Plasma Physics and Controlled Fusion* 52: 124048. <https://doi.org/10.1088/0741-3335/52/12/124048>.

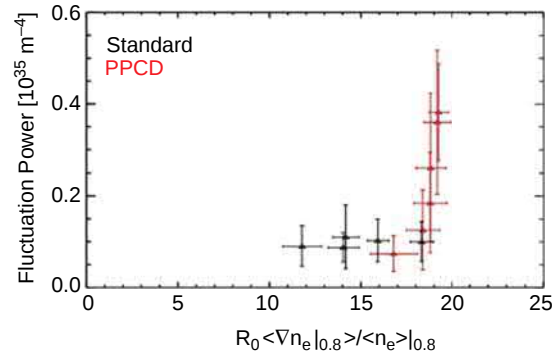


Fig. 17 Density fluctuation power (line-integrated density fluctuations measured at $r/a = 0.86$) by FIR scattering) as a function of the normalized inverse density gradient scale length evaluated at $r/a = 0.8$. In PPCD plasmas (red points) TEM emerge above a critical-gradient threshold close to that predicted by GENE. From Duff JR, Williams ZR, Brower DL, Chapman BE, et al. (2018) Observation of trapped-electron-mode microturbulence in reversed field pinch plasmas. *Physics of Plasmas* 25: 010701. <https://doi.org/10.1063/1.5010198>.

If both toroidal and poloidal loop voltages are sinusoidal at a given frequency, then net helicity injection, averaged over a cycle, is possible and maximized for a phase shift $\delta = \pi/2$. The frequency, must be greater than the inverse resistive diffusion time, $\tau_R^{-1} = \eta/\mu_0 a^2$, but lower than the typical time scale of magnetic relaxation for the RFP, i.e. the resistive-Alfvénic hybrid timescale ($\sqrt{\tau_R \tau_A}$), typically in the audio frequency range.

A demonstration of full current sustainment by OFCD in present experiments is not feasible due to the relatively low Lundquist number. 3D, nonlinear, resistive MHD modelling (DEBS code) was used to study the effect of OFCD at moderate $S = 10^5$ in conditions with a large AC ripple (Ebrahimi et al., 2003).

The first OFCD tests were performed on the ZT-40M device (Schoenberg et al., 1988). An extensive experimental analysis has been devoted to the OFCD technique on the MST device over a range of plasma current. The OFCD drive and anti-drive results from MST are shown in Fig. 18 (McCollam et al., 2006), where a continuous current ramp-up during the flat-top of the discharge is induced in the former case. On the basis of the experimental results, a limit of 15% of the total current is estimated to be drivable by OFCD. The maximum OFCD-driven current in MST plasmas occurs for oscillator phasing $\delta = \pi/8$ and not $\delta = \pi/2$, as expected for a global helicity balance, which has been interpreted as being due to an increase in the dissipation of magnetic helicity (McCollam et al., 2010), related to the dynamics associated with relaxation and confinement.

Nevertheless, the good agreement between the simulations and experiment suggests a robust physics basis for OFCD and further tests of OFCD in RFP plasmas with larger size and temperature, i.e., at larger Lundquist number, are envisaged to be performed in the RFX-mod2 device, presently under development. In particular, in high Lundquist number RFP plasmas, the efficacy of OFCD in the quasi-single-helicity (QSH) regime typical of such conditions needs to be verified.

Plasma-wall interactions issues

In the RFP the magnetic topology strongly influences the plasma-wall-interaction (PWI). The locked mode, due to nonlinear tearing mode interactions, produces deformations of the last closed magnetic surface, which can induce strongly local power deposition on the first wall, leading to uncontrolled release of both main fuel and impurities. An example is shown in Fig. 19, where the image taken from a CCD internal camera is compared to the reconstruction of the magnetic perturbation for a RFX-mod discharge.

In RFX-mod the tearing mode mitigation by the active control system previously discussed allowed a much more uniform PWI to be obtained, compared to that of the previous RFX, while in the EXTRAP T2R active feedback control allowed an improvement of the performance (Brunsell et al., 2004).

In the RFX-mod the helical ripple of QSH states modulates the plasma edge, so that the PWI is distributed over a helical region much larger than in the MH state, with a helical modulation of edge plasma parameters also found (Scarin et al., 2011, 2019). The helically deposited power density is $\approx 5 \text{ MW/m}^2$ even at the highest current (Puiatti et al., 2013b), with plasma radiation more symmetric than in MH (Spizzo et al., 2001). However, even in the purest QSH, the phase locking of the secondary modes induces a residual PWI, which it is planned to avoid in the next RFX-mod2 device (Scarin et al., 2019).

PWI is of particular relevance in devices with a carbon wall, such as RFX-mod, as the plasma density level of the discharge is mainly determined by wall desorption, with a recycling coefficient R greater than 1, rather than controlled by the fueling (Bergsäter et al., 1995; Puiatti et al., 2013a).

With the aim of avoiding a strong (uncontrolled) release of particles in high current discharges, a wide range of wall conditioning treatments are routinely applied in the various RFPs. These include Glow Discharge Cleaning (GDC), Pulse Discharge Cleaning, boronization, lithization, hot wall operations (Bergsäter et al., 1997; Canton et al., 2013). As an example, the positive effects of boronization obtained in MST include prolonged phases without discrete reconnection events, because of a lower plasma resistivity, and a factor of 3 increase of the energy confinement time (Chapman et al., 1996).

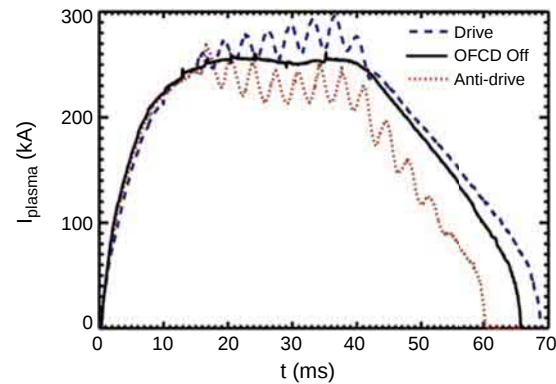


Fig. 18 Time trace of plasma current in MST discharges with OFCD (drive and antidrive) compared to a standard plasma (without OFCD). From McCollam KJ, Blair AP, Prager SC, and Sarff JS (2006) Oscillating-field current-drive experiments in a reversed field pinch. *Physical Review Letters* 96: 035003. <https://doi.org/10.1103/PhysRevLett.96.035003>

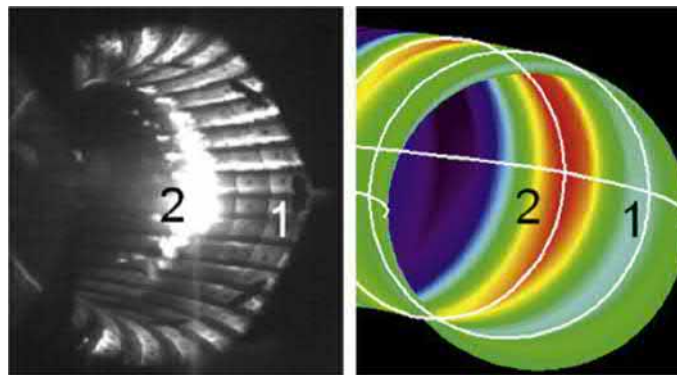


Fig. 19 (left) image taken with a CCD camera, where the footprint of the perturbation associated with the Locked Mode in a RFX-mod discharge is clearly visible; (right) magnetic reconstruction of the perturbation. Colors refer to the distance between the non-axisymmetric plasma and the wall, ranging from blue (furthest) to red (closest). Adapted from Martini S, Agostini M, Alessi C, Alfier A, et al. (2007) Active MHD control at high currents in RFX-mod. *Nuclear Fusion* 47: 783. <https://doi.org/10.1088/0029-5515/47/8/008>

Lithization, implemented in RFX-mod by a lithium capillary-pore system (Puiatti et al., 2013b) and lithium pellet injection (Munaretto et al., 2015; Innocente et al., 2015) induces significant decrease of oxygen and carbon influxes, improvement of density control, a slight peaking of density profiles, though a clear confinement improvement has not been observed.

As described previously, a close fitting conducting wall is mandatory in RFPs. For this reason, present RFPs are not equipped with divertors, which would, in any case, present additional difficulties due to the non-stationarity and non-uniformity of the edge topology. The feasibility of a pumped limiter as an alternative to a divertor has been proposed (Sonato et al., 2002), but not yet experimentally applied.

However, the similarity of the 3D field structures of RFPs and stellarators suggests the possibility of exploiting the coil system to induce a magnetic configuration (Martines et al., 2010) resembling the island divertor of the Stellarator (Konig et al., 2002). This concept would rely on the chain of $m = 0$ magnetic islands that develop at the reversal surface with the same helicity as the dominant mode in QSH-SHAX states. The X-points of $m = 0$ islands, which would need to be actively controlled, would act as the X-point of a divertor with localized plates.

Density and β limit

In the RFP both low and high density limits characterize the operational parameter space. The lower limit is posed by the generation of runaway electrons and the growth of magnetic fluctuations. As the purest and longest helical states are found in low density conditions, this regime has been extended due to the exploitation and optimization of the Clean Mode Control feedback system (Gobbin et al., 2015). The upper density limit was compared to the Greenwald limit (Greenwald, 2002) in both RFX-mod (Valisa et al., 2004) and MST standard plasmas (Wyman et al., 2009). Indeed, the RFP is found to exhibit an empirical limit, that somehow resembles that of tokamaks, being related to the current density, which, as already mentioned, can be very large in RFPs.

Recently, a power balance model accounting for neutrals and impurity radiation has been developed, which successfully describes density limits in the stellarator, RFP and L-mode (low confinement mode) tokamak (Zanca et al., 2017, 2019).

In the RFX-mod discharges at values close to the Greenwald value, a temperature decrease is measured together with a density accumulation at the edge (Spizzo et al., 2010). For densities above a critical value, $n/n_G \approx 0.35$, a back transition from the QSH to the MH regime occurs (Scarin et al., 2011). A poloidal radiating belt forms (Valisa et al., 2004; Puiatti et al., 2009c), with similarities with the highly radiating MARFE in tokamaks (Lipschultz, 1987) or Serpens in the LHD stellarator (Miyazawa et al., 2006).

It is important to note that the density limit is not a disruptive one in the RFP, but that normally only a soft decrease of plasma current occurs. Indeed, in the RFP, when the density increases, the plasma cools, so that resistivity increases along with the loop voltage, i.e. the input ohmic power necessary to sustain the discharge (Spizzo et al., 2010).

The role played by the $m = 0$ mode magnetic topology, associated with the maximum deformation due to the locking in phase of $m = 1$ modes, in determining density accumulation and hence the density limit has been the focus of dedicated experimental campaigns in RFX-mod (Spizzo et al., 2012), (Spizzo et al., 2015). This is summarized in Fig. 20, where the maximum H_α light emission can be correlated with the $m = 1$ mode deformation, (green and red lines in panel (d)), thanks to the application

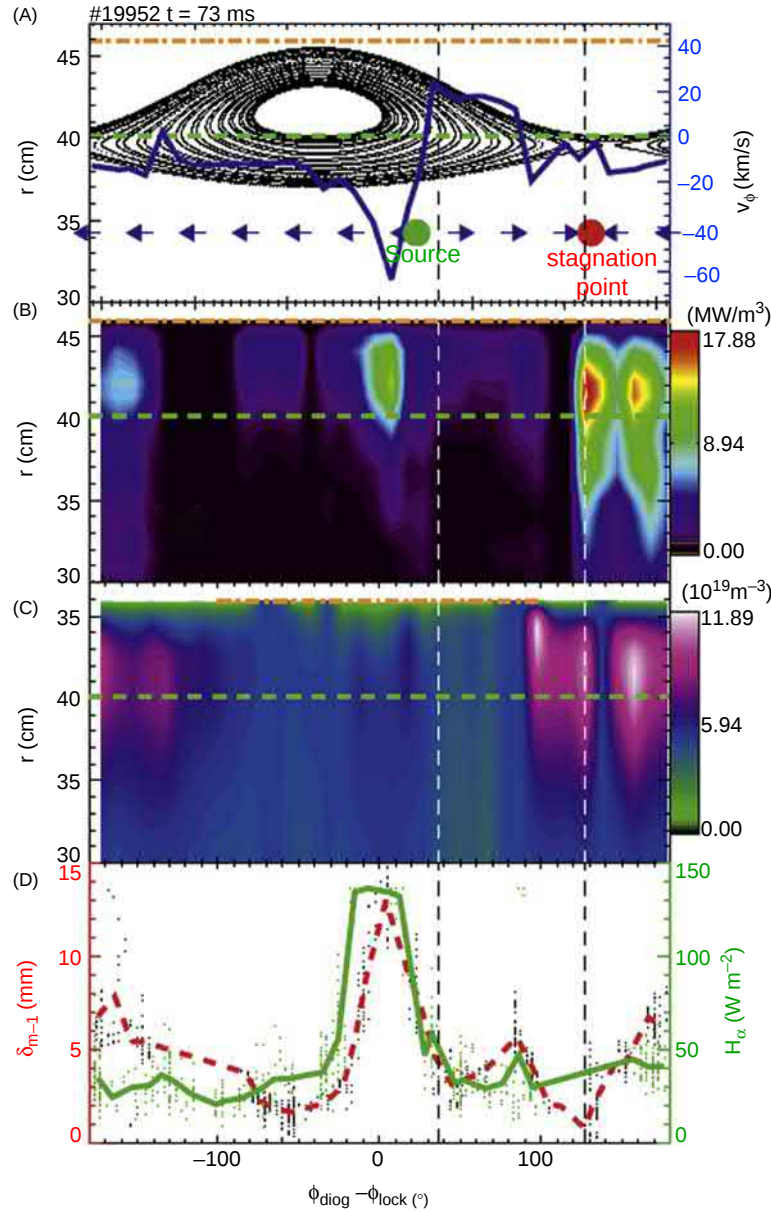


Fig. 20 A high density, $n/n_G = 0.8$, RFX-mod discharge: (a) plasma flow as a function of the normalized toroidal angle, $\phi - \phi_{\text{lock}}$ and Poincaré plot of the $m = 0, n = 1$ island; (b) tomographic map of total radiation; (c) map of electron density; (d) H_α emissivity (solid) and $m = 1$ deformation (dashed). The stagnation point for the plasma flow corresponds to the X-point of the island and to the maximum radiation and density. The reversal surface, i.e. where $B_\phi = 0$, is indicated by a dashed, horizontal line in panels (a)–(c). From Spizzo G, Agostini M, Scarin P, Vianello N, et al. (2012) Edge topology and flows in the reversed-field pinch. *Nuclear Fusion* 52: 054015. <https://doi.org/10.1088/0029-5515/52/5/054015>

of an upgraded version of the guiding-center code ORBIT (White and Chance, 1984), including the effect of a recycling wall (Spizzo et al., 2012).

In RFP plasmas $n/n_G > 1$ regimes have been produced in several conditions. In RFX-mod, the Greenwald value has often been exceeded at low current levels (≤ 0.3 MA) (Spizzo et al., 2015), when the amount of available loop voltage, due to the input ohmic power limit, is maximum. Moreover, helium plasmas favour $n/n_G > 1$ conditions (Valisa et al., 2004; De Masi et al., 2012).

In MST, $n/n_G = 1.4$ ($n_e = 0.9 \cdot 10^{20} \text{ m}^{-3}$) has been obtained by exploiting cryogenic pellet injection during PPCD phases (Sarff et al., 2015; Wyman et al., 2009).

No disruptive β limit exists for the RFP, but a “soft” degradation of confinement is observed, likely when pressure driven emerges during improved confinement regimes induced by active current profile control (Wyman et al., 2009).

Experimentally, high beta values of up to $\beta_p \approx 18\%$ had been obtained in the TPE-1RM20 RFP with a reduction of magnetic fluctuation associated with the dynamo activity (Hirano et al., 1996) and in TPE-RX as a result of fast gas puffing (Sakakita et al., 2004). Also in RFX-mod, the highest poloidal beta values of $\approx 15\%$ have been observed when n/n_G approaches 1 (Innocente et al., 2007), while a small degradation of β_p with increasing plasma current and a more significant increase as n/n_G raises is predicted by statistical analysis (Innocente et al., 2009). In MST, a modest decrease of β_p with increasing current was reported (Stoneking et al., 1998), while 3D resistive MHD simulations by the DEBSP code, based on stochastic transport scaling (Scheffel and Schnack, 2000), predict a dependence of β_p on I_p which is stronger than the experimental RFX-mod findings.

In various RFP devices record values of transient β enhancement were achieved by applying current profile techniques in high-density plasmas, as, for example, in RFX, where a 50% β_p increase was induced by Oscillating Poloidal Current Drive (Martini et al., 1999a). In TPE-RX and MST, PPCD leads to an enhancement of β_p by up to a factor of 2 (Sarff et al., 2003a), while record values were measured when during PPCD phases in which deuterium pellets were injected (Chapman et al., 2009; Koguchi et al., 2009).

By optimizing the pellet parameters in MST, pressure gradients exceeding the Mercier limit have been produced (Wyman et al., 2009; Chapman et al., 2009), corresponding to a value of $\beta_p \approx 40\%$. In such regimes both pressure-driven interchange modes and tearing modes are predicted to be linearly unstable (Chapman et al., 2009). Interchange modes have not been observed in MST, while in RFX-mod the observation of edge magnetic fluctuations, which correspond to linearly unstable pressure driven modes with tearing parity, has been reported (Zuin et al., 2010).

Neoclassical bootstrap current at high β

In an axisymmetric RFP, the fraction of trapped particles in the core is similar to that in a tokamak with the same aspect ratio, while it falls to lower values at the edge, depending on the current profile (Stoneking et al., 1998). Due to the higher poloidal field, the corresponding bootstrap current is much lower than in tokamaks (Gobbin et al., 2009a; Zanca and Terranova, 2004; Guazzotto et al., 2004). In present day circular RFPs the bootstrap mechanism can provide only a negligible additional current (Guazzotto and Paccagnella, 2009). In RFP Quasi Single Helicity states, because of the additional helical field ripple, a modest increase of the bootstrap current is envisaged (Gobbin et al., 2009a), which could only be slightly improved by shaping effects (Guo et al., 2013).

The situation is considerably different in low aspect ratio RFP equilibria, due to neoclassical viscosity enhancement (Sanpei et al., 2009). In a reactor relevant regime, with an aspect ratio of 2, with appropriate shaping it is predicted that an equilibrium with 94% bootstrap current and $\beta = 63\%$ which is stable to ideal kinks and Mercier localized modes could be sustained (Shiina et al., 2005).

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Magnetic Confinement Fusion—Technology: Fusion Core

Gianfranco Federici^{a,*}, Christian Bachmann^{b,*}, Valentina Corato^c, Samuel Jimenez^d, and Jeong-Ha You^e, ^a Fusion for Energy, Garching, Germany; ^b Technical University of Denmark, Lyngby, Denmark; ^c ENEA, C.R. Frascati, Rome, Italy; ^d CCFE Culham Science Centre, Abingdon, United Kingdom; and ^e Max Planck Institute for Plasma Physics, Garching, Germany

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Glossary

Breeding blanket It is the region surrounding the fusion reactor core. The blanket of a fusion reactor has the threefold purpose of breeding the tritium by interactions of the fusion neutrons with lithium, of converting the energy of the neutrons to high-grade heat for electricity generation, and of shielding the super-conducting magnets from energy deposition and radiation damage.

Cable-in-conduit conductor Current carrying element made by a cable surrounded by a metallic jacket.

Cassette body (divertor) A cassette body (made of stainless steel in ITER) mechanically supports and feeds coolant to the divertor plasma facing components (PFCs). The entire cassette assembly must be capable of withstanding the thermal loads

*Currently on assignment to EUROfusion.

from the plasma and from the neutron flux, and the loads generated by electromagnetic forces. Also, it performs the function of neutron shielding.

Critical heat flux The heat flux limit at the cooling circuit inner wall at which the onset of a phase change (boiling) of coolant occurs forming a layer of bubbles covering the wall surface and thus causing a severe damage of the overheated wall material due to hindered heat transfer.

Central solenoid coils Components of a tokamak that produce inductive magnetic flux to ramp up and maintain the plasma current and contribute to shape the plasma.

Correction coils Components of a tokamak that correct magnetic field errors.

DEMO Acronym for Demonstration Reactor. The requirements vary from nation to nation. In Europe the main aims of DEMO are to demonstrate the production of few hundred MWs of net electricity, the feasibility of operation with a closed-tritium fuel cycle, and maintenance systems capable of achieving adequate plant availability.

Divertor Component of a magnetically confined toroidal fusion device that diverts charged particles on the outer edge of the plasma into a separate chamber where they strike a barrier and become neutralized. In a reactor, the divertor would incorporate a system for pumping out the neutralized particles as exhaust from the machine. A divertor, like a limiter, prevents the particles (including helium ash) from striking and degrading the chamber walls and dislodging secondary particles that would cool and contaminate the plasma.

Double pancake Methodology for winding the conductor.

First wall Innermost shield of the main vessel wall of a fusion reactor. It forms the plasma-facing liner of the blanket structure and protects the underlying steel structure from hot plasma.

High-heat-flux Heat fluxes expected for the plasma-facing components in a fusion reactor typically ranging from a few MW/m² to several tens MW/m².

High temperature superconductors Superconducting materials with a critical temperature above the liquid nitrogen temperature (77 K).

In-vessel components Components installed inside the vacuum vessel of a fusion reactor. Divertor, blanket, pumping systems, port plugs and diagnostic sensors belong to this category.

ISI In-service inspection. Regular inspections of the reactor pressure vessel to verify that no damage has occurred.

Limiter Structures placed in contact with the edge of a confined plasma, which are used to define the shape of the outermost magnetic surface (see entry for LCFS). See also entry for divertor.

Low temperature superconductors Superconducting materials with a critical temperature lower than 20 K. The term generally refers to Nb-based superconductors that were already in use prior to the discovery of HTS.

Plasma-facing components In-vessel components mounted onto the vessel and facing plasma or standing in direct contact with plasma: first wall and divertor.

Poloidal field coils Components of a tokamak that assist in shaping and stabilizing the plasma.

Radial plates Thick steel plates to embed the insulated superconductors in a magnet winding.

React&Wind Manufacturing approach for Nb₃Sn coils.

Scrape-off layer Plasma edge layer outside the boundary separating those magnetic field lines that intersect the wall (open lines) from the closed field lines that never intersect the wall (closed lines).

Toroidal field coils Components of a tokamak that assist in stabilizing the plasma, by creating a “magnetic bottle” for confinement.

Vacuum vessel The vacuum vessel maintains a high vacuum/low impurity volume for the production of the plasma. In a reactor it also acts as a first confinement barrier for tritium and contributes to limiting the heat flux to the toroidal field coils.

Wind&React Manufacturing approach for Nb₃Sn coils.

Winding pack Region of the cross-section of a magnet that contains the (super)conducting cables.

Abbreviations

BB Breeding blanket

CHF Critical heat flux

CC Correction coils

CICC Cable-in-conduit conductor

CS Central solenoid

Cu Copper

DEMO Demonstration power reactor

DP Double pancake

FW First wall

HHF High-heat-flux
 HT Heat treatment
 HTS High temperature superconductors
 ISI In-service inspection
 IVCs In-vessel components
 JET Joint European Torus
 LTS Low temperature superconductors
 Nb₃Sn Niobium-tin
 NbTi Niobium-titanium
 PF Poloidal field
 PFC Plasma-facing components
 R&D Research and development
 R&W React&Wind
 RP Radial plates
 SC Superconducting
 S-He Supercritical helium
 SOL Scrape-off layer
 SS Stainless steel
 TF Toroidal field
 VV Vacuum vessel
 W&R Wind&React
 WP Winding pack

Introduction

This chapter provides a brief account of the current status of the design and development of the magnetic fusion reactor core systems, such as the superconducting magnets, the divertor, the vacuum vessel and remote maintenance. Other important systems, such as the breeding blanket are not discussed here but the interested reader can consult other sources (see for example [Boccaccini, 2021](#)).

ITER technology is discussed to illustrate the current status of technology R&D toward a fusion power plant, and the DEMO studies are presented to illustrate the considerations involved and progress being made in extrapolating the technology toward the even more exacting demands of a power plant generating electricity with high availability. The main differences between ITER and DEMO and the outstanding DEMO design and technology challenges, where knowledge gaps exist, are discussed by [Federici et al. \(2014\)](#) and by [Federici et al. \(2017\)](#). In this sense, the DEMO technology can be considered as ‘prototypical’ of that used in a power plant, but with the expectation that there may be further evolution in the development of industrial scale fusion power plants.

Conceptual designs have been developed for both tokamak and stellarator approaches to a fusion power plant. Although the magnetic configurations and detailed engineering designs differ, much of the technology under development for the tokamak DEMOs currently under study is also applicable to stellarator designs. Therefore, the discussion in the chapter is focussed on the tokamak configuration, where the engineering analysis and R&D is more advanced, but the considerations are relevant to the stellarator line of development.

Conceptual tokamak power plant designs have considered both pulsed and steady-state operation of the fusion core. For the EU DEMO technology, which is discussed as prototypical of power plant technology in the present chapter, a pulsed design has been chosen to accelerate the progress to a device capable of electricity production ([Federici et al., 2019](#); [Donne, 2021](#)).

Superconducting magnets

Superconducting magnets in fusion devices

In the framework of magnetic confinement fusion, magnetic fields are used to initiate, confine, shape and control the fusion fuel in the form of [plasma](#). Most of the experimental devices built until now are toroidal machines, such as tokamaks and stellarators.

The properties, advantages and challenges of the various magnetic configurations are discussed by [Galvão and Canal \(2021\)](#).

Starting from the 1980s relevant milestones in plasma engineering were achieved in pulsed operation by devices made with copper windings, such as the Tokamak Fusion Test Reactor (TFTR) ([Meade, 1995](#)), The Joint European Torus (JET) ([Wesson, 2000](#)) and the JAERI Torus-60 (JT-60) ([Ando et al., 1988](#)). Many deuterium-tritium (D-T) experiments have been conducted on

TFTR and JET, which have produced fusion powers of 11 MW (in 1994) and 16 MW (in 1997), respectively. The JET device is still in operation to test a beryllium/tungsten plasma facing wall and new D-T experiments for ITER.

Despite evident advantages of technological maturity and low costs, copper coils cannot generate steady-state magnetic fields higher than a few Tesla in very large volumes. The need for large, reliable, high-field superconducting (SC) magnets has been recognized since the 1980s to satisfy the long-pulse or steady-state demands of power-producing devices.

The intrinsic advantage of using SC magnets is the possibility of carrying higher current and producing stronger magnetic field than conventional copper (Cu) coils. Compared to traditional technology, the use of superconductivity allows the power consumption to be reduced, even if SC coils must be cooled with supercritical helium (S-He) in the range of 4 K (-269°C). However, for large-scale reactors such as ITER or DEMO that aim to confine and shape the plasma for long burning periods, superconductivity is an essential technology (Bruzzone, 2015).

Superconducting coils are generally manufactured using low-temperature superconductors (LTS), such as niobium-titanium (NbTi) for magnetic fields up to about 6 T or niobium-tin (Nb_3Sn) for fields between about 6 T and 16 T.

In the manufacturing process of Nb_3Sn magnets a specific heat treatment (HT) is required for allowing the formation of the SC material. Since Nb_3Sn becomes brittle after the HT and experiences degradation of its performances if subject to tensile strain, the HT generally follows the winding phase of the manufacturing procedure. This is the so-called Wind&React (W&R) approach, which is used for ITER coils. However, also the React&Wind (R&W) technology is considered for the manufacture of Nb_3Sn coils. In this second approach the HT is performed before the winding, which requires extreme care in manufacturing and handling of the magnet winding. On the other hand, the R&W approach allows the reduction of the effective compressive strain acting on Nb_3Sn with a notable improvement of the SC performances.

Although there are many examples of stellarator experimental devices, in particular the Large Helical Device (Imagawa et al., 2009) in Japan and Wendelstein 7-X (Rummel et al., 2012) in Germany, which both make use of superconducting coils, most of the existing magnetic fusion devices are tokamaks. Some fusion devices implementing SC magnets are already in operation, such as Tore Supra (Equipe Tore Supra, 1989) in France, KSTAR (Kim et al., 2005; Kwon et al., 2011) in Korea and EAST (Wu and EAST team, 2007; Wei et al., 2010) in China. Other fusion devices based on superconducting Tokamaks are in construction, like JT-60SA (Yoshida et al., 2010) in Japan, and ITER (Mitchell et al., 2012) in France.

A new generation of compact tokamaks is currently under study, i.e. ARC (Sorbon et al., 2014) and SPARC (Greenwald et al., 2018), that opens an attractive path for fusion energy development through high-field (>20 T) and small-volume devices. For achieving magnetic field higher than 16 T high-temperature superconductors (HTS) are the unique enabling technology. In compact tokamaks it is expected that all coils are manufactured with HTS conductors. The long-term technological steps toward using HTS in device design will be addressed in Section “HTS superconductors.”

It is worth noting that HTS are presently used in the design of ITER current leads, which is a component of the feeders that brings current from the power supply to the magnet. In ITER the leads consist of a section of a bismuth strontium calcium copper oxide (BSCCO 2223) high temperature superconductor (HTS) and a conventional copper section connected together through an electrical joint (Ballarino et al., 2012). The joint and copper are cooled by helium gas at 50 K, the cold end of the BSCCO 2223 at 4.5 K.

Magnetic system of ITER

The magnet system of the ITER fusion reactor is presented in Fig. 1 as an example of SC magnet system in a tokamak (Mitchell et al., 2008, 2012).

The Toroidal Field (TF) coils, distributed toroidally around the vacuum vessel, produce the dominant component of the confining magnetic field. In ITER there are 18 D-shaped Nb_3Sn coils measuring $9\text{ m} \times 16.5\text{ m}$, capable of producing 41 GJ of magnetic energy and a field of 5.3 T on the plasma center. In order to achieve such a high value of magnetic field over a major radius of 6.2 m, the current flowing in the TF conductors is 68 kA, which produces a maximum magnetic field of 11.8 T on the coil. As reported in Fig. 2 (left side), each TF magnet is made by a winding pack (WP), that contains the superconducting cables, enclosed by a stainless steel casing, which is the principal structural component of the magnet system. The weight of each TF coil is about 360 t, distributed among its components: 110 t for the WP, 190 t for the casing and 60 t for pre-compression systems, shear keys and bolts.

The Central Solenoid (CS) coils produce inductive flux to ramp up and maintain the plasma current and contribute to shape the plasma. In ITER the CS is made of six independent Nb_3Sn modules, with a total height of 13 m, a diameter of 4 m and a total weight of about 1000 t. The maximum operating current in the CS modules is 45 kA, leading to a maximum field of 13 T reached in the center of the stacked modules and a magnetic energy stored in CS coils of 7 GJ. This generates a sufficient magnetic flux to allow initiating and sustaining a plasma current of 15 MA for about 400 s under burning plasma conditions.

The Poloidal Field (PF) coils control the radial position of the plasma, contributing to its stability by maintaining the clearance between the plasma and the first wall. PF magnets are also essential elements of the slow vertical position control and of the plasma shaping capability. In ITER the six annular NbTi PF coils are located outside of the TF magnet structure. They are designed to operate at a maximum current of 45 kA, which generates a maximum magnetic field of 6 T on the coils and a total magnetic energy of 4 GJ. The largest magnet has a diameter of 24 m; the heaviest is 400 t, with a total combined weight of 2163 t.

All of the coils mentioned above are toroidally symmetric. There are also asymmetric coils, namely the superconducting Correction Coils (CCs), with the role of correcting magnetic field errors (Hugué, 1997) potentially caused by deviations from target dimensions during manufacture and imprecise positioning during assembly. The concept of magnetic error fields and their impact

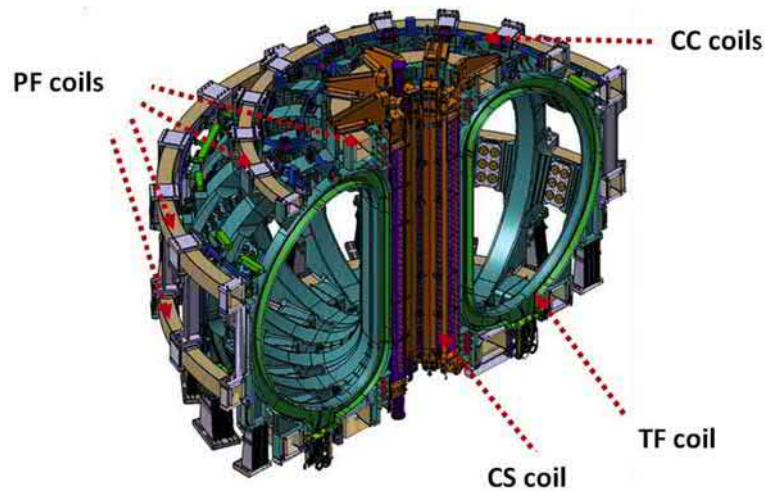


Fig. 1 ITER Magnet system. Courtesy of Mitchell N, Bessette D, Gallix R et al. (2008). The ITER magnet system. *IEEE Transactions on Applied Superconductivity* 18: 435–440.

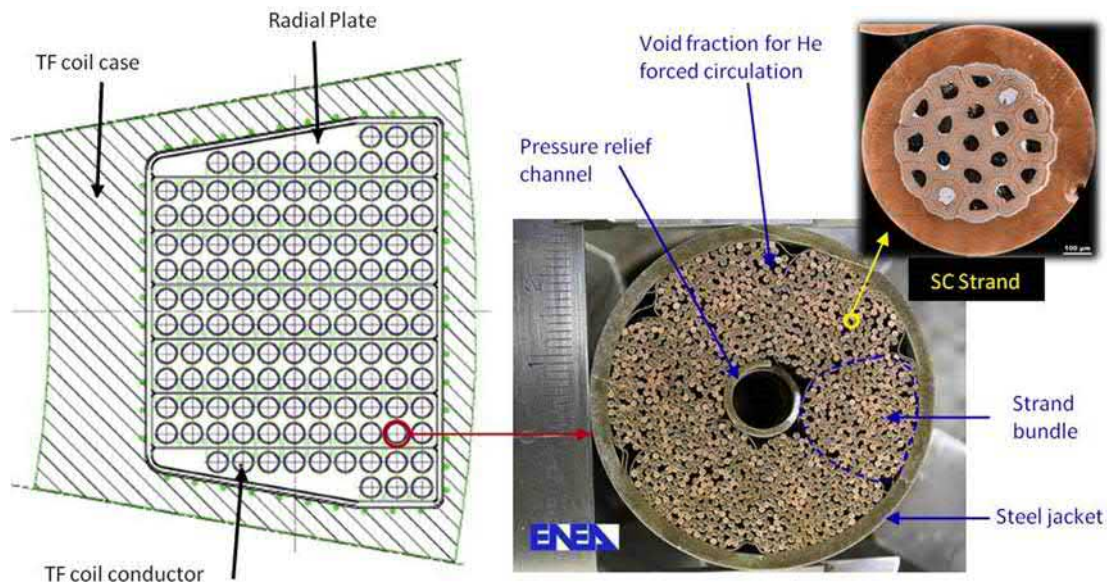


Fig. 2 Cross section of a TF coil (left side); layout of the ITER TF conductor with the detail of a SC strand (right side).

on plasma performance is discussed elsewhere (Lao, 2020). In ITER 18 CCs, divided into three independent sets of six coils each, will be installed around the TF magnets.

Superconducting magnet technology

Superconducting magnet technology is developed at industrial level thanks to the realization of large scale projects, such as the fusion reactors mentioned in Section “**Superconducting magnets in fusion devices,**” and high-energy accelerators, like the Large Hadron Collider (LHC) realized at CERN. However, nuclear fusion devices have certain criticalities that led to the development of specific technologies, as reported below.

Cable-in-conduit conductors

The cable-in-conduit conductor (CICC) concept is based on the observation that the cooling efficiency in a conductor is improved if S-He (at about 4.5 K) flows in the turbulent regime in direct contact with the SC strands (Muzzi et al., 2015). In nuclear fusion reactors SC magnets are subject to several sources of heat loads, such as the nuclear heat load and the thermal load coming from gravity supports. In order to maintain good magnet stability, large heat removal capabilities are required and this explains why the design of the WPs for fusion reactors mostly relies on CICCs.

In a CICC, superconducting and copper wires are cabled in a multi-stage, transposed, and twisted structure. The cable is then inserted and compacted inside a stainless steel (SS) jacket, maintaining a void fraction that guarantees the forced circulation of the refrigerating S-He. Depending on conductor size and hydraulic length, one or more spirals exchanging He with the strand bundles might be introduced in the cable as pressure relief channels.

The TF conductor of ITER is shown in Fig. 2, right side, as an example. It includes 900 Nb₃Sn SC strands and 522 Cu strands, whose role is to prevent the burning of the SC wires during the quench from the superconducting to the normal state. Each SC strand is made of thousands of SC filaments in order to reduce the AC losses that are generated in the presence of variable magnetic fields (Wilson, 1987).

The use of Nb₃Sn as superconductor allows the target field of 11.8 T for ITER TF coils to be achieved. However, in contrast to NbTi, its performance is sensitive to mechanical loads (Muzzi et al., 2012) and suffers degradation because the material is brittle after the heat treatment required for Nb₃Sn formation. In fact, during the magnet operation, the SC cable inside the jacket is mechanically loaded by high Lorentz forces of about 800 kN/m. This translates into a combination of bending strain and contact stress on the wires (Mitchell et al., 2013) that might lead to the fracture of Nb₃Sn filaments, especially in case of tensile stresses. The irreversible degradation of the SC performance occurs during repeated electromagnetic or thermal loading cycles, until stabilization is reached. The effect was observed on most of the TF short samples tested in the SULTAN facility (Breschi et al., 2012) by applying 1000 electromagnetic cycles and various warm-up/cool-down cycles.

Increasing the stiffness of the cable and reducing the void fraction can mitigate the issue, as demonstrated by qualification tests of ITER CS conductors (Martovetsky et al., 2017). However, degradation remains a critical aspect of the conductor design, that must be solved for guaranteeing the reliability of future power plants.

Radial plates

A specific feature of the ITER TF coil design is the use of thick austenitic SS radial plates (RP) to embed the insulated conductors in machined groves (Hugué, 1997). ITER is the only fusion device to date that has adopted this solution, with the aim to transfer the Lorentz forces acting on each conductor directly to the case. The RP prevent the accumulation of forces on the conductor and its turn insulation, reducing the probability of insulation degradation due to mechanical fatigue. The role of the SS jacket around the cable is therefore limited to support the local conductor forces and provide a containment for the forced flow helium. In order to manufacture the ITER TF coil (Mitchell et al., 2012), the conductor is wound in “double pancakes” (DP) making use of a mold. After the heat treatment, the mold is removed and the DP is transferred to radial plate grooves with an accuracy of a few millimeters. These operations are very complex and require special equipment to handle and move the different components. All technical issues were successfully solved in the manufacturing phase, but resulted in an overall cost of the RP corresponding to 1/3 of the cost of all TF coils.

Mechanical structures

Another specific aspect of the magnet system for fusion reactors is the need for mechanical structures to sustain the enormous forces acting on the coils. For example, for ITER the centering force per TF coil is 403 MN and the vertical force on one half of a TF coil is 202 MN. The maximum vertical force per coil is 327 MN and 160 MN for the CS and PF magnets, respectively (Mitchell et al., 2012).

For supporting the out-of-plane forces, in the inboard region the straight legs are wedged all along their side walls and the curved parts are linked by six pre-compression rings. In order to counteract the twisting moment of the TF coils due to the out-of-plane forces, the outer legs have been connected by the outer intercoil structures (OIS), which form toroidal shear belts around the magnets. The overall force balance in the TF set is maintained by the TF cases, including the central cylinder formed by the wedging, the inner pre-compression rings and the OIS.

The CS assembly includes the stack of six modules and a preload structure that compresses the CS magnet vertically during the assembly and the cool-down of the machine. The preload structure is made by a set of tie-plates located at the inner and outer diameters of the CS stack. The CS conductor has a very thick square jacket which is self-supporting against the radial and vertical magnetic loads.

Sliding supports allowing radial displacements are attached to the TF coil cases to support the PF coils. The structural support of each PF magnet is guaranteed by the thick square jacket of the PF conductor.

The future of magnet technology

The development of nuclear fusion power plants for supplying energy to the grid requires a consistent increase of the plasma volume or of the value of the magnetic field on the plasma compared to ITER, which is currently the largest reactor under construction. New challenges are expected for superconducting magnets in the design of the DEMOnstration fusion power plants. In the following sections the strategies adopted by the European DEMO are presented (Corato et al., 2018; Sedlak et al., 2020).

Improving performances of LTS magnets

The European DEMO design (Federici et al., 2019), whose major radius is about 9 m, is conceived to provide a plasma burn with a duration of 2 h. Scaling-up the ITER magnet design to DEMO is not impossible, but two relevant aspects must be improved. Firstly, the degradation of the performance of the conductor under electromagnetic cycles is not acceptable in a power plant that is expected to efficiently operate for tens of years. Secondly, the cost of magnet manufacturing must be reduced as much as possible to make a fusion power plant an economically viable source of energy.

In the European DEMO design the issue of degradation was successfully addressed using rectangular samples with a low void fraction and an increased stiffness of the cable. A first set of conductors has been realized using the R&W technology for Nb₃Sn superconductor (Sedlak et al., 2019), while a second type of superconductor has been manufactured with the W&R approach (Muzzi et al., 2017). For both types no degradation was observed after 1000 electromagnetic cycles and various thermal cycles.

The R&W technology presents an additional feature that might also reduce the cost of TF magnets compared to the ITER experience. Since the reaction HT is performed before the jacket is applied, the thermal strain applied to the cable during the cool-down is lower and this reduces the effective compressive strain acting on the superconductor. Since the performance of Nb₃Sn improves with lower strain (Muzzi et al., 2012), the net consequence is a reduction in the number of the required SC strands, compared to that of a W&R conductor carrying the same current.

Another strategy for optimizing the costs is to wind the conductor in (radially distributed) layers instead of (poloidal cross-section) pancakes. In fact, this approach allows grading of both SC and SS cross-sections in the different layers, increasing the amount of SC material and decreasing the amount of SS on the inner layers (i.e. facing the plasma), with the opposite trend on the outer layers.

HTS superconductors

The debate on the possibility of building a fusion power plant with HTS coils is very active (Whyte et al., 2016; Bruzzone et al., 2018) but the technology for large HTS magnets is not yet mature and requires an engineering demonstration, that includes the development of the following elements: high current HTS conductors, magnet protection devices and methods specific to HTS magnets, advanced radiation-tolerant insulating materials and coil fabrication technology.

In addition, to achieve demountable TF coils in a compact tokamak, such as ARC (Sorbom et al., 2014), the first objective is the development of the engineering design of WP joints for demountable coils. The key points to be solved in the design are the remote handling operations, the demounting activity, and the subsequent restoration of the cable connection with continuous insulation and structural support. All steps require quality and assurance procedures and experimental validation.

At present, an additional objection to the use of HTS is the high cost, one order of magnitude higher than that of LTS, which makes large HTS TF coils extremely expensive. However, in the European DEMO design a hybrid solution (HTS-Nb₃Sn-NbTi) based on layer winding has been proposed for the CS coil (Wesche et al., 2019). Compared to an LTS CS coil with the same outer radius, the hybrid configuration is more complex from the manufacturing point of view, but provides a more cost-effective design and a slightly higher magnetic flux.

It is worth noting that in high-field applications even HTS conductors need to operate at low temperatures, below 25 K, because the critical current density at high field dramatically drops above 25 K.

Plasma facing components

Plasma power exhaust and divertor protection

The description of the divertor operating concept and its underlying physics is discussed elsewhere (Reiter, 2020).

One of the crucial points in the dimensioning of a power producing fusion plant remains the size of the device and the amount of power that can be reliably produced and controlled within it. This heavily depends, among other things, on the heat load that can be tolerated by the divertor under normal and off-normal operation (Federici et al., 2019).

A Demonstration Power reactor to follow ITER must typically exhaust ~500 MW (~150 MW in ITER) of heating power (fusion alpha particle power and auxiliary heating power). If 40% of the exhaust power is radiated inside the magnetic separatrix, 300 MW will flow into the SOL. In order to achieve an acceptable heat load on the divertor targets, the latter are inclined at a shallow angle with respect to the magnetic field lines and located in a region near the separatrix X-point (magnetic null point) with significant magnetic flux expansion. In this way, the wetted area of the divertor targets in DEMO can be increased up to 1–2 m². Thus, if all the heat entering the SOL ultimately arrived at the divertor target (attached divertor regime), the power load would be above the capability of the highest performance state-of-the-art high-heat-flux technologies for steady state power load based, for example, on water-cooled copper alloys. To further reduce the heat load, part of the power flowing into the SOL has to be radiated in the divertor, leading to the so-called partially detached regime. This requires plasma temperatures in the proximity of the divertor target below 10 eV (Reiter, 2020).

Low temperature, detached divertor conditions also reduce the erosion of the divertor armor. The main erosion mechanism in the divertor is physical sputtering by plasma and impurity ions. These impinging particles transfer energy to the atoms of the armor materials. If the transferred energy is large enough, the target atoms can overcome the surface binding energy and leave the surface. The plasma temperature in front of the target defines the energy of the impinging particles by their Maxwell distribution and by the additional acceleration that the charged particles experience in the so-called plasma sheath in front of the surface. For plasma temperatures below 5 eV, physical sputtering approaches zero for tungsten as an armor material.

Another important function of the divertor is particle control, i.e. to provide adequate pumping capability to exhaust the neutralized gas, most notably the He ash, as well as to retain eroded impurities such that they will not enter the main plasma, which reduces fusion performance and can lead to plasma instabilities.

Siccinio et al. (2019) discusses the criteria to be employed in the preliminary phases of a tokamak fusion reactor dimensioning to ensure the integrity of the divertor for sufficiently long operating times, without, at the same time, compromising the stability of the

core plasma or the fusion power generation. It is shown that two high-level requirements must be fulfilled: (1) the concentration of seeded impurities in the SOL has to be lower than some critical value in order not to compromise the fusion plasma performance or stability; and (2) the design of the divertor target must be able to withstand accidental re-attachment of the plasma for a sufficiently long time to recover detachment or to ensure a reliable, controlled termination of the plasma discharge. The main conclusion of Kemp et al. (2018) is that, for a given fusion power level, the simultaneous fulfillment of both requirements limits the viable device size in terms of both major radius, R , and toroidal magnetic field, B .

Several solutions for the heat exhaust in DEMO are currently being explored in Europe 2020 (Turnyanskiy et al., 2015):

1. Baseline divertor solution—a combination of radiative cooling and detachment;
2. Innovative magnetic divertor configurations to achieve higher flux expansion thereby spreading the heat over a larger area, or to achieve longer divertor connection lengths and larger divertor radiated power;
3. Alternative solutions (e.g. liquid metals) that could exhaust higher heat loads.

However, the physics basis and the technology readiness level of solutions (2) and (3) remain low and it is to be expected that their relevance in terms of design constraints arising from DEMO integration and operation issues will require deeper scrutiny, if they are ultimately proved to be effective. A review of liquid metals as an alternative solution for the power exhaust of future fusion devices is provided by Coenen et al. (2014). The most attractive candidates under investigation are Sn and Li, and an R&D program is in place in Europe to investigate the power handling of realistic engineering solutions such as the capillary porous system (CPS). Critical issues that need to be resolved include, resilience to transients, design integration, vaporization/redeposition and tritium retention.

Although subjected to lower power and particle loads than the divertor targets, the first wall protecting the breeding blanket will also receive power from radiation and particles, and will undergo erosion. Robust engineering solutions are required for the components that directly surround the plasma to ensure that they can withstand the thermal and mechanical loads during normal and off-normal operation and that they contribute to protection of the outer components, especially the vacuum vessel and the magnets, from the effects of neutrons.

Considerations here are limited to the portion of the PFC surface that protects the main chamber (the so-called first-wall), which, for both JET and ITER, is made of beryllium. The strategy and the criteria adopted to design the PFCs of DEMO and ITER are substantially different. The problems associated with the design of the first-wall at JET and ITER are described by Federici et al. (2016). One of the main differences between ITER and DEMO is that for the latter to achieve efficient breeding (and therefore tritium self-sufficiency), the first-wall must be relatively thin (several mm) to avoid adversely depleting the neutron flux entering the breeding regions. In addition, conversion of this energy at adequate thermodynamic efficiencies requires that the coolants be at high temperature and pressure. This has a strong influence on reactor engineering. The ITER design uses a relatively thick Be-clad PFC actively cooled with water at low temperature.

The power handling capability of first wall in DEMO (i.e. the non-breeding part of the breeding blanket) is rather limited (~ 1 – 1.5 MW/m² for a water cooled concept and < 1 MW/m² for helium cooling) due to the requirement to use radiation-hard materials such as EUROFER steel for the structures and the cooling pipes of the breeding blanket, and high-temperature, high-pressure coolant for efficient energy conversion. Operation within these heat-flux limits is deemed to be achievable in nominal conditions in the present DEMO blanket designs, but the occurrence of plasma transients that may severely damage the first wall or even lead to a breach of the cooling pipes and a consequent loss of coolant remains a significant issue. Preliminary results of the R&D launched to address this problem are described by Maviglia et al. (2018). Tungsten foams are being investigated for use as armor for the protection limiters in the EU DEMO design (see De Luca et al., 2019). These limiters, in contrast to the plasma start-up limiters, will see very low heat loads during normal operation, but very high off-normal heat transient heat loads, and the benefit of a low thermal conductivity armor is the prevention of damage to the underlying cooling structures.

Design/engineering challenges for reactor application

Thermal loads

There are two major limitations to peak plasma heat exhaust.

- The continuous flow of power to the divertor plates and nearby surfaces that, for state-of-the-art technology with water cooling in a tokamak environment, is limited to 20 MW/m² peak heat-flux.
- The transient peak heat-flux that can be tolerated for a short time (< 1 ms) before substantial ablation and melting of the surface occurs; such common large transient events are Edge Localized Mode (ELMs) and disruptions (Lao, 2021). The material limit imposed by these events is of the order of ~ 40 MJ/m² s^{1/2} (Federici et al., 2003).

Accommodation of such high heat loads is very challenging and gives rise to several issues affecting the performance and lifetime of the PFCs, and influencing the choice of materials and design configurations. These issues are related to the armor and joint behavior as well as to the heat removal capability of the coolant. One of the key design drivers that has led to the selection of water as coolant for the divertor in the EU DEMO has been the much lower heat removal capability of helium-cooled components compared to water-cooling (typically 5–10 MW/m² vis-à-vis 10–20 MW/m² for water) (You, 2015a). Even more importantly, the ability of water-cooled components to absorb much larger transient heat fluxes (e.g. transition from detached to attached plasma lasting several seconds) easily exceeds the heat removal capability of helium-cooled targets, which cannot tolerate such excursions.

Nuclear loads

Neutron irradiation produces significant volumetric thermal power by gamma ray generation and causes serious material damage (crystal defects, transmutation). For example, in the EU DEMO, roughly 130 MW of fusion power is transferred to the divertor in form of nuclear heating in solid structures and in coolant. The maximum nuclear heating power density in the steel structure of the cassette body reaches roughly 5 W/cm^3 , with 6 W/cm^3 in the shielding liner heat sink (You et al., 2017).

In most metals, crystal defects (vacancies, interstitials, dislocation loops) cause hardening and embrittlement of the material when the irradiation temperature is not high enough to activate thermal recovery processes. A measure of material damage due to crystal defects is the average number of lattice displacement per atom (dpa). In the case of the DEMO divertor, the target armor (tungsten) and heat sink (copper) is predicted to experience 2 dpa and 7 dpa per full power year (fpy), respectively. The steel structure of the cassette body experiences damage of approximately 2 dpa per fpy (You et al., 2017).

Helium bubbles produced by nuclear transmutation are potentially the most critical cause of embrittlement. In a DEMO divertor target, a maximum helium production rate of up to 2 appm and 60 appm is foreseen for the target armor (tungsten) and heat sink (copper), respectively.

Mechanical loads

Under the thermal loads, a temperature gradient develops between heat sources and heat sinks (cooling circuit). In a divertor target, a steep temperature gradient builds up during stationary operation ($\sim 80^\circ\text{C/mm}$) and a slow transients ($\sim 200^\circ\text{C/mm}$). There will also be a temperature gradient along the longitudinal axis of a target element due to the non-uniform heat flux footprint. Such strong temperature gradients produce thermal stresses due to restraint. Additionally, significant thermal stresses (several 100 MPa) can be produced by a thermal strain mismatch between dissimilar joined materials (armor and heat sink) (Li et al., 2014; Li and You, 2016). The thermal stresses may cause material damage of different types leading to structural failure. Embrittled materials, in particular, are highly vulnerable to cracking and tend to exhibit a reduced strain resilience.

Design concepts

A vigorous R&D program on PFCs engineering is underway in Europe, as part of the DEMO Development Program (Federici et al., 2014) benefitting from lessons learned from ITER (see, for example Merola et al., 2010). This includes: (i) the design of the divertor cassette body, considering a number of variant layouts; and (ii) the development of several candidate divertor target concepts, including fabrication trials and high-heat flux tests (You et al., 2018). The crucial design requirements for the target are to ensure sufficient margins for power handling, including slow thermal transient events, and to accommodate the relatively high neutron irradiation dose expected for a DEMO divertor. The PFCs will operate in a harsh loading environment characterized by extremely severe and multiply combined loads. Cyclic loading due to pulsed operation causes material fatigue. Moreover, the DEMO divertor is required to fulfill further key functions, namely, nuclear shielding of adjacent vacuum vessel and superconducting magnets, and guiding gas flow for particle exhaust.

Divertor cassette system

In the case of a tokamak-type fusion device, the divertor consists of many discrete cassettes arrayed along the toroidal orientation. The European DEMO will have 48 divertor cassettes (ITER has 54 cassettes). Each cassette consists of two vertical targets (to block and baffle SOL particles on the outboard and inboard side), cassette body (to hold targets and to route gas flow), shielding liner or reflector plates (to protect the underlying structure) and a cooling circuit pipework (to feed coolant) (Marzullo et al., 2019). Fig. 3 shows a typical cassette system of the European DEMO divertor. The big duct located in the cassette body (left figure) is an opening for fostering gas flow toward a pumping port. This pumping duct is covered by a shielding liner (with a vertical gap) to protect the underlying vacuum vessel from neutron streaming.

Individual cassettes are deployed by remote handling operations via the lower ports of the vacuum vessel and distributed toroidally. During fusion operation, a strong radial inward magnetic force is generated in the cassette body (ferromagnetic steel) and exploited for fixing the cassette, which is attached to the inboard vessel.

The structural material of the cassette body, shielding liner and reflector plates is reduced activation EUROFER97 steel. It is ferritic-martensitic 9Cr steel where W and Ta are used as carbide-forming alloying elements. The upper service temperature limit is about 550°C , above which tensile strength begins to drop rapidly. The ‘desired’ lower service temperature limit is around 300°C , below which the steel undergoes a transition to non-ductile state after neutron irradiation (Tavassoli et al., 2004). The question whether a lower operation temperature, below 300°C , would be applicable for EUROFER97 depends on stress intensity and the applied structural design criteria (Mazzone et al., 2017).

A critical design element in divertor engineering is the target design. The baseline design of the DEMO divertor target mostly inherited the preceding design of the ITER divertor target, which is a so-called tungsten mono-block type (You et al., 2018) (see Fig. 4). The monoblock type design was firstly developed for the carbon fiber composite plasma facing material of the NET and ITER reactor designs (Toschi et al., 1988; Janeschitz et al., 1995) and it was later adapted to beryllium and tungsten armors. In this design, the armor surrounds the heat sink, yielding a particularly robust design, in particular for a high number of high heat flux cycles (Cardella et al., 1991, 1993).

The structural part of the target is the embedded cooling pipe (heat sink). Precipitation-hardened copper alloy (CuCrZr) is the baseline structural material, while tungsten wire-reinforced copper composite is considered as a back-up option (You et al., 2018). The ideal service temperature range proposed for irradiated CuCrZr ranges roughly from 250°C to 300°C with some margins (You, 2015b). The lower limit is determined by thermal recovery of ductility and the upper limit by the onset of thermal

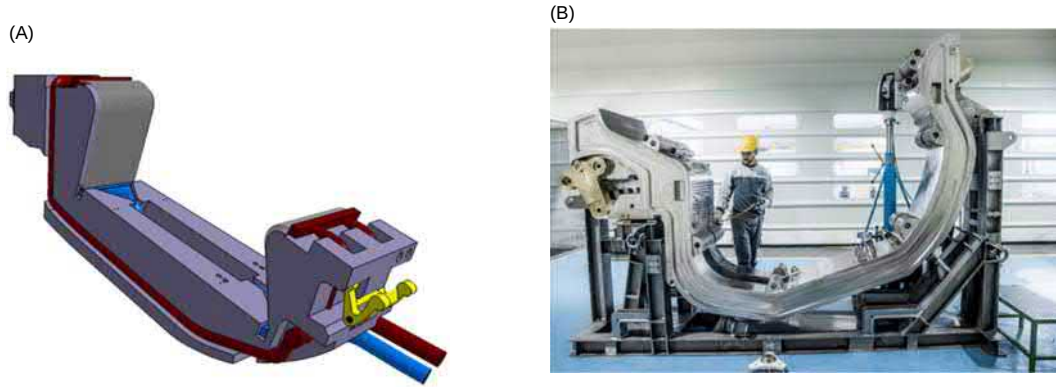


Fig. 3 (A) Cassette system of the European DEMO divertor; (B) Full scale divertor cassette body manufactured for ITER. Source: ITER IO.



Fig. 4 A small mock-up showing the baseline design of the DEMO divertor target.

softening and irradiation-assisted creep. In recent design studies of target PFCs, the operation temperature range of CuCrZr pipe had to be extended (130 °C–430 °C) far beyond the ideal service limits (You et al., 2018). Dedicated design rules should be applied to cover the non-ductile design regime. Using a copper-base material for the target heat sink is deemed indispensable, as other relevant structural materials (e.g. steel) have a thermal conductivity that is too low to be used for high heat flux (HHF) applications.

Tungsten is used as plasma-facing armor (blocks) because tungsten possesses several unique, outstanding properties such as the highest melting point (3422 °C), very low sputtering yield, extremely low hydrogen solubility, relatively high thermal conductivity and high-temperature strength, and high thermal shock resistance, etc. A major drawback is its brittle nature at lower temperatures and strong embrittlement by irradiation. Under normal operation (10 MW/m²), tungsten armor reaches a maximum temperature of roughly 1100 °C at the surface.

Divertor cooling scheme

For the sake of longevity of the targets, early failure of the structural material (pipe) due to premature damage by coolant dry-out (or film boiling) should be avoided. To this end, a valid cooling condition needs to be defined in such a way that, at the region of the plasma strike point, the maximum wall heat flux of the cooling pipe does not reach the critical heat flux (CHF). There should be a sufficient margin (40%) to guarantee that either single-phase flow or subcooled nucleate boiling is maintained even during slow (~10 s) thermal power transient events where the heat load increases up to 20 MW/m² and the armor temperature reaches up to 2200 °C.

A similar rule is applied to the cassette body and the shielding liner. Considering the significantly higher nuclear loads and irradiation damage dose expected in DEMO than in ITER, two separate cooling circuits, each with a different thermo-hydraulic condition, are employed, as illustrated in Fig. 5, where the inlet water coolant temperature (with predicted temperature rise) and inlet pressure for each circuit is indicated.

HHF technologies

State-of-the-art technologies for divertor targets

In addition to the ITER-like target technology currently favored for the design of the DEMO divertor in Europe, (albeit with a reduced dimension), advanced technologies have been developed to improve structural reliability and material performance under

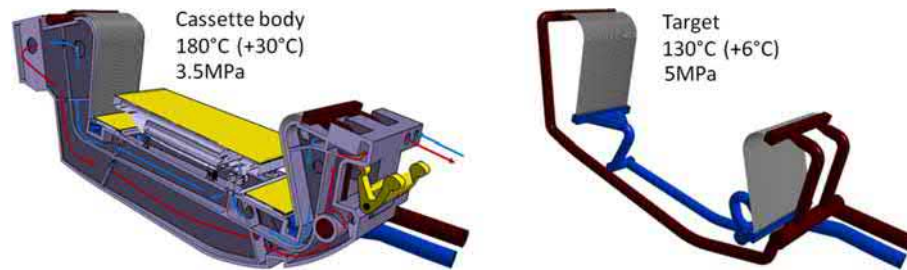


Fig. 5 Cooling circuits of DEMO divertor (left: cassette body, right: targets) and inlet coolant conditions.

combined HHF and irradiation loads. **Fig. 6** shows four selected examples of novel alternative options (together with the current baseline), where the essential features of design or material are highlighted. The first four cases refer to a mono-block type structure while the last one is a flat-tile type. Fabrication technologies are now quite mature for all cases and have demonstrated a good quality of production, as confirmed by non-destructive inspection (Visca et al., 2018).

HHF tests

All monoblock-type HHF technologies have been verified in an extensive HHF testing campaign using small-scale mock-ups at a neutral beam (hydrogen) test facility, exhibiting an excellent performance at 20 MW/m² (500–1000 pulses, 130 °C coolant), 25 MW/m² (100 pulses, 20 °C coolant), up to 32 MW/m² (5 pulses, 20 °C coolant) without any discernible major failure or damage (Greuner et al., 2019, 2020; Dose et al., 2019).

Fig. 7 shows in-situ infrared pyrometer thermographic images displaying the surface temperature of four different variants of tungsten monoblock-type target mock-up (shown in **Fig. 6**) in the course of HHF tests conducted at 20 MW/m² with hot coolant (130 °C). A distinct increase of surface temperature or occurrence of a hot spot may reveal a site of impaired heat conduction indicating material failure (mostly at a joining interface). The images exhibit no significant evolution of temperature for all target designs, demonstrating a reliable HHF performance at the targeted (i.e. specified for design) HHF load (note that the minor changes in color shading is attributed to the changing surface emissivity due to surface layer modification.). **Fig. 7** shows the photographs of two ITER-like target mock-ups, each with a different tungsten material, after an overload screening test (5 cycles at 32 MW/m²) followed by an HHF fatigue test (25 MW/m², 20 °C coolant). All armor surfaces were intact, exhibiting neither discernible cracks nor severe damage.

Vacuum vessel

Functions and configuration

In a tokamak, the vacuum vessel (VV) is a D-shaped toroidal pressure-retaining vessel with ports allowing access to the plasma. It is enclosed by the toroidal field (TF) and poloidal field (PF) coils, which form a cage-like structure whose interspaces are used to integrate the VV ports.

Vacuum and confinement: The VV maintains the ultra-high vacuum ($< 10^{-5}$ Pa) required for plasma operation. It is made of stainless steel with surfaces finished with stringent cleanliness requirements. Gases and impurities adsorbed on the VV inner surfaces after venting, during in-vessel maintenance, or after a vacuum leak, need to be desorbed to achieve the required quality and purity of the vacuum. Hence the VV is heated before pump-down to a temperature of 150–200 °C (Härtl et al., 2019). In devices subject to nuclear regulation, radioactive dust and tritium are confined within the primary vacuum of the VV and the structure must therefore be designed according to nuclear pressure vessel codes when pressurized water is used as coolant (Choi et al., 2014). These require a component reliability so high that failure is practically excluded and hence rigorous qualification, as well as in-service inspection, is required, as discussed below.

Neutron shielding: Fusion devices with copper coils typically do not require the VV to reduce the neutron flux. ASDEX Upgrade has a single-walled VV (Steibl et al., 2003). In JET the interspace of the double-walled VV is not filled with water, a neutron moderator, but is used as a secondary vacuum (Duesing, 1987). In fusion devices with superconducting magnets, however, the VV must contribute to the shielding of neutrons to limit the neutron heating of the magnets operated at cryogenic temperatures. To reduce the distance between the plasma and the TF conductor, and hence maximize the high field region accessible to the plasma, the space occupied by the VV (and dedicated to reducing the neutron flux on the inboard side) must be minimized (Bachmann et al., 2016). In devices with high lifetime neutron fluence, such as ITER (0.3 MWyr/m² Juárez et al., 2018) or DEMO (approx. 6 MWyr/m² Fed-erici et al., 2019), the neutron flux must be further reduced, in particular on the outboard side, to avoid excessive activation of the components inside the cryostat and hence to allow the access of personnel in exceptional situations requiring unplanned maintenance. For the VV to shield neutrons efficiently a double-wall steel structure, filled with liquid cooling water acting as neutron moderator, has proven to be most suitable (**Fig. 8**). Additional plates of borated steel are stacked in the interspace to reach a favorable mixture of steel and water of about 60/40% by volume (Loughlin et al., 2011). The significant weight of the ITER VV associated

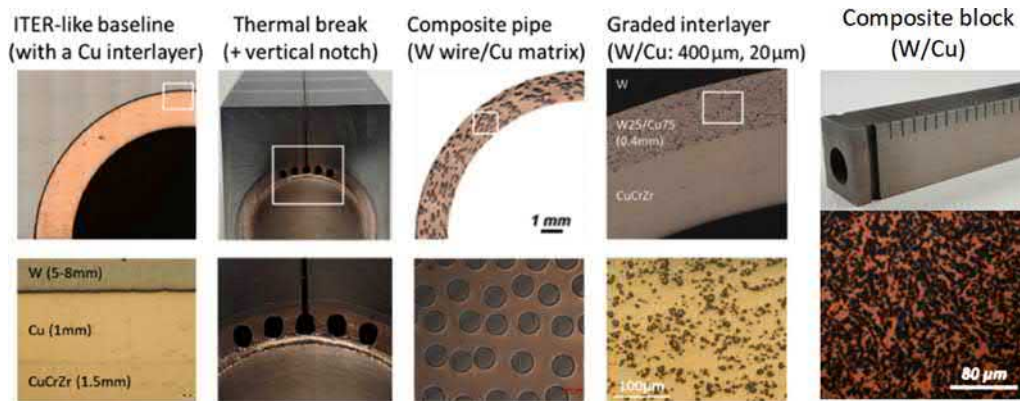


Fig. 6 Selected examples of novel alternative target technology options.

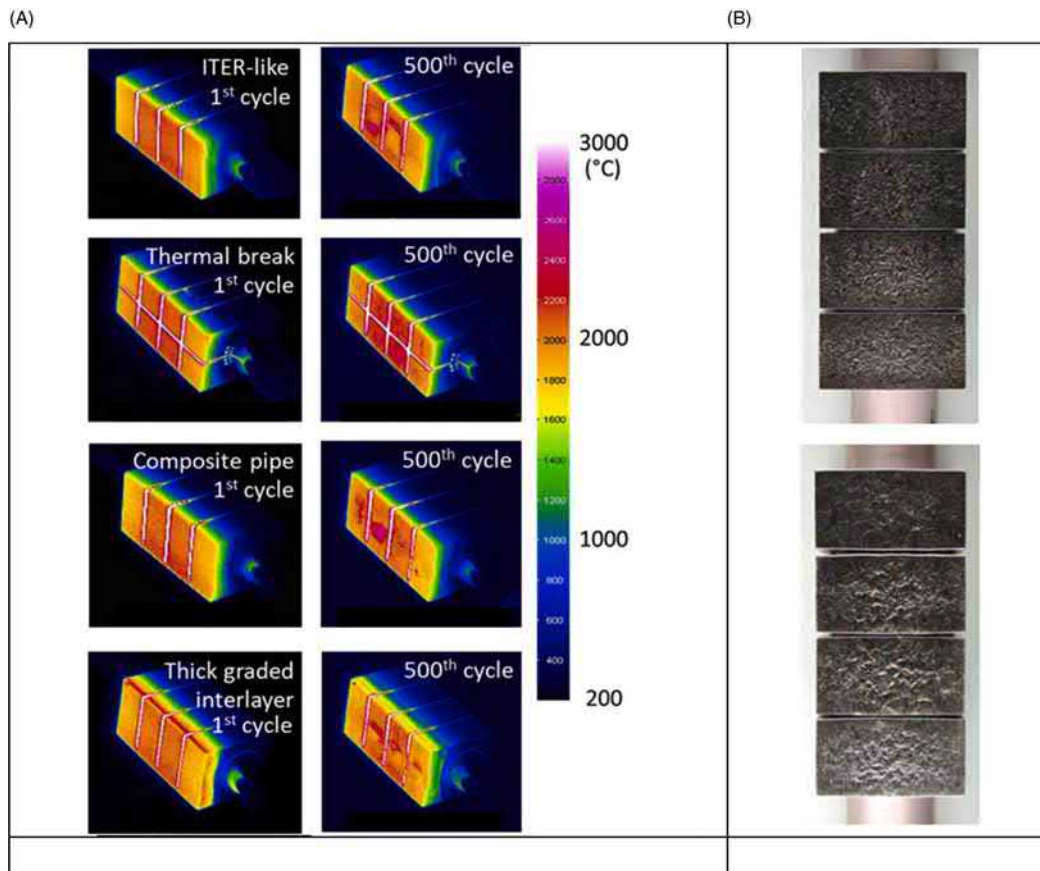


Fig. 7 (A) In situ infrared pyrometer thermographic images displaying the surface temperature of four different variants of tungsten monoblock-type target mock-up in the course of HHF fatigue tests at 20 MW/m^2 with hot coolant (130°C). (B) Photographs of two ITER-like target mock-ups, each with a different tungsten material (lower: AT&M, upper: ALMT), after an overload screening test at 32 MW/m^2 followed by an HHF fatigue test at 25 MW/m^2 (You et al., 2020).

with this design approach is shown in Table 1. Note that in KSTAR (a non-nuclear device) the cooling water is also borated to further enhance the neutron absorption (Kim et al., 2008).

Mechanical integration: In order to avoid high thermal stresses due to differential thermal expansion, different high stress interactions and thermal conduction from the VV to the cryogenic coils (Farfaletti-Casali et al., 1984), the VV is supported independently of the magnets. Moreover, in nuclear devices such as ITER, the safety-classified VV cannot be supported by the non-safety-classified magnet system.

Precision: The VV supports the in-vessel components (IVCs) and hence the plasma-facing components, mainly the blanket first wall (FW) and the divertor targets. The VV has stringent manufacturing and assembly requirements. Leading edges of the PFCs

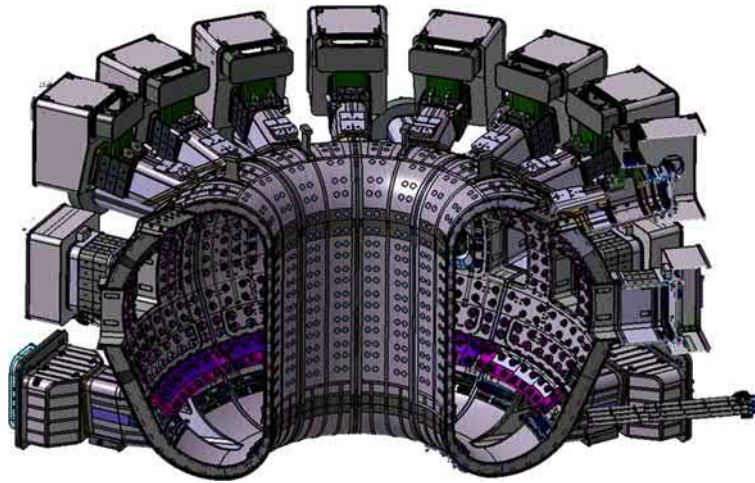


Fig. 8 View of four sectors of the assembled ITER VV with upper, equatorial and lower ports and IVC support structures (pink) on the inner shell. Support structures below the lower ports and port plugs not shown. The splice plates used to assemble the sectors and some of the ports are highlighted in dark gray. From ITER organisation.

Table 1 Major dimensions and weight of the ITER VV.

Major dimensions		Weight (tons)	
Outer diameter (m)	29.0	Main vessel	1611
Height (m)	11.3	Shielding	1733
Double wall thickness (m)	0.34–0.75	Ports	1781
Interior surface (m ²)	850	Supports	111
Interior volume (m ³)	1600	Total	5236

would be subject to excessive heat loads due to charged particles escaping from the plasma confinement (Maviglia et al., 2017). It is therefore necessary to align the PFCs to the toroidal field with a precision that, for components of this size, is uncommonly tight: in the EAST tokamak the alignment of the PFCs to the magnetic axis is less than ± 0.5 mm (Wenge et al., 2005). The ITER FW has a prescribed alignment tolerance of ± 5 –10 mm (Wilson et al., 2015) and the ITER divertor targets of ± 2 mm (Escourbiac et al., 2012). Given the foreseen integration of protection limiters the alignment requirement of the DEMO BB FW might be somewhat relaxed but still high, currently estimated as ± 5 –10 mm (Vizvary et al., 2019). The customization of the individual IVC support structures can compensate misalignments of the VV inner surface to some degree (Wilson et al., 2015). However, for the successful installation and alignment of the PFCs, small VV tolerances are required, e.g. ± 3 mm in JET (Duesing, 1987), ± 5 mm in JT60SA (Shirai et al., 2017), and ± 10 mm in ITER (based on large-scale fabrication studies Nakahira et al., 2008).

Electromagnetic integration: In tokamaks, to enable plasma operation, the currents in the PF and CS coils are varied for three different purposes: (i) to provide toroidal magnetic flux to break down the hydrogen gas into a plasma and initiate the discharge, (ii) to drive the plasma toroidal current, and (iii) to actively control the plasma shape and the plasma radial and vertical positions. The VV is subject to the resultant magnetic field variations as it is a toroidally continuous structure. To reduce its EM shielding effect during plasma breakdown, several existing tokamaks integrate bellows or electrical breaks between the VV sectors, increasing the toroidal resistivity, e.g. JET (Crisanti et al., 2003), ASDEX Upgrade (Steibl et al., 2003) or JFT-2M (Mori et al., 1988). Shunts can prevent arcing at the bellows and at electrical breaks between the VV sectors (Mori et al., 1988). On the other hand, in the case of plasma vertical motion, the toroidal electrical conductivity of the VV causes currents to be induced that counteract the plasma movement, providing passive vertical stability (Albanese et al., 2004). Due to their flexibility, bellows between VV sectors prevent the VV acting as a ring structure that would be well suited to bear vertical and out-of-plane loads. Operational experience with tokamaks having vertically elongated plasmas showed that the VV's role in the passive stabilization of vertical plasma motion was an important factor in limiting the loss of vertical position control. In addition, vertical displacement events (VDEs) associated with such loss of control produced significant forces on the VV and in-vessel structures. Consequently, the occurrence of VDEs required the JET VV to be reinforced (Buzio, 1998). With the increase of the plasma currents, e.g. toward JT-60U and ITER, the VV design approach incorporating bellows or electrical breaks was abandoned. In NET, it had been attempted to maintain the structural toroidal continuity while at the same time integrating bellows between sectors (Anon, 1993). The concept was later abandoned in ITER (Ioki et al., 1998). The fact that the use of bellows was unprecedented in nuclear containment vessels and the expected difficulties in obtaining a nuclear license was also an important consideration. Both shells of the ITER VV sectors will be fully welded at the sector field joints using splice plates (Choi et al., 2014) and the same principle is envisaged for DEMO. Due to progress in

operational techniques for establishing plasma breakdown, the use of a toroidally continuous VV is no longer considered a major issue (Lloyd et al., 1996; Ambrosino et al., 2015; de Vries and Gribov, 2019).

Material and sustainability

Vacuum vessels in existing devices are fabricated from a range of austenitic steels for their good mechanical properties and good weldability, e.g. 316 LN (EU 1.44xx steels) in KSTAR (Kim et al., 2008), JT60SA (Shirai et al., 2017), Wendelstein 7-X (Cardella et al., 2007) and ITER (Ioki et al., 2010). Nicrofer 7216 LC, which is similar to Inconel 600, was used in JET (Duesing, 1987), Inconel 625 in JT-60U and AISI 321 in T-15M (Mironiv et al., 2004). In DEMO the VV will be subject to a significant level of neutron irradiation, which introduces two additional aspects to be considered in the material selection, namely the degradation of the material properties and the material activation. These considerations relate mainly to the inner shell since the neutron flux decreases through the VV thickness, by about one order of magnitude per 10 cm (Flammini et al., 2016). The irradiation resistance of austenitic steels is judged (Stork et al., 2014) as sufficient even for fusion power plants with 30 years of operation time. The degradation of the fracture toughness of 316 LN austenitic steel is defined in the nuclear code RCC-MR as negligible up to a damage level of 2.75 dpa (Bachmann et al., 2018), which is somewhat higher than is predicted to occur in DEMO (Fischer et al., 2016). A similar radiation resistance can be expected for the higher strength austenitic steel XM-19 (used for the ITER inner divertor target body) (Pokrovsky and Fabritsiev, 2011). One drawback is the high nickel content of austenitic steels and its long-life isotopes, which make the VV a major contributor to the intermediate-level waste produced by DEMO (Gilbert et al., 2019) and hence impact on the sustainability of fusion energy, see Fig. 9.

To date, the search for a non-nickel steel for the DEMO VV has not been successful: ferritic steels, as used in fission reactor pressure vessels and experiencing a dose of approximately 0.1 dpa, seem—based on (IAEA, 2009)—to have an radiation resistance of no more than 0.1–1 dpa. On the other hand, the radiation resistance of ferritic-martensitic steels is sufficient (> 5 dpa Brachet et al., 2001) and the reduced-activation steel EUROFER (Tavassoli et al., 2004) would be a candidate material for the VV from the point of view of neutron irradiation. The material is, however, ferromagnetic and shows poor machining, poor forming and poor welding characteristics. Welding EUROFER necessitates complying with small temperature ranges for preheating (100–150 °C), intermediate layer temperature (< 280 °C), soaking against hydrogen-induced cold cracking (250–300 °C), cooling down and post-weld heat treatment at 760–800 °C to recover the material's ductility. In particular, no technical concept has been found that would allow adequate control of the temperature of the sector field-joint welds during assembly, those on the inboard being accessible only from the plasma side due to the presence of the TF coils. Based on current fabrication capabilities, the use of EUROFER does not appear technologically feasible for a complex DEMO VV sector, which will be larger than that in ITER and require numerous sequential welds, see Fig. 9. Therefore, it is foreseen to use austenitic stainless steel for the structural parts of the DEMO VV, as in ITER.

To reduce the irradiated waste associated with the DEMO VV, the following principles are envisaged: (i) In order to reduce the amount of steel used in the most exposed part, i.e. the inner shell, high-strength steels might be considered. XM-19 might be a suitable candidate but further qualification is needed. (ii) Since no welding is required for the in-wall shielding plates, non-nickel steel is foreseen for those below the inner shell and still exposed to significant neutron flux. The European Power Plant Conceptual

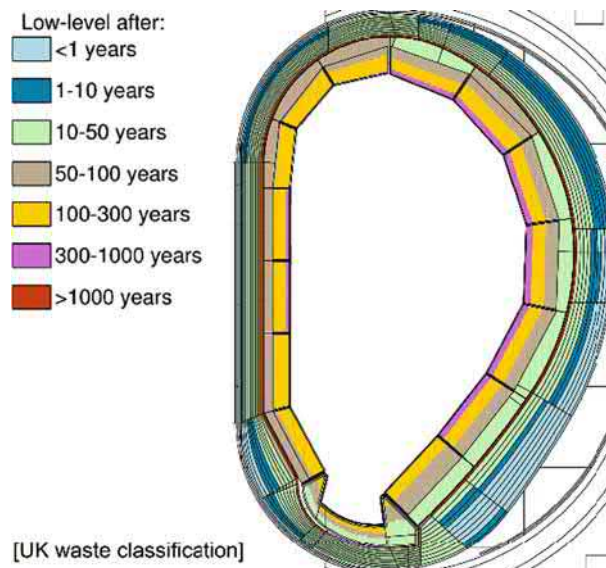


Fig. 9 Vertical cross-section through the DEMO VV and in-vessel components, showing the predicted time to acceptance as low-level waste based on the UK waste classification, (the red line represents the VV inner shell). In-wall shielding plates are assumed here to be made of non-nickel steel.

studies (Maisonnier et al., 2005) foresaw a semi-permanent EUROFER shield covering the VV. This additional in-vessel component could reduce the activation of the VV sufficiently to allow its classification entirely as low-level waste.

Fabrication

The VV is typically fabricated as individual sectors that are welded together in the tokamak pit after the installation of the TF coils. The VV ports are manufactured separately and later joined to the sectors (Utin et al., 2013). In ITER the manufacturing processes include cutting, machining and forming of parts of the shells and ribs, welding, installation of in-wall shielding plates, weld inspection and leak testing (Choi et al., 2014; Ioki et al., 2010). To reduce weld shrinkage, welding technologies are chosen with minimal heat input such as narrow gap TIG and electron beam (Dans et al., 2014). In the design of the VV, provisions need to be taken to allow for non-destructive inspection of the welds, e.g. by radiography or ultrasound testing (Martinez-Ona et al., 2013). The challenging fabrication of the ITER VV sectors and ports is already underway, see Fig. 10.

Operation

Cooling

Tokamaks with short pulses and low neutron generation did not require an active cooling system for the VV. Temperature control systems were installed mainly to allow heating of the VV for vacuum conditioning via baking. In stellarators (e.g. W7-X) and in tokamaks with high performance and long pulses (e.g. JT-60SA) the VV is provided with active cooling. In ITER and future devices with long pulses and high neutron production rates, a significant amount of heat is deposited in the VV, e.g. in DEMO about 65 MW (Bachmann et al., 2016). A cooling water system removes this heat during plasma operation and is also used to heat the VV for baking. Since the IVCs will be activated, but are not expected to be safety-classified in accident scenarios, the VV cooling system will be essential for removal of the decay heat from the reactor. In the event of a loss of off-site power, the natural circulation of the VV cooling water might be sufficient. In ITER, however, a relatively small safety pump (44 kW) will be used, powered by safety-classified emergency diesel generators.

Loading conditions

The VV of ITER and large future devices with design points similar to the European DEMO (Federici et al., 2019) must be designed to withstand a range of mechanical, thermal, electromagnetic and seismic loads, which, for ITER, are summarized in (Martinez et al., 2014). The external pressure of 1 bar as well as the accidental internal over-pressurization condition (up to 2 bar in ITER Martinez et al., 2014) are not design drivers for the vessel with the exception of some parts of the port structures and the port closure plates. Instead, the pressure required to prevent boiling of the coolant water during baking at $\sim 150\text{--}200^\circ\text{C}$ (Härtl et al., 2019) defines the number of ribs to be welded between the two shells. Due to the neutron shielding function, which requires large amounts of steel in the VV, and due to the blanket weight, in particular in devices with a full tritium-breeding blanket, the dead weight carried by the VV supports is very large (~ 8900 tons in ITER and $\sim 27,000$ tons in DEMO). Seismic loads are usually less severe than EM loads, even at the ITER site, which has a rather severe ground spectrum (Sannazzaro et al., 2009).

Plasma disruptions generate large electromagnetic forces on the VV, as well as on the IVCs (Noll et al., 1997; Hender et al., 2007). Possible consequences include significant pressure loads on the VV shells, vertical and sideways net forces, and challenging interface loads on the supports of the IVCs (Martinez et al., 2014).



Fig. 10 ITER VV sector during manufacturing at Hyundai Heavy Industry in South Korea (September 2019). Source: ITER IO.

In order to protect the TF superconductor from over-heating in the event of a quench, a TF coil fast discharge is initiated to rapidly reduce the coil current using a dump resistor. The consequent poloidal current that is induced in the vessel interacts with the magnetic field that is still present in the initial phase of the discharge. This generates a pressure load on the VV acting away from the plasma that is particularly strong on the inboard side. In devices larger than ITER, this load can become critical, since the induced current increases with the size of the TF coils and the stress level in the VV is proportional to the inboard radius of the VV, which is large in devices with long inductively-driven (i.e. by the central solenoid) plasma pulses (Kovari et al., 2016). A slower discharge of the TF coils might be necessary to mitigate this issue in future tokamaks, which would, however, require an increase of the copper fraction in the superconductor (see Section “Cable-in-conduit conductors”).

In-service inspection (ISI)

Design codes for fission reactor pressure vessels, e.g. ASME Section XI in the US and AFCEN RSE-M in France, require regular pressure and leak tests, as well as the regular inspection of all welds, at an interval of 4–10 years, or individual inspections following accidents upon request of the regulatory authority. In the same way, ISI is mandatory for fusion reactor VVs, which are defined as nuclear pressure equipment. Ultrasonic testing might be a candidate technique for weld inspection; due to the high residual gamma radiation, radiographic inspection is excluded. Both shells are fully covered by other tokamak components, which restricts the access to most VV welds: the outer shell by the thermal shield, the inner shell by the IVCs. In particular, on the inboard side the compactness of the tokamak radial build is essential for the plasma performance (Kovari et al., 2016) and the provision of space for inspection tools has a substantial cost impact. In DEMO and future fusion plants, however, regular survey of the inner shell might be possible when the IVCs reach the end of their irradiation lifetime (Federici et al., 2019) and are being replaced. The design of future fusion plants needs to incorporate access for ISI tools and must aim to reduce the number of welds in the VV, e.g. in ITER the VV inter-modular keys are machined from a forged piece (Ioki et al., 2013) and complex 3D formed parts are used for parts of the shells (Jones et al., 2007; Ioki et al., 2010). A potential solution is that some welds with poor accessibility might be exempted from ISI if they are designed with additional margins. In ITER, ISI of welds is emphasized for those welds in higher stress areas (Ioki et al., 2013).

Remote maintenance

Maintenance in fusion reactors

The reliability of in-vessel components is among the foremost considerations for the successful development and deployment of future fusion reactor systems.

The effects of thermal loads and of highly energetic neutrons on the plasma-facing components lead to a progressive degradation of their properties. This reduces the lifespan of plasma-facing components, so that they require regular replacement. Hence, fusion reactors have inherently high planned maintenance frequency requirements compared to other types of power plants.

Neutrons also lead to material transmutation, and hence the activation of the plasma-facing components. The activation levels for commercially relevant power plants will be such that human access will be precluded anywhere in the plant where activated components may be present. This is not simply limited to the confines of the bioshield: many of these components must be extracted from within the bioshield for maintenance and processing in other parts of the plant. The presence of activated erosion dust and tritium also implies that any systems, which enter the vacuum vessel, such as remote maintenance equipment, may also be contaminated.

Aside from radiation risks, fusion reactors also contain hazards from a wide range of sources, including hazardous substances (e.g. beryllium), cryogenics, high voltage, industrial machinery, etc. The combination of all these hazards means that human access must be severely restricted in many parts of the plant. Carrying out maintenance activities safely therefore requires the use of remote equipment.

The availability of a power plant is the fraction of time it is operational and producing useful power (and hence, profit) over its lifetime. It is a key driver for commercial viability, which means uptime must be maximized and maintenance durations minimized. When considering also the neutron-induced degradation and the limited safe access available, it follows that an efficient maintenance strategy, supported by robust remote maintenance technology, is mission-critical for the commercial viability of fusion power.

Maintenance strategy

Maintenance is not limited only to in-vessel components: almost all systems inside the plant will require some form of maintenance. This can include anything from statutory inspection of welds, replenishment of consumables, replacement due to wear or corrosion, calibration, repair of damaged parts, etc.

For a reactor to be able to meet all its operational, commercial and regulatory requirements over its entire lifetime, a very high level of integration is needed between component, reactor, and maintenance system designers. This can only be achieved if the maintenance strategy is incorporated from the outset in a Maintenance Management Plan (Tesini and Rolfe, 2009) and used to drive the design of the reactor and its subsystems (Raimondi, 1989a; Rolfe, 2007; Tesini and Palmer, 2008). An example of this is ITER's

maintenance classification scheme (David et al., 2009), which assists component designers in providing a first assessment of the maintenance requirements early in the design process, and in communicating these to the maintenance specialists.

Designers must consider that, during maintenance, components will be subjected to a wide range of non-operating conditions over several decades. For instance, at times they may be only partially assembled, moved to new locations, held in different positions, stored in hot/cold rooms, require aggressive decontamination, etc.

Contamination control and safety are driving considerations because maintenance activities often involve the relocation of components between different parts of the plant. This can increase the likelihood of contaminants such as radioactive dust being mobilized, so that activities must be carefully planned, and additional control measures deployed as needed.

Maintenance activities often need a variety of specialist tools and equipment, each with individual requirements. Two major considerations which affect the layout of the plant are space (requirements for transporting components, temporary storage, installing temporary contamination control barriers, access and operation of maintenance equipment, etc.) and provision of services (electricity, hydraulic power, cooling fluids, cleaning fluids, laser sources, etc.). Both are typically required close to where the maintenance activity is taking place. Retrofitting space and services into the design once the plant is built is costly and often unfeasible.

Remote maintenance technology

In locations where human access is prohibited, remote means are required to perform maintenance activities. Historically, these have involved using teleoperated robots, which are controlled by human operators from the safety of a control room.

The remote maintenance system of the Joint European Torus (JET) (Raimondi, 1989b) is the most developed and comprehensive of its type, having accumulated over 30,000 h in operation (Buckingham and Loving, 2016). It uses two snake-like booms to enter the vacuum vessel (Fig. 11, left). These move around the vessel controlled remotely by operators, carrying other subsystems and payloads such as the MASCOT telemanipulator (Fig. 11, right). The booms are reconfigurable to adapt to a wide range of missions. They measure 12.5 m at full extension and are capable of carrying 650 kg payloads. A similar deployment principle is used for the Articulated Inspection Arm, used in the EAST and WEST devices (Villedieu et al., 2016). Although limited to 10 kg payloads, it can operate under vacuum conditions.

Some complex tasks may require highly bespoke handling equipment, such as divertor cassette replacement in ITER. The reference design of the Divertor Remote Handling System (Fig. 12) consists of two movers (Esqué et al., 2014). The Cassette Multifunctional Mover transports the ~10 ton divertor cassettes radially to and from the vessel through the maintenance port. Several interfacing end-effectors also allow it to transport a range of other hardware and remote handling equipment. The Cassette Toroidal Mover locates the cassettes inside the vessel, moving them along toroidal rails. A variety of tools are used to cut and weld divertor cooling pipes, lock the cassettes in place, and collect contaminated dust from the work area. These tools are deployed with a dexterous, teleoperated manipulator arm, mounted on the movers.

Remote operations, maintenance equipment and reactor components must all be designed together, as all associated constraints must be considered in an integrated way to achieve a successful design satisfying the lifetime operational requirements. As a result, design verification and validation typically require physical mock-up testing, where operations, equipment and components can be shown to work together as intended in a representative setup (Rolfe, 2007; Tesini and Palmer, 2008). The JET In-Vessel Training Facility is a full-scale mock-up of the JET vessel that has been used extensively to support reliable operations inside the reactor, and to train JET operators to the necessary levels of competence (Cusack et al., 1997). The ITER Divertor Test Platform 2 facility

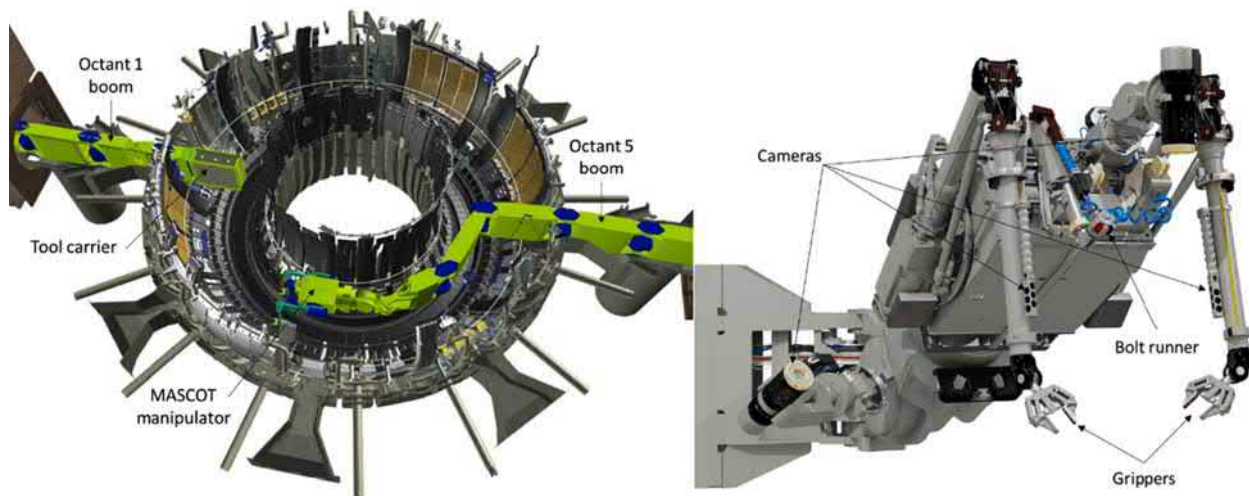


Fig. 11 JET Remote Maintenance System. Left: Overview of in-vessel equipment. Right: MASCOT teleoperated servo manipulator. Image courtesy of EUROfusion.

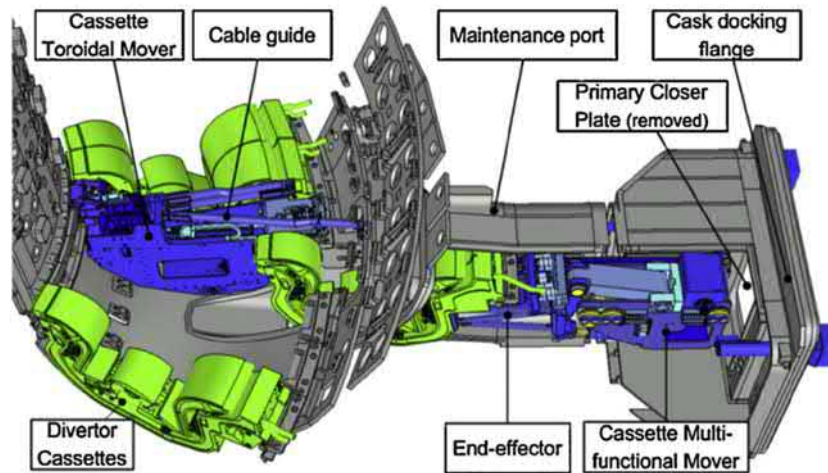


Fig. 12 Overview of ITER's divertor remote handling system (Esqué et al., 2014). From Esqué S, van Hille C, Ranz R. et al. (2014) Progress in the design, R&D and procurement preparation of the ITER Divertor Remote Handling System. *Fusion Engineering and Design* 89(9–10): 2373–2377.

provides a further example. This has allowed extensive testing to ensure that the design concept for the Divertor Remote Handling System is feasible (Saarinen et al., 2011), given that it must be able to operate in a highly constrained space and in the presence of gamma radiation.

The cost of remote equipment failure can be prohibitive, as the limited access makes rescue and recovery operations very complex and time-consuming. The more likely equipment failure modes, or those with particularly negative consequences, are eliminated from the system at the design stage. If a failure mode cannot be eliminated, the likelihood of it occurring can be reduced by providing a redundant system to prevent loss of function. Systems should fail in a known and controlled manner, such as lifting interfaces remaining closed in the case of a power failure to prevent payloads being dropped. Where possible, all remote maintenance equipment should allow recovery, which is defined as the ability of the system to extract failed subsystems from the restricted area using only its existing capabilities. In contrast, rescue is defined as the removal of failed equipment using external means, such as additional robots. All remote equipment must be rescuable.

In future reactors, the boundary of the restricted access area will expand as activation levels increase. In JET, the prohibited access area during maintenance is limited mostly to the vacuum vessel. Plasma-facing components are handled in the dedicated Beryllium Handling Facility, which is restricted to personnel wearing pressurized suits due to the toxicity hazards of beryllium particulates. However, radiation levels of ITER components will be much more significant (than in JET), so a dedicated hot cell with specialized remote equipment is required to process components leaving the reactor (Friconneau et al., 2017). In future systems it is expected that an entire Active Maintenance Facility will be required to process the highly activated components, with many large areas being out of bounds for human access (Thomas et al., 2014). This expanding boundary implies that the number of maintenance activities that must be performed remotely will increase dramatically.

The future of remote maintenance

Fusion reactors following ITER will face further challenges, related to the additional requirements as designs approach commercial deployment.

Higher fusion power will lead to increased activation and to higher radiation levels during maintenance, which will be sufficient to damage electronic systems that are not properly shielded. Deployment of robotic systems inside the vessel, while still possible for ITER, will be unfeasible for DEMO and future power plants unless adequate radiation-resistant technology is developed (Bachmann et al., 2018).

A key challenge for future fusion reactors is their larger size, which means very large components need to be handled with high precision. For instance, ITER requires a bespoke deployable rail system to handle the $1\text{ m} \times 1.4\text{ m} \times 0.5\text{ m}$, 4.5 t blanket modules (Raffray et al., 2014; Noguchi et al., 2018). The largest components in DEMO could be $\sim 10\text{ m}$ tall, have masses in the order of several tens of tonnes, and require a positioning accuracy in the order of $\pm 20\text{ mm}$ (Keep et al., 2017; Keech et al., 2019). The handling strategy for such large components is a fundamental driver for the architecture of the entire plant (Utoh et al., 2015; Wolff et al., 2017).

To reach commercially-relevant levels of plant availability, the speed of maintenance systems and the ability to perform tasks in parallel must improve significantly. Human-in-the-loop systems are inherently limited in these areas, and so it will be necessary to automate many maintenance activities, leveraging the productivity improvements already demonstrated in manufacturing industries worldwide. However, deployment of automated and autonomous systems with reduced human supervision in such harsh environments is a significant technology challenge, particularly considering the need for acceptance by regulators.

Given that many remote systems will need to be deployed, they will have to be highly reliable to minimize equipment failures during maintenance operations, which would otherwise reduce plant availability. This will require the adoption of more intelligent condition monitoring techniques which can give advanced warning of failures to better mitigate their consequences.

The connection of services (power, cooling & breeding fluids, etc.) is time consuming (Keogh et al., 2018). This problem is exacerbated by the higher temperatures that the components will experience, which increases the number and size of active cooling connections needed. Techniques are required which allow faster making and breaking of these connections, such as in-bore laser cutting and welding (Keogh et al., 2018).

Concluding remarks

Magnetic fusion is an attractive source of abundant and clean energy, but the endeavor to harness fusion power is technologically challenging. A limited number of fusion facilities have been built and successfully operated around the world during the last decades to investigate fundamental plasma physics issues that underpin the design of a new generation of power producing designs. Sustained progress over several decades has taken fusion close to the demonstration of the scientific feasibility of large-scale fusion energy production, with ITER expected to achieve operating conditions approaching those of a power reactor. ITER will also be a critical test of the reliable operation of a number of important engineering systems and core technologies.

Focus is now shifting toward the design and R&D of systems that, after ITER, will demonstrate the production of net electricity, the feasibility of operation with a closed tritium fuel cycle, and maintenance systems capable of achieving adequate plant availability. This is essential to overcome the remaining daunting technological hurdles to practical fusion energy.

In several key technologies further R&D will be essential to improve performance and reduce costs.

Superconducting Magnets—The main technological challenge for superconducting magnets is reducing the degradation of the performance of the conductor under electromagnetic and thermal cycles. This issue could prevent the optimal and reliable operation of magnetic fusion reactors over tens of years. Future research will be driven by the development and qualification of SC conductors to mitigate this risk.

The cost of magnets manufacturing must also be reduced where possible to make a fusion power plant an economically viable source of energy. Cost-effective solutions, such as the R&W approach to Nb₃Sn magnet manufacture and an efficient use of HTS coils, need to be pursued and validated with dedicated R&D. The effectiveness of the selected solutions should be demonstrated by long-length industrial production and, in the engineering phase, through the manufacturing and the testing of model coils.

Plasma facing components—R&D is required to substantiate the power handling performance of the high-heat-flux solutions under development, including the influence of neutron irradiation. The key elements in this regard would be: (1) full high-heat-flux qualification of a state of the art baseline PFC concept at relevant irradiation dose (e.g. as a post-irradiation test); and (2) development and qualification of novel neutron-resistant materials and joints for an enhanced lifetime. An early engagement of a dedicated irradiation testing campaign would be necessary to reduce the gap between the technological readiness and the design progress. Attention must be given to specific combinations of heat flux (i.e. temperature level) and irradiation fluence (i.e. damage level) at different areas of PFCs so that material requirements being pursued are in accordance with the envisaged design space. In parallel, dedicated structural design rules need to be developed on the basis of valid materials irradiation test data and failure criteria tailored for PFCs. Finally, adaptation to industrial manufacturing processes and incorporation of innovative fabrication technologies will remain future objectives.

Vacuum Vessel—The design and technology of the ITER VV are largely suitable for future fusion power plants. The stainless steels used in ITER and other existing machines are suitable for the higher neutron fluence expected in such devices. The fully welded double-wall design is an established concept and it is expected that the installation procedure of the ITER tokamak will validate the assembly of the sectors and ports structures. Construction in accordance with nuclear pressure vessel design codes, which require very high levels of qualification, is a significant cost driver. The ITER operational experience could result in the possible use of a more specific “Fusion Code” in future devices that might mitigate code requirements and reduce costs. To improve the sustainability of the VV some measures can be taken to reduce the amount of VV material that will remain activated over a long period. In future fusion devices, such as DEMO, that are larger than ITER, the VV inboard wall will be subject to higher EM forces and may require the use of a higher strength steel. To allow the in-vessel components to be installed with high alignment precision, the acceptance tolerances of the inner VV shape must be small and are usually at the limit of what can be achieved. Consequently, difficulty, risk and cost of the VV fabrication and assembly increase. The designers of future fusion power plants may succeed in including a greater customization range in the design of the support structures of the in-vessel components and hence relieve the requirements on the VV shape tolerances. Moreover, overall standardization and uniformity of the VV design and manufacturing methods should be emphasized in future fusion plants to reduce cost, to increase reliability and to mitigate tolerance issues. In addition, the implementation of plasma limiters can reduce the alignment requirements of the blanket first wall. The development of the VV of DEMO and other future devices will greatly benefit from the lessons learned and the solutions adopted in the nuclear licensing process of the ITER VV, e.g. regarding fabrication processes and their qualification and in the facilitation of in-service inspection, resulting in manufacturing cost reductions.

Remote Maintenance—As the radiation boundary expands inside fusion power plants, so too will the number of maintenance tasks that will have to be performed remotely. Combined with the increasing size and mass of reactor components and the need to achieve commercially-relevant plant availability, it follows that future remote maintenance systems will need to be faster,

more capable and more automated. To achieve this, future fusion plants will need to evolve highly integrated designs where the maintenance strategy is considered from the outset and built into the very architecture of the plant.

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Magnetic Confinement Fusion Technology: “Plasma Engineering”

Takashi Inoue^a, Hiroyuki Tobari^a, Koji Takahashi^a, So Maruyama^b, Ryota Imazawa^a, and Kenichi Kurihara^a, ^a National Institutes for Quantum and Radiological Science and Technology (QST), Naka Fusion Research Institute, Ibaraki-ken, Japan; and ^b ITER Organization, Saint-Paul-lez-Durance, France

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Glossary

Bootstrap current A non-inductive and spontaneous plasma current which arises when plasma pressure gradient is high. When the bootstrap current accounts for the majority of the plasma current, it will improve the economic efficiency of fusion reactors. For economical fusion reactors, the ratio of bootstrap current to the total plasma current needs to be 75% or more.

Electron cyclotron resonance frequency (ECRF) Uses RF waves with frequencies in the range from several tens to hundred GHz and mainly heats plasma electrons. It is also used for current drive in both central and peripheral region of plasma.

ELM Edge Localized Modes (ELMs) is an intermittent and periodic heat and particle release phenomenon that occurs in the plasma edge of the improved confinement mode called H-mode. The ELMs are necessary to keep the H-mode plasma steady, but there is a concern that the large energy released intermittently will damage the divertor material, and research to control the ELMs to small amplitude has been vigorously carried out.

ELMy H-mode Improved confinement mode called H-mode having ELM. H-mode without ELM is called ELM-free H-mode.

Fusion gain factor Q Ratio of fusion output power to input external power. The condition with $Q = 1$ is called break-even plasma condition, the fusion reaction produces energy equal to the energy injected externally. $Q = \infty$ is called self-ignition condition, the fusion reaction can be sustained without further external heating input in the condition. ITER aims to achieve burning plasma with $Q=10$ or more.

Internal transport barrier (ITB) ITB is a phenomenon in which the thermal conductivity of the plasma is drastically reduced at a certain radius inside the plasma, resulting in a significant improvement of the confinement performance and the formation of a steep temperature gradient, which was first discovered in JT-60. Since then, similar phenomena have been reported in many

tokamaks and are common to many high-performance plasmas, and have been studied worldwide as a key phenomenon to improve plasma confinement.

Ion cyclotron range of frequency (ICRF) uses RF waves with frequencies typically in the range from 10 to 100 MHz and mainly heats plasma ion. This heating method is usually referred to as ion cyclotron radiofrequency (ICRF) heating.

Landau damping In a non-collisional plasma where collisions between particles are negligible, wave energy is attenuated by the energy exchange between waves and particles. Particles traveling at a speed approximately the same as the phase velocity of the wave can exchange energy by interacting with the electric field of the wave. It means that particles slower than the phase velocity of the wave are accelerated by the wave and particles faster than the phase velocity of the wave are decelerated by the wave.

Lower hybrid range of frequency (LHRF) The LH waves have been used for current drive (LHCD) in the peripheral region of plasma, with typical frequencies in the range of several GHz.

Negative magnetic shear (or Reversed magnetic shear) In the negative magnetic shear, the magnetic shear becomes negative in the plasma confinement magnetic field configuration of the torus system. The negative magnetic shear configuration is considered to be a suitable configuration for steady-state tokamak reactors because it is compatible with a high bootstrap current fraction.

Neoclassical tearing mode (NTM) instabilities A type of resistive instability found in tokamak plasma, and likely to occur in high beta plasma with a positive magnetic shear and a large plasma pressure gradient. NTM forms magnetic islands on its resonant rational surface around the plasma, causing a plasma pressure drop. In order to stabilize NTMs, an external current injected into the inside of the magnetic island is applied, for example, using electron cyclotron wave power, which can be injected effectively within the width of the magnetic island.

Plasma disruptions There are two kinds of plasma disruption caused by thermal decay and plasma current collapse. The former impacts the thermal load on plasma facing components and the latter generates an electromagnetic force on the vacuum vessel and an in-vessel component.

Introduction

This article provides an overview of key technologies required for plasma production, initiation, and the sustainment of fusion reactions by Fueling, Heating and Current Drive, Diagnostics and Plasma Control.

The fusion reaction is initiated once the plasma reaches a certain density and temperature. To increase the plasma density, the Fueling System is firstly required to supply D_2 gas of a certain amount into the vacuum vessel and subsequently to provide a mass flow (in the form of gas or solid cryogenic pellets) at a certain rate to raise and maintain the plasma density. In a reactor, this mass flow would consist of a mixture of D_2 and T_2 . Once the plasma is initiated, it is sustained utilizing a magnetic field generated by the coils surrounding the vacuum vessel. In some magnetic confinement systems, such as the tokamak, a magnetic field generated by an electric current flowing within the plasma is also utilized as part of the confinement magnetic fields. This plasma current is initially generated within the plasma by inductive action of some of the external coils, but in a reactor it would be preferable to drive the current continuously by non-inductive means, including a contribution provided by Heating and Current Drive Systems. A key role of the Heating and Current Drive Systems is to heat the plasma to a required temperature of the order of 10^8 K to achieve fusion reactions. Plasma Diagnostics provide measurements to characterize the plasma, which are necessary to understand the plasma state. These measurements are also utilized in the Plasma Control System to sustain the plasma under feedback control for long pulses or in steady state. The Control System performs not only plasma control but also functions as part of investment protection and for nuclear safety. Thus, these systems play the essential roles described above in ITER, and such functions are also expected within a DEMO fusion reactor.

Heating and current drive systems

Neutral beam injection heating and current drive

Principles

Neutral beam injection (NBI) is the workhorse of plasma heating in magnetic confinement fusion reactors. Its function is to deliver high-energy particles to the heart of the fusion plasma, which is achieved by firstly neutralizing a high-energy ion beam (in a gas cell neutralizer) and then injecting the neutralized particle beam into the fusion plasma. With proven results in legacy fusion devices such as JT-60U (Ikeda et al., 2006), JET (Duesing et al., 1987), and the LHD (Kaneko et al., 2003), which include achieving high performance plasmas having a (DT equivalent) energy gain factor, Q , of 1.25 and highly efficient current drive, the output expected of future NBI systems such as those of ITER and DEMO are highly anticipated.

The principles of NBI are illustrated in Fig. 1. In ion sources, plasmas are the medium from which the ions are extracted. The extracted ions are then accelerated in an accelerator to generate an ion beam. The accelerators used for NBI are electrostatic

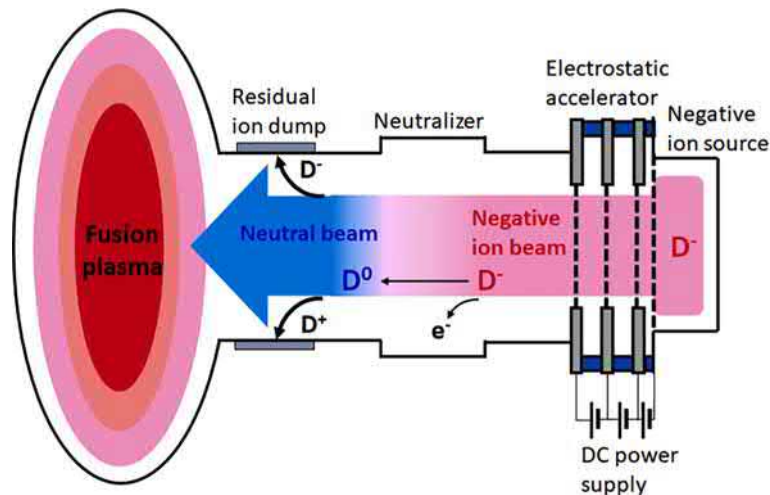


Fig. 1 Principle of NBI using a negative ion source.

accelerators, the same type of accelerators used at universities. While beam energies are relatively low, ranging from a few 10 keV to 1 MeV, the current required for the ITER negative ion beam is in the tens of amperes, which is about six orders of magnitude higher than the current required for conventional accelerators. Big numbers require big solutions and thus a feature characteristic of the ITER NBI accelerators is that they use electrodes over 1 meter in length to accelerate multiple (1280) beamlets simultaneously.

Only atoms with a positive or negative charge (ions) can be accelerated by an electric field. However, electrically charged ions cannot be injected into plasmas of fusion devices because they will be deflected by a “cage,” that is, the magnetic field of the plasma. In the ITER NBI system, negative ion beams must pass through a gas cell where the excess electrons will be stripped from the negative ions so they can be injected into the plasma as a neutral beam—a beam that is not affected by a magnetic field. Once deeply inside the plasma, the neutral beam particles collide with ions and electrons and are re-ionized, becoming trapped in the magnetic cage. Through these repeated rapid collisions with ions and electrons in the plasma, energy is transferred to the plasma particles, a process which heats the plasma.

NBI will also play a key role in driving the plasma current required to sustain the fusion plasma. The established operating mode for tokamak devices is the pulsed mode, where current is applied to plasmas via electromagnetic induction by means of a transformer, and this also creates a component of the magnetic field used for plasma confinement. However, new fusion reactors are aiming for steady state, or continuous, operation. Neutral beams also provide momentum to the plasma and current drive by tangential injection into the torus. Through ionization by collisions with torus plasma electrons and ions, the toroidally circulating fast positive ions run around inside the circular chamber as large nested helices to make the magnetic cage stable (Dumont, 2021).

NBI systems were originally developed for use in fusion experiments in the 1970s (Dawson et al., 1971). The beam energies required for tokamaks at that time were ~ 30 keV. Accordingly, the very first NBI systems were based on positive precursor ion beams of hydrogen or deuterium, which were then converted in the injector to a “neutral” beam by neutralizing the ions in a gas cell neutralizer. As plasmas started getting larger and denser, beam energies of 0.5 MeV or more were required to inject the beams into the heart of the plasma. Fig. 2 shows the relationship between the neutralization efficiency of the ion beams and the beam energies. At beam energies above 200 keV, the neutralization efficiency of positive ion beams is essentially zero. On the other hand, negative ion beams can achieve a high neutralization efficiency of about 60% even at beam energies above 1 MeV. Therefore, it is crucial that the NBI systems of ITER and DEMO use negative ion beams for their primary beams.

Developing the negative-ion-based NBI for ITER

Performance requirements and technical challenges

Table 1 compares the values achieved by legacy NBI systems that use negative ion sources with the values required of the ITER and DEMO NBIs. Comparing the three main performance parameters of NBI systems—beam energy, beam current, and pulse duration—the large beam current required of ITER has already been achieved by the system developed for the Large Helical Device stellarator (LHD), but the beam energies and pulse durations demanded of ITER are very substantially greater. In other words, accelerating beams to 1 MeV or more for pulses lasting 3600 s are crucial engineering challenges for future NBI systems such as those in ITER or DEMO.

Negative ion source technological improvements

The negative ion sources used in JT-60U (Ikeda et al., 2006), JT-60SA (Ikeda et al., 2007), and the LHD (Kaneko et al., 2003) were designed to produce negative ions in an arc discharge plasma using filaments as the cathodes and the ion source walls as the anodes. Although negative ions are generated in arc discharge plasmas, their density is 1/10 or less than that of the positive ions (negative ion current density: 100 A/m²). The negative ion density had to be increased to obtain a high-current beam. Thus, in an effort to

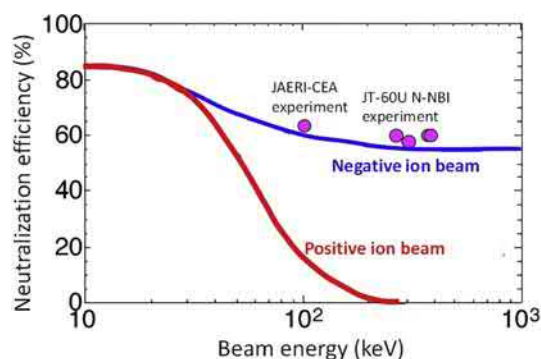


Fig. 2 Dependence of the neutralization efficiency via charge exchange neutralization for positive and negative deuterium ions vs. the beam energy.

Table 1 Requirements of legacy and planned future NBI systems.

Parameters/Machines	LHD (Kaneko et al., 2003)	JT-60 U (Ikeda et al., 2006)	JT-60SA (Dawson et al., 1971)	ITER	DEMO
Status	Achieved		Under Development		In Planning
Beam energy (MeV)	0.18	0.5	0.5	1	1 ~ 1.5
Beam current (A)	40	22	22	40	~40
Pulse duration (s)	10	30	100	3600	Steady state
Technical issues				reduced maintenance frequency Twice a year	Continuous operation of appx. 1 year • RF-driven negative ion source • Cesium-free ion source • Plasma neutralizer • Photo (laser) neutralizer

increase current density, ion sources based on the “surface-ionization” principle were developed. Using this principle, atoms are converted into negative ions by receiving electrons at electrode surfaces, whose work function can be reduced by coating the ion-extracting electrodes with alkali metals, such as cesium, resulting in a high conversion rate (Leung et al., 1990; Hanada et al., 2006). By adding about a gram of cesium to the arc discharge plasma, a highly dense negative ion current density of 380 A/m² was obtained, which satisfies the current density required of ITER. The Max Planck Institute for Plasma Physics (IPP) in Garching, Germany, took a different approach to developing negative ion sources. They developed a new technology in which radiofrequency (RF) waves are injected through an external antenna to generate plasmas (Speth et al., 2006). Compared with the arc discharge method, RF ion sources have advantages such as

- requiring less maintenance and achieving higher reliability, removing the issue of filament lifetime,
- being clean because of the absence of evaporation of tungsten (filament material), removing the issue of cesium consumption by covering it with tungsten, therefore requiring no maintenance even after long operations.

Thus, RF negative ion sources were chosen as the baseline design for producing plasmas in the ITER negative-ion-based neutral beam injection source. The NBI design of ITER has eight RF drivers arranged in parallel, as shown in Fig. 3, to uniformly generate negative ions near the 1 × 0.6 m electrode surfaces and produce a negative ion current of 40 A. To make these RF negative ion sources a reality, research is ongoing at IPP on the ELISE test rig (Extraction from a Large Ion Source Experiment), which features an ion source one-half as large as ITER’s (4 drivers). With the goal of improving extracted ion current and stability, so far, a 190 A/m² deuterium negative ion beam has been successfully extracted for 2700 s, which is about 70% of the ITER requirement (Kraus et al., 2018). The key aspects of the ion sources will be tested on SPIDER (Source for the Production of Ions of Deuterium Extracted from RF plasma), an ITER-scale negative ion source at the Neutral Beam Test Facility (or NBTF) in Padua, Italy (Toigo et al., 2017). The NBTF started operations in June 2018 to demonstrate the required performance of ITER.

Accelerator technological improvements

Typical accelerators, such as those used at universities for particle experiments, are filled with an insulating gas like SF₆ to insulate the high voltage that is applied to the accelerator. However, because the beamlines of the ITER NBI system are straight, the accelerators are located directly in front of the plasma. Consequently, radiation from the fusion reactions in the form of neutrons and gamma rays flows into the NBI system through the beam injection ports. If the ITER NBI system were a conventional system that

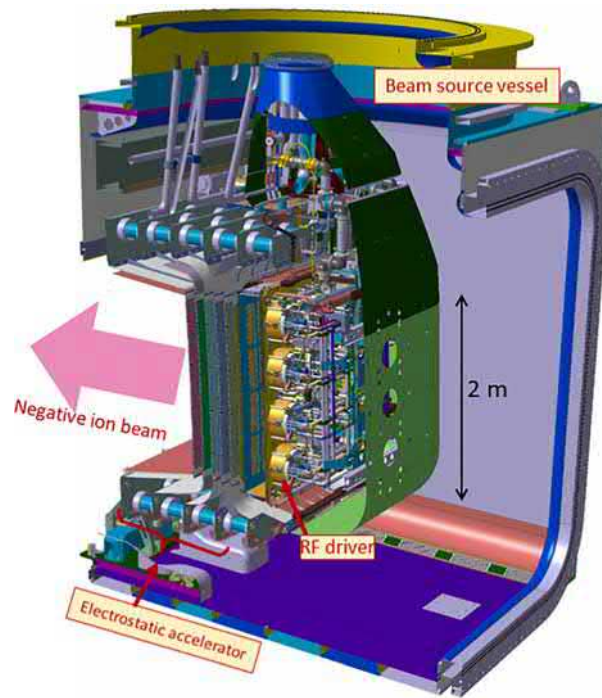


Fig. 3 Schematic of the beam source designed for the ITER NBI.

used an insulating gas for applied voltages, a leakage current would flow through the insulating gas because of radiation-induced conductivity, resulting in decreased insulation performance and an exorbitant heating loss of over 100 kW. ITER overcomes this issue because its negative ion sources and accelerators are installed in a high vacuum, which insulates the system from the 1 MV acceleration voltages. As previously mentioned, for the ITER NBI to obtain sufficient current, the extraction takes place through 1280 beamlet openings distributed over a very large area (1×0.6 m).

In the early years of R&D for the ITER negative ion accelerator (around 1994), there were no inorganic insulator rings larger than 1 m to provide a sufficient insulation distance to ensure vacuum insulation for high voltages of 1 MV in the accelerator. Fiber reinforced epoxy (FRP) insulator rings were used, but they proved to be unsuitable to hold off 1 MV voltages due to frequent discharges along the vacuum side. Experiments showed that the discharges originated at the triple junction where three elements (metal, dielectric and vacuum) with different permittivities come in contact with one point. The electric field on such specific point tends to increase. However, through finite element analysis, it was possible to reduce the triple junction electric field to under 1 kV, resulting in the first successful hold-off of 1 MV for 1 h (Taniguchi et al., 2003; Inoue et al., 2006a).

The insulation distance between the electrodes was based on the standoff voltage data of Rogowski electrodes having a diameter of about 200 mm. The standoff voltage, V , was assumed to be proportional to the one-half power of the electrode gap (g), according to clump theory ($V = C g^{0.5}$). The proportionality factor, C ($= (V \times E)^{0.5}$), is a function of electrode area, and therefore electrodes having various surface areas were tested to derive a relation between the proportionality factor, C , and the electrode area, S . Results of these tests are illustrated in Fig. 4 (Tobari et al., 2017; Kojima et al., 2016). The voltage dependence on the surface area turned out to be rather weak, and thus vacuum insulation of a 1 MV accelerator became plausible.

When multiple negative ion beamlets are accelerated simultaneously in parallel, the beams are deflected due to space charge repulsion between the beams, which causes the beams to collide with the accelerating electrodes. The resulting high local thermal load can actually melt the electrodes, which can trigger degassing, and this, in turn, can cause breakdown discharges—a typical engineering issue with long pulse beam acceleration. By simulating the negative ion trajectory using 3D beam trajectory analysis, a novel method of creating an electric field to compensate the deflected beam trajectory by deliberately shifting the axis of the electrode aperture through which the beam passes was developed and tested at the National Institutes for Quantum and Radiological Science and Technology (QST) in Naka, Japan. The thermal loads on the electrodes, which in the past typically accounted for 25% of the acceleration power (acceleration current $\times$ acceleration voltage), were reduced to 10%. As a result, a negative ion beam of 0.97 MeV, 190 A/m^2 was successfully accelerated for 60 s (Kojima et al., 2017) and a beam of 0.5 MeV, 154 A/m^2 for 118 s (Hiratsuka et al., 2020), satisfying the requirements of ITER and JT-60SA, respectively.

Advanced technology development for NBI in demo and future reactors

The technical specifications of the NBI system for DEMO have not yet been finalized, but a logical extrapolation of ITER NBI technology is assumed, with estimated values shown in Table 1. The most substantial change in the step from ITER to DEMO is the neutron and gamma radiation doses. The radiation intensity of DEMO is expected to range from several to 10 times higher than

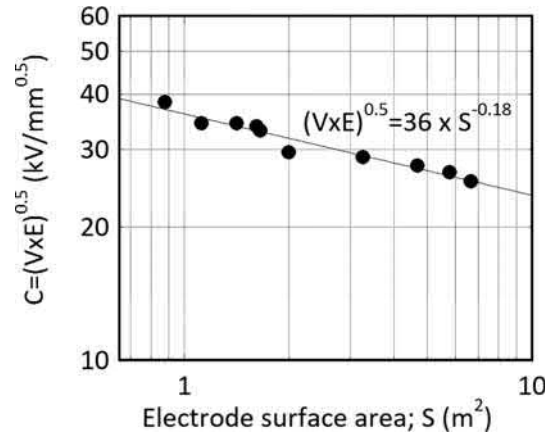


Fig. 4 Experimental measurements of the proportionality factor, C , used to determine standoff voltage in a vacuum-insulated configuration, as a function of the electrode surface area, S .

that of ITER. To prevent the vital functional materials used in the ion source from degrading (such as the permanent magnets and ceramic insulator rings), shielding must be installed in the beamlines. A design has been proposed in which shielding is installed in the empty spaces created by focusing the multiple beams accelerated by the large-area electrodes into several converging beams (Inoue et al., 2006b).

DEMO must also prove to be economically efficient to demonstrate the commercial viability of fusion and so an NBI system efficiency (=NB injected power/NB operating power) of 50% or more is required to reduce the circulating power of the facility (Okano et al., 1997). NBI system efficiency is dependent on neutralization efficiency, which has a maximum of ~60% in the case of conventional gas neutralization cells. ITER's NBI system has a system efficiency of about 40%. A need for alternatives to gas cell neutralizers has therefore been recognized. Plasma neutralizers (neutralization efficiency: 80%) are a possible candidate and, accordingly, proof-of-principle tests have been conducted (Hanada et al., 2004). Another promising candidate is the photo (laser) neutralizer, which, in theory, could achieve a neutralization efficiency of 90% (Inoue et al., 2006b).

In terms of DEMO's beam acceleration, using electrostatic accelerators similar to those of ITER, in addition to vacuum insulation of 1–1.5 MV, gas insulation, involving a gas flow into the ion source to produce ions, has been verified (Inoue et al., 2006b). Operating the negative ion sources stably during the continuous 1-year operation campaign required of DEMO will entail the development of advanced technologies, ideally technologies that reduce maintenance frequency (RF-driven negative ion sources) and do not require seeding with cesium.

Radio frequency heating and current drive

Principles

Just as with communication devices such as mobile phones, radios, and televisions, RF heating and current drive systems also use radio frequencies, in the range from a few MHz to several 100 GHz, to heat and drive plasmas. The heating principle is similar to that of the microwave oven. In magnetically confined plasmas, electrons and ions spiral around a magnetic line of force (hence the name cyclotron). Energy is fed to these charged particles in the form of electromagnetic waves at frequencies that match the oscillations of the different particles inside the plasma—the resonant cyclotron frequency $\Omega_{i,e} (=|q_{i,e}| \cdot B / 2\pi \cdot m_{i,e})$: ion, electron charge, B : magnetic field, $m_{i,e}$: ion, electron mass). The charged particles in the plasma absorb the electromagnetic energy and as a result, increase in kinetic energy. By way of collisions, ions will in turn transfer the absorbed energy to electrons, thus heating the plasma. Radiofrequency waves are also used to induce currents in the plasma in a technique known as radio-frequency current drive. RF waves are injected into the plasma and the consequent wave electric field accelerates the electrons as the wave and particles move together around the torus, resulting in the electrons developing a net motion, or current (Fig. 5: Landau damping).

RF heating systems have been used in various magnetic confinement fusion devices, and particularly in tokamaks and stellarators. RF heating systems are composed of an RF source, a transmission system for transmitting the generated RF power to the vacuum vessel, and a coupling system (antennas and launchers) installed in the vacuum vessel port to inject the RF power into the plasma. RF heating systems have the advantage of being less susceptible to the effects of radiation (such as from neutrons) because their RF sources can be installed far away from the plasma, which may be advantageous in future fusion reactors. There are three kinds of RF heating systems such as ICRF, LHFR and ECRF.

Ion cyclotron and lower hybrid radio frequency systems

An ion cyclotron radio frequency (ICRF) system uses commercially available tetrodes with MW-level output power, which were already available by the second half of the 20th century. In addition, radio/TV broadcasting equipment can be used for the transmission systems. The JET (EU) and TFTR (US) tokamaks have also exploited ICRF heating extensively to explore fusion plasma

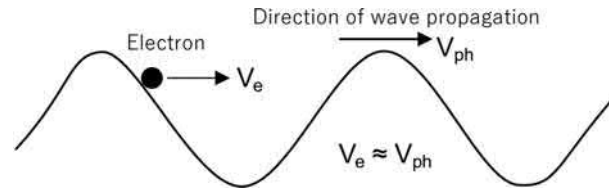


Fig. 5 Landau damping. Particles traveling at a speed approximately the same as the phase velocity of the wave can exchange energy by interacting with the electric field of the wave. The event is called as Landau damping.

performance in the parameter range relevant to fusion power production (The JET Team, 1998; Phillips et al., 2000). Based on these experience, ICRF heating has developed into a mature RF heating technology. In the MHz range, however, the wavelength within the plasma differs substantially from that in vacuum. Therefore, the design of a suitable RF matching system is important in designing a suitable ICRF antenna in order to obtain a good coupling with plasma. Consequently, ICRF antennas must be installed very close to plasma boundaries, which raises the engineering challenges of how to mitigate the high thermal loads from the plasmas and the effects of neutron irradiation on a high voltage insulator. Thus, there are significant challenges in the development of ICRF antenna and matching systems for ITER or DEMO reactors, and major design and R&D activities are underway to address these challenges for ITER (Hoang et al., 2009).

In lower hybrid radio frequency (LHRF) systems, klystrons are used for the RF sources. In the case of JT-60UI, current drive in the peripheral region of plasmas was successfully demonstrated using MW-level klystrons. However, as with ICRF systems, the launchers of LHRF systems also must be installed near the plasma to ensure efficient coupling of the waves to the plasma, and this raises the engineering challenges of not only mitigating high thermal heat loads from the plasmas but also of limiting impurity contamination into plasma (due to sputtering of the plasma-facing material of the launchers). Novel solutions, such as the passive-active multijunction launcher (Kasugai et al., 1998; Hillairet et al., 2013), are being investigated to address the issues associated with applying LHCD technology in the reactor plasma environment.

Electron cyclotron radio frequency system

An electron cyclotron radio frequency (ECRF) system uses high-frequency RF waves in the range of 100 GHz. In this frequency range, RF power is generated by a gyrotron, transmitted either by waveguides or quasi-optical (via a mirror arrangement) means toward a vacuum vessel port and finally injected into plasma by a launcher installed at the port. There are two types of transmission lines; by means of waveguides and quasi-optical mirror. The waveguide transmission is commonly applied to the ECRH system of many machines and ITER (Cengher et al., 2017; Omori et al., 2011). Power density can be increased with focusing the beam shape by using metal mirrors. Thus, RF power of about 1 MW can be injected through a relatively small aperture (a few to several tens of centimeters square) and allows the launcher mirrors to be installed at some distance from the plasma, with improved potential for shielding and cooling than is the case for other RF systems. Further, these mirrors are rotatable and can control the direction of RF power injection, which allows the location of power deposition to be controlled selectively, from the center to the edge of a plasma for plasma heating and current drive by taking advantage of both cyclotron resonant and Doppler-shifted absorption, as discussed by Dumont (2021). Taking the advantage of this feature, neoclassical tearing mode instabilities (This instability is caused by a magnetic island induced by magnetic fluctuation.), which cause plasma disruptions (collapse events of plasma due to instabilities) due to the formation of magnetic islands in the peripheral region of plasma, can be suppressed. The mitigation and the stabilization of magnetic islands by performing local current drive within magnetic islands can only be accomplished using ECRH systems due to the narrow spatial localization of the wave absorption in the plasma.

Because of these advantages, ECRH was selected as the single RF heating system for JT-60SA (Kobayashi et al., 2009), a joint Japan and EU tokamak which is expected to achieve plasma operation in 2021. The ITER project plans to install both an ECRH and an ICRF system, each capable of launching 20 MW of power into the plasma, with later expansions of this capability possible.

As an example of an initial reactor-grade ECRH system, an overview of the ITER ECRH system configuration is illustrated in Fig. 6. High power microwaves at 170 GHz are transmitted over a distance of ~ 100 m using several arrays of oversized cylindrical corrugated waveguide and the power is focused into the plasma by means of groups of mirrors situated in five launchers, one situated on the tokamak midplane and the other four situated in upper ports. To generate the 24 MW of source power, providing an injection power of 20 MW, ITER will use 24 “gyrotrons” each generating 1 MW of RF power at 170 GHz. An example of an ITER gyrotron is shown in Fig. 7, while a schematic cross-section of the device is illustrated in Fig. 8. The gyrotron concept, in which stimulated emission is generated and amplified by a relativistic electron beam passing through a cylindrical cavity, was invented in the 1960s in Russia and the United States (Gaponov, 1959; Schneider, 1959). The technology has been under development since that time with several design improvements and substantial increases in power generation in the last two decades.

One can understand the basis operating principle of the gyrotron from the schematic in Fig. 8: the electron gun at the bottom of the gyrotron extracts a “hollow” beam of electrons having 70–80 keV of energy and this is accelerated in the magnetic field generated by the superconducting coils; the beam is subsequently injected into the cavity resonator, where the rotational energy of the electrons is converted into RF energy by means of the electron cyclotron maser principle (Gaponov, 1959; Schneider, 1959); the RF excited within the cavity has a complex electromagnetic field distribution which is then converted into a Gaussian power

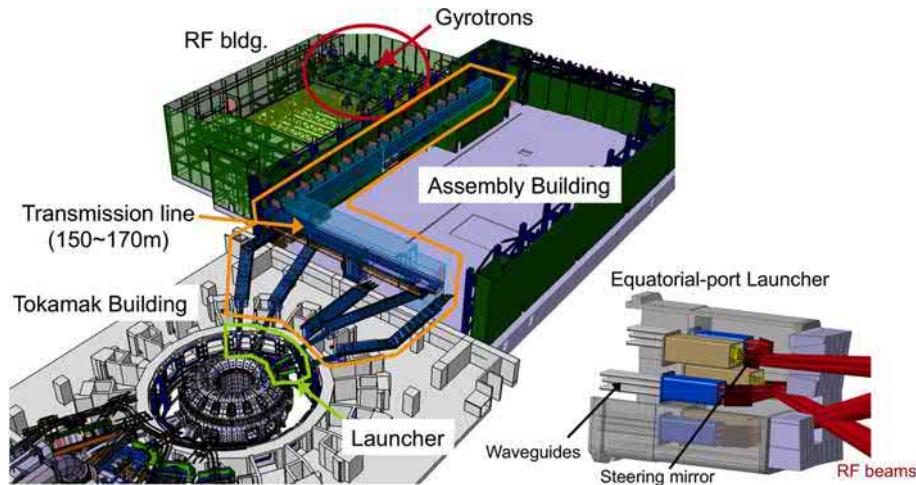


Fig. 6 An overview of the ITER ECRH system configuration.



Fig. 7 An example of an ITER gyrotron, one manufactured in Japan.

distribution by a mode converter installed at the exit to the cavity, and the resulting RF beam is emitted from the gyrotron. The electron beam component retaining the energy fraction not converted into RF power is transmitted to the collector located at the top of the gyrotron, where the residual beam power is recovered and/or thermally dissipated.

Research and development of gyrotrons for fusion plasma heating has been underway since the early 1980s, and a high power, long pulse capability has been achieved by overcoming several specific challenges: (1) high-order mode oscillation due to the high frequency of operation; (2) improvement of gyrotron operational efficiency; and (3) the need for vacuum windows to allow low loss transmission of MW-level RF energy from the gyrotron to the transmission system. Development of sophisticated numerical codes capable of modeling nonlinear cavity oscillations (Sakamoto et al., 1996) allowed the derivation of conditions for stable mode oscillation, eliminating competing modes. This analysis provided the basis for the manufacture of the internal components with precision machining on the order of microns. Combined with the use of position control of the electron beam injection into the cavity, conditions for stable oscillation of the 170 GHz gyrotrons in ITER were established. The ITER R&D activities also demonstrated that by insulating the gyrotron collector from the gyrotron body and applying a high voltage to the body of the collector to decelerate the electron beam and recover its kinetic energy, gyrotron operation efficiency could be increased to 50%.

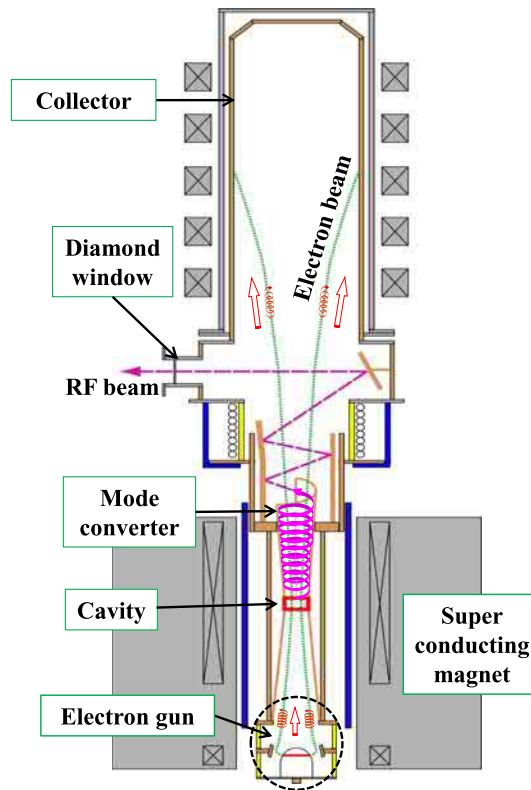


Fig. 8 A schematic cross-section of the Gyrotron device.

Development of a vacuum window capable of handling high power density with low losses in long pulse operation has been a long-standing constraint on the application of ECRH to fusion plasma heating, particularly with a perspective toward its application in fusion reactors. In earlier experiments an alumina or a sapphire disk was used for gyrotron vacuum windows, but the transmitted power was limited to a few hundred kW for pulses of less than a second, even with improved cooling structures. In 1990, a high-quality, large-diameter (~ 100 mm) synthetic diamond disk produced by means of CVD (plasma chemical vapor deposition) became available (Brandon et al., 2001). By using vacuum windows formed from these synthetic diamonds, which have about five times higher thermal conductivity than copper and therefore allow efficient edge cooling of the disk, 1 MW steady-state gyrotron operation could be demonstrated. The high mechanical strength of diamond also allows very thin windows, ~ 1 – 1.5 mm, “tuned” to the gyrotron radiation wavelength to be fabricated, minimizing reflection and absorption losses.

Developing prototype reactor ECRF systems

In case of both ICRF and LHRF system, their antenna design is the most critical issue because of high thermal radiation and large neutron flux from plasma, which lead the damage on the antenna front. Although the technical issue exist, ICRF is a candidate of Heating and Current Drive system of DEMO (Federici et al., 2019). On the other hand, an ECRF antenna requires only a small aperture to inject RF power and, in this sense, ECRF is a promising candidate of RF heating system for a DEMO reactor. The various design options for DEMO are currently under development, ranging from high-field tokamaks (11 T) to devices with more modest fields (around 6 T, which is close to ITER’s magnetic field strength of 5.3 T) with the aim of reducing the construction cost. Considering ECRF over this range of magnetic fields, an ECRF system at 300 GHz would be required for a high-field design, while a frequency close to 200 GHz would be required in the latter case. Numerous technologies and advances acquired in the development of the ITER gyrotrons can be applied to these future designs, while further increasing their output power, perhaps into the range of 1.5–2 MW. Regarding to the development of the launcher for DEMO, the use of a movable launcher mirror to control the injection angle of the beams for changing the location of RF power deposition is unlikely to be practical in DEMO and fusion reactors due to the significant increase of neutron loading. Instead, a gyrotron capable of multi-frequency oscillation could be used to control the resonance location of RF power in plasmas by variation of the RF frequency, since the resonant location for absorption of the RF power depends on both the wave frequency and the magnetic field. Initial developments in multi-frequency gyrotrons (Ikeda et al., 2017) have demonstrated the concept successfully and, while significant further development is required for their application to CW high power operation in future reactors, this would provide the possibility of varying the radial deposition location while simplifying the launcher design.

Fuelling systems

Introduction

Fuel supply systems, which are designed to maintain the required operational plasma density and to replenish the DT fuel consumed in fusion reactions, mainly consist of a gas injection system (GIS [Boccaccini and Day, 2021](#)) and a pellet injection system (PIS). These systems are capable of supplying fuel particles (H, D, T) and impurities (Ne, Ar, etc.) into the vacuum vessel, which is necessary for plasma burn control. These fuel and impurity species are provided by the storage and delivery system of the tritium plant, as described by [Boccaccini and Day \(2021\)](#). A further function now incorporated into the fuel supply system which is not directly related to the supply of fuel particles, but in tokamak experiments is just as important, is the disruption mitigation system (DMS [Lehnen et al., 2018](#)), which rapidly (~ 10 – 100 ms) injects a large number of particles into the plasma when the plasma control system flags an approaching disruption. The operation of the DMS is designed to mitigate the effects of thermal loading and electromagnetic forces, and to suppress the generation of runaway electron (RE) beams associated with a sudden plasma quench.

In this article, the features of the PIS and DMS, which play key roles in the operation of large fusion devices, and the basic technologies of the respective systems are described. The GIS, which is the standard fuel supply system for small to medium scale machine, employs piezo-electric actuating valves or mass flow controllers compatible with nuclear environment. In ITER, newly developed mass flow controller is chosen for mass flow control.

Pellet injection system (PIS)

As magnetic fusion devices have increased in physical size, it has become increasingly challenging for the GIS to supply fuel particles to the center of the plasma efficiently, since injected gas particles are ionized at the plasma edge and must be transported to the center by (collisional and turbulent) transport processes within the plasma. Experiments showed that the required plasma density control efficiency could not be achieved. Thus, in many fusion devices, pellet injection systems capable of injecting solid, high-density fuel in the form of small pellets of frozen hydrogen or deuterium play a central role in controlling plasma density. It should be noted that a significant R&D issues for fusion reactors (including ITER, of course) is that tritium injectors must sustain cryogenic operation in spite of the heat generated by the radioactive decay of tritium (which is, of course, not the case for hydrogen and deuterium injectors).

In fusion plasmas, particles injected into the plasma become ionized and are subjected to plasma-induced forces which drive the cloud of particles resulting from the ablation of the pellet from the high field side (HFS) to the low field side (LFS), an effect associated with the magnetic field in the plasma. Therefore, when injecting pellets from the LFS, the injection speed must be extremely fast (> 2000 m/s) to penetrate deeply into the plasma and to overcome this process, which not only complicates the system but also makes it difficult to support continuous operation at the pellet injection frequencies of 1–10 Hz required to meet the fuelling requirements. In many fusion devices, including ITER, pellets are injected from the HFS side, which can supply fuel particles at a sufficient rate to the center of the plasma even at low injection speeds (300–500 m/s), which is favorable in comparison to the injection velocity requirements from the LFS side.

A brief description of the pipe gun pellet injector capable of both generating and accelerating pellets is given below, followed by a description of the main features of the PIS, i.e., the extruder which generates and injects solid fuel and the accelerator which accelerates pellets to the required speed.

Pipe gun pellet injector

The simplest type of pellet injector is the pipe gun injector. These have a proven track record and have been used in fusion devices for over 30 years. As shown in [Fig. 9](#), the section of the cylinder of the pipe gun without the extruder is cooled below the triple point of

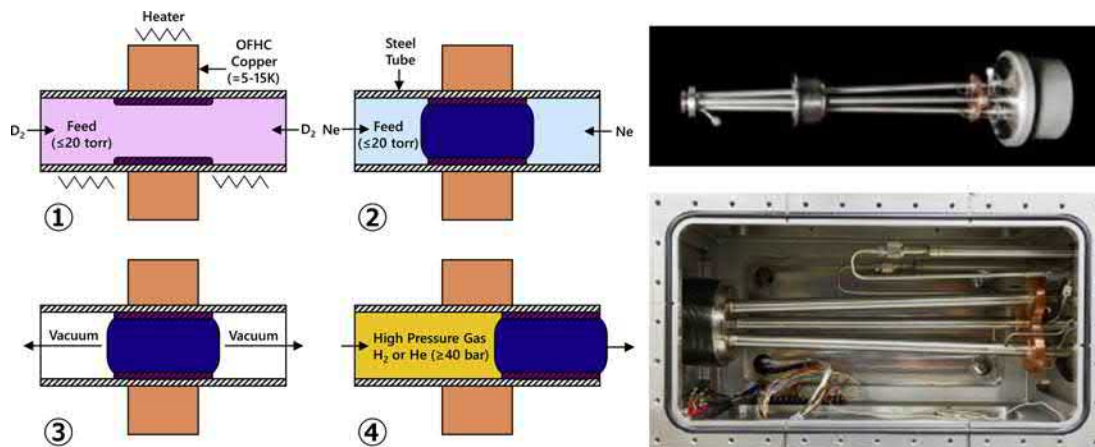


Fig. 9 Pipe gun pellet injector concept and application to Shattered Pellet Injection DMS. From Baylor LR, et al. (2019) *Nuclear Fusion* 59: 066008.

the fuel gas and the fuel gas is then introduced to form pellets directly. High-pressure gas is supplied from behind to detach the pellets and accelerate them to the required speed. Thus, pipe gun injectors are not suitable for continuous fuel particle supply and are mainly used for intermittent operations such as injecting impurity pellets.

Extruder

Extruders can be divided into two main types, batch and continuous, also known as piston and screw extruders, respectively. With piston extruders, fuel particles are pre-solidified and stored in the cylinder, and the solid fuel is injected by the piston. With screw extruders, as shown in Fig. 10, fuel gas is supplied to the extruder which is cooled below the triple point and the solidified fuel is continuously injected by a screw mechanism. Using this principle, solid fuel can be injected continuously as long as there is a supply of fuel gas and coolant.

Accelerator

There are two types of pellet accelerators, the pneumatic gas gun type, driven by compressed gas, and the centrifuge type, driven by centrifugal force. With pneumatic gas gun accelerators, a solenoid drives a pellet cutter back and forth, solid fuel on a ribbon ejected from the extruder is loaded into the cylinder, and the compressed gas accelerates the pellets to the required speed. With centrifuge accelerators, the solid fuel injected from the extruder is cut into the required size by a pellet cutter, the fuel is dropped into a pellet catcher at the center of a centrifugal accelerator, and then accelerated to the required speed by using centrifugal force.

ITER, for example, uses the gas gun method, which has been proven in fusion devices such as the DIII-D tokamak and the LHD stellarator. Research and development in support of the ITER system is being conducted at the Oak Ridge National Laboratory in the U.S. using equipment at 1/5th scale.

Disruption mitigation system (DMS)

Current fusion devices such as JET and the ASDEX Upgrade have used Massive Gas Injectors (MGI) that use high pressure gas valves (52 bar D_2 or Ar/D_2 :0.1/0.9 in JET) driven by piezoelectric or eddy current forces for their disruption mitigation system. These have

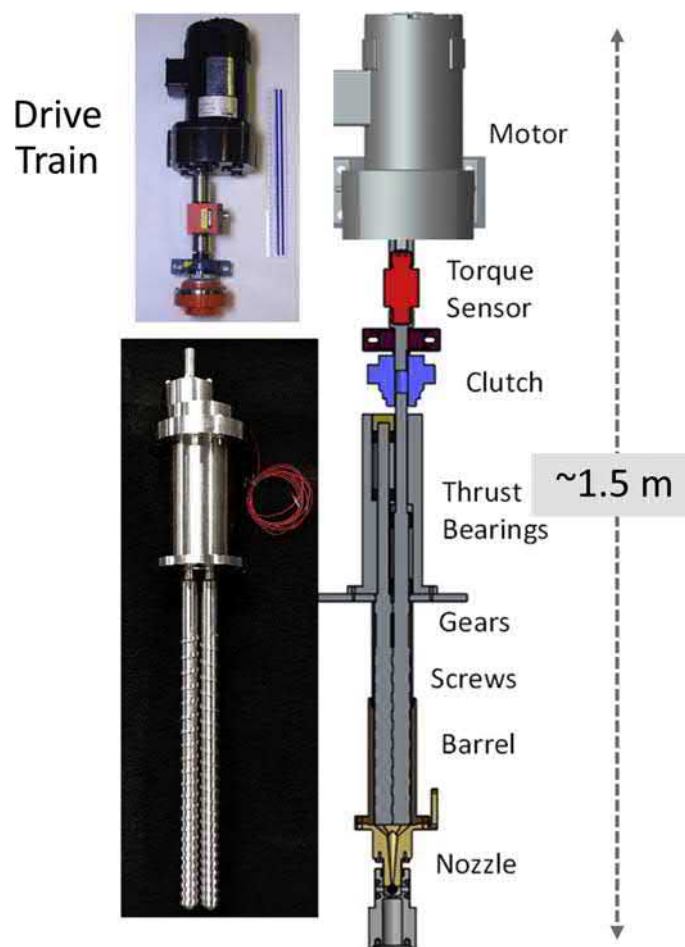


Fig. 10 Twin screw extruder developed for W7X. From Baylor LR, et al. (2019) *Nuclear Fusion* 59: 066008.

succeeded in mitigating the effects of thermal loading and electromagnetic forces caused by plasma disruptions by injecting Ne, Ar or a gas mixture involving one of these impurities and D₂. However, JET experiments verified that MGI did not suppress or mitigate Runaway Electron (RE) beam generation because the penetration of injected particles into the plasma was insufficient. Thus, ITER will use shattered pellet injectors (SPI)—installed in seven locations around the tokamak—capable of injecting a larger number of particles than MGI into the center of the plasma. See Fig. 11.

SPI is a pipe gun driven system developed at Oak Ridge National Laboratory. Solid pellets made of argon, neon, deuterium, or a mixture of these materials fly through a guide tube at a speed of 300 m/s to 500 m/s in order to reach the edge of the plasma. These pellets are crushed by a shattering tube installed at the end of the guide tube before being injected into the plasma. Although the effectiveness of SPIs have been verified in the DIII-D device, the ITER Organization has taken the lead to acquire the data necessary to extrapolate the data to the ITER scale, and is equipping the JET, KSTAR, and ASDEX Upgrade devices with SPI systems to explore the effectiveness of this approach to disruption mitigation in experiments starting in 2020.

Fueling systems for a nuclear device face significant additional challenges in sustaining the required fuel mixture to maintain high fusion gain operation under quasi-stationary conditions compared to the demands made on fueling systems for existing devices. Therefore, successful development and implementation of fueling and disruption mitigation systems are critical to the success of ITER operation and will establish the technological basis both for the DEMO project and for future fusion power plants.

Plasma diagnostics

Introduction

Magnetic confinement reactors are equipped with an array of various diagnostic instruments to provide the measurements necessary to control, evaluate, and optimize plasma performance (both to operate the reactor safely and stably and to maintain high fusion power for a long period of time) and to further the understanding of plasma physics (to develop better operating modes). Plasma diagnostics serve the following four main functions.

- #1: Protect reactor systems and components by ensuring their safety and functionality.
Measuring the surface temperature of plasma-facing components, such as the blanket or divertor, to verify that the temperature is within the allowable range, is but one example.
- #2: Control plasma
Parameters necessary to control and maintain dense, high-temperature plasmas, such as plasma current, electron density, and the location of the last closed flux surface, are measured throughout the entire process from start-up and steady-state operation to shutdown.
- #3: Evaluate plasma performance
Parameters used to evaluate plasma performance, such as assumed confinement improvement H_{Hy2} , normalized beta β_N , normalized density n_e/n_{GW} , and bootstrap current fraction f_{BS} , are determined to create optimal operation scenarios. Reference Wade (2021) explains these parameters.
- #4: Enhance the understanding of plasma physics
Diagnostic instruments are used to establish the underlying physics of fusion plasmas, such as determining the responses to plasma instabilities and the basis for heat and particle transport. Diagnostics are indispensable tools in terms of developing fusion reactors: they provide the scientific foundation upon which reactor design tools and operation simulators are based.

A well-known example of function #3, evaluating plasma performance, relates to the T-3 tokamak from the former Soviet Union. Soviet scientists announced that the T-3 could produce plasmas with temperatures above 1 keV, but doubts were cast on the

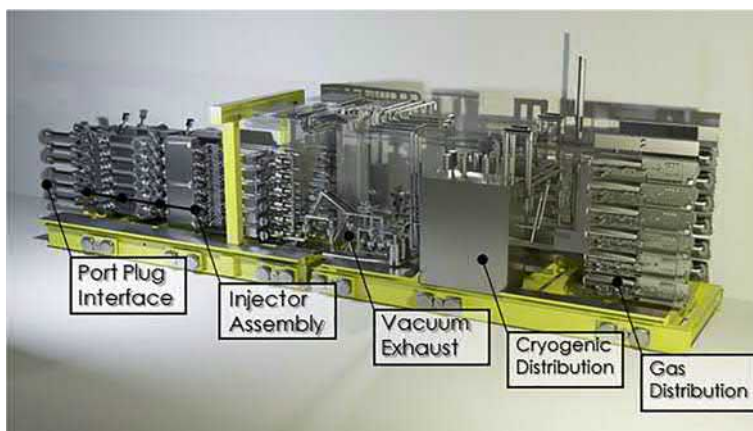


Fig. 11 DMS system to be installed in ITER equatorial port.

reliability of their plasma temperature evaluation method. British scientists at Culham had perfected a reliable method for measuring very high temperatures based on the then novel method of laser light scattering on plasma electrons (Thomson scattering). They brought their Thomson scattering apparatus to the Soviet Union to validate the T-3 performance, which proved successful and confirmed the high temperatures of the T-3 plasmas (Peacock et al., 1969). This blew a fresh wind through fusion research worldwide in the 1970s and was a major breakthrough for the tokamak concept. A quintessential example of function 4, enhancing the understanding of plasma physics, would be sawtooth instabilities. Sawtooth instabilities are relaxations commonly observed in the core of magnetically confined plasmas, which occur quasi-periodically and cause a sudden drop in the temperature and density in the center of the plasma. A theory was created to explain the mechanism of these sawtooth instabilities, that is, until new diagnostics provided contradictory data, which ultimately improved upon and made the theory more concrete. The full story is summarized in reference Wesson (2011).

Various types of plasma parameters need to be measured to ensure functions #1 to #4: Plasma current (I_p), plasma position and geometry, energy (W , β), magnetic field (B), electric field (E), electric potential (ϕ), radiant energy (P_{rad}), ion (i)/electron (e) density (n)/Temperature (T)/flow velocity (v)/distribution function (f), ion species, fusion power (P_{fus}). Further, non-contact measurements are required because of the high temperature of fusion plasmas. Plasma diagnostics can be classified according to their measurement method as follows.

- *Electrical/magnetic diagnostics*: magnetic probes (I_p , LCFS, V_{loop} , β_p), Langmuir probes (n_e , T_e).
- *Systems based on propagation characteristics of electromagnetic waves*: interferometers (n_e), polarimeters ($n_e B$), reflectometers (n_e).
- *Systems based on radiation/scattering of electromagnetic waves*: CYCLOTRON emission diagnostics (T_e), scattering diagnostics (n_e , T_e , n_{α} , $f(v_i)$).
- *Plasma emission diagnostics*: passive spectroscopy (impurity, Z_{eff} , T_i , n_H/n_D , n_T/n_D), active spectroscopy ($B_{\text{pol}}/B_{\text{tor}}$, v , Z_{eff} , n_{He}/n_e), bolometers (P_{rad}).
- *Particle/neutron diagnostics*: electrostatic energy analyzers (n_T/n_D , ϕ), neutron diagnostics (P_{fus} , n_T/n_D), lost alpha particle diagnostics.
- *Plasma-facing and operational diagnostics*: thermocouples, pressure gages, residual gas analyzers, visible light cameras, IR cameras, erosion monitors.

As listed above, a variety of physics principles and technological techniques are used to ensure function #1 to #4. Multiple instruments may be combined to obtain certain specific plasma parameters. For example, a highly accurate determination of safety factor can be calculated by combining measurements made by magnetic probes, interferometers, polarimeters, Thomson scattering diagnostics, and active spectroscopy.

ITER diagnostic systems

ITER will be equipped with 58 diagnostic systems (Vayakis et al., 2012; ITER Physics Expert Group on Diagnostics and ITER Physics Basis Editors, 1999; Donne et al., 2007), and will measure several physical quantities for physics research purposes. Many of these single physical quantities will be measured by using multiple diagnostic instruments having different measurement principles, hence the large number of diagnostic systems.

Each individual diagnostic instrument must be designed in consideration of the following: (1) radiation heat ($< 1 \text{ MW/m}^2$) and nuclear heat ($< 10 \text{ MW/m}^3$) from the plasma, (2) electromagnetic forces ($< 100 \text{ T/s}$) and acceleration load due to plasma disruptions, (3) electromagnetic noise, (4) effects of neutron and gamma irradiation on optical and electronic devices, (5) neutron and gamma shielding for optical and electronic devices, (6) minimization of shutdown dose rate (ALARA (as low as reasonably achievable) principle to minimize exposure to humans), (7) containment of activated pressurized cooling water (application of nuclear pressure vessel regulations), (8) ensuring tritium confinement boundaries are maintained in case of accidents such as earthquakes or fires (e.g., vacuum windows and vacuum electrical feedthroughs), (9) structural designs that are based on nuclear energy standards, (10) position alignment, calibration, and maintenance methods that can be implemented in a radiation environment.

In particular, (4) "the effects of neutron and gamma irradiation on optical and electronic devices" is of special importance for ITER (and any subsequent fusion reactors, such as DEMO). Because of the harsh environment inside the ITER vacuum vessel—where many of the detectors are located—the diagnostic systems must cope with a range of phenomena not previously encountered in diagnostic implementation, while performing with great accuracy and precision. (Thermocouples, bolometers, electromagnetic and neutron diagnostics will be heavily affected by radiation.) The levels of fast neutron flux, gamma ray dose rate, and neutron fluence are respectively about $1 \times 10^{17} \text{ n/m}^2\text{s}$, $\sim 100 \text{ Gy/s}$, and $2 \times 10^{24} \text{ n/m}^2$ behind the shielding blanket. In addition, the diagnostic instruments must be designed in consideration of radiation-induced conductivity (RIC) (Zinkle and Hodgson, 1992; Hobbs, 1994), radiation-induced electrical degradation (RIED) (Hodson, 1992; Shikama and Pells, 1994), radiation-induced thermoelectric sensitivity (RITES) (Nishitani et al., 2004), and radiation- and temperature-induced electromotive forces (RIEMF, TIEMF) (Vayakis et al., 2003). What is more, electronic devices located outside the vacuum vessel (detectors, amps, cameras, etc.) must be radiation hard. The electronics of vital equipment necessary for plasma operation must be shielded so that the total ionizing dose is less than 1 Gy , the equivalent neutron fluence is less than $1 \times 10^8 \text{ n/cm}^2$, and the total neutron flux is less than $0.01 \text{ n/cm}^2\text{s}$ (EEE, 2017). Optical elements such as windows, lenses, and optical fibers must be designed in consideration of the irradiation effect, because their transmittance changes with neutron and gamma irradiation. R&D results for ITER are summarized in Reference (Campbell et al., 2019).

To expand on (5) “neutron and gamma shielding” and (10) “alignment and maintenance procedures,” recall that magnetic confinement reactors use non-contact measurement methods to measure electromagnetic waves and light. Although there are holes in the shielding to allow electromagnetic waves and light to propagate, the optical path must be in the form of a labyrinth to prevent the high neutron flux from leaking out of the vacuum vessel through these holes. The mirrors installed at the bends of the labyrinth optical path are, along with the shielding, located inside the vacuum vessel, where they will become extremely hot because of nuclear and radiation heat. Thus, these components located in-vessel must be made of metals having excellent cooling properties. Glass substrate lenses, prisms, or mirrors cannot be used. This means that optical components of ITER diagnostics are installed inside the vacuum vessel and their transmission optics are to be composed only of water-cooled planar or curved metal mirrors. Besides, deposition of impurities such as beryllium and tungsten on the mirrors are also problem because the deposition causes reduction of reflectance and change of spectral reflectance. Such a feat has never before been achieved by any other experimental device. There are also significant constraints on how these mirrors can be maintained, replaced, or adjusted. The ITER vacuum vessel will be radioactive and inaccessible to humans. The shielding to which the mirrors are mounted weighs more than 20 t, and thus the optics in the vacuum vessel needs to be designed without relying on maintenance, refurbishment or replacement demanding remote-handling removal of 20 t shielding blocks. Also, in consideration of the electromagnetic loads during plasma disruptions, the mirrors are fixed to a rigid mirror mount that cannot be adjusted angularly, except for some special cases. The inability to maintain or replace optics together with the inability to adjust front-end optics angles represent a major challenge in diagnostic development, so as to ensure reliable operation over the lifetime of the experiment. One of techniques developed for the mirrors is in situ mirror cleaning system (Litnovsky et al., 2019). The in situ mirror cleaning system uses plasmas generated by radio frequency to sputter contaminants from the mirror surface.

Diagnostic systems for DEMO

Studies on the diagnostics for the DEMO reactor have begun (Donne et al., 2012; Todd, 2014; Biel et al., 2015, 2019), but no specific designs have been decided. What is certain, though, is that the number of diagnostic instruments installed in DEMO will be significantly less than in ITER or other devices currently in operation. The space in the blanket that can be allocated to diagnostics must be limited to ensure a tritium breeding ratio (TBR) of one or more. The type of instruments that can be installed for front-end optics will be constrained due to the harsh environment (such as neutron radiation, accumulation of impurities and dust, and high temperatures). The “control knobs” for self-sustaining burning plasmas are also limited, so potentially having only a diagnostic that determines whether the plasma is in steady state or not may be sufficient. The following instruments are considered to be the minimum requirements: (1) magnetic diagnostics for plasma current, position control, and for detecting instabilities, (2) line-averaged electron density measurement to control the fuel supply, (3) spectroscopy for impurity and helium monitoring, and (4) neutronic diagnostics to measure the ratio of fuel to total neutron yield. Burn control and control of the detached divertor plasma are also very important factors, but the selection of the minimum instrumentation required for these controls is dependent upon the results of ITER experiments.

Other items that must be factored into DEMO diagnostics are radiation hardness, high thermal loads, high displacement per atom due to neutron flux, measurements during steady-state operation, deposition of impurities and dust, relativistic effects on electromagnetic wave propagation characteristics, attenuation of measurements due to the blanket, and compliance with blanket maintenance procedures. As a specific example relevant to the requirements for DEMO magnetic measurements, magnetic diagnostics in present devices record data by using the signals from pickup coils via integrators. However, since DEMO will operate in steady state for periods of several months, using integrators will result in large errors (integration systems can measure reliably only for periods of up to 1000 s). It is also entirely possible that Hall effect sensors cannot be used because of the radiation environment. Hence, either these integrator errors will have to be corrected somehow, or the in-vessel conditions may have to be estimated from data gathered by Hall effect sensors installed outside the vacuum vessel. Even if the errors of the pickup coils can be corrected, issues may arise from high thermal loads and irradiation effects if the coils are located near the plasma. And if the pickup coils are installed behind the blanket, the magnetic field would be shielded by the blanket, making plasma vertical position instabilities difficult to detect. Even if all of these issues have been resolved, it begs the question: how would pickup coils installed behind the blanket undergo maintenance. These challenges are not limited to only magnetic diagnostics but common to most plasma diagnostics. Considerable R&D is required to develop a reliable diagnostic capability for DEMO and beyond. ITER will provide significant input to this R&D, but it is likely that a variety of developments will have to be pursued in parallel to establish the science and technology foundations to design and construct robust diagnostics for DEMO.

Plasma control

Introduction

A magnetically confined plasma is a complex state which cannot be sustained without a sophisticated feedback control system. In addition, free energy which is available within the plasma tends to generate turbulence and magnetohydrodynamic (MHD) instabilities which reduce plasma confinement quality. In a thermonuclear plasma, with significant fusion power produced by deuterium-tritium reactions, further control requirements are imposed in ensuring that the fusion power can be sustained for extended periods, or even continuously.

Plasma control methodology does not differ from that applied in control systems found in all technologies: it requires a means for measurement of the parameter to be controlled (in the fusion plasma environment via "diagnostic" systems), an "actuator" to modify the required parameter, which for fusion plasmas might be heating and current drive systems, fueling systems or magnetic coils, and an electronic system to determine the required action via an analog or, now more commonly, digital computation. Particular challenges for magnetically confined thermonuclear plasmas (i.e., plasmas primarily heated by DT reactions) include the fact that the physical parameters to be controlled must, in some cases, be derived from multiple diagnostic measurements, which must themselves be adapted to the fusion nuclear environment, and that only a limited number of actuators are available to control numerous parameters, and so each actuator might be shared among several control functions.

Among others, there are three functional requirements for the plasma control: they are (i) plasma control to achieve and maintain high fusion power output stably for long period as described above. This is to be achieved by the PCS. And moreover, (ii) investment protection, and (iii) nuclear safety. These functions of plasma control are incorporated in the control system of a fusion reactor. Failures of such plasma control system lead to conditions which almost invariably tend to reduce fusion power, or extinguish the plasma.

According to the functions required above, 3 levels of hierarchies are applied in the plasma control systems in the nuclear environment of ITER (Aymar et al., 2002; Donné et al., 2021) as an example. The three levels of systems as follows have differing functions and separate implementation:

- (i) Plasma Control System (PCS),
- (ii) Investment Protection System (ICS), and
- (iii) Safety Protection System (Central Safety System, (CSS) in ITER).

Plasma control system (PCS)

The fundamental variables most relevant to achieving the high values of fusion gain, Q , required in a fusion reactor are the ion temperature at the plasma core (T_i), plasma density (n_e), and energy confinement time (τ_E). In particular, the confinement time involves a complex interplay of various distributed quantities and states of the plasma, which are mainly determined by the toroidal magnetic field, position and shape of the plasma, total plasma current, the current density profile, etc. In addition, the fundamental quantitative variables of the plasma are the shape of the last closed flux surface; plasma current density profile; electron density profile (or n_e at a representative point); ion temperature profile (or T_i at a representative point); and electron temperature profile (or T_e at a representative point). All of these physical quantities have associated fusion reaction rates (and their distributions in energy). And so all of these factors must be controlled, either directly or indirectly, to control the fusion output.

These plasma control variables can be determined by directly measuring or processing the signals measured by the diagnostic instrumentation. Rudimentary electromagnetic measurements include plasma current, coil current, poloidal magnetic field, poloidal magnetic flux, toroidal magnetic flux. From these measurements, various equilibrium quantities such as the current, position, and shape of the plasma can be obtained. Using these equilibrium quantities and other physical quantities, such as plasma density, electron and ion temperatures, number of neutrons emitted, electromagnetic radiation from the plasma, impurities, plasma internal magnetic field distribution, the three fundamental variables n_e , τ_E , T_i , (for determining Q value) can be calculated.

Furthermore, plasma behavior determined by fusion reactions (Kurihara et al., 2008) is characterized by:

- (a) high self heating (70–90%) and high bootstrap current fraction, the interactions of which make for a self-sustaining system;
- (b) a system in which local and global interactions and fluctuations with different time scales (MHD, turbulence) intertwine. A wide range of time constants are also prevalent, for example, ideal MHD (typical time scale in the order of microseconds), energy confinement time (seconds), current diffusion time (in the order of hundreds of seconds), and particle wall saturation (in the order of thousands of seconds). ITER is expected to operate in a range of "advanced" plasma regimes in order to meet its mission goals. For example, in "ELMy H-mode" plasmas, MHD instabilities localized near the plasma edge (ELMs) that produce periodic loss of energy from the edge plasma must be controlled, while in plasmas having negative magnetic shear, which are characterized by internal transport barriers, the current profile shape must be controlled for stable long-pulse operation (ITER Physics Expert Group on Disruptions, Plasma Control, and MHD et al., 1999; ITER Physics Expert Groups and ITER Physics Basis Editors, 1999).

Turning to the plasma control methods themselves, PCS of fusion reactors must have the capability to;

- (a) adjust actuator commands in response to various changes in the plasma to sustain the planned operation; and
- (b) to terminate the plasma in a controlled fashion in the event a plasma cannot be adjusted back to the standard operating mode.

Using feedback control, plasmas must be precisely maintained in, or adjusted to, their optimal state to achieve high fusion performance, which requires the use of highly customized computer systems for real time control (Kurihara, 1993). Some control methods combine control techniques, for example by switching between:

- (a) Simple setup control: a waveform of actuator outputs are given directly in advance (pre-programmed waveform: ppw) and executed during plasma discharges, and
- (b) Tracking control: a waveform of certain plasma parameters are given in advance (ppw) and actuators are controlled to follow these pre-programmed plasma parameter waveform.

In the following are some example of plasma control for events with specific time scales.

Fast control of plasma: Equilibrium control

"Equilibrium control" is a control of the plasma current, position, and shape, where the poloidal field (PF) coils (including the central solenoid (CS)) are used as actuators. From all appearances with a time constant of about 0.1 ms, we can approximate that a microscopic dynamic state of equilibrium always exists at every point in a plasma. In addition to the microscopic equilibrium of the forces in a plasma, the macroscopic plasma motion is an important factor in equilibrium analysis because a magnetic field is formed when magnetic lines of force pass through the conductors, which can induce eddy currents in structures such as the vacuum vessel.

The macroscopic objective of equilibrium control is to control the plasma current and plasma shape.

- (a) Plasma initiation and plasma current control: ppw commands for the Center Solenoid voltage is sent to a power supply, and its output is applied as a one-turn voltage for plasma start-up, and subsequently, the plasma current ramp-up rapidly. Feedback control is applied according to the given ppw of plasma current after a few tens of milliseconds from plasma initiation until the discharge ends.
- (b) Plasma shape and position control: Feedback control is performed in accordance with the predetermined position and shape of the commanded input ppw. A detached plasma, so called the divertor configuration is achieved accordingly in this series of plasma shape and position control.

The equilibrium control in the JT-60U was a typical feedback control on a basis of proportional-derivative (PD) control, in which the control signal was generated in proportional to the deviations between the target ppw of plasma current, position, and shape and the measured values. PD gain matrices are used so that a multivariable linear feedback control method can be applied.

By controlling the plasma current profile precisely during start-up, it is well known that a plasma current profile with central hollow current, can be obtained. This is so called negative magnetic shear configuration, where an internal transport barrier (ITBs) is formed in the plasma leading to a high plasma confinement performance. Using this method, a fusion energy gain factor of 1.25 (when converted for the DT-equivalent plasma) was achieved in the JT-60U. Thus, the advancement of plasma equilibrium control is a key to achieving and maintaining high performance plasmas. And hence, now a days, it has been established that the plasma shape and position, and equilibrium parameters are precisely reproduced in real-time by the method actually employed in the experiment ([Kurihara, 2000](#)).

Comparatively slow control of plasma

"Density control" is one of the three indicators of plasma performance, and high plasma density is required to achieve high performance plasmas. The method for controlling plasma density is to manipulate the amount of fuel (deuterium and tritium) injection by gas or cryogenic pellets into the main plasma and apply feedback control so that the average values of plasma electron density match the expected waveform. The fuel injection can be achieved rather quickly in the order of less than seconds. However, the gas pumping and wall saturation requires longer time scale of up to minutes to hours. The density control together with particle exhaust including helium ash is a future issue to be tackled in long pulse operations of magnetic confinement fusion machines.

The "fusion reaction rate control" corresponds to power control in a fusion device. The power of the heating systems, such as neutral beam injectors (NBI), is adjusted and feedback control is applied so the increased neutron generation rate, caused by the increase in plasma ion temperature, matches the expected waveform.

As previously mentioned, the plasma heating by alpha particles in fusion reactors is high (70% in ITER and 90% in prototype reactors such as DEMO). This will require rather meticulous and maybe non-linear control, since the pressure profile, which determines the fusion power profile (and hence the fusion power profile equal to the alpha-heating profile), is itself dependent on the alpha-heating profile.

In steady-state reactors, plasma current must be driven in a steady-state manner without using electric fields that are induced by poloidal field coils or Center Solenoid. Thus, non-inductive current drive is achieved by electron cyclotron heating (ECH), ion cyclotron heating (ICH), and N-NBI (negative ion based neutral beam injectors), which act as the actuators. In the negative magnetic shear regime, the direct external current drive is to be reduced by increasing the fraction of bootstrap current, that is, to drive additional bootstrap current in the plasma by creating a large pressure gradient in the plasma. This will again require sophisticated control against non-linear variation of plasma current.

Investment protection system (IPS)

Tokamak machines built recently, such as JT-60SA and ITER, are large investment with superconducting magnets, powerful heating systems capable of operating long pulses. So the control system of such machines equip investment protection functions.

In ITER as an example, the Investment Protection System is provided in the control system, which is largely implemented in a separate, more "robust" hardware and software framework than PCS. But even PCS is utilized for the investment protection: The PCS detects whether the plasma has strayed too far from the planned operating regime and so the PCS decides to implement a controlled shutdown (which might avoid a later disruption). These actions can also be considered as an element of the investment protection system, through "defense in depth." In ITER the IPS is primarily implemented through the Central Interlock System (CIS). CIS also triggers the Disruption Mitigation System, though most of the triggering decisions are made by PCS.

It should be noted here that neither PCS nor CIS are formally part of the “safety” protection system, except in the sense that it is the first step in the “defense in depth” philosophy which is an element of the overall safety framework.

Safety protection system

Safety Protection System in ITER, so called Central Safety System, (CSS), applies exclusively to personnel, population, and environmental safety. This uses even more robust and simpler and independent hardware and software systems than PCS and CIS.

In relation to the plasma shutdown requirement at abnormal events, the safety control system could be structured as:

(i) Controlled slow or fast equilibrium shutdown via PCS;

In the case of an abnormal event in a plant, the power injection systems, such as the heating systems, are stopped and the plasma current is shutdown immediately by using an equilibrium control system. In case of failures in control systems, coils, power supplies, or plasma anomalies, if plasma position and shape control is functional, the plasma is controlled so that it does not come into contact with the wall and the energy of the plasma is gradually discharged by resistive heating or radiation outflow from a plasma.

(ii) Fast plasma termination via “disruption mitigation system (DMS),”

In tokamak plasmas, “disruptions,” the sudden termination of a plasma due to a rapidly growing instability, must be taken into account. Comparatively slow disruptions require plasma current control to be performed for as long as possible to match the decreasing plasma current and recover as much of the plasma’s magnetic energy as possible. In particular, coil current must be controlled at acceptable levels so as not to exceed PF coils allowable stress.

Sudden disruptions rapidly change the position and shape of the plasma, which can induce currents in in-vessel components, generate large magnetic forces on the structures surrounding the plasma, and dump a lot of heat on the vessel wall, all at the same time. Disruptions must be avoided at all costs, but nevertheless must be planned for should an abnormality occur. For fast disruption, ITER CSS can directly intervene to shutdown the plasma to satisfy *safety* requirements (essentially using a very simple approach to “disruption mitigation”).

A disruption mitigation system (DMS in ITER) is mediated by the plasma control, but physically triggered by the interlock control, as it is essentially an investment protection measure. One could also note that in a reactor, this is essential, but that the plasma operation should be organized so that this needs to be used very rarely.

The development of disruption avoidance control and mitigation systems is paramount. Research is already underway on applications of artificial intelligence (AI) and real-time processing of instability prediction. A system that can catch the signatures of disruptions and release the energy in the plasma via radiative process by injecting impurities such as D, He, Ar, thus preventing the in-vessel components from being damaged, is also in development.

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Fusion—Reactor Materials

A. Litnovsky^{a,b}, I. Duran^c, J.W. Coenen^{a,d}, Yu Gasparyan^b, M.R. Gilbert^e, E. Hollmann^f, Ch Linsmeier^a, S. Nogami^g, C.H. Skinner^h, and S. Zinkleⁱ, ^a Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung, Jülich, Germany;

^b National Research Nuclear University MEPhI, Moscow, Russian Federation; ^c Institute of Plasma Physics of the Czech Academy of Sciences, Praha, Czech Republic; ^d Department of Engineering Physics, University of Wisconsin Madison, Madison, WI, United States; ^e United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, Oxfordshire, United Kingdom; ^f University of California—San Diego, La Jolla, CA, United States; ^g Department of Quantum Science & Energy Engineering, Graduate School of Engineering, Tohoku University, Sendai, Japan; ^h Princeton Plasma Physics Laboratory, Princeton, NJ, United States; and ⁱ University of Tennessee, Knoxville, TN, United States

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Introduction

The aim of controlled fusion is to produce electricity from the energy released by fusion reactions with a competitive cost efficiency, maximum safety and with least environmental impact. In order for nuclear reactions to become effective, the fusion particles of relevant energies in keV range must be brought together at sufficient concentration for sufficient time defined by Lawson criterion (Lawson, 1957), as discussed in detail in Wade (2021). The Lawson criterion defines the condition, at which the energy balance is positive by comparing the energy released in fusion reactions to the energy lost to the environment. Quantitatively, for deuterium (D) and tritium (T) reaction, assuming 50%–50% mixture and temperature above 10 keV (1.16×10^8 K), the Lawson criterion can be written as:

$$n\tau_E > 10^{20} \text{ s} \times \text{m}^{-3}, \quad (1)$$

where n is the number density of reacting particles and τ_E is the energy confinement time. From the formula (1) we may see that there are two possible ways to attain the positive balance of energy gained from fusion compared to energy losses:

1. by increasing the confinement time at the given density of reacting particles;
or
2. by increasing the density of reacting particles possibly, interacting within the very short time.

These fundamentally different approaches have led to two main technological concepts in realizing the controlled production of nuclear fusion energy: magnetic confinement concept and inertial confinement concept.

In magnetic confinement fusion (MCF) devices, such as tokamaks (Artsimovich, 1972; Wesson and Campbell, 2011), described in detail by Zohm (2021) and stellarators (Spitzer et al., 1954), discussed by Yamada (2021) and relatively rarely, magnetic traps (Ioffe, 1965). The long-pulse or quasi-stationary fusion plasmas are confined in the vacuum vessel by a strong magnetic field. In tokamaks and stellarators, such a magnetic field is created using the external coils surrounding a donut-shaped vacuum vessel. The variety of magnetic configurations is discussed in detail by Galvao (2021).

Inertial fusion (IFE) relies on the exposure of deuterium—tritium targets, implosively compressed and heated by intensive laser or ion beams. In the IFE devices, plasma is free to expand after the implosion of the D-T target. More details on inertial fusion concepts and underlying physics can be found e.g. in [Atzeni \(2021\)](#).

In both concepts, MCF and IFE, fluxes of fusion plasma particles and intensive radiation come into contact with materials of fusion device. Depending on their function, several material classes are used in the components of the fusion facility illustrated schematically in [Fig. 1](#):

Plasma-facing materials. These materials are exposed to plasma particle and radiation fluxes directly. The role of these materials is to protect the inner structure of the fusion device and to guarantee the low level of plasma impurities in the plasma.

Blanket materials. These materials are located behind the plasma-facing materials. Their main role is to ensure tritium breeding needed for “refilling” the fusion plasma with tritium and to conduct the heat generated by the products of fusion reaction to the coolant—in order to produce electricity from it later.

Diagnostic materials. These materials are used for multiple measurement (i.e. diagnostic) systems of a fusion device. The main role of diagnostic materials is to transfer the information: electromagnetic radiation (e.g. light) from the main fusion plasma towards the corresponding detectors and sensors.

Materials for magnets. These materials are used in the main field magnets and in various magnetic subsystems needed for plasma confinement, positioning and stabilization and for distribution of plasma loads.

Structural materials. The role of these materials is to support the entire first wall, blanket, diagnostic infrastructure and magnets in the most robust and reliable manner. Structural materials are also used to provide the required vacuum conditions in the fusion facility. These materials have to preserve the entire infrastructure integrity under challenging pressure, temperature and seismological conditions.

Magnetic confinement fusion devices

Edge plasma and scrape-off layer

In magnetic confinement fusion devices, in most of the volume occupied by the plasma, the trajectories of plasma particles never intersect the wall of plasma vessel, creating the so-called confined plasma volume, as illustrated in [Fig. 2](#). The total plasma thermal energy accumulated in the central, core confined plasma in a modern tokamak such as ITER ([ITER Fusion Reactor Website, 2020](#)) can be as high as several hundreds of megajoules (MJ) ([Shimada et al., 2007](#)), whereas in a fusion power plant thermal energy is

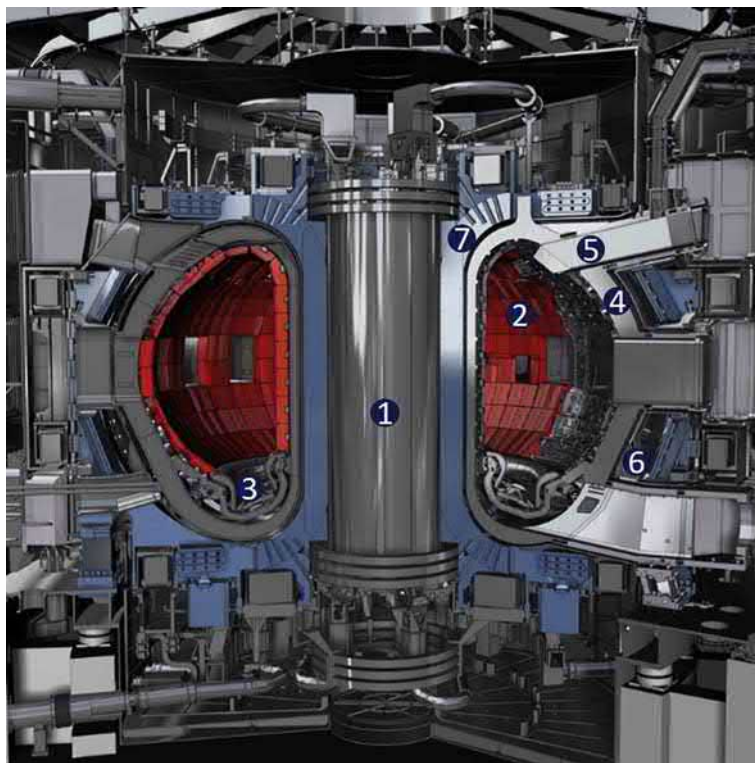


Fig. 1 Main components of the MCF device: (1) central solenoid, (2) plasma-facing first wall, (3) divertor, (4) blanket, (5) one of diagnostic ports, (6) inner infrastructure and (7) magnetic coil. Figure courtesy of ITER Organization.

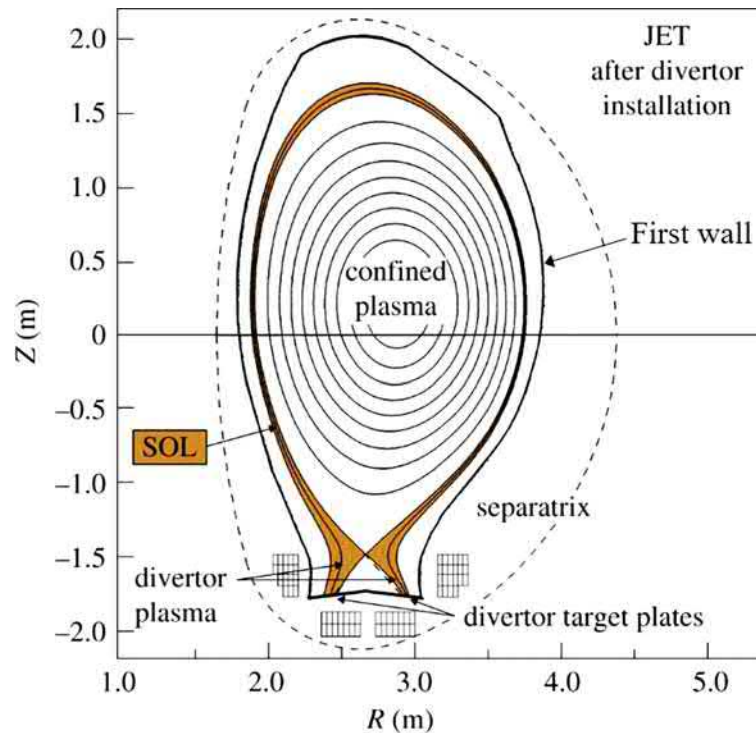


Fig. 2 Main areas occupied by fusion plasmas in a magnetic fusion device: a cross-section of JET tokamak with outlined scrape-off layer (SOL) in between the main confined plasma and the first wall. Adapted from Costley AE (2019) Towards a compact spherical tokamak fusion pilot plant. *Philosophical Transactions of the Royal Society A: Mathematical, Physical and Engineering Sciences* 377(2141). doi: 10.1098/rsta.2017.0439.

expected to be of order of one gigajoule (GJ) (Federici et al., 2017, 2018). To confine large accumulated energy, magnetic fields with strengths of several Tesla are used in modern magnetic confinement fusion devices. However, complete confinement cannot be achieved even with the most modern magnet technology.

Due to processes such as collisions between particles and turbulence, heat and particles are transported outwards across the separatrix. In the narrow region outside the confined volume towards the wall, the plasma particles and the heat get in contact with the wall. This happens in the thin, so-called “Scrape-off layer, SOL” between the confined plasma and the vessel wall (Stangeby, 2000) as shown in Fig. 2 adapted from Costley (2019). Particles which cross the separatrix onto the open magnetic field lines in the narrow, typically < 1 cm, SOL can then be transported rapidly to the divertor. A smaller fraction diffuses further outwards and these particles are intercepted by the first wall.

Stationary and transient power loads

The plasma energy in the SOL despite effective magnetic confinement, is nevertheless high enough to cause severe damage to the materials facing the plasma. The most severe stationary power loads are expected to occur in the divertor of magnetic fusion devices at the so-called target plates shown in Fig. 2. There the highest heat loads of the order of $10\text{--}20 \text{ MW} \times \text{m}^{-2}$, are expected in steady-state (Federici et al., 2001; Loarte et al., 2007; Pitts et al., 2011).

Moreover, instabilities in the plasma can lead to significant pulses of heat and particles being ejected into the SOL. In tokamaks, severe adverse effects may be caused by the so-called Edge-Localized Modes or ELMs (Hill, 1997; Stangeby, 2000) formed during the high confinement regime or H-mode (Wagner et al., 1982) and reviewed in greater detail in Zohm (2021) and in Jenko (2021). ELMs originate from plasma instabilities and eject the energy from the central confined plasma, shown in Fig. 2, towards the wall components (Stangeby, 2000). Edge-Localized Modes are capable of delivering several megajoules of energy within milliseconds to the wall components (Hill, 1997; Stangeby, 2000; Loarte, 2006) which can lead to localized melting and ablation of plasma facing surfaces at the divertor targets (Federici et al., 2017, 2018). Significant research activities have been implemented to establish means for either controlling the ELM frequency and amplitude, e.g. (Rapp et al., 2002; Baylor et al., 2012; Evans, 2013; Lang et al., 2013) or suppressing them completely, as described e.g. in (Evans, 2013; Loarte et al., 2015; Suttrop et al., 2018). ELM suppression is seen as a crucial aspect of designing plasma scenarios for a fusion reactor.

In addition to ELMs, tokamak disruptions can cause severe divertor thermal loads. Disruptions are global instabilities which can release the entire plasma stored energy on millisecond timescales, resulting in surface heat loads approaching $1 \text{ GW} \times \text{m}^{-2}$. Extensive research is underway to predict, avoid, and mitigate disruptions in tokamaks (Hollmann et al., 2011; Lehnen et al., 2015).

Both the quasi-stationary power loads and transient power loads must be controlled carefully to ensure reliable operation of the plasma and to limit heating and erosion of the plasma facing surfaces thereby ensuring an acceptable lifetime for these surfaces.

The impinging plasma particles and intensive radiation from the plasma trigger complex plasma-material interactions (PMI) which will be described in detail later in the article. In the course of this explanation, we will review the particular impact of these processes on materials and describe the material concepts in use in present-day magnetic confinement fusion devices. Furthermore, we will describe solutions currently foreseen for the next generation of fusion devices and challenges to be addressed on the way to building a fusion power plant. We will review all material classes of magnetic fusion devices in this article. Due to the versatility of plasma-material interaction processes, their complexity and their crucial impact on the lifetime of fusion reactor, the plasma-facing materials will be discussed in greater detail.

Material challenge in magnetic confinement devices

Plasma-facing materials

Plasma—Material interaction processes

Plasma-material interaction (PMI) plays a crucial role in the choice of candidate materials suitable for the environment of fusion facilities. Among the most important are the following PMI processes:

- Sputtering of the wall materials by plasma particles
- Re-deposition of sputtered material, dust formation and transport
- Thermal loads to the plasma-facing components which can cause material melting
- Accumulation of radioactive tritium from the fusion plasma particles in the wall materials and its possible transport inside the materials of a fusion reactor
- Surface modifications and microstructure changes caused by the plasma particles and radiation

The main PMI processes see e.g. (Kirschner et al., 2016) are schematically shown in Fig. 3 (Kirschner, 2020) for the examples of beryllium (Be), molybdenum (Mo) and tungsten (W).

Physical sputtering

Electrons and ions in the core confined plasmas have energies of order of several keV. In the scrape-off layer plasmas, both electron and ion energies decrease significantly. Electrons, due to their low mass, cannot cause sputtering of material. In some cases the residual energy of plasma ions can exceed the threshold energy needed for sputtering of the wall material. The physical sputtering coefficient in the binary collision approximation depends on the masses of impinging and target particles as shown in the pre-factor for energy transfer:

$$\gamma = 4 m_t m_p / (m_t + m_p)^2 \quad (2)$$

where m_t and m_p are the masses of the target material and the projectile respectively (Eckstein et al., 1993). Therefore, the most efficient sputtering occurs in the case of interaction of impinging and target particles of similar masses. Data on sputtering coefficients for most fusion materials can be found e.g. in Eckstein et al. (1993). The sputtering process leads to the reduction of the thickness of the plasma-facing material (PFM), which may adversely affect the lifetime of the entire component. The lifetime estimates for beryllium plates in the ITER fusion reactor (ITER Fusion Reactor Website, 2020), e.g. predict a surface recession of order of 0.01 mm for each plasma discharge due to sputtering (Borodin et al., 2011, 2019). Physical sputtering is frequently followed by processes of re-deposition of sputtered material and its subsequent re-erosion, leading to the stepwise process of so-called material migration in the fusion device.

The ejection of sputtered atoms into the edge plasma leads to core plasma contamination causing an adverse impact on plasma performance (Gruber et al., 2009; Mayer et al., 2009; Neu et al., 2009; Philipps et al., 2010; Tsitrone et al., 2011). Whereas sputtering increases with decreasing target mass, the plasma contamination impact increases super-linearly with increasing mass of the ejected atoms. Therefore, either low mass, i.e. minimized plasma impact or high mass i.e. minimized sputtering rate PFMs are preferred.

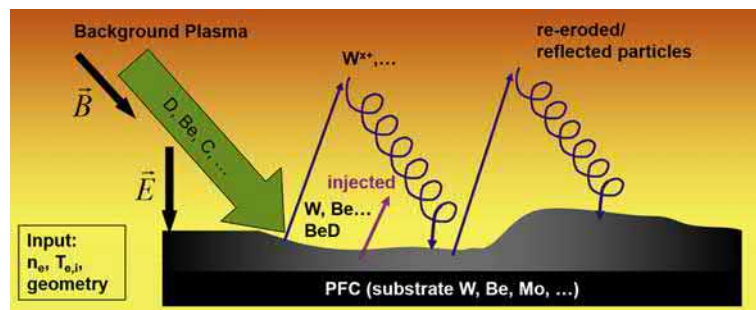


Fig. 3 Main processes of plasma-material interaction. Kirschner A (2020) PWI processes. Private communications.

Chemical erosion

The main plasma particles and the wall materials may possess a chemical affinity to each other and thereby undergo chemical reactions and build chemical compounds. The wall material will be then “consumed” in forming these compounds i.e. it will be eroded by chemically active particles. The corresponding process is called chemical erosion. The detailed description of chemical erosion can be found e.g. in Roth (2005). One of the most pronounced examples of plasma-facing materials prone to chemical erosion is carbon. Deuterium and tritium ions and atoms from the plasma interacting with carbon wall readily produce a variety of volatile hydrocarbons (Haasz et al., 1987; Pospieszczyk et al., 1999; Philipps et al., 2003; Roth et al., 2004; Jacob, 2005; Brooks et al., 2014). The hydrocarbons formed are then transported into the plasma and further to plasma-remote areas, such as e.g. pumping ducts. Chemical erosion processes are temperature-activated and their resultant erosion rate can even exceed that of physical sputtering. This happens e.g. in cases when the energies of impinging ions are still below the threshold for sputtering, whereas the temperature of the material and surrounding environment is already sufficient for chemical reactions to occur (Von Keudell and Jacob, 1996; Bohmeyer et al., 2005; Rudakov et al., 2007; Litnovsky et al., 2008; Wong et al., 2013).

Chemical reactivity is, strictly speaking, not necessarily an adverse effect in fusion plasmas. For instance, the chemical affinity of beryllium (Be) to oxygen—a common impurity in fusion plasmas—is effectively used in the JET tokamak to bind the oxygen to the beryllium walls thus forming an extremely stable and hard melting beryllium oxide, removing the oxygen from the plasma and thus purifying the edge plasma.

Fuel retention

Besides the material loss via chemical erosion, compounds formed may contain the plasma fuel particles: deuterium and tritium. Fuel accumulation represents a severe issue, since radioactive tritium e.g. in the form of volatile tritiated carbon compound, may travel long distances in the vacuum vessel until it finally reaches remote areas. Cleaning and conditioning of these remote areas are frequently challenging or even impossible. With time, such material transport and fuel co-deposition will lead to the undesirable accumulation of the radioactive fuel in the reactor (Dittmar et al., 2011; Litnovsky et al., 2011b; Koivuranta et al., 2013; Pégourié et al., 2013). Besides the radioactivity issue, transport of tritium to remote areas contributes to the loss of this costly element, which is indispensable for sustaining fusion reactions.

Some materials, like tungsten, have a low efficiency of tritium accumulation. Here, another issue may arise though. Once implanted into the material, tritium may start to diffuse deeper into the bulk and eventually penetrate into the coolant of the corresponding cooling systems. Increased chemical reactivity leading to a rapid degradation of the performance of cooling material in the tritium-containing environment is a known effect (Hayashi et al., 2006).

Fuel retention, accumulation and transport impose therefore, a significant concern. A set of measures is foreseen in order to minimize the tritium-driven effects. Certainly, a minimal affinity to tritium is desired from the plasma-facing materials and hence, low tritium retention. It is also known that increased temperatures lead to the prompt desorption of accumulated hydrogenic atoms (Roth et al., 2012; Atsumi et al., 1985; Brezinsek et al., 2013; Ryabtsev et al., 2016; Maier et al., 2019; Zlobinski et al., 2019). It will therefore be desirable to keep the operational temperature of the plasma-facing materials at the level of several hundreds of degrees centigrade. As an example, the first wall temperature of the future DEMO fusion power plant is expected to be in the range 600°C–730°C (Igitkhanov et al., 2013). Once tritium has penetrated into the bulk of plasma-facing materials, its transport can be stopped at the interface to the underlying structural material. For this purpose, so-called permeation barriers (Livshits et al., 1990; Levchuk et al., 2004, 2007; Pisarev et al., 2006; Nemanič, 2019; Houben et al., 2020) are being designed. The permeation barriers act, in a sense as a filter for tritium, preventing it from entering the structural material.

Recrystallization and melting

The heat loads on the PFMs can lead to temperatures significantly exceeding melting temperatures of corresponding materials. Plasma-facing materials in the divertor are at most risk. Melting in the divertor area has been extensively studied both experimentally (Coenen et al., 2011, 2015; Matthews et al., 2016; Krieger et al., 2018) and by modeling (Bazylev et al., 2011, 2014). However, accidental melting in the main chamber has also been reported (Matthews et al., 2016). A photograph of a re-solidified melted fragment from the inner wall limiter in JET is presented in Fig. 4. In tokamaks, severe adverse effects may be caused by the ELMs and disruptions described above.

Melt damage is dangerous in itself. After melting, the damaged area often solidifies in a shape which protrudes towards the plasma and intensifies melting and splashing of molten material during the subsequent plasma operation. Such behavior was observed in melting experiments reported both in divertor (Coenen et al., 2015, 2017; Matthews et al., 2016; Krieger et al., 2018) and limiter tokamaks (Sergienko et al., 2007; Bazylev et al., 2011; Coenen et al., 2011). The molten layer experiences current flowing through it. Being located in a magnetic field, the motion of the molten layer is dominated by the $\mathbf{J} \times \mathbf{B}$ force which, depending on geometry, may even exceed the gravity force (Sergienko et al., 2007; Coenen et al., 2011; Matthews et al., 2016). The molten layer of e.g. heavy tungsten may exhibit an unusual property of moving upwards, depending on location of the molten surface.

Resistance to thermal loads and the corresponding avoidance of surface damage and melting is not solely a function of specific material choice, but also of material geometry. Shaping of plasma-facing components becomes necessary for fusion devices. Here, experience from dedicated studies in MCF devices (Litnovsky et al., 2005b, 2011b; Tanabe et al., 2005; Wong et al., 2007; Krieger et al., 2007; Litnovsky et al. 2007a,b, 2009; Rudakov et al., 2009b; Ding et al., 2014; Hong et al., 2014) coupled with modeling (Dejarnac and Gunn, 2007; Komm et al., 2011; Matveev et al., 2014), can be harnessed for construction of the advanced shapes



Fig. 4 A photograph of a re-solidified area of a beryllium plasma-facing component in the JET tokamak following a melting event. Matthews GF, et al. (2016) Melt damage to the JET ITER-like wall and divertor. *Physica Scripta* T167: 14070. doi: 10.1088/0031-8949/t167/1/014070.

required for future plasma-facing components (Litnovsky et al., 2007b; Pitts et al., 2011, 2019; Stangeby, 2011; Hirai et al., 2016). An example of shaping of the tungsten plasma-facing components in the ITER divertor is provided in Fig. 5.

Dust formation and transport

Co-deposition of sputtered material with the fusion fuel, sometimes followed by prompt re-erosion of freshly deposited layers, their further re-deposition and hence, the further transport of the material frequently leads to the formation of thick deposits. Adhesion of those deposits to the surface is usually rather poor, which leads to their flaking. Flakes or larger fragments of material are usually called dust. The dimensions of dust particles varies greatly, from several tens of nanometers to several millimeters. Hence the effects caused by dust are rather diverse: dust may be fairly harmless, disappearing in the remote areas of the machine and sometimes, not even being noticed and, in the other extreme, dust can cause severe plasma contamination, finally leading to radiative collapse and to a plasma disruption and quench (Winter, 1998, 2004; Krasheninnikov et al., 2005, 2008; Rudakov et al., 2009a; Rubel et al., 2018).

The effects of dust were extensively investigated using various diagnostic techniques (Smirnov et al., 2009; Bergsaker et al., 2011; Kantor et al., 2013; Litnovsky et al., 2013; Shalpegin et al., 2015; Tolas et al., 2016). Dedicated studies show that the fraction of dust mobilized from wall surfaces into the plasma is rather small and not exceeding 20% of the entire amount of dust (Litnovsky et al., 2013). Most dust particles travel to the pump ducts or become stuck in hidden areas and gaps of the first wall and divertor. At the same time, dust mobilized into the edge plasma may travel distances as large as several meters in the plasma volume before they are ablated (Krasheninnikov et al., 2005, 2008; Smirnov et al., 2007; Litnovsky et al., 2013; Lazzaro et al., 2020). Dust entering the core plasma volume is rather rare and its effect usually vanishes after several milliseconds of plasma discharge (Rudakov et al., 2009a; Litnovsky et al., 2013). If dust does enter the core plasma, its ablation will provide a massive source of impurities which causes

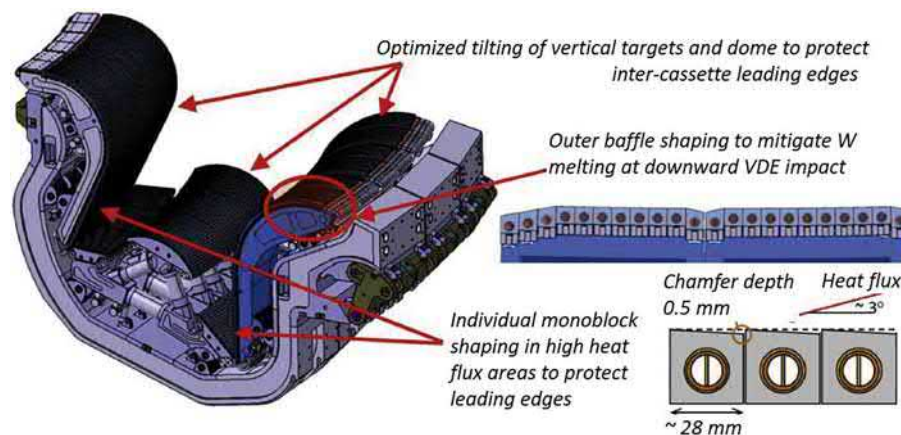


Fig. 5 Shaped plasma-facing components of the tungsten divertor of ITER. Hirai T, et al. (2016) Use of tungsten material for the ITER divertor. *Nuclear Materials and Energy* 9: 616–622. doi: 10.1016/j.nme.2016.07.003.

unacceptable radiative losses and eventually, leads to plasma collapse due to the loss of density control. The amount of allowed impurities in the core plasma is negligible. For instance the allowable amount of tungsten estimated for a core plasma of ITER is limited to some 0.001% of the main fuel particles (Pütterich et al., 2010). The detected re-mobilization of dust particles increases the residence time of dust in plasmas and degrades the plasma performance further (Ratynskaia et al., 2013; Tolias et al., 2016).

Besides the issues of radiation and plasma control impact, dust may impose a radioactivity hazard by the accumulation and transport of tritium to remote areas. Furthermore, because of its small size, great particle number and hence, large total reaction area, under some conditions dust may pose an explosion hazard, as reviewed in e.g. Grisolia et al. (2009) and may introduce safety hazards during off-normal events that mobilize the in-vessel inventories (Sharpe et al., 2002, 2005; Petti et al., 2006; Skinner et al., 2008). Dust handling represents a cornerstone element in licensing of future fusion facilities (Van Dorselaere et al., 2012; Perrault, 2016; Wu et al., 2016).

Surface and structure modifications, blistering and fuzz formation

Interaction with plasma particles leads to a series of changes both in microstructure and on the surface of plasma-facing materials (Ueda et al., 2014). Corresponding investigations are underway both in the MCF devices (Bernard et al., 2015; Tokitani et al., 2017) and laboratory facilities (Baldwin and Doerner, 2008). Dedicated modeling (Krasheninnikov, 2011; Martynenko and Nagel', 2012; Trufanov et al. 2015) is used to address these changes and to reveal the underlying physics mechanisms.

Plasma-material interaction may lead to e.g. formation of bubbles and blisters (Shu et al., 2007a,b; Miyamoto et al., 2009; Kajita et al., 2011; Yamagiwa et al., 2011; Juslin and Wirth, 2013; Ueda et al., 2014), needle-like structures, sometimes called whiskers (Chih et al., 2011; Doerner et al., 2014; Kreter et al., 2014) and the more complicated structures, called fuzz, see e.g. Ueda et al. (2014). The formation of blisters and fuzz are among the most widespread effects and presently attract the most attention. Blisters are cavities created by implanted and diffused plasma particles: H, D, T and He accumulated beneath the surface of metals. Blisters are therefore filled with a gas. The size of blisters varies depending on environmental conditions, ranging from several nanometers up to several hundreds of micrometers. Depending on conditions, blisters can either grow or break leading to the gas release and, more importantly, trigger crack formation (Hong et al., 2014). Numerous studies (Shu et al. 2007a,b; Miyamoto et al., 2009; Hong et al., 2014; Ueda et al., 2014) were conducted showing that blister evolution is very sensitive to the elemental composition of impinging particles (Shu et al., 2007b; Ueda et al., 2014), their energy (Buzi et al., 2014), structure of plasma-facing material (Manhard et al., 2017), the temperature of the material (Ueda et al., 2014) and its crystal orientation (Shu et al., 2007a). A concise physics understanding and evaluation of blister formation are still to be attained.

Fuzz formation is another effect triggered by plasma-material interaction. Under specific conditions of high flux and correspondingly, high fluence of helium ions, and at surface temperatures in the range of several hundred degrees centigrade, the formation of the thin, nanometer-sized and long (hundreds of nanometer in length) column-like structures—"fuzz" is routinely observed (Baldwin and Doerner, 2008; Kajita et al., 2009; Gasparyan et al., 2016). Fuzz is usually formed on the surface of heavy metal plasma-facing materials, like tungsten. Unlike blisters with clearly detrimental effect on consistency and mechanical properties of the exposed material, there is no consistent opinion on the role of the fuzz in plasma-material interaction. On one hand, the surfaces with fuzz are expected to be vulnerable to heat loads (Ueda et al., 2011; Brezinsek et al., 2017b). On the other hand, modeling studies e.g. Krasheninnikov (2011) predict the fuzz in fact, to be more stable than the solid substrate material to the sputtering due to relatively small partial area of the fuzz, which can be hit by plasma particles and become sputtered without a prompt re-deposition to the neighboring fuzz structure. As in the case for blisters, the formation of fuzz is highly dependent on environmental conditions in a fusion device. Presently, it is assumed that the growth of fuzz is dependent on competing erosion of fuzz structure and its annealing due to transient heat loads (De Temmerman et al., 2019). The formation of deposits covering fuzz was detected on the tungsten surface of a material test-sample in the divertor of ASDEX Upgrade (Brezinsek et al., 2017b), whereas the pre-grown fuzz placed in the outer SOL plasmas of TEXTOR vanished within a few plasma discharges (Ueda et al., 2011).

Plasma-facing materials: Material choice

The processes of plasma-material interactions discussed above represent a severe challenge for candidate materials needed for the next generation of MCF devices like ITER and further, for a fusion power plant. The optimum plasma facing material should simultaneously possess several, in part contradictory, properties in order to be compatible with the harsh fusion environment. The most important, almost compulsory, properties include:

- Low tritium retention
- Low sputtering yield by plasma particles
- High thermal conductivity at elevated temperatures above 200°C
- As low as possible neutron-induced radioactivity in fusion environment
- Wherever possible, low chemical activity
- Wherever possible, ability to capture or bind plasma impurities

Here we briefly review past, current and prospective candidate materials and concepts.

Carbon and carbon composites

For several decades, carbon (Merola et al., 2004) was one of the most widespread PFMs for fusion devices (Barabash et al., 1999; Duffy, 2009). Its extraordinary properties, such as absence of melting, very high thermal conductivity, even further enhanced in the so-called carbon fiber composites (Snead and Ferraris, 2020), low brittleness and hence, easy machinability are some of the most attractive features of carbon. However, carbon is prone to chemical erosion of carbon (Mech et al., 1998; Philipps et al., 2003; Roth, 2005) described above. Volatile hydrocarbons can travel long distances. Since radioactive tritium reacts with carbon, the so-called long range transport (Wienhold et al., 1999, 2003; Matthews, 2005) can lead to tritium accumulation in poorly accessible areas of the fusion reactor. Being a light element, carbon is prone to physical sputtering by plasma particles (Eckstein et al., 1993). These properties and their implications for reactor operation, have ultimately disqualified carbon as a prospective fusion material (Neu et al., 2009; Pitts et al., 2011; Hong et al., 2015).

Beryllium

Beryllium (Be) has received significant interest as a plasma-facing material within the fusion community, in particular in existing fusion experiments such as JET. Beryllium, still being a light element with rather high physical sputtering vulnerability (Eckstein et al., 1993), features greatly suppressed chemical erosion when compared with that of carbon (De Temmerman et al., 2008; Carpentier et al., 2011). At the same time, beryllium possesses an excellent capability to binding oxygen (Winter, 1996; Tomastik et al., 2005; Konings et al., 2012; Wauters et al., 2020)—a highly deleterious impurity for fusion plasma performance responsible for undesirable radiative cooling of the plasma (Breton et al., 1976; Rogerson et al., 1977; Morozov et al., 2007) and for the enhanced sputtering of plasma-facing materials. Beryllium is therefore, the candidate material for the first wall of fusion devices where particle, thermal and neutron loads are rather benign.

The application of beryllium in a fusion power plant would be however, impossible. As a light material, Be would be subject to intolerable sputtering and material losses during a quasi-stationary operation. A fusion power plant will feature extremely high neutron fluence as compared to that of any present MCF device including ITER. In the course of neutron irradiation, Be will be a subject to transmutation leading to the formation of primarily ^{10}Be , ^7Li , and ^6Li (M R Gilbert and Sublet, 2016), as well as significant gas formation, particularly from helium ^4He , ^3He which could cause embrittlement problems (Martin and Ellis, 1963; Gelles et al., 1994; Kesternich and Ullmaier, 2003; Gilbert et al., 2012).

Tungsten

Achievements in plasma position control (Cenedese and Sartori, 2004; De Tommasi, 2019), well-developed scenarios for auxiliary heating (Gruber et al., 2009) have returned heavy metal options as future plasma-facing materials (Mayer et al., 2009; Neu et al., 2009; Brezinsek et al., 2017a; Pitts et al., 2019) into consideration. Tungsten (W) with its excellent thermal conductivity, low sputtering yield with respect to light plasma particles, highest melting temperature among non-ceramic materials, low tritium retention and moderate activation by fusion neutrons is deemed presently to be the best choice for plasma-loaded areas, like the divertor (Loarte et al., 2007; Pitts et al., 2011).

A material mix: Be for the first wall and W in the divertor, has been successfully used for plasma-facing materials in JET (Matthews et al., 2007; Philipps et al., 2010). In recent years, tungsten was successfully applied as a PFM in large MCF devices: besides JET, tungsten is used in ASDEX Upgrade (Neu et al., 2009) for the entire coverage of the first wall and divertor as well as in WEST (Missirlian et al., 2014; Firdaouss et al., 2017). There are also plans to make a tungsten divertor in JT 60SA. After the successful attempt with tungsten in LHD (Tokitani et al., 2019), this material is currently under the consideration for the future divertor in the Wendelstein 7-X advanced stellarator in Germany.

There are also disadvantageous properties of tungsten. Fusion plasma is highly sensitive to tungsten as an impurity (Pütterich et al., 2010), since this material radiates very effectively leading to undesirable radiative loss of fusion energy in the core plasma. Melting of tungsten may impose the issue, especially during transient events (Coenen et al., 2011; Matthews et al., 2016; Krieger et al., 2018) or re-attachment of the divertor plasma (Bonnin et al., 2017; Eldon et al., 2017). Recrystallization of tungsten in the course of thermal excursions (Guo et al., 2018; Morgan et al., 2020) may change its mechanical properties and affect the exploitation of the plasma-facing components. Tungsten exhibits an extensive crack network developed under repetitive heat loads (Klimov et al., 2011; Herrmann et al., 2017; Zammuto et al., 2018), which may lead to thermal fatigue failure. Tungsten oxidizes readily at elevated temperatures and the oxide of tungsten is volatile. At room temperature, pure tungsten is a very brittle material and its machining is a challenging task. Under these circumstances, the use of pure tungsten in a future fusion power plant is deemed to be problematic. New, advanced materials are being created, significantly expanding the functionality of pure tungsten as needed for the power plant application.

Advanced plasma facing materials

We have reviewed the current challenges in selecting a suitable plasma-facing material for the construction of near-future fusion experiments. For the next ultimate step—the fusion power plant, PFM challenges will increase significantly. Whereas the plasma parameters are not expected to differ greatly from those expected in the ITER experiment, the key difference is much higher particle and especially, neutron fluence. The severe neutron environment will bring new effects adversely affecting material performance. Another significant challenge will be to maintain the thermo-mechanical functionality of plasma-facing materials throughout the entire service life of the power plant. This may lead, possibly, to an avoidance of specific plasma scenarios. A third significant

challenge is due to significantly increased safety requirements imposed on a quasi-stationary power plant. The costs and availability of the source materials and technological feasibility of the component manufacturing are important factors, too. The analyses of new aspects influencing the fusion materials (Maisonnier, 2005; Coenen et al., 2016; Federici et al., 2017; Wenninger et al., 2017; Coenen, 2020) have outlined the necessity of the advanced material concepts. Advanced materials possess unique features, significantly extending the properties of conventional materials in critical areas of their functionality. The development of advanced materials has been focused on four main directions:

- Doped tungsten
- Tungsten composites
- Binary solid solution alloys
- Multi-component alloys

An extensive summary of the mentioned concepts is provided in e.g. Rieth et al. (2019). Usually, advanced material is being developed using a combination of element selection and advanced technology thus representing the material concept. We will briefly review a few mostly advanced material concepts.

Intrinsic properties of tungsten, such as its brittleness at room and elevated temperature were improved using cold rolling combined with thin plate preparation technology, applied successfully for the production of tungsten laminates (Reiser et al., 2017). The remarkable feature of these laminates is their engineered ductility even at room temperature.

Tungsten fiber composites or W_f/W , have been introduced in order to extend the applicability and robustness of tungsten, especially during cyclic high heat loads (Riesch et al., 2014; Jasper et al., 2016; Coenen et al., 2018). Millimeter-size tungsten fibers with a diameter of about 150 microns are introduced into a tungsten matrix (Jasper et al., 2015; Coenen et al., 2018; Mao et al., 2018, 2019a,b, 2020). Fibers are coated with an interface layer of erbia Er_2O_3 or yttria Y_2O_3 having a typical thickness of several microns, ensuring a weak interface between the fiber and the matrix. The main idea behind the tungsten fiber materials is in the dissipation of energy, which otherwise would feed into the buildup of stresses, crack propagation and finally, to the destruction of the component. The W_f/W features an unique pseudo-ductility and therefore, is much more resilient against damage than monolithic tungsten (Coenen et al., 2018; Mao et al., 2020). Another promising concept is the so-called microstructured W, where an array of millimeter-size tungsten tubes with diameter of $\sim 150 \mu m$ is directly integrated into the heat sink material. Microstructured W has already demonstrated impressive results by surviving extreme repetitive high heat loads expected in the divertor area (Terra et al., 2019) without a damage.

Significant progress was attained in production of tungsten components using the so-called power-injection molding (PIM). In this process, the tungsten powder is fed into a cavity where it is pressed into the desired form using high pressure. The significant advantages of PIM is the speed of the process and the ability to obtain components of practically any shape and geometry (Antusch et al., 2011).

Another promising concept, self-passivating (Koch and Bolt, 2007; Koch et al., 2009, 2011), so-called “smart” tungsten-based alloys (López-Ruiz et al., 2011; García-Rosales et al., 2014; Litnovsky et al., 2017a,c,d) has been proposed for the first wall of the future fusion power plant. During regular operation, smart alloys should exhibit advantages of a pure tungsten plasma-facing surface. During an accident, the alloying elements in the bulk form a protective layer and prevent the sublimation of neutron-activated tungsten oxide into the atmosphere. Modern tungsten-chromium yttrium smart alloys feature at least 10^4 -fold suppression of oxidation (Wegener et al., 2016, 2017; Litnovsky 2017c; Klein et al., 2018a) and more than a 40-fold reduction of sublimation (Klein et al., 2018b). Photographs of smart alloys and pure tungsten before and after exposure under accident conditions are shown in Fig. 6. Plasma tests (Litnovsky et al., 2017d; Schmitz et al., 2018, 2019) confirmed a similar sputtering resistance of smart alloys and tungsten to deuterium plasma under conditions expected at the first wall of a fusion power plant (Litnovsky et al., 2020; Schmitz, 2020).

New alloys with a controlled microstructure were developed for application in a high neutron flux environment feature reduced recrystallization and high tensile strength under extreme conditions (Kurishita et al., 2010; Nogami et al., 2020). This promising concept of so-called nano-engineered alloys was pursued in (Zhang et al., 2017). The small nanometer-size chromium inclusions are formed between the tungsten grains in tungsten-chromium and tungsten-chromium-titanium alloys. The formation of nano-size chromium-rich phase effectively opposes recrystallization and grain coarsening under neutron irradiation and temperature excursions.

Studying the effects caused by neutron irradiation are a vital part of the testing and qualification of materials for a future fusion power plant. In particular, the damage produced by neutrons and corresponding transmutation is a focus of these investigations. Several approaches exist to study in particular, the neutron damage, such as e.g. ion self-damage see e.g. Markelj et al. (2013) and exposure to fission neutrons. However, comprehensive testing under fusion relevant conditions is only possible with a fusion-relevant neutron source (Gilbert and Sublet, 2017). The International Fusion Materials Irradiation Facility—DEMO Oriented Neutron Source (IFMIF-DONES) is the major facility presently under construction to address the effects caused by fusion neutrons on materials (IFMIF DONES Facility, 2020). The IFMIF-DONES facility is reviewed in Donne et al. (2021).

Liquid metals

Liquid metals represent another, unique class of plasma-facing materials under investigation. Unlike the solid metals described above, where extreme care was taken to preserve the initial surface and microstructure, in the concept of the liquid first wall these

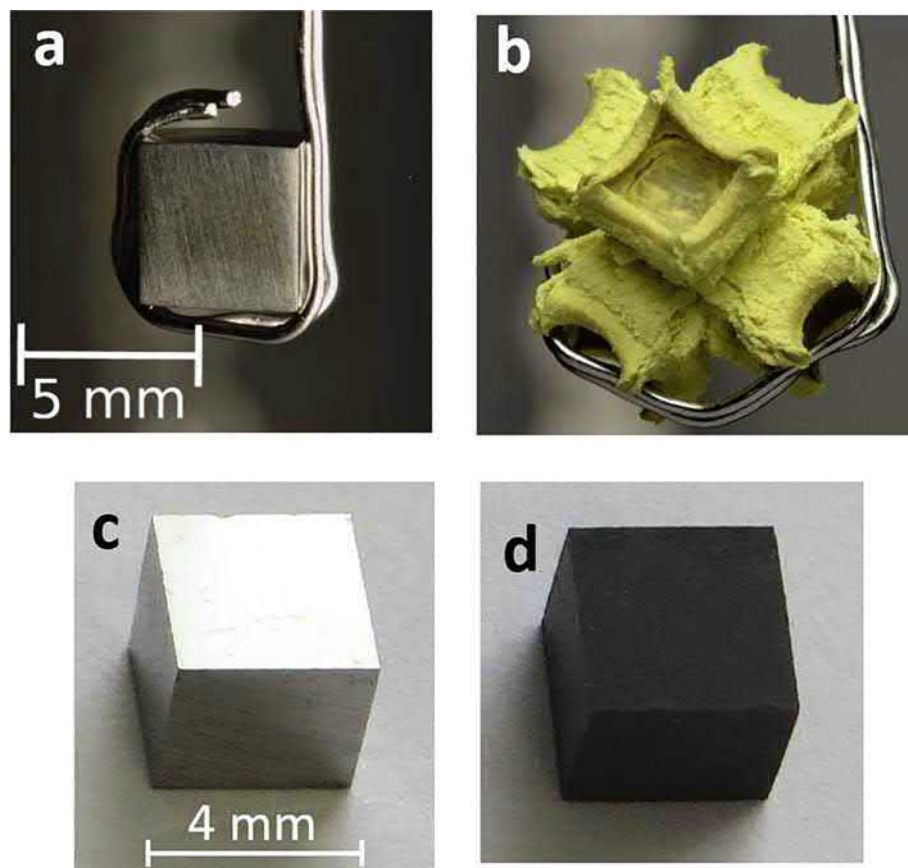


Fig. 6 Photographs of a pure tungsten sample and a tungsten-based smart alloy exposed under accident conditions as expected in DEMO power plant: (A) tungsten sample before exposure; (B) oxidized tungsten sample after exposure; (C) smart alloy sample before exposure and (D) smart alloy sample after exposure.

properties are discarded from the beginning. Instead, a constant renewal of the liquid metal surface is the fundamental feature of liquid metal application (Nygren and Tabarés, 2016; Ruzic et al., 2017; Zakharov, 2019), thus mitigating the issues mentioned, such as erosion and surface and transient damage. Many approaches to realize the liquid metal first wall have been investigated, ranging from a free molten surface (Ono et al., 2017; Yang et al., 2017) to a rather sophisticated, but the most promising, self-sustaining surface realized via a capillary porous structure, CPS (Evtikhin et al., 2002b; Khripunov et al., 2003; Mirnov et al., 2006; Lyublinsky et al., 2007; Coenen et al., 2014). In the CPS, the stability and the adherence of the metal molten layer is maintained by the capillary forces acting on the metal melt that soaks through the supporting metal mesh of submillimeter size. Several candidate materials are presently under investigations (Mazzitelli et al., 2011), such as lithium (Lyublinsky et al., 2007) and tin (Lyublinski et al., 2016; Roccella et al., 2020). Lithium is among the most promising materials, due to its low melting temperature, low Z number and the possibility of using it as tritium breeder in the fusion power plant (Evtikhin et al., 2002a; Nagayama, 2009). Laboratory studies have demonstrated the superior ability of a lithium liquid surface to withstand the high heat loads exceeding 20 MW/m^2 (Khripunov et al., 2001) by evaporating and creating a dense and cold lithium plasma in front of the liquid metal surface (Khripunov et al., 2001). It was shown that most of the power dissipation is realized through the vaporization of lithium. Successful realization of prototype first wall components have been reported from present day devices (Mirnov et al., 2006; Mazzitelli et al., 2011; Ono et al., 2017), where liquid metal structures, mostly CPS, were installed both in the divertor area (Ono et al., 2017) and as a part of the first wall. Among the challenges imposed by realization of the liquid metal solution are the recycling of lithium and re-liquidation of the molten layers and the collection and re-feeding of lithium to the first wall structures. An important issue is the transport of the vapor created by plasma-material interaction and the re-solidification of lithium in remote, usually cold and hardly accessible parts of the MCF device, causing metal contamination of the sensitive, mostly optical, diagnostic components (Khripunov et al., 2001). Uptake of deuterium and tritium via lithium layers, prompt creation of lithium hydrides and their subsequent transport to the distant areas and, potentially, into the inner structures of the device represent a critical issue for the application of liquid metals as PFMs for a fusion power plant (Ono et al., 2017). Intensive R&D efforts are currently underway to address the mentioned issues and several proposals for the first wall and divertor components of the power plant have been created and are proceeding towards realization (Evtikhin et al., 2002a; Nagayama, 2009; Morgan et al., 2018).

Blanket materials

In a fusion power plant, the plasma-facing first wall and the breeding blanket will constitute a common structure. The purpose of the blanket is the breeding of tritium needed for the tritium self-sufficiency of the fusion power plant as reviewed in detail by L.V. Boccaccini in (Boccaccini, 2021). The blanket must, in addition, play a major role in transferring the heat to the heat exchangers and finally, in converting the heat to electricity. The blanket structure is, therefore, crucial for the success of the controlled nuclear fusion concept (Stork et al., 2014a,b; Stieglitz et al., 2018; Federici et al., 2019). Depending on the blanket construction, a dedicated neutron multiplier may be used. Alternatively, the breeding material of the blanket itself will be used as a neutron multiplier. Presently, there are several blanket concepts under the development. These concepts can be divided into three main types:

- liquid metal concepts;
- ceramic breeder concepts;
and recently proposed
- concepts based on molten salt, is considered in the end of this section

In the liquid metal concepts, either pure lithium or lead-lithium eutectic alloy (PbLi) is used. There are two essentially different design solutions for the liquid metal blankets. In the single-coolant design, the liquid metal will be used both for breeding and for cooling of the blanket structure, whereas in the dual-coolant concept, liquid metal is used for breeding only and a separate coolant, water or helium, is used for cooling.

Among the most developed liquid metal breeder concepts are:

- Water-cooled lead-lithium blanket (WCLL)
- Helium-cooled lead-lithium blanket (HCLL)
and
- Dual-cooled lead-lithium blanket (DCLL)

In the solid breeding blanket concepts, a solid medium in the form of ceramic pebbles or spheres is enclosed in the body of the beryllium neutron multiplier. Among the known and most developed concepts are:

- Helium-cooled ceramic blanket (HCPB)
- Helium-cooled ceramic breeder graphite collector (HCCR)
- Water-cooled ceramic breeder (WCCB)

Blanket construction is rather complex, comprising the cooling, tritium extraction, coolant purification and instrumentation systems and control subsystems. The principal blanket concepts are presented and assessed in e.g. Giancarli et al. (2012), and Abdou et al. (2015). It is planned to test the major breeding blanket concepts: HCLL, WCCB and HCCR, in ITER environment using the so-called test blanket modules (Boccaccini et al., 2004; Giancarli et al., 2012; Boccaccini, 2013; Abdou et al., 2015; Zmitko et al., 2017). In ITER, however, the tritium breeding efficiency and effective heat production cannot be tested due to insufficient neutron fluxes (Giancarli et al., 2012; Abdou et al., 2015) at the ITER first wall. Nevertheless, the entire construction of the blanket can be evaluated, including the consistent functioning of all essential subsystems and the results can be used for further extrapolative modeling (Giancarli et al., 2012). There are several proposals for the future full functional test of the breeder solutions, including the building of a dedicated Fusion Science Nuclear Facility, as proposed by the United States.

Operation of the breeding blanket is accompanied with several, in part crucial, issues and risks. Among those, the interaction of a slowly moving conductive liquid metal with a strong magnetic field, which causes significant Lorentz forces. In several concepts, such as DCLL and HCLL, insulating so-called flow channel inserts (FCIs) are foreseen in order to minimize these conductive circuits and hence, to limit the electromagnetic forces acting on the blanket. The structure of the blanket is expected to be made from magnetic steel. The use of magnetic material leads to stresses in the construction and may distort the magnetic field configuration in the power plant, especially during the ramp-up and ramp-down phases of plasma operation (Zmitko et al., 2017). Finally, general safety and tritium isolation impose severe requirements on tritium permeation, efficient evacuation and containment. In general, the following requirements are specified for blanket materials:

- Neutron robustness, reduced activation
- Tritium opacity
- Reduction of electromagnetic forces
- Thermo-mechanical stability of the entire construction

Generally, the principle of Reliability/Availability/Maintainability/Inspectability (RAMI) (Abdou et al., 2015) is applied to the entire blanket construction. The application of RAMI to materials used in blanket construction has led to the material selection described below. There are three classes of functional blanket materials: the structural materials, the materials for flow channel inserts and the tritium permeation barrier (TPBs) materials.

For the structural blanket materials, reduced activation ferritic-martensitic steels (RAFM) are selected. In the European HCLL and HCPB concepts, the use of Eurofer (Van der Schaaf et al., 2003; Möslang et al., 2005; Boccaccini, 2013; Tassone et al., 2018) is foreseen, whereas in the WCCB concept proposed by Japan, the F82H RAFM steel is planned as a structural material (Giancarli et al., 2012; Tanigawa et al., 2017). Significant R&D activity is underway in order to assess, among others, the interaction

of helium coolant with RAFM as well as the complex interaction with the PbLi (Flament et al., 1992; Tortorelli, 1992; Malang and Mattas, 1995; Moreau et al., 2011; Abdou et al., 2015; Zmitko et al., 2017).

For flowing channel insulation the main candidate solution is silicon carbide (SiC). The use of SiC for FCIs, in bulk or in dense foam form brings several advantages, such as significant reduction of the MHD loads and a rise of operation temperature to about 700°C. Such a high operation temperature however, exceeds the current temperature window of the RAFM application (Möslang et al., 2005; Abdou et al., 2015; Tanigawa et al., 2017; Tassone et al., 2018). In addition, the high temperature response of SiC to neutron irradiation, as well as technological concerns of manufacturing the complex geometries of FCIs out of SiC, need further exploration. The potential interaction of FCIs with the e.g. PbLi eutectic and RAFM are among potential concerns (Pint et al., 2007; Muroga, 2012; Abdou et al., 2015; Smolentsev et al., 2015). The use of alumina (Al_2O_3) as an alternative to SiC is still under assessment. Here, the radiation hardness and activation properties are among major concerns (Abdou et al., 2015).

Tritium permeation barriers have to be implemented within the blanket structure. Their role in the breeding blanket becomes even more important, since the TPBs must warrant efficient containment, and refill the bred tritium to the power plant, including the reservoir systems to be used in case of blanket failure (Abdou et al., 2015). Permeation barriers under development (Livshits et al., 1990; Levchuk et al., 2004, 2007; Pisarev et al., 2006; Houben et al., 2020) are already providing an efficient tritium isolation. Modern barriers made of yttria (Houben et al., 2020) may provide more than a 10^6 -fold decrease of tritium flux permeation to the bulk of the facility in the temperature range of 300°C–600°C (Houben et al., 2020).

In the blanket environment, the reactions with liquid breeder and coolant, as well as possible interactions with structural materials and FCIs, must be investigated. Performance studies of neutron-irradiated TPBs are of crucial importance for the breeder blanket.

As mentioned above, the interaction of the working fluid or solid breeder and neutron-multiplier materials with the functional elements of the blanket, is important. The chemical and mechanical durability of breeder and multiplier materials themselves represents a crucial topic for the blanket feasibility. We will not provide a comprehensive overview of all issues and open questions in the section, but mention a few. The use of beryllium as neutron multiplier in e.g. ceramic breeders faces issues due to the radiation-induced volumetric swelling, reaching 16% at 650°C (Abdou et al., 2015; Fedorov et al., 2016). The use of the beryllium intermetallic phase Be_{12}Ti seems advantageous in this respect. However, water-beryllium interaction in the case of liquid blankets imposes problems (Abdou et al., 2015). For the dual-coolant systems, such as DCCL, the permeation of bred tritium into the helium-coolant circuit is considered as critical, imposing stringent requirements for TPBs.

In relation to the molten salt concepts noted above, the main potential advantage of using e.g. F-Li-Be or F-Li-Na-Be salts is their low electrical conductivity. This would alleviate the MHD issues of the conductive medium. Molten salts are also relatively chemically inert. At the same time, their high viscosity and high melting point limit the operating window severely. In addition, under neutron irradiation, the corrosive interaction of molten salts with the interior of the blanket, leading to the potential release of highly active and toxic fluorine, represents a significant concern.

Diagnostic materials

Another class of materials in MCF devices which are at least partly exposed to plasma particles are the materials of diagnostic components. Here, the difference between the present-day devices and the coming experiments, e.g. ITER, is rather distinct. The particle, neutron and radiation environment in present-day devices is rather benign and the plasma pulses are usually limited to several seconds. A wide range of materials can be used without any significant risk or degradation of performance or with relatively easy exchange possibilities. On the contrary, in the devices of ITER-class, the neutron and radiation fluxes are orders of magnitude larger (Costley et al., 2001, 2005; Walker et al., 2005; Donné et al., 2007) than those in any of the present MCF devices, and hence cannot be neglected. In the following description, we will focus on diagnostic material challenge for fusion devices of the ITER-class and for the fusion power plant.

Due to the harsh neutron environment, the use of refractive optical elements i.e. lenses and prisms, will be impossible in these MCF devices, mostly due to the inability of the corresponding materials to withstand neutron irradiation (Voitsenya et al., 2001; Voitsenya and Litnovsky, 2009). A reflective, mirror-based option will be used instead. Diagnostic mirrors will guide the plasma radiation towards spectrometers and detectors (Costley et al., 2001; Donné et al., 2007; Litnovsky et al., 2019a). The first optical element directly viewing the plasma is the so-called first mirror. It is one of the most important diagnostic elements overall, given the number and importance of optical diagnostics and the fact that the performance of the entire corresponding diagnostic largely relies on the performance of its first mirror and hence, on the mirror material. Moreover, it will become necessary to clean the mirrors from the deposited material eroded from the first wall and the divertor—these processes are described earlier in the subchapter “Plasma-material interaction processes”. The repetitive cleaning will be performed in situ using a local plasma for sputtering the impurities from the mirror surface (Litnovsky et al., 2011a, 2015, 2019b; Mukhin et al., 2012; Moser et al., 2015; Leipold et al., 2016; Peng et al., 2018; Yan et al., 2018). Single crystal mirrors made from highly reflective materials, such as molybdenum and rhodium are among the main candidates for the diagnostic mirrors (Voitsenya et al., 2001; Litnovsky et al., 2005a, 2007c, 2017b, 2019b; Matveeva et al., 2010; Peng et al., 2018). Mirrors coated with a reflective molybdenum or rhodium layer are under investigation (Marot et al., 2007; Eren et al., 2011). As for the further, more distant, secondary mirrors, the environmental conditions are not as restrictive and therefore the choice of materials increases. Currently, stainless steel and aluminum mirrors coated with protective sapphire or tantalum oxide Ta_2O_5 layers are being considered (Mukhin et al., 2008a). Alternatively, secondary mirrors with titanium dioxide and silica highly reflective layered coating oxide are under study (Krimmer et al., 2013); however, attaining the desired long-term performance is challenging.

For some diagnostics (Mukhin et al., 2008b, 2009), protective windows are foreseen. Windows will be used as well as vacuum isolation for several diagnostics. Fused silica, sapphire and quartz are presently the main candidate window materials (Aymar, 2002). As for duct and corresponding support structures, most diagnostics are expected to use steel (Kalinin et al., 2000; Voitsenya et al., 2001; Krasikov et al., 2011). For fusion power plants, the choice will most probably be made in favor of reduced-activation steels due to the increased neutron irradiation. These steels are described in detail in the next section.

Material choice for the other elements of the reflective optical paths, such as fiber bundles, appears to be challenging. A range of neutron-induced changes or effects, such as radiation-induced electromotive effect, radiation-induced conductivity, radiation-induced electric degradation and generally, non-ohmic behavior, may affect the performance and degrade the lifetime of the fiber materials (Zinkle, 1994; Aymar, 2002). Current studies show advantages from using mineral insulated cables (Vayakis et al., 2008). However, care must still be taken to avoid radiation-induced effects (Vila and Hodgson, 2009), depending on their chemical and elemental purity of the fiber material (Hodgson, 1998). Similar issues with radiation-induced effects will be experienced by the other diagnostic components such as e.g. imaging bolometers (Peterson et al., 2008).

In a fusion power plant, the requirements for diagnostic materials and assemblies will likely be even more demanding. Only a few crucial optical diagnostics will remain, and mirrors should be of reduced size and placed far away from the plasma (Orsitto et al., 2016; Biel et al., 2019). In addition, reserve optical channels should be foreseen to ensure continued operation in case of the failure of the main diagnostic (Litnovsky et al., 2010; Biel et al., 2019).

Structural materials

The main function of structural materials in current plasma devices is in providing the mechanical support to components, as well as defining the vacuum vessel boundary. Furthermore, the structural components contain the cooling infrastructure of MCF devices and ensure the required vacuum conditions. The functionality of structural materials in current fusion machines and hence, their choice, depends largely on interface properties to PFMs, blanket materials and chemical resistance to the reactive, humid or corrosive environment (Lässer et al., 2007; Zinkle and Busby, 2009). Presently, austenitic stainless steel of different types is the most widespread structural material (Kalinin et al., 2000; Barabash et al., 2007). In ITER, most of the supporting structures will be made of type 316L stainless steel (Kalinin et al., 2000). The heat sink interface material to the water-cooling infrastructure will be provided by a high strength, high conductivity copper alloy (CuCrZr) (Kalinin et al., 2000; Barabash et al., 2007).

Requirements for structural materials increase significantly in a fusion power plant due to the long required lifetimes at high temperatures in the presence of high fluxes of energetic neutrons, corrosive fluids, and large cyclic thermomechanical stresses. Superior mechanical properties and resistance to radiation induced property degradation such as embrittlement and void swelling are required to ensure acceptable reliability and component lifetimes before replacement, and low residual activation following prolonged exposure to energetic neutrons is required for safety and recycling/waste disposal considerations (Bloom, 1998). Key properties include superior elevated temperature thermo-mechanical creep-fatigue properties, resistance to neutron-induced embrittlement and stress corrosion cracking, resistance to radiation-induced dimensional changes due to irradiation creep or void swelling, and high temperature coolant corrosion resistance (Bloom et al., 2007). A particularly daunting challenge is the high levels of transmutant hydrogen (H) and helium (He) that are produced in most materials by the 14 MeV neutrons generated in the deuterium-tritium fusion reaction. These H and He gaseous transmutants tend to enhance radiation degradation processes such as low temperature embrittlement, intermediate temperature void swelling, and high temperature grain boundary embrittlement (Dai et al., 2012; Zinkle et al., 2019) dramatically. Due to the worldwide lack of a suitably intense prototypic 14 MeV neutron source (Zinkle and Möslang, 2013; Knaster et al., 2016), estimates of potential degradation rates are mainly based on limited simulation experiments and computational modeling predictions. Therefore, the structural material suite being considered for fusion power plants is quite broad.

Reduced activation ferritic-martensitic (RAFM) steels developed for fusion are the current leading candidates. These include Eurofer (Lindau et al., 2005; Möslang et al., 2005), Rusfer (Chernov et al., 2010; Rogozhkin et al., 2012; Tyumentsev et al., 2012), CLAM (Huang, 2014), F82H (Jitsukawa et al., 2002; Tanigawa et al., 2011) and JLF-1 (Kohyama et al., 1998). The historic evolution of the RAFM steel development and current status is provided e.g. in Kohyama et al. (1996); Tanigawa et al. (2017). The favorable reduced activation properties of RAFM steels is largely due to replacement of undesirable solute elements (Mo, Nb) in commercial high performance steels with similar-functioning reduced-activity elements (W, V, Ta) (Klueh and Bloom, 1985; M.R. Gilbert and Sublet, 2016). A wide variety of ultra-high-performance alternatives to the baseline RAFM steels are under intense investigation, due to concerns that higher performance properties and/or radiation resistance may be needed for next step fusion energy devices (Lindau et al., 2005; Klueh et al., 2007; Zinkle et al., 2017). These high performance alternative steels generally utilize ultra-high densities of nanoscale particles such as oxides, carbides or nitrides to improve high temperature strength and neutron radiation resistance.

Promising risk mitigation alternatives to RAFM structural materials under investigation include silicon carbide ceramic composites (Snead et al., 2007; Katoh et al., 2014) and refractory alloys such as V-4Cr-4Ti (Zinkle, 2005; Zinkle et al., 2013; Muroga et al., 2014). These structural materials offer the potential for improved thermodynamic efficiency due to $\sim 200^{\circ}\text{C}$ – 400°C higher operating range than that of RAFM steels, along with potentially improved radiation resistance compared to RAFM steels. However, the industrial experience with these materials is much less compared to that for steel production and use in engineering systems.

For the use of low-activation steels, good interface to the PFMs plays a crucial role. Due to significant difference in a wide variety of crucial parameters and, primarily, thermal conductivity and thermal expansion coefficients, direct joining of tungsten to low-activation steel is challenging (Driemeyer et al., 1999; Bachurina et al., 2018). Various joining techniques are under study, including

brazing (de Prado et al., 2017; Bachurina et al., 2018), hot isostatic pressing (Driemeyer et al., 1999), field-assisted sintering technology, creation of functionally graded materials (Weber et al., 2013; Heuer et al., 2020); the use of tungsten laminates (Reiser et al., 2017) and additive manufacturing look promising. Here, the strength of the interface, the inter-diffusion of steel elements and tungsten, the phase formation and the durability under cyclic heat loads are the prime criteria for feasibility.

Tritium safety must be applied to the structural materials. The development of tritium barriers reviewed earlier in the article must suppress the undesirable diffusion of tritium in the bulk materials and prevent tritium interaction with the coolant.

Structural materials must also provide an efficient and robust interface to the cooling structure. Here, depending on coolant type, various materials or alloys with a high thermal conductivity are under investigation. For water-cooled high heat flux structural applications, the most promising are high-strength, high conductivity Cu alloys such as CuCrZr (Pintsuk et al., 2010; Barabash et al., 2011; Stork et al., 2014a; You, 2015; Richou et al., 2017, 2020) and other Cu alloys or composites are under development (You, 2015; Zinkle, 2016; Wang et al., 2020).

Different joining methods are under development (Nicholas et al., 2018) for providing robust interfaces to the heat sink materials. Joints initially created by hot isostatic pressing (HIP) face challenges, mainly due to severe thermal expansion mismatch of tungsten and heat sink materials (Barabash et al., 2000; Rigal et al., 2000). Recently, melt infiltration (Muller et al., 2017) and field-assisted sintering technology (FAST) (Galatanu et al., 2017, 2019) were applied successfully. Direct joining of tungsten to the heat sink material is another strategy of interest for divertor applications, etc. Additive manufacturing as a technique for direct joining of tungsten and the heat sink material is also under consideration (Müller et al., 2019).

Materials for magnets

Despite years of research, the magnet systems in fusion reactors are still considered to be one of the primary life-limiting components. Difficulties associated with replacing, in particular, toroidal field coils are considered a barrier to commercialization of fusion (Chislett-Mcdonald et al., 2019). The current material of choice as the superconducting material for EU-DEMO applications is niobium (Nb)—tin (Sn) alloy Nb₃Sn, but long-term reliability is an issue because of its brittleness. Nb-Ti, which is commonly used in magnetic resonance imaging machines, is a potential ductile alternative, which was recently proven applicable for advanced fusion systems with well-controlled plasma confinement requiring lower magnetic fields (Chislett-Mcdonald et al., 2019).

Meanwhile, “High temperature” (20–40 K) Rare-Earth-Ba-Cu-O (REBCO) superconducting tapes are being explored for conventional (Heller et al., 2015) and compact (spherical) fusion devices because of their ability to carry more current at higher field compared to conventional low temperature (4 K) superconductors, which is critical for compact devices (Sorbom et al., 2016; Sykes et al., 2018). These materials are immature and inherently brittle and understanding of how such materials withstand neutron irradiation damage is still largely missing. Since no ductile solution for high-field superconductors has so far been found, there is active research in devising economic engineering solutions to the problem of replacing field coils made from REBCO or otherwise (Tsui et al., 2016), which could also benefit the lower temperature options. Demountable joints are an attractive solution, allowing a modular construction of coils and making replacement easier (Sorbom et al., 2016). Soldering of the joints, e.g. using tin-based solders (Tsui et al., 2016) is the most promising solution. However, challenges are still to be overcome concerning the ability of the joints to carry currents and the reliability and repeatability of the joining process that could be required in thousands of locations in a typical fusion reactor system (Sagara et al., 2014).

Materials for inertial fusion devices

Introduction

The development of the facilities for inertial fusion is mainly focused on reaching the prime goal of attaining the ignition of D-T targets. Material development for the inertial fusion facilities is accompanying the mentioned mainstream research. Due to the different operation modes of IFE and MCF facilities, major requirements deviate significantly for these two concepts. At the same time, there are also distinct similarities. Specifically, the similarities can be noticed in requirements for the materials suggested for IFE facilities. Information on material research for IFE facilities is not as abundant as that for MCF devices. Nevertheless, the intensive pilot research conducted provides a sufficient degree of specification of the envisaged material concepts.

A schematic view of the inertial fusion facility is presented in Fig. 7 on the example of the European facility HiPER (HiPER facility website, 2020). Laser beams focused on the frozen D-T target with dimensions of several millimeter will ablate the outer layers of the target compressing it and causing the fusion D-T reaction reviewed in details in (Atzeni, 2021). The reaction volume is contained in a spherical reaction chamber where, as in MCF devices, the first wall will be exposed to the products of a fusion reaction. The spherical reaction chamber is the inner sphere in Fig. 7. The first wall will be fixed to the structural material, providing the mechanical support and acting as a chamber shield. The so-called final optical assembly (FOA) is contained within the second, larger spherical shield. The massive structure will hold optical tubes used to direct the powerful laser radiation beamed through the set of mirrors and lenses to the D-T target. Finally, the external cylindrical wall will encapsulate the entire volume of the inertial fusion facility. Depending on requirements and functionality the suggested materials for an IFE facility will be divided into several different classes:

- a. First wall materials
- b. Structural materials
- c. Optical materials

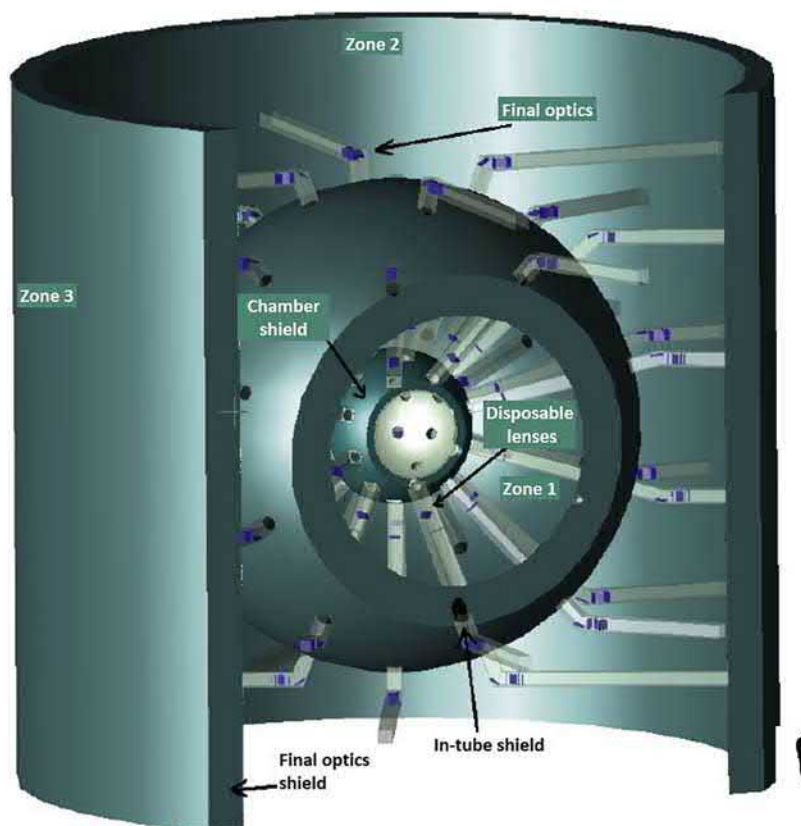


Fig. 7 A preliminary design of HiPER. Adapted from Ruiz JA, Rivera A, et al. (2011a) Materials research for HiPER laser fusion facilities: Chamber wall, structural material and final optics. *Fusion Science and Technology* 60: 565–569. doi: 10.13182/FST11-A12443.

In this sub-section we will review the main challenges and requirements for these material classes and will survey the candidate materials suggested for IFE.

Material challenges and requirements in IFE

Owing to the nature of the inertial operation, significant heat loads are expected inside the IFE facilities. Presently, the laser systems with a total power ranging from 0.5 to 1.5 megajoules will beam onto the D-T target with very short pulses of one to several nanoseconds (Chanteloup et al., 2010; Sethian et al., 2010; Ruiz et al., 2011a; Colier et al., 2013; Miquel et al., 2016; Norimatsu et al., 2017). The entire IFE facility is expected to operate with a shot frequency of several hertz. The resulting fusion energy of order of tens of megajoules, will be unloaded within the first few hundreds of nanoseconds onto the components of the first wall in the reaction chamber. The massive particle, heat and radiation loads impose a significant challenge for first wall materials. In HiPER, the expected loads to the first materials can reach about 200 GW/m^2 for the He ions originating from fusion reactions. The debris particles originating from the non-ablated remainder of D-T capsule will lead to loads of order of 40 GW/m^2 (Perlado et al., 2012) exposing the first wall for about $2 \mu\text{s}$ with a repetition rate of 1–10 Hz (Ruiz et al., 2011b).

Pulsed operation of the IFE facility will provide an additional complication. The nearly instantaneous release of a large amount of energy will impose the risk of melting and sublimation and splashing of plasma-facing materials (Norimatsu et al., 2010, 2017; Sethian et al., 2010; Ruiz et al., 2011a). Additionally, the cyclic mode of operation with a high repetition rate of shots, 24 h a day and 7 days a week in the timeframe of the envisaged 20–30 years of operation of IFE (Sethian et al., 2010; Ruiz et al., 2011a; Norimatsu et al., 2017) will significantly increase the probability of material failure as a consequence of, e.g. thermal fatigue.

Finally, intensive particle and neutron radiation originating from the D-T reactions will induce the substantial radioactivity in the components of the IFE reactor. The radiological aspects of nuclear fusion facilities are reviewed in the Chapter 01229 by N. Taylor (2021).

Material choice for IFE facilities

Plasma-facing materials

Material choice for the first wall of the reaction chamber varies strongly depending on particular design (Perlado et al., 2010, 2018; Sethian et al., 2010; Norimatsu et al., 2017). The two essential material options under consideration are:

- Solid (dry) wall
- Liquid wall

In some cases, in the course of design development there is a transition between the first wall options: a solid wall to be exchanged for a liquid wall as it was in case for the Japanese laser fusion commercial reactor project KOYO-F (Norimatsu et al., 2017). An additional options to reduce damage of the first wall includes magnetic diversion of fusion-generated ions (Sethian et al., 2010).

The choice of a solid wall usually is justified by the required robustness of the design from the beginning, based largely on the experience obtained at the MCF facilities and corresponding MCF materials testing program. Here, the reader may notice distinct similarities in material choice for IFE and MCF facilities. Because of known problems with tritium retention and formation of volatile hydrocarbons, carbon was eliminated as a candidate material for the first wall armor. Due to its natural advantages mentioned earlier, tungsten is currently being considered as the prime candidate material of the solid first wall (Juarez et al., 2010; Perlado et al., 2010, 2018; Sethian et al., 2010; Ruiz et al., 2011a). Depending on the specific design, the thickness of the first wall varies between 1 mm and 1 cm (Juarez et al., 2010; Perlado et al., 2010, 2018; Sethian et al., 2010). Alternatively, tungsten alloys or nano-structured tungsten are suggested to gain the advantage of reduced recrystallization and to relieve neutron effects (Perlado et al., 2018). It is expected that the use of tungsten with its advantageous properties will relieve the effect of thermal loads on the first wall of the envisaged ICF devices—a similar approach is followed for MCF facilities. At the same time, there is still a risk of tungsten melting, which was predicted for the particular conservative scenario in HiPER (Ruiz et al., 2011a,b). Here, the peak pulse of X-rays created after the reaction pushes the temperature of the W-armor to 3300°C within several microseconds. Detailed calculations reveal an increased risk of temperature excursions of the plasma-closest layers of the armor. Owing to the very fast response to the reaction event, the power load from the fusion burn reaches the first wall within several nanoseconds and after 1 μ s up to 50% of generated power is absorbed within the first 1 μ m of the tungsten wall. More realistic modeling taking into account the time evolution of the heating pulse predicts, fortunately, peak temperatures of around 1530°C (Ruiz et al., 2011a). Another essential issue is tritium permeation through the first wall material. The corresponding research is briefly reported in (Ruiz et al., 2011a). The repetitive heat loads lead to extensive crack formation and cause degradation of the thermo-mechanical properties of tungsten, as well as provoking mechanical fatigue of W (Sethian et al., 2010), as known from the MCF research (Linke et al., 2019). The corresponding studies reported for HiPER in (Ruiz et al., 2011a,b) outline the risk of cracking at the 1.5 μ s after the laser shot in HiPER, when the Von Mises stress of tungsten armor exceeded 3 GPa (Blanchard and Martin, 2005). The evolving cracking can degrade the overall mechanical stability of the tungsten first wall and lead, among other consequences, to dust and debris formation.

Nevertheless, the solid wall option is treated as a relatively robust one and is incorporated in the majority of the first wall concepts. The liquid wall concepts are usually based on lithium and either represent pure Li wall solutions or the lithium-lead (Pb) combination—which is advantageous since it integrates the blanket solutions. An extensive study was reported in (Norimatsu et al., 2010, 2017). The repetitive short-pulse power loads will lead to expected effects including liquid splashing and vaporization and, eventually, ablation. The ablation of a plasma-facing 3- μ m thick layer of the first wall per a single laser power pulse was predicted. The ablated materials then become split and in part strike the first wall and can subsequently re-enter the inner volume of the reaction center. Modeling predictions were experimentally validated. The propagation velocity of the ablated plume of first wall materials is of order of $1 \text{ km} \times \text{s}^{-1}$, as detected using the Mie scattering technique (Norimatsu et al., 2010).

Important information was obtained on the nuclear decay heat in a LiPb first wall. For the LIFT test reactor (Japan) with a nominal projected fusion energy of 40 MJ per pulse operating at 1 Hz repetition rate, the resulting decay heat in the first wall still exceeds 1 MW after 10,000 h (> 1 year) of operation.

Structural materials

Structural materials in IFE facilities will naturally experience, less intense power and radiation loads than the first wall. At the same time, since structural materials support the first wall armor, their long-term durability under cyclic repetitive mechanical loads and eventually, significant temperature excursions, is required. Various types of steels were therefore proposed for different IFE facilities. In the LIFT project (Japan) an F82H martensitic steel structure will hold the liquid first wall of the blanket, whereas the stainless steel SS 316 will encapsulate the cooling channels. Here, the calculations (Norimatsu et al., 2017) show a temperature periodic rise to 1500°C which will potentially endanger the steel supporting structures.

An extensive study on possible structural material options was undertaken for the HiPER project. Among others, the radiology aspects were considered. A comparative analysis of the Eurofer reduced activation steel developed for MCF (Lindau et al., 2005; Möslang et al., 2005) and the commercial SS304L steel has been performed for the standard expected HiPER operation scenarios. The scenarios comprised 100 MJ of neutron yield per shot and up to 100 shots in one burst repeated at 10 Hz, with one burst per month for 20 years. The results showed clearly that the low contact dose rate attained with SS304L reaches $1.5 \times 10^{-5} \text{ Sv} \times \text{h}^{-1}$ in 50 years which is sufficiently low to qualify this relatively inexpensive steel for the HiPER environment. More details on the reported comparative analysis can be found in (Juarez et al., 2010).

Optical materials

By their nature, all projected IFE facilities require an extensive set of optics to guide the laser radiation to the D-T targets. The optical components comprise both the refractory and focusing optics and corresponding waveguides and supporting systems (Sethian et al., 2010). The structure of optical components is presented in Fig. 7 for the example of the HiPER facility (Ruiz et al., 2011b). In the case of optics, the radiation level and the thermomechanical loads dictate the conditions and largely govern the

design of components. In HiPER, the region between the inner spherical reaction chamber and the outer spherical shield, the outer sphere in Fig. 7, is the location of the so-called disposable optics in the Final Optical Assembly (FOA). It is assumed that optical components of the FOA located within Zone 1, Fig. 7 in the course of the operation become either damaged or displaced and must be regularly exchanged. Optical components outside Zone 1 are stationary and are not supposed to be exchanged.

For optical elements in Zone 2, the prime material for the mirrors and focusing lenses is deemed to be a pure (single crystal) silica based on results of investigations summarized in (Juarez et al., 2010; Ruiz et al., 2011b). At the same time, recent calculations briefly reported in (Ruiz et al., 2011b) suggest the necessity of using cooling for at least a part of optical components. Other final optic options include grazing incidence metal mirrors (Sethian et al., 2010), similar to the choice adopted for MCF systems. For the optical tubes, several materials are under consideration including both low-activation and regular SS304L stainless steel.

Summary and outlook

In this article, we summarized the development of fusion reactor materials—one of the key pillars in attaining the ultimate goal of controlled nuclear fusion, the realization of a fusion power plant. For magnetic confinement devices: tokamaks and stellarators, we have seen the variety of processes occurring during the interactions of fusion plasmas with materials. Among them are: physical sputtering and chemical erosion, recrystallization and melting, tritium accumulation, prompt re-deposition of eroded material leading to the dust formation, and significant changes imposed by plasma particles and radiation on the microstructure and surface of plasma-facing materials. The majority of these processes are also expected to occur in inertial confinement devices. The processes of plasma-material interactions have imposed severe requirements on the candidate material concepts. It is interesting to see a kind of “loop” in the development of fusion materials from metals widely spread in fusion facilities in 1970s, to the light, carbon-based materials and beryllium dominant until 2000s and then, a return to the metals nowadays. Advanced plasma control scenarios, and improved plasma heating knowledge have largely contributed to this return to high-Z refractory metals, such as tungsten. However, achievements were scored not on the plasma side only. The development of new material concepts, such as fiber components, suitable as plasma-facing materials, and significantly advanced engineering of components, have contributed to the return of the high-Z metal option, an option which was even hard to imagine for fusion devices before.

Presently, we are witnessing the next, remarkable stage in the development of plasma-facing materials—the creation of the advanced, entirely novel, high-Z material concepts and alloys. Their features are unique: from the capability of handling enormous heat loads without serious damage, to the resistance to cracking and recrystallization and even a capability to adapt the material properties to the environment—an unbelievable set of properties is now in our hands. A liquid first wall and divertor—options that looked like a fantasy before, are now becoming a reality and we see first design solutions for the liquid metal components for a future fusion power plant.

Material development for the inertial fusion facilities is largely harnessing the intensive synergetic research results gained for magnetic confinement fusion devices. As for the first wall of the reaction chamber, two primary concepts are presently under consideration. The so-called “solid” wall is planned to be built either from pure tungsten or from the advanced nano-grain tungsten or tungsten alloys. Alternatively, for several IFE reactor concepts, a self-cooling lithium-based liquid wall has been suggested. Here, issues may arise potentially with the recycling of liquid material and with the stability of a liquid under cyclic high thermal, mechanical and radiation loads. The advantages of a liquid wall are the ability to act as a blanket for tritium breeding and the absence of a requirement for active cooling.

Optical materials for IFE facilities, such as like mirrors, frequency multipliers and focusing lenses will largely rely on established material concepts. Presently, it is planned to use pure silica for all these elements, although adjustments may become necessary for optic components located in final optical assembly.

Optical systems of the MCF facilities will rely on mirror-based reflective optics. Here, the use of unique, highly reflective material concepts, such as single crystal molybdenum and rhodium for diagnostic mirrors—with their unprecedented capability to preserve the reflectivity under plasma sputtering, have opened the way for mirror-based diagnostics which can last for the entire lifetime of future MCF devices. The dedicated facilities for mirror surface recovery, including in situ mirror cleaning and calibration systems, will support the retention of high mirror performance.

Development of efficient and reliable blanket technology represents technological, physics and engineering challenges. Several promising concepts are under development and their simultaneous testing in the ITER test blanket modules is foreseen as a part of the R&D and qualification program.

Last, but not least, exciting progress has been achieved in structural materials. Whereas, conditions in inertial facilities luckily allow the use of conventional martensitic steel, the creation of the entire class of the reduced activation steels including Eurofer, Rusfer, CLAM, F82H—has dramatically changed the R&D and opened the way to their unmatched functionality in magnetic fusion devices.

Of course, as we have seen, the material choice and especially, the realization of the components for a fusion power plant are still far from the final stage. Significant remaining challenges include the suppression and effective control of transient power and particle loads in MCF and IFE facilities and the long-term degradation of materials. The design of MCF power plants will also benefit from the development of ductile, robust superconducting materials, or from the reliable engineering solutions for existing superconducting materials, an issue which will be addressed by ITER operation.

In a fusion power plant, materials will have to withstand, and retain their functionality under a very high fluence of highly energetic 14 MeV neutrons—and neutron-resistant material solutions must still be thoroughly tested in the burning plasma environment. However, with the solid portfolio of advanced materials developed to date, and considering the impressive progress already made on fusion materials, a significant basis exists for the development of the materials required for construction of a fusion power plant and finally, for the production of fusion—based electricity to the grid.

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Magnetic Confinement Fusion—Reactor Blanket Technologies

Lorenzo Virgilio Boccaccini and Christian Day, Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany

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Glossary

Divertor Dedicated structure within the vacuum vessel onto which the highest plasma exhaust heat fluxes are directed and through which unburned fuel, helium produced in fusion reactions and gaseous impurities are extracted, usually by vacuum pumps (the divertor is toroidally symmetric in a tokamak, but has a more complex geometry in a stellarator).

Dual coolant lead lithium (DCLL) Blanket concept based on lead-lithium breeder with steel as structural material; it is cooled both by the breeder itself flowing in a dedicated loop and by an additional helium loop that provides the cooling of the steel structure.

Ductile-to-brittle transition temperature (DBTT) The temperature at which there is a pronounced decrease in a material's ability to absorb force without fracturing.

Full power day (or year) The equivalent of 24 h (or 365 days) of full power operation by a reactor.

Fusion power (FP) The power directly released by the thermonuclear D-T reaction in the plasma (the energy released per single event is 17.6 MeV)

Heating, ventilation and air conditioning (HVAC) In nuclear technology the HVAC system must provide the functions of maintaining ambient conditions within acceptable limits within the nuclear buildings (temperature, humidity, contamination), protecting the staff and equipment against specific risks from inside the buildings (anoxia, explosion, fire) and, outside of the nuclear buildings, monitoring the release of air from the controlled areas in normal operation, and containing the radioactivity released during accidental events.

Helium cooled pebble bed (HCPB) Blanket concept based on solid breeder in form of a pebble bed and cooled by helium.

Magnetohydrodynamics (MHD) As applied in blanket technology, refers to the influence of the electromagnetic forces on liquid metal breeders, arising from the interaction of the confining magnetic field of the reactor with currents induced in the liquid metal as it flows across the magnetic field.

Self-coolant lithium lead (SCLL) Blanket concept based on lead-lithium used both as breeder and coolant. To allow this feature the selected structural material is SiC_f/SiC.

Test blanket module (TBM) A small (volume $\sim 1 \text{ m}^3$) tritium breeding module which will be exposed to the DT neutron flux in the ITER tokamak to test the performance of prototypical tritium breeding blankets for a DEMO fusion power plant.

Tritium breeding ratio (TBR) This represents "the ratio between the number of tritium atoms produced in the reactor and the number of neutrons made available from the thermonuclear reaction".

Water cooled lead lithium (WCLL) Blanket concept based on lead-lithium breeder and cooled by water at fission pressurized water reactor (PWR) coolant conditions.

Introduction

The breeding blanket (BB) is a key sub-system of the D-T fusion power reactor (FPR) with the main functions of removing heat to use for electrical production, and generating tritium that is one of the fuel elements used for sustaining the thermonuclear D-T reaction. The blanket will also fulfill several secondary functions, i.e., neutron shielding, provision of the first wall-plasma interface and containment of radioactive products. To perform all of these functions effectively, the blankets will surround the burning plasma

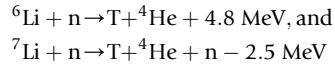
with a high coverage factor (usually greater than 0.85), representing the largest in-vessel system of the fusion reactor; their coolant and breeder features strongly influence the technologies employed in the plant design for the overall primary heat transfer system (PHTS) and for tritium management within the facility.

This article presents an overview of the principles of the breeding blanket design and of the main related technologies. Although we have tried to keep the description of this system as general as possible and therefore applicable for any type of fusion reactor, the particular design solutions and the quantitative values presented and discussed here refer to recent studies for a demonstration reactor (DEMO) which is currently considered as the next step after ITER. Our present understanding of DEMO is a fusion device based on the tokamak magnetic configuration of about 1–2 GW of fusion power,¹ relatively large dimensions (major radius of 8–9 m), low power density with an average neutron wall load (see section “Heat removal”) not larger than 1–1.5 MW/m² and a plant life time of about 6–7 full power years. We refer to (Federici et al., 2018) for an overview of the general design and performances of this kind of DEMO device that is considered in the European fusion road map for the breeding blanket development. Extension to blanket application to other magnetic configurations (e.g., stellarator) or inertial confinement will be briefly considered in section “Blanket for tokamak and stellarator” and “Blanket for inertial fusion applications”.

Tritium production and extraction

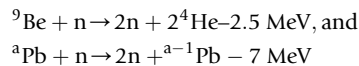
A fusion power reactor consumes about 153 g of T to sustain each GW of installed fusion power in a full power day; this means that for 25 full power years (FPY) a 2-GW power plant requires about 2.8 t of tritium. A comparable quantity of D (about 1.9 t) must also be provided. While D is present in nature and can be extracted with already consolidated technologies (e.g., it is used in fission technology in the CANDU reactors), tritium is rare in nature due to its half-life of 12.3 years. Industrially usable quantities of this radioactive hydrogen isotope can be produced as result of human activity only. At present it has been estimated that about 25 kg of tritium produced by heavy water reactors (mainly in Canada, Korea or Romania) are available for fusion technology applications (Pearson et al., 2018). However, it is not feasible to supply a D-T reactor externally with the required quantities of tritium, and therefore an FPR should be able: (1) to generate itself all the tritium that it will use during its operation; (2) to rely on an initial charge of tritium (estimated to be several kg) to start the plant and sustain the fueling for the first few days to maintain operation until the breeding blanket and its associated Tritium Systems are able to supply the Fuel Cycle with newly-bred tritium; (3) to produce, at the end of the reactor lifetime, a charge of tritium able to support the start-up of new power plants. These three requirements are usually encompassed under the general plant requirement of “tritium self-sufficiency.”

The production of tritium in the blanket exploits the neutrons generated during D-T reactions within the burning plasma. Lithium is central to this process, as its neutron capture reaction results in the production of tritium. Natural lithium is composed of two isotopes ⁶Li and ⁷Li, with an abundance of ⁶Li of about 7.5% (atomic). Both isotopes can generate tritium via neutron capture, but ⁷Li releases an additional neutron that can be used for a further breeding reaction:



The efficiency with which the reactor produces tritium can be quantified by the Tritium Breeding Ratio (TBR); this is commonly defined as “the ratio between the number of T atoms produced in the reactor and the number of neutrons made available from the thermonuclear reactions.” It is clear that self-sufficiency can be reached only with a TBR > 1. Taking into account that only one new neutron is made available by a D-T reaction (that consumes one atom of tritium and one of deuterium), but that not all of these neutrons will react with a lithium atom, this balance can be sustained only if a considerable neutron multiplication is achieved in the blanket to compensate losses. The loss of neutrons is due to the incomplete coverage of the plasma by the blankets, the limited thickness of the blanket layer and the unavoidable parasitic absorption in other blanket elements, such as structural material and other constituent elements in Li-compounds.

Although ⁷Li can contribute to the neutron multiplication function, the reaction cross-section is small and concentrated only in the medium energy spectrum, so that the use of natural lithium for this purpose does not, in general, satisfy the requirements. Only in combination with very low neutron absorbing materials and large blanket thickness would it be possible to obtain a sufficient TBR. A better strategy is to substitute ⁷Li with a more effective neutron multiplier and enrich the fraction of ⁶Li in the lithium used (to 50–90% according to various designs) in order to capture as large a fraction as possible of neutrons in the low energy region of the neutron spectrum, where ⁶Li has the higher cross section. This leads to the use of elements such as Be or Pb in the blanket material composition. These exhibit a more effective (n, 2n) reaction in a broader neutron energy spectrum:



The physical form of lithiated material for tritium production in the blanket provides a first classification of blanket concepts, namely “solid” and “liquid” breeder blankets. The former make use of solid Li compounds like ternary ceramics (e.g., orthosilicate,

¹Fusion power is defined here as the power directly released by the thermonuclear D-T reactions in the plasma (with an energy of 17.6 MeV per single reaction).

metatitanate, metazirconate) or Li-oxide. The ternary ceramic can be adopted up to $\sim 1000^\circ\text{C}$, but somewhat less for the oxide, which exhibits degradation of mechanical properties. Under thermal cycles and neutron irradiation solid breeders suffer from damage and fragmentation; for this reason, a common use of these materials is in form of a pebble beds, to reduce the modification of their basic properties (e.g., effective thermal conductivity of the pebble bed) in case of a further fragmentation and to allow a control of the breeder temperature during the whole lifetime. The thermal conductivity of these solid beds is in any case low; if cooled by surrounding plates or tubes containing the coolant, they can be used only as thin beds of about 1–1.5 cm. There are several criteria for the selection of the solid breeder material, namely high Li density, low activation (for rapid recycling after irradiation), better mechanical and thermal properties, high chemical compatibility with structural materials, less hygroscopicity for easy handling and mature manufacturing processes. None of the candidate materials are ideal, so that compromises among the various properties are required; among the materials mentioned above, the current preferences in the design are for lithium orthosilicate (Li_4SiO_4) produced by a melting-spraying process, lithium metatitanate (Li_2TiO_3) produced by wet-chemistry and sintering (Tsuchiya et al., 2007), or variants such as the KALOS material ($\text{Li}_4\text{SiO}_4 - 30\% \text{Li}_2\text{TiO}_3$) that is the reference in the EU program (Knitter et al., 2013). Be and Be alloys are the natural neutron multiplier choice to complement the solid breeder; they can be used in form of rods, slab or pebbles. A disadvantage of Be is its chemical reactivity with water, causing hydrogen production; at temperatures higher than 600°C this reaction is exothermic and constitutes a serious safety issue if the blanket uses water cooling. At high temperature the mechanical properties are poor and the swelling excessive; this suggests an upper operational limit of about 650°C . However, tritium retention constitutes the major concern: the simultaneous production of He and tritium in the material during the irradiation chain produces an inventory of tritium trapped in the helium bubbles that grows in the material grains. This tritium is difficult to release at the operational temperatures and can accumulate over time, creating safety concerns (Chakin et al., 2020). The recognition of these issues has led to the development of Be intermetallic compound (Nakamichi et al., 2018), such as Be_{12}Ti or Be_{12}V . In particular, titanium beryllide has been proposed as a solution for the present design of the European HCPB Blanket (see section “Blanket architecture”) in the form of hexagonal rods; it reduces the neutron performance slightly, but improves the compatibility with water, increases the operational temperature and allows a rapid release of the tritium. Beryllides are proposed in Japanese, European, Korean and Chinese blanket concepts, also in combination with water cooling.

Liquid breeder blankets use breeders in the form of liquid lithium or lead lithium eutectic (Pb-15.8% at Li, labeled usually as PbLi). Molten salts, such as FLiBe or FLiNaBe, have also been proposed in several concepts. Liquid breeders can be recirculated outside the blanket, carrying tritium to an external processing plant that allows tritium extraction and purification. All these breeders can also be used as a coolant (see later) in advanced blanket configurations. The advantage of liquid breeders is the practically unlimited lifetime, which contributes to long blanket availability; in fact, breeder damage due to creation of impurities or Li burn-up can be compensated in the ex-vessel reprocessing plant. High thermal conductivity, as in liquid metals, enhances heat removal features. Principal disadvantages are chemical reactions with water, air and other materials; this can be a serious safety issue in the case of accidents involving liquid Li, and a moderate risk for PbLi reacting with steam. Corrosion of structural material is an operational issue; usually coating is necessary to mitigate this effect. The dose rate level of the PbLi is high level; at short cooling times (less than 1 month) the dose rate is dominated by the Pb activation products and afterwards by radionuclides generated from the specified impurities, in particular ^{54}Mn . Volatile transmutation products such as ^{210}Po , a strong alpha emitter, are also generated in chain reactions from Pb. Helium, which can form bubbles, is also produced in the Li reactions. Helium should be removed to avoid accumulation in the PbLi loop; otherwise, if it accumulates in the blanket, it can cause local blockage of the flow. A further issue for electrically conducting liquid breeders is the influence of MHD in the flow. This issue is critical when the liquid metals are used as a coolant (see later), but it can also affect the local flow of the breeder and impact important functions such as heat and tritium transport (Bühler and Mistrangelo, 2017).

The tritium extraction process from the breeder is also related to the type of breeder used. In the case of solid breeder blankets, gas purge flow can be used to extract the tritium from the breeder material in blanket; usually an addition of H or D to the carrier gas (e.g., He) is foreseen to facilitate the tritium extraction per isotopic exchange. The resulting composition of the purge flow shows the presence of HT and HTO at low concentrations ($\sim 1 \text{ Pa}$) in the outlet gas flow. HTO can be extracted by cold traps ($< 0^\circ\text{C}$) or reactive molecular sieve beds (RMSB); for HT the extraction can be performed using cryogenic technology at liquid nitrogen temperature (cryogenic molecular sieve bed, CMSB) or using getter beds at room temperature (Cristescu and Draghia, 2020). In general, all the mentioned technologies are “batch processes” in which the individual getters are installed as a pair: one operates in absorption mode treating the purge gas coming from the reactor, while the other is in regeneration, releasing the H isotopes to the fuel cycle. The dimensioning of these getters and the frequency of switching between the two operating modes are critical parameters for maintaining low tritium inventories and high extraction efficiency of the systems. This technology exists and is used in industrial applications; its application in fusion must be optimized in terms of energy consumption, especially if cryogenic technologies are adopted. Some studies have been undertaken to develop continuous extraction processes, e.g., using “membrane reactors” (Demange, 2011).

In the case of liquid breeders, the extraction process is performed outside the vacuum vessel; hence, the liquid has to recirculate in a dedicated external loop where an extraction unit transfers tritium to the fuel cycle. The recirculating loop should be located near the tokamak to reduce the breeding quantities and to be strongly confined and efficiently shielded within the nuclear island to avoid the spread of the highly activated breeder and its transmutation products. If PbLi is used, $\sim 1000 \text{ t}$ of breeder (e.g., in the WCLL presented in section “Blanket architecture”) has to be recirculated outside the reactor core, leading to a replacement of the blanket inventory approximately 10 times per day. Several processes have been proposed for the tritium extraction (Utili et al., 2019), such as Permeation Against Vacuum (PAV), gas stripping in Gas Liquid Contactors (GLC), or direct exposition to vacuum in the

so-called Liquid Vacuum Contactors (LVC). All of these are continuous processes that are installed in a bypass of the recirculating loop. The principal issue is to demonstrate a technology at high specific extraction efficiency so that it is possible to scale it to systems suitable for operation in DEMO conditions. Extraction processes from liquid Li and molten salts for blanket applications are less studied and in general more challenging e.g., (Tebus et al., 1998) or (Fukada et al., 2017).

The fuel cycle

The processes described above concerning tritium production in the blanket and its transport to an external plant for extraction and removal is an integral part of the general architecture of the fuel cycle for the FPR; it constitutes the so called “outer cycle,” to distinguish it from the main part of the plant (“inner cycle”) that is responsible for the function of managing the exhaust gas coming from the plasma chamber, so as to remove helium produced in the D-T reactions, to extract other impurities formed by plasma-wall interactions, to provide the “fueling” for the plasma and to inject plasma enhancement gases (PEG) needed for plasma stability control. Furthermore, the fuel cycle in dedicated systems will recover tritium from the blanket coolant, which accumulates a tritium inventory due to permeation at the high operation temperatures within the blanket. The “outer cycle” supplies the “inner cycle” with the amount of tritium necessary for the FPR operation; in ITER where the “outer cycle” is missing, the “inner cycle” will have to cope only with the externally supplied tritium provided from the long-term storage.

The design of the fuel cycle is guided by several strict requirements: firstly, to reduce as much as possible the tritium inventories in storage, in the tritium plant and in the vacuum vessel. Regulatory limits (such as the no-evacuation criterion) will set upper boundaries for such inventories which then have to be further reduced following the ALARA approach (as low as reasonably achievable). Furthermore, the scale and cost of the plant should be considered and minimized. The key to reaching a suitable operational point is to select architectures and technologies able to reduce as much as possible the cycle time of the associated processes; in addition, a suitable requirement for the TBR² has to be selected to optimize quantities such as the initial storage, but also to minimize the build-up of inventories. The quantification of the required TBR, as well as that for the initial charge of tritium or for the storage of tritium for a new DEMO start-up, is the result of an optimization of the fuel cycle calculations; it depends on the plasma reaction efficiency, the time constant of the fueling processes, the trapping of T in materials, T decay and performance of the blanket extraction systems. Safety requirements will play a large role in prescribing limitations of tritium inventory in the various parts of the reactor (e.g., vacuum vessel, tritium plant, long term storage). From recent assessments performed for the EU DEMO, a blanket TBR performance of 1.05 is specified (Fischer et al., 2020). The minimum required TBR is defined mainly by the amount of tritium sequestered throughout the fuel cycle systems (saturation of structures and materials) and the time needed to reach this quasi steady-state. Plant lifetime estimates of the tritium inventory evolution do clearly show that under the assumption of a constant TBR above the minimum performance, and for reasonable integral availabilities above 30%, there will be net production of tritium over the lifetime (Coleman et al., 2019). This allows the export of tritium to feed the start-up of the next reactor.

It can be shown by simple mass balances that the design driving load on the fuel cycle is given by one performance indicator, the so-called tritium conversion rate, which is defined as the product of the effective tritium burn-up fraction times the fueling efficiency. The D-T injection rate the fuel cycle has to provide is inversely proportional to the tritium conversion rate. Due to losses of the injected fuel from the core and to limited reactivity of the fuel in the core, only a small fraction of the D-T injected into the plasma will burn (1% in ITER, targeted to be higher than 2.5% in DEMO), while the remainder, together with helium and impurities, will be exhausted and sent to the tritium plant for treatment. The plasma D-T reaction is fed by the fueling system that injects cryogenic D-T pellets. The fueling efficiency, which is defined as the ratio of the plasma fueling rate to the fuel injection rate, is limited. This reflects the fact that there are significant matter losses of the injected pellets on their way to the plasma and only part of the ablated pellet mass remains inside the plasma core due to particle transport processes near the plasma edge. For ITER, the achievable fueling efficiencies are unlikely to be higher than 50%.

A direct consequence of the limited tritium conversion rate is the fact that the tritium throughput that the inner fuel cycle has to remove from the plasma chamber is about a factor of 100 higher than the tritium throughput that the outer fuel cycle takes from the blankets; significant increases of plasma performance in this area cannot be expected. A simple scaling of the ITER fuel cycle to FPR dimensions will result in excessive tritium inventories and does not lead to a suitable solution for DEMO in terms of operational complexity, facility size, but especially for the avoidance of prohibitive inventories. Recently an innovative architecture of the fuel cycle has been proposed for the EU DEMO design in (Day et al., 2016) and (Day et al., 2019), see Fig. 1, which is currently considered as the most promising solution to mitigate the inventory issue.

The suggested architecture features three loops and introduces a separation function close to the divertor, which extracts pure fuel and recycles it via a bypass to the tritium plant, the Direct Internal Recycling (DIR). This shortcut re-routes, within a short processing time, the major part of the tritium in the exhaust gas, so that only a minor fraction will have to be routed through the tritium plant, resulting in reduced plant size under nominal operation conditions. This fraction is again distributed in a faster loop with protium

²“Required TBR” defines the tritium production that should be performed in the fusion reactor to enable continuous FPR operation in the tritium self-sufficiency regime. The “required TBR” should be not confused with other values (generally higher values) that are reported in blanket design reports. Usually these latter values refer to calculated performances in neutronic analyses that are affected by data and modelling uncertainties, and use heavy modelling assumptions, e.g., absence of auxiliary systems that require a reduction of blanket coverage. Whether this level of performance can cover the needs of “required TBR” must be carefully checked in the design.

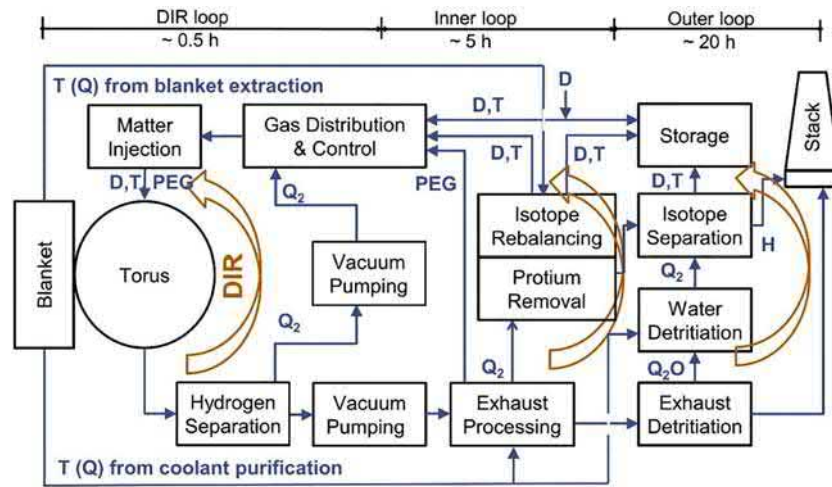


Fig. 1 Schematic of the fuel cycle architecture for a FPR ($Q = H/D/T$, PEG denotes plasma enhancement gas). This picture has been done by the co-author Day C (2020). It is his intellectual property, but it has been already used also for another publication. *Fusion Engineering and Design* 155: 111553.

removal and isotope re-balancing, which produces DT in the specified composition, and sends the remaining gas to an outermost loop for the tritium recovery, which is very similar to the classical outer fuel cycle (Lawless et al., 2017). This last loop also includes the detritiation of tritiated water. From the perspective of the tritium plant, the introduction of the DIR loop with a nominal separation fraction of 80% would leave a load on the DEMO tritium plant which is a factor of four lower than in a once-through architecture. The introduction of DIR decouples the tritium plant operation from the tokamak operation. Any potential improvement in the tritium conversion rate would further improve the fuel cycle efficiency. The DIR concept builds, moreover, on continuous processes, avoiding accumulation of tritium in each process stage. This is also the reason why a second loop (isotope rebalancing and protium removal) has been introduced to establish, in a quasi-steady-state manner, the fuel mixture that is needed to provide, in combination with the DIR return stream, the fuel composition requested from the pellet injectors. This architecture therefore results in significantly reduced D-T cycle times and tritium inventories, as indicated in Fig. 1, and helps to relax the TBR requirement for the same performance in terms of start-up inventory.

The technical realization of DIR (KALPUREX process in Giegerich and Day, 2014) involves metal foil pumps and linear diffusion pumps for primary pumping, thereby replacing the batchwise cryopumps that are used at present in most fusion devices. The metal foil pump is a novel technology for hydrogen separation that allows the extraction of a pure hydrogenic stream from the exhaust gas mixture under the low pressure conditions close to the divertor, i.e., conditions which exclude the use of the classical membrane separation technology (Hanke et al., 2020). It exploits the phenomenon of superpermeation, which enables suprathermal hydrogen particles to permeate through a metal foil with a surface barrier, whereas all other species in practice do not permeate through the foil. Both product streams of the metal foil pump need further vacuum pumping for compression. To meet the requirements of this process step, continuous pumping systems are being developed, which would make the processing time and resulting inventories negligible. A development program of mercury based continuous vacuum pumping solutions is under way (as mercury has practically zero solubility for tritium) using mercury vapor diffusion pumps backed by liquid ring pumps with mercury as the working fluid.

The main function of the exhaust processing sub-system is to separate the arriving gas mixture into three streams, namely (i) the fuel stream for re-use, (ii) the stream of PEGs via appropriately sized decay tanks for the activated species, and (iii) the remaining detritiated impurity stream. For inventory and continuity reasons, classical permeation membrane technology with a number of suitably sized palladium-silver permeators will be employed for bulk hydrogen separation. Also for tritium recovery from tritiated impurities, ITER technologies based on membrane reactors may well serve as a good candidate. For PEG separation there exist several technologies that have to be chosen once the type of PEGs to be used is known: options include cryogenic trapping or molecular sieving by porous membranes (Tosti et al., 2017).

Classically, the central technology in the outermost fuel cycle loop is isotope separation via cryogenic distillation to separate the incoming hydrogenic gas stream into the different pure hydrogen isotopes. Cryogenic distillation suffers from the inventory issue in that there are significant quantities of hydrogen isotopes being liquefied, and this is the main factor responsible for the long processing times assumed for this fuel cycle loop. ITER has established the requirement to provide the different isotopes in pure form, mainly to provide experimental flexibility within the ITER research program. This is not required for DEMO operation, where essentially one and the same plasma discharge will be repeated. It is therefore sufficient to provide mainly D_2 and T_2 at the requested composition (around 1:1). This situation brings back a technology which had been discarded at ITER, because it does not achieve the same purity requirements as cryogenic distillation, but it does have the advantage of shorter processing times. It is embedded in the additional system block which provides the intermediate processing loop prior to the classical isotope separation in Fig. 1. The

task of isotope rebalancing is to re-establish the required fuel mixture composition before reinjection into the torus, while protium removal separates protium, which inevitably enters the fuel cycle via outgassing or replacement reactions. The resultant fuel stream with adjusted mixture composition directly feeds the gas distribution and control box, as does the permeated stream from the metal foil pumps. The operation principle of the intermediate loop is based on membrane-coupled temperature swing absorption (Neugebauer et al., 2020). The main feature is the combination of two separation principles—Isotope dependent gaseous diffusion through porous membranes, and hydrogen-metal interactions by temperature cycling of the gas between two materials, one with a positive isotope effect that absorbs the heavier isotopologues more readily, and the other with an inverse isotope effect which preferentially absorbs the lighter isotopologues. The addition of a membrane acts as pre-separation, thereby reducing the tritium inventory and separation time. The joint unit for the two functions making up the intermediate processing loop consists of two columns. In the first, a flow enriched in tritium will be sent to the gas management system. In the second column all the protium plus a certain amount of deuterium are separated and sent to isotope separation. One advantage of the absorption-based technology is that it provides some flexibility to work at different target points so that the composition of the fuel injected into the tokamak could be adjusted—although slowly.

The discharge of tritium must be minimized in accordance with the ALARA principle and must, at least, remain below the regulatory discharge limits. The exhaust detritiation system is a key component in ensuring that this objective is met, by capturing any tritium, from waste process gas or potentially contaminated air, in both gas and water vapor form. This includes input streams coming from heating, ventilation and air conditioning (HVAC) systems, glove box vent systems and air from other rooms with tritiated components. The tritium is prevented from leaving the facility into the environment and instead will be made available for re-use. As in ITER, it is expected that exhaust detritiation on DEMO will operate on the principle of converting tritium to tritiated water, consequentially, this is the main feed of water into the water detritiation system.

The leading technology choice for water detritiation is combined electrolysis and catalytic exchange (CECE), which is an industrialized process well developed for heavy water detritiation of CANDU reactors (Spagnolo and Miller, 1995). It is believed that the design for DEMO can be derived from that developed for ITER, however at a clearly larger size. Due to the direct electrolysis, the CECE process is very energy intensive, as are alternative technologies at a similar readiness level. A more efficient technology is desirable, and R&D is ongoing in this area, although, at present, at a rather fundamental level (Lozada-Hidalgo et al., 2016).

To complete the overview of the tritium systems connected to the fueling function, we present in the following the tritium recovery from the main coolant. This “purification” technology is essential in the design of DEMO reactors that use water or gas as coolant (see section “Blanket architecture”) where the contamination (mainly due to permeation in the blanket structures) of the PHTS with several % of the tritium produced in the breeder is an issue for the tritium control in the reactor. Obviously, the coolant purification system (CPS) will be first-of-a-kind at the DEMO scale (Santucci et al., 2020). The main functions of the CPS are (i) extraction of tritium from the coolant to below specified levels, (ii) removal of solid, liquid and gaseous impurities, and (iii) control of the coolant chemistry (by adjusting the oxidation-reduction potential of the coolant). The largest challenge comes from the huge coolant flow rate (several 1000 kg/s of water or helium). The general strategy is, therefore, that only a fraction of the coolant flow is processed by the CPS, to be determined by the assumed CPS efficiency, the total coolant flow rate, the tritium permeation rate from the blanket to the coolant loop, and the allowable tritium concentration inside the coolant.

For identification of a promising helium CPS technology, one can start from the CPS of the helium-cooled test blanket modules (TBM) of ITER, in which the tritium removal foresees the transformation of HT into HTO by the use of high temperature oxidizing beds and subsequent adsorption of the resulting tritiated water in molecular sieve beds at room temperature. The process proposed for the ITER CPS uses mature and consolidated technologies (that are also being used in fission applications). In principle such a process can be scaled up to fulfill DEMO requirements, but it comes with some drawbacks and operational complexity. An alternative solution is therefore being explored based on new, high capacity, non-evaporable getter (NEG) materials which would avoid the oxidation step.

For detritiation of the water coolant, one may use the same CECE technology mentioned above. However, the water CPS functions are only manageable if a suitable strategy for reducing T permeation from the blanket into the coolant is applied; initial estimates specify required permeation reduction factors between 100 and 1000. In the latter case, an ITER-sized system would be sufficient to manage the tritium inventory in the coolant loop. Given the importance of permeation barriers, a strong R&D program is currently ongoing world-wide, including demonstration under reactor conditions. The alternative approach for the water CPS would be an off-line plant, assuming that the water coolant in the primary cooling system is replaced with fresh water after 1 year of DEMO operation.

Heat removal

As described more in detail in the thermal load conditions in operation for in-vessel components like the blanket are characterized by three major sources of heat. 80% of the fusion power leaves the plasma in form of neutron kinetic energy. The neutrons then interact with the surrounding materials releasing their energy progressively while attenuating the flux with an exponential law. The plasma source intensity can be quantified locally by the neutron wall load (NWL): NWL is the energy flux transported by the 14 MeV neutrons crossing a reference closed surface surrounding the plasma ring (usually a reference surface that follows the FW contour). In a tokamak this distribution is almost axisymmetric so that the neutron wall load can be described as a function of the poloidal angle. In a stellarator the distribution will depend on the poloidal and toroidal angles as the closed magnetic surfaces

are helical with fully 3D geometry. In the present European DEMO device, the maximum NWL will be around 1 MW/m^2 for a total fusion power of $\sim 2 \text{ GW}$. The heat deposition decays exponentially through the radial blanket thickness, with a peak value in the plasma-facing region not exceeding 20 MW/m^3 .

The remaining 20% of the fusion power associated with the alpha particles, together with the additional power injected by the external heating provided by the H&CD Systems, leaves the plasma either in the form of electromagnetic waves or via plasma thermal conduction and particle convection processes. The electromagnetic waves are emitted mainly via Bremsstrahlung and synchrotron radiation and via impurity line radiation: the radiation will produce a power load of about 0.3 MW/m^2 on the surface of the blankets with peaks at the equatorial plane. The plasma particles (ions and electrons) contributing to conduction and convection follow the magnetic field lines in the scrape-off layer (SOL), moving toroidally and poloidally in the direction of the divertor targets. In the course of this process (which can occur over a field line length of $\sim 100 \text{ m}$ in the case of DEMO) particles can impact tangentially on the blanket surface, producing localized hot spots.

To remove and transport the heat to a power generation plant, a range of materials and corresponding technologies have been proposed for the blanket cooling: water, gases, liquid metals or molten salts. Water has been proposed at the thermo-hydraulic conditions of fission PWRs that foresee a pressure of 15.5 MPa and water temperatures in the range $285\text{--}335^\circ\text{C}$; this specification allows retention of the liquid phase with sufficient margin to critical heat flow. The advantages are excellent cooling characteristics (high density, heat capacity and heat transfer coefficient) that lead to systems with low pumping power and low temperature differences between inlet and outlet. The level of temperature is, however, relatively low, leading to low thermo-hydraulic efficiency of the power generation systems (a minimum of $\sim 33\%$ is desirable) and issues with ferritic-martensitic steels under irradiation (DBTT shifts to room temperatures after $20\text{--}30 \text{ dpa}$, as described in Litnovsky, 2021). An increase in temperature is not possible while maintaining PWR operating conditions; to overcome this limitation supercritical water conditions can be selected, but the performance deteriorates and the issue of compatibility with steels (corrosion) becomes critical. The use of PWR technology ensures the availability of components and equipment (i.e., pumps, steam generators, pressurizers, valves) with high technological readiness level (TRL). In spite of the extensive use of these technologies in fission applications, several issues are still open for their application to fusion technology. In particular the energy of the neutrons is higher than in fission, enhancing water radiolysis and activation, and, in addition, tritium can be trapped in water, increasing its inventory in the coolant if not efficiently removed.

Among the gas coolants, helium is the preferred option; it is chemically compatible to all the other materials in operational and accident conditions and is not modified significantly under exposure to neutron or ionizing radiation. Helium has no intrinsic limitation in temperature range of application, and a high temperature range is favorable to increase the thermo-hydraulic efficiency of the power generation cycles; the limitations are due to the materials that constitute the pressure containment (degeneration of chemical and mechanical properties at very high temperature). As a gas, the cooling capability is less effective than water, since the density is much lower; this increases the resulting volumetric flow and the dimensions of the components of the cooling system. To remove comparable heat power without increasing too much the dimensions of the associated flow area (and then the dimensions of the components), helium systems have to operate at high pressure, high velocity and with a larger difference between inlet and outlet temperatures. However, a large thermal difference makes the mechanical design more demanding in order to minimize potentially high thermal stresses, and, to sustain high volumetric flow, high pumping power is required so that the pressure drops have to be minimized. As helium is almost transparent to neutrons, it doesn't contribute to the shielding; this should be carefully considered in the blanket design, avoiding possible neutron streaming through penetrating pipes from the plasma side to the rear part of the structure. For blanket configurations of the first generation, it is proposed to use helium in a moderate temperature range of $300\text{--}500^\circ\text{C}$ (inlet-outlet) at a pressure of 8 MPa . Velocities of up to 80 m/s and turbulence promoters (Arbeiter et al., 2017) are required in the FW to remove high heat fluxes.

In spite of exceptional cooling characteristics (high density, low operational pressure, low volumetric flow, etc.), liquid metal cooling is highly penalized in MCF reactors. The presence of strong magnetic fields generates relevant magnetohydrodynamic (MHD) issues, such as very substantial pressure drops and turbulence suppression, that frustrate any other advantages (Smolentsev et al., 2010). In fact, electrical potentials are generated in the liquid metals as the result of interaction between the magnetic field and liquid flow. In particular, if electrically conducting materials such as steel are used for the containment structure, the conducting walls allow the closure of the flow path of eddy currents that interact with the magnetic field, producing forces that oppose the liquid movement. With a magnetic field of $4\text{--}9 \text{ T}$ experienced in the blanket, the velocity of liquid metal should be limited, in the worst case, to only several mm/s. The only way to allow a sufficient flow velocity (that is essential to reduce duct dimensions and increase heat transport) is to use ceramic structural materials (like SiC_f/SiC) or, alternatively, electrical insulators between the liquid metal and the metallic structure (e.g., alumina coatings or ceramic inserts). The attempt to develop this technology is illustrated in advanced blanket proposals such as the DCLL or SCLL blanket concepts (see Section "Blanket architecture").

Molten salts have been proposed as alternatives to liquid metal in order to maintain the same advantages in term of low pressure, low volumetric flow and the possible use as both coolant and breeder, while avoiding MHD effects. This is possible as their electrical conductivity is several orders of magnitude lower than that of liquid metals. The use of molten salts requires, however, a complex chemical control; the TBR is not exceptional, the melting temperatures relatively high and, in particular, they are extremely corrosive with structural materials. Even though blanket concepts based on the molten salt technology have been proposed several times since the 1970s, this technology has never provided the basis for a concrete DEMO design concept.

Neutron shielding

In the design of a blanket the neutron balance leading to a sufficient TBR is the high priority. To achieve this, the selection of blanket materials and their distribution in at least the first 50–80 cm are crucial to avoid large parasitic absorption. For instance, in a solid breeder blanket like the HCPB, 20% of the volume within the first 50 cm of the outboard region is occupied by ceramic breeder, while 50% is occupied by Be₁₂Ti, 15% by EUROFER steel and the rest helium. Suitable neutron performances can be realized with a compact and light design only, in which the structural part is dimensioned to be as thin as possible, manifolds of helium are avoided in the front region, and the mechanical structure is highly optimized to reduce steel content (which is responsible for the large parasitic neutron absorption).

Following this rationale, it is clear that the shielding is not the first priority in the design of the front part of the blanket. In this zone the neutron flux can be reduced by only a factor 10 without compromising the TBR. Once beyond it, the quantity of highly neutron-absorbing materials can be increased with the aim of satisfying shielding requirements. The first goal of the blanket shielding is to protect the vacuum vessel and in particular the mechanical and hydraulic attachment of the blanket to the vacuum vessel. Usually requirements are to reduce the cumulated dpa in an austenitic VV to lower than ~ 3 and to avoid excessive heating of the structures (to keep low temperatures in the design, enhancing safety features). Much lower values are of interest to reduce the VV activation and the large associated waste costs. In the design of particular components (e.g., welded piping connection or the entire VV) the production of helium via transmutation in steel should also be strongly limited; in fact, as a result of crack formation, values exceeding several at.% can already impede repair and maintenance operations that require welding of already irradiated material. The protection of the magnet systems is a further priority, but the vacuum vessel plays a major role in this respect; however, the blankets have to contribute to the shielding to reach the required plant limits: in DEMO this means a fast neutron flux lower than approximately $10^{13} \text{ m}^{-2} \text{ s}^{-1}$ and a power density than 50 W m^{-3} (Fischer et al., 2015).

Blanket concepts

Several blanket concepts suitable for the MCF FPR have been proposed since the 1980s based on different combinations of structural, coolant and breeder materials. However, the design has not evolved to more than a pre-conceptual level, where the feasibility of key technologies has not yet been fully demonstrated. ITER, the first thermonuclear device, will not be equipped with a Breeding Blanket, but only with a shielding structure (ITER “Shielding Blanket”), which is necessary to protect the vacuum vessel and magnets from the neutron flux and to remove the heat generated by the burning plasma. ITER will use only an external supply of tritium from fission reactors (about 16 kg for the entire possible operating lifetime (ITER Technical Basis, 2002)) and will not produce electricity using the relatively low water coolant temperature, which would make the process inefficient. For this reason, the design of this ITER component is almost irrelevant for the development of the breeding blanket, which is characterized, on the contrary, by the production of tritium and by high temperature coolant (at least greater than 300 °C) suitable for efficient electricity generation. Blanket technologies will, however, be tested in ITER in the form of small test blanket modules located in a few equatorial ports (see Section “The ITER test blanket program”).

Considering recent international developments, leading fusion communities, e.g., China, Europe, Japan and Korea, are proposing, as the next step after ITER, the construction of a national demonstration reactor based on the tokamak configuration. For this reactor, only few blanket concepts are under consideration using technology that seems achievable in the next 10–20 years. A common feature of these first-generation blanket concepts is the use of reduced activation ferritic martensitic (RAFM) steels as structural material; the development of these grades of steel has been started on a worldwide basis about 30 years ago with the ambitious target of withstanding neutron irradiation of more than 100 dpa and with activation characteristics that may allow their complete recycling in 100–200 years after FPR operation end. RAFM are 9Cr-steels that replace high activation elements (e.g., Mn, Ni, Mo) in the alloy with elements (e.g., W, V) that give corresponding material properties to the original alloy, but which have reaction products with a shorter half-life. These types of steels are under development by nations with a large fusion program using slightly different compositions: EUROFER in the EU (van der Schaaf et al., 2003), F82H in Japan (Jitsukawa et al., 2002), CLAM or CLM-1 in China (Huang, 2014) and ARAA in Korea (Chun et al., 2018). These materials have been produced already at semi-industrial scale, but the final in-pile characterization at high doses and, in particular, the definition of design rules that will make possible the final design is not yet complete, pending irradiation at fusion-relevant neutron energies. See (Litnovsky, 2021) for a more detailed discussion on fusion materials.

Based on these materials, manufacturing technologies for blankets have been tested in several programs. For the construction of the ITER TBMs these technologies are already being transferred at industrial scale, with final procurement foreseen towards the end of 2030. The technologies identified for the construction of the TBMs are, at present, the reference to which the DEMO manufacturing program is oriented; studies to extrapolate these technologies to DEMO conditions (e.g., large dimensions, longer irradiation time, stricter tolerances, reduced space, difficult inspection, etc.) have, at the moment, priority in the attempt to define a sound assembly route for these components. Taking into account the long time necessary for the qualification of a structural material for pressure vessel components within a nuclear environment, no materials other than RAFM steels are likely to be available for the structure of the first generation of breeding blankets. Alternatives, such as vanadium (Muroga, 2005) or SiC/SiC (Raffray, 1999), will be not qualified in this period, and typical ITER materials such as copper alloy and austenitic steel will not survive even the relatively low dpa levels foreseen for the proposed DEMO reactors (20–50 dpa). However, RAFM steels have critical issues, for

example, unexceptional thermal properties, which limit the heat flux that can be removed, and embrittlement under irradiation. In particular, the shift of the Ductile-to-Brittle Transition Temperature (DBTT) up to, or even above, room temperature during irradiation and the role of He-transmutation in increasing the total embrittlement are considered the most severe issues, and these are still not perfectly understood. What is known is that the major design parameter that impacts the damage is the operational (irradiation) temperature, which limits the dpa that the material can tolerate (and thus the operational lifetime). Only at relatively high irradiation temperatures (over 350–400 °C) is the shift in DBTT moderate (Gaganidze et al., 2006). Unfortunately, the temperature window—as foreseen for EUROFER in the present blanket concepts—ranges from 300 °C (even 285 °C, if water is used) up to 550 °C. Specifically, the worst conditions are at the front of the blanket, with the structure in contact with the coolant being at lower temperature but exposed to higher values of dpa. The issue is at present common to all the concepts; for blankets that use water cooling this level of temperature is intrinsic to the cooling regime adopted (where the temperature ranges over 285–335 °C in the PWR cycle). For blankets adopting helium cooling, the lower temperature can be increased, but this implies that the outlet temperature should be increased correspondingly to maintain a suitable ΔT to ensure high coolant performance; to accommodate these higher helium temperatures, the development of optimized RAFM steels for a higher operational window of, e.g., 400–650 °C, is required.

Blanket architecture

Fig. 2 show the typical architecture that has been developed for these first-generation blankets. The blanket is a box that contains the breeder, the coolant and shielding material. This box has a wall in front of the plasma referred to as the “first wall.” The rear part is a supporting structure, the “back plate”; the remaining walls are the side walls and two closing structures generally called “caps.”

The first wall (FW) protects the blanket from the plasma, removes the surface heat and confines the breeders and coolant; in addition, it must be relatively thin to avoid excessive absorption of neutrons before tritium breeding. High plasma radiation fluxes of 0.3–0.5 MW/m² have to be removed with an appropriate design of the first wall, which leads to a plate with a maximum of 30 mm thickness incorporating small cooling channels. In addition, interaction with the plasma ions and electrons of the SOL can produce local heat fluxes exceeding 5 MW/m² (Mitteau et al., 2013); taking into account constraints in the present structural materials (e.g., necessity of high neutron damage resistance or low activation criteria), these heat flux levels from the plasma cannot be accommodated with a solid wall (see 10 for similar considerations on an alternative “wetted-wall” design). For this reason, protection limiters are under development in the EU DEMO to limit the maximum local flux to values not exceeding about 1 MW/m², which is the limit of heat extraction with current technology using RAFM. The presence of an effective “limiter protection strategy” is critical for the realization of the present DEMO blanket concepts (Maviglia et al., 2018).

The zone beyond the FW and contained by the blanket box is the Breeding Zone (BZ). The function is essentially to breed tritium and remove heat, so that the main materials present in this region are the breeder/neutron multiplier (solid or liquid), the coolant

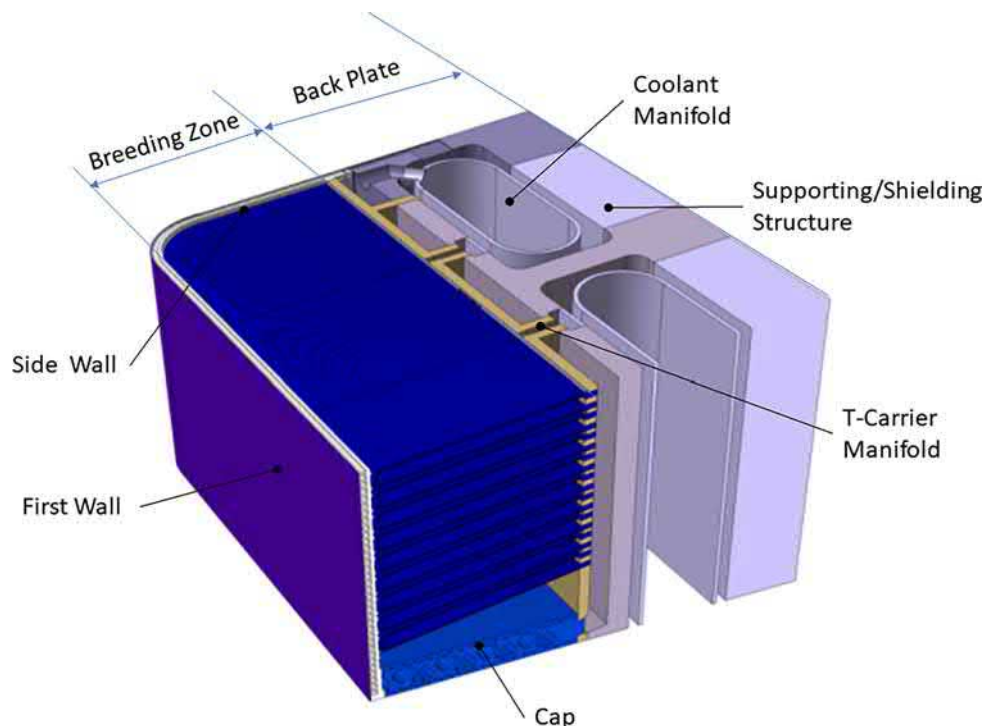


Fig. 2 Blanket architecture (functions of the various structures is described in the text).

and the structural material. All of the most-studied DEMO BB concepts share a common architectural design for the BZ: the breeder and the coolant are contained into two completely separate circuits with separate functions (“independent loop configuration,” see Fig. 3). The coolant loop (i.e., the Primary Heat Transfer System, PHTS) has the function to remove the heat; this heat has to be transferred to secondary systems that use it for electricity production. The coolant in these concepts is high pressure helium or water (8 and 15.5 MPa, respectively) at a temperature higher than $\sim 300^\circ\text{C}$. The breeder loop has to produce tritium and allow its

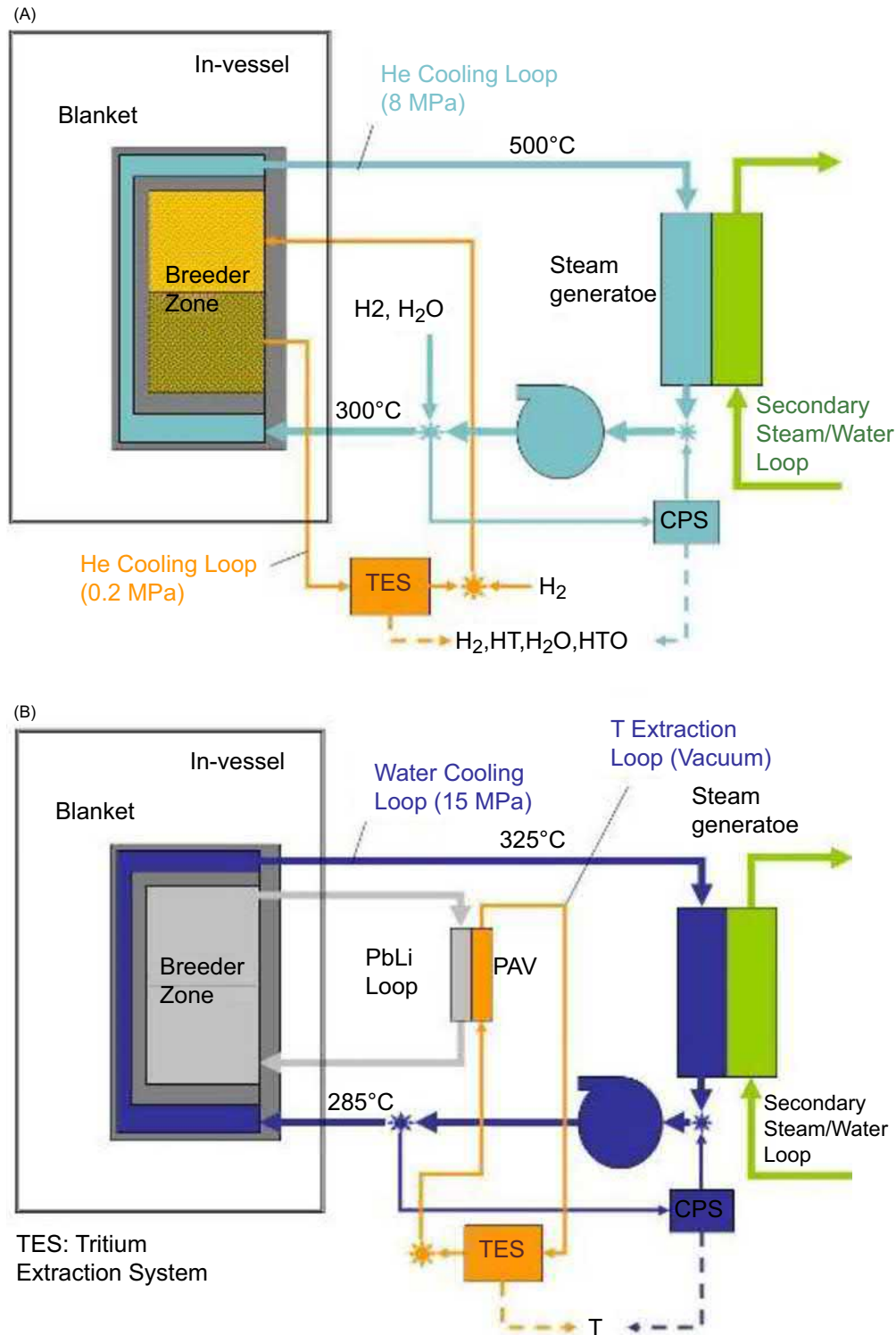


Fig. 3 Typical “independent loop configuration” for (A) a helium cooled solid breeder and (B) a lead-lithium water cooled blanket concepts.

transport (with a tritium carrier) outside of the vacuum vessel, where it will be pre-treated (e.g., tritium is removed from the carrier and concentrated) and sent to the Fuel Cycle. The advantage of this architecture is that tritium is concentrated in the carrier, where it can be more easily extracted/removed, avoiding dilution in the large volume of coolant. The coolant is not subject to direct contamination by tritium, and this both mitigates the issue of a potentially high tritium inventory accumulating (since achieving a simple and economic extraction would be challenging) and eliminates the possibility of tritium reaching the secondary circuit through a steam generator (if the direct cycle is used) or via a heat exchanger to a storage system (e.g., molten salts); should tritium reach the secondary loop, its possible release to the environment would be a major safety risk. A serious drawback of this configuration is the complex design of the blanket breeding zone, where the two loops are strongly interconnected. In fact, the coolant has to be distributed by capillary action to ensure the cooling of the breeder material; the coolant flows inside small tubes, or plates with channels, that are surrounded by the breeding materials. This means that a large surface will be necessary at the interface of the two independent loops (a figure of $\sim 13,000 \text{ m}^2$ is estimated in the whole blanket system for these concepts). This increases the design complexity (e.g., number and lengths of welding seams) and poses issues of tritium permeation from the breeder to the coolant loop.

Almost all of the most studied “near term” blanket concepts for DEMO follow this architecture; among these we can mention: the EU Helium Cooled Pebble Bed (Hernández et al., 2018), the Japanese Water Cooled Solid Breeder (Someya et al., 2019), the EU Water Cooled Lead-Lithium blanket (Del Nevo et al., 2019), the Chinese water-cooled blanket (Liu, 2019) and the Korean helium-cooled ceramic reflector blanket (Cho, 2018).

Various architectures have been proposed for advanced blanket concepts using different configurations for the breeder and coolant; the “self-cooled” configuration uses a coolant that is simultaneously the breeder; in this way the two loops can be unified and the overall design becomes very compact. The simultaneous transport of heat and tritium in a unique loop generates several concerns, however; as already noted, the tritium is diluted in the large volume of coolant and its removal necessitates very efficient plants with large dimensions. On the other hand, the high temperature of the coolant makes tritium confinement difficult; issues arise in the, potentially substantial, permeation into the steam generator/heat exchanger at the interface with the secondary systems. Fig. 4 shows an example of a self-cooled reactor that uses PbLi in the double function of coolant and breeder (Giancarli et al., 2003). In addition, a third configuration is the basis of a very interesting blanket concept developed in parallel in KIT (Germany) and in the United States, the “Dual coolant blanket.” This concept adopts two coolant circuits: one contains a pure coolant (usually helium) in direct contact with the steel structures, but the second uses a liquid breeder, as in in the “self-coolant” configuration. The electrical and thermal insulation between the two loops is ensured by means of insulating inserts (e.g., Alumina or SiC) inside the liquid breeder channels. This concept allows both the use of liquid metal cooling (at least for removing power generated in the breeder) and also the use of low temperature (up to 600°C) steels that are directly cooled by the non-(electrically) conducting coolant (i.e., helium). An example of a design using this blanket concept can be found in (Malang, 2011) and it is illustrated in Fig. 5.

The rear part of the blanket that contributes to closing the containment box is the back plate; this region has to incorporate the large manifolds for the coolant and the tritium carrier, the necessary shielding capability, the structural integrity of the blanket

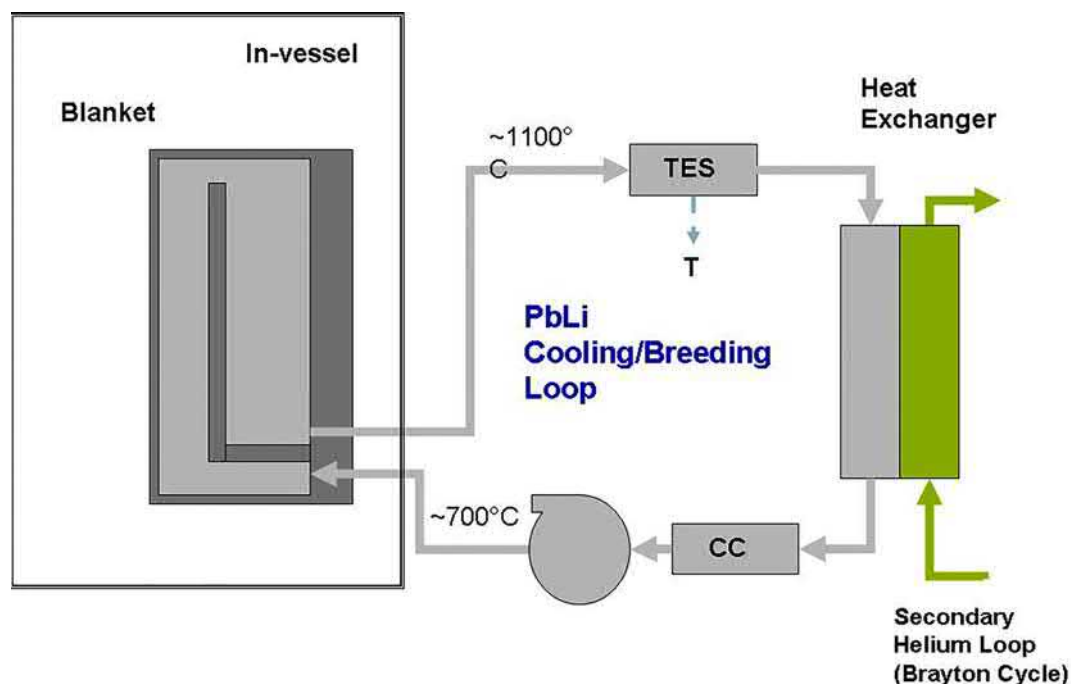


Fig. 4 Self cooled configuration, as discussed in the text.

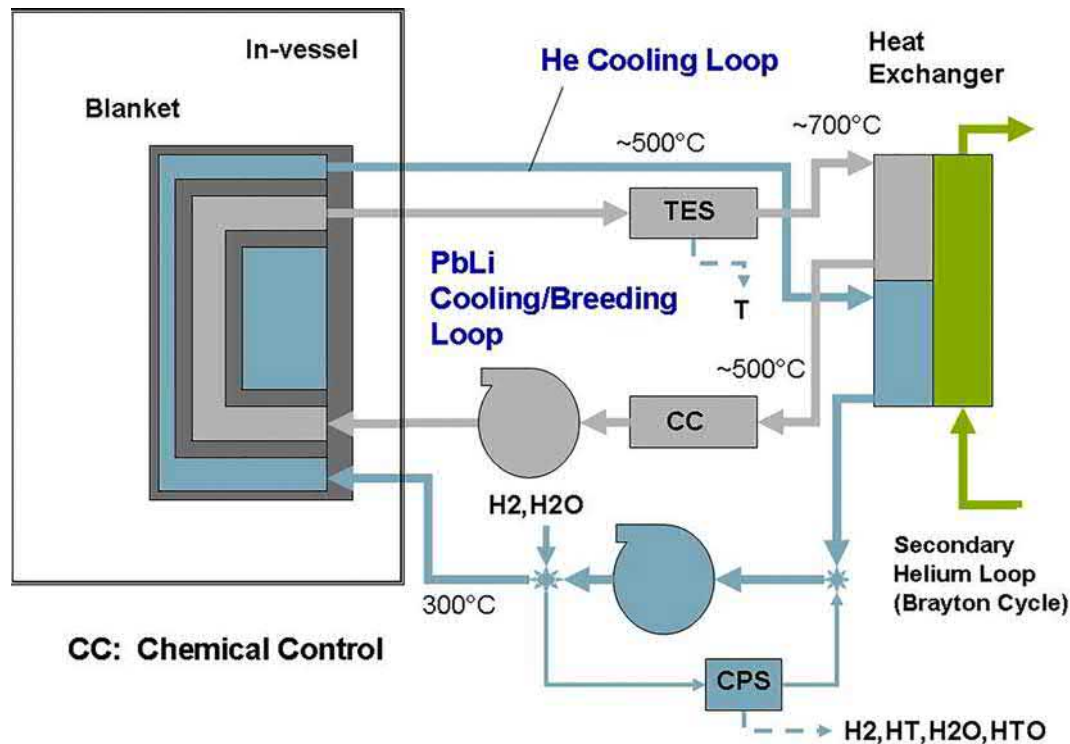


Fig. 5 Dual Coolant configuration, as discussed in the text.

replacement units for remote handling and the interface to the mechanical attachment system to the vacuum vessel. The design is strongly impacted by the concept of blanket replacement foreseen for the reactor; if a fully vertical maintenance concept (Keep et al., 2017) is adopted in DEMO, the back plate is a structure about 12 m high and 1.5 m wide, with a thickness of about 40 cm. It contains long poloidal manifolds to distribute the coolant and the tritium carrier to the FW and BZ from the connection to the external piping.

The ITER test blanket program

As already mentioned in Section “Blanket concepts”, ITER offers a unique opportunity to test blanket concepts in a full fusion environment before their application in a DEMO reactor. For this scope, two equatorial ports of ITER will be used to host blanket test objects (Giancarli et al., 2020). A Test Blanket Module has the dimension of about 1 m³ with a surface exposed to the plasma of about 1 m². Two of these objects can be allocated inside an ITER equatorial port and provided with independent coolant and tritium carrier loops; they will be fixed to the port plug structures that can be inserted into the equatorial port using the ITER standard transporter system. The port plug acts as primary containment insulating the TBMs from the rest of the device; only coolant, tritium carrier pipes and diagnostic lines are routed through the port plug to reach other locations in the ITER buildings that host, e.g., coolant systems, tritium extraction systems and diagnostic cubicles.

Each TBM reproduces the blanket architecture described in Fig. 2 and will use materials that are selected for a DEMO concept (e.g., use of RAFM steels as structural materials); the TBM consists of an external box that includes a BZ connected to specific coolant and tritium carrier loops. The BZ, in particular, is entirely DEMO-relevant, allowing experimental studies of structures, coolant, breeder/multiplier and tritium carriers under the same operational conditions (in terms of temperatures, pressures, flow velocities) as the DEMO blanket concept that is the subject of the test. In this respect, the fusion reactor conditions provided by ITER for neutrons and magnetic field intensity are not so different from those described for the DEMO (e.g., NWL of 0.78 MW m⁻² in comparison to ~1 MW m⁻² in the EU DEMO). Of course, the neutron fluences will be very limited in the ITER test, not more than 1 dpa in the FW structures for each irradiated TBM; this will allow a view of the behavior of the system in a “start of life” phase, but no information can be obtained for components at DEMO-relevant levels of integrated irradiation. The length of the pulses will typically be several minutes long, much shorter than the ~2 h foreseen for, e.g., the EU DEMO (though, in later phases of ITER operation, significantly longer pulses might be accessible as the ITER research program explores fully non-inductive long-pulse operation); this duration will be sufficient barely to reach temperature asymptotic conditions in all the blanket subcomponents and materials of interest.

A particularity of the TBM location in the port plug is that its FW is recessed by about 3 cm in comparison to the ideal FW surface dictated by the arrangement of the ITER shielding blanket. This implies that the exposure conditions of the TBM FW will involve strong “shadowing” by the other ITER FW systems and by the port plug. The TBM FW will be impacted only by (somewhat reduced) direct radiation from the plasma surface and will be almost completely shielded from the tangential plasma ion flux. How much the test can reproduce DEMO conditions (also if the “limiter protection” concept is adopted) it is an open point. In addition, the manifold and support elements of the TBM box are designed mostly to cope with the ITER requirements for interfacing with the port plug and to allow an easier access of diagnostic lines to monitor the TBM operational conditions.

A specific discussion is necessary for the use of RAFM in the TBM (Zmitko et al., 2017). Even if not strictly necessary for withstanding ITER irradiation conditions (as already mentioned they will not exceed several dpa in the entire lifetime), the test of these materials is a key aspect in the testing of the blanket system. For this, the TBM requires the development of manufacturing and joint technologies for this class of materials applied to a component design that follows the basic architecture of a blanket that will be adopted for a power plant.

The TBM auxiliary systems are designed to feed the TBM with coolant and tritium carrier at conditions comparable with DEMO in order to allow the full test of the blanket BZ. The technologies used in the cooling systems are not necessarily DEMO relevant: preference is given to standard proven technologies, robust, but perhaps not highly efficient for a large power plant. For example, helium-cooled TBMs (with a helium temperature of up to ~ 500 °C) use cooling loops with an 8-loop configuration with an intermediate recuperator that is typical for research loops where the helium blowers operate at low temperature (~ 50 °C), to reduce cost and simplify the operation mode. Tritium extraction technologies selected for the TBM (Ricapito et al., 2018) are essentially dictated by ITER (to limit the impact on the facility and to avoid jeopardizing the safety concept) and by the testing requirements (e.g., to perform accurate measurements of tritium species involved in the process). Once again, performance requirements are not of the highest priority, although several technologies are also incorporated to be tested in view of corresponding DEMO systems (e.g., GLC for tritium extraction from PbLi, or non-cryogenic systems for the tritium extraction of HT from the helium purge flow).

The role of TBM testing in the general DEMO development time schedule is a key consideration for the implementation of the ITER program. The DEMO programs of Europe, China and Japan have been proposed with the ambitious goal of starting operation in the middle of this century; the results of ITER (in terms of both device performance and TBM testing) should therefore be available on a timescale supporting this time schedule, as they are intended to guide and confirm the DEMO program. To what extent the development of the various DEMO sub-systems can be conducted separately from ITER, especially as concerns the development of those technologies that are not foreseen in ITER operation and testing (e.g., the fuel cycle architecture, with an “outer cycle” as discussed in Section “The fuel cycle”) remains under review. Even the present phase of the ITER TBM Program plays an important supporting role for the DEMO BB program via a constant Return of Experiences (RoX) in strategic topics such as the development of fabrication technologies (structural and functional materials), modeling tools, and safety and licensing concepts. In the 2030s, the TBM tests in ITER will allow the completion of out-of-pile integral testing with in-pile experiments where unique aspects of the fusion environment can be fully explored (e.g., neutron and magnetic field). A more detailed discussion of the scope of the TBM testing program and its benefits for the BB program can be found in (Poitevin et al., 2007).

Blanket for tokamak and stellarator

In the previous Sections the blanket has been described from the perspective of its application to a tokamak type MFC reactor, as is currently foreseen for DEMO, but this component is necessary also for the other possible MFC concepts, like the stellarator. The stationary operational mode of this reactor and the absence of major disruptions will simplify some loading conditions, but not in such a way that the general operational conditions will be so different from those in a tokamak; hence, the design principles of stellarator blankets will not differ much from what has been developed for the tokamak. However, the reactor integration will be more demanding, involving a fully 3D geometry in which the inherent axisymmetry of the tokamak configuration is absent. In addition, a blanket maintenance concept for stellarators has not yet been developed and faces the challenge of combining a complex geometry with intrinsically reduced space between magnet and plasma, and very limited space for large ports between the magnets for accessing the blanket position in the vacuum vessel. Only the absence of the internal solenoid and the large major radius—about 20 m in HELIAS (Warmer et al., 2016)—can, in principle, allow direct access to the VV also from the inboard side with ports. Overall, the stellarator blanket studies can be considered to be about 20 years behind those of tokamak reactors. Even if substantial elements of the blanket technology can be derived from the tokamak studies, further constraints due to the fully 3D geometry have not been studied in any depth and their consequences for the blanket technology are still not well understood.

Blanket for inertial fusion applications

As far as the breeding blanket for ICF reactors is concerned, very few specific power plant studies can be found in the literature; one such study was conducted in the framework of the ARIES-IFE (2004) to which we refer in this section. As in MCF reactors, it is assumed that a blanket with high coverage surrounds the ICF reaction chamber, with a thick BZ protected by a FW directly exposed to the D-T reactions. The major difference compared to MCF blankets is in the loading generated by the fusion process. In an ICF reactor, the driver energy (laser or heavy ion beam) induces a thermonuclear reaction via the implosion of single D-T target; a pulse

of fusion energy (from some tens to some hundreds of MJ) is released in a single target implosion on the nanosecond time scale; the repetition frequency of the implosions is in the range of 5–15 Hz. Depending on the target design and materials, several reactions can occur which determine the final composition of the particles and radiation emerging from the target: in the typical targets considered in the ARIES-IFE, $\sim 70\%$ of the energy is transported by neutrons that will reach the FW of the reactor chamber, penetrating it and depositing their energy into a large blanket BZ. Apart from a slight difference in the neutron spectra caused by the target interaction, the principles of thermal deposition (and nuclear reactions) in the blanket volume are similar to those already discussed in relation to the MCF blankets and can be accommodated by standard BZ technology. The major difference relating to blanket design is the absence of magnetic fields (and of the external structures associated with the magnets and their supports), which can enable an efficient design based on liquid metal cooling and facilitate the accessibility of blanket component from outside the reaction chamber; these features could simplify the design, even if the complex arrangement of laser or heavy ion beams imposes other constraints associated with the conflicting requirements of providing simultaneously external penetration for the beams and confinement and shielding of the internal reaction products.

A major difference with magnetic fusion reactors relates to the transport of the remaining 30% of the emitted energy; this will be divided between the kinetic energy of ions and soft X-rays (< 1 keV); both energy contributions will be absorbed in the chamber FW as surface heat flux, almost entirely within the first 100–200 μm of FW material. The short deposition time (i.e., nanoseconds) of this energy presents a key challenge, since the energy pulse can produce ablation (e.g., evaporation) in a solid surface; at a deeper level within the FW, the heat flux tends to become uniform in time due to the thermal capacity of the wall materials, converging to absolute values comparable with those in MCF devices. In ICF reactors, the energy emitted in the form of X-rays can reach $\sim 25\%$ of the total fusion energy released in the case of indirect-drive targets (using high-Z materials in the target hohlraum to increase the yield of the reaction); in direct-drive targets the ions tend to retain their energy so that only a small fraction ($\sim 1.5\%$) of the energy is transmitted as X-rays. As all photons transport energy to the first wall on the same nanosecond-timescale, the resulting power deposition can be very severe; in contrast, ion energy (of fusion reaction products and target debris) is emitted with a wide range of energies and speeds, so that the deposition time scale is expanded to the microsecond-scale, reducing the first wall ablation per shot. In addition, the chamber environment (i.e., presence of residual products of previous reactions or buffer gases) can play an important role in modifying the load; i.e., the controlled use of a sufficiently dense high-Z gas, xenon as described in (Dunne et al., 2011), can efficiently protect the FW from X-rays via reactions that can disperse the energy pulse over a period of several microseconds. To what extent the use of buffer gases is compatible with the propagation and focusing of driver beams, as well as the survivability of the targets, must be further investigated.

This explains why ICF blanket studies have focused on FW heat removal and protection (“chamber concepts”). The issue of FW protection has emerged as a driving requirement in the design of this component and in the material selection (Alvarez et al., 2011). In particular, the use of indirect-drive targets (high X-ray content in the resultant heat flux) brings the technology of solid walls (“dry-wall”) to its limits (Najmabadi et al., 2004). For this reason, alternative solutions have been proposed adopting, e.g., the “wetted-wall” technology, where the supporting FW structure is protected by an external liquid film that acts as a renewable armor; if the liquid is selected from among potential liquid breeders (liquid metals or molten salts) the neutronics aspects of the design can be optimized, avoiding any extraneous materials. Here, two different concepts can be adopted to create a protective liquid film: (a) a porous FW structure that allows the injection into the reaction chamber of the liquid to compensate the evaporation and maintain a thin protective film, or (b) a forced flow of a liquid film tangential to a solid wall that supports and drives it. In case (b) the absence of magnetic fields allows a better use of liquid metals, avoiding large MHD flow instabilities, a key issue for MCF applications. But this is the only advantage: in general, the use of a “wetted-wall” technology encounters many open questions on the film dynamics, tritium compatibility, compatibility with and recovery of evaporating liquid in the reaction chamber, etc.

An integrated ICF power plant design with breeding blankets can already be found in the literature more than 25 years ago, the HYLIFE-II baseline (Moir, 1994). The paper proposes the use of a “waterfall” of FLiBe for wall protection and heat removal; the FLiBe flow is then passed into a heat exchanger where it heats water for use in the turbines. FLiBe acts also as tritium breeder and neutron multiplier: the tritium produced in the reaction with lithium is also extracted in the external loop in order to close the power plant’s thermonuclear fuel cycle.

An example of a “dry-wall” chamber is presented in (Garoz, 2016): in this paper the behavior of a tungsten first wall is studied under the irradiation conditions predicted for HiPER (High Power Laser Energy Research Facility) with direct drive targets.

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Magnetic Confinement Fusion—Development Facilities

A.J.H. Donné^{a,b}, G. Federici^a, A. Ibarra^c, J. Menard^d, and F. Warmer^e, ^a EUROfusion, Garching, Germany; ^b Dutch Institute for Fundamental Energy Research, Eindhoven, The Netherlands; ^c CIEMAT, Madrid, Spain; ^d Princeton Plasma Physics Laboratory, Princeton, NJ, United States; and ^e Max-Planck Institute for Plasma Physics, Greifswald, Germany

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Glossary

Beta normalized Operational parameter indicating how close a tokamak plasma is to reaching the empirical pressure limit, involving, in some cases, destabilization of a major magnetohydrodynamic (MHD) instability.

Bootstrap current A self-generated current due to collisions between trapped and passing plasma particles. The bootstrap fraction is strongly influenced by the density gradient.

DCLL Dual-Coolant Lithium Lead—one of the possible concepts for the tritium breeding blanket.

Disruption In a disruption the plasma confinement is lost in milliseconds. Often this occurs after a vertical displacement event of the plasma column or after the growth of non-axisymmetric magnetohydrodynamic instabilities.

ECRH Electron Cyclotron Resonant Heating.

Edge Localized Mode (ELM) Magnetohydrodynamic instability in the plasma edge region associated with a quasi-periodic relaxation of the edge transport barrier formed during the transition from low (L) to high (H) confinement mode.

Greenwald scaling Empirical upper limit for the central line-averaged density in a tokamak plasma which depends on plasma current and minor radius.

HCCB Helium-Cooled Ceramic Breeder—one of the possible concepts for the tritium breeding blanket.

HCLL Helium-Cooled Lithium Lead—one of the possible concepts for the tritium breeding blanket.

HCPB Helium-Cooled Pebble Bed—one of the possible concepts for the tritium breeding blanket.

HTSC High-temperature superconductor.

ICRF Ion cyclotron radiofrequency heating.

IFMIF/DONES International Fusion Materials Irradiation Facility/DEMO Oriented Neutron Source.

IFMIF/EVEDA International Fusion Materials Irradiation Facility/Engineering Validation and Engineering Design Activities.

LHH Lower hybrid heating.

LTSC Low-temperature superconductor.

Martin scaling Empirical scaling for the input power to the plasma required to access the H-mode.

NBH Neutral Beam Heating.

Snowflake divertor A divertor geometry that effectively spreads the heat load from the plasma over four instead of over two footprints, thereby effectively lowering the heat load on the divertor materials.

WCLL Water-Cooled Lithium Lead—one of the possible concepts for the tritium breeding blanket.

Introduction—The path towards fusion electricity

A number of countries have developed fusion roadmaps or strategic plans, in order to clarify which steps need to be taken to achieve electricity generation from fusion (Donné et al., 2018; Okano, 2019; Wan et al., 2017; National Academy of Sciences, 2019). Although there are differences between the various fusion roadmaps, there are also many commonalities, which are depicted in Fig. 1. All roadmaps include ITER (Bigot, 2019) as an essential step. ITER will demonstrate that energy from fusion is feasible by aiming at a plasma power multiplication ($Q = P_{\text{fusion}}/P_{\text{in}}$) of 10 and it will demonstrate essential fusion technologies, such as tritium breeding.

A second important facility included in basically all roadmaps is the demonstration reactor that will for the first time deliver electricity to the grid. In Europe and Japan this device is named DEMO (Federici et al., 2018; Tobita et al., 2019), in Korea K-DEMO (Kim et al., 2015), and in China it is known as the China Fusion Experimental Test Reactor (CFETR; Zhuang et al., 2019). These devices are all based on relatively conservative extrapolations from ITER following the rationale that obtaining a nuclear license for a new device is facilitated by exploiting technologies and components that have already been demonstrated elsewhere. Reactors following DEMO and CFETR are foreseen as first-of-a-kind fusion power plants (FPPs).

Some countries, however, prefer to pursue smaller, more flexible, facilities based on spherical tokamaks to follow ITER (Menard et al., 2016), such as the Fusion Nuclear Science Facility (FNSF; Kessel et al., 2018) in the USA and the Spherical Tokamak for Energy Production (STEP; Wilson, 2020) in the UK. Most likely these facilities must be followed by a real demonstration plant, before fusion can be commercialized. Several privately funded enterprises (Sykes et al., 2018; Sorbom et al., 2015) are also following this approach, with the special feature that high-temperature superconducting magnets are proposed to make it possible to build rather small fusion reactors, featuring a high power density. In a certain sense one could say that these approaches are depicted by the green line in Fig. 1.

Since the neutron fluence in the devices following ITER (e.g., DEMO and CFETR) will be much higher than those in ITER, new materials need to be developed and validated. At present, this is largely pursued in Materials Test Reactors (basically fission reactors), but for testing the plasma facing components an intense neutron source with a dedicated fusion neutron spectrum is needed. The International Fusion Materials Irradiation Facility (IFMIF) is such a facility, and prototype components are presently being tested in a collaboration between Europe and Japan (Knaster et al., 2017). To accelerate the path to the realization of such a neutron source, Europe and Japan are now each proposing smaller neutron sources, IFMIF/DONES in Europe (Ibarra et al., 2018), and the Advanced Fusion Neutron Source in Japan (Ochiai et al., 2020). Spherical tokamak based devices have also been proposed as volumetric neutron sources, denoted ‘Component Test Facility’.

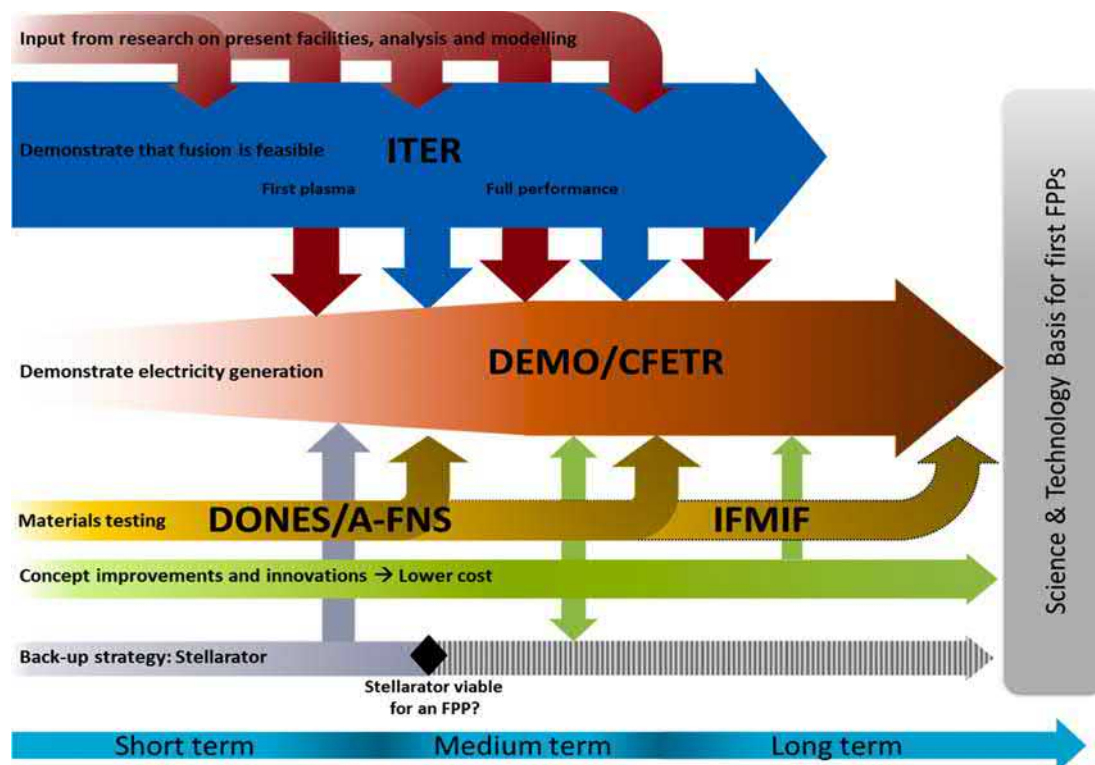


Fig. 1 Schematic view of the main elements of a fusion roadmap that is common to several countries. The stellarator line is mainly in the European and Japanese roadmaps.

As indicated by the red arrows in Fig. 1, many operational facilities within the international fusion research program are providing input to ITER and to DEMO. Tokamaks predominate, but devices such as stellarators also give important input to the main fusion roadmap. All devices limit operation to hydrogen and deuterium plasmas to minimize nuclear activation, facilitating upgrades and the continuous evolution of experimental capabilities. The Joint European Torus (Joffrin et al., 2019), nevertheless, retains a capability for experiments with deuterium-tritium plasmas. As discussed later, several significant new non-nuclear facilities are currently in the development phase: JT-60SA, a Japan-EU collaboration (Barabaschi et al., 2019), the Divertor Test Tokamak (DTT) in Italy (Albanese et al., 2019), HL-2M (Song et al., 2019) in China and COMPASS-Upgrade (PANEK et al., 2017) in the Czech Republic.

This chapter is organized as follows. Section “ITER” briefly describes the key objectives, design features, major component systems and overall research framework of the ITER project. Section “Demonstration reactors” summarizes the major considerations influencing the design of demonstration reactors based on conservative extrapolations from ITER (the DEMO reactors by Europe, Korea and Japan and the Chinese CFETR), discusses a range of concepts developed within the framework of the ‘Fusion Nuclear Science Facility’ and ‘Pilot Plant’ studies, the majority based on the spherical tokamak configuration and several seeking to exploit recent developments in high temperature superconductors, and finally comments on key prospects for stellarator-based DEMO reactors. Conceptual designs for facilities intended to provide a nuclear irradiation capability for fusion materials and components are described in section “Fusion materials test facilities”, while several of the non-nuclear facilities under development are briefly discussed in section “Non-nuclear development facilities”.

ITER

Introduction

ITER is the next step fusion device, and at this moment the largest international scientific project world-wide (ITER, 2002; Bigot, 2019). It is a collaboration between China, the European Union (including Switzerland), India, Japan, South-Korea, the Russian Federation and the United States. The initial ITER collaboration was launched in November 1985 at the Geneva summit meeting between Secretary-General Gorbachev of the Soviet Union and President Reagan of the United States. After two decades of design studies, R&D and international negotiations, the ITER Agreement among the seven members was signed in November 2006 and the ITER Organization was officially established in October 2007 with its headquarters at the ITER construction site in St Paul-lez-Durance, southern France. The history of ITER is described in Ikeda (2010) and Claessens (2020).

As cited in the ITER Research Plan (ITER Organization, 2018), the ITER scientific goals, defined in terms of plasma fusion performance, are:

- ITER should achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power, Q , of at least 10 ($Q \geq 10$) for a range of operating scenarios and with a duration sufficient to achieve stationary conditions on the timescales characteristic of plasma processes.
- ITER should aim at demonstrating steady-state operation using non-inductive current drive with the ratio of fusion power to input power for current drive of at least 5.
- In addition, the possibility of controlled ignition should not be precluded.

The technical goals, aiming to provide much of the technological basis for the design of future fusion power plants capable of generating electricity, are:

- ITER should demonstrate the availability and integration of technologies essential for a fusion reactor (such as superconducting magnets and remote maintenance).
- ITER should test components for a future reactor (such as systems to exhaust power and particles from the plasma).
- ITER should test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high-grade heat and electricity production.

Demonstration of the safety and environmental advantages of fusion energy is a further significant aspect of the ITER project mission. In this respect, it should be noted that ITER has undergone the full licensing procedure appropriate to a ‘Basic Nuclear Installation’ (INB) within the French nuclear regulatory system (see, e.g., Taylor et al., 2013). The French Government granted the Decree of Authorization of a nuclear facility in November 2012 and ITER is now established as INB-174 within the French regulatory framework.

A cutaway view of the ITER tokamak is given in Fig. 2 and the main parameters of the ITER-plasma are listed in Table 1. Extensive details on the ITER design and its technical systems can be found in papers published during the project’s design and R&D phase, e.g., (ITER, 2002; Aymar et al., 2002; Shimomura, 2001), while more recent developments in ITER technology are discussed by Campbell et al. (2019).

Tokamak core systems

As illustrated in Fig. 2, the principal tokamak core components are the vacuum vessel, plasma-facing components (divertor and shielding blanket/first wall), in-vessel control coils, superconducting coil systems (toroidal field (TF), poloidal field (PF), central solenoid (CS) and correction (CC) coils), thermal shield and cryostat.

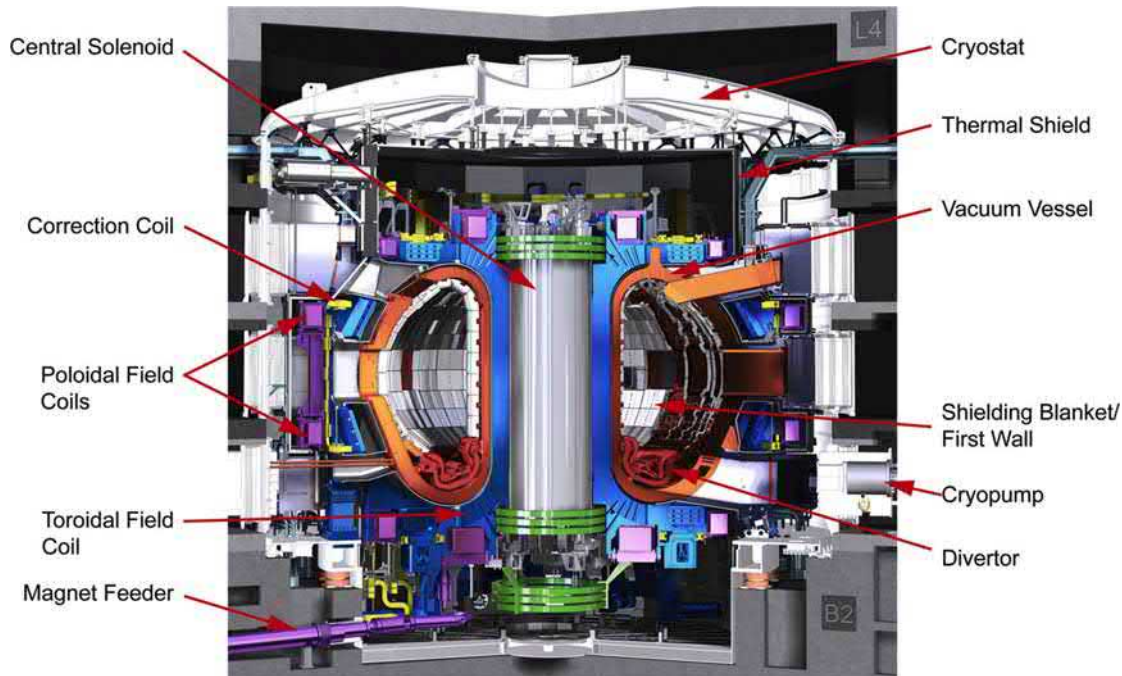


Fig. 2 Schematic view of the ITER tokamak with the main components indicated. Copied with permission from Campbell DJ, Akiyama T, Barnsley R, et al. (2019) Innovations in Technology and Science R&D for ITER. *Journal of Fusion Energy* 38: 11–71. <https://doi.org/10.1007/s10894-018-0187-9> (web archive link,).

Table 1 Main parameters of ITER.

Parameter	Value
Major radius, R_0 (m)	6.2
Minor radius, a (m)	2.0
Toroidal field on axis, B_T (T)	5.3
Plasma current, I_p (MA)	15
Plasma elongation, κ	1.85
Plasma triangularity, δ	0.49
Edge safety factor, q_{95}	3.0
Neutral beam power, P_{NB} (MW)	33
Radiofrequency power, P_{RF} (MW)	40
Fusion power, P_{fusion} (MW)	500
Fusion gain, Q	10
Burn time (s)	300

The ITER vacuum vessel (Martinez et al., 2016), consisting of the main vessel, port structures and supporting system, provides a hermetically sealed plasma container which also fulfils the critical role of first tritium containment barrier. It is a water-cooled austenitic stainless steel (SS 316L(N)) pressure vessel with an outer diameter of 19.4 m, a height of 11.4 m and an inner volume of 1400 m³. The main vessel is a double-walled steel structure strengthened by interspace stiffening ribs, with the space between the two 60 mm walls filled by ‘in-wall’ shielding in the form of modular blocks of borated and ferromagnetic steel. These blocks contribute additional neutron shielding for components external to the vacuum vessel (e.g., the superconducting magnets), while also reducing the toroidal field modulation (‘ripple’) caused by the finite number (18) of toroidal field coils. 44 ports in the vessel (18 upper, 17 equatorial and 9 lower ports) allow access for diagnostic systems, neutral beam injection, radio-frequency heating, fueling and vacuum pumping, as well as providing access for remote handling systems used for maintenance and repair. The total weight of the vacuum vessel, including ports and in-wall shielding is approximately 5200 tons.

The inner walls of the vacuum vessel, with an area of ~ 600 m², are covered by 440 actively-cooled shielding blanket modules that protect the vacuum vessel and the superconducting magnets from the heat and neutron fluxes produced by fusion reactions. Each water-cooled blanket module, fabricated from 316L(N) stainless steel, has a plasma facing surface area of ~ 1.5 m², is ~ 45 cm thick radially and weighs up to 4.5 t. The blanket cooling water system operates at 4 MPa, with an inlet temperature of 70 °C, and is capable of removing 750 MW of thermal power. A detachable first wall, fabricated from 8 to 10 mm thick beryllium, is mounted on

the module plasma facing surface to intercept heat fluxes arriving in the form of plasma particles and electromagnetic radiation. Beryllium has been selected as the first wall material due to its low atomic number, which limits plasma contamination, and low affinity for hydrogenic isotopes, which limits fuel retention.

The divertor extracts heat, helium ash produced by the fusion reactions, unburned fuel and impurities in such a way that plasma contamination is minimized. It is formed from 54 stainless steel cassettes, each weighing ~ 8 t, arranged to form a toroidal ring. Plasma facing surfaces ('divertor targets') are mounted on the inboard and outboard regions of the cassette designed to withstand steady state heat fluxes of 10 MW m^{-2} and slow transients of up to 20 MW m^{-2} , while the 'dome' region lying between the inner and outer targets can handle heat fluxes of up to 5 MW m^{-2} . Solid tungsten, ~ 8 mm thick, forms the plasma facing surfaces, due to its high melting point, low sputtering yield under plasma bombardment and low fuel retention. High pressure water cooling is used for heat exhaust, as for the shielding blanket.

The superconducting magnet systems consist of 18 TF coils and 6 CS modules wound from Nb_3Sn superconductor plus 6 PF and 18 CC coils fabricated from NbTi superconductor. All coils are cooled with supercritical helium at temperatures in the range 4.2–4.5 K. The D-shaped TF coils are 17 m high and 9 m wide, and each weighs ~ 330 t. They are wound in 'double pancakes'—layers of spiraled conductor embedded in radial plates and each TF winding is encased in stainless steel. The maximum magnetic field on the coil is 11.8 T and, when fully energized, the total magnetic energy of the TF is 41 GJ. The ring-shaped PF coils, the largest having a diameter of 24 m, are mounted outside of the TF coils and operate at a maximum magnetic field of 6 T at the coil. The CS coil, with an overall height of 13 m and weighing ~ 1000 t, consists of six independent modules, each 4 m in diameter. With a maximum magnetic field of 13 T in the center of the coil stack and a stored magnetic energy of 6.4 GJ, the CS can sustain a 15 MA plasma current for 300–500 s during a high power fusion burn. The CC are much lighter, as they simply compensate for residual magnetic field errors, but the largest of these nevertheless have an area of $8 \times 8 \text{ m}^2$. An overview of the ITER magnet system is provided by Mitchell and Devred (2017), while Devred et al. (2014) describe the challenges in the production of the superconductor.

The height and diameter of the ITER cryostat (Sekachev et al., 2015; Bhardwaj et al., 2016) are both ~ 30 m, while the structure, including the thermal shields, weighs ~ 4000 t. With an internal volume of $16,000 \text{ m}^3$, this stainless steel pressure vessel provides the high vacuum ($\sim 10^{-4}$ Pa), cryogenic environment for the ITER tokamak core components, in particular the superconducting magnets. The cryostat has over 200 penetrations to allow for maintenance access and replacement of blanket modules and divertor cassettes, and to provide access for heating systems, diagnostics, cooling systems, and magnet feeders. It is connected to the vacuum vessel via large bellows that allow for thermal contraction and expansion during ITER operation.

Auxiliary and plant systems

To achieve the initial conditions for achieving high fusion gain, three auxiliary heating and current drive (H&CD) systems provide a total baseline heating power of 73 MW (Singh, 2016):

- Two Neutral Beam Injectors (NBI) (Hemsworth et al., 2017) will provide a total input power of 33 MW, with provision for a third injector to allow a total beam input power of 50 MW. The injectors are based on negative ion technology (Inoue et al., 2021) operating at 1 MeV injection energy, allowing central deposition of the beam power. Near-tangential injection into the plasma allows torque input and current drive.
- The Ion Cyclotron Radiofrequency Heating (ICRH) system (Beaumont et al., 2017) injects 20 MW of power with a frequency in the range 40–55 MHz from two antennas located in the equatorial port plugs.
- The Electron Cyclotron Resonant Heating (ECRH) system (Darbos et al., 2016) will deliver 20 MW of power to the plasma at a frequency of 170 GHz. The gyrotron power sources are each designed to deliver 1 MW for periods of at least 1000 s. The ECRH power is injected into the plasma via launchers with steerable mirrors located in both the upper and equatorial ports of ITER.

A nominal input power level of 50 MW is foreseen for sustainment of the reference burning plasma scenario, which is predicted to generate 500 MW of DT fusion power:

ITER will have approximately 55 different diagnostic measurement systems to measure a large range of plasma and machine parameters (Donné et al., 2007; Walsh et al., 2015). The diagnostics are classified into three groups: (i) diagnostics for basic operation and machine protection, (ii) diagnostics for advanced control and (iii) diagnostics for physics evaluation and performance optimization. A large range of different diagnostic techniques is used (Inoue et al., 2021), for example, magnetic measurements, neutron emission measurements, optical systems, radiation bolometers, spectroscopy across the electromagnetic spectrum, microwave emission, transmission and reflection, and plasma-facing and operational diagnostics. For some diagnostics, the detectors must be mounted inside the vacuum vessel and great emphasis is given to ensuring their radiation hardness. Most in-vessel detectors are located deep in slots between adjacent blanket modules, to provide protection against the hostile plasma, while some diagnostics for plasma edge measurements are mounted on the divertor cassettes. Many other diagnostics have optical or microwave interfaces incorporated into dedicated vacuum vessel port plugs with complex folded optical or microwave transmission lines to reduce neutron streaming through the port plugs. Detectors and active sources, such as lasers and microwave generators, are then located in the diagnostics building, far from the plasma. The hostile environment within the proximity of the plasma necessitates the development and application of in-situ techniques to counteract the effects of plasma bombardment of the first, plasma facing, mirrors in many optical systems, ensuring high reflectivity of the mirror surface.

A sophisticated plasma control system (PCS) combines measurements made by these diagnostic systems to sustain fusion power production via real-time control of the plasma parameters implemented via various ‘actuator’ systems, i.e., magnet systems, H&CD systems, fueling and impurity injection systems (Snipes et al., 2017). At high plasma performance, the control of magnetohydrodynamic (MHD) instabilities (Lao et al., 2021) is particularly critical in ITER to maintain the fusion burn and to avoid potential damage to the first wall. For example, edge localized modes (ELMs), which expel short bursts of heat and particles, are controlled by a set of 27 in-vessel copper coils (Vostner et al., 2019). Disruptions, which produce high transient heat and electromechanical loads on in-vessel structures, must be avoided or their effects mitigated (see Lao et al., 2021; Zohm, 2021). A dedicated system, based on fragmentation of cryogenic deuterium or noble gas pellets, and referred to as Shattered Pellet Injection, SPI (Lehnen et al., 2015) will operate under PCS control to implement disruption mitigation. The PCS can also control the injection of localized H&CD to suppress MHD instabilities responsible for triggering disruptions.

Demonstration of the feasibility of producing tritium within the reactor system is a key mission of ITER, and the project will provide a unique opportunity to test prototypical in-vessel tritium breeding blankets in a real fusion environment (Giancarli et al., 2018). Several concepts for tritium breeding and high grade heat extraction will be tested in ITER using two dedicated port plugs, each allowing two distinct ‘test blanket module’ (TBM) designs to be exposed to the fusion neutron flux. Initially, two TBMs are planned to use water-cooling and two helium-cooling. Further details of the ITER TBM program are discussed by Boccaccini et al. (2021).

The ITER plant systems provide key services to the tokamak and its auxiliary systems, and, in many respects, are constructed on a scale commensurate with that required in a fusion power plant. Plant systems include the Fuel Cycle (fueling, vacuum, exhaust gas reprocessing), Remote Handling, Cryopant, Cooling (water cooling systems, heat exchangers, waste heat exhaust), Power Supplies (steady-state and pulsed) and Waste Processing (including detritiation).

As noted above, ITER will test concepts for tritium breeding. However, the significant amount of tritium fuel, ~ 14.5 kg over a period of ~ 20 years, required meet the project lifetime goal of achieving an average first wall neutron fluence of 0.3 MW yr m^{-2} means that tritium must be externally supplied from existing civil fission stockpiles and that fuel reprocessing on an unprecedented scale will be required to ensure efficient consumption of tritium. A large-scale tritium reprocessing plant therefore forms the core of the ITER Fuel Cycle system (Boccaccini et al., 2021): an annual tritium reprocessing capability corresponding to ~ 25 times the onsite tritium inventory will be required. A substantial high vacuum pumping capability is a further critical element. The total volume of the ITER vacuum system (Day et al., 2008) exceeds $10,000 \text{ m}^3$ (cryostat: 8500 m^3 , vacuum vessel: 1330 m^3 , neutral beam injectors: 180 m^3 each). The central part of the roughing system consists of newly developed cryogenic viscous flow compressors in combination with Roots mechanical pumps, while the facility also features a set of six cryopumps for the main vacuum vessel, four for the neutral beam system and two for the cryostat. These pumps, cooled by supercritical helium, are coated with activated charcoal as sorbent material, in particular for helium pumping. Finally, fueling of DT plasmas by gas injection and cryogenic pellet injection closes the fuel cycle. Further considerations related to the design of reactor-scale fuel cycle systems can be found in Boccaccini et al. (2021).

ITER is equipped with one of the world’s largest cryopants (Monneret et al., 2017)—only the distributed system for the Large Hadron Collider has a greater cooling capacity. It consists of two large nitrogen refrigerators and three large helium refrigerators, which provide the necessary cooling for the superconducting magnets. It also supplies the six cryopumps that operate at a temperature of 4 K, the magnet current leads that operate at 50 K and the thermal shields within the cryostat that are cooled to 80 K. The installed cooling power is 75 kW at 4.5 K (helium) and 1300 kW at 80 K. The ITER tokamak and the auxiliary systems will produce on average 500 MW of heat during a typical plasma pulse cycle, with a peak of 1.1 GW during the plasma burn phase, and this needs to be dissipated to the environment. Heat generated by the plasma is transferred via the Tokamak Cooling Water System, TCWS (Lioce et al., 2019), to the intermediate Component Cooling Water System (CCWS; Kumar et al., 2013), which is a closed loop system that transfers the heat to the Heat Rejection System (HRS). Water from the HRS is ultimately discharged into the nearby canal after monitoring to ensure that it fulfils the stringent environmental release criteria.

Robust remote handling techniques are being developed to support in-vessel maintenance and repairs (Damiani et al., 2018; Federici et al., 2021). Special remote handling tools will be able to replace heavy components such as the blanket modules, divertor cassettes and the port plugs, with weights of up to 50 tons. A substantial Hot Cell Facility, HCF (Friconneau et al., 2017), will permit maintenance and processing of irradiated and tritiated components that have been removed from the tokamak. Transportation between the tokamak and the HCF will be carried out using a sealed transport cask. Radioactive waste will be produced during ITER operation and decommissioning (Rosanvallon et al., 2016) due to the replacement of components, as well as from process and housekeeping waste. The waste consists of components activated by neutrons with energies of up to 14 MeV, and contaminated by activated corrosion products, activated dust (including beryllium and tungsten dust) and tritium. To satisfy French regulatory requirements, facilities are included in the ITER Hot Cell complex for the treatment and interim storage of the waste. This complex, equipped with an extensive remote handling capability, will provide a secure environment for the processing, repair or refurbishment, testing, detritiation and disposal of activated ITER components.

ITER research plan

The ITER Baseline Schedule foresees a phased transition from facility construction to plasma operation (the ‘Staged Approach’), with the scope of the associated research program, as defined by the ITER Research Plan (IRP) (ITER, 2018), expanding as the facility capabilities are increased over a period of approximately 10 years (Fig. 3). Within the IRP, these first 10 years of ITER operation are

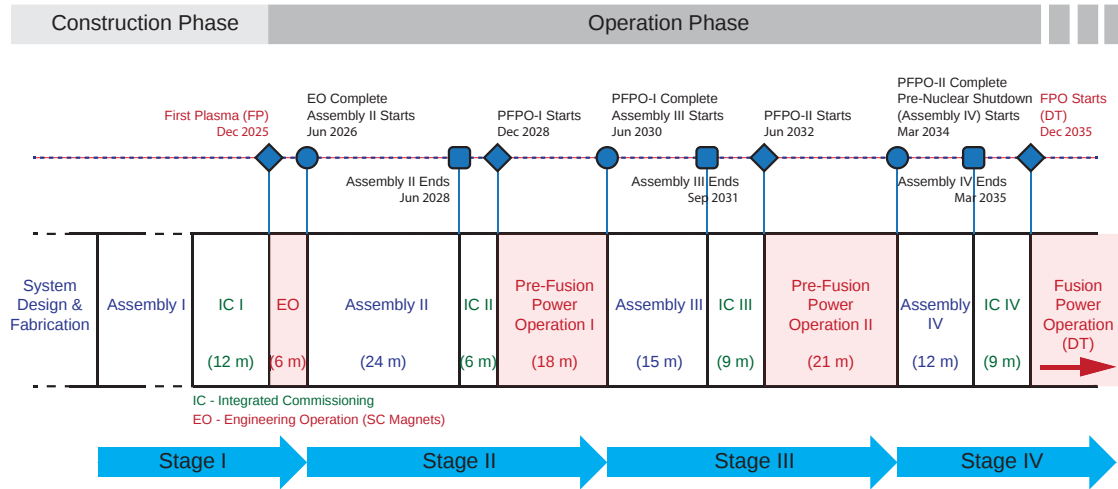


Fig. 3 Schematic time sequence of the first 10 years of ITER operation in the Staged Approach. Adapted with permission from Fig. 5 of Campbell DJ, Akiyama T, Barnsley R, et al. (2019) Innovations in Technology and Science R&D for ITER. Journal of Fusion Energy 38: 11–71. <https://doi.org/10.1007/s10894-018-0187-9> (web archive link,).

sub-divided into three phases: First Plasma (FP), Pre-Fusion Power Operation (PFPO, subdivided into two stages) and Fusion Power Operation (FPO). During the FP and PFPO phases ITER will operate only with hydrogen and helium plasmas, while the FPO phase will focus on optimization of fusion power production in DT plasmas.

During the First Plasma phase, scheduled for late 2025 and targeting plasma currents in the range 100–500 kA, the magnets, the vacuum vessel and the cryostat will be qualified and the most basic auxiliary systems commissioned. These commissioning and experimental activities terminate in an Engineering Commissioning phase to commission the superconducting magnet systems to full current. The PFPO-I and PFPO-II programs are dedicated to a phased commissioning of key hardware systems associated with a gradual increase in plasma parameters and in the sophistication of plasma operating scenarios. Thus, the blanket modules/first wall, divertor, full ECRH system and a range of diagnostic and dedicated control systems will be installed in advance of PFPO-I, allowing routine operation at plasma currents of up to 7.5 MA and toroidal fields in the range 1.8–5.3 T with heating powers of up to 20 MW. A range of plasma operating regimes and plasma control schemes, including disruption avoidance and mitigation, will be commissioned and explored during this program. The full H&CD capability and an extended set of diagnostic systems will be added in advance of PFPO-II. This will support operation at full technical performance (15 MA/5.3 T) with up to 73 MW of heating power and the implementation of an extensive program of plasma and plasma control scheme development, with plasma pulse durations of at least 100 s, in preparation for the transition to DT plasma operation.

Long-range plans foresee final preparations for DT operation being made in 2034–2035, including the final commissioning of the DT fuel cycle and the installation of additional DT-relevant diagnostics. Following authorization by the nuclear authorities, deuterium plasma operation would be launched in early 2035, followed by the transition to full DT operation in the course of 2035. Operating on a two-year cycle of 16 months plasma operation and 8 months maintenance, a series of experimental campaigns would explore optimization of fusion power production in a range of ‘baseline’ plasma scenarios, as detailed in Table 2. Initially, the program is planned to address the achievement of a fusion gain, Q , of at least 10 at fusion powers of up to 500 MW using nominal plasma parameters of 15 MA/5.3 T, and with the burning plasma duration being extended towards the range 300–500 s as operational experience accumulates. Alternative scenarios which current experiments indicate as particularly promising include the ‘hybrid’ scenario, operating in the range 12–14 MA with a significant component of non-inductive current drive. A further key aim of the DT experiments is the achievement of fully non-inductive steady-state plasma operation at lower plasma currents, ~ 9 MA, with $Q \geq 5$ and pulse durations of up to 3000 s.

Table 2 Main parameters of the ITER baseline reference scenarios (Campbell et al., 2019).

Parameter	Inductive	Hybrid	Non-inductive
Plasma current, I_p (MA)	15	13.8	9
Energy confinement time, τ_E (s)	3.4	2.7	3.1
Confinement enhancement $H_{98}(y,2)$	1.0	1.0	1.6
Pressure figure of merit, β_N	2.0	1.9	3.0
Average electron density $\langle n_e \rangle$ (10^{19} m^{-3})	11.3	9.3	6.7
Fusion power, P_{fusion} (MW)	500	400	360
Fusion gain, Q	10	5.4	6
Burn time (s)	300–500	> 1000	~ 3000

While the fully non-inductive steady-state scenario (i.e., with zero flux swing in the CS) is anticipated to provide the operational basis for future fusion reactors, ITER can also explore alternative operating scenarios to those listed in [Table 2](#), whether developed in ongoing experiments in smaller devices, or in ITER's experimental program. The DT operations phase will, in addition, incorporate significant fusion technology studies in support of DEMO and future reactors, the most significant being the demonstration of tritium breeding and high grade heat extraction from the TBMs which will be exposed to the DT neutron flux.

Demonstration reactors

Background

With the ITER project well under way, the nations engaged in magnetic fusion R&D are now determining the necessary science and technology developments required to harness fusion energy. This includes, primarily, progression towards a device capable of demonstrating the production of few hundred MW of net electricity, feasibility of operation with a closed-tritium fuel cycle, maintenance systems capable of achieving adequate plant availability, a high level of safety and low environmental impact. This is known as a DEMOnstration Fusion Power Plant (DEMO).

Europe and Japan envision that ITER should be succeeded by the construction and exploitation of DEMO, the final step before the first commercial power plant. A dedicated divertor test tokamak (DTT), to investigate the problem of power exhaust and demonstrate performance of advance divertor solutions, and a neutron-irradiation facility to qualify materials (IFMIF-DONES) are included in this strategy. Stellarators are also being pursued, the largest devices being LHD (Japan) and Wendelstein 7-X (Europe). China has an ambitious plan to exploit fusion energy for electricity production as quickly as possible to offset the foreseeable large increase in energy demand. The “Chinese Fusion Engineering Test Reactor (CFETR)” facility is currently in the conceptual design development phase with construction planned by the end of the 2020s. South Korea has begun pre-conceptual design work on a DEMO device to develop technology solutions, denoted ‘K-DEMO-1’, with the intention of starting construction in the late-2020s and beginning operation in 2040. After a reconfiguration of the internal components, the second phase (‘K-DEMO-2’) will be launched, leading to a full demonstration of industrial scale electricity generation from a fusion plant. Other communities, e.g., the US and India, view DEMO as the last step before commercialization, but both consider that an intermediate step between ITER and DEMO is required. There is considerable deliberation over the scope of such a device and delay in exploiting fusion energy implied by an additional intermediate step is an issue. Finally, the Russian Federation is considering both ‘pure’ fusion machines and fission–fusion hybrids in their strategy.

The construction, commissioning and operation of ITER are central elements in all fusion development roadmaps. However, while the contribution of ITER to fusion technology is undeniable, large gaps towards DEMO remain and there is an urgent need for continued and vigorous R&D in areas such as fusion grade materials, tritium breeding, plasma exhaust solutions, heating and current drive systems (especially in terms of efficiency), and remote handling. The integration of these technologies to build a reliable device is recognized as being, probably, the greatest challenge ahead. ITER design and construction activities have, nevertheless, yielded several important lessons for those responsible for DEMO design:

- (1) the importance of thermal transients for in-vessel component design;
- (2) strengthening of nuclear design integration and nuclear safety culture during all design phases;
- (3) implications of design changes imposed on ITER by regulatory bodies, derived from increasingly stringent nuclear safety regulations;
- (4) the criticality of a robust maintenance plan with clear provisions to access areas where contamination and activation could be higher than initially considered;
- (5) the need to address more systematically plant layout considerations, including cooling loops and auxiliary systems, and to provide adequate space to integrate all equipment, particularly in the tokamak building.

Demonstration reactors based on extrapolations from ITER

Defining appropriate plant design parameters and technical features starts with the definition of the plant requirements and involves trade-offs between the attractiveness and technical risk associated with the design options. Some of the physics assumptions (e.g., energy confinement, plasma pressure, H-mode access threshold, bootstrap current fraction, etc.), and technology assumptions (e.g., allowable divertor heat loads, structural material neutron-load limits, maximum field in the superconducting magnets, plant thermodynamic efficiency, wall-plug efficiency of Heating and Current Drive systems, etc.) play a major role in the tokamak dimensioning process. Schematic drawings of the current designs of the EU DEMO and CFETR are shown in [Figs. 4 and 5](#), respectively. The considered design parameters of these devices, along with those of JA-DEMO and K-DEMO are shown in [Table 3](#).

A major difference between the EU DEMO and CFETR ([Zhuang et al., 2019](#)) design concepts is that the CFETR design assumes steady-state operation, with a significant current-drive fraction, and hence low-density operation (which is a challenge for the divertor). The EU DEMO design is based on operation in 2-h pulses and relies on a conventional, ideally ELM-free, H-mode scenario with an emphasis on power exhaust issues. This is not yet a fixed design choice, but rather a “proxy” to be used to identify and resolve crucial design integration problems ([Federici et al., 2016; Federici et al., 2018](#)). Considerations are also given to a design capable of

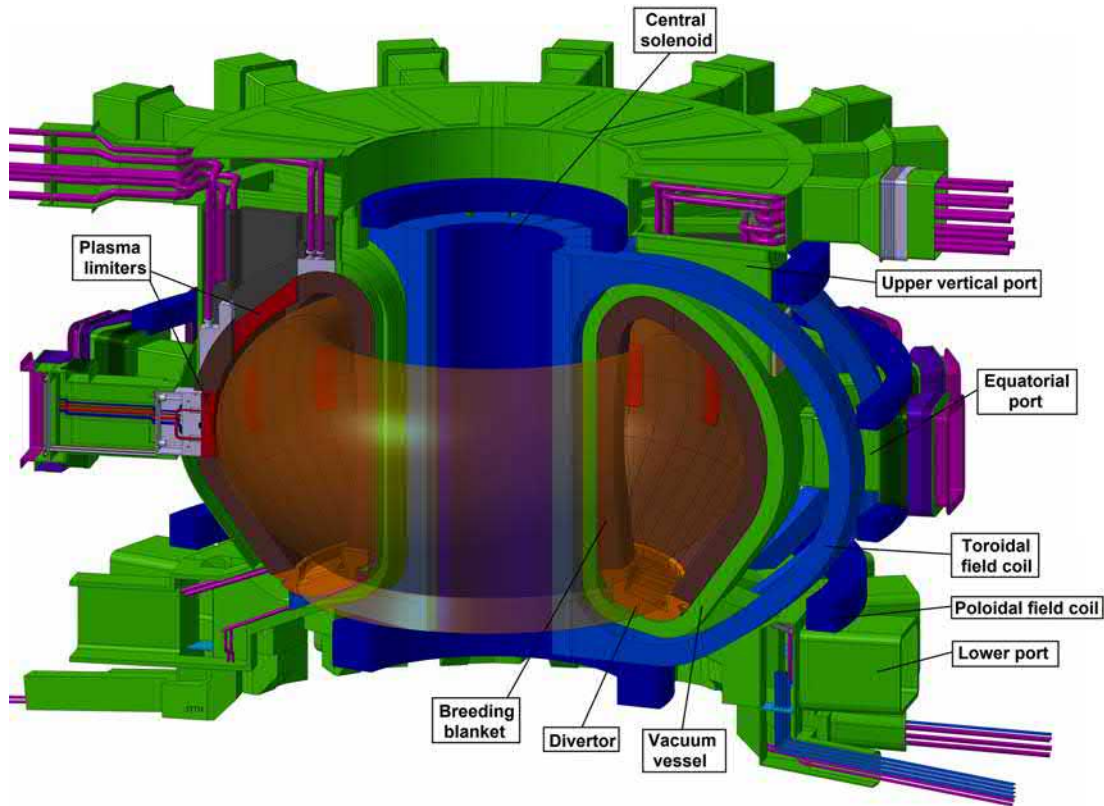


Fig. 4 Schematic drawing of EU DEMO with the various component indicated. © EUROfusion.

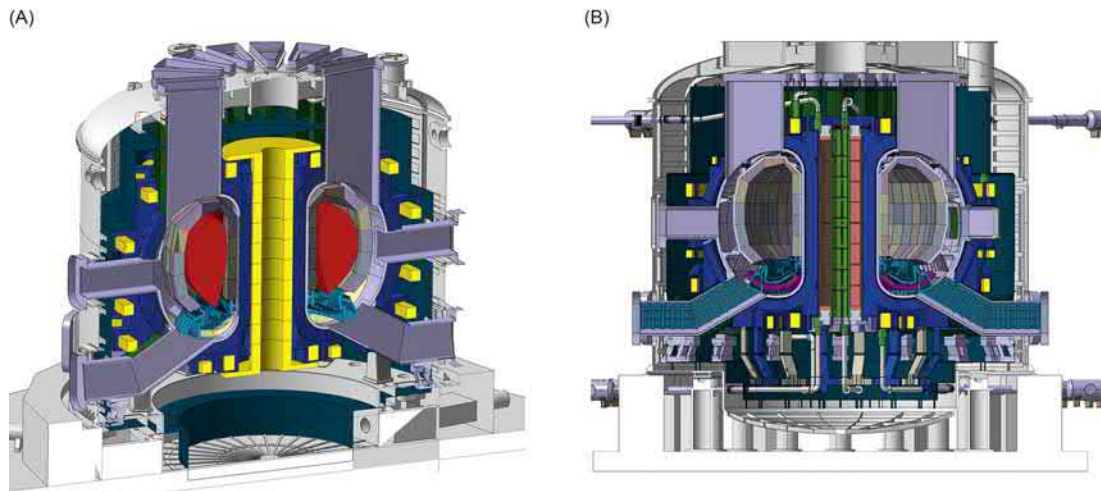


Fig. 5 Schematic drawing of CFETR. Courtesy ASIPP, China.

a short pulse mode operation (e.g., 1 h) for nominal extrapolated performance ($H_{98} = 1.0$) and capable of moving to steady-state operation, while maintaining the same fusion power and net electrical production, when it proves possible to access plasma conditions with better confinement. However, this option requires greater confidence in physics extrapolation and highly reliable and efficient current-drive and control systems, which will require further development. In contrast, a key design consideration for CFETR is the reliance on ‘staged operation’ spanning a wide operating range (see Table 3), for the progressive development of the plasma physics, materials science and technology over the operational lifetime of the plant to achieve performance improvements through system and/or component upgrades.

Although many discussions are ongoing regarding the possibility to achieve fusion power from smaller and cheaper devices to be built on a fast timescale, there is no magic bullet to solve the integrated design problems: squeezing one element of the design

Table 3 Main machine parameters for future devices in EU, Japan, China and South Korea DEMOs.^a

Parameters	Symbol	EU DEMO (pulsed/2 h)	JA DEMO (steady-state)	CFETR (steady-state)	K-DEMO (steady-state) ^b
Major, minor radius	R_p, a (m)	9.0, 2.9	8.5, 2.42	7.2, 2.2	6.8, 2.1
Aspect ratio	A	3.1	3.5	3.3	3.2
Magnetic field on axis	B_T (T)	5.9	5.9	6.5	7.4
Plasma current, edge safety factor	I_p (MA), q_{95}	18, 3.9	12.3, 4.1	12–14/5	>12, <8.0
Elongation, triangularity	κ_{95}, δ_{95}	1.65, 0.33	1.65, 0.33	2, 0.4–0.6	2.0, 0.63
Fusion and auxiliary power	P_{fus} (GW), P_{aux} (MW)	2.0, 50	1.5, 83.4	0.1–2.0/80	2.2–3.0, 80–130
Net electrical power	$P_{el,net}$ (GW)	0.5	>0.25	0–0.8	0.4–0.7
Confinement factor	HH_{95y2}	~1	1.3	0.7–1.6	1.6
Normalized beta	β_N	2.6	3.4	1–3	<4
Bootstrap fraction	f_{BS}	0.39	0.61	0.4–0.6	>0.6
Relative density to Greenwald density	n_e/n_{GW}	1.2	1.2	0.5–1.3	1.25
Average power load on structure	A_{NW} (MW/m ²)	~1	~1	0.1–2.5	1.6–2.0
Structural material		EUROFER for the blanket and AISI 316 for the vacuum vessel	FH82 for the blanket and AISI 316 for the vacuum vessel.	ODS Ferritic Martensitic steel for the blanket	Reduced-Activation Ferritic Martensitic steel for the blanket
Divertor		Single-null water-cooled; W-armor.	Single-null water-cooled; W-armor.	Advanced divertor	Double-null water-cooled W-armor
Breeding blanket (TBM)		WCLL/HCPB	WCPB	WCLL/HCPB	HCSB/WCCB
Toroidal field coils		LTSC magnets Nb ₃ Sn (grading)	LTSC magnets Nb ₃ Sn	NbTi/ITER Nb ₃ Sn/ high Nb ₃ Sn	LTSC magnets Nb ₃ Sn
Lifetime 1st /2nd blanket	(dpa)	20/50	20/TBD	10/20	TBD
Remote maintenance		Blanket vertical RH/divertor cassettes	Blanket vertical RH/divertor cassettes	Blanket vertical RH/divertor cassettes	Blanket and divertor cassettes vertical RH
Balance-of-plant		Yes	Yes	Yes	Yes

^aData of JA DEMO largely extracted from Tobita et al. (2019); data of CFETR largely extracted from Zhuang et al. (2019); data of K-DEMO extracted from Kim et al. (2015), Neilson et al. (2014) and Kang et al. (2017).

^bResults of an optional study.

inevitably exacerbates design challenges elsewhere. The design of EU DEMO is based on the present physics knowledge—i.e., the H-mode scaling, exhaust and technology—and on the experience gained in the design and construction of ITER. It is not based on an a priori decision to design a large-scale device (Federici et al., 2016).

It is essential to address integration issues in the design activity as early as possible to avoid the development of design solutions that cannot be integrated in practice. In addition, fusion is a nuclear technology and will be assessed with full scrutiny by nuclear regulators. In this context, shielding, remote handling, safety, and licensing considerations all play significant roles in the design.

System codes (see Kovari et al., 2014; Kovari et al., 2016; Reux et al., 2018) for the full DEMO power plant are being used in Europe to underpin DEMO design studies and find meaningful design points (Kemp et al., 2018). These codes are used to find solutions with a minimum major radius, which is a rough proxy for the cost of the device. Determining the available design space for DEMO has been the subject of recent European studies (Federici et al., 2016; Federici et al., 2019; Federici et al., 2017; Siccinio et al., 2019). Three overarching limitations preventing further reductions in major radius are: (1) the divertor power handling, (2) the maximum field in the conductor of the toroidal field (TF) coils and the stress in the coil casing, and (3) the access to the H-mode:

- (1) The minimum size of DEMO is currently limited by divertor exhaust power handling for a given machine size, represented in systems code terms as $P_{sep} B/qAR_0$, where P_{sep} is the power crossing the separatrix, B is the toroidal field in the plasma, q is the safety factor at the plasma edge, A is the aspect ratio, and R_0 is the machine major radius (Federici et al., 2019).
- (2) The magnet performance is a further limit on DEMO sizing: the achievable field is limited by the stresses in the mechanical structures of the coils, rather than the superconductor performance. Forces vary with B^2 , and since the coil cannot expand toroidally, it must become radially larger, rapidly limiting how small the machine can become. With an aspect ratio of 3.1, space for a breeding blanket, and stress limits of <700 MPa in the structural coil materials, targeting a field of 5T in the plasma leads to a device with $R_0 > 7$ m without considering other limitations. A growth in the coil allows higher fields representative of high-temperature superconductors (HTS), but without a corresponding increase in the stress limit, results in a larger machine (albeit, one with improved plasma confinement). To an extent, this can be overcome by excluding tritium breeding from the inboard side to reduce the plasma-magnet distance, but this seriously compromises the ability to breed fuel.

- (3) In order to access H-mode, it is assumed that the amount of power crossing a flux surface just inside the separatrix must exceed the L-H transition threshold power P_{LH} . At present, it is assumed that $P_{sep} > P_{LH}$ for DEMO, as it is likely that P_{sep} will need to be higher than P_{LH} in order to achieve sufficient controllability and confinement quality. Using the Martin scaling (Martin et al., 2008) for P_{LH} and the Greenwald scaling (Greenwald et al., 1988) for the density limit, it can be found that for a fixed P_{sep}/R_0 (i.e., divertor protection figure of merit) increasing the ratio of P_{sep}/P_{LH} can only be satisfied by reducing the magnetic field (Wenninger et al., 2015).

Allowing a variation in aspect ratio, R/a , appears to overcome some of these limits. As aspect ratio falls, elongation can be increased and a higher β_N is achievable; however the increased minor radius means that the field in the plasma is lower and the actual plasma pressure does not change much. For a given achievable field at the TF coil, there is no significant change in power density, although lower aspect ratio designs can deliver higher absolute power due to increased plasma volume (but must still respect power exhaust constraints). This increased power comes at the cost of TF coils that have a larger cross-sectional area, to accommodate the increased plasma volume.

Taking all these elements into account using more detailed models, and allowing for some conservatism, a device of $R_0 \sim 9.0$ m is required. To significantly reduce the size requires advances in plasma physics (higher energy confinement, control and diagnostics in a fusion environment, plasma scenarios that reduce the power density to the divertor target, and highly reliable techniques to mitigate the transient effects of ELMs or exploitation of ELM-free scenarios); materials and design solutions to handle higher power densities in multiple parts of the reactor during steady-state operation and transients; remote handling maintaining high availability with restricted access; and improved magnets capable of higher fields and handling the resulting structural stresses. This must be achieved while integrating nuclear safety in the design ab initio, and without jeopardizing reliability. This does not mean that the EU-DEMO is low-risk, but that the approach chosen is to minimize the risk in extrapolation.

An increased appetite for risk on fusion performance would lead to design changes, which may ultimately reduce the size of DEMO. The first is a reduction in conservatism—a more complete scientific and technical basis allows a reduction in performance margins on extrapolation and increased confidence in plasma control at high radiative fraction, plasma-facing component (PFC) surface erosion rates, or higher β_N . This may be offset by the need to operate in e.g., ELM-free regimes. It is anticipated that the ITER and DEMO research programs will improve understanding and reduce uncertainties before the DEMO design point is finalized. If the high-level goals of DEMO are relaxed (e.g., through a reduction in target electricity production or tritium self-sufficiency, or less targeted technology transfer to a fusion power plant), size savings can be achieved. Pulse length could also be shortened (to save solenoid space) or lower aspect ratio explored (this can generally achieve higher bootstrap current fraction, supporting longer pulse length without additional auxiliary current drive). In the first case, the DEMO mission is compromised and in the second, the design is based on a reduced scientific basis.

Fusion nuclear science facilities and pilot plants

Studies have been performed to evaluate what additional science and technology advances would be required to significantly reduce the scale of the devices summarized in Table 3 and/or to pursue a complementary mission to those of ITER and DEMO to make unique contributions to the world fusion development program. The Fusion Nuclear Science Facility (FNSF) (Peng et al., 2009; Menard et al., 2016; Kessel et al., 2018) aims to create a fusion-nuclear environment representative of fusion power plants to establish a materials and component operational database before proceeding to larger electricity producing devices (see Fig. 6). To be relevant to future fusion power plants, an FNSF facility and program must also utilize and advance fusion-relevant materials for radiation resistance, low activation, operating temperature range, chemical compatibility, and plasma-material interaction damage resistance.

In an FNSF program, blanket material and component development and testing leading to tritium self-sufficiency and high thermal conversion efficiency are high priority mission elements. For FNSF concepts pursued in the U.S., full poloidal Dual Coolant Lead Lithium (DCLL) blankets are pursued as the baseline blanket concept due to the potential for high operating temperature and thermal conversion efficiency. Helium Cooled Lead Lithium (HCLL) and Helium Cooled Ceramic Breeder/Pebble Bed (HCCB/PB) are considered alternate blankets to be developed in FNSF if necessary to mitigate DCLL risks in liquid metal breeder technology and the flow channel insert development (Kessel et al., 2018). Detailed 3D neutronics and tritium breeding ratio calculations are critical to assess the effects of test blanket modules and non-breeding regions (e.g., materials test modules, ports for heating and current drive systems, diagnostics, and disruption mitigation) on the viability of T self-sufficiency in FNSF.

A conventional aspect ratio ($A = 4$) and steady-state FNSF operating range has been identified (Kessel et al., 2018), and detailed analysis has been carried out for an operating point with $R = 4.8$ m, $a = 1.2$ m, $I_p = 7.9$ MA, $B_T = 7.5$ T, $\beta_N < 2.7$, $n/n_{Gr} = 0.9$, $f_{BS} = 0.52$, $q_{95} = 6.0$, $H_{98} \sim 1.0$, and $Q = 4.0$. This operating space is found to be robust to parameter variations and operates $\sim 35\%$ below the divertor exhaust power parameter limit of $P_{sep} B/q A R_0 = 9.3$ MW/m used for EU DEMO (Federici et al., 2019). At full performance the $A = 4$ FNSF generates 1.2 MW/m² average neutron wall loading, which is comparable to the EU DEMO1 value, but with a major radius approximately half that of DEMO1 (4.8 m vs. 9.1 m) and with one quarter of the fusion power (0.52 GW vs. 2.04 GW). The FNSF facility discussed here (with $A = 4$) was not designed to generate net electricity but has an equivalent engineering gain near 1 ($Q_{eng} = 0.86$) and could access $Q_{eng} > 1$ (equivalent) with increased confinement enhancement factor, i.e., $H_{98} > 1.0$. It also aims to access higher shaping (higher elongation and triangularity) and exploits a double-null divertor configuration to increase stability limits. The design uses a radial port maintenance scheme, which differs from the vertical

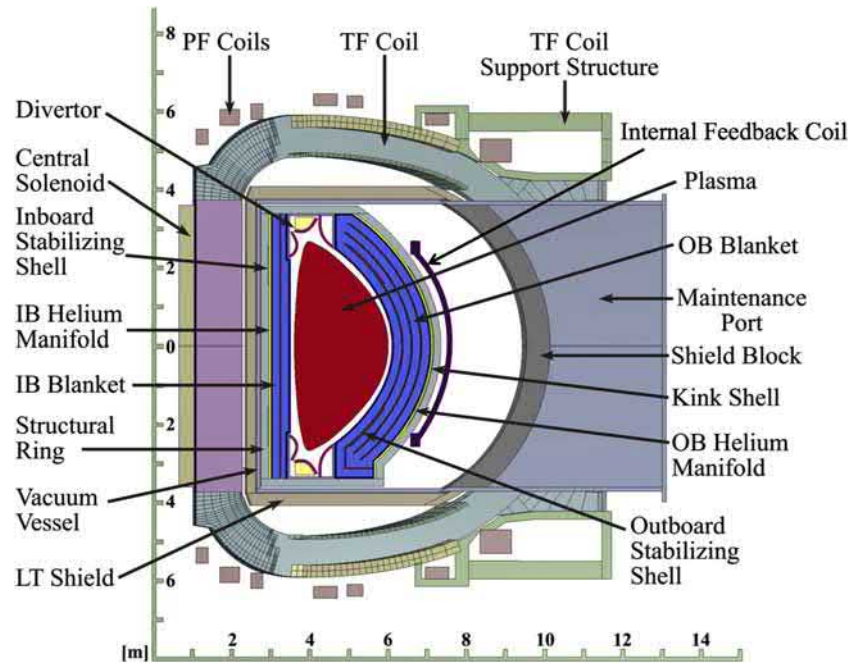


Fig. 6 Cross section of the Fusion Nuclear Science Facility. Reproduced with permission from Kessel CE, Blanchard JP, Davis A, et al. (2018) Overview of the fusion nuclear science facility, a credible break-in step on the path to fusion energy. *Fusion Engineering and Design* 135B: 236–270. <https://doi.org/10.1016/j.fusengdes.2017.05.081> (web archive link,).

port maintenance approach used by EU DEMO, CFETR, and several other DEMO concepts. The device maintainability and availability dependence on maintenance approach is an active area of research.

Lower aspect ratio and smaller major radius FNSF concepts based on the spherical tokamak (ST) have also been explored. An ST with copper toroidal field coils is a promising candidate for the FNSF application due to its potentially high neutron wall loading and modular configuration (Menard et al., 2016), with fusion-relevant neutron wall loadings of $\sim 1 \text{ MW/m}^2$ possible in devices as small as $R = 1 \text{ m}$, at fusion powers of 50–100 MW, and a vertical maintenance scheme enabling removal of all in-vessel fusion core elements. An ST-based FNSF with $R = 1 \text{ m}$ cannot achieve a Tritium Breeding Ratio (TBR) ≥ 1 (TBR without Be enrichment = 0.85–0.9) and would require T from external sources. Larger ST devices with $R \geq 1.7 \text{ m}$ and $P_{\text{fus}} = 150\text{--}200 \text{ MW}$ are projected to achieve $\text{TBR} \geq 1$. Potential disadvantages of the ST-based FNSF device include the lack of a central solenoid for current initiation or sustainment, neutron embrittlement of the central Cu TF magnet, hundreds of MW of resistive power dissipation in the TF magnet, and a less developed physics basis for energy confinement and non-inductive current sustainment. The NSTX Upgrade (Menard et al., 2012) and MAST Upgrade (Chapman et al., 2015) facilities and research programs are strongly aligned with addressing these ST physics issues, with NSTX-U emphasizing core transport and stability and non-inductive sustainment, and MAST-U emphasizing power exhaust mitigation using novel long-legged divertor capabilities and core-edge integration with such divertors.

While the FNSF approach emphasizes nuclear component and blanket development to demonstrate tritium self-sufficiency and steady-state plasma operation, a further related mission is that of achieving electricity break-even ($Q_{\text{eng}} \geq 1$) in a so-called “Pilot Plant” (Dean et al., 1991; Dean et al., 1992; Menard et al., 2011). Such a mission places lower priority on very high neutron fluence operation. For advanced tokamak (AT)-based Pilot Plant operation, both higher maximum field and higher effective current density within the inboard leg of the toroidal field coil was found to be a key enabling capability for achieving electricity break-even in a smaller-major-radius $R = 4 \text{ m}$, $A = 4$ tokamak (Menard et al., 2011). Toroidal field magnets using REBCO HTS tapes with very high peak on-coil magnetic field, $B_{\text{max}} = 23 \text{ T}$, have been proposed to access very high toroidal magnetic field $B_0 = 9.2 \text{ T}$ at the plasma geometric center in an $R = 3.3 \text{ m}$ compact pilot plant, “ARC”, producing 190 MW of net electric power (Sorbom et al., 2015). Compact pilot plant concepts, such as ARC (see Fig. 7) motivate the development of several potential innovations including high-field-side lower-hybrid current drive for steady-state operation, demountable superconducting TF magnets for fusion core access and maintenance, an immersion blanket using the molten-salt FLiBe as a single-phase, low-pressure, single-fluid coolant, and a long-legged tightly-baffled divertor power exhaust scheme (Wigram et al., 2019) compatible with the immersion blankets and elevated core confinement ($H_{98} = 1.8$).

Lower aspect ratio fully-non-inductive tokamak pilot plant concepts with $A = 2$, $R = 3.0 \text{ m}$, $B_T = 4 \text{ T}$, $P_{\text{fus}} = 500\text{--}550 \text{ MW}$, $Q = 10$, and $P_{\text{net}} = 50\text{--}100 \text{ MW}$ which exploit the projected very high current density of REBCO-based HTS superconducting TF magnets have also been developed (Menard et al., 2016). A proposed $A = 2$ HTS pilot plant uses full poloidal DCLL blankets, as proposed for the $A = 4$ FNSF, but has a vertical maintenance scheme similar to the EU DEMO approach. Neutronics calculations for the $A = 2$ HTS pilot plant indicate that an inboard water-cooled shield combined with a thin inboard DCLL breeding region

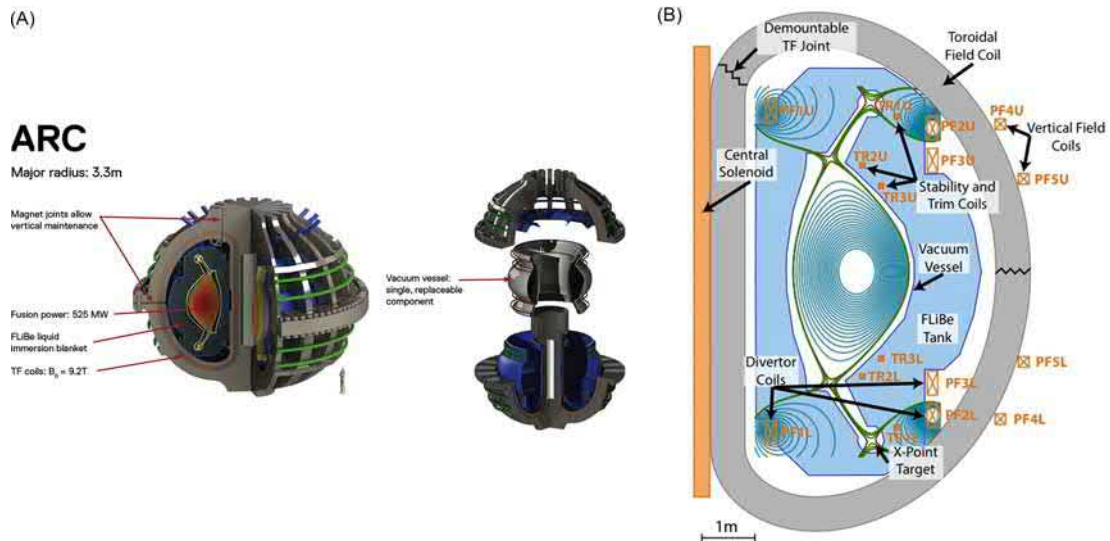


Fig. 7 (A) Rendering of the ARC power plant; and (B) schematic diagram of the long-legged X-point divertor. (A) Modified from Sorbom BN, Ball J, Palmer TR, et al. (2015) ARC: A compact, high-field, fusion nuclear science facility and demonstration power plant with demountable magnets. *Fusion Engineering and Design* 100: 378–405. <https://doi.org/10.1016/j.fusengdes.2015.07.008> (web archive link,). Figure credit: MIT PSFC. (B) Reproduced with permission from Kuang AQ, Cao NM, Creely AJ, et al. (2018) Conceptual design study for heat exhaust management in the ARC fusion pilot plant. *Fusion Engineering and Design* 137: 221–242. <https://doi.org/10.1016/j.fusengdes.2018.09.007> (web archive link,).

(and a thick outboard breeding blanket) can achieve $TBR > 1$, high neutron flux and fluence, and tritium and electrical self-sufficiency. Such a device may reduce the magnet mass per unit fusion power relative to higher aspect ratio configurations and may offer core stability and confinement advantages (Menard, 2019). However, at these low aspect ratios there is only room for a small central solenoid, so non-inductive current formation and/or overdrive would be needed to access the full flat-top operating current of 12–13MA. Furthermore, as is the case for the ST-based FNSF, the physics basis for higher-field low-A tokamaks with $A = 1.7$ to 2.2 has not yet been established and will need to be supported by research on NSTX-U, MAST-U, and ST-40 (Windridge, 2019).

A key question to be addressed in choosing the optimal aspect ratio for fully-non-inductive pilot plants is how energy confinement scales versus aspect ratio between spherical and more conventional tokamak aspect ratios, i.e., covering the range $A = 1.7$ to 2.4 (Menard, 2019). If high confinement and high-bootstrap fraction scenarios can be realized in near-term STs and if high-field high-current-density TF magnets can be realized (Windridge, 2019; Sorbom et al., 2015), a more modular path to fusion energy development may become feasible (Chuyanov and Gryaznevich, 2017). The UK government has also expressed interest in advancing the ST for fusion energy through the Spherical Tokamak for Energy Production (STEP) program (Wilson, 2020). STEP is proposed as a staged program to design and build the world's first compact fusion reactor based on the ST by 2040.

Prospects for stellarator-based demonstration reactors

The stellarator (Yamada, 2021) is a promising alternative to the tokamak, offering inherently steady-state operation and enhanced plasma operational reliability without severe disruptions. It is therefore being pursued internationally as a potential fusion power plant candidate. The freedom to shape a stellarator plasma in three dimensions opens up a path to resolve plasma and engineering issues by computational design rather than by relying on specific plasma regimes. Specifically, stellarators possess 40–50 degrees of freedom in shaping the magnetic field, providing a virtually infinite configuration space. However, while significant progress has been made in the physics optimization, the challenges in the area of stellarator engineering remain to be addressed. A variety of stellarator concepts exist, the most studied concepts being heliotrons and advanced stellarators (including compact designs) (MFE16, 2016).

As for tokamaks, some of the greatest challenges are in engineering and technology (e.g., 3D non-planar magnet design, heterogeneous neutron loads, divertor heat loads, 3D blanket design and segmentation, remote maintenance, etc.). Detailed studies of these issues are limited, but these aspects have gained more interest in recent years and it is expected that, with the success of devices such as W7-X and LHD, further attention will be drawn to stellarator-specific reactor-relevant engineering and technology aspects. Similar to the physics optimization, stellarator engineering offers tremendous potential to address challenges by computational design and optimization.

Design studies for a stellarator-based DEMO device are less advanced than those for tokamak-based devices, though conceptual studies of fusion reactors based on extrapolations of the most favored stellarator configurations have been developed (see Fig. 8), e.g., HELIAS 5-B (Schauer et al., 2011, 2013), extrapolated from the W7-X design (Grieger and Milch, 1993; Grieger et al., 1992), and

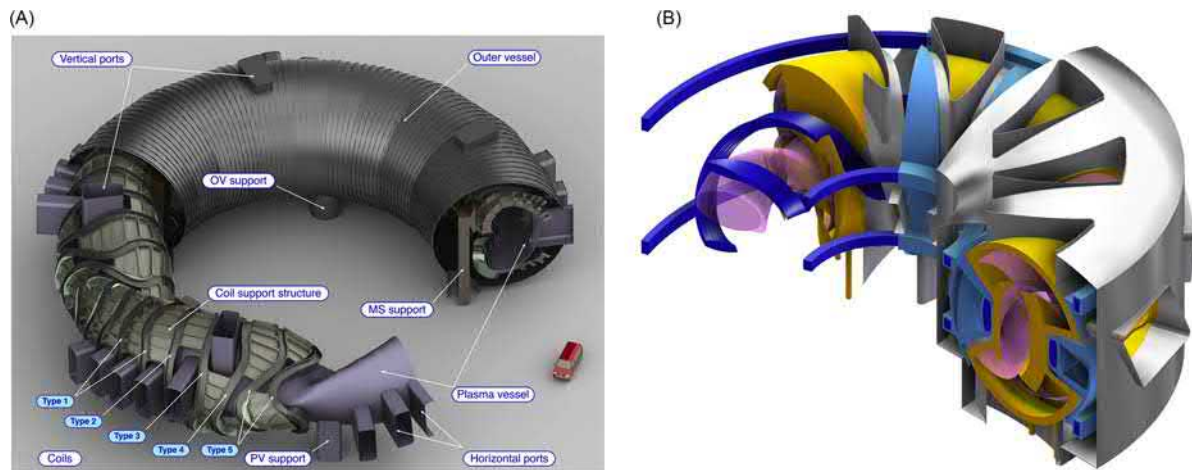


Fig. 8 HELIAS 5-B, design study for an advanced stellarator fusion power plant and FFHR-d1, a heliotron type design based on LHD. (Left) Reproduced with permission from Schauer F, Egorov K, and Bykov V (2013) HELIAS-5B magnet system structure and maintenance concept. *Fusion Engineering and Design* 88: 1619–1622. <https://doi.org/10.1016/j.fusengdes.2013.01.035> (web archive link,). (Right) Reproduced with permission from Tamura H, Tanaka T, Goto T, et al. (2015) *Fusion Engineering and Design* 98–99: 1629–1633. <https://doi.org/10.1016/j.fusengdes.2015.01.025> (web archive link,).

FFHR (Sagara et al., 2012) a refinement of the heliotron concept, where extensive experimental experience has been accumulated, including more than two decades of operation of the Large Helical Device, LHD (Iiyoshi et al., 1999).

With a major radius, $R = 22$ m, HELIAS-5B has a slightly larger aspect ratio, $R/a = 12$, than that of W7-X to accommodate space for the blanket. The space between the plasma and the magnetic field coils is a limiting factor in stellarator designs and space requirements for a blanket impose a lower limit on the device size. The space between the coils is also an important factor for port-based maintenance, limiting the maximum size of the blanket segments. Therefore, optimised coil designs are under investigation that offer a higher accessibility without compromising the magnetic configuration (Gates et al., 2017). Using Nb_3Sn superconductor technology similar to ITER, which allows a field of 5–6 T on-axis, the 50 non-planar field coils are roughly the same size as the ITER toroidal field coils, allowing off-site fabrication. High-temperature-superconductors are also under consideration, but the associated quench protection and electromagnetic forces are challenging aspects that need detailed investigations.

Winding of the large helical coils, which must be prepared on-site, is a particular challenge for the FFHR design. The large poloidal field coils also require on-site assembly. Joint-winding of large-current capacity conductors employing high-temperature superconducting tapes that could be connected on site by mechanical connection offers a potential alternative. Sufficiently low joint resistance has been confirmed by a prototype conductor (Yanagi et al., 2015) and a large-scale fabrication method is under investigation. On the other hand, this concept incorporates large gaps between the helical windings, which serve as access points for port-based remote maintenance.

To provide tritium breeding in HELIAS-5B, adaptation of tokamak DEMO blanket concepts is under study. The HCPB concept (Boccaccini et al., 2021) appears favorable due to its low space requirements and very high tritium breeding ratio ($\text{TBR} = 1.38$ for HELIAS 5-B; Häussle et al., 2018). The FFHR blanket design uses a molten salt (LiF-BeF_2) self-cooled liquid tritium breeder, referred to as FLIBE (Sagara et al., 2000), though a more refined FLINABE (LiF-NaF-BeF_2) concept has been proposed to increase the acceptable temperature range, achieving a higher thermal efficiency (Sagara et al., 2017).

In HELIAS configurations, magnetic islands are naturally generated at the plasma edge. These islands are intersected by divertor plates forming the so-called ‘island divertor’. Successfully tested in Wendelstein 7-AS (McCormick et al., 2003) and the subject of further detailed investigations in W7-X (Niemann et al., 2020), this is a promising exhaust concept for a stellarator reactor. In the FFHR heliotron concept the divertor configuration also originates naturally from the helical geometry, producing a continuous helical ‘double null’-like configuration. However, plasma detachment is necessary to protect the divertor against local hotspots produced by toroidal asymmetries. Selection of a favorable magnetic configuration and/or application of resonant magnetic perturbations (RMP) are options to achieve reduced heat loads (Yanagi et al., 2019). Advanced divertor concepts using molten tin shower jets or a metal pebble flow are also under investigation (Miyazawa et al., 2017; Ohgo et al., 2019).

To bridge the physics and engineering gaps between W7-X and a HELIAS reactor, several options for an intermediate-step stellarator have been proposed (Warner et al., 2016). An intermediate device could either be: (a) a physics-focused device with significant fusion power to study burning plasmas and which can also be used as a steady-state volumetric neutron source; (b) a technology focused DEMO-like device, including a full prototype blanket, aiming to provide an integrated systems solution with a moderate amount of net electricity. The HELIAS line, features a high aspect ratio, but studies of more compact stellarator designs with two or three field periods are also ongoing (Henneberg et al., 2019). Although no such device exists, a small prototype experiment is planned in China (Isobe et al., 2019). A recent study showed that it is, in principle, possible to substantially simplify the

magnetic field coils of such a device by introducing additional permanent magnets that can generate a part of the poloidal flux and rotational transform (Helander et al., 2020).

For the FFHR concept, a step-by-step development strategy is considered focusing on system integration to overcome the engineering and technology challenges: (1) FFHR-a1, a device for demonstration of advanced engineering in a non-nuclear environment; (2) FFHR-b2, a steady-state neutron source for material qualification and testing, as well as providing an early demonstration of DT fusion in the stellarator configuration; (3) FFHR-c1, a heliotron DEMO; and (4) FFHR-d1, a full-scale power plant (Miyazawa et al., 2019). High-temperature superconductors are an option for these designs, allowing higher magnetic field and more compact devices.

Fusion materials test facilities

Introduction

Radiation damage in materials is a complex function of two competing basic phenomena (nuclear reactions and elastic collisions) for the primary damage and the specific characteristics of each material components, as well as temperature, for the later evolution of the initial effects (see Litnovsky et al., 2021 for a more detailed description of reactor materials). Elastic collisions induce atomic displacement of the ions inside the materials giving rise to point defects and nuclear reactions give rise to material activation and impurification. The relative relevance of both types of reactions is a strong function of the irradiating particles, their energy and the material composition. This complexity makes it almost impossible at present to predict the behavior of materials in a radiation environment, and systematic characterization experiments are required in the future, especially for the case of high-dose irradiation conditions (Knaster et al., 2016b).

The radiation environment of a fusion reactor is also a complex function of the reactor design and it is strongly dominated by the presence of the 14 MeV neutrons produced by fusion reactions in the areas close to the first wall. This complex environment very soon raises the need for a fusion-like neutron source focused on the development and qualification of the materials to be used in a fusion reactor. This source will mainly be used during the design phase of the fusion device (in order to define the design window of the different components) and during the licensing phase (in order to demonstrate to the licensing body that the properties of the structural materials used in the fabrication of the device are sufficiently robust under nuclear conditions, to ensure that the safety of the reactor is guaranteed). Taking this into account, the requirements for the neutron source facility can be summarized as: “the facility should be able to produce fusion-like neutrons with an intensity large enough to allow accelerated testing, up to a damage level above the expected operational lifetime in a reactor, and in an irradiation volume large and uniform enough to allow the characterization of the macroscopic properties of the materials of interest required for the engineering design of the fusion machine”. Here it is important to emphasize that the need to characterize the behavior of engineering macroscopic properties requires the use of a relatively large irradiation volume. In order to minimize this volume, the use of small sample testing techniques is nowadays strongly being developed.

The specifications of the irradiation requirements are a function of the specific fusion power plant design. In most countries with a significant fusion program, the present strategy is based on a stepped approach in which it is planned to qualify radiation effects up to the dose required by the different DEMO designs and, at a later stage, the fusion power plant needs will be addressed. So, for example, in the case of the European Roadmap and the EU-DEMO currently under development, the facility should provide a neutron flux in the high flux region of the source enabling the accumulation of fluences of 20–30 Norgett-Robinson-Torrens (NRT; Norgett et al., 1975) displacements per atom (dpa) in <2.5 years applicable to a 0.3 L irradiation volume, and fluences of 50 NRT-dpa in <3 years applicable to a 0.1 L volume.

It is important to note that several neutron sources are already available (e.g., materials test reactors), which are able to provide the required neutron flux. However, their energy characteristics are very different from those of a fusion reactor. Consequently, they can be of interest to study materials irradiation effects, but they are not useful for the qualification of the materials in a fusion reactor. For example, neutrons produced in fission reactors have a lower energy (typically up to 1–2 MeV) and the number of transmutation reactions will therefore be much smaller than in the case of neutrons produced in a fusion-like environment. On the other side, the energy spectra of the neutrons produced in spallation sources have a very high-energy tail and, as a consequence, the number and type of transmutation reactions are much higher than in the case of a fusion-like environment (Zinkle and Möslang, 2013).

Fusion-like neutron sources

In recent decades, a variety of concepts have been proposed for building a neutron facility which satisfies the testing requirements discussed above. All are based either on the production of neutrons by the interaction of D and T, or on the production of neutrons by using the interaction between a light ion (usually D) and a light target (like Li, C or Be). In the latter case, the neutron spectra produced are not exactly the same as those in a fusion reactor, but the predicted effects in the materials have been assessed to be similar. The two approaches have given rise to several different proposals, although the only one currently in a relevant engineering design phase is the so-called IFMIF-DONES (International Fusion Materials Irradiation Facility-DEMO Oriented Neutron Source) project described in more detail in section “IFMIF-DONES”.

The most obvious way to produce fusion-like neutrons is by using the D-T reaction. Usually such facilities are based on a low energy deuteron beam (a few hundreds of keV) impinging on a tritium target. The target can be solid, with tritium embedded in a solid matrix (SORGENTINA (Pillon et al., 2014), HINEG-II (Wu, 2016)) or gaseous (PNL/UW (Kulcinski et al., 2017)). The main difficulty of this approach is linked to the fact that, to obtain the required neutron flux, a high reaction rate is required. For solid targets, this is only possible with a very high power density, which is beyond present technological capabilities. Alternatively, a high number of smaller neutron sources can be applied (PNL/UW). Nevertheless, conceptual designs which have been proposed are not able to reach the required neutron fluxes.

Thus, for example, SORGENTINA is a project in which an intense 14 MeV D-T neutron source is based on continuous D and T ion beams at 40 A, 200 keV energy, impinging onto a 2 m diameter water-cooled rotating target operated at a rotational speed of about 1000 rpm. It produces up to about 10^{15} n/s (about 1×10^{13} n/cm² s over about 50 cm³ and dpa rates corresponding to 2–4 dpa per full-power year (fpy)). The facility makes use of an ion source and accelerator stages from neutral beam injectors utilized at large experimental tokamaks together with a properly scaled rotating target technology.

The PNL/UW proposal is able to produce peak damage levels of ≈ 4 –6 dpa/fpy in 15 cm³ and has a test volume of ≈ 600 cm³ covering the damage range from 1 to 6 dpa/fpy. This design is based on a quasi-commercial linear 14 MeV DT neutron source generated by injecting a 300 keV deuterium beam (of a few hundred mA intensity) into a 30 Torr tritium gas target (the pressure in the target can be tuned to optimize the irradiation parameters). A number of these small neutron sources can be arranged in a lattice configuration in which the materials to be irradiated can be located.

An interesting alternative approach is represented by the so-called “volumetric” neutron sources in which a plasma device with no energy gain ($Q < 1$), but with external heating and an external supply of D and T is used to produce 14 MeV neutrons. The Gas Dynamic Trap (GTD) (Anikeev et al., 2015) is based on a special magnetic mirror system for plasma confinement. The main advantage of this type of neutron source is that the available irradiation volume is high (typically volumes of 0.3 L at > 15 dpa/full power year; 17.5 L at 10–15 dpa/fpy; 6.9 L at 5–10 dpa/fpy and 24.7 L at < 5 dpa/fpy; Fischer et al., 2000). The main difficulties are related to the plasma regime (still not fully demonstrated), the required availability (continuous operation) and all the aspects related to the device engineering (especially tritium and remote handling management). Also some of the ST-based concepts discussed in section “Fusion nuclear science facilities and pilot plants” have been proposed as Component Test Facilities as they exhibit the proper fusion neutron spectrum.

The production of neutrons by using the interaction of D with other light ions in solid targets is utilized in FAFNIR (Surrey et al., 2014) in which a C target is used. Its performance is limited by the power handling capability of the target. Facilities in which a D-beam interacts with light ions in liquid metals constitute the IFMIF family of devices, including the IFMIF proposal (Knaster et al., 2016a), IFMIF-DONES (Ibarra et al., 2017), A-FNS (Ochiai et al., 2020) and BISOL (Liu et al., 2019). The principle is based on Li(d,xn) nuclear reactions taking place in a liquid Li target when bombarded by a deuteron beam. To obtain neutron spectra comparable to that in the first wall of a fusion power reactor, the deuteron beam is accelerated up to 40 MeV. The main difference between the various proposals is linked to the accelerator current and, as a consequence, with the corresponding irradiation performance for each.

IFMIF-DONES

The IFMIF-DONES facility will be the neutron source that can provide conditions such as anticipated for EU-DEMO. To obtain irradiation conditions comparable to those at the first wall of a fusion power reactor, the DONES Facility will produce a 125 mA deuteron beam, accelerated up to 40 MeV (5 MW beam power) and shaped to have a footprint in the range from 100 mm $\times$ 50 mm to 200 mm $\times$ 50 mm, providing an irradiation volume of up to 0.5 L that can house around 1000 small specimens irradiated to a high dose rate (over 10 dpa/fpy). The beam will impinge on a 25 mm thick liquid lithium target intersecting the beam with a transverse velocity of about 15 ms^{-1} . The Li(d,xn) stripping reactions will generate a large number of neutrons (up to 5×10^{18} n/m²s) that will interact with the material samples located immediately behind the lithium target, in the Test Modules (Ibarra et al., 2018).

The site-specific engineering design of the IFMIF-DONES facility has been underway since 2015 with the objective to be ready for construction of the facility in the early 2020's (Ibarra et al., 2019) in Granada (Spain). It is based on the IFMIF engineering design (Knaster et al., 2015) developed in the framework of the IFMIF/EVEDA (acronym that stands for Engineering Validation and Engineering Design Activities) project, but updated to take into account the validation results obtained plus modelling calculations, and simplified as much as possible in order to reduce cost. Fig. 9 shows a schematic view of the operational principles and a simplified CAD drawing of the facility.

The IFMIF/EVEDA project also developed a number of prototyping sub-projects (Knaster et al., 2013), including: (i) an Accelerator Prototype (LIPAc), assembling IFMIF components up to its first superconducting acceleration stage (9 MeV energy, 125 mA of D⁺ continuous wave current); and (ii) the EVEDA experimental Lithium Test Loop (ELTL), which integrated all elements of the IFMIF Li target facility that was used to test the long-time stability of the liquid lithium jet forming the target. Results obtained to date have not identified any significant showstoppers.

Taking all these results into account, the construction phase of IFMIF-DONES could start in the early 2020's and the facility could be operational around 10 years later. The aim is to produce first preliminary data around 2035, on time for the engineering design of EU-DEMO.

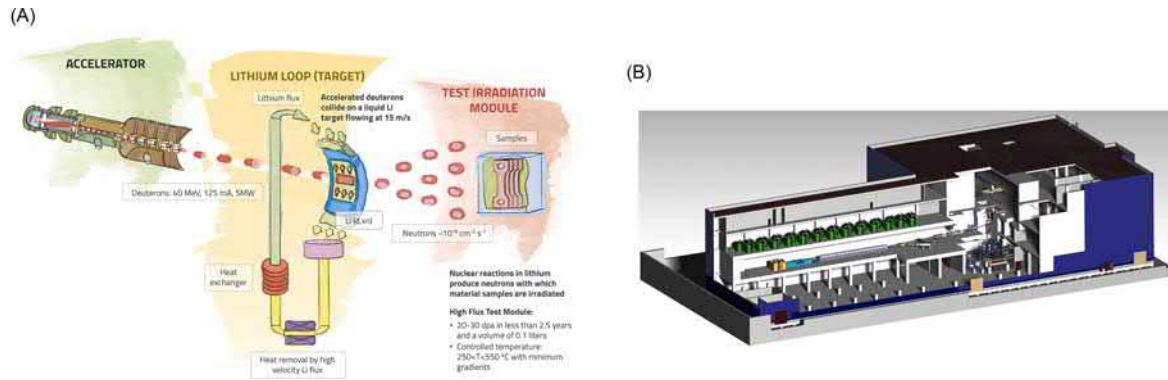


Fig. 9 IFMIF-DONES facility: (A) Schematic view of the operational principles, (B) a simplified CAD drawing. Courtesy CIEMAT, Spain.

Non-nuclear development facilities

Several significant non-nuclear development facilities are currently under construction, or in planning, which are expected to deliver important complementary R&D information to the nuclear devices described in earlier sections (see Table 4). All of these devices are non-nuclear in the sense that they will not operate with a deuterium-tritium mixture.

The largest of these devices is the Japanese-European JT-60SA tokamak (Barabaschi et al., 2019). The mission of JT-60SA is to contribute to the early realization of fusion energy by addressing key physics issues for ITER and DEMO. It is a fully superconducting tokamak capable of confining high-temperature (100 million degree) deuterium plasmas, under plasma conditions equivalent to 'breakeven' (where the fusion power would be equal to the input power if the plasma were operated in D-T). It is designed to support the optimization of the plasma configurations for ITER and DEMO and has a high plasma heating and current drive power, using both positive and negative ion based neutral beams, as well as ECRH. It will typically operate for 100 s pulses once per hour, subjecting water-cooled divertors to maximum heat fluxes of 15 MW m^{-2} . The device will be able to explore fully non-inductive steady-state operation. The assembly of JT-60SA was completed in 2020 and, on entry into operation in spring 2021, it will be the world's largest operating tokamak, in succession to the Joint European Torus.

COMPASS-U in Prague (Pánek et al., 2017) is a high field, high-density tokamak, with a flexible set of poloidal field coils for generation of single-null, double-null and snowflake divertor configurations. In addition, COMPASS-U will be equipped with a closed divertor operated at high plasma and neutral density and with high optical opacity similar to future reactors. COMPASS-U will address some of the key challenges in the field of the plasma exhaust physics, reactor-relevant edge plasma physics, advanced confinement regimes, advanced magnetic configurations and materials for next-step devices. The copper coils are cooled with liquid nitrogen to allow for higher magnetic fields.

The Divertor Test Tokamak in Frascati (Albanese et al., 2019) is a high-field superconducting tokamak aimed at exploring and qualifying alternative power exhaust solutions for DEMO. The device will have the possibility to test several different magnetic divertor topologies in relevant reactor regimes. Various plasma facing materials will be tested (e.g., tungsten, liquid metals) at divertor heat fluxes in excess of 20 MW m^{-2} . The final target of the experiment is the realization of an integrated solution (bulk and edge plasma) for the power exhaust in view of DEMO. For this purpose, the DTT tokamak has edge conditions that are similar to those of DEMO in terms of dimensionless parameters.

HL-2M (Li and HL-2M Team, 2015) is a copper-conductor tokamak under construction in Chengdu, China and is dedicated to studying the critical physics and engineering issues of ITER and CFETR.

Table 4 Main parameters of the most significant non-nuclear fusion development devices.

	COMPASS-U	DTT	HL-2M	JT-60SA
Major radius (m)	0.84	2.10	1.78	2.96
Minor radius (m)	0.28	0.65	0.65	1.18
Toroidal field (T)	5.0	6.0	2.2	2.25
Plasma current (MA)	2.0	5.5	2.5	5.5
ECRH (MW)	4	20	2 (4)	7
ICRF (MW)		15		
NBH (MW)	4–5	– (10)	2 (4)	34
LHH (MW)			– (2)	
Expected operation	Late 2021	2025	Late 2020	Early 2021

Acknowledgement

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Magnetic Confinement Fusion—Power Plant Concepts

David J. Campbell*, Munich, Germany

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Glossary

(Magnetic) Safety factor The safety factor, q , of a given magnetic flux surface within a toroidal plasma equilibrium is most simply thought of as the “magnetic winding number,” i.e., the number of toroidal passages a (helical) magnetic field line requires within the flux surface before it closes on itself poloidally. It is related to the “rotational transform,” ι , more commonly used to describe stellarator plasmas, by $\iota = 2\pi/q$ (Galvão and Canal, 2021).

(Poloidal) divertor A toroidally continuous in-vessel structure in tokamaks which is designed to intercept (at “divertor target plates”) the peak heat fluxes flowing into the scrape-off layer and to allow extraction (i.e., by pumping) of helium produced by fusion reactions (Zohm, 2021). Stellarators use modified versions of this concept referred to as a “helical” or an “island” divertor (Yamada, 2021).

Aspect ratio The ratio R/a , where R is the torus major radius and a the (horizontal) minor radius.

Beta β Ratio of the volume averaged plasma thermal pressure, $\langle p \rangle$, to the pressure exerted by the toroidal magnetic field at the plasma center, $\beta = 2\mu_0 p B_t^2/0$. β is typically quoted in percent (Wade, 2021; Lao et al., 2021).

Beta-normalized β_N A figure of merit used to renormalize and compare the values of β achieved in different tokamak experiments, $\beta_N = \beta\%aB_t0I_p$, where a (m) is the plasma minor radius, $B_t(0)$ (T) the toroidal magnetic field at the plasma center and I_p (MA) the plasma current.

Beta-poloidal β_p Ratio of the volume averaged plasma thermal pressure, $\langle p \rangle$, to the average pressure exerted by the poloidal magnetic field at the plasma edge, $\beta_p = 2\mu_0 p B_p^2/line$, where a is the plasma minor radius and $\langle \rangle_{line}$ represents a poloidal line integral around the last closed flux surface (LCFS).

Bootstrap current A non-inductive current which arises within the plasma due to the existence of a radial pressure profile and an exchange of momentum between the trapped and passing particles.

Burning plasma Plasma in which heating is dominated by fusion products of thermonuclear reactions, i.e., α -particles in deuterium-tritium plasmas.

Central Solenoid (CS) Tokamak coil which produces magnetic flux swing generating electric field for plasma breakdown and (inductive) plasma current drive.

COE Cost of electricity (produced by a power plant).

Disruption In tokamak plasmas a sudden loss of plasma thermal energy and plasma current triggered by the growth of MHD instabilities (Lao et al., 2021).

Double-null X-point (DNX) An up-down symmetric plasma configuration in which the last closed flux surface (LCFS—see definition) is formed by a magnetic separatrix with symmetric poloidal field nulls at the top and bottom of the plasma.

Electrical gain (Q_{Elec}) $P_{e,gross}/P_{e,circ}$ —ratio of the gross electrical output power of a power plant to the recirculating electrical power required to maintain the plant operation.

Energy confinement time $\tau_E = W_{th}/P_{loss}$ Ratio of the total thermal energy of the plasma, W_{th} , to the total power lost by all mechanisms (transport, radiation, etc.), P_{loss} ; under steady-state conditions P_{loss} is equal to the total plasma heating power, i.e., the sum of the α -particle heating power resulting from DT fusion reactions, P_{α} , and the external heating power injected into the plasma, P_{heat} (Wade, 2021).

*Retired.

Full power year (FPY) Integrated power output from a power plant equivalent to 1 year of uninterrupted operation at nominal power.

Fusion gain (Q_{th}) P_{DT}/P_{heat} —ratio of the DT (thermal) fusion output power of a plasma to the input heating power from external sources.

Greenwald density n_{GW} A figure of merit used to normalize and compare the values of plasma density, n (particles per m^{-3}), achieved in different tokamak experiments, $n_{GW} = \frac{I_p}{\pi a^2}$, where n_{GW} is expressed in units of $10^{20} m^{-3}$, a (m) is the plasma minor radius and I_p (MA) the plasma current.

H-mode “High confinement mode” of operation in tokamaks, which usually requires operation above a certain level of input power (denoted the “threshold power”); also observed in stellarators.

Ideal MHD instability A magnetohydrodynamic instability which grows in an “ideal,” i.e., perfectly conducting, plasma.

ITER Major tokamak-based facility designed to produce 500 MW of fusion power using deuterium-tritium plasmas; ITER, which means “the way” in Latin, is an international project under construction in southern France.

kiloelectronvolt (keV) Temperatures, T , in thermonuclear plasmas are normally quoted in keV ($T(\text{keV}) = 10^{-3} k_B T(\text{K})/e$, with k_B the Boltzmann constant and e the electron charge): $1 \text{ keV} = 1.1605 \times 10^7 \text{ K}$

Last closed flux surface (LCFS) Magnetic flux surface within which the magnetic field lines form closed magnetic surfaces and confine the plasma. This may be created by contact with a solid “limiter” surface, or magnetically (referred to as a “magnetic separatrix”), in which case the open flux surfaces in the SOL are channeled towards a divertor target (Galvão and Canal, 2021).

L-mode “Low confinement mode” of operation in tokamaks, typically observed at low values of input heating power.

MHD Magnetohydrodynamic (relating to stability behavior of magnetically confined plasma).

Neutron wall load Power flux on the first (i.e., plasma facing) wall of a reactor due to neutrons produced by fusion reactions, typically expressed in $MW m^{-2}$.

Ohmic heating Plasma heating produced by the flow of the plasma current within the plasma.

Passing particle Plasma particle with sufficiently large component of parallel velocity to follow the helical field lines around the torus (Sarazin, 2021).

Plasma facing component (PFC) In a reactor, a cooled component facing the plasma, either on the first wall or divertor, designed to withstand heat loads in the form of electromagnetic radiation and plasma particles, and surfaced with a material (e.g., tungsten) chosen to withstand plasma bombardment.

Poloidal field coils (PF coils) Set of (toroidally oriented) coils in a tokamak which produces magnetic fields for plasma shaping and control.

RAFM Reduced-activation ferritic-martensitic: A family of steels in which the alloying elements are chosen to lower the susceptibility to radiation damage and to reduce the long-term activation under neutron irradiation.

Resistive MHD instability A magnetohydrodynamic instability whose growth is dependence on the finite resistivity of the plasma.

Scrape-off layer (SOL) In a toroidal magnetic confinement device, the layer of open magnetic field lines outside the last closed flux surface which eventually intersect (material) plasma facing surfaces.

Silicon carbide composite (SiC_f/SiC) SiC-matrix ceramic material reinforced with SiC fibers (SiC_f) that is under development for nuclear applications and is considered as a potential structural material for fusion power plants.

Single-null X-point (SNX) An up-down asymmetric plasma configuration in which the last closed flux surface (LCFS—see definition) is formed by a magnetic separatrix with a single poloidal field null, either at the top or bottom of the plasma.

Toroidal field coils (TF coils) Set of (poloidally oriented) coils in a tokamak which produces the principal component of the confining magnetic field, the toroidal field.

Trapped particle Plasma particle which is reflected from high magnetic field region on torus inboard side and circulates toroidally on torus outboard side (Sarazin, 2021).

Tritium breeding ratio (TBR) Ratio of the number of tritium atoms bred in the breeding blanket per atom of tritium burned in fusion reactions within the plasma.

Vertical displacement event (VDE) Occurs in an elongated tokamak plasma when control of the plasma vertical position is lost: the plasma moves rapidly up or down, leading to a disruption.

Abbreviations

FPP Fusion power plant

HTS High temperature superconductor

LTS Low temperature superconductor

MCF Magnetic confinement fusion

Introduction

In view of the availability of abundant fuel resources (Campbell, 2021) and the benign safety and environmental features of fusion reactors (Taylor, 2021), numerous studies of the physics and technology characteristics of power plants based on magnetic fusion devices have been conducted during the last half century. As discussed by Donné et al. (2021), several of the major research communities engaged in the development of fusion energy have, in recent years, provided a framework for progress towards construction of fusion power plants (FPP) via the development of roadmaps delineating the major steps in the progress towards the exploitation of fusion energy for commercial electricity production. Key aspects of “burning plasma” physics and “reactor-grade” technology will be tested by the ITER project (ITER, 2002) which are then expected to be developed into prototypical concepts for an FPP via the DEMO-generation of fusion devices, which should demonstrate significant electricity generation, at the level of several hundred MW to 1 GW, e.g., (Donné et al., 2021). Following the successful operation of the DEMO-class devices, including the confirmation of the safety and environmental advantages of fusion energy and the demonstration of a viable economic potential, it is expected that commercialization of fusion energy would be undertaken and one or more types of FPP would be constructed in the period after 2050.

Initial fusion reactor concepts developed in the late 1960s and early 1970s, based on the deuterium-tritium-lithium fuel cycle (Boccaccini and Day, 2021) in toroidal magnetic confinement devices (primarily tokamaks), were large (in contemporary terms), with major radius, R , in the range 10–15 m and fusion thermal output powers of up to 6 GW. However, by 1980, progress in understanding of physics and technology aspects of tokamaks had been refined sufficiently that conceptual power plant designs had settled into the range, $R \sim 5$ –8 m, around which the majority of contemporary DEMO and power plant designs have been developed. The following discussion will, in the interests of space constraints, concentrate on the more recent developments in power plant concepts, but excellent summaries of the early progress in fusion reactor studies are provided by Conn (1981) and Braams and Stott (2002).

Conceptual studies for magnetic FPPs have encompassed a wide variety of configurations in the 70 years or so since the controlled fusion research program was launched. Nevertheless, due to the achieved levels of plasma performance, toroidal magnetic confinement configurations have largely eclipsed linear configurations across the fusion research program. Linear devices based on the magnetic mirror configuration, such the Gas Dynamic Trap (GDT), e.g., (Galvão and Canal, 2021; Ivanov and Prikhodko, 2013), and on the field reversed configuration, e.g., (Hirano et al., 2018), nevertheless, remain under investigation. The following discussion focuses on the three principal toroidal configurations, where power plant studies have reached the most advanced level of detail: the tokamak, including spherical tokamaks (ST), the stellarator and the reversed field pinch (RFP).

To develop the framework and motivation for the conceptual power plant designs discussed later, the section “**Requirements for commercial fusion power**” outlines the principal issues relating to safety and environmental considerations, operational reliability and availability, economics and timescale. In the section “**Physics issues**”, the key physics considerations influencing power plant design are discussed, while an overview of the major technology issues is presented in section “**Key technologies**.” “**Conceptual power plant designs**” discusses several examples of contemporary power plant designs based on the tokamak, stellarator and reversed field pinch configurations, and the section “**Conclusions**” develops some brief conclusions on the current status and future development of power plants.

Requirements for commercial fusion power

While the generic concepts of widespread fuel availability and the low environmental risk associated with fusion energy production have provided a long-term motivation for the development of fusion power plants, the acceptability and, indeed, viability of large-scale production of electricity from fusion power requires not only the resolution of the outstanding science and technology challenges (discussed later), but a detailed evaluation of the likely properties of fusion power plants in terms of safety and environmental impact; public acceptance; practicality of nuclear licensing, encompassing radiation confinement and waste disposal; operational reliability and availability, which must include an assessment of the achievable device component lifetime in the burning plasma nuclear environment. These issues will influence the capital cost of a power plant, which will dominate the ultimate cost of electricity (COE) and will itself feedback on both public and industrial/commercial acceptability of fusion energy. Finally, the timescale on which fusion energy can make a significant contribution to the world’s energy consumption will have a substantial impact on the likelihood of its incorporation into the resource mix, particularly in view of the current emphasis on climate change and the expansion of energy resources with reduced or zero CO₂ emission.

These issues have increasingly been integrated into FPP studies since the early 1990s, in particular through a sequence of studies in the EU (Raeder et al., 1995; Maisonnier et al. 2007; Pamela et al., 2009), the United States (Conn et al., 1990, 1998; Miller, 1998; Najmabadi et al., 1997) and Japan (JAERI, 2000; Tokimatsu et al., 2003), and the socio-economic aspects of FPP analysis, relating primarily to safety, environmental and public acceptability considerations, are now a central aspect of conceptual power plant studies. Practical experience in this area has been gained through the licensing activities for the ITER facility (Taylor et al., 2013), the first fusion facility to undergo a full nuclear licensing procedure and defined as a nuclear facility within the relevant national (French) regulatory regime. As discussed by Taylor (2021), the detailed analyses performed in support of the ITER licensing application have provided confidence that even highly unlikely accidents will have very low consequences. Although the national

nuclear regulatory frameworks can differ internationally, key elements of the safety and environmental requirements for an FPP would be:

- high confinement of radioactive materials;
- low exposure to ionizing radiation for personnel, the public and the environment.

A fundamental implication of these requirements is that there should be no need for public evacuation following even the most severe accidents. Furthermore, it is widely accepted within the fusion community that environmental impact/public acceptability will require that waste disposal on geological timescales should not be necessary, i.e., that no high level waste (HLW) should be generated and, if at all possible, waste should ultimately fall within the classification for low level waste (LLW). Both tritium contamination of in-vessel components and material activation by fusion neutrons influence the longer term waste classification. Material processing schemes exist to remove tritium from such components within plant decommissioning procedures and, as discussed by Litnovsky et al. (2021), the principal structural materials being developed for fusion power plants are designed as “low activation” materials. The aim has been to design materials which would satisfy the LLW classification and which would permit, perhaps ~100 years after decommissioning of a power plant, much of the structural material to be recycled into the construction of new FPPs. A detailed analysis of radioactive waste from EU DEMO designs (Gilbert et al., 2019) has highlighted several issues related to the composition of in-vessel materials which could enhance the residual radiation levels beyond the LLW limit on longer timescales and has proposed mitigation measures to guide future reactor designs in this area.

Operational reliability and high availability are necessary characteristics for a power plant if fusion is to find wide acceptance as a commercial source of electricity. As discussed below, analysis of the potential cost for electricity generated by an FPP shows that these factors have a significant impact on the projected COE. Key issues which affect this aspect of fusion power generation include the application of novel technologies (e.g., superconducting magnets, breeding blankets, etc.) in “industrial scale” power generation, the inherent technical complexity of the magnetically confined reactor core and “auxiliary systems” (e.g., fuel processing plant, heating and current drive systems), the challenges of sustaining operation of a burning plasma with high fusion (thermal) power gain, Q_{th} (i.e., $Q_{th} \sim 20\text{--}50$), the component lifetime in the nuclear environment, and the associated ease of maintenance. These challenges in the physics and technology areas are discussed further in the sections “Physics issues” and “Key technologies”.

Assuming that the physics and technology challenges end route to the construction of an FPP can be successfully resolved, fusion electricity must ultimately compete in the energy market with a range of other power generation technologies. A significant rise in total electricity consumption can be expected in the coming decades: the increasing global population, particularly the (increasing) fraction in the developing world, will continue to strive to achieve a higher standard of living, and the decarbonization of industry and transportation is likely to translate into a greater demand for electricity (e.g., MacKay, 2009). The energy market is currently very dynamic, as renewable resources such as wind and solar power are gradually penetrating the market in various parts of the world (particularly the developed world). Predicting the target COE in perhaps 50 years, when a fleet of FPPs could be available, therefore implies significant uncertainties. In addition, the evolution of nuclear regulatory requirements also complicates the prediction of future construction and operational costs of both fission and fusion power plants. Nevertheless, it is likely that some form of decarbonized “baseload” electricity resource will be required to complement the fluctuating electricity production associated with renewables (e.g., Wagner, 2013).

Given the current status of fusion energy research and the resultant uncertainties in the final configuration of an FPP, plus the ongoing evolution in the energy market, the specification of a definitive target for the cost of fusion electricity that would make fusion an economically attractive energy source is premature. Nevertheless, several systematic studies have been performed to examine how fusion might penetrate into the energy market and the implications for both FPP construction costs and the COE, e.g., (Miller, 1998; Tokimatsu et al., 2003; Ward et al., 2005; Lopes Cardozo et al., 2016; Takeda et al., 2018). In spite of the prevailing uncertainties, the value of determining the factors with the greatest influence on the expected COE, as a means of focusing research on optimizing these factors and on developing conceptual FPP designs which incorporate this optimization, is recognized. This analysis has shown that the capital cost of the fusion core, including the replacement of components such as the reactor first wall, blanket and divertor, which will suffer erosion or radiation damage, will dominate the COE, e.g., (Kikuchi et al., 1991; Galambos et al., 1995; Hender et al., 1996; Ward et al., 2001, 2005; Wong et al., 2001; Najmabadi et al., 2006).

To examine how the COE might vary with key parameters related to the construction and operation of an FPP, Ward et al. (2001) analyzed the predicted COE for a range of power plant designs, deriving the scaling,

$$COE \propto \left(\frac{rF}{A} \right)^{0.6} \frac{1}{\eta_{th}^{0.5}} \frac{1}{P_e^{0.4} \beta_N^{0.4} N^{0.3}}, \quad (1)$$

Where r is the discount rate, F the learning factor (i.e., cost reduction factor for a 10th of a kind plant, ranging from 0.5 to 0.8 in this study), A the FPP availability, η_{th} the thermodynamic efficiency, P_e the electrical unit size (GW), β_N the normalized β and N the normalized density ($N = n/n_{GW}$). As might be expected, factors such as β_N and N , which are essentially figures of merit for the level of plasma performance that can be achieved, figure prominently. It is notable also that the COE typically decreases with increasing unit size, although studies have, in the main, settled on a unit size with a net electrical output of 1 GW, considered to be the largest unit which would be acceptable to electricity utilities. The importance of plant availability, and therefore, component reliability and ease of maintenance, clearly also figures strongly, as was noted above. It is also not surprising that “economic” factors such as the thermodynamic efficiency, discount rate and learning factor should have a strong influence on the final COE. Fig. 1 illustrates the

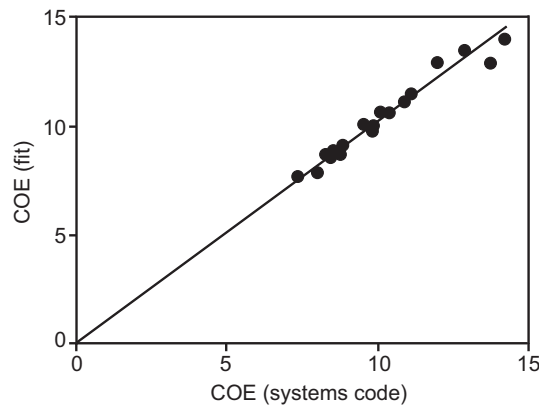


Fig. 1 Comparison of scaling for predicted cost of electricity (Eurocents/kW-h) shown in Eq. (1) versus the predicted cost of electricity for the individual tokamak-based power plant concepts evaluated using the PROCESS systems code. Reprinted from Figure 2 from Ward DJ, Cook I, Lechon Y and Saez R (2005) The economic viability of fusion power. *Fusion Engineering and Design* 75–79: 1221–1227, <https://doi.org/10.1016/j.fusengdes.2005.06.160>, Reproduced with permission of publisher.

scaling shown in Eq. (1) plotted against the predicted COE for individual FPP designs derived using the PROCESS code (Hender et al., 1996) validated against ITER costings and using ITER reference plasma performance. The predicted range of €0.07–0.14/kW h is indicative of the range of uncertainties in the key parameters, both those derived from fusion physics and technology and those derived from economic conditions that may prevail when fusion electricity is finally available in the second half of the 21st century. Similar analysis derived from the ARIES power plant studies program aimed at a COE in the range \$0.065–0.08/kW h (Najmabadi et al., 1997), while the later EU Power Plant Conceptual Study (PPCS) yielded a range of designs spanning a COE of €0.05–0.9/kW h (Maisonniere et al., 2007). It should be emphasized that these costs are determined entirely by the “internal” costs of constructing, operating (including fueling), maintaining and disposing of an FPP. The “external” costs for fusion electricity, i.e., those associated with safety, health and environmental effects, are very low and would make a negligible contribution to the COE if included in this estimate, e.g., (Ward et al., 2005). This reflects the attractive safety and environmental aspects of fusion electricity generation.

Given the range of physics and technology challenges which are still the subject of ongoing R&D programs and which will need to converge via the ITER and DEMO programs to provide a reliable basis for the construction of an FPP, the absolute costs quoted above can only be considered as “ball-park” estimates based on assumptions on whether “conservative” or “advanced” physics and technology solutions will be derived from ITER and DEMO experience (and accompanying programs). Nevertheless, analysis such as that outlined here provides guidance on key issues for future R&D to take magnetic fusion from the realm of research to the practical application of electricity production, and also provides some confidence that fusion electricity can satisfy commercial boundary constraints which would complement its safety and environmental advantages and pave the way to its integration into the future energy market.

Physics issues

The international fusion research community has assembled a comprehensive physics basis for the design of the ITER device (ITER, 1999; ITPA, 2007) which provides confidence in the ITER reference physics scenario for long-pulse burning plasma operation with $Q_{th} \geq 10$. However, the analysis of plasma performance requirements for tokamak FPPs, e.g., (Kikuchi, 1990; Sheffield, 1994; Jardin et al., 2000; Campbell et al., 2006; Zohm et al., 2017) has shown that improvements in plasma performance are typically required beyond the ITER baseline operating level to ensure that electricity can be generated at economically attractive rates. Moreover, while plasma β and density appear explicitly in Eq. (1), several additional elements of fusion physics are implicit in the analysis underlying this scaling, such as the ability to handle the power exhausted from the plasma while maintaining high energy confinement quality, high β and acceptable plasma contamination, the achievement of sufficiently high current drive efficiencies to maintain steady-state operation, the exploitation of auxiliary heating and current drive systems to maintain the current profiles required for high confinement and MHD stability, and the development of plasma control capabilities to ensure an acceptable disruption frequency (typically estimated to be in the range 0.1–1 per operational year) and to suppress β -limiting instabilities.

Integrating these requirements at the reactor scale is a central challenge for the development of a viable FPP. It is not yet possible to assign a quantitative probability to the likelihood that these various issues can be satisfactorily resolved at the level that will guarantee that the conceptual designs discussed here can be realized, particularly when critical elements of the solution, the achievement of a burning plasma with significant fusion gain and the characterization of its behavior, will be demonstrated only once ITER makes the transition to DT operation in the 2030s. Nevertheless, it is possible to briefly summarize the goals of the physics research program which leads via ITER to DEMO and ultimately to an FPP. Key issues which must be addressed to finalize the physics basis for a tokamak FPP are:

- **Steady-state operation:** several of the issues discussed below derive from the aim of achieving continuous, or “steady-state,” operation of the tokamak reactor core of a power plant. Continuous production of electricity does not rely on continuous tokamak operation, but, as discussed later, pulsed operation of the tokamak within a power plant framework tends to increase the physical size of the reactor, to provide a sufficiently large central solenoid (CS) to support long-pulse operation (e.g., several hours), and creates electromagnetic and thermal stresses in the reactor components, with implications for fatigue lifetime of the reactor core. In general, therefore, the majority of tokamak FPP conceptual designs are developed around a steady-state tokamak operating scenario.
- **Plasma confinement quality:** the operational baseline for ITER operation is the so-called H-mode confinement regime (Wagner et al., 1982), which has been studied extensively in present devices and for which an energy confinement time scaling (from present devices to ITER) has been developed (ITER, 1999). The predicted energy confinement time for a given device derived from this scaling is referred to as $\tau_{E,IPB98(y,2)}$. For any FPP design study, it is conventional to define the plasma confinement quality required to reach a specified level of fusion performance by a “confinement enhancement factor,” $H_{98(y,2)} = \tau_E / \tau_{E,IPB98(y,2)}$, where τ_E is the targeted energy confinement time at the reactor operating point. It is often found that FPP plasmas need to operate with $H_{98(y,2)} > 1$, i.e., with better confinement than predicted by the H-mode confinement scaling, a condition which can be achieved under some tokamak operating conditions, often referred to as an “advanced scenario.” This impacts, among other factors discussed below, the achievable values of normalized plasma pressure, β , and fusion gain, Q_{th} .
- **Plasma pressure:** it is recognized that increasing the plasma pressure towards the limits predicted by MHD stability theory allows the development of more compact and economic power plants. Theory determines the stability limits (Lao et al., 2021) associated with pressure in terms of β , the plasma thermal pressure normalized against the magnetic field pressure. The pressure limit for a given plasma regime can be characterized in terms of a figure of merit, $\beta_N = \beta(\%) \frac{a(m)B_t(T)}{I_p(MA)}$, which, for any given plasma configuration, has an absolute limit determined by so-called “ideal” MHD instabilities. The actual value of β_N which can be achieved depends on specifics of the plasma configuration, the electromagnetic interaction with the reactor wall and whether active control measures are implemented. Since, under optimum conditions (i.e., for average ion temperatures in the range 10–20 keV—see (Wade, 2021)), the fusion power scales as $P_f \propto \beta^2 B_t^4$, efficient use of the magnetic field (a major cost driver for the reactor core) requires operation at the highest β_N for the as-designed engineering configuration. This, in turn, implies optimization of the plasma scenario to maximize the confinement and the stability limit. As an example, for “conventional” tokamak reactors (i.e., with an aspect ratio, R/a , of ~ 3) it is typically assumed that β_N in the range 4–5 can be achieved, while for the spherical tokamaks discussed later (i.e., roughly with $R/a \leq 2$) β_N values of up to ~ 8 have been assumed.
- **Plasma density:** since the fusion reaction rate, and hence fusion power, depends quadratically on the density of reacting ions, operation at the highest achievable density is desirable. In addition, as noted above, the fusion power scales as $P_f \propto n_i^2 T_i^2$, with n_i the ion density T_i and the ion temperature, so that increasing the density to hold the ion temperature around an optimum value makes efficient use of the plasma β . Operation at high plasma density also facilitates handling of exhaust power by increasing impurity radiation for fixed impurity density (see below). In tokamaks, however, an operational limit on average plasma density is observed for H-mode operation which is characterized by the Greenwald density, $n_{GW} (10^{20} \text{ m}^{-3}) = I_p (\text{MA}) / \pi a^2 (\text{m})$ (Greenwald et al., 1988). Tokamak reactor concepts tend, therefore, to be designed around this point. As discussed later, stellarator plasmas do not suffer from this constraint, but appear to exhibit a density limit determined by the balance between plasma heating power and core plasma radiation (Sudo et al., 1990), allowing greater flexibility in determining the operational density.
- **Power exhaust:** the power deposited in the plasma by α -particles, together with the power injected to heat the plasma to thermonuclear conditions initially and then to contribute to current drive for steady-state operation (see below), must be exhausted via the internal surfaces facing the plasma. These consist of the “first wall” in the main (plasma) chamber of the vacuum vessel and the “divertor,” a specialized structure, generally in the lower region of the vacuum vessel (see discussion in the section “Key technologies” and Fig. 2 later), designed to handle the highest heat fluxes, but constrained by engineering limitations to $\sim 10 \text{ MWm}^{-2}$. The majority of designs rely on (water- or helium-cooled) tungsten as the plasma facing surface (PFC), as discussed by, e.g., Litnovsky et al. (2021). To limit tungsten sputtering, the temperature of the edge plasma in contact with these surfaces must be held below approximately 20 eV (Wu and Mszanowski, 1995). Otherwise, penetration of sputtered tungsten into the core plasma would produce an unacceptable level of contamination: the core concentration of tungsten must be held below $\sim 10^{-5}$ for acceptable fusion plasma performance, e.g., (Federici et al., 2001). The solution foreseen, both to limit the peak heat flux and to reduce the plasma temperature immediately in front of the material surfaces, is a development of that proposed for ITER in which low concentrations (1–2%) of impurities such as neon or argon are introduced (“seeded”) to radiate most (perhaps 80%) of the exhaust power to the entire first wall. Recent experimental results indicating that the width of the SOL may scale inversely with the poloidal gyro-radius (and therefore inversely with the plasma current) (Eich et al., 2011) imply an increase in the expected peak heat flux in the SOL, adding a further element to the challenge of developing a satisfactory solution for handling the power exhaust in a magnetic fusion reactor.
- **Non-inductive current drive:** the plasma current in tokamaks is normally produced inductively via the flux swing in the CS. To achieve steady-state operation, the plasma current must be driven fully non-inductively by a combination of currents generated in the plasma by the heating and current drive (H&CD) systems (Dumont, 2021) and by the “bootstrap” current (Sarazin, 2021). In order to minimize the recirculating electrical power required for the H&CD systems, plasma operating scenarios are designed to minimize the contribution required from the externally driven current by maximizing the bootstrap current, in effect by maximizing the achievable value of β_N . Tokamak reactor designs with $A \sim 3$ typically aim to generate 80–90% of the plasma

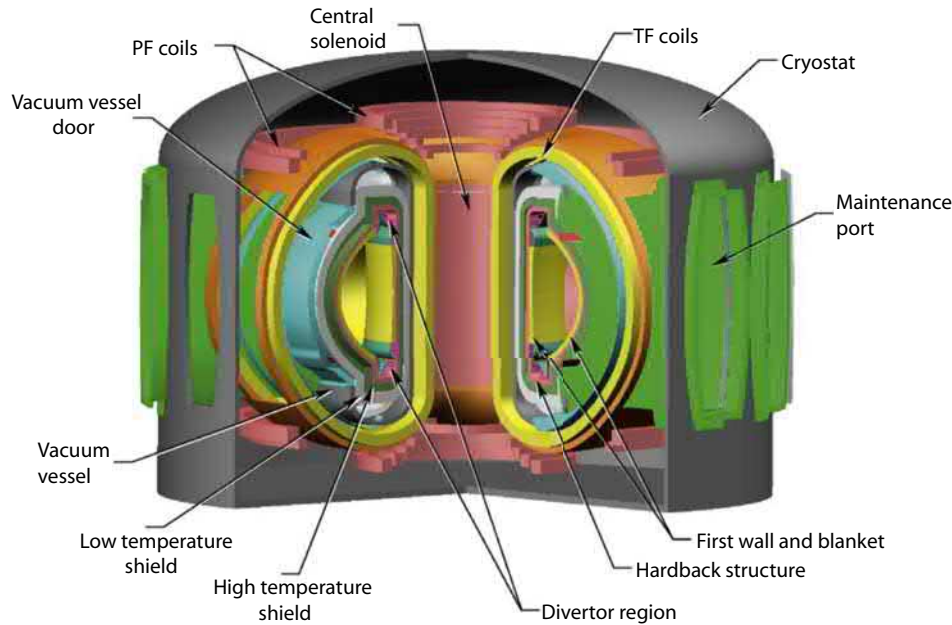


Fig. 2 A 3-D cutaway view of the fusion core of the ARIES-RS power plant study. Figure 1 from Najmabadi F and the ARIES Team (1997). Overview of the ARIES-RS reversed-shear tokamak power plant study. *Fusion Engineering and Design* 38: 3–25, [https://doi.org/10.1016/S0920-3796\(97\)00110-5](https://doi.org/10.1016/S0920-3796(97)00110-5). Reproduced with permission of publisher.

current via the bootstrap process, while concepts based on the spherical tokamak aim to achieve bootstrap current fractions of at least 90%. The remaining fraction of non-inductive current must be generated with high efficiency, to minimize the injected power, and must also provide some level of current profile control to maintain MHD stability. Indeed, for tokamak FPPs at conventional aspect ratio, the achievable value of Q_{th} is more typically determined by the current drive power requirement than by the thermal power balance.

- **Disruption avoidance:** Maintaining overall stability in the plasma against a range of possible MHD instabilities is fundamental, not only to sustaining high fusion performance, but also to protecting the in-vessel structures of a tokamak reactor: some classes of MHD instability can grow to a level at which they trigger a disruption, an event in which the total thermal energy of the plasma can be lost in milliseconds, the plasma current decays in tens or hundreds of milliseconds (at the reactor scale) and beams of high energy (~ 10 MeV) electrons can be generated which carry total currents of several mega-amperes (Lao et al., 2021). Such events produce high localized heat loads on PFCs, possibly causing localized melting, and high electromagnetic forces on in-vessel structures, which must be designed to withstand the most severe cases to ensure the integrity of the in-vessel components and of the reactor vessel itself. The high energy electrons can also cause local damage to PFCs when they are lost from the residual post-disruptive plasma. Since tokamak plasmas are usually vertically elongated, disruptions are often accompanied by loss of control of the plasma vertical position, leading to a vertical displacement event (VDE), which increases the electromagnetic forces on the vessel and in-vessel structures. Avoidance, control and mitigation of disruptions is a major element of the tokamak research program and concepts for application at the reactor scale will be tested in ITER, e.g., (Lehnen et al., 2018). Nevertheless, in addition to the potential damage to the tokamak internal structures, the loss of operational time would be a severe disadvantage to a fusion reactor if disruptions remained as common as in present devices. Therefore, development of operating scenarios and control techniques to reduce the frequency of disruptions to the range 0.1–1/year will be a key element in the development of the burning plasma research program via ITER and DEMO to an FPP. While tokamaks and RFPs are subject to the large scale MHD instabilities which lead to disruptions, it is a significant advantage of the stellarator approach to fusion energy production that stellarator plasmas are free from such events.
- **Burning plasma operation:** In present magnetic confinement experiments, the heating power to the plasma is dominated either by “ohmic heating” or, more commonly, by additional heating systems injecting particle beams and radiofrequency waves. ITER is designed to be the first magnetic confinement experiment in which 3.5 MeV α -particles produced by DT fusion reactions will dominate these conventional heating methods: in a $Q_{th} = 10$ plasma, the α -heating will be twice as large as the injected heating power, a regime referred to as a “burning plasma.” Many experiments have demonstrated that energetic particle populations such as α -particles behave as expected when present at sufficiently low densities to be treated as individual particles, e.g., (Gorelenkov and Sharapov, 2021). However, in the burning plasma environment, both nonlinear physics and collective effects may start to play a role, so that the behavior of a plasma dominated by α -heating might be more complex than is currently observed. This may be particularly important in the favored steady-state regime, the so-called “advanced scenario,” where optimization of confinement and stability can involve a nonlinear interaction between the pressure and current profiles,

e.g., (Kikuchi and Azumi, 2012). Thus, establishing robust plasma scenarios to maintain high fusion gain under the plasma conditions required to sustain steady-state operation will be a major element of the ITER program, providing a basis for their further development in DEMO-class devices in preparation for an FPP.

Concepts for stellarator and RFP reactors require the resolution of the majority of the challenges outlined above, although the detailed physics relating to some aspects of plasma behavior may differ from those in tokamaks. Stellarator plasmas differ notably from tokamak plasmas in several respects, in particular, in the ease of achieving steady-state operation, the absence of plasma disruptions and access to higher density operation than in equivalent tokamaks (Yamada, 2021). Since the confining magnetic field of the stellarator is produced by the magnetic coil configuration, there is no requirement to drive current in the plasma. Once a burning plasma is established by initial injection of auxiliary heating, the plasma equilibrium can, in principle, be sustained without the need for continuous application of H&CD power to maintain the magnetic equilibrium. As discussed in the section “Conceptual power plant designs”, several concepts have been developed for stellarator FPPs based on “ignited” plasmas, i.e., plasmas sustained purely by α -heating after an initial phase of auxiliary heating designed to reach thermonuclear conditions. The absence of a plasma current also frees stellarators from the MHD instabilities associated with the current, in particular disruptions. The nature of the density limit also allows more flexibility in the choice of operating point for stellarator plasmas. On the other hand, control of the 3-D stellarator equilibrium, to preserve closed magnetic surfaces across the entire plasma as the plasma β increases and to minimize losses of high-energy α -particles, is a significant challenge for high fusion gain plasmas.

A key advantage of RFP plasmas (Zuin, 2021) is operation at lower magnetic fields than is required by tokamak plasmas, therefore allowing exploitation of plasmas with substantially higher values of β , opening the path towards a fairly compact reactor core (see the section “Reversed field pinch concepts”). However, the existence of large-scale MHD modes in RFP plasmas, which currently limits the energy confinement to values which are perhaps two orders of magnitude lower than those in equivalent scale tokamak plasmas, represents a major challenge in progressing to reactor-relevant experiments. The contemporary RFP research program is therefore focused on the development of techniques for active control of the MHD modes and of the plasma current profile to suppress the mode amplitude and enhance the energy confinement.

Key technologies

While ITER will provide a significant testing ground for most key technologies required for DEMO and FPPs, it is recognized that, in some areas, significant further developments will be required to satisfy the increased demands of operation in the burning plasma environment of a power plant core, with, e.g., substantially higher neutron wall loads and plasma exhaust power, while maintaining the required electrical output with high availability. Fig. 2 illustrates a cutaway view of the (tokamak) fusion core of the ARIES-RS conceptual design (Najmabadi and The ARIES Team, 1997), which, in overall structural terms, is prototypical of toroidal FPP designs. A stellarator would have an inherently more complex 3-D geometry (whereas the tokamak and RFP are essentially toroidally symmetric), would lack a CS and would have a set of either modular or helical magnetic field coils in place of the TF/PF coil combination of the tokamak. Nevertheless, the overall structure, involving PFCs (first wall and divertor), breeding blanket, shielding structure, vacuum vessel, superconducting coils, thermal shield and cryostat, with access ports for operational (H&CD, Diagnostics) and maintenance systems, would follow the same pattern. An essential consideration that must be integrated into the design is the degree of shielding required to protect personnel and certain reactor components against the fusion neutrons: the breeder modules should provide a shielding factor of $\sim 10^2$ (in terms of neutrons m^{-2}) while the dedicated shield structure between the breeder modules and the vacuum vessel must further reduce the neutron flux by a factor of at least 10^4 . A total neutron attenuation factor of 10^6 should ensure survival of the superconducting magnets against radiation damage for the plant lifetime, but a radiological shield (typically concrete) surrounding the fusion core contributes a further attenuation factor of $\sim 10^2$ for personnel protection, see, e.g., (Najmabadi, 1999).

One detail in which the ARIES-RS configuration differs from many current devices and ITER is the use of both upper and lower divertors. This “double null X-point” (DNX) configuration, as opposed to the “single null X-point” (SNX) configuration of most major contemporary devices, appears in many FPP design concepts due to the use of highly shaped plasmas (i.e., high plasma elongation and triangularity), for which the β -limit, and therefore achievable fusion power density, is increased. The high shaping leads naturally to the development of up-down symmetric plasma configurations and favors the simultaneous formation of divertor magnetic configurations at the top and bottom of the plasma chamber.

Magnet systems are critical components in determining the overall performance of the fusion reactor and are also a significant factor in the cost of the FPP core. Numerous studies have been developed around the current “standard” low temperature superconductors (LTS) foreseen in ITER (Mitchell and Devred, 2017), with Nb_3Sn used in the TF and CS, where maximum values of magnetic field in the range 12–16 Tesla (T) are required, while NbTi is used in the PF coils, which operate at lower magnetic fields. The lack of space and shielding in the central core of spherical tokamaks, however, has hitherto prevented the use of LTS in the TS systems of conceptual FPPs based on the ST concept and therefore the use of water-cooled copper operating at room temperature, or helium-cooled aluminium operating at 30 K has been proposed, e.g., (Voss et al., 2002; Najmabadi et al., 2003). More recent studies, with somewhat higher aspect ratios of ~ 2 , have developed FPP concepts at higher toroidal field using LTS, e.g., Nb_3Al (Gi et al., 2015). Conceptual stellarator FPPs generally aim to limit the maximum field to ~ 12 T, allowing the use of NbTi , but, as discussed later, some design studies have targeted higher field values, in the range 14–16 T, and therefore make use of Nb_3Sn .

In RFPs, the significantly lower TF requirements allow NbTi to be applied, although in several cases a significant fraction (or all) of the magnet system operates at normal temperatures.

LTS is now considered as a mature technology for fusion devices, with several current tokamaks and stellarators in routine operation with both Nb₃Sn and NbTi magnet systems (Zohm, 2021; Yamada, 2021) and ITER's reactor-scale magnets under construction. Nevertheless, FPP designs would benefit from high temperature superconductors (HTS): operation in the region of 20 K (required to allow the target values of current density and magnetic field to be achieved), rather than at the 4.2 K typical for LTS, would avoid the need for a liquid helium cryogenic circuit and lower the cryogenic power requirements by a factor of several, reducing the recirculating electrical power fraction; in addition, the higher magnetic fields which can be achieved, in principle, with HTS would allow a reduction in the size of the FPP for a given power output, e.g., (Zohm, 2019). Although HTS have been used, e.g., in the current feeds for the ITER magnet system, and FPP concepts such as A-SSTR2 (Nishio et al., 2001) and ARIES-AT (Najmabadi et al., 2006) have been developed around the use of HTS such as Bi-Ag and YBCO (yttrium-bismuth-copper oxide) respectively, no experimental fusion device or model coil with comparable performance requirements has been demonstrated. More recently, the development of REBCO (rare-earth-barium-copper oxide) tape with the potential for high current density and magnetic field has stimulated the conceptual development of proposals towards more compact tokamak reactors with HTS magnet systems (Sorbon et al., 2015; Sykes et al., 2018). Nevertheless, substantial R&D programs will be required to develop the performance level required of HTS magnets for FPPs based on this technology and to reduce the superconductor material cost, which is an order of magnitude greater than that for the LTS material currently in use (Bruzzone et al., 2018).

As discussed by Litnovsky et al. (2021) (see also (Stork and Zinkle, 2017) and related papers), reactor materials are subject to very significant performance demands, in terms of heat loads, neutron exposure and operational temperature range. Plasma facing materials on the first wall and divertor targets are subject to high stationary heat loads (ranging from $\sim 1 \text{ MWm}^{-2}$ in the first wall region to $\sim 10 \text{ MWm}^{-2}$, or beyond, at the divertor target plates), bombardment by plasma particles (causing sputtering and erosion of the surface), and neutron fluxes which correspond to power fluxes of several MWm^{-2} (the precise value depends on the assumed plasma fusion performance), leading to a range of radiation-induced effects and transmutation reactions with potential formation of helium bubbles. In addition, plasma MHD instabilities may give rise to significant electromagnetic loads, as well as transient heat loads which can produce localized melting, though, as noted previously, the frequency of such events in a reactor must be limited to < 1 per year.

There is no ideal solution to this problem, but the majority of reactor designs have selected tungsten or tungsten composites for the divertor targets, the regions exposed to the highest heat fluxes and plasma particle fluxes. Tungsten is also often chosen for the first wall. Alternatively, on the assumption that means of controlling the plasma-wall interaction will be available at the time of FPP construction, the wall material is simply taken to be the same as the structural material for the breeding blanket. Liquid metals, such as lithium and tin, are under investigation as possible solutions to the challenges of providing resilient plasma facing components for fusion reactors, but R&D in present programs is at an early stage (see, e.g., (Mazzitelli et al., 2011; Ono et al., 2017)) and very few conceptual FPPs have been proposed which exploit liquid metals as the plasma facing materials.

Structural materials for in-vessel components, in particular the breeding blanket, shielding structure and divertor, are exposed to many of the same issues as plasma facing components, although, in general, the plasma facing surface is typically considered distinct and designed to handle the specific issues associated with heat and particle fluxes from the plasma. Nevertheless, these structures must absorb the fusion energy produced by the plasma and provide it to the heat exchangers at a sufficiently high temperature to allow a high thermal conversion efficiency in the generation of electricity. The most favored, and most highly developed, materials for this application are the reduced-activation ferritic-martensitic steels (RAFM), e.g., (Tanigawa et al., 2017), which are capable of satisfying the FPP operational specifications, as well as meeting safety and environmental requirements (Taylor, 2021). However, the limited operational temperature range of the present generation of these steels, typically 350–550 °C, constrains the achievable thermal conversion efficiency, though proposals have been advanced to raise the coolant outlet temperature by inserting thermal insulation between the breeder and the ferritic steel structure, allowing a coolant outlet temperature of 700 °C (Najmabadi et al., 2003).

Research is underway to develop a further generation of advanced RAFM steels and also oxygen dispersion strengthened (ODS) steels which would have the potential to offer higher thermodynamic efficiency, superior resistance to neutron damage, improvements in reactor operational lifetime and reliability, and reduced construction costs (Zinkle et al., 2017). Alternatives to the RAFM approach have also been investigated as possible FPP solutions: the use of both vanadium alloy and silicon carbide composites (SiC/SiC) has been developed within several FPP concepts, including the ARIES power plant study program, e.g., (Najmabadi et al., 1997). When combined with appropriate breeding and cooling materials, such approaches could allow significantly higher thermal conversion efficiencies to be achieved while retaining attractive safety and environmental properties.

Considerable R&D is required to fully characterize the impact of neutron irradiation levels expected in an FPP and to demonstrate that changes in properties such as thermal conductivity and structural integrity and strength are acceptable, e.g., (Van der Schaaf et al., 2006). The achievable lifetime of near-plasma components will be a key issue: FPP first wall neutron loads are typically in the range $1\text{--}6 \text{ MWm}^{-2}$, i.e., $0.44\text{--}2.7 \times 10^{18} \text{ neutrons m}^{-2} \text{ s}^{-1}$, and the annual neutron fluence of $1\text{--}6 \text{ MW year m}^{-2}$ corresponds (for steels) to an annual displacement per lattice atom (dpa) of $10\text{--}60 \text{ dpa}$ for plasma facing surfaces. The tolerable irradiation lifetime of structural materials is typically assumed to lie in the range $100\text{--}150 \text{ dpa}$, implying a requirement to exchange near-plasma components (i.e., the breeding blanket, including the first wall, and divertor) every 3–10 years, a period which is significantly shorter than the assumed lifetime of the power plant. This has an impact on both the lifetime costs and availability of an FPP.

The key issue of tritium breeding technology is intimately connected with the choice of structural materials, the design of the structures facing the plasma and the blanket coolant scheme. Boccaccini and Day (2021) discuss the present considerations in

relation to reactor breeder blankets and the R&D program via ITER designed to identify the most promising options—despite the critical nature of tritium breeding from lithium in the fuel economy of an FPP and the implications for fusion as a viable source of large-scale energy production, ITER will be the first opportunity to test breeding module concepts as an integrated technology. Abdou et al. (2021) have recently completed a detailed analysis of the tritium breeding ratio (TBR) required in a fusion reactor, indicating that a target value in the range $1.05 \leq \text{TBR} \leq 1.15$ is likely to be required, but that extensive further physics and technology R&D will be necessary to ensure that such values are attainable in the reactor environment. Additional challenges will need to be addressed to achieve such values of TBR in the spherical tokamak environment, where the breeding blanket is limited to the outboard side of the torus, and in stellarator FPPs, where the limited clearance between the plasma and superconducting coils prevents a uniform coverage of tritium breeding modules, as discussed in the section “Conceptual power plant designs”.

Within the framework of the breeding blanket R&D, the coolant scheme is a major consideration, since it is closely coupled to the design of the breeding blanket, has an impact on the extraction of high grade heat for electricity generation, i.e., the thermal conversion efficiency, affects the recirculating electrical power fraction (with helium particularly demanding in this respect) and also has safety implications, e.g., via the consequences of water leaks into the vacuum vessel, due to the reactivity of liquid metals with steam or air, or permeation of tritium into water or gaseous coolants. Assumptions related to integrated breeder blanket designs which satisfy the requirements on TBR while allowing for efficient heat extraction at high temperature consistent with the operating temperature limits of structural and breeder materials are therefore a significant element in the analysis of overall FPP performance.

Extension of H&CD technology to the reactor scale will be an essential element of the R&D activity to ensure that the required level of plasma fusion performance can be achieved and sustained in steady-state tokamak FPPs, e.g., (Inoue et al., 2021). Developments required for H&CD applications in ITER will advance the technologies in this area towards meeting the requirements for DEMO and FPPs. Nevertheless, substantial R&D will be necessary to ensure that the key H&CD systems considered viable in the reactor environment will achieve the required level of electrical conversion efficiency while providing the specified power continuously for periods, say, of order a year, implying significant progress in system reliability. Since radiofrequency sources can be sited remotely from the fusion core, RF system performance will be less sensitive than that of neutral beam injection systems to the high radiation environment of the FPP burning plasma, where the first wall neutron flux will be a factor of several above that in ITER and the annual neutron fluence will be approximately two orders of magnitude greater. The quasi-optical propagation of radiowaves in the electron cyclotron frequency range ($\sim 100\text{--}200$ GHz) will be particularly advantageous, while the close coupling to the plasma required by ion cyclotron radiofrequency (ICRF) H&CD (operating in the frequency range $\sim 100\text{--}200$ MHz) will present challenges in ICRF antenna design.

Fig. 3 provides an illustration of the impact of assumptions on plasma performance and reactor technology on the scale of tokamak FPPs. The figure shows the poloidal cross-section of a series of design concepts developed within the EU's fusion power plant conceptual study (PPCS) program, with the ITER cross-section for comparison (Maisonnier et al., 2007). The largest of the devices, models A, AB and B, assume somewhat better (by $\sim 30\%$) plasma performance parameters than would be derived from

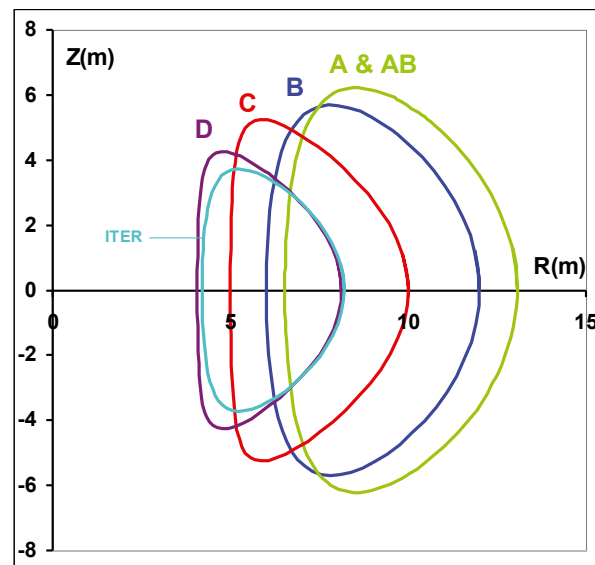


Fig. 3 Variation of poloidal plasma cross-sections of the conceptual power plant concepts developed in the EU PPCS studies, with the ITER plasma cross-section shown for comparison: R is the major radius coordinate and Z the vertical height. As the physics and technology assumptions were made more ambitious, the sequence of devices varied from A/AB to D (Note that device B has been slightly increased in size to renormalize the net electrical output to 1.5 GW, providing a common basis for the scale comparison). Figure 1 from Maisonnier D, Campbell D, Cook I et al. (2007) Power plant conceptual studies in Europe. *Nuclear Fusion* 47: 1524–1532, <https://doi.org/10.1088/0029-5515/47/11/014>. Reproduced with permission of copyright holder (IAEA).

the ITER Physics Basis (ITER, 1999). However, the “conventional” plasma regime assumed limits the plasma β and the bootstrap current fraction, requiring a high level of auxiliary heating to achieve 100% non-inductive current drive. The use of “conventional” blanket technology, i.e., RAFM structural material, which limits the coolant temperature and therefore the thermal conversion efficiency, together with the significant fraction of recirculating electrical power, necessitates a high fusion power production (5 GW in the case of model A) to provide a net electrical output of 1.5 GW.

The large dimensions of these devices are also driven by constraints on the divertor power loading—indeed, there is a complex interplay between plasma confinement, H&CD power requirements and acceptable divertor heat fluxes. If more ambitious targets are allowed for fusion plasma performance, i.e., an “advanced” plasma scenario, and β is allowed to rise accordingly, the bootstrap current contribution increases, reducing the auxiliary heating requirement. In addition, models C and D make use of more advanced blanket concepts, either a He-cooled RAFM structure with SiC_f/SiC insulators and an LiPb breeder (model C), or an LiPb self-cooled breeder blanket with a SiC_f/SiC structure. Both approaches increase the assumed thermal conversion efficiency and reduce the electrical pumping power. The “advanced” technology assumptions of the two smaller models include an increase in the electrical efficiency of the H&CD systems (0.7 vs. 0.6 for the larger devices), also contributing to a reduction in the recirculating electrical power fraction. Not surprisingly, the projected COE of these devices decreases by almost a factor of two between model A and model D.

Conceptual power plant designs

Tokamak/Spherical Tokamak concepts

Detailed studies of concepts for tokamak fusion power plants have been carried out internationally in recent decades. The US ARIES program has developed a series of concepts, comparing pulsed and steady-state tokamak designs, analyzing also the spherical tokamak concept and, as discussed later, stellarator and reversed field pinch configurations. FPPs based on both the tokamak and ST concepts have also been developed by the EU and Japanese fusion programs. Table 1 provides a list of principal parameters of several tokamak and spherical tokamak FPP concepts developed during this period.

The PULSAR-I concept (Bathke et al., 1994) illustrates an FPP developed around a tokamak core having a pulse length of ~ 2.5 h. Although an earlier proposal than the steady-state concepts discussed below, it is nevertheless relevant, since some contemporary options for a DEMO, e.g., (Federici et al., 2017) propose a pulsed device as a conservative approach, potentially allowing more rapid construction of DEMO as soon as ITER results provide an adequate physics and technology basis. The need to limit cyclic fatigue in the TF coil, which then constrains the maximum value of toroidal field at the coil inner leg, is a significant factor in determining the large major radius of the PULSAR-I device. The resultant scale of this device might be more typical of a first generation FPP, since the physics assumptions incorporated in the design are more characteristic of those underlying ITER, rather than of the more advanced assumptions, in terms of plasma confinement and β_N , incorporated in the steady-state FPPs listed in the table. The scale and more conservative technology applied in the design of the PPCS-A, PPCS-AB and PPCS-B concepts illustrated in Fig. 3 could also be considered to be in this category, though they are designed around steady-state tokamak operation. On the other hand, the use of SiC_f/SiC as a first wall, blanket and shield material in PULSAR-I is likely more appropriate to later FPP generations.

A noteworthy aspect of the PULSAR-I physics operation is that it is developed around “ignited” operation, i.e., with no additional H&CD after an initial application to heat the plasma to the state where fusion reactions take over and fully maintain the burn point. In contrast, the majority of tokamak FPP designs foresee the tokamak plasma operating as a “power amplifier” with a typical thermal fusion gain in the range 20–50. To address the issue of continuous electricity production, essential for a “baseload” power plant, the design makes use of the thermal storage capacity of the neutron shield to supply thermal power to the heat exchangers for a period of 200 s while the plasma is terminated, the tokamak CS “recharged” and the plasma reheated to ignition. Nevertheless, as a result of the design elements associated with pulsed operation, the projected COE associated with PULSAR-I was found to be $\sim 30\%$ higher than for equivalent FPP concepts developed around steady-state tokamak operation.

The PPCS-C (Maisonier et al., 2007), ARIES-AT (Najmabadi et al., 2006) and A-SSTR2 (Nishio et al., 2001) devices represent designs based around “conventional,” i.e., moderate to high, aspect ratios. Advanced plasma regimes, to allow access to high values of β_N , were combined with assumptions on future technological developments to produce moderately compact devices operating in steady-state. There are, nevertheless, significant differences in the technology exploited. For example, in PPCS-C, the TF magnets used the well-established Nb₃Sn superconducting strand, while the A-SSTR2 designers proposed to fabricate the TF windings from an HTS strand, Bi-Ag alloy, operating at 20 K and allowing a substantially higher field (23 T) at the TF coil inner leg than Nb₃Sn. The design of the ARIES-AT TF magnet was developed around the YBCO HTS operating at 77 K, but at a moderate value of TF field, since the concept incorporated advanced physics assumptions, including (self-consistently) operation at a particularly high value of β_N . In addition, while the PPCS-C design assumed that the principal in-vessel structures were fabricated from a reduced activation ferritic-martensitic steel, EUROFER, which has been the subject of extensive R&D, the ARIES-AT and A-SSTR2 designs made use of SiC_f/SiC, allowing a higher blanket coolant temperature and therefore higher thermal efficiency. A further unusual feature of the A-SSTR2 design is the elimination of the central solenoid, so that the plasma current ramp-up would be performed fully non-inductively—this represents the end-point of conceptual developments in which the central solenoid is “slimmed” to reduce the radial build of the tokamak, requiring an increasing non-inductive contribution to the current ramp-up. Simulations of the current ramp-up for A-SSTR2 showed that the sequential application of RF waves and neutral beam injection could allow the operating current of 12 MA to be achieved in ~ 22 h.

Table 1 Principal parameters of several tokamak and spherical tokamak power plant studies.

Parameter	<i>PULSAR-I</i> (Bathke et al., 1994)	<i>PPCS-C</i> (Maisonnier et al., 2007)	<i>ARIES-AT</i> (Najmabadi et al., 2006)	<i>A-SSTR2</i> (Nishio et al., 2001)	<i>STPP</i> (Voss et al., 2002)	<i>ARIES-ST</i> (Najmabadi et al., 2003)
Toroidal field (axis) (T)	6.7	6.0	5.8	11	1.8	2.1
Plasma current (MA)	14.2	20.1	13	12	31	28.4
Major, minor radius (m)	9.2, 2.3	7.5, 2.5	5.2, 1.3	6.2, 1.5	3.42, 2.44	3.2, 2.0
Aspect ratio (R/a)	4.0	3.0	4.0	4.1	1.4	1.6
κ_{95} , δ_{95}^a	1.6, 0.35	1.9, 0.47	2.1, 0.8	1.8, 0.4	3.2, 0.55	3.4, 0.64
q_{95}^a	3.5	4.1	≤ 4	5.2	13	11
β , β_N	2.8%, 3.0	5.4%, 4.0	9.2%, 5.4	2.9%, 4.0	59%, 8.2	54%, 7.4
Plasma volume (m ³)	1540	1750	350	470	1175	840
Auxiliary power (MW)	0	112	35	85	50	31
Fusion power (MW)	2030	3410	1720	4000	3260	2860
Total thermal power ^b (MW)	2310	4000	1900	5000	4090	3370
Net electrical power (MW)	1000	1450	1000	2550	1220	1000
Av. neutron wall load (MWm ⁻²)	1.1	2.2	3.3	6.0	3.5	4.1
Operation mode	2.5 h Pulse	Steady-state	Steady-state	Steady-state	Steady-state	Steady-state
Thermal cycle	Brayton	Brayton	Brayton	Brayton	Rankine	Brayton
Thermal efficiency, η_{th}	0.49	0.42	0.59	> 0.50	0.43	0.45
Recirculating power fraction (1/ Q_{elec})	0.1	0.15	0.14		0.30	0.34
First wall material		ODS-s ^c	SiC _f /SiC		C	RAFM ^d
First wall structure	SiC _f /SiC	RAFM ^d	SiC _f /SiC	SiC _f /SiC	RAFM ^d	RAFM ^d
First wall coolant	He	He	PbLi ^e	He	He	He
Breeding blanket structure	SiC _f /SiC	RAFM ^d	SiC _f /SiC	SiC _f /SiC	RAFM ^d	RAFM ^d
Breeding material ^f	Li ₂ O	PbLi ^e	PbLi ^e	Li ₂ TiO ₃	Li ₂ O	PbLi ^e
Breeding blanket coolant temp. ^g (K)	< 1200	750/990	1370	1170	870	800/990
Breeding blanket coolant	He	He/PbLi ^e	PbLi ^e	He	He	He/PbLi ^e
Shield blanket material	SiC _f /SiC	RAFM ^d	SiC _f /SiC	TiH ₂	RAFM ^d	RAFM ^d
Vacuum vessel structure		RAFM ^d	RAFM ^d		RAFM ^d	RAFM ^d

^aIn plasmas with a magnetic separatrix, it is common to characterize the plasma equilibrium by quoting the elongation κ , triangularity, δ , and safety factor, q , at the magnetic flux surface which encompasses 95% of the poloidal magnetic flux in the plasma (i.e., just inside, but close to, the separatrix).

^bTotal thermal power exceeds the fusion power due to the exothermic tritium-breeding reactions.

^cODS-s: oxygen dispersion strengthened (reduced activation) steel.

^dRAFM: reduced-activation ferritic-martensitic steel.

^eThe eutectic form referred to as PbLi varies in composition across the literature: in the EU literature, the composition Li_{15.8}Pb_{84.2} is favoured (also referred to as Pb-15.8% at Li), while in the US literature the composition Li₁₇Pb₈₃ is favoured.

^fLi₂O and Li-ceramic (e.g., Li₂TiO₃) breeder materials generally require a neutron multiplier (e.g., Be).

^gOutlet temperature of coolant—temperature of breeder material may be higher.

As discussed in the section “Requirements for commercial fusion power”, a key issue critical to the credibility of fusion as a major energy resource, is the frequency with which PFCs and breeding blankets must be replaced (and the duration of the associated “maintenance” activity), since this is assumed to be the determining factor in the power plant availability. As well as the impact on the annual operating costs of the FPP, this also influences the volume of waste for disposal. It would be preferable to match the PFC and blanket replacement cycles. However, this is not always possible. In the PPCS-C design, for example, the low neutron wall loading and the use of RAFM steel for the blanket structure allowed a 5-year interval between blanket replacements (assuming an irradiation lifetime of ~ 150 dpa), while the divertor PFC lifetime was estimated to be 2 years. In the ARIES-AT concept, it was assumed that the lifetime of the blanket structure was limited by 3% burnup of SiC, leading to an estimated 4-year replacement cycle (4 full power years (FPY), to be precise). It was also assumed that the divertor PFC replacement cycle could be matched to this. The very high neutron wall load in the A-SSTR2 design limited the lifetime of the blanket to 2 FPY, and it was assumed that the blanket could be replaced within 2 months on a 2-year cycle. Clearly an important aspect of sustaining high availability is the capability of replacing major in-vessel components rapidly on a regular cycle and this is therefore a significant consideration in the design of the fusion power core and of the maintenance (remote handling) technology.

The scale of the devices listed in Table 1, in terms of the electrical power output, is large, typically of order 1 GWe or greater, and this reflects the relationship between COE and unit size given in Eq. (1). The combination of advanced physics and technology in the ARIES-AT and A-SSTR2 concepts implies that they might be considered as long-term developments of the tokamak concept

towards an optimized FPP, whereas the PPCS-C design, relying on near-term technological readiness and physics assumptions which are less ambitious than those of ARIES-ST or A-SSTR2, might be achievable at an earlier phase of FPP implementation.

The ST approach to fusion energy production aims to develop more compact, and possibly cheaper, solutions for the fusion power core. Physics analysis also inspires this approach since both experimental results and theoretical studies, e.g., (Peng and Strickler, 1986; Menard et al., 2016; Harrison et al., 2019; Kaye et al., 2019) indicate that low aspect ratio favors both improved confinement and enhanced MHD stability in comparison with conventional tokamaks (aspect ratio $\sim 3\text{--}4$). Experimental progress in STs is more limited than in the conventional approach, but the potential physics and cost advantages have motivated a study of FPP concepts at low aspect ratio, as illustrated by the STPP (Voss et al., 2002) and ARIES-ST (Najmabadi et al., 2003) designs listed in Table 1. As is evident from the physics parameters listed in the table, the fusion core of an ST power plant would be expected to operate at substantially higher values of β and β_N than conventional tokamak designs, but at values that are, nevertheless, plausible on the basis of the relevant experimental and theoretical background.

An important consequence of operation at such high values of β_N is that the pressure driven currents in the plasma (primarily bootstrap current, but other physics processes also contribute) generate 95–99% of the total plasma current, implying that only a small fraction of the total current must be driven by the auxiliary heating systems to support steady-state operation. This would resolve a key challenge of steady-state operation in the ST configuration: due to the large fraction of trapped particles typical of low aspect ratio plasmas, the achievable current drive efficiency with auxiliary heating systems is significantly lower than in plasmas at conventional aspect ratio. A viable scenario for plasma start-up and (predominantly non-inductive) current ramp-up must also be developed for such low aspect ratio ST FPP concepts, since the limited space on the inboard side of the plasma does not allow the inclusion of a central solenoid. Experimental R&D is underway to demonstrate this capability, e.g., (Bongard et al., 2019; Ono et al., 2019). Satisfactory dissipation of the plasma exhaust power flowing to the divertors during high fusion power operation at low aspect ratio must also be addressed. Redistribution of the exhaust power via impurity radiation, as is foreseen for conventional tokamaks, will probably form the basis of the solution, but maintaining the required radiated power fraction while sustaining the necessary level of fusion performance will be even more challenging. Therefore, novel magnetic configurations for the divertor geometry to mitigate divertor heat loads in the ST configuration are under study, e.g., (Harrison et al., 2019).

Many elements of the technology used in these concepts are shared with the conventional tokamak designs. However, as indicated in the section “Key technologies”, an important difference is that the low aspect ratio imposes space constraints on the inboard side of the plasma, limiting the achievable neutron shielding factor and preventing the use of magnets fabricated from LTS. The water-cooled copper magnets in the TF system then make a significant demand on recirculating electrical power and this can be seen in the table. As noted in the section “Key technologies”, use of a somewhat higher aspect ratio, ~ 2 , would allow the use of LTS and, if HTS technology such as REBCO eventually confirms its potential, this could have a significant impact on the design of FPPs with an ST fusion core (Sykes et al., 2018). The space constraints on the inboard side of the plasma restrict the use of a breeding blanket to the outboard side, but nuclear analysis of the two design concepts shown in the table indicated that, with neutron multiplication and adequate ${}^6\text{Li}$ enrichment, an acceptable TBR of ~ 1.1 could be achieved. The high neutron wall loads, $3.5\text{--}4\text{ MWm}^{-2}$, imply a short replacement time for the blankets, and indeed much of the structure in close proximity to the plasma: for STPP the blanket lifetime was estimated as ~ 2 FPY, while the ARIES-ST designers considered a lifetime of almost 3 FPY possible.

Stellarator concepts

Although stellarator power plant concepts were studied during the early decades of magnetic fusion research, the rapid development of plasma performance in tokamaks led to a neglect of the stellarator line for several decades. Stimulated by conceptual improvements in the stellarator magnetic configuration, which were accompanied by promising developments in plasma performance during the 1980s and 1990s, the recognition that the potential for intrinsically steady-state plasma operation and disruption-free plasmas represented significant advantages for a power plant reinvigorated physics and engineering analysis of the stellarator as the core of an FPP. The term “stellarator” encompasses several subtly different magnetic configurations (Yamada, 2021). The examples of stellarator FPP concepts listed in Table 2, therefore, illustrate options developed around the three most promising configurations: the torsatron/heliotron (FFHR-d1A), the helias (HSR4/18, HSR5/22) and the quasi-axisymmetric stellarator or QAS (ARIES-CS). The relative merits and challenges associated with each of these configurations are detailed by Yamada (2021).

A key challenge in the design of stellarator FPPs is that the higher order components of the confining magnetic field, which is produced entirely by currents in coils external to the plasma, decay rapidly with distance from the coils to the plasma. This implies that the radial space available for breeding and for shielding of the coils is limited. This is particularly problematic on the inboard side of the plasma, due to the helical shape of the magnetic flux surfaces. Since it is nevertheless necessary to provide sufficient radial space (ideally $\sim 1.3\text{ m}$) for the breeding and shielding blankets between the plasma and coils, stellarator reactor designs tend to be rather larger in scale than tokamaks with equivalent net electrical power output. Of course, the plasma scale required to achieve significant fusion gain—the stellarator concepts listed in the table are all designed to operate in “ignited” mode, i.e., with infinite gain—and fusion power production must be integrated into this engineering constraint. In addition, the complex shape of the coils required to produce the helical configuration can generate large electromagnetic forces on the coil system, requiring significant mechanical restraint within a limited space. Finally, it should be noted that the need to facilitate maintenance access to in-vessel components within the helical configuration leads to a significant increase in the complexity of the fusion core components compared to those in tokamak reactor concepts. The FPP studies illustrated in the table have exploited advances in the conceptual design of helical systems to address these problems and to allow practical solutions to the reactor constraints.

Table 2 Principal parameters of several stellarator power plant studies.

Parameter	FFHR-d1A (Sagara et al., 2014, 2017)	HSR4/18 (Wobig and Wagner, 2005; Beidler et al., 2001)	HSR5/22 (Wobig and Wagner, 2005; Beidler et al., 2001)	ARIES-CS (Najmabadi et al., 2008; Lyon et al., 2008)
Toroidal field (at coil axis) (T)	4.7	5	4.75	5.7
Plasma current (MA)	0	0	0	4 ^a
Major, (average) minor radius (m)	15.6, ~2.5	18, 2.1	22, 1.8	7.75, 1.70
Aspect ratio (R/a)	~6	8.5	12.2	4.6
$\beta_{0r} < \beta >$	8%, —	13.2, 4.2%	15.7, 4.24%	—, 6.4%
Plasma volume (m ³)	1421	1570	1407	440
Auxiliary power ^b (MW)	0	0	0	0
Fusion power (MW)	3000	3000	3000	2440
Total thermal power ^c (MW)		~3480 ^d	~3480 ^d	2920
Net electrical power (MW)	~1000	~1123 ^d	~1123 ^d	1000
Av. neutron wall load (MWm ⁻²)	1.5	≤1.2	≤1	2.6
Operation mode	Steady-state	Steady-state	Steady-state	Steady-state
Thermal cycle				Brayton
Thermal efficiency, η_{th}		~0.35 ^d	~0.35 ^d	0.43
Recirculating power fraction (1/Q _{Elec})		0.08 ^d	0.08 ^d	0.2
Winding magnetic polarity, L	2	N/A	N/A	N/A
Toroidal field periods, M	10	4	5	3
Number of modular coils	N/A	40	50	18 ^e
First wall material				RAFM ^f
First wall structure		RAFM ^f	RAFM ^f	RAFM ^f
First wall coolant		He or H ₂ O	He or H ₂ O	He
Breeding blanket structure	RAFM ^f	RAFM ^f	RAFM ^f	RAFM ^f
Breeding material ^g	Li ₂ TiO ₃	Li ₂ TiO ₃ or PbLi ^h	Li ₂ TiO ₃ or PbLi ^h	PbLi ^h
Breeding blanket coolant temp. ⁱ (K)		720 or 600	720 or 600	730/1010
Breeding blanket coolant	H ₂ O	He or H ₂ O	He or H ₂ O	He/PbLi ^h
Shield blanket material ^{j,k}	WC/RAFM ^f + B ₄ C	RAFM ^f	RAFM ^f	WC + RAFM/RAFM ^f
Vacuum vessel structure		SS	SS	

^aPlasma current is driven by bootstrap effect and depends on plasma pressure profile.^bAuxiliary power during stationary fusion burn; 20–100 MW of auxiliary power is used transiently to initiate plasma and to reach burn point, depending on reactor design.^cSee Table 1, footnote b.^dValues based on assumption of blanket power gain of 1.2 and of thermal conversion efficiency of 0.35.^eMagnetic configuration also includes 8 poloidal field coils for equilibrium shape control.^fSee Table 1, footnote d.^gSee Table 1, footnote f.^hSee Table 1, footnote e.ⁱSee Table 1, footnote g.^jWC/RAFM + B₄C: inboard shield is WC and outboard shield is RAFM + B₄C.^kWC + RAFM/RAFM: WC + RAFM shield is used in areas with limited radial space and RAFM elsewhere.

The Force-Free Helical Reactor, FFHR-d1A (Sagara et al., 2017), is a member of a family of similar reactor-scale devices which has been developed from the heliotron configuration of the Large Helical Device (LHD) (Iiyoshi et al., 1999; Yamada, 2021), in which two helical coils, carrying currents in the same direction, encircle the toroidal plasma. The magnetic equilibrium also requires a small number (four in the case of FFHR-d1A) of poloidal field coils to complete the equilibrium configuration. The resultant magnetic equilibrium naturally generates two helical divertors on either side of the plasma. The “basic” design option of this device was based on ITER technology wherever possible, e.g., the superconducting coils are fabricated from Nb₃Sn, the breeding blanket concept was based on the water-cooled ceramic breeder concept under development by Japan for testing in ITER and the divertor targets were assumed to exploit the water-cooled tungsten technology being developed for ITER. A technologically “challenging” option making use of HTS (e.g., REBCO) magnets, a molten salt (FLiNaBe/metal powder mixture) breeding blanket and a liquid metal (Sn) divertor target was also studied. A notable feature of this and the other stellarator concepts listed is that the average neutron wall loading is modest compared to the majority of tokamak concepts discussed previously, implying a significantly longer period (possibly ~10 years) between blanket replacements than in the tokamak reactors. The volume of blanket to be replaced is, however, larger. Nevertheless, as in tokamaks, regular replacement of high power handling PFCs would be required on a shorter timescale than the blanket replacement.

The Helias (helical advanced stellarator) reactor concepts, HSR4/18 and HSR5/22 (Beidler et al., 2001; Wobig and Wagner, 2005), are developments of the W7-X optimized advanced stellarator (Grieger et al., 1992; Yamada, 2021). HSR5/22 is an analogue of the W7-X device, with five toroidal periods, while HSR4/18, has four toroidal periods, resulting in a more compact device. In addition to the physics optimization designed to minimize several limitations of the conventional stellarator,

a particularly attractive feature of this concept is the use of modular (NbTi) superconducting coils to produce the required magnetic equilibrium. The coils could therefore be manufactured off-site, rather than onsite, as in the heliotron concept. The magnetic configuration also produces an associated “island divertor” via a series of magnetic islands generated at the plasma edge, thereby generating a means of power and particle exhaust. Alternative blanket concepts studied for the HSR devices are a helium-cooled ceramic breeder or an LiPb breeder with helium- or water-cooling. A self-cooled LiPb option was also considered. As in the FFHR design, the blanket segmentation was arranged to allow replacement of the blanket without disturbance of the superconducting coil system.

The final concept illustrated in the table, ARIES-CS (Najmabadi et al., 2008), is based on the concept of quasi-axisymmetry (QAS), see, e.g., (Yamada, 2021), an alternative form of plasma optimization to that applied in the Helias concept. The magnetic configuration, with three toroidal periods, is produced predominantly by a set of modular coils, but a small plasma current (4 MA) is also exploited and the equilibrium was designed to generate this via the pressure-driven, i.e., bootstrap, current in the plasma to avoid the need for an external source of steady-state non-inductive current drive, as is required in tokamaks. The aim of this approach was to allow the design of a more compact reactor design, on a scale comparable to typical tokamak reactor concepts, targeting a COE comparable to that derived for tokamak FPPs. Optimization of the plasma configuration aimed to obtain a smaller aspect ratio than possible with the heliotron and Helias designs while simultaneously minimizing the α -particle losses, a key physics issue for helical systems. The compact size of the device, resulting in a peak magnetic field of ~ 15 T at the coils, necessitated the use of Nb₃Sn superconductor operating at 4 K, and this implied a significantly more complex manufacturing process than for tokamak TF and CS coils. The breeding blanket geometry is also complex and in areas with insufficient space between the plasma and the coils, a tapered breeding blanket with an additional WC shielding blanket was employed. The breeding blanket used a self-cooled PbLi breeder with an additional He circuit to cool the RAFM steel structure. Due to the complex plasma equilibrium, the peak neutron wall loading is a factor of two greater than the average figure of 2.6 MWm^{-2} given in the table. This limits the blanket lifetime to ~ 3 FPY. A further challenge relates to the divertor concept and control of PFC heat loads: the divertor configuration approximates a toroidally threefold version of the tokamak double-null divertor configuration, but the design was not fully optimized to ensure PFC power handling constraints were entirely respected.

Reversed field pinch concepts

Interest in the development of reactor concepts around the reversed field pinch (RFP) has been limited by the poor plasma energy confinement associated with the dynamo action which generates the magnetic configuration. As discussed by Zuin (2021), RFP research is addressing this limitation through the use of active control of the current profile and of MHD instabilities. Nevertheless, several FPP studies using an RFP fusion core have already been developed, motivated by the possibility of operation at high values of β (several tens of percent), as already demonstrated experimentally, implying a more economic exploitation of the confining magnetic field. Early studies on this approach undertaken in the 1970s and 1980s are discussed by Conn (1981) and Krakowski et al. (1986a,b). Perhaps the most detailed approach to the development of an RFP FPP design concept was undertaken within the TITAN study. Parameters of the TITAN-I device (Najmabadi et al., 1993a,b) are listed in Table 3. A variant of this concept, TITAN-II, with essentially the same device parameters and similar electrical power output, but using somewhat different technology, was also analyzed (Wong et al., 1993).

Striking features of this design concept, scaled to produce nearly 1 GWe, include the very compact dimensions of the device, with a major radius of 3.9 m, and the high value of β_p , 23%, resulting in a very high average neutron wall load, 18 MWm^{-2} . Although this neutron wall load would now likely be considered unacceptable, the overall aim of the study was to investigate the implications of exploiting the potential for a compact high power FPP associated with the RFP configuration, in particular to analyze the technical feasibility and principal development issues for such a concept. Several key physics assumptions were incorporated in the design that are still to be resolved within the RFP research program, e.g., (Zuin, 2021), including sustainment of sufficiently high energy confinement to achieve the required fusion gain and fusion power production, the efficiency and feasibility of exploiting oscillating field current drive (OFCD) to maintain the plasma current, which is the sole source of plasma heating and is an essential element of the magnetic configuration, and the feasibility of the approach to handling power and particle exhaust using toroidal divertors (as opposed to the poloidal divertors used in tokamaks, which are toroidally symmetric). The magnet system used both helium-cooled copper conductors, in particular for the ohmic heating circuit, and NbTi superconductors. Vanadium alloy was selected for all in-vessel structures and for the vacuum vessel due to its radiation resistance and capability for high temperature operation. The former feature was critical given the high average neutron wall load. Nevertheless, it was foreseen to replace the entire torus structures annually, and this requirement was incorporated into the plant layout. Although it was noted that certain features of the vanadium alloy might mitigate corrosion by the liquid lithium, this issue would require further R&D.

Overall, the TITAN concept was a highly ambitious approach to an FPP design which, nevertheless, sought to understand the implications of fusion power production in a very compact system. Further developments of several physics and technology aspects of the concept, including the implications for the COE, were discussed by Müller (2009). Subsequently, RFP research has focused on resolving the key physics issues limiting plasma fusion performance and there has been no further conceptual developments towards an FPP with a pure fusion RFP at its core. However, a proposal for a fusion-fission hybrid reactor using an RFP as the fusion neutron source has recently been developed (Piovan et al., 2018, 2020).

Table 3 Principal parameters of the TITAN reversed field pinch power plant study.

Parameter	TITAN-I (Najmabadi et al., 1993a,b)
Toroidal field (plasma surface) (T)	0.36
Plasma current (MA)	17.8
Major, minor radius (m)	3.9, 0.6
Aspect ratio (R/a)	6.5
β_p	23%
Plasma volume (m ³)	28
Auxiliary power ^a (MW)	0
Fusion power (MW)	2301
Total thermal power ^b (MW)	2935
Net electrical power (MW)	970
Av. neutron wall load (MWm ⁻²)	18
Operation mode ^c	Steady-state
Thermal Cycle	Rankine
Thermal efficiency, η_{th}	0.44
Recirculating power fraction (1/Q _{Elec})	0.25
First wall material	V-3Ti-1Si
First wall structure	V-3Ti-1Si
First wall coolant	Liquid Li
Breeding blanket structure	V-3Ti-1Si
Breeding material	Liquid Li
Breeding blanket coolant temp. (K)	970
Breeding blanket coolant	Liquid Li
Shield blanket material	V-3Ti-1Si
Vacuum vessel structure	V-3Ti-1Si

^aPlasma is heated entirely by ohmic heating produced by plasma current.

^bSee Table 1, footnote b.

^cSteady-state power production is maintained by oscillating field current drive (OFCD).

Fusion-Fission hybrid reactors

The fusion-fission hybrid reactor concept noted in the previous section was originally developed during the initial decades of fusion research, see, e.g., (Moir, 1981). A hybrid reactor has a fusion device at its core which is surrounded by a blanket which not only breeds tritium from lithium to sustain the fusion fuel cycle, but has a sub-critical assembly of fissile material which uses the fusion neutrons to sustain the fission reactions and converts fertile material, such as ²³²Th or ²³⁸U, into fissile material, i.e., ²³³U or ²³⁹Pu. A hybrid reactor is capable of net electricity production, and a significant electrical gain factor, Q_e , from a low fusion gain core, $Q_{th} \sim 1-3$, while producing fissile material and “burning” nuclear waste. It would allow the exploitation of fusion energy with significantly lower fusion performance (and hence more modest plasma parameters) than for a pure fusion device. However, the technical complexity of the hybrid core, as well as the environmental, safety and nuclear proliferation issues associated with the hybrid concept, are more challenging than for a pure fusion device. The following paragraph touches on several references which can provide an introduction to this area of research without attempting to address the scientific, technical and environmental issues involved in the conceptual designs or their implementation.

The key concepts associated with the hybrid reactor are presented in tutorial form by Stacey (2012). In addition, a detailed review of the research program required to advance the realization of this approach can be found in Freidberg et al. (2009). The scope of contemporary research activities on this topic, which are perhaps most vigorously pursued in China and Russia, can be found in the proceedings of the International Conferences on Fusion-Fission Sub-Critical Systems for Waste Management and Safety, e.g., (Pizzuto and Orsitto, 2017, 2019). A range of toroidal and linear fusion devices have been considered as suitable neutron sources within conceptual designs for hybrid systems and more detailed presentations of specific concepts can be found in, e.g., (Wu et al., 2006; Ryutov et al., 2010; Stewart and Stacey, 2014; Kuteev et al., 2017; Stacey, 2017; Piovan et al., 2018).

Conclusions

The magnetic fusion research program is still addressing a range of significant challenges in both science and technology, and the realization of a fusion power plant remains several decades in the future. Nevertheless, the conceptual studies of FPPs undertaken in recent decades, of which only a subset has been reviewed here, have been essential in deepening the understanding of how the scientific and technological solutions resulting from the ongoing research must converge to a self-consistent system which is capable of large-scale electricity production at a cost which is likely to be economically acceptable. It must also be emphasized that the various conceptual designs reviewed here are likely to be representative of different phases in the eventual implementation of a fleet of FPPs.

As discussed by [Donné et al. \(2021\)](#), the scale and complexity of an FPP is likely to require a rather conservative development path towards commercial electricity production, and this is reflected in current concepts for a DEMO reactor—the “proof-of-principle” step for fusion electricity production and the next step beyond ITER towards construction of the first FPPs. However, if this development program is successful, it can be anticipated that the extensive natural fuel resources available for fusion energy, together with its favorable safety and environmental characteristics, will encourage the exploration of more advanced, potentially more cost-effective, concepts for fusion reactors. Even the most scientifically and technologically ambitious of the concepts reviewed in the preceding discussion can be expected to evolve under the influence of results from the current R&D program, in particular from ITER, which will provide the first opportunity for the study of burning plasmas at the reactor scale. They nevertheless underline the options available in the longer term for refining the design of FPPs towards lower COE.

The timescale for construction of the first (commercial) FPP remains uncertain and there are opportunities in the ongoing R&D programs that could accelerate the steps to demonstrating the scientific, technical and commercial viability of fusion. Examples include the development of large-scale HTS-based magnets, which might support more compact reactor designs, nuclear qualification of improved radiation-resistant structural materials that can operate at higher temperature, improving the power plant thermal efficiency, and a demonstration of liquid metal plasma facing components that can operate in the burning plasma environment, which might allow longer operational periods between in-vessel interventions for component replacement. Such opportunities should therefore be pursued in parallel with the current major thrusts of the magnetic confinement program developed around ITER and DEMO, the further exploration of the reactor potential of the stellarator configuration and the construction of a high-intensity fusion neutron source for the qualification of reactor-relevant materials.

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Inertial Confinement Fusion – Physics Principles

Stefano Atzeni, Dipartimento SBAI, Università degli Studi di Roma “La Sapienza”, Roma, Italy

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Glossary

Coulomb logarithm A quantity characterizing binary collisions of plasma particles, defined as the logarithm of the ratio of the maximum to the minimum value of the impact parameter.

CPA Chirped pulse amplification.

DT Equimolar mixture of deuterium and tritium.

D-T Deuterium-tritium fusion reaction.

ICF Inertial confinement fusion.

IFAR In-flight-aspect ratio.

IFE Inertial fusion energy.

RTI Rayleigh-Taylor instability: instability of a fluid layer accelerated by the pressure exerted by a less dense fluid.

SI Shock ignition.

FI Fast ignition.

IFAR In-flight-aspect ratio.

MCF Magnetic confinement fusion.

TNSA Target normal sheet acceleration.

Introduction

Net release of energy by fusion reactions requires the attainment of very high temperature as well as appropriate *confinement* of the reacting materials, at the plasma state. By confinement we refer to a mechanism allowing to maintain burning fusion conditions for a sufficiently long time, so that the energy released exceeds the energy spent to create and sustain the burning plasma configuration.

In magnetic confinement fusion (MCF), a relatively low density plasma is kept in a quasi-steady-state in which power losses from the plasma are compensated by power delivered by fusion products and, if required, by auxiliary heating systems. The relevant necessary condition is expressed by the Lawson criterion, $n\tau_E > \zeta(T)$, requiring that the product $n\tau_E$ of ion density n times energy confinement time τ_E exceeds a certain function of the temperature $\zeta(T)$. Characteristic reference values for an equimolar deuterium-tritium (DT) plasma, at temperatures of 10 – 15 keV, are $\zeta \approx 10^{14} \text{ cm}^{-3} \text{ s}$, $n = 10^{14} \text{ cm}^{-3}$ and $\tau_E \approx 1 \text{ s}$. (Here and in

most of this Chapter we will consider DT fuels; advanced fusion fuels are briefly dealt with in Section “*ICF with deuterium and advanced fuels*”).

Inertial confinement fusion, ICF, takes a radically different approach: fusion conditions are achieved in a strongly compressed and hot fusion fuel, with such a large pressure (hundreds of billion of atmospheres) that no containment is feasible, other than that due mass inertia. The fuel remains in the compressed state for the time τ a rarefaction wave (proceeding with the isothermal sound speed c_s) takes to propagate through it. For a sphere of radius R , we may consider a fuel averaged confinement time $\tau \approx R/4c_s$, with the factor 4 in the denominator accounting for the earlier rarefaction of the outer layers, which contain most of the sphere mass. For a fully ionized DT plasma, $c_s = 2.8 \times 10^7 \sqrt{T}$ cm/s, with T the temperature in keV. It follows that the confinement time of an ICF burning plasma, with $R \approx 100 \mu\text{m}$ and temperatures of tens of keV is as short as 100 ps or less. Notice that while for steady state MCF plasmas we consider an energy confinement time, here the relevant time τ is a matter confinement time.

As detailed in the following, typical values of particle density are in the range of $10^{25} - 10^{26} \text{ cm}^{-3}$ in ICF. Fusion reactions occur in a time interval comparable to (or shorter than) such a confinement time. A prospective inertial fusion energy (IFE) reactor will therefore release energy in a sequence of micro-explosions, to be contained within the reactor. If the energy release is limited to 1 GJ per target, the burnt mass in a target cannot exceed 3 mg, since the specific yield (energy released by complete burn) of DT is $Y_{\text{DT}} = 337 \text{ MJ/mg}$.

In an IFE power plant suitable *targets* containing small quantities of fusion fuels will be burned at a rate of 1–10 Hz. Energy to heat and compress the target is provided by a powerful pulsed *driver*, such as a laser. Of course, operation of the plant requires that the produced energy exceeds (by far) the energy necessary to operate the driver. In this context, target energy gain, or simply *gain* $G = E_{\text{fus}}/E_d$, ratio of fusion energy released by a single target to the energy in the driver pulse, is a particularly important figure of merit to gauge target performance.

In the following sections of the present Chapter we discuss confinement and compression requirements, ignition conditions, as well as conditions for the achievement of high gain. We show how compression requirements, ignition conditions, and ignition modality follow from the need of significant gain in a containable micro-explosion. We also briefly illustrate different schemes which have been proposed to compress and ignite ICF fuels. The reader interested in more detailed treatments is referred to the textbook by [Atzeni and Meyer-ter-Vehn \(2004\)](#) and to the review papers by [Brueckner and Jorna \(1974\)](#), [Lindl \(1995\)](#), [Lindl et al. \(2004\)](#), [Craxton et al. \(2015\)](#), and [Betti and Hurricane \(2016\)](#). A pedagogical treatment of the basic plasma physics of ICF can be found in an article by [Tikhonchuk \(2021\)](#).

Basic requirements

IFE plant energy cycle and target gain

We now consider a simplified IFE power plant energy cycle. The fuel containing target is irradiated by driver (e.g. laser) pulses of energy E_d , obtained by converting electrical energy into beam energy (photons) with efficiency η_d . The irradiated target releases fusion energy $E_{\text{fus}} = GE_d$, with a gain G . Such an energy is first converted into thermal energy in a suitable blanket, and then into electrical energy through a thermal cycle, with efficiency $\eta_{\text{th}} \approx 40\%$. A fraction f of this electrical energy is used to energize the laser, hence $\eta_d G \eta_{\text{th}} f = 1$. Requiring recycled electrical energy fraction $f \leq 25\%$, we obtain

$$G \eta_d \geq 10. \quad (1)$$

For driver efficiency of 10%, as predicted for future reactor-size lasers ([IAEA, 1995](#)), target gain $G \geq 100$ is required. Smaller gains can be allowed by potentially more efficient drivers, such as heavy ion accelerators, which could achieve $\eta_d \approx 25\%$.

The power plant delivers to the grid electrical power $W = \nu_d(1 - f)\eta_{\text{th}}GE_d$, where ν_d is the shot frequency. A set of possible values for a 900 MW power plant is $E_d = 3 \text{ MJ}$, $\eta_d = 10\%$, $\nu_d = 10 \text{ Hz}$, $G = 100$, $f = 25\%$. For simplicity, the above considerations do not include an additional power multiplication, by a factor of about 1.1, due to exothermal neutron induced reactions in the reactor blanket.

Key requirements for high gain ICF are identified by simple considerations on target gain G . The driver pulse energy is used to compress and heat the target; however (see, e.g. [Tikhonchuk, 2021](#)) only a fraction η_c , typically 5–10% in the conventional direct-drive schemes discussed in Section “*Conventional beam-driven spherical implosion*,” is actually transferred to the fuel mass m . We can then write

$$G = \frac{E_{\text{fus}}}{E_d} = \frac{m\Phi Y_{\text{DT}}}{me/\eta_c} = \eta_c \frac{\Phi}{e} Y_{\text{DT}} = 100 \left(\frac{\eta_c}{0.1} \right) \left(\frac{\Phi}{0.3} \right) \left(\frac{10^{15} \text{ erg g}^{-1}}{e} \right) \quad (2)$$

with m the fuel mass, Φ the fuel fractional burn-up, Y_{DT} the specific DT yield, and e the specific energy (energy/mass) delivered to the fuel. According to Eq. (2) gain of the order of 100 implies fractional burn of 20–30% from a fuel with mass averaged specific energy of the order of $10^{15} \text{ erg g}^{-1}$. In the next two subsections, we show that two basic ingredients of high gain ICF, namely strong compression and hot spot ignition, just follow from the requests of relatively large Φ and relatively small e .

Fractional burn, confinement and compression

A first estimate for the fractional burn Φ is given by the ratio τ/τ_b of the confinement time τ defined earlier to the characteristic nuclear burn time. For a deuterium or tritium nucleus in an equimolar DT plasma with nuclei density n , $\tau_b^{-1} = (1/2)n \langle \sigma v \rangle$, where $\langle \sigma v \rangle = \langle \sigma v \rangle(T)$ is the Maxwellian averaged reactivity of the D - T fusion reaction, with σ the reaction cross section and v the relative velocity of the reacting nuclei.

For a homogenous spherical fuel we then have

$$\Phi = \frac{nR \langle \sigma v \rangle}{8c_s} = \frac{\rho R}{H_b}, \quad (3)$$

where ρ is the mass density, ρR is the so called ICF confinement parameter, and $H_b = H_b(T) = 8m_i c_s / \langle \sigma v \rangle$ is the ICF burn parameter. Eq. (3) only applies for small values of Φ and homogeneous fuels. A more general expression, accounting for fuel consumption and inhomogeneity is

$$\Phi = \frac{\langle \rho R \rangle}{\langle \rho R \rangle + H_b}, \quad (4)$$

where $\langle \rho R \rangle = \int \rho dR$ over the whole fuel. For an ignited DT fuel, with $T = 20 - 80$ keV, the temperature dependence of H_b is weak and one can take $H_b = 7 \text{ g cm}^{-2}$ as a reference value. Fractional burn of 30% then requires $\langle \rho R \rangle = 3 \text{ g cm}^{-2}$. Achieving such a value of the confinement parameter with a mass of at most a few mg, implies strong compression. For a homogenous sphere $m = (4/3)\pi(\rho R)^3/\rho^2$. It follows that

$$\rho = \sqrt{\frac{4\pi(\rho R)^3}{3m}} = 194 \left(\frac{\rho R}{3 \text{ g cm}^{-2}} \right)^{3/2} \left(\frac{m}{3 \text{ mg}} \right)^{-1/2} \text{ g cm}^{-3}. \quad (5)$$

Average density of at least 200 g cm^{-3} , i.e. 800 times DT solid density is required. Even stronger compression is necessary for the smaller targets to be irradiated in experiments aiming at demonstrating ignition.

Hot spot ignition

A second key requirement follows from limiting the average fuel specific energy at ignition to $e_{\max} \sim 10^{15} \text{ erg g}^{-1}$ (see Eq. 2). According to the ideal gas equation of state, $e = C_v T$ with the DT specific heat $C_v = 1.15 \times 10^{15} \text{ erg g}^{-1} \text{ keV}$, this corresponds to a DT temperature of 870 eV. Such a temperature is well below the ideal ignition temperature of 4.3 keV at which the power associated to the 3.5 MeV alpha-particles released by the DT fusion reaction equals the bremsstrahlung power loss. Heating the whole fuel to temperature of 4 – 5 keV or larger would result in unacceptably low gain. The solution is *spark ignition* or *hot spot ignition*. Only a small part of the fuel is heated to the high temperature necessary to trigger effective fusion burn. Such a hot spot first self-heats and ignites due to the energy delivered by the fusion alpha-particles, and then drives a burn wave which ignites the remaining cold fuel. Of course, as discussed in the next section, the hot spot parameters must satisfy specific conditions, concerning confinement parameter and temperature, in order to self-heat and ignite. In addition, in order to make use of the driver energy in the most efficient way, the main fuel should be compressed by keeping its temperature as low as possible. Ideally, one would like to produce a cold plasma. In practice, the main fuel will be heated to some (relatively low) temperature T_c and will have pressure $p_c = p_c(\rho_c, T)$. Compression is characterized by the ratio

$$\alpha = p_c(\rho_c, T)/p_d(\rho_c) \geq 1, \quad (6)$$

i.e. the ratio of the actual fuel pressure to the pressure $p_d = A_d \rho^{5/3}$ of a cold gas with degenerate electrons. Here $A_d = 2.15 \times 10^{12} \text{ (erg g}^{-1}) \text{ (g cm}^{-3})^{-2/3}$. Since this ratio is proportional to entropy and is therefore constant during isentropic compression, it is usually called *isentropic parameter* or *adiabat*. The achievement of strong compression with low entropy generation, i.e. with small values of α is discussed, e.g., by Tikhonchuk (2021).

Hot spot ignition

Hot spot ignition conditions depend on the specific mechanism leading to the generation of the compressed fuel assembly. However, relevant information is obtained by considering the simple case of a central hot spot in a spherical fuel, initially at rest. It is assumed that such a configuration is generated as the target implosion stagnates, i.e. at the time of maximum compression of the fuel. In the spirit of this over-simplified model we assume that the hot spot is a homogeneous sphere of DT of radius R_h , with mass density ρ_h and temperature T_h . The surrounding fuel, with mass $m_c \gg m_h$, is also homogeneous, with density ρ_c and temperature $T_c \ll T_h$. Due to its high temperature the hot spot specific energy is that of an ideal gas, $e_h = C_v T_h$, while the colder fuel energy can be parametrized as $e_c = (3/2)\alpha A_d \rho_c^{2/3}$. It follows that the average specific energy of the igniting fuel assembly can be written as

$$e \approx (m_h/m_c) C_v T_h + (3/2)\alpha A_d \rho_c^{2/3}. \quad (7)$$

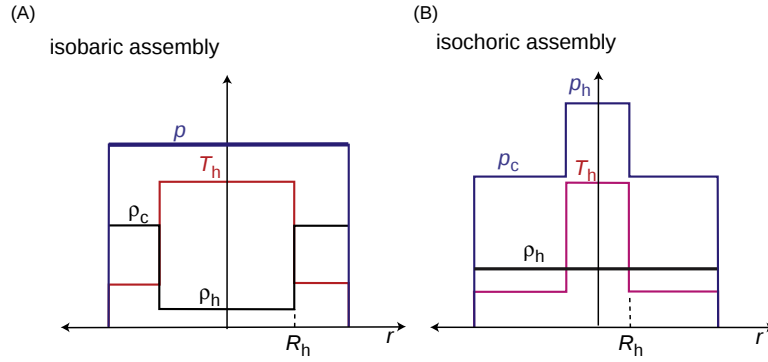


Fig. 1 Fuel configuration at ignition: (A) isobaric; (B) isochoric, illustrated by radial profiles of density (black), temperature (red) and pressure (blue). The significance of these two approaches is discussed in Section “Hot spot ignition.”

Two limiting cases are of particular interest (see Fig. 1). In the first one the compressed assembly is isobaric, i.e. the pressure is uniform over the whole fuel. In this case the hot spot is surrounded by much denser fuel. In the other case (isochoric assembly) the density is uniform and then the hot spot pressure substantially exceeds the cold fuel pressure. Isobaric conditions are a good approximation to the actual state of the compressed fuel in conventional ICF implosions, sketched in outline in the following Section “Conventional beam-driven spherical implosion.” Isochoric conditions, instead, are relevant to fast ignition schemes (Section “Fast ignition”), in which the fuel is first compressed to high density and then the hot spot is generated by an ultra-intense external particle beam pulse of duration shorter than the energy confinement time of the hot spot itself.

Hot spot power balance and hot spot self-heating conditions

The time evolution of the hot spot internal energy is ruled by a competition between heating by fusion alpha-particles (delivering a power density $W_{\alpha-in}$ inside the hot spot), and cooling by bremsstrahlung radiation (W_b) and thermal conduction (W_{th}). In the isochoric case, an additional loss is due to the mechanical work (with power density W_m) performed by the hot spot that expands, since its pressure exceeds that of the surrounding fuel. Neutron interaction is energetically negligible for typical hot spot parameters. The 3.5 MeV alpha-particles, which carry 1/5 of the DT fusion energy, have a finite range l_α and then deposit a fraction f_{in} of their energy in the hot spot itself. A first, rough evaluation of the ignition conditions can then be obtained as follows. Alpha-particle range, determined by Coulomb interactions with plasma electrons and ions, can be written as $l_\alpha \approx 0.025 T_h^{1.25} \rho_h^{-1} \text{ cm}$ (Fraleigh et al., 1974), for temperatures below 30 keV and densities in the range $10 - 100 \text{ g cm}^{-3}$. The fraction f_{in} then turns out to be a function of the ratio $\xi = R_h/l_\alpha$; in particular, for $\xi > 1/2$, $f_{in} \approx 1 - 1/4\xi$ (Gus'kov et al., 1974). It follows that, at temperatures 5 – 7 keV, when the alpha-particle power starts exceeding bremsstrahlung losses, a necessary condition for hot spot self-heating is a substantial containment of alpha-particles, which requires $\xi \approx 1$. The hot spot confinement parameter (product of density and radius) should satisfy $\rho_h R_h \approx \rho_h l_\alpha$. At temperature $T_h = 7 \text{ keV}$, $\rho_h R_h \approx 0.3 \text{ g/cm}^2$.

For a more accurate calculation, let us now consider the rate of change of the hot spot energy density ε_h

$$d\varepsilon_h/dt = W_{\alpha-in} - W_b - W_{th} - W_m, \quad (8)$$

where the individual power terms can be written as

$$W_{\alpha-in} = f_{in} A_\alpha \rho_h^2 < \sigma v >, \quad (9)$$

$$W_b = A_b \rho_h^2 T_h^{1/2}, \quad (10)$$

$$W_{th} = A_{th} T_h^{7/2} / R_h^2 = A_{th} \rho_h^2 T_h^{7/2} / (\rho_h R_h)^2, \quad (11)$$

$$W_m = (p_h/V_h)(dV_h/dt) \quad (12)$$

We have used standard expressions for fusion power density and bremsstrahlung power density. The conduction term is obtained using dimensional arguments and the usual Spitzer (Spitzer and Härm, 1953) formula for electron thermal conductivity, $\chi \propto T^{5/2}/\ln \Lambda$, where $\ln \Lambda$ is the Coulomb logarithm, weakly depending on both temperature and density. The quantities A_α and A_b are constants, with values $A_\alpha = 8 \times 10^{40} \text{ erg g}^{-2}$ and $A_b = 3.05 \times 10^{23} \text{ erg cm}^3 \text{ g}^{-2} \text{ s}^{-1} \text{ keV}^{-1/2}$, $A_{th} \approx 2.9 \times 10^{20} / \ln \Lambda \text{ erg s}^{-1} \text{ cm}^{-1} \text{ keV}^{-7/2}$, with $\ln \Lambda \approx 5$ in cases of interest here. Approximate expressions for $f_{in} = f_{in}(\xi)$ are given by Gus'kov et al. (1974). Mechanical power is zero in the isobaric case, since the hot spot initially does not expand. In the isochoric case (with $p_h \gg p_c$), instead, the hot spot expands suddenly, and an outward directed shock wave is driven. The expansion velocity can be

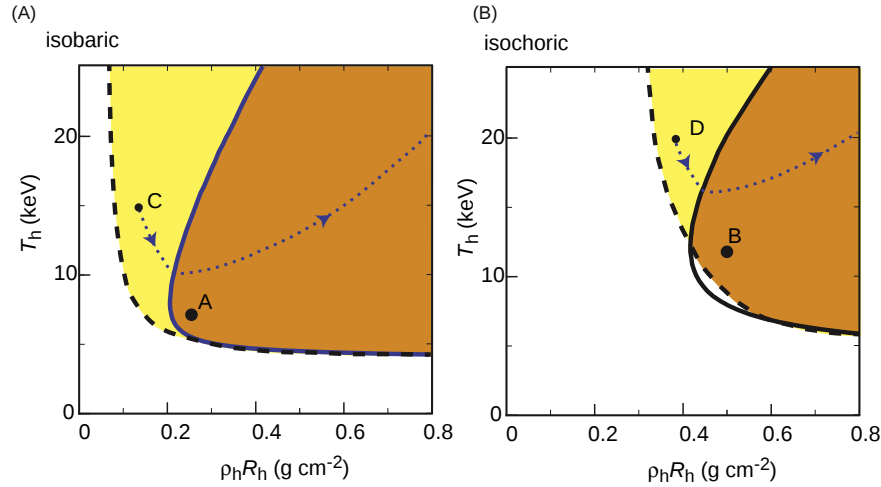


Fig. 2 Hot spot self-heating boundaries (thick curves) and ignition boundaries (dashed curves) in the $\rho_h R_h - T_h$ plane, for (A) isobaric initial conditions and (B) isochoric initial conditions. Hot spots with initial parameters in the orange areas self-heat; hot spots with initial parameters in the yellow areas eventually self-heat and ignite after an initial cooling stage, as shown by the dotted curves originating from point C in frame (A) and from point D in frame (B). Points A and B indicate reference cases leading to ignition with hot spot initial energy close to a minimum.

approximated as that of the material behind a strong shock wave propagating in a homogeneous medium (Zel'dovich and Raizer, 1967), $u_h = dR_h/dt \approx (3p_h/4\rho_c)^{1/2} = (3p_h/4\rho_h)^{1/2} = (C_v T_h/2)^{1/2}$. We then write

$$W_m = A_m \rho_h T_h^{3/2} / R_h = A_m \rho_h^2 T_h^{3/2} / (\rho_h R_h), \quad (13)$$

with $A_m = 5 \times 10^{22} \text{ cm}^3 \text{ s}^{-3} \text{ keV}^{-3/2}$. From Eqs. (9)–(11) and (13) it is apparent that a self-heating condition $dE_h/dt > 0$ can be written in the form

$$\rho_h R_h > g(T_h), \quad (14)$$

when the weak dependence of the Coulomb logarithm on temperature and density is neglected. The self-heating boundaries computed for both isochoric and isobaric ignition conditions are shown in Fig. 2, and found to agree with the results of detailed hydrodynamic simulations. Hot spots with initial parameters in the orange areas self-heat.

Hot spot ignition conditions

It is important to notice, however, that also initial conditions falling outside the self-heating region can eventually lead to fuel ignition and propagating burn. Indeed, for initial conditions in the yellow area in the figure, the hot spot at first cools, but the energy lost by thermal conduction and the energy deposited by the alpha-particles just outside the hot spot heats a layer of initially colder fuel surrounding the hot spot (notice that the cold fuel is a more efficient stopper than the hot spot). This fuel is ablated and enters the hot spot, increasing its density and then its ability to contain the alpha-particles. This may lead, as in the cases represented by the dotted curves originating from point C in Fig. 2A) and from point D in Fig. 2B), to eventual self-heating and ignition (Atzeni and Caruso, 1984). We can then define an ignition boundary of the same form as (14) as the separatrix between initial conditions finally leading to fuel burn and those leading to cooling. Ignition boundaries, determined by detailed numerical simulations, are represented by the thick dashed curves in Fig. 2. The curve for isobaric conditions, which also depends on the density ratio $r_\rho = \rho_c/\rho_h$, is plotted for $r_\rho = 16$. The figure also shows two points, A and B, corresponding to non-marginal ignition at nearly minimum energy for isobaric and isochoric conditions, respectively. In case A, $\rho_h R_h = 0.25 \text{ g cm}^{-2}$, $T_h = 7 \text{ keV}$; in case B, $\rho_h R_h = 0.50 \text{ g cm}^{-2}$, $T_h = 12 \text{ keV}$. The corresponding hot spot energy is

$$E_h = (4\pi/3) R_h^3 \rho_h C_v T_h = E_0 \left(\frac{\rho_h}{100 \text{ g cm}^{-3}} \right)^{-2}, \quad (15)$$

with $E_0 = 5.2 \text{ kJ}$ for case A (isobaric), and $E_0 = 72 \text{ kJ}$ for case B (isochoric).

Designed ignition experiments and prospective reactor targets exploiting isobaric ignition aim at generating hot spots with density of $50 - 100 \text{ g cm}^{-3}$, radius of $30 - 50 \mu\text{m}$, surrounded by a denser fuel with density up to 1000 g cm^{-3} . Fast ignition schemes consider nearly uniform density of $300 - 500 \text{ g cm}^{-3}$, and then hot spot radii of $10 - 20 \mu\text{m}$.

The ICF ignition conditions, in the form $\rho_h R_h > g(T_h)$, are analogous to the Lawson criterion $n\tau_E > \zeta(T)$ for magnetic confinement fusion. Interestingly, also for ICF, we can identify a metric for progress towards ignition analogous to the MCF triple product $n\tau_E T$. In Fig. 3, we plot the quantity $\rho_h R_h T_h r_\rho^{1/2}$ needed for ignition according to full simulations, for the isochoric case and for an

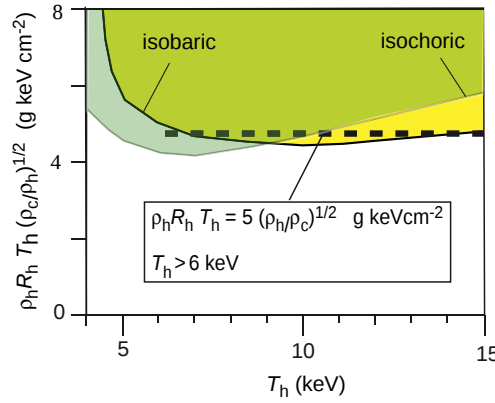


Fig. 3 Ignition condition in a form similar to the triple product considered in magnetic fusion. The thick dashed line represents the ignition condition (16); the green and yellow areas, denote hot spot ignition regions for isochoric and isobaric initial condition, respectively.

isobaric case with $r_\rho = 16$. We see that the two curves are close to each other for temperature $T_h > 6$ keV. A general, approximate ignition condition can then be written as

$$\rho_h R_h T_h \geq 5/\sqrt{r_\rho} \text{ g cm}^{-2} \text{ keV}; \quad T_h > 6 \text{ keV.} \quad (16)$$

or, using $p_h = (2/3)\rho_h e_h$,

$$p_h \geq 400 \left(\frac{10}{r_\rho} \right)^{1/2} \frac{30 \text{ } \mu\text{m}}{R_h} \text{ Gbar,} \quad (17)$$

where the hot spot radius is normalized to a typical value of $30 \mu\text{m}$. Eq. (17) shows that the hot spot pressure at ignition in typical conventional ICF cases is somewhat larger than the pressure at the Sun center (250 Gbar).

Following ignition, a burn wave propagates outward. In its first stage, its dynamics are characterized by the self-regulation of the size of the burning region to range of the alpha-particles (which increases with temperature, roughly as $T^{5/4}$). Other regimes of burn propagation can occur at very high temperatures. In any case, the fractional burn is essentially a function of the confinement parameter $\langle \rho R \rangle$ only.

Low temperature volume ignition

The previous discussion on ignition conditions assumes that emitted bremsstrahlung radiation escapes from the hot spot. This is a good approximation for the hot spots considered so far, since the characteristic Rosseland mean free path of thermal photons (Zel'dovich and Raizer, 1967) in DT is $l_R = 400 T^{7/2} \rho^{-2} \text{ cm}$, with the temperature in keV and the density in g cm^{-3} . However, a uniformly heated mass of dense hot fuel, with radius comparable to or larger than l_R , reabsorbs most of the emitted energy. In this case volumetric emission is replaced by surface emission. The fuel can now self-heat also at low temperature, if the confinement time is sufficiently long (Caruso, 1974). At large $\rho R \geq 5 \text{ g cm}^{-2}$ the fuel ignites at temperature of the order of 1.5 keV or lower. Note however that, even neglecting the contribution due to compression, the fuel specific energy is still too large to allow for large fusion gain. In addition, the large value of ρR required implies either large mass (and then large drive energy) or extremely high density, and self-heating is relatively slow, so that fractional burn up is smaller than the value computed using Eq. (4).

ICF with deuterium and advanced fuels

The preceding discussion only concerns DT fuels and the D-T reaction



Other fusion reactions and therefore other fuels are of potential interest. We refer in particular to the reactions



Table 1 Key properties of thermonuclear fusion fuels.

	T_{id} (keV)	$H_b(g\text{ cm}^{-2})$	Y (MJmg $^{-1}$)
DT	4.3	7	341
D	35	52	88
D (full catalysis)	25	35	350
D 3 He	28	51	357
p 11 B	–	73	70

Adapted from the original in Atzeni and Meyer-ter-Vehn (2004) *The Physics of Inertial Fusion*. Oxford University Press, reproduced with permission of Oxford University Press through PLSclear.

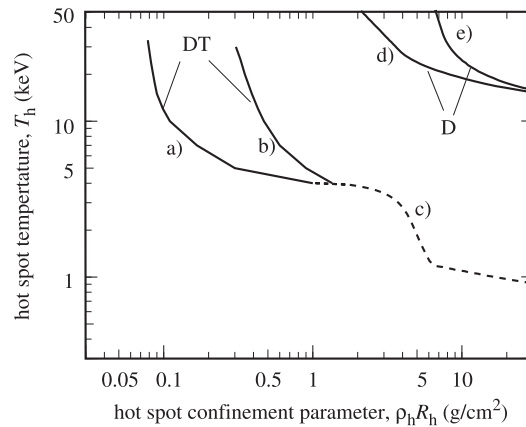


Fig. 4 Ignition boundaries for: a) DT fuel, isobaric initial conditions; b) DT fuel, isochoric initial conditions; c) DT fuel, volume ignition; d) pure D, isobaric initial conditions; e) pure D, isochoric initial conditions. Curve d) was first computed by Basko (1990). Adapted from the original in Atzeni and Meyer-ter-Vehn (2004) *The Physics of Inertial Fusion*. Oxford University Press, reproduced with permission of Oxford University Press through PLSclear.

The DT mixture is the main candidate fuel for IFE because the reactivity of the D-T reaction is by far larger than that of any other reaction at temperature $T < 100$ keV. Pure deuterium is an attractive fuel, due to its practically unlimited availability. D-D reactions also produce T and ^3He , which can react with other D nuclei, through D-T and D- ^3He reactions. This so-called *catalytic burn* increases significantly deuterium reactivity, but to levels still much below that of DT. The reactivity of D- ^3He is higher than that of deuterium at $T > 25$ keV, and the reaction D- ^3He does not release neutrons. However in a D ^3He fuel D-D reactions occur. Additionally, ^3He is only found in extremely small quantities on Earth. A hydrogen-boron mixture would release a negligible amount of neutrons, but p- ^{11}B reactivity becomes comparable to that of the other, D-based, fuels only at temperatures of the order of 100 keV.

Key parameters to evaluate the potential of fusion fuels, namely, ideal ignition temperature assuming equal electron and ion temperatures, T_{id} , minimum value, H_b , of the burn parameter, and specific yield, Y , are listed in Table 1. It is seen that p- ^{11}B cannot be ignited, while D-based fuels have ignition temperatures significantly higher than that of DT. The burn parameter of DT is much smaller (and then burn-up of a fuel with given ρR is much higher) than that of all other fuels. Hot spot ignition boundaries for DT and DD (including partial catalytic burn of T and ^3He) are shown in Fig. 4. In conclusion, high gain burn of pure D or advanced fuels does not appear feasible, at least using the ICF schemes discussed in this Chapter. Large and dense assemblies of deuterium or D- ^3He mixtures could instead be ignited by the burn wave generated by thermonuclear burn of a DT seed (Tabak, 1996; Atzeni and Ciampi, 1997).

Compression and ignition options: Conventional and advanced schemes

Conventional beam-driven spherical implosion

Qualitative description

According to the previous section, an IFE target should contain a few mg of DT fuel, which must be compressed to an average density of at least 1000 times solid density and low average entropy. In addition a hot spot with a pressure of 300–500 Gbar and sufficiently large radius has to be generated.

The standard scheme to generate such a compressed fuel configuration is beam-driven implosion (Nuckolls et al., 1972) of a millimeter sized hollow shell, schematically illustrated in Fig. 5. Here we consider laser-driven implosion, but the laser beams can be replaced by thermal X-rays or, possibly, by heavy ion beams. The target is in the form of a hollow shell (Kidder, 1976), rather

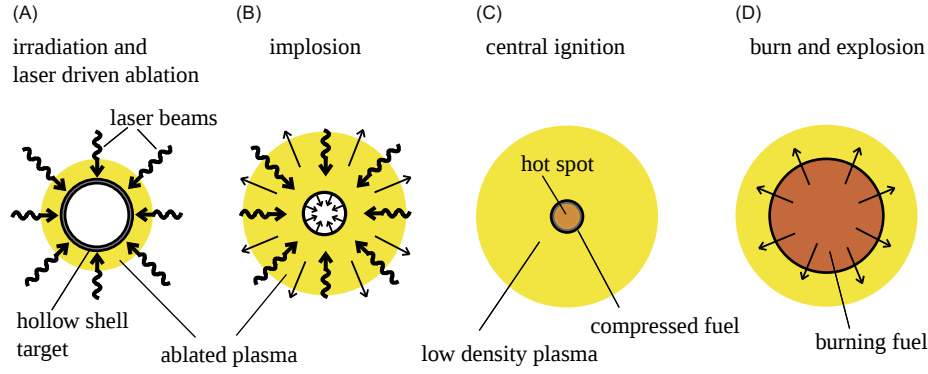


Fig. 5 Laser-driven implosion: (A) irradiation; (B) ablatively driven implosion; (C) compression and hot spot formation; (D) ignition and burn. Adapted from the original in Atzeni and Meyer-ter-Vehn (2004) *The Physics of Inertial Fusion*. Oxford University Press, reproduced with permission of Oxford University Press through PLSclear.

than a full sphere, to increase its outer surface, thus allowing for less intense and longer pulses (see below). The initial fuel aspect ratio (ratio $R_0/\Delta R$ of the fuel outer radius to the thickness of the fuel layer) is of the order of 10; larger values make the target too sensitive to instabilities. The fuel is solid DT (at temperatures of 17 – 19 K) in order to minimize the required compression. It is surrounded by a layer of material (plastic, beryllium, carbon) which absorbs laser light and provides structural robustness. DT vapor, with pressure equal to the vapor pressure of the solid DT, fills the shell interior.

We now briefly describe the implosion process, obtain the values of a few characteristic parameters, and outline a few key issues for the feasibility of ICF. Upon uniform laser irradiation the target outer layers are heated, removed (ablated) and expand at high velocity. As irradiation proceeds the hot ablated material becomes transparent to laser radiation and new material is ablated for the whole duration of the pulse. As a reaction to the outward motion of the ablated material, a shock wave is sent through the solid shell. As this shock wave reaches the shell inner surface, the whole shell is set in motion and accelerates inward. The mechanism is similar to a spherical rocket, where the exhausts are the ablated materials and the payload the DT fuel and the non ablated part of the other material. In this way, a fraction $\eta_c = \eta_a \eta_h$ of the laser beam energy is transformed into kinetic energy of the imploding fuel. Here η_a is the absorption efficiency of laser light and η_h is the hydrodynamic efficiency of rocket acceleration, i.e. the ratio of the kinetic energy of the imploding fuel to the absorbed laser energy. As the imploding fuel reaches the center, it stagnates, and its kinetic energy is converted into internal energy. The fuel is then heated and compressed and the central hot spot is created, in a roughly isobaric fuel assembly. The hot spot pressure at ignition is therefore the stagnation pressure, which is a strong function of the implosion velocity. According to an analytic model (Kemp et al., 2001), confirmed by simulations,

$$p_h \propto \alpha_{if}^{-0.9} u_{imp}^3 p_a^{0.4}, \quad (23)$$

where α_{if} is the in-flight adiabat parameter of the shell, i.e. the adiabat at the end of inward acceleration. (Adiabat at stagnation will be somewhat higher, due to the generation of entropy during shell deceleration, Hermann et al., 2001.) An implosion velocity of 300–450 km s^{−1} turns out to be necessary to achieve ignition, depending on adiabat and ablation pressure.

Parameter estimate

While complex multi-physics simulations are required for a complete description of the implosion (and still, they are not fully predictive) a few main parameters can be obtained by simple considerations. A prospective reactor-size target with fuel outer radius $R = 2 - 2.5$ mm, and fuel aspect ratio 10 contains 2.5–4 mg of DT. According to Eq. (23) the target must be imploded at a velocity $u_{imp} = 300 - 450$ km s^{−1} to achieve the pressure of 400 Gbar required for ignition. The corresponding fuel specific energy is $e = v^2/2 = (5 - 10) \times 10^{14}$ erg g^{−1}, as anticipated in Eq. (2). Laser beam parameters can be estimated as follows. The pulse energy is $E_{laser} = (1/2)mv^2/\eta_c$ or

$$E_{laser} = 1.84 \frac{m}{3 \text{ mg}} \left(\frac{u_{imp}}{350 \text{ km s}^{-1}} \right)^{20.1} \frac{1}{\eta_c} \text{ MJ}. \quad (24)$$

Assuming, for simplicity, implosion at constant acceleration from the initial radius to half the initial radius, we find a beam pulse duration $\Delta t = 5 - 8$ ns. Order of magnitude values for pulse energy, time averaged laser power and intensity are then $E_L = 1 - 5$ MJ, $W_L = E_L/\Delta t = 300 - 500$ TW and $I_L = W_L/4\pi R^2 = 3 - 5 \times 10^{14}$ W cm^{−2}.

Next, we consider the ablation pressure required to accelerate the fuel at the required implosion velocity. We assume that a uniform and constant pressure p_a is applied to the outer surface of the fuel up to the time the radius is halved. We assume that internal energy is negligible with respect to kinetic energy during the implosion acceleration stage. For the kinetic energy

theorem, $p_a \Delta V = (7/8)p_a V_0 = (1/2)mv^2$, with $V_0 = (4/3\pi R_0^3)$. Writing the fuel mass as $m \simeq 4\pi R_0^2 \Delta R \rho_{DT}$, where $\rho_{DT} = 0.25 \text{ g cm}^{-3}$ is the solid fuel density, we have

$$p_a = 53 \left(\frac{u_{\text{imp}}}{350 \text{ km s}^{-1}} \right)^2 \frac{\Delta R/R_0}{0.1} \text{ Mbar.} \quad (25)$$

The above value of the ablation pressure is underestimated because the initial shell mass is greater than the final one. In addition, this is a time-averaged value. In fact, actual laser pulses must be time-shaped. Their power has to increase gradually, according to precise prescriptions, in order to avoid the generation of too strong shocks, which would cause substantial fuel preheat, thus increasing the isentrope α and hindering substantial compression; see, e.g., [Tikhonchuk \(2021\)](#). It turns out that peak value of the required ablation pressure p_a is about twice that computed according to Eq. (25).

The generation of ablation pressure of the order of 100 Mbar upon laser irradiation of a solid material at intensity $I_{\text{laser}} \leq 10^{15} \text{ W cm}^{-2}$ is discussed, e.g., by [Tikhonchuk \(2021\)](#).

Four key issues

The benefits of spherical implosion of a hollow shell are clearly evidenced by comparing the ablation pressure driving the implosion, which peaks at 100–150 Mbar, and the pressure achieved at the end of the implosion, 300–500 Gbar. Spherical implosion, with properly timed pulses, can multiply the applied pressure by a factor of several thousand! However, four critical issues have to be addressed in order to achieve ignition and high gain ([Bodner, 1981](#); [Rosen, 1999](#)).

- i) Laser light must effectively (i.e. with adequate coupling efficiency η_c) drive target implosion, which firstly requires good light absorption and then generation of ablation pressure of about 100 Mbar. It is now established that this can be achieved using pulses of vuv laser light (with wavelength λ in the range 250–350 nm), at intensities $I_{\text{laser}} \simeq 10^{15} \text{ W cm}^{-2}$.
- ii) The coupled energy must be mainly used to compress the fuel, with small entropy generation, i.e. at small value of α (but at the center of the fuel, where the hot spot has to be created). This requires careful timing of the laser pulse, to avoid generation of too strong shocks. Any other form of fuel preheating, i.e. heating during the stage of inward acceleration, should be avoided. This concerns, in particular, preheating by non-thermal electrons generated by plasma instabilities, and preheating by hard X-rays.
- iii) Departures of implosion from spherical symmetry have to be limited, to allow for the generation of a small hot spot, with a radius of the order of 1/40–1/20 of the initial shell radius. This requires extremely uniform irradiation. Long-scale relative irradiation non-uniformities should indeed be limited to about 1%.
- iv) Beam-driven implosions are subjected to hydrodynamic instabilities, in particular the Rayleigh-Taylor instability ([Taylor, 1950](#)). RTI amplifies small initial target perturbations, and may cause shell breaking, mixing of different materials, or mixing of hot and cold fuel. The deleterious effects of RTI can be reduced by (a) reducing possible instability seeds, such as target defects and short-scale laser irradiation non-uniformities (b) reducing RTI growth by suitable choices of laser pulse shaping and target materials; (c) by reducing the target in-flight-aspect-ratio (IFAR, ratio of radius to thickness during the implosion), in such a way as to keep the target thickness larger than the amplitude of the perturbations. It should be observed however, that IFAR grows with implosion velocity and decreases with α . Controlling RTI therefore conflicts with the requirements of high implosion velocity and low entropy compression required for ignition and high gain.

The above critical aspects are discussed in some detail by [Tikhonchuk \(2021\)](#). For in-depth discussions the interested reader can refer to [Atzeni and Meyer-ter-Vehn \(2004\)](#).

Direct- and indirect-drive

In the previous section we have illustrated a directly laser-driven implosion: laser beams impinge on the target, generate a high plasma pressure, and drive the implosion. An indirect-drive scheme ([Lindl, 1995](#)) is also studied (actually this is the approach favored, at least so far, by the largest laser installations). In indirect drive the fusion capsule is placed at the center of a cavity (or *hohlraum*) of a heavy material, see [Fig. 6](#). The laser beams penetrate inside the cavity through entrance holes and irradiate the inner side of the cavity, producing a layer of plasma that emits X-rays. Such X-rays are confined (actually, absorbed and re-emitted) in the cavity, which acts as a black-body. The resulting quasi-thermal radiation bath irradiates the capsule, driving its implosion.

Indirect-drive is less sensitive than direct-drive to laser nonuniformities. X-ray driven acceleration is more robust to instabilities than laser-driven acceleration ([Lindl, 1995](#)). However, the intermediate steps of X-ray generation and X-ray transfer to the capsule reduce significantly (by a factor of three or larger) the overall laser-to-fuel coupling efficiency η_c . All main aspects of indirect-drive are discussed in two extensive review papers by [Lindl \(1995\)](#) and [Lindl et al. \(2004\)](#).

Conventional ignition vs advanced ignition schemes

In the conventional implosion scheme the laser pulse provides both fuel compression and generation of the ignition hot spot at the center of a quasi isobaric assembly. Other, *advanced ignition schemes* have been proposed, in which the two stages of compression and ignition are completely or partially separated. In these cases the igniting compressed configuration is isochoric, or, anyhow, non-isobaric. Here we just outline the basic principles of advanced ignition schemes. More details can be found in [Tabak et al. \(2014\)](#) and in the references quoted therein.

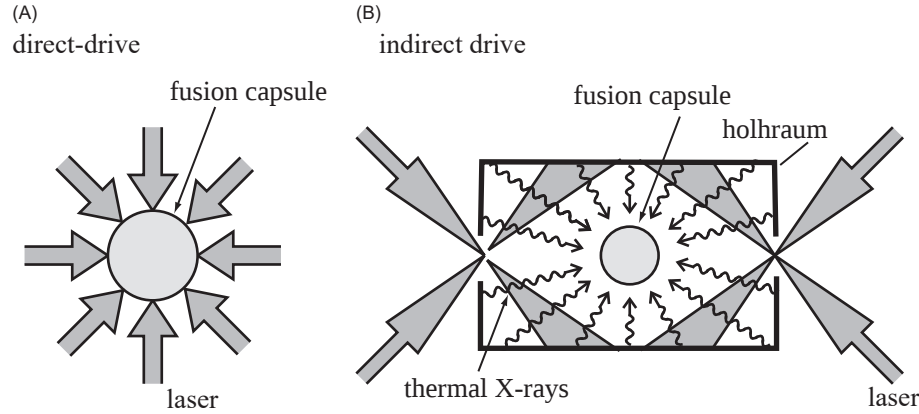


Fig. 6 Inertial fusion main schemes: (A) direct-drive; (B) indirect-drive.

Eq. (17) shows that the hot spot ignition condition is essentially a condition on the pressure of the hot spot. In the conventional scheme, hot spot pressure is the pressure at implosion stagnation, which is a strong function of the implosion velocity (see Eq. 23). However, the larger the implosion velocity the more difficult is control of RTI. The confinement parameter $\langle \rho R \rangle$, instead, depends essentially on laser energy and in-flight adiabat, and very weakly on implosion velocity, $\langle \rho R \rangle \propto E_{\text{laser}}^{1/3} u_{\text{imp}}^{0.06} \alpha_{\text{if}}^{-0.55}$ (Zhou and Betti, 2006). Reducing the implosion velocity has therefore the potential advantage of reducing risks associated with the instabilities, without compromising fuel confinement. In addition, provided ignition is achieved efficiently by other means, reducing the implosion velocity could allow for significantly larger gain, since it would reduce the fuel specific energy $e = v^2/2$ (see Eq. 2). The price to be paid is that now the ignition hot spot has to be created by an additional laser pulse. We now consider separately the advanced schemes of fast ignition and shock ignition.

Fast ignition

In fast ignition (Tabak et al., 1994), the hot spot is created in a nearly homogeneous compressed fuel by an ultrapowerful particle beam. Such a beam has to produce a hot spot in an isochoric assembly, with parameters as for point B in Fig. 2, namely $\rho R_h = 0.5 \text{ g/cm}^2$ and $T_h = 12 \text{ keV}$, in a time shorter than the hot spot confinement time R_h/c_s . According to Eq. (15) the energy required scales as $E_h \propto \rho^{-2}$; pulse duration and beam radius should scale as $1/\rho$. Detailed simulations substantially confirm the above scalings, but indicate that Eq. (15) underestimates the required beam energy. The minimum beam energy, beam power, and beam intensity for fast ignition are indeed given by (Atzeni, 1999).

$$E_{\text{ig}} = 18 \hat{\rho}^{-1.85} \text{ kJ}, \quad (26)$$

$$W_{\text{ig}} = 0.9 \times 10^{15} \hat{\rho}^{-1} \text{ W}, \quad (27)$$

$$I_{\text{ig}} = 7.2 \times 10^{19} \hat{\rho}^{0.95} \text{ W cm}^{-2}, \quad (28)$$

where $\hat{\rho} = \rho / (300 \text{ g cm}^{-3})$. For a DT fuel precompressed at density $\rho = 300 \text{ g cm}^{-3}$ the igniting particle beam must deliver the above 18 kJ, in 20 ps, onto a spot of radius of $20 \mu\text{m}$. The ignition energy decreases strongly with increasing density; however, this implies reducing the beam focus and increasing beam intensity. We then take $\rho = 300 \text{ g cm}^{-3}$ as a reference density, since it is unlikely that laser beams delivering pulses of tens of kJ can be focused onto a spot with radius smaller than $15 - 20 \mu\text{m}$. An additional requirement concerns the range of the igniting particles, which should be comparable to or smaller than the desired hot spot size. The above beam parameters indeed apply to particles with mass penetration depth ρl in the range $0.15 - 1.2 \text{ g cm}^{-2}$, such as fast electrons with energy of 1–2 MeV or protons with energy of up to about 10 MeV. Notice that the above Eqs. (26)–(28) refer to deposited energy. The corresponding laser parameters should be a factor $1/\eta_{\text{ig}}$ larger, where η_{ig} is the efficiency of generation of the fast particles.

Intense electron and proton beams can be produced by ultraintense chirped-pulse-amplified (CPA) lasers (Strickland and Mourou, 1985). Electron fast ignition schemes foresee direct irradiation of the precompressed target with an ultraintense laser; see Fig. 7A. Laser interaction around critical density generates relativistic electrons, which should then reach the dense fuel and generate the hot spot. Notice that the intermediate step of propagation from the low density plasma corona to the compressed fuel involves transport of a giga-ampere current. Success of the scheme requires that hot electrons are produced with good efficiency (e.g. 20–30%), in a tightly focused, nearly parallel beam, can reach the dense fuel, and have appropriate range. Laser accelerated beams of multi-MeV protons or light ions have also been proposed as fast ignitor drivers (Roth et al., 2001). Protons can be produced by laser irradiation through many different schemes. The most studied is the TNSA mechanism (target normal sheet acceleration, from the rear side of a laser accelerated foil) (Snavely et al., 2000). A possible target configuration is shown in Fig. 7B. Being

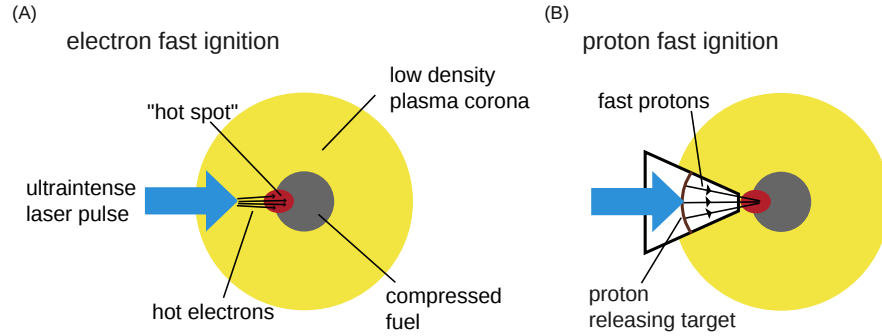


Fig. 7 Fast ignition of precompressed fuel: (A) electron fast ignition; (B) ion fast ignition.

much more massive than electrons, protons are easier to transport and focus than electrons. Feasibility of the scheme mainly concerns the efficient generation of the ultraintense proton beam.

Shock ignition

In shock ignition (Betti et al., 2007), a first laser pulse drives implosion at a velocity somewhat lower than in conventional direct-drive. At a time close to implosion stagnation the target is irradiated (still over the full solid angle, or at least a large portion of it) by a short, intense laser pulse. This pulse, of duration of a few hundred ps and intensity of about $10^{16} \text{ W cm}^{-2}$, generates an imploding shock wave of pressure of a few hundred Mbar. The shock wave strengthens as it implodes, due to spherical convergence. When this shock-wave collides with the shock wave bouncing from the center of the previously compressed fuel the pressure is further amplified and the implosion hot spot is generated. Unlike fast-ignition, shock ignition relies on well understood hydrodynamics, and laser parameters for a demonstration are in principle compatible with present technology. Physics issues, instead, concern the interaction of the intense laser pulse with the plasma. In particular, it has to be proved that pressures of 300–500 Mbar can be generated efficiently and without producing deleterious hot electrons.

Concluding remarks

In the previous sections we have discussed the basic requirements for the achievement of net energy production by inertial confinement fusion. We have also outlined the principles of the schemes proposed to first demonstrate the feasibility of ICF, and then proceed to energy production. An important question for the current phase of research concerns laser parameters required for ignition demonstration using conventional implosion schemes. According to Eq. (15), the hot spot energy at ignition scales as $E \propto p_h^{-2}$. In turn pressure p_h scales with implosion parameters following Eq. (23). If we assume that the total fuel energy at stagnation is proportional to the hot spot energy, and that laser energy is coupled to the fuel with efficiency η_c , we obtain (Kemp et al., 2001)

$$E_{\text{laser}}^{\text{ignition}} \propto \eta_c^{-1} \alpha_{\text{if}}^{1.8} u_{\text{imp}}^{-6} p_a^{-0.8}, \quad (29)$$

This scaling is in very good agreement to the fit of a large set of detailed numerical simulations (Herrmann et al., 2001):

$$E_{\text{laser}}^{\text{ignition}} = 200 \frac{0.1}{\eta_c} \alpha_{\text{if}}^{1.88 \pm 0.05} \left(\frac{u_{\text{imp}}}{350 \text{ km s}^{-1}} \right)^{-5.89 \pm 0.12} \left(\frac{p_a}{100 \text{ Mbar}} \right)^{-0.77 \pm 0.03} \text{ kJ}. \quad (30)$$

These equations evidence the critical role of both implosion velocity and adiabat. In turn, reducing the adiabat and increasing the implosion velocity increases the risks associated with RTI growth. Furthermore, Eq. (30) assumes perfect spherical symmetry and absence of material mixing induced by RTI. In general, indirect drive allows for lower adiabat, but implies much smaller η_c . Increasing the latter in indirect-drive conflicts with symmetry requirements. Eventually, only experiments (with ingenious technical solutions) will answer the questions concerning pulse energy required for ignition and selection of the target and drive concept.

Shock ignition, which can be seen as a slightly modified conventional scheme could in principle lead to ignition at somewhat lower energy than conventional direct-drive, but the physics of strong shock generation still deserves investigation. Fast ignition potentially allows for ignition (and high gain) at much lower energy, but anyhow requires ultraintense laser pulses delivering at least 50 – 100 kJ in a few ps for hot spot generation.

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Inertial Confinement Fusion – Key Elements of Plasma Physics

Vladimir T. Tikhonchuk^{a,b}, ^a CELIA, University of Bordeaux, CNRS, CEA, Talence, France; and ^b ELI-Beamlines, Institute of Physics, Czech Academy of Sciences, Dolní Břežany, Czech Republic

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Glossary

BGK Born-Green-Kirkwood approximation for the collision integral.
CH Plastic made of carbon and hydrogen.
CHIC Hydrodynamic code for simulation of ICF implosions.
HiPER High power energy research project.
RM Richtmyer-Meshkov instability.
RT Rayleigh-Taylor instability.
SNB Schurtz-Nicolai-Busquet electron heat transport model.
SSD Smoothing by spectral dispersion of laser beams.
TNSA Target normal sheath acceleration of ions.
TPD Two plasmon decay instability.
YOC Yield over clean of fusion reactions characterizing performance of ICF targets.
WKB Wentzel-Kramers-Brillouin approximation in solution of differential equations.

Abbreviations

CBET Cross beam energy transfer

DT Deuterium-tritium
 ICF Inertial confinement fusion
 IFE Inertial fusion energy
 NIF National Ignition Facility
 RPP Random phase plate
 SBS Stimulated Brillouin scattering
 SRS Stimulated Raman scattering

Introduction

Laser-driven inertial confinement fusion is a multi-physics project involving phenomena on different spatial and temporal scales, ranging from macroscopic scales of several millimeters and nanoseconds, corresponding to the fusion target size and the whole time of the process, to the picosecond time scales, corresponding to the energy transport by charged particles and down to femtosecond times and sub-micron scales, where electromagnetic processes dominate. A good comprehension of each of these processes and their interactions is indispensable for controlling the fusion reactions and producing commercially usable energy.

A detailed presentation of all related physics goes far beyond this short review, which aims to recall the basic notions of the physics at each temporal and spatial scale, and to guide the reader to up to date developments and results with a critical analysis of unresolved issues and problems. The article is divided into two sections, addressing the macroscopic and microscopic physics, respectively.

The framework of macroscopic physics is classical hydrodynamics. Under the enormous pressure produced by tightly focused high power laser beams, the target made of a solid material behaves as a compressible fluid. It can be compressed and heated to the conditions where fusion reactions can be excited and release more energy than invested by the laser. The first half of the first section considers the ideal case of spherical symmetry. It addresses issues such as: how the laser radiation can be used for creating unique (in laboratory) pressures exceeding 100 Mbar, how this pressure should be shaped in time in order to accelerate the target to velocities exceeding 300 km s^{-1} needed for efficient fuel compression, how to implode the fuel and to achieve conditions for ignition of fusion reactions.

The real world is, however, three-dimensional, and all deviations of the target from spherical symmetry need to be well-understood and properly controlled. The analysis of three-dimensional effects in inertial fusion is presented in the second half of the first section. Small initial deviations from spherical symmetry due to the target imperfections and laser intensity modulations are unstable and grow in time during the target implosion. The two major instabilities involved in the implosion process—Rayleigh-Taylor and Richtmyer-Meshkov—are, however, strongly modified by the ablation process, and this effect provides the possibility for their control and mitigation.

The second section is dedicated to the description of microscopic processes of laser propagation, absorption and scattering and of the energy transport beyond the range accessible to the laser. One of the major particularities of laser fusion is spatial separation of the zone where laser energy can be absorbed in a relatively low density and high temperature plasma and of the high density zone where fusion reactions take place. The energy is transported between these two zones by charged particles, electrons and, in some particular cases, by ions. This is a kinetic process which cannot be reduced to a classical diffusion. The methods of description of energy transport in inertial fusion plasmas are discussed in the second half of the second section. This is one of the outstanding problems of inertial fusion theory for which an adequate solution is yet to be found.

Efficient laser energy absorption is the key to achieving energy production from fusion reactions. The classical collisional absorption of electromagnetic waves in a plasma strongly decreases as the temperature increases. Laser propagation in a plasma and the methods of improving the collisional absorption are described in the first half of the second section. These include an appropriate choice of the laser wavelength and different laser beam smoothing techniques. Laser intensities required for target acceleration are very high and nonlinear processes are an indispensable part of the laser-plasma interactions. Laser interaction with plasma waves results in stimulated scattering and resonance absorption. Furthermore, coupling of plasma waves to electrons results in fast electron generation and undesired fuel preheating. Control and mitigation of these processes are the subjects of present day research. They are discussed in this review along with reduced models that are designed to represent the microscopic physics in the large scale macroscopic models.

Physics of target implosion

This section describes physical processes related to the target implosion: how the ablation pressure is created, how the fuel is compressed and how the fusion reactions are ignited. The major results consist in the relations between target parameters, ablation pressure and characteristics of the compressed fuel. These relations allow to optimize the efficiency of transfer of the absorbed laser energy to the internal energy of the assembly of the fuel. More detailed descriptions of the hydrodynamic processes in application to

ICF can be found the textbook by Atzeni and Meyer-ter Vehn (2004) and reviews by Brueckner and Jorna (1974), Lindl et al. (2004), and Craxton et al. (2015). Physics principles and schemes of fusion energy production are described in the chapter 12.20 “ICF physics principles”.

Choice of the target and the compression method

To be efficient, the compression of the target must be nearly isentropic. This can be performed by two methods: (i) either launching a converging compression wave to the center of a homogeneous sphere (ii) or by taking the fuel in a form of a thin hollow spherical shell, accelerating it to a certain velocity and letting it converge freely to the center. It is then compressed by the geometric effect of spherical convergence, and its kinetic energy is transformed into the internal energy at the moment of stagnation. The free flight phase is isentropic as the sum of the kinetic energy and of the internal energy is conserved. In both cases the fuel is accelerated by a pressure created by the laser ablation of the external part of the target, called ablator. It also plays the role of a structural element by maintaining the mechanical stability of the target. These two options of possible ICF target architectures are presented in Fig. 1.

In panel A, a gaseous DT fuel is placed inside a spherical shell made of plastic or glass, which acts as ablator and piston. The ablation pressure has to have a special temporal profile so the isentropic compression wave converges to the center of the target, producing there the density and temperature needed for ignition, as described by Kidder (1974). The major advantage of this design is that the target can be maintained at room temperature, but it is very fragile because of a small ablator thickness and high gas pressure inside. Moreover, the ablation pressure needs to be strongly increased at the end of implosion, which imposes strong constraints on the laser system, and the imploded fuel has a cold core and a hot external part. This is not the fuel assembly that is needed for ignition of fusion reactions with a hot core and a cold periphery. Therefore, such targets are used in the physical experiments dedicated to studies of laser absorption and shock formation, but not suited for the energy production.

The second design (panel B) is better suited for the production of energy. Here, the DT fuel is deposited as a thin shell inside the ablator. The residual gas pressure inside could be very low and the target is structurally more stable, but it has to be maintained at a cryogenic temperature of 17 K, because of a very low melting temperature of hydrogen. This design has an advantage of relaxing constraints on the laser power—the shell compression is separated in two stages of acceleration and free flight, and it naturally produces the desired fuel assembly with a hot spot in the center and a cold shell outside. The first step of the shell acceleration can be accomplished by a succession of shocks. This method allows controlling the shell entropy as it discussed in section “Shell acceleration by a succession of shocks.” The second step is an isentropic free flight, and the desired compression is reached by the geometric convergence. The hot spot conditions are defined by the shell implosion velocity and the quantity of gaseous DT inside the shell.

Creation of the ablation pressure by laser

Conversion of the laser pulse energy incident on the target surface into the shell kinetic energy proceeds in several subsequent steps. Firstly, a front part of the laser pulse is absorbed in the outer layer of the target. There, the material is heated, vaporized and transformed in a low density, hot expanding plasma. At the second step, the remaining part of the laser pulse propagates through this plasma and is partially absorbed near its reflection point, called the critical density. For the laser wavelengths in the optical domain, that is, from 0.3 to 1 μm , the plasma critical density is very low, 10–100 times smaller than the solid density. The absorbed laser energy is transported further from this low density hot plasma to the cold solid material by electrons and partially by X-ray radiation. The energy flux arriving on the cold dense material causes its ablation and ejection of the vapors. This is the final step of the laser energy absorption process—the reaction force of ablated vapors creates a pressure on the remaining solid part of the shell.

For the laser intensities of interest for inertial fusion, that is, $I_{\text{las}} \sim 10^{14} - 10^{15} \text{ W cm}^{-2}$, the plasma expansion velocity is about 300 km s^{-1} , and the ablation process becomes quasi-stationary on the nanosecond time scale. It is shown schematically in Fig. 2. Laser energy is absorbed in the hot and underdense corona near the critical surface. The energy flux is then transported

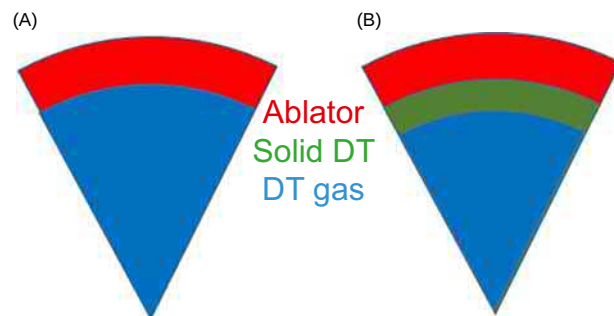


Fig. 1 Target architecture for inertial fusion with gaseous (A) and solid (B) fuel. The external part of target (red) serves as ablator and structural material. A DT may be in a gaseous (blue) or in a solid (green) form.

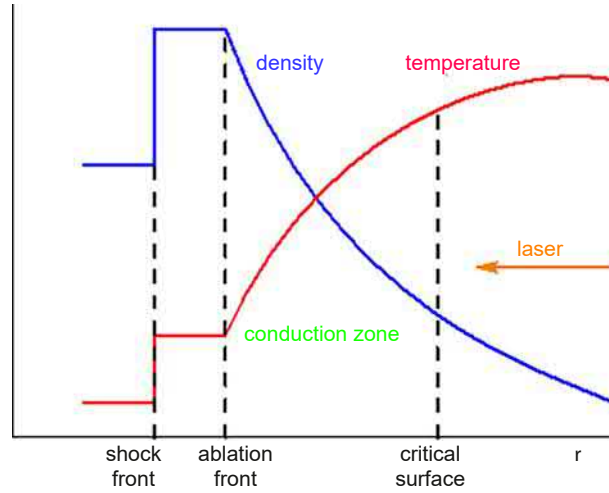


Fig. 2 Profiles of density (blue) and temperature (red) of the target irradiated by a laser (coming from the right): dashed lines show the radial position of the critical surface, ablation front and shock front. The conduction zone separates the plasma corona to the right and the shocked target to the left.

through the conduction zone to the compressed shell by electrons and photons. It is balanced by the outflow of hot ablated material. Behind the ablation front, there is a shock wave propagating in the solid shell.

Although the fluxes of energy in the laser-irradiated target are equilibrated, this is not true for the mechanical momentum. The pressure of laser radiation is negligible in the ICF conditions. Indeed, assuming that all laser photons are absorbed, the radiation pressure, I_{las}/c , is less than 1 Mbar, while ablation pressure could be 100 times larger.

The ablation pressure can be calculated from the law of energy flux conservation at the target surface. Assume that a fraction η_{abs} of the incident laser radiation is absorbed in the plasma near the critical surface. The absorbed energy is transformed in a plasma outflow ejected from the target with the velocity u , density ρ and temperature T . The conservation of the energy flux reads:

$$I_{\text{abs}} = \eta_{\text{abs}} I_{\text{las}} = \left(\frac{1}{2} \rho u^2 + \frac{5}{2} \rho c_s^2 \right) u \quad (1)$$

The left hand side in this relation presents the absorbed energy flux. The explicit expression for η_{abs} is discussed in section “**Laser propagation in an inhomogeneous plasma**,” devoted to the laser plasma interaction physics. The right hand side presents the flux related to the plasma kinetic energy and enthalpy. Here, $c_s = ((ZT_e + T_i)/m_i)^{1/2}$ is the local ion acoustic velocity, Z is the ion charge, m_i is the ion mass and $T_{e(i)}$ is the electron (ion) temperature. As the thickness of the conduction zone is smaller than the shell radius, we consider here a planar expansion. Then, the mass conservation implies $\rho u = \text{const}$.

While Eq. (1) is valid throughout the conduction zone, there is an additional relation, called the Chapman-Jouguet condition, which follows from the stability of the plasma expansion in the case of localized energy deposition (see [Atzeni and Meyer-ter Vehn \(2004\)](#) for more details). Namely, a transition from subsonic to super-sonic expansion may take place only in the domain where the laser energy is absorbed, that is, near the critical density. Therefore, Eq. (1) has to be complemented by a relation, $u = c_{s,\text{cr}}$ at $\rho = \rho_{\text{cr}}$, and it then takes the following form: $I_{\text{abs}} = 3\rho_{\text{cr}}c_{s,\text{cr}}^3$.

This relation needs one more correction before applying it to the ICF target. In fact, this relation assumes that all absorbed laser energy goes into the conduction zone, which is not the case. The laser corona is not adiabatic, but isothermal, because of a high electron mobility. Therefore, a part of the absorbed laser energy is transported by electrons downstream of the critical density. Therefore, to the expression in the right hand side of this relation one should add the electron heat flux, $\rho_{\text{cr}}c_{s,\text{cr}}^3$, corresponding to the isothermal rarefaction wave in a hot plasma corona. Thus, we obtain the following relation between the absorbed laser energy and the plasma parameters at the critical density:

$$I_{\text{abs}} = 4\rho_{\text{cr}}c_{s,\text{cr}}^2 \quad (2)$$

Here, the critical density is defined by equating the laser frequency, ω_{las} , to the local electron plasma frequency, ω_{pe} (see section “**Laser propagation in an inhomogeneous plasma**” for further details). It can be conveniently expressed as $\rho_{\text{cr}} = n_{\text{cr}}m_i/Z$, where $n_{\text{cr}} = 1.1 \times 10^{21} \lambda_{\text{las}}^{-2} \text{ cm}^{-3}$ is the electron critical density and λ_{las} is the laser wavelength in microns.

Eq. (2) provides a relation of the plasma acoustic velocity at the critical density, $c_{s,\text{cr}} = (I_{\text{abs}}/4\rho_{\text{cr}})^{1/3}$ (it is called also a blowoff velocity, u_{bf}), and, consequently, the ablation rate, $\dot{m}_a = \rho_{\text{cr}}c_{s,\text{cr}}$, to the laser parameters. As the plasma pressure is approximately constant in the conduction zone, one can also evaluate the ablation pressure on the target as a momentum flux, $p_{\text{abl}} \approx \rho_{\text{cr}}c_{s,\text{cr}}^2$. In practical units and for a plastic (CH) plasma one can deduce the following expressions:

$$T_{e,\text{cr}} = 13.7 I_{\text{abs}}^{2/3} \lambda_{\text{las}}^{4/3} \text{ keV}, \quad \dot{m}_a = 3.2 \times 10^5 I_{\text{abs}}^{1/3} \lambda_{\text{las}}^{-4/3} \text{ g s}^{-1} \text{ cm}^{-2}, \quad p_{\text{abl}} = 57 I_{\text{abs}}^{2/3} \lambda_{\text{las}}^{-2/3} \text{ Mbar} \quad (3)$$

where the laser intensity is in units of $10^{15} \text{ W cm}^{-2}$ and a laser wavelength is in microns. Consequently, a laser with such intensity at a wavelength of $\sim 0.3\text{--}0.5 \text{ }\mu\text{m}$ may create pressure on the order of 50–100 Mbar, as needed for an efficient ICF implosion.

It is important to notice that the ablation pressure and the mass ablation rate are inversely proportional to the laser wavelength. The shorter the laser wavelength, the more efficient is the laser action. For this reason and because of a better absorption, the short wavelength laser systems are of major interest for ICF. Such a dependence of the ablation pressure on laser wavelength motivated the indirect drive approach, where the laser radiation is transformed into black-body X-rays in a cavity, known as a *hohlraum*, with a target inside it, see the chapter 12.20 “ICF physics principles” and Lindl et al. (2004). X-ray radiation penetrates deeper into the target and deposits its energy directly in the solid shell. Moreover, black-body radiation is naturally isotropic and thus produces a more homogeneous ablation. A similar indirect drive approach is used in heavy ion inertial fusion, where the energy of an accelerated ion beam is converted into the X-ray radiation which is then directed onto a spherical pellet.

Creation of the ablation pressure with thermal X-rays

X-ray radiation used in the indirect drive approach can be considered as black body radiation. Ablation pressure and mass ablation rate in this case can be also evaluated from Eq. (2). The absorbed energy flux is related to the X-ray black-body temperature, T_X , as $I_{\text{abs}} \approx \sigma_{\text{SB}} T_X^4$, where $\sigma_{\text{SB}} \approx 10^5 \text{ W cm}^{-2} \text{ eV}^{-4}$ is the Stefan-Boltzmann constant. The position of the absorption zone in a stationary regime can be found from the condition that the electrons are in thermal equilibrium with radiation, that is, $T_e = T_X$. Consequently, the plasma blowoff velocity in the absorption zone is known, $u_{\text{bf}} = (ZT_X/m_i)^{1/2}$, and Eq. (2) defines thus the blowoff plasma density, $\rho_{\text{bf}} = \sigma_{\text{SB}} T_X^4 / 4u_{\text{bf}}^3$, which is about 1 g cm^{-3} for the radiation temperature of 300 eV.

From these relations one obtains directly expressions for the mass ablation rate and ablation pressure in practical units:

$$\dot{m}_a = 1.4 \times 10^5 \left(\frac{T_X}{300 \text{ eV}} \right)^3 \text{ g s}^{-1} \text{ cm}^{-2}, \quad p_{\text{abl}} = 170 \left(\frac{T_X}{300 \text{ eV}} \right)^{7/2} \text{ Mbar} \quad (4)$$

So, in order to produce radiation pressure at the level of 100 Mbar, needed for ICF implosion, one has to create black-body radiation with a temperature of 300 eV. This corresponds to the energy density of about 10^5 J cm^{-3} , that is, one needs to deposit a laser energy of about 100 kJ in a cavity of a volume of 1 cm^3 . The plasma density where absorption takes place is comparable to the initial target density, which is favorable for efficient ablation.

Shell acceleration by ablation pressure

Because of shell compressibility, application of an ablation pressure does not lead to an instantaneous acceleration. Ablation pressure creates a shock wave that propagates inside the target. It is shown in Fig. 3 as a zone to the left of the ablation front. Only after the shock traverses the shell, is the latter moving as a whole with a fluid velocity behind the shock u_s . For a material with a polytropic index γ , the velocity D_s of a strong shock and the downstream velocity u_s are defined by the ablation pressure and the shell initial density ρ_0 :

$$D_s = \sqrt{\frac{\gamma + 1}{2} \frac{p_{\text{abl}}}{\rho_0}}, \quad u_s = \frac{2}{\gamma + 1} D_s \quad (5)$$

In the particular case of a mono-atomic gas with $\gamma = 5/3$, the shock velocity is $D_s = 1.15 (p_{\text{abl}}/\rho_0)^{1/2}$ and $u_s = 0.87 D_s$.

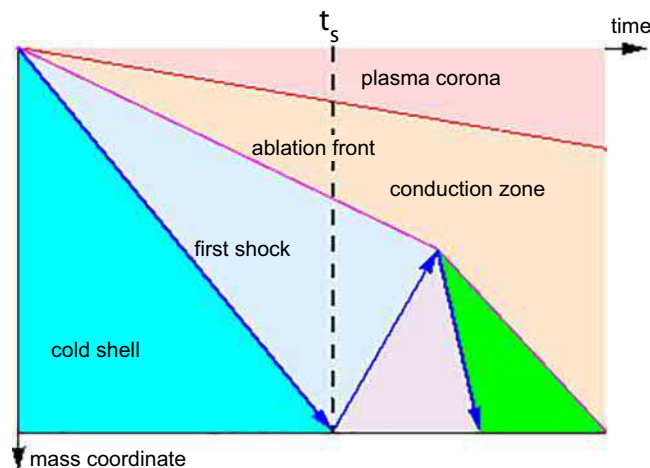


Fig. 3 Target acceleration shown in mass coordinates: light red—plasma corona, yellow—conduction zone, cyan—cold shell, light blue—compressed target behind the first shock, rose—rarefaction wave, green—target compressed by the second shock. Red line shows the trajectory of critical surface, magenta—trajectory of the ablation front, blue—trajectories of the shock and rarefaction waves.

One can make a distinction between two regimes of laser-target interaction. If the shell thickness, d_s , is sufficiently large and laser energy deposition ends before the shock crosses the shell, the latter is compressed but does not move. Otherwise, if the laser pulse duration is longer than d_s/D_s , the target as a whole gains a velocity u_s . This latter regime is the one of interest for ICF.

The target acceleration scheme is presented in Fig. 3. The first shock propagates with a velocity, D_{s1} , and arrives at the rear shell interface at the time t_s . It is reflected there and propagates back as a rarefaction wave. The shell behind the first shock is already compressed and accelerated to the velocity, u_{s1} , defined by the ablation pressure. When the rarefaction wave arrives at the ablation front it is reflected as a second shock. It propagates through the shell with the velocity $D_{s2} > D_{s1}$, compresses it and accelerates the shell once more to the velocity $u_{s1} + u_{s2}$. This process of consecutive propagation of the shock and the rarefaction waves across the shell repeats itself several times. After each shock reflection the shell gains an additional velocity. So the shell acceleration proceeds as a step-like process. If the shell is sufficiently thin and the shocks are crossing it many times, one can consider the interaction as in a continuous acceleration of the shell as a whole. This is described by a rocket model presented in section “Shell acceleration by ablation pressure”. The acceleration ends in the moment when the ablation front reaches the rear side of the shell.

Shell acceleration by a succession of shocks

The target acceleration is a non-isentropic process. The shock dissipates its energy and produces entropy in the compressed material. The larger the shock amplitude, the more entropy is injected in the fuel. Thus, the first shock defines a large part of the entropy of the shell. However, material compression in a shock is limited to a factor $(\gamma + 1)/(\gamma - 1)$, which is 4 for a mono-atomic gas, where $\gamma = 5/3$. That is much less than one needs for the ICF implosion. Thus, it is important to understand how one can accelerate a shell and compress it to high densities without injecting too much entropy.

Suppose that the initial state of the shell is characterized by a pressure p_0 and a density ρ_0 . If it would be compressed by an external pressure, p isentropically, its density will increase according to the power law:

$$\rho = \rho_0 \left(\frac{p}{p_0} \right)^{1/\gamma} \quad (6)$$

However, to realize such a compression, the pressure should be applied to all material at the same time. Instead, the ablation pressure is applied to the edge of the shell. It produces a shock, which deposits in the material a certain quantity of energy, and compression in a shock is described by the Rankine-Hugoniot relation:

$$\rho = \rho_0 \frac{(\gamma + 1)p + (\gamma - 1)p_0}{(\gamma - 1)p + (\gamma + 1)p_0} \quad (7)$$

The curves described by Eqs. (6), (7) departing from the same point are shown in Fig. 4. The isentropic curve (red) allows an arbitrary strong compression depending on the applied pressure. Conversely, the Rankine-Hugoniot relation (black) limits the maximum compression by a factor $(\gamma + 1)/(\gamma - 1)$. Thus, one shock cannot compress the target more than a factor of 4. (A stronger compression is possible in radiatively cooling shocks, but this is not suitable for ICF targets that use low atomic number materials.)

A much more efficient compression can be achieved with a succession of several shocks. Fig. 4 shows a comparison between the compression with one shock with a pressure jump, $p/p_0 = 216$, and a succession of three shocks of an equal strength, $p_1/p_0 = p_2/p_1 = p_3/p_2 = 6$, with the same total pressure jump, $p_3/p_0 = 6^3 = 216$. Each shock creates a smaller compression, $\rho_1/\rho_0 = \rho_2/\rho_1 = \rho_3/\rho_2 = 2.5$, but a sequence of three shocks allows the achievement of a much higher compression of $\rho_3/\rho_0 = 2.5^3 \approx 15.6$, compared to 3.9 in the single shock case, while maintaining the same overall pressure jump.

This difference is due to a smaller target entropy in the second case. It can be characterized by a dimensionless factor called adiabat, $\alpha = p_0^*/p_0$. Here, p_0^* is the initial pressure of the isentrope corresponding to the final state p_3, p_3 . It is shown with a dashed red line

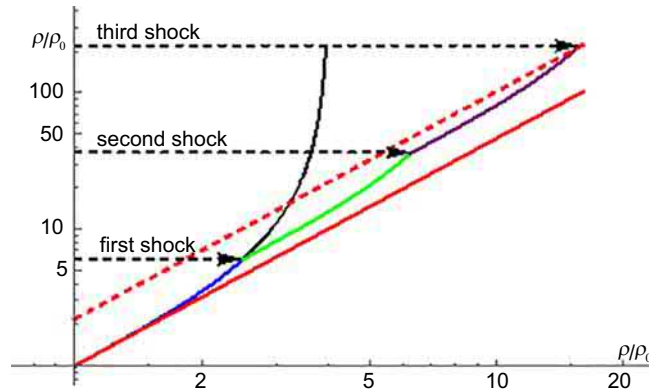


Fig. 4 Compression diagram of a plane target described as a mono-atomic gas: red—isentropic compression from the initial pressure; dashed red—isentrope corresponding to the initial pressure $p_0^* = 2.2p_0$; black—Rankine-Hugoniot adiabat; blue, green, purple—compression with a sequence of three shocks with a strength $p_{i+1}/p_i = 6$.

in Fig. 4. Adiabatic characterizes how much entropy has been introduced into the material downstream of the shocks. In our example with three shocks, the initial pressure corresponding to the new isentrope is 2.2 times higher than p_0 , and therefore the succession of three shocks increases the adiabat by a factor of $\alpha = 2.2$. This should be compared with the factor 22 of the entropy increase in the case of a single shock with the same pressure jump. For the ICF target compression, the shell initially is cold and p_0 corresponds to the pressure of degenerate electrons. Thus, α characterizes how much entropy is introduced by the shocks into the shell compared to the ideal isentropic compression. One says that a sequence of three shocks puts the shell on the adiabat $\alpha = 2.2$.

Fig. 4 shows that each subsequent shock has a higher pressure and correspondingly propagates faster than the preceding one. So, it is possible to adjust the launch time of each subsequent shock in such a way that all of them merge at the rear side of the shell. This condition corresponds to the most efficient shell acceleration with the minimum deposited entropy. Thus, the timing of the shock launch is the most important adjustable parameter of the implosion process. It depends on the laser intensity and wavelength, the thickness and the material of the shell, and it is chosen by using theoretical models and numerical simulations.

Rocket model

The rocket model describes a thin shell acceleration as a continuous process, considering it as a rigid object with a variable mass. We, therefore, neglect the time that it takes the shock to cross the target, assuming that the acceleration process takes a longer time. Although this is not exactly the situation in the ICF targets, the rocket model is a convenient approximation that describes the major features of the process.

Shell acceleration is described by the laws of conservation of mass and momentum in a system containing a solid target and plasma ejecta. The model is similar to the equations describing the acceleration of a rocket. In the case of a planar foil, this equation applies to a unity of surface with the areal mass M_σ :

$$\frac{d}{dt}(M_\sigma U_{\text{imp}}) = \dot{m}_a(u_{\text{bf}} - U_{\text{imp}}) \quad (8)$$

Here, the right hand side describes the change of the momentum of the target, $M_\sigma U_{\text{imp}}$, due to ejection of a plasma with a relative velocity $u_{\text{bf}} - U_{\text{imp}}$. A decrease of the target mass is defined by the ablation rate (3):

$$dM_\sigma/dt = -\dot{m}_a \quad (9)$$

Excluding the time derivative from the first equation by using the mass conservation, we obtain a relation between the target velocity and its mass, $dU_{\text{imp}}/dM_\sigma = -u_{\text{bf}}/M_\sigma$. Assuming a constant blowoff velocity, this equation can be integrated giving the target velocity as a logarithmic function of the mass, which decreases linearly with the time:

$$M_\sigma = M_{\sigma 0} - \dot{m}_a t, \quad U_{\text{imp}} = -u_{\text{bf}} \ln \frac{M_{\sigma 0}}{M_\sigma} \quad (10)$$

A negative sign in the second equation indicates that the target is accelerated in a direction opposite to that of the ejection of vapors. The maximum time of the acceleration process depends on the target thickness, $t_{\text{acc,max}} = M_{\sigma 0}/\dot{m}_a$, where $M_{\sigma 0}$ is the initial target areal mass.

According to Eq. (10), one can accelerate target to an arbitrary large velocity, but the efficiency of this process depends on the ratio of the residual payload to the initial mass $X = M_\sigma/M_{\sigma 0}$. The mechanical efficiency of the acceleration process can be defined as the ratio of the kinetic energy of the remaining solid part of the shell, $\frac{1}{2}M_\sigma U_{\text{imp}}^2$ to the kinetic energy of ejecta:

$$\eta_{\text{acc}} = \frac{M_\sigma U_{\text{imp}}^2}{(M_{\sigma 0} - M_\sigma) u_{\text{bf}}^2} = \frac{X \ln^2 X}{1 - X}. \quad (11)$$

It is shown in Fig. 5A. The maximum efficiency of 65% is achieved when 80% of the initial mass is ablated. In that particular case the target speed is $\ln 5 = 1.6$ times larger than the ejecta velocity.

Implosion of a spherical shell

The rocket model can be readily modified to account for the spherical geometry. The velocity of a shell of a mass M , radius R and thickness $\Delta R \ll R$ is defined as $U_{\text{imp}} = dR/dt$. Then Eqs. (8), (9) take the following form:

$$\frac{d}{dt}(MU_{\text{imp}}) = 4\pi R^2 \dot{m}_a(u_{\text{bf}} - U_{\text{imp}}), \quad \frac{dM}{dt} = -4\pi R^2 \dot{m}_a \quad (12)$$

Excluding the time derivative from these two equations, we find again a logarithmic relation between the shell velocity and the mass, (10). That leaves us with one differential equation relating the shell mass to the radius:

$$\ln X \frac{dX}{dR} = -4\pi R^2 \frac{\dot{m}_a}{u_{\text{bf}} M_0} \quad (13)$$

It depends on a dimensionless parameter:

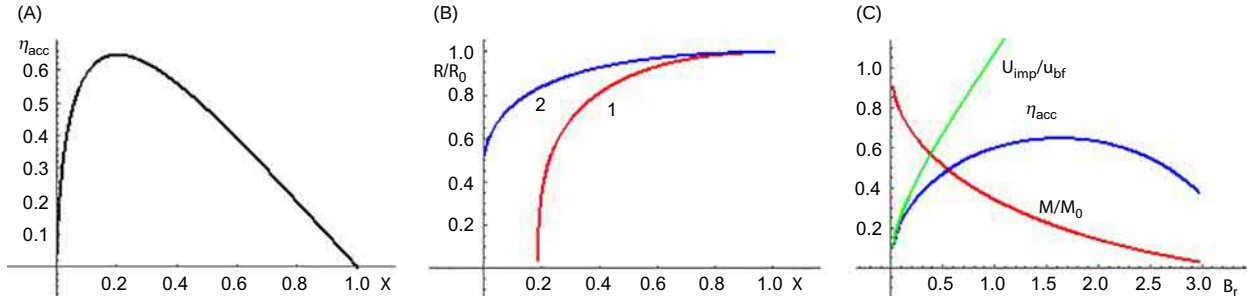


Fig. 5 (A) Mechanical efficiency of a planar foil acceleration by the ablation pressure, η_{acc} (11) as a function of the residual mass, $X = M_r/M_{r0}$. (B) Radius of a spherical shell accelerated by the ablation pressure as a function of the residual mass, X , for the value of implosion parameter (14) $B_r = 1.5$ (1, red) and $B_r = 3.5$ (2, blue). (C) Parameters of an imploding spherical shell at the moment when its radius, R , decreased two times with respect to the initial value, R_0 , as a function of the implosion parameter B_r : energetic efficiency (blue), ablated mass (red) and shell velocity (green).

$$B_r = \frac{4\pi\dot{m}_a R_0^3}{u_{\text{bf}} M_0} = \frac{\dot{m}_a R_0}{\rho_1 u_{\text{bf}} \Delta R_1} \quad (14)$$

Here, we accounted for the relation between the shell mass and its thickness, $M_0 = 4\pi\rho_0 R_0^2 \Delta R_0$, and for the fact that the acceleration starts after the first shock crosses the shell and compresses it from the initial value, ρ_0 , to the value, ρ_1 , while its areal mass remains the same, $\rho_0 \Delta R_0 = \rho_1 \Delta R_1$. According to section “**Rocket model**”, the maximum compression in the case of an ideal mono-molecular gas is 4, but in real experiments the solid shell is compressed by $\rho_1/\rho_0 \approx 2 - 3$ times.

Parameter B_r has a simple interpretation as a ratio between the time of the shell collapse to the center and the burnthrough time. The solution of Eq. (14) gives an implicit relation between the residual mass of the shell and its radius

$$1 + X(\ln X - 1) = \frac{B_r}{3} \left(1 - \frac{R^3}{R_0^3} \right) \quad (15)$$

The left hand side term in this equation is limited in the interval between 0 and 1. The term in parenthesis in the right hand side is limited to the same interval. Thus, the value of B_r defines the final state of the acceleration process. The dependence of the position of the shell as a function of the shell mass, $R(X)$, is presented in Fig. 5B. For $B_r < 3$, the shell collapses in the center before it is completely ablated. Conversely, for $B_r > 3$, the target is too thin, or the ablation rate is too high, so the target mass goes to zero before the shell reaches the center. Thus, the value $B_r = 3$ separates two regimes of shell acceleration.

It can be seen in Fig. 5B that a notable shell acceleration occurs when the shell radius decreases by at least a factor of 2. At that time, acceleration can be stopped and the shell will continue its implosion in free flight under its own inertia. Fixing the shell convergence at $R_f/R_0 = 1/2$, one can solve the system (13) numerically and obtain the efficiency of the implosion as a function of the shell parameter B_r , shown in Fig. 5C. The range $0.5 < B_r < 2.5$ corresponds to the most efficient implosion. The limit $B_r < 0.5$ corresponds to a very light target, and the case $B_r > 2.5$ to a very massive target. The value $B_r = 1.6$ corresponds to the maximum hydrodynamic efficiency of 65%. In that case the residual mass is 20% of the initial mass, and, according to Eq. (10), the shell implosion velocity $U_{\text{imp}} = 1.6 u_{\text{bf}}$.

Practically, it is too risky to vaporize 80% of the total mass. The shell becomes too fragile and can be easily destroyed by hydrodynamic instabilities. Aiming on a convergence factor of the order of 10, it is reasonable to vaporize not more than half of the initial shell mass. As a consequence, the implosion parameter in the direct drive scheme cannot be much bigger than $B_r = 0.5$. This, however, reduces the acceleration efficiency to about 50%. The shell speed is then only $0.7 u_{\text{bf}}$.

From this analysis we can obtain a practical formula for the shell implosion velocity. According to Fig. 5C, shell velocity depends approximately linearly on the implosion parameter B_r . Therefore, according to Eq. (14), $U_{\text{imp}} \approx B_r u_{\text{bf}} = \dot{m}_a R_0 / \rho_1 \Delta R_1$. This formula can be further simplified recalling that the ablation rate is related to the ablation velocity, $D_a = \dot{m}_a / \rho_1$. The parameter $A_{\text{if}} = R_0 / \Delta R_1$ is called the shell in-flight aspect ratio. It is different from the shell initial aspect ratio, $A_0 = R_0 / \Delta R_0$, because of the shell compression in the first shock. Using the definition of the aspect ratio, we obtain a simple relation between the implosion and ablation velocities:

$$U_{\text{imp}} = D_a A_{\text{if}} \quad (16)$$

Thus, one can increase the implosion velocity by an appropriate choice of the shell aspect ratio. It depends on the initial aspect ratio and the amplitude of the first shock. The latter is limited by the condition on the shell entropy and adiabat parameter. A cold shell can be assimilated to a degenerate electron gas and its pressure can be estimated as a Fermi pressure, $p_F = (3\pi^2)^{2/3} \hbar^2 n_e^{5/3} / 5m_e$ Landau and Lifshitz (1980), where $\hbar$ is the Planck constant and m_e is the electron mass. For the DT gas of a density ρ expressed in g cm^{-3} , the Fermi pressure reads $p_F = A_F^{5/3} \text{ Mbar}$, with $A_F = 2.16$. The density of solid DT is 0.25 g cm^{-3} and the corresponding Fermi pressure is $\sim 0.2 \text{ Mbar}$. Assuming a compression factor of ~ 3 and an in-flight adiabat parameter $\alpha_{\text{if}} = 2$, the first shock pressure is 1.3 Mbar . According to Eq. (3), that ablation pressure corresponds to relatively low laser intensities $\sim 3.5 \times 10^{12} \text{ W cm}^{-2}$. By contrast, in order to achieve implosion velocities of about 300 km s^{-1} needed for fuel ignition, according to Eq. (16), one needs

more than two orders of magnitude higher laser intensities. This example shows that a careful choice of the laser pulse temporal profile is very important for an efficient implosion.

Mechanical efficiency as defined by the rocket model, Eq. (11) and Fig. 5, accounts only for the kinetic energy of the blowoff plasma. However, as is discussed in section “Creation of the ablation pressure by laser”, the internal energy of the ejected plasma is larger than its kinetic energy. According to Eqs. (1), (2), in the Chapman-Jouguet point, the plasma density is equal to the critical density, ρ_{cr} , and its hydrodynamic velocity, u_{bf} , is equal to the ion acoustic velocity, $c_{s,cr}$. Thus, the total plasma energy, $(M_0 - M)\left(\frac{1}{2}u_{bf}^2 + \frac{3}{2}c_{s,cr}^2\right) = 2(M_0 - M)u_{bf}^2$, is four times its kinetic energy. Consequently, the hydrodynamic efficiency of the ablative acceleration is four times smaller than the mechanical efficiency, and is given by:

$$\eta_{hydro} = \frac{MU_{imp}^2}{MU_{imp}^2 + 4(M_0 - M)u_{bf}^2} = \frac{\eta_{acc}}{4 + \eta_{acc}} \quad (17)$$

The maximum mechanical acceleration efficiency of 65% obtained in the rocket model corresponds thus to only 16% of the total hydrodynamic efficiency. In reality, there are additional losses related to: (i) incomplete laser energy absorption; (ii) vaporization and ionization of ejecta and (iii) radiation cooling of expanding plasma, which further reduce the achievable level of hydrodynamic efficiency. In practice its value is less than 8%.

Shell deceleration and hot spot formation

After compression by the first shock and further acceleration by the ablation pressure according to the rocket model, the shell enters the coasting phase: a free flight toward the center. As is discussed above, it is reasonable to end acceleration at the moment when shell radius decreases by approximately a factor of 2, from R_0 to $R_1 \approx \frac{1}{2}R_0$. Assuming a constant in-flight aspect ratio, the fuel density at the acceleration phase is increased by a factor of $(R_0/R_1)^2 \approx 4$. Moreover, the shell is already compressed before the deceleration phase by a factor of $\sim 2-3$ in the first shock. Therefore, the shell is compressed by a factor of about 10 at this stage, $\rho_1 \approx 10\rho_0$. Accounting also for the increase of adiabat by a factor of approximately 2, the shell pressure, $p_1 \approx \alpha A_F \rho_1^{5/3}$, is increased by more than a factor of 100 to $\sim 30-50$ Mbar. Although the acceleration is terminated, the laser pulse intensity has to be maintained at a constant level in order to prevent the shell from expanding.

The shell initial deceleration phase can be described by assuming the pressure to be homogeneous across the shell. The free flight is an isentropic process and, therefore, the kinetic energy gained by the shell during acceleration is gradually converted into its internal energy as the shell decelerates. This process is described by a dynamic equation:

$$M_{sh} dU_{sh}/dt = -4\pi R_{sh}^2 (p_{sh} + p_h) \quad (18)$$

where $U_{sh} = dR_{sh}/dt$ is the shell velocity and M_{sh} is the shell mass, which is approximately constant at this phase. The right hand side accounts for the shell internal pressure, which is considered as a partially degenerate electron gas, $p_{sh} = \alpha_{if} A_F \rho_{sh}^{5/3}$, and the pressure of the DT gas inside, which can be considered as an ideal classical gas, $p_h = 2n_i T_h$, assuming equal electron and ion temperatures.

Let us first consider an empty shell. It slows down under its own internal pressure and stops at a minimum radius, R_c , when the pressure work, $4\pi \int R_{sh}^2 p_{sh} dR$, is equal to its initial kinetic energy, $\frac{1}{2} M_{sh} U_{imp}^2$. Assuming the shell thickness and adiabat are constant during the flight, the shell density is inversely proportional to the square of the radius, then one can express the maximum converging factor $C_R = R_1/R_c$ as follows:

$$C_R = \left(1 + M_{sh} U_{imp}^2 / 4 A_{if} \mathcal{E}_{sh0}\right)^3 \quad (19)$$

where $\mathcal{E}_{sh0} = 6\pi p_1 R_1^3 / A_{if}$ is the shell internal energy at the end of the acceleration phase. According to this expression, the shell compression is proportional to the cube of the ratio of shell kinetic and internal energy.

Hot spot energy balance

In the ICF schemes with ignition from a central hot spot, the internal gas pressure dominates the shell pressure at the end of implosion. The hot spot is created during implosion, when the first shock exits the shell and enters the gas inside. Because of a low gas density, $\rho_g \sim 0.1 \text{ mg cm}^{-3}$, the transmitted shock pressure decreases by a factor $\rho_g/\rho_{sh} \sim 10^{-3}$, but it increases as the shock converges to the target center. The reflection of a spherical shock is described by a self-similar solution of the equations of ideal hydrodynamics found by Guderley (1942), see also Atzeni and Meyer-ter Vehn (2004). According to this solution, after reflection, the DT gas pressure, p_{h0} , increases by a factor of 40 to about 0.1 Mbar and the gas is heated to a temperature of about 100 eV. Subsequent hot spot evolution takes place during the implosion phase. For simplicity, in the following estimates the shell is considered as a rigid piston.

The hot spot dynamics are described by the equation of energy conservation. The gas implosion is subsonic as the implosion velocity is small compared to the acoustic velocity in the hot spot. Consequently, we may account only for the gas internal energy, $\frac{3}{2} p_h V_h$, with p_h being homogeneous across the hot spot of a volume $V_h = \frac{4}{3} \pi R_{sh}^3$.

There are two sources of increase of the hot spot energy: the work performed by the piston, $-4\pi R_{sh}^2 p_h U_{sh}$, and the energy released in fusion reactions and deposited in the hot spot. There is also a channel of energy loss: bremsstrahlung emission. The shell is typically transparent for the fusion neutrons and X-ray photons. Consequently, the energy equation for the hot spot takes the form:

$$(3/2)d(p_h V_h)/dt + 4\pi R_{sh}^2 p_h U_{sh} = (W_\alpha - W_{br})V_h \quad (20)$$

The rate of bremsstrahlung emission, W_{br} , is proportional to the product of electron and ion densities, $n_e = n_i$, and increases as the square root of temperature. For a DT plasma it can be written as

$$W_{br} = 5.3 \times 10^{-31} n_e^2 T_e^{1/2} \text{ W cm}^{-3} \quad (21)$$

where plasma density, n_e , is in units of cm^{-3} and temperature is in keV, see [NRL \(2007\)](#).

Assuming equipartition of deuterium and tritium with densities, $n_D = n_T = \frac{1}{2}n_e$, the rate of energy release by α -particles having energy, $\varepsilon_\alpha = 3.5$ MeV, can be written as $W_\alpha = \frac{1}{4}n_e^2 \varepsilon_\alpha \langle \sigma_{DT} v \rangle$. The last term in this expression is the rate of DT reactions, which depends on the ion temperature, T_i . Near the ignition threshold, in the temperature range between 8 and 20 keV, it can be interpolated as a quadratic function, $\langle \sigma_{DT} v \rangle \approx 1.1 \times 10^{-18} T_i^2 \text{ cm}^3 \text{ s}^{-1}$, where T_i is keV. Consequently, the energy release rate can be presented in the form:

$$W_\alpha \approx 1.5 \times 10^{-31} n_e^2 T_i^2 \text{ W cm}^{-3} \quad (22)$$

where density is in the units of cm^{-3} and temperature in keV.

From these expressions one can obtain several conclusions. Firstly, by comparing the two terms in the right hand side of Eq. (20), we find the minimum hot spot temperature needed for ignition fusion reactions. The condition $W_\alpha > W_{br}$ defines the minimum temperature, $T_{\min} \approx 4$ keV. Secondly, above this temperature, the rate of fusion reactions increases so rapidly that one can neglect the radiation losses and analyze Eq. (20) without the loss term. Then, noting that the product $n_i T_i$ is equal to $\frac{1}{2} p_h$, assuming equal temperatures of electrons and ions, the energy balance in the hot spot takes the form:

$$dp_h/dt = -5p_h U_{sh}/R_{sh} + S_\alpha p_h^2 \quad (23)$$

where $S_\alpha = 0.1 \text{ ns}^{-1} \text{ Gbar}^{-1}$. It is important to note that this equation does not contain the terms of energy exchange between the hot spot and the shell. This is explained by the fact that energy transferred by electrons and α -particles to the shell is recovered as the flow of ablated mass to the hot spot. Thus, the mass of the hot spot increases with time, but it does not have an effect on the hot spot pressure, which depends only on the compression work and the fusion energy production, see [Betti et al. \(2002\)](#).

The two terms in the right hand side of Eq. (23) represent the piston work and the fusion energy production. The piston work is reversible: the hot spot pressure increases as the fifth power of compression and then decreases. By contrast, the fusion energy production is irreversible: if this term dominates, the pressure increases explosively, $p_h = p_{h0}/(1 - p_{h0} S_\alpha t)$. This explosive behavior can be identified as ignition. Thus, by comparing the two terms in the right hand side we can define the ignition threshold as

$$p_{h,ig} \approx 5U_{imp}/R_c S_\alpha \quad (24)$$

Here, for the estimate of implosion time we consider the shell implosion velocity and its radius at the stagnation. For typical values of $U_{imp} \sim 300 \text{ km s}^{-1}$ and $R_c \sim 50 \text{ }\mu\text{m}$, the ignition pressure is on the order of 300 Gbar. This corresponds to the hot spot energy of a few tens of kJ, that is, about 20–40% of the shell kinetic energy. The shell compression can be obtained by solving Eqs. (18), (23) while neglecting the shell internal pressure and fusion reactions:

$$C_R = \left(1 + M_{sh} U_{imp}^2 / 2\mathcal{E}_{h0}\right)^{1/2} \quad (25)$$

where $\mathcal{E}_{h0} = 2\pi p_{h0} R_1^3$ is the initial energy of the hot spot.

The pressure in the left hand side of expression (24) can be estimated as the initial hot spot pressure, p_{h0} , multiplied by the compression ratio to the fifth power. Thus, Eq. (24) can be represented as a condition on the initial hot spot pressure and compression ratio:

$$p_{h0} > 5U_{imp}/R_1 S_\alpha C_R^5 \quad (26)$$

In addition, we need to impose the condition of validity of Eq. (23), namely, that the hot spot temperature is sufficiently high and that the T_i^2 dependence of the fusion reaction rate applies. This means that α -particles have to deposit a significant fraction of their energy, f_α , inside the hot spot.

The factor f_α depends on the ratio, $\tau_\alpha = l_\alpha/R$, of the α -particle stopping range, l_α , to the hot spot radius, R . In the temperature domain of interest, the stopping range can be expressed as $l_\alpha \approx 0.025 T^{5/4} / \rho_h \text{ cm}$, where the plasma temperature is in keV and density is in g cm^{-3} . The fraction of fusion energy deposited in the hot spot can be interpolated as proposed by [Krokhin and Rozanov \(1973\)](#) and [Gus'kov et al. \(1975\)](#):

$$f_\alpha = \begin{cases} 1.5\tau_\alpha - 0.8\tau_\alpha^2, & \text{for } \tau_\alpha < 0.5, \\ 1 - 0.25\tau_\alpha^{-1} - 0.00625\tau_\alpha^{-3}, & \text{for } \tau_\alpha \geq 0.5. \end{cases} \quad (27)$$

Consequently, the condition $\tau_\alpha \geq 0.5$ is needed for achieving ignition.

Moreover, the initial mass of DT gas is not sufficient to achieve ignition. The hot spot mass is provided by the shell internal ablation under the electron energy flux. The mass ablation rate can be evaluated similarly to the X-ray-driven ablation discussed in section ["Creation of the ablation pressure with thermal X-rays"](#) by equating the energy flux entering the solid shell to the energy

carried with the ablated plasma, $4\rho_{bf}u_{bf}^3$. The only difference from Eqs. (2), (4) is that the energy flux is provided by the electron heat conduction and α -particles. The electron heat flux, $q_e = -\kappa_{SH}dT/dR$, can be estimated as $\approx \kappa_{SH} T_h/R_{sh}$, where

$$\kappa_{SH} = 3.24n_e v_{te}^2 / \nu_{ei} = \kappa_{e0} T_h^{5/2} \quad (28)$$

is the Spitzer heat conductivity, with $v_{te} = (T_e/m_e)^{1/2}$ being the electron thermal velocity and ν_{ei} the electron-ion collision frequency. For a DT plasma and taking a value 5 for the Coulomb logarithm, $\kappa_{e0} \approx 1.2 \times 10^{28} \text{ cm}^{-1}\text{s}^{-1}\text{keV}^{-5/2}$. Estimating then the blowoff velocity as $u_{bf} \approx (T_h/m_i)^{1/2}$, we obtain the following expression for the mass ablation rate: $\dot{m}_e = \rho_{bf}u_{bf} \approx 0.25m_i\kappa_{e0}T_h^{5/2}/R_{sh}$.

Similarly, the energy flux of α -particles can be estimated as $(1-f_\alpha)W_\alpha V_h/S_h$, and the corresponding mass ablation rate reads: $\dot{m}_\alpha \approx 0.25(1-f_\alpha)W_\alpha m_i T_h/R_{sh}$.

At the shell deceleration stage, the electron driven ablation dominates. The hot spot temperature can be related to the pressure and mass: $T_h = 2\pi m_i p_h R_{sh}^3/3M_h$, and, assuming that that hot spot pressure evolves adiabatically, $p_h = p_{h0}(R_1/R_{sh})^5$, the equation for the hot spot mass, $dM_h/dt = 4\pi R_{sh}^2 \dot{m}_e$ can be expressed as follows:

$$\frac{dM_h}{dt} = \pi \kappa_{e0} m_i^{7/2} \frac{\mathcal{E}_{h0}^{5/2} R_1^5}{M_h^{5/2} R_{sh}^4} \quad (29)$$

As the shell trajectory $R_{sh}(t)$ is known from Eq. (18), Eq. (29) can be readily integrated, thus providing the hot spot mass at stagnation, M_{hc} :

$$M_{hc} \approx m_i (\pi \kappa_{e0} R_1^2 / U_{sh})^{2/7} \mathcal{E}_{h0}^{5/7} C_R^{6/7}$$

A typical value of hot spot mass is on the order of 10 μg , that is, about 1% of the total fuel mass. Similarly, one obtains expressions for the hot spot areal density and hot electron temperature at stagnation:

$$\rho_h R_c \approx M_{hc} C_R^2 (3/4\pi R_{sh0}^2), \quad T_h \approx (m_i/M_{hc}) C_R^2 \mathcal{E}_{h0}$$

By using these two expressions along with Eq. (25) for the compression ratio, one can present the two ignition conditions, Eq. (26) and $\tau_a \gtrsim 0.5$, as two criteria on the shell implosion velocity. In practical units they read:

$$\left(\frac{M_{sh}}{1 \text{ mg}}\right)^{1/2} \left(\frac{U_{imp}}{300 \text{ km s}^{-1}}\right)^{4/5} > \left(\frac{R_1}{1 \text{ mm}}\right)^{13/10} \left(\frac{p_{h0}}{1 \text{ Mbar}}\right)^{3/10} \quad (30)$$

$$\left(\frac{M_{sh}}{1 \text{ mg}}\right)^{1/2} \left(\frac{U_{imp}}{300 \text{ km s}^{-1}}\right)^{11/20} > \left(\frac{R_1}{1 \text{ mm}}\right)^{5/4} \left(\frac{p_{h0}}{1 \text{ Mbar}}\right)^{1/4} \quad (31)$$

These analytic estimates provide the values of characteristic parameters that must be attained at the acceleration stage in order to achieve ignition from the hot spot. They are confirmed with more detailed numerical simulations.

Burn-wave propagation in a cold shell

As soon as ignition conditions are achieved in the hot spot, the hot spot pressure and temperature increase and a strong shock is launched into the shell. Behind the shock, the hot spot energy and mass increase because of energy released in fusion reactions and enhanced shell internal ablation driven by an energy flux of α -particles escaping the hot spot. The burn front velocity is typically smaller than the shock velocity, thus the burn time is defined by the time of the shell confinement, that is, the time of shock propagation across the shell, $\Delta t_{sh} = \Delta R_{sh}/D_s$ and the time of rarefaction wave propagation backwards from the shell edge to the burn front, $\Delta t_{rw} = \frac{1}{4}\Delta R_{sh}/c_{sh}$. Here, $D_s \approx \sqrt{p_h/\rho_{sh}}$ is the shock velocity and $c_{sh} \approx \sqrt{p_h/4\rho_{sh}}$ is the acoustic velocity in the shocked shell compressed by approximately a factor of 4 in the shock.

The dynamics of the burning hot spot is described by equations of energy and mass conservation similar to Eqs. (20), (29), with the difference that one has to account for the work performed by the shock and for the ablation due to the α -particle energy flux. The kinetic energy of the hot spot can be neglected because of its subsonic expansion. Consequently, the hot spot dynamic equations read:

$$(3/2)d(p_h V_h)/dt = W_\alpha V_h - 4\pi R_h^2 p_h u_s \quad (32)$$

$$dM_h/dt = (1-f_\alpha)m_i W_\alpha V_h/T_h \quad (33)$$

Here, $u_s = \frac{3}{4}D_s \approx dR_h/dt$ is the fluid velocity behind the shock and we neglected the bremsstrahlung losses and the electron heat flux.

This system of equations has been analyzed by Atzeni and Caruso (1984) and more recently by Christopherson et al. (2018). If fusion reactions are ignited, the temperature in the hot spot rises quickly to a value of $T_{\max} \approx 40\text{--}50 \text{ keV}$, corresponding to the maximum of the DT reaction rate. This value defines the pressure in the burning plasma and the burn time, $t_b \approx \Delta t_{sh} + \Delta t_{rw}$, which is proportional to the shell thickness. Its typical value is about 100 ps.

Knowing the plasma temperature and the burn time, one can also evaluate the fraction of the fuel that can be burnt during the confinement time: $\Phi_b \approx W_\alpha V_{sh}(R_{sh}/D_s)/(\epsilon_\alpha \rho_{sh} V_{sh}/2m_i)$. As the fusion rate is proportional to the square of the fuel density, the burn fraction is proportional to the shell areal density:

$$\Phi_b = \rho_{sh} R_{sh} / H_b$$

where $H_b \approx 8m_i D_s / \langle \sigma_{DT} v \rangle \approx 7 \text{ g cm}^{-2}$. Such a high value of the fuel areal density is very difficult to achieve. The present day designs aim at shell areal densities of about $1.5\text{--}2 \text{ g cm}^{-2}$, which correspond to a fuel compression of more than a factor of 1000. Correspondingly, the expected burn fraction is on the order of 20–30%. The physics of target implosion and burn is demonstrated in the numerical simulation presented in the next section.

Numerical simulations of an ICF target

Numerical simulations of target implosion are based on the hydrodynamic model considering the plasma as a single species, multi-material, two-temperature fluid. The equations of ideal hydrodynamics present the basic framework of the model. They are completed with modules describing all the physics involved in the ICF process. The following physical effects are typically included in the model: (i) laser energy deposition in plasma corona; (ii) realistic equations of state; (iii) electron and ion heat conduction; (iv) radiation transport with tabulated opacities; (v) generation of spontaneous magnetic fields and their effect on transport coefficients; and (vi) fusion reactions and α -particle energy transport. The equations are solved on an unstructured grid with Lagrangian or Eulerian solvers in one-, two- or three-dimensional geometry depending on the problem considered.

Theoretical estimates of the target implosion are illustrated here with numerical simulations conducted with the radiation hydrodynamic code CHIC, see Ribeyre et al. (2008). We consider a directly driven “all DT” target containing a very thin plastic shell of $3 \mu\text{m}$, to ensure the mechanical support, and a DT layer with a thickness of $211 \mu\text{m}$. The total fuel mass is 0.59 mg , external radius is $1047 \mu\text{m}$ and gas density is 0.1 mg cm^{-3} (see Fig. 1B). The external part of this DT shell is used as an ablator and the internal part as a fuel. So the effective fuel mass is about a half of the initial mass, that is, 0.3 mg . This target produces in the numerical simulation a thermonuclear energy of 19.5 MJ , with an energy gain of 150.

The laser pulse shape shown in Fig. 6A is chosen in such a way that it creates the level of pressure needed for acceleration of the shell with a desired value of the in-flight shell adiabat, $\alpha_{if} \approx 1$. The formulas for the ablation pressure and ablation mass rate (3) allow definition of the necessary laser intensities and powers. We consider a laser wavelength of 351 nm (third harmonic of the Nd:glass laser) and a coefficient of absorption, $\eta_{abs} \approx 70\%$. The first shock with an amplitude of 2.4 Mbar , is created by a pre-pulse (a “foot”) with an intensity $4 \times 10^{12} \text{ W cm}^{-2}$, corresponding to a laser power of 0.6 TW . The duration of the pre-pulse, $t_1 \approx 4 \text{ ns}$, is defined by the time needed for the first shock to cross the shell. The second shock, that propagates faster than the first, catches the first one at the inner edge of the shell. This allows maintenance of a low adiabat in the fuel and preheating the gas inside the shell. The total energy of the laser pulse is 134 kJ , and the absorbed energy is 92 kJ .

The rising part of the pulse in the time interval between 4 and 8 ns is dedicated to acceleration of the shell to the implosion speed, $U_{imp} \approx 240 \text{ km s}^{-1}$. The ablation pressure is rising gradually to 40 Mbar , which corresponds to increasing laser intensity to the level of $3 \times 10^{14} \text{ W cm}^{-2}$ and the pulse power to 40 TW . This level is maintained during the implosion time, which is approximately 3.5 ns in the present example. Fig. 6B shows the implosion diagram—the trajectories of the fluid elements. Initially, all the mass of the target is distributed homogeneously in the interval $0.833 < R < 1.044 \text{ mm}$. Upon the laser pulse arrival, the outer cells are ablated and shifted outwards. The shock generated by the pre-pulse ablation pressure is moving inwards with a velocity of $\sim 30 \text{ km s}^{-1}$, and at time $t_2 = 7 \text{ ns}$ it arrives at the rear side of the shell. Starting from this moment, the shell as a whole is accelerated by the rising part of the laser pulse. The acceleration process continues for 3 ns , and, at a time of 10 ns , the shell enters into the

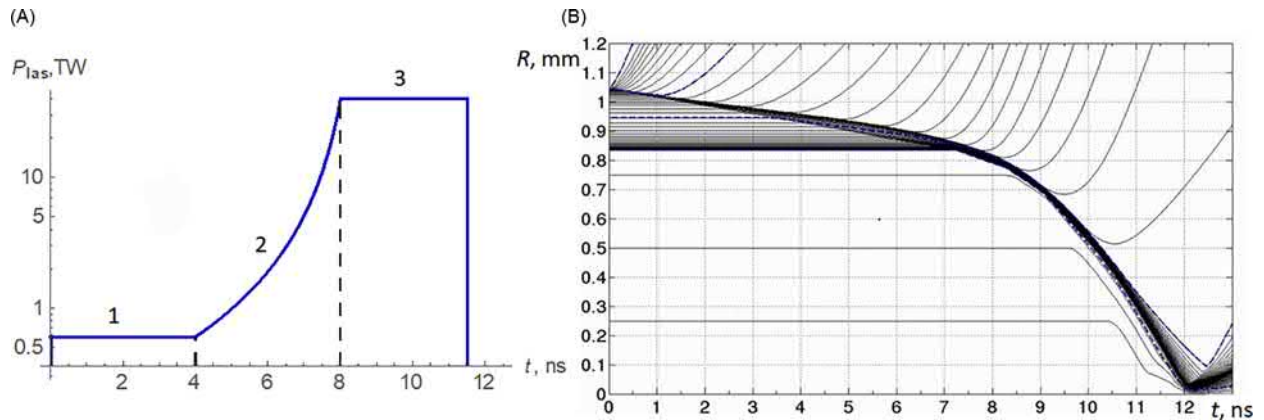


Fig. 6 (A) Temporal profile of laser pulse: dependence of the laser power on time, including the pre-pulse (1, $t < 4 \text{ ns}$), rising part (2) and full power (3, $t > 8 \text{ ns}$). (B) The shell implosion diagram: dependence of the radial position of the shell on time. Dashed blue line shows the first shock trajectory.

implosion phase. Its radius at this moment is ~ 0.5 mm and it is already compressed in the shocks. The acceleration is delayed by 3 ns with respect to the rising part of laser pulse, which is explained by the delay of energy transport between the absorption zone and the ablation layer, where the shock is formed.

The implosion phase takes a time of ~ 2 ns, and it ends with stagnation, which occurs at $t_3 \approx 12$ ns. At the beginning of the phase of implosion, the mass of the shell is 0.3 mg, its velocity is 240 km s^{-1} , and its kinetic energy is 8.6 kJ. Thus, the hydrodynamic efficiency of the acceleration of the shell is less than 9%.

A more detailed understanding of the evolution of the shell can be obtained by considering the trajectories of the first and second shocks in the mass coordinates shown in Fig. 7A. The bright line in the bottom left corner shows the trajectory of the first shock arriving at the inner edge of the shell at 7.3 ns and reflected back as a rarefaction wave (dark line). The line starting in the top left corner shows the ablation front that slowly moves downwards. The rarefaction wave is reflected at the time 8 ns from the ablation front, and this moment is synchronized with the arrival of the second shock. It propagates much faster than the first shock and arrives at the inner edge of the shell at 9 ns. After the launch of the second shock the shell is accelerated almost isentropically.

The details of the stagnation phase are presented in Fig. 7B. The shell density achieves its maximum value of 500 g cm^{-3} at 12.15 ns and then it decreases because of the expansion of the hot spot under the increasing pressure produced by fusion reactions. The hot spot radius at the maximum compression time is $28 \mu\text{m}$, its temperature is 10 keV and its density $\sim 35 \text{ g cm}^{-3}$. The areal density of $\sim 0.15 \text{ g cm}^{-2}$ in that case is sufficient for ignition. The areal density of the fuel is about 1 g cm^{-2} . This is a relatively low value, so the produced fusion energy of 19.5 MJ corresponds to a burn fraction of less than 10% of the initial target mass. The combustion time is extremely short in this example, the fusion energy yield increases in 20 ps from almost zero to its maximum value.

Hydrodynamic stability of the implosion process

This section presents the effects that may degrade the performance of the ICF target. They are related to a deviation of the shell from the ideal case of perfect spherical symmetry assumed in the previous sections. Any deviation from the spherical geometry compromises the conversion of the shell kinetic energy into the internal energy of compressed fuel and heating of the hot spot. The ablation pressure and the distribution of laser intensity over the target surface must be as uniform as possible. Hydrodynamic stability of the imploding shell is a subject of prime importance for ICF, it is discussed in detail in reviews and textbooks by Brueckner and Jorna (1974), Atzeni and Meyer-ter Vehn (2004), Lindl et al. (2004) and Craxton et al. (2015).

A perturbation of the target surface can be developed in a series of spherical functions characterized by the polar number, $l \geq 0$, and azimuthal number, $|m| \leq l$. Consequently, the distribution of the laser intensity over the target surface and shell density perturbations are characterized by the amplitudes of corresponding spherical harmonics.

Deviations from spherical symmetry are divided in two groups depending on their origin. Large scale perturbations with characteristic length larger than the shell initial thickness, corresponding to $l \leq 20$ –30, are due to target misplacement and asymmetry of laser irradiation. The small-scale perturbations with wavelengths of the order of the shell in-flight thickness, or smaller, depend strongly on the quality of the target surface and the smoothness of the laser beams. The initial perturbations are amplified during the implosion process for two reasons. Firstly, the relative amplitude of a perturbation, $\delta R/R$, increases because the shell radius decreases. The initial perturbation is thus multiplied by the convergence ratio at stagnation. Secondly, the amplitude of an initial perturbation increases with time because of hydrodynamic instabilities.

Two instabilities are of prime importance for ICF, the Rayleigh-Taylor and Richtmyer-Meshkov instabilities. The former operates during the shell acceleration and deceleration phases, while the latter is excited when the shock front crosses a modulated interface between two materials of different density. These hydrodynamic instabilities are identified as the major threat for the shell

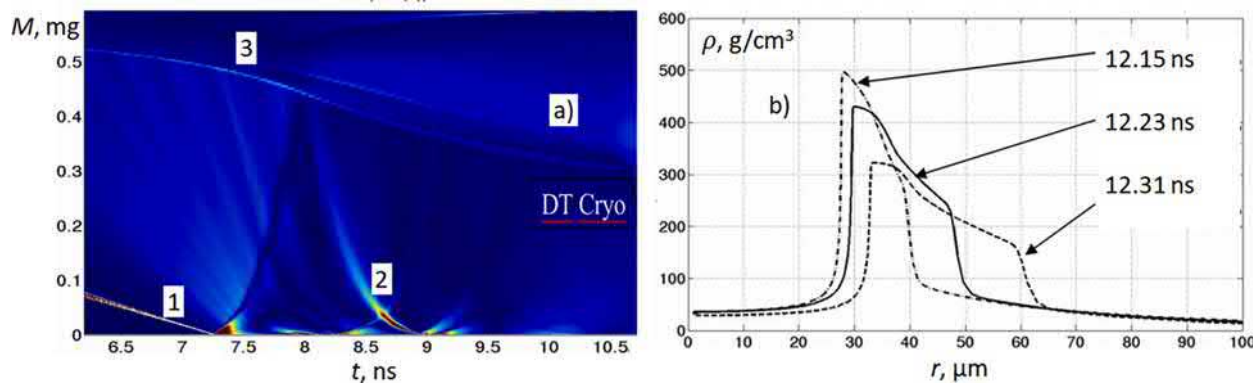


Fig. 7 (A) Trajectories of the first (1) and second (2) shocks in the mass coordinates in the time interval from 6 to 11 ns. The first shock arrives at the inner edge of the shell at $t = 7.3$ ns. The white line (3) at the top shows the position of the shell outer edge. (B) Density profiles in the target during the stagnation showing the hot spot ($r < 30 \mu\text{m}$) and the compressed shell ($r \approx 30$ – $50 \mu\text{m}$).

implosion for two reasons. Firstly, the shell can be destroyed if its radial displacement becomes larger than the in-flight thickness. Having in mind that half of the shell is ablated and that the remaining part is compressed by at least a factor of 2–3, the typical value of the in-flight aspect ratio is $\mathcal{A}_{if} \sim 20$. This corresponds to a shell thickness of less than 5 μm at stagnation and imposes a severe constraint on the acceptable initial perturbation amplitude. Secondly, development of the instability during the deceleration phase leads to mixing of cold shell material with the gas in the hot spot. This effectively decreases the hot spot areal density and temperature, and penalizes ignition. The effect of the hot spot contraction because of shell deformation and fuel mixing is described empirically in one-dimensional simulations with a factor “yield over clean” (YOC), which is defined as a ratio of the measured neutron yield (in multidimensional simulations or in experiments) to that predicted in the corresponding one-dimensional simulation assuming ideal spherical symmetry.

Symmetry of implosion

The low modes are controlled by the focusing conditions and intensity profiles of the laser beams. The requirements on the symmetry of illumination depends on the shell compression. For a given convergence ratio C_R , the perturbation of the initial shell radius, δR_0 , is proportional to the perturbation of the implosion velocity, $\delta R_0/R_c = C_R \delta U_{\text{imp}}/U_{\text{imp}}$. The perturbation of the hot spot that can be tolerated at ignition is a fraction of the hot spot radius. Setting $\delta R_0/R_c \leq 0.2$, the condition on the implosion velocity is $\delta U_{\text{imp}}/U_{\text{imp}} \leq 0.2/C_R$. According to Eq. (16), the implosion velocity is proportional to the mass ablation rate, which depends on the laser intensity as $I_{\text{las}}^{1/3}$. Consequently, the constraint on the uniformity of laser irradiation is $\delta I_{\text{las}}/I_{\text{las}} \leq 0.6/C_R$. For realistic values of $C_R \sim 20\text{--}30$, the uniformity should be better than 1–2%.

Homogeneity of irradiation depends on the number of laser beams and their radial shape. Fig. 8 shows an example of optimization of the target irradiation performed by Hallo et al. (2009) in the framework of the project HiPER (Edwards and Danson, 2015). The facility is planned to have 48 laser beams distributed over a spherical chamber in three cones from each side of the chamber at angles 21.23 degrees (4 beams), 47.03 degrees (8 beams) and 74.96 degrees (12 beams) with respect to the North-South axis. The beam intensity profile

$$I_{\text{las}}(r) = I_0 \exp[-(r/R_{\text{las}})^{2m}] \quad (34)$$

is characterized by two dimensionless parameters, $a = R_{\text{las}}/R_0$ and m , defining the ratio of the beam radius to the target radius and steepness of the intensity profile, respectively. These parameters provide a flexibility for controlling the irradiation symmetry, which is characterized by a standard deviation

$$\sigma_{\text{rms}} = \left[\int (I - \langle I \rangle)^2 d\Omega \right]^{1/2} / \langle I \rangle \quad (35)$$

where the integral is performed over the target surface and $\langle I \rangle$ is the average intensity. The red point in the figure shows the optimal choice with $a \approx 0.6$ and $m \approx 1.0$, which gives homogeneity of irradiation of 0.15% with 94% of laser energy incident on target. It withstands an energy imbalance of 10% and beam pointing imbalance of 5% without compromising the target performance. The amplitude of the most dangerous mode, $l = 12$, defined by the geometry of beam distribution over the chamber, is 0.5% in this case.

Laser imprint

The perturbations at small scale, the high modes of the spherical functions, are dangerous because they can be unstable and grow significantly during the implosion. There are two sources of these perturbations: roughness of the target—small variations of the shell thickness, typically of a few nanometers, and the laser imprint—small perturbations of the laser beam intensity that produce

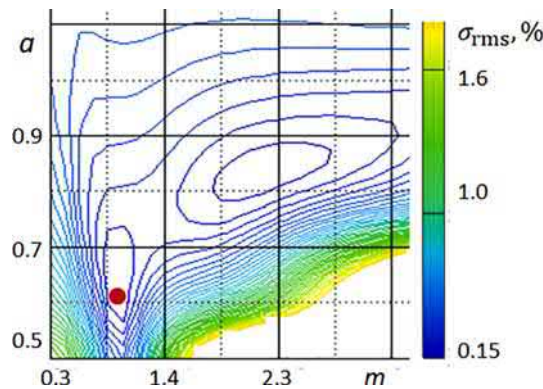


Fig. 8 Homogeneity of target irradiation σ_{rms} (35) in percent as a function of the laser beam radius $a = R_{\text{las}}/R_0$ and profile steepness m by using 48 laser beams with a super-Gaussian intensity profile calculated within the HiPER project (Edwards and Danson, 2015). Red point shows the optimized configuration with $\sigma_{\text{rms}} = 0.15\%$ and 94% of laser energy incident on target.

perturbations of the ablation pressure during the launch of the first shock. The latter is considered as a major issue for the direct drive approach.

A typical nanosecond laser pulse carries energy of a few kilojoules and contains several resonator modes. While focused on target, the mode interference produces a complicated intensity modulation with an amplitude incompatible with requirements on the homogeneity of ablation pressure. Several optical techniques have been developed for improving the laser intensity distribution, commonly called “laser beam smoothing”. They include spatial smoothing using phase plates, polarization smoothing by overlapping two spatially incoherent beams with perpendicular polarizations and temporal smoothing with laser beam spectral broadening.

Spatial smoothing is the method that involves splitting the laser beam entering into the interaction chamber into many (from a few hundred to a few thousand) spatially incoherent beamlets with a random phase plate (RPP) as proposed by [Kato and Mima \(1982\)](#). This allows conversion of an uncontrolled beam interference pattern into an ensemble of small-scale speckles with a transverse scale of about 10 laser wavelengths and homogeneously distributed over the laser focal spot. The speckle statistics follows an exponential distribution and is rather reproducible. A combination of two laser beams with orthogonal polarizations allows a doubling of the number of speckles while maintaining the same average intensity, thus reducing the amplitude of intensity fluctuations on the target.

While spatial laser beam smoothing significantly improved the quality of laser plasma interaction since it was introduced in the 1990s, see [Lindl et al. \(2004\)](#) and [Craxton et al. \(2015\)](#), it is not sufficient for reducing the laser imprint. Temporal smoothing aims at reducing the speckle lifetime to a few tens of picoseconds, thus reducing significantly the time averaged amplitude of intensity modulations. This is achieved by the method of smoothing by spectral dispersion (SSD)—a phase modulation of the laser beam across its transverse profile, as proposed by [Skupsky et al. \(1989\)](#). For a typical frequency of the electro-optical modulator of 17 GHz and depth of modulation of 5, the effective speckle lifetime can be decreased to about 12 ps. Larger spectral widths are incompatible with the existing technique of frequency conversion of the Nd:glass laser, operating on a wavelength of $\sim 1 \mu\text{m}$, into the third harmonic used in the ICF research.

Modulations of laser intensity in the absorption zone can be related to the modulations of the ablation pressure by using Eq. (3), but one should also account for the smoothing effect due to the heat transport in the conduction zone. As suggested by [Bodner \(1981\)](#), this can be described by a “cloudy day” model, namely, due to the lateral electron heat transport the amplitude of intensity modulations in the conduction zone decreases by a factor $\exp(-kd_c)$, where k is perturbation wave number and d_c is the thickness of the conduction zone. Thus, one can evaluate the amplitude of ablation pressure modulations as

$$\delta p_{\text{abl}}/p_{\text{abl}} \approx (2/3) \exp(-kd_c) \delta I_{\text{las}}/I_{\text{las}} \quad (36)$$

As the thickness of the conduction zone increases with time because of plasma expansion, the most dangerous laser imprint is produced at the beginning of the laser pulse, during the first nanosecond, when the ablation surface is very close to the critical surface. Unfortunately, temporal smoothing does not sufficiently improve the situation, because the intensity modulation pattern is established during the first modulation period, which is, in practice, approximately a few tenths of a nanosecond, comparable with the plasma expansion time.

Several methods have been developed for imprint mitigation, all of them attempting to increase the thickness of the conduction zone before the laser pulse arrival. They include: the use of an X-ray flash or low intensity laser pulse to ionize the external layer of the shell, covering the shell with a low density foam layer which can be ionized by X-rays or a laser pre-pulse, or by adding mid-Z elements as dopants in the ablator in order to increase the contribution of radiation transport. These techniques, which are of prime importance for direct drive ICF, are described by [Craxton et al. \(2015\)](#).

Rayleigh-Taylor instability

The hydrodynamic Rayleigh-Taylor (RT) instability develops at the interface of two fluids of different densities under a gravitational force or an acceleration. In the ICF context, this configuration occurs either during the shell acceleration by the ablation pressure acting on the outer shell interface, or during the shell deceleration on the inner shell interface under the pressure of the compressed hot spot. A particular case of the RT instability is the Richtmyer-Meshkov (RM) instability, occurring at a perturbed interface crossed by a planar shock. In all cases the perturbation amplitude is proportional to the initial perturbation and depends on the perturbation wavelength, $2\pi/k$, in the interface plane. The particularity of these instabilities in the ICF context is the mass flow across the interface due to ablation, which plays an important stabilizing role.

The basic mechanism of this instability can be understood by considering an interface between two liquids, a heavy one with a mass density, ρ_1 , and a light one with a mass density, ρ_2 , in a gravitational field with acceleration g directed along the vertical axis z . This configuration is stable if the interface is plane and perpendicular to the z axis, as the gravitational forces $\rho_{1,2}gz$ are perpendicular to the interface and the liquids can be considered as incompressible.

However, the stability is lost if the interface $z = 0$ is perturbed, $\xi(x) = \xi_0(t) \sin kx$. Then, there is an uncompensated tangential component of the force producing acceleration of fluids parallel to the interface according to the Euler equation:

$$\rho_{1,2} dv_{x1,2}/dt = \rho_{1,2} kg \xi \quad (37)$$

where the factor $k\xi$ accounts for the component of the gravitational force acting along the surface. As the fluids are incompressible, the tangential components of the velocities have opposite signs, $v_{x1} = -v_{x2} = v_x$, while the normal components are equal,

$v_{z1} = v_{z2} = v_z = d\xi/dt$. Moreover, incompressibility implies that $v_x \approx v_z$. Consequently, subtracting Eqs. (37) at both sides of the interface, we have:

$$(\rho_1 + \rho_2)d^2\xi/dt^2 = (\rho_1 - \rho_2)kg\xi \quad (38)$$

This equation describes gravitational waves with frequency, $\omega = \sqrt{|A_t|kg}$, if the Attwood number, $A_t = (\rho_1 - \rho_2)/(\rho_1 + \rho_2)$, is negative, that is, the light fluid is above the heavy one. In contrast, it corresponds to an instability growing exponentially, $\xi_0(t) \propto \exp(\gamma t)$, with the growth rate, $\gamma = \sqrt{A_t kg}$, if the heavy fluid is above the light one, $\rho_1 > \rho_2$.

The Richtmyer-Meshkov instability is a special case where a modulated interface is accelerated for a short time at the shock wave front. Then Eq. (38) can be written as $dv_x/dt = A_t kg\xi$, and, integrating this equation, we obtain that there is a modulation of the flow velocity behind the shock: $\delta v_z \approx A_t k D_s \xi_0$, where D_s is the shock front velocity. Consequently, the interface perturbation grows linearly with time:

$$\xi_0(t) = \xi_0(0)(1 + A_t k D_s t) \quad (39)$$

This expression applies, for example, to the interface between ablator and fuel, where perturbations can grow with time downstream of the first shock and will be further amplified exponentially by the RT instability during the acceleration phase. Both instabilities are strongly modified at the ablation surface.

Hydrodynamic instabilities at the ablation front

The presence of the conduction zone in close contact with the ablation surface strongly modifies the hydrodynamic instabilities. Firstly, the density of the blowoff plasma is much smaller than the solid density, so the Attwood number is close to 1 and thus the second equation in (37) can be neglected. Secondly, the conduction layer has a finite thickness, d_c , typically of a few microns, and perturbations with shorter wavelengths, $kd_c > 1$, are suppressed due to the efficient lateral electron heat conduction, similarly to the “cloudy day” model (36).

Moreover, for the long wavelength modes, $kd_c \ll 1$, there are two stabilizing factors related to the mass flow across the ablation front and the electron heat flux. Firstly, the fluid element remains in the unstable zone only a finite time, which can be estimated as the ratio of the thickness of the perturbed layer, $1/k$, to the flow velocity, D_a . Consequently, the RT growth rate is reduced by a factor kD_a , where $D_a = \dot{m}_a/\rho_1$ is the ablation velocity (3).

Secondly, the ablated plasma is a compressible fluid, and modulations of the ablation front result in a variation of the thickness of the conduction zone. Namely, a protrusion of the ablation front into a hot blowoff plasma makes the conduction zone thinner and, correspondingly, results in a higher temperature gradient, a higher heat flux and a higher ablation rate. This is called the “fire polishing” effect. This effect results also in an increase of ablation pressure, which acts in the direction opposite to the acceleration and, therefore, reduces the instability growth rate. The force related to this overpressure effect can be estimated as a projection of the pressure gradient, $k p_{abl}$, on the direction normal to the perturbed ablation front, thus resulting in the restoring force $\sim k^2 p_{abl} \xi$.

These two stabilizing effects bring two additional terms in Eq. (38) describing the temporal evolution of the perturbation of the ablation surface. It takes the following form:

$$\rho_1 d^2\xi/dt^2 + k\rho_1 D_a d\xi/dt = \rho_1 kg\xi - k^2 p_{abl} \xi \quad (40)$$

The second term on the left hand side accounts for the mass flow. It is similar to the corresponding term in the rocket model, section “Shell acceleration by ablation pressure”. The second term on the right hand side corresponds to the overpressure effect. While the complete theoretical expression for the RT growth rate is quite complicated, it turns out that it can be interpolated by a simple empirical formula proposed by Takabe et al. (1983):

$$\gamma \approx \alpha \sqrt{kg/(1 + kd_c)} - \beta k D_a \quad (41)$$

where $\alpha \approx 0.9$ and $\beta \approx 3.1$ are two adjustable parameters, dependent on the shell material. The term under the square root describes the effect of suppression of short wavelength modes due to the finite thickness of the conduction layer, and the second term in the right hand side is related to the mass flow. A remarkable feature of this formula is the stabilizing role of the ablation for long wavelength perturbations and the prediction of a cutoff wave number. Typically, in Omega and NIF implosion experiments, the RT instability cutoff corresponds to the spherical mode number $l \approx 100$ –200.

The stabilizing effects related to ablation are very important for the ICF target design. They suggest a spatial shaping of the ablator density and entropy in such a way that the external part of the shell has a lower density and high temperature, while the internal part of the shell, including fuel, remains cold and does not compromise the implosion. This shaping of the ablator entropy is achieved with a short laser prepulse which heats and compresses the external part of the shell, thus stabilizing the ablation front. This technique of adiabat shaping has been developed and implemented at the Omega laser facility in the context of direct drive implosions, see Craxton et al. (2015).

The exponential growth of the ablation front perturbation corresponds to the linear stage of the RT instability development; it is limited to a perturbation amplitude smaller than 10% of the wavelength. For longer times and larger amplitudes, the instability enters into a nonlinear phase with a slower growth of the perturbation amplitude followed by mixing of the cold material with the hot blowoff plasma. This is a highly undesirable effect for ICF and has to be avoided.

The proximity of the conduction zone also has a notable effect on the Richtmyer-Meshkov instability. After the shock crosses the ablation surface, the latter is accelerated to the velocity, $\delta v_z \approx k D_s \xi_0$, but its further evolution is affected by the overpressure effect. The restoring force in the right hand side of Eq. (40) stops the linear growth of the initial perturbation and results in oscillations of the ablation surface with a frequency $\omega_{RM} \approx k \sqrt{D_a v_{bf}}$. The amplitude of oscillations is related to the initial shock perturbation as $\delta v_z / \omega_{RM} \approx \xi_0 D_s / \sqrt{D_a v_{bf}}$. These oscillations evolve during the time the first shock crosses the shell. They produce a seed for development of the RT instability when the shell is accelerated as a whole. Furthermore, the oscillations of the ablation surface penetrate inside the shell and produce feedthrough perturbations at the inner shell surface where the RT instability develops at the deceleration stage. The most dangerous perturbations are those having a wavelength on the order of the shell thickness, corresponding to the spherical mode number $l \sim \mathcal{A}_{if}$. They grow more rapidly than lower modes and efficiently penetrate through the shell. This fact explains the fragility of very thin shells and is one of the major limiting factors of target compression.

Laser plasma interactions in the ICF context

Laser radiation is the best suited for driving inertial fusion targets. It is a unique source capable of providing extreme powers, on the level of a few hundred terawatts, needed to bring the DT fuel to ignition conditions. However, the physics of laser plasma interactions is complicated and is a source of several serious issues. Firstly, the electromagnetic temporal and spatial scales of a few femtoseconds and a few microns are incompatible with the ICF scales of hundreds of picoseconds and hundreds of microns. So, the physics of laser plasma interactions is considered separately and introduced in the global ICF model in a simplified fashion. Secondly, laser radiation in the optical domain cannot reach the dense plasma, it is absorbed or reflected, at the critical density, which is 10–100 times smaller than the solid density. Therefore, one should ensure an efficient laser energy absorption in such a low density plasma and efficient energy transport from the absorption zone to the solid densities. Thirdly, the laser intensities needed for target implosion are rather high, at the level of 10^{14} – 10^{15} W cm⁻², so that the electron quiver velocity in the laser field is comparable to (but smaller than) the electron thermal velocity, so the laser plasma interaction becomes nonlinear, resulting in excitation of parametric instabilities. This leads to modification of the laser propagation, enhanced scattering and the generation of suprathermal (hot) electrons, which are undesirable for ICF.

These three major issues for ICF are discussed in this section. A more detailed description of laser plasma interactions in the ICF context can be found in reviews and textbooks by [Kruer \(2003\)](#); [Atzeni and Meyer-ter Vehn \(2004\)](#); [Lindl et al. \(2004\)](#) and [Craxton et al. \(2015\)](#).

Laser propagation in an inhomogeneous plasma

Laser radiation is quasi-monochromatic and the laser period is much shorter than characteristic hydrodynamic scales. Thus it can be described by Maxwell's equations in the envelope approximation with the electric field written as $\text{Re} E \exp(-i\omega_{las}t)$, with the complex amplitude, E , slowly varying over the laser period, $2\pi/\omega_{las}$, and wavelength, λ_{las} . The magnetic field is then defined by the Faraday's and Ampere's equations:

$$\mathbf{B} = -(i/\omega_{las}) \text{rot} \mathbf{E}, \quad \text{rot} \mathbf{B} = -i(\omega_{las}/c^2) \varepsilon(\omega_{las}) \mathbf{E} \quad (42)$$

Here $\varepsilon(\omega_{las})$ is the plasma dielectric permittivity, which has a simple form, $\varepsilon = 1 - \omega_{pe}^2/\omega_{las}^2$, where $\omega_{pe} = (e^2 n_e / m_e \varepsilon_0)^{1/2}$ is the electron plasma frequency, ε_0 is the vacuum permittivity and the ion contribution is neglected because of a large ion-to-electron mass ratio.

In the geometrical optics approximation, $\mathbf{E} \propto \exp(i \int \mathbf{k}_{las} \cdot d\mathbf{r})$, from these two equations one obtains a dispersion equation,

$$k_{las}^2 c^2 / \omega_{las}^2 - \varepsilon = 0 \quad (43)$$

As the first term in the left hand side is positive, the laser may propagate only in a transparent plasma where $\varepsilon > 0$. As the laser propagates in a plasma with increasing density, the dielectric permittivity and the wave number decrease and they are equal to zero at the critical density, $n_{cr} = m_e \varepsilon_0 \omega_{las}^2 / e^2$. Thus, the laser cannot propagate deeper into plasma and is reflected. The plasma critical density can be represented as $n_{cr} = 1.1 \times 10^{21} \lambda_{las}^{-2}$ cm⁻³, where the laser wavelength λ_{las} is in microns. In fact, this density is accessible only at normal laser incidence, otherwise the laser radiation is reflected at lower densities.

The expression on the left hand side of Eq. (43) can be considered as the Hamiltonian of a classical particle of unit mass and with a dimensionless momentum, $\mathbf{k}_{las} c / \omega_{las}$, moving in a potential of $\frac{1}{2} \varepsilon(\mathbf{r})$. Consequently, the trajectory of an optical ray can be found from the principle of minimum action

$$d\mathbf{r}/ds = \mathbf{k}_{las} c / \omega_{las}, \quad d\mathbf{k}_{las}/ds = (\omega_{las}/2c) \nabla \varepsilon \quad (44)$$

where s is an effective time, which, in this case, is a coordinate along the ray. According to these equations, the trajectory of a laser ray incident obliquely on a plasma with a linear density profile, $\varepsilon = 1 - z/L_n$, is a parabola, similar to the particle trajectory in a constant gravitational field. For the angle of incidence θ , the perpendicular component of the wave vector is conserved, $k_{las \perp} = (\omega_{las}/c) \sin \theta$, while the parallel component decreases to zero at the turning point, $z_t = L_n \cos^2 \theta$.

The equations of geometrical optics (44) constitute the basis for modeling of laser propagation in ICF hydrodynamics. Namely, the laser beams incident on a plasma are split into many rays, each of them carrying a fraction of the total beam power and propagating along its own trajectory defined by the electron density profile provided by solution of the hydrodynamic equations. As the light velocity is much larger than the plasma fluid velocity, the ray propagation is considered to be instantaneous. The laser energy deposition in the plasma is calculated at each time step as a sum of the energy deposition of each ray along its trajectory according to the imaginary part of the dielectric permittivity:

$$dP/ds = -\kappa_{IB}P \quad (45)$$

where P is the ray power and $\kappa_{IB} = (\omega_{las}/c) \text{Im}\epsilon (\text{Re}\epsilon)^{-1/2}$ is the absorption coefficient. In the ICF-relevant conditions, it is defined by electron-ion collisions, $\kappa_{IB} = (\nu_{ei}/c)(n_e/n_{cr})(1 - n_e/n_{cr})^{-1/2}$, where

$$\nu_{ei} = \frac{Ze^4 n_e \ln\Lambda}{3\epsilon_0^2 m_e (2\pi T_e)^{3/2}}$$

is the electron-ion collision frequency and $\ln\Lambda$ is the Coulomb logarithm. This is the inverse bremsstrahlung process.

In an inhomogeneous plasma, the collisional absorption rate, κ_{IB} , increases as the square of the plasma density, so the maximum of absorbed energy is localized near the critical density. In particular, in the case of a linear density profile, $n_e(z) = n_{cr}z/L_n$, Eq. (45) can be integrated explicitly providing the following expression for the absorption coefficient

$$\eta_{abs} = 1 - \exp\left(-\frac{32}{15}\tau_{IB}\cos^5\theta\right) \quad (46)$$

where $\tau_{IB} = \nu_{ei,cr}L_n/c$ is the plasma optical thickness, and $\nu_{ei,cr}$ is the collision frequency at the critical density. The value of τ_{IB} defines the efficiency of collisional absorption. Its dependence on the ion charge and density favors short laser wavelengths and high ion charge. At the wavelength of $0.35\text{ }\mu\text{m}$ and at normal incidence, the absorption coefficient increases from 60% for a DT plasma to 70% for a CH plasma. An efficient laser collisional absorption is one of motivations for converting Nd laser radiation generated at $1\text{ }\mu\text{m}$ wavelength to the second and third harmonics.

Laser collisional absorption depends strongly on the angle of incidence. According to Eq. (46), absorption is reduced by a factor of 2 for a ray incidence angle of $\theta = 30$ degrees with respect to the density gradient. Oblique incidence is inevitable in ICF-relevant conditions, where the laser beam width is comparable to the target size. Moreover, the laser beams arriving from different directions on the target cross over in plasma, thus creating a complicated intensity pattern. Fig. 9 shows an example of two laser beams incident on a spherical target.

The geometrical optics model provides, however, a very basic description of the laser plasma interaction, limiting it simply to collisional (inverse bremsstrahlung) absorption. Each ray propagates independently and interference effects are not accounted for. Moreover, each ray carries a power, but the intensity of laser radiation within the plasma is not known. Therefore, nonlinear and polarization effects are out of scope of this model. This is not adequate for ICF conditions, where effects such as resonance absorption, cross beam energy transfer, excitation of parametric instabilities have to be considered. These effects are discussed in the next sections.

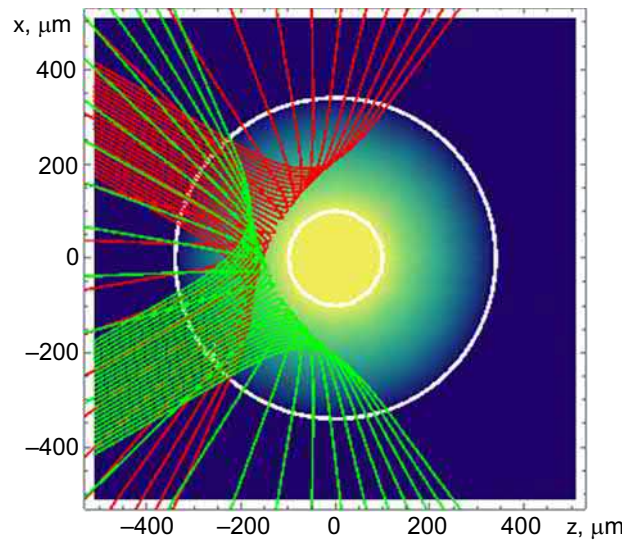


Fig. 9 Example of ray trajectories of two laser beams incident at angles ± 30 degrees on a spherical target with electron density increasing linearly from 0 to $1.2n_{cr}$ between the two white circles.

Resonance absorption

In the case of oblique incidence on a plasma of the laser wave polarized in the plane of incidence there is another mechanism of absorption related to a resonance excitation of plasma waves with frequency equal to the laser frequency. Because of the resonance condition, $\omega_{\text{las}} = \omega_{pe}$, this process takes place near the critical density. However, the absorbed energy does not go to plasma heating, but to acceleration of a relatively small group of resonant electrons to high energies.

Let us consider a plane laser wave with amplitude, E_0 , incident at angle, θ , in the y, z plane on a plasma with a linear density profile, $n_e = n_{cz}/L_n$. The resonant absorption occurs for the wave having the electric field in the plane of incidence, y, z . This wave is reflected at the point, $z_r = L_n \cos^2 \theta$, but the evanescent field component, E_z , parallel to the gradient penetrates to the critical point, $z_r = L_n$, and excites there an electron plasma wave. Solution of the Maxwell's equations for a linear density profile in the case $\omega_{\text{las}} L_n / c \gg 1$ provides the following expression for the driver field, see [Kruer \(2003\)](#):

$$E_z(z_{cr}) = E_{\text{las}} (2\pi\omega_{\text{las}} L_n / c)^{-1/2} \phi(q) \quad (47)$$

where $q = (\omega_{\text{las}} L_n / c)^{1/3} \sin \theta$ and ϕ is the resonance function, $\phi(q) \approx q \exp(-2q/3)$. It is zero for the case of normal incidence and increases with θ , achieving a maximum for an optimal angle $\theta_{\text{res}} \approx 0.7 (\omega_{\text{las}} L_n / c)^{-1/3}$, then decreases exponentially.

The function ϕ characterizes the laser absorption, that is, the efficiency of conversion of the incident laser energy flux into the energy of plasma waves: $\eta_{\text{res}} \approx \frac{1}{2} \phi^2(q)$. The maximum value of the resonance absorption coefficient is approximately 50%, but it is limited to relatively small angles of incidence. The energy transferred to the plasma waves is eventually absorbed in plasma. Depending on the plasma temperature two absorption mechanisms are possible. In a low temperature plasma, if the characteristic collision frequency is sufficiently large, $\nu_{ei,cr} / \omega_{\text{las}} \gg (\lambda_D / L_n)^{2/3}$, where λ_D is the electron Debye length, the width of the resonance is determined by collisions, $\Delta z_{\text{res}} \approx L_n \nu_{ei,cr} / \omega_{\text{las}}$, and the electron distribution remains approximately Maxwellian. In the opposite case, corresponding to ICF conditions, Landau damping is dominant and the absorbed energy is transferred to the resonant electrons. The characteristic width of the plasma resonance in that case, $\Delta z_{\text{res}} \approx (\lambda_D^2 L_n)^{1/3}$, corresponds to the phase velocity of the plasma wave, $v_{ph} \approx \omega_{pe} \Delta z_{\text{res}}$. Thus, the characteristic energy of accelerated electrons is a factor of $(L_n / \lambda_D)^{2/3}$ higher than the temperature of cold electrons, T_c , at the critical density. The energy flux of these electrons is directed into the lower density plasma, opposite to the density gradient, but the accelerated electrons are returned by the self-consistent electric field and penetrate into the overdense plasma behind the ablation front. They may preheat fuel and compromise compression.

This linear model of resonance absorption applies to relatively low laser intensities, when the ponderomotive pressure of resonantly excited plasma waves is smaller than the plasma thermal pressure. At intensities exceeding $10^{14} \text{ W cm}^{-2}$, the resonantly excited plasma wave produces a steepening of the density profile, so that the characteristic density scale length decreases and becomes comparable to the laser vacuum wavelength. Analytic estimates and numerical simulations conducted by [Estabrook et al. \(1975\)](#) and [Forslund et al. \(1977\)](#) have led to the following conclusions concerning the nonlinear resonance absorption: (i) plasma density profile steepening leads to increase of the absorption efficiency and optimal angle of incidence; (ii) the energy of hot electrons produced due to the resonance absorption significantly decreases; (iii) parametric instabilities that may be excited near the critical density are suppressed. The energy distribution of accelerated electrons has approximately an exponential form and can be characterized by a hot electron temperature, see [Forslund et al. \(1977\)](#):

$$T_h \approx 30 (I_{\text{las}} \lambda_{\text{las}}^2)^{1/3} T_c^{1/3} \text{ keV} \quad (48)$$

where T_c is the temperature of bulk electrons in keV, the laser intensity at the turning point is in units of $10^{15} \text{ W cm}^{-2}$ and the laser wavelength in microns. A combination of $I_{\text{las}} \lambda_{\text{las}}^2$ is proportional to the quiver energy of electron in the laser field; this is the main parameter characterizing nonlinear laser plasma interaction processes.

Resonance absorption was considered as an important process in the early years of ICF, when long wavelength lasers at 10 and 1 μm were used in experiments. Transition to short wavelengths, 0.5 and 0.35 μm , has significantly diminished the role of the resonance absorption for two reasons: firstly, the collisional absorption is more efficient, significantly decreasing the laser intensity near the critical density and, consequently, the fraction of laser energy transferred to resonant electrons to a few percent; secondly, the laser irradiance, $I_{\text{las}} \lambda_{\text{las}}^2$, decreases at shorter wavelengths, so the hot electron temperature (48) typically does not exceed 20–30 keV and does not present a notable risk of fuel preheat.

Parametric instabilities

When the laser irradiance, $I_{\text{las}} \lambda_{\text{las}}^2$, exceeds a few times $10^{14} \text{ W } \mu\text{m}^2 \text{ cm}^{-2}$, the temperature of the plasma corona increases above 1 keV, collisional absorption rapidly decreases and the laser plasma interaction is dominated by the nonlinear effects. The most important under ICF-relevant conditions are the following nonlinear processes: laser beam self-focusing and filamentation, and parametric instabilities: stimulated Raman and Brillouin scattering (SRS and SBS) and two plasmon decay (TPD). All these processes take place in the laser propagation zone, to the right of the critical surface in [Fig. 2](#). The domain of operation of parametric processes is shown in more detail in [Fig. 10](#).

Filamentation and SBS (stimulated scattering on low frequency ion acoustic waves) may develop at all densities in the range $n_e < n_{cr}$; they are responsible for laser intensity modulation and reflection of laser light before it accesses the zone of efficient absorption. TPD develops in a narrow zone of plasma with a density close to one quarter of the critical density. It corresponds to a resonant

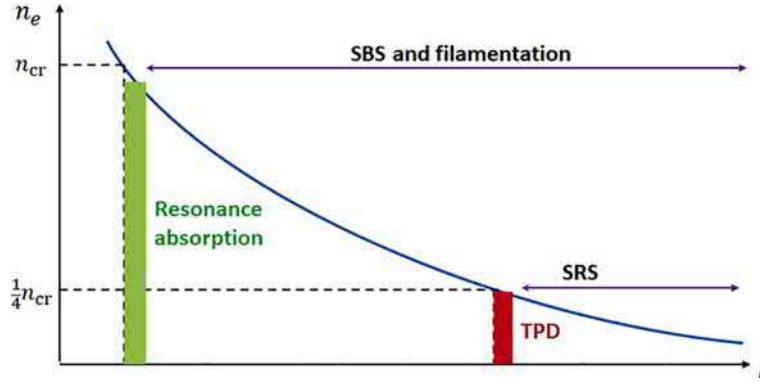


Fig. 10 Domains of operation of different parametric processes in terms of the plasma density profile, $n_e(r)$. Laser radiation comes from the right. Green and red marks show the positions of the resonance absorption near the critical density, n_{cr} , and two plasmon decay near one quarter of the critical density, $\frac{1}{4}n_{cr}$.

excitation of plasma waves with frequency equal approximately to one half of the laser frequency. These waves typically have a phase velocity much larger than the electron thermal velocity and thus may accelerate electrons to high energies. TPD excitation is manifested by the appearance in the spectrum of reflected light fractional harmonics with the frequencies $\frac{1}{2}\omega_{las}$, $\frac{3}{2}\omega_{las}$ and $\frac{5}{2}\omega_{las}$, which are produced due to scattering of the incident and reflected laser waves on the TPD-driven plasma waves. SRS, stimulated scattering on electron plasma waves, develops in plasmas with densities smaller than one quarter of the critical density. It transfers a part of laser energy to plasma waves, and consequently to energetic electrons, and another part to scattered electromagnetic waves. The ratio between these two components, according to the Manley-Rowe relation, is proportional to the ratio of the frequencies of the corresponding waves, $\omega_{pe}/(\omega_{las} - \omega_{pe})$. So, up to a half of the incident laser energy can be transferred to plasma waves.

Often these parametric processes are considered as undesirable because they reflect a significant portion of the laser light (SBS and SRS), modulate the laser beam intensity thus creating high intensity spots (filamentation), redistribute the laser energy over the absorption zone due to the cross-talk between the overlapping beams arriving from different directions (cross beam energy transfer, CBET), and generate large amounts of energetic electrons due to excitation of large amplitude plasma waves (TPD and SRS).

While parametric processes have been studied experimentally and theoretically for more than 50 years, understanding of their role in the ICF context is still far from being complete. This is explained first of all by the fact that plasma instabilities develop on a microscopic scale, which cannot be resolved in hydrodynamic simulations and in integrated experiments. There is a disparity, therefore, between the detailed local microscopic models and kinetic simulations and their implementation in the ICF hydrodynamic codes. The major obstacle is related to the geometrical optics model describing laser energy deposition in the plasma, which does not account for the interference pattern of laser beams and does not provide access to the local values of laser intensity in the plasma, which is indispensable for characterization of nonlinear effects. Another issue is related to suprathermal (hot) electrons, which have a large mean free path and their energy deposition in the plasma has to be described with special modules for nonlocal energy transport.

The role of parametric instabilities in scattering of laser energy and generation of hot electrons has been demonstrated in early ICF studies in the 1970s and 1980s by McCall (1983) and Lindl et al. (2004), and resulted in the complete reorientation of the ICF programs. Namely, the CO₂ gas lasers operating at a wavelength of 10.6 μm were abandoned, and Nd:glass lasers operating at a wavelength of 1.06 μm were converted into the third harmonic. Moreover, laser beam smoothing techniques have been introduced (see section “Laser imprint”), providing a much better control on the overall intensity distribution on the target. While these measures have certainly improved the quality of laser energy deposition and suppressed undesirable nonlinear processes, they did not completely eliminate them. Large scale integrated experiments conducted during the last decade on the Omega and NIF facilities have demonstrated significant energy losses due to SBS, SRS and CBET, and generation of large populations of hot electrons due to SRS and TPD, see Kirkwood et al. (2013) and Craxton et al. (2015). In what follows, we describe the role of parametric instabilities in the ICF context and methods for their modeling with hydrodynamic codes.

Laser beam filamentation

Self-focusing is an important process that can modify the propagation of a high power laser beam in a low density part of the plasma located in front of the absorption region. The technique of laser beam smoothing employed at high energy laser facilities includes random phase plates and smoothing by spectral dispersion. It allows suppression of the global laser beam self-focusing by splitting the whole beam into multiple mutually incoherent beamlets, each of them carrying a much smaller power. In the focusing plane these beamlets create an interference pattern that is called *speckles*. Each speckle can be considered locally as a Gaussian beam with a width, $w_s \sim \lambda_{las} F/D$, defined by the whole laser beam aperture, D , the focal distance, F , and length, $l_s = \pi w_s^2 / \lambda_{las}$, with the maximum intensity, I_s , following an exponential probability distribution, $p_s(I_s) \propto \exp(-I_s/\langle I \rangle)$, where $\langle I \rangle$ is the average laser intensity on target.

Self-focusing of each laser speckle takes place if the speckle power, $P_s = I_s \pi w_s^2$, exceeds the critical power for the ponderomotive self-focusing, $P_c = 1.86 v_g n_{cr} T_e / r_{cl} n_e$, where v_g is the group light velocity and r_{cl} is the classical electron radius. It can be written in a convenient form as

$$P_c = 32 T_e (n_{cr} / n_e) \sqrt{1 - n_e / n_{cr}} \text{ MW} \quad (49)$$

where the electron temperature is in keV. For the typical ICF conditions with a laser wavelength $0.35 \mu\text{m}$ and focusing optics $F/D \approx 8$, the speckle radius is $w_s \approx 2.5 \mu\text{m}$ and the average speckle power, $\langle P \rangle \approx 0.2 \text{ GW}$, is of the order of the critical power for the laser intensity, $I_{las} \sim 10^{15} \text{ W cm}^{-2}$. Thus, speckles with intensity above average may undergo self-focusing. This leads to a local increase of laser intensity in a speckle, a modification of the speckle probability distribution with an increase of the number of high intensity speckles, which become the site of secondary parametric instabilities.

The time of self-focusing development can be estimated as the time for ion acoustic wave propagation across the speckle, $t_s = w_s / c_s$. For the typical ion acoustic velocity, of the order of $0.3\text{--}0.4 \mu\text{m ps}^{-1}$, this time is about $5\text{--}10 \text{ ps}$. It is comparable to the laser pulse correlation time introduced by smoothing using spectral dispersion (SSD, see section “Laser imprint”). Thus, this temporal smoothing technique introduced by Skupsky et al. (1989) is an efficient tool for control and suppression of speckle self-focusing under ICF-relevant conditions.

Stimulated Brillouin scattering

Stimulated Brillouin scattering (SBS) corresponds to scattering of the incident laser light on an ion acoustic wave in the plasma. The scattered light may propagate in different directions, but SBS in the backward direction usually dominates, thus reducing the amount of laser light accessing the zone of absorption. SBS is a convective instability, that is, the scattered wave of a given frequency, ω_s , is amplified as it propagates in a spatially inhomogeneous plasma through the resonance region where the beat frequency, $\omega_{las} - \omega_s$, equals the ion acoustic frequency, $\omega_{iaw} = k v_s$. The resonance wave number k_{iaw} is defined by a zero detuning condition, $\Delta k(z) = 0$, where the wave number detuning for the backward scattering reads:

$$\Delta k(z) = k_{iaw}(\omega_{las} - \omega_s) - k_{las}(\omega_{las}) - k_s(\omega_s) \quad (50)$$

The width of the resonance, $\Delta z_{res} \approx (\Delta k')^{-1/2}$, is defined by the condition of small detuning, $\int_{\Delta z_{res}} |\Delta k| dz \sim 1$, where $\Delta k'$ is the spatial derivative of the wave numbers of three interacting waves in the resonance point, $\Delta k(z_{res}) = 0$. The gain factor of the scattered wave, $G = \ln(I_s/I_{s0})$, has been calculated by Rosenbluth (1972) in the WKB approximation, assuming that the width of the resonance region is larger than the wavelengths of the interacting waves:

$$G = \frac{2\pi\gamma_0^2}{|\Delta k' v_{g1} v_{g2}|} \quad (51)$$

where $v_{g1,2}$ are the group velocities of the ion acoustic and scattered waves at the resonance, $v_{g1} = v_s$ and $v_{g2} \approx k_{las} c^2 / \omega_{las}$. The SBS temporal growth rate in a homogeneous plasma, $\gamma_0 \approx 2^{-3/2} \omega_{pi} v_{las} (k_{las} / \omega_{las} v_s)^{1/2}$ is proportional to the electron quiver velocity in the laser field, $v_{las} = e E_{las} / m_e \omega_{las}$.

For SBS, the major factor limiting the width of the amplification region and gain is the inhomogeneity of the ion acoustic wave velocity, $v_s = c_s + u$, where $u(z)$ is the plasma expansion velocity. The phase detuning can be written as $\Delta k' \approx k_{iaw} / L_u$ where $L_u = c_s / (dv_s/dz)$ is the characteristic velocity scale length, and the expression for the SBS gain reads, see Liu et al. (1974):

$$G_{SBS} = \frac{\pi}{8} \frac{n_e}{n_{cr}} \frac{v_{las}^2}{v_{te}^2} \frac{\omega_{las}^2}{k_{las} c^2} L_u \quad (52)$$

The condition $G_{SBS} \gg 1$ corresponds to a strong amplification of the scattered wave from the thermal noise level. The characteristic time of SBS development, $\sim \gamma_0^{-1}$, is on the order of a picosecond. Similarly to self-focusing, it can be controlled by SSD or other methods of laser spectral broadening.

Cross beam energy transfer

SBS is considered as a moderately dangerous nonlinear process. The measured level of backward reflected light in the focusing optics rarely exceeds 10–15% in integrated ICF experiments, with approximately the same amount of light scattered at other angles. More attention was attracted recently to a special form of SBS in the zone of overlapping of two or more laser beams. This situation is common for any laser fusion scheme. In the direct drive approach, many beams are overlapping at the target surface with the goal of creating as homogeneous an irradiation field as possible (see section “Laser imprint” and Fig. 9). In indirect drive, many laser beams are overlapping in the laser entrance aperture of the hohlraum, producing a complicated interference pattern. These beams are interacting strongly in the sonic point where the local plasma expansion velocity is equal to the acoustic velocity, $c_s + u = 0$. As the plasma expands toward the laser, the incident wave is upshifted in the plasma reference system and it transfers its energy to the downshifted reflected wave. As both waves have comparable amplitudes, this process of cross beam energy transfer (CBET) is efficient even at modest gains, $G_{SBS} \sim 1$, and results in energy losses of up to 40%, see Myatt et al. (2014). The effect of speckle self-focusing on CBET has been considered by Raj and Hüller (2017).

Because of such a strong impact on the energy balance, methods for CBET numerical modeling and mitigation have been developed. In the indirect drive experiments on NIF, a frequency shift between the outer and inner laser beams was introduced. This

permitted compensation of the excessive energy losses of the inner beams due to Raman scattering and improved the implosion symmetry, see Michel et al. (2010). In direct drive experiments, in addition to the frequency shift between the laser beams, other CBET mitigation methods have been tested—reduction of laser beam width on the target and doping the ablator with high-Z elements. Unfortunately, none of these methods provided satisfactory results because of secondary effects: reduction of the laser beam waist degrades the low mode implosion symmetry (see section “Symmetry of implosion”), while doping of the ablator decreases the hydrodynamic efficiency (see section “Rocket model”).

Online numerical modeling of CBET with hydrodynamic codes requires knowledge of the laser intensity distribution in the plasma. Several methods of laser intensity evaluation have been developed. The ray-based method proposed by Igumenshchev et al. (2012) evaluates the distance between the adjacent ray trajectories and thus accounts for the laser beam intensity variation due to plasma refraction. The energy exchange between beams has been calculated by pair-wise coupling of all intersecting beams with the corresponding SBS gain. Although this method shows an improved agreement with experiment, it is numerically cumbersome and does not conserve the energy. Another method of “thick rays” considers each ray as an elementary Gaussian beam and solves an equation for the beam front curvature along the ray trajectory, see Colaitis et al. (2016). The pair-wise coupling in this approach is energy conserving and the calculations are less time consuming. However, spatial resolution of these methods is limited by the hydrodynamic mesh size and it cannot resolve the speckle size. Other numerical methods promising high resolution CBET simulations are under development, see for example Colaitis et al. (2019).

Two plasmon decay

Two plasmon decay (TPD) develops in a narrow region near one quarter of the critical density, where the laser frequency is equal twice the plasma frequency, see Fig. 10. The daughter plasma waves are excited in the plane of polarization of the pump wave and they are spatially localized. Consequently, TPD is an absolute instability—the daughter wave amplitudes grow in time and saturate due to nonlinear effects: secondary parametric decay, electron trapping or density profile modification.

TPD is an electronic instability with a high growth rate, $\gamma_{\text{TPD}} \approx \frac{1}{4} k_{\text{las}} v_{te}$, which corresponds to a growth time of typically less than one tenth of a picosecond. Electron collisions are not sufficient to stabilize such a fast growth, but TPD is sensitive to dephasing due to the density inhomogeneity. Thus, the TPD threshold is proportional to the density gradient:

$$v_{\text{las,th}}^2 / v_{te}^2 = 12 / k_{\text{las}} L_n \quad (53)$$

where $v_{te} = (T_e/m_e)^{1/2}$ is the electron thermal velocity and L_n is the density scale length at one quarter on the critical density. The TPD threshold can be written in practical units as:

$$I_{\text{TPD,th}} = 8.2 T_e L_n^{-1} \lambda_{\text{las}}^{-1} 10^{15} \text{ W cm}^{-2} \quad (54)$$

where the electron temperature at one quarter of the critical density is in keV, and L_n and λ_{las} are in microns. For the typical plasma temperature in the corona of 2–4 keV and a density scale length of 300–500 μm , the TPD threshold is in the range of several $10^{14} \text{ W cm}^{-2}$ and it is routinely excited in integrated experiments conducted on the Omega laser facility. It is, however, suppressed by SRS at larger energies in the experiments conducted on NIF, as demonstrated by Rosenberg et al. (2018).

TPD corresponds to excitation of a broad spectrum of plasma waves, from $k_p \sim k_{\text{las}}$ to the Landau damping limit, $k_p \sim 0.3 \lambda_D^{-1}$. They are efficiently coupled to electrons and produce a hot population with an effective temperature, $T_{h,\text{TPD}}$, and energy conversion factor, η_{TPD} , characterized in a series of numerical simulations by Vu et al. (2012):

$$T_{h,\text{TPD}} \approx 15.5 + 17.7 \zeta \text{ keV}, \quad \eta_{\text{TPD}} \approx 2.6 \times 10^{-2} \left(1 - \exp\left(-\sqrt{\xi - 1}\right) \right) \quad (55)$$

where $\zeta = I_{\text{las}}/I_{\text{TPD,th}}$. Typically, a few percent of laser energy is transferred to hot electrons with characteristic energy of the order of 100 keV. These electrons are considered as a serious threat for the direct drive implosions. It is, however, observed that TPD is suppressed in large scale experiments by the competing instability—SRS.

Stimulated Raman scattering

Stimulated Raman scattering (SRS) corresponds to a decay of the incident laser wave into another electromagnetic wave and a daughter electron plasma wave. The instability temporal growth rate, $\gamma_{\text{SRS}} \approx \frac{1}{4} k_{\text{las}} v_{\text{las}} (\omega_{pe}/\omega_s)^{\frac{1}{2}}$ where $\omega_s = \omega_{\text{las}} - \omega_{pe}$ is the frequency of scattered wave, is comparable to that of TPD, but in an inhomogeneous plasma it can be excited in a broad density range from one quarter of the critical density down to a few percent of the critical density, see Fig. 10. The minimum density is defined by the Landau damping of the daughter plasma wave.

SRS is a convective instability at densities below the quarter critical density. The resonance conditions for a given scattered wave frequency can be fulfilled in a narrow zone depending on the density scale length. The spatial detuning is defined by the plasma wave dispersion, $v_{g1} \Delta k_{\text{res}}' \approx d\omega_{pe}/dz = \omega_{pe}/2L_n$. Then, for scattering in the backward direction, according to Eq. (51), the SRS spatial gain reads:

$$G_{\text{SRS}} = \frac{\pi}{4} \frac{v_{\text{las}}^2}{c^2} \frac{k_{\text{las}}^2}{k_s} L_n \quad (56)$$

The wave number of the scattered wave, $k_s = (\omega_{\text{las}}/c) \sqrt{1 - 2\omega_{pe}/\omega_{\text{las}}}$, decreases with the plasma density and, consequently, the gain coefficient increases. This expression is not valid near one quarter of the critical density, where the group velocity of the

scattered wave goes to zero and the WKB approximation used in deriving Eq. (51) fails. At this particular density, SRS becomes an absolute instability, with the threshold lower by a factor of $(k_{\text{las}} L_n)^{1/3}$ than the threshold of the convective SRS, see Liu et al. (1974). In practical units, the SRS threshold near one quarter of the critical density can be written as

$$I_{\text{SRS,th}} = 100 L_n^{-4/3} \lambda_{\text{las}}^{-2/3} 10^{15} \text{ W cm}^{-2} \quad (57)$$

where the laser wavelength and the density scale length are in microns. It is quantitatively comparable to the TPD threshold (54), but the dependence of the density scale length and temperature is different. The TPD threshold is lower at lower temperatures and shorter plasma scale lengths, while the absolute SRS is excited at higher temperatures and in larger plasmas. Side scattering SRS is also an absolute instability below the quarter critical density.

Nonlinear saturation of SRS has been investigated in a large number of numerical simulations. Compared to TPD, absolute SRS saturates at a higher level, reflecting about 10–15% of the incident laser energy in the spectral domain around one half of the laser frequency. It also converts a similar amount of laser energy into a flux of forward propagating hot electrons. The energy distribution of such hot electrons can be approximated by an exponential distribution in energy with a characteristic temperature of 30–50 keV, as shown by Klimo et al. (2014), which is significantly lower than the effective temperature of TPD-generated hot electrons. The dominant role of the absolute SRS instability in hot electron generation at laser intensities exceeding $10^{15} \text{ W cm}^{-2}$ has also been confirmed in experiments by Theobald et al. (2017) and Rosenberg et al. (2018).

Electron energy transport

Laser energy deposited in the underdense plasma, $n_e \leq n_{\text{cr}}$, is transported to the ablation zone at the surface of a solid shell by electrons and radiation, see Fig. 2. Bremsstrahlung emission from the underdense plasma is much smaller than the laser energy flux, so that the electron heat flux dominates. However, it has been recognized at an early stage in ICF research that the electron energy flux in the conduction zone cannot be described by the classical Spitzer heat conductivity, $\mathbf{q}_{\text{SH}} = -\kappa_{\text{SH}} \nabla T_e$ (28). The absorbed energy flux of the order of $10^{15} \text{ W cm}^{-2}$, with a temperature decreasing from a few keV to a few eV, implies that the thickness of the conduction zone has to be comparable to the electron mean free path. Thus, the energy flux in the conduction zone is nonlocal and its appropriate description is one of the permanent preoccupations in ICF modeling.

As the full kinetic description of electron transport is incompatible with the hydrodynamic model, several simplified descriptions have been developed. The most simple approach consists of limiting the Spitzer heat flux with a constant fraction f_{fs} of the free streaming electron energy flux:

$$\mathbf{q}_e = \min \left\{ -\kappa_{\text{SH}} \nabla T_e, -f_{\text{fs}} n_e v_{te} T_e \nabla T_e / |\nabla T_e| \right\} \quad (58)$$

This model, however, is oversimplified, while many experimental observations may be simulated with a hydrodynamic code by choosing an appropriate value of flux limit, the chosen values of f_{fs} vary in a broad range from 0.01 to 0.5, depending on the numerical scheme and physics studied in a particular experiment.

Another model of nonlocal transport proposed by Luciani et al. (1983), describes the electron heat flux at a point, z , in plasma as a convolution of Spitzer fluxes originated at another points, z' , with a kernel, $w(z, z')$, localized on a distance comparable to the electron mean free path, $\lambda_e = \alpha(Z + 1)^{1/2} v_{te}/v_{ei}$:

$$q_e = - \int dz' w(z, z') q_{\text{SH}}(z'), \quad \text{where } w(z, z') = \frac{1}{2} \lambda_e^{-1}(z') \exp \left(- \left| \int_z^{z'} dz'' \lambda_e^{-1}(z'') \right| \right) \quad (59)$$

Comparison of this model with a full kinetic Fokker-Planck simulation shows a good agreement for the numerical factor $\alpha = 32$. Such a large value of the numerical coefficient is explained by the fact that the electrons contributing the most to the energy transport have a velocity 3–4 times larger than the thermal velocity. Their collision length scales as the fourth power of velocity and, consequently, the delocalization length increases by a factor of ≥ 30 compared to the mean free path of a thermal electron.

This nonlocal transport model (with an improved kernel defined in Fourier space) has found applications in describing parametric instabilities, in particular, SBS and filamentation, in a weakly collisional plasma, see Bychenkov et al. (2000). It is not, however, used in hydrodynamic modeling for two reasons. Firstly, such an integral representation of the heat flux is not compatible with a differential form of hydrodynamic equations, and the integral form of Eq. (59) strongly penalizes the performance of numerical simulations. Secondly, this model is intrinsically one-dimensional, which is not sufficient for modeling of three dimensional ICF implosions.

The most successful nonlocal model is that proposed by Schurtz et al. (2000), called the SNB model, which is based on the multigroup approach. Similarly to the Spitzer and Härm approach, this model starts from the approximation of a small anisotropy of the electron distribution function, $f_e(\mathbf{v}, \mathbf{r}) = f_0(v, \mathbf{r}) + (\mathbf{v}/v) \cdot \mathbf{f}_1(v, \mathbf{r})$, where $f_1 \ll f_0$, and uses a simplified BGK form of the collision integrals with the velocity-dependent electron-ion $\nu_{ei}(v)$ and electron-electron $\nu_{ee}(v)$ collision frequencies. Consequently the stationary Fokker-Planck equation is written in the following form:

$$\frac{v}{3} \nabla \cdot \mathbf{f}_1 - \frac{e\mathbf{E}}{3m_e v^2} \cdot \frac{\partial(v^2 \mathbf{f}_1)}{\partial \mathbf{v}} = -r \nu_{ee}(f_0 - f_M) \quad (60)$$

$$v \nabla f_0 - \frac{eE}{m_e} \frac{\partial f_0}{\partial v} = -\frac{\nu_{ei}}{\xi} f_1 \quad (61)$$

Here, the coefficients, $r \sim 1$ and $\xi(Z) = (Z + 0.24)/(Z + 4.2)$, account for the contribution of electron-electron collisions and the self-consistent electric field, E , is defined by the zero current condition, $\mathbf{j} = -4\pi e \int_0^\infty dv v^3 \mathbf{f}_1 = 0$. In contrast to the Spitzer-Härm approach, where the isotropic part of the electron distribution is assumed to be a Maxwellian function, f_M , with density, n_e , and temperature, T_e , depending on the spatial coordinate, the SNB model accounts for the deviation of f_0 from f_M . The solution is constructed in three steps. Firstly, Eq. (61) is solved for $\mathbf{f}_1$ with $f_0 = f_M$. Defining the electric field from the zero current condition, one obtains the Spitzer-like expression for the anisotropic part of electron distribution function:

$$\mathbf{f}_{1M} = -\xi \frac{v}{\nu_{ei}} \left(\frac{m_e v^2}{2T_e} - 4 \right) f_M \frac{\nabla T_e}{T_e} = -\left(\frac{m_e v^2}{2T_e} - 4 \right) g_1 \quad (62)$$

This is the integrand of the classical collisional heat flux, $\mathbf{q}_{SH} = (2\pi m_e/3) \int_0^\infty dv v^5 \mathbf{f}_{1M}$. The term in the parenthesis in (62) accounts for the balance between the current transported by hot electrons and the return current of cold electrons. In the second step, this expression for the anisotropic part of the electron distribution function is used to calculate the corrections $\delta f_0 = f_0 - f_M$ and $\delta \mathbf{f}_1 = \mathbf{f}_1 - \mathbf{f}_{1M}$:

$$r\nu_{ee}\delta f_0 + \frac{v}{3} \nabla \cdot \delta \mathbf{f}_1 = \frac{v}{3} \nabla g_1, \quad v \nabla \delta f_0 = -\nu_{ei}^* \delta \mathbf{f}_1 \quad (63)$$

This set of equations follows from (60), (61) with several simplifying assumptions. The return current and the electric field are neglected on the right hand side of first equation and the collision frequency of hot electrons in the second equation is corrected by the electric field contribution to the friction force, $\nu_{ei}^* = \nu_{ei}/\xi + 2e|E|/m_e v$, with the first order electric field. Then, in the third step, a correction to the electron heat flux is calculated: $\mathbf{q} = \mathbf{q}_{SH} + \delta \mathbf{q}$, where $\delta \mathbf{q} = (2\pi m_e/3) \int_0^\infty dv v^5 \delta \mathbf{f}_1$.

In spite of empirical modifications in the set of equations for the perturbation of the electron distribution function (63), the SNB model has demonstrated a very good performance in a semi-collisional plasma, where the electron-ion mean free path is smaller than a few tenths of the characteristic electron temperature length, $L_T = T_e/|\nabla T_e|$. Modification of the isotropic part of the electron distribution function in the velocity range $(1-10) v_{te}$ was found to be the major cause of the heat flux limitation. A diffusive form of the kinetic equation (63) enables its easy implementation in the hydrodynamic codes, so that the transport simulations are performed inline at each time step.

This model is implemented in many ICF hydrodynamic codes. Fig. 11 shows the performance of the SNB nonlocal model compared to the Spitzer-Härm heat flux and several other models for two representative tests. Firstly, one considers small amplitude temperature modulations, $T_e(z) = T_{e0} + \Delta T_e \sin kz$, with amplitude less than 1%. Panel A shows reduction of the thermal conductivity, defined as $q_e = -\kappa(k)k\Delta T_e \cos kz$, compared to the Spitzer thermal conductivity, see also Brodrick et al. (2017). Points indicate

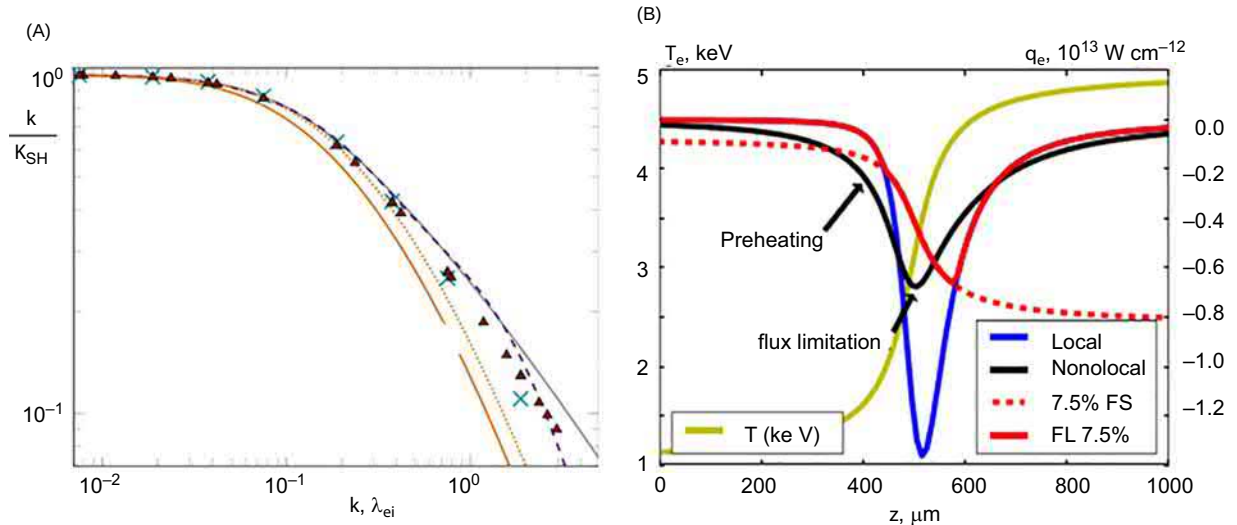


Fig. 11 Performance of the nonlocal transport model: (A) Ratio of the heat transport coefficient, κ , to the value calculated with the Spitzer-Härm theory, κ_{SH} , for the periodic temperature modulations in a plasma with the ion charge $Z = 1$: Points are the results of kinetic simulations; black line—a theoretical fit; red and yellow lines—SNB model with two correction coefficients $r = 2$ and 3, respectively. (B) Calculations of the electron heat flux, q_e , across a steep temperature profile varying from 1 to 5 keV over a characteristic length of $50 \mu\text{m}$ with the electron mean free path of $\sim 3 \mu\text{m}$. Yellow line shows the temperature profile, $T_e(z)$; black line— q_e calculated with the SNB model; blue line—Spitzer heat flux, q_{SH} ; red line— q_e calculated with the flux limit model with $f = 0.075$. For more details see Brodrick JP, Kingham RJ, Marinak MM, et al. (2017) Testing nonlocal models of electron thermal conduction for magnetic and inertial confinement fusion applications. *Physics of Plasmas* 24: 092309. doi:10.1063/1.5001079.

the results of Fokker-Planck calculations, while red solid and dotted lines show the results of the SNB model with two different correction coefficients, $r = 2$ and 3 . A notable deviation starts already at very large wavelengths, $k\lambda_{ei} \sim 0.01$, and the SNB model provides sufficiently accurate results up to $k\lambda_{ei} \sim 0.2$ – 0.3 . This collisionality range is pertinent for ICF. By contrast, the SNB model fails in the collisionless limit, $k\lambda_{ei} \geq 1$, where κ scales as $1/k$.

Fig. 11b shows performance of the SNB model in the case of a large temperature variation from 1 to 5 keV over the characteristic length of 50 μm in a plasma with $Z = 1$ and an electron mean free path of 3 μm . In this case, the Spitzer heat flux is reduced by 40%, which can be accounted for by the flux limiter model with $f = 7.5\%$. However, the spatial profile of heat flux is not reproduced. The SNB model shows an enhancement of the heat flux in a cold plasma, which is produced by energetic electrons and cannot be reproduced with a simple flux limiter. Recent studies by Brodrick et al. (2017) confirmed a good agreement of the SNB model with full kinetic simulations in a broad range of plasma parameters.

This good agreement is, however, limited to cases without external magnetic field. A nonlocal transport in the presence of a magnetic field is not well described by the nonlocal models. This is still an open question. The BGK form of the collision integral is not adequate to describe the joint variation of the temperature and magnetic field profile in a plasma. An advanced kinetic model with a more accurate representation of the electron-electron collisions and entropic closure proposed by Del Sorbo et al. (2017) could be a promising option, but it is not yet sufficiently developed. Such a kinetic model can also be used for describing the energy transport by electrons generated in the plasma corona by parametric instabilities. At present, these hot electrons are described in a Monte Carlo or continuous slowing down approximation, which does not account for self-consistent electric and magnetic fields and thus is not sufficiently accurate.

Relativistic laser pulses

Conventional ICF approaches, both direct and indirect drive, are considering laser energy deposition on a nanosecond time scale and ablation pressures in the range of 100 Mbar. This implies the use of laser intensities in the range of 10^{14} – $10^{15} \text{ W cm}^{-2}$ at a wavelength of 0.35 μm (corresponding to the third harmonic of the Nd:glass laser). The physics of laser-plasma interactions in these conditions is non-relativistic as the electron quiver energy in the laser electric field is in the hundred electron-volt range, more than three orders of magnitude smaller than the electron rest energy. There are, however, alternative approaches to inertial fusion, which aim to create an ignition spot by a direct deposition of laser energy in the compressed fuel, see chapter 12.20 “ICF physics principles” and Kemp et al. (2014), Robinson et al. (2014), Norreys et al. (2014), and Fernández et al. (2014). As discussed in section “Hot spot energy balance”, the hot spot energy at ignition is about 10 kJ and the stagnation time is about 10 ps. Thus, the required laser power is in the range of several petawatt. When focused in a spot of a size less than 100 μm , the laser intensities exceed typically $10^{19} \text{ W cm}^{-2}$ at a wavelength of 1 μm . The electron quiver energy under such conditions is in the mega-electron-volt scale, and thus the laser plasma interaction is relativistic.

The laser radiation cannot directly access the compressed core, and the interaction process proceeds in two (or three) steps: firstly, the intense laser pulse is absorbed and its energy is transferred to a group of relativistic electrons; secondly, these electrons transport this energy to a dense fuel and deposit it preferentially in a small volume thus raising the pressure and temperature locally to the ignition conditions. This scheme is known as electron fast ignition, see Tabak et al. (1994). Alternatively, in the fast ion ignition scheme, see Bychenkov et al. (2001) and Roth et al. (2001), the laser-accelerated electrons transfer their energy to ions, which then penetrate to the dense core and create the ignition spot.

An energy transfer from laser to electrons is very efficient at relativistic intensities. Extensive numerical simulations and experiments show that electrons acquire an exponential energy distribution with an effective temperature, T_{eh} , comparable to their quiver energy in the laser electric field, see Wilks et al. (1992):

$$T_{eh} \approx m_e c^2 \left(\sqrt{1 + a_{\text{las}}^2/2} - 1 \right) \quad (64)$$

where $a_{\text{las}} = eE_{\text{las}}/m_e\omega_{\text{las}}c$ is the dimensionless laser amplitude. The conversion efficiency of laser energy to electrons increases with laser intensity from about 20–30% at a near relativistic intensity of $I_{\text{las}} \sim 1.5 \times 10^{18} \text{ W cm}^{-2}$ (corresponding to $a_{\text{las}} \sim 1$) to up to 80% at $a_{\text{las}} \sim 10$ – 20 .

There is, however, a serious caveat with fast electron ignition. While the electrons are accelerated predominantly in the laser propagation direction, the electron bunch entering into the dense plasma becomes strongly divergent, representing a cone with a characteristic opening angle of 30–40 degrees. This is explained by the electromagnetic Weibel-like instability excited by the return current. Indeed, a beam of relativistic electrons carrying an energy flux of $q_e \sim 10^{19} \text{ W cm}^{-2}$ also carries an enormous current $j_e \sim 10^{12} \text{ A cm}^{-2}$. It is compensated by the return current of plasma electrons ($j_{\text{ret}} = -j_e$) moving in the opposite direction with a velocity, $v_e \sim 10^{-3}c$. The growth rate of an electromagnetic instability, $\sim \omega_{pe}v_e/c$, belongs to the subpicosecond domain and results in a fast beam filamentation and the generation of a strong magnetic field producing a significant beam divergence.

A strong return current in the plasma leads also to an enhanced dissipation rate of the relativistic electron beam, as shown by Bell et al. (1997). In contrast to the laser accelerated relativistic electrons, the plasma electrons are collisional and the return current produces a strong electric field, $E_s = j_e/\sigma_e$, inversely proportional to the plasma conductivity, σ_e . For a typical value of plasma conductivity of 10^7 S , this electric field attains values of about 1 GV m^{-1} , thus slowing down the relativistic electrons over a distance of 1 mm. Moreover, according to the Faraday’s law, the curl of this electric field results in the generation of a magnetic field with a growth rate of about 100 T ps^{-1} that may deflect the electrons and prevent their focusing in a small spot.

The electrons are not well suited to the creation of a compact hot spot. As explained in section “Hot spot energy balance”, one of the conditions for fuel ignition is that the hot spot areal density is comparable with the stopping range of α -particles produced in DT fusion reactions, which is on the order of $0.2\text{--}0.4\text{ g cm}^{-2}$. This areal density corresponds to the stopping range of electrons with energy less than 1 MeV. However, according to Eq. (64), such relatively low energy electrons are produced at a laser intensity of several units of 10^{18} W cm^{-2} , which is not sufficient for providing the petawatt power needed for ignition. Thus, relativistic electron heating inevitably creates a hot spot, which is too large. Although recent integrated experiments on the Japanese laser system GekkoXII-LFEX by Sakata et al. (2018) have demonstrated an efficient guiding of a laser-generated electron beam and energy deposition in the compressed target, the use of relativistic electrons for ignition of fusion reactions remains problematic.

Energetic protons are better suited for creation of a compact hot spot. Their stopping length is shorter than that of electrons and can be adjusted to the optimum size of the ignition spot, as each proton carries more than ten times more energy than an electron and, therefore, less protons are needed. In addition, protons can be better focused in a small spot. However, laser acceleration of protons is less efficient and methods of proton acceleration are less developed.

Laser-ion acceleration proceeds in two steps: firstly, electrons are accelerated in the laser pulse and then they transfer their energy to ions. The most reliable method, called target normal sheath acceleration (TNSA), involves ion acceleration in a plasma rarefaction wave driven by hot electrons. It operates via an interaction of intense and short laser pulses with thin solid foils. The electrons accelerated in a laser pulse on the front surface penetrate the foil and create a Debye sheath, $\lambda_{Dh} = (\epsilon_0 T_{eh}/e^2 n_{eh})^{1/2}$, at the rear surface. The potential drop, $\Phi_{es} \sim T_{eh}/e$, could be as large as 10 MV over the thickness of a few microns, thus creating an electrostatic field, $E_{es} \approx \Phi_{es}/\lambda_{Dh}$, exceeding 1 GV m^{-1} . This field instantaneously ionizes the atoms at the rear surface and accelerates the ions, preferentially protons as they have the largest charge-to-mass ratio, to energies of up to a few times the electrostatic potential. Extensive reviews of the ion acceleration can be found in Daido et al. (2012) and Macchi et al. (2013).

Numerical simulations and experiments show that the ions are accelerated normally to the rear target surface in a narrow cone of $10\text{--}20$ degrees; their energy distribution is broad, exponential-like, with an effective temperature of the order of the hot electron temperature (64). So laser intensities on the order of 10^{20} W cm^{-2} are needed for ion acceleration to tens of MeV energies. Two major obstacles for using the TNSA ions for fast ignition are: (i) relatively small efficiency of ion acceleration, typically a few percent of laser energy is transferred to ions; and (ii) relatively small number of accelerated ions, typically $10^{12}\text{--}10^{13}$ particles per shot. These issues are related to the very small depth of penetration of the sheath electrostatic field in a solid target, which is less than one tenth of micron. In contrast to electron fast ignition, no integrated experiments on ion fast ignition have been performed yet. This could be a challenge for the next generation of high-power, high-energy laser facilities.

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Inertial Confinement Fusion—Experimental Physics: Laser Drive

Sean P. Regan and E. Michael Campbell, Laboratory for Laser Energetics, University of Rochester, Rochester, NY, United States

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Glossary

Areal density The product of the plasma density and thickness.

Ignition Ignition occurs when the yield amplification from the alpha heating exceeds twenty.

Plasma A plasma is a quasineutral gas of charged and neutral particles which exhibits collective behavior due to the long-range electromagnetic forces.

Introduction

In the last seven decades research and development of controlled thermonuclear fusion has been pursued in the laboratory as an alternative energy resource to fossil fuels using magnetically-confined plasmas (Meade, 2009), such as the tokamak (Rutherford, 1980). Since the advent of the laser in the 1960s, inertial confinement fusion schemes using lasers have been investigated (Nuckolls et al., 1972; Kidder, 1974, 1976; Lindl et al., 1992). In the last few decades, inertial confinement fusion using pulsed power has been pursued (Slutz et al., 2010; Gomez et al., 2021). Thermonuclear fusion of the hydrogen isotopes deuterium (D) and tritium (T) involves nuclear fusion of the ions in a high-temperature plasma, where the thermal energy of the ions in the plasma overcomes their long-range Coulomb repulsion and the short-range nuclear force fuses the ions. The high fusion reaction rate of deuterium and tritium (i.e., $D + T \rightarrow {}^4\text{He} + n$) relative to other fusion reactants makes it a practical fuel choice for laboratory fusion plasmas and the majority of fusion research has used this fuel. Nuclear energy is released in the reaction, since the DT fusion converts almost 0.4% of the mass of the hydrogen isotopes into energy through Einstein's mass-energy relationship, $\Delta E = \Delta mc^2$, where ΔE is the energy liberated in the nuclear fusion reaction, Δm is the difference in the mass of the reactants and the fusion products, and c is the speed of light. The alpha particle (${}^4\text{He}$) and the neutron fusion products have birth energies of 3.5 and 14.1 MeV, respectively. Thermonuclear fusion plasmas are designed to absorb the energy of the alpha particle to further increase the plasma temperature leading to more fusion reactions (i.e., alpha heating). The viability of energy production in a fusion plasma is quantified by the Lawson criterion, which compares the rate of energy generation from nuclear fusion to the rate of energy loss in the plasma (Lawson, 1957). For ICF, these loss mechanisms include radiation, thermal conduction, and hydrodynamic expansion. Future inertial fusion power plants would use the energetic neutrons to bombard a blanket containing lithium surrounding the fusion plasma to breed tritium fuel using the following nuclear reaction: $n + {}^6\text{Li} \rightarrow {}^4\text{He} + T$ (Atzeni, 2021). The energy that the neutrons deposit in the blanket can be extracted with a heat exchanger and used boil water for a steam turbine to generate electricity. The three main approaches of inertial confinement fusion are laser direct drive (Goncharov et al., 2014, 2017; Craxton et al., 2015; Regan et al., 2016, 2019; Bose et al., 2016; Campbell et al., 2017, 2021; Gopalaswamy et al., 2019; Campbell, 2021), laser indirect drive (Moses et al., 2009; Glenzer et al., 2010; Meezan et al., 2010; Michel et al., 2010; Haan et al., 2011; Edwards et al., 2011; Landen et al., 2011; Edwards et al., 2013; Hurricane et al., 2014, 2019a,b, 2021; Meezan et al., 2015; Le Pape et al., 2018; Landen et al., 2020; Patel et al., 2020; Town, 2020; Zylstra et al., 2021; Landen et al., 2021), and magnetic direct drive (Slutz et al., 2010, 2018; Sinars et al., 2020; Gomez et al., 2021). This chapter concentrates on the current status of the experimental physics in inertial confinement fusion involving lasers, while magnetic direct drive is covered elsewhere (Gomez et al., 2021).

Laser-driven inertial confinement fusion creates thermonuclear conditions in the laboratory by irradiating a spherical target containing a layer of thermonuclear fuel (Gibson et al., 2009; Johal et al., 2009; Harding et al., 2005, 2018) as shown in Fig. 1 with high-intensity lasers or X-rays (Atzeni, 2021; Tikhonchuk, 2021). The physical dimensions and composition of the ablator depends on the energy incident on the target and the photon wavelength of the driver. Inertial confinement fusion implosions are analogous to a spherical rocket, where the rocket fuel (the ablator) is energized by the incident driver energy and the DT fuel is the payload. The initial pursuit of laser fusion involved laser direct drive, where the outer surface of the target is uniformly irradiated with multiple, temporally-shaped laser beams having a peak, overlapped intensity of $<10^{15}$ watts/cm² on the nanosecond time scale. As mentioned above, the resulting laser-ablation process causes the target to accelerate and implode via the rocket effect, reaching

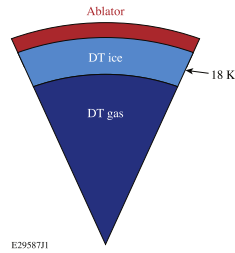


Fig. 1 Schematic of a laser-driven inertial confinement fusion target for central hot-spot ignition, consisting of an ablator surrounding a cryogenic layer of deuterium (D) and tritium (T). The central void contains DT gas.

a peak implosion velocity in the 300 to 500 km/s range depending on the implosion design (Tikhonchuk, 2021). In the early days of inertial confinement fusion research, challenges with nonuniformities in the laser drive incident on the target surface motivated the development of laser indirect drive, whereby the laser energy is converted to X-rays in a high-atomic number radiation case, called a hohlraum, with a near-Planckian spectrum having a temperature of about 300 eV to drive the ablation process (Lindl et al., 1992, 2004; Lindl, 1995, 1998). The two laser-driven, inertial confinement fusion techniques are highlighted schematically in Fig. 2. The spherical concentric layers of a laser-direct-drive inertial confinement fusion target typically consist of a central region of a near equimolar DT vapor surrounded by a cryogenic DT-fuel layer and a thin, nominally plastic (CH) layer, called the ablator (see Fig. 1). The laser-indirect-drive target is similar except the ablator is thicker, due to the efficient x-ray driven ablation, and has multiple concentric layers designed to optimize the implosion (Lindl, 1998; Town, 2020; Tikhonchuk, 2021). In both cases, as the DT-fuel layer decelerates, the initial DT vapor and the fuel mass thermally ablated from the inner surface of the ice layer are compressed and form a central hot-spot plasma, in which fusion reactions occur for a few tenths of a nanosecond around stagnation. The energy coupling from the laser to the hot-spot plasma can be adversely affected by laser plasma instabilities (Kruer, 2003; Tikhonchuk, 2021), hydrodynamic instabilities (Tikhonchuk, 2021), and low- and high-mode drive asymmetries (Town, 2020; Campbell, 2021; Tikhonchuk, 2021). Inertial confinement fusion relies on the 3.5 MeV DT-fusion alpha particles depositing their energy in the hot-spot plasma, causing the hot-spot temperature to rise sharply and a thermonuclear burn wave to propagate out through the surrounding nearly-degenerate, cold dense DT fuel as highlighted in Fig. 3, producing significantly more energy than was used to heat and compress the fuel. The inertia of the compressed DT shell confines the hot-spot plasma for a timescale that is long enough for alpha heating to trigger the ignition instability and achieve energy gain (Town, 2020; Campbell, 2021; Tikhonchuk, 2021; Atzeni, 2021). A burning plasma has a yield amplification due to alpha heating greater than 3.5 (Betti et al., 2015; Hurricane et al., 2019b). Alpha amplification is a metric to describe the increase in overall fusion beyond that due to the hydrodynamic work done during the implosion. An ignited plasma has a yield amplification due to alpha heating in the range of 15 to 25 depending on the implosion design (Christopherson et al., 2019). The total fusion yield depends on the burn up fraction of the DT fuel, which is

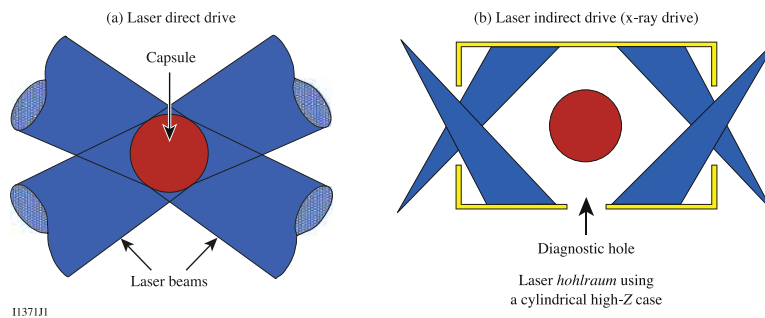


Fig. 2 Schematics highlighting the two laser-driven, inertial confinement fusion techniques: laser direct drive (left panel) and laser indirect drive (right panel).

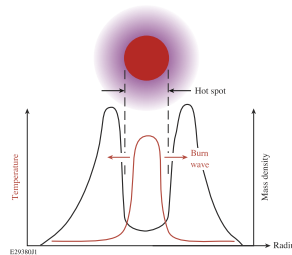


Fig. 3 Inertial confinement fusion relies on the 3.5 MeV DT-fusion alpha particles depositing their energy in the hot-spot plasma, causing the hot-spot temperature to rise sharply and a thermonuclear burn wave to propagate out through the surrounding nearly-degenerate, cold dense DT fuel as highlighted with the spatial profiles of the temperature and density.

a function of the total areal density of the compressed DT fuel (Lindl, 1998). The onset of central-hot-spot ignition is predicted to occur when the product of the temperature and areal density of the hot-spot plasma reach a minimum of $5 \text{ keV} \times 0.3 \text{ g/cm}^2$ (Betti et al., 2002; Lindl et al., 2004; Atzeni and Meyer-ter-Vehn, 2004). A generalized Lawson ignition parameter (hot-spot pressure $\times$ hot-spot confinement time) is used to quantify the proximity to ignition, where a value of unity corresponds to ignition (Betti et al., 2015). Here the pressure and confinement time are estimated without accounting for alpha heating to assess the pure hydrodynamic performance of the implosion. The ignition threshold factor (Spears et al., 2012; Lindl et al., 2014, 2018) used for laser indirect drive implosions is related to the generalized Lawson ignition parameter (Patel et al., 2020).

Central-hot-spot ignition is presently the main line research of laser-driven inertial confinement fusion. Other laser driven inertial confinement fusion techniques under investigation include shock ignition (Betti et al., 2007; Atzeni et al., 2014), fast ignition (Tabak et al., 1994; Kodama et al., 2002; Theobald et al., 2011, 2014), and volumetric ignition with multiple-shell targets (Varnum et al., 2000; Amendt et al., 2002; Molvig et al., 2016; Montgomery et al., 2018; Hu et al., 2019).

This chapter concentrates on the laser-driven central-hot-spot ignition experiments involving implosions of layered DT cryogenic targets (Gibson et al., 2009; Johal et al., 2009; Harding et al., 2005, 2018) conducted at the 192-beam, 2.1-MJ, 500-TW, 351-nm National Ignition Facility (Campbell and Hogan, 1999; Spaeth et al., 2016) and the 60-beam, 30-kJ, 30-TW, 351-nm OMEGA Laser System (Boehly et al., 1997). These lasers are shown in Fig. 4 and described elsewhere (Finnegan, 2021). A primary goal of the National Ignition Facility is to achieve ignition and modest gain using the laser-indirect-drive ignition approach first and then laser direct drive.

Laser-indirect-drive ignition-scale implosions have been performed on the National Ignition Facility since 2009 (Moses et al., 2009; Glenzer et al., 2010; Meezan et al., 2010; Michel et al., 2010; Haan et al., 2011; Edwards et al., 2011, 2013; Landen et al., 2011, 2020, 2021; Hurricane et al., 2014, 2019a, b, 2021; Meezan et al., 2015; Le Pape et al., 2018; Patel et al., 2020; Town, 2020; Zylstra et al., 2021). The laser-indirect-drive implosions on the National Ignition Facility are designed to investigate alpha heating, burning plasma, and ignition target designs (Hurricane et al., 2014, 2019a, b, 2021; Patel et al., 2020; Town, 2020; Zylstra et al., 2021). These experiments have demonstrated alpha heating and are accessing the burning plasma regime, which is an encouraging development in the pursuit of the grand research challenge of ignition.

Several facility enhancements are required on the National Ignition Facility to conduct laser-direct-drive ignition-scale implosions. Progress made with laser direct drive over the last decade motivates laser direct drive implosions on the National Ignition Facility (Goncharov et al., 2014, 2017; Craxton et al., 2015; Regan et al., 2016, 2019; Bose et al., 2016; Campbell et al., 2017; Gopalaswamy et al., 2019; Campbell, 2021; Campbell et al., 2021), since laser direct drive increases the amount of energy coupled to the hot spot by a factor of several over the laser indirect drive, which enables increased mass in the hot spot, leading to a potential increase in the margin for ignition. Energy coupling from the laser to the implosion capsule is an important consideration in inertial confinement fusion. The larger imploded mass and increased coupled energy results in lower required convergence and hot-spot pressure and the possibility of achieving MJ yields at implosion adiabats greater than three (Goncharov et al., 2014, 2016, 2017). The adiabat is the ratio of the pressure in the cold DT fuel to the Fermi-degenerate pressure, which is the minimum energy density due to the degeneracy of the electrons that can be achieved (Tikhonchuk, 2021). Understanding the laser and target physics requirements for ignition is important for the inertial confinement fusion research program on the National Ignition Facility and for the research and development of a next-generation laser facility designed to achieve a robust burning-plasma platform and high yield (i.e., $>200 \text{ MJ}$).

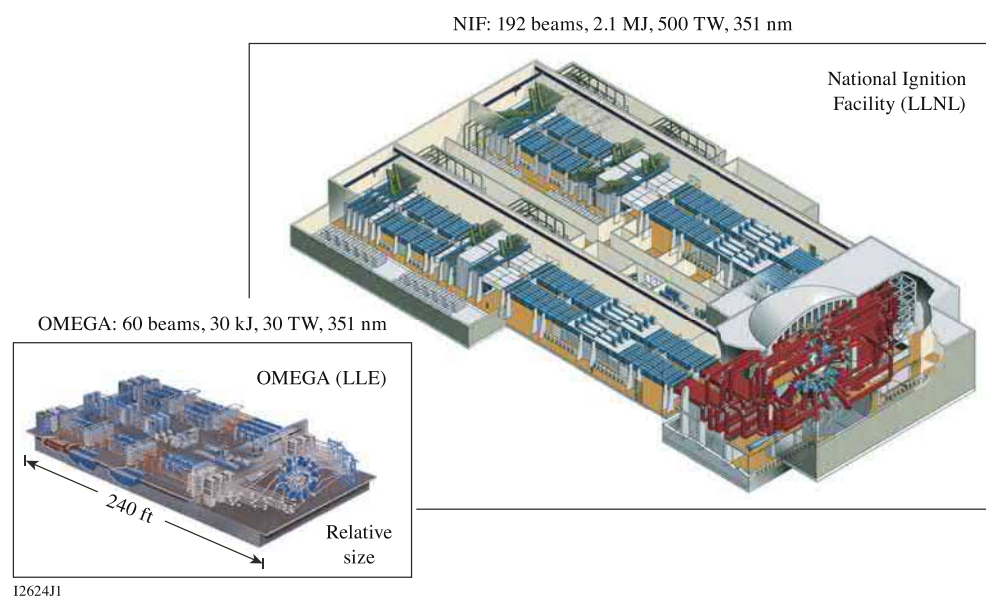


Fig. 4 Schematics of the 60-beam, 30-kJ, 30-TW, 351-nm OMEGA Laser System at the University of Rochester's Laboratory for Laser Energetics (LLE) and the 192-beam, 2.1-MJ, 500-TW, 351-nm National Ignition Facility at the Lawrence Livermore National Laboratory (LLNL).

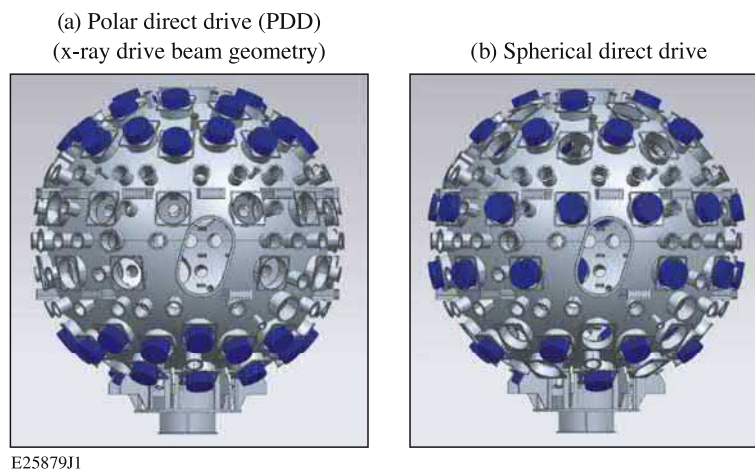


Fig. 5 Schematics of the NIF target chamber with the blue circles highlighting the beam configuration for laser indirect drive (X-ray drive) and Polar Direct Drive (left panel) and the spherically-symmetric beam configuration for laser direct drive, called spherical direct drive (right panel) on the National Ignition Facility. The current beam configuration on the National Ignition Facility is optimized for laser indirect drive (left panel).

Although the beam configuration on the National Ignition Facility is optimized for laser indirect drive with a right circular cylindrical hohlraum by arranging the beams around the poles of the target chamber as illustrated with the blue circles in the left panel of Fig. 5, it can also be used for a variant of laser direct drive called Polar Direct Drive (Skupsky et al., 2004; Craxton et al., 2005; Craxton and Jacobs-Perkins, 2005; Cobble et al., 2012; Radha et al., 2012, 2013, 2016; Krashenninnikova et al., 2014; Hohenberger et al., 2015; Rosenberg et al., 2018; Marozas et al., 2018a, b). The optimal arrangement for laser direct drive with the beams uniformly distributed around the target chamber is highlighted with the blue circles in the right panel of Fig. 5, called spherical direct drive. Although the NIF target chamber has ports for spherical illumination, switching between these two beam geometries requires a major, resource-intensive engineering effort. The laser direct drive ignition approach on the National Ignition Facility is Polar Direct Drive for the next five to ten years. In the Polar Direct Drive configuration, which is also studied in experiments on OMEGA using only 40 beams (Marshall et al., 2006, 2016; Radha et al., 2012), a subset of the beams is repointed toward the equator of the target to recover the spatial uniformity of the on-target, overlapped laser-drive intensity. Several of the key physics aspects of laser direct drive, including energy coupling, laser plasma instabilities, and hydrodynamic instabilities, at ignition-relevant scale lengths are studied on the National Ignition Facility using a polar-direct-drive configuration (Radha et al., 2013, 2016; Krashenninnikova et al., 2014; Hohenberger et al., 2015; Rosenberg et al., 2018; Marozas et al., 2018a, 2018b; Campbell, 2021).

Significant laser and cryogenic target upgrades are required to perform high-convergence (i.e., convergence ratio > 15), laser-direct-drive implosions using the polar-direct-drive configuration for burning plasma and ignition on the National Ignition Facility (Campbell, 2021). Planning is underway to implement these polar direct drive facility enhancements to the National Ignition Facility over the next 10 years (Campbell, 2021). Hot-spot formation for spherically symmetric, laser-direct-drive, DT-layered implosions is studied on the OMEGA laser system using hydrodynamically-scaled ignition targets (Nora et al., 2014; Goncharov et al., 2014; Regan et al., 2016; Bose et al., 2016; Gopalaswamy et al., 2019), since OMEGA does not have enough laser energy to assemble an ignition-scale hot-spot plasma. A primary goal of OMEGA is to investigate subscale laser-direct-drive implosions and the underpinning fundamental physics of implosion designs that are hydrodynamically equivalent (i.e., maintaining the same shell convergence, hot-spot pressure, shell adiabat, and implosion velocity) to burning plasma and ignition designs on MJ-scale lasers in both polar and spherical illumination geometries (Campbell, 2021).

This chapter presents a concise overview of the current status of the experimental physics research for central-hot-spot ignition using laser-driven inertial confinement fusion. More detailed descriptions can be found in recent review reports for laser direct drive (Campbell, 2021) and laser indirect drive (Town, 2020). The energy coupling from the laser to the hot-spot plasma and the degradation mechanisms to implosion performance from laser plasma instabilities, hydrodynamic instabilities, and low-mode drive and target asymmetries (Tikhonchuk, 2021) are discussed for the two laser-driven inertial confinement fusion approaches using central hot-spot ignition. Section “**Laser-Direct-Drive Research**” presents the performance of laser-direct-drive implosions on OMEGA that are hydrodynamically-scaled to the 2.0 MJ of laser energy on the National Ignition Facility (Goncharov et al., 2014; Igumenshchev et al., 2016; Regan et al., 2016; Bose et al., 2016; Gopalaswamy et al., 2019; Campbell, 2021; Campbell et al., 2021). These hydrodynamically-scaled, spherical direct drive implosions are predicted to achieve a threefold amplification in fusion yield due to alpha heating, corresponding to a total fusion yield of about 0.6 MJ (Gopalaswamy et al., 2019; Campbell, 2021). The energy-scaled, generalized Lawson ignition parameter (Betti et al., 2015; Bose et al., 2016) for laser direct drive implosions on OMEGA is approximately 0.75 (Campbell, 2021). The research and development of high-energy, solid-state laser drivers that mitigate laser-plasma instabilities and laser-beam imprint via enhanced spectral bandwidth to increase the implosion performance are also discussed in this chapter (Follett et al., 2018, 2019; Bates et al., 2018; Dorner et al., 2020). Section “**Laser Indirect Drive Research**” summarizes the progress of the laser-indirect-drive approach on the National Ignition Facility demonstrating alpha

heating and accessing the burning plasma regime with ~ 2 MJ of laser energy, corresponding to ~ 0.06 MJ of total fusion yield, and the quest for the ignition challenge (Moses et al., 2009; Glenzer et al., 2010; Meezan et al., 2010, 2015; Michel et al., 2010; Haan et al., 2011; Edwards et al., 2011, 2013; Landen et al., 2011; Hurricane et al., 2014; Le Pape et al., 2018; Hurricane et al., 2019a, b; Landen et al., 2020, 2021; Patel et al., 2020; Town, 2020; Zylstra et al., 2021; Hurricane et al., 2021). The generalized Lawson ignition parameter for laser-indirect-drive implosions on the National Ignition Facility is 0.75 (Patel et al., 2020). The path forward for ignition on the National Ignition Facility with laser indirect drive is also discussed.

Laser-Direct-Drive Research

As mentioned in the introduction, the high-intensity lasers are incident directly on the target for the laser-direct-drive approach. There are four stages to the resulting implosion: initial plasma formation, acceleration, deceleration, and stagnation, which are highlighted in Fig. 6 (Campbell, 2021; Tikhonchuk, 2021). Depending on the implosion design, the peak shell velocity at the end of the acceleration phase is in the 300–500 km/s range. After the laser ionizes the target surface and the coronal plasma is formed, the laser light is absorbed in the under-dense plasma via inverse Bremsstrahlung and a fraction of the absorbed energy flows to the ablation surface via electron thermal conduction. The coronal plasma typically absorbs 60–70% of the laser energy and hydrodynamic efficiency of converting that absorbed laser energy to inward kinetic energy of the shell via the rocket effect is about 9%. This gives a conversion efficiency of incident laser energy to shell kinetic energy of about 6% for laser direct drive. The laser ablation launches a shock wave into the target. As the coronal plasma forms, spatial nonuniformities in the laser irradiation can imprint on the ablation surface (i.e., laser imprint) and mass perturbations on the target from microscopic debris, the support stalk, and the fill tube seed the Richtmyer-Meshkov hydrodynamic instability (Richtmyer, 1960; Meshkov, 1969; Goncharov, 1999). The key target physics during the initial plasma formation are laser uniformity, laser absorption, energy coupling, ablation, shock propagation, laser imprint, and the Richtmyer-Meshkov hydrodynamic instability. The time history of the laser intensity on target, shown in the right panel of Fig. 6, is designed to minimize the rise in the entropy of the compressed shell, quantified by the shell adiabat, in order to minimize the amount of laser energy needed for ignition (Tikhonchuk, 2021). Although lower-adiabat ignition target designs require less laser energy to achieve ignition (Herrmann et al., 2001; Betti et al., 2002), they are more susceptible to hydrodynamic instabilities (Betti et al., 1998).

After the shock wave reaches the inside layer of the DT shell and the rarefaction wave returns to the ablation surface, the DT shell begins to accelerate (see Fig. 6). Mass perturbations at the ablation surface due to target defects and laser nonuniformities can seed the Rayleigh Taylor hydrodynamic instability (Rayleigh, 1945; Taylor, 1950; Cole et al., 1982). During the acceleration phase, as mentioned above, the intensity of the laser irradiation may exceed the thresholds to excite electron waves in the plasma via the Stimulated Raman Scattering or the Two Plasmon Decay instability and ion waves in the plasma via the Stimulated Brillouin Scattering (Kruer, 2003). Excitation of ion waves in the plasma can divert energy flow from the laser to the ablation front. In the cross-beam energy transfer process (Igumenshchev et al., 2010, 2012; Froula et al., 2012), nonabsorbed light that is reflected or scattered from its critical surface or refracted from the under-dense plasma acts as an electromagnetic seed for the stimulated Brillouin scatter of incoming (incident) light (Randall and Albritton, 1981). Cross beam energy transfer has been shown to reduce the target absorption and resulting ablation pressure of laser direct drive targets by as much as 40% on OMEGA (Igumenshchev et al., 2010, 2012, Froula et al., 2012) and 60% on polar direct drive targets for the National Ignition Facility (Radha et al., 2016). Excitation of electron waves

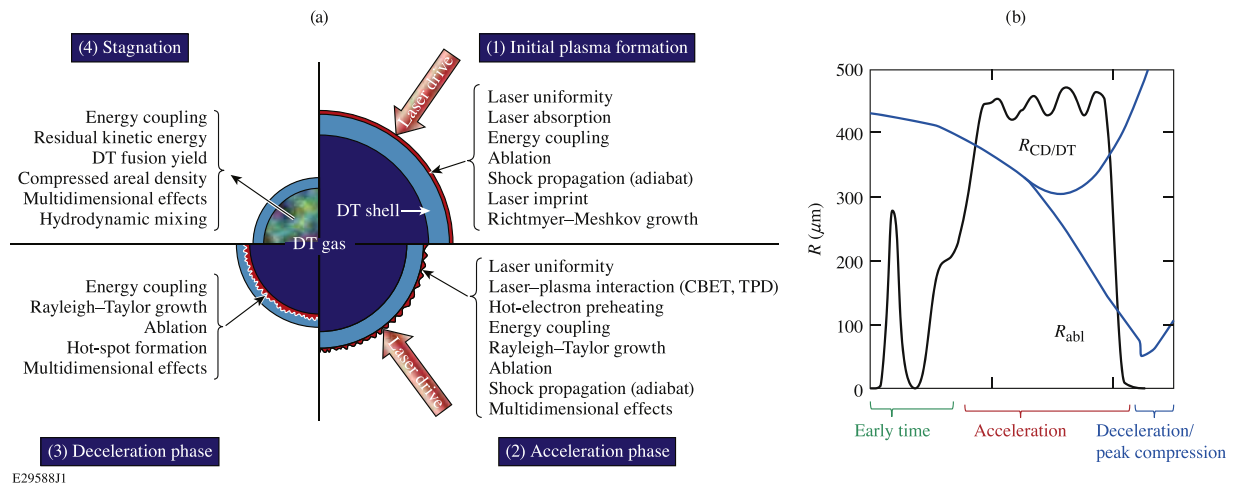


Fig. 6 (Left panel) The four stages of the laser direct drive implosion—initial plasma formation, acceleration, deceleration, and stagnation—and associated key target physics areas. The peak neutron production occurs at stagnation. (Right panel) The time history of the laser power (black curve) and the trajectory of the ablation surface of the imploding shell R_{abl} (lower blue curve) and the trajectory of the ablated CD/DT interface $R_{CD/DT}$ (upper blue curve).

can cause suprathermal electron generation, which has the detrimental effect of preheating the DT fuel and raising the adiabat (Rosenberg et al., 2018; Solodov et al., 2020; Tikhonchuk, 2021). Deviations from a spherical implosion or multidimensional effects can be caused by hydrodynamic instabilities and laser-plasma instabilities. The key target physics during the acceleration phase are laser uniformity, laser-plasma interactions, hot-electron preheating, energy coupling, Rayleigh-Taylor hydrodynamic instability, ablation, shock propagation, and multidimensional effects.

The displacement work that the converging DT shell performs on the nascent hot-spot plasma increases the hot-spot pressure and causes the DT shell to decelerate (see Fig. 6). The inner wall of the DT shell can become unstable to the Rayleigh-Taylor hydrodynamic instability (Tikhonchuk, 2021). Multidimensional effects are amplified during the deceleration phase and can lead to perturbations to the hot-spot plasma (Woo et al., 2018; Shah et al., 2020; Mannion et al., 2021). The key target physics during the deceleration phase are energy coupling, Rayleigh-Taylor growth, ablation, hot-spot formation, and multidimensional effects. The peak values of DT fusion yield and the compressed areal density can be limited at stagnation by hydrodynamic mixing of the ablator and DT ice layer with the hot-spot plasma at stagnation (see Fig. 6) and incomplete conversion of shell kinetic energy into internal energy of the hot spot due to low-mode asymmetries (Woo et al., 2018; Mannion et al., 2021). The key target physics at stagnation are energy coupling, residual kinetic energy, DT fusion yield, compressed areal density, multidimensional effects, and hydrodynamic mixing.

The performance of a laser direct drive ignition target depends on the implosion symmetry (Woo et al., 2018; Mannion et al., 2021), the hydrodynamic instabilities seeded by laser imprint and target features (e.g., microscopic surface debris, fill tube or stalk) (Igumenshchev et al., 2013), the laser-plasma instabilities in the coronal plasma (Kruer, 2003; Igumenshchev et al., 2010; Igumenshchev et al., 2012; Froula et al., 2012; Randall and Albritton, 1981; Rosenberg et al., 2018; Marozas et al., 2018, b; Kruschwitz et al., 2019; Turnbull et al., 2020a, b; Solodov et al., 2020; Tikhonchuk, 2021), and the fundamental material properties (i.e., equation of state, opacity, and conductivity) (Hu et al., 2011, 2015, 2017, 2018; Ding and Hu, 2017; Campbell, 2021). In the near term, mitigation strategies for laser plasma instabilities include laser (i.e., wavelength detuning) (Marozas et al., 2018a, Marozas et al., 2018b, Kruschwitz et al., 2019, Turnbull et al., 2020a, Turnbull et al., 2020b,) and target solutions (i.e., buried Si layer in ablator) (Solodov et al., 2020) that can be implemented on the National Ignition Facility. In the longer term the Laboratory for Laser Energetics is exploring broadband ultraviolet (351-nm) laser technologies to mitigate laser plasma instabilities (Follett et al., 2018, 2019; Bates et al., 2018) and laser imprint, leading to more stable implosions with a higher fraction of laser energy coupled to the hot spot. Recently, the Laboratory for Laser Energetics has demonstrated a significant breakthrough with a novel efficient broadband tripling scheme ($\Delta\omega/\omega > 1.5\%$, > 13 THz) (Dorrer et al., 2020). A staged plan has been devised to develop this new laser technology over the next few years for laser direct drive as well as laser indirect drive (Campbell, 2021).

As mentioned in the introduction laser and target upgrades are required on the National Ignition Facility to perform high convergence ($CR > 15$) polar-direct-drive implosions. The current laser-direct-drive research strategy for estimating the performance scaling from OMEGA to the National Ignition Facility in the absence of these upgrades starts from the theoretical scaling to the NIF, positing the same core conditions achieved on OMEGA can be reproduced at the NIF scale (i.e., the same pressure and scaled density and temperature) (Campbell, 2021). This scaling approximates conservatively that engineering target features (e.g., the stalk or the fill tube) and target-fabrication imperfections that degrade the implosion, scale with size. The impact of all the sources of degradation, including energy coupling, hot-electron preheat, target imperfections, and laser-drive asymmetry, expected at the National Ignition Facility scale are assessed and evaluated through dedicated laser plasma interaction and hydrodynamic experiments on the National Ignition Facility and OMEGA. A comparison of polar direct drive and spherical direct drive laser-irradiation geometries on OMEGA is investigated. The design parameter space available for the laser direct drive approach is determined, and the hydrodynamic scaling is verified within the energy range accessible to the OMEGA laser.

The performance of laser-direct-drive implosions on OMEGA that are hydrodynamically-scaled to the 2.0 MJ of laser energy on the National Ignition Facility assuming spherical direct drive (Goncharov et al., 2014; Regan et al., 2016; Bose et al., 2016; Gopalaswamy et al., 2019; Campbell, 2021; Campbell et al., 2021) were evaluated. The best performing hydrodynamically-scaled, spherical direct drive implosions are predicted to achieve a threefold amplification in fusion yield due to alpha heating, corresponding to a total fusion yield of 0.6 MJ of fusion yield (Gopalaswamy et al., 2019; Campbell, 2021). The energy-scaled, generalized Lawson ignition parameter (Bose et al., 2016) for laser direct drive implosions on OMEGA is approximately 0.75 (Campbell, 2021). A statistical approach was used to design and quantitatively predict the results of the highest-performing laser-direct-drive implosion on OMEGA, leading to tripling of the fusion yield (Gopalaswamy et al., 2019).

A 10-year plan with four physics goals is being developed to implement upgrades to the National Ignition Facility to optimize the performance of polar direct drive DT cryogenic implosion targets in the burning-plasma regime, and to understand the laser facility and target requirements (i.e., energy, power, size, beam smoothing, phase plates, and wavelength detuning) needed to achieve ignition and multi-MJ yield. The first physics goal is to optimize the symmetry of convergence ratio ≤ 10 , polar direct drive implosions of room-temperature DT-filled capsules using the current NIF capability and innovative target designs with a goal of achieving a 50-kJ fusion yield. The second physics goal is to improve with modest investment the energy coupling and symmetry for those implosions using wavelength detuning with beam remapping and dedicated polar direct drive phase plates. The third and fourth physics goals require significant modifications on the National Ignition Facility, including a laser direct drive cryogenic target-handling system and upgrades in laser beam smoothing, to increase the convergence ratio to 15 and then to 20 with polar direct drive DT cryogenic target implosions in order to achieve an alpha burner and to reach the burning-plasma regime (i.e., 1-MJ fusion yield), respectively.

Laser-Indirect-Drive Research

Ignition-scale implosions are being performed on the National Ignition Facility using laser indirect drive. As mentioned in the introduction, the challenges with nonuniformities in the laser drive incident on the target surface in the early days of inertial confinement fusion research motivated the development of laser indirect drive (Lindl et al., 1992, 2004; Lindl, 1995, 1998). The laser beams incident on the inner wall of the high-Z hohlraum wall (see right panel of Fig. 2) create an ablation plasma, which converts the laser energy to the x-ray drive (Lindl, 1998). Many of the target physics areas described in the Laser-Direct-Drive Research section are common to Laser-Indirect-Drive Research; however, there are important distinctions related to radiation hydrodynamics of the hohlraum (Lindl, 1998; Barrios et al., 2018; Callahan et al., 2018, 2020; Tikhonchuk, 2021), laser plasma interactions within the hohlraum (Glenzer et al., 2010; Kritcher et al., 2018; Callahan et al., 2020; Tikhonchuk, 2021), conversion of laser energy to shell kinetic energy (Lindl, 1998; Tikhonchuk, 2021), and the seeds of hydrodynamic instabilities (Barrios et al. 2013; Nagel et al., 2015; MacPhee et al., 2017; Martinez et al., 2017; Hammel et al., 2016, 2018; Tikhonchuk, 2021).

Two of the main design considerations for the hohlraum radiation hydrodynamics are the expansion of the high-Z ablation plasma into the hohlraum and the plasma conditions in the laser entrance hole region of the hohlraum, which can adversely affect the laser to x-ray energy coupling and the implosion symmetry (Glenzer et al., 2010; Barrios et al., 2018; Kritcher et al., 2018; Callahan et al., 2018, 2020; Tikhonchuk, 2021). The hohlraum is filled with a low-Z gas, which is ionized by the lasers forming a plasma that tamps the expansion of the high-Z ablation plasma. Target design considerations are made to optimize the energy coupling and implosion symmetry, while mitigating the laser plasma interactions in the hohlraum. Because of the area ratio of the capsule to the inner wall of the hohlraum, approximately 10% of the laser energy is converted to X rays in a hohlraum that are incident on the implosion capsule. The X rays deposit their energy close to the ablation surface of the implosion capsule, which increases the hydrodynamic efficiency of converting the absorbed X-ray energy to inward kinetic energy of the shell via the rocket effect to about 15% versus 9% for laser direct drive. This gives a conversion efficiency of laser energy to shell kinetic energy of about 1.5% for laser indirect drive versus 6% for laser direct drive. Consequently, laser indirect drive solves the laser imprint problem at a cost of lowering the conversion efficiency of laser energy to shell kinetic energy. Initiatives to increase the energy coupling to the capsule are explored using innovative hohlraum designs (Amendt et al., 2019) and increasing the size of the capsule, while keeping the size of the hohlraum fixed (Callahan et al., 2018, 2020). However, these initiatives will only lead to a modest increase in the energy coupling.

Over the last decade the primary neutron yield for laser indirect drive implosions on the National Ignition Facility has been increased from 2×10^{14} to 2×10^{16} (Moses et al., 2009; Glenzer et al., 2010; Meezan et al., 2010, 2015; Michel et al., 2010; Haan et al., 2011; Edwards et al., 2011, 2013; Landen et al., 2011, 2021; Hurricane et al., 2014; Le Pape et al., 2018, Hurricane et al., 2019a, b, 2021; Landen et al., 2020; Patel et al., 2020; Town, 2020; Zylstra et al., 2021). After an initial series of implosions on the National Ignition Facility, where the DT fuel was diluted with hydrogen (H) to reduce the yield (i.e., THD implosions), low-adiabat ignition designs having predicted yields around 3.6×10^{17} or greater than 1 MJ of fusion yield were explored (Haan et al., 2011). When the high neutron yields were not realized in the laboratory, a more phased approach using higher-adiabat implosion designs was undertaken to understand the implosion results (Hurricane et al., 2014). Using the higher-adiabat implosion, that is less susceptible to hydrodynamic instabilities, driven with a low-fill density hohlraums, that have lower levels of laser plasma interactions, alpha heating was demonstrated and burning plasma is now being explored on the National Ignition Facility with laser indirect drive. The highest yield published to date corresponds to 56 kJ of thermonuclear fusion energy, where alpha heating has boosted the fusion yield by a factor of three from that caused by the implosion system alone (Town, 2020; Patel et al., 2020). The generalized Lawson ignition parameter for laser indirect drive implosions on the National Ignition Facility is 0.75 (Patel et al., 2020).

The current hypothesis is the performance of the current laser indirect drive implosions on the National Ignition Facility is limited by the low amount of energy coupled to the hot-spot plasma and two physical degradation mechanisms (Town, 2020). The mode-1 asymmetry caused by the laser imbalance or capsule nonuniformities (Rinderknecht et al., 2020; Casey et al., 2021) and hydrodynamic instabilities seeded by engineering features such as the fill tube (Hammel et al., 2011; Regan et al., 2013; Ma et al., 2013; Pak et al., 2020), capsule support (Nagel et al., 2015; MacPhee et al., 2017; Hammel et al., 2016, 2018; Martinez et al., 2017), and capsule imperfections (Barrios et al., 2013; Town, 2020). This hypothesis is based on the current understanding of experimental results and modeling of alpha heating (Zylstra and Hurricane, 2019; Patel et al., 2020), hohlraum performance (Hall et al., 2017; Kritcher et al., 2018; Callahan et al., 2018, 2020), laser plasma interactions (Kruer, 2003; Regan et al., 2010; Glenzer et al., 2010; Dewald et al., 2016; Kritcher et al., 2018; Callahan et al., 2018; Callahan et al., 2020; Tikhonchuk, 2021), low-mode asymmetry (Kritcher et al., 2014, 2018; Rinderknecht et al., 2020; Casey et al., 2021), engineering feature such as fill tubes, capsule support, capsule quality (Martinez et al., 2017), and the quality of the DT cryogenic fuel layer (Gibson et al., 2009; Johal et al., 2009), low compression (Landen et al., 2020, 2021), implosion reproducibility (Town, 2020).

The thrust of the research over the next five years for laser indirect drive on the National Ignition Facility will concentrate on the following: (1) increasing amount of laser energy coupled to the implosion capsule (Amendt et al., 2019); (2) optimizing implosion symmetry (Callahan et al., 2018; Callahan et al., 2020); (3) mitigating hydrodynamic mixing of material into the hot-spot plasma; (4) increasing the convergence ratio of the implosion and the compressed fuel density (Landen et al., 2020, 2021); (5) improving predictive capability of implosion performance (Marinak et al., 2001; Clark et al., 2011, 2017, 2019a, b; Callahan et al., 2018, 2020; Gaffney et al., 2018, 2019); (6) increasing the energy and power of the National Ignition Facility (Town, 2020).

Ignition is a grand research challenge. Estimates for the amount of laser energy needed on the National Ignition Facility path forward to ignition with laser indirect drive on the National Ignition Facility range from 1.5 times the current 2 MJ of available laser

energy, if lower-adiabat implosion with higher energy coupling are viable, to 10 times the current 2 MJ of available laser energy, if no improvements are made to the current implosions (Town, 2020). Understanding the science and improvements in target fabrication with systematic experiments over the next several years will reduce the uncertainty in required driver energy and increase the fusion performance at the National Ignition Facility. Although incremental improvements to the performance are ongoing, understanding predictable ignition is crucial for the laser-driven inertial confinement fusion research.

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Inertial Confinement Fusion—Experimental Physics: Heavy Ion Beam Drive

Peter A. Seidl, Lawrence Berkeley National Laboratory, Berkeley, CA, United States

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Introduction

The idea of using accelerators for inertial fusion energy (IFE) was born during the “energy crisis” of the mid-1970s. The enthusiasm in the accelerator community was due in part to the fact that particle accelerators had been built for decades producing ever increasing particle energy and beam current. Many were reliable facilities generating high intensity beams, running nearly non-stop for weeks or months for nuclear and particle physics research and a growing number of industrial applications. Megajoules of beam energy (ion kinetic energy $\times$ number of particles in the beam), and terawatts of beam power had been demonstrated for high-energy physics and nuclear physics research. The ability to steer and focus such beams to sub-mm spots was well established.

As described in earlier chapters covering inertial fusion energy or inertial confinement fusion, the energy to compress and heat the fusion fuel is delivered with high intensity laser beams, ion beams or X-rays. While many of the first IFE experiments used lasers, the prospects for accelerator driven IFE gained momentum. High-level reviews of the US inertial fusion program through the 1990s concluded that accelerators were the most promising long-term approach (NRC, 1986, 1990; FPAC, 1990; FEAC, 1994; Sheffield et al., 1996). Ion driven inertial fusion is particularly compatible with the protection of the structural wall with neutron-absorbing, thick flowing liquid. In contrast to laser IFE, the final optical elements (electromagnets) can be straightforwardly shielded from line-of-sight irradiation and debris from the target.

In 1994 the US Department of Energy decided to build the National Ignition Facility (NIF), a laser-based facility designed to evaluate the feasibility of inertial fusion in the laboratory and as a testbed for nuclear weapons topics. Completed in 2009, a campaign of integrated ignition experiments began in 2010. The 2 MJ laser system has performed as designed, operating very reliably and has led to an evolution in the understanding of target physics. The results have pointed to issues that have, to date, impeded the demonstration of ignition at NIF. Nevertheless, as discussed by Regan (2021), there are reasons for optimism for ignition at NIF.

In 2003, the Department of Energy decided to de-emphasize IFE research, including heavy ion research, pending the demonstration of ignition on the NIF. But it continued accelerator research to study high energy density physics (HEDP) and warm dense matter physics (WDMP). This means creating, for a short time, states of matter exceeding temperatures of 10^4 K and pressures above the GPa range. This intersection of plasma and condensed matter physics is of interest for the study of giant planets, and stellar interiors. Also, as inertial fusion fuel is heated it passes through this regime.

This article aims to show the experimental progress supporting the heavy ion fusion (HIF) approach to inertial fusion energy, and is an introduction to the more detailed descriptions in the references. Bangerter et al. (2013) and Hofmann (2018) are recent reviews of HIF aimed at accelerator experts, including target physics and specific designs which in turn set the accelerator requirements. More experimental results than described here have helped develop HIF to the present promising status.

Target physics for HIF

The physics of compressing and igniting inertial fusion targets is described by Atzeni (2021), Tikhonchuk (2021) and Bangerter et al. (2013). Since different drivers (lasers, pinches, heavy ions) might have particular considerations, we briefly summarize the important issues and put heavy ion targets in perspective.

As for all IFE targets, it is necessary to trap the energy of the fusion reaction products in the fuel to assist with ignition and burn. This sets a lower limit on the size of the target (the fuel capsule radius is around 2 mm), and therefore the energy required to compress and ignite the target should be above roughly 1 MJ, deposited in ~ 10 ns. Target designs call for multiple beams, partly to compress the fuel uniformly, and also because it is very difficult to produce the required total beam power without multiple beams. Energetic ions are rapidly stripped of their electrons, greatly increasing their stopping power. Heavier elements have a shorter range at a given kinetic energy (Fig. 1). The suitable range for HIF targets is between 0.03 and 3 g/cm². Which ions and what beam current should be used? Maschke (1975) proposed that beams of heavy ions with a kinetic energy of several GeV would minimize the required beam current while meeting the desired ion range for the target dimension noted above. (Low kinetic energy—e.g., 20 MeV—protons have an appropriate range but the required beam current leads to greater problems with the ability to focus the beam.) A repetition rate of ~ 10 Hz is required to generate enough fusion energy yield to generate 1 GW of electrical power, a common output for an electrical power plant. Target, driver, and reactor designs can also be adapted for lower or higher electrical power in response to evolving electric grid paradigms.

HIF targets may be characterized by the implosion mode and the ignition mode. Direct drive and indirect drive are implosion modes, each with advantages and disadvantages. Direct drive targets, in which the ions penetrate the outer layer of the fuel capsule, ablate the fuel to produce high pressure to compress the fuel. This approach can be two to three times more efficient in the transfer of beam energy to the imploding fusion fuel than indirect drive targets. However, the tolerances on positioning and beam spot smoothness are more demanding.

For indirect drive targets, the beam heats a material inside a high-atomic-number hohlraum or cavity, which in turn produces radiation which heats the ablator material on the fuel capsule at the center of the hohlraum cavity. This transfer of energy has an energy penalty, and indirect drive targets will require a correspondingly higher total beam energy to compress and ignite a similar fusion pellet than direct drive targets. For indirect drive IFE targets the physics of the capsule implosion is the same for HIF targets and laser IFE targets, an important synergism with the NIF ignition. Indirectly driven targets might be more complex to fabricate than direct drive targets, consisting of various materials precisely located in the hohlraum to achieve uniform implosion of the fuel pellet. On the other hand, direct-drive fuel capsules likely require more stringent smoothness, and must survive the trajectory to the center of the chamber.

For direct drive targets the beam energy must be delivered in a time close to the implosion time, while for indirect drive targets, the beam energy is deposited in a time longer than the capsule implosion time. Some of the challenges in NIF experiments are in the implosion stage. Possible reasons include asymmetry in the drive pulses, interactions between the laser and the created plasma in the target when individual lasers cross (laser-plasma interaction), and asymmetries in the target-hohlraum geometry. Not all of these (e.g., laser-plasma interactions) are germane to HIF targets.

The most commonly considered ignition mode is fast ignition, shock ignition and hot-spot ignition. In fast ignition, a separate beam pulse from the implosion beams ignites a fraction of the target. In shock ignition, a high-power beam pulse creates a shock wave to help ignite the fuel. In hot spot ignition, the compression beams themselves heat and ignite part of the compressed fuel. Indirect drive, hot-spot ignition is the approach most explored at the NIF.

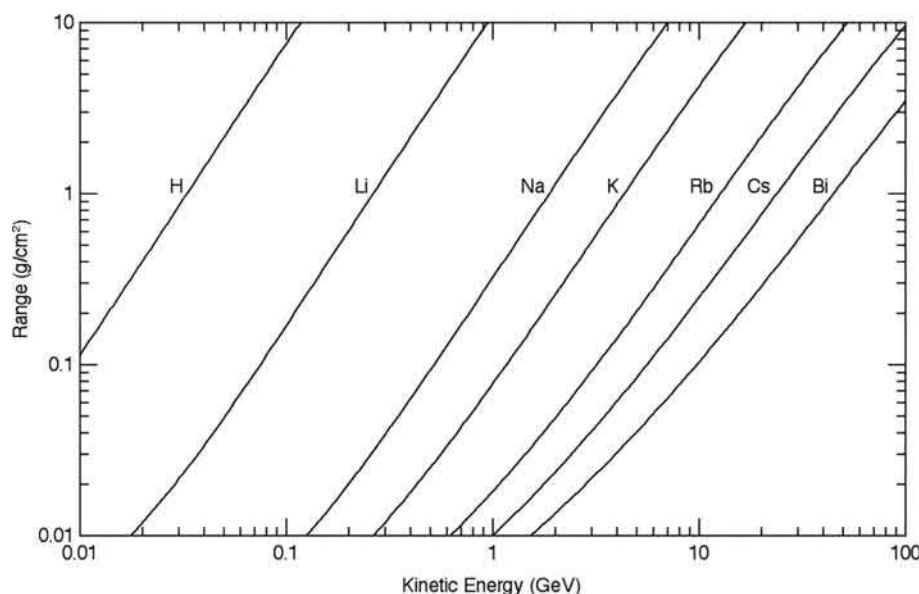


Fig. 1 Range as a function of kinetic energy in aluminum (at a temperature of 300 eV) for a variety of ions. From Bangerter RO, Faltens A and Seidl PA (2013) *Accelerators for Inertial Fusion Energy Production, Reviews of Accelerator Science and Technology*. vol. 6, pp. 85–116, World Scientific Publishing Company. <https://doi.org/10.1142/S1793626813300053>.

The choice of implosion and ignition modes leads to various trade-offs between target gain (ratio of the fusion energy yield to total beam energy), required beam energy, power, and focal spot radius. Detailed simulations of the target physics for particular designs have been carried out. In Bangerter et al. (2013), several designs are summarized, varying in beam energy between 3.5 MJ and 6.7 MJ, peak beam power from 470 TW to 30,000 TW, focal spot radius from 0.26 mm to 4.5 mm, ion kinetic energy from 3.5 GeV to 20 GeV and fusion gain from 60 to 300. These beam requirements span a wide range in fusion gain with implications for the accelerator requirements. The most demanding beam requirements are for fast ignition, and the more relaxed requirements are for hot spot ignition. To decide among these target design and accelerator combinations, an encompassing evaluation of the engineering challenges, reliability and economics of the whole system is required: accelerator, fusion reactor chamber, target fabrication and balance of plant. These system studies are discussed in the accompanying chapter by Dunne (2020).

Target requirements establish the ion beam and accelerator specifications

To illustrate how the target design is coupled to the accelerator requirements, we use a specific example, the close-coupled target (Callahan-Miller and Tabak, 2000), summarized in Table 1. This target design is attractive because it has a relatively high fusion gain and low beam energy for an indirect drive target and, relative to some other targets, moderate peak power requirements. The focal radius of 1.7 mm is somewhat demanding, putting it in the realm of requirements for some direct drive target designs.

We first examine two effects which limit the minimum focal radius. (Limitations due to chromatic aberrations in the lenses will be discussed later.) One is the electrostatic repulsion among the ions, or the space charge effect. The other is the spread of ion coordinates in position and momentum space transverse to the beam direction, which may be characterized by the beam phase space density, brightness, and emittance (Reiser, 2008).

Unnormalized brightness and emittance are related via the beam current, I :

$$B = I / (\epsilon_x \epsilon_y \epsilon_z) \quad (1)$$

Where ϵ_x and ϵ_y are the emittance in the two directions orthogonal to the direction of propagation (z). From Liouville's theorem we know that for a system under conservative forces, the six-dimensional phase space density is conserved. The six dimensional phase space density is proportional to the normalized brightness $B_n = B / (\gamma\beta)^3$, where γ and β are the usual relativistic factors. This means that in an accelerator with ideal focusing systems the density is conserved, while imperfections in the accelerator components (e.g., focusing lens aberrations) lead to an increase in emittance (by engulfing empty phase space) and sometimes particle loss. As we will see, both emittance and space charge place important performance constraints on the accelerator system.

From Table 1 we deduce that the number of ions per pulse is $3.5 \times 10^6 \text{ J} / 3.5 \times 10^9 \text{ eV} = 1 \text{ mC}/e = 6 \times 10^{15}$, and the peak current is $I_{\text{peak}} = 134 \text{ kA}$. These are unusually high values for any ion accelerator. These demanding values must be achieved at the target, while for much of the accelerator before the final beam compression (power amplification) and focusing, the radius and beam power may be relaxed.

To further discuss the roles of the beam space charge and brightness, we will separate the accelerator into three main parts: The ion source, the main accelerator, and the final focusing system. Ions are created in the ion source at low kinetic energy (0.1–2 MeV). In the main accelerator the ions are accelerated to the final kinetic energy and then compressed to a higher density, principally in the longitudinal direction, by compressing the bunches or “stacking” a series of bunches in, for example, an accumulator ring. In the final focusing system the beam is focused radially to meet the desired spot size at the target.

Because the driver must accelerate ions to the GeV range and the maximum acceleration gradient is 1–10 MeV/m for most of the accelerator (with the upper range being speculative at this point), most driver designs call for accelerators at least a kilometer in length. Since the accelerator is a narrow and long structure, the area occupied by the driver is still competitive with large baseload electrical power plants. Note that there can be a trade-off between beam current and ion energy to meet a fusion target power requirement. Lower kinetic energy implies a shorter accelerator, but more challenging space charge repulsion. Circular accelerators, where the beam passes through a small number of acceleration stations many times, would be attractive by reducing the overall length. They also have special beam physics considerations as we will discuss below.

Ion sources

The need for reproducibility at the fusion target means that the ion source should produce nearly identical pulses as characterized by the initial current, energy, size and convergence or divergence angles. Sources usually extract ions from a spark, plasma or a hot

Table 1 Beam parameters for the close coupled target, an indirect-drive, hot spot ignition target.

Energy (MJ)	Peak power (TW)	Focal radius (mm)	Gain	Ion	Kinetic energy (GeV)
3.5	470	1.7	130	Pb	3.5

From Bangerter RO, Faltens A and Seidl PA (2013) *Accelerators for Inertial Fusion Energy Production, Reviews of Accelerator Science and Technology*. vol. 6, pp. 85–116, World Scientific Publishing Company. <https://doi.org/10.1142/S1793626813300053>.

surface (surface or contact ionization). The ions are then accelerated across one or a handful of high-voltage gaps to 10's of keV to the MeV range (Brown, 2004). It appears to be necessary to operate in the space-charge limited mode, so that the variations in beam current that would otherwise be caused by the fluctuations in the ionization process are eliminated. Thus, the emission current density (or ionization rate) should be greater than the space charge current density by some margin to accommodate emission fluctuations. The Child-Langmuir formula for space charge limited flow in a planar gap (gap distance, d) shows the dependence of the ion current density (J) on the geometry and voltage (V) across the gap.

$$J_{CL} = (4\epsilon_0/9) (2qe/M)^{1/2} V^{3/2} / d^2 \quad (2)$$

where J is the current density, ϵ_0 is the permittivity of free space and M is the ion mass. In order to maximize the current density and the current ($I = J \bullet a^2$, where a is the source beam radius), the source should operate at the highest voltage allowed by high-voltage breakdown considerations while still delivering space-charge limited flow. The formula also shows that the voltage must be very stable to prevent unwanted current variations.

Various effects contribute to the phase space volume occupied by the beam. The ions are created or born with a spread of momenta which may be approximated by a temperature (though the distribution may not be Maxwellian). The associated rms velocity per degree of freedom is $v = (T/M)^{1/2}$ where T and M are the temperature and mass in eV and eV/c², respectively. One can then define the normalized transverse emittance, ϵ_n , and relate it to the rms velocity (for a beam not undergoing expansion):

$$\epsilon_n \equiv a\beta\gamma\theta = 2a(T/M)^{1/2} \quad (3)$$

where a is the beam radius. For most heavy ion fusion drivers the final ion energy is mildly relativistic, so $\gamma \approx 1$. θ is the beam spread or divergence angle. The non-relativistic beam velocity allows for manipulation of the beam current by modulating the velocity of different parts of the pulse. More generally for an expanding beam, $\epsilon_n = \beta\gamma\epsilon_x \equiv 4\beta\gamma(\langle x^2 \rangle \langle x'^2 \rangle - \langle xx' \rangle)^{1/2}$, where the derivatives are with respect to the direction of beam motion (z) and the brackets indicate rms quantities of the ensemble of particles.

The longitudinal phase space volume or emittance is often expressed as the product of the pulse duration (t , in seconds) and the energy spread, ΔE , in eV. If the particles have a longitudinal velocity spread v in the beam frame of reference, then in the laboratory frame the velocity is $v_0 \pm v$ and the energy spread ($2\bullet$ rms) in the laboratory frame is $\Delta E = 2Mv_0v$. If the velocity spread is due to the aforementioned source temperature, T , then the longitudinal emittance, ϵ_L , is:

$$\epsilon_L = (2ET)^{1/2} t \quad (4)$$

One can see from the expressions for ϵ_L and ϵ_n and their dependence on T and E that, in addition to the ion temperature at the point the ions are formed, rapid fluctuations in the extraction voltage during the pulse will also contribute to the emittance. Another source of emittance increase is imperfection in the ion extraction and acceleration fields.

The maximum high voltage against breakdown in the vacuum system varies as $d^{1/2}$ for $d > 1$ cm and linearly with d for $d < 1$ cm (Kwan, 2005), and for extraction optics with low aberrations $a \leq d$ should be satisfied. It follows that the six-dimensional phase space density, or beam brightness,

$$B = 1/(\epsilon_L \epsilon_n^2) \quad (5)$$

increases with decreasing a or d . Thus, small radius beams with short extraction gaps are brighter, but will also have lower current, which must be reconciled with the current requirement set by the target design and final ion kinetic energy. The lower current can be compensated by configuring the source with many smaller, higher-brightness beams which are very closely spaced, and merged into a single beam after extraction. The empty space between beams means that the average brightness of the merged beam will be lowered somewhat than that of the individual beams, but is still attractive.

It is tempting to design sources to create higher charge state ions ($q > 1$) because the ions can be brought to the final kinetic energy in the accelerator in a shorter distance. For example, a factor of two, for $q = 2$ vs. $q = 1$ could be attractive economically. Note that the current density (Eq. 2) increases as $q^{1/2}$, while the particle current (or number density) of the beam decreases inversely with q for a given current, thus decreasing brightness. Since space charge is especially challenging at low kinetic energy, an alternative is to start with $q = 1$ ions and strip the ions to a higher charge state at an intermediate kinetic energy. One difficult trade-off is that the stripping process leads to a distribution of charge states and a lower brightness, since only one charge state is utilized. To date, the brightest sources for HIF tend to be $q = 1$ plasma sources or aluminosilicate (surface ionization) sources.

For induction accelerator drivers, several candidate sources for HIF are summarized by Kwan (2005). Surface ionization sources have generated MeV and ampere-level beams (Bieniossek et al., 2005) as illustrated in Fig. 2. Beams formed from the merging of many, small radius beams with a total current of ≈ 0.5 A are also promising, based on experiments and simulations. The current/beam for RF drivers is usually lower, and suitable high brightness beams with 40 mA current have been demonstrated (Hofmann and Plass, 1998; Ratzinger, 2001). The CHORDIS ion source is filament driven with a small plasma chamber and a permanent magnet array to maintain a high plasma density.

The goal of developing laser-driven ion beams with high efficiency and low energy spread has been advancing in the past decade. For example, intense lasers (few ns, $\sim 10^9$ W/cm²) produce ions in an expanding plasma plume after striking a solid target. Extraction of these ions into a longer, low emittance bunch has been proposed for HIF ion sources and injectors (Okamura, 2018), where the production of low charge state ions has been demonstrated. Remaining challenges include the time variation of the beam during the 20–50 μ s extracted pulse and clearing unwanted charge states.

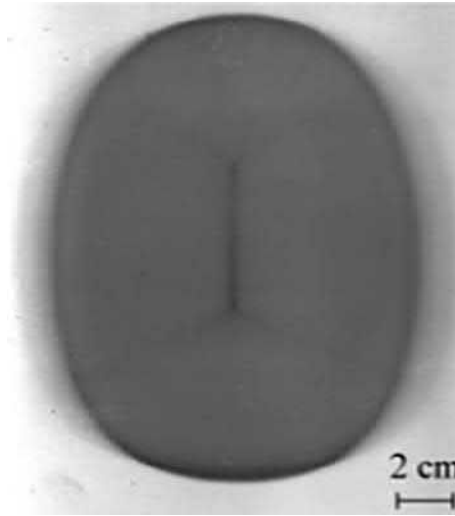


Fig. 2 Beam footprint from a prototype ion source and injector for heavy ion fusion (Bieniosek, et al., 2005). At 2 MeV, the injector generates a 5 μ s, 0.8 A pulse. From Bieniosek FM, Celata CM, Henestroza E, Kwan JW, Prost L et al. (2005) 2-MV electrostatic quadrupole injector for heavy-ion fusion. *Physical Review Accelerators and Beams* 8: 010101. <https://doi.org/10.1103/PhysRevSTAB.8.010101>.

Main accelerator

High energy accelerators generate beams of charged particles by acceleration due to electric fields, while the beams are focused with magnetic or electric lenses. Magnetic quadrupole lenses, arranged in an alternating gradient pattern are particularly common in particle accelerators. The fields in the horizontal plane switch from focusing to defocusing in each successive quadrupole. The focusing is reversed in the vertical direction. This pattern repeats with a period $2L$, where L is the distance between successive quadrupoles. The acceleration fields are typically generated between electrodes in a gap between quadrupoles. These accelerating gaps are realized with either RF cavities (RF accelerators), or from high-voltage pulses (e.g., induction accelerators). We will first discuss the focusing of the beam in the accelerator.

Ion-driven fast ignition requires shorter pulses of high brightness beams, with a pulse energy of ≈ 100 kJ to a few MJ (Henestroza and Logan, 2012) and meeting the stringent requirements of rapid and highly localized target heating is a more difficult challenge due to chromatic and other aberrations (Bangerter et al., 2013). Laser driven ion beams can produce 10 MeV to 100 MeV ions within a small transverse radius. This method has been studied for the heating of matter to high temperatures for HEDP and for ion-driven fast ignition (Fernandez, et al., 2014), where the laser pulse would create the ion pulse very near the target without the need for post acceleration. In this case recent results have shown an improved overall efficiency and reduced energy spread. The lasers themselves can be about 20% efficient (wall plug to photons), with efficiencies up to 50% envisioned (Hooker, 2013). An efficiency of energy transfer from the laser pulse to the ions of 5% has been demonstrated (Palaniyappan et al., 2015). Higher brightness and efficiency are needed and might come into reach with laser technology advances and demonstrations of advanced laser-ion acceleration mechanisms (Park et al., 2019).

Space charge, transverse emittance and transportable current

Though space charge is usually the limiting factor on transportable current in the main accelerator, the emittance must also be kept low enough to meet the focusing requirements at the target. This is non-trivial; it has to stay low over many focusing elements in an accelerator where the beam path length may be several kilometers and where deleterious effects will accumulate, lowering brightness. Driver designers routinely stipulate that the brightness at the ion sources should be greater than that required at the target by a substantial margin to accommodate these real-world effects.

The contributions from the transverse emittance and space charge can be seen with the beam envelope equation, which describes the evolution of the transverse shape of the beam through the focusing lenses (quadrupoles):

$$\frac{d^2a}{ds^2} = \pm k_x^2 a + \frac{2Q}{a+b} + \frac{\epsilon^2}{a^3} \quad (6)$$

The equilibrium beam shape is elliptical, where a and b are the semi-minor and semi-major beam radii. k_x^2 contains the focusing effect from the quadrupoles—its sign depends on whether the quadrupole is focusing or defocusing along the a axis. A similar equation describes motion in the orthogonal (b) direction. The perveance is $Q = I_a q e / (2\pi\epsilon_0 M c^3 \beta^3 \gamma^3)$, a dimensionless quantity, representing the defocusing of the beam due to its electric and magnetic fields. The transverse emittance, ϵ , is assumed here to be equal in both transverse planes.

The focusing term can be related to an important characteristic of accelerator focusing systems, the phase advance. Individual particles oscillate with characteristic periods in focusing systems with repeating structures. In an alternating gradient quadrupole channel, a single charged particle's phase advances by σ_0 per full lattice period ($2L$) and

$$k_x = \frac{\sigma_0}{2L} \quad (7)$$

σ_0 is referred to as the single-particle phase advance, which characterizes the oscillations in the case of low space charge or current. Higher strength quadrupole magnets correspond to higher σ_0 . Due to the onset of instabilities there is a limit on the phase advance at and above $\pi/2$, and practical accelerator designs tend to keep the phase advance to $\leq 85^\circ$.

Increasing perveance (beam current or space charge) depresses the particle phase advance to lower values. Since unusually high current beams are needed for HIF, an important early question concerned the lower limiting value of the ratio σ/σ_0 , or tune depression, that could be realized without beam instabilities or other problems. This was answered by experiments with simulations of high perveance and low kinetic energy ion beams which showed that $\sigma/\sigma_0 \approx 0.1$ is possible for $\sigma_0 = 85^\circ$ (Tiefenback and Keefe, 1985; Tiefenback, 1986). In a later experiment near driver scale (1 MeV, 0.2 A), a K^+ beam injected into a quadrupole channel showed no emittance growth while $\sigma/\sigma_0 < 0.2$ (Prost et al., 2005).

The space charge repulsion of the beam must also (in addition to the defocusing due to emittance) be balanced by the electromagnetic focusing elements (electromagnets, e.g., quadrupoles, or electrostatic equivalents). For a fixed focusing strength, a beam with higher space charge repulsion (higher beam current) will have a larger equilibrium radius and, with further increase of space charge, some particles will graze the beam pipe and be lost.

The fundamental limit on the transportable beam current in an accelerator focusing system is referred to as the Maschke limit (Maschke, 1979). Maschke proposed to set the maximum transportable current such that the space charge forces were one-half the average focusing forces. Here we give the maximum transportable current for a series of magnetic quadrupoles. Assuming that the quadrupoles occupy one-half of the beamline, respecting the known limits of beam stability, and for an average beam radius one-half the bore radius ($a/b = 0.5$), the maximum transportable current is (Bangerter et al., 2013):

$$I < 5 \times 10^5 * B_b * \beta^2 \gamma^2 a \quad (\text{Amperes}) \quad (8)$$

where B_b is the magnetic field in Tesla at a distance b (in meters) from the axis (the beam pipe radius). This expression is approximately the same as the Maschke limit. Similar expressions may be derived for solenoid lenses or electrostatic quadrupoles. $B_b = 5$ T is representational of a modern superconducting quadrupole. For the beam parameters in Table 1, and taking as an example $a = 0.03$ m, the Maschke limit is 2.8 kA, small compared to the total beam current of 134 kA. This is a practical reason to distribute the total current into many lower current beams. Eq. (8) shows that for a practical limit on the peak magnetic field, the transportable current increases with the particle velocity in magnetic quadrupoles.

Superconducting coils (vs. normal conducting coils) are essential for overall electrical efficiency of the driver. The technology is advancing, with numerous research groups pursuing the development of higher focusing strength to match the strong interest in higher kinetic energy and higher intensity accelerators for high-energy and nuclear physics research. This kind of synergistic research is of high value. Earlier generations of superconducting quadrupoles used NbTi conductors, but have been superseded by higher current density Nb₃Sn conductors (Schoerling and Zlobin, 2019). Pursuit of higher fields from high-temperature superconductors such as BSSCO (Bi-221) and rare earth barium copper oxides (REBCO) is becoming more common (Wang et al., 2021).

The loss of particles is of concern in accelerators because of the decrease in beam power, and the possible radioactive activation of accelerator components. For a fixed physical aperture, the fractional loss increases with the increase in the rms beam size. Prost et al. (2005) showed no particle loss while the beam filled 60–80% of the clear aperture. Particle loss is also caused by field imperfections and misalignments distorting the distribution and causing the beam centroid to wobble about the primary axis. The lost particles also cause the desorption of electrons, ions, atoms and molecules from the vacuum chamber wall leading to a transient increase in pressure (Cohen et al., 2005). In circular accelerators, particle loss is a particular concern which can lead to a beam-destroying runaway situation: the pressure can build up with successive laps of the beam bunch around the machine leading to beam-gas collisions and further beam loss. This has been addressed in experiments and will be discussed below.

Radiofrequency accelerators

Radiofrequency (RF) accelerators, invented almost a century ago, are a mature technology with a large international research community continuing to push the performance to higher energy and beam intensity. RF accelerators add energy to particles through a series of voltage steps where the fields in the acceleration gap are driven by an RF source (e.g., a klystron). The driven load is often a high-impedance and high-Q resonant cavity connected to the acceleration electrodes and therefore high fields can be established in the acceleration gap. In the case of cyclic accelerators there are no more than a few acceleration gaps in the ring and very high kinetic energy can be achieved by many circulations of the charged particle bunch. In an RF linear accelerator there are typically many gaps to attain the desired ion kinetic energy, and particles pass through each gap only once. Since many gaps in the accelerator are driven by a common RF frequency, the charged particles must arrive during the accelerating part of the wave at nearly the same phase. The particles tend to reside in a bunch around this synchronous phase. The phase window in which the particles reside and are stably accelerated through many acceleration gaps is referred to as the RF bucket. Thus, the spacing between successive gaps must be $n\beta\lambda/2$, where n is an odd integer and λ is the wavelength of the RF field. There is a tendency for longitudinal spreading (defocusing) of

the bunch due to space charge and emittance. This is counteracted by locating the bunch on the negative slope of the RF wave. In the transverse plane the beam is confined by magnetic quadrupoles between the acceleration gaps.

Since the acceleration gaps in a cyclic accelerator are traversed many times by each particle, the acceleration volts are relatively inexpensive compared to a linear accelerator. Synchrotrons are a type of cyclic RF accelerator where the guiding magnetic field is varied to match the increasing momentum of the accelerated particles. They have generated some of the highest kinetic energy ions and electrons and are the basis for state-of-the-art facilities for high-energy physics research. The Large Hadron Collider at CERN is an iconic example (Llewellyn Smith, 2015). Acceleration of ions with synchrotrons followed by the accumulation of charge in storage rings (a particular type of synchrotron where the kinetic energy is kept constant) were among the first HIF driver concepts in the 1970s and 1980s. There are some important hurdles with synchrotrons for HIF. More detailed and sophisticated target designs showed that the desired ion range was lower than originally anticipated, corresponding to lower ion kinetic energy (and higher beam current, which is more challenging in cyclic accelerators). Furthermore, there is great sensitivity in a synchrotron to collisions among the ions and with the background gas molecules, since the beam must circulate repeatedly while stray ions striking the accelerator wall desorb neutral particles. Recently this has become an urgent problem to be solved for the Facility for Antiproton and Ion Research (FAIR), a high-intensity heavy-ion synchrotron in Germany. Experiments and simulations are guiding the development techniques to trap desorbed particles within the accelerator. These are promising developments for storage rings in HIF drivers (Spiller and Barth, 2014).

Following the earlier synchrotron approach, RF linear accelerators in combination (usually) with storage rings have become the preferred design choice for the RF approach. An important advantage of the linear accelerator (LINAC) is that it allows for higher current per beam bunch compared to a synchrotron. Also, the preservation of brightness is much less sensitive to collisions with background gas molecules, since it is a single-pass structure with ample time between driver pulses (~ 100 ms). Since the spacing between RF acceleration stations increases in proportion to the particle velocity, with increasing velocity the spacing becomes sparse and the average acceleration rate per unit length decreases. Therefore, it is advantageous to occasionally double the RF frequency and to then accelerate with the more densely spaced bunches. This can be accomplished with parallel RF accelerators upstream (at the lower frequency). Beams from pairs of accelerators operating at the lower frequency fill stable phase intervals downstream of the transition, thereby occupying each period at the higher frequency. This funneling of bunches into a single beamline was experimentally demonstrated by Zimmermann et al. (2001) by deflecting beams from two parallel accelerators into a single beamline (Fig. 3).

Most of the acceleration occurs after funneling, in a single accelerator at a fixed RF frequency (~ 400 MHz). The peak current in the LINAC is on the order of 1 A, and is constant through the main accelerator, with an electrical (wall plug to beam) efficiency of about 25%. Once the ions reach the top kinetic energy the bunches are injected into a storage ring for current amplification. There the number of ions is accumulated to meet the fusion target specifications and extracted into a structure which compresses the beam longitudinally before focusing onto the target. Fig. 4 shows an RF driver layout from the European HIDIF (Heavy Ion Driven Inertial Fusion) study (Hofmann and Plass, 1998).

Induction accelerators

Induction linear accelerators were invented well after RF accelerators by Christofilos et al. (1963) to generate very high current pulses of energetic electrons for a fusion energy experiment. Since the kiloampere-level beam current exceeded what was (and is) possible with RF accelerators, it became the main approach developed for HIF in the United States. The acceleration field is generated by pulsed power to each acceleration station and, unlike the RF system described above, the modulator drives the beam directly. The pulsed power drives a low-impedance circuit comprised of cylindrically symmetric toroidal cells loaded with ferromagnetic material. An electric field appears across the gap in the bore of each cell, accelerating ions, and the process is repeated until the final kinetic energy is achieved. A toroidal hoop of ferromagnetic material limits the needed drive current over the pulse duration until

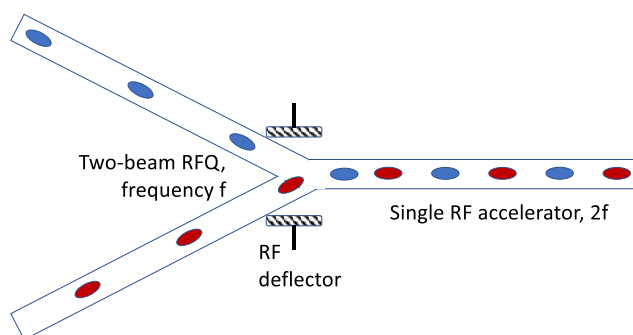


Fig. 3 Schematic of the RF funneling principle. Beams from pairs of accelerators operating at the lower frequency (f) fill stable phase intervals downstream of the transition, thereby occupying each period at the higher operating frequency ($2f$). Experiments showed the funneling of two beams into a single RF accelerator. Zimmermann H, Bartz U, Fieck D, Fischer P, Vossberg M et al. (2005) The frankfurt funneling experiment. In: *Proceedings of the 2005 Particle Accelerator Conference, Knoxville, TN, USA*. pp. 677–679, <https://doi.org/10.1109/PAC.2005.1590525>.

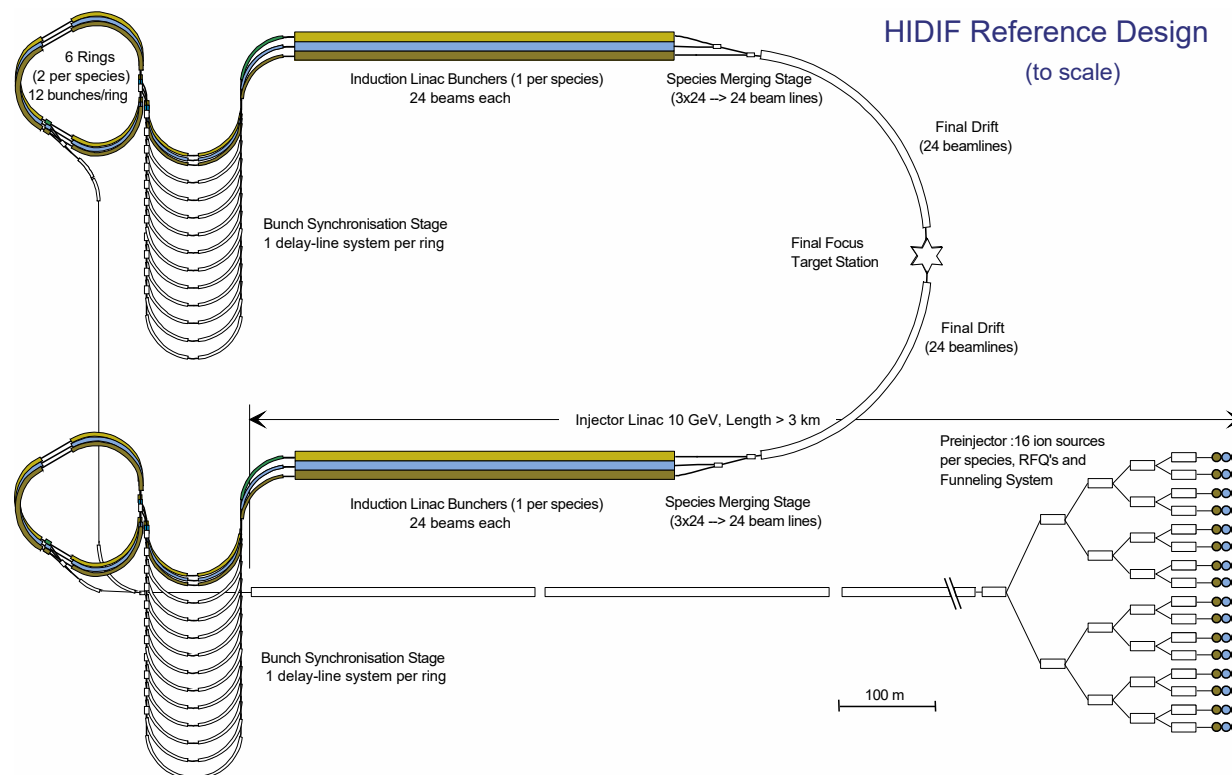


Fig. 4 The RF-accelerator based HIDIF driver design begins with 48 ion sources and low-frequency RF sections funneling into a main LINAC. At the end of the accelerator the beam is fed to multiple storage rings followed by an induction LINAC buncher. Hofmann I and Plass G (1998) The HIDIF Study: Report of the European Study Group on Heavy Ion Driven Inertial Fusion for the Period 1995–1998. GSI Report 98-06; <http://www-alt.gsi.de/documents/DOC-2009-Feb-44.html>

the saturation flux density is reached. Electrical efficiency of 50% (wall plug to beam power) has been demonstrated in high-current electron accelerators (Takayama and Briggs, 2011). The induction cell has low impedance and requires a very high (kA range) drive current to achieve the desired acceleration fields.

The voltage waveform may be customized at each station to accelerate and compress the beam pulse. This allows for current amplification during acceleration without the need for the additional beam manipulation of injection into a storage ring for current amplification. Thus, the ability to transport greater current for a practical limit on the magnetic field, B , (Eq. 8, a consequence of the Lorentz force) allows for current amplification to keep pace with the transportable current in the quadrupoles. Between successive acceleration cells, quadrupoles keep the beam transversely confined against the defocusing from space charge and emittance. The timing of the pulsed power is adjusted to match the arrival time of the beam, and so the placement of successive cells is chosen to keep the spacing between cells compact while the kinetic energy increases. The high beam current creates a longitudinal self-field which is greatest at the ends of the bunch, pushing particles away from the center of mass of the moving bunch. This is counteracted with specialized induction cells creating “ear” or “barrier” pulses to keep the pulse contained. The ear pulses counteract the longitudinal electrostatic field of the bunch, predominantly at the ends of the pulses.

Inspired by the potential cost savings of circular accelerators, a driver design based on a recirculating induction accelerator was developed, and a scaled experiment explored some of the beam physics issues (Ahle et al., 2001). Though the ring was only partially completed, the experiment demonstrated coordinated bending and acceleration of a space-charge dominated heavy-ion beam. Using a low energy beam of electrons, similar physics was explored at the University of Maryland Electron Ring, with induction acceleration cells, quadrupoles and dipoles. High brightness beams with 1000 turns of circulation were achieved (Kishek et al., 2014). A related experiment at a high-energy physics lab in Japan accelerated protons from 500 MeV to 6 GeV with repeated induction acceleration steps of 6.4 kV per turn in a ring with a circumference of 340 m. To counter the tendency of the beam to spread longitudinally, the beam was confined with barrier bucket pulses with a dedicated induction cell, where the barrier pulses applied a repulsive potential barrier to particles near the ends of the bunch.

Multiple, parallel beams in the accelerator

The high-current capability of induction accelerators led Maschke (1979) to suggest the acceleration of many closely spaced, parallel beams of ions with arrays of quadrupoles between acceleration cells. All the beams pass through common induction cells, a distinct

feature of induction linear accelerators for HIF compared to most induction accelerators (Fig. 5). The choice of multiple beams in the induction accelerator is driven in part by the need to focus to a small radius at the fusion target. As will be discussed later, focusing at the target is less challenging for lower current per beam. There is also a cost advantage: Consider a total current I_t distributed equally among N beams, with the quadrupoles nested compactly in a four-fold symmetric array. Averaged over the transverse dimensions of a multiple beam array within the bore of the induction cell, the array is more compact and leads to lower cost than a single beam of current I_t . This arises due to practical limits on the fields which can be generated in superconducting quadrupoles and on the focusing force required to counteract the dominant space charge term (Eq. 6). Arrays of tens to hundreds of beams are usually proposed. For greater values of N , the quadrupole size decreases, the fraction of space occupied by conductor, insulation and support structure leaves less space for the beams and the average beam current density of the N beams eventually drops. Also, practical alignment tolerances become more important as the channel size decreases. Motivated by the multi-beam approach, compact RF accelerator structures (mm-scale) which accelerate many parallel beams (Vinayakumar, et al., 2019) are in early development.

Induction accelerator designs usually start with initial beam pulses from the injector in the 1–30 μs range from many identical ion sources. Each beam is injected into individual channels of a multiple beam accelerator. The bunch compression continues until the bunch duration is 0.1–0.25 μs . This compression leads to a high beam current, where the electrical efficiency of the pulsed power and induction cells is high. The acceleration proceeds at a constant bunch duration until the final kinetic energy.

A disadvantage of induction accelerators is that the technology is less developed than RF accelerator technology. The main advantage of linear induction accelerators is in preserving beam brightness, since the beams keep their identity through the accelerator and to the target, and there are fewer beam manipulations than in RF designs. The induction accelerator approach would benefit from advances in lower-cost, long-life pulsed power, higher-gradient insulators, and robotic manufacturing of superconducting quadrupole arrays (Seidl et al., 2013).

Cost is a critical issue for the feasibility of fusion energy. Not surprisingly, the main accelerator is a major cost center of the overall driver. For RF designs the focusing quadrupoles, RF cavities, and RF power are important (Hofmann, 2018). The required peak power, from a few to 100 GW, is generated by gridded tubes or klystrons. For the highest peak power driver designs (fast ignition), the capital cost of the RF power is particularly challenging (Bangerter et al., 2013).

Superconducting RF cavities would lower the operating cost of the driver. For example, the Spallation Neutron Source accelerates 1.4 mA (average current) proton pulses with a 6% duty cycle, and 23 mA within the macro-pulses (Kim et al., 2017). Higher intensities are planned for future neutron sources (Garoby et al., 2018).

Beam compression

Compressing the beam in the direction of motion is essential because the beam current from the best ion sources is much less than that required by the target. RF LINACs are constant current devices, while induction LINACs can have varying current along the accelerator, leading to different beam compression approaches.

In the RF case the beam current remains constant from the source to the top kinetic energy of several GeV. Current amplification happens subsequently, as shown by the HIDIF example (Fig. 4). By using multiple ion species with the same momentum but slightly different velocity the beam compresses. The species are generated in separate ion sources and funneled into a common RF accelerator, as described above. The lighter ions have a higher velocity and catch up to the heavier species as the ions approach the fusion target, resulting in a “telescoping” effect, amplifying the beam current. Modeling, supported by the established energy acceptance of RF accelerators, suggest that three (Hofmann, 2018) species with a mass difference of a few percent accomplishes

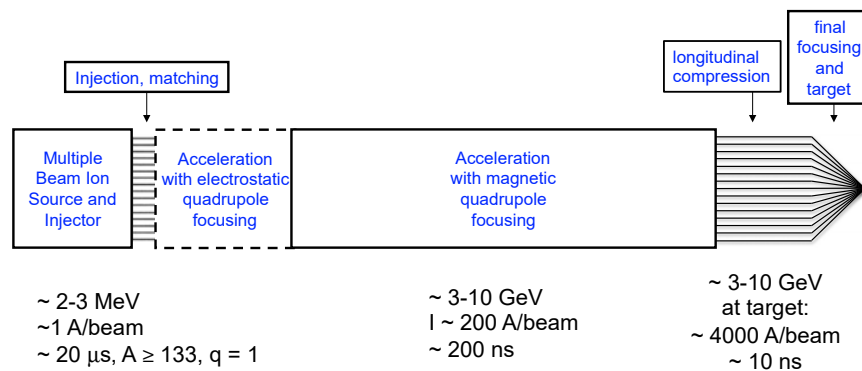


Fig. 5 Schematic of an induction accelerator driver begins with multiple ion sources injected into separate focusing beamlines. All the beams are accelerated by common induction cells. After acceleration to the top kinetic energy the beams are compressed longitudinally. The last beam manipulation is the final focusing onto the fusion target. Bangerter RO, Faltens A and Seidl PA (2013) *Accelerators for Inertial Fusion Energy Production, Reviews of Accelerator Science and Technology*. vol. 6, pp. 85–116, World Scientific Publishing Company. <https://doi.org/10.1142/S1793626813300053>.

the goal. The telescoping of different species in a common RF accelerator has yet to be demonstrated. The species eventually overlap near the target, this leads to some issues yet to be tested by simulations or experiment—for example, that of the mutual space charge repulsion between bunches and whether associated brightness dilution is within limits.

Since the beam brightness decreases due to injection of the beam into, and extraction of the beam from, the storage rings, there has been some effort to design around this step. [Burke \(2014\)](#) proposed a driver with higher peak beam current in the main accelerator while telescoping six ion species (no storage rings, so “single-pass”). The design is based on a fast-ignition fusion target, which is part of the reason for higher peak current.

In the induction accelerator, the current increases during the acceleration by bunch compression. This is accomplished by applying acceleration waveforms with a small positive slope. Within the center of mass of the moving bunch, particles move inward and the bunch shortens, or compresses. As a result, the bunch current increases gradually as kinetic energy (E), increases, while staying below the transport limits in the quadrupoles.

Once ions have reached the final kinetic energy, the beam enters a drift section (only focusing quadrupoles) and the beam compresses further to the duration and pulse shape required at the target. This is accomplished via compression waveforms designed to compress the beam from ~ 100 ns to < 1 ns (fast ignition) to ~ 10 ns (indirect drive) as required by the particular target design. This penultimate beam manipulation is often referred to as drift compression. It is followed by the final focusing of the beam with lenses near the entrance of the target chamber. In drift compression, the minimum bunch duration is limited by the space-charge repulsion of the beam in the longitudinal direction. At this point, electrostatic repulsion has stagnated the inward longitudinal motion induced by the compression waveforms.

Experimental support for the simultaneous acceleration and compression of the beam comes from MBE-4 ([Fessenden, 1989; Fawley et al., 1997](#)). Four parallel space-charge-dominated beams were accelerated and compressed for a nine-fold increase in current in an induction accelerator using custom waveforms with 24 induction cells. Though emittance growth was observed during the compression, the causes of (and corrections for) the decrease in brightness were identified with detailed multi-particle simulations.

To compress beams having space charge forces too high to meet the target requirements, it is possible to mostly eliminate the repulsion of the beam by passing it through a plasma. The electron distribution of the plasma responds to the ion beam and effectively cancels out the space charge forces of the beam. The compression proceeds to the minimum bunch duration limited by the longitudinal emittance of the beam. Ion energy variation from the compression is still present at the target, contributing to greater chromatic aberrations in the final focusing to the fusion pellet.

Neutralized drift compression and focusing has been experimentally demonstrated in experiments NDCX-I ([Coleman et al., 2007](#)) and NDCX-II. In these studies, a plasma ignited near the surface of the drift tube filled the tube to a number density somewhat higher than the peak ion beam density, providing sufficient electrons to neutralize the space charge forces. The bunch duration and focal spot sizes after the final lenses were in good agreement with multi-particle beam and plasma simulations.

NDCX-II is a new ion induction accelerator for the study of rapid heating and radiation effects in materials from high-brightness ion beams ([Barnard et al., 2014](#)). Starting with ~ 1 μ s duration injected pulse, it simultaneously accelerates and compresses the beam to a few ns duration and mm-radius at the target, including neutralized drift compression and focusing ([Fig. 6](#)). The results confirm predictions from multi-particle simulations ([Seidl et al., 2017](#)).

Final focusing

The ion beam is accelerated with a much larger radius than the target focal radius. The beam is then compressed longitudinally, and then the final focusing lenses near the reactor chamber impart a convergence angle to the beam. The final focusing lenses have a focal length slightly greater than the standoff ($f \approx 10$ m) between the target and the reactor wall. The minimum focal spot radius will be limited by space charge and the emittance. To see how space charge limits the focal spot, briefly ignore the emittance term and integrate the envelope equation (Eq. 6). For a convergence angle θ_o , initial beam radius r_o , and radius at the target r_t :

$$\theta_o = (2Q \ln(r_t/r_o))^{1/2}$$

This can be rewritten in terms of a current limit using the definition for the perveance, and the degree to which the beam is neutralized by a factor $(1 - f_n)$, representing the difference between the density of the beam ions and the plasma electrons:

$$I_e < 8 \times 10^6 \left(\frac{A \beta^3 \gamma^3 \theta_o^2}{f_n q \ln(r_t/r_o)} \right) [\text{Amperes}],$$

where A is the ion mass in amu. If the degree of neutralization is high ($f_n \rightarrow 0$) then the spot will be limited by the emittance and chromatic aberrations, and other effects. The above result is important because it shows the effect of mass, ion charge state and ion velocity on the current per beam on target.

The target sets a lower limit on the brightness or six-dimensional phase space density. The limit on transverse momentum spread (part of the transverse emittance) is $r_t p / f$. This is the fraction of the beam momentum allowed in the transverse plane for a required r_t . In the transverse direction, the phase space volume is constrained by the fraction of the total solid angle available for focusing at the reactor wall. At the wall of the reactor chamber the transverse spatial area of the phase space volume is Ωf^2 , where Ω is the solid

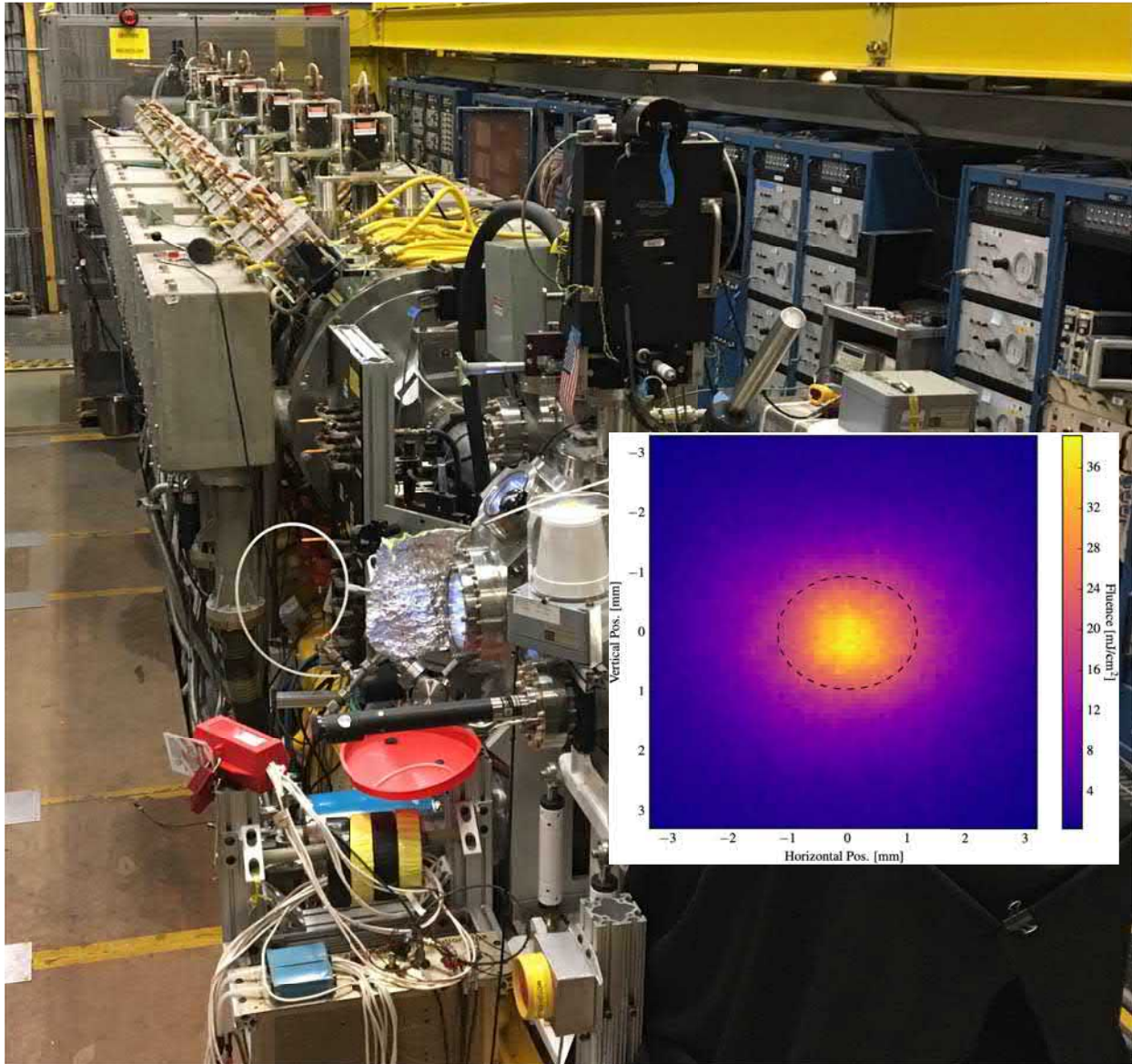


Fig. 6 NDCX-II (LBNL) compresses and focuses ions in an induction accelerator to nanosecond bunches and millimeter focal spots. It uses many of the beam manipulations needed in an induction accelerator driver for fusion. The inset shows a 2 A beam focused to 2-mm FWHM.

angle occupied by all the beams. As discussed by [Dunne \(2021\)](#), in order to allow space for breeding tritium and for collecting the energy released by the fusion reactions, Ω is necessarily much less than 4π . Indirect drive targets with cylindrical symmetry typically call for two-sided illumination and the solid angle of the two groups of beams must be ≤ 0.5 . One must consider the packing of beams at the wall, and clearance required to accommodate beam halo and alignment tolerances. This clearance requirement reduces the solid angle occupied by all the beams to ≈ 0.1 .

The longitudinal phase space components are the energy spread and pulse duration. Usually the most important issue driven by energy spread is chromatic aberrations in the final focusing lenses, leading to a broadening of the focal spot compared to that of a monochromatic beam. For most driver designs these focusing elements are quadrupoles and the increase in the focal radius due to the chromatic effect ([Bangerter et al., 2013](#)) is approximately $\xi = \alpha \delta f \theta_0$, where δ is the fractional momentum spread of the beam, and α is a parameter dependent on the design of the focusing system. Thus, targets that require small focal spots (and small ξ) have stricter requirements on the momentum spread.

Several significant experiments have explored the issues above and have validated specific final focusing designs. They have been carried out at reduced kinetic energy and beam current. [MacLaren et al. \(2002\)](#) showed that the equations that governed the focusing of a driver beam when scaled down correctly gave the expected focal spot. The focusing design was based on a specific RF driver design for 10 GeV Bi^+ beams at 1.25 kA/beam. Several experiments have explored the neutralization of space charge for transverse focusing and drift compression. [Welch et al. \(2008\)](#) showed ion beam space charge neutralization with a background plasma,

greatly reducing the focal spot size compared to un-neutralized focusing. Simultaneous with the focusing, a significant bunching waveform was applied to the beam, compressing the beam longitudinally by a factor of > 60 in a few meters.

NDCX-II showed simultaneous drift compression and focusing in the presence of a neutralizing plasma. The beam is accelerated by 12 induction acceleration and compression cells, while transversely confined by 29 solenoid lenses before entering the background plasma of the drift compression and final focusing lenses. It is a significant test of the main beam manipulations of an induction linear accelerator driver for HIF (injection, acceleration, compression and focusing).

As implied above, the overall 6D brightness constraint is a minimum, but not sufficient condition. The beam must have the right shape at the target, typically longer in the direction of motion. The beam for an induction LINAC driver is considerably longer than it is wide and initially has a higher temperature in the transverse direction and will transfer energy to the longitudinal direction, up to some fraction of the former. In the case of RF drivers the bunches are shorter, but still greater in length than the transverse dimension. RF buckets contain ion bunches with duration typically < 1 ns, occupying a small fraction of the RF wave at, say, 200 MHz.

Summary and outlook

Nuclear and high energy physics accelerators with total beam energy of ≥ 1 MJ have separately exhibited efficiencies, pulse repetition rates (> 100 Hz), power levels, and durability required for IFE. The two principal approaches to heavy ion drivers are RF accelerators and induction accelerators. RF accelerators are appealing because of the extensive experience in high energy and nuclear physics; and induction accelerators, because of their much higher efficiency and higher particle beam current (≈ 10 kA). While recent results in from the laser-plasma acceleration of ions are encouraging and their application to advancing high energy density science, the application to IFE will require advancements in laser technology for efficient acceleration of high ion current.

The beam brightness and alignment must be such that the beam can be focused onto the target to a radius of a few millimeters from a distance of several meters. Limitations due to space charge, emittance growth, beam-gas, and beam-plasma interactions must be sufficiently controlled throughout the driver. Because of the high charge per bunch, the general approach is to accelerate a longer pulse and then compress it to the short length required at the target. The strategies for doing so differ depending on the main accelerator. Several of the experimental results were performed at reduced energy and current in such a way as to preserve the relative emittance and space charge effects expected in a driver. These experiments addressing main components of the driver are encouraging. **Table 2** summarizes a subset of the experimental results that test the HIF approach, including research not described above.

Table 2 Ion experiments in support of heavy ion fusion.

<i>Institution, Country</i>	<i>Experiment</i>	<i>ion</i>	<i>E [MeV]</i>	<i>I (peak) [A]</i>
GSI, Germany	Halo/desorption and pressure rise mitigation in high intensity storage rings and synchrotrons (Omet et al., 2006).	U^{28+} , U^{73+}	$2.7\text{E}+03$	0.01
GSI, Germany	CHORDIS: low charge state ion source for an RF accelerator driver (Ratzinger, 2001)	Bi^+	$2.8\text{E}-02$	0.07
GSI, Germany	Energy loss of heavy ions in a plasma target at the UNILAC (Hoffmann et al., 1990)	^{40}Ca , ..., ^{238}U	56-333	
GSI, Germany	Energy loss of heavy ions in a solid neon target in the SIS-18 synchrotron (Varentsov et al., 2003)	^{238}U	$4.5\text{E}+04$	1.75×10^4
Goethe Univ., Germany	Funneling of two beams into a single RF beamline (Zimmermann et al., 2001)	He^+	$1.6\text{E}-01$	1.0×10^{-3}
Hiroshima Univ., Japan	Paul Trap: Compact platform to test space-charge phenomena (Ohtsubo et al., 2010)	Ar^+		
KEK, Japan	Digital Synchrotron Accelerator: Barrier bucket capture and acceleration of protons in an induction synchrotron (Takayama et al., 2007)	p	$5\text{E}+03$	$1.30\text{E}-07$
ITEP, Russia	Study of intense heavy ion beams in synchrotrons and accumulator rings (Sharkov et al., 2016)	C^{4+} , C^{6+}	$3.6\text{E}+03$	0.005
BNL, USA	Laser ion sources: Demonstration of low charge state ion generation of heavy ions for heavy ion fusion drivers (Okamura, 2018)	Bi^+	$1.4\text{E}-02$	$9.40\text{E}-03$
LBNL, USA	Neutralized Drift Compression Experiment-II (NDCX-II): Ion beam compression to $> 10^{20}$ ions/cm ² /s (Seidl et al., 2017)	He^+	1.1	2.1
LBNL, USA	NDCX-I: Ion beam compression and focusing with space charge neutralization (Coleman et al., 2007)	K^+	0.3	3

(Continued)

Table 2 Ion experiments in support of heavy ion fusion.—cont'd

<i>Institution, Country</i>	<i>Experiment</i>	<i>ion</i>	<i>E [MeV]</i>	<i>I (peak) [A]</i>
LBNL, USA	Neutralized Transport Experiment: Plasma neutralization and focusing (Roy et al., 2005)	K ⁺	0.3–0.4	0.08
LBNL, USA	2-MV Heavy Ion Injector: Driver-scale beam (Bieniosek et al., 2005)	K ⁺	2.2	0.8
LBNL, USA	High Current Experiment: Driver-scale beam in a quadrupole channel (Prost et al., 2005)	K ⁺	1	0.2
LBNL, USA	Cs ⁺ ion injector: Driver scale beam generation (Abbott et al., 1979)	Cs ⁺	0.3–1.2	1
LBNL, USA	Multiple Beam Experiment (MBE-4): Induction compression (9x) of high space charge beams (Fawley et al., 1997)	Cs ⁺	0.2	3.20E-02
LBNL, USA	Single Beam Transport Experiment (SBTE): Stability limits of an intense, space-charge dominated beam in an 87-quadrupole channel (Tiefenback, 1986)	Cs ⁺	1.2E-01	1.50E-03
LBNL, USA	Merging of four beams into a single quadrupole channel (Seidl et al., 2003)	Cs ⁺	0.16	0.012
LBNL, USA	Multi-beamlet ion source (STS-500): a low emittance, high-current driver-scale beam from > 100 plasma sourced beamlets (Kwan et al., 2006)	Ar ⁺	0.4	0.07
LBNL, USA	Scaled Final Focusing Experiment: Scaled replication of a driver final focusing system (MacLaren et al., 2002)	Cs ⁺	0.16	4.00E-04
LLNL, USA	Circular induction accelerator for space charge dominated beams (Ahle et al., 2001)	K ⁺	0.08	0.002
Univ. Md, USA	University of Maryland Electron Ring (UMER): Ion storage ring with high space charge (Kishek et al., 2014)	e [−]	1E-02	0.1
PPPL, USA	High space charge confinement in a Paul trap, equivalent to > 10 ⁴ quadrupoles (Chung et al., 2009)	Cs ⁺	Low	2.5x10 ⁶ ions
Tokyo Inst. Tech., Japan	Laser ablation plasma ion source with high ion flux (Okamura, 2018)	Ti	3E-03	3.E-03

The opportunity to address remaining issues and to reduce risk in the development of HIF is timely given the progress at the NIF and advances in accelerator technology and computer modeling of intense beams and plasmas. Scaled experiments at heavy ion accelerators (e.g., Schoenberg et al., 2020) and with laser driven ion pulses can support the development of target designs and benchmarking simulation codes. Beam experiments to, for example, tackle remaining beam physics related to compression and focusing are possible with a modest research investment. Any future integrated experiment capable of testing beam physics at full scale, including heating, compressing and igniting targets will require the development of technology to reduce cost. Since particle accelerator research and development continues to be a vibrant field, there will be numerous opportunities to leverage ongoing research.

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Inertial Confinement Fusion - Experimental Physics: Z-Pinch and Magnetized Liner Inertial Fusion

Matthew R. Gomez, Mary Ann Sweeney, David J. Ampleford, Stephen A. Slutz, Gregory A. Rochau, and Daniel B. Sinars, Sandia National Laboratories, Albuquerque, NM, United States

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Glossary

Areal density The integral of the density over a path length traversed by a particle or photon. The areal density of both the fuel and the tamper are of interest in a fusion capsule.

Beam-target fusion In order for fusion to occur, nuclei must overcome the electromagnetic repulsion between them. Beam-target neutron production occurs when nuclei (the beam) are electromagnetically accelerated into target nuclei and fusion reactions occur. This process cannot be energetically favorable, making it undesirable for fusion energy.

Case-to-capsule ratio The ratio of the inner radius of the hohlraum surrounding an inertial confinement fusion capsule to the outer radius of the capsule. The smaller the ratio, the greater the potential for implosion symmetry issues.

Convergence ratio The ratio of the initial target radius to the target radius at stagnation.

DT-equivalent yield The estimated fusion yield that would have been produced by a target if it had been fielded with 50% deuterium and 50% tritium fuel (referred to as DT fuel). The conversion factor from neutron yield produced with pure deuterium to the neutron yield with 50% deuterium and 50% tritium is typically between 50–100 and is dependent on the plasma temperature. The DT-equivalent yield is reported as the total fusion energy released (17.6 MeV per fusion reaction).

Hohlraum A shell whose walls are heated to produce x-rays. Indirect drive fusion uses the uniform soft x-ray emission from the hohlraum walls to drive a fusion capsule implosion.

Laser entrance hole (LEH) window A sub-micrometer to few-micrometer thick membrane that is used to contain gas within a target.

Liner Any conducting shell, typically a cylindrical metal tube, used as the initial configuration of a z-pinch implosion (e.g., as in Magnetized Liner Inertial Fusion targets).

Mix Material other than the fusion fuel, often with an atomic number greater than one, that is introduced into the fuel. This material radiatively cools the fuel at an increased rate, reducing the achievable yield.

PdV work Pressure-driven work done on the fuel through volumetric compression.

Pulsed power driver efficiency The ratio of electrical energy delivered to the target to the energy stored in the capacitor banks of a pulsed power driver.

The classical Z-pinch

The classical z-pinch, sometimes referred to as the Bennett pinch (Bennett, 1934), consists of a steady-state cylindrical plasma column conducting an axial current (see Fig. 1A). The axial current generates an azimuthal magnetic field that exerts a radially inward pressure given by

$$P_{mag} = B^2/2\mu_0 = \mu_0 I^2/8\pi^2 r^2, \quad (1)$$

where B is the azimuthal magnetic field generated by the axial current I flowing through the column, and r is the radius of the plasma column. This magnetic pressure is balanced by a radially outward plasma pressure given by

$$P = nk_B T, \quad (2)$$

where n is the plasma density, k_B is the Boltzmann constant, and T is the plasma temperature. In the classical z-pinch, this pressure balance is established quickly relative to the overall timescale of the pinch, with most time spent in a steady state. The classical z-pinch is more akin to magnetic confinement fusion (MCF) schemes than to inertial confinement fusion (ICF) schemes.

The current flowing through the z-pinch also ohmically heats the plasma. With a large enough current, the energy added through ohmic heating balances the energy lost through radiative losses (Pease, 1957; Braginskii, 1958; Pereira, 1990). For hydrogen, this "Pease-Braginskii" current is 1.4 megaAmperes (MA), independent of the plasma density and pinch radius (Ryutov et al., 2000).

The steady-state z-pinch was one of the first geometries suggested for controlled thermonuclear fusion (Anderson et al., 1958). The aim was to drive the z-pinch to sufficient temperature and density and hold it in that state while fusion energy was produced. Fusion neutron production was observed in this configuration but was the result of **beam-target fusion**. Magnetohydrodynamic instabilities like the "sausage" ($m = 0$) and "kink" ($m = 1$) modes (see Fig. 1B and C) resulted in the plasma column breaking up, generating a large voltage and driving **beam-target fusion** reactions. The realization that such neutron production did not support net-energy production caused classical z-pinch fusion research to dwindle.

The pulsed Z-pinch

Advances in pulsed power technology led to renewed interest in z-pinchs with many studies focused on a configuration known as the pulsed z-pinch (Pereira and Davis, 1988; Ryutov et al., 2000; Sweeney, 2002). The pulsed z-pinch (sometimes called a "fast" or "imploding" z-pinch) also employs an axial current to generate an azimuthal magnetic field and a radially inward force. A key difference between the pulsed z-pinch and the classical z-pinch is that the lifetime of the pulsed z-pinch is short compared to the time to assemble the pinch. In contrast with the classical z-pinch, the pulsed z-pinch is an ICF concept, despite the large magnetic fields

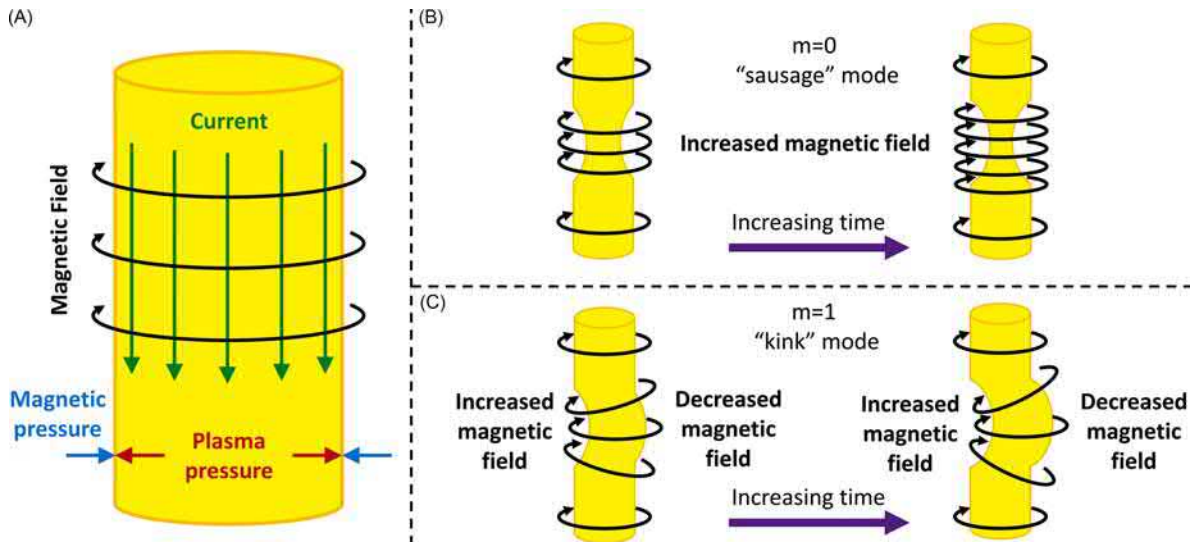


Fig. 1 (A) Cartoon depicting the classical z-pinch geometry. Current flows axially through a cylindrical plasma column and generates an azimuthal magnetic field. The radially inward magnetic pressure is balanced by a radially outward plasma pressure. (B) Cartoon depicting development of the $m = 0$ (sausage) instability. The magnetic field at the plasma surface is inversely proportional to plasma radius, so a small dip in the surface increases the magnetic field. A larger magnetic field applies a greater magnetic pressure locally, decreasing the radius further and creating a positive feedback loop that causes surface perturbations to grow. (C) Cartoon depicting development of the $m = 1$ (kink) instability. If a portion of the plasma has a small offset from the axis, the magnetic field will be greater on the side of the plasma with the dip than on the side with the bulge, resulting in greater magnetic pressure on the side of the dip, which increases the offset and creates a positive feedback loop.

involved in driving the implosion. A subset of pulsed z-pinchs known as magneto-inertial fusion concepts incorporate magnetic fields to retain heat within the fuel during the ICF implosion and stagnation, blurring the lines between ICF and MCF.

In the pulsed z-pinch, typically an annular shell of material is magnetically imploded onto the z-axis, with the implosion duration comparable to the risetime of the driving current pulse. Assuming the radial extent of the annular shell is small compared to the shell radius (i.e., $r \gg \Delta r$), the force acting on the material can be approximated as

$$F_{mag} = -P_{mag}2\pi rh, \quad (3)$$

where r is the outer radius of the material and h is the axial extent of the z-pinch. Substituting Eq. (1) gives

$$F_{mag} = -\frac{\mu_0 I^2}{8\pi^2 r^2} 2\pi rh, \quad (4)$$

which can be simplified to

$$F_{mag} = -\frac{\mu_0 h}{4\pi} \frac{I^2}{r}, \quad (5)$$

indicating that the magnetic force accelerating the material diverges as the material approaches the axis. For a sense of scale, if 20 MA is flowing through a 1-mm-diameter plasma column, the magnetic pressure acting on the column is approximately 250 megabar (Mbar).

When this imploding material reaches the axis, it briefly stagnates, reaching high temperatures and densities for a duration much shorter than the implosion time, and rapidly disassembles. The short stagnation time significantly reduces the importance of the sausage and kink instabilities; however, implosion instabilities, such as the magneto-Rayleigh-Taylor instability (Harris, 1962), take their place. In the magneto-Rayleigh-Taylor instability, much like the classical Rayleigh-Taylor instability (Rayleigh, 1882; Taylor, 1950), when a light fluid (the magnetic field) pushes on a heavy fluid (the plasma), small perturbations in the interface between the two fluids will grow. In a pulsed z-pinch, the outside surface of the plasma is magneto-Rayleigh-Taylor unstable and develops a “bubbles and spikes” pattern during the implosion. This results in a less coherent implosion that impacts the efficiency with which the material can be driven to small radius.

Pulsed power drivers

Pulsed power drivers store electrical energy for long times (seconds to minutes) and discharge that energy over much shorter time-scales (nanoseconds to microseconds) resulting in extremely high electrical powers for short times. For example, the electrical power of the discharge for the world’s largest pulsed power system, the Z machine at Sandia National Laboratories (also referred to as “Z”), can be greater than 80 terawatts (Savage et al., 2011). For a sense of scale, this is roughly an order of magnitude greater than the global power generation capacity, though the discharge lasts for less than one millionth of a second.

Pulsed power development during the 1960s was primarily driven by a desire to conduct radiation effects testing and flash radiography for nuclear weapons applications (Martin, 1996; Van Arsdall, 2007). Radiation effects testing was dominated by multi-megavolt, high impedance ($> 1 \Omega$) systems, but increased interest in low photon energies and high fluences began to grow during that period, leading to lower impedance ($< 1 \Omega$), higher current (~ 1 MA) generators. Many low impedance systems primarily drove electron beams for x-ray production, but early studies of pulsed z-pinchs were also conducted (Turchi and Baker, 1973). Advances in capacitors and pulsed power architectures led to continued increases in the drive current from ~ 1 MA in 100 ns in the 1970s to ~ 27 MA in 100 ns in the 2000s.

Marx generator systems

The Marx generator (Marx, 1924) is the basis of most modern z-pinch pulsed power drivers (see Table 1). A Marx circuit consists of resistors, capacitors, and switches (Fig. 2A and B) arranged such that the capacitors are charged in parallel to the same voltage but discharged in series, creating a voltage equal to the number of capacitors times the individual charge voltage. This is an excellent method to generate large voltages, but because of the relatively high inductance, achieving high currents (multi-MA) with fast rise-times (~ 100 ns) is challenging. To increase the current of a Marx-based pulsed power driver, the current from several Marxes are combined in parallel, resulting in a peak current roughly equal to the number of Marx generators times the individual Marx generator current. The risetime of the Marx current pulse can be reduced by pulse charging a large, low-inductance, water capacitor (McBride et al., 2018) and using a multi-megavolt (MV) laser-triggered spark gap switch (LeChien et al., 2010); however, the faster risetime comes at a cost of reduced pulsed power driver efficiency.

The Z machine at Sandia National Laboratories is an example of a Marx-based system (Savage et al., 2011; Rose et al., 2010). Each Marx has 60 2.6- μ F capacitors, which are routinely charged to 85 kV, resulting in a voltage of 5.1 MV. Z has 36 parallel Marx generators, the combination of which can produce a peak load current of up to 27 MA. Pulse compression allows the $\sim 1 \mu$ s risetime of the Marx current pulse to be reduced to ~ 100 ns. At 85 kV, the energy in the capacitors is ~ 20 megaJoule (MJ), and ~ 2 MJ in electrical energy is delivered to the load; hence, the pulsed power driver efficiency is roughly 10%.

Table 1 An incomplete list of today's pulsed power drivers with the peak currents and risetimes from the references listed.

Driver	Location	Peak current (MA)	Risetime (ns)
Z (Savage et al., 2011)	Sandia National Laboratories, U.S.	27	85
Julong 1 (JianJun et al., 2013; Deng et al., 2016)	China Academy of Engineering Physics, China	8–10	90
Saturn (Sanford et al., 1996)	Sandia National Laboratories, U.S.	7	35
GIT-12 (Bugayev et al., 1997)	Institute of High Current Electronics, Russia	6	1700
ANGARA-5 (Branitsky et al., 1997)	Troitsk Institute of Innovative and Thermonuclear Investigations, Russia	4	90
ZEBRA (Chuvatin et al., 2010)	University of Nevada-Reno, U.S.	1–1.6	100
MAGPIE (Lebedev et al., 1999)	Imperial College, UK	1.4	200–260
COBRA (Greenly et al., 2008)	Cornell University, U.S.	1	95–230
Mykonos (Mazarakis et al., 2010)	Sandia National Laboratories, U.S.	1	70
MAIZE (Gilgenbach et al., 2009; Yager-Elorriaga et al., 2015)	University of Michigan, U.S.	0.6–1	75–140
CESZAR (Conti et al., 2020)	University of California-San Diego, U.S.	0.5–0.9	160–200

Because of feedback from the load to the driver, some variability occurs in the amplitude and risetime of the load current for each driver. All drivers listed are Marx-based with the exception of Mykonos, MAIZE, and CESZAR, which are LTD-based.

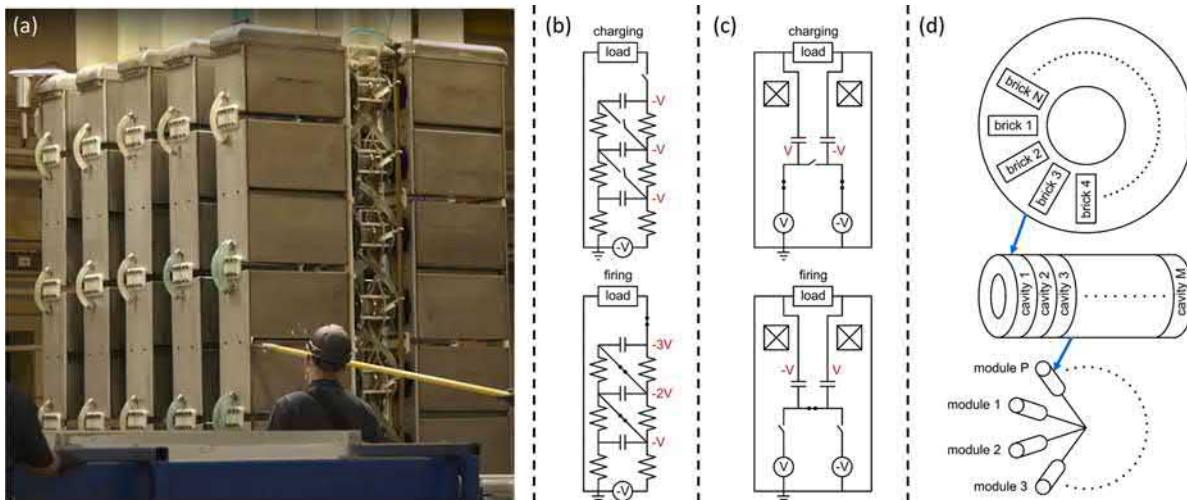


Fig. 2 (A) Image of one of the 36 Marx generators that form the Z machine. (B) Circuit diagrams of a 3-stage Marx generator in charging (top) and firing (bottom) states. When charging, all capacitors are connected in parallel to the charging supply, providing the same voltage to each capacitor. When the Marx generator is fired, spark gap switches are closed, connecting the high voltage end of one capacitor to the ground end of the next one. As a result, the voltage is equal to the number of stages times the charge voltage. (C) Circuit diagrams of the smallest element of a linear transformer driver, a “brick,” in charging (top) and firing (bottom) states. The two capacitors are oppositely charged and, when the switch closes, twice the charge voltage is applied across the load. The iron cores inductively limit the current path to ground. (D) Many bricks are connected in parallel to create a “cavity,” which can deliver a 1 MA, 100 kV, 100 ns risetime pulse to a matched load. If a greater voltage is required, several cavities can be stacked to form a “module” with the same current and risetime as an individual cavity but a voltage equal to the charge voltage times the number of cavities. Multiple modules can then be connected in parallel to increase the current of the driver (McBride et al., 2018).

Fast linear transformer drivers

In the late 1990s, an alternative to generate high currents with ~ 100 ns risetimes, called the fast linear transformer driver (LTD), was developed (Kim and Mazarakis, 2020). This concept avoided the less efficient pulse compression schemes needed with Marx systems by using a much larger number of smaller components. In a linear transformer driver, the smallest building block of the system (a “brick”) has the desired risetime, and these bricks are stacked in series and parallel to achieve the desired voltage and current (see Fig. 2C and D). This allows the driver efficiency to be as high as 40% (Mazarakis et al., 2010). In addition, high repetition rate (~ 0.1 Hz) operation of a single LTD cavity has been demonstrated with component lifetimes $> 10^4$ shots (Mazarakis et al., 2009). These characteristics are necessary but not sufficient for a z-pinch-based inertial fusion energy concept. A significant challenge with LTDs is that a high current (~ 60 MA) driver would contain over 10^6 capacitors and 5×10^5 spark gap switches (Stygar et al., 2007; Douglass et al., 2018), making individual component failure likely and tracking failures very challenging. Additionally, the magnetic cores required for each cavity in a large LTD-based driver would significantly add to the cost and timeline.

Fast Marx technologies (Lam et al., 2003) such as impedance-matched Marx generators (Stygar et al., 2017) have also been proposed as z-pinch drivers. Similar to linear transformer drivers, these concepts use a large number of smaller components that enable shorter risetimes than conventional Marx systems, eliminating the need for a pulse compression stage. In contrast to linear transformer drivers, these systems do not utilize magnetic cores.

Explosive pulsed power

An alternate method to generate pulsed high currents is through explosively-driven flux compression (Fowler et al., 1960; Shearer et al., 1968; Sakharov, 1966). Flux compression generators leverage explosives to implode a conductor, amplifying a seed magnetic field within the conductor; this technology provides a path to extremely high currents (~ 100 MA), but with longer risetimes ($\gg 1$ μ s). Flux compression generators are single use as the device is consumed in operation. An example of the use of flux compression generators in pulsed z-pinch studies is given below in the section on magneto-inertial fusion.

Radiation production with pulsed Z-pinch

Fusion can be produced directly with a pulsed z-pinch through the implosion of a deuterium column. This is the closest analog of the pulsed z-pinch to the classical z-pinch. Typically, the load is gaseous deuterium that has been injected into the load region, spanning the anode-cathode gap (Bailey et al., 1982). Deuterium gas puffs can be fielded on large drivers to create significant neutron yields (Klir et al., 2014; Coverdale et al., 2007). A drawback to this approach is that the quantity of gas required is large, so the fielding of tritium is somewhat impractical. Excluding tritium reduces the potential neutron production by nearly 2 orders of magnitude. Moreover, thermonuclear neutron production in gas puff z-pinch scales as the current to the fourth power (Welch et al., 2011), so scaling to large yields ($\gg 10$ MJ) is not feasible even with tritium. Additionally, the fuel density required at high current (> 80 MA) is beyond what is possible to generate with a puffed gas nozzle. Lastly, current coupling has proven problematic (Harvey-Thompson et al., 2016a) and may remain an issue on future high current drivers.

A z-pinch can also be used for x-ray production (e.g., metal wire array, argon gas puff) (Stallings et al., 1976), where the x-ray energy spectrum is dependent on the wire material. Low atomic number materials like aluminum, argon, iron, and copper produce K-shell x-rays (Ampleford et al., 2014) for radiation effects tests. High atomic number materials like tungsten produce a softer spectrum of continuum x-rays to drive ICF capsules (Sanford et al., 1999) and laboratory astrophysics studies (Rochau et al., 2014; Bailey et al., 2015).

In the 1990s and 2000s, z-pinch ICF research focused on arrays of fine metal wires (Matzen, 1997; Matzen et al., 2005), based largely on record-setting soft x-ray yields on Saturn (Sanford et al., 1996) and Z (Deeney et al., 1998). The innovation that led to significant increases in x-ray production was using many (~ 100 – 500) fine wires of ~ 10 μ m diameter to produce a more uniform implosion. X-ray yields as high as 2 MJ and x-ray powers as high as 330 TW have been produced on Z with a peak load current of 20–21 MA (Jones et al., 2014). The success of high-wire-number z-pinch on Saturn and Z led to numerous university studies of wire initiation and array implosion dynamics at ≤ 1 MA (Duselis and Kusse, 2003; Kantsyrev et al., 2006; Bland et al., 2007; Shelkovenko et al., 2007; Douglass et al., 2008; Gomez et al., 2008; Hall et al., 2008; Zier et al., 2009). Additionally, z-pinch studies at universities led to spin-off designs to study astrophysically-relevant plasmas in the laboratory (Bott-Suzuki et al., 2015; Hare et al., 2017; Lebdev et al., 2019).

The main goal of z-pinch studies on Z in the 1990s and 2000s was to develop an x-ray source capable of imploding a spherical ICF capsule while demonstrating the necessary shock timing and symmetry control. Many z-pinch concepts were investigated, and two promising concepts were identified. The cylindrical double-ended hohlraum (Matzen, 1997; Hammer et al., 1999; Matzen et al., 2005) and the cylindrical dynamic hohlraum (Smirnov, 1991; Matzen, 1997; Brownell et al., 1998) were thoroughly studied (see Fig. 3A and B, respectively).

The double-ended hohlraum consists of two wire arrays, each housed in a primary hohlraum, with a secondary hohlraum containing a fusion capsule between them. X-rays from the z-pinch drive the primary hohlraums. Those two hohlraums then drive the secondary hohlraum, and the secondary hohlraum drives the fusion capsule. The secondary hohlraum and capsule geometry share a lot of similarities with the laser indirect drive (Regan, 2021) configurations being tested on the National Ignition Facility (NIF) (Finnegan, 2021). A key difference is that the hohlraum containing the capsule is driven with soft x-rays, which avoids or reduces some issues associated with heating the hohlraum with laser beams (e.g., cross-beam energy transfer). Heating up the primary hohlraums to heat a secondary hohlraum causes some inefficiency; however, this design provides excellent irradiation symmetry of the capsule (Cuneo et al., 2012). As a point of reference, two 9 MJ z-pinch are believed to be sufficient to drive a capsule capable of creating ~ 500 MJ of fusion yield; these z-pinch would require a 65–70 MA driver (Vesey et al., 2007).

The dynamic hohlraum consists of a wire array surrounding a low-density foam that houses the capsule. The wire array implodes and strikes the foam, generating a radiative shock. The tungsten plasma surrounding the foam acts as a hohlraum and continues to implode toward the capsule, increasing the radiation temperature. On Z, this configuration produced a deuterium-deuterium neutron yield up to 3.5×10^{11} (0.07 kJ DT-equivalent yield), which was the highest at the time from an indirect drive concept (Ruiz et al., 2004; Rochau et al., 2007). Numerical scaling studies also indicated the dynamic hohlraum could produce ~ 500 MJ yields with ~ 60 MA drive currents (Mehlhorn et al., 2003); however, the lower load impedance would require a driver with significantly less stored energy than the double-ended hohlraum (Cuneo et al., 2012). Although the dynamic hohlraum would be more efficient at coupling energy, the capsule implosion symmetry is challenging as a result of the shrinking case-to-capsule

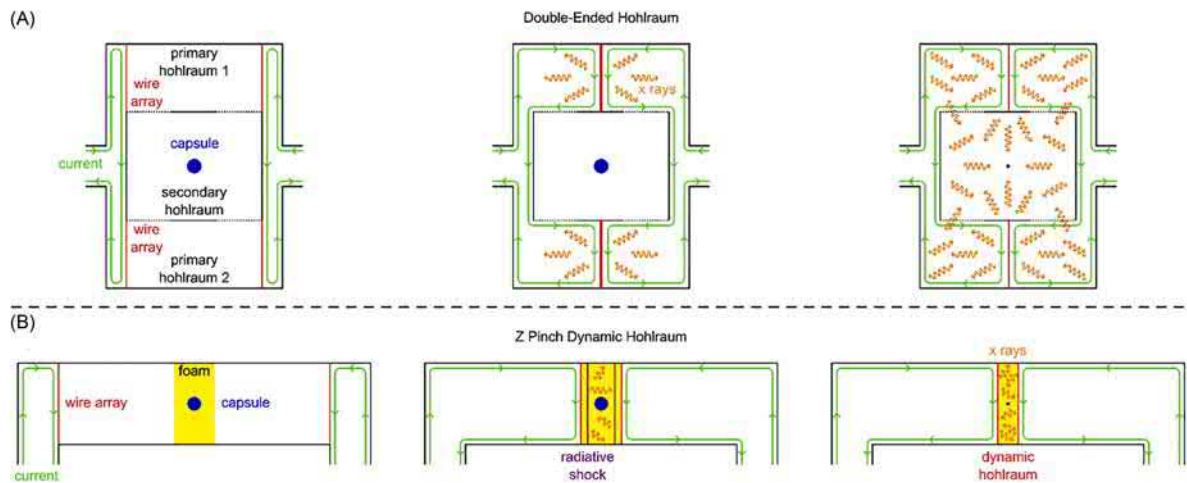


Fig. 3 (A) Cartoon depicting the key stages of the cylindrical double-ended hohlraum concept. Initially current (green) flows through two arrays of fine tungsten wires (red) as shown in the left image. The tungsten wire arrays become plasma, implode, stagnate on axis, and begin emitting x-rays (orange) as shown in the middle image. The **hohlraum** walls are heated and begin radiating soft x-rays, which implode a spherical fusion capsule (blue) as shown in the right image. (B) Cartoon depicting the key stages of the cylindrical dynamic hohlraum concept. Initially a current (green) flows through an array of fine tungsten wires (red) as shown in the left image. The tungsten wire array becomes plasma, implodes, and collides with a foam (yellow) generating a radiating shock (purple) as shown in the center image. The tungsten continues to implode while serving as the walls of the **hohlraum** to implode the spherical fusion capsule (blue) as shown in the right image.

ratio (Cuneo et al., 2012). Efforts to improve the dynamic hohlraum on Z ceased in the 2000s, but innovation of the concept has continued on the Julong-1 facility in China (Huang et al., 2017; Meng et al., 2017; Wu et al., 2018).

While wire array z-pinch source development on Z has continued for radiation effects sciences testing, efforts to use z-pinzches for indirect drive fusion on Z largely stopped in the mid-2000s. Two major factors contributed to this dramatic change in focus - the driver-target coupling efficiency and the start of operations on the NIF.

For indirect drive, the coupling efficiency includes (1) energy stored vs. electrical energy delivered to the load, (2) electrical energy to the load vs. x-ray energy radiated by the load, (3) x-ray energy radiated vs. energy absorbed by the capsule, and (4) energy absorbed vs. fuel thermal energy at stagnation (Cuneo et al., 2012). In an effort to address these inefficiencies, a magnetic direct drive configuration was developed, where the magnetic pressure supplied by the drive current is directly applied to a capsule. In magnetic direct drive only (1) energy stored vs. electrical energy delivered to the load and (2) electrical energy to the load vs. thermal energy at stagnation must be considered. Cutting out the steps where x-rays are generated and absorbed potentially increases the efficiency significantly, although it is important to note that indirect drive has been thoroughly studied for many years, making that concept much more mature than magnetic direct drive (Cuneo et al., 2012).

The second event that drove a shift in ICF research on Z was start of operations at the NIF. Significant progress was made in z-pinch-based indirect drive (sometimes called x-ray indirect drive) as a viable path to producing fusion, but the **hohlraum** temperatures generated in Z experiments did not allow the study of ignition scale capsule implosion physics, which was the largest remaining uncertainty for the concept. Efforts on the NIF, which was expected to generate significantly higher **hohlraum** temperatures, would be focused on studying implosion physics in ignition scale capsules. A goal of the ICF studies on the NIF was to achieve ignition, but there was an expectation that achieving > 10 MJ yields would require a larger driver than the NIF, which could take the form of z-pinch-based x-ray indirect drive on a next-generation pulsed power facility.

Magneto-inertial fusion

Between the limit of high density, small volume, and short duration plasmas characteristic of ICF and the opposite limit of low density, large volume, and long duration plasmas typical of MCF, exists a continuum of magneto-inertial fusion (MIF) concepts (Lindemuth and Kirkpatrick, 1983; Basko et al., 2000). In all MIF concepts, a magnetic field is applied to a fusion fuel-containing target, which is then imploded. The implosion increases the fuel temperature and density, and the shell surrounding the hot fuel acts as a tamper, providing the inertial confinement. The magnetic field reduces the rate at which thermal conduction losses occur between the fuel and the outer layers of the target, and the magnetic field also helps trap charged fusion products within the fuel, reducing the **areal density** requirement for self-heating. Implementations of MIF range from simply applying a magnetic field to an ICF capsule to improve its performance (Chang et al., 2011; Perkins et al., 2013) to imploding a low-density plasma torus within a metal shell (Degnan et al., 2013). MIF configurations in which current passes through a cylindrical target, causing it to implode, are a type of pulsed z-pinch and are referred to as magnetic direct drive fusion configurations.

One of the earliest MIF implementations was MAGO (magnitnoye obzhatiye, or magnetic compression), studied in Russia starting in the 1970s (Chernychev et al., 2016). In MAGO a hot, magnetized plasma is formed with an explosively-driven flux compression generator; a cylindrical or spherical **liner** is then imploded, compressing the hot plasma. This geometry is z-pinch-like, although the **liner** implosion is not necessarily driven by a current. Recently, research was conducted in the U.S. on Shiva Star to generate, transport, and crush a field reversed configuration; however, the lifetime of that configuration was insufficient for compression (Degnan et al., 2013). Z-pinch-like MIF experiments have also been conducted on the University of Rochester's Omega laser facility by illuminating the target surface to compress a cylindrical target containing magnetized fuel (Gotchev et al., 2009). Although magnetic pressure did not cause the implosion, the geometry was similar to that of z-pinch concepts. Neutron production in magnetized experiments was greater than in unmagnetized ones, but because of large variability in the performance of the unmagnetized targets, it is challenging to specify by how much. Other non-z-pinch research in MIF has included the Phi-target, which was tested on a predecessor of Z at Sandia National Laboratories (Chang et al., 1977; Sweeney and Farnsworth, 1981); the plasma liner experiment at Los Alamos National Laboratory, which creates standoff between the driver and target by injecting the fuel and **liner** with plasma guns (Hsu et al., 2012; Merritt et al., 2013); and acoustically-driven magnetized target fusion at General Fusion (Laberge, 2008).

Introduction to magnetized liner inertial fusion

In the late 2000s, a magneto-inertial fusion concept known as Magnetized Liner Inertial Fusion (MagLIF) was developed (Slutz et al., 2010). Similar to other MIF concepts, MagLIF relies on strong magnetic fields to limit thermal conduction losses and a **liner** implosion to compress and heat the fuel to fusion conditions. A major difference between MagLIF and most other MIF concepts (the Omega experiments listed above being the exception) is that the implosion occurs on a much shorter timescale (~ 100 ns), which enables the use of a significantly higher initial density fuel and open magnetic field lines within the target.

In MagLIF, an axial magnetic field of 10–30 tesla (T) is applied using Helmholtz-like coils to a 10-mm tall, 5-mm outer diameter, 0.5-mm wall thickness metal cylinder, which contains fusion fuel (see Fig. 4). Magnetization of the target is on a multi-ms timescale, which allows the magnetic field to diffuse through the conductors connecting the generator to the target as well as the target itself (Rovang et al., 2014).

A few-ns, multi-kJ laser pulse enters the magnetized target axially through a thin plastic foil, heating the fuel to hundreds of eV. The fuel density is roughly 10% of the critical density for a frequency-doubled Nd:YAG laser (527 nm), allowing the laser to penetrate and heat the fuel through inverse bremsstrahlung absorption. The current from Z passes axially through the target and causes it to implode with a peak velocity of 70–100 km/s.

The implosion velocity in MagLIF is slow by ICF standards but better matched to the 100-ns risetime of Z and allows for relatively thick target walls that are more robust to feedthrough of instabilities developed on the exterior of the target. The slow, stable

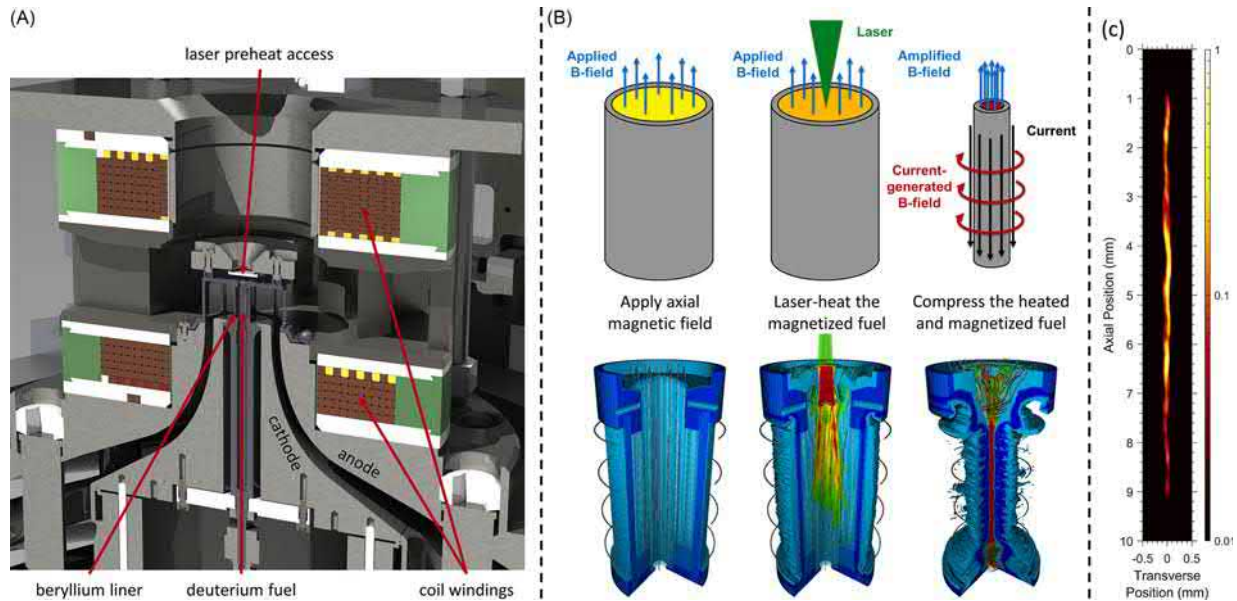


Fig. 4 (A) Cross section of MagLIF load hardware rendering showing the target, applied magnetic-field coils, and final transmission line. (B) Cartoon depicting the three MagLIF stages (top) and snapshots from a 3D magnetohydrodynamic calculation showing the same three MagLIF stages (bottom). Initially an axial magnetic field is applied to the target, which contains gaseous fuel. Next a laser enters the target through a thin plastic foil (a **laser entrance hole window**) to preheat the fuel prior to the start of the implosion. The metal walls of the target implode, compressing the fuel and magnetic flux contained in the target to achieve multi-keV temperatures and multi-kiloTesla magnetic fields. (C) X-ray self-emission image from the fuel at stagnation in a MagLIF experiment.

implosion necessitates the laser preheat; without it, the fuel could not reach multi-keV temperatures. Instead, a **convergence ratio** greater than 50 would be required, and instabilities would make such high **convergence ratios** unrealistic. The magnetic field serves (1) to limit thermal conduction losses from the preheated fuel to the target walls, enabling the preheat energy to be retained and augmented through the **PdV work** of the implosion, and (2) to trap charged fusion products at stagnation, enabling them to deposit their energy into the fuel to produce additional heating.

MagLIF experimental progress

The first Z experiments with the axial magnetic field, laser preheat, and **liner** implosion stages combined were conducted in November 2013 (Gomez et al., 2014a). The experiments used deuterium gas as the fuel, a 10 T axial field, 0.5 kJ laser preheat energy deposited, and 18 MA peak load current. The highest performing experiment produced a burn-averaged ion temperature of 1.9 keV and a deuterium-deuterium neutron yield of 5.3×10^{11} (0.07 kJ **DT-equivalent yield**). Over the next few months, subtle changes to the target geometry gave yields as high as 2.0×10^{12} (0.3 kJ **DT-equivalent yield**) (Gomez et al., 2014a).

Subsequent improvements to the target design have primarily focused on reducing radiative losses during the long (> 50 ns) delay between preheat and stagnation. Since bremsstrahlung radiation goes as roughly atomic number cubed, any **mix** with an atomic number greater than one will enhance radiative losses relative to the deuterium fuel, with higher atomic number materials having a significantly greater effect. The **laser entrance hole (LEH) window** foil thickness was reduced to (1) decrease the amount of hydrocarbon window material available to be driven into the fuel and (2) roughly double the preheat energy coupled to the fuel to around 1 kJ (Gomez et al., 2015; Harvey-Thompson et al., 2019). The enhancement in coupled preheat energy has increased the importance that all fuel-facing components within the target should be made from low atomic number materials (Knapp et al., 2019; Gomez et al., 2019). Switching several components from aluminum to beryllium increased the ion temperature to 2.2 keV and the neutron yield to 3.2×10^{12} (0.45 kJ **DT-equivalent yield**) by mid-2015.

Efforts to improve the quality of the laser beam spot and to modify the laser pulse shape (Harvey-Thompson et al., 2018) have increased the laser energy deposited to 1.4 kJ. An advanced applied magnetic field coil has enabled applied fields of 16 T without obstructing radial diagnostic access to the target, and modifications to the final transmission line have increased the peak load current to 20 MA. Experiments with these improvements have produced an ion temperature of 3.1 keV and a yield of 1.1×10^{13} (2.0 kJ **DT-equivalent yield**) (Gomez et al., 2020).

Scaling MagLIF results to high yields

The improvements to the MagLIF platform described above have enabled parametric scans of the applied magnetic field, coupled preheat energy, and peak current. The trends in the data are well matched by magnetohydrodynamic simulations (Slutz et al., 2018; Gomez et al., 2020), providing confidence in scaling these results to high yield. Additional insight into the scaling has been gained through experiments on the Omega laser at 1/10th spatial scale (Barnak et al., 2017; Davies et al., 2017; Hansen et al., 2020). In the Omega experiments, the **liner** compression was provided by laser direct drive of a plastic cylinder (Hansen et al., 2018) as opposed to magnetic direct drive of a metal cylinder.

Given the consistency between the data and the simulations, what might be achieved with a significantly larger pulsed power driver has been estimated. MagLIF scaling simulations indicate that multi-MJ yields and capsule breakeven might be achieved with a current pulse of ~ 45 MA and that tens of MJ of fusion yield (approaching facility breakeven) might be produced through volume burn with a current pulse of ~ 60 MA (Slutz et al., 2016). These two-dimensional simulations indicate ~ 60 MA is the threshold where burning DT ice becomes possible, which might enable high gain and hundreds of MJ of fusion yield (Slutz and Vesey, 2012; Sefkow et al., 2014; Slutz et al., 2016). In addition to the drive current being significantly increased, the coupled laser energy and applied magnetic field would also be increased to 30–60 kJ and 30 T, respectively, to achieve higher yields. These are large extrapolations outside of the explored parameter space, so careful work is needed to understand the uncertainties associated with these predictions.

Scaling uncertainties

To bolster confidence in scaling the MagLIF concept to high yields, the key uncertainties have been identified, and efforts to understand and mitigate these uncertainties are underway. These key uncertainties are laser preheat energy coupling, target magnetization, current coupling, implosion stability, and enhanced radiative loss through **mix**. Nearly 100 MagLIF experiments have been conducted on Z since 2013, with many focused on addressing these five scaling uncertainties.

Laser preheat energy coupling

Although laser preheat has been a consistent challenge for MagLIF, efforts to understand this stage via experiments (Geissel et al., 2016, 2018) with the Z-beamlet laser (Rambo et al., 2016), independent of Z, have led to continued improvement. Experiments (Harvey-Thompson et al., 2015, 2016b) have also been conducted on the Omega-Extended Performance laser facility (Waxer et al., 2005), which has enhanced beam smoothing capabilities as well as the ability to investigate laser wavelength effects.

To date, the maximum laser energy coupled to the fuel in a MagLIF experiment was 1.7 kJ with $\sim 70\%$ coupling efficiency. Further improvements in preheat energy coupling efficiency will require either cryogenic cooling to reduce the fuel pressure, which

will allow the thickness of the **LEH window** to decrease (Awe et al., 2017), or will require an alternative method to reduce the effects of absorption in or backscatter from the **LEH window**. One such alternative is the laser-gate concept (Slutz et al., 2018; Miller et al., 2020) in which the window is rapidly opened just before the laser arrives. Given the infrastructure of the Z-beamlet laser, the maximum laser energy that could be delivered to a MagLIF target is $\sim 5\text{--}6$ kJ; hence, 4–5 kJ preheat energies appear plausible on Z and are roughly optimal (Slutz et al., 2018) for the 20–22 MA maximum peak load current expected in Z experiments.

Given the order of magnitude difference in preheat energy between what is possible on Z and what is required for ignition, efforts to study ignition-scale laser preheat have become a focus for MagLIF. Preheat experiments are being conducted using four beams of the NIF. In these experiments, more than 30 kJ of laser light can reach a surrogate MagLIF target, enabling studies of ignition-scale preheat energy coupling. An applied magnetic field capability of > 20 T has also been developed at the NIF in support of this effort.

Target magnetization

Magnetization of a MagLIF target is challenging because radial diagnostic access is strongly desired. Magnetic coil designs that can reach 25–30 T but do not maintain diagnostic access are available (Rovang et al., 2014). Efforts to develop internally-reinforced coils to enable high magnetic fields while preserving diagnostic access began in 2017. Such a coil configuration, which magnetized a MagLIF target to 22 T, was first demonstrated in offline tests conducted in early 2020. Further improvements to this design, which are expected to provide 25–30 T, are under development. An alternative magnetization scheme has been proposed (Slutz et al., 2017); for that scheme, the main drive current is directed in a helical path through the target in order to generate an axial magnetic field of $\sim 50\text{--}200$ T within the target (Shipley et al., 2019). The implosion stability of such a helically structured target is being investigated. Scaling calculations indicate that the optimal field for MagLIF is close to 50 T when the peak load current is roughly 20 MA (Slutz et al., 2018) and drops to 20–30 T when the current is 40–60 MA (Slutz et al., 2016).

Current coupling

Understanding electrical power flow to a target is critical for pulsed power-based systems; hence, a focused effort to model and obtain data on Z's power flow began in the late 2010s. The potential for plasmas to form increases as the current density is increased in a transmission line, and these plasmas have been linked to current losses (Rose et al., 2008; Gomez et al., 2017; Hutsel et al., 2018; Bennett et al., 2019). Moreover, in higher current drivers, plasmas will form over a larger radial extent of the transmission lines, which increases the uncertainty in scaling the current coupling. About 10 dedicated power flow physics experiments have been conducted per year, with an initial focus on developing a suite of visible light and charged particle diagnostics (Gomez et al., 2014b; Gomez et al., 2017; Dolan et al., 2018; Datte et al., 2020). The primary goal of such experiments is to obtain constraining datasets against which power flow simulations can be vetted to increase confidence in extrapolating to a future facility that would deliver 40–60 MA to a target.

Implosion stability

Achieving implosion stability is critical as the disassembly time at stagnation in an ICF target depends on the integrity of the tampering material. Magnetohydrodynamic simulations of **liner** implosions can match the general behavior observed in magnetically-driven implosions (Martin et al., 2012), but some observations are puzzling. Identifying and understanding the missing physics in these simulations is ongoing. A few examples are given below.

Studies of the development of instabilities in experiments with a prescribed instability seed are well predicted by simulations (Sinars et al., 2010); however, a thorough study of development of instabilities in smooth surface targets has shown an unexpected level of azimuthal correlation of instability structures (McBride et al., 2012). The cause of such behavior was hypothesized to be an electrothermal instability (Oreshkin et al., 2004; Peterson et al., 2012) that created a large seed for the magneto-Rayleigh-Taylor instability. That hypothesis is consistent with experiments in which a dielectric coating was applied to the exterior of a target to tamp expansion of the electrothermal plasma, resulting in a significant reduction in the magneto-Rayleigh-Taylor amplitude (Peterson et al., 2014; Awe et al., 2016).

In **liner** implosions with an applied axial magnetic field, the instability structures that developed were helical (Awe et al., 2013). However, these structures were expected to be azimuthal because, at the time the **liner** began to implode, the azimuthal magnetic field was much larger than the axial field applied to the target (Awe et al., 2014). Several hypotheses to explain these structures have been proposed (Ryutov et al., 2014; Sefkow, 2016; Seyler et al., 2018), and efforts to determine the origins of the helical instability experimentally are underway. Understanding this behavior is important because helical structures are seen in the fuel column at stagnation, indicating feedthrough of the instability from the exterior surface of the **liner** to the interior surface. Prediction of such behavior in simulations may allow the development of mitigation schemes.

Mix

The introduction of mid- or high-atomic number material into the fuel is a concern for all fusion concepts; the enhanced radiative losses that result can significantly degrade target performance. In MagLIF, the implosion duration is significantly longer than in a typical ICF implosion, which would enhance the detrimental impact of **mix** introduced early in a MagLIF implosion. Typically, the laser preheat timing is chosen to maximize the work that the **liner** does on the fuel during the implosion; however, this also maximizes the radiative losses for any laser-induced **mix** in a MagLIF target. The two main sources of laser-induced **mix** are the **LEH window** material and material liberated from the inner surface of the target via direct laser strike. For a well-designed laser

pulse shape, laser spot size, and target fuel density, injection of the window material can be essentially eliminated (Harvey-Thompson et al., 2018). Similarly, a well-designed laser pulse should not strike any surface within the target (Weis et al., 2021) and, if all components within the target are coated in deuterium ice, any mix would have a minimal impact on target performance (Slutz et al., 2018).

The future of pulsed power and Z-pinchs

Z-pinch-based fusion efforts require a larger drive current than is available today. Several concepts explored on present-day pulsed power facilities are predicted to reach tens to hundreds of megajoule yields with ~ 60 MA drive current, according to simulations (Mehlhorn et al., 2003; Vesey et al., 2007; Slutz et al., 2016). A number of large-scale facilities have been proposed that could enable such tests. In Russia, the Baikal generator, a Marx-based facility designed to produce 50 MA with a 100–300 ns risetime, was proposed but has yet to be built (Grabovsky et al., 2001). In China, two facilities have been proposed based on fast linear transformer driver technology, with construction of one test module (out of 60) in progress for the larger, 60-MA, 120-ns risetime facility (Chen et al., 2019; JianJun, 2020). While the second proposed Chinese facility (CZ30) is a bit smaller at > 30 MA, it would still be the highest current pulsed power facility in the world (Mao et al., 2019). In the United States, both Marx-based and linear transformer driver-based concepts for a next generation pulsed power facility have been proposed (Stygar et al., 2007, 2015; Douglass et al., 2018; Sinars et al., 2020) for ~ 45 –65 MA drive currents.

A key milestone for z-pinch-based ICF, which would require a higher current driver than is operational now, is to produce more fusion energy from a target than the electrical energy invested in the target. Simulations suggest that this threshold could potentially be reached with a fusion yield of ~ 1 MJ and a peak current of ~ 30 MA (Slutz et al., 2016). A longer-term goal is to demonstrate high gain, which would require burning DT ice to produce hundreds of MJ (Hammer et al., 1999; Mehlihorn et al., 2003; Slutz and Vesey, 2012; Slutz et al., 2016). Demonstration of a single-shot, high-gain configuration is an objective for stockpile stewardship and inertial fusion energy; however, considerable additional research would be required to apply such a configuration to inertial fusion energy (Sinars et al., 2015).

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Overview of IFE Key Power Plant Technologies*

E. Michael Campbell^a, Jose Manuel Perlado^b, Javier Sanz Gozola^{b,c}, Susana Reyes^d, and Michael Tobin^e, ^aLaboratory for Laser Energetics, University of Rochester, Rochester, NY, United States; ^bInstituto Fusión Nuclear “Guillermo Velarde”, Universidad Politécnica Madrid, UPM, Spain; ^cDepartamento de Ingeniería Energética, Universidad Nacional de Educación a Distancia, UNED, Spain; ^dSLAC National Accelerator Laboratory, Stanford University, Menlo Park, CA, United States; and ^eJohns Hopkins University Applied Physics Laboratory, Air and Missile Defense Systems, Laurel, MD, United States

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Glossary

Activated Material Material that has experienced neutron irradiation resulting in neutron capture and the production of radioactive isotopes.

Additive Manufacturing Commonly referred to as “3-D printing” this refers to a manufacturing technique that builds components by layering a powder that is melted by a precision laser or electron beam according to a computer design drawing.

Artificial Intelligence The theory and development of computer systems to be able to perform tasks that normally require human intelligence such as decision-making.

*Dedication to Professor/General Dr. Guillermo Velarde.

Beta-Decay In nuclear physics, beta decay (β -decay) is a type of radioactive decay in which a beta particle (fast energetic electron or positron) is emitted from an atomic nucleus, transforming the original nuclide to an isobar of that nuclide, that is, a material with the same mass number but a different number of protons.

Blackbody The amount of radiation energy emitted from a surface at a given wavelength depends on the material of the body and the condition of its surface as well as the surface temperature. Various materials emit different amounts of radiant energy even when they are at the same temperature. A body that emits the maximum amount of heat for its absolute temperature is called a blackbody.

Debris After the target completes its fusion burn, the material remaining, consisting of unburned and very energetic T and D atoms, plastic CH material in a plasma state, Be and/or C from the capsule, and/or Pb (or other high Z material) in a plasma state that had been used for the hohlraum; expands and impacts the chamber first wall.

Deep Learning (DL) A subset of machine learning applications that teaches itself to perform a specific task with increasingly greater accuracy, without human intervention.

Displacements per Atom A measure of the amount of radiation damage in neutron-irradiated materials where 10 dpa means each atom in the material has been displaced from its site within the structural lattice of the material an average of 10 times due to interactions between the atoms and the energetic neutrons irradiating the material.

Efficiency-Gain Product The efficiency of whatever driver is being used in an IFPP multiplied by the gain that this driver creates, economic studies have shown, should be a value of about 10 in order to be economical and lead to a competitive cost of electricity eV Condition: Temperature greater than an eV or 11,000 Kelvin.

FAIR Facility for Anti-proton and Ion Research being built at the GSI in Germany, will be one of the largest and most complex accelerator facilities in the world. The FAIR accelerator facility will have the unique ability to provide particle beams of all the chemical elements (or their ions), as well as antiprotons. The particles will be accelerated to 99% of the speed of light. At the heart of the facility is an underground ring accelerator with a circumference of 1100 m. The existing accelerator facility will serve as the injector for the new FAIR facility which will start operation in 2025.

Fermi degenerate state Fermionic matter in highly dense state in which the Pauli exclusion principle exerts significant pressure in addition to, or in lieu of thermal pressure. The description applies to matter composed of electrons, protons, neutrons or other fermions.

Fermi-Dirac Statistics In quantum statistics, a branch of physics, Fermi-Dirac statistics describe a distribution of particles over energy states in systems consisting of many identical particles that obey the Pauli exclusion principle. It is named after Enrico Fermi and Paul Dirac, each of whom discovered the method independently.

Fermions In particle physics, a fermion is a particle that follows Fermi-Dirac statistics and generally has half odd integer spin such as $1/2$, $3/2$, etc. These particles obey the [Pauli exclusion principle](#). Fermions include all quarks and leptons, as well as all composite particles made of an odd number of these, such as all baryons and many atoms and nuclei. Some fermions are [elementary particles](#), such as electrons, and some are composite particles, such as protons. According to the [spin-statistics theorem](#) in relativistic quantum field theory, particles with [integer](#) spin are [bosons](#), while particles with half-integer spin are fermions. Composite fermions, such as protons and neutrons, are the key building blocks of [everyday matter](#). The name fermion was coined by English theoretical physicist Paul Dirac from the surname of Italian physicist Enrico Fermi.

First Wall The first surface in a IFPP chamber that receives the IFE target emissions.

Flash-Lamped Pumped Tubes filled with xenon gas discharge its line emissions into Nd-doped phosphate glass acting as laser amplifiers for the NIF.

Frequency Up-Conversion The Nd laser emission line of 1.056 nm is frequency doubled to 528 nm and the frequency tripled to 352 nm to use for driving an ICF direct or indirect drive target.

Fuel Configuration at Stagnation This is a key measurement to make during current NIF experiments that shows whether the hotspot and dense fuel regions are properly formed and are free of mix.

Fusion A nuclear reaction where two light nuclei merge and produce products with different mass numbers than the reactants.

Gain The yield of an ICF target divided by the driver energy that caused the yield.

GSI GSI Helmholtzzentrum für Schwerionenforschung is a German research center in Darmstadt, Germany that operates a unique large-scale accelerator for heavy ions.

Heavy Ion "Heavy" by convention is an atom that has an atomic number greater than 13 and is ionized so that it can be accelerated by magnetic fields.

HIAF The High Intensity heavy ion Accelerator Facility is a new facility planned in China for heavy ion related research. The key features of HIAF are unprecedented high pulse beam intensity and versatile operation mode. The facility will be commissioned in 2025.

Hotspot Mix The baseline approach to ICF creates a "hotspot" in the center of the assembled fuel where the fusion reactions initiate leading to a propagating burn wave into the cold dense fuel. If material enters the hotspot region because it mixes in during the compression phase, then the hotspot is degraded and cannot ignite the fusion fuel.

ICF Target This refers to either a directly or indirectly driven capsule encased in other materials to convert the particular driver energy being used into an ablative force to drive the capsule to fusion conditions.

Ignition The condition achieved in an ICF capsule where hot DT fuel in a low density region in the center of the fuel region undergoes fusion and the alpha particles released heat the nearby much higher density but colder fuel resulting in a substantial increase of the ion temperature and more fusion reactions leading to about 250 kJ of net fusion yield for a NIF scale target.

Inertial Fusion Power Plant A facility that generates electricity by harnessing the fusion reaction in a pulsed mode.

ITEP Institute for Theoretical and Experimental Physics located in Moscow, Russia.

Laser Light Amplified by the Stimulated Emission of Radiation that, applied to IFE, means collimated photons of a single wavelength that can be focused to single spot.

Low Activation Tendency of a material to activate with radioisotopes with short half-lives.

Machine Learning (ML) A field within artificial intelligence that keeps a computer's built-in algorithms current regardless of changes in its environment.

Material Damage Fusion neutrons penetrate into materials causing atomic displacements in the lattice structure of the molecules making up the material; when excessive this can change materials properties such as tensile strength.

Molecular Dynamic Simulation Molecular dynamics (MD) simulation is a widely used approach to study the time evolution of a system of "particles," typically atoms or molecules with defined properties.

Neutralized Drift Compression Experiment II (NDCX-II) A linear induction accelerator located at Lawrence Berkeley National Laboratory where 0.8–1.2 MeV helium ion pulses with intensity up to 1.5×10^{18} ions/mm²/s in a few ns pulses are available where the number of ions is tunable up to 10^{11} ions/pulse and the beam spot sizes can be tuned from 1 mm² to 1 cm² and the repetition rate is 1.3 shots/min (Seidl et al., 2018).

Neutron Heating This describes a process where neutrons undergo elastic or inelastic collisions with atoms in a material the neutron has entered the result of which is energy is transferred to the target atom and the neutron moves on with less energy. The material realizes this effect as an increase in its temperature or heating.

Neutron Shielding Materials that are effective in slowing down and/or absorbing neutrons.

Neutronics Analyses that describe the effects of neutrons.

Neutrons One of the products of the fusion reaction between Deuterium and Tritium; sub-atomic particle with nearly the mass of a proton but with no charge.

Object Kinetic Monte Carlo Simulation A Monte Carlo method computer simulation intended to simulate the time evolution of some processes occurring in nature - typically processes that occur with known transition rates among states. These rates are inputs to the KMC algorithm, the method itself cannot predict them.

Pauli Exclusion Principle The assertion that no two fermions can have the same quantum number that is occupy the same quantum state at the same time.

Polar Drive Using the NIF beams to irradiate a direct drive target. It is called "polar" because there are no "equatorial" beams on the NIF that would complete a direct drive configuration.

Pulsed Power Applied to IFE, the delivery of a very large currents of electricity (> Mega-Amps) in a very short time (100 ns) to an anode-cathode gap that contains a direct drive DT target.

Relativistic Heavy Ion Collider The RHIC is the first machine in the world capable of colliding heavy ions, which are atoms which have had their outer cloud of electrons removed. RHIC primarily uses ions of gold, one of the heaviest common elements, because its nucleus is densely packed with particles, and collide them at nearly the speed of light.

Residual Radioactivity Activated material that has isotopes that are long enough that decay radiation is persistent for a period of time.

Shell Asymmetry Any aspect of an ICF capsule that would make it deviate from a perfect spherical surface.

Tritium A radioactive isotope of hydrogen with a half-life of 12.32 years. The nucleus contains one proton and two neutrons.

Ultra-high Vacuum Vacuum of 10^{-8} Torr or lower.

Warm Dark Matter A popular (if speculative) "dark matter" particle, one that has some ability to interact with very light particles that move close to the speed of light forms one of several possible warm dark matter (perhaps more accurately called interacting dark matter) scenarios (Bose et al., 2019).

X-Rays The region of the capsule where the fusion reactions take place emits X-rays in conjunction with these reactions; in addition, the very hot debris also emits X-rays. ICF X-rays are photons from energies as cold as 1 eV and as hot as several hundred keV.

Yield The net energy released by an IFE target from fusion reactions.



Dr. Guillermo Velarde was Professor Chair of Nuclear Physics since 1973 at the Industrial Engineering College of the Universidad Politécnica de Madrid and General in the Division of the Spanish Air Force. Professor Velarde began at the Spanish Atomic Energy Commission in 1958. In the early 1970s he initiated research in Inertial Confinement Fusion Physics (ICF) with a small but outstanding group developing several integrated/coupled codes for Radiation-Hydrodynamic and Particle Transport Physics (NORMA, CLARA and NORCLA) to study the implosion and burning of fusion-fission and pure fusion targets. The codes' results, published in various journals starting in 1976, had a large international impact in the fusion scientific community, attracting the interest of laboratories such as Kernforschungszentrum Karlsruhe (Germany), Lawrence Livermore National Laboratory (USA), Institute of Laser Engineering Osaka University (Japan), Laboratory for Laser Energetics at the University of Rochester, Lebedev Institute Moscow (Russian Federation) and Commissariat d'Énergie Atomique (France). Guillermo published in 2007 to critical acclaim a history of early ICF research activities with original contributions by early researchers called "Inertial Confinement Nuclear Fusion: A Historical Approach by Its Pioneers" (Velarde and Carpintero-Santamaria, 2007). In 1981 Guillermo founded the Instituto de Fusión Nuclear (IFN) (today re-named Instituto de Fusión Nuclear "Guillermo Velarde") within the Universidad Politécnica de Madrid with a small group of scientists, C. Ahnert, J.M. Aragones, J.M. Martinez-Val, E. Minguez, N. Carpintero, J. M. Perlado, and later P. Velarde and J. Sanz. From its start the IFN initiated many high level international scientific relationships in research fields dedicated to generation of electrical power from Inertial Confinement Fusion. This effort played a fundamental role in the promotion of research for Inertial Confinement Fusion in EURATOM, through its decision-making committees such as the Consultative Committee EURATOM Fusion and the Keep in Touch ICF Programs throughout the Framework Programs V, VI, and VII. Guillermo was president of the Keep in Touch from 1998 to 2006. These programs continue today in EuroFusion with the "Towards Inertial Fusion Energy Project." Radiation-Hydro codes for target design together with new multiscale modeling of key mechanisms in reactor development in Heavy Ion and Laser Inertial Fusion Energy were the goals under his direction. Professor Velarde also reinforced the role of the International Atomic Energy Agency for Inertial Fusion Energy (IFE). He, together with many other outstanding scientists he recruited, wrote the IAEA book on Energy from Inertial Fusion. He also proposed and led several successful IAEA Coordinated Research Projects in Inertial Fusion. Guillermo wrote/co-wrote 14 books, including "Quantum Mechanics" (2002) and published 325 papers. He was a committee member of more than 55 International Conferences and addressed more than 300 Conferences in 18 different countries. Guillermo received many awards including the Edward Teller Award in 1997; the Archie Harms Award in 1998 and the Marqués de Santa Cruz de Marcenado Award for his outstanding scientific work in the Armed Forces in 2011. In 2007 he was appointed Member of the European Academy of Sciences. Professor Velarde passed away on January 12, 2018 but is always with us inspiring our future endeavors.

Introduction

Technology is the sum of the skills, tools, machines, materials, techniques, and processes used to accomplish a technical goal. Here we present an overview of key technologies necessary to succeed in delivering Inertial Fusion Power to the world.

The chapter is divided into five sections. The first section provides a general introduction to the components of an Inertial Fusion Power Plant (IFPP) based on the deuterium-tritium fuel cycle and goes into some detail about each component. The second section describes key technologies that need to be addressed for a successful IFPP. The third section describes the interrelationships between chamber, driver, and target technologies that provide options for IFPPs. The fourth section describes more of the conventional technologies that are also required for IFPPs. The last section discusses emerging technologies that are expected to make major impacts on IFPP technology: additive manufacturing and artificial intelligence/machine learning.

An IFE Power Plant schematic is shown in Fig. 1. The IFE Driver sends beams of energy in short pulses into the IFE Chamber where they irradiate an IFE Target producing fusion yield. The rate of target irradiation and yield can span from less than one to many times per second. The IFE Targets are produced in the IFE Target Factory. Deuterium and tritium filled capsules are mated with the Target for injection into the Chamber. The Chamber exports heated material to the Electricity Production which converts

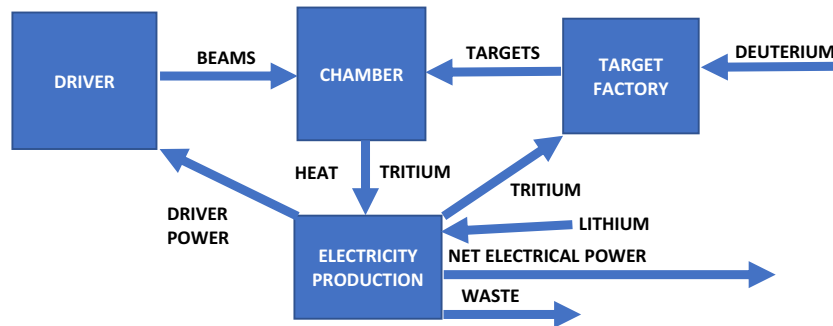


Fig. 1 Schematic of an Inertial Fusion Power Plant (IFPP).

this to electricity. Some power is sent to the Driver and Power Plant Operations (the recirculating fraction) while the remaining power goes out onto the Electric Distribution Grid as net power produced.

There are important aspects of implementing ICF into IFPP that impacts technology development. For ICF the fusion reaction is studied on the scale of a cubic centimeter. This means that key physics research moves forward through changes in target design or driver energy delivery. Such improvements could be implemented into a change in targets being manufactured for the IFPP or a change in the details of the beams to target engagement. The activated material and tritium are generally confined to the chamber systems. Also, the total level of residual radioactivity can be kept small for an IFPP due to being able to provide neutron shielding protection of the first wall through careful selection of in-chamber materials. Depending on material selections and the degree of first wall protection, IFPPs may be able to have first walls last an entire plant lifetime. A relatively small amount of tritium is put at risk at any given time for an IFPP since the major tritium concentration is in the target fabrication facility and not in the IFPP Chamber. IFPP Chambers do not require ultra-high vacuum levels to support beam propagation. IFPP may have materials selected having low activation to allow some hands-on maintenance. Finally, IFPPs can separate its driver, chamber, target factory and balance of plant, reducing the risk to any single item.

IFE and MFE share many development issues and both technologies remain aware of dual benefit research. Similar principles are in play for both approaches for fusion energy related to the neutrons released in the fusion process. For example, neutron deposition and tritium production in the breeding blanket of the chamber. The capture of neutron energy in the fluid in the primary thermodynamic cycle of the power plant and at the same time can produce undesirable effects such as activation and materials damage. A major difference however is in considering the first wall which is affected in a very different way in magnetic and inertial fusion designs because of the ability for inertial fusion to protect the first wall.

Key technical challenges for IFE

This section describes especially key technical challenges for the realization of Inertial Fusion Power drawing on the last few decades of ICF and IFE research.

Demonstration of ignition and gain with an affordable driver

The most important next step in implementing Inertial Fusion Energy is for a facility to achieve target ignition and gain. This is important for three fundamental reasons. First, this would prove that the physics of capsule implosion has been proven to be correct. Since the Multi-Mega-joule National Ignition Facility at LLNL is the only present facility in the world with the driver energy to potentially achieve ignition and gain, this is the facility, using its flash-lamped pumped, frequency up-conversion to the ultra-violet (350 nm wavelength) solid-state laser, that is the most likely to demonstrate this first step toward Inertial Fusion Energy with its indirect drive approach. As discussed below, the indirect drive approach uses X-rays produced in a laser heated high Z enclosure (hohlraum) to implode the fusion capsule. Directly illuminated targets, where the fusion capsule is directly illuminated by the laser, are also being explored on the NIF utilizing the irradiation pattern for indirect drive. This “polar drive” configuration has the least impact on facility modifications. The NIF target chamber also has ports for spherical illumination but would require redirecting some of the NIF beams. Physics research over the next decade will motivate more investments in the direct capability on NIF. Second, a demonstration of ignition and gain would also show that the target to achieve this was able to be built and successfully fielded. Given that the target is at cryogenic temperatures and must be built to sub-micron tolerances, this is a significant challenge. This target would then evolve to examine simpler approaches where the “performance cliff” would be understood where ignition and gain is possible and where it is not. Furthermore, the basic physics for other types of targets such as heavy ion targets and direct drive targets would also see important research as the ignition “cliff” for alternative options that may turn out to be more economical would also be studied. Finally, on a single shot basis, some fundamental aspects of emissions of debris, X-rays, and the neutrons could be studied to aid in chamber confinement studies.

Basic inertial confinement fusion target physics

This section describes the basic processes for achieving inertial confinement fusion (ICF) using a suitably formatted driver energy.

In ICF, equal amounts of deuterium (D) and tritium (T) fusion fuel contained in a capsule are compressed and heated to a very high density, temperature and pressure so that the fusion reaction ($D + T \rightarrow n + {}^3\text{He}$) rate is quick enough to cause a significant portion of the fuel to be consumed in these reactions, perhaps as much as one-third, before the capsule is destroyed (Nuckolls et al., 1972; Lindl, 1998). A successful implosion requires fuel pressures of tens to hundreds of Gbars, fusion fuel temperatures of 55 million degrees K, and a density of a few hundred g/cm^3 . In this fusion reaction, 17.6 MeV of energy is released in the form of a 14.1-MeV neutron and a 3.5-MeV alpha particle. The specific energy release from DT is 4×10^{11} J/g. The neutron, with an energy substantially greater than that released by fission by a factor of 5 or more, escapes the high-density burning fuel after depositing a small fraction of its energy. The alpha particles however are stopped and deposit all of their energy and rapidly further heat the fuel. When this occurs, the fusion reaction rate rapidly increases since the rate scales as the square of ion temperature, T_i . To guarantee that the alpha particle is stopped and “bootstraps” the reaction, the thickness $\times$ the density ($\rho \times \Delta R$) must be at least $0.3 \text{ g}/\text{cm}^2$ while the fuel temperature is 5 keV. This is called the Lawson criteria for ICF and is equivalent to the $n \times \tau \times T_i$ for magnetic fusion, where τ is the energy confinement time and n is the particle density. Any driver considered for an IFPP must create these conditions for the capsule. There are five approaches to achieve these conditions: Laser Indirect Drive (LID), Laser Direct Drive (LDD), Heavy Ion Direct Drive (HIDD) and Heavy Ion Indirect Drive (HIID), and Pulse Power Direct Drive (PPDD). These approaches are discussed in the rest of this section along with comments as to their level of progress to achieve IFPP needed conditions in an ICF capsule.

Indirect drive using solid state lasers

A laser driver must compress sufficient energy in space and time and apply it to the IFE target. In LID, the laser energy is first converted into X-rays in a high-Z enclosure, named hohlraum from the German word for cavity, where a near Planckian radiation source with an effective radiation temperature in the resulting plasma of 300 eV is created. The X-ray flux then ablates the outer surface of the capsule. Fig. 2 illustrates the concept of an indirect drive target and also shows the capsule design for 1.8 MJ laser energy deposition. The National Ignition Facility (NIF) is the world’s leading facility for Laser Indirect Drive research. The NIF followed a series of ICF research facilities at LLNL starting with the Argus laser facility in 1976, the Shiva laser facility in 1977, the Nova laser facility in 1985, and the Beamlet facility in 1995. Each facility demonstrated sufficient progress to lead to the next larger laser facility.

The LID approach is the most researched and uses a solid-state laser at 1056 nm that is frequency tripled to 350 nm using potassium di-hydrogen phosphate crystals and used in an indirect drive configuration of three cone angles of beams per side to distribute the energy within the hohlraum. The method of indirect drive is inherently inefficient but was chosen because it was believed to be the lowest risk for successful capsule drive to ignition and gain conditions. The target is a hohlraum with a small 2.2-mm cryogenic capsule filled with D-T at 18 K and surrounded by a low density He gas to prevent the hohlraum wall material from moving in and affecting the capsule implosion. A large number of laser beams (NIF has 192 beams) are carefully placed in such a way that laser spot intensities of order $5\text{--}10 \times 10^{14} \text{ W}/\text{cm}^2$ are produced. Each of these spots are a source of X-rays that then drive the capsule to fusion conditions.

To minimize the driver energy to achieve fusion conditions in the imploded fuel the following physics strategy has been developed. If all of the DT was heated to 5 keV, it would require $\sim 375 \text{ kJ}$ of energy coupled to the fuel since the specific heat of DT is $\sim 100 \text{ MJ}/\text{KeV-gram}$, assuming equal electron and ion temperatures. To put this in perspective, consider that the 1.8 MJ National Ignition Facility (NIF) laser at LLNL at present, with the inefficiencies described above, with laser indirect drive targets, couples only $\sim 10\text{--}20 \text{ kJ}$ to the compressed fuel (hot spot and main fuel) (Soures et al., 1996). The minimum energy required to compress DT occurs when the fuel is in a “Fermi degenerate state.” When the fuel is in this state the pressure is dominated by the degeneracy

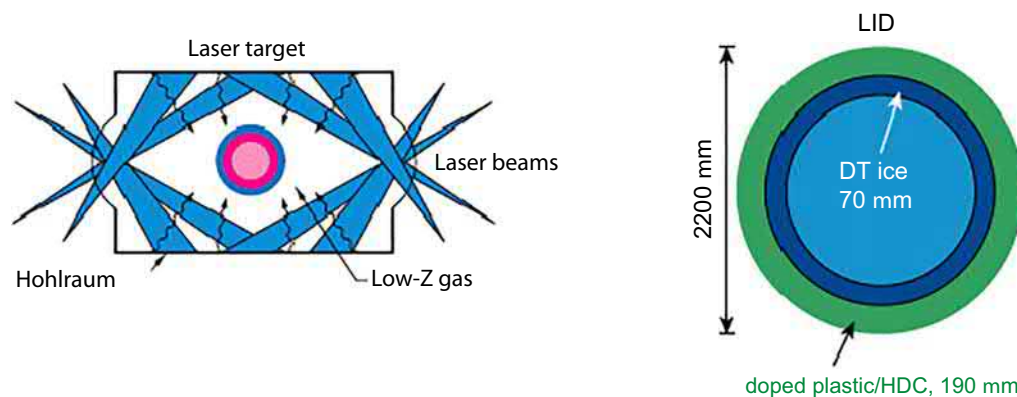


Fig. 2 An indirect drive capsule uses a hohlraum to convert laser driver energy to x-rays to drive the capsule (DOE, 1999). On the right the design of the capsule is shown (Campbell, 2020).

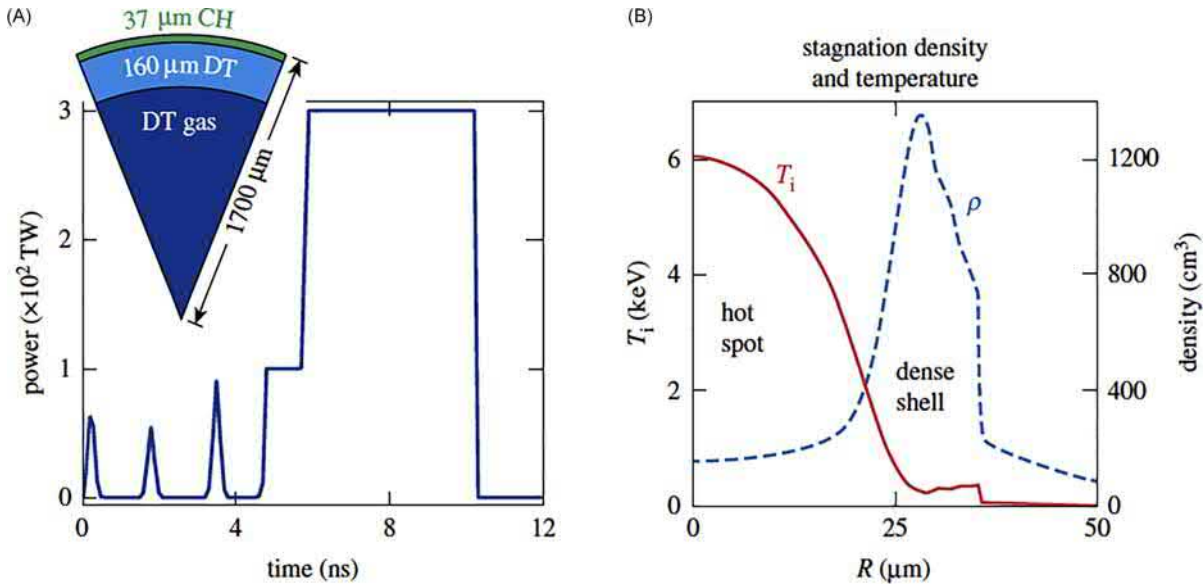


Fig. 3 On the left (a), the target and laser temporal profile is shown that produces the fuel conditions shown on the right (b) at stagnation. This shows the ion temperature as function of radius and the density of the imploded DT fuel that has stagnated.

pressure of the electrons, which are fermions—particles with half integer spin obeying Fermi-Dirac statistics. The specific energy required for compression of Fermi degenerate fuel is approximately given by $\epsilon_{\text{Fermi}} (\text{J/g}) \sim 3 \times 10^5 \rho^{2/3}$ (where the units for ρ are g/cm^3). Thus, the energy to compress 750 μg of Fermi degenerate DT to 1000 g/cm^3 , the nominal peak density of the cold but dense fuel is $\sim 1000 \text{ g/cm}^3$, is approximately ~ 22.5 kJ or about 6% of that required to heat an equivalent mass to 5 keV (Sinars et al., 2020; Campbell and Hogan, 1999).

This simple energetics argument explains why the implosion strategy commonly used in the majority of high gain laser driven ICF target designs has been adopted. Recognizing the overall inefficiency in which the energy from ICF laser drivers is coupled to the fusion fuel, in order to maximize the mass of DT that can be imploded and to minimize the size (and cost) of the laser, the implosion dynamics must be chosen to both ignite the fuel and to compress sufficient mass so that significant energy release is possible. A nominal high gain LDD target, pulse shape and fuel density and ion temperature radial profile at stagnation are shown in Fig. 3. For a given target design, this near isobaric imploded fuel assembly is achieved through temporal shaping of the laser pulse. As shown in the figure, the implosion is designed so that a small fraction ($< \sim 5\%$) of the fuel mass is compressed and heated to satisfy the generalized Lawson criteria to ignite and to propagate a fusion “burn wave” into the majority of the fuel, which as described above, must be maintained as close as possible to Fermi degeneracy. Target and driver temporal pulse format are designed so that the partition of the implosion energy between the igniting hot spot and cold fuel is approximately equal. The surrounding cold fuel confines and even tamps the igniting central hot spot and provides the majority of fusion fuel that enables high yield and gain (Sinars et al., 2020; Campbell and Hogan, 1999; Lindl, 1998; Betti and Hurricane, 2016; Miller et al., 2004).

Achieving ignition conditions using the 1.8 MJ NIF laser, where unambiguous alpha particle energy deposition results in a substantial ion temperature increases over that achieved by the laser alone, requires a neutron yield from an irradiated indirect drive target of 250 kJ of neutrons, or 10^{17} neutrons. Until November 2020, the largest yield achieved on NIF was in 2017 when 55 kJ of fusion yield was produced. Recently, however, LLNL achieved a record neutron yield of 100 kJ from a HYBRID-E target shot on November 1 and 100 kJ from an I-raum target shot on November 22. These results are shown in Fig. 4.

The main reasons for the improvement in yield was increasing the X-ray energy coupled to the fuel. Additionally, much improved diagnostics are allowing for improved shell asymmetry measurements, hotspot mix imaging, and measuring fuel configuration at stagnation (Herrman, 2020). These improved diagnostics that enable three dimensional measurements of the imploded fuel configuration will be critical to achieve better target designs and increased fusion performance.

One of the most important attributes of a particular target design for IFPP is its gain—the multiplication of the driver energy times the gain gives the fusion yield of each shot. Multiplying by the repetition rate provides the fusion power for the IFPP. Fusion gains for a specific class of indirect drive targets is given on the right side of Fig. 5.

The DOE is considering increasing the energy of the NIF to try to achieve ignition and gain and to develop robust models of the ignition and gain parameter space for various other options for IFPP including LDD, HIDD, and HIID.

Direct drive using solid state lasers

LDD involves using highly uniform laser beams that each fully engulf the cryogenic fusion capsule as shown in Fig. 6. The driver energy required to create high gain from direct drive can be substantially smaller than that required for indirect drive high gain. The number of beams required for sufficiently uniform illumination is based on the necessary RMS amplitude which has been found to



Fig. 4 NIF Indirect Drive targets achieving record gains in 2017 and 2020 (Herrman, 2020).

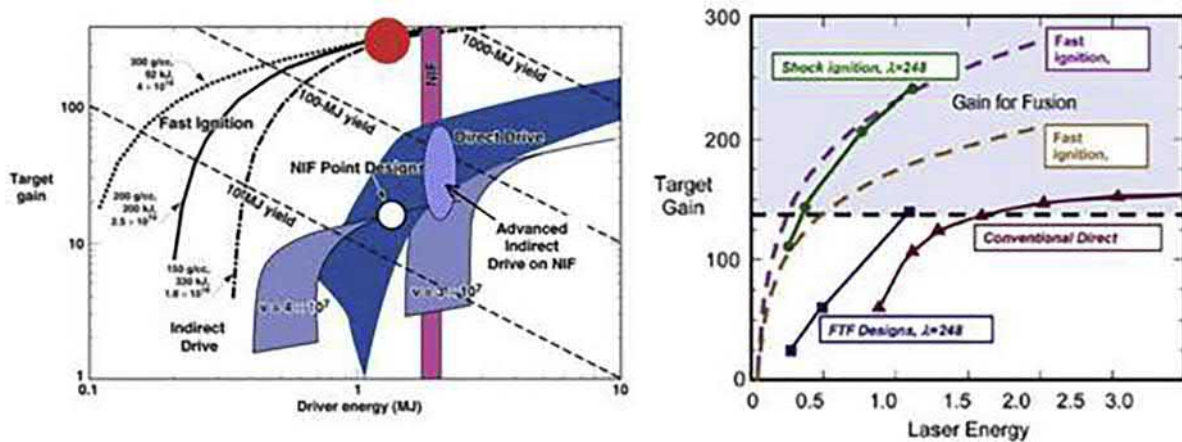


Fig. 5 On the right, several target gain curves for indirect and direct drive are shown with varying implosion velocities. A gain of 60 is achieved at 2 MJ. The red dot is a prediction for what is called a fast ignition target. A fast ignition target relies on a special injection of very short pulse duration energy into the target fuel once the target achieves compression. On the left, target gain for laser direct drive is shown in triangles with 300 km/s implosion velocity. A gain of 60 needing less than 1 MJ driver energy would suffice for DD IFE (NRC, IFE, 2013a,b; Lindl, 2003).

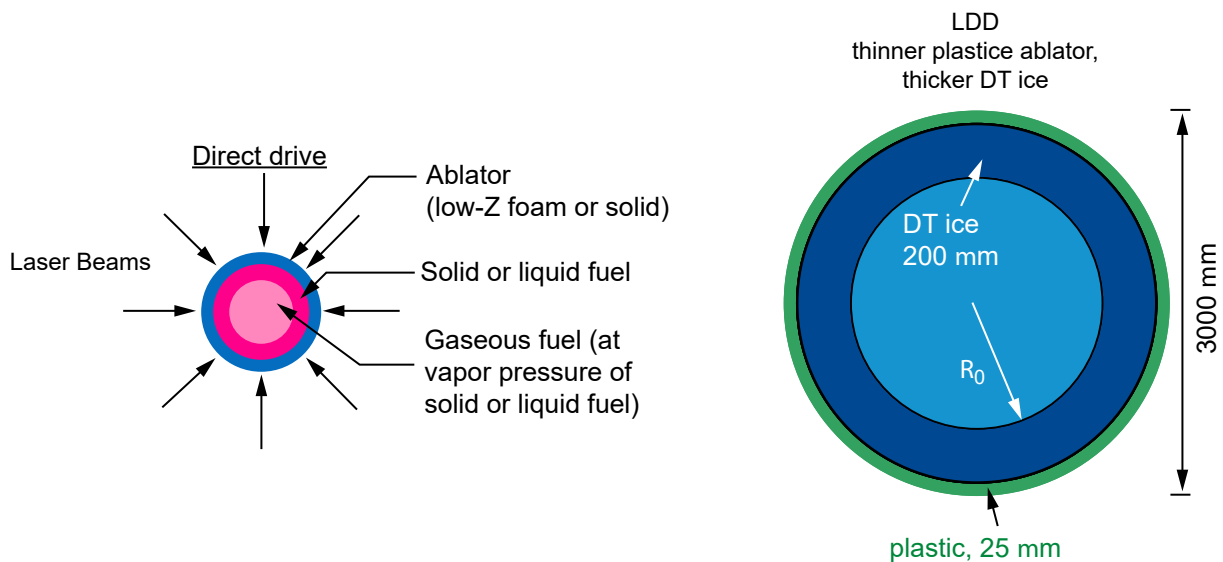


Fig. 6 To the left, the concept for a Direct Drive capsule is shown. (DOE, 1999) On the right is the capsule design that would be driven to gain conditions with a 1.8 MJ laser driver.

Table 1 Quad arrangement by cone angle to shift from indirect to direct drive experimentation on the NIF. A total of 12 quads would need to be re-directed from the 30 and 50 degrees cone angles to the 74.85 degrees cone angle. The NIF chamber has complete symmetry between the upper and lower hemispheres.

Cone	# Ports	Indirect drive layout	Direct drive layout
1	4	23.5°	23.5°
2	4	30°	
3	8	46.5°	46.5°
4	8	50°	
5	12		74.85°

be 0.4–0.6% during the initial to final drive to ignition and gain conditions requiring a focus ratio (beam size to target radius) from less than 1.0–1.8 at the smallest radius.

For LDD, the concentrated laser energy flux generates pressure on the exterior of the target that implodes the fuel to the conditions required for fusion (Betti and Hurricane, 2016). The pressure is produced by the rapid ablation of the outer target surface by a high-intensity laser pulse (see Fig. 6, left side). The peak laser intensity is typically 10^{15} W/cm². The capsule design for a LDD capsule driven by 1.8 MJ of energy to conditions of ignition and gain is shown on the right side of Fig. 6.

The LLE research goal on OMEGA is to achieve the highest possible scaled condition for ignition and to understand degradation mechanisms that limit the implosion performance. This will be done both with spherical illumination for which OMEGA is optimized and then in the polar geometry that exploits the existing illumination geometry of the NIF (Collins et al., 2012). OMEGA energy and power (30 kJ and 30 TW) are insufficient to achieve ignition conditions in the imploded fuel so both simulations and modeling are employed to scale OMEGA results to NIF energies (2.15 MJ and 500 TW). The LDD research on the NIF is focused on laser–target coupling and laser–plasma interaction (LPI) physics in ignition-scale plasmas. Implosions with limited convergence (less than 10), given the present capabilities of the NIF, are also a component of the research. The ability to conduct research over a large range of energy, power and scale size, using both Omega and the NIF, and the modeling and simulation tools developed at the Laboratory for Laser Energetics (LLE) are major positive aspects of LDD research that reduce the risk in scaling from OMEGA to megajoule-class lasers. The goal over the next decade is to develop and demonstrate a convincing physics case for modifying the NIF to enable burning-plasma experiments to be conducted, producing greater than 1 MJ fusion yields from a direct-drive target illuminated in polar geometry (Collins et al., 2012). An additional longer-term goal is to develop a quantitative understanding of LPI physics that limit the parameter space for all laser-driven ICF approaches and to develop and demonstrate advanced laser technology that can expand the parameter space and potentially lead to multi-megajoule yields and high fusion energy gain at MJ-scale laser energies (Campbell, 2020, p. 5).

The NIF was designed and built in such a way that it would be possible to convert the irradiation scheme from indirect drive to direct drive should the decision be made to pursue LDD on the NIF (Tobin et al., 1996). This would involve re-directing 12 quads of beams from the polar angles to around the equator of the NIF chamber. Table 1 shows what the beam layout would be for direct drive experimentation on the NIF. The 12 quads would have re-positioned with six just above the equator of the chamber and six just below to “fill in” the equatorial region of driver beams.

Direct drive gain curves are shown in Fig. 5. This shows that while a 1.8 MJ indirect drive laser target would achieve a gain of about 60 for a fusion gain of 120 MJ, a 1.8 MJ direct drive laser target (such as in Fig. 6) would achieve a gain of about 150 for a fusion gain of 270 MJ.

Direct drive using excimer lasers

It is well known that improved laser-target coupling such as reducing/eliminating electron pre-heat for driving an inertial fusion target and other instabilities is made possible by a shorter wavelength driver. This has generated great interest to examine so-called excimer lasers (more properly called an exciplex laser)—namely the KrF laser which emits at 248 nm and the ArF laser which emits at 193 nm.

The excimer laser was invented by Nikolai Basov et al. at the Lebedev Physical Institute in Moscow (Basov et al., 1970). Exciplex lasers work by having the mixture of noble gas (in this case Kr or Ar) and fluoride absorb energy from an electron beam. The energy causes the noble gas to react with the fluoride producing a temporary atomic “complex” in an excited state (for example, $2\text{Kr} + \text{F}_2 \rightarrow 2\text{KrF}$) that can undergo spontaneous and/or stimulated emission, reducing its energy state back to individual atoms of noble gas and fluoride molecules.

KrF lasers have a very short energy storage time (2 ns) and low saturation intensity (2 MW/cm^2). To extract a useful amount of energy from each amplifier, the pulse length therefore needs to be much longer than the energy storage time. The amplifier is energized by microsecond duration electron beams and to achieve the ~ 10 ns laser pulse required for IFE, angular multiplexing, in which a number of short duration laser pulses, whose temporal format is matched to the IFE target, are amplified and then optically combined has been developed by Naval Research Laboratory scientists to achieve the needed on-target power (Obenschain et al., 2020).

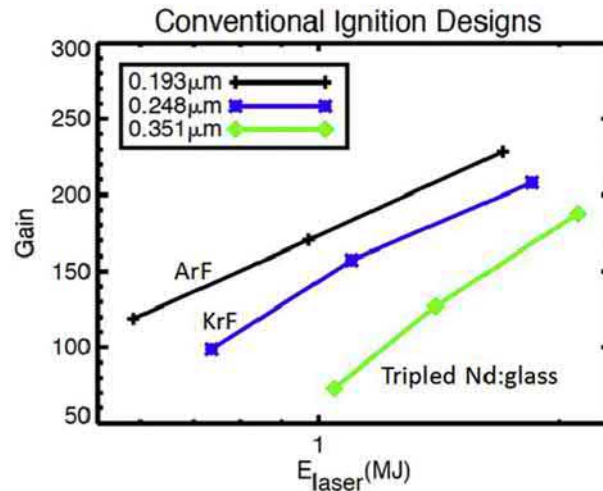


Fig. 7 One-dimensional hydrocode simulations of target energy gains as a function of laser energy for conventional direct drive targets. The simulations employed zooming of the laser focal diameter (Obenschain et al., 2020, p. 5).

A KrF laser for an IFE power plant has a number of potential advantages over a Nd-solid state laser: (1) the shorter wavelength reduces laser-plasma instabilities and maximizes absorption and ablation pressure; (2) electron beams can easily operate at a high pulse rate; (3) a KrF laser on target intensity can be made very uniform using the technique of induced spatial incoherence with many overlapped beams and (4) and since the far field image as the beam leaves the amplifier is projected onto the target, it is possible to “zoom” the beam as the direct drive target implodes such that an ever-smaller diameter beam continues to drive the compression significantly reducing the laser energy required (Obenschain et al., 2020). The overall laser efficiency of KrF, however, is only about 7% such that higher gain targets (e.g. gain of 140) is required for IFE economics compared to, for example, solid state lasers (e.g. gain of 60 at 16% efficiency).

Recent studies indicate that an ArF laser (193 nm) may have substantial advantages over a KrF laser (248 nm). At the shorter ArF wavelength the following are achieved; the greatest pressure, the highest ablation rate, the lowest electron temperature, the lowest susceptibility to laser plasma interactions (LPI), and the highest bandwidth (10) which should avoid cross beam energy transfer (CBET) (Obenschain et al., 2020).

Fig. 7 shows how these benefits aid in the gain of direct drive targets comparing ArF, KrF, and solid-state Nd where the frequency has been tripled to 351 nm in one-dimensional hydrocode simulations. For example, while tripled Nd glass would achieve again of 75 at 1 MJ laser energy, KrF would achieve a gain of 150 and ArF a gain of 175. This shows that the ArF option, with the lowest driver energy for the same modest net electrical power, may be the least expensive and also the lowest risk option for a IFPP using an excimer laser.

While the benefits of KrF and ArF have been more established in recent years, there is no current plan for excimer laser research to be expanded at this time. These comparisons need much more peer review to establish excimer lasers as a potential viable candidate for IFE. It is expected that the achievement of ignition and gain on the NIF will provide more key confirming data as to the possible role of this driver technology for IFPP.

Indirect drive using heavy ion beams

As early as 1980 the proposal to use heavy ions (HI) to illuminate an ICF target and produce an efficient implosion fuel burning with high gains using DT fuel was considered. From 1981 to 1985 along with the design of the HIBALL I-II power plant (Böhne et al., 1982) a significant number of other power plant studies can be found in the literature: HIBLIC-I (Yamaki et al., 1985), HYLIFE I-II (Moir et al., 1994), OSIRIS (Meier et al., 1992), and HIDIF (Hofmann and Plass, 1998).

One key seminal scientist in proposing Heavy Ion Fusion, Al Maschke (Martin, 1996) started by proposing research in high-energy proton synchrotrons early in the 1960s. Later it was understood that the range of those protons in a fusion target was too long and so Maschke switched to heavy ions of energy about 100 MeV/u that have shorter range and had the possibility to design efficient drivers exploiting traditional linear accelerators. An advantage of HIF is that the coupling of the driver energy does not involve the excitation of plasma instabilities which remains a major issue with all laser approaches that presently limits the laser-target parameter space for achieving high fusion gain.

The mechanisms for target implosions are essentially the same as in the case of lasers but different when considering the energy deposition in the target as seen in Fig. 8. The interaction of heavy ions with matter (ablation of target materials) is quite well understood, guided by Coulomb interaction with screening effects due to the plasma conditions of the matter during its interaction. Mehlhorn explains how to incorporate “finite temperature effects by scaling the relevant bound electron parameters with the degree of material ionization as well as by including the free-electron stopping power. In this article he discusses both the phenomenon of range shortening and range lengthening in heated material. The calculations indicate that the minimum ion range in heated

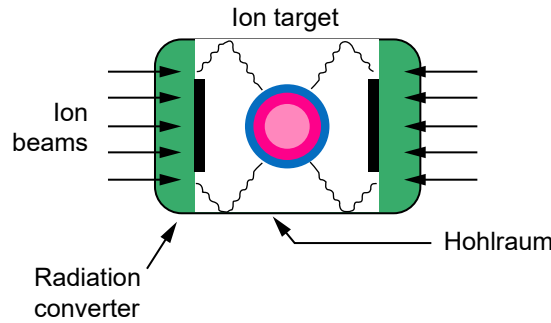


Fig. 8 Heavy ion beam indirect drive target; ion beams shine on the radiation convertor; X-rays from the radiation convertors fill the inside of the hohlraum and heat the capsule (DOE, 1999).

material is approximately one-half that in cold dense targets, independent of the identities of the projectile ion and the host material" (Mehlhorn, 1981). The ions effectively deposit all of their energy in the plasma and the penetration of the ions at the relevant energies, charge, and mass is in the range of hundreds of microns (Kothe et al., 1990).

For the direct drive approach, we can compare a heavy ion driver and laser driver from the point of view of coupling energy to the target that is the efficiency in which incident driver energy is converted into kinetic energy of the imploding target. Heavy ions can have a total efficiency up to 25–30% while lasers will be in the range (at best) of 8–10%.

If we indirect drive for target implosion for HIF and consider the losses occurring in the conversion of laser energy to X-rays in the case of the hohlraum target and the coupling of X-ray energy to the capsule which depends on the area ratio of the hohlraum to the capsule, the integral efficiency will be $\sim 2\text{--}4\%$. This low efficiency motivates the option of Heavy Ion Direct Drive. It is true that the number of beams is limited for Heavy Ions because of the architecture of final focusing of accelerators and the potential hydrodynamic instabilities created by a non-uniform illumination that can destroy the appropriate implosion efficiency limiting target gain. Low mode non-uniformities can also result in the inefficiency of converting the implosion kinetic energy into pressure at the fuel stagnation. As discussed earlier, for an acceptable recirculating energy fraction, economics suggests that a value of driver efficiency multiplied by the target gain should be around 10. For a heavy ion driver efficiency in the 30–40% range, this suggests the heavy ion fusion target gain need only be 30–40.

Research in Heavy Ion Fusion has been experimentally devoted to the area of accelerator technology, and very little to target implosion studies or associated physics except basic research in High Energy Density Physics (HEDP) and interactions in order to achieve the magnitudes required for the power plant application. The most explored case for considering heavy ions as the driver for an ignition facility is the case of the European Heavy Ion Driven Inertial Fusion Facility (HIDIF) but laser facilities such as the National Ignition Facility (NIF) are more adequate for this step and heavy ions are better considered for a power plant (Hofmann and Plass, 1998).

Of the two techniques to build heavy ion accelerators the induction linac is favored today over the radio-frequency technique due to high peak currents possible and proven reliability and efficiency as well as being able to achieve short pulses and support multiple beams.

Different types of heavy ion targets have been proposed since the 1980s and 1990s. Because of the large effort in X-ray driven targets and the common capsule hydrodynamics indirectly driven ion targets have received the most attention as shown in Fig. 9 (Callahan-Miller et al., 1999; Tabak et al., 1998; Basko and Meyer-ter-Vehn, 1993).

Standard hohlraum-to-capsule radius ratio design (HCR – 2.1) Gain 60 @ 6 MJ

M. Tabak and D. Callahan-Miller, Phys. Plasmas 5 (1998)



(Drawn at same scale)

Close-coupled design (HCR – 1.6) Gain 130 at 3.3 MJ

D. Callahan-Miller, and M. Tabak, Phys. Plasmas 7 (2000)



Hohlraum is smaller – requires smaller beam spots

Fig. 9 Two heavy-ion fusion indirect drive target designs have evolved (references in figure).

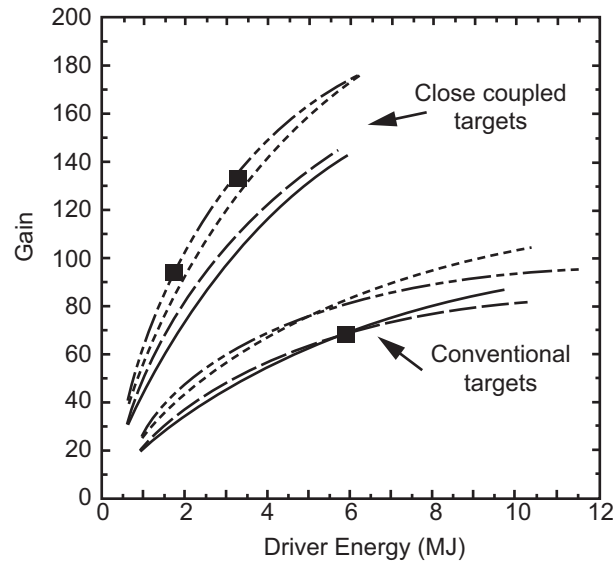


Fig. 10 Gain curves for distributed radiator targets. The lower set of curves is for the conventional case-to-capsule ratio while the upper set is for the close-coupled design. For each set, the solid curve is 250 eV drive and fixed ion range, the dashed curve is 250 eV drive and varying ion drive, and the dotted curve is 240 eV drive and fixed ion range, and the dot-dash curve is 240 eV drive and varying ion range. The squares represent the three integrated LASNEX calculation point designs (Callahan-Miller and Tabak, 2000, p. 2083).

Full study of these targets, both two-sided radiator and distributed radiator, consider a driver energy range of 4–10 MJ and Pb + ions from 3 to 10 GeV. As discussed by Callahan-Miller and Tabak, for the original distributed-radiator-target, it was assumed that all the ion current was carried by only two beams, one on each side, with “top hat” beam density distributions. Quadrupoles cannot focus 3 MJ of ion beam energy in a single beam. Since more than two beams are then needed, all of the beams need to enter the hohlraum at an angle relative to the hohlraum centerline axis to allow sufficient space for final focusing magnets. Chamber calculations also show that a top hat density distribution is not realistic for conventional ballistic chamber transport. The beam distribution was important originally in the distributed radiator target because the central region of the hohlraum was a beam block. For a top hat distribution, 25% of the beam energy hit this beam block but was wasted. With a Gaussian distribution, 50–55% of the beam energy would hit the beam block. If two Gaussian beams were used, the energy required to drive the original target design would therefore increase from 6.5 MJ to more than 10 MJ (Callahan-Miller and Tabak, 1999, pp. 1–2).

In Fig. 10, the gains are shown for these designs. The baseline design allows large heavy ion beam spot sizes but requires more driver energy to achieve a gain of 55 for 6.7 MJ of beam energy. The close coupled design requires smaller ion beam focal spots but produces more than adequate gain at low driver energy giving a gain 130 from 3.3 MJ of beam energy. To achieve closely coupled results, the hohlraum dimensions were reduced by 27% while driving the same capsule. This reduced the beam energy required to 3.27 MJ. This close coupled target can also be scaled down for an Engineering Test Facility. LASNEX calculations predict a gain of 94 can be achieved from 1.75 MJ of beam energy (Callahan-Miller and Tabak, 2001, 1998).

One reason for significant improvement over prior gain curves 20 years earlier is the fact that now instead of 12 beams each delivering 417 kJ in 100 ns for 5 MJ total energy and for a gain of 87, 120 beams were now delivering nearly 7 MJ of energy using six different pulse durations and two different ion energies (3.3 and 4 GeV) for a gain of 55. This is shown in Fig. 11.

The national USDOE strategy that was developed in the last decade was to achieve ignition and yield on the NIF and then to conduct research evaluating heavy ion indirect drive targets from which to select the optimum development path for the heavy ion accelerator. The lack of ignition to date on NIF has reduced the urgency and motivation for a robust HIF effort. However, the Neutralized Drift Compression Experiment II (NDCX-II) at Lawrence Berkeley Laboratory was completed and is conducting research relevant to HI IFE by heating targets to eV conditions for the study of warm dark matter. The experiments in High Energy Density Physics in SIS-18 at GSI Darmstadt (Germany) and those at ITEP (Russia) also have been relevant for Heavy Ion Fusion (Hofmann, 2018).

High intensity heavy ion accelerators are being developed for nuclear structure and other areas of research—namely the FAIR facility at GSI in Germany (Schoenberg et al., 2020), the Relativistic Heavy Ion Collider at Brookhaven National Laboratory in the United States (Ozaki and Roser, 2015), and the HIAF in Lanzhou, China (Zhao et al., 2014). These developments will benefit future HIF programs by advancing accelerator technology and resolving issues such as activation of the accelerator structure by uncontrolled heavy ion beam loss and “loss-induced” degradation of the vacuum by gas desorption (Hofmann, 2018).

Direct drive using heavy ion beams

While as mentioned above heavy ion beam drivers have been primarily associated with indirect drive, there has also been limited consideration for direct drive concepts. In any target design, it is theoretically well known that the most efficient implosions would

Block	No. of Beams	Power, TW	Pulse width, ns	Energy, MJ
A (foot)	16	70	6.5	0.46
B (foot)	16	20	38.3	0.77
C (foot)	16	53	10.1	0.54
D (main)	24	120	13.7	1.64
E (main)	48	388	9.3	3.61

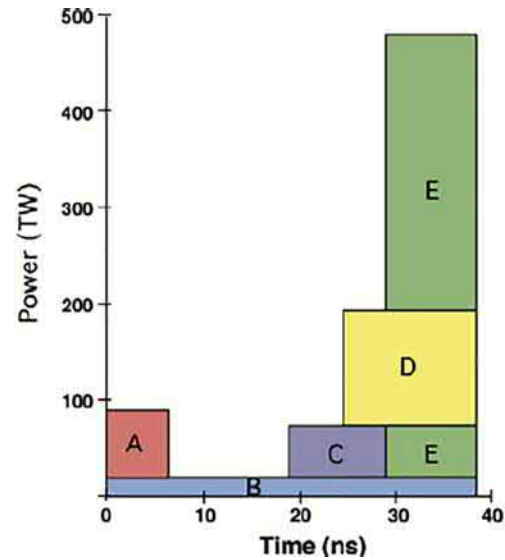


Fig. 11 Heavy Ion Beam timing and energies for the Robust Point Design-2002 IFE Power Plant (Yu et al., 2003, p. 2).

be done at the peak of rocket efficiency with a low-Z ablator having small ionization energy. For heavy-ion direct-drive target coupling efficiency, there are two areas that may be improved upon according to recent investigations: first, decoupling the ion-beam deposition away from the ablation front during the implosion; and second, beam deposition non-uniformity and resulting RT instability (Logan et al., 2008).

As shown in Fig. 12, LASNEX calculations indicate that for this direct drive target and heavy ion beam pulse target coupling efficiency in direct drive is 16% with a CH:DT ablator, and 18% for a pure DT ablator. These impressive results are the highest overall coupling efficiencies published for 1-D heavy ion capsule implosion calculations that ignite with a total beam input energy of only 1 MJ. The resulting implosion velocity (4.45×10^7 cm/s) is higher than the minimum needed for ignition (3.68×10^7 cm/s). In the future, we will consider driving low-Z NIF-size capsules with heavy ion beams assuming we can focus heavy-ion beams to 1 mm radius spot size, with heavier krypton ions for the lower ranges required, and with short-focal-length copper final-focus magnets. If successful designs emerge along with NIF achieving ignition, the prospects for heavy-ion fusion development would look more promising since it would suggest a gain of 100 at just 200 kJ total drive energy. Further if we can optimize coupling efficiencies to 25% for larger mass targets, then igniting large fuel assembly energies with 1 MJ might be possible with 4 MJ of beam drive energy. Such large fuel assemblies with T-lean fuel (Tabak, 1996; Atzeni and Ciampi, 1997) (DD fuel with a small inner DT spark-plug) at compressed $\rho \sim 10$ g/cm³ would self-breed tritium without external blankets, and internally capture the neutron energy into mostly plasma energy for direct conversion. These would create great interest in heavy ion fusion (Logan, Perkins, and Barnard, 2008, p. 5).

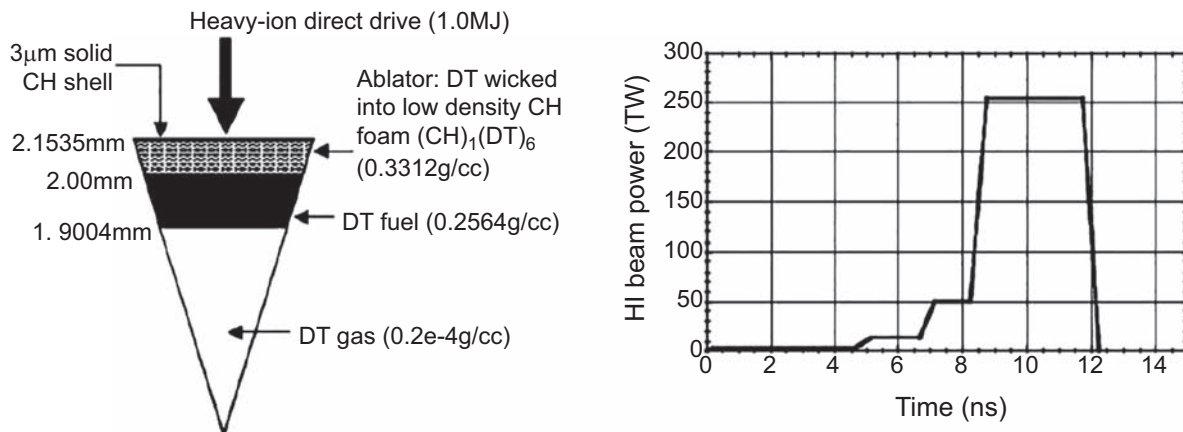


Fig. 12 A "pie" diagram of the target construction for the LASNEX calculation (a), and the 50 MeV argon ion-beam power pulse shape used (b). This case gives a fusion gain of 50 for 1 MJ of beam drive, assuming the incident beam profile zooms in a factor of 2 in radius (from 2 to 1 mm) during the drive.

If ignition and gain are achieved on NIF then motivation for further studies and research of DD-HIIF would certainly grow.

Direct drive using pulse power inertial fusion

The main approach to producing ignition and gain using pulsed power is Magnetized Liner Inertial Fusion (MagLIF). In this approach pulsed-power-driven IFE would utilize ≥ 50 MA of current from a pulsed power accelerator to generate sufficiently high magnetic field pressures to compress and heat magnetized, pre-heated fusion fuel contained in a cylindrical target to ignition conditions. The laser energy needed for preheating the fuel is only about 50 kJ. An external magnetic field of ~ 100 kG is also used to insulate the fuel as it is imploded. The magnetic Reynolds number is large trapping the internal B field so that it is significantly increased to > 10 MG. This field significantly reduced radial heat conduction and can trap the 3.5 MeV alpha particles that then deposit their energy and ignite the DT fusion fuel. The pulsed-power approach has relatively low-cost and high-efficiency driver technology that appears to be scalable in a straightforward way to the peak power and total energy presently estimated to be needed for IFE (National Research Council, 2013a,b). Experiments on the NIF to explore laser heating and other integral experiments using laser compression rather than magnetic pressure for the MagLIF program are ongoing (Barnak et al., 2016).

The primary conceptual approach to achieving pulsed-power IFE, Magnetized Liner Inertial Fusion (MagLIF), is a direct-drive approach—that is, fuel compression and subsequent heating (laser heating is used to preheat the fuel) are driven directly by magnetic pressure. This approach offers the potential benefits of a relatively simple cylindrical target geometry and highly efficient delivery of driver energy to fuel implosion and heating. However, there is considerable uncertainty (i.e., technical risk) surrounding all aspects of this approach owing to a limited experimental data on target physics and ignition and a lack of in-depth design studies on inertial fusion reactors at the proposed multi-GJ yield and ~ 0.1 Hz repetition rate called for by advocates. In addition to MagLIF, there are other promising approaches to pulsed power fusion energy, including one called magnetized target fusion (MTF). While MagLIF operates on the 100-ns timescale, is ~ 1 cm in size, and involves open magnetic field lines, MTF operates on a ~ 1 μ s timescale, is tens of centimeters in size, and involves closed (field-reversed) magnetic field lines. A pulsed-power fusion reactor system would be very different from both laser and heavy-ion fusion systems. As such, the technological or economic failure modes are likely to be very different (National Research Council, 2013a,b).

Since 2018, simulation and experiments at the Z facility at Sandia produced 1 kJ of DT fusion yield using 21 MA of current to produce a 15 T magnetic field where results agreed with LASNEX 2-D predictions. Sandia's goal is to demonstrate 2 kJ and 4 kJ DT fusion yields using 20 T and 25 T magnetic fields generated by 20–21 MA of current by 2022 (Sinars, 2020).

Direct drive pulsed power inertial fusion remains a viable candidate for future IFPPs. In addition to the physics challenges, the driver, unlike lasers, must be physically connected to the fusion target. Concepts such as the “recycling transmission line” or RTL have been developed but significant engineering and operational challenges remain.

IFE chamber technology

The chamber of an IFPP that confines the fusion target yields has an advantage over most magnetic fusion concepts in that a “first wall” (first surface that target emissions make contact with) can be a replenishable material or perhaps a material that can be easily periodically replaced. This opens up the design space to allow a greater range of options for the structural wall that holds vacuum and confines the repeated fusion yield. Along with this advantage, however, comes a key difference between the magnetic fusion neutron loading and the inertial fusion neutron loading of chamber components.

Inertial Confinement Fusion has had a very large number of proposals for the design of Power Plants since the early 1970s. More recently proposals include: LIFE (Latkowski et al., 2011), HiPER (Perlado et al., 2011), KOYO-F (Norimatsu et al., 2009), FALCON (Ogawa et al., 2008), and a large effort in the United States due to the High Average Power Laser (HAPL) and the ARIES initiatives (see <http://aries.ucsd.edu/ARIES/>).

The chamber of an IFPP is the location of the implosion-ignition-burnup of the ICF target at its center. The chamber could be filled with a protecting gas at a low density such as 5 $\mu\text{g}/\text{cm}^3$ or alternatively be designed to be at a much lower ~ 1 μTorr vacuum. The first surface of the chamber wall (the so-called First Wall) receives the radiation in the form of target ions, X-rays and neutrons. The chamber must also host the system to extract the neutron energy as heat and at the same time support materials to cause tritium to be produced which is called “breeding” tritium.

The degrading effects of X-ray radiation and target debris ions can be mitigated or eliminated through the use of liquid layers adhered to the first wall or by a chamber gas to absorb these target emissions. For examples, the LIFE IFPP employs Xe gas, the HIBALL and KOYO IPFFs employ thin liquid layers, while the HYLIFE-II IPFF uses a thick liquid layer. These are in contrast to non-protected first wall chambers such as HiPER.

We now describe in more detail, as examples of alternative chamber technologies, the IFPP chamber technologies for the HiPER concept (Juarez et al., 2013), the LIFE concept (Latkowski et al., 2011), the HYLIFE-II concept (Moir et al., 1994) and the KOYO-F concept (Norimatsu et al., 2009).

The HiPER concept employs a dry First Wall (FW) made of 1-cm of Ferritic-Martensitic Steel (EUROFER97) with a 1-mm coating of nanostructured W. The liquid metal $\text{Li}_{17}\text{Pb}_{83}$ is the coolant. The breeding blanket is also made of EUROFER97 and also with $\text{Li}_{17}\text{Pb}_{83}$ circulating inside. The HiPER concept also calls for a second heat removal system using He to remove heat from the FW while maintaining $\text{Li}_{17}\text{Pb}_{83}$ in the main blanket. There are two heat removal approaches using the Self-cooled Lithium Lead (SCLL) concepts. One fully integrates the He and SCLL into the Integrated First Wall Blanket (IFWB) while the other keeps these two systems separate in the Separated First Wall Blanket (SFWB) arrangement, as shown in Fig. 4.

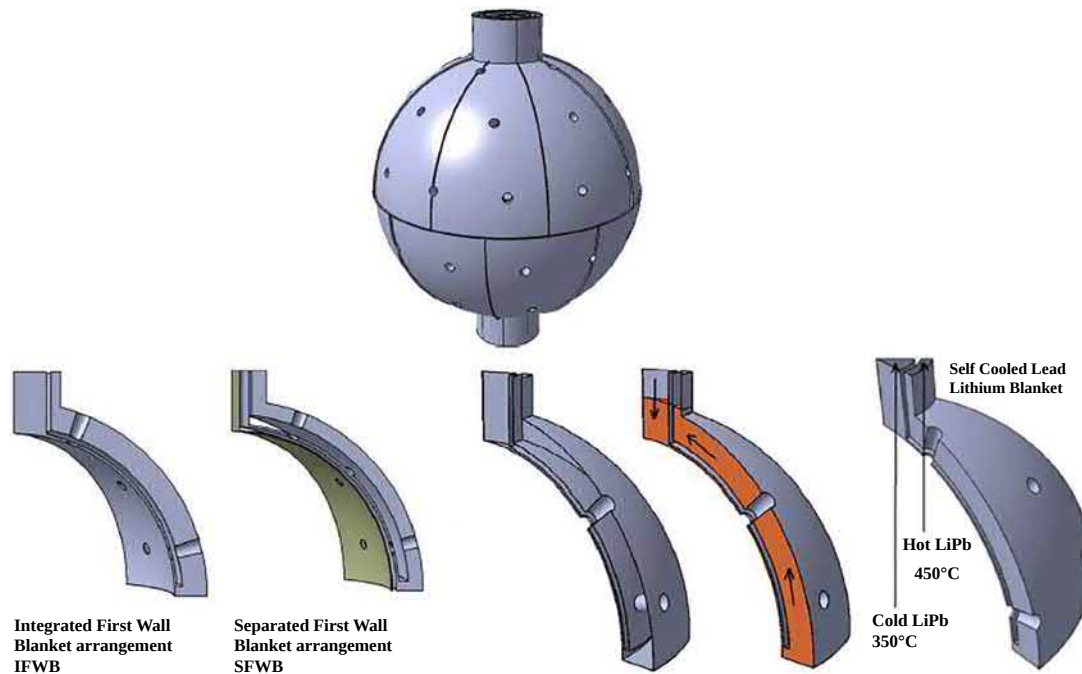


Fig. 13 In the HiPER concept, both an Integrated First Wall Blanket and a Separated First Wall Blanket arrangement is under consideration (Manas, 2013). The chamber has 48 beam ports to support the Direct Drive approach.

The chamber structure is made up of two hemispheres with an internal radius of 6.5 m with each hemisphere composed of eight sections as shown in Fig. 13. The chamber has 48 laser beam penetrations each 40 cm in diameter. As shown in Fig. 4, each blanket module has a thin 8-cm layer of LiPb coolant closest to the chamber which flows in cool $\text{Li}_{17}\text{Pb}_{83}$ leaving from the heat exchangers. Just behind this layer there is a second layer 42 cm thick that flows out of the chamber to the heat exchangers. Together these $\text{Li}_{17}\text{Pb}_{83}$ coolant flows not only remove neutron heating but ensures a tritium breeding ratio (TBR) of at least 1.10. In a study to select a thin (42 cm) or thick (67 cm) layer for the second layer of the blanket, the thinner option was chosen to reduce weight and hazardous nuclide production. However, this also results in the need to enrich the Li^6 concentration to about 35% in the blanket and increase the shielding protection for the vacuum vessel by adding graphite shield behind the blanket enlarging the overall structure (Juarez, 2013). Pure Li was avoided as a coolant to reduce the safety risk.

The LIFE chamber was also designed in a modular fashion and constructed from eight identical sections as shown in Fig. 14 below, configured into a sphere with a radius of 3.4 m. Inside each section, in the region nearest the chamber center, there are rows of 10-cm diameter steel tubes through which flow the liquid lithium, the selected coolant for the FW. The coolant carries away the heat from neutron deposition and from the ions and X-rays not absorbed by the chamber FW Xe gas protection. The flow is very rapid through the pipes and they create a “skin cooling” effect. This is shown in the left side of Fig. 14. The coolant then is turned around at the bottom of the blanket and then flows up the through the bulk of the section, the breeding blanket region, at a much slower speed. It is here the main tritium breeding occurs.

The coolant enters into the LIFE first wall coolant tubes at a temperature of 550 °C and arrives at bottom of each section where it turns to enter the blanket region at 600 °C. It is further heated during its pass in the bulk of the blanket with a design exit point of 800 °C. Steel has a limit of 600 °C and so insulating panels of W must be used to allow the lithium to rise to its maximum design temperature. Refractory coatings such as W are compatible to 1300 °C. A total of 48 penetrations for lasers go through the chamber wall system comprising a 3% solid angle. Notably, there are no beam tubes attached to the chamber system since an exterior wall outside the chamber region is the vacuum barrier.

The KOYO-F IFPP concept (Norimatsu et al., 2009) is representative of the thin protection or wetted FW design options similar to the HIBALL (Badger et al., 1981) and Prometheus-L (Waganer et al., 1992) IFPP concepts. The main idea is to have FW structures where a thin sacrificial layer is formed on its face thick enough to absorb the X-ray and debris/ion target emissions and be able to be replenished before the next shot. The KOYO-F IFPP concept uses $\text{Li}_{17}\text{Pb}_{83}$ as this sacrificial layer on the first wall.

The KOYO concept employs an unusual technique to protect the ferritic steel first wall with this wetted surface. The chamber first wall is formed by a structure of integrated panels formed into boxes that allow the hot liquid $\text{Li}_{17}\text{Pb}_{83}$ to flow from the top of the chamber toward the bottom in a cascade of flows where the overflow of certain boxes spills out and coats the first surface. The lower the cascade falls in the chamber the wider the structure necessarily becomes to ensure continuous and uniform coating. The maximum diameter of the chamber is 3 m but is reduced from the top to the bottom of the chamber by the overall thickness of the cascade structure. Fig. 15 shows these details. The top of the chamber is formed in a conical shape that studies show will result

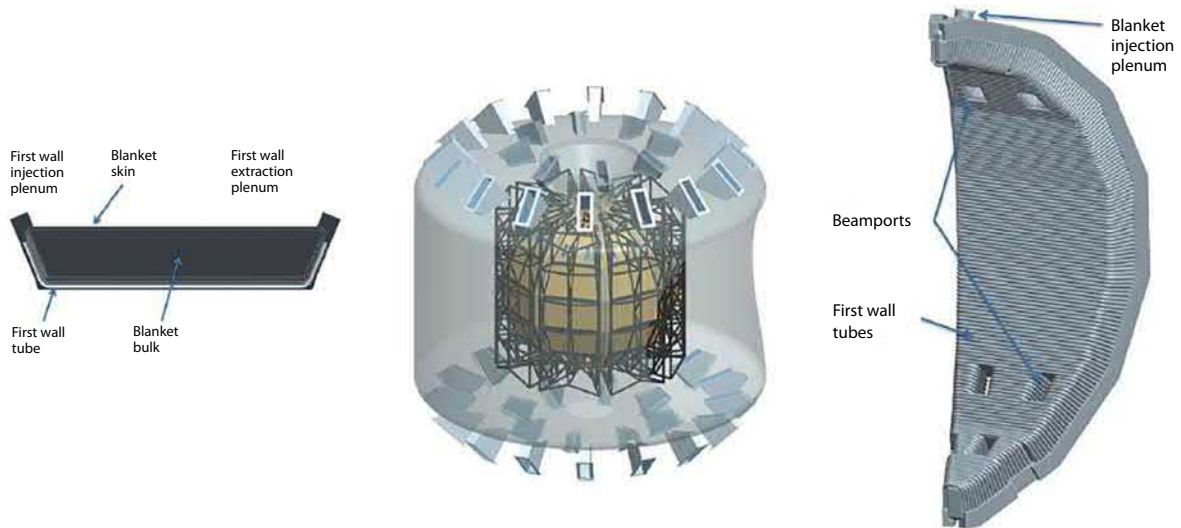


Fig. 14 (Left) The LIFE first wall is composed of steel tubes that are mounted to coolant plena on the sides of the blanket. (Center) The LIFE chamber consists of eight identical modules assembled into sections for transport to the engine bay. (Right) The LIFE blanket utilizes skin cooling to maintain structural material strength and corrosion resistance (Latkowski et al., 2011).

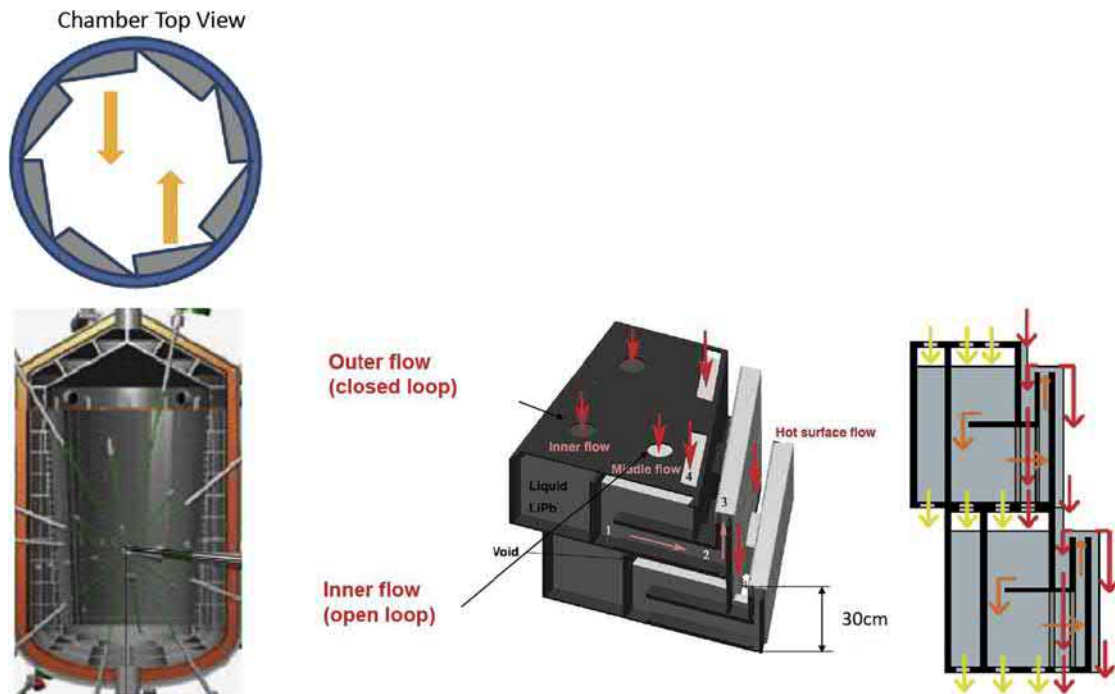


Fig. 15 (Left) Cross section of the 3-m diameter chamber shows the cascade structure thickening from the top to the bottom. (Center) Two views of the details of the cascade structure shows the interior flow of $\text{Li}_{17}\text{Pb}_{83}$ to and how the exterior surface is coated to 5-mm thickness. The height of each cascade small module is 30 cm that correspond to 0.25 s synchronized with the chamber repetition rate of 4 Hz. (Upper Right) Top chamber view shows the irregular pattern the cascade structure makes (Norimatsu et al., 2009).

in a layer of condensed $\text{Li}_{17}\text{Pb}_{83}$ to coat this surface as well. KOYO is designed to contain 300 MJ yields at a rate of 4 Hz in four separate chambers supported by a single driver (Norimatsu et al., 2009).

The IFPP concept HYLIFE-II is a thick FW protection scheme. This IFPP uses a material called FLiBe (Li_2BeF_4) as a coolant. This FLiBe is a molten salt that is a 2:1 combination of LiF and BeF_2 with a melting point of 459 °C. In this concept, liquid FLiBe coolant is formed in stable jets using nozzles in the top of the chamber that taken together provide a thick shield for the chamber FW yet still allow driver beam access and target injection to the chamber center. In Fig. 16, there is an image of molten FLiBe flowing and it a completely clear liquid when melted. The flowing FLiBe in the chamber also absorbs all X-rays and ionic debris emitted from



Fig. 16 Molten FLiBe flowing in a tube while being heated. This sample has a slightly green tint due to entrained uranium tetrafluoride.

the target as well as most of the neutron energy and therefore acts as the coolant in addition to protecting the first wall. The FLiBe also breeds the tritium in the chamber. Studies show that rapid enough condensation occurs to support shot rates exceeding 5 Hz in the presence of single 1 GJ fusion yields (Moir et al., 1992).

Fig. 17 shows pulsed flows and oscillating jets flows. Pulsed flow is used in the center region to create a slug of the FLiBe coolant. The role of this slug is to clear the beam and target injection path from the splash generated from the prior shot. The slug is created every 1/6th to 1/8th second and flows downward in a height of 1-m and about 60 cm by 60 cm on a side with a speed of 12–16 m/s. The oscillating jets are generated by nozzles oscillating at 8 Hz to support high chamber repetition rates. The oscillating jets are a neutronically relevant 70-cm thick. All working together this creates a pocket designed to let targets and beams be injected to the center of the chamber to create the micro-explosion and the FLiBe jets shield the components and also clear the chamber between shots, all the while producing tritium and generating electricity.

From an economics point of view, it is very desirable for the FW of a IFPP to last the plant lifetime. The lifetime of the first wall depends on the type of protection and design of the first wall. Research focused on ion-induced damage to the first wall for direct drive target ion emissions suggests that the first millimeter of the first wall will be severely damaged and need replacing within days of full power operation if there is not a protective gas, a wetted wall, or thick interior flowing solid or liquid blanket to attenuate the ions (Gonzalez-Arrabal et al., 2020). In a dry direct drive chamber where there is no protection for the first wall the key damage to the first wall and the limiting lifetime factor is ion bombardment that in a fairly short time embrittles the W coating protecting the steel (Garoz et al., 2016). The extent and rate of the damage depends on the pulse conditions such as yield, duration, repetition rate, and whether the surface experiences a thermal load (X-ray loading) concurrent with the pulse. Direct Drive/Dry-Non-Protected Wall studies (Gonzalez-Arrabal, 2020) indicate that the lifetime of the coarse-grained W protective FW layer in the presence of ion and X-ray exposure is: for 20 MJ per shot only 1000 shots; for 50 MJ/shot at 1 Hz only minutes; and at 150 MJ/shot at 10 Hz only seconds.

Studies have examined the use of nanostructured materials to be used (Andrievski and Khatchoyan, 2016) in extreme conditions and, in particular, columnar nanostructured W was proposed to replace coarse grained W as the protective 1-mm coating for the EUROFER97 FW (Gonzalez-Arrabal et al., 2014). A nanostructured material is where the grain size is on the nanometer scale size. There appears to be some resistance in nanostructured material to radiation effects as well as an increase in yield strength and lower ductile-to-brittle transition temperature (Zhang et al., 2009). The main aim of nano-structuring is to create a large density of Grain Boundaries (GBs), which will promote self-healing and delay blistering by increasing the effective area where light ions can be accommodated or by creating effective diffusion channels that allow the light species to escape. More research is needed to know the influence of the GBs in the transport of light species such as H or He. Migration energies of light species in GBs under realistic operational conditions need to be well understood since the retention of light species in GBs makes nanostructured materials useless for FW protection applications (Padmanabhan, 2001).

Engineered surfaces such as needles and foams may also offer a solution. Implementing an engineered surface greatly increases the surface area upon which the ions will be depositing. This reduces the concentration and delays the onset of effects. Experiments exposing samples to ion bombardment performed on needle-like carbon fibers covering a W layer showed that neither the fibers nor the W were damaged (Renk et al., 2012).

The use of W-hollow nanospheres (W-hNPs) has also been proposed for W protection over the FW of a direct drive IFPP chamber. Using molecular dynamic and object kinetic Monte Carlo simulations, it was shown that W-hNPs are able to withstand very high temperatures (~ 3000 K) and very large internal pressures (> 5 GPa at this temperature) before rupturing. Even more interestingly, a nominal pressure increase as for an ICF exposure stimulates openings in the material that allow trapped gas to be released but then close within a few nanoseconds. Also, defects within the W-hNP shell migrate to surfaces at the operating temperatures

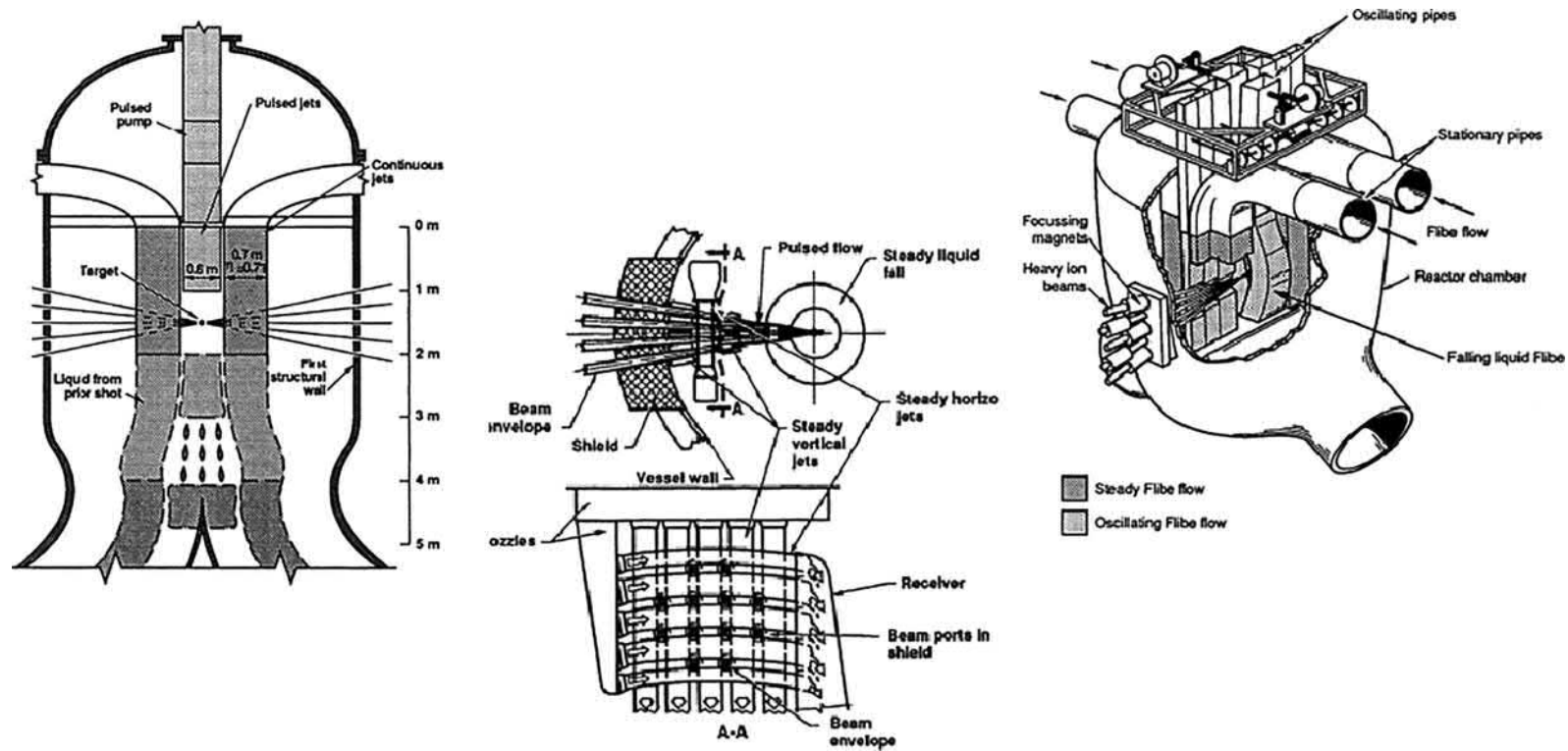


Fig. 17 (Left) HYLIFE-II pulsed flow. The flow speed for 8 Hz (chamber repetition rate) is 16 m/s with a 2-m fall height giving a flow rate of 34 m³/s. (Center) HYLIFE-II oscillating flow. (Right) Thick horizontal and vertical liquid jets protect the beam ports from radiation and help clear splash liquid for the next shot (Moir et al., 1992).

experienced and then annihilate. Long term integrity of such materials under long term irradiation remains a concern (Díaz-Rodríguez et al., 2020).

The behavior of helium in the thin tungsten layer protecting the FW in an unprotected chamber scheme is a key issue. If He builds-up due to both transmutation (n, α) reactions or direct penetration into the FW of He burn products from the ICF target burning, this degrades the metal. If He escapes regularly and avoids build-up the life of the tungsten layer would be extended. One mechanism for escaping would be if the He atoms diffuse out of the metal due to the elevated temperatures of the W. It is also important to understand if pulsed impingement (as for ICF) has any difference from continuous impingement (as for MFE). To explore this, Object Kinetic Monte Carlo code calculations were performed to evaluate the difference between continuous and pulsed He irradiation of W (Rivera et al., 2013). These results are shown in Fig. 18 and show that at temperatures over 1300 K, pulsed irradiation leads to larger He retention and creates a much higher density of vacancies that are highly occupied by He atoms compared to continuous irradiation under the same conditions (ion energy, temperature, and fluence). These results account for the decrease in the W damage threshold experimentally observed.

In depth the limiting damage for the blanket structure will be due to neutrons. The duration of the neutron pulse that is released from an ICF target is so short that it can be assumed to be released in a step function or instantaneously. There is neutron scattering that takes place in the fuel region such that the 14.1 MeV neutrons share energy with alpha particles and unburned deuterium and tritium, and this results in a spread of neutron energies that peaks at 14.1 MeV. The average neutron energy leaving the capsule is about 12 MeV. This neutron takes about 50 ns to arrive to a 5-m radius chamber wall. With a spread in neutron energies of 2 MeV, the neutron pulse (FWHM) is about 100 ns in duration. Therefore, for an IFE power plant operating at 1 Hz, 1 GJ yield and a chamber radius of 5 m, the first wall would experience a neutron fluence of about $1\text{E}14 \text{ n/cm}^2$ —but a neutron flux of about $1 \times 10^{21} \text{ n/cm}^2/\text{s}$ —an extremely high value. The areal atomic density of iron—the major constituent of steel is $8.5 \times 10^{22} \text{ atoms/cm}^2$ per cm thickness so that in a year with an IFPP at 80% availability the $2.5 \times 10^{21} \text{ n/cm}^2$ neutrons will have collided with 3% of the atoms in the chamber wall. This also means that the first wall for an IFE power plant under these conditions would experience a year's worth of neutron damage during a year's worth of power operation, but this damage will be accumulated over a total irradiation period of 250 milliseconds. So, there are therefore very different damage rates for IFE and for MFE.

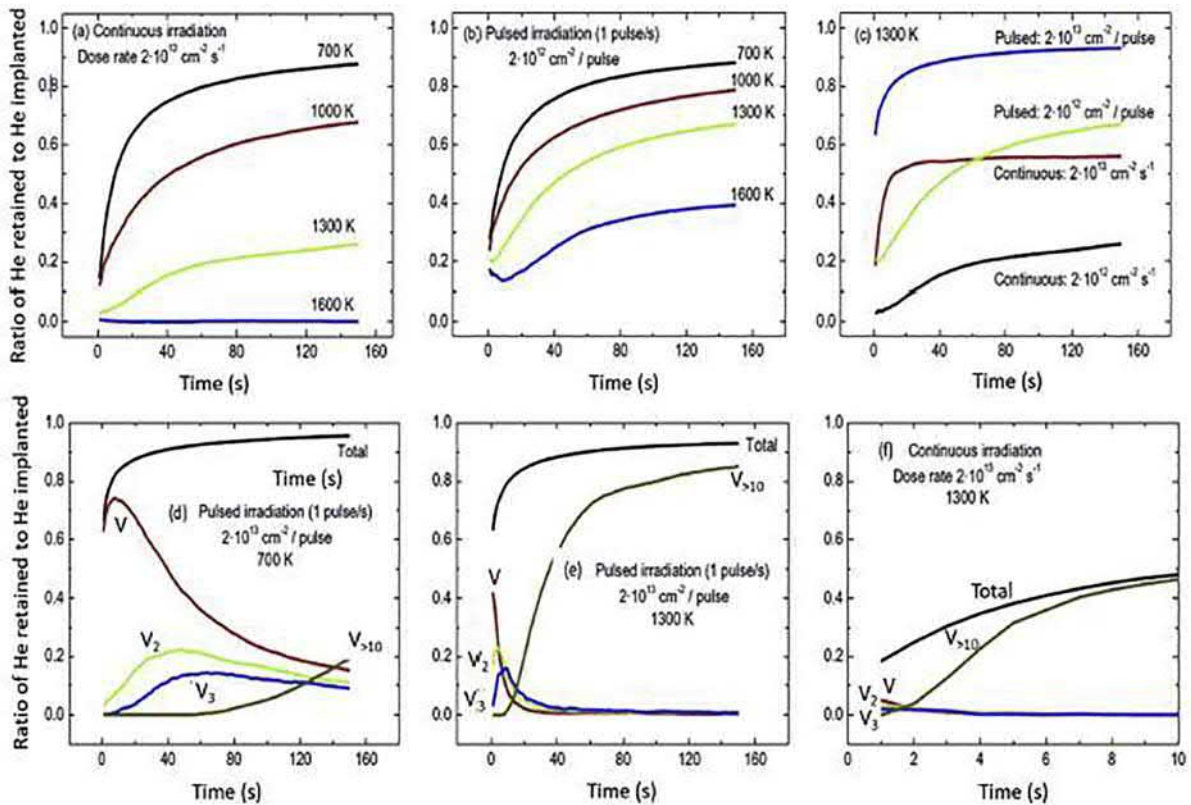


Fig. 18 (a)–(c) Ratio of He retained to total implanted He for W samples irradiated with He at 3 keV under different conditions: (a) Continuous irradiation at a dose rate of $2 \times 10^{12} \text{ ions/cm}^2\text{-s}$. (b) Pulsed irradiation, with $2 \times 10^{12} \text{ ions/cm}^2$ per pulse at a repetition rate of 1 Hz. (c) Comparison of continuous and pulsed irradiation at 1300 K and for different fluxes. In all cases, the number of He ions per cm^2 averaged over 1 s is the same for $2 \times 10^{12} \text{ ions/cm}^2$ or $2 \times 10^{13} \text{ ions/cm}^2$. (d)–(f) Fraction of He retained in trapping sites for W samples irradiated with He at 3 keV under different conditions. (d) Pulsed irradiation at 700 K, with $2 \times 10^{13} \text{ ions/cm}^2$ per pulse at a repetition rate of 1 Hz. (e) Pulsed irradiation at 1300 K, with $2 \times 10^{13} \text{ ions/cm}^2$ per pulse at a repetition rate of 1 Hz. (f) Continuous irradiation at a dose rate of $2 \times 10^{13} \text{ ions/cm}^2\text{-s}$ (Rivera et al., 2013).

Further, “a recent review of pulsed damage effects...concluded that there is sufficient experimental and theoretical evidence to be concerned about this phenomenon. Pulsing effects on precipitate phase stability and large changes in swelling and interstitial loops have been observed in stainless steel. Large changes in the void microstructure in Ni irradiated under pulsed conditions have also been observed. While there is no experimental evidence with a high neutron fluence at damage rates shown in Figure 8 it should be clear that ... the damage produced by 1 MW y/m² exposure of metal under steady state conditions may bear no resemblance to 1 MW y/m² applied in a [IFE-like] pulsed mode” (Sawan, 2011, p. 448).

Fig. 19 shows the instantaneous damage rates in a SiC first wall. “Furthermore, the pulsed nature of IFE results in extremely high instantaneous damage production rates as high as 40 dpa/s. The ratio of instantaneous-to-average dpa rate is about 50 million, with implications to the accumulation of surviving defects. In addition, the time spread of displacement damage production is longer than that for helium production leading to peak instantaneous He/dpa ratio that is higher than the average value. The strong directional bonding and the mass difference between Si and C atoms result in unique displacement features in SiC as compared to metals. The useful lifetime of SiC/SiC composites in a fusion neutron environment depends on the magnitude of swelling and the overall impact of the transmutation products on strength that are not presently understood. As facilities are not currently available to experimentally simulate the fusion neutron environment, active modeling and experimental activities in this area will provide significant insight into the physics of damage production and accumulation in SiC/SiC composites” (Sawan, 2011, p. 449).

The conditions of a first wall due to neutron damage at the end of a single year in a Magnetic Fusion Power Plant where a continuous flux of 4.5×10^{14} neutrons/cm²/s is impinging the first wall will be quite different than for an Inertial Fusion Power Plant unshielded first wall damage where, for the same neutron loading, the total irradiation time is less than 1 s and the wall has been annealing for the remainder of that second. For each single 100 ns pulse of neutrons striking the first wall, there is an annealing time of nearly a full second until the next 100 ns pulse arrives. Analyses using molecular dynamics simulations can provide insight into these very different irradiation conditions to gain some understanding of the outcome but have yet to be done. These would have to be validated with experiments in a facility that also does not exist yet. Prudence may suggest assuming materials will not stand up well to this very irregular neutron irradiation and take every possible step to reduce the neutron loading of the first wall in developing IFE power plant concepts. This suggests that a material that is continuously replenished and about a half to a full meter in thickness be placed between the first wall and the burning target. IFE concepts that employ liquid FLiBe or lithium or solid LiO₂ granules would reduce this pulse and create a more benign first wall outcome. The instantaneous concentration of both helium and hydrogen atoms in the first wall will also be much greater than for MFE. It is unclear how the combination of large dpa rates, helium production, and hydrogen build up inside the first wall will affect the first wall material lifetime.

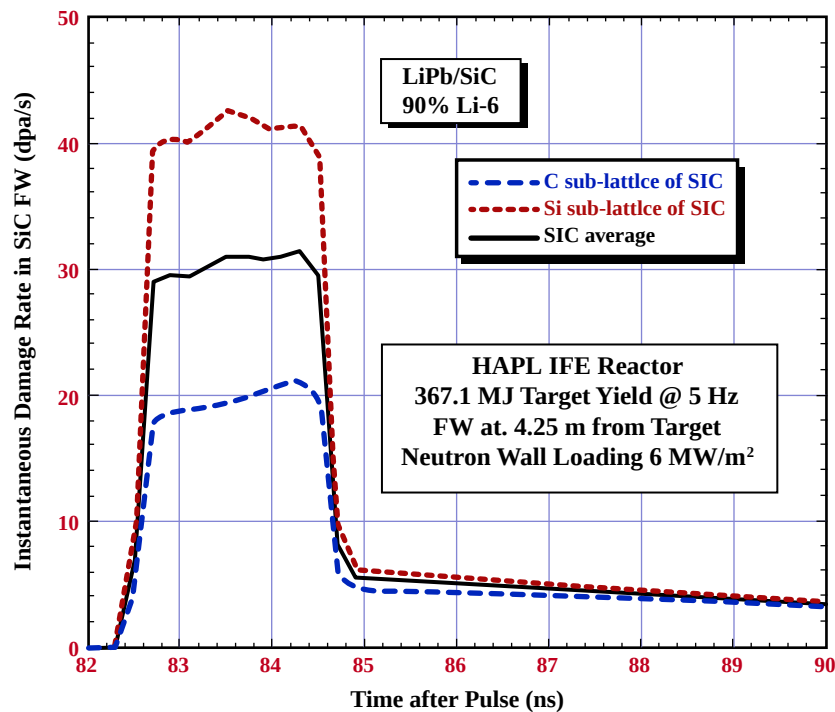


Fig. 19 Instantaneous damage rate in SiC first wall for LiPb blanket (Sawan et al., 2011, p. 449).

Robust chamber clearing

For direct drive, target emissions for a 1 GJ yield direct drive target will include about 700 MJ neutrons, 290 MJ of ions (debris and target fuel) with the remaining 10 MJ of energy (1%) in X-rays (El-Guebaly and Malang, 2009). This level of X-ray loading to first surfaces inside the chamber, assuming 3.5 to 5 m radius, will be of order $\frac{1}{4}$ to 1.5 cal per cm^2 and little material will be liberated by these X-rays. While the D-T fusion reaction itself emits a 14.1 MeV neutron, 80% of the net energy release of the reaction, and a 3.5 MeV He atom nucleus or alpha particle with the remaining 20% of the energy, neutron scattering in the high-density region of the burning target results in a net transfer of about 10% of neutron energy to the debris and fuel ions. The target emissions are important because whether the chamber has a bare first wall, wetted first wall, or a flowing blanket inside the chamber, the target material itself and any material liberated by the intensities of X-rays and/or ions must be sufficiently cleared to allow both the next direct drive target injection into an environment where it will survive to chamber center and until the driver energy arrives in a sufficiently cleared environment to reach the target and properly irradiate it.

For indirect drive, target emissions for a 1-GJ indirect drive target will again include 700 MJ of neutrons and a much more substantial portion of the emission in X-rays, some 250 MJ, leaving only 50 MJ of debris that is emitted over tens to hundreds of microseconds. The X-ray emission is relatively short—with emission through the target beam entrance holes (for laser indirect drive) completed in about 20 ns—and X-rays emitted from the walls of the target over some 100 ns since this is simply the radiation of energy from the hot debris—and characterized as a ~ 100 eV blackbody—while the entrance hole emission will be hotter— ~ 300 eV blackbody. This level of X-ray loading to first surfaces inside the chamber, assuming 3.5–5 m radius, will be of order 20–40 cal per cm^2 and substantial material will be liberated by the X-rays on each shot—thicknesses of tens to hundreds of microns. For example, if the first wall is a wetted wall of liquid Li, then this would liberate 2 kg of vaporized Li per shot that must be removed or condensed prior to the next target being injected and the next irradiation taking place. For a spherically shaped chamber, this material would also be launched perpendicular to the surface toward chamber center perhaps increasing the difficulty of quickly clearing this material.

The direct drive emissions therefore produce fewer challenging conditions for chamber clearing than indirect drive due to its substantially less X-ray emission.

Much more research is needed to be done to resolve whether the liquid first wall options using FLiBe or FLiNaBe eutectic salts that are liquids at the chamber operating temperatures discussed in this section are viable candidates for an IFPP. This is also extremely dependent on IFPP design parameters such as repetition rate and fusion yield. While no research is fully conclusive on these points, indications are that a small repetition rate IFPP (1–2 Hz) that uses modest yields (~ 1000 MJ yield) might succeed based on predicted condensation times. The reference (Raffray et al., 2006) includes an excellent summary of all of the computational tools (4) and experimental facilities (7) around the United States that have been used to research aspects of this very complex issue.

Robust target design and affordable target fabrication

Target fabrication, executed for the past 50+ years for a few shots per day and where target cost is seldom a driver for ICF research, will be a challenging task for Inertial Fusion Energy. First, the DT fuel being loaded into capsules must be maintained at ~ 18.5 K with temperature fluctuations less than ± 0.5 K after fabrication and during transport to the target injection system. Direct and indirect drive capsules are filled by using a very small fill hole (typically 5 μm in diameter) and fill tube to pass the cryogenic DT into the capsule. Tritium beta-decay is then used to smooth the fuel into layers that meet the 1- μm RMS surface finish. The mating of the capsule to its hohlraum is a slow high precision step where quality assurance is present in each step as two hohlraum halves are mated trapping the capsule that has been emplaced in a membrane. An IFE power plant operating at 0.1–20 Hz will need to fabricate and inject into the chamber from 8640 to 1,728,000 targets per day.

The LIFE project effort at LLNL from 2010 to 2014 conducted the most extensive study of indirect drive target manufacture to support IFE power plant needs. LLNL concluded that a Pb-based indirect drive target with a total mass of 4 g could be made for \$0.30 each to support a 16-Hz repetition rate. Lead was chosen as an affordable substitute for the gold that has been used in the vast majority of the thousands of hohlraums employed in experiments at LLNL on the NIF and at other facilities such as the Omega laser (Miles et al., 2011). This target is shown in Fig. 20.

Target performance modeling and supporting experiments on Omega were conducted to indicate similar target performance for lead as for gold for a somewhat larger laser driver energy of 2.2 MJ over the 1.8 MJ. Another excursion in building LIFE targets compared to NIF would be to employ 20–30 mg/cm^3 CH foam to wick liquid DT into a stable fuel layer in lieu of the time consuming and therefore unaffordable beta-decay layering and smoothing techniques. A modified gain of about 60, which met the LIFE requirement, is therefore assumed in lieu of the clean 1-D calculation resulting in a gain of about 100. A “rugby ball” shape also replaces the traditional cylinder reducing the interior surface area by some 30% thereby reducing the total laser energy required.

The LIFE effort assumed it would be less expensive to simply throw away the 35 kg to up to 7000 kg per day of target material than to develop a means to recycle since lead was described as being only \$1 per pound or a daily throwaway cost of \$77 to \$15,400 or an annual cost of \$28 k to \$5.6 M. Further substantial scaling up of current target fabrication capabilities will be required such as chemical Carbon Vapor Deposition machines with single batch sizes of about 45,000 capsules per batch.

The 2013 US National Research Council Report on ICF Targets said:

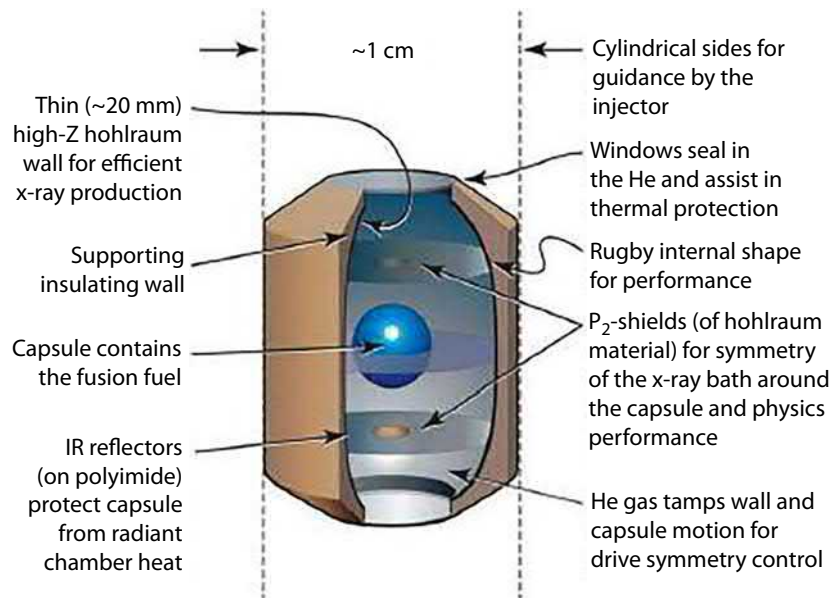


Fig. 20 LIFE (laser inertial fusion energy) target concept (Miles et al., 2011).

The primary concern of this panel with regard to ICF target fabrication relates to the technical feasibility of various proposed fabrication methods and the remaining technical risks and uncertainties associated with these methods. The question of whether the targets can be made cost-efficiently for a power plant is beyond the purview of this panel and is addressed by the National Research Council's IFE committee. [A] promising approach [is]:

Microfluidic Methodologies for Manufacturing Targets. The polymer shell that contains the DT fuel for DD laser and heavy-ion-beam fusion is proposed to be manufactured using a microfluidic droplet formation method. This is an established technology that is used to make ICF capsules for current DD and ID experiments. The principle is to flow three immiscible fluids coaxially through two nozzles where the Rayleigh-Plateau instability that occurs in the region where they intersect produces individual droplets. Each droplet is an emulsion consisting of a thin shell of water surrounding a spherical oil droplet; these droplets are collectively immersed in oil. The thin shell of water contains the polymer precursors that form the plastic capsule. The final phase of the production process is to remove the fluids using supercritical drying.

This process has a very high production rate that is needed for a fusion energy program. However, the repeatability and precision of the process must be improved if the process is to be a viable option for an energy program. (The repeatability of the current process does not ensure that each capsule meets the required specifications, so each capsule is individually measured to determine its suitability; this raises the cost of the targets, which is acceptable for ICF experiments but not for an IFE program.) In all other aspects, this production process offers a potentially viable method for producing targets cost-effectively.

One modification to the current microfluidic method that may improve the reliability is to introduce electromechanical control into the process (Cho et al., 2003). This process, referred to as "lab-on-a-chip," has demonstrated the feasibility and benefits of using electric fields and electronics to control important steps in the target production process (Bei et al., 2010; Wang et al., 2011). This concept can potentially reduce the production time and physical size of a target production facility and address the precision and reliability concerns with the existing process. Further development of the process is needed.

The lab-on-a-chip concept is being evaluated as a method to accomplish the cryogenic operation of loading the DT fuel into the capsule. Preliminary proof-of-concept experiments show that it is possible to form individual droplets of liquid deuterium of the correct size and wick them into a foam capsule in a short period of time. This would have the benefit of simplifying the target fueling process and shorten the process time, which would reduce the tritium inventory that is required by an IFE plant. Additional work is required to further develop this concept—specifically, to demonstrate that the process works with tritium and that it is practical to apply a condensed gas (argon, neon, or xenon) seal-coat onto the capsule once the fuel is loaded (NRC, 2013a,b, pp. 85–86).

In 2013 the US National Research Council Assessment of the Prospects for Inertial Confinement Fusion Energy Systems stated:

The scientific challenges to IFE target fabrication lie primarily in understanding the physics behind the specifications for inertial fusion target requirements: sphericity, uniformity and smoothness (How good is good enough?), and understanding the physics and chemistry behind the ability to achieve those requirements (Which physical processes control sphericity, uniformity, and smoothness?) Experiments with IFE targets at the NIF can help provide the physics understanding. The engineering challenges lie in selecting and developing materials that can achieve these requirements and in developing the processes and equipment needed to do so reliably and repeatedly with very high yield at reasonable cost. The specific requirements appear at present to include these:

The ability to fabricate IFE targets that meet specifications such as those for indirect drive:

- Capsules with 4 mm diameter, $< 1 \mu\text{m}$ sphericity, $\sim 100 \mu\text{m}$ wall with $< 0.5 \mu\text{m}$ tolerance, $< 200 \text{ rms}$ surface smoothness, surface power spectrum below the NIF capsule profile.
- Hohlräume fabricated to $\leq 10 \mu\text{m}$ accuracy. Targets assembled to $\leq 10 \mu\text{m}$ accuracy.

And those for direct drive:

- Foam shell capsules with $\sim 150 \mu\text{m}$ thick with $< 0.5 \mu\text{m}$ tolerance and $\sim 4 \text{ mm}$ diameter with $< 1 \mu\text{m}$ sphericity. Foam density $\leq 100 \text{ mg/cm}^3$ with cell size $< 1 \mu\text{m}$. A seal coat on top of the capsule having a $1\text{--}5 \mu\text{m}$ wall with $< 0.5 \mu\text{m}$ tolerance, $< 200 \text{ rms}$ surface smoothness, and a surface power spectrum meeting the NIF/NIC required profile.

A projected cost of mass-producing IFE targets for a power plant of $\leq \$0.50$ each (NRC, 2013a,b, p. 99).

The objectives of IFE target fabrication R&D must be to understand the physics behind the specifications for inertial fusion target requirements and understand the physics behind the ability to achieve those requirements to such a depth that target materials can be selected and/or developed that meet target specifications, and processes and equipment can be developed to do so reliably and repeatedly with very high yield at reasonable cost. The report concluded that target fabrication at the quality and production rate needed appears possible with continued development.

Robust high rep-rate target injection and tracking

The challenges for injection of a direct drive target to chamber center for IFE are: first, provide enough velocity on each target to maintain the required shot rate; second, accelerate the target in a way that avoids damage to the fragile frozen layers in the capsule; third, allow no more heating from the chamber walls or “atmosphere” to the target than 0.5 mK ; fourth, detect the target location to within $20 \mu\text{m}$ at shot time to ensure small beam alignment adjustment gives optimal irradiation; and fifth, ensure the driver energy arrives at the target in the designed power format and pulse duration to drive the target to the design gain and yield.

If the distance to chamber center is $3\text{--}6 \text{ m}$, then to support a shot rate from 0.5 to 20 Hz , target velocities to chamber center are shown in Fig. 21.

So, in addition to chamber clearing times, the IFE power plant shot rate will be limited not only by driver repetition rate capabilities but also by successful target survival to chamber center due to both chamber wall and/or atmospheric heating and acceleration forces.

The NRC 2013 Assessment on IFE made the following comments on rapid target injection:

After the targets have been fabricated, they must be injected into the chamber. For laser drivers and accelerators, several methods of ballistic injection have been suggested, including gas guns and electromagnetic accelerators. For present pulsed-power fusion system designs, the targets are attached directly to the end of a transmission line. In this case, the targets and a replaceable transmission line are inserted into the chamber mechanically. [We] consider only ballistic injection.

Gas guns have been built at LBNL and at General Atomics. These have been used to accelerate surrogate targets to high velocity ($> 100 \text{ m/s}$). In the case of direct drive, the targets must be carried by some kind of sabot to protect the target as it is accelerated in

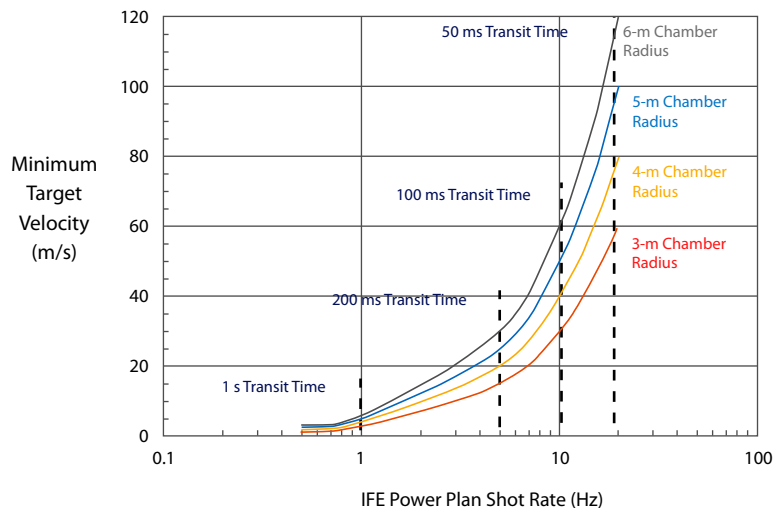


Fig. 21 Minimum target (direct or indirect drive) velocities needed as a function of chamber radius and shot rate.

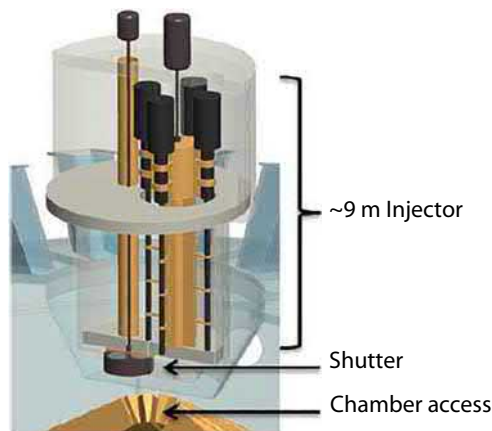


Fig. 22 The LIFE concept would use a 9-m long injection system to launch indirect drive targets to chamber center (Miles et al., 2011).

the gun barrel and injected into the chamber. The sabot is removed either mechanically (with a spring) or magnetically. The gas gun experiments have demonstrated high-repetition-rate injection, including separation of the sabots from the targets, in a burst mode (Goodin et al., 2003). In these experiments, the placement accuracy at a distance of 20 m was about 10 mm. This 10 mm includes the contributions from the accuracy of the gun and from the separation of the target from the sabot. Estimates of the placement accuracy for indirectly driven targets (no sabots required) are much better than 10 mm. This is adequate for subsequent target tracking and beam steering, as discussed in the next section. In summary, one can unquestionably build devices to inject the targets at adequate velocities and repetition rates. The remaining challenges are associated with wear and long-term reliability and durability, particularly in a fusion environment. The report concluded that target injection techniques have been developed in the laboratory that are adequate for subsequent target tracking and steering and that appear to be scalable to meet the inertial fusion energy requirements for speed and accuracy (NRC, 2013a,b, pp. 100–101).

A target injector for the LIFE concept was also designed and is shown in Fig. 22. It uses a 9-m long injection system with a rotating shutter shield to prevent neutron heating of targets waiting to be injected. The goal of the injection system is to place the target within 500 μm of chamber center so that final optics can steer beams in the final 25 μm to engage the target (Miles et al., 2011).

The target tracking system (see Fig. 23) detects the target when it is only 6-mm away from chamber center using a tracking laser beam array that produces a glint from the target that can be detected using the same final optics as for the main beams. The accuracy of beam engagement to the target is calculated to be as good as 10 μm using this system and well within the target specification of 100 μm (Miles et al., 2011). This system may be challenging to be implemented successfully for a chamber concept that employs liquid first walls and has a high shot rate.

The 2013 NRC Assessment of IFE further stated:

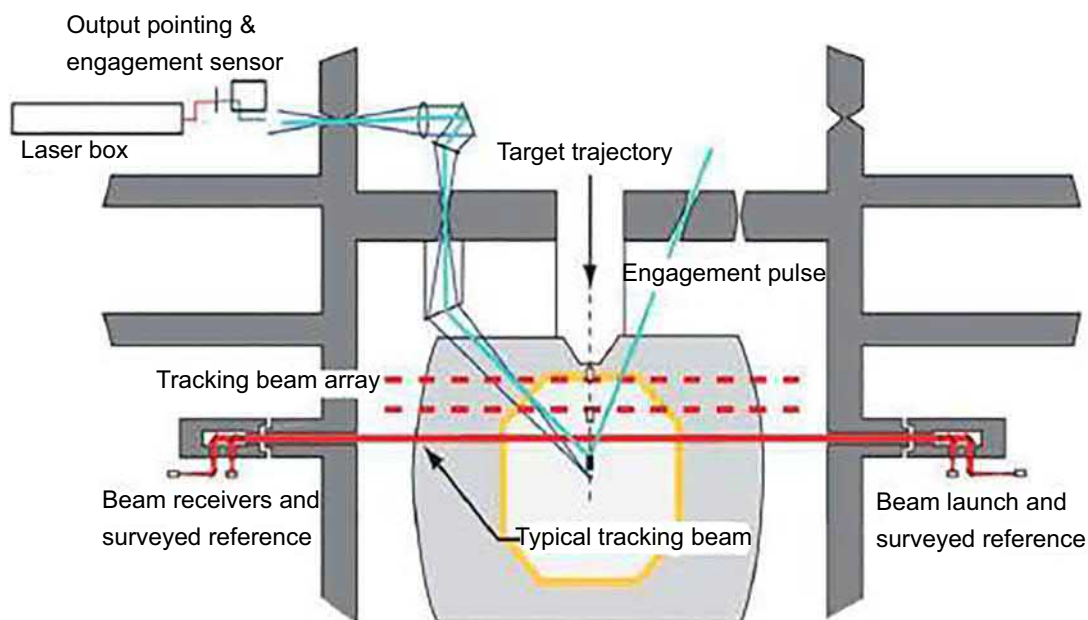


Fig. 23 The schematic of the LIFE target tracking and engagement sensor system (Miles et al., 2011).

The uncertainty in position of the targets when injected is much larger than the alignment precision of the driver beams relative to the target needed for ignition. Typically, the required alignment precision is approximately 20 μm for both laser and ion direct drive (Carlson et al., 2010). For NIF-like, indirectly driven targets, the required precision is approximately 80 μm . For ion-beam indirect drive, the requirement is calculated to be 100–200 μm , depending on the size of the hohlraum. Given this situation, it is necessary to track the position of the target and to point the driver beams at the target. At least two methods of target tracking have been demonstrated. One tracks the shadow of the target using light-sensitive sensors. The other relies on the reflection (“glint”) off the target. A scaled experiment performed by the University of California at San Diego and General Atomics demonstrated a beam alignment of 28 μm (Carlson et al., 2010). An alignment precision of 28 μm is nearly good enough, even for direct drive. Improvement to 20 μm seems possible, although advanced concepts such as shock-ignition targets may require still more precise alignment. The remaining challenge is to scale the technique to full size and full target velocity and demonstrate that it works reliably in a fusion environment. In a fusion environment one will undoubtedly have to deal with rapidly changing temperatures, mechanical vibration, and degradation of components by radiation.

The pointing of laser beams is usually done mechanically using a rapidly moving optical element. For accelerators, the beams can be pointed by pulsing relatively weak dipole magnets. For the beam parameters usually associated with ion indirect drive, this technique does not appear to be challenging. On the other hand, it may be necessary to put a significant energy spread on the ion beams to achieve the beam pulse durations needed for shock ignition or fast ignition. Energy spread produces dispersive effects in magnetic fields, so more work is needed to establish pointing feasibility for these options. [The report concludes that] target tracking and laser-beam-pointing methods that are adequate for indirect drive have been developed in the laboratory; direct drive will require higher precision (NRC, 2013a,b, p. 101).

Affordable activated material management

The challenges with target material management mainly apply to indirect drive targets. For an indirect drive target, in a 50-year plant lifetime at 90% availability, operating at 20 Hz and using the lead 4-g LIFE target, some 300 tons of low-level radioactive material would be produced if the lead was a once-through material expense. Studies (Latkowski et al., 2000) also suggest that while fusion will largely avoid material that can be rated high level waste, it will overwhelm low level waste repositories with extremely large volumes of material—even not including the activated target material.

The 2013 NRC Assessment of IFE states:

All targets produce radioactive materials—unburned DT fuel if nothing else—that must be recycled. Nevertheless, targets for laser direct drive produce orders of magnitude less high-Z material than indirectly driven targets for both lasers and ion beams. Although the indirectly driven targets have the advantage in terms of injection, direct drive has the advantage in terms of recycling. Most direct-drive (actually mixed-drive) ion targets also contain significant quantities of higher-Z material. In the case of pulsed-power fusion, the target materials themselves are dwarfed by the transmission line structure that is destroyed on each pulse.

There is currently little agreement on how to handle the high-Z materials such as Pb, Au, and Pd. These materials will be activated to some extent and will have to be considered as radioactive waste. Some researchers believe that it is preferable to use new material, such as lead, for each target (El-Guebaly et al., 2006). In this case, there is a significant waste stream, but it is only mildly radioactive. In contrast, the LIFE team proposes to recycle the lead used for the hohlraums (Dunne et al., 2011a,b; Latkowski et al., 2011). All surfaces in the reactor and vacuum chamber are designed to operate at temperatures exceeding the melting point of lead. The molten lead is collected and recycled. For liquid-wall chambers using lithium or molten salt, the hohlraum materials would have to be removed from the liquid. There are a number of trade-offs involved in the choice of hohlraum material. Some materials are better than others in terms of target performance. Some are better in terms of activation, toxicity, and cost. Finally, some are easier to separate from the chamber liquid.

For IFE concepts with wetted or liquid wall chambers, it may be possible to make the targets from materials that are constituents of the chamber coolant. Lead hohlraums for use with LiPb coolants and frozen-salt hohlraums with a high-Z liner for use with liquid-salt coolants may be possible.

There has been significant research on nearly all of the issues associated with handling and recycling the target materials (El-Guebaly et al., 2006). Determining the optimal methods and materials and demonstrating commercial feasibility remains an important challenge. [The report concludes that] target materials recycling issues depend strongly on the inertial fusion energy concept, the target design, and the chamber technology. Direct-drive targets have fewer concerns in the area of recycling and waste management; indirect-drive target materials handling, recycling, and waste management will need further development (NRC, 2013a,b, pp. 103–104).

Of particular importance are those waste streams that are considered “mixed waste.” Mixed waste has both a chemical hazard and a radiation hazard; irradiated lead is an example of a mixed waste. Lead is a coolant candidate as well as a target material candidate. Mixed waste currently has no disposition path in the United States, but regulations governing its disposal are under development and would likely be in place before deployment of the first commercial fusion plant.

[There report concludes that] design studies of inertial fusion energy power plants indicate that, with the use of low-activation materials, it will be possible to minimize high-level waste. However, the amount of waste that requires disposal, albeit near the surface, may be very large. Low-level waste disposal in the United States is becoming increasingly difficult [and further recommends continuing] studies that examine the potential for recycling and reuse of radioactive materials within the fusion system to reduce the amount of material that must be disposed of (NRC, 2013a,b, p. 134).

Target materials for inertial fusion energy (IFE) power plant designs might be selected for a wide variety of reasons including wall absorption of driver energy, material opacity, cost, and ease of fabrication. While each of these issues are of great importance, target materials should also be selected based upon their safety and environmental (S&E) characteristics. Previous studies have traditionally focused on the recycling, waste management, and accident dose characteristics of potential target materials (Latkowski et al., 2000; El-Guebaly et al., 2004).

It is clear that indirect drive targets, typically designed within a high-Z material hohlraum or enclosure around the D-T pellet, contain materials that need to be disposed or recycled after each target use. Previous studies on indirect drive targets have proposed the use of Au, Gd, W, Pb, Hg, Ta and combinations of these (El-Guebaly et al., 2004). The choice of target materials has a definite impact upon the fuel cycle design and the power plant material disposal/recycling strategy.

Analyses published to date (El-Guebaly et al., 2004) show that target materials represent a small stream compared to that for the nuclear island. However, recycling continuously during plant operation will contribute to public acceptance and may improve the economics of the plant life-cycle through more favorable waste categorization.

Efficient tritium production, recovery, and management

From the NRC Report on IFE:

Tritium production, recovery, and management are key to the success of an IFE system. The supply of tritium on Earth is limited (half-life ~ 12.3 years), so tritium “breeding” is required to ensure a ready supply of fuel for IFE. Tritium self-sufficiency (the “closed” fuel cycle for fusion) is necessary for commercial success or even large-scale test facilities. This requirement brings with it a range of issues, including target performance, tritium breeding potential of the blanket, and the tritium inventory in the IFE system. Because tritium is hazardous and readily mobile under certain conditions, there is a trade-off between tritium inventory and safety.

Several design studies have evaluated tritium-breeding performance and associated tritium inventories (DOE, 1992; Dunne, 2011a,b). These studies provide a useful first examination of these issues. The quantitative conclusions from all such studies must be viewed as somewhat uncertain, because they are at a relatively high level and miss many of the issues that become apparent when a system is actually built at engineering scale, revealing, for example, the actual area available for tritium breeding once all the equipment, manifolding, and such are considered.

The tritium inventory in the target fabrication plant is highly dependent on the target performance (lower performance means higher tritium inventory in the targets) and on the process used for target fabrication. Depending on the target fabrication process, tritium inventories in the target fabrication plant can be as large as 10 kg. Important in the consideration of tritium inventories is the ability to recover the unused tritium from the unburned DT fuel as higher burn fraction results in less tritium to recover. In the LIFE concept, estimates suggest that about half of the tritium inventory will be in the target fabrication plant, and total tritium inventory in the LIFE system is < 600 g (Dunne, 2011a,b). The SOMBRERO design study claims a similar (300 g) tritium inventory in the target fabrication plant (DOE, 1992).

Tritium breeding is accomplished in the blanket. IFE and MFE share tritium breeding needs and basic blanket concepts. The section on reaction chambers above summarizes the types of chambers under consideration for IFE; they fall into two main categories: solid walls and liquid walls. Liquid lithium is an option for liquid walls and has the advantage of relatively high tritium solubility, thus reducing tritium permeation concerns; however, that high solubility can result in undesirably high tritium inventories. Tritium recovery systems have been partially developed and tested at laboratory scale and indicate that acceptable tritium removal and thus inventory limits can be met with these processes; further testing at laboratory and engineering scales is needed to confirm this. Liquid lithium is a superior tritium-breeding medium (compared with molten salt and LiPb) and is therefore attractive from a tritium self-sufficiency point of view. Molten salt (e.g., FLiBe) and LiPb have the advantage from a safety point of view (El-Guebaly and Malang, 2009) of reduced tritium inventories and less chemical activity; however, they have low tritium solubility and thus a higher driving force for permeation (a safety disadvantage) and may require tritium permeation barriers to control the movement of tritium throughout the system. The SOMBRERO design is considerably different from most other IFE designs: it utilizes a granular Li_2O blanket (using gravity to move the particles through the system) that serves as both the coolant and the tritium breeder (DOE, 1992). Low-pressure helium removes the tritium from the Li_2O and transports the granules to and from the intermediate heat exchangers. The tritium inventory in the SOMBRERO design was originally estimated at just under 200 g; however, later analysis indicated that the inventory may be 1–2 kg of tritium in the carbon structure, emphasizing the potential for uptake of tritium in structural materials. A large tritium inventory requires an engineering or materials solution to ensure safety under off-normal conditions. Tritium removal from ceramic breeder blankets is also a topic of interest to the ITER test blanket module (TBM) program (Albrecht and Hutte, 2000) and the IFE program can leverage those activities.

Each of these studies shows tritium self-sufficiency. However, in actual application, losses due to uptake in structure, process losses, and actual neutron economy will likely be greater than accounted for in the studies. While there are a number of ways to adjust the tritium-breeding ratio (blanket thickness, ${}^6\text{Li}/{}^7\text{Li}$ ratio, neutron multiplier), until tritium breeding studies are done for detailed designs, including testing at engineering scale, the tritium self-sufficiency of any design must be considered uncertain. Tritium management will benefit from NIF and OMEGA studies to a limited extent (particularly target fabrication, tritium management, tritium handling, and tritium processing). However, the lack of a breeding blanket in NIF leaves an important area in a state of uncertainty. There has been limited work on liquid and solid breeder blankets in the IFE context. The work in the MFE program could be leveraged. Much could be gained from taking advantage of the larger MFE blanket programs under way in other countries. Further, a workshop was held in 1992 to brainstorm uses of the NIF to advance Inertial Fusion Energy to ensure its design did not preclude any of the ideas generated. This body of ideas, which included experiments with tritium, should be reviewed today, updated, and implemented (Tobin et al., 1994).

The report concluded that tritium-breeding performance has been considered in several design studies. These provide a useful initial examination of these issues. As these studies are at a preconceptual design level, they miss many of the issues that become apparent when a system is actually built at engineering scale, tritium recovery systems have been partially developed and tested at laboratory scale, and the signs are that acceptable tritium removal—and thus inventory limits—can be met with these processes. Further testing at laboratory and engineering scale is needed to confirm this, and tritium management will benefit from National Ignition Facility (NIF) activities, particularly target fabrication, tritium management, tritium handling, and tritium processing. However, the lack of a breeding blanket in NIF leaves an important area uninvestigated (NRC, 2013a,b, pp. 125–128).

Technology for IFPP driver—Chamber—Target integration

The separability of the IFE Power Plant components is considered one of its advantages compared to MFE. However, an interface must still be made, and a key technology is the ability to allow driver beams into the chamber area without chamber emissions contaminating or damaging the IFE Power Plant driver. The neutrons, as previously discussed, are traveling too fast for a mechanical shutter to be successful, as are the X-rays. Chamber gases and target debris must also be prevented from traveling up the beamlines as well. The transport systems such as focusing optics for lasers and magnets for heavy ion beams must be protected.

A major challenge for each driver technology proposed for IFE power plants—solid state laser, excimer laser, heavy ion beams, and pulsed power—is connecting to the target chamber in such a way as to pass the driver energy into the chamber environment and successfully irradiate the target to achieve the design yield without the chamber environment (e.g., neutrons, X-rays, debris, ablated material, tritium, etc.) being realized within the driver area. This section describes the technologies that allow this to happen.

Indirect and direct drive IFPP using solid state lasers

Table 2 gives a set of laser parameters for a diode-pumped solid-state laser for the Laser Inertial Fusion Energy (LIFE) IFPP using indirect drive targets.

A continuous phase plate (CPP) is used to produce a focal spot of 3.1 mm enclosing 95% of each beam's energy. The CPP also modifies the far field beam from a peak to a flat top for target drive purposes. A spectral bandwidth at 0.351 nm of 180 GHz suppresses Stimulated Raman Scattering (SRS), Stimulated Brillouin Scattering (SBS) and with a diffraction grating for Smoothing by Spectral Dispersion (SSD) of the laser speckle created by the CPP onto the target. There are 384 separate beams divided into 48 groups of 8 beamlines (Bayramian et al., 2011).

Table 2 Laser system requirements for diode-pumped solid-state laser IFE for indirect drive (Bayramian et al., 2011).

<i>Characteristic</i>	<i>Requirement</i>
Total laser energy at 351 nm (MJ)	2.2
Total peak power (TW)	633
No. of beam lines (48 groups of 8)	384
Energy per beam line at 351 nm (kJ)	5.4
Wall plug efficiency at 351 nm (%)	15
Repetition rate (Hz)	16
Lifetime of system (shots)	3×10^{10}
Availability (99%)	99
Beam pointing (μm RMS)	100
Beam group energy stability (8 beams) % RMS	< 4
Beam-to-beam timing at target (ps RMS)	< 30
Focal spot (mm)	3.1
Spectral bandwidth, 3ω (GHz)	180
Prepulse at 20 ns prior to main pulse (W/cm^2)	< 10^8

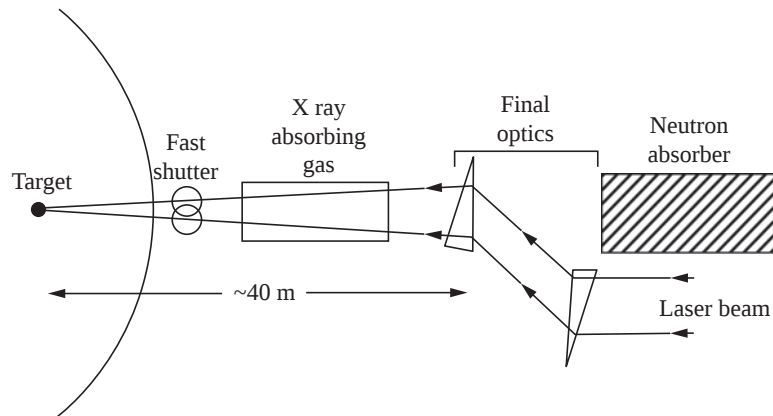


Fig. 24 Fused silica wedges can be used to deflect the beam out of the line of sight of the target (IAEA).

For laser drivers, approximately 3% solid angle of the chamber is open due to beamlines. The most plausible option to protect the final optics from chamber emissions for either direct or indirect drive is to use two optical wedges in line to shift the most sensitive optics out of the line of sight of the target chamber center as shown in [Fig. 24](#).

Fused silica will form color centers due to the neutron irradiation levels for IFE Power Plants at distances of about 40 m from chamber center and fusion power levels of 2000 MW_{fusion} which results in $\sim 3.5 \times 10^{12}$ neutrons cm²/s. However, it is possible to anneal the color centers even as they are made to keep the transmission reduction to well less than a few tenths of a percent if the optic can be kept at temperatures of about 350 °C. X-rays from indirect targets must also be mitigated from reaching the fused silica at fluences above 1 J/cm. This can be easily done by puffing a noble gas such as xenon into the beam line on a continuing basis maintaining ~ 1 mg/cm² gas density. This is a sufficiently small amount injected well up beam lines that vacuum pumps will remove this gas before it reaches chamber center and can affect aspects like keeping the target cold and/or injecting the target correctly. Direct drive targets do not need to protect final optics from X-rays but must protect them from debris ions emitted from the target. The same level of gas protection for X-rays will also stop debris ions from reaching the fused silica ([IAEA, 1995](#)).

Grazing incidence mirrors are also an option for 0.351 wavelength laser light to bend the light out of the direct line of sight of the target emissions exiting the chamber beam port. Grazing incidence mirrors work by accepting light at a very shallow angle (typically 6 degrees) where for a sufficiently polished and flat surface the light will be reflected at the same angle such that the beam is 12 degrees off axis from the chamber center. In [Fig. 25](#), the final optics elements design for the Prometheus IFE Power Plant concept, the grazing incidence mirror employment can be seen in conjunction with turning mirrors. A dielectric optic focuses the beam onto the first turning mirror setting the laser pulse spot size.

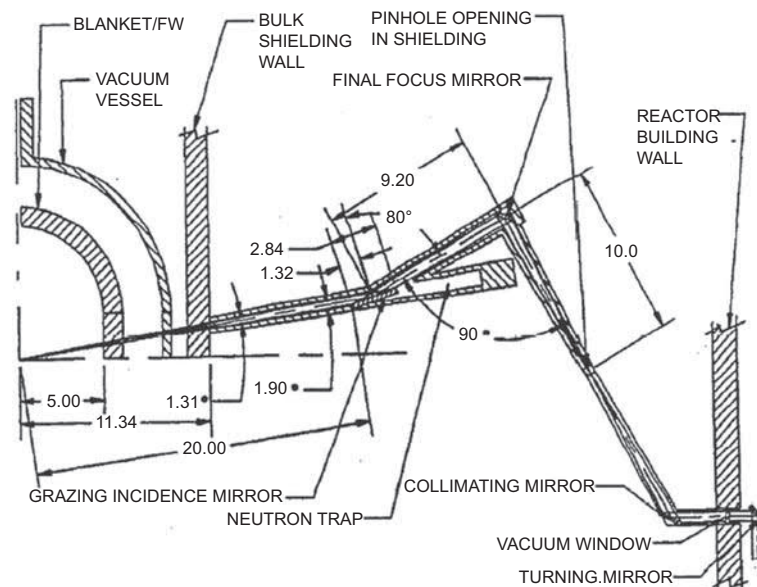


Fig. 25 Final optical [GIMM] elements in the Prometheus IFE Power Plant design ([Moir, 1999](#)).

Direct drive IFPP using excimer lasers

The KrF driver system consists of (1) a front-end which produces a pulse with the desired band width and temporal and spatial intensity characteristics; (2) several stages of intermediate amplification and progressive temporal/angular multiplexing; (3) final amplification by large electron-beam pumped 2-pass amplifiers; and (4) demultiplexing and beam delivery to the IFE power plant chamber (DOE, 1992).

Since electron beam amplification typically is most efficient in pulses of 500–1000 ns, significant temporal compression of each beam is required before it can be sent to the target. This is accomplished through a multiplexing system that uses arrays of mirrors to slice the long pulse beam into 100 smaller beams each 1/100th of the pulse duration of the long pulse to recombine the beams into the shorter pulse length. In Table 3, for example, the 600 ns pulse is shortened to 6 ns but generating 100 beamlets per beamline and then recombining them into 60 pulses at a pulse length shorter by a factor of 100.

Due to the very short wavelength of KrF and ArF, the use of fused silica wedges to direct beams out of the line of sight of chamber emissions is not feasible due to the much lower laser damage threshold and the added B integral. Multi-layer dielectric mirrors experience high damage rates due to neutrons and are therefore a poor candidate. The only credible means to adjust the directionality of the short wavelength beams (< 300 nm) such that streaming neutrons, debris, and X-rays from the chamber can be avoided is by the use of a Grazing Incidence Mirror (GIM) for each beam line. Two types have been studied: Grazing Incidence Metal Mirrors (GIMM) and Grazing Incidence Liquid Metal Mirrors (GILMM).

The leading choice for a GIMM are solid solution aluminum alloy mirrors due to the very high reflectivity of Al deep into the UV part of the spectrum (Tillack et al., 2009). Operation at a grazing angle such as 85 degrees with properly polarized light substantially increases reflectivity and increases the beam footprint by $10 \times$ thereby reducing the incidence fluence as shown in Fig. 26. It appears that defects (such as surface contamination) can be tolerated up to 1% of the laser wavelength suggesting that for ArF such contaminants would need to be limited to 2 nm in size (Tillack et al., 2002).

An option in lieu of a GIMM was proposed, the GILMM.

[For grazing incident liquid metal mirrors (GILMM),] a thin film of liquid metal serves as the robust final optics. The amount of laser light the GILMM can withstand is 57 J/cm² normal to the beam for a 20 ns pulse of 0.35 μ m light set by restricting the surface temperature rise to 200 degrees C to limit liquid ablation. Liquid aluminum can handle 106 J/cm². The liquid surface is kept to the required flatness by a combination of polished substrate, adaptive optics, surface tension and a low Reynolds number laminar flow in the film. The film's substrate must be polished to 15 nm. Surface tension keeps the surface smooth over short distances (~10 mm) and low Reynolds number laminar flow keeps the surface smooth by keeping the film thickness constant to less than 10 nm over distances > 10 mm. Adaptive (deformable) optics techniques keep the substrate flat to within 60 nm over 100 mm distances and 600 nm over 1 m distances. The mirror can withstand the X-ray pulse when located 30 m away from 400 MJ micro-explosions (50 MJ X-rays) when Li is used but for higher atomic number liquids like Na and Al there may be too high a temperature rise forcing use of other X-ray attenuation methods such as xenon gas, which may be needed for first wall protection anyway. The cumulative damage from neutrons causing warpage of the liquid film's substrate can be compensated by adaptive optics techniques giving the mirrors long life, perhaps 30 years. The GILMM should apply to both direct and indirect drive laser pulse lengths (Moir, 1999, p. 1).

Table 3 KrF laser parameters for the direct drive sombrero life power plant concept (DOE, 1992, p. 25).

Characteristic	Requirement
<i>Laser</i>	
Total laser energy at 248 nm (MJ)	3.6
Number of beam clusters	60
Beamlets per cluster	100
Final pulse duration (ns)	6
Repetition rate (Hz)	6.7
Total peak power (TW)	600
Wall plug efficiency (%)	7.5
Spectral bandwidth, 3ω	3 THz
<i>Electron-beam amplifier</i>	
Amplifier Energy (kJ)	60
Ar in Kr (%)	50
Pressure (Atm)	1
Initial temperature (°C)	500
Pumping (kW/m ³)	400
Extraction time (ns)	600
Amplifier gain	16
Lengths in optical, flow, and E-beam directions (m)	1/2/1
Fluence (J/m ²)	5
Electron-beam voltage (kV)	610
Diode current (A/cm ²)	40.6
Inductance (nH)	23
Applied field (kG)	6
Intrinsic efficiency (%)	14.5

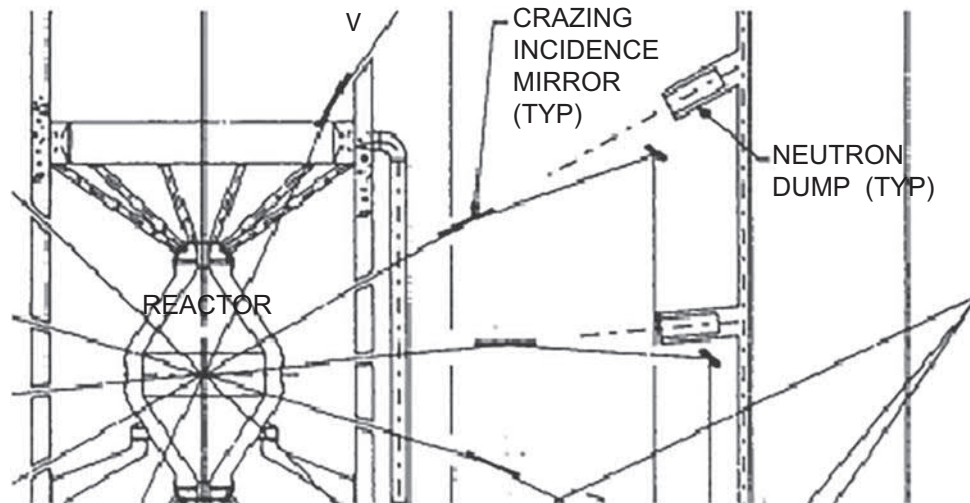


Fig. 26 A section of the elevation view of the Sombbrero IFE Power Plant showing the orientation of the grazing incidence mirror and the final turning mirror (DOE, 1992, p. 226).

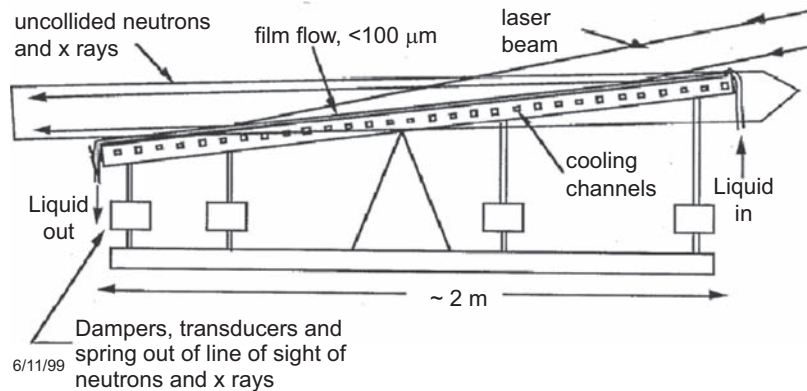


Fig. 27 Grazing incidence liquid metal mirror (GILMM) concept (Moir, 1999, p. 6).

A grazing incidence liquid metal mirror typical layout is shown in Fig. 27.

Direct and indirect drive IFE using heavy ions

The Robust Point Design-2002 (RDP-2002) Heavy Ion IFE Power Plant concept, the most recent HI IFE Power Plant study, adapts the most recent heavy ion target design to an updated power plant concept by now including 120 penetrations into the chamber (Yu et al., 2003). First wall protection and shielding for key components of the accelerator beam line is still achieved by using eutectic salt jets as in the HYLIFE-II IFE Power Plant concept but now a substantially larger number of jets are required as shown in Fig. 27.

The heavy ion beam chamber interface is complicated. First the final beam transport employs a bending magnet to direct only the last 10–20 m of beamline into line of sight with the chamber ports. This minimizes the region that must be protected by neutron shielding but of course includes the superconducting focusing magnetics. Just past the last focusing magnet in the RBD-2002 heavy ion beam driver they designed a 50-cm section that had 20-cm dipole at 1 kG and a 20 cm pulsed plasma source with an average of 10^{14} ions per cc. These performed the dual function of plasma “plug” for beam neutralization and as a magnetic shutter to remove target debris. The plasma shutter was timed to be on as the beam passed into the chamber assisting beam neutralization in the 6-m transport to the target. This also prevents plasma from passing into the beam transport section where good vacuum is needed. A second plasma “plug” is at the end of the tube enhancing neutralization after transit. In the reverse, the same components act as a magnetic shutter preventing target debris from passing up the beam tube. Vortices of the liquid eutectic salt are also set up that coat the beam tubes that penetrate the solid shielding and chamber walls. Scaled experiments using water have demonstrated that each of these three jet types can be created (Fig. 28) (Yu et al., 2003; Abbott et al., 2001; Pemberton et al., 2001).

IFE using pulsed power

Pulsed-power-driven IFE would use ≥ 50 MA of current from a pulsed power accelerator to create very large magnetic field fields that would in turn compress and heat magnetized D-T fuel to produce fusion yield. This approach has relatively low-cost, high-efficiency

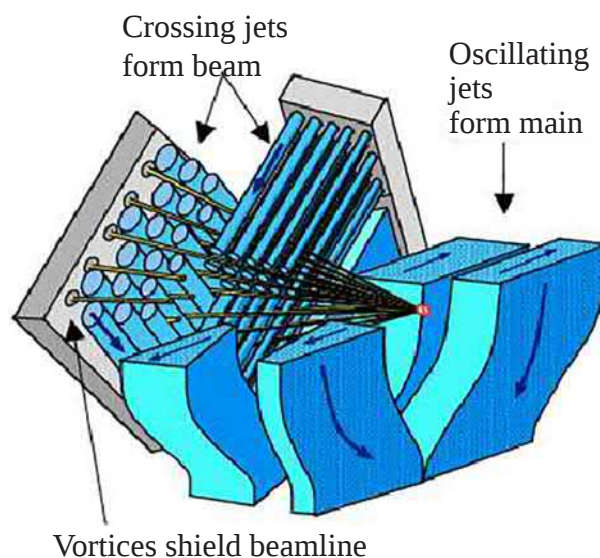


Fig. 28 A complex arrangement of crossing jets and oscillating jets protect the first wall (Yu et al., 2003, p. 4).

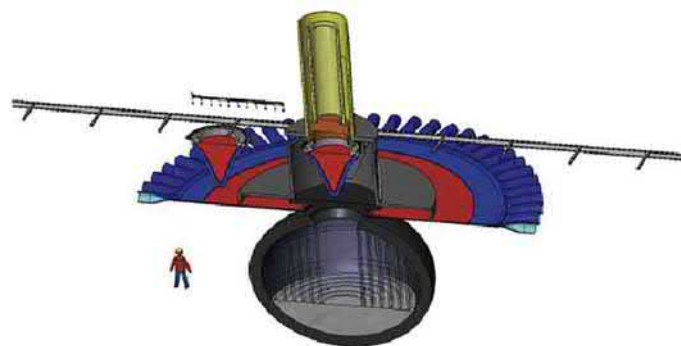


Fig. 29 The magnetize liner inertial fusion (MagLIF) concept using a liquid wall chamber. The target is called a recyclable transmission line (RTL) and is approximately 2-m long and 1-m wide (NRC, 2013a,b; Cuneo et al., 2011).

driver technology that may be directly scalable to the peak powers and energy needed for IFE. It appears feasible to perform these functions at a repetition rate of perhaps 0.1 Hz. This approach is called Magnetized Liner Inertial Fusion (MagLIF) as shown in Fig. 29 (NRC, 2013a,b).

While the approach appears relatively simple with a cylindrical target geometry and highly efficient delivery of driver energy to fuel implosion and heating, there is considerable uncertainty and technical risk surrounding all aspects of this approach limited relevant experimental data on target physics and ignition and a lack of in-depth design studies on inertial fusion reactors at the proposed multi-GJ yield and ~ 0.1 Hz repetition rate.

In addition to MagLIF, there other promising approaches to pulsed power fusion energy, including one called magnetized target fusion (MTF). While MagLIF operates on the 100-ns timescale, is ~ 1 cm in size, and involves open magnetic field lines, MTF operates on a ~ 1 μ s timescale, is tens of centimeters in size, and involves closed (field-reversed) magnetic field lines (NRC, 2013a,b, p. 78).

Unlike any other concept for IFPP, the pulse power option does not require any special mechanism to protect the driver from chamber emissions since it is a closed system and brings its energy to initiate the fusion reaction through conducting media rather than by transport through air. The only consideration would be appropriate load-matching to the Replaceable Transmission Line (RTL) to drive the 50 + MA of current through to the fusion fuel. Since the RTL weighs some 50 kg, the main issue with this concept is rapid enough replacement and disposal of this mass and fabricating them at the rate of one every 10 s.

Summary of IFE chamber technologies

Table 4 summarizes the information in section *Supporting Technologies for IFPP* in a way that shows the logical possible technology options for the combined driver-target-chamber interfaces.

Table 4 Technology options for IFE power plants (NRC, 2013a,b, p. 114).

	<i>Solid wall</i>		
	<i>Thick liquid wall</i>	<i>Protective gas</i>	<i>Vacuum</i>
Primary advantage	Heavy ions (HI); Pulsed power (Z) Fewer materials issues with X-rays, ions, or neutrons. Thick liquid also breeder/coolant.	Laser indirect drive Fewer first wall X-ray or ion material issues.	Laser direct drive Simplicity
Primary challenge	Chamber clearing/beam propagation, target placement	Chamber clearing, laser propagation in non-uniform gas density	First wall resistance to helium retention, surface morphology change, and mass loss.
Target survival	Hohlraum thermal insulation	Hohlraum thermal insulation	IR protective layer, start target cold
Driver target coupling	(HI) Accurate target injection. (Z) Target part of RTL: automatically aligned	Inject target close enough to chamber center to allow laser mirrors to be steered to required accuracy.	Inject target within 1 cm of chamber center, detect glint from target, and steer laser mirror to required accuracy.
Resistance to emissions of X-rays, ions, and neutrons	Thick liquid resistant to all emissions, including neutrons	6 $\mu\text{g}/\text{cm}^3$ xenon gas (760 mTorr at STP). Modeling: gas stops X-rays, reemits later peak wall $T < 850^\circ\text{C}$	Engineered tungsten or magnetic intervention
Chamber recovery: Rep-rate and clearing	(HI) Oscillating liquid jets sweep chamber (Z) Metal “waterfalls” protect walls; RTL obviates clearing	Recycle 0.5% of gas between shots.	Evacuate the chamber; well within commercial technology
Breeder; coolant	Thick liquid	Lithium, behind first wall	FLiBe or PbLi behind first wall
Chamber repetition rate and clearing issues	(HI) Do oscillating jets sweep out enough ionized/atomized liquid for driver propagation and target injection? (Z) Demonstrate RTL concept with scaled experiments.	Target survival and adequate quality laser propagation through residual hot Xe or Xe/Pb gas/plasma	Only gas load is from vaporized direct-drive target ~ 0.025 mTorr per shot
Chamber chemistry issues	Proposed liquid: FLiBe also maybe Na. All are very reactive. Must stay “chemically locked up” when subject to X-rays, ions, and heat	Effect of lead liquid / vapor (from hohlraum) on wall and optics. Deposition of carbon-tritium on “colder” surfaces	Should be no chemistry issues with tungsten wall. Deposition of carbon-tritium on colder surfaces
Other critical issues	(Z) RTL “insertion hole” needs protection from emissions	Target survival/laser focusing experiments	He retention; finish target warm-up

Supporting Technologies for IFPP

Fusion neutronics and modern tools and methods

Basics of fusion neutronics and radiation transport simulations

In IFPP using tritium-deuterium fuel, 14.1 MeV neutrons are produced in the burning target. Depending on the fuel density where they are born, the neutron *energy* spectrum emitted could differ from that of the *neutrons* produced in lower or higher densities. This must be taken into account for accurate neutronics calculations in IFPP and accurate calculations from target physics models are used.

Once the neutrons escape the target, they pass into the chamber components. Nearly all will interact with the materials encountered. As result of the neutron transport and interactions with matter different effects are produced, all of which must be carefully tracked in order to accurately predict all of their relevant effects.

Neutrons carry the energy that during propagation in matter is transferred to an absorbing medium according to various interactions. This deposited energy is the nuclear heat source that will be used to generate electricity. As neutrons move within the system, they induce reactions to produce tritium inside the chamber in a process called breeding. Fusion power plants must be self-sufficient in T fuel production as well as produce start up fuel for IFPPs coming on-line.

Neutrons irradiation can damage materials by causing atomic displacements within the materials and inducing changes in their chemical composition in a process called transmutation. This is responsible for the accumulation of H and He gases and solid impurities in materials. Displacements and transmutation products make changes in physical dimensions and degrade physical and mechanical properties of the materials. Selecting materials that will withstand the fusion neutron environment is one of the greatest challenges in the history of materials science. Neutron irradiation can lead to health-damaging effects to people. Neutron shielding must be designed using neutronics to protect people and equipment. Shielding design is challenging since neutrons penetrate deep into materials and also interact and produce gamma rays which also must be shielded.

In neutron activation nuclei in materials are transmuted through nuclear reactions into radioactive nuclei through various reactions, for example, in an (n, γ) reaction, the material increases in mass number by one but stays as the same element. In an (n, p) reaction, the mass number stays the same but the atomic number of the material is increased by one making it a different element. Safety, maintenance and waste related analyses require: the radioactivity inventories by nuclide accumulated during operation of an IFPP, the related decay heat generated, and the resulting radiation doses from decay radiation as function of time and location in the IFPP.

The fusion neutronics goals are to establish a firm understanding by modeling the neutrons and therefore support the IFPP design with complex analyses, suggestions of design configurations and materials to be used. Computer codes and databases enable these complex analyses to be performed.

There are two main parts of fusion neutronics: radiation transport and time-dependent evolution of the radioisotope inventory. Advances in radiation transport and transmutation/activation calculations have been significant in recent years and are described here.

The Monte Carlo method is preferred in fusion neutronics to simulate neutron and gamma transport. It enables integral quantities to be estimated in a detailed space, energy, angle and time mesh, through statistical sampling of individual neutron histories from birth in the fusion reaction to death by absorption or leakage and registering any single event through the neutron life. Simulating a sufficient number of neutron histories is what makes it possible to quantify the macroscopic physical quantities of interest.

In simulating a single source particle's movement, we compute the score for a quantity of interest and call this a "tally," such as neutron energy deposited in a particular region. The average of the scores corresponding to an infinite number of histories would be the true value (true mean) of the physical quantity represented by the tally quantity scored in the simulation, again such as energy deposition. A real MC simulation that employs a finite number of histories, N , computes this average (mean value) as an estimate of the physical quantity, and also estimates the error, that is, calculates how far the estimated mean is likely to be from the true mean for a sample of length N . This difference is the variance, which is necessary to be small in most cases to have a good estimate of the value or tally.

The Monte-Carlo method is applied to solving the neutron transport problem using a continuous representation of independent variables (i.e., of the phase space domain). It enables that any physical system can be modeled in full 3D geometry without the need for approximations, and that continuous energy nuclear interaction cross-sections are used. This means that with a high-quality nuclear data base, the accuracy of the MC calculation is only affected by the statistical uncertainty of the calculation itself. However, to obtain results of sufficient precision via Monte Carlo, this is proportional to the number of followed particles, and can therefore be extremely computationally expensive. Methods must be introduced to optimize MC neutron transport simulations in order to get precise results.

Major advantages in the application of Monte Carlo transport codes for fusion neutronics over the last decade are due to the development of: (i) global and very efficient Variance Reduction Techniques; (ii) parallel computing; and (iii) use of available CAD geometry data in the Monte Carlo calculations.

Calculations that a decade ago would have been heroic or impractical, such as those involving large and complex geometry configurations with important deep-penetration and streaming effects are today tractable efficiently and reliably. One of the more difficult problems is that of coupling neutron transport and activation simulations to provide radionuclides inventories, decay heat and shutdown doses in those complex systems. Such calculations are done routinely today.

Acceleration of MC transport calculations

There are two general approaches for the MC simulation of the physical transport process. In the first, the analogue MC simulation uses probability densities corresponding exactly to the physical transport process; in the second the non-analogue MC approach uses well-founded modified probabilities or sampling distributions in such a way that can lead to the same result in the estimates of the quantities than those using the natural probabilities. The approach that converges faster to the correct results will be the preferred one, i.e., the one getting the smaller variance for the same computational time. Analogue MC transport simulations have a very limited range of application. Non-analogue MC can reduce the amount of computer time required to obtain results of sufficient precision. That reduction in time gets larger the more efficiently modified probability distributions are selected and used to obtain estimates of the quantities on interest (i.e., select a good Variance Reduction, VR, model). Parallel computing is also a means to speedup MC calculations, as the MC method is inherently parallel (particles are simulated independently). The combination of these two acceleration approaches have dramatically increased the performance of modern MC codes.

Traditionally, VR techniques were used to reduce the variance in the estimate of physical quantities confined to a localized energy, angle and spatial (phase-space) region, and a considerable expertise was needed to use them. Automated VR methods were later implemented to make their use easier. However, when trying to calculate/tally a quantity in multiple localized regions, or even to compute distributions of a quantity, such as flux or dose rate distributions, users had to accept different levels of uncertainty among the estimated quantity in the different locations or run separate calculations optimized for each individual location. For these analyses deterministic methods were preferred. The Modern VR techniques developed over the last 10 years called global variance reduction (GVR) enable determining a quantity (e.g., flux, dose rate, etc.) with uniformly low statistical uncertainty throughout the entire problem space, and not only confined to a localized region.

The most successful GVR methods rely on using information previously obtained, as fast and effortlessly as possible, on the distribution of the neutron fluxes. Two types of strategies have been implemented to obtain this a priori information: stochastic ([Davis and Turner, 2011](#)) and deterministic - giving raise to the denominated hybrid Monte Carlo/Deterministic methods ([Wagner](#)

et al., 2014). Currently, ADVANTG is the most advanced tool implementing deterministic strategies to generate VR parameters to be used by MCNP (Mosher et al., 2015). Furthermore, solutions for an efficient combination of parallel computing and the application of strong GVR techniques have been developed, avoiding a significant reduction in the efficiency of parallel computation (Turner and Davis, 2013). Optimization of GVR techniques is an active field of research.

Modern geometry modeling in Monte Carlo transport calculations

The MC method has the potential to be applied to any arbitrary 3D geometric configuration. Traditional transport simulation codes have limited geometry modeling capabilities and hence only an approximation of the realistic geometry configuration could be made available.

The efforts to overcome this problem were aimed at making use of available CAD geometry data in Monte Carlo calculations. Three types of approaches have been developed. In what is named indirect or translational approach, the use of CAD geometry data is achieved by converting the CAD data into the geometry representation used by MC codes (Wu et al., 2009; Lu et al., 2014). In the direct approach, direct tracking of MC particles on the CAD geometry is employed (Wilson et al., 2010). More recently, the unstructured Mesh (UM) geometry description approach is available in MCNP6 (Martz, 2014).

The current status is that the mentioned approaches have reached the maturity level to allow their application to real and complex design analyses not possible just a few years ago. Further advances will take full advantage of using modern computational geometry generation tools in applying MC methods to physics problems with complicated geometries.

Coupling radiation transport-activation schemes and shutdown dose rate calculations

The advances in geometry modeling and acceleration of MC transport calculations have very important implications in many types of analysis. One of the most complex is that of the shutdown dose rate (SDR) mapping, a very important issue in the nuclear design on any fusion device. Two approaches have been proposed to address the problem.

The Rigorous Two-Step (R2S) approach is of general application to most of problems and involves three steps: a neutron transport calculation to determine space- and energy-neutron flux distributions, an activation calculation to compute the decay gamma source, and finally a photon transport calculation for determining the photon flux distribution in space and energy and the resulting SDR mapping. The “two-step” denomination specifies that neutron and photon transport steps are done separately. Transport by MC enable the geometry to be represented continuously, but as for the R2S activation a discrete mesh geometry representation is required, which can limit the accuracy.

The implementation of this approach has followed different improved methods. Let’s just mention the following for reader reference. Pioneering code systems were based on the cell-based method (Chen and Fischer, 2002) and they were followed by those under the “super-imposed mesh-based” method (Davis and Pampin, 2010). The limitations due to the spatial gradients of the neutron flux and the decay gamma emission from radionuclides produced have been overcome more recently with the “cell-under-voxel” method (Sauvan et al., 2016), which does not require use of a conformal mesh. Finally, R2S systems have been developed based on the use of a conformal mesh to tally neutron fluxes and carry out activation (Eade et al., 2015; Biondo et al., 2016).

The other approach, called Direct One-Step (D1S) (Mariano et al., 2020; Sauvan et al., 2020), performs both the neutron and decay-photon transport simultaneously in the same Monte Carlo simulation. This approach is the most accurate because no discretization is needed for any variable in problems where all the radioisotopes of interest are generated from the initially present stable nuclides by pathways including only one neutron reaction; in addition, the composition change in those stable nuclides should be negligible, and the disappearance of the radioisotopes is only due to radioactive disintegrations. These conditions are fulfilled in many practical situations.

Advances on both D1S and R2S methodologies are still going on. Efforts are being devoted to increase the efficiency of MC calculations by more suited VR techniques to sample more particles in the phase-space regions that contribute to the tally SDR, not just to the determination of the neutron flux distributions (Ibrahim et al., 2015). As for R2S new capabilities, the problem of how uncertainties in the MC neutron transport calculation impact the SDR uncertainty is another subject of current interest (Ibrahim et al., 2015; Alguacil et al., 2019).

All of the fusion neutronics developments mentioned here are having and will have an impact on the nuclear analysis and design activity of future inertial fusion (as well as magnetic fusion) facilities, providing time-integrated neutronics calculations with an accuracy never thought possible before.

As fusion neutrons are produced in an extremely short time in IFE, time dependent neutron transport calculations can be also necessary to evaluate time-dependent responses, such as material damage.

The calculation workflow for advanced CAD-based Monte Carlo transport and activation simulations for IFPP neutronics is given in Fig. 30. Examples are from ITER as recognition for its driving the impressive advances in computational fusion neutronics.

Technology for IFE safety and environment

This section explains how through careful design choices, the inherent safety features of IFPPs can be optimized. By including safety and environmental considerations early in the design phase, the following objectives can be achieved: minimization of accident source terms, and therefore of radiation releases related to credible accidents, qualification for shallow land burial (which reduces the lifecycle cost), and control of tritium inventories by efficient tritium extraction from the blanket and by keeping the breeding ratio close to one.

Neutronics workflow

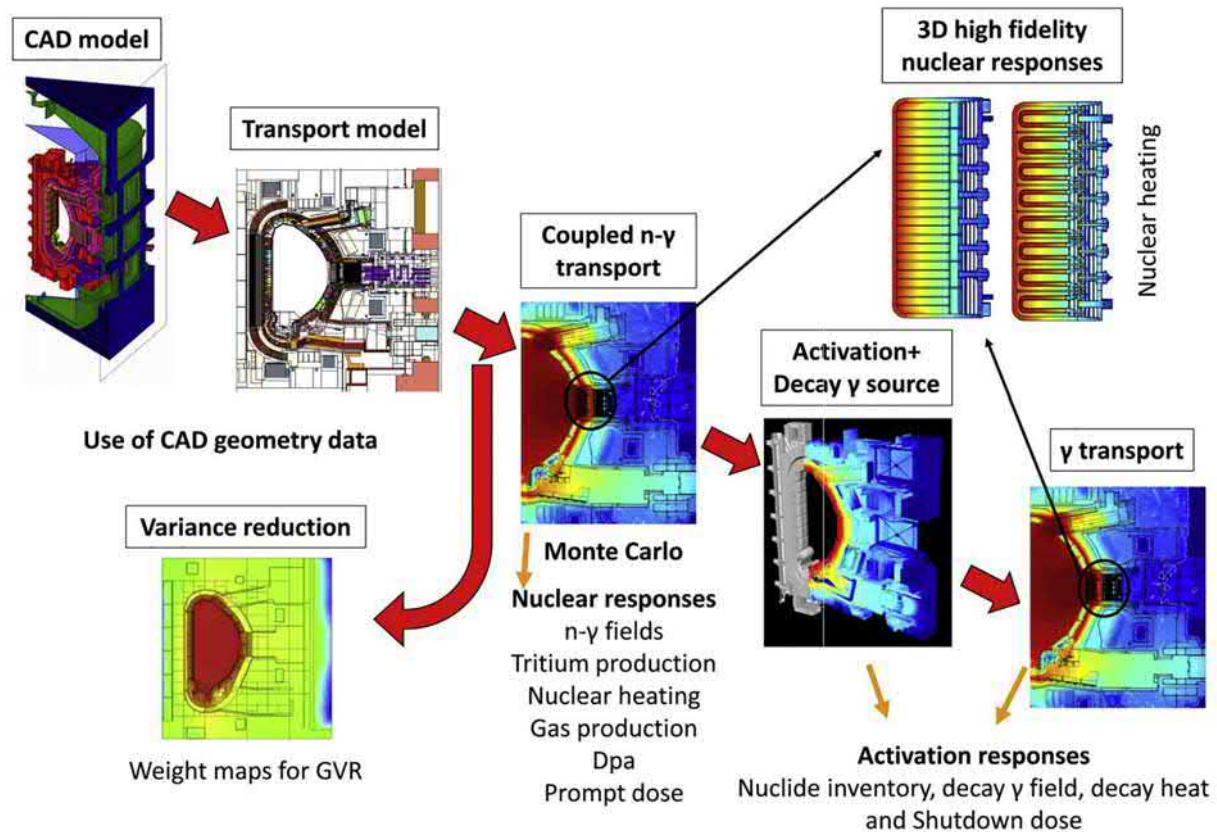


Fig. 30 Work-flow for modern neutronics calculations. A CAD model is converted to a radiation transport model. Variance reduction techniques such as Global Variance Reduction (GVR) substantially improve the efficiency and reduce the run times required for 3-D calculations. Coupled neutron-photon transport calculations provides nuclear responses such as tritium production, nuclear heating, H or He production in structures, displacements per atom in materials and radiation loads and dose fields. Coupling schemes for radiation transport and activation calculations provide the radionuclide inventory, decay gamma ray source and the resulting decay heat and radiation dose rate fields during maintenance and shut-down periods. Neutronics today can accurately determine nuclear responses such as nuclear heating distributions for use in an integrated multi-physics approach (Sanz-Gozalo, private communication).

Significant progress has been made in the area of IFPP safety and environment over the past two decades. Focus has been driven toward the minimization of inventories, accident safety, development of safety guidelines and licensing considerations, and waste management approaches (El-Guebaly et al., 2004; Reyes et al., 2013a,b,c, 2016). An important inherent safety advantage of IFE when compared to magnetic fusion energy (MFE) concepts that have been proposed to-date (including ITER), is that loss of plasma control events with the potential to damage the first wall, such as disruptions, are non-conceivable, therefore eliminating a number of potential accident initiators and radioactive in-vessel source term generation. The main operating risks in a fusion nuclear facility relate to the presence of radioactive materials (tritium and activation products) that could conceivably be released to the environment following an accident. The hazards are primarily managed by the implementation of the main safety principles: as low as reasonably achievable (ALARA) and defense in depth. In this way, existing hazards are identified and maintained ALARA, and a confinement strategy is implemented based on multiple barriers to provide an appropriate level of protection, according to the level of risk.

When considering the construction and operation of a future fusion nuclear science facility or a DEMO (either IFE or MFE), it is recognized that much work needs to be done in the safety and licensing area. The US DOE Fusion Safety Standards (DOE-STD-6002-96, 1996; DOE-STD-6003-96, 1996) provide general safety requirements and guidance for fusion experimental facilities, such as the non-evacuation criteria used above. Nevertheless, these Standards were issued in 1996 and must be brought up to date to ensure the congruency with current regulations, and with expanded current understanding of fusion safety challenges. Regarding the ITER precedent set by the French regulator it is important to recognize that regulations developed for this first-of-a-kind tokamak in France may turn out to be overly conservative or simply inappropriate for practical application to next step fusion facilities. Potential key items that need to be examined/re-visited based on experience in ITER include load cases, acceptance criteria, safety margins, materials qualification approaches and the so-called beyond-design-basis (hypothetical) event requirements (Taylor and Cortes, 2014).

Table 5 In-vessel radioactive source term for ITER and various IFE designs (Reyes et al., 2016; Taylor et al., 2009; Reyes et al., 2001).

Mobilizable source term (kg)	ITER	HYLIFE-II	SOMBRERO	LIFE
In-vessel tritium	1	0.14	1.17	0.06
Activation products (material)	1000 (W)	0.5 (SS)	50 (C/C Composite)	13.9 (HT-9)

Furthermore, future fusion power plant operations will present different safety characteristics compared to current pulsed fusion experimental facilities such as ITER. It is also clear that the safety features of future IFE and MFE facilities could differ significantly, not only in those aspects related to the behavior of the plasma, but also in terms of amount of radioactive source term, available energy sources and potential initiating events. As an example, Table 5 gives a comparison of estimated accident source terms in various IFE facilities compared to ITER.

Initial safety studies indicate that hazards associated to a future IFPP can be appropriately managed. Simulation of worst-case scenarios indicate that the accident consequences are limited and that the potential release of radioactive material would remain below the limit for public evacuation (Reyes et al., 2016; Taylor and Cortes, 2014). It is recommended that future work continues to advance the design of the specialized equipment through experimental R&D in order to reduce existing technical uncertainties. The trade-offs between the tritium plant subsystems need to be carefully analyzed, and the design improved and optimized along the ALARA principle for tritium inventories, and the implementation of the confinement approach. Although the future licensing process for a commercial fusion facility is unclear at this stage, moving forward it will be important to develop and/or revisit fusion safety guidelines so that they are as congruent as possible with updated regulations and with current understanding of fusion safety challenges.

Technology for IFE fuel cycle and tritium control

A 2018 report by the Fusion Energy Science Advisory Committee (FESAC) on “Transformative Enabling Capabilities for Efficient Advance Toward Fusion Energy,” describes a series of transformative technologies toward fusion energy development, including advanced technologies for tritium fuel production, extraction, and processing. Here we describe the technologies that are relevant for the IFE fuel cycle.

The IFE fuel cycle has certain unique characteristics, like the use of only 1 mg of tritium per fusion target, limited target debris-wall interaction due to chamber gas protection schemes, and a high expected burn-up fraction ($\sim 30\%$), which impacts the design and operation of IFE tritium processing systems (Reyes et al., 2013a,b,c). These intrinsic features result in limited inventories inside the vacuum vessel and throughout the facility simplifying reprocessing requirements when compared to what has been traditionally proposed for MFE. Table 6 provides data showing the large differences between MFE and IFE as far as tritium on site for a given facility. For example, for an MFE demonstration power plant, the daily fueling rate is 13.6 times higher for MFE than for IFE. The tritium recovered from plant operations is 16.5 times higher for MFE than for IFE.

Regarding tritium inventories, pre-conceptual level designs of the complete IFE fuel cycle have previously been published according to the functional requirements and technology selection for each of the fuel cycle sub-systems (Reyes et al., 2016). These studies describe a path toward design optimizations and R&D efforts that could allow for substantial inventory reduction compared with traditional technologies. In particular, the IFE fuel cycle has been modeled using a chemical flowsheet designed to predict steady state tritium inventories using the Aspen simulation platform (LANL/SRNL, 2011). The IFE fuel cycle flowsheet was established to support estimating the effects of design modifications upon fuel cycle inventories and predicting the impact of alternate technologies. Simple models of process operations within each subsystem (e.g., diffusers, oxidizers, molecular sieve beds, etc.) were developed and moved in and out of the flowsheet to allow for these iterations. We emphasize that the results presented here are based on pre-conceptual level design assumptions and detailed models of process operations within each subsystem should be performed as the design matures. These assessments show the potential for large reductions in the plant tritium inventory by utilizing

Table 6 Example MFE and IFE fuel cycle parameters (Reyes et al., 2016).

Feature	ITER	MFE DEMO	IFE INTERIM	IFE DEMO
Fusion power (MWth)	500	2700	1200	2200
Fueling rate (kg/h)	1.1	1.0	0.03	0.06
Burn fraction (%)	0.3	2	23	27
T consumption (g/day)	76	412	169	366
T recovery (kg/day)	25.3	20.2	0.6	1
T breeding (g/day)	0.4	450	203	439

Note that the tritium consumption and recycling figures quoted for ITER assume that it is operating continuously (in the mode of the MFE DEMO quoted in the table). However, the real operational numbers would be smaller (the “as designed” duty cycle of the device once it is in its long-pulse operation mode is 25%, i.e. the ratio of fusion burn duration to the period between the start of each burn) (David Campbell, Private Communication).

technologies that have been proven on an experimental scale or are already under development and yet not fully commercialized, such as mini-TCAP (Andrievski and Khatchoyan, 2016), Palladium Membrane Reactors (PMRs) (Heung et al., 2011), and Cryogenic Viscous Compressors (CVCs) (Gallucci et al., 2013).

In particular, two systems showing significant inventory reduction potential are the Light Isotope Purification (LIP) and Isotope Separation (IS). The initial LIP design contained 83% of the total facility inventory due to the long regeneration time of the molecular sieve (MS) beds based on an over-simplified nearly spherical tank with desorption controlled by band heating on the tank exterior. The regeneration time of the MS beds would be greatly improved by simply optimizing bed geometry and the regeneration methodology. Furthermore, an even more significant tritium inventory reduction would be realized if the MS beds in LIP could be replaced by diffusers (typically palladium-silver alloy membrane permeators). Diffusers take advantage of the fact that molecular hydrogen can rapidly diffuse through palladium alloy membranes while other compounds are completely excluded (Heung et al., 2011). Diffusers do not include any significant quantity of condensed phases allowing for a reduction of over 90% in the LIP major equipment inventory. The Thermal Cycling and Adsorption Process (TCAP) in IS holds 10% of the total tritium inventory. The conventional TCAP consists of a Pd/K packed column and a plug flow reverser (PFR) column. Mini-TCAP replaces the PFR with an active column that provides the inverse isotopic effect of the palladium column doubling the process throughput while producing a higher purity raffinate in addition to the high purity product (Andrievski and Khatchoyan, 2016).

Finally, another possible design improvement has been identified in the Chamber Gas Handling (CGH) system, which purifies the xenon chamber gas for recycle into the chamber and recovers the unspent fuel. This initially was accomplished by passing a portion of the chamber exhaust through a series of condensers to first separate out compounds lighter than xenon (like water vapor, carbon dioxide, and methane), and then condense xenon from hydrogen and helium. Due to the cost of compression and the uncertainty in condenser design, the pre-conceptual design only processed 10% of the exhaust stream for fuel recovery. Currently under development for other fusion applications, Cryogenic Viscous Compressors (CVC) are tritium-compatible, single stage condensers that could handle the entire IFE chamber gas exhaust flow (Gallucci et al., 2013). Designed to condense molecular hydrogen from helium, CVCs could more readily handle the requirements for Xe purification and recycling and tritium recovery. Future work will focus on additional optimization and design iteration, with the goal of advancing the design toward the conceptual stage. Trade-offs between the tritium plant subsystems need to be carefully analyzed, and the design improved and optimized along the ALARA principle for tritium inventories, and the implementation of the confinement approach.

Technology for IFE thermal management and electrical power production

There are different proposed options for heat extraction materials to support electrical power production for IFPPs. The option that shows the greatest thermal to electric conversion efficiency is the Brayton cycle with superior performance over Rankine cycles used in current nuclear power plants.

Different blanket materials can flow to create the primary coolant cycle of the power plant. There could be two main blanket material coolant states: a gas or a liquid. The gases He or CO₂ are used in IFPP concepts as well as Li₁₇Pb₈₃ and FLiBe or FLiNaBe coolants for liquids. Lithium must be a main constituent of the blanket in either a liquid form or in a ceramic solid form. Gas would flow through this media confined to tubes for flow purposes and to extract heat. The IFPP concepts LIFE, HiPER and KOYO-F chose to use Li as a liquid metal coolant.

The thermal deposition and management in the power plant cycle of the European HiPER concept is discussed here in detail (Sanchez et al., 2013). Technologies available in the short term were considered and studied to minimize risks. For the HiPER reactor proposal, the option of a self-cooled lead lithium (SCLL) blanket was proposed and two different arrangements of first wall (FW) cooling were laid out: an integrated first wall-blanket (IFWB) and a separate first wall-blanket (SFWB). In the IFWB configuration all the fusion energy of the reactor is extracted with a single coolant (Li₁₇Pb₈₃) whereas in the SFWB there are two different coolants, one cooling the FW (He) by removing about 20% of the energy and the other cooling the blanket (Li₁₇Pb₈₃) removing the remaining 80% of the energy. A simple ideal helium Brayton power cycle was analyzed for both IFWB and SFWB options in order to compare them in terms of power cycle efficiency. The SFWB yielded a higher thermal to electrical conversion efficiency than IFWB so we examine this design more closely (Sánchez, 2011).

In the IFWB, the Li₁₇Pb₈₃ carries to the outside of the reaction chamber all of the energy deposited in the blanket and first wall, a thermal power of 1702.5 MW. Thus, the cold and hot temperatures of the Li₁₇Pb₈₃ loop, 350 °C/450 °C is the operating temperature window for the power cycle. It is worth remembering the origin of these limits. When irradiating EUROFER97 below 350 °C, it suffers radiation induced hardening and a DBTT shift above room temperature. EUROFER97 is a ferritic/martensitic steel with reduced activation properties that has good thermo-mechanical properties and good dimensional stability under irradiation (i.e., resistance to swelling) (De Meis and Mazzone, 2016). On the other side, above 450 °C, the corrosion rates of the Li₁₇Pb₈₃ with EUROFER may become unaffordable. If the corrosion rates were exempt, the upper temperature limit would be 550 °C. Above that, the EUROFER97 suffers from creep rupture.

The SFWB configuration, as mentioned, makes use of a helium branch from the secondary circuit to cool the first wall independently from the blanket so here there are two heat sources. The helium Brayton power cycle for the SFWB layout is shown in Fig. 31. In this case, the first wall receives 25% of the fusion energy and the blanket 75%.

In addition to the heat magnitude difference, the heat sources operate at different temperatures. The blanket temperature range is, as was the case of the IFWB, 350 °C/450 °C. However, the first wall, being exempt from Li₁₇Pb₈₃, does not present any corrosion threat. If the first wall was built out of EUROFER97 protected with aluminum coatings, the upper temperature, 450 °C could be

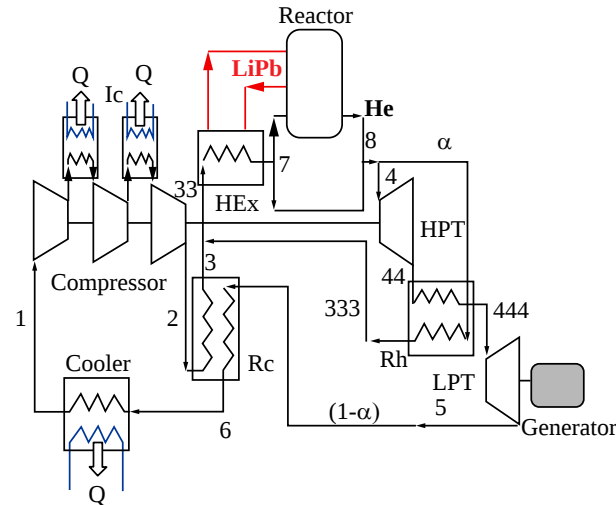


Fig. 31 HiPER power plant cycle schematic (Sanchez et al., 2013).

raised to 550 °C, or this purpose, the use of ODS-EUROFER97 could be justified (ODS stands for Oxide Dispersion Strengthened). The creep rupture threat of the ODS-EUROFER97 starts at 650 °C and the radiological performance is the same as EUROFER97. It is remarkable that the ODS-EUROFER97 makes sense as structural material only for an independently cooled first wall, as the upper temperature in the blanket is limited to 450 °C for corrosion reasons. Then, in the SFWB configuration, there are two heat sources: the first one is 375 MW operating at 350 °C/650 °C, and the second one is 1327.5 MW operating at 350 °C/450 °C. In this study, a simple helium Brayton cycle is designed, making use of the same basic cycle scheme for the more efficient SFWB case, in order to compare power cycle efficiencies when adding inter-cooling and re-heat processes.

The power cycle diagram is shown in Fig. 31. The following assumptions are made for the SFWB power cycle analyses considering additional inter-cooling stages and a re-heating process step as in nuclear power plants.

The inputs to the data analyses included $\text{Li}_{17}\text{Pb}_{83}$ coolant inlet/outlet temperatures of only 350/450 °C limited to this rather low value by effects to the first wall structural material, helium pressure of 8 MPa, inlet helium temperature to both compressors of 30 °C, reactor thermal power of 1702.5 MW and the coolant temperatures inlet and outlet for the first wall of 350/650 °C. Further, pressure drops in pipelines, in junctions, and in flow splitting were not considered. Pressure drops were considered across intercoolers, recuperator, heat exchanger and cooler. Changes in kinetic and potential energies in each component were neglected. The turbines isentropic efficiency was assumed to be 93% with 90% for the compressors. Recuperator and heat exchanger effectiveness were considered to each be 95%.

The results showed that for the optimum compression ratio of 1.8 the maximum conversion efficiency was 35.3%. However, with a re-heat cycle, the efficiency was raised to 37% and if a rise of 100 °C could be achieved for the outlet coolant temperature, for example, by using Al plating on EUROFER97, then the efficiency could reach 42.5% (Sánchez, 2011).

Future technologies

Some exciting new technologies are going to make a major impact on the development and successful implementation of Inertial Fusion Power. These include Additive Manufacturing, Artificial Intelligence, and Machine Learning and are described further below.

IFE and additive manufacturing

Fusion introduces many materials science challenges related to the behavior of tritium, the effects of the build-up of He and hydrogen, and the effects of displacements per atom (dpa) on lattices for strength and other properties—among many others. Additive manufacturing, which makes parts from three-dimensional models typically in layer-by-layer “builds,” could make major contributions to fusion systems (Nygren et al., 2018) in a similar way it is making major contributions in many industries today such as making jet engines (Dutta and Sam Froes, 2015). Fig. 32 shows five examples of products made from Metal Additive Manufacturing (MAM) that suggest that some crossover into fusion applications may be possible—especially for thick parts that are used in a fusion chamber. AM is excellent for making complex shapes, like enclosed void spaces as for coolant channels, and for when novel materials are used requiring jointless transitions from one material to another. Fabrication of refractory metals without melting may mitigate issues such as distortion, residual stresses, and crack formation in large structures. AM can also include processes called Direct Write where an example is depositing a thick slurry (ink) to build a fragile “green” preform that can be machined easily and then cured later to gain its finished properties. AM and Direct Write and similar processes are together known as AM+. The reference (Nygren et al., 2018) provides 45 references that cover a broad range of applications of AM that could have use of benefit for fusion.

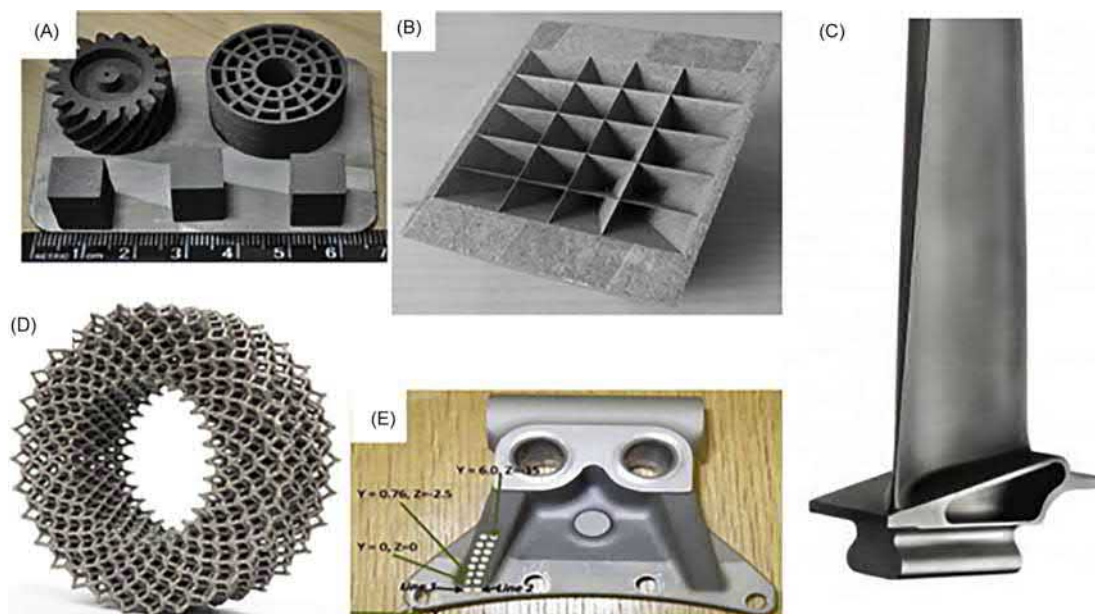


Fig. 32 Some AM metal products are shown and the method to produce them described: (a) TiC-Ferritic-Martensitic Steel preforms made by the binder jet process (Levy et al., 2017); (b) W Multi-pinhole collimator, made by selective laser melting (Deprez et al., 2013); (c) Ti turbine blade made by e-beam (Dutta and Froes, 2018); (d). W complex shape made by e-beam (Hancock et al., 2017); (e) Inconel 718 engine mount made by direct metal laser sintering (Hazeltine, 2009).

Nygren goes on to suggest specific contributions AM could make to fusion:

Features that may be useful for Plasma Facing Components (PFC) and that AM+ can easily provide include controlled porosity, functionally graded materials, and complex cooling channels. Enabling designs with a fine array of cooling jets for PFCs is one example. Another is controlled porosity or other nano-features (micro-fibers and/or nanoparticles) in the region of material close to the plasma-facing surface. Such features might increase recombination and recycling of D/T or mitigate crack growth into the deeper structure (an identified concern for W-based PFCs). Features that can collect He and transmutation products and limit tritium trapping, as well as graded compositions and transitions in materials may be important in developing PFC solutions and their development paths (Nygren et al., 2018, p. 2).

In tritium breeding blankets, most programs give solid breeders a higher priority for R&D than liquid breeders (e.g., Li or Li-Pb). Solid breeders are typically ceramics with relatively poor thermal conductivity arranged in pebble beds. AM+ offers the possibility to make open porous structures that could offer much improved performance for tritium breeding blankets and eliminate issues such as compaction in pebble beds. Also, with a radially graded structure, adjusting the breeder fraction and the cooling to the gradients in the nuclear heating and tritium production would be possible. A standard pebble bed design would require additional structure to make separate zones (Nygren et al., 2018, pp. 2–3).

Youchison and Garde (2012) have made open porous structures such as (d) in Fig. 33 and has modeled details of the flow and heat transfer and also fabricated open celled structures using ceramics. Manifolds and shielding could also be supported by AM+. Load pads for cryogenics could also be made for robustness under high mechanical loads while providing thermal isolation by using layers that transition from load bearing to thermally isolating. Large parts could be re-envisioned as collections of smaller parts giving more flexibility in design approach, a reduced volume of special shapes needed, and a reduction in the volume of replacement parts.

AM+ may provide solutions at lower cost than alternative options for some fusion applications but fusion engineers also need to keep in mind that AM “is rife with mega-hype and overblown promises.” “Many would-be users view AM as a silver bullet to fix all. Fortunately, AM experts can easily advise where AM can be useful and where it is not” (Nygren et al., 2018, p. 3).

Fusion engineers also need to be aware that the benefits of AM may require an R&D effort to be realized if other applications have not developed the material or fabrication process needed. Early consideration of AM and engagement with AM engineers can identify R&D pathways for those applications seen as feasible and enable the development of an AM Roadmap.

IFE and artificial intelligence and machine learning

Artificial Intelligence (AI) is the theory and development of computer systems to be able to perform tasks that normally require human intelligence such as decision-making. Closely coupled is machine learning where a computer program can learn and adapt to new data without human intervention. Machine Learning (ML) is a field within artificial intelligence that keeps a computer’s

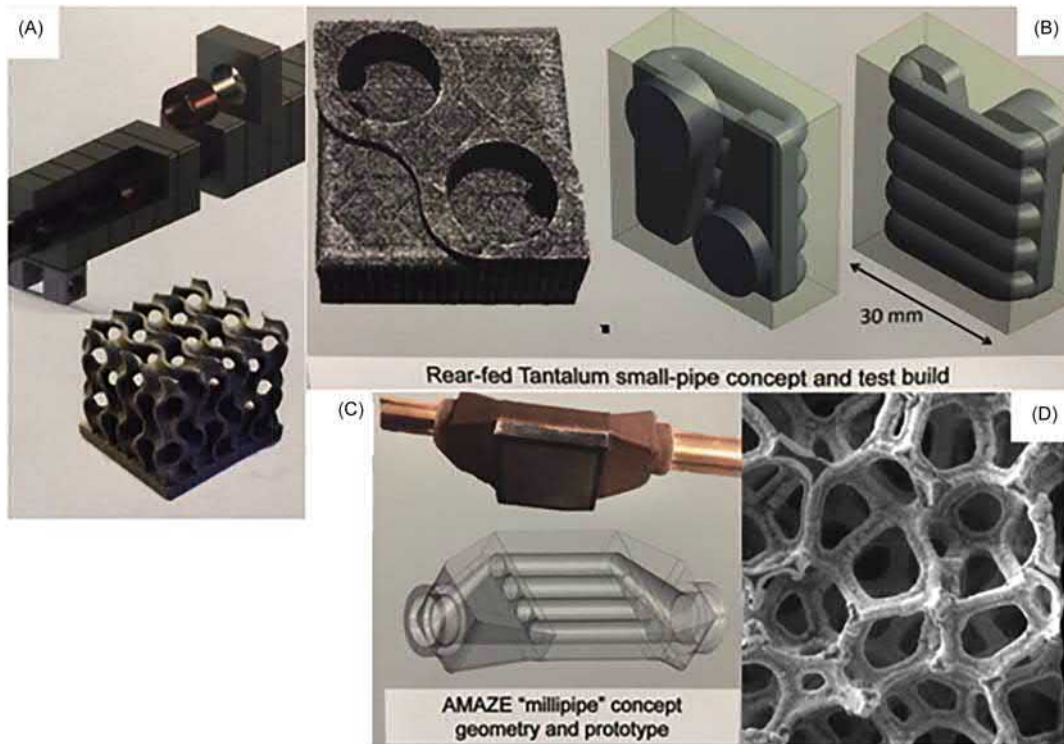


Fig. 33 AM parts for made for fusion (a–c) (Hancock et al., 2017); (d) Mo ligament structure formed in CVD (Youchison and Garde, 2012).

built-in algorithms current regardless of changes in its environment. A complex algorithm or source code is built into a computer that allows for the machine to identify data and build predictions around the data that it identifies. Machine Learning is useful in parsing the immense amount of information that is consistently and readily available in the world to assist in decision making. Machine Learning is being applied in a variety of areas and here we discuss applying it to the operations of an IFPP. Deep Learning (DL) is a subset of machine learning applications that teaches itself to perform a specific task with increasingly greater accuracy, without human intervention, and will also be very useful in an IFPP setting.

There will be many highly precise and highly repeatable functions that take place inside an IFE power plant. Then net fabrication of targets will need to be conducted very rapidly, perhaps 5–10 per second, in the IPFF Target Fabrication Facility. This will require a number of target fabrication stations working in unison to meet the local target demand. The tolerances are sub-micron on dimensions meaning that this rapid process must also be very precise. The process will need to be fully automated. Each target will need to be highly mensurated during its fabrication process through the use of highly precise and optimized sensors. The overall feedback and control relying on AI/ML/DL will be required for successful performance. To further complicate the process, the fabrication step that mates materials to the target capsule will need to be done at cryogenic temperatures. The target capsule fabrication and cryogenic filling step will also be done using AI/ML/DL processes combined with robotics. The current movement toward “lab-on-a-chip” at LLE for target fabrication is also a natural setting to implement robotics, machine learning, and artificial intelligence leading to accurate, precise, rapid, and economic target fabrication.

The transport/delivery of the target to the target injection station will need to be done where further sensor outputs allow the AI/ML/DL system to ensure that the target stays at the right temperature (18 K) with no more than 0.5 K fluctuation and with exquisite control over the robotic system providing the transportation.

The target will need to be properly positioned within the target injection system and integrated into the chamber processes seamlessly such that it does not experience any deleterious heating due to chamber emissions. The target, designed to robustly withstand the acceleration processes and housed in a sabot, is then accelerated at the right instant to within less than a tenth of a second to enter the chamber. Systems relying on AI/ML/DL will continually “sense” the chamber conditions to ensure they are suitable for target insertion.

Once inserted, the AI/ML/DL system is continually aware of the target position, the removal rate of any attached sabot, and the timing and coordination of all beamlines of energy that need to adjust to the predicted irradiation location in the chamber such that the irradiation takes place to within tens of microns in a chamber meters in diameter. This could only be done using AI/ML/DL systems to control the mechanical and electrical systems performing the functions.

The emissions of the target will also be diagnosed and the knowledge the AI/ML/DL systems has acquired will allow it to very accurately predict when chamber clearing will have occurred to keep the timing of targets into the chamber synchronized so that targets robustly survive to chamber center.

Advanced algorithms will be the backbone of machine learning systems and transform our vision of feedback control for a IFPP. This will enable operation at optimized operating points whose achievement and sustainment are impossible without high performance feedback control. The area of advanced algorithms includes the related fields of mathematical control, machine learning, artificial intelligence, integrated data analysis, and other algorithm-based R&D. Given the pace of advances, control solutions that establish IFPP operations will become within reach, as will the discovery and refinement of physics principles embedded within the data from experiments at current facilities such as NIF, Omega, and LMJ.

In February 2018, the Department of Energy's Office of Science published a report by the Fusion Energy Sciences Advisory Committee on Transformative Enabling Capabilities for Efficient Advance Toward Fusion Energy. While the report only examined Transformative Enabling Capabilities (TEC) for magnetic fusion energy, many TECs identified in the report were equally applicable to inertial fusion energy such as Advanced Algorithms and its comments on Artificial Intelligence (AI) and Machine Learning (ML) (DOE, FESAC, 2018).

The Report on Advancing Fusion with Machine Learning Research Needs Workshop held April 30–May 2, 2019, by FES and ASC stated:

Science Discovery with Machine Learning involves bridging gaps in theoretical understanding via identification of missing effects using large datasets; the acceleration of hypothesis generation and testing and the optimization of experimental planning. Essentially, machine learning is used to support and accelerate the scientific process itself.

Machine Learning Boosted Diagnostics is where machine learning methods are used to maximize the information extracted from measurements, systematically fuse multiple data sources and infer quantities that are not directly measured. Classification techniques, such as supervised learning, could be used on data that is extracted from the diagnostic measurements.

Model Extraction and Reduction includes the construction of models of fusion systems and the acceleration of computational algorithms. Effective model reduction can result in shorten computation times and mean that simulations (for the tokamak fusion reactor for example) happen faster than real-time execution.

Extreme Data Algorithms involves finding methods to manage the amount and speed of data that will be generated during the fusion models.

Data-Enhanced Prediction will help monitor the health of the plant system and predict any faults, such as disruptions which are essential to be mitigated.

Fusion Data Machine Learning Platform is a system that can manage, format, curate and enable the access to experimental and simulation data from fusion models for optimal usability when used by machine learning algorithms (DOE, Report on ML, 2019, pp. 2–3).

Data science methods from the fields of machine learning and artificial intelligence offer opportunities for enabling or accelerating progress toward the realization of fusion energy by maximizing the amount and usefulness of information extracted from experimental and simulation output data.

Magnetic fusion energy research is working very hard to try to use AI/ML techniques to try to solve one their most intractable problems—predicting when disruptions will occur and be able to mitigate or prevent their occurrence to prevent damage to Plasma Facing Components. Inertial fusion energy does not have an analogous problem to solve but the solution by magnetic energy researchers may provide insight into addressing IFPP-related control issues.

The collaborative development of the sensors and data collection system coupled with the Artificial Intelligence and Machine Learning/Deep Learning techniques are critical research activities in the near future to realize Inertial Fusion Energy.

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Inertial Confinement Fusion—Major Facilities

S.M. Finnegan, Los Alamos National Laboratory, Los Alamos, NM, United States

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Glossary

Direct-drive An ICF approach for which the driver energy is directly applied to the target.

High energy density plasma (HEDP) High energy density plasma is a plasma in which the internal energy density exceeds 1 Mbar of pressure.

Ignition The condition for which the energy deposition from thermonuclear reaction products within a volume of plasma is sufficient to maintain the plasma temperature and fusion reactivity in the absence of external heating.

Indirect-drive An ICF approach in which the driver energy is first converted to secondary energy such as thermal X-rays which then compress a target.

Inertial confinement fusion Fusion concept in which energy is delivered to the outer surface of a target, initiating an inward implosion. The imploding target compressionally heats fusion fuel located within an inner cavity. Inertia of the imploding mass is used to confine the thermonuclear-burning gas.

Pulsed-power The application of the rapid discharge of energy to generate large electrical current.

Introduction

All inertial confinement fusion (ICF) approaches rely on compression of the fusion fuel to increase the fuel density and temperature¹ towards the initiation of fusion self-heating, thermonuclear burn, and energy gain. ICF concepts systematically rely on generating large fuel pressures to maintain thermonuclear self-heating rates in excess of the net power lost by the gas from various processes, making them inherently pulsed and not steady state systems. Furthermore, ICF designs rely on three principal methods for compression and heating: light (either optical or X-ray) generated ablation pressure, electro-magnetic forces, and particle-beam (electron or ion) heating.

The earliest laboratory ICF experiments compressed hot plasma using magnetic fields to “pinch” (or stagnate) the plasma along a particular axis (e.g., Z-pinch, theta-pinch, screw-pinch, etc.). These approaches were unsuccessful as a result of various plasma instabilities that limited the achievable plasma densities, temperatures, and energy-confinement-times,² limiting fusion energy production.

¹Some concepts include additional external heating sources in the form of photons or particles.

²The energy-confinement-time for a thermonuclear plasma is the time-scale on which the global energy density changes as a result of various cooling mechanisms.

The demonstration of the laser in 1960 revolutionized the pursuit of inertial confinement fusion as it provided a potential means to directly compress the fuel to the extreme densities and temperatures necessary to generate significant fusion reactions, using only the inertia of the system for energy confinement (Nuckolls et al., 1972). Since that time, the laser has played a significant and dominant role in the development of inertial fusion energy (IFE). In addition to directly applying the laser energy to the compression of the target, commonly referred to as laser-direct-drive (LDD) (Craxton et al., 2015), the energy in the laser can also be converted to X-ray drive pressure, using a hohlraum (Lindl, 1995; Lindl et al., 2004), to compress the target. This indirect compression approach is commonly referred to as laser-indirect-drive (LID) and is the most developed approach to inertial fusion being pursued today. Additionally, alternative approaches apply both a long-pulse laser to compress the fuel to high density followed by a short-pulse (< 10 PS), high-intensity ($> 10^{18}$ W cm $^{-2}$), peta-watt lasers (Danson et al., 2019) to generate particle beams of protons or electrons that are then delivered to the hot deuterium-tritium (DT) gas in the center of the target to provide additional collisional heating to the gas in order to initiate thermonuclear ignition of the fusion fuel (Tabak et al., 2005). These “fast ignition” concepts require the co-location of both a target compression driver and a particle-beam source.

Along with laser systems, other dominant inertial fusion drivers include pulsed-power, or magnetic drive systems, as well as heavy-ions. While early magnetic-drive (or pinch) concepts were limited by magnetohydrodynamic instabilities, today’s systems are substantially more advanced, powerful, and diverse in their utility and application (Sinars et al., 2020). Today, magnetic drive is being used to study both indirect-drive target concepts (Sanford et al., 1999; Cuneo et al., 2012; Olson et al., 2020) as well as magnetic-driven concepts that also exploit laser heating and externally applied magnetic fields (Slutz and Vesey, 2012). As no implosion system coupling heavy-ion driver energy to an inertial fusion target exists today, heavy ion fusion research continues to work towards independently establishing control and conditioning of the ion beams themselves (Hofmann et al., 2018).

In addition to being uniquely configured and scaled for ICF implosions and the development of IFE, today’s major inertial fusion facilities have also led to the emergence of high-energy-density (HED) physics as a scientific frontier in the field of plasma physics. Today’s generation of major ICF facilities field experimental platforms³ that deliver the most extreme states of matter on Earth, and produce an ever increasing number of breakthrough discoveries ranging from the formation of exotic material states only found in the cosmos, such as metallic hydrogen (Celliers et al., 2018) and diamond rain (Kraus et al., 2017), to the self-generation of magnetic fields from turbulent dynamo processes (Tzeferacos et al., 2018) believed to play a prominent role in establishing the intergalactic magnetic field.

While laser and pulsed power-based facilities comprise the major implosion facilities in ICF today, in the past two decades there has also been significant development in alternative inertial confinement concept drivers and approaches (Thio et al., 2001), and new investments from private sector start-ups are increasingly exploring alternative instrument configurations. These new alternative concept approaches are not yet at a scale that could be consider “major” and as such are not reviewed here.

This article discusses the major experimental inertial confinement fusion facilities both in use today and under development. Major inertial confinement fusion facilities are defined as: laser facilities capable of delivering 10 kJ or more of energy to an imploding fusion target, and pulsed-power facilities capable of providing a main load current greater than 5 MA. Additionally, this article focuses on facilities and not experimental programs such as MAGO/MTF (Chernychev et al., 2016; Chernyshev et al., 2018) that, while satisfying the current criteria for pulsed power systems, are not executed using a fixed facility capability. Here, they are organized by driver technology, principally laser and pulsed-power facilities.

Laser facilities

The only “ignition”-scale facilities operational today are laser-based instruments such as the National Ignition Facility (NIF) at Lawrence Livermore National Laboratory in Livermore, CA, United States. Laser based ICF facilities are generally configured, with regard to laser injection orientation around the central target chamber, to support a specific LDD or LID target concept. Here, the major laser drivers for ICF are reviewed and listed in Table 1.

The National Ignition Facility (Livermore, CA, United States)

The National Ignition Facility (NIF) at Lawrence Livermore National Laboratory in the United States is the world’s most energetic laser, capable of delivering more than 2 MJ of energy onto a target with a peak power of 500 TW. Plans for the construction of the NIF began in 1993 (Paisner et al., 1994, 1995), with the official ground-breaking occurring in 1997, and the facility was completed in 2009. Following completion, the National Ignition Campaign (NIC) was initiated to develop and integrate all capabilities required for pursuing fusion ignition and, if possible, to demonstrate fusion ignition and energy-gain by the end of 2012 (Lindl et al., 2014). While fusion ignition remains elusive and has yet to be achieved on the NIF, researchers continue to make progress (Hurricane et al., 2019) with the NIF holding the record for having achieved the highest laboratory inertial fusion neutron yields to date at $\sim 3 \times 10^{16}$. Today, NIF remains the only mega-Joule class laser ICF facility in the world, fields in excess of 400 experiments

³A “Platform” is a unique experimental hardware set/configuration for the generation and study of the formation and dynamics of a particular state of matter, including integrated implosions.

Table 1 List of major laser ICF facilities worldwide.

Facility	Location	Systems/type	# Beams	Configuration	Key parameters (energy/time/wavelength)	Status
NIF	United States	NIF/Nd:glass	192	Polar/4 cones	2.2 MJ/4 ns/351 nm	Operational
		ARC/	4		0.4 kJ/1.3 ps/1053 nm	Operational
LMJ	France	LMJ/Nd:glass	176	Polar/2 cones	1.5 MJ/4 ns/351 nm	40 Beams operational
		PETAL/Nd:glass			0.85 kJ/0.7 ps/1053 nm	Operational
UFL-2M	Russia	UFL-2 M/Nd:glass	192	Symmetric	2.8 MJ/tbd/527 nm	Under construction
SG-III	China	SG-III/(Yb-fiber, Nd:glass)	48	Polar/4 cones	180 kJ/3.5 ns/351 nm	Operational
Omega	United State	Omega-60/Nd:glass	60	Symmetric	40 kJ/0.8 ns/351 nm	Operational
		Omega-EP/Nd:glass	4		0.5 kJ/0.7 ps/1053 nm (5 kJ/351 nm)	Operational
Iskra-5	Russia	Iskra-5/iodine	12	Symmetric	30 kJ/0.3 ns/1316 nm	Operational
					2.6 kJ/5 ns/657.6 nm	Operational
SG-II UP	China	SG-II UP/Nd:glass	8	Polar/1 cone	24 kJ/3 ns/351 nm 1 kJ/	Operational
		SG-II PW/Ti:sapphire	1		1.7 ps/1053 nm	Operational
Gekko XII	Japan	Gekko XII/Nd:glass	12	Symmetric	10 kJ/1.7 ns/527 nm	Operational
		LFEX/Nd:glass	4		2.8 kJ/1 ps/1053 nm	Operational

annually, and continues to serve as a key element of the National US Nuclear Security Administration's science-based Stockpile Stewardship Program. Images of the NIF are shown in Fig. 1.

The NIF is a flash-lamp pumped neodymium (Nd) glass (Nd:glass) laser, capable of producing up to 20 kJ of energy in each beam at the fundamental (1ω) wavelength of 1053 nm (infrared) (Moses et al., 2008). The system consists of 192 individual beams that are transported in groups of four, called "quads," to the target chamber where the beams in each quad are frequency tripled (3ω) to the wavelength of 351 nm (ultraviolet). After frequency conversion to 3ω , each beam has approximately 10.5 kJ of energy. The beams are focused through a final focus lens onto the target at target-chamber-center (TCC), delivering up to 2.2 MJ of total energy in a temporally shaped pulse with peak power of approximately 500 TW.

The NIF is currently configured for research on LID fusion concepts that convert the incident laser light using cylindrical hohlraums to X-ray drive pressure for compressing the fusion fuel (Lindl et al., 1995, 2004). As a result, the 48 final optics assemblies that house each quad of beams are positioned around the top and bottom poles of the target chamber in 4 cones at polar angles 50 degrees, 44.5 degrees, 30 degrees, and 23.5 degrees. The target chamber is a 10 m diameter, 10 cm thick aluminum vessel, designed specifically to withstand a 45 MJ yield output, 1200 MJ annual-yield, and 50 shots of 20 MJ per year (Paisner et al., 1994). The vessel contains 48 indirect-drive ports, 24 direct-drive ports, and 85 diagnostic ports. The additional direct-drive ports allow for the modification of the NIF to accommodate a symmetric laser-direct-drive configuration.

Each beam within a quad can be steered individually, enabling each beam to be precisely positioned with regard to the target. Providing a further level of control, the NIF also has the capability to tune the wavelength of the laser beams by approximately ± 2.3 Å. At present, the NIF laser has the capability to do this with three separate wavelengths simultaneously, one wavelength for each of the inner beam cones (23.5 degrees and 30 degrees) and one for all of the beams in the outer cones (44.5 degrees and 50 degrees). NIF beam also employ Random phase plates and limited smoothing by spectral dispersion to improve on target beam quality.

Additionally, the NIF laser provides a great deal of flexibility in pulse length and laser pulse shape. The NIF has the capability to deliver a variety of precision laser pulse shapes. Each quad may be independently configured for both timing and laser pulse shape, with the shape and timing of a pulse being common for all four beams within a single quad. As a result of the amplifier configuration, the flexibility of the pulse shaping is limited between the two quads within a single bundle.



Fig. 1 (A) Image of the NIF laser-bay. (B) Image of the NIF target chamber. (C) Image of the inside of the NIF target chamber including target position. Image (A) <https://www.flickr.com/photos/lnl/3676978870/in/album-72157607099824019/>; (B) <https://www.flickr.com/photos/lnl/3421927305/in/album-72157607099824019/>; (C) <https://www.flickr.com/photos/lnl/3421902499/in/album-72157607099824019/>

Targets are held and positioned by either cryogenic or non-cryogenic TARget POSitioners (TARPOS), see Fig. 1C. For ignition targets, a Cryo-TARPOS is necessary to complete the formation of the fuel ice layer within the target as deuterium-tritium fuel is transferred to the capsule, using temperature differentials, through the capsule fill-tube. Beta-layering⁴ is used to ensure a smooth inner ice surface within the capsule (Moses, 2008).

The NIF hosts a wide array of some of the world's most advanced diagnostics for evaluating HED systems, and indirectly-driven ICF implosions, in particular. These include, but are not limited to, full aperture backscatter for hohlraum energetics, velocity interferometers for shock timing, calibrated soft X-ray spectrometers to measure radiation drive, gamma ray detectors to measure burn history, and a magnetic recoil spectrometer to measure neutron spectra. Further information regarding the NIF diagnostics is available on the NIF website [<https://lasers.llnl.gov>].

In addition to the 192 long-pulse drive beams, the NIF also hosts the ARC (Advance Radiography Capability) laser (Di Nicola et al., 2015). The ARC laser uses four beams (1 quad), each beam being split into two, producing a total of 8, petawatt (PW) class beams delivering between 0.4 kJ and 1.7 kJ in 1.3 ps and 38 ps respectively. As such, the NIF ARC can provide high-energy X-ray energies for radiographic imaging using the backlighting technique. By staggering the delivery timing for each of the eight beams onto a backlighter target, the NIF ARC system can deliver a time-sequence (i.e., movie) of radiographic images showing the evolution of target materials and the core through the implosion.

Laser Mega-Joule (Le Barp, France)

The Laser Mega-Joule (LMJ) was developed by the Commissariat à l'Energie Atomique et aux Energies Alternatives (CEA) and is located at the Centre d'études scientifiques et techniques d'Aquitaine (CESTA) in Le Barp, France. Construction on the LMJ was started in 2003 and continues today. In addition to serving as a frontier instrument for the study of HED science and ICF, the LMJ is part of the French Simulation Program to improve physics models through experimental validation of high-performance numerical simulation (Miquel et al., 2016). The current design for the LMJ is to deliver a total of 1.4 MJ of energy with a peak power of 400 TW to a fusion target using 176 laser beams. At present, 40 of the planned 176 beams have been commissioned and are operational (Nicolaizeau and Miquel, 2019). Images of the LMJ facility are found in Fig. 2.

Similar to the NIF laser in the United States, the LMJ laser is a flash-lamp pumped, Nd:glass laser producing 15 kJ of energy per beam at the fundamental (1ω) wavelength of 1053 nm. The beams are organized into "bundles" of eight beams that are divided into two quads (four beams per quad). Each quad is transported to the target chamber where it enters the frequency conversion and focusing system (SCF) (Nicolaizeau and Miquel, 2019). Within the SCF, each beam is frequency tripled to 3ω , or 351 nm, and focused to TCC using the 3ω focusing grating. After frequency conversion to 3ω , each beam has approximately 9 kJ of energy. Like the NIF, LMJ will use both cryogenic and non-cryogenic target positioning systems to hold and position targets at TCC (Miquel et al., 2016).

LMJ is configured for experiments supporting laser-indirect-drive concepts, with the beams being positioned in cones in the upper and lower hemispheres of the target chamber. The target chamber is a 10 m diameter, 10 cm thick aluminum sphere containing 200 ports for both laser injection and diagnostics. Forty quads will enter the target chamber through ports located on the two cones at 33.2 degrees and 49 degrees polar angles. The other four quads will be positioned at a polar angle of 59.5 degrees for

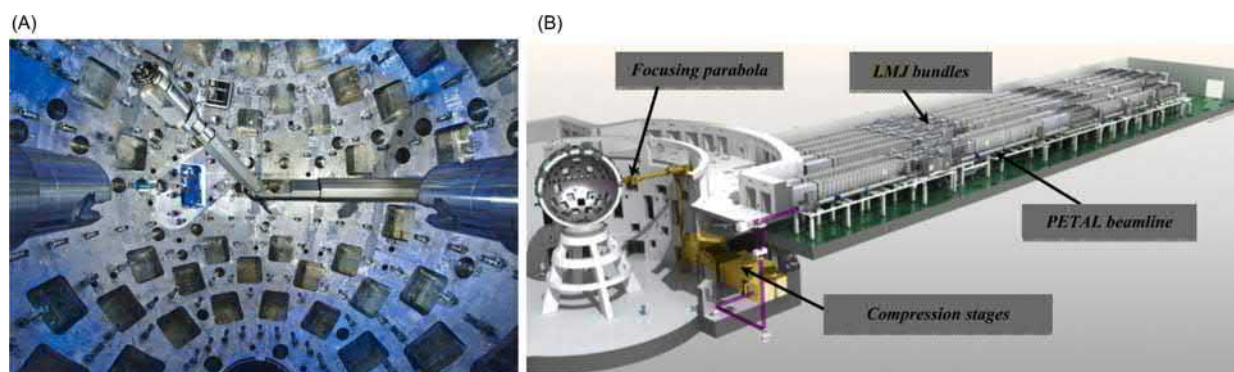


Fig. 2 (A) Image of the inside of the LMJ target chamber. (B) Cartoon schematic depicting implementation of the PETAL laser. Image (A) <https://www.science-et-vie.com/archives/laser-mega-joule-la-nouvelle-arme-de-dissuasion-nucleaire-40582>; (B) Nicolaizeau M and Miquel JL (2019) LMJ status: Fifth bundle commissioning and PW class laser coupling. In: *Proceeding of the SPIE 10898, High Power Lasers for Fusion Research* vol, 1089802 (4 March 2019). doi: <https://doi.org/10.1117/12.2507193>.

⁴Beta-layering occurs in a hot-spot filled with DT and cooled below the triple point. Self-heating due to the absorption of beta particles redistributes solid matter as a net sublimation of material from thick to thin regions takes place (Miller, 1976).

radiography (Miquel et al., 2016). The LMJ laser is capable of providing sophisticated, ignition-design-relevant laser pulse shapes having a minimum duration of 0.7 ns and maximum duration of 25 ns. Furthermore, as with the NIF, complex shapes such as a rising pulse, decreasing pulse, multiple pulse, with pedestal, etc. are possible (Casner et al., 2015).

An array of various photon and particle diagnostics are being considered and developed for LMJ. These include, but are not limited to, hard (keV) and soft (100's eV) X-ray imaging systems, a calibrated X-ray spectrometer, VISAR,⁵ Streaked Optical Pyrometer, a full aperture backscatter system, etc. To date, nine plasma diagnostics have been deployed and commissioned (Nicolaizeau and Miquel, 2019).

In addition to the long-pulse, compression lasers, the LMJ facility also hosts PETAL (PETawatt Aquitaine Laser). PETAL is short-pulse (500 fs to 10 ps), ultra-high-power (> 1 PW), high energy (> 1 kJ) laser. The PETAL design incorporates a mode-locked titanium-sapphire (Ti:sapphire) oscillator front end, and is based on chirped pulse amplification (CPA) combined with optical parametric amplification (OPA) (Nicolaizeau and Miquel, 2019). The PETAL beams are synchronized with the main LMJ lasers, and are injected in the equatorial plane of the LMJ target chamber using an off-axis parabolic mirror. The system was commissioned in 2015 with a performance of 1.15 PW, 700 fs, yielding 850 J. PETAL represents a significant capability for the LMJ facility as it extends the facility's diagnostic capabilities. Furthermore, LMJ-PETAL is open to research by members of the academic communities for 20–30% of the facility operating time, providing a rich resource for a wide array of HED science applications, particularly in laboratory astrophysics (Casner et al., 2015). Since 2014, more than 300 experiments have been fielded using LMJ and PETAL.

UFL-2M (Sarov, Nizhny Novgorod, Russia)

In 2012, the construction of the UFL-2M was initiated at the Russian Federal Nuclear Center, All-Russian Scientific Research Institute of Experimental Physics (RFNC-VNIIEF) in Sarov, Nizhny Novgorod, Russia. The UFL-2M facility is designed and scaled to achieve laboratory fusion ignition by delivering 2.8 MJ of laser energy onto a target using 192 beams (Rozanov et al., 2016). Unlike the NIF and LMJ, the UFL-2M facility will be configured for geometrically symmetric illumination of a target, accommodating the pursuit of direct-drive ignition concepts as well as spherical-indirect-drive designs.

The UFL-2M laser will be a Nd:glass laser, delivering 12.5 kJ per beam at the second harmonic (2ω), or 527 nm, an important distinguishing characteristic from NIF to LMJ, which use 3ω light. Completion and commissioning of the facility is anticipated in the first half of the 2020 decade (Rozanov et al., 2016).

Shen-Guang III & IV (Mianyang, China)

The Shen-Guang III (SG-III) laser is located at the Laser Fusion Research Center (LFRC) of the China Academy of Engineering Physics (CAEP) in Mianyang, China. SG-III is the largest laser ICF facility in China, consisting of 48 beams capable of delivering a total of 180 kJ, 60 TW of power, at 351 nm in 3.5 ns (Zheng et al., 2016). Construction of SG-III was initiated in 2007 and completed in 2015. SG-III was developed specifically for ICF studies (Jiang et al., 2019) in China. Images of the SG-III facility are found in Fig. 3.

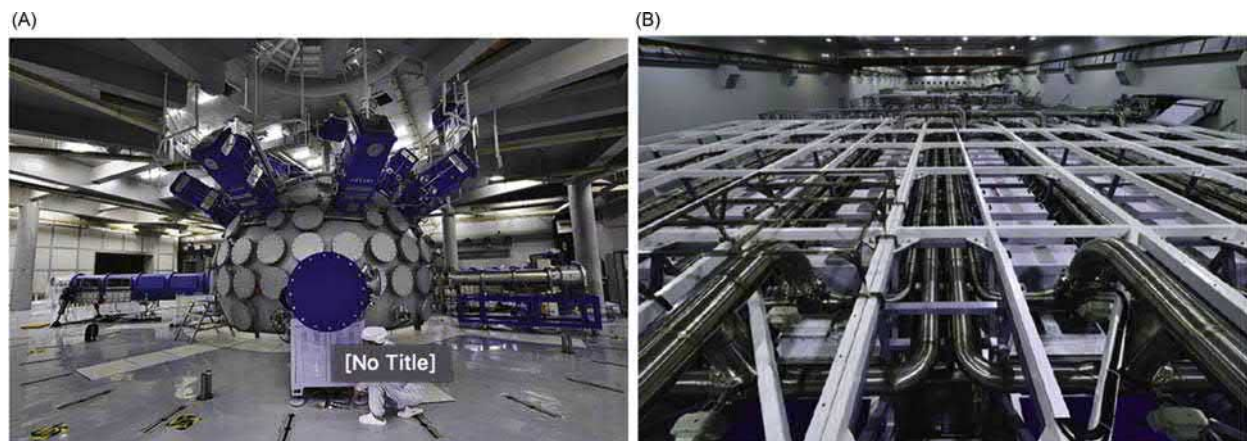


Fig. 3 (A) Image of the target area for the Shen-Guang III laser. (B) Image of the laser bay for the Shen-Guang III laser. Image Lan K et al. (2016) Progress in octahedral spherical hohlraum study. *Matter and Radiation at Extremes* 1(1): 8–27. doi: <https://doi.org/10.1016/j.mre.2016.01.003>.

⁵VISAR stands for Velocity Interferometer System for Any Reflector and measures the relative speed of a moving surface by recording its Doppler wavelength shift.

SG-III is a Yb-fiber seeded, Nd:glass laser produces 7.5 kJ of energy per beam at the fundamental (1ω) wavelength of 1053 nm (Zheng et al., 2008). The beams are transported to the target chamber, where the final optics assembly frequency triples the beams to 3ω , or 351 nm. After the frequency is tripled to 351 nm, each beam has approximately 3.2 kJ of energy. The SG-III laser systems have a pulse shaping capability for delivering ignition-design-relevant laser pulse shapes. Additionally, smoothing of the beams using a continuous phase plate (CPP), smoothing by spectral dispersion (SSD), and polarization smoothing (PS) have all been adapted (Zheng et al., 2016).

SG-III is principally configured to support research on indirect-drive target concepts. The beams are arrayed around the top and bottom of the target chamber surface in four cones having polar angles of 28.5 degrees, 35 degrees, 49.5 degrees, and 55 degrees (Lan et al., 2016). The target chamber is 6 m in diameter. The SG-III laser facility has in excess of 80 diagnostics (Jiang et al., 2019). These diagnostics include, but are not limited, to: soft X-ray spectrometers, transmission grating spectrometers, 4ω Thompson scattering imaging system, full aperture backscatter diagnostics, and optical diagnostic systems such as VISAR. For a more complete listing, see Ref. (Wang, 2020).

The Shenguang-IV laser has been proposed with an initial design goal of achieving 1.5 MJ or greater energy. While the design for SG-IV has yet to be finalized, the intention is for the facility to demonstrate the achievement of fusion ignition. Initial design concepts of the target chamber is to enable spherical illumination (He and Zhang, 2013; Zheng et al., 2008).

Omega (Rochester, NY, United States)

The Omega laser facility is located at the University of Rochester's Laboratory for Laser Energetics (LLE) in Rochester, NY, United States. The facility hosts both the Omega-60 and Omega-EP laser systems. The Omega-60 laser was last upgraded in 1995 (Soures, 1991; Boehly et al., 1995), while the Omega-EP system was completed in 2008 (Waxer et al., 2005). The Omega laser facility plays an important role in the development of experimental techniques, platforms, and diagnostics that are eventually fielded on the National Ignition Facility. The facility fields in excess of 2300 experiments a year, supporting both ICF and fundamental HED research. Images of the Omega facility are shown in Fig. 4.

The Omega-60 laser is a Nd:glass laser producing approximately 978 J per beam at the fundamental (1ω) wavelength of 1053 nm (Boehly et al., 1997). The beams are frequency tripled to 3ω , or 351 nm, before being transported to the target chamber. Following frequency conversion to 3ω , the resulting energy per beam is approximately 750 J. Thus, the 60-beam system can deliver more than 40 kJ with peak powers up to 45 TW (Boehly et al., 1997). The facility is capable of producing arbitrary laser pulse shapes from 0.1 ns to 4 ns in length. Additionally, the lasers employ an array of different beam smoothing capabilities including spectral dispersion (SSD) in two dimensions, polarization smoothing, and random phase plates. Omega has the most advanced beam smoothing capabilities of any implosion system. Such smoothing techniques are critical for direct-drive approaches to ICF, as spatial and temporal smoothing of the laser energy deposition onto the target is essential for minimizing the seeding of hydrodynamic instability growth.

The Omega-60 system is specifically configured for laser-direct-drive studies with the beams being arranged symmetrically around the target chamber with a fivefold rotational symmetry about the vertical axis (Boehly et al., 1992). The target chamber is 3.3 m in diameter and has a total of 92 ports; 60 beam ports and an additional 32 ports for diagnostics (Boehly et al., 1992). As a result of the role that the Omega laser facility plays in the development of diagnostics for the NIF, the facility hosts some of the most advanced particle and radiation diagnostics in the world for characterizing ICF implosions and HED plasma. In addition to the advanced diagnostics, the facility also has a pulsed magnetic field generator called the magneto-inertial fusion electrical discharge system (MiFEDS), capable of generating 15 T fields for the study of magnetized inertial fusion and magnetized-HED plasma science (Fiksel et al., 2015).

Alongside the Omega-60 facility, the LLE also hosts the Omega Extended Performance (EP) system (Waxer et al., 2005). The OMEGA-EP system has four, frequency-tripled (3ω), kilojoule class, independently configurable beamlines, two of which can be



Fig. 4 (A) Image of the target area for the Omega-60 laser. (B) Image of the laser drivers for the Omega-60 laser. (C) Image of the Omega-EP beam lines. Image (A) <https://www.lle.rochester.edu/index.php/omega-laser-facility-2/omega-laser-system/omega-experimental-systems/>; (B) <https://www.lle.rochester.edu/index.php/omega-laser-facility-2/omega-laser-system/omega-laser-drivers-2/>; (C) <https://www.lle.rochester.edu/index.php/omega-laser-facility-2/6271-2/omega-ep-beamlines/>

compressed for short pulse, petawatt-class operation enabling a wide range of experimental configurations (Meyerhofer et al., 2010). The integration of Omega-EP with Omega-60 provides an expanded capability for enhanced radiography as well as the study of fast ignition concepts.

As of 2019, the two short-pulse capable beams can provide 1ω , 1053 nm, light with up to 0.5 kJ in 0.7 ps, 1.25 kJ in 10 ps, or 2.3 kJ in 100 ps. When operated together, the two short-pulse beams propagate to two perpendicular off-axis parabolas in the OMEGA-EP target chamber, or alternatively are combined to co-propagate along a single axis. In any short-pulse configuration, the two remaining beams can provide simultaneous long-pulse operation for target preconditioning. It is also possible to transport the short-pulse beam(s) to the OMEGA chamber for integrated operation with the OMEGA-60 laser. Alternatively, all four of the OMEGA-EP beams can provide long-pulse (0.1–10 ns) operation to the OMEGA-EP target chamber. Each long-pulse beam is frequency-tripled to 351 nm, having energy up to 5 kJ, and with the ability to individually configure each laser pulse shape. Over 80 qualified diagnostics are available for experiments at the OMEGA-EP chamber. Three dedicated Target Positioning Systems (TPS) enable control for multi-target geometries.

Iskra 5 (Sarov, Nizhny Novgorod, Russia)

The Iskra 5 laser is located at the Russian Federal Nuclear Center, All-Russian Scientific Research Institute of Experimental Physics (RFNC-VNIIEF) in Sarov, Nizhny Novgorod, Russia. Completed in 1989, Iskra 5 is a 12 beam, 30 kJ (100 TW) facility created to study indirect drive target physics as well as the behavior of hot and dense plasma (Kirillov et al., 1990).

The Iskra 5 laser is a flash-lamp pumped, photodissociative iodine laser, producing 2.5 kJ of energy per beam at the fundamental (1ω) wavelength of 1316 nm (Kirillov et al., 1990). The beams are arrayed around a target chamber measuring 2 m in diameter. In 2002 the Iskra 5 laser was modified with a conversion to the second harmonic at 657.6 nm, giving it the capability of operating at either 1ω (1316 nm) or 2ω (657.6 nm) wavelength (Annenkov et al., 2003). The energy in each beam at 2ω is 220 J in 5 ns, and thus, when operating at 2ω , the facility can deliver a total energy of 2.6 kJ in 5 ns to a target. The facility hosts an array of diagnostics for measuring X-ray radiation, studying flows of charged particles, neutron counters, and plasma temperature (Gus'kov et al., 2010; Kirillov et al., 2000; Abzaev et al., 1998).

Shen-Guang-II Upgrade (Shanghai, China)

The Shen-Guang II Upgrade (SG-II UP) is located at the National Laboratory on High Power Laser and Physics (NLHPLP) of the Shanghai Institute of Optics and Fine Mechanics (SIOM), Chinese Academy of Sciences (CAS), in Shanghai, China. The SG-II UP facility was designed specifically for studying ICF and, particularly, fast-ignition concepts, as well as other basic HED science. The SG-II Upgrade Facility began operation in 2016, delivering eight beams with a total energy of 40 kJ in 3 ns at 1053 nm and 24 kJ in 3 ns at 351 nm (Zhu et al., 2018). Images of the SG-II UP facility are shown in Fig. 5.

The SG-II UP nanosecond laser is a Nd:glass laser capable of delivering a maximum of 8.5 kJ in 5 ns per beam at the fundamental (1ω) wavelength of 1053 nm. The beams are transported to the final optics assembly located at the 2.6 m diameter target chamber, where the frequency is tripled to 3ω , 351 nm, before being focused to TCC using a wedged focus lens (Zhu et al., 2018). The SG-II U facility is configured for studying indirect-drive target concepts, with the beams being clustered near the polar regions of the top and

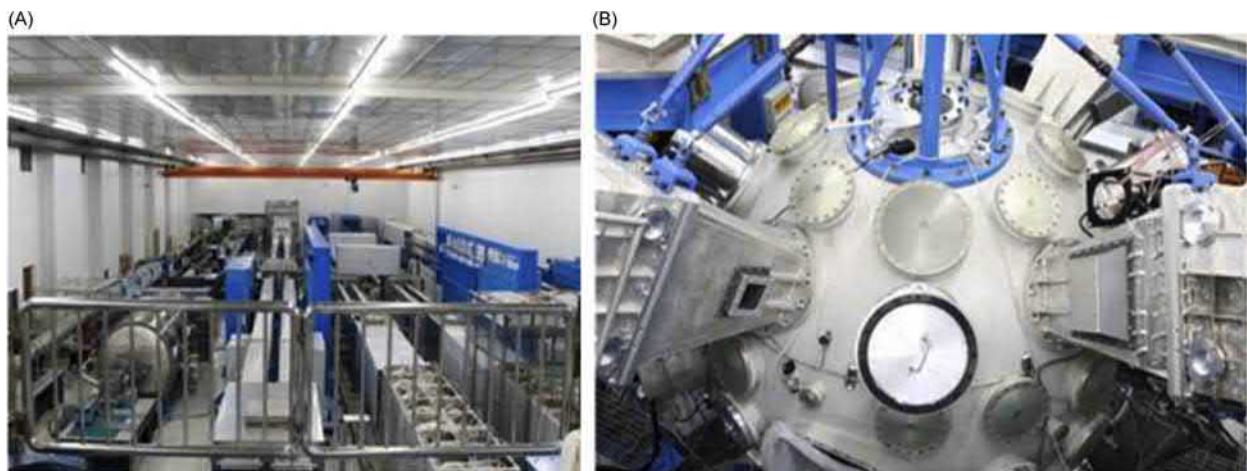


Fig. 5 (A) Image of the laser hall for Shen-Guang II UP laser. (B) Image of the Shen-Guang II UP target chamber. Image Zhu J et al. (2018) Status and development of high-power laser facilities at the NLHPLP. *High Power Laser Science and Engineering* 6: 1–23. doi: <https://doi.org/10.1017/hpl.2018.46>.

bottom hemispheres. Four beams are injected in each hemisphere, with the beams being equally distributed in the azimuthal direction and being situated on a single cone of polar angle 50 degrees (Tian et al., 2019).

The SG-II UP system supports a pulse shaping capability for delivering sophisticated ignition-relevant laser pulse shapes. The facility hosts a number of important ICF diagnostics, including hard X-ray imaging, X-ray spectrometers, and neutron time-of-flight (nTOF) (Zhu et al., 2018; Cui et al., 2021). Recently, the SG-II UP facility has added a pulsed magnetic field generator capable of producing an 8.8 T field using a double-turn, i.e., Helmholtz, coil (Hu et al., 2020), enabling studies of magneto-inertial fusion as well as magnetized HED plasmas.

Accompanying the SG-II UP nanosecond laser system is the SG-II UP PW picosecond laser system that delivers 1 kJ of energy in 1–10 ps at the fundamental (1ω) wavelength of 1053 nm (Zhu et al., 2018). SG-II UP PW employs a hybrid technology that combines optical parametric CPA (OPCPA) and Nd:glass CPA. This short-pulse, PW class laser system provides enhanced research capability to the facility as it expands the range of diagnostics that can be fielded, while also serving as an independent heating beam for fast ignition studies.

Gekko XII & LFEX (Osaka, Japan)

The Gekko XII facility is located at the Institute for Laser Engineering (ILE) at Osaka University, Osaka, Japan. The Gekko XII facility was completed and became operational in 1983 (Chiyo Yamanaka et al., 1981). The facility comprises 12 beams, capable of delivering approximately 10 kJ of total energy in 1.7 ns (Nakai et al., 1992), though, at present, routinely operates at approximately 4.5 kJ in 1.5 ns (Shiraga et al., 2011). Today, the Gekko XII facility houses two separate target chambers and maintains the capability to either symmetrically drive targets (Chamber 1) or illuminate a target from a single side (Chamber 2) at high power by combining the 12 beams into a single beam (Miyanaaga et al., 2000).

The Gekko XII laser is a Nd:glass laser producing 2.5 kJ per beam at the fundamental (1ω) wavelength of 1053 nm. The beams are frequency doubled (2ω) to 527 nm for routine operations, and are capable of delivering approximately 800 J per beam. The lasers are arranged with a dodecahedron orientation around target chamber 1. The facility maintains an array of plasma diagnostics for diagnosing implosions such as X-ray streak cameras, X-ray framing cameras, neutron spectrometers, and electron spectrometers.

Within the Gekko XII facility, the Laser for Fast Ignition Experiment (LFEX) (Miyanaaga and Kawanaka, 2011) has been developed and deployed as an auxiliary heating beam for fast-ignition studies as part of the Fast Ignition Realization Experiment (FIREX) project (Azechi, 2016). The LFEX laser is a four beam, Nd:glass laser based on CPA technology and capable of delivering up to 700 J in 1 ps per beam, or 2.8 kJ in 1 ps total, at the fundamental (1ω) wavelength of 1053 nm (Shiraga et al., 2015). In 2015, The LFEX laser achieved the record for being the world's most powerful laser when it output 2 PW of power. The record was recently broken in 2019 by the ELI-NP (Extreme Light Infrastructure—Nuclear Physics) laser that output 10 PW of power [https://optics.org/news/10/3/41].

Pulsed Power Facilities

Alongside laser facilities, pulsed-power instruments are the only other class of major ICF facility capable of imploding targets with fusion fuel. The term pulsed-power is a general reference to the use of high-power, high-current systems that utilize the Lorentz force to “magnetically” compress material. The technology used to create the drive current varies, ranging from conventional electric power storage in capacitor banks to the use of high-explosives. Here, the major pulsed-power drivers for ICF are reviewed and listed in Table 2.

Z-Facility (Albuquerque, NM, United States)

The Z-facility is located at Sandia National Laboratories in Albuquerque, NM, United States. In 2007, the Z-facility completed a refurbishment that delivered improved precision and laser pulse shape variability, increased the shot capacity, and increased the delivery current (Weinbrecht et al., 2007). In addition to ICF research, the Z-facility is a significant capability for a wide range of HED science, including radiation source development, radiation-driven science, and dynamic material properties (Sinars et al., 2020). Images of the Z-facility are shown in Fig. 6.

Table 2 List of major pulsed-power ICF facilities worldwide.

Facility	Location	Systems/type	Key parameters (Current/time)	Status
Z-facility	United States	Z-pinch/Marx	30 MA/100 ns	Operational
		Z-Beamlet/Nd:glass	0.6 kJ/0.2 ns/527 nm	Operational
Julong-I (PTS)	China	Z-pinch/Marx	10 MA/90 ns	Operational
Angara-5-1	Russia	Z-pinch/Marx	5 MA/100 ns	Operational

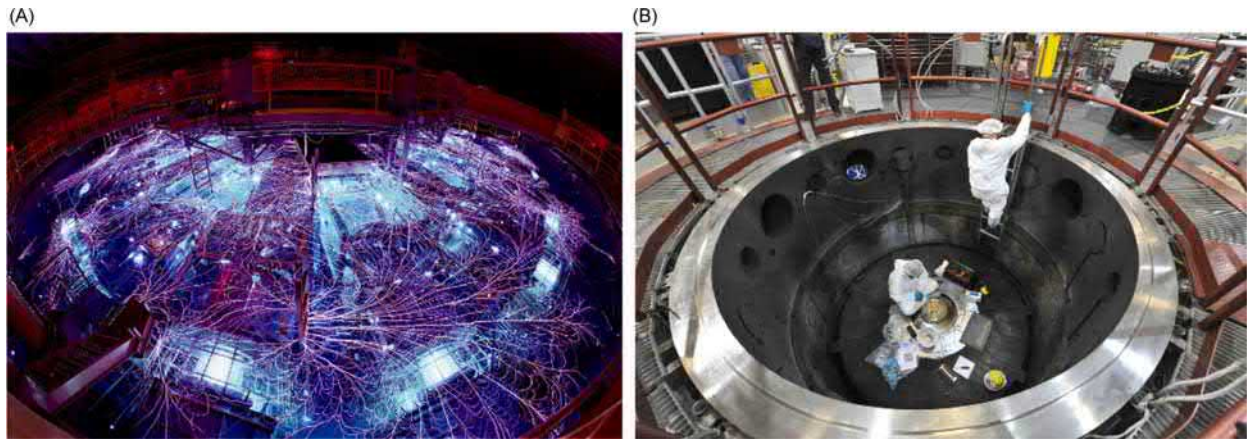


Fig. 6 (A) Z-Facility. Photo by Randy Montoya (B) Center load section of the Z-facility. Image <https://www.energy.gov/nnsa/articles/meet-machine-z-machine-creates-conditions-found-nowhere-else-earth>

The Z-facility can store up to 22 MJ in a ring of Marx generator capacitor banks around the perimeter of the machine, each of which is capable of storing 632 kJ. This energy can be discharged in 100 ns, and can deliver a peak current of up to 30 MA and peak power of 80 TW. This translates to the ability to deliver approximately 3 MJ of energy to a target and the surrounding load hardware (Sinars et al., 2020). The Z-facility maintains a wide array of X-ray and optical diagnostics, both spectroscopic and radiographic, for characterizing material behavior and properties.

While the Z-facility was optimized to drive the Z-pinch dynamic hohlraum for magnetic-indirect-drive ICF concept studies (Matzen et al., 2005), the leading ICF concept being developed on the Z-facility today is called Magnetized Liner Inertial Fusion (MagLIF) (Slutz and Vesey, 2012). Rather than the implosion of a wire-array target, as in the case of a dynamic hohlraum, the MagLIF concept entails the implosion of a solid metal “liner,” in conjunction with “pre-heating” of the fusion fuel by the Z-beamlet laser, described below, and the application of a background magnetic field to reduce thermal conduction losses from the fusion fuel thereby relaxing the compression requirement for achieving ignition.

In addition to the Z-pinch, the Z-facility also hosts the Z-beamlet laser (Rambo et al., 2005). The Z-beamlet laser is based largely on the Beamlet laser built by Lawrence Livermore National Laboratory as the prototype “unit laser cell” for the National Ignition Facility. Upon decommissioning of the Beamlet laser in 1998, it was moved to SNL for integration with the Z-facility. The Z-beamlet laser is a Nd:glass laser that delivers 0.6–2 kJ of energy in 0.2–2 ns at 527 nm (2ω). The Z-beamlet laser provides a powerful tool for both radiographing implosions on the Z-pinch, and heating the central fuel in MagLIF targets.

Julong-I, Primary Test Stand (Mianyang, China)

The Julong-I, Primary Test Stand (PTS) facility began operations in the 2012–13 time frame, and is located at the Key Laboratory of Pulsed Power, Institute of Fluid Physics, China Academy of Engineering Physics (CAEP), in Mianyang, China. The PTS is a multi-terawatt facility capable of delivering 8–10 MA of current with rise-time of approximately 90 ns (Deng et al., 2013, 2015, 2016). The system comprises 24 modules that include, among other things, a Marx generator, an immediate storage capacitor, as well as a laser-triggered spark-gap switch. The facility hosts a variety of X-ray and optical diagnostics for characterizing pinch plasmas as well as material behavior and properties (Deng et al., 2016). Images of PTS facility are shown in Fig. 7.

Angara-5-1 (Moscow, Russia)

The Angara-5-1 facility is located at the Kurchatov Institute, National Research Center, Moscow, Russia. The facility has been in operation since 1984 (Al’bikov et al., 1990). The Angara-5-1 is a Z-pinch facility capable of delivering 5 MA with a rise time of 100 ns (Aleksandrov et al., 2008). The facility comprises eight modules, placed radially, occupying an area 50 m in diameter. In the center is a concrete, radiation-shielding ring 10 m in diameter (Al’bikov et al., 1990). The facility has a maximum storage capacity of 650 kJ, and power of 9 TW (Al’bikov et al., 1990).

The Angara-5-1 facility routinely fields experiments using wire-array (Mitrofanov et al., 2020) targets for ICF studies, but also facilitates a wide range of HED (Grabovski et al., 2016) and materials (Aleksandrov et al., 2019) research. As such, the facility maintains a wide array of both X-ray and optical diagnostics, as well as magnetic probes, for characterizing pinch plasmas as well as material behavior and properties.

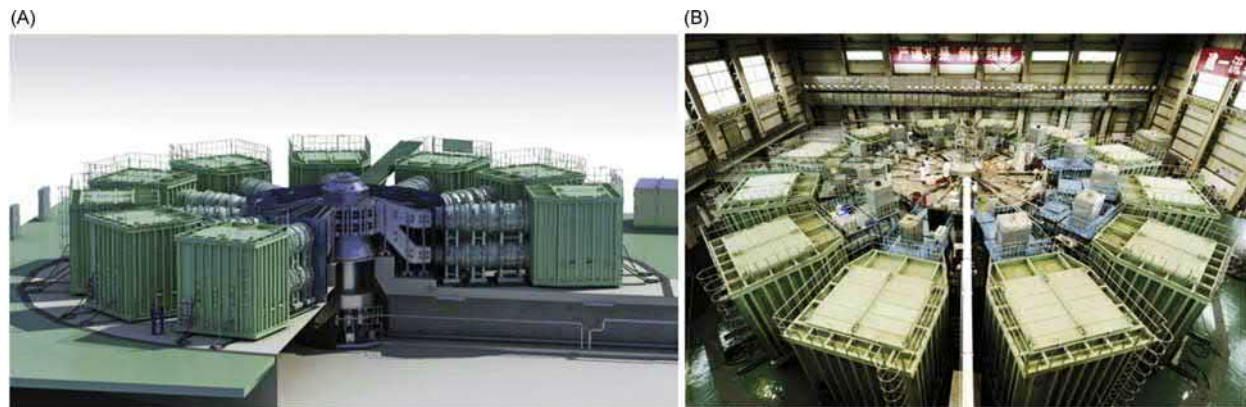


Fig. 7 (A) 3D cutaway schematic of the Primary Test Stand facility. (B) Image of the Primary Test Stand target area. Image Deng J et al. (2016) From concept to reality—A review to the primary test stand and its preliminary application in high energy density physics. *Matter and Radiation at Extremes* 1(1), pp. 48–58. doi: <https://doi.org/10.1016/j.mre.2016.01.004>.

Other ICF relevant physics facilities

In addition to the major ICF implosion instruments described in the preceding sections of this article, key physics R&D supporting the advancement of ICF science is executed on a wide array of smaller facilities. These facilities, while not considered major facilities for the purposes of this article, are providing essential data for the advancement of ICF and IFE. This includes facilities such as ORION (Hopps et al., 2015), LULI 2000 (Zou et al., 2008), and NIKE (Obenschain et al., 2015).

At this time, there are no major integrated facilities for prominent alternative approaches to IFE such as heavy-ion fusion. Nevertheless, research continues on facilities such as the Neutralized-Drift-Compression-Experiment II (NDCX-II) (Waldron et al., 2009), and will be featured at future facilities such as the Facility for Antiproton and Ion Research (FAIR) in Germany (Rosner, 2007), and the Heavy Ion Research Facility at Lanzhou (HIRFL) China (Zhan et al., 2008).

Lastly, the emergence of coherent X-ray light sources, such as the European XFEL [<https://www.xfel.eu>], LCLS [<https://lcls.slac.stanford.edu>], and SACLA [<http://xfel.riken.jp/eng>], coupled with high-energy and high-intensity optical laser systems, stands to provide unique and essential capabilities for unraveling, at a fundamental level, the complex interactions that occur in the environments in which energy couples to the capsule and is transferred to the fusion fuel in ICF targets.

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Inertial Confinement Fusion Power Plants

Mike Dunne^a, Tom Anklam^{b,*}, and Wayne Meier^{b,*}, ^a SLAC National Accelerator Laboratory, Stanford University, Menlo Park, CA, United States; and ^b Lawrence Livermore National Laboratory, Livermore, CA, United States

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The enduring commitment to nuclear fusion research has been motivated in large part by its potential to deliver large scale commercial energy production. Achieving this will require a deep understanding of the difference between achieving meaningful fusion gain in the laboratory and delivering commercially viable power plants. Issues of technology, economics, safety and public acceptance can make or break an energy source, even one with a sound scientific basis.

The purpose of the chapter is to discuss the requirements for commercial fusion energy systems and to describe some of the design tradeoffs involved. Because this is a chapter on IFE power plants, the focus is on pulsed systems operating above the threshold to achieve fusion ignition and gain (i.e., when there is a propagating burn wave that yields net energy output). However, much of the material is sufficiently general to apply to both IFE and MFE systems.

This article is divided into four sections:

- (1) A brief introduction of the basic principles and design considerations for an economic plant;
- (2) A summary of “end user” requirements that need to inform the design of power-producing plants and any intermediate prototype steps;
- (3) Explanation of how the different technology elements of the system interact with each other to impact the overall design and viability of the plant;
- (4) Example design studies and parameters for integrated IFE power plants.

Introduction

Plant electrical gain and minimum fusion gain

All power plants require electrical energy to run the various plant systems to extract energy from the fuel and deliver electricity to the marketplace. In most types of power plants, this energy is relatively low, a few percent of the output of the plant. But in fusion power plant designs the electricity required to run the plant is typically much larger, perhaps 20–30% of plant output, and is called the recirculating power, as illustrated in Fig. 1.

The reason that the recirculating power in a fusion plant is so high is that fusion is a threshold process. Typically, a large amount of energy is expended to heat and compress the fuel to achieve the conditions for fusion, and only after the ignition threshold is

*Visiting Scientist.

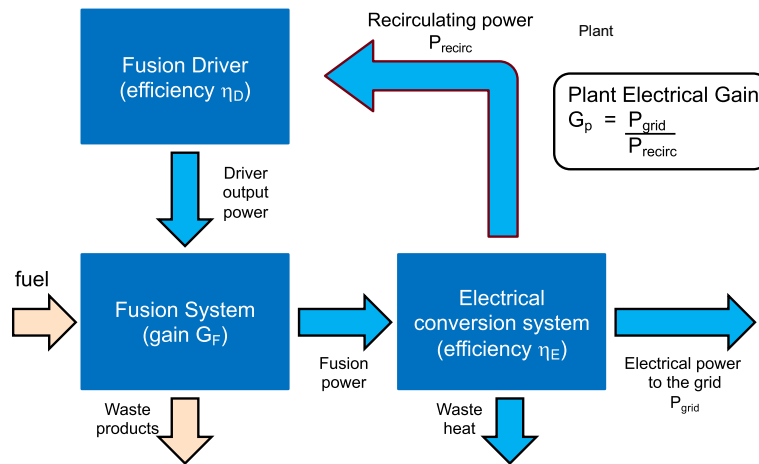


Fig. 1 Basic flow diagram for an IFE power plant, highlighting the fact that fusion power requires a non-trivial amount of electricity to generate electricity.

reached is there meaningful energy gain. Oftentimes, once threshold has been reached, the marginal cost of producing additional gain is low. As a result, the largest operating cost in the plant can be the cost of providing the driver energy needed to achieve threshold.

The relatively high recirculating power in fusion systems leads to a fundamental requirement concerning the total amount of power generated by the plant and that delivered to the grid. **Fig. 2** shows a basic calculation of relative plant cost as a function of plant electrical gain, for a plant with fixed electrical output to the grid. At gains less than about 3, the plant cost rises sharply. The reason is that as plant electrical gain becomes smaller, more and more of the plant electrical generation is being used to offset the power needed to run the fusion driver. In the extreme case of a plant electrical gain of zero (i.e., net breakeven), all the plant's output is being recirculated to operate the driver and nothing is sent to grid. Commercially viability is a complex topic, but as a general guide a fusion plant must have a net electrical gain greater than about 3. When one considers the likely electrical efficiency of the fusion driver and the systems that convert fusion energy to electrical energy, this leads to a requirement for minimum commercially viable fusion gain.

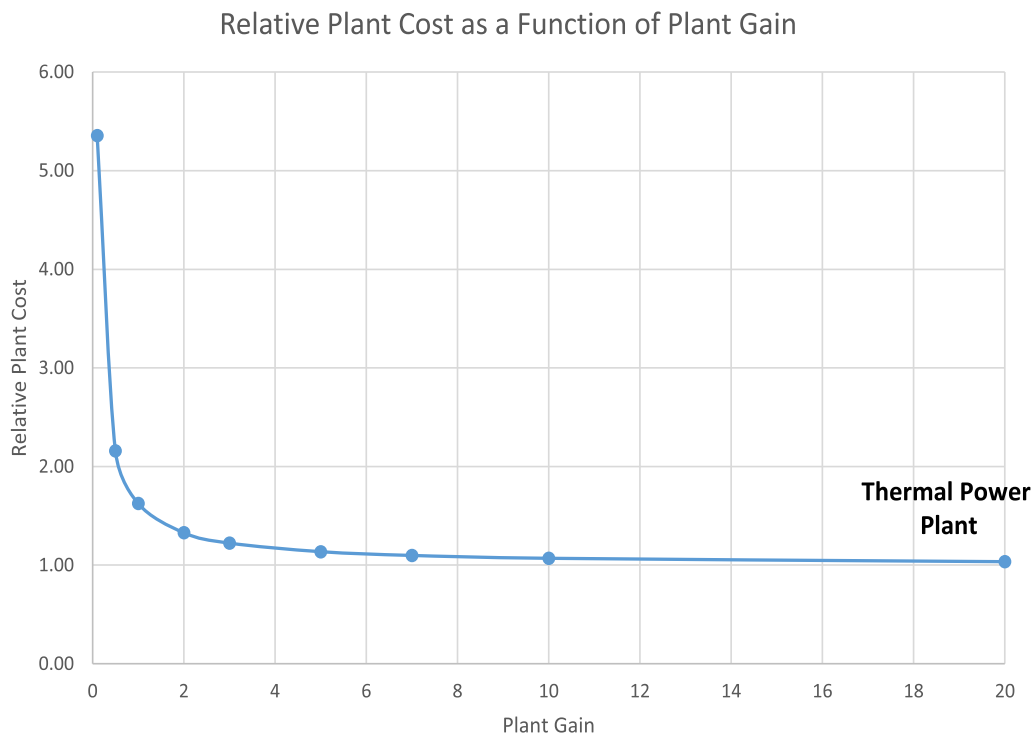


Fig. 2 Relative thermal plant cost (excluding driver cost) versus net plant electrical gain, G_p , which is defined as the ratio of the energy supplied to the grid to the energy needed to run the plant, $G_p = P_{\text{grid}}/P_{\text{recirc}}$.

The required gain from the fusion system (G_F) can be expressed in terms of the driver efficiency (η_D), the electrical conversion efficiency (η_E), blanket gain (G_B) and the net plant electrical gain (G_P). As an illustrative example, if we set the electrical conversion efficiency to 0.4, typical of steam turbine systems, the driver efficiency to 0.1, representative of laser drivers, blanket gain multiplication from exothermic reactions (M) of 1.1, then the required fusion gain required to close the loop is about 100. The required fusion gain, G_F , in this context means the ratio of the useful fusion output power to the incident driver power (noting that the useful output needs to account for any losses due to incomplete geometrical coverage by the blanket—for example due to driver penetrations—and whether the power associated with ion and X-ray output is able to be transferred efficiently to the heat exchange system).

$$G_F = \frac{(G_P + 1)}{M \cdot \eta_D \cdot \eta_E} \quad (1)$$

Eq. (1) provides required fusion system gain in terms of net plant electrical gain and major system efficiencies. The reasoning that underlies this conclusion is simple, but the conclusion is of profound importance to fusion power plant designers. In essence, the first question that the developer of a candidate fusion system should ask is whether there is a plausible case for achieving the minimum required fusion gain. This is a necessary, although not sufficient requirement for commercial viability. One consequence of this requirement is that, for pure fusion systems,¹ ignition and self-propagating burn is required in order to achieve adequate fusion gain for commercial purposes. In turn, this highlights a second important parameter, the threshold energy for fusion ignition, which will be considered in the next section.

Note that increasing the fusion gain offers the best chance of increasing plant gain, since (1) for a given driver type there is limited ability to increase driver efficiency, (2) the thermal conversion efficiency is limited by practical limits on coolant temperature, and (3) for pure fusion the blanket gain is limited to about 1.1–1.2. In contrast, fusion gain can be increased significantly by increasing the driver energy, but this comes at the cost of a more expensive driver. As a result, we typically find a minimum in the total cost of the power plant (thermal plant, driver and target factory) and cost of electricity as a function of driver energy (and corresponding plant gain) for fixed net electric output

Threshold energy, minimum plant size, minimum single shot fusion yield.

The threshold energy for commercially meaningful fusion gain is one of the most important parameters affecting fusion power plant design. Threshold energy, which is a function of the physics solution adopted in the target design, determines the minimum size for the fusion driver. Below-threshold drive energy produces no revenue. A large threshold energy requires a large fusion driver (with its up-front capital cost), which then requires a larger power plant in order to generate the revenue needed to pay for it. As a result, threshold energy is one of the primary determinants of minimum viable plant size and is the reason why many fusion power plant designs optimize at large plant sizes.

Less obvious is that, for IFE plants, threshold energy is a primary determinant of the minimum “single-shot” fusion yield. Recall that economic considerations require a minimum plant electrical gain, which in turn requires a minimum fusion system gain. Achieving the minimum required fusion gain requires going above the fusion threshold energy. Because single shot fusion yield is the product of driver energy and fusion gain, the threshold energy sets a lower limit to the fusion yield.

If the minimum single shot yield is high, it complicates the fusion chamber design. For example, consider a 1 GWe fusion power plant that has a 1 GJ (10^9 J) single shot fusion yield. One GJ is the energy equivalent of 220 kg of TNT, albeit with lower blast due to the small mass of the target. To generate 1 GWe would require 65 million such fusion events each year. The challenges to power plant mechanical and thermal designs are readily apparent. Plant designs with such high single shot yields often rely on the development of technologies such as flowing liquid metal curtains, in an attempt to deal with the thermal, mechanical and radiation consequences.

If the single shot yield is low, the target injection rate needs to go up in order to produce the sustained output required for a power plant. This makes the already difficult problem of target manufacture, injection, tracking, ignition and waste disposal even harder and perhaps out of reach. However, once threshold energy has been reached, single shot fusion yield can be increased by increasing driver energy, providing a path to an acceptably high single shot yield, if all the other terms can be made self-consistent.

It's worth noting that the conclusions just discussed are derived solely from economic and simple energy balance considerations:

- Economics require that the net plant electrical gain be in the range of three or greater so that the costs of operating the fusion driver are not overwhelming.
- Once values for driver and electrical conversion efficiencies are determined, one can specify a minimum fusion gain required to achieve the desired plant gain.
- Achieving minimum fusion gain requires operating above the fusion ignition threshold energy for the driver.
- Threshold energy is a primary factor in determining minimum economically viable plant size; high threshold energy requires large plant size to pay the cost of achieving ignition.

¹Fission-fusion hybrid systems can in principle achieve sufficient gain without ignition.

- Threshold energy, combined with minimum fusion gain, determines the lower limit to single shot fusion yield. Single shot fusion yield is a primary driver for the design of the fusion chamber.
- Finally, single shot fusion yield, combined with power plant output power, sets the requirement for target injection rate.

In summary, quite a lot can be learned about the commercial viability of different fusion energy schemes from just these simple considerations. We turn now to other considerations that drive the power plant design, and then assess how these requirements impact the choice and integration of different paths to fusion. At the end of the chapter, an overview is provided of different IFE plant concepts that have sought to address these challenges.

IFE power plant design considerations

End-user requirements from utilities and suppliers

For any complex system, it is essential to guide the development path with a clear set of end-user requirements, even if many aspects may not be able to be met in intermediate demonstration facilities. This is in order to maintain a robust understanding of how different options for technology choices or system configurations could affect the ability to progress to the end product, and thus to avoid dead-end research directions. In the case of fusion, these requirements largely derive from the long-term needs of the electric utility companies (or process heat users), along with the power plant vendors (and associated supply chain) and primary stakeholders such as the public and the regulatory agencies.

An illustrative starting point is the Utility Requirements Document for Advanced Light Water Reactors (EPRI, 1999). The comprehensive nature of this type of approach can be coupled with industrial perspectives of fusion power delivery requirements (Vondrasek, 1982; Kaslow et al., 1994; Dean, 1999; Kadak et al., 2012), analyses of the anticipated safety licensing requirements for fusion power delivery, and the commercial impact of different technology options for a power plant (Anklam et al., 2011; Goto et al., 2006).

Working from these criteria allows the power plant design process to focus on operational characteristics that are familiar to the end-users (utilities), delivery industries (vendors) and regulators. It allows trade-off decisions to be made on the wide array of possible development and risk-reduction activities, and enables down-selection of options consistent with the final goal of robust, economically competitive electricity production.

Criteria that drive the design process for an IFE power plant should include the following:

- i. Ability to supply baseload power:
 - This typically translates into > 100 MWe scale output with high availability ($> 85\%$ service factor), and the need for predictable shutdown and rapid restart (the importance of which scales with the power capacity of the plant). Ideally the power would be dispatchable (i.e., able to respond promptly to varying demand), if financially acceptable.
- ii. Ability to meet the highest levels of urban environmental and safety standards:
 - Economics and logistics preferentially drive the co-location of power plants with load centers such as cities, while public acceptability requires that there be minimal need for off-site emergency planning, and an operator safety level comparable to existing technologies. An acceptably small footprint for such a site requires detailed consideration of the consequences of hazards such as fire (which can disperse radioactive materials), a breach of the shield wall (which can lead to exposure to prompt radiation from activated materials in the reactor volume), and permeation of tritium into the local environment. In turn, these drive a wide range of system choices, detailed below, with examples such as chamber designs that minimize residual decay heat in the event of system shutdown, and target and chamber designs that permit small and segregated tritium inventories.
- iii. Licensing predictability and global deployability:
 - This is a complex and evolving area, and is perhaps the principal non-technical hurdle that could inhibit deployment if not addressed in a timely manner, as has been seen for many advanced nuclear fission schemes. Licensing considerations directly drive the plant design in areas such as radioactive materials inventory (from tritium, activated target and chamber materials, and byproducts in the cooling loops), as well as the blanket design (e.g., choice of coolant) and shield wall construction.
- iv. Capital build—duration, cost (\$) and intensity (\$/W), with acceptable return on equity (ROE):
 - The up-front (capital) cost for a fusion plant can present a major barrier to entry, particularly for commercial markets such as the US where the market capitalization of the utility companies militates against financing a multi-\$B construction project. The necessary presence of specialized equipment (e.g., lasers) and uncertain duration for licensing approvals impose constraints on the plant design that translate into physics parameters such as the need for high fusion gain (to minimize driver cost), and avoidance of novel nuclear-grade structural materials (that could challenge timely regulatory approval).
- v. Levelized cost of electricity, LCOE (\$/kWh):
 - Price-sensitive markets such as electricity generation drive fusion to large-scale systems (nominally > 0.5 GWe, with the upper bound set by an acceptable impact grid stability, which is typically < 2 GWe). This can be in direct conflict to the difficulties of financing such large plants (as noted above). System models point to the impact of the choice of structural materials (which drives the overall plant performance and maintenance regime); the cost of manufacturing

targets; the importance of high gain; and a surprising insensitivity to some parameters such as driver efficiency under certain conditions.

vi. Low environmental impact:

- A primary attraction of fusion, compared to conventional fission, is the prospect of a manageable waste stream of nuclear byproducts during operation and at decommissioning. This outcome is far from assured, requiring targeted development of low-activation structural materials, mitigation of first wall bombardment (that can lead to highly radioactive “dust”), and (for IFE) suitable choice of target materials.
- As a thermal energy plant (for conventional DT reactions), the use of water for cooling towers and the impact of local heating on the associated river/ocean, is an environmental consequence that can be mitigated by driving to higher temperature operations (and possibly direct electrical conversion in the long term).

vii. Protection of capital assets against operational risk and innovation:

- Often overlooked, this term can impact whether the overall approach to fusion would be acceptable to a commercial market. Failure modes need to be such that they do not hold the entire plant at risk of write-off or prolonged downtime. For example, the probability of an event that compromises the operability or structural integrity of the reactor vessel needs to be vanishingly small.
- Conversely, given the pace of progress in the high technology areas such as lasers, the owner/operator needs to be confident that they could upgrade their systems in the event of a disruptive advance in efficiency or cost (e.g., the ability to adapt to improved target designs).

viii. Robust supply chain, including fuel supply, technology availability, and materials scalability:

- The plant should be constructed from materials that have sufficient availability so as to not constrain the rate of rollout of a fleet, or overly limit its geographical extent.
- Where possible, the plant equipment should be transportable using conventional modes of shipping over public highways, including during decommissioning and waste disposal.

ix. Widespread availability of operations staffing:

- Widespread deployment requires a plant design consistent with an operations and maintenance (O&M) regime that can make use of a relatively conventional power plant workforce, with minimal requirement for scarce expertise (e.g., plasma physicists).

While a number of these criteria (e.g., cost of electricity) are difficult to predict in an absolute sense with high accuracy, the relative benefit of different design choices can still be assessed. For example, from a financial point of view, an option in which additional capital investment leads to a reduced cost of electricity can be assessed by comparing the operational savings with the cost of amortization of the increased capital.

Other criteria (e.g., availability and maintainability) drive fundamental design choices in the overall power plant architecture, sub-system configurations, and acceptability of certain technology. For example, as will be shown below, the impact of designing a laser sub-system that can be maintained while the plant remains operational is of over-riding importance. This allows the power plant availability to remain high even if the reliability or longevity of a critical sub-system is relatively low.

Overall, it is instructive to compare the anticipated market positioning of fusion with that of other current and emerging low-Carbon energy technologies, particularly for the primary use-case of baseload electricity provision (MacKay, 2008). Fusion has many inherent advantages due to the quasi-continuous nature of the power generation that is consistent with baseload supply to existing grids. It also offers the potential for security of supply if the plant design avoids the need for geographically restricted materials (such as rare earths), in addition to the avoidance of enrichment and reprocessing cycles that may be needed for fission.

Attributes of IFE to address the end-user requirements

The above “end-user” requirements can be translated into a set of attributes to drive technical design choices for an IFE power plant, at both a qualitative and quantitative level—with the latter needing system-level optimization models to assess aspects such as performance, cost, availability, and risk.

While the primary near-term focus of the fusion community is necessarily on achieving the desired thermonuclear performance, prime importance needs also to be on ensuring that the physics solution can ultimately be made consistent with a path to the required plant availability, maintainability and reliability, costs of electricity and construction, protection of capital investment, licensing simplicity, safe operations, and timeliness of rollout.

To this end, the following attributes can be adopted to inform any given plant design.

Compatibility of the integrated performance cycle with power production at scale

At its most simplistic level, there needs to be a self-consistent set of parameters for the major elements of the system, including:

- Driver² output and efficiency (from wall-plug to energy on target, accounting for all ancillary systems, beam transport losses, and sustained operating parameters).

²Throughout this chapter, the term “driver” is used as shorthand for the chosen pulsed device, whether Z-pinch, particle beam or optical laser.

- System repetition rate (which is likely limited by the reset time of the interaction chamber).
- Fusion target gain (energy output/driver energy incident), in which the output energy is only that fraction which is available to heat the thermal blanket.
- Blanket performance (accounting for the absorbed fraction of incident neutrons, and any inelastic reactions that can amplify the power), and in particular the need for tritium breeding at a rate to assure self-sufficiency.
- Thermal power cycle efficiency (accounting for any intermediate loops that may be needed for safety isolation of the primary circuit).
- Power needed for auxiliary systems ("house power"), mitigated by any energy capture from the low-enthalpy rejected heat.

An example power performance cycle is presented in Fig. 3.

Meeting the RAMI³ requirements for baseload power

It is inevitable that in a system as technologically complex as fusion, there will be failures or degradation of individual components throughout the life of the plant, and the need for periodic maintenance to ensure suitable longevity. It is likely that the mean time between failure (MTBF) for high technology components could be relatively short, particularly in the early phases of plant deployment. It is imperative that such events do not hold the power production at risk at a level that substantially impacts the overall plant availability, or in a manner that can lead to unpredictable reductions in power output that could compromise the stability of the electrical grid. Plant intermittency resulting from failure of individual components in any fusion system would likely be deemed unacceptable for a baseload source of energy.

The need to avoid relying on the integrated operation of a highly complex and coupled system should be a primary driver for any plant design. One primary mitigating approach is to configure the plant as a set of modular, parallelized Line Replaceable Units (LRUs) at a scale that allows single units to fail (or be taken out of service for maintenance) while allowing the rest of the plant to continue to function. Where possible, this approach should be used to decouple the *reliability* of high-threat, limited life components (such as final optics) from impacting the overall plant *availability*.

The quantitative requirements for each sub-system (e.g., laser box) will depend on their MTBF and mean time to repair (MTTR), along with any headroom designed into the sub-systems to compensate for non-working neighbors. See Fig. 4 for an example from the LIFE plant design (Dunne et al., 2011) that shows how power plant availability can be largely decoupled from sub-system reliability (Bayramian et al., 2011).

For IFE, this modular approach can help enable a separable development program in which the performance of individual sub-systems can be locally optimized. In the longer-term, it also allows for the creation of a cost-effective supply chain via off-site factory manufacture and truck-based delivery of the specialized equipment.

Availability of key materials

A primary materials problem is the first wall of the fusion chamber, which has to withstand intense and sustained bombardment of neutrons, X-rays and ionic debris from the exploding targets. The pace of fusion delivery could be driven in part by the long time-scales associated with advanced materials development and certification for high-threat components (e.g., low-activation, long-lifetime structural materials). Such advanced materials are still highly beneficial, but intermediate solutions would permit more timely deployment of an initial fleet. Some chamber designs may permit liquid walls, or wetted walls (if a self-consistent solution

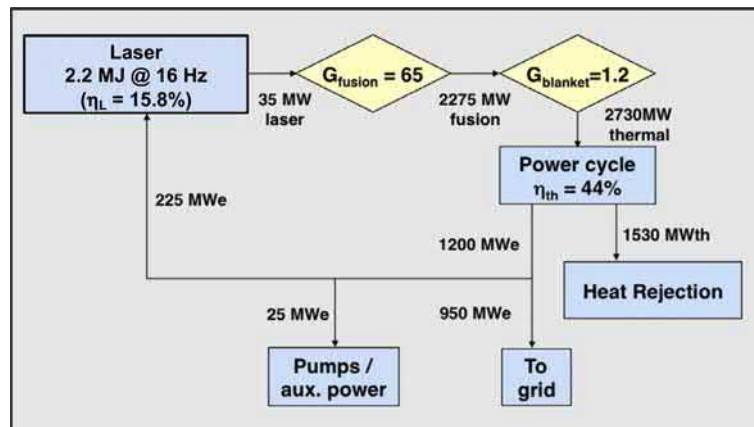


Fig. 3 Illustrative power cycle for a laser-driven IFE plant.

³RAMI stands for reliability, availability, maintainability, inspectability.

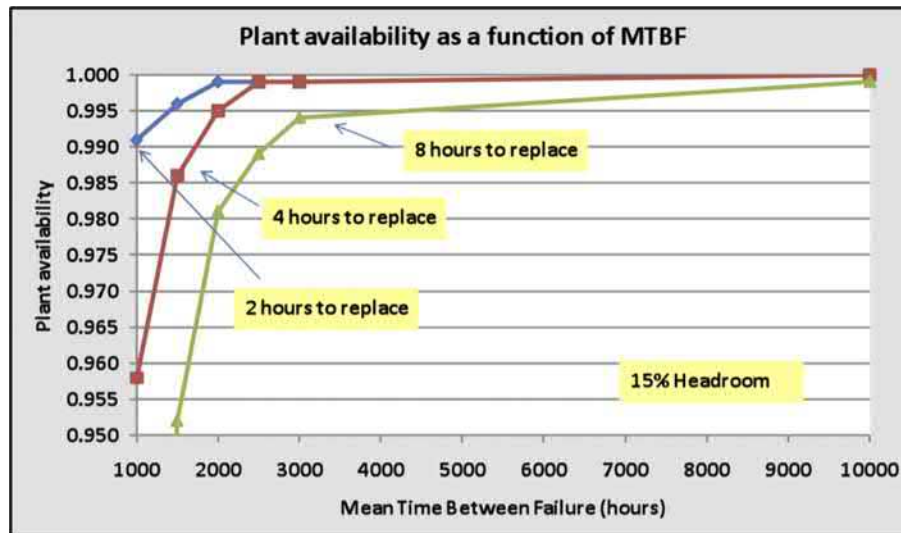


Fig. 4 Illustrative example of the impact of having a line replaceable unit “laser box” on power plant availability. The graph shows the result of a Monte Carlo simulation of plant operations as a function of beam-box mean time between failure (MTBF) and mean time to replace (MTTR). Plant availability is impacted by less than 1% even at very conservative failure rates for this design that incorporates a sufficient number of parallel line-replaceable units (384 laser boxes) and 15% headroom in total laser power.

can be found with the other sub-systems, as discussed below), but where this isn’t possible then means must be found to treat the first wall and blanket as Line Replaceable Units themselves, consistent with a few-year (rather than few-decade) lifetime.

Another key materials problem is the need for a non-trivial inventory of tritium for start-up and initial operations, with some fusion power plant designs requiring levels comparable to the total available global inventory. This presents a substantial obstacle to fusion power plant delivery. By virtue of the high fractional burn-up of fuel within an IFE capsule, then through suitable choice of target manufacturing approach, and selection of a high-solubility coolant in the blanket, the tritium inventory may be able to be reduced by an order of magnitude or more, to less than 1 kg per plant (Reyes et al., 2017).

As an example of a broader materials consideration, the design of the specialized systems (such as lasers and targets) should be compatible with low cost, mass markets production using materials that can be sourced around the world. Sub-systems tradeoffs between alternate designs should not just assess raw performance, but should account for the ability to scale-up and be deployed on a global scale at an acceptable cost and risk.

A primary example is the target design, for which exotic materials and manufacturing processes are acceptable for the “single-shot” proof of performance, but which may need substantial changes (and likely compromises in performance) in order to achieve high volume, low-cost targets that are also compatible with injection into the harsh environment of a cycled IFE chamber, and subsequent extraction and recycling of the inert target materials.

Criteria for intermediate demonstration systems

It is likely that one or more intermediate facilities will be needed to drive progress from an initial demonstration of physics performance (likely in a “single shot” mode within a highly instrumented and optimized science facility), to a prototype that can demonstrate integrated system performance in a power-plant-compatible mode (e.g., performance at a suitable repetition rate), to a demonstration plant that delivers net power output (likely in a non-commercial mode initially).

The integrated plant design and its associated technology program should explicitly define the required performance levels at each stage, and understand the nominal path of progression from one step to the next and ultimately to a commercial plant. Without this guiding principle, it is likely that dead-end solutions will be adopted whose removal may impose unacceptable design changes, costs and delay.

Example requirements that should be met *prior* to the design and construction of a demonstration power plant (in order to reduce risk to acceptable levels) are a function of the risk appetite of the investors, but could include the following, which includes the use of Technology Readiness Levels, TRL (NASA, 2007) as a quantitative metric of maturity:

- Demonstration of “single shot” physics performance at the required level (or within an acceptable margin based on high-assurance modeling).
- Advanced prototypes (TRL 6) for non-nuclear sub-systems (e.g., fusion driver module, target injector, target tracking/engagement).
- Validation in laboratory or relevant environments (TRL 4 or 5) for elements that have to exist in the locality of the chamber (e.g., final optics, structural materials, tritium recovery).
- Production path for consumables (e.g., fuel targets, short-life optics).

- Quantitative operating model for the fusion chamber, including non-radioactive demonstrations of important operations such as first wall replacement.
- Agreed upgrade paths for rapidly evolving sub-systems (e.g., driver efficiency, target performance).
- Indicative regulatory framework for plant operations.
- Nominal compatibility of sub-system requirements and anticipated performance.

Subsequently, an example set of requirements for a pre-commercial demonstration plant could include the following, with an overall objective to demonstrate integrated performance at a level needed to attract vendors and utilities for subsequent deployment:

- Achieve full system operation (fusion output, blanket operation, closed tritium cycle, waste management, consistent with coupling to a thermal balance of plant).
- Demonstrate a path to acceptable sub-system efficiencies, consistent with progression to overall power delivery to the grid.
- Sub-system demonstration in the required environment (TRL 7 or 8) for elements that need a working reactor (e.g., first wall, blanket, tritium recovery).
- Full operational demonstration (TRL 9) for non-nuclear subsystems.
- Quantify reliability and maintainability of components, consistent with projection to an operating power plant.
- Demonstrate ability for safe and environmentally sustainable operations (e.g., no impact outside of the plant; worker safety; operational model; waste stream).
- Test accident scenarios to level required by the regulator.
- Test inspection and certification methodology for the regulator.
- Determine anticipated rate and cost of build, and M&O costs.
- Identify material and sub-system supply chain for roll-out.
- Demonstrate a path to acceptable grid compatibility (e.g., start up, shut down, availability), and/or high-temperature applications compatibility (e.g., for chemical processing).
- Demonstrate path to an acceptable cost of electricity, capital build cost, revenue stream, and return on investment.
- Demonstrate path to low-cost manufacturing of targets at scale of tens to hundreds of millions per plant-year.
- Full design model for roll-out plan.

IFE power plant systems integration and interrelationships between the primary technology areas

The integration of individual technology components required for an IFE power plant can be depicted as a functional diagram, as in Fig. 5. This shows the main process flow via interconnected boxes, along with process support functions and the governing regulatory framework. Boxes shaded blue represent systems and processes at a level of maturity that substantively meet the needs of an IFE power plant today. Boxes shaded orange represent IFE-specific technologies, but which are largely within the bounds of the immediate fusion community to address. Boxes shaded yellow represent areas with technical and programmatic risk that largely rely on external factors to determine solutions and ensure progress.

In many ways, development of these sub-systems can proceed in a relatively independent manner (e.g., improvements in driver efficiency, or performance testing of fusion capsules), and this is a primary hallmark of IFE that has allowed substantial progress without the need to build power-plant-like facilities as is the case for magnetic fusion energy, MFE. But there are some key interdependencies imposed by the integration requirements of an IFE reactor. Examples are highlighted below, for which any plant design needs to ensure self-consistency, as well as an equitable distribution of tolerances (for efficiency, availability, cost, performance, etc.). These interdependencies include:

Driver (e.g., laser, ion beam, Z-pinch)

- *Performance* consistent with the target design: (i) at the single-shot level, (ii) at the required repetition rate, and (iii) with acceptable degradation over time. This typically translates to the need for 10–30% headroom in single-shot performance at the start of plant operations, given the reduction in damage threshold for optical components subjected to repeated operation. 10 Hz operation corresponds to a billion pulses every 3 years.
- *Wall-plug efficiency* consistent with the target gain and overall plant performance requirements. This dependency is non-trivial and highly system dependent. It is not linear since if driver efficiency is reduced but remains reasonably high (~10%), the plant electrical output just needs to be increased slightly to provide the additional power to run the driver.
- *Reliability* consistent with required plant availability. Even for a robust driver, the high capacity factor and predictability demanded for baseload power translate to the need for periodic maintenance *during* full plant operations, including “hot swap” replacement of limited lifetime components such as final optics. For multi-beam technologies like lasers, this requirement can conceivably be met by installing an excess number of beamlines, such that a fraction can be offline at any given time. For integrated technologies such as a Z-pinch, it drives the need for high reliability across the full system, with sufficiently high mean-time-between-failure and rapid replacement when needed.
- *Robustness* and *accessibility* consistent with the fusion environment. The requirement for any final optics to survive direct view of the fusion target in a sustained manner is a primary challenge for laser IFE. Degradation over time is common in most designs

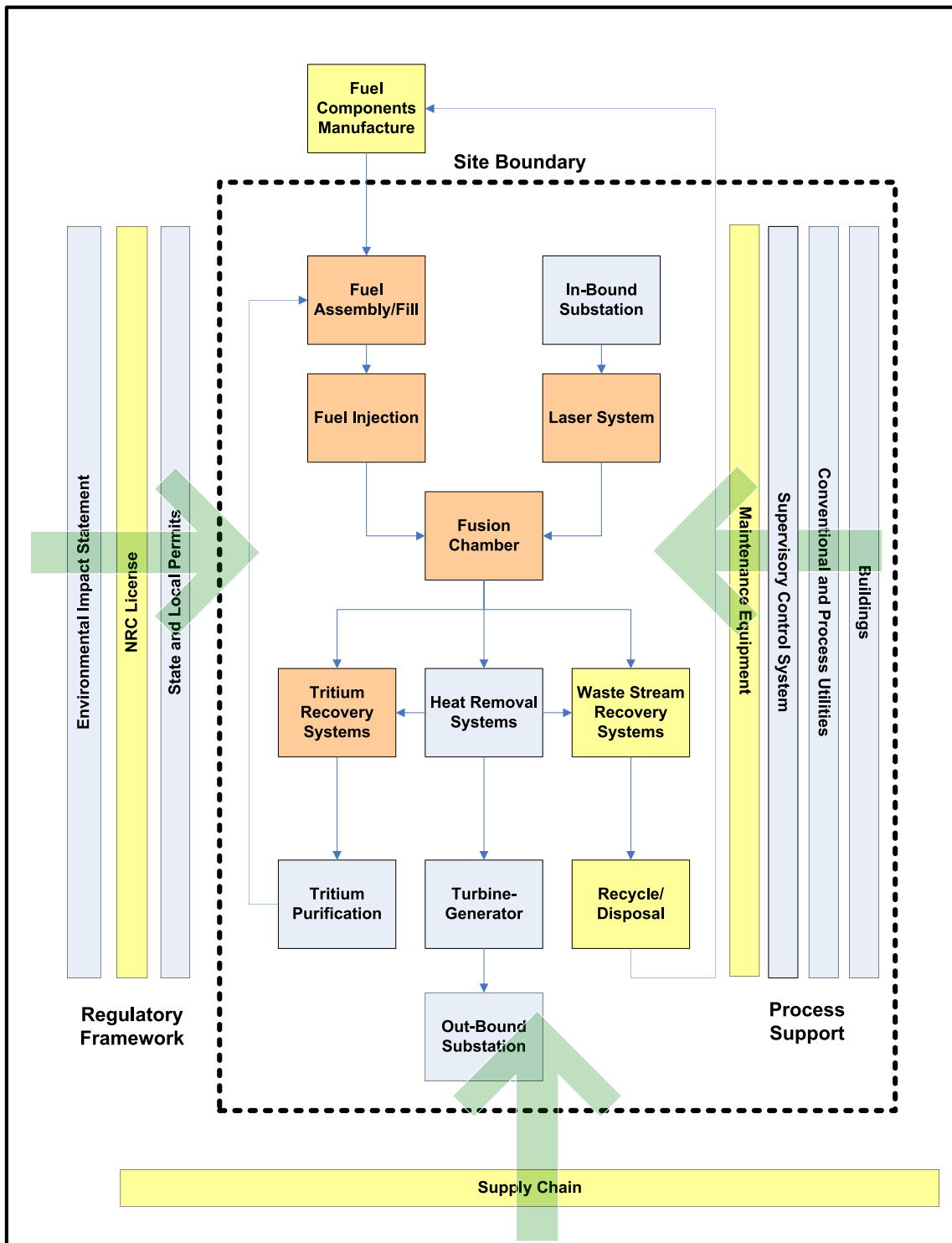


Fig. 5 IFE power plant functional diagram.

(with replacement cycles measured in weeks or months). But access to the final optics area is hindered by the extreme radiation environment during operations, the need for near full solid angle coverage by the chamber blanket system, and tight angular and position tolerances and delicacy of the optical components themselves. Three further constraints on the final optic system are that it needs to decouple the driver from the chamber vacuum; it needs to ensure tritium does not permeate into the laser system; and the placement of optics must be consistent with any chamber maintenance activities. For example, this could indicate a desirability for small transmissive optics confined to the polar regions, rather than large grazing incidence optics surrounding

the entire chamber. Similarly for pulsed power designs, there needs to be sufficiently low activation to enable robotic clean-up and replacement activities between shots and for periodic maintenance.

- *Beam pointing and focusing* needs to be resistant to the threat spectrum from the fusion reaction. One example is the formation of color centers in transmissive Fresnel optics from neutron bombardment that leads to laser energy loss and heating of the lens (Marshall et al., 1997; Latkowski et al., 2003). Another is the sensitivity of the wavefront to target debris on grazing incidence reflective optics, along with the difficulty in pointing and replacing such large components and the response of thin film coatings to neutron and gamma irradiation.
- *Beam propagation* consistent with the post-fusion chamber environment. The interaction of laser or particle beams with residual gas in the chamber and any deposits on the final optics needs to be benign, enabling suitably high transmission and beam quality on target. This is non-trivial given the sensitivity and tight tolerances of short-pulse beams propagating through fusion chambers whose scale size (hundreds of m^3) means that a timely return to pristine vacuum conditions is very challenging. This is further complicated by the complex byproducts from the target explosions and their inhomogeneous distribution around the chamber volume. In particular, chamber designs that call for the sustained presence of a residual gas to protect the walls from X-rays and ions, or that make use of liquid blankets or wetted walls, need to demonstrate consistency with the demands placed on beam propagation (as well as consistency with the injection of cryogenic fuel pellets, as discussed below).

Fusion fuel target

The primary focus of inertial fusion research has been on developing a target design that can demonstrate the required performance on a single shot basis. For an IFE power plant, this needs to be extended to incorporate the following:

- *Primary output*—The useful fusion gain and yield obviously need to be consistent with the requirements of the power plant cycle, and ultimately with an acceptable cost of electricity. The former translates to the focus on DT, not only because of its relative high reactivity, but also because of its high inherent gain (17.6 MeV fusion output from thermal ions at ~ 10 keV). In contrast, some “advanced” fuels do not have an adequate inherent gain for integrated plant operations. As an example, $p(^{11}\text{B},\alpha)2\alpha$ has an optimized once-through energy amplification factor of just ~ 12.9 , which is the ratio of the fusion output energy (8.7 MeV) to the energy of the incident particles (0.675 MeV at the peak of the fusion cross-section resonance). This low ratio is grossly incompatible with the requirement for overall power plant gain to be ≥ 3 once all the system inefficiencies are accounted for.
- *Acceptable byproducts*—The secondary output (X-rays, ions, chemical and particulate debris) needs to be consistent with the design of the wider system—typically the chamber first wall and final optics, but also the in-chamber chemistry and the waste processing system. This translates into a tight set of constraints on materials that can be used in the target. The high energy particle output needs to be tailored to be consistent with first wall survival; the target mass needs to be such that high momentum debris is not generated from unevaporated segments; the choice of elements needs to prevent the activation of long-lived isotopes; and the chemical byproducts need to be benign. As an example of the latter point, gold or uranium cannot be used since they would condense in the chamber and plate the inner walls (but these are currently the materials of choice for ICF hohlraums; alternatives such as Pb with a lower melting point would be needed). Similarly, carbon (an ablator of choice for many pellet designs due to its compatibility with manufacturing methods) can be highly problematic due to its ability to condense on the chamber walls and optics, form carbides that could embrittle the wall, and generate problematic compounds for the exhaust/waste systems, unless actively managed (DeMuth and Simon, 2009). The use of hohlraums (for indirect drive) or sabots (to protect direct drive pellets) with mass of order a gram translates to of order 1 metric ton of daily throughput, imposing demands on any storage and recycling system, including the need to treat any activated elements.
- *Mass production*—ICF targets used in research today are highly complex and expensive, requiring expert human intervention throughout the manufacture and assembly chain. For an IFE reactor, their designs need to be consistent with mass manufacture and delivery methods (~ 1 million per day per plant), and at a cost that does not preclude economic operation of the plant, which is typically $\ll \$1/\text{target}$ (Schultz, 1998; Goodin et al., 2005; Frey et al., 2007; Miles 2009).
- *Tritium inventory*—The ability to burn up a substantial fraction of the injected tritium fuel on each shot (tens of percent) is a key advantage of IFE, resulting in a reduction in overall tritium inventory in the plant. However, the target design needs to be consistent with such performance if this advantage is to be realized. Furthermore, the target manufacturing plant can account for a sizeable fraction of the onsite tritium inventory, impacting the safety stance of the facility, perhaps predicating the need for wetted foam designs rather than slow fuel filling processes such as diffusion.

Fusion fuel target injection

Inertial fusion research experiments use massive mechanical arms to hold the fuel targets at the center of the interaction chamber, and require extensive pre-shot alignment and characterization. For most approaches to an IFE plant, the fuel will need to be remotely injected in a manner that allows the pellet to arrive structurally intact at a well-defined location at a well-defined time and in the desired thermal state and orientation (if non-spherical). This imposes a number of new constraints on the target design:

- *Mechanical survival*—The target design needs to be consistent with the required acceleration in the injector without undue deformation, for example of the DT layer inside the capsule, or the position of a capsule within a hohlraum. This can be

challenging given the lack of strength of DT ice, and the need for extremely thin support membranes to avoid perturbing hohlraum-driven implosions.

- *Thermal survival*—The steady-state environment of an IFE chamber presents considerable challenges to the survival of a DT pellet, which needs to maintain its cryogenic state in the presence of radiation from the hot walls and conduction from any residual chamber gas. The pellet must preserve its spherical symmetry, surface smoothness and internal vapor pressure to a high degree of precision (Schultz et al., 2001). A principal difficulty is that the heat capacity of IFE pellets is rather small, such that less than 1 mJ of energy will heat the DT ice by more than the typical 100 mK limit. Increasing the driver energy can compensate for some level of imperfection, but this would be at considerable cost. Potential solutions have been explored, such as invoking a very low chamber pressure and low first-wall temperature (Petzoldt et al., 2002; Raffray et al., 2003) which must then be self-consistent with a chamber solution for first-wall protection. Alternatively, the use of a thermal shield can be invoked, either within the pellet, as a transient overlayer, or in a protective sabot. But any such scheme needs to be consistent with the constraints imposed by manufacturing (such as cost), injection (size and shape), target performance, and waste management constraints (activation and recycling), as is the case anyway for indirect drive targets (Miles et al., 2011). The timing of removal of any sabot or overcoat needs to be self-consistent with the in-chamber environment, with typical timescales for exposure of an unprotected pellet being < 100 ns (~1 cm of flight).
- *Tracking*—Precise timing of the driver is needed so that the target arrives at the chamber center synchronously with the driver energy. The gating event is likely to be the flight of the target, such that an in-chamber tracking system needs to be able to provide a timely trigger for the pulsed driver. The need for an active interrogation system may impose additional requirements on the target design, such as adequate reflectivity to a probe, as well as requirements on the chamber design to provide lines of sight, as well as protection of the tracking components. This may be incompatible with some approaches, such as the use of liquid sheets.
- *Placement*—The fuel needs to be placed at a location consistent with the focused energy of the driver. The absolute position may be relatively relaxed if there are active alignment systems for the driver, but the relative position tolerance (e.g., beam placement on the pellet or hohlraum) is typically very tight. Bare spherical pellets have some disadvantages with regard to their poor flight characteristics in a non-vacuum environment, and particularly in an inhomogeneous and potentially unpredictably turbulent gas. The thermalization provided by a hohlraum conceivably offers some lessening of the positional tolerance, although evidence to date is that beam placement is at least as difficult as for a spherical direct drive implosion.
- *Reliability*—The integrated target injection, tracking and ignition system needs to perform with a high degree of system reliability. Targets that do not ignite produce no energy and so degrade plant capacity factor. Targets that absorb driver energy but do not ignite can also produce high velocity debris that can damage other equipment. If ignition interruptions are sufficiently long, they can also create thermal transients that create cyclic thermal mechanical loads and challenge plant control systems.

Overall, the target survival, tracking and placement tolerances impose substantial self-consistency constraints on the overall design of the fusion chamber, driver and exhaust systems, as well as on the target design itself.

Alternative approaches that do not require injection (e.g., Z-pinch designs) impose stringent requirements on rapid, low-cost replacement of mechanical components, recycling of the irradiated byproducts, and often the need for multiple parallel interaction areas with all the associated infrastructure.

Fusion chamber

The heart of an IFE power plant is the chamber that serves to host the fusion reaction. This is a complex interdependent system in its own right that needs to: enable successful target and driver propagation to the chamber center; capture and transmit thermal power to the balance of plant; breed sufficient tritium; remove residual target debris; reset for the next pulse; and maintain sufficient longevity and availability for sustained multi-year operation. Many different conceptual approaches have been explored, and are summarized in the final section of this article. The chosen approach needs to be consistent with the other sub-systems, taking into account:

- *Chamber survival vs target survival*: A key topic that was underappreciated in some early designs of IFE chambers (Conn et al., 1977; Meier et al., 1994; Sviatoslavsky et al., 1993) was the need for self-consistency between mechanical survival of the first wall (which typically calls for a back-fill gas to range-out the problematic ions and X-rays, but not too high a pressure to induce blast wave damage) and thermal survival of the injected fuel pellet (which typically calls for as low a density as possible, particularly for direct drive approaches). Solutions have focused on the use of armor plated or nano-textured walls in very large chambers with a low-density gas, coupled to magnetic deflection of ions and aggressive exhaust systems (in designs that favor preservation of a simple pellet design); or the use of a more conventional steel wall in a smaller chamber with a higher-density gas for designs that invoke indirectly driven targets. There are also various approaches to incorporating liquid sheets (to protect the wall from X-rays, ions, and debris), for which the vapor pressure and potential presence of droplets must not interfere with the target survival, and which themselves must recover from the disruption from the target best prior to the next injection.
- *Chamber survival vs driver performance*: Similarly, any gas or liquid in the chamber needs to be consistent with high quality propagation of the driver beams (limiting the allowable gas density, heterogeneity/turbulence, and presence of droplets and condensation on optics, as well as placing constraints on the optical geometry for liquid sheet designs). Temperature stability of the optics is required in order to preserve the focusing and pointing characteristics, as is isolation of the laser systems from the

harsh neutron environment in order to enable a practical maintenance regime consistent with the overall plant availability requirements.

- *Chamber maintenance vs system integration:* Separate to the issue of first wall protection from ions and X-rays is the need for all the in-chamber components and associated structural materials to withstand the integrated neutron dose for a full maintenance cycle of the plant (which may be as short as a few years for line replaceable components, or as long as the lifetime of the power plant for structural components). The interdependencies here are with the overall concept for tritium containment and vacuum preservation in the chamber volume, likely predicated a separability of the first wall from the hermetic sealing system (Malang et al., 1998; Latkowski et al., 2011), and potential interference between periodic removal of chamber components and other sub-systems such as the final optics assemblies.
- *Chamber environment vs target design:* As noted above, the chemistry and radiochemistry of target by-products and any in-chamber gas and first wall materials needs to be self-consistent, avoiding unwanted condensation/plating; minimizing activated waste streams; ensuring an ability for timely reset to acceptable “ambient” conditions for the next pulse; and compatibility with cold-start/restart scenarios when the chamber temperature drops below its steady-state level and potentially below the freezing point of any residual chemicals.
- *Chamber design vs heat extraction system:* In the fluid-mechanics and structural design of the chamber and associated blanket/heat extraction system, there is a balance to be struck between: (i) running the thermal loop at high temperature for efficient heat extraction; (ii) the need to limit the temperature to allow use of cost-effective conventional steel piping and commercial turbines; (iii) avoidance of temperature regimes that accelerate corrosion in the primary heat exchange loop (which is a problem for liquid metals such as $\text{Li}_{17}\text{Pb}_{83}$ and salts such as FLiBe (a 2:1 eutectic mixture of LiF and BeF_2)); (iv) use of materials that provide sufficient strength at the high operating temperatures; that are resistant to neutron-induced swelling; and that can withstand the fatigue resulting from the pulsed thermal cycling inherent to IFE; and (v) choice of an efficient single heat exchange loop versus a dual loop to isolate the tritium.

Tritium system

- The inter-relationship between the chamber design, the associated breeding blanket, thermal loop, chamber exhaust, and the tritium processing and containment systems are all extensively covered elsewhere. These need to ensure there is overall tritium self-sufficiency (Abdou et al., 1986), and a safety regime consistent with regulatory and public acceptance requirements.
- In addition, there is a direct impact on target design due to the need to maintain an acceptable site inventory. There must be a sufficiently high burnup fraction ($\sim 30\%$), compatibility with rapid fill schemes (e.g., via wetted foams rather than diffusion and layering), and elimination of materials that could lead to tritium retention in the chamber (e.g. via the formation of metal hydrides with the walls, tritium implantation in the wall, binding to lithium, or formation of organic molecules via interaction with carbon from the target). Similarly, the exhaust chemistry must avoid tritium bonding into species that would compromise efficient reprocessing.

Waste stream processing system

- Every element of the plant design needs to account for its eventual disposition, and be consistent with the overall value proposition of the power plant to the owner and other stakeholders (Fetter et al., 1990). Particular consideration needs to be given to design choices that can result in day-to-day operational waste, and whether these will impose a need to store, transport and dispose of low-level nuclear waste (El-Guebaly et al., 2006). Examples are the chamber exhaust (resulting from byproducts from the target assembly, as well as any chemical and nuclear interactions with the gas and walls); byproducts from the tritium processing systems, and the fate of life-limited components from the nuclear environment that can be activated and/or tritiated (e.g., final optics, first wall panels, valve seals, etc.).

Safety and environment

- Similarly, every element of the design needs to ensure rigorous consistency with the eventual regulatory licensing requirements and corresponding accident scenarios (Reyes et al., 2001, 2016). This impacts a very broad array of facility subsystems, including: target design (e.g., due to on-site tritium inventory); chamber design (e.g., to minimize decay heat after reactor shutdown, and to limit radioisotope dispersal in the event of a fire, and if possible to mitigate the prompt gamma hazard in the event of a shield wall breach); coolant loop design (e.g., tritium permeation hazard); beam path design (e.g., to ensure sufficient shielding to minimize dose to workers); and the impact of maintenance cycles and waste stream options on onsite storage of hazardous materials.
- An important environmental design consideration for a success-based approach is to ensure the raw materials required by a worldwide fleet of fusion plants is consistent with known reserves, likely production rates, competing markets, and any geopolitical limitations. There is a well-known issue with the start-up need for tritium, but prior analyses have also shown that problematic raw materials may also include: beryllium (used in blanket designs as a neutron multiplier and/or coolant),

tantalum (used in some high temperature heat exchange designs, in power conditioning capacitors, and as a corrosion inhibitor layer when used with lithium salts), gallium (used in diode-pumped lasers), and enriched lithium (if needed for the blanket design). More parochially there would be intense constraints if materials were limited to domestic supply chains, leading to considerations of strategic reserves of key materials. Finally, the production rate of specialized components (e.g., optics, laser diodes, low activation and damage resistant steels) and the need for a suitably qualified workforce for the supply chain, plant construction, and plant operations are all nontrivial issues that can have a substantial impact on the growth rate and cost of the plants if not managed appropriately, as has been seen in other parts of the energy sector.

IFE power plant design studies

Many IFE design studies have primarily focused on the complexities of the fusion chamber. These typically assessed the choice of the chamber first wall and breeding blanket materials, chamber environment compatibility with delivering the driver energy to target, the response of the chamber to the pulsed fusion energy releases, and nuclear performance (e.g., tritium breeding, activation, radiation damage).

There have also been a number of integrated IFE studies since the 1970s that have sought to incorporate the full range of considerations for a power plant. These have typically been multi-year efforts with teams of physicists and engineers from universities, National Laboratories and private industry, including architect engineering firms with expertise in the design and construction of conventional fossil and fission electric power plants. Summaries are provided by Meier (1988), Craxton et al. (2015), Olson et al. (2005), as well as Davidson et al. (2013) for the National Academies, along with industrial perspectives from the Energy Power Research Institute (Hirsch et al., 1992; Kadak et al., 2012).

Table 1 gives key fusion chamber parameters for a sample of the major IFE power plant studies, organized by driver type. Heavy-ion-beam driven designs include Osiris (Meier et al., 1994), Prometheus-H (Waganer et al., 1994) and HYLIFE-III (Moir et al., 1994). Laser driven studies include Sombrero (Meier et al., 1994), Prometheus-L (Waganer et al., 1994) and LIFE (Dunne et al., 2011; Anklam et al., 2011; Meier et al., 2014), with the first two based on KrF lasers and LIFE based on diode-pumped solid-state lasers (DPSSL). Important work on integrated technologies for a direct-drive KrF laser plant was performed in the HAPL program (Sethian et al., 2010). Note that the Z-IFE power plant concept consists of 10 chambers, each operating at 0.1 Hz with yields of 3000 MJ, for a total plant thermal power of 3000 MWt (Olson et al., 2005). Details of the power conversion cycle were not given, so the thermal, gross electric and in-plant powers listed in the following tables were scaled from HYLIFE-II, a similar thick liquid wall chamber using FLiBe.

Table 2 summarizes the key power plant features and design parameters. Osiris and HYLIFE-II selected FLiBe as the primary coolant and tritium breeding material. The two Prometheus designs split the first wall and blanket cooling functions, using Pb for first wall cooling and high pressure He for cooling the Li₂O based tritium breeding blanket. Sombrero circulated Li₂O breeding pebbles and low pressure He through the blanket structures and then through steam generators. LIFE utilized liquid Li for tritium breeding and cooling of the chamber first wall and blanket. All these plants were sized for a nominal 1 GWe net output and all proposed conventional steam-driven power conversion cycles. Power conversion efficiencies ranged from 42–47%. The laser driven design have significantly higher recirculating power requirements due to the lower driver efficiency compared to heavy ion beams. As a result, the laser-driven plants have lower total plant gain (net electric output/total electric input) as discussed previously. Z-IFE Plants have significantly higher fusion threshold energy than for ion beam and laser-based systems, which leads to single-shot fusion yields roughly an order of magnitude higher than for the other two driver technologies. Another feature of the Z-IFE design is that it has 10 fusion chambers operating in parallel, all using the same driver. This allows each chamber to operate at a lower pulse rate, facilitating mechanical replacement of the fusion targets.

Table 1 Key fusion engine parameters.

	Year	Driver energy, MJ	Driver eff., %	Target gain	Target yield, MJ	Pulse rate, Hz	Fusion power, MW	Chamber first wall material	Chamber structural material	Breeding material
<i>Heavy ion beam driven</i>										
Osiris	1992	5.0	28.2	86.5	432	4.6	1987	C fabric	C	FLiBe
Prometheus-H	1992	7.0	18.5	103	719	3.54	2543	SiC	SiC	Li ₂ O
HYLIFE-II	1994	5.0	35	70	350	6.0	2100	Steel	Steel	FLiBe
<i>Laser driven</i>										
Sombrero	1992	3.4	7.5	118	400	6.7	2677	SiC	SiC	Li ₂ O
Prometheus-L	1992	4.0	6.5	124	497	5.65	2807	SiC	SiC	Li ₂ O
LIFE	2011	2.2	15.8	60	132	16.7	2200	Steel	Steel	Li
<i>Z-pinch driven</i>										
Z-IFE (10 chambers)	2006	30	15	100	3000	0.1	3000	Steel	Steel	FLiBe

Table 2 Key power plant parameters.

	Primary coolant	Primary coolant T_{max} , C	Plant eff., %	Thermal power, MWt	Gross electric power, MWe	Driver power, MWe	In-plant power, MWe	Net electric power, MWe	Plant gain
<i>Heavy ion beam driven</i>									
Osiris	FLiBe	650	45	2504	1127	82	45	1000	7.9
Prometheus-H	Pb & He	525 (Pb) 650 (He)	42.7	2780	1189	137	53	999	5.3
HYLIFE-II	FLiBe	650	43	2500	1075	85	50	940	7.0
<i>Laser driven</i>									
Sombrero	He + Li ₂ O	700	47	2891	1359	304	55	1000	2.8
Prometheus-L	Pb & He	525 (Pb) 650 (He)	42.3	3264	1382	349	61	972	2.4
LIFE	Li	575	47	2640	1217	248	64	905	2.9
<i>Z-Pinch driven</i>									
Z-IFE	FLiBe	680	43	3572	1536	200	71	1264	4.7

Further examples and references for specific IFE reactor designs include:

i. Laser IFE:

- Blascon (Fraas, 1971; Booth, 1973)—Early consideration of the approach to laser IFE.
- Solase (Conn et al., 1977)—Gas filled chamber and ceramic breeder.
- Prometheus-L (Waganer et al., 1994)—Laser variant of an integrated plant design providing various tradeoff analyses.
- Sombrero (Meier et al., 1994)—An integrated design using direct drive and a granular Li₂O blanket as the coolant and breeder.
- Sirius-P (Sviatoslavsky et al., 1993; Sawan, 1995)—A direct drive system with a dry wall chamber and SiC blanket cooled by a flowing granular bed of solid ceramic material.
- HiPER (Dunne et al., 2008)—A pan-European design study to integrate direct laser drive incorporating the “fast ignition” approach to target design.
- FALCON-D (Goto et al., 2009)—Laser “fast ignition” design.
- HAPL (Sethian et al., 2010)—An integrated program to develop technologies for a dry wall, direct-drive reactor.
- LIFE (Dunne et al., 2011)—An integrated design for a full-scale power plant based on indirect drive and performance that could be tested on the National Ignition Facility.

ii. Heavy Ion IFE:

- HiBALL (Böhne et al., 1982)—Thick liquid lead-lithium blanket in porous SiC fiber tubes.
- HYLIFE/HYLIFE-II (Blink et al., 1985; Moir et al., 1994)—Cylindrical chamber with thick liquid lithium first wall from a series of jets to provide lifetime survival of the structural elements.
- HIDIF (Hofmann and Plass, 1998)—European heavy ion plant design.
- Osiris (Meier et al., 1994)—Heavy ion counterpart to Sombrero.
- Prometheus-H (Waganer et al., 1994)—Heavy ion variant of an integrated plant design providing various tradeoff analyses.

iii. Z-Pinch:

- MagLIF (Slutz et al., 2010)—Concept for pulsed power performance consistent with IFE plant requirements.

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Fusion—Safety and Environmental Considerations

Neill P. Taylor, UK Atomic Energy Authority, Culham Science Centre, Abingdon, United Kingdom

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Abbreviation

ACP	Activated corrosion product
ALARA	As low as reasonably achievable
ASN	Autorité de Sûreté Nucléaire, the nuclear safety authority of France
BDBA	Beyond design basis accident
FMEA	Failure Modes and Effects Analysis
FMMEA	Functional Failure Modes and Effects Analysis
HAZOP	Hazard and operability study
HEPA	High efficiency particulate-air (filters)
HVAC	Heating Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
INB	Installation Nucléaire de Base (basic nuclear installation)
IVC	In-vessel component
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Analysis
RF	Radio frequency
SIC	Safety Importance Class
SSC	System, structure or component

Introduction

One of the main motivations for the development of fusion power for base-load electricity generation is its perceived favorable safety and environmental performance. This expectation derives from the characteristics of the fusion process, with the absence of hazardous reaction products and the strong tendency for a fusion plasma to terminate naturally in the event of any abnormality. But radiological hazards arise from the use of tritium as fuel, and from the products of activation by the high-energy neutron produced in the D-T fusion reaction (Morse, 2021; Campbell, 2021a). These hazards must be controlled and minimized.

Compared with a fission reactor, a fusion power plant will contain no actinides, no fission products, and no equivalent of reactivity excursions or core melt events. Accident scenarios are limited to events in which there is a degradation of the confinement of the radiological inventories of tritium and neutron activation products, potentially allowing some release to the environment.

Confinement of these radiological inventories must be maintained to prevent an accidental release that could pose a risk to the public, and also to prevent their spread within the plant and create an occupational hazard to personnel. Routine operation, including maintenance activities, must not result in significant gaseous or liquid effluents containing radioactive species, particularly tritium. Operation must not give rise to the accumulation of radioactive waste through replacement of components and, together with that generated at decommissioning, the waste must not become a burden on future generations.

These aims for the safety and environmental impact of a fusion power plant may be achieved by setting them as top-level objectives during the entire life cycle starting with the conceptual design, together with the adoption of safety criteria to check that they are fulfilled. One such criterion, commonly cited for fusion, is the ‘no-evacuation criterion’ that requires that there will never be the need to evacuate public from around the site, in any event no matter how unlikely. By reference to IAEA guidelines for evacuation (IAEA, 2014), this sets a quantitative limit on the consequence of any accident in terms of the biological dose expressed in units of milliSievert (mSv) (Apostolaei, 2021). The limit is 100 mSv maximum dose to the most exposed individual in 7 days (50 mSv is often assumed to provide a margin).

The safety design and operation of a fusion facility is commonly based on the application of recognized principles such as defense in depth and maintaining hazards as low as reasonably achievable (ALARA). Wherever possible, safety functions are provided by passive means, for example by employing strong static barriers that are part of the plant design (such as the vacuum vessel) to provide the confinement of radioactive materials.

In order to demonstrate that fusion can achieve its expected safety and environmental performance, safety studies of fusion have been performed to guide fusion power plant design in optimizing the safety implementation, and to ensure successful nuclear licensing for construction and operation. These include studies of conceptual designs for fusion power plants, particularly in Europe (Cook et al., 2005) and the US (Petti et al., 2006), and the extensive studies of ITER (Taylor et al., 2011) for the safety case presented to the French nuclear safety authorities in order to obtain the license to construct the facility in the south of France (Taylor et al., 2013).

Main safety and environmental concerns for fusion

Radiological hazards in a D-T fusion plant, whether magnetically or inertially confined, arise from two sources: the tritium used as fuel and generated in breeding blankets (Boccaccini, 2021), and the products of neutron activation.

Tritium hazards and their amelioration

Tritium is a pure beta-emitter with a half-life of 12.3 years. The beta particle has a low energy, averaging 5.7 keV. This particle has a range of only 6 mm in air, or typically 6 μm in organic matter. The principal radiological impact is therefore not from exposure but from inhalation, ingestion or through skin absorption. The biological hazard coefficient for ingestion of tritium is 4.2×10^{-11} Sv/Bq, some three orders of magnitude lower than that of ^{131}I . The difficulty with tritium arises from its nature as an isotope of hydrogen; it diffuses readily into and through materials, leading to retention of tritium in structures and challenges in its confinement. Furthermore, it exchanges with hydrogen in molecules such as water, creating tritiated water HTO, and also in organic matter and other hydrogen-containing material.

Significant inventories of tritium are likely to be found at many locations in a fusion power plant:

- Retained in the vacuum vessel, where it will be adsorbed onto surfaces, permeated into the structure of components inside the vessel (in-vessel components, IVCs) and absorbed into dust generated by erosion of surfaces;
- In equipment used to fuel the plasma, to pump exhaust from the plasma chamber, and to recover and process tritium from that exhaust, collectively known as the fuel cycle system;
- In breeder blankets where it is generated, together with the system to extract tritium from the blankets and to process and store it for use as fuel;
- As contamination of remote maintenance equipment used to remove components from the vessel and transport them to a maintenance facility;
- In the storage facility for in-vessel components awaiting maintenance or disposal;
- In the maintenance facility where components are treated, repaired, dismantled and prepared for disposal;
- In coolants, due to permeation through cooling channel walls;
- In the atmospheres of any rooms containing tritium-contaminated items.

The hazards presented by these inventories are three-fold: the risk of inhalation by plant personnel (i.e. occupational radiation exposure), routine effluents in gaseous or liquid form during normal operation or maintenance, and the risk of an accidental release to the environment in an off-normal situation. The approach to minimizing each of these hazards is the same: reduce tritium inventories and provide effective confinement. Minimization of inventories needs to be done at the design stage, by designing systems that can perform their function with the lowest possible quantity of tritium, avoiding features that may lead to tritium retention (including the choice of materials), and segregating tritium inventories so as to limit the maximum quantity that could be mobilized in case of a failure or degradation of confinement systems.

Confinement systems to prevent or minimize occupational radiation exposure utilize room ventilation based on nuclear-grade HVAC (Heating Ventilation and Air Conditioning) systems. These maintain a pressure cascade between adjacent rooms to ensure that air flow is always in the direction towards the more highly contaminated spaces. Conventional nuclear HVAC systems remove airborne contamination by filtering the air before it is recirculated or vented to the environment. However, they cannot remove tritium. To be able to do this, large and complex atmosphere detritiation systems are required typically based on a series of scrubber

columns. Such systems have limited maximum flow rates compared with the regular HVAC systems, for which reason flow is likely to be diverted to them only when a tritium concentration above some threshold is detected in a room.

For the minimization of routine effluents and of accidental releases, confinement systems comprise a series of strong barriers together with the same ventilation and detritiation systems. Providing the confinement function by passive means is an important objective, to reduce dependency on active safety systems. Nevertheless, the atmosphere detritiation systems mentioned above will always be necessary to reduce tritium concentration in effluent being vented to the environment. Where possible, strong barriers that are present in the design are utilized as confinement barriers, for example the vacuum vessel itself to protect the in-vessel inventory of retained tritium. Such barriers, however, may have a number of penetrations; in the case of the vacuum vessel these include cooling pipes, ducts for neutral beam systems, waveguides for RF heating systems and diagnostics, feeders for in-vessel coils, fueling and pumping systems. At each of these penetrations, seals, bellows, windows, isolation valves and parts of the ducts or pipework become part of the confinement barrier and need leak-tightness with high reliability.

Neutron activation and its consequences

14 MeV neutrons from the D-T fusion reaction (Campbell, 2021a) penetrate all structures and components around the plasma and are slowed-down through scattering reactions, resulting in a broad energy spectrum down to thermal energies, but always with a dominant peak at 14 MeV. This moderation to low energies is deliberately promoted by typical breeder blanket designs, since the major tritium-producing reaction in lithium-6, ${}^6\text{Li}(n, \alpha)\text{T}$, has its highest cross section at thermal energies.

During the transport and slowing-down of these high-energy neutrons, they take part in many reactions that cause transmutations in the materials of the plasma-facing components, and to a lesser extent the vacuum vessel and components outside of it. The high-energy part of the neutron spectrum enables many reactions with threshold energies that leave them unimportant in fission reactors, including charged-particle producing reactions such as (n,p), (n,d) and (n, α) (Wietfeldt, 2021). The outcome, for irradiation of typical stainless steel, is a material composition comprising hundreds of nuclide species. These include many radioactive nuclides which may be broadly classed into three categories according to their half-lives:

Short-lived nuclides that decay on the timescales of minutes or hours. These may provide intense gamma radiation as they decay, but only in the short term. They are of potential importance in postulated accident scenarios involving a loss of confinement; if they may be mobilized and released from the plant, they may provide radiological doses to members of the public in the vicinity of the plant.

Medium-lived nuclides with lifetimes of days or weeks. Their gamma radiation has the potential to cause occupational doses to personnel involved in maintenance activities when the plant has been shut down. To avoid such exposure, remote handling techniques will be deployed for the removal and replacement of components including their transport to the maintenance facility. The remote handling equipment must be qualified to operate in the high radiation fields.

Long-lived nuclides with half-lives of months, years or longer. These constitute a large volume of active material that must be handled and stored during final plant decommissioning and dismantling. The longer-lived nuclides will result in some of this material, together with that of components removed during operation, ultimately requiring disposal as radioactive waste.

One important approach to minimizing the impact of neutron activation is through the choice of materials in those components exposed to the neutron flux (Litnovski, 2021). Specially designed materials for fusion systems have been developed as alternatives to conventional steels (Tanigawa et al., 2017). Referred to as low-activation or reduced-activation materials, they have a composition that omits or minimizes those elements whose activation products are dominant. It is also important to restrict the levels of some impurities in steels, as these often contribute significantly to the activation. For example, cobalt, a typical impurity in stainless steel, generates ${}^{60}\text{Co}$, which has a 5.3 year half-life and emits high-energy gammas. At typical cobalt concentrations in conventional stainless steels, ${}^{60}\text{Co}$ dominates the gamma dose rate arising on the timescale from 6 months to 50 years after shutdown.

With regard to activation products providing a potential source term for release in postulated accidents, the degree to which the active material may be mobilized must be considered. Although much of the active material is stable inside solid structural or other materials, some of it may be mobilized if the accident scenario involves high temperatures. Furthermore, the plasma-facing surfaces become eroded during plasma operation generating dust that accumulates inside the vessel and may be easily mobilized in some accident scenarios. This dust is not only radioactive but may also absorb tritium.

Active dust accumulated inside the vessel, and any other mobilized activation products, require confinement by exactly the same means as those discussed above for tritium. The important difference compared with tritium is that airborne dust may be readily removed in ventilation systems by the use of high efficiency particulate-air (HEPA) filters.

Although in normal operation the products of neutron activation are well confined inside the tokamak and their radiation blocked by a biological shield, there is one significant exception. In water cooling circuits corrosion of the surface of coolant channels, pipework and components of the cooling system produce corrosion products that are carried around the cooling loop. These are largely in solution in the water, but also include some particulate matter. If the origin of the corrosion is material in active components these products are a source of gamma radiation, and material originating in any part of a cooling loop becomes active when it is carried by the coolant through the region under neutron flux. The result is an inventory of activated corrosion products (ACPs) that are transported throughout the cooling circuit, including those parts outside of the bio-shield. Some of this ACP inventory deposits on surfaces throughout the loop so that, even after the water is drained for maintenance, the pipework and components provide a significant source of gamma radiation.

Looking further into the future it is conceivable that instead of D-T fusion, an aneutronic fusion reaction could be employed, i.e. one that does not directly produce neutrons. However, reactions of this type, for example $D-^3\text{He}$, all require conditions that are more difficult to achieve, and some production of neutrons cannot be precluded due to other reactions, such as D-D, also taking place within the plasma.

Implications from safety analyses

Studies of the safety of fusion facilities are done during design activities in order ensure that the design includes all required safety provisions and that these are optimized to reduce the safety and environmental impact of operation. Safety analyses not only provide feedback to the design process, but also prepare essential information used in a safety case as part of the licensing process to authorize construction, commissioning and operation.

Safety functions and classification

It is essential to define the functions that must be provided to ensure that the facility remains in a safe state, even in off-normal conditions or with the failure or malfunction of one or more systems. For fusion, two top-level safety functions are generally defined:

- The confinement of radioactive materials;
- Limitation of exposure to ionizing radiation.

For the first of these, every inventory of tritium and activation products, as outlined in Section “**Main safety and environmental concerns for fusion**” above, must be protected by a series of strong barriers. These prevent the spread of radioactive contamination within the plant and ultimately avoid or limit the amount released to the environment in accident conditions. The second safety function is focused on protecting personnel from radiological doses, and is achieved by means of radiation shielding coupled with strict access control to hazardous areas.

To ensure that these safety functions can be provided in all plant conditions, a set of supporting functions are defined, for example the control of energies that might present a challenge to confinement barriers.

The safety functions and their supporting functions are provided by systems, structures and components (SSCs) in the plant that have a role to play in maintaining safety. Such SSCs are classed as importance to safety, and have a Safety Importance Class (SIC) assigned to them according to the consequences if they fail to provide their function. The highest level, SIC-1, is assigned to SSCs that are needed to ensure that the plant remains in a safe state, or is returned to a safe state following the initiation of an accident scenario. Lower levels, SIC-2 and sometimes SIC-3, indicate an SSC that provides mitigation of the consequences of an accident, but whose function is not essential to reach a safe state. Quantitative definitions of the SIC levels are commonly used.

The classification of SSCs as SIC implies a set of requirements for their design, manufacture, qualification and monitoring or inspection during service. It also allows their correct functioning to be credited in analyses of postulated accident sequences. If, in the course of those analyses, it is found necessary to assume that some SSC behaves in a particular way to obtain a favorable outcome, then this behavior becomes a requirement and the SSC is classified as SIC.

Accident analysis for fusion

A central part of safety analysis is the evaluation of the possible consequences of postulated accident scenarios. This is done by detailed computer modeling of the systems that could be involved, and their response to the abnormal situation. Usually, for fusion systems, this is done using deterministic accident analysis. It involves the determination of source terms, i.e. the quantities and characteristics of the radioactive inventory vulnerable for release, which typically requires neutronics and activation modeling to establish full nuclide-by-nuclide compositions. In scenarios involving the flow of coolant or other fluid, it requires thermal-hydraulic modeling. It may require aerosol modeling to represent the transport of airborne radioactive species through the plant. And it may need calculations of the effects of over-pressure on structures, including, in some cases, explosion modeling. Throughout these calculations, conservative assumptions are made to ensure that results represent the maximum consequences of the accident scenario. The outcome is an evaluation of the maximum releases of radioactive material from the plant to the environment in the given scenario, typically tritium (both elemental, T_2 and as tritiated water, HTO), neutron activated material (particularly dust), and in some cases ACPs. These are then used as source terms for the modeling of dispersion in the atmosphere, transport to locations where people may be present (e.g. the site boundary), and evaluation of the potential radiological dose to the most exposed individual.

This kind of analysis must be performed for each accident scenario for a number of postulated events, ensuring that the set of scenarios studied cover the complete range of accidents that are possible—even with very low likelihood—in the plant. To ensure that the range of events analyzed is fully comprehensive it is essential that accident identification studies are performed using systematic methods. Typically this is done with Failure Modes and Effects Analysis (FMEA), in which all conceivable failure modes in all components of the plant are listed and their potential impact on the plant assessed (Pinna et al., 1998). Failures in different components that lead to a similar impact on the plant are grouped together into Postulated Initiating Events (PIEs), which are then

used to develop event sequences, usually with the aid of event trees. For each PIE, the event sequence with the greatest possible consequence in terms of radiological release is chosen as one of accident scenarios to be the subject of full analysis as outlined above.

FMEA requires sufficient detail in the design that individual components and their failures may be assessed; at early stages in the conceptual design of a facility this may not be available. In this case a functional approach may be used, using Functional Failure Modes and Effects Analysis (FFMEA) which identifies the functions provided by elements of the systems and the consequences of the loss of these. Hazard and Operability (HAZOP) studies provide an alternative systematic approach to the identification of off-normal conditions that could initiate an accident sequence. These so-called bottom-up studies of accident sequences may be supplemented by top-down approaches using fault trees or Master Logic Diagrams, to determine all the ways in which an unwanted event (radiological release) could occur. Combining different approaches to accident identification provides greater confidence that the final list of scenarios is complete.

An alternative to the deterministic accident analysis approach outlined above is Probabilistic Safety Analysis (PSA) (Modarelli and Kim, 2021). However, PSA requires quantitative knowledge about failure rates and reliabilities of components in systems, usually derived from operational experience of similar plant. At this stage in fusion development, many key systems are novel and first-of-a-kind, so that such data is not available. For this reason, it is currently not feasible to apply PSA to fusion facilities.

As well as studying the potential consequences of faults and failures within plant equipment, the potential impact of internal hazards such as fire or flooding, and external hazards such as earthquake, extreme weather events, or aircraft crash must be evaluated. Such studies must ensure that the plant will be resistant to all such hazards and that no significant radiological consequence will result.

ITER safety analyses

The most comprehensive safety analysis yet performed for an experimental fusion facility is that performed for ITER (Taylor et al., 2011). The analysis was used as the basis of a Preliminary Safety Report that formed a key part of the safety submission to the French nuclear safety authorities in order to obtain the license to begin the construction of the facility (Taylor et al., 2013). The safety studies included all the elements outlined above in Sections “Safety functions and classification” and “Accident analysis for fusion.”

Using extensive FMEA studies of plant systems, a set of 25 “reference events” were selected as accident scenarios to be analyzed in detail. The outcome of these analyses showed that in all cases the public consequences in terms of radiological dose to the most exposed individual remained well below the limits and targets set in the project safety objectives, with the most severe event giving rise to less than 0.1 mSv dose. Further analyses were performed of hypothetical sequences, or beyond-design-basis accidents, in which additional failures were postulated. Even in the worst case scenario the maximum long-term public dose was evaluated as 4 mSv, well within objectives and below the level at which counter-measures such as evacuation would be considered.

Accident scenarios typically involve a large leak of water coolant from the system providing heat transfer from the in-vessel components. One of the most severe scenarios begins with a postulated large fracture of a coolant pipe in the room that houses the water pumps and primary heat exchangers. Taking as an example the system that cools the divertor, this failure leads to an immediate under-cooling of the divertor and rapid temperature rise due to the high heat flux impinging on the divertor surface from the plasma. Safety systems should shut down the plasma within a few seconds of the coolant abnormality being detected. Nevertheless, temperatures may continue to rise due to decay heat from the radioactive decay of activated material in the divertor structure. If this temperature rises high enough, the structure of the divertor is weakened and fails, releasing cooling water or steam into the vessel. Now the same cooling loop is breached at two locations, both inside the vessel and outside, where the initial failure occurred. Once all of the cooling water has drained from this circuit, the pipework presents a direct path from within the vessel to the heat exchanger room, thus bypassing the first confinement barrier. The coolant that has entered the vacuum vessel pressurizes it, and the pressure continues to rise as it is heated by decay heat in the plasma-facing components. This pressure can create a flow through the voided coolant pipes, carrying with it a proportion of the in-vessel inventory of tritium and active dust, as well as the ACPs in the water itself.

Such an event results in some part of the radioactive inventory being released into the room where the initial break occurred, here assumed to be the heat exchanger and pump room. The walls, floor and ceiling of this room now have to fulfill their role in the second confinement system, preventing an uncontrolled release to the environment. Instead, the contaminated atmosphere in the room is vented through the HVAC and detritiation systems, so that only a small fraction of the inventory is released in a controlled way via the elevated release point at the stack.

A potential complication of the accident scenario described above results from the use of beryllium as the plasma-facing surface in ITER. Material eroded from this surface becomes dust that is expected to accumulate at the bottom of the vessel, around the divertor. When the coolant water or steam enters the vessel in the event scenario described, it is likely to chemically react with this beryllium dust, producing hydrogen. This beryllium-steam reaction is exothermic, so that if a sufficiently high temperature is reached (around 600 °C), there is a potential for it to become self-sustaining even with no additional input of heat. The quantity of hydrogen produced in this case could be sufficient to create an explosive mixture, if an air ingress into the vessel is also postulated through the empty cooling pipe. Such an explosion, if an ignition source is available, could produce a peak pressure that challenges other parts of the first confinement barrier. To avoid this eventuality, it is necessary for the plasma shutdown system to operate before the temperature of the divertor, and thereby the beryllium dust, has reached 600 °C.

Detailed analysis of this scenario, as outlined in Section “[Accident analysis for fusion](#),” has shown that even in the worst case, the radiological release to the environment is small, and the maximum dose to a member of the public would be less than 0.1 mSv. Analysis of all other ITER reference events yielded lower maximum public doses than this worst case.

Beyond design basis accidents

The ultimate safety margin provided by a fusion power plant may be demonstrated by postulating hypothetical event sequences involving multiple independent and low-probability failures. By taking a scenario corresponding to a reference event considered within the design, and postulating the failure of safety important class components that have been credited in the analyses, a beyond design basis accident (BDBA) scenario is defined. Analyses of the consequences of such a highly unlikely event gives confidence that the outcome of the worst sequence will be acceptable, for example by not violating the no-evacuation criterion mentioned in Section “[Introduction](#)” above.

An important demonstration in the study of BDBA scenarios is the absence of “cliff-edge” effects. This means that as progressively more and more independent failures are postulated, leading to lower and lower probability of the complete event sequence, a point is not reached at which there is a sudden increase in the consequences of the accident.

The accident scenarios involving a loss of cooling, such as the sequence described above in Section “[ITER safety analyses](#),” may be aggravated by postulating additional failures. After shutdown, the only source of heat is from radioactive decay, principally in the materials of the plasma-facing components within the vacuum vessel. Although this decay heat falls rapidly, if not rejected by some cooling system it may promote temperature rises in structures that could challenge their integrity. Typically, in the event of failure of all the cooling systems of in-vessel components, the cooling system of the vacuum vessel has adequate capacity to remove the decay heat and prevent damaging temperature rises. Only a small flow rate in the vacuum vessel cooling circuit is required to remove this decay heat, so that separate emergency pumps that can be powered by diesel generators provides protection against loss of electric power and other failures.

An example of a BDBA scenario related to this loss-of-cooling event is to postulate that even this emergency cooling system of the vacuum vessel fails, so that there is now no active cooling system operating in any part of the tokamak. Heat transfer away from the hot in-vessel components is then only by passive conduction and radiation, towards the cooler components (including cryogenic superconducting magnets) outside the vessel. Passive heat transfer across the volume of the cryostat is very limited, as it is designed, in normal operation, to provide good thermal insulation of the magnets and features thermal shields with low emissivity. Analyses of this BDBA event for ITER have shown that filling the cryostat with some gas, such as air or helium, provides a convective heat transfer path that results in adequate rejection of decay heat and avoids excessive temperatures in in-vessel components. Because of the large thermal masses involved and the only modest level of decay heat, temperatures rise only slowly, meaning that the action of allowing air (or helium) into the cryostat is not urgently required, but may be initiated on the timescale of days or even weeks after the onset of the accident.

Licensing and regulations for fusion

Although there have been many experimental tokamak and other fusion devices, only two have so far used both tritium and deuterium as fuel and have achieved D-T fusion reactions. These are TFTR at the Princeton Plasma Physics Laboratory in the USA, which completed operations in 1997, and JET the European tokamak located at the Culham Science Centre in the UK. Only limited tritium inventories were used at TFTR and JET, and a restricted number of D-T neutrons generated, so that neither required licensing as a nuclear facility. Many other tokamaks have used deuterium fuel and generated neutrons via D-D fusion, but none at a level significant enough to warrant nuclear licensing. When considering the licensing of future fusion power plants and the regulations that may apply, it is useful to look at nuclear licensing in an international context and consider the principles that are common in the approach of different countries ([Williams, 2021](#)).

The construction of ITER is the first project requiring nuclear licensing of a fusion facility ([Taylor et al., 2013](#)). At its site at Cadarache in the south of France it is subject to the same regulations as any other nuclear facility in France, and its construction, commissioning, operation and decommissioning require authorization by the French nuclear safety authority (Autorité de Sûreté Nucléaire, ASN). Any facility in France categorized as a basic nuclear installation (Installation Nucléaire de Base, INB) is regulated by the same law, the *loi no. 2006–686 relative aux installations nucléaires de base et au contrôle, en matière de sûreté nucléaire, du transport de substances radioactives*, known as the TSN law (for Transparency in Nuclear Safety). ITER is categorized as an INB because of its expected significant on-site tritium inventory.

Whereas in France all nuclear facilities, from power reactors to fuel stores, research laboratories and accelerators, fall under the same law and new nuclear fusion plant are readily included in the scope, in many countries the existing nuclear legislation is more limited in scope. In such cases existing laws and regulations are typically focused on fission power plant, and new fusion facilities do not fall within their scope. It will then be necessary to enact new laws and regulations specifically to authorize the construction and operation of a fusion plant. In the development of such fusion-specific regulation, two important principles are that regulations should be *targeted* and *proportionate*. This means that the requirements arising from regulations developed for fusion should be concomitant with the nature and level of hazards presented by fusion facilities. This may lead to a simpler licensing process

than an approach in which attempts are made to adapt existing nuclear fission regulations to fusion plant. For example, the absence of reactivity accidents and of core melt events removes some of the safety provisions required in a power plant.

Another important absence in a fusion power plant is that of actinides. There is no requirement for any fissile or fertile material in the facility, which may significantly simplify requirements for nuclear security. Although tritium has its application in nuclear weapons technology, it is not currently a safeguarded material and is not subject to any provision in the Non-Proliferation Treaty or related agreements. It is however subject to export control provisions in some countries, not only tritium itself but also tritium technologies. As there will be significant quantities of tritium (order of hundreds of grams) stored, processed, burnt and generated in the systems of a fusion facility, a very strict tritium accountancy system will need to be established to guard against the diversion of small quantities for clandestine use.

The other area of possible proliferation concern for a fusion plant is in the potential use of neutrons from fusion to breed fissile material that could be used for weapons. Introducing fertile material such as ^{238}U into a blanket would result in the generation of ^{239}Pu , for example. As noted above, however, no fissile or fertile material is legitimately present in a fusion plant. Thus applying safeguards is straightforward, requiring only the verification that no fertile material is present on site. This contrasts with a fission reactor where a fine balance of large inventories of fissile and fertile material on the site, or entering and leaving it, must be maintained. The legal framework for this is the Non-Proliferation Treaty and an Additional Protocol signed by many states that allows for the verification of the absence of nuclear material. The technology to detect fertile material is readily available and may need to be applied to verifying fresh blanket materials that arrive on the site.

Additional considerations for fusion-fission hybrids

A combined fusion-fission power plant concept may be conceived to comprise a D-T fusion plasma, typically in a tokamak configuration, surrounded by a sub-critical fissile blanket that provides energy multiplication using the 14 MeV D-T neutrons. The enhanced neutron population could provide adequate breeding of tritium fuel as well as conversion of fertile material into fissile, or other transmutation of actinides.

The presence of fusion and fission technologies in a single plant would lead to a combination of hazards requiring additional safety provisions. The attractiveness of a pure fusion system, with the absence of actinides, fission products and potential reactivity accidents, is diminished. Confinement systems have to be designed not only to protect inventories of tritium and activation products, but also actinides and fission products. Accident scenarios that may challenge confinement systems will include those arising from additional initiating events and with additional consequences of, for example, loss of cooling functions. The spectrum of radioactive material that could be released now includes a substantially wider range of nuclides.

The replacement of in-vessel components of a fusion plant, first wall, blankets and divertors, is the most challenging maintenance operation of the plant and introduces additional risks that must be mitigated by specific additional confinement and shielding provisions. In a fusion/fission hybrid plant the difficulty and complexity is augmented by the need to replace fission fuel assemblies in an operation similar to that of refueling a fission reactor.

Finally, the simple approach to avoiding proliferation concerns in a fusion facility, by requiring no fertile or fissile material on site, is also lost. A fusion/fission hybrid plant would be subject to the same safeguards procedures as any fission reactor, but with the additional concern of a significant inventory of tritium.

Management of radioactive waste from fusion

The quantity of material in a fusion facility that experiences a neutron flux is potentially quite large. The in-vessel components (i.e. the first wall, blankets and divertor) experience a high level and accumulate significant neutron activation, and the vacuum vessel and some components outside of it are also irradiated by neutrons with sufficient fluence to become active at a level too high to allow direct handling by human intervention, at least at early times after shutdown. The ex-vessel components that may become active include supporting structures of toroidal field coils, other structures and shields inside the cryostat, and the cryostat vessel itself. Parts of the bioshield, in performing its function of absorbing neutrons that have penetrated the various components providing shielding, will also acquire some level of activation.

There may therefore be a large volume of active material arising from decommissioning at the end of plant life, augmented by those in-vessel components that have been removed from the plant to be replaced during the operational life. The majority of this material is at a low level of activation and would be categorized as low level or very low level waste. Components that have been irradiated in the high neutron flux inside the vessel can be expected to be categorized as intermediate level waste. No high-level waste is foreseen from a fusion plant. Much of the material decays with a relatively short half-life, so that storage for some tens of years before disposal may allow a more favorable waste categorization.

To minimize the quantity and level of radioactive waste, the use of reduced-activation or low-activation steels, particularly for the structure of in-vessel components such as the blankets, is commonly proposed. The minimization of some impurities in materials compositions can be important, as some have activation products that contribute significantly on the longer timescale. An important example is cobalt, in which the reaction $^{59}\text{Co}(n, \gamma)^{60}\text{Co}$ generates the high-energy gamma emitter ^{60}Co , with a 5.3 year half-life. This nuclide typically dominates gamma dose rates on the timescale of relevance to maintenance and decommissioning operations,

up to 50 years after shutdown. At ITER, although in-vessel components are fabricated from 316L(N) stainless steel, the specification for an ITER Grade of this steel limits the cobalt content to 0.05% by weight, compared with the 0.2% more common in standard steels.

For ITER, a detailed characterization of radioactive waste has been carried out (Rosanvallon et al., 2016), according to the waste classification scheme employed in French regulations. The in-vessel components become classified as intermediate-level long-lived waste (Moyenne Activité à Vie Longue, MAVL), the vacuum vessel as low/intermediate-level short-lived (Faible et Moyenne Activité à Vie Courte, FMA-VC), and most of the ex-vessel structures within the cryostat as very low-level waste (Très Faible Activité, TFA). There is no material qualifying as high-level waste (Haute Activité à Vie Longue, HAVL). At decommissioning, the total quantity of TFA may exceed 20,000 tonnes, FMA-VC around 13,000 tonnes and MAVL 2500 tonnes.

Those components that have been in-vessel will be not only activated by neutrons but also contaminated with tritium, both on the surfaces and by permeation into the bulk of the material. The expected level of tritium contamination may initially exceed the level at which this MAVL material could be accepted into an existing waste repository in France, so it is planned to store the material in an interim storage facility for some tens of years to allow for some tritium decay (Decanis et al., 2016). Other techniques for the detritiation of solid wastes have been studied as it may be necessary for future fusion facilities such as DEMO (Taylor et al., 2019). These techniques generally involved bulk heating, baking or even melting of the material to release the tritium permeated into the bulk.

As well as material that will become activated, there are components of the fuel cycle systems that have no exposure to neutrons, but will be treated as radioactive waste purely because of their contamination with tritium.

For DEMO and future fusion power stations (Campbell, 2021b), the quantities of radioactive waste could be higher than in ITER. The potentially increased level of activation due to higher neutron fluence levels will be offset by the use of low-activation materials such as specially developed ferritic-martensitic steels. It has long been the aim of conceptual power plant studies that the amount of waste requiring permanent repository disposal may be minimized by applying the principles of clearance and recycling (Zucchetti et al., 2001). Clearance allows material that has fallen below defined radioactivity levels to be released from regulatory control and re-used for any purpose, for example by applying the criteria defined by the IAEA (2014). Recycling of material that does not meet this criterion may still be possible in applications where a level of residual activity would not preclude its use, for example within future nuclear facilities.

An illustration of the decay of the activation products in waste material is provided in Fig. 1. This shows how the total beta and gamma-ray activity in the steel structural material of an in-vessel component decays during the 1000 years after shutdown of the plant. This is from analysis of a proposed breeder blanket design for a European DEMO reactor (Gilbert et al., 2018), which uses Eurofer, a reduced-activation ferritic-martensitic steel, as its structure. Also plotted are the contributions of the dominant nuclides. It is clear that, apart from ^{55}Fe , most of the long-lived activity comes from neutron activation of impurities and trace elements in the steel composition. Also indicated is the limit below which the material would be classified as Low Level Waste (LLW) according to regulations in the United Kingdom. It is clear that only ^{14}C , which is generated by the $^{14}\text{N}(n,p)^{14}\text{C}$ reaction, prevents the material from becoming LLW after around 100 years of storage. Nitrogen is included in the steel composition at typically 70 ppm to increase strength and resistance to corrosion (Litnovski, 2021). Removal of ^{14}C , which has a half-life of 5700 years, would therefore be beneficial, and this may be possible during smelting of the material for casting into ingot form for storage or recycling (Di Pace et al., 2019), at which stage tritium could also be removed.

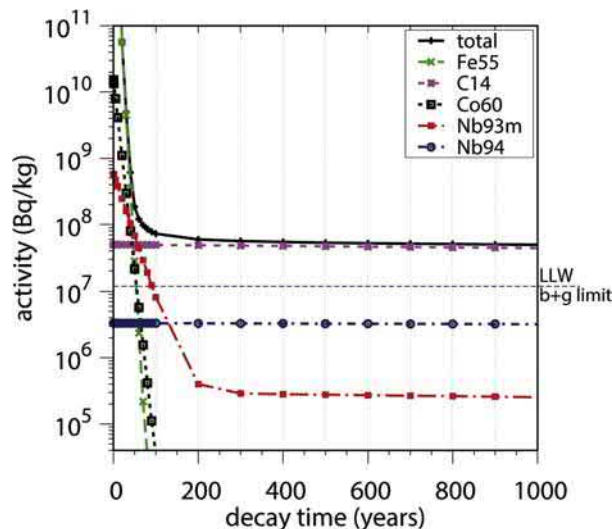


Fig. 1 Evolution of beta and gamma activity in steel structure of a DEMO breeder blanket following shutdown of the plant.

In addition to the large quantity of solid radioactive material, albeit most of it at a low or very low level of activity, there is likely to be some liquid waste stream from the plant, as well as routine gaseous effluent as mentioned in Section “**Tritium hazards and their amelioration.**” The liquid waste will mainly be fluids into which tritium has permeated, as well as coolants containing activated corrosion products. Tritium will be recovered from this waste stream to the maximum extent possible and returned to the fuel cycle. Detritiated waste water may be discharged from the plant if the activity level is below permitted levels.

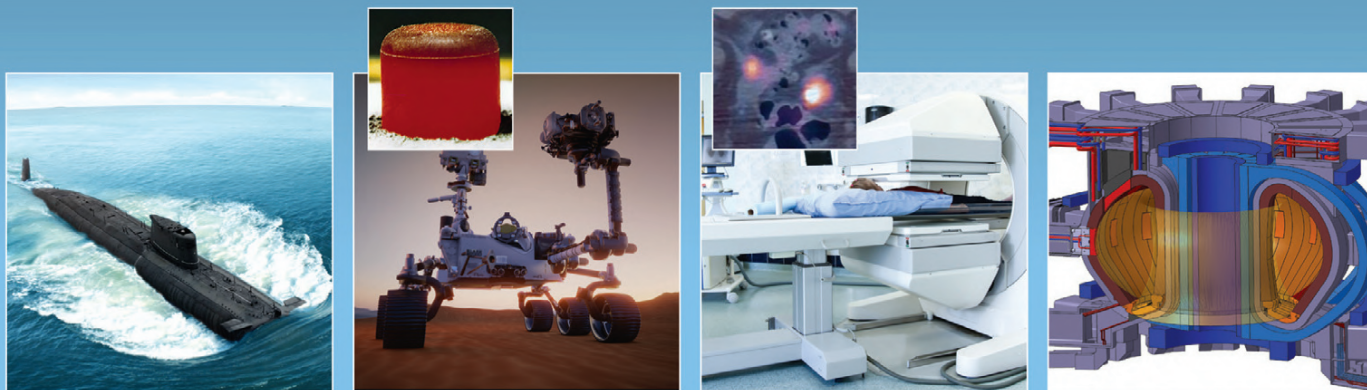
See Also: Ensuring Safety by Design and Evaluation; Safety of Nuclear Reactors; Safety, Regulation, and Decommissioning of Commercial Nuclear Power Reactors—Introduction.

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*Retired Professor, Nuclear Engineering Department,
University of California at Berkeley*

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COVER PAGE

Bottom: View of a nuclear power plant by the sea in Vandellos, Spain. It is a 1043 MWe Pressurized Water Reactor.
Top, from left to right (sample applications of nuclear energy):

- a submarine powered by nuclear energy
- an exploration vehicle/lab on MARS powered by a Plutonium-238 nuclear heat source (insert)
- a nuclear medical diagnostic machine for humans with a sample of PET-CT scan (insert)
- an artist vision of a future Tokamak-type fusion energy demonstration power reactor

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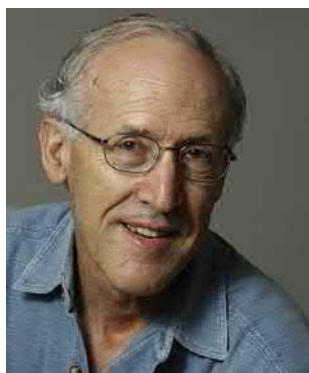
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University of California at Berkeley
gehud@nuc.berkeley.edu

Professor Greenspan (Ph.D., 1966, in Nuclear Science and Engineering from Cornell University, Ithaca, N.Y., USA) retired from the Department of Nuclear Engineering of the University of California at Berkeley (UCB) in 2014 but served as a Professor in the UCB Graduate School until 2019. He has nearly 60 years of nuclear energy related research experience starting in 1961 as an M.Sc. student at the Department of Nuclear Engineering of the Israel Institute of Technology. His research spans a wide range of nuclear engineering related disciplines including: theoretical and experimental reactor physics; computational methods development including development of new optimization methods for determining the minimum critical mass and for finding optimal design of radiation shields, fusion reactor blankets and medical installations; conception and analysis of advanced fusion fuel cycles and of fusion-fission hybrid reactors; as well as advanced fission reactors and nuclear fuel-cycle conception, design and analysis. The latter activity was the focus of his research during the last twenty-five years. The thrust of this

research is improving sustainability of nuclear energy along with improving economics, safety, and proliferation resistance. He has been the Principal Investigator on studies of dozens of advanced nuclear systems, including those corresponding to Generation-IV reactors and to Small Modular Reactors (SMRs). He was one of the early advocates of SMRs and led a multi-institutional international team that developed a preliminary conceptual design of the Encapsulated Nuclear Heat Source (ENHS)—an innovative nuclear battery-type SMR that is free of pumps, pipes and valves in the primary reactor system. The advanced reactor technologies addressed in his research include: heavy water reactors, pressurized and boiling water reactors, prismatic and pebble-bed liquid-salt cooled reactors, molten-salt reactors, sodium and lead-alloy cooled fast reactors, breed-and-burn reactors, heat-pipe reactors as well as fusion-fission hybrid reactors and accelerator driven nuclear reactors. His research is documented in over 400 refereed publications, over 200 non-refereed publications, 5 book chapters, and a number of patents.

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Introduction to the Encyclopedia of Nuclear Energy

Ehud Greenspan, Department of Nuclear Engineering, University of California, Berkeley, Berkeley, CA, United States

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Introduction

From the beginning of the Universe through to the present day, nuclear energy, nuclear reactions, and nuclear radiations have played major roles in the evolution and shaping of our universe and planet Earth. All the chemical elements we find on Earth were produced by nuclear processes in the Universe. Planet Earth is continuously exposed to nuclear radiation from primordial radioactive isotopes produced in cosmological processes that followed the Big Bang. A number of primordial radioactive isotopes are still among the Earth's constituents, including thorium ^{232}Th , the uranium isotopes ^{235}U and ^{238}U as well as potassium ^{40}K . The spontaneous radioactive decay of the nuclides of these isotopes releases nuclear energy that provides a major contribution to the heat balance of planet Earth. The internal heat source is responsible for keeping the outer core of the Earth in a molten magma state and played a major role in the geologic evolution of the Earth's continents and oceans. The primordial radiation and Earth's heat source may have also contributed to the creation as well as evolution of life on Earth. Since its birth, the Earth has been also constantly bombarded by galactic cosmic nuclear radiation, which primarily consists of protons (hydrogen nuclei) and alpha particles (helium nuclei). Hence, mankind as well as any organism on Earth, were and are being continuously exposed to background nuclear radiation throughout their evolution. The background radiation level as well as the radioactive heat source due to the decay of the primordial radioactive isotopes were significantly higher at the time of Earth formation and are slowly decreasing with time.

It is only within the last hundred and twenty years or so that mankind discovered the structure of the atom and its nucleus and learned how to harness the enormous energy potential contained within the nucleus as well as the different types of radiation nuclei and atoms can emit. Nuclear radiation was discovered in 1896, when Henry Becquerel and Marie Curie discovered that salts of uranium emitted radiation that could be detected with photographic plates. In the 125 years since this discovery, mankind deciphered the structure of the atom and its nucleus, learned how to produce artificial radioactive isotopes (Irène Joliot-Curie and Frédéric Joliot in 1934), discovered how to fission certain heavy isotopes (Otto Hahn, Fritz Strassmann, Lise Meitner and Robert Frisch, 1938/39) and learned how to establish a fission chain reaction (Leo Szilard, Enrico Fermi et al., 1942) and developed numerous applications of nuclear radiation and nuclear energy. The feasibility of releasing nuclear energy by fusing nuclei of light isotopes was postulated already in 1920 (Arthur Eddington) who speculated that such nuclear reactions provide the enormous energy released in the sun and stars. Although research of harnessing nuclear fusion for controlled generation of nuclear energy is being pursued since the late 1940s, mankind has not yet succeeded in producing sufficiently more energy from fusion reactions than the energy invested in inducing these reactions. Additional information on the history and fundamentals of harnessing nuclear energy can be found in (Norman, 2021).

Encyclopedia scope

The Encyclopedia of Nuclear Energy (The Encyclopedia) provides a comprehensive authoritative coverage of a wide range of topics related to the generation, utilization and safe handling of nuclear energy, nuclear radiation and associated technologies, for the benefit of mankind and planet Earth. It covers past developments, the present status, and research and development of advanced innovative technologies that may enable improvements in the generation and utilization of nuclear energy (including nuclear radiation). The Encyclopedia consists of 272 thematic articles written by experts and arranged in 13 Sections; each Section covers the subject area identified in Table 1.

The fundamental concept of this encyclopedia is to provide all this vast and very diversified information in convenient to read cross-referenced articles within a single publication. Each article provides an effective solid foundation of its topic and references for readers interested in deeper exploration of the subject matter.

Table 1 Encyclopedia sections, number of articles per section and the section editors.

Sec. #	Section title (# of articles)	Section editor	Section editor affiliation
1	Fundamentals and history of development of nuclear energy (17)	Eric B. Norman	Professor in the Graduate School, Nuclear Engineering Department, University of California at Berkeley
2	Commercial nuclear power reactors and their design (22)	Russell E. Stachowski	Chief Consulting Engineer, Global Nuclear Fuel
3	Advanced nuclear reactor concepts under R&D (28)	Alexander Stanculescu	Retired Generation Four International Forum Technical Director; Retired Idaho National Laboratory Division Director
4	Safety, regulation, and decommissioning of commercial nuclear power reactors (24)	Joy L. Rempe	Rempe and Associates, LLC
5	Fuel cycle front-end: from mining through utilization (17)	Douglas C. Crawford	Director, Transient Reactor Test Facility, Idaho National Laboratory.
		Colby B. Jensen	National Technical Lead for Fuel Safety Research, Idaho National Laboratory
6	Fuel Cycle back-end: from recycling to deep geological disposal (19)	Christophe Poinssot	Deputy General Director (CEO) and Scientific Director of the French Geological Survey (BRGM) ^a
7	Radiation protection (21)	Laurence Miller	Professor Emeritus, Department of Nuclear Engineering, the University of Tennessee
8	Non-electric applications of nuclear energy (15)	Shannon M. Bragg-Sittou	Lead for Integrated Energy Systems, Idaho National Laboratory ^b
9	Nuclear power for space (16)	David I. Poston	Chief Reactor Designer, Los Alamos National Laboratory
10	Research reactors (23)	Sean O'Kelly	Associate Laboratory Director, Idaho National Laboratory. Chair of IAEA Technical Working Group on Research Reactors
11	Medical, industrial and agricultural applications of nuclear technology (26)	Alan E. Waltar	Retired Professor and Head, Department of Nuclear Engineering, Texas A&M University, College Station
12	Nuclear fusion R&D (26)	David J. Campbell	Munich, Germany. Former Director for Science and Operations, ITER Organization
13	Social issues of nuclear energy (18)	Andrew C. Kadak	Kadak Associates, Inc. Former Professor of the Practice Massachusetts Institute of Technology

^aAlso: Prof. of Nuclear Chemistry at the National Institute of Nuclear Science and technology (INSTN).

^balso National Technical Director for the DOE-NE Integrated Energy Systems program.

Intended readership

The articles are written at a level that users, with a bachelor degree in engineering, scientific and medical disciplines, especially in nuclear engineering, will find understandable and helpful. The Encyclopedia is intended to be an invaluable resource for students and will make an important contribution to the preservation of the vast amount of knowhow developed since the dawn of the nuclear era in the nuclear energy related fields and to its transfer to the new generation.

The Encyclopedia will also be very useful to industry and academia researchers and practitioners in the many scientific and engineering fields related to the generation and use of nuclear energy, including nuclear radiation. It will enable researchers and practitioners to refresh their memory in their field of specialization or expand their knowhow on less familiar subject areas.

The Encyclopedia will also be useful to members of the public with education in engineering or sciences who are interested in learning about specific nuclear energy or nuclear radiation related topics without having to struggle with textbooks or journal articles.

On the need for this Encyclopedia

The publication of this Encyclopedia is very timely for a number of reasons, including the following:

- Nuclear energy can play a very important role in drastically reducing the amount of fossil fuels burned for generation of electricity and, hence, the amount of greenhouse gas (GHG; primarily CO₂) emissions while, at the same time, making more electricity available for developing countries and for supporting the expected growth in the world population. Moreover, nuclear energy can make an important contribution to drastically reducing the amount of GHG emission from the transportation, industrial, residential and agricultural sectors of the economy. Hence, nuclear energy can make a vital contribution to meet the goal set by the 2015 Paris Agreement on climate change and ratified by many nations—to limit global warming to well below 2, preferably to 1.5 degrees Celsius, compared to pre-industrial levels. This point is elaborated upon below in the section: “On the Importance of Nuclear Energy in Avoiding Climate Change Catastrophe”. The state of the art in nuclear energy generation is covered in (Stachowski, 2021; Crawford, 2021).

- The Generation IV (GEN-IV) International Forum (GIF) was created in 2001 as a co-operative international endeavor of research, development, demonstration and deployment of advanced fourth generation innovative nuclear reactors that, hopefully, will be available for commercial deployment by 2030. The objectives of the innovative GEN-IV nuclear energy systems include improved sustainability, economics, safety and reliability, as well as proliferation resistance and physical protection. This encyclopedia provides an update on the state of the GEN-IV R&D efforts and explains the benefits the innovative GEN-IV reactors may provide. The Encyclopedia also describes (Stanculescu, 2021) the advanced nuclear fuel cycles that offer highly improved resource utilization, greatly reduced nuclear waste, and reduced proliferation risks.
- Over the past decade or so, there has been a surge in the number of private companies, mostly supported by venture capital, that are developing advanced innovative Small Modular Reactors (SMRs); reactors designed to generate between 10 to 300 MW of electricity per unit (Stanculescu, 2021). The SMRs offer a new paradigm in the generation and delivery of nuclear power; they promise to improve all aspects of design, engineering, construction and operation of commercial nuclear power plants, including enhanced safety performance, better economic affordability, and reduced financial risk. The components (modules) of SMRs are to be of a standard design, shop fabricated in assembly lines subjected to high quality control, and then transported as complete modules to the power plant site for rapid installation. The SMRs also offer flexible power generation for a wider range of users and applications—central power stations can consist of multiple units installed at a rate that best matches the growth in power demand; alternatively, a single SMR unit can be installed in the vicinity of the power customer using a so-called distributed (localized) power grid. It is even being proposed that the hundreds of shuttered coal power plant sites could be repurposed for SMRs. This encyclopedia describes (Stanculescu, 2021) a sample of the SMRs under development and explains the incentives for their development and the benefits they offer. The Encyclopedia also describes the motivation for, and examples of, so called “microreactors”—reactors producing an electrical power of less than 10 MW. Such reactors are primarily intended for deployment in remote areas which are not connected to the grid and to which the transport of fossil fuel is very expensive, as well as to provide reliable power to sensitive installations even in case of failure of the central power grid.
- The very ambitious space explorations being planned by leading industrial countries cannot succeed without nuclear energy. Space Fission Power (SFP) systems offer the potential to enable truly ambitious exploration of outer space, and ultimately a permanent human presence in space. Most of the benefits of SFP is tied to the very high amount of energy that can be extracted per unit weight of fissile material, which can allow high powers and/or long lifetime at a relatively low mass. SFP systems also offer great flexibility in terms of being able to operate in diverse environments, including environments with no or limited solar radiation, and the ability to provide abundant electrical power and heat from kilowatts to megawatts. Applications of SFP include propulsion and/or power for spaceships, electricity and heat for exploration rovers for landing on Mars (other planets and moons) or for human colonies on the Moon or Mars. This encyclopedia describes (Poston, 2021; Bennett, 2021) the many benefits nuclear energy offers to space exploration and the many challenges that need to be overcome.
- Controlled thermonuclear fusion is the focus of major international research programs that aim to demonstrate the scientific and technological feasibility of large-scale generation of nuclear energy by the fusion of light atomic nuclei. The widely distributed fusion fuel resources together with the favorable anticipated safety and environmental characteristics of future fusion power plants provide a significant motivation for the continued development of the challenging science and technology towards the exploitation of fusion energy for the benefit of mankind. In addition to the description of the fundamental elements of the fusion process and of the principal approaches to the controlled and sustained generation of fusion energy being pursued, this Encyclopedia presents the state-of-the-art for fusion research and development and the progress towards commercial fusion energy generation (Campbell, 2021). The advantages and challenges of nuclear fusion as an energy source are summarized.
- This encyclopedia also describes the many ways nuclear radiation can be harnessed to provide enormous benefits in the fields of medicine, agriculture, modern industry, environmental protection, and public safety (Waltar, 2021). Described in the Encyclopedia are more than a dozen unique properties of radiation technology that allow for these benefits to be achieved. Even though enormous and widespread, the benefits to society from these technologies are largely unknown and unappreciated. The global markets for these technologies exceed USA \$500 billion annually and are growing rapidly.
- Also provided in this Encyclopedia is information on dozens of research reactors (O’Kelly, 2021) that are operating or under construction around the world and provide unique research capability for scientists pursuing advanced research in physics, chemistry, biology, material and other sciences. Another invaluable utilization of research reactors is the production of special radioactive isotopes for numerous applications in medical, industrial and agricultural applications, and scientific research. Some research reactors are also used for demonstrating the performance of evolutionary as well as advanced nuclear fuels and of advanced nuclear power reactor components.
- Public perception of nuclear energy varies from country to country. It tends to be most negative in wealthy democracies like Germany, Australia, and Japan. Contributing to the negative perception is a fear of reactor accidents, a fear of nuclear radiation, a fear of long-lived radioactive waste, and a fear of the nuclear technology misuse for weapons proliferation. The encyclopedia addresses these concerns and provides authoritative objective information that, hopefully, will contribute to dispelling the public misconceptions (Kadak, 2021).
- The Encyclopedia describes measures implemented to ensure the health and safety of the public from nuclear power (Rempe, 2021). These measures evolved as new lessons were learned from operating experience and unplanned events, in particular events with core damage and radionuclide release (including in Three Mile Island, Chernobyl, and Fukushima). Clearly, the

experience gained from the design, evaluation, operation, decommissioning, and regulation of nuclear power plants has led to significant safety improvements. Future advanced reactor designs are expected to incorporate innovative inherent safety features (Stanculescu, 2021).

- The encyclopedia provides a detailed description of the biological effects of nuclear radiation, of how we reliably measure radiation levels and ensure that they do not exceed the permissible levels (Miller, 2021). The Encyclopedia describes the theory and practice of radiation protection that provide the basis for regulatory requirements that are essential for the safe and beneficial use of the many advanced technologies that involve exposure to nuclear radiation. The Encyclopedia explains that everyone is exposed to natural background radiation on a daily basis; that radiation is routinely in use for diagnostics of health conditions, such as for dental and medical X-rays, and that this is the largest source of radiation exposure for almost all human population in developed countries and that the benefits from these practices far exceed their risk. The regulatory permissible level of nuclear radiation is harmless much like a moderate exposure to sunshine is harmless (and may even be beneficial, whereas over-exposure can cause sunburns). In other words, the regulatory limits of doses induced by exposure to nuclear radiation are already substantively protective, and there is no evidence of health improvements by reducing the doses below these limits. Insisting to reduce radiation exposure to as low as possible below the regulatory limits may result in harm.
- Also provided in this Encyclopedia is detailed information about the nuclear waste and how it can be disposed of safely (Poinssot, 2021; Kadak, 2021). Using advanced fuel cycles (Stanculescu, 2021; Poinssot, 2021) the amount and activity of the long-lived nuclear waste can be reduced to few percent of the amount that needs to be disposed of per unit of energy generated using the contemporary “once-through” fuel cycle.

On the importance of nuclear energy in avoiding climate change catastrophe

The world’s population is expected to grow from current ~6.7 billion people to over 9 billion by 2050. As recognized in the United Nations Resolution 70/1 “Transforming our world: the 2030 Agenda for Sustainable Development” (Stanculescu, 2021; United Nations, 2015), providing “universal access to affordable, reliable and sustainable energy” and in particular to reliable electrical energy, is central to human development, social stability and even world peace. Currently, about 2.8 billion people are still lacking modern energy services (International Energy Agency, 2017), and more than one billion do not have access to electricity. Not only the growth of worldwide population, but also the economic growth in developing countries will lead to a continuing increase in the demand for energy.

The tremendous increase in energy generation required for the benefit of the global population has to be accomplished while reducing the rate of GHG (primarily CO₂) emission. Already, “human activities are estimated to have caused approximately 1.0 °C of global warming above pre-industrial levels” according to the Intergovernmental Panel on Climate Change (IPCC, 2018). In order to limit the global temperature increase to around 1.5 °C, total CO₂ emissions need to be reduced to net zero by around 2050.

Germany, for example, had decided to achieve this goal by installing renewable energy generators made of solar photovoltaic panels and wind turbines. Germany also decided to close down its nuclear power plants. As a result, according to Goldstein et al. (2019), “Germany, which went all-in for renewables, has seen little reduction in carbon emissions, and, according to our calculations, at Germany’s rate of adding clean energy relative to gross domestic product, it would take the world more than a century to decarbonize, even if the country wasn’t also retiring nuclear plants early. A few lucky countries with abundant hydroelectricity, like Norway and New Zealand, have decarbonized their electric grids, but their success cannot be scaled up elsewhere; the world’s best hydro sites are already dammed.”

Goldstein et al. (2019) are also saying “we actually have proven models for rapid decarbonization with economic and energy growth: France and Sweden. They decarbonized their grids decades ago and now emit less than a tenth of the world average of carbon dioxide per kilowatt-hour. They remain among the world’s most pleasant places to live and enjoy much cheaper electricity than Germany to boot. They did this with nuclear power. And they did it fast, taking advantage of nuclear power’s intense concentration of energy per unit weight of fuel. France replaced almost all of its fossil-fueled electricity with nuclear power nationwide in just 15 years; Sweden, in about 20 years. In fact, most of the fastest additions of clean electricity historically are countries rolling out nuclear power.”

In the words of World Nuclear Association Director General Agneta Rising in 2019 (Bisconti, 2021): “The 445 nuclear reactors in 30 countries are the low-carbon backbone of electricity systems, operating in the background, day in day out, often out of sight and out of mind, capable of generating an immense amount of clean power. They are the silent giants upon which we rely today.” Table 2 lists the fraction of electricity generated during 2018 in selected countries and globally from energy sources that do not emit greenhouse gases (Pioro et al., 2021). According to Poinssot and Bourg (2021), nuclear energy is within the top three of the most environmental-friendly energy sources; in the same range as hydropower and wind-power. Measured by CO₂ emissions, nuclear energy is even the most environmentally friendly energy source.

It is being suggested (Conca, 2021) that recent research has revealed the potential for extracting practically unlimited supplies of uranium from the oceans at a cost that may come close to that of land mining. If successful, this may make fission nuclear energy practically as renewable as wind and solar (Conca, 2021).

It is not sufficient to decarbonize the electricity generation sector. In the U.S., for example, 33% of CO₂ emissions are attributed to energy generated to support the electricity sector but an additional 36% of emissions result from transportation and 19% result

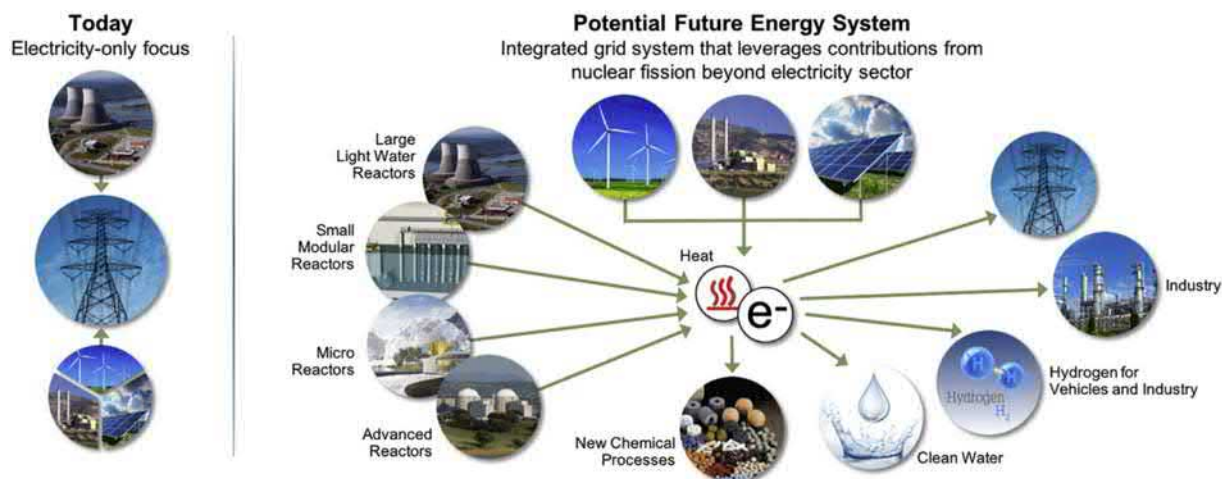


Fig. 2 Conceptualized coordination of energy generation sources and demand to maximize flexibility and economic performance while ensuring grid reliability and resilience.

Most energy experts around the world are advocating the development of a portfolio of low-carbon multi-energy technologies including, in addition to hydro-power plants (with very limited potential for expansion in capacity) and the currently favored “renewable” electricity generators, nuclear power plants. Because the power output of the renewable generators greatly fluctuates over the day and over the year, these generators need to be supplemented by effective energy storage devices that significantly add to the cost of energy.

Nuclear power plants, on the other hand, can generate power at a constant rate throughout the day and the year (excluding a shutdown period of ~1 month for refueling and maintenance once every 18–24 months), practically independent of weather conditions, while requiring a smaller land area per unit energy generated relative to the area required by renewable energy generators.

Moreover, nuclear energy can be generated in the form of high temperature heat (Stanculescu, 2021; Bragg-Sitton and Boardman, 2021) that is the most economical form of energy to store and to use as a backup for the intermittent nature of the renewable energy generators. Therefore, a suitable mix of nuclear and renewables is likely to be the optimal strategy for carbon-free energy generation.

Fig. 1 (Bragg-Sitton and Boardman, 2021) is a simplified illustration of the evolving perception of integrated energy systems using contemporary nuclear power reactors (primarily Light Water Reactors, LWR) in combination with solar and wind renewable energy generators to provide electricity and heat energy for transportation and many industries including non-grid energy users.

Fig. 2 (Bragg-Sitton et al., 2021) is a simplified conceptual perception of the proposed transition away from the use of contemporary large central nuclear power plants from their traditional electricity-only production mode to their exploitation for the provision of a variety of outputs. Future energy systems that seek to maximize energy utilization, generator profitability, and grid reliability and resilience must necessarily consider all available carbon-free energy sources, using each available resource in a way that maximizes its benefits while minimizing less-desirable attributes. The future energy systems should include advanced nuclear reactors including Gen-IV reactors, SMRs and even microreactors and should use distributed in addition to centralized electricity grids.

In summary, the need to meet the objective of zero net CO₂ emission by 2050 combined with the need to greatly increase the total global energy generation rate is an enormous challenge that can be achieved only by making use of all available clean energy generation technologies including nuclear in addition to solar and wind electricity generators. At the 2021 Earth Day climate summit, U.S. president Biden called (WSJ, 2021) for slashing U.S. greenhouse gas emissions by 2030 by 50%–52% compared with the baseline year of 2005, as he seeks to put America at the forefront of world-wide efforts to combat climate change. According to WSJ (2021), meeting this objective would require dramatically reshaping key sectors of the economy, from energy to transportation to agriculture.

Concluding remarks

This Encyclopedia of Nuclear Energy provides a very comprehensive, up-to-date and authoritative information related to nuclear energy, including nuclear radiation—from the discovery of the structure of the atom and its nucleus; through the discovery of three different ways to extract energy from nuclei (fission of heavy nuclei, fusion of light nuclei and radioactive decay); through the state-of-art of the nuclear industry for energy generation and for exploitation of several nuclear technologies for medical, industrial, agricultural and other applications; to the research and development of innovative nuclear technologies that are expected to further

increase mankind benefits from nuclear energy. The reader can find in the Encyclopedia a wide range of topics ranging from the contribution of nuclear energy to the evolution of the continents and oceans of planet Earth to the enabling of very ambitious space explorations including the provision of electricity and heat to human colonies on the Moon, Mars and beyond. The Encyclopedia explains why it is essential to greatly increase the nuclear energy generation rate in order to prevent a global warming catastrophe and, at the same time, provide sufficient energy to the growing world population and to the billions of people in underdeveloped countries.

It is hoped that this Encyclopedia will make an important contribution to transferring of the huge amount of technical knowledge accumulated in the field to the young generation, will inspire students to join research and development of innovative nuclear technologies and will contribute to better understanding of the nuclear technologies, their merits and safety by the general public and policy makers. The appreciation of the enormous benefits in the fields of steady reliable clean energy generation, medicine, agriculture, modern industry, environmental protection, and public safety that mankind enjoys every day and which is described in this Encyclopedia will, hopefully, contribute to much greater public acceptance and support of enhanced utilization of nuclear energy in the mix of clean energy technologies.

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CONTRIBUTORS TO VOLUME 4

Hamid Aït Abderrahim

*Belgian Nuclear Research Centre (SCK CEN),
Boeretang, Mol, Belgium*

Md Arzoo Ansari

*Isotope and Radiation Application Division, Bhabha
Atomic Research Centre, Mumbai, India*

Nikolay V Arkhangelskiy

*State Atomic Energy Corporation "Rosatom", Joint Stock
Company "Science and Innovations",
Moscow, Russia*

Suzanne Baker

*Nuclear Engineering and Radiological Sciences
Department, University of Michigan, Ann Arbor, MI,
United States*

RA Bari

*Brookhaven National Laboratory, Upton, NY, United
States*

Roger H Bezdek

*Management Information Services Inc., Virginia,
United States*

Gilles Bignan

*French Atomic and Alternatives Energies Commission
(CEA), Paris, France*

Ann Stouffer Bisconti

*Bisconti Research, Inc., Chevy Chase, MD, United
States*

Tor Bjørnstad

*Institute for Energy Technology, Kjeller, Norway; and
University of Oslo, Oslo, Norway*

P Brisset

*International Atomic Energy Agency (IAEA), Vienna,
Austria*

Christopher D Bryan

*Oak Ridge National Laboratory, Oak Ridge, TN, United
States*

Giovanni Cattoli

*Animal Production and Health Subprogramme, Joint
FAO/IAEA Division of Nuclear Techniques in Food and
Agriculture, Department of Nuclear Sciences and
Applications, International Atomic Energy Agency,
Vienna, Austria*

David Chandler

*Oak Ridge National Laboratory, Oak Ridge, TN, United
States*

L-Y Cheng

*Brookhaven National Laboratory, Upton, NY, United
States*

Andrzej G Chmielewski

*Institute of Nuclear Chemistry and Technology,
Warsaw, Poland*

R David Clovis

*Sandia National Laboratories, Albuquerque, NM,
United States*

Caroline Cochran

Oklo, Inc., Sunnyvale, CA, United States

Didier De Bruyn

*Belgian Nuclear Research Centre (SCK CEN),
Boeretang, Mol, Belgium*

Vitaly Fedchenko

*Stockholm International Peace Research Institute
(SIPRI), Stockholm, Sweden*

Rafaël Fernandez

*Belgian Nuclear Research Centre (SCK CEN),
Boeretang, Mol, Belgium*

Daniel Flynn

*NIST Center for Neutron Research, National Institute of
Standards and Technology, Gaithersburg, MD, United
States*

Leslie P Foyto

University of Missouri, Columbia, MO, United States

Pasquale Fernando Fulvio

Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

Mario Garcia Podesta

Animal Production and Health Subprogramme, Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, Department of Nuclear Sciences and Applications, International Atomic Energy Agency, Vienna, Austria

Heiko Gerstenberg

Technische Universität München, ZWE FRM II, Garching, Germany

Michael W Golay

Department of Nuclear Science and Engineering, Massachusetts Institute of Technology, Cambridge, MA, United States

Tsahi Gozani

Gozani Consulting, Playa Vista, CA, United States

Michael W Gregson

Sandia National Laboratories, Albuquerque, NM, United States

Dominique Greneche

Nuclear Consulting Company, Marcoussis, France

Michał Gryziński

National Centre for Nuclear Research, Świerk, Poland

Bumsoo Han

International Atomic Energy Agency, Vienna, Austria

Ayman I Hawari

Nuclear Reactor Program, Department of Nuclear Engineering, North Carolina State University, Raleigh, NC, United States

Lee Kheng Heng

Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, Vienna, Austria

J Stephen Herring

Center for Space Nuclear Research, Universities Space Research Association, LLC, Idaho Falls, ID, United States

GiHoon Hong

GH Institute, Seoul, South Korea

Ian Hore-Lacy

World Nuclear Association, Blackburn North, VIC, Australia

Lin-wen Hu

Nuclear Reactor Laboratory, Massachusetts Institute of Technology, Cambridge, MA, United States

George R Imel

Idaho State University, Pocatello, ID, United States

Daniel T Ingersoll

Oak Ridge National Laboratory, Oak Ridge, TN, United States; and NuScale Power, LLC, Corvallis, OR, United States

Colby Jensen

Nuclear Fuels and Materials Division, Idaho National Laboratory, Idaho Falls, ID, United States

Deepa Joseph

Microtrol Sterilisation Services Pvt. Ltd., Mumbai, India

Andrew C Kadak

Kadak Associates, Inc., Port St. Lucie, FL, United States; and Massachusetts Institute of Technology, Cambridge, MA, United States

Vikram Kalia

Microtrol Sterilisation Services Pvt. Ltd., Mumbai, India

Keon Wook Kang

Department of Nuclear Medicine, Seoul National University College of Medicine, Seoul, South Korea

Si-Yong Kang

Advanced Radiation Technology Institute, Korea Atomic Energy Research Institute, Jeongju, Jeonbuk, Republic of Korea

CG Karhadkar

Reactor Group, Bhabha Atomic Research Centre, Mumbai, India

Aruna Korde

Division of Physical and Chemical Sciences, International Atomic Energy Agency, Vienna, Austria

U Saravana Kumar

Isotope Hydrology Section, Division of Physical and Chemical Sciences, Department of Nuclear Sciences and Applications, International Atomic Energy Agency (IAEA), Vienna International Centre, Vienna, Austria

Maciej Lipka

National Centre for Nuclear Research, Świerk, Poland

Lance Lippert

Sandia National Laboratories, Albuquerque, NM, United States

Jessica R Lovering
Carnegie Mellon University, Pittsburgh, PA, United States

Jessica Lovering
Good Energy Collective, Brentwood, MD, United States

Gaweł Madejowski
National Centre for Nuclear Research, Świerk, Poland

Lonnie E Martin
Sandia National Laboratories, Albuquerque, NM, United States

Harold F McFarlane
Idaho National Laboratory, Idaho Falls, ID, United States

MLE Nagai
Nuclear and Energy Research Institute—IPEN, São Paulo, Brazil

Thomas Newton
NIST Center for Neutron Research, National Institute of Standards and Technology, Gaithersburg, MD, United States

Sooyoul Oh
Korea Atomic Energy Research Institute, Daejeon, South Korea

D Sean O’Kelly
Idaho National Laboratory, Idaho Falls, ID, United States

Edward J Parma
Eden Radioisotopes, LLC, Albuquerque, NM, United States

James Parry
TREAT Division, Idaho National Laboratory, Idaho Falls, ID, United States

John E Parsons
Sloan School of Management, Massachusetts Institute of Technology, Cambridge, MA, United States

Gregory W Patton
Pacific Northwest National Laboratory, Richland, WA, United States

Rui Pereira
Insect Pest Control Subprogramme, Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, Department of Nuclear Sciences and Applications, International Atomic Energy Agency, Vienna, Austria

Nuno Pessoa Barradas
International Atomic Energy Agency, Physics Section, Vienna International Centre, Vienna, Austria

Winfried Petry
Technische Universität München, ZWE FRM II, Garching, Germany

Eric T Pillai
Department of Industrial Engineering and Operations Research, University of California, Berkeley, CA, United States

Suresh D Pillai
National Center for Electron Beam Research, Department of Food Science & Technology, Texas A&M University, College Station, TX, United States

Bartłomiej Piwowarski
National Centre for Nuclear Research, Świerk, Poland

Pavel P Povinec
Faculty of Mathematics, Physics and Informatics, Comenius University, Bratislava, Slovakia

Rafał Prokopowicz
National Centre for Nuclear Research, Świerk, Poland

Steve Reese
School of Nuclear Science and Engineering, Oregon State University, Corvallis, OR, United States

Geoffrey S Rothwell
Department of Economics, Stanford University, Stanford, CA, United States

Sunil Sabharwal
Radiation Technology Development Division, Bhabha Atomic Research Centre, Trombay, Mumbai, India

Maria Helena Sampa
IPEN, Sao Paulo, Brazil

Shyam Shrivastava
Radiation Oncology, Apollo Hospitals, Navi Mumbai, India

Tej Singh
Reactor Group, Bhabha Atomic Research Centre, Mumbai, India

Sobhana Sivasankar
Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, Vienna, Austria

Rachel Slaybaugh
Department of Nuclear Engineering, University of California, Berkeley, Berkeley, CA, United States

Valeriia N Starovoitova
Radioisotope Products and Radiation Technology Section, Division of Physical and Chemical Sciences, Department of Nuclear Sciences and Applications, International Atomic Energy Agency, Vienna International Centre, Vienna, Austria

John G Stevens
*Nuclear Science and Engineering Division, Argonne
National Laboratory, Lemont, IL, United States*

Dan Strellis
Rapiscan Systems, Andover, MA, United States

Mark Summerfield
*Australian Nuclear Science and Technology
Organisation, Lucas Heights, NSW, Australia*

Anna Talarowska
National Centre for Nuclear Research, Świerk, Poland

J Thereska
*International Society for Tracer and Radiation
Applications (ISTRA), Vienna, Austria*

Gert Van den Eynde
*Belgian Nuclear Research Centre (SCK CEN),
Boeretang, Mol, Belgium*

Steven Van Dyck
SCK•CEN, Mol, Belgium

Prabhakar V Varde
*Reactor Group, Bhabha Atomic Research Centre,
Mumbai, India*

PAS Vasquez
*Nuclear and Energy Research Institute—IPEN, São
Paulo, Brazil*

Meera Venkatesh
*Division of Physical and Chemical Sciences,
International Atomic Energy Agency, Vienna, Austria*

Gerrit J Viljoen
*Animal Production and Health Subprogramme, Joint
FAO/IAEA Division of Nuclear Techniques in Food and*

*Agriculture, Department of Nuclear Sciences and
Applications, International Atomic Energy Agency,
Vienna, Austria*

David Vittorio
*Australian Nuclear Science and Technology
Organisation, Lucas Heights, NSW, Australia*

Marc JB Vreysen
*Insect Pest Control Subprogramme, Joint FAO/IAEA
Division of Nuclear Techniques in Food and Agriculture,
Department of Nuclear Sciences and Applications,
International Atomic Energy Agency, Vienna, Austria*

Daniel Wachs
*Nuclear Fuels and Materials Division, Idaho National
Laboratory, Idaho Falls, ID, United States*

Alan E Waltar
*Department of Nuclear Engineering, Texas A&M
University, College Station, TX, United States*

Spencer Weart
*American Institute of Physics, College Park, MD, United
States*

Emilia Wilińska
National Centre for Nuclear Research, Świerk, Poland

Robert Williams
*NIST Center for Neutron Research, National Institute of
Standards and Technology, Gaithersburg, MD, United
States*

Nicolas Woolstenhulme
*Nuclear Fuels and Materials Division, Idaho National
Laboratory, Idaho Falls, ID, United States*

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An Introduction to Research Reactors

D. Sean O'Kelly, Idaho National Laboratory, Idaho Falls, ID, United States

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Introduction

To appreciate the wide variety of reactors considered to be research reactors requires an understanding of the history and the categorization of these reactors. The categorization is important to differentiate research or testing reactors from demonstration, prototype, propulsion, or power reactors. Globally, over 600 research reactors have operated and contributed to the fields of nuclear engineering and science over the more than 70 years since the first sustained chain reaction ([IAEA Research Reactor Database, 2018](#)). A broad historical survey of all research reactors built would be interesting but many of the reactors constructed in the past were superseded quickly by more advanced designs or were simply research dead ends that did not progress. This chapter does not attempt to describe each of those facilities as many have been shut down or decommissioned with only about 200 research reactors operating today. However, many of the world's currently operating research reactors are based on early research reactors or are essentially the same design sold or provided by companies or governments worldwide in the 1950s and 1960s (e.g., TRIGA, IRT, or VVR reactors).

In several cases, application or test environment specific reactors were built that are unique and continue to be operated today. Many reactors designed and operated during the early years of nuclear energy development were specifically used to demonstrate various reactor concepts. In some ways this is a gray area in the definition of research reactors and many of these would be better classified as demonstration or experimental reactors. For example, reactors that demonstrated liquid metal and molten salt coolants or nuclear power production in a boiling water reactor would not be considered true research reactors. However, there are liquid metal cooled fast spectrum reactors currently operating, under construction, or being considered that would be used to test materials for deployment in fast spectrum power reactors that would be classified as research or test reactors.

Today, the utilization of research reactors has matured, and many research reactors designed to be multi-purpose have generally focused on key research or utilization strengths. Research reactors continue to be important to nuclear science and engineering education with many graduate students that were trained on a university reactor going on to perform advanced research using the limited number of high-performance reactors in the world. Research reactors operating at powers of tens of megawatts have sufficient flux for the efficient production of medical radioisotopes in addition to the research activities of the facility. The value of small research reactors as an important educational tool for the public and students cannot be underestimated. University research reactors are essential for demonstrating nuclear physics and reactor operations as an entry into the nuclear science and engineering field.

First research reactors and development during World War II

The first reactor was used to prove that a controlled nuclear chain reaction was possible. After much preparation, Chicago Pile 1 (CP-1) achieved the first criticality of a man-made nuclear reactor (a natural critical chain reaction occurred at the Oklo site in Gabon, Africa almost 2 billion years ago) on December 2, 1942. This remarkable achievement was led by Enrico Fermi and occurred in a large room, a squash court, under the Stagg Field Stadium at the University of Chicago. The pile, as it was called, consisted of stacked blocks of graphite with natural uranium slugs in holes drilled into the graphite. The stack was made in the approximate shape of a sphere and only operated for a few minutes for that first experiment before Fermi ordered the insertion of a cadmium control rod to stop the chain reaction and shutdown CP-1. The reactor was used for initial experiments but could only operate to a maximum level of 200 W to minimize radiation exposure to researchers from the unshielded nuclear reactor. CP-1 ceased operation on February 28, 1943 to dismantle it and move the reactor out of the city for further experiments ([American Nuclear Society, 1992](#)).

The reactor was moved in 1943 to a building called Site A, in an area of the Argonne Woods outside of Chicago. This was to have been the originally proposed location for the first criticality experiment, but the building was still under construction in 1942. The

reconstructed reactor was now called CP-2 and was reconfigured to be more block-like and surrounded by 1.5 m of concrete for shielding, except for the top which was primarily lead and wood (Fig. 1 below). CP-2 was large, about 9 m wide and long and 6 m high. The reactor contained 52 tons of natural uranium and was air-cooled. Heating of the graphite was a problem and the CP-2's normal maximum power level was 1 kW, but it occasionally was operated to 5 kW or 10 kW for short periods for specific experiments. CP-2 should be considered the first true research reactor with an extensive experimental program measuring reactivities of various materials as well as the reactor's operating characteristics. The early experimental program supported the development of later reactors designed to support the war effort to produce plutonium but CP-2 continued to operate until 1954.

The X-10 reactor was constructed at the Clinton site near Oak Ridge, Tennessee and started up on November 4, 1943. The X-10 was based on the CP-2 and was originally designed to be a prototype or pilot plant for the plutonium production reactors to be built at the site near Hanford, Washington. The X-10 was air-cooled by large blowers and could operate up to 4 MW. The X-10 was built as a demonstration facility and was used to produce small amounts of plutonium and for research developing the separation chemistry needed for the materials expected to come from the production reactors. Another important use of the X-10 was as a training platform for the operators and engineers who would be needed to operate and maintain the reactors at Hanford that were under construction. The X-10 operated for research purposes until 1963. The Belgian BR-1 reactor located in Mol, Belgium started up in 1956 and is very similar in design to the X-10 reactor. The BR-1 continues to operate today for research and education with a maximum power of 4 MW.

The CP-3 (shown in Fig. 2 below) reactor was designed and built at the Argonne site near CP-2 under the direction of Walter Zinn as an alternative to the graphite-natural uranium reactor for plutonium production. This reactor used heavy water, D_2O , as the moderating material and started up on May 15, 1944. As the heavy water was a more efficient moderator and reflector, CP-3 was much smaller than the CP-2 reactor. Being more compact, the average neutron flux in CP-3 was much higher for a given power than the graphite-natural uranium reactors. The later Canadian ZEEP (Zero Energy Experimental Pile) and NRX (National Research Experimental) reactors were of this design although NRX was light water cooled. The JEEP (Joint (Norwegian-Dutch) Establishment for Nuclear Energy) reactor which operated (critical in 1951) at Kjeller, Norway was very similar to the CP-3.

The final type of research reactor constructed and operated during World War II was the Water Boiler reactor which was a homogeneous mixture of enriched uranyl sulfate and light water operated at the Los Alamos site. The reactors were operated below the actual boiling point of water but the bubbles produced from radiolysis made them appear to be boiling. The first reactor of this design started up on May 9, 1944 and was called LOPO (Low Power) because it operated at less than 1 W. This reactor provided valuable information to improve critical mass calculations for the nuclear bomb design. LOPO only operated for a few months before it was reconfigured in December 1944 as HYPO (High Power) and began operation to a maximum power of 5.5 kW fueled by enriched uranyl nitrate in light water. A version of HYPO was the first research reactor to operate in the United States outside the U.S. government and the first civilian licensed research reactor and began operation in 1953 at what is now known as North Carolina State University.

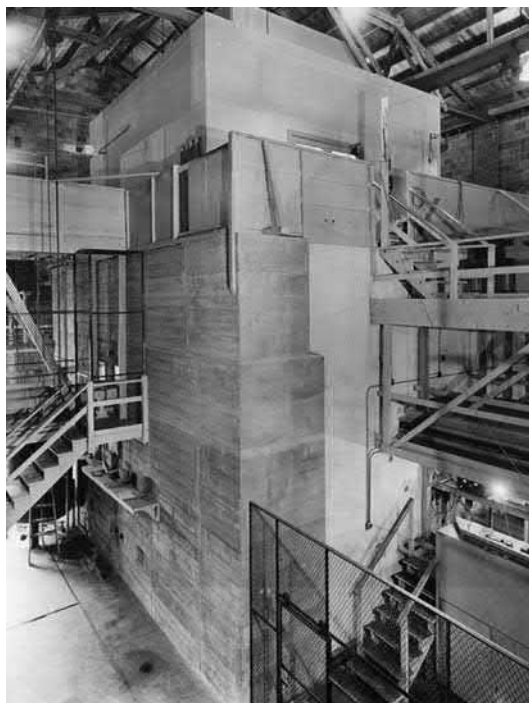


Fig. 1 CP-2 (reconstructed CP-1) research reactor in 1943.



Fig. 2 CP-3 operating in 1944.

Post-War research reactor development and activities

In the years from 1946 to 1954 there was great interest in expanding the use of nuclear reactors as a research instrument and the production of propulsion power and electricity from steam generated by the heat of nuclear fission. The use of fissile materials was strictly controlled by the government and much of the research was classified at the time. During this period, some of the higher power reactors were used to demonstrate liquid metal cooled fast spectrum reactors (Experimental Breeder Reactor-1, EBR-1) and the possibilities of operating submarines using nuclear power (Submarine Thermal Reactor, STR). These demonstration or prototype reactors were important to show engineering scale-up of concepts but are not considered true research reactors as they were built for a single purpose with very little flexibility for experimentation.

Canadian scientists had participated in many early experiments using the CP-2 and CP-3 reactors during and immediately after World War II. The National Research Experimental (NRX) reactor located at Chalk River in Ontario started up in August 1947. The NRX was based upon the CP-3 design and initially operated at 10 MW fueled by natural uranium, cooled by light-water, and neutron moderated by heavy water surrounding the fuel elements. For a time, it was the most intense neutron source in the world. The NRX suffered an operator-initiated accident in December 1952 which caused a rapid power transient and a cladding rupture of several fuel rods in the reactor. Lessons learned from that accident concerning reactor safety and redundant protection systems were applied towards future reactor designs. The NRX core was replaced and the reactor power was raised to 42 MW in 1954. A later Canadian-designed research reactor, the National Research Universal (NRU), began operation in 1957 fueled with natural uranium and operating at 200 MW. The NRU was converted to enriched uranium in 1964 and eventually operated using low-enriched uranium fuel at 135 MW before it was permanently shutdown in 2018.

The CP-3 reactor operated for experiments until 1950 when the fuel was then removed due to suspected corrosion of the aluminum cladding around the fuel rods. The reactor was upgraded and redesigned using enriched uranium fuel and restarted as CP-3' (CP-3 Prime) in May 1950. CP-3' was shutdown permanently 4 years later on the same day as the CP-2 shutdown. CP-3' was replaced by the CP-5 reactor at the Argonne National Laboratory in 1954 and Site A was decommissioned. The CP-5 research reactor was based upon the CP-3' design and was also a heavy-water reactor with enriched uranium fuel. CP-5 operated at a power of 5 MW and was used as a strong neutron source for materials irradiations and to create neutron beams. The CP-5 reactor design was later used as a model for many research reactors that used heavy water coolant built around the world (e.g., NBSR in the US, DIDO in the UK, and HIFAR in Australia).

It was identified very early on while testing in CP-2, CP-3, and the higher power production reactors that materials exposed for a long time to intense neutron and gamma radiation could change physical properties. In 1944, work had begun at the Clinton Laboratories near Oak Ridge, Tennessee to design a high flux reactor that could be used to test materials that would potentially be used in later reactor designs. The project was approved in 1946 but the U.S. Atomic Energy Commission decided in 1949 that the 30 MW reactor should be built in a remote area near Arco, Idaho. The Materials Testing Reactor (MTR) was fueled by highly enriched (93%) uranium and cooled and moderated by light water. Beryllium and graphite were used as neutron reflectors. The MTR went critical on March 31, 1952 and was the highest flux research reactor for many years. The MTR power was eventually raised to 40 MW in 1955 with an average thermal neutron flux of 3×10^{14} neutrons/cm²/s. The MTR was designed to be a flexible testing reactor and had over 100 irradiation facilities and six horizontal neutron beam tubes. The MTR fuel was designed to produce a more

compact reactor by using plate fuel rather than pins or slugs. The high thermal conductivity of the flat plates permitted higher power densities and the resultant high neutron densities while still being sufficiently cooled. The MTR fuel plates were an aluminum-uranium mixture clad in a “sandwich” of aluminum cladding. The fuel development was led by Eugene Wigner at the Clinton Laboratories (Rosenthal, 2010). This fuel was so robust and useful that the basic design was used for many reactors constructed world-wide and became universally known as “MTR-type” reactor fuel. Prior to MTR’s construction, the reactor vessel and piping were mocked-up at Oak Ridge facilities to perform mechanical and hydraulic testing of the MTR cooling system design before it was constructed. In 1950, fuel and beryllium were loaded into the vessel to perform low-power reactor physics testing to evaluate and improve the neutronic design of MTR prior to its initial criticality. The reactor was called the Low-Intensity Test Reactor (LITR) and was eventually modified and improved to operate up to 3 MW.

As nuclear power and propulsion reactor programs advanced there was a need to evaluate how to shield the intense gamma and neutron radiation produced from the operating reactors. Materials inserted into or irradiated by reactors had provided information on nuclear cross-sections and radiation effects but could not provide a test platform to evaluate new radiation shielding materials. The Bulk Shielding Reactor (BSR), built at ORNL, used the MTR-type fuel and was constructed in a large tank or pool of light water (6 m by 12 m and 7 m deep) to provide flexible access and insertion of large radiation shield prototypes. The BSR (Fig. 3 below) went critical in December 1950 and initially operated up to 100 kW on natural circulation cooling. A down-flow forced cooling system was later installed that allowed operation up to 2 MW. The BSR design was low cost to construct and was often called a “swimming pool reactor.” The fundamental BSR design was used in dozens of research reactors worldwide because it was inexpensive to construct and relatively easy to operate.

On December 8, 1953, U.S. President Dwight D. Eisenhower gave his famous “Atoms for Peace” speech at the UN General Assembly which helped establish the International Atomic Energy Agency (IAEA) in 1957. His speech and the 1955 International Conference on the Peaceful Uses of Atomic Energy in Geneva propelled the rapid growth of peaceful nuclear research around the world and the accelerated design and deployment of research reactors over the next two decades. The number of research reactors operating globally in 55 countries peaked around 1975. Most of the research reactors operating today are over 50 years old so there are ongoing efforts to maintain and upgrade these facilities when possible. In several cases, countries with highly utilized research reactors are planning to build replacement facilities that are designed with advanced safety features and enhanced utilization facilities.

Research reactor design approaches

Research reactors are fundamentally neutron sources, but some operate at such low powers (less than 1 W) that they are called zero-power or critical facilities. A key differentiator of research nuclear reactors from other reactors is whether the heat of the fuel is used



Fig. 3 ORNL bulk shielding reactor (1950).

to generate steam. The steam may be then used to generate electricity or provide propulsion. There have been nuclear thermal rockets designed and tested for propulsion using high temperature gases, but these would still not be classified as research reactors. The use of the research reactor is somewhat related to the intensity of the neutron flux in the reactor core. There is a significant variety in the design of research reactors. The following provide the scope for possible design approaches when designing research reactors:

- Fuel materials: U-Al alloy; Uranium silicide U_3Si_2 ; Uranium-Zirconium hydride UZrH_x (x is typically 1.6); Uranium dioxide UO_2 ; metallic uranium. UZrH_x is being used exclusively in TRIGA-type research reactors; its unique feature is that it provides a strong negative reactivity feedback upon fuel temperature increase that enables operating TRIGA reactors in a pulsed mode. U_3Si_2 has been used as a substitute for U-Al dispersion fuels and alloys when converting some reactors to operate with less than 20% enriched uranium rather than with HEU.
- Uranium enrichment: Highly enriched uranium HEU—had been a common fuel in many research reactors; Low enriched uranium (LEU)—up to 20% of ^{235}U —the common fuel in most contemporary research reactors; Natural uranium—feasible only in heavy-water moderated research reactors.
- Fuel element geometry: plate type fuel or fuel rods.
- Moderator type: Regular water H_2O ; heavy water D_2O . Most research reactors use H_2O moderators.
- Coolant type: Most research reactors use H_2O but some use D_2O for coolant.
- Reflector type: Most research reactors use H_2O reflector but D_2O cooled reactors use D_2O for reflector as well. Some H_2O cooled research reactors use beryllium or D_2O reflectors. Since the probability of neutron absorption in D_2O and in Be is significantly smaller than in H_2O , the use of D_2O or Be enables to have a higher thermal neutron flux in various experimental facilities in and around the core.
- Cooling mode: Most research reactors use natural coolant circulation for removing the energy released by fissions. High flux reactors usually use forced circulation cooling.
- Core containment: Most research reactors use open water pools with the core installed near the bottom. Heavy-water moderated contain the core in a closed vessel. High flux research reactors use a closed reactor vessel with forced circulation water coolant/moderator.
- Designed for high thermal neutron flux: The neutron flux level is directly proportional to the power generated per unit fuel volume (power density) and inversely proportional to the fissile fuel concentration per unit fuel volume. The power density can be maximized in two ways: (1) By maximizing the ratio of the fuel surface area in contact with the coolant to fuel volume; that is, by using plate-type fuel. (2) By maximizing the heat flux from the fuel to the coolant; that is, by using forced circulation cooling.
- Maximized neutron flux to core power ratio: As the primary product of many research reactors is the neutron flux available for multitude of experiment, it is desirable to minimize the reactor power for a given neutron flux level; a higher power implies higher rate of fuel consumption and more expensive heat removal systems. This objective can be achieved by minimizing the outer core surface area. The most compact critical core using fuel with a given enrichment can be achieved using water moderator and beryllium or heavy-water reflector. Using heavy water moderator enables designing research reactors that are fueled with natural rather than enriched uranium but the minimum core volume of such reactors is much larger than that of a light-water cooled reactors; the latter must use, though, enriched uranium fuel. Increasing the enrichment of the fuel will minimize the parasitic absorption by uranium-238 and other core materials which will enable a smaller core volume.

Low-power, low-flux reactors operating at universities are used for teaching fundamentals of reactor physics and operations but are often also used for neutron activation of materials for testing and analysis in student research projects. Higher power research reactors enable more complicated experiments that require higher neutron intensities. These higher flux reactors produce most of the world's industrial and medical radioisotopes. Lower power (< 5 MW) research reactors may have horizontal neutron beam tubes but the beam intensity is relatively low for high quality neutron scattering experiments with a low signal to noise ratio. The highest thermal flux reactors operating today can only achieve approximately 10^{15} neutrons/cm²-sec flux levels due to cooling limits for higher power densities.

Categories of research reactors

There are many ways to categorize research reactors (fuel, coolant, moderator, etc.) but the function or utilization of the reactor is generally related to the power level and neutron flux available ([International Atomic Energy Agency, 2016](#)). The lowest powered research reactors operate at “just critical or near-critical” with minimal thermal power (generally less than 1 W) generated. These critical assemblies or zero-power reactors are reconfigurable to test different reactor designs, fuels, or moderators, and benchmark computational models. Some universities have these reactors and use them for teaching reactor operations and performing reactor physics laboratories. In the last decade, neutron producing particle accelerators have been coupled to some of these facilities to evaluate models for accelerator-driven subcritical systems (ADSS) that could be used to transmute or “burn” high level nuclear waste. The design of most of these zero-power reactors allows precise control of reactor physics measurements to confirm models. These reactors are not used for irradiation experiments because of the very low neutron flux available.

One such low power research reactor that would be used more for training of engineers and scientists is the Argonaut reactor. The name is actually an acronym for ARGOnne Nuclear Assembly for University Training. The first Argonaut went critical in February 1957 and operated at 10 kW. The reactor was used to teach reactor theory and physics and operated at Argonne National Laboratory (ANL) until 1972. The reactor was not decommissioned at that time but was dismantled and shipped to Taiwan to become a training

platform for that country. A second Argonaut was built specifically for the second International Conference on the Peaceful Uses of Atomic Energy in 1958. The reactor was assembled for conference attendees, brought to criticality at the conference, and then dismantled and shipped back to ANL. There only a few Argonaut reactors still operating but one is actively used today for teaching students at the University of Florida.

Another research reactor that was designed specifically as a training platform are the AGN-201 reactors. These reactors were designed and built by Aerojet-General Nucleonics in the late 1950s. The AGN-201 is factory manufactured, relatively portable, homogenous, thermal reactor. The reactor core is unusual because it composed of a stacked series of circular disks composed of a mixture of polyethylene and uranium oxide (19.75% enriched). A horizontal tube passes through the reactor to allow irradiation and activation of materials for experiments. A maximum power level of less than 5 W limits the level of experimentation to nuclear engineering classes or laboratories. Many universities and colleges used these small reactors in the 1960s and 1970s but most have been decommissioned. There a few AGN-201 reactors still operating around the world. Three university programs that actively use the AGN-201 reactor for teaching are at Idaho State University, New Mexico State University, and Texas A&M University.

The next level of research reactor type and use would fall under a category of education and research reactors. These reactors operate from roughly 5 W to 5 MW and are often associated with a university nuclear education program. Reactors built for university research are often a pool-type reactor with horizontal beam tubes for maximum access and use of the reactor for irradiation experiments. Because the highest neutron flux intensity would be found in the reactor interior, special irradiation locations or pneumatic systems may be designed to utilize this higher flux. Thermal neutrons leaking from the core may be put to use by collimation in a horizontal guide tube and creating a neutron beam. Such beams may be used for neutron radiography or cross-section measurements. The lower neutron flux available in the areas external to the reactor core and in the beams usually require longer irradiation times to compensate and provide sufficient fluence or counting data. Research reactors include the pulsing or transient reactors which operate at relatively low steady-state powers but are able to rapidly raise reactor power for high-power (> 1200 MW) transients that last less than 1 s.

The final grouping or category of research reactors would be considered to be the high-performance research reactors. These reactors were originally designed to provide intense neutron fluxes for irradiation testing of materials or radioisotope production. Many of these facilities were also designed with numerous neutron beam ports which have enabled great advances in neutron scattering science. Unlike lower power facilities at most universities or many research centers, the high-performance research and testing reactors will usually operate 24 h a day over several shifts. These larger research reactors were usually constructed as the main facility of a multidisciplinary nuclear laboratory. Ancillary facilities supporting the reactor may include hot cells for examining irradiated materials and radiochemistry laboratories.

Fuel conversion of operating research reactors

The 1954 Atoms for Peace Program encouraged the export of research reactors to many other countries to support the development of peaceful uses of nuclear materials. Many of the early exported research reactors utilized low-enriched ($< 20\%$) uranium (LEU) or natural uranium fuels. However, these fuels limited research reactor capabilities, so the United States and the Soviet Union began to export reactors and reactor fuels utilizing highly enriched uranium (HEU). In 1978, an international effort lead by the United States to reduce the civilian use of HEU in research reactors worldwide was created. This was called the Reduced Enrichment for Research and Test Reactors Program (RERTR) which had an initial goal to convert as many higher powered (> 1 MW) research reactors as possible to LEU fuels. While many low-powered research reactors were relatively straight-forward to convert to LEU fuels without reducing the research capabilities, other research reactors were more challenging to convert. Higher powered and higher flux reactors were designed to operate with the higher U-235 densities of HEU. These fuels had already reached the uranium concentration limits of the current technologies available at the time. Under RERTR, a fuels development program was initiated that could allow conversion of the higher performance reactors without a loss of research capability. Early successes included the qualification of an LEU silicide-aluminum dispersion plate fuel in 1988. This fuel achieved uranium densities up to 4.8 g/cm^3 which allowed the conversion of a number of research reactors that could not convert previously ([U.S. Nuclear Regulatory Commission, 1988](#)). A high density, LEU TRIGA fuel that achieved 45-wt% was qualified and accepted in 1987 and permitted conversion of all U.S. TRIGA reactors from HEU FLIP (Fuel Life Improvement Program) fuels ([U.S. Nuclear Regulatory Commission, 1987](#)).

In 2004, the U.S. Department of Energy and National Nuclear Security Agency (NNSA) started a new program, the Global Threat Reduction Initiative (GTRI), that would continue to support the original RERTR goals but would now include efforts to address nuclear and radioactive material security and control. The GTRI program transitioned to become the Materials Management and Minimization (M3) program within NNSA. The M3 program compliments and supports LEU fuel development and conversion efforts in the European Union (under EURATOM), the International Atomic Energy Agency (IAEA), and the Russian activities (Rosatom). As of 2019, several potential high-density LEU fuels are under development that could allow the conversion of operating high-performance research reactors and become potential fuels for next generation research reactors.

Regulation and safety of research reactors

Each country that uses nuclear materials, radiation, or operates a nuclear reactor of any kind is expected to develop methods or laws that will regulate the use of nuclear materials and the resulting byproducts to ensure the safety of the public. The International

Atomic Energy Agency (IAEA) was not created to regulate the use of nuclear materials in member countries. The IAEA's purpose is to promote the peaceful and safe use of nuclear energy science and technology, and to inhibit its use for any military purpose, including nuclear weapons. There are currently 171 IAEA member states and while some of these countries have operating nuclear power plants, many more of the member states only operate research reactors for education and research and are only planning a nuclear power program. The IAEA supports the member states regulatory bodies and provides training and coordination to ensure a reasonable level of nuclear safety is maintained.

Because research reactors operate at lower powers, have a lower fission product inventory and lower cooling requirements after shutdown they are considered lower risk facilities as compared to nuclear power plants. Many countries that have both operating nuclear power plants and research reactors apply a graded approach to regulation and oversight due to the lower risk to the public and the environment. The IAEA works to harmonize regulation of research reactors and their applications across all member states by providing workshops, support missions, and documents generated from multi-country work groups.

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Applications of Research Reactors

Nuno Pessoa Barradas, International Atomic Energy Agency, Physics Section, Vienna International Centre, Vienna, Austria

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Glossary

BNCT Boron neutron capture therapy
DNC Delayed neutron counting
E&T Education & training
ENAA Epithermal neutron activation analysis
FNT Fast neutron therapy
GMP Good manufacturing practices
HEU Highly enriched uranium
I&C Instrumentation and control systems
IAEA International Atomic Energy Agency
ICP-MS Inductively coupled plasma mass spectrometry
INAA Instrumental neutron activation analysis
LEU Low enriched uranium
NAA Neutron activation analysis
NDT Non-destructive testing
NTD Neutron transmutation doping
PALS Positron annihilation lifetime spectroscopy
PGA Prompt gamma analysis
PGAA Prompt gamma activation analysis
PGNAA Prompt gamma neutron activation analysis

PIXE Particle-induced X-ray emission
 RI Radioisotope
 RNAA Radiochemical neutron activation analysis
 RR Research reactor
 SANS Small angle neutron scattering
 SPND Self powered neutron detectors
 STEM Science, technology, engineering and mathematics educational pathways
 U-SANS Ultra small angle neutron scattering
 XRF X-ray fluorescence

Introduction

Research reactors (RRs) contribute to improve the health of people, to combat climate change, to develop the materials that will underpin the technologies of the future, to educate the new generations of professionals in nuclear science and technology and in many other fields, to generate new knowledge for humanity, and to enhance prosperity and development of society ([International Atomic Energy Agency, 2014a](#)).

Millions of people benefit from diagnosis and treatment of health disorders with radioisotopes produced in RRs. RRs have been essential in developments in nuclear power, and are used for research and development of other energy sources such as hydrogen cells and batteries. Materials irradiated at RRs as part of their development are present in numerous devices and technologies. Elemental analyses with neutrons contribute to archaeological and cultural heritage studies, to forensics and food safety, to study air quality and the environment, to industrial applications in mining, forest and oil industries. Neutron beams enable leading edge scientific research and non-destructive materials characterization and testing in nearly every area of application including microelectronics, aerospace industry, soft and biological matter, astrophysics and innovative medical techniques.

The number of RRs and their main applications and countries involved are shown in [Table 1](#), as reported in the International Atomic Energy Agency (IAEA) RR database ([International Atomic Energy Agency, n.d.-a](#)). Not all RRs can implement all these applications, as some require certain neutron flux levels ([Table 2](#); [International Atomic Energy Agency, 2014a](#)), specific design features and supporting infrastructure ([International Atomic Energy Agency, 2007a](#)). A RR of a given power level will likely not engage in all possible applications, but will choose those for which it finds users and customers ([International Atomic Energy Agency, 2013a](#)). Such decisions should be underpinned by an effective and achievable strategic plan for utilization ([International Atomic Energy Agency, 2017a](#)), to support the long term sustainability of the RR facility in a changing world.

Education and training

Lectures and courses at universities and professional training courses in nuclear sciences and technologies are often complemented by practical experiments at RRs. Around two thirds of operational RRs are engaged in education and training (E&T) (see [Tables 1 and 3](#)), making this the most common application of RRs worldwide. Education concerns activities with students as part of an

Table 1 Number of RRs and countries involved in the different applications of RRs, as reported in the IAEA RRDB ([International Atomic Energy Agency, n.d.-a](#)) in November 2020. NAA includes Prompt Gamma Analysis (PGA). Neutron therapy includes Boron Neutron Capture Therapy (BNCT) and Fast Neutron Therapy (FNT).

<i>Application</i>	<i>Number of RRs involved</i>	<i>Number of countries</i>
Education & Training (E&T)	162	50
Neutron Activation Analysis (NAA)	117	49
Radioisotope production	83	41
Neutron radiography	69	37
Material/fuel testing/irradiations	68	26
Neutron scattering	44	28
Geochronology	24	21
Si doping	23	15
Gem coloration	20	12
Nuclear energy research	17	10
Neutron therapy	15	11
Nuclear data measurements	14	7

Table 2 Potential capabilities of RRs in given applications, for different indicative RR power levels (International Atomic Energy Agency, 2014a). I&C is instrumentation and control.

Power level	Isotope				Geo-chronology	Transmutation effects			Neutron imaging	Neutron scattering	Positron source	Neutron therapy		Materials testing		
	E&T	NAA	PGA	production		Silicon doping	Gamma irradiation	Gemstone Coloring				BNCT	FNT	I&C	Materials	Fuels
<1 kW	F	S														S
100 kW	F	F	S	S	S		S		S	S		F			S	
1 MW	F	F	S	S	S	S	S	S	S	S	S	F	S	F	S	
10 MW	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	S
>10 MW	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F

S: Some capability (e.g. mostly for development or demonstration purposes); F: Full capability (e.g. for state of the art R&D or for commercial activity). See the glossary for definition of the acronyms.

Table 3 University and non-university RRs involved in Education & Training, as reported in the IAEA RRDB (International Atomic Energy Agency, n.d.-a), out of 236 RRs in operational or temporary shutdown status. E(≥ 100): RRs involved in E, reporting at least 100 students per year; T(≥ 20): RRs involved in T, reporting at least 20 trainees per year.

	E&T	E	E(≥ 100)	T	T(≥ 20)
University	59	57	20	43	14
Non-University	102	82	7	98	18
Total	161	139	27	141	32

academic program for undergraduate and postgraduate degrees. Training concerns activities with professionals for acquisition of specific professional competences and skills. It can address initial training of new staff or retraining of experienced staff, including those from nuclear utilities.

Low power RRs and sub-critical assemblies are often well suited for basic E&T (Boeck et al., 2015). Some university RRs are exclusively dedicated to E&T. Other RRs, often not directly affiliated to a university, offer their experimental facilities, regularly or sporadically for (under)graduate programs or training courses. Medium and high power RRs, primarily involved in provision of commercial product and services, may lack the flexibility of operation required for E&T. Some high flux RRs neutron sources operate as user facilities dedicated to research and development (R&D) with research mainly done by post-graduate students.

Education

Being part of a higher education institution or being located close to one with easy access to students of relevant subjects and courses is per se not sufficient to attract courses and students. RR managers and staff involved in experiments usually need to approach the university departments and propose courses, experiments and research opportunities that are adequate to the curricula and classes already taught. In some cases, RR staff are also appointed as professors.

University RRs are nearly always involved in education at some level (see Table 3). Nearly two thirds of RRs that report activity in education (International Atomic Energy Agency, n.d.-a) are non-university RRs (82 out of 139). This is reduced to around one quarter (7 out of 27) when only RRs that report 100 or more students per year are considered.

RRs are ideally suited to teach nuclear and radiological engineering, radiation protection and related subjects such as radio- and nuclear chemistry, nuclear energy, nuclear safety and security, radioactive waste management, nuclear applications, medical physics, nuclear medicine, etc. Typically, reactor physics and reactor operation experiments, radiation protection and radiation monitoring activities are conducted. The offer can range from a (series of) half-day experiments to an extended experimental course lasting several weeks.

RRs are also often used to teach other engineering studies such as power, electrical, mechanical and other engineering specialities that require, for example, non-destructive testing (NDT). Affiliations with universities allow for introducing the opportunities of neutron-based research and irradiations in applied sciences such as material science, biology, forensic science, environmental and earth sciences, and even humanities such as archaeology, anthropology, cultural heritage, etc. Besides physics, where reactor physics experiments can also be offered, utilization experiments are usually most appropriate for these subjects. Radiation protection is a cross-cutting subject to all (Fig. 1).

Typical reactor physics experiments at RRs include theoretical background, experimental methods, requirements and recommended equipment and safety precautions are described in International Atomic Energy Agency (2014b) and Rataj et al. (2010):

- Measurement of neutron flux spectra
- Measurement of reactor kinetics parameters



Fig. 1 Teaching students at the Czech Technical University VR-1 research reactor. Courtesy of Czech Technical University, Czech Republic.

- Subcritical multiplication and neutron source effects
- Control rod worth measurement and calibration
- Determination of reactivity, shutdown margin and reactor period
- Critical experiment and approach to criticality
- Reactor power calibration
- Power decay and delayed neutron measurements
- Temperature, void and material effects on reactivity.

Reactor operation experiments comprise the above while putting the emphasis on operations and reactor control. Examples of additional experiments are:

- Research reactor operation, including cold start-up to a certain power, stabilization of reactor period, change of power level at different rates, manual and automatic control, shut down, hot restart.
- Reactor instrumentation and control, both for analogue and digital (I&C) systems.

Radiation protection and monitoring experiments take advantage of the RR as a neutron and gamma-ray source and of the availability at the RR of a range of radiation detection and monitoring systems, shielding, irradiated samples, standard sources and personal protective equipment. Many relevant activities and experiments can be performed, including development of emergency scenarios and evacuation drills.

Utilization experiments can be based, in principle, on all experimental activities and R&D at the RR. The latter is often carried out by graduate and post-graduate students under supervision. Utilization experiments commonly offered include:

- NAA for trace element analysis, from sample preparation to irradiation and counting
- Radioisotope production, including preparation and use of radiotracers
- Neutron imaging
- Neutron diffraction, neutron scattering and time-of-flight experiments.

Training

Nearly all non-university RRs involved in E&T report activity in training, compared to only around 70% university RRs (see [Table 3](#)). More non-university than university RRs report training of 20 or more professionals per year.

The nuclear workforce can be divided in three categories ([OECD, 2012](#); [UK Department of Energy and Climate Change, 2015](#)):

- Nuclear: subject matter experts with specialized formal education in nuclear subjects
- Nuclearized: staff with formal education in a non-nuclear area, requiring development of nuclear skills
- Nuclear-aware: staff with generic skills requiring nuclear awareness training.

This categorization was developed for nuclear power. However, training of nuclear power plant staff has been reduced in RRs following the widespread use of reactor simulators by the nuclear energy industry. The categorization can be also used for staff from radiation protection and related professions, regulatory bodies, nuclear safety and security, research reactor operators and users, nuclear applications, etc., taking advantage of the access at the RR to different types and sources of radiation, radiation detection and monitoring systems, and flexibility in the use of the facility.

Training at RRs addresses well-defined training needs of nuclearized or nuclear-aware staff in short duration training courses, from a few days to some weeks (see [section “Education”](#)). It can also be aimed at the development of highly specialized competences for nuclear staff in extended training lasting up to 1 or 2 years, for instance through the IAEA International Centres based on Research Reactors (ICERRs) scheme ([International Atomic Energy Agency, n.d.-b](#)). In these cases, training in specific areas of utilization is also common.

Public tours and visits

Public tours and visits are one form of education. They can be offered to the general public (usually in Open Door Days), high school and university students, and to specific targeted groups such as local or regional decision makers and stakeholders. This can lead to increased support by the public for nuclear science and technologies, including nuclear energy, as it has been shown that people become increasingly supportive of nuclear when they feel better informed and their level of knowledge about the subject is higher ([Nuclear Energy Agency, 2010](#)).

Exposure of high school students to the exciting science, research and development done at RRs can increase their interest in science in general, and in nuclear sciences and technologies in particular. Students often exchange their experience to their colleagues, friends and relatives, with increasing use of school blogs and social media. Such advocacy can lead in the short term to a positive perception of the RR within the community, and more students can become encouraged to choose STEM (science, technology, engineering and mathematics) educational pathways. In the long term, increased awareness of the facility's capabilities may result in expansion of the RR utilization.

Procedures for access control during tours and visits need to be flexible and allow for entrance and movement of groups of people in designated areas of the RR hall, while respecting radiological protection, safety and security requirements, including age limits and constraints for pregnant and breastfeeding women. The tours, conducted by authorized operations staff, are usually preceded by a lecture with technical level adapted to the audience, focusing on the RR purpose, capabilities and contribution to societal goals. The tour can include observing a start-up, viewing the experimental hall, given instruments, the control room, hot cells, and other areas, use of models and mock-ups of the reactor, dummy fuel assemblies and other components. Displays such as posters or digital screens illustrate the work done at the RR and the explanations provided. The visits can also include demonstrations and practical exercises, for instance on radiological protection procedures and radiation detection.

Neutron activation analysis

Neutron Activation Analysis (NAA) is an isotope-specific analytical technique for the qualitative and quantitative measurement of the total masses of major, minor and trace chemical elements present in samples weighting from a few micrograms to several kilograms ([International Atomic Energy Agency, 2001a](#)). The technique is based upon the conversion of stable atomic nuclei into radioactive nuclei by irradiation with neutrons, and the subsequent measurement of the radiation (often gamma rays) emitted during the decay of these radioactive nuclei ([Fig. 2](#)). In principle, over 70 elements can be detected in amounts as small as micrograms to nanograms, or even less, depending on the physical characteristics of the element and other elements present in the object analysed ([Parry, 1991](#); [Ehmann and Vance, 1991](#); [Greenberg et al., 2011](#); [Grdeń, 2020](#)).

NAA is the second most common application, and the most common analytical technique at RRs ([Table 1](#)), from subcritical and critical assemblies to RRs with power ratings up to 200 MW ([International Atomic Energy Agency, n.d.-a](#)). RRs with power above 10 kW have thermal neutron fluxes already sufficient for most of the applications of NAA, and NAA is mostly done at RRs with power rating between 10 kW and 5 MW. The higher fluxes at these reactors lead to lower detection limits.

Most samples are multi-elemental, leading to a mixture of radioactive nuclei with short, medium and long half-lives after irradiation. The energy of the emitted gamma ray is characteristic of the emitter and is used to identify the element present. Some gamma rays of radionuclides of different elements have close-by energies, and a high intensity from one radionuclide can mask the signal from another. This may be resolved by taking advantage of the differences in half-lives; the initially masked signals



Fig. 2 NAA Technician loading a sample in the automated short-lived cyclic system at the University of Texas research reactor. Courtesy of the University of Texas Nuclear Engineering Teaching Lab, United States of America.

may become visible once the interfering activity has decayed away. In instrumental NAA (INAA), successive measurements are therefore taken after different decay times (for instance immediately after irradiation, 3–5 days later, and 2–3 weeks later). The combined analysis of these measurements may result in information on 50–60 elements. Less commonly, chemical separation of the irradiated sample leads to split it into fractions with only a few non-interfering radionuclides in each, leading to better detection limits. This destructive technique is called radiochemical NAA (RNAA) (Heydorn, 2000).

One common variant of INAA is epithermal NAA (ENAA), where only epithermal (and fast) neutrons are used by shielding the sample with a thermal neutron absorber such as cadmium or boron (Alfassi, 1985). This can increase the sensitivity to elements with higher epithermal than thermal activation rate, or in the presence of interfering radionuclides with different epithermal and thermal activation rates. NAA with fast neutrons is rarely done at RRs as it requires more complex absorbers for both the thermal and the epithermal neutrons; it is more common with accelerator based neutron sources, D-D or D-T generators and even with isotopic sources.

Delayed neutron counting (DNC) is a form of NAA to measure thorium and uranium by measurement of neutrons emitted by fission products or their daughters.

NAA requires appropriate irradiation positions with a well characterizable neutron spectrum. Short irradiations are usually done with a pneumatic (less commonly, hydraulic) sample transfer system, which for NAA and DNC should have transfer times around 1–2 s. Long irradiations can be done also with such systems or by manual insertion and removal of samples from the irradiation position. For ENAA, irradiation positions surrounded by thermal neutron shielding (e.g., a 1 mm thick cadmium sheet) are common. Counting of the induced activity is done with a gamma ray spectrometer, usually employing a high resolution germanium semiconductor detector. Compton suppression systems can significantly improve the detection limits for some radionuclides, but increase the complexity and cost of the system. Automation of the NAA workflow and use of multiple counting stations can lead to capacities up to 1000 samples per month (International Atomic Energy Agency, 2017).

NAA is complementary to other analytical techniques for determination of elemental compositions, such as inductively coupled plasma mass spectrometry (ICP-MS), X-ray fluorescence (XRF), particle-induced X-ray emission (PIXE), and others. Strengths of NAA include (Grdeń, 2020):

- Real total bulk analysis for sample masses from micrograms to several kilograms (International Atomic Energy Agency, 2018) of objects of various dimensions and shapes (Grdeń, 2020);
- No dissolution of sample (except in RNAA), therefore preservation of sample and no losses of measurand;

- Accuracy not affected by the chemical state and physical form of the elements;
- Matrix independence; no need for matrix matched calibrators;
- Many adjustable experimental parameters for optimization to suit specific needs;
- High degree of isotopic and elemental specificity;
- High degree of accuracy; all sources of uncertainty can be quantified (Greenberg et al., 2011);
- Neutron activation of elements such as H, C, N, O, Si do not result in measurable induced activity. Many sample matrices become “transparent” for the signals from the trace elements of interest;
- RNAA is done with direct measurement of the chemical yield of the separation (by adding a known amount of non-radioactive carrier prior to the chemical separation), leading to highly accurate measurements;
- The measurement instrumentation is relatively inexpensive, robust and with very little maintenance requirements.

NAA is particularly well suited to application niches with solid samples that are either

- Difficult to dissolve completely;
- Easy to contaminate during preparation of the test portion;
- Unique and should keep their integrity; or
- For which other techniques are lacking matrix matched calibrators.

Some examples are plastics, soil, rock, bulk analysis of coal, ores and minerals, plant material, glass, catalysts, difficult elements such as halogens, arsenic, antimony and mercury, high purity compounds of carbon fiber, silicon and composites, air particulate matter, human or animal tissue, forensics, archaeological and cultural artefacts, and entire or very large test portions.

Prompt gamma analysis

In prompt gamma analysis (PGA) the gamma radiation emitted in the neutron capture (n,γ) reaction is detected (Grdeń, 2020; Molnár, 2004; Lindstrom et al., 2017). The terms prompt gamma NAA (PGNAA) or prompt gamma activation analysis (PGAA) are also commonly used even if the radiation emitted by the activated reaction product does not play a role.

Like NAA, PGA is a non-destructive isotope-specific analytical technique for the determination of presence of elements and of their amount in samples weighting from a few micrograms to several kilograms. The IAEA database for PGA has data for more than 80 elements (International Atomic Energy Agency, 2003). In a high flux RR, up to 40–50 elements may be measured with detection limits ranging typically from 0.01 μg to 1 mg (Paul et al., 2015). In practice, PGA is most useful for trace elements that cannot be easily (or at all) measured with NAA, such as H (Couet et al., 2012), B (Paul, 2005), C, N, Si, P, S, Cd and Gd. Some of these elements can be determined with other analytical techniques, but PGA has the same advantages that were listed for NAA above.

While in NAA the samples are first irradiated and then the resulting activity is counted after a certain decay time, in PGA the counting takes place while irradiating the sample. Hence PGA is performed in neutron beam positions, requiring adequate shielding, and only one sample can be measured at a time. Irradiation times of several hours are often required to attain the desired sensitivity, particularly for multielemental analysis, which leads to a much lower throughput of samples than can be achieved with NAA.

Activation rates are in general higher for cold neutrons. The most sensitive set-ups therefore use cold neutrons and suppress fast and epithermal neutrons and background gamma rays by using a curved neutron guide.

PGA with isotopic sources such as ^{252}Cf or portable DD/DT generators are widely used in the coal, power, cement, iron ore and steel industries for real time analysis of bulk materials, as well as for detection and identification of explosives, land mines and unexploded ordnance (Martin et al., 2000).

Radioisotope production

Radioisotopes play a key role in many areas of human activity with very high economic and social impact, including medicine, industry, agriculture and R&D. Most radioisotopes used in those fields are produced in RRs or accelerators (usually cyclotrons), with production in nuclear power plants being limited to a few important RIs with high demand, such as ^{60}Co and ^{137}Cs .

Research reactor radioisotopes

Some radioisotopes produced in RRs are shown in Table 4, together with the applications for which they are used (International Atomic Energy Agency, 2003). The list is by no means exhaustive, and many other RIs can be produced in RRs. Medical applications include diagnostic—imaging with radiolabeled molecules, therapy—using the radiation emitted by the RI to destroy affected cells, and palliation, i.e. pain-reducing therapeutic relief. Radiotracers are also used in a wide array of industrial applications including process control, chemistry and agriculture. Strong neutron emitters such as ^{252}Cf are used in industrial PGA, as start-up source in nuclear reactors and in well logging in the oil industry. In all these applications, including medicine and industry, there are opportunities for R&D, including radiotracer studies in scientific research.

Table 4 Some radioisotopes produced in RRs as a function of the thermal neutron flux. For each flux range, the radioisotopes indicated are additional to those shown in lower flux ranges. The operation cycle also determines the RIs that can be produced in a given RR.

Thermal neutron flux ($\text{cm}^{-2} \cdot \text{s}^{-1}$)	Some radioisotopes produced	Typical application
$< 10^{12}$	^{24}Na	Radiotracer in industrial and medical applications
	^{38}Cl	Radiotracer in industrial applications (organic chemistry)
	^{41}Ar	Radiotracer in industrial applications
	^{56}Mn	Radiotracer in industrial applications
	^{64}Cu	Therapy and diagnostic: cancer imaging and theranostic agent
$10^{12} - 10^{14}$	^{198}Au	Radiotracer in industrial applications Therapy: different cancers
	^{32}P	Radiotracer in industrial applications (hydrology) Therapy: radiosynoviorthesis, ^a blood cancer
	^{35}S	Radiotracer in industrial applications (agriculture)
	^{51}Cr	Radiotracer in biological and medical research Diagnostic: hematology
	$^{60}\text{Co}^b$	Radiotracer in industrial applications Industrial: sterilization; process control; materials science
	^{82}Br	Industrial: chemistry applications
	^{90}Y	Palliation: metastatic bone pain Therapy: neuroendocrine cancers, hepatocellular carcinoma, lymphoma
	^{99}Mo	Diagnostic: $^{99\text{m}}\text{Tc}$ (from ^{99}Mo) for oncology and cardiology
	^{125}I	Therapy: brachytherapy and prostate cancer
	^{131}I	Therapy: thyroid cancer, hyperthyroidism, neuroblastoma
	^{133}Xe	Diagnostic: lung function
	^{153}Sm	Palliation: metastatic bone pain
	^{165}Dy	Therapy: radiosynoviorthesis ^a
	^{169}Er	Therapy: radiosynoviorthesis ^a
	^{169}Yb	Therapy: brachytherapy
	^{177}Lu	Therapy: solid tumors
	^{186}Re	Therapy: cardiology Palliation: metastatic bone pain
	^{192}Ir	Therapy: brachytherapy Industrial: radiography
	^{203}Hg	Industrial: chemistry applications
$> 10^{14}$	^{75}Se	Palliation: metastatic bone pain
	^{89}Sr	Palliation: metastatic bone pain
	^{103}Pd	Therapy: brachytherapy for prostate and brain cancers
	$^{117\text{m}}\text{Sn}$	Therapy: radiosynoviorthesis ^a Palliation: metastatic bone pain
	^{188}W	Therapy: ^{188}Re (from ^{188}W) cardiology
	^{252}Cf	Industrial: strong neutron emitter

^aRadiosynoviorthesis is a procedure for the local treatment of chronic inflammatory joint diseases.

^bMost high specific activity ^{60}Co is produced at NPPs.

In RRs, RIs are produced by neutron induced reactions, capture or fission on a target material, leading either to activation of the target material or to fission. Fission has the advantages that many different RIs can be obtained and that higher specific activities can be reached with appropriate processing. In capture the common (n, γ) reaction is often used; in that case, chemical separation of the daughter from the target is often not possible, leading to lower specific activities. Whatever the method, three parameters are normally important to consider for the RIs produced, needing to be within the limits prescribed by the end user: the specific activity; the chemical purity (contamination by stable isotopes); and the radiochemical purity (contamination by other RIs).

The neutron flux determines which radioisotopes can be produced with given desired specific activities in a given RR. Commercial production of RIs usually requires the availability of high neutron fluxes. Medium neutron fluxes can be sufficient for many applications, even low scale commercial production or for local supply. Low flux RRs are also often used to produce RIs, for R&D or for demonstration (often within E&T programs) purposes. The operation cycle plays an essential role: to reach high specific activities of long-lived RIs continuous operation during extended periods is needed. For instance, commercial producers of ^{99}Mo typically operate in cycles of 3–4 weeks, but other RIs may require longer periods of several months. Operation in daily shifts limits the capability of an RR to short lived RIs only, which may be appropriate for low power RRs. Sequential neutron capture, needed to produce some RIs such as $^{186}\text{W}(n, \gamma)^{187}\text{W}(n, \gamma)^{188}\text{W}(\beta^-)^{188}\text{Re}$, is only efficient in very high flux reactors.

RI production involves a series of interrelated steps that go much beyond actual irradiation in the RR: fabrication of appropriate targets and their encapsulation; their irradiation; transportation of the irradiated targets to host cells and radiochemical processing

facilities; processing or encapsulation into sealed sources; quality control; and transportation to end users. Processing of the irradiated targets can be a complex and costly activity, requiring hot cell facilities and associated equipment. Particularly in large-scale commercial production, it is often done elsewhere by a specialized company. In such cases the reactor operator only needs the capacity to irradiate and handle the targets, including their packaging and transport, according to national (and international) regulations, respecting radiation protection requirements and ensuring safety at all times. Commercial medical RI production requires the establishment of a QA/QC system and respecting good manufacturing practices (GMP) requirements (World Health Organization, 2014).

Molybdenum-99 production

Radiopharmaceuticals are radioisotopes bound to biological molecules able to target specific organs, tissues or cells within the human body.

On a global scale, there are some 50 million nuclear medicine procedures with RIs per year (World Nuclear Association, 2020). Developed countries account for about one quarter of the world's population and over 80% of the demand of RIs. Correspondingly, estimated market growth rates were 0.5% for mature markets and 5% for developing markets (at the end of 2018 (Nuclear Energy Agency, 2019)), and production of RIs was one of the major motivations behind a majority of projects for new RRs in developing countries, in spite of the fact that entry of newcomers to the market is difficult due to the large investment needed, regulatory hurdles and commercial factors.

Diagnosis accounts for 90% of medical procedures involving RIs, and Technetium-99 (^{99m}Tc) alone is used in about 85% of diagnosis scans with radioisotopes. The short (6 h) half-life of ^{99m}Tc means it cannot be produced, processed and delivered in time to the end users. Instead, ^{99}Mo , which decays to ^{99m}Tc with a half-life of 2.75 days, is used as the source in $^{99}\text{Mo}/^{99m}\text{Tc}$ generators. The widespread use of ^{99}Mo is due to its ease of production and handling, high specific activity, reliable product quality, and the fact that ^{99m}Tc can be attached to several specific molecules, forming diverse radiopharmaceuticals for the diagnosis of many diseases, including certain types of cancers (Fig. 3).

Large scale production of ^{99}Mo , which decays to ^{99m}Tc with a half-life of 2.75 days, is made in a small number of RRs by fission of ^{235}U . Historically, highly enriched (HEU) targets were used, but low enriched uranium targets (enriched in ^{235}U by less than 20%) are available and most major producers either are already using LEU or intend to convert from HEU to LEU targets (International Atomic Energy Agency, 2013b; Nuclear and Radiation Studies Board et al., 2017). Low scale production can also be made from activation of natural Mo or enriched in ^{98}Mo targets (International Atomic Energy Agency, 2015a).

There are also only a few major processing facilities, and a global supply chain $^{99}\text{Mo}/^{99m}\text{Tc}$ generators is established taking into account the irradiation and processing capacities, the complex logistics established, as well as planned RR operation and outage schedules (e.g. due to planned maintenance or refueling). Nevertheless, unexpected simultaneous shutdowns of some of these RRs led to a global ^{99m}Tc shortage in 2009–10, and in 2019 the supply chain was more stable but still frail (Nuclear Energy Agency, 2019). The 2020 covid-19 pandemic led to some difficulties in transport and distribution of radioisotopes worldwide. At the same time, new concepts are being developed, including direct target irradiations at particle accelerators (International Atomic Energy Agency, 2017b) or using accelerator driven homogeneous aqueous solution subcritical assemblies (International Atomic Energy Agency, 2008a).

Geochronology and thermochronology

Research reactors can be used as tools for radiometric dating in geochronology and in thermochronology, which study, respectively, the age and the thermal evolution of rocks, minerals, fossils and sediments.



Fig. 3 Left: Molybdenum plate in holder. Centre and right: radiopharmaceuticals ready to be shipped. Courtesy of NECSA, South Africa.

Ar geochronology

Potassium is the eighth most abundant element in the Earth's crust, with around 2 wt%, and it is a major or minor component in many minerals. It has three natural isotopes including the radioactive ^{40}K , with 0.012% abundance and a 1248 million years half-life. It decays to the stable isotopes ^{40}Ca (89.28%) and ^{40}Ar (10.72%). By measuring the amount of ^{40}K and of radiogenic ^{40}Ar in a mineral, their age can be determined. ^{40}Ca is not useful because Ca is a common element in minerals.

K–Ar dating is used in geological problems since 1950. It relies on absolute determination of the abundances of both isotopes (requiring separate aliquots) and on several assumptions that cannot always be reliably verified. $^{40}\text{Ar}/^{39}\text{Ar}$ dating, developed in the 1960s, is based on a relative measurement on the same aliquot and is much more accurate. It also requires much smaller masses, on the sub milligram range, and is capable of determining the age of micrometer sized grains, from volcanic rocks as young as two millennia to rocks from the moon or as old as Earth itself.

$^{40}\text{Ar}/^{39}\text{Ar}$ dating is based on irradiating the sample with fast neutrons. The amount of potassium is determined using the $^{39}\text{K}(\text{n}, \text{p})^{39}\text{Ar}$ reaction, which has a threshold of 1.2 MeV. The ratio of ^{40}Ar to ^{39}Ar is determined with mass spectrometry (with outgassing due to heating of the sample), and is used to calculate the date. Interfering reactions on different elements and atmospheric ^{40}Ar not of radiogenic origin can be accounted for.

Minerals formed at high temperatures only started to retain the ^{40}Ar after a certain period of cooling. The age determined by $^{40}\text{Ar}/^{39}\text{Ar}$ corresponds to the time where diffusion loss stopped. Thermochronology takes advantage of this to determine the thermal history of the samples by stepwise heating them while performing mass spectrometry (McDougall and Harrison, 1999). Laser heating for spot analysis leads to age mapping of samples with micrometer spatial resolution (Kelley, 2002).

Research reactors are typically only involved in the irradiations and the analysis is often done off-site in specialized laboratories. The main requirement is a sufficiently high fast neutron flux. Typically a 1 MW power is needed for relatively “young” samples and 10 MW for “older” samples.

Fission track geochronology

Fission track dating is a relatively simple method for determining the age and thermal history of uranium-rich minerals such as apatite, zircon, titanite and natural glasses, with ages starting at around 100,000 years.

When ^{238}U undergoes spontaneous fission, the fission products create tracks in the material. By measuring the fission track density and the amount of uranium in the sample, its age can be determined. Irradiation with thermal neutrons leads to fission of ^{235}U , which creates more tracks. Counting the track density before and after irradiation, the ^{235}U content is also determined, providing the total uranium content through its known natural abundance (Tagami and O'Sullivan, 2005).

The fission tracks are revealed by chemical etching and counted visually. Since ^{238}U and ^{232}Th undergo fission when irradiated with fast neutrons, a highly thermalized spectrum is needed. In the thermal column of a medium power RR with fairly low flux, long irradiation times may be needed, up to several days. In a heavy water moderated RR the thermal neutron flux can be much higher leading to much lower irradiation times.

The fission tracks are progressively shortened and disappear when the rock is heated, depending on the temperature and time. Track fading and track length therefore give indication on the thermal history of the sample, and fission track thermochronology, usually done in off-site in specialized facilities, has been applied to problems such as tectonics, petrology, stratigraphy, hydrocarbon exploration and geomorphology (Malusa and Fitzgerald, 2019) (Fig. 4).



Fig. 4 Inspection of fission tracks after irradiation at the IEA-R1 research reactor of IPEN. Courtesy of IPEN/UNICAMP, Brazil.

Transmutation effects

Neutron and gamma radiation can induce changes in the properties of materials, opening fields of application of research reactors not only in research, but mainly in the provision of commercial products and services. Relatively high and uniform fluences are often required to enact the desired changes, and this activity usually requires a high or medium power RR with adequate irradiation positions.

Silicon transmutation doping

Neutron transmutation doping (NTD) is a method to introduce dopants in intrinsic or extrinsic semiconductors ([International Atomic Energy Agency, 2012](#)). It is widely used to dope silicon with phosphorus via the reaction $^{30}\text{Si}(n,\gamma)^{31}\text{Si}$ with subsequent beta decay to ^{31}P ($T_{1/2} = 2.62$ h). Irradiated silicon is used in high quality devices for high power applications such as thyristors and high-voltage power rectifiers, due to the very high accuracy and uniformity of the dopant concentration that can be achieved. Other semiconductors such as Ge ([International Atomic Energy Agency, 2012](#); [Malar et al., 2014](#)) can also be doped with NTD, but large scale commercial activity is concentrated in irradiation of silicon.

NTD requires specially designed irradiation positions, which should have fast neutron and gamma fluxes as low as possible, since they cause electrically charged defects in the silicon lattice and heating, respectively. Uniformity of irradiation at the ingot is achieved by rotating the ingot (and in some facilities, inverting it at half the irradiation time), and with flux screens designed taking into account the flux's axial profile to create a flat flux profile at the ingot. Cooling must be provided. The irradiation positions must be sufficiently large to accommodate ingots with 4, 5, 6 or 8 in. diameter, with demand currently concentrated on 6 and 8 in. ingots.

The main producers of NTD silicon have power ratings of 20 MW or more, but several RRs with power rating from 3 MW also have or are planning activity in the area ([International Atomic Energy Agency, 2012](#)) ([Fig. 5](#)).

Gemstone coloring

The color of gemstones can be changed by irradiation with high energy neutrons, electrons, gamma radiation or a combination of these, creating intrinsic color centers in the gemstones and leading to an increased market value. Neutron irradiation is most often applied to colorless topaz, that acquire a blue color with intensity that increases nearly linearly with the fluence ([Leal et al., 2007](#)). Typical fast neutron fluences for commercial purposes range from 10^{17} to 10^{18} cm^{-2} . The thermal neutron fluence must be minimized.

The resulting gemstones are activated, and decay times from several months to a few years are needed before they are introduced in the market, depending on the impurities in the original material ([Zhang et al., 2011](#)). The practice is forbidden in some countries and regulated in others, such as in the United States by the US Nuclear Regulatory Commission.

Gamma irradiation

Irradiated fuel assemblies can be utilized as source for a gamma irradiation facility, usually located in the service pool. The achievable dose rates can range from around 2 to 200 Gy/h, depending on the reactor power and fuel history. Applications range from irradiation of plants and seeds to the study of radiation effects on materials.

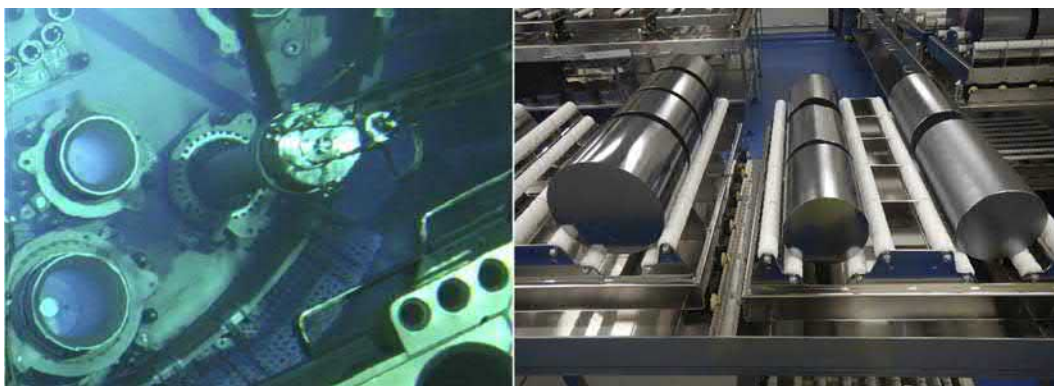


Fig. 5 Left: a silicon array being loaded into an irradiation facility of the OPAL research reactor. Right: irradiated silicon ingots. Courtesy of ANSTO, Australia.

Transmutation of minor actinides

Long-lived minor actinides are a component of irradiated nuclear fuel with significant radiotoxicity. Subcritical accelerator driven systems have been designed for transmutation of the minor actinides into shorter lived isotopes with reduced radiotoxicity and thermal output (Abderrahim et al., 2000). Research reactors can also be utilized to provide needed experimental data (Guidez et al., 2004).

Neutron imaging

Neutron imaging exploits the penetration of neutrons in objects to study their internal structure (International Atomic Energy Agency, 2008b, 2015b). Neutrons interact mainly with the nuclei and with magnetic fields. They are strongly attenuated by light elements such as hydrogen (and by a few specific heavier elements such as cadmium), contrarily to X-rays, which interact with the electron clouds and are more attenuated by heavier elements. Therefore, neutron imaging is often complementary to the corresponding X-ray technique. During the last two decades, most of film-based detection has been replaced by digital imaging, which led to the widespread adoption of 3D computed digital tomography, where the inner structure of the object is reconstructed based on many 2D images taken at different angles (Schillinger et al., 1999).

With time-series or stroboscopic techniques with frame acquisition time from a fraction of a second down to 10 μ s, dynamic processes can be followed (Tremsin et al., 2010; Tötzke et al., 2017, 2019). Advanced imaging techniques beyond imaging by absorption were developed, opening up new possibilities in NDT (Lehmann et al., 2017). These techniques take advantage of the availability of cold neutron beams, monochromators and beam conditioning and detection systems such as grating interferometers and neutron polarizers, to visualize the crystalline structure, crystal orientation, strain, magnetic domain walls and magnetic fields. Spatial resolutions down to 10 μ m have been achieved in some systems. Fields of view commonly range from 10 to 40 cm diameter or side length.

Neutron imaging, in its diverse variants, finds preferential fields of application in NDT, in particular for visualization of the inner structure of intact objects, which are either unique, precious or costly to replace, when operando analysis of dynamic processes is needed, and when it can provide unique information not available with other techniques. Applications include, for instance: 3D hydrogen distribution in fuel cells and other technological devices; efficiency and performance of engines and batteries; quality control of turbine blades and other mechanical components in the car, aviation and building industries; NDT of nuclear fuel and other nuclear materials; provenance and technology of objects in cultural heritage; and diverse applications in life sciences, geology, soil physics and materials research (Fig. 6).

A number of medium and low flux RRs have some neutron imaging capabilities, often using thermal neutrons. Important niches of applications can be found in such cases, including 2D radiography of industrial components or cultural heritage. At low fluxes on

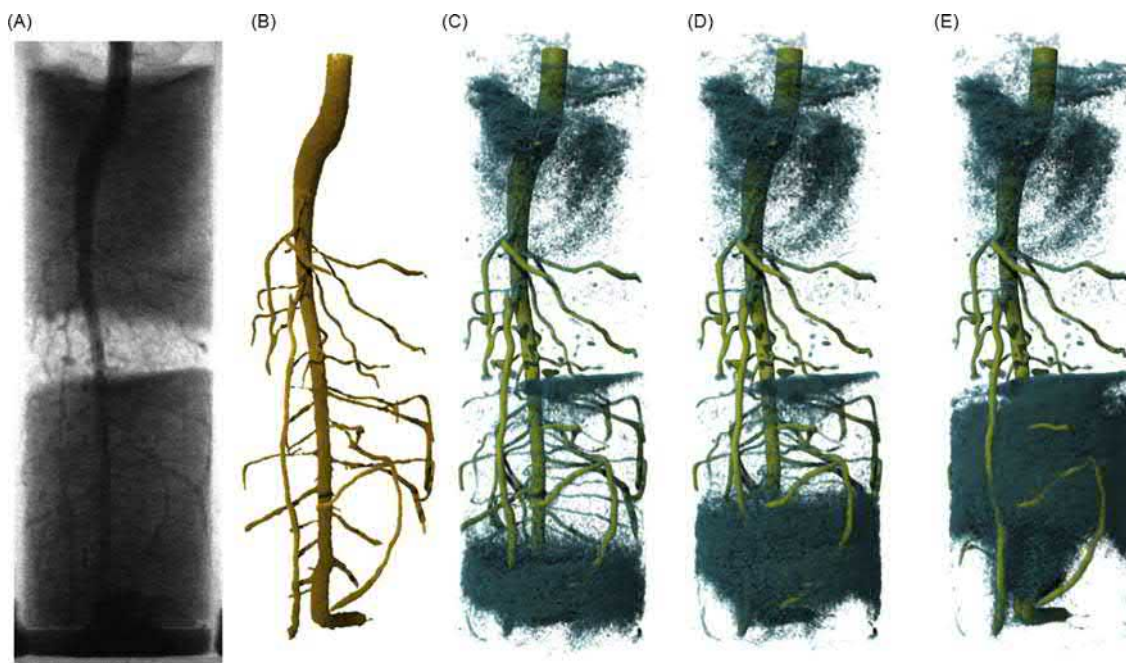


Fig. 6 Real time high speed neutron imaging of water uptake in roots of lupine (a) Radiographic projection image, (b) segmented root system; (c–e) water ascent. Courtesy of University of Potsdam and Helmholtz-Zentrum Berlin für Materialien und Energie GmbH, Germany.

the sample, long image acquisition times may impede 3D tomography. Advanced neutron imaging techniques are normally found at high flux RRs and spallation neutron sources.

Materials research

Innovative technological applications depend on the development of new materials with specifically designed and optimized properties, for which understanding the complex interplay between their properties and structure plays a crucial role. Materials research thus depends on appropriate characterization techniques capable of providing insights on matter from the atomic to large scale structure and ordering, 2D and 3D nanosystems, and on atomic and molecular dynamics and electric and magnetic structure.

Neutron scattering

Neutrons have wavelengths similar to interatomic spacing, kinetic energies similar to that of atoms and a magnetic dipole, which makes them ideal probes of atomic structure, atomic dynamics and magnetism, respectively (Guidez et al., 2004; International Atomic Energy Agency, 2006a,b; Furrer et al., 2009; Scattering et al., 2012). They have very large penetration depths (in the absence of strong neutron absorbers), which means they can probe bulk materials, and they are isotope-specific. Their energy and polarization can be controlled. These properties are of particular importance for neutron scattering, because they can be used in diverse variations of the technique, using specialized instruments, to probe different aspects of matter and length scales. Similarly to neutron imaging, neutron scattering is performed optimally at high flux RR or spallation neutron sources, and is often complementary to scattering techniques that use X-rays, light or electrons as probe. In soft matter studies, isotopic substitution by exchange or selective labelling of part of the system by deuteration is a key technique, taking advantage of the strong contrast in neutron scattering between hydrogen and deuterium.

The study of structural and magnetic properties at the atomic scale typically uses wide angle diffraction. The main techniques used are powder diffraction and single crystal diffraction, often using thermal neutrons, but hot or cold neutron sources are needed for some applications. Applications are, among others, phase, texture and structure determination in materials such as metals and alloys, inorganic compounds and ceramics and liquid and amorphous materials; location of light elements such as hydrogen in complex materials; stress analysis in mechanical components; determination of magnetic and spin structures in e.g. high temperature superconductors and molecular magnets.

Large scale structures, on lengths from the nm to the near- μm range, are typically studied with small angle neutron scattering (SANS) and neutron reflectometry, usually with cold neutrons. SANS is the most widely used neutron technique for structure determination, and applications include the study of molecular structure of macromolecules such as polymers, proteins and other biological molecules; cell membranes, surfactants and other surface systems; micelles, dendrimers, colloidal particles and microemulsions; and magnetic and spin systems such as magnetic nanoparticles, nanolayers, clusters, vortices and superconductors. Ultra-small angle neutron scattering (U-SANS) extends the scale range to $10\ \mu\text{m}$ (at the cost of loss of 2D information) and is applied to study micro-structural properties of polymer blends, structural steels, natural composites, complex fluids and colloidal crystals and alloys and other systems with complex morphology. Neutron reflectometry is used to characterize the thickness, structure, density, roughness and magnetic properties of thin films, multilayers and interfaces at the nanometer scale and to depths up to 100 nm. This finds applications in many fields, including the study of magnetic nanosystems; surfactants, complex fluids and liquid-liquid interfaces; kinetic studies of layer and interface evolution, polymer diffusion, protein unfolding and others; self-assembly and self-organization.

Atomic and molecular dynamic processes are studied with inelastic and quasi-elastic scattering, where the neutrons exchange kinetic energy with the lattice nuclei, thus creating or annihilating an excitation probing collective excitations of the lattice (phonons) and of the electron spin structure (magnons), as well as atomic diffusion. Both direction and energy of the scattered neutrons must be detected, and several techniques have been developed, such as triple axis spectrometry, time-of-flight (TOF) spectrometry, backscattering spectrometry and neutron spin echo. Thermal, cold or hot neutrons may be used depending on the system and properties under study. In spite of the complexity of some of the techniques, inelastic and quasi-elastic scattering have found application in many areas, including, for instance, determination of interatomic forces and magnetic coupling; behavior of quantum magnets and superconductors; dynamics of soft and biological matter; local and long range diffusion, including hydrogen in batteries and fast ion conductors; phase transitions (Fig. 7).

Neutron depth profiling

The elemental and isotopic composition of materials often determines their properties. NAA and PGAA provide the quantity of matter in samples, but are not sensitive to the change with depth of elemental concentrations, which is essential in the layered systems that make up nearly all modern technological materials. There are many depth profiling techniques based on ions, electrons and photons as probing particles. In many cases, successive sputtering of surface layers is employed in conjunction with a surface probing technique.

Neutron depth profiling provides quantitative depth profiles of some light elements such as He, Li, Be, B and others, independently of the sample matrix and non-destructively. Thermal neutrons lead to a nuclear reaction with emission of charged particles



Fig. 7 A Triple Axis Spectrometer installed at BATAN's RSG-GAS research reactor. Courtesy of BATAN, Indonesia.

(typically a proton, tritium, alpha, or other light nuclei) with energies ranging from a few tens or hundreds of keV to several MeV. These particles lose energy as they traverse the sample, and so emerge and are detected with reduced energy. The yield as function of energy can be easily translated into the concentration as function of depth.

Neutron depth profiling has been extensively used for real-time in-situ quantification of the changing lithium distribution in rechargeable lithium-ion batteries during charging and discharging. The Li depth profile can be determined over a range of 20 μm or more using the reaction ${}^6\text{Li}(n,\alpha){}^3\text{H}$, with acquisition times of a few minutes, which is sufficient for operando analysis (Liu et al., 2014), and 3D reconstructions of high and low regions of Li concentration are also possible (He et al., 2015).

Positron sources

Positrons are used as probes in the positron annihilation lifetime spectroscopy (PALS) which is a non-destructive technique that probes defects, voids, and open-volume defects in materials from the sub-nm scale to large vacancy clusters, with application in solid state physics and materials science, including semiconductors, metals and alloys, polymers and biological systems (Siegel, 1980). The depth distribution of the defects can be determined and, employing a positron microbeam, a sub-micrometer resolution lateral distribution can also be obtained. Doppler-broadening spectroscopy and angular correlation of annihilation radiation take advantage of conservation of momentum during the annihilation process to provide information on electron momentum distributions and are sensitive to the chemical environment of the annihilation site. Intense positron beams are used in atomic physics and nuclear astrophysics.

There are several paths to produce positrons, from β^+ isotopic sources, accelerator based sources, and RRs. In RRs, positrons can be produced by neutron activation of suitable targets such as copper, or by high energy gamma rays (from fissions in the core, or from neutron capture e.g. by cadmium) that are then absorbed by a high atomic number target such as platinum or tungsten (Hugenschmidt, 2017). The positrons produced are moderated to the desired energy. RRs with 100 kW can create around 10^7 moderated positrons per second. Well-designed facilities at high power RRs can reach two orders of magnitude higher values.

Neutron therapy

Neutron therapy uses neutron capture reactions such as ${}^{10}\text{B}(n,\alpha){}^7\text{Li}$, ${}^6\text{Li}(n,\alpha){}^3\text{H}$ or ${}^{157}\text{Gd}(n,\gamma){}^{158}\text{Gd}$. The reaction products (or, in the case of ${}^{157}\text{Gd}$, gamma rays and electrons) have MeV energies and a range in tissue comparable to cell dimensions, resulting in high Linear Energy Transfer (LET). The increased biological effects will be localized at the tumor cells if the ${}^{10}\text{B}$ is selectively concentrated there, with minimized damage to healthy tissue.

Distance to hospitals and public acceptance of nuclear reactors in a context of health care have hindered widespread use of RR based neutron therapy, with not more than a few thousand patients treated, many in the context of clinical trials. Strong side effects and inconclusive or insufficient clinical results, together with the development of new techniques including low-LET and high precision dose delivery methods also led to a reduction of the number of available facilities and of patients treated (Wagner et al., 2012). Current efforts are directed towards accelerator based neutron sources adequate for use in a hospital context.

Boron neutron capture therapy

Boron neutron capture therapy (BNCT) uses a thermal neutron beam for treatment of shallow tumors, sometimes combined with epithermal neutrons to extend the effect up to 10 cm depth (International Atomic Energy Agency, 2001b). Nearly 1400 patients were treated by 2020 in over 20 RRs, mainly in clinical studies in Japan, Finland, the United States, China (mainland and Taiwan), Netherlands, Sweden, Argentina, Czech Republic and Italy, with basic research done in further countries. Most of these facilities are no longer operating. Variable results have been obtained in treatment and palliation of a number of different cancers, and the efficiency of BNCT is still unproven. Research focus on malignant melanomas and brain tumors. Development of appropriate radio-pharmaceuticals, uptake and distribution of boron, treatment planning and treatment schedules, mixed neutron and gamma fields and tolerable doses and dosimetry are still challenging areas of research (Fig. 8).

Fast neutron therapy

Around 30,000 patients have been treated with fast neutron therapy (FNT) with accelerator based and isotopic neutron sources. Less than 2000 were treated in three RRs in Germany and the Russian Federation, of which one continues this activity (Specht et al., 2015), together with a number of accelerator based neutron sources. There is some clinical evidence that the addition of FNT to gamma treatment leads to superior results in some specific situations. FNT has been used in treatment and palliation of different tumors including salivary gland tumors, recurrent malignant melanoma and metastatic skin lesions.

Materials testing

Rrs can use the neutron and gamma radiation for testing and calibration of materials and instruments (Giot et al., 2017; International Atomic Energy Agency, 2007b, 2011). Even low power RRs can perform instrument testing and calibration, while testing of nuclear materials such as nuclear fuel and reactor structural components is normally done in high power RRs operated by experienced organizations such as national laboratories or research centers.

Neutron and gamma detectors and instruments need to be calibrated and tested to ensure correct operation. This can be made at nearly any RR, independently of power. Some of these instruments are self-powered neutron detectors (SPND), fission chambers,



Fig. 8 The BNCT facility at the RA-6 research reactor in Bariloche, Argentina. Courtesy of CNEA, Argentina.

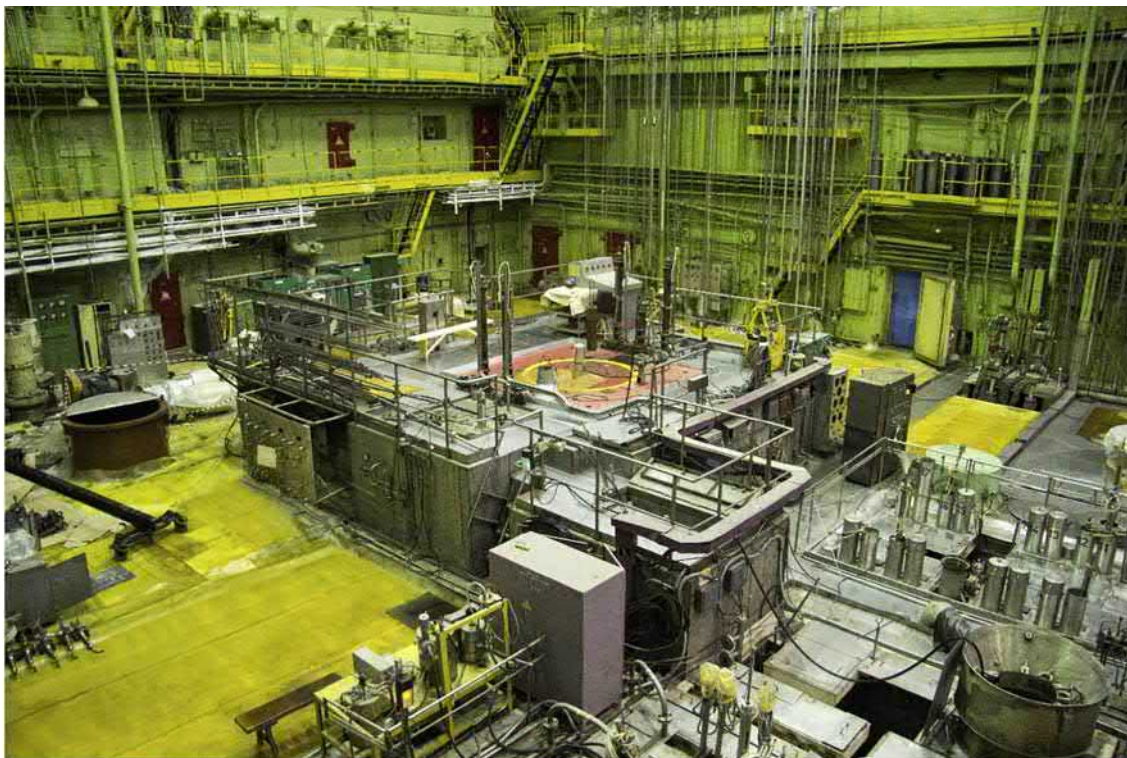


Fig. 9 The loop-type MIR.M1 research reactor, mainly used to test different fuels for nuclear reactors. Courtesy of the Joint Stock Company “State Scientific Centre—Research Institute of Nuclear Reactors”, State Atomic Energy Corporation Rosatom, Russian Federation.

nuclear heating instrumentation and radiological protection instruments (International Atomic Energy Agency, 2017c). A RR where calibration of instruments is an important activity may require accreditation.

Testing of materials typically uses fast neutrons with fluence in the range from 10^{17} to 10^{21} cm^{-2} and higher. The purpose is to relate the neutron fluence to the ageing and degradation of the material's properties. Typical applications include study of embrittlement, stress corrosion and ageing of reactor pressure vessels, other nuclear reactor materials, fusion reactor materials, and materials used in spallation neutron sources. Study of the failure of electronic components under irradiation is also common, including those for space applications.

Testing of nuclear fuels is usually made in instrumented closed loops, where environment parameters such as temperature, pressure and coolant flow and chemistry can be controlled to resemble the expected operating conditions of the fuel. The neutron spectrum should also be matched to real operating conditions. Two main types of experiments are performed, namely to study burnup under steady state conditions, and to study the behavior of fuel and cladding under transient conditions (Fig. 9).

Other applications

R Rs have been widely used in other areas, namely determination of nuclear data such as cross sections, branching ratios, decay constants, etc. which are essential for nearly all fields of application, and provision of benchmarking data for validation and verification of codes for calculation of e.g. neutronics, thermohydraulics and activation of materials. A number of R Rs were built to model proposed new concepts of power reactor or accelerator driven system, e.g. to validate purpose-developed codes, to determine parameters such as flux and power, and assess safety margins. R Rs are also used for shielding experiments, namely to test new shielding designs and materials. Some R Rs have been used to generate heat and electricity, or to test design concepts for new generation NPPs.

See Also: Global Market for Radiation Applications.

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Historical Survey of Test Reactor Programs at INL Over 70 Years

Harold F. McFarlane*, Idaho National Laboratory, Idaho Falls, ID, United States

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Abbreviations

A1W Aircraft carrier, first prototype, Westinghouse
AEC Atomic Energy Commission
ANP Aircraft Nuclear Propulsion
ARA Army Reactor Area
ATR Advanced Test Reactor
BORAX Boiling Water Reactor Experiment
CP-1 Chicago Pile Number One
DOE Department of Energy
EBR Experimental Breeder Reactor
ERDA Energy Research and Development Administration
FFTF Fast Flux Test Facility
GCRE Gas Cooled Reactor Experiment
GE General Electric
HTRE Heat Transfer Reactor Experiment
IFR Integral Fast Reactor
JCAE Joint Committee on Atomic Energy
LOCA Loss of Coolant Accident
LOFT Loss of Fluid Test
ML-1 Mobile Low-Power Reactor
NASA National Aeronautics and Space Administration
NRAD Neutron Radiography Facility
NRC Nuclear Regulatory Commission
NRF Naval Reactors Facilities
PBF Power Burst Facility
RTG Radioisotope Thermoelectric Generator
S5G Submarine reactor, 5th prototype, General Electric
SL-1 Stationary Low-Power Reactor
SM-1 Stationary Medium-Power Reactor
SNAP Systems for Nuclear Auxiliary Power
SPERT Special Power Excursion Reactor Test
STEP Safety Test Engineering Program
STR Submarine Thermal Reactor (also known as S1W)

SUSIE Shield Test Pool Facility
TAN Test Area North
TMI Three Mile Island
TREAT Transient Reactor Test
ZPR Zero Power Reactor
ZPPR Zero Power Physics Reactor (originally known as Zero Power Plutonium Reactor)

Introduction

This survey of Idaho's experimental, test, and demonstration reactor programs describes their history from the perspective of why they came to be, and what impact they had, or failed to have. Many of the technical details of the reactor technologies, and in a few cases the Idaho reactors themselves, are described elsewhere in this encyclopedia as well as in readily available reference materials. The U.S. Department of Energy (DOE) published the official compendium of Idaho's fifty-year, fifty-plus reactor history ([Stacy, 2000](#)), where more details can be found.

Idaho National Laboratory

INL is the acronym for Idaho National Laboratory. Officially created in 2005, for simplicity this history of the site since 1949 has been pulled together under the INL moniker, although the spirit of the story resides within the original designation—the National Reactor Testing Station. Many details of research-sponsoring agencies, site contractors, and key personalities are skipped in this short account of Idaho's role the development of nuclear energy.

Atomic Energy Commission selects Idaho

By the time that the Atomic Energy Commission (AEC) took over from the Manhattan Project in 1947, some of the nation's brightest people were considering power applications for nuclear energy. There were questions that could only be answered by experiments. Based on the lack of knowledge at the time, some of these experiments would pose unacceptable risks if sited near a metropolitan area. Aggressive reactor testing required a remote location. On March 1, 1949, the AEC announced that it had selected Idaho for the National Reactor Testing Station.

A proliferation of reactor technology concepts

With the capability to enrich uranium developed during the Manhattan Project, the U.S. could consider any conceptual reactor technology for commercial development. Technologies that did not require enriched uranium, such as heavy water moderated or graphite moderated reactors, were commercially pursued abroad. Exotic concepts for using solution fuels or very high temperature graphite reactors were investigated at other U.S. national laboratories. INL's principal focus was on light water cooled and liquid metal cooled concepts.

How many reactors were built at INL?

It depends. The commonly accepted number is 52. However, two of those reactors did not operate before the commercial concepts were eliminated. Another small reactor was simply borrowed from and returned to California. On the other hand, a critical facility was counted as a single reactor, whereas 63 unique benchmark and detailed engineering mockups reactors were built in the ZPR-3 facility and 21 were built in the large ZPPR facility ("reactors" ne.anl.gov). This survey highlights the reactors that had the biggest impact.

The right stuff

Among the oral histories of how Idaho was selected for the National Reactor Testing Station, there are three dominate threads: the local perspective of a successful political campaign; the dry official perspective of the AEC; and the one selected here—the perspective of the scientists and engineers responsible for the tests. Enrico Fermi's chief engineer for the world's first nuclear reactor CP-1, Argonne National Laboratory Director Walter Zinn recommended the Idaho site to the AEC. It had the right stuff for testing reactors under any conditions:

- Remote from any permanent development
- Access to cooling water
- Railroad infrastructure for heavy components
- Previously disturbed land
- Local support

Land and water

More than 61% of Idaho land is federally owned. Some 2250 km² of this land was carved out for reactor testing. The site is large enough to accommodate several widely separated operational hubs, while maintaining a nominal 50 km buffer zone from population centers. INL sits atop the northeast corner of the massive Snake River aquifer, which is capable of supplying cooling water for multiple medium-size power reactors.

Naval proving ground

Prior to its adoption for reactor testing, the site tested large naval guns that were refurbished in Pocatello, Idaho, transported by rail to a central location, and fired at buttes that jut from the desert plain. Several large conventional explosions contributed to testing safe storage of military ordinance. B-17 and B-24 bombers regularly blasted two small areas of the site prior to the end of World War II. There was little doubt that this rugged landscape could support a few test reactors.

No permanent settlement

There was no record of long-term human occupancy on the reactor test site. Subsequent cultural resource investigations revealed a rich history of hunting, gathering, crisscrossing travelers including Oregon Trail migrants and fur traders, Native American encampments, and failed homestead attempts ([DOE/ID, 2011](#)).

Strong community support

Supportive of the atomic dream like the rest of the country that in that era, the surrounding communities enthusiastically backed the test station and competed for the local AEC headquarters.

Bellwether achievements

Having spent 1948 drawing up plans, the AEC wasted no time in starting construction of key facilities in Idaho. By 1951, the first Idaho reactor, Experimental Breeder Reactor No. 1 (EBR-I), commenced operation. Without supercomputers, established supply chains, or modern construction methods, 17 reactors were operating on the site within 10 years. With national expertise in short supply, INL's first four reactors were designed at Argonne National Laboratory near Chicago, Illinois.

First nuclear electricity

Illuminating four light bulbs, EBR-I produced the first nuclear-generated electricity on December 20, 1951. It also demonstrated that breeding—the production of more fuel than is consumed—was attainable in a reactor with a high-energy neutron spectrum. Today EBR-I is an historical landmark open to visitors.

First materials testing reactor

Much larger and more complex than EBR-I, The Materials Test Reactor (MTR) began operating in 1952. Besides demonstrating a new high radiation flux technology, the MTR accelerated the screening of potential materials for use in promising reactor technologies. It was also the first building block of INL's sustaining research mission.

First large-scale application of nuclear energy

The Submarine Thermal Reactor (STR), also known as S1W, started up in 1953. This prototype reactor demonstrated the viability of the pressurized water reactor (PWR) technology to generate the mechanical power needed to turn a large propeller shaft. The U.S. Navy uses this technology in its nuclear ships. Two-thirds of the world's commercial nuclear powerplants employ PWR technology.

First town lighted by nuclear energy

The first boiling water reactor experiment (BORAX-I) also reached power in 1953. The five BORAX tests ran through 1958, demonstrating the viability of the boiling water reactor (BWR) concept. Today BWRs comprise 20% of the world's commercial nuclear powerplants. BORAX-III scored a notable first in 1955 when it supplied all the electricity to the town of Arco, Idaho, and several of the INL facilities.

Solid foundation for pursuit of commercial nuclear power

The early results from these four INL reactors put the AEC in a position to encourage private enterprise and international cooperation in future nuclear energy development ([American Nuclear Society, 1992](#)). President Eisenhower signed the Atomic Energy Act of 1954, the enabling legislation for cost sharing between government and industry in the Power Demonstration Reactor Program. Demonstration reactors began connecting the U.S. power grid in 1957.

Nuclear energy for the armed services

Drawing on the success of the Manhattan Project, all branches of the US military wanted to capitalize on nuclear energy to help achieve operational superiority. The Navy, the Air Force and the Army initiated projects to pursue their objectives. However, only the Navy had a mission for which there was no alternative to nuclear. The ambitions of the Army and the Air Force would be overtaken by competing technologies, but not before they made a significant footprint in Idaho.

Marine propulsion

The most successful reactor program with a footprint in Idaho is also the most difficult to document because of its military origin. The United States Navy, in conjunction with the AEC, initiated a project in 1949 to develop a nuclear-powered submarine under the leadership of Captain Hyman G. Rickover. ([U.S. DOE and U.S. Department of the Navy, 2014](#)). The fast-paced project bypassed established engineering development processes and set up simultaneous activities in New York, Pennsylvania, Connecticut, Illinois, and Idaho. Idaho's role was to construct and operate a prototype powerplant for the USS NAUTILUS, a submarine under construction in Groton, Connecticut. Westinghouse and the Bettis Atomic Power Laboratory were responsible for the engineering design, with core design support from Argonne National Laboratory. General Electric and the Knolls Atomic Power Laboratory were responsible for a competing sodium-cooled reactor design which was to power the USS SEAWOLF.

Naval prototype reactors

Nuclear power enabled the first true submarines by allowing ships to drive submerged at high speed for long periods without the need to resurface to obtain oxygen. Early submarines were essentially surface vessels that could submerge for short periods before resurfacing to run diesel engines to recharge batteries and oxygen supplies. Nuclear power also enhanced the capabilities of large aircraft carriers. Since 1955 the U.S. Navy has built more than 225 nuclear-powered ships, primarily submarines.

Submarine thermal reactor prototype

The Submarine Thermal Reactor, also known as S1W (submarine, first prototype, Westinghouse), was constructed at the new Naval Reactors Facility (NRF) on the INL site. STR was an exact prototype of the powerplant used in the USS NAUTILUS. Two precisely sized hull sections were used to house the reactor and the power equipment. The hulls were placed into a large water tank in order to replicate the radiation shielding conditions that would be necessary for the submarine. Following startup in early 1953, S1W simulated a 96-h submerged test run across the North Atlantic that eliminated any residual doubt about the feasibility of nuclear propulsion.

Aircraft carrier reactor prototype

Similar to the submarine program, the aircraft carrier first prototype A1W (aircraft carrier, first prototype, Westinghouse), simulated the essential features of a nuclear power system aboard a large aircraft carrier. Because of the size of these ships, multiple reactors were required. For the Idaho test, two reactors were built inside a large building at NRF. The novel feature was the use of both reactors to turn a single shaft. The arrangement simulated one engine room of the USS ENTERPRISE, which was to have a total of eight reactors. A1W achieved full power in 1959 and the ENTERPRISE went to sea in 1961.

Fifth generation submarine reactor prototype

NRF constructed a third prototype reactor in 1965. Known as the Natural Circulation Reactor, or S5G (submarine, fifth prototype, General Electric). S5G was built in a submarine hull section in a water tank. The hull could be rocked to simulate more realistic ocean conditions. The purpose of the new design was noise reduction by using natural circulation rather than pumps to circulate the cooling water for all but high-speed operation. This prototype supported the USS NARWAHL, which launched in 1967.

The Navy's continuing mission in Idaho

The naval reactor prototypes also found purpose in training navy personnel and testing new designs and equipment. The training mission phased out with the end of the cold war in the 1990s. Nevertheless, the Naval Reactors Program continued to have a major presence in Idaho as the primary experimenter in the materials testing reactors and supporting facilities. NRF continues to operate large nuclear facilities in support of the Navy's missions.

Dominance of the pressurized water reactor

Although the sodium cooled SEAWOLF performed satisfactorily, pressurized water technology was chosen for future US nuclear ships. These ships have compiled an outstanding performance and safety record. In the mid-1950s, the AEC began to transfer the technology to the private sector under its Power Demonstration Reactor Program. The AEC selected five companies for the construction of the first PWR plant, with the overall responsibility for the project assigned to Naval Reactors Branch, Division of Reactor Development ([American Nuclear Society, 1992](#)). The Shippingport, Pennsylvania plant began to deliver power to Duquesne Light Company customers in 1957. Operated by the utility, the plant reached its full power of 60 MW-electric on December 23, 1958. As of 2019, 297 PWRs were operating worldwide, with another 51 having shut down at the end of economic life ([Nuclear News, 2019](#)).

Failure to transfer

The enthusiasm for U.S. Navy nuclear ships did not transfer to merchant ships. Idaho conducted a brief program from 1962 to 1964 for a high temperature, air-cooled marine propulsion reactor. This low-power critical experiment, called 630-A, was discontinued when the nuclear power merchant ship program lost priority due to a projected lack of economic competitiveness.

Fantasy of flight

The US Air Force envisioned that a nuclear-powered aircraft could stay aloft without a need to refuel. The first priority would be to fly reconnaissance missions around Russia, each mission lasting multiple days. Take-off and landing would be accompanied by escorts prepared to deal with a crash. The Aircraft Nuclear Propulsion (ANP) program included activities across the US, with General Electric (GE) taking the design lead. The ANP program spawned a flurry of activity and investment at INL, which came to a crashing halt in March 1961 when the Kennedy Administration abruptly cancelled it. With the advent of satellites and intercontinental ballistic missiles, the rationale for nuclear-powered flight waned.

One global runway

The Air Force briefly considered Idaho for the operations of the ANP airplane. A 7 km runway was envisioned and a giant hanger capable of handling the radioactive plane and its crew was actually constructed. Today that hanger has been repurposed for classified manufacturing for the U.S. Army. The program established the facility infrastructure at Test Area North, which included the capability to tow highly radioactive experiments by a shielded locomotive to a massive hot cell examination facility, which would have important later application.

Coupling a nuclear reactor with turbojet engines

GE conducted three high temperature reactor experiments (HTRE 1–3) at INL. The first experiment in late 1955, called the Initial Engine Test, proved that a reactor's heat could run a modified J-47 engine. More work was needed to develop fuels and materials for the high-temperature operating regime, but the test specimens were too large for MTR. GE modified the HTRE for irradiating materials in high neutron flux at temperatures of 1500 °C in the HTRE-2 tests. GE built a new HTRE-3 reactor for the next phase, which achieved a major milestone in December 1960 when the reactor started and ran two turbojet engines at 1100 °C.

Plans for managing radiation and contamination

The ANP program addressed challenging radiation issues. If everything went right on a flight mission, the plane would return with highly radioactive nuclear reactor. The shielded locomotive was to move it into the hanger, which had a complex shielded subterranean maze for the crew's egress. SUSIE, the Shield Test Pool Facility, was built at TAN for ANP shielding studies, but served a similar function for later programs. The most unusual experiments were the Fuel Element Burn Tests, better known as the Wiener Roasts, which simulated a crash and a jet-fuel driven fire on the desert floor. The nuclear fuel proved resistant to contamination release.

Powering remote bases

The U.S. Army was also motivated in the 1950s to find nuclear power solutions to operational issues. Given sufficient safety and reliability, small nuclear powerplants would be advantageous for replacing diesel technology that required a steady resupply of diesel fuel in challenging locations. The Army's interest was in small units that could be transported and set up relatively easily, as well as mobile powerplants that could be deployed where needed. With low total weight a priority, the Army pursued BWR technology for the stationary plants and air-cooled technology for mobile units. Idaho's role was one of many activities integrated across the nation.

Army reactor experimental area

Like the Navy and Air Force, the Army got its own INL test site, the Army Reactor Experimental Area, later known simply as ARA. The first test for the mobile units was the Gas Cooled Reactor Experiment (GCRE) designed by Aerojet General Corporation. GCRE went critical in 1959, producing the fuel pin data for the Mobile Low-Power Reactor, ML-1. The first reactor at ARA was the Stationary Low-Power Reactor No. 1, which went critical in 1958. Designed by Argonne National Laboratory and operated by Combustion Engineering, SL-1's mission was to train crews for future reactor operations in the Defense Early Warning Line along the Arctic Circle. Crews were rotated between Idaho and Fort Belvoir, Virginia, where they trained on the prototype Stationary Medium-Power Reactor, SM-1.

Disruptive disaster

SL-1 experienced a lethal accident in early January 1961. The catastrophe destroyed the reactor, contaminated ARA, and forced the ML-1 work to move to Test Area North, home of the cancelled aircraft propulsion program. Results from the ML-1 program were not encouraging, with the reactor not able to sustain power production long enough to reach its performance goals. The following year the Army ended its reactor development program. Apparently undeterred by the SL-1 experience, the Army continued to operate a few small reactor plants in remote locations for several more years. Decades later, there is revived interest in advanced versions of these microreactors to power remote communities and military stations.

Testing the limits of safety

Safety was the compelling argument for locating reactor testing in remote Idaho. The case had two parts—assuring public protection against unexpected radiological incidents and gauging the safety limits of the reactor concepts themselves. Both the reactor designers and the nuclear regulators needed hard data on the practical physical limits of reactor designs, with major emphasis on fuel and thermal properties.

Reactors designed for testing accident scenarios

In simplest terms, reactor safety experiments focus on runaway reactions, such as happened at Chernobyl in 1986, and heat removal once the cooling system is inoperable, such as the case with the Fukushima Daiichi powerplants in 2011. Transient test reactors examine the effects on nuclear fuel when subjected to sudden overpower events. The majority of INL's safety test reactors were of this type. Following the 1979 Three Mile Island nuclear accident in Pennsylvania, emphasis shifted to a reactor facility constructed to investigate loss of coolant flow following reactor shutdown. The materials testing reactors also contributed to the safety database by putting candidate fuels through less severe operational transients of temperature and power. Collectively these tests provided indispensable knowledge for reactor design and licensing.

Transient test reactors

In the earliest transient test reactors, the reactor itself was the experiment. A fairly simple reactor design with the desired characteristics would be subjected to a rapid power excursion, after which the damage would be assessed and analyzed. The Boiling Water Reactor Experiments (BORAX) and the Special Power Excursion Reactor Tests (SPERT) I-III were in this category. Later, more versatile transient test reactors were built to test different types of reactor fuels under varying conditions. The reactors consisted of a driver core and a central region for the targeted fuel. Very rapid power transients could destroy the test articles but had little or no effect on the reactor. The Transient Reactor Test Facility (TREAT), SPERT-IV, and the Power Burst Facility (PBF) comprised this category.

BORAX

No reactor program at INL had a more colorful history than the Boiling Water Reactor Experiments. INL was the only place where scientists could deliberately cause a reactor to blow up to test their predictions. In the early 1950s there was technical disagreement about the potential effect of steam bubbles on a reactor's controllability. Some scientists thought the voids in the core would cause erratic behavior, while others thought that boiling might actually make the core self-regulating, thus preventing accidental overheating. The BORAX experiments tested this theory from 1953 through 1958. BORAX-I and II tested the basic BWR concept using available test reactor fuel. The larger BORAX-III experiment demonstrated the first useful production of nuclear electricity in 1955. At 20 and 40 MW, respectively, the BORAX-IV and V reactors tested high-temperature fuels and higher quality steam for commercial application. DRESDEN-1, the first commercial BWR, started operation in 1960 in Morris, Illinois.

BORAX-I&II

BORAX-I was a simple, open pool-type reactor with aluminum fuel and controls located in a remote trailer. Samuel Untermyer was the main proponent of the boiling water reactor concept, with essential backing and hands-on participation from Argonne Laboratory Director Walter Zinn. From the control trailer, Untermyer could pull a control rod from the core, sending the reactor on a power excursion, spewing steam and water high enough to be visible from the public highway that runs through the site. After more than 200 successful tests that demonstrated self-regulation of reactor power, Untermyer designed the final test to demonstrate

that a BWR could be pushed too far. The test, which blew the reactor apart in a steam explosion, was the first to quantify the impacts of an accident and remove some speculation about the resultant nuclear hazards. BORAX-I became a video sensation (“BORAX” [youtube.com](https://www.youtube.com/watch?v=Ugk1vYUg8j0).) Cleanup of the destroyed reactor was unceremoniously accomplished, and BORAX-II was constructed a short distance away, only to be similarly destroyed in a violent explosion at the end of its useful life. Convinced by the BORAX-I experiments that aggressive safety testing under controlled conditions could provide vital information without disaster, the AEC approved new projects for other reactor concepts, leading to a rapid proliferation of reactors on the desert.

BORAX-III

BORAX-III provided electricity to the city of Arco, Idaho, on July 17, 1955—a nuclear first. The accomplishment would appear simple enough, given a stream-producing 15 MW nuclear reactor and the electrical infrastructure of the INL. However, the project was under intense pressure to be completed prior to the first International Conference on Atomic Energy scheduled for August. Finding and assembling a compatible turbine generator on short notice proved challenging, as did connecting to utility transmission lines. (Haroldson, 2008) Success came at midnight, unknown to most residents, but captured on film by the AEC. The citizens of Arco were invited to view the film on August 12, 1955—the same day that it was shown to the conference delegates in Geneva, Switzerland.

SPERT

Having demonstrated the stability and safety characteristics of boiling water reactors, Argonne transitioned to technical support for their commercialization. The Phillips Petroleum Company took over transient testing for water cooled and moderated reactors, building the Special Power Reactor Excursion Test (SPERT) facility near the Central Facilities area of the site. Thousands of SPERT experiments, conducted between 1964 and 1970, were increasingly more sophisticated than the proof-of-principle BORAX experiments. While each of the four SPERT facilities contributed to power reactor development, they also advanced instrumentation needed to extract data from challenging test environments. SPERT-I tested the damage resistance of low-enriched uranium oxide fuels similar to the type that would be used in commercial powerplants. SPERT-II conducted transient experiments in heavy water systems, providing essential data for reactor development in Canada and Europe. SPERT-III enabled testing under a wide range of temperature, pressure, and flow conditions. SPERT-IV introduced a driver reactor core with central test region, setting the stage for the next generation of transient test facilities.

PBF

The Power Burst Facility (PBF) took over the water reactor safety test program from SPERT in 1972, operating until 1985. It was built during the time at which commercial LWR designs were being scaled by an order of magnitude larger than the AEC-supported early demonstration plants. Safety analysts conjured a new suite of potential accidents for the large powerplants. Experimental data were needed to set safety limits and license these new plants. Sufficiently isolated southeast of the site’s Test Reactor Area, PBF was a large, powerful, versatile test reactor. Capable of large bursts of power within milliseconds, it could simulate a wide range of transients in a central test fuel capsule. But it could also test slower events, such as loss of coolant. Destroyed test fuel was moved to INL hot cells for examination. Data from PBF was a boon for safety code developers. Following the Three Mile Island meltdown in 1979, a PBF simulation accurately predicted the state of the melted core.

TREAT

The first facility constructed at INL’s easternmost site, now called the Materials and Fuels Complex (MFC), the Transient Reactor Test Facility (TREAT) began operation in 1959. Unlike BORAX, SPERT, and PBF, which tested the transient behavior of water-moderated reactors, TREAT was primarily intended for fast reactor fuel research. The reactor is an air-cooled graphite cube, lightly salted with highly enriched uranium, a central test region capable of accommodating sodium or water-cooled test loops as well as simple capsules, and hodoscope slot to allow real-time observation of fuel behavior during a transient test. Capable of a wide variety of transient shapes and operating modes from steady-state to one-second pulses of 20 gigawatts, TREAT conducted hundreds of experiments that led to improved fuel performance and enhanced development of severe accident simulation codes. TREAT claims two bits of history that are unique on the INL site: some of the graphite for its reflector came from CP-1 (Holl, 1997), and after being mothballed since 1994, the facility resumed operation in 2017.

Slow accident testing

While the transient test reactors were capable of simulating some scenarios involving loss of coolant, regulatory interest completely refocused on slow-evolving events following the severe accident at the Three Mile Island powerplant in 1979. Scenarios that took hours or even days to evolve required tests with whole-core involvement, something that could not be fully achieved in the modern transient reactors. Concern about the consequences of a cooling pipe break in an LWR plant began in the mid-1960s as the plant designs began to exceed 1000 MW. While testing in SPERT showed that the fission reaction would automatically shut down due to physics, the worry about a loss-of-coolant (LOCA) accident was that the fuel would fail due to continued heating by decay of the massive fission product inventory in the core. The term “China Syndrome” was coined to describe a scenario in which the molten reactor core bore through the containment into the soil beneath the vessel. A secondary system would be needed to remove the decay heat and ensure adequate safety. In 1963 the AEC initiated planning to test LOCA events in a new facility called the Loss-

of-Fluid-Test (LOFT) reactor in Idaho, a key addition to the Commission's capabilities for its Safety Test Engineering Program (STEP).

STEP stumbles

Initial plans for the LOFT facility followed the typical Idaho script, except that the components would be off-the-shelf, not custom fabricated in INL shops. A small version of a PWR would be constructed, with initial testing of all the non-nuclear components, followed by a destructive event that could be analyzed in detail. Two developments sidetracked these plans. First, the LOFT design incorporated stainless-steel clad fuel rods, just as the industry was changing to a zirconium alloy, which potentially would affect an accident involving high-temperature steam. But the major impact came from a change of direction in the AEC's reactor development program. The AEC shifted emphasis to assuring the highest levels of quality, rigid procedures, transferring responsibility for reactor development to the private sector at the earliest opportunity, and designing redundant engineered safety systems to prevent major accidents. The concept of a dramatic, but unrepeatable destructive experiment did not fit this strategy, causing the LOFT program to drift. STEP progress did continue using an electrically heated facility called Semiscale which was ideal for providing data for codes that analyzed pipe breaks. Unexpectedly, in 1971 Semiscale demonstrated that under some conditions steam pressure would prevent emergency cooling water from reaching the core, raising troublesome issues for licensing large LWR plants. More federally funded safety research was needed after all.

LOFT

LOFT finally got on track in 1974 with a new testing strategy when all safety research was put under a new Division of Reactor Safety Research in the AEC. The facility was redesigned to provide multiple LOCA tests that would inform safety and accident codes. Located at Test Area North, the shielded locomotive was pressed into service to move the reactor between its containment dome and the massive TAN Hot Shop for examination. LOFT's first experiment was a worst-case event—a double-ended guillotine pipe break in December 1978. The emergency core cooling system performed better than predicted, limiting the fuel temperature rise to about 540 °C. A variety of additional tests were planned to provide data for the continuously improving computer models.

The TMI-Idaho connection

Three months after the initial LOFT test, the core meltdown at the Three Mile Island 2 (TMI) caused an abrupt change of plans. The testing program for LOFT and Semiscale shifted from large LOCAs to examination of small leaks and operational transients. PBF results had quickly determined the likely state of the TMI fuel; LOFT focused on the cooling water. Within 2 years, LOFT had successfully simulated the TMI accident, verified the associated safety codes, and attracted international attention to INL's unique safety testing capability. The Organization of Economic Cooperation and Development contributed technically and financially to the program from 1983 to 1986. LOFT testing concluded in 1985 with the deliberate melting of its test fuel bundle, almost as originally planned nearly two decades earlier. The TMI fuel rubble was eventually transported to Idaho for further study and interim storage.

Developing fast reactor technology

From the earliest days of the AEC, nuclear energy leaders anticipated that fast breeder reactors would be deployed as soon as possible, eventually dominating the commercial nuclear market. This assumption was grounded in the belief that uranium was scarce, and the knowledge that thermal reactors could only consume less than 1% of mined uranium ore. On the other hand, breeders could extend the resource by two orders of magnitude. EBR-I was the initial fast reactor, constructed at its namesake site. As development plans matured, all fast reactor work transferred to a more suitable site, now known as the Materials and Fuels Complex (MFC). By 1969 the move was complete.

An integrated research and development complex

The Materials and Fuels Complex had all the facilities, materials, and infrastructure needed for fast breeder reactor development, with capacity to support other nuclear programs. The Transient Reactor Test facility kicked off the operations in 1959. The Experimental Breeder Reactor-II (EBR-II), a 62.5 MW powerplant, supplied electricity to the INL and cogenerated steam heat to the MFC site, while performing fast neutron irradiation services for a variety of programs. The Fuel Cycle Facility recycled used metal alloy EBR-II fuel. The Fuel Manufacturing Facility fabricated fresh driver and experimental fuel. The Analytical Chemistry Laboratory performed detailed analysis of test samples. The Zero Power Physics Reactor (ZPPR) facility assembled 21 reactors varying from small space systems to huge breeder reactors, supplying direct design physics data as well as benchmark data for the International Reactor Physics Handbook Database ("IRPhE Project" [OECD-NEA.org/science](https://www.oecd-nea.org/science)). Starting in 1975, the last major facility, the Hot Fuel Examination Facility (HFEF), supported reactor programs from across INL. The Neutron Radiography Facility (NRAD), a unique feature of HFEF was built in the basement around a TRIGA reactor, which in 1978 was the last reactor added to the INL fleet.

EBR-II: A versatile demonstration platform

EBR-II was to be the first major step toward commercialization of sodium-cooled reactors. Construction started in 1958 with an initial objective to operate reliably using recycled fuel to drive a superheated steam cycle to produce about 20 MW of electricity. Most systems and components were first of a kind, as were the maintenance and operating procedures. EBR-II's unique design enclosed all primary heat transport systems, two centrifugal pumps, a heat exchanger, and a fuel storage basket in a sodium-filled vessel. An electromagnetic sodium pump was used for the secondary sodium loop that connected the in-pool heat exchanger and the steam generator (Koch, 2008). Fuel was handled remotely under sodium within the vessel. Approximately 35,000 fuel pins were melt-refined, recast, and remotely fabricated in FCF for reuse in EBR-II (Stevenson, 1987). From 1963 to 1965 EBR-II gradually increased power to its thermal design rating of 62.5 MW, making modifications as needed. Fuels and materials irradiation testing, in place from the outset, became EBR-II's primary mission in 1969 after the AEC had decided on an accelerated path to rapid fast reactor commercialization. The next test reactor, the Fast Flux Test Facility (FFTF), would be built on the Hanford site in Washington, to be closely followed by the Clinch River Breeder Reactor (CRBR), a 350 MW-electric utility-AEC cost-shared prototype in Tennessee. The AEC directed EBR-II and the other MFC facilities to support those two projects.

Opportunity from chaos

Electric Utilities agreed to fund \$257 million of the \$400 million estimated cost of the CRBR project. The project started in 1972 without the technical support expected from FFTF, which was finally completed in 1978 after extensive delays and cost overruns. The extended schedule delay for FFTF operation gave EBR-II new opportunities as a flexible technology demonstration platform. Advanced fuels, cladding and structural materials were irradiated in EBR-II. From 1975 to 1986 the Run Beyond Cladding Breach (RBCB) program demonstrated that due to innovative technology to remove contamination from the coolant and inert cover gas, sodium-cooled reactors could continue to operate with minor cladding failures. The results of the RBCB program also dispelled the widely held belief that significant cladding breaches would lead to flow blockage or damage propagation. Operational experience demonstrated that the EBR-II design with metal fuel was inherently self-regulating. Many of EBR-II's innovations were picked up by designers in other countries, even though the U.S. decided on a different design approach. Loss of utility interest and an estimated completion cost exceeding \$3 billion contributed to Congress' cancellation of the CRBR project in October 1983. Though jarring to the nuclear community, the sudden cancellation created an opportunity for adapting breakthrough technologies.

The integral fast reactor

Argonne's Associate Laboratory Director, Charles Till, assembled his most gifted leaders to put together a fast reactor program proposal that would address all of the major public concerns about fast reactor technology. Less than a year after CRBR's demise, Till introduced the Integral Fast Reactor Program (IFR) as a complete technical solution (Till and Chang, 2011). Inherent safety was addressed by the proper choice of fuel, materials, and design. Proliferation resistance was inherent in an on-site electrorefining recycling scheme that would not separate refined plutonium. Waste would be compact, with the long-lived constituents remaining in the fuel cycle. The scientific underpinning was already in place through the EBR-II experience. The highlight of the IFR program took place in 1986 when the EBR-II staff executed two landmark safety experiments witnessed by a collection of distinguished international, national laboratory, and industry observers. With safety systems disabled, two severe accident precursor events were initiated. In both cases, the reactor temperature rose slightly and then stabilized. Either event would have destroyed a conventional reactor, but there was no damage to the EBR-II fuel or components. These landmark tests were overshadowed by the Chernobyl disaster 3 weeks later. The IFR program was cancelled in 1994, but not before planting the conviction that advanced reactors could be designed to be impervious to accidents under almost all conditions.

Unplanned accidents

The brilliant minds of the 1940s knew that they didn't have all the answers. The best protection for the public while the AEC pursued those answers was to locate experiments for the new, the untried, and the unestablished limits of safety well away from population centers. The Idaho site was the location of two notable unplanned accidents.

SL-1

In early January 1961 the SL-1 reactor explosively disassembled, killing all three technicians on late-night duty. The resulting contamination spread over several kilometers. The cause of the accident was extensively studied. Design shortcomings, procedural violations, and inadequate maintenance may have each played a role. This was the only deadly accident in U.S. nuclear powerplant history.

EBR-I

The EBR-I highly enriched uranium core has been described as about the size of a football. During a transient experiment in November 1955, about one-half of the core melted. As horrific as a core meltdown may sound today, it was a manageable incident. The damaged core was packaged, sent to Illinois for examination, and replaced with a new one.

Fuels and materials development

No INL reactor activity has been more sustaining, varied, and impactful, nor required more ingenious solutions, than fuels and materials development. It is actually the only reactor activity that has continued unabated for almost 70 years. While this is an enterprise broadly distributed across industry and research laboratories, the irradiation and examination of samples are centered in Idaho. Through clever design, this class of test reactors can substantially accelerate aging in materials and fuel samples.

Idaho's irradiation facilities

The Materials Test Reactor, one of the original big four, operated from 1952 to 1970. As with other first reactors, it was its own experiment, demonstrating plate-type fuel for future test reactors. It also demonstrated plutonium-based driver fuel at a power level up to 30 MW. The 175 MW Engineering Test Reactor was a more powerful and versatile experimental reactor that operated between 1957 and 1982. A sodium cooling and a helium cooling system were added in order to accommodate tests for a broader array of concept fuels. The 250 MW Advanced Test Reactor (ATR) began operating in 1967 and is expected to continue as long as needed. In addition to its fuel testing services, ATR produces isotopes for medical and industrial applications. From 1965 through 1994 EBR-II served as fast-spectrum fuels irradiation facility, servicing a variety of customers including the fusion program.

Notable fuel development

Given the tens of thousands of experiments in INL's materials testing reactors, it is impractical to catalogue all the fuel concepts that have passed through. Idaho's unique capabilities for examining irradiated material, in whatever detail necessary, provides the data that are actually applied in production and licensing. Four accomplishments illustrate the impact. Naval Reactors Program is the main customer of ATR; it has developed fuel that lasts for the entire 30-year life of a submarine. The IFR ternary uranium-plutonium-zirconium metallic fuel is the reference for whatever U.S. fast reactors are developed in the future. Several variants of Fukushima-inspired accident-tolerant LWR fuel under industrial development are being tested in Idaho. A reliable tri-isotopic (TRISO) high-temperature fuel has been qualified for use in helium-cooled, water-cooled, or molten-salt reactor applications.

Powering space exploration

Since its inception in 1958, the National Aeronautics and Space Administration (NASA) has been intrigued about the possible mission-enhancing benefits of nuclear power. Numerous universities, NASA research centers, and DOE national laboratories including INL have been engaged. Three types of nuclear systems have been investigated at various times: nuclear rockets for human exploration of the solar system; tiny reactors for power-intensive satellites; and long-lasting, reliable radioisotope thermoelectric generators (RTG). INL played a significant, but not dominant role in each of these technology programs.

Nuclear thermal propulsion

From 1967 to 1972 INL conducted cavity reactor critical experiments at the Test Area North to determine whether a gaseous uranium core surrounded by flowing hydrogen could achieve criticality. Simulants were used in the low-power, low-temperature experiments. Temperatures in this ambitious concept's gaseous core were expected to reach as high as 50,000 °C. During the following decades INL had little direct involvement with NASA's nuclear propulsion concepts until fuel testing in TREAT began in 2019. The current propulsion concept expects the engine to run several times for a few seconds at 3000 °C, parameters that can be readily achieved in a TREAT transient experiment.

Reactor-powered satellites

Russia and the U.S. both had an infatuation with reactor-powered satellites. Russia launched 31 of them between 1967 and 1988. The U.S. launched a single reactor, SNAP-10A, which operated for about a month. NASA's major safety concern was having the highly enriched uranium system crash into the ocean, possibly causing a runaway reaction and substantial radiation release. During the mid-1960s, INL destroyed three SNAP-10 reactors at Test Area North in order to characterize the radiation release from accidents. During the mid-late 1980s, NASA pursued development of a 100 kW-electric space power system called SP-100 that could be used for nuclear-electric propulsion or stationary power. Critical experiments to support the basic concept and the ground test facility

were conducted in ZPPR in 1986 and 1988–89. The tests included simulating accident scenarios such as water immersion and being imbedded in sand. Development of SP-100 was terminated due to the estimated high cost to completion.

Radioisotope thermoelectric generators

Radioisotope thermoelectric generators, better known as RTGs, have been an integral part of NASA missions for almost six decades. The NASA RTGs generate heat by the radioactive decay of plutonium 238. In the early 2000s, the Department of Energy terminated RTG assembly, testing and transportation at Mound, Ohio, and reestablished the capability at the Materials and Fuels Complex in Idaho. INL subsequently provided the RTG for the Pluto New Horizons mission and the Mars Science Laboratory Curiosity rover. In 2019 INL began preparing the RTG for NASA's Mars 2020 mission.

The arc of reactor history in Idaho

Understanding the rise, fall, and possible rebound of Idaho's civilian reactor programs requires putting the events into a national context. The Naval Reactors Program, with its own imperatives and stable funding, had an independent, illustrious trajectory.

Naval reactors program

Although public information about naval reactors is limited, the program has maintained a visible INL presence from the outset. Major achievements include the spinoff of PWR technology to private industry, providing thousands of well-trained operators to nuclear utilities, and developing submarine reactors that don't require refueling during the ship's lifetime. In order to accomplish the later, the naval reactors program dominated fuels and materials testing in ATR.

Rapid growth of test reactors

In the 1950s Americans were fascinated with the promise of atomic power just as electricity demand accelerated in the post-World War II era. In Congress the Joint Committee on Atomic Energy streamlined financial support for nuclear development. The race for nuclear reactor commercialization and technology applications was on; reactors mushroomed in the Idaho desert. By 1960 twenty-three reactors were operating in Idaho, with five more already shuttered. Most of these reactors were built quickly and inexpensively, not intended for long-term use. Technology shakeout favored PWR and BWR development for the near term.

Shift to commercialization

From 1960 through the early 1970s, the AEC focus shifted to the Power Demonstration Reactor program. The number of INL's operating reactors dropped to 17, despite the addition of 15 new units that supported physics, safety, and space research. Culled projects included the beryllium-oxide moderated reactor and the organic-cooled/moderated concepts, while demise of the aircraft propulsion program and the army program terminated several INL reactors. INL's two most venerable reactors, EBR-II and ATR, started operating during this era.

Rise and fall

Significant reactor program activity from the 1974 through 1986 belied the future for INL. Major political changes affected the mission and funding of the laboratory. The Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC) replaced the AEC. The Joint Committee on Nuclear Energy, a stalwart bi-partisan supporter of nuclear development, was abolished in 1977, the same year that the Department of Energy replaced ERDA. With broader energy, science and weapons research portfolios, neither ERDA nor DOE built new test reactors. In 1975 cancellations exceeded new orders for commercial nuclear plants. The last new order in the 20th century was in 1978. With safety experiments winding down by 1986, INL's reactor program future was fading.

Loss of mission

The next 10 years brought an end to INL's civilian reactor development. Anti-nuclear sentiment in the Clinton Administration prevailed. IFR development was terminated in 1994. By 1998, the already shrinking national nuclear research and development budget was eliminated. The Navy no longer needed to train sailors in Idaho. INL's new mission was to decommission facilities and prepare radioactive waste for disposal. Only ATR, with its naval research mission, and two small support reactors remained in operation by 1996. INL's definitive 50-year history ([Stacy, 2000](#)), published at the end of the 20th century, could be read as a 375-page obituary for Idaho's reactors. But with more than 100 commercial reactors operating across the U.S., some national laboratory capabilities were still needed, including regulatory support. National laboratories and universities rallied to put together a reactor research plan for the 21st century.

Mission restored

With his nuclear budget vanishing, William Magwood, IV, then-Director of DOE's Office of Nuclear Energy, crafted a long-range advanced nuclear development plan with his counterparts in France and Japan. The Administration and Congress backed the idea of a multi-national effort to develop 4th generation reactors, possibly leapfrogging the third generation which had yet to be built. The Generation 4 International Forum held its inaugural meeting in January 2000. National laboratories, corporations, and universities set to work on advanced reactors that would meet a robust set of inherent safety, economic, sustainability and nonproliferation goals. Reminiscent of the 1950s, feasibility studies were launched for sodium, lead, helium, molten salt and supercritical water-cooled systems. Magwood then set about consolidating R&D contracts in Idaho, creating a new national laboratory with the Office of Nuclear Energy as its principal sponsor, leaving the decommissioning activities to a separate contractor and DOE office. The Idaho National Laboratory officially began operating in 2005 under the Battelle Energy Alliance.

Stability and purpose

During the next decade, INL matured into its role as a multi-program national laboratory, still focused on nuclear energy, but with significant expertise and funding in cyber security, integrated energy systems, battery technology, supercomputer applications, and biofuels science. Fuels and materials research evolved sophisticated computational capability on the macro, *meso* and micro levels, supported by characterization of irradiated samples at similar levels of detail using new instruments and facilities. Cooperation and coordination became the new norm in nuclear research as public-private partnerships supplanted most government-only R&D funding. More than 40 companies began pursuing advanced nuclear systems. With ATR as its centerpiece, INL coordinates the National Scientific Users Facility (NSUF) to conduct hundreds of research projects for qualified applicants in 50 partner facilities across the U.S. INL also partnered with Argonne and Oak Ridge National Laboratories to create DOE's Gateway for Accelerated Innovation in Nuclear (GAIN), whose aim is a radical reduction in development time for nuclear technology.

Back to the future

As INL celebrated 70 years of landmark nuclear research, the laboratory planned for impactful reactor facilities once again to rise in the Idaho desert. Transient testing resumed at TREAT in 2017. NuScale Power and Utah Associated Municipal Power Systems were moving forward with plans to build a powerplant consisting of 12 small modular reactors on the site in the mid-2020s. One of the modules is planned to be set aside for the Joint Use Power Module project, which will enable research on more efficient ways to utilize a reactor's thermal energy. DOE announced plans to build a Versatile Test Reactor in Idaho or Tennessee for fast spectrum irradiations. INL's 70th year celebration event included DOE's announcement of the creation of the Nuclear Research Innovation Center (NRIC), which could host reactor demonstrations within a few years. NRIC demonstrations will be funded in whole or in part by the private sector. NRIC could be the National Reactor Testing Station reincarnate, with a fresh focus on economic competitiveness and rapid deployment of advanced technology.

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Historical Survey of Fast Test Reactors

George R. Imel, Idaho State University, Pocatello, ID, United States

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Glossary

BN, BOR, BR Russian designations of fast systems, as in BN 350.
CRBR Clinch River Breeder Reactor.
DFR Dounreay Fast Reactor.
DOE Department of Energy (US).
EBR Experimental Breeder Reactor as in EBR I.
FBR Fast Breeder Reactor.
FCF Fuel Cycle Facility.
FFTF Fast Flux Test Facility.
FSU Former Soviet Union.
GenIV Generation IV, advanced reactor concepts.
HM Heavy Metal; usually uranium and plutonium.
IAEA International Atomic Energy Agency.
IFR Integral Fast Reactor.
IHX Intermediate Heat Exchanger.
INL Idaho National Laboratory.
LWR Light Water Reactor.
MBIR Russian designation of fast test reactor for fuels and materials.
MOX Mixed Oxide (fuel).

MWth Megawatt thermal.
MWe Megawatt electrical.
NEI Nuclear Engineering International.
PFR Prototype Fast Reactor at Dounreay.
SEFOR Southwest Experimental Fast Oxide Reactor.
SRE Sodium Reactor Experiment.
UK United Kingdom.
w/o Weight percent.

Introduction

This survey will briefly address the conceptualization world-wide of the fast neutron spectrum reactor (fast reactor). Liquid metal fast reactors are the most common system, and typically can be separated according to coolant constituent (sodium, sodium-potassium eutectic, lead or lead-bismuth eutectics, etc.), heat removal systems (pool vs. loop), fuel type (oxide vs. metal), and fuel composition (plutonium, uranium, MOX, etc.). As the early reactors were introduced, most every combination of the above have been tested. As the survey is presented, these important characteristics will be highlighted where possible.

The promise

The early promise of the fast breeder reactor was one of virtually unlimited fuel through breeding of fissile fuel in fertile blankets composed of depleted uranium or thorium ([Shusterman, 2020](#)). The ability to achieve a breeding ratio (ratio of fissile fuel produced by neutron capture in fertile fuel to fissile fuel consumed) of greater than one was demonstrated quite early by some of the first reactors like EBR II and Phenix. The EBR I reactor was fueled with enriched uranium and was able to obtain a ratio of just about one.

Overview of design differences

Coolant

The most common coolants have been various liquid metals, such as sodium or lead. They were chosen for their low moderation power and usually very high thermal conductivities. For example, the thermal conductivity of sodium is roughly 120 times that of water at 200 °C, and heat removal is not as sensitive to turbulence in the coolant. That has given liquid metal reactors some safety advantages in a transient situation. A partial list of liquid metal coolants is: sodium-potassium (EBR I), sodium (EBR II), lead or lead-bismuth (Soviet nuclear submarines of the 1970s) and even mercury (Clementine). There are also some fast reactor designs proposed using helium or molten salts, but these have not had wide-spread favor. In fact, by far the most common coolant choice has been sodium in spite of its chemical reactivity. Because of this, most of the discussion below will consist of sodium cooled reactors, although the Soviet experience with lead-bismuth eutectic will be briefly discussed. See also ([Heidet, 2021](#)).

Loop or pool?

In the loop system, several parallel sodium circuits pump sodium through the core to remove heat. Each circuit goes to a separate intermediate heat exchanger (IHX) which is still inside the reactor building. The secondary loop then goes out of the reactor building to a steam generator building, which then goes to the turbines, etc. for power generation. Thus, for safety reasons sodium reactors consist of three loops ensuring that any possible sodium-water reactor would be out of the reactor building. However, this can escalate the cost and complexity of sodium cooled reactors.

The pool system differs in that the primary pump and IHX are contained within a large tank of sodium. The primary pump forces sodium up through core where it is discharged over the IHX. The IHX then transfers heat from the tank which goes to a secondary loop passing outside the reactor building to a steam generator, etc. EBR II was a pool system; the secondary sodium was pumped by means of an electromagnetic pump while the primary pumps in the tank were centrifugal.

It is debatable which system is better ([Kohler et al., 1990](#)). The tank system has an advantage of inherent heat removal just through conduction in the event of pump failure as might happen in a loss of on-site power accident. A series of such experiments were performed in EBR II in the late 1980s demonstrating inherent safety ([Planchon et al., 1987](#)).

Fuel type

Fuel can be characterized by isotopic composition, including fissile enrichment, and structure (metal vs. oxide or nitride). Composition is usually plutonium or uranium, or a mix (MOX or separate oxide elements). The primary reason that SFRs feature higher HM inventory per unit core volume is to achieve hard spectrum—good neutron economy (high breeding gain). The need to use higher enrichment than in LWR is due to the smaller fission cross section. It is possible to design a fast reactor with less than 20% enrichment (% fissile out of fissile + fertile isotopes), but in the early years enrichments were above 50%. The metal fuel has distinct advantages in its heat transfer capabilities and its easy fabrication relative to oxide; metal fuel can be cast and in fact was the basis for the Integral Fast Reactor (IFR) concept developed at Argonne National Laboratory. Metal fuel was the choice of the early fast reactors such as EBR I, EBR II, and Dounreay Fast Reactor (DFR) in Scotland. However problems with swelling led to the adoption of the oxide fuel type which also has a much higher melting point than metal. Balancing that is the superior heat conductivity of metallic fuel, so one can argue that the safety arguments cancel each other out. Metal fuel also offers the highest possible fissile fuel (uranium, plutonium, etc.) density and enables one to design the core to have a harder neutron spectrum and, therefore, better neutron economy. While the world at that point was concentrating on oxide fuels in the late 1980s and 1990s, new designs demonstrated that the swelling problems could be mitigated by using sodium bonding of the fuel to the clad, and having a large plenum above to accept expansion due to swelling (Till and Chang, 2011; Porter, 2021). Thus the metal fuel was a key component of the IFR.

The early years

Clementine

According to Los Alamos reports, Clementine was the world's first plutonium fast reactor (Bunker, 1983). Construction was begun in 1946 under the direction of Phillip Morrison. Apparently, he named the reactor after the song, "My darling Clementine" which has the line "in a cavern, in a canyon." The reactor site was in one of the Los Alamos canyons, and to Morrison the workers were modern '49ers. Additionally, 49 was code for Plutonium-239 (94 protons, 239 nucleons). The fuel was plutonium rods clad in steel, and the coolant was mercury. The core had a six-inch natural uranium reflector followed by an additional 4 in of steel and 4 in of lead shielding. Reactor control rods were fueled, thus withdrawn to achieve criticality.¹ Although criticality was achieved early, actual construction was not completed until March of 1949 and the reactor was operated at 25 kW. (Note: fast reactors can achieve criticality without any coolant, and startups as dry critical are common.) Several shutdowns were necessary due to fuel problems, and at the last one in 1952 it was discovered that fuel had leaked into the mercury coolant. The decision was made to disassemble the reactor completely at that time. During its short life, 41 total cross-section values were measured at Clementine to an accuracy of 10% and over an energy range of 3 eV to 13 MeV. These data provided valuable information to the Los Alamos weapons designers. Additionally, engineering contributions were made to the fast reactor program in general. For example, mercury turned out to be a bad choice for a coolant because of its poor thermal properties.

EBR I

EBR I first went critical in 1951 at the National Reactor Testing Station, now known as Idaho National Laboratory (INL), in December of that year it demonstrated the first production of useable electricity from nuclear power by lighting four 200 W light bulbs (Westfall, 2004; McFarlane, 2020). To be fair, the first actual electricity was produced at the Graphite Test Reactor at Oak Ridge in 1948, by generating enough steam to run a toy steam engine and light a small Christmas tree bulb, but EBR I was the first reactor to have a complete heat removal and electricity production system. The reactor coolant was a sodium-potassium eutectic (NaK). The core, highly enriched U-235 metal, clad in stainless-steel was very small (football size). The core was surrounded by a natural uranium blanket. EBR I had a total thermal power of 1.4 MW and generated 200 kW of electricity (actually enough to power the site). It operated from 1951 to 1963 and was then replaced by EBR II also at the INL, although at about the mid-point of its operating years an operator error during an experiment led to a core meltdown/destruction. The core was replaced rather quickly and operations continued. EBR I served its mission well demonstrating the possibility of a fast reactor producing electricity. It was able to demonstrate a breeding ratio of about one, which is probably close to the limit of a U-235 reactor surrounded by a simple blanket. Using Pu-239 would significantly improve the breeding ratio, and various mixes of U-238 in the core region would also help.

¹The criticality, or multiplication, of a reactor is controlled by altering the neutron balance. If there are more neutrons in a given generation than the preceding generation, the power will increase. The neutron balance is controlled by sources and losses; we can alter this by changing the losses (leakage or absorbent materials inserted into the core), or changing the source of neutrons by inserting fueled control rods. The latter method is used when neutron economy is important, as in a breeder reactor.

Other early reactors

In the Former Soviet Union, the physicist A.I. Leipunsky recognized the favorable neutron balance of the fast system, and began to study the possibilities (IAEA, 2012). He led the effort to develop the fast reactor there, BR 5 was one of the first constructed in 1959. In contrast to other types around the world, the Soviet design utilized PuO_2 as the fuel type instead of metal. BR 5 (5 MWth loop type) operated from 1959 having had two major upgrades over its lifetime, including replacement of the vessel and components of the primary circuit. It operated with nitride fuel in its later years. The mission had been mostly materials irradiation and instrumentation testing, although there were some cancer treatment experiments and fission product release experiments over its years. Leipunsky also oversaw BOR 60 and BN 350 (their first prototype) which was built on the Caspian Sea in what is now Kazakhstan. The BR 10 reactor was shutdown in 2002.

The growth years

EBR II

EBR II followed the success of EBR I, its predecessor. Initial design was essentially complete in 1956, and an architectural engineering firm, H.K. Ferguson, was chosen (Westfall, 2004). Funding for construction was authorized in 1957, and construction begun with Chicago Bridge and Iron being the lead. The first dry critical experiments were begun in December 1960 while the heat transport system was being constructed and tested. The reactor had its first wet criticality in November 1962. After a long period of testing, the reactor was brought to 30 MW in August 1964, and incremented to its final thermal power of 62.5 MW by September 1969. The original mission of EBR II was to further demonstrate breeding based on the experience of EBR I, to be a complete power plant, and to operate as a closed fuel cycle. To that end, the Fuel Cycle Facility (FCF) was constructed adjacent to the reactor building, and a complex cask system developed to move hot spent fuel into FCF for processing and recasting into fuel elements to go back to the reactor. That phase of the mission was completed, and in the 1970s the reactor was used primarily as a fuels and materials test reactor, including testing fuel developed for the Fast Flux Test Facility (FFTF). With breeding demonstrated, the inner blanket region was changed from U-238 to a stainless-steel reflector. In the 1980s the mission changed again, and EBR II became an important component of the Integral Fast Reactor (IFR) program. As part of that program, two landmark tests were conducted in April 1986: (1) loss of flow without scram, and (2) loss of heat sink without scram. In both tests, the reactor experienced mild temperature transients but the reactor shutdown due to strong negative reactivity feedback (Planchon et al., 1987). Ironically, the Chernobyl accident occurred just several weeks after the EBR II tests. In spite of the technical successes, the reactor and the entire IFR research were terminated by Congress in 1994, and reactor operation ceased on September 1994.

Dounreay fast reactor (DFR)

This test reactor, located in northern Scotland, was a loop-type reactor. It was designed to produce 60 MW thermal power and 15 MW electrical power, very similar to EBR II (Metz, 1976). Construction began in 1958 and the reactor achieved its first criticality in 1959. It exported electricity to the UK National Grid beginning in October 1962 until its shutdown in 1977. It was replaced by the Prototype Fast Reactor (PFR) which went critical in 1974 at the same site. PFR is described in the section on prototype reactors.

Rapsodie

Rapsodie was the first step in the French breeder program (Zaleski, 1980). The design started in 1957 and construction began in 1962 at the Cadarache site in southern France. It achieved its first nominal power level of 20 MW thermal in 1967. The fuel was a mixed plutonium and uranium oxide. It had a very simple heat removal system; heat was released to the atmosphere via sodium-air heat exchangers. The power level was increased in several steps ending up at 40 MW in 1971. From 1967 to 1976 Rapsodie operated virtually continuously with about a 90% capacity. It was shut down for about 6 months, upgraded, and operated again until closure in 1982. In April 1994, a fatal explosion occurred during neutralization of some residual sodium. A multi-stage cleaning process using ethylcarbitol was used creating over time an explosive mixture that ignited (Marmonier and Del Negro, 1995).

SEFOR

The Southwest Experimental Fast Oxide Reactor (one loop, mixed oxide plutonium uranium fuel) operated for a very short time, 1969–72 (Hixson, 1972). As the name suggests, it was fueled by MOX and no electricity was produced. SEFOR used a reflector control system: nickel reflector segments were dropped below the core for shutdown, and brought up for operation. Pre-operational testing began in May 1968 continuing until April 1969. The facility had a fairly specific set of objectives for a three-year experimental program: (1) to determine the physics parameters of sodium-cooled fast oxide reactors, particularly concentrating on Doppler and other quantities important to safety, and (2) to demonstrate the reactor operating characteristics. To these ends, four phases were planned and executed which included static testing, oscillator experiments, delayed-critical transients, and prompt-critical transients. Upon completion of the program, the reactor was shut down in 1972. The site was taken over by the

University of Arkansas after nuclear material was removed and used for research purposes until 1986. The University finally received decommissioning funding from the Department of Energy (DOE) and dismantling was completed in 2019.

BOR 60

BOR 60 was commissioned in 1967 designed for 60 MW thermal and 15 MW electrical (very similar to EBR II and DFR). Its fuel is MOX, with highly enriched uranium. BOR 60 has served mostly as a fuels and materials irradiation facility (Trojanov et al., 1990) serving as a testbed for new fuels before going into BN 350 or BN 600. Vibro-packed MOX fuel has been tested in this reactor, and an extensive structural materials irradiation program conducted, concentrating on low-swelling ferritic steels. Additionally, cesium traps with graphite as absorbent have been tested and refined in this reactor, and knowledge applied to BN 350 and BN 800. BOR 60 is still operating after 50 years and will likely stay operational until the MBIR (a ‘multi-purpose sodium cooled research reactor’) is completed.

Joyo

Joyo is a 100 MWth fast reactor in Japan, operating since 1977 (Sawai et al., 1990). It has been used as a fuel and material irradiation facility as well as a test bed for other components of the reactor. Its fuel is a plutonium-uranium oxide mix enriched to a little over 20% at equilibrium. Joyo has been used to test decay heat removal by natural convection as part of a program to demonstrate inherent safety of fast breeder reactors. Joyo is in a temporary shutdown following a general slowdown in reactor development in Japan particularly following the Fukushima accidents; however, the decision has been made to decommission its planned successor (Monju) but continue research at Joyo (NEI, 2016).

Some prototype reactors

Fermi 1

The Fermi 1 reactor was designed to be a 400 MWth reactor, but with its first (and only) core it only generated 200 MWth with 61 MW electric. It was built by and operated by Detroit Edison to be the U.S.’s first prototype breeder reactor. Construction began in 1956, and first criticality achieved in 1963. However, the facility was beset with many problems and useable power not generated until 1966 (Reindl, 2016). Within months, a zirconium catch plate had broken off and partially blocked some of the coolant inlet flow plates causing two fuel subassemblies to partially melt (about 1% of the core). The reactor was immediately shut down, and there was no subsequent radioactive release beyond the containment building. It still took almost 4 years to investigate the problem, remove the damaged subassemblies and restart the reactor. The reactor then began operation, generating an overall modest amount of electricity until the core hits its burnup limit in 1972. At that point the utility decided to permanently shut the facility leaving the U.S. with no operating prototype fast breeder reactor. Fermi 1 was to be followed by the demonstration reactor, the Clinch River Breeder Reactor (CRBR) but that project was canceled in the early 1980s.

BN 350

BN 350 could be considered the world’s first (successful in contrast to Fermi 1) industrial scale breeder which was also used to power a large desalination plant. The reactor is located on the Mangyshlak Peninsula in the Former Soviet Union country, Kazakhstan, on the Caspian Sea. It is located on the site of a “secret uranium mining center” (Guth, 2018). The design was started in 1960 and the reactor first went critical in 1972. It was connected to the grid in 1973 and initially operated at powers of around 300 MWth until issues with the steam generators as well as other start-up problems were solved. From about March 1975 to January 1996 it operated at a 650–750 MWth level at around 90% capacity (IAEA, 2006). Leaks in two of the steam generators cause water incursion into the sodium, so two out of six loops were taken out of service and the plant derated to about 420 MWth. It was permanently shutdown in April 1999. One of the original goals of BN 350 was to be a testbed for improvements in burnup. It started with around 9% burnup (fissioning ~9% of the loaded heavy metal) and gradually improved that to better than 12% through introduction of various martensitic-ferritic steels for subassembly and fuel wrappers (Trojanov et al., 1990). To help flatten the power profile, the core had three zones of enrichment, from 17% to 21%. Various fuels of plutonium, uranium, and MOX were tested over the years feeding valuable data to its follow-on power reactor, BN 600.

Phenix

Phenix was also a successful prototype, built and operated in Marcoule, France (on the Rhone River near the city of Orange). It was a 250 MWe pool-type with three secondary loops and modular steam generators (Carle, 1973). The design began in 1964, construction in 1968, and operation commenced in 1973 followed by so-called industrial operation in 1974 (Zaleski, 1980). Following a few years of startup problems with the steam generators in particular, in April 1978 Phenix achieved full operating power and continued to run for years at typically 85% capacity factors. In the beginning, operation of the pool-type reactor proved to be “easy,” which was one of the early objectives of the facility. The reactor was rated at 590 MWth over a core volume of 1200 L. It

demonstrated a breeding ratio of 1.12 with PuO_2 fuel. Because of the higher coolant temperatures, the reactor operated with a thermal efficiency of 44% (Kohler et al., 1990). Operation could be divided in roughly three phases: (1) 1974–90 Demonstration of Na cooled FBR technology, (2) 1990–93 Investigation of reactivity transients, and (3) 1994–2002 Renovation with two full operating cycles. From 2003 the reactor was derated to 350 MW operating with only two secondary circuits. It was shutdown in 2009 and is in the decommissioning phase. Phenix was followed by Superphenix, a large technological step at 1200 MWe.

PFR

The Prototype Fast Reactor (PFR) was located at Dounreay, UK (northern Scotland). It began operation in 1975, designed for 250 MWe and 630 MWth (Jensen and Olgard, 1995). It was shutdown in 1994. The active core was 0.915 m in height, 1.47 m diameter yielding a core volume of around 1500 L. It was fueled with a combination of PuO_2 and UO_2 with an inner zone enrichment of 23.5 w/o (Pu) and outer zone enrichment of 32 w/o (Pu). The reactor was a pool-type with three primary and three secondary loops. Above and below the active core was a depleted uranium blanket, a radial blanket surrounded the core as well. The original target burnup was 7.5% (heavy metal) but through improvements in cladding and plenum designs 15% was achieved. Operation of PFR can be grouped into two phases (IAEA, 2006). In the first decade, operation was plagued by steam generator leaks (37 in all) preventing any load factor from exceeding about 12%. After 1984 most of the problems were fixed through improvements in welding and other design changes, and the load factor increased to about 57%. In 1985, the facility could operate with a full set of steam generators for the first time. In this second (and last) decade of operation there was only one major outage of 18 months caused by leakage of oil from the primary pump into the sodium. This occurred in 1991; the reactor was operated for the last time in 1994.

Russian submarine reactors

While they fall outside of the definition of test reactors, we mention the lead-bismuth eutectic systems used by the Soviets for submarines from the mid-1960s to the mid-1990s because of the importance of their design and operating experience with a promising reactor type for the next generation of advanced reactors. In the period mentioned above, the Soviets built 12 reactors and 15 cores and accumulated 80 reactor-years of operations (Alemerti et al., 2014). While the U.S. chose sodium as the coolant for their first liquid metal submarine reactor (Sea Wolf) because of problems with lead-bismuth corrosion, the Soviets invested the time to solve the problems. They ultimately succeeded mostly through better engineering of coolant systems and better control of the coolant chemistry (Zrodnikov et al., 2000). The advantages are the very high boiling point of the lead-bismuth eutectic and its chemical inertness, in contrast to sodium on both counts. Lead-bismuth reactors do not suffer from positive reactivity void coefficients because of the high boiling points. Also the very low absorption of the coolant allows the reactor to operate with less excess reactivity (fuel mass). There were three accidents that resulted in the reactor systems being unusable, but only one is directly attributable to the actual coolant type. That occurred in 1968 when the formation of lead oxides accumulated in the core causing local coolant blockage. A series of events lead to partial core meltdown. The solution was better oxygen control of the primary system. In the last 10 years of operation there were no further problems due to corrosion or purity control.

It is true that the lead-bismuth eutectic reactor does produce Polonium-210, a dangerous alpha emitter with a 138 day half life. However the Soviets found that the creation of aerosols would be the main source of release and transport of polonium is mitigated by rapidly lowering the temperature and solidifying the eutectic. Also the low concentration of polonium in the system leads to the formation of polonium-lead compounds which reduce the polonium pressure by a factor of 1000.

Other reactors

This survey has attempted to focus on the pioneer facilities both in the experimental reactor and prototype reactor phases. Additionally, some reactors that were purely fuels and materials test reactors, with no electricity generation, have not been highlighted. Notably, there is no description of the Fast Flux Test Facility (FFTF) which operated at 400 MWth from 1980 to 1992, but Joyo in Japan is described because Joyo is Japan's first experimental fast reactor. If one is interested in more information there is a comprehensive list compiled by the IAEA and available as a technical document (IAEA, 2006). Additionally the World Nuclear Organization maintains a searchable database.

Summary

Mixed success

It is fair to say that the fast reactor program has a mixed record regarding its development as a commercial power source. There have been some successful programs, notably EBR II which was well on its way to becoming a test facility for the advanced IFR program until it was canceled by the U.S. Congress in 1994. Some had very difficult start-up periods (cf. Phenix and PFR), often with steam generator problems exacerbated by the sodium/water boundary that must not be breached. Fermi 1 suffered a partial core melt

within the first few months of power operation. And although Phenix is considered a success, it did turn out to be much more expensive to construct and operate than originally envisioned.

Early cancelations or closure

In the U.S, the CRBR never made it to the final design stage before being canceled by Congress. The Monju reactor in Japan suffered a series of accidents starting with a sodium fire in 1995 that prevented any significant hours of electricity production since the first criticality. In 2016 the decision was made to decommission the reactor. The SNR-300 reactor in Germany was constructed, but never operated. Finally Superphenix in France, a very large step from the successful Phenix, was closed in 1997 by the French government after a number of years of mishaps, including a roof collapse in an auxiliary building due to snow load.

Future

At the present time there is not much need for a breeder reactor so it is likely that the future of the fast reactor lies in its use as a fuels and materials test reactor. The inherently higher fast neutron fluxes allow accelerated testing of advanced fuels and materials in support of the extension of the current LWR fleet as well as in support of Gen-IV systems if any country decides to actively pursue such systems. Examples are Joyo (which has survived while Monju has not), the MBIR under construction in Russia, the planned Versatile Test Reactor in the U.S. and Astrid (in France).

Summary Table.

Reactor	MWth	Fuel	Coolant
Clementine	0.025	Pu	Hg loop
EBR-I	1.2	U	NaK loop
BR-5	5	PuO ₂ -UO ₂	Na loop
EBR-II	62.5	U-Zr	Na pool
DFR	60	U-Mo	Na loop
Rapsodie	40	PuO ₂ -UO ₂	Na loop
SEFOR	20	PuO ₂ -UO ₂	Na loop
BOR-60	55	PuO ₂ -UO ₂	Na loop
Joyo	100	PuO ₂ -UO ₂	Na loop
Fermi-1	200	U-Mo	Na loop
BN-350	750	UO ₂	Na loop
Phenix	560	PuO ₂ -UO ₂	Na pool
PFR	650	PuO ₂ -UO ₂	Na loop
Soviet submarine	155	–	Pb-Bi loop

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The Transient Reactor Test (TREAT) Facility at the Idaho National Laboratory

Daniel Wachs^a, Nicolas Woolstenhulme^a, Colby Jensen^a, and James Parry^b, ^a Nuclear Fuels and Materials Division, Idaho National Laboratory, Idaho Falls, ID, United States; and ^b TREAT Division, Idaho National Laboratory, Idaho Falls, ID, United States

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Introduction

Transient reactors are a unique branch of materials test reactors designed for investigation of material behavior that rapidly evolves in extreme nuclear environments. While many of these facilities have been and continue to be used for the study of nuclear physics, most are dedicated to the investigation of reactor and fuel safety research. These test reactors played a particularly prominent role in the early days of nuclear technology development when the foundation of reactor safety was being set. As light water reactor (LWR) technology matured and become widely implemented, the need for transient testing waned and many facilities were decommissioned. However, a resurgence in advanced reactor technology development and exploration into higher performance LWR fuel designs has returned these facilities to the forefront of the field (Wachs, 2012).

Licensing of nuclear fuels for reactor applications requires empirical demonstration of their performance limitations and confirmation that those limits are not exceeded during all anticipated operating occurrences (AOOs) and design basis accidents (DBA). The nuclear regulator community often extends this exploration to include fuel behavior under severe accident conditions. In all cases, transient testing reactors are essential to replicate this unique nuclear environment and, thus, empirically demonstrate the relevant behavior in the most prototypic environment possible.

The Idaho National Laboratory's (INL) Transient Test Reactor (TREAT) facility shown in Figs. 1 and 2 is a foundational piece of the U.S. advanced fuel technology development and licensing enterprise. Constructed in 1959 (Freund et al., 1958; Pope et al.,



Fig. 1 Aerial photograph of the Transient Reactor Test (TREAT) facility.

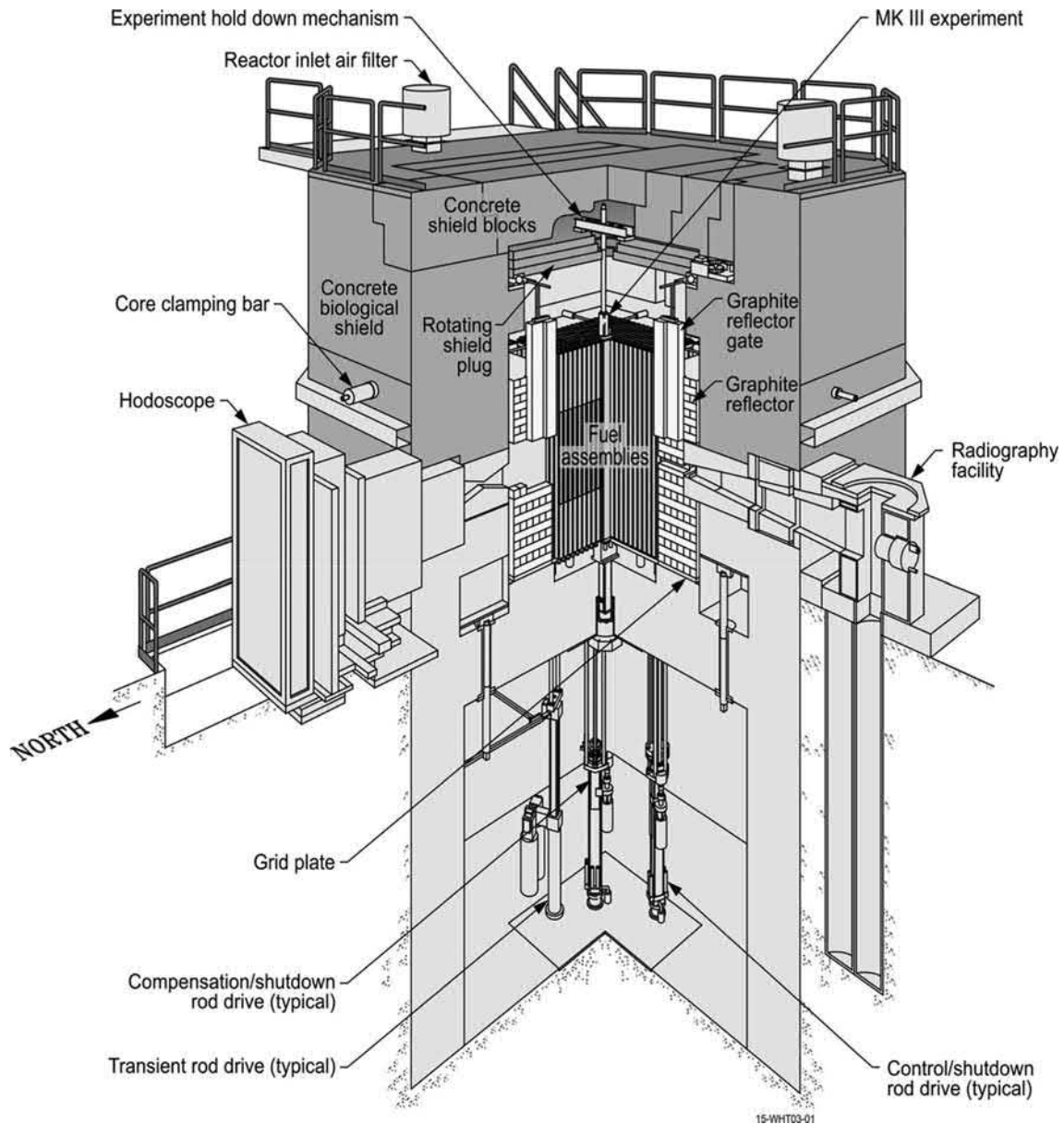


Fig. 2 TREAT reactor layout.

2019), TREAT was one of the world's earliest nuclear reactors and was the first to be dedicated to nuclear fuel safety testing. TREAT conducted thousands of experiments in the following decades and helped form the basis for the reactor fuel safety limits still applied worldwide in both commercial power generating stations and research reactors. TREAT was also critical in the evolution of fuel systems projected for use in advanced reactor concepts, sodium-cooled fast reactors (SFR) in particular. This significant versatility across multiple reactors types, in spite of TREAT's thermal neutron spectrum, is due to the fact that fuel safety testing is driven by the study of thermal-mechanical response of fuel systems to time dependent fission energy deposition. Thus, spectral effects are relatively unimportant when compared to testing under steady operating conditions.

However, after enabling decades of critical nuclear research and development, TREAT was placed in standby mode in 1994 due to massive reductions in U.S. nuclear energy research funding. TREAT was subsequently returned to operation in 2017 (Bumgardner and Wachs, 2019; Heath and Race, 2019) following a remarkable resurgence in interest in both enhanced fuel designs for LWR and advanced reactors applications.

TREAT is designed to deliver a wide array of nuclear transients that mimic expected and hypothetical events of interest to nuclear fuel and reactor technology developers. These nuclear transients are complemented by the ability to insert experiment specific test rigs into the reactor that are capable of simulating whatever unique thermal hydraulic environment that is required to mimic the reactor system of interest.

Nuclear reactor design fuel assembly

The TREAT core consists of a 19×19 array of 10-cm (4-in) square fuel assemblies arranged on a grid plate forming a 1.9-m (6-ft 4-in) square reactor core region. The fuel assemblies consist of a 1.2-m (4-ft) long fueled region with 0.6-m (2-ft) long graphite reflectors attached above and below (Fig. 3). The fueled region consists of 8 axially stacked graphite blocks containing a 0.3 wt% UO_2 powder (93% enriched) such that each assembly contains approximately 37.4 g U-235. The fueled segment is encapsulated in Zr-3 cladding. The graphite reflectors are similarly clad in aluminum and are attached to the fueled segment by rivets. The upper and lower heads of the fuel assembly are fitted with a handling fixture and a grid plate locating pin, respectively.

Pathways for the air coolant flow from the upper plenum to the lower plenum are formed within the reactor through slightly chamfered corners on the fuel assembly and a small standoff built into the end plugs. The chamfered corners create 16-mm (0.625-in) square coolant channels on the corners between fuel assemblies. The standoffs effectively separate the fuel surfaces between each fuel assembly by 2-mm (0.080-in.) and create additional cooling channels as well as protecting the fuel from damage during handling. Several of the fuel assemblies are also outfitted with thermo-couples to allow for core temperature monitoring during transients. The thermocouples are embedded in various internal locations including the cladding inner surface, fuel surface, and fuel centerline. These elements are strategically located in limiting temperature positions in the TREAT core to validate safety calculations associated with any given core configuration.

Nuclear physics parameters and reactivity control system

The TREAT facility is capable of generating peak thermal power approaching 2 GW during a limiting step reactivity insertion. At this power level the peak instantaneous neutron flux is greater than 10^{17} n/cm²/s¹. The flux nominally scales with the reactor power, depending on specifics of the experimental configuration, such that an instantaneous power of 10–50 MW (0.5–2.5% of peak power) can deliver fluxes prototypic of either an LWR or SFR, thus providing clear capacity to drive fuel up to and beyond the failure thresholds. However, these remarkable values do not adequately describe TREAT's critical nuclear physics parameters. The TREAT reactor is designed to deliver large shaped nuclear transients while leveraging its natural response to protect the plant from damage. This is accomplished through a fast acting, computer-controlled reactor control system that can introduce and remove reactivity based on a specific user's needs. This active control is backed up by a very strong negative reactivity feedback coefficient that ultimately terminates transients without control rod insertion. This behavior is a result of the unique thermal-nuclear response of the TREAT reactor (which is not the typical doppler shift seen in other reactors). During operation, the core is effectively adiabatic due to the high rate of power generation and the low rate of heat rejection. Significant negative reactivity is therefore inserted when the neutron energy spectrum shifts up toward higher energy as the graphite heats up, thereby reducing the effective fission cross section in the fuel phase. The reactor control system is therefore not relied upon for safety functions which provides the experimenter with remarkable flexibility in test design.

The TREAT reactivity control system consists of three independent types of control rods that are manipulated from the bottom of the core. The various drive systems push the control elements into the upper plenum such that downward movement, via gravity or control action, inserts negative reactivity into the core. The control elements consist of 4.45-cm (1.75-in) diameter B_4C rods clad in carbon steel. Each rod is 1.5-m (60-in) long and is followed by an upper and lower graphite extension clad in mild steel. Each rod penetrates the lower grid plate and is mated to special purpose fuel assemblies that have a central hole extending through its entire axial length that is designed to accommodate the rod and its movement.

The three types of TREAT reactor control rods, including (1) the compensation/shutdown rods, (2) control/shutdown rods, and (3) transient rods, each serve a specific reactor control function. The compensation/shutdown rods include four rods used to adjust the overall core reactivity worth when it is loaded with an experiment. The position of each rod is independently set prior to operation using a screw-driven drive mechanism. The rods are also used to terminate transients using a fast-acting, pneumatically accelerated scram system. The control/shutdown rods consist of eight rods assembled into two independent banks that are used to support reactor control during startup and steady state operations. These are operated by systems similar to the compensation/shutdown rods (e.g., screw driven control with pneumatic assisted scram). Finally, eight transient rods operated in four banks are used to initiate and control the reactor transient. The transient rods can be controlled either by an operator in "low-speed" mode or computer controlled in "high speed" mode. The transient rods are hydraulically driven at rates up to 3.56-m/s (140 in/s). The rods can be moved in either direction (in or out of the core) at this speed.

The reactor is typically operated in one of three modes, depending on the user needs, including steady state, temperature limited transient, or shaped transient. The reactor can be operated indefinitely in steady state mode at roughly 120 kW where the heat rejection and generation from the air cooling system and reactor core, respectively, are in balance. A temperature limited transient is induced through a step reactivity insertion achieved by rapid control rod ejection (Chase et al., 2019). During this event, the reactor power will exponentially increase until the negative reactivity insertion due to increasing core temperature is sufficient to fully suppress the power excursion. The size and length of the semi-Gaussian pulse shape will depend on the size of the specific reactivity insertion. The pulse width, typically characterized by its full-width at half-max, can range from 100s of milliseconds to several seconds when operated in this mode.

The most common TREAT operating mode is the shaped transient. In this mode the reactor control rod movements are pre-programmed into the automatic reactor control system in order to deliver the unique transient prescription required for a given experiment. A wide array of shaped transients can be executed with this system (Holschuh et al., 2019). The most common shaped transients are based on a series of simply segmented reactivity control actions. For example, a standard temperature limited transient

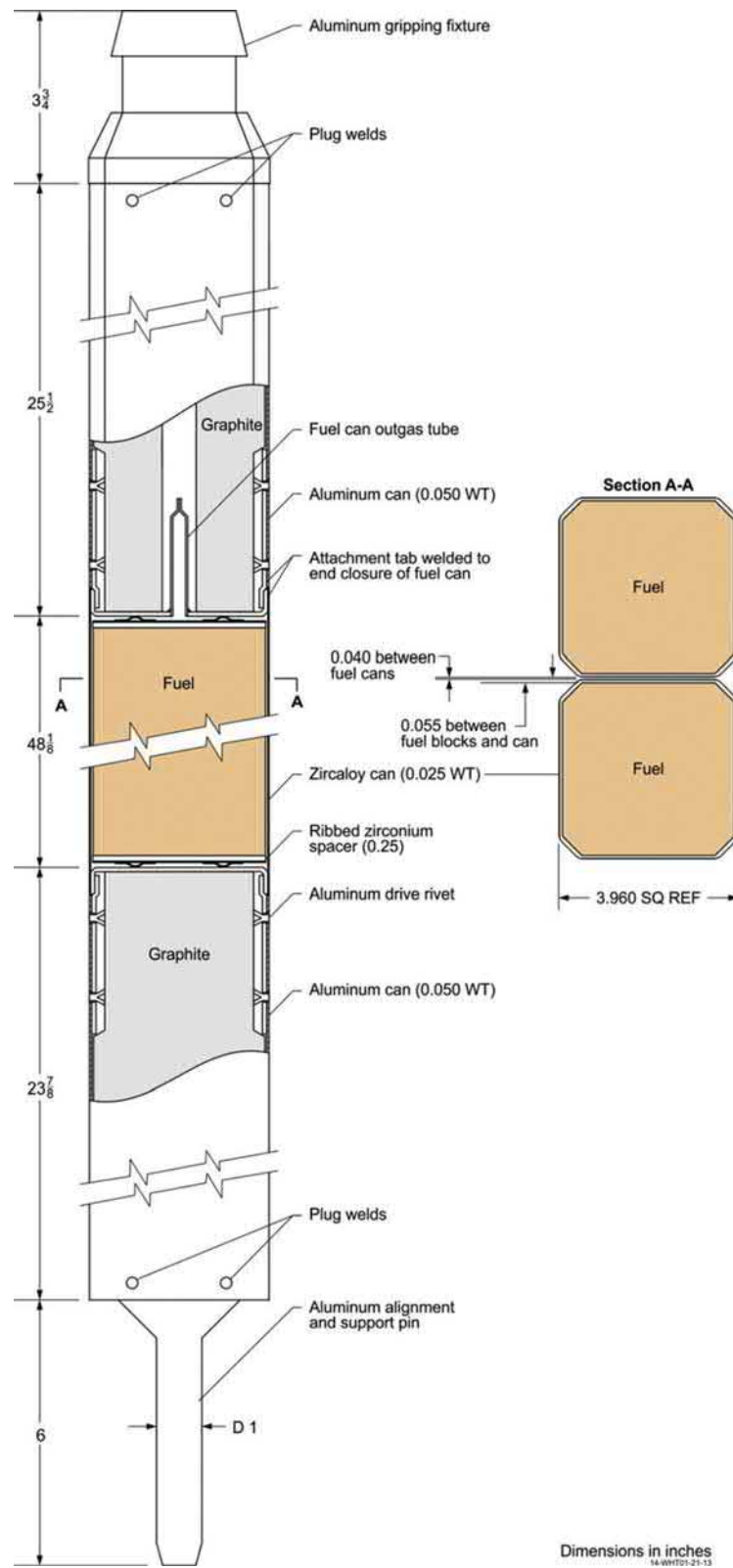


Fig. 3 Schematic of the TREAT fuel assembly (dimensions in inches).

can be “clipped” by early insertion of the control rods to terminate the transient and reduce pulse widths to below 80–90 ms (Bess et al., 2019). Addition of a He-3 based control “rod” operating in a similar mode is expected to shorten the pulse even further, perhaps as low as 40 ms. The reactor can also be operated at constant power greater than the steady state limit, known as a “flat top,” or at a fixed power ramp by gradually removing the control rods to make up for the core reactivity loss as the core heats up. While operating in this mode the reactor gradually heats up until reaching either the operating set point limits (2500 MJ of energy addition) or the excess reactivity is exhausted by the negative reactivity feedback changes that accompany core heating.

In more sophisticated shaped transients, multiple segments of these types can be combined to imitate complex events in nuclear reactors. This includes simulation of LWR loss of coolant accidents (LOCA) or SFR loss of flow (LOF) events that require a period of normal operation followed by a period of decay heat simulation. Shaped transients are typically only constrained to a maximum integrated TREAT core energy addition to limit core heating such that cladding temperatures do not exceed ~ 600 °C during normal operation or ~ 800 °C during accident conditions to prevent degradation of the Zry-3 cladding due to accelerated oxidation.

Core configuration

Within the core boundary, the fuel assemblies can be reconfigured within the constraints of the 19×19 grid plate to create test sections of various sizes (Fig. 4). The core is typically arranged to accommodate a 10-cm by 20-cm (4-in by 8-in) test section by removing two fuel assemblies. However, it can also be configured to accommodate up to a 25-cm (10-in) diameter test section (by removing 9 assemblies) that matches the core shield and the inner diameter of the experiment handling cask. This geometry is the effective mechanical constraint for experiment size. The experiment hardware, as discussed in a later section, is self-contained and readily inserted and removed from this test position.

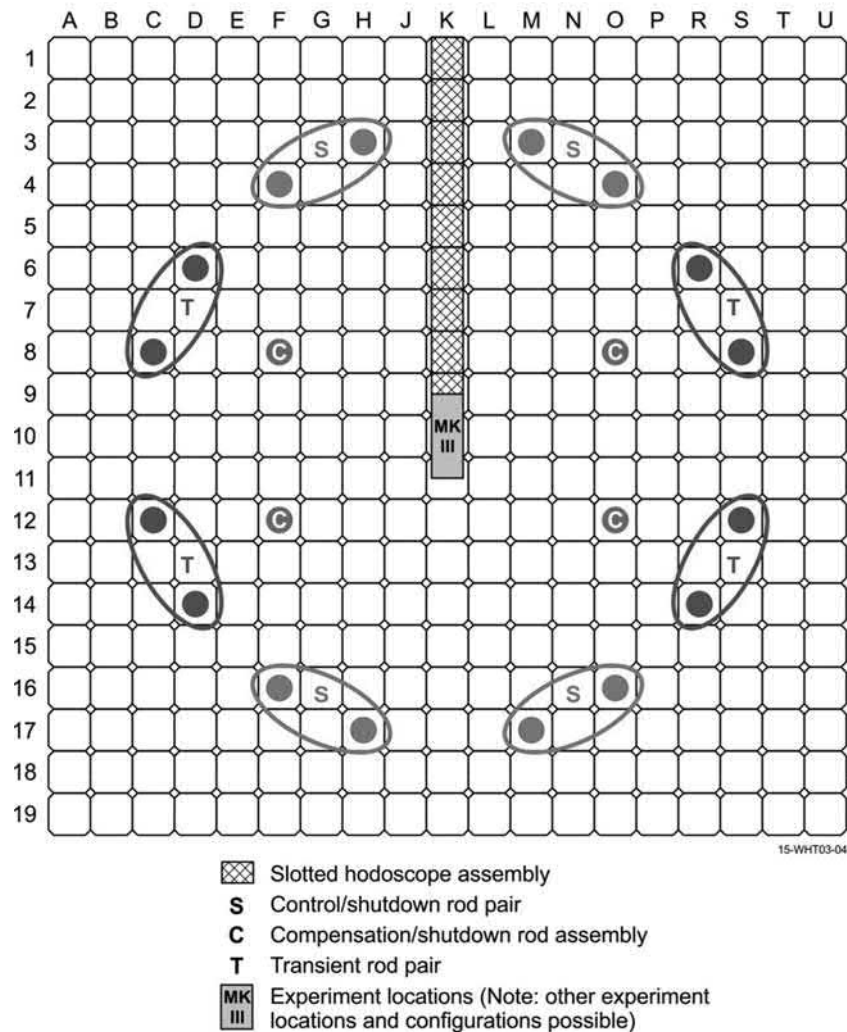


Fig. 4 TREAT core layout including hodoscope slot, experiment position, and reactor control rod types and locations.

The TREAT core is surrounded successively by a 0.6-m (2-ft) permanent graphite reflector and 1.5-m (5-ft) concrete biological shielding region. Openings in both the biological shielding and reflector are available on each side of the reactor to create a direct line-of-sight to the core. These positions can be used to support specialized beamlines or for direct observation of the experiment. The West penetration is occupied by a neutron radiography station that is used to image experiments before and after irradiation (Jensen et al., 2019). The North penetration is occupied by the fuel motion monitoring system (also known as the fast neutron hodoscope) (Hix et al., 2017) which is used to monitor fuel motion in the test section during irradiation. This in-situ viewing is accomplished by creating an open channel between the experiment and the hodoscope detectors by removing a full row of fuel assemblies. Fast neutrons emitted from the test section propagate down this channel through a collimator and impinge on the hodoscope's detectors. These ZnS scintillators (Ocampo Giraldo et al., 2019) are sensitive to fast neutrons but are transparent to gamma radiation and thermal neutrons. Thus, the intensity of the fast neutron signal can then be correlated with the number of fissions occurring (and thus the fuel mass) in a particular region of the test section. While qualitative motion assessment is valuable and available for all experiments, advanced users can also achieve quantitative determination of the fuel mass through knowledge of both the local effective fission cross sections and reactor flux. The remaining two penetrations, on the South and East, remain available for special purpose applications.

Cooling system

TREAT's thermal management strategy is unique amongst the research reactor community in that, while the facility is characterized as air-cooled, during operation it is essentially adiabatic. TREAT's large, very dilute core (1:10,000 uranium to carbon atom ratio) provides a substantial thermal mass that absorbs virtually all of the fission heat generated during transient operation. Core temperatures commonly approach the operating limits of 600 °C during large transients while staying well below the safety limits of 820 °C where minor cladding damage can begin occur due to oxidation. The core heating is critical to providing the passive shutdown mechanisms described above. The heat is subsequently rejected over several hours through the air cooling system. This system nominally draws air from the reactor building into the upper plenum through two pre-filters on top of the reactor. However, the system is not leak tight and air can be drawn in from various openings in the core reflector and shielding. Air is then pulled into the core cooling channels and the lower plenum before being passed through a HEPA filtration system and exhausted through the building stack. This system is powered by two parallel blowers that are capable of driving 170 m³/min (6000 cfm) of air through the core. With full operation of the core cooling system, the core can be cooled and ready for additional transient operations in roughly 4 h.

The maximum threshold for total energy injected into the core during a transient is limited to 2500 MJ during a prompt pulse (<1 s) and 2900 MJ during a shaped transient to keep the cladding temperature below 600 °C. Below this temperature threshold there is virtual no oxidation of the Zry-3 cladding in air. If the temperature were to exceed roughly 800 °C, rapid oxidation would occur and lead to fuel damage. Although the fuel elements contain a negligible amount of radionuclide due to the extremely small amount of lifetime exposure associated with transient operations, fuel handling and continued use of the fuel for reactor operations could be problematic following exposure to temperatures greater than these thresholds.

Experiment systems

TREAT delivers a unique nuclear environment that can be representative of various transients of interest to a wide variety of nuclear systems. However, this must be complemented by experimental devices that deliver the desired thermal-hydraulic-mechanical environment required to meet the experiment objectives. These devices are entirely self-contained and operated independently from the TREAT driver core with their own control, instrumentation, and thermal management systems. A suite of experiment enabling devices has been developed (Woolstenhulme et al., 2019) to support cross-cutting missions that are tailored for a wide range of experimental objectives. These include integral tests that carefully match a projected accident scenario as well as simplified separate effects tests targeted at understanding general transient fuel behavior.

The simplest form of TREAT testing is conducted using a modular experimental system that separates components focused on reactor safety from components focused on experiment objectives (shown in Fig. 5). During the transient, the TREAT core must be protected from potential damage associated with postulated failures in the test hardware due to reactor or experiment design basis accidents. The broad use specimen transient experiment rig (BUSTER) module is built around a robust pressure vessel that meets these reactor safety function requirements. This pressure vessel takes the form of a long cylindrical 316-SS "pipe" with closure flanges that is mounted within a 10-cm by 20-cm (4-in by 8-in) package designed to interface with standard TREAT core structures and support routine handling (including transport to hot cell facilities if necessary). Insertion of the device is shown in Fig. 6. This device is reusable for multiple experiments and, thus, minimizes the repeating cost of experimental systems with significant safety implications.

The test hardware, called the sample environment rodlet transient testing apparatus (SERITA), is then inserted into the BUSTER pipe. This engineered system is loaded directly into the BUSTER pipe and is tailored to the unique experiment objectives by combining several experiment modules as necessary. The base unit for a SERITA device typically consist of a static capsule that is designed to contain a test sample and is equipped with several penetrations to support in-situ instrumentation (Jensen and Fleming, 2019). The separate effects tests holder (SETH) module represents the simplest capsule configuration available. This sealed capsule is produced through direct laser metal sintering (DLMS) of titanium alloy which allows for use of complex internal geometries defined by the experimenter. The capsule can be back filled with a user selected gas and offers limited environmental control

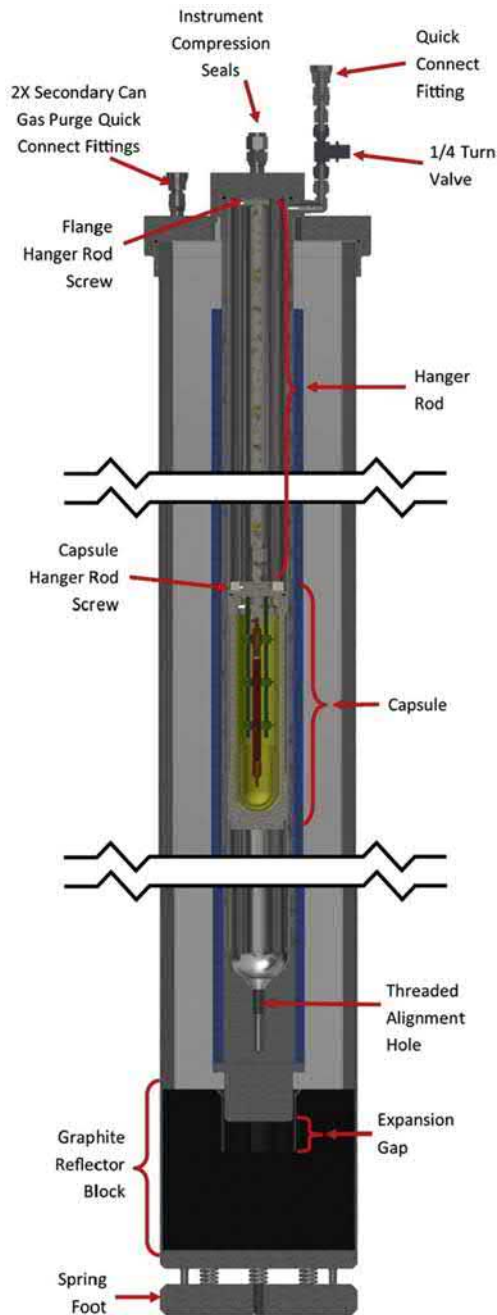


Fig. 5 The Multi-application reusable capsule holder system (including the BUSTER and SETH modules).

(e.g., electric pre-heat up to 700 °C prior to the nuclear transient). A variety of unique separate effects experiments can be accomplished using this test configuration; recent examples are described in [Woolstenhulme et al. \(2020a,b\)](#), [Schulthess et al. \(2020\)](#), [Spencer et al. \(2019\)](#), [Hernandez et al. \(2020\)](#), [O'Brien et al. \(2019\)](#), and [Brown \(2020\)](#).

A variety of additional modules can be added to the SETH-SERTTA device to enhance the range of experiment conditions available. For example, a pressurizer tank module was added to the top of the SETH capsule to allow for transient tests with the experiment submerged in pressurized water. This test configuration can be used to simulate the effects of both design basis accident and anticipated overpower events on light water reactor (LWR) fuels ([Kameraman, et al., 2020](#); [Emerson, et al., 2019](#)). Further, an expansion tank module with a fast acting relief valve can be attached to the bottom of this capsule to allow for its rapid, controlled depressurization. This test configuration can be used to simulate the early stages of an LWR loss of coolant accident (LOCA) ([Woolstenhulme et al., 2020a,b](#)). Following this methodology, modules have also been developed to support advanced reactor fuel testing applications.



Fig. 6 Photograph of the MARCH system being inserted into the TREAT reactor core.

More sophisticated experiment devices can also be deployed to conduct fully integral transient test simulations on engineering scale specimens. Self contained “cartridge”-style loops capable of recirculating flow with active control have been used extensively in TREAT to conduct both overpower and undercooling transient tests that simulate complex events ranging from anticipated operational occurrences through severe accidents. Loops to support both LWR (Woolstenhulme and Epiney, 2019) and sodium-cooled fast reactors (SFR) (Jensen et al., 2017) will be available for transient testing in the near future. These loop devices are prepared and loaded with pre-irradiated fuel at the INL’s hot fuel examination facility (HFEF) and are transferred to TREAT in a specialized transport container.

This model can be further extended beyond fuel system testing to support the study of integral reactor systems behavior. For example, TREAT’s unique design can also be leveraged to support investigation of integral micro-reactor systems. In this mode the TREAT core is configured to allow for incorporation of a large enough micro-reactor segment to capture all the key reactor systems. A dedicated heat rejection system is attached to provide prototypic cooling to the test section. The TREAT automatic reactor control system (ARCS) is programmed to respond to signals from the test segment (i.e., thermocouples embedded in the micro-reactor segment) with nuclear kinetics that mimic the full micro-reactor core. The integral system experiment can be driven by the TREAT core in either steady-state or transient mode to investigate dynamic response to a variety of events. These tests provide a significant incremental test bed that lies between benchtop demonstrations and prototype reactor facilities that can help accelerate both development and licensing of new designs.

Summary

The Transient Test Reactor (TREAT) at the Idaho National Laboratory is a special purpose test reactor designed to support transient testing of nuclear fuel and reactors components. These transients are primarily used to safely simulate the extreme environment inside a nuclear reactor core under off-normal or accident conditions. The versatility enabled by TREAT's unique design consisting of an air-cooled dispersed UO_2 fuel phase in a graphite pile and computer controlled, fast acting control rods is a foundational resource to nuclear reactor system developers. A variety of irradiation test devices and in-situ instruments can be inserted into TREAT's core to simulate the desired thermal-hydraulic conditions associated with virtually any reactor system during a user defined nuclear transient. This unique flexibility has allowed TREAT to support nuclear fuel development and testing activities for reactors ranging across standard light water reactors, research reactors, sodium cooled fast reactors, gas cooled reactors, and even space reactors.

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The Advanced Test Reactor

D. Sean O’Kelly, Idaho National Laboratory, Idaho Falls, ID, United States

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Introduction

The Advanced Test Reactor (ATR) is an evolutionary research reactor that was conceived just over 16 years from the first criticality of Chicago Pile-1 (CP-1) and began operation less than 10 years later. The history and the evolution of ATR begins with the design, construction, and operation of the two high-flux test reactors that preceded it, the Materials Testing Reactor (MTR) and the Engineering Test Reactor (ETR) (Stacy, 2000). Lessons learned from the operation of those two earlier facilities were applied to the design of the ATR. All three test reactors were built and operated in one area, now called the ATR Complex, within what is now the Idaho National Laboratory (INL) site near Arco, Idaho. This INL area was originally a World War II naval ordnance proving ground and aerial bombing practice range. The US Atomic Energy Commission expanded the site and created the National Reactor Testing Station (NRTS) in 1949. The MTR planning began in 1947 but it was not clear where the reactor was to be constructed until the NRTS was created in Idaho and the AEC chose that location (due, in part to its remoteness) to build the MTR. Groundbreaking for the 30 MW MTR was in 1950 and the reactor went critical in March 1952. To achieve such a rapid construction schedule required the reactor facility to be designed almost at the same time as it was being built. The MTR was a highly-enriched-uranium fueled, beryllium-reflected and light-water cooled reactor. It was designed as a multipurpose reactor with over one hundred irradiation positions, including neutron beam tubes but was primarily focused on accelerated materials irradiation testing (Stacy, 2000).

For years, MTR was the highest neutron flux research reactor in the world, but scientists and engineers desired higher fluxes for testing and MTR power was raised to 40 MW in September 1955. However, it was already clear that even higher fluxes were required for some testing programs than MTR could provide so an improved test reactor design effort began in 1955. The Engineering Test Reactor would use fuel similar to the MTR, but the power level was raised to 175 MW which would increase the neutron flux above the MTR maximum by a factor of four or more. There were many design improvements in the ETR, but two significant changes allowed experimental loops in the reactor interior and moved the location of the control rod drive motors below the reactor. The MTR only permitted relatively small experiments to be irradiated and only on the periphery of the reactor core lattice. By 1955, larger samples, up to 10 in. in length, containing more fissionable materials required higher cooling and larger experimental volumes than MTR could provide. The ETR effective core height was 0.9 m versus the 0.6 m of the MTR core and the core lattice allowed approximately 7.6 cm by 7.6 cm locations for experiment insertion. Moving the control rod drives below the reactor allowed more access from above the reactor but still caused perturbations of the neutron flux as control rods were moved during operation (Gilbert and Stacy, 2006).

Phillips Petroleum Company was the first contractor company for the operation of the MTR and ETR and was asked in late 1959 to design an advanced engineering test reactor that would have a neutron flux in test positions greater than $1\text{E}15$ neutrons $\text{cm}^{-2} \text{ s}^{-1}$. The original conceptual design was referred to as ETR-II but the name eventually was changed to the Advanced Test Reactor (deBoisblanc et al., 1960).

The ATR is a unique purpose-specific reactor design and, with a maximum power of 250 MW, one of the highest power research reactors ever operated in the world. The building exterior of the ATR is shown below in Fig. 1. The ATR typically operates at lower powers, in the range of 115 MW, for normal experiments but does operate at higher power for specific experimenter needs. To achieve the high neutron flux, the fuel operates at high power densities of approximately 1 kW cm^{-3} . The ATR was designed to provide variable neutron flux in different experiment locations and achieves this by operating in a radially unbalanced power condition. The unbalance ratio may be as high as four to one across the core for specific experiment needs; however, this large of a power split is rarely used. As with many high power, high performance research reactors, the ATR core may contain all fresh fuel or



Fig. 1 External view of ATR building.

a mixture of fresh and recycled fuel elements to achieve experiment and cycle length demands. A typical experimental cycle for ATR is from 2 to 8 weeks but cycles as long as 9 weeks have occurred.

Originally conceived by Deslonde deBoisblanc, a senior Phillips Petroleum reactor physicist, the ATR is a very unusual design for a nuclear research reactor. Computer modeling of nuclear reactors was not advanced in the 1950s and the ATR physics parameters were initially calculated using two-group diffusion theory and with a two-dimensional computer code. Following the standard practice for the MTR and ETR facilities, a low power but physically identical version of the ATR core was constructed to validate these computational models and measure core physics characteristics prior to ATR's initial startup. This reactor was named the ATR Critical Facility (ATRC) and started up in 1964. The ATRC typically operates at 600 watts or less and is still used to benchmark computational models and measure reactivity effects of experiments prior to insertion into ATR. These tests are still required today, even with the available advanced computational modeling and simulation capabilities to assure safety is not compromised. As an aside, the ATR design was first mocked-up using the ETR Critical Facility (it had a rectangular grid array) to validate the conceptual five-lobe design and the computational approximations for a cylindrical flux trap.

Reactor internals

The ATR internals consist of a series of tanks or support structures internal to the reactor vessel. These tanks support the reactor fuel and reflector and direct primary coolant flow through the reactor core quadrants. The ATR operates with a pressurized primary cooling system (approximately 2.56 MPa) and uses demineralized light water for cooling. A significant portion of the cooling flow is directed downward through the fuel elements with the remainder providing cooling for the non-fuel reactor internals and experiments. At full flow the ATR primary coolant flow rate is $3.2 \text{ m}^3 \text{ s}^{-1}$. When three pumps are operating with approximately $2.2 \text{ m}^3 \text{ s}^{-1}$ flowing through the fuel.

Reactor fuel

The ATR reactor core contains 40 fuel elements arranged in a clover leaf or serpentine pattern that creates four lobe areas as shown in [Fig. 2](#) below. This geometry maximizes the neutron flux for cylindrical experiments located in the formed flux traps. ATR was originally designed to have nine experiment loops utilizing the flux traps but was reconfigured in 1994 to allow some flux trap locations to have experiments cooled by the reactor primary coolant. The ATR has been reconfigured several times to accommodate different experiments in the flux traps.

Each fuel element forms a 45-degree sector of a right circular cylinder and contains 19 fuel plates roll-swaged into the side plates. The side plates have machined vents that provide interconnection of flow channels from fuel element to fuel element and reduces pressure differentials. Each fuel plate contains an active fuel length of 1.22 m and is loaded with highly enriched uranium matrix (UAl_x) clad with 6061 aluminum alloy. Aluminum was selected due to its high thermal conductivity and low neutron cross section. The fuel elements are subjected to preconditioning at the manufacturer to establish a boehmite (a stable form of aluminum oxide) corrosion layer prior to use in the reactor for repeatable heat transfer. Small amounts of boron in the form of B_4C powder is added to some fuel plates as a burnable poison which allows longer cycle lengths and minimizes power peaking.

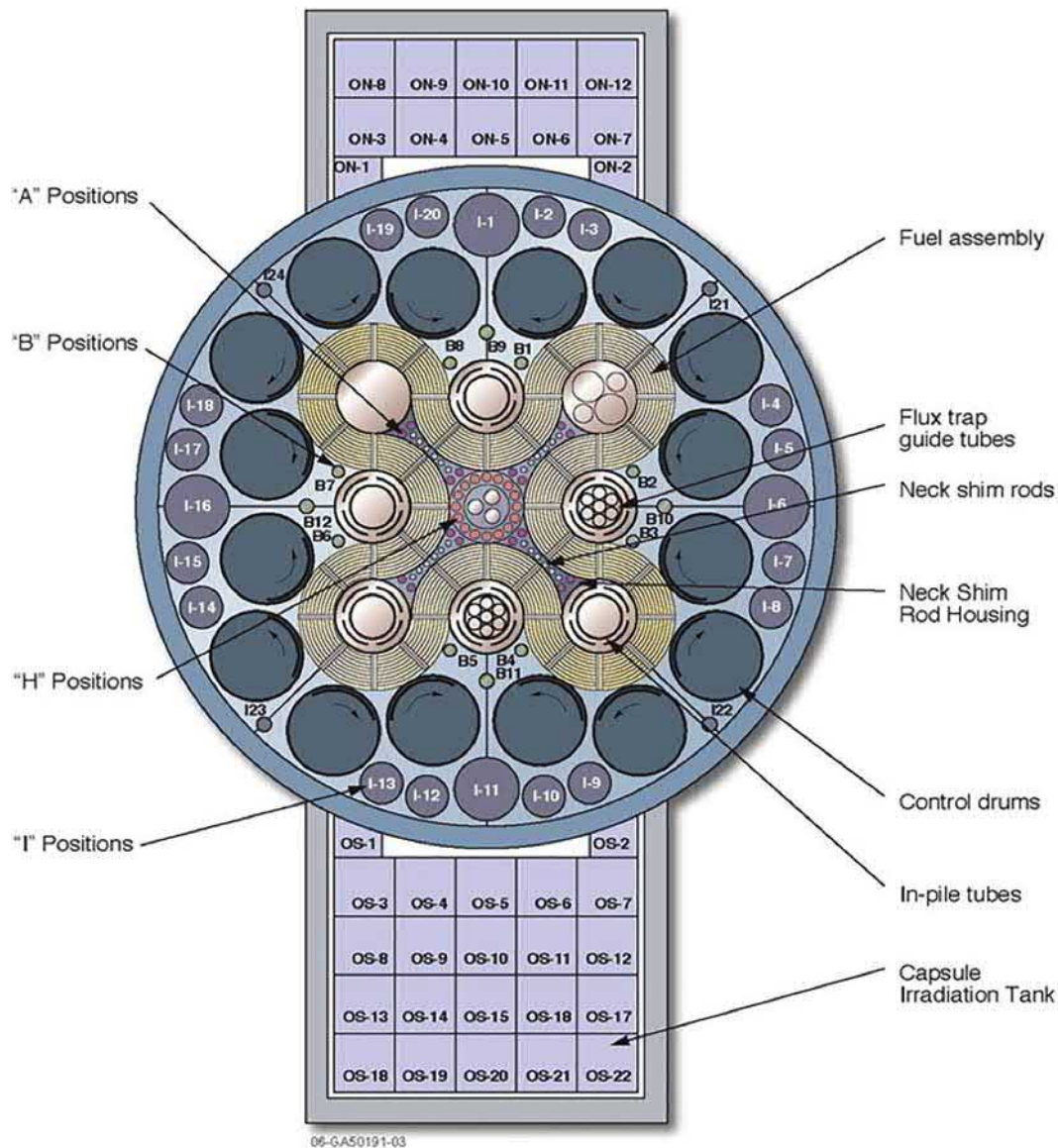


Fig. 2 ATR radial cross-section.

Reactivity control elements

The ATR reactivity control is unique in that it was designed to maintain a near cosine-shaped axial neutron flux profile throughout the operation cycle. The ATR has four independent reactivity control systems, and each has a separate function. These systems are the safety rod system, the neck (fine control) shim rod system, the regulating rod system, and the outer shim control cylinder (OSCC) system. The safety rod system consists of aluminum tubes with hafnium inserts and provides for the automatic or manual rapid shutdown capability. The safety rods are located in six of the nine flux traps areas and are nearly fully withdrawn (7.62 cm remain) above the core region prior to starting up the reactor. There are 24 neck shim and regulating rods located in the neck shim housing located in the center of the reactor. Two of the neck shim rods are used as regulating rods to maintain power automatically via a servo system. All neck shim rods are fully inserted at the beginning of a cycle and gradually pulled out throughout the cycle to compensate for local fuel burnup. The OSCC system consists of 16 beryllium cylinders and each cylinder is 1.19 m in length to be nearly the same height as the active fuel region. The cylinders each have a 120-degree hafnium insert and are operated in pairs to establish lobe power splits and compensate for fuel burnup. Lobe power splits refer to the ability of the ATR to adjust local lobe power for experiment specific power levels. Each lobe consists of the eight fuel elements surrounding a flux trap. ATR is currently potentially able to operate with up to a four to one ratio split across the core for certain experimental programs but generally operates with a 1–10-megawatt difference across the reactor.

Beryllium reflector

The ATR reflector is formed from eight high-grade beryllium manufactured pieces that fill the space between the fuel and the internal diameter wall of the core reflector tank. Primary coolant fills the area between the reflector tank and the reactor pressure vessel. Space is provided between beryllium segments and four holes are drilled through the beryllium sections for cooling water flow. Beryllium in a nuclear reactor will suffer irradiation damage that causes swelling and cracking. Horizontal saw cuts in the ATR beryllium reduce the tensile stresses and increases the reflector life by reducing crack lengths. Evaluation of reflector end of life is performed after stressed-induced cracking is first observed during routine examinations and consists of finite element stress analysis, more frequent inspections, and comparison to neutron fluence of previous reflectors. Approximately every 10–20 years, as determined by the swelling and cracking amount, the ATR beryllium is entirely replaced with new material. During this time, the majority of the other in-core components are also replaced in an evolution called a core internals change-out or CIC.

Instrumentation monitoring

ATR neutron power is monitored as with conventional research and power reactors by monitoring leakage neutrons which are proportional to total reactor power. These neutrons are detected by fission and neutron sensitive ion chambers in detector wells between the reactor reflector and the vessel wall. The ion and fission chambers are contained in 5-in. stainless steel instrument thimbles which are shown clearly in Fig. 3 below. Radiation streaming is minimized by installed shield plugs at the tube exit. Because the ATR typically operates in a power unbalanced condition with non-uniform power distribution, additional local power monitoring is required to monitor and control lobe power. Many neutron detector designs such as self-powered neutron detectors would quickly fail in the extreme neutron fluxes within the ATR. The ATR uses nitrogen-16 (N-16) and water power calculators to determine local lobe powers. The N-16 power monitoring system consists of 10 channels which use reentrant flow tubes in various core lobe regions to sample water flowing through these areas. Fast neutron flux, which is directly proportional to fission power, converts some oxygen in the water to N-16 with a 7.2 s half-life. The decay of the N-16 is detected using beta sensitive detectors and that signal is used to determine relative lobe power. Output from water power calculators and the N-16 power monitoring system are reduced by the reactor data acquisition system to yield the lobe powers.

Reactor vessel

Another unique feature of the ATR is that the reactor pressure vessel is entirely manufactured from 304 stainless steel. The vessel is 3.66 m in diameter, 10.7 m tall, and 50.8 mm thick at the core mid-plane. Stainless steel was selected, rather than clad carbon steel,

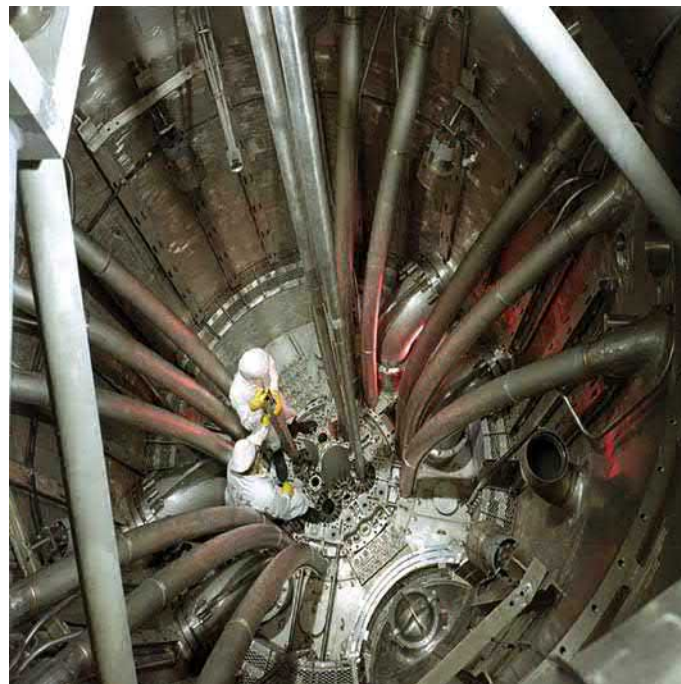


Fig. 3 Men standing on ATR during initial construction.

for corrosion resistance and a lower probability of brittle fracture after long radiation exposures. It is well known that the interaction of fast neutrons from the reactor with the stainless-steel increases tensile strength and reduces the ductility of the material. However, the fast neutron fluence on the ATR vessel is relatively low due to the intervening thicknesses of beryllium and reactor coolant which moderates and slows down fission neutrons as shown in Fig. 4. Calculated total fluence through the year 2060 only results in weldment fracture toughness reduced to 15% below the unirradiated materials; therefore, neutron embrittlement of the reactor vessel is not an issue in the foreseeable future for ATR.

Fresh fuel and new experiments may be loaded into the reactor vessel through penetrations in the reactor vessel head. Used fuel and experiments that are not in experiment loops are discharged to the working canal through a drop or discharge tube assembly. The penetration into the reactor vessel is closed during pressurized reactor operation using a double O-ring cover.

Reactor coolant enters the vessel through two 0.61-m pipes welded to the bottom vessel hemispherical head and then flows up through the area between the reactor core tanks (tanks are defined as enclosed areas, similar to pipes, that separate inlet and outlet coolant flows) and the vessel wall, including cooling the thermal shields. Thermal shields are used in higher power research reactors and nuclear power plants to minimize gamma radiation heating of the reactor pressure vessel. The reactor coolant then flows down through the core and into a flow-distribution tank which divides the flow into quadrants which then flow into four 0.41-m diameter outlet pipes which transport the coolant above the reactor core and out the four outlet nozzles. The coolant flow path then

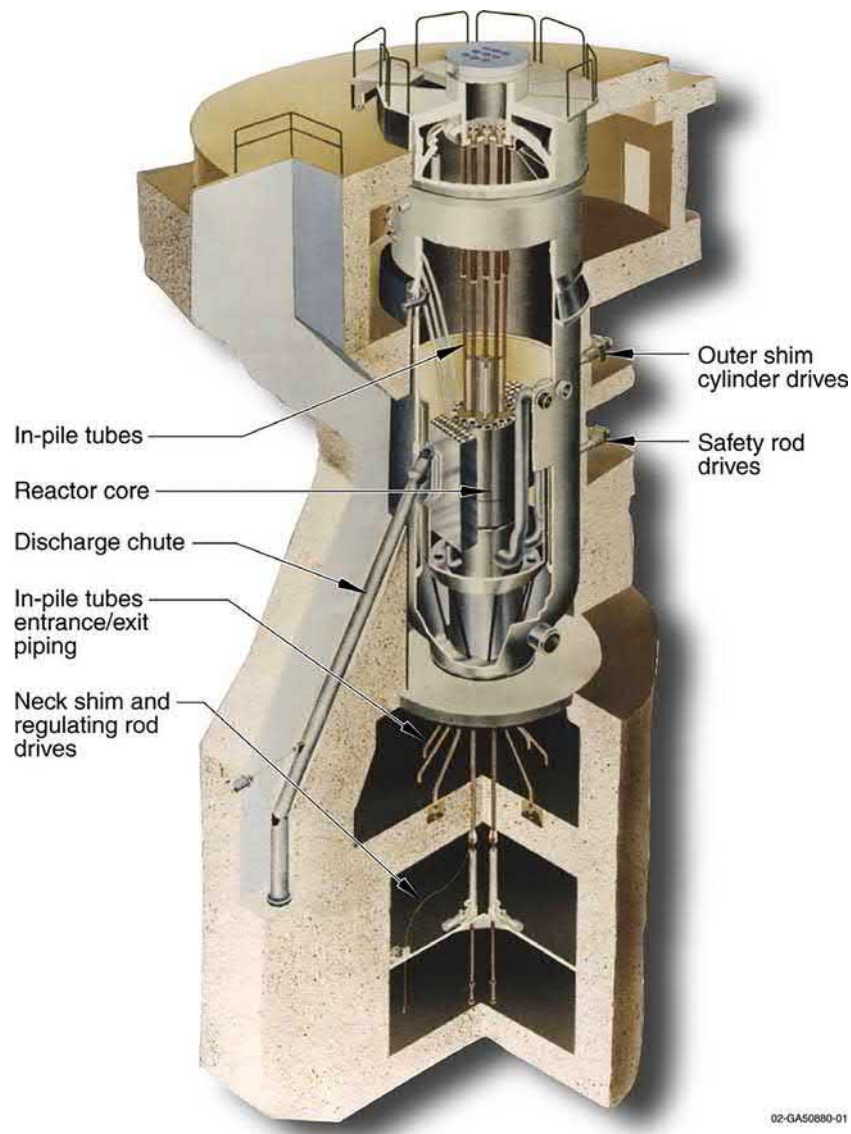


Fig. 4 ATR vessel cut-away.

transitions to 0.46-m pipes which combine into two 0.61 m return lines and finally into a single 0.91-m line that distributes flow to the five primary to secondary coolant heat exchangers.

ATR primary cooling system pressure is maintained by a feed and bleed pressure control system. An automatic pressure control valve bleeds coolant through a degassing tank that removes dissolved gases from the coolant. A pressurizing pump draws coolant from the degassing tank and discharges through a flow control valve back to the primary cooling system. Rapid pressure changes are dampened by a surge tank which has a pressurized air and water volume to provide pressure control (Table 1).

Experiments locations and associated neutron fluxes

As seen in Fig. 2 above, the prime experimental locations in ATR are in the higher flux regions inside the serpentine fuel area, especially the nine-flux trap locations. There are 77 ATR experiment locations with B- and I-positions located outside the core periphery in the beryllium reflector. The position diameters vary from 1.59 to 12.7 cm, but all locations have a 1.22 m available height providing large experiment irradiation volumes. Table 2 summarizes the neutron flux levels in several ATR experiment locations. The peak flux values are given at 110 MW because the ATR rarely operates at full design power of 250 MW for an experiment cycle (ATR User Guide, 2008).

The ATR utilizes three basic in-core experiment configurations: static capsule, instrumented lead experiments, and pressurized water loops (ATR User Guide, 2008). Static or drop-in capsules are a more familiar type of irradiation experiment to users of other research reactors. A static capsule may contain small samples stacked within the welded volume or for large diameter locations (i.e. a flux trap or I-positions) there may be engineering scale components or materials within the capsule. Static capsules are cooled by the primary coolant system and temperature is controlled by a gas gap between the inner capsule and the outer capsule in contact with reactor coolant that is adjusted with pressures and gas mixtures to have a known thermal conductance. Further temperature control would be by selection of the core location to account for local power or gamma heating. No direct temperature monitoring is available for capsule experiments. Melt-wires may provide peak temperature indication and neutron-activated flux wires provide integrated neutron flux measurements when the sample capsule is opened.

Instrumented lead experiments allow control of some experiment parameters and real-time monitoring of the experiment conditions. These experiment capsules are not completely sealed and have small tubes leading in and out of the reactor vessel that carry instrument wires and control or monitoring gases. These experiments are more complicated than simple static capsules and require some type of monitoring and control station to automatically adjust parameters as needed. Temperatures of these experiments may be adjusted by varying the ratio of helium to neon gases in the gas gap to change the thermal conductivity of the gap or by adjusting local reactor powers if agreed to by other experimenters.

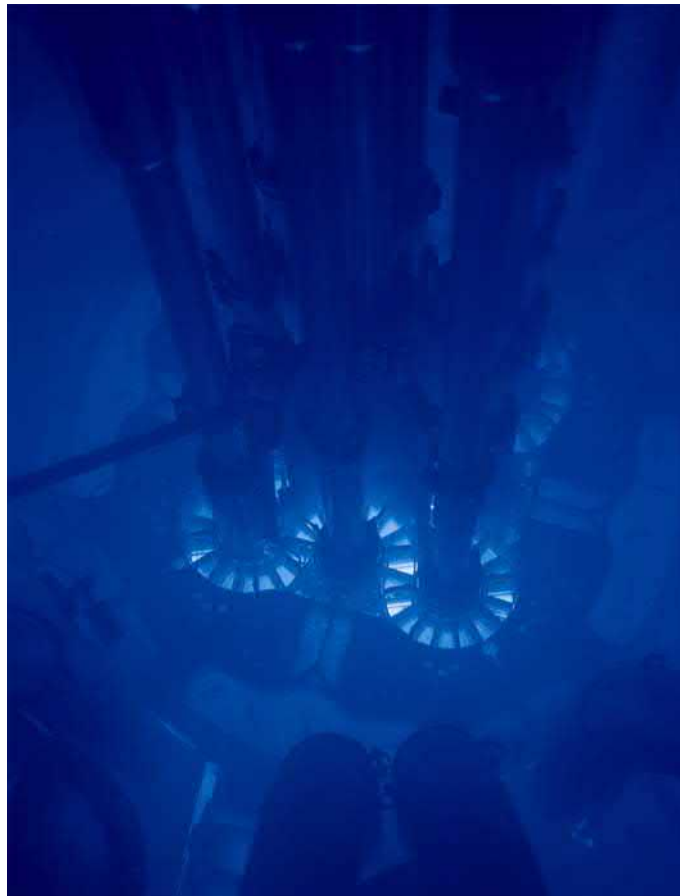
The final and most complicated experiment type used by the ATR is the pressurized water loop. These facilities are referred to as In-Pile Tubes (IPT) and are shown in Figs. 2, 3, and 5. These loops are designed to simulate the conditions (pressure, temperature, chemistry, etc.) of a pressurized water nuclear power reactor (PWR) within the low pressure, low temperature environment of the ATR. To achieve the PWR conditions, each loop experiment has a dedicated system that cools or heats the experiment and maintains high pressure conditions utilizing a scaled down pressurizer system. A low-pressure helium gas gap surrounds the pressure boundary in the reactor vessel to insulate the loop cooling system from the lower temperature ATR primary coolant. These experiments have significant real-time monitoring capability but are quite long (approximately 7.5 m) and must be removed from the

Table 1 ATR general characteristics.

<i>Reactor</i>	
Thermal power	250 MW _{th}
Power density	1.0 kW cm ⁻³
Maximum thermal neutron flux	1.0×10^{15} n cm ⁻² s ⁻¹
Maximum fast neutron flux	5.0×10^{14} n cm ⁻² s ⁻¹
Number of flux traps	9
Number of experiment locations	77
<i>Core</i>	
Number of fuel assemblies	40
Active fuel length	1.2 m
Number of fuel plates per assembly	19
Uranium-235 content (93%) in fresh assembly	1075 kg
Peak fuel temperature	253 °C
Peak heat flux	555 W cm ⁻²
<i>Coolant</i>	
Design pressure	2.7 MPa
Design temperature	115 °C
Light water maximum flow rate (3 pumps)	3.36 m ³ s ⁻¹
Coolant temperatures	<52 °C (inlet), 71 °C (outlet)

Table 2 Approximate peak flux values at 110 MW (22 MW in each lobe).

Core position	Diameter (cm)	Thermal flux ($n\text{ cm}^{-2}\text{ s}^{-1}$)	Fast flux ($E > 1\text{ MeV}$) ($n\text{ cm}^{-2}\text{ s}^{-1}$)
Northwest and Northeast flux traps	~ 12.7	4.4×10^{14}	2.2×10^{14}
Other flux traps	~ 7.6	4.4×10^{14}	9.7×10^{13}
A-positions (16)	1.6	2.0×10^{14}	$1.7\text{--}2.3 \times 10^{14}$
B-positions	2.2 to 3.8	$1.1\text{--}2.5 \times 10^{14}$	$1.6\text{--}8.1 \times 10^{13}$
H-positions	1.6	2.0×10^{14}	1.7×10^{14}
I-positions			
Large (4)	12.7	1.7×10^{13}	1.3×10^{12}
Medium (16)	7.6	3.4×10^{13}	1.3×10^{12}
Small (4)	3.8	8.4×10^{13}	3.2×10^{12}

**Fig. 5** Shutdown ATR showing IPTs and Cherenkov radiation from fuel.

reactor by withdrawing the experiment into a shielded cask on the reactor top for discharge to the adjacent cooling and storage canal or packaged for shipment.

Additional information

In 2020, the ATR was one of the most utilized test reactors in the world with over 90% of the key test locations inside or near the reactor fuel scheduled for at least 5 years into the future. The ATR supports advanced reactor projects by irradiation materials testing of proposed liquid-metal cooled small modular reactors (SMR) materials and fuels, high-temperature gas-cooled reactor fuels (TRISO) (and structural graphite materials), and potential fuels for space propulsion systems. The ATR continues to provide irradiation testing of next generation LEU fuels that might replace the HEU fuels used in many high-performance research reactors and materials testing of advanced materials that could be used in current or future nuclear power plants. Finally, ATR

provides some radioisotope production capability for high specific activity materials and produces Pu-238 as a power source for deep-space probe missions.

Simulation of a fast spectrum reactor in a thermal reactor may be provided by inserting or designing a thermal neutron absorber around the experiment area to shield or filter the experiment neutron energy profile. Some common thermal neutron filters are boron, cadmium, and hafnium. The filters don't enhance the fast neutron spectrum but only remove the thermal energy neutrons. This may be done to test fast spectrum reactor fuels or to separate fast and thermal neutron damage effects for some materials. It is important for the experiment to be designed such that it is cooled by a non-thermalizing coolant, such as a gas or liquid metal to maximize the neutron energy interacting in the experiment. As was already noted, thermal neutron filters only remove low energy neutrons from the reactor energy spectrum and do not increase the numbers of fast neutrons at a given reactor power. Therefore, a thermal test reactor is not an ideal substitute for a fast spectrum test reactor which may have several orders of magnitude higher values of fast neutron fluxes. It is for this reason that countries considering building fast spectrum reactors will require a true fast spectrum test reactor capability. Russia is currently operating the BOR-60 fast spectrum test reactor and the US has begun a design and construction effort to build the Versatile Test Reactor (VTR).

Although the ATR has been operating for over 50 years, every operational cycle needs a different set of experiment conditions that must be established by variable power levels, nuclear material loading of the experiment, and length of the cycle. Because of this flexibility, every reactor core operated over the lifetime of ATR had to be re-designed completely to accommodate the particular experiments that were planned to be in the reactor. In the early several decades of operation, ATR was modeled using legacy computer codes with relatively little fidelity of the output. Today with the use of much more powerful computers, ATR is modeled in three-dimensions when it is necessary and in two-dimensions routinely to establish expected parameters and ensure they will be below safety limits.

Future of ATR

In 2016, the US Department of Energy began an investment program to replace or refurbish equipment at ATR that was operating beyond design lifetimes. This plant health program established long-term goals and plans to improve ATR operational reliability and provide a firm foundation to operate ATR for another two decades. The unique irradiation testing capabilities of the ATR are unlikely to be replaced in the near-term so this national and international asset will be maintained and upgraded as required to serve its diverse user community and provide essential long-term irradiation test program support well into the future.

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High Flux Isotope Reactor (HFIR)[☆]

David Chandler and Christopher D. Bryan, Oak Ridge National Laboratory, Oak Ridge, TN, United States

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Introduction

The High Flux Isotope Reactor (HFIR) is a versatile, multimission research reactor that is operated at Oak Ridge National Laboratory (ORNL) in Oak Ridge Tennessee, United States. With world-class capabilities, this US Department of Energy (DOE) Office of Science User Facility serves a variety of national missions and a broad range of science and technology communities. Neutrons produced in HFIR are used to support cold and thermal neutron scattering experiments, isotope production, materials irradiation research, and neutron activation analysis (NAA). Additionally, the gamma flux from used fuel cores is utilized for gamma irradiation studies, and neutrinos produced in the core are utilized for antineutrino studies. Fueled by 10.1 kg of highly enriched uranium (HEU) (9.4 kg ^{235}U) and operated at 85 MW_{th}, HFIR provides one of the highest steady-state neutron fluxes of any research reactor in the world.

Vision into reality

In October 1957, Glenn Seaborg, the co-discoverer of plutonium, californium, and other heavy elements (Rosenthal, 2010), wrote a letter to the Atomic Energy Commission (AEC) chairman that addressed the need for a new domestic very high-flux reactor capable of producing substantial, weighable quantities of heavy elements (e.g., berkelium, californium, einsteinium) to advance the field of transuranium element research. Seaborg was especially interested in the production of ^{252}Cf . A high-flux reactor is also advantageous in producing high specific activities of short-lived isotopes and accelerating the production of long-lived isotopes.

[☆]Change History: September 2020. D Chandler and CD Bryan updated text and figures to incorporate reviewers' comments and suggestions.

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A series of meetings were initiated in late 1957 to explore this need and to examine the technical problems of designing and constructing a new reactor capable of producing neutron fluxes an order of magnitude greater than any other domestic reactor at the time (Lane, 1958). In November 1958, it was recommended that a new high-flux reactor be designed, built, and operated at ORNL. ORNL concluded that an annular, HEU-fueled flux trap-type reactor consisting of a beryllium reflector, light water coolant, and a centrally located light water “island” (i.e., the flux trap) would provide the required thermal neutron fluxes. The HFIR core configuration is illustrated in Fig. 1. Construction was initiated in Melton Valley in 1961. HFIR was the thirteenth reactor to be built and operated at ORNL, and it is ORNL’s only remaining operating reactor.

Operational history

A series of fuel qualification irradiations and critical experiments were performed to qualify HFIR’s concept of a U_3O_8 -Al dispersion fuel and to finalize the design parameters of the fuel elements in the early 1960s. HFIR achieved criticality on August 25, 1965, and, following a low-power testing program, the reactor was brought to 100% design power (100 MW_{th}) in September 1966. The original operating inlet water coolant pressure of 4.24 MPa was raised to 5.27 MPa in 1977.

In November 1986, HFIR was temporarily shut down because tests on irradiation surveillance specimens indicated that the horizontal beam tube nozzle regions of the carbon steel vessel were being embrittled at a rate faster than predicted. As the result of embrittlement concerns, changes to accepted nuclear safety standards, and the incidents at Three Mile Island and Chernobyl, several committees were appointed to review various aspects of the reactor. In response, the power level was derated to 85 MW, and the operating pressure was reduced to 3.33 MPa. HFIR restarted in April 1989 for low-power operations and for training and achieved 85 MW in May 1990.

Facility evolution

The name of the reactor, High Flux Isotope Reactor, is expressive of its original isotope production mission; however, because of HFIR’s high-flux and spatially varying neutron energy distribution coupled with its easy accessibility for in-vessel experiments, it has evolved to become a multission neutron science research facility. The AEC expected HFIR’s sole mission would be heavy isotope production; however, Dr. Alvin Weinberg, the director of ORNL from 1955 to 1973, saw additional value in the inclusion of beam tubes emanating from the beryllium reflector that would allow neutrons to be beamed into experiments outside the reactor shielding. HFIR’s primary mission today is to conduct neutron scattering research, which is enabled because of Dr. Weinberg’s forethought to incorporate beam tubes into the design. Today, HFIR’s missions include cold neutron scattering, thermal neutron scattering, radioisotope production, materials irradiation, NAA, gamma irradiation, and fundamental physics research. Fig. 2 illustrates the versatile in-core experiment layout.

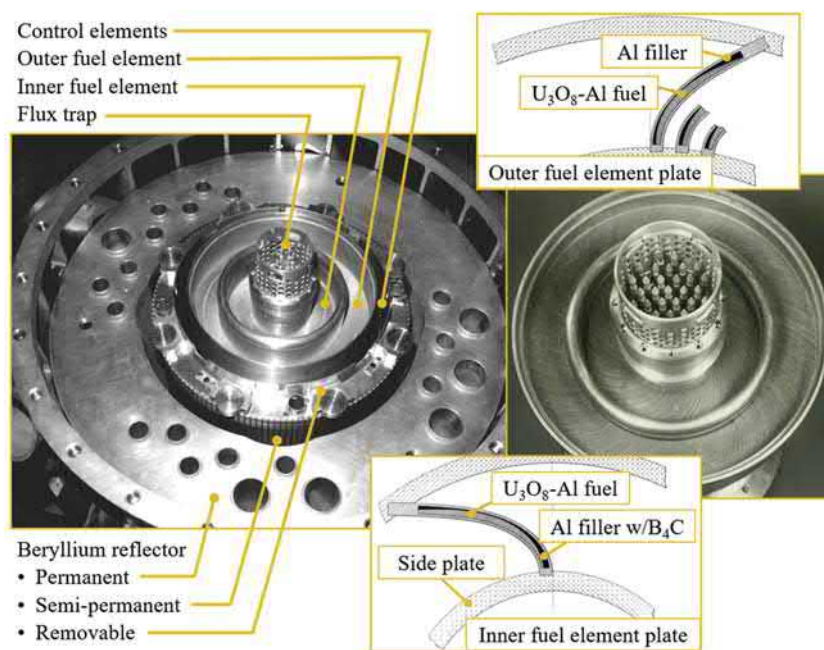


Fig. 1 High Flux Isotope Reactor core mockup.

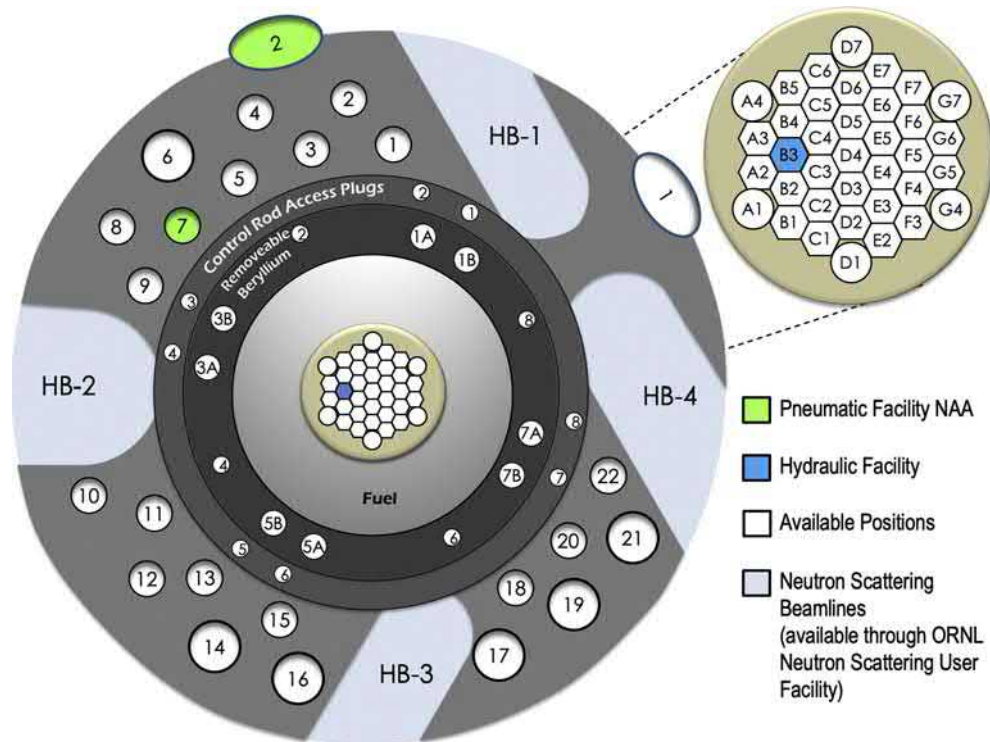


Fig. 2 In-core experiment facility layout.

The first pneumatic tube was installed in 1970 to enable NAA capabilities; in 1987, the second pneumatic tube was installed to further enhance its world-class NAA capabilities. The number and size of material irradiation facilities were expanded in 1987 to improve the efficiency and throughput for materials irradiation research. In 2000, horizontal beam tube number 2 (HB-2) was replaced with a larger beam tube, which resulted in a threefold increase in thermal neutron flux. Horizontal beam tube number 4 (HB-4) was converted into a powerful liquid hydrogen cold neutron source beamline in 2007, resulting in a continuous high-brightness source of cold neutrons being delivered to the cold guide hall instruments. In preparation for the ~2024 permanent beryllium reflector replacement outage, the permanent reflector was redesigned to increase the number of irradiation facilities from 22 to 28, enhance irradiation experiment versatility, and improve thermal-structural performance ([Chandler et al., 2019](#)).

HFIR nuclear reactor design

Nuclear design, experiment, and analysis activities were performed to determine the HFIR core design. The reactor operates at 100% nominal 85 MW_{th} power for ~23–26 day fuel cycles. Currently, HFIR operates seven cycles per year. Both the inner and outer fuel elements are replaced during the end-of-cycle outages, which typically vary between about 2 and 7 weeks. Extended outages are required for some activities, such as replacing the permanent beryllium reflector and making significant updates to the horizontal beam tubes.

Reactor core assembly

The HFIR reactor core assembly design consists of a series of concentric regions about 61 cm in height, including, from the core centerline outward, a flux trap target region, an inner fuel element (IFE), an outer fuel element (OFE), a control element region, and a beryllium reflector ([Fig. 1](#)). The core is axially and radially reflected by light water and is contained in a pressure vessel, which itself is in a pool of light water.

Flux trap

The flux trap, which is located in the centermost region of the core, is 12.70 cm in diameter and utilizes a basket-type design to provide 37 target positions: 31 inside the basket assembly and 6 positions around the periphery of the basket, close to the IFE. The peripheral target positions are typically occupied by materials irradiation experiments that require high fast neutron fluxes. The interior flux trap positions are occupied by capsules containing materials for isotope production and materials irradiation experiments. Experiments requiring high thermal neutron fluxes are often placed near the center of the flux trap.

A hydraulic tube facility allows for up to nine small irradiation capsules (called “rabbits”) to be transported to and from the core on demand using the pressure drop in the core. The hydraulic tube consists of the necessary piping, valving, and instrumentation to move capsules during reactor operation, which enables short to full cycle irradiation periods. The hydraulic tube is especially advantageous for isotopes that reach an equilibrium concentration sooner than the nominal HFIR cycle length.

Fuel elements

The HFIR fuel assembly consists of an integral two-element configuration composed of two concentric fuel annuli. The IFE and OFE contain 171 and 369 fuel plates, respectively, and each of the 540 involute-shaped plates contain fuel meat in the form of U_3O_8 -Al cermet encapsulated in aluminum-6061 clad (Fig. 1). The fuel plate design incorporates a dispersion core with fuel particles dispersed in a continuous aluminum matrix metallurgically bonded to the aluminum alloy cladding. The involute shape is advantageous for several reasons: it provides a uniform metal-to-water ratio with a constant coolant gap thickness, it allows for circular fuel elements with only one unique plate per element, it provides a means for varying the fuel concentration in the radial direction, it minimizes mechanical instability issues (e.g., buckling), it provides a high ratio of heat transfer surface area to fuel region volume, and it maintains azimuthal symmetry and adequate coolant channel thickness when considering channel narrowing effects such as thermal expansion (Cheverton and Sims, 1971).

The fuel is HEU enriched to ~ 93 wt% ^{235}U and ~ 2.6 and 6.8 kg of ^{235}U are loaded into the IFE and OFE, respectively (9.4 kg total). The fuel is contoured along the arc of the involute (i.e., nonuniformly distributed), with thinner fuel at the lateral ends to reduce edge power peaking. The IFE’s aluminum filler region, which is adjacent to the fuel meat, contains B_4C burnable poison for reactivity hold-down and edge power suppression purposes. The lateral edges of the fuel meat directly face water and beryllium moderators/reflectors (e.g., flux trap, labyrinth between elements, and beryllium reflector), which cause thermal flux and power density peaks at the edges. Thin fuel edges and burnable poisons suppress the edge power thereby flattening the power profile and increasing the thermal safety margin.

For both plate types, the fuel plus filler region thickness is 0.762 mm, and the clad thickness is 0.254 mm. The fuel plates are thus 1.27 mm thick and, because of their involute shape, adjacent fuel plates are separated by a constant 1.27 mm coolant channel width (metal-to-water ratio of 1). Each fuel plate is 60.96 cm in length, and the active fuel zone length is 50.80 cm. The fuel plates are inserted into cylindrical side plates and are attached via welds. The IFE has inside and outside diameters of 12.88 and 26.87 cm, respectively; the OFE has inside and outside diameters of 28.54 and 43.53 cm, respectively. The active volume of the core is ~ 50 L, and thus the average power density is ~ 1.7 MW/L.

Control elements

Two concentric control elements, in the form of thin (0.635 cm thick) neutron poison-bearing cylinders, are located in the annular region between the OFE and the beryllium reflector. The inner control element consists of four curved control plates welded together to form a cylinder, which is used for shimming and power regulation purposes. Four individual shim-safety plates (i.e., quadrants), each having an independent drive and safety release mechanism, form the outer control element, which is used for both power regulation and safety purposes.

Each aluminum-cladded control plate consists of three distinct longitudinal core regions with various neutron absorption capabilities. The white regions located at the upper and lower extremes of the plates are made of aluminum, which is essentially transparent to neutrons. A 12.70 cm long gray region consisting of tantalum-aluminum has a moderate neutron absorption cross section and is used for fine power regulation. The 55.88 cm long black region contains Eu_2O_3 -Al, which is a strong neutron absorber and is used for power regulation and fast scram purposes.

Beryllium reflector

A large beryllium reflector (~ 30.49 cm thick), as shown in Fig. 1, provides very efficient neutron moderation and reflection properties. To accommodate the replacement frequency associated with radiation damage, the reflector is subdivided into three regions to allow the innermost regions to be more easily replaceable. These three regions, in order of their distance from the core centerline and their operational lifetime, are the removable beryllium reflector (RB), the semi-permanent beryllium reflector (SPB), and the permanent beryllium reflector (PB). The reflector experiment facilities can accommodate static experimental capsules, complex-testing engineering loops, and special experimental isotope irradiations.

The RB has eight large aluminum-lined RB experiment facilities and four small unlined RB facilities. The large RB facilities can be used for instrumented or uninstrumented experiments. The SPB contains four control rod access plugs, each providing two small unlined irradiation facilities. The PB, which is the largest of the three reflector regions, is a single large annulus of beryllium penetrated by 22 vertical experiment facilities (VXF), four horizontal beam tubes, and two engineering slant facility tubes. One VXF facility and one engineering slant facility each contain a pneumatic tube facility that uses differential pneumatic air pressure to shuttle small capsules from shielded work areas to the reactor and back for NAA. Four horizontal beam tubes penetrate the reflector and terminate at instruments in the beam room and cold guide hall, where thermal neutron scattering and cold neutron scattering are performed, respectively.

Pressure vessel

The reactor pressure vessel cylinder (i.e., shell) has an inner diameter of about 2.39 m and is constructed of carbon steel clad with austenitic stainless steel on both sides. A stainless steel extension is attached to the lower end of the vessel to permit control

element access to the core through the 2.14 m thick pool floor. The flat lower head is nominally 15.24 cm thick. The upper vessel head is made of 35.56 cm thick carbon steel clad with stainless steel and is secured to the vessel with bolts and sealed with an O-ring (SAR, 2019). Illustrations of the pressure vessel and the flow distribution within it are provided in Fig. 3.

A quick-opening access hatch with a diameter of 76.20 cm is located in the center of the upper head to enable movement of fuel elements, targets, control elements, and inner reflector pieces without the upper head being removed. The openings (i.e., nozzles) where the four horizontal beam tubes penetrate the vessel are exposed to higher radiation damage rates than the other regions of the vessel because of the ability of radiation to stream down the beamlines with limited shielding. With an expected lifetime of ~ 50 effective full power years at 100 MW, the vessel will need to be replaced in the 2050–70 timeframe, depending on future operating schedules. If a future vessel replacement occurs, it will likely be of a similar design, but employ stainless steel to avoid future embrittlement issues.

Nuclear physics overview

Highly energetic neutrons are created from fission reactions in the fuel and are subsequently slowed down via interactions with materials such as the water moderator and the beryllium reflector. At the beginning of cycle (BOC), the peak IFE and OFE fission rate densities are $\sim 6.9 \times 10^{14}$ and 6.0×10^{14} fissions- $\text{cm}^{-3} \text{ s}^{-1}$ and occur on the core midplane at the inner and outer radius of the elements, respectively (Chandler et al., 2019). As the fuel depletes over the course of the cycle, the power distribution flattens, and the peak IFE and OFE fission rate densities decrease to $\sim 3.2 \times 10^{14}$ and 4.7×10^{14} fissions- $\text{cm}^{-3} \text{ s}^{-1}$, respectively, at the end of cycle (EOC).

The unirradiated HFIR fuel assembly contains ~ 10.1 kg total uranium and therefore reaches an average burnup of about 220 MWd/kg U during a 2200 MWd operating cycle. At a recoverable energy of 200 MeV per fission, this is equivalent to about 0.23 fissions per initial uranium atom. The peak EOC cumulative fission density for the IFE and OFE are $\sim 1.1 \times 10^{21}$ and 1.3×10^{21} fissions- cm^{-3} , respectively (Chandler et al., 2020). Both peaks occur on the core horizontal midplane; the IFE peak occurs at the inner radial edge adjacent to the flux trap; the OFE peak occurs at the outer radial edge adjacent to the beryllium reflector.

Fig. 4 illustrates the core midplane BOC and EOC radially dependent three-energy group neutron flux profiles with thermal, epithermal, and fast neutron energy boundaries of 0.625 eV, 0.1, and 20 MeV, respectively. The peak fast neutron flux of $\sim 2.0 \times 10^{15}$ neutrons- $\text{cm}^{-2} \text{ s}^{-1}$ occurs in the IFE. Neutrons that leak into the flux trap slow down via interactions with water, which is why the thermal neutron flux increases from $\sim 1.5 \times 10^{15}$ to 2.5×10^{15} neutrons- $\text{cm}^{-2} \text{ s}^{-1}$ from the outer radius of the flux trap to the centerline of the flux trap and the fast neutron flux decreases from $\sim 1.5 \times 10^{15}$ to 1.0×10^{15} neutrons- $\text{cm}^{-2} \text{ s}^{-1}$. Similarly, as neutrons leak out into the beryllium reflector, their energy is reduced due to interactions with the beryllium. At BOC and EOC, the peak thermal neutron flux in the beryllium reflector is about 1.1×10^{15} and 1.4×10^{15} neutrons- $\text{cm}^{-2} \text{ s}^{-1}$, respectively.

The spectral brightness of the HB-4 cold source operating with a liquid hydrogen moderator temperature of 22.5 K and a reactor power level of 85 MW is described in (Robertson and Iverson, 2008). The peak brightness of $\sim 1\text{--}2 \times 10^{13}$ neutrons- $\text{cm}^{-2} \text{ s}^{-1} \text{ str}^{-1} \text{ \AA}^{-1}$ occurs in the 2–3 Å energy range.

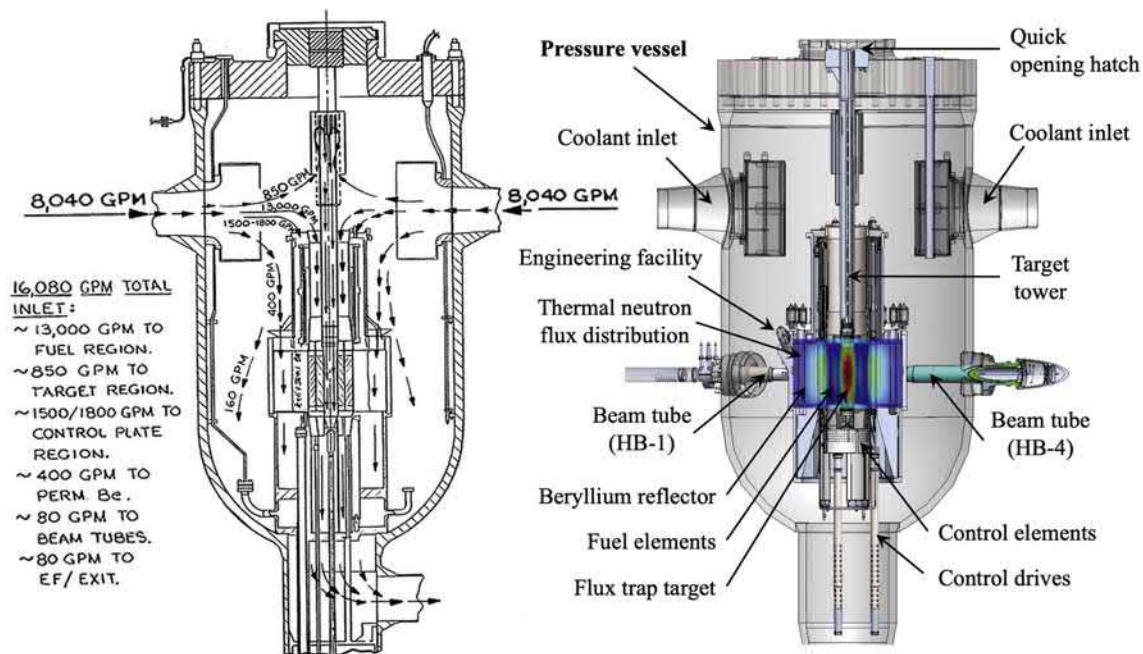


Fig. 3 Reactor pressure vessel and flow distribution.

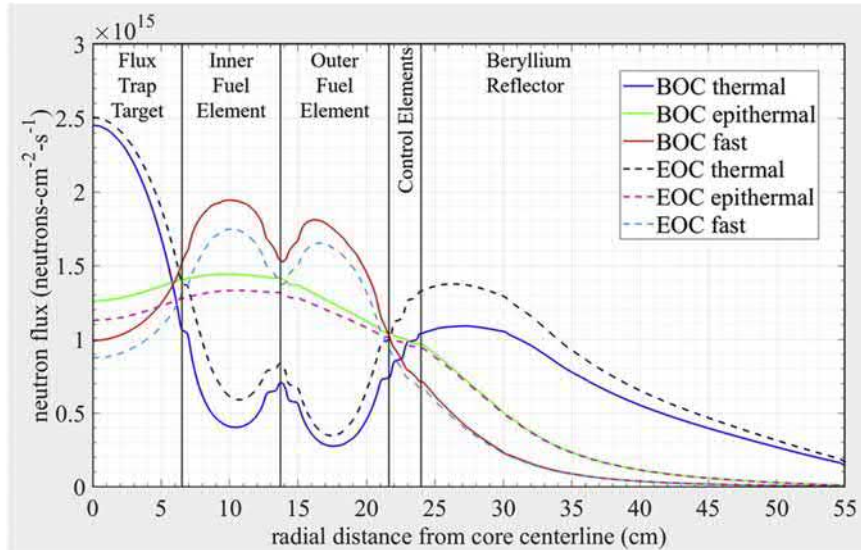


Fig. 4 Radial three energy group neutron flux profiles.

Reactivity control and safety systems

The fuel assembly cannot achieve criticality without being contained within the beryllium reflector. Therefore, the control elements are in the annulus between the fuel and the reflector to vary the efficiency of the reflector. The control element drive mechanisms, located in the subpile room below the pressure vessel, drive the control elements in the axial direction. When the reactor is shut down, the europium-bearing black regions of the control elements are adjacent to the active fuel zones. However, during operation, the elements are symmetrically withdrawn in opposite directions with the shim-safety plates being withdrawn vertically upwards and the shim-regulating cylinder being withdrawn downwards. The symmetric movement drives the poison-bearing regions away from the core midplane, increases the “neutron window,” maintains criticality, and maintains axial symmetry in the power distribution. The reactor achieves criticality when a small window opens about the core midplane. The control elements continue to withdraw throughout the cycle to compensate for fuel depletion and fission product buildup.

A servo-control system maintains constant power by moving the inner shim-regulating cylinder, which is accomplished through a system of three independent servo channels, each supplied with a servo reset flux signal. Each servo channel compares the operating power level indicated by the reset flux signal with the requested power indicated by its servo demand setting (SAR, 2019). The neutron flux level is monitored by six ionization chambers (three associated with the servo-regulating system and three associated with the safety system) located in three thimbles that are positioned around the periphery of the beryllium reflector, each containing one servo and one safety ion chamber.

The reactor instrumentation and control system design reflects the emphasis placed on the importance of continuity of operation while maintaining safe operation. It is an independent protection system whose function is to monitor reactor conditions and provide a method of initiating a rapid shutdown should any of the parameters affecting safety exceed preset values. Three independent safety channels are arranged in a coincidence system that requires agreement of two of the three for reactor shutdown. The shim-safety plates and drives are designed to ensure that the reactor can be reliably shut down whenever a protection system parameter reaches its trip threshold. Any single shim-safety plate or the shim-regulating cylinder can shut down the reactor or maintain the reactor subcritical under clean conditions with the other plates fully withdrawn. The shim-safety plates travel from fully withdrawn to fully inserted within ~ 250 ms.

Primary and secondary cooling systems

The primary coolant system’s key safety functions include heat removal, pressure boundary integrity of the high-pressure system, reactor vessel overcooling protection, containment of fission products, prevention of coolant flow blockage through the core, and maintenance of core exit pressure. Down-flowing demineralized water coolant enters the pressure vessel through two inlets above the reactor, passes through the core, and then exits the vessel through the outlet below the core. The inlet coolant temperature and pressure are ~ 48.89 °C and 3.33 MPa, respectively. Approximately $0.82 \text{ m}^3 \text{ s}^{-1}$ of the total $1.01 \text{ m}^3 \text{ s}^{-1}$ coolant flow passes through the fuel elements where a ΔT and ΔP of ~ 20 °C and 0.76 MPa, respectively, are nominally maintained. These conditions result in a coolant velocity of $\sim 15.54 \text{ m s}^{-1}$ through the fuel elements.

The flow passes through the target, fuel, control, and reflector regions in parallel paths, and, after exiting the pressure vessel, the coolant branches into four parallel headers. The coolant flow is typically distributed to three of four identical shell and U-tube-type heat exchanger and single-speed vertical centrifugal primary circulation pump combinations. The primary water from each heat

exchanger enters the associated main circulating pump and then recombines in a return line before passing through the inlet strainer and entering the reactor vessel. The primary coolant passes through the tube side of the heat exchangers, where the primary coolant gives up its heat to the secondary coolant, which is circulated through the shell side of the heat exchangers. The heat in the secondary coolant is dissipated to the atmosphere via a four-cell conventional induced-draft cooling tower mounted over a concrete basin.

Decay heat removal following reactor shutdown is normally provided by continuing to operate the primary coolant pumps to reject heat to the secondary; however, if the primary pumps aren't available, decay heat is removed by circulating the primary coolant via the dc-powered pony motor pumps.

Containment system

HFIR containment is defined as the primary coolant system pressure boundary as it is employed to contain fission products (SAR, 2019). A dynamic containment design, referred to as a confinement system, is employed. The system maintains a constant negative internal pressure with respect to atmosphere. Controlled ventilation moves air from areas of less potential contamination to areas of high potential contamination. Establishing and maintaining a controlled in-leakage within the confinement safety-related boundary are accomplished by the special building hot exhaust (SBHE) system, which establishes a negative pressure within the physical boundary envelope. Air exhausted by the SBHE system is routed from intake headers in the reactor bay to an underground filter pit. After passing through the filters, exhausted air is monitored and discharged up the exhaust stack, and any residual activity (i.e., noble gases, filter penetration) is dispersed by diffusion.

Current path to conversion to low-enriched uranium fuel

In collaboration with the US Reactor Conversion Program, ORNL has been performing engineering evaluations on the conversion of HFIR from HEU to low-enriched uranium (LEU) fuel since 2005. ORNL is funded by the US DOE National Nuclear Security Administration's Office of Material Management and Minimization to perform conversion studies as part of its nuclear nonproliferation mission. HFIR is committed to converting to LEU once a fuel form capable of ensuring that operations will continue at the current functional and scientific performance level in a safe, reliable, and affordable manner.

HFIR LEU conversion is planned to take place in five phases with the appropriate safety basis preparation by ORNL and with the approval for each by the US DOE Office of Science. The first phase is to present the US DOE Office of Science with a Preliminary Documented Safety Analysis. The second phase consists of running at least one cycle with the current HEU fuel at the higher power required for LEU conversion (e.g., 95 MW) to demonstrate the ability of the reactor cooling systems to remove the higher heat load. In the third phase, in-vessel, zero- and low-power tests of the LEU lead test core (LTC) will be conducted to confirm calculated neutronic parameters. Then, a single cycle with the LTC will be run at the higher power to demonstrate the continued ability to perform HFIR's scientific mission safely. The fifth and final phase consists of operating HFIR long term with LEU fuel at the higher power. The current HFIR conversion date is November 2033 (NNSA, 2019).

Fuel design studies have been conducted by ORNL to evaluate performance and safety metrics for two proposed fuel systems: uranium-molybdenum monolithic alloy (U-10Mo) and uranium-silicide dispersion (U_3Si_2 -Al). Results to date indicate that HFIR could convert to LEU fuel and maintain both performance and safety with the U-10Mo or U_3Si_2 -Al dispersion fuel systems if, among other considerations, the reactor power is increased, fabrication of the specific design features are successfully demonstrated, and the fuel forms are qualified for HFIR-specific irradiation conditions.

Experiment capabilities and highlights

ORNL maintains the missions of high-impact research about the structure and properties of materials across the spectrum of biology, chemistry, physics, materials science, and engineering as well as isotope production, neutron and gamma-induced damage testing of materials, and NAA. Neutrino and other fundamental physics research experiments are also performed at HFIR.

Cold and thermal neutron scattering

Neutrons are generated from fission reactions in HFIR's fuel at very high energies and are subsequently slowed down as they traverse through and interact with the water coolant/moderator, beryllium reflector, and cold source liquid hydrogen moderator. The low-energy neutrons that are directed down the horizontal beam tubes travel to instruments at the end of the beamlines, which allow users to study the structures and dynamics of materials. Originally, all four horizontal beam tubes (HB-1, HB-2, HB-3, and HB-4) delivered thermal neutrons to the thermal beam room; however, in 2007, HB-4 was converted into a powerful cold neutron source and has since been producing the brightest continuous source of cold neutrons in the world. A liquid hydrogen cold source moderator was installed in HB-4, and a new cold neutron guide hall was built to support cold neutron scattering. The layout of the four beam tubes is illustrated in Fig. 5.

Currently, there are seven instrument stations in the thermal beam room and five instruments at the end of the cold source beam tube that are operational and available in the user program. Additionally, there are currently three cold instruments and one thermal instrument in commissioning or operating as development beamlines. The intense neutron flux is used by more than 500

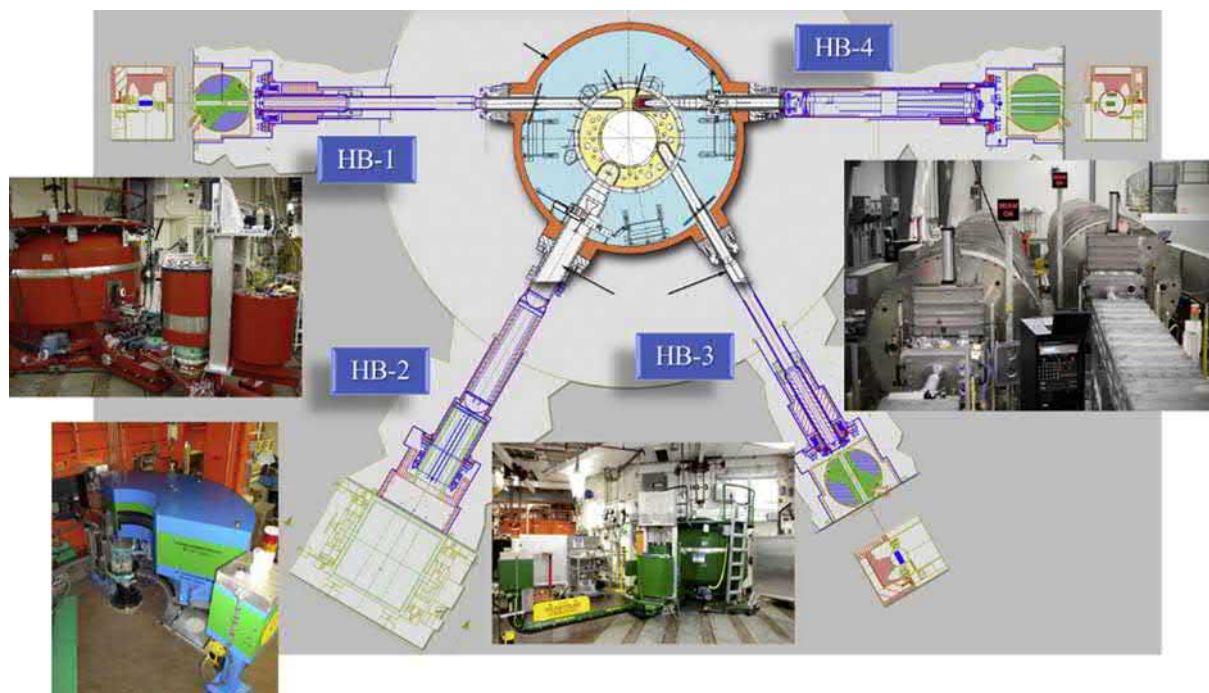


Fig. 5 Horizontal beam tube layout.

researchers each year for neutron scattering research. The neutron scattering research facilities contain a world-class collection of instruments used for fundamental and applied research (HFIR, 2020).

The HB-1 and HB-3 beam tubes are situated tangential to the core while HB-2, the largest of the four beam tubes, extends radially outward from the core centerline such that it directly faces the fuel elements. These thermal beam tubes terminate in the beam room where the thermal neutron instrument suite, consisting principally of triple-axis spectrometers (e.g., fixed-incident-energy, polarized), diffractometers (e.g., four-circle, neutron powder, wide-angle), and a residual stress mapping facility, resides. Versatile sample environments allow for measurements under extreme conditions (e.g., temperature, pressure, magnetic field). Thermal neutrons are used for research focused on impactful scientific discovery across many disciplines, such as studies of magnetic structures, high-temperature superconductors, thermoelectrics, potential quantum spin liquid materials, powdered and ceramic materials, diffuse scattering in single crystals, static and dynamic properties of spin and lattice, and residual stresses in materials.

The HB-4 tangential beam tube has an aluminum moderator vessel that contains supercritical hydrogen that is nominally maintained at 17 K. Thermal neutrons from the reflector scatter into the moderator vessel, where they are further moderated to cold energy levels (i.e., wavelengths of $\sim 4\text{--}12$ Å). The stream of neutrons delivered down HB-4 is split into four individual beams, each of which is directed down a cold guide that terminates in the cold guide hall. The cold neutron instrument suite principally includes instruments for imaging, small-angle neutron scattering, triple-axis spectrometer, and diffractometer studies. The imaging beamline is used for neutron radiography and computed tomography for applications in additive manufacturing, materials science, geoscience, biology, energy, and transportation. The general purpose and biological small-angle instruments are vastly used for soft condensed matter, hard condensed matter, magnetic systems, bio-macromolecules, bio-membranes, complex systems, biomass, biofuels, and biomaterials studies. A triple-axis spectrometer is ideal to study static and dynamic properties of spin and lattice in materials. The neutron image plate, single-crystal diffractometer provides atomic-resolution information on structures in biology, chemistry, and complex materials.

Isotope production

The in-core experiment facilities, as illustrated in Fig. 2 and previously discussed in the Flux Trap and Beryllium Reflector sections, are used for medical, industrial, and research isotope production. The ^{244}Cm , ^{248}Cm , ^{249}Bk , ^{249}Cf , ^{252}Cf , and ^{254}Es isotopes are the most commonly produced heavy isotopes at HFIR, are chemically separated at ORNL's Radioisotope Engineering Development Center (REDC), and are utilized for heavy element research. The isotopes are produced by irradiating aluminum-cladded targets containing a stack of curium oxide pellets with a stack height of about 50 cm and a composition typically consisting of a mixture of plutonium, americium, and curium isotopes as feed material.

About 80% of the world's supply of ^{252}Cf is produced at HFIR (Roberto, 2019). The ^{252}Cf isotope has a half-life of 2.645 years and decays via alpha emission and spontaneous fission (3.09% of the decays are through spontaneous fission). With a high neutron emission rate of $\sim 2.3 \times 10^6$ neutrons $\cdot\text{s}^{-1}\cdot\mu\text{g}^{-1}$, ^{252}Cf is used for a variety of unique purposes, such as treating certain types of brain

and cervical cancers, detecting metal fatigue in aircraft, and university education. In the nuclear industry, it is used to determine fuel enrichment and as a reactor startup source. For national security purposes, it is used to inspect shipping containers and to calibrate detectors. It is also used for prompt gamma neutron activation analysis in the cement, coal, mineral, and power plant industries for process control.

Isotopes produced in HFIR via neutron transmutations have been utilized in research that has led to element and isotope discoveries. Recently, as part of a US-Russian collaborative effort, ^{249}Bk produced in HFIR and isolated by REDC was sent to Russia, where it was bombarded by a calcium ion beam to form element 117. This element was named tennessine because of the efforts from both ORNL and Vanderbilt University, both of which reside in the state of Tennessee. In addition to element 117, ORNL-produced materials and isotopes have been used in the discoveries of superheavy elements 104 (rutherfordium), 105 (dubnium), 106 (seaborgium), 114 (flerovium), 115 (moscovium), 116 (livermorium), and 118 (oganesson). Ongoing research is being performed with ^{248}Cm , which upon bombardment with a vanadium beam, will produce element 119, if successful.

The ^{63}Ni and ^{75}Se isotopes, in high specific activities, are routinely produced in HFIR's flux trap. The ^{63}Ni isotope is used in electron capture technology for national security applications such as detection of explosives and drugs in airports. The ^{75}Se isotope is used for commercial and industrial gamma radiography (e.g., nondestructive testing applications such as pipeline weld inspections). The ^{133}Ba isotope is also produced at HFIR because it is an ideal isotope for calibration of instruments used in the oil and gas industries.

Radioisotope production is also very important for the medical industry for use in treating conditions such as cancer, arthritis, bone metastases, and restenosis. Other US-based research reactors, such as the Missouri University Research Reactor, produce medical radioisotopes; however, HFIR produces high specific activity radioisotopes that cannot be efficiently produced in other reactors. The medical radioisotopes produced at HFIR include ^{14}C , ^{89}Sr , ^{166}Ho , ^{177}Lu , ^{188}W , ^{227}Ac , and ^{229}Th . Additional isotopes are being considered as the medical field surrounding targeted alpha and beta therapies matures. Furthermore, ^{99}Mo research and development relies on HFIR to support aspiring domestic producers to develop their production technologies, to support medical diagnostics and treatments that utilize this radioisotope.

The production of ^{238}Pu takes place in HFIR's beryllium reflector VXF's via irradiation of ^{237}Np in the form of $\text{NpO}_2\text{-Al}$ cermet or NpO_2 oxide pellets contained in target assemblies. Loaded with PuO_2 fuel, radioisotope thermoelectric generators (RTG) convert heat energy produced from the alpha decay of ^{238}Pu (half-life of 87.7 years) into useable electricity and are used as a reliable power source for deep space and planetary National Aeronautics and Space Administration (NASA) missions. An RTG provided electrical power for NASA's Cassini spacecraft and is used to charge the batteries that power the Mars Perseverance and Curiosity rovers and their instruments.

Materials irradiation

The in-core experiment facilities are also used for research on severe neutron and gamma damage to materials. Because of its very high neutron flux environment, HFIR can simulate the full life of radiation exposure for commercial, fission-based reactor materials and components in a reasonable timeframe. HFIR is typically loaded with more than 100 separate experiment capsules and thousands of test specimens of various materials for research of future reactor materials as well as life extensions of current power reactor fleets. Data obtained from the irradiations are used for the qualification of new reactor component materials, including fuel cladding and reactor vessel materials. The HFIR materials irradiation mission also supports fusion energy development. Specifically, areas including neutron interactive materials (structural materials and ceramics), high-heat flux materials, and plasma-interactive materials are supported.

The materials research is enabled by HFIR's ability to accelerate damage testing using the highest fast neutron flux in the Western world, producing up to 2 displacements per atom (dpa) in steel during a single HFIR operating cycle of ~ 25 days. Additionally, a unique irradiation facility allows experimenters to actively monitor and control temperatures inside experiment capsules during irradiation via resistance heaters, variation in sweep gas compositions, and thermocouple measurements.

Neutron activation analysis

Two pneumatic tube facilities are available in the beryllium reflector as illustrated in Fig. 2. Each facility consists of a flight tube, air supply and exhaust lines, a loading station in the NAA laboratory, and irradiation stations. NAA offers a combination of ultrasensitive measurements and fast turnaround for determining the presence of certain isotopes/elements in materials. The facilities offer thermal neutron fluxes of up to about 4×10^{14} neutrons $\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$. Additionally, the neutron spectra in these facilities are very soft (i.e., high thermal-to-total flux ratio). The NAA lab provides irradiation and detection services for a wide variety of programs, including nuclear forensics and criminal forensics. Some of the major applications include research of material impurities, detection of trace quantities of nuclear materials, understanding the nature and quantity of pollution in the environment, and qualifying ultrapure materials used in highly sensitive detectors.

Gamma irradiation

The Gamma Irradiation Facility employs used fuel elements to irradiate experiments with high doses of gamma radiation to study the effects of radiation on materials. With the flexibility to select fuel elements based on their decay time, users can obtain dose rates up to 1.8×10^8 rad $\cdot\text{h}^{-1}$ with used-fuel gamma spectra for rapid materials testing. The facility has been used for a variety of

applications, from testing materials for a nuclear reactor on the moon to research of radiation effects on artificial spinal disc materials. Gamma irradiation research has been used to test the effects of gamma dose on selected resins that are being evaluated for removing ^{137}Cs from high-level waste solutions. The facility has also been used for investigating radiation resistance of circuit boards for lunar reactor environments for NASA, determining radiation-induced changes in the conductivity of high-voltage insulators, and evaluating new radiation-resistant concrete formulations for nuclear power applications.

Fundamental physics

Fundamental physics research is ongoing at HFIR in support of its multimission objective. One example is the Precision Reactor Oscillation and SPECTrum Experiment (PROSPECT), which is a short-baseline reactor antineutrino experiment with the goals of discovering eV-scale neutrinos through oscillation effect observations, testing reactor antineutrino spectrum predictions, demonstrating antineutrino detection techniques, and developing technology for nonproliferation applications (Ashenfelter et al., 2019). HFIR is an ideal test bed for this type of experiment because it has a high-power-density, well-understood compact core with a ^{235}U fission fraction greater than 0.99 throughout the entire fuel cycle.

Conclusion

As a US DOE Office of Science User Facility, the HFIR, located in Oak Ridge, Tennessee, provides world-class neutron science capabilities to serve a variety of science and technology communities. HFIR is a pressurized, light-water-cooled, and beryllium-reflected research reactor that has a unique flux trap-type design. The central flux trap basket is surrounded by a cylindrical fuel assembly consisting of two concentric fuel elements that collectively bear 540 involute-shaped fuel plates. Fueled by 9.4 kg ^{235}U and operated at 85 MW, HFIR provides one of the highest steady-state neutron fluxes of any research reactor in the world. The peak, unperturbed thermal and fast neutron fluxes available in the flux trap are $\sim 2.5 \times 10^{15}$ and 1.4×10^{15} neutrons-cm $^{-2}$ -s $^{-1}$, respectively. The high-power density core, intense neutron flux, and versatile experiment configuration enable world-leading science in mission areas such as cold and thermal neutron scattering, radioisotope production, materials irradiation research, and neutron activation analysis.

HFIR was designed principally to produce heavy isotopes at the highest possible rate in the flux trap; however, the forethought of the reactor designers to include horizontal beam tubes and other experiment facilities in the reflector has enabled the mission capabilities to evolve as the neutron science research landscape has expanded and advanced since HFIR went critical in 1965. The current primary mission of providing neutrons for condensed matter physics research will continue to grow with improvements to HFIR facilities and detection instruments. The ability to damage materials with neutrons grows ever more important as the United States forges ahead with next-generation power reactor designs. The original isotope production mission will also continue to grow due to HFIR's efficient ability to produce high specific activities of desirable medical, industrial, and research isotopes.

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The National Bureau of Standards Reactor at the NIST Center for Neutron Research

Thomas Newton, Daniel Flynn, and Robert Williams, NIST Center for Neutron Research, National Institute of Standards and Technology, Gaithersburg, MD, United States

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Introduction

Neutrons have long been recognized as very valuable tools in probing the structure and dynamics of materials. Their unique properties of selectivity, magnetism, and neutrality make them useful in many diverse fields of science.

The need for a neutron source at the U.S. National Bureau of Standards was recognized and the initial planning for a reactor-based neutron science facility began in March 1958 with a construction permit being issued by the U.S. Atomic Energy Commission in 1963. This permit was converted to Operating License TR-5. The reactor achieved initial criticality on December 7, 1967 and began full-power operation at 10 MW on February 9, 1969. At the time, the US was building a new generation of high-performance research and test reactors: HFIR, HFBR, and ATR which all began operations at about the same time.

The NBSR received a 20-year extension of the existing operating license along with a power increase from 10 to 20 MW on May 16, 1984, and achieving 20 MW for the first time on April 3, 1985. In 1988, the National Bureau of Standards became the National Institute of Standards and Technology (NIST). Because of licensing concerns, it was decided at that time to retain the title of the reactor as the NBSR. There have been some suggestions that the acronym now stands for the 'Neutron Beam Split-core Reactor,' but the official title remains the 'National Bureau of Standards Reactor.' Another license renewal for 20 years of operation was granted on July 2, 2009 (NIST, 2004; Rush and Cappelletti, 2011).

Neutron research has always been the focus of the NIST Center for Neutron Research (NCNR). There have also been efforts in neutron activation analysis, development of standard reference materials, and some isotope production, but the main focus has been on neutron scattering. The NBSR thermal neutron scattering research was very successful and had a number of collaborative efforts in materials research.

The reactor was built with a large thimble for a cold neutron source adjacent to the reactor (shown in Fig. 1), but it was not until 1987 that a D₂O ice cold neutron source was installed, allowing new capabilities for research using longer wavelength neutrons. A guide hall with seven cold neutron guide tubes was completed in 1989, with first neutrons being received in 1990. In 1995, the cold source was upgraded, with the D₂O ice source being replaced by a liquid hydrogen source. This was again upgraded to an advanced liquid hydrogen source in 2002, and, in 2012, the guide hall was enlarged, and a second liquid hydrogen cold source was installed in a separate beam tube. Today, the NCNR has 24 cold neutron instruments, 6 thermal neutron instruments and hosts over 2500 research participants with over 300 research publications annually in the fields of biology, chemistry, condensed matter, engineering physics, measurements, soft matter, neutron physics and geology (NIST Center for Neutron Research, 2018).

Nuclear reactor design

Fuel element

The 1.75 m long NBSR fuel element, shown in Fig. 2, consists of 19 curved plates, of which the inner 17 contain U₃O₈ fuel plus aluminum dispersant with aluminum cladding. In these 17 plates, there are upper and lower fueled regions 33 cm length by 7 cm width with a 17.8 cm gap between upper and lower fuel regions, which allows for highly thermalized flux in the center of the core. Thus, there are a total of 34 individual fuel plates in each element. Each plate contains about 13 g of U₃O₈ and 19 g of Al in the 8.9 cm³ volume available for the fuel meat. Adding curvature to the plate minimizes the effects of heating on the



Fig. 1 Interior view of NBSR beam tubes, including the 55 cm cryogenic thimble, used for cold sources. Courtesy of NIST Center for Neutron Research.

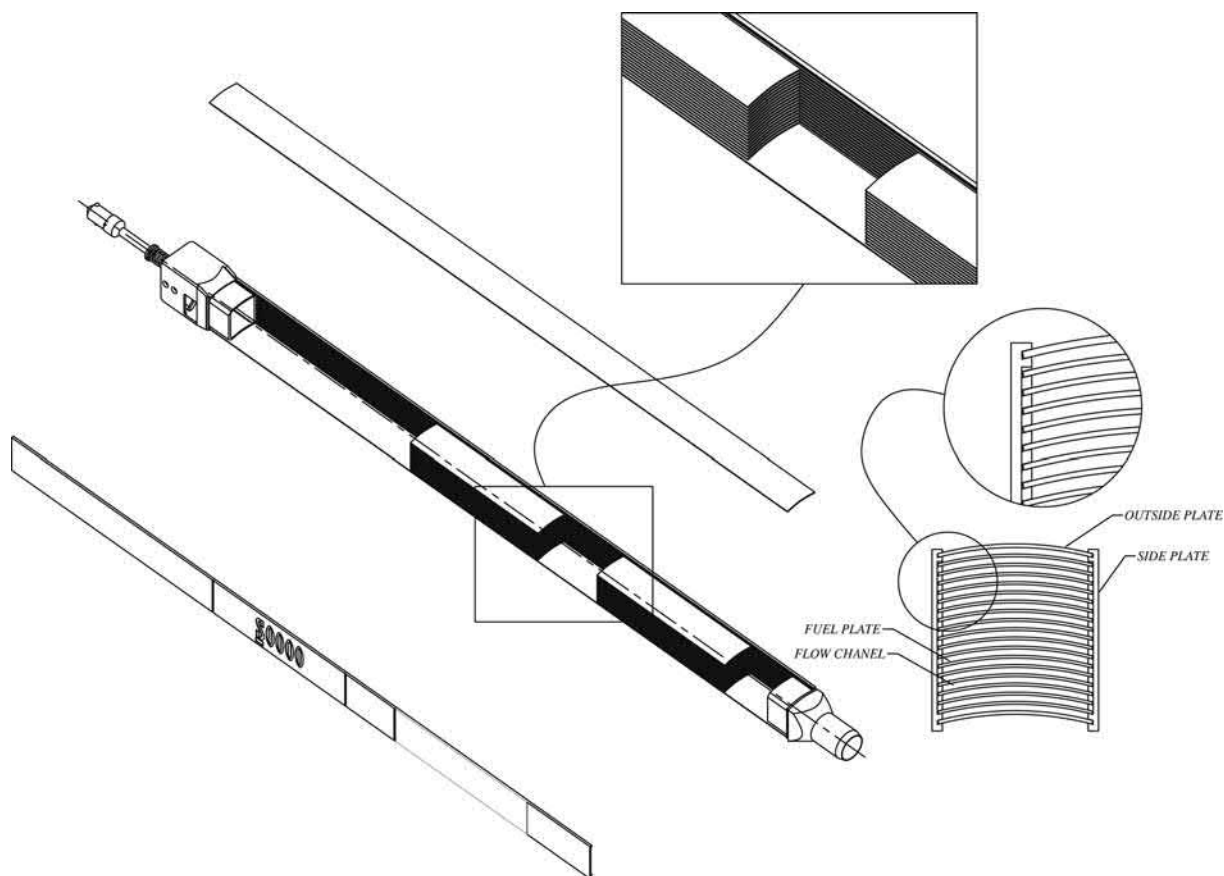


Fig. 2 View of NBSR fuel element, with 17.8 cm central gap. NIST Center for Neutron Research (2004) *Safety Analysis Report for License Renewal for the National Institute of Standards and Technology Reactor (NBSR 14)*. Gaithersburg, MD: U.S. Department of Commerce.

mechanical joints that hold the fuel plates in place in the fuel element assembly. No burnable poisons or neutron moderators are added to the fuel elements. The fuel is enriched to approximately 93% and each element contains about 350 g of ^{235}U (NIST, 2004).

Coolant enters the internal passage of the lower adapter, flows up through the internal conical transition section, through the 18 channels defined by the curved fuel plates and unfueled curved plates of the lower fuel section, into and through the gap, through the 18 channels of the upper fuel section, and then out the upper adapter. The bottom adapters also act as check valves, allowing a small amount of upward coolant flow via hydraulic force against a spring when there is flow, but preventing the draining of the outlet side of the fuel elements if there is no flow.

Nuclear physics parameters

With the reactor operating at 20 MW, the peak thermal flux in the unfueled region at the core midplane is about $3.5 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$, with the thermal neutron flux being over four times the fast neutron flux. Nuclear physics parameters are listed in [Table 1](#) ([NIST, 2004](#)). These provide sufficient available reactivity that allows operation for a 38.5-day fuel cycle, while assuring a robust ability to shut down the reactor under all circumstances.

LEU conversion

Studies have shown that it will be feasible to use monolithic U-10Mo fuel with an enrichment of 19.75% in the same configuration as the existing HEU fuel, with slightly thinner fuel meat thickness. This 20 MW LEU core will have the same fuel cycle length and have almost identical shutdown margins as the HEU core. A Preliminary Safety Analysis Report for the use of this fuel was submitted to the U.S. Nuclear Regulatory Commission (NRC) in 2014 ([Brookhaven National Laboratory, 2014](#)). Conversion of the NBSR awaits qualification of this fuel and approval by the NRC. Thermal flux losses to the beam tubes of about 10% are expected with this conversion. This will be mitigated by the installation of a liquid deuterium cold source, detailed below.

Core configuration

The NBSR core, shown in [Fig. 3](#), is moderated and cooled with D₂O and consists of 30 fuel elements in a 3 concentric ring hexagonal pattern with the inner 2 rings having 6 fuel elements each and the outer ring having the remaining 18 elements. In addition, there are seven experimental thimble positions in a circular pattern about the core center, not currently in use. The center axial fuel gap, mentioned above, allows the installation of large radial beam tubes that surround the reactor and extend right to the edge of the core without looking at significant fast flux.

Running at 20 MW 24/7, the NBSR fuel cycle lasts 38.5 days. At this point, the reactor is refueled by removing four spent fuel elements and adding four new elements. Each fuel element in the reactor is cycled into a new position with each refueling, transitioning closer to the core center with each cycle. After running for either seven or eight cycles, the fuel elements are discharged. As a result of this fuel management scheme, the average ²³⁵U burnup in a 7-cycle element is 65% and the burnup in the 8-cycle elements is 71%, with a typical fission density of $1.9 \times 10^{27} \text{ fissions m}^{-3}$.

Reactivity control and safety systems

The NBSR is controlled by four shim arms, containing 0.2 cm thick cadmium clad with aluminum on both sides. Each shim arm is 2.54 cm thick by 12.1 cm wide with a poison length of 132 cm. Because of the need to minimize shadowing of the beam ports in the core center, movement of the shim arms are in a semaphore-type (angular) direction with an operational travel of 41°. The full-in position is 41° below horizontal. A reactor scram is accomplished by disengaging an electromagnetic clutch which, when energized, holds the shim arm in position against a ball screw jack and compression spring. Release of the clutch will result in complete shim arm insertion in less than 1 s. Neutron absorption by the shim arms is sufficiently large so that the reactor could be shut down by any one of them. Lowering the level of the D₂O in the vessel is an additional mechanism for emergency shutdown and provides sufficient negative reactivity to make the core subcritical even with all four shim arms withdrawn ([NIST, 2009](#)).

In addition to the shim arms, there is a fine-tuning regulating rod in the core, consisting of a solid aluminum cylinder 6.4 cm in diameter by 74 cm long. Displacement of the heavy water by aluminum and neutron absorption provides sufficient fine-tuning control.

Cooling systems

A simplified diagram of the D₂O primary system is shown in [Fig. 4](#). There is a total of four primary pumps (three normally in use) which drive a flow of about 560 l s⁻¹ into the main heat exchangers. Then the D₂O flows through a strainer to two inlet plena for the

Table 1 Summary of NBSR Nuclear Characteristics.

<i>Moderator and Coolant</i>	<i>D₂O</i>
Reactor power	20 MW
Reactivity, clean, cold, all shims withdrawn	6.91% ΔK/K
Total reactivity all shim arms	− 23.7% ΔK/K
Temperature coefficient of reactivity	− 0.029 ΔK/K/°C
Effective delayed neutron fraction	0.00757
Prompt neutron lifetime	800 μs

NIST Center for Neutron Research (2004) *Safety Analysis Report for License Renewal for the National Institute of Standards and Technology Reactor (NBSR 14)*. Gaithersburg, MD: U.S. Department of Commerce.

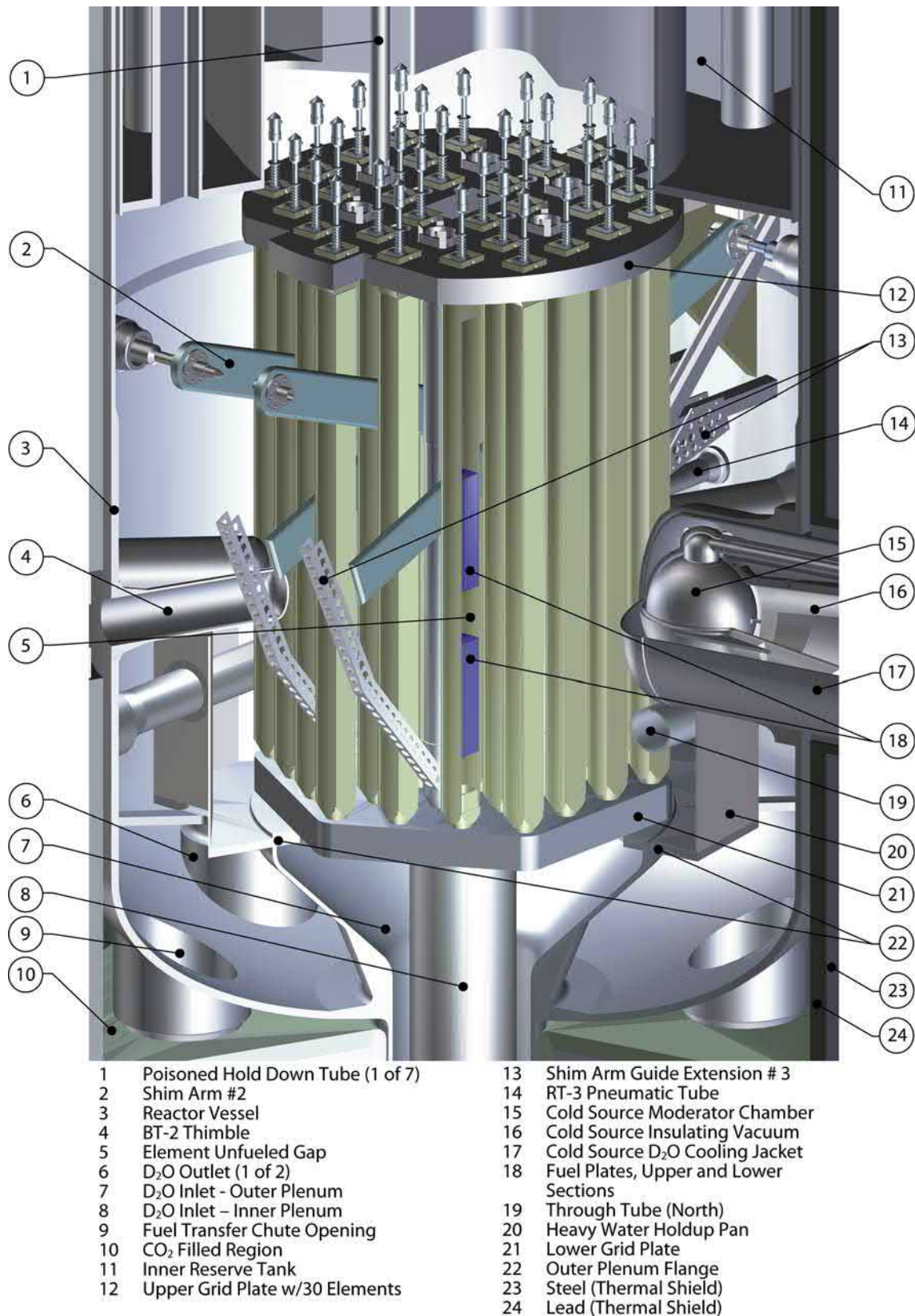


Fig. 3 Diagram of the NBSR core. Courtesy of NIST Center for Neutron Research.

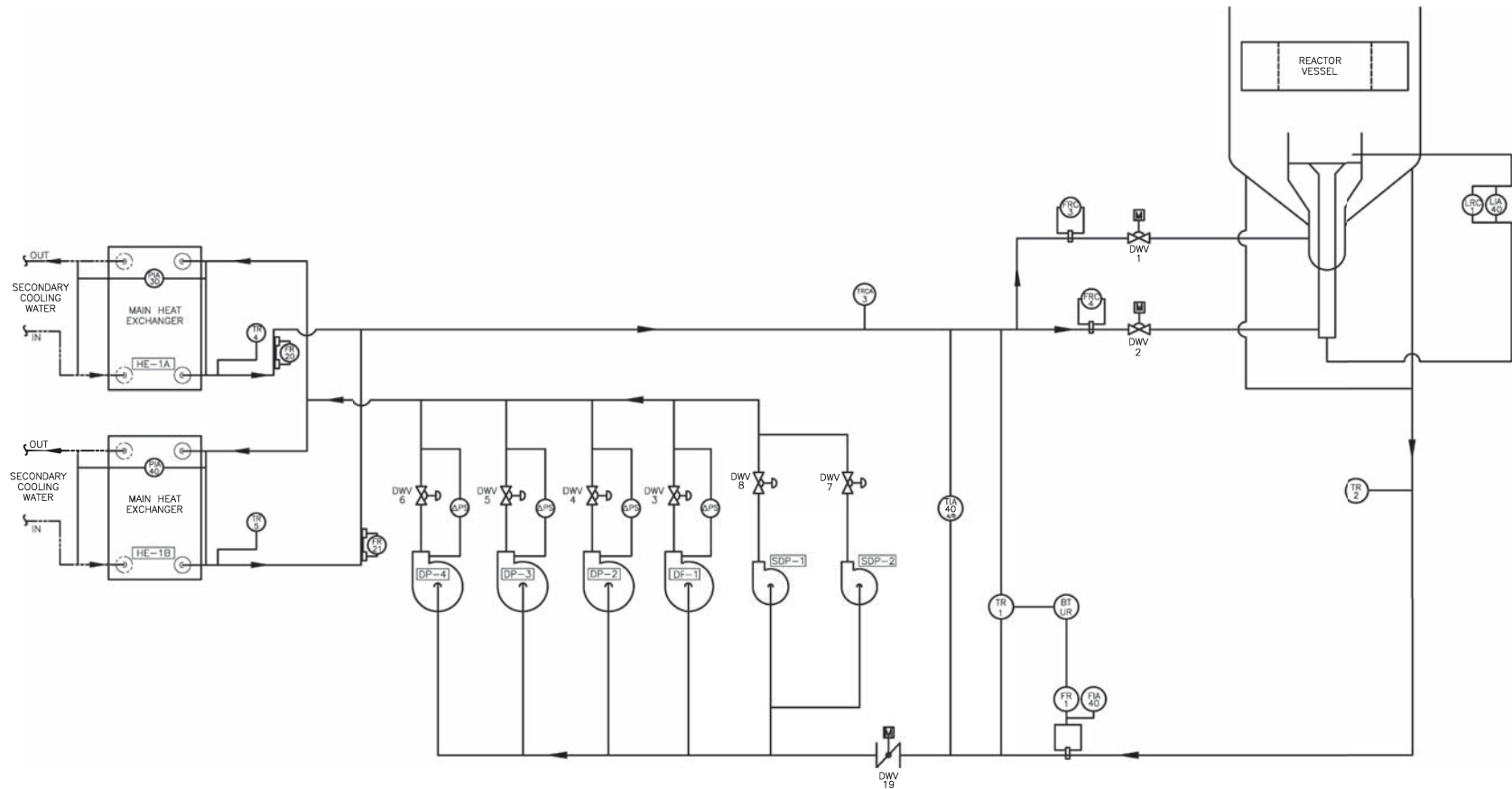


Fig. 4 NBSR D₂O primary coolant system. NIST Center for Neutron Research (2004) *Safety Analysis Report for License Renewal for the National Institute of Standards and Technology Reactor (NBSR 14)*, Gaithersburg, MD: U.S. Department of Commerce.

reactor vessel. The plena direct water up through the vessel into the core, with the flow being divided into an inner plenum of the inner six fuel elements (140 l s^{-1}), and the outer plenum of the remaining 24 elements (420 l s^{-1}). The coolant exits the fuel elements via the open top of the elements, then flowing to the bottom of the vessel where it exits through two outlet pipes. These are joined to a single outlet header containing a venturi for outlet flow measurements. The outlet header joins the suction header of the main circulating pumps, completing the cooling cycle. A closed helium sweep system is used to prevent degradation of heavy water isotopic purity and to minimize diffusion of tritiated heavy water vapor to the confinement building atmosphere as well as removing gases formed by radiolysis of the primary coolant and recovering any heavy water lost due to evaporation.

The D_2O system also contains a purification loop, an experimental cooling loop for cooling the cold source cryostats, and also supplies reserve D_2O to external and internal emergency cooling tanks. The main heat exchangers are located in the process room in the basement; the entire D_2O system is within the confinement building.

A thermal shield is installed just outside the reactor vessel, consisting of a 5 cm thick lead shield, which absorbs gammas from the core, preventing thermal damage to the concrete biological shield. The thermal shield is cooled by a series of 188 water (H_2O) cooling lines imbedded in the lead. Heat is transferred to the secondary cooling system via a plate-and-frame heat exchanger.

The secondary system removes heat from the primary coolant systems as well as several ancillary systems. System flow is generated by three of four main secondary pumps, with a nominal flow of about 660 l s^{-1} . The heat load assumed by the secondary coolant is transferred to the atmosphere via two cooling towers, a hybrid wet/dry tower with three cells and a smaller tower with two cells erected to provide additional cooling capacity. The secondary pumps and several additional auxiliary pumps are located in a separate building adjacent to the cooling towers.

Experiment systems

The layout of NCNR neutron instruments is shown in Fig. 5. There are five thermal neutron scattering instruments and one thermal radiography instrument (BT2), labeled BT1 through BT8, whose beams are aligned with the D_2O (non-fueled) gap between the upper and lower fuel sections of the core. In addition, there is a single instrument which uses cold neutrons from a small liquid hydrogen cold source in BT9. The remaining 23 neutron scattering instruments, listed in Table 2, all use cold neutrons emanating from a liquid hydrogen cold source, which can be seen in Fig. 3.

The liquid hydrogen cold source consists of a cryostat with a 3 cm annulus of liquid hydrogen. This is connected to a helium-cooled condenser mounted on the face of the reactor above the cryostat. Liquid hydrogen flows from the condenser to the cryostat via a natural convection thermosiphon loop wherein hydrogen vapor produced in the cryostat moves up to the condenser, completing the cycle.

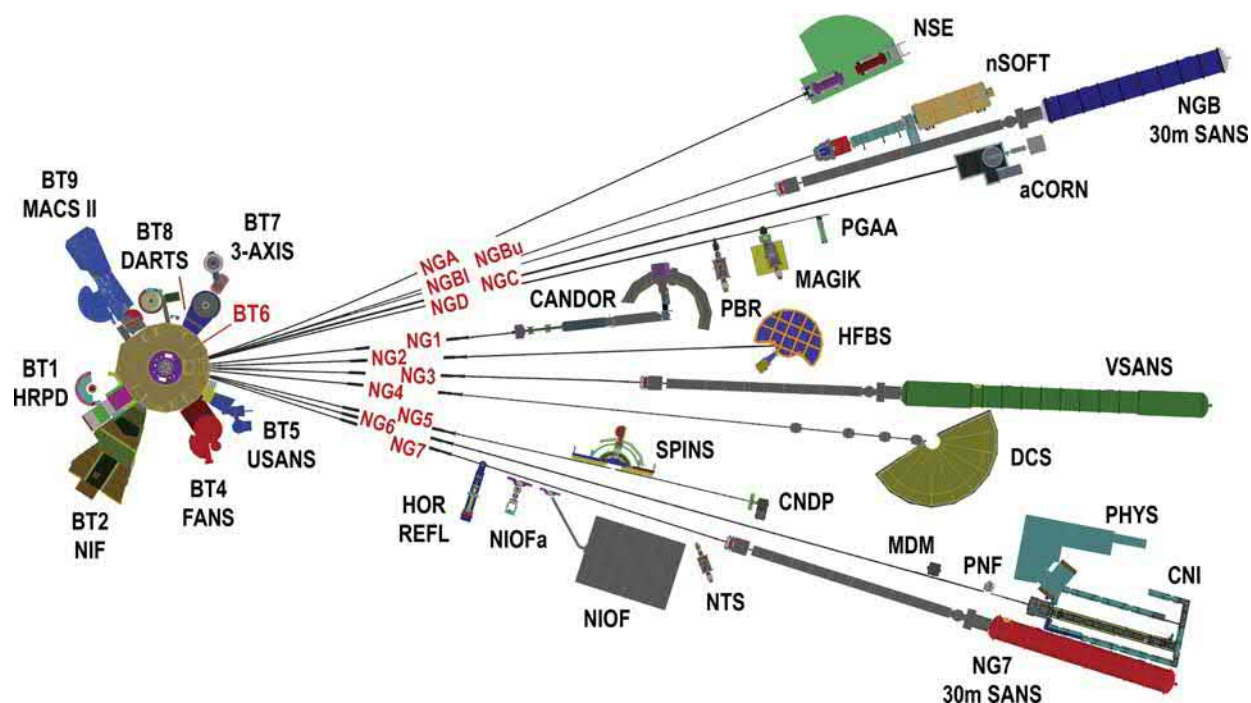


Fig. 5 Layout of NCNR neutron instruments. NIST Center for Neutron Research (2018) 2018 NIST Center for Neutron Research Accomplishments and Opportunities. Gaithersburg, MD: National Institute of Standards and Technology.

Table 2 NCNR Neutron Instruments.

BT-1 HRPD	High resolution powder diffractometer
BT-2 NIF	Thermal Neutron Imaging Facility
BT-4 FANS	Filter Analyzer Neutron Spectrometer
BT-5 USANS	Ultra Small Neutron Spectrometer
BT-7 3-Axis	Triple Axis Spectrometer
BT-8 DARTS	Residual Stress Diffractometer
BT-9 MACS-II	Cold Neutron Multi Axis Crystal Spectrometer
NGA NSE	Neutron Spin Echo Spectrometer
NGBu nSOFT	10 m Small Angle Neutron Scattering
NGBI	30 m Small Angle Neutron Scattering
NGC ^a	NIST Beam Neutron Lifetime Experiment
NGD PGAA ^b	Cold Neutron Prompt Gamma Activation Analysis
NGD MAGIK	Off-specular reflectometer
NGD PBR	Polarized Beam reflectometer
NG1 CANDOR	Chromatic Analysis Neutron Diffractometer or Reflectometer
NG2 HFBS	High Flux Backscattering Spectrometer
NG3 VSANS	Very Small Angle Neutron Scattering
NG4 DCS	Disc Chopper Time-of-Flight Spectrometer
NG5 SPINS	Spin-Polarized Triple-Axis Spectrometer
NG5 CNDP ^b	Cold Neutron Depth Profiling
NG6 MDM ^a	Precision measurement of the neutron's Magnetic Dipole Moment
NG6 NF ^a	Precision measurement of neutron flux
NG6 CNI ^a	Cold Neutron Imaging Facility
NG7 SANS	30 m Small Angle Neutron Scattering
NG7 NTS	PHADES Neutron Test Station
NG7 NIOF ^a	Neutron Interferometry and Optics Station
NG7 NIOFa ^a	Neutron Physics Interferometry Test Bed
NG7 HOR REFL	Horizontal Sample Reflectometer

^aPhysics.^bChemistry.

NIST Center for Neutron Research (2018) *2018 NIST Center for Neutron Research Accomplishments and Opportunities*. Gaithersburg, MD: National Institute of Standards and Technology.

Replacement of the cold source with a liquid deuterium cold source is planned for 2023. This will result in a gain of up to a factor of 2 for long wavelength/low energy neutrons ($> 5 \text{ \AA}$ or $< 4 \text{ meV}$) and is intended to mitigate the effects of the thermal neutron flux loss upon conversion of the core to LEU fuel. The proposed liquid deuterium cold source will contain a volume of about 35 l of liquid deuterium, compared with about 5 l in the current liquid hydrogen cold source. This is shown in Fig. 6. This will result

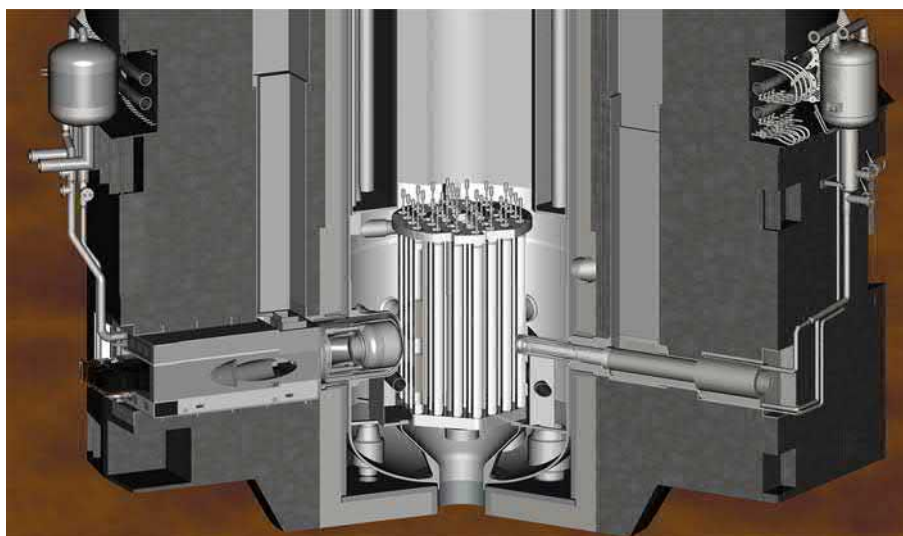


Fig. 6 NBSR core, showing large proposed liquid deuterium cold source (left) and small liquid hydrogen cold source (right). Courtesy of NIST Center for Neutron Research.

in the cold source heat load increasing from about 3 kW to about 6 kW. A new 7 kW helium refrigeration system was installed in 2018 in preparation for the upcoming deuterium cold source installation and eventual conversion to LEU.

Most of the beam time on NCNR instruments is made available through a peer-review proposal process. The NCNR issues calls for proposals approximately twice a year and proposals are reviewed at several different levels by both outside reviewers, NCNR staff, and a Beam Time Allocation Committee.

There are several sponsors of neutron instruments at NCNR, including the Center for High Resolution Neutron Scattering (CHRNS), which is jointly funded by the U.S. National Science Foundation. CHRNS supports a portfolio of neutron instruments at NCNR and provides capabilities for a very diverse scientific program in material science, chemistry, biology, geosciences and condensed matter physics. CHRNS also supports a variety of educational programs and activities, including a summer school on “Methods and Applications of Neutron Spectroscopy” for graduate and post-doctoral students ([NIST Center for Neutron Research, 2018](#)).

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Relevant websites

- www.ncnr.nist.gov—NIST Center for Neutron Research.
- www.nist.gov—National Institute of Standards and Technology.
- https://ws680.nist.gov/publication/get_pdf.cfm?pub_id=908047—Informal History of the NBSR.

The Annular Core Research Reactor (ACRR) Description and Capabilities[☆]

Edward J. Parma^a, Michael W. Gregson^b, R. David Clovis^b, Lance Lippert^b, and Lonnie E. Martin^b, ^aEden Radioisotopes, LLC, Albuquerque, NM, United States; and ^bSandia National Laboratories, Albuquerque, NM, United States

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Glossary

Reactivity A measure of the deviation of the reactor core from a critical state in which the rate of neutron generation equals the rate of neutron loss; that is, the fission rate is constant and the so-called effective multiplication factor, k_{eff} , is 1.0. Reactivity is sometimes measured in units of $\$$'s, where a $\$1$ increase in reactivity implies that the chain reaction can be sustained by only the prompt neutrons; that is $k_{\text{eff}} = 1 + \beta$ where β is the fraction of the delayed neutrons (emitted from certain fission products). For ^{235}U $\beta = 0.0064$.

Abbreviations

ACRR Annular Core Research Reactor
Al Aluminum
B Boron
B₄C Boron carbide
Bucket A hollow right-circular cylinder spectrum modifier
Cd Cadmium
eV Electron volt
Flux Number of neutrons crossing a unit area per unit time
Fluence Time integrated flux
FREC-II Fueled Ring External Cavity—version 2
FWHM Full-width at half-maximum
Gy Gray—unit of absorbed dose equal to 100 rad
keV kilo-electron volt
 k_{rad} Kilorad—unit of absorbed dose equal to 1000 rad
LB44 A 44-in.-tall, lead-boron lined right circular cylinder
MCNP Monte Carlo N-Particle
MeV Mega-electron volt
MJ Megajoule
ms Milliseconds
MW Megawatts
 n/cm^2 Units of fluence (neutrons per cm^2)
Pb lead
PCD Photo-conducting detector
PLG Polyethylene-lead-graphite

[☆]Change History: October 2020. EJ Parma, MW Gregson, RD Clovis, L Lippert, LE Martin, 2/29/2020: Added additional author to chapter; uploaded figures 6, 7, and 15 to EMSS. 8/20/20: Deleted seven figures and associated text. Deleted 2 references, added 1 reference. 9/02/20: Added 1 figure.

Poly Polyethylene
 rad Unit of absorbed dose equal to 100 erg/g in a specified material
 SNL Sandia National Laboratories
 SPND self-powered neutron detector
 TA-V Technical Area V
 TLD Thermoluminescent dosimeter
 TRIGA Training, Research, Isotopes, General Atomics
 UO₂-BeO Uranium dioxide—beryllium oxide
 UZrH Uranium zirconium hydride

Introduction

The Annular Core Research Reactor (ACRR) is a unique, one-of-a-kind nuclear reactor facility operated by Sandia National Laboratories (SNL) for the National Nuclear Security Administration that is considered by many to be a national treasure. There is no other research reactor in the world that has the attributes and capabilities comparable to that of the ACRR. The ACRR was specifically designed to meet the irradiation testing needs of the U.S. nuclear defense program. The ACRR has four major attributes that make it unique: (1) a large (9", 22.86 cm) dry central cavity; (2) an epithermal neutron flux; (3) a large pulsing capability; and (4) a fueled-ring external cavity with a larger (20", 50.8 cm) dry cavity.

The ACRR is typically used to perform irradiation testing when a high neutron fluence is required over a short period of time. Historically, the ACRR has been used for a wide variety of experiment campaigns in addition to supporting its main mission—irradiation services of components for the nuclear defense program. Some of these campaigns have included radiation damage in materials testing, nuclear pumped laser testing, nuclear fuels testing, space nuclear thermal propulsion testing, and medical isotopes production.

The ACRR is located in Technical Area V (TA-V) at Sandia National Laboratories in Albuquerque, New Mexico. The ACRR is currently fully operational. The reactor, in its current fueled configuration, was assembled in 1978 to accommodate large experiments at the center of its core and have large pulsing capabilities. The fuel elements for the ACRR are similar in size and shape to TRIGA fuel (Reese, 2021). However, the fuel is unique in that the form of the fuel is uranium dioxide/beryllium oxide (UO₂-BeO) that was specially designed to have a large heat capacity in order to allow for larger pulsing capabilities as compared to other TRIGA-like pulsed reactors. This type of fuel, containing beryllium oxide, is no longer capable of being produced anywhere in the world.

ACRR description

The ACRR can operate in a pulse or steady-state mode. Most customers desire the pulsing capabilities of the reactor. This has allowed the ACRR to have no significant fuel burnup over its 40 years of use. Under the same operating conditions, the reactor fuel is expected to last at least another 40 years. Fig. 1 shows the ACRR looking into the pool during a 2 MW steady-state power operation. The ACRR core is shown on the left in the figure. The 9-in. (22.86 cm) diameter dry cavity extends from above the pool through the center of the core. The reactor facility also accommodates the fueled ring external cavity-II (FREC-II), shown on the right in the figure, which maintains a larger 20-in. (50.8 cm) diameter dry cavity and uses uranium/zirconium-hydride (UZrH) TRIGA fuel as a subcritical multiplier. As currently configured, the FREC-II is not capable of achieving nuclear criticality. FREC-II provides experimenters with a larger test volume, and the fuel arrangement limits the neutron fluence gradient across the test volume. The TRIGA fuel used in FREC-II was originally used in ACRR's predecessor core, the Annular Core Pulse Reactor.

The dry central cavity extends from above the pool and goes directly through and below the core region. Another view of the ACRR and FREC-II is shown in Fig. 2, with the reactor shut down, FREC-II (on the left) tilted back in its "decoupled" configuration, and the radiography tube on the right.

The ACRR core is located in an open pool 10 ft (3.0 m) in diameter and 28 ft (8.5 m) deep. The pool is filled with 64,000 L of deionized water. The core is cooled by natural convection of the pool water. The pool water is cooled by a heat exchanger and cooling tower. For steady-state mode operations, the ACRR operates continuously up to 4 MW. The pool heat rejection system is rated to support steady-state operations up to 5 MW.

The ACRR is fueled by a 236-element array of UO₂-BeO fuel elements. The fuel is uranium enriched to 35 wt% U-235, with 21.5 wt% UO₂ and 78.5 wt% BeO. The ACRR fuel elements are stainless steel clad, 1.5 in. (3.8 cm) in diameter and 21 in. (53.34 cm) in fuel length. Within the elements are niobium cups that hold the UO₂-BeO fuel pieces. The ACRR is controlled by two fuel-followed safety rods, three poison (void-followed) transient rods, and six fuel-followed control rods. The control rods (safety and control) make up part of the 236 elements for the normal core configuration.

- 236 UO_2 -BeO fueled elements
1.5 in (3.8 cm) dia. x 20.5 in (52 cm)
100 g U-235 per element – 35% enriched
- Operating Power level
2-4 MW_{th} Steady-State Mode
300 MJ Pulse Mode (6 ms FWHM)
300 MJ Transient Mode (Programmable)
- Dry cavity 9 in (23 cm) diameter
Extends full length of pool through core
Neutron Flux $4\text{E}13 \text{ n/cm}^2\text{-s}$ at 2 MW
Neutron Fluence $6\text{E}15 \text{ n/cm}^2$ at 300 MW
90% > 1 eV, 58% > 10 keV, 46% > 100 keV
- Epithermal/Fast Spectrum
Flux in cavity can be tailored for desired energy spectrum (Poly, B4C)
- Open-pool type reactor
Core cooled by natural convection
Pool cooled by HX and cooling tower
- FREC-II uses previous ACPR fuel
TRIGA type (UZrH)
Dry cavity 20 in (51 cm) diameter
- Fuel burnup is minimal
- Reactor used for short duration power runs, pulses, and transients

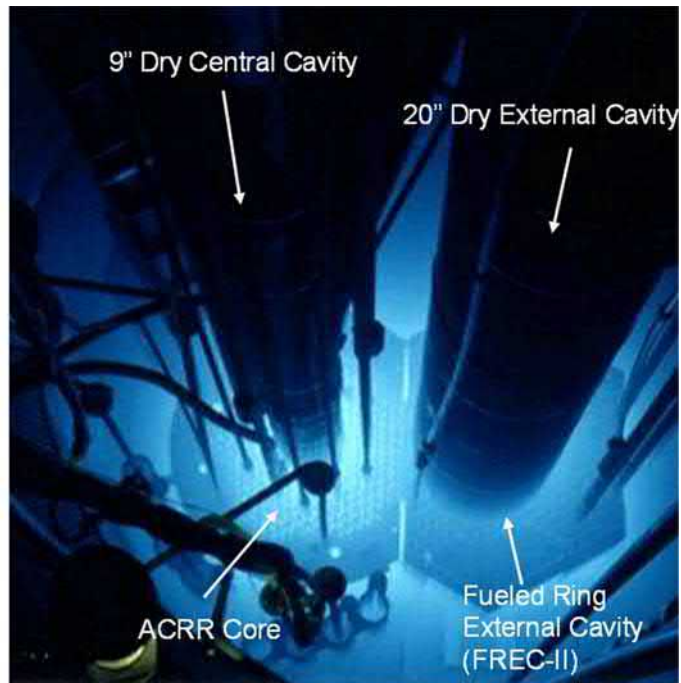


Fig. 1 The ACRR and FREC-II operating at 2-MW steady-state power.



Fig. 2 ACRR with FREC-II decoupled.

The ACRR central cavity and FREC-II cavity allow for a high degree of experiment flexibility, along with in situ and real-time experiment instrumentation and diagnostics. The large volume of the cavities allows for the possibility of acceleration/deceleration (High-G) apparatus, flow loops and other complex experimental hardware to be fielded within the high-fluence region of the core. Because the ACRR is under-moderated, the neutron spectrum has a large epithermal component within the central cavity. This epithermal spectrum can be modified using thermal absorbers to harden the spectrum or by the use of moderator materials to give a softer spectrum. The ACRR can operate in a steady-state, transient, or pulse mode. In the steady-state mode, the operating power level is limited to $\sim 4 \text{ MW}$. In the pulse mode, a maximum pulse yield of $\sim 300 \text{ MJ}$ (resulting from reactivity insertion of $\$3.50$) with a full-width half-maximum (FWHM) of 6.8 ms can be attained. In the transient mode, the reactor power shape can be tailored to the desired requirements for a total reactor energy yield of $\sim 300 \text{ MJ}$. The transient mode can be used to increase the reactor power quickly, as in a linear ramp increase in power levels from low to high power level.

Cavity spectrum modification capability

The ACRR maintains an epithermal neutron fluence spectrum in the core and central cavity. This allows for the neutron fluence energy spectrum to be tailored to the desired specifications of an experiment. Moderators can be used within the cavity to thermalize the neutron spectrum. These spectrum modifiers are referred to as “buckets.” Polyethylene or water can be added to a bucket environment in an annulus geometry to increase the thermal neutron population. Boron and lead can be used to increase the fast neutron fluence ratio and decrease the gamma-ray fluence, respectively. Boron can be added to a bucket environment in an annulus geometry to remove the thermal neutron population. Gamma rays can be attenuated or increased by adding lead or cadmium, respectively, in an annulus geometry. These perturbations allow for the neutron to gamma ray ratio to be modified in the cavities, adding to the versatility and flexibility of the ACRR to meet the customer’s radiation testing requirements.

In addition to the free-field irradiation condition (no bucket environment) of the ACRR and FREC-II cavities, several buckets exist and are available for use by the experimenter to allow for neutron and gamma-ray spectrum modification. New buckets can also be built and tested as future needs are required by the experimenters. The irradiation space within each bucket is thoroughly characterized prior to use in order for the experimenter to have the utmost confidence in the neutron and gamma ray irradiation conditions. Characterization involves both MCNP modeling and experimental testing (Parma et al., 2013, 2015a,b, 2016a,b, 2017). ACRR test measurements in the irradiation volume includes using neutron activation analysis of foils and pellets, irradiation of thermoluminescent dosimeters (TLDs), and active measurements using calorimeters, photo-conducting diodes (PCDs), self-powered neutron detectors (SPNDs), and fission chambers. The neutron energy spectrum is adjusted using least-squares fitting techniques with activation foil measurements to produce the characterized neutron spectrum at the irradiation location within a bucket.

Figs. 3–7 show the neutron energy spectrum and the prompt gamma energy spectrum for the ACRR cavity free field, PLG bucket, LB44 bucket, Cd-Poly bucket, and the FREC-II cavity free field axial centerline positions. Fig. 8 shows simplified drawings of some spectrum modifiers. As previously discussed, the use of polyethylene, boron, cadmium, and lead can significantly alter both the neutron spectrum and the gamma-ray spectrum and magnitude. The PLG bucket enhances the thermal neutrons and reduces the gamma-ray intensity as compared to the free field case. LB44 removes the thermal and some of the epithermal neutrons and reduces the gamma-ray intensity as compared to the free field case. Cd-Poly removes the thermal neutrons and enhances the gamma-ray intensity. The FREC-II cavity has a greater thermal neutron component as compared to the ACRR cavity.

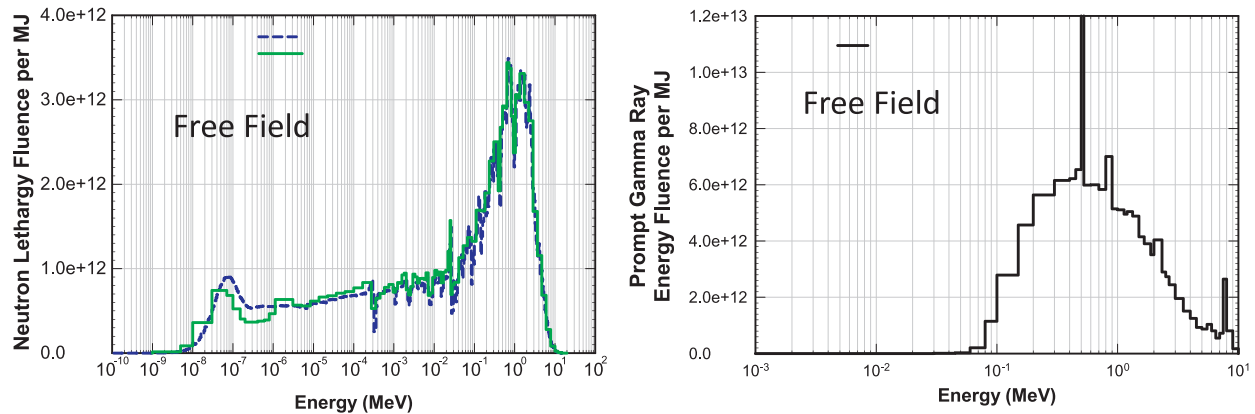


Fig. 3 Neutron and prompt gamma-ray energy spectra for the central cavity free-field environment.

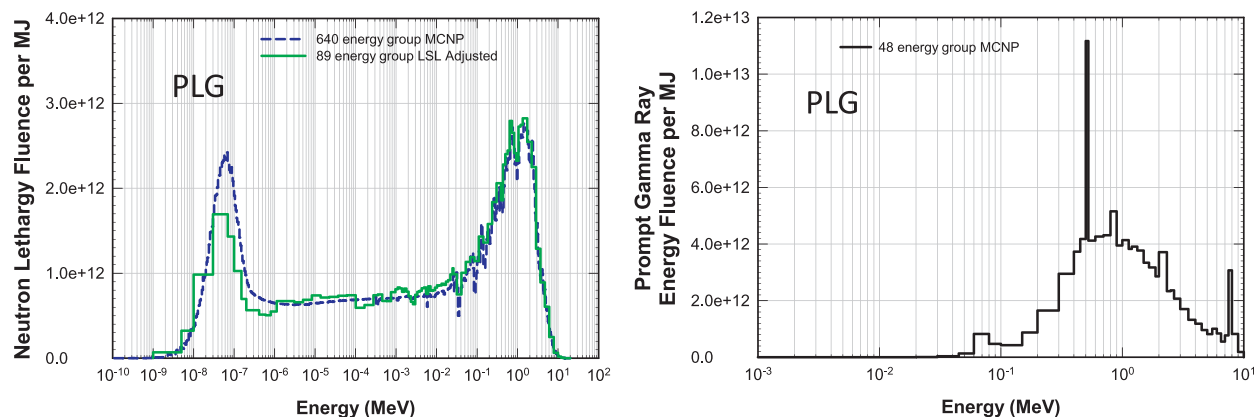


Fig. 4 Neutron and prompt gamma-ray energy spectra for the polyethylene-lead-graphite (PLG) bucket environment.

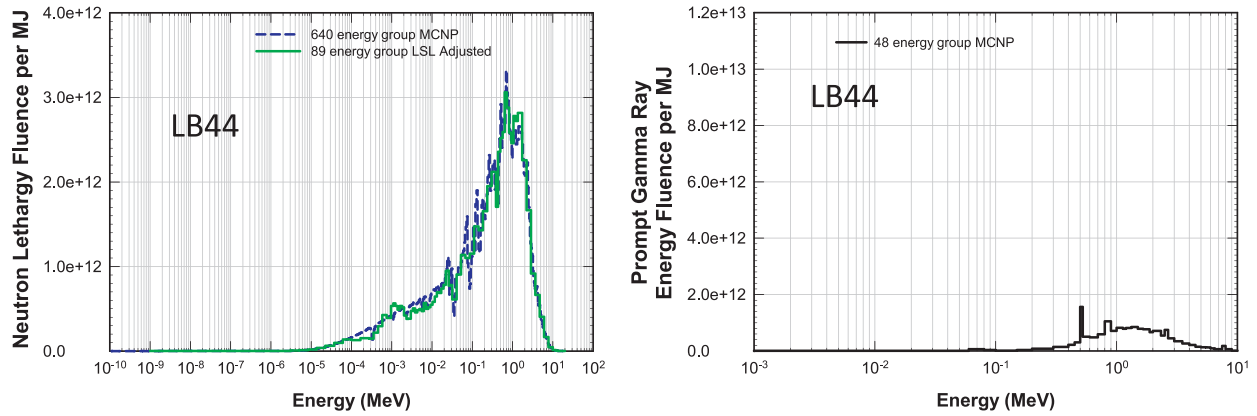


Fig. 5 Neutron and prompt gamma-ray energy spectra for the Lead-Boron (LB44) bucket environment.

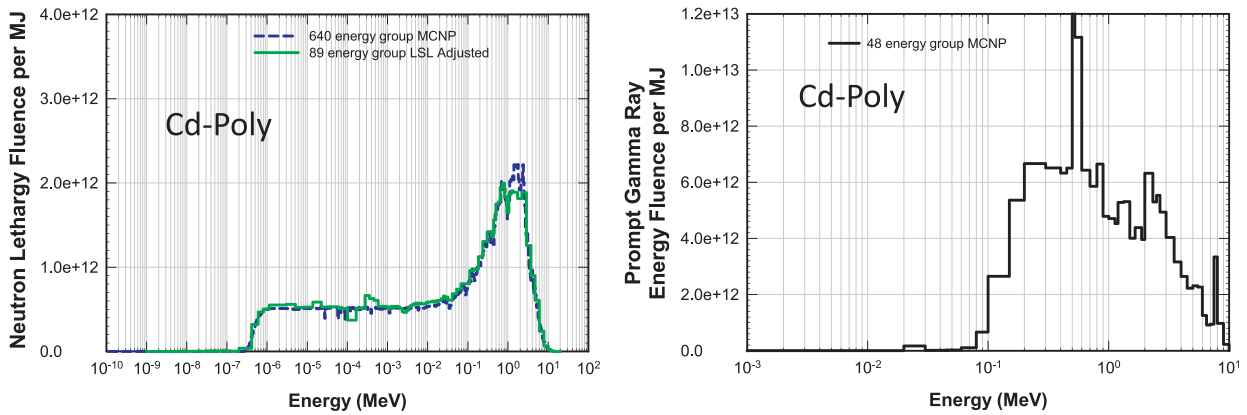


Fig. 6 Neutron and prompt gamma-ray energy spectra for the Cadmium-Polyethylene (Cd-Poly) bucket environment.

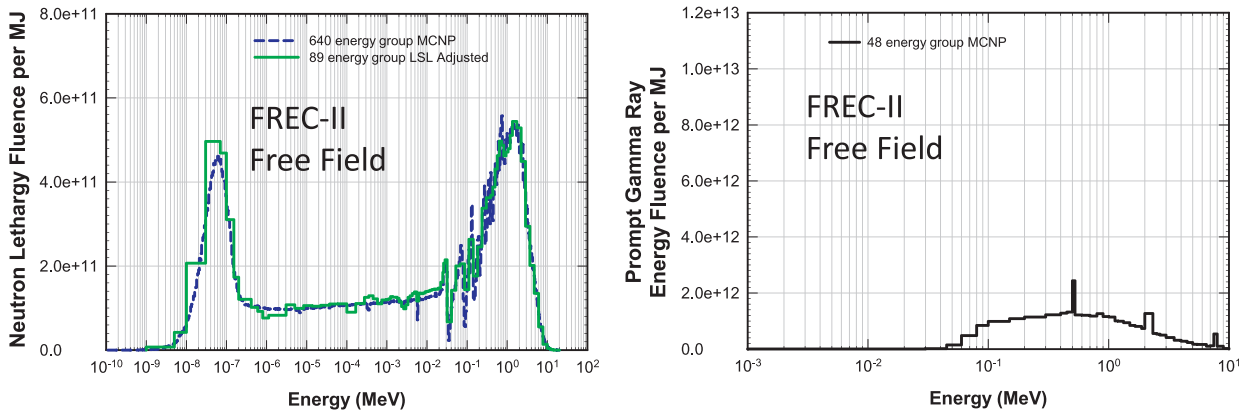


Fig. 7 Neutron and prompt gamma-ray energy spectra for the FREC-II cavity free-field environment.

Table 1 shows some of the important metrics associated with the bucket environments that are of interest to experimenters. These metrics include the total neutron fluence, the 1-MeV damage equivalent in silicon (DES) neutron fluence, the prompt gamma-ray absorbed dose in silicon (Si), and the ratio of the 1-MeV DES neutron fluence to the prompt gamma-ray dose. From this table the largest 1-MeV DES neutron fluence attainable is $\sim 8.2\text{E}12 \text{ n/cm}^2$ per MJ. The largest prompt gamma-ray dose is $\sim 11 \text{ k}_{\text{rads}}$ (Si) per MJ. The 1-MeV DES fluence to gamma-ray dose can be varied by about a factor of 13, with the Cd-Poly bucket having the smallest ratio and the LB44 bucket having the largest ratio.

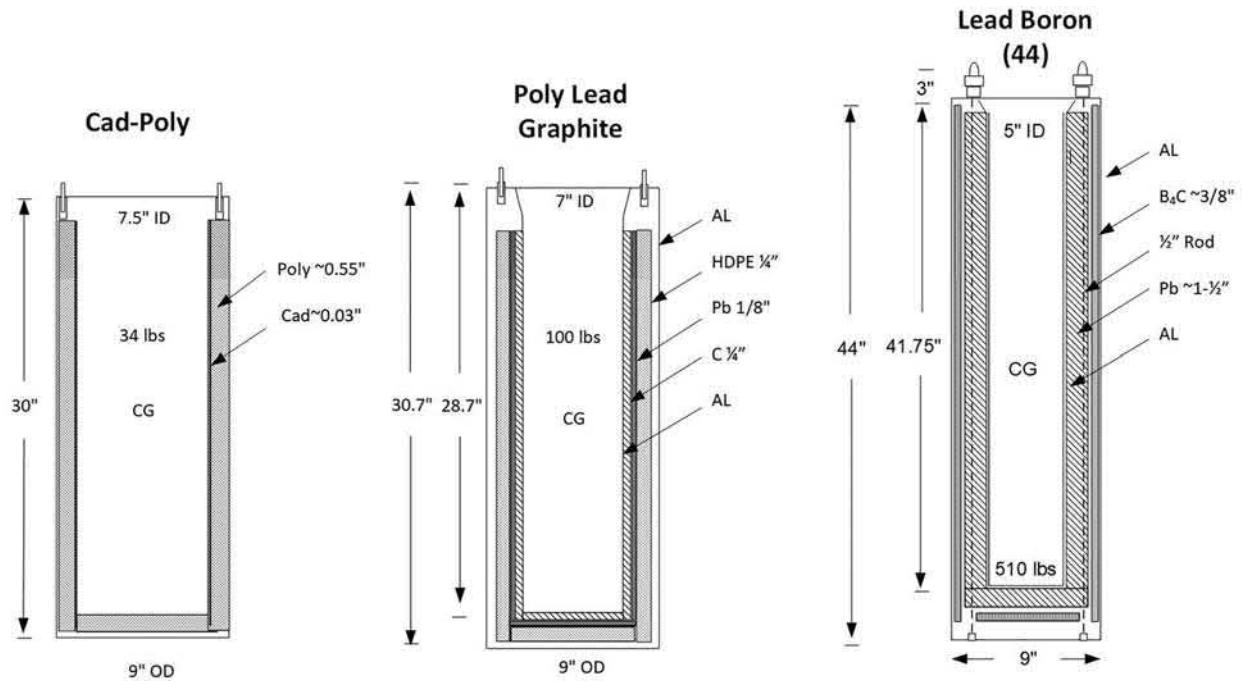


Fig. 8 ACRR spectrum modifiers.

Table 1 Neutron fluence and prompt gamma-ray dose for the characterized environments.

Bucket Environment	Neutron Fluence Total (n/cm^2 -MJ)	Neutron Fluence 1-MeV DES (n/cm^2 -MJ)	Prompt Gamma-Ray Dose in Si (k_{rad}/MJ)	Ratio of 1 MeV DES Fluence to Gamma-Ray Dose (Si) $\times 1E+12$
Central Cavity Free Field	$2.05E+13$	$8.19E+12$	7.94	1.03
PLG	$2.12E+13$	$6.90E+12$	6.10	1.13
LB44	$1.15E+13$	$6.47E+12$	0.98	6.60
Cd-Poly	$1.26E+13$	$5.24E+12$	11.05	0.47
FREC-II Free Field— FREC Rods Up	$3.98E+12$	$1.34E+12$	1.71	0.78

Table 2 Neutron Fluence and Prompt Gamma-Ray Dose for Free Field ACRR and FREC-II Cavities for a 300 MJ Maximum Pulse.

Environment	Neutron Fluence Total (n/cm^2)	Neutron Fluence 1-MeV DES (n/cm^2)	Prompt Gamma-Ray Dose in Si (k_{rad})
Central Cavity Free Field	$6.15E+15$	$2.46E+15$	$2.38E+03$
FREC-II Free Field - FREC Rods Up	$1.19E+15$	$4.02E+14$	$5.13E+02$

Table 2 shows the neutron fluence and gamma-ray dose for a 300 MJ pulse for the ACRR and FREC-II free-field environments. For a maximum pulse, the total neutron fluence in the ACRR central cavity is $6.15E15$ n/cm^2 , the 1-MeV DES neutron fluence is $2.46E15$ n/cm^2 , and the prompt gamma-ray dose in Si is $2.38E3$ k_{rad} s. For a 300 MJ pulse, the peak power level attainable is $\sim 50,000$ MW, corresponding to a neutron flux of $1.0E18$ n/cm^2 -s.

Pulsing capabilities

Pulse mode capability

The ACRR has unique pulse mode capabilities, significantly greater than other pool-type pulse reactors. The three pulse rods can be simultaneously or sequentially ejected from the core region using a pneumatic ejection system, within ~ 80 ms. A maximum pulse at the ACRR is a reactivity addition of $\$3.50$, producing a pulse width of 6.8 ms, a peak power of $\sim 50,000$ MW, and a total reactor

energy deposition of 300 MJ. This is at least three times larger than what TRIGA-type reactors can attain. Smaller pulses, as low as 10 MJ, can also be attained with the ACRR. Fig. 9 shows the experimental results for several different size pulses for the ACRR. The full-width half-maximum (FWHM) for a pulse is directly proportional to the neutron generation time for the reactor, and inversely proportional to the reactivity added. For reference, a 300 MJ pulse yield from a reactivity insertion of \$3.50 would produce a FWHM pulse width of ~ 7 ms; a 100 MJ pulse (\$2.05) would have a pulse width of ~ 14 ms; and a 50 MJ pulse (\$1.40) would have a pulse width of ~ 32 ms. With spectrum modifying buckets inserted in the central cavity, the pulse characteristics do not change significantly because the neutron generation time does not change appreciably. With FREC-II coupled to the ACRR and the FREC rods in the “down” position, the pulse characteristics are also similar. With the FREC-II coupled to the ACRR and the FREC rods in the “up” position, the neutron generation time increases to ~ 32 μ s, from ~ 24 μ s for the ACRR with FREC-II decoupled, and the pulse width increases proportionately.

Transient mode capability

The ACRR also has unique transient mode capabilities. The three pulse rods can be driven out of the core using stepper motors as opposed to pneumatically ejected. Under these conditions, up to \$4.25 of reactivity can be inserted incrementally over time. Power ramps and steady-state transients can be generated for up to 300 MJ of reactor energy. An example is an instantaneous step power of 50 MW that can be held constant for 6 s, or 100 MW for 3 s in order to achieve the 300 MJ of energy. These types of conditions are typically desirable for advanced fuels testing experiments where the cladding or fuel material integrity of the test specimen is desired to be challenged.

Double pulse mode capability

The ACRR also has unique double-pulse mode capabilities. Using the pneumatic ejection system, the three pulse rods can be simultaneously ejected from the core for a single pulse or timed for a unique double pulse. Each of the pulse rods is worth greater than

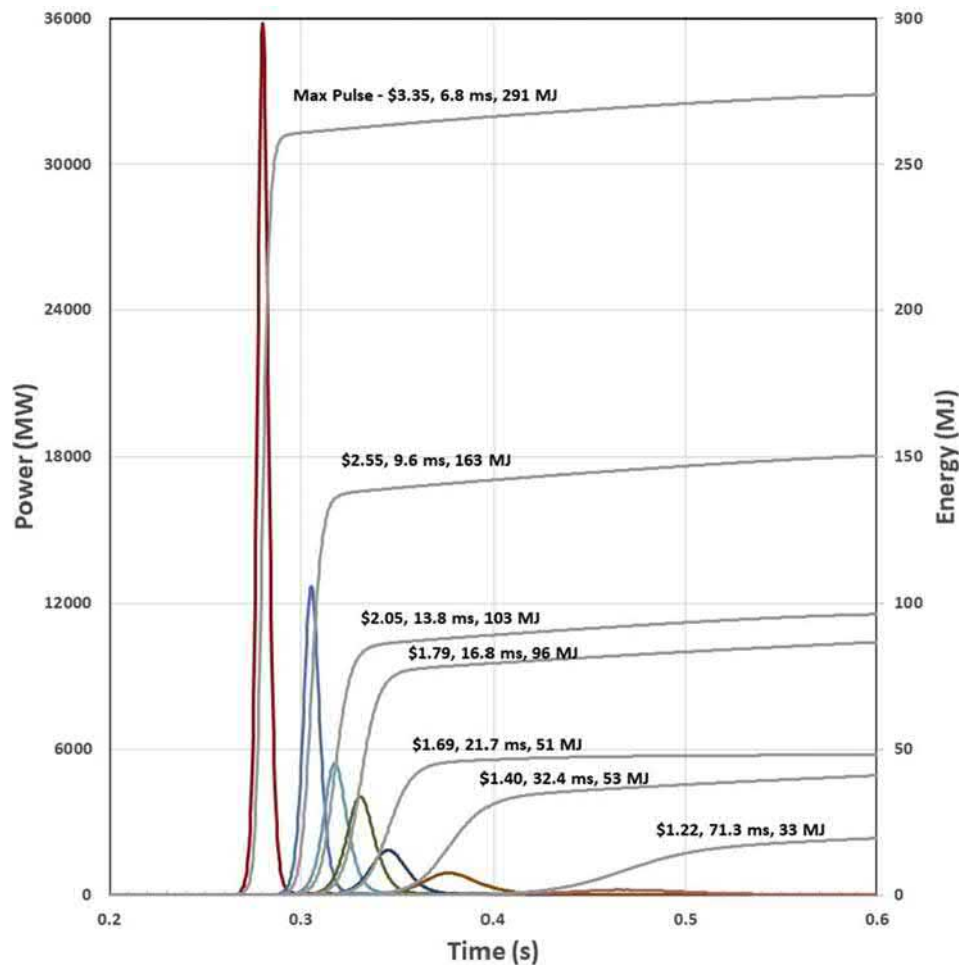


Fig. 9 ACRR pulsing characteristics.

\$1.00 of reactivity allowing for one rod to be ejected, followed by the other two. This capability is desirable for experiments where rapid heating is desired in an experiment, followed by an additional pulse.

Conclusion

The ACRR with its FREC-II subcritical multiplier is a unique research reactor facility unlike any other in the U.S. Its large pulsing capabilities and dry irradiation cavities allow for both passive and active experiments to be fielded at the peak neutron and gamma-ray flux in the core. Its epithermal/fast neutron spectrum in the central cavity allows for neutron spectrum modification and gives experimenters the ability to attain their unique irradiation requirements. The ACRR will continue to provide invaluable pulse and steady-state irradiations services to the U.S. defense program for at least another 40 years.

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MIR.M1 (100 MW)

Nikolay V. Arkhangelskiy, State Atomic Energy Corporation “Rosatom”, Joint Stock Company “Science and Innovations”, Moscow, Russia

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History

The first Russian research reactor, F-1, was commissioned in 1946. It was used only for the demonstration of sustainable nuclear fission chain reaction. Later in the 1950s and 1960s, Russian designers developed many diverse research reactors projects which demonstrated the vast creative potential of the national school of nuclear engineering. In the early 1950s the development of nuclear power program began in USSR. It became absolutely necessary to create powerful material testing reactors for the study of structural and fuel materials intended for the use in nuclear power reactors, propulsion facilities and other installations. The study of such materials for reactors with different types of coolant (water, liquid metal, gas) required the creation of channels equipped with autonomous circulation circuits called “loops.” In 1952, the Laboratory $\mathcal{N}^{\circ} 2$ (now the National Research Centre “Kurchatov Institute”) brought into the action the RFT—a material testing loop-type reactor which gave rise to extensive studies on the behavior of reactor materials and provided essential data for the design of different nuclear reactors. But experimental capabilities of this reactor were insufficient keeping in mind the proposed intensive development of fuel and structural materials for nuclear power facilities. By this reason it was necessary to replace a reactor RFT for a new more powerful loop-type reactor.

The idea of designing a new reactor for to carry out experimental activities in the field of atomic energy belongs to academician I.V. Kurchatov, scientific leader of Russian nuclear program. In 1956, at his suggestion, the Institute of Atomic Energy (IAE, former Laboratory $\mathcal{N}^{\circ} 2$) began a reactor feasibility study called MIR. Then in 1958, NIKIET, the leading Russian nuclear designer organization started designing the reactor MIR under the scientific supervision of IAE.

However, in 1960 IAE also began the development of a loop-type material testing reactor MR to replace the reactor RFT that was finally shutdown in 1962. The development of MIR and MR reactors occurred in parallel. It was a very productive competition between the two projects. The physical startup of the MR reactor in IAE was held in December 1963, and full power achieved in July 1964. The physical startup of the MIR reactor took place on December 24, 1966; the reactor was brought to full power on August 11, 1967.

After commissioning of these reactors, they provided almost all the reactor material testing research in the USSR for nuclear power and other tasks until the early 1990s when MR reactor was permanently shut down and the MIR reactor became the main Russian material testing reactor. At this time, the power level of MIR was raised to 100 MW.

The MIR (later renamed to MIR.M1) research reactor is a heterogeneous, loop, channel, thermal neutrons reactor installed in a water pool (Arkhangelsky et al., 2012; Fedulin and Vinogradov, 2016). It is designed for:

- loop tests of fuel elements and fuel assembly mock-ups of different nuclear reactor types;
- tests of structural and absorbing materials and products of different nuclear reactor types and purpose;
- production of radionuclides.

At first, the reactor had four core loop channels connected in pairs to two loops installed by that time: water loop PV-1 and steam-water loop PVK-1 of 2 MW capacity each. Later two 500 kW loops with a lead-bismuth coolant, each having one experimental channel, were put into operation as well as a 2 MW sodium-cooled loop with two loop channels. Once one more channel was connected to the water and steam-water facilities, the total number of loop channels in the core achieved 10.

In 1975, the reactor was refurbished and changed the name to MIR.M1. As a result, each loop channel was surrounded with six fuel assemblies and four or five control rods. It made it possible to make all the core loop cells practically equivalent and to improve the regulation of the test modes in each of the channels. After the refurbishment, the reactor was equipped with water loops PV-2 and PVK-2. After the start of the fast reactor BOR-60, also in RIAR, the sodium-cooled loop was converted into a loop with a high-temperature organic coolant.

In 1983, a loop with a steam-water coolant PVP-1 was commissioned with parameters close to those of the RBMK reactor coolant. In 1989–1990, the following loop facilities were commissioned: gas-cooled loop PG-1 and steam-water-cooled loop

PVP-2. Their sealed containments serve as additional safety barriers. It allows conducting specific experiments to simulate and study accidents of fuel elements in the loop channel.

Future prospects

Constantly conducted systematic and purposeful work on the improvement and extension of the reactor lifetime will allow operating the MIR.M1 reactor for research purposes up to 2027.

In 2016 RIAR received designations as IAEA International Centre based on Research Reactor (ICERR). Designation of RIAR as the IAEA ICERR confirmed its worldwide recognition as a reputable research organization. Reactor MIR.M1 includes in the perimeter of ICERR.

Future perspectives of the utilization of MIR.M1 reactor deal with the irradiation tests of materials for innovative water-water reactors such as VVER-C, VVER-SCWR in particularly:

- in-pile testing and research in developing the promising structural materials for Gen IV reactors;
- lifetime extension for the existing water-cooled power reactors;
- development and pilot demonstration of innovative nuclear technologies.

The MIR.M1 reactor used in the research with IAEA, China, Republic of Korea and other countries. The international cooperation activity includes bilateral commercial contracts for in-pile experiments and post irradiation experiments of structural, fuel and absorbing materials in support of innovative fission reactor technologies; long-term research programs for innovative light water reactor materials under way based on national programs and bilateral commercial contracts (Accident Tolerant Fuel Concepts, radiation resistant steels for light water reactor vessel and in-vessel devices, etc.); irradiation research and post irradiation experiments of materials and elements of equipment to be used in prospective fusion projects; long-term national research programs on radioactive wastes, spent fuels and minor actinides management technologies and bilateral commercial contracts.

Reactor fuel

The fuel for Russian reactors and MIR.m1 is fabricated according to original Russian extrusion technology (Alexandrov et al., 1996). This technology was successfully developed in Russia from beginning of 1950s and has made significant progress. This technology allows producing tubular fuel elements of various shapes adapted to different reactors and experimental devices. A MIR.M1 assembly is shown in Fig. 1. (See Table 1.)

A MIR.M1 element is structurally made from 4 coaxially located annular fuel elements fastened in the upper and lower grids. Each fuel element is a three-layer tube, in which the fuel layer from both sides is enclosed into the aluminum alloy cladding. In the cavity of the inner fuel element there is an annular aluminum displacer $\varnothing 34 \times 2$ mm. The fuel elements are 2 mm thick and on the outer surface along the whole length there are three azimuthally oriented ribs. The outer fuel element is 70 mm in diameter. Height of the fuel is 1000 mm. The fuel material is UO_2 of 90% ^{235}U enrichment in aluminum matrix. The fuel layer is 0.56 mm thick, the claddings are 0.72 mm thick and made of aluminum alloy. Inter fuel element gap is 2.5 mm. In total one fuel assembly contains 345–350 g of ^{235}U .

The MIR.M1 reactor uses highly enriched uranium (HEU) and Russian researchers participate in the uranium enrichment reduction program for research reactors (Podvig, 2017). From 2010 they have investigated consequences of the reactor conversion from HEU to low enriched uranium (LEU). To compensate for additional neutron absorption in ^{238}U after the conversion, it will be necessary to increase the density of ^{235}U from 0.9 to 1.0 g cm $^{-3}$ and total uranium density from 1.0 to 5.2 g cm $^{-3}$, respectively. For this purpose the material of the meat will change from UO_2 to U-Mo both in aluminum matrix.

Core configuration

By physical features, the MIR.M1 reactor is a thermal heterogeneous reactor with a moderator and a reflector made of metallic beryllium and water coolant. According to its design features it is a channel-type reactor located in a pool with water. Such design allowed for a combination of the main advantages of pool-type and channel-type reactors. A fundamentally new designer solution of the reactor MIR.M1 is the withdrawal of loop channels pipelines from the reactor into dry chambers under the pool water layer. It reduced the overall dimensions of the loop channels, simplified the reloading operations and improved operation and maintenance safety (Fig. 2).

The MIR.M1 core is arranged of hexagonal blocks of beryllium stacked along the axis of which straight-through zirconium alloy channels are installed to accommodate working and experimental fuel assemblies. The core design is chosen based on the condition of minimal mutual influence of neighboring test rigs since their operation modes can differ greatly. Over time, the beryllium has a negative effect on the operation of the reactor due to the accumulation of byproduct nuclides such as ^8Li and ^3He which are strong neutron absorbers. This effect reduces the reactivity margin and increases the power perturbation across the core.

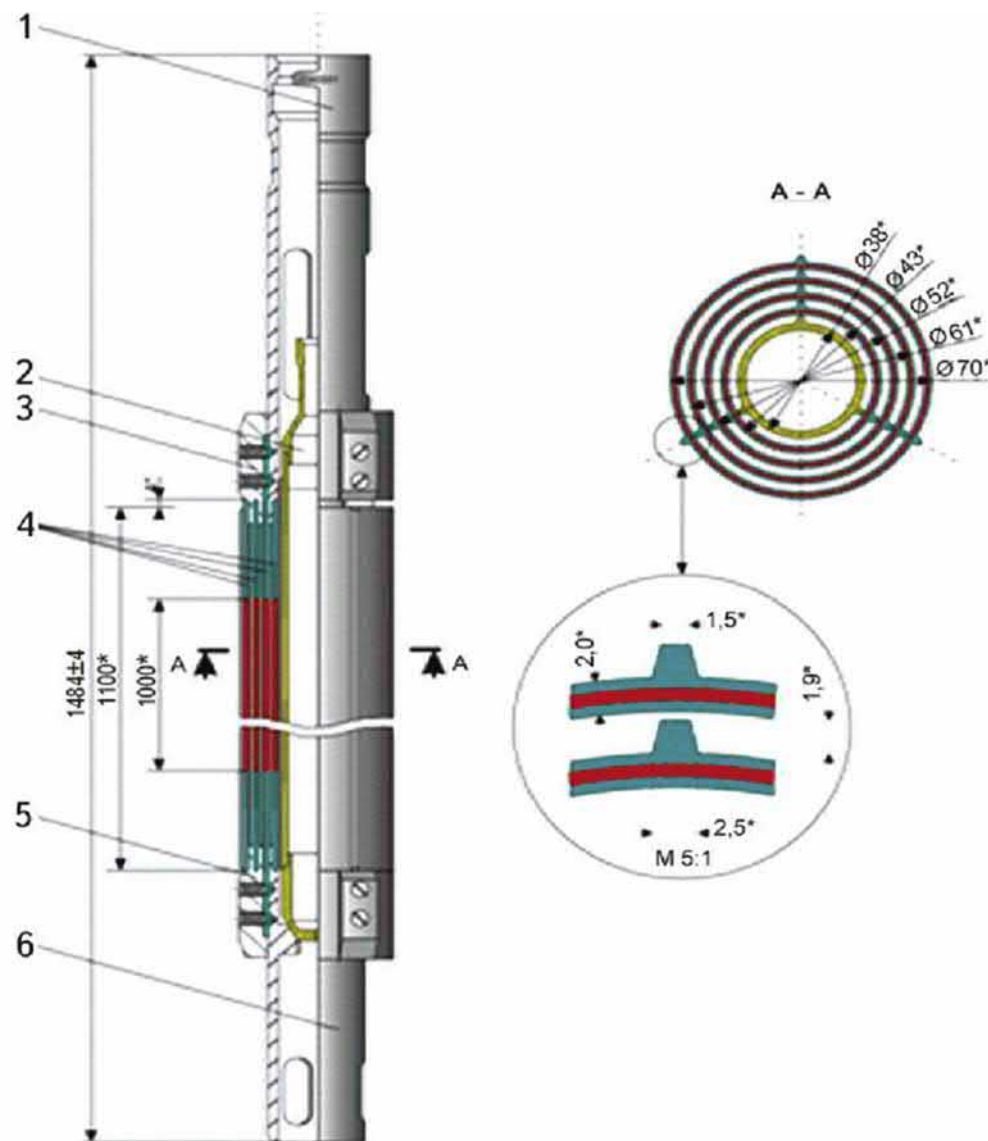


Fig. 1 MIR.M1 standard fuel assembly (1) top-end component; (2) displacer; (3) upper grid; (4) fuel tubes; (5) lower grid; (6) bottom-end component.

Table 1 MIR.M1 reactor parameters.

Name	Units	Value
Thermal power, max.	MW	up to 100
Maximum power of working channel	MW	3.2
Loop channel diameter, max.	mm	120
Core height	mm	1000
Number of loop channels, max.		11
Thermal neutron flux density, max.	$\text{cm}^{-2} \text{s}^{-1}$	5×10^{14}
Fast neutron flux density, max.	$\text{cm}^{-2} \text{s}^{-1}$	1×10^{14}
Average volumetric power	MW l^{-1}	0.85
Coolant:		water
pressure at the reactor inlet	M Pa	1.25
temperature at the reactor inlet	K	303–343
temperature at the reactor outlet	K	up to 371
Operation cycle duration	days	up to 40

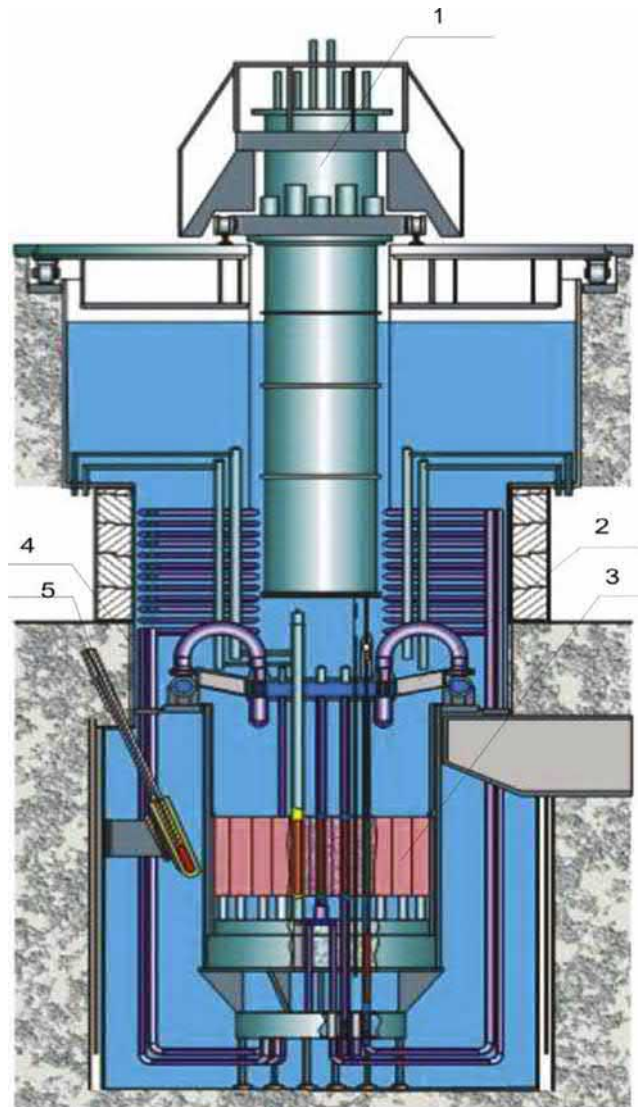


Fig. 2 MIR.M1 reactor cross-cut. (1) control rods drives; (2) divided inlet manifolds; (3) core; (4) outlet pipelines; (5) ionizing chamber.

Simultaneously in the reactor there can be performed tests of several trial fuel assemblies and experimental devices differing by the design, material composition, content of fissionable materials in fuel elements, power, type and parameters of the coolant. The main requirement set to the reactor is the possibility of provision, maintenance and control of the specified irradiation conditions simultaneously for all the investigated fuel assemblies. According to the reactor purpose the core structure is chosen from the condition of minimum mutual effect of the adjacent loop devices, since their operation modes can significantly differ. The cartogram of the MIR reactor core is presented in Fig. 3.

Control and safety rods

The distinctive peculiarity of the MIR reactor is the use of a great number of control rods. It is related to the necessity of creation of the energy release profile providing and maintaining the experimental work. These control rods are located in zirconium channels at the edges of the adjacent beryllium blocks. They are distributed by the performed functions as follows:

- 2 automatic control rods (AC);
- 27 safety rods and shim rods (SR-ShR).

The control scheme provides the possibility of selection of any 6 rods from 27 as safety rods (SR).

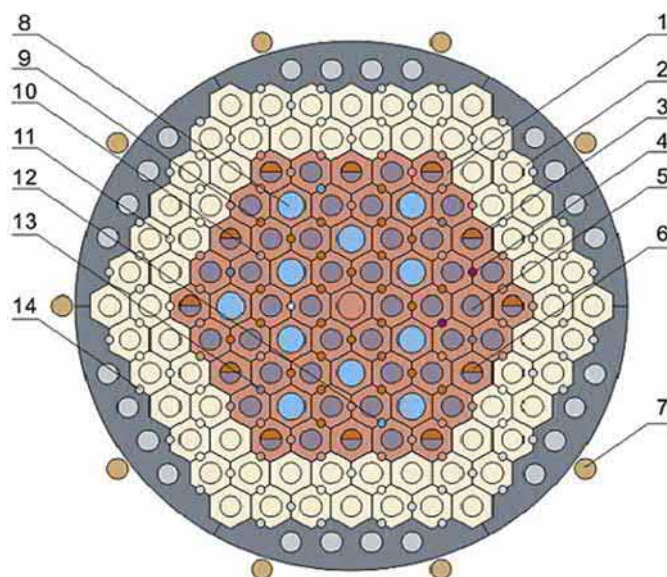


Fig. 3 Reactor core cartogram. (1) Core Be-block; (2) reflector Be-block; (3) loop channel Be-block; (4) emergency control rod; (5) operating channel; (6) channel for additional loading; (7) ionizing chamber; (8) loop channel; (9) scram rod; (10) core Be-plug; (11) reflector Be-plug; (12) control rod tube; (13) control rod with a plug; (14) Al-plug.

Reactivity compensation is also made by compensators that are located in the central channels of beryllium blocks of the core stacking peripheral row.

The shim and safety rods present the design consisting of the jointly connected parts: double-element absorber and displacer. On the upper SR end there is an adapter, with the help of which it is weighed to the drive exhaust. All SR units are connected to each other by spherical joints.

The SR absorbing part consists of two jointly connected cylindrical absorbing elements with nominal length of the absorber - 512 mm each. The absorbing element consists of the cladding having the outer diameter of 24 mm and wall thickness of 0.5 mm made from stainless steel. In the inner cladding cavity there is a fuel column from inserts of dysprosium titanate ($\text{Dy}_2\text{O}_3 \times \text{TiO}_2$) with diameter of ~ 22.8 mm, density of $6.0\text{--}6.3 \text{ g cm}^{-3}$. On top above the absorber column there is a gap for compensation of thermal lengthening and swelling of the absorber.

The absorber unit consists of a rod with diameter of 9 mm made from the zirconium alloy, on which bushes are threaded that supported by the lower plug. The bushes have the outer diameter of 24 mm and are made from CAB1 aluminum alloy. The displacer is 1100 mm long.

MIR.M1 experimental capabilities

The MIR.M1 reactor was designed to test experimental fuel, fuel assemblies and structural materials of various nuclear installations (power, propulsion) operated under different loads in different coolants (water, liquid metals, gas, organic compounds). The main feature of the reactor is 11 experimental loop channels in the core connected to autonomous loop facilities with different types and parameters of coolants to perform testing under various thermal and hydraulic modes.

Key areas of research at the reactor:

- tests of fuel elements and fuel assemblies under designed conditions for the cores of advanced reactors of increased safety;
- tests of irradiated and unirradiated fuel elements under accident-simulating conditions;
- study of behavior of leaky fuel elements under stationary and transient conditions.

Irradiation of promising fuel elements with claddings from improved alloys and fuel compositions based on intermetallics and cermets.

Irradiation and post-irradiation examinations of loop fuel assemblies of various types that were used in experiments with power ramps, loss of cooling and simulated leakage of fuel rod claddings.

At present, seven loop facilities are under operation at the MIR.M1 reactor, each connected to one or two loop channels.

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The BR2 Reactor: Short History, Characteristics and Utilization

Steven Van Dyck, SCK•CEN, Mol, Belgium

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Introduction

History of nuclear research and development in Belgium

The history of nuclear R&D is mostly the history of the Belgian Nuclear Research Centre, today known as SCK•CEN (Verwimp and Verledens, 2002). Today as in the past SCK•CEN plays a central role in research and development on the applications of nuclear technology and the effects of radiation, reaching out to academics, industry and research institutes in Belgium and abroad.

In September 1951, Pierre Ryckmans commissioned a group of scientists to set up a new organization to study the applications of nuclear energy. The founders came from various circles of the scientific world, universities, government and industry. After consultations they decided to set up a non-profit association that would be called “Studiecentrum voor de Toepassingen van de Kernenergie” (Research Centre for the Applications of Nuclear Energy) or abbreviated to STK. The choice of this form of company shows that they wanted to stimulate the peaceful development of nuclear energy in the public interest. The statutes were published in the Staatsblad (State Gazette) of April 19, 1952.

In 1954, construction at the site of Mol started. On Friday May 11, 1956, the BR1, Belgian Reactor 1, produced its first chain reaction. The reactor used graphite as a moderator, natural uranium as nuclear fuel and air as the coolant, with a power of 4 MW. The first design of the reactor was installed in Rhode-Saint-Genèse and was built in the premises of the “Administratie voor Luchtvaartkunde” (Department of Aeronautics). The employees of the Centre carried out the final design in close co-operation with specialists from Harwell (United Kingdom) and Belgian engineering consultancies. The BR1 reactor is still operational today.

After the successful start-up of the first nuclear reactor in Belgium, the research center decided to construct a high flux test facility, the BR2 (Fig. 1) and a nuclear power reactor, the BR3. The former would be a complement to the BR1 low flux reactor in order to support the development of nuclear energy applications in Belgium. The latter was initially planned to supply energy to the 1958 world exhibition in Brussels, but this plan was cancelled and the plant, the first pressurized water reactor on the European continent, was constructed in Mol as the BR3 reactor.

On July 23, 1957 the Royal Decree that set up the Belgian Nuclear Research Centre as an organization of public benefit with a legal personality appeared in the Belgisch Staatsblad (Belgian State Gazette). The new statutes of the organization described the role of SCK•CEN as follows:

- The collecting and keeping of scientific and technical documentation relating to the application of nuclear energy;
- Undertaking research of a scientific and technological nature regarding applied nuclear energy;
- Stimulating the training of specialized personnel relating to the application of nuclear energy;
- Performing supervisory and monitoring operations of a technical nature.

In the second half of the 1960s, facilities for studying and demonstrating options for the nuclear fuel cycle were established, as well as laboratories to study the effect of irradiation on materials and on living organisms. A pilot fuel reprocessing plant and

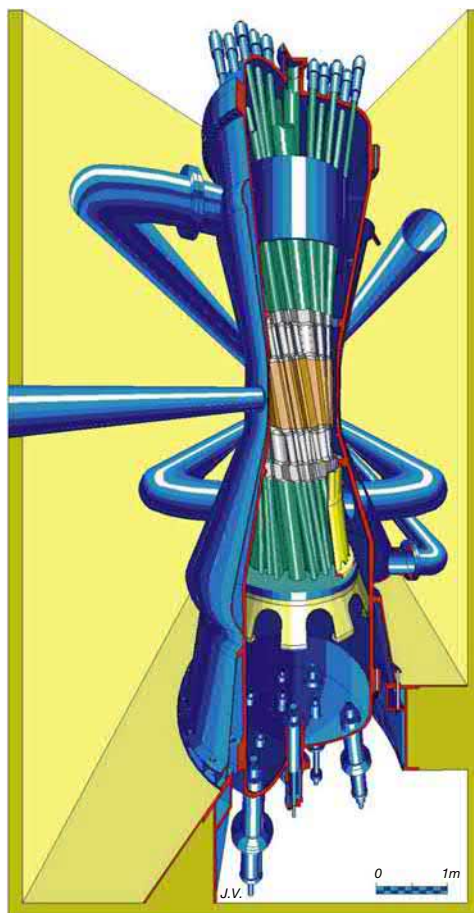


Fig. 1 Schematic of the BR2 reactor, showing its particular geometric design with the central beryllium matrix, containing the 79 irradiation channels.

underground laboratory complemented this infrastructure in order to support studies on the closed fuel cycle for nuclear power. This effort was part of the European development of nuclear power generation, following different reactor technical paths, including light water reactor, gas cooled reactors and liquid metal cooled reactors.

Since the 1970s, Belgium installed nuclear power generation capacity to cover about 55% of the electricity needs. Belgium is also one of the world's leading suppliers of radio-isotopes for industrial and medical use. The institute for Radio Elements (IRE) was established for conditioning, distributing and commercializing radioisotopes, mainly for medical applications, and also for developing applications in other fields, for example for the biomedical and industrial sector. SCK•CEN was responsible for the production.

Operational history or milestones of significance

The first criticality of the BR2 reactor was reached on July 6, 1961. This event was preceded by a design and construction period, which started in 1956. The Belgian Nuclear Research Centre contracted Nuclear Development Corporation of America in White Plains (New York, USA) to design the BR2. Managers of the STK, engineers from industry, engineering consultants and electricity producers took part in the design. The unique design of the BR2 is shown in **Fig. 1**. The construction of the reactor began in September 1957. In support of the commissioning and later operation, a zero power assembly, the BR02, went into operation at the end of 1959.

Belgian industry was used as much as possible for the construction of the reactor. On July 6, 1961 BR2 produced its first chain reaction. The Prime Minister, the chairman of EURATOM, the American ambassador and a representative of the Ministry of Economic Affairs attended the event. The reactor was further tested and was declared fully operational in December 1962. Irradiation experiments began in January 1963 and have continued to today. In 1971, the cooling capacity of the primary and secondary circuits was increased to 125 MW. The beryllium matrix was replaced for the first time from 1979 to 1980.

Since 1986, the BR2 reactor has been subject to decennial safety reassessments required by the regulator. The initial design life of 25 years has been extended under the same regulations as for the nuclear power plants and all class 1 nuclear facilities in Belgium.

There is currently no end date for operation of the BR2 in the license. The periodic safety reassessments are imposed by Belgian regulation, implementing the European Council directive 2014/87/EURATOM. The content of the safety reassessment is defined following the IAEA guide SSG-25.

In 1995, the beryllium matrix was replaced for the second time. The matrix of the BR02 zero power mock-up was inserted into the reactor and the BR02 was dismantled. Physics testing to determine core characteristics are now measured utilizing the BR2 reactor at zero power. The reactor was restarted in 1997 after a successful refurbishment.

For the 2016 periodic safety reassessment, a formal ageing management program was established for the BR2 reactor. The outcome of this program resulted in a second refurbishment program with replacement and modernization of many components (Van Dyck, 2017). As a consequence of the European Stress Test that became required after the Fukushima Daichi accident in 2011, additional improvements were made. A new and upgraded emergency electricity feed network was installed in support of defense in depth criteria for the passive safety system of the BR2 reactor.

In 2016, the BR2 reactor restarted and was again the most powerful operating research reactor in Western Europe. The 2016 periodic safety reassessment covers the operational period up to 2026 and the strategy for operation until 2036 is currently under preparation.

How the facility mission or design has evolved over its lifetime

The initial mission of the design of the facility was expressed in the initial mission to the reactor designer, “a test facility of greatest overall usefulness in a future power reactor development program ... a basic objective of this program was to provide high flux test facilities of ready accessibility.”

This mission has basically not changed during its history (Baugnet et al., 1987). The main design change has been the power upgrade in 1971. During the refurbishment of 2015, the beam tubes, out of use since a long time, have been removed. With the change of the societal and industrial focus from power production to non-power applications, the utilization of the reactor has shifted over the years, but the basic mission to provide a flexible platform for performing irradiations has remained.

Nuclear reactor design

Fuel design

Fuel construction and fabrication

The BR2 driver fuel elements consist of six concentric cylinders (Fig. 2). The fuel meat is currently based on highly enriched Uranium (90–93%). The outer diameter of the standard fuel elements is 8.4 cm; the inner diameter is 2.5 cm, which allows the insertion of irradiation devices inside the fuel elements. Non-standard elements may have less plates (inner fuel ring or two rings removed) or a different outside diameter to be loaded in non-standard channels (20 cm outer diameter elements or six elements with 5.7 cm outer diameter elements were used in 20 cm channels).

The initial fuel design was based on seamless or welded tubes, containing 224 g of ^{235}U as UAl_4 . This fuel design was replaced by a new design, using UAl_x cermet fuel meat. This development was coupled to the power upgrade of the reactor to 100 MW. In a second stage, cermet fuel of the same density, containing burnable absorbers in the fuel meat was introduced in order to prolong the reactor cycle and improve the fuel economy. At this same time, the tube design was replaced by a three-segment bent plate design using stiffeners, so the central aluminum tube could be removed and experimental capsules could be inserted into the center of the fuel elements more easily.

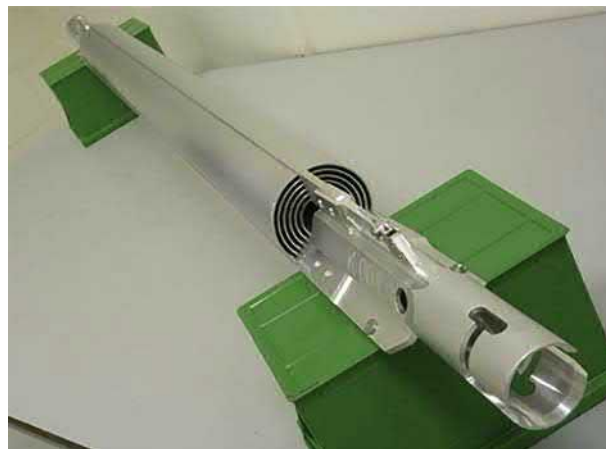


Fig. 2 A standard driver fuel element of the BR2 reactor, consisting of six concentric tubes, formed by three plates, swaged into stiffener plates.

The fuel elements are constructed according to the standard industrial practice of powder metallurgy to prepare the fuel compacts, then rolling and forming the plates and finally swaging the fuel plates into the stiffener plates to form the elements. In the past, extrusion of tubes was also used as a fabrication technique, as well as welding three 120-degree segments into tubes.

Nuclear physics parameters

The availability of irradiation positions inside the driver fuel elements and the flexibility of optimizing the reactor core configuration yields a high variety of possible irradiation conditions. The neutron flux can reach $10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$ thermal flux and $6 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$ fast flux ($E > 0.1 \text{ MeV}$). Typical neutron spectra are given in Fig. 3. The use of absorbing screens to tailor the neutron spectra has been applied mainly for fast reactor fuel experiments (Fig. 4).

The BR2 driver fuel elements are currently used to a burn-up at final discharge of about 55% of the initial fissile metal content. Together with the consumption of the burnable absorbers, this implies a very significant evolution in nuclear physics parameters of the elements during their useful lifetime in the reactor. The effect of the burnable absorbers is twofold:

- The initial loading of the elements with fissile material can be increased without jeopardizing the safety in terms of reactivity for storage or loading of the reactor. The elements with burnable absorber produce a lower k_{eff} in an infinite hexagonal lattice than the initial fuel design with no burnable absorbers.
- As the burn-up of the fuel elements increases, the reactivity of the elements increases. This allows BR2 to use the elements for a significantly longer time in the core of the reactor, as their reactivity increases in the initial stage of life in the reactor.

Path to conversion to LEU

At the initial stage of the Reduced Enrichment for Research and Test Reactors (RERTR) program, it was recognized that the BR2 driver fuel could not be converted to a fuel system based on low enriched uranium (LEU) as it was available at that time. Recently,

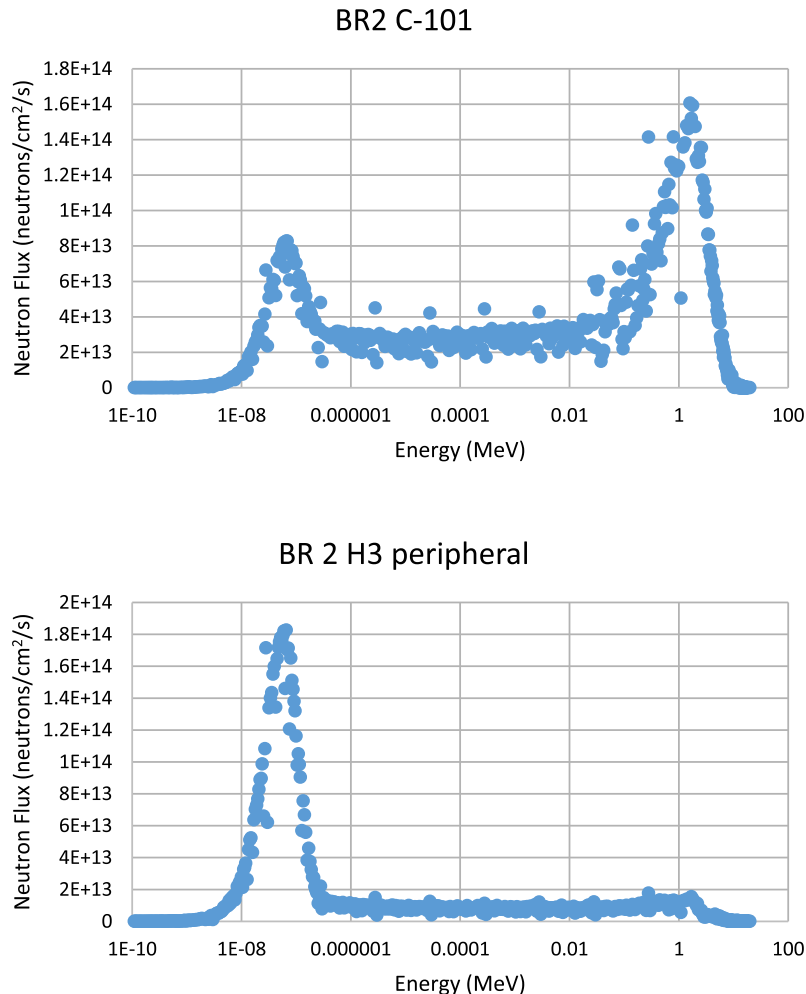


Fig. 3 Neutron spectra in the axis of a fuel element (top) and in a 20 cm Be plug (bottom).

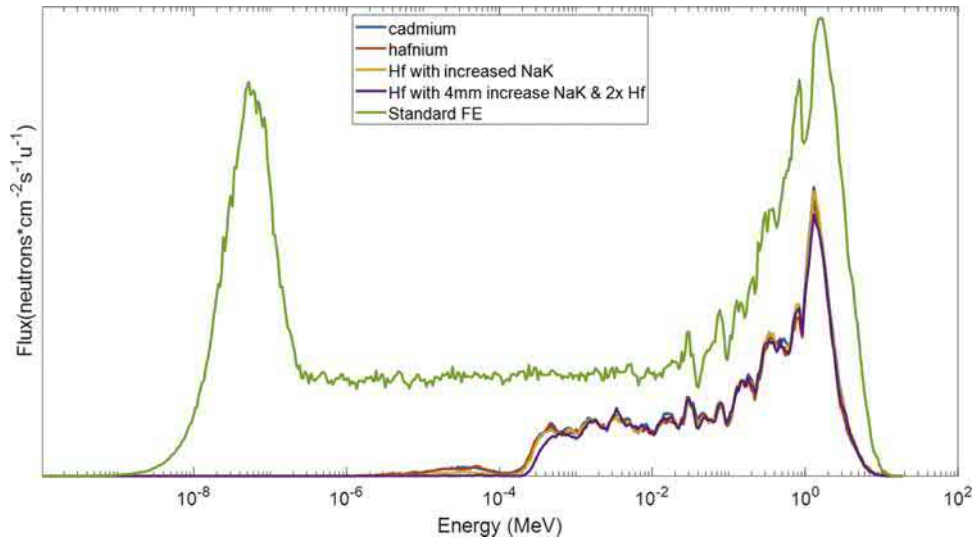


Fig. 4 Application of Cd screen to harden the neutron spectrum in a BR2 loaded experiment inside a fuel element.

advances in fuel development and minor redesign of the fuel element have led to two potential fuel systems, suitable for conversion of the BR2 driver fuel.

- Dispersed UMo with a density of 7.5–8 g of uranium per cm^3 . Due to the absorbing nature of molybdenum, this type of fuel requires a higher volume fraction of Uranium in the fuel meat, but due to the higher density of the uranium-molybdenum alloy, this is feasible. The UMo with 7.5 g cm^{-3} uranium has the same equivalent fissile content. The fuel meat volume remains the same in the plate design as in the HEU case. However, due to poor irradiation behavior, this type of fuel could not be qualified to the required extent of power level and burn up requirements for the BR2. A back end solution (i.e. recycling of the material) is not available for this type of fuel. Under Belgian regulation, the availability of a back-end is a requirement.
- Dispersed U_3Si_2 : a higher density than the 4.8 g cm^{-3} which is described in NUREG-1537 is required in order to meet the performance goals for the BR2 reactor. In order to limit the volume fraction of the dispersed phase to less than 50 vol%, the uranium density in the selected fuel system has been limited to $5.3 \text{ g of uranium per cm}^3$. The details of the fuel element design evolution and qualification can be found in reference (Kalcheva et al., 2018).

Core configuration

One of the unique features of the BR2 reactor is that the core configuration is not fixed in the design nor the license. The core consists of a metallic Beryllium matrix, in which 79 irradiation channels are present. There are three types of irradiation channels:

- Standard channels have a diameter of 8.5 cm; 64 channels of this size are present in the matrix. These channels may contain standard fuel elements, control or regulation rods, experiments or a Beryllium plug.
- Large channels with a diameter of 20.3 cm, five in total, of which one is the central channel of the reactor. These channels may also contain experiments or plugs. In standard conditions, Beryllium plugs with either three channels of 8.5 cm diameter or one channel of 8.5 cm, surrounded by six channels of 3.3 cm diameter can be loaded in such 20 cm channels. Plugs with different configuration can be inserted for dedicated experiments. Also, dedicated fuel elements with outer diameter of 20 cm can be loaded in such channels. The number of fuel plates in such elements is optimized according to the required space inside this fuel element to insert an experiment.
- Small channels with a diameter of 5.0 cm are located at the periphery of the core. There are 10 such channels. These channels cannot contain fuel elements nor control rods.

The selection of the core configuration depends on the demand for irradiations, the reactor cycle length and the number and position of the control and regulating rods. The reactor cycle length is typically between 21 and 35 days. Shorter cycles may be performed for dedicated experiments.

The configuration of each cycle needs to fulfill the safety criteria:

- The minimum available anti-reactivity (shutdown margin necessary to keep the reactor subcritical) needs to be above $\$2.00$, in the most unfavorable conditions, including malfunctioning of the most reactive control rod.
- The heat flux on the driver fuel element plates may not exceed 470 W cm^{-2} . This limit is evaluated for the hottest fuel element at the most penalizing position of the shim rods.

Respecting these criteria, the configuration is composed with the following objectives:

- to provide the required flux levels in the different irradiation devices loaded in the reactor core. This is done by selecting the appropriate fuel elements and beryllium plugs, containing the irradiation rigs and surrounding them. The total reactor power can be adjusted as additional degree of freedom, within the limit of the cooling capacity of the primary and secondary cooling loop (125 MW maximum).
- to provide the required cycle length by loading a sufficient amount of uranium; this is done by selecting the appropriate number of fuel elements and their burn-up at the beginning of the cycle.

Reactivity control and safety systems

Scram safety control rods or other shutdown systems

A number of shim-safety rods needs to be present into the core configuration, as required by the technical specifications. These rods are inserted from the top cover of the reactor and have an absorbing element consisting of metallic Hafnium with a length of 94 cm (exceeding the fueled zone by more than 10%). The rods are withdrawn by electromagnetic motors, which are fail safe upon an inadvertent loss of electrical power. Rods are dropped by spring assisted gravity. The absorbing part is connected to a Beryllium follower. A maximum number of eight shim rods can be loaded in a reactor core configuration, with one additional safety rod (extracted during reactor operation, no shim function). Eventually less shim rods can be used (partially replaced by safety rods).

Reactor trips are engaged either by fast electronic interruption of the electromagnet current in the rods drive mechanisms or by opening of relays. The former method is fastest (response time less than 50 ms) and reserved for the nuclear safety instrumentation action and the primary cooling system pressure. The relay SCRAM is actuated by process variables, detection of failure of support systems or experiments.

Regulating rod and control systems

One or two regulating rods can be loaded in the reactor vessel but only one rod can move at a time. Only one rod is required and operational in the regulation control loop. The regulating rod is loaded in a reactor channel where its reactivity worth is less than \$0.48. The regulating rod provides the following functions:

- automatic start-up of the reactor from 1% to 100% power;
- stabilization of the reactor power at a constant level;
- automatic slow or fast set back of the reactor power.

The power regulation system is based on a double loop measurement of neutron flux in the reactor and the relative power change, measured by the ^{16}N in the primary water. The absolute power calibration is done using the thermal balance of the primary cooling loop.

Primary and secondary cooling loops

The reactor core is cooled by demineralized water, flowing at about 10 m s^{-1} along the fuel plates. The inlet water temperature is maintained below 40°C ; the outlet temperature is maintained below 60°C . The inlet pressure in the reactor is $1.2 \times 10^6 \text{ Pa}$ (12 bar), with a pressure drop over the reactor of $3 \times 10^5 \text{ Pa}$ (3 bar) (about $2.1 \times 10^5 \text{ Pa}$ (2.1 bar) pressure drop over the fuel elements). As the reactor channels are fed from a common top water plenum, the pressure drop across all channels needs to be the same. Experiments therefore need to be designed not to create a hydraulic by-pass through the core, which would reduce flow through the fuel elements. The total flow rate depends on the configuration of the core, as fuel elements typically require more flow than plugs or experiments. For a typical configuration, the flow needed to maintain the set pressure drop is around $25.2 \times 10^6 \text{ m}^3\text{s}^{-1}$.

The primary cooling system loop contains four main pumps (three in operation, one in standby) and two shut down pumps which are used when the reactor is not operating. The primary pressure is maintained by an electrical pressurizer, which also serves as degassing unit. The three primary heat exchangers operating in parallel have a total cooling capacity of 125 MW. The heat is transferred to the atmosphere in open cooling towers with forced draft. In order to limit the environmental impact if a sudden rupture in the main heat exchangers would occur, the acceptable contamination level of the primary water is 400 Bq cm^{-3} of ^{131}I . The primary water is continuously purified by resin ion exchange. The pH is maintained typically between 5.5 and 6.5. The residual heat from the BR2 driver fuel can be evacuated either by forced convection (with the main or shutdown primary cooling pumps) or by natural convection if there is a loss of electrical power. In the latter case, the flow will reverse from top-bottom flow to bottom-up flow.

In order to avoid fuel damage, the maximum heat flux on the driver fuel plates has been limited to 470 W cm^{-2} . However, for some experiments, 600 W cm^{-2} can be allowed when the experiments are cooled by the primary water. (600 W cm^{-2} corresponds to the onset of nucleate boiling). This limit only applies in nominal conditions of operation of the primary loop and cooling water gaps which are not smaller than the BR2 driver fuel plate spacing (3 mm water gap between the plates). For similar reasons, experiments inserted inside driver fuel elements may not generate more than 25 W cm^{-2} , 23 s after reactor SCRAM.

Experiment cooling loop

This is a closed loop available as heat sink to experimental devices. The experiment primary cooling loop operates at 1.5×10^6 Pa pressure, with a maximum allowable temperature of 109°C . The flow rate in the primary experiment cooling loop is $3.24 \times 10^5 \text{ m}^3 \text{ s}^{-1}$. The total cooling capacity of this experiment cooling loop is 25 MW; typically, individual experiments can generate up to 5 MW, with the heat transferred to the experiment cooling loop.

Eventually, demineralized water from the primary experiment cooling loop may be fed directly into experiments at low flow rate; this mode is a bleed mode and the water is evacuated towards the liquid radwaste system of BR2 afterwards. Such experimental devices are cooled by the BR2 main coolant loop and the water of the experiment cooling loop only provides a separate environment to the experiment (mainly used to maintain different pressure and temperature than the main loop in the experiment).

In-core experimental capabilities (thimbles, loops, rabbit, etc.)

As pointed out in the section on core design, the BR2 reactor channels can accommodate different types of experimental rigs as shown in Fig. 5. These rigs can either be installed in an entire channel or inside the central cavity of a fuel element (standard fuel element or element with five plates). All channels are accessible from the top of the reactor, so experiments are typically loaded through the reactor pool into the reactor. Experiments can be cooled by the primary water of the BR2, by pool water (in a thimble tube) or by independent cooling loops. For the installation of loops, the five channels of 20 cm diameter and 12 standard channels of 8.5 cm can also be accessed from the sub pile room. This allows for the installation of through loops in the reactor.

Experiments in the BR2 reactor can be loaded in four types of irradiation positions:

- Irradiation positions inside fuel elements: these can be in standard six plate elements (F1) (diameter 2.54 cm) or dedicated five plate elements (F2) (diameter 3.2 cm) for more space. Typically, these positions yield the highest fast flux levels but limited space. The fluxes quoted in the table scale with the power level of the reactor and can vary depending on the position of the element and the burn-up level of the surrounding elements.
- Irradiations in standard channels (S): flux levels will depend on the position of the channel in the reactor (total flux) and the thermal to fast flux ratio will be optimized by the number and burn-up of the surrounding fuel elements. All standard channels have a diameter of 8.4 cm; the flux level generally varies with the distance to the reactor's central flux trap.
- Irradiation in large channels: there are five large channels (H), offering space up to 20 cm in diameter. These channels can contain a single irradiation rig (20 cm), 1–3 standard channels (8.4 cm) or a combination of a standard channel with six small channels (3.3 cm).
- Irradiation in peripheral channels (P; 5.0 cm diameter), located at the edge of the matrix.

The dimensions of the different irradiation channels and typical available flux ranges are given in Table 1.

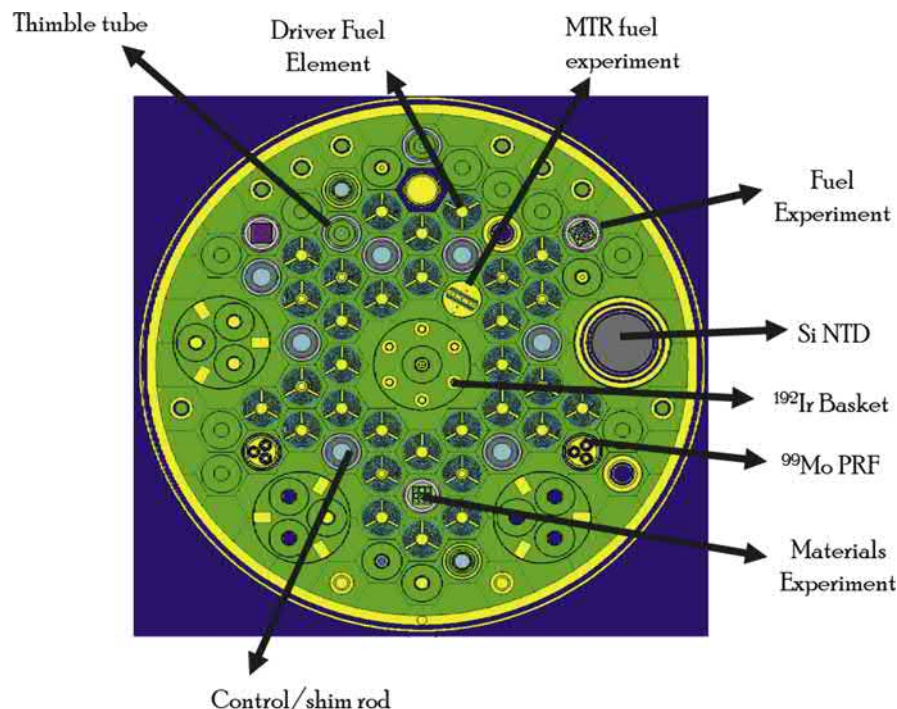


Fig. 5 Typical core configuration of the BR2, combining the irradiation of production devices and experiments.

Table 1 Nuclear characteristics of the irradiation channels.

Channel type	Thermal flux range ($10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$)	Fast flux range ($10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$) ($E > 1 \text{ MeV}$)	Gamma heating (W/g Al)	Diameter (mm)	Typical number available
F1	1–3.5	0.5–2.8	1.7–8.8	25.4	30
F2	Up to 2.5	Up to 2.5	Up to 6.8	32	2 ^a
S	1–3.5	0.1–0.7	0.9–2.3	84	24 ^b
Central large channel H1	Up to 10	Up to 1.8	3	200	1 ^c
Peripheral large channel Hi	3	1.3	0.1	200	4 ^d
Peripheral small channel P	0.7–1.5	0.05–0.1	0.4–1	50	9

^aThe five plate elements are loaded upon experimental request; the amount in the core depends on the number of used/available rigs requiring a five-plate element.

^bThe number of available standard channels depends on the configuration (number of fuel elements, control rods and isotope irradiation facilities loaded).

^cThe 20 cm central flux trap can be configured to hold one 20 cm rig, or one 8.4 cm rig and six 3.3 cm rigs. In the 8.4 cm rig also a fuel element in the central flux trap can be loaded with an irradiation rig inside.

^dThe available peripheral 20 cm channels are configured with three inner 8.4 cm channels in the standard configuration. One channel is reserved for silicon doping.

Loading and unloading of thimble tube experiments can be done during reactor operation. Experiments inserted into the primary water of the BR2 cannot be unloaded during reactor operation (the devices for production of ^{99}Mo by irradiation of uranium targets being an exception).

For unloading experiments to the pool, either the entire experimental capsule can be unloaded or the capsule/loop can be opened in the reactor pool. The reactor loading crane can handle loads up to 360 kg; for larger loads, a polar crane with hoists of 5 tons and 25 tons is available. Materials and rigs can be stored in pools in the reactor building or transferred to the adjacent process building storage canal. On this storage canal, a large hot cell is available for loading and unloading irradiated materials from irradiation rigs. This connection allows flexible and fast handling of irradiated materials. For irradiated fuels, cooling down to remove decay heat may be required before transfer to the hot cell. The storage pools in the reactor building have a separate cooling loop.

Thimble tubes are available in two sizes, one with diameter 3.6 cm (standard size) and one with diameter 7.0 cm (large size). The standard thimble tubes are (amongst others) used to irradiate materials in standard capsules, having an inner diameter of 2.3 cm and a useful inner length of 7.0 cm. In support of irradiation capsules for materials and fuel, a water chemistry control unit is being built to be connected to the primary experiment cooling loop. This unit will provide water with controlled pH (through LiOH addition) and hydrogen content to irradiation capsules for materials and fuel pin irradiation.

Irradiation devices loaded inside the reactor's primary circuit can in general not be unloaded during the reactor cycle (Mo-99 production devices being an exception). Hence, all rigs or baskets loaded inside fuel elements and most other rigs are irradiated for the entire reactor cycle.

Several rig designs are available, which can be recharged with fresh or pre-irradiated specimens. An overview of these rigs is given in [Van Dyck and van den Berghe \(2018\)](#). The main rig types are the following:

- Capsules for material irradiation: un-instrumented capsules can be inserted in the primary water of the BR2 reactor. They can either be loaded in a fuel channel (F1 in [Table 1](#)) or a reflector channel (S or P in [Table 1](#)). The irradiation temperature is not controlled but determined based on the predicted nuclear heating rate and the design of the specimen holder in the capsule. In that way, an engineered gas gap between the specimens in their holder and the capsule wall in contact with the BR2 primary water creates a temperature gradient at constant reactor power to obtain the desired irradiation temperature. Capsules with temperature control are available according to the desired irradiation temperature range, environment and neutron flux. For low neutron flux, capsules with electric heating can be inserted in a thimble tube, loaded in a reflector position (S position). The irradiation temperature is controlled by a combination of variation of electric power to the integrated heaters and water flow in the thimble. For high flux applications and water environment, a boiling water capsule is available, inserted in a fuel element (F2 type channel). This capsule offers temperature control by adjusting the pressure to the desired boiling point and operates from 150 to 350 °C. For higher temperatures, a gas filled capsule in an F1 type channel can be used, controlling the irradiation temperature by variation of the pressure of inert gas in the gap between the specimens and the outer wall of the capsule, cooled by primary BR2 water. Such capsule can operate up to 1000 °C.
- Capsules for fuel irradiation: small fuel samples can be irradiated in capsules with controlled gas gaps in order to achieve fast burn up accumulation at various temperature. The temperature can be measured and is set by engineering the geometrical properties of the fuel holder ([Williams et al., 2019](#)). Light water reactor fuel pins can be irradiated in a pressurized water capsule. This capsule allows to control either irradiation temperature or pressure to the desired set point. The fuel pin is cooled by boiling water, with radial heat evacuation. The heat, produced in the rig is measured on line, so the device can be used for steady state or transient irradiations ([Wéber et al., 2010](#)). Finally, plate type fuel can be irradiated in baskets in the BR2 primary water. This device is frequently used in the development of material test reactor fuel ([Van Berghe et al., 2012](#)).

Conclusions

Today, the BR2 reactor remains one of the highest performing and versatile neutron irradiation platforms available to the international community. It allows the utilization of a number of irradiation devices for providing irradiation services to a wide spectrum of clients, from users of radioisotopes and semiconductors to nuclear material and fuel researchers. Thanks to a long-term vision of SCK CEN, supporting the continuous investment and improvement of the installation, an increased and long-term availability of up to 210 days per year up to 2036 can be supported. The available infrastructure and services on site also allow an integrated service to users, from preparation of irradiation devices up to the post irradiation examination on site and shipment of irradiated materials.

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The Jules Horowitz Reactor[☆]

Gilles Bignan, French Atomic and Alternatives Energies Commission (CEA), Paris, France

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Abbreviations

BWR Boiling water reactor (Western Style Nuclear Power Plant)
DA Destructive analysis
EPR European pressurized reactor
GEN II/III Generation 2 and Generation 3 Nuclear Power Plants (water coolant reactors)
Gen IV Generation IV reactors (innovative concepts)
I-C & C Instrumentation-Command & Control
JHR Jules Horowitz reactor
LOCA Loss of coolant accident
MTR Material test reactor
NDA Non-destructive analysis
NPP Nuclear power plant
PWR Pressurized water reactor (Western type Nuclear Power Plant)
R&D Research & development
RPV Reactor pressure vessel
TSO Technical support organization
VVER Водо-Водяной ЛнерГетический Реактор (Russian word for Vodo-Vodianoï Energuetitcheski Reaktor meaning Water Energy Reactor)—(Nuclear Power Plant developed by Russian Federation)

Introduction

Material Test Reactors (MTR) have provided an essential support for nuclear power programs over the last 50 years within the Nuclear Community. However, the large majority of these MTRs are more than 50 years old, leading to the increasing probability of some shutdowns for various reasons (e.g., life-limiting factors, heavy maintenance constraints, possible new regulatory requirement) as one may have seen recently with the permanent shut-down of OSIRIS (Greek: “The King”) in France (2015), JMTR (Japan Materials Testing Reactor) in Japan (2017) and Halden project in Norway (2018). Such a situation is not sustainable in the long term.

On the other hand, when associated with hot laboratories for the post irradiation examinations, MTRs remain key infrastructure facilities for the research area in the field of nuclear fission energy. MTRs address the development and the qualification of materials and fuels under irradiation with sizes and environment conditions prototypical for nuclear power plants in order to optimize and demonstrate safe operations of existing power reactors as well as to support future reactor design:

[☆] Jules Horowitz (1921–1995) was a prominent French scientist who founded the reactor physics division at CEA; this research reactor's name is a tribute to his intense contribution to nuclear physic.

- Nuclear plants will follow a long-term trend driven by the plant life extension and management, reinforcement of the safety, waste and resource management, flexibility and economic improvement.
- In parallel to extending performance and safety for existing and power plants to come, R&D programs are taking place in order to assess and develop new reactor concepts (Generation IV reactors), innovative fuel and materials that meet sustainability purposes.
- In addition, for most nuclear countries, keeping competences alive is a strategic crosscutting issue; developing and operating a new and up-to-date research reactor appears to be an effective way to train a new generation of scientists and engineers.

It is important to note that new facility construction today is much more complicated due to safety and environmental regulations plus the higher costs of labor and materials and, currently, there are only two new high power (above 30 MW) MTR under construction worldwide:

- JHR at CEA Cadarache (France) mainly in support to LWR
- MBIR in RIAR-Dimitrovgrad (Russian Federation) mainly in support of fast spectrum reactors.

JHR at a glance

JHR will offer modern irradiation experimental capabilities to study material & fuel behavior under irradiation (Bignani et al., 2011). JHR will have a flexible experimental infrastructure to meet industrial and public needs related to present and future nuclear power reactors. JHR is designed:

- to provide high thermal and fast neutron fluxes,
- to run highly instrumented experiments,
- to support advanced modelling allowing prediction beyond experimental data points,
- and to operate experimental devices under specific environmental conditions (pressure, temperature, flux, coolant chemistry) typical of light water power reactors (PWRs, BWRs, VVERs), but with some limitations as it is a water research reactor-in support of non-water reactors R&D (sodium cooled fast reactors).

These objectives require representative tests of structural materials and fuel components as well as in-depth investigations with “separate effects” experiments coupled with advanced modelling. For example, the JHR design accommodates improved on-line monitoring capabilities such as a fission product laboratory directly coupled to the experimental fuel sample under irradiation.

As a modern research infrastructure, JHR will contribute to the development of expertise and to the training of the next generation of scientists and operators with a positive impact on nuclear safety, competitiveness, and social acceptance. The JHR is designed mainly to meet these technical and societal objectives. As an associated objective, the JHR will also contribute to secure the production of radioisotopes for medical applications. The following Fig. 1 gives a schematic view of the facility and a photo of the building site as of mid-2020.

JHR, as a future international user facility, is funded and steered by an international consortium gathering industry (Utilities, fuel vendor, reactor designers) and public bodies (R&D centers, Regulator and academic institutions). The generic model of JHR consortium is the following:

- CEA remains the owner and the nuclear operator of the nuclear facility with all liabilities,
- JHR Consortium Members are (for the whole operation life of the reactor) the owners of Guaranteed Access Rights to the experimental capacities in proportion to their financial commitment to the construction and with a proportional voting right in the Consortium Governing Board,
- A member can use totally or partly his access rights for implementing proprietary programs with full property of results and/or for participating to the Joint International Programs open to non-members
- JHR consortium membership is open to new members until completion of the reactor.

CEA is encouraged by the consortium to enlarge JHR membership and, as of mid-2020, the present members list of JHR consortium is the following: CEA (France), EDF (France), AREVA.SA (France), FRAMATOME (France), TECHNICATOME (France), European Commission-JRC, SCK-CEN (Belgium), UJV-NRI (Czech Republic), VTT(Finland), CIEMAT(Spain), Studsvik (Sweden), DAE(India), IAEA (Israel), and NNL (United Kingdom).

JHR general description

This facility is a 100 MWth pool type reactor with a compact core (60 cm fissile length, 70 cm diameter) cooled by a slightly pressurized primary circuit (Coulon et al., 2019). The nuclear facility comprises (Fig. 1) a reactor building with all systems dedicated to the reactor and experimental devices; and an auxiliary building dedicated to tasks in support for reactor and experimental devices operation (hot cells, storage pools, laboratories).

The reactor building consists in pre-constraint concrete with a diameter of 37 m. The nuclear auxiliary building consists in three storage pools for spent fuels, irradiated experimental devices and in four hot cells for preparation, conditioning of experiments and

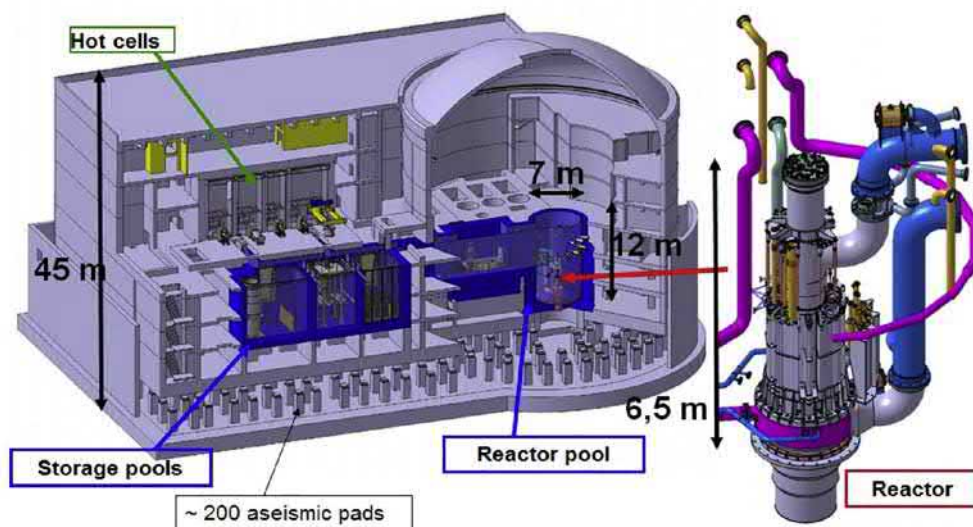


Fig. 1 Jules Horowitz Reactor (JHR) schematic view and photo as of mid-2020. Courtesy CEA.

non-destructive examinations on irradiated samples. A transfer channel between the reactor building and the nuclear auxiliary building allows the underwater transfer of spent fuels and experimental devices between the two buildings. In support to the nuclear island, additional ancillary facilities include the following:

- 1 support building for cooling system
- 1 support building for the fluids and ventilation facilities
- 2 emergency diesel generator buildings
- 1 building for assembly and test of experimental devices before entering the nuclear island ("cold workshop").

The design of the core and Beryllium reflector provides various irradiation locations and cavities; the core consists of 37 silicide fuel elements (U_3Si_2 -cylindrical shape) with several control and pilot rods made of Hafnium (see Fig. 2 below).

The reactor facility is designed to operate 20 experiments simultaneously in the various testing locations:

- Located in the core (7 of small diameter; 3 of large diameter) which are characterized by a high fast neutron flux ($5.5 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$ with energy above 1 MeV corresponding to 16 dpa (displacements per atom)/year).
- Located in the beryllium reflector zone surrounding the core, (about 20, most of them of 10 cm diameter), with a high thermal neutron flux (up to $5.5 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$). Material experiments requiring a low radiation exposure rate (such as the reactor pressure vessel steel) will be installed in the reflector because of the low fast neutron flux corresponding to about 0.1 dpa/y.

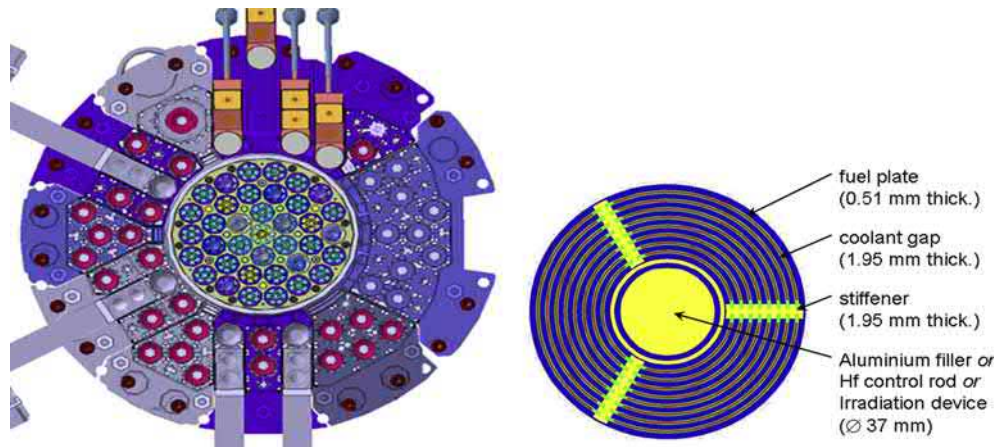


Fig. 2 JHR cross-section with central reactor core surrounded by reflector and sketch of fuel element. Courtesy CEA.

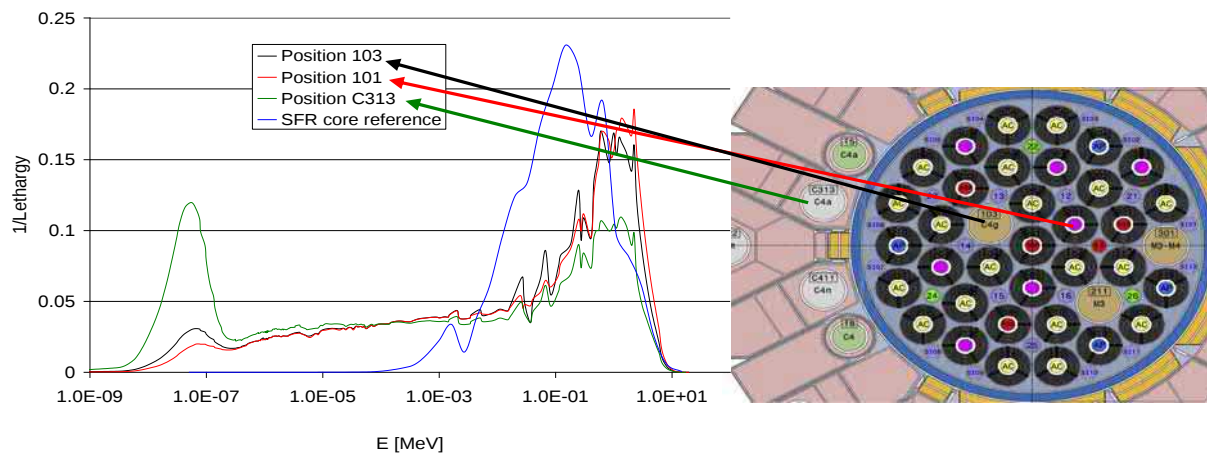


Fig. 3 Neutron flux normalised per unit of lethargy $[\text{Log}(E_{\text{ref}}/E)]$ versus Neutron Energy for JHR core. Courtesy CEA.

- Four to six water channels (depending on the core configuration) through the reflector are equipped with displacement devices to accurately control the distance to the core and therefore the irradiation flux (e.g., for an accurate stable power, for power ramps, or for power cycling).

As shown in the Fig. 3, the JHR design allows both high thermal neutron flux and fast neutron flux allowing high performance experiments of fuel and structural material.

The JHR experimental capacity will also include non-destructive examination (NDE) benches: a coupled gamma scanning and X-ray tomography bench located in the reactor pool, a similar bench located in the storage pool of the Nuclear Auxiliary Building (and a neutron imaging system located in the reactor pool).

JHR construction update status as of 2020

The civil work of the reactor was completed in spring 2018 and the project is now in a new phase focused on “components integration”. Engineering studies were contracted to Technicatome, which supervises the construction site, and which is also in charge of providing key reactor components. More than 20 other suppliers in the fields of civil works, mechanics, heating, ventilation, air-conditioning, and electric components contribute to the construction of the facility. In March 2020, a new organization was put in place consisting of CEA, Technicatome and Framatome in an integrated team for enhancing this second phase of mechanical erection and completion of the reactor.

JHR safety design

As a new construction facility, JHR incorporates safety analysis right from the design phase, based on a modern reference system and methodologies. It can be linked to those used in contemporary projects such as the current next generation NPPs under construction, but adapted to the characteristics and situation of a research reactor project.

Following the Fukushima-Daichi Accident (March 2011), the French Regulator (Autorité de Sûreté Nucléaire-ASN) also asked CEA to perform complementary safety assessments to meet objectives under extreme situations exceeding licensing basis (with focus on “cliff-edge” effect prevention; that is, avoiding an incidental scenario leading to several others) (Bignan et al., 2011). The complementary safety assessments confirmed the sound design bases of the newly designed JHR under construction. A few selected needs for extra equipment were also identified, and, as an answer to French nuclear regulatory requirements, CEA proposed a set of “hardened core” measures including for example:

- An ultimate recirculation pump on the reactor main cooling circuit.
- Pipes and valves for ultimate pool water supply from outside the containment building
- Ultimate valve actuation on some ventilation lines.
- Dedicated sensors to independently measure pressure and radio-activity level in the containment building.
- Specific ultimate power set for the above-mentioned equipment's.

Developing a modern experimental capacity

JHR is designed as a high performance MTR (thermal power up to 100 MW) with the capacity to perform about 20 experiments at the same time. Characteristics (at full 100 MW capacity) are as follows:

- thermal neutrons flux in reflector: up to $5.5 \text{ E14 neutrons cm}^{-2} \text{ s}^{-1}$
- fast neutron flux in the core: up to $5.5 \text{ E14 neutrons cm}^{-2} \text{ s}^{-1}$ for $E > 1 \text{ MeV}$ and/or up to
- $\text{E15 neutrons cm}^{-2} \text{ s}^{-1}$ for $E > 0.1 \text{ MeV}$
- material radiation damage: up to 16 displacements per atom (dpa) per year
- six displacement systems to adjust fissile power and perform power transients
- power transients for fuel limit to clad failure studies: linear power up to 600 W cm^{-1} .

At nominal operation JHR is to operate with 8 cycles a year (representing about 180 EFPD-Equivalent Full Power Days).

CEA with its partners is preparing the first experimental capacities by developing some modern experimental devices for fuel and material behavior studies under irradiation such as the following (Gonnier et al., 2018; Young, 2021):

- the MADISON™ loop for fuel investigation under normal condition (evolution of the fuel micro structure, clad corrosion, fission gas releases ...),
- the ADELINÉ™ loop for power transient studies (ramps) allowing clad failure for up-to-limit situations,
- the LORELEI™ loop for safety LOCA (Loss Of Coolant Accident) studies for accidental scenarios,
- the MICA™ capsules for material investigation under high fast neutron flux and high dpa rate,
- the MELODIE™ device for on-line bi-axial constraint analysis (analysis in both axial and radial directions) on material,
- the OCCITANE™ device for RPV (Reactor Pressure Steel) degradation studies,
- the CLOE™ loop for material corrosion studies.

The following Figs. 4 and 5 indicate the location of some experimental devices within JHR core and reflector.

Compared to the existing experimental capacities worldwide, a great effort is ongoing to improve the performance of such loops and to develop new devices with innovative concepts by

- better monitoring and follow-up of the irradiation conditions,
- having much more on-line instrumentation to measure key parameters (e.g., fast and thermal neutron fluxes, gamma heating, temperature, fission gas release for fuel investigation, and material elongation),
- having modern well-equipped post-irradiation examinations either directly within JHR nuclear building (for non-destructive assay) or in Cadarache Hot Laboratory (for non-destructive and destructive examinations) or in consortium members hot laboratories.

Radioisotopes production

Molybdenum 99 production

The JHR objectives for this radioisotope are to be able to produce an annual volume of 25% of European needs on an average basis and up to 50% of European needs in peak production. For this, production must be flexible according to customer's needs.

Irradiation capacity for Moly production will be flexible according to customer's orders and may be extended for limited periods as following:

- Weekly maximum capacity: up to $4800 \text{ }^{99}\text{Mo Curies (Ci)}/6\text{-days week}$
- Yearly maximum capacity: up to $115,200 \text{ }^{99}\text{Mo Ci/year}$.

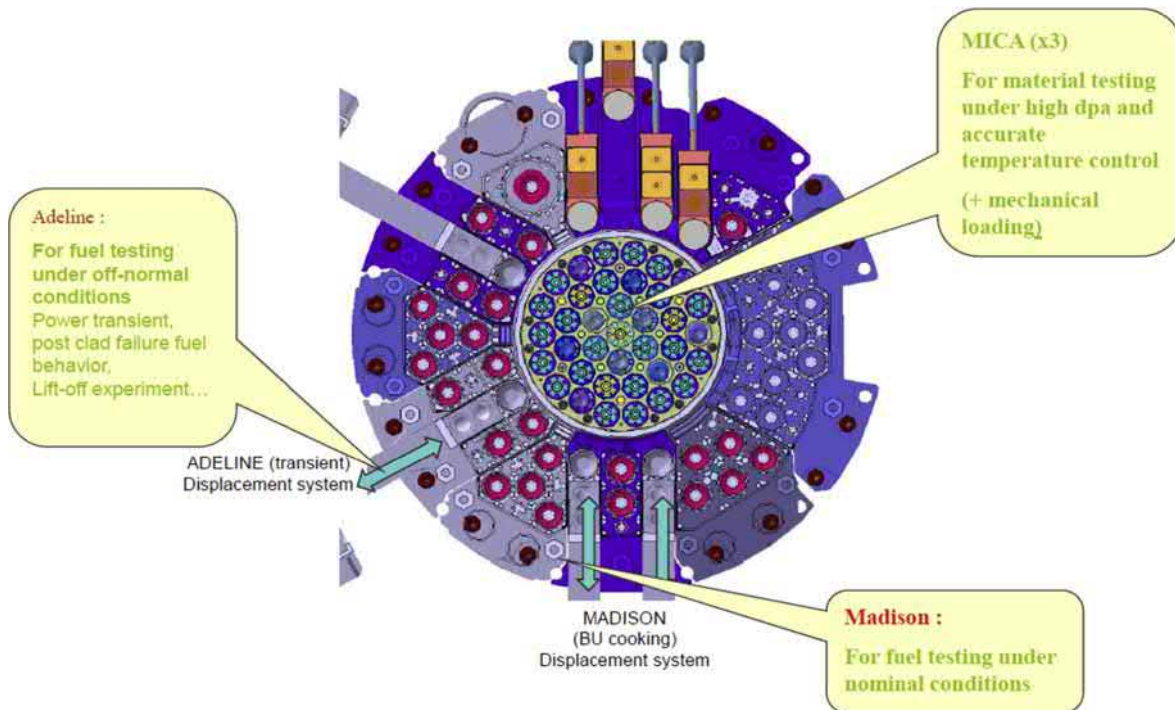


Fig. 4 Location of several experimental devices in JHR core and reflector.

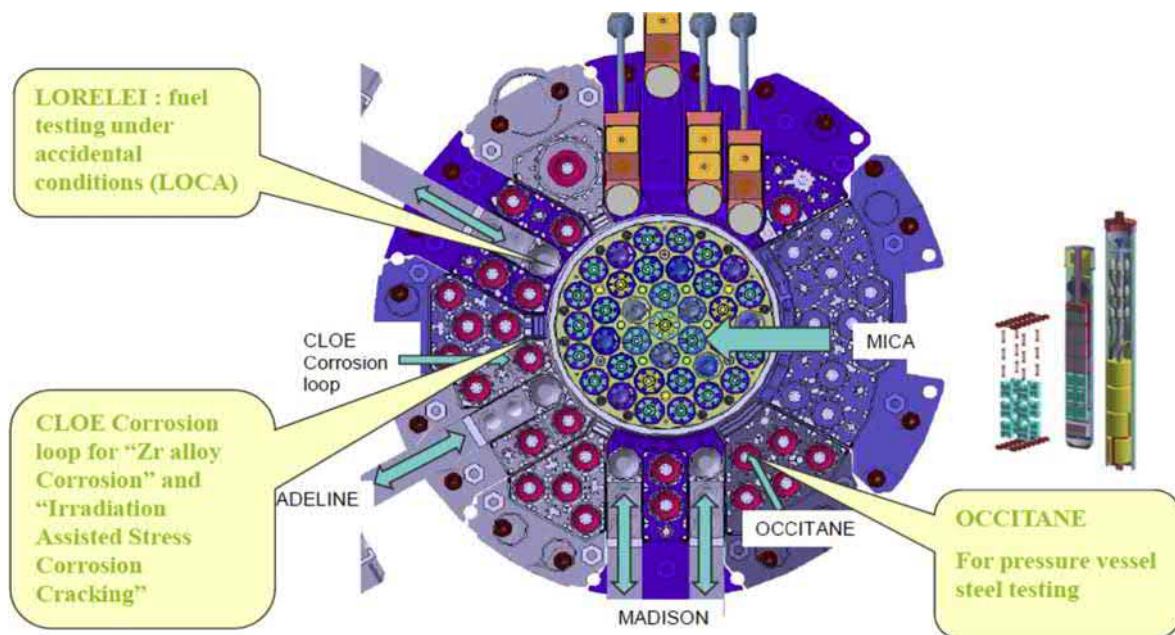


Fig. 5 Location of several experimental devices in JHR core and reflector.

The Moly facility has been designed for this production capacity, while integrating safety and operational constraints. From the beginning, the Moly irradiation devices will use Low Enriched Uranium (LEU) targets.

The facility is composed of two main parts:

- the in-pile part in the reactor pool, which includes four Moly device vessels with the target holder, a movable system, underwater lines up to the piping penetrations and two safety pumps;
- the out-of-pile part is composed of the piping penetrations allowing the junction between the in-pile cooling circuit and out-of-pile cooling circuit, equipment for instrumentation and control system (I & C) and power supply.

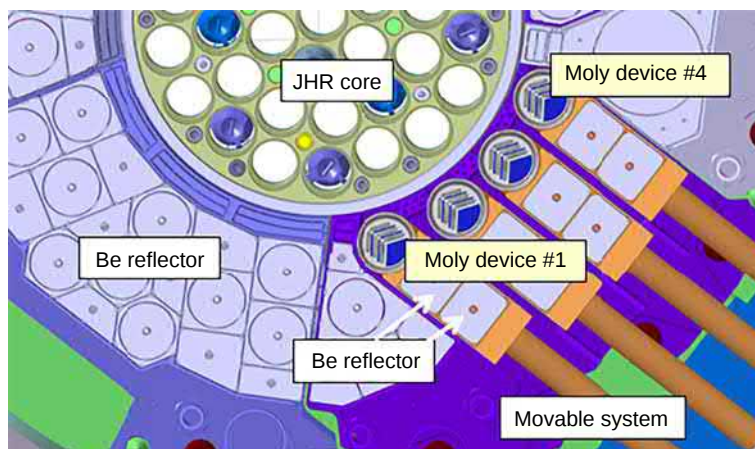


Fig. 6 Implementation of the Moly devices in the JHR reflector.

In the JHR beryllium reflector, a reflector sector has been designed to accommodate the four Moly devices and their movable system (Fig. 6 below), in order to provide a large production capacity with the greatest flexibility.

Other radioisotopes

Other isotopes, such as ^{131}I and ^{133}Xe , are produced in the uranium targets through fission reactions and could be extracted. Such radioisotopes are important for nuclear medicine applications. In parallel and in accordance with customer needs, feasibility studies could be performed in order to develop the production equipment for radioisotopes production based on neutron capture reactions. Many irradiation positions are located in the Beryllium reflector. The potential thermal neutron fluxes would allow JHR to produce by neutron activation ^{90}Y , ^{153}Sm , ^{166}Ho , ^{169}Er , ^{177}Lu , ^{186}Re , ^{192}Ir , ^{103}Pd , and ^{125}I (as examples) for medical purpose. Production of radioisotope in fast and epithermal neutron fluxes could also be investigated. Finally, JHR could also produce radioisotopes such as ^{192}Ir , ^{60}Co , ^{75}Se and ^{169}Yb for industrial purposes. Due to JHR neutrons fluxes, high production rates are expected.

JHR as an international user-facility through international joint programs and/or academic support

Parallel to the completion of the reactor construction, the establishment of an international community around JHR is continuing. This is important because, as already indicated, building and gathering a strong international community in support of MTR experiments is a key-issue for R&D in the nuclear energy field.

According to the consortium agreement, JHR will become a user reactor at an international level with multinational projects and proprietary experiments. As anticipated preparatory actions, the JHR consortium has set-up a yearly scientific seminar and three working groups (Fuel, Material and Technology) to identify R&D topics of common interest and to prepare the first international joint programs addressing fuel and material issues that are key for operating plants and future NPP (mainly focused on LWR).

This latest seminar (the 10th seminar was held in April 2019) gathered about 100 participants and is a unique opportunity for the future end users to share and discuss progress on the latest developments on JHR experimental capacities. The main outputs of such seminars are to identify scientific needs leading to proposal of future R&D programs with precise requirements regarding the management of irradiation conditions and the performances of the instrumentation of the experimental devices. Current planning focus is on the mastering and follow-up of the irradiation conditions in terms of local neutron flux and spectrum, temperature homogeneity and thermal gradients, with the possibility to un/re/load irradiated materials in the JHR experimental devices.

The experimental priorities are directed toward Generation II-III nuclear power systems with needs related to:

- LWR fuel element performance under nominal operation (fuel rod, cladding, fuel assembly) for various new fuels (UO₂ with additives, burnable absorbers) and clad candidates, including Enhanced Accident Tolerant Fuels, for various chemical environments and various power conditions (power cycling, load following and extended operation at reduced power level).
- LWR fuel testing in normal operational conditions: power ramps (crack initiation and propagation, clad integrity thresholds, pellet-cladding chemical interaction, fission products release and radiological source-terms), power to melt, and fuel assembly lift-off phenomenon.
- LWR fuel behavior under accident conditions such as LOCA-type conditions (clad ballooning and hydriding, fuel fragmentation, pulverization and ejection, grid-spacer effects, radioactive source term, and fuel assembly geometry after quenching).

For the LWR fuel assemblies and core structures, special attention is given to behavior of guide tubes and grid spacers: grid-spring interaction, guide tubes axial creep, and effect of rod bowing. For the control rods and burnable absorbers: thermal-mechanical and

geometrical evolutions and neutron absorber consumption (Young, 2021). Specific irradiation testing needs related to core materials:

- For the cladding: Mechanical behavior (creep test) with on-line biaxial (axial and radial) control (stress and strain), effect of environment on mechanical behavior, effect of dose accumulation on cladding (microstructure, hardening, embrittlement, creep for Gen II/III, and microstructure, swelling, embrittlement, creep for Gen IV).
- For the Reactor Pressure Vessel (RPV): Dose accumulation with the Irradiation effect on the microstructure and mechanical properties.
- For core internals: dose accumulation and environment effect on mechanical behavior.
- For Absorbers: Dose accumulation, effect of irradiation on the microstructure.

The JHR scientific team at Cadarache is already welcoming scientists, engineers (called *Secondees*) from various organizations and institutes, who are integrated within the JHR team for a limited period of time (typically 1 year) for various projects such as physics studies for the development of the experimental devices (neutron physics and thermal-hydraulics) and for support to future operations (e.g., safety analyses).

This Secondment program is important for countries willing to invest in nuclear technology as it helps to create and sustain key technical competences.

Conclusion

JHR will be a major international research infrastructure regarding fuel and material behavior under neutron irradiation; it will perform R&D programs for the optimization of the present generation of Nuclear Power Plants (NPPs) but also support the next generation of NPPs.

The JHR construction will continue with completion expected in accordance with project plans to start normal operations by the second part of this decade. Beyond current construction activity, the facility, especially regarding the experimental capacity, is already open (and will be more and more so in the future) to international collaboration: As some examples of current collaborations are the output of the various working groups who identified R&D topics of common interest such as:

- LWR fuel testing up to limits and in incidental conditions (fuel element thermal-mechanical behavior, fuel-cladding chemical interaction, fission gas and volatile fission products release and associated source term, transient swelling)
- Dose accumulation in low-alloyed steels for reactor pressure vessel and new cladding studies for accident tolerance.

Typically, such R&D programs could be managed through international joint programs open to non-Members of the JHR consortium. To summarize, JHR prepares to be a key research and development infrastructure in the European and International nuclear materials research areas to support the use of nuclear energy during this century.

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FRM II (20 MW Germany)

Heiko Gerstenberg and Winfried Petry, Technische Universität München, ZWE FRM II, Garching, Germany

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Introduction

The nuclear era in Germany started on June 11, 1956 when Prof. Heinz Maier-Leibnitz of the Technische Hochschule München—now the Technical University of Munich—was empowered by the Bavarian Minister-President to purchase a swimming pool reactor in the United States of America. This research reactor, called Forschungsreaktor München (FRM), was built in Garching, at that time a small village close to Munich, and taken into operation at a power of 1 MW on October 31, 1957. During the next decades it was upgraded stepwise to a power of 4 MW and became the starting point of many important applications and developments in neutron physics, e.g. neutron guides, ultra-cold neutrons, high resolution spectroscopy, gravity spectrometry using neutrons, neutron irradiations at liquid He temperature, hadron therapy of tumors by fast neutrons, etc. Last not least it was the nucleus of the present large university campus in Garching, populated every day by thousands of students and employees.

With the success of new very intense neutron sources in the United States and Europe, it turned out that the FRM did not meet the requirements for a modern research reactor anymore. Consequently, in 1993 the decision was taken by the Bavarian government to replace the Atomic Egg, as FRM is commonly called, by a high performance neutron source on the same site. The basic design feature of the Forschungsneutronenquelle Heinz Maier-Leibnitz (FRM II), how this new reactor is called, is its single cylindrical compact fuel element providing a thermal neutron flux density of up to $8 \times 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$ (unperturbed) at a moderate thermal power of 20 MW. Its first criticality was achieved on March 2, 2004 and its specified normal operation for scientific users and interested commercial parties started in May 2005 (Schreckenbach and Gerstenberg, 2005). The present external appearance of the reactor site at the Campus in Garching is shown in Fig. 1. For actual information see FRM II homepage.

The dedicated mission of the FRM II is to provide neutrons for all kinds of scientific, industrial and medical applications. For this purpose, it is equipped with 11 beam-tubes and 5 irradiation facilities. In addition to the use of thermal neutrons a cold and a hot neutron source operated at $T = 24 \text{ K}$ and $T > 2200 \text{ K}$ respectively are available to provide the desired neutron spectrum for any kind of beam tube experiment. Further, a setup of two fuel plates located at the edge of the moderator tank “converts” thermal neutrons into an intense beam of fission neutrons ($2.3 \times 10^8 \text{ n}_f \text{ cm}^{-2} \text{ s}^{-1}$) in a narrow energy range around 2 MeV. Moreover, the FRM II is equipped with the worldwide most intense source for thermal positrons ($10^9 \text{ e}^+ \text{ s}^{-1}$) being generated in a beam-tube after thermal n-capture in Cd and subsequent pair-production from high energy gamma quanta. Finally, the installation of an ultra-cold neutron source in a beam tube crossing the moderator tank is in progress.

As of today about 30 beam-tube instruments are in operation at the FRM II, built and maintained by skilled scientists from TUM, the Helmholtz centers Forschungszentrum Jülich and Geesthacht, the Max Planck Society and nine other German universities. Nineteen of these instruments are fed by the cold source, clearly indicating the dominant interest of today's neutron science on long wavelength neutrons. Since 2011 the scientific exploitation of the neutron source happens under the umbrella of the Heinz Maier-Leibnitz Zentrum (MLZ) (Petry, 2019). Actual information is presented on the MLZ homepage. Twice a year measuring proposals are reviewed by a committee of independent international experts. Measuring time at MLZ is allocated solely based on scientific merit. Often, however, the applicants are suffering from the overbooking of the requested instrument by a factor 2–3.

Although the FRM II has clearly been optimized by its design to provide excellent conditions for beam-tube experiments using cold, thermal, hot and fission neutrons there was a consent already during the planning phase that it has to be made available for industrial, technical and medical applications as well. Consequently, commercial companies are using the FRM II as a service provider for the production of radioisotopes or the neutron transmutation doping of Si. In particular, the demand for radioisotopes to be used in nuclear medicine has increased considerably during the last few years. Among other considerations, this development led to the decision to launch a licensing procedure for the production of $^{99}\text{Mo}/^{99\text{m}}\text{Tc}$. This facility is supposed to contribute considerably to the growing European needs for $^{99\text{m}}\text{Tc}$ in the near future.



Fig. 1 View of the FRM II showing from left to right: the neutron guide hall (east), the FRM II reactor building, the neutron guide hall (west) and the “Atomic Egg” in the background. Wenzel Schuermann/TUM



Fig. 2 View of the FRM II compact fuel element from different perspectives. FRM II/TUM

Nuclear reactor design

The entire reactor core of the FRM II consists of a single very compact fuel element with a maximum outer diameter of 23.7 cm and an active length of 70.0 cm. It contains approximately 8 kg of uranium in the form of U_3Si_2 grains embedded in an Al matrix with a maximum uranium density of 3 gU/cm^3 (see Fig. 2). Like other high performance research reactors, the FRM II is fueled with highly enriched uranium at a degree of enrichment of $>90\%$ in U-235. Each fuel assembly is built up of 113 involute-shaped fuel plates that are clamped between two concentric tubes made from AlMg3 with diameters of $\varnothing_i = 11.8 \text{ cm}$ and $\varnothing_o = 23.7 \text{ cm}$. This geometry guarantees the cooling slits between two adjacent plates to be of equal width over the entire diameter of the fuel assembly. The fuel assemblies are manufactured by the company FRAMATOME-CERCA in France.

The fuel element sits in a light water cooled central channel, which separates the light water from the surrounding heavy water being kept in a moderator tank with height and diameter of equally 2.5 m. Due to the compactness of the fuel element the moderation of the neutrons takes place mainly in the surrounding moderator. This setup makes it necessary to reduce the fuel density in the outer part of the fuel element from 3 to 1.5 g cm^{-3} in order to avoid unacceptable power peaks at its outer surface. On the other hand, due to the same effect the maximum of the thermal neutron flux density shows up some 12 cm outside the core region in the

moderator tank. This feature provides the important advantage that major experimental installations like the cold and the hot neutron source may be installed right in the maximum of the neutron flux density.

Originally, the lifetime of a fuel element in FRM II was 52 full power days (1040 MWd). Fortunately, however, extensive irradiation experiments for the certification of the fuel had been carried out already during the project phase. Their results covered the safe operation of the fuel for a higher burn-up. Consequently, taking advantage of the experience from the first 12 reactor cycles and after evaluating further important parameters like e.g. the efficiency of the control rod at the end of cycle, the maximum burn-up was extended to 1200 MWd or 60 full power days. Ideally, the operational schedule of the FRM II consists of four reactor-cycles per year of equally 60 days in a row.

As mentioned above each reactor-cycle starts with the installation of a fresh fuel element. Consequently, an external neutron source is required to start the reactor. For this purpose, the FRM II is equipped with a primary source based on the spontaneous fission of ^{252}Cf . Since, however, the half-life of ^{252}Cf is 2.65 years only and an exchange of the ^{252}Cf source is expensive and technically challenging a secondary starting source has been installed that uses the photo neutrons being emitted by Be under intense gamma radiation. In more detail: The secondary neutron source consists of 55 compressed pellets containing an equal volume of Sb and Be. The source has a total length of 825 mm and is installed in the moderator tank in a distance of 340 mm from the center of the fuel element. During reactor operation the stack of Sb/Be-pellets is irradiated by about $10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$ and the resulting activation product ^{124}Sb ($T_{1/2} = 60.2$ days) provides the necessary intense gamma radiation to induce the emission of photo neutrons from the Be.

The reactor FRM II is equipped with a single control rod dedicated to regulate the reactor power and to compensate changes in reactivity due to increasing burn-up or variations in the concentration of fission products acting as neutron poisons. In addition, the control rod system is one out of two independent systems that allow to shut down the reactor and to keep the fuel assembly permanently subcritical at any time. Because of their crucial importance, both shutdown systems are implemented in the reactor safety instrumentation and its control and communication system.

The control rod system consists of three major components: drive, connecting rods and the control rod itself, which uses Hf as neutron absorber. During standard reactor operation the control rod is driven with a velocity of approximately 0.08 mm s^{-1} within the inner tube of the fuel assembly in order to balance the reactor power. In case of serious interferences or emergency the control rod drops and the reactor is shut down.

Formally, the control rod is the secondary shut down system of the reactor FRM II. The primary shut down system consists of five independent units being installed on a common radius on top of the heavy water moderator tank. An individual casing protects each of them against contact with the light pool water. Like the control rod, the absorbers of the shutdown-rods are tubes made from Hf. During reactor operation, all shutdown rods are held in their top position by electromagnetic coils. In this position they are dipped into the moderator tank above the fuel element. In case, a relevant signal from the reactor safety instrumentation initiates a SCRAM the holding current of the magnetic coil is interrupted. In this case, stretched springs assisted by gravity accelerate the shutdown rods into their lowest position. In parallel, also the control rod drops into its lowest position. No need to say, of course, that independently of each other the control rod as well as the shutdown rods provide sufficient reduction in reactivity to shut down the reactor reliably and keep it permanently in a subcritical state. Fig. 3 shows schematically a vertical cut through the FRM II indicating main components required for reactor operation.

As a whole, the cooling system of the FRM II consists of three independent circuits safely separated by heat exchangers. The fuel element is cooled by pumping light water at a rate of 300 kg/s through the slits between the fuel plates. Consequently, at nominal reactor power of 20 MW the temperature increase of the cooling water is limited to only 15°C , i.e. from $\approx 35^\circ\text{C}$ to $\approx 50^\circ\text{C}$ between the inlet and the outlet of the fuel assembly. Four primary cooling pumps are necessary to provide the cooling stream. By means of two heat exchangers the nuclear heat is transferred to a secondary cooling circuit acting mainly as a barrier against the possible leakage of contamination into the environment. Finally, a tertiary cooling circuit is transferring the heat to a small cooling tower where it is released to the air.

In case of the unavailability of the standard cooling system the FRM II is relying on a threefold emergency cooling system. Its main components are three identical pumps for the removal of residual heat mounted in the storage pool. After shut down of the reactor in an emergency case each of them is independently capable to provide sufficient cooling to avoid any damage by overheating the fuel assembly. The pumps are battery-buffered in order to allow their operation for at least 3 h. After that period two emergency check valves open automatically due to gravity and the further cooling is driven by the thermosiphon effect. For this purpose, the sieve, a deep lying pipe section of the primary cooling circuit, is intentionally perforated. This construction allows pool water to be sucked through the fuel element due to its residual heat and to be returned into the pool through the check valves. The volume of light water in the reactor pool amounts to about 700 m^3 and is by far enough to provide sufficient cooling after an unintended shutdown for the fuel assembly in the core as well as for all of the in maximum 50 spent fuel assemblies in the pond for any necessary timeline.

A schematic of the FRM II cooling circuits is given in Fig. 4. Further, Fig. 5 provides a view to the reactor hall.

Experimental support systems

The compact core of the reactor with its surrounding heavy water moderator creates a source term of thermal neutrons with a shape of a hollow cylinder where the maximum thermal flux is some 12 cm away from the outer side of the central channel. This design

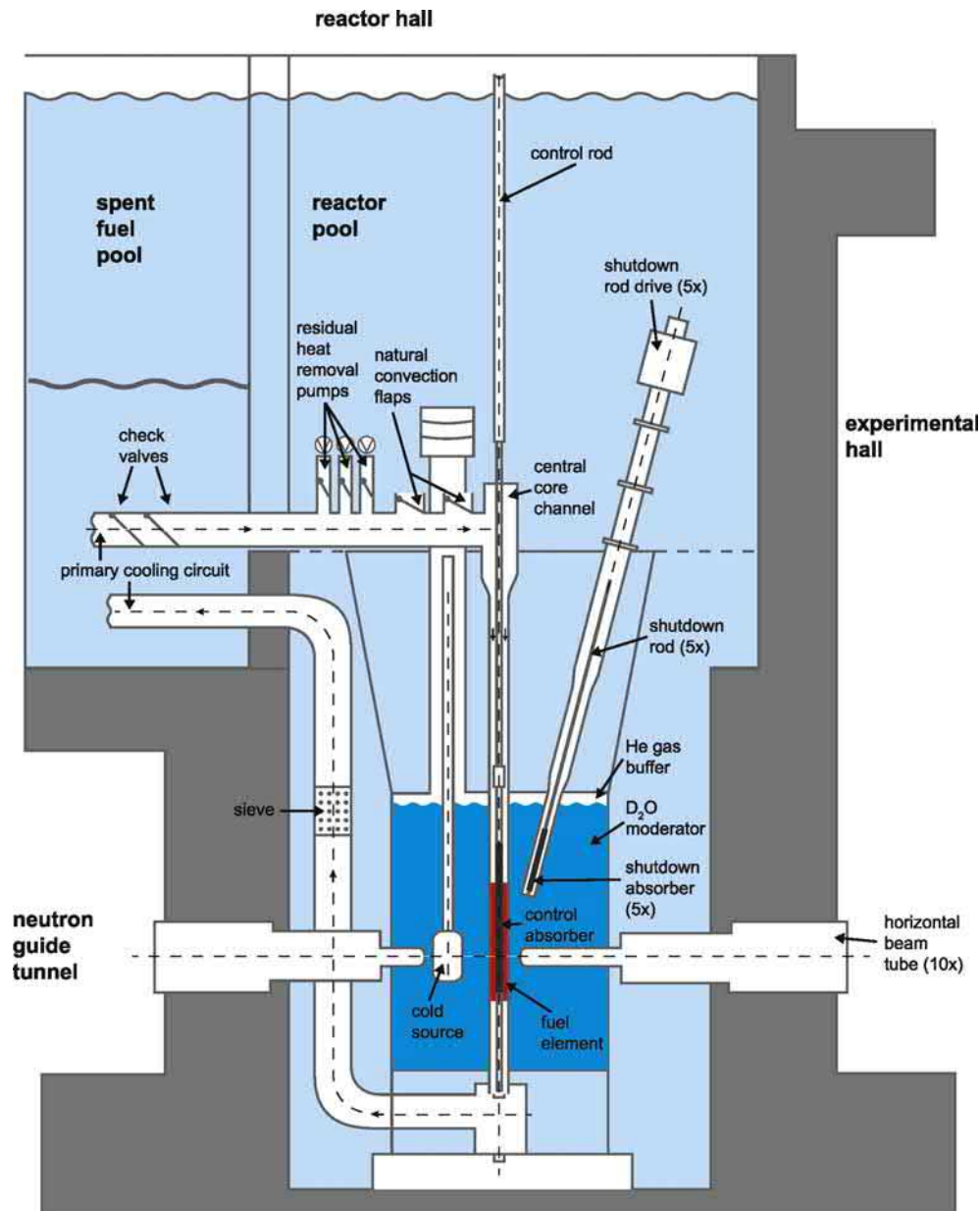


Fig. 3 Schematic vertical cut through the FRM II reactor indicating major technical components and scientific support systems. FRM II/TUM

provides an ideal geometry for the extraction of neutrons by beam tubes and the installation of spectrum shifters in an optimum position. Further, all beam tubes extracting thermal neutrons are oriented tangentially with respect to the cylindrical source term, thus giving no direct view to leaking fast neutrons towards the beam ports outside the primary concrete shielding. This is one important measure to suppress background from fast neutrons. Like any light microscope neutron instrumentation does not so much ask for intensity, but for brilliance, i.e. neutrons per cm^2 , s, rad and $(\Delta\lambda/\lambda)$. One important measure to achieve this is shifting the neutron spectrum already at the source to the wavelength range needed for a particular instrument or experiment. At FRM II neutrons are shifted to cold (24 K), thermal (60 °C), hot (2000 °C), to fission spectrum (2 MeV), in near future to ultra-cold (mK) and even “converted” to positrons. Eleven beam tubes are available to extract all these particle beams.

The by far most important experimental installation in the moderator tank is the Cold Neutron Source (CNS). Its major component is the moderator chamber located in a distance of 400 mm from the center of the fuel element right in the maximum of the neutron flux density. During cold operation, it contains about 12 l of liquid deuterium at a temperature of ≈ 24 K. A He refrigeration device provides the required cooling power of about 7 kW at 20 MW reactor power. It feeds the cold part of a He/D₂ heat exchanger at a temperature of ≈ 19 K. The deuterium is liquefied in the heat exchanger and rinses down into the moderator chamber where it evaporates again due to the nuclear heat. Finally, by means of the thermosiphon effect the deuterium gas is

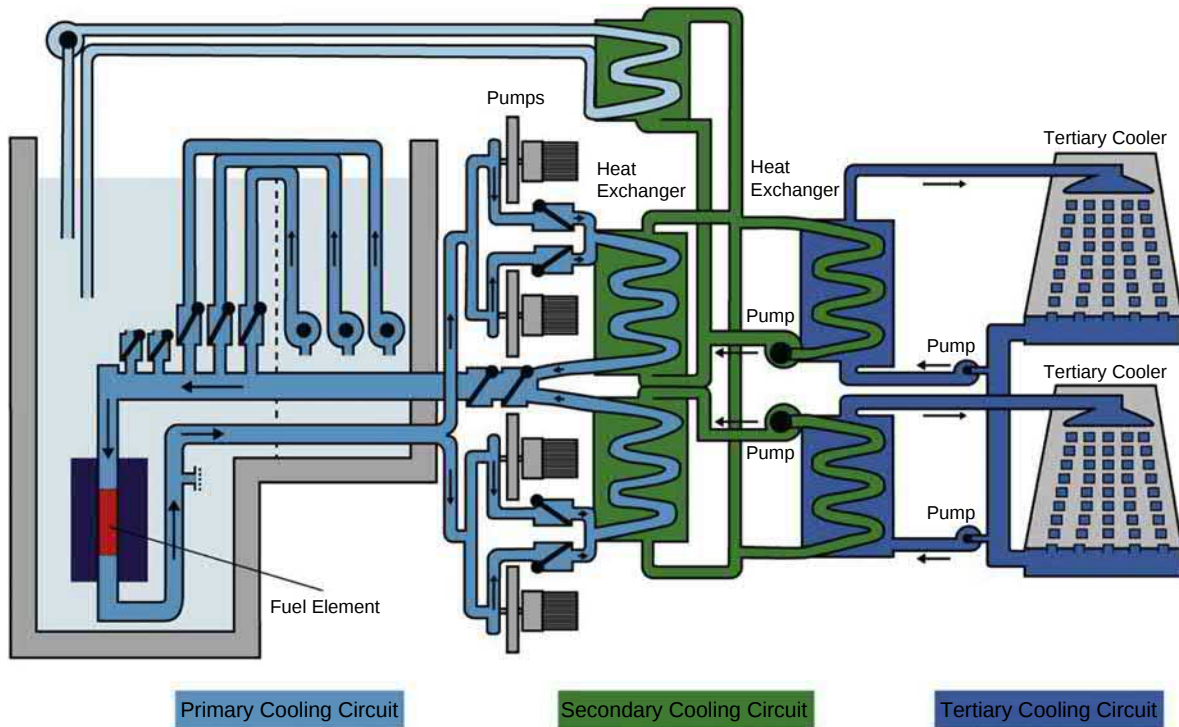


Fig. 4 Scheme of the three independent cooling circuits of the FRM II. FRM II/TUM



Fig. 5 View into the FRM II reactor hall. Bernhard Ludewig

led back to the heat exchanger for re-liquefaction. Important out of pile components of the CNS are the He-refrigerator with its compressor, a buffer tank and hydride storage tank to preserve the deuterium during maintenance periods of the reactor when the CNS is heated up. The CNS feeds three beam-tubes and 19 scientific instruments in the neutron guide halls and experimental

hall of the FRM II. Its spectrum is slightly undermoderated with the advantage of providing a broad wavelength spectrum from about $1.8 \text{ \AA}$ to $> 10 \text{ \AA}$. It is important to note that in order to prevent the CNS from thermal damage it must be in operation before the start of the reactor, i.e. malfunction of the cold source automatically triggers an unavailability of the reactor (Müller et al., 2005).

The second spectrum shifter in the FRM II moderator tank is the Hot Neutron Source (HNS). It is dedicated to the provision of short wavelength neutrons in a wavelength range of $0.3\text{--}1 \text{ \AA}$ mainly for the structural investigations in condensed matter samples. As only a small fraction of those neutrons is present in the thermal spectrum a hot moderator is required for their up-moderation. At the FRM II this hot moderator consists of a cylindrical graphite block ($\varnothing = 200 \text{ mm}$; $h = 300 \text{ mm}$) with a weight of about 14 kg . It is contained in a double-walled zircaloy container. The inner volume of this container houses the carbon block being held in position by carbon felt with a low heat conductivity and kept under an inert gas atmosphere at low pressure. In contrast, the volume between the zircaloy walls of the container is kept under about 3 bar He atmosphere in order to guarantee the detection of a possible leak in one of the walls in a timely manner. During reactor operation the carbon block heats up by the nuclear radiation from the reactor core to a value of approximately 2200 K . The temperature within the HNS is measured using a thermometer based on the thermal noise of a carbon resistor. In addition, 36 thermocouples are connected to the zircaloy container in order to supervise its temperature and, as a by-product, indirectly the correct position of the carbon block within the container. The HNS feeds hot neutrons into one beam tube, which supplies two single crystal diffractometers in the experimental hall of the FRM II. The different neutron spectra provided by the CNS, the HNS with respect to the thermal neutron spectrum are shown in Fig. 6 (Müller et al., 2005).

In order to obtain a high intensity neutron beam with an unmoderated fission spectrum, an arrangement of two fuel plates containing in total about 540 g of highly enriched uranium, the converter facility, is inserted as a secondary source in front of one beam tube. Slow neutrons from the moderator tank interact with the uranium in those fuel plates and generate fission neutrons for medical and technical purposes. Without moderation, the neutrons are led through a horizontal beam tube into a room dedicated to the treatment of cancer by direct irradiation with fast neutrons (Wagner et al., 2008). Supplementary, the converter facility can also be used for the radiography and tomography of technical objects using fast neutrons. Furthermore, the converter plates may be lifted out of their position in front of the beam tube. In this operational mode the conversion to fission neutrons does not happen and the beam is used for radiography and tomography with thermal neutrons.

In addition to neutrons FRM II provides the world's most intense permanent source for positrons. The high intensity beam of positrons is generated by pair production from absorbed high-energy gamma-rays. This source is positioned in the cusp of an inclined beam tube, where the required gamma-rays are produced by the absorption of neutrons in an isotopically enriched cadmium layer. The pair-conversion itself occurs in proximate platinum foils (see Fig. 7). The positrons are extracted by means of electric and magnetic fields and guided to several experimental stations attached to the positron beam facility named NEPOMUC (Hugenschmidt et al., 2012). Low mass and adjustable kinetic energy makes the positron a probe perfectly suited to solid state research. As the anti-matter particle of the electron, positrons annihilate electrons with a characteristic lifetime. The emerging annihilation radiation provides detailed information of the local surroundings at the annihilation site. As an example, the positron life

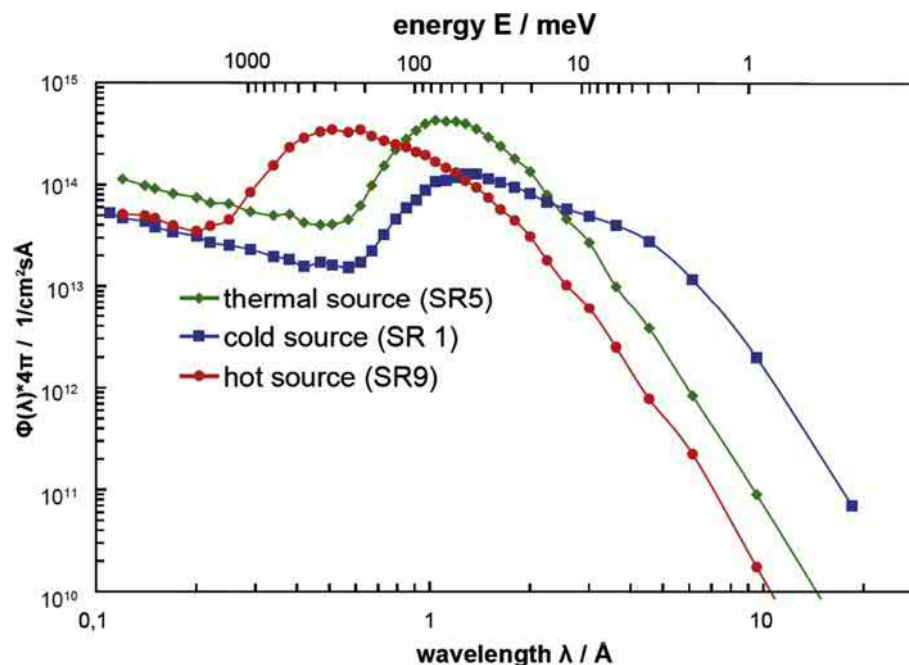


Fig. 6 Comparison of the neutron spectra in beam tubes fed by a thermal neutron source, the cold neutron source and the hot neutron of the FRM II reactor. FRM II/TUM

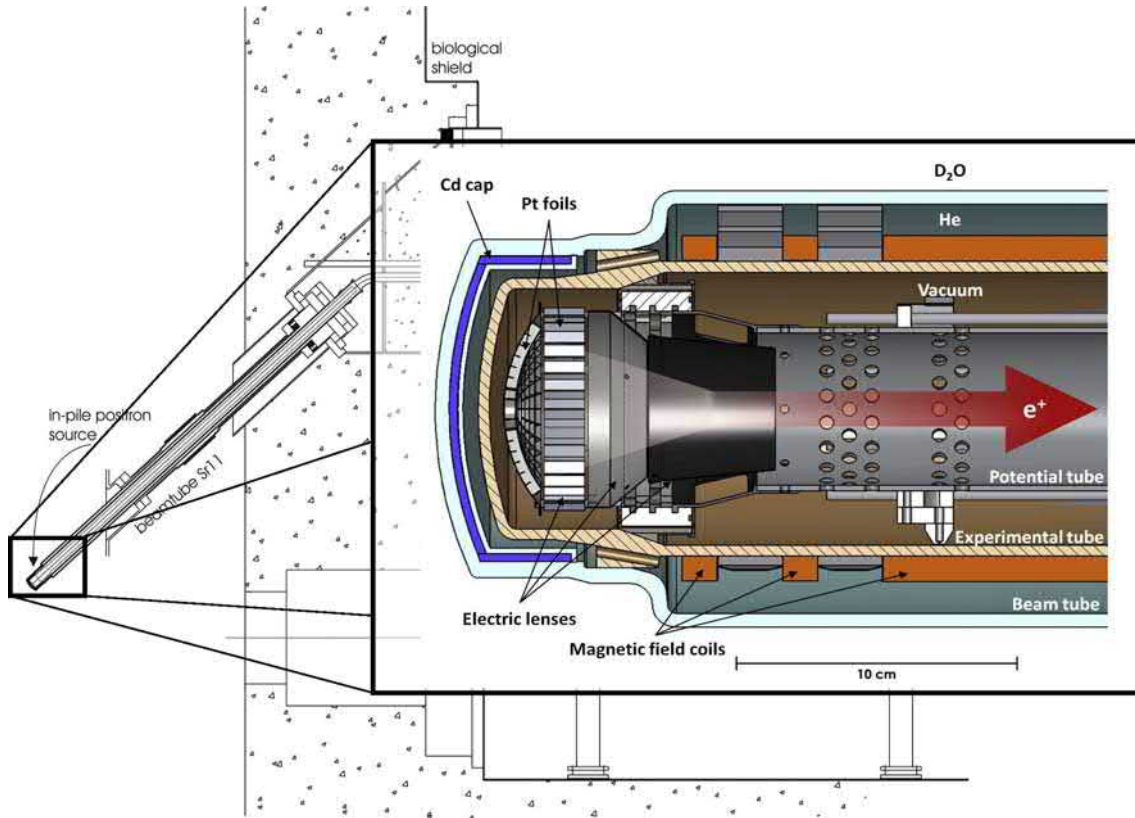


Fig. 7 Experimental setup of the in-pile positron source NEPOMUC installed in beam tube SR 11 of the FRM II. FRM II/TUM

time is slightly larger in vacancy-like defects than in the unperturbed crystal lattice. In this way, vacancy concentrations as low as 10^{-7} can be detected.

Besides the installations for neutron scattering the FRM II is also equipped with a couple of irradiation facilities. They all have in common that they are installed in light water filled thimbles reaching down from the reactor floor into the moderator tank. The biggest of those installations is the silicon doping facility. Si ingots with a diameter between 125 mm to 200 mm are moved semi-automatically from a loading station into the irradiation position being located in a vertical thimble in the moderator tank. In order to achieve uniform neutron irradiation, an indispensable prerequisite of successful transmutation doping of Si, the ingots are rotated around their central axis during neutron exposure at a rate of typically 5 turns/min. In addition, the irradiation position is equipped with a properly designed flux screen to guarantee the required axial homogeneity of the doping profile. Furthermore, the irradiation position follows the vertical shift of the flux profile with increasing burn-up of the fuel assembly. By accurate adjustment of the exposure time a resistivity between 30 Ωcm and 1200 Ωcm can be achieved with a deviation of less than 5% from target. It is noteworthy as well that the very well thermalized neutron spectrum containing only a low proportion of fast neutrons helps to avoid an undesirable high amount of extended lattice defects within the Si. Of course, the FRM II is also equipped with all necessary tools for the cleaning and release of the irradiated ingots required for the delivery to customers outside the nuclear business (Gerstenberg et al., 2009).

At the FRM II two different rabbit irradiation systems serve not only the production of radioisotopes but also the activation of samples for various topics like e.g. neutron activation analysis or the supply of small sources for education and training. The so-called capsule irradiation facility is used for long-term irradiations at a thermal neutron flux density of approximately $10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$. During irradiation the sample is typically sealed twice in a quartz ampoule and in addition in an Al-capsule with a welded lid in order to avoid its contact with pool water. After insertion into the facility the capsule is led automatically into the irradiation position by a stream of pool water for a pre-determined irradiation time. After completion of the irradiation it is returned the same way into a position dedicated to allow the decay of short lived isotopes. For further processing the capsule irradiation facility is connected under water with the hot cell where the irradiation capsules are opened and the samples are packaged according to the requirements of the regulations for the transport of radioactive goods on public roads. A high demand for this type of irradiation service is presently coming from customers of the radiopharmaceutical industry in particular companies interested in the production of ^{177}Lu (Gerstenberg et al., 2013).

The second rabbit irradiation system at FRM II is a standard pneumatic dispatch using CO_2 as process and transport gas. The samples are contained in individually labelled polyethylene capsules foreseen for single use. The facility provides six individual irradiation channels exhibiting thermal neutron flux densities between $3 \times 10^{12} \text{ cm}^{-2} \text{ s}^{-1}$ and $5 \times 10^{13} \text{ cm}^{-2} \text{ s}^{-1}$. A typical

application is the supply of samples for neutron activation analysis. For this purpose, the facility is connected directly with the neighboring Institute for Radiochemistry (Lin et al., 2006). On the other hand, the facility is used for production of isotopes as well, e.g. the activation of ^{166}Ho dedicated to the treatment of liver cancer.

Guiding neutrons to the instruments

The MLZ makes extensive use of modern neutron guides to transport and distribute the neutrons over large distances in the Experimental Hall, situated around the primary reactor block, as well as in the Neutron Guide Halls West and East. Radiation resistant metallic neutron guides start neutron extraction already within the primary shielding of the reactor block. From the beam ports at the outside of the primary shielding onwards coated glass guides continue neutron transportation to the instruments. Adapted to the needs of the instruments with respect to wavelength distribution and angular dispersion the guide elements are coated with ^{58}Ni or supermirror coatings with m values up to 3.0; on focusing sections up to $m = 3.6$. The value of m refers to the increase of the angle of total reflection for a given supermirror with respect to the angle of total reflection for elemental Ni. For example $m = 2$ means a fourfold increase in transported neutrons compared to an elemental Ni coated guide. Of course, according to the Liouville theorem increase in m goes with a similar increase in divergence. Because the measurable signal, i.e. the ratio of peak-to-background is the interesting quantity in an experiment and not alone intensity in the peak, stringent background suppression is priority in the lay-out of the guide systems at MLZ. For that purpose:

- (i) All guides are curved in order to have a cut-off for undesired short wavelength neutrons. For some instruments in Guide Hall West even S-shaped curvatures are used to strictly suppress fast neutrons running along the outer side of a one-sided curved guide.
- (ii) The value of m is chosen in a way, that only the needed divergence is delivered to the particular instrument.

Further, extreme attention is paid to shielding of the neutron guides.

Cold neutron guides: Beam tube SR-1 facing the cold neutron source delivers the neutron beams for the entire Neutron Guide Hall West. The in-pile unit of SR-1 consists of a mirror box with $m = 2.2$ supermirror coating on Al plates and a dividing section of 2.1 m length, where the beam is divided into the six principal neutron guides. Each of those is further split in order to serve a maximum number of instruments, especially with terminal positions. Altogether 15 instruments are served out of SR-1. Two other beam ports looking in the experimental hall face the cold source. SR-2 serves the three-axes spectrometer PANDA. SR-4 serves the imaging station ANTARES and in the near future the nuclear and particle physics beam line MEPHISTO in the Neutron Guide Hall East.

Thermal neutron guides: Beam tubes SR-8a and SR-8b are equipped with supermirror guides with coatings up to $m = 3$. At SR-5b a polarizing supermirror bender provides the instrument TRISP with polarized neutrons. In the near future, elliptical focusing thermal guides will provide neutron beams for the instruments in the Neutron Guide Hall East.

The entire setup of neutron guides and connected instruments is shown in Fig. 8 and described in Petry and Brückel (2015). The experimental stations in the Neutron Guide Hall East are presently still under construction. In addition, Fig. 9 gives an impression on the occupancy level in the experimental hall of the FRM II.

Instrument classes

As already mentioned, instruments at MLZ are designed to have highest brilliance and lowest background. A broad range of techniques is used for defining the incoming wavelength and analyzing the scattered wavelength such as double focusing Bragg crystals, time-of-flight, wavelength selectors, classical as well as resonant spin-echo. Most instruments have the option to polarize the incoming beam and to analyze the polarization of the scattered beam. As polarizing devices Heusler crystals, polarizing supermirrors and He absorption filters are in use. All instruments belong to the best of their class, some of them are unique or world champion.

Large angle diffraction: Large angle diffraction is the working horse of any decent neutron source. Questions like “Where are the atoms?” and “What is their magnetic moment?” are tackled. The diffraction instruments include the high resolution powder diffractometer SPODI (thermal neutrons), the strain scanner STRESS-Spec (thermal neutrons), the single crystal diffractometers HEIDI (hot neutrons), POLI (hot neutrons, three-dimensional polarization analysis) and RESI (thermal neutrons), the protein diffractometer BioDiff (cold neutrons) and a Laue camera (thermal neutrons). A high intensity time-of-flight powder diffractometer POWTEX (thermal) and an extreme pressure diffractometer SPHIRE are under construction in the Guide Hall East.

Small angle diffraction and reflectometry: Small angle scattering characterizes the ensemble average of mesoscopic structures from typical 1 nm to 10 μm , i.e. it fills the gap between atomic resolution and structures visible with light. At MLZ the classical pinhole instruments KWS 1, KWS 2 and SANS-1 are as long as 40 m, have large two dimensional detectors and cover routinely reciprocal space from $1\text{--}10^{-3} \text{ \AA}^{-1}$. KWS 3 works on the focusing mirror principle and extends the range in reciprocal space to $< 10^{-4} \text{ \AA}^{-1}$. The reflectometers NREX, REFSANS and MARIA cover similar dimensions like the small angle cameras, but now in reflecting geometry exploring the vertical dimension of surfaces and interfaces. Furthermore, all three instruments allow the so-called grazing incident mode thereby exploring the lateral dimension of surfaces.

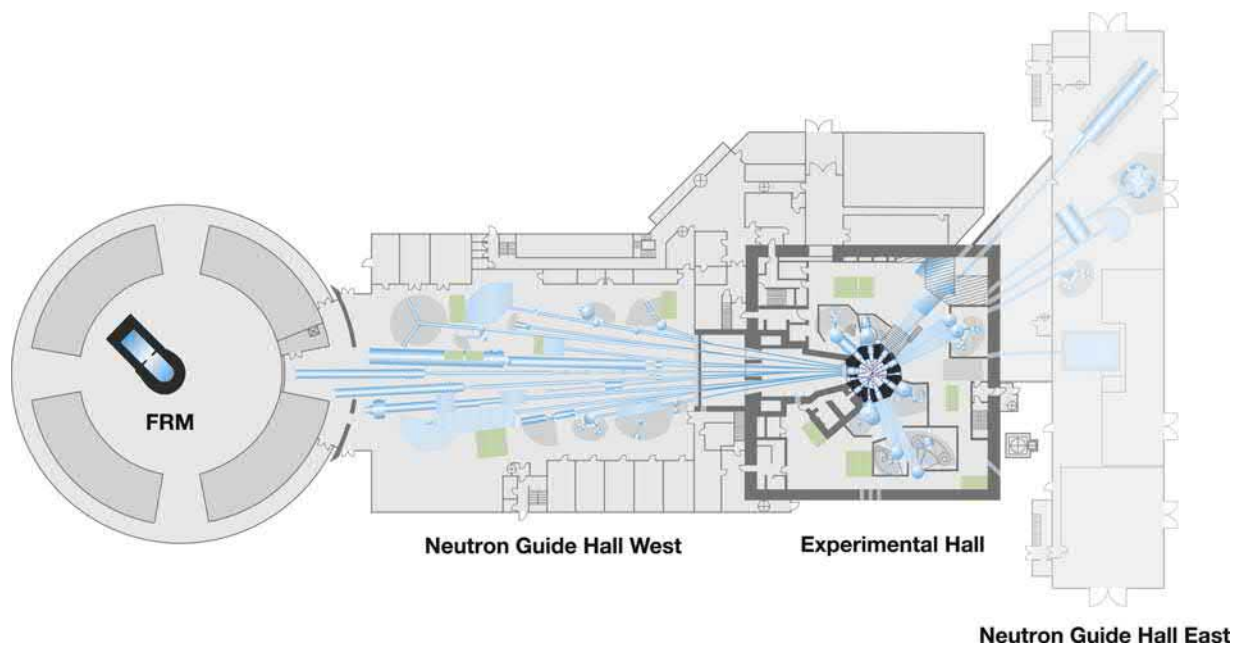


Fig. 8 Floorplan of the FRM II with the neutron guide system. FRM II/TUM



Fig. 9 View into the experimental hall of the FRM II. Bernhard Ludewig

Inelastic neutron scattering: Inelastic neutron scattering is the method of choice to tell how the atoms or their magnetic moments move. At MLZ five 3-axis spectrometers PUMA, PANDA, TRISP, MIRA and KOMPASS complement each other in time-spatial resolution and spin analysis. Among these, TRISP is the world champion for analyzing phonon lifetimes. To

reveal stochastic movements in hard as well as soft matter even higher resolution in energy or time is asked for. This is achieved by NSE (spin echo), RESEDA (resonance spin echo), SPHERES (backscattering) and TOFTOF (time-of-flight). NSE and TOFTOF stand out having the lowest background for their kind of instrument. Finally, DNS focuses on the detection of magnetic fluctuations. Under construction is TOPAS, a thermal time of flight spectrometer featuring polarization analysis and high flux.

Imaging and element analysis: whereas all above mentioned instruments measure in reciprocal space neutron imaging displaces objects in direct space. ANTARES (cold neutrons) and NECTAR (thermal and fission neutrons) are doing so by transmission measurements. Collecting typically some hundred transmission pictures under different angles enables tomography, i.e. producing a three-dimensional picture of the attenuation coefficients of the object. Highest spatial resolution achieved so far at ANTARES is 12 μm . One cold neutron beam is dedicated to Prompt Gamma Activation Analysis (PGAA). By means of focusing optics the world most intense beam of cold neutrons ($4\text{--}6 \times 10^{10} \text{ n cm}^{-2} \text{ s}^{-1}$) hits the sample with a spot size $< 1 \text{ mm}^2$. In principle, all the chemical elements can be analyzed with PGAA. The major advantage of the method is its capability in the analysis of light elements. Elemental analysis as low as 0.1 ppbw can be achieved.

Nuclear, particle and astrophysics: Going back to Heinz Maier-Leibnitz nuclear, particle and astrophysics with neutrons have a large tradition at TUM. Some of the questions in this field can be dealt with the instruments PGAA and the positron source NEPOMUC. However, the most versatile tool would be an intense beam of cold neutrons and an ultra cold neutron source. Both are currently under construction. MEPHISTO, as the cold neutron beam is called, will deliver a gold capture flux of $2 \times 10^{10} \text{ n cm}^{-1} \text{ s}^{-1}$ on a beam area of $60 \times 106 \text{ mm}^2$. The UCN source will be installed in the through-going beam tube SR 6, it will operate with a solid D_2 target and expects UCN flux densities in the order $10^5 \text{ cm}^{-2} \text{ s}^{-1}$ in an energy band of 100–230 neV. Both sources will serve for high precision experiments like the β -decay of the neutron, its life time or eventual electric dipole moment. The ultimate goal of these experiments is to find evidence for physics beyond the standard model.

Science with positrons: NEPOMUC provides a high-intensity low-energy positron beam for applications in solid state and surface physics as well as for fundamental research in nuclear and atomic physics. The intensity amounts to $> 10^9$ moderated positrons per second at a beam energy of $E = 1 \text{ keV}$. The remoderation device of NEPOMUC enhances the brightness of the positron beam and hence enables positron experiments which are highly resolved in space or/and in the time domain. The remoderator is based on the stochastic positron cooling in a W(110) single crystal and reemission of thermalized positrons into the vacuum with discrete energy. For most of the measurements the brightness enhanced positron beam is used. By a central, fivefold beam switch the positron beam is delivered to one of the five experiment beam lines:

Pulsed low-energy positron system (PLEPS), Coincident Doppler-broadening spectrometer (CDBS), Positron annihilation induced Auger-electron spectrometer (PAES), Scanning Positron Microscope Interface (SPM Interface). The fifth beam line is for the multi-purpose open beam port (OP) [<https://www.mlz-garching.de/nepomuc>].

Instrumentation at MLZ is under continuous development. The actual status and detailed information can be found on the MLZ homepage under “Experimental facilities – Heinz Maier-Leibnitz Zentrum”.

An outlook

Despite the decision of the German government to phase out of nuclear energy by the end of 2022 there is no doubt that research reactors are not affected by this development. On the other hand, as of 2020 only two research reactors are still in operation in Germany, namely the small 100 kW TRIGA reactor of the Johannes Gutenberg University in Mainz and FRM II. Consequently, science as well as technical, industrial and medical applications based on the use of neutrons will be focused even more on FRM II than today. Numerous precautions have been taken in order to guarantee the availability and attractiveness of FRM II for a long-term future.

With respect to operation the supply of fuel elements and the external storage of spent fuel have been secured by means of long-lasting contracts. The scientific use will be extended by means of one or maybe even two additional neutron guide halls housing additional experiments. The fleet of secondary sources is about to be extended in the near future by an ultra-cold neutron source providing neutrons with temperatures in the mK range.

In parallel, also the use of neutrons for industrial and medical applications will be of growing importance. Neutron scattering will continue to provide important contributions to material science dealing with questions as different as the better understanding of the charging/discharging of Li-ion batteries or the knowledge about the ultimate lifetime of industrial components. The irradiation facility dedicated to the production of ^{99}Mo is under construction and will contribute considerably to satisfy the growing European need for $^{99\text{m}}\text{Tc}$, the indispensable “working horse” in nuclear-medical diagnostics. The demand for further radioisotopes like e.g. ^{166}Ho is increasing and needs to be met. The same applies to neutron transmutation doped Si that is needed e.g. for high-power switches for the DC long distance current transmission from the off shore wind based energy production to the users in the industrial centers of the country.

Taking into account all what has been said above it is obvious that a multi-purpose research reactor providing neutrons for a broad band of applications will be needed urgently also in the future and FRM II and MLZ will contribute their share for the benefit of science as well as for the entire society.

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MARIA Research Reactor

Bartłomiej Piwowski, Michał Gryziński, Rafał Prokopowicz, Anna Talarowska, Maciej Lipka, Emilia Wilińska, and Gawel Madejowski, National Centre for Nuclear Research, Świerk, Poland

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Introduction

The MARIA research reactor is named after Maria Skłodowska-Curie, a great Polish physicist, twice Nobel prize winner (1903 in physics, jointly with H. Becquerel, and 1911 in chemistry). It is currently the only nuclear reactor in Poland; however, it is the outcome of a long history of atomic research and development. The nuclear site NCBJ (National Centre for Nuclear Research, former IBJ i.e. Institute for Nuclear Research A. Sołtana) is located in Otwock-Świerk, 30 km from Warsaw, Poland (Fig. 1). The technical specifications for the reactor, at that time with the codename R-2 (second reactor), were initially proposed in 1964, and they were based upon the experience with the VVR-S type (a Soviet design) EWA research reactor that was operational since 1958. EWA was an acronym for (in Polish) Experimental-Water-Atomic but it was effectively a modified (2–10 MW) Soviet VVR-S design. Requirements for the new reactor included: production of radionuclides for the needs of industry and medicine, testing of nuclear reactor materials, coolants and reactor fuels in rigs and loops, neutron modification of materials, activation analysis and autoradiographic techniques, and basic and applied research using neutron beams. Additionally, in-core irradiation positions and neutron beams were designed for medical research and training in reactor physics and technology.

Because of the political situation at the time, the only supplier of nuclear technology could be the USSR, which initially proposed the RFT type (water cooled uranium-graphite reactor) nuclear reactor technology. During the negotiations (and as a result of Polish lobbying), this was changed to the design of the MR-type reactor. The original Soviet project was modified in Poland to meet the specific needs by adding horizontal neutron beams, and vertical channels for the production of the radioisotopes. Also, the individual solutions differ with the MR, including the reactor pool cooling system and in-core structures. Some of the designs were tested in the MARYLA (general purpose zero power reactor), and two specially built for that purpose: ANNA (RFT model) and AGATA (MR model).

The construction of the MARIA reactor began in 1969 and was carried out by the Institute for Nuclear Research (pol. Instytut Badań Jądrowych), the reactor went critical on December 18, 1974. MARIA operated until 1985 and then it underwent a thorough modernization, including replacement of the reactor control system, inspection of graphite reflector blocks, supplementing the core configuration with additional beryllium blocks, construction of a biological radiation shield, modernization of the reactor cooling system equipment, modernization of reactor technological ventilation systems and installation of the fuel channels flow and temperature control system. During the modernization, the Chernobyl disaster happened, so its initial scope was broadened: an emergency fuel channels cooling system and redundant covers over the horizontal neutron beams were added (Pytel, 2015). The reactor resumed regular operation in 1993 and is the only nuclear reactor operating in Poland today (the EWA reactor was shut down in 1995).

MARIA is constantly being modified in order to improve its safety and scope of scientific and technical applications. The most significant changes concern nuclear fuel. Initially, the MARIA reactor used fuel enriched in U-235 to 80%. In 2003, the reactor was converted to 36% enriched fuel and in 2005–06 the decision was made to use a fuel designed with the same enrichment but with a smaller mass of uranium in a single fuel element. In 2009, tests began on a new type of French manufactured MC-5 fuel element with enrichment below 20%. Beginning in 2012, the process of a phased conversion started and the total conversion of MARIA into low-enriched uranium fuel (LEU) was completed in 2014. This change of fuel required a refurbishment of the reactor cooling system in 2013 following a cooling tower replacement in 2012. That same year, the low-enriched uranium Russian MR-6 fuel elements were



Fig. 1 MARIA reactor and technological-administrative buildings. Courtesy of NCBJ (National Center for Nuclear Research).

tested in MARIA which allowed the reactor to have two sources of fuel elements for flexibility. Additionally, MARIA received a license in 2020 for a special purpose MR-2 fuel design that enables sample irradiation inside the central cooling channel of the fuel element in high fast neutron flux. In 2016, repatriation of the used and spent HEU fuel elements to the Russian Federation were completed as part of the International Global Threat Reduction Initiative (GTRI) coordinated by the US Department of Energy and the International Atomic Energy Agency.

In 2014, the INSARR (Integrated Safety Assessment of Research Reactors) of the International Atomic Energy Agency (IAEA) mission recommended an upgrade of the fuel temperature measurement system to decrease the delay time of safety signals; In 2017, follow-up INSARR mission positively assessed the state of nuclear safety of the reactor and implementation of recommendations and suggestions.

Currently, the MARIA reactor has a license to operate until 2025, and thanks to regularly carried out modernizations and upgrades, it is expected to operate at least until 2050.

MARIA reactor design

Fuel design

Two types of low-enriched fuel elements are currently being used in the MARIA reactor (**Fig. 2**):

- MC-5/485, French production (by Cerca-Areva), with 19.75% enrichment, a dispersion of a uranium silicide alloy powder (U_3Si_2) in Al, and
- MR-6/485, Russian production (by Novosibirsk Chemical Concentrates Plant, TVEL), with 19.70% enrichment, a dispersion of UO_2 in Al.

Both fuel element designs are clad with aluminum. Each fuel element has a form of six (MR) or five (MC) concentric tubes.

For the French MC-type fuel, U_3Si_2 powder is produced by melting uranium and silicon together and then grinding into powder of a specific size. Then it is mixed with aluminum powder and compressed between two Al covers, forming a sandwich, and transformed into a thin plate by hot and cold rolling and then cut and curved to obtain the right dimensions. Each fuel element tube is fabricated using several curved fuel plates. Plates in the outermost tube, acting as a support tube, are welded together, while in the others the curved fuel plates are attached with stiffeners. The tubes (except for the outer tube) are embedded in guides with a clearance provided for axial thermal expansion.

The Russian MR-type fuel is produced as a tube in the process of direct extrusion. The fuel element consists of six concentric tubes, the third of which from the outside also plays a supporting role, because the structural elements of the fuel element are attached to it. The other tubes can move along the axis within a certain thermal clearance. Each tube consists of three layers: inner and outer aluminum alloy cover and the fuel core. In both types of fuel, the tubes are spaced equally, due to the use of spacers, in order to ensure appropriate cooling channel dimensions. Parameters of both types of fuel are presented in **Table 1**.

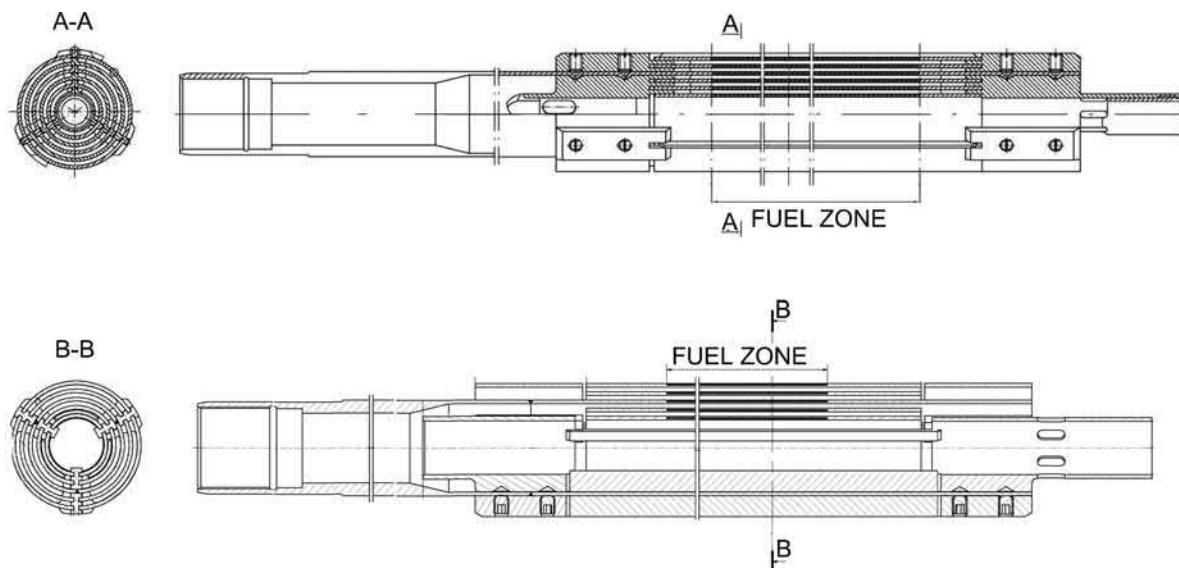


Fig. 2 Fuel elements cross section. At the top there is the French MC-5 design and at the bottom there is the Russian MR-6 design. Courtesy of NCBJ (National Center for Nuclear Research).

Table 1 Parameters of fuel currently used in MARIA reactor.

Parameter	MC-5/485 (19.75%)	MR-6/485 (19.70%)
Total length of the fuel element	1315 mm	1380 mm
Nominal fuel length	1000 mm	1000 mm
Assembly outer diameter	74.6 mm	74.8 mm
Fuel type	Dispersion of U_3Si_2 in Al	Dispersion of UO_2 in Al
U-235 content	485 g	485 g
Uranium density in the fuel layer (tube)	4.79 g cm^{-3}	3.59 g cm^{-3}
Fuel layer thickness	0.8 mm	0.8 mm
Heat exchange surface	1.29 m^2	1.72 m^2
Mass ratio U-235/ H_2O in the fuel element at $T = 20^\circ$	0.233	0.940
Diameter of the outermost fuel tube	70 mm	70 mm

Reactor characteristics

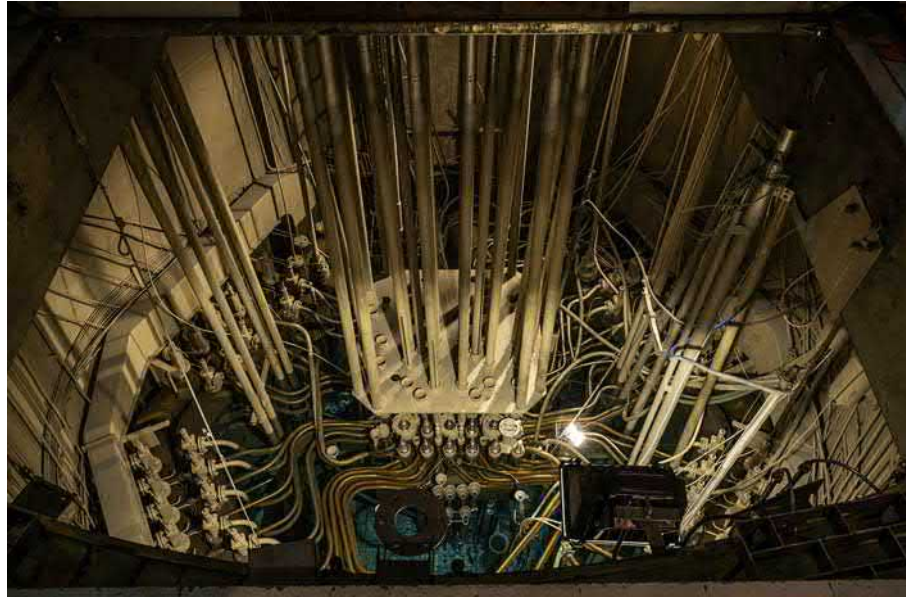
The critical parameters of the two types of fuel used in the MARIA reactor slightly differ. In the MC fuel there is angular unevenness of power density distribution in fuel tubes, related to discontinuities in the fuel distribution at the boundaries of fuel plates as well as with the presence of plate stiffeners. The problem does not occur in the MR fuel because of the different design. Parameters critical for MARIA reactor operation are listed in [Table 2](#).

Fuel conversion

Initially, the reactor operated on Russian fuel MR-6, in the form of a dispersion of UAl in an Al matrix with 80% enrichment. In response to IAEA recommendations, the fuel was gradually replaced with less enriched material. The conversion took place in two steps: from 80% to 36% and from 36% to below 20%. The 80% enriched fuel was gradually replaced by lower enriched elements (36% enrichment, in the form of a dispersion of UO_2 in Al, from the same manufacturer). The fuel design remained unchanged with the amount of U-235 increased, with concurrent enhancement of the fuel matrix density. To maintain fuel performance, more low-enriched uranium was required to be used in the fuel matrix which significantly increased the amount of U-238 in the fuel. As a result, a decrease in available neutron flux occurred (due to the additional parasitic U-238 content) while maintaining the thermal characteristics of the fuel. Maintaining neutron flux at the previous levels was only possible by increasing the thermal power of the fuel and raising the total power of the MARIA reactor. With the first stage of conversion completed, the next step was replacing the fuel containing 540 g of U-235 with new elements of the same enrichment but with a U-235 content of 430 g (same manufacturer). This process took place in years 2005–06. The result of this change was a significant decrease in the release of fission products to coolant and the possibility of higher burn-up with the more robust fuel design. Since 2005, a full core conversion to low enriched fuel was planned and scheduled as part of the US Global Threat Reduction Initiative (GTRI). In 2009, tests of French fuel in the form

Table 2 General parameters of MARIA reactor core.

Parameter	Value
Neutron flux:	
• Thermal, in core	• $2.5 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$
• Thermal, in reflector	• Less than $3 \times 10^{13} \text{ n cm}^{-2} \text{ s}^{-1}$
• Fast, irradiation position	• Up to $1 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$

**Fig. 3** The reactor pool view with the various fuel channels cooling pipes in the foreground. Courtesy of NCBJ (National Center for Nuclear Research).

of U_3Si_2 dispersion in Al with 19.75% enrichment were started. Positive test results led to the decision to approve the fuel for reactor operation. However, hydraulic calculations showed that the introduction of the new fuel would result in an increase in the pressure drop across each fuel channel, as the new fuel was characterized by higher hydraulic resistance and smaller heat exchange surface. To maintain a safe thermo-hydraulic regime an increase of the coolant flow through the fuel channels was needed. An upgrade of the fuel channel cooling system was designed and installed in 2013 which replaced the single-speed main circulation pumps with two-speed pumps. In addition, a newly designed decay heat removal system with separate circulation pumps was installed at the same time. The full HEU to LEU core conversion program was completed in 2014.

Core configuration

MARIA is a 30 MW pool type reactor of a unique design with individual pressurized fuel channels for cooling the reactor fuel. Each channel consists of an outer aluminum tube containing an individual fuel element and cooled by dedicated piping connected to the reactor cooling system. Graphite is used as a reflector surrounding the reactor core and water and beryllium as moderators between and around fuel channels. The core consists of 28 fuel channels situated in a beryllium matrix, enclosed by graphite blocks forming the reflector. It is also equipped with approximately 26 vertical channels used for irradiation and 8 horizontal channels (seven operational and one still only in conceptual design) for other experiments requiring a radiation beam. The whole core is placed in an aluminum housing, called the basket which is supported on a table or platform at the bottom of the reactor pool. The top view of the core is shown in Fig. 3. For physical reasons, the fuel grid pitch in the middle of the height of the fuel in MARIA reactor core is 13 cm. In contrast, structural considerations, to provide more space, require a much larger grid pitch at the level of connection of the cooling water connections as well as at the level of control rod drives above the pool water level. Effectively, the MARIA reactor core was designed as a truncated cone with spacing between fuel channels smaller at the bottom of the core than at the top. This is another unusual feature of MARIA as most research reactors, with the Belgian BR2 being a notable exception, are designed with fuel elements or assemblies in vertical orientation.

The beryllium matrix consists of 45 blocks and moldings of 10 different uncladded shapes. The size of the full block (precisely the aluminum cap) at the top is 14 cm, and at the bottom 12 cm. All the fuel channels have an outer tube diameter of 7.9 cm. They

are placed in the spaces created by the corner cuts in the beryllium blocks and embedded in the mounting plate. The control rods of the reactor, also used as shim rods, have B_4C for neutron absorption and are placed in the central channels of the beryllium blocks. The design of the control rod drives and channels for all types of rods are the same which allows moving them or changing the configuration as necessary. Fuel assemblies and core configuration are usually shuffled before each reactor start-up. The number of necessary fuel elements is usually 28; however, the total number of fuel elements can be changed as needed for a particular reactor cycle. A typical core configuration is shown in Fig. 4. Graphite blocks have an analogous design as beryllium blocks, except that they do not have recessed corners. The graphite blocks are clad with aluminum. Some graphite blocks also have holes in the center (along the entire height of the block) that can accommodate isotope production channels. To allow irradiation of large experiments or materials such as for silicon doping, a large 4 by 4 grid area is established with a special neutron moderation curtain between the reactor and the experiments.

Cooling systems

The MARIA reactor is cooled by two separate primary cooling systems: the fuel channel cooling system and the reactor pool cooling system. During regular operation of the reactor, coolant flow in the fuel channels is provided by two main pumps (with two in standby) providing a total flow of $660 \text{ m}^3 \text{ h}^{-1}$. Additionally, two auxiliary pumps operate in the bypass mode, and their role is to provide forced cooling after the reactor shutdown. A small by-pass flow is sent through the pressurizer to degas the coolant of dissolved gases. The fuel channel cooling system is shown in Fig. 5. In the MARIA reactor, each fuel element is placed in an individual fuel channel connected to inlet and outlet manifolds through pipelines and valves. The channels are pressurized with the inlet pressure of 1.7 MPa, and the individual flow is set at $25\text{--}30 \text{ m}^3 \text{ h}^{-1}$, the overpressure is supplied by the main pumps and a pressurizer. A set of thermometers installed in the cooling system enables measuring coolant inlet temperature and outlet temperature of individual fuel channels. This capability enables the measurement of thermal power, coolant flow, and temperatures in each fuel element. During normal operation, the maximum temperature of the fuel element cladding does not exceed 155°C . Each of the fuel channels is also connected to the failed fuel detection system. The system enables monitoring of the fission products release from individual fuel elements.

The pool cooling circuit ensures safe operating conditions for the beryllium matrix, graphite reflector, control rods, target materials in the irradiation channels, experimental devices, and structural elements of the reactor core. The heat generated in non-fuel core elements comes mainly from gamma radiation emitted from the fuel, as well as from neutron induced reactions in the core materials. A significant part of the heat received by the pool cooling circuit comes from the heat exchange between the fuel channels and the pool. The pool water itself provides radiation shielding and acts as a biological shield. It is a forced circulation, low pressure, open circuit. The secondary cooling circuit is mutual for both systems. It is connected to them by a set of heat exchangers, and the heat is dissipated in the atmosphere via a forced draft cooling tower.

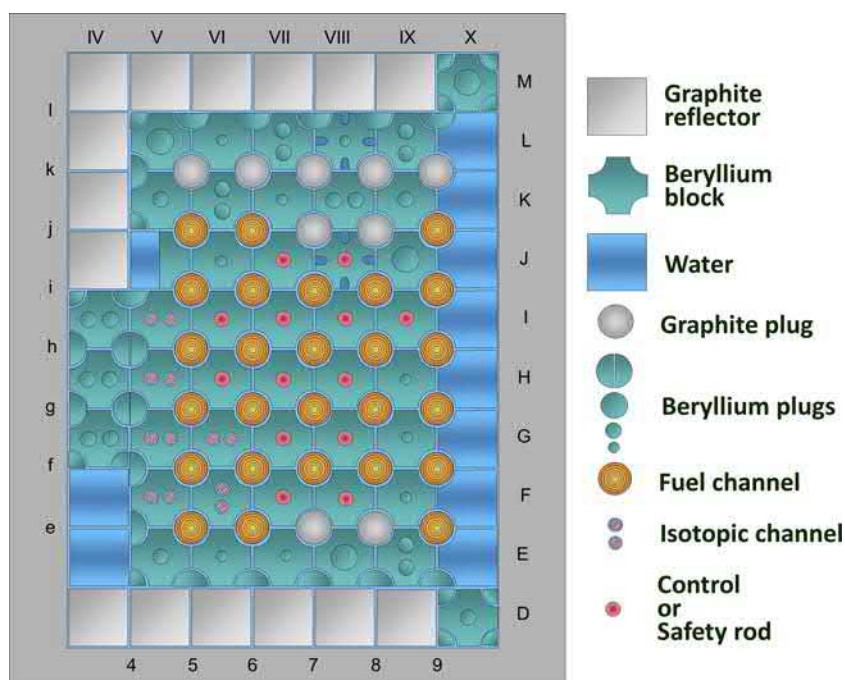


Fig. 4 The typical core configuration for MARIA reactor. Courtesy of NCBJ (National Center for Nuclear Research).

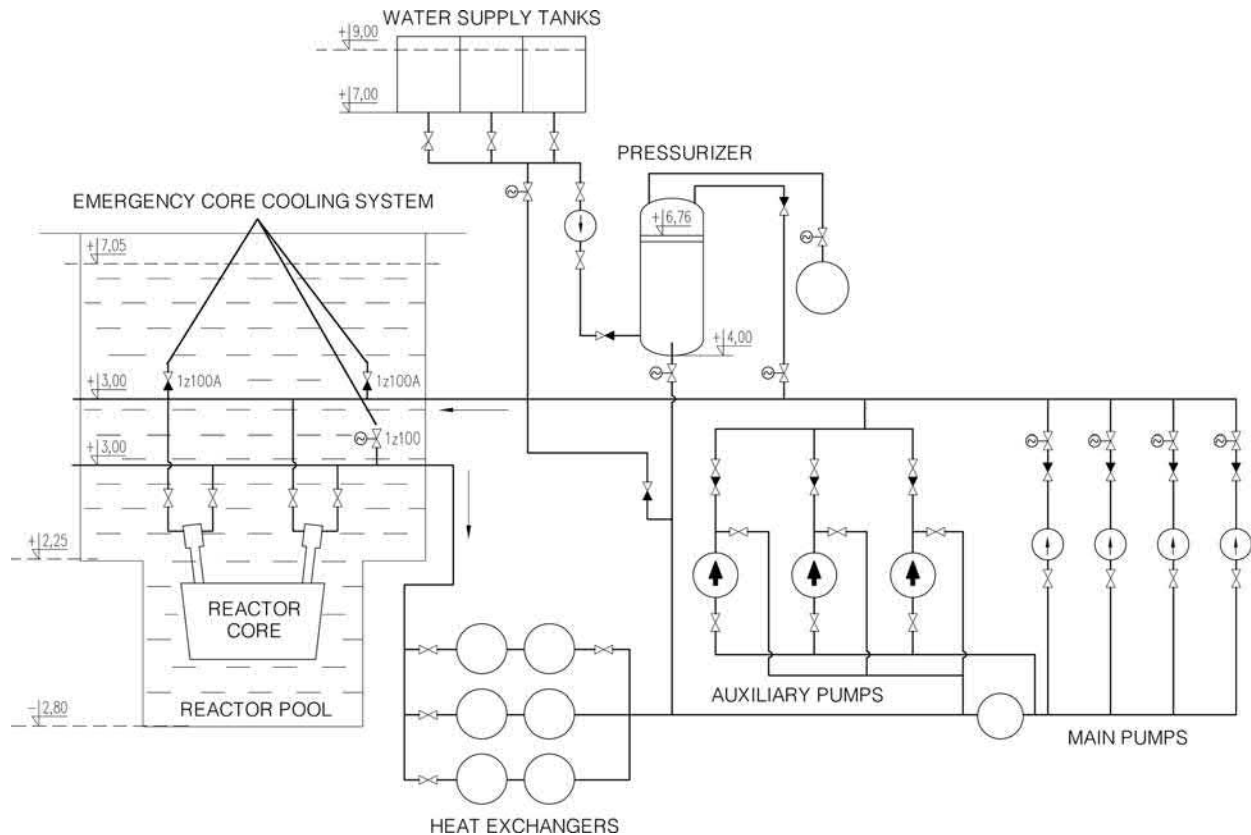


Fig. 5 Fuel channels cooling system diagram including main and auxiliary pumps, pressurizer, heat exchangers, water supply tanks and emergency core cooling system marked. Courtesy of NCBJ (National Center for Nuclear Research).

Safety systems

The reactor protection system (RPS) is based on 13 absorbing rods divided into three categories: safety rods, shim rods and an automatic control rod. Although the safety rods are the main protection system, both safety rods group and shim rods group are capable of scramming the reactor. The RPS receives information from the following safety systems: reactivity control, SSC's readiness information, instrumentation and power supply. Moreover, several signals deriving from the control room buttons including manual emergency fuel channel cooling system activation, are included in RPS.

Reactivity control system

The reactivity control system functions range from control of slight deviation from criticality during normal operation to its safety role during postulated initiating events. The following devices are included in the system:

- Start-up channel based on a fission chamber with the following outputs: reactor period and neutron flux (low and medium measurement ranges);
- Log amplified lines based on ionizing chambers with the following outputs: neutron flux (medium and high measurement ranges);
- Linear amplified lines based on ionizing chambers with the following outputs: reactor period and neutron flux (medium and high measurement ranges).

Other instrumentation systems

There is a group of additional measurement systems which provide the global temperature, pressure and flow measurements as well as water conductivity, vacuum pressures and stack effluences monitoring. However, unlike the pool type reactors, the MARIA reactor design included provisions for nuclear power plant fuel assemblies testing. Therefore, an additional surveillance was needed and fulfilled by the following instrumentation systems:

- Surface contamination and flawed fuel elements monitoring in each fuel channel;

- Inlet and outlet temperature measurements of coolant in each fuel channel;
- Flow measurement in each fuel channel;
- Pressurizer water level.

Experimental facilities

Experimental facilities

The MARIA reactor was designed to operate with several experimental devices in parallel. The most important of them are:

- vertical irradiation channels for radionuclide production;
- reactor test rigs for structural materials, biological samples and reactor fuel studies under stationary conditions ([Krzysztozek, 2013](#));
- large sample channels for electronic, concrete and other materials irradiation;
- horizontal experimental channels for neutron beam studies.

The reactor construction allows installing various kinds of irradiation facilities and neutron sources.

The following channels are available:

- 2.3 cm diameter channels in beryllium blocks;
- 2.8 cm diameter channels in the graphite reflector;
- 3.8 cm diameter channels in the graphite reflector;
- 4.6 cm diameter channels inside the modified fuel element MR-2;
- bulk material irradiation facility for large-size (up to 8.5 cm in diameter and 90 cm height) target materials
- channels equipped with a hydraulic transport ("rabbit") system located in the beryllium and graphite area of the reactor core.

The customized fuel channel can be used to irradiate fission materials, like targets or fuels, under strictly controlled conditions. Such technology has been applied for irradiation of uranium targets for ^{99}Mo production. This activity plays a special role in the utilization of the MARIA reactor.

Following the shortage of the key medical radioisotope ^{99}Mo and its daughter $^{99\text{m}}\text{Tc}$ related to a long-term reliability, the MARIA reactor declared its readiness to irradiate the uranium targets. The program has been successfully launched and the irradiation of uranium targets has been performed since 2010 (HEU targets). Since 2017 LEU targets have been irradiated.

The technology implemented in the MARIA reactor for ^{99}Mo production employs a specific construction of fuel channels with a cooling system, mechanical hardware and measuring system. The system is linked to the measuring system of thermal-hydraulic parameters of the each of the reactor fuel element.

Thanks to the versatility of the MARIA reactor construction, there are plenty of possibilities of neutron beam and core neutron field utilization. A set of irradiation channels located in the beryllium moderator matrix can provide neutron flux density of up to $2.5 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$. Their performance and neutron spectrum may vary depending on the distance from the fuel elements and the position in the core; however, the spectrum is still typical for the thermal reactor core. Four of the in-core irradiation positions comprise the hydraulic rabbit system, allowing irradiation time control with one-second resolution. One of the rabbit system channels is located aside of the core, therefore providing a well-thermalized neutron flux of up to $2.5 \times 10^{13} \text{ n cm}^{-2} \text{ s}^{-1}$.

The MARIA reactor is equipped with bulk material irradiation facility located at the peripheral part of the core, behind the thermal neutrons shielding screen. The fast neutron flux reaches $3 \times 10^{12} \text{ cm}^{-2} \text{ s}^{-1}$ in the most powerful position, while the thermal neutron flux density is reduced below $3 \times 10^{10} \text{ cm}^{-2} \text{ s}^{-1}$. The facility consisting of eight channels enables to irradiate large-size (up to 85 mm in diameter and 900 mm height) target materials, devices or its components.

A special 14 MeV neutron irradiation channel is operated in the MARIA reactor core. The device consists of thermal-to-14-MeV neutron converter based on lithium-6 and deuterium compounds ([Prokopowicz, 2015](#)). The principle of its operation is based on two-step reactions induced by thermal neutrons. The 14 MeV neutron flux density exceeds $1 \times 10^9 \text{ n cm}^{-2} \text{ s}^{-1}$ inside the converter with cut-off thermal neutrons at the same time. The neutron energy spectrum depends on the converter location in the reactor core and is similar to the one in thermonuclear devices. Its loading capacity is ca. 60 cm^3 (four containers, $\varnothing 15 \times 90 \text{ mm}$ each). The converter is dedicated to material research, i.e. irradiation tests of new structural materials for Generation IV nuclear reactors and fusion facilities. It can also be used to obtain radiopharmaceuticals produced by fast neutrons in threshold reactions.

The MARIA reactor design offers an excellent irradiation capabilities in thermal neutron spectrum with a flux up to $2.5 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$, but fast neutrons flux is limited below $5 \times 10^{13} \text{ n cm}^{-2} \text{ s}^{-1}$. To increase the irradiation capabilities in the fast flux a modified fuel assembly (MR-2) design was proposed with a place for target irradiation in the middle of the element – this feature would increase the fast flux up to $1 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$.

The MR-2 fuel can hold an irradiation container with a diameter of $\delta = 34 \text{ mm}$ $\delta = 34 \text{ mm}$ along the whole active length of the fuel that is 1000 mm, the total volume of irradiation equals 90 dm^3 .

The reactor is also equipped with a facility for neutron transmutation doping of silicon single crystals. The vertical neutron flux distribution to be available in graphite reflector together with additional shaping by means of titanium absorber screens has been adopted for achieving high degree of axial and radial doping distribution uniformity. The above solution allowed to achieve the

simplest technical construction and very high quality level of doped silicon crystals. The reactor design allows installing various in-pile experimental facilities like rigs and loops. Recently, thermostatic irradiation rigs are being prepared to be installed in the MARIA reactor core. The rigs can be inserted to the vertical openings inside or between beryllium blocks. The mounting slab over the core allows fixing the upper part of the rig.

Neutron beams characteristics

The neutron beam in the MARIA reactor is streamed through horizontal channels which pass through the biological shield and the reactor pool to the graphite reflector surrounding the core. Some of them are directed towards the core, while others are arranged tangentially, they provide more thermalized yet weaker beam. In the past, the horizontal channels used to be utilized for fundamental research, supplying the research equipment with the beam of intensity up to $3\text{--}5 \times 10^9 \text{ n cm}^{-2} \text{ s}^{-1}$. The research equipment consisted of a wide array of facilities for different applications: a radiography facility, an autoradiography facility for examining paintings, an abiaxial diffractometer for studying crystalline and magnetic structures, a diffractometer for low angle scattering experiments, a low-angle neutron scattering spectrometer, a polarized neutron spectrometer, a triaxial spectrometer for testing network dynamics and a triaxial spectrometer for inelastic scattering.

Currently the equipment undergoes modernization. One of the channels will provide a newly designed biomedical research facility with a neutron beam. It will be primarily dedicated for boron-neutron capture therapy research and other medical purposes such as development of therapeutic and diagnostic carriers or for beam treatment planning and surrounding dosimetry. A neutron converter, which will be installed at the channel inlet, will amplify epithermal and fast neutron flux. The remaining channels are going to be equipped with state-of-art experimental instruments for neutron scattering biological studies (Gryziński, 2015).

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HANARO

Sooyoul Oh, Korea Atomic Energy Research Institute, Daejeon, South Korea

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Overview

The High-flux Advanced Neutron Application Reactor or HANARO is a 30 MW research reactor. It is located at the site of the headquarters of Korea Atomic Energy Research Institute (KAERI) in Daejeon, Republic of Korea. Based largely on its previous experience of operating two TRIGA reactors for more than two decades, KAERI established the design requirements, developed the detailed design, and constructed HANARO. At the conceptual design stage, KAERI collaborated with the AECL of Canada, and later on it realized the design concepts explored during the collaboration.

The HANARO complex consists of a reactor building, Cold Neutron Research Facility (CNRF), Radioisotope Production Facility (RIPF), Irradiated Materials Examination Facility (IMEF), and several auxiliary buildings such as cooling towers and a cryogenic equipment building for the cold neutron source. Fig. 1 shows the HANARO complex.

In addition to the complex, KAERI has a fuel fabrication factory for research reactors, which manufactures uranium silicide fuels for HANARO, plate-type uranium-molybdenum fuels, and low enriched uranium targets for ^{99}Mo production by fission for medical applications. HANARO is also supported by KAERI's other infrastructures. For example, a thermal hydraulic test loop outside of the HANARO complex is used for testing the mechanical integrity of newly developed irradiation capsules that will be loaded in HANARO.

Since the initial criticality in February 1995, HANARO has been gradually evolving. After resolving an early stage regulatory issue on the fuel integrity that allowed power operation only up to 80% of the rated power, HANARO began the operation at full power of 30 MW in 2004. Four KAERI-made fuel assemblies were first loaded in 2005, and since then HANARO has used KAERI-made fuels in full. Meanwhile, the neutron science in Korea met a turning point with the inauguration of the CNRF in 2010. Instruments



Fig. 1 HANARO complex. RX: reactor building; IMEF: irradiated materials examination facility; RIPF: radioisotope production facility; AU: auxiliary utility building for cold neutron source; PH: pump house; CT: cooling tower; CNL: cold neutron laboratory (cold neutron research facility); AE: auxiliary equipment building for cold neutron source.

moved to the CNRF from the reactor hall began to utilize cold neutrons, and new instruments have been installed one by one. The areas of neutron irradiation for industrial and medical sectors have been growing as well. For instance, since HANARO began the service of neutron transmutation doping of silicon ingots in 2003, its share in the world market has grown significantly. Production of radiopharmaceuticals for domestic use is another example. In 2014, HANARO celebrated 3000 days of operation. Furthermore, as an International Center based on Research Reactor (ICERR) designated by the IAEA in 2019, KAERI provides researchers from around the world with access to HANARO and associated facilities more effectively than before.

Design features

HANARO is a 30 MW, multipurpose, pool type research reactor that uses rod-type, low enriched uranium silicide fuels (Kim et al., 1996). Table 1 shows its design features.

HANARO uses two types of fuel assemblies and the full core consists of 20 hexagonal and 12 circular assemblies. Each assembly is loaded into a flow tube and locked in the bottom grid plate. Four control rods called CAR are for the power control while four shutoff rods are fully withdrawn during power operation. Each control or shutoff rod is a hafnium tube and it travels vertically, encompassing a circular fuel assembly. The core is cooled by the upward, forced convection of light water and flap valves provide a loop for natural convection through the reactor pool when necessary. The reactor concrete island consists of three pools: a reactor pool, a service pool, and a spent fuel storage pool. The reactor building is a confinement maintaining a negative pressure of about 50 Pa.

Fig. 2 shows the core and the heavy water reflector along with the arrangement of horizontal beam ports and vertical irradiation holes. The core is inside of the curvy rectangle in the center, called the inner shell of the heavy water vessel, and a large circular heavy water vessel surrounds the core. Hexagonal fuel assemblies are shown in blue and circular assemblies in yellow. Four circular assemblies are loaded outside of the core.

Three vertical irradiation holes in the core are for the tests of structural materials or nuclear fuels that usually require high neutron fluence. At the central hole (CT), the thermal neutron flux measures up to $4.5 \times 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$, and the fast neutron flux above 1 MeV is about half of the thermal flux. As shown in Fig. 2, the reactor core is rectangular in order to increase neutron leakage from the core into the reflector. Along with the superior moderating ratio of heavy water, this design results in high thermal neutron flux in the reflector region, 0.3 to $2.0 \times 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$ in irradiation holes and 0.4 to $2.5 \times 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$ at beam port noses. The fast neutron flux level in the reflector is lower than the thermal flux by an order of two or more.

In the reflector region, there are a total of 33 vertical holes available for neutron utilization. These include one hole for producing cold neutrons (CNS in Fig. 2), three holes connected to pneumatic transfer systems for the neutron activation analysis (PTS), two for

Table 1 Design features of HANARO.

Category	Description
Type of reactor	Open tank in pool
Thermal power (MW)	30
Thermal neutron flux ($\text{cm}^{-1} \text{ s}^{-1}$)	Up to 4.5×10^{14} at core center Up to 2.5×10^{14} in reflector region
Fuel	
Shape of fuel element	Finned rod
Shape of fuel assembly	Hexagonal (36-rod assembly), Circular (18-rod assembly)
Fuel meat	U_3Si in aluminum matrix, 19.75% enriched uranium
Effective fuel length (mm)	700
Control element	
Absorber material and shape	Hafnium, cylindrical
Number of elements	4 for control, 4 for shutoff
Drive mechanism	Stepping motor, top-mounted
Fluidic system	
Coolant	Light water
Core cooling method	Forced convection, upward flow
Reflector	Heavy water
Structural material of reactor	Mostly aluminum, Zircaloy-4 (flow tubes and heavy water vessel)
Instrumentation and control	Digital except reactor protection system
Type of reactor building	Confinement
Neutron beam port	
Angle to the core	Tangential
Number of ports	7
Number of irradiation holes	3 in core, 33 in reflector region

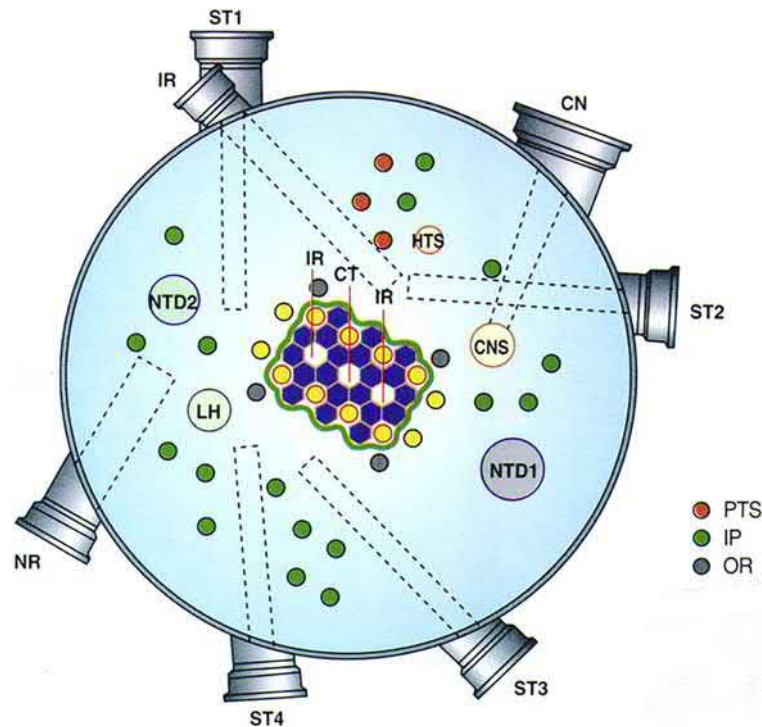


Fig. 2 Plan view of core and reflector region.

the neutron transmutation doping of silicon ingots (NTDn), two large diameter holes mainly for the material testing (LH and HTS), and other small holes of 6 cm in diameter for radioisotope production and other irradiation purposes (IPs).

The neutron beam ports in HANARO include one cold neutron port (CN in Fig. 2), one for neutron radiography (NR), one for ex-core neutron irradiation (IR), and four standard ports (STn). HANARO produces cold neutrons by cooling down thermal neutrons in a double-wall container of liquid hydrogen, called a moderator cell, in the reflector region. A helium refrigeration system maintains liquid hydrogen at 22 K, and it results in the cold neutron characteristic temperature of about 26 K, which corresponds to a wavelength of 0.6 nm (Lee et al., 2012). The CN port views the cold neutron source, and cold neutrons travel to the CNRE, a building next to the reactor building, through cold neutron guides extended from the CN port. There are five, multi-layer nickel-titanium coated supermirror guides that eventually branch into seven (Cho et al., 2011). Each guide has a different curvature for removing gamma rays and fast neutrons at arrival at the neutron instruments. The cold neutron fluxes at the monochromator and neutron velocity selector position are in the range of 0.4 to $1.4 \times 10^9 \text{ cm}^{-2} \text{ s}^{-1}$, depending on the neutron guide and instrument.

Reactor operation

A usual cycle of operation consists of 28 days of continuous operation and 14 days of shutdown for maintenance and fuel reloading. Among the 32 fuel assemblies in the core, typically five assemblies are replaced every cycle. The operators work in three shifts and each shift is staffed with three operators including one senior reactor operator. The control room must be attended always, even during a shutdown period.

The HANARO complex is regulated by the Nuclear Safety and Security Commission, which is the governmental nuclear regulatory body. The regulatory activities include biennial inspection of the facilities, biennial quality assurance audit, periodic safety review every 10 years, and investigation of incident if any. In the field, two technical support organizations of the Commission, the Korea Institute of Nuclear Safety and the Korea Institute of Nuclear Nonproliferation and Control, carry out the regulatory activities for the Commission.

In addition to the formal organization in KAERI, there are several committees in different levels. The HANARO steering committee reviews, recommends, and decides the policy for facilitating the utilization, installation or modification of user facilities, and so on. The members are representatives of each user group outside KAERI, a government officer, several KAERI managers, and others such as an expert in regulation. The HANARO safety review committee, chaired by the head of HANARO, reviews the safety concerns of activities such as conducting a new kind of experiment, making revisions to operating procedures, and restarting of the reactor after an unplanned shutdown. The HANARO users association is a group of delegates from individual utilization areas—neutron science, fuel and material test, neutron activation analysis, radioisotope research, and so on.

Utilization

Neutron beam utilization

HANARO has 17 beam instruments in total including nine instruments that utilize cold neutrons (Park, 2013). In addition, it operates the Neutron Radiography Facility (NRF) and the Ex-core Neutron Irradiation Facility (ENF). Figs. 3 and 4 show the instruments and facilities in the reactor hall and the CNRF, respectively.

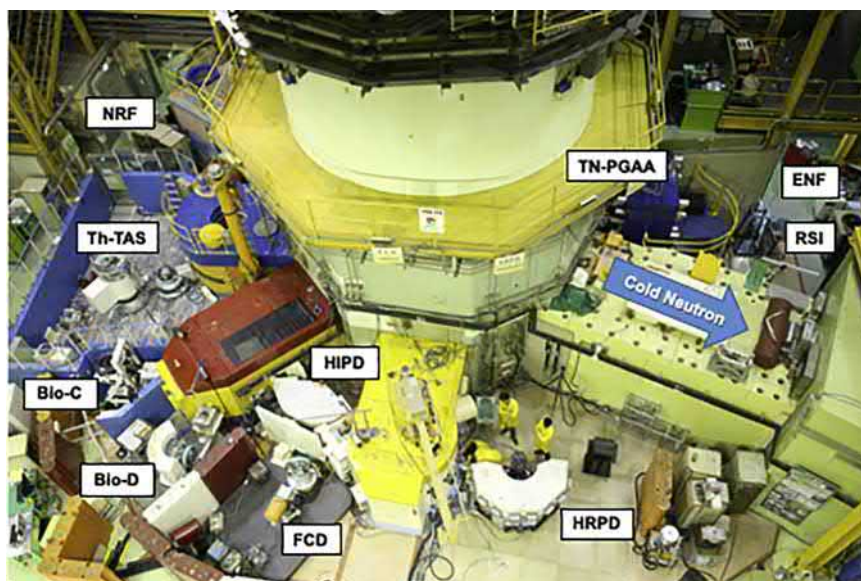


Fig. 3 Instruments and facilities in reactor hall. NRF: neutron radiography facility; Bio-C: bio-diffractometer with neutron image plate camera; FCD: four circle diffractometer; HRPD: high resolution powder diffractometer; RSI: residual stress instrument; Thermal-TAS: thermal neutron triple axis spectrometer; Bio-D: bio-diffractometer; HIPD: high intensity powder diffractometer; ENF: ex-core neutron irradiation facility; TN-PGAA: thermal neutron prompt gamma activation analysis (above RSI).

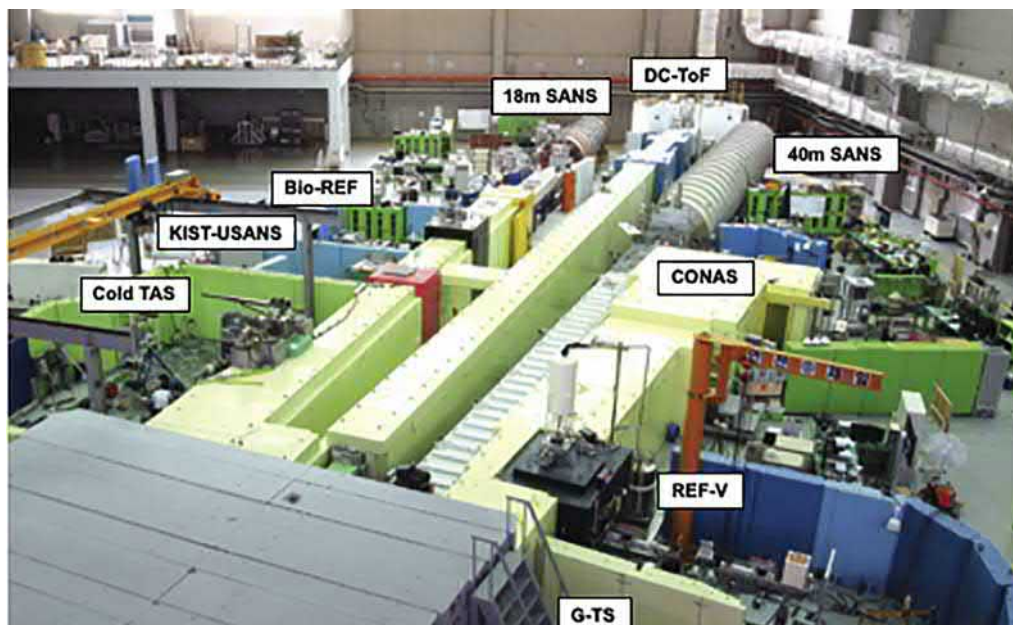


Fig. 4 Instruments in cold neutron research facility. G-TS: guide test station; CONAS: cold neutron activation station; DC-ToF: disk chopper time-of-flight spectrometer; Bio-REF: bio-reflectometer; Cold TAS: cold neutron triple axis spectrometer; REF-V: vertical type neutron reflectometer; 40 m SANS: 40 m small angle neutron scattering instrument; 18 m SANS: 18 m small angle neutron scattering instrument; KIST-USANS: KIST ultra small angle neutron scattering instrument.

Neutron diffraction is the only way to distinguish a microscopic structural deformation of light elements such as lithium and oxygen ions in a unit cell. The High Resolution Powder Diffractometer (HRPD) is a tool used to investigate the magnetic structures, multiferroic materials in which ferroelectric and magnetic properties coexist, and an electrochemical reaction mechanism.

The Four Circle Neutron Diffractometer (FCD) is used for a single crystal diffraction. The Bio Diffractometer with neutron image plate camera (Bio-C) and Bio Diffractometer (Bio-D) are dedicated to the investigation of the crystal structure of biological macromolecules such as proteins and nuclei acids. The Residual Stress Instrument (RSI) is used to examine the behavior of material in-situ under load. The RSI is an engineering neutron diffractometer dedicated to the measurement of stress in polycrystalline metals and alloys, and it is useful for investigating defects in, for instance, welding of thick steels. HANARO plans to move most of the above thermal neutron instruments to the CNRF after installation of thermal neutron guides in the future.

Meanwhile, HANARO operates two facilities that utilize thermal neutron beams in the reactor hall. The NRF uses neutrons to provide a transmission image. An example using the NRF is an investigation of water behavior in a fuel cell where hydrogen should be burned efficiently. The ENF is a facility used for various purposes such as a dynamic neutron radiography and developments of neutron detectors such as a SiC detector.

In the CNRF, the neutron instruments utilize cold neutrons produced in HANARO with wavelengths of 0.4–2 nm. A small angle neutron scattering (SANS) technique provides a unique opportunity to study the structure and dynamics of materials at nanometer scale. It is applied to problems involving soft matters such as polymer conformation, morphology, and biological structure. SANS is also powerful in studies on flux-line lattices in superconductors and precipitations in alloys. Three SANSs are in the CNRF (Kim et al., 2013): 40 m SANS (Han et al., 2013), 18 m SANS, and Ultra SANS (USANS). The former two instruments are a conventional pinhole type, while the USANS uses a pair of modified Bonse-Hart-Agalian channel cut crystals as a monochromator and an analyzer. The Korea Institute of Science and Technology has developed and is managing the USANS. Research on the homogeneity and mass density of amorphous metallic glasses is one example of achievements using USANS (Kim et al., 2018). On the other hand, the vertical type neutron reflectometer (REF-V) and Bio-Reflectometer (Bio-REF) are used to study the property of interfaces. The disc chopper time-of-flight neutron spectrometer (DC-ToF) and the triple axis spectrometer (TAS) are the neutron inelastic spectrometers in HANARO (So and Park, 2013). The DC-ToF can provide a wide dynamic range by adjusting incident neutron energy and instrument resolution, while the TAS measures low energy magnetic dynamics. Two kinds of TAS are available in HANARO: a thermal neutron TAS (Thermal-TAS) in the reactor hall and a cold neutron TAS (Cold-TAS) in the CNRF. Lastly, the cold neutron activation station, called CONAS (Sun and Park, 2013), accommodates the instruments for the prompt gamma activation analysis (CN-PGAA) (Hoang et al., 2013), neutron induced pair spectrometer (CN-NIPS), and neutron-induced prompt charged particle depth profiling (CN-NDP) (Park et al., 2014b). The cold neutron PGAA has an advantage of significantly reduced gamma ray background while the capture reaction rate in a sample is comparable to the rate of thermal neutron PGAA.

Neutron irradiation using vertical holes

Fuel and material test

HANARO uses either instrumented or non-instrumented capsules for testing nuclear fuels, new materials under development such as semi- and super-conductor, and materials for current as well as coming nuclear power industry (Choo et al., 2014; Kim et al., 2017). In case of an instrumented capsule, the temperature is controlled up to 700 °C. The thermal and fast neutron fluences are also monitored typically using Nb-Ag and Fe-Ni-Ti wires, respectively. After an irradiation in HANARO, structural parts of the capsule including fluence monitors are dismantled in a hot cell, and specimens are transported to the Irradiated Materials Examination Facility where the fuel burn-up, mechanical properties, and so on are measured and analyzed. Further development of the capsule technology is underway, targeting a fast neutron induced material damage of up to five displacements-per-atom. The Fuel Test Loop (FTL) is the other way to test nuclear fuels (Ahn et al., 2008).

Neutron activation analysis

HANARO provides three stations for neutron activation analysis (NAA). One station is for the instrumental NAA that uses pneumatic transfer systems with which an encapsulated sample in a polyethylene rabbit is irradiated in the reflector region (Chung et al., 2008). There are three irradiation holes with different neutron flux levels and spectra. The Cd ratio, which is a measure to show how well the neutrons are thermalized, is in the range of 10–85 for gold. The other two stations are for the prompt gamma activation analysis (PGAA) using the standard beam port #1 in the reactor hall and cold neutrons in the CNRF, instead of using vertical holes. The thermal neutron PGAA uses the Bragg diffraction technique using pyrolytic graphite crystals (Byun and Choi, 2000). The thermal neutron flux at the sample position is of the order of $10^8 \text{ cm}^{-2} \text{ s}^{-1}$.

Neutron transmutation doping

Neutron transmutation doping (NTD) is regarded as the most effective way to obtain uniform resistivity in a single crystal silicon ingot. HANARO has two vertical holes for NTD. The NTD facility irradiates silicon ingots from 12.7 cm (5 in.) to 20.3 cm (8 in.) in diameter and up to 60 cm in length. A neutron screen inside of an ingot container is designed to yield a flat axial resistivity distribution in an irradiated ingot, and the container rotates at a rate of 20 rpm or less in order to obtain the radial resistivity as uniform as possible. The irradiation time is determined by the initial and target resistivities and some constants. The constants were empirically obtained in HANARO for not only initially n-type but also initially p-type silicon (Kim et al., 2009). A self-powered neutron detector, which is mounted at the top of hole sleeve, monitors the neutron fluence during irradiation. In addition to commercial

services, HANARO conducts R&Ds that focus on how to enhance the uniformity of resistivity and how to reduce crystal defects due to fast neutrons and gamma heating.

Radioisotope (RI) production and R&D

The RIs being produced in HANARO include ^{131}I , ^{99}Mo by (n,γ) reaction, ^{192}Ir , ^{166}Ho , and so on for the medical and industrial sectors (Park et al., 2010). A target material is encapsulated in an aluminum RI capsule for the irradiation in HANARO. An irradiated capsule is put in a cask and transported to the RI production facility (RIPF). The RIPF has four banks and each bank consists of various number of concrete or lead hot cells. Each bank, except Bank II that is a multi-purpose bank, has a production line dedicated to specific RI products: Bank I for ^{192}Ir source for non-destructive test equipment, Bank III for iodine pharmaceuticals such as ^{131}I metaiodobenzylguanidine (mIBG), and Bank IV for $^{99\text{m}}\text{Tc}$ and ^{188}Re generators.

The RI-related R&Ds in HANARO extend to radioisotope thermoelectric generators, micro battery, radiolabeled compounds for medical use, and radiotracers and systems for the diagnosis of, for instance, a petro-chemical plant during operation. In addition, HANARO has developed a technology of ^{99}Mo production by fission using a low enriched uranium target. The technology will be implemented in the Kijang research reactor (Park et al., 2014a), which is under construction in Korea.

International collaboration

As a multipurpose and high neutron flux research reactor, HANARO provides not only domestic but also international users with facilities and instruments in a wide range of R&D areas and neutron irradiation services. The HANARO complex, incorporating the Nuclear Training and Education Center of KAERI, is one of the International Centers based on Research Reactor (ICERRs) designated by the IAEA. Through bilateral arrangements facilitated by the IAEA, the HANARO resources are available to organizations and institutions of IAEA member states. The Korean Neutron Beam Users Association (KNBUA) is a member of the Asia-Oceania Neutron Scattering Association (AONSA), which is an affiliation of neutron scattering societies and represents users in the Asia-Oceania region. HANARO serves the local KNBUA as well as international researchers in the region by providing the facilities, sharing R&D results, and disseminating knowledge through, for instance, the neutron school. HANARO is also involved in the Forum for Nuclear Cooperation in Asia (FNCA). HANARO recognizes that these international collaborations are mutually beneficial. In addition, KAERI provides foreign countries with its research reactor technology on either commercial or non-commercial bases. Examples include construction of a research reactor complex, installation of a specific unit facility such as a cold neutron source, and supply of equipment such as a tracing equipment using radioisotopes. In addition to HANARO as an ICERR, Korea more broadly provides international users with the infrastructures for the education and training through the AGN-201K reactor at Kyung-Hee University which is the Internet Reactor Laboratory designated by the IAEA.

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Dhruva—A 100 MWth Indian Research Reactor

Prabhakar V. Varde, C.G. Karhadkar, and Tej Singh, Reactor Group, Bhabha Atomic Research Centre, Mumbai, India

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Introduction

In Sanskrit, Dhruva means ‘immovable or unshakable’ and refers to ‘pole star,’ a reference to robust, reliable and safe reactor system. This multi-purpose reactor was built to provide the needed support to an ambitious ever evolving Indian nuclear program. The proposal for building this reactor, referred as R-5 during the project stage that stands for 5th reactor at BARC (Bhabha Atomic Research Centre), was initiated and discussed in 1971 and the first criticality was achieved on August 8, 1985 at 0220 h. **Fig. 1** shows an external panoramic view of Dhruva reactor located at BARC campus. The high power and high thermal neutron flux of $2.0 \times 10^{14} \text{ ncm}^{-2} \text{ s}^{-1}$ and host of experimental beam hole positions and irradiation facilities makes this reactor as one of the top operating research reactors now in the world (IAEA, 2019). Dhruva is a vertical tank type reactor, located in a pool of water called a vault and surrounded by heavy density pile block concrete that works as the biological shield. The reactor uses natural uranium as fuel with aluminum as cladding, heavy water as coolant, moderator and reflector. The primary shutdown devices



Fig. 1 Dhruva reactor—An external view.

comprising of nine shut-off rods having cadmium as absorber of thermal neutrons; and six control/dump valves that dumps moderator from reactor vessel to a dump tank; each provides independent, redundant and diverse modes that assure long term sub-criticality. To isolate the reactor from the outside environment, the reactor and associated process and safety systems are housed in a concrete containment system. For the postulated very unlikely accident scenario that might pose threat of radiation release from the reactor systems, the containment along with reactor building isolation features acts as physical barrier to mitigate and eliminate radiation consequences in public domain. The demineralized water acts as secondary coolant while sea water is used as tertiary coolant or ultimate heat sink during normal operation of the reactor.

The reactor has logged in ~35 years of smooth and safe operation and continues to provide needed research and development support to the nuclear power program and isotope requirements in healthcare, agriculture and industries. Material irradiation studies in support of science and engineering developments has also been one of the areas where Dhruva has been utilized extensively.

Indian research reactor program – A brief overview

For initiating a nuclear programing any country, building of research reactor is the most effective option to make the robust foundation for development and harnessing the application of nuclear energy (IAEA, 2016a). Research reactors in India provided the right ecosystems for, not only Indian nuclear power program, but also R&D support toward meeting the nation's aspirations in application of radioisotope in health care, agriculture and industry.

In 1955 construction of the first Indian research reactor, Apsara—a swimming pool type, 1.0 MW research reactor started and in 1956, the reactor achieved its First Criticality. Apsara operated till 2009, when it was decommissioned after more than 50 years of dedicated service to the nation in the area of nuclear science and technology (Agarwal et al., 2006a). Even when Apsara was in operation, a need was felt for a reactor having higher power and neutron flux more than 10^{13} ncm⁻² s⁻¹. In July 1960 Cirus—a 40 MWth reactor was commissioned (BARC, 2010a). When Apsara on one hand offered a facility for nuclear R&D and training of staff, the Cirus became, as it is perceived, the workhorse of Indian nuclear program. In fact, the initial staff for Indian nuclear power and research reactor design and operations program got their training in reactor design, operation and radiation protection aspects in Cirus reactor. Along with building of Cirus, the fuel fabrication facility, low level waste, water treatment facility and also fuel reprocessing facilities were also developed, at BARC. Between 1970s and 1980s small research reactors, like Zerlina, Purnima series (I, II and III) were developed and operated for lattice physics experiment to firm up the design of Dhruva, PHWRs, Fast breeder reactors and ²³³U fuelled KAMINI reactor.

The Dhruva reactor was commissioned in 1985 and the country had a high flux facility that enabled the needed enhanced capacity of reactor systems for neutron activation analysis, isotope production and engineering experiments which required higher flux (BARC, 2010b). BARC has also been involved in developing the Advanced Heavy Water Reactor (AHWR). To support the experimental requirements for AHWR, a Critical Facility was built and operated mainly to carry out AHWR lattice experiments and core optimization studies. Recently, on September 10, 2018, a 2.0 MWth swimming pool type reactor to augment experimental, isotope and detector testing requirement of the nation has been commissioned (BARC, 2018).

Most of the research reactor program for thermal research reactors, as above, was developed at BARC. India has a three stage nuclear program where the first stage envisages, development of thermal reactors that utilize natural uranium fuel for power production; the second stage comprises a Fast Reactors program that utilizes reprocessed fuel from thermal reactors like Plutonium in the form of Pu-U-C or Oxide along with utilization of Thorium as blankets and breeds fissionable inventory (U-233); the third stage reactor are again thermal reactors using the fissionable inventory available from fast reactor after reprocessing is utilized as fuel like U-233 as fuel in third stage systems (Sinha and Kakodkar, 2003); Here, the Indira Gandhi Centre for Atomic Research (IGCAR), located at Kalpakkam, India has also developed fast research reactors systems. There are two research reactors located at IGCAR, Kalpakkam, viz. a 40 MWe Fast Breeder Test Reactor (FBTR) and 30 kW Kalpakkam Mini (Kamini) thermal reactor. The Kamini reactor is the first research reactor facility in the world operating on U-233 as fuel (IGCAR, 2019). The Expertise available on FBTR has been valuable for construction and commissioning of 500 MWe Prototype Fast Breeder Reactor (PFBR), Kalpakkam (Bhawini, 2019), India.

Major design features of Dhruva

There are over 27 systems in Dhruva that includes (a) process systems required for reactor normal operation like main coolant system, secondary and tertiary coolant system, reactor regulation system, Off-site or Class IV power supply systems, etc.; (b) reactor safety systems which remains standby and gets actuated when there is a demand due to reactor reaching some limiting conditions or a parameter reaches safety system settings, like reactor shutdown systems, decay heat removal system. Systems like Emergency Core Cooling System (ECCS), reactor building Emergency ventilation system are called engineered safety features (ESFs) that come into action only to eliminate/mitigate the consequences of emergency scenarios. There are systems which are also required for actuation and operation of safety systems, called safety support systems, like emergency power supply systems, and compressed air systems have also been provided for the reactor. A simplified representation of process flow schematics of Dhruva is shown in Fig. 2.

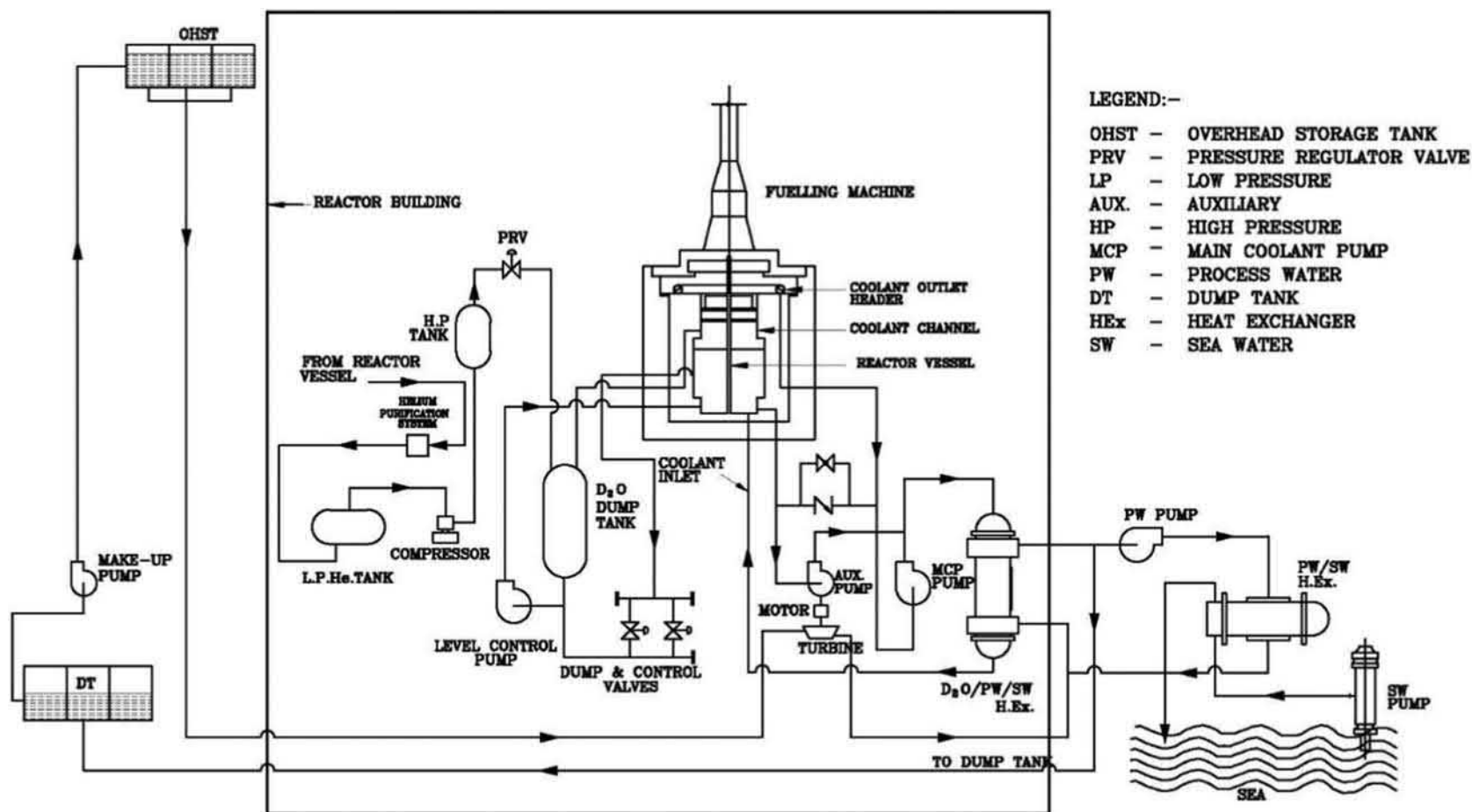


Fig. 2 Simplified representation of process flow schematic of Dhruva (Agarwal et al., 2006b).

Reactor physics

As mentioned above, Dhruva is a thermal neutron, heterogeneous, natural uranium fueled reactor, moderated, cooled and reflected by heavy water and has a rated power of 100 MWth. The reactor is natural uranium fueled with seven pin cluster assemblies. The reactor core is contained in a stainless-steel vessel and the core consists of 146 lattice positions including irradiation positions and 9 shut off rods (SOR). The Pneumatic Carrier (PC) facility in pile facilitate short-term irradiation, ≤ 1.0 min, of sample. The core configuration of Dhruva showing apart from fuels, shut-off rods, other experimental assemblies, is given in Fig. 3. Heavy water in the reactor vessel outside the fuel region serves the purpose of top and bottom axial reflectors (~ 30 cm) and also of radial reflector (~ 60 cm). The reactor has a maximum thermal neutron flux of $2.0 \times 10^{14} \text{ ncm}^{-2} \text{ s}^{-1}$ corresponding to a maximum fuel channel power of 1170 kW to meet the demands of research and development in the areas of science and engineering.

The reactivity control and shut down margins are estimated for the most reactive loading with the coolant present in the channels and experimental and irradiation assemblies absent. The reactor power regulation is achieved by varying the moderator level with constant inflow and variable outflow in the reactor vessel. Three independent channels of instrumentation are given for the important safety parameters.

Pile block

The reactor cylindrical pile block is made of heavy density concrete (density 3.5 g/cc). The cylindrical reactor vessel, made of 19.0 mm thick stainless steel—304 (SS-304-L) having an internal diameter of 3720 mm and height of ~ 7000 mm, is located in

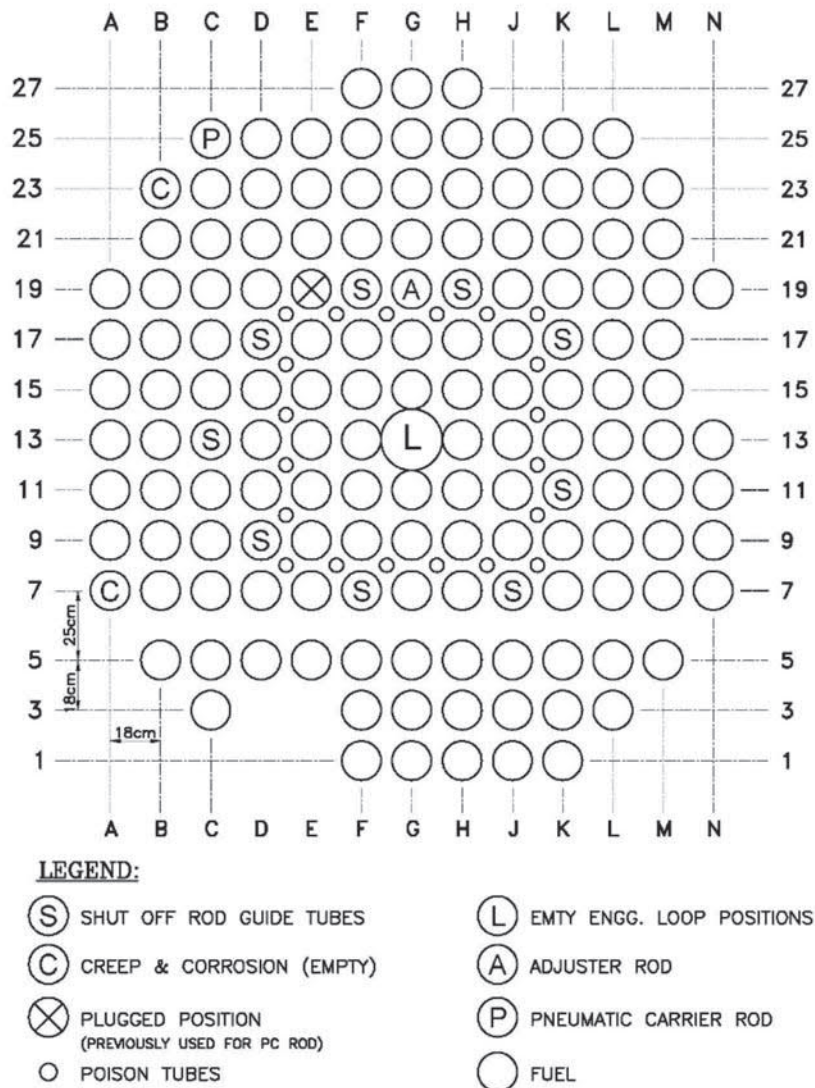


Fig. 3 Dhruva pile positions configuration (BARC, 2010b).

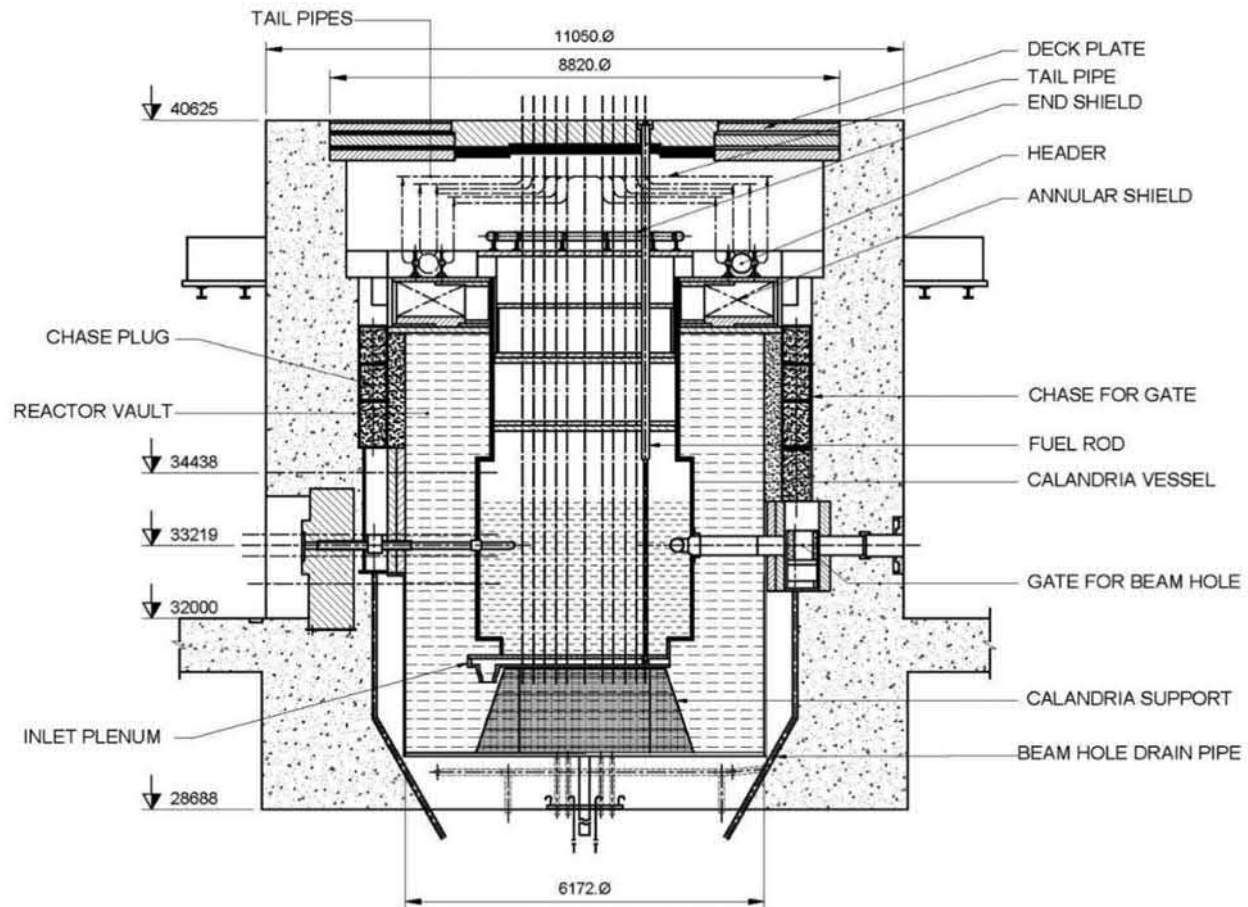


Fig. 4 General arrangement of reactor pile block of Dhruva (Agarwal et al., 2006b).

reactor vault as shown in Fig. 4. The pile block concrete and vault water thickness measure 2.4 m and 1.2 m, respectively and provide shielding around the reactor vessel such that radiation field on contact with the concrete surface is well within the limits i.e. < 0.1 mR/h (1 mSv/h). The reactor vessel is supported on the reactor support structure in the vault. The vault has SS-304 lining to reduce the chance of seepage of vault water. The carbon steel end and annular shield ensures that radiation field on reactor top is well within the stipulated limit. The SS inlet plenum and top tube sheet of the reactor vessel has lattice holes to accommodate 146 lattice positions with precision alignment in axial and radial direction to enable smooth entry of ~ 10 m fuel into the Zircaloy guide tubes in each lattice position and other assemblies. Typical core configuration is comprised of 128 fuel assemblies, 9 shutoff rods, and rest for Adjuster Rod, Pneumatic carrier rod, isotope tray rods and corrosion loop experiments and one in-pile loop. The Shutoff rod positions are cooled by compressed air while the tray rods are cooled by heavy water at reduced rate compared to fueled core positions.

The pile and reactor vessel have 16 positions distributed around for beam tube experiments in addition to three positions which are reserved for neutron detectors for nuclear instrument channels as shown in Fig. 5. The beam hole positions in reactor vessel has Zircaloy re-entrant cans.

Fuel

Dhruva fuel comprises of three major parts, a fuel cluster, an aluminum shielding assembly, and a fuel locking arrangement on top that secures the fuel to the guide tube. All of these fit together to form a fuel assembly of ~ 9 m length. The cluster is made by assembling seven natural uranium fuel pins of ~ 3.0 m length assembled and placed inside aluminum flow tube as shown in Fig. 6. The spacer in the fuel cluster ensures adequate fuel pin separation. The top of the flow tube is connected to aluminum shielding portion by a pin welded to assure positive connection and the top of the aluminum shielding rod is connected to the SS shielding and locking arrangement. The locking arrangement along with backup features keeps the fuel in position within the guide tube. The top and bottom split bulge in the fuel provides adequate flexible support in guide tube bottom and middle position to arrest vibration of the fuel.

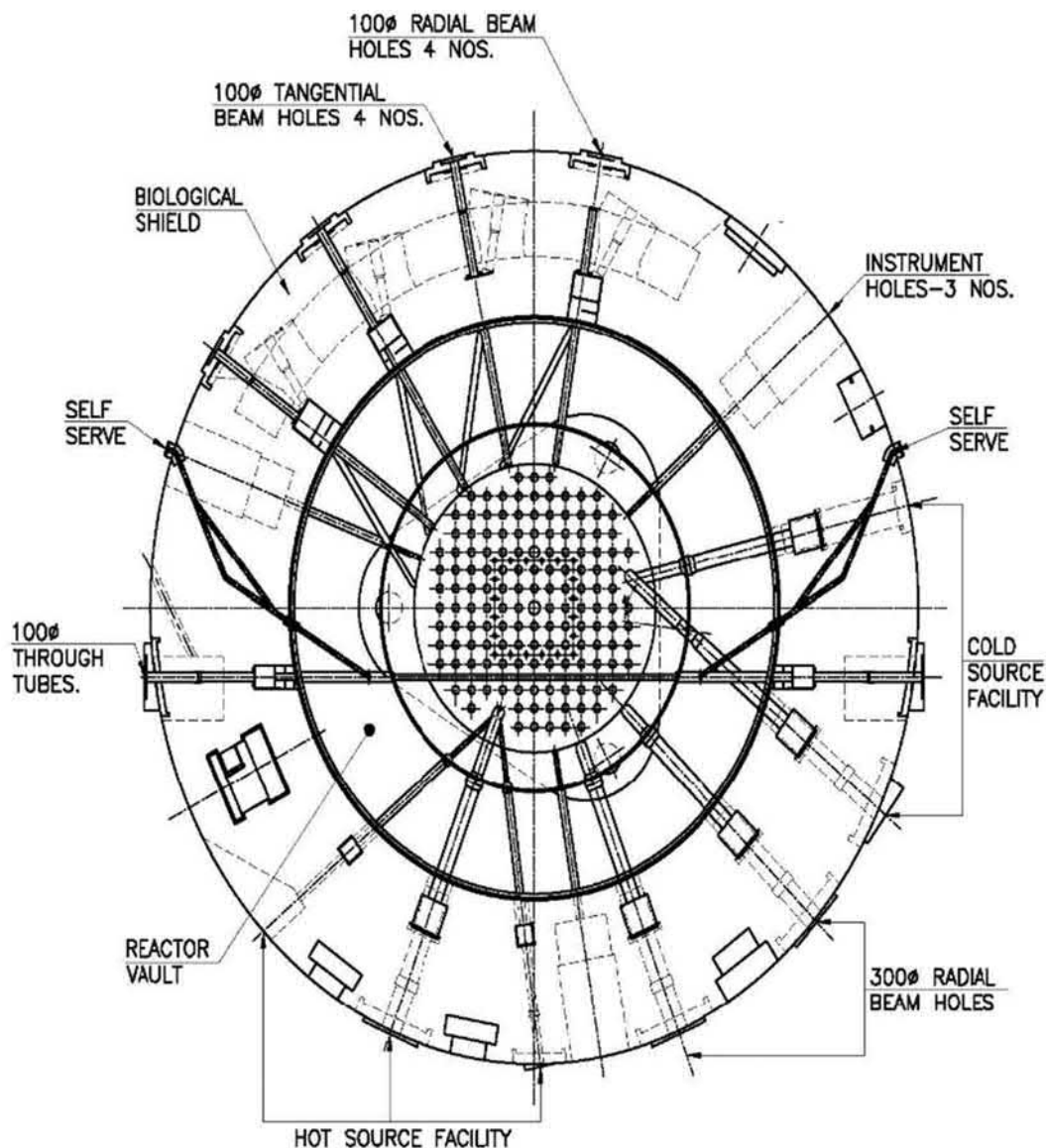


Fig. 5 Beam hole arrangement in Dhruva—A plan view (BARC, 2010b).

Operating experience

The reactor has operated with a high availability factor adequately meeting the research and development needs, isotope requirements for healthcare, agriculture and industrial and irradiation studies on materials. The management of heavy water system that serves as coolant, moderator and reflector is one of the major systems from operation and maintenance considerations along with reactor fuel handling. The major incident free operation of these systems along with reliable operation of other systems has been a significant hallmark of Dhruva operations. The aging studies and refurbishment experience at the Cirus reactor facility have provided insights to initiate an aging management and refurbishment program for Dhruva.

Even after the service of more than 30 years while maintaining high availability factors, system modifications and retrofit activities were performed to continue to safely operate Dhruva. These upgrades considered national and international practice and standards, lessons learned from operating experience, changes to regulatory requirements, situations encountered due to obsolescence, additional insights available from periodic safety analysis of the plant, and international events.

In 2017, Dhruva reactor completed over 32 years of operation but some mechanical structural systems needed immediate attention to extend the operation by at least 5 years. Hence, a 2 month planned shutdown was scheduled and major jobs performed during this period were (a) rectify a minor leak in one of the three primary to secondary heat exchangers, (b) a complete overhaul of the fueling machine—a 300 Ton movable and complex mechanical structural system, and (c) replacement of four 900 mm rubber expansion joints in the light water secondary cooling and shutdown cooling header. With these projects completed, it is expected that the Dhruva refurbishment plan can be extended by another 5 years.

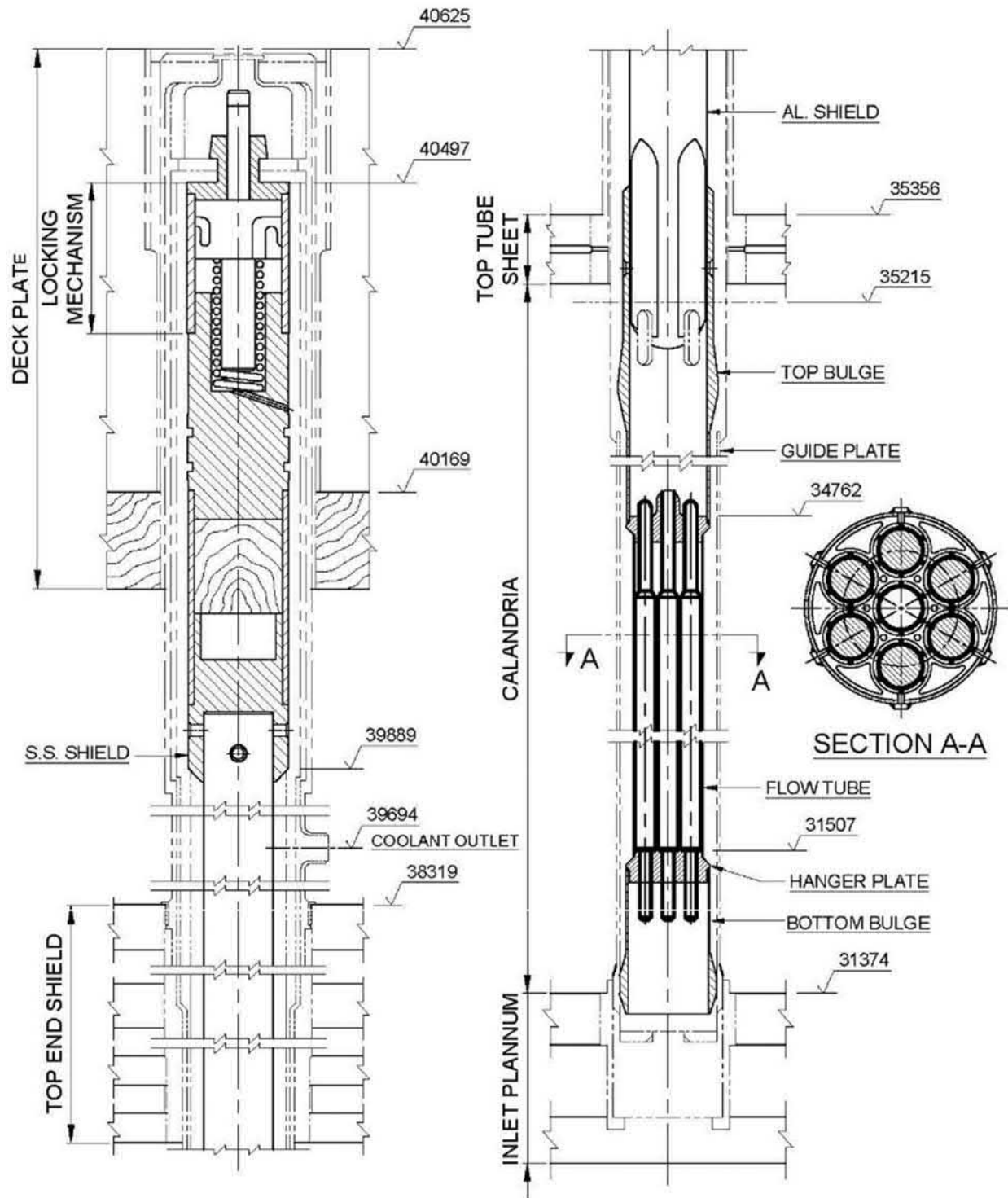


Fig. 6 Simplified representation of Dhruva fuel assembly located in reactor guide tube (Agarwal et al., 2006b).

Reactor utilization

Given the international concern regarding under-utilization of research reactor (IAEA, 2016b), the number of irradiation and experimental facilities provided in Dhruva reactor makes this reactor well utilized facility. The isotopes produced in the reactor are used for healthcare, agriculture and industrial applications. The isotope irradiation tray rods enable production of isotopes and regular



Fig. 7 A view of Dhruva reactor hall showing experimental facilities connected to the neutron beam-hole positions (Rao, 2004).

delivery to cater to the market demands. The pneumatic carrier facility of the reactor facilitates short term irradiation of the order ranging from few seconds to a couple of minutes of material for many experimental studies including neutron activation analysis.

The inter-university consortium acts as link between the academic institutions and reactor facility to enables the students and faculties to perform research and experiments using neutron beam facility in the reactor. The 16 neutron beam lines in the reactor and associated setups and instrumentation offer the researcher with source of experiments for R&D in the area of materials, physics, and chemistry research. The neutron activation analysis performed enable understanding the atomic arrangements in the material. There are beam holes for fission fragment studies and neutron tomography towards understanding internal structure of material and components. Apart from this, a provision also exists for engineering experiments in support of new reactor development program. Fig. 7 shows a view of reactor hall showing utilization of beam holes for various scientific and engineering studies.

Final remark and conclusion

Indian experience has been that establishment of research reactor programs is not an option but a necessity for initiating and maintaining nuclear program for societal applications. Dhruva reactor has been utilized effectively and efficiently to support research programs in the area of neutron scattering, activation analysis, material testing and characterization on one hand and isotope production for societal and industrial applications on the other. The safe operation of over 34 years of Dhruva and an accumulated operating experience of research reactors over 140 years testifies to the maintenance of highest standard of safety in design and operations in research reactors at BARC.

Since the Dhruva reactor will be completing 40 years of operation in 2025, as an advance action, the aging studies for reactor systems have been performed and a draft document on refurbishment program has been prepared and undergoing safety reviews. It is expected that after completion of the refurbishment plan, that the reactor life will be extended by another 20 years.

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The OPAL Research Reactor

David Vittorio and Mark Summerfield, Australian Nuclear Science and Technology Organisation, Lucas Heights, NSW, Australia

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Introduction

History and context

The major nuclear research and development activities in Australia, including the operation of infrastructure and businesses associated with these activities, are conducted by the Australian Nuclear Science and Technology Organization (ANSTO) (<https://www.ansto.gov.au/>), which is an Australian Government organization that has been in existence under its current Act of Parliament (<https://www.legislation.gov.au/Details/C2017C00304>) for a little over 30 years. In practice, the nuclear organization has existed since the mid-1950s as ANSTO was created from the Australian Atomic Energy Commission (AAEC) that came into existence in 1953. The AAEC was transformed into ANSTO in 1987, with the recognition that the basis for its existence had changed from being an organization that conducted research and development focused toward realizing nuclear power for Australia, into an organization that undertook scientific research and development, and commercial activities in a range of non-power nuclear technology applications. Australia has no nuclear power plants and currently has legislation in place preventing these plants from being constructed and operated.

The AAEC owned and operated two research reactors and both were transferred in operational states to ANSTO's ownership in 1987. The first and largest was the HIFAR reactor, which was a UK designed DIDO class reactor that was cooled and moderated by heavy water, had a thermal power of 10 MW, and which operated safely and successfully between 1958 and 2007. HIFAR has been shutdown and is currently in a care and maintenance state. It is planned that HIFAR will be decommissioned in the future. The other research reactor was a 100 kW thermal Argonaut class reactor called MOATA that which operated between 1961 and 1995. MOATA was the first reactor to be decommissioned in Australia in 2009.

ANSTO's current operations are primarily based around large-scale scientific infrastructure that are largely been funded by Government, with the rationale for the funding being scientific research and development, production of radio-isotopes, particularly for health and medicine and the safe and secure management of radioactive wastes. The most important piece of nuclear infrastructure in Australia is the OPAL research reactor, which is multi-purpose by design with its main applications being neutron beams for research and irradiation of many different materials, some of which allow manufacture of radioisotopes for health and medicine. OPAL is a key part of the supply chain for the manufacture of radioisotopes and radiopharmaceuticals for the health sector and other radioisotopes for a number of other industrial and research sectors.

The need for the OPAL reactor was established in the late 1990s when the Australian Government provided funding for a "replacement research reactor" for the HIFAR reactor. The rationale for the new reactor was a continuation of the decades of nuclear based research and development work performed in Australia, a recognition that the facilities available for that work could be upgraded and enhanced with a new facility, and a strategic objective for Australia to remain as an engaged and knowledgeable participant in the international nuclear science and technology community.

Operational history and major milestones

It was decided to build the OPAL reactor (as a replacement to HIFAR) on the same Lucas Heights site as HIFAR to take advantage of the existing site infrastructure. After a rigorous preparation and tender process, in 2000 ANSTO signed a contract with the Argentine company INVAP S.E and its Australian alliance partners, John Holland Construction and Engineering Pty Ltd. and Evans Deakin Industries Limited, for the design, construction and commissioning of OPAL. ANSTO managed the overall design and construction process.

Design and construction were undertaken between 2000 and 2006. The most significant issue that occurred during the construction period was the discovery of two geological features in the excavation zone for the reactor. Construction was halted between April and June 2002 while investigations were undertaken to establish whether the features had significant potential for displacement at or near the ground surface (Pillans, 2003). The outcome of the investigations, which used paleomagnetic dating techniques among other methodologies, were that the faults were older than 7,80,000 years with the three sampled sites yielding interpolated paleomagnetic ages of 2 ± 1 million years, 9 ± 4 million years and 14 ± 8 million years. This meant the faults were classified as “non-capable” and regulatory approval was given to proceed with construction at the reactor site.

An operating license was granted in July 2006, with OPAL achieving first criticality on August 12, 2006, and full operating power in November 2006. The Prime Minister of Australia officially opened OPAL in April 2007.

OPAL is used for:

- Irradiation of target materials to produce radioisotopes for medical and industrial applications;
- Research in the field of materials science using neutron beams and associated instruments;
- Analysis of minerals and samples using neutron activation techniques and delayed neutron activation techniques;
- Irradiation of silicon ingots for use in the manufacture of electronic semiconductor devices.

In July 2007 (<https://www.arpansa.gov.au/sites/default/files/legacy/pubs/regulatory/opal/modifiedfuelstatement160608.pdf>), during a routine reactor shutdown and following a fuel change, the reactor core was examined by video camera as part of the process for returning to power and to ensure correct fuel configuration. During this examination, it was observed that a fuel plate was displaced from its position in a fuel assembly. The subsequent investigation found that several fuel plates had been displaced by varying amounts. The reactor was placed in a safe shutdown condition to undertake detailed investigation and rectification. A design change for the fuel assemblies was proposed and the associated safety case for restarting reactor operations was submitted to the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) in late 2007. ARPANSA considered the submission and approved restart of OPAL using the new approved fuel assembly design in May 2008.

Since 2008, the combination of full power operational time, excellent reliability, high levels of utilization, particularly for supply of radioisotopes, high quality research publications for neutron beam applications, neutron activation analysis for research and industrial clients and reliable irradiation of more than 50 tonnes of silicon ingots per year, has positioned OPAL as one of the most successful research reactors in the world. The OPAL Reactor aims to operate for 300 days per year at a reliability of over 98%. Between 2010 and 2019 (10 year period), OPAL has operated for 2942 days.

A Periodic Safety Review (PSR) was conducted in 2011 as part of a license condition imposed by ARPANSA. The PSR re-examined the safety of the OPAL reactor taking into consideration operating experience from commissioning and the first 2 years of operation. Apart from submitting the PSR to ARPANSA, ANSTO was also required to seek an international peer review of OPAL. The outcomes from the PSR and an action plan were reviewed and accepted by ARPANSA in 2014 (<https://www.arpansa.gov.au/sites/default/files/legacy/pubs/regulatory/opal/sor-opal2014.pdf>).

Design and operational developments

The major design changes and operational developments instituted at OPAL since it was granted a license in 2006 are summarized below.

i) Fuel Assembly design

In 2008, ARPANSA considered the evidence and assessments provided by ANSTO and other parties on the cause of the fuel plate displacements in 2007 and the proposed rectifications to prevent the displacements from occurring again. The CEO of ARPANSA concluded that the displacement of the fuel plates was a result of a combination of design factors, primarily in that there was no secondary stopper mechanism to prevent movement of fuel plates if a primary fixing failed. A new fuel design with a mechanical stopper incorporated into it was proposed and approved by ARPANSA. OPAL returned to power operation in May 2008, and the new fuel design has been shown to be satisfactory for long term operations.

ii) Heavy Water Isotopic Purification system

Following the discovery of the ingress of light water into the heavy water reflector vessel during the early period of OPAL operation, ANSTO and INVAP proposed to design, construct and operate a heavy water purification system to assist with maintaining the heavy water purity and hence the neutron flux performance of OPAL. The heavy water isotopic purity system was commissioned in 2011 and is operated for approximately 60 days per year to maintain a high level of heavy water purity.

iii) Cold Neutron Source (CNS) system

During the first 5 years of operation, the CNS system operation and reliability was affected by recurring failures of the rotating screws inside the helium gas compressors of the system. After operational and engineering investigations, it was ascertained that the compressors were not being cooled sufficiently and that the subsequent material expansion was causing mechanical failure of the screws. The solution was to redesign the heat exchanger and modify the oil operating temperature used in the helium compressors. This work resulted in significant performance improvements for the CNS system, which since 2013 has been operating reliably.

iv) Maintenance Strategy Optimization

As part of the OPAL operational plan, an asset management program for OPAL has been implemented that utilizes modern engineering asset management methodologies and techniques. A significant component of the OPAL Asset Management program was a project on maintenance strategy optimization. The central aim of the project was to define the optimum mix of maintenance tasks for reactor systems and the use of the resources, both human and engineering based, to achieve the optimum mix. Reliability centered maintenance is the main methodology used and the main components of the maintenance strategy optimization were to specifically identify systems and components that:

- could be maintained via Condition Monitoring by using appropriate surveillances, inspections, testing, monitoring, trending or analysis tasks. Failure modes for these systems and components are addressed with defined potential to failure (P-F) intervals;
- should be overhauled, restored or replaced at fixed time intervals, and which primarily address failure modes that exhibit a predictable wear-out failure pattern;
- could be corrected if an unexpected failure occurs or an expected failure at the end of life which did not penalize performance.

The strategic approach to maintenance has been one of the main contributors to OPAL's excellent operational and reliability performance results.

OPAL utilization

OPAL continues to be utilized at a high level in the main application areas for which it was designed.

i) Radioisotopes (<https://www.world-nuclear.org/information-library/country-profiles/countries-a-f/appendices/australian-research-reactors.aspx>; <https://www.ansto.gov.au/business/products-and-services>)

ANSTO manufacture and supply a range of radiopharmaceuticals, radiochemicals, kits and accessories for use in research, industry and the health sector. The neutron-rich radioisotopes are sourced from OPAL, which has an extensive range of irradiation facilities located in the reflector vessel surrounding the reactor core, with a broad range of neutron fluxes, neutron energies and cooling capabilities available so that a range of different radioisotopes for medicine, industry and research can be produced.

The most extensively used radioisotope in the world for nuclear medicine is Technecium-99 m (Tc-99 m), which is mainly used as diagnostic tool for many different types of patient studies. OPAL was one of the first research reactors in the world to only use low-enriched uranium as the target for neutron irradiation in the production of Molybdenum-99 (Mo-99), which is the parent radioisotope to Tc-99 m. OPAL has 12 irradiation facilities which can be used for Mo-99 production and this capacity, combined with the ANSTO Nuclear Medicine Mo-99 facility, enables ANSTO to supply bulk Mo-99 within Australia and throughout the world, including major exports to the USA and Asia.

Other medical radioisotopes produced by OPAL include, I-131, Sm-153, Cr-51, Y-90 and Lu-177.

ii) Neutron scattering (<https://www.ansto.gov.au/research/facilities/australian-centre-for-neutron-scattering>)

The Australian Centre for Neutron Scattering (ACNS) is a major research facility for neutron science that comprises a suite of neutron instruments that use a range of techniques for scientific investigations in physics, chemistry, materials science, medicine, environmental science and other fields. There are 15 neutron beam instruments at the ACNS, which are classified into four main groups: diffractometers, small-angle spectrometers, imaging and reflectometry instruments and inelastic spectrometers. Most of these instruments have been funded by the Australian Government with some designed and constructed coincident with the design and construction of the OPAL reactor with others funded, designed and constructed over subsequent years as the science program grew. The construction of one instrument (a cold neutron triple-axis spectrometer) was funded by the Taiwanese Ministry of Science and Technology. Another instrument (a BioRef reflectometer) was originally from the Helmholtz Zentrum Berlin and was transferred to the ACNS following the shutdown of the Helmholtz Zentrum Berlin reactor.

Most of the instruments and operating cabins are housed in a Neutron Guide Hall located next to the OPAL reactor. The neutrons from the reactor reach the instruments in the Neutron Guide Hall via various neutron guides that penetrate into the reactor reflector vessel. The Neutron Guide Hall also accommodates sample preparation areas, laboratories and other technical support facilities. Some of the neutron beam instruments are housed in the Reactor Beam Hall which surrounds the concrete reactor block in which the reactor is located. The ACNS also operates three X-ray instruments, a helium polarizing instrument and a physical properties measurement system. Since OPAL commenced operations in 2007, there have been more than 1200 research publications from ACNS and their international and national collaborations.

- iii) Neutron activation analysis (<https://www.ansto.gov.au/research/facilities/opal-multi-purpose-reactor/overview/neutron-activation-analysis-and-neutron>)

There are two major facilities that use neutron activation analysis (NAA) techniques at OPAL. The first is dedicated to research and commercial clients with a wide variety of end uses including environmental science, geoscience and mineralogy, forensics and counter-terrorism, archaeology, agriculture and materials science. NAA is a sensitive method of quantitative multi-elemental analysis where samples are irradiated in the OPAL reflector vessel and after irradiation, the specific activity of radionuclides in the sample can be determined by measuring the energy and intensity of characteristic gamma rays that are emitted. To offer the greatest flexibility to clients and collaborators, both the relative method of standardization and the k_0 -method of standardization have been implemented.

The second facility is dedicated to the Delayed NAA (DNAA) technique for uranium ore assays for clients from the minerals and mining sectors.

- iv) Neutron Transmutation Doping of silicon (<https://www.ansto.gov.au/business/products-and-services>)

There are six vertical irradiation facilities in the OPAL reactor's reflector vessel that are used to irradiate silicon ingots for commercial clients. The irradiation facilities in OPAL have a high 'thermal to fast' neutron ratio (or Cd ratio) that assists with minimizing silicon crystal lattice damage. The facilities have a range of diameters, including the capability to irradiate 8" diameter ingots. OPAL has a high irradiation capacity with more than 50 tonnes being undertaken per annum.

OPAL reactor design

The OPAL Reactor is an open pool type research reactor as shown in **Fig. 1** with an operating power of 20 MW thermal. The fundamental safety objective in the design is the protection of the public, the facility personnel and the environment from exposure to radiation due to its operation. A "defense-in-depth" approach is applied throughout the facility, providing multiple levels of protection against the accidental release of radioactive materials.

The reactor has a compact core, which maximizes the flux of neutrons available for radioisotope production, irradiation services and research. Heavy water is used as the reflector to sustain the nuclear reaction. The heavy water is contained in the Reflector Vessel shown in **Fig. 2** that surrounds the core. The core consists of 16 Fuel Assemblies (FAs) that contain Low Enriched Uranium (LEU). Fission heat is removed by water circulating through coolant channels within the FAs. Reactivity is controlled by five control rods,



Fig. 1 OPAL research reactor pool.

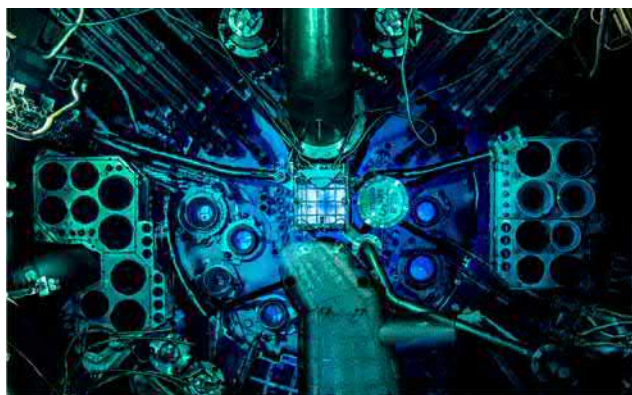


Fig. 2 OPAL research reactor core and reflector vessel.

four of which have neutron-absorber plates inserted into the core in a cross-shaped array and the fifth has a central cruciform shaped absorber plate. The core is thus divided into four portions that each contain four FAs.

The core and the Reflector Vessel are positioned close to the bottom of the 12.6 m deep reactor pool (RPO). The RPO is connected to a Service Pool (SPO) by means of a Transfer Canal. The SPO provides a general working area and enough space to store the spent fuel generated from 10 years of reactor operation.

The Primary Cooling System (PCS) removes the heat from the core by forced upward circulation of water and transfers the heat to the Secondary Cooling System (SCS). A Core Chimney above the Reflector Vessel contains the core coolant before it enters the pump suction line of the PCS piping and provides an additional enclosure for water that protects the core in case of a Loss of Coolant Accident (LOCA).

The RPO inventory is cooled by the Reactor and Service Pool Cooling System (RSPCS), whose main function is the cooling of irradiation rigs. This system also provides long-term pool cooling to the RPO and SPO to extract decay heat.

Fuel design

The reactor core is loaded with plate-type FAs containing uranium silicide powder dispersed in pure aluminum with a maximum uranium density of 4.8 gcm^{-3} . The FAs are square-shaped in cross-section, each with 21 flat fuel plates separated by a channel for coolant flow. Each fuel plate consists of a meat of U_3Si_2 powder (19.75% of U^{235} in weight enrichment) dispersed in an aluminum matrix. The fuel meat is sealed between two aluminum alloy covers (the cladding). The FAs have burnable poison in the form of Cd wires, which are designed to reduce variation in neutron flux over the core life of the FAs. The end box of the FA has an external square section with a circular internal centering hole for coolant flow. The lower end of the inner fuel plates is held by a comb located at the middle of the plates along their width. The side plates of the FA are flat with grooves in their internal faces to allow the positioning and fixing of fuel plates by swaging. The side plates are manufactured in aluminum. There is a handling pin at the top of the FA, attached to the side plates, which allows the fuel to be moved and manipulated with a special handling tool. There are two mechanical stoppers held in place on the side plates and above the fuel plates to provide a secondary mechanism to halt potential movement of the fuel plates.

i) Nuclear physics parameters

The main reactor characteristics and associated core parameters for OPAL are given in [Table 1](#).

OPAL was designed to operate exclusively using LEU fuel and all the uranium targets that are used for Mo^{99} production also use LEU.

Core configuration

The core grid has 16 FA positions in a 4×4 configuration. The core grid is divided into four zones by guide boxes that accommodate five control rod plates in a cross array. The maximum residence time of the FAs in the core is about 180 full power days.

Reactivity control and safety systems

The reactivity control elements consist of Control Rod Plates (CRPs), which are driven by the Control Rod Drives (CRDs) located in the CRD room below the Reactor Pool. The active components of the CRPs are plates made of neutron absorbing material that can regulate the reactor power and shut down the reactor. The CRPs are the core reactivity control part of the First Shutdown System (FSS), which provides the capability to decrease the reactor power level very rapidly as demanded by the First Reactor Protection System (FRPS) or the operators.

Table 1 Main reactor characteristics and core parameters.

<i>General Data</i>	
Type of reactor	Open pool
Nominal thermal power	20 MW
<i>Nucleonic</i>	
<i>Core</i>	
Number of fuel assemblies in equilibrium core array	16
Core dimension	$35 \times 35 \times 61.5$ cm
Net core volume	0.075 m^3
Grid array	4×4
Number of control rods	5
Absorbing material	Hafnium
Core fuel load, average over five Beginning of Cycles (BOCs)	6.23 kg uranium-235
Average at power operation cycle length, reference core	28.8 full power days
Average cycle length, reference core	30.8 days
Maximum power peaking factor, reference core	2.4
<i>Neutronic Data</i>	
Average core thermal flux (average over five BOCs)	$1.07 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$
Average core fast flux (average over five BOCs)	$1.31 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$
Average core thermal flux (average over five EOCs)	$1.18 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$
Average core fast flux (average over five EOCs)	$1.32 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$
Prompt neutron life-time	
BOC, hot/cold ^a	161/165 μsec
EOC, hot/cold ^b	166/171 μsec
Effective delayed neutron fraction (β effective)	
BOC, hot/cold ^a	734/734 μsec
EOC, hot/cold ^b	722/722 μsec
Total reactivity worth (control rods), cold/hot - average over cycle length	19,599/20600 pcm
<i>Nuclear Fuel</i>	
Fuel type	19.75% enriched, U_3Si_2 -Al dispersion fuel
<i>Operational Data</i>	
Full assembly residence time	Approximately 190 full power days
Average discharge burn-up	78,100 MWd/Tn U (46.0%)
<i>Fuel Assembly</i>	
Fuel element type	Plate
Number of fuel elements per fuel assembly	21
Active length	615 mm
Plate thickness	1.35 mm (internal plates) 1.5 mm (external plates)
<i>Thermal-Hydraulics</i>	
<i>Core Thermal Data</i>	
Inlet temperature	Nominal value $37^\circ\text{C} \pm 1^\circ\text{C}$
Average outlet temperature (1900 m ³ /h through the core)	Nominal value $46^\circ\text{C} \pm 1^\circ\text{C}$
Core power density	250 kW/L
<i>Core Hydraulic Data</i>	
Effective coolant flow, minimum	1900 m ³ /h
Coolant velocity in core coolant channel (internal channel)	8.2 m/s
Core pressure drop	205 kPa
<i>RPO Data</i>	
Internal pool diameter	4.5 m
Internal pool height	14.1 m
Internal pool water depth	12.6 m
RPO water inventory	186 m^3

^aAveraged over five BOC.^bAveraged over five EOC.

OPAL has a completely independent Second Shutdown System (SSS) that when operated, partially drains the Reflector Vessel of heavy water. The subsequent neutron leakage from the core is sufficient to shutdown the reactor in a short time. The Second Reactor Protection System (SRPS) controls the SSS and is independent from the FRPS. It also uses diverse technology from that used in the FRPS, thus providing independent and diverse functionality for reactor protection. The SSS will also actuate in circumstances where the FSS fails to actuate.

Reactor control and monitoring

The Reactor Control and Monitoring System (RCMS) is a distributed, computer-based, high-availability system which reads all plant and reactor information and presents it to operators in the reactor control room and other selected workstations. The RCMS enables reactor control, process commands and overall data-management for the operation and monitoring of the facility in normal conditions and to ensure that safety actions are executed under interlock conditions or when limits are exceeded.

From a safety perspective, the FRPS and SRPS have priority over the RCMS and the three systems are functionally, physically and electrically independent.

Primary and secondary cooling system

The Primary Cooling System (PCS) removes the heat from the core during operation at power by forced upward circulation of water and transfers the heat to the Secondary Cooling System (SCS) through the heat exchangers of the Primary Cooling System. The reactor can also be operated in a low-power state where the PCS removes heat from the core via natural circulation and in this case the RPO inventory is enough for safe low power operation.

During forced circulation a core chimney above the Reflector Vessel contains the core coolant before it enters the pump suction line of the PCS piping and provides an additional enclosure for water that protects the core in case of a Loss of Coolant Accident (LOCA).

The RPO inventory is cooled by the Reactor and Service Pool Cooling System (RSPCS), whose main function is the cooling of irradiation rigs. This system also provides long-term pool cooling to the RPO and SPO to extract decay heat. Heat from the RSPCS is transferred to the SCS.

Significant ancillary systems

i) Containment

The reactor containment is the physical barrier that prevents the accidental release of radioactive material from the areas within the containment, namely the RPO and SPO, Reactor Hall, and areas below the RPO that contain RPO water systems and Reflector Vessel heavy water systems, to the environment. The Containment consists of a physical barrier, containment isolation valves, a containment energy removal system and other sub-systems that when initiated, provides automatic closing of valves, heat removal and filtered air circulation for which the containment system is required to provide its protective function.

ii) Electrical

The electrical system is designed to meet demands during operational states and accident conditions. It provides diverse, reliable power sources that are physically and electrically isolated so that any single failure affects only one source of supply and does not propagate to other sources. During normal plant operation, power is delivered from normal electrical supply, and when normal supply is not available, all Safety Category 1 systems receive power from standby diesel generators. Uninterruptible Power Supply (UPS) units also provide reliable uninterruptible power for instrumentation systems to ensure that information is continually available on the plant state.

iii) Radioactive Waste Management

Waste management of radioactive solids includes spent resin handling and storage, the handling and storage of waste from irradiation facilities and the replacement of active components and filters. The Radioactive Liquid Waste Management System (RLWMS) classifies, collects, stores, controls and monitors all liquid waste originating within OPAL. Liquid waste is classified and monitored prior to transfer to the ANSTO site waste treatment facilities. Gaseous emissions from OPAL are monitored by the Air Effluent Monitoring (AEM) System. Any gaseous emission above a preset level will generate signals within the FRPS that will isolate the containment.

iv) Heating, Ventilation and Air Conditioning

Various HVAC Systems are provided for areas outside of the containment for air conditioning, circulation and renewal and to provide the required environment for personnel and equipment.

v) Physical Security System

The physical security system within the facility integrates with protection arrangements at the ANSTO site to provide physical protection to persons, nuclear materials, commercial, scientific and national security information, hardware, software, documentation and operational arrangements. The system also provides protection against unauthorized entry into the Reactor Building.

Experimental support systems

OPAL was designed as a multi-purpose facility with a compact reactor core that has no facilities within the reactor core. All facilities for research and business purposes are either located wholly or partially in the Reflector Vessel that is filled with heavy water and surrounds the reactor core.

Neutron beam facilities

The neutron beam facilities of the OPAL Reactor include:

- i. Cold and thermal neutron source systems.
- ii. Neutron guides for thermal and cold neutrons.
- iii. Neutron beam shutters.
- iv. Auxiliary systems.
- v. Neutron beam instruments.

The neutron guides are used to condition and transport neutrons from the high-density neutron areas located in the Reflector Vessel to the research instruments located in the Reactor Beam Hall and the Neutron Guide Hall.

Other support infrastructure and services include workshops and laboratories, offices, a viewing gallery, cranes, air conditioning and ventilation, electrical supply, communication systems, water supply and drainage and gas supply.

Cold neutron source

The Cold Neutron Source (CNS) consists of a liquid deuterium moderator that operates at a temperature of approximately 20 K. It is located close to the reactor core that increases the neutron yield in the “cold” energy range. Cold Neutrons moderated in this cold fluid are then transported through the neutron guides which transport neutrons to dedicated instruments. The moderator is liquefied and maintained in the liquid state within a natural circulation thermosiphon loop, which has a heat exchanger and a cooling jacket fed with cryogenic helium at 19 K. Both the CNS Refrigeration Cryogenic System and the reactor are capable of operation if the other is shutdown. The CNS is automatically monitored and controlled by a dedicated control and monitoring system that also has a functionally separate protection system designed to protect the CNS.

Thermal neutron sources

The thermal neutron sources comprise two heavy water zones located close to the region of peak thermal flux in the Reflector Vessel. Two neutron beams originate at the thermal neutron source. The neutron spectrum has a temperature in the range from 40 °C to 60 °C. Thermal neutrons are transported through the neutron guides to dedicated instruments.

Irradiation facilities

Radioisotopes are produced by introducing targets into dedicated vertical irradiation positions in the Reflector Vessel. The following irradiation facilities are available at OPAL along with associated laboratories and hot cells.

- a. Bulk Production Irradiation Facilities;
- b. Long Residence Time General Purpose Irradiation Facilities;
- c. Short Residence Time Irradiation Facilities;
- d. Large Volume Irradiation Facilities (currently used for silicon irradiations);
- e. Hot Cells and Auxiliary Facilities;
- f. Interbuilding Pneumatic Transfer System.

The irradiation facilities are controlled and monitored by the previously described RCMS. The range of irradiation facilities offer ability for different sized, targets to be irradiated in different thermal flux energies. Irradiation facilities are also supported by mechanisms to transfer targets into shielded hot cells.

Reference

Pillans, B., 2003. Dating ferruginous regolith to determine fault capability at Lucas Heights, Sydney. In: Roach, I.C. (Ed.), *Advances in Regolith*. CRC LEME, pp. 324–327.

University of Missouri Research Reactor

Leslie P. Foyto, University of Missouri, Columbia, MO, United States

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Introduction

At the close of World War II, the United States Federal Government created the Atomic Energy Commission (AEC) to control the peace-time uses of uranium and its fission products. Everyone saw the overwhelming and devastating results of the “Manhattan Project” by the detonation of the first atomic weapon (Code name “Trinity”) in the Jornada del Muerto desert in New Mexico on July 16, 1945. The AEC immediately began to make public scientific information which had heretofore been in the secret category. The purpose being to stimulate peace-time scientific research for the practical application of nuclear reactors for electrical power generation and for the use of fission products for industrial and medical purposes.

Realizing the stimulus to scientific research in many divisions of the University of Missouri which would result from the installation of a nuclear facility at the University, Dean Huber Croft wrote to University of Missouri President Elmer Ellis in October of 1955 suggesting that he may wish to appoint a committee to survey the need for a nuclear reactor at the University. In November of 1955, President Ellis appointed a committee for this purpose consisting of Dean Bent of the Graduate School, Professor Gingrich of the Physics Department, and Professor Brody of the College of Agriculture, with Dean Croft of the College of Engineering as the Chairman (Croft, 1965; Gingrich, 1983).

In June 1956, Senator Jackson introduced a bill in the Senate to provide financial support from the AEC to universities to furnish nuclear facilities. This action was supported by letters to the AEC by Senators Symington and Hennings from Missouri. To indicate the interest of the University in the possibility of a nuclear reactor in Columbia, President Ellis wrote to both Dr. Alan Waterman, Director of the National Science Foundation (NSF), and Admiral Strauss, Chairman of the AEC, in May and December of 1956. This had been preceded by a visit to Dr. Waterman in Washington by President Ellis, Dean Bent and Dean Croft in November 1956.

During 1957, the committee was busy determining estimates of initial costs and annual operating expenses for reactors as well as advantages and disadvantages of the various types of reactors. It was initially recommended that the reactor should be a “research reactor” and not a “training reactor” with a power level of 1000 kW and a neutron flux of at least 10^{13} neutrons $\text{cm}^{-2} \text{s}^{-1}$. Of the kinds available at that time, the CP-5 reactor at Argonne National Laboratory (ANL) and the Massachusetts Institute of Technology (MIT) reactor, which was under construction at the time, were considered. Additionally, different locations near the University were being studied for the reactor site. Later it was also decided that the reactor design should have the flexibility to allow an increase in power at some future date without a major change in the shielding or the building.

On July 7, 1960, the Internuclear Company presented the “Preliminary Design Report” to the University. The design recommended a modified tank-type reactor (pressurized) having an annular core of fuel elements with a light water moderated center test hole, or “flux trap,” having a neutron flux of 2.23×10^{14} neutrons $\text{cm}^{-2} \text{s}^{-1}$ at 5 MWs and 4.51×10^{14} neutrons $\text{cm}^{-2} \text{s}^{-1}$ at 10 MWs. In March of 1961, the “Preliminary Hazards Summary Report” was submitted to the AEC for their review. On November 21, 1961, the AEC issued Construction Permit No. CPRR-68 to The Curators of The University of Missouri authorizing the University to begin construction of a 10 MW research reactor.

On May 28, 1966, Dr. Glenn Seaborg, Chairman of the AEC and 1951 winner of the Nobel Prize in Chemistry, makes a speech titled “The University Research Reactor—A Sign of the Times” at the dedication of the new research reactor at the University of Missouri (United States Atomic Energy Commission, 1966). Following are selected excerpts from Dr. Seaborg’s speech:

"The reactor facility we are dedicating today is an excellent testimonial to both the fruits and promise of the interdisciplinary approach to scientific and technological achievement. Its construction was motivated to a large degree, I am sure, by the contributions it could make to a broad range of activities on this campus and in this area. As a result, we see here not only the reactor itself but a cobalt-60 irradiation unit, chemistry laboratories, an engineering research laboratory, a physics laboratory, and agriculture and medical research laboratories—all facilities which will be essential to those making the fullest use of the reactor itself." ...

"The University of Missouri Research Reactor Facility is an excellent "sign of the times," then, in our present day world of science and technology. The research reactor's ancestral past, and its plans and promise, are interdisciplinary to the highest degree, and also well exemplify the new and dramatic symbiotic relationship between the pure and the applied aspects of science. We see here research and education going hand-in-hand, and we see embodied here the cooperative interest in both of the activities of the State of Missouri and several agencies of the Federal Government. But above all, we see a stimulus and a tool for the continuing pursuit of excellence in which we all have a vital stake." ...

On October 13, 1966, the AEC issued reactor operating license No. R-103 to The Curators of The University of Missouri authorizing the University to operate the 10 MW research reactor at steady-state power levels up to a maximum of 5 MW. The final design of the University of Missouri Research Reactor (MURR) is a pressurized, beryllium and graphite reflected, open pool-type, light water moderated and cooled heterogeneous system with a central flux trap. A flux trap-type reactor is characterized by a thin fuel region adjacent to a good moderator which thermalizes the neutrons and causes the thermal neutron flux to peak in a region.

On October 13, 1966, MURR establishes initial criticality with the 5.2 Kg core. The fuel material at time of initial startup was a uranium-aluminum alloy with each fuel assembly loaded to a maximum of 650 g of ^{235}U . This type of fuel system had performed very reliably in the Materials Test Reactor (MTR) and the Engineering Test Reactor (ETR) as well as in other reactors throughout the world.

On August 13, 1971, in order to reduce the fuel cycle cost and the amount of ^{235}U needed per MWD of energy produced at MURR, the reactor begins the initial use of the 6.2 Kg uranium aluminide (UAl_x) dispersion core with a maximum loading of 775 g of ^{235}U per assembly. The UAl_x dispersion fuel system was developed at Idaho National Laboratory (INL) for the high flux, high power Advanced Test Reactor (ATR) and subsequently used at the MTR and ETR prior to its use at the MURR.

On July 9, 1974, the AEC issues Amendment No. 2 to reactor operating license No. R-103 authorizing the University of Missouri to operate MURR at steady-state power levels up to a maximum of 10 MW. On July 18, 1974, the reactor is upgraded to 10 MW.

On September 1, 1977, MURR starts a 10 MW, 150 h per week operating schedule. Since that day, MURR has continued to operate on this schedule of being online approximately 90% of available hours. No other research or test reactor, domestically or internationally, operates that many hours per year.

MURR continues to be one of safest, most reliable and versatile university-operated research reactors in the world providing critical isotope production capabilities, neutron scattering facilities and a broad range of analytical and irradiation services to the research community and the commercial sector. Scientific programs include research in archaeometry, epidemiology, health physics, human and animal nutrition, nuclear medicine, radiation effects, radioisotope studies, radiotherapy, boron neutron capture therapy and nuclear engineering; and research techniques including neutron activation analysis, neutron and gamma-ray scattering and neutron interferometry.

Nuclear reactor design

Introduction

The University of Missouri Research Reactor (MURR[®]) is a pressurized, reflected, heterogeneous, open pool-type, which is light-water moderated and cooled (Fig. 1). The reactor is designed and licensed to operate at a maximum thermal power level of 10 MW with forced cooling, or up to 50 kW in the natural convection mode. A unique design feature provides an experimental position (flux trap) through the center of the core. A flux trap-type reactor is characterized by a thin fuel region adjacent to a good moderator which thermalizes the neutrons and causes the thermal neutron flux to peak in a region (center test hole) accessible for experiments. A relatively high neutron flux is provided for beamport experiments as well (Foyto et al., 2006).

The reactor core assembly is located eccentrically within a cylindrically shaped, aluminum-lined pool, approximately 10 ft (3.0 m) in diameter and 30 ft (9.1 m) deep. The reactor core consists of three major regions: fuel, control blade, and reflector. The fuel region has a fixed geometry consisting of eight fuel elements having identical physical dimensions placed vertically around an annulus in between two cylindrical reactor pressure vessels. Each fuel assembly is comprised of 24 circumferential plates containing uranium enriched to approximately 93% in the isotope uranium-235 (^{235}U) as the fuel material. The control blade region is an annular gap between the outer pressure vessel and the beryllium reflector, so that no penetration of the pressure vessels is required. Five control blades operate vertically within this gap, controlling reactor power by varying neutron reflection. The control blades are coupled to drive mechanisms by means of a support and guide extension. The drive mechanisms, mounted on the upper bridge over the reactor pool surface, consist of a motor and a reduction gear drive assembly. The reflector region consists of two concentric right circular annuluses surrounding the control blade region. The inner reflector annulus is a 2.71-in. (6.9-cm) thick solid ring of beryllium metal. The outer reflector annulus consists of vertical elements of graphite canned in aluminum, having

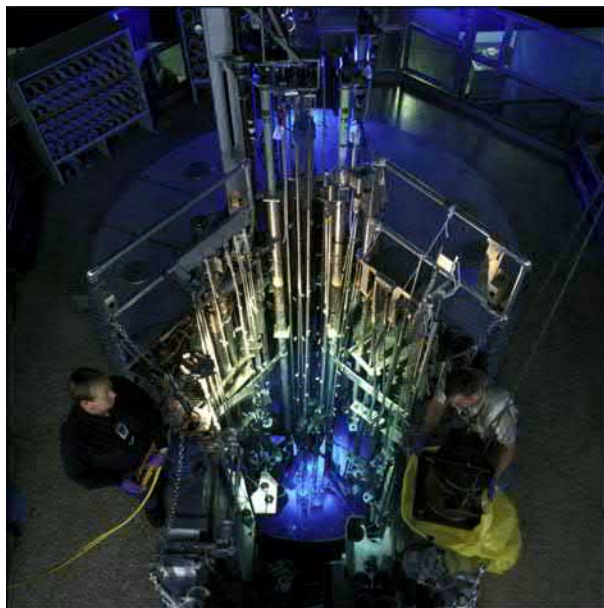


Fig. 1 University of Missouri Research Reactor.

a total thickness of 8.89 in. (22.6 cm). The graphite elements are designed to accept the source ends of four radial and two radial-tangential neutron beam tubes.

The fuel region is cooled by a pressurized primary coolant system which, at 10 MW, circulates a nominal 3750 gpm ($0.237 \text{ m}^3 \text{ s}^{-1}$) of light-water coolant through the reactor pressure vessels. The reflector region, the control blade region, and the center test hole are cooled by pool water which is drawn through these regions and circulated through the pool coolant system at a nominal flow rate of 1100 gpm ($0.069 \text{ m}^3 \text{ s}^{-1}$). The heat from the pool and primary coolant systems is transferred to a secondary coolant system by means of separate pool and primary heat exchangers. The heat is then dissipated to the atmosphere through a cooling tower.

The primary design features of the MURR provide the maximum neutron flux to the greatest number of users. The reactor offers the following experimental facilities:

- Center Test Hole (Flux Trap);
- Six Beamports;
- Pneumatic Tube System;
- Graphite Reflector Region Irradiation Positions;
- Bulk Pool Region Irradiation Positions; and
- Thermal Column.

Fuel design

The fuel material at time of initial startup was a uranium-aluminum alloy with each fuel assembly loaded to a maximum of 650 g of ^{235}U . This type of fuel system had performed very reliably in the Materials Test Reactor (MTR) and the Engineering Test Reactor (ETR) at the Idaho National Engineering Laboratory (INEL), as well as in other reactors throughout the world. However, in order to reduce the fuel cycle cost and the amount of ^{235}U needed per MWD of energy produced at the MURR, a conversion was performed in 1971 to switch to a uranium-aluminide (UAl_x) dispersion fuel material with a maximum loading of 775 g of ^{235}U per assembly.

The MURR 775-g fuel element is shown in **Fig. 2** and is a product of the UAl_x dispersion fuel system development. The UAl_x dispersion fuel system was developed at the INEL for the high flux, high power Advanced Test Reactor (ATR) and subsequently

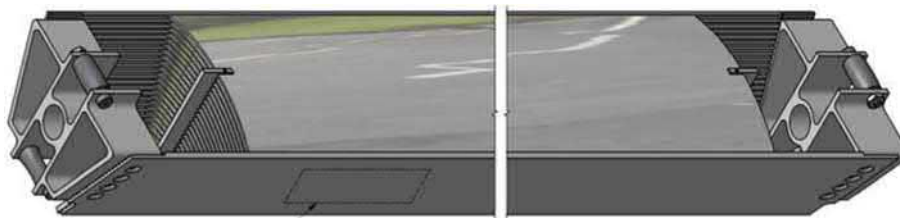


Fig. 2 MURR fuel element.

used at the MTR and ETR prior to its use at the MURR. Several features of the UAl_x dispersion fuel system increase the fuel performance capability in high flux reactors. One of these features is powder dispersion in the fuel matrix causing voidage which will accommodate fission products. The UAl_x structure has exceptional tolerance for fission gas retention and burnable poisons can be readily dispersed in the fuel matrix. Due to the compact core size, the fuel uses high enriched uranium (HEU) to obtain sufficient excess reactivity to be able to operate at 10 MW.

The fuel elements have a radial dimension of 3.21 in. (8.15 cm) as determined from radial length from the inside of the inner roller to the outside of the outer roller and an overall length of 32.5 in. (82.55 cm). Each element is longitudinally-symmetrical with 24 fuel bearing plates. The fuel plates are segments of concentric circles 0.050 in. (0.127 mm) thick separated by a gap of 0.080 in. (0.203 cm). These gaps provide the coolant channels necessary for the removal of the heat generated during the fission process, and for neutron moderation. Additional coolant gaps exist between the innermost fuel plate (No. 1) and the island tube wall (inner pressure vessel), and the outermost fuel plate (No. 24) and the outer pressure vessel wall. These gaps are 0.095 in. (0.241 cm) and 0.075 in. (0.191 cm), respectively. The nominal radius to the innermost plate is 2.77 in. (7.04 cm), and 5.76 in. (14.63 cm) to the outermost plate.

Each fuel plate is fabricated using the "picture frame" method which consists of the fuel compact (fused UAl_x and aluminum powders) encased in a one-piece window-shaped aluminum frame with an aluminum cladding plate welded on both sides, sandwiching the fuel compact. This sandwich is then heated and repeatedly rolled until the fuel plate is 0.050 in. (0.127 cm) thick. This process ensures that a metallurgical bond develops between the fuel material, aluminum powders, and cladding at all interfaces. The fuel meat in each plate is 0.020 in. (0.051 cm) thick. The fuel plates are supported along their vertical edge by slotted aluminum side plates. The fuel plates are permanently fastened into the side plates using a mechanical binding procedure that provides a tensile strength of greater than 150 pounds per linear inch (26.79 kg cm⁻¹) of the side plate joint. This ensures a rigid assembly fully capable of withstanding the hydraulic forces imposed by the primary coolant design velocity of 23 ft s⁻¹ (7.01 m s⁻¹). In fact, fuel assemblies of similar construction have withstood severe hydraulic tests at flow velocities up to 50 ft s⁻¹ (15.24 m sec⁻¹) without distortion, thus indicating an adequate design margin in this regard. The side plates are 31.75 in. (80.65 cm) long by 3.16 in. (8.03 cm) wide and 0.15 in. (0.381 cm) thick.

Tables 1 and 2 provide summaries of the description and specifications of the 775-g MURR fuel elements. A drawing of the MURR fuel element is shown in Fig. 2. Eight (8) such fuel elements comprise the fixed MURR core as shown in Fig. 3. This design

Table 1 Summary sheet for 775-g fuel elements.

Fuel Content (grams ²³⁵ U per element)	775
Type of Fuel	Aluminide-UAl _x mostly UAl ₃ Phase
Fuel Density (grams of ²³⁵ U loaded per cubic centimeter)	1.5–1.6
Boron Content (natural boron per element)	Trace Impurities
K _{eff} (clean core—control blades full out)	1.079
Control Blade Worth (Δk)	0.165
Peak Burnup Density (fissions per cubic centimeter)	< 2.3 × 10 ²¹
Energy per Element at Peak Fission Density (MWD per element)	≈ 180

Table 2 Summary of fuel element specifications.

<i>Description</i>	<i>Nominal Value</i>
<i>Fuel Material</i>	
Enrichment ²³⁵ U	93%
Thickness	(0.020 in.) (0.0508 cm)
<i>Cladding</i>	
Material	Aluminum
Thickness	(0.015 in.) (0.0381 cm)
<i>Fuel Assembly</i>	
Number of Fuel Plates	24
Innermost Fuel Plate Center Radius	(2.795 in.) (7.099 cm)
Outermost Fuel Plate Center Radius	(5.785 in.) (14.694 cm)
Overall Fuel Assembly Length	(32.5 in.) (82.550 cm)
Overall Fuel Plate Length	(25.5 in.) (64.770 cm)
Overall Active Fuel Length	(24.0 in.) (60.960 cm)
Fuel Plate Thickness	(0.050 in.) (0.127 cm)
Distance Between Fuel Plates	(0.080 in.) (0.203 cm)
Maximum ²³⁵ U Loading UAl _x Dispersion Fuel	775 g
Weight	≈ 6 Kg

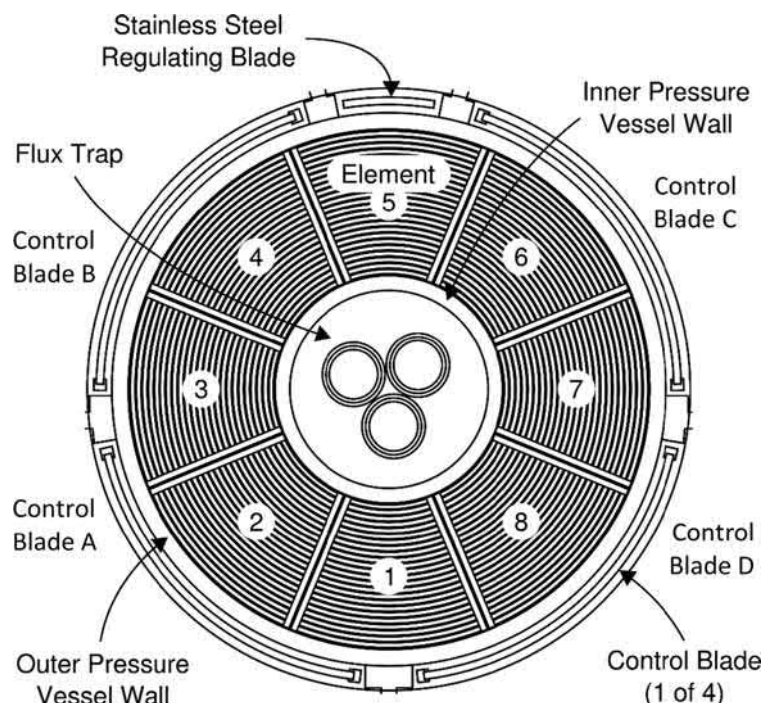


Fig. 3 MURR core layout.

has proven to be reliable with more than 48 years of continuous operation at the MURR, and in other research reactors with no significant failures.

Core loadings

To ensure that the validity of the safety limits is maintained, the reactor core will consist of eight fuel elements. However, operation at power levels up to 100 watts above shutdown power using a core loading of less than eight fuel elements is permitted for the purposes of reactor calibration or multiplication studies. Reactor core loadings are normally comprised of a mixture of eight fuel elements with varying stages of burnup (fissions cm^{-3}). Generally, this mixture consists of two relatively new fuel elements, four fuel elements that are approximately at the “halfway” point, and two fuel elements that are approaching their burnup limit. A mixed core allows the fuel elements to be used to a higher burnup while not exceeding their licensed burnup limit, as specified in the Technical Specifications. **Fig. 3** identifies the fuel element positions and control blade locations in the MURR core.

Fuel conversion

Because of its compact core design (33l), which requires a very high loading density of ^{235}U , MURR could not perform its mission with any previously qualified low-enriched uranium (LEU) fuels. However, in 2006 with the prospect of the National Nuclear Security Administration (NNSA) Global Threat Reduction Initiative Reactor Conversion Program validating the performance of U-10Mo monolithic LEU foil fuels, MURR began actively collaborating with what is currently the NNSA Material Management and Minimization (M3) Office’s Reactor Conversion Program, along with four other U.S. high-performance research and test reactors that use HEU fuel, to find a suitable LEU fuel replacement. It was concluded that the proposed LEU fuel assembly design, in conjunction with an increase in power level from 10 to 12 MW_{th} , will (1) maintain safety margins during steady-state and transient conditions, (2) allow operating fuel cycle lengths to be maintained for efficient and effective use of the facility, and (3) preserve an acceptable level and spectrum of key neutron fluxes to meet the scientific mission of the facility. Current projected conversion date is July 2028 (Foyto et al., 2017).

Control blades and control system

The reactivity and power of the reactor are controlled by five neutron-absorbing control blades, which are all located external to the pressure vessels. Each control blade is coupled to a control blade drive mechanism by means of a support and guide extension (offset mechanism). Four of the control blades, referred to as the shim blades, are used for coarse adjustments to the neutron density within the reactor core. The fifth control blade is a regulating blade. The low reactivity worth of this blade allows for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level.

The shim blades are constructed of formed BORAL[®] plate which is, by weight, nominally 50% boron carbide and 50% aluminum. The boron carbide-aluminum mixture is clad with 0.0375 in. (0.09525 cm) of aluminum-alloy 1100 for a nominal blade thickness of 0.175 in. (0.4445 cm). All four sides of the blades have a 0.25-in. (0.635 cm) aluminum frame where the blade thickness is 0.25 in. (0.635 cm). The active length of the neutron absorbing material is 34 in. (86.35 cm), and the overall blade length is approximately 40 in. (101.6 cm). The upper 6 in. (15.25 cm) of the shim blade is a 0.25-in. (0.635 cm) thick aluminum mounting plate that is curved to the shape of the blade. Each shim blade occupies approximately 72 degrees of a circular arc around the pressure vessel.

The regulating blade is constructed of stainless steel. The blade has an overall length of approximately 30 in. (76.2 cm) and occupies approximately 18 degrees of the circular arc.

The shim blades are positioned by four control blade drive mechanisms mounted on the upper bridge over the reactor pool surface. Each control blade drive mechanism consists of a 0.02-HP, 115-V, one-amp, single-phase, 60-cycle motor connected to a lead screw assembly through a reduction gear box and overload clutch. The lead screw assembly converts the rotating motion of the drive motor to the linear motion of the control blades. The overload clutch allows slippage of the lead screw should a shim blade become bound within the control gap. A ball nut coupled directly to a drive tube is driven inward and outward by the lead screw. Connected to the bottom end of the drive tube is an electromagnet which engages a cadmium-plated carbon steel anvil attached above the water level to the end of a lift-rod assembly. The lift-rod assembly allows the positioning of a shim blade through a support and guiding mechanism (offset mechanism) mounted on a pedestal attached to the reflector tank. Limit switches mounted on each control blade drive mechanism stop the drive motor at the top and bottom of travel. These limit switches also provide indication on the reactor control console that the drive mechanism is either in the full-in or full-out position.

The shim blade can be withdrawn when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized, releasing the anvil, and allowing the shim blade and lift-rod assembly to drop. Loss of electrical power to the control blade system results in a reactor scram and safe shut down of the reactor. A dash pot assembly cushions the fall of the shim blade during the final 20% of travel.

The regulating blade is positioned by a control blade drive mechanism similar to the ones used for the shim blades with a few exceptions. A scram signal does not insert the regulating blade; therefore, an electromagnet and an anvil are not required. The lift-rod assembly is pinned directly to the drive tube. Also, the regulating blade is driven at a faster speed than the shim blades, requiring a different reduction gear box.

The reactor is operated from the reactor control console in either of two control modes: manual or automatic. Manual control is used for reactor start-up, changes in power level, and steady-state operation for short periods of time. Automatic control is selected only after a minimum power level has been attained and is used for long term steady-state operation.

Control blade movements, interlocks and bypasses, and control modes are managed by the Rod Control System. The Rod Control System is a relay and switch logic system used to prohibit accidental or incorrect operation which could result in an unsafe condition. Rod position indication is determined and displayed in the reactor control room by a system of microcomputer controlled instruments. The system consists of a central control chassis and a remote display unit which provides the reactor operator with control rod height in inches.

The controlling parameter for automatic operation of the Rod Control System is the neutron flux monitored by a compensated ion chamber (CIC). The neutron flux is essentially proportional to the reactor power level. The pulse signal generated by the CIC is routed to the wide range neutron flux monitor located in the reactor control room. An output signal produced by the wide range signal processor drawer is then routed to a solid state servo amplifier. This output signal voltage is proportional to that of the input signal generated by the CIC.

The servo amplifier is mounted on the control room instrument panel. The servo amplifier automatically controls the power level of the reactor by comparing the output signal from the wide range signal processor drawer with the power schedule setting. The power schedule set point is adjusted by a spring return switch and displayed on a meter mounted on the reactor control console. The output from the signal processor drawer is a 0 to +10 VDC signal, corresponding to 0 to 125% reactor power level (+8.0 VDC = 100% power). The output from the power schedule potentiometer is a 0 to -10 VDC signal depending on the setting. This $\pm$ voltage window represents the $\pm$ control sensitivity of the servo amplifier. If a mismatch does exist, a positive or negative output signal is generated and sent to the relay-controlled drive motor of the regulating rod drive mechanism, which repositions the regulating blade, stepwise, in a direction which minimizes the discrepancy between the power schedule setting and the actual power level.

The Reactor Scram System consists of the electronic circuitry which can initiate the instantaneous drop of the reactor control blades (reactor scram) by interrupting power to their electromagnets should a monitored parameter (power level, temperature, flow rate, etc.) exceed a predetermined value. A reactor scram may also be initiated manually by depressing a push button on the reactor control console.

The protection channels and protective responses are sufficient to ensure that no safety limit, limiting safety system settings, or Reactor Scram System limiting condition for operation will be exceeded. The system design ensures that the design bases can be achieved, and that the system can be readily tested and maintained in the designed operating condition. Sufficient safety system isolation and independence is provided to avoid malfunctions or failures caused by other systems. The Reactor Scram System reasonably achieves safe reactor shutdown in the event of a single random malfunction within the system. System operation prevents or mitigates hazards to the reactor or the escape of radiation, so that the full range of normal operations poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment.

The Rod Run-In System is designed to initiate the automatic insertion of the shim blades at a controlled rate of 0.085 cm s^{-1} (2 in. per minute) should a monitored parameter exceed a predetermined value. This system is not part of the Reactor Scram System; however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to actuating a Reactor Scram System set point trip. The Rod Run-In System also inserts the control rod drive mechanisms in the event of a scram condition. A rod run-in may be initiated manually by depressing a push button on the reactor control console.

Neutron moderator and reflector

In addition to removing the heat generated from the fission process, coolants from the primary and pool systems also serve as the neutron moderator. Neutron reflection is provided by two concentric right circular annuli surrounding the reactor pressure vessels. The reflector materials have a rigid structure that retains size and shape, and supports all projected forces and weights. The inner reflector annulus is a 2.71-in. (6.88 cm) thick solid ring of beryllium metal which forms the outer wall of the control blade gap. Additional reflection is provided by an outer reflector annulus consisting of canned vertical elements of graphite having a total thickness of 8.89 in. (22.58 cm). Each graphite element is protected from water penetration by a leak-tight welded aluminum can. The graphite elements are designed to allow horizontal penetration of the beam tubes perpendicular to the beryllium reflector ring. The overall height of the graphite reflector region is 36 in. (91.44 cm), and it is centered vertically with respect to the fueled region. The height of the beryllium reflector is 37 in. (93.98 cm) and it extends 1 in. (2.54 cm) below the graphite reflector.

Reactor coolant systems

MURR utilizes three coolant systems: Primary, Pool, and Secondary. The primary and pool coolant systems are designed for three (3) modes of operation:

- (1) Mode I—At power levels of up to 10 MW with the primary coolant system pressurized and at a flow rate of approximately 3800 gpm ($0.240 \text{ m}^3 \text{ s}^{-1}$), and a pool coolant flow rate of approximately 1200 gpm ($0.0757 \text{ m}^3 \text{ s}^{-1}$); used when all heat exchange and pumping capacity is available;
- (2) Mode II—At power levels of up to 5 MW with the primary coolant system pressurized and at a flow rate of approximately 1900 gpm ($0.120 \text{ m}^3 \text{ s}^{-1}$), and a pool coolant flow rate of approximately 600 gpm ($0.0379 \text{ m}^3 \text{ s}^{-1}$); utilizing only half the design heat exchange and pumping capacity available; being the maximum operating power level at the time of initial startup; and
- (3) Mode III—At power levels of up to 50 kW with the primary coolant system open to the reactor pool, the reactor pressure vessel head removed, the flanged port open, and the pool water level at the elevation of either the upper or lower reactor bridge; used for core flux calibrations following the loading of a new core, or after a fuel rearrangement.

Heat from the primary and pool coolant systems is transferred to the secondary coolant system by means of heat exchangers. This heat is then dissipated to the atmosphere through a cooling tower. Auxiliary systems which support the three principal reactor coolant systems are also discussed in this chapter.

Experimental facilities and utilization

Introduction

The experiment program at MURR provides a broad range of analytical, radiographic, and irradiation services for use by the research community and the commercial sector. Experiments conducted at the MURR are subdivided into two general classifications: (1) neutron beam, and (2) neutron irradiation and isotope production. The neutron beam experiments are those research projects which utilize one of the beamports. The neutron irradiation and isotope production experimental facilities include the center test hole, the graphite reflector region, the pneumatic tube system, and in-pool locations external to the graphite reflector (bulk pool). Some of the irradiation services provided by these experimental facilities include isotope production for the development of radiopharmaceuticals, neutron activation analysis (e.g., archeological samples), and transmutation doping of silicon. The beamports are primarily used for neutron scattering useful in determining the structure of solids and liquids.

The principal purpose of the MURR is to provide neutrons to the experimental facilities, and the design effort has been directed toward accomplishing this end. In order to accomplish the goals of the experiment program, the design of the reactor and the experimental facilities must be rather unique, but must also emphasize safety as a paramount concern. This latter requirement further dictates design criteria. A major safety feature of the reactor can be found in the design of the beryllium reflector, which effectively decouples the reactor core from experiment variations in the beamports, bulk pool and graphite reflector irradiation positions, the pneumatic tube system, and the thermal column. The only experimental facility not neutronically decoupled from the reactor in this manner is the center test hole (flux trap). However, this experimental facility is subject to a high degree of administrative control as discussed in the following section.

Center test hole

The Center Test Hole (Flux Trap) is that portion of the reactor through the center of the core which is bounded by a 4.5-in. (11.43-cm) inside diameter inner pressure vessel (island tube) and which extends 15 in. (38.1 cm) above and below the core centerline. A specially-designed test hole canister is inserted into this region of peak thermal flux ($6 \times 10^{14} \text{ n cm}^{-2} \text{ s}^{-1}$) for the purpose of material irradiations.

Beamports

The reactor is designed to accommodate six (6) horizontal beamports. These hollow tubes channel neutrons and gamma radiation from the reactor core with minimal scattering and attenuation between the beryllium reflector and the experiment equipment. Each beamport designation, size, elevation, and description is shown in [Table 1](#).

[Fig. 4](#) provides the current beamport layout and associated instruments.

Each beamport assembly consists of three major components: a fixed beamport liner, a removable beamport liner (beam tube), and typically, a removable collimator liner. The fixed beamport liner provided the necessary form work during the pouring of the biological shield, therefore it is the only component of the beamport assembly which is in direct contact with the magnetite concrete of the biological shield. The fixed liner is integrally welded to the reactor pool liner and serves as an extension of the reactor pool into the beamport vestibule. The fixed beamport liner allows penetration of the removable liner through the biological shield from the recessed vestibule inward, penetrating the graphite reflector region and terminating adjacent to the beryllium reflector. The portion of the beam tube within the graphite reflector region is cooled by pool water flowing downward through the gaps between the graphite reflector elements and around the beam tube. When the removable liner is fully inserted, a gap exists between it and the fixed liner. A $\frac{1}{2}$ -inch line, which penetrates the fixed beamport liner, returns water from the pool skimmer system into this area and then back into the reactor pool volume. This flow path prevents the stagnation of pool water in the beamport, helping minimize corrosion to the fixed and removable liners. The removable liner is attached to the fixed liner by means of a bolt ring located in the recessed vestibule. A packing gland assembly seals the removable liner to the fixed liner. Typically, a removable collimator liner is located within the removable beamport liner. The removable beamport liner provides a path for the neutron beam to travel prior to entering the removable collimator liner, where the beam is further shaped and defined by the collimator for the experiment equipment. The collimators are designed to accept different neutron filters. Crystal-sapphire and silicon are the predominant choices for filters utilized in the MURR neutron scattering experiment program. The removable collimator liner is attached and sealed to the removable beamport liner similarly to the way the removable liner is attached and sealed to the fixed liner. This allows filling the removable beamport liner with helium if a neutron beam is to be accessed, or demineralized water if not.

Pneumatic tube system

The Pneumatic Tube (P-Tube) System is designed to quickly transfer individual samples into and out of the graphite reflector region of the reactor core assembly. The samples are placed in small high density polyethylene sample carriers, or "rabbits," and

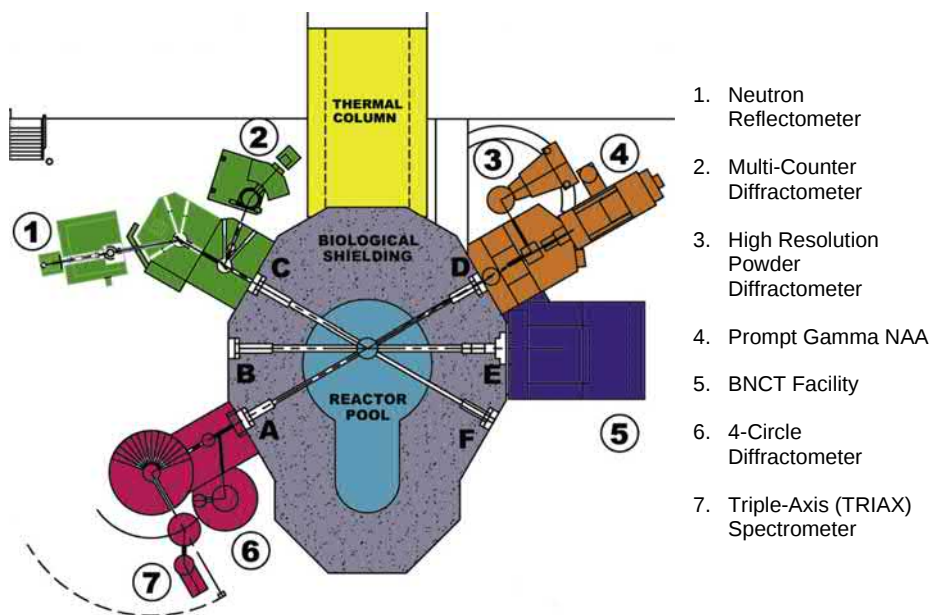


Fig. 4 MURR beamport layout.

transported at velocities of 30 to 45 ft per second ($9.1\text{--}13.7\text{ m s}^{-1}$) from designated laboratories into a region of relatively high thermal flux (approximately $1 \times 10^{14}\text{ n cm}^{-2}\text{ s}^{-1}$) for the purpose of material irradiation.

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MIT Research Reactor (MITR-II)

Lin-wen Hu, Nuclear Reactor Laboratory, Massachusetts Institute of Technology, Cambridge, MA, United States

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Introduction

The Massachusetts Institute of Technology Research Reactor (MITR) is located on MIT campus in Cambridge, Massachusetts. The first reactor, MITR-I, started construction in June 1956 and achieved criticality in July 1958. The MITR-I was a heavy-water cooled and moderated tank-type reactor using MTR-type fuel elements. It was one of the first university research reactors built on university campus to accomplish missions in education, training, nuclear science and technology, and fundamental and applied research such as metallurgy, radiochemistry, geology, and medicine. Following a year of low power tests after initial criticality, the reactor began a routine three-shift schedule at a power level of 1 MW in July 1959 with the exception of weekend shutdowns for maintenance and for refueling and experiments. In July 1961, the power was raised to 2 MW, and in November 1965 to 5 MW, the power level for which the MITR-I was designed ([MITR Staff, 1968](#)).

The MITR-I was shut down in 1974 for dismantling, construction and pre-operational testing until the second reactor, MITR-II, went critical in August 1975. The MITR-II is a light-water cooled and moderated, heavy water reflected tank-type reactor, located in the same shielding and containment structure as the MITR-I. It was designed to operate at 5 MW with a new finned aluminum clad MTR-type fuel element design in a compact hexagonal core housing to increase neutron flux, improve fuel economy and reduce personnel dose during refueling ([Allen, 1976](#)).

In July 1999, a formal application was submitted to The U.S. Nuclear Regulatory Commission (NRC) to relicense the reactor for an additional 20 years and to upgrade the power level from 5 to 6 MW with the existing fuel design, reactor operation conditions, and containment structures. The NRC issued a license renewal along with approval for 6 MW operation in November 2010 ([MITR Staff, 2011](#)). In order to accomplish the transition to higher operating power, major improvements were made since the late 1990s to reactor cooling systems including primary coolant pumps, heat exchangers and cooling towers as well as modern instrumentation.

The MITR has contributed to many research projects through the years, such as production of medical isotopes, closed-loop digital control of spacecraft and terrestrial reactors; boron neutron capture therapy for the treatment of cancer; material studies for advanced reactors; and neutron activation analysis for the study of environmental contaminants.

The MITR currently operates at its licensed power continuously up to 10 weeks per cycle and a nominal 3-week outage for refueling and maintenance. It has one of the strongest in-pile irradiation programs on a university campus, supporting experimental studies in the areas of advanced nuclear fuel, materials and instrumentation which are necessary for both existing and advanced power reactors. With renewed interest in developing advanced nuclear power systems, many using novel materials and advanced forms of fuels, facilities such as the MITR are needed to test material and fuel behavior in a variety of radiation environments. The MITR is one of the best suited research reactors for carrying out such studies because of its relatively high-power density (similar to a light water reactor), the capability to control chemistry and thermal conditions to reflect prototypic conditions, its easy-access geometric configuration, and being located on a university campus that offers easy access to researchers and students.

The MITR is integrated into national programs through various Department of Energy's initiatives, such as the Nuclear Science User Facilities (NSUF) ([NSUF, n.d.](#)), and through collaboration with national laboratories, industry, and other universities. Research and service areas currently supported by the facility include:

- Experimental infrastructure to support the US initiatives to design and build the next generation of nuclear reactors, such as Molten Salt Reactors and High-Temperature Gas Reactors;
- Advanced nuclear materials and fuel research,
- Trace element analysis, isotope production, and irradiation services,

- Neutron transmutation doping of silicon,
- Neutron scattering, and
- Design, licensing, and planning for conversion from Highly Enriched Uranium (HEU) fuel to a high-density Low Enriched Uranium (LEU) fuel.

MITR reactor design

The MITR operates at 6 MW with light water primary coolant temperature around 50 °C at atmospheric pressure, and nominal flow rate of 125 kg/s (2000 gpm). Secondary coolant transfers the fission power from the primary cooling system, heavy water and graphite reflector, and auxiliary cooling systems and rejects the heat to atmosphere through forced draft cooling tower.

a. Fuel Design

The MITR utilizes flat, plate-type, finned, aluminum-clad fuel elements with highly enriched uranium (HEU). Each fuel element consists of 15 identical fuel plates. The fuel plates are assembled between two grooved side plates. The fuel plates are 0.20 cm thick with longitudinally-milled fins on both sides to increase the heat transfer surface area. The fuel plates are 6.48 cm wide, 58.42 cm long and are spaced 0.40 cm apart to form 14 cooling water channels 0.20 cm wide fin-to-fin. The fins are 0.025 cm high and 0.025 cm wide separated by 0.025 cm wide grooves. Within an element, the spacing between the fuel plates is maintained by the grooves in the side plates. Between elements, end nozzles which are designed to provide structural rigidity for the element maintain the necessary spacing. The fuel elements are rhombic in shape with a perpendicular distance between the flats of 6.03 cm. The overall length, including the end nozzles, is 66.68 cm. Each element has same nozzles at both ends, thereby allowing it to fit into the lower grid plate and to mate with the upper hold-down grid. The fuel element design is both radially and axially symmetric so that they can be both rotated and inverted in order to equalize the effect of flux peaks on burnup.

Each fuel element contains about 510 g of U-235. The amount of U-235 is evenly distributed among each fuel plate. Each plate contains fuel meat which is approximately 93% enriched uranium in the form of a UAl_x cermet. The fuel meat is 55.88 cm to 57.79 cm long, 5.29 cm wide, and 0.08 cm thick and clad with 6061 aluminum alloy. The nominal thickness of the cladding is 0.038 cm, not including the fin height. The UAl_x powder sintering process results in a final product that is at less than theoretical density, specifically the void fraction is between 4% and 7%. This provides space for fission product gases and hence prevents fuel plate swelling. The MITR fuel elements do not contain any burnable poisons or moderating materials.

b. MITR Core Design

The reactor core consists of 27 positions, up to 24 positions are filled with fuel elements. The remaining three positions are filled with either a solid aluminum “dummy” element or an in-core experiment. The MITR uses six coarse-control shim blades and one fine-control regulating rod located at the core peripheral to adjust reactivity during reactor operation. The core map and location of these control elements is shown in Fig. 1 below.

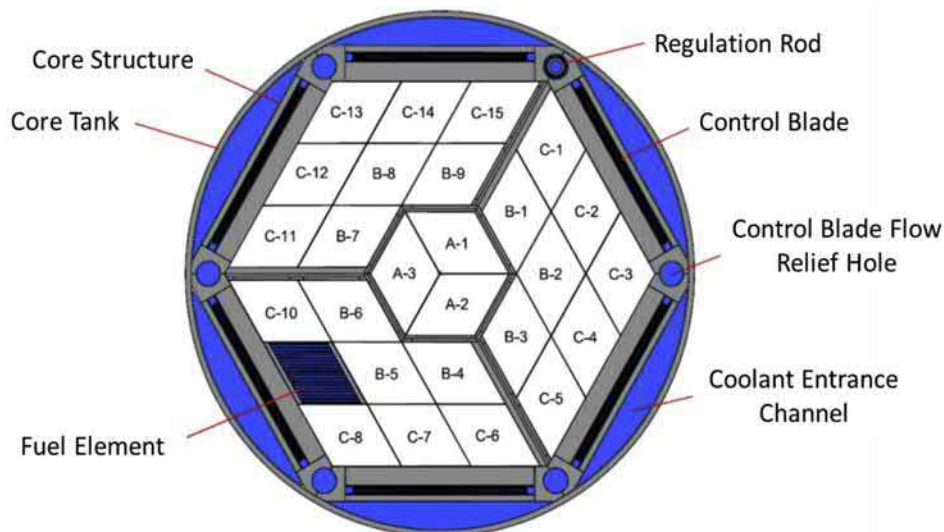


Fig. 1 Schematic of the MITR core showing 27 fuel positions, shim blades, and regulating rod.

c. Reactivity Control and Safety Systems

There are six identical shim blades arranged symmetrically with respect to the core. These blades are contained within the light-water core tank and occupy the space between the outermost ring of fuel, the C-Ring, and the core-housing wall. The full range of travel is such that the tip of the blade is never withdrawn from its guide slot. The reactor shutdown margin requirement is such that it is possible to shut the reactor down from its most reactive state with the most reactive blade (and also the regulating rod) in the full-out position. In addition, the reactor can operate with only five blades being functional provided that the remaining one is at or above the average height of the functional ones. Thus, only five are needed to shut down and to operate the reactor. Each blade can be operated independently of the others. Another reactor protection is further provided by the capability to dump the heavy water reflector and thereby shut down the reactor.

The shim blades, which measure 77.8 cm by 17.78 cm by 0.76 cm, are made from 1.1% natural boron impregnated stainless steel. Each blade consists of two 0.32 cm-thick plates separated by spacers so that a 0.13 cm gap exists in the lower 59.36 cm of the blade. This gap, which allows water to circulate between the plates, provides cooling and helps to prevent warping because of radiation damage. An aluminum connecting arm is attached to the guide rod of each shim blade assembly. Affixed to this connecting arm is an armature that is plated with corrosion-resistant material such as nickel or chrome. The shim blade assemblies move up or down at a constant speed of 0.18 cm s^{-1} . The maximum travel of the shim blades is 53.34 cm. When the current to the electromagnet on the drive mechanism is cut off, the armature is released thereby dropping the shim blade assembly by gravity.

There is one regulating rod for fine reactivity control and is connected to automatic controller. It is located exterior to the core in one of the small circular holes at the corner of the core support housing. The regulating rod consists of an extension and an absorber section. The absorber section is 63.18 cm long with a 2.22 cm outer diameter. The bottom 45.7 cm of the absorber section consists of an inner aluminum rod, a cadmium wrap, and an outer aluminum clad. The mechanism for moving the regulating rod in or out is similar to the shim blade mechanisms. The travel speed of the regulating rod is 0.18 cm s^{-1} . The regulating rod mechanism differs from the shim blade mechanism in that there is no electromagnet assembly. Hence, the regulating rod does not drop into the core upon automatic scram. Rather, a scram signal causes it to be driven inwards at its normal speed.

d. Reactor Tanks

The reactor core, the light-water coolant and moderator, the heavy water reflector and its helium cover gas, and the control devices are all contained within two concentric tanks. These are the light-water core tank and the heavy-water reflector tank. All penetrations through the light-water tank are above the core. All penetrations that are capable of creating a siphon that could drain the tank are equipped with a siphon-break to prevent that occurrence. Both tanks are designed against failure that could cause loss of integrity, and both would have to fail in order for the core to be uncovered. A pipe break or a single tank failure could not result in such a scenario.

The light-water core tank was manufactured and installed in 1974. The MITR's core shroud and flow guide, which are attached to the core tank and the core support housing respectively, are also described here. The core tank contains more than 2700 l of light water coolant which serves as the heat sink for fuel decay cooling. There are four natural convection valves located at the bottom of the core tank outside the core shroud. These valves and two anti-siphon valves open upon loss of primary coolant flow and allow natural circulation flow in the core tank to remove decay power from the core. Fig. 2 are pictures of the hexagonal compact core with 24 fuel elements installed, and top view of the light-water core tank. Fig. 3 is a cross-sectional view of the MITR showing the core tank, heavy water reflector tank, and graphite reflector.

e. Reactor Coolant Systems

The MITR's primary coolant is light water. The purpose of the primary coolant system is to remove the heat generated in the core by fission, to moderate neutrons, to provide shielding for the reactor top, to serve as a reservoir of coolant for possible emergencies, and to act as a barrier that would prevent the release of fission products in the event of multiple failures. The system, which consists of pumps and heat exchangers as well as valves and piping, is located within the reactor containment building. A storage tank allows for the expansion and contraction of the coolant that results from temperature variations. An air purge that is drawn through the space between the surface of the coolant and the underside of the core tank's lid precludes the buildup of any radiolytic gases associated with reactor operation. The air purge system automatically isolates if abnormal radiation levels are detected by radiation monitors. Hence, the primary coolant system operates at closed to the atmospheric pressure, and it serves as one of three barriers to fission product release. The MITR is normally operated under forced convection. However, the natural convection mode of operation is also possible at low power levels ($< 100 \text{ kW}$).

Beyond the core region, gamma heating is also deposited in the heavy water and the graphite reflectors. This energy is removed by the reflector coolant and shield coolant systems, respectively. Each system has its own dedicated pump, heat exchanger, and cleanup system. The secondary coolant system also uses light water. It transfers heat from the primary, reflector, and shield coolant systems, and it discharges the heat to the atmosphere via the cooling towers.

The MITR has operated at its licensed power for at least 90 h per week during the 1970s through 1990s. In 1990s through late 2000s, it has operated 24 h per day, 7 days per week with a shutdown of a few days per month for maintenance, test and calibration, refueling, etc. Since 2010s after receiving 6 MW operating license, the MITR operates with quarterly cycles, continuously up to 10 weeks per cycle and a nominal three-week outage for maintenance and refueling, in order to support in-pile materials and instrumentation tests. The power history of the reactor maintains a strong photo-neutron source generated from decay gammas reaction

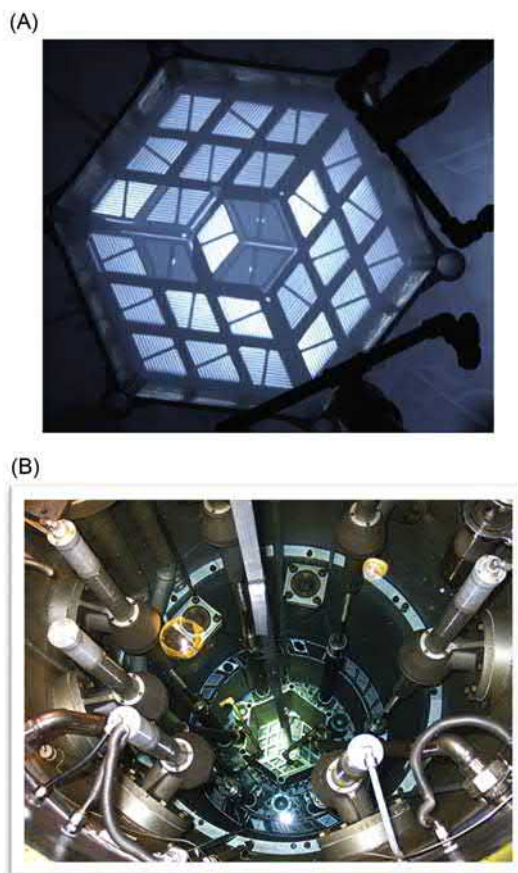


Fig. 2 (A) MITR core with Cherenkov radiation (B) Top view of the MITR core tank.

with the deuterium in the heavy water reflector tank. Accordingly, there is no need for an installed neutron source with the exception of the initial fresh fuel startups as was performed in 1958 and 1975 and may be required for initial startup of the LEU fuel when conversion occurs in the future (Hu et al., 2020).

Experimental facilities

The MIT Nuclear Reactor Laboratory (NRL), an interdepartmental laboratory which operates the MITR, has an extensive array of facilities for performing neutron and gamma irradiations, studying neutron physics, and instruments for examining radioactive materials. A comprehensive description of the irradiation facilities which contains pneumatic tubes, beam ports, in-pile irradiation facilities, associated neutron activation analysis laboratory, hot cells and post-irradiation examination instruments can be found online ([MIT Nuclear Reactor Laboratory, n.d.](#)). A summary of the MITR's experimental facilities is given below.

In-Core irradiation facilities

The MITR has the capability to perform a wide range of experiments in the reactor core which has neutron flux and energy spectrum similar to a light water power reactor. In-core experimental facilities are not permanently installed in the reactor. Facilities that can be reused are stored in hot cells or shielded locations in the reactors containment building. A brief list of the types of facilities that are available includes:

- An inert gas-filled irradiation tube (ICSA) for sample capsule irradiation at $<900^{\circ}\text{C}$ (instrumented/lead-out or un-instrumented),
- Forced-circulation coolant loops that replicate conditions in both pressurized and boiling water reactors (up to 300°C and 10 MPa),
- High temperature ($>900^{\circ}\text{C}$) irradiation facility for materials irradiations in inert gas (He/Ne mixture),
- Custom, dedicated facilities for irradiations in unique conditions (e.g., molten fluoride salts).

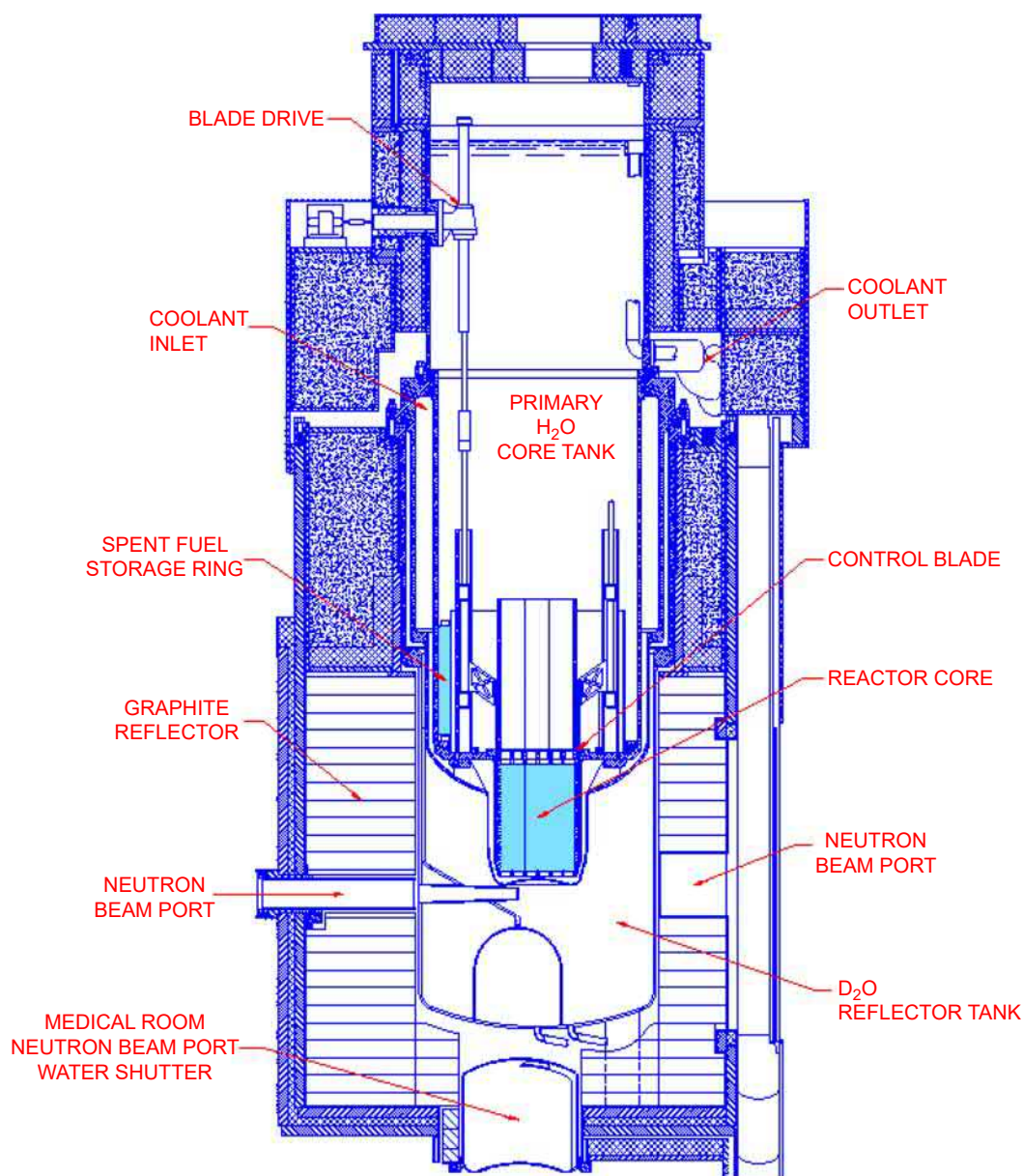


Fig. 3 Cross-sectional view of the MITR.

Beam ports and pneumatic tubes

The MITR has 11 neutron beam ports surrounding the reactor. The biological shield ranges from 7.62 to 30.18 cm (3" to 12") in diameter. There are also two tangential through tubes that pass near the reactor heavy water reflector. Two systems are available to perform neutron irradiations on small samples by pneumatically inserting them into positions near the MITR core, a 5.08 cm (2") high flux pneumatic tube, and a 2.54 cm (1") intermediate flux pneumatic tube.

Neutron scattering

The MITR has two neutron scattering facilities: a Triple-axis diffractometer equipped with Pyrolytic Graphite (PG) monochromator, PG analyzer, He-3 monitors, and detector and standard collimators, slits, and PG filters, and a neutron optics test station with polychromatic neutron beam.

Time-of-flight experimental facility

This facility can be operated locally or remotely over the internet for student experiments in neutron energy distributions, attenuation, and scattering. Also available for neutron detector testing or cross-section measurements.

Post-irradiation examination (PIE) facilities

The MITR has a wide range of facilities that are useful for handling and examination of highly radioactive materials. These include:

- Hot cells and for remote handling of irradiated components and materials,
- Hot sample preparation facilities,
- Helium-filled glove box with high-temperature furnace and gamma shielding
- Electron microscopy facilities,
- Access to electron microscopy facilities can be arranged for irradiated samples.

Vital nuclear engineering research

The in-pile irradiation experiments at the MITR is an integral part of the Department of Energy Office of Nuclear Energy's effort to improve current generation nuclear reactors and develop new reactor technologies with improved economics and safety. It is the only university research reactor engaged in testing significant irradiation damage levels in environments representative of commercial power reactors, and is an excellent complement to national laboratory test reactors. Current irradiation experiments cover four major areas:

a. Accident tolerant fuel development

Developing accident tolerant fuel became a focal point for the nuclear industry following the 2011 earthquake and tsunami that led to the Fukushima nuclear accident in Japan. This fuel would be able to withstand "beyond design basis" accidents without releasing radioactivity into the environment. At most commercial reactors, nuclear fuel cladding is currently made from a metallic alloy of zirconium called Zircaloy™. Ceramic composite material, however, is known for having better high temperature strength and corrosion resistance compared to Zircaloy™ and is being investigated as an alternate cladding. The MITR carried out one of the first irradiation tests in high temperature, high pressure water of a ceramic composite material based on silicon carbide as an alternative to Zircaloy™. These early results demonstrated that testing materials in an irradiation environment is crucial to understanding their behavior and the program has played a major role in developing a viable ceramic composite cladding.

b. Advanced instrumentation testing

The extreme environment of a nuclear power reactor, or recreated environment in a research or test reactor, proves challenging for instrumentation. Instrumentation, such as sensors of various types, however, are vital to fully exploit expensive irradiation experiments in a reactor. They are used to characterize the test environment (temperature, pressure, radiation intensity and various chemical parameters) and to measure the response of the test materials in real time. The alternative to real time measurements is the periodic removal of samples for testing, which is expensive, time-consuming, may affect the parameters being measured and gives no information about the changes that occur between removals. Sensors capable of withstanding extreme nuclear environments also have applications in operating power reactors to monitor normal and accident conditions. The MITR has an ongoing program of in-core testing of advanced ultrasonic and fiber-optic sensors, as well as more traditional thermocouples optimized for nuclear use at high temperatures. In-core irradiation tests have also included detectors for neutron and gamma irradiation, measurements of electrochemical corrosion potential (an important parameter for understanding the corrosion and cracking of metals in water or molten salt coolants). A specialized sensor for measuring the crack growth in a stressed sample in-core was adapted from Halden reactor and successfully demonstrated in the MITR.

c. Molten Salt Reactor (MSR) research

A promising MSR design, Fluoride salt-cooled High-temperature Reactor (FHR) utilizes coated particle fuel, was shown to be safe and economical. In addition to generating electricity, it would also be able to operate at temperatures high enough to provide process heat for industrial applications. The design is based on combining several technologies: the fuel from high temperature gas reactors, modern gas turbines for power production, and a molten lithium fluoride/beryllium fluoride salt (flibe) as the coolant. Many aspects of the technology have been previously tested, but many technical challenges remain to demonstrate a complete reactor. Understanding the behavior of the molten flibe salt and its interaction with the other reactor components under irradiation is vital. Four irradiations of flibe salt at 650–700 °C have been carried out at the MITR since 2013 to advance the technologies needed for MSR development ([Dolan et al., in press](#)). These were the first irradiations of flibe since the MSR Experiment at Oak Ridge National Laboratory in the 1960s.

d. Nuclear fuel testing

Advances in nuclear fuel design to improve safety or efficiency require extensive testing to reach the point where they can be demonstrated and more widely adopted in operating power reactors. In the U.S., in-core testing of actual fuel materials was once only possible in test reactors at national labs. The MITR was licensed in 2003 to test actual fuel materials in its core, allowing new fuel concepts to be tested on a small scale and at significantly reduced cost and shorter turnaround compared to larger scale experiments at larger test reactors. Two types of fuel samples have been tested. The first tested the impact of providing an additional coolant channel on the inside of a nuclear fuel rod to reduce the temperature of the fuel and improve safety. The second tested the response of a different form of uranium fuel, which is a better heat conductor than the fuel typically used in power reactors. The improved conductivity reduces the temperature of the fuel and improves safety. It may also allow the fuel to operate longer in the reactor to more fully utilize the uranium it contains.

LEU fuel conversion program

The U.S. Department of Energy (DOE) National Nuclear Security Administration (NNSA) Office of Material Minimization and Management (M3) Reactor Conversion Program has been working to convert research reactors globally from highly enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel. Several U.S. high performance research reactors (USHPRRs), including the MITR, could not be converted using a previously qualified LEU fuel. A new type of LEU fuel based on a uranium and molybdenum (U-10Mo) alloy is being developed to allow the conversion of five USHPRRs. The conversion program activities include the qualification of the proposed fuel, U-10Mo alloy foil clad in an aluminum alloy, and the development of the fabrication techniques (Wilson et al., 2019). The primary changes due to the MITR LEU conversion are the fuel element design from 15-plate finned HEU fuel to 19-plate un-finned LEU fuel, reactor operating power increases from 6.0 to 7.0 MW, and a higher primary coolant flow rate from 125 to 150 kg/s (2000–2400 gpm). The new LEU “FYT” fuel element design consists of three types of fuel plates with different fuel meat thicknesses while maintaining the same fuel element outer geometry. MITR conversion study concluded that the proposed LEU fuel design with power increase will maintain sufficient safety margins during steady-state and transient conditions, fuel cycle lengths, and reactor performance to support its experimental program (Sun et al., 2018).

Fig. 4 is a diagram of the new MITR LEU fuel element design. It utilizes flat, plate-type, aluminum-clad fuel elements that are U-235 enriched to 19.75 wt%. Each fuel element consists of 19 fuel plates. The fuel plates are assembled between two grooved side plates which are 0.48 cm thick. The fuel plates are 0.12 cm thick, 5.86 cm wide, 58.42 cm long and are spaced 0.19 cm apart to form 18 cooling water passages. Within an element, the spacing between the fuel plates is maintained by the grooves in the side plates. End nozzles, which are designed to provide structural rigidity for the element, also maintain the necessary spacing between elements.

Each LEU fuel element will contain 968 g of U-235. The amount of U-235 is not evenly distributed among each fuel plate, where three different fuel core thicknesses are adopted. T-plates (Plate No. 1 and 19, yellow-color plates) have a fuel core thickness of 0.033 cm. Four Y-plates (Plate No. 2, 3, 17, and 18, orange-color plates) have 0.043 cm. Thirteen F-plates (Plates No. 4–16, red-color plates in Fig. 4B) have a fuel thickness of 0.064 cm. The fuel core is LEU enriched to 19.75 wt% U-235 in the form of a U-10Mo monolithic alloy. Like the HEU fuel, each plate has fuel 58.42 cm long and 5.29 cm wide. Plates remain clad with 6061 aluminum alloy (Al-6061) with 0.0025 cm zirconium interlayers between the fuel and cladding. The nominal thickness of the Al-6061 cladding with zirconium for T-plates, Y-plates, and F-plates are 0.043, 0.038, and 0.028 cm, respectively, so that the total thickness of each plate is identical (0.124 cm). Fig. 4 shows the LEU fuel element design. The comparison between the MITR HEU and LEU fuel is shown in Table 1. Table 2 is a summary of major operational parameters of HEU and LEU core.

The MITR LEU core startup plan is developed to follow the pre-operational testing, criticality studies, start-up experiments and stepwise power increase tests outlined in the MITR-II Startup Report after its major modifications (MITR Staff, 1977). Because of the

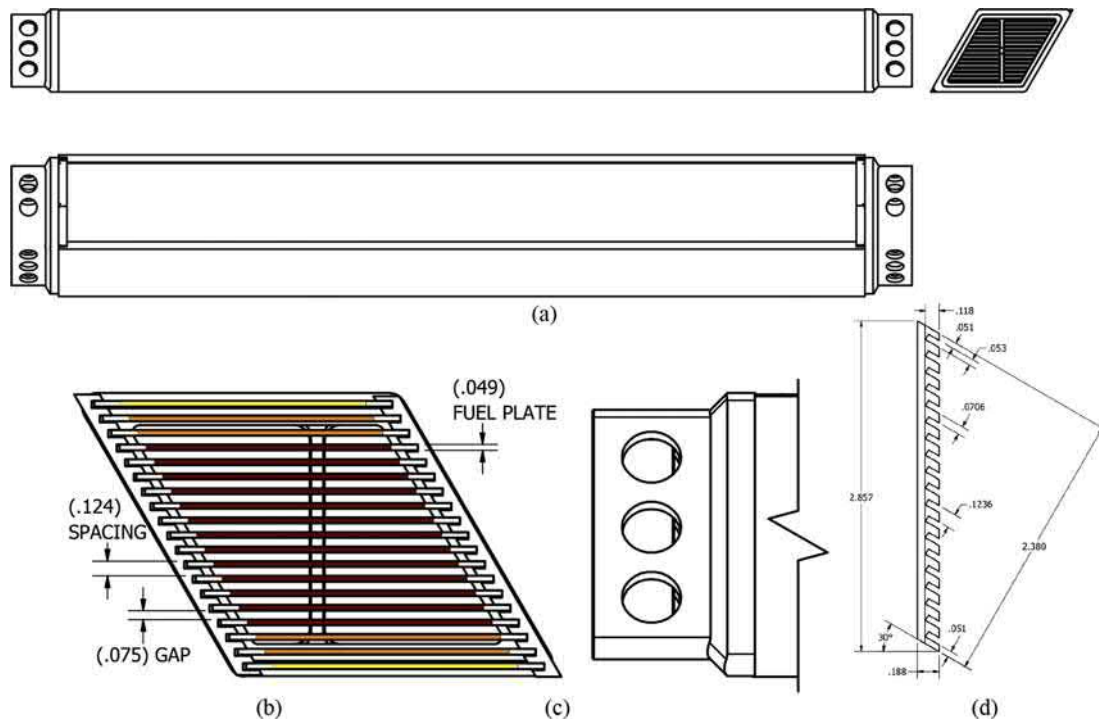


Fig. 4 Diagram of an MITR LEU fuel element (A) full element assembly, (B) assembly cross-section, (C) assembly nozzle, and (D) side-plate (dimensions given in inches).

Table 1 Major design parameters of the MITR HEU and LEU Fuel.

No.	Parameters	HEU fuel	LEU fuel
1.	Fuel Form	UAl _x Cermet	Monolithic U-10Mo Alloy
		1.59 g-U/cc	15.3 g-U/cc
2.	Fuel Thickness	0.076 cm	0.033 cm for T-plates (2) 0.043 cm for Y-plates (4) 0.064 cm for F-plates (13)
3.	Interlayer	N/A	0.0025 cm Zirconium between fuel and cladding, bonded by co-rolling
4.	Cladding	Aluminum w/ Fins Nominal thickness 0.038 cm at base of groove between fins	Aluminum w/o Fins Nominal thickness: 0.028 cm (F-plates) 0.038 cm (Y-plates) 0.045 cm (T-plates)
5.	Fin Depth	0.025 cm	N/A
6.	U-235 Loading	510.0 g per Element 34.0 g per Plate	967.7 g per Element 57.74 g (F-plates) 39.26 g (Y-plates) 30.02 g (T-plates)
7.	Void Volume	4% to 11%	N/A
8.	Limiting Fission Density	2.3×10^{21} fissions/cc	2.0×10^{21} fissions/cc (F-plates) 2.6×10^{21} fissions/cc (Y-plates) 3.0×10^{21} fissions/cc (T-plates)

Table 2 Major reactor parameters of the MITR HEU and LEU cores.

No.	Parameters	HEU Core	LEU Core
1.	Steady-State Operating Power	6.0 MW	7.0 MW
2.	Primary Coolant Flow	125 kg/s	150 kg/s
3.	Coolant Outlet Temperature	55 °C	55 °C
4.	Limiting Safety System Settings		
	Power	7.4 MW (Max.)	8.68 MW (Max.)
	Flow	112.5 kg/s (Min.)	137.5 kg/s (Min.)
	Temperature	60 °C (Max.)	60 °C (Max.)
	Coolant Level	10 cm (Min.) below overflow, or 305 cm above top of the fuel plates	10 cm (Min.) below overflow, or 305 cm above top of the fuel plates
5.	Coolant and Moderator	Light Water	Unchanged
6.	Reflector	Heavy Water/Graphite	Unchanged
7.	Reactor Type	Tank	Unchanged
8.	Fuel Type	U-Al _x Cermet	U-10Mo Monolithic Alloy
9.	Fuel Geometry	Finned Plate, Rhomboidal Cross-section	Unfined Plate, Rhomboidal Cross-section

existing systems and extensive operational experiences acquired during several decades of operations, only confirmation tests will be performed for auxiliary systems that will not be affected by LEU fuel conversion, such as heavy water reflector and shield cooling system (Hu et al., 2020).

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North Carolina State University PULSTAR Reactor

Ayman I. Hawari, Nuclear Reactor Program, Department of Nuclear Engineering, North Carolina State University, Raleigh, NC, United States

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Introduction

The history of nuclear research reactors on the campus of North Carolina State University (NCSU) is extensive. In 1949, Dr. Clifford Beck (Professor and Head of the Physics Department) developed a proposal to establish the first university based educational nuclear reactor in the world (Beck, 1949). Partial funding for this reactor was obtained from the North Carolina based Burlington Mills Foundation and ground was broken on the current "Burlington" building in 1951. The total cost of establishing this reactor, and the associated laboratories, was \$630,000 US Dollars. On September 5, 1953 at 12.59 a.m., the Raleigh Research Reactor went critical and operated under the R-1 license as issued by the US Atomic Energy Commission (AEC). The R-1 reactor had a maximum power of 10 kWth. It operated with a homogenous core of uranyl sulfate that contains 836 g of uranium 235 (U-235) in the form of 1.39 kg of UO_2SO_4 (with an enrichment of 93.7% in U-235) diluted in 12.5 L of H_2O . R-1 operated for 2 years until June of 1955 when it was shut down and decommissioned due to corrosion issues with a total output of 60,000 kW-hours. Subsequently, three reactors were operated on the NCSU campus until the early 1970s. Fig. 1 shows pictures for establishing the R-1 reactor.

In 1969, NCSU initiated construction of the PULSTAR reactor based on a design that was demonstrated by American Machine and Foundry (MacPhee and Wett, 1963). The PULSTAR became operational with a maximum power of 1-MWth and achieved first criticality on September 9, 1972 at 1.52 a.m. This was a major facility that had all the elements of a modern education and research reactor including beam tubes for experimental facilities. In 1978, the Nuclear Reactor Program (NRP), where the PULSTAR reactor is housed, was formally established as a university center. The mission of the NRP derives directly from the institutional mission and focuses on the three pillars of education, research and discovery, and outreach and extension. Consequently, on annual basis the PULSTAR serves the educational and research essentials of hundreds of students (undergraduate and graduate) and faculty, supports the technological needs of various industrial and governmental organizations, and provides many public outreach opportunities for K-12 students and teachers, technical 2-year colleges, and the community at large. Since 2010/2011, the NRP carries the designation of "Board of Governors' Center" within the University of North Carolina (UNC) system. In this capacity, it continues to serve its mission objectives with focus on the 17 campuses of UNC.

The PULSTAR has been operational for 47 years and is currently undergoing a license extension review by the US Nuclear Regulatory Commission. As part of this review, the PULSTAR's power is to be upgraded to 2-MWth. Over the past 18 years, the PULSTAR initiated major upgrades of its operational capabilities and facilities (Hawari, 2017). This included enhancing core reflection, licensing of new fuel, and major digital upgrades of its control console and instrumentation. In addition, several facilities for non-destructive materials examination and fundamental science were established. This includes an intense positron beam facility, a neutron imaging facility, a neutron powder diffractometer, and an ultracold neutron source. Recently, a hot cell facility was installed. A fuel testing loop is expected to be operational by the end of 2019. Fig. 2 shows a picture of the PULSTAR reactor core, and its bay as it currently stands with its various facilities. Details of the reactor design and its facilities are presented in sections *PULSTAR reactor design* and *PULSTAR experimental and educational systems*.

PULSTAR reactor design

PULSTAR core design and characteristics

The PULSTAR is a light water moderated and cooled reactor. It is fueled with uranium dioxide (UO_2) fuel that has an enrichment of 4% and/or 6% in U-235 by mass. The fuel is configured as sintered pellets in zircaloy two cladding and arranged into pins with every



Fig. 1 The biological shield and bay of the R-1 reactor (left), the Burlington building (center), and Clifford Beck with the R-1 reactor core in 1952 (right).

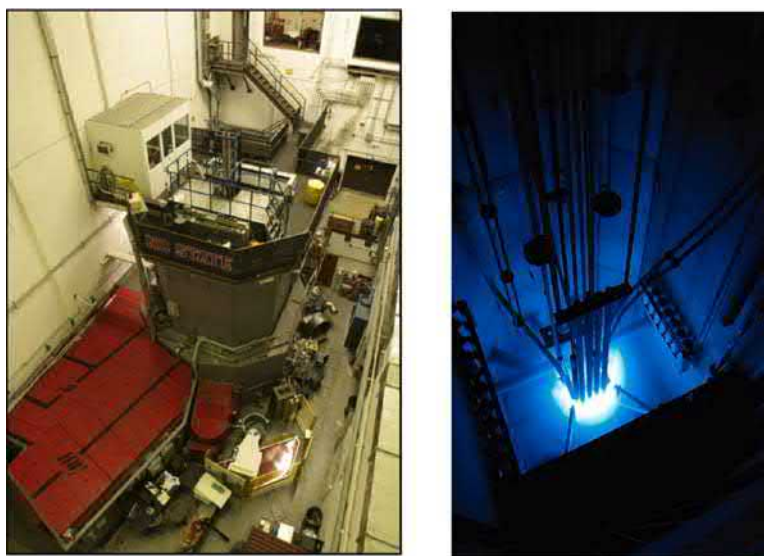


Fig. 2 The PULSTAR reactor bay showing various facilities (left). The core of the PULSTAR reactor at 1-MWth full power (right).

pin containing 25 pellets. The pins are configured into assemblies with a square array of 25 pins per assembly. The core of the PULSTAR is configured of a square array of 25 assemblies. The core is reflected on two sides with beryllium reflector assemblies that are used to improve its neutron economy and enhance excess reactivity. **Fig. 3** below shows the details of the PULSTAR fuel and the current configuration of the core.

A typical PULSTAR core is loaded with 360 kg of U-238 and 13 kg of U-235. This results in a H_2O -to- UO_2 volume fraction of near unity, which results in a highly under moderated core. Consequently, the thermal neutron flux does not peak in the core but on the periphery, while the fast and epithermal neutron fluxes peak in the core. **Fig. 4** shows the results of an MCNP simulation illustrating the spatial distribution of thermal, epithermal, and fast neutrons in the PULSTAR core. Furthermore, due to the large loading of U-238, the PULSTAR has a significant component of fuel temperature feedback reaching $-0.03 \text{ mk}/^\circ\text{C}$ with an overall power defect reaching -3.3 mk . Originally, the PULSTAR was designed to allow pulsing of the reactor to a maximum power of nearly 2200 MW. However, the NCSU PULSTAR is no longer licensed to pulse its power. **Table 1** below gives fundamental parameters that characterize the PULSTAR core ([PULSTAR SAR, 2019](#)).

PULSTAR control and safety systems

Control of reactivity in the PULSTAR core is achieved using four control rods that are fitted with silver-indium-cadmium alloy (80%–15%–5%) absorber blades. Each absorber blade is 0.46 cm thick and 61 cm long. One of the rods serves as a regulating rod, while the other three rods serve as safety rods. The regulating rod may be positioned automatically based on a power setting entered into the Automatic Power Control System. Each control rod is connected to a drive mechanism by an electromagnet, which can be de-energized within 50 ms. The drive speed for the control rods is 32 mm/s and the maximum reactivity insertion rate is limited to 2 mk/s. **Fig. 3** shows a schematic of a control rod and its drive mechanism.

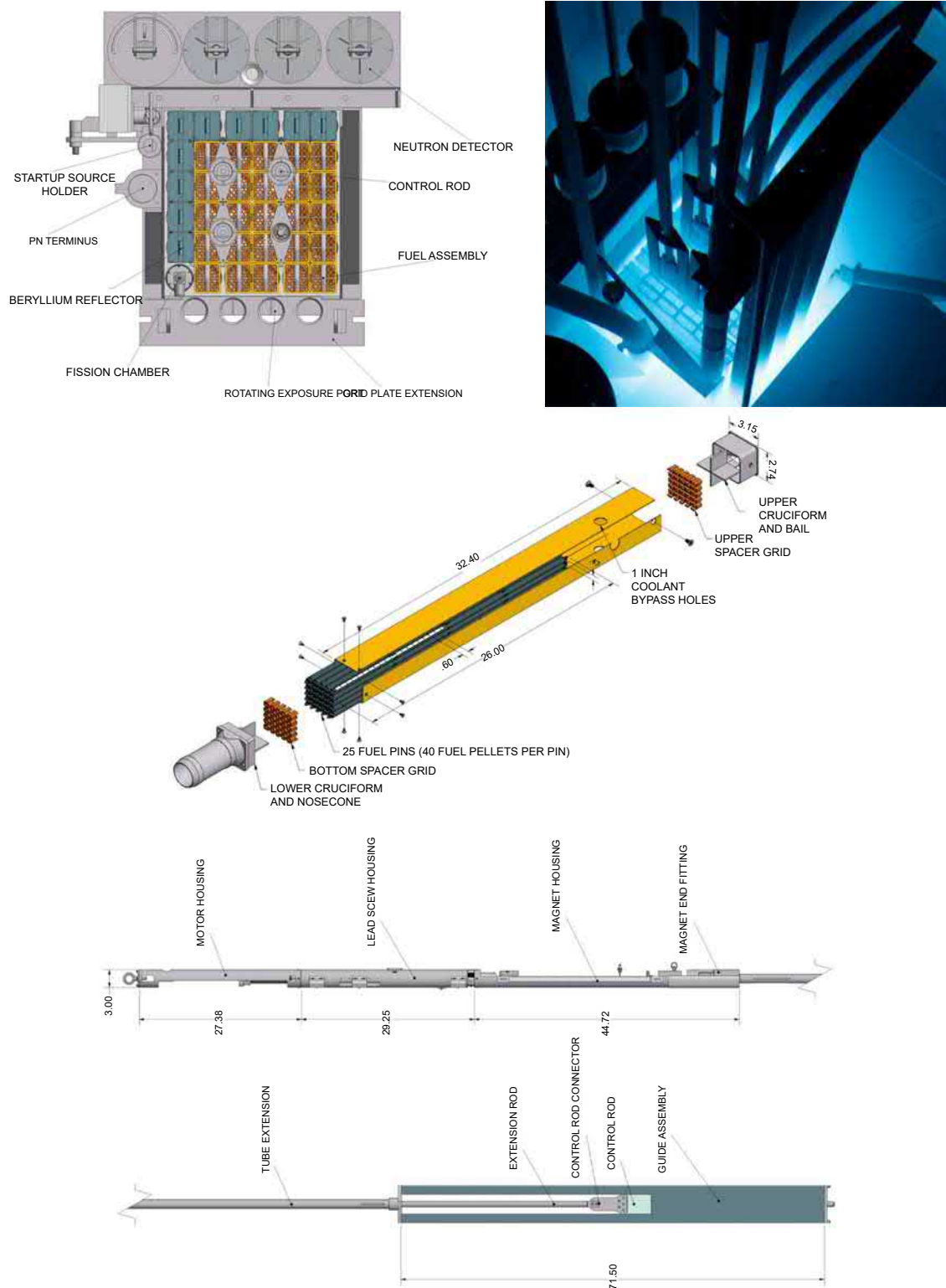


Fig. 3 The PULSTAR core and its layout (top). A schematic of the PULSTAR fuel (center). A schematic of the PULSTAR control rod and its drive mechanism (bottom, see section *PULSTAR Control and Safety Systems*). All dimensions are given in inches.

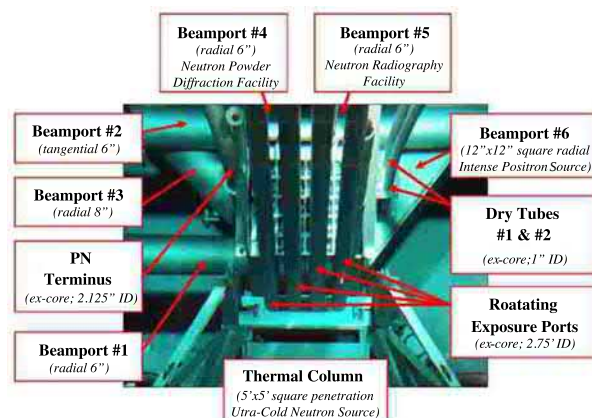
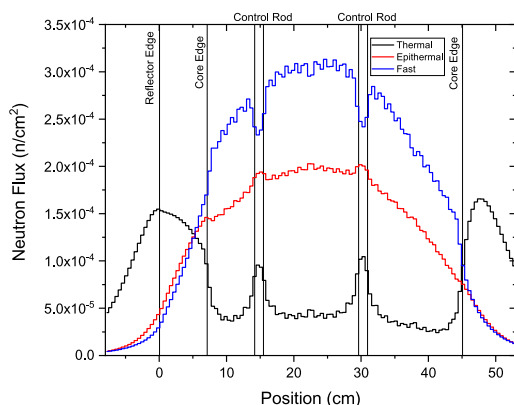


Fig. 4 The calculated energy and spatial distribution of neutrons in the PULSTAR core illustrating the significant flux of thermal neutrons (energy less than 0.5 eV) on the core's periphery (left). A picture showing the locations of the PULSTAR's in-pool rotating exposure ports (REPs), dry exposure ports (DEPs), and pneumatic transfer system (right). In addition, the PULSTAR's beam tubes are shown. By comparing the left and right images, the advantage of the resulting PULSTAR flux distribution can be observed.

Table 1 Parameters of the NCSU PULSTAR reactor that define its operational behavior. Unless indicated the values apply to both 1-MWth and 2-MWth operation.

Type of reactor	PULSTAR
Rated power (MWth)	1 (under upgrade to 2 MWth)
Type of fuel	Uranium dioxide (UO ₂)
Fuel enrichment (weight % U-235)	4% and/or 6%
Coolant/Moderator	Light water/light water (H ₂ O/H ₂ O)
H ₂ O-to-UO ₂ volume fraction	1.1
Core fuel loading (kg U-235)	13.3 (as of 2019)
Core inlet temperature (°C)	41
Average fuel temperature (°C)	145 (1-MWth), 223 (2-MWth)
Average coolant temperature (°C)	44
Fuel temperature coefficient of reactivity (mk/°C)	− 0.029 (1-MWth), − 0.028 (2-MWth)
Isothermal temperature coefficient of reactivity (mk/°C)	− 0.067
Moderator void coefficient (mk/cm ³) (reactivity feedback measured due to displacement of moderator)	− 0.01
Prompt neutron lifetime (μs)	80
Effective delayed neutron fraction, β _{eff} (mk)	7.33
Power defect (mk)	− 3.29 (1-MWth), − 5.60 (2-MWth)
Shutdown margin (mk)	35

PULSTAR cooling system

The PULSTAR reactor utilizes demineralized light water as coolant. The cooling system operates in forced convection mode, with downward flow through the core, and is designed to support the current power of 1 MWth and future operations at 2 MWth. In addition, at a power of 100 kWth or less, PULSTAR cooling can be supported using natural convection mode. The cooling system is comprised of primary and secondary subsystems. The primary subsystem has an open pool loop configuration and operates at atmospheric pressure. Water flows downward, at a rate of 32 L per second, through the core and transfers the heat generated by fission to the secondary subsystem using a plate type heat exchanger. Subsequently, the heat is removed by the secondary subsystem to the ultimate atmospheric heat sink using a standard counter-flow evaporative cooling tower. As a result, a pool inlet temperature below 41 °C is maintained with a temperature rise across the core of less than 7.5 °C. Fig. 5 below shows a rendering of the cooling system of the PULSTAR reactor and associated major components.

PULSTAR experimental and educational systems

In-pool irradiation capabilities

Facilities and locations for performing irradiations exist in the PULSTAR pool and in the vicinity of the core. This includes four vertical Rotating Exposure Ports (REP), which are typically used to perform neutron activation analysis (NAA), isotope production

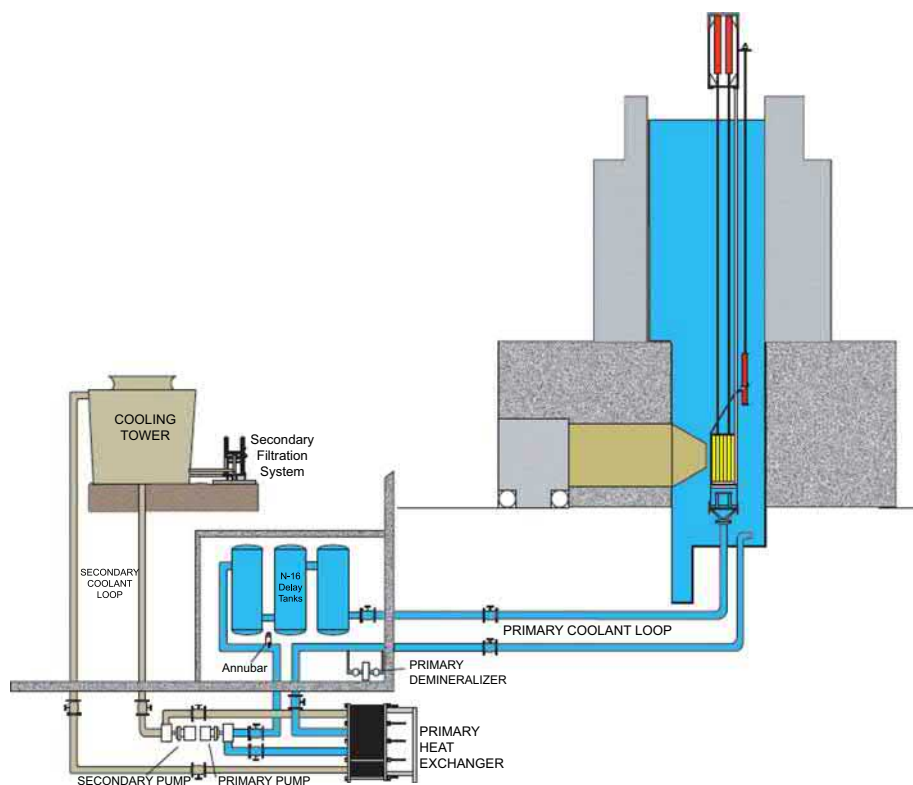


Fig. 5 A schematic of the current PULSTAR reactor's primary and secondary cooling systems.

Table 2 PULSTAR reactor's in-pool facilities and available neutron flux levels.

Facility/capability	Description	Neutron flux
Rotating exposure ports (4 in-pool, ex-core)	6.4 cm ID wet sample irradiation terminus.	1.0×10^{13} thermal n/cm ² /s 1.0×10^{12} fast n/cm ² /s
Dry exposure ports (2 in-pool, ex-core)	3.2 cm ID dry sample irradiation terminus.	4.0×10^{12} thermal n/cm ² /s 1.0×10^{11} fast n/cm ² /s
Pneumatic terminus (1 in-pool, ex-core)	3.2 cm ID fast rabbit sample transfer system	1.0×10^{13} thermal n/cm ² /s 1.0×10^{12} fast n/cm ² /s
Pool standpipes (4 in-pool, ex-core)	8.9 cm ID dry sample irradiation positions	1.0×10^{11} thermal n/cm ² /s 5.0×10^9 fast n/cm ² /s

experiments, and materials exposure experiments (e.g., for radiation damage studies). Samples of interest are usually encapsulated, loaded into irradiation containers and rotated during irradiations to ensure exposure uniformity. In addition, the PULSTAR's pool contains Dry Exposure Ports (DEP) and standpipes that are typically used for instrument and detector exposure and testing. Another capability is the pneumatic transfer system, that allows for performing rapid irradiations of small samples, e.g., for performing NNA, and is used in industrial and medical applications. Fig. 4 above shows the location of the in-pool capabilities. Table 2 below gives typical neutron fluxes expected in these ports.

Beam tube instruments and facilities

Over the past 18 years, unique beam tube instruments and facilities have been established at the PULSTAR reactor. Fig. 6 below shows a layout of the PULSTAR bay floor. A fuel testing loop dedicated to fission product gas release measurements is located on beam tube #1 (Liu et al., 2018), a neutron powder diffractometer is located on beam tube #4 (DiJulio and Hawari, 2008), a neutron imaging facility is located on beam tube #5 (Mishra et al., 2005), an intense positron beam is located on beam tube #6 (Hawari et al., 2011; Liu et al., 2013), and an ultracold neutron source is located in the thermal column enclosure (Korobkina et al., 2014). Fig. 7 shows pictures of the intense positron beam and neutron powder diffraction instruments that are often used in pre and post irradiation examination of materials. Table 2 below lists the instruments and basic characteristics that are of importance to their performance and operations. In addition, the bay floor also contains auxiliary facilities such as the recently installed

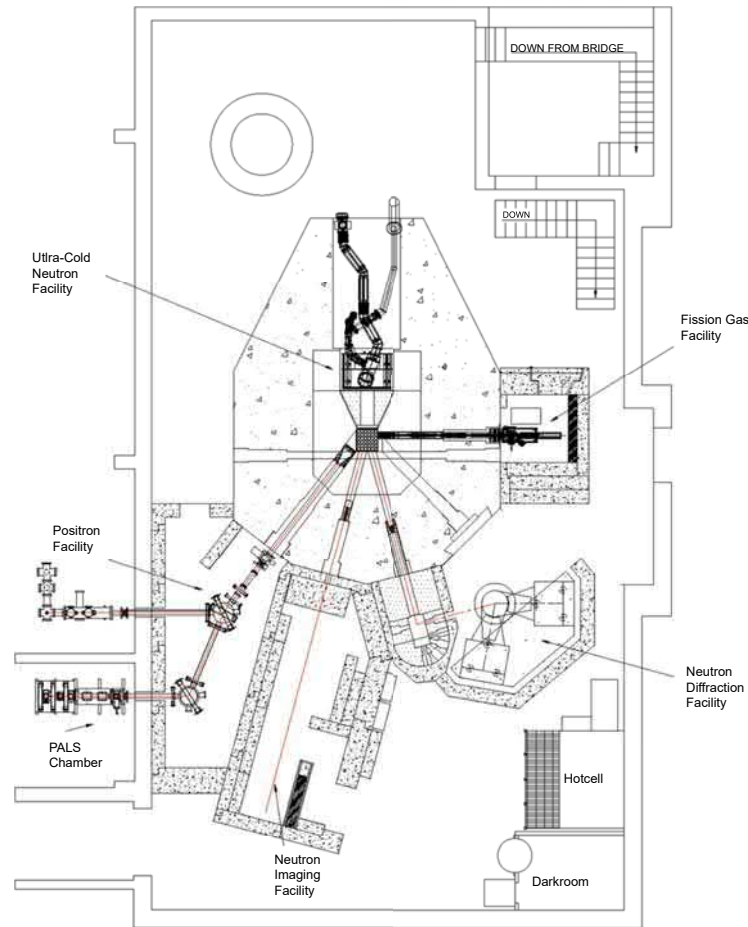


Fig. 6 The PULSTAR bay floor showing the beam tube instruments and various facilities.

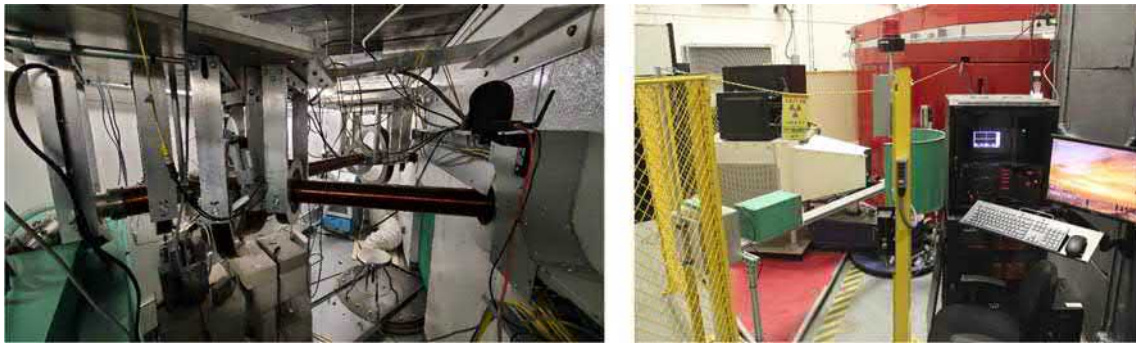


Fig. 7 Pictures of the Intense positron beam (left) and neutron powder diffraction (right) instruments at the PULSTAR reactor.

hot cell, which is used to support irradiation testing and isotope production, and a dark room that supports neutron radiography (Table 3).

Education and training capabilities

In addition to providing radiation experimental capabilities, the NRP with its PULSTAR reactor serves as a center for education in nuclear reactor and radiation physics. In this capacity, support is provided for typical reactor physics and engineering coursework in nuclear engineering. In addition, various training workshops are held annually to provide experience in reactor operations, radiation protection, and nondestructive examination to students at the K-12 and university levels.

Table 3 PULSTAR reactor instruments that are utilized in various testing and materials examination exercises.

Instrument/facility	Characteristics/performance
Fission gas measurement loop	Located on beam tube #1. Performs temperature dependent measurements of fission gas (krypton and xenon) release into flowing helium sweep gas using gamma-ray spectrometry.
Neutron powder diffractometer	Located on beam tube #4. Performs powder diffraction on samples with $\Delta d/d$ of 0.3%. The flux on sample is 10^5 n/cm ² ·s. Uses bent Si single crystal monochromator and beam energy of 30 meV (1.48 Angstroms). Able to extract diffraction patterns and pair distribution functions.
Neutron imaging facility	Located on beam tube #5. ASTM I ^A thermal neutron beam with flux on sample of 10^6 – 10^9 n/cm ² ·s. Spatial resolution in the range of 30–100 μ m depending on imaging modality. Capable of real time radiography and tomography.
Intense positron beam	Located on beam tube #6. Mono energetic positron beam generated using fission gamma-rays from the PULSTAR reactor's core in combination with capture gamma-rays in a cadmium shroud. Energy can be 1–17 keV and intensity reaches 10^9 e ⁺ /s. The beam is used in Doppler broadening and positron annihilation lifetime spectroscopy.
Ultracold neutron source	Located in the thermal column enclosure. Solid deuterium source (at 5 K) based on close coupling with fast neutrons leaking from the PULSTAR's core. Anticipated ultracold (300 nano eV) density of 30/cm ³ .

**Fig. 8** A remote IRL session (top). The main experimental interface displaying PULSTAR's control console telemetry such as power and control rod positions (bottom).

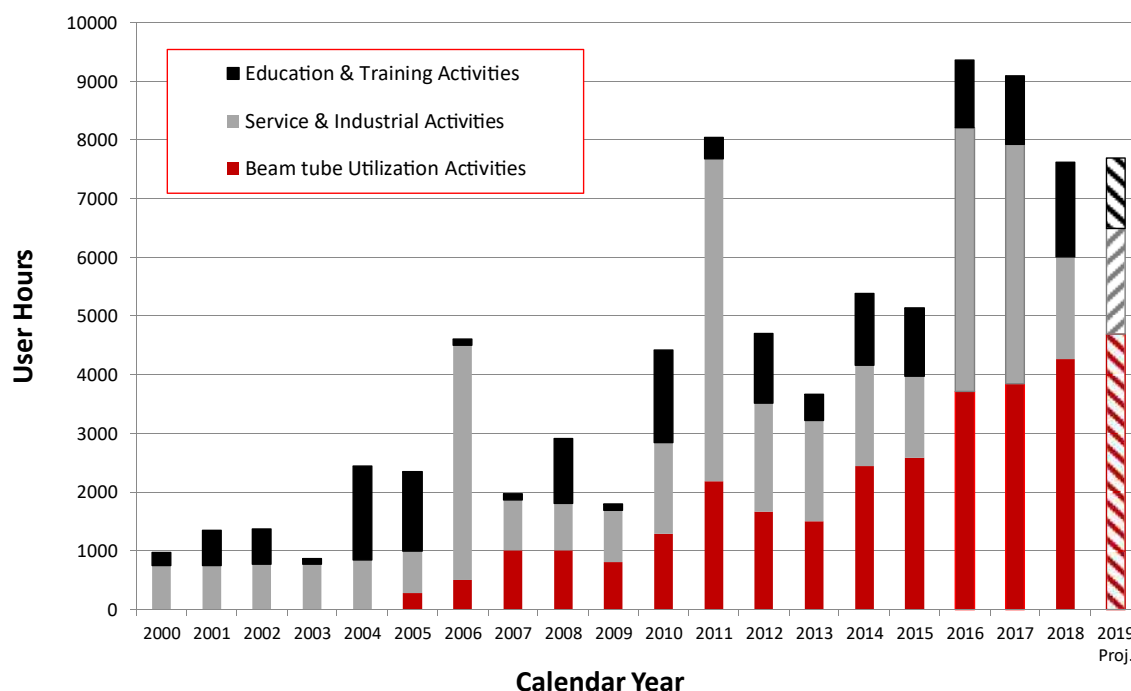


Fig. 9 The utilization of the PULSTAR reactor and associated laboratories represented as annual user hours covering education, service, and beam tube activities.

A unique modality that has been developed and offered at the PULSTAR reactor is the Internet Reactor Laboratory (IRL), which provides students at remote sites access to performing reactor physics and engineering experiments (e.g., control rod worth measurements, approach to critical analysis, etc.) (Hawari et al., 2020). Laboratory classes were first offered in 2004 within the United States. In 2010, and in collaboration with the US Department of State and the International Atomic Energy Agency (IAEA), classes were offered internationally. Recently, a second generation system (hardware and course modules) for remote education has been implemented. Fig. 8 shows pictures of an IRL session performed with the Jordan University of Science and Technology, and the main graphical interface representing the PULSTAR's control console.

Utilization and services

The Nuclear Reactor Program (NRP) is currently designated as a Board of Governors' Center within the University of North Carolina system of 17 campuses. In addition, it is available as a nuclear education and research center for users on a global scale. As indicated above, a development effort, that benefits from the characteristics and design of the reactor, took place over the past 18 years. This resulted in establishing various unique facilities and instruments and, in turn, drove unprecedented enhancement in the utilization levels of the NRP and the PULSTAR reactor. Specifically, utilization increased by nearly an order of magnitude for the period extending between the years 2000–19. Furthermore, the development of such capabilities allowed the NRP to become a partner in various US based national consortia including the US Department of Energy's Nuclear Science User Facilities (NSUF) and the US National Science Foundation's Research Triangle Nanotechnology Network (RTNN), and to participate in international efforts with organizations such as the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) of the Organization of Economic Co-operation and Development (OECD).

Fig. 9 below shows the utilization data presented as annual user hours. As it can be seen in the figure, a major component of utilization is due to beam tube instruments. This is a direct indicator of the diversity of utilization as the beam tubes present users from various disciplines (engineering, science, etc.) with options to perform materials examination studies. Consequently, this clearly demonstrates that the utilization of a research reactor depends on expanding its availability to a multi- and inter-disciplinary user base.

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TRIGA[®] Reactors: A Unique Approach in Research Reactor Fuel and Design[☆]

Steve Reese, School of Nuclear Science and Engineering, Oregon State University, Corvallis, OR, United States

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Introduction

TRIGA[®] (Training, Research, Isotope, General Atomic) reactors were conceived and developed by the General Atomic division of General Dynamics Corporation. They had the objective of providing a training, research, and isotope production reactor with intrinsic safety features designed to significantly reduce the probability and consequences of a nuclear accident. This was termed a “walkaway safe” reactor.

General Atomics developed a unique fuel-moderator element consisting of a solid homogeneous alloy of uranium fuel and a zirconium hydride moderator (U-ZrH_n). These unique fuel elements provide the TRIGA[®] reactor with a large prompt negative temperature coefficient. Any increase in power immediately increases temperature in both the fuel and the moderator, instantly causing the fuel to become less effective (a decrease in reactivity). This decrease causes the reactor to return to safe power levels within milliseconds, much faster than an operator or electronic circuit could react.

This intrinsic self-regulating characteristic is integral to the TRIGA[®] reactor and permits safe, steady-state operation in addition to pulsing to very high power levels without the use of any mechanical or electronic control devices to return the reactor to a safe power level.

History

By 1956, the year General Atomics was formed, considerable progress had already been made in devising safety systems to minimize the possibility of reactor accidents. All nuclear reactors had some type of mechanical device to shut down the reactor if power levels exceeded a set limit. These systems usually employed circuits that would electronically analyze the level of radiation received by a detector. If the radiation level or its rate of increase exceeded preset limits, the safety circuits would initiate a reactor shutdown by dropping the control rods. However, out of a desire to make the reactor *inherently* safe, three prominent nuclear scientists, Freeman Dyson, Massoud Simnad and Edward Teller, set out to make the reactor fuel itself the primary means for ensuring safety (Dyson, 1979).

Rather than depend entirely on electronic circuitry or moving parts, General Atomics incorporated physical properties inherent to the fuel itself. These properties provide feedback which limits the reactor to safe power levels, or causes it to shut down. Initially proposed by Dyson, the “warm neutron principle” (which would later be named the Dyson Effect) described a way of creating an extreme negative temperature coefficient by coupling the moderator with the fuel itself.

In roughly 2 years of effort developing the methodology for the metallurgical development of this novel fuel design, a working prototype reactor called the TRIGA[®] Mark I (Fig. 1), was built. It was taken critical on May 3, 1958. The Mark I ran at a power level of 10 kW at the brand new General Atomic research park being developed in La Jolla, California (IAEA, 2016). Remarkably, this reactor operated until 1995.

This reactor design can arguably be said to have become renowned when General Atomics set up operating demonstrations of the Mark I reactor at the Second United Nations Conference on the Peaceful Uses of Atomic Energy in Geneva, Switzerland in 1958 and at the World Agricultural Conference in New Delhi, India in 1960. At this later conference, President Dwight Eisenhower famously pressed the button which began the operation of the reactor amongst the many dignitaries present at the event (Fouquet et al., 2003).

[☆] *Change History:* February 2021. Steve Reese updated explicit locations for the figures in revision 5 and removing Figure 7. Figure 8 in revision 5 is now Figure 7 in revision 6.

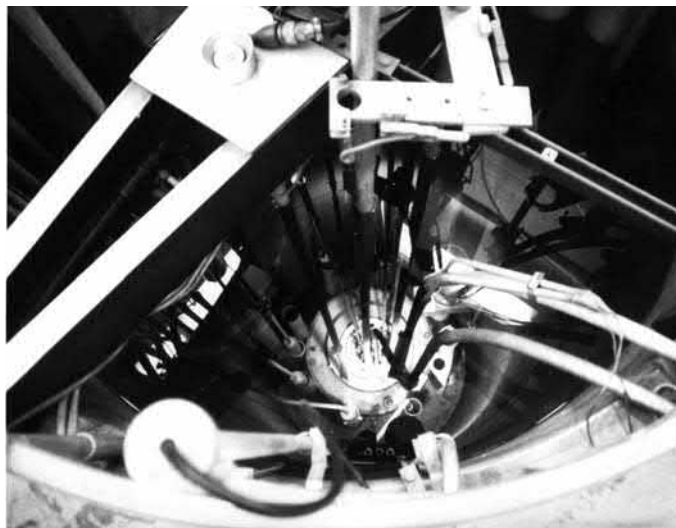


Fig. 1 The original TRIGA[®] Mark I reactor at the General Atomics research park in La Jolla, CA.

TRIGA[®] reactor basic design features

The heart of a TRIGA[®] reactor is its core. The core contains the fuel elements, graphite reflector elements, reflector assembly, and control rods. As a thermal reactor, the TRIGA[®] depends on thermal (low energy) neutrons to cause fission chain reactions in its fuel. Because the neutrons born in the fission of a ^{235}U atom are at a high energy, the probability of them causing another fission event before they leak out from the core is very low. A moderator is employed to thermalize (slow down) the fast (high-energy) neutrons. Moderation is the removal of energy from fast neutrons by collisions with light atoms of moderating material. In the TRIGA[®], neutron thermalization is achieved with zirconium hydride, water, and graphite as moderating materials; most of the moderation occurs in the fuel itself (General Atomics, 1958).

In any reactor, a certain percentage of the neutron population tends to “leak” out of the core: neutrons physically escape from the core and are absorbed elsewhere, such as in water or structural materials. To reduce this leakage, the core is surrounded by a graphite reflector, which reflects some of the escaping neutrons back into the core.

When a TRIGA[®] reactor is at power, the largest portion of energy being released by fission in the fuel appears as heat (Fig. 2). Water, moving by natural convection, is the primary reactor cooling mechanism except with the highest power designs (i.e., > 2 MW) where force flow is required. A secondary water system using a water-to-water heat exchanger removes heat from the primary coolant, and a forced air (or some other ultimate heat sink) cools the secondary water.

Normally a TRIGA[®] reactor is not operated for long periods of time at a high neutron flux. As such, many of the atoms born directly from fission that readily compete with the uranium in the fuel for neutrons are not present in high enough concentrations to tangibly affect the operation of the reactor. Some of these atoms have a large affinity to absorb neutrons (the physics term for this is called a cross section) that were intended to cause a fission reaction in the uranium fuel. Of these atoms, a radioisotope of xenon (^{135}Xe) has such a large cross section (i.e., affinity for absorption of neutrons) that it can prevent the startup of a reactor until it radioactively decays away. This phenomenon is referred to as “xenon preclusion”. If operation of the reactor does require long, sustained operations where xenon preclusion is potentially an issue, TRIGA reactors can simply add enough uranium fuel to overcome or override the ^{135}Xe that is created during the operation of the reactor.

Reactor fuel

The active part of each fuel-moderator element (Fig. 3), is approximately 3.05 cm in diameter and 38.1 cm long. TRIGA[®] fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing 8–30 wt% of uranium enriched up to 19.75% in ^{235}U . The hydrogen-to-zirconium atom ratio is approximately 1.6–1.0. To facilitate hydriding, the addition of super-saturated hydrogen in the fuel matrix, a 0.48 cm diameter hole is drilled through the center of the active fuel section; a zirconium rod is inserted into this hole after hydriding is complete (Simnad, 1980).

Each element is clad in 0.0508 cm of stainless steel, and all closures are made by heliarc welding. Two 8.9 cm sections of graphite are inserted into this stainless steel cladding, one above and one below the fuel, to serve as top and bottom reflectors for the core. Stainless steel end fixtures are attached to both ends of the can, making the overall length of the fuel-moderator element nominally 73.7 cm.



Fig. 2 The Oregon State TRIGA® Reactor operating at 1-MW.

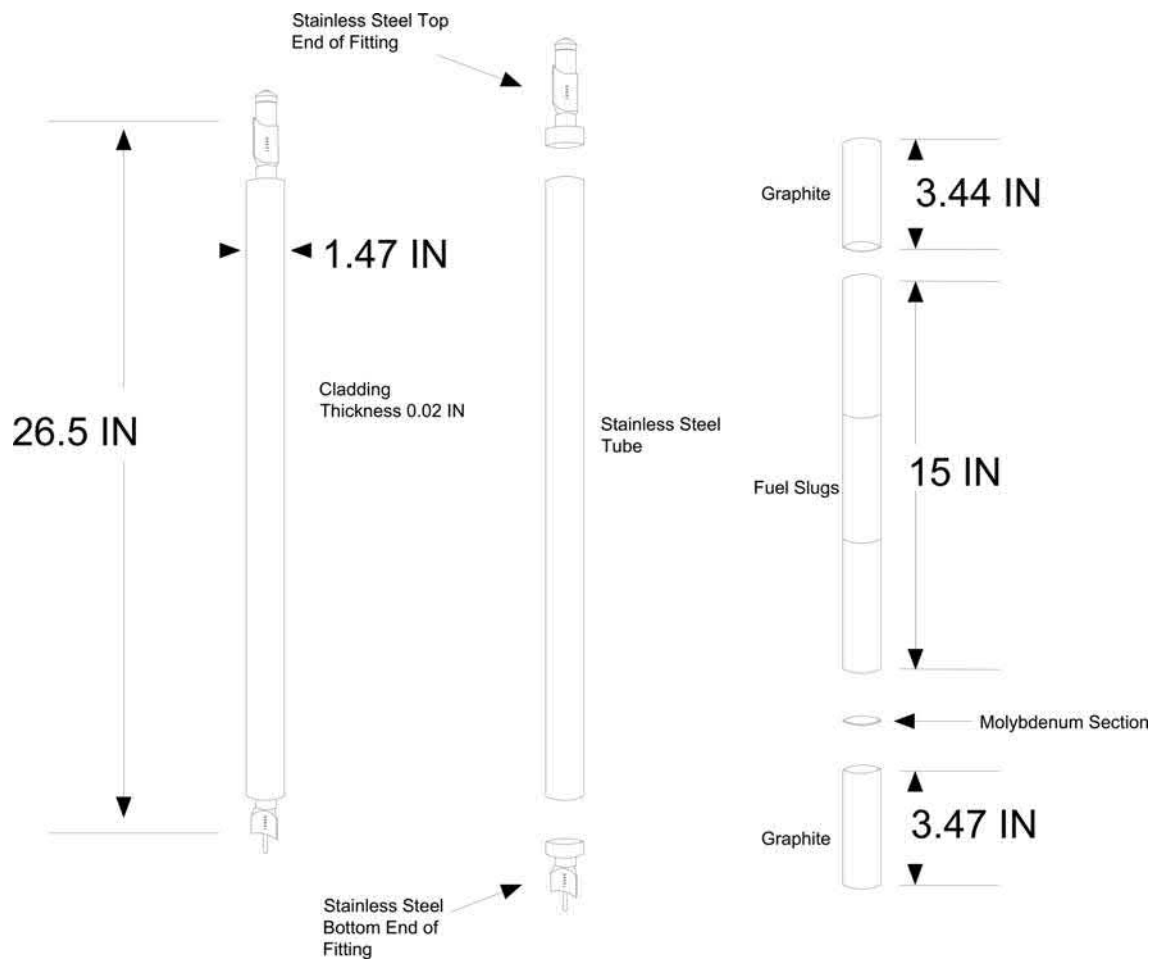


Fig. 3 Illustration of a typical standard TRIGA® fuel element showing the orientation of the fuel slugs, molybdenum disks, and graphite inside the stainless steel cladding.

The lower end fixture supports the fuel-moderator element on the bottom grid plate. The upper end fixture consists of a knob for attachment of the fuel handling tool, and a triangular spacer, which permits cooling water to flow through the upper grid plate. The total weight of a fully loaded fuel element is only about 3.2 kg (7 lbs).

Instrumented fuel elements are also commonly used in TRIGA® reactors. An instrumented fuel element has three thermocouples embedded in the fuel (Fig. 4). The sensing tips of the fuel element thermocouples are located halfway between the outer radius and the centerline at the axial center of the fuel section, and 2.54 cm above and below the axial center. The thermocouple lead-out wires pass through a seal contained in a stainless steel tube welded to the upper end fixture. This tube projects about 7.62 cm above the upper end of the element, and is extended by two 3 m lengths of tubing connected by unions to provide a watertight conduit carrying the lead-out wires above the water surface in the reactor pool. In other respects, the instrumented fuel-moderator element is identical to the standard element.

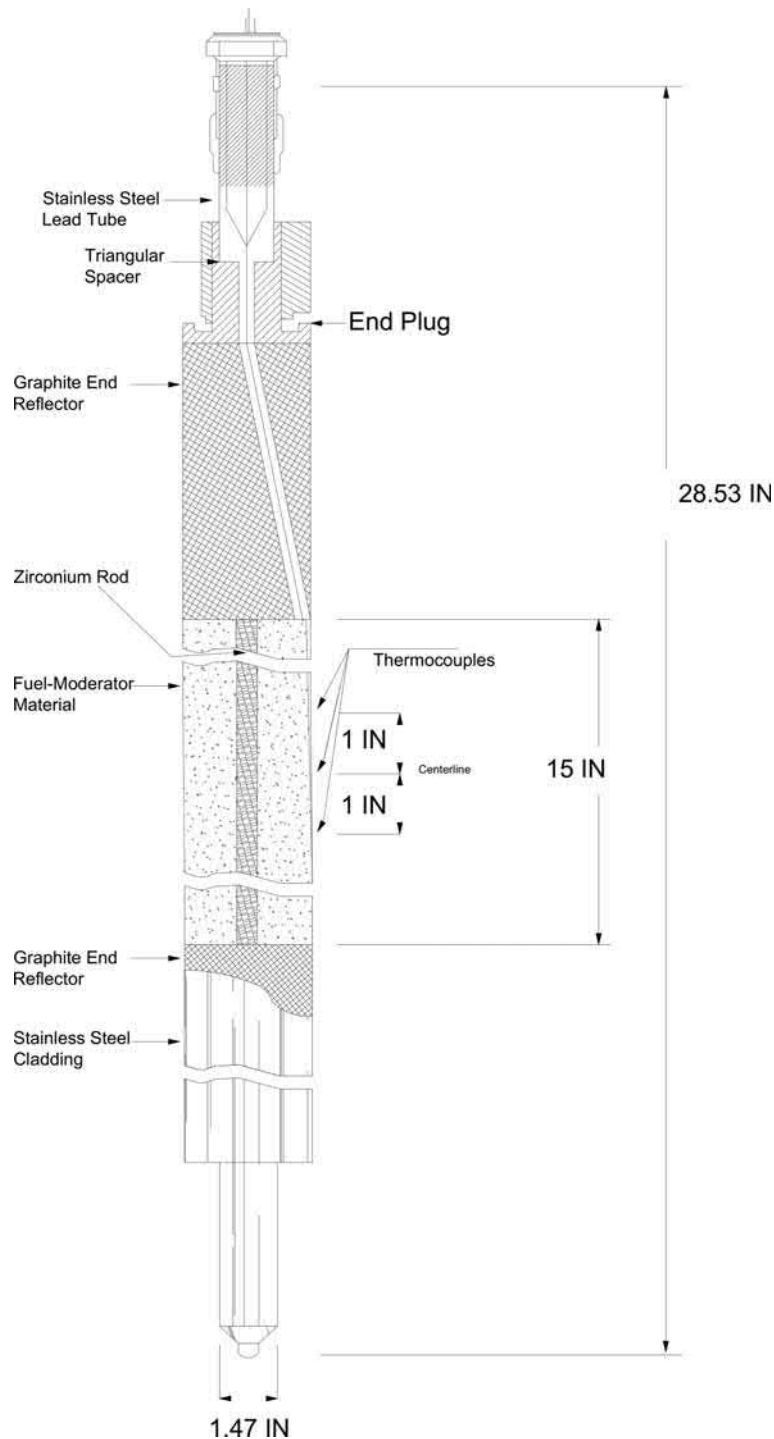


Fig. 4 Illustration of a typical TRIGA® instrumented fuel element, used for measuring fuel temperature during irradiation, showing the path and location of the embedded thermal couples.

The specific characteristics that make TRIGA®-type fuels uniquely suited for use in extremely safe research-type reactors are covered in detail later in this article. A summary of the characteristics is given below:

- $\text{ZrH}_{1.6}$ is single phase up to 649 °C [delta phase region];
- low hydrogen equilibrium disassociation pressure at normal fuel temperatures;
- high hydrogen retention;
- high heat capacity;
- low thermal expansion coefficient;
- relatively low chemical reactivity with water;
- high fission product retention;
- very large negative prompt temperature coefficient of reactivity;
- high burnup possible by addition of burnable poison; and
- high loading of uranium possible with insignificant change in fuel material properties.

Pulsing capability

With the development of the inherently safe fuel itself, a remarkable ability to operate reactors utilizing this fuel in a prompt critical condition was created. This ability comes from the heart of the mechanism that creates the intrinsic safety of the fuel.

When a reactivity insertion exceeds the fraction of delayed neutrons, the fission chain reaction is supported by prompt neutrons. Because the mean time between generations of prompt neutrons born from fission is so short, it is impossible to mechanically control the rate of increase of this reaction.

Several mechanisms within the uranium-zirconium hydride fuel matrix allow for the safe, controllable rapid insertion of a large amount of negative reactivity. The molecular structure of the fuel matrix and the presence of the burnable poison erbium both contribute to a large prompt negative fuel temperature coefficient of reactivity. The ^{167}Er absorption cross section has a strong, broad, resonance that peaks at 0.5 eV. As the fuel temperature increases after a large positive reactivity insertion, the spectrum of thermal neutrons in the core is shifted to higher energies caused by scattering collisions with the hydrogen in the fuel matrix. This spectral shift pushes more neutrons into the resonance absorption peak of the erbium, increasing the probability of neutron capture in the erbium prior to fission. This strong, negative fuel temperature feedback ($\sim 10^{-4} \Delta k/k/^{\circ}\text{C}$) allows for the TRIGA® to be safely pulsed to a power thousands of times larger than that of steady-state operation (Simnad et al., 1976). Pulses as high as \$5.50, obtaining peak powers on the order of 6400 MW were demonstrated at the General Atomics Mark F reactor.

It is not clear when the first pulse of a TRIGA® reactor happened but it is likely it occurred in late 1958 with the original Mark I reactor at General Atomics. That reactor was famously first publically pulsed by Nobel Laureate Niels Bohr by the simple push of a button at the inauguration of the new General Atomics research facilities in Ja Jolla in 1959. By the time that reactor was decommissioned, it had performed over 13,000 pulses (IAEA, 2016).

TRIGA® fuel development

The development and use of U-ZrH fuels for the TRIGA® reactor began at GA in 1957 and continues today. Over 6000 fuel elements of seven distinct types have been fabricated for the more than 60 TRIGA® research reactors operated in various countries around the world. The earliest of these has now passed 50 years of operation. U-ZrH fuel has exhibited unique safety features including a large negative prompt temperature coefficient of reactivity, high fission product retention, chemical stability when quenched from high temperatures in water, and dimensional stability over large swings of temperature.

TRIGA® fuel was developed around the concept of inherent safety. A core composition was sought which had a large negative prompt temperature coefficient of reactivity such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature increase would automatically cause the power excursion to terminate before any core damage resulted. Experiments performed in the late 1950s demonstrated that zirconium hydride possessed the basic mechanism needed to produce this desired characteristic. Additional advantages include ZrH's high heat capacity, allowing construction of a reactor with a relatively small core size and high flux values due to the high hydrogen content of the fuel rods, and can be used effectively to fabricate rugged fuel rods.

The standard TRIGA® fuel contains 8.5 wt% uranium (20% enriched) as a fine metallic dispersion in a zirconium hydride matrix. The hydrogen to zirconium (H/Zr) ratio is nominally 1.6 (in the face-centered cubic delta phase). The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For U-ZrH_{1.6}, the equilibrium hydrogen pressure is 1 atm at about 760 °C. The single-phase, high-hydride composition eliminates the problems of density changes associated with phase changes and with thermal diffusion of the hydrogen. A highly-enriched version of TRIGA® fuel titled "Fuel Lifetime Improvement Program" or "FLIP" (with a 70% enrichment) contained up to about 3% erbium as a burnable poison in order to increase the core lifetime and also contribute to the large negative prompt temperature coefficient (Simnad, 1980). The calculated core lifetime for FLIP fuel in a typical TRIGA® reactor was approximately 9 MW-years. However, FLIP fuel is no longer available because the enrichment exceeds 20%, the accepted maximum enrichment of low-enriched uranium (LEU) fuel.

In early 1976, GA undertook the development of fuels containing up to 45 wt% uranium (3.7 g U/cc) in order to allow the use of LEU fuel to replace the highly-enriched fuels while maintaining long core life. The 45 wt% fuel contains 19.75% uranium (referred to as 20-wt% although the actual number is just under 20-wt%). These fuels were fabricated successfully, with the required hydrogen content and erbium loading. The structural features of the hydrated LEU fuel were similar to those of the well-proven 8.5- and 12-Uwt% fuels, as shown by metallographic, electron microprobe analysis, and X-ray diffraction examination (Simnad, 1980). Detailed evaluations of the new LEU fuel have shown that it performs essentially identically to the older standard TRIGA[®] fuel.

Nuclear design and analytical studies have shown that the prompt temperature coefficient for the 20-wt% uranium fuel is essentially the same as that for standard fuel over the temperature range of interest (20–700 °C). The prompt temperature coefficient for the more highly uranium-loaded LEU fuel shows a temperature dependence (i.e., increasing with temperature), whereas the coefficient is relatively constant for standard fuel. The value of the prompt temperature coefficient of reactivity is slightly lower for the 30-wt% uranium fuel compared to the highly-enriched fuel it replaces; however, it is still large and significantly higher than the prompt temperature coefficients for any other type of reactor fuel.

Inclusion of erbium burnable poison in the TRIGA[®] LEU fuel has enabled core lifetimes of up to 3000 MW-d to be predicted for the 30-wt% fuel; this is the core life from the time of initial refueling to end of useful life. There are various TRIGA[®] fuel forms available today with a variety of enrichments and weight percents (Table 1). Although some early versions of 8.5/20 and 20/20 fuels had aluminum cladding, all fuels available today have Type 304 stainless steel cladding or Incoloy-800 for the 45/20 fuels. The most common TRIGA fuel types made today are the 30/20 and the 20/20 TRIGA fuels.

TRIGA[®] fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction from temperatures to 1150 °C (Foushee and Peter, 1971). After this temperature, hydrogen migration can lead to plastic deformation of the cladding. The basic limit for TRIGA[®] fuel was therefore determined by this dissociation of hydrogen from the U-ZrH_n fuel. A fuel temperature safety limit of 1150 °C for pulsing precludes loss of cladding integrity when the cladding temperature is below 500 °C. When the cladding temperature equals the fuel temperature, the fuel temperature limit is 950 °C.

Reactor versions

Beginning with the original Mark I reactor at the General Atomics site in La Jolla, General Atomics went on to design six unique versions of the TRIGA[®] reactor. Fundamentally, the defining common characteristic of each was the use of the U-ZrH_n fuels. All are open pool-type reactors that, except for the highest power design, have very similar core lattice configurations. The differences between the designs are the irradiation facilities and capabilities.

The original Mark I was a below-ground version of the open pool TRIGA[®] reactor with a circular grid and annular reflector. The reactor core is a cylinder consisting of cylindrical rod-type fuel-moderator elements and graphite reflector elements organized in rings. Water occupies approximately one-third of the core volume. An annular radial reflector surrounds the core and is supported by an aluminum platform at the bottom of the tank. The reflector consists of graphite with a 5.08 cm layer of lead on the periphery to reduce the gamma dose emanating from the core. A typical Mark I core utilizes approximately 90 elements in a circular grid array. The number of elements varies depending on the power level, burnup, and grid lattice but generally range from 85–120 elements for all versions of TRIGA[®] reactors. With this design reactor powers can reach as high as 2 MW, but typically are 250 kW to 1 MW. All are cooled by natural convection. The reactor is controlled by three or four neutron-absorbing control rods, classically called the shim, safety, regulating, and for reactors capable of pulsing, a transient control rod. The shim, safety and regulating rods are equipped with a fueled follower while the transient rod has an air follower. Typically three or four neutron detectors monitor the neutron flux, which is then used to indicate reactor power level. An automatic control system is incorporated to maintain the reactor at desired power levels by automatically adjusting the regulating rod height.

One of the main purposes of this reactor is to perform neutron irradiation of various materials. Irradiation facilities of a typical Mark I consist of a 40-position rotating specimen rack, a pneumatic transfer system, three in-core irradiation tubes, and a central thimble. The rotating specimen rack, also known as the “Lazy Susan,” is in a watertight assembly located in a well in the top of

Table 1 Nominal characteristics of TRIGA[®] fuels (NUREG-1282, General Atomics Technologies, 1987).

Fuel type (U wt%/enrichment%)	LEU 8.5/20	LEU 12/20	LEU 20/20	LEU 30/20	LEU 45/20	HEU FLIP
Uranium content (mass %)	8.5	12	20	30	45	20
²³⁵ U enrichment (mass % U)	19.75	19.75	19.75	19.75	19.75	70
Erbium content (mass %)	0	0	0.5	0.9	1.8	1.6
Original ²³⁵ U Mass (g)	39	55	99	162	282	137
Core lifetime (MW-d)	100	145	1200	3000	4000	3500
Uranium volume (%)	2.6	4.7	6.8	11.2	19.5	2.6

the reflector. The pneumatic transfer tube terminates into one of the lattice positions of the core. Dry, vertical in-core tubes are also commonly incorporated. A center location, usually called the central thimble, penetrates the center of the core, along its vertical axis; this location produces the highest thermal neutron flux.

The Mark II design (Figs. 5 and 6), is essentially an above-ground version of the Mark I, able to utilize the same core configurations and geometries as its predecessor. A large semi-octagon shaped concrete bioshield provides shielding in the horizontal plane, and contains a cylindrical reactor tank nominally 2 m in diameter by 8 m in depth. The above ground configuration allows for a substantial number of additional irradiation capabilities, most notably beam ports. A typical Mark II has four beam ports projecting from the reactor core through to the outside of the bioshield. Three of the beam ports are oriented radially out from the core; the fourth is tangential to the core to promote a beam with a high thermal neutron content while minimizing epithermal and fast neutrons as well as photons. This configuration is ideally suited for thermal neutron radiography. Mark II designs also typically have a thermal column, a large square compartment filled with graphite that produces a neutron field with a very high cadmium ratio (i.e., thermal to fast neutron ratio). It allows for samples to be inserted in the middle through the use of a large and heavy (~9 tons) wheeled door. Reactor power typically found in a Mark II design range from 500 kW to 2 MW, and utilize natural convection cooling. Thermal neutron fluxes of greater than $1\text{E}13 \text{ n cm}^{-2} \text{ s}^{-1}$ are easily achievable for a 1 MW TRIGA® reactor.

An early prototype design, designated the Mark F, was built at the General Atomics facility in 1960 and served as a testbed for the fuel development program. Nearly identical to the Mark F was the Mark III design. The Mark III, is a modified version of the Mark II which incorporates a number of novel features not found in other TRIGA® reactors. While the core lattice arrangement is similar to Mark I and Mark II designs, the most notable unique characteristic of the Mark III is the core attached to a moveable bridge gantry. This allows for the core to be moved in close proximity to beam facilities, some of which are similar to the Mark II while others are unique to the Mark III design. One such unique facility is an exposure room designed to accommodate large experimental setups. This design is also associated with a large elongated pool, with nominal dimensions of $7.6 \text{ m} \times 3.0 \text{ m} \times 7.6 \text{ m}$. These reactors were designed for steady-state power levels from 1–2 MW under natural circulation conditions.

The Annular Core Pulsed Reactor design was created to take full advantage of the unique pulsing capability of TRIGA® fuel. Typically, the U-ZrH_n lattice is arranged in a hexagonal grid with a large dry central irradiation cavity. These cores were designed to operate at modest steady-state power levels of 500 kW but produce pulses on the order of 22,000 MW (Dee et al., 1966). The fuel is unique in that the outside cladding of the fuel elements used on TRIGA®-fueled annular core reactors are dimpled to allow for the higher peak fuel temperatures observed on the large pulses.

The Multipurpose Reactor design was created to enable reactor powers much higher than previous designs. Natural convection cooling of standard TRIGA® fuel is limited to no more than approximately 2.5 MW. Above this, forced cooling is required. Force cooling for these higher-powered reactors takes the form of a simple reverse flow configuration (water is forced down through the core lattice from the top) at moderate flow rates. Additionally, the TRIGA® fuel form is changed considerably to handle the higher

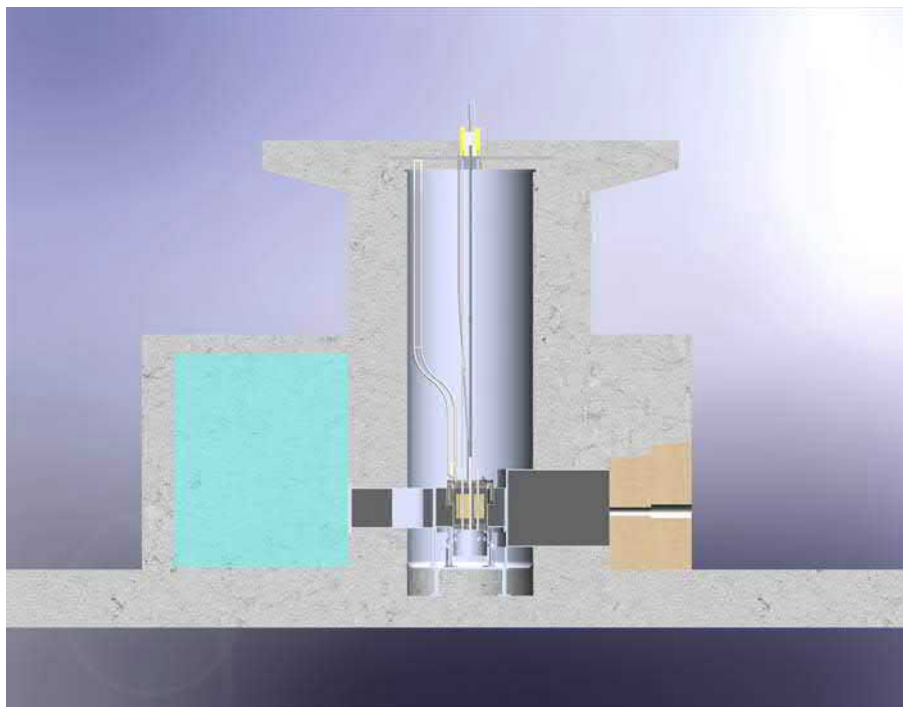


Fig. 5 Vertical cutaway illustration of a typical Mark II TRIGA® reactor. The orientation of the core at the bottom of a cylindrical tank surrounded by the concrete bioshield with the embedded irradiation facilities can be seen.

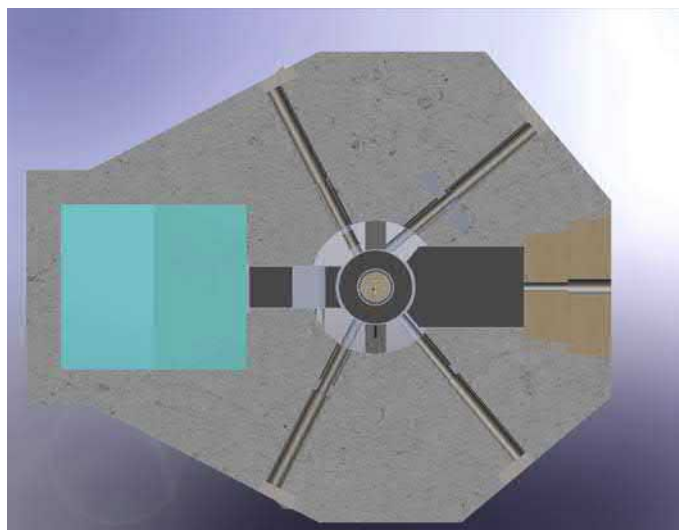


Fig. 6 Horizontal cutaway illustration of a typical Mark II TRIGA[®] reactor. The orientation of the core in the middle of the concrete bioshield with the embedded irradiation facilities (beam tubes and thermal column) can be seen.

heat flux off the surface of the fuel. In these reactors, the same U-ZrH_n is utilized but with two important differences. First, the fuel is much thinner, created in the form of 13 mm diameter fuel pins placed in a tightly packed square pitch configuration. Second, the weight percent of the fuel is much higher at 45%. Although this design is intended to operate from 5 to 20 MW, pulsing is not allowed in reactors operating above 3 MW. As of 2019, the only Multipurpose Reactor built is at the Institute of Nuclear Research in Pitesti, Romania.

The final design are the converted reactors. This is a group of reactors that were originally fueled with plate-type uranium-dispersion fuels (sometimes referred to as material test reactors or MTR) and subsequently converted to TRIGA[®] reactors by replacing the plate-type fuel with U-ZrH_n fuel (Fig. 7). Usually, the square pitch core lattice structure was retained. To fit the existing MTR core dimensions and orientation, General Atomic designed TRIGA[®] fuel with a slightly smaller diameter which could then be arranged in a square cluster assembly composed of four of these thinner TRIGA[®] fuel elements. Once converted, their physics and operational characteristics are nearly identical to original TRIGA[®] reactor designs. In all, 13 reactors were converted in this manner.

TRIGA[®] reactor utilization

TRIGA[®] reactors provide an optimum blend of capabilities ideally suited for a research or academic environment. As a result, 69 TRIGA[®] reactors have been built or converted to TRIGA[®] fuel in more than 30 countries, making it the most widely produced reactor of any kind. First and foremost this proliferation was driven by the inherent safety features offered by the fuel design: this is perhaps the most significant consideration for siting these reactors which are commonly found in urban environments or on university campuses. Even in the worst credible accident the resulting dose to the general public would be on the order of 1 mSv for a 2 MW TRIGA[®] fueled reactor due to the excellent fission product retention and high melting point.

In an academic setting, research reactors at universities have to balance the competing needs of research, academia, and service. Which need dominates is largely dependent on reactor power. Lower power reactors tend to place more weight on the academic rather than the research mission, with service work being a distant third. Successful research reactors of medium power usually place more weight on research, followed by the academic uses and with service work being the least important. Medium power research reactor programs tend lean towards the academic mission. Research reactors of higher power typically emphasize the research and service work more than an academic mission. The main reason for this is the notably higher operational tempo at high power research reactors. The increased neutron flux is best utilized for research or service, such as isotope production, only achievable in substantial quantities through continuous full power operations (IAEA, 2014).

Summary

TRIGA reactors are the most widely produced of any research reactor in the world, owing largely to the incredible versatility of the reactor design. From small 100 kW educational focused reactors to 14 MW research oriented designs, TRIGA reactors have proven to be immensely valuable instruments of science and engineering at universities and research centers. Furthermore, TRIGA[®] reactors

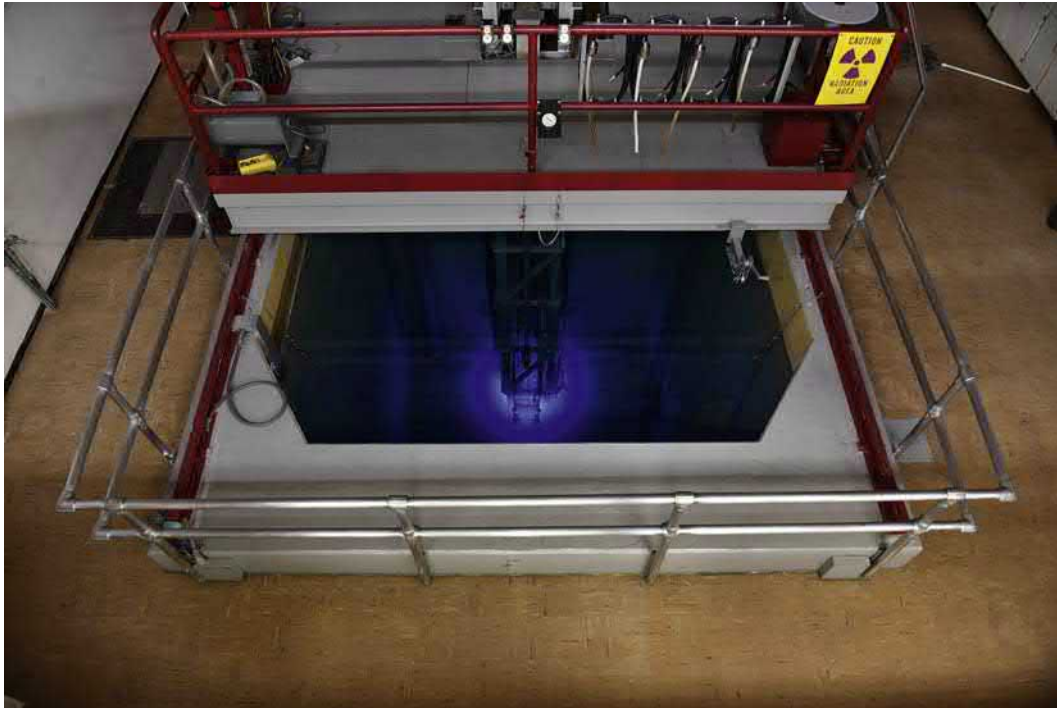


Fig. 7 Overhead view of the TRIGA® reactor at Washington State University. Called a conversion TRIGA®, this is an example of a reactor that was originally designed to utilize plate fuel but whose fuel was later replaced with TRIGA® fuel.

also provide ideal in terms of safety. The inherent safety of TRIGA® reactors has been demonstrated by the extensive experience acquired from similar TRIGA® systems throughout the world. This safety arises from the extremely large negative prompt temperature coefficient that is characteristic of uranium-zirconium hydride fuel-moderator elements used in TRIGA® systems. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. This results in a mechanism whereby reactor power excursions are terminated quickly and safely.

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Accelerator Driven Subcritical Systems

Hamid Aït Abderrahim, Didier De Bruyn, Gert Van den Eynde, and Rafaël Fernandez, Belgian Nuclear Research Centre (SCK CEN), Boeretang, Mol, Belgium

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Glossary

Beam Trip Interruption of the accelerator beam, resulting to the shutdown of the reactor. If this interruption is sufficiently long, the temperature within the reactor internals will decrease. Thermal unloading and immediate reloading of the internals is one of the most severe cases in the reactor design.

Diaphragm Reactor internal, which function is to separate the cold, high pressure LBE from the hot, low pressure LBE.

Abbreviations

ADS Accelerator Driven System
CW Continuous Wave
DC Direct Current
EFIT European Facility for Industrial Transmutation
FC Fuel Cycle
FP Fission Product
HLW High-Level Radioactive Waste
IPS In-pile Test Section
LBE Lead-Bismuth Eutectic
LFR Lead Fast Reactor
LINAC Linear Accelerator
LLFP Long-Lived Fission Products
MA Minor Actinide
MOX Mixed Oxide
MTBF Mean Time Between Failure
MYRRHA Multi-purpose hYbrid Research Reactor for High-tech Applications
P&T Partitioning and Transmutation
PWR Pressurized Water Reactor
RF Radiofrequency

Introduction

Energy in the world

One of the major challenges that our society faces is the increasing demand for energy in general and electricity in particular. According to international references (DOE/EIA, 2019; OECD, 2016) global energy requirements will increase by 50% by 2035. Many factors play a role in this like the expected growth of world population, making electricity available to the population that currently has not yet access to electricity, the higher consumption in developing countries and the average life expectancy increase, to name a few.

During the last century, our energy supply was mainly based on fossil fuels. While the increase of the global energy requirement is in a continuing trend, the available sources of energy are also being increasingly tapped. Fossil fuels are still in sufficient supply, are easily accessible and at a reasonable price. However, we are confronted today with excessive CO₂ emissions, so the use of fossil fuels needs to be limited. At the same time, renewable energy sources such as solar and wind based electricity generators cannot fully satisfy the global demand. Therefore, different countries recognize that nuclear energy needs to be included in the “energy basket” of the future.

Nuclear energy

Nowadays thermal nuclear reactors use nuclear fission of the fissile isotope of uranium (²³⁵U, and in a smaller amount, plutonium) induced by thermal (slow) neutrons to generate heat and electricity. During this process, the fissile atom is split. The current nuclear work horse for electricity production, the light water reactors, only fission a small portion (<1%) of the raw uranium. When nothing changes from a technological point of view, the known uranium resources may become scarce around the end of the century (OECD/IAEA, 2018) although there are scientists who believe that uranium is practically renewable (Conca, 2021; Herring, 2021). Reactors in which the neutrons have higher energies (so-called fast reactors) can utilize close to 100% of the natural uranium and hence need a smaller amount of uranium ore per unit of energy generated. Moreover, the fast reactors can also utilize depleted uranium huge quantities of which had been accumulated as “waste” in the uranium enrichment plants. These reactors would allow more efficient use of the uranium resources.

During their operation present nuclear power reactors produce high-level radioactive waste (HLW). For this high-level radioactive waste, a technical and socially acceptable solution is necessary. Even though the solutions that are being considered at this moment, like geological disposal, might offer a technically viable solution, the time scale needed for the radiotoxicity of the waste to drop to the level of natural uranium remains very long, which is not acceptable for part of the public.

In a nuclear reactor the released neutrons from a fission reaction can undergo various reactions; they can induce a new fission leading to the birth of new neutrons, they can be absorbed in the structures of the reactor or even in the fuel without inducing fission leading to the production of heavier elements than uranium (like neptunium (Np), plutonium (Pu), americium (Am), curium (Cm), etc.) or they can leak out of the reactor core and lost to the chain reaction. To maintain the chain reaction one needs a critical mass and critical dimensions of the reactor and these correspond to quantities of fissile materials arranged in a given geometry in such a way that the amount of neutrons produced by fissions is balanced by the amount of neutrons lost either by sterile absorptions or by leaking out of the reactor core. When these conditions are met, and the chain reaction is sustained by the system itself, the reactor is “critical.”

The core in an Accelerator Driven System (ADS) will be subcritical; it will not be able to establish a self-sustained chain reaction. In this context the term subcritical means that, on average, for each generation of fission neutrons, less than one neutron will be able of initiating a subsequent nuclear fission. This situation can be achieved by limiting the amount of fissile material in the core, in this way the chain reaction is not self-sustaining. One can operate such a reactor in continuous mode (i.e. constant power), by supplementing the fission neutrons by neutrons from an external source. In an ADS, these supplementary neutrons are produced by spallation reactions (Fig. 1), during which high-energy protons, coming from a particle accelerator, are impinging on a spallation target made of a heavy metal like lead. In the course of the spallation process, the liquid metal nuclei emit a large number of neutrons whose energy spectrum is made of two parts: a conventional fission spectrum and a high-energy tail up to the energy of the incident proton. At 600 MeV, each impinging proton on a lead target would produce about 10 spallation neutrons.

As already mentioned, the generation of nuclear energy from uranium produces, besides energy, also high-level nuclear waste. The HLW quantity one speaks about depends of the considered fuel cycle. To illustrate this let us consider the various fractions



Fig. 1 Spallation vs. fission.

produced by using one ton of uranium as fuel in a Pressurized Water Reactor (PWR) after four and a half years (corresponding to an average burn up of 50.000 MWd tHM⁻¹):

- 935 kg of uranium;
- 12 kg of plutonium;
- 2.4 kg of Minor Actinides (MA), divided into
 - 1 kg of neptunium;
 - 0.8 kg of americium and
 - 0.6 kg of curium;
- 50 kg of fission products (FP), from which a small part (3.5 kg) is long-lived (and called long-lived fission products or LLFPs).

Partitioning and transmutation

Partitioning is the term used for the chemical processes needed to separate the different elements present in spent nuclear fuel (actinides like uranium and plutonium; minor actinides like neptunium, americium and curium; fission products like cesium and strontium) such that a dedicated treatment can be applied to each stream (Bart, 2021). This is analogous to the concept of recycling in household waste: we separate paper from glass; compostable from plastic, etc.

Transmutation is the process where we subject the minor actinides and/or plutonium, to a flux of fast, high energetic neutrons. Why? Because the fission-to-capture ratio of fast neutrons is significantly larger than for thermal neutron. That is, a fast neutron has a higher probability to induce a fission of the actinides, and hence transmute them into fission products, than the probability that it would be captured by the actinide nucleus and convert it to a nucleus of an even heavier element (what is more likely to happen with slow, thermal neutrons).

Fig. 2 below illustrates the capture and fission cross-sections of ²⁴¹Am as function of energy. We can see clearly that below 1 MeV the capture cross-section is 2–3 orders of magnitude higher than the fission cross-section. The latter becomes predominant beyond a high-energy threshold a little bit below 1 MeV. Thus it is obvious when considering MA's transmutation one needs to conduct it in fast neutron spectrum reactors. This is the first condition for a successful transmutation of MA's. Hence, transmutation requires systems that run on fast neutrons. Plutonium is currently being recycled in LWRs in a few countries, like France (Boullis, 2021; Marin, 2021).

These fast neutron systems cannot be cooled by water since this would slow down the neutrons too much. Therefore, typical coolants for these fast neutron systems are liquid metals like sodium or lead/lead-bismuth.

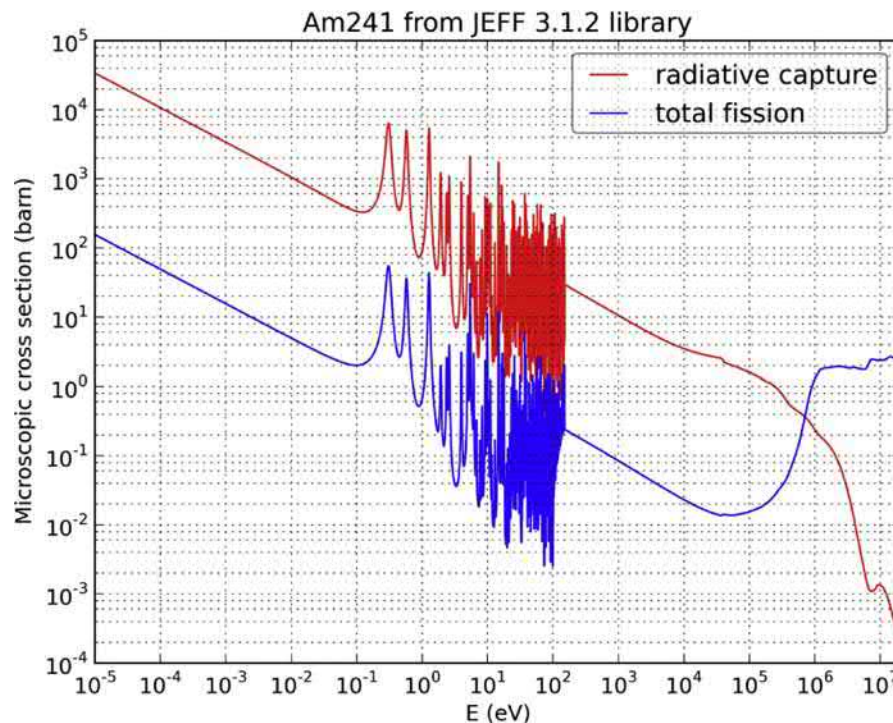


Fig. 2 Neutron spectrum of ²⁴¹Am. Figure is produced using JANIS 3 (Java-based Nuclear Information Software: <http://www.oecd-nea.org/janis>).

Another issue that one has to consider when dealing with transmutation of minor actinides like neptunium, americium and curium in a critical fast neutron reactor is guaranteeing the reactivity safety control of the reactor. The reactivity control being driven by the weighted average value of the effective delayed neutron fraction of the reactor core $\langle \beta_{\text{eff}} \rangle$ and due to the fact that minor actinides are having smaller values of delayed neutrons compared to uranium isotopes (see [Table 1](#)).

The doubling time of the neutron population (and fission power) in the reactor or the period of the reactor being proportional to this delayed neutron fraction:

$$T = \frac{1}{\rho} \left[\frac{\ell}{K_{\text{eff}}} + \sum_{i=1}^M \frac{\beta_i}{\lambda_i} \right]$$

It is obvious that the minor actinides will have a negative effect on the time constants of the system. The higher the amount of minor actinides in the core inventory is, the faster kinetically the reactor becomes. Therefore, with the knowledge and technology of today, we consider only a loading of 2% to maximum 5% of minor actinides inventory in a critical sodium fast reactor core.

This would mean that, in order to transmute the legacy waste of more than 40 years of PWR/BWR operation, numerous fast reactors have to be deployed. In a country that wants to continue with nuclear energy, this transition to a fast reactor fleet is credible (since these reactors have other advantages like less uranium resource use). But in countries that have decided to phase-out nuclear power, the option of building a fast reactor fleet is not acceptable.

For those countries in phase out, accelerator driven systems provide a solution. Since in these systems the time constants are driven by the accelerator and not by a self-sustained chain reaction, they can be loaded with much more minor actinides: up to 40% of the core contents and the remaining 60% core inventory can be made of Pu that should be also dealt with of a MOX made of the Pu and depleted U. Studies ([Van den Eynde et al., 2013](#)) have shown that for all European countries in phase out, seven 400 MWth ADS units would be required to transmute all minor actinides in a time frame of about 100 years.

Partitioning and transmutation (P&T) has three main goals:

- A reduction of the radiological hazard associated with spent fuel by reducing the inventory of minor actinides and plutonium;
- A reduction of the time interval required to reach the radiotoxicity of natural uranium;
- A reduction of the heat load of the HLW packages to be stored in the geological disposal leading to its efficient use.

P&T will not replace geological disposal but it is a means to ease the engineering of such a disposal and to reduce the life-time of the disposal site to times that are maybe apprehensible by humans.

The time scale needed for the radiotoxicity of the waste to drop to the level of natural uranium will be reduced ([Fig. 3](#)) from a “geological” value (500,000 to 1 million years) to a value that is comparable to that of human activities (several hundreds of years) ([OECD/NEA, 2002, 2006](#)).

In Europe, research and development has been built on the so-called four building blocks:

1. Advanced partitioning of PWR/BWR fuel. This is known technology, much of it already at semi-industrial scale.
2. Fabrication of minor actinide loaded fuel.
3. Transmutation in dedicated industrial transmutation facilities
4. Reprocessing of the transmuted fuel.

MYRRHA, Multipurpose hYbrid Research Reactor for High-tech Application ([Aït Abderrahim et al., 2010](#), described further on in this chapter), aims to be a prototype for the conceptual industrial transmuter, EFIT (European Facility for Industrial Transmutation), designed in the IP-EUROTRANS project ([Mansani et al., 2012](#)).

Table 1 Values of delayed neutron fraction β (pcm) for various actinides isotopes ([Plompen et al., 2020](#)).

Isotope	β (pcm) ^a
²³⁵ U	650
²³⁸ U	1480
²³⁸ Pu	120
²³⁹ Pu	210
²⁴⁰ Pu	270
²⁴¹ Pu	490
²⁴² Pu	573
²³⁷ Np	334
²⁴¹ Am	113
²⁴³ Am	208
²⁴² Cm	33
²⁴⁴ Cm	100

^a1 pcm = 0.00001 (one part in 100,000).

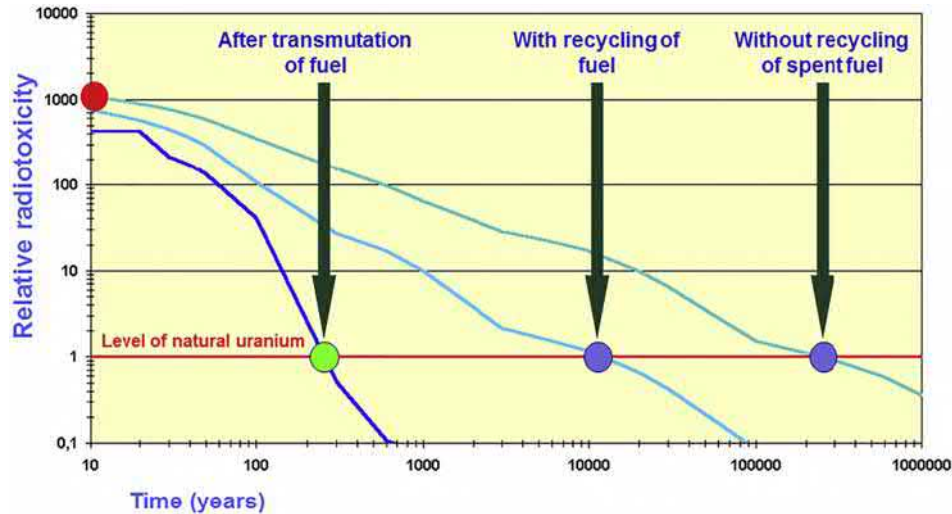


Fig. 3 Radiotoxicity of radioactive waste. Reference: Fig. 171, « Le traitement-recyclage du combustible nucléaire usé », M. Lecomte, CEA, Editions Le Moniteur, 2008, ISBN 978-2-281-11376-1.

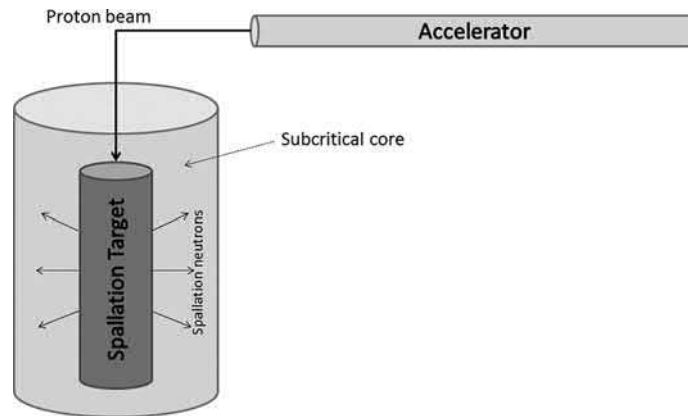


Fig. 4 Concept of an accelerator driven system.

ADS technology

The concept of ADS combines a particle accelerator, a spallation target and a nuclear reactor with a subcritical core (see Fig. 4). In most proposals, a proton accelerator can be found. The protons are injected by the accelerator onto a spallation target, where they produce neutrons for driving the subcritical core. Except for the subcritical state, the core of an ADS is very similar to that of a critical reactor. The different components of an ADS are discussed in detail in the following sections.

Accelerator

The accelerator is the driver of the ADS. The accelerator provides the high-energy protons that are used in the spallation target to create neutrons, which practically “drive” the subcritical core. The accelerator can be either a linear accelerator (LINAC) or a circular accelerator (cyclotron). The accelerators used in an ADS must be high-power accelerators and must have an extremely high reliability and availability. These accelerators are under continuous development and the construction of these machines seems feasible (see also Garcia, 2021).

Particle accelerators delivering a beam on a target can be organized according to the time structure of their beam:

- Direct Current (DC) beam delivery: this beam with no time structure is typically obtained from a DC accelerator;
- Continuous Wave (CW) beam delivery: in order to be accelerated by radiofrequency (RF) fields, the beam has to be subdivided in small packets called bunches and the RF has to be synchronized with the bunch progression through the accelerator. If the resulting bunch train is delivered continuously to the target, the beam delivery is called continuous wave. For providing such a beam, the accelerator must operate in a steady state characterized by constant guiding fields and constant RF frequency;

- Pulsed beam delivery: if the bunch train is interrupted in a periodic way (but with a frequency that is much smaller than the RF frequency) a beam on/off cycle is obtained, leading to pulsed operation. Within the pulse, the bunch structure of the beam remains present.

The choice between a LINAC and a cyclotron depends on many factors. In both cases, it is important to specify the beam characteristics, and more specifically to minimize the amount of beam interrupts. For most ADS the accelerator delivers the particles in continuous mode, although pulsed operation could be used as well (e.g. for testing purposes).

Linear accelerators have the possibility to place the transverse focusing elements (quadrupoles) at frequent intervals along a LINAC, as well as longitudinal focusing, which set a higher limit to the charge per bunch that can be accelerated without significant beam losses. This results in the fact that a LINAC could accelerate a current one to two orders of magnitude higher than a cyclotron, without having problems of beam extraction. The major drawback of a LINAC is its length, which depends on the final beam energy and accelerating gradient. This length of the LINAC is an important factor in the cost of the facility, as most of the accelerator needs to be shielded against radiations.

The base of a cyclotron is the so called “cyclotron resonance”: in a magnetic field, constant in time and perpendicular to the beam orbit, the particle revolution frequency is fixed and independent of the particle energy. The major drawbacks with cyclotrons are the absence of good focusing and the beam extraction. The deflecting system, which guides the beam out of the accelerator, is only permitted to touch a negligible fraction of the beam. The current limit of the cyclotron is given by a design requirement to produce a clean beam at the outer radius of the machine. The maximum proton energy that can be reached by cyclotrons is 1 GeV.

The accelerator must have a high reliability and efficiency. Indeed any beam trip occurring will lead to a reactor shut-down equivalent to an unforeseen SCRAM. As after a SCRAM the operational procedure of the restart of the reactor would require hours if not days this will be very penalizing the ADS operation in terms of availability. Therefore, the proposed operation approach would be to ignore very short beam trips with duration below 3 s (in the case of MYRRHA) as the thermal inertia of the lead-bismuth coolant will be protecting the reactor and the spallation target guide tube from temperature mechanical stress transients. When the accelerator cannot be restarted within 3 s leading to thermal stresses than one has to apply a “SCRAM” restart procedure with progressive ramping up of the current hence the power of the reactor. As such within MYRRHA we are limiting the number of beam trips longer than 3 s to 10 of them for an operating cycle of the reactor of 3 months.

Spallation target

The spallation target is a neutron source providing primary neutrons that are multiplied by the surrounding subcritical core. The primary neutrons are produced by the spallation reaction of heavy-metal target nuclei, bombarded by the high-energy particles generated by the accelerator.

The spallation target is one of the most stressed components of an ADS. The spallation target is designed to generate the maximum amount of neutrons while it must be able to safely evacuate the heat generated by the spallation process. In addition, the mixed proton-neutron irradiation field that typically can be found in the spallation target of an ADS imposes very specific conditions in the design and operability of the target and effects all thermo-mechanical characteristics for these targets. The target of an ADS must be cooled, mostly by gas, heavy water or liquid metal.

There are two options for the spallation target: a solid target or a liquid metal one. Within the operational, lower-power facilities there is a lot of experience with solid targets, especially when combined with an accelerator which delivers a pulsed beam. These solid targets are difficult to cool and suffer from radiation damage. The target should be cooled even after the accelerator is shut off. The path for solid targets for use in an ADS is abandoned in favor for the liquid metal targets.

For higher-power facilities ADS, the use of a liquid metal spallation target is preferred. Both lead as lead-bismuth eutectic (LBE) are primary candidate liquid metals for spallation targets in an ADS. The cooling of a liquid metal spallation target is inherent to the design of the system; there is a constant flow of liquid metal coolant. At the same time, it is possible to achieve higher power densities without having significant irradiation damage to the target. The drawback of a liquid metal target is that the coolant, and especially lead or LBE, can cause erosion or corrosion to the structural materials that are in contact with the coolant.

Design difficulties of liquid metal spallation targets include the following: First, we have the choice of structural materials for the target. The materials must withstand the thermo-mechanical loads on the targets and must be compatible with the liquid metal. Secondly, a reliable interface between the target and the beam guide tube must be designed. This may take the form of a solid beam window or a windowless design. A beam window is exposed to a significant flux of high-energy protons and neutrons at high temperatures and to a very corrosive environment. Therefore, it is very likely that the beam window will have to be replaced periodically because of material damage. The use of a windowless target will alleviate the window embrittlement issue. Nevertheless, when the proton beam will interact with the heavy liquid metal serving as the spallation target material, volatile spallation products will be produced. The later can escape out of the reactor via the proton beam guide line and will be inducing the contamination of the accelerator. Therefore, in a windowless target design a so called “cold window” should be introduced in the last meters of the proton beam line far away from the place where spallation reactions are occurring. As such this “cold window” protecting the accelerator from contamination will be submitted only to proton irradiation and will result in approximately 1.5–2 times lower radiation dose less harmful in comparison with the window target design.

Subcritical core

The subcritical core of an ADS is constructed in such a way that it does not have enough fissile material to reach criticality. Non self-sustaining fission chain reaction is established by the spallation neutrons. The subcritical core of an ADS acts as an amplifier: on average ~ 15 neutrons are produced for each proton.

The energy spectrum of the subcritical core depends on the application of the core. For the transmutation of minor actinides, a fast neutron spectrum has various advantages among them a larger excess of neutrons, a reduced amount of minor actinide production in the core and more fission energy generation per spallation neutron.

The evolution of core composition with time (i.e., with burnup) and temperature feedback effects will have an effect on the multiplication factor of the core, as in critical systems. However, for a subcritical core, the behavior is dominated by the external source and its variation in time. The closer to a critical system, the feedback elements of the core will become more important. There are no final criteria to define an optimal level of subcriticality.

In critical cores, reactivity of the core and the power level of the reactor are adjusted using control rods. With an ADS, it should be possible to control the power level of the subcritical reactor through the external source. When a core is working in “source dominated” mode, the shutdown of the source has an instantaneous effect to reduce power. The inverse effect, a sudden increase of the source has the consequence of an instantaneous increase in the reactor power.

Advantages of an ADS

The main advantages of an accelerator driven system compared to critical reactors are twofold: an ADS allows a greater flexibility with respect to the fuel composition and it has a potentially enhanced safety.

ADS are able to burn fuels that are problematic from the standpoint of critical reactor operation, namely fuels that would degrade neutronic safety characteristics of the critical core to unacceptable levels due to small delayed neutron fractions and short neutron lifetimes. An ADS is capable to cope minor actinide fuel as well as with ^{233}U fuel. Additionally, an ADS allows the use of non-fissile fuel such as thorium without the incorporation of U or Pu (as spike fissile material) in the fresh thorium fuel.

The enhanced safety of an accelerator driven system is due to the fact that once the accelerator is turned off, the system shuts down. When the margin towards criticality is sufficiently large, reactivity-induced transients will never result in a super-critical accident.

Since the accelerator is the driver of the system, the power of the ADS is controlled by the beam current of the accelerator. This feature can be utilized to compensate for the reactivity effect fuel burnup.

In the context of transmutation, another advantage of the ADS is increased core design and fuel management flexibility. The subcriticality also opens opportunities for new reactor concepts, including concepts that are otherwise ruled out by insufficient neutron economy. An ADS allows transmuters to be designed as pure minor actinide burners. This minimizes the fraction of specialized transmuters in the reactor park.

ADS developments in the world

Many ADS concepts have been developed in the world, involving small, medium-size or even large installations ([Kadi and Revol, 2001](#); [Revol, 2001](#)). The first practical attempts to promote accelerators to generate potential neutron sources were made in the late 1940s by E.O. Lawrence in the United States, and W.N. Semenov in the former Soviet Union. During the period 1950s, the MTA (Materials Testing Accelerator) program at Lawrence Livermore investigated in detail the use of accelerators to produce fissionable material. This project was soon abandoned when high-grade uranium ores were discovered in the United States.

A materials production accelerator—the Electronuclear Reactor—was patented in 1960 by Lawrence et al. to provide adequate quantities of material which can only be produced artificially. The targets considered were natural uranium and thorium and the artificially produced materials were ^{239}Pu and ^{233}U respectively. This concept of accelerator breeding was also studied by Russian scientists. A relatively realistic concept of an “Accelerator Driven System” in the present meaning, where safety issues and transmutation of waste play an important role, was also developed in the late 1980s by a research group at Brookhaven National Laboratory.

The Energy Amplifier developed at CERN in the 1990s was devoted to a better use of natural resources and has evolved then to the XADS (eXperimental ADS) developed by Ansaldo Nucleare. At the same time the ADONIS project in Belgium (1995–96) was solely devoted to the production of radioisotopes for the medical industry; ADONIS ([D'hondt et al., 1997](#)) evolved to MYRRHA in 1997; this project still exists and will be detailed in [Section 0](#).

Both XADS and MYRRHA were regrouped in the frame of the EUROTRANS project of the sixth framework program from the European Commission (2005–10). EUROTRANS considered two different but complementary concepts:

- MYRRHA, as smaller scale ADS project for the short term (see below);
- EFIT (for European Facility for Industrial Transmutation), as larger (but still mid-scale) ADS project for the longer term.

EFIT ([Mansani et al., 2012](#)) comprises a large-scale accelerator (800 MeV, 20 mA) coupled to a pure lead target, the subcritical reactor (400 MWth) being also cooled by liquid lead. The optimum burning capacity of such an industrial ADS has been evaluated to be 35 kg MA/TWh leading to a yearly burning capacity, with an availability factor ranging between 80% and 85%, of ~ 100 kg MA/year.

In the meantime (De Bruyn, 2012), the Japanese JAEA was investigating an even larger “commercial-size” installation, with an (1.5 GeV, 20 mA) accelerator coupled to an LBE target, the 800 MWth reactor being also cooled by liquid LBE.

At present time, those large-scale facilities have been put on-hold and attention has been focused on the construction and the operation of smaller installations. A zero-power installation, GUINEVERE, has been successfully operated at SCK CEN, since 2011 in the framework of various research projects of the European Commission.

MYRRHA

Introduction

MYRRHA (Ait Abderrahim et al., 2010), the flexible experimental accelerator-driven system in development at SCK•CEN, will demonstrate the ADS full concept by coupling the three components (accelerator, spallation target and subcritical reactor) at reasonable power level to allow operation feedback, scalable to an industrial demonstrator and allow the study of efficient transmutation of high-level nuclear waste. The MYRRHA-facility is conceived as a flexible irradiation facility, able to work in both subcritical and critical modes. Both modes of operation have their specific energy and flux distributions, which permit, besides transmutation experiments, a wide range of applications, from fuel developments for innovative reactor systems, material developments for GEN IV systems, material developments for fusion reactors, to radioisotope production for medical and industrial applications. Since MYRRHA is based on the heavy liquid metal technology, lead-bismuth eutectic, it will be able to significantly contribute to the development of Lead Fast Reactor (LFR) Technology (see Alemberti, 2021) and in critical mode, MYRRHA will play the role of European Technology Pilot Plant in the roadmap for LFR.

The MYRRHA accelerator

The MYRRHA accelerator is to provide a proton beam with energy of 600 MeV and a maximum current of 4 mA, which will be delivered to the core in continuous wave mode. The beam is delivered to the core from above through a beam window.

High accelerator availability, that is, a long Mean Time Between Failure (MTBF), is commonly obtained by a combination of over-design and redundancy, as well as fault tolerance. Fault tolerance will allow the accelerator to recover the beam within a beam trip (no beam is delivered) duration tolerance after failure of a single component. In the MYRRHA case, the beam trip duration tolerance is 3 s. Within an operational cycle of MYRRHA reactor (90 days) the number of allowed beam trips exceeding 3 s must remain under 10, shorter beam trips are allowed without limitations (Rimpault et al., 2013). The combination of redundancy and fault tolerance should allow obtaining a MTBF value in excess of 250 h.

At present proton accelerators with megawatt level beam power in CW mode only exist in two basic concepts: sector-focused cyclotrons and linear accelerators. Cyclotrons are an attractive option with respect to construction costs, but they do not have any modularity which means that a fault tolerance scheme cannot be implemented. Also, an upgrade of its beam energy is not a realistic option. A linear accelerator, especially if made superconducting, has the potential for implementing a fault tolerance scheme and offers a high modularity, resulting in the possibility to recover the beam within a short time and increasing the beam energy.

The MYRRHA reactor

MYRRHA consists of a pool-type ADS with the ability to operate in critical mode cooled by liquid lead-bismuth eutectic. Consequently, all the primary coolant systems are housed within the reactor vessel. The reactor is located in the reactor pit (Fig. 5) which features a liner able to serve as secondary containment in case of a reactor vessel leakage or break. The reactor cover closes the reactor vessel and supports all the components. The reactor core (Fig. 6) is included in the core barrel and consists of 211 positions. Such positions are filled, among others, with fuel assemblies, control rods, scram rods, the spallation window, in-pile sections, reflector assemblies, instrumentation and surveillance capsules. From these core positions 55 are accessible from the cover and are called Multifunctional Channels (MFC).

The multi-functional channels (MFC) include control rods, scram rods, spallation window target, in-pile sections, instrumentation and surveillance capsules; all installed from the top. Because of the structures above the core, the fuel (driver MOX fuel assemblies as well as MA fuel assemblies) and reflector assemblies have to be loaded from underneath. Thanks to the buoyancy, the fuel and reflector assemblies do not need locking devices to hold them on the core support structure. However, in order to avoid any small position changes of the fuel assemblies inside the core during operation, a core restraint system fixes the radial position of the fuel assemblies.

The MYRRHA reactor core is made of MOX “driver” fuel assemblies with fast reactor typical plutonium content ranging between 30% and 35%. The MYRRHA core is designed to accept up to six fuel assemblies heavily loaded with MA (orange positions in Fig. 6). This MA fuel type is for industrial ADS, such as EFIT mentioned in Section 0; it is composed of 50% Pu, 46% Am, 2% Np and 2% Cm. Cm will not be included in the MA fuel for practical reasons.

Two in-vessel fuel-handling machines, installed permanently in the reactor, handle the loading and the unloading of the fuel assemblies to the in-vessel fuel storages, which are integrated into the diaphragm, the internal reactor structure separating the cold zone of the reactor (down comer) from the hot zone (upper comer). Due to the chosen manipulator concept, the most compact design is obtained by using two independent machines, each covering one half of the core and serving half of the in-vessel fuel

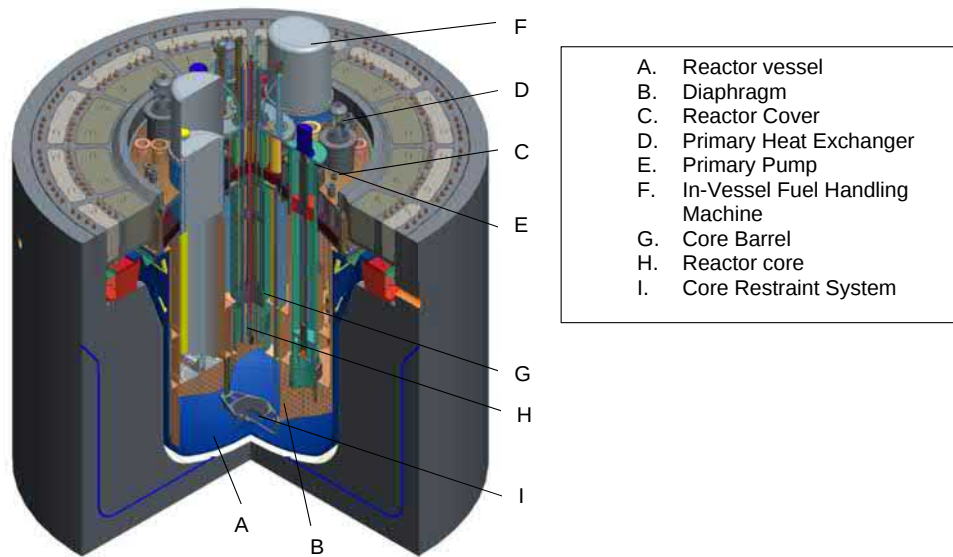


Fig. 5 Overview of the MYRRHA reactor.

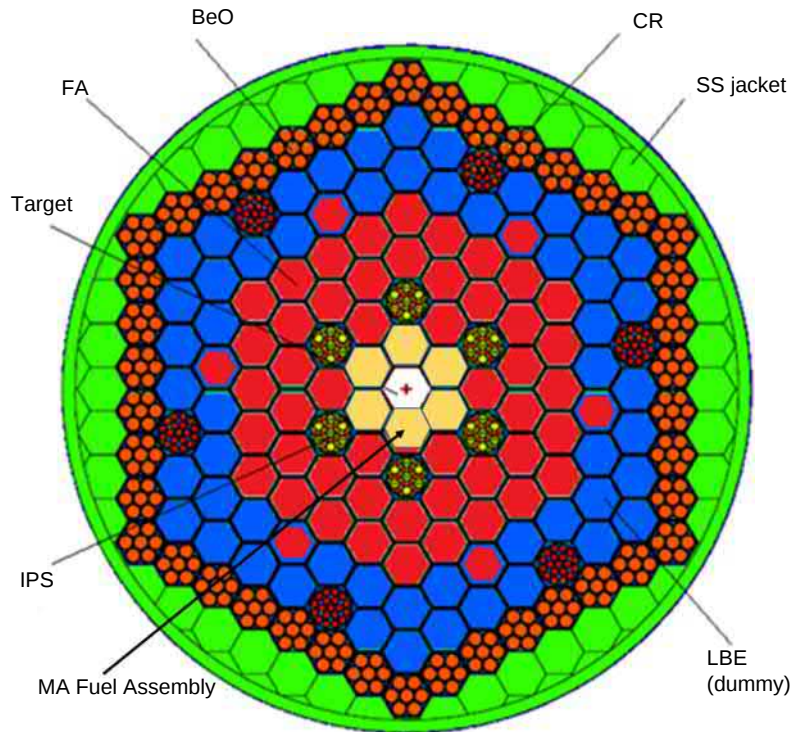


Fig. 6 Horizontal cut in the MYRRHA core.

storage positions. The diaphragm separates the cold, high pressure LBE from the hot, low pressure LBE and contains the in-vessel fuel storages and the pump casings.

The diaphragm is connected, together with the cover, to the reactor vessel. The diaphragm consists of two horizontal plates connected to each other by vertical shells allowing some components to reach the lower plenum. Due to this design, several volumes exist in-between the two horizontal plates. Two of these volumes house part of the pumps and the heat exchangers. Each pump casing lodges one pump and two heat exchangers. Four volumes are dedicated for the in-vessel fuel storage. These storages are cooled during operation mainly by the forced circulation of LBE imposed by the pumps, while during shutdown cooling is ensured by natural circulation.

The four primary heat exchangers extract the power of the primary system exchanging it with the secondary system. The head imposed by the pumps circulates the hot LBE from the hot plenum into the heat exchangers. After being cooled, the LBE flows into the pump casings and then through the pumps themselves. This concept mitigates the LBE corrosion by reducing the temperature of the fluid inside of the pumps. In particular, these are sized in order to be able to pump the LBE through the core balancing also the head losses of all the other sections of the primary system. The number of heat exchangers and pumps is optimized taking into account the space utilization inside the reactor vessel and the symmetry of the coolant flow. This leads to a configuration with two heat exchangers and one pump on each of the two sides of the reactor.

R&D in support to design and licensing

SCK CEN conducts since the beginning of the program an ambitious R&D program to address the main design and licensing challenges. In this frame SCK CEN has constructed and commissioned various LBE test facilities which are used for qualification of the key materials and components of MYRRHA. These facilities can also be used for the development of materials and components for fast reactors of all power ranges, including SMR type systems, working with LBE or Lead as coolant.

The main challenges of the MYRRHA development are in particular related to the use of liquid Lead-Bismuth Eutectic as reactor coolant. Consequently various LBE test facilities for research on

- heavy liquid metal chemistry and conditioning,
- heavy liquid metal corrosion of materials,
- thermal hydraulics in heavy liquid metals,
- instrumentation in heavy liquid metals, and
- testing of components in heavy liquid metals

have been constructed and commissioned (Fernandez et al., 2019).

For the components and thermal hydraulics program the *E*-SCAPE facility, European SCALe Pool Experiment, a thermal hydraulic 1/6-scale model of the MYRRHA reactor, has been commissioned in spring 2017. The main objective of this installation (Fig. 7) is the study of the thermal hydraulic behavior of liquid metal in a complex pool geometry. The characterization of the pool thermal hydraulic phenomena is needed for qualification of computational fluid dynamics code and other system tools used for the safety assessment of a reactor system.

The COMPLIT, COMPOnents LOOp Testing, facility is a large-scale, closed loop isothermal experimental loop in operation since 2014, used for hydraulic experiments of full-scale reactor components in flowing LBE. The goal of COMPLIT is to simulate a full-scale hydraulic flow path through the MYRRHA core allowing the characterization of hydraulic and hydrodynamic behavior of full-scale MYRRHA core components such as the fuel assembly, the spallation target, the control and safety rod systems and other in-pile instrumentation, systems and experiments.

HEXACOM, Heat EXchanger at COMplot, steam loop is a two-phase water-steam cooling circuit that provides temperatures and flow conditions representative of the MYRRHA Secondary Cooling System. The main objectives of the facility is to investigate the heat transfer performance of different primary HX configurations within the operational ranges relevant to MYRRHA, to develop and validate heat transfer correlations specific to bayonet tube applications in flowing LBE, to improve the level of knowledge on the phenomena that occur in the steam/water side of double wall bayonet tube heat exchangers that operate at low pressure,



Fig. 7 Overview of the *E*-SCAPE facility.

to obtain experimental data suitable for model development and/or the validation and verification of system codes. Since HEXA-COM is a small version of the MYRRHA secondary and tertiary system, additional phenomena, typical for steam/water, natural circulation cooling and anti-freezing strategies can be studied in support of the design of these systems in MYRRHA.

Finally, RHAPTER, the Remote HANDling Proof-of-principle TEST Rig, in operation since 2011, is designed to test and validate mechanical components submerged in LBE.

The LBE chemistry and conditioning program uses the MEXICO, Mass EXchanger In Continuous Operation, loop to test different oxygen control systems, such as the gas phase, solid oxide phase and electrochemical oxygen pumping systems for regulating dissolved oxygen in liquid lead-bismuth eutectic. It is also used to evaluate the efficiency and expected life time of filtration systems for purifying the LBE of oxides and possible impurities from the liquid metal flow while minimizing the created heavy metal waste stream.

HELIOS, HEavy Liquid metal Oxygen conditioning System, is a LBE conditioning and storage setup which is used to investigate LBE conditioning schemes. It can also serve to study calamity mitigation strategies after a possible steam ingress due to a tube rupture in a heat exchanger or exposure to air.

The main infrastructure for materials research is CRAFT, Corrosion Research for Advanced Fast reactor Technology. It is an installation for long term corrosion experiments on MYRRHA candidate materials in liquid LBE. The loop type installation operates at representative conditions of temperatures, LBE velocities and dissolved oxygen concentrations of MYRRHA.

LIMETS, Liquid METals Test Stands, are experimental set-ups designed for mechanical testing of materials in a stagnant LBE environment in order to investigate mechanisms and kinetics of material-liquid metal interactions that influence the mechanical properties of materials.

Conclusions

Partitioning and transmutation is an attractive solution to reduce the amount of long-lived radioactive waste legacy and to reach the goal of sustainable nuclear energy systems. P&T could address the high-level radioactive waste issue and initiates the strategy towards a more resource-efficient nuclear energy system in the future. P&T will not replace the need for appropriate geological disposal of high-level waste, but it will reduce the long-term radiotoxicity of the waste significantly and thus reduce the burden of HLW on the safety and required volume of geological disposal increasing highly its acceptability by the public.

All transmutation strategies with closed fuel cycles could achieve high reductions in the inventory of actinides and the radiotoxicity of the nuclear waste by making use of fast neutron spectrum systems such as fast critical reactors or accelerator driven systems.

Nevertheless, the sub-critical ADS offers more degrees of freedom with respect to the core design and fuel management as well as flexibility in the policy towards the use of nuclear energy.

With the extensive development of accelerator technology, it appears today that proton beam powers of 10–100 MW with the needed reliability for operating ADS systems are within the reach in the coming decade using the LINAC technology.

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Research Reactor Conversion to Low Enriched Uranium (LEU) Fuel

John G. Stevens, Nuclear Science and Engineering Division, Argonne National Laboratory, Lemont, IL, United States

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Introduction

While many think of nuclear reactors only as a source of power for electricity generation or propulsion, there is a distinct class of hundreds of remarkable reactor systems that use the sustained fission chain reaction to produce neutrons and gamma radiation for scientific experiments and/or to transmute target isotopes into medical or industrial isotopes (IAEA, 2014).

The range of science and isotope production research reactors provide is not just broad, but touches our lives in many ways every day (DOE, 2020; Barradas, 2021). Small reactors may focus on education and training, but they also allow tracking of environmental pollution or industrial quality control for industries like insect-resistant lumber or mold-resistant drywall. Larger reactors allow experiments in the core to study how neutron and gamma radiation damage materials, testing new fuels or core components, and assuring safety of power reactors and facilitating space applications of materials. In addition to making isotopes that detect and fight cancer, research reactors also produce large semiconductor materials vital to hybrid electric cars. And since neutrons do not have any electric charge but do have magnetic spin, scientists in many fields can use beams of neutrons delivered from the reactor core to study details of materials and biological processes in unique ways complementary to X-rays.

The three core images of Fig. 1 hint at the tremendous variety of research and test reactors. The RA-6 research reactor in Argentina uses fuel in the form of flat plates grouped into “elements” (i.e., a set of plates secured in a support structure) that are placed on a grid plate at the bottom of an open pool. The MARIA reactor in Poland uses fuel elements in which each plate is curved so that the elements are a set of concentric circles. The MARIA elements are sealed into individual cooling loops, and it is the loops that are loaded onto the grid plate at the bottom of a pool. The KUCA critical facility dry cores use small “coupons” of uranium sealed in epoxy that are stacked as combinations of fuel, polyethylene plastic, and spacers or test materials.

International nuclear security requires keeping weapons-usable material out of the wrong hands. Many of the research reactors in the world have used fuel composed of High Enriched Uranium (HEU, U-235 enrichment $\geq 20\%$) (NAS, 2016). The HEU fuel provided the density of the fissile isotope U-235 needed with the types of fuels available at the time the systems were designed and deployed. However, as described by Glaser (2006), the risk of misuse of fuel increases with enrichment. National and international norms treat uranium with an enrichment of the isotope U-235 $\geq 20\%$ as a security risk that should be minimized, and to the extent possible eliminated, for civilian reactors and medical isotope production.

The process of changing the fuel used by the reactor from HEU to Low Enriched Uranium (LEU, U-235 enrichment $< 20\%$) is defined as reactor conversion. The US technical effort to convert began in 1978 as the Reduced Enrichment for Research and Test Reactors (RERTR) program. From the beginning, the effort organized a global community of reactor operators, fabricators, and their governments.

As described at the first RERTR International Meeting in 1978 (Travelli, 1978), the program has been focused upon preservation of reactor mission capability as a central tenet of the conversion program. Indeed, the reason the reactor conversion program continues forty years later is because the community continues to collaborate on projects to preserve mission capability, rather than forcing degradation of performance. The RERTR program management within the Department of Energy (DOE) matured to the Global Threat Reduction Initiative (GTRI, 2004–15) and now the National Nuclear Security Administration (NNSA) Office of Material Management and Minimization (M3, 2015–present).

As indicated in Fig. 2, there has been a great deal of success converting reactors around the world. A total of 71 reactors converted from HEU fuel to LEU fuel in the period 1978–2019, and an additional 31 reactors that used HEU fuel have been confirmed permanently shutdown. Fig. 2 highlights the fact that 24 high impact reactors with thermal neutron fluxes in experimental positions higher than 10^{14} n/cm²/s have been converted without significant reduction of the maximum available fluxes.

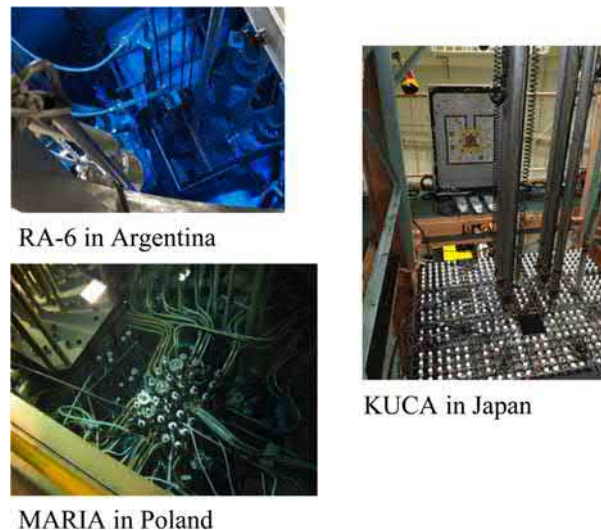
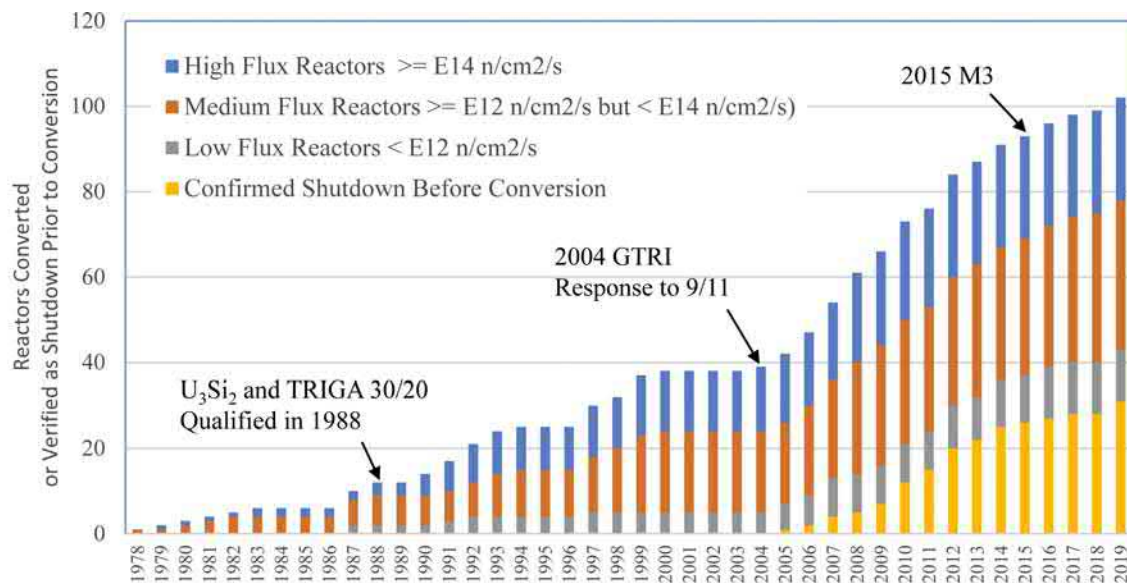


Fig. 1 Examples of research reactor variety. Source: John Stevens, Argonne National Laboratory.



U_3Si_2 and TRIGA 30/20 fuels were key higher-density fuels to allow conversion from HEU to LEU

GTRI: Global Threat Reduction Initiative organization within NNSA

M3: Material Management and Minimization organization within NNSA

NNSA: National Nuclear Security Administration

Fig. 2 Global reactor conversion pace. Source: John Stevens, Argonne National Laboratory.

Challenges and Approaches to Reactor Conversion

Converting a reactor while maintaining experimental performance is not an easy task, but it is an exciting technical challenge.

Nearly all of the fissions in a uranium fueled reactor occur in the isotope U-235. So if the enrichment is reduced from 93% U-235 to 19.75% U-235, the density of uranium in an unchanged geometry would need to be increased by at least a factor of 4.7. In fact, most uranium that is not U-235 is the isotope U-238, which is more likely to absorb a neutron without fission than to fission. That means U-238 acts as a “parasitic” absorber, reducing the number of neutrons available for fissions and experiments, so the overall LEU uranium density would need to increase by more than 4.7 if the geometry is not changed. Because of this, the search for fuels with different uranium alloys that have higher density than the original HEU fuels has been a major focus of the reactor conversion effort.

Fortunately, the many minds around the world that have worked on reactor conversion found ways to apply physics of core modifications in addition to uranium density to preserve experimental performance. For instance, changes of the fuel geometry, reflectors, and mix of fresh and partially-used fuels within an existing reactor adjust the core volume and the energy of the neutrons that interact with the uranium and experiments.

Changing the geometry of fuel and/or core does mean that the cooling of the fuel is changed. Adequate cooling is the main constraint for how small a core can be made and how intense local experimental fluxes can be. So the computational methods that predict the details of where power is produced in the reactor core over time and the methods that predict how the associated heat is removed have been continuously improved over the decades, with experiments to verify that the methods predict details of reality well.

Changing the fuel materials and geometry also requires the manufacturing of fuel to be improved over time, with enhanced precision, predictability and quality verification methods.

No two conversions have been the same since each research reactor is unique in design, mission, and/or operational cadence. Global collaboration through the International Atomic Energy Agency (IAEA), bilateral, and multilateral teams has always been vital to address both technical challenges and the logistics/legalities of the multinational nuclear fuel cycle (particularly the transportation aspects of the fresh fuel supply chain and spent fuel repatriation).

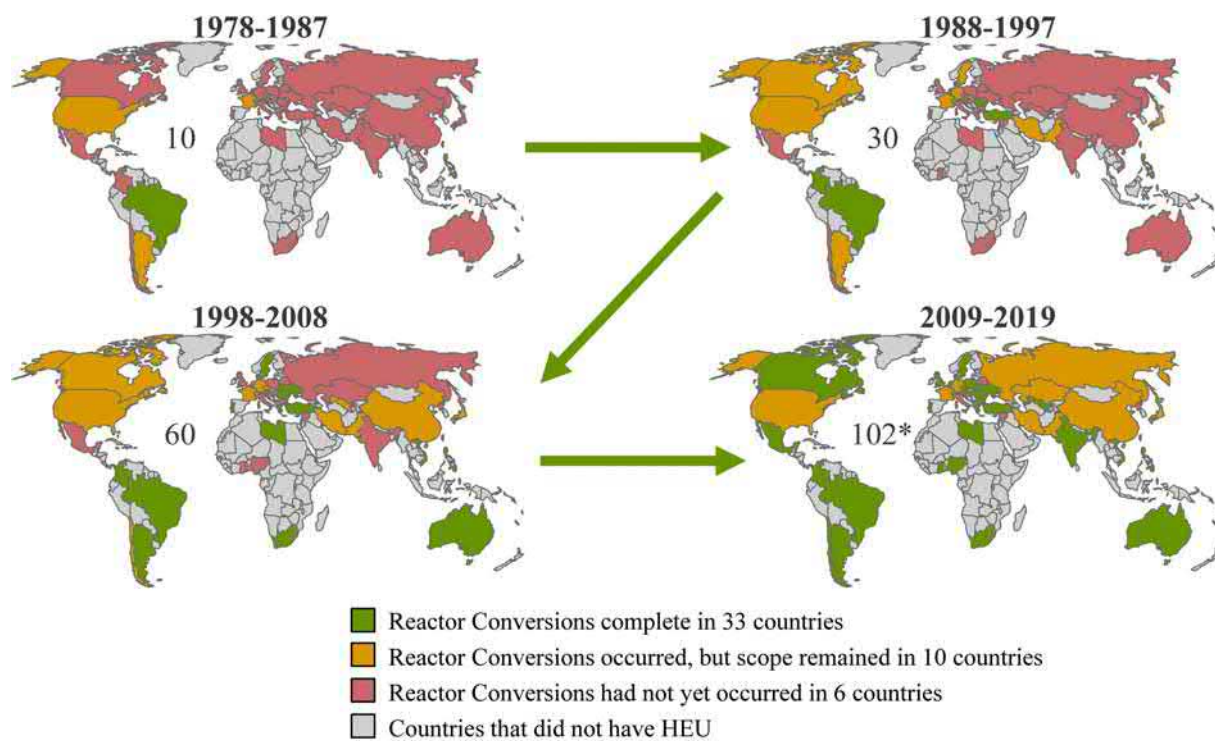
Reactor Conversion Historical Overview by the Decades

Fig. 3 restates the progress of global reactor conversion in terms of geography and time. It is clear that international cooperation existed from the start, but then accelerated as improvements to fuels, methods, and collaboration structures all improved with time.

1978–87: Pragmatism

Ten conversions were accomplished in seven countries during the first decade of the program by:

- Conversion to UAl_x and U_3O_8 fuels fabricated with higher uranium density than had originally been deployed. This approach could only suffice for a few reactors, but showed the importance of fabrication improvements.



* Cumulative number of conversions to LEU (71) + shutdowns confirmed prior to shutdown (31) in the years 1978-2019

Fig. 3 Global reactor conversion progress. Source: John Stevens, Argonne National Laboratory.

- Conversion of HEU plate-type cores to low-density LEU TRIGA UZrH rod clusters. This was a clear demonstration of how physics-based modifications beyond simple density would allow conversion.

The US Nuclear Regulatory Commission (NRC) issued regulation 10 CFR 50.64 in 1986, formalizing the expectation of the use of LEU fuels in civilian reactors once the ability to preserve acceptable experimental performance is demonstrated with an “acceptable fuel,” and funding to perform the conversion is provided.

LEU TRIGA 20/20 & 30/20 UZrH fuel was qualified by the US NRC in 1987 (where the 20/20 indicates 20 weight percent uranium in the UZrH at 20% enrichment U-235, 30/20 indicates 30 weight percent uranium in the UZrH). These higher density UZrH fuels had reactivity similar to the prior TRIGA HEU fuels to facilitate TRIGA reactor conversions.

1988–97: Momentum and Improved Methods to Open Design Space

Twenty conversions were accomplished in 12 countries during the second decade of the program.

The most significant breakthroughs included:

- U_3Si_2 fuel was qualified by the US NRC in 1988.
- The “Schumer Amendment,” Section 903 of the Energy Policy Act of 1992 formalized restrictions on exports of HEU from the US related to the expectation to convert to LEU.
- The US NRC published NUREG 1537 “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors” in 1996, specifically addressing the scope of safety analysis expected for a reactor conversion.
- U_3Si_2 and TRIGA LEU 20/20 and 30/20 fuels were deployed to a variety of reactors.

Many methodological improvements for design, operational, and safety analyses were pursued during the second decade in order to reduce the penalties of overly “conservative” approaches. The improved methods evolved to treat physics detail, replacing linearizations, assumptions of separability, and correlations wherever possible.

- Neutronics
 - o Improved Cross-Sections for Deterministic Steady State and Depletion Calculations
 - o Monte Carlo methods (e.g., MCNP) to replace steady state deterministic methods 1990–now (Fig. 4 illustrates MCNP models for two very different research reactor cores)
 - o Monte Carlo-based Depletion (1998–now)
- Thermal-Hydraulics methods have been extended to treat multiple dimensions and to apply less conservative safety margin correlations that have been shown more appropriate to distinct research reactor flow conditions.
 - o PLTEMP steady state analyses (1985–now)
 - o PARET transients (1982–now)
 - o RELAP system transients (1996–now)
- Fuel Performance Modeling: both Correlation-based and Finite-Element methods evolved as irradiation test data were collected for each of the new fuel systems.
- Linkages & Automations between physics domains and levels of detail have been established. For instance, historic separability assumptions between “Pin Cell” and the core have been replaced by more Full-Core and Full-System analyses in order to avoid conservatism or potential errors of a priori judgements about how the power distributions and cooling interact.

1998–2008: Celebrating Successes, Sobered by 9/11

Twenty-six conversions were accomplished in 19 countries during the third decade of the program. An additional four shutdowns-prior-to-conversion were confirmed in three countries.

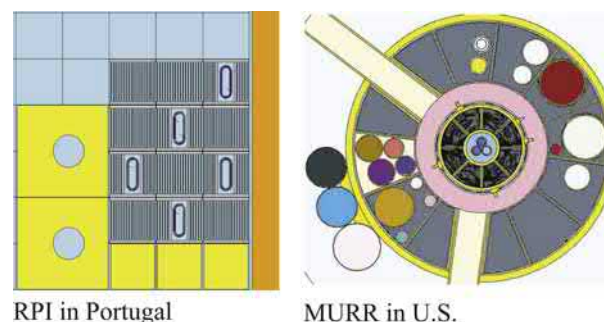


Fig. 4 Examples of research reactor core models. Source: John Stevens, Argonne National Laboratory.

The pace of conversions was accelerated in response to the terrorist attacks of 9/11 in 2001. Eight conversions had occurred in the four years 1998–2000. The Global Threat Reduction Initiative (GTRI) was formed in 2004. The associated budget increases, improved project management, and global cooperation after GTRI was formed led to eighteen conversions in the 5 years (2004–08).

Collaboration between East and West on conversion of reactors exported by Russia and China was a major breakthrough after 9/11. While there had been productive engagement on nonproliferation matters for years, new frameworks were established in the third decade. The Russian Research Reactor Fuel Return (RRRFR) program was launched in 2001 to repatriate Russian origin HEU, analogous to the US Foreign Research Reactor Spent Nuclear Fuel Acceptance Program that had begun in 1996.

Synergistic Working Groups for several “sets of similar reactors” were established to accelerate conversion projects and improve efficiency of supply. The TRIGA conversions (both original TRIGA and Plate-Type to TRIGA) had shown the benefits of the working group approach.

The Miniature Neutron Source Reactors (MNSR) exported by China to five countries were identified as a clear “set” with single supplier and very similar technical issues. While each MNSR has a distinct regulator, political, and logistical matters, those project aspects would benefit from the creativity and “lessons learned” of their peers. The MNSR working group approach was vital due to the nuclear political complexities of United States and European technical collaboration with some of the countries involved (China, Ghana, Nigeria, Pakistan, Syria, and Iran). An IAEA Coordinated Research Project for MNSR conversions was established in 2006 that helped lead to several subsequent conversions (2016, 2018, and 2019 in China, Ghana, and Nigeria).

Russian designed reactors that used IRT fuel (concentric extruded tubes of fuel) in Russia and abroad were another group able to benefit from common IRT-3M or IRT-4M fuel qualification efforts, and analytical collaborations. The Czech VR-1 was converted in 2006. The Libyan IRT-1 critical facility and reactor were converted in 2006 and 2007. The Uzbek VVR-SM was converted in 2008. The Czech Rez reactor converted in 2010.

Two large and enduring working group structures were established to address the most challenging Western research reactor conversion projects, those reactors that could not use any prior qualified LEU fuel to convert while maintaining their critical missions.

- US High Performance Research Reactors (USHPRRs): The MITR at Massachusetts Institute of Technology in Cambridge, Massachusetts; the MURR at University of Missouri in Columbia, Missouri; the NBSR at the National Institute of Standards and Testing (NIST) in Maryland; the ATR at the Idaho National Laboratory (INL); and the HFIR at Oak Ridge National Laboratory (ORNL) in Tennessee. (The ATRC critical facility will also be converted to remain appropriate for the ATR reactor programs). The NNSA USHPRR sub-program that began in 2005 has assembled expertise from nine national laboratories and sites (Argonne, Brookhaven, Idaho, Los Alamos, Oak Ridge, Pacific Northwest, Sandia, and Savannah River national laboratories and the Y-12 national security complex) as well as the reactor operating institutions, fuel fabricator BWXT, and the US NRC.
- European High Flux Reactors (EUHFR): the BR2 at SCK CEN in Mol, Belgium; the RHF at Institut Laue-Langevin (ILL) in Grenoble, France; and the JHR under construction at the French Alternative Energies and Atomic Energy Commission (CEA) in Cadarache, France; and the fuel fabricator Framatome CERCA. The US technical effort to support the EUHFR conversion group that began in 2008 has always involved a team of Argonne and Idaho National Laboratories, under the International Reactor Conversion sub-program at NNSA. The Y-12 national security complex has been instrumental for fresh uranium supply and transport. Savannah River National Laboratory provides strategic assistance.

2009–19: Tackling Ongoing Challenges Together

Fifteen conversions were accomplished in 12 countries during the fourth decade of the program. An additional 27 shutdowns-prior-to-conversion were confirmed in 13 countries.

The decade saw the fruition of several projects that had taken more than a decade due to the combination of fuel qualification and fabrication improvements, newly optimized designs and the associated safety analyses, and complex supply chains and transportation logistics:

- The global TRIGA Conversion Scope was completed—the first class of reactor conversions to be successfully completed.
- The Russian-domestic ARGUS reactor converted.
- Three of the seven MNSR reactors converted, with the technical path for the remaining four well defined (if not the political/financial path).
- The VVR-K in Kazakhstan converted, and the success of that multinational project has facilitated the IVG project expected to be completed in 2021.

An impressive span of multilateral projects demonstrated continued resolve to address the full range of challenges for remaining conversions with creative persistence. The tradition of operators and nations that successfully converted helping their peers—or even competitors—to successfully convert continued. For instance, the SLOWPOKE reactor at Polytechnique Montreal in Canada converted in 1997, but their experts were vital to the success of the JM-1 SLOWPOKE conversion in Jamaica in 2015. Similarly, a number of the technical experts helping to implement the Arak Modernization Project of the Joint Comprehensive Plan of Action (JCPOA) “Iran Agreement” leveraged reactor conversion expertise in reactor design, fuel qualification, and multinational project implementations to modernize the Iranian IR-40 reactor as a system with much higher experimental performance, but much lower risk of proliferation.

The Continuing Effort to Achieve High Performance With Low Risk

The decade before us will see some of the most exciting reactor conversions in terms of both the technical accomplishments and the impact toward minimization of HEU in civilian use.

In 2021 and 2022 the IVG in Kazakhstan and the KUCA in Japan should both convert after successfully implementing reactor-specific LEU fuel qualification.

Table 1 indicates the variety of research reactor fuels used in reactors that have converted or that are under development and qualification for conversion of the remaining reactors. The fuel qualification and fabrication campaigns described in DOE (2020) will allow the USHPRR and EUHFR reactors to convert in the 2026–34 timespan.

The global team does also continue to develop and apply higher-fidelity multiphysics methods for design and safety analysis. The combination of high density fuels, supply chain, methods, expertise, and implementation skills that the reactor conversion program has developed and deployed will surely be leveraged to assure high experimental performance, safety, and low risk of proliferation for the new research reactors of the future.

Acknowledgement

As noted above, the reactor conversion effort has always been collaborative. The technical “giants” that paved the way to conversion include Armando Travelli, Jim Snelgrove, Jim Matos, Bill Woodruff, Pablo Adelfang, Jordi Roglans, and many more. The IAEA has been an important partner to the 43 countries in which conversions have occurred. The community of technical and policy experts have met at 40 RERTIR International Meetings hosted by 21 countries. We are grateful that those that have converted in the past continue to assist the global effort to complete the remaining conversions. Argonne’s reactor conversion work is sponsored by the US Department of Energy, Office of Material Management and Minimization in the US National Nuclear Security Administration Office of Defense Nuclear Nonproliferation under Contract DE-AC02-06CH11357.

Table 1 Uranium density of research reactor fuels qualified and under qualification.

	Uranium loading density (gU/cm ³)			
	1978 (Travelli, 1978)	Now (Travelli, 1995)	In Progress	
TRIGA				
U ₂ ZrH pins	0.5	2.1	–	30/20 ^a qualified in NUREG-1282 (1987)
UO ₂ pins	9.3	–	–	Not fabricable for plate geometry
UAl _x -Al dispersion	1.7	2.3	–	HEU fuel in use ^b at MITR, MURR, ATR, BR2, RHF
U ₃ O ₈ -Al dispersion	1.3	3.2	–	HEU fuel in use ^b at NBSR and HFIR
U ₃ Si-Al dispersion	–	6	–	Only suitable for limited irradiation conditions
U ₃ Si ₂ -Al dispersion	–	4.8	–	Qualified fuel in NUREG-1313 (1988), but at much lower power density than USHPRR and EUHFR
	–	–	4.8–5.6	HERACLES/LEU FOREVER at EUHFR Conditions (experiment irradiated, awaiting PIE) (Valance et al. (2020))
	–	–	5.3	SCK COBRA EUHFR Conditions including Gd integral burnable absorber (Kalcheva et al., 2018)
	–	–	4.8	USHPRR HFIR-Relevant Conditions (experiment being fabricated) (Wilson et al., 2019)
UO ₂ -Al Dispersion	–	2.5	–	WWR-M2 Qualified (2002) (Konoplev et al., 2002)
	–	2.8	–	VVR-KN Conversion (2013) Coextruded tubes (Shaimerdenov et al., 2018)
	–	3.0	–	IRT-4M Qualified (2002) Coextruded tubes (Chernyshov et al., 2002)
UMo-Al dispersion	–	5.4	–	IRT-3M Qualified (2016) Coextruded tubes (Izhutov et al., 2016)
	–	–	8	KJRR LTA (2 LTAs irradiated, PIE in progress) (Yim et al., 2017)
	–	–	8	HERACLES/US (Van den Berghe and Lemoine, 2014; Valance et al., 2020) (EUHFR experiments irradiated, PIE in progress)
UMo Monolithic	–	–	15.3	USHPRR HIP Clad Process (in qualification phase of experiments) (Wilson et al., 2019; Rabin et al., 2017; Woolstenhulme et al., 2016; Wight et al., 2018)
	–	–	15.3	HERACLES C2TWP Clad Process (in development, miniplates irradiated) (Valance et al., 2020; Stepnik et al., 2014)

^aTRIGA 30/20 fuel name denotes 30 weight percent uranium in U₂ZrH at 20% enrichment U-235

^bThe USHPRR and EUHFR have not yet converted. The densities of UAl_x and U₃O₈ indicate the maximum qualified uranium density of those fuel forms.

See Also: Non-Proliferation Aspects of Nuclear Energy; Nuclear Criticality Safety.

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Harnessing Nuclear Radiation

Alan E. Waltar*, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

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Glossary

Activation Analysis A method to identify an unknown material by irradiating it with neutrons and measuring the characteristic radiation from the reaction products.

Alpha Particle A positively charged particle emitted by certain radioactive material, made up of two neutrons and two protons (a helium nucleus).

Beta Particle A negatively charged particle (a high-speed electron) emitted in certain types of radioactive decay. Some types of these particles have a positive charge and are called positrons.

Deuterium A nonradioactive isotope of hydrogen having one proton and one neutron in its nucleus.

Fission The process of splitting a heavy atom into two or more lighter atoms, usually upon absorbing a neutron.

Fusion Fusing two light nuclei into one heavier nuclei.

Gamma Ray A high-energy, short-wavelength electromagnetic radiation emitted in the radioactive decay of many nuclides. Gamma rays originate from within the nucleus.

Ionizing Radiation Radiation capable of causing ionization of the atoms through which it travels.

Irradiation The result of nuclear radiation impinging upon a material.

Isotope An atom of the same element but with a different mass number. Isotopes of a given element have the same number of protons but a different number of neutrons in their nuclei.

MeV Million electron volts = amount of kinetic energy gained by an electron that is accelerated, in a vacuum, across an electric potential difference of one million volts. $1 \text{ MeV} = 1.602 \times 10^{-13} \text{ J}$.

Neutron A basic particle residing in the nucleus of an atom that is electrically neutral. It has a mass approximately equal to a proton.

Nuclear radiation The process in which an unstable atomic nucleus loses energy by emitting nuclear radiation – usually gamma-rays, beta particles or alpha particles.

Nuclide A species of atom characterized by the number of protons, number of neutrons, and energy content of its nucleus.

Photon A quantum of electromagnetic radiation of energy $h\nu$ (h being Planck's constant and ν the frequency of the radiation).

*Retired.

RHU Radioisotope heater unit.

RTG Radioisotope thermoelectric generator.

Sievert The international unit used for measuring effective radiation dose: the health effect of ionizing radiation on the human body. (1 Sievert = 100 Rem).

T_{1/2} Half-life of a radioactive substance (the time when only one-half of the original substance is still in existence).

Transmutation The process of changing one isotope into another isotope through irradiation with neutrons.

X-Ray Electromagnetic radiation of energy greater than that of visible light. X-rays emanate from outside the nucleus.

Introduction

Welcome to the section of this encyclopedia dealing with the myriad applications of harnessed nuclear radiation not directly related to the production of nuclear power. Whereas nuclear power is generally recognized as the most obvious application of nuclear technology, a relatively unknown fact is that non-power applications resulting from the incredible discoveries of Wilhelm Roentgen, Henri Becquerel, Marie Curie and their successors far outdistance nuclear power in terms of the number of jobs created and the financial contribution to national economies (Waltar, 2004, 2021; Bezdek, 1994, 2021).

Radiation technology has already delivered enormous contributions to medicine, agriculture, modern industry, environmental protection, public safety, and cultural heritage, and the future continues to look very bright for the broadening of these applications to benefit our ever-growing global population. This section of the encyclopedia is dedicated to a review of the variety of contributions of radiation technology and their impact to modern life. In order to cover the extensive spectrum of radiation applications in a coherent manner, the following chapters have been generally grouped according to the categories just mentioned.

Medicine

Of the various areas of where radioisotopes have been used in our modern world, medical applications are among the earliest and the most widely known. Shortly after the discovery of artificial radioactivity in 1934 by Frederic and Irene Joliot (Irene being the daughter of Pierre and Marie Curie), medicine began to benefit from the atom. One of the first uses was Iodine-131 for the treatment of thyroid cancer—followed by a phenomenal growth of diagnostic and therapeutic applications, along with new drug developments. Nuclear reactors and particle accelerators soon came along to produce a wide variety of radioisotopes capable of emitting alpha, beta, and gamma particles—as well as neutrons for specialized cases. Radioisotopes are now in common use for both diagnostic and therapeutic purposes.

We begin this section on medicine with the applications of radiation technology to sterilization, followed by four chapters on nuclear medicine.

Chapter 2: Medicine: Sterilization (Kalia, 2021) focuses on the use of radiation techniques to sterilize most of the surgical supplies readily available at hospitals or medical clinics. The original processes utilized to ensure sterility were based on chemical or heat treatments. But once radiation processes were developed, radiation has generally become the preferred treatment modality for sterilizing bandages, syringes, and other materials commonly used.

Chapter 3A: Medicine: Radionuclides Used in Nuclear Medicine (Venkatesh and Kang, 2021) provides a broad perspective on the numerous radionuclides used for nuclear medicine diagnostic and therapeutic procedures. Specialty radioisotopes using either X-rays, negative or positive beta rays, or gamma rays at low doses are employed in SPECT or PET devices for imaging the site of malignancies during diagnostic procedures, whereas high-energy radiation is deployed in high doses for therapeutic uses. One out of every three patients entering US hospitals or medical clinics benefit from nuclear medicine for either diagnostic or therapeutic purposes.

Chapter 3B: Medicine: Radiopharmaceuticals and Their Use in Nuclear Medicine (Kang and Venkatesh, 2021) is the specialty in which a radiolabeled substance known as a radiopharmaceutical is administered into the patient (“in-vivo”) either for diagnostic or therapeutic purposes. Since the behavior of the radiopharmaceuticals inside the body is governed by the biochemistry of the radioactive molecule and the biochemical functioning of the concerned body organs, nuclear medicine studies provide unique functional information unlike most other imaging modalities. Thus, nuclear medicine imaging is also referred to as functional imaging or molecular imaging. Studies that analyze a patient’s sample outside the body, such as determining for tissue malignancy, are referred to as “in-vitro.”

Chapter 4: Medicine: New Drug Developments (Venkatesh and Korde, 2021) provides the basis for the development of many new drugs coming on the market that require the employment of radiation tracers in order for both the researchers and the licensing agencies to certify that the new drug is reaching and treating the intended target tissue, while minimizing any damage to surrounding healthy cells. Approximately 80% of all new drugs rely on radiation tracer technologies before being approved by the FDA for commercial use.

Chapter 5: Medicine: Therapeutic and Other Applications Using Sealed Radiation sources ([Shrivastava and Venkatesh, 2021](#)) articulates the use of sealed radiation sources for the treatment of cancer. External (outside the body) focusing of radiation on cancerous lesions is known as Teletherapy, whereas Brachytherapy is accomplished by placing the sealed sources in close contact with the lesion (inside the body). These, and other therapeutic techniques using sealed radiation sources, constitute the focus of this article.

Agriculture

Chapter 6: Agriculture: Improving Crop Production ([Sivasankar et al., 2021](#)) reveals the incredible impact of how X-ray, beta, gamma, and neutron irradiation have transformed the food industry. Adequate supplies of food are absolutely critical to the health and actual survival of global citizens. Though not widely known, perhaps the largest impact of harnessed radiation is in the field of agriculture. Well over 200 new varieties of grains and vegetables have been developed using such irradiation techniques to improve crop production with wheat, rice, and cotton leading examples of enhanced, world-wide commerce. Radiation sources harnessed to improve the efficient utilization of both fertilizers and water have saved enormous quantities of these precious resources.

Chapter 7: Agriculture: Improving Livestock Production ([Viljoen et al., 2020](#)) reveals how radiation sources have been employed to enhance the attributes of a huge variety of animal stock. Included is a description of the methods deployed to improve the digestion of food intake, the development of vaccines, and the eradication of harmful insects.

Chapter 8: Agriculture: Electron Beam Irradiation Technology Applications in the Food Industry ([Pillai and Pillai, 2021](#)) details methods currently in global use to preserve food that would otherwise be wasted. Despite new radiation technologies now available to enhance both plant and animal production, a huge volume of farm products spoils before reaching the kitchen table. Hence, preserving the food that has been produced is becoming more essential for everyday commerce, considering the continually expanding global population and the necessity for extending the shelf life of so many perishable products. Food irradiation is now more prevalent in the kitchen than normally recognized—providing sanitized spices and even those special encapsulated coffee creamers that do not require refrigeration.

Modern industry

Chapter 9: Modern Industry: Diagnostics and Process Control ([Thereska and Brisset, 2021a](#)) focuses on the myriad uses of radiation thickness and density gauges to monitor the production of everything from paper production to the control of concrete pours in construction. It is difficult to imagine modern life without the application of radiation techniques to so many products that we now almost take for granted.

The next three chapters focus on nuclear techniques for the testing and inspection of new materials as they become available on almost a daily basis:

Chapter 10: Modern Industry: Gamma Methods for Materials Testing and Inspection ([Bjornstad, 2021a](#)) focuses on the unique properties of gamma irradiation for testing and inspection.

Chapter 11: Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection ([Bjornstad, 2021b](#)) provides a detailed overview on how neutron sources are developed, along with appropriate detection techniques, for use in testing and inspection.

Chapter 12: Modern Industry: Applications of Neutrons for Materials Testing and Inspection ([Bjornstad, 2021c](#)) provides vast coverage on how neutrons are utilized for materials testing and inspection.

Chapter 13: Modern Industry: Development of New Materials ([Fulvio, 2021](#)) focuses on the application of radiation technology to develop a myriad of new products that enter our market on a frequent basis—products all the way from kitchen conveniences (e.g., shrink wrap created by radiation cross-linking) and the semiconductors that provide the key ingredient of our computers and cell phones.

Chapter 14: Modern Industry: Non-Nuclear Power Applications ([Thereska and Brisset, 2021b](#)) provides an overview of how radiation techniques are applied to the fossil fuel industry (coal and oil) and the renewables (solar, wind, and geothermal). Whereas the bulk of this encyclopedia is properly focused on nuclear energy, it is noteworthy that radiation techniques have enormous applications to the development and utilization of sources of energy not tied to nuclear power.

Chapter 15: Modern Industry: Personal Care, Conveniences, & Safety ([Waltar, 2021](#)) is included to reveal the numerous ways that radiation devices are employed to provide for our personal comfort and care—from smoke detectors to the brilliant colors of precious gems.

Environmental protection

Chapter 16: Environmental Protection: Managing Fresh Water Resources ([Kumar and Ansari, 2021](#)) is focused on perhaps the most pressing global challenge today; namely, finding ways to discover, guard, and produce adequate supplies of potable

water—especially for those areas of the globe where fresh water is rapidly vanishing. There are likely many people who would be surprised by how many ways radiation is used to protect and improve our precious environment.

Chapter 17: The Oceans—Formation and Global Climate Change (Hong and Povinec, 2021a) focuses on the formation and significance of our oceans to life on our planet—and how natural contamination takes place. It is continually becoming clear that it is essential to protect the greatest water mass in the world; namely the oceans.

Chapter 18: Environmental Protection: The Oceans—Implications of Man-Made Contamination (Hong and Povinec, 2021b) continues the background provided in Chapter 17 by focusing on how modern life continues to contaminate our precious oceans, and why we need to utilize radiation devices to detect and remove pollutants that could significantly alter the productivity of these massive reservoirs of water.

Chapter 19: Environmental Protection: Reducing Environmental Pollution (Chmielewski et al., 2021) reveals the many ways our current population continues to pollute our only life-giving planet. It is very important, therefore, to determine the sources of pollution and the remedies for reducing and cleaning up such sources.

Public safety and security

Chapter 20: Public Safety and Security: Border Security (Patton, 2021) is focused on the radiation technologies currently in use to make sure the massive daily quantity of goods entering our ports around the world do not contain illicit materials. We all now live in an increasingly fragmented world. Every day we hear of new threats to our desired way of life, and we need assurances that we are being protected.

Chapter 21: Public Safety and Security: Emerging Inspection Techniques—Active Interrogation for Nuclear Materials (Strellis and Gozani, 2021) provides an overview of how even more powerful radiation techniques can be developed and used to thwart the increasingly sophisticated intentions of terrorists and others with criminal intentions.

Chapter 22: Public Safety: High Radiation Sources (Starovoitova, 2021) is included to help us all understand both the benefits of properly employing high radiation sources, along with the precautions necessary to minimize any wrongdoing. Although radiation can, indeed, be very harmful if delivered inappropriately in high doses, that same feature can and is being used very effectively to kill cancer, eradicate harmful insects, and greatly extend the shelf life of perishable food products.

Chapter 23: Public Safety and Security: Nuclear Forensics (Fedchenko, 2021) describes some of the most interesting uses of radiation to help solve crime scenes. Included is both the history of how radiation has been used in criminal investigations, as well as a description of some evolving new technologies to make further use of these unique capacities.

Cultural heritage

Chapter 24: Cultural Heritage Preservation (Varquez and Nagai, 2021) is included to provide insight on how precious works of art can be protected against both natural degradation processes as well as intentional forgery. History is very important to all segments of our global population, and having ways to understand our history, and the associated artifacts, provides a precious contribution to fulfilled living.

Chapter 25: Global Market for Radiation Applications (Bezdek, 2021) is a most fitting way to conclude this section. It provides an assessment of how incredibly important this “silent servant” is to modern life, etc. and an assessment of how the global market for radiation applications continues to grow. Data indicate the growth ranges from 5% per year to over 20% per year, depending upon the sector—with a current contribution of over 500 billion US dollars to our global economy. As such, both the number of jobs and the economic impact of radiation applications—NOT DIRECTLY CONNECTED TO THE PRODUCTION OF HEAT AND ELECTRICITY—widely exceed the comparable numbers associated with nuclear power production. Marie Curie can be rightly proud of the legacy she created when she first discovered and coined the term “radioactivity!”

So how is this all done? How can radiation, which cannot be seen, heard, or felt, accomplish such enormous tasks? To the uninitiated reader, this may be very difficult to grasp. Hence, before moving on to the remaining 24 chapters in this section that cover these many applications in considerable detail, the present chapter will be focused on some of the unique characteristics of radiation that allow it to be such a helpful servant. This requires defining the “hooks” upon which to fasten the harness.

Our forefathers have known for centuries that oxen and horses possess strength far beyond that of humans. Consequently, they built harnesses to translate their energy into pulling plows and to perform other useful work. Chemical energy, originally exploited in campfires for heating caves and cooking meals, has been recognized to have explosive characteristics when certain fuels are mixed. This “hook” gave rise to the internal combustion engine—allowing tractors of enormous “horsepower” to considerably increase efficiency on the farm. This same internal combustion engine continues to be the primary source of power widely used today for cars, trucks, ships, and locomotives. Early observations that electrons could be directed to move in a preferred direction by applying a voltage to one end of a conductor has given rise to vacuum tubes and then computer chips capable of transmitting almost unfathomable amounts of data in modern computer systems.

As we now look to radiation, we find that there are over a dozen unique aspects of nuclear radiation and of radioactive decay process that can be and have been tapped and harnessed for beneficial human use.

Material penetration

Different types of particles are emitted during the radioactive decay process and they have unique material penetration powers. For instance, if a beam of gamma rays of a known energy is focused upon a thin sheet of aluminum, the precise thickness of that sheet can be determined by measuring the reduction in the beam current on the other side of the metal. This property (also referred to as attenuation) is used in countless commercial operations such as making rolled metals or paper, filling containers, and so on (Fig. 1).

The most commonly recognized application of radiation penetration, and its associated attenuation property, is for imaging. A perfect example is an ordinary dental X-ray. Here the dentist focuses an X-ray beam on one side of the patient's mouth and places a film on the other side of the tooth. That's the little "wad" that we are asked to bite down on and then hold still. As the X-ray traverses the area of the tooth, it easily penetrates through the gum and exposes the film. But the tooth is more massive, and very little of the X-ray can penetrate. Hence, that area of the film is not exposed. Should there be a cavity in the vicinity, a bit more of the X-ray can get through to the film. Hence, a "gray" spot appears at that point—and the dentist then knows precisely what to do.

X-rays are also commonly used to check the health of lungs (chest X-rays), as well as to assist the orthopedist should the patient have the misfortune of breaking a bone somewhere in his or her body. The amount of radiation normally received from such procedures is now quite small (a few micro Sieverts) and certainly worth it. Most people would not want the dentist to pull the wrong tooth or to have a broken bone set improperly.

Extensive use of X-rays, gamma rays, and beta particles to measure thicknesses, densities, etc. in a myriad of medical and industrial processes are covered by (Venkatesh and Kang, 2021; Thereska and Brisset, 2021a). Modern industry simply would not be able to function without the employment of this material penetration attribute of radiation technology.

Heat source

Alpha and beta particles are stopped within short distances of their origin as they collide with atoms in their surrounding material, thus converting their initial kinetic energy into heat. Substantial heat is generated in this process, much like the heat generated in a skidding tire on dry pavement. This heat can be used to generate electricity, and it is precisely this technique that is used to power deep space probes. The vivid pictures returned to Earth from Mars during the *Viking* mission were made possible by radioactive heat sources. This is the technique utilized to power the *Voyager* "twins" and other space probes—a topic covered by Bennett (2021) in Chapter 4, Section 9 of this encyclopedia. Properly designed, such power sources can last for decades—without any moving parts (Fig. 2).

Pu-238 is often used as the alpha emitter to provide this heat source. This material is encased in a unit called a radioisotope heater unit (RHU). Pu-238 is not fissile, but it has a half-life of 87.5 years, allowing the RHU to safely provide 50% of its initial heat after 87.5 years of use. These units are productively employed in space probes and other spacecraft to keep them from freezing in the unforgiving environment far removed from the atmosphere of the earth. By utilizing the thermoelectric principal of attaching two insulated wires made of dissimilar materials (i.e., nickel and copper) and attaching one end to the heat source and the other end to a cooling junction that can reject heat to space, an electrical current can be generated. This unit is called a radioisotope thermoelectric generator (RTG). Fig. 3 is an illustration of a hot glowing pellet of Pu-238 encased in an RTG.

It may be of interest to the reader that it is the radioactive decay of uranium-238 ($T_{1/2} = 4.46$ billion years), thorium-232 ($T_{1/2} = 14$ billion years), and potassium-40 ($T_{1/2} = 1.27$ billion years) that provide the earth with its enormous heat source just below the surface of the globe.

Material Penetration

- Degree of beam attenuation determines thickness of material
- Metal foil and paper manufactured by this process

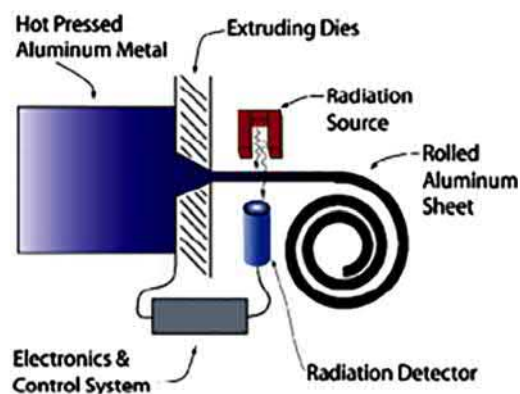


Fig. 1 Material penetration.

Heat Source

- Alpha and beta particles stop within short distances of their origins
- Substantial heat is generated in this process
- Heat can be used to generate electricity (e.g. thermoelectric effect)
- Such sources used in space probes (e.g. Viking mission to Mars)

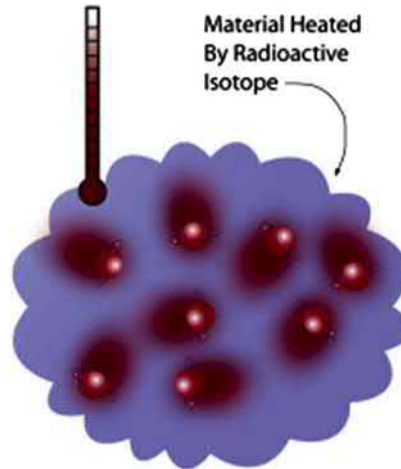


Fig. 2 Heat source.



Fig. 3 A glowing Pu-238 pellet encased in an RTG. Source: Wikipedia.

Emission

Many isotopes emit gamma rays during the radioactive decay process, often in conjunction with other particle emissions. Since gamma rays can penetrate through a relatively large mass, it is possible to attach specialty gamma emitters to materials via chemical means that can easily be transported by fluids. Such a linkage allows the gamma-emitting radioisotopes to flow through a set of pipelines or blood vessels, thus allowing the location of this radioactive material to be measured as a function of time. This *tracer* or *labeling* technique is routinely used to map groundwater movement, detect pipe leaks in the petroleum industry, and diagnose ailments in the human body. As might be expected, this technique has made a powerful impact on the world of medicine (Venkatesh and Kang, 2021). It likewise is prolifically used in many aspects of both agriculture and modern industry (Sivasankar et al., 2021; Viljoen et al., 2020; Thereska and Brisset, 2021a) (Fig. 4).

Transmutation/activation

By focusing a beam of neutrons on unknown materials, it is possible to transmute (fundamentally change) the target material into another isotope (or possibly an isotope of a different element). If this new isotope is radioactive, it will emit a radiation signature that uniquely determines its identity. It is then a very simple process to identify the target material (the original unknown material) that was converted to the measured resulting product created by absorption of the neutron beam. This property is used routinely for

Particle Emission

- Gamma rays can penetrate through substantial material
- Hence, isotopes that emit gamma rays can be attached to fluid materials & then continuously monitored vs time
- Such tracing techniques are used to map groundwater movement, detect pipe leaks, & diagnose ailments in the human body

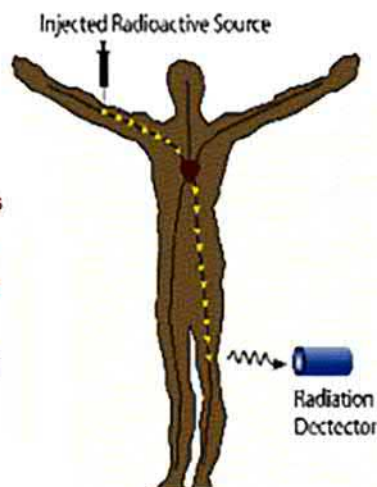


Fig. 4 Particle emission.

criminology investigations (Patton, 2021; Strellis and Gozani, 2021; Fedchenko, 2021). It is also useful in determining the impurity levels in materials (Bjornstad, 2021b,c) (Fig. 5).

Change chemical structure

Bombarding some materials with intense beams of gamma rays, electrons, or neutrons can break molecular bonds and evolve new types of materials (i.e., change the chemical or physical structure). Such treatment is used to produce stronger plastics or increase the rate of chemical reaction (Fulvio, 2021). This approach is even used to enrich the color and brilliance of some types of gemstones by slightly modifying their structure (Waltar, 2021) (Fig. 6).

Cell destruction

In a manner similar to the above, controlled amounts of beta particles, X-rays, or gamma rays can be used to kill microorganisms or sterilize unwanted insects. This property is used in millions of operations per year for sterilizing hospital equipment (Kalia, 2021), killing food pathogens (i.e., any microorganisms that can cause disease) and suppressing sprouting in vegetables, and eradicating cancer cells (Venkatesh and Kang, 2021). It is this property that allows for a potential transformation in our massive food industry—to provide safer produce with a longer shelf life (Pillai and Pillai, 2021) (Fig. 7).

Transmutation/Activation

- Neutrons focused on certain materials will transmute the target material into a new radioisotope
- The target material can then be identified by analyzing the radioactivity of the newly formed radionuclide
- This technique is often used in criminology investigations and for revealing explosives in airport luggage

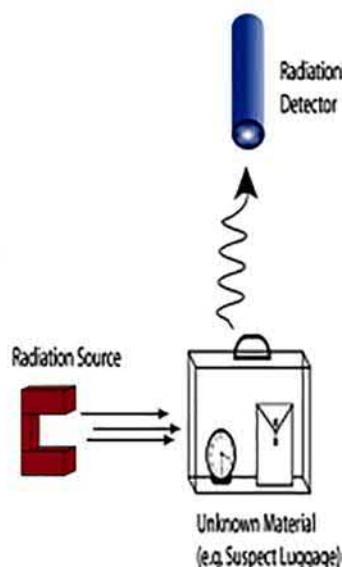


Fig. 5 Transmutation/activation.

Change Molecular Structure

- Bombarding some materials with high enough levels of tailored radiation will break molecular bonds
- This changes the chemistry and structure of materials
- Sample products are tough plastics & special rubber for tires

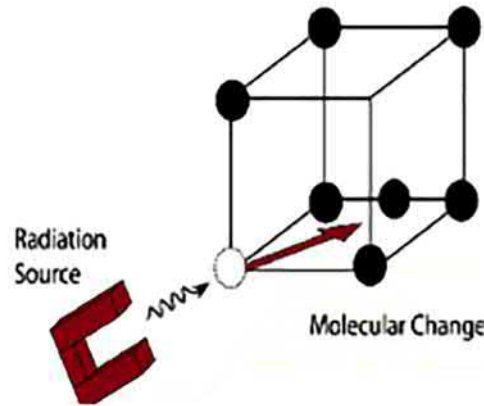


Fig. 6 Change chemical structure.

Cell Destruction

- In a more macro scale, controlled amounts of beta particles or gamma rays can be used to break DNA bonds, thereby killing undesirable cells (e.g. insects & pathogens)
- This process is used to sterilize hospital equipment, suppress sprouting in vegetables, and kill cancer cells

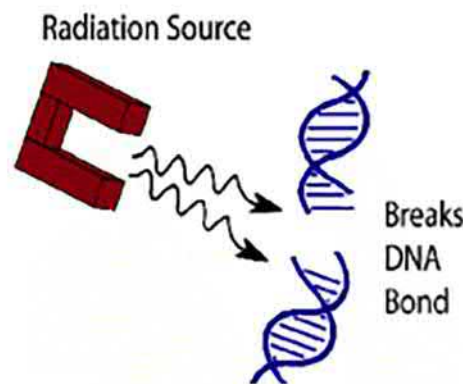


Fig. 7 Cell destruction.

Decay time

Knowing the “half-lives” of certain radioactive isotopes (i.e., the time it takes for one-half of the atoms of a certain species to decay into another species), it is possible to perform aging analyses on a variety of archeological artifacts—including the age of Earth (Varquez and Nagai, 2021). This is the technique used to determine prehistorical climate changes, the role of the industrial age in contributing to greenhouse gases, and so on.

Carbon dating is the process that is the most familiar. Natural carbon (C-12) is stable, i.e., it does not decay. It constitutes 98.89% of all carbon in existence. However, carbon-13 also exists in nature (comprising essentially the remaining 1.11%). But of most interest to us within the present context is that carbon-14 also exists in nature as a result of cosmic radiation. Neutrons from outer space react with nitrogen in the upper atmosphere to form carbon-14 plus a proton. The amount of carbon-14 is exceptionally small. In fact, for every carbon-14 atom existing in nature, there are one trillion atoms of carbon-12.

Even though the amount of carbon-14 is exceedingly small, there exists a constant ratio of carbon-14 to carbon-12 naturally occurring in our environment at all times. All living beings (whether plant or animal) have this same ratio in their systems, since they all have a constant intake of carbon while still alive. But when such plants or animals die, there is no more intake of carbon in any form. Carbon-12 is stable, but carbon-14 decays with a half-life of 5715 years. Consequently, the ratio of carbon-14 to carbon-12 diminishes with time in dead species. This change in the ratio of carbon-14 to carbon-12 can then be used to determine the age of the dead matter.

Since the half-life of carbon-14 is 5715 years, there is a practical limit of time for which the standard carbon dating method is accurate. This is generally in the range of 50–50,000 years. However, it is now possible to directly detect individual carbon-14 atoms in a sample—one carbon-14 atom out of 10^{16} atoms of C-12. This allows a significant refinement in determining the age of the dead object.

Other radioactive techniques have been developed to measure artifacts up to several million years of age, based on the bombardment of cosmic rays that create radioisotopes of beryllium and aluminum. When an earth movement (such as a volcano, earthquake, or mud slide) buries sediments containing these metals, the cosmic ray bombardment stops. However, the radioactive isotopes continue to decay. Hence, the “clock” is set at this time. Beryllium-10 has a half-life of 1.5 million years, and aluminum-26 has a half-life of 0.71 million years. The age of the sediment specimen can be determined by measuring the change in the ratio of these two radioisotopes.

Many other applications utilizing the long half-lives of substances such as uranium and potassium have allowed accurate measurements to be made for the age of the earth, the effect of an asteroid hitting the earth some 65 million years ago, etc. More detail for the numerous applications of exploiting the property of radioactive decay is provided in (Varquez and Nagai, 2021) (Fig. 8).

Luminescence

Particles emitted from radioactive decay can cause the emission of visible light when interacting with certain target materials. Tritium emits beta particles as it decays and the beta radiation can excite a luminescence material upon impact to produce visible light. Tritium, mixed with a suitable luminescence material, is routinely used in the form of a “paint” for lighting airport runways (Waltar, 2021). It is likewise used for exit signs or other safety equipment in remote areas where it is not economical to deliver electricity or where 100% reliability must be guaranteed—even when all power is lost (Fig. 9).

Ionization

Radiation striking outer-shell electrons can, if sufficiently energetic, knock such electrons out of orbit and cause the targeted material to become ionized. This is called ionizing radiation. The targeted material then contains free radicals. When the striking radiation is very intense, it can enhance polymer chains to form useful products such as polyethylene, PVC pipes, and heat-shrink plastics. Conversely, it can also degrade polymers to make products such as Teflon and cellulose (Fulvio, 2021) (Fig. 10).

Decay Time

- Knowing the half-life of certain radioactive species, aging analyses are possible
 - Archeological artifacts
 - Age of the earth
 - Prehistoric climate changes
 - Industrial age climate stability

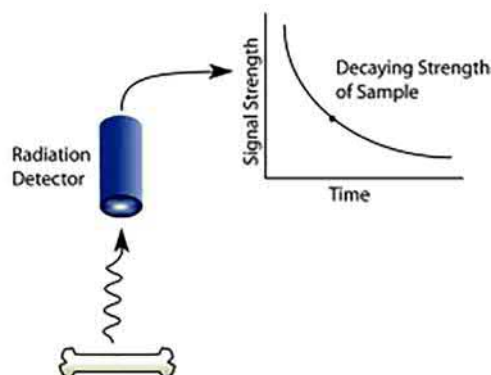


Fig. 8 Decay time.

Luminescence

- Some radioactive materials (e.g. tritium) can produce wave lengths in target materials that are visible to the human eye
- Routine uses include lighting for airport runways
- 100% reliable & self contained

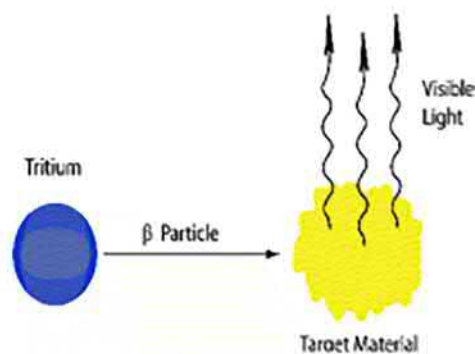


Fig. 9 Luminescence.

Ionization

- Sufficiently high energy radiation will knock off electrons in surrounding media
- Media becomes electrically charged

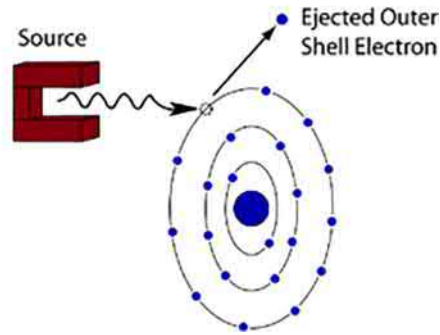


Fig. 10 Ionization.

X-ray fluorescence

X-rays focused on a material of unknown composition can knock out inner electrons from the atomic structure of the unknown material. When this happens, electrons from the outer shells will refill the vacated site. The movement of these electrons causes the emission of X-rays with energies and relative intensities characteristic of each element comprising the unknown material—thus allowing precise identification of the unknown composition (Patton, 2021; Strellis and Gozani, 2021; Fedchenko, 2021). This is a powerful method for measuring concentrations of trace elements and is used in any fully equipped analytical laboratory. It is often employed for criminology work (Fig. 11).

Neutron slowing down

If a burst of energetic (fast) neutrons is introduced into a media, these neutrons will collide with the nuclei of surrounding atoms—thus losing energy and slowing down to velocities typical of ordinary materials (i.e., thermal speeds). This slowing-down process can take place very rapidly if there is a substantial water component in the surrounding media, since water contains hydrogen (an element consisting of a single proton). A fast neutron colliding with a proton can lose all of its energy in a single collision—in the same manner as a cue ball that is stopped in its tracks upon a “bull’s-eye” collision with the eight-ball. However, when a fast neutron collides with a heavier element, such as iron or lead, it loses its energy gradually and many collisions are required for the neutron to reach thermal speeds. This property can be used to obtain an accurate measurement of how much moisture is contained in a particular medium (such as soil), since the number of slow (thermal) neutrons detected is directly proportional to the number of water molecules in the immediate vicinity (Sivasankar et al., 2021).

This property is extensively used for prospecting good sites for oil wells (Thereska and Brisset, 2021b; Bjornstad, 2021b,c) and the discovery of other productive mineral sites (Fig. 12).

Neutron or photon scattering

As a variation to the neutron slowing-down property noted above, if a neutron colliding with a nucleus of unknown identity does not make a “bull’s-eye” hit, it will scatter forward or backward at some angle relative to its original direction. This information can be

X-Ray Fluorescence

- Monochromatic X-rays focused on a material of unknown composition
- Inner shell electron ejected
- Outer shell electrons move to inner shells, emitting an X-ray characteristic of unknown material

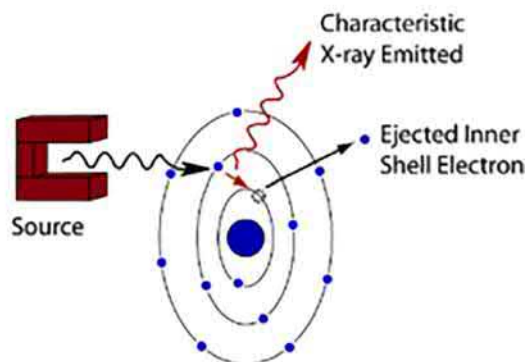


Fig. 11 X-ray fluorescence.

Neutron Slowing Down

- A burst of fast neutrons injected into a medium
- Neutrons will slow down rapidly if the medium is rich in protons (i.e. water)
- Excellent way to determine moisture content

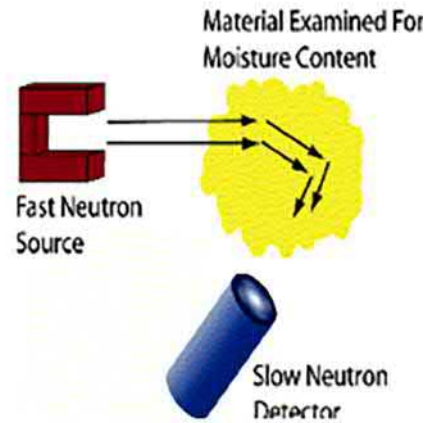


Fig. 12 Neutron slowing down.

used to determine the identity of the unknown scattering material (Patton, 2021; Strellis and Gozani, 2021). Gamma or X-ray backscattering is a variation of the same theme (Bjornstad, 2021a). The difference between gamma or X-ray scattering and neutron scattering is that gamma and X-rays interact with the atomic electrons whereas the neutrons interact with the nuclei of the atoms. These techniques can be used to analyze materials on the surface of bulk matter, such as the thickness of plastic coatings on metals (Theriska and Brisset, 2021a). X-ray backscattering is routinely used in airports to screen passengers for hidden materials (Radiation and Airport Security, <https://www.epa.gov/radtown/radiation-and-airport-security-scanning#about-radiation>) (Fig. 13).

Fission

For certain heavy isotopes such as uranium-233, uranium-235, or plutonium-239, the absorption of even a low-energy ("thermal") neutron can cause a transformation to take place that produces a nucleus sufficiently unstable that it will literally break apart, generally resulting in the release of two fission fragments, several new neutrons, and an enormous amount of energy (~ 200 MeV per fission event). When the fission event takes place, the mass of the two fission fragments (or fission products) and associated free neutrons is slightly less than the mass of the original fissionable element and the neutron it absorbs, and this loss of mass is instantly converted into energy via the famous Einstein equation:

$$E = mc^2$$

Neutron or Gamma Scattering

- The identity of unknown material can sometimes be determined by measuring the scattering angle of a focused beam of incident neutrons or gamma rays

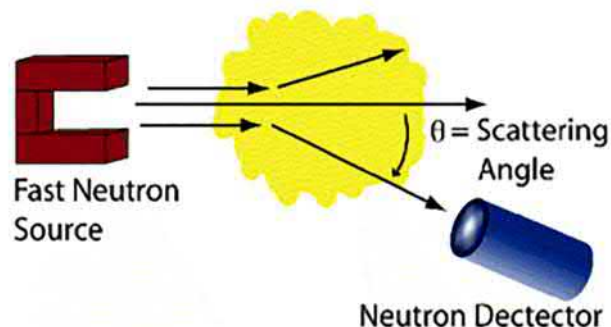


Fig. 13 Neutron or gamma scattering.

where E is energy, m is the mass lost, c is the speed of light.

The same equation can quantify the energy released in radioactive decay, such as the decay of Pu-238 described in the “Heat Source” section above. A nuclear reactor operating on the fission concept is designed so that a sufficient number of neutrons released during a fission event will cause subsequent fissions, resulting in a self-sustaining chain reaction. The fuel and absorber assemblies are arranged and controlled such that the number of fission events per unit time remains constant—producing a steady-state generation of power.

Fig. 14 is a very simplified illustration of a U-235 fission event, whereas Fig. 15 is expanded to illustrate the eventual production of Pu-239, which is another fissile isotope. The latter production is a result of a neutron capture in U-238 to form U-239, followed by a beta emission to form Np-239, and then followed by another beta emission to form Pu-239. The result is that a reactor initially fueled with only uranium ends up with fission occurring from both U-235 and Pu-239. After several months of operation, about a third of the power is generated by plutonium.

The heat generated by the fission process is transferred to a coolant that, in turn, converts its energy to a rotating turbine-generator system to produce large amounts of electricity. The bulk of this encyclopedia is dedicated to this topic.

Fusion

For very light isotopes, such as deuterium (a hydrogen atom containing a neutron in addition to a proton) or tritium (a hydrogen atom containing two neutrons in addition to a proton), it is possible that a collision of deuterium and tritium at high enough speeds (i.e., exceedingly high temperatures!) will cause the two nuclei to fuse and form a new isotope—yielding an energy release (in the case of a D-T reaction) of 17.6 MeV—via the same mass/energy equivalence equation noted in the previous section. This energy release is substantially greater than in the fission process outlined above if measured per unit mass of the reacting nuclei. This process is called nuclear fusion (Fig 16). The fusion of light isotopes is the energy source for our sun (primarily converting ^2H nuclei into He) and all the other stars.

The principal challenges of harnessing the nuclear fusion process for commercial use include the extremely high temperatures (~ 14 million degrees Kelvin) required to initiate the fusion process, and the difficulty in constraining this extremely hot plasma.

Fission

- Very heavy fissile isotopes (e.g. U-235, Pu-239) will likely undergo fission (i.e. break apart) upon absorbing a neutron
- The mass lost in the process is converted into energy
 $E = mc^2$ (energy = mass times the speed of light squared)
- Two fission fragments usually result, along with 2 or 3 neutrons to continue the chain reaction

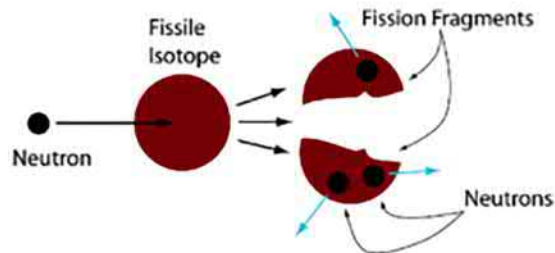


Fig. 14 Highly simplified nuclear fission.

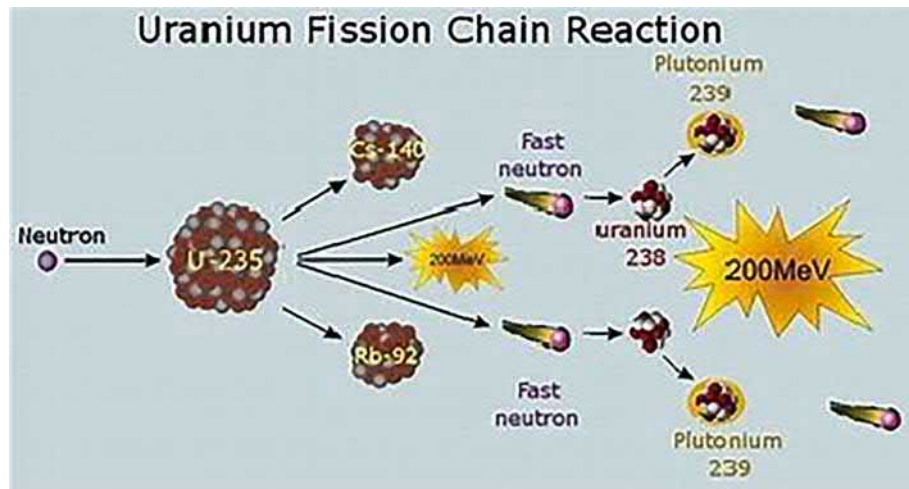


Fig. 15 ^{235}U fission process to produce the fission fragments ^{92}Rb and ^{140}Cs and eventually ^{239}Pu . Source: Wikipedia.

Fusion

- Very light isotopes (e.g. deuterium, tritium), if brought together at very high velocities (i.e. very high temperatures), will fuse together, forming a new element.
- The mass lost in the process is converted into an immense amount of energy
 $E = mc^2$ (energy = mass times the speed of light squared)

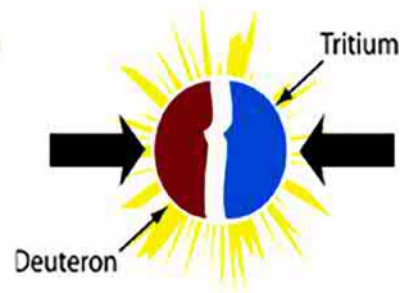


Fig. 16 Highly simplified nuclear fusion.

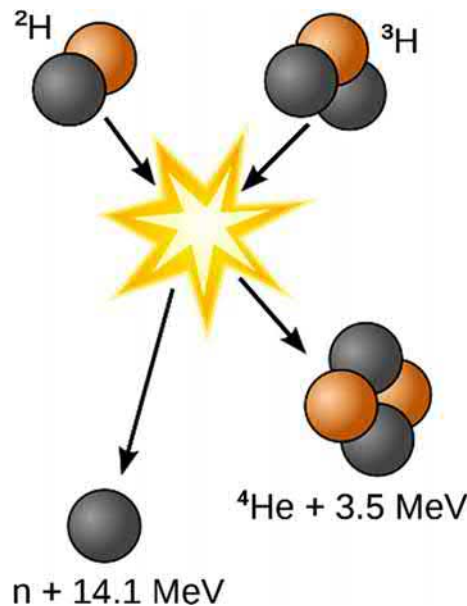


Fig. 17 The fusing of deuterium (^2H) and tritium (^3H) (Wikipedia).

However, the lure is very strong for ultimate success. Accordingly, progress to harness this process continues in many parts of the world (Fig. 17). The state-of-art in the pursuit of ways for harnessing nuclear fusion to the benefit of mankind is covered in detail in Section 12 (Nuclear Fusion R&D) of this encyclopedia.

Hopefully, this brief overview has helped take some of the mystery out of how nuclear radiation can be productively employed. Its many unique characteristics afford a plethora of opportunities for scientists and engineers to tap its power for beneficial use. Now that we have a better understanding of where to “strap the harness,” we can begin our investigation of the numerous ways that radiation has so effectively improved our lives.

See Also: Natural and Man-Made Sources of Radiation; Radiation Sources; Radioactive Decay.

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Medicine: Sterilization

Vikram Kalia and Deepa Joseph, Microtrol Sterilisation Services Pvt. Ltd., Mumbai, India

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Glossary

Absorbed dose The amount of energy absorbed per unit mass of irradiated matter,

$D = dE/dm$, where

D = Absorbed dose

dE = mean energy

dm = mass of matter

Beam energy The kinetic energy of the accelerated electrons in the beam. The beam energy determines the depth of penetration of electrons. Beam energy is measured in Electron volts (eV).

Beam spot profile The shape of the unscanned electron beam incident on the reference plane.

Bio burden The population level of viable microorganisms on or in raw materials, products, and labeling/packaging materials determined before sterilization.

Becquerel (Bq) The derived unit for radionuclide activity. One Becquerel equals one nuclear transformation per second. The unit 'Curie' was earlier used to indicate activity

$1 \text{ Curie} = 3.71 \times 10^{10} \text{ Bq}$

Conversely $1 \text{ Bq} = 2.7 \times 10^{-11} \text{ Curies}$

Typically, a commercial gamma irradiator will have a source strength of 1 million Curies ($3.71 \times 10^{16} \text{ Bq}$).

Calibration A set of operations that establish, under specified conditions, the relationship between values of a quantity indicated by a measuring instrument or measuring system, or values represented by a material measure or a reference material, and the corresponding values realized by standards.

D value A value indicating the extinction rate of microorganisms killed under defined conditions.

D_{10} value The time or radiation dose required to inactivate 90% of a population (only one tenth of the starting value remaining) of the test microorganism under stated exposure conditions.

Dose Limit Ratio (DLR) The ratio of the upper regulatory dose limit to the lower regulatory dose.

DNA Deoxyribonucleic acid; the information-carrying genetic material that comprises the genes. DNA is a macromolecule composed of a long chain of deoxy-ribonucleotides joined by phosphodiester linkages. Each deoxy-ribonucleotide contains a phosphate group, the five-carbon sugar 2-deoxyribose, and a nitrogen-containing base.

Dose Quantity of ionizing radiation energy imparted per unit mass of a specified material. The unit of absorbed dose is Gray (Gy), where 1 Gy is equivalent to the absorption of 1 J/kg.

Dosimeter A device having a reproducible, measurable response to radiation, which can be used to measure the absorbed dose in a given system.

Dose Uniformity Ratio (DUR) The ratio of the maximum dose absorbed by the product to the minimum dose absorbed by the product. In a gamma irradiator, this ratio increases with the density of the product as well as with the size of the container. This term is used as a measure of the dose distribution within a product that is irradiated. In an irradiation plant, a low DUR avoids unnecessary treatment, while a high DUR indicates better efficiency of energy utilization.

Dose Limit Ratio (DLR) The upper regulatory dose limit to the lower regulatory dose limit.

Electron accelerator A device that emits high energy electrons.

Enzyme A protein that accelerates a specific chemical reaction in a living system.

Electron volt (eV) A unit of energy. One electron volt is the kinetic energy acquired by an electron in passing through a potential difference of one volt in vacuum. $1 \text{ eV} = 1.60219 \times 10^{-19} \text{ J}$.

Pharmaceutical Formulation This is the process, in pharmaceuticals, in which different chemical substances, including the active drug, are combined to produce a final medicinal product.

Free radicals An uncharged molecule (typically highly reactive and short-lived) having an unpaired valency electron.

Gray (Gy) This is the SI unit for absorbed dose. Gray has replaced 'Rad,' which was earlier used to measure absorbed dose.

$$1 \text{ Gy} = 1 \text{ J/kg} = 100 \text{ Rad}$$

$$1 \text{ kGy} = 1000 \text{ Gy}$$

Inherent product bio burden The bio burden present on a product after its manufacturing process before the product is sterilized.

Installation Qualification The process of obtaining and documenting evidence that equipment has been provided and installed in accordance with its specification.

Ionization A process in which a charged particle is created from a parent atom or molecule or other bound state.

Isotope Nuclides having the same atomic number, Z (i.e. the same chemical element) but having a different mass number, A .

Irradiation Operator The company or body responsible for the irradiation of a product.

Maximal resistance Components identified as having high resistance, among the reference set of resistances of micro-organisms. This value helps in substantiating the dose as per the $VD_{\max}$ method.

Maximum acceptable dose The dose given in the process specification as the highest dose that can be applied to a defined product without compromising safety, quality or performance.

Molecular bond A molecular bond, usually a covalent bond, is a chemical bond that involves the sharing of electron pairs between atoms.

Operational Qualification A process of obtaining and documenting evidence that installed equipment operates, within predetermined limits, when used in accordance with its operational procedures.

Performance Qualification Process of obtaining and documenting evidence that the equipment, as installed and operated in accordance with operational procedures, consistently performs in accordance with predetermined criteria and, thereby, yields a product meeting its specification.

Poisson distribution A discrete probability distribution that expresses the probability of a given number of events occurring in a fixed interval of time or space if these events occur with a known constant mean rate and are independent of the time since the last event.

Practical electron range (r_p) The depth in homogeneous material to the point where the tangent at the steepest point (the inflection point) on the almost straight descending portion of the depth-dose distribution curve meets the extrapolated X-ray background (see Fig. 4).

Primary Manufacturer The body responsible for the design and manufacture of a medical device, together with the safety and performance of that medical device when placed on the market.

Process thickness A term usually used in relation to electron beam radiation. The process thickness is defined as the depth at which the dose equals the entrance (surface) dose. See "useful range" in Fig. 4.

Radiation resistance of a population The resistance shown by a population of micro-organisms to radiation. It is dependent on the amount of absorbed radiation energy required to inactivate the mixture of microorganisms present on a product. Radiation resistance, even under comparable conditions, varies widely among different microorganisms.

Radioisotopes The unstable form of an element that emits radiation to transform into a more stable form. Emitted radiation brings about a change on the product when exposed.

Of the 118 elements enlisted in the periodic table, only 94 radioisotopes occur naturally, and radiation emitted by them are of different types, most often alpha, beta and gamma.

Most radioisotopes used for industry are artificially produced in research reactors and accelerators by exposing a target material to 'intense particles,' such as neutrons or protons (neutron flux in a nuclear reactor), followed by different chemical processes to bring them into the required chemical form (Previtali, 2021).

Sterility Assurance Level (SAL) The probability of a single viable microorganism occurring on an item after sterilization.

Scan width The dimension of the irradiation zone perpendicular to the direction of product movement at a specified distance from the accelerator window.

Standard distribution of resistances (SDR) A reference set of resistances of microorganisms and corresponding probabilities of occurrence.

Sterilization A validated process used to render a product free from viable microorganisms.

Sterilization dose The minimum dose to achieve the specified requirements for sterility.

Terminal sterilization A process whereby a product is sterilized in its final container or packaging, which permits the measurement and evaluation of quantifiable microbial lethality. It is an important process, as it ensures the product remains sterile.

Uncertainty of measurement A parameter associated with the result of a measurement that characterizes the dispersion of values that could reasonably be attributed to the measurement

VD_{max}¹⁵ The maximal verification dose for a given bioburden, consistent with the attainment of an SAL of 10^{-6} at a specified sterilization dose of 15 kGy.

VD_{max}²⁵ The maximal verification dose for a given bioburden, consistent with the attainment of an SAL of 10^{-6} at a specified sterilization dose of 25 kGy.

Introduction

The word ‘sterilization’ brings to mind terms like pure, clean, aseptic, safe, hygienic and hospitals, hospital equipment and, rightly so, as it is associated with medicine, medical devices, etc., key for an under-treatment patient’s recovery in a hospital, or even otherwise.

We address the seven questions i.e., Why, How, How Much, What, Where, When and Who related to sterilization of medicines using irradiation.

Why

Comparatively, medical practices were relatively unhealthy until the late 1800s, and once the germ theory of diseases gained credibility, practitioners understood the cause for increased cases of disease in a hospital environment. Antiseptic surgical dressings, importance of disinfection and sterilization processes, were introduced towards the end of the 19th century. For example, Ernst von Bergmann introduced the autoclave in the 1870s, a device for sterilizing surgical instruments. A few decades later, in the 1920s, wide-scale production of syringes and needles began. Since these early beginnings, hygienic practices and sterilization continued to evolve and have helped control the spread of disease and infections. The first use of Ethylene oxide gas as a hospital sterilant occurred in 1940, radiation sterilization was introduced in 1956 and, in particular, gamma radiation sterilization in 1964 facilitated industrial scale sterilization.

A sterilization process renders products free from viable micro-organisms such as bacteria, fungi and their spore forms and viruses.

Sterilization is an important stage in the field of medicine and healthcare. Its application starts right from the medicine that is administered to a patient and to various accessories, devices, instruments and equipment used for treating a patient. While medicines such as peptides, proteins and some chemotherapeutic agents, when ingested, are naturally sterilized in the patient’s gastrointestinal tract, medicines that are administered through intravenous/intramuscular injections are required to be necessarily sterilized to prevent complications, even mortality.

In addition to medicines, various components in the chain also need to be sterile, whether these consist of large equipment, packaging components, pre-assembled kits, etc.

How

a. Radiation effect on micro-organisms

The binding energy of molecular bonds is known to be below 12 eV (eV). During the radiation process, an energy level of 10 eV or more is deposited onto the micro-organisms, which leads to a chemical reaction and breakage of molecular bonds—leading to inactivation of micro-organisms. This inactivation takes place either due to direct energy deposition (ionization) on vital components of the micro-organisms, or indirectly due to formation of highly reactive chemical radicals that, in turn, bring about random detrimental reactions in the micro-organism.

Targets for direct ionization could be the DNA molecule, or an enzyme, or any other crucial molecule in the cell causing significant change in the cell. Such multiple changes will result in complete inactivation of the cell. Damage to the cell membrane structure or cell respiratory functions are some examples of changes that result in complete inactivation of micro-organisms.

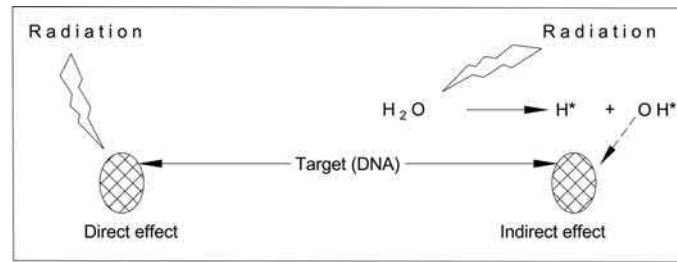


Fig. 1 Example of direct and indirect effect of radiation resulting in cell damage.

All micro-organisms have a certain amount of water. Radiation causes formation of free radicals (OH^* and H^*) and molecules such as H_2O_2 , from water, which are highly reactive and react with the vital components of micro-organisms. These reactions are lethal and, thus, indirectly cause inactivation of micro-organisms (Fig. 1).

b. Concept of SAL

Inactivation or sterilization of micro-organisms by irradiation follows an exponential law, i.e., no matter how high the irradiation dose is, there is always a chance that a micro-organism has survived the process. Thus, evidence that a product is sterile is proved only in terms of probability. Sterility of a product is defined in terms of the probability of existence of an unsterile product in a population, and the term used to indicate this probability is Sterility Assurance Level (SAL). SAL is a function of the number and types of micro-organisms on the product, the lethality of the sterilization process, and sometimes the environment in which micro-organisms exist during treatment.

Each log reduction—a reduction by a factor of 10—represents 90% reduction in micro-organism population. Thus, a 6-log reduction indicates that the microbial population has reduced by a million, which is theoretically close to zero. SAL is normally expressed as 10^{-n} . Some countries mandate a maximum SAL of 10^{-6} in sterile medical devices. Different values of SAL may also be selected; however, these matters are usually regulated by national licensing bodies.

How much

It is essential to first determine the maximum acceptable radiation dose that a product could withstand without affecting its intended functions. Next, the sterilization dose at which the product is rendered sterile is determined. The sterilization dose identified should be less than, or equal to, the maximum dose determined—failing which, the product cannot be sterilized by irradiation.

As mentioned above, a SAL of 10^{-6} is accepted by most countries, although it begs the question as to what is the optimal dose of irradiation required to achieve a SAL of 10^{-6} . This dose is product-specific and depends on the bio burden, D_{10} values and the SAL required.

The sterilization dose may be determined using any of the three methods, i.e., Method 1, Method 2 or VD_{max} method. The Product performance qualification mentioned earlier is conducted using the sterilization dose thus determined.

Approaches to determine the Sterilization Dose.

- Method 1:** In this method, the radiation resistance of the inherent product bio burden is compared with the radiation resistance of a population (mixture of micro-organisms) having a standard distribution of resistances (SDR).

The various doses to achieve a SAL of 10^{-2} , 10^{-3} , 10^{-4} , 10^{-5} and 10^{-6} , for increasing levels of an average bio burden, of a microbial population having SDR, are arrived at using computational methods (Tables 5 and 6 in ISO 11137 – 2).

Once the average bio burden in the product is determined, the dose required to achieve a SAL of 10^{-2} in the SDR population is arrived at referring to Tables 5 and 6 of the above Standard. One hundred units of the product are subjected to this dose and individual sterility tests are conducted on these 100 units. If ≤ 2 positive sterility tests are determined in these 100 product units, it verifies that the product bio burden is more sensitive than the SDR population and, thus, the dose for the desired SAL is arrived at, for routine processing, after referring to Tables 5 and 6. The rationale for allowing 2 positive sterility tests is arrived at by applying the Poisson distribution, which indicates that there is a 92% probability that 0, 1 or 2 positive tests will occur.

- Method 2:** The sterilization dose in this Method is also arrived at by using the radiation resistance of the product bio burden. Incremental doses are subjected to 100 product samples. Of these incremental doses, the dose at which ≤ 2 positive sterility tests are achieved will be the dose at which a SAL of 10^{-2} is achieved. Based on this finding, the dose for SAL values below 10^{-2} is extrapolated and determined. The validity of the extrapolation is confirmed using computer simulations.

- VD_{max}^{25} method or VD_{max}^{15} method:** In this method, the dose of 25 kGy* or 15 kGy is selected. It is further substantiated that this dose ensures sterility to the product. The rationale, however, is similar to the one used for Method 1. The radiation resistance of a population with maximal resistance is then determined and the doses to achieve various SAL for such a population is tabulated in Tables 9 and 10 of the standard ISO 11137-2. The average product bio burden is thereby determined. The dose (VD_{max}^{25} or VD_{max}^{15}) required for a SAL of 10^{-1} in a population with maximal resistance is identified and 10 units of the product are subjected

to this $VD_{\max}^{25}$ or $VD_{\max}^{15}$ dose. If there is $\leq$ one positive sterility test from these 10 irradiated product units, then the dose of 25 kGy or 15 kGy stands substantiated.

*Historically, 25 kGy was identified to achieve a SAL of 10^{-6} . Thus, this was the preferred dose by most primary manufacturers. However, with the introduction of the standard ISO 11137 in the year 2006, doses lower than 25 kGy have been identified and permitted to provide a SAL of 10^{-6} .

What

Methods of sterilization

The established methods of sterilization are:

- o High temperature/pressure sterilization (by dry heat or moist heat)
- o Chemical sterilization (e.g. Ethylene oxide gas)
- o Filtration
- o Radiation sterilization (Gamma/Electron beam)

In selecting the sterilization process, due weightage should be given to ensure product quality is maintained. The method of choice should not affect polymers, seal strength, label and box adhesion, corrugated and paperboard strength, and also the product's color. Radiation sterilization, being a cold process, helps address these issues more effectively compared to others. Furthermore, this facilitates sterilization of pre-packed, ready-to-use products without any quarantine restrictions.

The method covered in this chapter is about Radiation sterilization using:

- o Radioactive gamma sources and
- o Electron beam accelerators

In both cases, ionizing radiation facilitates sterilization. Described below are some aspects of such facilities.

Irradiation facilities using radioactive gamma sources

Two radioisotope sources can be used in irradiation facilities; namely Cobalt – 60 (^{60}Co) and Cesium – 137 (^{137}Cs). Of these, ^{60}Co is commercially more popular as it can be conveniently produced from nuclear reactors in a large quantity. Typically, a ^{60}Co pencil can be used for about 20 years in an irradiator. (Pencils containing slugs or pellets of ^{60}Co are prepared by irradiating 99.9% pure sintered Cobalt powder in Zirconium alloy capsules in a nuclear reactor for about 18–24 months, during which ^{59}Co is converted to ^{60}Co). Periodical replenishment of the source is necessary to compensate for the decay of the radioactive source material (the half-life of Cobalt-60 is ~ 5.28 years).

Various types of commercial irradiators can be designed. The irradiators mainly have two variables; namely, the activity level of radiation source and the product movement around the source for uniform absorption of radiation. The activity level (capacity) of the radiation source may vary from a few thousand Curies to several million Curies. The installed capacity of a facility is always less than or equal to the designed capacity. The dose that a product receives depends on the installed source activity and the residence time in the irradiator. The dose is controlled by adjusting the time that the product is exposed to the source, i.e., time in an irradiation cell, which depends upon the conveyor speed (the product moves around the source). The irradiation time needs a periodic adjustment, which depends on the source strength, since the source decays as a function of time.

It is prudent to design irradiators to maximize energy utilization and dose uniformity in products. This generally poses a challenge due to the wide range of products to be irradiated.

There are two types of Irradiator designs popularly used by the trade:

1. *The Product overlap* design—where two rows of stacked metal containers (placed one above the other) known as totes are used, which move in four/six paths around the source (two/three on either side). A tote is versatile to be able to accommodate bags, cartons or drums. The combined height of the two totes, stacked one above the other, is greater than the height of the source rack, and thus the term 'Product overlap.' Shuffling of the totes is essential to ensure dose uniformity in the product. Since most of the radiation emitted by the source is intercepted by the tote arrangement (and, therefore, absorbed by the product), energy utilization is high in such facilities. It is important to align the center of the product stacks to the center of the source rack to achieve optimum dose uniformity.

An example of a product overlap design is the pallet irradiator where a loaded pallet moves around the source, which helps save time as one can irradiate the packed pallet as a whole in its final form (Fig. 2A).

2. The second design is the *Source Overlap* design, where the height of the product carrier is shorter than the height of the source rack. Though energy utilization is lower, the product conveying systems are less complex, with comparable dose uniformity (Fig. 2B).

Fig. 3 provides an isometric view of a Product overlap Gamma irradiator.

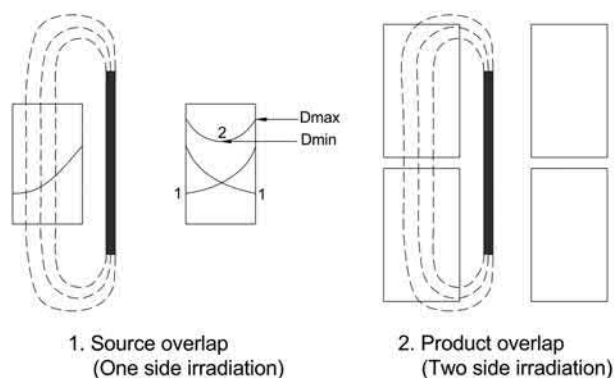


Fig. 2 Diagrammatic representation of Source Overlap and Product Overlap Irradiator design.

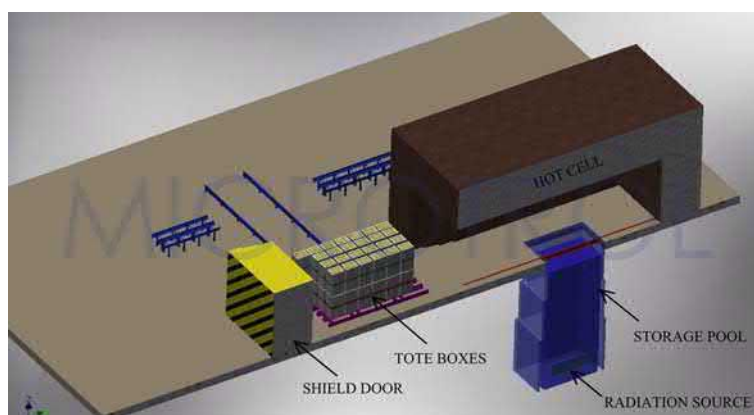


Fig. 3 Diagrammatic representation of a Batch Irradiator.

Irradiation facilities using Electron beam (EB) accelerators

Electron beam accelerators produce intense high energy electron beams (up to 10 MeV), enabling such irradiation facilities, the advantage of processing higher throughputs compared to gamma sources. Energy utilization in such facilities is also higher as the beam is directed onto the product and energy wastage is minimum, as the EB facility can be switched ON or OFF depending upon the treatment requirements.

These accelerators are broadly classified, based on the energy range: Low, Medium and High energy accelerators of which medium (4–5 MeV) or high energy (up to 10 MeV) are ideal for sterilizing packed products. In EB irradiators, optimal utilization is achieved if the products are uniformly packed.

Industrial accelerators could either be Direct Transformer Accelerators operating in DC mode (ELV, Dynamitron, Cockroft-walton etc.), or RF based single resonant cavity accelerators (ILU type), or Linear accelerators (linacs) operating in pulse mode (IAEA, 2008). The accelerators today are designed to operate on 24 × 7 basis with minimal manual intervention and maintenance.

Product packs (cartons, etc.) are conveyed below the beam window where the scanned electron beam is made to impinge on the product. The energy absorbed by the product per unit mass varies, based on the extent to which the atoms (of product) scatter the electrons (from the beam) in all directions. The electrons gradually slow down after each scattering, and this slowing down increases the energy absorption efficiency by the product.

A typical depth dose profile within a product is shown below (Fig. 4).

From the figure, it can be observed that the dose is minimum at the surface, which progressively increases (with increasing thickness of the product) due to scattering that happens within the product. The dose peaks at a particular product thickness referred to as D_{max} , and subsequently the dose decreases with increasing product width. The dose at the surface of the product is known as the entrance dose. The depth within the product that equals the entrance dose is called the useful range, which is the effective thickness of the product that electron beam can bring about the desired sterilization/change in the product, with minimal DUR (Dose Uniformity Ratio).

The D_{min} position is usually at the surface of the product. The above (Fig. 4) one-sided radiation profile is for water. However, for products having lower density, the useful range would increase whereas the Dose Uniformity ratio (DUR) would decrease. The inverse happens for products with higher density i.e. the useful range decreases but DUR increases. From the profile, it can be

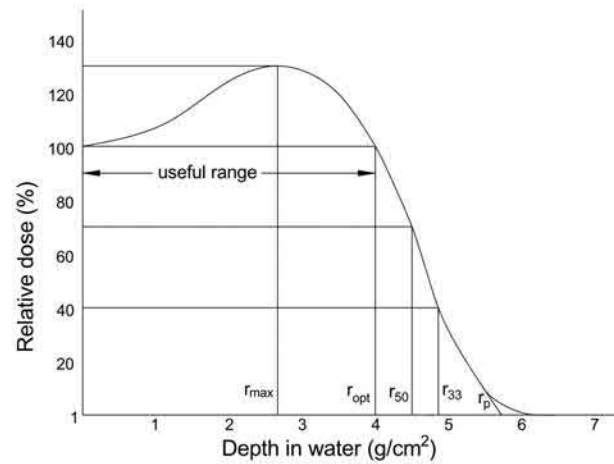


Fig. 4 Depth dose profile for 10 MeV electron beam.

inferred that the useful range for a unit product density is 4 cm thickness and the DUR i.e. $D_{\max}/D_{\min} = 130/100 = 1.3$. In order to increase the process thickness (the useful range), the product needs to be exposed from both sides. The depth dose profile for a 2-sided irradiation is shown in **Fig. 5**. The treatment efficiency is improved since all the energy is deposited within the product.

Upon exposure to electron beams from both sides for a product pack, the minimum dose received would be at the surface—as well as the point where the curves from both sides intersect. As shown in **Fig. 5**, if the product pack thickness for one-sided irradiation is 'x', then for double – sided irradiation the thickness of the subject product pack would exceed twice as much.

Electron penetration is proportional to its energy and inversely proportional to the product density, which is represented in the basic formula (for electron energies greater than 1 MeV):

$$\text{Penetration (cm)} = (0.524 E - 0.1337) / \rho$$

where E = Beam energy in MeV, ρ = density of product in g/cm^3 .

During routine operations of the EB irradiator, two parameters are essential for delivering a desired dose; namely, beam energy and current. The beam energy determines the maximum thickness of a product that can be effectively penetrated, whereas the current helps regulate the dose rate and, thus, the time of irradiation at which the product is irradiated.

The following gives the relation between the process thickness, d (in cm) and energy for 1-sided irradiation:

$$E = 2.63 d \rho + 0.32 \text{ (for 1-sided irradiation)}$$

However, as explained earlier, 2-sided irradiation would help in slightly more than doubling the product thickness that can be subjected to a particular dose with a known DUR as follows:

$$E = 1.19 d \rho + 0.32 \text{ (for 2-sided irradiation)}$$

A typical layout of an EB accelerator with various equipment is shown in **Fig. 6**.

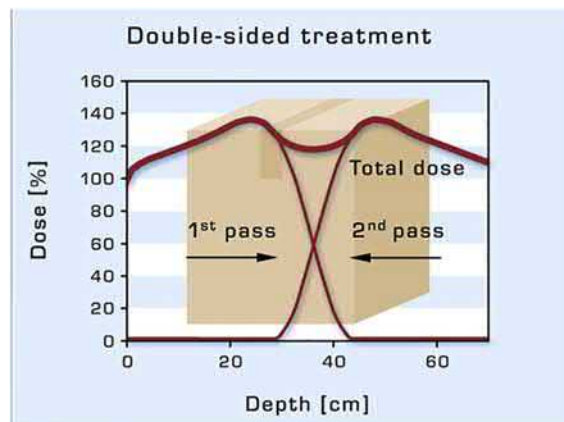


Fig. 5 Depth dose profile for electron beam radiation for two side irradiation.

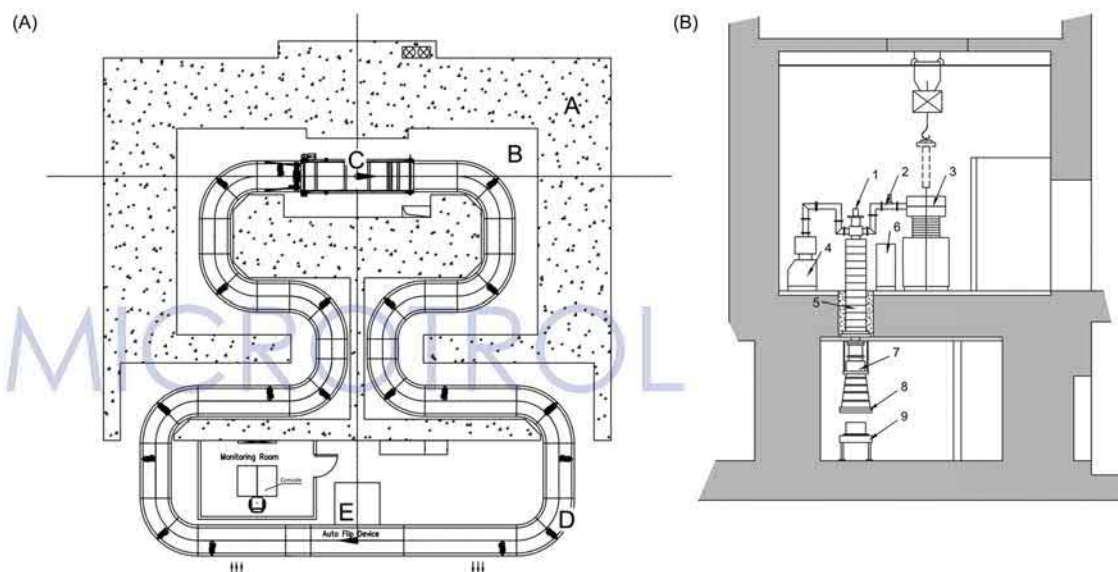


Fig. 6 (A) Layout 'Plan' of an EB facility where the product on conveyor, passes under the beam window (A) Shielding Cell (B) Irradiation Cell (C) EBeam (D) Conveyor (E) Product Flipping System. (B) Sectional layout of an EB Facility. 1. Electron gun, 2. Waveguide, 3. UHF generator, 4. Forevacuum pump, 5. Accelerating section, 6. Gun modulator, 7. Scanning magnet, 8. Output window, 9. Conveyor.

Commissioning (installation qualification and operational qualification)

In order to deliver a particular dose repeatedly, the operating conditions need to be set at the time of installation of an irradiator. This is achieved during commissioning studies where dose-distribution, throughput and reproducibility of absorbed dose for a homogenous material in a typical range of density, is related to operating parameters.

Installation qualification establishes that the entire equipment and its associated processing and measuring instruments are installed as per specification, whereas Operational Qualification establishes that the installed irradiator can operate within specified limits.

Most commercial irradiation facilities involve movement of product carriers around the source. Due to this, some positions in the product carrier would absorb maximum dose and some, the minimum dose. In order to locate these positions in the product carrier, extensive dose mapping is carried out.

Gamma irradiators

In a commercial gamma irradiator, these studies involve 3 sets of trials; namely, 'X' run, 'Y' run, and 'Z' run.

The 'X' run helps determine the alignment of the source rack with respect to product, the dose received at the center of product carrier, and the cycle time for the run (Fig. 7).

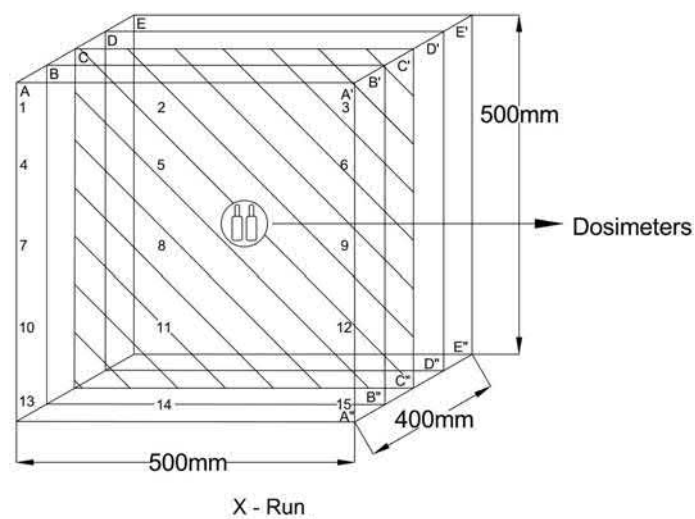


Fig. 7 Placement of dosimeters in X-Set at C-8 position.

The 'Y' run helps determine location of minimum and maximum dose inside product carrier. A considerable number of dosimeters (around 10–50 dosimeters) are placed in a systemic 3-dimensional form within the product carrier, with homogenous reference material, and irradiated for the cycle time determined in the 'X' run (Fig. 8).

The 'Z' run provides a statistical evaluation of the minimum and maximum dose positions and proves repeatability of dose distribution throughout the irradiator (Fig. 9).

Parameters like source strength, source arrangement, conveyor speed, dwell time, multi-pass mode, and irradiation geometry and bulk density of product are to be characterized for its complete range before commercial operations commence in an irradiator.

During gamma irradiation, gamma rays imparts energy (i.e. radiation dose) through atomic interactions such as photoelectric effect, Compton scattering (more predominant) and pair production (Maruyama, 2021). Due to these interactions, the intensity of the gamma rays decreases as they travel through the product. To attain a uniform dose in the product, the product container is rotated 180° during irradiation. The depth-dose distribution during one-side and two-side pass, around the source rack, is depicted in Fig. 10. The one-sided pass around the source results in a decrease in absorbed dose with increased product depth. However, the total dose due to irradiation from both sides is indicated by the curve 'a + b'. The dose achieved in the product by 2-sided irradiation is more uniform than by single side radiation processing (Fig. 10).

EB irradiators

The Operational Qualification of an electron beam irradiator facility involves characterization of parameters such as the mean energy of the electron beam, beam spot profile and scan width (Fig. 11). The energy of the electron beam is determined by the

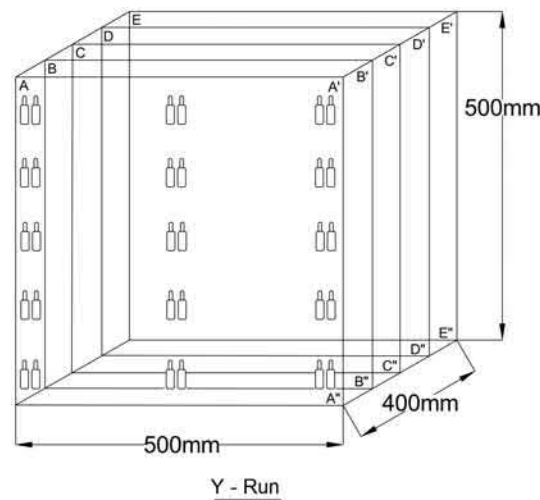


Fig. 8 Placement of dosimeters in Y-Set at 1–15 position in planes A - E.

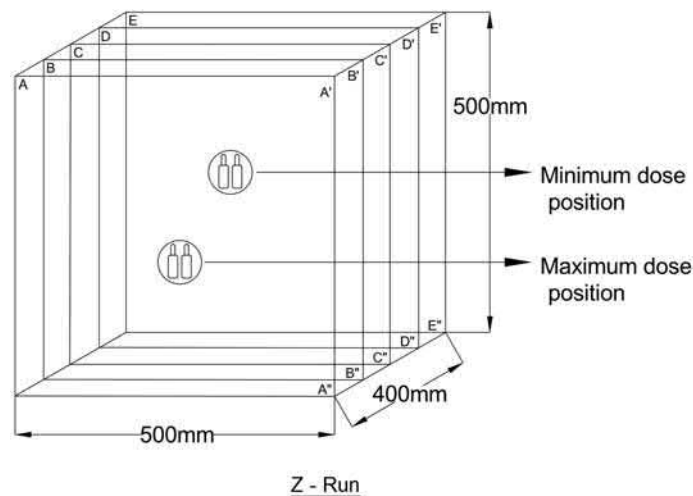


Fig. 9 Placement of dosimeters in Z-Set at A – 8 and C -8 positions, which are the maximum and minimum dose positions respectively.

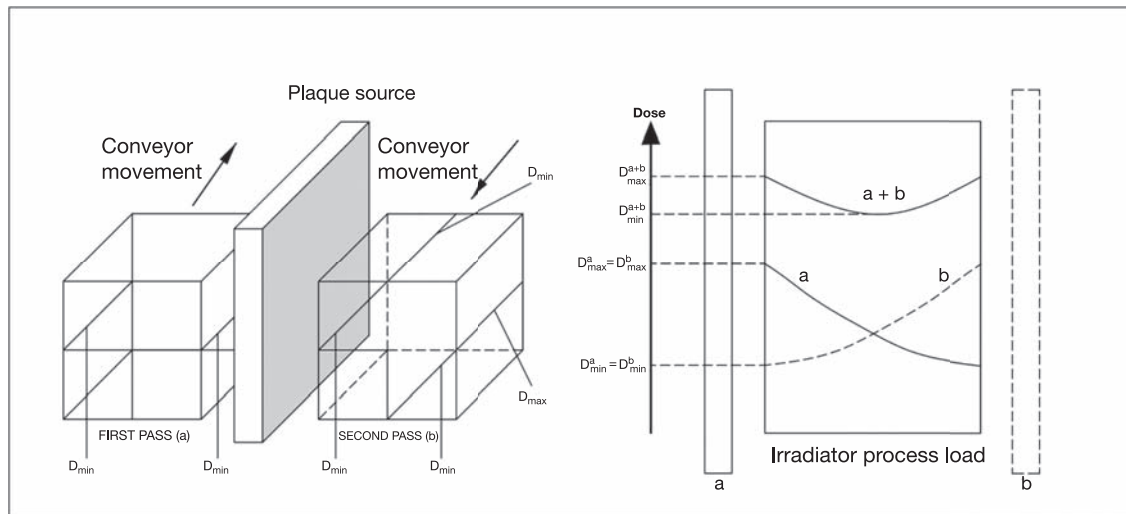


Fig. 10 Dose distribution in a two - pass gamma radiation facility. Maximum and Minimum dose positions in a rectangular product that is irradiated and its cumulative dose distribution curve.

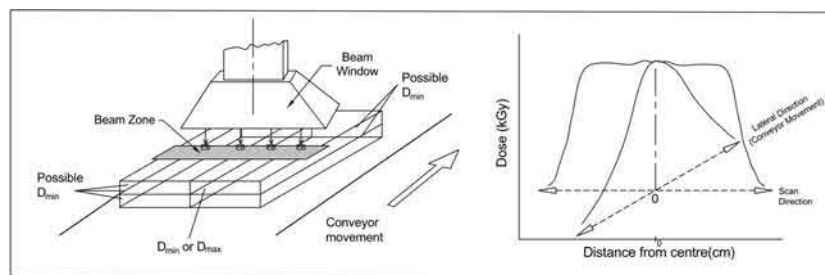


Fig. 11 Dose distribution along the Scan and Lateral direction of an electron beam irradiator.

depth-dose distribution study, which is conducted along the beam axis, with reference material that is generally polystyrene, water, graphite or aluminum.

The mean electron beam energy ($\bar{E}_0$) and the most probable beam energy (E_p) are determined using the formula

$$E_0 \text{ (MeV)} = 2.33 \times R_{50}$$

$$E_p \text{ (MeV)} = 0.22 + 1.98 R_p + 0.0025 R_p$$

where R_{50} is the depth at which the dose becomes half its maximum, expressed in cm in water; R_p is the practical range expressed in cm in water.

Dosimetry system

A dosimetry system helps in determining the absorbed dose. The dosimetry system consists of

- Dosimeter: This is any material wherein irradiation brings about a change that is measurable and repeatable.
- A measuring instrument: An instrument that measures the change in a dosimeter post irradiation.
- Procedure for analysis: A manual calculation of absorbed dose may result in errors. Thus, suitable software is available for computation and analysis of absorbed dose.

Based on intrinsic accuracy and application, dosimeters are classified into Primary, Reference, Transfer or Routine dosimeters. Various aspects of the dosimetry system are given in [Table 1](#) below.

Most gamma irradiators use alanine or ceric-cerous dosimeters for routine process monitoring, whereas thin film dosimeters are the preferred dosimeters in Electron beam irradiation facilities for routine process control ([Eckerman, 2021](#)).

Table 1 Classification of dosimetry systems.

<i>Class and salient features</i>	<i>Calibration</i>	<i>Uncertainty</i>	<i>Examples</i>
Primary • Absolute dose determination with reference only to SI units • Generally maintained by national standard laboratories	Not required	1%	Calorimeter, ionization chamber
Reference • High metrological quality • Effect of parameters need to be characterized	Required	1–2%	Calorimeter, alanine, dichromate, ceric-cerous, Fricke
Transfer • Used to transfer dose information from accredited calibration laboratory to irradiation facility • Generally reference standard dosimeters	Required	1–2%	Alanine, dichromate, ceric-cerous, Fricke
Routine • Used for dose mapping and routine dose monitoring • Shows variation from one manufacturing batch to another	Required	3–5%	PMMA, radiochromic and cellulose triacetate (CTA) films, ceric-cerous, ethanol-chlorobenzene (ECB)

Performance qualification

During Performance Qualification, values for various process parameters (operational as well as product and its packaging) are characterized. These parameters consistently provide sterile product with a high degree of confidence. Performance Qualification involves Dose mapping with actual product.

The position in at least one container in each irradiation path receiving the maximum and minimum dose is identified during Performance Qualification. The product, its packaging and its loading configuration in the irradiation container, is specified during the Performance Qualification studies and the same is to be followed during routine processing.

Depending on product homogeneity, a considerable number of dosimeters are labeled and affixed in an irradiation container. After irradiation, these dosimeters are carefully removed and the dose received is analyzed. The positions with maximum and minimum doses are identified, as well as the DUR. Ideally, the DUR calculated should be lower than the DLR. If it is vice versa, then process parameters would have to be adjusted in order to comply with this criterion, which is a pre-requisite for routine processing.

In some cases, the minimum dose could be in the center of the irradiation container. Hence, placing dosimeters for routine monitoring would require tampering with product packing. In such situations, a reference point at the surface is identified. The co-relation between this reference point and the minimum point is to be determined during Performance Qualification studies. The absorbed dose at both these positions should be reproducible.

During the Performance Qualification exercise, the exposure time requirement and processing rates are also determined.

This exercise is to be repeated whenever there is a change in source configuration/replenishment.

Routine sterilization and its control

After the Performance qualification studies are completed, the process parameters that a product category is to be subjected for sterilization is determined. Once the product is received for treatment, the incoming documents are reviewed for correctness with respect to count and product damage. Discrepancies, if any, are to be resolved before further processing of the consignment. Upon receipt, the consignment is stored in a designated area for Unprocessed Material with appropriate labels. These labels should contain details like the product name, consignment size, product lot number, etc. These labels may also be affixed with 'Go – No go' chemical indicators, which would help identify the stage of processing.

During processing, dosimeters are to be placed at positions identified as minimum dose and maximum dose identified during Performance Qualification studies. If the minimum dose position is inaccessible, the dosimeter is to be placed at the reference position, which is also identified during Performance Qualification studies. Loading of product into irradiation containers should be the same as done during Performance Qualification studies.

Irradiation of the product is closely monitored for the dose, exposure time in the radiation field, the source activity, product loading pattern and bulk density and distribution of product within the irradiator. Dosimeters placed within the product are retrieved carefully and analyzed for absorbed dose, post-irradiation.

At post treatment, the consignment is placed in the area demarcated for storage of processed products. The products received for treatment should follow unidirectional flow from Unprocessed to In-process to Processed storage areas. The products are then dispatched after review of processing records by authorized personnel.

Recent developments

The universally accepted SAL of 10^{-6} originated from the food canning industry, which was further adopted by the aerospace industry and the healthcare industry. A recent thought process that is emanating among researchers is challenging the requirement

of such a high SAL. Probabilistic models have been analyzed that suggest a lower SAL of $10^{-5}/10^{-4}/10^{-3}$ or for that matter even 10^{-2} may suffice for terminal sterilization of a medical device—especially those that have been aseptically manufactured.

Where

Irradiation facilities are designed and constructed after considering various factors at a site approved by the Regulators. The load bearing capacity of soil is an important factor, since the facilities are comprised of heavy concrete structures essential for biological shielding for personnel safety.

When

For pre-packed medical products, terminal sterilization radiation processing is the method of choice due to its inherent advantages, i.e., processing at ambient temperatures and adequate penetration.

Who

In the irradiation industry, two bodies are identified; namely, the primary manufacturer and the irradiation operator.

The *primary manufacturer* is the body who shall be responsible for the design and manufacture of the medicine, i.e. medical device or drug. They are also responsible for the safety of the medicine and to ensure its performance when it is placed in the market. Apart from controlling the medical devices manufacturing process (specifications like product density, orientation, dimensions etc.), the primary manufacturer is also responsible to develop product families, establish the maximum acceptable dose for the various product families, establish their sterilization dose, product performance qualification and product release into the market.

The *irradiation operator* is responsible for irradiation of the product. Their responsibilities are limited to the Installation Qualification, Operational Qualification, controlling the irradiation process, certification of the radiation dose subjected to the product, develop processing categories, and controlling changes to the irradiator.

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Medicine: Radionuclides Used in Nuclear Medicine

Meera Venkatesh^{a,*} and Keon Wook Kang^b, ^a Division of Physical and Chemical Sciences, International Atomic Energy Agency, Vienna, Austria; and ^b Department of Nuclear Medicine, Seoul National University College of Medicine, Seoul, South Korea

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Glossary

Alpha particles Alpha (α) particle or alpha rays, are particles with two protons and two neutrons, identical to a helium-4 nucleus, with an overall charge of +2. It is represented as ${}^4\text{He}^{+2}$.

Beta particles Beta (β) rays or beta particles are a type of particulate radiation with the mass of an electron emitted by certain radioactive nuclides. There are two kinds of beta rays. β^- rays have a charge of -1 and are identical with electrons. They are also known as negatrons. β^+ rays have a charge of $+1$ and are identical with positrons, which is the anti-matter of electron. Generally, the term “beta rays” is used to refer to β^- rays (negatrons) and β^+ rays are referred to as positrons. β^+ particle when emitted, travels until it encounters an electron and they get annihilated, their mass emitted as two photons of 511 keV (equivalent to the mass of an electron in accordance with $E=mc^2$) each at opposite (at 180 degrees) directions.

Computed Tomography (CT) Tomography is a three-dimensional image of a structure. When a series of plane cross-sectional images are used to construct a tomographic image using a computer to process the data, it is known as Computed Tomography or CT. In medical imaging field, X-rays are used for imaging to construct CT scans. As the CT image gives precise anatomical information, a combination of NM imaging and CT image is used for accurate diagnostic imaging.

Decay constant (λ) The decay constant of a radionuclide is the probability of the radioactive decay of a radionuclide. If there are N atoms of a radionuclide, then the number of nuclei that will undergo decay per unit time is $N\lambda$. This number is also

*Retired.

referred to as the “activity.” The larger the λ , the larger the activity per unit mass (also known as specific activity), and shorter is the half-life $T_{1/2} = 0.693/\lambda$.

Diagnostic nuclear medicine The branch of nuclear medicine used for diagnostic purposes.

Dosimetry Calculation of the radiation dose delivered to the target tissue and the various vital.

Electron capture-EC Electron capture is a mode of transmutation wherein a neutron deficient nucleus of a radionuclide captures an electron from one of the orbiting electrons, resulting in conversion of a proton into a neutron along with the emission of a neutrino (to balance the momentum). The transmuted/transformed nuclide has the same number of nucleons (mass number A) but atomic number (Z) less by one.

^{18}F FDG [^{18}F]-2-Fluoro-2-deoxyglucose, a radioactive compound used extensively in PET procedures.

Gamma rays Gamma (γ) rays are photons and are electromagnetic radiation. They have no charge or mass and are emitted from the nucleus of an atom when the atom undergoes transmutation. Emission of gamma rays does not change the atomic number or the mass number of the nuclide. The change is only in the energy levels within the nucleus.

Half-life The half-life of a radionuclide, denoted as $T_{1/2}$, is the time taken for one-half of the nuclei to decay (transmute spontaneously into the daughter nuclide by emission of radiation). The half-life depends on (and inversely proportional to) the probability of transmutation, which is also denoted as the “decay constant” represented as λ ; the larger the λ , the shorter the $T_{1/2}$. ($T_{1/2} \propto 1/\lambda$ and $T_{1/2} = 0.693/\lambda$).

Isotope Isotopes are atoms of the same element (same atomic number or number of protons(Z) in the nucleus), differing in the numbers of neutrons and hence different mass numbers(A). An atom is represented as ^A_ZX where X is the symbol of the element. However, since each element has a unique fixed “Z,” often the isotopes/radionuclides are expressed as ^AX . In this article, the radionuclides are expressed as ^AX or X-A or the full form of element name followed by its mass number. Example: Technetium-99m is represented either as Tc-99m or $^{99\text{m}}\text{Tc}$; Iodine-131 is represented either as I-131 or ^{131}I .

In Vitro In glass or outside the body.

In Vivo Inside the body.

LET Linear energy transfer—This term is used to indicate the capacity of a particulate radiation (such as alpha or beta rays) to transfer their energy into the matter they travel. Particulate radiations lose their energy upon interaction with atoms in their path of movement.

Molecular imaging This term defines an imaging modality in which the signal for imaging arises because of bio-molecular action/path of the tracer/probe used. Nuclear medicine imaging, wherein the radiopharmaceutical tracer acts at molecular level to provide information of bio-chemical action inside the body, is the most used “molecular imaging” modality.

Monoclonal antibodies-MAB These are antibodies that are derived from the same “clone” or a single antibody producing cell that has been multiplied, having identical characteristics and behavior.

Positrons Beta particles with +1 charge. See above the note on Beta particles for details.

Radioactive decay The process of transmutation/transformation of the nuclide of the radioisotope to a more stable isotope with emission of radiation (such as α , β , γ , X radiations or electrons). The radioisotope is referred to as the “parent” and the resulting nuclide after the transmutation is referred to as the “daughter.” When the parent nuclide transmutes from a higher nuclear state (metastable state) to the ground state of the same isotope, the decay is referred to as “Internal Transfer” (IT). The parent nuclide is denoted as $^A\text{mX}_Z$ and the daughter nuclide as $^A\text{X}_Z$ or $^A\text{gX}_Z$, wherein “m” indicates metastable state and “g” the ground state. Example: Technetium-99m ($^{99\text{m}}\text{Tc}$) decaying by IT to $^{99\text{g}}\text{Tc}$ (generally represented as ^{99}Tc).

Radionuclide An isotope of an element, the nucleus of which is unstable and transmutes (transforms; decays) into a more stable nucleus by either emission of one or more kind of radiations (α , β^- , β^+ , γ) or by electron capture.

Radiopharmaceuticals A drug formulation containing radioactivity, used in nuclear medicine for diagnostic imaging or therapy.

RBC Red Blood Corpuscles or Red blood cells are also known as erythrocytes and carry oxygen throughout the body.

RID RadioImmuno Diagnosis; a nuclear medicine procedure wherein a radiolabeled antibody (generally monoclonal antibody) is used for diagnostic purpose.

RIT RadioImmuno Therapy; a nuclear medicine procedure wherein a radiolabeled antibody (generally monoclonal antibody) is used for therapeutic purpose.

Therapeutic nuclear medicine The branch of nuclear medicine in which radiopharmaceuticals are labeled with radionuclides with therapeutic potential (high linear energy transfer).

Abbreviations

PRRT Peptide receptor radionuclide therapy

PET Positron emission tomography

SPECT Single Photon Emission Computed Tomography

Introduction

Nuclear Medicine (NM) is a specialty in which a radiolabeled molecule known as a “Radiopharmaceutical” is used for the management of diseases, either through diagnosis or treatment or monitoring treatment progress. Historically, radionuclides were used for the treatment of diseases soon after the radioisotopes were available, with radium being used to treat tuberculosis lesions as early as 1901. Several brilliant inventions and innovative ideas laid the foundation for the use of radioisotopes in medicine (SNMMI, 2020; EANM Publications-Tech Guide, 2019).

In brief, the invention of the cyclotron (Ernest Lawrence, 1930), the artificial production of radioisotopes (Irene Joliot Curie and Frederick Joliot, 1934), the concept of using radioisotopes as tracer inside a living being (Georg de Hevesy, 1935), and the achievement of controlled nuclear fission that formed the basis of nuclear reactors (Enrico Fermi, 1942), could be considered as the major factors that accelerated the growth of nuclear medicine. Radioisotopes produced using cyclotrons and nuclear reactors were explored for use in medicine, both for diagnosis and for therapy. Among the radioisotopes explored and used initially, Iodine-131 used in the treatment of thyroid cancers is an outstanding example, which continues to be used to this date (Fahey et al., 2017). With the growth in the nuclear industry and establishment of nuclear reactors and cyclotrons, which produce radioisotopes, the use of radioisotopes in nuclear medicine grew and expanded rapidly both in volume and variety. The advances in related fields such as molecular biology, immunology, cancer biology, radiolabeling techniques, automation, radiation detectors, computation, etc. aided significantly in the growth of nuclear medicine.

While nuclear medicine involves the ‘in-vivo’ administration of radiolabeled molecules, “in-vitro” use of radiolabeled molecules played a crucial role during 1970s and 80s. Rosalyn Yalow won a Nobel Prize in 1977 for discovering *Radioimmunoassay* (RIA), a system using radioisotopes for measuring minute quantities of hormones and biomolecules in human samples, like serum, with high sensitivity and specificity. This revolutionized the practice of endocrinology and oncology (Yalow Rosalyn, 1973).

A related procedure known as “Immunoradiometric assays” (IRMA), with higher specificity, were developed in later years. RIA and IRMA grew rapidly in the following decades and the availability of monoclonal antibodies gave a boost to development of such assays. “RIA/IRMA kits” that could be used in pathology labs for measuring nearly all the important hormones and several cancer antigens were developed. Iodine-125 is the most used radionuclide in these “in vitro” tests. While RIA/IRMA kits are still used for measurement of a few hormones in a few countries, most of these assays have shifted to the use of other tracers such as enzymes or fluorescent markers. The longer shelf-life of enzyme or fluorescent labeled biomolecules in comparison with radiolabeled molecules and the possibility to automate enzyme/fluorescent assays to have huge throughput are the reasons for the shift away from radiometric assays in pathology labs. However, radiometric assays are deemed to be the gold standard and are still developed and used to measure biomolecules in research (such as in new drug development).

Nuclear medicine has steadily grown over the past decades (Anderson et al., 2019; Ramamoorthy, 2020). Diagnostic nuclear medicine is predominantly used in oncology, cardiology, neurology, and functional imaging of many of the human systems. It is noteworthy that the World Health Organization (WHO) survey in 2018 showed that ~71% of the deaths globally (~41 Million people/year) are due to non-communicable diseases, predominantly cardiovascular diseases (17.9 M/year), followed by cancers (9.0 M/year). In the management of these diseases, diagnostic nuclear medicine provides invaluable information to the doctors for making an accurate diagnosis, planning treatment and monitoring patients for effective therapy.

Therapy in nuclear medicine is most often used for treating cancers and a few other conditions such as hyperthyroidism. Though therapy aims at delivering radiation dose to the target organ/lesions, it is important to ensure that the radiation dose is accurately delivered to the desired tissues. Hence, monitoring therapy through imaging is commonly followed.

Currently, nuclear medicine offers diagnostic procedures to image all the important organs and functioning of vital organs and therapeutic options for several disease conditions. Nuclear medicine imaging is also known as “molecular imaging” as it enables anatomical as well as functional (characterization and measurement of biological processes at molecular/cellular/whole organ level) evaluation. It has evolved as an important branch of medicine, providing unique and unequivocal support for patient management, especially for patients with cancers or heart diseases. This article focuses on the radionuclides currently used in nuclear medicine, as well as others now under development.

Radionuclides used in nuclear medicine

The radionuclide (RN) used in radiopharmaceuticals primarily depends on the use to which it is put (Mettler and Guiberteau, 2019). The mode of decay and the type, as well as energy of radiations emitted, are the important factors in guiding whether a radionuclide can be used in diagnostic or therapeutic NM. For diagnostic imaging, the radiation from the RN should provide information on its location, while for therapy the radiation should destroy the cells at its targeted location. The interaction of photons (Gamma rays or X-rays) with matter is quite different from the interaction of charged particles (beta electrons, positrons or alpha particles), primarily due to the charge and mass they carry. Photons are chargeless electromagnetic rays with negligible mass and are highly penetrating. When a radiopharmaceutical with a gamma emitting RN is localized inside the body, the emitted gamma radiation travels through the layers of body tissues and can provide information about the location of the radiopharmaceutical. Hence, radionuclides that decay with the emission of gamma rays can be effectively used in diagnostic imaging. On the other hand, radiations that have charge as well as mass are quickly stopped through interaction with

surrounding atoms, and they transfer their energy through such interactions. When a radiopharmaceutical with a RN that emits beta or alpha particles or Auger electrons is localized in a tissue, these radiations deposit their energy very close to their location, which in turn results in the damage or destruction of the nearby cells. The range of action would depend on the energy they carry (Saha, 2006; Nichols and Byrne, 2021). While the easy traceability of radiation and its quantitative measurement aid in diagnostic imaging, high energy transfer to the diseased cells to destroy them is necessary for therapeutic effect. Thus, the RNs ideal for use in diagnostic radiopharmaceuticals are completely different from those suitable for internal therapy.

Radioactive nuclides are unstable isotopes that undergo nuclear transformation (also referred to as transmutation or decay) to form a new nuclide at a lower energy level by emitting radiations. The difference in the energy levels of the two nuclides is carried by the radiations that emanate. The nature of the radiation emitted (α , β or γ or any combination of these) will depend on the nuclear state of the radionuclide. Radiations are energetic and travel with varying speeds, depending on their energy and their masses. The way they traverse in a medium and the mode of their energy loss will depend on the nature of the charge they carry, their mass and the medium in which they traverse. Being the most massive among the radiations mentioned above, the particulate α rays are generally highly energetic and lose their energy by collisions within a very short distance. Thus, their Linear Energy Transfer “LET” is very high and their range is very short (in microns). The particulate β rays (electrons or positrons) travel a little further than the α rays and have ranges of few millimeters, while the electromagnetic γ rays traveling at the speed of light have the maximum range.

The various applications of the radionuclides are due to one of the following characteristics:

1. radiations that are energetic and deposit their energy in the medium they travel;
2. radiations that are easily measurable at very low levels and, hence, can act as tracers; or
3. radiations that can alter the material in which they travel due to the energy deposition.
4. the radiation itself undergoes changes in its intensity and energy (attenuation of energy) depending upon the material in which it travels.

Apart from the mode of decay, the other physio-chemical parameters such as their half-life ($T_{1/2}$) and chemical feasibility to label different biological molecules influence the use of RNs in nuclear medicine. Further, for regular use in clinics, the RNs must be available in an economically viable manner. Hence, an amenable production route to avail required amounts of the RN in a pure form is important. In this context, key parameters include the route of production, their rate of production, the availability, the purity, and the isotopic abundance of the target nuclide. Radionuclidic impurities are of great concern due to the unnecessary radiation dose these impurities can impart to the patient. Based on the radiation doses delivered to critical organs, tolerance limits are set for RN impurities in RN preparations.

Desirable radionuclide properties for diagnostic nuclear medicine

For diagnostic nuclear medicine, RNs of high desirability include the following properties: they (1) emit photons that can be imaged with high sensitivity, (2) have a short half-life ($T_{1/2}$) (generally few hours to a few days), (3) exhibit amenable chemistry, and (4) are devoid of particulate emission. Gamma rays in the range of 100–200 keV energy are suitable for imaging using a gamma camera (which generally employs NaI(Tl) crystals for detection). Annihilation photons (511 keV) from positron emitters, when measured (using appropriate detectors) in a coincidence mode, provide images with high resolution. Hence, positron emitters are very useful in diagnostic nuclear medicine.

In the early stages of nuclear medicine, planar gamma cameras were used to obtain two-dimensional images of the distribution of the gamma ray emitting radiopharmaceutical inside the body and were useful in delineating abnormalities in organs, aiding in diagnosis. But, with the advances in detector and computation technologies, suitable techniques were quickly used in NM imaging to image the body with array of detectors and compute the readings to obtain three-dimensional imaging with high sensitivity. The two 3-D imaging techniques include

- A. Single Photon Emission Computed Tomography (SPECT)—where the RN is a gamma emitter used for imaging (Khalil et al., 2011); RNs emitting a single photon in high yield are preferred, and
- B. Positron Emission Tomography (PET)—where the RN is a positron emitter (Shukla and Kumar, 2006).

These two systems are commonly used in clinical NM imaging and, in many clinics, have replaced the planar gamma cameras.

In comparing these two approaches, PET images have better resolution (2–3 mm) than SPECT (4–5 mm), since the positrons result in a pair of diametrically opposite photons on annihilation and, hence, can be measured in a coincidence mode to reduce noise. Diagnostic imaging is very often used to investigate any malfunctioning tissues/lesions in the organ systems, and quantification of the radioactivity distribution is important for accurate diagnosis and treatment planning. In this context, spatial resolution and precise anatomic localization of the diseased tissue are very important. Hybrid imaging, wherein the NM imaging is carried out in conjunction with an anatomical imaging modality, such as Computed Tomography (CT), enables accurate location imaging. In the past two decades, hybrid imaging using SPECT-CT and PET-CT have become very popular and used in most NM centers regularly (Mariani et al., 2010; Farwell et al., 2014).

The following Table 1 lists the radionuclides commonly used and those with potential for use in diagnostic nuclear medicine.

Table 1 Radionuclides used in diagnostic nuclear medicine.

Gamma radiation emitting RNs suitable for single photon emission computed tomography (SPECT)					
Radionuclide	$T_{1/2}$	Decay mode	Major E_γ keV (%)	Common production route/s	Remarks
<i>Widely used</i> ^{99m}Tc	6 h	IT	140.5 (89)	$^{235}\text{U}(\text{n},\text{f})\ ^{99}\text{Mo}(\beta^-)^{99m}\text{Tc}$ $^{98}\text{Mo}(\text{n},\gamma)\ ^{99}\text{Mo}(\beta^-)^{99m}\text{Tc}$ $^{100}\text{Mo}(\text{p}, 2\text{n})^{99}\text{Mo}$ $^{100}\text{Mo}(\gamma, \text{n})^{99}\text{Mo}$	Generally availed through the ^{99}Mo - ^{99m}Tc RN Generator. ^{99}Mo $T_{1/2}$ 66 h. Tc-99m is the “work horse” of diagnostic NM. The (n,f) route yields very high specific activity Mo-99, which is preferred The (n, γ) route is used by a few institutes. Over the past decade non-fission routes for Tc-99m production are being pursued
^{111}In	2.8 days	EC	171 (93) 247 (94)	$^{112}\text{Cd}(\text{p},2\text{n})^{111}\text{In}$ $^{111}\text{Cd}(\text{p},\text{n})^{111}\text{In}$ $^{109}\text{Ag}(\alpha,2\text{n})$	The use of In-111 has been reduced after the availability of Ga-68 through a generator for PET use. But, the experience with In-111, a metallic radionuclide with +3 charge in ionic form, has been valuable to label bio-molecules with trivalent radiometals such as Ga-68, using chelating agents such as DTPA
^{123}I	13.3 h	EC	159 (83)	$^{124}\text{Xe}(\text{p},2\text{n})\ ^{123}\text{Cs} \rightarrow ^{123}\text{Xe} \rightarrow ^{123}\text{I}$ $^{124}\text{Xe}(\text{p},\text{pn})\ ^{123}\text{Xe} \rightarrow ^{123}\text{I}$ $^{123}\text{Te}(\text{p},\text{n})^{123}\text{I}$ $^{127}\text{I}(\text{p}, 5\text{n})^{123}\text{Xe} \rightarrow ^{123}\text{I}$	Reactions on Te-123 and I-127 generally yield I-123 contaminated with other isotopes of I
<i>Used in specific studies</i> ^{67}Ga	78 h	EC	93 (37) 185 (20)	$^{68}\text{Zn}(\text{p},2\text{n})^{67}\text{Ga}$ $^{67}\text{Zn}(\text{p},\text{n})^{67}\text{Ga}$	Was used for imaging acute inflammatory lesions and later to image cancer lesions. With availability of F-18-FDG (PET RN), which has better features for imaging cancerous lesions, use of Ga-67 has reduced
^{81m}Kr	13 s	IT	190 (100)	$^{82}\text{Kr}(\text{p},2\text{n})^{81}\text{Rb}(\text{EC}/\beta^+)^{81m}\text{Kr}$	Rb-81/Kr-81m generator; Rb-81 $T_{1/2}$ 4.58 h; 190 keV photons Internally converted. The short half-life of the generator and the product limits its use
^{131}I	8 days	β^-	364 (81); 637 (7)	$^{130}\text{Te}(\text{n},\gamma)\ ^{131}\text{Te}(\beta^-)^{131}\text{I}$; $^{235}\text{U}(\text{n},\text{f})^{131}\text{I}$	Useful in diagnosis as well as therapy owing to abundant gammas and the energetic beta particles. With availability of better diagnostic RNs (non-particulate emitting) use of I-131 is mostly in imaging prior to therapy of thyroid cancers (theragnostic)
^{133}Xe	5.3 days	β^- , γ	81	$^{235}\text{U}(\text{n},\text{f})^{133}\text{Xe}$	Used exclusively in Lung perfusion studies
^{201}Tl	73.1 h	EC	69–82 (95) Hg X-rays 135(3) & 167 (11)	$^{203}\text{Tl}(\text{p},3\text{n})^{201}\text{Pb}(\beta^-)^{201}\text{Tl}$ $^{\text{nat}}\text{Tl}(\text{p},\text{xn}) \dots ^{201}\text{Tl}$	Tl-201 has been specifically used only in Myocardial perfusion studies and is considered to be a gold standard
<i>Limited use</i>					

^{124}I	4.2 days	EC & β^+ (23%)	EC 603 (62.9) $\beta^+ + E_{\text{ave}}$ 2.14, 0.82 MeV	$^{124}\text{Te}(p,n)^{124}\text{I}$	Complex decay scheme of EC & β^+ ; low yield of 511 keV photons; and possible interference due to the 609 keV photon of EC. But could be a good theragnostic RN as it can be used for imaging as well as therapy with the copious Auger electrons
$^{113\text{m}}\text{In}$	1.7 h	IT	290 (100)	$^{112}\text{Sn}(n,\gamma)^{113}\text{Sn}(\beta^-)^{113}\text{In}$	^{113}Sn —118 days $T_{1/2}$; radionuclide generator; but production of Sn-113 has extremely low yields, making practical feasibility difficult

Positron emitting RNs suitable for positron emission tomography (PET)

Radionuclide	$T_{1/2}$	E_{β^+} MeV	511 keV rays%	Common production route/s	
<i>Widely used</i>					
^{18}F	110 months	0.635	194	$^{18}\text{O}(p,n)^{18}\text{F}$	Most widely used PET RN F-18-FDG is the most used PET radiopharmaceutical
^{68}Ga	68 months	1.899	178	$^{68}\text{Ga}(p,2n)^{68}\text{Ge}(\beta^-)^{68}\text{Ga}$ $^{68}\text{Zn}(p,n)^{68}\text{Ga}$	^{68}Ge $T_{1/2}$ 271 days; $^{68}\text{Ge}/^{68}\text{Ga}$ Radionuclide generator
<i>Limited specific use</i>					
^{11}C	20.4 months	0.96	196	$^{14}\text{N}(p,\alpha)^{11}\text{C}$	Since ^{11}C has a short half-life of 20 min, its use in regular clinics is limited. But, since carbon is a native element of all bio-molecules, C-11 labeled molecule would be identical to the unlabeled molecule and, hence, it is a valuable RN to study the biological behavior of molecules such as drugs
^{13}N	9.97 months	1.19	200	$^{13}\text{C}(p,n)^{13}\text{N}$	The short half-life of ^{13}N makes its routine clinical use challenging. But ^{13}N -ammonia has proven to be useful in cardiac studies and is being used via production in hospital cyclotrons
^{82}Rb	1.26 months	3.4	192	$^{85}\text{Rb}(p,4n)^{82}\text{Sr}(\beta^-)^{82}\text{Rb}$	^{82}Sr $T_{1/2}$ 25.5 days; $^{82}\text{Sr}/^{82}\text{Rb}$ Radionuclide generator; Used specifically in heart perfusion studies
<i>Emerging RNs</i>					
$^{44\text{g}}\text{Sc}^a$	3.97 h	0.632	188	$^{44/\text{nat}}\text{Ca}(p,n)^{44}\text{Sc}$ $^{45}\text{Sc}(p,2n)^{44}\text{Ti}$	^{44}Sc is of great interest as it can be a theragnostic pair with ^{47}Sc . ^a Availing ^{44}Sc through a generator from parent ^{44}Ti ($T_{1/2}$ —60 years) is attractive for clinics due to long generator shelf life. However, production of Ti-44 requires high energy cyclotrons
^{45}Ti	3.08 h	1.04	190	$^{45}\text{Sc}(p,n)^{45}\text{Ti}$	Produced using 11–16 MeV protons to avoid long-lived Ti-44. A promising RN due to ease of production

(Continued)

Table 1 Radionuclides used in diagnostic nuclear medicine.—cont'd

Positron emitting RNs suitable for positron emission tomography (PET)					
<i>Radionuclide</i>	<i>T_{1/2}</i>	<i>E_{β+} MeV</i>	<i>511 keV rays%</i>	<i>Common production route/s</i>	
⁶⁴ Cu	12.7 h	0.656	38.6	⁶⁴ Ni(p,n) ⁶⁴ Cu	Unique RN; decays by β ⁺ , β ⁻ as well as Electron Capture. Hence suitable for diagnostic as well as therapeutic NM. A good “Theranostic” RN
⁸⁶ Y	14.7 h	3.141	68	⁸⁶ Sr(p,n) ⁸⁶ Y	Theragnostic pair for Y-90, a pure beta emitter used in therapy
⁸⁹ Zr	3.27 days	0.9	45.6	⁸⁹ Y(p,n) ⁸⁹ Zr	Long half-life suitable for labeling antibodies that require long time for localization. Production amenable
¹²⁴ I	4.18 days	2.138	50	¹²⁴ Te(p,n) ¹²⁴ I	Complex decay scheme of EC & β ⁺ ; low yield of 511 keV photons and possible interference due to the 609 keV photon of EC
<i>RNs with potential for use</i> ¹⁵ O	2.03 months	1.72	200	¹⁵ N(p,n) ¹⁵ O	The half-life of ¹⁵ O is too short for routine clinical use. Yet, ¹⁵ O labeled water is deemed an ideal tracer for myocardial perfusion studies
⁷⁵ Br	96.7 months	1.721	151	⁷⁶ Se(p,2n) ⁷⁵ Br	⁷⁵ Br was explored two decades ago as a promising PET RN. But it has not progressed as expected

^aThere are two Sc-44 RNs. The metastable Sc-44m decays to the ground state Sc-44 with a half-life of 58.6 h, while Sc-44 ground state decays via positron emission or EC, with a half-life of ~4 h. Production of Sc-44 using protons or deuterons leads predominantly to the Sc-44g state. Sc-44m is produced using high energy alpha particle reactions. Currently, Sc-44g is being explored for use in PET, while Sc-44m is in the research stage. The Ti-44/Sc-44 generator route always gives Sc-44 in the ground state. Hence, it is relevant to mention Sc-44g here.

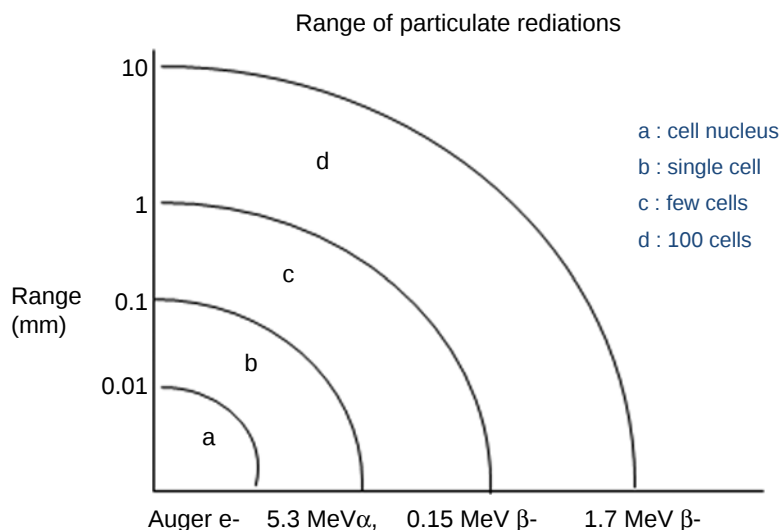


Fig. 1 Range of particulate radiations in human tissues.

Desirable radionuclide properties for therapeutic nuclear medicine

For internal therapy, radiation with high “Linear Energy Transfer” (LET) is effective. So, RNs that emit particulate radiations such as β^- , α , or electrons, combined with a half-life of a few days and amenable chemistry, are suitable for therapeutic nuclear medicine. Many RNs that decay through an electron capture path emit several low energy electrons (Auger electrons/Coster-Kronig electrons and conversion electrons) as the electron capture from the inner orbits leads to subsequent emission of X-rays and Auger or conversion electrons. These low energy electrons are also useful for therapy if the RN is in close proximity to the lesion. For selection, the therapeutic range of the RN is of importance and a RN with therapeutic range matching with the size of the lesions to be treated is chosen.

Fig. 1, showing the range of different particles in tissues, illustrates the importance of choosing the appropriate radionuclide based on the target tissue size.

A wide range of RNs are suitable for NM therapy applications (Goldsmith, 2019). But the factors mentioned earlier, such as half-life, feasibility of production, availability in reasonably pure form without other RN impurities, and amenability to tag to suitable bio-molecules, dictate the actual use in clinical practice. Most of the current approved therapeutic radiopharmaceuticals are tagged with beta emitting RNs, with a preference to those that also emit an imageable photon—since this allows monitoring of the localization of the radiopharmaceutical as well as the ability to calculate the dose delivered to the target and different vital non-target organs.

In recent times, there has been a lot of attention on targeted alpha therapy, as alpha particles have high LET. One alpha emitting radiopharmaceutical ($^{223}\text{Radium chloride}$ for bone pain palliation) is approved (Deshayes et al., 2017), while several are under clinical trials.

The commonly used and significantly explored radionuclides in therapeutic nuclear medicine are listed below in **Table 2**.

In the case of therapy, the type and energy of radiation influence the range of therapeutic action. **Table 3** provides a brief perspective of therapeutic RNs based on the range of action.

Thera(g)nostics

The practice of therapeutic NM is moving towards personalized therapy. The key for effective therapy includes (1) ensuring that the therapeutic radiopharmaceutical will localize in the target tissue, and (2) the ability to calculate accurate delivery of the radiation dose. This is achieved by an appropriate diagnostic evaluation, followed by the administration of a suitable therapeutic radiopharmaceutical in an amount calculated based on the diagnostic evaluation. For the best outcome, the same biomolecule is used for imaging as well as therapy, but labeled with suitable radionuclides of the same element or surrogate nuclides. This combination of diagnostic and therapeutic procedures for planning personalized treatment is known as “Thera(g)nostics.” Here a matched pair of nuclides (which could either be of the same element or a surrogate element) are used (Yordanova et al., 2017; Ballinger, 2018). See more on Thera(g)nostics in (Kang and Venkatesh, 2021). **Table 4** lists the RNs for use as “thera(g)nostic” pairs.

Production of radionuclides

Naturally occurring RNs are seldom useful in healthcare applications. The RNs used in medicine are artificially produced either in research reactors (Barradas, 2021) or charged particle accelerators (Garcia, 2021) or availed from a radionuclide generator (the parent nuclide in the radionuclide generator is produced either in a reactor or an accelerator and the daughter nuclide is the RN used in NM).

Table 2 Radionuclides used in therapeutic nuclear medicine.

Radionuclide	$T_{1/2}$	Decay mode	$E_{\beta\max}$ (MeV)	Common production route/s	Remarks
<i>Widely used</i> ^{131}I	8.03 days	β^- , γ	0.606 (major beta)	$^{130}\text{Te}(n,\gamma) \rightarrow ^{131}\text{Te} (\beta^-) ^{131}\text{I}$; $^{235}\text{U}(n,f) ^{131}\text{I}$ $^{176}\text{Lu}(n,\gamma) ^{177}\text{Lu}$; $^{176}\text{Yb}(n,\gamma) ^{177}\text{Yb} \rightarrow \beta^- ^{177}\text{Lu}$	Well established for treatment of thyroid cancers. I-131 labeled molecules used in certain specific cancers
^{177}Lu	6.64 days	β^- , γ	0.497		Growing use in therapy of different cancers
<i>Established, limited specific use</i> ^{32}P	14.3 days	β^-	1.71	$^{32}\text{S} (n,p) ^{32}\text{P}$	High energy neutrons needed Primarily used in the past for treating polycythemia vera. In some countries currently used for palliative treatment of skeletal metastases
^{89}Sr	50.5 days	β^-	1.46	$^{89}\text{Y}(n,p) ^{89}\text{Sr}$ $^{88}\text{Sr}(n,\gamma) ^{89}\text{Sr}$	High energy neutrons needed for (n,p) reaction; well established for palliative treatment of skeletal metastases
^{90}Y	2.67 days	β^-	2.27	$^{235}\text{U}(n,f) \rightarrow ^{90}\text{Sr} (\beta^-) ^{90}\text{Y}$ $^{89}\text{Y}(n,\gamma) ^{90}\text{Y}$	^{90}Sr $T_{1/2}$ 29.9 years; $^{90}\text{Sr}/^{90}\text{Y}$ Radionuclide Generator; Used for treatment of large size lesions, especially for liver cancers
^{153}Sm	1.9 days	β^- , γ	0.81	$^{152}\text{Sm}(n,\gamma) ^{153}\text{Sm}$	Primarily used as phosphonate for palliative treatment of skeletal metastases
^{169}Er	9.3 days	β^-	0.34	$^{168}\text{Er}(n,\gamma) ^{169}\text{Er}$	Extremely specific use in radiation synoviorthesis to treat small phalangeal joints
^{188}Re	17.0 h	β^- , γ	2.12	$^{186}\text{W}(n,\gamma) ^{187}\text{W}(n,\gamma) ^{188}\text{W} (\beta^-) ^{188}\text{Re}$ $^{187}\text{Re}(n,\gamma) ^{188}\text{Re}$	^{188}W $T_{1/2}$ 69 days; $^{188}\text{W}/^{188}\text{Re}$ Radionuclide Generator; ^{188}W production involves successive neutron capture and, hence, feasible in high flux reactors
$^{225}\text{Ac} \downarrow$	9.92 days	α	5.83	Radiochemical extraction of ^{225}Ac from ^{229}Th ($T_{1/2} = 7917$ years) from spent fuel/stock of nuclear materials	RN of great interest in recent times; excellent results in treatment of prostate cancers when labeled with PSMA, a peptide specific to prostate cancer; availability in huge amounts remains a challenge
^{213}Bi (see Fig. 2 below)	45.6m	β^- (97.8%) α (2.2%)	1.42(β^-) 5.875(α)	$^{226}\text{Ra}(p,2n) ^{225}\text{Ac}$ $^{227}\text{Ac} (\beta^-) ^{227}\text{Th}(\alpha) ^{223}\text{Ra}$ $^{226}\text{Ra}(n,\gamma) ^{227}\text{Ra}(\alpha) ^{223}\text{Ac} \downarrow (\beta^-)$ $^{223}\text{Th}(\beta^-) ^{223}\text{Ra}$	^{213}Bi is a daughter product in the decay series and, hence, an Ac-225 tagged radiopharmaceutical would have the therapeutic effect of Bi-213 Bi-213 as a therapeutic RN is also being pursued strongly
^{223}Ra	11.43 days	α	5.72		$^{227}\text{Ac}-^{227}\text{Th}-^{223}\text{Ra}$ generator system is used; but ^{223}Ra is expensive, as ^{227}Ac is difficult to obtain. The $T_{1/2}$ of ^{227}Ac is 22 y and of ^{227}Th is 19 days, which is also a potential therapy RN Specifically used in palliative treatment of skeletal metastases in the form of Radium chloride. ^{223}Ra decays in a chain through emission of 4 α and 2 β -particles to stable ^{207}Pb . The total energy emitted in the sequence is 28.2 MeV (see Fig. 3 below)

<i>Emerging RNs</i> ^{47}Sc	3.35 days	β^- , γ	0.6	$^{46}\text{Ca}(n,\gamma)^{47}\text{Ca} \rightarrow ^{47}\text{Sc}$ $^{47}\text{Ti}(n,p)^{47}\text{Sc}$	Thera(g)nostic pair of Sc-4
^{67}Cu	2.58 days	β^- , γ	0.57	$^{67}\text{Zn}(n,p)^{67}\text{Cu}$ $^{68}\text{Zn}(p,2p)^{67}\text{Cu}$	Thera(g)nostic pair of Cu-64; growing interest
<i>RNs with therapeutic potential-explored/limited use/identified</i>					
^{105}Rh	1.47 days	β^- , γ	0.57	$^{104}\text{Ru}(n,\gamma)^{105}\text{Ru} (\beta^-)^{105}\text{Rh}$	The beta particles from ^{105}Rh are attractive for therapy; however, radiolabeling bio-molecule with the soft metal Rh(I) is not very amenable
$^{117\text{m}}\text{Sn}$	13.6 days	IT and IC	IC e^- 130, 160, E_γ 159 keV	$^{117}\text{Sn}(n,n\gamma)^{117\text{m}}\text{Sn}$	Very low reaction cross section and yields. Primarily used for palliative treatment of skeletal metastases
^{143}Pr	13.6 days	β^-	1	$^{142}\text{Ce}(n,\gamma)^{143}\text{Ce} (\beta^-)^{143}\text{Pr}$	One of the radiolanthanides with therapeutic potential; but not explored much
^{165}Dy	2.33 h	β^- , γ	1.285	$^{164}\text{Dy}(n,\gamma)^{165}\text{Dy}$	The photon emission is very low and of low energy
^{166}Ho	26.9 h	β^- , γ	1.85	$^{165}\text{Ho}(n,\gamma)^{166}\text{Ho}$	γ rays of 81 keV (6.4%) helps in imaging and dosimetry
^{170}Tm	128.6 days	β^- , γ	0.968 (82%), 0.883 (18%)	$^{169}\text{Tm} (n,\gamma)^{170}\text{Tm}$	Although the half-life of 129 days is generally not preferred, it is useful in pain palliation
^{175}Yb	4.18 days	β^- , γ	0.47	$^{174}\text{Yb}(n,\gamma)^{175}\text{Yb}$	The $T_{1/2}$ and 390 keV γ emission in low fraction are advantageous for therapy
^{186}Re	3.72 days	β^- , γ	1.07	$^{185}\text{Re}(n,\gamma)^{186}\text{Re}$	Small percentage (9.5%) γ with energy 137 keV is very useful. But, production logistics not amenable
^{198}Au	2.69 days	β^- , γ	0.96	$^{197}\text{Au} (n,\gamma)^{198}\text{Au};$	Useful for therapy; but E_γ 412 keV is high
^{211}At	7.21 h	EC(58.2%) α (41.8%)	0.687 (EC- γ) 5.87 (α)	$^{209}\text{Bi}(\alpha,2n)^{211}\text{At}$	At-211, a radiohalogen, has been explored for labeling in a similar way as I-131
^{213}Bi	45.6 months	β^- (97.8%) α (2.2%)	1.42(β^-) 5.875(α)	$^{225}\text{Ac}(3\alpha)^{213}\text{Bi}$ (see Fig. 2)	^{213}Bi is short lived α emitter availed from Ac-225. Therapy using both the RNs is an approach followed
^{227}Th	18.7 days	α	6.038	$^{227}\text{Ac} (\beta^-)^{227}\text{Th}(\alpha)^{223}\text{Ra}$ (see Fig. 3)	^{227}Th is an intermediate product in the decay chain of Ac-227 and has the potential to be used for therapy

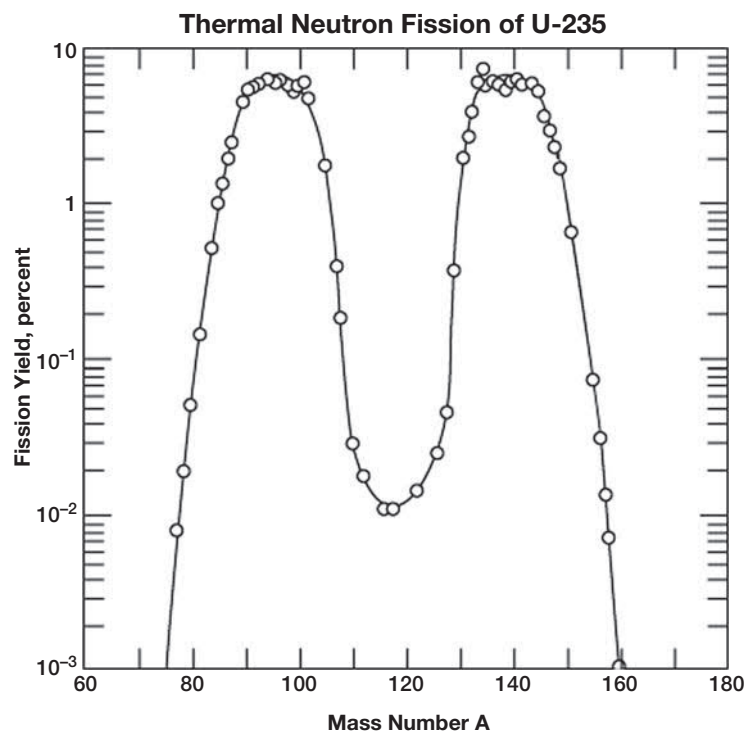


Fig. 4 U-235 Fission Yield Curve.

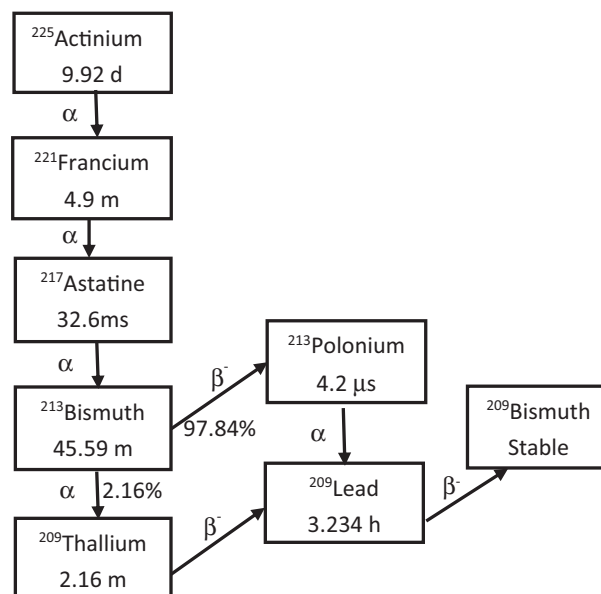


Fig. 2 Decay scheme of Ac-225.

Radionuclide production in nuclear reactors

As mentioned earlier, the use of radionuclides in medicine has a long history. With the establishment of research reactors, several radioisotopes were produced and supplied to the hospitals. The most used were Phosphorus-32, Iodine-131, Gold-198, Chromium-51, Sodium-24 and Mercury-198; and among these, I-131 continues to be an important therapeutic radioisotope, and Phosphorus-32 is still used in a few countries for therapy. The rest are currently seldom used in radiopharmaceuticals, although in recent years there has been a revival of interest in the use of the emitting **what?** for in vivo therapy.

In nuclear reactors, reactions induced by neutrons (such as radiative neutron capture (n, γ); neutron capture followed by beta decay ($n, \gamma \rightarrow \beta^-$); neutron capture followed by a proton emission (n, p); neutron capture followed by an alpha particle

Table 3 Important therapeutic radionuclides in terms of nature of emission and tissue range.

	Radionuclides
ALPHA EMITTERS ($E_\alpha = 5\text{--}8$ MeV; 40–80 μm range)	^{211}At , ^{212}Bi , ^{213}Bi , ^{256}fm , ^{225}Ac , ^{223}Ra
BETA EMITTERS (range: 1–12 mm)	
Hard betas (E_β max over 1.5 MeV)	^{32}P , ^{90}Y , ^{188}Re , ^{166}Ho , ^{89}Sr , ^{142}Pr
Medium betas (E_β max 0.5–1.5 MeV)	^{131}I , ^{153}Sm , ^{186}Re , ^{89}Sr , ^{170}Tm , ^{67}Cu , ^{143}Pr
Soft betas (E_β max less than 0.5 MeV)	^{33}P , ^{169}Er
Auger Electrons/Conversion Electrons Emitters (range: nm to μm)	^{124}I , ^{125}I , $^{117\text{m}}\text{Sn}$

Table 4 Thera(g)nostic pairs: therapeutic radionuclides and surrogate radionuclides for diagnostic imaging for internal dosimetry.

Therapeutic RN	$T_{1/2}$	Decay mode	E_γ/keV (%) for imaging	Diagnostic pair	$T_{1/2}$	Decay mode	Surrogate RN E_γ/keV (%)
<i>Therapeutic radionuclides with gamma emission which can be imaged</i>							
^{131}I	8 days	β^-	365 (82)	^{123}I	13.3 h	EC	159 (83)
				^{124}I	4.15d	EC and β^+	603 (62.9)
							511 (23)
^{47}Sc	3.35 days	β^-	159 (68)	^{44}Sc	3.97 h	β^+	511 (188)
^{67}Cu	2.58 days	β^-	186 (40)	$^{64}\text{Cu}^a$	12.7 h	EC, β^- , β^+	511 (38.6)
$^{117\text{m}}\text{Sn}$	14.0 days	EC	159 (86)				
^{153}Sm	1.94 days	β^-	103 (30)	$^{99\text{m}}\text{Tc}$	6 h	IT	140.5 (89)
				$^{94\text{m}}\text{Tc}$; ^{94}Tc	52 min; 4.9 h	EC, β^+	871 (99)
							511 (135)
^{186}Re	3.7 days	β^-	137 (9)	$^{99\text{m}}\text{Tc}$	6 h	IT	140.5 (89)
				$^{94\text{m}}\text{Tc}$; ^{94}Tc	52 min; 4.9 h	EC, β^+	871 (99)
							511 (135)
^{188}Re	17 h	β^-	155 (15)	$^{99\text{m}}\text{Tc}$	6 h	IT	140.5 (89)
				$^{94\text{m}}\text{Tc}$; ^{94}Tc	52 min; 4.9 h	EC, β^+	871 (99)
							511 (135)
^{211}At	7.2 h	α	79 (21)				
^{213}Bi	46 min	β^-	441 (926)				
<i>Therapeutic radionuclides that do not have a suitable imageable photon emission</i>							
^{90}Y	2.7 days	β^-	–	^{111}In	2.8d	EC	171 (90)
							245 (94)
				^{86}Y	14.7 h	β^+	1077 (82.5)
							511 (66)
^{125}I	60 days	EC	–	^{123}I	13.3 h	EC	159 (83)
				^{124}I	4.15d	EC and β^+	603 (62.9)
							511 (23)
^{89}Sr	50.5 days	β^-		^{85}Sr	64.9d	EC	514 (96)
				$^{87\text{m}}\text{Sr}$	2.8 h	IT	388 (82)

^a ^{64}Cu decays by three different modes, namely β^- , β^+ , as well as Electron Capture and emits beta rays, positrons as well as gamma rays, which can be used for both diagnostic and therapeutic purpose.

emission (n, α); and nuclear fission (n, f)) are used for the production of RNs. Often, the products are neutron rich isotopes, which would transmute through beta decay. In nuclear reactors, the most commonly used routes for production of RNs are (n, γ) or (n, γ) $\rightarrow \beta \rightarrow$ reactions.

But, the fission of U-235, which results in a wide range of radioisotopes with varied yields, peaking around mass numbers 100 and 130, as shown in the adjoining Fig. 4, is an attractive route for producing RNs in the peak regions as the yields are excellent. Fission has been the principal route for production of Mo-99 (parent of Tc-99m) and a few other radioisotopes such as I-131, Xe-133.

Radionuclide production in accelerators

Protons and deuterons of energies between 10–30 MeV are the most employed charged particles for the production of clinically used RNs in dedicated medical cyclotrons. Often, small cyclotrons, also known as baby cyclotrons, are installed in hospital set-up to avail a smooth supply of such RNs. (The IAEA Medical cyclotrons directory gives the list of cyclotron facilities in the world). The proton/deuteron induced reactions with emission of neutrons ((p, n), (d, n), (p, xn) etc.) are the most common routes used, leading to neutron deficient RNs, which predominantly decay through positron emission or electron capture routes.

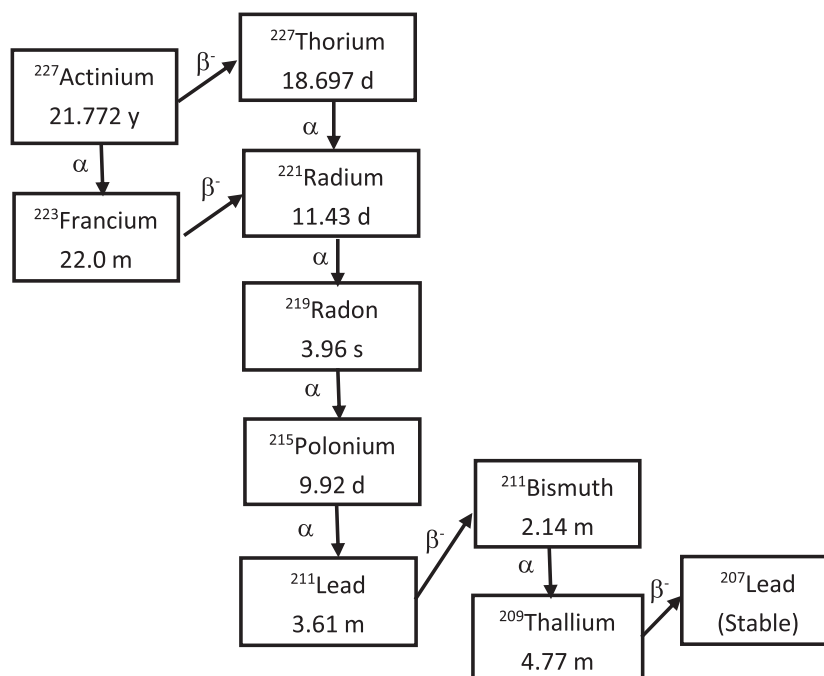


Fig. 3 Decay scheme of Ac-227.

Table 1 lists several examples of such reactions. Alpha particles have also been used for the production of a few RNs. For example, production of Ga-68 and Sm-153 have been reported through (α, n) reaction ($^{65}\text{Cu}(\alpha, n)^{68}\text{Ga}$; $^{150}\text{Nd}(\alpha, n)^{153}\text{Sm}$) (Qaim, 2017). However, heavy ion induced reactions often require higher energy cyclotrons and may not be economically attractive for regular production, especially if there is an alternate method. With the quest for additional suitable RNs for use in NM, higher energy particle reactions in larger cyclotrons (including spallation reactions), linear accelerators, and electron accelerators are being increasingly explored and employed for their production.

Although several RNs could be used in NM, as seen from the tables above, not all of them are used in clinical settings on regular basis, despite good features. Several factors that influence the feasibility of production of a RN in NM in clinical practice were mentioned earlier, underlining the need for producing the RN in high purity. In this context, the use of enriched targets often helps to avoid unwanted radionuclide impurities. But, enriched targets are generally expensive and, in some cases, where the natural abundance is low, enriched targets can be extremely expensive—making the cost of the procedure very high and, hence, unfavorable for routine clinical use. In the case of therapy using RNs, the therapeutic range depends on the type of radiation and its energy, and often the RN is chosen depending on the desired volume to be irradiated.

Radionuclide generators in nuclear medicine

“Radionuclide generators” are very convenient devices to produce a RN at the user end (hospital or a radiopharmacy). Such RN generators are possible when a parent RN with a longer half-life decays to the daughter RN with shorter half-life. The daughter nuclide grows and reaches a maximum—beyond which its growth and decay match and nullify each other and, hence, reaches an equilibrium state after a time that is dictated by the half-lives of the pair. The daughter nuclide can be repeatedly harvested after suitable time gaps and used for preparation of the radiopharmaceuticals, which makes the RN generators an attractive option to produce the RN at will. Availability as a generator has been one of the important factors that steered the growth of the generator availed RNs in NM. The $^{99}\text{Mo}/^{99\text{m}}\text{Tc}$ generator is an excellent example.

Radionuclide generators are very convenient to avail short lived medical radionuclides at will. Hence, they provide an attractive option for the hospitals/radiopharmacies since it is not dependent on transport from a reactor or cyclotron—even though the parent radionuclides have to be produced in reactors or cyclotrons (Lambrecht, 1983; Rösch and Knapp, 2011). As mentioned earlier, for a useful generator, the parent nuclide has to be longer-lived than the shorter-lived daughter nuclide, such that the daughter nuclide can be periodically separated and used in preparation of radiopharmaceuticals. The daughter nuclide, which is of interest, is produced due to the decay of the parent nuclide. Its potency is then depleted due to its own decay.

Consider the decay constant of the parent nuclide to be λ_1 and that of the daughter to be λ_2 . If there are N_1 atoms of the parent nuclide, the rate at which it decays to the daughter nuclide can be expressed as $N_1\lambda_1$ (and this is the same as the rate of production of the daughter nuclide). The rate at which the daughter nuclide decays is $N_2\lambda_2$. The overall growth rate of the daughter nuclide can be expressed as

$$\frac{dN_2}{dt} = N_1\lambda_1 - N_2\lambda_2$$

If the half-life of the daughter nuclide is shorter than the parent nuclide, the daughter nuclide grows initially from the point when there is just the parent nuclide ($N_1 = N_1^0$ and $N_2 = N_2^0 = 0$) and with time the amount of N_2 increases, until the rate of growth and depletion of the daughter nuclide are nearly equal. This state is known as "equilibrium." The time taken to reach this equilibrium would be a function of the decay constants of the parent-daughter pair.

We can write mathematical equations to represent the amounts of the parent and daughter at any point of time "t," wherein at time $t = 0$, the beginning, $N_1 = N_1^0$ and $N_2 = N_2^0$ (which is equal to 0 when one starts with pure parent nuclide. However, this equation is valid only when N_2^0 amount of daughter nuclide is present in the system at time $t = 0$).

$$\frac{dN_1}{dt} = -N_1\lambda_1 \text{ and } \frac{dN_2}{dt} = N_1\lambda_1 - N_2\lambda_2$$

On integration of these equations, one gets $N_1 = N_1^0 e^{-\lambda_1 t}$ and

$$N_2 = \lambda_1 / (\lambda_2 - \lambda_1) N_1^0 (e^{-\lambda_1 t} - e^{-\lambda_2 t}) + N_2^0 e^{-\lambda_2 t}$$

When $N_2^0 = 0$, $N_2 = \lambda_1 / (\lambda_2 - \lambda_1) N_1^0 (e^{-\lambda_1 t} - e^{-\lambda_2 t})$

If λ_2 is considerably greater than λ_1 , and when the time elapsed is long (several folds longer than the half-life of the daughter nuclide), then the term $e^{-\lambda_2 t}$ is negligible in comparison to $e^{-\lambda_1 t}$, and the equation can be approximated to

$$N_2 = \frac{\lambda_1}{\lambda_2 - \lambda_1} N_1^0 e^{-\lambda_1 t}$$

Since $N_1^0 e^{-\lambda_1 t} = N_1$, the equation can be written as

$$N_2 = \frac{N_1\lambda_1}{(\lambda_2 - \lambda_1)}; \quad \text{which is} \quad \frac{N_1}{N_2} = \frac{\lambda_2 - \lambda_1}{\lambda_1}$$

Thus, it can be seen that when a considerable amount of time elapses, the ratio of parent and daughter nuclides reaches a constant value (equilibrium). When the half-life of parent is far greater (there is no specific number, but can be considered to be orders of magnitudes greater) than that of daughter, the equilibrium is known as "Secular" equilibrium. When it is moderately greater, the equilibrium is known as "Transient."

If the above equation is expressed in terms of "activity" (disintegrations per second or Bq), it will be $\frac{N_2\lambda_2}{\lambda_2} = \frac{N_1\lambda_1}{(\lambda_2 - \lambda_1)}$; or $\frac{A_2}{\lambda_2} = \frac{A_1}{(\lambda_2 - \lambda_1)}$; $A_2 = \frac{A_1\lambda_2}{(\lambda_2 - \lambda_1)}$

This indicates that when sufficient time has elapsed and equilibrium is established, the parent and daughter activities are in a constant ratio. If λ_2 is considerably greater than λ_1 , the above equation can be further approximated to $A_2 = A_1$. The time required to attain equilibrium and reach this maximum level, also known as $t_{\max}$ is given by the equation

$$t_{\max} = \frac{1}{\lambda_2 - \lambda_1} \ln \frac{\lambda_2}{\lambda_1}$$

As a general approximation, it is considered that the time to reach equilibrium is about four times the $T_{1/2}$ of the daughter RN in the case of transient equilibrium, and about seven times the $T_{1/2}$ of the daughter RN in the case of secular equilibrium.

The graph of N_1 and N_2 with time would look like the following Figs. 5 and 6 for both cases.

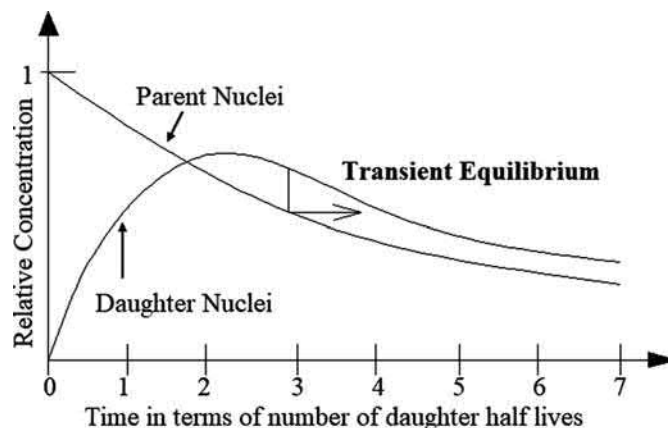


Fig. 5 Transient equilibrium.

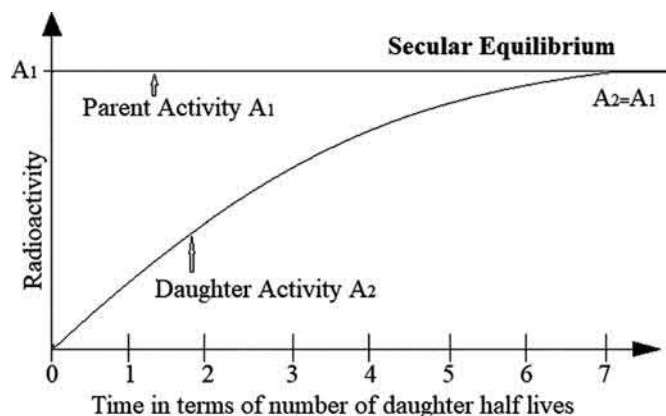


Fig. 6 Secular equilibrium.

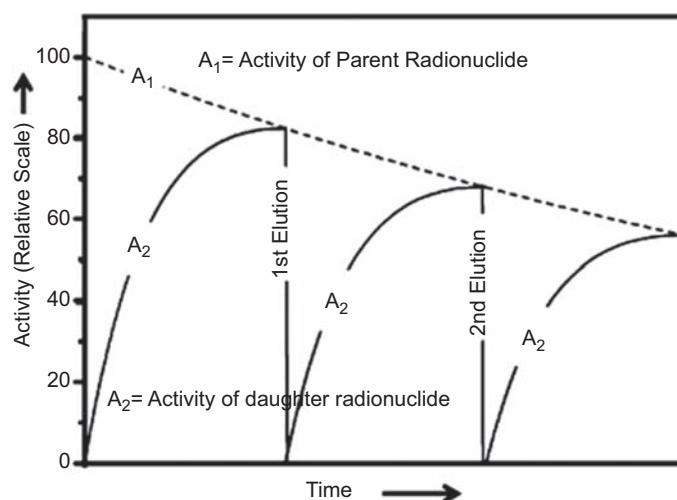


Fig. 7 Growth of daughter RN with time on repeated elution.

Once the daughter nuclide grows to the maximum (no effective growth when rate of growth and decay are equal), it is “harvested” by separating it from the parent, setting the time back to “zero” ($N_2 = 0$) and it starts growing again. A graph of the activity growth is represented below in **Fig. 7**.

The daughter nuclide is thus availed repeatedly, and this phenomenon has been compared with daily milking of cows. Separation of the daughter nuclide is at times referred to as “milking” and the generator as “cow.” In the case of Tc-99m, with a half-life of ~ 6 h, the t_{max} is ~ 23 h and the Mo-99/Tc-99m generator can be indeed “milked” every morning at the radiopharmacy. The useful shelf life of the generator will depend on the half-life of the parent; and the time required for the daughter RN to grow to the equilibrium state depends on the half-life of the daughter, which guides the frequency at which it can be separated and used. Isolation of the daughter nuclide from the parent is most often achieved through chromatographic elution from a column using suitable eluants, although a few other separation mechanisms such as solvent extraction, electrochemical separation, sublimation, conversion of the parent RN into a solid matrix are amenable for use as the column matrix or membrane separation have been used and explored ([Dash and Chakravarty, 2014](#)).

While several separation techniques may be used to efficiently separate the daughter nuclide from the parent, the practical usage will depend upon the ease of production as well as the ease of operation, with minimal radiation exposure to the personnel. Hence, generators that can be manufactured in large numbers, are portable, safe, cost effective, and which can be handled with ease by the users are the ones used widely. Column chromatography-based generators, where the daughter RN can be eluted by connecting an evacuated vial at the end of the column, are the most used.

It is important to separate the daughter nuclide of interest from the parent nuclide with high purity, both chemically as well as radio nuclidic purity. It is possible to avoid radionuclide impurities when the parent and the daughter nuclides are of different elements, which is the case in all the generators used in nuclear medicine. The permissible limit for the parent RN in the daughter RN (often referred to as parent RN breakthrough in the daughter eluate) depends on the undesired effects of the parent RN. The unwanted radiation dose from the parent RN, which in turn would be dependent on the target organs

where it would accumulate and its half-life, are the prime factors that govern the tolerance limits set for parent RN in the daughter RN product. For example, the limit for ^{99}Mo in $^{99\text{m}}\text{Tc}$ eluate is 0.15%, whereas the limit for ^{90}Sr in ^{90}Y is extremely low, less than $10^{-4}\%$, since ^{90}Sr has a half-life of 28.5 years and Sr^{+2} accumulates in bone and stays there for decades, giving a potentially serious radiation dose to bone marrows. The limits for ^{90}Sr in ^{90}Y are set based on the bone marrow radiation dose from ^{90}Sr (Pandey et al., 2008).

The quality control tests used are designed to measure such extremely low levels of the undesired RNs. Further, quality control tests are also performed to ensure that the levels of the possible chemical impurities are lower than the permissible levels. For example, if a column chromatography generator is used, the amount of adsorbent that gets eluted into the daughter product is limited; and if a solvent extraction separation is used, the amount of the solvent used for extraction is limited—the limits being set up based on the permissible levels of these chemicals in an injectable.

A list of RN generators used/potential for use in nuclear medicine is provided in Table 5 (not an exhaustive list):

Some of these generators are in regular use and commercially supplied, while several others have been and are being explored for use in NM, but are yet to be used in a sustainable manner in NM. Although the daughter RN may have the potential for use, the practical aspect, such as the feasibility of producing the parent RN in high purity and in adequate quantities, is an important factor for sustainable production and use of RN generators. Further, RNs with very short half-life pose a great challenge in their use. The concept of “in vivo” generator, wherein the parent RN is used for preparation of the radiopharmaceutical and administered in the patient, has been explored in a few cases where the daughter RN has a very short half-life (and, hence, would decay before the molecule reaches the target) (Mausner et al., 1989; Edem et al., 2016). However, this approach requires the molecules labeled with parent and daughter nuclides to have similar in vivo behavior, which is often not the case due to different chemical features of the parent and daughter RNs.

The commonly used generators are briefly described below.

Table 5 Radionuclide generators relevant to nuclear medicine.

Generator	Parent $T_{1/2}$	Decay mode	Daughter $T_{1/2}$	Decay mode	E_{γ} (keV)
<i>Generators for gamma emitting RNs for SPECT/gamma imaging</i>					
$^{99}\text{Mo}/^{99\text{m}}\text{Tc}$	65.9 h	β^{-}	6.01 h	IT	140.5
$^{81}\text{Rb}/^{81\text{m}}\text{Kr}$	4.57 h	EC	13.1 s	IT	190.6
$^{77}\text{Br}/^{77\text{m}}\text{Se}$	57 h	EC	17.4 s	IT	161.9
$^{109}\text{Cd}/^{109\text{m}}\text{Ag}$	1.265 years	EC	39.8 s	IT	88
$^{113}\text{Sn}/^{113\text{m}}\text{In}$	115.1 days	EC	1.66 h	IT	391.7
$^{178}\text{W}/^{178\text{m}}\text{Ta}$	21.6 days	EC	2.36 h	EC(99%), β^{+}	97.3
$^{191}\text{Os}/^{191\text{m}}\text{Ir}$	15.4 days	β^{-}	4.9 s	IT	171.3
<i>Generators for positron emitting RNs for PET imaging</i>					
$^{68}\text{Ge}/^{68}\text{Ga}$	270.9 days	EC	67.7 min	β^{+} (89%), EC	511
$^{82}\text{Sr}/^{82}\text{Rb}$	25 days	EC	1.25 min	β^{+} (96%), EC	511, 776
$^{44}\text{Ti}/^{44}\text{Sc}$	47/60.0 years	EC	3.93 h	β^{+} (95%), EC	511, 1157
$^{52}\text{Fe}/^{52\text{m}}\text{Mn}$	8.3 h	β^{+} (56.5%), EC(43.5%)	21 min	β^{+}	511, 1434
$^{162}\text{Zn}/^{62}\text{Cu}$	9.2 h	β^{+} (6.9%), EC(93%)	9.73 min	β^{+} (97.8%), EC	511
$^{72}\text{Se}/^{72}\text{As}$	8.4 days	EC	26 h	β^{+} (77%), EC	511, 834
$^{110}\text{Sn}/^{110\text{m}}\text{In}$	4.11 h	EC	69 min	β^{+} (62%), EC	511, 658
$^{118}\text{Te}/^{118}\text{Sb}$	25 days	EC	3.5 min	β^{+} (76%), EC	511
$^{122}\text{Xe}/^{122}\text{I}$	20.1 h	EC	3.6 min	β^{+} (77%), EC	511
$^{128}\text{Ba}/^{128}\text{Cs}$	2.43 days	EC	3.64 min	β^{+} (53%), EC	511
$^{134}\text{Ce}/^{134}\text{La}$	3.16 days	EC	6.45 min	β^{+} (62%), EC	511
$^{140}\text{Nd}/^{140}\text{Pr}$	3.37 days	EC	3.39 min	β^{+} (51%), EC	511
<i>Generators for therapeutic RNs</i>					
$^{90}\text{Sr}/^{90}\text{Y}$	28.5 years	β^{-}	2.67 days	β^{-}	E_{max} (MeV) 2.218
$^{188}\text{W}/^{188}\text{Re}$	69.4 days	β^{-}	16.9 h	β^{-}	2.118
$^{66}\text{Ni}/^{66}\text{Cu}$	54.6 h	β^{-}	5.1 month	β^{-}	2.6
$^{112}\text{Pd}/^{112}\text{Ag}$	21.0 h	β^{-}	3.1 h	β^{-}	3.94
$^{115}\text{Cd}/^{115\text{m}}\text{In}$	2.23 days	β^{-}	4.49 h	β^{-}	0.83
$^{140}\text{Ba}/^{140}\text{La}$	12.75 days	β^{-}	1.68 days	β^{-}	1.35
$^{194}\text{Os}/^{194}\text{Ir}$	6.0 years	β^{-}	19.28 h	β^{-}	2.23
$^{224}\text{Ra}/^{212}\text{Bi}^{\text{a}}$	3.63 days	α	60.55 min	α , β^{-}	α (6.0); β^{-} (2.25)
$^{225}\text{Ac}/^{214}\text{Bi}^{\text{a}}$	9.92 days	α	19.9 min	α , β^{-}	α (5.83); β^{-} (3.27)
$^{227}\text{Ac}/^{223}\text{Rn}^{\text{a}}$	21.77 years	α	11.43 d	α	5.72

^aIn these generators, the first parent indicated here undergoes a series of alpha decays to yield the daughter RN of interest, indicated here. This can be seen from the decay schemes of Ac-225 and Ac-227 given earlier.

$^{99}\text{Mo}/^{99\text{m}}\text{Tc}$ Generator

$^{99\text{m}}\text{Tc}$ is an ideal radionuclide, used in a wide variety of radiopharmaceuticals (details given under Tc-99m later in this article) and the $^{99}\text{Mo}/^{99\text{m}}\text{Tc}$ generator is the most used generator in nuclear medicine. The ^{99}Mo (half-life 66 h/2.75 days)— $^{99\text{m}}\text{Tc}$ (half-life 6 h) pair attains transient equilibrium and the amount of daughter $^{99\text{m}}\text{Tc}$ reaches a maximum at nearly 23.5 h, which is practically very convenient. The terms “cow” to indicate the generator and “milking” to refer to the separation of the daughter nuclide, fit well in this case, as generally $^{99\text{m}}\text{Tc}$ is separated daily for preparation of radiopharmaceuticals. Different techniques, such as column chromatography, solvent extraction, sublimation and electrochemical separation have been used to separate $^{99\text{m}}\text{Tc}$ from ^{99}Mo (Boyd, 1982; Le, 2014; IAEA TECDOC-852, 1995). Among these, column chromatography generators with alumina as the sorbent and saline as the eluant are the most widely used. The specific activity of the Mo-99 is a key factor that has an impact on the separation efficiency.

In brief, the adsorbent used in the column binds to Mo-99 strongly while the daughter Tc-99m is not bound, which makes it possible to elute the daughter from the column using a suitable solvent, such as saline. As the capacity to bind to the parent Mo-99 is limited, an adequate amount of the sorbent needs to be used in the column to hold the amount of molybdenum. Mo-99 obtained from fission of uranium is of high specific activity and has extremely low amounts of molybdenum, enabling use of a small amount of sorbent (few grams of alumina) per generator. Low specific activity Mo-99 from the neutron activation route would have far higher amounts of molybdenum and would need much larger amounts of adsorbent making the column long and the eluted Tc-99m dilute (low specific volume). For radiolabeling molecules, Tc-99m must be adequately concentrated. Hence, column chromatography generators with alumina as the adsorbent are not suited to avail Tc-99m from low specific activity Mo-99 without post elution concentration of the eluted Tc-99m. Other adsorbents based on metallic oxides and nanocrystalline materials have been explored with limited success; but none have been adopted to commercial production and supply. Several other separation technology options have also been explored in this context.

In the other technologies, such as sublimation, solvent extraction, and electrochemical separation, Tc-99m is obtained in a concentrated form. Converting the low specific activity Mo-99 into a chemical form that is amenable to be used as the column matrix has been explored, and zirconium molybdate has been successfully used to prepare column generators, also termed as “gel generators,” but deployed in a few limited institutes (Evans et al., 1987; Saraswathy et al., 1998). Mo-99-Tc-99m generators continue to be of interest to researchers, especially as production of Mo-99 through non-fission is being explored.

 $^{81}\text{Rb}/^{81\text{m}}\text{Kr}$ Generator

Kr-81m, a noble gas with a half-life of just 13 s, is used predominantly in lung imaging (Koyama et al., 1980). As the half-life of the parent Rb-81 is also short, 4.7 h, the useful life of the generator is just about a day (Ruth et al., 1980), making it a challenge to produce and transport the generator to hospitals. Hence, the manufacture and the use of this generator is limited. The permissible level of Rb-81 in the Kr-81m product is less than 0.1%.

 $^{68}\text{Ge}/^{68}\text{Ga}$ Generator

Ga-68 has gained a lot of attention in the past decade (more on Ga-68 under radionuclides below) (Rösch, 2013). A key factor for its increased use is the availability through the generator, especially because the long half-life of 271 days of the parent Ge-68 translates to long useful shelf-life of the generator and assured supply of the short lived (67.7 min half-life) Ga-68 over several months. The currently available Ge-68/Ga-68 generators are of the column chromatography type and different adsorbent matrices such as titanium dioxide or tin dioxide, as the adsorbent matrix, and hydrochloric acid as the eluant (Romero et al., 2020). The daughter Ga-68 growth reaches the maximum level in 7 h and, hence, can be eluted a few times every day. The gallium, eluted as Ga(III) chloride. The permissible amount of Ge-68 in the Ga-68 product is not strictly specified, but a value of less than 0.01% breakthrough of Ge-68 is considered acceptable. Most commercial generators report far lesser breakthrough values.

 $^{82}\text{Sr}/^{82}\text{Rb}$ Generator

Rb-82, a positron emitter, is used in myocardial perfusion imaging studies (Alvarez-Diez et al., 1999; Yoshinaga et al., 2010). Rb-82 has a half-life of just 1.25 min., making its administration into the patient a challenge. But, as the parent RN Sr-82 has a half-life of 25 days, the generator has a reasonably long useful life of about 6–8 weeks. The current Sr-82/Rb-82 generators are chromatographic generators with titanium oxide as the column material and eluted with normal saline. The permissible limits for the strontium radionuclides Sr-82 and Sr-85 in the Rb-82 eluate are $2 \times 10^{-4}\%$ and $2 \times 10^{-3}\%$ respectively. The use of Sr-82/Rb-82 generator, which was limited owing to the short half-life of Rb-82, witnessed a renewed interest in catering to the myocardial imaging that faced challenges due to the shortages of supply of Mo-99/Tc-99m generators, during 2007–08 period. Although the supply of Mo-99/Tc-99m generators has resumed to cater to demands, several users continue to use the Sr-82/Rb-82 generators.

 $^{90}\text{Sr}/^{90}\text{Y}$ Generator

Y-90, with a half-life of 64 h, emits high energy beta particles and is ideal for RN therapy of large lesions. The parent Sr-90 has a long half-life of 28.5 years, and, hence, the Sr-90/Y-90 generators would theoretically have an exceptionally long shelf life. However, in

practice, the usability of a generator will depend on the type of generator and the storage of the parent RN. If the generator is of column chromatography type, in which the parent RN is adsorbed in a solid matrix, the integrity and performance of the matrix could be affected by radiation and, hence, this aspect will have to be considered in the use of the generator. Nevertheless, the long half-life of Sr-90 makes it a possible to separate the daughter Y-90 repeatedly for several decades, longer than the half-life of Sr-90. Different types of generators have been reported (IAEA TECDOC TRS 470, 2009), based on separation systems such as ion-exchange column chromatography (Chinol et al., 1997), electrochemical separation (Chakravarty et al., 2008), and membrane-based separation (Dhami et al., 2007). Currently, the Sr-90/Y-90 generators are prepared and operated at the nuclear institutes or radiopharmaceutical manufacturing centers and not commercially available, although Y-90 is commercially available for radiolabeling at the user end. As Sr-90 is also a beta emitter and an analog of calcium, it would accumulate in the bone and stay there for long if administered along with Y-90. Hence, based on the maximum permissible body burden of 74 kBq (2 μ Ci) for Sr-90, the permissible limit for Sr-90 in Y-90 is extremely low (less than $10^{-4}\%$), as the total amount of Y-90 administered is generally large over several cycles of treatment.

¹⁸⁸W/¹⁸⁸Re Generator

Re-188 emits high energy beta particles and is well suited for therapy of large lesions. The 16.9 h half-life of Re-188 and the 69 days half-life of parent W-188, makes this generator system a very convenient one with a long shelf life, with the possibility of availing Re-188 frequently. The major challenge is the production of W-188 by double neutron capture on W-186, which yields a low specific activity of W-188. Column chromatography generators with alumina or zirconium tungstate gel have been reported (Knapp et al., 1997; Boschi et al., 2014), but currently there is only a sole commercial supplier.

Widely used radionuclides

Technetium-99m

Technetium-99m is the most widely used radionuclide in nuclear medicine, in about 80% of all the diagnostic procedures, amounting to approximately 40 Million studies annually conducted throughout the world, earning it the name “Work-horse” of nuclear medicine. This is because of the excellent features of Tc-99m—such as, (a) $T_{1/2}$ of 6.02 h, decay by internal transition with emission of a single gamma ray of 140 keV energy that can be imaged with great sensitivity using the NaI(Tl) detectors, (b) availability from the Mo-99/Tc-99m generator system for repeated elution at very convenient time intervals (daughter RN Tc-99m grows to maximum level in about 24 h—most convenient for everyday elution and preparation of radiopharmaceuticals), (c) high production feasibility of Mo-99 in reactors through the fission route in very high yields, and (d) amenable chemistry of Tc-99m, making it feasible to label a large variety of molecules through complexation.

Technetium-99m, was discovered in 1939 (by Seaborg and Segre) soon after the discovery of the element technetium (the first man-made element, discovered in 1937, for which there are no naturally occurring stable or radioactive isotopes). The use and steep growth of Tc-99m started around the late 1960s when the Mo-99-Tc-99m generator was introduced in the market (Brookhaven National Lab developed the first Mo-99/Tc-99m generator in 1958—shown in Fig. 8, which was proven to be useful and commercialized in 1966) (Eckelman, 2009).

The period after introduction of a commercial Mo-99/Tc-99m generator witnessed an exponential growth in the development of new radiopharmaceuticals labeled with Tc-99m and their use in clinics. The Mo-99/Tc-99m generator yields ^{99m}Tc in the form of ^{99m}TcO₄[−] (pertechnetate) ion, which was the first product tested for its usefulness as a radiopharmaceutical. The monocharged anion behaved like an iodide ion and accumulated in the thyroid, enabling thyroid imaging. The ^{99m}TcO₄[−] pertechnetate ion was also shown to be useful in imaging the brain when the blood-brain-barrier is breached due to brain tumors. The versatile chemistry of Tc (with several oxidation states and amenability to complex with organic molecules) led to the development of a wide range of ^{99m}Tc based radiopharmaceuticals. The availability of gamma cameras, and later the SPECT cameras, gave a boost to the Tc-99m use.

The availability of the Mo-99/Tc-99m generator in adequate quantities throughout the year all around the world was a crucial factor in making Tc-99m the most used RN. In addition, with the availability of Tc-99m generators in the market during the 1970s, a wide range of Tc-99m tagged radiopharmaceuticals were developed for anatomical as well as functional imaging of the important organs. The development of numerous Tc-99m based radiopharmaceuticals to image all important diagnostic needs and making it possible for the hospital radio pharmacies to prepare these varied radiopharmaceuticals in the required quality (radiopharmaceutical purity, specific activity, biological purity—sterility and absence of pyrogens) by developing “cold-kits” (containing all the ingredients that can be used as per the instructions provided along with the Tc-99m eluted from the generator) have also been equally important for the versatile use of Tc-99m. In all, the entire supply chain involved in the use of Tc-99m (nuclear reactors, Mo-99 processors, Mo-99-Tc-99m generator producers, Cold kit producers, Hospital radiopharmacy) are important for uninterrupted use of Tc-99m based radiopharmaceuticals. (Note: The fragility of the medical radioisotope supply came to the fore during 2007–09, when the key medical radioisotope producing reactor NRU in Canada, which was supplying bulk of the global Mo-99, had a prolonged shutdown. The situation turned into a crisis when another major reactor in Netherlands also had an unplanned shut down. The global supply of Tc-99m generators was severely hit and triggered a global alarm for developing a more robust medical isotope supply strategy, which brought all the stake holders together to address the issue. The efforts through OECD is

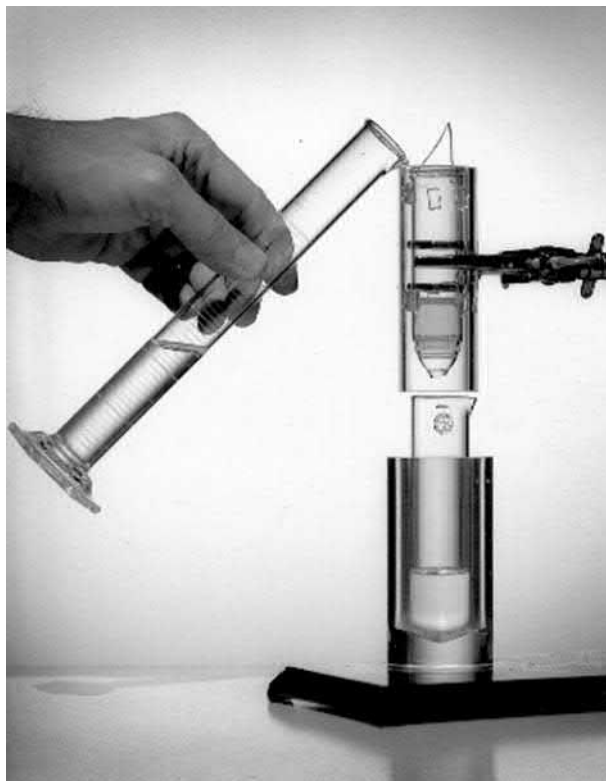


Fig. 8 The first Tc-99m Generator (shown here without the shield) developed at the Brookhaven National Laboratory.

noteworthy in this context (OECD/NEA, 2019). Fortunately, the coordination among the key players in the supply chain of Mo-99 and Mo-99/Tc-99m generators and the entry of new/refurbished reactors and augmented capacities in existing reactors have now ensured the secure supply of the most important medical isotope, Tc-99m.)

^{99m}Tc based radiopharmaceuticals are currently available for imaging nearly all the important organs and functions of the vital organs (Table 1).

Fluorine-18

Fluorine-18, with a half-life of 110 min, is the most used PET nuclide. The high percentage (97%) of emission of positrons of low energy (0.638 MeV) enable high resolution images. The use of F-18 grew very quickly towards end of the last century, when the remarkable utility of the molecule F-18 labeled, 2-Deoxy-2-Fluoro-D-Glucose (F-18-FDG) in several ailments was established and PET cameras were available for use in clinics. While F-18-FDG is undoubtedly the most used PET tracer, several other F-18 labeled molecules are regularly used in diagnostic PET imaging (confer Table 2).

Radioiodines

I-127 is the only stable isotope of the element iodine. Of the numerous (37) radionuclides of iodine, I-131, I-124 and I-123 serve a valuable role in in-vivo nuclear medicine. These three are considered together, owing to some common strategies in their use. I-125, used in in vitro assays is not considered here. Iodine chemistry is very well understood and, hence, labeling a variety of biomolecules with radioiodine is well established.

I-131: Historically, I-131 has the longest period of widespread use and, hence, has a unique place in the history of nuclear medicine.

The first RN of iodine to be produced was ¹²⁸I (by Enrico Fermi) in 1934, and as the role of iodide in thyroid metabolism was established, it was used to study thyroid metabolism and disease starting in 1937. However, the discovery of I-131 in 1939 (by Glenn Seaborg) and its availability in the years that followed, made I-131 (with 8.06 days half-life, emission of several gamma rays and decay through beta emission,) a RN of choice for thyroid studies and for treatment of thyrotoxicosis (hyperthyroidism) and thyroid cancer. Radioiodine I-131 is, thus, the first radionuclide used in nuclear medicine—both for diagnosis and for therapy and is the first theranostic agent (Silberstein, 2012). The biochemical path of iodide ions causing it to concentrate in thyroid enables the use of I-131, in the form of inorganic iodide, to image thyroid as well as treat thyroid disorders. I-131

radioiodine has been advantageously and continuously used since 1940s. It was the first FDA approved radiopharmaceutical (1951) (SNMMI Historical Timeline).

I-131 is generally produced in reactors by the irradiation of a tellurium-130 target, which decays by beta emission to I-131. As the parent and daughter are different elements, separation of I-131 from a tellurium target yields high specific activity I-131, and most manufacturers use dry distillation method to separate radioiodine from the target (Shikata and Amano, 1973; IAEA-TECDOC-1340, 2003). The use of natural tellurium oxide target results in small amounts of other radioiodine impurities, most of which decay away due to short half-lives, barring I-129, which has a long half-life of 1.57×10^7 years. I-131 is also produced through the (n, fission) route using U-235 target material, similar to Tc-99m, since the fission yield for the mass numbers around 130 is also high, as for mass numbers around 100 (see earlier section on fission production of isotopes). With the wide availability from nuclear reactors, I-131 was tagged to different molecules for imaging vital organs. I-131 labeled human serum albumin for imaging blood pool in heart, I-131-Hippuran for kidney imaging, and I-131-rosebengal for liver imaging were used in the decades followed, until suitable Tc-99 based agents replaced them, as Tc-99m was far superior to I-131 for diagnostic imaging. An exceptional case is I-131 labeled meta-iodobenzyl guanidine (^{131}I -mIBG), which behaves like norepinephrine and, hence, accumulates in adrenergic neurons, making this agent suitable for imaging neural crest tumors including neuroblastomas.

Although I-131 is used for diagnostic imaging, it is not an ideal RN for imaging, as I-131 decays by beta emission and emits several gamma rays, rendering the image quality somewhat inferior for dosimetry. But, the beta particle with E_{max} 606 keV is suitable for therapy, and I-131 in the form of iodide is advantageously used in the treatment of thyrotoxicosis and thyroid cancer. Apart from thyroid related diseases, I-131 is used in the treatment of a few other malignancies, among which ^{131}I -mIBG in therapeutic doses has an unequivocal role in treatment of neuroblastomas, including in children. Several monoclonal antibodies (MAb) labeled with I-131 have been used in the treatment of specific cancers, among which I-131-anti CD-20 MAb tositumomab is an FDA approved radiopharmaceutical for treatment of non-Hodgkin's lymphoma (Kaminski et al., 2005). The use of I-131 labeled lipiodol for treatment of liver cancers (Kobayashi et al., 1986; Leung et al., 1994) are the few other established therapy applications of I-131.

I-124: ^{124}I , a positron emitter with a half-life of 4.17 d, is attractive for studies that require long times, as most other PET RNs are short lived. Targeting molecules such as antibodies, which have high molecular weight, take a much longer time (few days) than the small molecules for reaching the target. In such cases, it is desirable to have a RN that has comparable half-life, and I-124 fits the requirement very well (Cascini et al., 2014). While the positron emission allows PET imaging of I-124, the copious Auger electron emission can be used for therapy in the range of a few nanometers by using appropriate molecules to target the DNA of the cancer cells. Further, although the decay scheme of I-124 is complex, PET imaging with I-124 has better resolution than I-131 and allows accurate quantification for pre-therapy dosimetry calculations, making I-124 a valuable diagnostic RN as a theragnostic pair with I-131. Thus, I-124 is the only long-life positron emitter isotope of iodine that is amenable for both imaging and therapy as well as for ^{131}I dosimetry. However, although I-124 has the above-mentioned attractive features, the high energy of emitted positron with long range and a high proportion (77%) of decay via electron capture route pose challenges in obtaining high resolution images. I-124 is produced in cyclotrons using an enriched Te-124 target through (d, 2n), or more recently and preferably a (p, n) reaction (Lambrecht et al., 1988). Several I-124 labeled molecules include ^{124}I -MIBG, ^{124}I -IAZA, ^{124}I -IAZG, and ^{124}I -DRFIB. (Mahajan and Divgi, 2016).

Despite the several advantages of I-124, its use is currently limited.

I-123: Iodine-123, which decays by electron capture with a half-life of 13.3 h, is used in delineation of thyroid functional status and assessment of thyroid nodules as the 159 keV photon emission is ideally suited for SPECT imaging. But its short half-life limits the study to metabolic processes that are quick.

Gallium-68

Gallium-68, a positron emitter with 67.7 min half-life and 89% decay through positron emission, is currently a highly used PET RN, with a huge potential to grow further (Rösch, 2013). Its availability from a generator (see the previous section on Ge-68/Ga-68 generator), makes it an attractive PET RN for clinics that do not have access to a medical cyclotron. The $^{68}\text{Zn}(p,n)$ reaction is also used for direct production of Ga-68 when a suitable cyclotron is accessible. Ga-68 was explored as a diagnostic imaging RN to be derived from a Ge-68/Ga-68 generator long before F-18 was used (Gleason, 1960; Greene and Tucker, 1961; Yano and Anger, 1964). However, since in the past Ga-68 was eluted in the form of Ga-EDTA (Ethylene Diamine Tetra Acetic acid), which is a very strong complex, hard to replace with other ligands, its utility remained very limited to using the ^{68}Ga -EDTA complex as the tracer. Moreover, the steep growth in the use of Tc-99m based radiopharmaceuticals, and later of F-18 based molecules, resulted in Ga-68 taking a back seat, until a generator that provided cationic $^{68}\text{Ga}(\text{III})$ became available, ushering in a new phase of Ga-68 radiopharmaceuticals. Being a trivalent ion similar to $^{111}\text{In}(\text{III})$, Ga-68-DOTATATE was soon developed as a somatostatin receptor binding agent for PET imaging of neuroendocrine tumors, replacing In-111-DTPA-octreotide. The trivalent chemistry of Ga (III) enables its complexation with various suitable chelating agents, such as DOTA(1,4,7,10-tetraazacyclododecane-1,4,7,10-tetraacetic acid) (Breeman et al., 2011). This is attractive since several therapeutic RNs such as Lu-177 are also trivalent and Ga-68-radiopharmaceutical can form a "Theranostic pair" with the same molecule labeled with a therapeutic RN. In recent years, $^{68}\text{Ga}(\text{III})$ -(HBED-CC)-PSMA-11 has gained a lot of attention as an agent for imaging prostate cancer lesions. Several ^{68}Ga labeled molecules are being explored as PET tracers, but are not covered here.

Radioisotopes of copper

Among the several copper radioisotopes, five of them, namely, Cu-64, Cu-67, Cu-62, Cu-61 and Cu-60 are considered suitable for use in nuclear medicine. But, of these, Cu-64 and Cu-67 have been explored well, Cu-62 to some extent, while the others have limitations (Boschi et al., 2018) and are not expanded here. Cu-62, which can be availed from a Zn-62/Cu-62 generator, has a short half-life of 9.74 min and the parent Zn-62 is also short lived with a half-life of 9.2 h. This makes the practical utility of Cu-62 very limited and is not detailed further here.

Copper-64: Cu-64 is a unique RN that decays by three different modes—positron (17.6%), electron capture (43.9%) and beta(−) decay (38.5%), accompanied by several Auger electrons. This makes it a suitable RN for PET imaging, as well as therapy—a theranostic RN, like I-131. A lot of work on Cu-64 has been reported in the past decade, especially as Cu-64 as Cu(II)chloride could target tumors (Boschi et al., 2018). Among various molecules labeled with Cu-64, Cu-64-ATSM has attracted attention for imaging tumor hypoxia. Cu-64 is generally produced in cyclotrons through the (p,n) reaction on Ni-64. Production via the (n, p) reaction on enriched Zn-64 target has also been reported.

Copper-67: Cu-67, a beta emitter with a half-life of 61.8 h, along with several imageable gamma rays, is another radionuclide that can be used as a theranostic RN, like I-131. But Cu-67 is predominantly proposed for use in therapy, with a theranostic pair Cu-64 for imaging.

Cu-67 is produced either through the (n, p) reaction on enriched Zn-67 in a reactor, or a (p,2n) reaction on enriched Zn-68 in a cyclotron. However, both routes have low cross sections and require high flux reactors or high energy cyclotrons. These pose limitations in the use of Cu-67.

Indium-111

In-111, a metallic radionuclide decaying by EC with a half-life of 2.8 days and emission of gamma rays of 171 and 247 keV, is a valuable RN for SPECT imaging. The amenable production in cyclotrons (through (p,n) or (p,2n) reaction on Cd-111 or Cd-112 target) and trivalent chemistry makes In-111 suitable for labeling biomolecules. Among the various molecules explored, In-111-Pentetreotide, for imaging neuroendocrine tumors expressing somatostatin receptors (Ivancevic et al., 1992), In-111 labeled leucocytes for infection imaging (Datz, 1994), and a few In-111 labeled MABs against tumor antigens (for imaging colorectal, ovarian and prostate cancers), were the most used. But, despite being a useful SPECT RN, the use of In-111 has reduced significantly after the wide availability of Ga-68.

Samarium-153

Sm-153, a beta emitter ($E_{\max}$ 0.81 MeV) with a small percentage of imageable gammas, half-life of 1.8 days, facile production in reactors through (n, γ) reaction on Sm-152, and convenient chemistry, render it a useful RN for therapy of medium sized lesions. Sm-153 labeled EDTMP has been used in pain palliation in skeletal metastases for many decades (Holmes, 1992). However, other products labeled with Sm-153 have yet to find wide use.

Lutetium-177

Lu-177, a beta emitter with a half-life of 6.64d, is an attractive therapeutic RN ($E\beta$ -max 0.5 MeV; imageable gammas) and can be produced amenably through the (n, γ) reaction on Lu-176 or the (n, p) reaction on Yb-176. Hence, the use of Lu-177 has grown exponentially in the past two decades. The (n, p) route yields 'no carrier added' (n.c.a.) grade high specific activity Lu-177, and is devoid of the long-lived Lu-177m (half-life 160.4 days), but the yields are far lesser than that for (n, γ) route (Dash et al., 2015). The major impact of Lu-177 has been due to the use of Lu-177-DOTATATE (a theranostic pair for In-111-Octreotide), for treatment of neuro endocrine tumors, though Lu-177-phosphonates were among the early radiopharmaceuticals to be used in palliative therapy. In recent times, Lu-177-PSMA-17 for treatment of prostate cancer has been used increasingly. The trivalent radiolanthanide chemistry is advantageously used to pair with trivalent diagnostic RNs such as Ga-67 and In-111 to develop Lu-177 based radiopharmaceuticals for therapy (Das and Banerjee, 2016).

Thallium-201

Tl-201, which decays through electron capture with a half-life of 73.1 h, was one of the early RNs used for myocardial perfusion imaging (Ritchie et al., 1977). Though considered the gold standard since Tl + 1 is analogous to K + 1, the use of Tl-201 had reduced considerably with the advent of Tc-99m-MIBI, primarily because Tc-99m from the Mo-99/Tc-99m generator is easy to access in comparison to Tl-201 produced in cyclotrons. But the use of Tl-201chloride returned in clinics after the Mo-99 supply shortages experienced during 2007–10 period and continues to be used (Pagnanelli and Basso, 2010).

Yttrium-90

Y-90, a pure beta-emitting RN (2.27 MeV $E_{\max}$) with a half-life of 64 h, availed from a Sr-90/Y-90 generator, is an ideal isotope for radionuclide therapy of large lesions (refer to the section on the Sr-90/Y-90 generator). The trivalent chemistry of yttrium makes it a suitable theranostic pair of Ga-68, and a good option for the treatment of large lesions for which Lu-177 is not the best choice. Yttrium-90 labeled monoclonal antibody ibritumomab tiuxetan (trade name Zevalin) is one of the two radiolabeled MABs for the treatment of non-Hodgkin's lymphoma. Apart from this product, use of Y-90 labeled microspheres (glass or resin) for selective internal radiation therapy (SIRT) of liver cancers (Salem and Thurston, 2006), Y-90 labeled peptide analogs of somatostatin in treatment of large neuroendocrine tumors (Otte et al., 1998; Bodei et al., 2004), and Y-90 phosphonates for palliative therapy in skeletal metastases have been reported with promising results. Use of Y-90 labeled particles for radiation synoviorthesis of large joints is also practiced in selected clinics (Chrapko et al., 2007).

Rhenium-188

Re-188 is another ideal therapeutic RN, as it emits high energy beta particles, ($E_{\max}$ 2.12 MeV) along with a low percentage of gamma emission (15% of 155 keV). It can be availed from a generator (refer W-188/Re-188 generator) and chemically belongs to the same group as Tc-99m, making it a theranostic pair of Tc-99m. However, its use depends on the generator availability, as the direct neutron activation production of Re-188 by irradiating enriched Re-187 has limitations of RN impurities and low yields. Re-188 labeled lipiodol for treatment of hepatocellular carcinoma (Jeong et al., 2001) is the most reported application, while use of Re-188 phosphonates for palliative treatment of skeletal metastases (Lepareur et al., 2019) and Re-188 particulates in radiation synoviorthesis of large joints are also practiced in limited centers (Knapp and Dash, 2016).

$^{227}\text{Ac}/^{223}\text{Ra}$ and $^{225}\text{Ac}/^{213}\text{Bi}$

Alpha radiation with high LET is attractive for therapy when the RN can be lodged close to the cancer cells. Among the various alpha emitters explored for therapy, Ra-223 and Ac-225 have shown promising results (Morgenstern et al., 2020; Kratochwil et al., 2020) and are detailed below:

Ra-223: Radium-223, in the form of radium chloride, is the first targeted alpha therapy radiopharmaceutical approved for clinical use for the palliative treatment of castration-resistant prostate cancer (CRPC) patients with symptomatic bone metastases (Brito and Etchebehere, 2020). Radium-223 is selectively accumulated in the bone, specifically in areas of high bone turnover, by forming complexes with the mineral hydroxyapatite (the inorganic matrix of the bone). The alpha radiation generated during the radioactive decay of radium-223 produces a palliative anti-tumor effect on the bone metastases. The treatment has been shown to be useful in life span extension and pain palliation in patients. Ra-223 chloride was the first commercial alpha therapy product. However, the radiolabeled phosphonates (Sm-153; Lu-177; Re-188) and ^{89}Sr -chloride, which have been used for the same purpose, have not been replaced by Ra-223 chloride—both due to availability and affordability.

Ac 225: The initial success of ^{223}Ra chloride led to the exploration of other alpha emitting RNs for therapy—some of which had been studied in the past, but abandoned due to practical difficulties in obtaining the RN as well as targeting. ^{225}Ac , with a half-life of 10 days, emits alpha particles and has a sequential decay pattern resulting in several alpha particles (as shown below Table 5). ^{225}Ac sequentially decays with α particle emissions to ^{221}Fr (4.9 min); ^{217}At (32.3 ms) and ^{213}Bi (46 min), and all the four α particles aid in therapy. But, such high energy alpha emission could result in recoil energy that is much higher than the energy of the chemical bond of the nuclide, and this could lead to the release of a daughter nuclide from the radiopharmaceutical molecule that is usually a conjugate of the target specific vector with a chelating moiety that holds the radionuclide. In such a case, the radionuclide will no longer be carried to the target, and could lead to normal tissue toxicity. For example, if ^{221}Fr (4.9 min), the decay of the product of ^{225}Ac , gets detached from the target specific molecule (it is similar to potassium, same group in periodic table and analog), it can be taken up in myocardium, which is not desirable. To avoid such problems, an option is to consider using ^{213}Bi , which results after four successive alpha emissions from ^{225}Ac .

The current research on ^{225}Ac is also focused on production of ^{225}Ac , since the current supplies from ^{229}Th are limited (60–70 GBq/year) and can be used to treat only a few thousand patients. ^{229}Th is obtained from the decay of ^{233}U , availed from the old legacy stocks existing in a few countries in the world (US-DOE Isotope Program; Russian Federation-Institute of Physics and Power Engineering, Rosatom; Europe- European Commission-Joint Research Centre and Canada-Canadian Nuclear Lab). Currently, the methods for production of ^{225}Ac that have shown some promise are (i) bombarding high energy protons on thorium targets in high energy proton LINACs (^{232}Th (p, spallation)), (ii) ^{226}Ra (p,2n) in a high energy accelerator, (iii) photon induced (γ , n) reaction on ^{226}Ra in electron accelerators where in the ^{226}Ra obtained decays to ^{225}Ac , and (iv) (n,2n) reaction on ^{226}Ra using high energy neutrons. All these methods have inherent challenges related to low yields, radionuclide impurities, and handling radioactive gas (^{222}Rn), which must be overcome to make them practical and sustainable.

^{213}Bi has also been widely studied as a therapeutic radionuclide and is availed from ^{225}Ac , in the form of a generator. But ^{225}Ac has the therapeutic advantage of alpha cascade and a much longer half-life (10 days) compared to that of ^{213}Bi (46 min), which is favorable for bio-distribution within the body and is, hence, avidly pursued for targeted therapy (Scheinberg and McDevitt, 2011). The short half-life of ^{213}Bi will also be a constraint in preparing clinical doses of ^{213}Bi radiopharmaceuticals, although generator production has logistic advantages.

Several other RNs such as Zr-89 (Zhang et al., 2011), Sc-44 (Rösch, 2012), Ti-45 (Chaple and Lapi, 2018) have been reported with promising results or high potential for use and are yet to be used in regular clinical practice. Hence, they are not detailed here. Other RNs, such as Rb-82, are used regularly for specific purpose and have already been detailed in the earlier section on generators.

As mentioned earlier, the concept of “Thera(g)nostics” is increasingly used for planning therapy that is personalized for better efficiency. Therapeutic radionuclides with imageable gamma emission have been used for imaging followed by therapy since long. I-131 iodide is the oldest in this context. Among the several “theragnostic” pairs listed in Table 4, a few (Ga-68 and Lu-177/Ac-225; I-123 and I-131) are commonly used.

Conclusion

Nuclear medicine is a unique modality, enabling accurate diagnosis, therapy as well as monitoring patients post treatment in several major diseases and, thus, significantly contributes to human welfare and longevity. Radionuclides, a vital part of the radiopharmaceuticals used in nuclear medicine, have constantly evolved. A wide variety of radionuclides are currently available for imaging, making it possible to obtain unequivocal diagnostic information as well as for efficient targeted therapy.

See Also: An Introduction to Research Reactors; Applications of Research Reactors.

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Medicine: Radiopharmaceuticals and Their Use in Nuclear Medicine

Keon Wook Kang^a and Meera Venkatesh^{b,*}, ^a Department of Nuclear Medicine, Seoul National University College of Medicine, Seoul, South Korea; and ^b Division of Physical and Chemical Sciences, International Atomic Energy Agency, Vienna, Austria

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Glossary

Those in italics are defined in more detail in Chapter 3A (Venkatesh and Kang, 2021).

Alpha rays *Alpha particles, identical to a helium nucleus.*

Amyloid plaques A pathological feature of Alzheimer's disease, which is aggregates of misfolded proteins that form in the spaces between nerve cells.

Becquerel (Bq) Standard International (SI) derived unit of radioactivity, equal to one disintegration per second (1 dps). The non-SI unit of radioactivity "curie" (denoted as Ci, which is the radioactivity in 1 gram of radium) is related to Bq as $1 \text{ Ci} = 3.7 \times 10^{10} \text{ Bq}$. Becquerel is an extremely small amount of radioactivity and in practice, the amounts of radioactivity are several orders of magnitude higher. They are represented as kilo Bq (kBq), mega Bq (MBq), giga Bq (GBq), tera Bq (TBq), etc.

Beta particles *Particles with a mass of electrons; Most have a negative charge and are called negatrons; Those with a positive charge are called positrons.*

Biological half-life Molecules that have biological action (such as drugs, radiopharmaceuticals, etc.) get depleted from the body (either due to excretion or conversion to another molecule or disintegration due to instability) with time. The time taken for the original amount of the active component to decrease to half its original amount is known as biological half-life. In the case of radiopharmaceuticals, both the physical half-life of the radionuclide as well as the biological half-life of the biomolecule are applicable.

C-11-PIB Carbon-11-labeled Pittsburgh Compound B, (C-11 PIB), is a radiotracer used with positron emission tomography (PET) scanning to image the build-up of beta-amyloid plaques in the brain, which are associated with Alzheimer's disease.

CAD/IHE Coronary Artery disease/Ischemic Heart disease—Heart diseases could be due to different causes (such as malfunctioning valves or shunt), among which the most common is due to abnormal (often blockage) coronary artery leading to poor blood flow and affected regions (ischemia) in the heart muscle. Heart diseases due to defective coronary artery leading to ischemia of heart muscles are called "Coronary Artery Disease or Ischemic Heart Disease." A diagnostic imaging test to visualize the coronary arteries and the blood flow to heart muscles helps the cardiologist treat the patient in an appropriate way.

Castration resistant prostate cancer (CRPC) Prostate cancer is one of the common cancers in males. As prostate cancers are often driven by the male sex hormones androgens, the first line of treatment for prostate cancer patients is androgen deprivation therapy, either through surgery or with drugs. However, in some cases, the cancer continues to grow and at times metastasize despite hormone therapy and irrespective of the levels of androgen. This disease state is referred to as castration-resistant prostate cancer (CRPC).

Chelate Chelate is a compound wherein a metal ion is attached through two or more coordination bonds to atoms of a ligand, which is usually an organic molecule. The word chelation is derived from *chēlē*, meaning "claw" in Greek. In a chelate, the central atom is attached with one or more ligands called chelators like the claws of a lobster.

Chemisorption A kind of adsorption that involves a chemical reaction between the surface and the adsorbate.

*Retired.

Cisternography An imaging test to diagnose a cerebrospinal fluid (CSF) leak or identify blockages in the CSF pathway. By injecting a small amount of radioactivity into the spinal fluid via lumbar puncture, the flow of the spinal fluid from the spine into the brain can be evaluated.

Decay constant (λ) *The probability of transmutation of a radionuclide.*

Diagnostic nuclear medicine *The branch of nuclear medicine used for diagnostic purposes.*

Dyspnea Shortness of breath, a feeling of not being able to breathe well enough.

Ejection fraction (EF) *The fraction of blood pumped out of the heart every time the heart contracts.*

Electrocardiogram (ECG) *A test done to measure the electrical activity of the heartbeat.*

Electron capture (EC) *A mode of transmutation wherein an orbital electron is captured, causing a proton in the nucleus to become a neutron.*

Etiology The study of causation or origination.

Gamma rays *Photons with an electromagnetic radiation.*

Half-life *The time taken for one-half of the original decaying radionuclide to transmute into its daughter product.*

Hydrocephalus An abnormal accumulation of cerebrospinal fluid (CSF) within cavities in the brain called ventricles.

Ictal phase The middle stage of a seizure.

In vitro *In glass or outside the body.*

In vivo *Inside the body.*

Interictal phase The period between seizures, or convulsions, that are characteristic of an epilepsy disorder.

Linear energy transfer (LET) *The transfer of energy through a medium by radiation (such as alpha or beta rays).*

Lipophilicity The ability of a chemical compound to dissolve in fats, oils, lipids, and non-polar solvents.

Molecular imaging *Signaling via a bi-molecular action of the tracer used.*

Monoclonal antibodies (MAB) *Antibodies derived from the same "clone," having identical characteristics and behavior.*

Neuroblastoma A cancer that develops from immature nerve cells. It most frequently starts from one of the adrenal glands but can also develop in the neck, chest, abdomen, or spine. It is the most common cancer in babies and the third-most common cancer in children after leukemia and brain cancer.

Norepinephrine secreting cell Norepinephrine is a hormone and neurotransmitter, excreted during situations of stress or danger, in the so-called fight-or-flight response. Norepinephrine secreting cells are in the sympathetic ganglia near the spinal cord or in the abdomen and the adrenal glands.

Osteomyelitis An infection of bone.

Paraganglioma A rare neuroendocrine (cells that secrete hormones in response to signals from nervous system) neoplasm (neoplasm is abnormal growth of cells/tissues—could be cancerous or non-cancerous) that may develop at various body sites (including the head, neck, thorax and abdomen). When the same type of tumor is found in the adrenal gland, they are referred to as a pheochromocytoma.

Peptide receptor radionuclide therapy (PRRT) *A therapy of cancers that overexpress specific peptides on cell surfaces using radiolabeled peptides that bind specifically to these receptors.*

Phagocytosis The process by which a cell uses its plasma membrane to engulf a large particle. It is a main mechanism of the innate immune defense.

Pheochromocytoma A rare, chromaffin cell tumor of the adrenal medulla. It results in the release of too much epinephrine and norepinephrine—hormones that control heart rate, metabolism, and blood pressure. This can lead to high blood pressure.

Positron emission tomography (PET) *A tracer technology based on using positrons to isolate a tomographic image.*

Positrons *Beta particles with a positive charge.*

Pyelonephritis Inflammation of the kidney, typically due to a bacterial infection.

Radioembolization A cancer treatment in which radioactive particles are delivered to a tumor through the bloodstream.

Radiolabeled diagnosis (RID) *A nuclear medicine procedure wherein a radiolabeled antibody is used for diagnostic purposes.*

Radiolabeled therapy (RIT) *A nuclear medicine procedure wherein a radiolabeled antibody is used for therapeutic purposes.*

Radionuclide *An unstable isotope that decays into a more stable isotope via radioactive emission.*

Radiopharmaceuticals *A drug formulation containing radioactivity.*

Radiosynoviorthesis A treatment for the restoration (orthesis) of the synovium (the connective tissue that lines the cavities of joints and makes synovial fluid, which lubricates the joints) by local administration of radioactive agents.

Red blood corpuscles (RBC) *Red blood cells that carry oxygen throughout the body.*

Reticuloendothelial system A part of the immune system that consists of the phagocytic (eating) cells located in reticular connective tissue, such as lymph nodes, the liver, and the spleen. The phagocytic cells, such as macrophages (white blood cell that engulfs and digests cell debris, foreign substances, microbes, etc.) and endothelial cells (cells lining the interior surface of blood vessels), clear macromolecules and nanoparticles from the blood circulation.

Single photon emission computed tomography (SPECT) *A tracer technology based on a gamma emitter to isolate a tomographic image.*

Sodium iodide symporter (NIS) NIS is a membrane protein that aids in the transport of *iodide* ions (and sodium ions) into thyroid follicular cells. It plays a crucial role in *iodine* metabolism and thyroid regulation.

Somatostatin receptor imaging A molecular imaging system that targets somatostatin receptors, which are over-expressed in neuroendocrine tumors.

Tachypnea A respiration rate greater than normal, resulting in abnormally rapid breathing.

Tau protein A microtubule-associated protein in neurons. Aberrant assembly of tau into insoluble aggregates is accompanied by synaptic dysfunction and neural cell death in a range of neurodegenerative disorders, such as Alzheimer's disease, Parkinson's disease, etc.

Therapeutic nuclear medicine *The branch of nuclear medicine in which radiopharmaceuticals are labeled with radionuclides for therapeutic purposes.*

Trans arterial radioembolization (TARE) A catheter-based liver-directed therapy that involves injection of micron-sized embolic (blocking caused by pieces of material inside a blood vessel) particles loaded with a radioisotope by use of percutaneous transarterial techniques.

Introduction

Nuclear medicine (NM) is the specialty in which a radiolabeled substance known as a *radiopharmaceutical* is administered into the patient (also referred as “in vivo” administration), either for diagnostic or for therapeutic purpose. While the “traceability” of radioisotopes allows the imaging of radiopharmaceuticals, which in turn can provide information about an organ (size, shape, abnormalities, functioning), the deleterious nature of high-energy, high dose-rate radiation is used in therapy to destroy diseased cells, such as cancer cells. Since the behavior of the radiopharmaceuticals inside the body is governed by the biochemistry of the molecule and the biochemical functioning of the concerned body organs, nuclear medicine studies provide unique functional information unlike several other imaging modalities. Thus, nuclear medicine imaging is also referred to as functional imaging or molecular imaging (Venkatesh and Kang, 2021).

Currently, nuclear medicine offers diagnostic procedures to image all the important organs and functioning of vital organs and therapeutic options for several disease conditions. Nuclear medicine imaging enables anatomical as well as functional (characterization and measurement of biological processes at molecular/cellular/whole organ level) evaluation. It has evolved as an important branch of medicine providing unique and unequivocal support for patient management, especially for patients with cancers or heart diseases.

Radiopharmaceuticals

Radiopharmaceuticals are radiolabeled molecules administered into the patient, most often through intravenous route and in some cases orally; hence, termed “in vivo” radiopharmaceuticals. The basis of the use of radiopharmaceuticals is the unique biological behavior of the radiopharmaceutical inside the body and localization in target organs/lesions. This enables either diagnostic imaging to look at various organs and their functioning or therapeutic dose delivery to diseased tissues. Radiopharmaceuticals have two components; namely, the radionuclide and the molecule that is radiolabeled. The choice of the radionuclide will depend on the intended use—either diagnosis or therapy. The molecule used for radiolabeling will determine the fate of the radiopharmaceutical when it is administered into the body.

At the beginning of the practice of nuclear medicine, the radiopharmaceuticals were often simple chemicals in their inorganic form, which had the desired biological behavior inside the body. This could be considered as the first-generation radiopharmaceuticals, and Iodine-131 in the form of iodide is an excellent example. Iodide ions, when ingested, get absorbed in the blood stream and quickly concentrate in the thyroid gland. This attribute was used advantageously to image the thyroid as well as to treat thyroid cancers. A few other radiopharmaceuticals, which are simple inorganic molecules currently in use, are Tc-99 m as pertechnetate ($^{99m}\text{TcO}_4^-$, used in thyroid imaging), Tl-201 as Thallous chloride ($^{201}\text{TlCl}$, used in myocardial perfusion (blood flow to the heart muscle) imaging/imaging the flow of blood in the heart), and Ra-223 as radium chloride ($^{223}\text{RaCl}_2$, used in palliation of pain from skeletal metastases).

However, not all organs could be so easily targeted using a simple molecule. The next range of radiopharmaceuticals were radiolabeled molecules consciously designed to have desired charge, size, lipophilicity (affinity to fatty molecules), etc. to target the organ/tissue of focus. Here, the knowledge about the biochemical functioning of the organ guided the choice of molecules. For example, it was well known that the liver processes lipophilic (fatty) molecules and kidneys process hydrophilic (water soluble) molecules. Such insight formed the basis to design radiopharmaceuticals that can be used to image these organs. The knowledge that potassium cation (K^+) binds to heart muscles (myocardium) formed the basis for the use of Tl-201 as a Tl^+ ion and cationic complexes of Tc-99 m for imaging myocardial perfusion. In a few cases such as spleen, lungs and bone marrow, which have a reticuloendothelial system, physical trapping has been used to target and localize the radiopharmaceuticals.

As the understanding of the biochemistry expanded, and with advances in molecular targeting of various organs/disease conditions, a huge variety biochemicals and biologically active molecules have been radiolabeled and used as radiopharmaceuticals. An outstanding example is the use of radiolabeled glucose to image metabolic activity, which has proved to be an excellent way to deduce information on active tumors/cancers—since they have metabolic rates far higher than normal tissues. Biologically active specific targeting molecules such as peptides and antibodies have had a great impact on the management of cancers and have also influenced the field of radiopharmaceuticals. Radiolabeled peptides and antibodies that can bind to cancer cells with great specificity are now used widely, and consequently, the terms radio immuno diagnosis (RID), radio immuno therapy (RIT), and peptide receptor radionuclide therapy (PRRT) have been coined.

One way to classify the targeting mechanism of the current radiopharmaceuticals is as follows:

- a. Active/metabolic transport (example: I-131 used in thyroid imaging/cancer therapy).
- b. Phagocytosis/entrapment of particles (example: Tc-99 m sulfur colloid for liver/spleen imaging).
- c. Entrapment due to size/blockade (example: Tc-99 m serum albumin macroaggregates for lung perfusion imaging).
- d. Diffusion across cell membranes (example: F-18 fluoride in bone imaging).
- e. Compartmental localization- the radiopharmaceutical distributed in an enclosed volume such as the blood pool, GI tract, lung airways, lymphatic channels, urinary tract, etc. (example: Xe-133 gas in lung perfusion imaging).
- f. Chemisorption (example: binding of radiolabeled phosphates to the bone).
- g. Ion exchange (example: binding of calcium⁺² analogs such as Sr⁺² to the bone).
- h. Cell sequestration—as in spleen, use of heat-treated RBC labeled with Tc-99 m.
- i. Immune binding—specific binding of the antibody and antigen used for targeting (example: radiolabeled monoclonal antibodies in RIT and RID).
- j. Receptor binding (example: use of peptide analogs to target tissues/lesions expressing specific receptors—such as In-111 or Tc-99 m Octreotide to target somatostatin receptor expressing neuro endocrine cancers).

A major difference between the “radiopharmaceuticals” and regular pharmaceuticals is in the amount of the targeting molecule administered. The amount of the biomolecule in radiopharmaceuticals is in nanomolar quantities, which is several magnitudes lower than the regular pharmaceuticals. The radionuclide component is the one that provides the information (in diagnostics) or causes cell damage (in therapy). The role of the bio-active molecule is primarily to carry the radionuclide to the target tissue. Hence, radiopharmaceuticals are not expected to cause physiological effects when administered.

An important aspect of radiopharmaceuticals is compliance with regulatory requirements for administration into patients (IAEA, 2018; Saha, 2018). These quality control measures are extremely important and are crucial for radiopharmaceutical preparation and supply. The quality control (QC) tests are aimed at testing and ensuring the quality from physical, chemical and biological angles. In general, the QC tests include testing for (i) radionuclidic purity to ensure that the levels of unwanted radionuclides that can give undesired radiation dose are below the allowed levels, (ii) radiochemical purity to ensure that the radionuclide is present in the desired chemical form and that the unwanted radiochemical impurities are below the allowed limits, (iii) sterility of the product, (iv) (in injectables) absence of pyrogens (substances that cause fever), (v) chemical purity to ensure that the levels of chemicals used in preparation of the radiopharmaceuticals are below the allowed limits, and (vi) the parameters such as pH, molarity, etc. are as desired as given in the product manual.

Tc-99 m Based Radiopharmaceuticals: Radiopharmaceuticals tagged with ^{99m}Tc and ¹⁸F are the most used. ^{99m}Tc based radiopharmaceuticals, which are currently available for imaging nearly all the important organs and functions of the vital organs, are listed in **Table 1**.

Technetium does not occur naturally and, hence, Tc-99 m is a foreign entity to all molecules. The versatile chemistry and multiple oxidation states of Tc are great advantages for radiolabeling molecules with Tc-99 m. Nevertheless, it is often a challenge to radiolabel a biomolecule with Tc-99 m to get a radiolabeled molecule that retains its integrity after administration into the body. Tc-99 m, a radiometal, is best incorporated into a molecule through complexation using suitable chelating atoms/moieties. In the case of Tc (a soft metal) oxygen, nitrogen, or sulfur are good chelating atoms that can form strong complexes. The radiometal center with the surrounding chelating groups is of significant size and could influence the in-vivo behavior of the molecule, especially if the molecule is small. Tc-99 m radiopharmaceuticals could be categorized as follows:

- 1) Radiopharmaceuticals that are bare ^{99m}Tc-complexes owe their specificity to the whole complex and not the chelating molecule alone. Here, Tc-99 m is an important component of the biological action. These are “radionuclide essential” radiopharmaceuticals. The brain imaging agents ^{99m}Tc-ECD (ethyl cysteinate dimmer), ^{99m}Tc-hexakis methoxy isobutyl isonitrile (^{99m}Tc-MIBI), ^{99m}Tc-HMPAO (d, l hexa methyl propylene amine oxime), and several other small complexes of Tc-99 m in the above table, belong to the RN essential radiopharmaceutical category.
- 2) In the case of physical particles, such as colloids labeled with Tc-99 m, the mechanism of radiopharmaceutical action is based on the physical size. Here, the labeling is not specific, and the radiopharmaceutical is not “radionuclide essential” type. An example is sulfur colloid labeled with Tc-99 m.
- 3) Most large molecules, such as proteins, are radiolabeled using chelating moiety either already present in the molecule or specifically attached for complexation of the RN. Such specifically attached chelating moiety is known as “bifunctional chelating agent” (BFCA), as this moiety is linked to the bioactive molecule on one end and holds on (chelates) the RN at the other. These are also “radionuclide non-essential” radiopharmaceuticals. Peptide and antibodies labeled with Tc-99 m belong to this category.

Table 1 Tc-99 m based radiopharmaceuticals.

Radiopharmaceutical	Use
^{99m}Tc -pertechnetate	Thyroid and salivary gland imaging
^{99m}Tc -hexakis methoxy isobutyl isonitrile (^{99m}Tc -MIBI);	Myocardial perfusion imaging
^{99m}Tc -ethoxy ethyl 3,12-dioxo-6,9-diphosphate tetradecane (^{99m}Tc -tetrofosmin)	
^{99m}Tc -pyrophosphate	Myocardial infarct (tissue that is damaged and not functional due to lack of blood supply) imaging
^{99m}Tc -HSA microspheres	Lung perfusion studies
^{99m}Tc -aerosols	Lung ventilation studies
^{99m}Tc -phytate	Liver scintigraphy/imaging
^{99m}Tc -mebrofenin	Hepatobiliary tract scintigraphy (imaging of the liver, gallbladder and the ducts)
^{99m}Tc -diethylene triamine pentaacetic acid (^{99m}Tc -DTPA)	Renal function—glomerular (a network of small blood vessels capillaries, located at the beginning of urine flow in the kidney) filtration rate; renal scintigraphy
^{99m}Tc -glucoheptonate (^{99m}Tc -GHA);	
^{99m}Tc (III)-dimercaptosuccinic acid (^{99m}Tc (III)-DMSA)	Renal tubular function imaging
^{99m}Tc -mercaptoacetyl triglycine (^{99m}Tc -MAG3);	
^{99m}Tc -ethylene L,L, dicysteine (^{99m}Tc -EC)	
^{99m}Tc -L,L,ethyl dicysteinate dimer (^{99m}Tc -ECD);	Brain perfusion imaging
^{99m}Tc -d,l-hexamethylene propylene amine oxime (^{99m}Tc -HMPAO)	
^{99m}Tc -TRODAT-1 (a tropane derivative)	Dopamine transporter imaging in brain (in Parkinson's disease)
^{99m}Tc -methylene diphosphonate (^{99m}Tc -MDP)	Skeletal imaging; imaging skeletal metastases, osteomyelitis, arthritis, fractures, bone metabolism, etc.
^{99m}Tc -hydroxy ethylene diphosphonate (^{99m}Tc -HEDP)	
^{99m}Tc -hydroxy methylene diphosphonate (^{99m}Tc -HMDP)	
Tin-pyrophosphate followed by $^{99m}\text{TcO}_4^-$	In vivo RBC labeling for cardiac blood pool studies, etc.
^{99m}Tc -leucocytes	Infection imaging
^{99m}Tc -HSA nanocolloid	Sentinel node imaging; lymphoscintigraphy and bone marrow imaging
^{99m}Tc -sulfur colloid	Lymphoscintigraphy and bone marrow imaging
^{99m}Tc (V)-dimercapto succinic acid (^{99m}Tc (V)-DMSA)	Tumor imaging- medullary carcinoma
^{99m}Tc -HYNIC-[D-Phe1, Tyr3-Octreotide] (^{99m}Tc -Tektrotyd)	Imaging somatostatin receptor expressing neuro endocrine tumors

F-18 Based Radiopharmaceuticals: The growth in the use of F-18 was phenomenal owing to the remarkable utility of F-18-FDG in several ailments and the excellent PET images, earning it the name “Molecule of the Millennium.” F-18-FDG, in which the hydroxyl group in glucose is replaced with a F-18 atom, was originally developed to study the glucose metabolism in brain (Anderson et al., 2019). Although ^{18}F -FDG is taken up by cells, just as glucose, the lack of a 2-hydroxyl group in ^{18}F -FDG makes it different from glucose in the next steps of metabolism. Consequently, after the first step of phosphorylation, the F-18-FDG-6-phosphate formed is trapped in the cell. Thus, ^{18}F -FDG was established as a tracer to image glucose metabolism, which was used in brain/neurological research to understand disorders such as Alzheimer's disease, epilepsy, drug addiction, etc.

Towards end of the 1970s and beginning of the 1980s, the use of ^{18}F -FDG soon expanded to reveal information about the functioning of different tissues/organs, including myocardial metabolism and tumor metabolism. With the development of a radiolabeling method to obtain ^{18}F -FDG in high yields around mid-1980s, its use in cancer management grew steeply. Since cancer cells divide in an uncontrolled manner and have a high demand for glucose, the cancerous lesions light up clearly in ^{18}F -FDG imaging, much before any anatomical change is noticed. Currently, millions of ^{18}F -FDG scans are performed annually in cancer patients for diagnosis, treatment monitoring, and also for anti-cancer drug evaluation.

While F-18-FDG is the most used PET tracer currently, the attractive features of PET imaging and the availability of F-18 from several medical cyclotrons led to the quest for more F-18 labeled PET tracers for use in diagnosis of other conditions. Currently, there are some F-18 based PET radiopharmaceuticals used in clinics, as listed in Table 2, along with several F-18 based radiolabeled molecules in research that are not listed here.

Table 2 F-18 based radiopharmaceuticals.

Product	Use
^{18}F -fluoro deoxy glucose (^{18}F -FDG)	Glucose metabolism-widely used in oncology for imaging tumors
^{18}F -fluoro DOPA (^{18}F -DOPA)	Brain imaging for diagnosis of Parkinsonism and Alzheimer's, whole body imaging for neuroendocrine tumors
^{18}F -choline	Prostate cancers, parathyroid adenoma
^{18}F -fluoride	Skeletal imaging (for tumors; bone metastases)
3'-deoxy-3'-[^{18}F]fluorothymidine (^{18}F]FLT)	Thymidine metabolism-used in imaging cell proliferation—especially in brain tumors, sarcomas
O-(2-[^{18}F]fluoroethyl)-L-tyrosine (^{18}F FET)	Tumor imaging (amino acid metabolism)
^{18}F -fluoromisonidazole (^{18}F -FMISO)	Hypoxia imaging; oncology

The other commonly used radiopharmaceuticals are detailed further in the later section on the radiopharmaceuticals used in targeting specific organs.

Diagnostic nuclear medicine

Diagnostic imaging is aimed at obtaining information about (i) shape and size of organs—such as thyroid, liver, etc.; (ii) functioning of organs—such as heart, kidneys, liver, lungs, brain, etc.; and (iii) shape, size and location of tumors/nodules—in various cancers, etc. Such imaging is used to diagnose the diseases/cause for the diseases and follow up response to treatment. In addition, diagnostic imaging is also used to study the fate of the molecule of interest when administered into the body in research-oriented studies, such as drug research (Venkatesh and Korde, 2021).

In routine clinical practice, diagnostic imaging is predominantly used in cardiology and oncology, and to a significant extent in neurology to determine the functioning of vital organs such as liver and kidneys. In the earlier sections of this article, while describing the important radionuclides used in nuclear medicine, a significant amount of information on some of these RNs in diagnostic NM has been given. In the present section, the focus is on the disease conditions and the use of different radiopharmaceuticals in diagnosis. The important aspects in the imaging of major organs/systems/diseases are briefly described below.

Cardiovascular system—heart

The heart, one of the most vital organs, is assessed for its functioning utilizing several parameters for diagnosis of heart diseases, all of which can be estimated using nuclear medicine imaging and, hence, nuclear cardiology has an important role in management of cardiac patients (Taillefer and Harel, 2018; Manabe et al., 2018). Coronary artery disease (CAD), also called ischemic heart disease, is a major health concern globally. Nuclear medicine imaging aids in diagnosis/confirmation of CAD, stratification of risk, and treatment response assessment. NM imaging is highly valuable as a prognostic tool in asymptomatic patients at high risk of CAD, such as diabetes patients. Another important value of nuclear medicine imaging is in identifying patients with an extremely low risk of cardiac events, even if an invasive test such as coronary angiography suggests CAD. In addition, in all cases where stress ECG test findings are equivocal or inconclusive, nuclear medicine imaging is useful to decide if the patient needs to go for an invasive coronary angioplasty test. Thus, the negative predictive value of nuclear medicine scan can save unnecessary investigations and save resources.

Myocardial perfusion imaging (imaging blood flow in the heart) is used to delineate areas in the myocardium that are damaged (ischemic—meaning tissue that is still viable, or infarcted—meaning tissue that is dead and not viable) from normal healthy parts by imaging the heart under normal conditions and stressed (under exercise or induced through drug) conditions (Iskandrian et al., 2018). This is a valuable non-invasive evidence-based tool for cardiac care management. The radiopharmaceuticals used in perfusion imaging are $^{201}\text{TlCl}$, $^{99\text{m}}\text{Tc}$ -sestamibi, $^{99\text{m}}\text{Tc}$ -tetrofosmin, $^{82}\text{RbCl}$, and $^{13}\text{NH}_3$. $^{201}\text{TlCl}$, where the Tl^+ ion mimics K^+ ion was the first cardiac perfusion imaging agent introduced in 1973 and soon after used in clinics regularly. $^{201}\text{TlCl}$ is considered the gold standard in myocardial perfusion imaging due to the superior performance features (high: $\sim 85\%$ myocardial extraction, good: 3–4% uptake in myocardial tissues, and slow wash out from viable but ischemic tissues) that allows a single injection of the radiopharmaceutical—unlike the other radiopharmaceuticals used in perfusion imaging.

Although the corresponding features for $^{99\text{m}}\text{Tc}$ radiopharmaceuticals are not as good, the image quality is better. Accessing $^{99\text{m}}\text{Tc}$ is much easier and cheaper than ^{201}Tl , and, hence, currently most clinics use $^{99\text{m}}\text{Tc}$ based radiopharmaceuticals for myocardial perfusion imaging. The efforts to develop an ideal myocardial perfusion imaging $^{99\text{m}}\text{Tc}$ -based radiopharmaceutical were aimed at designing a cationic complex of $^{99\text{m}}\text{Tc}$ that will mimic K^+ ion (just as $^{201}\text{Tl}^+$) and be taken up by the myocardium. $^{99\text{m}}\text{Tc}$ -sestamibi ($\text{Tc}(\text{I})$ hexakis isonitrile mono-cationic complex), was the first successful molecule that came to market in 1986, followed by another monocationic complex $^{99\text{m}}\text{Tc}$ -tetrofosmin ($\text{Tc}(\text{V})$ [6,9-bis(2-ethoxyethyl)-3,12-dioxo-6,9-diphosphatetradecane]). Both these $^{99\text{m}}\text{Tc}$ radiopharmaceuticals need to be injected twice in patients to get the perfusion images at rest and stress (Fig. 1).

A few other complexes of $^{99\text{m}}\text{Tc}$ that were promising for myocardial perfusion imaging did not make it to the clinics. One example is $^{99\text{m}}\text{Tc}$ -teboroxime, a neutral complex of $^{99\text{m}}\text{Tc}$, which had the best performance with respect to myocardial extraction (higher than $^{201}\text{TlCl}$ and far better than $^{99\text{m}}\text{Tc}$ -sestamibi and $^{99\text{m}}\text{Tc}$ -tetrofosmin), but was too quickly washed out making it difficult to image. However, with advances in imaging systems that can acquire images very quickly (within 5 min—ultrafast systems), $^{99\text{m}}\text{Tc}$ -teboroxime could find a place in regular clinical use for myocardial imaging in the future.

With the advent of clinical PET imaging in the 1990s, $^{82}\text{Rb}^+$ and $^{13}\text{NH}_3$ were used in myocardial perfusion imaging. PET images have superior resolution of the “defective” regions and better quantification features. However, owing to the very short half-lives of $^{82}\text{Rb}^+$ (1.26 min) and ^{13}N (9.97 min), their use was very limited. In the last decade (triggered by the shortage in supply of $^{99}\text{Mo}/^{99\text{m}}\text{Tc}$ during 2008–10 period), there was a resurgence in their use, facilitated by the availability of $^{82}\text{Sr}/^{82}\text{Rb}$ generators and hospital cyclotrons that can provide $^{13}\text{NH}_3$.

Although a quantitative assessment of cardiac pumping (left ventricular) function is mostly done using echo-cardiography, it is possible to carry out regional wall motion study and ejection fraction (EF) computation as a part of the myocardial perfusion study. This involves initial imaging with ECG gating, with an injection given to a patient already well-positioned for myocardial perfusion imaging using a radiopharmaceutical. A tracer such as $^{81\text{m}}\text{Kr}$ is used for assessing right ventricular function too, when required.

While perfusion imaging gives information on blood flow, imaging with ^{18}F -FDG or $^{123/124}\text{I}$ -fatty acid analogs provides additional information on the viability of heart muscles, as these mimic glucose metabolism or fatty acid metabolism, respectively.

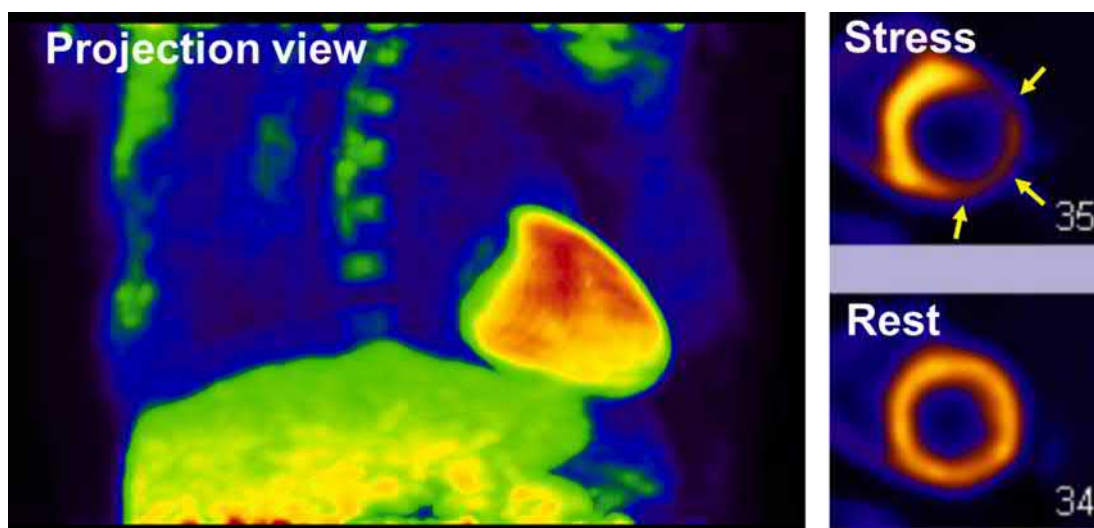


Fig. 1 Myocardial perfusion SPECT imaging using ^{99m}Tc -sestamibi. A projection view shows the heart on the left side of chest. Short axis views reveal decreased perfusion (yellow arrows) at stress state while normalized at rest.

^{18}F -Flurpiridaz (structural analogue of pyridaben, which is an enzyme inhibitor) is currently emerging as the superior PET tracer for myocardial perfusion (beyond $^{13}\text{NH}_4^+$, $^{82}\text{Rb}^+$), and is in an advanced phase of evaluation. ^{18}F -Flurpiridaz is taken up by binding to a mitochondrial complex 1 (MC1) present in myocardial cells (Liga and Neglia, 2020).

Neurology—brain

The brain is another vital organ, where various functional aspects are of great diagnostic value in determining several disorders. Brain imaging, which was limited to finding out if there is a breach in the blood brain barrier (BBB) due to presence of a tumor, has come a long way with several specific molecules now available to study various functional aspects (EANM Technologist's Guide, 2015).

Brain perfusion imaging can be carried out using radiolabeled molecules that are readily diffused into brain tissue across the BBB. Tc-99 m HMPAO and Tc-99 m ECD are used for SPECT imaging, and O-15 water is used for PET imaging. These imaging procedures are applied for evaluating ischemic brain diseases or localizing epileptic focus in the brain. Blood flow in the epileptic focus is reduced during the interictal phase and increased during the ictal phase. Since O-15 water has a short half-life of 20 s, repeat imaging is possible with less radiation dose. So, O-15 water PET is often used for visualizing brain function of the volunteers in neuroscience studies.

In-111 DTPA or Tc-99 m DTPA can be used for cisternography. If the radiopharmaceutical is injected into the spinal canal by lumbar puncture, a serial imaging shows the dynamic distribution of the radiopharmaceutical in the cerebrospinal fluid (CSF) space. RI cisternography is effective in evaluating the cause of hydrocephalus or locating the point of CSF leakage.

F-18 FDG PET reflects regional glucose metabolism in the brain. Its practical applications are in the diagnosis of dementia and localizing epileptic focus. Dementia has many different etiologies such as Alzheimer's disease, fronto-temporal dementia, vascular dementia, etc. The pattern of decreased regional glucose metabolism differentiates the cause of dementia. The glucose metabolism of epileptic focus is reduced during the interictal phase.

Multiple types of molecular nuclear imaging are capable of targeting amyloid plaques or tau protein allowing it to be practically applied in degenerative brain imaging (Chandra et al., 2019). Radiopharmaceuticals available for amyloid PET imaging are C-11 PiB, F-18 florbetaben, F-18 florbetapir, F-18 flutemetamol, and F-18 florapronol. Amyloid plaque specific imaging is not only useful for the early diagnosis of Alzheimer's disease, but also for the development of amyloid targeting therapeutics. Accumulation of radiopharmaceuticals from Tau PET imaging is well correlated with the symptom severity of Alzheimer's disease.

Dopamine transporter imaging using radiopharmaceuticals, such as [I-123] N- ω -fluoropropyl-2 β -carbomethoxy-3 β -(4-iodophenyl) nortropane (I-123 FP-CIT SPECT), and F-18 fluorinated-N-3-fluoropropyl-2-b-carboxymethoxy-3-b-(4-iodophenyl) nortropane (F-18 FP-CIT) PET, are useful in the diagnosis of Parkinson's disease (Brooks, 2016). It differentiates Parkinson's disease from other causes of Parkinson's syndrome, such as essential tremor, progressive supranuclear palsy, dementia with Lewy bodies, etc.

F-18 FDG PET is not practical for the diagnosis of brain tumor, except for glioblastoma or CNS lymphoma, since the normal glucose metabolism of the brain is high. Instead, O-(2- ^{18}F -fluoroethyl)-L-tyrosine (^{18}F -FET) or (S- ^{11}C -methyl)-L-methionine (^{11}C -MET) is used since they are taken up by the tumor through L-amino acid transporters and, in normal brain, the background is low. These imaging procedures are used for the initial diagnosis, recurrent detection, and treatment monitoring of the various neurological disorders.

Oncology—cancers

F-18 FDG PET is the most popular imaging choice for oncology (Nabi and Zubeldia, 2002). It reflects glucose metabolism and malignant tumors that consume glucose through aerobic glycolysis, which is known for the Warburg phenomenon. Since molecular imaging does not show anatomical structure in detail, in recent years most F-18 FDG imaging procedures are done in a combined system of PET and CT (PET/CT). FDG highlights the tumors on the anatomical imaging from CT scans (Fig. 2). Since oncologic PET/CT imaging covers the torso or the whole body, it is useful for staging work-up by detecting distant metastasis. F-18 FDG PET/CT is also applicable in detecting recurrence and in treatment response monitoring of cancers. Presently, a new hybrid system combining PET and MRI (PET/MRI) is commercially available. It has an advantage of reducing the radiation dose from CT and better delineation of anatomical structure in the soft tissue and the brain (Singnurkar et al., 2017).

As new molecular targets are discovered, new molecular specific cancer imaging systems have been developed. Somatostatin receptor imaging was developed for neuroendocrine tumors (NET). NET cells are highly expressed somatostatin receptors on the surface. In the last decade, In-111 octreotide scan has been used for the diagnosis of NET. Recently, somatostatin receptor PET imaging using Ga-68 DOTATOC (DOTA-Tyr³-octreotide), DOTATATE (DOTA-Tyr³-octreotate), and DOTANOC (DOTA-Nal³-octreotide) were introduced. While well differentiated NET shows high uptake of these radiopharmaceuticals, poorly differentiated NET favors the uptake of F-18 FDG, rather than the former. These somatostatin receptor imaging techniques are useful for selecting appropriate patients for peptide receptor radionuclide therapy (PRRT) with the same target (Fani et al., 2012).

Prostate cancer is generally not well visualized by F-18 FDG PET. Prostate cancer expresses prostate specific membrane antigen (PSMA) on the surface of the tumor cell. Recently, PSMA specific PET imaging is under clinical trials (i.e., Ga-68 Glu-urea-Lys(Ahx)-HBED-CC (PSMA-11), Ga-68 Glu-CO-Lys[(Sub)DLys-DPhe-DTyr(3I)-DOTAGA] trifluoroacetate (PSMA I&T), Ga-68 DOTA-GUL (Glu-urea-Lys), F-18 PSMA-1007, etc.). These imaging techniques have roles for staging, detecting recurrence, response monitoring of treatment, and selecting patients for PSMA targeting PRRT (Ahmadzadehfar et al., 2019).

Whole body bone scan has been used for detecting bone metastasis. Radiolabeled bisphosphonates such as Tc-99 m methylene diphosphonate (MDP), Tc-99 m hydroxy methylene diphosphonate (HDP), and Tc-99 m 2,3-dicarboxy propane diphosphonate (DPD), etc. are administered and images are obtained by a gamma camera or a SPECT/CT system. The incidence of bone metastasis from breast cancer, prostate cancer, and lung cancer is high. Thus, even if there are no signs or symptoms suspecting bone metastasis, imaging is indicated in order to detect hidden bone metastasis. Bone scan is not only helpful for malignant lesions but also for benign bone and joint lesion since the radiopharmaceuticals accumulate in inflammatory tissues. Such bone scans are also capable of detecting osteomyelitis, and various arthritis, such as degenerative, rheumatoid, and septic arthritis. F-18 bone PET shows whole body bones and joints and is also useful for malignant and benign bone and joint lesions.

Tc-99 m nano-colloid is used for detecting sentinel lymph node (SLN) during the operation of cancers. SNL is a first draining lymph node from the primary tumor. Some tumors, such as breast cancer and melanoma, spread mainly through lymphatic vessels.

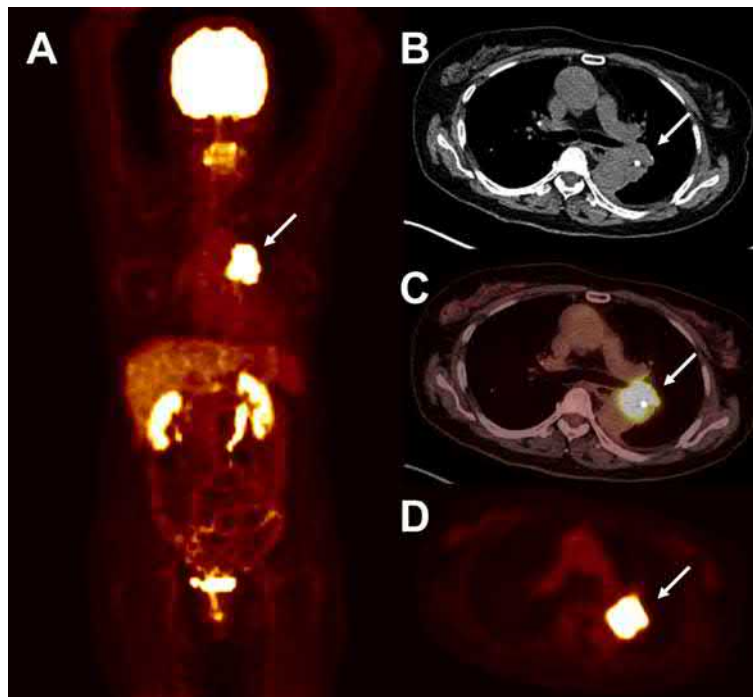


Fig. 2 Whole body F-18 FDG PET/CT scan shows lung cancer in the left hilar area (arrow). (A) is a whole body frontal view. (B and D) are transverse section images crossing lung mass in the chest from CT and PET, respectively. (C) is a fusion image of (B and D). The normal brain is bright since it is the most glucose consuming organ. There are no other abnormal hypermetabolic lesions in the scanned body.

If patients are in the early stage of cancer, minimal surgery is recommended. However, minimal surgery is only warranted when the primary tumor is proven not to spread via lymphatics. After Tc-99 m nano-colloid is subcutaneously injected near the tumor, it propagates via lymphatics. A surgeon can detect a SLN using a gamma probe during an operation (IAEA, 2015; Moncayo et al., 2019). After picking out the SLN, a pathologist may look at it under microscopy and decide the involvement of tumor. If SLN is free from metastasis, only minimal surgery is warranted.

Other organs

Various radiopharmaceuticals for organ specific physiologic imaging are also available. Thyroid tissue has sodium iodide symporters (NIS). NIS pumps in iodide, as well as technetium, to the thyroid cells. A thyroid scan using I-123, I-131, or Tc-99 m reveals the function of thyroid glands. It differentiates Grave's disease, and subacute thyroiditis, from patients with hyperthyroidism (Fig. 3).

Tc-99 m N-(2,6-dimethyl-phenyl-carbamoyl-methyl)-iminodiacetic acid (HIDA), Tc-99 m diisopropyl iminodiacetic acid (DISIDA), or Tc-99 m mebrofenin is used for hepatobiliary scans since they are taken up by hepatic cells and excreted through the biliary tract. It shows hepatobiliary function and can diagnose cholecystitis (inflammation of the gall bladder) in patients with right upper quadrant abdominal pain.

Pulmonary embolism (PE) is a blockage of an artery in the lungs by a substance that has moved from elsewhere in the body through the bloodstream. It requires emergency attention when patients have sudden dyspnea (shortness of breath) and/or tachypnea (rapid breathing). Deep vein thrombosis is the most common cause of PE. If the patient is not treated with heparin as soon as possible, the mortality rate is high. Mismatch findings in lung perfusion and ventilation scan is practical in the diagnosis of PE. PE shows the defect of pulmonary artery blood flow while ventilation is normal. Tc-99 m macro-aggregated albumin (MAA) is used for lung perfusion scans and Xe-133, Tc-99 m DTPA aerosol, or Tc-99 m Technegas are used for lung ventilation scans.

Some radiopharmaceuticals are readily excreted through the kidney and the urinary system. Renal functioning can be evaluated using Tc-99 m DTPA or Tc-99 m mercaptoacetyltriglycine (MAG3). A dynamic scan imaging after injection of these radiopharmaceuticals can then be administered to evaluate renal blood flow, cortical transit, and urinary flow. This is useful for the assessment of renal functions and obstruction of urinary system. Tc-99 m dimercaptosuccinic acid (DMSA) binds to the proximal convoluted tubules in the kidney. It can detect non-functioning cortical tissue in the kidney of acute pyelonephritis (Taylor, 2014).

Therapeutic nuclear medicine

In-vivo therapeutic radiopharmaceuticals are mostly used for the treatment of cancers and thyroid disorders such as hyperthyroidism. Radiation synoviorthesis (treatment of joint disorders where synovium is inflamed) is being practiced in a limited way. Historically, non-systematic treatment of diseases using radiation began soon after radioactive materials were available. Systematic treatment of thyroid cancers and hypothyroidism using ^{131}I and treatment of polycythemia vera (a kind of blood cancer arising from bone marrow) using ^{32}P began long ago. While the use of ^{131}I has been continued for more than 50 years, the use of ^{32}P to treat polycythemia vera is no longer practiced.

In the past 15–20 years, there has been a surge in therapeutic applications using radiopharmaceuticals. This is due to the knowledge about the molecular mechanisms of several cancers, advances in molecular biology, and the availability of highly specific molecules (especially monoclonal antibodies and peptides that bind to the cancer cell surfaces) that home in on the target tissues—most often cancer lesions. Such well-designed targeting molecules, when loaded with a radionuclide, help in specifically delivering the radiation dose to the target lesions. Hence, a huge amount of research has been going on to develop specific therapeutic radiopharmaceuticals to treat various cancers.

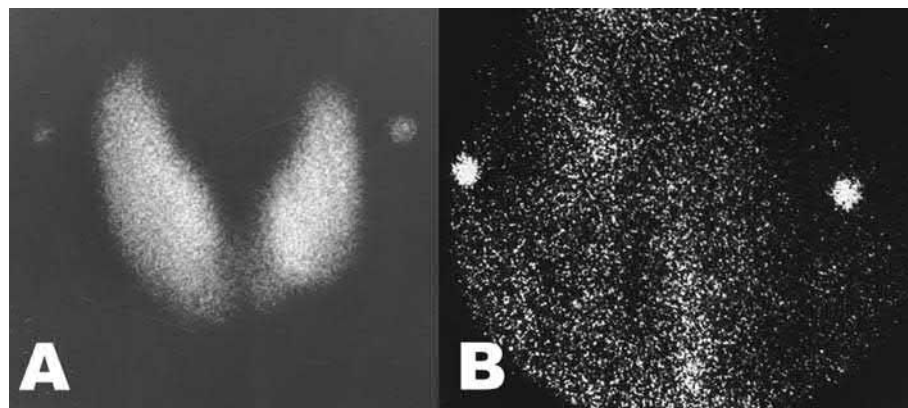


Fig. 3 Tc-99 m thyroid scan shows diffuse enlarged thyroid glands (A) in Graves's disease, while faint thyroid uptake (B) appears in the patients with subacute thyroiditis.

Unlike diagnostic radiopharmaceuticals, where a few radionuclides are found most suitable (^{99m}Tc , ^{123}I , ^{18}F , etc.), the ideal radionuclide for in vivo therapy will need to be selected based on the application. Generally high LET radiations such as α and β^- are preferred, but Auger & conversion electrons are also utilized for therapy. While α will be effective within a few cells range, β^- will be able to kill hundreds of cells. Auger and conversion electrons will act within nanometer distances and, hence, these have to be located in the DNA of the cancer cell in order to kill it. The energy of the radiation should match the volume to be treated, and the half-life should match the biological half-life of the product. Apart from these, the production feasibility, logistics, chemical amenability, etc. will also play a role in the selection of the nuclide for therapy. Generally, the radiation dose from a therapeutic radiopharmaceutical is magnitudes higher than that from a diagnostic radiopharmaceutical. Hence, caution is exercised in using therapeutic radiopharmaceuticals to avoid over-exposure of radiation to any non-target organ.

For in vivo therapy, apart from the old well-established ^{131}I and ^{32}P modalities, there are several radionuclides with high potential for therapy. Among these, ^{177}Lu , ^{90}Y , ^{188}Re and ^{153}Sm have evolved and are being used regularly.

Alpha therapy is gaining a lot of attention and several other RNs (Table 2-Chapter 3A, (Venkatesh and Kang, 2021)) have shown great promise and are being explored for clinical use. Some of them are being used in a limited way as investigational drugs. Loco-regional application of radiopharmaceuticals is recently gaining importance for treatment of well delineated tumors such as liver cancer or brain tumor, which can be treated by instilling the radiopharmaceutical within the organ/diseased tissue. In addition, internal radiation using a radiopharmaceutical has been used to treat certain joint disorders, such as rheumatoid arthritis and hemophilia. This treatment, known as "Radiation Synoviorthesis," is also a local application of a radiopharmaceutical. Another way of classifying the therapeutic radionuclides is based on the energy of the particles, which would influence the range of action in tissues (Table 3-Chapter 3A, (Venkatesh and Kang, 2021)).

The most common situations where therapeutic radiopharmaceuticals have been used successfully are briefly described below.

Radioiodine I-131 has been used for hyperthyroidism and differentiated thyroid cancer since the 1940s. It destroys both normal thyroid and thyroid cancer cells. This effect reduces the amount of thyroxine made by the thyroid gland and the size of the gland in patients with hyperthyroidism. For the treatment of advanced thyroid cancer, a range of 1.1 ~ 3.7 GBq I-131 is orally administered to ablate remnant thyroid tissues after the patients receive total thyroidectomy. It reduces the probability of thyroid cancer recurrence. If the patients have metastases initially, such as in their lungs, bone, etc., 5.5 GBq or higher dose is given to kill the metastatic tumors (Fig. 4).

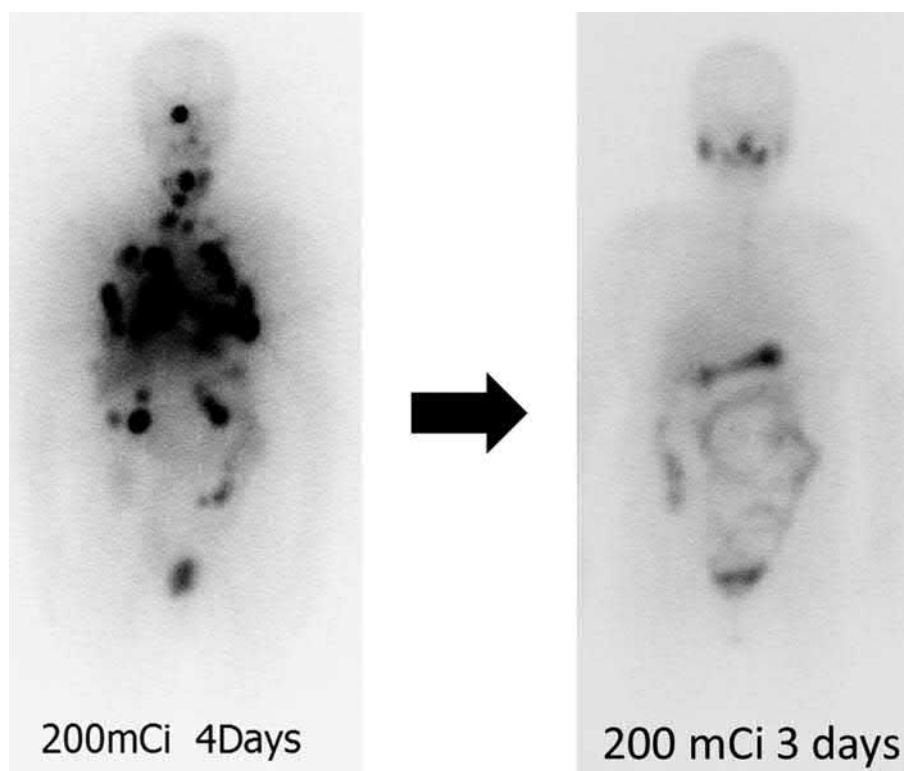


Fig. 4 Radioiodine whole body scans after 1st and 2nd 200 mCi (7.4 GBq) I-131 treatments. The first image shows multiple metastasis in the cervical lymph nodes, lung and bones. The second image 6 months after the first treatment shows only normal physiologic uptakes in the salivary glands and bowels. It tells us that the radioiodine treatment is successful to remove all metastatic lesions. Since I-131 emits both gamma rays for imaging and beta rays for treatment at the same time, whole body imaging 3–4 days after treatment is helpful to monitor therapeutic response.

If I-131 binds to other molecules, it follows that molecular behavior in the body. When I-131 is conjugated to meta iodo benzyl guanidine (mIBG), I-131 mIBG, it will channel to norepinephrine secreting cells, not to thyroid cells in the body. It also accumulates in certain neuroendocrine tumors such as pheochromocytoma, paraganglioma, and neuroblastoma. Hence, I-131 mIBG is an effective treatment for an advanced stage for such cancers when curative surgery is not indicated (George et al., 2016).

Bone metastasis is quite common in major cancers of organs such as breast, prostate, and liver. Most bone metastasis shows a high turnover rate of bone metabolism and such skeletal metastatic lesions are extremely painful. Palliation of bone pain using radiopharmaceuticals is an option. Hence, radiopharmaceuticals that are bone seekers are used for bone pain palliation therapy. Analogs of calcium ions, such as ions of Sr-89, Ra-223, or a radiolabeled bisphosphonate [such as Sm-153 ethylene diamine tetramethylene phosphonate (EDTMP) or Lu-177 EDTMP] accumulate at sites of increased osteoblastic activity and reduces cancer mediated bone pain.

Peptide receptor radionuclide therapy (PRRT) is a molecular radiation therapy targeting specific receptors that are expressed on the surface of cancer cells in the body (Fani et al., 2012; Kong and Hicks, 2019). It is one of the important treatment options for patients with somatostatin receptor-2 (SSTR2)-expressing neuroendocrine tumor (NET). Incorporating therapeutic radionuclides such as Lu-177 and Y-90 to the higher affinity octreotide analog (octreotate is associated with prolonged survival) has improved hormone associated symptoms and quality of life in patients with advanced NET. New PRRT is under development to target prostate-specific membrane antigen (PSMA), which is a transmembrane protein overexpressed in prostate cancer. Lu-177 PSMA targeted therapy is under clinical trials assessing its role in patients with metastatic castrate-resistant prostate cancer (mCRPC). PSMA-targeting alpha-particle therapy (TAT) is also in clinical studies. Ac-225 PSMA TAT demonstrates promising antitumor activity even in patients resistant to previous Lu-177 PSMA PRRT (Kratochwil et al., 2020).

Selective internal radiation therapy (SIRT) is a type of internal radiotherapy for liver cancers, sometimes called radioembolization or trans arterial radioembolization (TARE). In this procedure, glass or resin or albumin microspheres containing radioactive Y-90 are infused into the hepatic artery via a catheter, so that they preferably lodge in the vasculature of the tumor. These deliver high doses of ionizing radiation to hypervascular liver cancers such as hepatocellular carcinoma, metastatic colorectal cancer, or metastatic neuroendocrine tumors. The radiation will lead to damage of tumor tissue and control of inoperable liver cancers. Another strategy that is used to treat liver cancers is the localized infusion of radiolabeled (I-131 or Re-188) lipiodol through the hepatic artery (Uccelli et al., 2011).

Radiosynoviorthesis is an effective procedure for treating inflammatory joints, especially in hemophilia and rheumatoid arthritis (Fischer and Mödder, 2002; Chojnowski, et al., 2016). Depending on the size of the joint, various radiopharmaceuticals such as Y-90 (large joints-knee), Re-186 (medium joints-elbow, etc.), or Er-169 (small-finger phalangeal) in a colloidal or particulate form are injected into the joint space. The radiopharmaceutical accumulates in the inflamed synovial membrane by phagocytosis. Local irradiation prevents further destruction of the joint by immunological reactions.

Alpha therapy

Therapy using alpha particles has gained a lot of attention in the past decade. The use of radioactive radium in therapy, soon after its discovery and availability in limited quantities, was empirical and was abandoned owing to the undesired health effects caused due to high radiation levels. With advances in specific targeting, it is now possible to deliver the radiopharmaceuticals accurately in the target lesion, and alpha radiation can be extremely efficient in killing the cancer cells when it is in the lesion, due to their high LET. The alpha emitting radionuclides listed in Table 2 and Table 3 of Chapter 3A (Venkatesh and Kang, 2021) are those pursued currently, while several more have been explored. To date, Ra-223 and Ac-225 have been used in patients with impressive success.

Ra-223 chloride was the first commercial alpha therapy product (Xofigo). It accumulates in metastatic bone lesions and alleviates bone pain and extends survival in patients with prostate cancer [Brito and Etchebehere, 2020]. The initial success of ^{223}Ra chloride led to an exploration of other alpha emitting RNs for therapy, among which ^{225}Ac is currently explored avidly for therapy. ^{213}Bi , which results after four successive alpha emissions from ^{225}Ac , has also been widely studied as a therapeutic radionuclide and is availed from ^{225}Ac in the form of a generator. The longer half-life (10 days) of ^{225}Ac compared to that of ^{213}Bi (46 min) is favorable for bio-distribution within the body.

In recent years, ^{225}Ac -PSMA has shown impressive results in castration resistant prostate cancer patients with metastasis. The success reported by the initial researchers has encouraged and triggered similar work around the world. Those who can access Ac-225 from the stock available, or those who are able to produce some quantities of Ac-225 in advanced cyclotrons, are using ^{225}Ac to carry out clinical trials in treating patients with ^{225}Ac -PSMA. Several groups have reported promising results in scientific journals (Kratochwil et al., 2020; Zacherl et al., 2020; Yadav et al., 2020).

Peptide and antibodies for RN therapy

Targeted radionuclide therapy has grown manifolds in the past two decades, especially owing to the initial success with radiolabeled peptides. Octreotide, the octa peptide analog of Somatostatin, when labeled with In-111 and then with Tc-99 m and Ga-68, gave impressive results in diagnostic imaging of neuro endocrine cancer lesions, which was very useful for planning treatment when the lesions are spread all over the body. This led to the use of the same peptide labeled with a therapeutic RN such as Yttrium-90, and later with Lutetium-177, to treat the lesions, which gave good results. The quest for similar targeting peptides has continued, and in

recent times peptides that specifically bind to Prostate Specific Membrane Antigen (PSMA) that is expressed on prostate cancers have emerged as good candidates.

However, many other peptides (such as vasoactive intestinal peptide), which had also shown great promise, did not yet reach the clinics. Ga-68-PSMA is currently used for diagnosis and suitable patients have been treated with therapeutic RN labeled PSMA such as Cu-67, Lu-177, Y-90, and Ac-225. Among these, Ac-225-PSMA has shown excellent results in the clinical trials and these results have led to concerted efforts to make Ac-225 available for clinical use in adequate quantities (Kratochwil et al., 2020).

Antibodies, natural defense molecules that our body makes, specifically bind to the antigen against which it is directed. These are excellent targeting moiety when the antibody is directed against the tissue/lesion of interest. The invention of hybridoma technology (César Milstein and Georges Köhler invented in 1975; awarded Nobel Prize in 1984) to make antibodies in labs, and especially all of them identical, known as “monoclonal antibodies” (MAb), has been a boon in the treatment of cancers, which express unique antigens on their surfaces. Currently many MABs are used in regular clinics to treat specific cancers (Table 3).

Nearly all of these have also been radiolabeled and studied as a radiopharmaceutical for radioimmunotherapy. However, thus far only two radiolabeled MABs—both against CD20 antigen for treatment of B Cell Lymphoma—have been approved for regular clinical use; namely, Zevalin labeled with Y-90 and Bexxar labeled with I-131.

Theranostics

The concept of “personalized” treatment, based on the distribution of the radiopharmaceutical in the patient’s body, has grown in the past decade. This is especially important when a radiopharmaceutical accumulates to a different extent in the target lesions, in different patients, which is predominantly noticed in patients with neuroendocrine tumors treated with a radiolabeled (Y-90 or Lu-177) somatostatin receptor binding peptide. In these cases, the amount of therapeutic dose given is huge (in the ranges of 3–4 GBq) and the treatment is repeated 3–4 cycles. Dosimetry plays a very important role in therapy. If the exact distribution of the radiopharmaceutical molecule in the patient’s body is accurately determined, it helps immensely to calculate the exact amount of therapeutic radiopharmaceutical dose to be administered. Hence, the same targeting molecule, radiolabeled either with the same nuclide but in different dose levels, or a radionuclide of the same element that is suitable for diagnosis, or a surrogate radionuclide with very similar properties (to ensure the same molecular structure, which will have the same bio-distribution) is used for the diagnostic scan in planning the therapy.

This concept, now known as “Theranostics” was used in thyroid cancer treatment with I-131 iodide (see also (Venkatesh and Kang, 2021)). A small dose of ¹³¹Iodide used to be given for ensuring that the cancer lesions are indeed accumulating the radioiodide and to calculate the therapeutic dose to be administered. This practice is still followed. In the case of thyroid cancers, there are a few kinds that do not accumulate radioiodine, and such patients will need a different treatment.

Some therapeutic radionuclides that emit imageable photons can themselves be used in diagnostic quantities to get the diagnostic scan, as in the case of I-131. But, if there is a choice of another RN with better diagnostic features (no particulate emission, such as SPECT or PET RN), one may prefer to use it to avoid an undesired dose. The choice of I-123 in place of I-131 is such an option. In the case of RNs that are pure beta emitters, such as Sr-89 and Y-90, a suitable surrogate pair is sought for imaging. Although hard beta emitters, such as Y-90, could be followed using Bremsstrahlung radiations, the image quality is poor—leading to inaccurate dose calculations.

Nearly all the current therapeutic RNs have a suitable matched pair RN that can be used to label the same biomolecule and obtain a diagnostic image (see Table 4-Chapter 3A, (Venkatesh and Kang, 2021)).

Table 3 Monoclonal antibodies approved for use in clinics to treat patients.

Mab name	Brand name	Indication for treatment	Year approved
Rituximab	MabThera Rituxan	Non-Hodgkin’s lymphoma (NHL)	1997
Trastuzumab	Herceptin	Breast cancer	1998
Alemtuzumab	Campath Lemtrada	Chronic myeloid leukemia	2001
Cetuximab	Erbix	Colorectal cancer	2004
Bevacizumab	Avastin	Colorectal cancer	2004
Panitumumab	Vectibix	Colorectal cancer	2006
Ipilimumab	Yervoy	Metastatic melanoma	2011
Pertuzumab	Perjeta	Breast cancer	2012
Ramucirumab	Cyramza	Gastric cancer	2014
Nivolumab	Opdivo	Melanoma, NSCLC	2014
Pembrolizumab	Keytruda	Melanoma	2014

Conclusion

The applications of radiopharmaceuticals in nuclear medicine are vast, growing and constantly evolving to offer new products and new strategies to help the physicians from multiple specialties in the management of patients and monitoring the treatment. The United Nations Scientific Committee on the Effects of Atomic Radiation 2008 Report (UNSCEAR, 2008) estimated that 32.7 million global diagnostic nuclear medicine examinations are performed annually. The role of nuclear medicine is unique and often the nuclear medicine studies provide unequivocal contributions towards patient care.

See Also: Fundamentals of Health Physics; Health Effects and Radiation Biology; Radiological Control Practices.

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Medicine: New Drug Developments

Meera Venkatesh^{a,*} and Aruna Korde^b, ^a Division of Physical and Chemical Sciences, International Atomic Energy Agency, Vienna, Austria; and ^b Division of Physical and Chemical Sciences, International Atomic Energy Agency, Vienna, Austria

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Glossary

ADME The absorption, distribution, metabolism, and excretion—here used in the context of a drug and its behavior in the body.

Assay A method to measure a specific chemical/biochemical/hormone.

Inhibition In the biological assays and nuclear medicine, inhibition refers to the prevention of binding of the radioactive molecule with the receptor or to the tissue of interest.

Lead molecule A chemical entity that has the potential to be developed as a drug.

MAB Monoclonal antibody—antibodies that are derived from the same “clone” or a single antibody producing cell that has been multiplied, having identical characteristics and behavior.

MARG Microautoradiography—micro auto radio graphy—a technique wherein a frozen finely sliced (thickness in microns) section of a tissue is imaged to get the information on the distribution of the radionuclide present in it. This technique can provide information at cellular level and used to understand the migration and distribution of the radiolabeled molecule in animal/plant studies.

PET Positron emission tomography—An imaging technique in nuclear medicine wherein a positron emitter is used as the radiolabel.

QWBA Quantitative whole-body autoradiography—a technique in which the distribution of radioactivity (radiolabeled molecule) in the whole body (various tissues and organs) is imaged and quantitatively determined. Generally, QWBA studies are carried out on animal models (rats or mice, etc.) and the data obtained is very useful in the ADME studies in humans. QWBA studies are common in new drug development, to assess the drug distribution in various tissues and organs, which forms a part of comprehensive ADME studies.

Radioisotope Unstable isotope of an element that emits radiation.

Radiolabeling Attaching a radionuclide to a molecule. Generally radiolabeling is achieved through synthetic chemical methods.

Radiopharmaceutical A drug formulation containing radioactivity, used in nuclear medicine for diagnostic imaging or therapy.

Radioreceptor assay An assay that uses radiolabeled molecules that bind to specific receptors.

Receptor Molecular structures within or on the surfaces of cells in biological systems to which specific molecules bind and cause biological responses. For example, in the brain, the dopamine molecules bind to specific proteins on nerve cells in the central nervous system and induce neuro transmission actions.

RIA Radioimmuno assay—a competitive binding assay wherein a radiolabeled ligand is used as the tracer and a specific antibody of the ligand is used as the binding molecule (see [Venkatesh, 2020](#) for more details).

RN Radionuclide; an isotope of an element that is unstable and emits radiation (such as alpha, beta or gamma rays) to transmute into a different nuclide which is more stable.

SPA Scintillation proximity (radioligand binding) assays.

*Retired.

SPECT Single photon emission computed tomography—An imaging technique in nuclear medicine wherein a gamma ray emitter is used as the radiolabel.

Target organ/tissue The organ or tissue that is the focus of study.

Introduction

Drug development is a field of great interest. The efforts to combat diseases with better efficiency is a never-ending exercise owing both to the onslaught of new diseases as well as better understanding of the disease process. All of this inspires the strategy to combat the disease in the best possible way. Drug discovery and development is a multi-step process, and generally it is at least a decade (most of the times much longer) from the time of conception to the time a drug reaches the clinics. The various steps in new drug development and proving its clinical utility through multi-stage clinical trials are not only time consuming, but also very expensive, often leading to potential promising molecules (the key components of importance in a new drug) being abandoned half-way.

The various steps in a new drug development are

- Concept/design of a lead molecule, based on the understanding of the biochemistry of the disease/organs/mechanism;
- Synthesis and testing of a set of lead molecules (often could be hundreds of molecules with similar traits, but a few to be chosen based on actual efficacy);
- Studies on the biological half-life in the body—to ensure desirable retention period and adequate clearance from the body with minimal unwanted effects and maximal efficacy in treatment;
- Testing the hypothesis of drug efficiency—through pharmacological effects and biological distribution and metabolism of the drug inside the body, including localization and retention in target organ/tissue;
- Testing the absence of undesirable side effects—through none or minimal localization in non-target organs; and
- Clinical trials in multiple stages to prove the efficacy in actual clinical settings. This often starts with a small group of volunteers/patients and subsequently expanded to prove the usefulness of the drug beyond doubt in a larger disease population (**Fig. 1**).

The progress of a drug from laboratory to preclinical and clinical stages is an intensive process and it is essential to collect statistically valid data for analysis and interpret the results at each step for planning further experiments. New drug development is a multidisciplinary activity and pharmaceutical industries adopt suitable scientific and technical developments in related fields to increase the efficiency, reduce the cost, and to increase the percentage of the success rate. Taking advantage of the fact that radiotracers can be detected and measured in extremely low amounts (high sensitivity) and can be precisely located, quantified, and imaged, they have been used in new drug development over many decades (Lindner and Link, 2018).

Strategies in the use of radiolabeled molecules

Radiolabeled molecules can be used in each of stage of drug development (Uhl et al., 2015), albeit with different strategies/principles. The lead molecules, when labeled with a genuine radiolabel, would be chemically identical to the unlabeled counterparts in their biological action and, hence, can provide valuable information on the biological activities, physiological clearances, target/nontarget accumulation, etc. An example would be replacing the stable carbon atom, C-12, in the molecule with the radioactive C-14 or C-11 atom. In technical terms, these radioactive labeled compounds are used in nonclinical studies to assess drug absorption, distribution, metabolism, and excretion (ADME) (Marathe et al., 2004).

For example, Serotonin is an important neurotransmitter and optimum serotonin levels are essential for mental wellbeing. Molecules that can bind to a specific serotonin receptor (subtype 2C) in the brain hold promise for use as drugs to treat several disorders of the human central nervous system. These molecules were radiolabeled with carbon-11, and positron emission tomography (PET) imaging studies were carried out to study the kinetics of the proposed drug molecules in the body (Neelamegam et al., 2014).

Such information helps in the selection of promising molecules for taking to the next step. At an advanced stage of development, genuinely labeled drug molecules provide unequivocal proof of the drug action and localization in the desired target regions. This

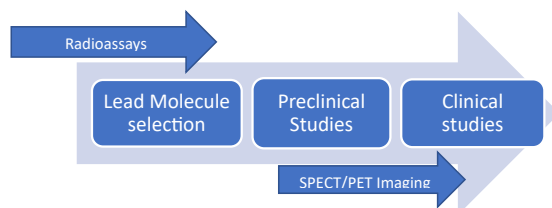


Fig. 1 Radiolabeled molecules and radiopharmaceuticals in drug development.

strategy is being increasingly used in drug discovery/development to understand the biological behavior of the molecule. This strategy is predominantly research oriented and calls for dedicated radiolabeling and testing facilities.

Another approach is the use of an established diagnostic radiopharmaceutical that is used to get information on the status of health of an organ or a system. If the efficacy of a new drug to treat the diseased condition is to be tested, diagnostic scans using the established radiopharmaceuticals before and after treatment can provide invaluable information. For example, ^{18}F -Fluoro deoxy glucose (^{18}F -FDG), is an analog of glucose, which is extensively used in oncology to image cancerous lesions in nearly all types of cancers. A promising new drug for treatment of a cancer can be tested for its efficacy by imaging the patient using ^{18}F -FDG at various time periods after administering the drug under trial. In this case, the radiolabeled molecule used is a well-established commercially available radiopharmaceutical, but the studies are research oriented. In this second approach, an established radiopharmaceutical that may be obtained commercially is used for checking the action/efficacy of the drug candidate either in preclinical or early phases of clinical trial.

In both approaches, these strategies help in screening a large number of molecules, ascertain biological in-vivo behavior (especially distribution and excretion pattern), and in certain cases, check the efficacy of a new drug. All these together reduce the time for new drug development.

Radiotracer-based assays

Yet another use of radiolabeled molecules in drug research is the use of radiotracer-based assays (Enna, 1984) such as radioligand receptor assays (or radioreceptor assays) and radio immuno assays (RIAs). While the use of RIAs is limited, radioreceptor assays have long been used as a part of new drug discovery process. With the advances in understanding the actions of hormones and drugs, for most drugs and hormones there exists a receptor site on membranes to which the substance must attach in order to evoke a biological response. This has been extremely useful in drug research to test if the molecule under study binds to the specific receptor, which is the first step for obtaining the desired drug effect.

Radioreceptor assays, based on the principle of competitive protein-binding methods, (similar to the radioimmunoassay described briefly in Venkatesh (2020)) were developed in this context. Radioreceptor assays are highly sensitive, reliable, and are simple and rapid to perform (Houghten, 1993). In a nutshell, in radioreceptor assay, a small fixed amount of the receptor protein on the membrane/cell to which the drug is designed/desired to bind is reacted with a fixed amount of the radiolabeled ligand that specifically binds to this receptor. Iodine-125 is the most commonly used radioisotope in radioreceptor assays and tritium (H-3) has also been used in some cases. At the end of the reaction, the receptor bound ligand and free ligands are separated using a suitable technique. If the reaction mixture contains another molecule that can bind to the same receptor, this would compete with the radio-labeled ligand to bind to the receptor. As the receptors are fixed and limited in number, such a competition would lead to decrease in the amount of bound radioligand and this decrease would be proportional to the amount of the competing molecule. When the molecule to be tested for its drug action is added in such a competitive reaction, the affinity of the molecule can be gauged from the extent of decrease in radioligand's binding. A receptor protein and a radioligand with high affinity for this receptor are the main ingredients for radioreceptor assays and the key to developing a successful receptor ligand assay is identifying the appropriate reagents, the reaction parameters, and the separation methods.

Radioligand receptor reactions are usually equilibrium studies, and the two commonly used designs for these assays are saturation assays and competition assays. These assays are sensitive (due to the high sensitivity with which the radioactive labels can be measured) and highly specific (due to the highly specific nature of receptor-ligand interaction). Different types of receptor preparations, such as proteins isolated from cell fractions or the whole cell or even tissue sections, could be used in these assays. With advanced developments in cellular and molecular biology, it has become possible to synthesize the receptor protein in the desired quality, which reduces the variations and increases the accuracy of the assay results.

Scintillation proximity radioligand binding assays (SPA) is a variant of radioligand receptor assay that does not need the separation of receptor bound and free fractions of radioligand. In SPA, the radioligand is labeled with a low energy beta emitter such as carbon-14 or tritium (H-3) or Auger electron emitter such as iodine-125 and the receptor is immobilized on a solid matrix which also encompasses a scintillant, that emits a light signal when it is excited by a beta particle or an Auger electron. In SPA, the light signal is emitted only when the radioligand is bound to the receptor and this is known as homogenous format, which allows monitoring association and dissociation of the radioligand without disturbing the reaction equilibrium. The progresses in biological and chemical sciences have resulted in conceptualization of a large number of target molecules and possibilities of synthesizing limitless drugs candidates to choose from. This has revolutionized drug screening methodology and achieved high throughput. SPA based radioreceptor assays, along with automation in radioactive counting systems, are being used widely in drug research (Cook, 1992, 1996) for determining drug affinity, drug interactions, receptor subtypes, and receptor concentrations. Radioreceptor assays have been applied to quantitative determination of hormones, neurotransmitters, and drugs. The radioreceptor assay of benzodiazepines is a typical example.

Radiolabeling of molecules and molecular imaging

In the early days, drug molecules were commonly labeled with long lived radioisotopes such as H-3 or C-14, which are pure beta emitters that emit low level beta radiation, to study drug localization in animal experiments. The radioisotopes were "visualized" by a simple autoradiography, which is a technique that uses X-ray film or phosphor imaging plates or a similar imaging system to study

localization of radiolabeled drug or drug fragments in organ or tissue sections of the animals (Solon, 2015). Autoradiography has been used for decades. With time, this technique has evolved into more sensitive and sophisticated phosphor imaging based quantitative whole-body autoradiography or autoradioluminography (QWBA) and microautoradiography (MARG) techniques, which have much higher resolution. QWBA uses phosphor imaging technology providing high resolution images of the spatial distribution and matching tissue concentrations of drug-related radioactivity throughout the body of laboratory animals. This provides tissue-specific pharmacokinetic (PK) compartmental analysis. MARG is another autoradiographic technique that qualitatively resolves the localization of radiolabeled compounds to the cellular level in a histological preparation. These techniques can provide spatial and temporal distributions in studies with tissue sections and are now used to quantify radioligands at cellular levels, which aids in identification and validation of drug targets.

Molecular imaging is playing an important role in drug development, especially after lead optimization when a drug candidate molecule enters the preclinical study stage. Nuclear medicine imaging, based on molecular interaction of the radiopharmaceuticals, is a truly “functional molecular imaging” and has far better penetration and resolution in comparison with other molecular imaging techniques. The current hybrid imaging systems, such as SPECT-CT or PET-CT, provide anatomical and physiological information, which aids immensely in assessing the localization of the drug molecules studied (Kristensen and Herth, 2017). The current availability of a wide range of radioisotopes and sophisticated, radiolabeling techniques makes it feasible to radiolabel the drug candidates with retention of its biological action and ensure its stability. The radiolabeled drug is generally prepared in very high specific activity and, hence, the actual drug concentration used for imaging studies is extremely low, usually in nano or pico molar concentrations. Hence, it is possible to obtain quantifiable information about the fate of the drug in the body, such as its distribution and retention in different tissues and excretion and metabolism, which helps in the selection of right candidates for further clinical studies.

Radiolabeling the drug molecules with a genuine label calls for labeling the radionuclide (RN) of one of the elements in the composition of the drug. As most of the drugs (biologically active molecules) are made up of elements carbon, nitrogen, oxygen (and occasionally phosphorus, sulfur or iodine), RNs of these elements need to be used in these studies. Further, radiolabeling a drug molecule will need to be carried out synthetically. As the radiolabeled drug is desired to behave exactly in the same manner as its unlabeled counterpart when administered into the body, it is important to retain the structural identity to the maximum extent possible. In the past, molecules have been labeled through “exchange route,” wherein the drug molecule is exposed to a radioactive reactive molecule (especially for labeling with ^3H or radioiodine). But this method has limitations and does not work for all molecules and all radiolabels. Most often, the radionuclide is introduced through a synthetic route, almost like synthesizing the drug.

Unfortunately, there is no RN that is ideally suited for drug related studies for the naturally occurring elements in biomolecules. The isotopes of carbon-11 ($T_{1/2} = 20.4$ min), nitrogen-13 ($T_{1/2} = 10$ min) and oxygen-15 ($T_{1/2} = 2$ min), are very short lived. Phosphorus and sulfur do not have suitable RNs, whereas Iodine isotopes I-123 and I-124 are quite amenable for such studies. However, not all drug molecules will have phosphorus or sulfur or iodine as one of the elements. Hence, C-11, a positron emitter used for PET studies, is the one used most widely in drug metabolism studies. Labeling with C-11 is desirable since the product is chemically identical with the unlabeled drug molecule (Boscutti et al., 2017; Miller et al., 2008). However, it could be challenging if the synthetic scheme is complex and with low yields. The short half-life of C-11 poses practical difficulties for synthesis and limits the studies to short spans. To incorporate C-11 into the drug molecule, a C-11 labeled reactive intermediate (e.g., C-11 methyl iodide, C-11 triflate, C-11 urea, C-11 carbamates, etc.) is used in an appropriate step of the synthetic route of the drug.

Drugs have been labeled with fluorine-18 ($T_{1/2} = 110$ min) as fluorine mimics the hydroxyl group and the radio-fluorinated drugs are studied as analogues. Since F-18 is the most widely used PET radionuclide, it is common to synthesize a series of radio-fluorinated compounds based on the pharmacophore of the drug, and evaluate them, in order to develop a potential diagnostic radiopharmaceutical.

Such radiolabeling demands good synthetic skills and a good synthetic laboratory with remote handling facilities. Besides the low yields, there are several additional challenges to ascertain their identity with the candidate molecule. These radiolabeled drugs are also required to fulfill the criteria of purity (often in the 95–98% range) and safety before they can be used in animals and humans for any studies.

New drug development and use of radiotracers

A large number of drugs have been studied for their efficacy using molecules labeled with F-18 and other PET radionuclides (Donnelly David, 2017; Matthews et al., 2012; Miyoshi et al., 2011).

As mentioned earlier, the drug molecule may be radiolabeled and administered in the body to study its behavior, or a well-studied radiopharmaceutical may be used to monitor the efficacy of treatment. For example, the drug Docetaxel was labeled with carbon-11 to study its localization and clinical action through PET imaging (van der Veldt et al., 2013).

There are several radiotracers molecules that are very well characterized for their binding affinity with certain targets. Such radiotracers, when produced as per regulatory requirements of safety and quality, can receive approval from regulatory authorities for use in clinical studies as exploratory new drugs/radiopharmaceuticals. Many such radiotracers are used for clinical trials of new drug development. Development of drugs for central nervous system (CNS) disorders, cancers, and other diseases has immensely benefitted from the use of such radiotracers. Some typical examples are given below.

Dopamine, synthesized by neurons in the brain, plays a vital role in various biochemical pathways. Association of dopamine with neuropsychiatry and movement disorders such as Parkinson's disease is now well recognized. The synthesis, storage, release and metabolism of dopamine is tightly controlled through various enzymatic processes and is also affected by concentration of

specific dopamine receptors. Currently, different radiolabeled molecules specific for these processes are identified. Among these, L-6-[^{18}F]fluoro-3,4-dihydroxyphenylalanine (also referred to as ^{18}F FDOPA) is used to study neuro degeneration of dopaminergic neurons associated with Parkinson's disease. ^{18}F FDOPA is approved by US FDA and is available from commercial producer for clinical use. Two other tracers, namely, ^{18}F Fallypride and ^{11}C Raclopride bind to specific dopamine receptor sub types (D2/D3) and provide information about the receptor concentrations. These established radiotracers are used to study the efficacy of new drugs or new therapies for Parkinson's disease and for comparative evaluation of the receptor binding capacity of potential drug molecules.

Serotonin is another important neurotransmitter and optimum serotonin levels are essential for mental wellbeing. SERT (serotonin transporter) plays vital role in serotonin system functioning. Many of the drugs for treating mental depression and anxiety are designed to act through SERT. The radiolabeled molecule ^{11}C -3-amino-4-(2-dimethylaminomethyl-phenylsulfanyl)-benzonitrile (^{11}C -DASB) binds to SERT with high affinity. ^{11}C -DASB is used to obtain information on the functioning of serotonin system in various neuropsychiatry and movement disorders, including the evaluation of new drugs to treat them.

Oncology is among the most benefitted fields by diagnostic nuclear medicine imaging. The use of ^{18}F -FDG, the most used PET tracer, in development of drugs for cancer was mentioned earlier. The emerging knowledge on cancer cell biology has shown that certain types of cancers overexpress specific types of proteins and this has paved way for development of new drugs that target these proteins. Radiotracers that can bind to such specific proteins on the cancer cells have been used for evaluation of these new drugs.

While PET imaging with ^{18}F -FDG is very useful in many cancers, it is not an option for certain cancers such as cancers of brain, prostate, etc. where the ^{18}F -FDG metabolism is high even in non-cancerous cells. In the case of brain, glucose is the main source of energy and, hence, ^{18}F -FDG accumulation in a normal brain is very high. Therefore, the uptake in a brain tumor does not show up in the image. This is overcome by the use of another tracer, 3'-deoxy-3'-[^{18}F]fluorothymidine (^{18}F -FLT), which is an analog of nucleotide thymidine and taken up by growing cells. However, ^{18}F -FLT gets trapped in the cell during cell division process, and rapidly growing brain tumors and other tumors accumulate ^{18}F -FLT and enable PET imaging.

Imaging prostate cancer lesions using ^{18}F -FDG is also difficult, as ^{18}F -FDG is normally excreted through renal route, which is close to the prostate gland location. Based on the knowledge that an enzyme known as choline kinase is elevated on prostate cancer cells, ^{11}C Choline and other analogues of choline labeled with ^{18}F (which act as a substrate for choline kinase enzyme) are used for imaging and early detection of prostate cancers. These tracers are also used to test the efficacy of new drugs against the cancers. Along similar lines, prostate specific membrane antigen (PSMA), which is found to be over expressed in prostate cancer cells, has been a successful target for treatment. Various molecules that act as inhibitor of PSMA are being identified, radiolabeled with diagnostic and therapeutic radioisotopes, and are being evaluated in clinical trials.

Table 1 below lists the examples of such radiotracers that have been used in drug research/development for treatment of some of the common diseases. Some of these are established radiopharmaceutical products, which can be obtained from authorized producers for clinical studies.

Radiolabeled antibody therapy

In recent times, there has been a paradigm shift in new drug development. Protein-based biologics, such as monoclonal antibodies (MAB), are increasingly being evaluated for therapy of various diseases. Antibodies have the inherent specificity to bind to the specific molecular targets against which they are produced. MABs that originate from a single source of antibody producing B-lymphocyte cell have well defined and controlled characteristics.

The hybridoma technology, invented in 1975 by Köhler and Milstein (see also [Venkatesh, 2020](#)), enables the creation of immortalized antibody secreting lymphocytes selection of single clones that secrete antibodies with desired binding characteristics that propagate the clones in-vitro. They produce the monoclonal antibodies on large scale, making them far superior to the conventional polyclonal antibodies for application as a drug. Polyclonal antibodies are derived/produced by different B cell lymphocytes and have different binding characters/affinities. This makes it difficult to predict the therapeutic efficiency when used. The hybridoma technology was quickly adopted to produce MABs specific to many target antigens in huge quantities. Currently, other molecular biology-based technologies, such as phage display methods, are also being adopted for large scale production of MABs. With the possibility of production of MABs specific to a large number of target antigens in huge quantities, immunotherapy has evolved as a very promising modality of treatment for several cancers and certain other diseases over the past two decades (see [Venkatesh, 2020](#)). For example, a MAB against head and neck tumors is used as one of the treatment options.

Table 1 Examples of radiotracers used in development of new drugs for some common diseases.

Target receptor/system for drug action	Radiotracer	Disorder/disease
Dopamine D2/3 receptor	^{11}C Raclopride	Neurology, psychiatry
Dopamine transporter	^{18}F Fallypride	Neurology, psychiatry
Serotonin receptors	^{11}C DASB(3-amino-4-(2-dimethylaminomethyl-phenylsulfanyl)-Benzonitrile)	Neurology, psychiatry
Glucose transporter	^{18}F Fluorodeoxyglucose (FDG)	Cancer, cardiology, neurology
Thymidine kinase	^{18}F Fluoro L-Thyroxine (FLT)	Cancer
Choline kinase	$^{11}\text{C}/^{18}\text{F}$ Choline	Prostate cancer
Prostate-specific membrane antigen (PSMA)	$^{18}\text{F}/^{68}\text{Ga}$ -PSMA inhibitors (PSMA11 or PSMA617)	Prostate cancer

In parallel, MABs were explored for use in nuclear medicine for specific targeting—especially cancers and radiolabeled monoclonal antibodies—as radiopharmaceuticals has grown as a specialty field. The increasing use of monoclonal antibodies in treatment of various cancers has often been supported through data obtained on using radiolabeled MABs to prove the targeting action and localization in the tumor tissues. In such cases, where a very large molecule (such as an antibody) is labeled, a foreign label such as radioiodine or Tc-99m has been used since the biological activity is not altered on radiolabeling if it is distant from the active site.

Radiolabeled monoclonal antibodies have in recent years evolved for use as therapeutic radiopharmaceuticals. Several radiometal based radioisotopes (such as Ga-68, Lu-177, Y-90) are being explored for radiolabeling antibodies, both for diagnostic and for therapeutic use (theragnostic), in turn leading to developments for producing these radioisotopes in adequate quantities and quality. The radiochemical strategies for labeling and related preclinical fields have also witnessed growth and various radiolabeled monoclonal antibodies are currently in clinical trials (Vugts Dinielle and van Dongen, 2019; Venkatesh and Kang, 2021). These studies are extremely useful for development of these drugs.

Rituximab, a MAB that binds to an antigen that is specifically expressed by B-cell non-Hodgkin's lymphomas, was the first MAB based drug approved by US FDA. B-cell non-Hodgkin's lymphomas (a type of blood cancer), being inherently radiosensitive, benefited from a radiolabeled rituximab—proving to be clinically significant as it delivered the therapeutic effect of ionizing radiation at the site of cancer tissues. Currently two radiolabeled MABs, Ibritumomab (labeled with Y-90) and Tositumomab (labeled with I-131) are approved as radiopharmaceuticals, both for treatment of B-cell lymphomas. Although the MABs by themselves are used for therapy, the addition of a radionuclide is like a payload that increases the therapeutic effect multifold. Along similar lines, Trastuzumab, which targets specific receptor protein expressed on certain types of breast cancers, is approved as a chemotherapeutic drug. Radiolabeled Trastuzumab is being used for in clinical trials for diagnosis and treatment of these types of breast cancer, with promising results.

Conclusion

The use of established radiopharmaceuticals in drug development is now quite common and often practiced in testing new drugs. With the advances in understanding of diseases at the molecular level, new strategies are evolving in treatment and testing their efficacy through a molecular imaging technique such as nuclear medicine imaging.

In conclusion, the role of radiolabeled molecules is very valuable in providing unequivocal proof of drug action and efficacy and, hence, increasingly used in drug discovery and drug development.

See Also: Fundamentals of Health Physics; Health Effects and Radiation Biology; Radiological Control Practices.

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Medicine: Therapeutic and Other Applications Using Sealed Radiation Sources

Meera Venkatesh^{a,*} and Shyam Shrivastava^b, ^a Division of Physical and Chemical Sciences, International Atomic Energy Agency, Vienna, Austria; and ^b Radiation Oncology, Apollo Hospitals, Navi Mumbai, India

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Glossary

3D-conformal radiation therapy (3D-CRT) Three-dimensional conformal radiotherapy is aimed at delivering a conformal dose distribution to tumors through the use of patient specific 3D images in the treatment planning process. Typically CT and/or MRI images are used in 3D-CRT.

Beta particles Beta (β) rays or beta particles are a type of particulate radiation with the mass of an electron emitted by certain radioactive nuclides. There are two kinds of beta rays. β^- rays have a charge of -1 and are identical with electrons. They are also known as negatrons. β^+ rays have a charge of $+1$ and are identical with positrons, which is the anti-matter of electron.

Boron neutron capture therapy (BNCT) A mode of therapy wherein boron-10 containing molecules are administered to localize in the tissues to be treated, after which an external beam of neutrons is focused on the tissue target to cause a B-10 reaction to emit alpha and Li-7 charged particles, which render the therapeutic effect.

Brachytherapy A therapy technique accomplished by placing a source of radiation inside the body, close to the diseased tissues, generally a cancer. (Brachy in Greek means "in contact").

Computerized tomography (CT) A mode of obtaining a three dimensional tomographic image using X rays.

Endovascular radiation therapy (EVRT) or endovascular brachytherapy (EBT) EVRT/EBT is a brachytherapy treatment modality wherein a radioactive source is placed inside the blood vessel for treatment. EVRT/EBT was used to prevent restenosis of coronary arteries of the heart and blood vessels of the extremities after the original stenosis, or blockage, was opened by angioplasty.

External beam radio therapy (EBRT) Use of an external source of radiation for treatment of diseases, generally cancers. Also known as "teletherapy".

Gamma knife A device in which several (typically 200) gamma radiation sources are placed in a spherical geometry for use in stereotactic radiosurgery of cancerous lesions in sensitive organs such as brain.

Gamma rays They have no charge or mass and are emitted from the nucleus of an atom when the atom undergoes transmutation. Emission of gamma rays does not change the atomic number or the mass number of the nuclide. The change is only in the energy levels within the nucleus.

Hadron therapy Use of charged particles such as protons or carbon ions in radiation therapy.

HDR High dose rate of radiation.

IGRT Image guided radiation therapy (IGRT) is a modality in radiation therapy wherein imaging techniques are used during each treatment session.

IMRT Intensity modulated radiotherapy (IMRT) is the mode of radiation therapy in which the intensity of radiation is varied according to the tumor volume so that the tumor tissues receive high doses while the surrounding tissues receive as low a radiation dose as possible.

KV Abbreviation of kilo electron Volt ($\text{keV} = 10^3 \text{ eV} = 10^{-3} \text{ MeV}$) to denote the energy of the X rays/photons derived from an X Ray machine used in radiotherapy.

*Retired.

LINAC Linear accelerator: An electron accelerator used in radiotherapy that can be used in two modes, to obtain either electron beams or photons.

Multi-leaf collimator (MLC) A hardware device having several movable leaf-like structures, usually made of an element with high atomic number such as tungsten, that can be used to shape the beam of radiation from a radiotherapy machine to match the size and shape of the tumor to be irradiated. The MLC is applied to target the beams at only the cancer cells in order to spare surrounding normal tissue. They are particularly useful when treating patients using IMRT (intensity modulated radiation therapy).

PET, positron emission tomography A tracer technology based on using positrons to isolate a tomographic image.

Proton therapy Use of protons of high energy from an accelerator as a focused beam for teletherapy.

Radiation dose The average energy deposited in tissue by ionizing radiation. Also known as the absorbed radiation dose. The SI unit of absorbed dose is Joules per kilogram of tissue (J kg^{-1}) and is known as "Gray" (Gy).

Radiation dosimetry The special field of calculating a radiation dose delivered to the body, including both the target tissues and the other areas when a source of radiation is used in a medical procedure such as radiotherapy or nuclear medicine.

Radiation sterilization The process of sterilization using nuclear radiation. Generally, gamma radiation from cobalt-60 source is used for sterilization.

Radionuclide An isotope of an element, the nucleus of which is unstable and transmutes (transforms; decays) into a more stable nucleus by emission of one or more kinds of radiations (α , β^- , β^+ , γ) or by electron capture. Sealed Radioactive sources: A hermetically sealed radionuclide source

Stereotactic radiosurgery (SRS) and stereotactic body radiotherapy (SBRT) Stereotactic radiotherapy is a non-surgical non-invasive procedure wherein multiple photon beams from different directions are focused to deliver a highly-concentrated dose of radiation to a precise target in a single session. 3D images of high quality and precision are used to program the irradiation machine for precise dose delivery. SRS is the term used for stereotactic radiotherapy of brain and SBRT when other parts of body are treated.

Teletherapy Use of an external source of radiation for treatment of diseases, generally cancers (the term "Tele"-is used to indicate "from a distance").

Introduction

Radioisotopes have been put to numerous applications owing to their unique properties, and medical applications are the most widely known. Radiation and radioisotopes have been used in the healthcare sector from almost the time when artificial radioactive isotopes were available. Over the past eight decades, the scope and varieties of uses of radiation and radioisotopes in healthcare have grown phenomenally and different radioisotopes/types of radiation are used for different applications. Medical applications of radioisotopes can be categorized based on their use, such as for therapy or diagnosis or for obtaining special materials for use in healthcare.

Radiation therapy using ionizing radiation to treat variety of malignant and benign conditions has come a long way since the discovery of X-rays by Wilhelm Conrad Roentgen in 1895 (Connell and Hellman, 2009), the discovery of the phenomenon of radioactivity by Henri Becquerel soon after in 1896 (Sekiya and Yamasaki, 2015), and the existence of radium and polonium in the pitchblende by Curie et al. (1898). Soon after the discovery of X-rays, they were explored for use in treating cancer. Treatment of cancer using radiation is among the most gratifying uses of radiation.

In the early period, low-energy cathode ray tube and radium filled glass tubes placed closed to the tumor, were practiced for giving radiation in a single dose. These initial phases of treatment with X-rays were associated skin toxicities, owing to the poor penetration of X-rays and limitations to treat deep seated tumors. The technological advances prior to 1945 focused on improving the quality and energy of the radiation beams. With the development of megavoltage radiation machines such as telecobalt (^{60}Co ; teletherapy with cobalt-60) and linear accelerators, it was possible to treat deeper seated tumors in the late 1950s. With the understanding of the effects of radiation, dividing radiation dose in multiple fractions was found to improve the tolerance of normal tissues towards radiation. Consequently, since then fractionated radiotherapy has been the common practice.

The approach taken in cancer treatment is either curative or palliative, depending on the extent of the disease and the health status of the patient. It involves one or a combination of the different modalities of surgery, radiation therapy and chemotherapy. The optimal treatment is tailored according to the type and stage of cancer, histopathology of tumor, anatomical location of tumor, and patient factors. In modern practice, generally the treatment is multimodality aimed at balancing the tumor control rate and the toxicities.

Radiation treatment is offered in two different formats: (1) external beam radiation therapy (EBRT) or *Teletherapy* where a sealed radiation source is at placed a distance from the tumor and the radiation is shone upon it, or (2) by placing the source in close contact with the lesion, also known as *Brachytherapy*.

The practice of teletherapy has undergone a sea-change over the past several decades, owing to the innovations/inventions in varied areas such as automation, computation, image processing/modelling, materials, and mechanization as well as advancements

in imaging and cancer biology. Currently, radiation oncology is an ideal non-invasive treatment and achieves maximal tumor control with minimum normal tissue damages.

The field of brachytherapy too has progressed, and sealed radiation sources are used either interstitially within the tumor or kept in the cavity close to the tumor or covering the tumor like a mold. Brachytherapy is continued to be explored as a treatment mode for well localized cancers and a few non-cancerous skin ailments with success in some cases.

Boron neutron capture therapy (BNCT) was conceptualized almost seven decades ago, but it has remained an option followed by very few clinics in few countries. In BNCT, therapy is achieved through alpha particles and lithium-7 ions being produced in situ from B-10 capture of neutrons. While the concept is very attractive, the practical challenges of targeting the cancerous lesion with a large number of B-10 atoms and shining a neutron beam on the lesion have been the major impediments. The research and development in these areas over the past years have yielded impressive results in certain types of cancers and BNCT is currently seen as a promising therapy mode.

Radioisotopes as a source of radiation for sterilization of gadgets/apparatus used in healthcare is termed as “*Radiation Medical Sterilization*” and is employed worldwide widely to sterilize most of the disposables such as syringes, needles, gauze, etc. as well as prosthetics (Kalia, 2021).

Based on similar principles, radiation, although at much lower dose rate, is used to treat the blood to be transfused, to prevent undesirable reactions. “*Blood irradiation*” is mandatory for transfusion in some countries, while in others radiation treated blood is used especially in immune compromised patients.

Another application of radiation is in the production of unique materials for use in healthcare. “*Radiation polymerization*” is well established in industrial applications for production of polymers for use in several areas (Fernando, 2021). In healthcare, radiation polymerized hydrogels are used in the treatment of burns and open wounds. In recent years, radiation polymerized nanoparticles are being explored for use in radiopharmaceuticals, drug delivery systems, and as scaffolds in tissue engineering.

External beam radiotherapy (EBRT) or teletherapy

The majority of radiation treatment is through external beam approach (EBRT or teletherapy), where the radiation is focused on the tumor. Such therapy started with the use of X-rays soon after X-ray tubes were available as a source of radiation, and X-rays of 200–400 KV energy were used in the early times. The availability of cobalt-60 teletherapy machines ushered in the next period of their extensive use followed by the linear accelerators for treatment with photons or electrons. Figs. 1 and 2 show the pictures of a linear accelerator and a patient prepared for radiotherapy treatment with custom made shields being used to protect the healthy regions organs in the face from being subject to irradiation.

The radiation source used in EBRT has further expanded to include a variety of radiation such as protons, carbon ions, neutrons etc. The effect of radiation depends on several factors such as inherent tumor sensitivity, susceptibility of adjacent normal tissue to radiation injury, quality of radiation, etc.

In order to precisely deliver the radiation only to the tumor, which in most cases is not of regular shape, multiple beams of radiation (or a beam of radiation rotated at different angles) are focused on the target from different angles. Further, if the tumor tissue is located in areas such as the chest (which moves while breathing), it is very important not to miss the target area and to ensure precise delivery of radiation. Advances in imaging techniques, computerized treatment planning and reproducible patient positioning systems have enabled to achieve accuracy in delivering the intended treatment. Currently EBRT can deliver the required radiation dose to the tumor volume very precisely.



Fig. 1 Linear accelerator for radiotherapy



Fig. 2 Patient prepared for radiotherapy.

Planning is extremely important in EBRT, to deliver the exact required dose to the tumor volume while protecting the surrounding healthy tissues to the maximum extent possible. Radiation treatment planning is generally performed using dedicated software, which derives input from an image of the tumor to be treated. The software makes precise calculations for passing multiple beams through the target volume (tumor) while protecting the surrounding normal tissues. Computed Tomography images are the most widely used in treatment planning. In addition, if MR (magnetic resonance imaging) scans and PET-CT scans of the patient are available, powerful software is now available to enable fusion of the images for more accurate and precise target delineation and dosimetry.

Several techniques have evolved from the conventional radiotherapy, aiming at maximizing the dose to the tumor with minimal effect on surrounding healthy tissues, while constantly accounting for involuntary body movements. The following are such advanced techniques, all useful and important to treat tumors in sensitive areas such as brain, head and neck, eye, etc.

3D-Conformal radiation therapy

Three-dimensional conformal radiotherapy (3D-CRT) aims to deliver a conformal dose distribution to tumors, while sparing surrounding normal structures. Development of hardware like a multi-leaf collimator (MLC), along with treatment planning software, was revolutionary in advancing radiotherapy—making possible the delivery of 3-Dimensional conformal radiation therapy (3D-CRT). This entails irregularly shaped treatment volumes, reducing radiation dose to the normal surrounding organs by shielding part in irradiated region during radiation dose delivery compared to conventional techniques ([Chopra et al., 2012](#))—Patient specific 3D images, typically, CT and/or MRI images, are used for defining the areas of concern as well as adjacent normal structures. A plan is developed so that the regions of high dose are concentrated within the tumors. 3D-CRT is used to treat tumors that are close to vital organs and structures, such as head and neck tumors, in a way that minimizes exposure of the spinal cord, optic nerve, salivary glands and other important structures.

IMRT and IGRT (intensity modulated radiation therapy and image guided radiation therapy)

Further innovations in inverse-planning software with robust computers enabled the complex computation to be faster and more accurate, leading to the development of intensity modulated radiation therapy (IMRT). With the IMRT, the intensity of radiation beam is modulated to provide tighter approximation of high dose of radiation to the target volume. An iterative algorithm is utilized to reach the specific dose coverage to the target and to minimize the possible dose to the normal tissues. The output defines the shape and distribution of radiation from multiple beam segments. This provides the high degree of conformity to an irregularly shaped target (tumor) and reduces the dose to tissues in close proximity of tumors ([Bortfeld, 2006](#)).

CT based planning provides the ability to precisely target radiation in tumor area; however, variations due to daily set-up and organ motion necessitate the treatment of a larger volume than that of the actual tumors. The technological advancements in the software, as well as hardware, have enabled the medical technicians to overcome the issues of positional errors and physiological motion and allow closer margins for treatment. The current software and hardware packages account for the physiological motions like respiratory motions, changing position of lung tumors, liver tumors, etc. The modern linear accelerators are fitted with an on-board imaging device, which provides diagnostic quality KV X-rays or cone beam CT Radio-opaque “fiducial” markers, which are small dot like structures made of gold, plastic, or another material (placed in the body within or near the tumor to help locate the cancer). They are used at times prior to pre-treatment imaging for better visualization of soft-tissue ([Shirato et al., 2003](#)). When images are obtained prior to treatment, and used to reduce positioning errors, this precision in treatment is known as, image guided radiation therapy (IGRT).

Clinically the IMRT/IGRT have been found to be extremely useful in allowing dose escalation to the tumor and reducing the toxicities. IMRT has become standard in “organ-sparing treatment” for radiation therapy of head and neck cancers. It helps in protecting the salivary glands and reduces radiation related xerostomia, leading to an improvement in the quality of life (Lin et al., 2003; Nutting et al., 2011). IMRT also offers techniques like “dose painting” and “simultaneous integrated boost (SIB)” to individualize dose planning (de Arruda et al., 2006; Konski et al., 2009). However, there are relatively limited randomized trials (Donovan et al., 2007) at present and several studies are on-going. IMRT is routinely used in treatment of cancers of head and neck, prostate and breast.

Volumetric modulated arc therapy (VMAT) is a mode within IMRT, in which the irradiation machine rotates to deliver the radiation dose continuously. This technique accurately shapes the radiation dose to the tumor while minimizing the dose to the organs surrounding the tumor. This mode is very accurate, shortens the treatment time, and uses a lower overall dose of radiation.

TomoTherapy is yet another mode within IMRT, similar to VMAT, in which the radiotherapy couch where the patient lies is moved through a donut-shaped machine. The radiation source in the machine rotates in a spiral pattern around the patient. It is also sometimes called helical tomotherapy.

SRS/SRT, SBRT (stereotactic radiosurgery and stereotactic body radiation therapy)

The “Stereotaxy” implies the use of multiple beams in a precise coordinate system. Although the word “surgery” is used in SRT, it is a non-surgical radiotherapy procedure. This modality of radiotherapy was conceptualized nearly 50 years ago by neurosurgeons and radiation physicists to treat brain tumors (American Association of Neurological Surgeons) and has since then evolved as a sophisticated, 3-D-computerized imaging to precisely focus multiple photon beams from different directions to deliver a highly concentrated dose of radiation to a precise target in a single session. Stereotactic surgery is often the choice to treat brain tumors and head and neck cancers that are difficult to operate. A device known as “Gamma knife” contains about 200 radiation sources placed in a spherical contour (Fig. 3) and is specifically used to treat tumors in brain/intracranial area. An equivalent device using linear accelerator is known as “cyber knife”.

SRS/SRT, when used to treat cancer lesions at other parts of the body, is known as stereotactic body radiation therapy or SBRT. SBRT is recommended for treating tumors involving the central nervous system, lung, head and neck, liver, abdomen, spine, and prostate. SBRT is used in patients who cannot tolerate surgery and when the tumor is difficult to remove.

The phenomenal improvement in precision in radiation treatment has led to a paradigm shift in favor of shorter treatment courses, allowing for the delivery of higher daily radiation dose, thereby reducing the total duration of treatment. This is known as hypofractionated radiotherapy. Hypofractionation treatment has been successfully used in certain cancers such as prostate and breast cancers. Conventional EBRT of Prostate cancers typically takes 8 weeks to deliver 70–78 Gy radiation dose. However, with hypofractionation, within 4–6 weeks it is possible to achieve biologically equivalent effects even though the radiation delivered is less than the conventional mode (Miles and Lee, 2008).

The extreme version of hypofractionation includes stereotactic radiosurgery and stereotactic body radiation therapy. SRS/SRT have been effective in primary and metastatic brain tumors, certain benign conditions such as pituitary tumors, arteriovenous malformations, trigeminal neuralgia, acoustic neuroma, etc., using a single large dose of ablative dose with a highly conformal radiotherapy dose distribution (Leksell, 1951). SBRT is a method used to deliver precise high dose radiation to extracranial site targets within body using either single or small number of fractions, particularly in liver or lung tumors (Potters et al., 2010). The clinical outcomes have been favorable across multiple diseases (Grills et al., 2010; Cox, 2012; Hoyer et al., 2005).



Fig. 3 Gamma knife. The patient's head is placed inside the hollow part wherein a large number of radiation sources are placed in the spherical covering part.

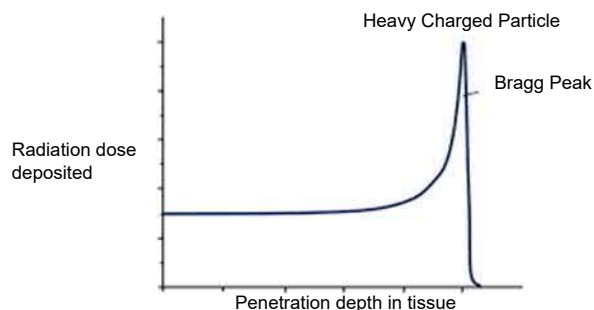


Fig. 4 Bragg peak showing the deposition of energy by charged particles within a short range in tissues.

Hadron (particle) therapy: Hadron therapy is a form of external beam radiotherapy using charged particles rather than X-rays (photons). Protons, carbon ions, or neutrons are generated by a particle accelerator and are delivered to the target volume (Durante and Loeffler, 2010). The charged particles have unique physical and biological properties. They traverse through matter/tissue with little interactions and deposit nearly all their kinetic energy just before the particle comes to rest within a very small volume. This allows a precise definition of the specific region to be irradiated. The peaked shape of the hadron energy deposition is called Bragg peak and has become the symbol of hadron therapy. Fig. 4 shows the Bragg peak.

With the use of hadrons, the tumor can be successfully irradiated with less healthy tissue damage than would be the case using X-rays. In therapy using charged particles, there is a reduction in integral dose, which reduces the risk of complications. There is a growing role of particle therapy in pediatric malignancies—particularly for brain tumors, tumors located near spinal cord, and relatively radioresistant tumors (Schulz-Ertner and Tsujii, 2007). In comparison with 3D-CRT and IMRT, proton therapy has yielded more promising outcomes in local control and survival for prostatic and head and neck and pediatric cancers. Carbon therapy in Japan has shown even more promising results than proton therapy in radioresistant cancers such as melanoma, non-small cell lung cancer and hepatocellular carcinoma (Jiang, 2012). The challenges for particle therapy are the high cost to build the facility, the need for significantly more physical space, and the need for high quality trained staff compared to photon facility centers (Jiang, 2012; Particle Therapy Co-Operative Group, n.d.).

Brachytherapy

In brachytherapy, sealed radioactive sources in the form of wires, needles, seeds or a surface mold/patch are placed directly into or near the cancer tissues to be treated. This procedure is generally limited to well-localized tumors (Skowronek, 2017). Brachytherapy allows a high radiation dose to be delivered locally to the tumor with a sharp decrease of radiation dose outside the tumor. Different types of cancers are treated through different modes of brachytherapy such as interstitial, intracavitary, intraluminal, surface and endovascular. Some brachytherapy treatments are temporary, with the dose delivered over a short time span, and others are permanent placement of the sources within the tissue wherein the dose is delivered over the lifetime of the sources. High dose rate (HDR) mode delivers high radiation dose within a very short time and low dose rate (LDR) mode uses smaller amounts of radiation over longer periods.

Cancers of cervix, prostate, breast and skin are common sites to be treated with brachytherapy. In certain situations, brachytherapy is combined with external radiation therapy. Studies suggest that disease outcomes after brachytherapy are superior to dose escalation with external beam radiation therapy (Jabbari et al., 2010).

In interstitial brachytherapy, the radioactive source in the form of needles or wires is placed within the cancer tissue. This could be temporarily placed for a short time as in the treatment of cancers of breast, head and neck, etc., or could be permanently placed as in the case of prostate cancers (Figs. 5 and 6).

In intracavitary brachytherapy, sealed radioactive sources are placed close to the tumor situated in a cavity, such as the uterus, for a short time. Intraluminal brachytherapy is a mode where the radioactive source is placed within the lumen—such as the trachea, for treatment of oesophageal and bronchial cancers. The surface brachytherapy is applied to cancers that are on the surface such as skin cancers or ocular cancers over which a radioactivity containing mold is placed for specified time to irradiate the cancers.

Endovascular brachytherapy (EBT) is a treatment modality to prevent restenosis of coronary arteries of the heart and blood vessels of the extremities after the original stenosis, or blockage, was treated by angioplasty. It has been proven that radiation decreases the occurrence of restenosis (Rubin et al., 1999). However, with the introduction and wide use of drug coated stents, which are expected to prevent restenosis, EBRT is currently seldom practiced. The following Table 1 lists the salient aspects of different modes of brachytherapy.

Boron neutron capture therapy (BNCT)

BNCT is a therapy modality wherein a boron (Boron-10) containing compound is administered to selectively accumulate in the cancerous tissues followed by irradiation of the tissue with neutrons. The B-10 nuclei capture neutrons and undergo a nuclear



Fig. 5 Interstitial brachytherapy for tongue cancers. Catheters are inserted around the tumor and maintain position with the help of buttons. Catheters are removed after radiation therapy is completed.

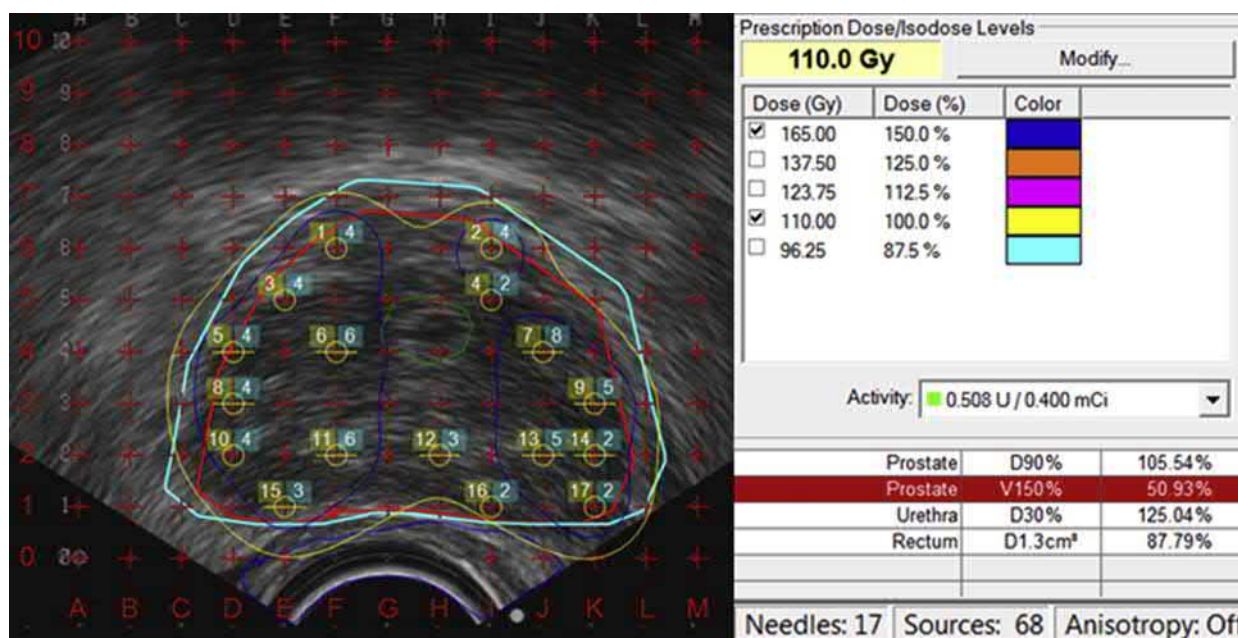


Fig. 6 Interstitial brachytherapy for cancer prostate with Iodine-125 seed (permanent implant) covering the entire prostate tumor adequately delivering approximately 150 Gy. Dose to rectum is substantially low.

reaction, emitting alpha particles and lithium nuclei (Fig. 7), both of which deposit the kinetic energy they carry within an extremely short distance, the size of a single cell.

BNCT is attractive for the treatment of well localized, not too deep-seated tumors, as it is possible to deliver huge amounts of energy precisely within the tumor without causing damage to the surrounding healthy tissues. Brain tumors such as glioblastoma and skin cancers have shown promising results with BNCT. The currently used boron compound is an amino acid derivative, *p*-boronophenylalanine (BPA), which has one boron atom per molecule of BPA. Unfortunately, this is not adequate for efficient therapy. Hence, designing a suitable compound with an adequate number of boron (B-10) atoms, delivering it precisely at the tumor site and having a neutron beam that is amenable to use in a clinical setting, constitute the major challenges in the growth of BNCT.

While neutron beams from reactors were used in the past, accelerator derived neutrons are now increasingly used in BNCT to treat patients (Ridikas et al., 2020; IAEA, 2020). Producing neutrons from a charged particle accelerator is cleverly conceived through the reaction of protons (from the accelerator) on a target to result in a nuclear reaction where neutrons are emitted. One such target is beryllium-11 ($^{11}\text{Be}_4 + ^1\text{p}_1 = ^{11}\text{B}_5 + ^1\text{n}_0$) which has been used for availing neutrons for BNCT. These neutrons emitted carry high energy and are generally passed through a slowing-down medium to reduce their energy to epithermal level for use in BNCT.

The potential of BNCT to treat well localized tumors efficiently without causing damage to tissues in the vicinity, is attractive and, hence, active research is underway to make BNCT an option in clinics. A schematic diagram of a BNCT set up for treating patients using accelerator derived neutrons is shown in Fig. 8 (Courtesy Sumimoto Heavy Industries, Ltd., Japan).

Table 1 Radionuclides used in brachytherapy.

Type of brachytherapy	Clinical conditions	Radionuclides used	Comments
Interstitial	Breast cancers	Ir-192; Co-60; Au-198	Useful in specific cases but not common in current times
	Head and neck cancers	In the form of needles, wires	
	Soft-tissue sarcomas	I-125; Pd-103; Cs-131	Small radioactive seeds are permanently placed for therapy of prostate cancers; practiced widely
	Prostate cancers	In the form of small encapsulated seeds	
Intracavitary	Uterine cancers	Ir-192 (HDR) Co-60 (HDR) Cs-137 (LDR) Sealed small pellets used through a device either remotely controlled (HDR) or manually operated (LDR)	Common; HDR is preferred over LDR
Intraluminal	Esophageal and bronchial cancers	Ir-192; Au-198	Most often used following EBRT as boost radiation therapy to increase the efficacy
Mold brachytherapy	Eye/ocular cancers	I-125, Ru-106	Used only in specific suitable cases; not frequent
Intra vascular	Endovascular Radiotherapy (EVRT) for prevention of restenosis	Re-188; P-32; Ho-166; Sm-153; Y-90	Radioactive liquid filled balloons or Ir-192 sources were placed inside the affected (restenosed) blood vessel. Rarely used now

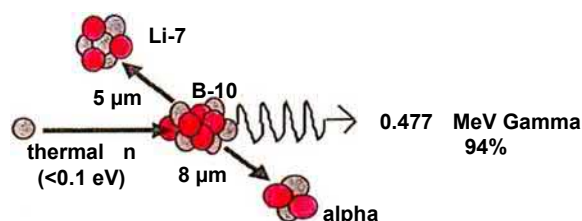


Fig. 7 Neutron capture reaction on Boron-10.

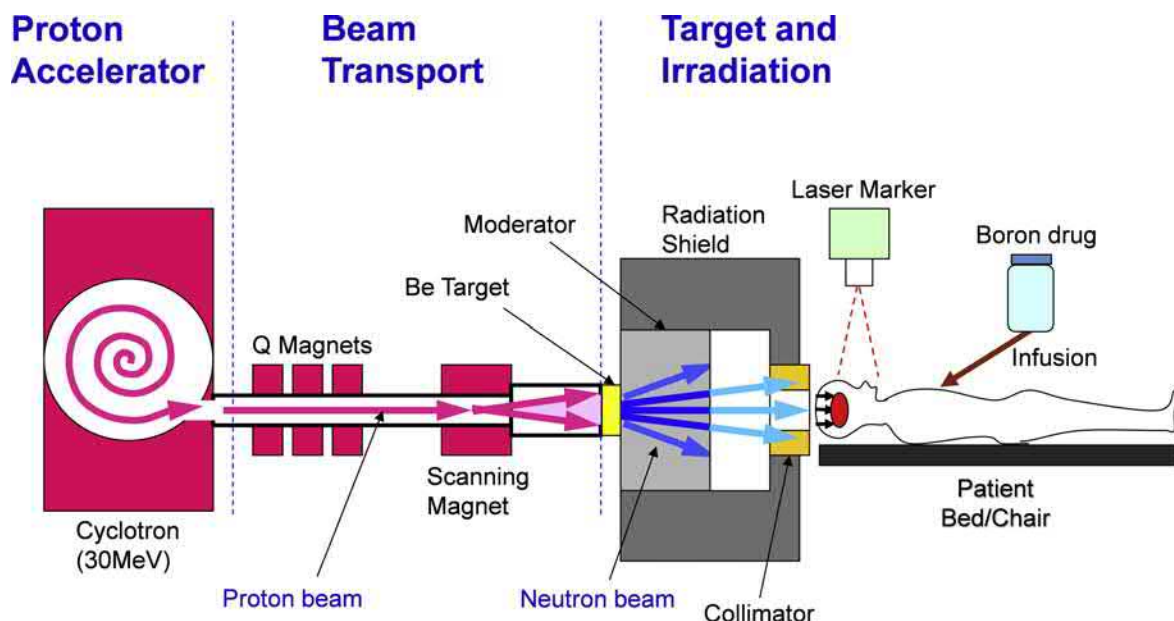


Fig. 8 A schematic diagram of a BNCT Set up for treating patients.

Advances in radiation therapy

The current research in radiotherapy is aimed at the challenges faced in radiotherapy practice.

A typical challenge with clinical radiotherapy is that tumors are frequently located adjacent to the radiosensitive normal tissues. Therefore, the drugs that can preferentially sensitize tumors so that they get destroyed by radiation easily, will be of great interest in oncology. Several radiosensitizer molecules play a critical role by enhancing the anti-tumor effects of radiotherapy. Standard chemotherapeutic agents are commonly utilized to increase the efficacy of radiation therapy.

Radiation causes single and double strand DNA breaks. The double strand breaks (DSB) are generally considered to represent the principal lethal event and desired for radiotherapy. The human body has existing mechanisms to repair DSB and this will decrease the efficacy of treatment. Hence, there has been a growing interest in understanding the mechanism of DSB repair pathways and the ways to block them. Many components of DSB repair pathways have been investigated as therapeutic targets and chemical inhibitors being explored as a possible lead compound for use as drugs to be given in conjunction with radiation (Russell et al., 2003). An example is PARP (Poly ADP-ribose polymerase) inhibitors, which are being studied in several clinical trials with chemotherapy and with radiotherapy (Chalmers et al., 2010; Powell et al., 2010). Similarly an enzyme compound of Histone deacetylase (HDAC) has shown radio-sensitizing properties (Munshi et al., 2005; Zhang et al., 2009).

The other challenge faced is the resistance to radiotherapy in the regions where tumors are hypoxic (low amounts of oxygen). Such hypoxic regions surround “necrosis” (dead cells) that occur due to the uncontrolled growth of cancers. Generally, hypoxic regions are radioresistant. Efforts to develop radiosensitizers that can render hypoxic cells susceptible to radiation are being continued, but clinical success is yet not achieved (Rischin et al., 2010).

There has been extensive preclinical work combining gene therapy and radiotherapy in recent past. Replication-defective viruses have been used to deliver radio sensitizing molecules (such as prodrug, cytokines, tumor suppressor genes and immune activating compounds), which have shown promising results (Harrington et al., 2010).

Applications of radiation processing

Apart from the therapy applications discussed above, radiation processing plays a unique role in healthcare, as briefed below.

Radiation sterilization: Sterilization of medical products such as syringes, sutures, needles, prosthetic implants, etc. through irradiation using gamma rays from Cobalt-60 is the most common practice for large scale medical sterilization (Kalia, 2021). Currently, over 200 gamma irradiators all over the world sterilize 12 Million cubic meters of single use medical devices annually. Radiation sterilization is an attractive option as it does not use or emit pollutants (hence, is environment friendly) and affords the possibility of large-scale operations.

Blood and tissue irradiation: Blood and tissues are irradiated with gamma radiation, most often using Cesium-137, to prevent the proliferation of viable T lymphocytes and render them safe for transfusion/grafting into patients, and to prevent the “Transfusion associated Graft-Versus Host Disease (TA-GVHD)”. TA-GVHD, though a rare complication, has a fatality rate greater than 90% and immune compromised patients are more susceptible.

Radiation processed materials in healthcare: The use of novel materials obtained through radiation processing is an actively growing area. The possibility to produce polymeric material of required physical characteristics, size and in a reproducible manner have led to the exploration of such novel materials in different applications in healthcare. Among the established products, “Hydrogel” is a polymeric material that is breathable and capable of absorbing large quantities of liquids. It is widely used to treat the skin conditions such as burns, bed sores, etc. Radiation processing is an established way to produce sterile hydrogels (Fernando, 2021). The use of polymeric scaffolding materials for tissue growth and the use of nano polymers as substrates for use in radiopharmaceuticals are some of the areas of application being actively pursued currently.

Conclusion

External beam radiation therapy for treatment has a huge impact on the management of cancer patients and the advances in radiation teletherapy over the past few decades have resulted in a phenomenal improvement in success rates. Currently EBRT is a component of therapy strategy in a huge fraction of cancer patients. The understanding of newer avenues such as biological, molecular, immunological and genetic intervention, all of which hold promise are being explored to enhance the efficacy. Brachytherapy has a unique role in treatment of malignant cancers and non-malignant situations. The potential of brachytherapy for treating uncommon rare cancers is explored resulting in new developments in this area, albeit of limited usage.

BNCT has shown promising results in the recent years and this could lead its use in clinics in more countries in near future.

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Agriculture: Improving Crop Production

Sobhana Sivasankar^a and Lee Kheng Heng^{a, a} Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, Vienna, Austria

Si-Yong Kang^{b, b} Advanced Radiation Technology Institute, Korea Atomic Energy Research Institute, Jeongeu, Jeonbuk, Republic of Korea

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Abbreviations

BNF	Biological nitrogen fixation
CRNS	Cosmic ray neutron sensors
CRP	Coordinated research projects at the IAEA
CSSI	Compound-specific stable isotope
DSS4AFA	Decision support system for nuclear emergencies affecting food and agriculture
FA	Food and agricultural organization
FRN	Fallout radionuclides
GHGs	Greenhouse gases
IAEA	International atomic energy agency
IPPC	Intergovernmental panel on climate change
IT-DSS	Information technology-data support system
LET	Linear energy transfer
PBG	Plant breeding and genetics
ppb	Parts per billion
SOC	Soil organic carbon
TAFRA-1	A new variety of groundnuts
T-DNA	Target element insertion into DNA
TILLING	Target induced local lesion in the genome
TUSC	Trait utility system for corn

Introduction

Crop production is central to food security, especially as global population is predicted to surpass the 9 billion mark by 2050. Adverse conditions for crop production resulting from a changing climate, such as increasing temperatures, frequent droughts, soil salinization, changing intensities and frequencies of disease and pest incidence and their transboundary spread, are all factors that further threaten food security. Genetic diversity of crop species achieved through breeding, and optimal soil and water conditions in the growing environment through management, constitute the essential requirements for increasing crop production.

Breeding improvements in crops include increasing crop yields, improving quality, and enhancing resilience to adverse conditions such as diseases, pests, and environmental stress including drought, heat, salinity etc. Management and agronomic practices address soil, water and nutrient requirements of the crop to ensure increased and sustained production. This chapter discusses nuclear, isotopic and related techniques used in creating new genetic diversity for crop improvement, and in the management of soil, water and nutrients for crop production.

Mutation breeding for crop improvement

Improvements in the productivity of plant species that are currently grown as food, feed or cash crops have been the result of domestication, diversification and breeding over centuries. Of particular note in the most recent past is the spectacular increase in productivity of rice and wheat from systematic plant breeding combined with increased fertilizers during the Green Revolution of the 20th century.

Mutations, or naturally occurring genetic variation, followed by genetic recombination, constitute the bases for crop improvement as it did for the domestication of crop plants more than 12,000 years ago. Currently about 2500 species have undergone selection by humans and domestication at one level or other, and 250 species are considered fully domesticated (Meyer and Purugganan, 2013). During the process of domestication, plant phenotypes (characteristics) or traits that were amenable to cultivation were selected by humans in a unique form of directed evolution. The selected plant phenotypes in the early stages of domestication included inflorescence architecture, non-shattering of seeds, loss of seed dormancy, size and number of seeds, determinate growth habit, and others. Subsequently, diversification traits were selected, and most recently crop improvement has been achieved through systematic plant breeding.

However, it was only as recent as the last few decades that the mutations or genetic variations underlying the selected phenotypes in domestication, diversification and breeding were identified, with the advent of molecular genetics and more recently, genomics technologies. One example is the evolution of present-day maize from its wild progenitor, teosinte, as a result of spontaneous mutation in the genetic locus, *teosinte branched 1 (tb1)*, encoding a transcription factor that controls shoot architecture or the branching pattern of the plant (Doebley et al., 1997). The dwarf phenotypes of rice and wheat that drove the Green Revolution during the 20th century were identified to be the results of mutations in the genes, *Reduced height (Rht)* in wheat, and *semidwarf1 (sd1)* in rice, involved in the regulation of the biosynthesis (synthesis of a molecule or compound in a biological system), or of the biosynthesis itself, of the plant growth hormone, gibberellin (Hedden, 2003).

Mutations are the ultimate source of all genetic diversity, and genetic diversity in turn is centric to crop improvement. Mutations in the DNA can occur spontaneously in nature or can be induced, and this together with genetic recombination leads to the development of new crop varieties. Since the last century, plant breeders have been systematically utilizing the existent natural diversity in germplasm pools (genetically divergent groups of plants) of crop species to select improved varieties. The extent of genetic gain or improvement achieved through plant breeding is dependent on the extent of genetic diversity in the base germplasm pool. In the major crops of the world such as maize and rice, the historic intensive breeding of locally adapted germplasm has led to a significant reduction of genetic diversity. Low genetic diversity is also encountered in vegetative propagated crops.

Induced mutations facilitate the creation of new genetic diversity in the germplasm allowing the development of new and improved varieties, while also enabling the acceleration of crop improvement. This accelerated genetic gain is critical to meet the food security needs of a growing global population. It is also crucial to facilitate crop adaptation to increasing temperatures, frequently occurring droughts, rapid salinization of soils, transboundary pests and diseases, and other adversities that accompany a changing global climate.

Nuclear techniques in mutation breeding—A brief history

Spontaneous mutations have been the bases of domestication for thousands of years. The term “mutation” and the practice of mutation breeding, however, evolved only as recently as the 20th century. Hugo de Vries, one of the rediscoverers of Mendel’s laws of inheritance, along with C. Correns and E. von Tschermak-Seysenegg, was researching on the evening primrose *Oenothera lamarckiana* during the late 19th century, when he discovered heritable variation that did not follow a Mendelian inheritance pattern. He coined the term “mutation” in 1901 to define the mechanism that creates heritable change in an organism. In the introduction to his publication, *Die Mutationstheorie* (Volume I), in 1901, De Vries speculated on the possibility of artificially inducing mutations in species. In 1904, he had already indicated the possibility of artificial mutation induction using the new types of radiation discovered in 1895 and 1900, respectively, X-rays and γ -(gamma)-rays. The work of De Vries can be considered as the starting point of the science of mutation breeding for crop improvement.

Irradiation of plant material for inducing mutations started soon after the discovery of X-rays. However, the first definitive proof for the induction of mutations within genes came from H. J. Muller in 1927 who showed that treating sperm in the male fruit fly, *Drosophila melanogaster*, with high doses of X-rays resulted in very high mutation rates of sex-linked recessive lethals (a hereditary change), as compared to untreated control parent flies (Muller, 1927). Around the same time, L. J. Stadler, a plant geneticist, was working on inducing mutations in barley, oats, wheat and maize using X-rays and radium rays, and published his first results a year later (Stadler, 1928). Stadler’s studies showed that irradiating germinating seeds produced an eightfold increase in mutation frequencies—relative to irradiating dry seeds, and that the mutation rate was proportional to the dose of irradiation, with the rate for a given dose differing between, and even within species. Stadler, however, was skeptical of the use of induced mutations for crop

improvement. He considered that sufficient variability existed within gene pools of crop plants to allow improvement by conventional breeding, and that newly induced mutations would be linked to other deleterious mutations. Notwithstanding Stadler's skepticism, contemporary plant geneticists including H. Stubbe and coworkers in Germany, and H. Nilsson-Ehle and A. Gustaffson in Sweden, used X-rays and UV-rays to induce mutations in plant species, and the latter became the first to successfully develop genetically superior cultivars in barley through mutation induction (Kharkwal, 2012). A. Gustaffson could probably be called the 'father of mutation breeding'. The first commercial mutant crop cultivar was developed in tobacco through irradiation of inflorescences (floral structures of plants) with X-rays as early as 1934 by D. Tollenaar, and this mutant cultivar, "chlorina," occupied 10% of the tobacco area in Indonesia already by 1936.

Several new mutagens, both physical and chemical, have since been used after the early use of X-rays, radium and UV rays for generating genetic variability. Their mechanism of action at the level of the DNA can vary from single point mutations or base changes to large deletions involving several megabases (large numbers of DNA base pairs). Next-generation genomics technologies have allowed, most recently, the deciphering of the effects of different mutagen sources on the DNA. The use of these various mutagens has given rise to the induction of a wide range of mutations in plant species, several of which have been used in fundamental studies to establish gene-to-phenotype relationships. Mutation studies in the model plant species, *Arabidopsis thaliana*, a small flowering plant with a small and simple-structured genome ($2n = 10$; ~ 120 Mb; few repeats) and the first plant to be fully sequenced, has yielded significant amounts of information over the last 50 years on the molecular biology of plants. These studies also shed light on genes involved in numerous biochemical and hormonal pathways, and morphological and physiological traits. More important, however, has been the use of induced mutations in practical plant breeding for the development of new and improved crop varieties for wide cultivation by farmers.

Mutation breeding took off in the 1950s with several countries, including China, Germany, India, Japan, the Netherlands, Sweden, UK and USA incorporating the technology into major plant breeding programs, resulting in the development of improved varieties of many crop species. Examples of these mutant varieties in food, oil seeds, cash crops, and horticultural crops developed and released for cultivation in the early years of mutation breeding are many. They include the rapeseed variety, Regina II, developed in Sweden and released in Canada in 1953; the induced mutant variety of sweet cherry, Compact Lambert, released in Canada in 1964; the first semi-dwarf rice variety, Reimei, released in Japan in 1966; the first semi-dwarf rice variety, Calrose 76, developed in the USA in the 1970s and integrated into breeding programs in the USA; Australia and Egypt to produce more than 20 new semi-dwarf rice varieties; the mutant grapefruit varieties, Star Ruby and Rio Red, developed in the USA and sold under the trademark Rio Star, which occupies 75% of the cultivated area under the crop in Texas; Todd's Mitcham and Murray Mitcham varieties of *Verticillium*-resistant peppermint released in the USA in 1971 and 1976; and others.

In 1964, the Joint Division of the Food and Agricultural Organization (FAO) and the International Atomic Energy Agency (IAEA) was established at the IAEA headquarters in Vienna to address Nuclear Techniques in Food and Agriculture. At that time, Plant Breeding and Genetics (PBG) was included as one of the five sections within the Joint Division, and this gave impetus to mutation breeding in many Member States. The technical support from the Section has enabled more than 70 developing countries to build capacities in mutation breeding, and improved mutant varieties have been developed and released especially by countries in the Asia Pacific region. The laboratory facilities of the Joint Division at Seibersdorf feature a Cobalt-60 γ -irradiator and an X-ray system, both of which are routinely used for irradiation of seed and vegetative plant material as a service provided by PBG to Member States. The wide establishment of Co-60 irradiation facilities in the 1950s made γ -rays a very popular mutagen source for mutation breeding during this early phase.

By the 1980s, with transgenic research starting to gain prominence, the role of induced mutations in plant breeding started to wane in North America. However, major seed and biotechnology companies in the USA continued to use various forms of mutagenesis to establish gene-to-phenotype relationships, which in turn lead to the identification of candidate genes that could be tested for transgenic trait improvement (Sivasankar et al., 2012). Mutation breeding also remains a popular method for the improvement of ornamentals and vegetables, although information of its use is not publicly available as most of this breeding is by seed companies. In the countries of the Asia Pacific, however, the use of induced mutations in mainstream breeding programs continued in the meantime, and a majority of mutant crop varieties of the world developed to date are recorded for this region. The IAEA maintains the Mutant Variety Database at <http://mvd.iaea.org/>, which is a repository of voluntarily contributed information of released mutant crop varieties. It includes both direct mutants (directly from mutation and selection) and indirect mutants (a mutant parent used in crosses to generate new varieties) (Fig. 1). The database, which is by no means complete, holds 3314 records across 220 plant species at the time of this writing, about 50% of which have originated from γ -ray irradiation. The majority of the records (over 2000) is from the Asia Pacific region, which is followed by Europe. China alone has 810 records in the database, and has been using several mutagen sources including, most recently, ion beam and space irradiation (taking seeds into space for exposure to cosmic radiation).

Mutagens in plant breeding

Mutations can be spontaneously occurring in nature or induced. Induced mutations are derived from specific treatments with physical or chemical mutagens. Mutagenesis with physical mutagens has been the most frequently used method to create genetic diversity for the development of new and improved crop varieties. Physical mutagens include non-ionizing radiation such as ultra-violet (UV) light, and ionizing radiation such as X-rays, gamma-rays, alpha and beta particles, neutrons and ion beams (FAO/IAEA, 2018). Further, cosmic irradiation and laser beam irradiation have also been used to create changes in the plant DNA.

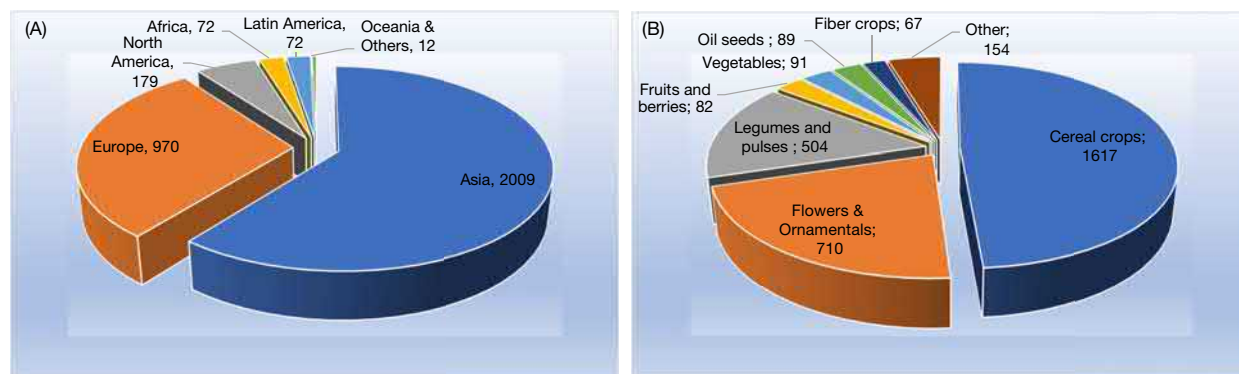


Fig. 1 Current records (voluntary reporting by IAEA Member States) of mutant varieties in the IAEA Mutant Variety Database. (A). Mutant varieties released from the different global regions across the years. Asia is the leading region, and China has the largest number of mutant varieties in the region, followed by Japan and then India. (B). Released mutant varieties across different crop categories. Cereal crops lead, with rice having the largest number of records. The categories of Fruits/Berries, and Flowers/Ornamentals, include tree species and shrubs. "Other" include medicinal species, fodder and forage, grasses, mushrooms etc.

UV light is a non-ionizing radiation with limited tissue penetration that has been frequently used for mutation induction in sensitive plant materials such as pollen grain, single cell-layer tissues, tissue culture material, cell suspension cultures, spores, etc.

Ionizing radiation can remove electrons from molecules in a cell and create free radicals. Free radicals and particle tracks of an ion-beam can break one or both DNA strands, which causes gene or chromosomal changes such as deletions, substitutions, insertions, translocations, duplications, and inversions. When compared with non-ionizing radiation, ionizing radiation penetrates deeper into the tissue and can induce a great number of different types of chemical changes. Hard X-rays with short wavelength are generally preferred rather than soft X-rays for mutation induction, due to the higher penetration of the former. Specific filters used in hard X-ray production help absorb unwanted soft radiation. The peak operating voltage, thickness and type of filter, distance of tube to target, and dose rate are factors that affect the usefulness of the mutagenesis event.

In the early days of mutagenesis, X-rays were mainly used, but with the advent of gamma irradiation facilities, gamma rays have been used to a great extent. Cobalt-60 (^{60}Co) with a half-life of 5.26 years, and Cesium-137 (^{137}Cs) with a half-life of 30.17 years are the main sources of gamma rays. Gamma rays have been used both for acute and chronic irradiation. Acute irradiation facilities irradiate plants with high doses in a short period of time in gamma rooms or gamma cells. Chronic irradiation facilities installed with gamma radiation sources (^{60}Co or ^{137}Cs) expose plants to low doses of gamma-rays for long periods while growing plants in fields, greenhouses, or environmentally controlled (phytotron) rooms. Such chronic irradiation facilities have been built and operated in several countries such as Japan, Malaysia, Korea and Thailand (Fig. 2).

Alpha and beta particles have been only used to a very limited extent in experimental mutation induction in plant tissues. Thermal (0.4–100 eV) and fast neutrons (200 keV–10 MeV) are also used in mutation induction, although their use has also been limited relative to X-rays and gamma-rays. The most famous example of a crop variety developed from mutation induction by thermal neutrons is Star Ruby, the red-fleshed, seedless variety of grapefruit, developed in 1970. More recently, heavy ion beams and cosmic irradiation are also coming into use in mutation induction for plant breeding.

Many research groups in Japan have created new varieties of various plants and microbes through mutations by heavy ion beams generated by big accelerators. The heavy ion and proton beams belong to high-LET (Linear Energy Transfer) radiation while the radiations of X- and gamma-rays are classified as low-LET. According to recent studies, high-LET radiation has a higher frequency and spectrum of induced mutations due to the ease of inducing single and double strand breaks, not achievable by low-LET radiation. Despite these advantages, high-LET is difficult to attempt in many countries because the construction of an accelerator facility for ion beam irradiation is too expensive, and the number and volume of samples that can be irradiated at one time is limited. Recently, securing gamma-ray sources has become more difficult due to the strengthening of international transportation laws and safety management. Therefore, there are opinions that X-rays and electron beams should be considered again for use in the future instead of gamma-rays. In addition, since the early 1990s, China has been developing several mutant varieties by space-induced mutagenesis using recoverable satellites, spaceships, and high-altitude balloons that carry plant seeds into space.

It should be emphasized that, unlike many people's concerns, radiation used for mutant breeding only uses its transmission energy to change the genes of the living cell nucleus. It does not remain behind or contaminate living things and the environment.

Chemical agents can also cause mutations in cells. These include a special class of alkylating agents such as ethyl methane sulphate, diethyl sulfate, ethylamine, ethyl nitroso urethane, ethyl nitroso urea, and methyl nitroso urea. Chemical mutagens most often affect single nucleotide pairs and generate point mutations. The benefit of using chemical agents is a high mutation rate, simple application, and less damaging effects at the DNA level. However, chemical mutagens have several disadvantages, including difficult penetration in multicellular plant tissues, uneven penetration, low reproducibility, and health-risks associated with poor handling. Beyond physical and chemical mutagenesis, mutations can also be induced by biological means, as with tissue culture techniques, and T-DNA or transposon element insertion (random insertion of the mobile DNA elements in the genome).



Fig. 2 Different gamma radiation facilities for acute and chronic irradiation: Gamma room (above) and Gamma Phytotron (below) of the Advanced Radiation Technology Institute, Korea Atomic Energy Research Institute in Korea.

Mutation breeding has several advantages. First, it is applicable to various plant species and materials such as annuals, perennials, seed or vegetative propagated crops, seeds, stem or root cuttings, tubers, pollen, cell cultures, tissue cultures and individual plants. In other words, radiation-induced mutation breeding can be used to increase the genetic diversity of crops that cannot be improved by conventional breeding through crossing. Second, it can be applied to the breeding of crops lacking genetic resources and can contribute to the expansion of breeding materials. Third, mutants have contributed greatly to the search for new genes in functional genomics approaches. Despite these advantages of mutation breeding, there are limitations in the easy generation of useful variations relative to cross breeding. Optimal dose rate and precise, trait specific selection is critical to capture mutants with useful trait variations, but with minimal effect on the agronomic performance of the plant.

Mutations and functional genomics platforms

The term “functional genomics” implies the determination of gene function at the genome-wide scale. It is essentially the establishment of gene-to-phenotype relationships. A phenotype or trait is a specific observable characteristic of an organism, and for crops it can include yield, stature, disease resistance etc. A trait can be the result of the action of a single gene (simple trait) or a large number of genes (complex trait). Prior to the genomic era, induced mutations were routinely explored in model plant species in order to identify the genes underlying specific phenotypes such as flowering time, morphological changes etc., and to unravel the molecular biology of biochemical and hormonal pathways. Establishment of gene-to-phenotype relations are important both for fundamental studies and for the identification of candidate genes responsible for specific traits. Candidate-gene information can be used to improve desirable crop traits through genome editing or marker-assisted breeding. The seed biotechnology industry uses various types of mutant populations for such candidate-gene discovery, including insertions of transposons or mobile genetic elements, activation tagging, chemical mutagenesis and others. Such candidate-gene discovery has been described from the perspective of developing drought tolerance in maize (Sivasankar et al., 2012).

A mutant population that is systematically developed, maintained, phenotyped and preferably, genotyped as well, can constitute a powerful functional genomics platform. Trait-specific phenotyping for candidate gene discovery in a forward genetics approach (from the phenotype to the gene) and mutant plant recovery through reverse genetics approaches (from the gene to the phenotype)

can both be achieved in such a platform with a single mutant population. An elegant example for such a platform is the Trait Utility System for Corn (TUSC) that was developed and maintained for both gene discovery and product development in the 1990s by the seed company, DuPont Pioneer, now Corteva. The mutations in this case involved transposon insertion. Subsequently, the reverse genetic approach, TILLING (Targeting Induced Local Lesion IN the Genome) has been used to recover mutant plants with point mutations in specific genes from chemically-mutagenized populations. TILLING is a high-throughput, sensitive and cheap approach to identify, from a mutant population, plants with alterations in a specific gene of interest.

A functional genomics platform involving a plant mutant population originating from physical mutagenesis employing X-ray, γ -ray or heavy ion beam applications can also be queried for mutant plants with point mutations or small indels (insertions and deletions) in genes of interest. Instead of the TILLING protocol described above, new high-throughput sequencing of amplicons (amplifications of DNA) of targeted regions in the genome in mutant DNA pools facilitates the recovery of mutant plants harboring mutations in the target genes. Such a functional genomics platform for sorghum, irradiated with γ -ray, has recently been developed at the Joint FAO/IAEA Division, and a bioinformatics package for querying the presence of mutants in genes of interest is in development.

The beginning of the present decade saw the advent of the Genome Editing technology using the CRISPR-Cas9 system and others for site-specific modification or editing of “candidate” genes known to underlie traits of interest. The technology enables a precise (site-specific) change in the genome, in a candidate gene or genetic element. A pre-requisite for genome editing is the identity of the gene that is responsible for a phenotype, the candidate gene. This information is available only from spontaneous or induced perturbation of the genome through mutations, and the subsequent establishment of gene-to-phenotype relationships. Genome editing currently has the advantage of improving simple traits (controlled by one or two genes), when the identity of the candidate gene is known. For all other applications, such as the improvement of complex traits (controlled by several genes) or the improvement of a trait when the identity of the candidate gene is not known, traditional breeding, or even better, mutation breeding, is the best resort.

Speed-breeding techniques for accelerated genetic gain

The time required to develop a new and improved variety in most public breeding programs at present is too long to meet the food needs of a growing population, especially in view of a changing climate that adversely affects crop production. In general, most food crops with a long generation time take about 10 years for the development of an improved variety. Vegetative propagated crops such as ornamentals require only half this time when variability is induced by somatic cell mutation.

Mutation breeding for seed propagated crops is started by subjecting seeds, whole plants, or gametes (pollen, egg cells) to mutagenic treatments. Seeds are the most commonly used initial materials for induced mutation. The seed of a parental line that has been treated to a mutagen is designated M_0 , which upon germination produces M_1 plants. Seed from M_1 plants are grown as M_2 where the first selection can be made for the phenotype or trait of interest. Subsequent advances of generations and selections in each generation is continued for three or four generations to obtain genetically stable mutants. In most countries, selected promising mutant lines are tested in multi-locational field tests under different environments. Promising selections then undergo registration tests of an official agency before registration and release as varieties.

Crops with longer duration can thus take up to 10 years for the development of a new variety. “Speed breeding” is a concept that includes a variety of techniques, some of which are guided by the crop and the trait of interest, that can significantly shorten breeding times and improve selection efficiency. These include doubled haploidy (doubling the number of chromosomes normally occurring in the mature germ cells), rapid cycling, shuttle breeding, molecular marker-assisted selection and genomic selection, all of which are used in the seed industry to accelerate the pace of breeding programs. These techniques, combined in recent times with “omics” technologies (technologies that address genome at the organism level or genomics, at the transcript/RNA level or transcriptomics, at the protein level or proteomics, at the metabolite level or metabolomics and at the phenotype/characteristics level or phenomics) and Big Data, have the potential to improved breeding efficiencies at increasingly affordable costs.

Impacts of mutation breeding on food security and income: Some recent examples

Induced mutations or variants have been routinely used in plant breeding for the improvement of a variety of traits, including increased yield, climate-change adaptation (tolerance to drought, heat, salinity etc.), resistance to transboundary diseases, and others. Much success has been achieved in crop improvement through mutation breeding in Member States of the IAEA through the technical support of the Joint FAO/IAEA Division. Mutation breeding, through nuclear techniques or chemical means, provides new genetic diversity, can be applied more easily compared to transgenic breeding, and can be used for step changes in crop improvement in countries that are not inclined, or do not have the regulatory framework, to adopt transgenic solutions.

Yield increase: Increased yield has been one of the main mutation breeding targets for many of IAEA’s Member States. In Vietnam, 20 mutant rice varieties have been released in recent years, which are grown by more than 300,000 farmers. Further, 50% of the soybean area in the country is covered by mutant varieties. In Peru, the barley mutant variety, Centenario II, covers 18% of the barely area and contributes USD 6.6MM annually, while the amaranth variety, Centenario, covers 47% of the amaranth area.

Climate-change adaptation: Mutation breeding for climate-change adaptation has mostly addressed tolerance to drought, heat and salinity. Sudan, a country that ranks fourth among the leading groundnut-exporting countries, produces 70% of the crop

in the western States of Kordofan and Darfur, which are prone to late-season drought stress. This has serious implications on groundnut pod-filling and maturation, and consequently production. A recent IAEA project that combined mutation breeding with farmer-participatory selection led to the identification of a new variety, TAFRA-1, which has shown 11% increased yields in the marginal dry areas above the prevailing local varieties. In Pakistan, four new heat-tolerant mutant cotton varieties have been identified since 2013, and these currently cover more than 25% of the 3.1 million hectares of the cotton production area in the country.

Disease resistance: Resistance to diseases is a major trait in crop plants for which genetic diversity is constantly sought, as existing resistance could break down, pathogen races can evolve, and transboundary spread can lead to extinction of popular crop varieties and seriously impact national economies. Mutation breeding has helped develop genetic variability for disease resistance in several instances. Resistance of barley to mildew developed through induced mutagenesis with X-ray was reported as early as 1942 by R. Freisleben and A. Lein. A mutant Japanese pear variety, Gold Nijisseiki, with resistance to the black spot disease, was developed in Japan in the 1990s. It combined the excellent quality of the popular variety, Nijisseiki, with resistance to the disease. A patent for its invention filed in 1992 was awarded in 1994 (US patent, USPP8529P). Disease resistance, when monogenic or the result of a single gene, is usually broken down within a short time. However, there have been exceptions, with the classical example being resistance conferred by mutation in the *ml-o* locus against all known fungal pathotypes of the powdery mildew, *Erysiphe graminis*, in barley, which has lasted a long time without breaking down.

Most recently, the disease Fusarium wilt in banana caused by the soil-borne fungus *Fusarium oxysporum* f. sp. *cubense* Tropical Race 4, that has been devastating plantations in Asia, spread to Africa in 2013, and to Latin America in August 2019. Mutation breeding efforts, through chemical mutagenesis in China, has already led to the development of a resistant banana variety, a breakthrough that holds promise for combating this disease with mutagenesis.

Nuclear and isotopic techniques in soil, water and nutrient management

Soil, water and nutrients are key natural resources that are critical for food security and sustainable development. Each year approximately 5–7 million hectares of surface soil (soil in the top 20 cm) (including nutrients) are lost through inappropriate soil management (FAO, 2019). Meanwhile more than 70% of the global water use is consumed for crop production and half of this is wasted through inappropriate fertilization and irrigation practices, polluting surface and groundwater. Climate change, with extreme weather events such as droughts and flooding, further exacerbate the issues. Improving the management of these resources through climate-smart agricultural practices can enhance food productivity and environmental sustainability.

Nuclear and isotopic techniques offer comparative advantages over conventional methods, as they often are more accurate, precise and specific, allowing effective assessment of the impact of climate change on agriculture and vice versa, so that technologies for adaptation and building resilience to climate change and the improvement of agriculture practices for potential mitigation can be developed.

In the following paragraphs, the unique role of nuclear and isotopic techniques that include the use of radioactive and stable isotopes, in soil, water and nutrient management is briefly discussed. Included are impact stories and case studies to showcase how these techniques can provide information essential to developing strategies and optimize soil-water-plant-nutrient management practices to increase crop productivity and foster environmental sustainability.

Integrated soil fertility and crop nutrition management

With the rapid population increase and the need for sustainable agricultural intensification in many parts of the world, improving soil fertility, nutrient use efficiency, and crop nutrition, is essential to enhance crop productivity and food security.

The stable isotope of nitrogen-15 (^{15}N) was discovered in the 1920s and the advantage of using it in agriculture, for instance for the determination of N use efficiency, has been recognized and applied very early on. It is estimated that half of the global application of 120 million tons of N fertilizer, which is worth billions of dollars, are not taken up by plants. Such N use efficiency study could be carried out because it is possible to label normal N fertilizer with ^{15}N isotopes. Hence the ^{15}N technique allows tracing N-fertilizer movement throughout the plant (IAEA, 2001). By identifying the pathways of the nutrient flow, substantial information can be drawn for N utilization efficiency. This allows for the development of integrated and sustainable soil and plant nutrient management practices for increasing soil fertility and crop yields.

The ^{15}N technique, when coupled with efficient Rhizobium bacterial strains, has been used to quantify the amount of N fixed through the process of biological N fixation (BNF), which is an alternative source to inorganic N fertilizer for crop growth (Chalk and Craswell, 2017). The N fixed by legume plant nodules not only promotes growth of the host plant but also leaves more N in the soil for subsequent crops. Major grain legumes have been estimated to fix approximately 11.1 million metric tons of N per year from the atmosphere in developing countries, as measured by the stable ^{15}N isotope methodologies (Hardarson et al., 2003). This helps reduce the need of inorganic N fertilizers for many poor farmers.

In tropical acid soils, isotopic and associated techniques such as the carbon-13 (^{13}C), phosphorus-32 (^{32}P) and phosphorus-33 (^{33}P) techniques have been of assistance to: (i) identify crop genotypes that are tolerant to aluminum toxicity and phosphorus (P) deficiency that are common in tropical acid soils; (ii) assess the effectiveness of a range of P fertilizers in providing P for crop requirements and (iii) quantify the enhancing effect of conservation tillage on soil quality.

Improving water use efficiency and water quality

Agricultural production and food security are highly dependent on the availability of water. Improving agricultural water management and its quality is key to sustainable and productive agriculture. Hence accurate measurement and assessment of the spatial and temporal dynamics of soil water content is crucial. This is the first step towards tackling agricultural water management problems.

Nuclear and isotopic techniques also play an important and unique role in providing information essential for developing strategies to improve agricultural water use efficiency (WUE), and hence in providing solutions to mitigate the increasing water scarcity. For example, the variation in the composition of water isotopes (^{18}O and ^2H) in the water vapour surrounding plants makes it possible to separate the process of evapotranspiration into individual component of plant transpiration and soil evaporation, so that measures can be taken to minimize the evaporation component to improve WUE. The soil moisture neutron probe is an accurate device for irrigation management—especially under saline conditions and cracking soil to improve irrigation management (IAEA, 2008). The newly established cosmic ray neutron sensor (CRNS) (Desilets et al., 2010; Wahbi et al., 2018) has been shown to be suitable for providing real-time and accurate landscape soil-water content measurements, making it ideal for long-term monitoring in agricultural ecosystems to aid in agricultural water and nutrient management decisions at larger scales (20 ha in area). This includes drought and flood forecasting, as it provides large scale data on saturation of near-surface soil layer, which is controlling the partition of infiltration and runoff. The principle employed here is to focus on fast neutrons onto the soil. If water is encountered, the neutrons are slowed down and easily registered by a thermal neutron detector.

The water stable isotopes (^{18}O and ^2H) are also suitable to evaluate groundwater sources and to quantify groundwater recharge rates as each water sources has unique water isotope composition, which remain constant as long as there are no phase changes along the flow-path. By knowing the origin and age of the water, sustainable groundwater extraction rates can be determined to avoid over-exploitation of aquifers. Isotopic compositions of water can furthermore be used to trace the return and loss of irrigation water, as evaporation changes the isotopic ratio. The use of ^{18}O and ^2H is therefore an important part of agricultural water management, allowing the identification of water sources and the tracking of water movement and pathways within agricultural landscapes as influenced by different irrigation technologies, cropping systems and farming practices. It also helps in the understanding of plant water use, including its efficiency, quantifying crop transpiration and soil evaporation allowing farmers to devise strategies to improve crop production, reducing unproductive water losses and minimizing land and water degradation (IAEA, 2017).

The stable isotope ratio of nitrogen in nitrate is distinct in different nitrate sources, for example synthetic fertilizer ($\delta^{15}\text{N} \approx -1\text{‰}$ to $+2\text{‰}$), animal waste or sewage ($\delta^{15}\text{N} \approx +8\text{‰}$ to $+16\text{‰}$) (Wassenaar, 1995), making it possible to trace the origin of nitrate and, hence, allows distinguishing between natural and synthetic inputs. Combined with other isotopic tracers such as boron and modeling approaches, it is further possible to distinguish between sewage, manure or mixed nitrate sources. Changes in the isotopic compositions of nitrate over time can also trace and quantify nitrification processes, which are important to evaluate the residence time and leaching of nitrate, and its natural remediation capacity within the soil column and groundwater.

Soil erosion control

Soil erosion by water results in an irreversible loss of fertile topsoil, reduction of soil productivity, damage of infrastructure, pollution of water resources and accelerated sedimentation in water reservoirs and harbors. Erosion leads to soil disruption and aggregate breakdown and hence, partial mineralization of soil organic carbon (SOC) and CO_2 release. The annual global cost of soil erosion and its associated downstream impact have been estimated at US \$400 billion (FAO, 2017). Accurate information on the extent and magnitude of soil erosion affecting agricultural land to strengthen soil conservation strategies is becoming crucial.

New techniques for evaluating soil redistribution processes are available to complement conventional methods. A great advance has been achieved due to the development and refinement of erosion assessment methods using specific radionuclides. Four of them are called fallout radionuclides (FRNs), because they are deposited in soil via atmospheric fallout from nuclear weapons testing, and are used as soil erosion tracers. The most commonly used radiotracer is the anthropogenic medium-lived ($T_{1/2} = 30.2$ years) cesium-137 (^{137}Cs) that originates mostly from past atmospheric nuclear weapon tests and nuclear power plant accidents (Zapata, 2002). Similar application is possible using long-lived isotopes of plutonium (i.e., $^{239} + ^{240}\text{Pu}$) originating also from the previous atmospheric nuclear weapon tests. Further information on long-term sediment budget and short-term erosion events can be gained by assessing the distribution of natural radionuclides such as the medium-lived ($T_{1/2} = 22.3$ years) geogenic lead-210 (^{210}Pb) and short-lived ($T_{1/2} = 53.3$ days) cosmogenic beryllium-7 (^7Be), respectively (Meusbürger et al., 2018).

The principle to investigate soil erosion and sedimentation is similar for all four FRN tracers. When deposited on the soil surface, they are strongly bound to soil colloid particles (clay and organic matter) in the uppermost soil layer and they move in the environment mainly by mechanical and physical processes such as tillage and soil erosion. Therefore, if the soil surface is eroded, the level of FRNs is significantly reduced. In deposition areas, the quantity of FRN increases. Erosion or deposition processes can be precisely estimated by comparing the level of selected FRN at a studied site to established inventory at a reference site that was undisturbed and represents the original fallout. The estimates of soil redistribution rates can then be derived using specific FRN conversion models.

Due to the different half-lives of these radionuclides, the combined use of FRNs (i.e., ^{137}Cs and ^{210}Pb) have been used, for instance, to estimate medium and long-term soil erosion rates and to establish sediment budgets in agricultural fields under Mediterranean climatic conditions in Morocco and for studying the effects of land use and the shift in tillage systems, i.e., from conventional to no-tillage in Chile. Moreover, in Madagascar the use of ^{137}Cs and ^{210}Pb highlighted that traditional terrace systems could

reduce soil erosion by 40%. On the other hand, the short-lived ^7Be has been used for evaluating the impact of short-term (up to 6 months) changes in land use practices and the effectiveness of soil conservation measures for ensuring sustainable management (Mabit and Blake, 2019).

In the last decade, an innovative method for investigating soil degradation processes and especially for establishing the origin of sediment emerged. It is based on compound-specific stable isotope (CSSI) analysis of carbon-13 (^{13}C) in targeted biomarkers such as fatty acids. Indeed, plant communities “label” the soil where they grow by exuding organic biomarkers, and plant species produce a similar range of organic compounds, but with different ^{13}C signature (IAEA, 2019).

FRNs and more recently the CSSI techniques have been tested and validated in various IAEA Coordinated Research Projects (CRPs) and the findings disseminated through the Technical Cooperation Program to more than 70 IAEA Member States to improve the effectiveness of their soil conservation strategies (Mabit et al., 2018).

Greenhouse gas emissions from agriculture

The rapidly changing global climate due to the increased emission of anthropogenic greenhouse gases (GHGs) is leading to an increasing occurrence of extreme weather events such as droughts, floods and heat waves. It is estimated that agricultural activities, deforestation and draining wetlands contribute to some 25% of the total GHGs emission. Agricultural emissions are generally linked to the management of agricultural soils, livestock, rice production and biomass burning.

The three major GHGs include carbon dioxide (CO_2), methane (CH_4) and nitrous oxide (N_2O). The major natural sources of CO_2 include ocean-atmosphere exchange, respiration of soils (microbial respiration) and plants, and volcanic eruption; while the anthropogenic sources include burning of fossil fuel (coal, natural gas and oil), deforestation, and the increasing conversion of peat-land and forest land for agriculture. Natural sources of CH_4 emission include wetlands and oceans. Paddy fields used for rice production, livestock production systems (enteric emission from ruminants), landfills, and the production and use of fossil fuels are the main anthropogenic sources of CH_4 .

Nitrous oxide, in addition to being a major GHG, is also an ozone-depleting gas with the highest global warming potential of 298 times that of CO_2 over a 100-year time period. Anthropogenic emissions of N_2O occurs mostly through agricultural and other land-use activities including the intensification of agricultural and other human activities such as increased use of synthetic fertilizer (119.4 million tonnes of N worldwide in 2018), inefficient use of irrigation water, deposition of animal excreta (urine and dung) from grazing animals, excessive and inefficient application of farm effluents and animal manure to croplands and pastures, and management practices that enhance soil organic N mineralization and C decomposition. Over the last decade the rate of N_2O increase was equal to $0.93 \text{ ppb year}^{-1}$.

The role of individual factors controlling the occurrence and rate of the key N transformation processes, such as denitrification and nitrification are uncertain. Nitrification results in ammonium (NH_4^+) being converted to nitrate (NO_3^-) under aerobic conditions, while denitrification is the reduction of NO_3^- to N_2 under anaerobic conditions. Nitrifier-denitrification occupies the niche between nitrification and denitrification and occurs as oxygen concentrations approach an anaerobic status. Under these conditions nitrifiers actually convert nitrite to N_2O and N_2 instead of nitrate.

These GHGs, CO_2 , CH_4 and N_2O , derived from various sectors, play a major role in regulating earth's temperature (Fig. 3). Without GHGs in the atmosphere, the average global soil surface temperature would be about -19°C , compared to the present values of 14°C (Karl and Trenberth, 2003; NASA 2009).

Recent data by the UN Intergovernmental Panel on Climate Change (IPCC) clearly show that anthropogenic emissions of GHGs are at the highest in history (IPCC, 2014). Since 1900, earth's average surface air temperature has increased by about 0.8°C , with much of these increases taking place since the mid-1970s. (Fig. 4).

Several non-isotopic approaches exist for measuring GHGs from agricultural soils; however, some of the problems with field GHG flux measurements are the large spatial variability of gas fluxes, and the high and often unpredictable temporal changes of fluxes. In addition, these non-isotopic techniques do not identify the sources (soil-N or fertilizer-N) nor the microbial process that generate N_2O emission in soil. Contrary to non-isotopic techniques, the use of ^{15}N stable isotope technologies offer the best choice for precise measurement of N_2O and N_2 and for identifying the exact sources of N_2O production.

Microbiologists have utilized the new possibilities that ^{15}N can offer, as it is now possible to quantify the turnover rates of individual processes in the N cycle based on dilution principles. Moreover, ^{15}N allowed for the first time the development of techniques to quantify the loss of N_2 against a huge atmospheric N_2 background. Also, the identification of which processes contribute to total N_2O emissions is unthinkable without the use of advanced ^{15}N tracing techniques (Müller et al., 2014). Various microbial processes (nitrification, denitrification, co-denitrification and organic N conversion to nitrite) produce N_2O . The two atoms of the N_2O (NO and N) molecule can be labeled with ^{15}N differently (i.e., alpha and beta positron). Therefore, the analysis of the two N atoms in ^{15}N labeled provides information on the exact microbial process that produces N_2O . This is commonly referred to the site preference (SP) which can be analyzed by modern analytical techniques (e.g., Isotope Ratio Mass spectrometers (IRMS) or Cavity Ring Down Spectroscopy (CRDS)) and therefore providing information on the origin of the N_2O molecules. There is a wealth of information we can obtain from using diverse isotopic approaches based on ^{15}N or ^{18}O labeling, but also based on natural abundance techniques that take advantage of the different metabolism with which for instance N_2O is produced. This, ultimately, provides us with the tools to identify emission pathways and in turn provide information on effective mitigation techniques.

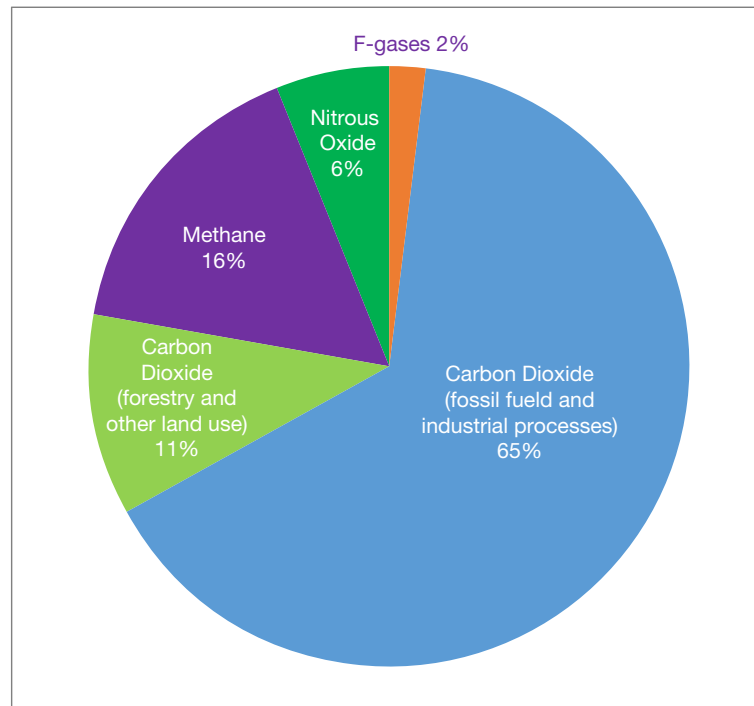


Fig. 3 Major greenhouse gas emissions and contribution by various sectors (IPCC, 2014).

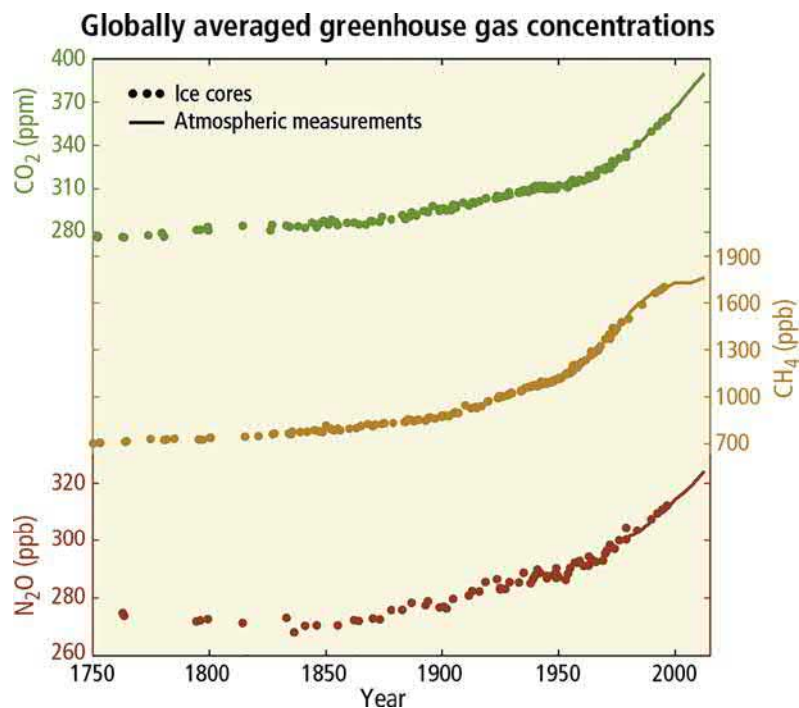


Fig. 4 Recent anthropogenic emissions of GHGs (IPCC, 2014).

Nuclear emergency and response affecting food and agriculture

Accidental releases of radionuclides to the environment (Miller, 2021) and their subsequent transfer to the food chain is a dynamic process that varies temporally and spatially across agricultural landscapes. Quickly identifying affected food production areas using up-to-date information of soil, water and plants are necessary for preventing potentially contaminated products from reaching consumers. This is especially the case for large-scale emergencies, whereby efficient management of big and diverse datasets determines the speed and quality of response.

Traditional processing of radioactive contamination data impacts both response times and accuracy as they are performed manually and not centralized. Developments in Information Technology-Decision Support Systems (IT-DSS) tools and algorithms allow for improved real-time management of large volumes of data and integrated decision-making support.

The modern IT-DSS provides clear visual aid for improving response capabilities. Examples include the potential to graphically represent the collection phase (maps showcasing status of sample collection or analysis), the analysis/validation phase (maps showcasing radioactivity concentration/deposition) and the decision-making phase (dashboard assisting in suggesting food restriction sites). Food contamination maps can be made available immediately, and the ability to visually assess real-time data helps weigh cost and benefits of possible response scenarios. These decision-making support functions increase the capacity of stakeholders to focus on the most important matters at hand—ensuring food and consumer safety. The IAEA recently developed an online Decision Support System for Nuclear Emergencies Affecting Food and Agriculture (DSS4NAFA) (Lee Zhi Yi and Dercon, 2019). DSS4NAFA is a cloud-based decision support system to manage large volumes of spatial and temporal data, real-time information processing and visualization, and enhanced aid to response actions and decision-making in case of a nuclear or radiological emergency affecting food and agriculture.

Future developments in innovative technologies used for response processes affecting food and agriculture include the use of artificial intelligence and machine learning to automate response processes and optimize resource-use efficiency.

Summary

Nuclear and isotope applications have played a key role in crop production over the last six decades in many countries, especially the Member States of the IAEA. Genetic diversity introduced through mutation breeding into more than 220 plant species using mostly physical mutagens (γ -rays) has led to the development of over 3000 improved crop varieties globally as recorded in the IAEA Mutant Variety Database. A majority of these are food crops contributing to food security of many developing nations. Harnessing the genetic potential of improved crop varieties require optimal agronomic practices including soil, irrigation water and nutrient management. Nuclear and isotope techniques have facilitated the determination of soil nutrient levels to plan optimal fertilizer rates, improve water and irrigation management, and the identification of the prevalence of soil erosion and redistribution to enable better landscape management. More recently these techniques are also being used for determining the rate and type of greenhouse gas emissions from agricultural soils, information that can be used to optimize farm management practices for reduced emissions. Finally, the Joint FAO/IAEA Division has also developed decision support tools to nuclear emergencies affecting food and agriculture.

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Agriculture: Improving Livestock Production

Gerrit J. Viljoen^a, Rui Pereira^b, Marc J.B. Vreysen^b, Giovanni Cattoli^a, and Mario Garcia Podesta^a, ^a Animal Production and Health Subprogramme, Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, Department of Nuclear Sciences and Applications, International Atomic Energy Agency, Vienna, Austria; and ^b Insect Pest Control Subprogramme, Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, Department of Nuclear Sciences and Applications, International Atomic Energy Agency, Vienna, Austria

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Abbreviations

AAT	African animal trypanosomosis
AI	Artificial insemination
AW-IPM	Area-Wide Integrated Pest Management
CBPP	Contagious bovine pleuropneumonia
DNA	Deoxyribonucleic acid
ELISA	Enzyme-linked immunosorbent assay
FAO	Food and Agriculture Organization of the United Nations
FMD	Foot-and-mouth disease
GIS	Geographic information systems
HAT	Human African trypanosomosis (sleeping sickness)
HPAI	Highly pathogenic avian influenza
IMP	Integrated pest management
ISO	International Organization for Standardization
LSD	Lumpy skin disease
MERS	Middle east respiratory syndrome
NIRS	Near-infrared spectroscopy
OIE	World Organisation for Animal Health (formerly Office International des Epizooties)
PAG	Pregnancy associated glycoproteins
PCA	Phased conditional approach
PCR	Polymerase chain reaction
PPR	Peste des Petits Ruminants (ovine rinderpest)
Purine	Type of chemical compound in food and feeds
RH	Radiation hybrid
RIA	Radioimmunoassay
RNA	Ribonucleic acid
Roughage	Plants or part of a plant rich in fiber
SI	Stable isotope

SIT Sterile insect technique
 SOP Standard operation procedure
 VETLAB Veterinary laboratory network
 WHO World Health Organization

Introduction

The global human population has almost doubled over the last 40 years, with the major proportion of the increase occurring in developing countries. Human population is expected to approach 10 billion by 2050 (United Nations, 2019). Today, one billion of the world's people face hunger and, therefore, are "food insecure" (FAO, 2019). Governments, international organizations, research centers and non-government organizations (NGOs) are placing individual and coordinated efforts to overcome this overwhelming situation (Nissanke and Thorbecke, 2007), for example, the 2030 Agenda for Sustainable Development Goals (FAO, 2018).

Livestock and crop production are major elements of support to the economy of most developing countries and provide livelihood for a significant proportion of the population. However, this production is highly dependent on the rural population. According to FAO, nearly 25% of the gross domestic product (GDP) derives from agriculture in many of the least developed countries. Furthermore, farmers in Latin America generate more agriculture value added than in other developing regions.¹

The role of livestock in developing countries is complex and significantly different from the situation in industrialized countries. Livestock are multi-purpose and contribute in many other ways to the socioeconomic welfare of people. Animals are valued for their hides and skins, and they provide manure, which helps to replenish soil fertility. For many rural farmers, livestock are a "living bank" and serve as a financial reserve during periods of economic distress. The development of the livestock sector would not only help developing countries to achieve self-sufficiency in animal products, but also prevent the migration of rural poor to urban areas in search of supposed economic benefits.

Nuclear and nuclear-related techniques

Nuclear applications provide added value to conventional approaches in addressing a range of agricultural problems and issues, including crop and animal production, animal disease diagnosis and control, insect pest control, food safety and sustainable use of natural resources. Nuclear and nuclear-related technologies are useful tools in research and development. They provide the basis to formulate appropriate management strategies for improving livestock productivity (Makkar, 2008; Odongo et al. 2010; Viljoen and Luckins, 2012).

Animal nutrition

In developing countries, livestock are fed mainly on low quality roughages, including natural grazing and agro-industrial by-products, which usually are seasonally dependent (Galyean and Hubbert, 2014). Research done using ¹⁴C, ³²P and ¹⁵N labeled material to measure the efficiency of rumen fermentation (chewing the cud) in cattle and buffalos fed the most commonly available feeds, along with data obtained from conventional methods (e.g. digestibility, rate of passage and feed intake) has allowed for the establishment of optimal combinations of feeds for use as supplements to available pasture and grazing (Leng and Nolan, 1984; Cantalapiedra-Hijar et al., 2015) (Fig. 1).

In animal nutrition research and feeding practices, several stable isotope and other radio-isotope techniques have definite benefits over conventional techniques: (1) carbon tracers (¹⁴C) can be used to develop models for the description of relationships between purine absorption and purine derivative excretion in urine (Liang et al., 2004), (2) infusion of ¹⁴C labeled acetic and propionic acid are being used to estimate volatile fatty acid production rates (Leng, 1970), (3) nitrogen (¹⁵N) labeled ammonium bicarbonate and ammonium chloride can be used to study microbial degradation of poor quality fibers, microbial mass, utilization of non-protein nitrogen, urea recycling, microbial protein synthesis and amino acid interconversions in rumen (Cantalapiedra-Hijar et al., 2015), (4) the rate of microbial protein synthesis is determined by phosphorus (³²P, ³³P), ¹⁴N or ³⁵P incorporation in rumen microorganisms (Kaswari, 2004), (5) cobalt (⁵⁸Co), chromium (⁵¹Cr) labeled forages (consumed plant materials) are used for passage rate determination (Uden et al., 1980), and (6) ¹⁴C/¹³C labeled sodium bicarbonate infusion techniques are used for estimating carbon dioxide production in the rumen (Duin et al., 2016). Additionally, methane emission by ruminants can be estimated by isotope dilution using either tritium (³H) or ¹⁴C labeled methane (Follet et al., 2001).

In pasture-based production systems, animals are highly selective of plants and parts of the plant that they eat and therefore the consumption and nutritional value of the total intake is hard to determine with any reasonable precision. However, conventional

¹<http://www.fao.org/3/i2490e/i2490e01c.pdf>.

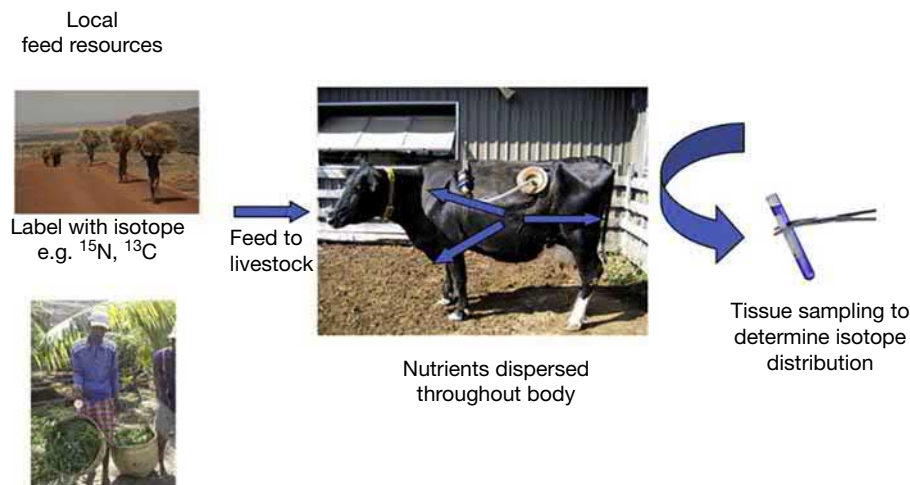


Fig. 1 Evaluation of nutritional value and efficient use of locally available feeds.

chemistry analysis and stable carbon isotope composition ($\delta^{13}\text{C}$ of n-alkanes)² can provide reliable data related to the nutritional value of those feeds, dry matter intake and plant proportions (Dove and Mayes, 2006). These data, in conjunction with information provided by the near infrared reflectance spectroscopy (NIRS), can allow the design of diets and supplements required to improve animal productivity under grazing conditions.

Animal reproduction

Low fertility is often identified as one of the primary constraints to livestock production. The monitoring of reproductive hormones, especially progesterone, through radio-isotopic techniques such as radioimmunoassay³ (RIA) has provided enormous advantages for a better understanding of the reproductive physiology of livestock species, for monitoring ovarian cyclicity, identifying anoestrus (animals without signs of oestrus for long periods) and non-pregnant females, and in assisting artificial insemination (AI) centers and AI service providers on management and detecting human errors in AI (Fig. 2) (Garcia and Edqvist, 1990; Rajamahendran et al., 1993; Garcia et al., 2001). Progesterone RIA can also be used for detecting non-pregnancies in cows as progesterone values are supposed to be very low or not detectable at 22–24 days of the oestrus (day of service).

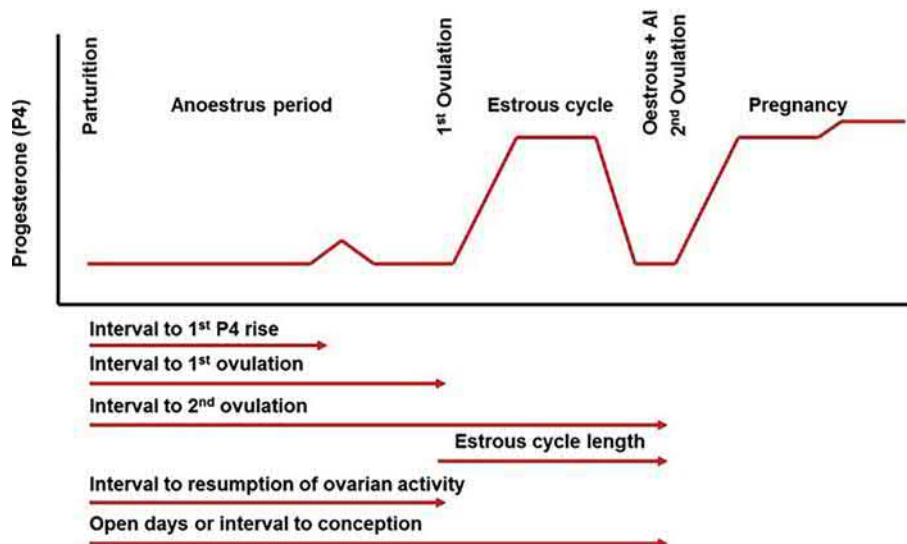


Fig. 2 Monitoring post-partum ovarian activity through radioimmunoassay (RIA) progesterone analysis.

²Isotopic signature, a measure of the ratio of stable isotopes ^{13}C : ^{12}C .

³Nuclear technique based on a competitive binding, where a radioactive antigen (in this case a radiolabelled progesterone) competes with a non-radioactive antigen (the progesterone hormone in the milk or blood sample) for binding sites in specific antibodies (e.g. against progesterone). The radioactivity of the reaction is measured in a gamma counter.

Novel uses of nuclear techniques for research purposes involves the identification of potential candidate protein biomarkers for early pregnancy detection by day 16 after AI. To identify these biomarkers, embryos are being experimentally cultured in vitro with radioactive labeled amino acids (e.g. ^{35}S), which will then be integrated into proteins synthesized by the conceptus. This approach offers the possibility of tracking de novo protein synthesis, and consequently proteins produced by the conceptus that could be used as early pregnancy biomarkers. These proteins, once identified, could be detected in maternal blood by RIA or other nuclear-related technique. RIA is already being used for determination of Pregnancy Associated Glycoproteins (PAG) concentrations in blood and in milk on days 35–42 after service (Ayad et al., 2007); and changes in micro-RNA in the bloodstream and milk may identify pregnancy in cattle as early as 24 days after service (Ioannidis and Donadeu, 2016).

Animal genetics

Genetic improvement of farm animals for productivity of milk, meat and fiber is traditionally based on identifying the high-performance animals. With the recent advances in nuclear derived genomic technologies, it is now possible to speed up the process of genetic improvement, by estimating breeding value of animals as soon as they are born, using the variations in an array of DNA markers spread across its genome (Hayes et al., 2009). These DNA markers have been identified through a range of nuclear-related approaches. Radiation hybrid (RH) mapping is a method for producing high resolution maps that can be used for integrating linkage maps and can serve as a link across species for comparative mapping (Perelman et al., 2018). Genome assemblies of major livestock species and humans have been fully developed, and other important farm species like the camel and the alpaca are in the process (Fig. 3).

Nuclear, nuclear-derived, and molecular techniques for genetic characterization of indigenous and/or locally adapted animal species, populations and breeds, and for the identification of genes and markers related to productive (i.e. milk, meat, wool and fiber) traits and those related to functionality, such as resistance to diseases (e.g. gastrointestinal parasitism in ruminants), are becoming important tools in understanding biology and enhancing genetic selection programs in cattle and small ruminants (sheep and goats). Genomic data of performance-recorded animals enable breeders and farmers to relate production traits with not only parentage, but also genetic admixture of animals when animals are a hybrid of multiple breeds, which is commonly the case in peri-urban dairy production systems. Such analyses can facilitate the identification of superior (both genotypically and phenotypically) crossbred genotypes for subsequent mating with artificial insemination.

Animal health

Nuclear and nuclear-related technologies have played an important role in animal health, particularly in relation to disease diagnosis and characterization of pathogenic organisms. Although many of the tests using radioisotope labels have been replaced, the new tests that have been developed rely in their inception on the radiolabeling methods that preceded them, so they are still very much nuclear-related (Viljoen and Luckins, 2012).

Vaccines

The control and prevention of diseases are of utmost importance, and vaccines play major roles. Vaccination is a cost-effective way of controlling animal diseases; however, there is a large variation in their efficiency, immunogenicity, shelf-life and storage requirements (Centers for Disease Control and Prevention, 2015).

Gamma irradiation of pathogens seemed to be a viable alternative nearly 40–50 years ago, but due to limitations in culture techniques, lack of molecular understanding of the genetics involved and limited irradiation sources avoided further research on this (Bain, 1999). Recent findings show that irradiation affects the ability of pathogens to replicate; therefore, producing sustainable immunity but preventing the occurrence of infection in the host. Current research using ^{60}Co and X-rays is showing flattering laboratory results for irradiated vaccines against some bacteria (Ahmed et al., 2015), parasites (*Haemonchus contortus*, *Salmonella gallinarum*) (Ammar et al., 2014) and viruses (avian influenza) (David et al., 2017).

Stable isotopes

The emergence of highly pathogenic strains of Influenza A virus (AIVs) in birds has been of considerable concern over the last few years as AIVs threaten animal health, livestock productivity and food security, and they can also evolve into dangerous human pathogens. In cases of outbreaks, millions of chickens are killed and many more are destroyed to control the spread of the disease affecting the availability of poultry meat in the country or region. Tracing the movements of migratory birds in relation to where they originated as well as their stopover points in their migration between breeding and non-breeding grounds render a better understanding of their role in spreading AIVs and facilitate its control (Fig. 4). A suitable technique to trace migrants is based on stable isotope (SI) signatures of tissues of birds, e.g. deuterium, especially those in feathers, as there is a strong correlation between levels of these SIs in the environment and the concentration of the same SIs in avian tissues (Meier et al., 2017). In order to improve the resolution of the migration patterns through spatial isoscapes,⁴ other isotopes may be included such as carbon (^{13}C) nitrogen (^{15}N), oxygen (^{17}O or ^{18}O) or sulfur (^{33}S , ^{34}S or ^{36}S) (Glew et al., 2018).

⁴The word isoscape is derived from isotope landscape. The IAEA defines the isoscape as “gridded isotopic landscape.” Please see http://www.naweb.iaea.org/naweb/ih/IHS_resources_rcwip.html.

Goat Radiation Hybrid Map Development

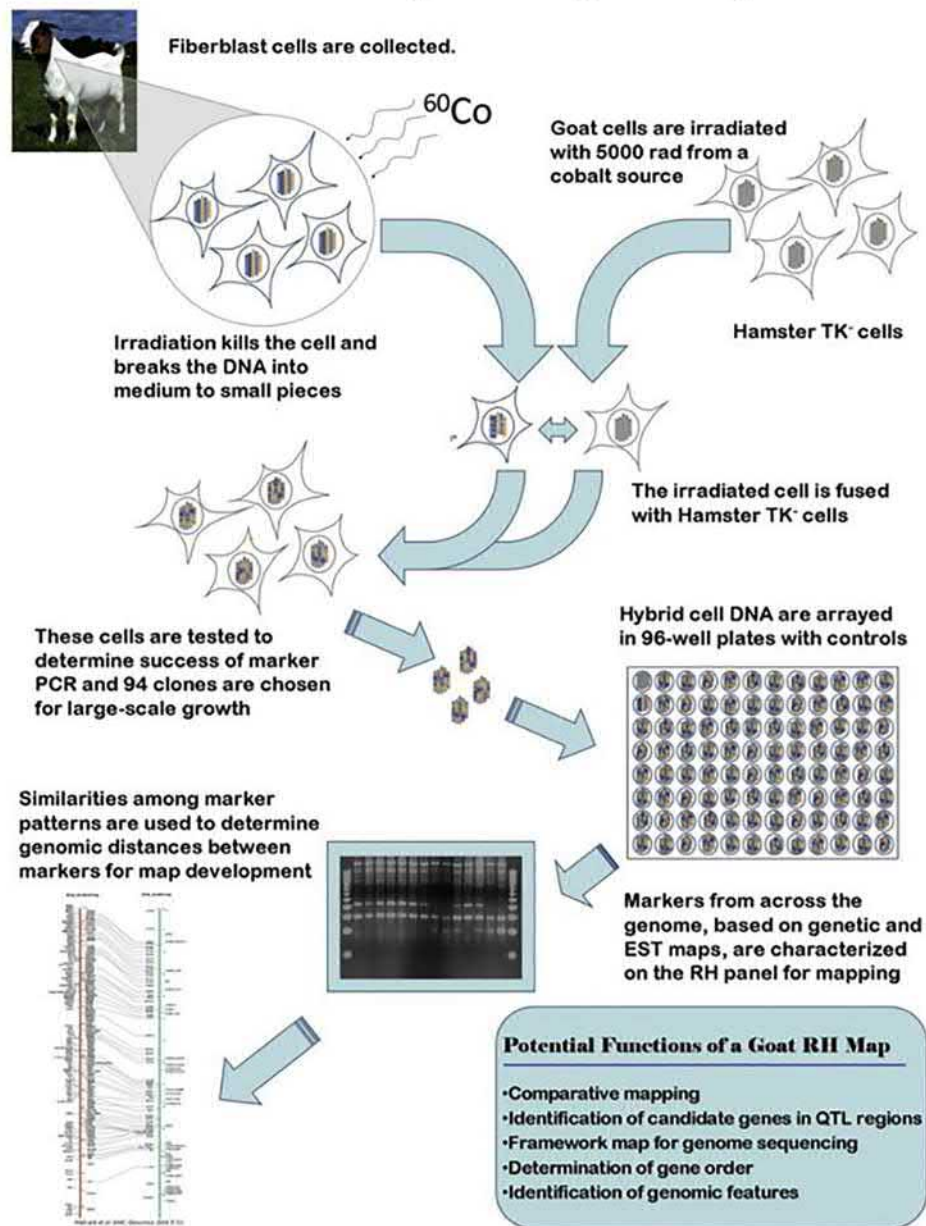


Fig. 3 Schematic overview of goat radiation hybrid map development.

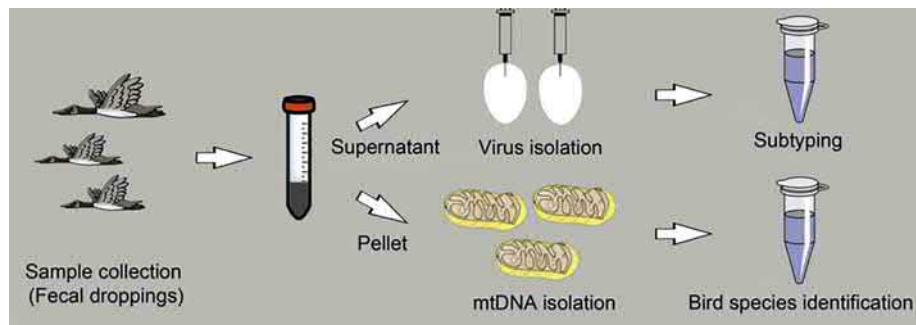


Fig. 4 Diagram showing the relationship of the stable isotope's role on the identification of migratory paths of wild waterfowl birds and detection of avian influenza virus.

Disease diagnosis

Isotopic techniques that included the use of ^{75}Se , ^{32}P , ^{125}I , and ^{35}S isotopes have enabled high levels of sensitivity and specificity for the detection of antibodies against animal pathogens (Rodák et al., 1981). The present-day extensive use of enzyme immunoassays (ELISA) and molecular methods (e.g. PCR) for diagnosis and characterization of animal pathogens has its origins in the use of isotope-labeled antigens and antibodies. ELISA is a technology used for detection of the antigen (pathogen) or antibodies induced by specific pathogens in the body of the host, where the isotope has been replaced by an enzyme⁵ that can be measured through color changes in the assay (Wright, 1995; Ma et al., 2011; Viljoen and Luckins, 2012). The PCR and the cyclic sequencing have their origin in purely nuclear techniques, where individual nucleic bases were labeled using radioactive isotopes, followed by transferring the signal from agarose gel⁶ onto a photographic film (Pestana et al., 2010). Although in many cases, due to safety reasons, the isotopic labeling has been replaced by enzymes (ELISA) or fluorophores (PCR), it has still major advantages over alternative labeling systems, as it does not affect the chemistry of the reactions and is, therefore, considered as the reference standard.

Nuclear and nuclear-related techniques, including immunological and molecular techniques, are highly valuable for control and prevention of major emerging transboundary animals and those of veterinary public health importance (Viljoen and Luckins, 2012). Main livestock and poultry diseases of interest in developing countries, some of them of zoonotic nature (affecting humans) include the highly pathogenic avian influenza (HPAI), foot-and-mouth disease (FMD), peste des petits ruminants (PPR), Rift Valley fever, brucellosis, contagious bovine pleuropneumonia (CBPP), Middle East respiratory syndrome (MERS), Newcastle disease, classical/African swine fever (CSF/ASF), lumpy skin disease (LSD), bluetongue, pox viruses and trypanosomiasis, as well as gastrointestinal parasites (i.e. *Haemonchus* sp. and *Fasciola* spp.). Moreover, several multi-pathogen detection systems for cost-effective syndromic surveillance and rapid differential diagnosis have been or are being developed and validated (Settypalli et al., 2016). Considering the variety of pathogens, differences among tests and chemicals involved, as well as the ample differences in expertise and infrastructure among laboratories, standardization of the methods is of great importance. The use of standard operation procedures (SOPs) and, if possible, the ISO accreditation of national disease diagnostic laboratories to build self-confidence on sample analysis and reporting is strongly recommended. The Veterinary Laboratory (VETLAB) Network, driven and supported by the IAEA involves veterinary diagnostic laboratories from 49 African and 19 Asian countries and it is expanding to other continents to facilitate sharing of expertise, data, information and material, development and harmonization of diagnostic methodologies, and supporting access to sequencing services and for international accreditation (<https://www.iaea.org/services/networks/vetlab>).

As an example of success can be mentioned the eradication of rinderpest (the cattle plague), where the ELISA kits were of high value for the monitoring of animals vaccinated and those carrying the disease. Rinderpest was for centuries a major cause of famine and poverty. In 2011 the FAO and OIE declared the disease officially eradicated from the world. This great achievement comes 30 years after the WHO declared that smallpox was eradicated, the only other viral disease to have been eliminated from the world. The estimated net annual economic benefit for the African region is at least US\$ 1 billion per year.

Area-wide integrated pest management (AW-IPM)

The publication in 1962 of Rachel Carson's "Silent spring" (Carson, 1962) was the start of growing concerns with respect to the massive use of broad-spectrum insecticides for the management of insect pests. These alarms have in the past decades resulted in increased demands for alternative methods of insect pest control that are not only effective but friendly or neutral to the environment. The concept of integrated pest management (IPM) offers a strategic approach to solving pest problems in an ecosystem context while taking into consideration any possible negative impact on human health and the environment (Brader, 1979). The IPM has remained the dominant paradigm of pest control in the last decades (Hendrichs et al., 2007). Against livestock pests, IPM programs can be implemented on a localized, herd-by-herd or field-by-field basis or through the management of entire insect populations in a delimited geographical area, i.e. area-wide IPM (Hendrichs et al., 2007; Vreysen et al., 2007; Klassen and Vreysen, 2020).

In a localized approach, individual livestock producers or farmers manage insect pests independently, without investing efforts to coordinate, collaborate or having to obtain the consent of their neighbors. The approach ignores pest individuals that frequently migrate into the treated area from infested areas in the untreated surroundings (Hendrichs et al., 2007). Localized pest control does not require a long-term commitment to a well-funded centralized management structure but relies on re-active interventions triggered when a pest population reaches a certain threshold and mostly rely on insecticide applications (Klassen and Vreysen, 2020).

Using simple mathematical population models, Knipling (1972, 1979) showed that small fractions (below 10%) of an insect pest population left uncontrolled can drastically nullify the benefits of suppressing a pest population in a large area (Fig. 5). These small foci or refugia of relic insect populations constitute permanent and prolific sources of immigrants into the suppressed or eradicated areas. The central paradigm of area-wide control or total population management recognizes this need to address all foci/refugia from which recruits can invade suppressed or pest-free areas (Bouyer and Vreysen, 2018). The approach also takes into consideration the movement of livestock as a host of the pest.

⁵ELISA has a similar principle than radioimmunoassay (RIA). In this case, an enzyme is used instead of radiolabelled isotope for the binding competition and the gamma counter is replaced by an ELISA reader where the change in color depending of the amount of substrate (e.g. antibodies against specific pathogen) is measured.

⁶Agarose is a polysaccharide. Agarose gel is a three-dimensional matrix formed of helical agarose that is used to separate DNA fragments according to their size.

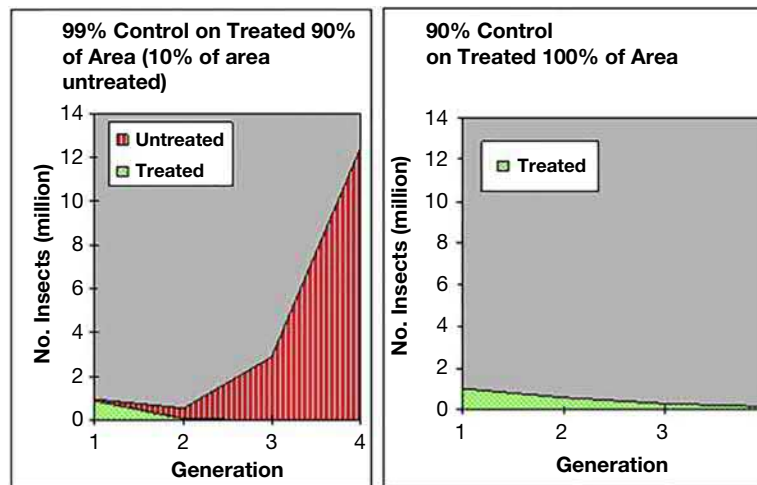


Fig. 5 Results of a model that shows the outcome of neglecting to suppress a small fraction of a pest population in an agroecosystem versus the effect of uniformly suppressing the entire pest population. Left: 10% of the population is untreated, and in four generations it produces a large number of individuals, while the 90% of the population that is treated declines. Right: Entire pest population in the agroecosystem is suppressed uniformly, and its numbers decline from generation to generation. Adapted from Knipling EF (1972) Entomology and the management of man's environment. *Austral Entomology* 11: 153–167.

Area-wide Integrated Pest Management (AW-IPM) efforts (1) focus on preventive pest management where action is taken before the pest population reaches damaging levels (preventive approach), (2) address the entire pest population throughout the ecosystem, (3) require several years of planning and an organization dedicated exclusively to the control effort, and (4) use advanced technologies such as geographic information systems (GIS), global positioning system, remote sensing and aerial release techniques (Klassen and Vreysen, 2020).

In some programs, political or financial pressures have sometimes resulted in starting an operational program without detailed knowledge about the biology, ecology and population dynamics of the targeted pest population. To avoid embarking on operational activities without sound baseline data, a phased conditional approach (PCA) should be followed (Bouyer et al., 2020). This requires that progressing to the next phase is conditional upon completion of all or most of the activities in the previous phase. Although the different phases of the PCA can vary with the target species, for tsetse flies the phases are: (1) training and commitment of all stakeholders, (2) baseline data collection and feasibility studies, (3) pre-operational activities and (4) operational activities (Vreysen et al., 2020).

The sterile insect technique (SIT)

In the 1930s, Dr. E. F. Knipling of the USA conceived the concept of managing economically important insect pests by manipulating their reproductive capacity. Knipling speculated that, if an efficient method could be found to sterilize insects, the release of large numbers of sterile males might be used to introduce sterility in the native insect pest population. Initial laboratory experiments in the 1950s showed that male and female New World screwworm flies, *Cochliomyia hominivorax* (Coquerel), could be sexually sterilized by exposure to X-rays (Bushland and Hopkins, 1951) or gamma rays (Bushland and Hopkins, 1953) without impairing their survival and field competitiveness. The technique would require the mass-production of the target insect in large factories, the sterilization of the male sex, and the sustained release of the sterile males in the target area. Mating of a sterile male with a virgin female would result in no offspring. The large numbers of sterile male insects would limit the reproduction of the natural population in proportion to the ratio of sterile to fertile insects. The sustained release of sterile insects would, with each generation result in increased levels of sterility in the target population, which could eventually culminate in the elimination of the target population.

The SIT acts inverse-density dependent, which is one of its main advantages, i.e. the control effort becomes more economical and efficient as the natural population declines and increasing ratios of sterile to wild males are achieved (Dame, 1970). This is in contrast with chemical methods of killing insects that are very effective at high density pest populations, but the continued use of the same treatment will result in the same percentage effect regardless of population density (LaChance et al., 1967) (Fig. 6).

Insect pest control programs that rely on the introduction of sterility have been based mainly on induced dominant lethal mutations (LaChance et al., 1967). A dominant lethal mutation is a nuclear change, e.g. chromosome breaks in the germ cells without affecting their maturation but killing the zygote during its development (Von Borstel, 1962). During cell division, chromosome fragments are lost because they lack a centromere,⁷ which is responsible for inclusion in the daughter nucleus. This leads to

⁷A specialized structure on the chromosome, appearing during cell division as the constricted central region where the two chromatids are held together and form an X shape.

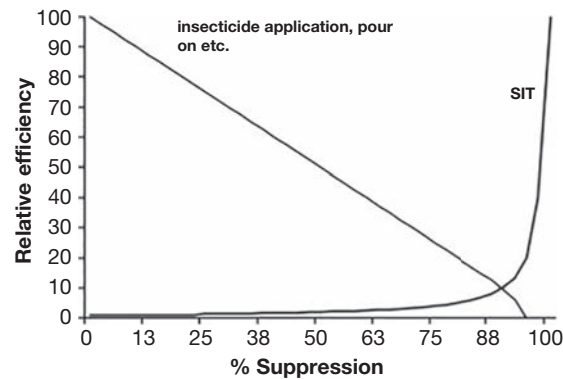


Fig. 6 Schematic representation of the sequential use of methods of suppression that efficiently decimate a pest population followed by use of the SIT, which becomes progressively more suppressive as the population declines. The term “efficiency” refers to the ease in further suppressing the population, and not to the cost of eliminating an individual pest (cost per kill). Adapted from Feldmann U and Hendrichs J (2001) *Integrating the Sterile Insect Technique as a Key Component of Area-wide Tsetse and Trypanosomiasis Intervention*. PAAT (Programme Against African Trypanosomiasis) technical and scientific series, no. 3. Rome, Italy: FAO. Available from <http://www.fao.org/DOCREP/004/Y2022E/Y2022E00.HTM>.

chromosome imbalance with the formation of dicentric chromosomes and breakage-fusion-bridge cycles during cleavage divisions in the zygote. Death usually occurs during the early divisions before the formation of the blastoderm (LaChance et al., 1967). In 95% of the eggs fertilized by sperm of irradiated male *Glossina palpalis palpalis* Robineau-Desvoidy, embryonic death occurred in the early cleavage division stages. This was then followed by lytic processes and the extrusion of the dead eggs 1–2 days after ovulation (Matolin and Van der Vloedt, 1982).

The SIT has several advantages: (1) it is non-intrusive to the environment with no adverse effects on non-target organisms, (2) it is species-specific, (3) it can easily be combined with other biological control methods such as parasitoids, predators and pathogens, (4) there is no risk of the development of resistance to the effects of the sterile males (provided that adequate quality assurance is practiced in the production process), and (5) the sterile insects cannot get established in the released areas as with other biological control programs (Vreysen, 2001).

The SIT also has its limitations: (1) the release of sterile insects is only effective and cost efficient when the target population density is low, (2) it requires detailed knowledge of the biology and ecology of the target pest, (3) the insect should be amenable to mass-rearing at a reasonable cost, (4) there is a delayed effect as the target insect is not killed but sterility introduced, and (5) the technique can only be used for those insects where the development phase that is used for the release (e.g. in most cases the adult insect or male mosquitoes that contrary to females do not transmit diseases) is not contributing to the damage. In addition, the SIT necessitates efficient release and monitoring methods, which must be applied on an area-wide basis (Vreysen et al., 2020).

The AW-IPM with a SIT component: Examples of successful implementation

Tsetse flies

Tsetse flies are the cyclical vectors of trypanosomes, protozoa that cause a disease occurring mostly in rural areas of sub-Saharan Africa where it gravely affects agro-pastoral activities. The fly infests an estimated area of 8.7 million km² where it transmits human African trypanosomiasis (HAT) or “sleeping sickness” in humans and African animal trypanosomiasis (AAT) or “nagana” in livestock. Besides the human suffering and death from sleeping sickness in certain areas, livestock production in rural areas is severely constrained by nagana (Feldmann and Hendrichs, 2001).

The first SIT-based operational program of significant size was carried out in the 1980s in the Tanga area of the United Republic of Tanzania. Two aerial applications of endosulfan⁸ (Williamson et al., 1983a) were used to suppress the targeted *Glossina morsitans* Westwood population. This was followed by the release of sterile male pupae⁹ for 15 months at a release density of 135 sterile males/km² resulting in an average sterile to wild male ratio of 1.12:1. This was too low to eradicate the population but low enough to keep the indigenous population at the 80–95% reduction level obtained after the insecticide applications (Williamson et al., 1983b).

Populations of the two tsetse fly species *Glossina palpalis gambiensis* Vanderplank and *G. tachinoides* Westwood were eradicated from 3500 km² of Guinea savannah in Sidéradougou, Burkina Faso in the 1980s. A combination of trapping techniques and insecticide-impregnated targets reduced the tsetse populations by 91–94%. Thereafter, sterile males were released at a density of 20–35 sterile males per linear km obtaining the desired sterile to wild male ratio of 10:1. The main river systems were free of tsetse flies 6 months after the start of the sterile male fly releases (Politzer and Cuisance, 1984).

⁸Chlorinated hydrocarbon insecticide of the cyclodiene subgroup, which acts as a contact poison in a wide variety of insects and mites.

⁹Complete, or holometabolous, metamorphosis is characteristic of beetles, butterflies and moths, flies, and wasps. Their life cycle includes four stages: egg, larva, pupa, and adult.

A similar approach was used to eradicate a *G. p. palpalis* population from 1500 km² of agro-pastoral land in Southern Plateau State of Nigeria. Insecticide-impregnated targets and biconical traps achieved 97–99% reduction of the fly population and this was followed by weekly releases of sterile males. In 1988, the fly was declared eradicated from the entire target area. The program demonstrated that a sterile to wild male ratio of 10:1 was a prerequisite to achieve eradication and that different habitats and different species demanded different release rates and different time frames (Oladunmade et al., 1990). Despite the successful eradication of the local tsetse populations in Burkina Faso and Nigeria, the tsetse-free status of the target zones could not be sustained, due to invasion pressure from the adjacent untreated areas. These examples exemplify the need for an area-wide approach to achieve sustainability.

The program against a *Glossina austeni* Newstead population on Unguja Island of Zanzibar (1994–97) was the first SIT-based eradication effort against a tsetse species that was implemented following AW-IPM principles. The fly population was suppressed using the live-bait technique in the agricultural areas of the northern half of the island (that had high cattle densities) and by deploying insecticide-treated blue cotton targets in the dense forested areas mainly in the south. This was followed by the aerial dispersal of sterile male flies over the entire surface area of Unguja Island, including the small offshore islands (Vreysen et al., 2000). The last indigenous *G. austeni* fly was trapped in September 1996 and Unguja Island has remained free of tsetse and AAT ever since.

In 2005, the Government of Senegal launched an AW-IPM project that aimed to create a sustainable *G. p. gambiensis* free zone in the Niayes, an area located north-east of the capital Dakar. Results of the initial phase of data collection, showed that there was only one species in the area with a limited distribution of 1000 km² (Bouyer et al., 2010), and the population was genetically isolated from the remainder of the tsetse belt (Solano et al., 2010). The impact of the removal of AAT was found to be significant with annual benefits of € 2.8 million for farmers in the tsetse-infested zone (Bouyer et al., 2014). The adopted AW-IPM strategy included suppression activities with insecticide-impregnated traps and the use of pour-on for cattle, followed by the release of sterile males to eliminate the remaining relic pockets.

New World screwworm

The New World screwworm is one of the most damaging insect pests of livestock in the tropical and subtropical areas of the western hemisphere. In 1934, 1.3 million cattle died in the south-eastern USA because of New World screwworm infestations (Dove, 1937). Between 1935 and 1937, intensive animal inspections combined with insecticidal wound treatment and good animal husbandry practices greatly reduced the incidence of wound infections, but eradication of the New World could not be achieved because wild-life and non-attended livestock served as reservoirs (Novy, 1991).

The huge economic losses result from direct reduction in productivity due to sickness and death, and from the labor and insecticide costs that are incurred from wound inspections and treatments. Gravid female screwworm flies are attracted to wounds and lay up to 400 eggs in or around the edges of these wounds. The eggs hatch and the larvae feed on the living flesh, making the wound larger and even more attractive for other females to oviposit. The larvae thrive on living warm blooded animals (including humans) and multiple infestations can kill a large cow if left untreated.

A first pilot trial using the release of sterile male New World screwworm was attempted on the Island of Curacao (Baumhover et al., 1955). Sterile males were released at a density of 300 flies per km² and eradication was achieved after only 6 months. This program led to the largest and most successful AW-IPM campaign with a SIT component ever undertaken against a major insect pest, i.e. the eradication of the New World screwworm from the southern USA, Puerto Rico, Mexico (Snow, 1988), Central America, and Panama (Wyss, 2000).

The economic benefits of this eradication program have been enormous. Although the total investment for the program has been estimated at US\$ 1300 million (over a period of almost 50 years), the estimated direct annual producer benefits are substantial and estimated at US\$ 896 million, US\$ 328 million and US\$ 87 million for the United States, Mexico and Central America, respectively (Wyss, 2000). The annual benefits for the livestock producers are, therefore, the same as the total project cost over 50 years.

In 1988, the New World screwworm was detected in the north-western part of the Libyan Arab Jamahiriya. This incursion originated most likely from the Americas although the mode of introduction remains unknown. By the end of 1990, an area of 25,000 km² was infested, which prompted the mounting of an internationally supported eradication campaign (FAO, 1992). An AW-IPM strategy to eradicate the screwworm from the infested area and to protect neighboring countries was developed that had the following components: intensive animal surveillance, insecticidal wound treatment and the sequential release of sterile insects. The sterile insects were produced at the mass-rearing facility in Tuxtla Gutiérrez, Chiapas, Mexico, and were transported by air to Libya. In February 1991, the full eradication phase was initiated with the release of 28 million sterile flies per week. By May 1991, this number was increased to 40 million/week, treating a total area of 40,000 km². Between December 1990 and October 1991, a total of 1300 million sterile New World screwworm flies were released, and in 1991 the area was declared free of screwworm (Lindquist and Abusowa, 1992).

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- <https://www.iaea.org/newscenter/multimedia/videos/using-nuclear-science-to-improve-animal-breeding>—Using Nuclear Science to Improve Animal Breeding. Now, with advances in genomics, it has become possible to estimate the breeding potential of an animal on the day of its birth, simply by looking at its DNA or genome map.
- <https://www.youtube.com/watch?v=KF0xjLUdDFU>—Using Nuclear Science in the Control of Avian Influenza. Outbreaks of avian influenza have led to the death or culling of millions of poultry and pigs and untold millions of dollars in losses to farmers and producers. Avian influenza is also a deadly disease that affects humans. Through the use of isotopic techniques to track migrating birds, and molecular techniques to detect the presence of viruses, researchers and veterinary services will have a head start in predicting sites at higher risk of outbreaks.
- <https://www.iaea.org/newscenter/news/tsetse-free-for-20-years-thanks-to-a-nuclear-technique-the-island-of-unguja-zanzibar>—The programme against a *Glossina austeni* Newstead population on Unguja Island of Zanzibar (1994–1997) was the first SIT-based eradication effort against a tsetse species that was implemented following AW-IPM principles. The last indigenous *G. austeni* fly was trapped in September 1996 and Unguja Island has remained free of tsetse and AAT ever since.
- <https://www.iaea.org/newscenter/news/productive-cattle-breeds-thrive-in-senegal-niayes-region-as-a-result-of-tsetse-suppression>—Productive Cattle Breeds Thrive in Senegal's Niayes Region as a Result of Tsetse Suppression. In Senegal, as in many other parts of West Africa, African animal trypanosomosis (AAT), a deadly disease carried by the tsetse fly, has long been a major obstacle to the development of more efficient and sustainable livestock production systems.

Agriculture: Electron Beam Irradiation Technology Applications in the Food Industry

Suresh D. Pillai^a and Eric T. Pillai^{b,*}, ^a National Center for Electron Beam Research, Department of Food Science & Technology, Texas A&M University, College Station, TX, United States; and ^b Department of Industrial Engineering and Operations Research, University of California, Berkeley, CA, United States

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Glossary

Aseptic Packaging Packaging foods in a sterile environment.

Becquerel One disintegration per second.

Codex Alimentarius International regulation for transboundary shipment of irradiated goods.

Curie 3.7×10^{10} disintegrations per second.

eBeam energy The energy of the electrons (which dictates the penetration ability).

eBeam power The processing speed of the eBeam system (which translates into the volume of food that can be processed per unit time).

EFSA European Food Safety Authority.

Electron beam technology (eBeam technology) Ionizing radiation technology that utilizes commercial electricity to generate energetic electrons to create ionization events.

FD&C Food, Drug, and Cosmetic Act.

FDA Food and Drug Administration (US).

FSMA Food Safety Modernization Act.

FSSAI Food Safety and Standards Authority of India.

Gamma radiation Ionizing electromagnetic radiation. In the context of this chapter is the radioactive decay of the isotope cobalt-60 that generates a stream of energetic photons.

Gray (Gy) One joule of radiation energy absorbed per kilogram of matter.

HACCP Hazard Analysis and Critical Control Point.

HEEB High energy electron beam irradiation technology that utilizes electrons that have 10 MeV (million electron volts) energy.

IAEA International Atomic Energy Agency.

*Currently at SpaceX, Hawthorne, CA.

Ionizing radiation Electromagnetic (or other type of nuclear) radiation that has sufficient energy to cause ionization events in atoms it encounters.

IPPC International Plant Protection Convention.

LEEB Low energy electron beam irradiation technology utilizing electrons that have energy ranging below 1 MeV.

LEEX Low energy X-ray irradiation technology that utilizes low energy X-ray radiation below 300 keV (kilo electron volts).

Linac Linear accelerator, a type of eBeam generating equipment.

NNSA National Nuclear Security Administration (US).

Phytosanitary Treatment The treatment of imported agricultural products to eliminate agriculturally harmful insects and pests.

USDA-APHIS US Department of Agriculture-Animal and Plant Health Inspection Service.

WTO World Trade Association.

Introduction

Professor Norman Borlaug, winner of the 1970 Nobel Peace Prize once said, “You can’t build a peaceful world on empty stomachs and human misery.” This prophetic saying is extremely relevant today, if not more so than 5 decades ago. Despite dramatic improvements in global food supply chains since the 70s, over a billion people (~14% of the global population) still go to bed hungry every night. While food insecurity stems from a wide combination of social, political, and economic factors, technological limitations also drive food insecurity. Among the most significant of these technical factors are spoilage of food during transport, food waste resulting from improper storage, and food contamination with microbial pathogens. These factors combine to hinder efforts at food relief and result in a tragic situation where logistical realities prevent food from reaching places where it’s needed the most.

Few technologies in the food industry, except perhaps genetically modified crops, have received more scrutiny and been the subject of as much controversy as food irradiation. The British scientists Appleby and Banks were awarded their patent on food irradiation over 115 years ago, in 1905. Their invention was intended to bring about an improvement in the condition of foodstuffs and their general keeping quality. This technology, however, has moved beyond just improving the quality and safety of foods. It is now being commercially used for improved food packaging, improved seed germination concepts, and improved seedling emergence as well (Pillai and Shayanfar, 2015a, b; Hertwig et al., 2018; Beyaz et al., 2016).

The globalization of foods has brought along challenges related to the spread of microbial pathogens via foods, the potential spread of invasive insects and pests, and the potential for significant losses during transportation across continents. Microbial pathogens continue to be a major issue even in developed countries. In the United States for example, the Centers for Disease Control and Prevention estimate that as many as 48 million people get sick and about 3000 die each year from microbially contaminated foods. The situation in other parts of the world may be significantly more dire. However, in a majority of countries, disease surveillance is not as developed and, therefore, the exact magnitude of foodborne illnesses is never known. Ionizing radiation technology is a proven technology to reduce or eliminate foodborne pathogens. Food irradiation for food safety applications is in commercial use in over 40 countries, with China leading in the volume of food that is irradiated. In addition to the potential spread of the pathogens via foods, ionizing radiation is growing in popularity for use as a technology for the phytosanitary treatment of agricultural commodities.

Phytosanitary treatment refers to the technologies to eliminate the potential of spread of invasive insects and pests from one geographical region to another. The spread of such insects can have devastating consequences in prime agricultural regions of the importing countries. Currently a variety of technologies including chemical fumigation (with methyl bromide), cold treatment and elevated temperature treatments are used for a variety of agricultural products. However, with the growing demand for higher quality and chemical-free fresh and processed foods, these conventional techniques are being replaced by ionizing radiation technologies. Besides the application of ionizing radiation technology on the food directly, this technology is finding increasing number of applications that one can describe as “food industry-related applications.” These will include the use of the technology for improving the quality of plastics used in food packaging in terms of polymer characteristics as well as the environmental persistence of food packaging polymers.

Besides improving the quality of polymers, ionizing radiation technology has now replaced hydrogen peroxide as the chemical sterilant for sterilizing aseptic packaging materials. Thus, it is evident that ionizing radiation technology is impacting the food industry in a number of ways not generally realized.

Ionizing radiation technology, however, is a catch all term for primarily three different technologies; namely, cobalt-60 based gamma irradiation technology, accelerator-based electron beam (eBeam) technology, and X-ray technology. Therefore, it is critically important to understand the key differentiating characteristics of these three distinct technologies, but also appreciate the nuanced differences in order to understand the reason(s) why the popularity of gamma irradiation technology is waning compared to eBeam and X-ray technologies. This chapter attempts to provide the reader a detailed discussion of eBeam and X-ray technologies and how they are being commercially used around the world. The reasons for the slow eclipse of gamma irradiation as the go-to technology for the food industry is also presented as part of this chapter.

Challenges the food industry face requiring ionizing radiation technology

To understand the challenges the food industry faces in terms of quality and safety requires an appreciation of the world of microorganisms. Microorganisms have been on this planet for almost 3.8 billion years, whereas modern humans have been around for only approximately 200,000 years. During this period, microbes have had to endure extremely adverse conditions—especially during this planet's origins. Not only have microorganisms adapted themselves to extreme conditions (temperature, pressure, oxygen concentrations, pH, etc.) they have ended up colonizing all known environments on earth, including the bodies of all species, insects, plants, and humans. They also outnumber humans at levels that are unimaginable. For example, a gram of soil contains at least an order of magnitude more microbes than the entire human population! Therefore, when one looks at the production of food in the natural environment and the processing of food in human-engineered processing facilities, foods will invariably harbor microorganisms.

While many species of microbes are benign, certain species can cause undesirable effects on food. Yeasts and molds cause extensive spoilage of foods, and their removal can greatly increase shelf life and improve storage of both fresh and processed foods. Besides spoilage, there are pathogenic microorganisms such as *Salmonella* spp., toxigenic *E. coli*, and *Listeria monocytogenes* that are known to cause illnesses and death in humans. According to the US Centers for Disease Control & Prevention (CDC), approximately 1 in 6 Americans (~48 million) get sick from foodborne pathogens annually and approximately 3000 die each year from foodborne pathogens (Scallan et al., 2011). Most of the illnesses are caused by viruses (58%), followed by *Salmonella* spp. (11%), *Clostridium perfringens* (10%), and *Campylobacter* (9%). Therefore, the food industry must rely on technologies that eliminate or at least reduce the exposure of humans to these pathogens. Many of these pathogens in the environment arise from humans and farm animals through improper disposal of animal and human wastes. These pathogen-laden wastes reach food crops through irrigation water, contaminated soil, manure applications, and via contaminated farm workers. In terms of the meat industry, gastro-intestinal pathogens within these animals contaminate the meats prepared from these animals. This is particularly a major concern when meat products are prepared from the mixing and grinding of meats from different parts of the animals (e.g. ground beef). Eggs can get contaminated from *Salmonella* infected chickens. Poultry meat can get contaminated from *Salmonella* spp. or *Campylobacter* spp. that are often present within the intestinal tract of poultry.

The food industry has relied on the HACCP (Hazard Analysis and Critical Control Point) system to manage the microbiological, physical, and chemical safety of foods. In the US, the FDA has instituted the Food Safety Modernization Act (FSMA) in 2011 to establish preventative measures to control microbiological hazards. FSMA covers several different but related topics such as prevention, inspection and compliance, response, food imports, and building partnerships among critical stakeholders. Food producers, processors, and retailers around the world are challenged with providing high quality, microbiologically safe, and environmentally sustainable foods to consumers (Pillai and Shayanfar, 2015a, b). Fresh produce is particularly vulnerable to microbial pathogens since they are very often grown in open fields, are handled extensively, and because of their perishability cannot be treated by any heat-based pathogen control technologies. For example, berries such as strawberries, blueberries and blackberries can be contaminated at the pre-harvest stage via irrigation, fertilizers, soil, wildlife, and domestic animals. Additionally, berry-associated outbreaks due to cross-contamination have occurred at the post-harvest stage through field workers, food handlers, contact surface, packaging materials, and during processing. It is also now known that viruses that are transmitted through the fecal oral route can get internalized into leafy greens such as lettuce (DiCaprio et al., 2012).

Besides foodborne pathogens, food spoilage due to lack of appropriate storage facilities or poor transportation infrastructure has global ramifications. The reasons for this significant loss are primarily due to lack of food-chain infrastructure, poor storage capabilities, poor transportation logistics, and, unfortunately, lack of knowledge or adoption of storage technologies (Nellemann, 2009). In India, the lack of cold storage at both wholesale and retail outlets results in almost a 50% loss of fresh produce (Nellemann, 2009). In Southeast Asia, more than one third of the rice production is lost due to pests and spoilage (FAO, 2011). The loss of food in the food supply chain due to food waste is quite different in developing countries as compared to countries in the developed parts of the world. The majority (60%) of the food in the developed parts of the world is lost at the end of supply chains, for example in homes and at retail. This is in contrast to the majority (~75%) of the food being lost much earlier in the supply chain (i.e., on farm and during transport and processing) in the under developed parts of the world (Godfray et al., 2010). Therefore, the food industry must be strategic in deciding where appropriate technologies are implemented depending on the region of the world.

Thanks to the growing buying power of the populace around the world, there is an increasing need for a variety of fresh produce all 12 months of the year. Therefore, it is not surprising anymore for retailers to be offering tropical fruits in middle of winter to consumers in western Europe. Similarly, consumers in the tropics are now able to purchase apples, cherries, and peaches many months of the year. Additionally, due to the growing migration of people, ethnic dishes are becoming mainstream around the world. For example, the growing Hispanic population in the US, with their increasing buying power and the growing popularity of Mexican cuisines in the US, have resulted in significant increases in the import of fruits such as guavas, mangoes, and limes from Mexico into the US.

However, the USDA-Animal and Plant Health Inspection Service (USDA-APHIS) has strict requirements in terms of preventing the accidental introduction of insects and pests via these agricultural commodities. The USDA-APHIS legacy treatments have included fumigation using methyl bromide, hot water dips and cold temperate storage. However, the quality of the fruits deteriorates or is not maintained when such treatments are applied. The USDA-APHIS has spent a considerable amount of resources to explore the implementation of ionizing radiation technologies for phytosanitary applications. Phytosanitary regulations pertain to the transboundary shipment of agricultural commodities around the world. The International Plant Protection Convention

(IPPC) is a global agreement that focuses on preserving standardization of plant health practices around the world and was established to prevent the introduction of regulated pests and pathogens to help protect cultivated and wild plants in different regions of the world (Pillai et al., 2014).

Food security (lack of adequate food supplies) is also of strategic importance to countries globally. The COVID-19 pandemic has heightened our awareness of the need to consider technologies to address food security as well. This pandemic has caused the closure of several meat and cheese processing plants across the USA. In April 2020, based on a survey of 115 meat and poultry processing facilities in 19 states, the CDC reported that there were approximately 5000 cases of COVID-19 and 20 deaths in food processing facilities (Dyal et al., 2020). By April 2020, almost 25% of the US' meat packing plants were closed down due to the rapid spread within these food processing facilities. In April 2020, Tyson, Inc. closed a meat processing plant in Iowa after hundreds of employees were infected. The situation is so dire that the chief executives of several leading food companies have alerted to the possibility of the US food supply chain breaking (Arkin, 2020).

So, ionizing radiation technology has a key role to play in preventing food spoilage, thwarting food security challenges, eliminating foodborne pathogens as well as controlling the accidental introduction of insects and pests. It must be noted, however, that the technology has to be customized to the specific application so that the technology is not only able to meet its intended application, but it is economically viable and sustainable. The conventional idea that the technology will be introduced and implemented by governmental agencies is flawed at the present time. One must consider the private sector to be able to finance and maintain the technology for the different food industry applications.

Ionizing radiation technologies in the food industry

As noted earlier, the use of ionizing radiation technology is not new. British scientists patented this technology for the preservation of foods in 1905. By 1921, scientists with the United States Department of Agriculture were recommending its use to control the protozoan pathogen trichinae in pork. Food irradiation is one of the mostly extensively studied food processing technologies, but, unfortunately, the least understood and often impugned technology ever (Pillai, 2016). In 1958, the US Food and Drug Administration amended the Food, Drug and Cosmetic Act (FD&C) to designate food irradiation as a food additive and not as a food processing technology (FDA, 2018). This was a pivotal regulatory decision because designation of this technology as a food additive requires that all foods (human and pet food) in the US that are treated with ionizing radiation be specifically labeled as such. Some in the food industry blame the labeling requirement as the key reason why the technology is not more widely adopted by the industry. However, on careful analysis it is evident that the reasons for the lack of widespread adoption are multi-fold. These include the lack of processing facilities dedicated for food processing, the capital and operational expense associated with this technology, as well as the general lack of deep actionable information about this technology among many of the decision makers in the food industry. Currently, gamma irradiation using cobalt-60, electron beam and X-ray irradiation using accelerator technologies are the main technologies used in the food industry. Gamma irradiation using cobalt-60 has historically been the mainstay of the irradiation technology in the food industry.

Underlying principles of how ionizing radiation technology controls microbial populations

Ionizing radiation, whether it is produced by cobalt-60, electron beam or X-ray technology, inactivates microbial cells by causing a number of irreparable double stranded DNA and RNA breaks in the microbial cells that are in the way of the electrons or photons. When photons from cobalt-60 encounter electrons within the orbital shells of atoms in the food materials, these electrons are ejected from their orbital shells (causing ionization events, hence ionizing radiation). These ejected electrons pick up some of the energy from these collisions and they themselves cause ionization events in adjoining atoms. Thus, each electron ends up creating a cascade of ionization events until the ejected electrons lose their energy to cause any further ionization events. Ionizing radiation does not discriminate between a pathogen or a non-pathogen or whether they cause damage to a DNA molecule or an RNA molecule, or a protein or a carbohydrate. The targets of ionizing radiation depend on the molecular weight. Since DNA is the largest molecule within microbial cells, compared to proteins or carbohydrates, at a given dose, most of the damage occurs in DNA molecules rather than in proteins. The electrons can cause direct effect on DNA molecules by the ionization events described above.

Additionally, the electrons will cause radiolysis of water molecules, which result in the release of highly reactive but short-lived free radicals such as hydrated proton, hydroxyl radicals, hydrogen gas, hydrogen peroxide, as well as aqueous electrons. The effects of hydrolysis depend on the amount of energy absorbed per unit mass of material (i.e., absorbed dose, measured in Grays/kilograys and denoted as Gy or kGy). The DNA is the largest biomolecule in the cell and, therefore, it is the most ionizing radiation sensitive molecule in cells. The microbial cell can repair single and double stranded breaks in its DNA. However, if the double strand breaks are juxtaposed across each other in different strands of the DNA, the microbial cells are incapable of repairing this type of damage. It is estimated that with each 1 kGy of ionizing radiation exposure, as many as between 10–100 double strand breaks occur. It is precisely for this reason that when food is treated with ionizing radiation, the decline in the bacterial bioburden results in extended shelf-life. Double strand breaks are the most lethal form of DNA damage because they halt DNA replication. With increasing dose,

ionizing radiation can also affect plasmid DNA, RNA molecules, cellular membranes, and even structural and functional proteins (e.g., enzymes).

Structural proteins, catalytic proteins (enzymes), and most vitamins are not damaged at doses regularly used in food irradiation (Smith and Pillai, 2004). Exposure to very high doses can, however, cause damage to macro molecules, such as proteins. The reason that maturation is retarded or inhibited is because the genes of many of these enzymes may have been structurally damaged during the double strand and single strand breaks. Previous studies have shown that for a dose of 100 Grays (0.1 kGy), 2.8% of the DNA, 0.14% of enzymes, and 0.004% of amino acids will be damaged (Pillai and Shayanfar, 2017). Therefore, the microbial DNA is damaged by both direct effects of the energetic electrons or from the highly reactive free radicals produced by the radiolysis of water molecules. Bacterial cells will try to repair the damage that has occurred during ionizing radiation. The cells attempt to repair their DNA damage using a variety of specific and non-specific repair mechanisms such as methyl directed mismatch repair, guanine oxidations, nucleotide excisions, base excisions, recombination repairs as well as the generic SOS repair systems.

Microbial inactivation in response to ionizing radiation is calibrated in terms of D-10 values. The D-10 value of an organism is the ionizing dose (measured in kGy) required to achieve a 90% reduction in its population. There is a linear relationship between the increasing doses and microbial inactivation. Factors controlling radiation resistance include the general physiological differences between the microbial cells (for example the D-10 value of *E. coli* is 0.1 kGy, while the D-10 value of *Salmonella* spp. is 0.2 kGy), the physical state of the food (fresh/refrigerated as compared to frozen), the type of food (fresh produce vs ground beef vs poultry), presence/absence of oxygen, and the atmosphere within the food package, as well as the physiological state of the pathogen (example if the cells had been exposed to acid stress or nutritional stress, etc.) (Clemmons et al., 2015). Table 1 is an illustrative example of how the D-10 values of organisms could vary, depending on the product temperature, organism type, and packaging conditions.

Historically, the food industry always targets a 5-log reduction of target pathogens in commodities such as spices and ground beef. However, with improved Good Agricultural Practices at the farm and improved food processing technologies within processing plants (associated with HACCP), the probability of large numbers of microbial pathogens on foods is very small. Therefore, the rationale for choosing a dose to target 5-log reduction is debatable. However, internationally recognized standards such as ISO 14470:2011 can guide the choice of an appropriate dose as well as requirements for the development, validation, and process controls for food irradiation.

It must be noted that the choice of the appropriate irradiation dose for food irradiation applications depends on the deep understanding of the expected pathogen bioburden as well as not employing doses that could cause sensory changes in the foods. It is for this reason that careful attention is paid to use this technology at the lowest possible dose. This implies that only foods with superior microbiological quality is processed with this powerful technology. Therefore, ionizing radiation technology should only be used to increase the value of a food by assuring microbiological safety rather than employing this technology to reclaim heavily contaminated foods. It is important that this concept be effectively communicated to those interested in adopting this technology.

Ionizing radiation technology related terminologies of relevance in the food industry

The penetrating power of ionizing radiation is a function of their energy (measured in electron volts or million electron volts) (keV or MeV). Energy dictates the penetration capability of the ionizing radiation. Cobalt-60 has gamma energies between 1.17 and 1.33 MeV. They are photons and, therefore, have no mass and no charge. Therefore, their penetration ability is very high. In addition to the ionizing radiation energy, another key parameter associated with food irradiation technology is “dose”. Dose is defined as the absorbed energy, which is measured in grays designated as Gy. 1 Gray is the deposition of 1 joule of energy per kg of matter. Dose determines the treatment parameter and it determines the specific application the technology is used for. In food irradiation, doses in the 1000 of Gy's are utilized. For example, the maximum dose that the FDA has approved for fresh produce in the US is 1000

Table 1 D-10 values of different microbial targets as a function of product type, product temperature, atmospheric conditions within package.

Product type	Product temperature (°C)	Pathogen	D-10 value (Gy)
Ground beef patties	5	<i>E. coli</i> 0157:H7	0.27–0.38
Ground beef patties	– 15	<i>E. coli</i> 0157:H7	0.32–0.63
Ground beef	3	<i>Salmonella enteritidis</i>	0.55–0.78
Beef	5	<i>E. coli</i> 0157:H7	0.30
Ground beef	5	<i>Campylobacter jejuni</i>	0.16
Deboned meat	5	<i>B. cereus</i> spores	2.56
Beef	5	<i>L. monocytogenes</i>	0.45
Poultry (air packed)	0	<i>Salmonella heidelberg</i>	0.24
Poultry (vacuum packed)	0	<i>Salmonella heidelberg</i>	0.39

Adapted from Pillai SD and Shayanfar S (2017) Electron beam technology and other irradiation technology applications in the food industry. In: *Applications of Radiation Chemistry in the fields of Industry, Biotechnology and Environment*, Springer.

Grays (1 kGy). Dose rate (measured in Gy/s) is the rate at which the energy is deposited in the target material. Dose rate, therefore, will translate into the processing line speeds when these technologies are employed.

The dose rate of gamma irradiation is significantly lower than commercial scale X-ray or eBeam irradiation processes (Miller, 2005). Many of the commercial eBeam facilities have dose rates in the range of 3000 Gy/s (3 kGy/s). Electron beam and X-ray equipment have an additional parameter, termed power, measured in kilowatts (kW). Beam power determines the processing speed of the equipment. The higher the beam power, the higher the product throughput (Brown, 2015). Therefore, when deciding on the appropriate eBeam and X-ray equipment, one has to have a clear understanding of the target doses for the specific application (kGy) as well as the energy (MeV), and the power (kW).

Current challenges associated with gamma irradiation in the food industry

Gamma radiation is used in about 40 countries around the world. The gamma radiation source (cobalt-60) is stored within a deep pool of water. The gamma source is placed under water when not in use because the source cannot be switched off.

The food in pallets are conveyed into the irradiation chamber using automated conveyor and product handling systems. Since photons have very good penetrating abilities, the primary advantage of the gamma irradiation process is that the pallets can be treated without the need to “break-down” the pallet into single cases (as in the case for electron beam facilities). For the actual irradiation process, the cobalt-60 source is raised out of the water and the pallets of food are automatically brought into the irradiation chamber and placed around the cobalt-60 radioactive isotope “pencils” at defined locations for defined periods and position so that they receive the minimum required dose, and ensure that the product receives as uniform a dose as possible. Since the gamma source cannot be switched off, there are obvious challenges associated with possible occupational exposure, radioactive source theft, storage of spent sources, and disposal of spent sources. These challenges have become more acute—especially due to the potential dangers posed by terrorism. Cobalt-60 isotope-based gamma sources have a 12% loss in specific activity each year due to radioactive decay. Gamma sources are priced on a per curie basis. Curie is 3.7×10^{10} disintegrations per second. Curie is, however, not an SI unit. The SI unit for ionizing radiation is the Becquerel.

The cost of gamma sources has increased substantially over the past decade, especially since there are only a couple of cobalt-60 suppliers around the world and because of the cost associated with the production of cobalt-60 gamma sources. The purchase of the cobalt-60 sources will invariably also include the cost of return for recycling as well as additional costs associated with the regulatory burden associated with radioactive materials. More so, smaller gamma irradiation facilities must pay substantially more compared to the large multinational gamma irradiation service providers. Current estimates for 1 curie of cobalt-60 is around \$5.00 (personal communication). A majority of the food irradiation facilities around the world require between 0.5 and 1 million curies of Cobalt-60. Therefore, the cobalt-60 cost alone (excluding shipping, installation, recycling/disposal) will be around \$2.5 million and \$5 million. In addition to these monetary costs, the legal, environmental, and security costs associated with maintaining and operating a cobalt-60 facility can be steep. There are anecdotal comments that suggest the security costs associated with a gamma facility is actually higher than the cost of cobalt-60. Because of the myriad of security challenges, the International Atomic Energy Agency (IAEA) and the US National Nuclear Security Administration (NNSA) are keen on replacing these high activity radioactive sources out of the commercial application arena (NNSA, <https://www.energy.gov/nnsa/office-radiological-security-ors>).

Basic principles of electron beam technology

There are several excellent books and book chapters discussing eBeam technology and how the technology can be integrated into food processing (Miller, 2005; Clemmons et al., 2015; Pillai and Shayanfar, 2015a, b; Shayanfar and Pillai, 2015; Brown, 2015).

The primary differentiation between cobalt-60 based gamma irradiation and electron beam (eBeam) technology is that eBeam technology does not involve radioactive isotopes. Electrons are generated using a cathode (electron gun) in a vacuum environment using commercial electricity. The electrons are accelerated to 99.99% of the speed of light at which time they have acquired energy of about 10 MeV. These 10 MeV electrons then exit the accelerator. After this beam of electrons exits the accelerator, it enters the scan horn. Within the scan horn, the electrons are oriented, focused and scanned in a sweeping pattern allowing the electrons to be “scanned” across the cases of food in a very defined and controlled manner.

There are basically three fundamental designs of electron beam equipment that could potentially be used in food irradiation; namely, pulsed machines (linear accelerators or linac), Continuous Wave (CW) machines (e.g., Rhodotron) and Direct Current (DC) machines (e.g., Dynamitron). Linacs is the generic term for accelerators that produce a high energy, high intensity burst of radiation. The bursts of radiation are the “pulses.” In the Rhodotron, the electron beam is not pulsed but composed of high intensity “bunches” that occur at very high repetition rate. In the Dynamitron-class machines, the electrons are neither pulsed nor in bunches. Instead, there is a constant stream of equally intense electrons. Table 2 below highlights some of the differences between linac, Dynamitron, and Rhodotron-class accelerators.

For most food irradiation applications, high doses and high throughputs are desired. Therefore, linacs and Rhodotrons are the ones that are finding increasing applications in food processing. Fig. 1 is a schematic and photo illustration of a linac-style accelerator. Electron beam accelerators are also used for sterilization processes in the medical industry (Kalia, 2021).

Table 2 Differentiating characteristics of the DC, CW, and pulsed type accelerators.

Parameter	DC accelerator	CW accelerator	Pulsed accelerator
Genre	Dynamitron-style	Rhodotron-style	LINAC style
Maximum energy used commercially	5 MeV	7.5–10 MeV	10 MeV
Power (commercial line speeds possible)	High power; as high as 100 kW	High power; as high as 800 kW	Limited; maximum around 20 kW
Electrical use efficiency	High	Medium	Low
Physical size	Large	Medium	Small

Originally adapted from Brown D (2015) Integrating electron beam equipment into food processing facilities: Strategies and design considerations. In: *Electron Beam Pasteurization and Complementary Food Processing Technologies*, Pillai SD and Shayanfar S (eds), 27–45.

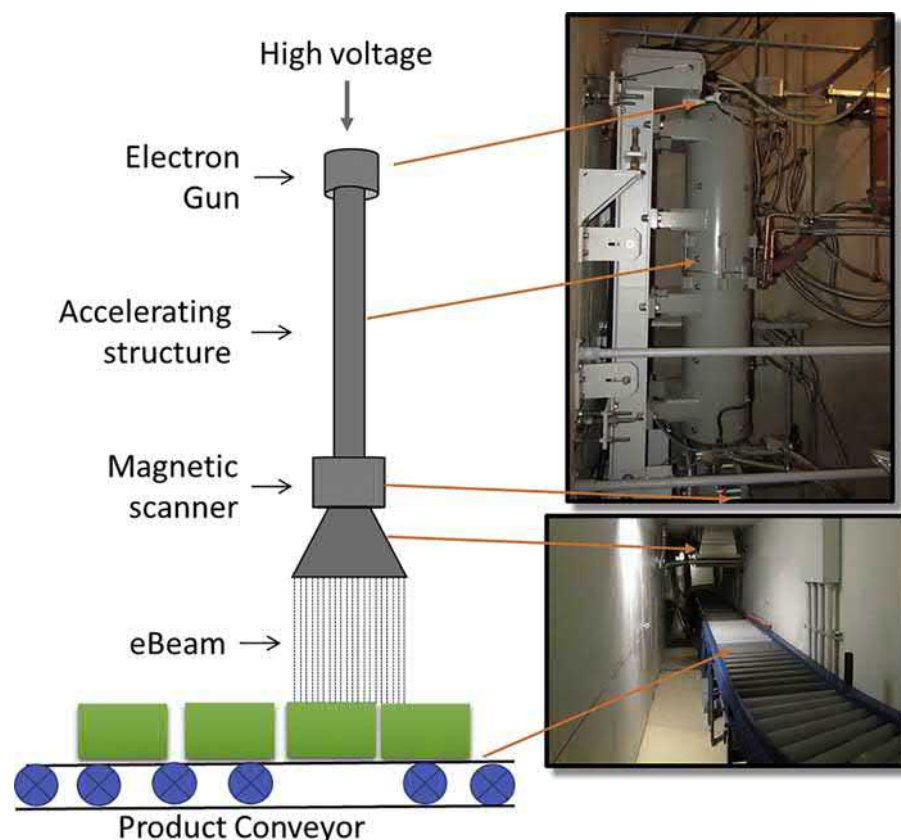


Fig. 1 Schematic and photo of a linear accelerator system used for panoramic food irradiation showing the cases of food on a conveyor being scanned by the electrons from the scan box. The photos are from a commercial scale panoramic food irradiation facility. Courtesy: Texas A&M University-National Center for Electron Beam Research.

The maximum electron energy that can be used for commercial food processing is 10 MeV. The penetration ability of 10 MeV electrons is significantly less than that of the cobalt-60 based gamma irradiation. In gamma irradiation, entire pallets of products can be treated at a time. However, in eBeam processing, the pallets have to be “broken-down” into individual cases and they need to be conveyed through the eBeam system (Fig. 2). In commercial eBeam food processing, the key parameter that needs to be considered is the maximum thickness of material that the beam can penetrate through. The areal density (g/cm^2) or mass of material in a certain area is the most common definition of “thickness” in food processing.

$$\text{Areal Density} = \frac{\text{boxweight}(\text{lbs}) \times 70.4}{\text{length of box} \times \text{width of box}} = \text{g}/\text{cm}^2.$$

A 10 MeV eBeam accelerator can penetrate up to $3.3 \text{ g}/\text{cm}^2$ using a single sided treatment or up to $8.3 \text{ g}/\text{cm}^2$ using a double-sided treatment. Double sided treatment refers to the product being treated with 10 MeV eBeam from top and bottom or the product is flipped over and treated from the other side. If the case of food has an areal density that is greater than $8.3 \text{ g}/\text{cm}^2$, it cannot be processed with eBeam (since the beam will not penetrate the product). In this situation one has the option to repackage the product so

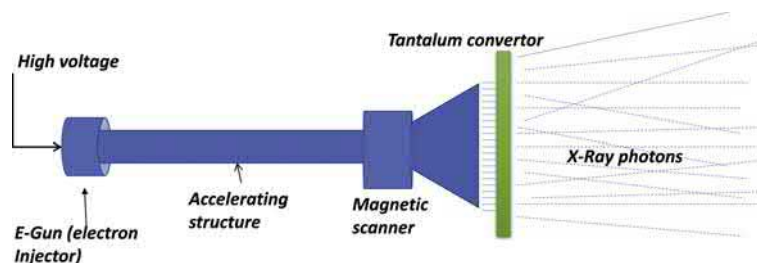


Fig. 2 Schematic of how X-rays are generated from an eBeam linac.

that the areal density falls between 3.3 and 8.3 g/cm² or, if this is not possible, then X-rays have to be used; if this is not possible, gamma irradiation is the only choice.

Worldwide, the maximum energy for eBeam technology that can be used for food irradiation is 10 MeV. The reason for this upper limit on energy is that higher energies could potentially induce transient radioactivity in highly dense materials such as bones, etc. The international harmonization entity, Codex Alimentarius regulates the use of irradiation technology for transboundary shipment of foods. This multinational body has also established 10 MeV energy (similar to the US) as maximum eBeam energy for food irradiation.

Basic principles of X-ray technology

A major drawback of eBeam based food processing is the requirement that the cases must fit within the areal density specifications and that pallets cannot be treated as such. Other than gamma irradiation, X-ray technology is the only other choice to process entire pallets. Miller (2005) is an excellent resource for details about the X-ray technology and the engineering principles of X-ray based food processing. Briefly put, X-ray generation from eBeam in a linac or Rhodotron is based on placing a very high atomic mass material such as tantalum or gold directly in the path of a stream of high energy (5 MeV or 7.5 MeV) eBeam. The collision of high energy electrons results in the formation of X-ray photons. X-ray photons are like gamma irradiation in that they have very high penetration capabilities (compared to eBeam). However, the energy of X-ray (either 5 MeV or 7.5 MeV) is significantly greater than the energy of cobalt-60 based gamma irradiation. Also, X-ray photon beams can have a much higher dose rate compared to gamma photon beams: ~ 100 Gy/s as compared with ~100Gy/min) (Hieke, 2015).

Fig. 2 is a highly simplified schematic of how X-rays are generated from a linac accelerator. In the US, eBeam energies as high as 7.5 MeV can be used to generate X-rays for food irradiation. However, worldwide, the maximum energy for X-ray is still set at 5 MeV. The major advantage of X-ray based food processing is that entire pallet can be treated at a time. Moreover, since the dose rate is significantly lower than that of eBeam processing, X-ray technology is ideally suited for low dose food processing applications such as phytosanitary treatment as well as for extending the shelf-life of fresh produce. However, the conversion of 7.5 MeV electrons into X-rays in X-ray processing is quite inefficient (~ 12.5%). Therefore, the cost of X-ray processing will be more than eBeam processing. The rule of thumb is that eBeam processing will be cheaper than X-ray processing. However, there are several other factors that can influence the overall cost of eBeam processing (discussed below).

Low energy electron beam processing technology

One of the major shortcomings of the high energy (10 MeV) eBeam technology platform is the capital expense (a turnkey system excluding building and land is approximately \$6–9 million). These high energy eBeam (HEEB) systems are also hampered by the need to have specialized concrete shielding to contain the ionizing radiation while the system is in operation. In general, the high capital investment has precluded the adoption of HEEB technology by most food processors. Around the world, the model for food irradiation currently involves transporting the food by truck to a commercial irradiation service provider. However, given the cost of transportation, turn-around time, and vulnerability to steep price increases, many of the major users of the technology in the food industry, such as spice and ingredient processors, are interested in bringing eBeam technology into their facilities as either in-line or end of line eBeam irradiation systems. Low energy eBeam (LEEB) technology offers this opportunity to be brought into commercial operations as either in-line systems or end of line sterilization/decontamination systems. Low energy eBeam systems can be broadly defined as generating electrons having energies ranging between 80 and 1000 keV. Since the electron energy in currently available commercial LEEB ranges between 180 and 300 keV, these are self-shielded units. However, as expected, LEEB can only be used for surface sterilization.

Within the food industry, the early adopters of LEEB have been the aseptic packaging and the printing ink industries. Low energy eBeam (LEEB) systems (~ 180 and 300 keV) are now commercially available for these industries (Pillai and Shayanfar, 2015b). The LEEB technology platforms are now offered as part of the original equipment by aseptic packaging equipment manufacturers such as TetraPak as part of their Tetra Pak® E3/Speed, E3/flex, and E3/CompactFlex platforms (<https://www.tetrapak.com/tetra-pak-e3->

Table 3 D-10 Values of Bacillus spores when exposed to 100 keV electrons and 10 MeV electrons.

Bacillus sp.	D-10 value	
	100 keV	10 MeV
<i>B. pumilus</i>	1.34 kGy	2.12 kGy
<i>B. megaterium</i>	3.46 kGy	4.11 kGy
<i>B. subtilis</i>	1.01 kGy	2.05 kGy

Adapted from Urgiles E, Wilcox J, Montes O, Osman S, Venkateswaran K, Cepeda M, Maxim J, Braby L, and Pillai SD (2007) Electron beam irradiation for microbial reduction on spacecraft components. In: *Proceedings of the IEEE Aerospace Conference*.

ebeam). According to the company's commercial marketing claims, the new eBeam based sterilization systems offer advantages such as higher production capacity without any increase in equipment footprint and up to 20% cost savings due to improved efficiency. A key advantage of LEEB for aseptic packaging as compared to the legacy hydrogen peroxide-based sterilization is the complete avoidance of using chemicals on packaging materials in direct contact with food and the ability to sterilize packaging without leaving any chemical residues. The electron energy in LEEB is approximately 200 keV. Low energy eBeam systems are characterized by very high dose rates compared to HEEB due to their short ranges. Previous studies in our laboratory have shown that higher dose rates could achieve greater microbial killing (Urgiles et al., 2007) (Table 3). The response of microorganisms in terms of their inactivation kinetics and their molecular responses (genomic and metabolomic) at varying dose rates needs detailed investigations (Bhatia and Pillai, 2019).

The LEEB platforms are gaining traction as in-line sterilization solutions within the single use device (e.g., syringe) manufacturers, by the pharmaceutical industry such as in vial filling, and lyophilization processes. Fig. 3 shows the self-shielded LEEB system (energy between 180 and 200 keV) for continuous syringe tub decontamination. Fig. 4 depicts how multiple LEEB systems, when configured appropriately, can be used to achieve complete in-line sterilization of a tub of syringes within the manufacturing space of a single use device manufacturer.

Besides the use in aseptic packaging and in the single use device manufacturing, commercial LEEB is now available for the surface sterilization of spices such as black pepper, cumin, aniseed, etc. Buhler, Inc., the Switzerland-based original equipment manufacturer of food industry equipment is now marketing Laatu™, a LEEB platform for the spice and ingredient industry (Fig. 5). The company's marketing literature states that 5-log reduction of microbial pathogens can be achieved with the Laatu™ LEEB platform. This, once again, is a self-shielded microbial reduction solution for industrial dry food processing. In this system, the spices/dry foods fall via gravity through an electron cloud generated from two low energy (≤ 300 keV) eBeam "lamps" (Fig. 6). The reported throughput is approximately 1 ton/h delivering a minimum dose to achieve a 5-log reduction of *Salmonella* spp. on spices and dry foods.

International regulations governing ionizing radiation technology in the food industry

United States

In the US, the use of ionizing radiation technology in food processing is governed by the FDA, USDA-FSIS and the USDA-APHIS. The FDA controls the labeling requirements and the approvals for food applications. The USDA-APHIS controls the minimum doses that can be used for phytosanitary applications. The USDA-FSIS regulates the use of the technology in the meat and poultry industries. Historically, food irradiation in the US is governed by the 1958 Food Additives Amendment of the Food Drug and Cosmetic (FD&C) Act as a "food additive". Per this Act, food additives must be specifically labeled with the radura symbol and must state the phrase, "treated with radiation" or "treated by irradiation." Thus, all irradiated foods that are sold at retail must have this labeling at the point of sale. The FDA also has specific stipulations as to the type of packaging materials that can be used, the maximum dose that these materials can receive, etc. Presently, in the US, the following foods (Table 2) are permitted to be treated with ionizing radiation (either gamma, eBeam or X-ray). The foods include fresh fruits and vegetables, meat (from cattle, sheep, swine and goat), poultry, shell eggs, iceberg lettuce and fresh spinach, spices, seeds for sprouting and molluscan shellfish. Table 4 shows the dose limits and the foods that can be treated with ionizing radiation in the US.

Per the FDA regulations, one cannot use the technology on fresh produce and make a claim that pathogens are eliminated (because the regulation is specifically approved for only the use of this technology for growth/maturation inhibition). Only iceberg lettuce and spinach are currently permitted for use for "pathogen elimination" by this technology. In the US, ionizing radiation is permitted for treating animal diets (bagged complete diets, packaged feeds, feed ingredients, bulk feeds, animal treats and chews). However, the dose cannot exceed 50 kGy. For complete poultry diets and poultry feed ingredients, the dose cannot be below 2 kGy and cannot exceed 25 kGy. The upper dose limit is based on the assumption that the irradiation treatment is being used to control *Salmonella* spp. Importantly, if an irradiated feed ingredient is less than 5% of the final product, the final product can also be irradiated without being considered re-irradiated.



Fig. 3 Self-shielded LEEB for continuous syringe tub decontamination. Courtesy SKAN US, Inc. Raleigh, NC.

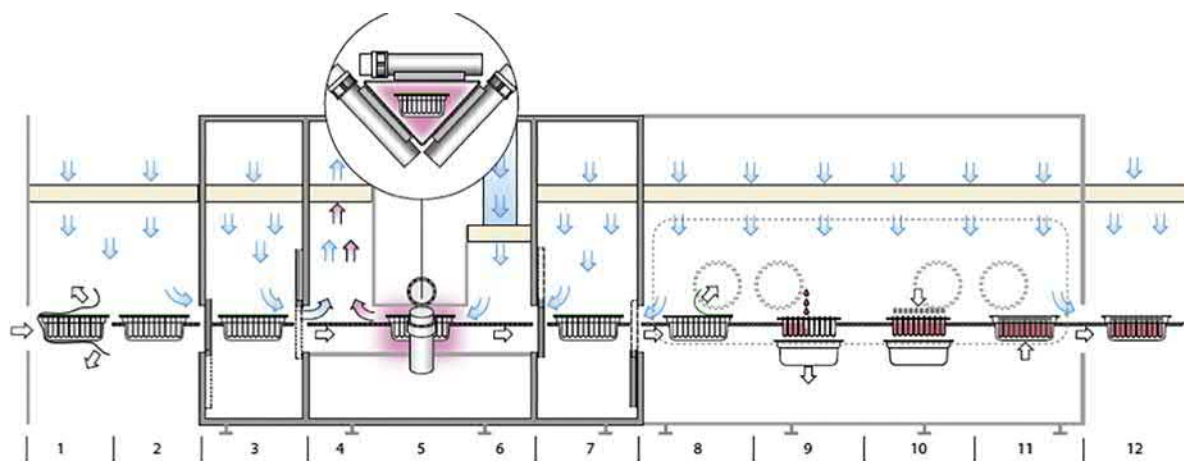


Fig. 4 Schematic of how 3 LEEB emitters (magnified in the upper circle) are configured to achieve the sterilization of the syringe tub (courtesy SKAN US, Inc. Raleigh, NC).

European Union

Interestingly, the European Union is more supportive of food irradiation than genetically modified foods. Contrary to popular belief, food irradiation is legal in the European Union per Articles 7(3) and 3(2) of Directives 1999/2/EC of the European Parliament. Food irradiation does occur in the EU, albeit, at significantly lower volumes than in other countries such as the US, China, and India. The irradiation of dried aromatic herbs, spices and vegetable seasonings is authorized across the EU by Directive 1999/3/EC. There is a community list of food and food ingredients that can be treated with ionizing irradiation. In addition to this “community



Fig. 5 Self-shielded LEEB Laatu™ for the surface decontamination of spices and dry foods. Courtesy of Buhler Inc, Uzwil, Switzerland.

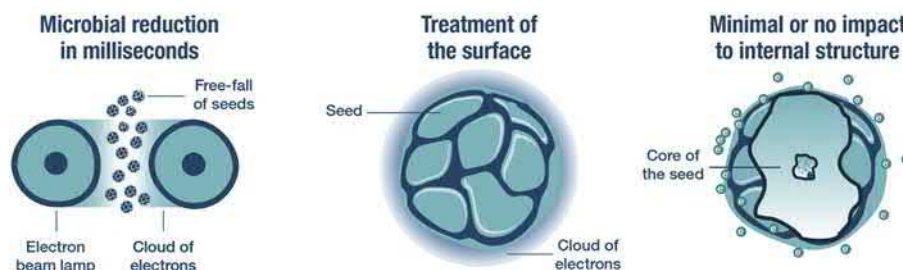


Fig. 6 Schematic showing the underlying principles of Laatu™. Courtesy of Buhler Inc, Uzwil, Switzerland.

Table 4 List of foods and food items permitted for ionizing radiation treatment in the United States (FDA, 2019).

<i>Food/food-related item</i>	<i>Specific application</i>	<i>Maximum allowable dose (kGy)</i>
Fresh, non-heated processed pork	Pathogen control	0.3–1.0
Fresh/frozen uncooked poultry products	Pathogen control	3
Refrigerated, uncooked meat products (sheep, cattle, swine and goat)	Pathogen control	4.5
Frozen uncooked meat products (sheep, cattle, swine and goat)	Pathogen control	7
Fresh/frozen molluscan shellfish	Pathogen control	5.5
Fresh shell eggs	Pathogen control	3.0
Dry or dehydrated spices and food seasonings	Microbial disinfection	30
Fresh produce	Growth and maturation inhibition	1
Fresh produce	Insect disinfection	1
Fresh iceberg lettuce and fresh spinach	Pathogen control	4.0
Seeds for sprouting	Pathogen control	8.0
Dry/Dehydrated spices and food seasonings	Microbial disinfection	30
Dry/dehydrated enzyme preparations	Microbial disinfection	10
Wheat flour	Mold control	0.5
White potatoes	Inhibit sprouting	0.15

list of foods and food ingredients”, seven-member states have their own list of foods and food ingredients that are above and beyond the community list (European Union, 2019a, b). Unlike the US, the EU’s has much stricter rules governing the labeling of irradiated foods. In the EU, any irradiated foodstuff containing one or more irradiated food ingredients must be labeled with the words “irradiated” or “treated with ionizing radiation.” Therefore, if an irradiated product is used as an ingredient (e.g., spices on a pizza) the notation that the ingredient was irradiated should be mentioned in the list of ingredients. Presently, Belgium, Bulgaria, Czech Republic, Germany, Estonia, France, Spain, Hungary, Netherlands, Poland, and Romania treat one of more foods or food ingredients by ionizing radiation. Table 5 is a listing of the countries and the food items that are being commercially irradiated in the European Union and Table 6 is the irradiation doses that are used in the EU countries.

The European Food Safety Authority (EFSA) in 2011 recommended that irradiation should be considered as one of the several approaches to reducing pathogens in food and this technology should be integrated into a multi-hurdle strategy, thereby assuring public health protection (EFSA, 2011a). The EFSA panel also confirmed the toxicology and chemical safety of irradiated foods (EFSA, 2011b). Importantly, they also recommended that food irradiation should be based on risk assessment and on the desired risk reduction rather than on predetermined food classes, commodities and doses. They also recommended that upper dose limits should not be specified, but rather based on undesirable sensory chemical changes that may happen at increasing doses. The EU is presently receiving public comments regarding to expanding the approvals of the technology across a wider array of foods and also possibly changing the stringent labeling requirements.

China

China irradiates the largest volume of food. By volume, the irradiated chicken feet sold as a snack in convenience stores in China is among the largest commodity that is treated by ionizing radiation anywhere in the world. There is a paucity of reliable datasets about the actual volume of food that is irradiated in China. The last report was based on 2005 datasets. Based on these values, a total of approximately 150,000 tons of food was being commercially irradiated in China in 2005 (Kume et al., 2009). Based on information presented at conferences, etc., it is thought that over 250,000 tons of food is irradiated in China. There are six mandatory national standards in terms of the foods that can be irradiated in China; namely, the maximum dose, the packaging, and the labeling requirements. The different food classes are (1) fruits and vegetables, (2) beans and grains, (3) poultry (fresh/chilled/frozen) and meats, (4) cooked meats, (5) spices and dehydrated vegetables, and (6) dried fruits and nuts. China’s approved list of packaging materials closely resemble that of US and Britain. The maximum dose for cardboard is set at 10 kGy, while for Polyethylene-polyvinyl acetate co-extruded film the maximum is 30 kGy. Nylon 11 (nylon is a part of some food packaging polymers) has a maximum dose limit of 10 kGy, while Nylon 6 has a maximum dose limit of 60 kGy (FDA, 2019).

India

In India, the regulations governing commercial food irradiation are governed by the Food Safety and Standards Authority of India (FSSAI). In November 2015, this agency proposed a revised set of standards for foods and allied materials. In January 2016, India notified the World Trade Organization (WTO) about the revised food irradiation standards. The Indian food irradiation regulations categorize foods and associated materials in different classes (Tables 7 and 8). The food irradiation regulations in India are prescriptive and provide significant commercial opportunities to expand the technology for foods. For example, the technology is approved for Class # 8 foods (ethnic foods, military rations, space foods, ready-to-eat, ready-to-cook and minimally processed foods). There is significant flexibility in terms of the doses that can be used and the ability to use the technology for fully cooked foods. The ability to use this technology for cooked foods opens significant opportunities for widespread commercial adoption, for example, in foods prepared for long-distance trains, airlines, buses, etc. The ability to use the technology for creating emergency stockpiles for natural

Table 5 Partial listing of EU countries and food items that are commercially irradiated (EU data from 2011).

<i>EU Country</i>	<i>Food and Food ingredients irradiated</i>
Belgium	Dehydrated blood, egg white, fish, shellfish, shrimp, frog legs, gum Arabic, herbs and spices, poultry, rice meal, vegetables
Czech Republic	Herbs and spices
Germany	Herbs and spices
Estonia	Herbs and spices
Spain	Herbs and spices
France	Frog legs, gum Arabic, herbs and spices, poultry
Hungary	Herbs and spices
Netherlands	Egg whites, fish, shellfish, shrimp, frog legs, herbs and spices, poultry, dehydrated products
Poland	Herbs and spices

Table 6 The administered doses of irradiation on aromatic herbs, spices and dried vegetable seasoning in some of the EU members.

<i>Country</i>	<i>Administered dose (kGy)</i>
Belgium	4–7.9
Czech Republic	5.66–9.92
Germany	5–10
Estonia	10
Spain	9.31
France	5–10
Hungary	2–10
Netherlands	4–8
Poland	5–10

Table 7 Indian regulations governing ionization irradiation dose limits and applications for different classes of foods.

<i>Class</i>	<i>Food Type</i>	<i>Application</i>	<i>Dose Limits (kGy)</i>	
			<i>Minimum</i>	<i>Maximum</i>
1	Bulbs, stem and root tubers and rhizomes	Inhibit sprouting	0.02	0.2
2	Fresh fruits and vegetables (other than class 1)	Delay ripening	0.2	1.0
		Insect disinfestation	0.2	1.0
		Shelf-life extension	1.0	2.5
		quarantine	0.25	1.0
3	Cereals and their milled products, pulses and their milled products, nuts, oil seeds, dried fruits and their products	Insect disinfestation	0.25	1.0
		Bioburden reduction	1.5	5.0
4	Fish, aquaculture, seafood and their products (fresh or frozen) and crustaceans	Pathogen elimination	1.0	7.0
		Shelf-life extension	1.0	3.0
		Control of protozoan parasites	0.3	2.0
		Pathogen elimination	1.0	7.0
5	Meat and meat products including poultry (fresh and frozen) and eggs	Shelf-life extension	1.0	3.0
		Control of protozoan pathogens	0.3	2.0
		Bioburden reduction	6.0	14.0
6	Dry vegetables, seasonings, spices, condiments, dry herbs and their products, tea, coffee, cocoa and plant products	Insect disinfestation	0.3	1.0
7	Dried foods of animal origin and their products	Pathogen elimination	2.0	7.0
		Control of molds	1.0	3.0
		Insect disinfestation	0.3	1.0
8	Ethnic foods, military rations, space foods, ready-to-eat, ready-to-cook and minimally processed foods	Bioburden reduction	2.0	10.0
		Quarantine application	0.25	1.0
		Sterilization	5.0	25.0

Adapted from Pillai, SD and Shayanfar S (2017) Electron beam technology and other irradiation technology applications in the food industry. In: *Applications of Radiation Chemistry in the fields of Industry, Biotechnology and Environment*, Springer.

and/or man-made disasters is another opportunity for commercial expansion of this technology. The specific mention of the min and max doses is empowering to commercial adoption. The irradiated foods have to be labeled with the radura logo in green color with information about the product identity, purpose of radiation processing, radiation processing facility and date of processing.

Future outlook

Choice of food irradiation technology- gamma or eBeam or X-ray?

Based on US and international trends, there is no doubt that gamma irradiation based on cobalt-60 or cesium-137 isotope-based irradiation technologies are on the way out of commercial use. The economics coupled with regulatory, environmental, and security challenges are factors against gamma irradiation. This brings up the question of the future of machine sources such as eBeam and X-ray technologies. Notwithstanding the challenges associated with the relatively small number of reliable suppliers of these technologies, the adoption of either of these technologies for food irradiation purposes has to be based on deep analysis that spans regulations, food industry adoption, decisions of who pays for the service, guaranteed volumes, capital costs, operating costs, as well as the location of the eBeam or X-ray food processing facilities. Further, the decision of adopting the appropriate technology

Table 8 Indian regulations governing dose limits for ionizing radiation processing of food-related allied products.

Allied Product	Application	Dose Limits (kGy)	
		Minimum	Maximum
Packaging materials for food and allied products	Bioburden reduction	5.0	10.0
	Sterilization	10.0	25.0
Food additives	Bioburden reduction	5.0	10.0
	Insect disinfestation	0.25	1.0
	Sterilization	10.0	25.0
Health foods, dietary supplements and nutraceuticals	Bioburden reduction	5.0	10.0
	Insect disinfestation	5.0	1.0
	Sterilization	10.0	25.0

**Fig. 7** Photo showing boxes of mangoes on the conveyor system being readied for eBeam processing. The mangoes arrive at the facility as pallets of mango boxes (shown on side).

depends on deep analysis of the engineering specifications and processing throughputs of the different technology platforms, costs (capital and operating).

The USDA-APHIS allows the technology to be used “in-country,” before the commodities are exported or “on arrival” in the US, subject to the commodity meeting strict quarantine packaging, transportation, and handling requirements. The eBeam facility of the National Center for Electron Beam Research at Texas A&M University is only one of 3 irradiation facilities that is approved by the USDA-APHIS for the phytosanitary treatment of regulated commodities on arrival in the U.S. This facility, in cooperation with the world’s largest retailer (Walmart, Inc.), treated millions of pounds of high value Mexican mangoes over 3 years using eBeam technology. The dose that was used was above 200 Gy and below 1 kGy (Fig. 7).

All of the market studies showed that the eBeam treatment was a superior treatment as compared to the legacy treatment of hot water for Mexican mangoes. The flavor profile of eBeam treated mangoes was better and the fruits could remain on the trees longer (since eBeam is gentler on the fruit compared to hot water). However, compared to hot water treatment, eBeam processing was costlier because of (a) the specialized transportation requirements, (b) paperwork to meet Mexican and US border personnel, and (c) eBeam processing costs. The mangoes were shipped to the facility as pallets. Since eBeam cannot treat entire pallets, the pallets had to be “broken down” and the single boxes had to be placed within carriers on the conveyor system for eBeam dosing (Fig. 7). There are significant labor costs associated with “breaking down” the pallet and for “re-building the pallets” for transport out of the facility after eBeam processing. These costs have to be reduced, to make the costs comparable to the current hot water treatment, to convince large retailers for wide scale technology adoption.

Based on an in-depth analysis of factors such as capital and operating costs, technology specifications (throughput, pallet level treatment, automation) it is clear that while eBeam processing of foods offers substantial benefits over status quo methodologies, the current design of eBeam facilities (built primarily for conventional meat pasteurization or medical device sterilization) are not suitable for phytosanitary treatment. With the construction of purpose-built facilities, eBeam phytosanitary treatment costs can be made comparable to or substantially lower than costs for hot water treatment. Table 9 below shows projected per-pound processing costs for low-dose phytosanitary treatment facilities.

Table 9 Projected annual eBeam throughput and costs (fixed and variable) for purpose-built phytosanitary treatment facilities.

<i>eBeam power (kW)</i>	<i>Calculated throughput</i>	<i>Variable costs</i>	<i>Fixed costs</i>	<i>Per pound processing costs (cents/lb)</i>
5	31,680 tons	\$485,000	\$2.34 million	4.55
20	126,720 tons	\$1.94 million	\$2.70 million	1.82
40	254,440 tons	\$4.84 million	\$2.94 million	1.36

Table 10 eBeam facility parameters by configuration.

<i>Parameter</i>	<i>5 kW</i>	<i>20 kW</i>	<i>40 kW</i>
Facility Cost	\$ 6.3 million	\$ 7.7 million	\$8.4 million
Annual amortized cost	\$ 1.6 million	\$ 1.9 million	\$ 2.1 million
Background power usage	4 kW	16 kW	32 kW
Hourly staff numbers	20	80	200
Annual production hours	2000	2000	2000

While facility fixed costs remain low even for high-powered facilities, variable costs (due to labor) rapidly eclipse the cost of equipment and electricity. These high labor costs are due to eBeam's primary limitation: the requirement for pallets of product to be broken down into individual boxes for processing due to the low penetrative power of eBeam phytosanitation. As facility throughput increases, tremendous work forces are required simply to de-palletize and re-palletize product. This work is physically demanding and must be maintained virtually non-stop to ensure that product continues to flow at a high rate through the electron beam. A breakdown of facility costs is included below (Table 10).

So, as facility size increases, facility staff requirements grow linearly with throughput, except for very large facilities where additional support staff and floor managers are required. While facility throughput can theoretically grow in linear proportion to beam power, this linear growth requires systemic changes to the design of a typical electron beam facility. Current electron beam facilities are designed with a single conveyor system for products. This conveyor then passes through the electron beam, and products receive a dose of irradiation. While this configuration is well-suited to applications where the power of the electron beam itself limits product throughput, the single-conveyor design imposes substantial throughput restrictions on products that require only a low dose of irradiation.

For example, in the National Center for Electron Beam Research at Texas A&M University, the 17 kW accelerator could theoretically process nearly 45,360 Kg (100,000 lbs.) of mango per hour. However, due to limitations on conveyor speed, this processing rate is reduced by almost 83% due to the inability of the conveyor system to "keep pace" with the maximum output of the electron beam accelerator. To rectify this problem, multiple (many) parallel conveyor systems are required. Instead of using a single conveyor, even the smallest of facilities (such as the 5 kW facility) would require four or more conveyors passing through a wide-aperture electron beam to take full advantage of the power output potential of the electron beam. In addition to changes to the conveyor system, reductions in the amount of labor required to de-palletize and re-palletize products would drastically improve per-pound processing costs.

Alleviating labor costs with mechanized or automated processes could result in even more significant cost savings. In our analysis, eBeam processing has the potential to significantly disrupt current industry norms for mango phytosanitation. However, it faces significant challenges to implementation as a result of the design of typical electron beam facilities that are not optimized for low-dose product processing at scale. To cost-effectively process mango, two major modifications are needed to conventional eBeam processing; namely, (1) implementation of many parallel conveyor systems in low-power facilities and (2) improved and/or automated de-palletization and re-palletization of product.

High power (~100 kW) X-ray systems for phytosanitary treatment are currently under construction in Australia and Mexico. Theoretically, the use of X-ray processing allows for pallet level irradiation treatment, thereby, avoiding the need for the excessive labor requirements that eBeam facilities need. However, the costs (capital and electricity costs) of X-ray systems can be high as compared to eBeam facilities. Also, since quick positive cash flow and a 3 to 5-year return of investment time frames are critical for the commercial success of such facilities, attention has to be paid to ensure that the facility is continuously operating. This means that there should be continuous supply of product to be treated. Given that many of these commodities are seasonal and grow at different locations within a country such as Mexico, strong cooperation among all entities in the supply chain is required. This cooperation should extend all the way from the choice of where the facility is built, the capacity, and guaranteed treatment volumes.

Location of phytosanitary treatment facilities

Globally, the use of these ionizing radiation technologies is growing the fastest for use in phytosanitary applications. The United States is the largest importer in the world of irradiated produce and Mexico is the largest exporter in the world of irradiated fresh produce. Given that phytosanitary treatment involves transboundary movement of the irradiated produce across international

borders, careful attention must be paid to the location of phytosanitary treatment facilities. Recent studies, using advanced operations management theory, have revealed the critical importance of cooperation among all the entities in the irradiated produce supply chain (Geismar et al., 2020). These studies show that with total cooperation among the different supply chain entities, overall profit increases by 8.6%. The concept of choosing the siting of phytosanitary treatment facilities based only on cursory maps of growing areas and transportation corridors and hubs can be fraught with serious financial ramifications.

Conclusions

The ionizing radiation technologies now available for the food industry include electron beam, (eBeam), X-ray and gamma irradiation. Gamma irradiation technology is slowly waning around the world. High energy eBeam (HEEB) technology is now slowly gaining market share. However, the need to bring the technology in-house has also catalyzed the commercial availability of low energy eBeam (LEEB) technology. X-ray technology-based food processing facilities are also being established to be able to treat entire pallets of food. The regulations in China and India favor large scale commercial adoption of the technology. In the United States, the sale of irradiated guavas and mangoes are expanding significantly. However, the regulations need slight modifications to enhance greater commercial adoption. Consumers do accept and purchase irradiated food all around the world. The future of this powerful technology looks promising in the food industry around the world.

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Modern Industry: Diagnostics and Process Control

J. Thereska^a and P. Brisset^b, ^a International Society for Tracer and Radiation Applications (ISTRA), Vienna, Austria; and

^b International Atomic Energy Agency, Vienna, Austria

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Abbreviations

CT	Computer tomography
DGNAA	Delayed gamma neutron activation analysis
E3M	Exploration, mining, mineral and metallurgical industries
GCT	Gamma computer tomography
NAA	Neutron activation analysis
NCS	Nucleonic control system
NG	Nucleonic gauge
NMR	Nuclear magnetic resonance
PEPT	Positron emission particle tracking
PGNAA	Prompt gamma neutron activation analysis
RPT	Radioactive particle tracking
RTD	Residence time distribution
SPECT	Single photon emission computed tomography
XRF	X-ray fluorescence

Introduction

Nuclear technologies such as radioactive tracers and nucleonic gauges have been widely used in various industries to optimize processes, improve product quality, save energy and reduce pollution release. Stimulated by an ever-increasing demand from large production plants, many nuclear techniques have been evolved to provide fast and effective solutions to plant and process problems.

The basics of how radiation can be used for diagnostics and control in modern industry are covered in this chapter.

Nuclear diagnostics and process techniques

Nuclear methods, which include nucleonic analysis and control systems, radiotracing technologies and other relevant nuclear technologies, are well suited to the optimization of a wide range of modern industrial processes (IAEA, 2004, 2005). Applications from single thickness and level gauges to complex devices in the exploration, mining, and metallurgical (E3M) industries are crucial in achieving huge economic and technical benefits both now and well into the future. The nuclear techniques being deployed and expanded include the use of radioactive tracer techniques for plant optimization, particularly in flotation and leaching, XRF for elemental characterization, and PGNAA for elemental analysis on bulk samples. Other techniques, currently deployed in the laboratory, may find plant use, e.g., NAA and NMR.

Major nuclear techniques applied for diagnostics and process control are presented in the Fig. 1 (IAEA, 2004).

Radioactive tracer methodology

The basic concept for radioactive tracer methodology is injecting a radionuclide into a stream of a moving fluid and employ radiation detectors downstream to detect the movement of the subject stream. Fig. 2 illustrates the tracer injection into a process vessel (IAEA, 2008a,b—TCS-31).

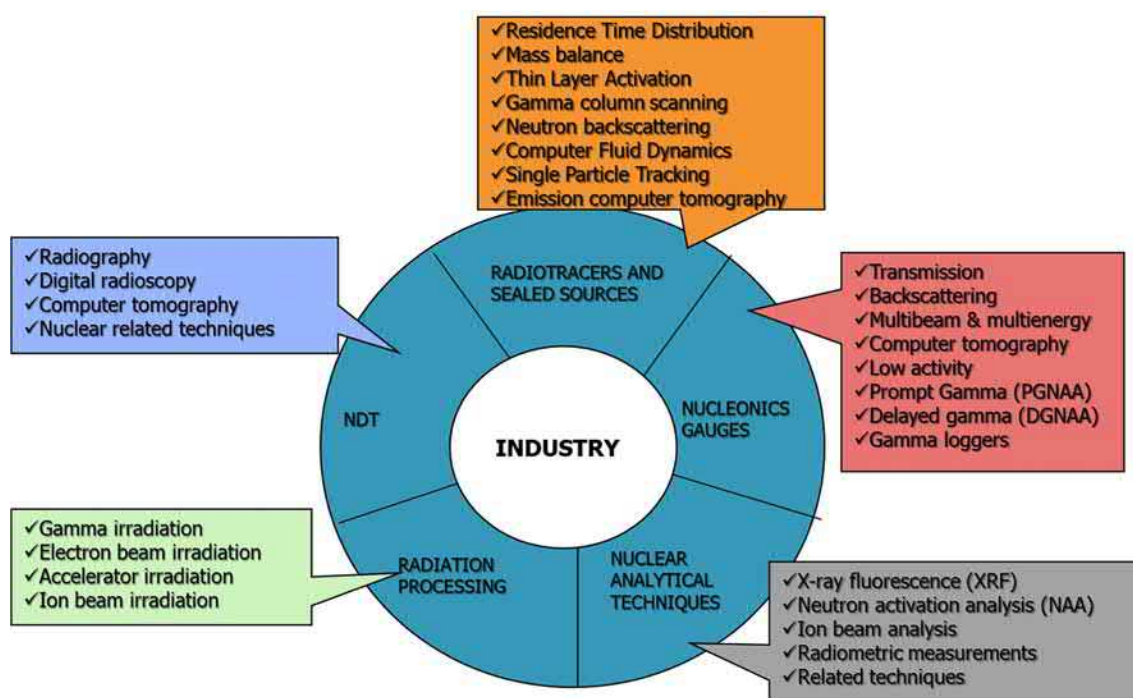


Fig. 1 Major nuclear techniques applied for diagnostics and process control (IAEA, 2004).

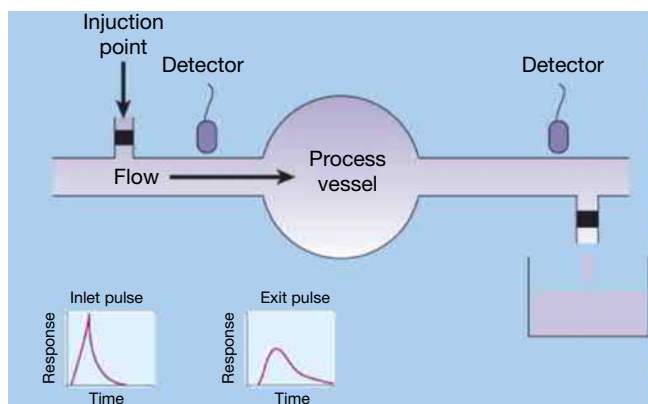


Fig. 2 Tracer principle (IAEA, 2008a,b—TCS-31).

A tracer is any substance whose atomic or nuclear, physical or chemical properties provide for following the behavior of various physical or chemical processes. Although there are many kinds of tracers, radioactive tracers are the only ones used for online diagnosis of industrial reactors.

Radioactive tracer methodology allows the hydrodynamics of a process to be determined through the observation of the behavior of a tracing substance incorporated to the process, during its progress into or at the outlet of the system (Charlton, 1986; IAEA, 2011).

Residence time distribution (RTD)

The residence time distribution (RTD) is extremely pertinent to the operation of chemical reactors, influencing both the throughput and the quality of the product (IAEA, 2008a,b—TSC-31). The residence time distribution (RTD) measured online by a radioactive tracer is becoming an important tool for the diagnosis of industrial processing units (Fig. 3). The experimental RTD curve provides many indications for reactor troubleshooting, like parallel flows, dead space, bypass or holding up. There are many kinds of tracers; radioactive tracers are the only ones used for online diagnosis and process control (IAEA, 2008a,b—TSC-31).

Using RTD to determine hydraulic detention times in wastewater treatment ponds

The operation of a wastewater treatment installation can be deceptively complex. Given the unsatisfactory state of current theoretical approaches, there is a need to be able to assess performance practically; that is, by measuring the RTD. Modeling by RTD will be a powerful tool, aiding both the design and performance optimization of wastewater treatment systems (IAEA, 2011).

The benefits are:

- ensuring thorough treatment of wastewater, thereby safeguarding the environment,
- operating existing ponds more effectively- saving money,
- providing data for the design of future ponds.

Leak testing using radioactive tracer methods

Leak testing using radioactive tracer techniques are probably the most widespread applications of radioactive tracers in industrial troubleshooting (IAEA, 2009). Radioactive tracer techniques for leak testing are applied in petroleum and chemical process industries for both identifying, locating and quantifying the leakage in pressured vessels, heat exchangers and pipeline systems in industrial process plants.

Leaks in heat exchangers are more frequent problems in many processing plants of the petrochemical industry (Fig. 4). Radioactive tracers are very efficient and most competitive for detecting small leaks inside the heat exchangers. Detection limits as low as 0.5% of stream flow can be achieved. Radioactive tracer methods are also very effective when the pipes are buried.

Advantages of radioactive tracer technology

The main advantages of radioactive tracers compared to conventional tracers are (IAEA, 2008a,b):

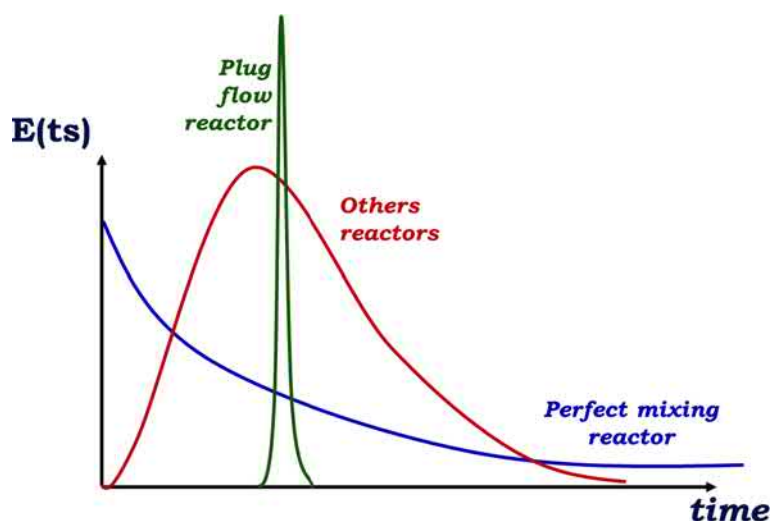


Fig. 3 Residence time distribution (RTD) for troubleshooting (IAEA, 2008a,b—TSC-31).

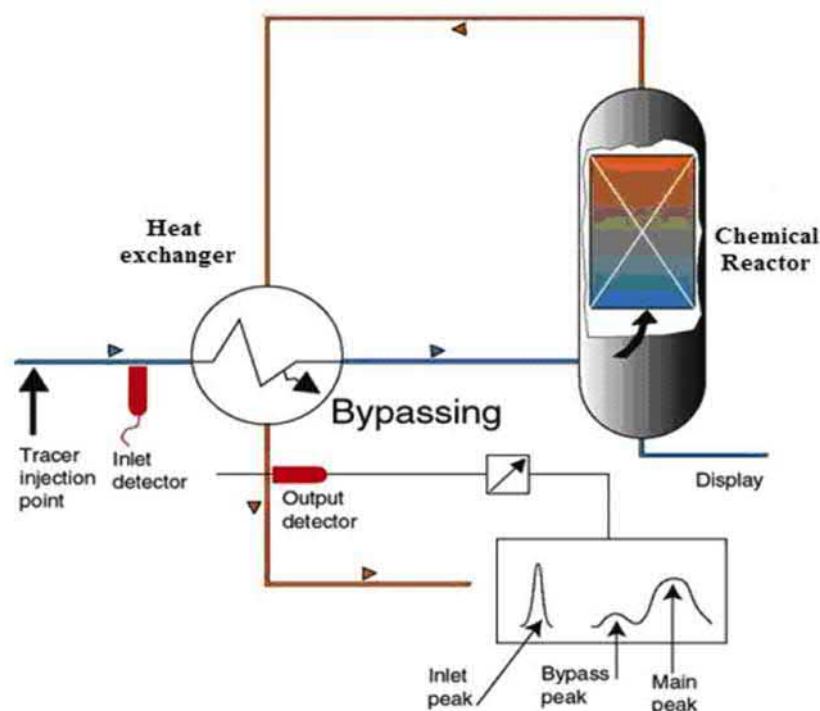


Fig. 4 Leak detection in a heat exchanger using radioactive tracer method (IAEA, 2009).

- Radioactive tracers have high detection sensitivity for extremely small concentrations. For instance, some radionuclides may be detected in quantities as small as 10^{-17} g.
- The amounts of radioactive tracers used are virtually insignificant. For example, 1 Ci of $^{131}\text{I}^-$ weighs 8 μg , while 1 Ci of $^{82}\text{Br}^-$ weighs only 0.9 μg . That's why, when injected, they do not disturb the dynamics of the system under investigation.
- Radioactive tracers offer the possibility of in situ measurements, from the outside of a pipe or vessel. This is of special importance for many opaque industrial facilities.

Today, most of the radioisotopes employed as radioactive tracers are still produced in nuclear reactors. However, cyclotron produced radioactive tracers are increasing in frequency. Iodine-123 is an example of a cyclotron produced radionuclide with widespread availability. Radioisotope generators have been found to be particularly suitable for radioactive tracer applications in industry (IAEA, 2013).

Table 1 contains a collection of some of the most frequently used radioisotopes for tracer technology.

Nucleonic gauges

Nucleonic gauges are nuclear instruments applied for process control and analysis (IAEA, 2005). The most important applications for them are in mining and mineral processing for in situ and online elemental analysis. These groups of nucleonic gauges are more specifically named as nucleonic control systems (NCS). There are several ways of applying the NCS, among them:

- ✓ on-line (process),
- ✓ off-line (process),
- ✓ in situ (well logging),
- ✓ used in laboratory (on samples), and
- ✓ portable, for site measurements.

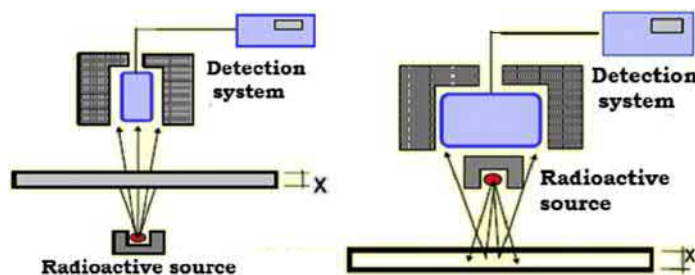
The principle of nucleonic gauges is given in Fig. 5.

Whenever a radiation beam interacts with matter, a fraction of it is transmitted, a fraction absorbed, and a fraction is scattered from its original path. In a transmission gauge, the media to be measured is placed between the radioactive source and the detector (Fig. 5, left). The radiation intensity in the detector is a function of several parameter characteristics of the material.

In a backscattering gauge, the scattering radiation (with angle greater than 90 degrees) emerges back from the investigated matter (Fig. 5, right) with degraded energy and intensity (count rate) characterizing the bulk density and the average chemical composition of the matter.

Table 1 Major radioisotopes used as tracer in industry (IAEA, 2008a,b—TCS-31).

Isotope	Half-life	Main energy (MeV)	Chemical form	Tracing of phase
Tritium (H-3)	12.6 years	Beta, 0.018(100%)	Tritiated water	Aqueous phase
Sodium-24	15 h	γ : 1.37 (100%) 2.75 (100%)	Sodium carbonate	Aqueous phase
Bromine-82	36 h	γ : 0.55 (70%) 1.32 (27%)	Ammonium bromide, <i>p</i> -dibrom-benzene, Dibrobiphenyl CH ₃ Br	Aqueous phase Organic phase Organic phase Gas phase
Gold-198	2.7 days	γ : 0.41 (99%)	Chloroauric acid	Solids/aqueous phases
Iodine-131	8.04 days	γ : 0.36 (80%) 0.64 (9%)	KI and NaI, Iodobenzene	Aqueous phase Organic phase
Iodine – 125	59.4 days	γ : 0.035	KI and NaI, Iodobenzene	Aqueous phase Organic phase
Iodine-123	13 h	γ : 0.159	KI and NaI, Iodobenzene	Aqueous phase Organic phase
^{99m} Tc from ⁹⁹ Mo/ ^{99m} Tc generator	6 h	γ : 0.14 (90%)	Sodium pertechnetate (TcO ₄ ⁻)	Aqueous phase
^{137m} Ba from ¹³⁷ Cs/ ^{137m} Ba generator	2.55 min	γ : 0.661	^{137m} BaCl ₂	Aqueous phase
^{113m} In from ¹¹³ Sn/ ^{113m} In generator	99.5 min	γ : 0.392	^{113m} InCl ₃ ^{113m} In-EDTA	Solid phase Aqueous phase
⁶⁸ Ga from ⁶⁸ Ge/ ⁶⁸ Ga Generator	68.3 min	γ : 1.08	⁶⁸ GaCl ₃ ⁶⁸ Ga-EDTA	Solid phase Aqueous phase
Xenon-133	5.27 days	γ : 0.08 (100%)	Xenon	Gas phase
Krypton-79	35 h	γ : 0.51 (15%)	Krypton	Gas phase
Argon-41	110 min	γ : 1.29 (99%)	Argon	Gas phase

**Fig. 5** Principles of nucleonic gauges, left transmission and right backscattering (IAEA, 2005).

Nucleonic gauges (NG) or nucleonic control systems (NCS) have been operating throughout the industrial world for quite some time. They are used by various industries to improve the quality of products by optimizing and controlling processes, saving energy and materials.

Neutron and gamma ray techniques are largely used for online analysis and control in:

- coal preparation
- coal-fired power stations
- cement manufacturing
- mineral processing and metal production
- oil industry
- density/thickness

Most of NCS are based on a few most common nuclear techniques:

- natural gamma-ray
- single energy gamma ray transmission
- dual energy gamma ray transmission
- gamma ray backscattering
- beta ray transmission
- neutron transmission
- neutron back scattering

- prompt gamma neutron activation analysis (PGNAA)

Many nucleonic gauges are now commercially available from several manufacturers. The development of supporting technologies such as compact electronics, fast computers, high-resolution detectors, small reliable neutron tubes, and dedicated computer modeling codes has resulted in increased technical viability and economic acceptability of nucleonic gauges. The development of the next generation of nucleonic devices is still taking place.

Relevant target areas are defined in international priority industrial sectors, such as oil and gas production, mining and mineral ore processing, environmental monitoring, cement and civil engineering industries, where the benefit is enormous, and the technology competes well with conventional techniques.

Industrial process gamma tomography

Industrial process gamma tomography has recently become popular in many laboratories. The gamma computer tomography (GCT) technique is the most suitable technique for visualizing the equipment contents of an opaque multiphase mixture. There are two gamma GCT methods (IAEA, 2008a,b—TECDOC-1589).

Gamma transmission tomography: The gamma transmission tomographic technique is based on the assumption that there is heterogeneity of a parameter in the system under investigation. In general, this parameter is the density of the medium. For example, determining the density of a packed bed or the concentration of a phase in a dual phase flow. There are many geometrical arrangements of source and detector(s). The simplest of these is the parallel, or pencil beam, configuration in which a source emitting a single pencil beam of radiation is coupled to a radiation detector, and this system is moved up and down surrounding the item under investigation (Fig. 6).

More complex gamma tomography systems also exist. The configuration depends upon the objectives in terms of space and time resolution. Fig. 7 shows a CT system with one ^{137}Cs source and 32 NaI (Tl) detectors (IAEA, 2009).

Gamma emission tomography: To investigate flow dynamics in multiphase flows, with or without density variations, it is generally better to use a radioactive tracer, labeling one phase and observing the behavior of this phase in the flow. Measuring radioactive tracer gamma radiation in a tomographic configuration allows the visualization of the flow patterns inside vessels to become possible.

Fig. 8 shows a gamma emission CT; 36 small NaI detectors are placed around the pipe. The radioactive tracer Tc-99 m is used and, as result, the flow image inside the pipe is obtained.

Typical fields of applications

Exploration, mining, mineral and metallurgical (E3M) industries are essential to the progress of modern society. The mining and mineral industries have contributed value-added services to the economy of many developing and developed countries. Securing a sustainable supply of raw materials is a key priority for every nation and becomes increasingly important to their economic

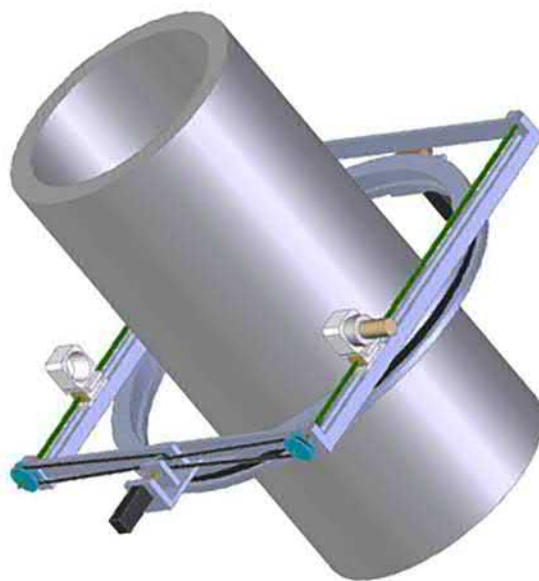


Fig. 6 Gamma transmission CT with 1 S—1 D (IAEA, 2008a,b—TECDOC-1589).

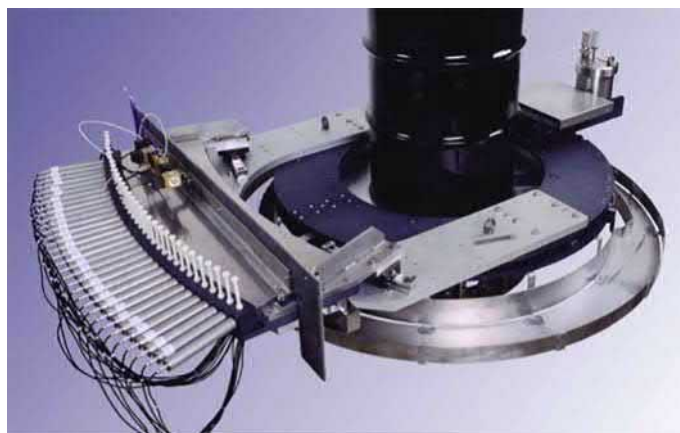


Fig. 7 CT system with one ^{137}Cs source and 32 NaI (TI) detectors (IAEA, 2008a,b—TECDOC-1589).

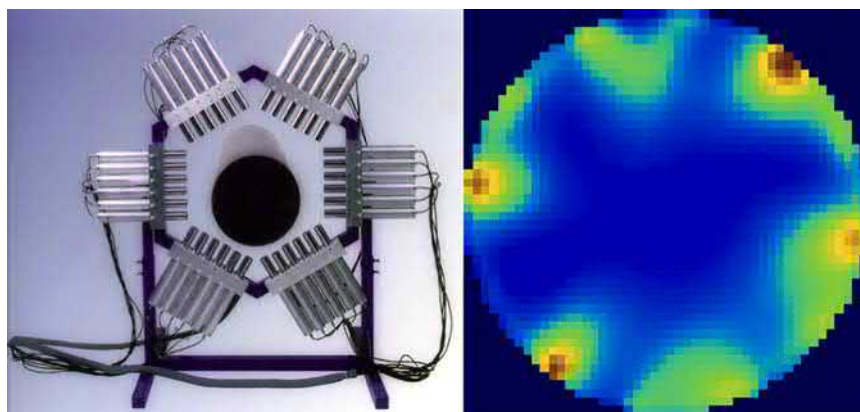


Fig. 8 Gamma emission CT (IAEA, 2008a,b—TECDOC-1589).

growth. Recent years have witnessed unprecedented growth in demands for raw materials worldwide, mainly due to rapid industrialization of emerging economies and continued material consumption (Cutmore, 2018).

In order to increase the production rates, exploitation has become more intensive over the years, which creates a variety of impacts on the environment before, during, and after mining operations. Easily accessible high-grade ore deposits are becoming less and less available over time. In many cases, high quality ores have largely been exploited and the ores remaining require more complex processing. As a consequence, mining operation has to be directed toward lower grade, deeper and/or more complex ores, which are considerably more difficult and expensive to extract. This has also led to the interest of reprocessing tailing materials.

Across the “exploration-to-product” chain, selection of the appropriate technologies for optimal recovery of the products within the process environment, their optimization for a particular system, compliance with statutory regulatory environmental conditions, as well as their implementation, are viewed as necessities.

Nucleonic gauges in mining and processing

Worldwide, new strategies and new technologies are developing and being implemented to create a translational shift in mining practice. The sensing techniques deployed and under development are wide and include both nuclear and nuclear related techniques. Some nuclear techniques are well suited to the analysis of bulk materials and play a key role in providing rapid sensing as part of an advanced systems for smart mining (Cutmore, 2018).

Neutrons provide the basis of several analysis techniques. Raw material can be analyzed by nuclear reactions, such as prompt gamma neutron activation analysis (PGNAA) and delayed gamma neutron activation analysis (DGNAA) (Fig. 9).

In the past, radioisotopes such as $^{241}\text{Am-Be}$ and ^{252}Cf have been used as source of neutrons, but these isotopes are now being replaced by intrinsically safer neutron generators (IAEA, 2012).

The PGNAA reaction (n, γ) usually happens with slow neutrons. When a slow neutron enters a material, the nucleus in the material can absorb the neutron, resulting in a compound nucleus with an excited energy that is identical to the sum of binding energy

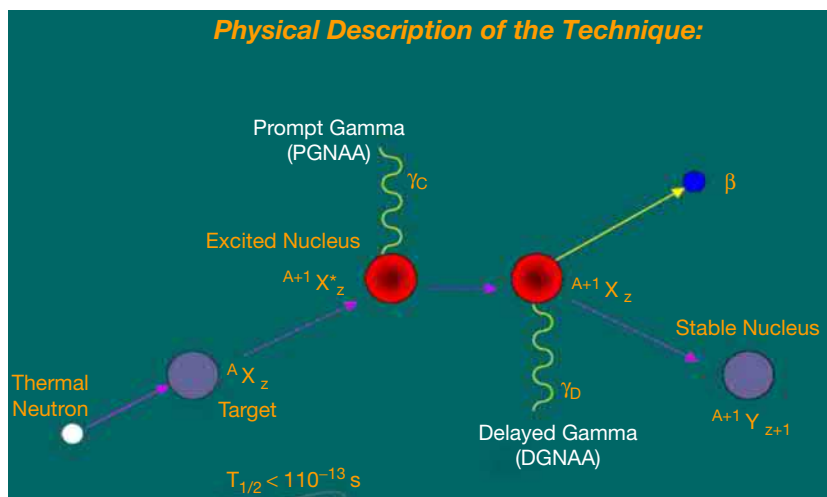


Fig. 9 Principles of prompt gamma neutron activation analysis (PGNAA) and delayed gamma neutron activation analysis (DGNAA) (Sowerby and Cutmore, 2008).

and the kinetic energy of the neutron. This compound nucleus then proceeds to reach its ground state by emitting a prompt gamma-ray in $\sim 10^{-12}$ s. The prompt gammas can be detected, which are characteristic of the emitting element.

NAA can be caused by either slow or fast neutrons. Reactions with the target materials and fast neutrons, such as $(n,p\gamma)$, $(n,n\gamma)$, $(n,\alpha\gamma)$ produce the radioactive daughter nuclei, which is then used to find out characteristics of the target materials. PGNAA techniques have been used in field instruments, and NAA in Laboratory based systems.

Another technique about to come available to the mining industry to determine ore characteristics is Nuclear Magnetic Resonance (NMR). This technique has been widely used in medicine and other industries but not the mining industry. A radiofrequency (RF) pulse at a specified frequency is emitted into the ore, the frequency matches a resonance of a target within the ore, and an echo pulse results that in magnitude indicates the amount of target material within the ore.

This technique offers the possibility of bulk ore sorting because of the speed of the measurement. An accurate measurement can be made in less than 5 s. This would be transformational for mining. By placing a sensor onto a conveyor belt as close to the mine as possible, the sensor could first determine the average grade of the ore. The next step would use the information gained to divert the ore into three separate streams. For example, (1) a low-grade ore is tagged as waste, (2) a high-grade ore for immediate downstream processing, and (3) a medium-grade ore for stockpiling and future downstream processing. Such an analyzer has been developed for copper mines and first deployments are expected soon.

Some examples of nucleonic control systems (NCS) for online raw material elemental analysis are presented in Fig. 10.

Neutronic gauges for monitoring processes

Table 2 contains a listing of some of the more common applications.

Nucleonic gauges in paper and sheet metal manufacturing

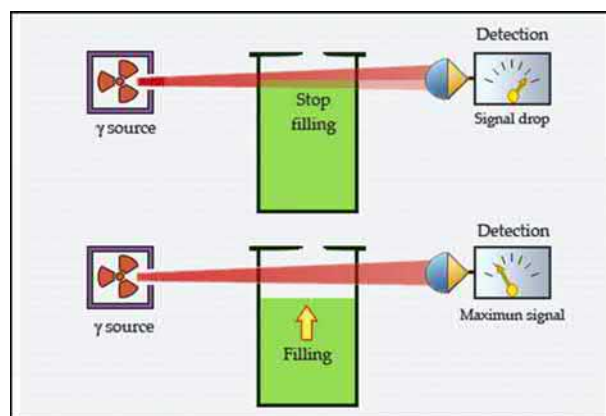
Simple nucleonic gauges are used routinely to measure the level of liquids or solids contained in vessels in a broad range of industries, especially in temperature and high-pressure conditions. The principle of gamma-ray level gauges employed in chemical, petrochemical, petroleum, paper/pulp, and iron/steel industries is shown in Fig. 11 (IAEA, 2005).



Fig. 10 Nucleonic control systems for online raw material elemental analysis in cement and mineral ore processing (Cutmore, 2018).

Table 2 NCS applied to the exploration, exploitation and processing of natural resources (IAEA, 2005).

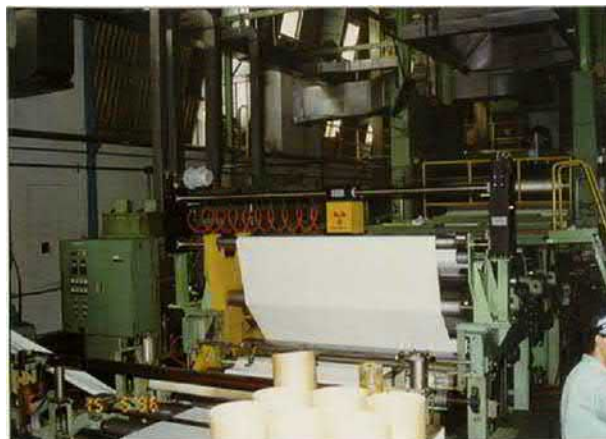
Techniques	Fields of application					
	Oil and gas		Logging		Mineral processing	
	Fluid flow	Delineation of deposit	In situ assaying	On and off belt analysis	Slurry analysis	Density, weight, level/fill
Natural γ radiation		X	X	X	X	X
γ -ray transmission	X			X	X	X
γ -ray backscatter		X	X	X	X	X
PGNAA and DGNAA		X	X	X		
XRF analysis			X		X	

**Fig. 11** Principle of a filling control (IAEA, 2005).

Nucleonic gauges are employed for measuring the thickness or area weight paper and metal processing industries (Fig. 12).

Soil compaction

One of the many pleasures that many of us now enjoy are good roads and highways upon which cars and trucks can traverse—even at high speeds. The key is developing a solid foundation before applying a topcoat of concrete or another preferred surfacing material. One of the principal ways to provide solid footings is to use radiation gauges to determine both the density and the moisture content of the soil beneath the foundations.

**Fig. 12** Principle of thickness monitoring using a transmission nucleonic gauge (IAEA, 2005).

A gamma source such as cesium-137 or cobalt-60 is often used for the applicable radiation gauge to determine soil density. For soil compactness measurements, the gamma source is lowered into a small test hole and the density of the soil is determined by the number of gamma rays reaching the detector at the surface. If the objective is to measure the density of thin layers of asphalt or concrete in road construction, the gamma source is placed on the surface of the layered material and the density is determined by the number of backscattered gamma rays.

For determining moisture in the soil, a neutron source, either californium-252 or a mixture of americium-241 and beryllium, is normally used. Fast neutrons, directed into the soil from the surface, are slowed after colliding with the hydrogen atoms present in the soil. A detector at the surface then records the number of thermalized (slowed) neutrons, which relates directly to the amount of soil moisture.

Mass flow dynamics in a copper melting process

In the shaft process of copper production, normally the raw materials—copper ore, concentrate and coke—are loaded continuously into the upper part of the shaft furnace. The slag and copper-matte flow from the shaft furnace into the settling tank, where the heavy phase-matte is separated from the slag (Fig. 13). For this example, a non-negligible decrease of copper recovery was observed in the copper processing line. Accordingly, a radioactive tracer experiment was incorporated to diagnose the hydrodynamics of the copper processing line (Plasari et al., 1999).

In this copper processing line, gold and cobalt are produced as copper by-products as well. Both are elements with industrial interest in copper raw mineral. The recovery of gold and cobalt was not known previous to the employment of nuclear processes. After this discovery, the radioactive tracer experiment was upgraded to measure their recovery as well. In this case, the cobalt radioactive tracer was employed with a double goal: to diagnose the process (through RTD measurement) and to measure the cobalt recovery (using mass balance method) as well. In solving this problem, the radioactive tracer technique was quite successfully applied. The Cobalt –58 radioisotope was used as the radioactive tracers in two chemical forms ^{58}CoO and ^{58}CoS (confer Figs. 14 and 15).

Fig. 14 shows the experimental RTD curves of both radioactive tracers in the shaft furnace were practically the same (were well superposed upon one another).

Fig. 15 represents the experimental RTD curves at the outlet of the settling tank and the model.

The experimental RTD curves indicate the existence of a lag (delay) and —a short circuit.

From the mean residence time of the RTD model, it was also estimated that nearly 30% of the volume of the settling tank was blocked by the solidified material. The decrease in the active volume could be one of the reasons for the short circuit. The other, not

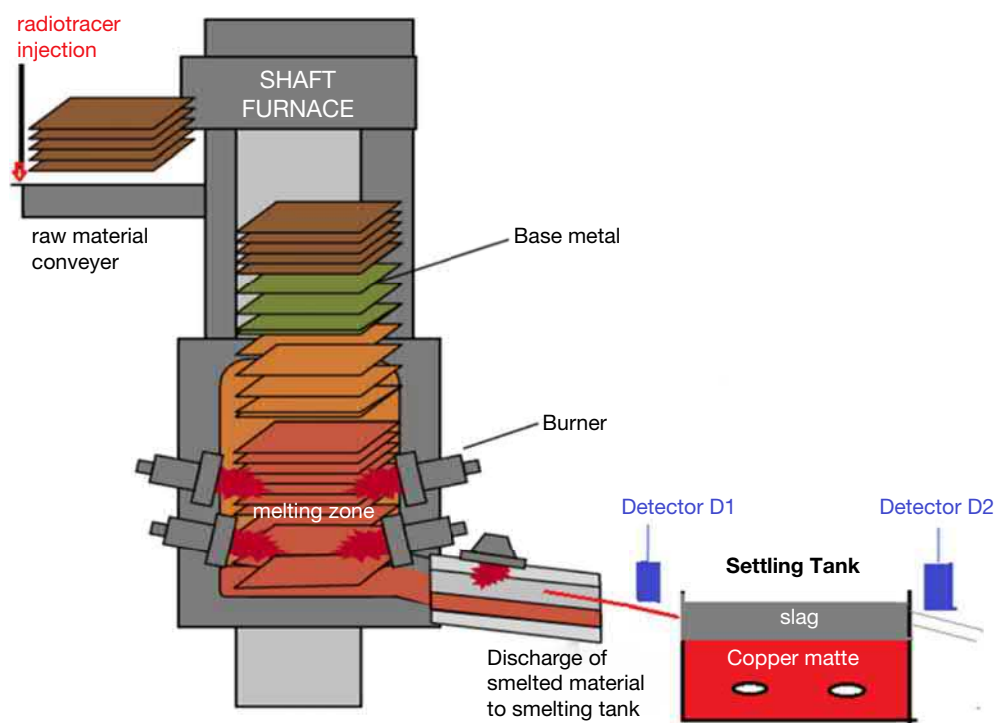


Fig. 13 A shaft furnace and settling tank where a radioactive tracer test was carried out. D1 and D2 radiation detectors were installed at the outlet of shaft furnace and settling tank (Plasari et al., 1999).

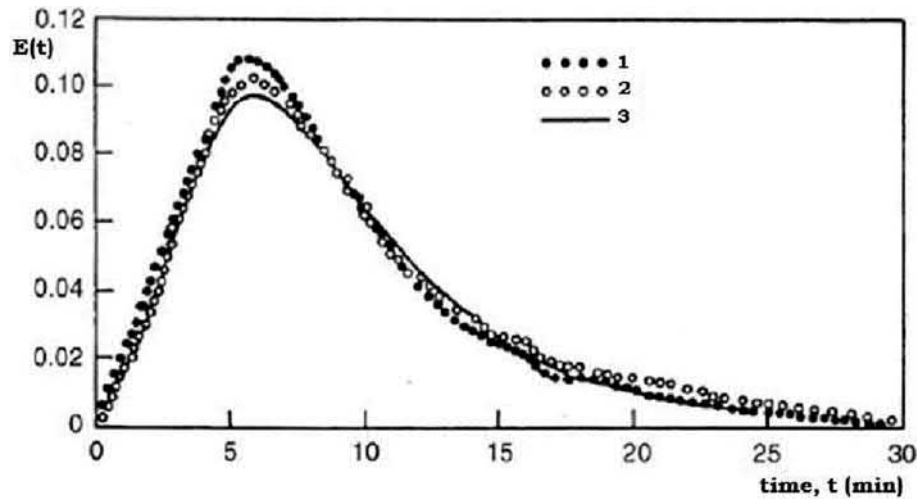


Fig. 14 Experimental RTD curves of ^{58}CoO and ^{58}CoS at the outlet of the shaft furnace (1— ^{58}CoO , 2— ^{58}CoS , 3—model) (Plasari et al., 1999).

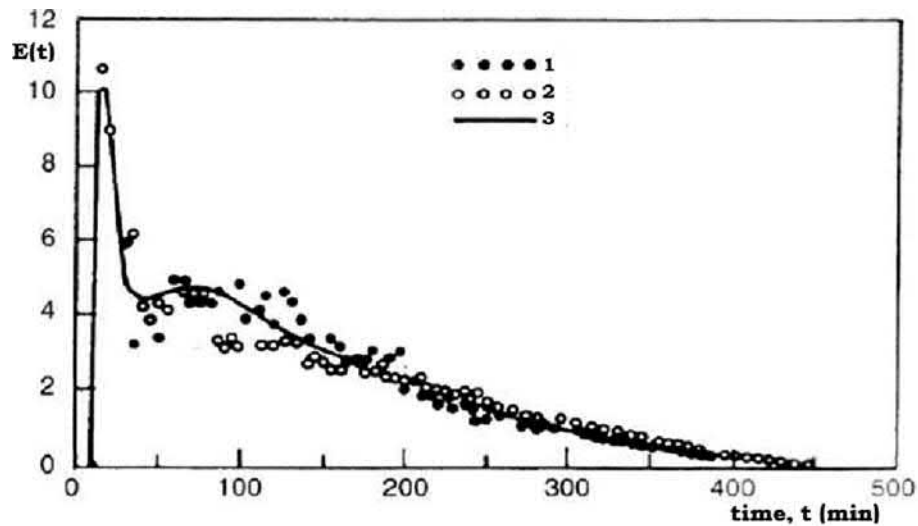


Fig. 15 Experimental RTD curves of ^{58}CoO and ^{58}CoS at the outlet of the settling tank (1— ^{58}CoO , 2— ^{58}CoS , 3—model) (Plasari et al., 1999).

less important response, was likely the geometrical form of the settling tank. The short circuit explains the decrease in copper recovery. In order to eliminate the short circuit, it was recommended the settling tank be reshaped.

Radioactive tracers in mineral processing

The objectives of radioactive tracer investigations in mineral ore processing industry include:

- ✓ troubleshooting
- ✓ process control and optimization
- ✓ investigation of flow patterns
- ✓ investigation of design and scale-up studies
- ✓ development/verification mathematical models.

There are certain processes found throughout mineral processing industries where radioactive tracers can be applied:

- ✓ grinding
- ✓ classification
- ✓ flotation

✓ homogenization

The most cost-intensive stage in a mineral processing plant is grinding, especially grinding, in which the ore particles are reduced in size by a combination of impact and abrasion, most commonly as a slurry.

Radioactive tracers in flotation

The use of radioactive tracers has proven to be a powerful tool to study the hydrodynamic behavior of large flotation machines (Lelinski et al., 2002). Mean residence time in pulp and froth zones can be evaluated from RTD measurements for liquid, solids and gas radioactive tracers. Relevant parameters such as mixing regime, mixing time, flow distribution in parallel flotation banks, gangue and gas entrainment, and particles segregation, have been evaluated using non-invasive sensors for radiation detection without disturbing the flotation operation. This information is fundamental for improving flotation machines operation, control and optimization.

Different radioactive tracer techniques have been used to measure a number of internal characteristics and the hydrodynamic behavior of flotation machines such as:

- residence time distribution (RTD) of liquid, solid and gas phases, in industrial cells and columns.
- real mean residence time evaluation.
- mixing regime in single cells, banks of cells and pneumatic columns.
- froth mean residence time of liquid, floatable and non-floatable solids.
- mixing time and internal pulp circulation in large industrial self-aerated cells.
- gas holdup and gas residence time distribution in flotation machines.
- flotation rate distribution.

Radioactive tracers in metallurgy

The critical raw materials, due to their high economic importance, include antimony, beryllium, cobalt, fluorspar, gallium, germanium, graphite, indium, magnesium, niobium, platinum group metals, rare earths, tantalum, and tungsten. Several other materials such as aluminium, bauxite, chromium, iron, magnesite, manganese, molybdenum, nickel, rhenium, tellurium, vanadium, and zinc may become critical as well into the near future.

The two main methods employed in order to process the ores for metal production are hydrometallurgy and pyrometallurgical process (IAEA, 2008a,b—TCS-31).

Hydrometallurgy is a technique of extractive metallurgy involving aqueous chemistry for the recovery of metals from the various raw materials such as low-grade oxide ores, concentrates, post-pyrometallurgy process tailings, electronic waste and recycled or residual materials, and other sources considered as a waste.

Radioactive tracer methods are the suitable tool for all hydrometallurgical process investigations. They provide important parameters such as:

- flow dynamics of hydrometallurgical systems,
- separation efficiency,
- process kinetics,
- process optimal time,
- agitation rate,
- liquid to solid ratio.

The *pyrometallurgy* processes involved in the mineral processing are based upon chemical reactions carried out at temperatures ranging from 500 °C to 3000 °C. The different chemical reactors that are used conventionally to carry out this process are shaft furnace, muffle furnace, hearth furnace and blast furnace. These processes may develop malfunctions with time—especially when the yield of product decreases. A huge amount of energy loss will occur if the process is carried on inefficiently in a malfunctioned reactor. Therefore, non-invasive, accurate and online techniques are required to monitor and identify any malfunction present in the pyrometallurgical reactors. Nuclear techniques (such as radioactive tracer and gamma ray densitometry) are the most suitable in these high density, large scale systems to provide various process information in the shortest possible time.

The most efficient nuclear techniques used in a laboratory setting for scaling up the design of pyrometallurgical reactors are Positron Emission Particle Tracking (PEPT) and Radioactive Particle Tracking (RPT) (IAEA, 2004). Fig. 16 illustrates the RPT method.

NCS in mining and mineral processing

Mining and mineral processing industries are the main end users of nuclear methods. Nucleonic control systems (NCS) are perhaps the most extensively employed (Cutmore, 2018). Fig. 17 illustrates some NCS applications in mining and processing industries.

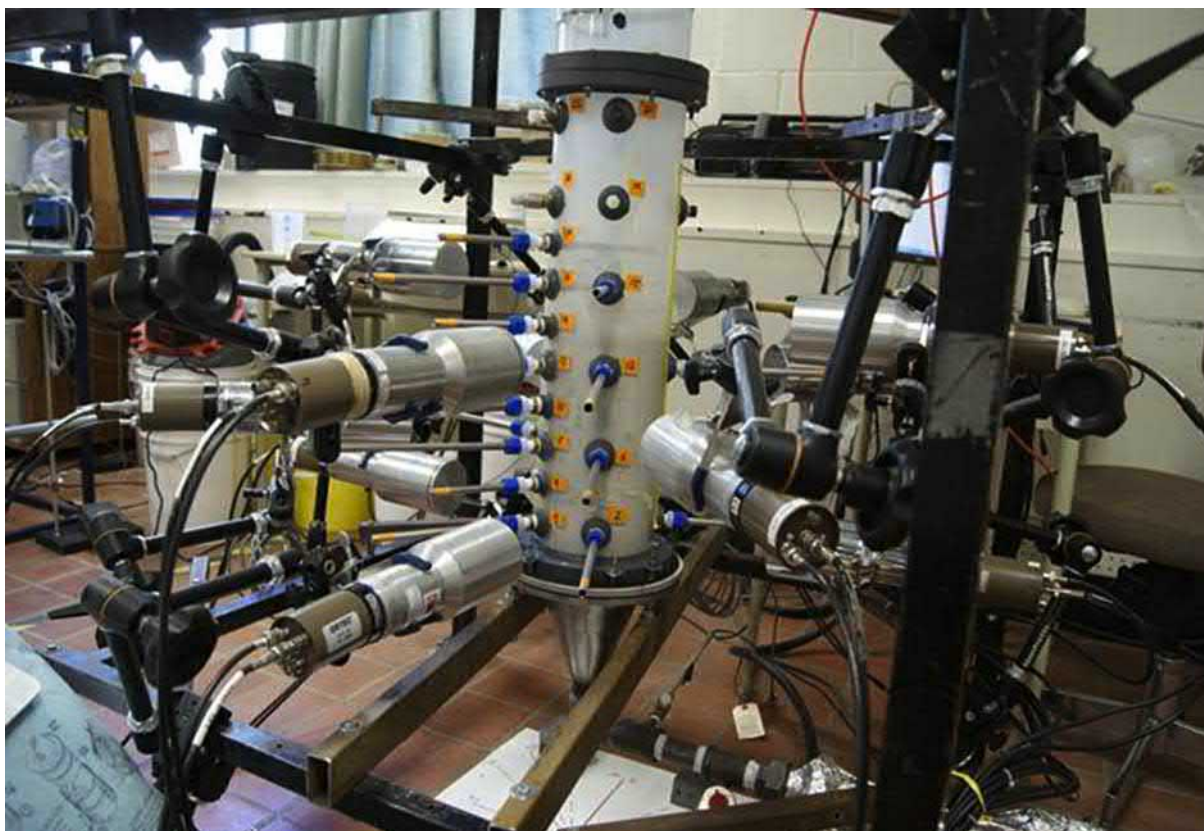


Fig. 16 Principle of radioactive particle tracking (RPT) method (IAEA, 2004).

Borehole logging in search of materials

Fig. 18 (left), shows a borehole logging process for copper ore analysis in a Chile open cut copper mine using a PGNAAP probe. The calibration curve shows good correlation in the full range of concentrations from 0.2 up to 2% copper (**Fig. 18**, right). Currently, the main interest for this method focuses on the range of concentration less than 1%, in particular around 0.5%.

Looking ahead

- Nuclear techniques are now conducted internationally with bases worldwide.
- Many nuclear applications are routinely carried out on petrochemical plants, but there is now extensive involvement upstream in refining and oil/gas production.
- The current amount of business not related to oil and gas is still rather small. Many applications, particularly those offline, involving sampling, are now carried out using conventional tracers.
- However, radioactive tracers are used widely for detailed residence time and distribution studies on complex units—made possible by the availability of increasingly sophisticated data acquisition and processing systems.
- Simple nucleonic gauges are still in use for monitoring processes in industrial harsh conditions.
- There is ongoing development in designing and manufacturing complex nucleonic control systems for online applications in exploration, mining, mineral and metallurgical (E3M) industries.

Predictions for the future

- Nuclear techniques will continue to constitute an important role in Modern Industry for Diagnostics and Process Control.
- Radioactive tracer studies will become increasingly wider in scope and more detailed.
- Techniques offering process visualization, such as tomography, will be widely used.
- Neutron techniques for inspection and analysis will continue to be refined and developed.

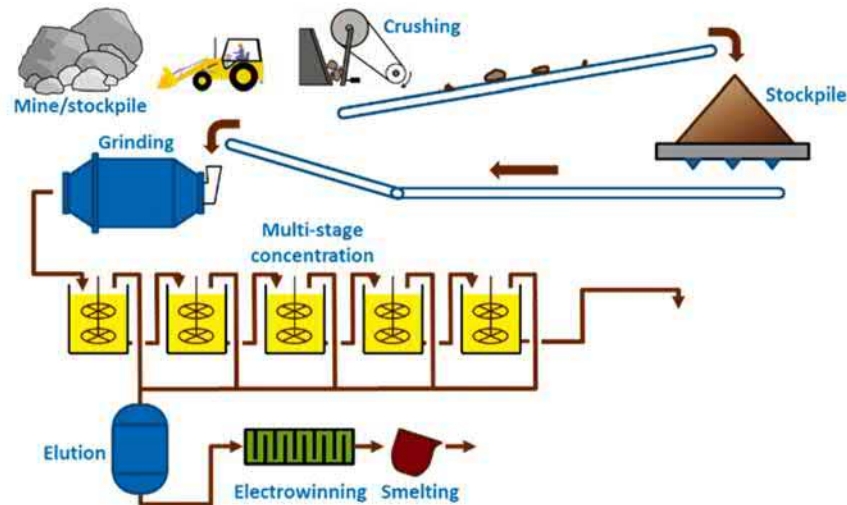


Fig. 17 Some typical NCS applications in mining and processing industries (Cutmore, 2018).

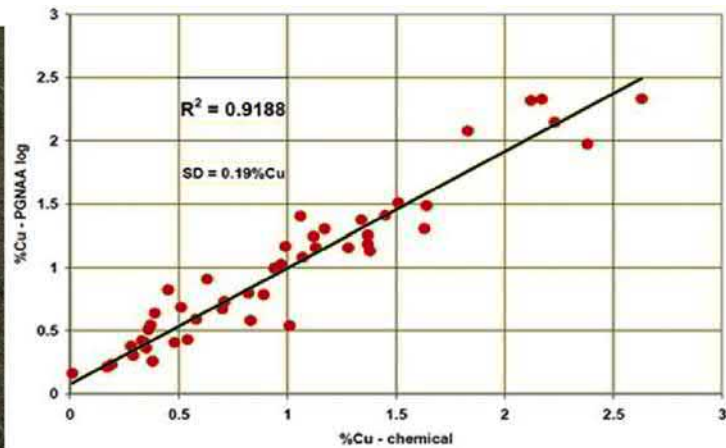


Fig. 18 Borehole logging process for cooper ore analysis in an open cut cooper mine using PGNA probe (left) and the calibration curve (right) (Sowerby and Cutmore, 2008).

- There will be an increased uptake of the technology by industries other than oil and gas. Minerals processing appears to offer the most promise.

We should not let unjustified fear of radiation create obstacles to continued progress and benefits. There remains a clear need for more public awareness. Adherence to responsible safety guidelines remains a necessity for the application of these valuable nuclear techniques to be successfully employed.

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Modern Industry: Gamma Methods for Materials Testing and Inspection

Tor Bjørnstad, Institute for Energy Technology, Kjeller, Norway; and University of Oslo, Oslo, Norway

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Glossary

Bq Becquerel, – SI unit of radioactive disintegration corresponding to 1 dps (disintegration per second) with dimension s^{-1} .

Ci Curie, – previous unit of radioactive disintegration, originally corresponding to the disintegration rate of 1 g of ^{226}Ra but later modified to 3.7×10^{10} dps. It was officially replaced by Bq in 1975 but is still in use in technical texts. Thus, $1 \text{ Ci} \equiv 3.7 \times 10^{10} \text{ Bq}$, conversely $1 \text{ Bq} = 2.7 \times 10^{-11} \text{ Ci}$.

Civil Engineering A professional engineering discipline that deals with the design, construction, and maintenance of the physical and naturally built environment, including public works such as roads, bridges, canals, harbors, dams, airports, sewerage systems, pipelines, structural components of buildings, and railways.

Compton Scattering The inelastic scattering of a photon (which may be an X-ray or gamma ray photon) by a charged particle, usually an electron in an atom. The incident photon is deflected through an angle with respect to its original direction and continues in the new direction with lower energy (Compton effect).

Condition Monitoring Monitoring of the performance of processing units (multiphase separators, scrubbers, heat exchangers and similar) relative to the design criteria.

Distillation Column Scanning Identification of the internal condition of crude oil distillation unit (column) using a gamma transmission scanning technique.

Distillation Tower Identical in basic operation to distillation column or fractionating column in general. This is an essential item used in the distillation of liquid mixtures to separate the mixture into its component parts, or fractions, based on the differences in volatilities. In the petroleum industry, these columns constitute dimensions (several tens of meters) and appear as towers.

dps Disintegrations per second, dimension s^{-1} .

eV Electron volt, – energy unit used in nuclear chemistry and nuclear physics to characterize the energy of nuclear radiation and the energy of accelerated charged particles. $1 \text{ eV} \equiv$ kinetic energy gained by one electron accelerated in a potential of 1 Volt.

Gamma Emission The process whereby an excited atomic nucleus emits gamma radiation, often after beta or alpha decay, but also after having been excited by interaction of a colliding particle like a neutron or a proton.

Gamma Radiation Gamma radiation or gamma rays are high-energy photons that are emitted by decay of excited atomic nuclei. Gamma radiation is high-energy form of ionizing radiation, with very short wavelength. Energies range from a few keV to several MeV.

Gamma Scattering In civil engineering, this term is related to the Compton scattering process where Compton-scattered gamma radiation is used to reveal internal structures of the irradiated matter. The intensity of the scattered radiation tells about electron density and the energy is related to spatial position.

Gamma Tomography Based on gamma transmission, this refers to the cross-sectional imaging of an object by the use of a gamma ray source on one side and a set of detectors on the opposite side of the object. Different arrangements are possible: The source-detector arrangement may be rotated around the object or everything may be stationary but based on the simultaneous use of several gamma sources. Internal structures of the object are revealed because different materials exhibit different gamma ray transmittance (see definitions of linear and mass absorption coefficients).

Gamma Transmission In principle, gamma radiation cannot be stopped, only attenuated. The degree of attenuation depends both on atomic number Z and on material density (g/cm^3). Gamma transmission then reflects the degree to which the intensity of an original gamma ray energy has survived the transmission through an object and may, therefore, be regarded as the inverse of the gamma attenuation.

keV Kilo electronvolts $\equiv 1000 \text{ eV}$.

Injection Well This is a well in an oil or geothermal reservoir constructed to inject various liquids, liquid mixtures or gases in order to enhance the production performance of the reservoirs and/or to remove and sequester toxic or otherwise potentially polluting compounds from the industrial processes.

Isotope This term denotes nuclides of the same chemical element (same atomic number Z) but with different number of neutrons N in the nucleus. Thus, isotopes of one defined chemical element have different mass numbers A since $A = Z + N$. Some isotopes of such a defined chemical element may be stable and some radioactive (most often artificially produced). Example of isotopes of cobalt ${}^A_Z\text{Co}_N$: ${}^{59}_{27}\text{Co}_{32}$ or only ${}^{59}\text{Co}$ (stable) and ${}^{60}_{27}\text{Co}_{33}$ or only ${}^{60}\text{Co}$ (radioactive).

Linear Absorption Coefficient The fraction of a beam of a defined gamma radiation energy absorbed per unit thickness (cm) of an absorber, dimension cm^{-1} .

Mass Absorption Coefficients The linear absorption coefficient divided by the density (ρ in g/cm^3) of the absorber, dimension cm^2/g .

MeV Million electronvolts $\equiv 1000.000 \text{ eV}$.

Mineral Scaling Precipitation of a mineral salt in a liquid due to oversaturation of one or more of the ionic components involved, – see definitions of solubility product and saturation ratio. Examples of mineral scaling experienced in the oil, geothermal and water production are: CaCO_3 , BaSO_4 , FeS and more.

Multiphase Flow Metering Monitoring of individual components of aqueous, oleic and gas fluids and solids in a continuous flow.

Nuclide An atomic nucleus is defined by its atomic number Z (number of protons, which defines the chemical element) and its neutron number N . It is illustrated by the symbol ${}^A_Z\text{X}_N$ where $A = Z + N =$ the mass number. For a known chemical element, say cobalt, we know that $Z = 27$. For a unique definition of the nuclide, notation of the mass number and the chemical symbol are sufficient, i.e. ${}^{60}\text{Co}$ is a uniquely defined nuclide.

PET Positron Emission Tomography is an imaging procedure that uses radionuclides emitting positrons to create 3-D pictures of the object in which it has been introduced. It is based on annihilation of the positron whereby two photon quanta, each of an energy of 511 keV, is generated and emitted with an angle of 180 degrees.

Porous Media Generally speaking, these are solid materials containing pores (voids). The skeletal portion of the material is often called the “matrix.” In the present context, the solid materials are minerals or mineral compositions of different kind, and the pores are typically filled with a fluid (liquid or gas).

Positron Positively charged electron emitted from a radioactive nucleus.

Production Well Well drilled into a reservoir for production of reservoir fluids (oil, gas, water, steam etc.).

Radionuclide A radioactive nuclide.

Reaction Kinetics In this context the term is concerned with understanding the rates of chemical reactions. Chemical reaction kinetics includes investigations of how various experimental conditions (temperature, pressure pH, chemical component concentrations etc.) influence the speed of a chemical reaction.

Reservoir Rock Porous rocks that have the ability to store fluids inside their pores. Accordingly, fluids like water, oil, and gas can be accumulated and transported. In petroleum geology, such rocks are mainly based on sandstones or carbonates. In the geothermal fields, fractionated granite rocks also belong to this category.

Saturation Ratio (SR) For a specific uncharged inorganic compound C_xA_y ($\text{C} =$ cation, $\text{A} =$ anion) saturation ratio is defined as $\text{SR} = [\text{C}]^x \cdot [\text{A}]^y / K_{\text{sp}}(\text{C}_x\text{A}_y)$; see definition of solubility product, K_{sp} , below. For $\text{SR} < 1$, no precipitation can happen. For $\text{SR} = 1$, onset of precipitation is thermodynamically allowed, for $\text{SR} > 1$, precipitation will take place.

Scaling Inhibitor A class of specialty chemicals that are used to slow down (kinetic inhibitor) or prevent (thermodynamic inhibitor) mineral scaling in water systems.

Solubility Product Cations and anions may combine to non-charged compounds which, in solution, may precipitate as a solid phase in equilibrium with dissolved ions. The concentration of cations and anions in this equilibrium defines a physical parameter known as the solubility product, K_{sp} . For the reaction $C_xA_y \rightleftharpoons xC + yA$ the $K_{sp} = [C]^x \cdot [A]^y$, where the brackets denote the concentrations. Each noncharged (salt) compound has a specific K_{sp} -value for defined experimental conditions. Examples for aqueous solutions at 25 °C: $K_{sp}(\text{CaCO}_3) = 3.3 \times 10^{-9} \text{ mol}^2$, $K_{sp}(\text{FeS}) = 4.9 \times 10^{-18} \text{ mol}^2$.

SPECT Single Photon Emission Computed Tomography is based on the use of radionuclides that emit single photons (as opposed to PET radionuclides) to picture activity distribution in an extended object. In its simplest version, a projection only is recorded. By recording projections at different angles, a 3D tomographic picture may be generated.

Three-Phase Flow Separator In this context, a processing device that enables separation of the single components water, oil and gas from commingled production in production wells.

TOF Time-of flight is the measurement of the time taken by an object, particle (for instance a neutron) or a wave (for instance electromagnetic) to travel a distance through vacuum or a medium.

X-ray Electromagnetic radiation where one may differentiate between two forms based on their different origin: 1. The elementary X-rays that originate from the electron shell structure in an atom with a definite narrow energy bands (specific for each chemical element) and with energies in the eV and keV range, and 2. the so-called bremsstrahlung, which is generated by charged particles, in practice electrons, accelerated (or bent) in an electromagnetic field. This latter "type" has a continuous energy distribution without definite structure with energies from eV to several tens of MeV, depending on the kinetic energy of the accelerated particle.

Introduction

This chapter describes the principles and practical uses of gamma radiation, i.e. gamma transmission, gamma scattering, gamma tomography and x-ray tomography methods in materials testing and inspection.

Gamma radiation originates from the atomic nucleus. When a nucleus at high excitation de-excites to lower energy states, gamma radiation is emitted. Such highly excited states may be populated in the decay process of radioactive nuclei and by induced nuclear reactions of different kinds. Gamma energy spectra consist of discrete peaks (photo peaks) on a ridge of scattered (Compton) radiation. X-rays may be regarded as a collective term composed of *elementary x-rays* originating from deexcitations in the electron structure around an atomic nucleus with discrete energies, and of *bremsstrahlung*, which occurs by fast acceleration of charged particles. The latter produces a continuous electromagnetic energy spectrum with no discrete structure. The energy distribution depends on the kinetic energy of the charged particles. This happens for instance when an electron passes through the electromagnetic field of an atomic nucleus or when a fast-moving electron beam is bent in an external electromagnetic field like in a ring-shaped accelerator (synchrotron radiation).

Gamma energies and x-ray energies are measured in electron volts, eV.¹ Gamma energies range normally from a few keV (kilo electron volts) to several MeV (million electron volts). Energies of elementary x-rays range from 54.3 eV for lithium ($K_{\alpha 1}$ -line) to 114.5 keV for uranium ($K_{\beta 2}$ -line) while bremsstrahlung energies may have maximum energies of several tens of MeV.

Gamma transmission and scattering

The gamma transmission method

This is a non-destructive and non-intrusive method to measure mass density or mass thickness (g/cm^2) of an object. Gamma radiation from an external gamma source penetrates the object, and the decrease in intensity, or the gamma attenuation over the object, is measured. We may, somewhat imprecisely, call it the principle for "the nuclear scale or the nuclear balance". The principle of this method is illustrated in Fig. 1A.

Transmission of a mono-energetic beam of collimated photons through a single and uniform absorption sample can be described by Lambert-Beer's equation

$$I_{xA} = I_0 \cdot e^{-\mu_A x_A} \quad (1)$$

where I_0 is the beam intensity before transmission, I_{xA} the beam intensity after transmission, μ_A the linear absorption coefficient of the absorbing material A with dimension cm^{-1} and x_A the material thickness (cm).

When the sample is composed of several materials, say A, B and C, each with their own linear absorption coefficients μ_A , μ_B and μ_C and thicknesses x_A , x_B and x_C , exponents are additive, and the formula becomes (Fig. 1B)

$$I_{x_A x_B x_C} = I_0 \cdot e^{-(\mu_A x_A + \mu_B x_B + \mu_C x_C)} \quad (2)$$

¹1 eV = the kinetic energy attained by an electron accelerated in a potential of 1 V.

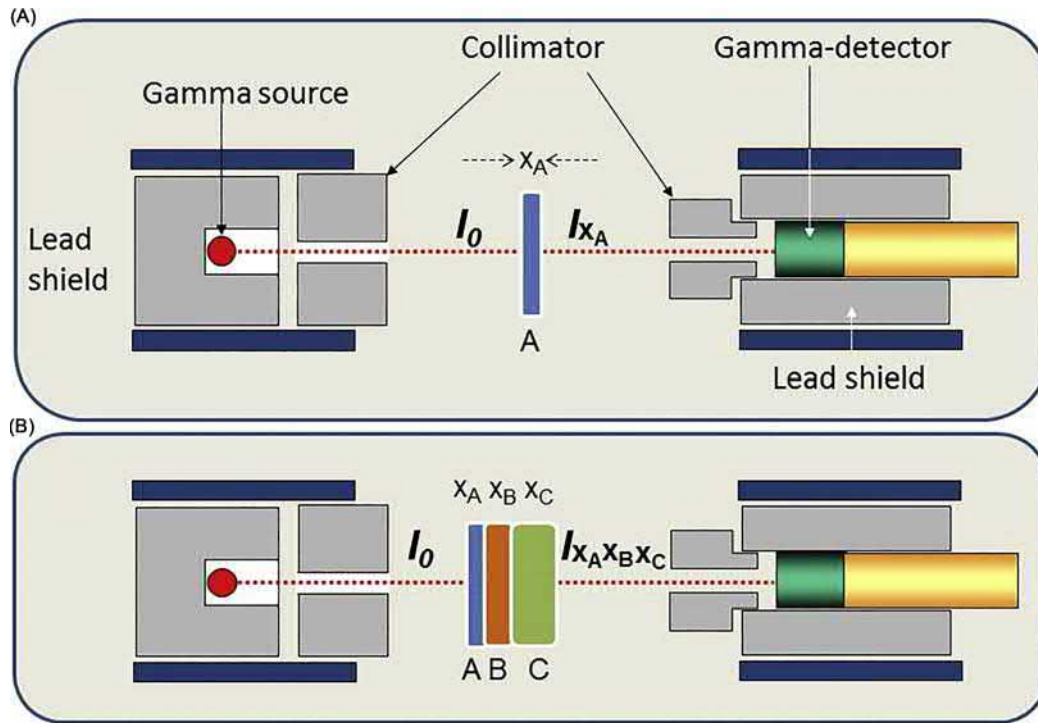


Fig. 1 Gamma transmission measurement principle illustrated on a single sample in (A), and on a composite sample in (B).

More frequently used than the linear absorption coefficients are perhaps the *mass absorption coefficients* defined as μ/ρ with dimension cm^2/g . Here, ρ is the material density (g/cm^3). Let us call this parameter μ_m . The corresponding x -parameter is then the *mass thickness* denoted with x_m , and with dimension g/cm^2 . Introducing this parameter into Eq. (2) gives:

$$I_{x_A x_B x_C} = I_0 \cdot e^{-(\mu_{mA} x_{mA} + \mu_{mB} x_{mB} + \mu_{mC} x_{mC})} \quad (3)$$

The relation between the linear thickness and the mass thickness is simply $x = x_m/\rho$.

Mass absorption coefficients for various chemical elements (and for many common compounds) are tabulated in (Hubbell and Seltzer, 2004). Values vary with gamma energies as illustrated in Fig. 2 below for aluminum, copper and lead.

Often, it is illustrative and informative to derive the thickness needed of a selected material to reduce the transmitted intensity to half the value of the initial intensity, – the so-called half-value thickness (HVT) or $x_{1/2}$. The procedure is as follows:

Eq. (1) may be rewritten as

$$\mu_m = \frac{\mu}{\rho} = \frac{\ln(I_0/I_x)}{x_m} \quad (4)$$

Solving for x_m and setting $I_x = I_0/2$ gives:

$$x_m = \frac{\ln(2)}{\mu/\rho} \quad (5)$$

Dividing this expression with the material specific density ρ gives the half-value thickness $x_{1/2}$.

Examples of such calculations for the three chemical absorber elements Al, Cu and Pb, and for gamma energies from the three radionuclides ^{241}Am , ^{192}Ir and ^{137}Cs are given in Table 1.

The gamma scattering method

This method relies on the Compton effect, where a gamma ray transfers parts of its energy to an atomic electron in the gamma ray path and is scattered away in an angle different from the incoming direction and with a lower energy $E'_\gamma = E_\gamma - E_e - E_{eb}$. Here, E_γ is the incoming gamma energy, E_e the energy transferred to the dispatched electron and E_{eb} the binding energy of the electron in the atom. This method is further explained in the applications section below.

These technologies are useful and utilized both in hand-held, transportable and stationary (permanently mounted) instruments with a wide range of applications. The radioactive sources used in this kind of equipment are medium and low activity sealed sources in the range of 10–100 MBq. Typical radionuclides are ^{241}Am (60 keV), ^{137}Cs (661 keV), and ^{60}Co (1174 and 1332 keV),

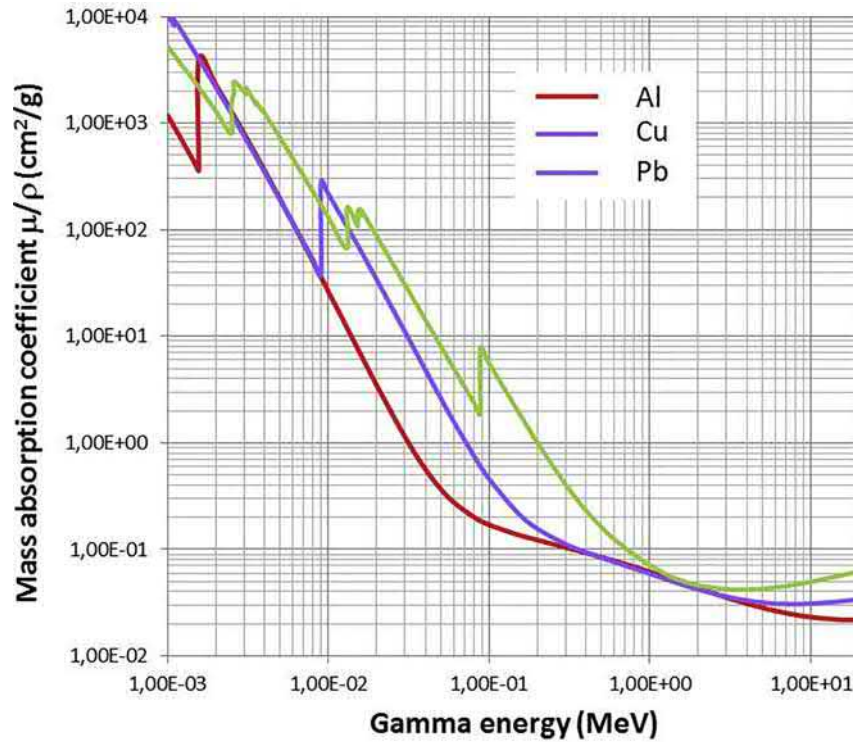


Fig. 2 Mass attenuation coefficients as a function of gamma energies for the three metals aluminum (Al), copper (Cu) and lead (Pb). The curves are derived on the basis of tabulated values from (Hubbell and Seltzer, 2004).

Table 1 Calculated values of mass thickness and half-value thickness of dominant gamma radiation from three radionuclides often used in practical gamma transmission measurements for the three metals Al, Cu and Pb.

Radionuclides		²⁴¹ Am (60 keV)				¹⁹² Ir (317 keV) ^a			¹³⁷ Cs (661 keV)		
Element	ρ (g/cm ³)	μ/ρ (cm ² /g)	x _m (g/cm ²)	x _{1/2} (cm)	μ/ρ (cm ² /g)	x _m (g/cm ²)	x _{1/2} (cm)	μ/ρ (cm ² /g)	x _m (g/cm ²)	x _{1/2} (cm)	
Al	2.699	0.278	2.50	0.92	0.100	6.93	2.60	0.075	9.24	3.42	
Cu	8.960	1.59	0.435	0.049	0.100	6.93	0.77	0.073	9.50	1.06	
Pb	11.350	5.02	0.138	0.012	0.350	1.98	0.17	0.090	7.70	0.68	

Data are based on values of mass absorption coefficients (μ/ρ) shown in Fig. 2.

^aEvaluated only for the strongest gamma line of 317 keV (82.9%) and not for the second strongest of 468 keV (47.8%).

depending on the individual application. The purchase, handling and use of such sources must be approved by the appropriate radiation protection authority in accordance with radiation protection regulations, and the practitioners must be licensed.

The practical use of this gamma transmission method, briefly outlined above, is best demonstrated by showing some examples of applications in science, industry and civil engineering.

Various application examples

Applications in civil engineering

Soil compaction and asphalt thickness

Nuclear density gauges are used by many building contractors, engineers and highway departments for compaction control of soil, aggregate, concrete and asphalt. They are used to determine compaction acceptance of earthworks, granular and stabilized pavement materials, concrete and asphalt (Whitepaper, 2020; APNGA Manual, 2009; IAEA, 2005). This means that buildings and roads are only as good as the foundations you build them upon, and this must be carefully controlled during building site and road preparation and construction.

Wet density, also known as bulk density, typically consists of the soils and moisture present in the ground being compacted. Dry density consists of only the soil solids. The most stable foundation consists of a mixture of solids and moisture. The key to

maximum density and stability is the percentage of moisture in a given soil. The moisture acts as a bond or “glue” for the soil particles. If the moisture is too high, the water will displace the denser soil particles. Hence, no bonding will occur, and a maximum compaction of the particles is not possible. One may end up with mud. If the soil is too dry one will lose the stabilizing effect of the moisture. The needed humidity for maximum soil stability varies with the type of soil.

In order to determine the soil density and humidity, various nuclear gauges have been constructed. These are commercially available today in several designs. They are based on the principles of gamma transmission and gamma scattering (Compton scattering) for density measurements and on neutron back-diffusion (or backscattering) for measurements of humidity. The gamma scattering principle is discussed below while the neutron back-diffusion method is treated in (Bjørnstad, 2021).

The gauge principle sketched in Fig. 3 consists of a shielded and sealed gamma-emitting radioactive source, detectors for gamma radiation and electronics to convert count rates to information. These conversion routines are based on a thorough laboratory calibration for various types of soil, humidity and pavement.

The source most often used is ^{137}Cs (661 keV) with a source strength in the region 200–400 MBq. In the simplest versions the detector is a Geiger-Muller tube.

In the direct transmission position the source rod extends through the base of the gauge into a predrilled hole up to 30 cm deep in the material being tested. The gamma rays from the source, penetrate the test material, are attenuated and subsequently counted by the detectors located within the gauge. The average density between the source and detectors is then determined.

When in use, this instrument is placed on top of the road surface at a position which has been carefully prepared. It can be operated in various modes, two of which are the soil mode and the asphalt mode. Operational parameters and calibration factors vary between the modes.

Soil mode is designed for measurements of soils, stone or other materials where both density and moisture content are desired. Direct transmission testing typically offers better precision and control of depth of measurement and is the preferred method.

Asphalt Mode is used to depths greater than 100 mm of asphalt. Typically, the source rod is in the backscatter position, i.e. slightly above the asphalt (see Fig. 4), but direct transmission can also be used if a hole can be drilled in the asphalt. After having carefully prepared the measurement site, the measurement itself takes only a few minutes. The measurement precision is good (<1%) but the accuracy may vary depending on operational performance and generation and application of correct calibration parameters.

Neither the source nor the detector is directionally collimated. This means that scattered radiation (Compton scattering) will be obtained from various positions and depths over a broad energy range. This enhances the sensitivity but diminishes the specificity, i.e. the sensitivity to special positions.

The energy of the scattered radiation depends on the scatter angle θ as defined by Eq. (6).

Here, E_γ is the emitted gamma energy from the source, E'_γ the energy of the scattered gamma quantum, θ the scattering angle (see Fig. 5), m_e the electron rest mass and c the speed of light. The factor $m_e c^2 = 511$ keV.

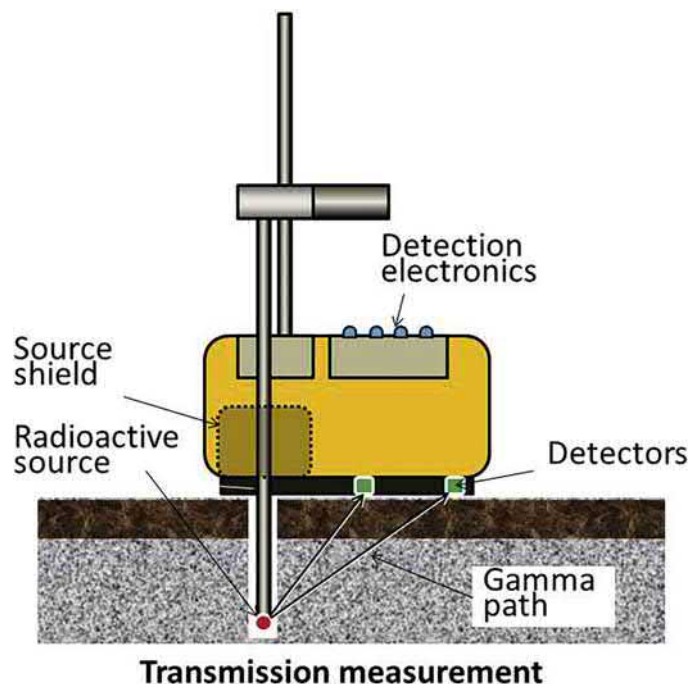


Fig. 3 Portable density gauge based on gamma transmission and scattering, here shown operating in the gamma transmission mode.

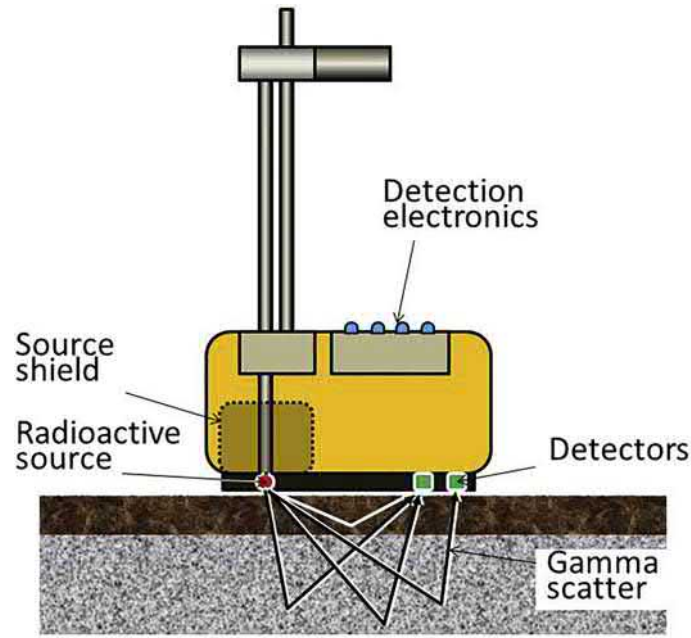


Fig. 4 Portable density gauge based on gamma transmission and scattering, here shown operating in the gamma scattering mode.

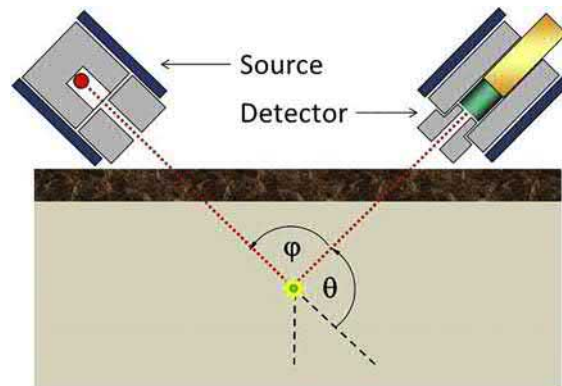


Fig. 5 Collimated source and detector showing the scattering angle θ , and where the angle ϕ can be varied.

$$E'_\gamma = \frac{E_\gamma}{1 + \frac{E_\gamma}{m_e c^2} (1 - \cos\theta)} = \frac{E_\gamma}{1 + \frac{E_\gamma}{511} (1 - \cos\theta)} \quad (6)$$

A more narrowly defined Compton-scattered gamma energy is detected. The energy of the scattered gamma ray can be calculated by Eq. (6). Varying the angle ϕ gives possibility of depth information. All equipment is above ground. By moving this surface equipment laterally in a predefined pattern, volumetric information may be retrieved. This is not readily possible with the equipment sketched in Fig. 4 and not practical with equipment pictured in Fig. 3.

Sediment studies in harbors

The investigation of sediment transport along seashores and in rivers is important for civil engineering and coast-near protection and management (Wentworth, 1922; Maa et al., 2006; Dyer, 1986; Corporate author, 2015). Littoral regions (along the shores) and sea beds are in dynamic change. Coastlines are undergoing periods of erosion with sediment transport, sedimentation and consolidation. The main causes for erosion in beaches include:

- storms leading to ocean waves that attack the consolidated coastline rock and, with time, breaks it down into grains and fines,
- human actions, such as the construction of quays and seawalls,
- jetting and the dredging of river mouths followed by dumping of the dredging deposit.

Each of these human actions disrupts the natural flow of sand. Current development activities and practices are accelerating the beach erosion process. However, there are viable options available to study the development of such disruptions in order to mitigate the damage and to provide for sustainable coastline control.

One useful method is the application of radiotracers for sediment transport studies (IAEA, 2014). Sealed source techniques can also provide information on the density of sediments deposited in a channel of navigation, as well as on the concentration of sediments circulating in suspension.

Sediment studies in seas, rivers and dams commonly deal with sand and silt. There are different sub-classifications of sands according to their sizes. The most widely used classification of sediments and sedimentary particles is the Wentworth's classification (Wentworth, 1922). It comprises four main classes of particles:

1. Gravel: particles greater than 2 mm (mm) in diameter;
2. Sand: particles between 63 μm (μm) and 2 mm in diameter;
3. Silt: particles between 2 μm and 63 μm in diameter; and
4. Clay: particles less than 2 μm in diameter.

Sedimentation and sediment density may be studied in controlled laboratory examination in two ways: (1) Directly on sediment columns extracted from the sea floor by specially designed sampling equipment in order to preserve the natural axial composition and density variation, and (2) on a sediment slurry that is set to sediment in the laboratory in order to measure sedimentation rates. Mud compaction rates, their initial concentration, height of the deposit and the salinity of the medium can be investigated in this way by a nucleonic gauge using gamma transmission. For this purpose, a source of ^{241}Am (3,7 GBq) may be positioned opposite a carefully collimated scintillation detector with the column in the middle. The source-detector pair can be scanned axially over the column to record the changing density with column height. Density precisions in the order of 1% may be obtained in term of concentration of the mixture at one moment and on a given level (Meyer et al., 1983).

However, useful in-situ information can also be obtained by realistic studies directly on the various costal sites of interest by using real-time NDT-methods. Three methods are

1. Gamma ray scattering

Here, a measurement tool consisting of a radioactive source and a gamma ray detector sealed in a water-proof housing is lowered by a wireline from a boat down towards the seabed where the tool sinks some distance into the sediments. The density of the surrounding sediment is measured by the gamma scattering technique as illustrated in Fig. 6. The source is typically ^{137}Cs (661 keV) with a strength of 18.5 MBq. The detector has spectroscopic properties and consists typically of a 50x50 mm NaI(Tl) crystal.

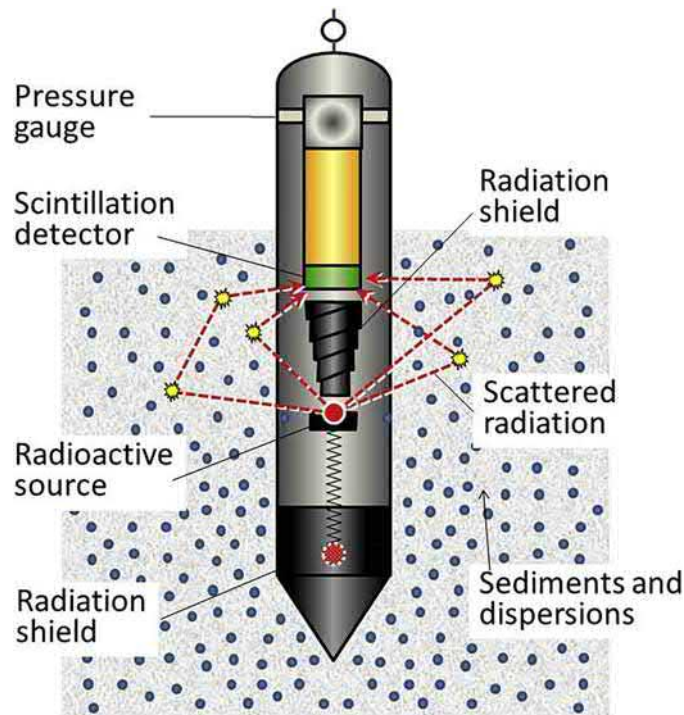


Fig. 6 Principle of a gamma ray scattering gauge designed for in-situ measurement of seabed sediment density and thickness.

2. Gamma ray transmission

A typical example of an instrumentation that can be moved, in principle, continuously over large areas is illustrated in Fig. 7. It consists of a sledge which is towed on the seabed behind a boat on the water surface. The sledge contains a gamma source (source type ^{241}Am or ^{137}Cs , source strength $\approx 220\text{ MBq}$) on one skid beam and a spectroscopic gamma detector (typically a $50\times 50\text{ mm NaI(Tl)}$ crystal) on the other skid beam. The space in-between will automatically be filled with sediments during operation. The measurement method is thus based on the gamma transmission principle. The layout of the sledge can be modified in order to account for various sediment, slurry or dispersion densities which may also need adjustment in the type and activity of the radioactive source.

The boat may traverse the sea in a defined pattern. Thus, extended areas can be monitored with reduced effort.

3. X-ray transmission

This technique is very similar to the gamma transmission technique but characterized by lower energy photons generated by an X-ray tube instead of a radioactive source. Typical photon energies are in the order of 30 keV . The tool itself consists of two parallel vertical legs with a spacing, one containing the X-ray generator and the other the x-ray detector in a very robust design. The tool is lowered stepwise down towards the seabed and finally into the sediments while measurements are recorded continuously. Thus, changing density is measured in a vertical setting. In order to map sediment densities over an area, the tool must be relocated to another position in a preselected grid pattern.

Examples of practical applications of sediment studies with nuclear-based methods are found in refs. (Crickmore et al. (1990) and Bandeira et al. (2014).

Applications in science

The examples are multiple, but we will here limit the description to only one example, – the laboratory-based studies of the basic mechanism of undesired mineral precipitation, or *scaling*, in various process equipment. This is a major concern both in civil engineering (clogging water supply systems) and various industries like the oil production industry (reduces permeability in the mineral rock close to production well and thereby the production rate, clogs production tubes and affects negatively the performance of 3-phase separators) and the geothermal energy production industry (clogging production wells, heat exchangers and valves). The costs for maintenance and replacement of vital equipment parts and components amount to billions of dollars each year. Hence, there is a considerable drive to understand the basic scaling mechanisms in order to come up with effective mitigation measures.

Study of mineral scaling and scaling kinetics

Natural waters contain cations and anions of different kinds. When the concentration of some of these ions in combination surpasses certain limits, precipitation of solid compounds takes place.

The solubility of a certain solid compound is defined by its solubility product K_{sp} . For a compound with the general formula C_xA_y the stoichiometric solubility product is defined as $K_{sp} = [C^{y+}]_{eq}^x \cdot [A^{x-}]_{eq}^y$ where $[C^{y+}]_{eq}$ and $[A^{x-}]_{eq}$ are the concentrations in mol/L of the cation and anion of the compound, respectively, at saturation, i.e. in equilibrium with the solid compound C_xA_y . The solubility product depends on temperature, pressure, ion strength etc. Hence, precipitation requires that the actual

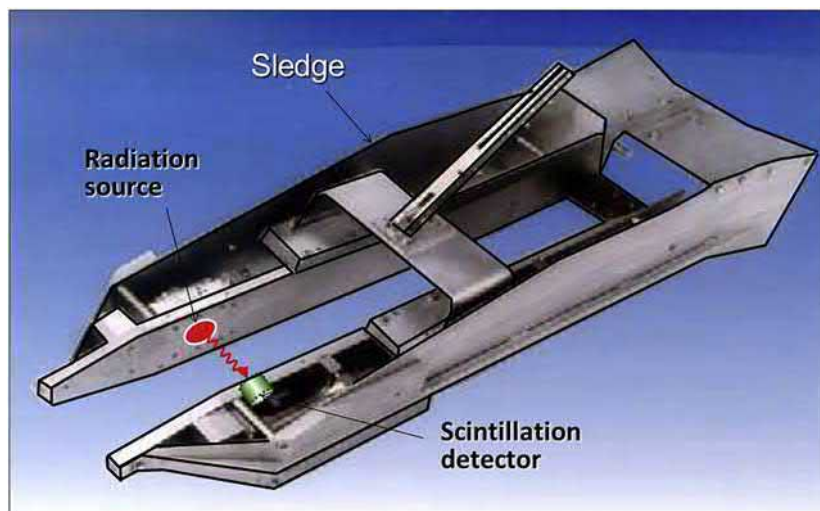


Fig. 7 Sketch of a sledge equipped with a collimated gamma source and a collimated scintillation detector for gamma transmission measurements of the sediments/sludges directly on the seabed as the sledge is towed behind a boat on the surface (illustration modified after ref. IAEA, 2005).

and real ion product in the water present $[C^{y+}]^x \cdot [A^{x-}]^y > K_{sp}$ for the existing conditions. This is expressed as the saturation ratio SR defined in Eq. (7):

$$SR = \frac{[C^{y+}]^x \cdot [A^{x-}]^y}{K_{sp}} > 1 \quad (7)$$

If $SR < 1$ we say that the solution is undersaturated and precipitation cannot take place.

There are several cations which, in combination with certain anions, may result in scaling leading to process equipment component fouling, see Fig. 8. Examples are Ca^{2+} , Sr^{2+} and Ba^{2+} in combination with CO_3^{2-} or SO_4^{2-} and Pb^{2+} , Zn^{2+} and Fe^{2+} in combination with CO_3^{2-} or S^{2-} . The main scaling type in fresh-water supply systems, especially in ground waters, is $CaCO_3$. In the petroleum sector the main scaling components are $CaCO_3$ (high), $SrCO_3$, $BaCO_3$, $FeCO_3$, $BaSO_4$ (high), FeS , ZnS and PbS . In the geothermal sector the main scale components are $CaCO_3$ (high), FeS , ZnS , PbS and, in high-temperature geothermal reservoirs ($T > 300^\circ C$), amorphous silica $[SiO_x]_y$ (high), which originate from dissolution of silicate minerals in the reservoir rock.

In the further text we will concentrate on $CaCO_3$ scaling. Aspects about the precipitation mechanism can be studied in dedicated laboratory experiments by nuclear techniques including gamma transmission.

Waters of different origin contain considerable amounts of calcium and bicarbonate ions. Although the ions are in equilibrium in the reservoirs, during the production and the associated pressure release downstream in the production tubing, bicarbonate decomposes into carbonate and CO_2 . If Ca^{2+} is present in sufficient concentration, $CaCO_3$ will precipitate according to the equation:



The scaling mechanism and rate (kinetics) can be studied partly with a setup as sketched in Fig. 9 (Bjørnstad and Stamatakis, 2006). Here, separate liquid flows of calcium (Ca^{2+}) and bicarbonate (HCO_3^-) representing $SR > 1$ meet in the fluid mixing

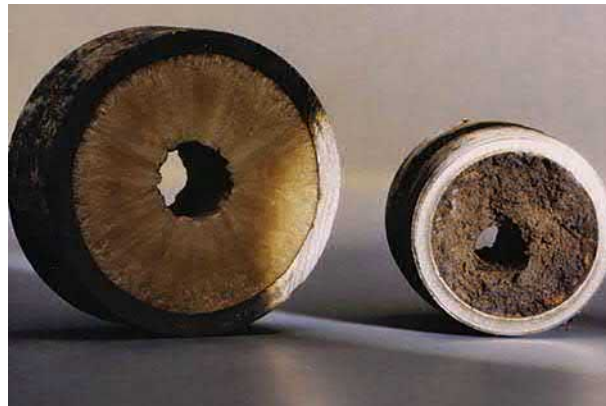


Fig. 8 Water transportation tubings nearly blocked by $CaCO_3$ precipitation.

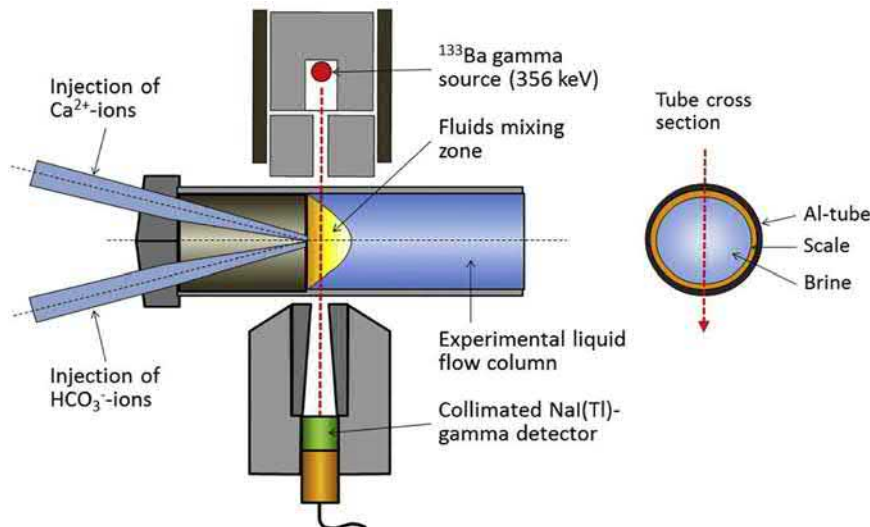


Fig. 9 Sketch of a gamma transmission setup to study mechanism and kinetics for precipitation of calcium carbonate, $CaCO_3$.

zone. Creation of tiny precipitation seeds will start immediately. Most of the seeds will be transported down-stream but some will attach to the flowline wall at the mixing zone. Here, they act as crystallization points for further particle growth, and a solid layer will start forming and continue growing on the inner wall at the mixing zone and further downstream as well. A stationary setup of a radioactive source (^{133}Ba) and a gamma detector monitors the mixing zone and the increasing mass of the formed scale by a decrease in the recorded count rate. The thickness of the scale can be calculated by Eq. (2) above, when we introduced the values of the linear attenuation coefficients μ for aluminum, calcium carbonate scale and brine that have been measured separately. The thickness of the aluminum walls is constant, and the known internal diameter is $x_{\text{scale}} + x_{\text{brine}} = 10 \text{ mm}$. Hence, $x_{\text{brine}} = 1 - x_{\text{scale}}$ and Eq. (2) may be solved with respect to the only unknown parameter x_{scale} . In this way one can study the precipitation reaction kinetics for changes in various parameters like temperature, concentrations of the precipitating components, effect of other ions present in natural waters (brines) and of various production chemicals that are often added to the industrial processes like, for instance, corrosion inhibitors.

But perhaps most important is the effect of adding chemicals to try to prevent the precipitation to take place, so-called scaling inhibitors. That is the main technical aim of such studies. Fig. 10 illustrates the effect of adding one possible scaling inhibitor in different concentrations to one of the liquid flows in a system where $\text{SR} = 20$. Scaling is significantly reduced with increasing inhibitor concentration, and at 100 ppm (ppm) no scaling and scale growth has been detected at the fluids mixing zone. This method can be used to investigate the effect of various scaling inhibitors in order to find the best alternative, which is a function of the lowest possible concentration, lowest environmental impact and lowest price.

Applications in the oil industry

There are multiple applications of the gamma transmission technique in the petroleum production industry (see also Thereska, 2021). Here, we will briefly touch upon four applications: 3-phase fluid separator condition monitoring, 3-phase flow separator separation performance, distillation tower condition monitoring, and multiphase flow metering.

3-Phase fluid separator condition monitoring

In general, oil well fluid flow consists of the three fluids oil, water and gas, but the flow may also contain suspended particles like sand from the reservoir and scaling as described above. The components are separated into individual phases in a 3-phase flow separator. Suspended particles settle at the bottom. The immiscible components (on the molecular level) oil and gas separates normally with water at the bottom and oil on top and is drained from the separator through different channels. Gas exits the separator at the top. Each of the phases continues to further purification steps: Oil to remove traces of emulgated water, water to remove traces of emulgated oil and, in some cases, water-dissolvable organic components originating from the oil, and gas to remove traces of wetness consisting of water and organic gaseous molecules with carbon number higher than about 4 (butane).

With time, performance of the separator may deteriorate; for instance, due to changes in the internal structure because of corrosion and mechanical breakdown or to accumulation of too much solid particles. Using gamma transmission monitoring, problems of this nature may be detected, calling for major maintenance.

Multiphase separators are mechanical structures that come in different configurations and sizes. A simple outline of a horizontal multiphase separator is sketched in Fig. 11. On offshore oil production rigs, these structures are large, i.e. more than 10 m long and

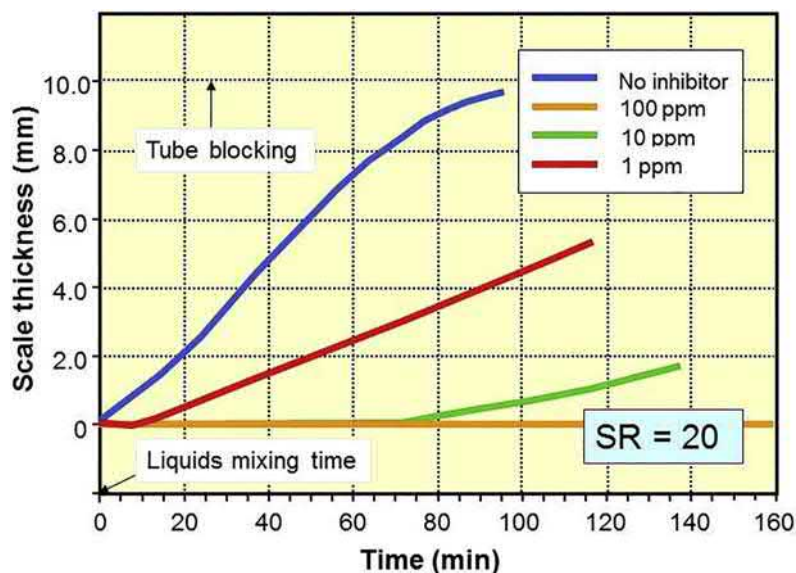


Fig. 10 Decrease in growth of scaling thickness with increasing concentration of an added scaling inhibitor.

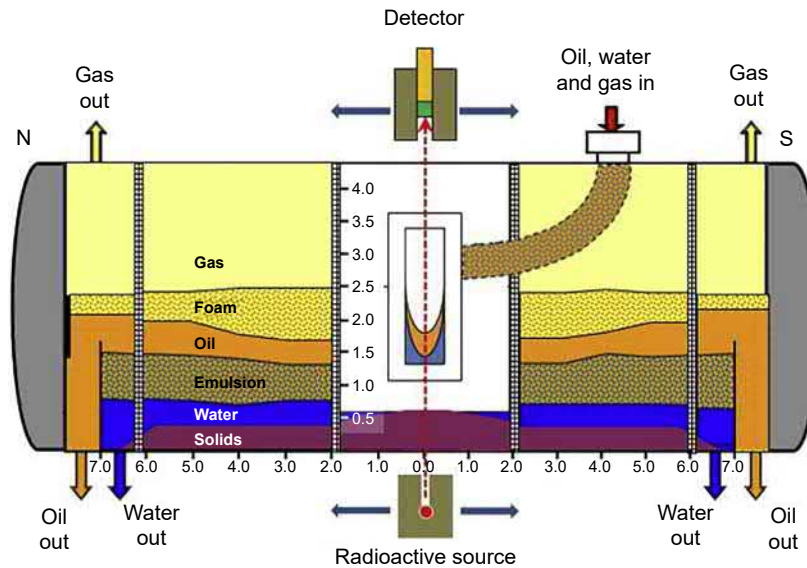


Fig. 11 Sketch of a multiphase separator for oil, water and gas and the use of a scanning setup of a gamma transmission monitoring equipment to detect operational anomalies.



Fig. 12 Off-shore field installation of a gamma source below a 3-phase separator for gamma transmission monitoring experiments. Mind the sizable dimensions.

3–4 m in diameter. A radioactive source below the separator tank (Fig. 12) and a corresponding gamma detector above the tank are scanned along the tank axis direction to record densities and detect any possible change in the internal metallic structures. Typical sources are ^{137}Cs (661 keV) and ^{60}Co (1174 and 1332 keV) with activities of a fraction of GBq up to some tens of a GBq.

3-Phase flow separator separation performance

Performance of a separator depends not only on the separator construction itself but also on molecular components and their concentrations in the incoming fluid mixture. Throughout the lifetime of an oil reservoir (20–40 years, sometimes longer) the natural compositions of oil, water and gas change. In addition, as reservoirs age, improved oil recovery methods are sometimes applied to keep up the production volume as long as possible. This implies that so-called production chemicals are injected into reservoirs and wells in relatively large amounts. Such chemicals comprise corrosion and scaling inhibitors and acids for wells, and polymers, surfactants and alkaline solutions for reservoirs. Some fraction of these chemicals enters the separator as well and may contribute to perturbation of the normal operational regime. Therefore, it is important to follow tightly the separator performance by frequently monitoring the level of the various phases in the separator, which, in addition to those already mentioned, contains an oil/water emulsion phase and an oil-gas foam phase. Equipment with different designs have been constructed based either on gamma transmission or neutron back-diffusion (backscattering) technology. The latter is treated further elsewhere on neutron methods (Bjørnstad, 2021).

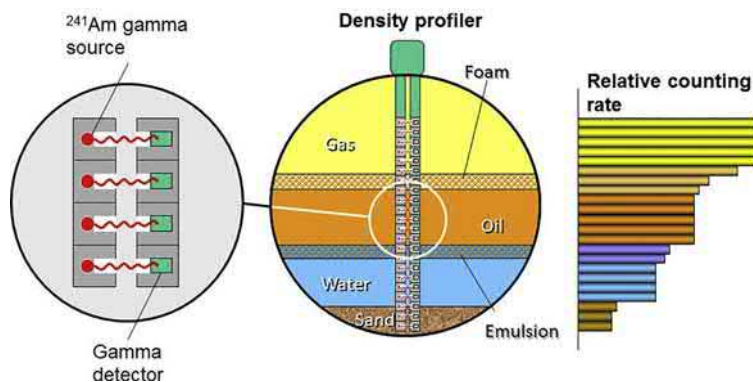


Fig. 13 Principle sketch of Tracerco's "density profiler" mounted in a horizontal 3-phase separator measuring the density profile (height) of the various phases inside the separator during operation.



Fig. 14 Illustration from an oil refinery showing distillation columns of various sizes.

Fig. 13 shows a principle sketch of an equipment developed by Tracerco² (Charlton, 2006). It consists of two parallel and vertically mounted titanium pipes with a spacing of a few cm. One pipe contains a set of ^{241}Am -sources in tungsten shielding facing a set of opposite detectors (Geiger-Muller (GM) tubes) in the other pipe. The number of source-detector pairs depends on the desired vertical measurement resolution and the diameter of the separator. The tubes may contain up to a maximum of 125 units. The monitoring unit is inserted into the separator from the top so that the pipes stick into the fluids all the way down to the bottom. The expanded view to the left in **Fig. 13** shows how the collimated radiation from the source penetrates the medium between the pipes and is detected by the collimated GM-tubes. The count rate is recorded for each source-detector pair whose individual characteristics are calibrated. It depends on the density of the traversed medium. Low count-rate means high density and higher count rate lower density. The composite results are shown as the relative counting rates in the panel to the right in **Fig. 13**. The instrument is continuously operating, but in practice the panel showing the relative counting rate is updated in intervals (counting times) at frequencies chosen by the operator from seconds and upwards. Such a surveillance gives the operator insight in when to modify the various running conditions like the flow rates and chemistry in order to rectify any malfunctions.

Distillation tower condition monitoring

Oil from the 3-phase separators described above is piped downstream for further processing. This processing is of various kinds: distillation to separate components, cracking to chop up long-chain compounds into smaller, chemical conversion to produce new compounds, etc. The present section will be limited to monitoring of distillation columns.

In an oil refinery, several types of columns are in operation. Two main types are trayed and packed columns. The sizes of the mechanical structures vary, from ~ 10 m up to >100 m in height and with diameters from ~ 2 to >10 m (see **Fig. 14**). In

²www.tracerco.com.

a distillation process, the molecular components are separated based on the *relative volatility* of the components. The relative volatility of component *a* relative to component *b* is defined as

$$\alpha_{ab} = \frac{x_{va}/x_{la}}{x_{vb}/x_{lb}} \quad (8)$$

where x_{va} is the mole fraction of component *a* in the vapor phase, x_{la} is the mole fraction of component *a* in the liquid phase x_{vb} and x_{lb} the corresponding mole fractions for component *b*.

The atmospheric distillation columns belong to the biggest units. They can process up to $> 30,000 \text{ m}^3$ of oil per day. The process pressure is slightly above atmospheric pressure.

A distillation column like the one sketched above is, of course, not able to resolve single components into individual fractions. Rather, boiling point fractions along the height of the column are collected with the least volatile in the bottom and the most volatile (like methane = C1, ethane = C2 etc.) at the top.

This is illustrated in Fig. 15. These fractions include methane, ethane, propane and butane mixtures, naphtha's/gasoline fractions, kerosene/aviation turbine fuel, light oil like diesel, heavy oil like lubrication oil and fuel oil for ships and so-called reduced crude oil (the least volatile fraction), which are further processed in vacuum distillation columns.

In a trayed column during normal operation, each tray should sustain an adequate liquid loading. If the operation shows that results deviate from the design output, the reason is often that the liquid loading in individual trays or in tray sections are too large (flooding) or too low, which again may be due to mechanical changes in the internal structure of the column. The incoming crude oil contains acidic components like CO_2 and H_2S , although to varying degrees. Over time these may attack internal steel structures, especially at high temperatures, and trays may be damaged.

Gamma transmission scanning is a useful method to identify reasons for such deteriorated separation efficiencies. These scans are carried out during full tower operation and enables detection of the problem in "true time" while it is being experienced. Flooding, entrainment, foaming or weeping (liquids drop downwards from tray to tray) may be detected together with any major

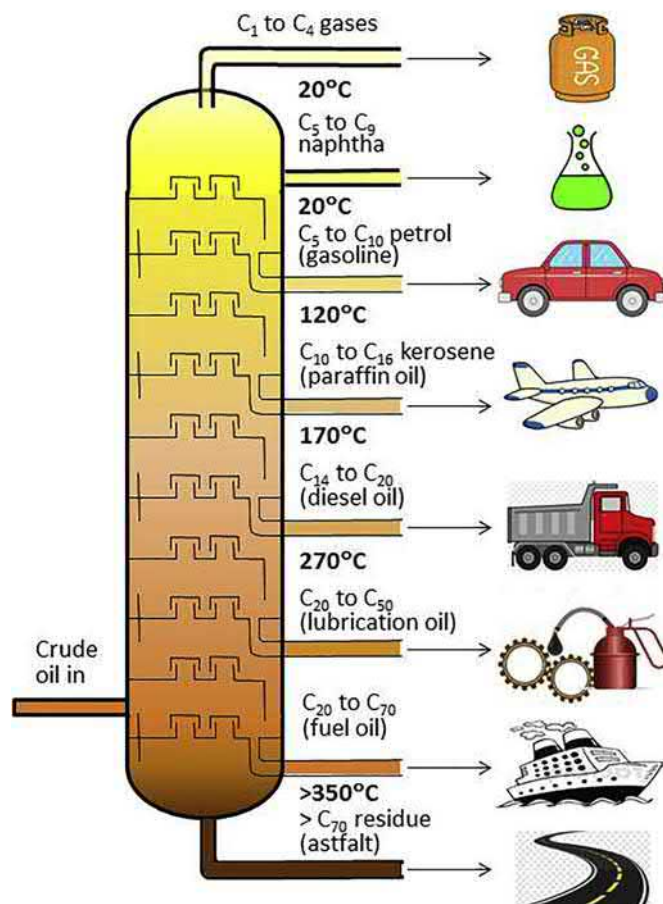


Fig. 15 Illustration of an atmospheric pressure distillation column where the preheated crude oil from the 3-phase separator enters near the bottom. The molecular components are separated according to their volatility which is related to the number of carbon atoms (C₁, C₂, C₃ etc.) in the molecules.

mechanical structure defects (Hills et al., 2002; Abdullah, 2005; Laraki et al., 2008; Walinjkar and Singh, 2011; Elsharkawy and Hammad, 2018).

A gamma scan is performed by positioning a gamma source placed in a 360° panoramic lead collimator (normally ^{137}Cs or ^{60}Co , source strength in the range 30–100 MBq) on one side of the column and a directional lead-collimated detector (normally NaI(Tl) scintillator) on the opposite side (see Fig. 16). The source and detector are aligned at the top of the column and synchronously lowered down the outside of the column in steps where counting rate recordings are taken for every step. The orientation is normally chosen to scan the column diameter or through another selected active area. The gamma attenuation is strictly dependent on the density of the material between the source and detector. The major factor affecting the amount of radiation passing through the column apart from the mechanical structure is the presence of liquid on the trays and the absence of liquid in the vapor spaces.

Repeating scans under different operating conditions such as temperature, pressure, feed flow rate and reflux ratios can reveal additional information on the degree of entrainment (carry-over of liquid), tray and “jet” flooding and foaming or “weeping.” However, often only one scan is performed to detect a malfunction.

In principle, two preparative scans should be performed on all columns: One before liquids is introduced to obtain a “background” density chart of the mechanical structure, and one after the column has come into normal operation right after commissioning of the column. These scans will then represent the normal operating conditions and will later ease the interpretation of a scan taken during some form of malfunction.

For packed columns, four scans are often performed (see Fig. 17) to detect malfunctions like packing material collapse or other types of heterogeneous redistribution, which affects the column performance.

Multiphase flow metering

Multiphase systems are fundamental to the operation of many industrial and environmental processes. They are to be found across a very wide industrial spectrum, embracing various types and sizes of production unit ranging from oil production facilities at one extreme, with throughputs of tens of thousands of tons per day, to pharmaceutical plants, with throughputs of the order of a few tons per day, at the other.

Generally speaking, it is large-scale and continuously operating units that are the main beneficiaries from radionuclide applications. Because of the sheer size of the processes, the economic benefits arising from on-line troubleshooting and basic process optimization procedures, such as de-bottlenecking are often very large, and this has acted as an incentive for large-scale industries to utilize the technology extensively. For this reason, these industries have long been the main target area for the exploitation of various radionuclide techniques of which gamma transmission monitoring is one.

Here, we will exemplify the usefulness of this technique by limiting the text to applications in the oil and gas sector (production, treatment and pipeline transport phases). However, the technique is also useful in mining and mineral processing, for petrochemical production, production of fertilizers, in power generation (hydropower, coal grinding, nuclear power, geothermal power), in the general chemical industry, cement industry, waste water treatment and food processing to mention some of the main areas.

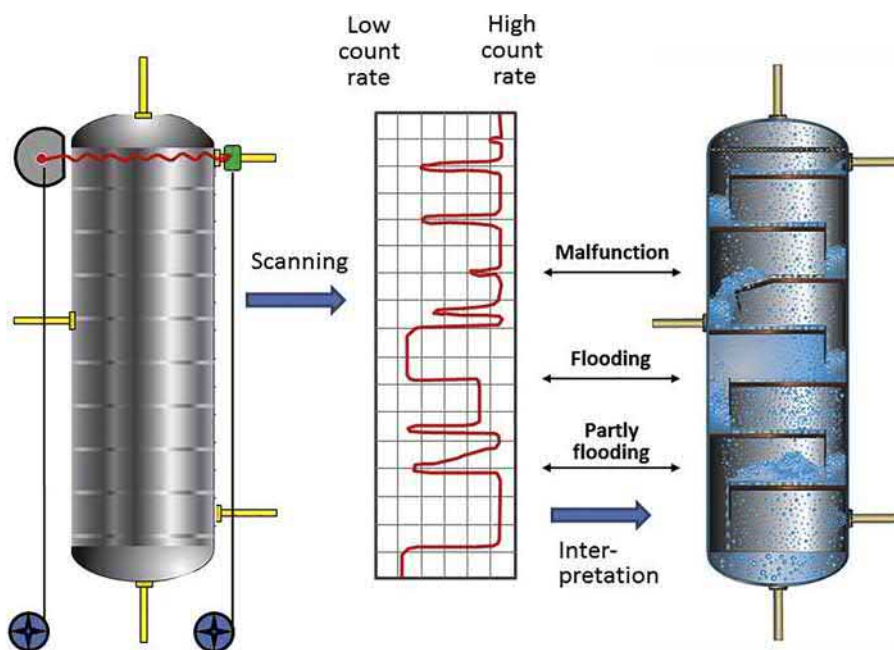


Fig. 16 Illustration of a distillation column gamma scan to the left in the panel, the results in form of a density chart in the middle and the associated interpretation to the right in the panel.

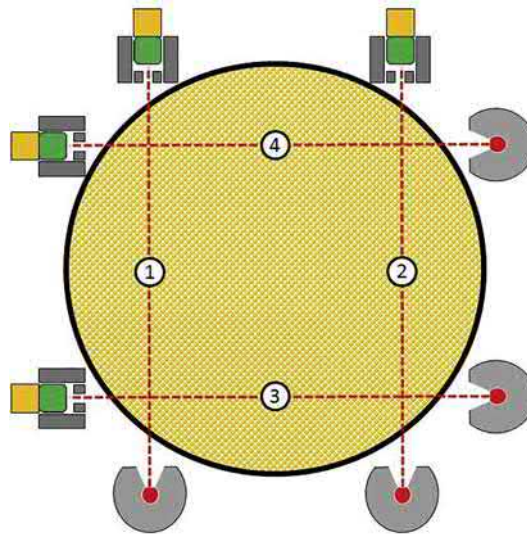


Fig. 17 Typical scanning sectors for packed distillation columns – top view.

In this text, multiphase flow is defined as simultaneous movement of two or more of the following main constituents or phases in a confined space:

- Water (fresh or saline, continuous phase)
- Oil (or other organic water insoluble liquid components, continuous phase)
- Gas and gas mixtures
- Solid particles (sand, inorganic mineral scales or precipitations consisting of inorganic cations and organic anions. These particles may exhibit water-wetting or mixed wetting properties)
- Emulsions (oil in water or water in oil). These may be characterized as micro emulsions, which may be stable over extended time (i.e. coalescence rate is low) or emulsions which are generated and broken down continuously (i.e. coalescence rate is high).
- Asphaltenes and resins, which exhibit bipolarity but are mainly oil wetting.
- Gas hydrates, which are compounds of water and natural gas arranged in a more or less solid rigorous lattice.

In the petroleum industry, however, multiphase flow is often defined as the commingled flow of oil, water and gas only.

The need for multiphase flow measurement in the oil and gas industry has been evident for many years. Several such meters have been designed and developed since the early 1980s by research organizations, meter manufacturers, services supply industry, oil and gas production companies and others. They are designed to apply for large spans in the relative mass flow of the various phases. Flow structure varies with change in the superficial phase velocities³ of the individual components. This is illustrated for two-phase (water-gas) horizontal and vertical flow in Fig. 18A and B.

A simple clamp-on gamma transmission system for multiphase flow measurements is illustrated in Fig. 19. A collimated gamma ray beam (narrow beam) traverses the transportation tube with the internal flow and is detected with a gamma detector with spectroscopic properties, – typically a NaI(Tl) scintillation detector. It is also possible to use a wide-angle beam (broad beam) with a correspondingly collimated detector for a larger cross section view. Gamma attenuation takes place according to Eq. (2), but here slightly reformulated for two-phase flow, for instance oil and water:

$$I_{ow} = I_0 \cdot e^{-(\alpha_o \mu_o d + \alpha_w \mu_w d)} \quad (9)$$

Here, I_0 is the count rate recorded with empty tube, I_{ow} the count rate recorded with commingled oil and water flow, α_o and α_w the fractions of oil and water, respectively, μ_o and μ_w the effective linear attenuation coefficients of oil and water for the applied gamma energy, respectively and d the internal diameter of the tube. In addition, we have that:

$$\alpha_o + \alpha_w = 1 \quad (10)$$

By combining Eqs. (9), (10), the unknown fractions of oil and water, also called the hold-up fractions, α_o and α_w can be calculated.

Hence, for two-phase flow a source with only one gamma energy is sufficient, but the energy must be sensitive to phase ratios over a wide range. For instance, if gas is one of the fractions and may be the dominating one, a low-energy gamma like from ^{241}Am (60 keV) can be the best choice. With increasing mass thickness (g/cm^2), which is a function of both phase fractions and diameter

³Superficial phase velocity = the flow velocity in m/s of one phase of a multiphase flow, assuming that the phase occupies the whole conduit by itself. It may also be defined by the relationship (Phase volume flow rate)/(Pipe cross-section).

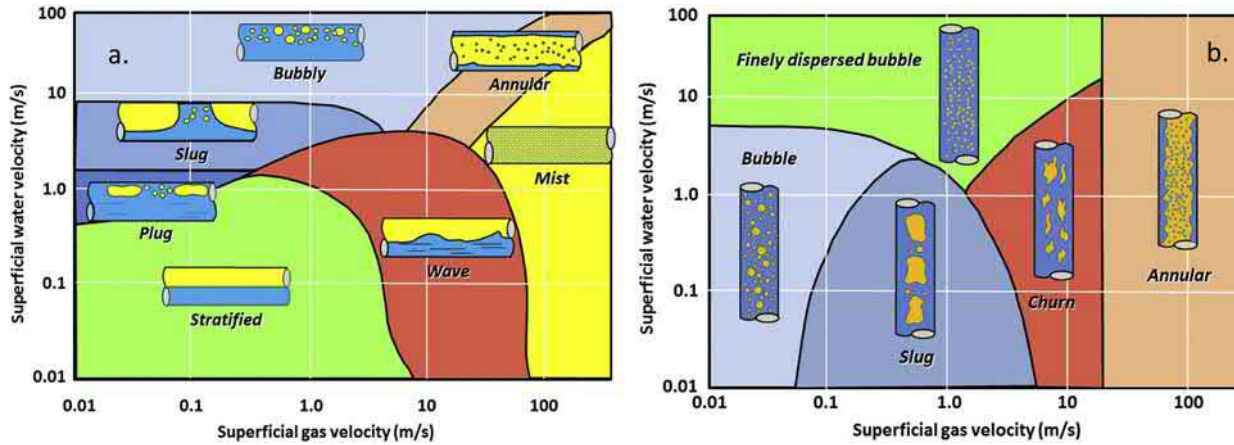


Fig. 18 Illustration of various flow structures in two-phase flow that dominate in certain areas of the map as a function of the superficial velocities of gas and water phases for a. horizontal and b. vertical flow. The figures are redrawn based on information in ref. [Corneliussen et al. \(2005\)](#).

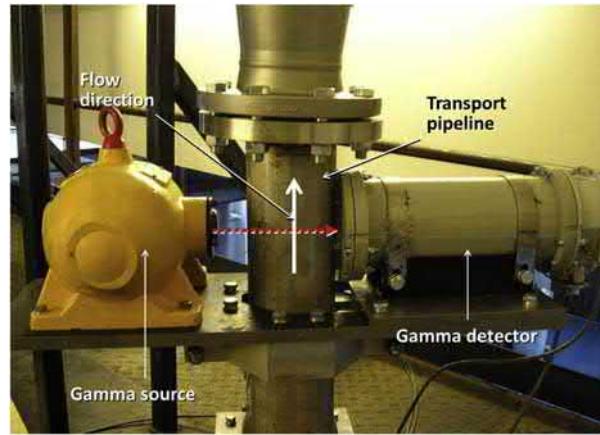


Fig. 19 Clamp-on source-detector system for measurement of individual phase fractions in a vertical multiphase transport pipeline.

on the transportation tube, a higher energy gamma ray may be required. Either ^{133}Ba (main energy at 356 keV) or even ^{137}Cs (661 keV) may be a better choice.

For three-phase flow, say oil, water and gas, at least two different gamma energies are required ([Scheers and Slijkerman, 1996](#); [Khayat and Afarideh, 2019](#)). The gamma energies must be chosen so that their attenuation coefficients are sufficiently different. Again, ^{133}Ba is a viable choice because, in addition to its main energy at 365 keV, it has strong emissions at 80 keV and 303 keV as well. If a higher gamma energy is required, a mixed source of ^{133}Ba and ^{137}Cs is possible.

The formalism for three phases (o , w and g) and two gamma energies (1 and 2) is as follows:

$$I_{ow1} = I_0 \cdot e^{-(\alpha_o \mu_{o1} d + \alpha_w \mu_{w1} d + \alpha_g \mu_{g1} d)} \quad (11)$$

and

$$I_{ow2} = I_0 \cdot e^{-(\alpha_o \mu_{o2} d + \alpha_w \mu_{w2} d + \alpha_g \mu_{g2} d)} \quad (12)$$

I_{ow1} and I_{ow2} are measured for the two gamma energies and the values of all the effective linear attenuation coefficients and the diameter d of the tube are known (measured separately). But this time we have three unknowns and need three equations for solving α_o , α_w and α_g . The third equation is:

$$\alpha_o + \alpha_w + \alpha_g = 1 \quad (13)$$

By combining Eqs. (11)–(13), the individual hold-up fractions may be calculated.

If a 4th component, like for instance sand, is part of the mixture, 3 gamma energies are required and so on. It is very important to check regularly the values of the linear absorption coefficients that may change somewhat during operation due to change in oil composition or water salinity.

Gamma transmission and emission tomography

Gamma transmission tomography

Tomographic measurements of extended objects can be done with different methods (Wang, 2015), but here we concentrate on gamma transmission tomography. Gamma transmission tomography is a refinement of the simple fixed-geometry clamp-on single-beam gamma transmission setup. The same basic gamma attenuation principle applies here. Tomographic measurements allow not only hold-up fractions to be measured but also their spatial distributions over the cross section viewed by the source-detector arrangement (Johansen and Jackson, 2004; IAEA, 2008; Kim et al., 2011).

In tomographic imaging, multiple measurements are carried out at different directions through the object being investigated as illustrated in Fig. 20. The tomographic representation is generated by an image reconstruction technique where a 2D distribution of parameters, in this case mainly densities, is produced. Several sequential 2D images may also be stacked together to form a 3D image. Cross sections of spatial distribution of masses/phases under transport averaged over some preselected counting time are recorded. The time resolution is the counting time needed to achieve sufficient counting statistics in the measurements.

This technology can also be used to map the internal configuration of static objects that do not change constitution or composition within the time resolution window. Examples may be severe corrosion attack on pipeline internal walls or the precipitation of mineral scales or other deposits in the pipeline under inspection. Such information is used in the design and optimization of industrial processes and process equipment, and to improve accuracy of multiphase system measurements in general.

The technique is also useful in quite different areas such as revealing the hidden internal structure in cultural heritage objects or, for instance, detecting basal stem rot in living oil palms (Abdullah et al., 2013).

Tomographic equipment based on gamma transmission has undergone considerable development over time from the relatively simple 1st generation with movable source-detector setups with "pencil beam" scanning to the static version where there are no movable parts except from the tomography object itself. With reference to Fig. 20, a short description of each method is offered below.

1st generation: This is the simplest setup based on parallel scanning in which a source, emitting a single pencil beam of radiation, and a detector are coupled together so that the detector is always facing the source as illustrated in Fig. 20. The source and detector are moved linearly in increments to acquire individual measurement. After the completion of the linear measurement series across the object, both source and the detector are rotated to the next angular position to acquire the next set of measurements. Alternatively, the object itself can be rotated a preselected angle if it is not tied up to equipment above and below.

2nd generation: This setup is based on an array of detectors (8 detectors shown here) facing a single source. The source is collimated in such a way that the pattern of the beam is a fan with a thickness of about 5 to 10 mm. Although this is still a translation-

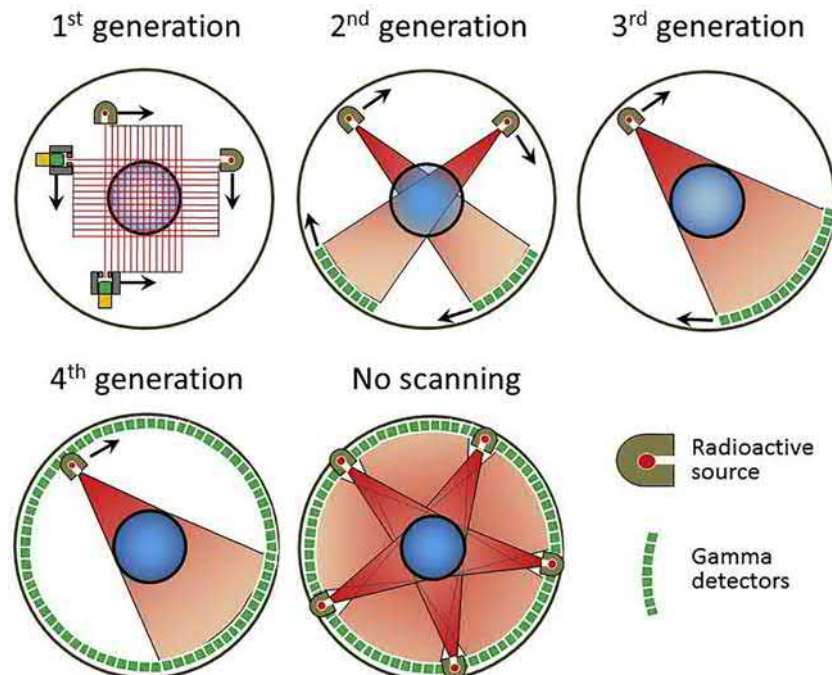


Fig. 20 Principles of different gamma tomography approaches from the simplest 1st generation setup to the most complicated with no movable parts. To some extent the figure also illustrates the developments which have taken place on a timeline. The figure is redrawn after information in ref. IAEA (2008).

rotation scanner, the number of rotation steps is reduced because the multiple pencil beams cover a larger area of the object in one single scan. Hence, the number of scanning angles can be reduced, and the scanning time considerably shortened.

3rd generation: This is perhaps the most popular scanner type. In this configuration, a larger number of detectors (here 18 individual detectors) are mounted in an arc concentric to the source. The size of the detector array is sufficiently large to cover the entire object in the detector field of view at all times. The relative position of the source and the detector remain stationary, while the entire apparatus rotates about the object. The data acquisition time is reduced relative to the two preceding scanners.

4th generation: This is a fixed detector-rotating source arrangement in which many detectors are mounted on a ring arrangement. The detector array forms a closed ring and remains stationary during the entire scan. The radiation fan beams sweep across the object while the source rotates continuously or stepwise in preset angle increments. One of the advantages of this scanner is less complicated mechanical setup, since the only moving component is the gamma source which, of course, is not wired, in contrast to moving wired detector arrays in the previous scanner versions.

5th generation: In fact, this is not a scanner in the sense that either source or detector or both move while scanning a pencil or a fan beam across the object. This version consists of several sources (5 sources in Fig. 20) positioned with equal spacing at a fixed distance from the scanning object. Each source emits a fan beam sufficiently broad to cover the whole diameter of the object. The transmitted radiation from each source is detected by a fixed array of individual detectors (12 detectors in Fig. 20) broad enough to cover the width of the fan beam. This setup can, in contrast to the previous 4, be used to scan objects with dynamic changes in nearly “true time” as demonstrated in ref. Johansen et al. (2014) (Fig. 21).

The type of tomographs described here can be applied for troubleshooting, for collecting information about internal structures of extended objects (beyond the range of traditional x-ray machines) and for monitoring dynamic processes in order to improve process modelling and optimization. However, each setup has its own limitations. Some of the limiting parameters or hardware requirements are tabulated in Table 2.

Gamma emission tomography

In gamma emission tomography there is no external radioactive source. The effective source here is a radioactive gamma-emitting component somehow distributed inside the object to be examined. The radioactive component can be a purposely located point source with isotropic gamma emission, a radioactive component more randomly distributed, or a fluid with a dissolved radioactive tracer compound. Outside the object, and in the simplest version, there is mounted a gamma detector that can be moved in a circular pattern around the object to detect radiation emitted from the source in different angles or the object itself can be rotated (Fig. 22). In more elaborate versions a ring of detectors is mounted around the object in a stable and regular pattern (Fig. 23). No scanning is needed for the generation of one slice tomogram. If the object is a vessel or tubing and the source is not a point source in fixed position, the ring of detectors may be moved axially to record new slice tomograms. In this way a 3D picture of the interior of the object may be generated.

One may distinguish between two somewhat different methods with acronyms SPECT and PET separately described below.

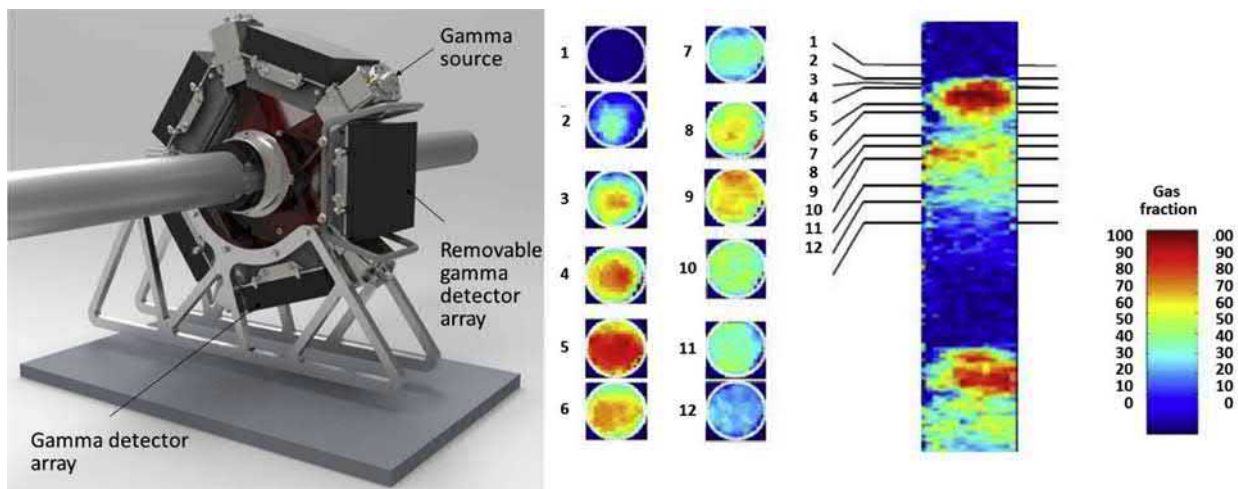
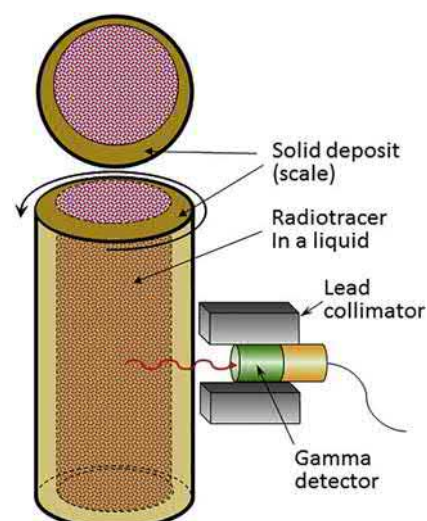


Fig. 21 5th generation gamma tomography system mounted on a transportation pipeline. Mind the removable gamma detector array that allows the scanner to be fitted around the pipeline. The pictured scanner is developed at the University of Bergen, Norway, by Johansen et al. (Johansen et al., 2014). The tomograph consists of 5 gamma sources of 18.5 GBq ^{241}Am (60 keV) each, and total of 85 CdZnTe semiconductor gamma detectors with 17 detectors in each of the 5 arrays. The tomogram slices shown to the right are extracted from a time series plot of a commingled oil and gas flow with data from one of the five detector arrays (17 detectors) of the high-speed gamma-ray tomograph. The time sequence is 2.5 s and the temporal resolution is 10 ms. The inner diameter of the pipe is 82 mm and the flow rates of the oil and gas are 100 and 50 cm/s, respectively. Courtesy of Prototech and University of Bergen.

Table 2 Approximate combinations of object sizes, needed gamma energies, suitable radionuclide sources and suitable tomography systems.

Object/vessel dimension	Approximate diameter/size	Needed gamma energies	Examples of radionuclides		Suitable for:
Small	< 150 mm	< 200 keV	x-rays, ^{241}Am (60 keV), ^{57}Co (122 keV)		All generations but special for 5th generation
Medium	150–600 mm	200–800 keV	^{133}Ba (356 keV), ^{137}Cs (661 keV)		All generations except 5th generation
Large	600–1500 mm	> 800 keV	^{59}Fe (1099 and 1292 keV)		Partly 1st and 2nd generation + 4th generation
Extra large	> 1500 mm	> 800 keV	^{60}Co (1174 and 1332 keV), ^{60}Co , ^{88}Y (1836 keV) and in special cases ^{56}Co (2598 keV)		4th generation but with narrow fan beam
Scanner Object	1st generation	2nd Generation	3rd generation	4th generation	5th generation
Static	Yes Rotation of object	Yes	Yes	Yes, Extended obj., narrower fan beam	Yes, but overkill
Dynamic (mass flow)	Very slow flow changes	Slow flow, time averaging	Slow flow, time averaging, Faster than 2nd	Slow flow, time averaging, Faster than 3rd	Yes, nearly true-time resolution.

**Fig. 22** Sketch of gamma emission tomography principle with radiolabelled internal volume, one detector and rotating object: Measurement of mineral scale distribution in a tubing section.

SPECT or Single Photon Emission Computed Tomography: This method utilizes radionuclides that emit one major gamma photon only. The optimal photon energy will vary according to the specific applications. In industrial setups where detector arrangements are as described above and for extended tomography objects, the energy may be relatively high (> 1 MeV), see Fig. 23. In research and development laboratories with smaller objects and where the detector consists of medical type gamma cameras (Anger cameras) with specially designed collimators, the photon energy must be much lower. The optimum energy is in the range of 140–150 keV, but some collimators can manage even 350 keV (Fig. 24). In addition to the shown parallel hole collimator, there are the converging hole collimator, the diverging hole collimator and the pinhole collimator, each with their own properties. An additional property to be considered for a SPECT-applicable radionuclide in industrial settings is its half-life. In contrast to the half-life of confined and encapsulated external sources used in gamma transmission tomography, which we want to be long (hundreds of days or rather years), the optimum half-life of SPECT-radionuclides are shorter, in the order of hours to a few days. These sources are distributed (and open) rather than confined and must be handled and taken care of after the experiment is finished. Short-lived radiotracers of this kind die away and leave no traces of radioactivity. Thus, radiological and environmental impact is strongly reduced.

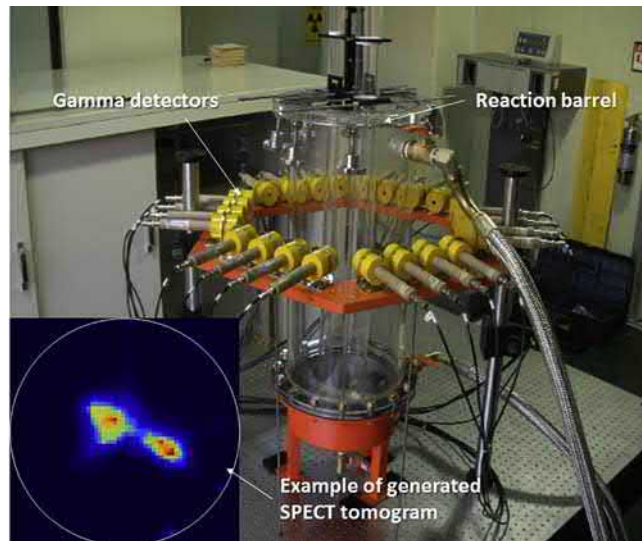


Fig. 23 SPECT setup with 24 gamma detectors positioned in a ring outside a reaction barrel. For demonstration: A phantom insert with a gamma emitter generates tomograms like the one in the lower left corner (Courtesy of Korean Atomic Energy Research Institute KAERI).

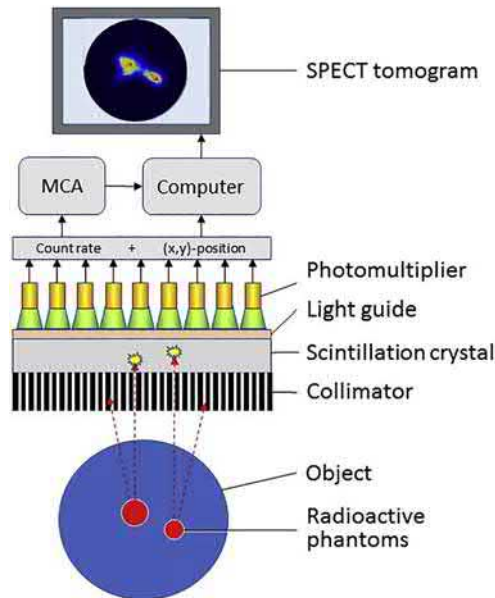


Fig. 24 Sketch of the main components in a gamma camera (Anger camera). Operation is exemplified by recording a 2D activity distribution of a phantom object.

Anger cameras are not frequently used in industrial settings except from experiments in laboratory environment. Hence, the energy limitation posed by the Anger camera does not apply. On the contrary: For examination of extended objects, higher gamma energies are in many cases better, and tomography arrangements like that sketched in Fig. 23, developed in various versions, are preferred. Such arrangements are more flexible with respect to type and size of detector, detector collimation and operations.

Table 3 lists some useful SPECT-radionuclides as tracers or label on tracers for fluids of different kind and for solid particles in industrial settings.

The table proposes a selection only of various possible alternative radiolabels where the timing, size and type of the examination object dictate the needed nuclear properties. They may all be produced by nuclear reactions induced by thermal neutrons in a reactor.

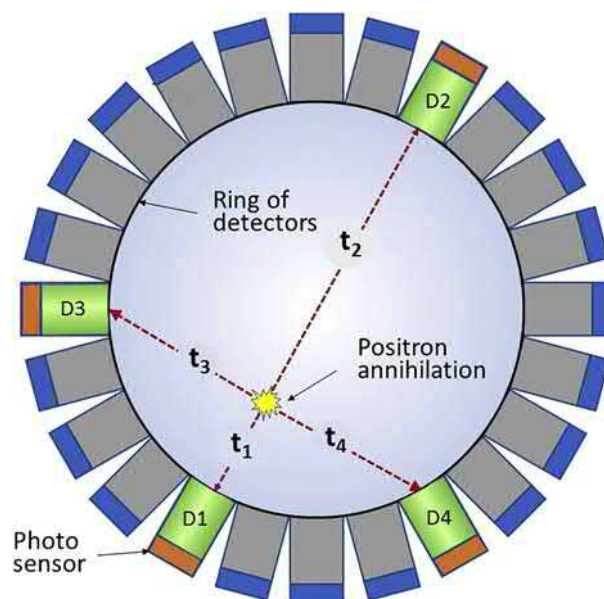
PET or positron emission tomography

This method utilizes radionuclides that decay by emission of a positron which is a positively charged electron or an anti-electron. By travelling some short distance in the surrounding medium and upon hitting an orbital electron in a nearby atom, the positron

Table 3 Important nuclear data for radionuclides useful for SPECT examination of industrial processes.

Radionuclide	Half-life	Gamma energy (% intensity)	Applications in:
^{24}Na	15.0 h	1369 (100), 2754 (100) keV	Aq
^{38}Cl	37.2 m	2167 (44), 1642 (33) keV	Aq, Org
^{41}Ar	109.6 h	1294 (99) keV	Gas
^{46}Sc	162.6 d	1121 (100), 889 (100) keV	Aq-C, Org-C, Sol-V
^{51}Cr	27.7 d	320 (10) keV	Aq-C, Org-C
^{56}Mn	2.58 h	847 (99), 1811 (27) keV	Aq-C, Org-C, Sol-S
^{82}Br	35.3 h	776 (84), 554 (72) keV	Aq, Org
^{79}Kr	35.0 h	261 (13), 398 (9) keV	Gas
^{86}Rb	18.6 d	1077 (9) keV	Aq
$^{99\text{m}}\text{Tc}$	6.0 h	141 (89) keV	Aq, Org-C, Sol-S
$^{113\text{m}}\text{In}$	99.5 m	392 (65) keV	Aq-C, Org-C, Sol-S
^{131}I	8.0 d	364 (82), 637 (7) keV	Aq, Org
^{140}La	1.68 d	1596 (95), 487 (46) keV	Aq-C, Org-C, Sol-V
^{198}Au	2.7 d	412 (96) keV	Sol-S
^{203}Hg	46.6 d	279 (82) keV	Aq-C, Org-C

Aq = aqueous solutions, Aq-C = complexes in aqueous solutions, Org = organic solutions, Org-C = complexes in organic solutions, Sol-S = solids surface labelling, Sol-V = solids volume labelling, Gas = gas labelling.

**Fig. 25** Principle sketch of a positron emission tomography system with detectors arranged in a ring around an object under examination.

combines with electron and both masses disappear under emission of two photons, both with energy of 511 keV, which is the equivalent energy of the mass of an electron. The two photons, which are generated simultaneously, are emitted with angle of (approximately) 180° with respect to each other (Fig. 25). This coincident and directional (relative to each other) photon emission is the basis for the technology called positron emission tomography or PET. Since around the mid-1980s it has found wide application in the medical sector, and most modern hospitals with radiological competence have implemented this technology. The development has been somewhat slower in industries not related to the medical sector.

The imaging principle is that the positron annihilation must be found somewhere on the straight line between the two detectors that have registered the two photons in coincidence. When many such events have been recorded between detectors in the detector ring, finally a "density" plot will appear where detector tie-lines are crossing each other most frequently. This represents a 2D map, tomogram, of activity distribution in this cross section. A 3D picture can be generated by moving the detector ring axially in increments or moving the object across the detector ring.

In recent PET developments, detector setups have become more and more sophisticated. Some types of instrumentation now use detector blocks which are combinations of several small detectors in a 2D array. This allows the possibility of recording not only a narrow 2D slice of the object for each position, but a 3D image over a defined volume. An additional important development is the introduction of time-of-flight, TOF, of the photons. In traditional PET technology, only photons coincident within a defined

time window and the corresponding detector position are recorded. In the new development, the *time differences* of the two photons are also recorded. With reference to Fig. 25, the time it takes for one photon to travel from the site of annihilation, say t_1 to detector D1, and the time it takes, t_2 , for the corresponding photon to reach detector D2, can now be recorded even though the photons travel with the speed of light. Thus, one can infer where on the tie-line between the two detectors the annihilation has taken place. This aids in improving the sharpness of the spatial resolution.

There is, however, one important physical factor in PET that is impossible to ignore and circumvent, and that affects the spatial resolution of this technology. Positrons are emitted from the radioactive nucleus with a certain kinetic energy. The energy depends on the specific radioactive nucleus. Therefore, the positron travels some distance in matter depending on the mass density, or rather on the electron density. The annihilation does not take place until the positron is near the end of its path. Thus, and in practice, the annihilation takes place at a position different from that of the nucleus that emitted the positron. This leads to blurring of the true position. However, modern picture reconstruction algorithms have come a long way to improve or compensate for this “built-in” physical limitation.

Some positron emitters, conditionally applicable in various industrial settings, are tabulated in Table 4. They may all be produced in simple proton- or neutron-induced nuclear reactions at small cyclotrons or in reactors, except from ^{82}Sr which requires somewhat higher bombarding energy.

Regarding detectors, some of the earlier technical applications of PET technology used position-sensitive wire grid chambers originally developed for high-energy physics. One such application was the examination of lubrication efficiency of Rolls Royce motors where the oil was labelled with a positron emitter (Stewart et al., 1988).

In practice, PET systems range from the very simple setups with two detectors only to the most complex pet scanners with hundreds of detector crystals systems most often based on solid scintillators. Originally, NaI(Tl)-detectors were dominating, and plastics based on organic scintillators like stilbene and anthracene were also in use. Today, a range of new detector crystals have been developed and implemented with improved overall properties for PET-applications.

The following is a brief listing of such detector scintillation crystals, some of them doped with cerium: Barium fluoride (BaF_2), bismuth germanate (BGO, $\text{Bi}_4(\text{GeO}_4)_3$), gadolinium orthosilicate (GSO, $\text{Gd}_2(\text{SiO}_4)\text{O}:(\text{Ce})$), lutetium-yttrium orthosilicate (LYSO, $\text{Lu}_{1.8}\text{Y}_2(\text{SiO}_4)_4\text{O}:(\text{Ce})$), lanthanum bromide ($\text{LaBr}_3:(\text{Ce})$), lutetium fine silicate (LFS, unknown (proprietary) chemical composition), lutetium aluminate (LuAP , $\text{LuAlO}_3:(\text{Ce})$), lutetium iodide ($\text{LuI}_3:(\text{Ce})$) and lutetium orthosilicate (LSO: $\text{Lu}_2(\text{SiO}_4)\text{O}:(\text{Ce})$). In addition, the semiconductor CZT ($\text{Cd}_{0.9}\text{Zn}_{0.1}\text{Te}$) is used. Detector technology is continuously improving.

A few industry-related applications

There are numerous applications of gamma emission and transmission tomography published in the literature. Here, a few examples are selected for illustration purposes.

Healthy or infected tree stems?

Detection of the oil palm stem rot diseases is a major issue in estate management and production in oil palm plantations. In order to avoid costly and time-consuming conventional manual diagnostic techniques based on chemical analysis of root stem tissue, an alternative method based on gamma transmission tomography has been developed (Abdullah et al., 2013). It consists of a transportable equipment that can readily be applied on living trees in plantations (see Fig. 26). This system produces high quality tomographic images that clearly differentiate between healthy and infected oil palm stems as shown in the inset in the upper left corner of Fig. 26. This method offers a promising alternative for in situ inspection of oil palm stem diseases compared to the more conventional methods, which, in extreme cases, include cutting down the trees.

Table 4 Selection of radionuclides with important nuclear parameters conditionally applicable for PET imaging of static objects and dynamic processes.

Radionuclide (nucl. react.)	Half-life	Positron energy (% intensity), ($E_{\beta\text{max}}$) (keV)	Gamma energy	Applications in ^a :
^{18}F	109.7 m	634 (97)	No γ	Aq-C, Org
^{22}Na	2.6 y	546 (90)	1275 (100) keV	Aq
^{44}Sc	4.0 h	1474 (94)	1157 (100) keV	Aq-C, Org-C, Sol-V
^{45}Tl	3.1 h	1040 (85)	Very weak γ	Aq-C, Org-C
^{48}V	16.0 d	694 (50)	984 (100), 1312 (98) keV	Aq-C, Org-C
^{64}Cu	12.7 h	653 (18)	Very weak γ	Aq-C, Org-C
^{68}Ga	67.7 m	1899 (88)	1077 (3.2) keV	Aq-C, Org, Org-C
^{82}Sr	25.3 d	1536 (82), 1169 (13)	776 (15) keV	Aq-C, Org-C
^{89}Zr	78.4 h	902 (23)	Weak γ , IT: 909 keV	Aq-C, Org-C
$^{94\text{g}}\text{Tc}$	4.9 h	811 (11)	871 (100), 703 (100) keV	Aq-C, Org-C, Sol-S
^{124}I	4.2 d	1534 (12), 2138 (11)	603 (63), 1691 (11) keV	Aq, Org

^aAq = aqueous solutions, Aq-C = complexes in aqueous solutions, Org = organic solutions, Org-C = complexes in organic solutions, Sol-S = solids surface labelling, Sol-V = solids volume labelling, Gas = gas labelling.

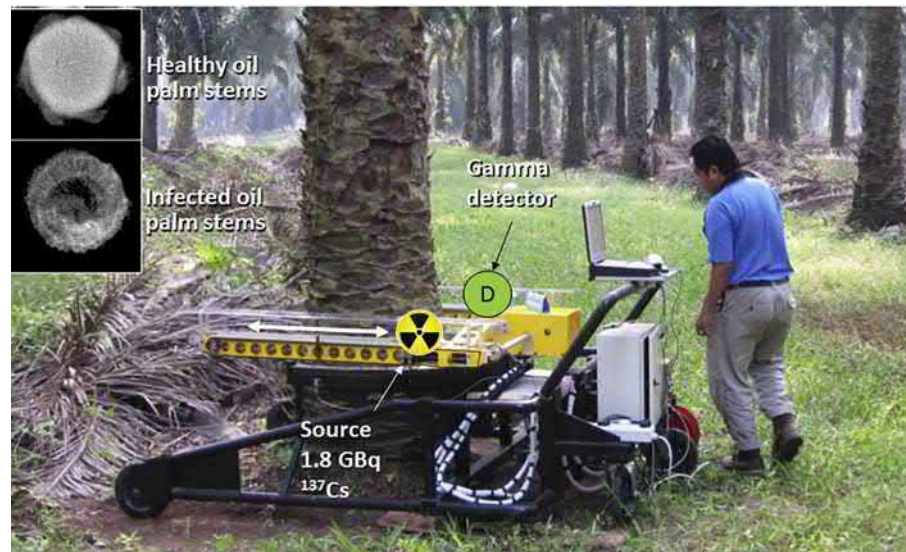


Fig. 26 Picture showing in-situ application of the gamma tomography system GammaScorpion to detect signs of stem rot in living trees on an oil palm plantation in Malaysia. Courtesy of Dr. J. Abdullah, Malaysian Nuclear Agency, Malaysia.

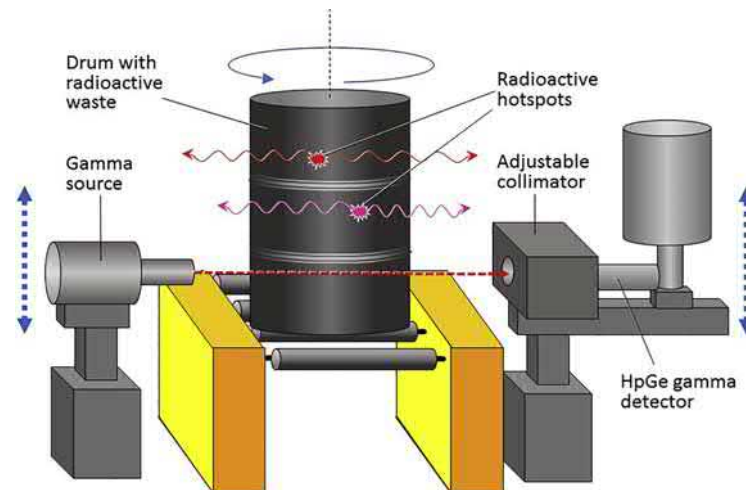


Fig. 27 Principle sketch of a scanning system for non-destructive assay of solid drummed radioactive waste from nuclear power plants. The system accommodates two scanning modes: Gamma transmission using an external radioactive source and gamma emission measuring on the gamma radiation from radioactive waste in the drum (Venkataraman et al., 2005).

Activity distribution in nuclear waste - combined transmission and emission tomography

An integrated system that combines gamma transmission and emission tomography was developed for the non-destructive assay of solid drummed radioactive waste from nuclear power plants (Venkataraman et al., 2005). The system is intended for assaying waste items up to 320 l with a contact dose rates up to a maximum of 2 Gy/h. A principle sketch of the scanning system is given in Fig. 27.

The scanning system is equipped with an adjustable aperture collimator on the detector side, and the distance of the detector from the drum can also be varied. The distance is chosen based on the measured dose rate of the item, and a selection of lead attenuators with different thickness may be applied to reduce count rate to reasonable levels.

The matrix density distribution is measured by monitoring transmitted gamma radiation from a source of ¹⁵²Eu with an activity of 0.55 GBq. ¹⁵²Eu has gamma emission with energies ranging from 122–1408 keV. It is a well-suited radionuclide for transmission experiments when used together with a detector with high energy resolution like the semiconductor HpGe (high-purity germanium).

In addition to the gamma transmission mode, the drum can also be scanned to map radioactivity distribution and hot spots by detecting the gamma radiation emitted from the drum itself, i.e. in the gamma emission mode. This is the main objective of the equipment. The object is scanned in 3-degrees of freedom; rotation, translation and elevation.

The derived density distribution is then used to correct gamma attenuation from the radiation emitted by the radioactive content in the drum. The system is not designed for a very high spatial resolution of the density or the radioactivity distribution, but rather for a high output. Resolution of approximately 50 mm voxel size is obtained, which is suitable for the purpose.

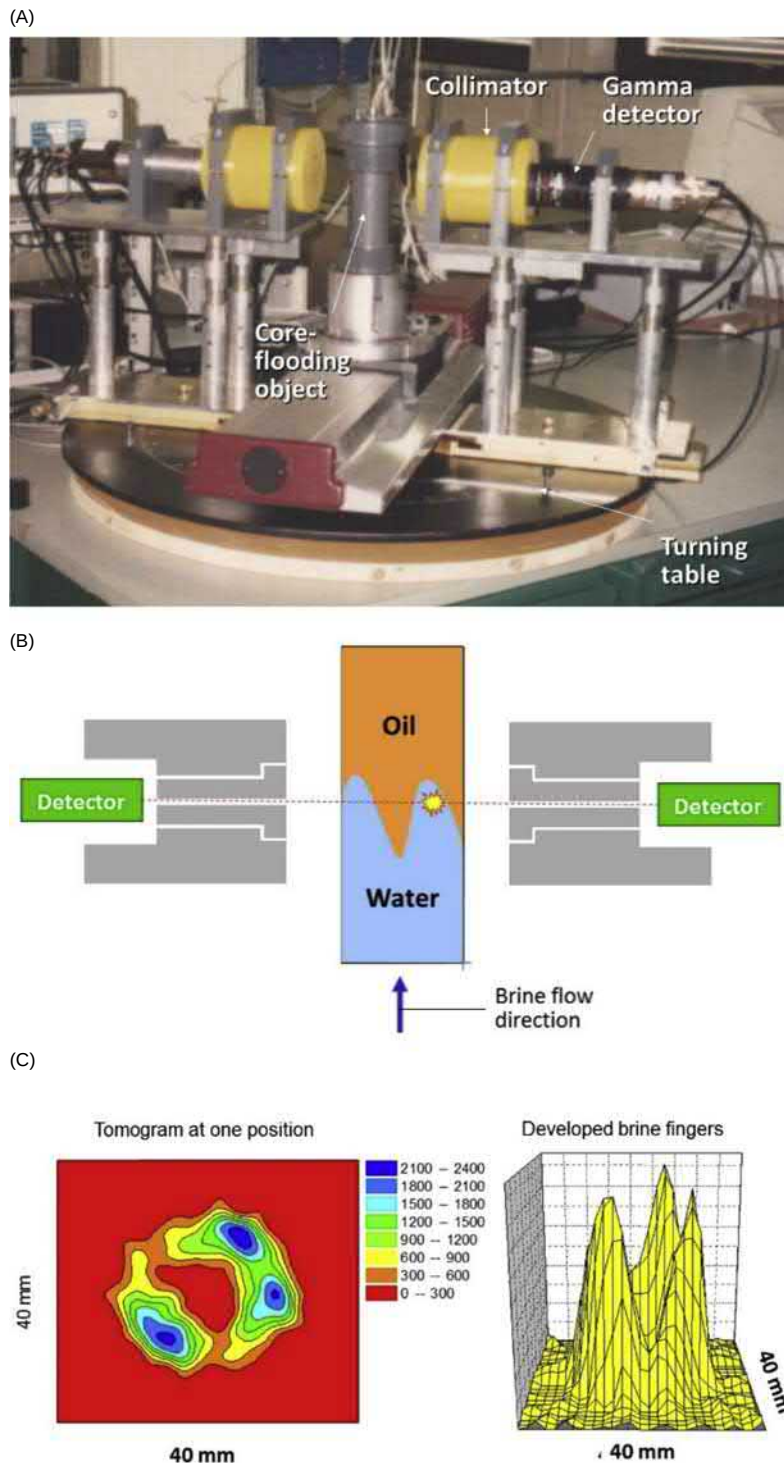


Fig. 28 Simple PET setup with two opposing detectors mounted on a 2D (semi 3D) scanner for detection of profiles of fluid flow during core-flooding in porous media Courtesy of Dr. A. Haugan, IFE, Norway.

Front instability when displacing oil with water - monitored with PET

In most oil reservoirs, water is injected in injection wells to displace oil from the reservoir rock and drive the oil forward to production wells. This is termed secondary oil production. However, if the viscosity of the water is considerably lower than that of the oil, the displacement front becomes unstable and so-called fingering develops. This results in a less efficient sweep of the reservoir and, thereby, lowers oil production. A considerable amount of movable oil is left untouched in the reservoir. One of the remedies is to increase the viscosity of water by injecting slugs of a water-soluble polymer in order to stabilize the displacement front (see also [Thereska, 2021](#)). This method belongs to the suite of so-called enhanced or tertiary recovery methods. During the development of improved oil recovery methods, it is important to increase the understanding of the mechanism and structure of fluid flow in

porous media like the reservoir rocks. Extensive laboratory experiments have been performed and are still being executed where such fluid flow and oil displacement mechanisms are studied. In this work, gamma (and X-ray) transmission and emission methods are attractive, and in some cases the only viable examination alternatives. X-ray transmission and gamma emission using SPECT and PET are in active use. While X-ray transmission cannot effectively be used if the density contrast between the fluids is absent or low, SPECT and PET are not sensitive to density contrasts. However, X-ray tomography (CT) is excellent to map permeability contrasts in the porous medium itself.

Fig. 28 shows a simple setup (“poor man’s PET”) with two NaI(Tl)-detectors that can be turned around the examination object and the object can be turned between the detectors. The object in this case is an encapsulated sandstone core saturated with oil. Brine labelled with the positron emitter $^{22}\text{Na}^+$ displaces the oil, and fingering develops. The flow is stopped at time intervals for recording of PET images.

Fig. 28C shows a recorded tomogram with developed brine fingers. Thus, brine properties and heterogeneities of the core plug may be modified to examine the effect on the brine front stability.

More sophisticated PET core-flooding experiments have been published using multi-modality systems where medical PET scanners have been combined with CT (Fernø et al., 2015; Brattekkås et al., 2020; Hu et al., 2020).

Studies like the ones described here generate data for a fundamental understanding of fluid flow in porous media and, thereby, for mathematical model development of these processes. Such models, which are developed on the pore-core scale, are again the basis for upscaling attempts and, thereby, for full-scale reservoir simulation models. Work continues along these lines.

See Also: Phenomenology of Gamma Ray and Charged Particles Interactions.

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Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection

Tor Bjørnstad, Institute for Energy Technology, Kjeller, Norway; and University of Oslo, Oslo, Norway

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Glossary

Barn A unit for reaction probability of a nuclear reaction, also called reaction cross section. $1 \text{ b} = 1 \text{ b} \equiv 10^{-24} \text{ cm}^2$ (or 10^{-28} m^2).

Bremsstrahlung Electromagnetic radiation (also sometimes called X-rays) generated by charged particles (normally electrons) when accelerated (or bent) in an electromagnetic field. This “type” of X-rays has a continuous energy distribution without definite structure, with energies stretching from eV to several tens of MeV, depending on the kinetic energy of the accelerated particle.

Excitation function A graphical representation (plot) of the reaction cross section as a function of the energy of the bombarding particle.

Gas proportional counter This is a type of gaseous ionization detector device used to measure particles of ionizing radiation including neutrons. The two most well-known are the ^3He -filled detector, based on detection via the nuclear reaction $n + ^3\text{He} \rightarrow ^3\text{H} + p$, and the BF_3 -filled detector, based on the nuclear reaction $n + ^{10}\text{B} \rightarrow ^7\text{Li} + \alpha$. The emitted charged particles (p and α) give rise to an electrical pulse that is amplified and detected. The Intensity of the pulse is proportional to the energy of the incoming neutron.

LIPAC Linear prototype accelerator—an initiative in a Japanese/European program called IFMIF (International Fusion Materials Irradiation Facility) to provide an accelerator-based, D-Li neutron source to produce high energy and intensity neutrons for fusion research. The initiative includes building a prototype accelerator with a 9 MeV deuteron beam with intensity of 125 mA.

MCP Microchannel plate, a detector concept for cold, thermal and epithermal neutrons consisting of a micro-perforated (microchannels) glass plate where the glass matrix contains nuclides with high neutron capture cross sections like ${}^6\text{Li}$, ${}^{10}\text{B}$ or ${}^{155,157}\text{Gd}$. Nuclear reactions induce emission of charged particles, which upon recoiling out of the matrix cross the micro-perforations and kick out an avalanche of electrons from the tube walls. These electrons are accelerated and multiplied in an electrical field and exit the plate with a measurable intensity. This detector is position sensitive with very high spatial resolution.

NAA Neutron activation analysis, a sensitive analytical technique for qualitative and quantitative analysis of chemical elements in nearly any type of matrix. It is based on irradiation of the sample with neutrons (various energies), thereby generating radioactive nuclides in the sample, followed by detection of the emitted radiation, which is element (and isotope) specific like a fingerprint.

PGNAA Prompt gamma neutron activation analysis, an analytical technique for qualitative and quantitative non-destructive analysis of chemical elements in nearly any type of matrix. It is based on irradiation of the sample with neutrons (various energies are possible), inducing nuclear reactions like absorption or scattering, thereby exciting the nuclei to higher energy levels followed by spontaneous and fast de-excitation by emission of gamma rays. Detection of the intensity and energy of this radiation uniquely defines the chemical element (even the isotope of this element) and its concentration.

Scintillation detector A detector principle for ionizing radiation consisting of the two concepts solid scintillation and liquid scintillation. A solid scintillator is a (mono) crystal or another solid material (doped plastics) which, upon hitting by gammas, neutrons or charged particles, generates isotropic emission of photons, which may be detected by a photomultiplier and further amplified into a measurable current pulse. A liquid scintillator works with the same principle, but is based on a liquid solution of the scintillating compounds.

Semiconductor detector This is a detector for nuclear radiation based on the semiconductor principle. When applied for neutron detection, typically a planar thermal neutron conversion layer consisting of a thin film of neutron reactive material is deposited on the surface of a semiconductor diode. ${}^{10}\text{B}$ and ${}^6\text{LiF}$ are frequently used as conversion layer materials. The reaction product particles emitted in the neutron-induced nuclear reactions with ${}^{10}\text{B}$ (α -particles) and ${}^6\text{Li}$ (α - and ${}^3\text{H}$ -particles) will create electron-hole pairs in the semiconductor material which will travel to the positive, respectively to the negative, pole in an electric field set up over the semiconductor thus creating a current pulse. The number of electron-hole pairs is proportional to the energy of the radiation to the semiconductor.

SONATE This is an on-going French initiative to develop an accelerator-based neutron source with sufficiently high neutron flux for neutron scattering investigations, thus replacing nuclear reactors for this purpose. This design is referred to as "SONATE." It is based on acceleration of a high-current proton beam to energies < 50 MeV impinging a lithium or beryllium target.

Introduction

A neutron is a non-charged sub-atomic particle with masses slightly greater than that of a proton. Together with positively charged protons, neutrons are bound with the strong nuclear force in atomic nuclei. In this state, neutrons are of no use for material testing. However, neutrons can also exist, one by one, outside the nucleus in a free state. In this form, the neutron has a half-life of ~ 10.2 min. It decays into a proton, an electron and an antineutrino according to the equation

$$n^0 \rightarrow p^+ + e^- + \bar{\nu}_e$$

The neutron has an internal structure (Garcia, 2021a) consisting of quarks (one up-quark and two down-quarks) as shown in Fig. 1. That is the reason why the neutron should no longer be categorized as an elementary particle as was common years ago, and

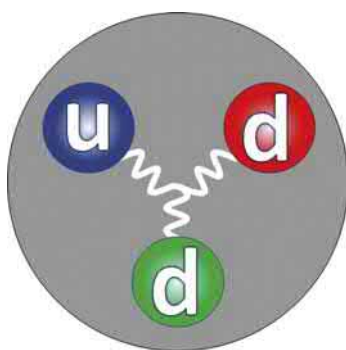


Fig. 1 Quark structure of the neutron: u = up-quark, d = down-quark. Neutron decay leads to formation of two up-quarks and one down-quark.

which can still be found in technical texts from qualified laboratories. This internal structure has, however, no relevance to the practical use of neutrons, and will therefore not be explained further here.

Since the neutron has no electric charge, it is not affected and cannot be accelerated in electric fields. However, the neutron has a magnetic moment and can, therefore, be influenced by magnetic fields. This interaction, together with other properties like scattering, allows slow neutrons to be guided from its source of production to an experiment several meters away (see also Garcia, 2021a; Dewji and Miller, 2021).

Neutrons with high energies (so-called fast neutrons) can be produced in various nuclear reactions, for instance with charged particles as the bombarding beam. The resulting neutron may then be slowed down (moderated) to energies lower than that resulting from the nuclear reaction. At these slow, cold and ultra-cold energies, the neutron shows a strong interaction with several chemical elements, quite in contrast to interactions with gamma rays.

These special properties make neutron interactions and gamma interactions with matter complementary methods for examination of materials. This complementarity is utilized in numerous practical applications since the two irradiation forms are influenced by different properties of an examination object.

Neutron interactions with matter

Fig. 2 systematizes the various nuclear interactions that a neutron can have with matter, i.e. with atomic nuclei. Fig. 3 schematically visualizes some of the reaction types mentioned in Fig. 2.

Scattering reactions

This term includes elastic scattering (n, n) and inelastic scattering (n, n'). In elastic scattering a fast neutron imparts a fraction of its kinetic energy to the target nucleus without transferring excitation energy to that nucleus. If the neutron is thermal, that is, in thermal equilibrium with the medium nuclei, there is no net energy transfer between the neutron and the nuclei. In inelastic scattering a fast incoming neutron is scattered away with a lower energy than it had upon arrival. The difference is given to the nucleus as excitation energy which can be followed by emission of gamma radiation. Very low energy neutrons can also undergo diffraction like waves (Wietfeldt, 2021). This very specialized area is covered in (Barradas, 2021; Wietfeldt, 2021). In general, the different forms of scattering reactions utilize neutrons with energies from super-cold to fast (see classification below). Each isotope of the various chemical elements has its unique probabilities for such interactions. We call this probability (microscopic) cross section. The unit is *barn* (b), and it has the dimension of area ($1 \text{ b} = 10^{-28} \text{ m}^2$ or 10^{-24} cm^2 which is more frequently used in formulas) denoted in Fig. 2 by σ_e for elastic scattering and σ_i for inelastic scattering. Cross sections for each type of nucleus depend on the neutron energy. The energy dependence curve is called the *excitation function*. The parameter Σ in Fig. 2 is explained in the text below.

Absorption reactions

This term describes two mechanisms: (1) compound nucleus formation without fission and (2) compound nucleus formation followed by fission. Common for the two is that the nucleus absorbs a neutron. The probability for this event is generally highest at low

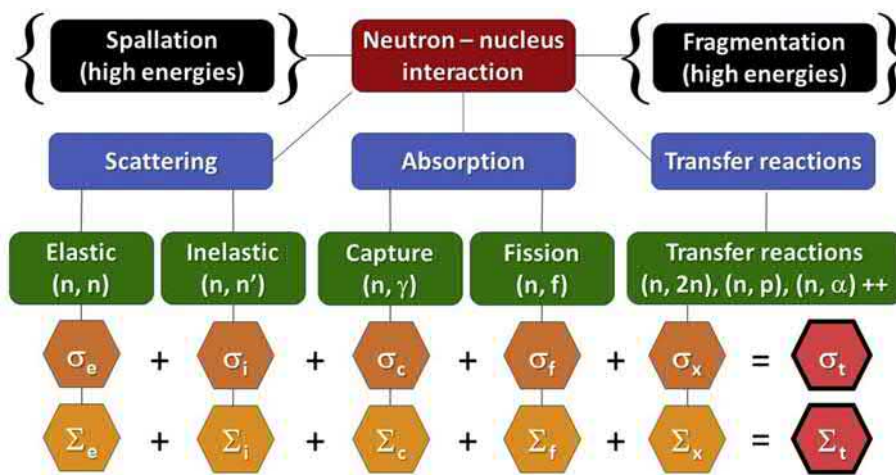


Fig. 2 The figure intends to show the various types of reactions that neutrons induce when interacting with atomic nuclei. The main reactions of relevance for material testing are the scattering, absorption and transfer reactions. Other reactions like neutron-induced spallation and fragmentation reactions can only take place with high-energy neutrons ($E_n > 100 \text{ MeV}$) and are, therefore, of no interest as direct probes in this text. However, proton-induced spallation installations are modern large-scale tools to provide high fluxes of spallation neutrons for a number of material investigations. Some of the features in the figure is explained in the text.

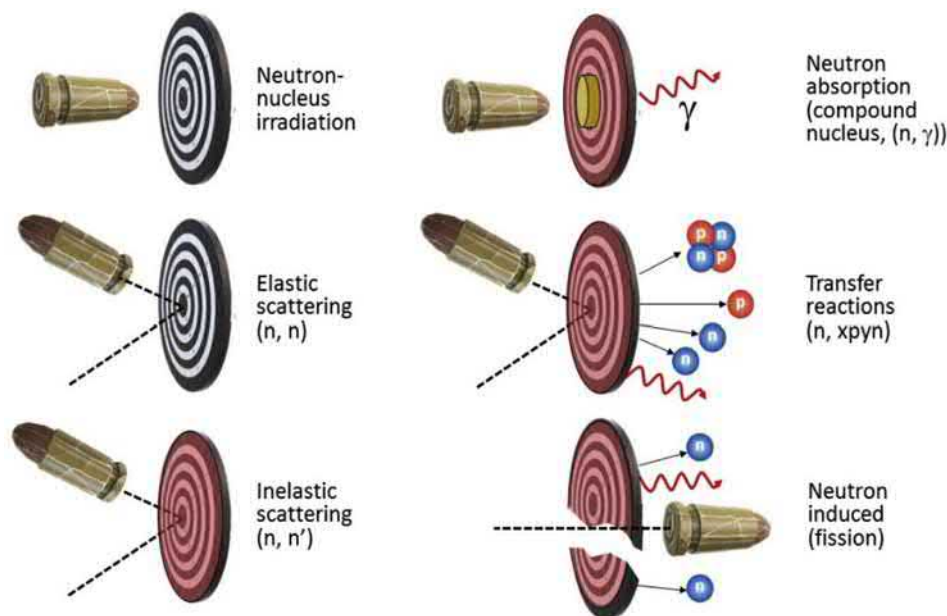


Fig. 3 Schematic illustration of some neutron reactions with nuclei. The mechanisms include neutron scattering, absorption (compound nucleus formation) with emission of prompt gamma rays, transfer reactions where charged particles, neutrons and gamma rays are promptly emitted and fission of certain heavy nuclei where additional neutrons and gamma rays are emitted.

and thermal energies. The process leaves the nucleus in a highly excited state corresponding to the binding energy of the neutron in the nucleus (5–8 MeV).

Let us first consider the reaction without fission. Most of the time the nucleus gets rid of the excitation energy by emitting an avalanche of gamma rays, more seldom particles and, “finally,” after some picoseconds, ends up in the ground state (**Fig. 4**). A new isotope of the bombarded element has been formed. Here, there are two features to be noted: (1) the energies and intensities of the prompt gamma rays are unique for the formed isotope and (2) the new isotope may (or may not) be radioactive ([Previtali, 2021](#)). Upon decay it may emit gamma rays where energies and intensities are also unique for the decaying radionuclide. We say that this gamma ray pattern is the “fingerprint” of the radionuclide. These properties are the basis for *prompt gamma neutron activation analysis* (PGNAA) and ordinary *neutron activation analysis* (NAA). Such processes add to the extensive suite of nuclear methods for material examination since they describe qualitatively and quantitatively the isotopic content, and thereby also the elemental content for elements that can be considered to have the natural isotopic composition.

For heavy nuclei such as ^{233}U , ^{235}U , ^{239}Pu , absorption of a thermal neutron makes the resulting nucleus so unstable that it tends to break up into two pieces, one fragment somewhat lighter than the other. The fission is accompanied by emission of 2–3 new neutrons per event. This process is the basis for running nuclear reactors, where the surplus of neutrons is used both in activation and scattering experiments. If high-energy neutrons are available as bombarding particle, several other heavy nuclei may undergo fission. Examples are ^{232}Th and ^{238}U .

Again, these processes are governed by microscopic cross sections, here called σ_c and σ_f .

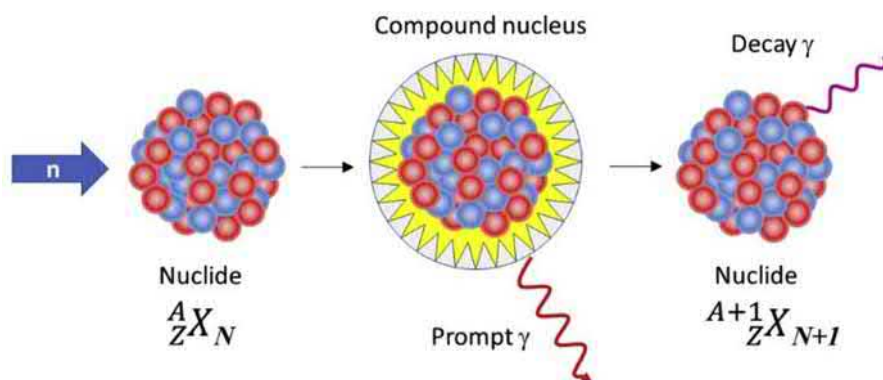


Fig. 4 Illustration of the neutron capture process forming a compound nucleus, which decays to its ground state by emission of prompt gamma rays.

Transfer reactions

These reactions are most relevant for neutrons with energies in the range of several MeV. The neutron transfers parts of its kinetic energy to neutrons and protons in the bombarded (target) nucleus. This energy may exceed the neutron or proton separation energies (binding energies) in the nucleus. The particles are emitted from the nucleus, and new nuclei are formed. Alpha particles may also be emitted in this process as illustrated in Figs. 3 and 5. Each of the reactions has a probability expressed by a cross section value for each individual nuclide. In Fig. 2 this is denoted as σ_x , which is the sum of all individual microscopic transfer reaction cross sections ($\sigma_{n,p}$, $\sigma_{n,\alpha}$, $\sigma_{n,2n}$, etc.). The individual probabilities differ in magnitude, i.e. some reactions proceed more readily than others. The probability for a certain reaction channel may be compared with a barrel of water with outlet channels of different sizes. The water outflow is directly proportional to the size of the opening. Thus, with reference to Fig. 5, the highest cross section for this special nuclide is for the (n,2n)-reaction.

Neutron overall interactions relevant for transmission experiments

It is possible to ascribe a total microscopic cross section, denoted σ_t in Fig. 2, for a chemical element by combining all the individual isotopic cross sections and multiplying it with the fraction that each individual isotope contributes to the chemical element, which we call the isotopic abundance. We define the total macroscopic cross section Σ_t as:

$$\Sigma_t = \sigma_t \cdot N \quad (1)$$

Here, N is the atomic density (atoms/cm³) of the chemical element. Strictly speaking, Σ is not a cross section since it has the dimension cm⁻¹, but it can be used as an attenuation coefficient for neutron transmission through matter. Thus, neutron transmission through a material slab consisting of only one single chemical element is, as for transmission of gamma radiation (see Bjørnstad, 2021a,b), simply described by Lambert-Beer's law

$$I_d = I_0 \cdot e^{(-\Sigma_t \cdot d)} \quad (2)$$

where d = average penetration length in cm. If the material slab consists of more chemical elements, say 3, homogeneously mixed, with macroscopic cross sections Σ_{t1} , Σ_{t2} and Σ_{t3} and individual penetration lengths d_1 , d_2 and d_3 , the expression becomes

$$I_d = I_0 \cdot e^{(-[\Sigma_{t1} \cdot d_1 + \Sigma_{t2} \cdot d_2 + \Sigma_{t3} \cdot d_3])} \quad (3)$$

By introducing the number of atoms per unit volume of element $i = N_i$, the effective transmission length or path of element $i = d_i$, the energy-dependent total microscopic cross section for the isotope j of element $i = \sigma_{ij}(E)$ and the isotopic abundance of isotope j of element $i = A_{ij}$, the detailed final expression becomes

$$I_t(E) = I_{t0}(E) \cdot e^{\left(-\sum_i N_i \cdot d_i \cdot \sum_j \sigma_{ij}(E) \cdot A_{ij}\right)} \quad (4)$$

Σ_i and Σ_j in the exponent of Eq. (4) are not the macroscopic cross sections but the symbols for summation.

Neutron energy classification

Neutron energies can span the range from super-cold to very high (> 10 MeV) energies. Still, classification of neutron energies into selected "energy bins" seems to be useful. The reason is that neutrons with energies within defined ranges show different interactions

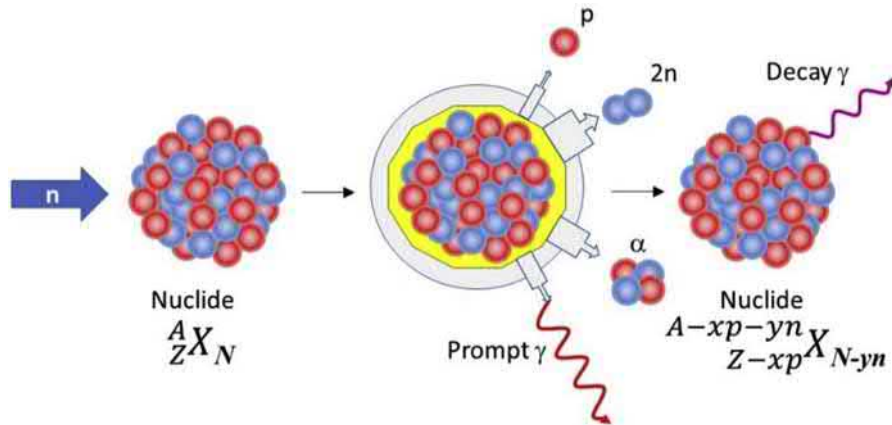


Fig. 5 Illustration of the neutron transfer reactions with emissions of protons, neutrons, alpha particles and prompt gamma rays before the resulting nuclide ends up as an isotope of a chemical element different from that of the target element.

with material. There is, however, no strict and unified agreement in the literature or in the nuclear community on how to classify. In the text below, one such classification is proposed.

Ultra-cold neutrons—Below 0.001 eV

The production, transportation and storage of ultra-cold neutrons (UCN) is currently motivated by their usefulness as a tool to determine properties of the neutron itself and to study various fundamental physical interactions. Such fundamental interactions will not be discussed in this text since they have no clear practical relevance for material examination. It can be mentioned that UCNs have been used to measure the presently accepted half-life of neutrons of ~ 10.2 min.

Cold neutrons

Neutrons of energies between 0.001 and 0.01 eV are termed cold neutrons and are used in several types of scattering (diffraction) applications such as studies on the structure of biological molecules and particulate objects in the nanometer range. Absorption reactions can also occur at these energies with high cross sections (see example of ^{238}U in Fig. 6). This is of value both for transmission experiments, for increased production of selected radionuclides, and for analytical purposes (NAA and PGNA).

Thermal neutrons

Neutrons that are in thermal equilibrium with the motion of the atoms of a medium at room temperature (20°C) are called thermal neutrons. Their energy is described by a Maxwell-Boltzmann distribution that has a maximum value at 0.0253 eV. In nuclear reactors with higher moderator and cooling medium temperature, this maximum shifts to higher values. Thermal neutrons are used in neutron diffraction studies of material molecular structures, in neutron scattering and radiography for mechanical structures and in activation analysis of elemental compositions of materials.

Epithermal neutrons

Epithermal neutrons have energies between thermal (0.025 eV) and a few hundred eV. The epithermal region is characterized by strong oscillations in the excitation function for most nuclei. These oscillations are related to the internal structure (excited states) of the bombarded nucleus: At some defined energies the probability of absorbing a neutron (n, γ) becomes high, corresponding to populating definite excited states. There are also fission and scattering resonances. As the neutron energy increases, the density of the oscillations increases because the spacing of excited states decreases. For some nuclei, especially among the lanthanide elements, these oscillations are very strong. Hence, in conventional neutron activation analysis of elemental contents of materials, one often prefers epithermal rather than thermal neutrons. Neutrons in this energy region can be separated from thermal neutrons by absorbing the thermal neutrons in a cadmium foil, which has a very large capture cross section for thermal neutrons and is relatively transparent for epithermal neutrons. Other reactions like elastic and inelastic scattering also take place in this energy range.

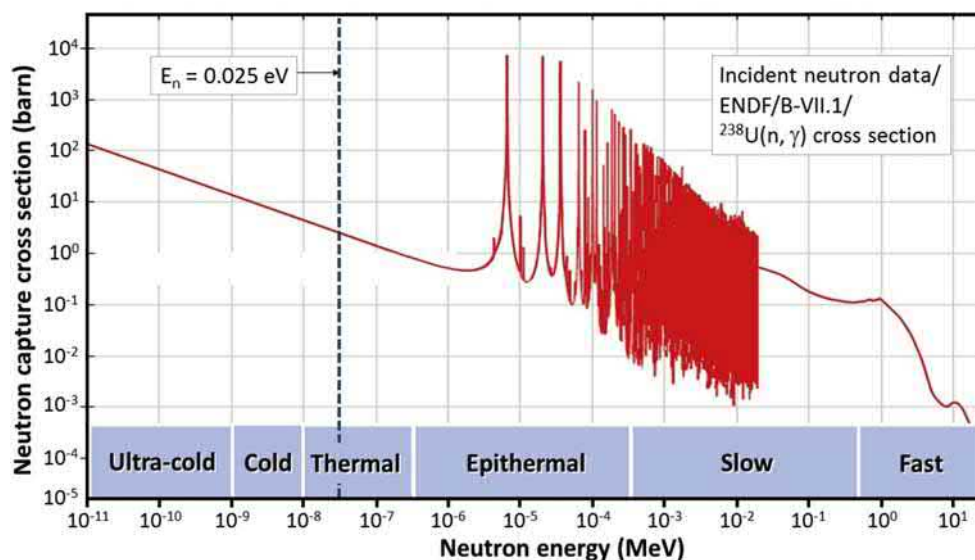


Fig. 6 Cross section for (n, γ)-reaction of ^{238}U as function of incoming neutron energy. Curves like this are called excitation functions for the actual nuclear reactions, here for the (n, γ)-reaction. The figure also defines approximate regions for ultra-cold, cold, thermal, epithermal, slow and fast neutrons.

Slow neutrons

Slow neutrons have energies between hundreds of eV to 0.5 or 1 MeV. This energy range overlaps into the resonance region and strong neutron absorption can take place. However, it is not the most useful energy region for material testing, although both elastic and inelastic scattering take place, which can be used in neutron radiography and tomography.

Fast neutrons

Generally, fast neutrons have energies between 0.5 and 10–20 MeV. This is the energy region where transfer reactions start to be possible. (n,p)- and (n, α)-reactions dominate for nuclei with $Z < \sim 40$. For most nuclei there is a threshold energy below which transfer reactions cannot take place. This is because the outgoing charged particle must penetrate the coulomb barrier set up by the nuclear charge. Threshold energies for different elements vary in the approximate region 2–12 MeV. There are a few examples of (n,p)- and (n, α)-reactions which, in fact, are exothermic, i.e. there is no energy threshold for the neutron. One such example is the formation of ^{14}C from ^{14}N by the reaction $^{14}\text{N}(\text{n,p})^{14}\text{C}$. Two examples for (n, α) reactions are $^6\text{Li}(\text{n},\alpha)^3\text{H}$ and $^{10}\text{B}(\text{n},\alpha)^7\text{Li}$, where there are no lower limits for the neutron energy; their thermal cross sections are exceptionally high, i.e. 940 b and 3840 b, respectively. (n,2n) reactions are dominant for elements with $Z > \sim 40$. The threshold energies start at higher values than those for the previously mentioned reactions because sufficient energy must be transferred to overcome the neutron binding energy in the nucleus. These reactions are of interest in nuclear activation analysis of elemental compositions of materials. Such neutron energies are also used in fast-neutron transmission radiography and tomography experiments.

High energy neutrons

High-energy neutrons have energies above 20 MeV and are found in cosmic radiation, where energies as high as 1000 MeV has been measured. In the laboratory, high-energy neutrons can be produced in high-energy spallation reactions. An example is the high-energy neutron facility at Los Alamos Lab (Nowicki et al., 2017). In Spallation reactions with 800 MeV protons on tungsten, they obtain a neutron flux in the energy range 10–600 MeV of $2 \cdot 10^6 \text{ n/cm}^2 \cdot \text{s}$, where it seems that about 25% have $E_n > 100 \text{ MeV}$. This neutron flux is $3.5 \cdot 10^8$ times larger than the cosmic neutron flux at sea level. These fluxes make possible dose effect experiments of high-energy neutrons on biological systems. Such neutrons may also be used for testing of the survival of electronic components to be used in space missions. For ordinary and routine material examinations, however, such high-energy neutrons have no obvious use at present.

Neutron sources

This sub-chapter provides a brief description of the main sources for free neutrons to be used in various forms of material testing (see also Wehe, 2021; Miller et al., 2021; Garcia, 2021b).

Isotopic neutron sources

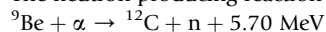
The simplest neutron producing devices are the isotopic (and isotropic) neutron sources. Three main types of such neutron sources can be distinguished. One type is based on alpha-emitters, usually mixed with beryllium metal, which produce neutrons through the (α ,n)-reaction. The second is based on high-energy gamma-emitters, usually mixed with deuterium or beryllium producing neutrons through the (γ , n)-reaction. The third is simply selected isotopes of heavy elements (actinides) that undergo spontaneous fission with emission of $\sim 3 \text{ n/f}$, where ^{252}Cf is the most prominent example. The reason why beryllium (^9Be) and deuterium are chosen is that these are the two radionuclides with the lowest neutron separation energies at 1667 and 2220 keV, respectively. Extensive overviews of such sources are found in Hoste (1988) and Knoll (2010).

Table 1 lists some possible isotopic neutron sources containing beryllium and provides also some information about a gamma-based source and a source based on spontaneous fission.

Neutrons can be generated in α -reactions with target material nuclides other than ^9Be , but with substantially lower yields. Here follow some examples, in brackets is the neutron yield in % of the yield from a beryllium target: ^7Li (3.3%), ^{10}B (16.3%), ^{11}B (32.5%), ^{13}C (12.5%), ^{18}O (36.3%) and ^{19}F (15.0%). Thus, the only practical target material for α -induced neutron production is beryllium.

For all aspects considered, the source based on $^{241}\text{Am}/\text{Be}$ stands out as the most versatile and general-purpose isotopic neutron source.

The neutron-producing reaction is as follows:



The source strength (n/s) is proportional to the activity of ^{241}Am . Sources with more than 50 Ci (1850 GBq)¹ have been produced. Sources in use in industry are typically in the range 50 mCi (1.85 GBq) to 10 Ci (370 GBq), smaller for hand-carried

¹1 Ci (Curie) = $3.7 \cdot 10^{10}$ disintegrations per second (dps) = $3.7 \cdot 10^{10}$ Bq (Becquerel, 1 Bq $\equiv$ 1 dps).

Table 1 Some isotopic neutron sources with selected nuclear data, average neutron energy and neutron yields.

Neutron source	Half-life	α -Energy (intensity) (MeV ^a)	Average neutron energy (MeV)	Neutron yield n/s·Ci (n/s·GBq)
²¹⁰ Po/Be	138 days	5.30 (100%)	4.2	$2.5 \cdot 10^6$ ($6.8 \cdot 10^4$)
²²⁶ Ra/Be	1600 years	4.78 (93.8%)	3.9	$1.5 \cdot 10^7$ ($4.1 \cdot 10^5$)
²²⁷ Ac/ ²²⁷ Th/Be	21.8 years	$5.76 + 5.97 + 6.06$ (68.1%)	3.8	$1.5 \cdot 10^7$ ($4.1 \cdot 10^5$)
²³⁸ Pu/Be	87.7 years	5.50 (70.9%)	4.0	$2.8 \cdot 10^6$ ($7.6 \cdot 10^4$)
²³⁹ Pu/Be	24,110 years	5.16 (70.8%)	4.6	$1.6 \cdot 10^6$ ($4.3 \cdot 10^4$)
²⁴¹ Am/Be	432.6 years	5.49 (84.8%)	4.5	$2.2 \cdot 10^6$ ($6.0 \cdot 10^4$)
²⁴⁴ Cm/Be	18.1 years	5.80 (76.9%)	4.3	$3.0 \cdot 10^6$ ($8.1 \cdot 10^4$)
¹²⁴ Sb/Be	60.4 days	Gamma: 1691 (47.6%)	0.02	$1.9 \cdot 10^5$ ($5.1 \cdot 10^3$)
²⁵² Cf	2.65 years	Spontaneous fission	1.0	$2.34 \cdot 10^6/\mu\text{g}$ ($6.3 \cdot 10^4$)

^aIntensity of the strongest α -energy.

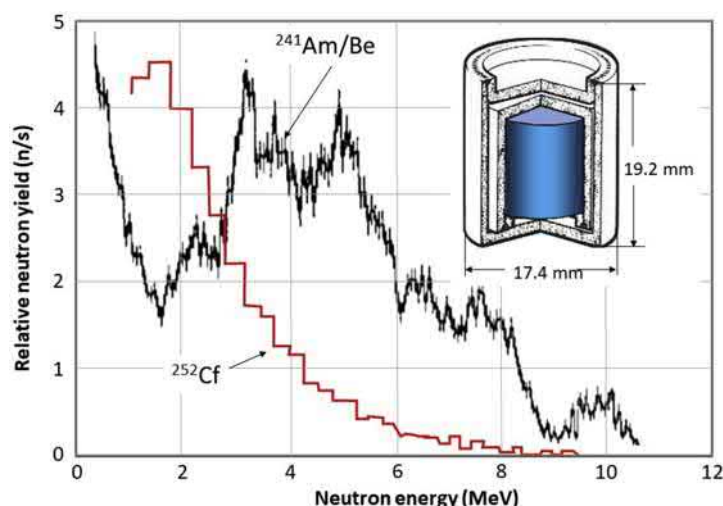


Fig. 7 Energy spectrum of neutrons from a ²⁴¹Am/Be isotopic neutron source reproduced from information in Marsh et al. (1995). Example of such a source is shown in the inset. For comparison, a neutron energy spectrum from a ²⁵²Cf fission neutron source is shown (red). Reproduced from Hassan AM (2001) Uses of isotopic neutron sources in elemental analysis applications. In: 3rd Conference on Nuclear & Particle Physics (NUPPAC 01), Cairo, Egypt, 20–24 October. Available from <https://www.osti.gov/etdweb/servlets/purl/20808702>.

devices. The physical size of the source can be tailormade to the various application but is generally small as indicated in Fig. 7. It is encapsulated in a double stainless-steel housing. Neutron energies vary from nearly thermal to about 11 MeV with an average around 4.5 MeV (Fig. 7). In the neutron-producing nuclear reaction a prompt gamma energy of 4.44 MeV is emitted from an excited state in ¹²C. The dose rate from an unshielded source, which is relevant in the human handling process, has been measured to 0.68 $\mu\text{Sv/h}^2$ at 1 m/GBq.

Table 1 also contains some information for the gamma-based ¹²⁴Sb/Be source. Other high-energy gamma emitting radionuclides may be used like ²⁴Na ($E_\gamma = 2754$ keV, used both with beryllium and deuterium) and ⁸⁸Y ($E_\gamma = 1836$ keV, used with beryllium). Both are challenged with some impracticalities. The first one can be produced in a nuclear reactor in sizable amounts but is too short-lived ($T_{1/2} = 15$ h). The second one ($T_{1/2} = 106.6$ days) must be produced by charged particle irradiation in a cyclotron or a linear accelerator with particle currents limited to the microampere (maximum 1 mA) region and with relatively small reaction cross sections.

The best among these is the ¹²⁴Sb/Be source. The radioactive ¹²⁴Sb-source is placed in a cavity in the beryllium block. When the activity becomes too low due to radioactive decay, it can in principle be removed, reactivated in a nuclear reactor and mounted back into the beryllium block. However, all isotopic neutron sources based on a high-energy gamma emitter are impractical in transportable devices due to the need for heavy shielding against gamma radiation.

The ²⁵²Cf source is more versatile. It is used both in industry (for instance well logging in the petroleum industry) and in permanent laboratory installations. The physical size of the source is comparable to the one for ²⁴¹Am/Be. The neutron yield per GBq or per μg is comparable to that of ²⁴¹Am/Be, but very strong sources can be made especially for laboratory (activation) use. The energy

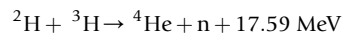
²Absorbed dose is measured in Gray (Gy) where $1 \text{ Gy} \equiv 1 \text{ J/kg}$. The effect of this dose in humans (equivalent dose) is measured in Sievert (Sv) where $1 \text{ Sv} \equiv 1 \text{ Gy} \cdot \text{WR}$ where WR = radiation weighting factor which expresses the harmfulness of the actual radiation type (different for β , α , γ , neutrons, etc.).

spread is narrower towards lower energies than for the $^{241}\text{Am}/\text{Be}$ source (Fig. 7), and this reduces the needed thickness for a moderator to produce epithermal or thermal neutrons.

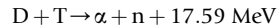
Sealed-tube neutron generators

Neutrons can also be produced by fusion of light hydrogen nuclei like deuterium and tritium. This may sound complicated and perhaps require large laboratory installations, but modern development has made it straight forward. Comprehensive overview texts are found in IAEA (2012), Valkovic (2015) and Chichester (2009). The following text concentrates around the use of so-called sealed tube neutron generators. These are small accelerators enclosed in sealed glass tubes. Depending on the generator design, one or more of three different fusion reactions $\text{D} + \text{T}$, $\text{D} + \text{D}$ or $\text{T} + \text{T}$ may take place. The working principle is illustrated in Fig. 8 and exemplified with a $\text{D} + \text{T}$ reaction. The tube contains a reservoir of deuterium bound as hydride in a thin and finely powdered metal layer (for instance in titanium as TiD_x where $x_{\text{max}} = 2$) on a molybdenum backing. Upon heating, some of the deuterium is released to a few (2–4) millitorr and ionized in an ion source, mainly to produce D_2^+ . In some generators, and depending on the type of ion source, D^+ is the dominating species. One type of ion source used is the Penning type (electron bombardment), another source is based on radiofrequency ion generation (RF-source).

A negative potential (typically 150 kV) accelerates the positively charged ions against a target containing tritium, for instance as TiT_x ($x_{\text{max}} = 2$). A nuclear fusion reaction takes place between deuterium and tritium resulting in production of a neutron according to the reaction



or



The 17.59 MeV is the total reaction energy divided as kinetic energy between the neutron and the α -particle.

The neutrons are nearly monoenergetic with energies in the range 14–15 MeV (Figs. 9 and 10).

With a deuterium-loaded target the reaction becomes



These neutrons are also close to monoenergetic with energy of 2.45 MeV (Fig. 9).

For thin targets the neutron energy also varies with the angles in which neutrons are emitted relative to the incoming deuterium particle and with the applied accelerating voltage. This is illustrated in Fig. 11. For the D–D reaction the neutron energy is approximately 2.45 MeV at 90 degree angle, while for the D–T reaction the neutron energy is around 14.1 MeV for the same angle.

The D–T reaction has the largest maximum cross-section of 5.0 b ($5 \cdot 10^{-24} \text{ cm}^2$) of all fusion reactions. Maximum cross-section of this reaction is reached at an acceleration potential of only 64 keV. At 100 keV acceleration voltage the cross sections are 3.43 b and 0.033 b for the D–T and D–D reactions, respectively. The cross section varies as a function of acceleration voltage (Fig. 12).

Neutrons produced from the D–T reaction are emitted isotropically from the target in the mass middle point system. Neutron emission from the D–D reaction is slightly peaked in the forward (along the axis of the ion beam) direction. In both cases, the He nucleus (^4He and ^3He , respectively) is emitted in the exact opposite direction of the neutron. This is a very interesting fact as it can be

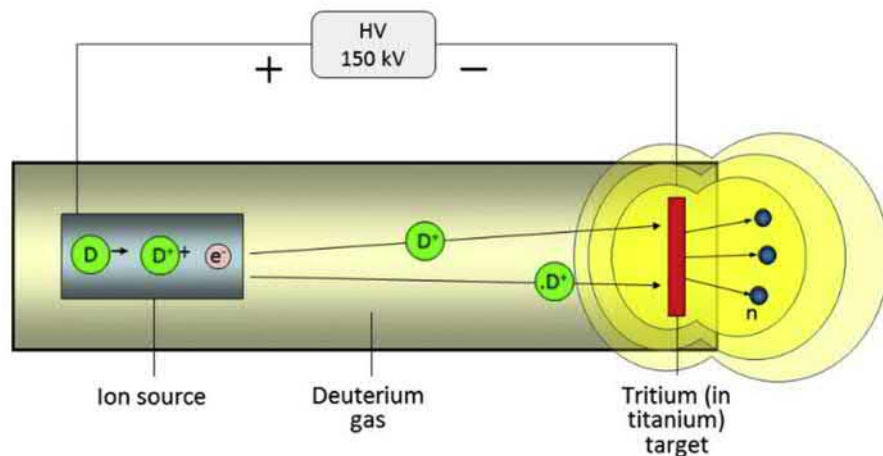


Fig. 8 Principle sketch of a sealed-tube neutron generator. Here, deuterium ions are accelerated against a target consisting of tritium bound as hydride in a metal (here TiT_x), and the resulting neutron flux distribution.

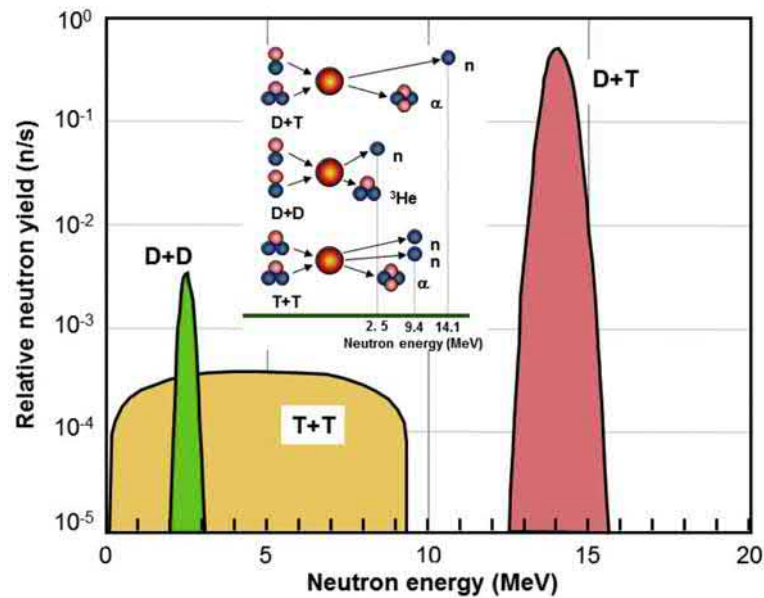


Fig. 9 Approximate energy distribution for the DT, DD and TT reactions that take place in a sealed-tube neutron generator.

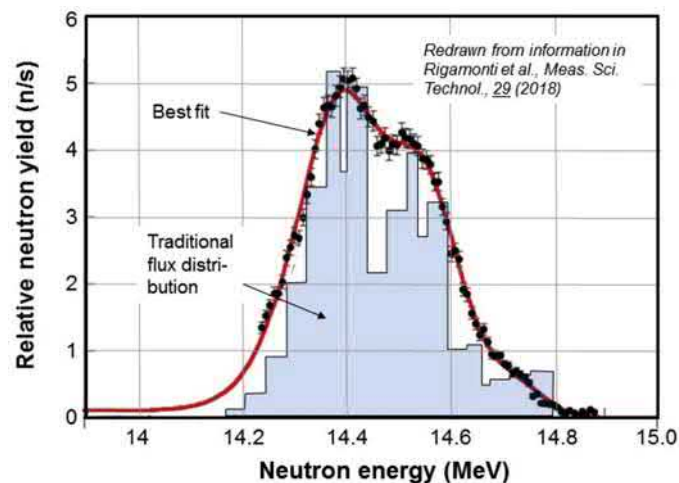
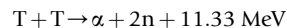


Fig. 10 Measured high-resolution energy distribution for produced neutrons from the DT-reaction. Redrawn from information in Rigamonti D, et al. (2018) *Measurement Science and Technology* 29.

utilized to perform space-resolved (tomographic) pictures of the neutron interaction with extended external objects (see Bjørnstad, 2021b).

In modern neutron generators the gas that is ionized and accelerated is a mixture of deuterium and tritium. The cross section for the T-T fusion reaction



is much lower than that for the D-T reaction, but similar in size to the D-D reaction (only 3.4×10^{-2} b at an acceleration voltage of 100 keV). The main purpose of the tritium gas is, therefore, to replenish used tritium in the target due to reactions, evaporation and sputtering. Because of this, some T-T reactions take place. Since two neutrons are emitted per reaction, the energy spectrum is not “monoenergetic” but shows a broader energy distribution as shown in Fig. 9, since the reaction energy now must be shared among three particles.

Sealed-tube neutron generators based on reactions described above come in different sizes, shapes and with different neutron output (n/s) from the smallest of ~ 200 n/s (“neutrons on a chip”) (Elizondo-Decanini et al., 2012) to more than 10^{13} n/s (Phoenix, 2020). They are all relatively small instrumentation. The footprint of the largest generators is 5–15 m² exclusive external shielding. These are mainly meant for permanent installation in a laboratory. However, the latest years have seen a vivid development of small-size generators that are transportable and even portable with neutron outputs in the range 10^6 – 10^9 n/s. In addition, those

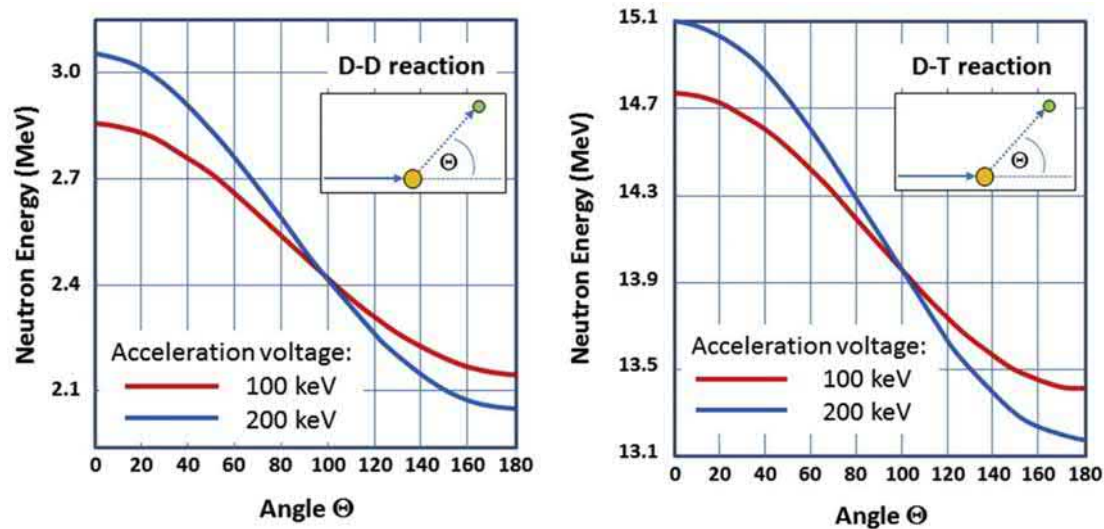


Fig. 11 Neutron energy as a function of the angle between the bombarding deuterium and the emitted neutron for D-D and D-T neutrons with the accelerating voltage as a parameter. Data extracted and reproduced from IAEA (2012) Neutron generators for analytical purposes. In: *Radiation Technology Reports No. 1*. Vienna: International Atomic Energy Agency, 145 pp. Available from <https://www.iaea.org/publications/8505/neutron-generators-for-analytical-purposes>.

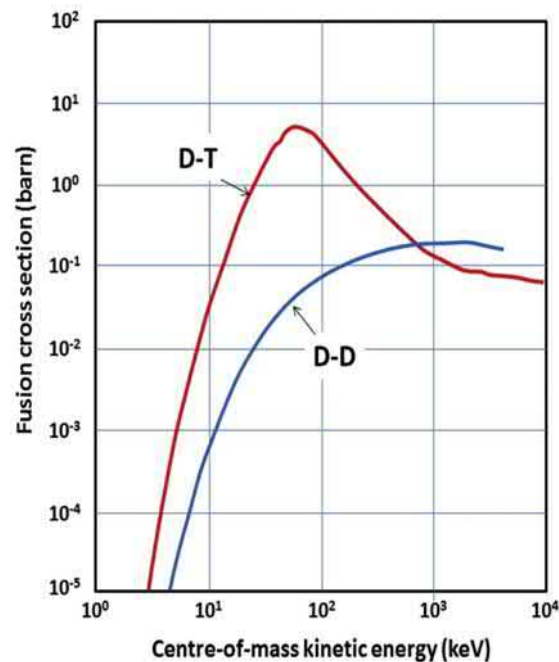


Fig. 12 Cross sections for the neutron-producing D-D and D-T nuclear fusion reactions as a function of the center of mass kinetic energy (acceleration voltage of the deuteron). Data extracted and reproduced from Atzeni S and Meyer-Ter-Vehn J (2009) Nuclear fusion reactions. In: *The Physics of Inertial Fusion*, pp. 1–30. Available from <http://www.fisicanucleare.it/documents/0-19-856264-0.pdf>.

which are based on the D-D reaction do not contain any radioactive material (in contrast to the D-T generator). This makes handling, logistics and transport easier, even across country borders. Thus, they can be implemented directly in the industry and civil engineering for resolving various operational challenges. A few examples are shown later. A typical portable generator is shown in Fig. 13. The weight of this one is only 15 kg.

Neutron generation based on small-to-medium size accelerators

Apart from sealed-tube neutron generators, three other types of accelerator-based neutron-producing devices are the electrostatic accelerators, the cyclotrons, and the linear accelerators (García, 2021b).



Fig. 13 Example of a portable sealed-tube neutron generator for use in industrial and civil service settings. Courtesy of Thermo Fisher.

Radiofrequency quadrupole (RFQ) linear accelerators (Schempp, 2000) or drift-tube Linacs (Wrangler, 2008) or a combination of the two are potentially a very good solution to the delivery of large proton (or deuteron) beam currents at energies of a few MeV. At present, neutron generating equipment based on the compact linear accelerator design can produce from 10^8 to more than 10^{13} n/s by bombardment of beryllium targets (Fig. 14). R&D and engineering work is now going on to develop linear accelerators with a proton current in the range 10–100 mA. Proton energies of 30 MeV can be obtained over an acceleration distance of 25–30 m.

One of the ongoing initiatives is the development of the SONATE project at CEA in France (Ott et al., 2020) with protons on a beryllium target. What is the achievable neutron yield? Although the melting point of beryllium is relatively high (1287 °C), the limit for power-dumping of a proton beam into a stationary target is probably ~ 100 kW. This means that there is a trade-off between the beam current (mA), the proton energy (MeV), and the duty cycle (%) of the pulsed beam. For example, for a proton energy of $E_p = 20$ MeV, a peak current of 100 mA and a 4% duty cycle, the power on the target would be 80 kW and the neutron yield would be 3.1×10^{14} n/s. If the proton energy is increased by a factor 2 to 40 MeV, while the power on the target remains limited at 80 kW, the duty cycle must be reduced to 2% and the neutron yield is 5.4×10^{14} n/s. For a given power deposited on the target, the neutron yield is roughly proportional to the proton energy. This neutron output is a factor of more than 10–50 compared to the highest yields achievable from state-of-the-art sealed-tube neutron generators.

Another modest-size linear accelerator, LIPAC (Cara et al., 2016), is at present being developed in a European-Japanese collaboration. It accelerates deuterons to 9 MeV with a peak current of 125 mA. This accelerator is mainly being developed for fusion research, but, obviously, it has potential as a general-purpose high-intensity neutron source as well.

Such neutron-generating equipment exhibits long lifetimes at full output, as there is little target or beam degradation. Since they do not use radioactive materials, licensing requirements are less stringent than for isotopic sources or tritium sealed-tube generators. These pulsed systems based on the RFQ accelerator are well suited for applications requiring a high peak neutron flux, including activation analysis of very short-lived reaction products and various forms of laboratory-based neutron imaging and scattering experiments.

Cyclotrons (Fig. 15) can also be used as a source of protons to generate neutrons from a target of lithium or beryllium. However, the obtainable proton flux for a state-of-the-art small and medium-size cyclotron with maximum energies in the range 18–30 MeV is considerably less than for the Linacs mentioned above. A small hospital-type cyclotron (for PET radionuclide production) with

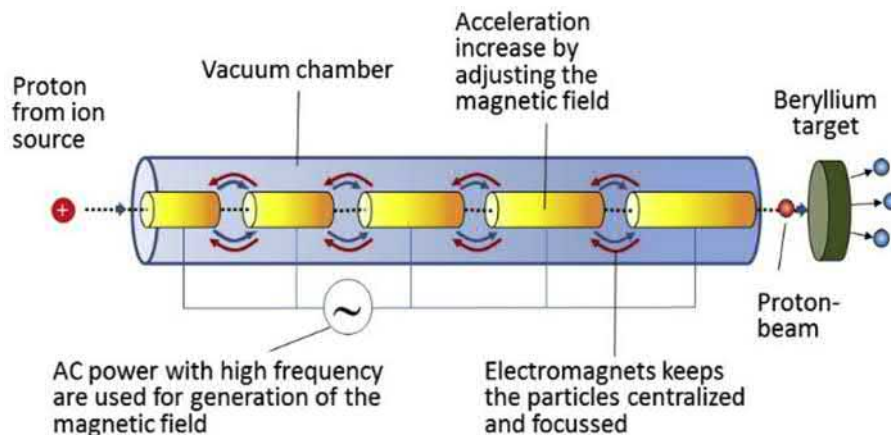


Fig. 14 Principle sketch of radiofrequency-based linear proton accelerator producing neutrons from a beryllium target.

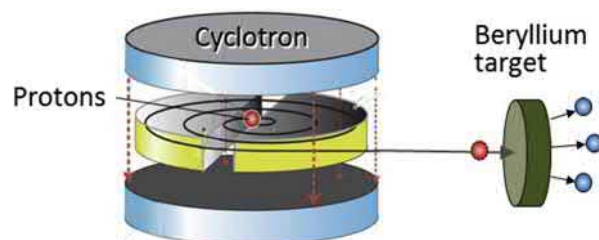


Fig. 15 Principle sketch of a small-to-medium size cyclotron accelerating protons against a neutron-producing target of beryllium.

$E_{\text{max}} = 18 \text{ MeV}$ provides a maximum proton output of $300 \mu\text{A}$ (IBA Cyclone[®] KIUBE³). A larger version, which provides an energy in the range 15–30 MeV, offers a maximum proton current of 1.5 mA (Cyclone[®] 30⁴).

Reactor neutrons from fission

Most research reactors today are conventionally based on fission of ^{235}U with isotopic enrichment in the range 4–20% and light or heavy water as the moderator and cooling medium. Obtainable neutron flux in the core of research reactors is in the range 10^{12} to $> 10^{15} \text{ n/cm}^2 \cdot \text{s}$ (Fig. 16). Research reactors provide many options for exposing target materials to high neutron flux and for extracting neutron beams for a variety of material property studies (O’Kelly, 2021).

Neutrons from bremsstrahlung reactions

This neutron source is based on the interaction of high-energy photons with the atomic nucleus. Then, where do the high-energy photons come from?

Photons are produced when high-energy charged particles, for instance electrons, imping on and are decelerated in material. The high-energy electrons are produced in electron accelerators. Two mechanisms are active. The first is that an incoming electron interacts with an orbital electron around a nucleus, kicks the orbital electron out of the atom leaving an electron “hole,” which subsequently is filled with electron transition from an outer electronic shell. This process is illustrated in Fig. 17 showing the removal of an electron from the K-shell. The resulting hole in the K-shell is filled by electron transition from the L-shell. The hole in the L-shell is

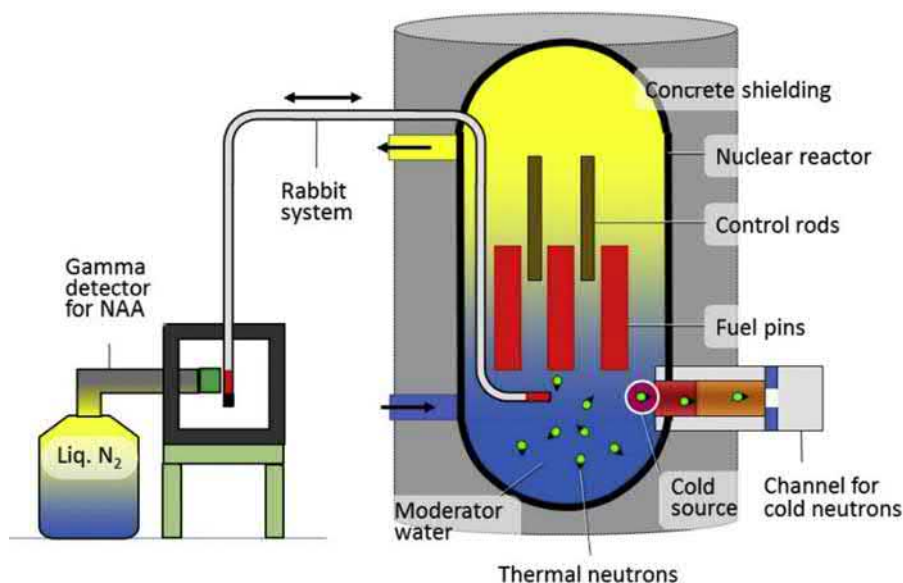


Fig. 16 Principle sketch of a nuclear reactor with fuel elements, control rods, circulating cooling water and moderator, exit channel for cold neutrons to scattering and diffraction measurements and with a rabbit system for activation of material samples for neutron activation analysis (NAA).

³https://www.iba-radiopharmasolutions.com/sites/default/files/resources/files/cbr_cyclone_kiube_v2_r05.pdf

⁴<https://www.iba-radiopharmasolutions.com/products/cyclotrons#cyclone-30>

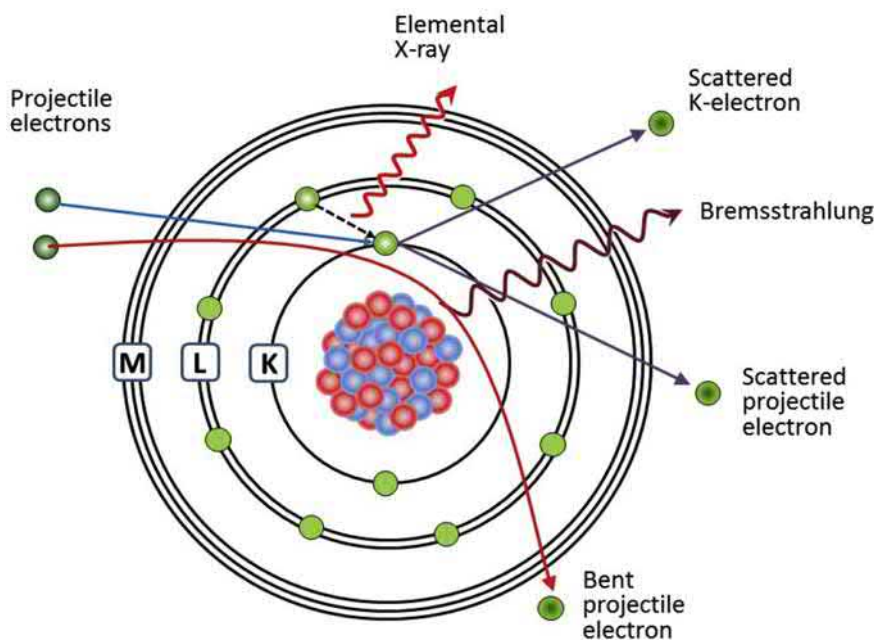


Fig. 17 Production of elementary X-rays and bremsstrahlung when high-energy electrons interact with orbital electrons in an atom or with the electromagnetic field of the nucleus.

subsequently filled from the M-shell and so on. Roughly speaking, one can say that the energy difference between the L- and the K-shells is emitted in the form of an element-characteristic X-ray, in this case a K-X-ray. Energy of elemental X-rays are generally below 100 keV (uranium and thorium has slightly higher energies), and these energies are too low for neutron production.

The other mechanism is that the incoming electron interacts with the electromagnetic field of the nucleus, resulting in deceleration of the speed and bending of its trajectory. The difference in electron energy before and after the bending is emitted as photons generally known as *breaking radiation* or *bremstrahlung*. This process gives rise to a continuous photon energy spectrum from its energy maximum close to the energy of the incoming electron down to very low energies. The intensity of the photons (photon flux or luminosity) increases towards lower energies. Increasing energy of the incoming electron results in shifting the photon energy spectrum towards higher energies. Thus, photon energies can be high enough to overstep the neutron binding energy in nuclei.

The high-energy photons impinge on a target typically of a high-Z element like tantalum or tungsten and interact with the nucleus of the elements. These photonuclear interactions, also known as the nuclear photo effect, are interactions where a photon disturbs the internal co-ordination of the nucleus. When the photons have energies above the typical binding energy range of the nuclei ($> 5\text{--}10$ MeV), these interactions generally lead to the emission of nucleons—both neutrons and protons. However, for heavy element targets like tantalum and tungsten, the emission of protons is suppressed due to the Coulomb barrier. The nucleon emission is mainly due to the four processes of giant nuclear dipole resonance, quasi-deuteron production, delta resonance, and vector meson dominance (Pisharody et al., 1998). Only the first process is of importance for photon energies in the 10–30 MeV range, and the various mechanisms will not be further discussed here.

When the same high-Z material is used both in the photon production and in the neutron generation, the whole process can take place simultaneously within the same physical target. For the higher photon energies, photo-fission (Methasiri and Johansson, 1971) may also be a component in this process leading to emission of fission neutrons as well (Fig. 18). Resulting neutron energies depend on the photon energy distribution and the neutron separation energy of the target element. Generally, the energy spectrum stretches from a few MeV down to low energies.

Neutrons from spallation reactions

At proton energies of 100 MeV and slightly higher, spallation reactions begin to take place, and at 1 GeV and higher, spallation is the dominating nuclear reaction process (Fig. 19). The process starts with a high-energy incoming proton hitting a target atom of a heavy element like lead, tantalum, tungsten, mercury or even thorium or uranium. The high-energy proton transfers kinetic energy to protons and neutrons in the nucleus. The result is that some of these sub-atomic particles are promptly knocked out of the nucleus leaving the rest-nucleus (in Fig. 19 called the excited nucleus) in an excited state. This part of the spallation process is called the intra-nuclear cascade. The rest-nucleus disposes off its remaining excitation energy by different mechanisms. One mechanism is the further evaporation of mainly neutrons and cascades of gamma rays until the excitation energy has reached a level below the neutron separation energy. Another mechanism is that the excited nucleus has sufficiently high energy to fission, leading to emission of fission neutrons and subsequently of delayed neutrons during beta-decay of the radioactive fission products. A third

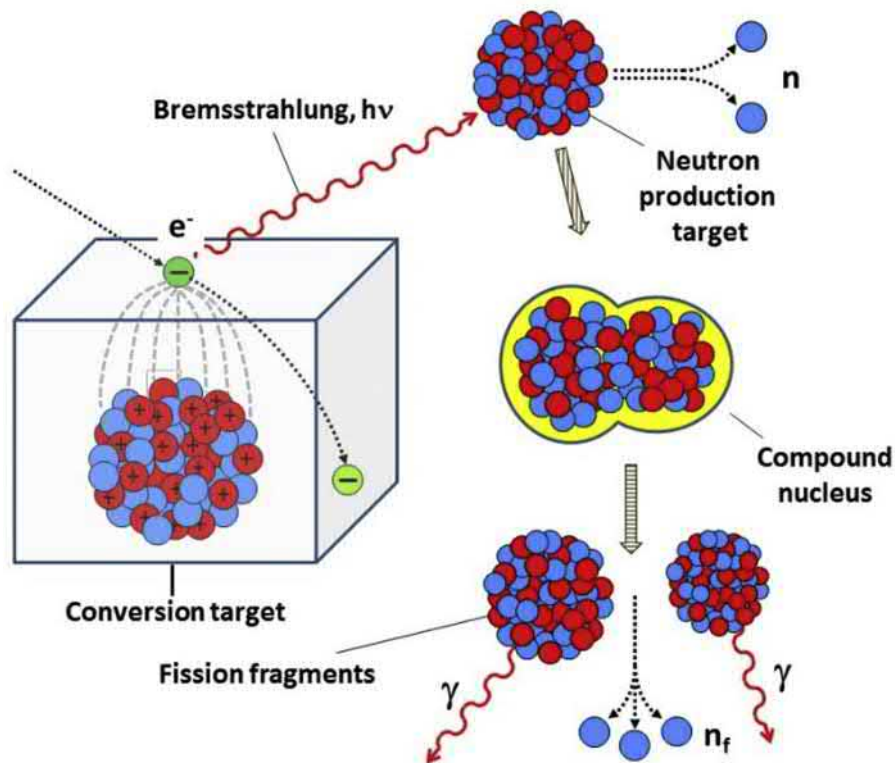


Fig. 18 High-energy electron induced generation of high-energy photons and the photon interaction with high-Z nuclei to produce neutrons directly and through an associated photo-fission process.

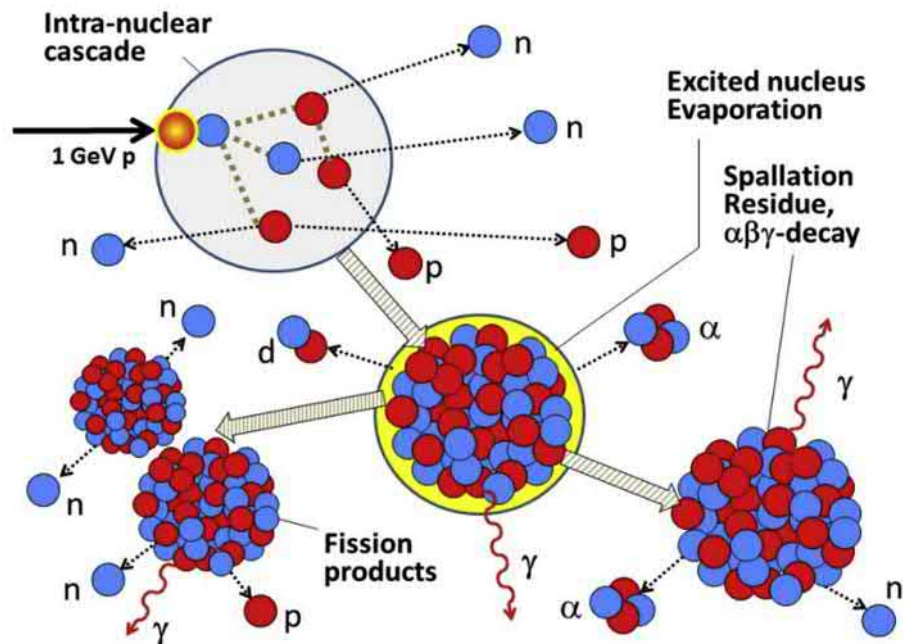


Fig. 19 Principle sketch of the spallation process leading to production of a shower of neutrons over a broad energy range.

mechanism is alpha, beta and gamma emission, leading to a nucleus (in Fig. 19 called the spallation residue), which has an atomic number (Z) a few units below the target nucleus. The new Z-value depends on the number of protons knocked out in the intra-nuclear cascade and on eventual alpha particle emissions until the spallation residue nucleus is reached. Thus, a number of neutrons over a broad energy spectrum are produced in the spallation process.

Facilities based on spallation sources provide the most intense pulsed neutron beams for scientific research and industrial development, and several facilities have been built through the years. One of the largest is the SNS⁵ at Oak Ridge in United States. A new European initiative is the ESS⁶ presently being built in Lund in Sweden where user programs are scheduled to begin in 2023.

Neutron detectors

There are three main categories of general-purpose neutron detectors (Han, 2021):

- *Gas proportional counters* use a gas to amplify the charge from the original charged particles generated by a neutron absorption reaction in the conversion material.
- *Scintillation detectors* use solid or liquid scintillating materials that emit light when struck by an incoming neutron.
- *Semiconductor detectors* consist of semiconductor chips where the charged particles emitted in the neutron absorption reaction deposit energy creating an electric signal.

In addition:

- *Microchannel plates (MCP)* has been in use in past years for the detection of, for example, X-rays and electrons. Development of this principle for position sensitive neutron detection in scattering and tomography experiments has taken place over the last 20 years.

Each of these principles, which are essential in any neutron transmission and scattering applications, are briefly described below. Additional material on neutron detectors can be found in Han (2020).

Gas proportional neutron counters

Gas proportional counters have been used for many years for the detection of nuclear radiation. Proportional counters operate at relatively high voltages of up to about 2000 V and with proper filling gases they can multiply the original ionization without introducing nonlinearities to obtain reasonable pulse sizes of several volts. When they include a filling gas of the correct composition, they can be used for neutron detection (Fig. 20).

Traditionally, these proportional counters were filled with BF₃, which provides good detection efficiency for thermal neutrons. These detectors exploit the high reaction cross section for ¹⁰B (3840 b) for thermal neutrons in the reaction:

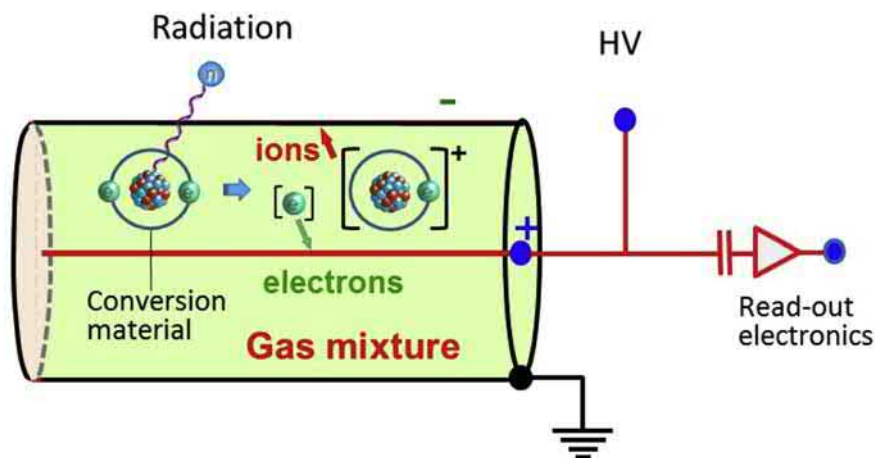
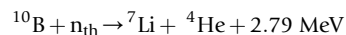


Fig. 20 Principle sketch of a gas proportional counter, which also can be used for detection of thermal neutrons if the filling gas contains BF₃ or ³He.

⁵<https://neutrons.ornl.gov/sns>

⁶<https://europeanspallationsource.se>

The discovery that ^3He provided better overall neutron detection capability, also for neutrons with higher energies, resulted in the almost unanimous use of this as the filling gas for proportional counters in neutron detection. Proportional counters with other filling gases can also be used by depositing solid forms of B or Li inside the proportional counters (Knoll, 2010).

When ^3He is the active material in the filling gas, the following nuclear reaction takes place (cross section for thermal neutrons is 5330 b):



The produced proton creates an electrical signal by ionizing the filling gas molecules.

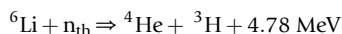
In the later years, the commercial availability of ^3He has been decreasing due to the huge demand for neutron detectors in border control and similar applications. One may say that ^3He has become a “strategic material” with limited free trade. This commercial shortage of ^3He , paired with high increase in prices, has caused a return to the use of BF_3 gas or the use of scintillation detectors for the detection of neutrons.

Solid scintillation neutron detectors

Scintillation neutron detectors include liquid organic scintillators, solid crystals, plastics, glass and scintillation fibers.

Neutron-sensitive scintillating glass fiber detectors

The scintillating glass fibers work by incorporating ^6Li and Ce into a glass bulk composition. The doped glass is extruded into fibers. The ^6Li has a high cross-section for thermal neutron absorption (940 b) through the reaction:



The alpha particle and triton interact with the glass matrix to produce ionization, which transfers energy to and excite the Ce-atoms. Upon deexcitation, a portion of the emitted scintillation light propagates through the glass fiber, which also acts as a light-guide (Fig. 21). The ends of the fiber are optically coupled to a photomultiplier tubes (PMTs) to detect photon bursts. The event results in a flash of light of several thousand photons for each neutron absorbed.

The scintillating fiber detectors have excellent sensitivity, they are rugged, and have fast timing ($\sim 60 \text{ ns}$) so that a large dynamic range in counting rates is possible. The detectors have the advantage that they can be formed into any desired shape and can be made very large or very small for use in a variety of applications. Further, they do not rely on any raw material that has limited availability, nor do they contain toxic or regulated materials. Their performance matches or exceeds that of ^3He -tubes for gross neutron counting due to the higher density of neutron absorbing species in the solid glass compared to high-pressure gaseous ^3He . Even though the thermal neutron cross section of ^6Li is lower compared to ^3He (940 b vs. 5330 b), the atom density of ^6Li in the fiber is 50 times greater. This results in an advantage in effective neutron capture probability ratio of approximately 10:1.

Crystalline neutron scintillation detector

LiCaAlF_6 is a neutron sensitive inorganic scintillator crystal, which like neutron-sensitive scintillating glass fiber detectors makes use of neutron capture by ^6Li . Unlike scintillating glass fiber detectors, however, the ^6Li is part of the crystalline structure of the scintillator, giving it a naturally high ^6Li density. A doping agent is added to provide the crystal with its scintillating properties. Two common doping agents are cerium (Ce) and europium (Eu). Europium doped LiCaAlF_6 has the advantage over other materials in that the number of optical photons produced per neutron capture is around 30,000–40,000, depending on the level of

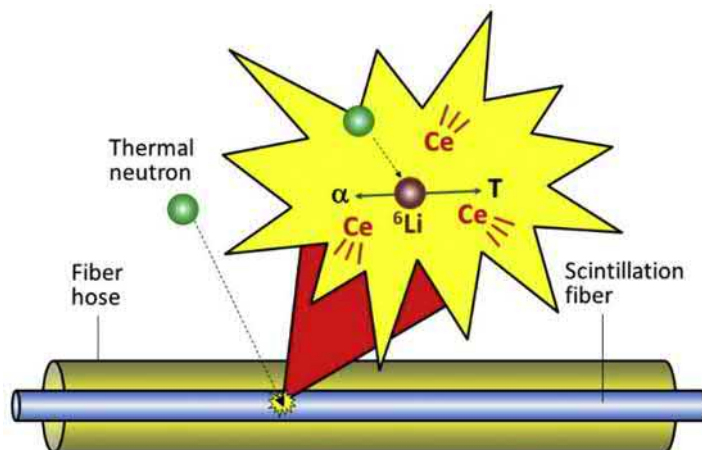


Fig. 21 Working principle of a scintillation fiber detector for thermal neutrons where the fiber has been doped with ^6Li .

Eu-dopant (0.5–2 mol%). This is more than five times higher than for example in neutron-sensitive scintillating glass fibers (~ 6000 photons/n).

Due to its high ${}^6\text{Li}$ density, this material is suitable for producing light weight compact neutron detectors. The long decay time of Eu-doped LiCaAlF_6 ($\sim 1.6 \mu\text{s}$) makes it less suitable for measurements in high radiation environments. The cerium-doped variant has a shorter decay time ($\sim 40 \text{ ns}$) but suffers from a somewhat lower light-yield (~ 4000 – 5000 photons/n).

Semiconductor neutron detectors

Solid state semiconductors represent one of the basic important detection principles for neutron measurement. When compared with gas-filled detectors, they have the potential to be more compact, operate at lower voltages, and be more robust against vibration-induced noise.

A typical planar thermal neutron conversion layer detector consists of a thin film of neutron reactive material deposited on the surface of a semiconductor diode. ${}^{10}\text{B}$ and ${}^6\text{LiF}$ are frequently used as conversion layer materials because of their stability, large thermal neutron absorption cross-sections, and primary reaction products.

The functional principle is sketched in Fig. 22 for the case of ${}^6\text{Li}$. In the case of ${}^{10}\text{B}$, the most common primary reaction products generated by neutron absorption are alpha particles (1.47 MeV) and ${}^7\text{Li}$ ions (840 keV). When a nuclear reaction occurs in the ${}^{10}\text{B}$ conversion layer, these charged particles will travel into the semiconductor material where they can create electron-hole (e-h) pairs.

A critical design parameter for planar conversion layer devices, which is ultimately responsible for limiting the maximum thermal neutron detection efficiency, is the thickness of the conversion layer material. The probability that a neutron will be absorbed in the conversion layer increases with increasing layer thickness. For ${}^{10}\text{B}$, $\sim 90\%$ of incident thermal neutron flux will be absorbed in a $\sim 50 \mu\text{m}$ thick layer, for ${}^6\text{Li}$ somewhat less. If the conversion layer is too thick, the reaction products may be absorbed before reaching the semiconductor. The average travel range in ${}^6\text{LiF}$ for the 2.05 MeV alpha particle from ${}^6\text{Li}$ -reactions is $6.3 \mu\text{m}$, while the range for the corresponding triton (T) is $34.7 \mu\text{m}$. Average travel ranges for the 1.47 MeV alpha particle and 840 keV ${}^7\text{Li}$ ion in ${}^{10}\text{B}$ are 3.9 and $1.7 \mu\text{m}$, respectively. Conversion layer thicknesses for planar conversion layer devices are limited by the average travel ranges of the reaction products. Detector efficiency may be improved by a detector configuration like the one in Fig. 23.

This design has been improved several times (McGregor et al., 2010; Engelmann and Martin, 2011; Bellinger et al., 2014), and development is continuously on-going to achieve higher energy resolution and sensitivity than other room temperature detectors currently available. The new detectors integrate one or more relatively thin films incorporating a neutron-sensitive element, such as gadolinium, boron, or lithium, with semiconductor detector materials. The configuration is a stacked multilayer diode design that allows for use of a wide variety of materials. The electrodes can be made of conventional metallic materials such as copper or chromium, and the semiconductor detector layers can be made of silicon, germanium, or gallium arsenide. For all possible forms, the design is easily adapted to well-established semiconductor manufacturing techniques.

Microchannel plate detectors (MCPs)

Microchannel plates are currently being applied for high-resolution neutron counting with spatial resolution as small as $\sim 20 \mu\text{m}$ and temporal resolution in the order of $\sim 1 \mu\text{s}$ (Tremis et al., 2008).

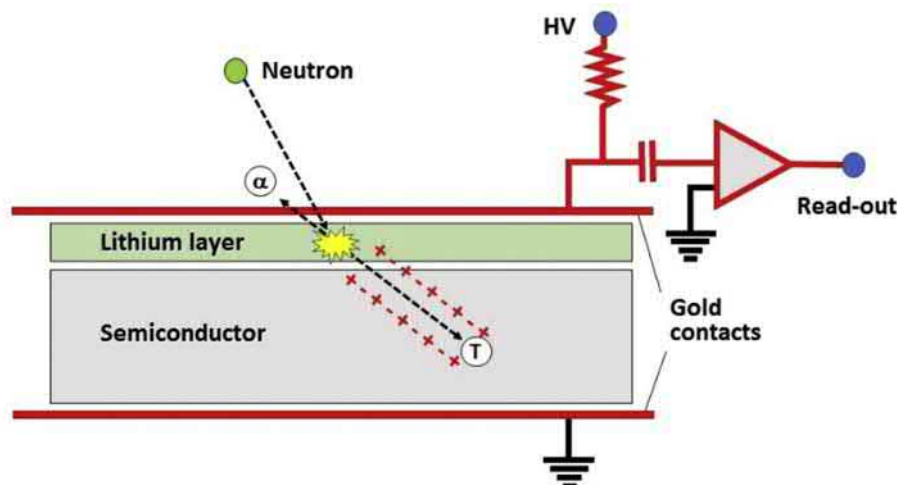


Fig. 22 Sketch of the working principle of a semiconductor detector for thermal neutrons where a thin layer doped with ${}^6\text{Li}$ reacts with the neutron to produce an α -particle and tritium, both ionizing a semiconductor material which, for instance, may be silicon (Si). A current pulse is produced and transmitted to read-out electronics.

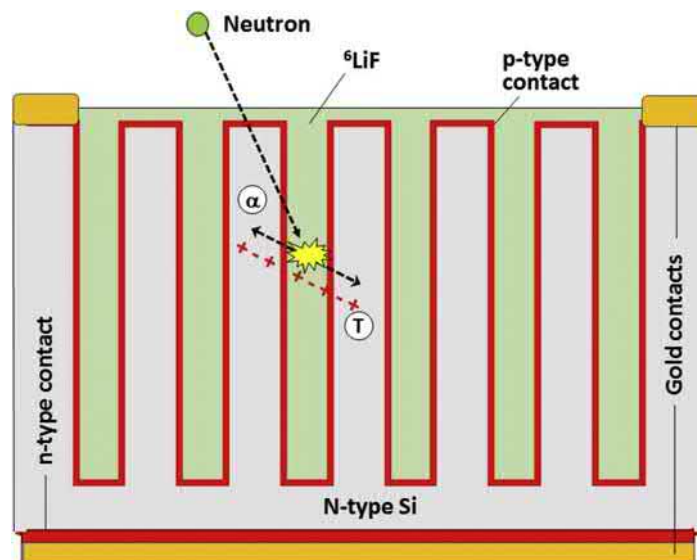


Fig. 23 Principle of a semiconductor detector with a multilayer structure for better detector efficiency. Layers are not to scale since the layer of LiF is only a few μm thick while that of silicon can be 40–50 μm .

MCPs upon inspection look like a thin glass disk perforated with microscopic holes. They typically consist of several million microchannels or capillaries fused together into a monolithic disk-like structure (a section of which is shown in Fig. 24). Each capillary is an independent, microscopic channel electron multiplier (Fig. 24).

The glass is doped with a neutron-absorbing material, typically either ${}^6\text{Li}$ or ${}^{10}\text{B}$. Upon reaction with an incoming thermal or epithermal neutron, the emitted alpha and triton particles penetrate the thin glass walls and enter into the evacuated capillaries where they hit the capillary wall (mostly two times). Electrons are kicked out from an electron-donating material on the wall surface of the capillaries. These electrons are accelerated in a potential put across the microchannel plate, hitting the wall several times—kicking out new electrons etc. An avalanche of electrons exits the channels with a current sufficiently large to be registered by backing electron-detecting devices. MCPs are able to amplify extremely weak signals by 10^3 – 10^7 , yielding output signals that can be easily handled by auxiliary electronic equipment.

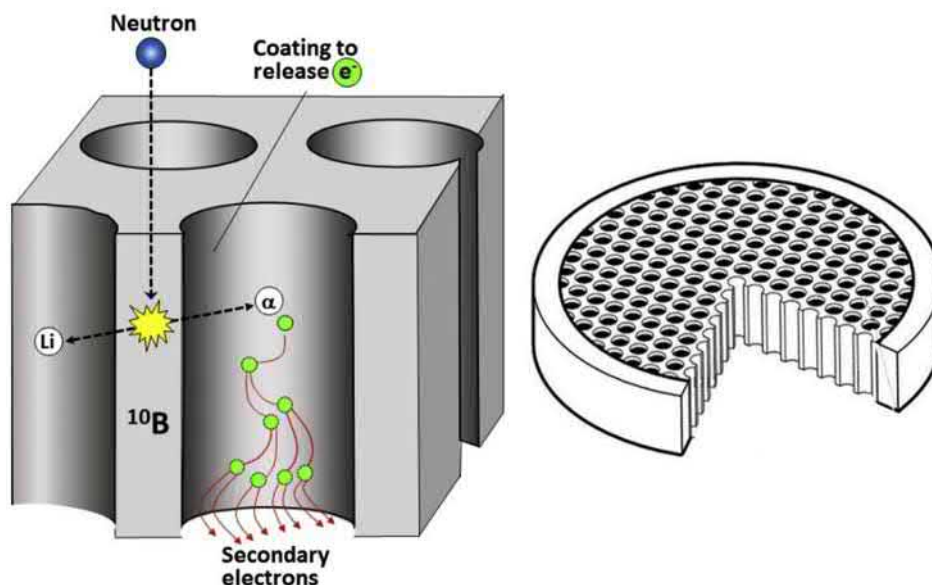


Fig. 24 Expanded view and cut of a microchannel plate detector where the thin glass matrix is doped with the neutron-absorbing nucleus ${}^{10}\text{B}$. Reaction products recoiling out of the glass into the evacuated microchannels induce an avalanche of electrons that are accelerated in an electric potential put across the plate and electrons exit the structure in the bottom. Here, the electrons are registered by auxiliary electronic equipment.

The microchannels are usually of a circular or square geometry with 6–12.5 μm wide pores with 7.2–15 μm center-to-center spacing. The thickness of the disk is typically 40–250 times larger than the pore diameter (the L/D or aspect ratio), resulting in typical wafer thicknesses of only 0.4–1 mm. MCPs can be fabricated with an active area $\geq 10 \times 10 \text{ cm}^2$.

Although the glass mixture can contain only a limited proportion of neutron-absorbing atoms, this deficiency is compensated in MCP collimators by the possibility to produce structures with very large aspect ratios. The ratio of pore length to its width LMCP/d for current technology can be as high as 250:1.

These detectors have several areas of use, but especially within the neutron scattering, radiography and tomography examinations since they provide position-sensitive properties with spatial resolution in the 20–50 μm range.

See Also: Phenomenology of Neutron Interactions.

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Modern Industry: Application of Neutrons for Materials Testing and Inspection

Tor Bjørnstad, Institute for Energy Technology, Kjeller, Norway; and University of Oslo, Oslo, Norway

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Glossary

CFD Computational fluid dynamics,—a branch of fluid mechanics that makes use of different mathematical algorithms and numerical analysis in order to analyze and solve problems involving fluid flows. It is based on the Navier-Stokes equations that describe how the velocity, pressure, temperature, and density of a moving fluid are related. Practical applications include analysis of air flow for an aerodynamic aircraft design, analysis of the hydrodynamic properties of a boat hull, industrial design of oil and water piping, and processing equipment like multiphase separators, and many more.

CNL Compensated neutron log,—a well logging tool for the determination of matrix porosity. The tool contains a strong neutron source and two detectors situated 15 in. and 25 in. from the source. The tool is designed to be sensitive to thermal neutrons and is, therefore, negatively affected by any chlorine content. The critical measurement for this tool is the *difference* in the thermal neutron population as registered by the two spaced detectors, which results from neutron capture and neutron scattering.

Cross section Each isotope of the various chemical elements has its unique probabilities for defined types of nuclear interactions like scattering, absorption, or transfer reactions. We call this probability the (microscopic) cross section. The unit is *barn* (b), and it has the dimension of area ($1 \text{ b} = 10^{-28} \text{ m}^2$ or 10^{-24} cm^2 , which is more frequently used in formulas). The symbol used is the Greek letter σ . The cross sections for each type of nucleus depend on the neutron energy. The energy dependence curve is called the *excitation function*. See also (Bjørnstad, 2021b).

GNT Gamma Ray/Neutron Tool,—a well logging tool for the determination of matrix porosity. It contains a neutron source and a single detector that is sensitive to high energy capture gamma rays and thermal neutrons. The measurement principle is that rock pores filled with hydrogen-containing liquids (water or oil) will thermalize and capture the neutrons. The degree of thermalization is a function of H-content and, thereby, the porosity. In addition, it detects high-energy capture gamma rays from thermal neutron capture in H, O, and C. Unfortunately, it is sensitive to the concentration of chlorine in the formation brine and other well fluids because chlorine has a relatively high cross section for capture of thermal neutrons, thereby overestimating the porosity.

Imbibition The process of absorbing a wetting phase, in this case water, into a porous rock is termed imbibition. Imbibition is important in petroleum production where reservoirs are under water injection (secondary oil recovery) and the reservoir rock is water-wetting. Spontaneous imbibition requires no external force: water intrudes and oil is expelled out of the rock pores.

NAA Neutron activation analysis,—a sensitive analytical technique for qualitative and quantitative analysis of chemical elements in nearly any type of matrix. It is based on irradiation of the sample with neutrons (various energies), thereby generating radioactive nuclides in the sample, followed by detection of the emitted radiation that is element (and isotope) specific like a fingerprint.

NCT Neutron-induced computed tomography,—a neutron imaging technique for revealing structure and elemental contents of extended objects by rotating the sample in the beam of neutrons and recording multiple 2D images through an angular range of 180 degrees with position-sensitive neutron detectors. From the data set, a 3D representation through the object (tomogram) can be constructed.

Neutron imaging The principle of neutron imaging is based on the attenuation, through both scattering and absorption, of a directional neutron beam by an object through which it passes. Since different materials vary in their ability to attenuate neutrons, both composition and structure can then be probed. Both NR and NCT are neutron imaging techniques.

NORM Naturally occurring radioactive material,—material found in the environment that contains long-lived radioactive elements of natural origin. Examples are uranium (mainly ^{235}U and ^{238}U), thorium (^{232}Th) and potassium (^{40}K) and any of their decay products such as radium (mainly ^{226}Ra and ^{228}Ra), radon (mainly ^{222}Rn) and lead (^{210}Pb). The material may exist in “natural concentrations” like in tailings from uranium or thorium mining or as a product of human or industrial activity where the concentrations have been physically or chemically enhanced like in radioactive scale from the petroleum production industry.

NR Neutron radiography,—a neutron imaging technique for revealing structure and elemental contents of extended objects. It involves placing an object in the path of the neutron beam and measuring the shadow image (projection) of the object with a position-sensitive neutron detector.

Penetration depth In the present text, the term *penetration depth* is taken to mean how deeply in mm a beam of thermal neutrons (here at 27 °C) penetrates a particular chemical element in its solid or liquid form before the intensity of the beam has been reduced by a factor $1/e$, i.e., to about 37% of its original intensity.

PGNAA Prompt gamma neutron activation analysis,—an analytical technique for qualitative and quantitative non-destructive analysis of chemical elements in nearly any type of matrix. It is based on irradiation of the sample with neutrons (various energies are possible), inducing nuclear reactions like absorption or scattering, thereby exciting the nuclei to higher energy levels followed by spontaneous and fast de-excitation by emission of gamma rays. Detection of the intensity and energy of this radiation uniquely defines the chemical element (even the isotope of this element) and its concentration.

RPM Rotations per minute,—describes the rotation speed for an object under angular rotation.

Saturation In this context, saturations is taken to mean water saturation. This is the fraction of the pore space in the reservoir rock occupied by water (denoted S_w). In a petroleum reservoir, two other phases may co-exist with water; namely, oil (with saturation S_o) and gas (with saturation S_g). We then have that $S_w + S_o + S_g = 100\%$ (of the available porosity). In a typical native oil reservoir S_w may range from 10% to 50%, and is 100% in a pure aquifer.

SNP Sidewall neutron porosity tool,—a well logging tool for the determination of matrix porosity designed for use in open holes only. The tool has a neutron source and a single neutron detector with a 16 in. spacing. The detector is sensitive to *epithermal* neutrons. These neutrons are not yet slow enough to take part in absorption reactions with hydrogen and chlorine. Hence, the SNP tool readings are unaffected by the presence of chlorine in high salinity muds and formation fluids.

TOF Time-of-flight,—the measurement of the time taken by an object, particle or wave (for instance a neutron, a charged particle or an electromagnetic photon quantum) to travel a distance through a medium, in many cases through vacuum.

Water-wetting This is the ability of a water to spread over a surface of reservoir rock, resulting from intermolecular interactions when the two are brought in contact. The degree of wetting is termed wettability. A water-wettable surface may also be termed hydrophilic and a non-wettable surface hydrophobic. One method for measurement of water wettability is the contact angle of a drop of water placed onto the surface. The contact angle is the angle at which the liquid-vapor interface meets the solid-liquid interface. A completely water-wetting surface has a contact angle of 0 degrees, i.e., the water spreads uniformly and covers the complete surface. Most oil reservoir rocks have a so-called intermediate wettability.

XCT X-ray computed tomography,—a nondestructive technique for visualizing interior structures within solid objects, and for obtaining digital information on their 3-D geometries and properties. The measurement principle is based on the attenuation of the X-rays, which mainly depends on density (including electron density).

Introduction

This chapter should be read back-to-back with (Bjørnstad, 2021b), which sketches the technical background for practical neutron applications. Portions of the material in this chapter are also covered in (Thereska and Brisset, 2021a,b).

Neutron interactions are used in many different scientific fields. The methods are all based on some form of scattering or diffraction or on absorption. Data like those plotted in Fig. 1 are fundamental for some of the applications. Neutron methods can be used to study a wide range of object sizes from macro-sizes to nano-particle sizes and molecular level sizes. Neutrons can penetrate and

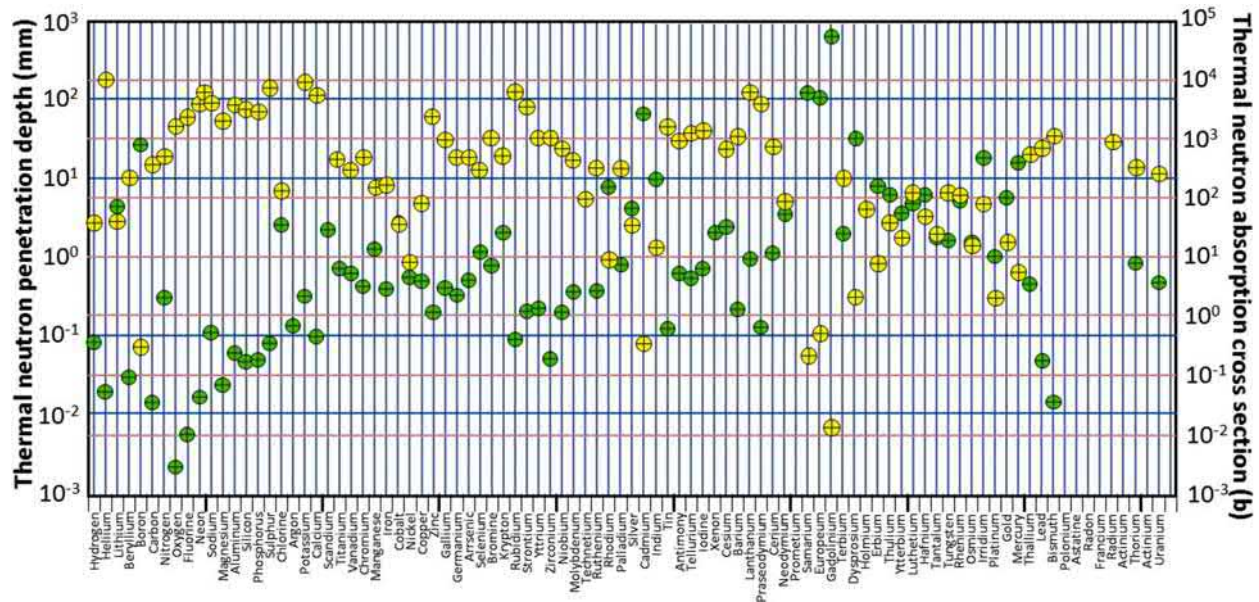


Fig. 1 Synopsis of the thermal neutron penetration depth (yellow points) and the thermal neutron absorption cross section (green points) for all chemical elements from H to U. A high absorption cross section is most often paired with a low penetration length. Data for penetration depth are redrawn from information in (Pynn, 1990) and corresponds to thermal neutrons at 27 °C.

probe into solid bulk material and reveal hidden structures, elemental compositions, and/or malfunctions and assist in testing the tolerance limits of crucial engineering components.

This chapter does not, however, discuss in detail studies on the nano or molecular level with neutron diffraction techniques (single crystal structure determination, powder diffraction studies, small angle neutron scattering (SANS) etc.). The examples of use given in the following text are all relating to the studies of macro-objects. Likewise, a major analytical technique like the beta-delayed neutron activation analysis, NAA, is not treated in any detail here. Several educational texts exist already in the published literature on this technology, see for example (Bode, 2017; Amiel, 2012; De Soete et al., 1972; Alfassi, 1990).

Neutron backscatter (or back-diffusion)

Soil water content

Knowledge about soil water content is important in agriculture (irrigation strategy) and in construction work of roads and buildings for ground compaction work. Water content may differ greatly from surface coverage where the water is “held” so strongly by the soil particles that it is not available to plants. When a soil is at this minimum “available” water content, it is said to be at the “wilting point.” On the other hand, there is a maximum concentration that can be maintained without the water draining rapidly. This is called the “field capacity.” Representative values for some soil types are given in Table 1.

It seems appropriate to define the soil water content as the volume of water per volume of soil (V_w/V_s). However, this will not be a perfect definition because different soils may swell or shrink with varying water saturation, but it is applicable within limits. A measurement method for water content should be able to handle the whole range of water saturation.

Various soil moisture measurement methods have been developed including heat capacity, electrical conductivity and electromagnetic pulse travelling time approaches. All have their own strengths and weaknesses, but these will not be discussed here. Below, we will exclusively refer to the neutron back-scattering (or back-diffusion) method.

This method uses the principle of neutron thermalization. Hydrogen nuclei are most effective in slowing neutrons down towards thermal energies due to similarity in mass (Wietfeldt, 2021). Thermalization time in water is of the order of a few μ s.

Table 1 Water content at minimum and maximum for a few soil types (IAEA, 2008).

Soil type	Wilting point (V_w/V_s in %)	Field capacity (V_w/V_s in %)
Coarse sand	2	6
Fine sand	4	10
Loamy sand	6	14
Loam	12	27
Clay	25	40

When the neutron has reached thermal energy, it runs the risk of being captured by any of the atoms in the close-by surrounding media. This risk depends on a combination of the absorption cross sections of the elements in the solid phase and the liquid phase and their respective abundances. For briny waters, chloride (Cl^-) may be one of the main neutron-capturing elements. For the solid particles, iron (Fe) may be abundant and one of the important neutron-capturing elements.

A few of the neutrons emitted from the neutron source bounce back, and even fewer of these backscattered neutrons have at this stage reached thermal energies without having been captured in compound nucleus formation of the surrounded media. These thermal neutrons will be registered by the detector, which is exclusively sensitive to thermal neutrons (BF_3 - or ^3He -counters).

The most frequently used isotopic neutron source for this instrument is the $^{241}\text{Am}/\text{Be}$ source with an energy spectrum, as shown in Fig. 7 of (Bjørnstad, 2021b), and an average energy about 5 MeV. The source strength or the neutron output is typically $\sim 10^5$ n/s corresponding to a ^{241}Am -activity of ~ 1.5 GBq (~ 40 mCi).

Fig. 2 displays the principle of a portable equipment for near-surface humidity measurement mostly used during road construction and soil compaction work and of construction material. The neutron source is lowered down from its shielded position to the surface of the object to be measured, for instance the soil. Emitted neutrons penetrate some 10–30 cm into the ground. Exact depth depends on water saturation. For 80% of the returned thermal neutrons, the penetration depth can be estimated by the empirical expression $\text{Depth (in mm)} = 280 - 2.7 F$ where F = fraction of water in % as defined above (<https://www.troxlerlabs.com>). For 10% water content (realistic for loamy sand, Table 1) this penetration depth amounts to ~ 25 cm. Some of the thermalized neutrons diffuse back towards the surface and are detected by the neutron detectors in the instrument.

In agriculture, another configuration of the same measurement principle has been developed shown in Fig. 3. This consists of a surface electronic equipment also housing a shielded neutron source with associated detectors for thermal neutrons. An access tube (made of aluminum) must be installed into the ground down to the depths of measurement interest. The source/detector arrangement is lowered on a wireline into the access tube and readings on thermalized back-diffused neutrons are recorded as a function of depth.

The detector reading is a function of the water saturation and the type of soil. Therefore, the instrument must be calibrated by varying these parameters. This can most conveniently be done in laboratory experiments where these factors can be precisely controlled.

Neutron well logging

We here consider only wells drilled in hydrocarbon reservoirs, although some of the technologies may also be used in fresh-water exploration and in geothermal energy production. There are several variations of neutron-based well logging. Some methods use isotopic neutron sources, and some apply neutron generators with 14 MeV neutron energy. In this text we will concentrate on neutron-based logging using neutrons from isotopic neutron sources. The main use of the neutron log is to provide porosity information of the near-well reservoir rock.

The well is drilled with the use of either water-based or oil-based mud with varying composition. A mud cake is built up on the well wall and mud filtrate is intruding into the near-well zone.

The porosity measurement is based on the assumption that the pore volumes in the formation rock is filled with either water or condensed hydrocarbon (oil). If free gas occupies some of the porosity, special corrections must be made to the instrument

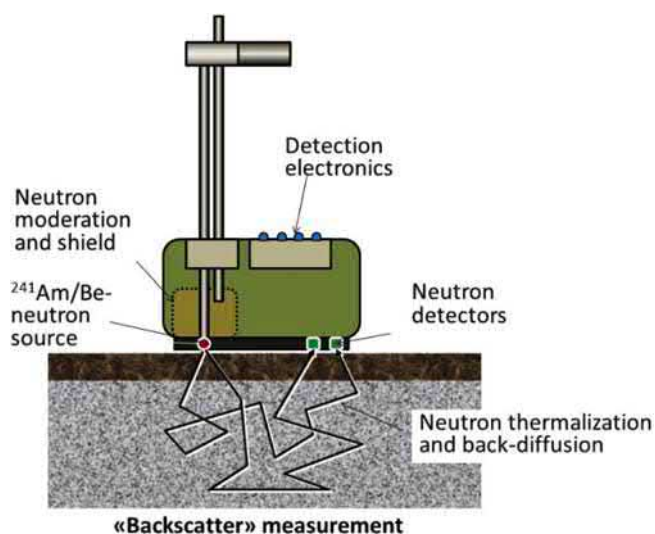


Fig. 2 Principle of a portable equipment for measurement of soil humidity in the near-surface layer especially used in road construction and soil compaction work.

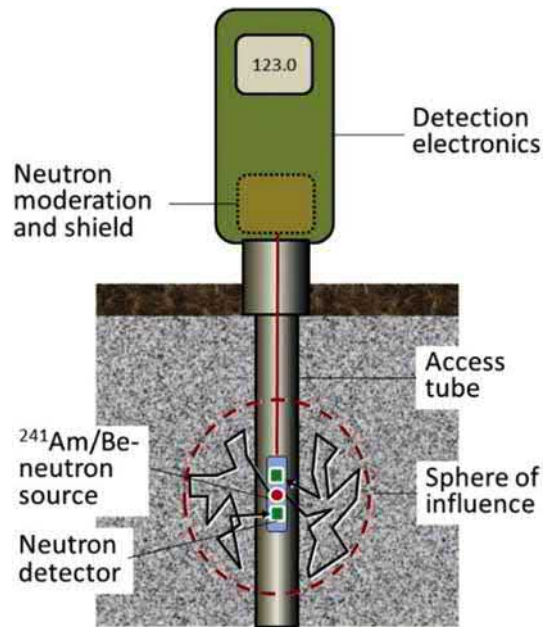


Fig. 3 Principle of a portable equipment for measurement of soil humidity used in the agricultural industry for irrigation control.

readings. The basic interaction is, as in the descriptions above, the slowing down of neutrons due to the collisions mainly with hydrogen atoms in the water and oil.

In neutron logging, some tools measure the epithermal neutrons, some the thermal neutrons, and some the gamma rays emitted when a neutron is captured by surrounding atoms.

Three methods are in use (Glover, 2014). These are

1. *The Gamma Ray/Neutron Tool (GNT)*: This tool has a neutron source and a single detector that is sensitive to high energy capture gamma rays (mainly from hydrogen and chlorine) and a detector for thermal neutrons. The tool is applied when centered in either open or cased holes. It is sensitive to, and must be corrected for, the borehole quality, drilling mud composition and mud cake thickness. The reading of the two detectors are both affected by the concentration of chlorine. This can give erroneous porosity values, which take the form of overestimated porosities.
2. *The Sidewall Neutron Porosity Tool (SNP)*: This tool can only be used in non-cased holes pressed against the well wall. The tool has a source and a single detector measuring epithermal neutrons. Since the cross section of chlorine for capture of epithermal neutrons is low, it is not particularly sensitive to chlorine contents of formation fluids. Source-detector spacing is ~ 40 cm. Rough well walls can cause misalignment of the tool and give erroneous readings.
3. *The Compensated Neutron Log (CNL)*: This tool carries a strong neutron source and two gamma-sensitive detectors spaced from the source at ~ 40 cm and ~ 65 cm, respectively.

The gamma detectors measure prompt gammas from the capture of thermal neutrons in hydrogen and chlorine. Thus, they are indirectly sensitive to thermal neutrons and at the same time to chlorine concentration, which must be corrected for. The tool is clamped against the well wall and can be used in cased holes. The measurement principle is recording of the difference in the thermal neutron population at the position of the two detectors. This difference is proportional to the concentration of hydrogen, which again is a function of the rock porosity. The principle of the CNL-method is illustrated in Fig. 4.

All tools need to be calibrated. The SNP and CNL tools are calibrated in blocks of limestone, sandstone and dolomite of high purity and accurately known porosity in test pits. They are calibrated to give the porosity directly in percent. Limestone is chosen as the calibrating lithology, so the readings are correct for limestone but need corrections for sandstone and dolomite or any other lithology. Fig. 5 shows an example of such corrected calibration curves.

Level gauging

Neutron back-scattering or back-diffusion techniques are also used to assess the status of tanks, silos, separators and other types of process equipment containing fluids and solid material of different composition. An example from the petroleum industry is given in Fig. 6, which shows the application of a neutron gauge to detect levels of different fluid fractions inside a fluid phase separator. Such large-scale process equipment is often opaque, thick-walled (2–3 cm) and made of metals like aluminum, titanium and steels. These metals are most often alloyed with other elements. Concentration of alloying elements may vary from 0.001% and below to 10s of %. Major alloying elements in steels, aluminum and titanium alloys are given in Table 2.

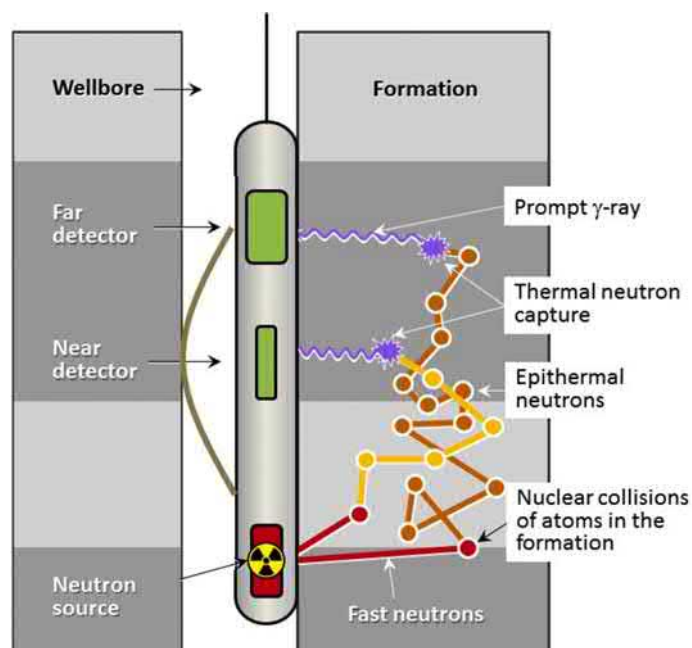


Fig. 4 Principle sketch of the Compensated Neutron Log (CNL) showing the combined wireline-operated tool with the neutron source and the two gamma detectors pressed up against the well wall for measurements.

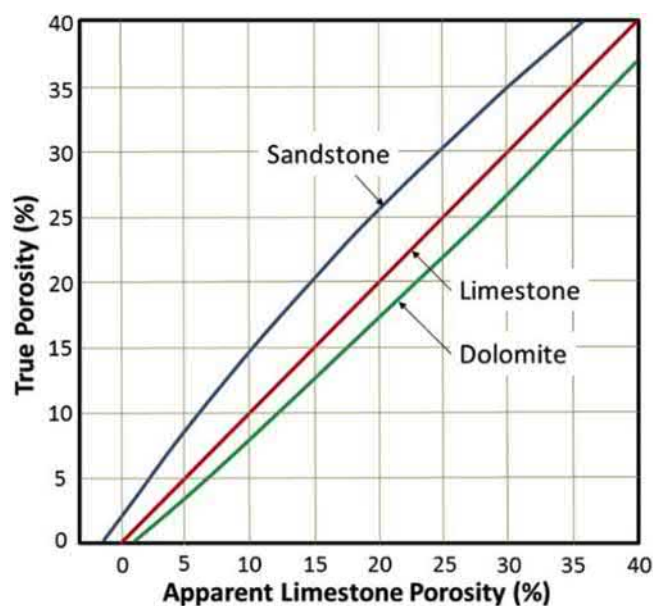


Fig. 5 Example of CNL tool where the calibration is based on Limestone with corrections for Sandstone and Dolomite. Figure modified from Glover P (2014) "Petrophysics MSc Course Notes—Chapter 15. The Neutron Log", pp. 150–171, UK: University of Leeds, http://homepages.see.leeds.ac.uk/~earpwjg/PG_EN/CD%20Contents/GGL-66565%20Petrophysics%20English/.

Although fast neutrons from the neutron source easily penetrate the thick metal walls without noticeable disturbance, the host metal and alloying elements may absorb a fraction of the returning thermal neutrons, thus reducing the flux that hits the detector. In steel alloys the absorption will be dominated by the major element Fe, in titanium alloys with the major element Ti, but in aluminum alloying elements may play a bigger role. In fact, the thermal neutron absorption in the metals will lead to compound nucleus formation with subsequent emission of prompt gamma rays. The intensity of this gamma ray emission is proportional to the flux of thermal neutrons hitting the metal wall, which again is proportional to the hydrogen content inside the vessel at the gauge position. Thus, detecting the gamma rays is an alternative to monitoring thermal neutrons. That will, however, require a more sophisticated detector arrangement with energy-dispersive gamma detector with both high efficiency and sufficiently

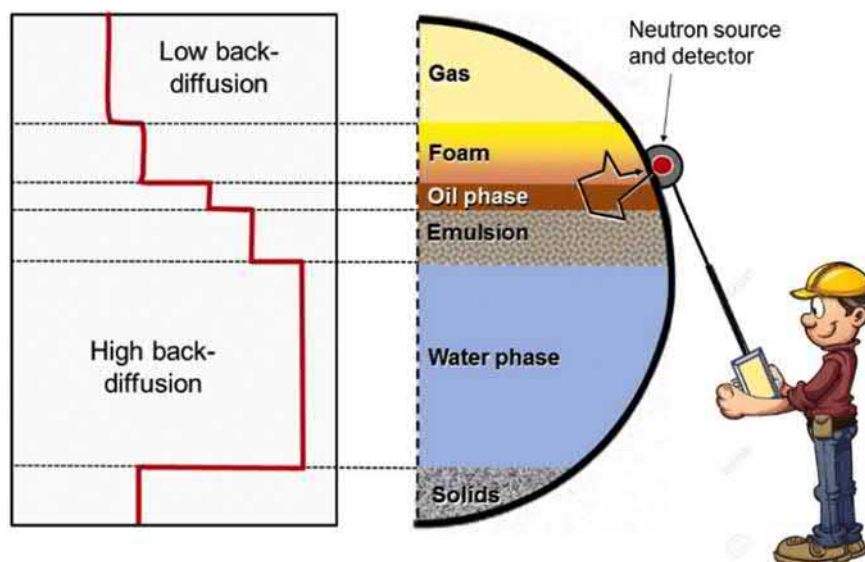


Fig. 6 Principle sketch of a neutron-based back-diffusion gauge applied on a phase separator for produced fluids in the oil production sector. The gauge includes an isotopic neutron source and a detector for thermal neutrons. Panel on the left indicates how readings change with gauge position on the separator outer surface, mainly dependent on the hydrogen concentration in the interior fluid phases.

Table 2 Main construction host metals in process equipment for harsh conditions and their common alloying elements with the metal absorption cross section (σ) in bracket.

Main element in the metal (σ in b)	Common alloying elements (thermal absorption cross section σ in barn in brackets)
Iron, Fe (2.7)	Mn (13.3), Ni (4.5), Cr (3.0), Mo (2.5), V (5.0), Si (0.17), B (760) (Less common: Al, Co, Cu, Ce, Nb, Ti, W, Sn, Zn, Pb, Zr)
Aluminum, Al (0.23)	Mg (0.07), Si (0.17), Mn (13.3), Zn (1.1), Cu (3.8)
Titanium, Ti (6.1)	Al (0.23), V (5.0), Mo (2.5), Nb (1.1), Zr (0.19), Sn (0.61), Cr (3.0), Fe (2.7), Pd (7.0)

high energy resolution, which poses some challenges. Therefore, and so far, thermal neutron detection is the major detection technique for this type of application.

The technique described here is extremely useful in that access need only be to one side of the vessel. This makes it very quick. It is also accurate with a vertical resolution typically of a few cm.

Typical applications of a portable neutron-based back-diffusion gauge mounted on an extension pole, in addition to the application described above, include:

- Determination of storage tank sludge and inventory levels.
- Determination of moisture behind isolation in transportation pipelines to prevent corrosion attack.
- Determining liquid heights in distillation column downcomers and trays.
- Provisional level measurements in tanks, silos and separators during repair of permanently installed gauges.
- Calibration of permanently installed level gauges.
- Locate hydrate, wax, asphalt, polymer and other solids build-up inside process lines.
- Examine levels of deposits on the shell side of heat exchangers.
- Locate voids or lack of solid support under tank floors.

Neutron radiography and tomography

General

A common name for neutron radiography and tomography can be *neutron imaging*. That is the process of producing an image of an object with the use of a neutron beam. The result is based on the neutron attenuation properties of the imaged object. Unlike for X-ray and gamma ray imaging, where the image is mainly based on density contrasts where the densest material gives the highest attenuation. The neutron-based images are mostly sensitive to light materials, and especially to those containing hydrogen atoms. In addition, some elements with exceptionally high probability of absorbing thermal neutrons like boron (B), cadmium (Cd), indium (In), and the lanthanide metals samarium (Sm), europium (Eu), gadolinium (Gd), and dysprosium (Dy), and some other

chemical elements (see Fig. 1) will also induce strong attenuation. Thus, X- and γ -ray based methods and neutron-based methods are close to be complementary (see Fig. 7), and a combination of the two constitutes a strong imaging tool.

A neutron radiograph is a two-dimensional image, a projection, generated by neutrons of an object at rest. Traditionally, but even today, the final image is generated on a radiation-sensitive film, especially when pictures with high spatial resolution are required.

Neutron transmission tomography is the process of generating a three-dimensional image of an object with volumetric resolution. This can be done by turning the object in selected angle increments and recording a picture for each new position. This process is normally not using radiation-sensitive films, as it would be too time-consuming. A recent (2013) review of significant advances in neutron-based imaging over time is given in ref. (Brenizer, 2013).

Below, we offer a brief description of both neutron radiography and neutron tomography.

Neutron radiography

As mentioned above, neutron radiography is based mainly on the application of a radiation-sensitive film to create an image of the interior of an object. The film technology generally produces the highest resolution form of neutron imaging, although digital methods with pixelized detectors are recently achieving comparable results.

The principle of the imaging process is illustrated in Fig. 8. Here, an object of a homogeneous material (which in principle can be composed of several chemical elements) with sections of different thickness and with an imbedded oblate-shaped void is irradiated by epithermal or thermal neutrons. The penetrated neutrons are recorded in a detection cassette arrangement, which, as one of the

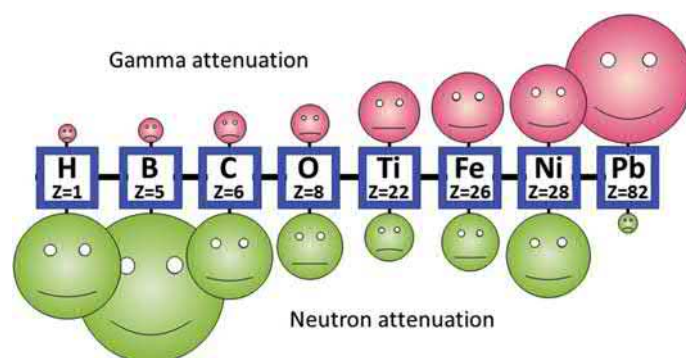


Fig. 7 Semi-quantitative illustration of the relative attenuation of gamma rays in different material and the corresponding thermal neutron attenuation in the same material. The smileys are not exactly to scale. They are solely meant to illustrate the attenuation tendency. Note the complementarity of the two imaging methods for hydrogen and lead.

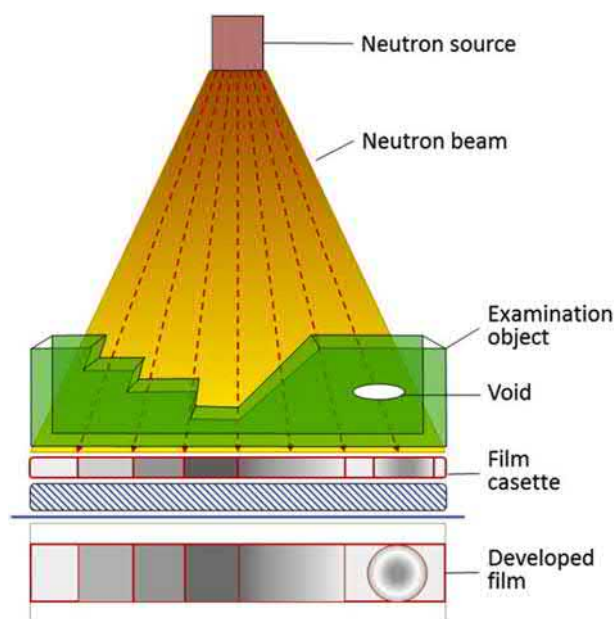


Fig. 8 Principle sketch of simple neutron transmission through an object of a homogeneous material with different material thickness and with an imbedded void.

components, may contain a radiation-sensitive film. The transmitted neutron flux is at its highest where the material is thinnest, and there is also where the “blackening” on the film is strongest. The analogue film image so produced may be digitalized afterwards using light transmission scanners.

Images are normally not generated by detecting the penetrated neutrons directly on the radiation-sensitive film but by application of so-called conversion screens (Mix, 2005). There are mainly two types of screens applied; namely, the *direct transfer screen* and the *indirect transfer screen*.

The most frequently used approach for direct transfer of the neutron intensity applies a screen with gadolinium (Gd) homogeneously distributed over the screen area. Gd has a very high thermal neutron capture cross section, in particular due to the two stable isotopes ^{155}Gd with a natural abundance of 14.8% and a thermal cross section of $\sigma_{n,\text{th}} = 61,000 \text{ b}$ and ^{157}Gd with a natural abundance of 15.65% and a cross section of $\sigma_{n,\text{th}} = 254,000 \text{ b}$. When a screen of Gd is hit by thermal neutrons, (n,γ) -reactions will take place for all stable Gd-isotopes, and for the two mentioned the products will be the two *stable* isotopes ^{156}Gd and ^{158}Gd , respectively. Thus, very little radioactivity is generated in the screen, so the same screen can be used repeatedly. However, upon emission of the prompt γ -rays (corresponding to release of the neutron binding energy), conversion electrons will also be generated and emitted. The γ -rays are not of interest here. They are more of a disturbance and a nuisance. It is the conversion electrons (Bjørnstad, 2021b) that are important and utilized; they will “blacken” the radiographic film when the film is put in close contact with the conversion screen during neutron bombardment of the object. The “blackening” (upon film development) is most intense in areas where the neutron flux ($\text{n cm}^{-2} \text{ s}^{-1}$) is highest. Thus, a 2D projected image of the internal parts of object is directly, but also indirectly via the conversion screen, transferred to the film. Although other elements like boron (B), lithium (Li) and cadmium (Cd) may also be used as screen material, it is the Gd-screen that provides the best spatial resolution and the best contrast. Resolution can be as good as $100 \mu\text{m}$.

The indirect transfer screen works in a different way. With this method, the radiographic film is not mounted in the neutron beam path during object bombardment. Instead, a screen with another chemical element with a high absorption and *activation* cross section, is applied. The screen is mounted behind the object. Upon neutron bombardment of the object, the screen will be activated, i.e., radioactive nuclides will be generated in the screen. The activity distribution is proportional to the transmitted neutron flux. The optimum half-life of the generated activity is typically a few hours. After bombardment, the radioactive screen is dismantled and placed in close contact with the radiographic film. Beta particles emitted from the screen bombard the film for a period of several hours. Areal film “blackening” is proportional to the areal activity distribution of the screen.

The preferred chemical element for this process is dysprosium (Dy). During neutron irradiation, the stable isotope ^{164}Dy with a natural abundance of 28.26% and a thermal neutron cross section of $\sigma_{n,\text{th}} = 2650 \text{ b}$ will capture a thermal neutron and produce the radioactive Dy-isotope ^{165}Dy .

This radionuclide has a half-life of 2.33 h, and decays with relatively high-energy β^- - and low-energy γ -emission. This method is more work intensive and takes longer time than the one with direct transfer screens. However, it is particularly useful for the examination of objects that are very radioactive themselves. Examples are radiography of used nuclear fuel pins and other radioactive mechanical components from nuclear installations. The indirect conversion screen method is not affected by this radiation, while the direct conversion screen method is seriously affected and cannot be used.

Neutron radiography can be applied on objects over a large size range from a few mm to truck-load sizes. Its remarkable ability to look into and reveal “secrets” behind thick and heavy metal walls and where X-ray and γ -ray radiography would completely fail, is demonstrated in Fig. 9, where a pair of tulips have been imaged inside a thick lead casket.

Neutron transmission tomography

Neutron transmission tomography is a three-dimensional imaging technique of extended objects. It provides a map in two- and three-dimensions of the neutron attenuation coefficient distribution inside an object. Thus, the internal macroscopic physical structure and material composition as a function of volumetric distribution may be reconstructed and visualized as a volumetric image.



Fig. 9 Left panel shows a photo and right panel a neutron tomograph of flowers inside a lead casket. Picture in the Public Domain.

The principle of neutron tomography has been known and applied for a considerable time already. However, the use has gained special momentum over the last years due to the development of digital detection techniques based on position-sensitive detectors with high sensitivity and acceptable spatial resolution paired with increased power in computing for picture reconstruction.

The most common neutron source for radiography and tomography has, over time, been the research nuclear reactors. But today, large installation spallation sources, medium-installation LINACs with neutron-generating targets and small-installation sealed-tube fusion neutron generators with high primary neutron output ($>10^{13} \text{ n s}^{-1}$) have become available for these purposes as well. Fig. 10 sketches the principle of the main components in a measurement setup based on a nuclear reactor.

In general, tomography provides many more details of the internal structure of the object, compared to the radiography image. After the dataset has been taken, software tools allow researchers to make virtual slices of the object in all directions.

However, there is a price to pay for this increased and detailed information. Sometimes, longer data-taking acquisition time is needed in total because the computer-based reconstruction of the third dimension needs many (hundreds) of single projection images taken during rotation of the object. Needed extra computer time for the image reconstruction may be in the range of minutes to hours. The size of the dataset and the reconstruction algorithm used determines the time. Thus, when it takes a only few seconds to record a single neutron transmission image, the tomography results are only available in time ranges of hours. Therefore, it is worthwhile to consider the need for a tomogram, instead of a 2D radiograph, in order to solve the problem at hand. It is of course also possible to accept a lower resolution image. That will reduce the total time for acquisition reconstruction.

The text below provides some examples or snapshots where only neutrons or neutrons in combination with X-ray or γ -ray transmission or X-ray fluorescence are used for material and object examination in biology, archeology, studies of cultural heritage objects, fluid flow in porous media, defense material, space mission application, detection of explosives and illicit materials and detection of smuggling across borders (homeland security).

Applications in biology

Dead and living plants and animals and other biological specimens may be subject to neutron imaging applications. An example for the use on plants is the time-resolved examination of water and nutrient uptake by the root system grown in soil and how the root systems develops with time (Tötzke et al., 2019).

Another striking example is the microtomography of a dead and dried horsefly illustrated in Fig. 11. The details are striking, considering that the size of the object is only 12 mm. These images have been produced at the ICON—Cold Neutron Imaging Facility—at SINQ, Paul Scherrer Institut (PSI), Villigen in Switzerland (ICON Instrument: http://neutra.web.psi.ch/facility/icon_index.html). This facility pretty much represents the state-of-the-art in neutron microtomography. A detection system based on a solid scintillator was built with an inherent pixel size as low as $13.5 \mu\text{m}$ (Lehmann et al., 2011). Light transmission from the scintillator to the camera-based sensor was optimized to ensure best achievable detection efficiency. This device is heavily in use for micro-tomography at a beam position at about 6.5 m from the cold source.

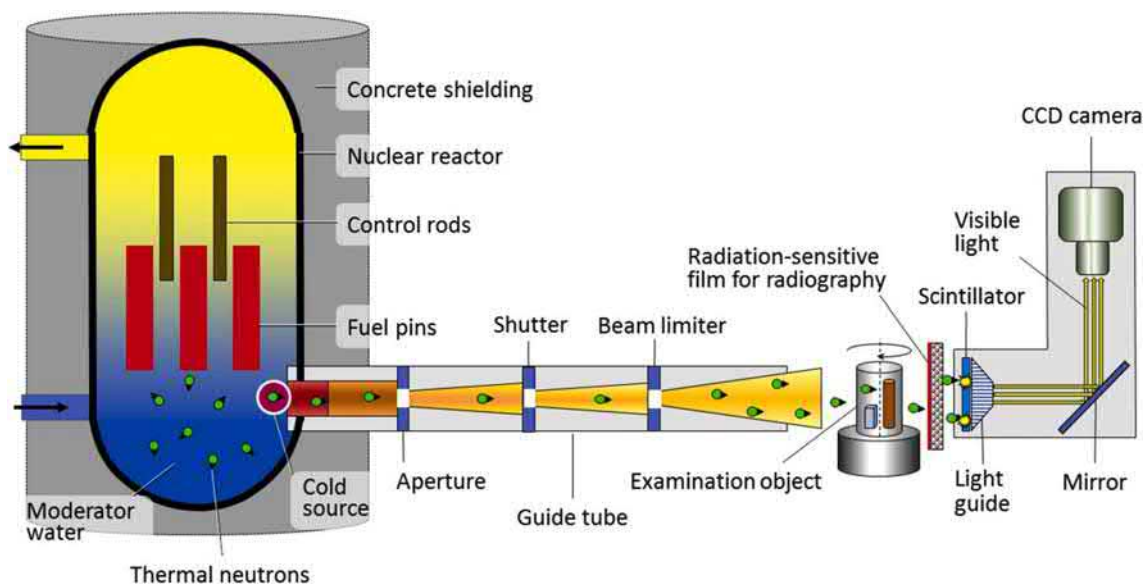


Fig. 10 Principle sketch of a nuclear reactor with a cold-source to convert thermal neutrons to cold neutrons, a subsequent beam transportation channel for the neutrons, a turn-table arrangement for tomographic measurement of an extended object followed by a detector system for the transmitted neutrons. In this case, the main components in the detector system is a cassette with a radiation-sensitive film for radiography and a radiation- and position-sensitive scintillator coupled by light guides and a mirror arrangement to a CCD camera for efficient tomographic measurements.

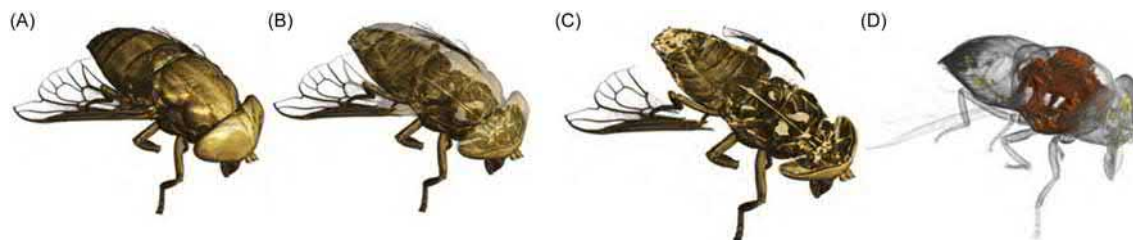


Fig. 11 Neutron microtomography of a dried horsefly. (A) Outer surface, (B) upper portion sliced away, (C) mid-level slice, (D) contour of wing muscles. The insect is approximately 12 mm long, and images have been taken with a resolution of 13.5 μm . Courtesy of Dr. Eberhard Lehmann, Paul Scherrer Institut, Switzerland.

Archeological and cultural heritage object examinations

In general, neutron radiography and tomography are often very useful for non-destructive examination and visualization of the internal structure of various archeological and cultural heritage objects embedded or not in other dense material non-transparent for visible light. The objective may be to verify its integrity, reveal hidden content and structure, assess the need of conservation and reveal restoration patches and manufacturing techniques.

A considerable fraction of available time for experiments on various neutron imaging facilities around the globe is dedicated to non-destructive study of archeological and cultural heritage objects (see also Varquez and Nagai, 2021). The literature in this area is overwhelming and specialized. To choose some examples of applications from the literature and leave others out feels nearly as a crime. Likewise, citing some references and leave others is also problematic. However, limited space for this article rules, and further reading to be enlightened in this area can begin with the references (Arif and Downing, 2006; Grabe et al., 2020; Kardjilov and Festa, 2017; Schillinger et al., 2018; Kichanov et al., 2018; Raymond and Bevitt, 2020; Festa et al., 2020).

The example chosen from archeology deals with tomographic examination of a mummified cat! The object was on loan to ANSTO (www.ansto.gov.au) from the Australian Institute of Archaeology (AIA) in Melbourne, Australia (Raymond and Bevitt, 2020).

The examinations comprised both X-ray and neutron-based computerized tomography (CT). Neutron CT appeared to be very useful for non-destructive studies of the ancient mummification techniques, and the neutrons revealed hidden contents otherwise not accessible. Both X-ray and neutron CT individually, and in combination, contributed to reveal valuable information on a partial skeleton, layers of different textile types and wadding and on bones and soft tissue (see Fig. 12). Results revealed age at death and that the animal remains belong to a small, juvenile feline.

Another example from this application area is the one represented by Fig. 13 below.

The object under study in this example is a small bottle made of red clay, with a smooth surface without any decoration. The bottle is 97 mm high and has a maximum diameter of 64 mm. A hieroglyphic seal is imprinted on its cap. The seal is identified as

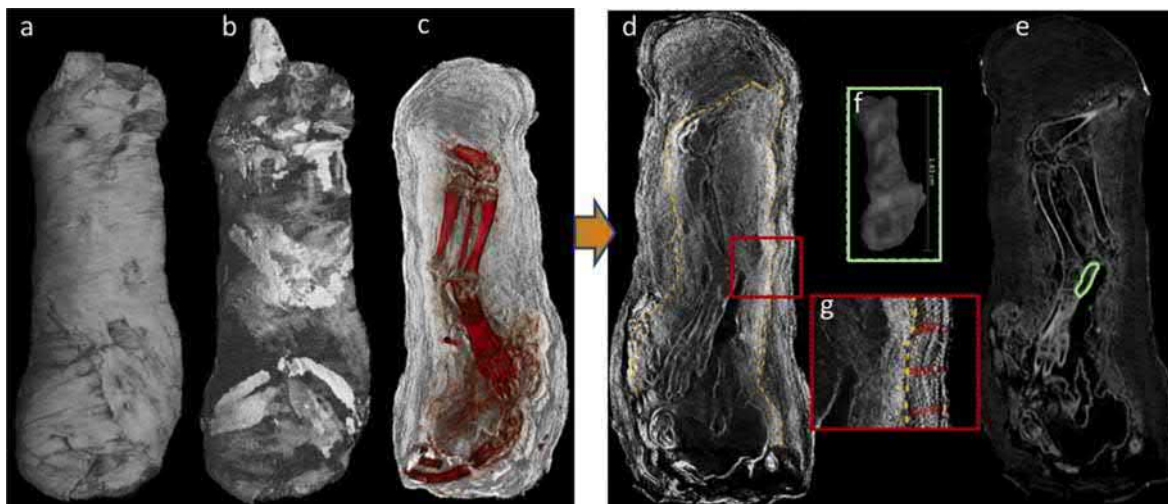


Fig. 12 (A) Surface reconstruction of a neutron CT scan; (B) surface reconstruction of an x-ray CT scan; (C) X-ray reconstruction (red overlay) on neutron reconstruction background to show the orientation of skeletal remains within the wrappings; (D) neutron CT slice revealing layers of textile wrapping; (E) X-ray CT slice yielding higher grayscale contrast; (F) segmented and visualized calcaneus bone (green area in E); (G) close-up of area in d showing wrapping layers of different thicknesses and coarseness (red box on D). Figure is composed of single pictures from the “open access” ref. (Raymond and Bevitt, 2020).

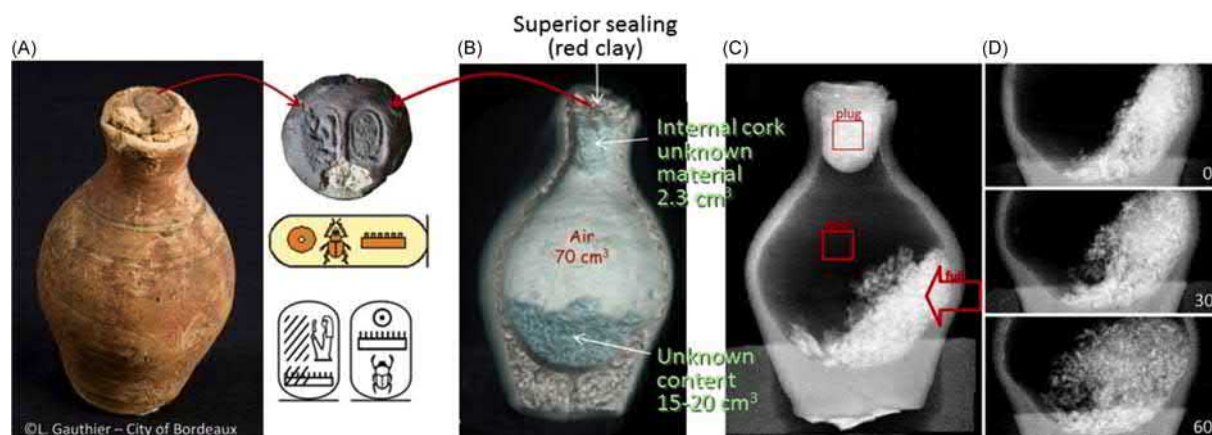


Fig. 13 (A) Photo of a sealed Egyptian pottery (on loan from Museum of Aquitaine) and details of the seal with the emblem of Menkheperre Tuthmosis III, pharaoh of the Eighteenth Dynasty (1479–1425 BC); (B) X-ray CT 3D summary; (C) global front view obtained with neutron CT. In areas with red markers, prompt gamma activation imaging (PGAI) has been performed; (D) front views showing the mobile contents by rotation the pottery in three angles. The figure is composed of single pictures from the “open access” ref. (Abraham et al., 2014).

the emblem of Menkheperre Tuthmosis III (https://en.wikipedia.org/wiki/Thutmose_III), pharaoh of the Eighteenth Dynasty (1479–1425 BC) in ancient Egypt.

The bottle and its unknown content have been examined by three different tomographic techniques. These are the THz (Tera-Hertz), X-ray, and neutron imaging. Terahertz imaging utilizes electromagnetic radiation in the wavelength range of 0.1–1 mm. This is an emerging nondestructive evaluation technique used for analysis of non-conducting (dielectric) materials. It has also proved to be applicable in the inspection of layers in paints and coatings, detecting structural defects in ceramic and composite materials and imaging the physical structure of paintings and manuscripts. The main advantage of THz radiation is the possibility of rapid and rather cheap on-site nondestructive and noninvasive diagnosis, even if it is limited by a millimeter spatial resolution.

The three techniques are largely complementary. Applied together they provide a better understanding of the studied object. THz computed tomography revealed nondestructively the presence of some content, whereas X-rays and neutrons analyzed more precisely the fabrication process and conservation of the pottery together with the nature of this content. In addition to the neutron transmission tomography studies of the whole object, it was possible to carry out prompt gamma neutron activation analysis (PGNAA) on three selected portions of the bottle. The selected regions were: The stopper/cork region, the empty region and the region with the bottle content represented by the red frames in panel c of Fig. 13.

The neutron imaging application informed the way the object has been closed by a double stopper (clay and probably a ball of linen or any other string-like organic material) and provided a more accurate representation of the mobile and heterogeneous content, which may be constituted by germinated seeds (high carbon content).

The bottle may be of some importance as it may be directly linked to the funeral ritual of the diseased, possibly by offerings of food (or hope for good crops in the next world??).

Examination of fluid flow in porous media

The nature of fluid flow in porous media is an important subject in the petroleum industry. It is directly linked to the prospects of oil recovery from sedimentary reservoir rocks, which are mainly sandstones and two major types of carbonate rocks. Sandstones are composed of silicate (SiO_2) minerals and the two types of carbonate are limestone composed of calcite or aragonite (different crystal forms of CaCO_3) and dolomite ($\text{CaMg}(\text{CO}_3)_2$). These rocks are not smooth and homogeneous in the sense that the permeability field is constant and isotropic. On the contrary. The pore volume and pore size distribution vary with position, and larger particles of impermeable rock may be imbedded. This inhomogeneity, in itself, may lead to instability in the displacement front even when displacing one brine with another brine. When displacing viscous oil with less viscous brine, this instability may be amplified and develop into a fingering pattern resulting in poorer volumetric sweep by the water and correspondingly lower oil production. This type of challenges has also been discussed previously in Bjørnstad (2021a)) and Thereska and Brisset (2021b).

The general aim of the work in the example mentioned below has been to map imbibition and saturation of water during forced constant flux experiments (forced imbibition) in porous carbonate rocks of various pore structure. One of the goals has been to map the wetting front as a function of time as the flow proceeds through the rock samples, which are initially dry and air saturated. Thus, a significant contribution of capillary forces is expected since the rock grain surfaces are water-wetting.

The experimental methods used here comprise: X-ray tomography (XCT) (see Bjørnstad, 2021a) on dry cores to describe the structure of the solid rock; and time-lapse neutron radiography (NR) to harvest information on transport of the water and the movement of wetting front related to the effect of heterogeneities (e.g., fractures and deformation bands) at the microscale; neutron

tomography (NCT) on dry and fully water-saturated rock in order to generate more information on the effective pore space contribution to fluid flow. Experiments were simulated with computational fluid dynamics (CFD) calculations.

The flow experiments were carried out by enclosing the core(s) in a flow cell of non-transparent Teflon, which was a compromise to suit both for X-ray and neutron transmission experiments.

Results from the neutron radiography experiments are displayed in Fig. 14. From these experiments one may conclude that: (a) fractures presented in samples seen in the X-ray experiments generate a non-uniform flow front, indicating the presence of some anisotropy on permeability, and (b) in unsaturated samples, water fills rapidly part of the pore-structure. The front advances with a faster speed than expected from a homogeneous filling of the pore space. This may indicate a capillary force control of the flow in this front area. Towards the end of the experiments the samples are uniformly water-saturated until the fluid spills out. This indicate a control by the macropores on fluid flow and storage in continuous flooding operation.

Applications to cargo inspection

The smuggling and unlawful transport of illicit goods poses a significant threat to the safety, security and economy of all nations (see also Patton, 2021; Gozani and Sterlis, 2021). Freight containers are potential means for smuggling (e.g., tobacco), illegal immigration, trafficking of drugs, mis-declared goods and dangerous illicit substances, including explosives, nuclear material, chemical and biological warfare agents and radioactively contaminated goods. Cargo in such closed containers for air, ship and road transport is subject to non-destructive inspection and examination methods of various kinds, in order to reveal any illicit or contraband material.

Two of the non-destructive methods in use separately are X-ray or γ -ray transmission and neutron radiography. An even more powerful combined method is achieved when integrating the electromagnetic methods with the neutron methods. Such a combined method has been developed at CSIRO in Australia during the last years (Sowerby et al., 2009). An example of this is the combination of a source of ^{60}Co (γ -energies of 1174 and 1332 keV) with a DT neutron generator (neutron energy 14.1 MeV), where the intensity changes of the radiations in the penetrating radiations are measured (combined radiography methods).

With reference to Eq. (1) in (Bjørnstad, 2021a), and Eq. (2) in (Bjørnstad, 2021b), which describe the transmission probability of X-rays (or γ -rays) and the transmissions of fast neutrons, respectively, we may rewrite these equations slightly, take the logarithm and producing the ratio:

$$R = \frac{\mu_n}{\mu_\gamma} = \frac{\ln(I_{n,x}/I_{n,0})}{\ln(I_{\gamma,x}/I_{\gamma,0})} \quad (1)$$

Here, μ_n is the mass attenuation coefficient for neutrons used, μ_γ the mass absorption coefficient for the used γ -radiation, $I_{n,0}$ the detected intensity of the neutrons without the object present, $I_{n,x}$ the measured intensity of the neutrons with the object present, $I_{\gamma,0}$ the measured intensity of the gamma radiation without the object present and $I_{\gamma,x}$ the measured intensity of the γ -radiation with presence of the object.

Fig. 15 illustrates the R -values for different selected materials showing a variation that can be utilized for enhanced characterization of mixed cargo.

Fig. 16 is an illustration of a private vehicle that has been examined by a method developed recently by CSIRO in Australia in cooperation with a private company using a 6 MV X-ray machine and a 14 MeV neutron generator. This development has included

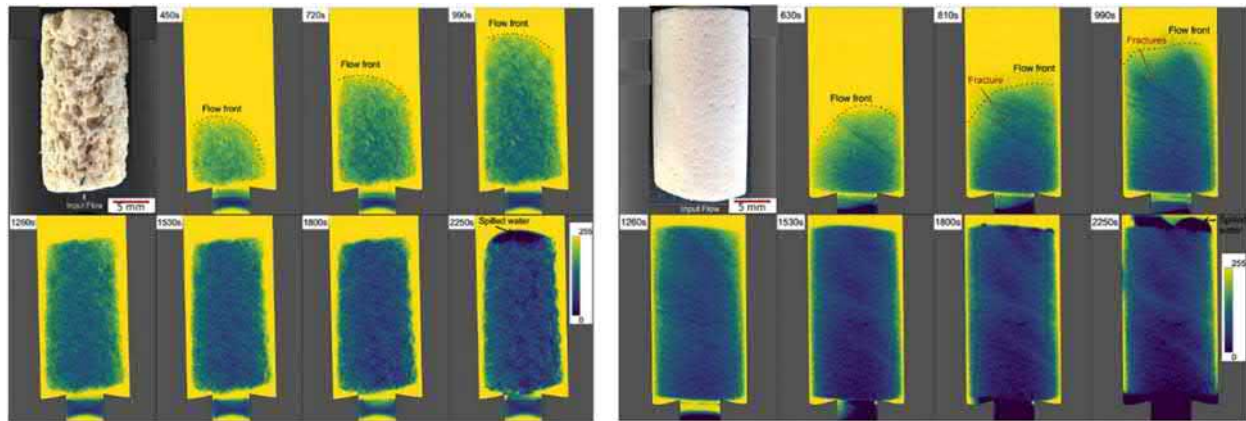


Fig. 14 Illustration shows the time-dependent neutron radiography of the flow profile of water injected in two air-filled carbonate cores with different grain size and pore structure: Core in panel to the left has much coarser pore structure than the core in panel to the right. Water enters from the bottom in these forced imbibition experiments. This type of measurements may reveal whether there is a homogeneous plug displacement or if fingering develops. Possible fluid channeling in fractures and indications of the contribution of capillary forces to the water migration may also be derived. The picture is composed of individual pictures from ref. (Zambrano et al., 2019).

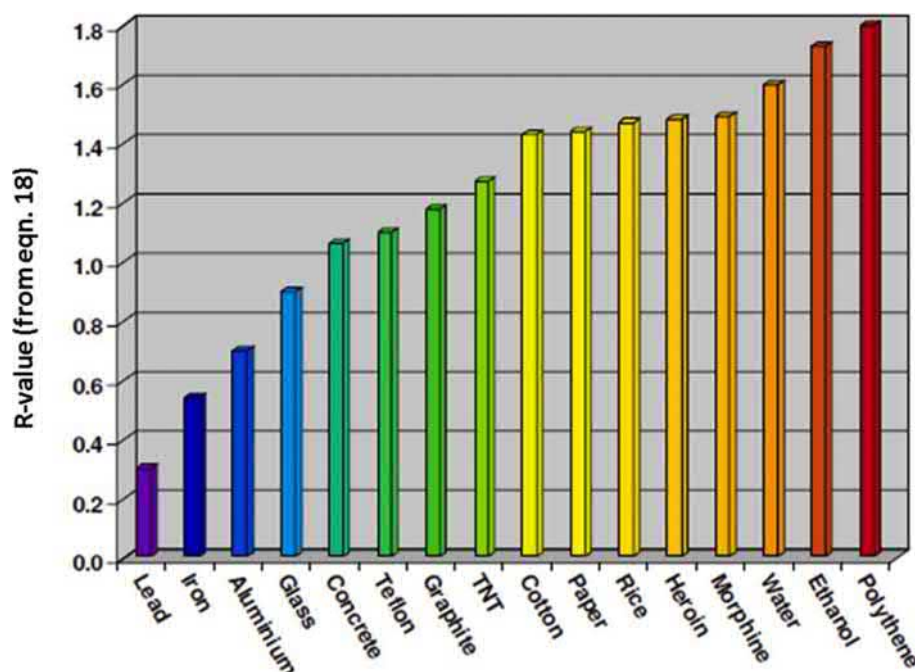


Fig. 15 *R*-values for different materials defined by Eq. (1). Courtesy of Dr. Nick Cutmore and CSIRO, Australia, Sowerby BD, Cutmore NG, Liu Y, Peng H, Tickner JR, Xie Y and Zong C (2009) "Recent Developments in Fast Neutron Radiography for the Interrogation of Air Cargo Containers", *International Topical Meeting on Nuclear Research Applications and Utilization Of Accelerators*, pp. 1–15. Vienna (Austria): IAEA, 4–8 May 2009; SM/EN-01, <http://citeseerx.ist.psu.edu/viewdoc/download?sessionid=6D0A7ED702673FD65DC67258FF7C199B?doi=10.1.1.470.795&rep=rep1&type=pdf>.

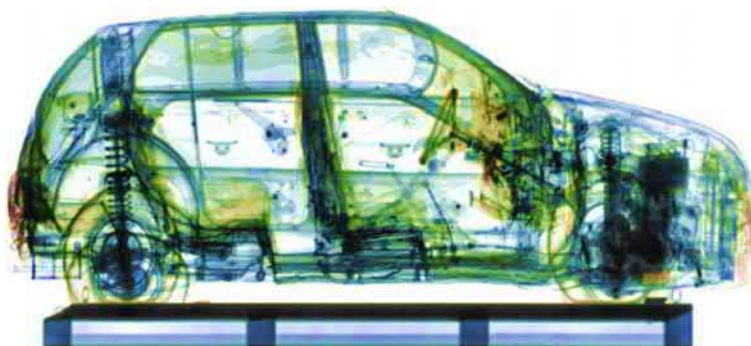


Fig. 16 Combined X-ray and neutron radiography image of a passenger car revealing both material structures and spatial distribution of some light-element materials. Courtesy of Dr. Nick Cutmore and CSIRO, Australia.

new, non-linear filtering techniques to decrease noise and increase image definition. The time for image reconstruction has decreased down to a factor of 1/10 while preserving the image quality.

One technique alone or a combination of two techniques cannot cope with all possible situations that might occur. Several initiatives and developing programs have been initiated both in United States and in Europe (<https://www.cbord-h2020.eu/page/en/publications.php>) to develop sets of combined non-destructive and non-intrusive techniques to control cargo transport (see also (Patton, 2021; Gozani and Sterllis, 2021)). Even so, it is fair to say that implementation of such advanced techniques is still in their infancy. Further technical development and possible issues about safety regulations have to continue to be sorted out.

Neutron-based elemental analysis

General about PGNA

Ordinary neutron activation analysis, abbreviated NAA, most often involves the use of thermal and epithermal neutrons, and proceeds via compound nucleus formation with the (n,γ) -reaction illustrated in Fig. 4 in (Bjørnstad, 2021b). When neutrons from isotopic neutron sources or neutron generators are used, various other reactions may take place. This is illustrated in Fig. 5

in (Bjørnstad, 2021b). Common for ordinary NAA-techniques is the subsequent detection of radiation, preferentially γ -radiation, emitted from a formed radioactive nucleus. We may term this radiation as β - or α -delayed γ -emission. The emitted γ -radiation pattern (energy spectrum) is individual for each radionuclide and has its origin in the excited energy levels in the nucleus.

The fact that all nuclei (not only the radioactive ones) have individual excited energy level patterns forms the basis for another analytical technique termed *Prompt Gamma Neutron Activation Analysis*, or PGNA in short. PGNA can for instance be performed by irradiating the sample by thermal neutrons and detecting the prompt gammas emitted in the neutron capture process (the gammas in the (n,γ) reaction, Fig. 4 in Bjørnstad, 2021b). The gamma spectrum resulting after such interactions can also be regarded as a “fingerprint,” not only of the chemical element to be analyzed, but also of the individual stable isotopes of this element. The strong features of this method are its non-invasive and non-destructive nature and the fact that it does not depend on the generation of radioactive nuclei. The excitation energy available in this process is the neutron binding energy for the bombarding neutron in the target nucleus. Neutron binding energies in nuclei close to stability are generally in the range 4–8 MeV. Thus, excitation of high-energy levels (below the neutron separation energy) is possible.

Prompt gamma rays are also generated in inelastic scattering reactions with fast neutrons ($(n,n'\gamma)$ -reactions) and in particle-emitting reactions like $(n,p\gamma)$ -, $(n,\alpha\gamma)$ - and $(n,2n\gamma)$ -reactions (Fig. 5 in Bjørnstad, 2021b). The ability to populate excited states with these reactions in the various nuclei depends on the neutron energy. D-D neutrons with an energy of 2.45 MeV do not allow excitation of higher-lying levels than this energy.

Fig. 17 shows some examples of excited states in a few relevant stable nuclei, often in macro-concentration, which can be analyzed with PGNA from inelastic scattering reactions of fast neutrons.

The nuclei shown in Fig. 17 have all high energies for the first excited levels. These levels cannot be populated in D-D reactions (except from ^{14}N), but they can all be populated in D-T neutron (14.1 MeV) scattering reactions.

Fig. 18 illustrates a simple experimental setup for PGNA analysis. For a flexible and transportable experimental setup, the neutrons may come from an isotopic neutron source or a portable neutron generator (like in Fig. 18). The exact layout of the experimental setup can then be improvised and fitted to the actual situation. Using neutrons from these sources without thermalization allows for the generation of prompt gammas by inelastic scattering reactions.

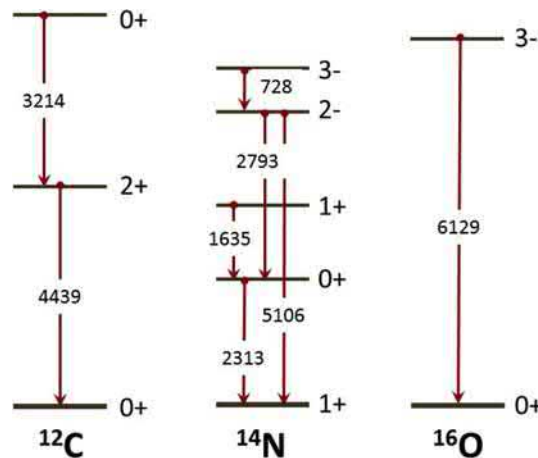


Fig. 17 Simplified schemes of excited levels for the three abundant isotopes ^{12}C , ^{14}N , and ^{16}O of the three abundant chemical elements carbon, nitrogen, and oxygen, reflecting some of the most important excitation levels and the γ -rays resulting from these during deexcitation.

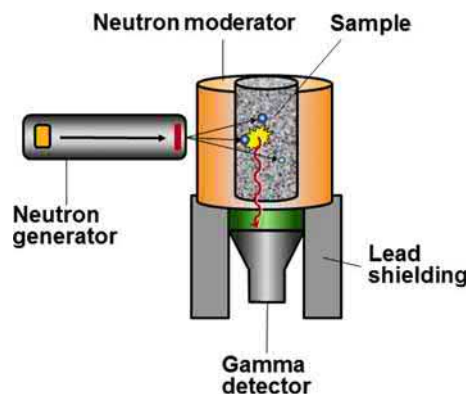


Fig. 18 Principle sketch of a PGNA experimental setup using neutrons from a neutron generator thermalized in a moderator before hitting the extended object.

Fixed PGNAA setups are often established at research reactors. Generally, higher neutron fluxes can be achieved, but only of thermal or cold neutrons. Thus, inelastic scattering reactions resulting in nuclear excitations are not possible. These setups are limited to utilization of the γ -rays emitted in the neutron capture process.

Sensitivities are not sufficiently high for most chemical elements for PGNAA to be characterized as an analytical technique for micro and trace elements. In the best cases, concentration in the low ppm range may be achieved. Still, it is a multi-element method for qualitative and quantitative analysis, but the particular strength lies in its ability to detect non-destructively light elements in bulky objects not easily achievable by other methods. A map of relative sensitivities for thermal neutron capture prompt γ -rays is provided in Fig. 19. Values on the lower limit of detection depend on a number of parameters, which may differ from one experimental setup to another. These include neutron flux, counting geometry, sample geometry, detector efficiency and energy resolution, etc. However, the general trend should be fairly similar for any setup applying thermal neutrons, and the values given may be considered as relative instead of absolute.

For quantitative analysis of a thin and small-size homogeneous object and for monoenergetic neutrons, the production rate R_p leading to the emission of a certain γ -ray energy is described by Eq. (2).

$$R_p = \sigma_\gamma \cdot \varphi \cdot N_T \quad (2)$$

Here, σ_γ is the partial cross section in barn for an excitation reaction that produces a defined γ -ray energy, φ is the neutron flux ($\text{cm}^{-2} \text{s}^{-1}$), and N_T is the number of target atoms of the analyte element. The counting rate involves a counting efficiency ε_γ composed of the counting geometry and the detector internal counting efficiency of the γ -ray energy in question. Hence, the counting rate R_c of a defined γ -ray energy is given by Eq. (3).

$$R_c = \sigma_\gamma \cdot \varphi \cdot N_T \cdot \varepsilon_\gamma \quad (3)$$

For extended objects the situation is more complicated. The neutron flux over the sample may vary ($\varphi(xyz)$) due to the self-shielding effect. In addition, the emitted γ -radiation is also subject to attenuation when penetrating the sample, and the degree of attenuation varies with the emission position ($\mu_\gamma(xyz)$). Further, if fast neutrons are applied and the sample consists of macro-amounts of light elements (plastics, aqueous or organic solutions), the neutron energy will also vary over the sample object ($E_n(xyz)$) and thereby also the reaction cross section ($\sigma_\gamma(xyz)$). For close distance between the extended object and the detector, the relative counting geometry will vary from different positions in the sample object and, thereby, also the counting efficiency ($\varepsilon_\gamma(xyz)$). The last factor is that the object may not be homogeneous, and then there is a volumetric variation also in the analyte

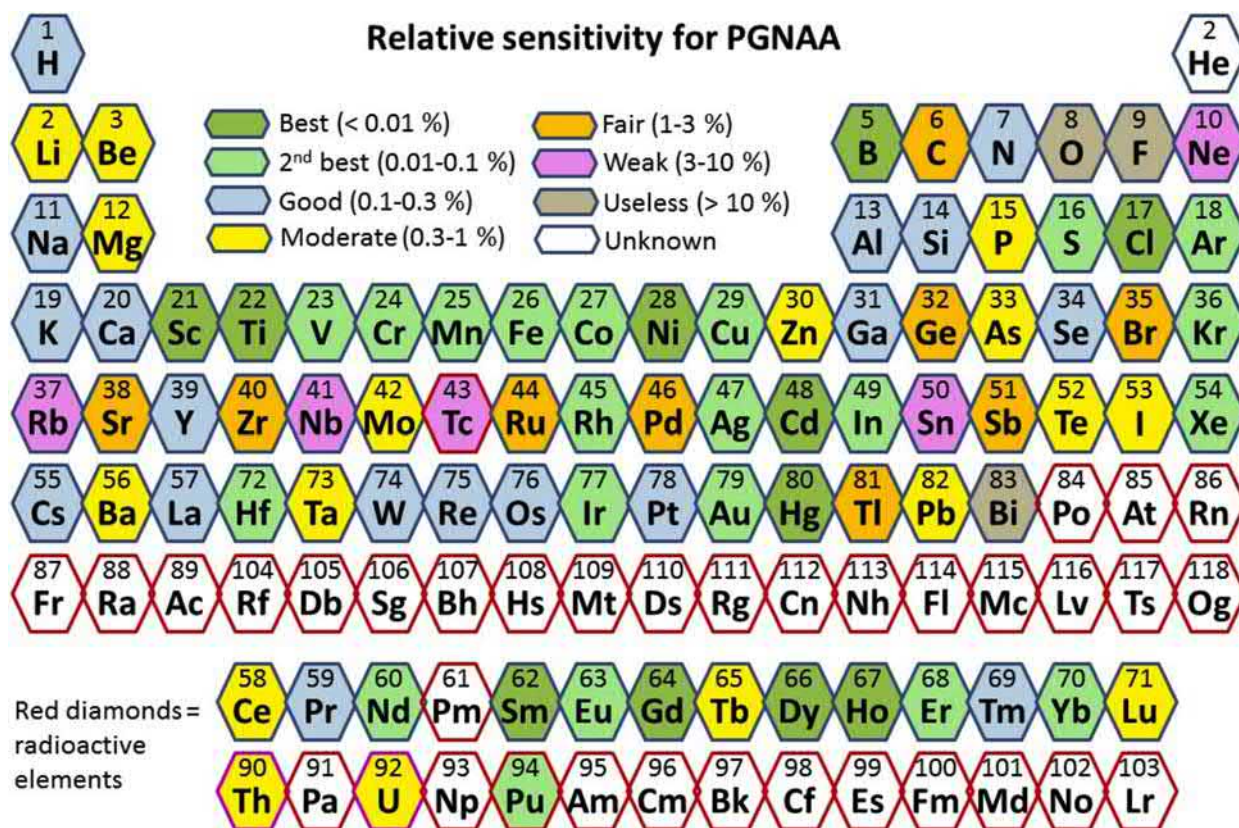


Fig. 19 Relative sensitivity of the PGNAA method for all chemical elements using thermal neutrons and detection of the prompt γ -rays from the neutron capture process.

concentration ($N_T(xyz)$). All these effects can be mathematically accounted for, but the present text is not the right place for an extended mathematical description.

Fig. 19 shows that one of the highest sensitivities for thermal neutron interactions is found for boron in the reaction $^{10}\text{B}(n, \alpha \gamma)^7\text{Li}$. The first excited state in ^7Li is at 478 keV, sufficiently low to be populated in reactions with all the relevant neutron energies.

This is, however, a special case. Because an α -particle is involved, the resulting 478 keV gamma line experiences a Doppler broadening. It is not appearing as a nice ordinary peak with normal energy resolution in the γ -spectrum. The size of the broadening is somewhat depending on the surrounding medium, broader in gases (like in BF_3) than in solid metallic boron.

The level energies and densities for the various isotopes of the chemical elements vary greatly over the nuclear chart. This is not the right place for a physical explanation of this variation, but a few general comments can be made.

For the lightest elements the excitation energies are in general higher than for heavier elements. Also, the level density for the first few MeV excitation energy is lower for the light elements than for heavier elements. However, close to magic numbers (Proton numbers 2, 8, 20, 28, 50, 82 and neutron numbers 2, 8, 20, 28, 50, 82 and 126) the situation is different. Examples of first excited states (energy in brackets) in magic proton number nuclides ^{58}Ni ($Z = 28$, 1454 keV) and ^{112}Sn ($Z = 50$, 1256 keV) and in magic neutron number nuclides ^{88}Sr ($N = 50$, 1836 keV), ^{138}Ba ($N = 82$, 1436 keV) and ^{140}Ce ($N = 82$, 1596 keV) all have higher first excited state than most of the surrounding nuclei. For the doubly magic nuclide ^{208}Pb ($Z = 82$, $N = 126$, 2614 keV), the first excited state is exceptionally high for a stable heavy nuclide.

With a high number of excited levels and a corresponding and even higher number of resulting gamma rays, the measurements may become a major challenge.

A gamma detector with high energy resolution is required if detailed spectroscopic measurements are deemed necessary. Fig. 20 shows the energy resolution achieved with three relevant gamma detectors (Seabury et al., 2006). The common, cheap and general purpose solid scintillator NaI(Tl) with a moderate energy resolution but high efficiency, the more expensive solid scintillator $\text{LaBr}_3(\text{Ce})$ with better energy resolution and even better efficiency and the semiconductor detector HpGe with the best energy resolution but lower efficiency for high-energy γ -rays.

An example of thermal neutron induced PGNAA on a real sample of natural rock minerals is shown in Fig. 21. The spectrum is accumulated with a NaI(Tl)-detector, and only the high-energy part is detailed enough for analysis revealing some of the macro-elements.

The neutron-based method described here is an essential part of the application examples that follow below.

Applications in the mining and mineral processing industries

World consumption of metals and other critical chemical elements like phosphorous are increasing. At the same time high-grade ores for some of these elements are decreasing. Thus, in order to meet larger demands, lower-grade and often inhomogeneous ores need to be mined. At the same time, the cost for mining and processing must be kept at reasonable levels. In response to these challenges, advanced online analytical techniques that support and ease selective mining and down-stream processing by better and faster characterization of ore composition are being developed and applied.

One may visualize the mining and further processing activities in a simplified way by the following steps (special procedure for metals):

Mining of rock → Crushing of rock → Transport to stockpile (on conveyor belt) → Separation of crushed rock according to mineral richness → Transport of rich rock to fine grinding → Flotation separation → Mineral dissolution → Chemical separation

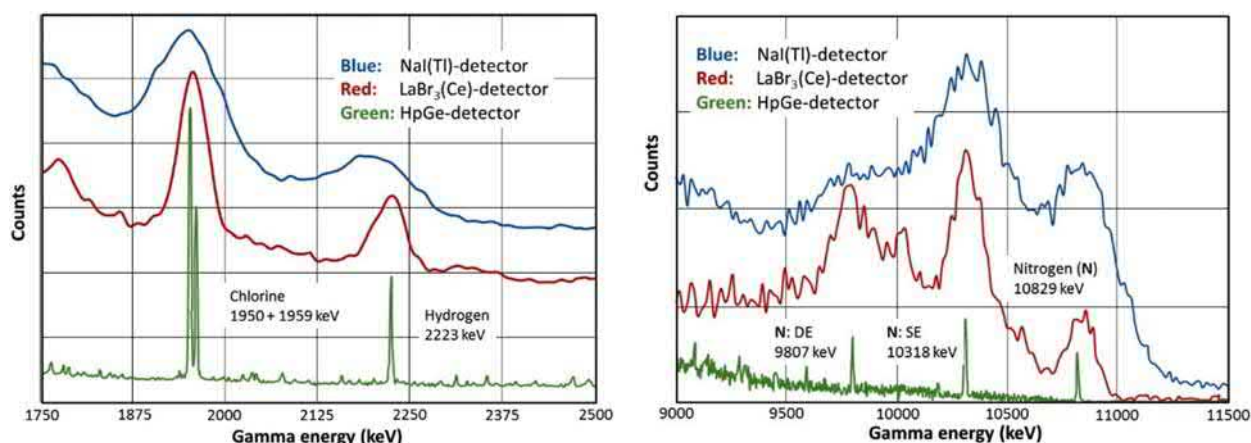


Fig. 20 Gamma spectra from detection of prompt γ -rays from chlorine, hydrogen and nitrogen accumulated with the solid scintillators NaI(Tl) and $\text{LaBr}_3(\text{Ce})$ and the semiconductor detector HpGe. Redrawn from data in Seabury EH, Wharton JC and Caffrey AJ (2006) "Response of a $\text{LaBr}_3(\text{Ce})$ Detector to 2–11 MeV Gamma Rays", IEEE Nuclear Science Symposium, San Diego, California, October 29–November 4, 2006, INL/CON-06-11300 preprint, <https://inldigitallibrary.inl.gov/sites/sti/sti/4460701.pdf>.

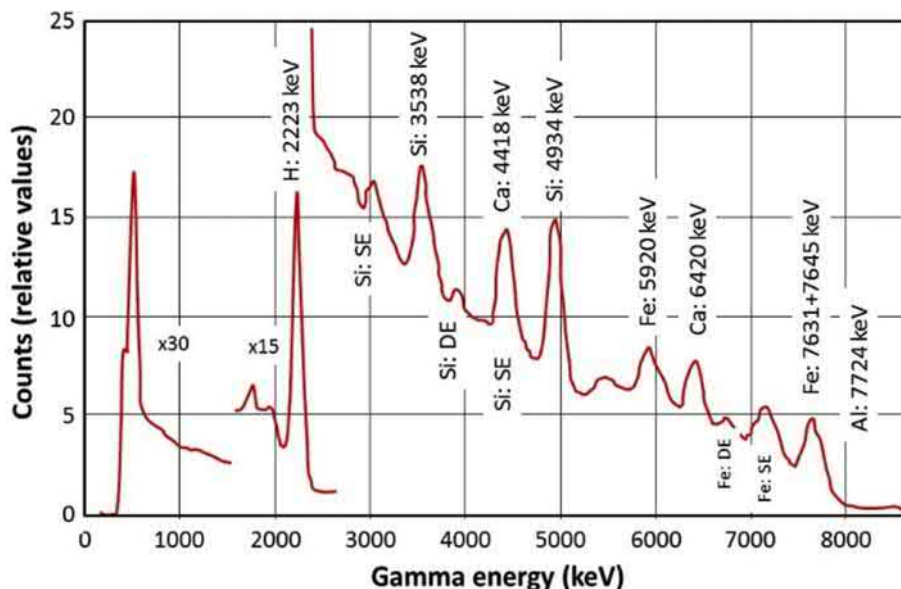


Fig. 21 Gamma spectrum from thermal neutron induced PGNA analysis of macro-components in a natural mineral sample accumulated with a NaI(Tl)-detector. Major high-energy γ -lines are seen together with single and double escape peaks.

step + ion exchange chromatography (liquid-liquid extraction) $\rightarrow$ Selective element elution $\rightarrow$ Electrowinning (of metals) $\rightarrow$ Smelting (of metals) $\rightarrow$ Final product.

Several of these processes are overseen by various sensing technologies comprising different nuclear methods. In this text, nuclear-based monitoring of two (three) of the steps in the industrial sequence above are briefly explained.

Mined ore is frequently transported on conveyors prior to stockpiling and milling and there is a need to analyze the ore with regard to ore grade and contaminating or otherwise disturbing chemical elements. Fig. 22 intends to show the principle of an industrial analytical setup where major elements in the crushed and grinded ore can be analyzed continuously and online on a running conveyor belt. Here, the analysis is based on the use of a 14 MeV neutron generator for generation of prompt γ -rays in inelastic scattering reactions. As opposed to the use of thermal neutrons for PGNA, this fast-neutron PGNA method is sensitive also to the contents of carbon and oxygen. In addition, it can be generally used for most of the chemical elements. Since the sensitivity is proportional to the reaction cross section, the sensitivities of some metals like B, Sc, Ti, Ni, Cd, Sm, Gd, Dy, Ho and some others are better for PGNA with thermal neutrons because of high cross sections for thermal neutron capture. This kind of measuring technology has only become possible in practice during the last 10–15 years due to the recent years' development of small portable neutron generators and has still not "invaded" globally the mineral processing industry.

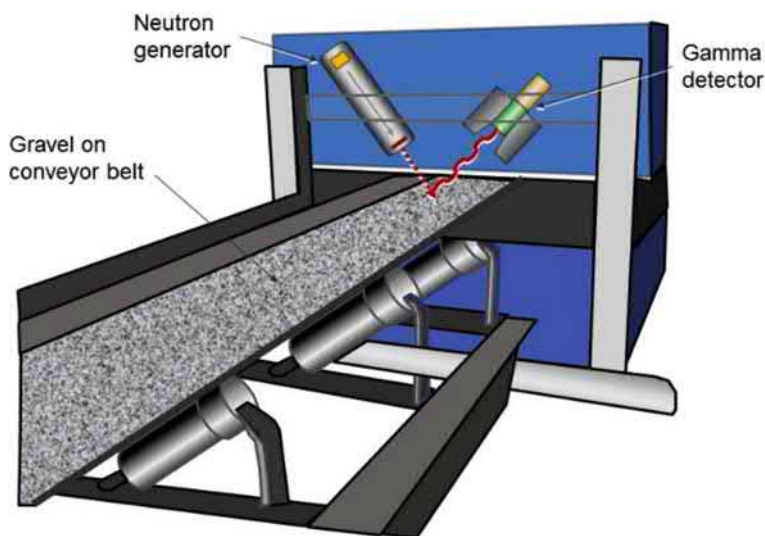


Fig. 22 Online analytical setup for bulk analysis of grained ore transported continuously on a conveyor belt. The method is based on PGNA using fast neutron from a neutron generator.

Another step in the mineral processing sequence is the chemical separation steps where liquid-liquid extraction, also termed solvent extraction, traditionally has constituted one of the major methods. It starts with the metal ore dissolved as ionic components in an aqueous solution (acidic or basic), complexation of the element (metal) of interest with a complexing agent, which makes the complex lipophilic and subsequent extraction of this lipophilic metallic complex into a contacting organic solution. Purification may take place in several steps, where the organic complex is back-extracted (stripped) into a new (virgin) aqueous phase, extracted again into an organic phase and so on. In each extraction step, contaminating elements lose ground and are finally reduced below the defined tolerance or concentration limit. The needed number of steps varies with the specific chemical procedure involved but can easily be >10 – 20 and even more. The process is illustrated in Fig. 23.

The elemental concentration in the various fluid streams in this industrial separation system may be overlooked by the use of neutron-based methods, and especially by the use of small portable neutron generators. Fig. 24 shows three principles for detection of elemental content of process fluids, which may be used at various stages in the chemical processing system of Fig. 23. The three methods are:

1. The first method in the left panel utilizes the transmission efficiency of thermal neutrons, and the targets for detection are chemical elements with a high thermal neutron absorption cross section like B, Sc, Ti, Ni, Cd, Sm, Gd, Dy, Ho and more. The intensity of the transmitted neutrons will decrease with increase in concentration of these elements. For small tube or pipeline diameters (<10 cm), it may be necessary to slow down the neutrons from the neutron generator by an external moderator (not shown in the figure).
2. The second method shown in the left panel is simply PGNA, with detection of prompt γ -rays generated both by thermal neutron absorption reactions and by inelastic scattering reactions. Both these methods are true non-intrusive, continuous and online adaptable methods.

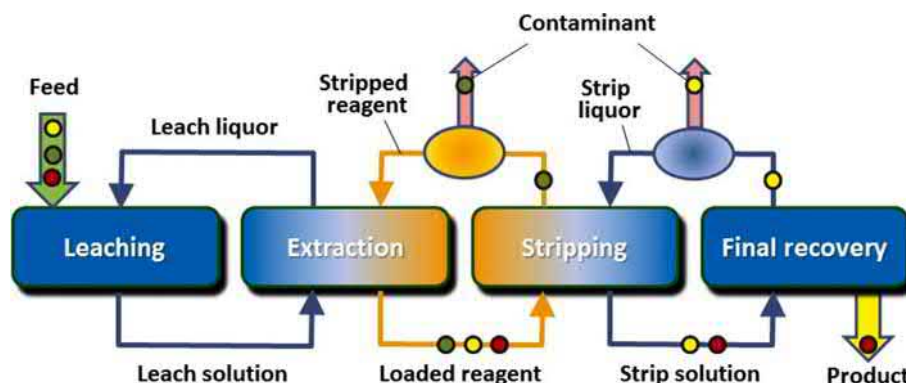


Fig. 23 Principle scheme of a chemical separation system based on solvent extraction with mixer-settler equipment where the element of interest (red circle) is separated from the contaminants (green and yellow circles) in a number of extraction/stripping steps.

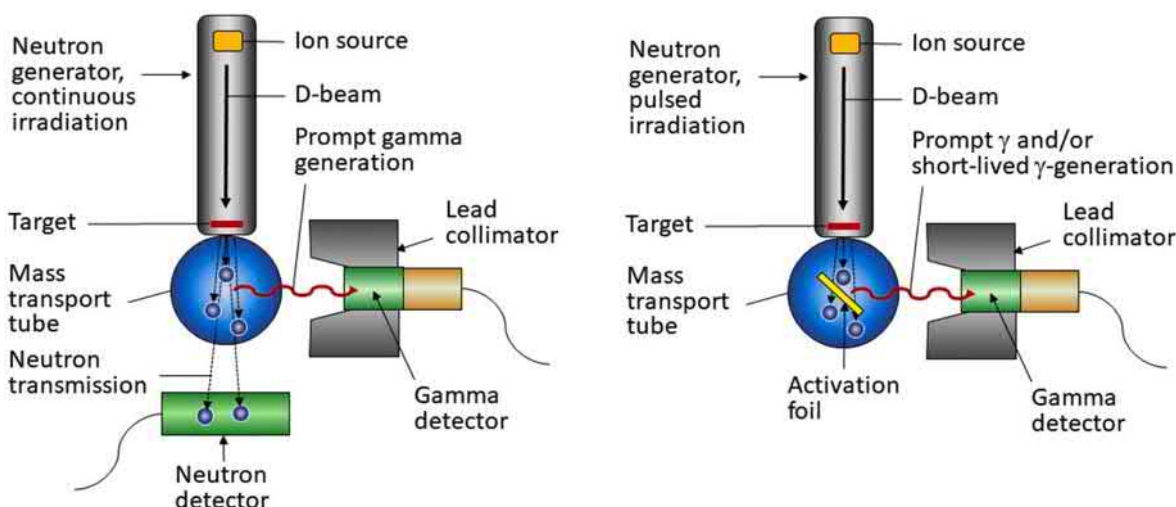


Fig. 24 Panel to the left: The figure sketches two neutron generator-based techniques for analysis of elemental content in fluid flows of process fluids,—one based on neutron transmission with detection of transmitted neutrons and one based on PGNA with gamma detection. Panel to the right: Special analytical technique of elements with high neutron capture cross section based on activation of a stationary activation metallic foil in a pulsed neutron beam.

3. The method in the right panel is also applicable for analysis of chemical elements (in macro-concentration) with a high cross section for interaction with the neutrons over an extended energy scale from fast to thermal energies. It is based on activation of a permanently installed “activation foil,” preferably a metal, which upon bombardment by neutrons from a neutron generator produces ultra-short-lived radionuclides with half-lives in the nanosecond (ns) to millisecond (ms) range. The neutron generator must be pulsed with a pulse duration in the low ms range. After a neutron pulse, the analytical system is silent for some picoseconds in order to get rid of any generated prompt γ -rays. Then it starts recording γ -radiation from the induced ultrashort-lived radionuclides in the activation foil for a period of 1–3 half-lives. Then a new neutron burst is generated, and the counting sequence repeated and so on. The concentration of the analyte in the flowing liquid is *inversely* proportional to the recorded intensity of the γ -radiation from the activation foil.

The fluid sample in point 3 above is considered homogeneous. The counting rate R_c from the generated activity in the activation foil may then be described by Eq. (4).

$$R_c = R_p \cdot \varepsilon_\gamma \cdot \left(1 - e^{(-\lambda t_i)}\right) \cdot e^{(-\lambda t_d)} \quad (4)$$

Here, R_p is the effective production rate of the radionuclide in question, ε_γ is the counting efficiency of a selected γ -ray energy, λ is the decay constant ($\ln 2/T_{1/2}$) of the radionuclide, t_i the irradiation time of the neutron burst and t_d the decay time from stop irradiation to start counting.

Candidate radionuclides to be generated in an activation foil are listed in Table 3. The methods described here must be regarded as “emerging” since some technical development remains.

Non-destructive mapping of element spatial distribution of extended objects

As mentioned previously in (Bjørnstad, 2021b), a DT-neutron generator is based on the nuclear reaction $^3\text{H}(\text{d},\text{n})^4\text{He}$, i.e., an α -particle is produced along with the neutron. This alpha particle is emitted in a direction 180 degrees opposite to each fast 14 MeV neutron produced. The α -particle has a kinetic energy of 3.5 MeV and moves back-to-back with the neutron in the center-of-mass system. In a modified form of the DT generator, these α -particles can be detected internally with a position-sensitive segmented (multipixel) array detector. By localizing the α -particle trajectory, the direction of the corresponding neutron is also determined. In addition, the multipixel α -detector will provide a time-tag for the α -particle generation, and of the accompanying neutron, as well as the direction relative to the neutron-generating deuterium target. Therefore, the direction of the accompanying neutron is also determined. The fast neutrons thus produced are, therefore, also defined as “tagged” (by the alpha particle), in time as well as in direction.

These neutron generators, with built-in α -particle detectors, are possible to apply for the identification of the volumetric distribution of selected chemical elements in extended objects with high sensitivity and position resolution. The neutron collides with a nucleus in the material under study and produces a prompt γ -ray, whose time of arrival at a position-sensitive and pixelized gamma detector can be precisely measured. This is, therefore, a Time-of-Flight (TOF) method that allows for the determination of the distance traveled by the neutron (as both the speed of the neutron and that of the γ -ray are known). Since its direction is also known, a three-dimensional volumetric resolution of elemental distributions in extended objects can be provided. This is then a form of tomography. The principles are sketched in Fig. 25.

Table 3 Some nuclei with ultra-short-lived isomeric states in the half-life region nanoseconds to microseconds, and which decay with measurable gamma energies.

Nuclide	Half-life	γ -energy (keV)	Nuclide	Half-life	γ -energy (keV)
$^{24\text{m}}\text{Na}$	20.2 ms	472.2	$^{174\text{m}}\text{Yb}$	830 μs	1518.2
$^{40\text{m}}\text{K}$	336 ns	1643.6	$^{175\text{m}}\text{Yb}$	68.2 ms	514.9
$^{101\text{m}}\text{Ru}$	17.5 μs	528	$^{176\text{m3}}\text{Lu}$	40 μs	1587.5
$^{114\text{m2}}\text{In}$	43.1 ms	502	$^{177\text{m2}}\text{Lu}$	155 μs	560.7
$^{116\text{m2}}\text{Sn}$	833 ns	3547	$^{177\text{m3}}\text{Hf}$	55.9 μs	1342.4
$^{122\text{m1}}\text{Sb}$	530 μs	137.5	$^{178\text{m3}}\text{Hf}$	68 μs	2572.4
$^{128\text{m1}}\text{I}$	845 ns	137.9	$^{179\text{m3}}\text{Hf}$	15 μs	3775.2
$^{151\text{m}}\text{Sm}$	1.4 μs	261	$^{180\text{m5}}\text{Hf}$	90 μs	3597.5
$^{153\text{m}}\text{Sm}$	10.6 ms	98.4	$^{181\text{m3}}\text{Hf}$	1.5 ms	1741.9
$^{156\text{m}}\text{Gd}$	1.3 μs	2137.6	$^{182\text{m1}}\text{Ta}$	283 ms	16.3
$^{166\text{m2}}\text{Ho}$	185 μs	190.9	$^{184\text{m1}}\text{W}$	8.33 μs	1285.0
$^{168\text{m}}\text{Er}$	109 ns	1094	$^{187\text{m}}\text{W}$	1.38 μs	410.1
$^{171\text{m}}\text{Er}$	210 ns	198.6	$^{194\text{m1}}\text{Ir}$	31.85 ms	147
$^{170\text{m}}\text{Tm}$	4.12 μs	183.2	$^{201\text{m}}\text{Hg}$	94 μs	766
$^{171\text{m}}\text{Yb}$	5.25 ms	95.3	$^{204\text{m}}\text{Tl}$	61.7 μs	1104
$^{172\text{m}}\text{Yb}$	3.6 μs	1550.4			

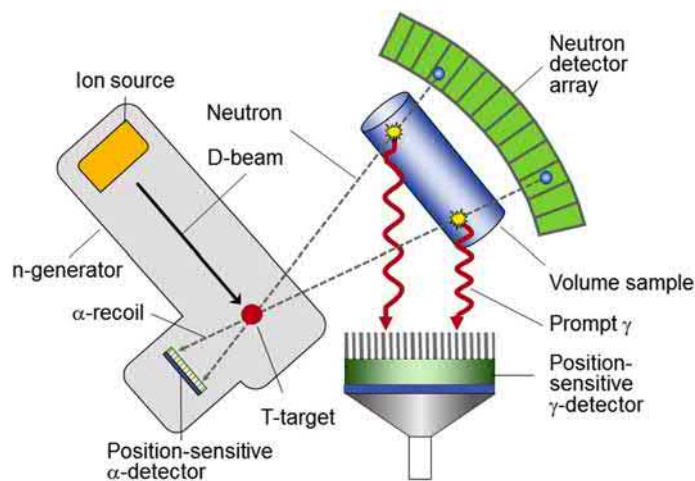


Fig. 25 Principle sketch of a 14 MeV neutron generator with a pixelized position-sensitive α -detector (the associated particle detector), for the α -particle generated in the $T(d,n)^4\text{He}$ reaction. The neutrons induce prompt γ -rays in an external extended object on a straight line determined both by the α -detector and a neutron detector array behind the sample. The position of the γ -emission in the object will then be determined by a collimated and position-sensitive gamma detector with spectroscopic properties.

Due to the use of an α -particle signal as a natural trigger, there is in principle no need to use a pulsed neutron generator for coincidence measurements. Instead, compact DT neutron generators with continuous Penning ion source can be used without expensive and heavy high-voltage pulse electronics. In some types of applications, the neutron production rate might even have to be kept low, so neutron interactions do not interfere in order to limit random coincidences.

Detection of the associated alpha particle requires that an alpha detector be built into the neutron tube. This places stringent requirements on the alpha detector material and design in that it must withstand the temperatures used in processing the neutron tube without introducing contaminants into the tube. Alpha detectors should also have very high radiation hardness for reliable operation during the lifetime of the sealed tube, which should be several thousands of hours. At present, neutron generators with built in alpha particle detectors can be provided by several companies, and we have until now only seen the beginning of the use of such equipment.

Various neutron-based applications in the defense industries

Ammunition and power devices based on explosives

Metallic capsules with energetic explosives are important in several areas, not least within the defense sector. For instance, they will power the instant removal of the canopy of a fighter plane and eject the pilot in case of emergency. In rockets that send satellites into orbit, similar explosive charges in the payload fairing separation system allow release of the payload. In war-related and similar conflicts the reliable operation of personal defense weapons like shotguns may be essential for the individual security. Thus, one must trust in the gun at hand and the corresponding ammunition. A single energetic device misfire can make the difference between success and a disastrous failure.

Such power devices, however, consist mostly of a dense outer shell encasing the powdery or solid compound mixture consisting largely of lighter chemical elements. Therefore, they are difficult to inspect for flaws with most methods for non-destructive testing. Defects such as lack of proper filling and voids, cracks, gaps and bubbles in the filling, can lead to misfires. However, the inspection is facilitated by using neutron radiography as illustrated for shotgun ammunition in Fig. 26.

Landmine detection

Landmines have over time been developed and used for two main purposes:

1. To create defensive tactical barriers, channeling attacking forces into predetermined fire zones (as in the Finnish defense against Russian invaders during the second world war) or slowing down an invading force's progress to allow reinforcements to arrive.
2. To act as passive area-denial weapons (to deny the enemy use of valuable terrain, resources or facilities when active defense of the area is not desirable or possible).

Landmines have been designed both for vehicles (tanks, trucks etc.) and for personnel to injure and kill.

Anti-personnel mines were, however, banned with an international treaty known as the *Ottawa Treaty* (https://treaties.un.org/Pages/ViewDetails.aspx?src=IND&mtdsg_no=XXVI-5&chapter=26&clang=_en) in 1997, and during the following years the

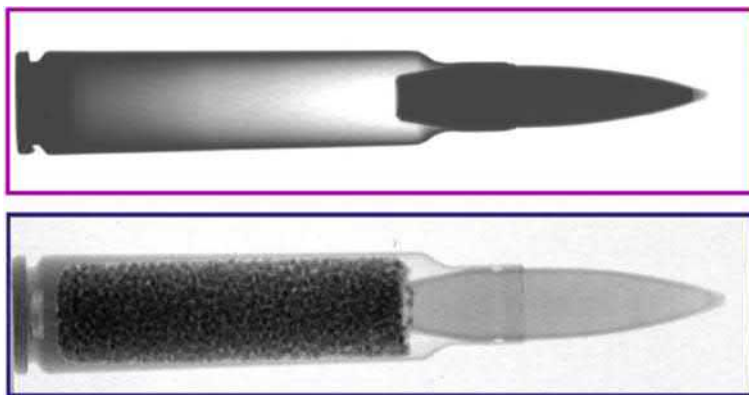


Fig. 26 X-ray (top) and neutron (bottom) image of shotgun ammunition. Note the difference in contrast where the X-rays image illustrates metal density, while neutrons are capable of penetrating nearly undisturbed through metals to image light-element materials in the gunpowder. Courtesy of Dr. Eberhard Lehmann, Paul Scherrer Institut, Switzerland.

majority of such mines were destroyed in the 164 nations that, to date, have signed the treaty. However, these do not include China, the Russian Federation, and the United States (<https://www.apminebanconvention.org/states-parties-to-the-convention/>).

As of 2013, the only governments that still laid landmines were Myanmar in its internal conflict, and Syria in its civil war (*The Economist*, 2013) but as of 2019, non-state armed groups have used landmines in additional 6 countries (*Briefing Paper*, 2019).

Despite the mentioned reduction in landmine inventory, buried and forgotten landmines still present a permanent humanitarian threat. It is estimated that there are over 110 million landmines scattered in over 78 affected countries in different regions of the world. In the period 1999–2017, more than 120,000 casualties were recorded, and countless number of people were injured.

Demining of previous battle fields is of high priority and importance but can be a tedious and dangerous work. Using traditional partly manual methods such as metal detectors, prodding and dogs is not sufficiently effective and safe. In addition, reasonably modern landmines are mostly made of plastics as opposed to older mines that contained substantial metallic parts. Thus, they cannot be detected with metal detectors.

Considerable resources have been allocated in several countries to develop new technology for landmine detection. One of the most successful and efficient developments is based on the use of neutrons. Three of the most common explosives are listed in *Table 4*. They are all composed of hydrogen, carbon, nitrogen and oxygen in various proportions. Thus, the solution to the problem of explosive identification may be based on identification of light elements suitable for neutron interaction techniques.

There are two techniques of main interest: (1) the application of thermal neutrons to detect nitrogen by thermal neutron capture and subsequent detection of prompt gamma rays, and (2) the use of fast (14 MeV) neutrons in inelastic scattering reactions followed by detection of the prompt gamma radiation for identification of all four elements.

Table 4 Three of the most common explosives in landmines.

Name/acronym	Sum formula	Structural formula
Trinitrotoluene/TNT	$H_5C_7N_3O_6$	
Cyclotrimethylene-trinitramine/hexogene/RDX	$H_6C_3N_6O_6$	
Pentaerythritol-tetranitrate/pentrite/PETN	$H_8C_5N_4O_{12}$	

Advantages of the first technique are: A relatively high neutron capture (n, p) cross section for nitrogen (1.93 b) paired with gamma emission in a high-energy region not substantially disturbed by prompt radiation from other elements (10.83 MeV). This offers a high and reliable sensitivity. The transportable neutron source may be ^{252}Cf or a neutron generator based on the DD-reaction, both equipped with a neutron moderator to slow down the neutrons to thermal energies. It is not appreciably disturbed by the common nitrogen content in soil of around 0.1%.

The second technique allows, in principle, detection of all four elements. However, it must overcome possible disturbances from soil humidity and an avalanche of γ -rays of other elements in the soil. A good energy resolution of the γ -ray detector may be more important than a high counting efficiency.

The demining process has nowadays become partly motorized and semiautomatic. Fig. 27 illustrates the principle of a remotely operated vehicle equipped with a neutron source and γ -ray detectors hunting for buried landmines. More details about neutron-based landmine detection can be found in refs. (Kregar, 2016; Cousins et al., 1998; Takahashi et al., 2011; Csikai et al., 2004).

Detection of illicit materials

Here we consider mainly illicit traffic of drugs and explosives. The detection of such concealed contraband has been based mainly on the use of X-rays, vapor detection by gas chromatography and sniffer dogs. However, since the principal elemental constituents of narcotic and explosive substances are the light elements hydrogen (H), carbon (C), nitrogen (N) and oxygen (O), the sensitivity of X-rays for these elements with low electron and mass densities is limited.

In addition, these materials are not easily detected by shape recognition because they can be formed in any shape and “hidden” behind cloths and other ordinary materials in suitcases and other types of closed containers.

This is particularly a problem in inspection of baggage for air transport. Here, the tolerance is zero for not detecting explosives when it is present.

The use of neutron-based technologies for the elemental analysis of these types of materials is a viable alternative. Fig. 28 shows that the principal elemental constituents of narcotic and explosive substances differ strongly from one-another. Explosives contain relatively high proportions of nitrogen and oxygen and relatively low proportions of carbon and hydrogen, as opposed to illicit drugs, which are generally rich in hydrogen and carbon and poor in nitrogen and oxygen.

Over the last two decades there has been much interest in exploring the use of neutron methods for the non-destructive and non-intrusive detection of explosives and illicit drugs—both in airline baggage and in large cargo containers. Such methods have already been in use for years at several locations, but methods are still continuously under development for improvement in speed, sensitivity and resolution.

Methods in use include thermal neutron-induced PGNA (isotopic neutron sources and neutron generators), fast neutron-induced PGNA (neutron generators), fast neutron induced PGNA with time and direction tagged neutrons (neutron generators with associated particle detection as sketched in Fig. 25), combined fast neutron and gamma radiography (neutron generator and gamma source).

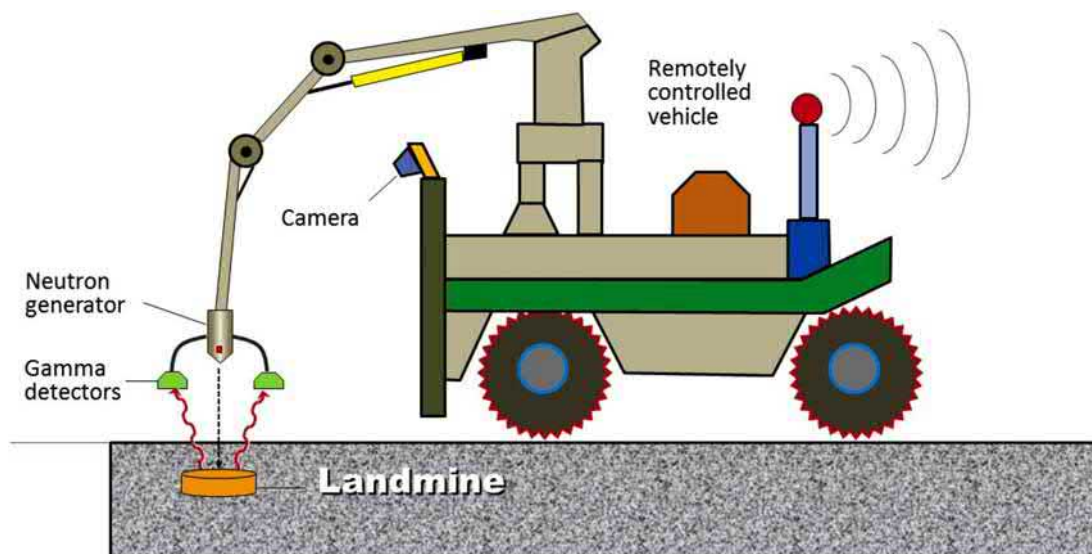


Fig. 27 Principle sketch of a motorized and remotely operated vehicle for detection of buried landmines. It is equipped with a set of gamma detectors and an isotopic neutron source (for instance ^{252}Cf) or a neutron generator of the DD (2.5 MeV neutrons) or DT (14 MeV neutrons) type. Neutrons from the two first neutron generating devices are slowed down to thermal energies before use in neutron capture reactions. The 14 MeV neutrons are used in PGNA based on inelastic scattering.

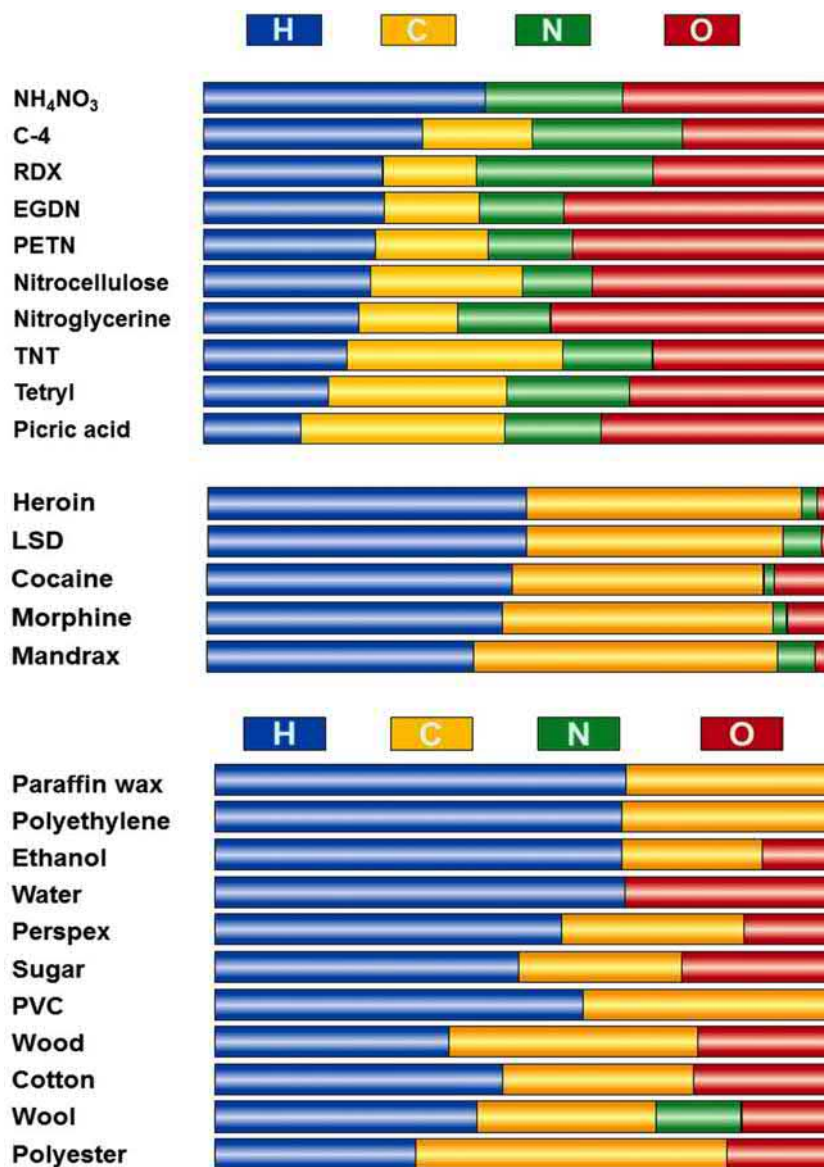


Fig. 28 Elemental composition of common explosives (upper panel), common drugs (mid panel) and other common materials (lower panel) that may be transported along with the illicit material. One observes that the relative concentrations of hydrogen, carbon, nitrogen and oxygen differ considerably between the various compounds, and this difference is basis for concealed drug and explosives detection.

Perhaps the most interesting and promising is pulsed fast and thermal neutron induced PGNA combined with detection of decay γ -rays. Further details of this technique are as follows:

Consider a situation with a concealed package with unknown but suspected content of illicit material. A pulsed sealed tube neutron generator, which produces a train of 14.1 MeV neutron pulses (DT generator), may be used here to reveal hidden secrets. Each neutron pulse has a duration of a few μ s. The inelastic scattering of the fast neutrons on elements like C and O produces prompt γ -emission with good sensitivity and main energies at 4.44 and 6.13 MeV, respectively. For each neutron pulse these are recorded with suitable γ -detectors, analyzed and stored. The sensitivity for H and N with fast neutrons is, however, not very high, so a "trick" must be performed. After recording of the prompt γ -rays (picosecond duration) a waiting time of about 100 μ s for the next neutron pulse is introduced. During this time a portion of the neutrons have slowed down, and some have become thermalized. Thermalization time for 14 MeV neutrons in an extended volume object with a considerable hydrogen content is in the order of 10–20 μ s. Thermalized neutrons are captured with usefully high cross sections by elements like H, N, Cl and P, and prompt γ -rays are emitted, detected, analyzed and stored in a file separate from the previous one. For H the important energy is 2.22 MeV and for N 10.83 MeV.

This procedure is repeated with a frequency of approximately 10 kHz. After a few hundred pulses, the neutron beam is turned off for a short period (3–5 ms). During periods with neutron pulses some short-lived radioactive components have partly grown in, the

decay γ -rays (β -delayed γ -emission) are detected after both the fast and thermal neutron induced prompt γ -rays have died away, analyzed and stored in another file separate from the two previous ones. Induced radioactive components from the presence of O will be ^{16}N ((n,p)-reaction on ^{16}O , $T_{1/2} = 7.13$ s, $E_\gamma = 6.13 + 7.12$ MeV), presence of F may generate the same radionuclide ((n, α)-reaction on ^{19}F) and in addition ^{19}O ((n,p)-reaction on ^{19}F , $T_{1/2} = 26.5$ s, $E_\gamma = 197 + 1357$ keV) and the presence of silicon, Si, may be visible through ^{28}Al ((n,p)-reaction of ^{28}Si , $T_{1/2} = 2.25$ m, $E_\gamma = 1779$ keV). Hence, by combining three different techniques, i.e., inelastic neutron scattering followed by prompt γ -ray emission, thermal neutron capture followed by prompt γ -ray emission and delayed neutron activation analysis, all the relevant light and several other chemical elements contained in an object can be measured in a continuous and non-intrusive mode without the need of sampling. Thus, illicit material may be discovered in the closed and sealed containers.

Applications for “homeland security” purposes

This section deals exclusively with the illicit transport and trafficking of radioactive and so-called “nuclear” material, where the term “nuclear material” is taken to comprise fissile (by thermal neutrons) and fertile materials (definitions are given in the text below). See also (Patton, 2021 and Gozani and Sterlis, 2021).

Control of unlawful transport and proliferation of radioactive and nuclear material has been internationally recognized as one of the most important concerted actions to take. The most direct proliferation threat has been seen to involve the transfer of completed nuclear weapons from one State to another. However, quite apart from the question of preventing transfers of complete explosive devices, there is a need to prevent transfers of radioactive and nuclear material as a whole (IAEA, 2007).

By radioactive material, in this connection, is meant longer-lived radioactive nuclei, which have the potential of contaminating large land areas when mixed with conventional explosives (like TNT) as “dirty bombs.” Such materials can also be dispersed by airplanes and drones or manually be put into reservoirs of drinking water. Examples of γ -emitting radionuclides (half-life in brackets) produced in considerable amounts mainly for industrial applications, but which can also be used in dirty bombs, are ^{57}Co (272 d), ^{60}Co (5.3 y), ^{75}Se (120 d), ^{133}Ba (10.6 y), ^{137}Cs (30.1 y), ^{152}Eu (13.5 y), ^{192}Ir (73.8 d), and ^{241}Am (432.6 y). Examples of corresponding pure β -emitters, which cannot easily be detected outside closed containers are ^{14}C (5730 y), ^{35}S (87.4 d), ^{63}Ni (100 y), and ^{90}Sr (28.9 y), while examples of α -emitters, are ^{210}Po (138.4 d), ^{226}Ra (1600 y), and ^{241}Am (432.6 y).

On border crossings, fixed radiation portal monitors (RPMs) are pass-through type monitors typically consisting of two pillars containing γ -radiation detectors both for vehicles, smaller pieces of goods and people, while pure β - and α -emitters cannot be seen by such equipment. These detectors do not necessarily exhibit spectroscopic properties, i.e., they are not able to accumulate energy spectra. However, they can be backed up by multipurpose handheld instruments for this purpose. Still, transport of illicit materials may be covered up, purposely or not, by lawful transport of the mentioned types of radionuclides aimed for industrial applications and transport of radionuclides for use in hospitals (nuclear medicine). These latter types of radionuclides are relatively short-lived and cannot effectively be used in dirty bombs. Examples of such radionuclides used in human examinations are ^{18}F (109.7 m), ^{67}Ga (78.3 h), ^{68}Ga (270.9 d), $^{99\text{m}}\text{Tc}$ (6.0 h), ^{111}In (2.8 d), ^{123}I (13.2 h), ^{125}I (59.4 d), ^{131}I (8.0 d), and ^{201}Tl (3.0 d). Other lawful transport of slightly radioactive material is goods containing naturally occurring radioactive material (NORM) mainly represented by ^{40}K (fertilizers, additives to drilling mud, foodstuff like bananas etc.) and the natural radioactive series originating from ^{238}U and ^{232}Th (raw materials/minerals for production of rare earth metals, building front cover bricks of granite, and tiles for road sidewalk pavement of granite).

However, the technology for detecting radioactive illicit material on a background of lawful radioactive material transport that does not involve neutron interrogation methods will not be further discussed here. Rather, the remainder of this section will be allotted to neutron-based methods for interrogation of illicit transport of nuclear material. Nuclear materials may be divided into two classes:

1. Those radionuclides or combinations of radionuclides that are fissionable by thermal neutrons and may be used for the development of a nuclear weapons. These include ^{233}U ($T_{1/2} = 1.59 \cdot 10^5$ y), ^{235}U ($T_{1/2} = 7.04 \cdot 10^8$ y) and ^{239}Pu ($T_{1/2} = 24,110$ y).
2. Radionuclides that can be used to produce fissionable materials. Those are said to be fertile. These include ^{232}Th ($T_{1/2} = 1.40 \cdot 10^{10}$ y) and ^{238}U ($T_{1/2} = 4.47 \cdot 10^9$ y). The first is raw material for production of ^{233}U by the reaction $^{232}\text{Th}(n_{\text{th}},\gamma)^{233}\text{Th} \rightarrow \beta^- \rightarrow ^{233}\text{U}$, and the second is raw material for production of ^{239}Pu by the reaction $^{238}\text{U}(n_{\text{th}},\gamma)^{239}\text{U} \rightarrow \beta^- \rightarrow ^{239}\text{Pu}$.
3. However, ^{232}Th and ^{238}U are also fissionable with high-energy particles like 14 MeV neutrons from a neutron generator.

The detection of concealed contraband items is a difficult challenge in most contexts. In many cases the contraband may constitute only a very small fraction of the total mass of the object being interrogated and screening often needs to be carried out rapidly. Here are a few brief descriptions of some non-intrusive neutron-based methods which have been developed or are still under development in order to meet these challenges.

- (a) *Spontaneous fission neutrons from uranium*: Passive detection of neutrons originating from spontaneous fission of ^{238}U , which has a spontaneous (unprovoked) fission decay branch of $5.4 \cdot 10^{-5}\%$. A sample of depleted uranium, ^{238}U , of 1000 cm² (corresponding to a weight of 19.16 kg) emits ~ 260 n s⁻¹. Significantly higher amounts than this of depleted uranium may be

transported in the same batch. Hence, this is readily detectable with a portal arrangement of high-efficiency neutron detectors (often based on ^3He - or plastic scintillators).

- (b) *Spontaneous fission neutrons from plutonium*: The spontaneous fission branch of the fissile radionuclide ^{239}Pu is only $3 \times 10^{-10}\%$, resulting in a neutron emission of only 6.9 n s^{-1} for a sample of 1 kg corresponding to a volume of 50.6 cm^3 . Samples much larger than this is not expected to be transported in one batch since the criticality of ^{239}Pu is around 10 kg. However, chemically processed plutonium may also contain sizable amounts of the heavier isotope ^{240}Pu ($T_{1/2} = 6561 \text{ y}$), which has a spontaneously fission branch of $5.7 \times 10^{-5}\%$. For a 1% (= 10 g) content of ^{240}Pu in 1 kg of ^{239}Pu , the neutron emission amounts to $4.8 \times 10^4 \text{ n s}^{-1}$. Normally, the isotopic ratio of ^{240}Pu is higher than 1%: Plutonium is graded by the proportion of ^{240}Pu : weapons grade (<7%), fuel grade (7–19%) and reactor grade (>19%). Therefore, samples containing chemically processed plutonium will be readily detectable by passive detection of the emitted neutrons from spontaneous fission of ^{240}Pu .
- (c) *Application of fast neutrons, the “nuclear car wash”*: Trucks with cargo containers to be inspected are towed along a track that passes the interrogation portal consisting of neutron generators (for instance with 14 MeV neutrons) on both sides together with a battery of gamma and neutron detectors (Fig. 29). Upon neutron interaction with the cargo, three effects may be observed and used to disclose the nuclear contraband:
1. Detection of high-energy γ -radiation (> 3 MeV) from short-lived fission products induced by a burst of fast neutrons from both fissile and fertile materials. The interesting half-lives are in the range 0.1 s to a few minutes. Prompt γ -rays from neutron interactions with other materials in the cargo are not disturbing because of a short delay from end of neutron burst till start of detection. Likewise, gamma radiation from natural sources is not interfering since the highest energy of some importance from natural radiation is 2.614 MeV and, therefore, not within the selected counting window. It can be argued that γ -radiation can be shielded and strongly attenuated either in passive high Z/high density shields or in the rest of the cargo in the container. However, for standard size containers (cross sections $D = 235 \text{ cm}$, $H = 239 \text{ cm}$) one has found that the maximum loading for an authorized road transport (weight limitation) is $\sim 0.86 \text{ g cm}^{-3}$ for a 20 ft. (5.87 m) container. For a centrally positioned γ -source, this corresponds to a mass density over the 117.5 cm distance to the container outer surface of 100 g cm^{-2} . In practice, mass density of actual loading in commercial traffic is in the range $0.1\text{--}0.3 \text{ g cm}^{-2}$ (Slaughter, 2016). These mass thicknesses are readily penetrated by high-energy γ -radiation. One advantage with this method is that high energy resolution of the gamma detectors is not required. Typically, several half-lives are registered in the counting window. Their decay characteristics may help to identify whether the target is highly enriched uranium, depleted uranium or plutonium, as these species have significantly different fission product yield distributions.
 2. Detection of β -delayed neutron emission from the fission products is also possible. If passive shielding against the induced γ -radiation has been implemented in the transported cargo, this may be the alternative method of choice, since neutrons penetrate normal shielding material for γ -radiation. The β -delayed neutron yield per fission for ^{235}U is ~ 0.015 , while that for ^{238}Pu is ~ 0.0061 . Half-lives of the neutron-emitting fission products are in the range of milliseconds to a few seconds. With a pulsed neutron generator with a pulse frequency in the range 0.1–10 Hz, repeated counting cycles can be performed on the same object while passing the interrogation portal, thus enhancing the sensitivity of the examination.
 3. Depending on the arrangement of the neutron detectors, a rough neutron tomogram may also be accumulated (see “Neutron transmission tomography” section earlier in this chapter). This may add positively to the experimental data inventory for detection of concealed contraband.

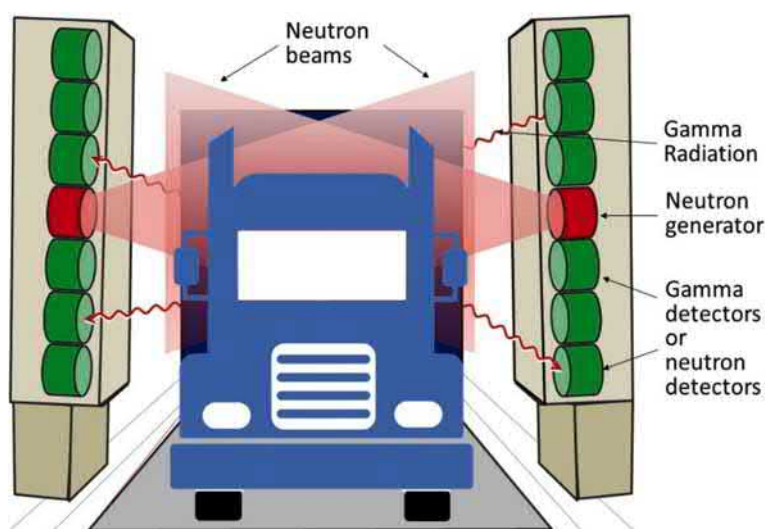


Fig. 29 Truck with cargo in sealed container are towed through an interrogation portal where the cargo is irradiated with beams of fast neutrons (like water in a car wash) to produce fission in fissile and fertile material concealed within the cargo. The portal also contains detectors for neutrons and γ -radiation.

The neutron-based “nuclear car wash” methods may be assisted by application of strong collimated γ -sources (for instance ^{60}Co) to reveal any high-density spots or volumes, for instance an active γ -shield, in the cargo (see The gamma transmission method in Bjørnstad, 2021a).

Applications in space research

The following text will concentrate on one example only: The Curiosity mission to Mars, launched by NASA, to measure evidence of water in Mars’ soil.

Already in 2004, the twin rovers Spirit and Opportunity discovered evidence that the planet Mars once hosted running water before becoming a frozen desert. This was a ground-breaking discovery. If there has been water, there might also be evidence of organic biological life. But when did this happen and why?

In order to improve data quality and, thereby, more solid conclusions, NASA launched the Curiosity mission (<https://mars.nasa.gov/msl/home/>) on November 26, 2011 carrying a supersized rover (Fig. 30) to learn more. It landed in the Gale Crater on Aug. 5, 2012. The rover carried various types of instrumentation, also comprising a pulsed 14 MeV neutron generator with a neutron output of 10^7 n/s and a corresponding ^3He -based neutron detector. This instrumentation, named DAN (Acronym for “Dynamic Albedo of Neutrons”), was developed and optimized for the space mission by Russian scientists. The rover and its instruments were powered by nuclear energy based on the radioactive decay of ^{238}Pu (Bennett, 2021a,b).

Thus, one of the question Curiosity was set out to answer is: did Mars ever have the right environmental conditions to support small life forms called microbes? The neutron generating and detecting instrumentation was included to analyze for hydrogen in the soil as a sign of water or hydroxyl (OH) as a prerequisite for life down to a maximum depth of 1 m. The analytical method is neutron scattering and back-diffusion spectroscopy, similar to that described previously in *Soil water content* and *Neutron well logging*. But neutron activation coupled with gamma spectroscopy was also used.

For the neutron scattering method the principle is in brief: The neutrons emitted from the neutron generator penetrate into Martian soil to interact with the nuclei of main rock-forming elements through several reactions including (n,n') , $(n,2n)$, (n,p) , (n,α) and (n,γ) . As a result of scattering interactions, fast neutrons are slowed down and lose part of their energy. Some of the moderated neutrons are absorbed in the subsurface, while the other part scatters and diffuses back. A fraction of these are registered by the neutron detector in a short period (milliseconds) after each neutron pulse as the time decay curve of the neutron flux. The returned epithermal neutrons have a short time distribution, while neutrons in the thermal region (who have had multiple collisions with hydrogen atoms) show a wider time distribution (Fig. 31). A high intensity of neutrons with back-diffusion times longer than ~ 2 ms is a sign of the presence of hydrogen.

The rover carried out detailed studies at the landing site, the Gale Crater, and then set out on a long journey to analyze soils in different areas and collect physical soil and rock samples. On August 19, 2015, NASA scientists reported that the DAN had detected an unusual hydrogen-rich area, at the so-called “Marias Pass,” on Mars. The hydrogen found seemed related to water or hydroxyl ions in rocks within 1 m beneath the rover. This indicated that the amount of hydrogen was three to four times as high as the amounts detected anywhere previously along Curiosity’s traverse of about 6.9 miles (11.1 km) since landing in August 2012. The hydrogen detected by DAN is interpreted as water molecules or hydroxyl ions bound within minerals or water absorbed onto minerals in the rocks and soil.

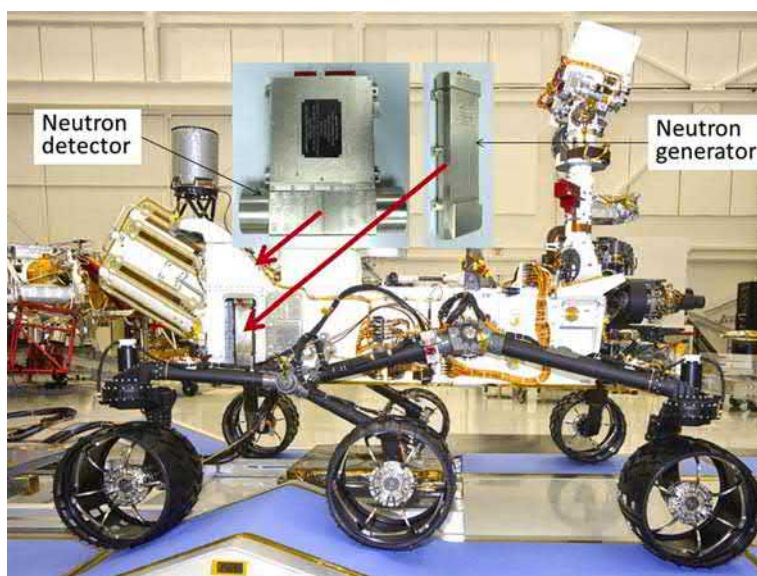


Fig. 30 The Curiosity rover for the NASA Mars mission launched in 2012. Insets are the pulsed 14 MeV neutron generator and the corresponding ^3He -based neutron detector developed by Russian scientists. Image Credit: NASA/JPL-Caltech/Russian Space Research Institute.

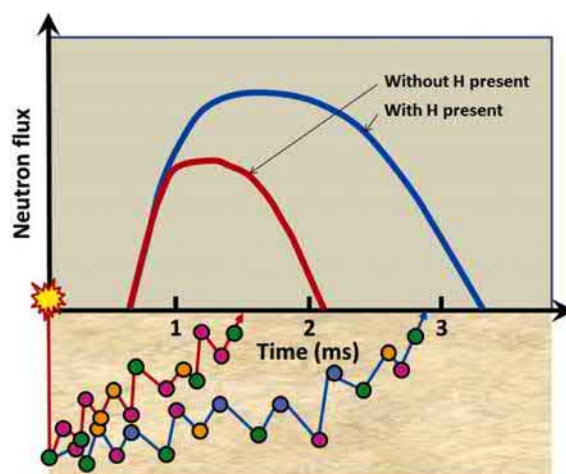


Fig. 31 Time distribution of backscattered epithermal neutrons (red curve) and back-diffused thermal neutrons (blue curve). Blue dots represent hydrogen atoms while the other colors represent other common chemical elements in the soil or rock. Figure reproduced based on information from NASA/JPL-Caltech/Russian Space Research Institute.

About a month later, NASA made the surprising announcement that the Mars Reconnaissance Orbiter (<https://www.jpl.nasa.gov/missions/mars-reconnaissance-orbiter-mro/>), launched August 12, 2005, had discovered signatures of hydrated minerals streaking down canyon hill sides and changing “colour” between warm and cold seasons (Carey, 2015). Could this be signs of flowing water?

At the time of writing this article (September 2020), the Discovery rover is still in operation and has traversed a distance of 14.12 miles (22.72 km). Each year DAN carries out around 120 measurements. About 1,300,000 neutron pulses of the neutron generator have been spent on this mission. This neutron-based method has proven beyond doubt that hydrogen-containing water-related molecules exist in the soil and rock.

See Also: Applications of Research Reactors; Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection.

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Modern Industry—Development of New Materials

Pasquale Fernando Fulvio, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

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Glossary

Adatoms Atoms that lie on a material's surface, as opposed to a surface vacancy defect.

Case hardening Process of hardening the surface of a material while keeping the core soft. This is done by implanting atoms that occupy interstitial sites. Also known as surface hardening.

CMOS Complementary metal oxide semiconductor. It is a metal-oxide-semiconductor field-effect transistor (MOSFET) fabrication process. It consists of complementary and symmetrical pairs of p-type and n-type MOSFETs for logic functions. These are connected to a common secondary voltage such that the pair operates in opposite (complementary) fashion. Thus, when one semiconductor is turned on, the other is turned off, and vice versa.

CVD Chemical vapor deposition. This is a vacuum deposition method used most often by the semiconductor industry to produce high quality, high-performance, thin films. The process makes use of compounds that evaporate when heated under vacuum or react from a gas mixture that flows over a solid substrate film that is heated. Evaporated compounds or gas mixtures decompose or react on the substrate surface forming a ceramic or metal thin film. The process is also adapted for growing organic compound thin films and polymer thin films. The film thickness can vary from a single-atom layer to micrometer thick films. Alternating layers of different chemical composition can be grown into superlattices.

DLC Diamond-like carbon. It is an amorphous carbon material with mostly sp^3 bonding and tetrahedral symmetry that exhibits many of the desirable properties of the diamond material such as hardness, wear resistance, and electrical resistivity.

EBL Electron-beam lithography, the direct writing lithographic process that uses a focused, high-energy beam of electrons (e-beam) to form patterns by material modification, deposition, or removal.

Electronegativity Electronegativity is a measure of the tendency of an atom to attract a bonding pair of electrons. The Pauling scale is the most commonly used. This property is affected by the atomic number of the atom and its radius.

EUV Extreme ultraviolet radiation. It consists of the portion of the electromagnetic spectrum with wavelengths in the range of 10 nanometers (nm) to 124 nm, therefore having photons with energies of 124 electron-volts (eV) down to 10 eV, respectively. EUV is used in the photolithography process.

FTIR Fourier-transform infrared spectroscopy. This type of spectroscopy excites the vibrational motion and sometimes rotational-vibrational motion of molecules in a gas, liquid and solid phase by means of an infrared radiation. The infrared region of the electromagnetic spectrum comprises the wavelengths of 700 nm to 1 mm. Part of the infrared radiation is transmitted by molecules, and another part is absorbed at specific wavelengths. The absorbed infrared is characteristic of particular chemical bonds between atoms forming a polar compound. Rather than exciting a single wavelength, this technique uses a Michelson interferometer, and all the infrared waves reflected back to the detector are analyzed by Fourier Transform. This allows for all infrared wavelengths to be scanned and computer analyzed within seconds and in a single measurement.

Graphene Allotropic form of carbon that consists of a single layer of carbon atoms arranged in a two-dimensional honeycomb lattice.

HDPE High-density polyethylene. In contrast to LDPE, the HDPE has high degree of crosslinking and is extruded using low pressure in the range of 6–7 atmospheres (atm) and low temperature in the range of 333–343 K.

IBAD Ion beam assisted deposition is a technique that combines ion implantation with simultaneous sputtering or another physical vapor deposition technique. It provides independent control of parameters, i.e., ion energy, temperature and arrival rate of atomic species during deposition. This technique is especially useful for thin films as it creates a gradual transition between the substrate material and the deposited film, resulting in films with a much more durable bond to the substrate.

LDPE Low-density polyethylene. This form of PE has low degree of crosslinking and it is extruded using pressures of 1000–5000 atm and a high temperature of 520 K.

Magnetron sputtering A high-rate vacuum coating technique that uses a specially formed magnetic field applied to a diode sputtering target.

MTJ Magnetic tunnel junction. It is a type of spintronic device that consists of two ferromagnetic layers separated by a thin insulating film. When the insulating film thickness is the order of a few nanometers, electrons can tunnel from one ferromagnet into the other. For magnetic layers having opposite large spin polarizations, large resistance changes occur through the junction due with tunneling of electrons with opposite spin polarization taking place. Consequently, the moments of the magnetic layers can be switched from a parallel to an anti-parallel alignment.

Photolithography This is a process used in microfabrication or formation of micropatterns on a thin film or the bulk of a substrate. It uses light to transfer a geometric pattern from a photomask to a photosensitive chemical photoresist on the substrate. The dimensions of the patterns match the wavelength of the light used, or that of the photomask.

Photoresist Light-sensitive material.

PII Plasma ion implantation, also known as pulsed-plasma doping. This is a technique for surface modification. Accelerated ions from a plasma are extracted by an externally applied high voltage and targeting them into a target substrate or electrode with a sample wafer placed over it. The high energy ions are, thus, implanted into the sample wafer.

Radiolysis Dissociation of molecules by ionizing radiation.

Sputtering Ejection of surface atoms and microscopic particles from a material upon bombardment with high energy particles from a plasma or gas.

Steric hindrance Slowing of a chemical reaction due to steric bulk and that relates to the spatial arrangement of atoms in molecules. The steric bulk is the relative volume of chemical functional groups neighboring the reactive site of a molecule or polymer and that are connected along its principal chain of carbon atoms.

SWNT Single-walled carbon nanotube. This carbon allotrope can be idealized as consisting of a single graphene sheet cutout that is rolled into a tube form along one of the Bravais lattice directions of the hexagonal system to form a hollow tube. The tube diameter, length and chirality affect the nanotube properties.

Thermoplastic, thermoset and thermo-responsive polymer Thermoplastic polymers soften when heated and harden when cooled reversibly, whereas thermoset polymers harden irreversibly upon heating as a consequence of crosslinking. Thermo-responsive polymers are a class of stimuli-responsive materials that exhibit conformational changes in response to temperature stimuli, especially near body temperature. These have unique sol-gel transition properties above a certain temperature.

UHMWPE Ultra-high molecular weight polyethylene is a subset of the thermoplastic polyethylene. This material is also known as high-modulus polyethylene (HMPE). It has extremely long chains, with a molecular mass usually between 3.5 and 7.5 MDa (1 Da = 1 amu or atomic mass unit; MDa = million Da).

YSZ Yttria stabilized zirconia is a ceramic material of zirconium oxide (ZrO_2) cubic crystal phase, also known as zirconia. The cubic crystal phase is only stable at temperature above 2370 °C. It then converts to the tetragonal phase below this temperature, and finally to the monoclinic phase below 1173 °C. The tetragonal to monoclinic transition is accompanied by a 5% volume change of the crystal lattice and it is stabilized by doping with yttrium oxide (Y_2O_3), or yttria. In order to stabilize the cubic crystal lattice, yttria is added since Y^{3+} has an ionic radius that is 17% larger than that of Zr^{4+} . Other materials associated with YSZ are calcia (CaO), magnesia (MgO), ceria (CeO_2), and alumina (Al_2O_3) stabilized zirconias.

Introduction

The irradiation of materials offers the possibility to tailor structures at wavelengths far below those of visible and ultraviolet radiation (Trautmann, 1999). The wavelengths of high energy electrons, neutrons, X-rays and gamma photons can match interatomic distances in solids. This property allows for the removal of single atoms from a crystalline lattice position, breaking and forming new chemical bonds, and consequently generating defects, and even nanosized patterns (Nayfeh, 2018).

Radiation is of major importance for developing commercial polymer materials. Polymers are composed primarily of chains of carbon atoms that are bonded to hydrogen. They may also contain functional groups of non-metal elements such as oxygen, nitrogen, silicon, sulfur, phosphorus, fluorine and chlorine. Polymers are classified as low temperature materials (Rösler et al., 2007; Li et al., 2012).

Electrons and gamma photons have enough energy to break covalent chemical bonds between carbon atoms or between carbon with hydrogen. Breaking these bonds leads to carbon atoms having an unpaired electron, a chemical species known as a free-radical. The rupture of a bond between two carbon atoms is a process known as *chain scission*. The generated radicals are highly reactive in nature and can react to form new chemical bonds. When carbon radicals react with other radical or ion species, such as oxygen and nitrogen present in the atmosphere, the reaction leads to shorter polymer chains and the formation of new chemical functional groups.

When carbon radicals form without breaking the polymer chain, the carbon radicals in one polymer chain can quickly form new bonds with other carbon atoms from neighboring polymer chains. This process is known as *crosslinking*, and it is responsible for the increase in molecular weight of polymers and their viscosity, mechanical and thermal stability (Farhataziz and Rodgers, 1987; Cleland et al., 2003). Crosslinking is also of major importance for the manufacturing of elastomers, tires, heat shrink plastics, to name a few. Chain scission and crosslinking are both used for the manufacturing of porous track-etched membranes, for the recycling of plastics, and for the blending of otherwise immiscible polymer phases. Both processes also find application in processing wood pulp into viscose fibers for clothing, and for manufacturing of wood-plastic products that are water proof (Clough, 2001; Cleland et al., 2003).

Metals and ceramics are of major technological importance and are classified as high temperature materials because they both have melting points greater than 500 °C. Metals are often formed by a single element or by combinations of different metal elements forming alloys and intermetallic compounds. In metals, lattice positions are occupied by individual atoms. The relative atomic position of each atoms with respect to the others is dictated by the atomic radius. This packing is the most efficient possible for a given metal. Typical lattice defects include vacancies, surface adatoms, and dislocation lines (Rösler et al., 2007). Irradiation of metals displaces atoms from their positions and atoms diffuse to occupy vacancies. To cause the displacement of atoms, the energy of the incident particles must exceed the binding energy of the atoms in the target (Zinkle and Matsukawa, 2004; Azevedo, 2011). Roughly 90% of all dislocated atoms diffuse to occupy vacancies left by other atoms in the lattice (Zinkle and Matsukawa, 2004). Defects are engineered in this way.

Doping of metals and ceramics is another important area of study. This technique is based on the use of ions in a plasma that are accelerated within an electric field and directed toward target metals. These accelerated ions penetrate the crystal lattice of the target material and diffuse to occupy specific positions. Some commercial examples include the use of nitrogen to harden steels, a process known as case hardening (Ensinger, 2006).

Ceramic materials have at least one element from the main group of metals, or a transition metal element paired with a non-metal element. These include metal oxides and sulfides, carbides and nitrides. Differences in electronegativity between the elements dictate if the ceramics will be ionic or covalent. Atoms in covalent ceramics share pairs of valence electrons, whereas the forces between charged elements are primarily electrostatic in ionic ceramics. Ionic ceramics assume crystal structures that most effectively balance the repulsive (between ions of the same charge) and attractive electrostatic forces (between ions of opposed charges). Ceramic materials also include those formed between a semimetal and a non-metal and these are often covalent. The spatial arrangement of atoms is determined by the elements forming the bond and number of bonds (Rösler et al., 2007). Carbon materials, i.e., diamond-like carbon (DLC) (Delplancke-Ogletree, 2006), graphene, graphite, fullerene and carbon nanotubes (Geim and Novoselov, 2007) are often treated as ceramics although these have only carbon atoms in the structure (Rösler et al., 2007).

Most commercial developments and ongoing research efforts are focused on high energy electrons for designing ceramics with unique properties. When high energy electrons hit a solid surface, only a fraction of energy will be transferred to a target atom. The transferred energy often matches the binding energies of atoms in solids, resulting in ionization of atoms and breaking of chemical bonds (Krashennnikov and Banhart, 2007). Besides allowing for the selective etching of surface atoms and formation of grids and patterns (Nayfeh, 2018), the removal of ions or of neutral atoms from an ionic or covalent ceramic material leads to a vacancy that is associated to a charge imbalance. This is what occurs in yttria stabilized zirconia (YSZ), an important industrial ceramic. These imbalances are often used for increased oxygen ion-conduction in YSZ, making these more active catalysts, electrodes for high temperature fuel cells and oxygen sensors (Dow et al., 1996; Resini et al., 2008).

Moreover, nuclear reactions can take place and lead to the transmutation of individual atoms in a crystal lattice. An example is that of silicon (Si), an intrinsic semiconducting ceramic. The ^{30}Si isotope in Si wafers transforms into phosphorus (P), yielding P-doped Si. (Janus and Malmros, 1976). This localized transmutation can take place at positions within the bulk of the Si wafer. This doping changes the electrical properties of Si and make it suitable for uses in energy storage and conversion devices such as solar cells and Lithium-ion batteries (Haas and Schnöller, 1976; von Ammon, 1992; Ozanam and Rosso, 2016).

These examples are presented in this article, along with some of the relevant theoretical background to enable an understanding of these processes. This article begins by covering polymers that are classified as low-temperature materials. It then proceeds to metals ceramics, composites, and nanomaterials. The use of radiation for the manufacturing of semiconductors, and for the quality control of sheet products and packaging industry are also covered. The use of irradiation for food, pharmaceutical and cosmetics industries are briefly mentioned, as these are covered in detail in other chapters. Some specific techniques and instruments used for gamma, heavy ions and focused electron beams are also described where appropriate.

Polymers (low temperature materials)

The number of new products that come into general commerce on an almost daily basis owe a good deal of their creation and development to radiation processes. Several of these products are highlighted in bold print, to help the reader focus on topics of most personal interest.

Hydrocarbon-based materials are classified as low temperature materials and are often referred to as soft materials. These include low molecular weight compounds, such as most liquid crystals, low and high-molecular weight polymers and resins, and biological tissues. These compounds undergo large deformation and various morphological changes in response to environmental stimuli (mechanical forces, temperature, humidity, pH, electric fields and van der Waals interactions). Polymers are selected for low temperature applications, being limited to temperatures below melting or decomposition (Li et al., 2012). These temperatures are often less than 500 °C.

Polymers, polymer hydrogels and polymer-matrix composites are the most important groups of commercial materials. The radiation dose (Dewji et al., 2021) of 10 kGy (1 Mrad) will increase the temperature of a polymer by 7 °C, and only $\sim 10^{-5}$ fraction of all chemical bonds will be broken. This is enough to reduce the average molecular weight of polymers from 10^6 down to 0.3×10^6 Da, leading to a drastic change in mechanical strength (Farhataziz and Rodgers, 1987). The polymer chain structure dictates whether it crosslinks (forming covalent bonds between neighboring polymer chains) or scissions (where the covalent bond between two carbon atoms is broken along the linear segment of the polymer chain). Polymers with sterically hindered chains undergo scission, whereas those with exposed carbon atoms along the main chain undergo crosslinking (Farhataziz and Rodgers, 1987). The fraction of broken bonds and the reactions that follow constitute the basis of radiation processing of polymers. Table 1 lists polymers that undergo crosslinking or chain scission due to steric hindrance (Cleland et al., 2003).

The specific energy (SE) required for polymers to undergo chemical reactions in units of kJ/kg is equal to the absorbed dose (D) in kGy (kiloGrays)

$$SE = N_A \times (100/G) \times q_0 = 6.022 \times 10^{23} \times (100/G) \times (1.602 \times 10^{-19}/10^3)$$

$$SE = 9.65 \times 10^6/G, [\text{kJ/mol}]$$

where N_A is Avogadro's number and q_0 the elemental electrostatic charge. The G-value is defined as the number of events (cross-linking or chain scission reactions) per 100 eV of absorbed energy. An expression for dose is obtained (Cleland et al., 2003) as a function of $10^3/\text{MW}$, in units of mol/kg, and where MW (g/mol) is the polymer molecular weight:

$$D = 9.65 \times 10^6/(\text{MW} \times G), [\text{kGy}]$$

Table 2 lists G values for most common polymers (Cleland et al., 2003).

Often times, the scission of main chain chemical bonds leads to the formation of intermediate reactive species that induce cross-linking of chains. For 1 in every 10^5 specific bonds that are broken by irradiation, the viscoelastic, mechanical and thermal properties of polymers can change drastically (Farhataziz and Rodgers, 1987). This is observed for the crosslinking of waste polyethylene/nitrile butadiene rubber (PE/NBR) with a gamma-radiation dose of 50 kGy, for which the thermal stability improves by means of covalently binding both polymers (Aly, 2016). For processing, the irradiation of Teflon reduces the particle size and molecular weight, allowing its incorporation into coatings, inks, etc. Also, broadening of the molecular weight distribution of polypropylene (PP) to enhance melt flow characteristics allows for easier molding (Clough, 2001).

Polymer radiation-processing technologies are dedicated to cross-linking of plastics and rubbers, curing of coatings and inks, heat-shrink products, fiber-matrix composites, chain-scission for processing control, surface modification, grafting, hydrogels, sterilization, natural product enhancement, plastics recycling, ceramic precursors, track-etched membranes and photoresist coatings in lithography for microdevice production. Gamma and e-beam sources, as well as ion beams (heavy ions, light ions, highly focused microscopic beams and high-intensity pulses), soft X-rays and e-beams undergo scattering to generate nanosized patterns (Farhataziz and Rodgers, 1987; Clough, 2001; Singh, 2001). Nanomaterials have at least one of their dimensions under 100 nm in size and have their properties (mechanical, electrical, thermal) impacted by size (Pokropivny et al., 2007).

Extreme ultra-violet (EUV) photons have wavelengths greater than 10 nm and do not allow for precise manipulation of materials approaching the molecular and atomic levels. Consequently, methods such as EUV lithography and photolithography, are limited by wavelength for nanomaterials design and nanopatterning. This limitation is overcome by gamma, X-rays, heavy ions

Table 1 Common commercial polymers and radiation reaction characteristic.

<i>Mainly crosslinking</i>	<i>Mainly scission</i>
Polyethylene	Polyisobutylene
Polyacrylates	Polymethacrylates
Polyvinyl chloride	Polymethylstyrene
Polysiloxanes	Polymethacrylamides
Polyamides	Polyvinylidene chloride
Polystyrene	Polytetrafluoroethylene
Polyacrylamides	Polypropylene ether
Ethylene vinylacetate	Cellulose

From Cleland MR, Parks LA and Cheng S (2003) Applications for radiation processing of materials. *Nuclear Instruments and Methods in Physics Research Section B: Beam Interactions with Materials and Atoms* 208: 66–73.

Table 2 Commercial polymers and their corresponding *G*-value for crosslinking and chain scission reactions.

Polymeric material	Crosslinking <i>G</i> (X)	Scission <i>G</i> (S)	Ratio <i>G</i> (S)/ <i>G</i> (X)
Low density polyethylene	0.8–1.1	0.4–0.5	0.47
High density polyethylene	0.5–1.1	0.4–0.5	0.56
Polyvinylidene fluoride	1.00	0.30	0.30
Polymethylmethacrylate	0.5	0.77	1.54
Polymethylacrylate	0.5	0.04	0.07
Nylon 6	0.67	0.68	1.01
Nylon 6,6	0.50	0.70	1.40
Polyvinylacetate	0.30	0.07	0.23
Atactic polypropylene	0.27	0.22	0.81
Isotactic polypropylene	0.16	0.24	1.50
Polystyrene	0.019–0.051	0.0094–0.019	0.41
Natural rubber	1.05	0.1–0.2	0.14
Polybutadiene	5.3	0.53	0.10
Polytetrafluoroethylene	0.1–0.3	3.0–5.0	20
Polyisobutylene	0.5	5	10
Cellulose	(low)	11	(high)

From Cleland MR, Parks LA and Cheng S (2003) Applications for radiation processing of materials. Nuclear Instruments and Methods in Physics Research Section B: Beam Interactions with Materials and Atoms 208: 66–73.

and e-beams (Trautmann, 1999). Radiation emitting instruments exist to control patterning at the nanometer scale. Selected methods are described in following sections, namely e-beam lithography and ion-implantation based on sputtering and magnetron sputtering. This process is also commonly used for metals and ceramics.

Crosslinking and chain scission

Crosslinking

Crosslinking by radiation is a widely used technology in industry for producing crosslinked wire insulation, heat-shrink products such as food wrap and tubing for electrical connections, and self-resetting polymer-based electrical fuses and self-regulating heat tape. Curing of coatings and inks is also an established use of radiation in the production of tubing, pipes and automobile tires, as well as fiber–matrix composites (Ueno, 1990; Gehring and Zyball, 1995; Clough, 2001). These include polyethylene terephthalate (PET) and Nylon (Ueno, 1990), polyethylene (PE) containing blends (Singh, 2001) and fluorinated polymers (Lyons, 1995). Crosslinked polymers maintain their shape, and have increased thermochemical stability, tensile and flexural strength, and better resistance to stress cracking and to creep (Nielsen, 1969).

In his seminal work, Charlesby observed that polyethylene formed an insoluble gel when exposed to reactor radiation. This contradicted the thought that crosslinking was only possible through the addition of chemical agents to polymers having suitable functional groups (Charlesby, 2016). A primary example of the latter is the vulcanization of rubber that proceeds through sulfur addition to double bonds along the principal chain of rubber. In contrast to the energy intensive thermal vulcanization, a minimum gamma radiation dose of 12 kGy is sufficient to vulcanize natural rubber latex in the presence of *n*-butyl acrylate (*n*-BA) (Haque et al., 1996). This offers substantial environmental and economic advantages over thermal and chemical processing of polymers and elastomers. Commercially used elastomers that are crosslinked using radiation include silicone (Basfar, 1997), styrene-butadiene block copolymers (Dakin, 1995), ethylene propylene diene monomer (EPDM) (Banik and Bhowmick, 2000; Nuinu et al., 2012), polybutylene terephthalate (Gehring, 2000), nitrile rubber, latex (Minoura and Asao, 1961), and polymethylmethacrylate/acrylonitrile-butadiene rubber (PMMA/ABN) blends (Cangialosi et al., 2001).

Crosslinking of polymeric foams include gamma irradiated polyethylene (Cardoso et al., 1998) and polypropylene blends (Tokuda and Kemmotsu, 1995), and e-beam (doses of 20–200 kGy) irradiated natural rubber-thermoplastic blends (Ghazali et al., 1999) and polyvinylchloride (PVC) (Sharma et al., 1995), used for heat insulation, automotive cushions, sporting goods and buoyance devices.

E-beam processing has been found to facilitate the blending of phase-segregated polymer phases into copolymers via crosslinking, as in between ethylene propylene/polyethylene (EP/PE). PE is hard plastic and susceptible to cracks during processing of wire insulation and of elastomeric automotive parts, but blending renders it flexibility and creep resistance (Spenadel, 1979). The processability of PP with EPDM forming PP/EPDM blends (van Gisbergen et al., 1989a) and of polystyrene/low density polyethylene (PS/LDPE) blends (van Gisbergen and Meijer, 1991) using e-beams allows for control over rheological properties. The e-beam induced crosslinking allows for fixation of the blended phases, whereas the chain scission of some non-crosslinked PP or LDPE leads to reduced viscosity and increased flowability for injection molding processing. Resulting polymers exhibit defined morphology for the blends and increased impact strength and elongation than low molecular weight PP with similar viscosity (van Gisbergen et al., 1989a).

Increasing temperature increases thermal vibrations in polymer chains. Flexible polymer chains adopt different configurations and move more freely. The entanglements between linear polymer chains disappear and reform along different points between neighboring chains. The number and lifetime of entanglements depend on temperature. These entanglements act momentarily as crosslinks. The minimum radiation dose required to induce covalent crosslinking between chains, and consequently gelation, depends on the molecular weight average and the viscosity. Increasing molecular weight leads to higher viscosity values, whereas at high temperature, the viscosity drops as a result of the decrease in entanglements between chains (Farhataziz and Rodgers, 1987). This allows crosslinking of fluorinated polymers to take place above their melting points under vacuum or inert atmospheres. When irradiated at ambient conditions, chain scission prevails and oxidation takes place, allowing PTFE to be used as an additive to **ink compositions** (Sun et al., 1994; Oshima et al., 1995; Lappan et al., 2000; Seguchi, 2000). The fact that high-molecular weight PTFE resin does not flow even above its melting point and that it is insoluble in any solvent make its processing difficult. This is facilitated by the use of ionizing radiation. A popular commercial name of this polymer is **Teflon** (DuPont). PTFE is commonly used as a **nonstick coating for cookware, in the manufacture of semiconductors and medical devices, as coatings for bulk chemical containers, eyeglasses, and shaver blades, and as an inert ingredient of pesticides.**

Heat shrink products rely on the ability to crosslink a polymer using e-beams while retaining any crystalline domain during the process, a process that is not often possible by chemical methods. After crosslinking, the polymer is heated above its melting point and stretched to about twice the original length and then cooled in a stretched state. The crosslinking prevents the material from flowing. The elongation is retained until the polymer is again heated, at which point the crosslinks will bring the polymer chains back to the original size (Cook, 1990). This is common for PE used in **food wraps** (Datta et al., 1997) and in **electrical tubing insulation** (Wenyuan et al., 1995).

Curing of inks, abrasion protective and diffusion barrier coatings, and adhesives is also achieved using low energy e-beam (Ohta et al., 1990; Thalacker, 1990; Holl, 1995). The penetration depth of the e-beam depends on dose and monomer polymerization can be monitored using Fourier transform infrared spectroscopy (FTIR), as demonstrated for acrylate monomer polymerization (Patacz et al., 2000). In particular, **oxygen diffusion barriers are of interest to food packaging industry.** Examples include the e-beam curing of methacrylate gelatin (Scherzer, 1997) and the ethylene vinyl alcohol (EVOH) and acrylonitrile that are compatible with LDPE films (Rangwalla and Nablo, 1993). Compared to ultraviolet (UV) curing, e-beam irradiation has advantages including energy savings, curing time, crosslinking of deeply pigmented dyes, and solvent-free systems (Läuppi, 1990). Nonetheless, the curing process can be combined with UV irradiation, demonstrating e-beam versatility toward combination of polymers that crosslink under e-beam or that require UV (Nablo, 1990).

Radiation has been applied to coating wood for **wood-plastic composites (WPC), wood-fiber reinforced plastics, and laminated WPCs** (Clough, 2001). Production of WPCs uses acrylates, styrene, vinyl acetate, and other monomers that crosslink (Khan et al., 1993). These are impregnated into wood that has 80% accessible porosity. The process is then followed by polymerization using gamma or e-beam. Composite materials have increased mechanical properties (Palm et al., 2015), including hardness (Solpan and Güven, 1999), tensile (Garnett and Ng, 1996) and flexural strength (Palm et al., 2016). Commercial products include **flooring and table tops, and laminated wood panels.**

Chain scission

The opposite of cross-linking is chain scission, which is the degradation of the main chain of a polymer via breaking covalent bonds. Many polymers undergo main-chain scission and experience degradation of mechanical properties. Side chain scission has only a minor impact in mechanical properties of polymers (Farhataziz and Rodgers, 1987). Chain scission may also be referred to as degradation, since polymer chains undergo fragmentation or decrease in the degree of polymerization (Clough, 2001; Cleland et al., 2003).

Polymers such as PP are irradiated with e-beams for inducing chain scission during its processing into fibers. Pristine PP has low melt strength, and irradiation leads to chain scission and long-chain polymer branching (Lugão et al., 2000). Chain scission prevails and yields double bonds between carbon atoms. The double bonds contain carbon atoms with sp^2 electronic hybridization. The latter increases electronic density over the C atoms, which are also less hindered due to the planar configuration adopted by the C atoms, as compared to C atoms forming single bonds and having tetrahedral configuration from the sp^3 hybridization (Ouellette and Rawn, 2018). These react and result in branching of chains. This process is also known as “viz-breaking” and it results in a melt with increased strength and processability (Clough, 2001). The improved rheological properties of the irradiated PP melt allow for melt mold and for fiber processing options (Lugão et al., 2000; Cleland et al., 2003).

E-beam irradiation has been used to induce the scission process to partially break cellulose from wood pulp into **viscose**. The viscose is then processed into **staple fiber, filament, cord, film, packaging, and non-edible sausage casings.** These materials are widely used in the **clothing, drapery, hygiene, automobile, food, and packaging industries.** E-beam irradiation eliminates hazardous chemicals and thermal energy required by conventional processing of wood pulp (Stepanik et al., 2000).

Functional polymers

The presence of air can lead to the oxidation of polymers at scission and introducing new functional groups. Cotton-cellulose is oxidized during fragmentation using gamma-irradiation with 15–100 kGy dose. The degree of polymerization decreases from 1200 down to 330, and carbonyl groups appear as a consequence of oxidation. Lower doses yield **fibers with enhanced crease resistance** (Takács et al., 1999).

E-beam irradiation of polytetrafluoroethylene (PTFE), copolymers of tetrafluoroethylene with hexafluoropropylene (FEP), and perfluoropropylvinylether (PFA) result in carbonyl groups forming. These groups quickly convert to carboxylic acid groups from contact with moisture in air (Lunkwitz et al., 2000). These oxidized fluoropolymers have better compatibility with polar compounds, make it easier to blend them, such as using PTFE as **ink additive**, for example.

The inert PE has been blended with other polymers and resins such as polyamide using chain scission and oxidation of PE by gamma-irradiation in the presence of air. The polymers were then blended with polyamide, resulting in materials with increased tensile strength (Valenza et al., 1993). The use of oxidizing conditions and N^+ ion beam also led to PE films having better adhesive properties, and binding with increased strength to **epoxy adhesive films** (Mesyats et al., 1996).

Ion beam irradiation of different polymers has also been reported to improve polymer metallization with Al and Cu. Besides improving interaction with the deposited metals, enhanced polymer crosslinking provides a diffusion barrier to the metals. These composites are of interest for **optoelectronics and microelectronic devices, and for electromagnetic radiation shielding** (Bertrand et al., 1997; Kupfer and Wolf, 2000; Rück, 2000).

Graft copolymerization with different monomers is a method for modifying the surface of polymer chains. Grafting can be promoted by e-beam, gamma and ion irradiation (Garnett, 1979). PTFE has been modified by styrene, acrylic acid, 4-vinylpyridine and *N*-vinylpyrrolidone to make **permelective membranes** (or ion-exchange membranes) having acid and basic surface groups (Chapiro, 1977). Grafting is a method used to modify polymer biocompatibility, make them suitable for artificial organs, soft and hard tissue substitutes, and diagnostic or therapeutic instruments and devices (Hoffman, 1977).

Grafting has also been used to modify PE fibers using gamma-radiation and copolymerized with amidoxime for **extraction of uranyl ions from seawater**. The uptake in batch experiments reached 3.3 mg of Uranium/g of sorbent after 8 weeks of contact with seawater. The expected cost of this process using the sorbent developed by Oak Ridge National Laboratory was of USD610/kg of Uranium (Kim et al., 2014). Advantages of this extraction method include the lower energy input required for passive mining compared to mining uranium ores, and the much greater amount of U present in seawater compared to terrestrial mines (Conca, 2016).

Fiber-matrix composites

Fibers are dispersed in a polymer matrix and processed into sheets that are rolled, stacked and molded for different applications. The matrix plays a protective role and provides a medium in which the fibers are oriented as to obtain isotropic or anisotropic properties. The strength of fiber-matrix composites depends on fiber length, relative orientation and in the strength of the interaction between fiber and polymer matrix.

Examples of e-beam crosslinking of fiber-matrix composites include epoxy resins with carbon fibers (Lopata et al., 1999) and with lignocellulosic fibers (Güven et al., 2016). The carbon and wood fiber-matrix composites find **structural applications for aerospace** (Saunders et al., 2000), **maritime** (Chappas et al., 1999), **automotive parts, and sporting goods** for their lightweight and high mechanical strength (Singh et al., 1996; Güven et al., 2016). Polymer coatings also make cellulosic fibers less hydrophilic, thus extending the lifetime use of these environmentally friendly and inexpensive composites in **furniture and laminated wood panels** (Güven et al., 2016). Overall, e-beam irradiation of composites provides reduced ambient temperature curing, reduced curing times, reduced volatile emissions, better material handling, and reduced costs over conventional thermal processing of composites (Singh et al., 1996; Chappas et al., 1999).

Track-etched membranes

Polymer track-etched membranes are porous polymer films formed from irradiation with high energy ions and subsequent chemical etching of damaged polymer regions by means of oxidizing agents. Track-etched membranes are commercially available and manufactured by companies such as Nuclepore™, Oxyphen GMBH, and Sterlitech Corporation.

Porous membranes having pore widths below 100 nm and in micron size range find many industrial applications including **purification and separation of gases and liquids, water filtration and desalination, separations of petrochemical products, processing of food and beverages, purification of biological and pharmaceutical compounds**. Control over the pore dimensions is of major importance for separations, as these are often dictated by size exclusion, as illustrated in Fig. 1A, and by hydrophilic properties of the pores (Virk and Kaur, 1998; Reber et al., 1999; Shumskaya et al., 2019). Pore size, shape and surface chemistry is also of major importance for the development of electrochemical biosensors. While selective binding of target elements, chemical compounds, DNA, proteins and other biomolecules depend on surface functional groups present in the membrane electrodes, the electrolyte accessibility and diffusion resistivity that is measured by electrochemical methods are a direct consequence of the membrane pore size and shape (Pérez-Mitta et al., 2017; Kaya and Keçeci, 2020).

Most chemical processes require multiple steps and the use of sacrificial templates to form pores, or use of polymer binders for inorganic porous materials prepared in multiple steps (Yang et al., 2011). Thus, track-etched membranes present several advantages over traditional membranes including the elimination of sacrificial templates, reduced number of steps to prepare, and a broad range of polymer products that can be irradiated. Moreover, the density of pores in these track-etched membranes can be regulated by the ion fluence, with higher fluences leading to greater densities of pores per unit area (Kaya and Keçeci, 2020).

For instance, irradiation of thin polymer films such as PET with heavy ions, i.e., Ar^+ , Kr^+ , Xe^+ , Au^+ and U^+ , and often having energies approaching 12 MeV, create micron and nanosized pores with uniform geometry. The pores form from extended chain

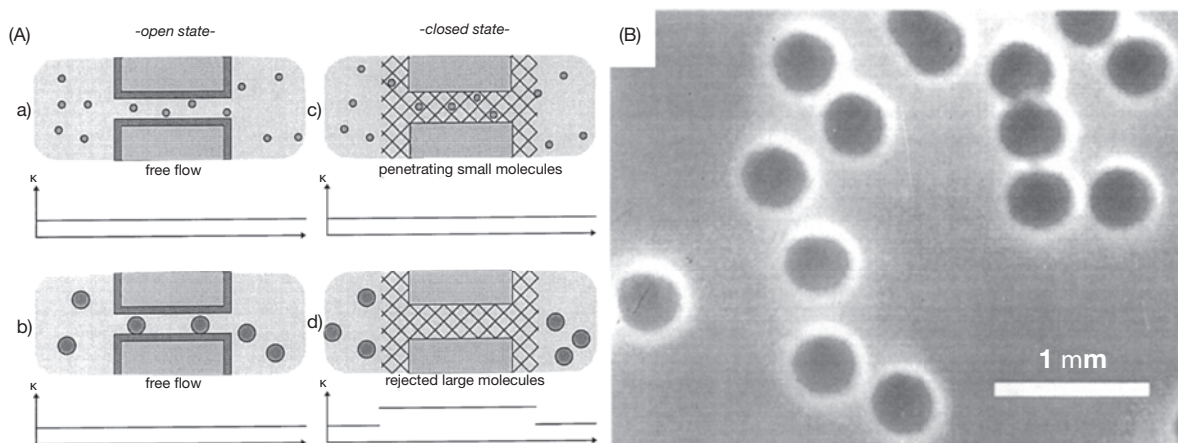


Fig. 1 (A) Size exclusion of molecules in the open (a, b) and closed (c, d) state of a responsive pore. In the closed state, only small pores formed between entangled polymer chains remain open and are accessible to small ions and molecules; larger biomolecules and polymer encapsulated compounds are excluded by size. In the open state, the polymer walls become denser, and larger pores form. This type of smart polymer material prepared using irradiation methods has been reported for both track-etched membranes and hydrogels sharing similar composition. (B) These are cylindrical pores in a polyimide track-etched membrane. From (A) Reber N, Omichi H, Spohr R, Wolf A, and Yoshida M (1999) Closure characteristic of thermally responsive ion track membranes. *Nuclear Instruments and Methods in Physics Research Section B: Beam Interactions with Materials and Atoms* 151 (1): 146–153. (B) Trautmann C (1999) Modifications induced by swift heavy ions. *Bulletin of Materials Science* 22(3): 679–686.

scission and oxidation reactions from chemical agents (NaOH solution) along the ion path. Competing crosslinking reactions help limit the scission and oxidation reactions and define the pores size at ranges up to 100 nm (Apel et al., 1998).

The scanning electron microscopy (SEM) image in Fig. 1B depicts a polymer track-etched membrane film cast on polyamide (Trautmann, 1999). Similarly, other polymers such as polyimide (PI) and poly(ethylene naphthalate) (PEN) with pores of 100 nm and less were prepared after film stacks were irradiated with Kr^+ , Xe^+ and Bi^+ ions with energy in the order of 10 MeV and chemical etching.

Regarding geometry, the pores of the track-etched membranes can be cylindrical, conical, bullet-like, hourglass (double-conical), and cigar-shaped, with the shape being controlled by the chemical etching method (Kaya and Keçeci, 2020). The cylindrical shape, for example, is evidenced by the shape of nanowires grown by electrodeposition within the nanopores of bisphenol-A polycarbonate (BPC) irradiated with Ar^{9+} ions (5 MeV/amu of energy) and etched using NaOH and a surfactant solution. After the BPC films were dissolved in dichloromethane, the nanowires had rod-like shape and their diameters were comparable to the widths of the BPC channels, 20–100 nm. As with other track-etched membranes, the pore widths were controlled by irradiation time (Ferain and Legras, 1997).

Membranes with ion channels mimicking those found in biological systems are of interest to the development of ionic sensors (Pasternak et al., 1995; Pérez-Mitta et al., 2017). This has been demonstrated for narrow pore polyethylene terephthalate (PETP) membranes. After etching to open the membrane pores, polymethylvinylpyridine (PMVP) was subsequently grafted on the pore surfaces to develop a functional material. The properties included rapid switching, selectivity between cations and anions, and modulation by H^+ and divalent Zn^{2+} and Ca^{2+} cations.

Track-etched membranes that are thermo-responsive are classified as smart-materials and are of major interest for encapsulating drugs for targeted therapies and for tissue engineering (Sponchioni et al., 2019). Above a threshold temperature, these materials lose water and consequently the polymer walls shrink. In the shrunk state, the pores open up. Below this threshold temperature, the polymer re-hydrates, thus swelling the polymer walls and closing the pores, as illustrated in Fig. 1A. An example of applications for these polymers in drug-delivery are cancer therapies. In these, encapsulated drugs used for killing cancer cells are absorbed exclusively by cancer cells. Since temperature and pH conditions in the blood plasma and inside cells differ, the drugs are then released inside the cancer cells, and healthy cells are preserved (Sponchioni et al., 2019).

In existing chemotherapy, these drugs are directly injected in the blood stream and circulate in the entire body, killing both cancer and healthy cells. Given the pore dimensions of track-etched membranes as poly-diethylene glycol-bis-allyl carbonate (CR-39) and a thermo-responsive gel acryloyl-L-proline methyl ester (A-ProOMe) (Yoshida et al., 1997), and of poly-N-isopropylacrylamide (Reber et al., 1999) that can reach 0.3 and 3.5 μm , respectively, small drugs, polyethylene glycol (PEG) encapsulated drugs, and even proteins can be safely delivered to target cells and organs. This is achieved with small temperature changes in a very narrow range around 30 °C (Yoshida et al., 1997; Reber et al., 1999).

Hydrogels

Hydrogels are polymer materials that have the ability to swell in water and retain 10–1000 times of their original weight or volume of water and water-based fluids within their structure. Given the highly crosslinked polymer network forming the

hydrogels, these materials do not dissolve in water. Hydrogels can be of natural or synthetic origin. Similar to other polymers, the hydrogel network can be formed by radiation induced crosslinking. Some of the most commonly used polymers forming hydrogels include polyvinyl alcohol (PVA), polyacrylamide (PAAm), polyvinyl-pyrrolidone (PVP), polyethylene oxide (PEO) and methyl cellulose (MC). Hydrogel technologies may be applied to **hygienic products, diapers, contact lenses, agriculture, drug delivery systems, sealing, coal dewatering, artificial snow, food additives and packaging, pharmaceuticals, tissue engineering and regenerative medicines, diagnostics, wound dressing, separation of biomolecules or cells and barrier materials to regulate biological adhesions, and biosensors** (Laftah et al., 2011; Peppas et al., 2012; Ahmed, 2015; Ullah et al., 2015).

Gamma and e-beam irradiation have been used to initiate the polymerization in aqueous solution for preparing hydrogels of unsaturated compounds. The irradiation results in the formation of radicals on the polymer chains. The parallel process is the radiolysis of water molecules which yield hydroxyl radicals that also attack the polymer chains resulting in the formation of macro-radicals. Recombination of the macro-radicals on different chains leads to forming of covalent bonds and finally a cross-linked structure (Farhatziz and Rodgers, 1987).

Examples of polymers crosslinked by gamma irradiation include the pH-sensitive PEO hydrogels grafted with acrylic acid (AAc) (Nho et al., 2004), copolymers of polyethylene glycol methacrylate and acrylic acid (Park et al., 2004a) and PVA networks grafted with AAc or methacrylic acid (Park et al., 2004b), the thermo-responsive copolymers of 2-hydroxyethyl methacrylate (HEMA) and itaconic acid (IA) (Tomić et al., 2007), and the superabsorbent nanocomposite of poly[acrylamide-co-(acrylic acid)]/montmorillonite (Luo et al., 2005). The major advantage of the radiation initiation over the chemical initiation is the production of relatively pure hydrogels and that are free of chemical radical polymerization initiators (Laftah et al., 2011; Ahmed, 2015; Ullah et al., 2015).

Plastic recycling

Plastic waste is a global problem impacting land and oceans. The USA is the world's leading nation in plastic waste generation, with an estimated amount of 42.0Mt produced in 2016. Out of this figure, up to 0.41 Mt were illegally dumped and 0.99Mt was inadequately managed by countries that imported plastic scrap that was previously collected in the United States for recycling (EPA, 2019). In 2010, the U.S. Environmental Protection Agency (EPA) reported that 75.4% of all plastic waste was treated by disposal in landfills, 15.3% was incinerated, and that only 9.3% was recycled (Law et al., 2020).

The most common plastics discarded in the US and considered of high value for recycling are LDPE, high density polyethylene (HDPE), mixed PE/PP, PP, and PET. Plastics deemed to be of relatively low value include PS (polystyrene), PVC (polyvinyl chloride), PC (polycarbonate), PLA (polylactic acid)/bio (Clough, 2001; Law et al., 2020).

A major challenge with plastic recycling includes the incompatibility between certain polymers. This means that some polymers will form segregated phases when molded for reprocessing. Chemical compounds known as compatibilizers are used to combined incompatible phases (Ignatyev et al., 2014). Another issue includes the oxidation and degradation of some of the polymers in landfills, all of which change their properties (Clough, 2001; Burillo et al., 2002).

Compared to other methods of plastic reprocessing for recycling, radiation offers many advantages, including lower energy requirements, the avoidance of chemicals and solvents, and the lack of a liquid neutralization waste. As previously discussed, irradiation of polymers can lead to cross-linking between different polymers without additives, or to chain scissions that lead to polymer decomposition. The possibility to partially degrade heavily crosslinked plastics through radiation also facilitates their melting and molding. Radiation also degrades toxic compounds adsorbed by plastics in landfills, which can cause undesirable color changes and odors (Burillo et al., 2002). Maximizing plastic recycling also prevents its incineration, thus avoiding the release of residual ash and of additional carbon emissions (Law et al., 2020).

Numerous examples have been reported of polymer crosslinking, compatibilization and morphology stabilization (overall form of polymer structure such as crystallinity, branching, molecular weight, and crosslinking) for single phase polymers, polymer blends and composites (Clough, 2001; Burillo et al., 2002). For instance, gamma irradiation of recycled PE (RPE) and of virgin PE (VPE) lead to VPE having a higher degree of crosslinking than RPE, when both samples had the same degree of crystallinity (Adem et al., 1998). Blends of LDPE/HDPE (75/25) were irradiated with gamma-doses up to 2000 kGy. Both tensile stress and elastic modules increased as dose increased also as a result of crosslinking (Miguez Suarez et al., 2000).

Radiation crosslinked EP blends with PP have been found suitable for use as **electrical insulators** (Spenadel, 1979). The use of an additional crosslinker monomer, trimethylolpropane trimethacrylate, was required for using EP/PP blends as **fascia in vehicles**. The crosslinked polymer blends had increased mechanical properties as a result of higher degree of crosslinking (Spenadel, 1979). Incompatible LDPE and polyamide-6 (PA-6) were blended after gamma-irradiation of LDPE. Irradiation oxidized LDPE and introduced polar functional groups that made it compatible with PA-6 (Spadaro et al., 1992). Blending both polymers allow for effective recycling into **food packaging materials**.

E-beam irradiation of PP at doses less than 100 kGy was found to cause chain scission. For the blend, the radiation simultaneously induced crosslinks in EPDM and grafting of PP to EPDM, thus stabilizing the morphology of this blend during injection molding (van Gisbergen et al., 1989b). The PP/EPDM blends find uses in **automotive and home appliances, such as airbag covers and housings, interior surfaces and functioning components (bin mats, buttons and knobs, consoles), static seals, gap filler and exterior finishing**.

Sterilization

Radiation sterilization of disposable plastic medical items (syringes, tubing, vials, gauze, etc.) is a growing segment of the modern industry in many industrialized countries (Kalia and Joseph, 2020). The sterilization by means of irradiation eliminates toxic residues found in chemical sterilization. Major challenges relate to the control of undesired property changes. These changes include post-irradiation degradation in mechanical properties, formation of color centers in transparent materials or discoloration in colored ones. Color arises from a combination of trapped radicals and permanent structural changes in the macromolecules which include conjugated chromophores. On the other hand, discoloration by bleaching occurs by reaction of diffusing oxygen with formed radicals. An in-depth discussion, including several examples of sterilization of materials by irradiation, have been described in review articles (Clough, 2001). This topic is well covered in Chapter 2 of this section (Kalia, 2021).

One example is the radiation sterilization of ultra-high molecular weight polyethylene (UHMWPE) materials used for hip joint replacement, for which long-term stability remains an issue. It has been reported that irradiation in air reduces the molecular weight due to chain scission and degradation. The extent of the oxidation has been found to be dependent on the irradiation time and on the absorbed dose. The oxidation was prevented by irradiating the UHMWPE under a nitrogen atmosphere, followed by its storage in an inert atmosphere. In this way, crosslinking of the PE chains prevailed, and the expected ductile material properties of this polymer were maintained (Streicher, 1995).

Another example is the irradiation of film packaging for food and pharmaceuticals. Preventing the formation of fragmented radiolysis products is a major concern, as these can migrate into the packaging contents. For instance, irradiation of a Nylon/Polyvinylidene dichloride (PVDC)/ethylene vinyl acetate (EVA) barrier film caused changes in mass transfers of the packaging material, with the development of off-odors and taints, even at 1 kGy dose. The results were dependent on irradiation source, gamma or beta (Deschènes et al., 1995).

Other packaging materials have also been investigated using gamma-irradiation. These include PS, polybutadiene and some copolymers, and wood pulp paper, both in the presence and in the absence of antioxidants and stabilizers. In the absence of additives, the effect of irradiation of PS is negligible even at high doses. Antioxidants and stabilizers have been found crucial in polybutadiene and butadiene-containing copolymers. Irradiated wood pulp paper sheets lose some of the bound water (Pentimalli et al., 2000).

Metals, ceramics and nanocomposites (high temperature materials)

High temperature materials are a class of materials having high melting points and that operate at temperatures above 540 °C. It includes some stainless steels, superalloys (alloys of iron, cobalt, or nickel), intermetallics (stoichiometric alloys with well-defined and homogenous composition), refractory metals containing nickel, molybdenum, chromium, and silicon, and certain ceramics (i.e., diamond, graphite, carbides, nitrides, and metal oxides).

Scattering of ions and neutrons by nuclei and electrons in solids can displace atoms from their lattice sites and leave metastable vacancies. These vacancies have distorted chemical bonds that rearrange by atom diffusion excited by other incident electrons (Kiritani, 1986). At higher energies, electronic slowing-down of incident particles prevails, and energy crossover between nuclear and electronic slowing-down depends on ion and particle mass. For Ar^+ ions this is 100 keV, and for Xe^+ , this crossover is 1 MeV (Krashennnikov and Banhart, 2007).

To cause the displacement of atoms, the energy of the incident particles must exceed the binding energy of the atoms in the target material crystal lattice. The binding energies vary from 20 to 40 eV. At lower energies below 1 MeV, inelastic scattering is dominant, and at this energy, roughly 25,000–50,000 atoms can be displaced for the lowest to the highest binding energies in metals, respectively (Zinkle and Matsukawa, 2004; Azevedo, 2011). When electrons with 100 keV of energy hit a solid surface, only a fraction of energy, typically 20 eV, will be transferred to a target atom due to conservation of linear momentum. This is at the threshold of displacing a carbon atom in a graphene hexagonal lattice (Krashennnikov and Banhart, 2007). Consequently, ionization of atoms and breaking of bonds between atoms prevail for high energy electrons.

The following discussion covers developments on ceramic materials, case-hardened steels, and nanomaterials as these comprise the bulk of all commercial products manufactured by different irradiation methods.

Nanocomposites

Design of nanopatterns smaller than 50 nm and manipulation of atoms on ceramics and ceramic nanocomposites require the use of high-energy radiation, such as focused electron beams, high-energy ions, and neutrons.

Among the different radiation-based techniques, electron beam (e-beam) lithography (EBL) and plasma ion implantation (PII) have become two of the most widely used for their simplicity and scalability.

EBL is a technique in which an e-beam is focused by means of external electrical or magnetic fields and used to scan a surface coated with an electron-sensitive film known as resist. It is a nanomanufacturing method used to draw fine lines on semiconductor surfaces, to create micro and nanodevices (Tennant and Bleier, 2011; Groves, 2014; Prakash and Yeom, 2014). This technique replaces photons (photolithography) for creating very small patterns and structures on the resist surface (photoresist).

An EBL instrument is schematically drawn in Fig. 2A. The advantages of EBL include the elimination of photomasks that are required when using photons (wavelengths greater than 50 nm). The resolution limit is not limited by diffraction because the wavelength of high energy electrons is very small. The resolution is affected instead by the spot size of the focused beam and by the backscattering of electrons by the substrate and the resist. Hence, EBL allows the drawing of patterns with resolution smaller than 10 nm (Tennant and Bleier, 2011; Groves, 2014). Aberration is corrected by focusing the e-beam with magnetic instead of electric field, further improving the resolution to about 4 nm (Martinez-Chapa et al., 2016; Manfrinato et al., 2017; Nayfeh, 2018).

The nanofabrication process can be combined with other chemical and physical deposition techniques to create heterostructures, as depicted in Fig. 2B for a magnetic tunnel junction (MTJ) (Singh et al., 2019). In this, the fabrication of $\text{Ni}_{80}\text{Fe}_{20}/\text{Co}_{75}\text{Fe}_{25}/\text{Al-O}/\text{Co}_{75}\text{Fe}_{25}/\text{Ta}$ MTJ begins by etching part of the MTJ structure using e-beam lithography and Ar ion milling (Panel A). The exposed substrate was covered with a thick $\text{Al}_2\text{O}_3/\text{Cu}$ film (Panel B) and liftoff in organic solvent. A Pt thin film of thickness 10 nm was deposited by chemical vapor deposition (CVD) on both sides of $\text{Al}_2\text{O}_3/\text{Cu}$ films. Finally, the junction area was defined by Ar ion milling, in which the Pt films were used as etching masks with $100\text{ }\mu\text{m} \times 10\text{ nm}$ (Panel C) (Niizeki et al., 2003).

Examples of resolutions of less than 4 nm in EBL have been reported for aberration-corrected high-resolution imaging in scanning transmission electron microscopy (STEM) instruments, as presented in Fig. 2C for patterning of ZnO substrates with nano-grid patterns. The resist typically consists of PMMA film. One of the advantages of PMMA, is that this polymer reduces the backscattered electrons as compared to hydrogen silsesquioxane, thus contributing to the enhanced resolution (Nayfeh, 2018). Because of the high electrical conductivity, transparency and precise patterning of these ZnO thin films by means of irradiation, these materials are important components of modern photovoltaic cells, thin film transistors, light-emitting diodes (LED), UV sensors, biosensors, gas sensors, and photodetectors (Kumar et al., 2015).

Boron nitride (BN) is an electrical insulator which is isoelectronic to carbon (diamond and graphene-based materials). As in carbon materials, tetragonal and hexagonal BN materials exist, where the hexagonal structure is the most common and stable at ambient conditions (Eichler and Lesniak, 2008).

Of recent interest to the aerospace industry is boron nitride nanotubes (BNNTs), which can be imaged as being formed by rolling a 2D hexagonal sheet of this material, in an analogous way as SWNTs (Golberg et al., 2007, 2010). The ability of B, particularly its isotope ^{10}B , to absorb neutrons, combined with the lightweight BNNTs that have extremely high tensile and compressive strengths, make this material desirable for future human space exploration (Estevez et al., 2014). Thus, knowledge of the radiation stability of BNNTs is crucial for its successful implementation as structural and as radiation shielding material for space flight technology (Jakubinek et al., 2019).

BNNTs have been irradiated with high-energy electrons and the selective elimination of B and N at specific locations has led to the formation of octahedral fullerene structures (Golberg et al., 1998), changes in nanotube diameter and chirality, and even to curled nanotubes (Zobelli et al., 2006).

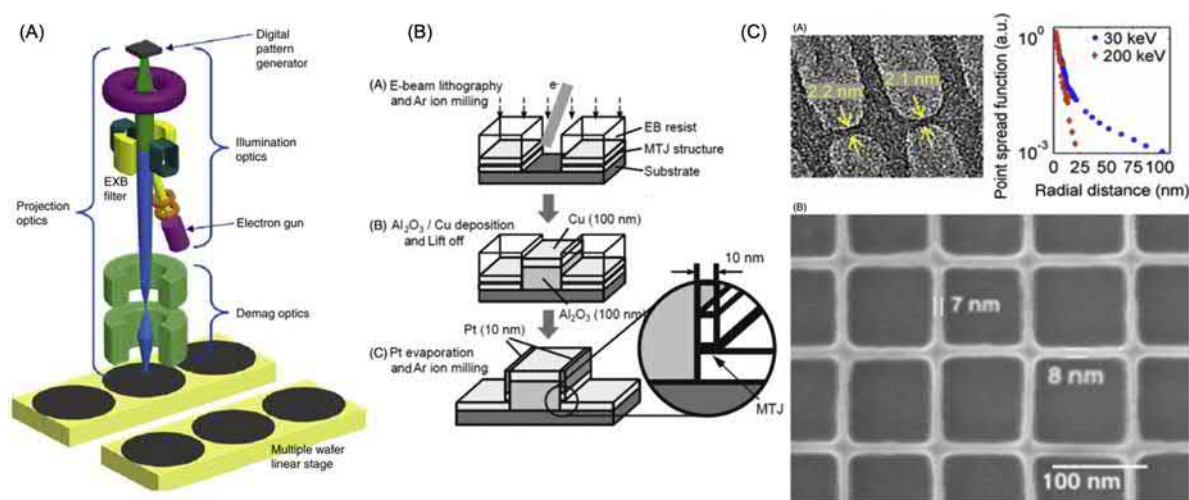


Fig. 2 (A) Schematic of a reflective electron beam lithography system. (B) Schematic illustration of the nanofabrication process. Various steps of this process are depicted in (Panel A), (Panel B), and (Panel C). (C) Aberration-corrected STEM images of 2 nm isolated feature size and 5 nm half-pitch (Panel A) and a grid formed in ZnO demonstrating a sub-10 nm resolution (Panel B). From (A) Martinez-Chapa SO, Salazar A and Madou MJ (2016) Chapter 13.2—Two-photon polymerization as a component of desktop integrated manufacturing platforms. In: Baldacchini, T (ed.) *Three-Dimensional Microfabrication Using Two-Photon Polymerization*, pp. 374-416. Oxford: William Andrew Publishing. (B) Niizeki T, Kubota H, Ando Y and Miyazaki T (2003) Nanofabrication of magnetic tunnel junctions by using side-edge thin film deposition. *Science and Technology of Advanced Materials* 4(4): 347. (C) (2018) Chapter 5—Manipulation and patterning of surfaces (nanolithography). In: Nayfeh M (ed.) *Fundamentals and Applications of Nano Silicon in Plasmonics and Fullerenes*, pp. 89-137. Elsevier.

Ceramics and metals

Yttria stabilized zirconia (YSZ) is an important ion-conducting material and largely used as thermal coatings (Chen, 2006), dental implants (Camposilvan et al., 2018), catalyst supports in automotive catalytic converters (Dow et al., 1996), as well as in high temperature fuel cells for energy conversion (Resini et al., 2008). The doping of zirconia (ZrO_2) with Yttria (Y_2O_3) yields Yttria stabilized Zirconia (YSZ) ceramics. The doping with a trivalent Y^{3+} cation of a ceramic of a tetravalent Zr^{2+} cation leads to anion vacancies in the crystal lattice for charge neutrality. The existence of these O^{2-} vacancies facilitates the diffusion of oxygen anions across the YSZ crystal, which is instrumental for fuel cell operation and for many catalytic reactions to take place. Thermal diffusion of O^{2-} further increases with decreasing grain size down to the nanoscale, making control over doping and crystal size simultaneously relevant for making materials with enhanced properties (Zakaria et al., 2020).

In YSZ irradiated with Kr^+ 400 keV, the concentration of defects was inversely proportional to the grain size of nanoceramics. The short migration distances in nanograined YSZ allowed radiation induced interstitials to reach the grain boundaries on the irradiation time scale, leaving behind only vacancy clusters distributed within the grain. Radiation-induced grain growth was observed as a consequence of the non-uniform ion implantation. The latter caused cracking of the nanosized samples induced by local stresses at the irradiated and non-irradiated grain interfaces (Dey et al., 2015). This knowledge could motivate the development of YSZ ceramics with engineered defects and at temperatures far below those required for thermal diffusion of doping species.

Ion-implantation is a technique for injecting energetic ions into the surface of a solid target to modify or change the near-surface chemical composition of a material. Sputtering, on the other hand, is the ejection of surface atoms and microscopic particles from a material upon bombardment with high energy particles from a plasma or gas. In ion implantation some sputtering occurs. Hence, ion implantation is also used for changing the defect state of or introducing defects to the target material. Ion implantation offers a clean surface treatment because of the little or no contamination of the target sample. Additionally, heat treatment of the sample is not required to induce transformation (Page, 1991). When plasma is used to generate ions, the process is known as PII. As shown schematically in Fig. 3A, an immersion variant of PII indicates that more than one surface of a material can be coated by this process (Gan and Berndt, 2015).

For thin films, ion beam assisted deposition (IBAD) is widely used, and instead of plasma, another physical vapor deposition technique is employed (Cui and Luo, 1999; Martín-Palma and Lakhtakia, 2013). The ions are accelerated between 20 and 200 keV toward the target sample surface. The ion penetration is limited to few nanometers, and a subsequent thermal diffusion step can be implemented for deposition of ions below surface levels (Gan and Berndt, 2015).

The ion energy and the microstructure of films produced by ion implantation are schematically shown in Fig. 3B. Low-energy ions of ~ 0.1 eV form implanted columnar structures and incomplete surface coverage that result in tensile stress. At intermediate energies of 10–100 eV, a dense film structure with stress that is compressive in nature is obtained. A keV energies, significant atomic rearrangement and relaxation of local stress and strain occur, and a mixing layer is formed at the film–substrate interface that improves adhesion (Gan and Berndt, 2015).

The hardness and stress level of ion-implanted ceramic surfaces change in near-surface, with significant consequences for the friction and wear behavior of ceramic surfaces (Ensinger, 2006). Similarly, the tensile and hardness properties of metals can be altered by induced stress, lattice defects, and carbon or nitrogen ion diffusion (case hardening), a process used to manufacture machine parts, tools, special screws, camshafts and of parts for the construction industry, such as girders, panels, doors and other products that have already been shaped in their final form and that require reinforcing (Pelletier and Anders, 2005; Ensinger, 2006).

For N-implanted steel by ion-implantation and subsequent thermal diffusion, the N-depth profile from Auger electron spectroscopy is shown in Fig. 3C for different ion implantation energy. The resulting hardness of the steels are then compared in Fig. 3D,

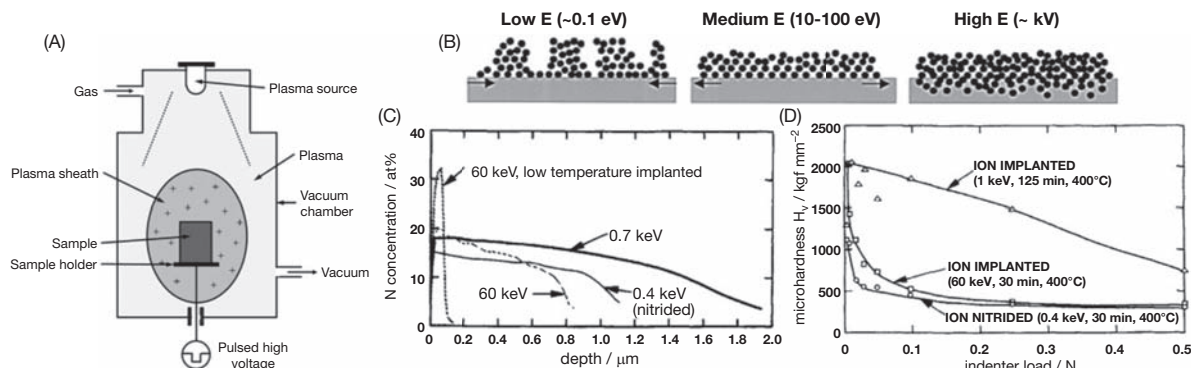


Fig. 3 (A) Schematic of a plasma immersion ion implantation system, and (B) schematics of film microstructure deposited using different ion energy levels. The arrows for the low- and medium-energy processes indicate the direction of strain and the resulting forces on the substrate. (C) Auger depth profiles of nitrogen-implanted and ion-nitrided stainless steel AISI 304 at 400 °C and (D) Vickers hardness of nitrogen-implanted and nitridated stainless steel AISI 304. From (A and B) Gan JA and Berndt CC (2015) 4—Plasma surface modification of metallic biomaterials. In: Wen C (ed.) *Surface Coating and Modification of Metallic Biomaterials*, pp. 103–157. Woodhead Publishing. (C and D) Ensinger W (2006) Chapter 18—Corrosion- and wear-resistant coatings formed by ion beam techniques. In: Pauleau Y (ed.) *Materials Surface Processing by Directed Energy Techniques*, pp. 595–628. Oxford: Elsevier.

showing the impact of ion implantation energy in sub-surface ion diffusion for increased hardness of the **nitridated steel casing** (Ensinger, 2006).

Diamond-like carbon (DLC) and DLC hybrid films have been prepared by arc plasma ion sputtering (Delplancke-Ogletree, 2006). During sputtering, the material from the sputter target is atomized by bombardment with ions or with neutral particles. The atmosphere during sputtering is an important parameter, since atomized material can lose energy by collisions with gas molecules. A negative voltage of typically -300 V or more is applied to the target, attracting positive ions as Ar^+ to the target surface. If the energy of bombarding ions exceeds the binding energy of surface atoms in the target, recoil atoms are created. The surface atoms of the target are atomized and travel to the surface that is to be coated. Almost no restrictions in target materials for coating exist, and these include pure metals, semiconductors and insulators (Adachi and Wasa, 2012).

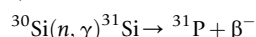
Plasma sputtering deposition of ion-doped DLC films can be carried out in either non-reactive (inert gas only) or reactive (inert and reactive gas mixtures) discharges with single- or multi-elemental targets. **DLC films are used as protective surfaces for prostatic implants, and implantation of ions such as nitrogen, calcium and phosphorus increase their biocompatibility** (Gan and Berndt, 2015).

DLC coatings are also used to protect atomically smooth superlattices and nanocomposites for **magnetic read/write applications** (Delplancke-Ogletree, 2006). Magnetic materials are implanted using magnetron sputtering, a variant of the sputtering technology. In magnetron sputtering, high energy ions (typically Ar^+) bombard and ionize a cathode target surface generated in a glow discharge plasma. Target atoms are ejected from the surface, and ion implantation occurs. Magnets are placed to generate close field loops. The increase in the magnitude of the magnetic fields can increase the plasma density over the target surface, consequently increasing the implantation rates (Coombs, 2012; Exter, 2015).

Semiconductors

Boron doping of DLC leads to **well structures, isolation regions, channel structures, and source and drain regions of Si complementary metal oxide semiconductors (CMOS) transistors**. These are examples of ion implantation for the semiconductor industry (Koizumi et al., 2018).

Another major commercial use of irradiation is that of neutrons from nuclear reactors for doping silicon (Si) with phosphorus (P) (Janus and Malmros, 1976). Si has 23 known isotopes with atomic masses from 22 to 44 amu and the most abundant isotope (92.2%) is the isotope with mass 28 (^{28}Si). Of interest is the isotope ^{30}Si , that has an abundance of 3.1% (Meija et al., 2016). The transmutation reaction involves the capture of neutron and gamma photon yielding ^{31}Si isotope. The latter has a half-life of 2.6 h and it transmutes into ^{31}P by β^- decay process (Haas and Schnöller, 1976). This decay is typical of neutron rich elements, and a neutron (n) is converted into a proton, an electron, and an electron antineutrino. The transmutation reaction is represented as



Silicon is an intrinsic semiconductor and the doping of Si with the electron rich P yields materials with distinct electrical properties. The electrical properties of P-doped Si change with the increasing levels of doping, which is controlled by the upper limit of the ^{30}Si abundance, and by the fluence of neutrons to which the Si wafers are exposed to (Janus and Malmros, 1976). The doped materials are commercially used in **Li-ion batteries and as semiconductors for uses in portable electronic devices** (Haas and Schnöller, 1976; von Ammon, 1992; Ozanam and Rosso, 2016). Advantages of the transmutation process include the energy and time for doping, and the uniform distribution of dopant in the Si lattice as compared to ion implantation or CVD methods that require a second thermal diffusion step to drive the doping elements from the Si surface into the bulk of the crystal (Haas and Schnöller, 1976; von Ammon, 1992).

Metallurgical, glass, paper and plastic industry

Laminated and sheet manufacturing requires strict quality control. Density is the physical property most impacted by the dimensions of laminated and thin-walled products. Online profilometry methods make use of gamma-radiation attenuation to verify the uniformity of the density profile of products. The most conventional method for density determination is the Archimedes method (ASTM B962-17, 2017). However, the existence of open and accessible pores in materials, issues with determining dry and wet densities of materials and their stability is water, make this method difficult to conduct at large industrial scales and with the required levels of precision and accuracy. Densitometry based on gamma-attenuation overcomes these issues, and it is capable of probing for density variations that result from the existence of inaccessible pores, voids and cracks within materials. Gamma radiation sources are ^{137}Cs and ^{141}Am , and a common scintillator detector is sodium iodide (NaI) (Schlieper, 2000, 2001).

The attenuation is proportional to the density of the material (ρ), and the signal intensity related to the mass attenuation coefficient (μ_m) and time (t) by Beer-Lambert's law:

$$I = I_0 e^{-\mu_m \rho t}$$

where $\mu_m = \mu_{m, \text{abs}} + \mu_{m, \text{scatt}}$ and $\mu_{m, \text{abs}}$ and $\mu_{m, \text{scatt}}$ denote the mass attenuation coefficients that result from the photoelectric, absorption and the Compton scattering effects, respectively. This process has been used by the metallurgical industry, which has used automated devices for radiographic density measurement, as represented in Fig. 4A (Schlieper, 2010).

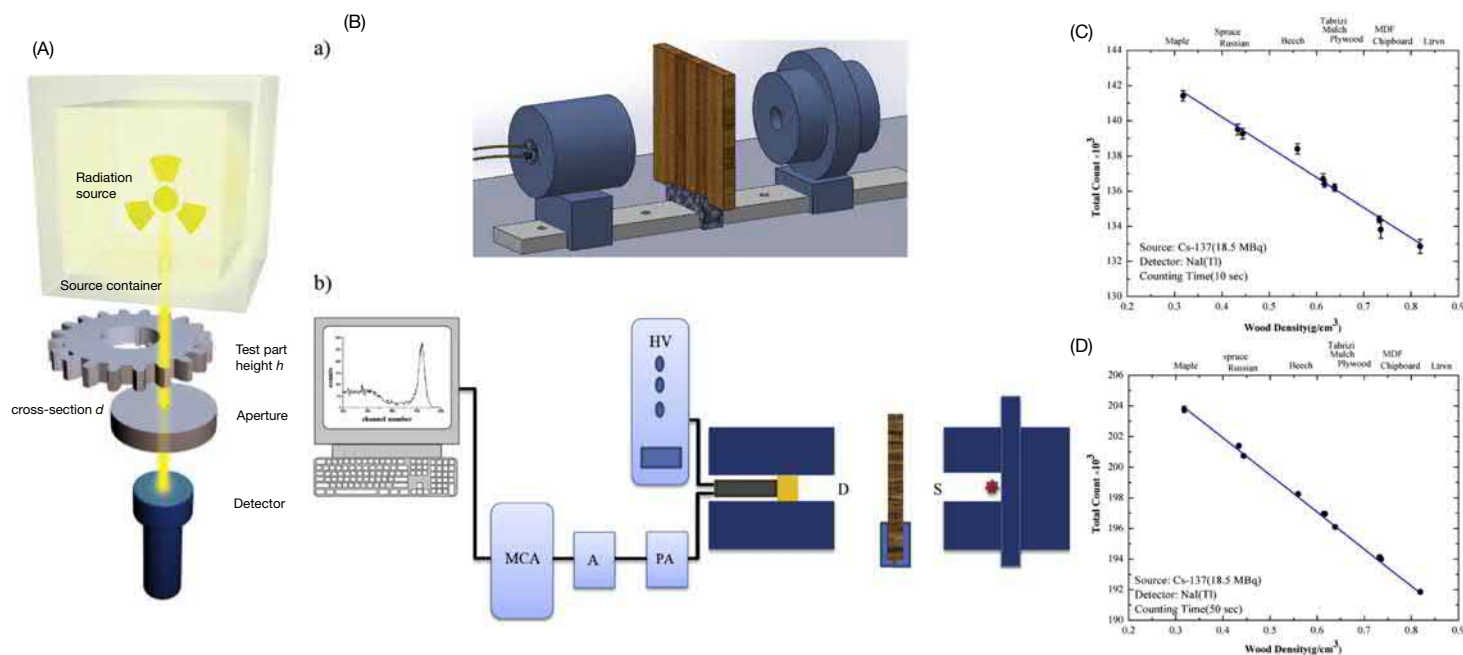


Fig. 4 (A) Schematic of automated gamma-radiation densitometer in use by the steel industry. (B) Schematic of densitometer adapted for measuring wood density in transmission mode (a) and electronics interface to generate a spectrum from the attenuation in gamma radiation transmission (b). (C) and (D) Gamma radiation counts versus calculated experimental densities for different wood samples at 10s and 50s of data collection time, respectively. From (A) Schlieper G (2010) Measuring PM density by going outside the visible spectrum. *Metal Powder Report* 65(3): 10–14. (B–D) Beigzadeh AM, Rashidian Vaziri MR, Soltani Z and Afarideh H (2019) Design and improvement of a simple and easy-to-use gamma-ray densitometer for application in wood industry. *Measurement* 138: 157–161.

Another example of such detector assembly is that for ^{137}Cs and NaI detector with a laminar wood sample placed halfway between detector and source, see Fig. 4B. In this setting, increasing the collection time from 10s to 50s improved the signal in the detector as summarized in Fig. 4C and D (Beigzadeh et al., 2019). However, the shorter detection times still offer reliable and reproducible results for **determining density of wood** from different sources. These changes are expected based on the different contents of cellulose and lignin polymers forming wood.

For materials having more uniform chemical composition and either an isotropic structure or anisotropic structure with preferred orientation of grains, such as plastics, glasses, ceramics, resin-fiber composites, metals and most steels, the same analysis can be applied. A similar principle is applied to measuring packaged solid and fluid levels in opaque containers, in which the attenuation coefficients of packaging materials and that of the packed substances are previously determined (Falahati et al., 2018). This method has been used by the **food and beverage, oil and gas, and construction materials industries**, to name a few.

Antimicrobial properties of irradiation

Gamma irradiation has been found as capable of killing microorganisms in different media. Gamma irradiation acts by breaking different types of chemical bonds in cellular media of the microorganisms. This bond-breaking process produces free radicals that attack the nucleic acid of the microorganism. Sterilization by gamma irradiation is achieved mainly by the alteration of nucleic acid, directly or indirectly, and that prevents the cellular division and microorganism proliferation (Hasanain et al., 2014). The radiation kills insects with doses as low as 500 Gy, and it kills mold and fungus and arrests the biodegradation processes above 10 kGy (Cortella et al., 2020).

Consequently, sterilization by gamma-irradiation has found many different uses in industry. The topics dedicated to **food preservation** and to **preservation of historic and archeological artifacts** are respectively covered in-depth in Chapter 8 (Pillai and Pillai, 2021) and in Chapter 22 of this section (Varquez and Nagai, 2021).

Gamma-irradiation has also been used for sterilization of pharmaceuticals and cosmetics. This includes products in either solid form or in aqueous solution, and polymers in the form of plastic medical products (Jacobs, 1995; Hasanain et al., 2014). Irradiation is particularly effective with heat sensitive pharmaceuticals, and with those incompatible with the most commonly used antimicrobial compound in industry, namely, ethylene oxide (Gopal, 1978). Several **cosmetics and hygiene products** have been safely irradiated with doses of 5–10 kGy. These include **baby powder, makeup products, solid and liquid soaps**. Starting materials include talc, bentonite, wheat and corn starch, and gelatin (Sivri et al., 2006).

Conclusions

The use of gamma radiation and focused e-beams, high energy ions, and neutrons have found many applications in developing a myriad of new commercial products, including new materials design and industrial processing techniques. These processing techniques include quality control, sterilization of medical equipment, of food, cosmetics and pharmaceutical packaging materials. Irradiation techniques have been employed extensively in the development of new polymers, including effective polymer recycling. They have also enabled the development of porous membranes with nanosized pores and of nanocomposites with fiberglass and carbon fibers. The use of radiation has paved the way for the precise manipulation of matter from the atomic level and up to tens of micrometers. Irradiation methods have been instrumental for the development of silicon semiconductor architectures below 10 nm. The use of radiation has enabled the commercial fabrication of non-siliceous electronic components and devices including MTJs, grid nanopattern waveguides for transparent electronic displays, YSZ catalysts and fuel cell electrodes. It also allows for hardening of steel and changing the mechanical properties of ceramics and polymers.

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Modern Industry: Non-nuclear Energy Applications

J. Thereska^a and P. Brisset^b, ^a International Society for Tracer and Radiation Applications (ISTRA), Vienna, Austria; and

^b International Atomic Energy Agency (IAEA), Vienna, Austria

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Abbreviations

FCCU fluid catalytic cracking unit

MEGRA multi-energy gamma ray absorption

PGNNA prompt gamma ray neutron activation analysis

RTD residence time distribution

Introduction

Whereas the main focus of this encyclopedia deals with nuclear power, along with the myriad applications of radiation technology to the subjects covered in this section, the other sources of generating heat and electricity such as oil, coal, natural gas, geothermal, and the renewables (hydropower, solar, and wind) also benefit from harnessed radiation. This article is devoted to providing a glimpse of how this is accomplished.

Oil and the petroleum industry

Oil is a fossil fuel that has been historically used for producing electricity. It was one of the principal means of producing electricity in the middle of the last century. However, the OPEC oil embargo of 1973 sent a shock wave through the industry and caused a significant reconfiguration of the global electricity-generating system. Oil currently plays a relatively minor role in the production of electricity in most countries—producing only about 5% or less in the United States today. However, oil is used in prodigious amounts for the transportation sector. The United States relies on oil for about one-third of its total energy consumption. Hence, oil remains a major energy source for transport and heating applications. Oil and petroleum industries are the main end users and beneficiaries of radiotracer and nucleonic gauge technologies (IAEA, 2002, 2004).

Borehole logging

The oil industry is heavily dependent upon the use of radiation to conduct business. Finding new oil fields is a constant effort, especially as traditional sites are becoming exhausted. The standard way to determine the siting of new oil fields is to drill boreholes

(wells) into the earth at strategic locations and then instrument these wells to make geologic measurements at various levels down the shaft. Borehole logging is a term used in conjunction with analyzing these test wells to determine the potential for economically viable oil deposits.

At least four methods based on radiation techniques have been developed to help determine which geologic strata being probed by the boreholes have the highest probability of being developed for oil production (IAEA, 2008). One is based on gamma ray backscattering, in which energetic gamma rays are emitted by two gamma sources placed into the borehole. One emits high energy gammas and the other emits relatively low energy gammas. The detector system placed near these sources in the bore hole can determine the bulk mass of the surrounding material from the intensity of the scattered high energy gammas (due to the Compton scattering effect). It can also determine the constituency of the bulk (i.e. the specific elements in the matter surrounding the borehole) from the intensity of the scattered low energy gammas (due to the photoelectric effect). The goal is to determine high hydrocarbon-bearing regions, characteristic of oil deposits.

A second method is to employ the prompt gamma ray neutron activation analysis (PGNNA) method. By identifying the carbon to oxygen ratio, a strong inference can be made regarding the presence of oil-bearing regions. The carbon to oxygen ratio varies from zero in the absence of hydrocarbons to a maximum value when the surrounding strata are saturated with hydrocarbons.

A third method is to utilize a neutron source in the borehole by employing the neutron backscattering principle to determine the hydrogen content. The americium-241/beryllium system yields neutrons in the 2–10 MeV range and can work well as an appropriate source. Neutron counters such as boron trifluoride detectors are sensitive only to thermal neutrons (neutrons that are thermalized most readily in a region of high hydrocarbons—due to the ability of the hydrogen atoms to rapidly slow down fast neutrons). Further, they are essentially insensitive to the fast neutrons coming from the source (since boron has a very low neutron capture cross section for fast neutrons). Hence, such detectors can be placed very close to the source without difficulty. Fig. 1 illustrates this method.

Yet another use of fast neutrons (though perhaps more expensive) is to utilize a pulsed neutron source by employing a high voltage ion accelerator. By accelerating deuterium onto a tritium target, 14 MeV neutrons are generated. Since the beam current (deuterium in this case) can be turned on and off in an accelerator, a fast beam of neutrons of only about 10 μ s in duration can be produced. A neutron detection system then measures the time it takes for backscattered neutrons to arrive, thus yielding a good estimate of the density of hydrocarbons in the vicinity of the region.

Radiotracers (radioactive tracers) for interwell connections in oil fields; interwell tracing

When primary oil production decreases in a field because of a reduction in the original pressure, water is usually injected to increase the oil production. Injected water in special wells (injection wells) forces the oil remaining in certain layers to emerge from other

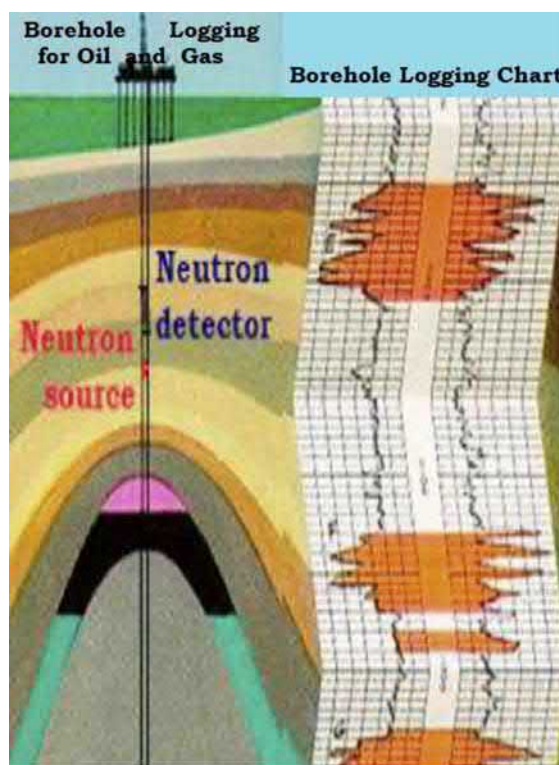


Fig. 1 Borehole logging for determining prime oil (IAEA, 2012).

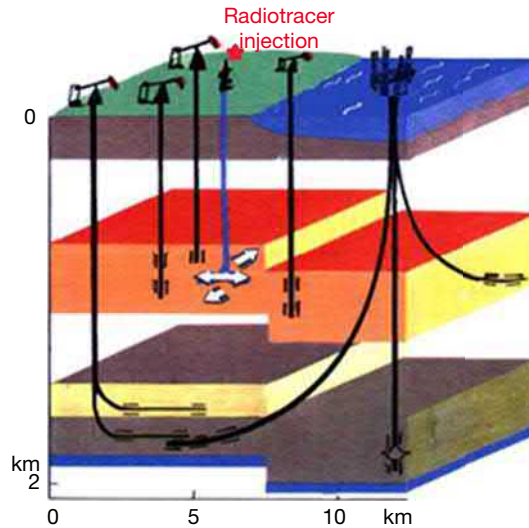


Fig. 2 Principle of radiotracer method for interwell communications (IAEA, 2012).

wells (production wells) surrounding the injector (Fig. 2). This technique, commonly called secondary recovery, provides extractions up to 50% of the original oil in place.

The main radiotracer technique is the measuring of the “time of travel” between injection and production wells. Tritium, as tritiated water, is ideal tracer for water (IAEA, 2012). Using radioactive (and other) tracers to tag the injected material, the following parameters can be determined:

- The percentage of injection water, or gas, flowing from an injector to a specific producer.
- The volumetric sweep efficiency of the flood.
- Reservoir preferential flow trends.
- Fault block or channel communication.
- Whether high-permeability channels are present.

Fig. 3 shows two typical experimental curves in production wells representing “strong channelling” and “normal dispersion” models of underground water transport.

A practical value of tracer tests could be also:

- shutoff of highly watered zones;
- better planning of injection and production wells configuration.

The enhancement of the oil production from the investigation of the secondary recovery processes by radiotracers is estimated to 10–15% of the residual oil for each oil field, which translates into an economic benefit of several million of US\$ per year (IAEA, 2012).

Nucleonic gauges for multiphase flow rate measurement

An increased number of applications with nucleonic gauges in the area of multi-phase flowrate measurement have been reported, mostly by international companies investing in oil production. Here the oil, water and gas flowrates are measured continuously without separating the various phases. Oil-water-gas flow meter nucleonic gauges replace expensive test separators, saving several million of US\$ in one oil field alone (IAEA, 2008).

Multi-Energy Gamma Ray Absorption (MEGRA) gauges for multi-phase flow measurements have been conceived to correct changing fluid parameters that might influence the calibration (e.g. salinity) (Fig. 4). The trend is to replace the radioactive sources with X-ray tubes for higher X-ray flux (increased accuracy) and safer use (e.g. during transportation and/or maintenance). A multi-phase flowmeter that employs the MEGRA concept does not require field calibration, which is a decisive advantage in subsea or marginal field developments (Sowerby, 2002; IAEA, 2008).

Oil refineries

Efficient refinery operation is also a very important part of the oil industry. Whereas it is difficult to install and maintain diagnostic probes inside the distillation towers (due to the extremely harsh chemical and temperature environment), gamma probes can be rather easily installed on the exterior of the towers, and then be moved up and down the tanks to record the composition of

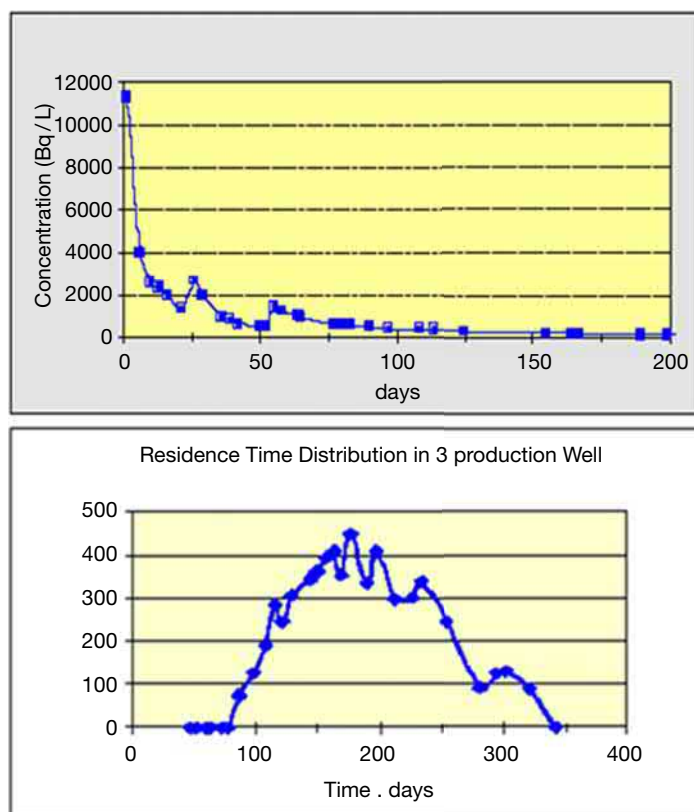


Fig. 3 Two typical experimental curves in production wells representing “strong channelling” (*up*) and “normal dispersion” models of underground water transport (*down*) (IAEA, 2012).



Fig. 4 Multiphase flow meter (MEGRA) installed in oil platform (IAEA, 2008).

ingredients at various vertical levels. Any malfunctions within the tanks can be readily detected. Many of these distillation columns are very large (measuring some 3 m in diameter and ranging to heights of 30 m). Hence, being able to obtain accurate measurements of the contents is very important to the overall economics of the operation (see Fig. 5) (IAEA, 2002).

Neutron backscatter gauges are also routinely used in the oil refinery industry to monitor the level of liquids in tanks and to determine the interface between different liquids. Because hydrogen is a highly efficient neutron moderator, the detection of backscattered thermalized neutrons can be used as an effective measure of the concentration of water or hydrocarbons in tanks.

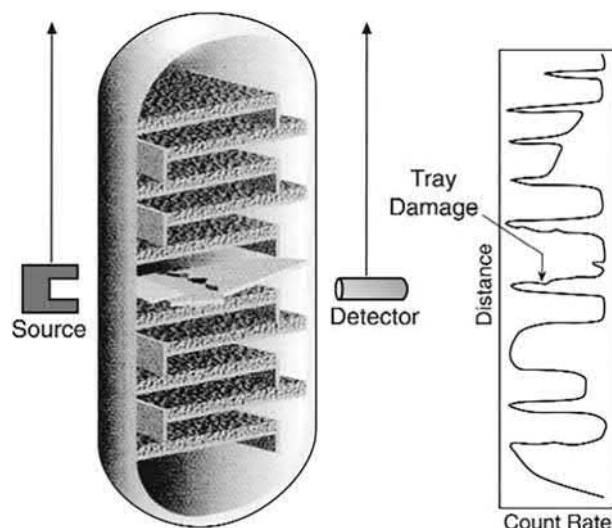


Fig. 5 Use of radiation gauges to check oil refinery operations (IAEA, 2002).

Petrochemical complexes

The petrochemical plants lie immediately downstream of the oil refinery and in many developing countries construction of the two types of facilities proceeds in parallel. Refineries and petrochemicals plants are generally continuously operating and technically complex. Thus, high economic benefits may be realized by the applications of radiotracer techniques on petrochemicals units. Though radiotracers are useful in solving a wide range of problems, the economic benefits become more pronounced the further “upstream” they are applied. This means that diagnosis of the cracking furnace, primary fractionator and gas separation chain, is of the highest potential value.

Fig. 6 shows a typical Fluid Catalytic Cracking Unit (FCCU) where radioactive tracers are employed to troubleshoot and diagnose the process. The fluid catalytic cracking (FCC) is one of the most important conversion processes used in petroleum refineries to convert the heavy portion of crude oil feedstock into lighter petroleum products, including liquefied petroleum gas and gasoline.

Coal

Fossil fuels provide the largest single source of energy for producing electricity, and coal is still the mainstay around the world. The principal attribute of coal is its availability and relatively low cost. Huge deposits still remain in many regions of the world.

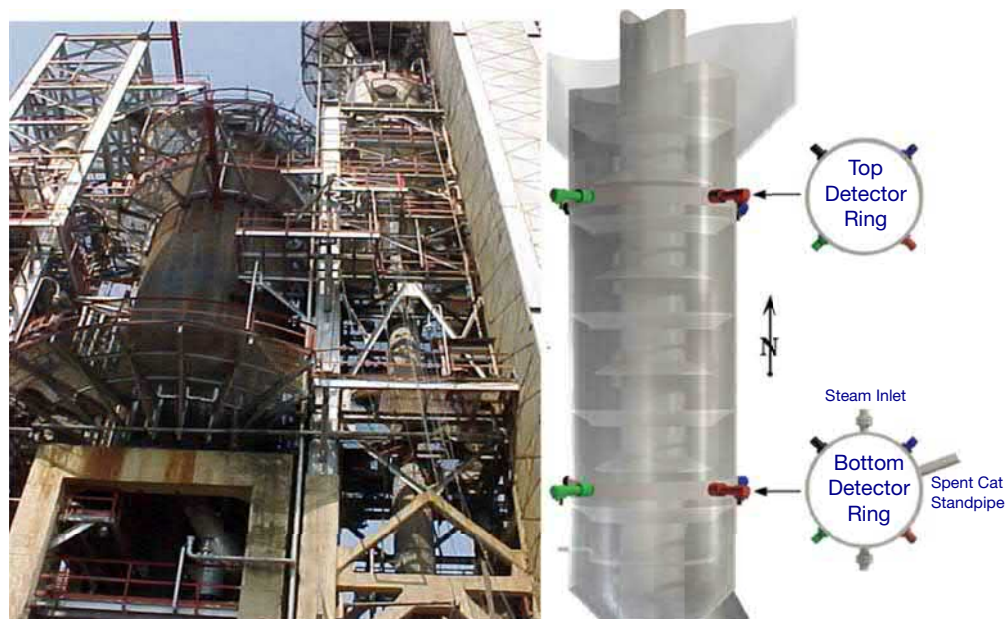


Fig. 6 Radiotracers for diagnosing FCCU (IAEA, 2002).

However, the best low-sulfur coal is being rapidly depleted—requiring more expensive procedures to mine the deeper and scarcer rich supplies. Also, transportation can become a major obstacle. China, for example, is thought to contain the largest deposits of coal of any nation, and these deposits are needed to serve the energy needs of its burgeoning population. Indeed, China is adding large numbers of coal plants for the production of electricity. However, the coal seams are mostly in the northwest region of the nation, far removed from the populous southeast region. The Chinese simply do not have the rail infrastructure capable of moving this source in adequate quantities to the areas requiring power.

But coal has several disadvantages. One is environmental pollution, with both sulfur and nitrogen oxide discharges (leading to acid rain) and carbon dioxide (leading to global climate change). Another is public safety. Far more deaths occur each year from the use of coal (mining and railroad accidents, plus human lung diseases caused by atmospheric smog) than any other fuel source. Furthermore, it is a non-renewable source—available for several more decades, but certainly not for the long term. Finally, coal is the chemical feedstock for numerous commercial products, ranging from laundry detergents to nylon hosiery. It is the key ingredient in the manufacture of iron and steel.

Borehole logging for searching for coal

Fig. 7 illustrates the borehole logging principle for searching coal and minerals, as well as showing a low activity borehole logger system for coal and mineral alignment (Sowerby, 2002).

Analyzing the quality of the coal

The coal industry, which currently accounts for well over half of the world's supply of electricity, benefits directly from using radiation gauges to measure the moisture content in coal, determine the ash content, determine the content of other ingredients (such as sulfur), determine the BTU (heat content), and measure the rate of flow of coal into the furnace (Cutmore et al., 1993; Sowerby, 2002). In order to obtain optimal burn efficiency, coal fired boilers must be fed with coal having a constant caloric value.

One way to obtain the above measurements is to bombard the coal entering the plant on a conveyer belt with cesium-137 and americium-241 (Fig. 8). The Cs-137 emits energetic gamma rays (662 keV), whereas the gamma rays emitted from the Am-241 are a factor of 10 times weaker (~ 60 keV). The high energy Cs-137 gammas are attenuated by Compton scattering (a process essentially independent of the composition of the material). The lower energy gamma rays from the Am-241 are attenuated by the photoelectric effect (which is very sensitive to the atomic number of the flowing materials), thereby easily revealing the presence of iron or silicon in addition to the desired carbon content. Hence, the flow rate of the coal being fed into the furnace is determined primarily by detection of the Cs-137 signal, whereas the percentage of ash in the mixture is determined by a computer analysis of the transmission of both the Cs-137 and Am-241 gamma rays.

More sophisticated radiation gauge systems can also be employed to measure other trace ingredients, including moisture content. One approach is to employ the X-ray fluorescence technique. Another is to employ the prompt gamma ray neutron activation analysis (PGNAA) (Cutmore, 2018). Here the trick is to irradiate the coal along the conveyer belt with neutrons (from a portable neutron generator, such as californium-252 or the Am-241/beryllium system) and cause some of the materials in the coal stream to become activated; thus, releasing prompt gamma rays characteristic of the materials to be identified.

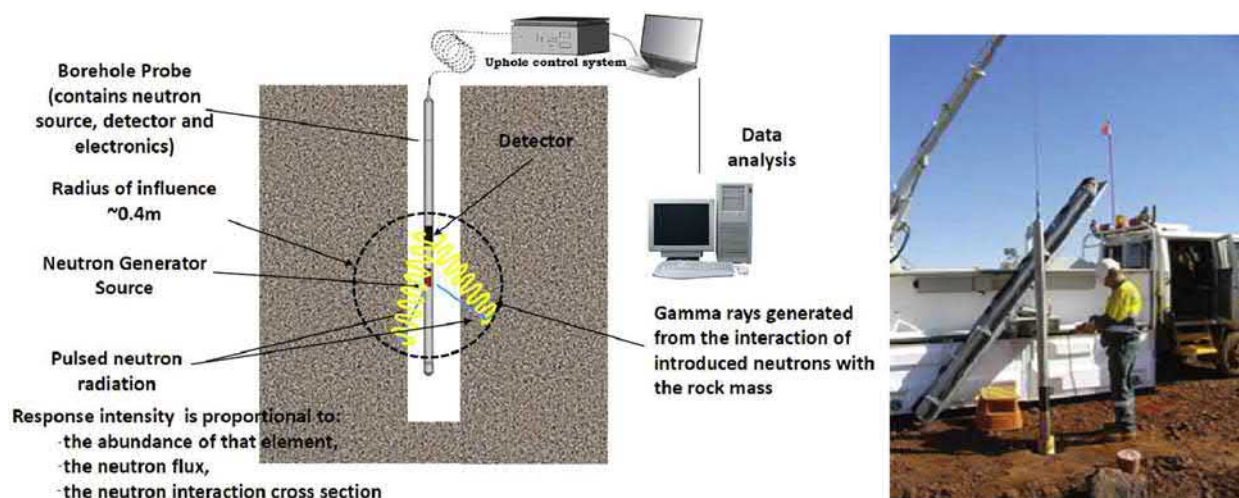


Fig. 7 Borehole logging probe principle (left) and low activity borehole logger system (right) (Sowerby, 2002).

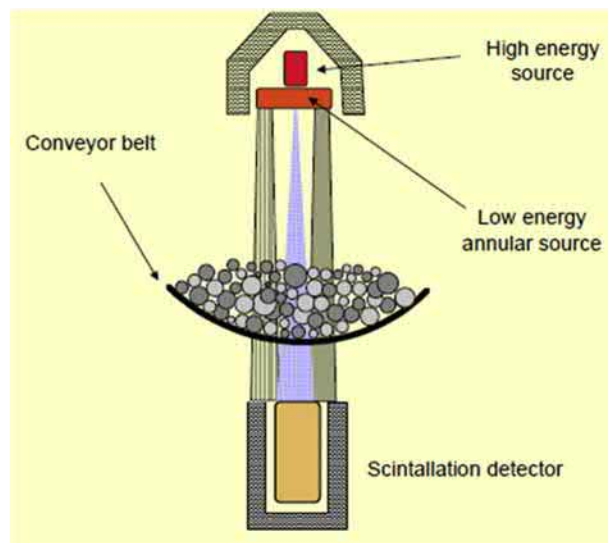


Fig. 8 Analyzing the quality of coal while entering the plant on a conveyor belt (IAEA, 2000).

It is important to determine the sulfur and nitrogen combustion products because these gases are of considerable environmental concern in causing acid rain. Sulfur that resides in the coal entering the plant combines with oxygen and forms a variety of sulfur oxides. Nitrogen is always present because it is the major component of air. Hence, because of the very high temperatures attained in the coal furnace, nitrogen oxides are also formed.

Radiotracer valuation of coal-ash dust cyclone efficiency

In the large boilers for steam production, the coal-ash is carried by air current into the cyclones, which constitute the principal parts of the cleaning system. To diagnose the cyclone efficiency, radiotracer tests have been performed. As an example, a case was studied where the granules of coal-ash were sprayed with an aqueous solution of Tc-99m and then evaporating to dryness. It was proved in laboratory testing that mass labelling had been attained. Thus, the tracer activity was proportional to tracer mass (necessary condition for mass balance). For each test, 30 g of labelled coal-ash was injected with an activity of 200 MBq. The experimental RTD curves for old and new cyclones are presented in Fig. 9. A significant difference was found between the efficiencies of the old and new cyclones for the grain class 0–40%. An improvement in the performance is noted for the new cyclone. For other two classes (40–80 and 80–100 μm), these differences were insignificant.

A new radiation technique, called electron beam (EB) processing, has been demonstrated to effectively remove both sulfur and nitrogen oxides in a single-stage process. By passing the sulfur and nitrogen flue gases under a strong beam of electrons, these pollutants become ionized (hence, electrically charged) and can then be collected on positively charged plates or diverted by a magnet—substantially ridding the flue gas from these environmentally objectionable contaminants. An additional attribute of this process is that these toxic substances can be converted into a commercially viable agricultural fertilizer.

Natural gas

Natural gas has become the recent “darling” for producing electricity in the United States, especially now that fracking technology has been effectively harnessed. The advantages are that these plants can be built quite rapidly and relatively inexpensively, and they can produce reasonably cheap electricity—when the price of natural gas is low. Unfortunately, natural gas supplies (along with the necessary pipeline network) are limited, and volatile gas prices can occur with little notice. The most serious spikes in natural gas prices occur when the demand is exceptionally high (e.g. huge cold snap) and the pipelines cannot handle the load.

Natural gas is a cleaner fuel than coal because it produces no sulfur oxides and about half of the carbon dioxide of a comparable coal plant. However, it does emit nitrogen oxides and, in particular, methane gas. The methane that leaks out in the thousands of miles of gas lines is becoming a sizable contribution to greenhouse gases, since methane is many times more effective than CO_2 in contributing to climate change issues. Perhaps of greater importance, gas is also a non-renewable source of energy. Furthermore, natural gas is a major feedstock to the chemical industry, which creates a mounting concern that an increased reliance on natural gas for electricity production could severely impact the ammonia industry. Enormous quantities of natural gas are required to produce fertilizers used throughout the world.

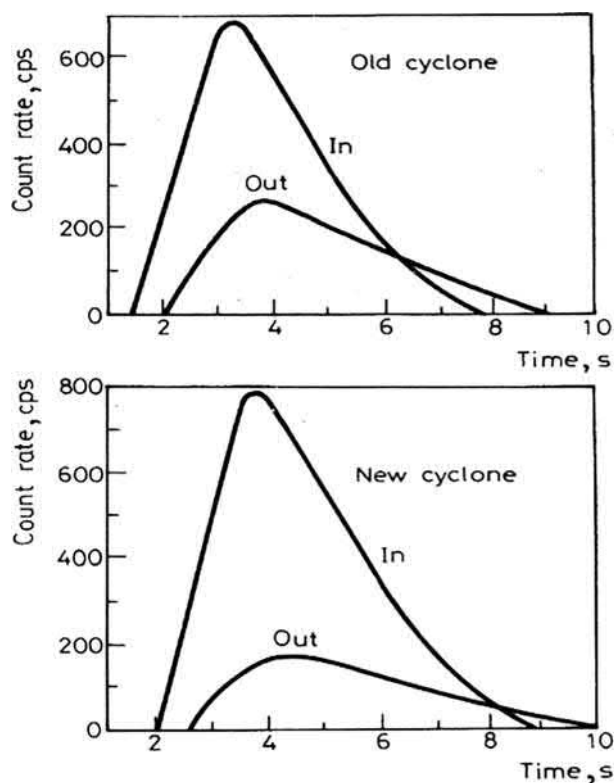


Fig. 9 Experimental RTD curves in the inlet and outlet of old and new cyclones for class 0–40 μm (IAEA, 2004).

Fluid flow rate calibration in closed conduits using radiotracer methods

Radioactive tracers have become important non-invasive tools for the measurement of flow rates in single phase flows inside closed conduits. Radioactive tracer methods are the most competitive and sometimes the unique method to carry out an online calibration of flow meters installed at any processing equipment without interrupting production, bringing considerable economic savings. The main customers of these methods are petrochemical and chemical plants, natural gas industries, the mineral processing industry, and wastewater treatment installations (IAEA, 2008).

Fig. 10 shows the principle of the radiotracer methods for single phase flow rate measurement in closed conduits for two case studies: on-site flow calibrations on a natural gas line and on a sulfuric acid production line.

Geothermal

Geothermal energy is a term used to characterize the harnessing of heat residing deep within our Earth. Hot magma, resulting from the slow decay of radioactive materials that make up the bulk of our planet (yes, the bulk of the mass beneath the crust of our Earth is heated by radioactive decay!), sometimes protrudes close enough to the outer crust of Earth that it can be accessed. Hence, there are limited places where it is feasible to drill deep enough wells to pipe water down to the magma and tap the steam that is created to subsequently spin electrical generators or heat homes. The principal problem with this source is that there are very few areas where this is practical. Still, geothermal energy may become a significant, long-term contribution to the generation of electricity.

A radiotracer study was carried out at the Rotokawa Geothermal Field, New Zealand (Addison et al., 2017). An injection of 200 GBq of Iodine-125 at specific locations was used to better understand the geology of the field and to optimize water injection and, consequently, to improve the thermal recovery. The Rotokawa geothermal field in New Zealand is one of approximately 15 known high temperature geothermal systems within the Taupo Volcanic Zone (TVZ). In 2010, the total water intake from the reservoir increased from around 15,000 t/day (which supplied the 34 MWe Rotokawa geothermal power plant) to nearly 65,000 t/day with the commissioning of the 138 MWe Nga Awa Purua ("NAP") geothermal power plant. Such radiotracer tests have been employed in this geothermal field many times. Fig. 11 shows the Rotokawa field layout as of the start of the 2015 reservoir tracer test.

Results from these tracer tests have assisted with improving the conceptual model understanding of the field. In particular the lack of any observed connection to the Western Compartment within 1 year of tracer results was consistent with expected

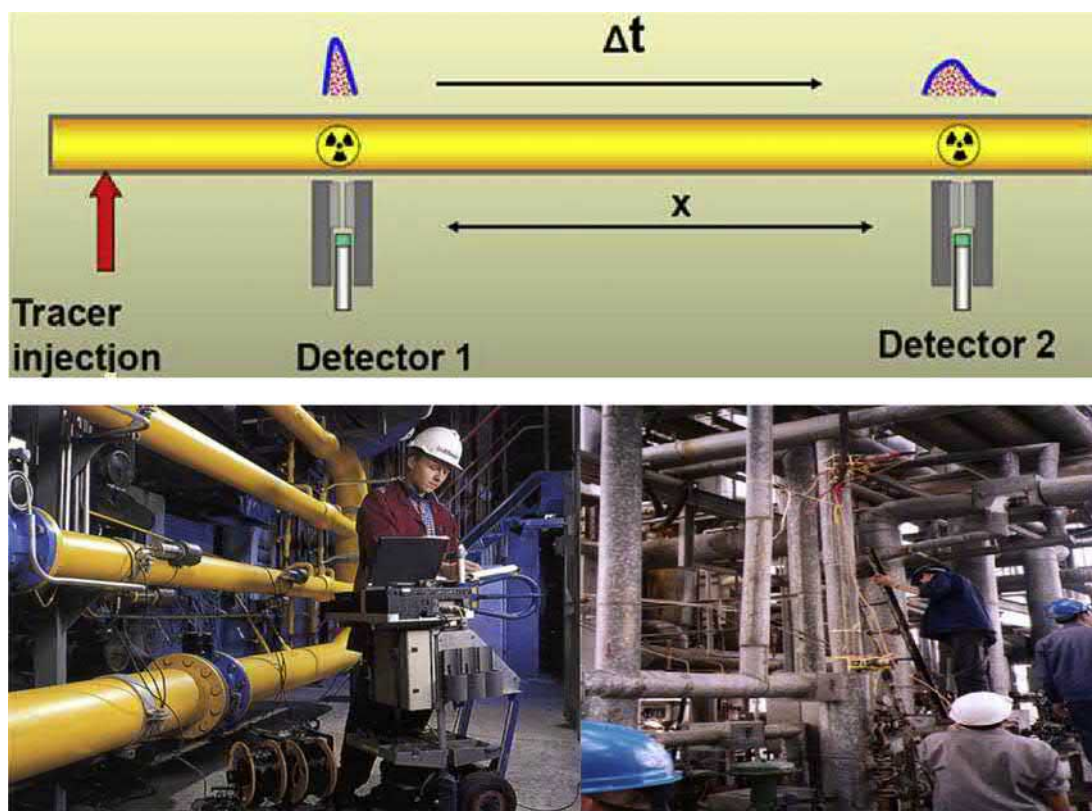


Fig. 10 On-site flow calibration on a natural gas line and on sulfuric acid production line (IAEA, 2008).

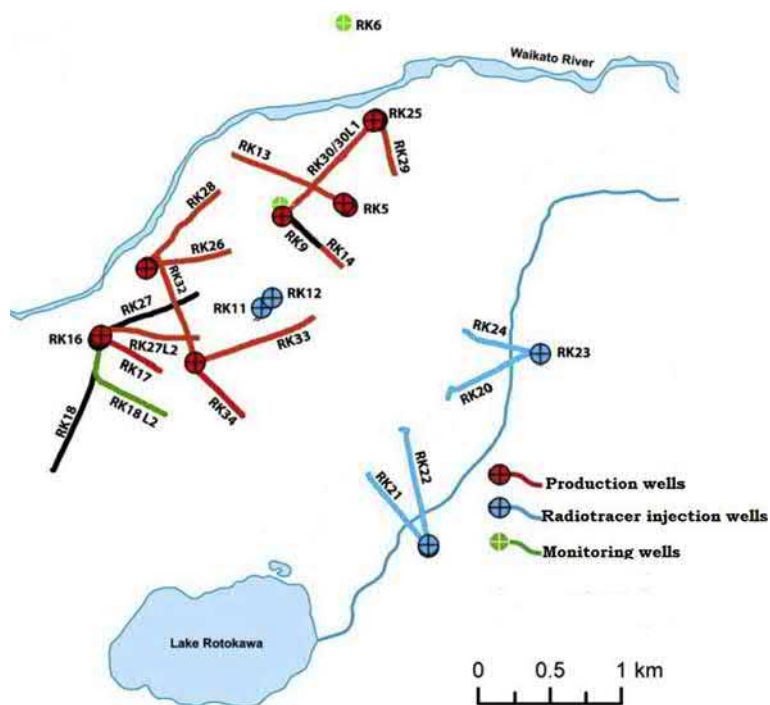


Fig. 11 Rotokawa field layout where a radiotracer was injected into injection wells (Addison et al., 2017).

permeability barriers within that part of the reservoir. In addition, the tracer tests assisted in explaining the generally low-pressure drawdown observed in that part compared to other parts of the Rotokawa reservoir.

Increased conceptual understanding from reservoir tracer tests have assisted in providing updates to the numerical model for the field that result in better matches of pressure in both the production and injection areas.

Hydropower

Many countries use water resources for electric power generation. The advantages of hydropower include its intrinsic renewal capability, long-term viability, minimal atmospheric pollution, and relatively low cost. However, dam silting-up problems create difficulties in providing a constant electric power supply, and shortfalls in electricity generation are sometimes experienced during peak demand periods. The increasing demand for electricity calls for expanded exploitation of the hydroelectric dams, which requires heavy dam maintenance. Dams and reservoirs are vital in terms of water supply, irrigation, flood protection, and electricity generation. Large investments are needed for maintaining the efficiency of dam and reservoir operations. Sustainable exploitation of dams requires an in-depth understanding of the silting-up process and monitoring of sedimentation rates using both nuclear and conventional techniques.

The nucleonic density gauge can measure continuously the suspended sediment concentrations providing data for optimal operation and better management of dams and reservoirs (IAEA, 2014). Monitoring the vertical profile of the sediment concentration in water bodies is necessary for dam reservoir flushing. Fig. 12 shows the Seditracker nucleonic gauge for monitoring the sediment density versus time employed in Kinguele dam, Gabon.

This nucleonic gauge can measure relatively large intervals of sediment density in water bodies (0.5–500 g/L). An ^{241}Am radioisotope source has been applied in some cases. In other cases, X-ray generators have been replacing radioisotope sources because of relatively simpler regulation aspects.

Solar

There are basically two types of solar plants in existence today: solar furnaces and photovoltaic (PV) devices.

Solar furnaces work on the principle of utilizing a large array of mirrors to focus direct solar rays onto a “point,” where water is then boiled to make the steam necessary to drive electrical generators. Photovoltaic devices utilize an ingenious application of photocells to convert sunlight directly into electrical current. Neither solar furnaces nor photovoltaic cells can operate without constant access to the sun.

There are applications where solar power can provide an ideal choice for heat and electricity. But the fundamental problem with all forms of solar energy is that it is a very diffuse energy source and it is completely unavailable during the night. Hence, enormous collection areas are needed to capture the quantities of energy needed for modern civilization, especially given the low efficiency of converting sunlight to electricity.

One of the main contributions of radiation technology to solar furnace electricity is in the manufacturing of the vast amounts of materials needed for the collection surfaces required. Radiation gauges are used extensively in making the rolled metal sheets that provide the base metal needed in forming the dishes required to collect the sun’s rays.



Fig. 12 Seditracker nucleonic gauge for monitoring the sediment density versus time in Kinguele dam, Gabon (IAEA, 2014).

For photovoltaic (PV) applications, the main ingredient required for making the system work is the semiconductor. Much the key technology required to manufacture these semiconductors is derived from nuclear technology, where ions are accelerated and implanted into the silicon matrix in a process called doping.

Wind

The wind, of course, is a direct result of the atmospheric pressure differences caused by the sun heating air masses at a different rate in various geographical locations. Because of the rapid fluctuations of the wind, and the low energy density of this source, a huge number of wind turbines must be fabricated and spaced over very large collection areas (much like solar plants).

To better appreciate the scales involved, for a wind farm to produce an amount of electricity equivalent to a comparably-sized nuclear power plant, the amount of material (aluminum, concrete, etc.) required is over 10 times that of the sister nuclear plant.

A perspective on the renewables

Whereas hydro, solar, and wind are generally considered the renewable sources for producing electricity, in reality there is one additional renewable source; namely nuclear power.

Nuclear power? How can this be? Technological advances in extracting uranium from sea water is getting close to becoming a commercial reality. Even though the amounts of uranium dissolved in sea water are very low (parts per billion), new techniques for extracting this nuclear fuel are becoming available to allow energy planners to foresee this as potentially an infinite supply.

See Also: Natural and Man-Made Sources of Radiation; Radiation Sources.

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Modern Industry: Personal Care, Conveniences, and Safety

Alan E. Waltar*, Department of Nuclear Engineering, Texas A&M University, College Station, TX, United States

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Glossary

Nucleotides Any of several compounds not found in nucleic acids, which function as coenzymes.

Abbreviations

DNA Deoxyribonucleic acid
FDA Federal Drug Administration (USA)
MRI Magnetic resonance imaging
NAA Neutron activation analysis
PCR Polymerase chain reaction technique for DNA analysis
PET Positron emission tomography
RFLP Restriction fragment length protocol technique for DNA analysis
SPECT Single photon emission computed tomography

*Retired Professor and Chair.

Introduction

Most of the chapters of this section of the encyclopedia focus on the various technical aspects of how nuclear radiation can be harnessed to address items of importance within our modern world. The present article is considerably “lighter” in scope. Here we attempt to capture just a glimpse of some of the way’s radiation can help in our everyday life—perhaps qualifying as our silent servant that we don’t normally recognize as even existing.

Personal care

General health

Our personal health is of precious value to every human being. Here we review some of the ways that harnessed radiation helps us get the most out of daily living. We start with perhaps the most valued part of our everyday surroundings: the air we breathe and the water we drink.

Clean air

Air pollution is still a serious problem in many parts of the world. One of the main sources of air pollution is the burning of fossil fuels for heating, the generation of electricity, and for transportation. Noxious byproducts of the combustion of oil, coal, and natural gas include sulfur and nitrogen oxides, as well as particulates that can lead to acid rain, lung disease, etc. All of these sources of energy conversion (including geothermal plants) use the earth’s atmosphere as their ultimate waste disposal medium. There are also many places in the world where chemical residues remain in the air due to the wide-scale use of pesticides in the agricultural industry. Large quantities of CO₂ and methane are emitted from agricultural activities and from primitive societies that burn dung and wood.

Fortunately, radiation technology is being used to reduce many of these pollutants from our atmosphere. The most obvious contribution is the electricity production from nuclear power plants, in which there is essentially no environmental discharge during normal operation. Some may question the issue of the final disposal of nuclear waste, but this (along with the other aspects that need to be included when evaluating the full life-cycle) are dealt with in-depth in another section of this encyclopedia ([Greneche, 2021](#)).

But fossil fuels still remain as the primary energy source in the world today, and will likely remain so for many more decades. Hence, it is important to reduce their polluting discharges as much as possible. Fortunately, one way that radiation technology is helping to clean the flue gas from fossil fuel plants is a flue gas purification system based on electron-beam irradiation that removes the noxious sulfur and nitrogen oxides and even converts them into useable fertilizer ([Chmielewski et al., 2021](#)).

Clean water

The continuing demand for potable water, combined with the gradual decline of fresh water resources, is becoming one of the most urgent global matters for life on this planet. Hence, finding new sources of fresh, drinkable water is a continuing struggle. Radiation tracer technology is a powerful tool in finding out how aquifers are being recharged, the age of such resources, and the purity of the water. More on this topic is included in ([Kumar and Ansari, 2021](#)).

A huge portion of the fresh water used on this planet is in farming. Much of the water now used for growing crops is wasted—i.e., it is simply not efficiently getting to the plant roots where needed. Fortunately, many farms are now employing radiation technology to optimize the use of both water and fertilizer. Neutron gauges are used to determine the moisture content of the soil, to ensure the level of moisture needed but avoid the overuse of irrigation. Tracer technology is used to ensure that the nutrients are getting to the plants in an optimal fashion, thereby avoiding the overuse of fertilizer. Runoff of unnecessary fertilizer results in both polluting streams and wasting the energy required to manufacture the fertilizer. These subjects are treated in more detail in ([Sivasankar et al., 2020](#)).

Given that over 97% of the water on the planet is seawater, and that many major populations centers are situated along the coastline, there is an obvious incentive to explore desalination (i.e., extract the salt and produce drinkable water) to help meet the increasing global need for additional potable water. The concept is not new. In fact, there have been many relatively small desalination plants in operation around the world. But most of them derive their energy from fossil fuel plants, using processes such as water vaporization (leaving the salt behind) or reverse osmosis (driven by electrical pumps that pressurize water and force it through a semi-permeable membrane against its osmotic pressure). But nuclear reactors are well-poised to provide the heat or electricity needed to enable such production of drinking water from the ocean without polluting our atmosphere, and many nuclear projects have provided successful demonstration (<https://world-nuclear.org/information-library/non-power-nuclear-applications/industry/nuclear-desalination.aspx>).

An additional way of utilizing nuclear processes is the potential for treating polluted water to kill bacteria and other pathogens, thus allowing the water to be sanitized for productive uses. This technology ([Macedonio and Drioli, 2010](#)) is especially appealing as we look for ways to deal with international emergencies where potable water is in very short supply and is needed to allow local populations to survive the crisis.

Treated sewage

Increasing urbanization over the last two centuries has been accompanied by an expansion of sewage collection systems that lack the capability and capacity for proper treatment. Liquid waste loads have become so large that the self-purification capacity of downstreams receiving the treated sewage can often no longer prevent adverse effects on water quality—especially for areas of large populations. These wastes now constitute significant sources of water pollution. The industrial effluents carry chemical contaminations such as heavy metals, organic pollutants, petrochemicals, pesticides and dyes, while the discharge of sewage and sludge gives rise to microbiological and chemical contamination of water bodies. The discharge of such materials into water bodies is responsible for risk of infection, health effects caused by contaminated drinking water, and offensive odors. Therefore, all these industrial and municipal wastewater systems need more adequate treatment.

Radiation processing of wastewater treatment is a non-chemical approach based on irradiating the waste stream with high-energy photons or electrons. The energy from these beams is absorbed in the electron system of the wastewater polluting molecules and results in breakage of interatomic bonds—thus, removing the objectionable biological contaminants from the effluent.

In addition to treating wastewater, research has shown that sewage sludge can be disinfected successfully by exposure to high-energy radiation, and the resulting product can be safely used as fertilizer. The technical basis for how all this works for both wastewater and sludge, including successful global demonstrations, is provided in [Chmielewski et al. \(2021\)](#).

Personal grooming and adornment

One of the major advances in the use of radiation is the development of super-absorbent materials. Radiation-grafting and cross-linking processes ([Fulvio, 2021](#)) have allowed the manufacture of disposable diapers used for a full range of ages (babies to those with aging incontinence issues). Feminine products such as tampons also benefit from this technology. Drug stores and grocery stores now often have a major segment of their offerings devoted to stocking this ever-increasing demand for super-absorbent materials.

An extensive variety of clothing materials is currently available—largely as the result of inexpensive cotton now grown in abundance, worldwide, due to new strains of cotton developed using radiation mutation breeding processes ([Sivasankar et al., 2020](#)). Clothes made from synthetics likely also derive benefit from radiation cross-linking or other materials-improvement processes somewhere in the development phase. In particular, we might select the blue polyester-cotton blended shirt because radiation was used to bind special chemicals to the fabric to assure a “fresh-pressed” look all day.

A very large percentage of our population require either glasses or contact lens for improving their eyesight. Without radiation applications, neither would be economical or safe. The glass used in eyeglasses very likely utilized radiation processes that improve the quality of glass (using neutron probes to measure the precise amount of water required during the manufacturing process of making the glass). People wearing contact lenses are likely unaware that the saline solution, used for washing and storing the lenses while not in use, is sanitized by gamma irradiation to kill any microbes that may irritate our tender eyes.

Similar radiation processing devices (including tracer technology) are used in the manufacturing process of cosmetics ([Fig. 1](#)). The exact ingredients used for the cosmetic industry are likely proprietary, but (like the sausage industry) one could be excused for not wanting to know precisely what went into the product. Fortunately, the cosmetics amply featured on the shelves of most drug stores and other convenience stores are safe and can be safely used due to the irradiation of the products to remove any unfavorable parasites ([Fulvio, 2021](#)).

Precious gems

Precious gems are universally admired and desired. But few people know that radiation plays a very significant role in transforming many of them into even more beautiful and desirable acquisitions.

Basically, gems derive their beautiful hues from mineralogical impurities. Jade, for example, derives its deep green because it contains chromium. This element absorbs red, blue, and violet light waves—but not green. Beryl, which is colorless in its pure mineral form, becomes emerald with chromium impurities. On the other hand, manganese impurities in beryl turn it into a pink color.

Gemstones can be enhanced from their natural condition by several means. The principal modes of enhancement are heating, oiling, diffusion, and irradiation. Heating is a widely accepted enhancement process used on rubies, sapphires, topazes, tourmalines, and zircons. Techniques range from simply throwing gems into a fire to employing sophisticated electric or gas furnaces at specific pressures and atmospheric conditions. Oiling is an ancient process that allows oil to seep into the fissures that reach the surface of the stone. This is not a permanent process, but it is often repeated—especially during cleaning or mounting. Diffusion is a method wherein a material such as an oxide form of titanium is coated onto a stone and left there until sufficient diffusion takes place near the surface to change the color. Colorless sapphire is transformed into blue sapphire by this method.

Irradiation is also now in common use to enrich the attractiveness of many precious gems. The following is representative of how radiation is used to enhance the value of several gems ([Matlins and Bonanno, 1989](#)).

diamonds _____ changes the color from an off-white to a fancy color (e.g., green or yellow).
kunzite _____ darkens the color.



Fig. 1 The large array of safe cosmetics now available (Waltar, 2004).

pearls _____ produces blue and shades of gray ("black" pearls).

topaz _____ changes from colorless or nearly colorless to blue; intensifies yellow and orange shades and creates green.

tourmaline _____ intensifies pink, red, and purple shades.

yellow beryl _____ creates yellow color.

In some cases, the irradiation effects will fade away. This is the case when trying to achieve blue, orange, or yellow colors by irradiating sapphire or by irradiating beryl to attain a deep blue color. However, for the cases listed above, the noted properties become permanent.

Perhaps the most successful application of radiation to the gem industry is in the creation of blue topaz. Tens of millions of carats of topaz are irradiated annually. Electron beam irradiation is used for converting white topaz to "Sky Blue" and converting "London Blue" to "Swiss Blue" type colors.

Cobalt-60 is often the radioisotope used to create "Cobalt Blue" topaz. The gamma rays cause electrons to be released from their normal locations in the gem. The resulting color then depends upon where the electrons relocate, as well as the charge of the atoms in their vicinity. These factors control the way the stone absorbs light—thus dictating the resulting color.

Topaz is normally colorless in its pure mineral form. It is only the rare case that blue topaz can be found in nature. When cobalt-60 irradiation was first attempted to convert the much-more-abundant colorless topaz to a rich blue color, the stone changed to a cinnamon brown—and then often faded when exposed to sunlight. However, the irradiation process has since been improved significantly, and the only way to distinguish blue topaz produced by irradiation from the rare natural blue topaz is to go through an elaborate analytical process to measure the light emitted by the gemstone when it is heated. There is no way to distinguish this with the naked eye. Hence, almost all blue topaz stones now on the commercial market are produced by irradiation.

Given the uncertainties in gem processing, there are no guarantees of success for gemstone enhancers. As noted earlier, trace minerals contained in the stones naturally affect gemstone colors, so the combination of trace minerals and irradiation can produce some unexpected results. Hence, many gemstones must be irradiated in order to obtain a few stones that are altered to the desired color (Fig. 2).

Irradiation is also used to change quartz, which is clear and colorless in its pure mineral form of silicon dioxide, into the more attractive and valuable smoky quartz. However, this transformation can be attained only when the colorless quartz contains traces of aluminum. Also, ultimate success depends upon where the electrons settle after irradiation.

The irradiation of diamonds is particularly interesting. New colors are attained in diamonds as a result of the changed crystalline structure. Given the universal attention accorded to diamonds, it is not surprising that diamond was one of the first gems to be tested with radiation. Indeed, the early tests were performed by Sir William Crookes in 1904 (the year after Marie Curie won her first Nobel Prize), when he used radium to produce gamma rays. He was able to achieve a permanent green color. However,



Fig. 2 Precious gems created by irradiation processing. Compliments of Wilson A.P. Calvo, Institute of Energy and Nuclear Research, Brazil.

substantially more success has been achieved in recent times with atomic particles accelerated in a cyclotron. Highly energetic protons, neutrons, or even alpha particles have been used to more efficiently obtain permanent color changes.

The stones that are irradiated today are often then heated to about 800 °C to free them of any residual traces of radiation. Indeed, the irradiation of stones can, under some circumstances such as neutron irradiation, leave the gems slightly radioactive. This has resulted in a potential health hazard in some past situations. Consequently, laws have been passed that require precious-gem enhancers to allow their products to decay to a very safe level of radioactivity before they can be marketed. Some instances have occurred in the past where less-than-scrupulous dealers have placed the enhanced stones on the market before allowing them to decay sufficiently, and there have been a few cases where this has resulted in skin burns. Hence, it is prudent to purchase gemstones from reputable dealers. There should be no concern if the stone is purchased in highly developed countries, but the buyer should beware of such purchases in countries where radioactive testing is not required.

Medicine

A key to the longevity of our lives is modern medicine. Many of the medical products we now take for granted—either over-the-counter or via prescription—are only possible because of harnessed radiation.

Medications

If we take just a few minutes to survey our medicine cabinet, we can now marvel at the incredible number of medications currently available. Most would not be on those shelves without the radioisotope tracers employed in so many parts of the development, testing, and FDA approval process (Venkatesh and Korde, 2021). Carefully selected radioisotopes are routinely used to trace the medical products through the body to be sure the product goes to the proper organs for treatment and that they accomplish the mission intended.

Even the food supplements, conveniently packaged for our use, likely benefited from nutrient tracing techniques afforded by the use of radioisotopes. It is estimated that nearly 80% of the prescriptions and over-the-counter medications and food supplements that we now take for granted would not be available without the testing conducted by the FDA that rely on radiation tracer technology—to be assured that the products accomplish their advertised goals in a safe and effective manner (Fig. 3).



Fig. 3 A typical array of pharmaceuticals at the local drug store (Waltar, 2004).

Sterilization

Customer demand for safe, sterilized products such as bandages, gauze, tissue grafts, burn dressings, surgical dressings, heart valves, plastic and rubber gloves, surgical instruments, etc. continues to increase—especially given the COVID-19 pandemic. The established methods of sterilization include (1) high temperature/pressure sterilization (by dry heat or moist heat); (2) chemical sterilization (e.g., ethylene oxide gas); (3) filtration; and (4) radiation sterilization (Gamma/Electron beam).

In selecting the most appropriate sterilization process, careful consideration needs to be given to ensure that product quality is maintained. The method of choice should not affect polymers, seal strength, label and box adhesion, corrugated and paperboard strength of the container, and also the product's color. Radiation sterilization, being a cold process, helps address these issues more effectively compared to other sterilization methods. Furthermore, this facilitates sterilization of pre-packed, ready-to-use products without any quarantine restrictions. Accordingly, radiation sterilization is becoming the preferred technology for commercial use. See (Kalia and Joseph, 2021) for details.

Routine X-ray applications

Most of us are familiar with the ubiquitous use of X-rays to determine the location of broken bones, tooth cavity detection, etc. This technology is simply taken for granted and the refinements now incorporated in modern X-ray machines are consistently able to provide the diagnostics necessary for an exceptionally small radiation dose.

Attention to serious diseases

Many of us are becoming aware of more advanced applications of nuclear technology for both the diagnosis and treatment of serious diseases. The bulk of this radiation technology to date has been focused on diagnosis, rather than therapy, although applications to the curing of serious diseases is making great progress. See (Venkatesh and Kang, 2021; Kang and Venkatesh, 2021; Shrivastava and Venkatesh, 2021) for substantially more in-depth coverage.

Diagnosis techniques

It is now possible to utilize radiation devices to provide excellent diagnostics of diseases such as cancer, heart, liver, and other serious ailments using tools such as SPECT, PET, and MRI scans. The fundamentals of each are as follows:

- **SPECT** (single photon emission computed tomography): This technique is based on injecting the patient with a radioactive substance, such as Technetium-99^m, and then detecting the traverse of this substance through the body to discover the “hot” spots (i.e., identify the location of the malignancies, where the rate of oxygen intake activity is especially high). SPECT is widely used for routine clinical work because it is relatively inexpensive and utilizes radioisotopes available from nuclear reactors.
- **PET** (positron emission tomography): This technique is based on injecting a different type of a radioactive substance, a positron emitter, into the body and utilizing coincidence counters surrounding the body to isolate the malignancy. PET systems tend to be more expensive than SPECT systems, partly because of the sophistication of the counting system and partly because the radioisotopes that emit positrons typically have a very short half-life (minutes or hours). This latter constraint means that the

equipment for creating the required radioisotopes must be located geographically close to the site of application. However, this technique is being increasingly used because of the preciseness of its diagnostic capability.

- **MRI** (magnetic resonance imaging): This technique is based on applying a strong magnetic field to flip the spin of a hydrogen atom within the imaged organ and utilize the resulting signal for diagnosis. This process does not involve ionizing radiation, but it does depend upon nuclear technology. It is particularly helpful in identifying the density of bone marrow and tissue, thereby obtaining excellent cross-sectional images of the body and its associated organs.

Therapeutic techniques

Much progress has been made over the past few decades to use radiotechnology for either curing serious diseases or at least providing comfort over much longer times than would otherwise be possible.

Most of us are aware of radiation treatments for several modes of cancer. Very high levels of radiation, normally X-ray or gamma ray, are focused on the malignancy to kill the cancerous cells. This can be done via external irradiation (teletherapy), i.e., placing the patient in a prone position and moving the radiation beam around the patient (taking care to focus the beam on the target area and minimize damage to surrounding healthy cells) or placing radioisotopes inside the body (brachytherapy) to get the radiation as near as possible to the malignancy. Brachytherapy normally consists of X-ray, gamma ray, or beta ray radioisotopes. However, an evolving new approach is to utilize alpha emitters, since alpha particles pack a substantially higher potency at close range. This appears particularly promising for diseases such as leukemia, where the malignant cells are not confined to a particular area of the body.

A rather astounding fact is that one out of every three patients entering a U.S. hospital or medical clinic derives direct benefits from nuclear radiation medical treatments. This is a rather stark reminder of how much we really owe to the medical benefits of harnessed radiation.

Food

One of the most significant advances in our global food supply over the past century is the use of radiation to develop and improve the production of a huge variety of foods that we now take for granted. As detailed in Chapter 6 (Sivasankar et al., 2020), basic foods such as new grains and vegetables have evolved over eons of time on a continuing (but slow) basis as they have adapted to their changing environment.

But, radiation techniques have been developed and put into full production to accelerate this natural mutation process by irradiating seeds or stocks, with resulting varieties that have substantially better yield, better disease resistance, better flavor and color, more efficient use of fertilizer, etc. (Sivasankar et al., 2020). In fact, from an economic perspective, radiation mutation processes have resulted in by far the largest economic impact to scores of nations around the world than from any other application of radiation technology. We are now blessed with an abundance of food varieties that are able to grow effectively in almost any environment. This technology will undoubtedly become even more important as we seek ways for field produce to survive and thrive to meet global climate change conditions.

When going to our local store or supermarket, we seldom worry about the quality of food we purchase. Why? For the most part, all products on the shelves have been sanitized and scrutinized to assure us that the food is safe to eat. Most raw products have gone through various screening processes and many arrive at the store wrapped by sanitized materials that have been irradiated to ensure cleanliness. As noted below, some of the products have been irradiated to ensure long shelf-life (Pillai and Pillai, 2021).

We can fry our eggs to our perfection on a pan equipped with a special coating (developed by a radiation cross-linking process) to inhibit sticking. Radiation gages likely determined the thickness of this coating and assured that it was properly adhered to the metal. The silverware used during our meal undoubtedly benefited from radiation thickness gages during the making of the sheet metal from which the utensil were subsequently stamped. Further, we can be assured that our full packages of breakfast cereal were weighed correctly or precisely measured by radiation leveling or density gages (Thereska and Brisset, 2020) to ensure that the weight and volume stated on the cardboard container were accurate (Fig. 4).

Despite the positive situation above, there remains caution for some foods, such as fish, poultry, and red meat—particularly for open air markets—since some bacteria and pathogens remain hidden and can pose a health risk if not properly cooked. Some estimates indicate one-third to one-half of all food harvested in the field becomes spoiled before reaching the dinner plate. In fact, this spoilage can reach up to 90% for fish. We can look forward to the day when there is sufficient public acceptance of food irradiation to further improve the safety of such products (Pillai and Pillai, 2021).

But even now, several food products benefit from irradiation prior to reaching our kitchen. Most spices are irradiated to kill unwanted pathogens, and numerous other products are undergoing approval by the Food and Drug Association (FDA) in the U.S. As one small example, sealed containers of the creamer that we may pour onto our coffee or tea can remain on our shelf for long periods without refrigeration because the container was irradiated prior to being filled in order to assure the absence of microbes. The shrink-wrap or aluminum foil that radiation processes help to produce provide convenient and safe ways to store unused food for consumption at a later time. Fortunately, food irradiation is slowing becoming more common—especially in many countries outside the U.S.

Fresh fruit and vegetables should constitute a significant portion of our daily diet. Such commodities cannot always be readily at hand during some seasons of the year. But, when available, we can be thankful for the much larger variety and volume of such foods

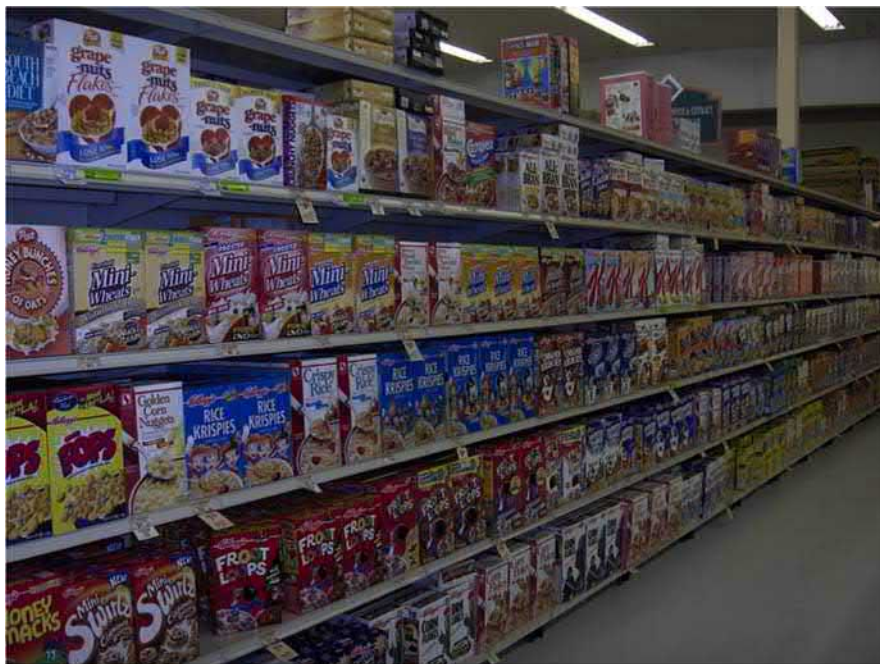


Fig. 4 Breakfast cereals, with proper fills assured via radiation gages (Waltar, 2004).

made possible by the greatly accelerated agricultural breeding processes now in practice, globally, by using either mutation or tracing processes (Sivasankar et al., 2020).

Much of the food on the grocery shelves is contained in bottles or cans. As noted earlier, a high volume of the glass and metal used in making these bottles and cans are made economically by radiation processes (e.g., improving the quality of glass). Radiation gages are often used to ensure proper thickness of many sheet metals used for manufacturing the cans. Given the numerous numbers of bottles and cans used in modern commerce, the employment of radiation devices to measure the proper filling and sealing processes for rapid automation is essential in keeping production costs to a minimum. We need not worry about whether our can of soda or beer has been filled to the proper level. That has been taken care of by the use of radiation level gages (Fig. 5).



Fig. 5 Bottled beverages—all assured of being filled to the brim via radiation gages (Waltar, 2004).

Personal conveniences

New products

A quick stroll through the modern cities of today bear little resemblance to the cities and towns of the “yesteryears.” Thick wooden plank floors of the “Old West” still have a lot of charm, but many of the floors now appearing in numerous office complexes, shopping malls, airports, and grocery stores have very different look. They are products of radiation cross-linked polymers. Such floors are not only beautiful, they are long-lasting (excellent abrasion resistance) and have low maintenance costs.

Radiation cross-linking is based on the effects of polymers being exposed to high-energy beta and gamma rays. In the process of cross-linking, when the material absorbs radiation energy, chemical bonds are broken and free radicals are formed. These free radicals then react to form new chemical bonds—forming an extremely resistant “network” (Fulvio, 2021).

Radiation cross-linking is basically suitable for all types of plastics that can be chemically cross-linked by using radical initiators such as peroxides. Unlike chemical cross-linking methods, radiation cross-linking takes place at low temperatures. The most voluminous commercial polymers to be radiation cross-linked are polyethylene (PE), polyamide (PA), polybutylene terephthalate (PBT), and polyvinyl chloride (PVC). The old process of chemical vulcanization of rubber for tires has largely been replaced by radiation cross-linking because the latter technique is done at low temperature and with greater precision and without fluctuations in quality.

This radiation technology has also been applied to developing new automotive primers and topcoats, and specialty coating for wood and paper.

Wood remains a highly valued material for making fine furniture, in addition to providing a key structural material for buildings. In order for such wood to provide long-term service without cracking, it is essential to remove the necessary amount of moisture before using. Neutron probes provide perhaps the most accurate way of determining when the wood has dried to the point for commercial use.

When cleaning furniture, floors, or pretty much everything around the house, new detergents and other cleaning solvents have emerged in a very cost-effective basis due the automation of mass production lines made possible by radiation level and density gages that allow the precise processing essential for quality products.

Transportation

The use of personal automobiles has transformed travel over the past century, and there is no indication that such mobility in freedom in travel will fade in the future. Knowing that our car will start instantly and provide reliable transportation for many, many miles is a luxury now taken for granted. But it was the use of radioisotopes to determine engine wear that was largely responsible for the fine-tuned engines now in common use. Plus, as outlined in (Thereska and Brisset, 2020), the materials used in the manufacture of the car were largely the result of radiation gages and measuring devices (i.e., the glass, body shell, etc.) As noted above, tires are now normally vulcanized by radiation cross-linking—rather than the older sulfurization process. Further, the pavements that are now available for smooth driving experiences have benefited by radiation compaction instruments prior to the final surfacing (Bjornstad, 2021).

Of course, any mode of travel involves the element of time. It may come as a surprise to some readers that the most precise clock ever invented and put into commercial use is the atomic clock. This device measures the electromagnetic signal that electrons in atoms emit when they change energy levels. Since 1968, the International System of Units (SI) has defined the second by measuring two energy levels of the ground state of the cesium-133 atom. It is accurate to within 10^{-9} s per day (approximately one part in 10^{14}) (https://en.wikipedia.org/wiki/Atomic_Clock).

Entertainment

Another great convenience in our daily life is to keep up with the news and be able to watch TV or listen to the radio for further entertainment. One of the main reasons that newsprint or magazine print is relatively inexpensive is due to the use of radiation gages in the high-speed automation in the production of the paper itself for the automation processes employed in making paper (Thereska and Brisset, 2020). Both the radio and television sets benefit from the wires made possible by radiation-treated heat-shrink insulation.

But even more common as we go through the day is our unchallenged addiction to computers and cell phones. The basic ingredient in both devices is the microchip, which requires specialty doped semiconductors to provide their power and convenience. The base material for the semiconductors is silicon, one of the most abundant elements on the planet. But raw silicon is neither a perfect insulator nor a perfect conductor. Rather, it is somewhere in between. By imbedding a smattering of elements such as boron, phosphorus, arsenic, gallium, or others into the silicon matrix, the flow of electricity through the silicon can be controlled. This allows the production of transistors, the building block for the production of microchips.

In order to make these specialty semiconductors, particle accelerators are used to propel these charged elements (ions) into the silicon matrix, forming N-type materials (molecules having an excess of free electrons) and P-type materials (molecular structural holes requiring free electrons). These units are then assembled into the transistors and microchips we require for use on a daily basis.

New technologies are being investigated for developing self-charging batteries, using radioactive carbon-14, that may 1 day allow users to keep cell phones charged essentially forever (<https://newatlas.com/energy/nano-diamond-self-charging-batteries-ndb/>).

One of the pleasures of modern life is to enjoy great art. Although we may not be aware of it, precious paintings are always vulnerable to forgery—as well as deterioration due to environmental factors. Hence, radiation has been used on a rather frequent basis to assure us that the paintings advertised as originals are indeed the real thing, and that they can be well preserved (Varquez and Nagai, 2021).

Home and garden

One of the marks of an inviting home is the incorporation of flowers in and around the house. Nothing stimulates our sense of sight and smell like the beauty and fragrance of a bouquet of a dozen roses or a flowering patch of multi-colored rhododendrons. Using the same radiation mutation technology outlined in the food section above, an incredible number of new flowers have been developed globally over the past several decades. The worldwide floriculture (flower) industry is estimated to be worth \$70 billion a year, and it is growing rapidly, especially in the developing nations. The actual value of this global industry will likely never be known very accurately, because much of this work has proprietary value. Developers are not generally willing to reveal their secrets on how, for instance, they have been able to utilize radiation to develop varieties of flowers having creamy strips with colors ranging from yellow to white, purple, yellow orange, and orange.

Any good gardener knows that the proper fertilizer is key to growing both fruits and vegetables, along with hearty flowers. The irradiation of sewage to both disinfect the raw effluent AND provide useable fertilizer in the process bodes well for the Master Gardener wishing to grow productive plants consistent with environmental stewardship.

One of the most common supplies in any home shop is sandpaper. This is another product we simply take for granted without realizing that there are normally three processing steps in the manufacture of sandpaper based on radiation technology: radiation thickness (depth) gages to (1) determine the proper thickness of the underlying paper, (2) determine the appropriate thickness of the glue, and (3) determine the size and thickness of the grit being bonded to the paper (Fig. 6).

Personal safety

Daily hazards

Perhaps the most noticeable application of radioisotopes to modern home and office safety is the common smoke detector. This device contains a small amount of a special radioisotope (often americium-241, although plutonium-239 has been used in numerous smoke alarms in Russia) that constantly emits alpha particles into an ion chamber. During normal atmospheric conditions, a small but steady electric ion current is actively recorded. However, if a fire occurs, smoke particles enter the chamber and reduce the ion current. The change in current triggers the smoke alarm. These compact smoke alarm devices have saved untold lives as well as valuable property throughout the world (Fig. 7).

Crowded theaters represent a hallmark of successful movies, live performances, concerts, and other artistic gatherings. But any time a throng of people becomes wedged into cramped quarters, there is always the potential for fire or other safety threats that provide an impetus for prompt evacuation. In such situations, it is comforting to know that the omnipresent exit sign will always stay illuminated, even in a blackened theater, because it is powered by radiation. By employing the 24-h reliability of the



Fig. 6 Sandpaper—made via three-step radiation manufacturing process (Waltar, 2004).

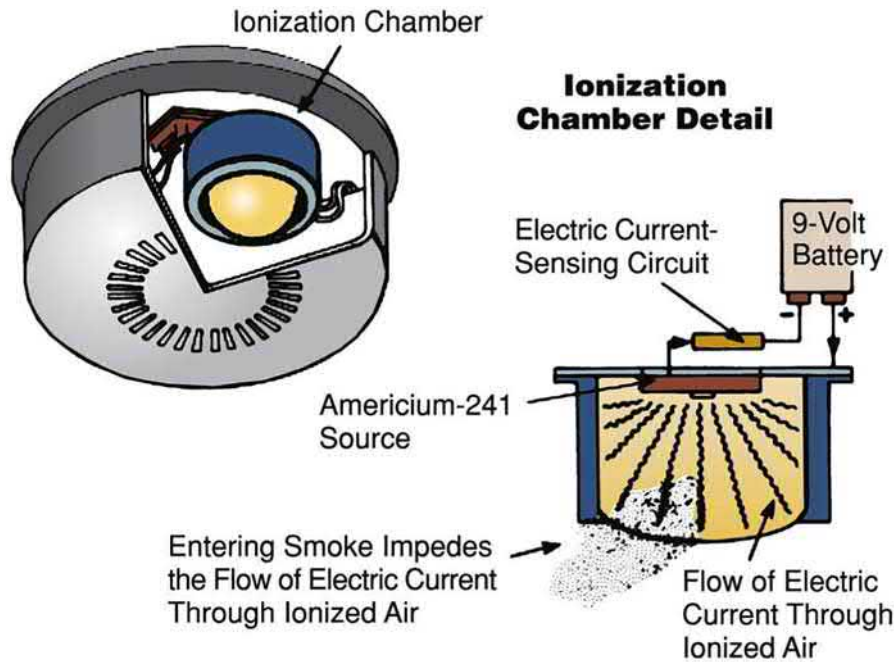


Fig. 7 The common household smoke detector (Waltar, 2004).



Fig. 8 A typical exit sign powered by radiation (Waltar, 2004).

luminescence property of radiation, these signs powered by radiation can be relied upon to operate under any and all circumstances. The glass tube spelling "exit" is painted with a luminescent material and is filled with tritium gas. The constantly emanating electrons from the tritium impact this special paint and cause the emission of light visible to the human eye. Such signs are now located in most public buildings in the Western world and have undoubtedly saved many lives when buildings need to be evacuated during power failures and other hazardous conditions (Fig. 8).

Airport screening

All of us are acutely aware of the passenger and luggage screenings that now routinely takes place in all modern airports around the world. Although this process is at times annoying, we recognize the need for taking such measures—especially since the terrorist attacks of 9/11. X-ray machines are an all-too-familiar fixture through which we must pass before heading to our departure gate. However, many of these machines are capable of producing only pictures that often cannot distinguish between a block of plastic explosives and a wedge of cheese. They did not prevent the hijackers from boarding airplanes on the morning of September 11, 2001 and carrying out their morbid plans. This placed considerably more reliance on the detection operations than is desirable—and it is why more-advanced detection machines are being designed for implementation.

One fairly recent development is the application of neutron activation devices to detect hidden explosives. A technique now employed in many airports is to direct a beam of neutrons on samples taken from passenger luggage. This transmutes a material such as nitrogen (a common ingredient in explosives) into a radioactive isotope that emits a characteristic radiation that can be

detected. Whereas this provides positive identification of suspicious materials, this procedure is generally not suitable for high-speed throughput of luggage inspections. Rather, such techniques are more applicable for border security operations (Strellis and Gozani, 2020). Hence, most airports currently utilize an ion mobility spectrographic system. In many cases, a beta emitter such as nickel-63 is used to create the ionization required for sample investigations. The most likely procedure used when airport security personnel take a “swipe” around your personal laptop computer or other carry-on luggage is based on X-ray backscattering (<https://www.epa.gov/radtown/radiation-and-airport-security-scanning#about-radiation>). We are comforted to know that a wide variety of radiation processes can be used to ensure our protection from terrorists when we board a commercial aircraft.

Airplane wings and other structures

It is difficult to imagine how 21st-century life could be accommodated without the convenience of air travel. We might put quotes around the word “convenience.” This mode of transportation has become considerably more cumbersome since the terrorist attacks of 9/11. Still, given the hectic pace of modern life, the time required to traverse the states in a train or cross the ocean in a ship for reasons other than pure leisure have become unthinkable. Air travel has transformed modern life, and there is no turning back.

Whereas air crashes still do occur, the reality is that air travel is by far the safest mode of transportation (as measured by life-lost per mile). Such a safety record seems counterintuitive, given the speed of such vehicles and the obvious concerns should some type of major failure occur while at 30,000 ft.

Certainly, there are significant safety factors built into the design of such aircraft, knowing that storms and other acts of nature can bounce these airfoils around. But logic dictates there are practical limits to the thickness of materials that can be used in making the wings and the body, lest the airplane be so heavy it will never leave the ground. Hence, it has long been a substantial comfort to know that the critical welds on those aircraft are routinely inspected by radiography. All commercially licensed aircraft have regularly scheduled inspections in which either neutron or gamma irradiation can be employed to inspect crucial areas such as the welds that attach the wings to the body. Any flaws revealed by this system result in immediate attention for appropriate repair. A popular source of neutrons for accomplishing this task is californium-252, a spontaneous neutron emitter that does not occur in nature but is made in nuclear reactors for such purposes. Neutrons are often needed for military aircraft that employ composite construction techniques, whereas gamma sources work well for the metallic construction normally employed for commercial aircraft.

Other structures in our normal travel patterns that demand careful attention to welds are the steel bridges that we count on for safety on our nation’s highways. There are now over 615,000 bridges in the US, the majority of which rely on steel structures for stability. Here again, radiation technology is periodically employed to ensure the structural reliability of key welds.

Runway lights

Another unique application of radiation is in providing the source of energy for airport runway lights. Given the possibility of a blackout at airports, one of the special safety features is using tritium to invoke the luminescent property of radiation [illustrated in Fig. 9 of (Waltar, 2021)] to assure a 24-h reliable light source—independent of weather or other confounding factors. Krypton-85 is also used for this purpose. Clearly, radiation provides a major contribution to the safety of modern air travel.

Safety from crime

Crime has undoubtedly been a distasteful part of humanity as long as history itself. And with the advancements in technology, crime has become increasingly more sophisticated. Consequently, it is necessary to employ new techniques in order to both thwart criminal activity and provide ways to track down and convict the offenders.

As might be expected, the activation property of radiation is a strong ally to aid in criminal investigations. Being able to trace the paths of guns, knives, and personal effects such as gloves and shoes can be exceptionally helpful in solving difficult criminal cases. Even trace amounts of “tell-tale signs” can be analyzed by using radiation techniques to help bring about justice. By sampling only a small amount of dirt on the shoes of a suspect in an appropriate radiation detector, the precise ingredients of the sample can often be matched to other soil samples to determine where the suspect previously has been.

Neutron activation analysis (NAA) was used as early as the 1950s by criminologists. This process involves placing a sample of evidence found at the crime scene into a nuclear reactor and irradiating it with neutrons (Fedchenko, 2021). The neutrons collide with components of trace elements contained in the sample, causing them to be transmuted into other isotopes that subsequently decay and emit gamma radiation of a characteristic energy level. This allows accurate identification of many constituents in the sample. The amazing thing about this procedure is that it will work for a small sample of hair. As many as a dozen or so different elements can be identified in a single strand of hair.

DNA testing

One modern technique that is being used more frequently to bring criminals to justice is DNA analysis. Exceptionally minute quantities of “residuals” associated with suspected persons can be used to provide positive identification. By matching residuals at the scene of the crime (such as blood stains, saliva, semen, or hair) with the suspect, prosecutors have powerful evidence in their arsenal.

Such analyzes have been used in famous cases and in other circumstances, such as in the O. J. Simpson murder trial, the President Clinton incident with Monica Lewinsky, and the positive identification of the deaths of Saddam Hussein's eldest sons. What may not be known to many people is that radiation often plays a key role in the analysis.

"DNA fingerprinting" has gained considerable popularity and glamor in the recent past because we now know that every person on Earth has his or her own unique variation of the DNA molecule—the only exception being identical twins. This affords forensic experts the possibility of providing a positive linking of crime scene trace DNA residue to only one person in the universe. However, the processes involved in "DNA fingerprinting" are quite complex (Bowling, 1996). Hence, DNA analyzes have sometimes been discarded in courts of law because of the lack of precision in the sample collection or in the measurement techniques. Also, since DNA analyzes are time consuming and expensive to perform, other "fingerprinting" techniques are being developed for more frequent applications.

The techniques used for DNA fingerprinting are based either on the type of genes an individual has or upon the amount of reoccurrence of a certain pattern of nucleotides in his or her genetic material. The former approach to determining a DNA fingerprint is called the polymerase chain reaction (PCR) technique, wherein the gene types of an individual are matched. The latter approach is called the restriction fragment length protocol (RFLP) technique, wherein the repetition of nucleotide patterns within a chromosome is measured to obtain a DNA fingerprint.

The PCR method does not involve radiation techniques in the DNA matching process. Rather, it is based upon duplicating a single strand of DNA in an exponential fashion, such that 20 replication cycles yield about a million copies of the original DNA. These new copies are placed on a sheet of nylon. Gene probes, which bind to certain genes, are then introduced, and they will attach to specific genes on the DNA strand. A special indicator solution, unique to each gene probe, is then added, and when this indicator locates a probe that is attached to a "target gene," a blue dot appears on the nylon strip. This series of blue dots and blank misses is the "DNA fingerprint." The advantage of the PCR method is that it requires only a very small sample size—perhaps the size of the tip of a pin. Furthermore, it takes only about 2 or 3 weeks to complete the procedure. However, the disadvantage of the PCR method is that it is considerably less accurate than the RFLP method discussed below. The probability of error in matching individuals with the PCR sample may be from one in a hundred to one in two thousand. Whereas this might sound quite precise, it may be insufficient to convince a jury "beyond a reasonable doubt." In fact, it was largely due to this possibility of error that the analysis provided by this approach was discarded in the O. J. Simpson trial (Cohen, 1995).

The RFLP method is based on employing an enzyme that can locate a certain string of nucleotides in the DNA molecule. After cutting the DNA into a series of variable lengths, these pieces are put into a special type of gel at one end of a container that has electrical plates affixed to both ends. An electrical current is then passed through the solution in a process known as electrophoresis. The pieces of DNA move through the gel at different rates because of their different sizes. The gel is subsequently removed and bathed in a solution containing four types of radioactive probes, each of which binds to a different nucleotide on the end of the cut-up DNA pieces. A piece of X-ray film is then placed on top of the gel, and a radioactive tag, normally phosphorus-32, is then used to expose the film. The strips of white at different locations on the exposed film indicate how far the different pieces of DNA have moved. This pattern is the classic "DNA fingerprint."

This procedure can take a much longer time to complete than the PCR method, primarily because of the small amount of phosphorus-32 that is tagged to the DNA. It may require up to 2 weeks to significantly expose a piece of X-ray film. In addition, the film must be separately exposed for each of five or more areas being tested, each taking an additional week. Upon completion of the procedure, the film is often scanned into a computer to obtain precise measurements. All of this could take 3 or 4 months. However, the reward is that the accuracy is exceptionally high. It is from five to a thousand times more accurate than the PCR method. In fact, some of the experts at trials have testified that the error in this method, based on radioactive techniques, may be only about one in a hundred million. When a prosecutor tries a case and can show that the chance of the DNA's not being that of the defendant is one in a hundred million, he or she has very compelling evidence.

To sum up the DNA-fingerprinting approach, this area is getting increasing attention as a way to match the DNA found in substances at the scene of a crime with the unique DNA of a suspect. Whereas very precise matching can be done using the approach based on employing radioisotopes, the process is expensive and time consuming. Hence, nonradioactive techniques are becoming popular for routine investigations. But for situations where preciseness really matters, the RFLP method, which relies on radiation techniques for crucial steps, remains an important forensic tool. An error rate of only one part in a hundred million can free an innocent person or send a murderer to jail.

Border security

With the ever increasing inflow of international goods crossing our borders on a daily bases, it is comforting to know that increasingly sophisticated methods are being developed and employed at our borders to thwart the attempts by terrorists and others with criminal intentions to threaten our safety. Two chapters are devoted to this important task in this Section (Patton, 2021; Strellis and Gozani, 2020).

Communications

It was shortly after the shock of 9/11 that the United States became immobilized with the news that anthrax bacteria had been sprinkled on some of the letters entering the domestic mail system. Anthrax-laced letters appeared in the offices of key members of the US

Senate, causing complete evacuation of several federal office buildings for a substantial length of time. As officials scrambled for ways to screen mail to be sure it was free of this deadly “dust,” inquiries were made to ascertain whether electron-beam accelerators could be used to kill the anthrax, thereby cleansing the mail. Quick demonstrations resulted in a positive answer and the use of radiation was quickly adopted as a safe and effective measure for such application.

Whereas there clearly would be logistical difficulties in requiring similar sterilization for the full mail system of any major country, and certainly so for the United States, this system did work for the limited volume of mail needed to keep the wheels of government moving in the United States. Improvements on the system have been subsequently made to refine the process and avoid many of the initial problems, such as causing some of the paper to be slightly discolored (*Nuclear News*, 2003). Many of these systems now remain in standby for similar emergencies.

Summary

It is impossible to go through a day living in our modern world where radiation technology has not provided a significant contribution to our personal care, conveniences, or safety. From the air we breathe, the water we drink, the food we eat, the medicines we take, the clothing we wear, the mobility we take for granted, the entertainment we enjoy, our improved understanding of our cultural heritage, and the personal protection we take for granted, harnessed radiation is there as our unseen but powerful servant.

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Environmental Protection: Managing Fresh Water Resources

U. Saravana Kumar^a, ^a Isotope Hydrology Section, Division of Physical and Chemical Sciences, Department of Nuclear Sciences and Applications, International Atomic Energy Agency (IAEA), Vienna International Centre, Vienna, Austria

Md. Arzoo Ansari^b, ^b Isotope and Radiation Application Division, Bhabha Atomic Research Centre, Mumbai, India

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Glossary

Absorption The natural assimilation or incorporation of fluids into interstices, i.e. liquids in solids. In absorption, the dissolved molecules are incorporated within the structure of the solid (such as soil or a rock mass) and the fluid is held mechanically.

Activity The rate of disintegration (transformation) or decay of radioactive material per unit time.

Adsorption The surface retention of solid, liquid, or gas molecules, atoms, or ions by a solid or liquid advection. A process by which solutes (dissolved constituents) and/or heat are transported by the bulk motion of groundwater flow.

Alpha particle A positively charged particle (helium atom) ejected spontaneously from the nuclei of some radioactive elements.

Aquifer A water-bearing or saturated formation that is capable of serving as a groundwater reservoir supplying enough water to satisfy a particular demand, as in a body of rock that is sufficiently permeable to conduct groundwater and to yield economically significant quantities of water to wells and springs.

Argillaceous formation Sedimentary formation (shale) that has fine grained clay minerals.

Becquerel (Bq) 1 Bq represents a rate of radioactive decay equal to 1 disintegration per second, and 37 billion (3.7×10^{10}) Bq equals 1 curie (Ci).

Beta particle A charged particle (with a mass equal to $1/1837$ that of a proton) that is emitted from the nucleus of a radioactive element during radioactive decay (or disintegration) of an unstable atom. A negatively charged beta particle is identical to an electron, while a positively charged beta particle is called a positron.

Carbon-14 dating A method for dating the age of a particular material using the ratio of carbon-14 (^{14}C) to carbon-12 (^{12}C) based on the half-life of ^{14}C .

Carbonate A sediment formed by the organic or inorganic precipitation of carbonate minerals.

Condensate Water that has been produced by the cooling of steam in a condenser.

Dating of groundwater The assessment of relative concentrations of radioactive environmental isotopes, such as tritium (^3H), chlorofluorocarbons (CFCs), and carbon-14 (^{14}C), have been used to determine the age of groundwater.

Daughter products Isotopes that are formed by the radioactive decay of some other isotope.

Delta (δ) notation Part per thousand (‰; per mil).

Denitrification A redox process by which nitrate (NO_3^-) in anaerobic systems can be converted by bacteria to gaseous nitrogen species such as elemental nitrogen gas (N_2), nitrous oxide (N_2O), or nitric acid (HNO_3).

Deuterium An isotope of hydrogen with one proton and one neutron in the nucleus.

Discharge area The portion of the drainage basin in which the net saturated flow of groundwater is directed toward the water table.

Discharge velocity/discharge rate The speed at which a given volume of water flows through a given unit of the porous medium's total area perpendicular to the direction of flow.

Dissolution The process by which minerals dissolve into water.

Distillation The removal of impurities from a liquid by heating it and condensing the vapor.

Evaporation The conversion of a liquid to the vapor state by the addition of latent heat below the boiling point of the liquid.

Evaporite A sediment produced from seawater or salt lakes in tidal areas under arid conditions as a result of extensive evaporation.

Evapotranspiration The process whereby water is evaporated from plants, soils, and surface water bodies.

Freshwater Surface water or groundwater that is not composed of or influenced by saltwater or salt-bearing rocks.

Geochemical processes The chemical changes and reactions that groundwater undergoes as it moves through the subsurface.

Groundwater The excess soil moisture that saturates subsurface soil or rock and migrates downward under the influence of gravity.

Half-life The time in which one half of the atoms of a particular radioactive substance disintegrate into another nuclear form.

Hydrologic cycle The worldwide process, without a true beginning or end, that graphically or mathematically represents the movement of water from the oceans to the atmosphere, to land, and back to the oceans through the process of evaporation, precipitation, infiltration, transpiration, overland flow, base flow condensation, and runoff.

Hydrology The study (or science) of water including its circulation, distribution, and chemistry.

Hydrosphere The water of the earth (the sphere), segregated from the lithosphere (rocks of the earth), biosphere (biota of the earth), or atmosphere (air of the earth).

Infiltration The process by which water enters the subsurface.

Isotope Two or more forms (or atomic configurations) of a given element that have identical atomic numbers (the same number of protons in their nuclei) and the same or very similar chemical properties but different atomic masses (different numbers of neutrons in their nuclei) and distinct physical properties. ^{12}C , ^{13}C , ^{14}C are isotopes of the element carbon, and the numbers denote the approximate atomic masses.

Isotopic enrichment A process by which the relative abundance of the isotopes of a given element are altered, thus producing a form of the element that has been enriched in one particular isotope and depleted in its other isotopic forms.

Lake A freshwater or saline inland body of water enclosed within a depression of the earth's surface, either natural or man-made.

Lithology The scientific study of rocks.

Magmatic Water The water derived from a magma or exists within magma.

Metamorphic water Water produced during rock metamorphism.

Meteoric water Groundwater that originates in the atmosphere and reaches the zone of saturation by infiltration and percolation.

Meteoric waterline The relationship between deuterium ($\delta^2\text{H}$) and $\delta^{18}\text{O}$ concentrations in water from global precipitation surveys.

Methanogenesis The process of generation of methane by microbes.

Nitrification The inorganic process whereby oxygen is consumed in groundwater through the oxidation of ammonia (NH_4), producing nitrate (NO_3^-).

Noble gas A gaseous chemical element that does not readily enter into chemical combination with other elements. An inert gas. Examples are helium, argon, krypton, xenon, and radon.

Paleoclimate assessment To assess the past climates.

Paleowater Also known as fossil water. The ancient body of water present in an aquifer for millennia.

Parent A radionuclide that upon radioactive decay or disintegration yields a specific nuclide (the daughter).

Permeability/permeable The property or capacity of a porous media (soil, rock, or sediment) for transmitting a fluid.

Porosity A measure of the void (i.e., "empty") spaces in an aquifer. The quantity of water that an aquifer can hold in storage is determined by its porosity.

Precipitation The falling of water in liquid or solid form that occurs when the atmosphere becomes saturated with water vapor.

Radioactive tracer A radioactive isotope used to track or monitor the movement and pathway of water.

Radioisotope (Radionuclide) An unstable isotope of an element that decays or disintegrates spontaneously, thereby emitting radiation.

Raindrop The form that water attains as it travels in the atmosphere.

Rainfall A form of precipitation, that is, the quantity or measure of water discharged from the atmosphere within a given time period.

Recharge The addition of water to the saturated zone, either naturally (by rainfall, precipitation, or runoff) or artificially (by spreading zones or injection).

Relative humidity (RH) The actual volume of water vapor in a given volume of air divided by the amount that would be present if the air were saturated at the same temperature.

Residence time The amount of time that water resides within its source area, aquifer, watercourse, or water body before moving to another phase of the hydrologic cycle.

Runoff The portion of precipitation that eventually appears in surface watercourses or water bodies.

Salinity The concentration of total dissolved solids (TDS) in water or, more specifically, the total concentration of dissolved minerals or inorganic salts in water.

Soil moisture The quantity (volume) of water occupying a given volume of soil, expressed as a percentage.

Solute An organic or inorganic constituent dissolved within a solvent.

Stable isotope An isotope that does not undergo radioactive decay.

Stream A conduit in the water cycle, with a current confined within a bed and stream banks that acts as a mechanism for groundwater recharge or discharge and also serves as a corridor for fish and wildlife migration.

Tracer A material or a substance that is introduced into a chemical, biological, or physical medium and used to track or follow a travel or flow path and/or determine the travel time or mean velocity.

Unsaturated zone The area below the ground where the void spaces are filled with a mixture of water under pressure less than atmospheric, which includes water held by capillarity and air (gases) under atmospheric pressure.

Vapor or moisture The gaseous state of water.

Abbreviations

3D-GCM 3-dimensional General Circulation model

3D-GW 3-dimensional groundwater model

AMS Accelerator mass spectrometer

AR Activity ratio

ATTA Atom Trap Trace Analysis

CDT Canyon Diablo Troilite

CFC Chlorofluorocarbon

GCM General Circulation model

GMWL Global meteoric water line

GNIP Global Network of Isotopes in Precipitation

ITCZ Inter Tropical Convergence Zone

LMWL Local Meteoric Water Line

NBS National Bureau of Standards

RCM Regional Climate model

SMOB Standard Mean Ocean Bromide

SMOC Standard Mean Ocean Chloride

SST Sea surface temperature

TIMS Thermal ionization mass spectrometer
 TU Tritium unit
 USGS United States Geological Survey
 VSMOW Vienna Standard Mean Ocean Water

Introduction

Freshwater is extremely essential for human survival. Since the last century, phenomenal growth in freshwater use has been observed worldwide. With the dramatic increase in the world's population, a plethora of water-related challenges are raised today in several regions. Each region has its unique problems of quantity and quality of water depending on its geographic, climatic, geologic and socio-economic conditions (Ansari et al., 2016a). With the advent of global warming and climate change, the rainfall pattern is likely to change worldwide and will almost certainly have a negative impact on most of the freshwater resources of the region. Many modeling studies have postulated that by 2050, the distribution of freshwater around the world is approaching a paradigm shift (Alcamo et al., 2007). The large-scale exploitation of groundwater and surface water to meet the growing demands has caused the depletion and contamination of the water resources. This has already affected the lives of millions of people in many regions (Rajaveni et al., 2016; Bhanja et al., 2017; Noble and Ansari, 2019; Ansari et al., 2019). In recent decades, the growing need for freshwater for drinking, domestic use, irrigation and industries, combined with the increasing scarcity of water have been recognized as major challenges for sustainable development. This situation has caused conflicts, social unrest and changes in migration patterns.

The establishment of the Sustainable Development Goal (SDG 6) "Guaranteeing the availability and sustainable management of freshwater and sanitation for all", reflects the increased attention on water and sanitation issues in the global political agenda. To maximize its scope, the objective focuses on all aspects of the water cycle: water, sanitation and health (WASH); water resources management and efficient use of water; wastewater management; water quality and protection of freshwater ecosystems. Therefore, a comprehensive understanding of freshwater resources is necessary for their coherent and prudent management and for the sustainable resource development, which can be acquired through scientific research efforts. In this regard, isotopic data along with hydrochemical data, provides one of the important tools for achieving the goal.

With the development of sophisticated instruments to accurately measure the isotopes of hydrogen and oxygen in water, the introduction of isotope hydrology tools (around 1950) to solve hydrological problems was made, which became one of the most useful modern tools for hydrological assessment. Since then, it has been applied to solve a wide range of problems related to water (surface and groundwater) and environmental studies (atmospheric circulation, climatology and ecology) and is currently a well-established scientific discipline.

Application of isotope hydrology

Isotope hydrology techniques constitute the most promising modern tools for providing critical hydrologic information that sometimes cannot be obtained with other conventional techniques. Its applications include tracing the evolution of the water mass from its beginning in the form of precipitation, through its infiltration below the surface (recharge processes), and finally its manifestation in an aquifer. In addition to this, these techniques can also be used to trace the origin of solutes in groundwater. Isotopic techniques employed to study the different aspects related to water resources management are shown in Table 1.

Different types of isotopes used for hydrological studies

Applications of isotopes in hydrology are based on the general concept of "tracing", in which either artificial radioisotopes (e.g., ^{82}Br , ^{198}Au , ^3H , ^{137}Cs , etc.) are intentionally introduced or naturally occurring environmental stable isotopes (e.g., ^2H , ^{18}O , ^{13}C , ^{34}S , ^{15}N , ^{11}B , ^{87}Sr) or environmental radioactive isotopes (^3H , ^{14}C , ^{36}Cl , ^{222}Rn , ^{226}Ra , etc.) are used (Table 2). Environmental isotopes (radioactive or stable) clearly have the advantage over injected (artificial) tracers because they facilitate the study of various hydrological processes on a much larger spatial and temporal scale due to their natural distribution in a system. The use of artificial tracers is generally effective for local and site-specific applications.

Principles of isotope hydrology

Isotopic fractionation

The application of stable isotopes for the evaluation of hydrological processes is based on the understanding of variations in the isotopic composition of precipitation and surface waters (Clark and Fritz, 1997; Kumar et al., 2010a; Ansari et al., 2018, 2020). At each stage of the hydrological cycle, a small change is recorded by natural physical processes that have resulted in a difference

Table 1 Application of isotope hydrology techniques in various fields.

<i>Surface water</i>	<i>Ground water</i>	<i>Meteorology</i>
Hydrographs separation	Soil moisture variation, movement and recharge	Variability of environmental conditions/climate change
River discharge measurements	Occurrence and recharge mechanism	Movement of clouds and variability in precipitation
Dynamics of lakes and reservoirs	Groundwater flow velocity and direction	Environmental pollution and mechanisms
Water balance	Identification of recharge sources and areas of deeper aquifers and springs	Prediction of arrival and retreat of monsoon
Leakage through dams, seepage losses from irrigation channels	Interconnections between groundwater bodies	
Sedimentation rate, soil erosion rate	Mixing and distribution	
Evaporation/evapotranspiration	Effectiveness of artificial recharge measures	
Surface water and groundwater interaction, sea water intrusion	Data on lithology, porosity, permeability, hydraulic conductivity, transmissivity of aquifers	
Sources and tracing of pollutants	Pollution source and mechanism, groundwater salinization	
Snow and glacial melt runoff	Groundwater dating, paleowaters, infiltration rate, rock-water interaction, geothermal investigations	

Table 2 Isotopes commonly used for different applications in hydrology.

<i>Environmental stable isotopes</i>			
<i>Isotope</i>	<i>Ratio</i>	<i>Reference (ppm)</i>	<i>Application in hydrology</i>
^2H	$^2\text{H}/^1\text{H}$	VSMOW (155)	Source and origin of water, identification of recharge area, surface water-groundwater interactions, groundwater salinization, inter aquifer connection, recycling of irrigation water, geothermal studies
^{13}C	$^{13}\text{C}/^{12}\text{C}$	VPDB	Source of carbonates, groundwater dynamics, rock-water interactions, identification of pollution and groundwater dating (^{14}C) models
^{15}N	$^{15}\text{N}/^{14}\text{N}$	Air N_2 (3677)	Source of pollution, origin of nitrates, microbial denitrification processes
^{18}O	$^{18}\text{O}/^{16}\text{O}$	VSMOW (2005)	Source and origin of water, identification of recharge area, surface water-groundwater interactions, groundwater salinization, inter aquifer connections, recycling of irrigation water, geothermal studies
^{34}S	$^{34}\text{S}/^{32}\text{S}$	CDT	Origin of salinity, identification of pollution sources, redox condition of aquifer, geothermal water investigations
^{37}Cl	$^{37}\text{Cl}/^{35}\text{Cl}$	SMOC (0.324)	Source of pollution, sources of salinity
^{87}Sr	$^{87}\text{Sr}/^{86}\text{Sr}$	USGS <i>Tridacna</i> ,	Provenance of water
^{11}B	$^{11}\text{B}/^{10}\text{B}$	NBS 951—(sodium borate)	Source of pollution/salinity
^{81}Br	$^{81}\text{Br}/^{79}\text{Br}$	SMOB	Origin and evolution of groundwater
^3He	$^3\text{He}/^4\text{He}$	Atmospheric He	Ages of young groundwater
Environmental radioisotopes			
Isotope	Half-life (years)	Type of radiation	Applications in hydrology
^3H	12.3	B	Young groundwater dating, identification of modern recharge, groundwater movement, infiltration rate in unsaturated zones
^{14}C	5730	B	Old groundwater dating, groundwater movement, identification of paleowaters
^{36}Cl	3.1×10^5	B	Very old groundwater dating
^{137}Cs	30	Γ	Sedimentation/deposition rate in watershed, lake, reservoir, water courses etc., estimation of erosion rates
^{222}Rn	3.8 days	α	Surface water-groundwater interaction
^{226}Rn	1600	α	Sub-marine groundwater discharge
^{234}U	246,000	α	Groundwater dynamics and weathering processes
^{39}Ar	269	B	Dating of groundwater
Artificial radioisotopes			
Isotope	Half-life	Chemical from	Applications in hydrology
^3H	12.3 year	HTO	Groundwater recharge rate and flow direction
^{60}Co	5.3 year	$\text{K}_3[\text{Co}(\text{CN})_6]$	Groundwater recharge rate, canal and river flow rate
^{82}Br	36 h	NH_4Br	Groundwater velocity, effluent dispersion
^{198}Au	2.7 day	HAuCl_3	Seepage entry and exit points in dams
^{131}I	8.04 day	I^-	Canal and river flow rate, measurement of aquifer parameters (velocity, hydraulic conductivity, porosity, transmissivity)
^{51}Cr	27.7 day	EDTA complex	Canal and river flow rate

in the isotopic composition in water that is as unique as a fingerprint (Clark and Fritz, 1997). The important processes that are responsible for these variations in the stable isotopic composition of water are the isotopic fractionation that occurs during physical and chemical processes. Evaporation of water and condensation of vapor are examples of the physical process that cause isotopic fractionation, while chemical fractionation occurs due to chemical bonding, because the heavy isotopes have stronger bonds than those of light isotopes.

Fractionation can be caused during equilibrium and non-equilibrium conditions. Evaporation occurs mainly under non-equilibrium conditions, while condensation is an equilibrium process. When seawater evaporates, the lighter isotopes of oxygen and hydrogen are preferably converted to vapor, while during the condensation process, the heavier isotopes are preferably condensed, leaving the residual vapor more and more depleted in heavy isotopes. Both these physical processes cause a variation in the isotopic composition of the water molecules. Fig. 1 shows the cooling of the precipitating air mass with an initial isotopic composition of vapor ($\delta^{18}\text{O} = -10\text{‰}$), the same as that of fresh evaporate generated in the ocean. The difference between the vapor (red) and condensate (blue) curves in Fig. 1 shows an enrichment of about 10‰ when the condensate turns into a liquid in the vapor reservoir due to the preferential condensation of heavy isotopes. As the rainout process continues with the decreasing residual vapor fraction (f), this preferential removal of heavy isotopes continues, causing gradual depletion of heavy isotopes in the vapor and the condensate that is formed. The $\delta^{18}\text{O}$ content of different components of the hydrological cycle are shown in Fig. 2.

Isotopic fractionation generally brings small differences in the isotopic composition and the rare isotopes abundance of these elements is low. Therefore, the absolute abundance of isotopes is generally not measured. Instead, the measurements are reported in parts per thousand (‰ ; per mil) as delta (δ) notation, which is the deviation from the standard reference, V-SMOW (Vienna-Standard Mean Ocean Water). The δ values are calculated using (Coplen, 1996):

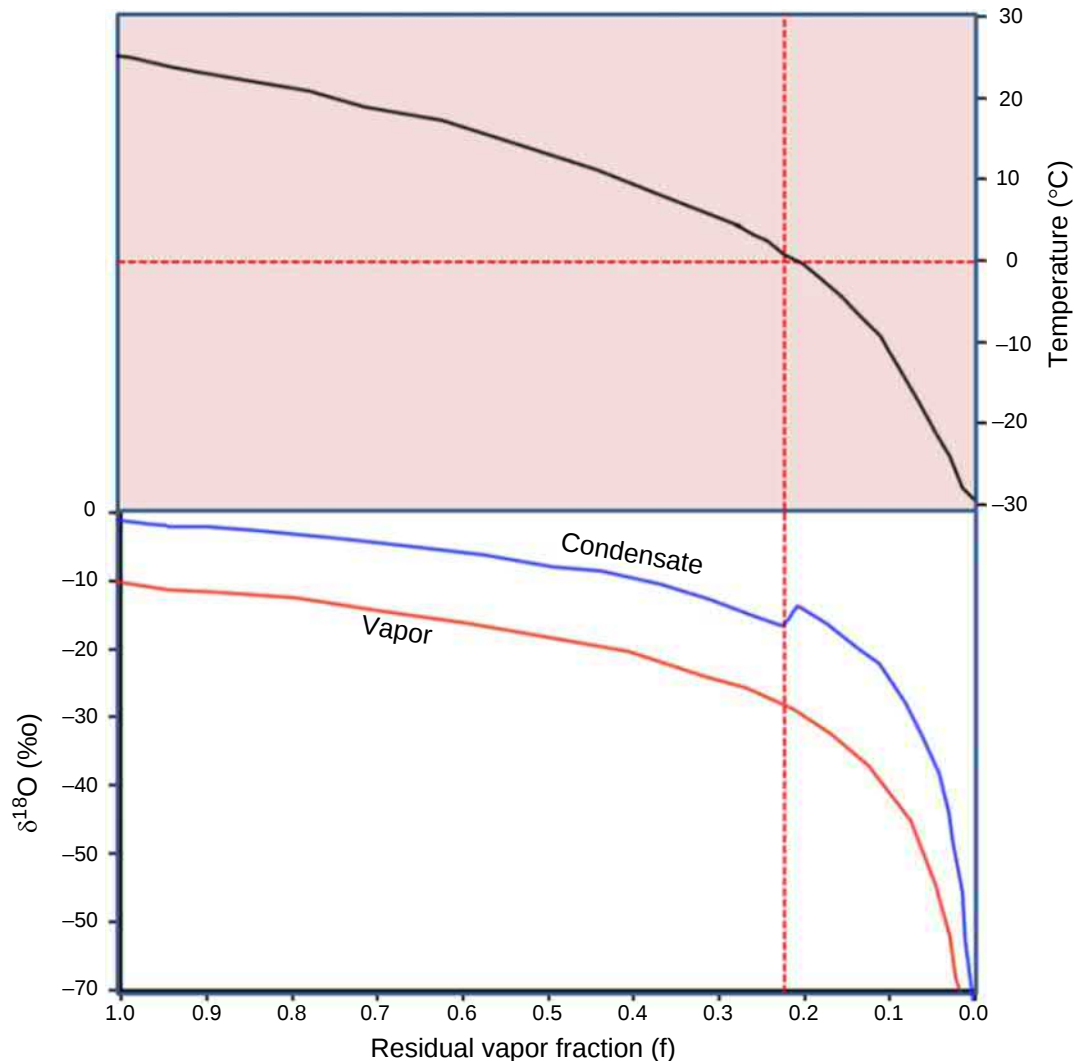


Fig. 1 Isotopic values of condensate (precipitating air mass) and vapor depleting as per the Rayleigh model. Modified based on Clark ID and Fritz P (1997) *Environmental Isotopes in Hydrogeology*. Boca Raton, FL: Lewis Publishers.

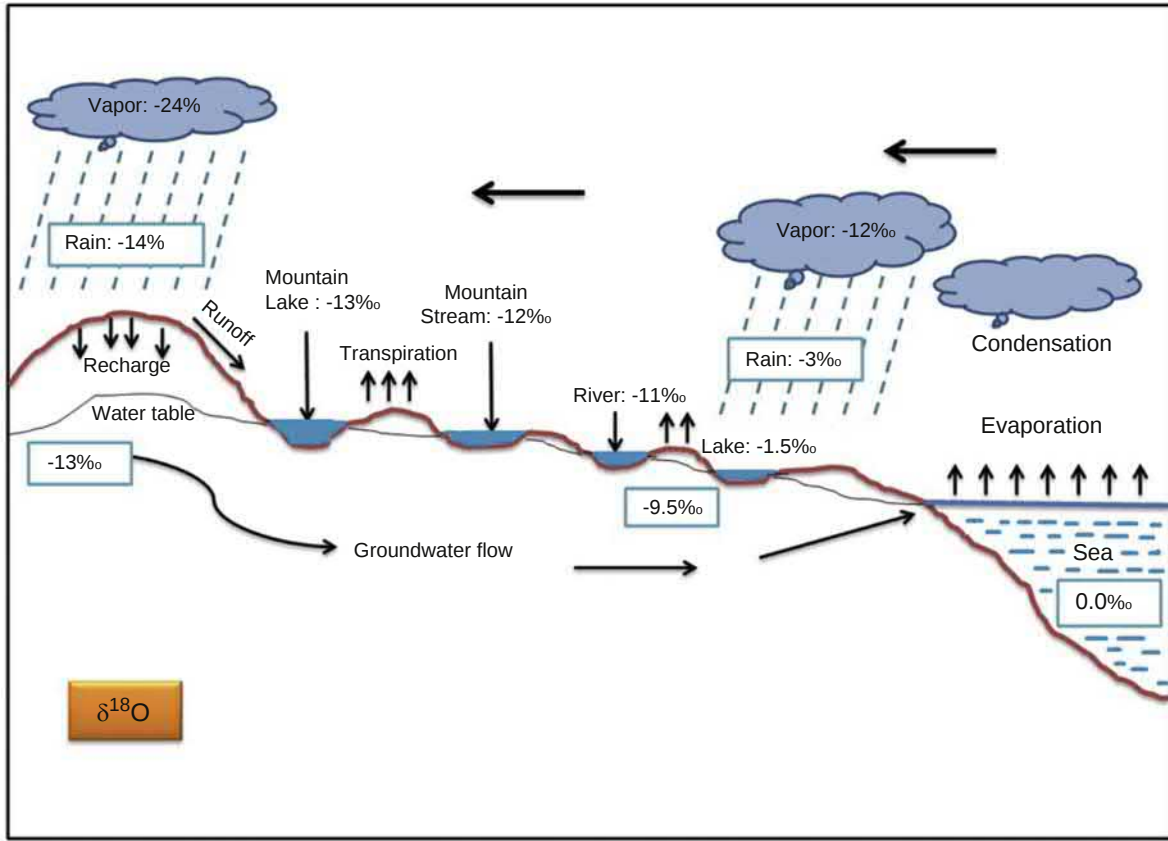


Fig. 2 $\delta^{18}\text{O}$ in different components of the hydrological cycle.

$$\delta(\text{‰}) = \left(\frac{R_x}{R_s} - 1 \right) \times 1000 \quad (1)$$

where R denotes the ratio of heavy to light isotope (e.g., $\frac{^2\text{H}}{^1\text{H}}$ or $\frac{^{18}\text{O}}{^{16}\text{O}}$) and R_x and R_s are the ratios in the sample and standard respectively. The isotopic compositions of the water samples are measured using an isotope ratio mass spectrometer (IRMS). Over the past decade, many significant advances have been made in the field of isotopes measurement. Now, inexpensive laser absorption spectrometers (Crosson et al., 2002), satellite based measurements of the isotopic composition of moisture (Frankenberg et al., 2009), and ground based remote sensing using spectra (Rokotyan et al., 2014) are available to perform the same function. At the same time, significant progress has been noted in numerical modeling through widespread use of the isotope enabled General Circulation model (Risi et al., 2012) and limited area models (Blossey et al., 2010). The range of isotopes ($\delta^{18}\text{O}$) of different hydrological bodies is illustrated in Fig. 3.

Global meteoric water line (GMWL)

The isotopic composition of precipitation measured worldwide has shown that stable isotopes ($\delta^{18}\text{O}$ and $\delta^2\text{H}$) values fall along a straight line on a $\delta^{18}\text{O}$ - $\delta^2\text{H}$ plot (Craig, 1961). This best fit line, termed as the global meteoric water line (GMWL), is given as:

$$\delta^2\text{H} = 8\delta^{18}\text{O} + 10 \text{ (‰VSMOW)} \quad (2)$$

Rozanski et al. (1993) further refined the Craig's GMWL equation by adding precision to it based on isotope data of precipitation from 219 sites (Gat, 1981) established by the IAEA since 1953 under the GNIP (Global Network of Isotopes in Precipitation) program. The refined relationship between $\delta^{18}\text{O}$ and $\delta^2\text{H}$ in global precipitation is given as:

$$\delta^2\text{H} = 8.17 (\pm 0.07) \delta^{18}\text{O} + 11.27 (\pm 0.65) \text{ (‰VSMOW)} \quad (3)$$

Various effects, such as evaporation of source water, recycling of moisture, and secondary evaporation, can cause deviations of the precipitation isotopes of any location from the GMWL (Breitenbach et al., 2010; Chakraborty et al., 2016; Kumar et al., 2010b), which is described as the Local Meteoric Water Line (LMWL) for that location (Table 3) and $\delta^{18}\text{O}$ - $\delta^2\text{H}$ plots in Fig. 4.

The deviation in the slope and intercept of the GMWL provides information regarding non-equilibrium processes caused during evapotranspiration, precipitation and moisture transport.

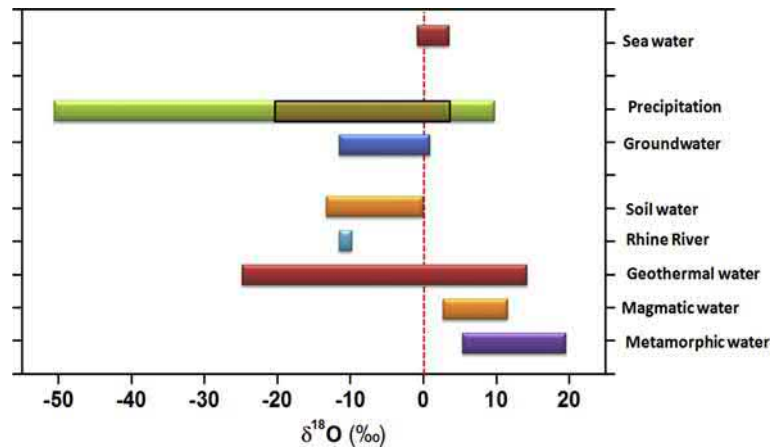


Fig. 3 Isotope ($\delta^{18}\text{O}$) range in the water cycle. Common values are marked with black a line. Modified based on Koniger P (2003) Tracerhydrologische Ansatz zur Bestimmung der Grundwasser-Neubildung. *Freiburger Schriften zur Hydrologie* 16: 89 p., Freiburg.; Clark ID and Fritz P (1997) *Environmental Isotopes in Hydrogeology*. Boca Raton, FL: Lewis Publishers.

Table 3 Regional meteoric water line of different region.

Region	Meteoric line (‰)	References
Global meteoric line	$\delta^2\text{H} = 8\delta^{18}\text{O} + 10$	Craig (1961)
Northern hemisphere (continental)	$\delta^2\text{H} = (8.1 \pm 1)\delta^{18}\text{O} + (11 \pm 1)$	Dansgaard (1964)
Mediterranean or Middle-East	$\delta^2\text{H} = 8\delta^{18}\text{O} + 22$	Gat (1971)
Northeastern Brazil	$\delta^2\text{H} = 6.4\delta^{18}\text{O} + 5.5$	Salati et al. (1980)
Northern Chile	$\delta^2\text{H} = 7.9\delta^{18}\text{O} + 9.5$	Fritz et al. (1979)
Tropical island	$\delta^2\text{H} = (4.6 \pm 0.4)\delta^{18}\text{O} + (0.1 \pm 1.6)$	Dansgaard (1964)
India	$\delta^2\text{H} = 7.93\delta^{18}\text{O} + 9.94$	Kumar et al. (2010a)
East India	$\delta^2\text{H} = 7.86\delta^{18}\text{O} + 8.81$	Ansari et al. (2020)
Thailand	$\delta^2\text{H} = 8\delta^{18}\text{O} + 9.4$	Wei et al. (2018)
Australia	$\delta^2\text{H} = 8.3\delta^{18}\text{O} + 14.1$	Hollins et al. (2018)

Oxygen-17 (^{17}O)-excess

Researchers are currently using a triple isotopic composition of oxygen ($\delta^{17}\text{O}$ and $\delta^{18}\text{O}$) in meteoric water and marine water vapor for climatological and groundwater studies (Landais et al., 2006; Uemura et al., 2010). The $\Delta^{17}\text{O}$ anomaly is defined as ^{17}O excess by Barkan and Luz (2007) as;

$$\Delta^{17}\text{O} = \delta^{17}\text{O} - 0.528 \delta^{18}\text{O} \quad (4)$$

where δ' is the modified δ notation and is defined analogously for ^{17}O . The $\delta^{17}\text{O} = 1n(\delta^{17}\text{O} + 1)$ and $\delta^{18}\text{O} = 1n(\delta^{18}\text{O} + 1)$, and 0.528 is the slope of GMWL. Since the magnitude of $\Delta^{17}\text{O}$ is very small, it was multiplied by 10^6 and reported as per meg rather than per mil with respect to VSMOW.

Deuterium excess (d-excess)

The $\delta^{18}\text{O}$ and $\delta^2\text{H}$ relation has been further described by another parameter, known as deuterium excess (d-excess) by Dansgaard (1964) and that is defined as:

$$\text{d-excess} = \delta^2\text{H} - 8 \delta^{18}\text{O} \quad (5)$$

It is used as a measure of the kinetic fractionation that occurs under non-equilibrium conditions, such as the evaporation of vapor at the sea surface (Zhou and Li, 2018). It also indicates the thermal condition and the equilibrium situation of vapor during the formation of air mass over the sea and the continent. It retains the signature of the climatic conditions under which rain forms, such as relative humidity (RH), temperature, sea surface temperature (SST), and supersaturated conditions (Pfahl and Sodemann, 2014). d-excess values can be used to understand the environmental conditions of moisture sources and is mainly influenced by the RH, SST, air temperature and sea surface isotopic composition at the moisture source region. It can also be used to understand the recycling of moisture over the continents and in the ocean (Rahul et al., 2018).

The value of d-excess is inversely correlated to RH and this is due to the molecular diffusion from the sea surface into the atmosphere at the air-sea interface, which is controlled by wind dynamics and temperature (Benetti et al., 2014; Rahul et al., 2018).

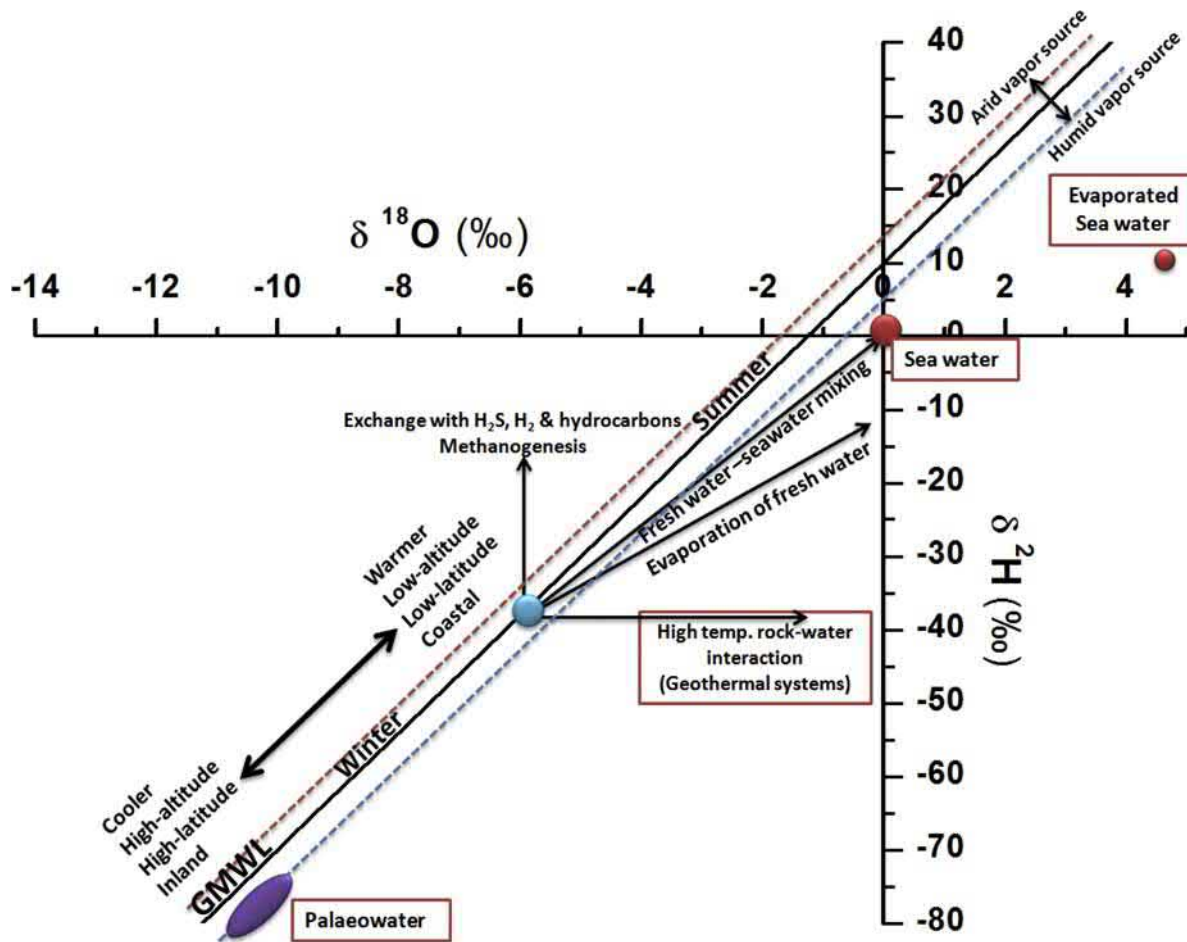


Fig.4 A graphic illustration of different water bodies falling along the global meteoric water line (GMWL).

d-excess is also positively correlated with sea surface temperature. That is because of the kinetic and equilibrium fractionation during phase changes (Munksgaard et al., 2015).

During the dry climatic conditions, the kinetic fractionation becomes robust because of the high evaporation. This is the reason why the residual water has lower d-excess, while the vapor produced has higher d-excess values and the precipitation from that vapor also has higher d-excess (Clark and Fritz, 1997; Kendall and McDonnell, 1998). During light precipitation, falling raindrops evaporate that result in lower d-excess in rainwater (Gupta and Deshpande, 2005).

The d-excess value is generally conserved in water vapor and the rain it produces will have the same values. However, it can change due to the sub-cloud evaporation or re-equilibration caused by the surrounding vapor when it descends to the surface (Balagizi et al., 2018). Therefore, d-excess acts as proxy and can be used to evaluate the climate change and meteorological conditions of any region. The average d-excess is 10‰ on the global scale.

Stable isotope effects

The isotopic composition ($\delta^{18}\text{O}$, $\delta^2\text{H}$) of precipitation has a unique characteristic, which varies with time and region. The isotopic signature of the precipitation depends on many factors or parameters such as temperature, relative humidity, isotopic values of moisture source, etc. When these factors are considered, the isotopic signatures provide information on the source and origin of moisture, precipitation and groundwater, as well as the climatic conditions at the time of recharge. Dansgaard (1964) observed that four important drivers for the isotopic variability in precipitation are distance from coast, altitude, rainfall amount, and latitude. These drivers depend principally on the temperature.

Altitude effects

The decrease in the isotopic composition of precipitation with increasing altitude is called altitude effects. This normally occurs during the orographic uplifting of humid air mass on the windward side of the mountain (Fig. 2). This depletion of the isotopic

composition may be due to the decrease in cloud temperature and the vapor saturation pressure with increasing elevation. A depletion of -0.15 to -0.6‰ in $\delta^{18}\text{O}$ has been observed per 100 m increase in elevation. The signature of altitude effects can be seen in groundwater in mountainous region and is used to distinguish its recharge at higher altitude (Kumar et al., 2012 & 2014). Therefore, it is useful for the hydrogeological studies.

Continental effects

The depletion of the isotopic ratio of precipitation with the increasing distance from the coast is called as continental effect. The continental effect is observed in precipitation, i.e. the precipitation is depleted in the heavier isotope of water molecules as we move away from the coastal parts. Since the oceans are the main source of moisture that causes most of the precipitation around the world, and as the moisture moves from its region of origin towards the interior of the continent, the rain out process depletes the precipitation in $\delta^{18}\text{O}$ and $\delta^2\text{H}$ with respect to the distance from the coast (Fig. 2). In addition, the depleted terrestrial moisture can be recycled, again depleting the isotopic ratios. As a result, coastal precipitation is isotopically enriched, while the inland regions receive isotopically depleted precipitation with strong seasonal differences. On average, there is approximately -2‰ depletion in $\delta^{18}\text{O}$ per 1000 km from the seacoast (Rozanski et al., 1993). Groundwater has a correspondingly depleted isotopic ratio as the distance from the coast is increased.

Amount effects

An area with a high amount of precipitation can sometimes have a significantly depleted isotopic composition. This amount effect is particularly observed in the region that is affected by the ITCZ and in hurricanes, as well as, (sometimes) in the convective precipitation regimes. The highly depleted isotopic ratio in heavy precipitation is caused by poor rainwater equilibration, which arises due to large rain droplets and high RH below the convective clouds, downdraughts and recycling of precipitation. This is also caused between the consecutive rain bands (Lawrence et al., 1998).

Temperature effects

The isotope ratios of precipitation are well correlated with the surface temperature. Dansgaard (1964) established its relation as:

$$\delta^{18}\text{O} = 0.62 T - 15.25 (\text{‰}) \quad (6)$$

This relationship between the stable isotopes and the temperature is currently used as a tool for the paleoclimate assessment of the region. The isotopic ratio of precipitation also depends on the temperature at which seawater evaporates. Seasonal variations in the isotopic ratios of local precipitation are caused by temperature fluctuations. The precipitation occurring during winter is isotopically depleted relative to the precipitation during the summer. In the geological past, when the climate was considerably different from today, the isotopic signatures of groundwater recharged at that time were very different from that of modern groundwater and precipitation. Fig. 5 illustrate how temperature influences the isotopic composition of any region.

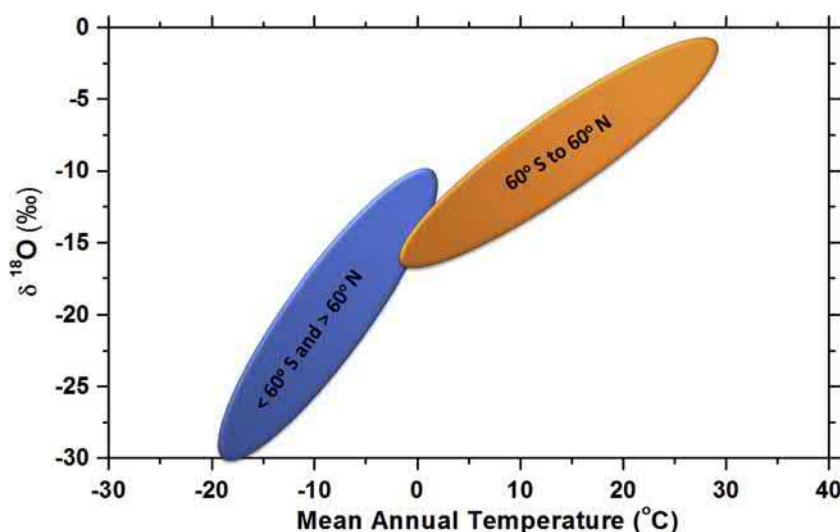


Fig. 5 Relation between weighted average $\delta^{18}\text{O}$ of precipitation and the average air temperature for 325 GNIP stations. Modified based on Gourey LL, Groening M, and Aggarwal PK (2005) Stable oxygen and hydrogen isotopes in precipitation. In: Aggarwal PK, Gat JR, and Froehlich KFO (eds.) *Isotopes in the Water Cycle: Past, Present and Future of a Developing Science*, pp. 39–53. Springer.

Furthermore, the latitudinal depletion of isotopic ratios (latitude effects) are also caused by temperature-dependent isotopic fractionation (Fig. 5). In semi-arid and arid regions, due to the low RH and low precipitation, the falling raindrops undergo evaporation, resulting in enrichment in heavy isotopes (evaporation effect). In addition, due to evaporation, surface waters (ponds, lake, rivers) are isotopically enriched.

Different environmental and radioactive isotopes for water resources development

Strontium-87 (^{87}Sr) isotope

Strontium is a minor constituent of groundwater. Strontium has four stable isotopes: ^{84}Sr , ^{86}Sr , ^{87}Sr and ^{88}Sr . Of these, only $^{87}\text{Sr}/^{86}\text{Sr}$ ratios are generally used for hydrological applications. The concentration of strontium in precipitation is generally low and can vary by two orders of magnitude from 2×10^{-3} to 3×10^{-1} mg/L. The much higher concentration in groundwater may be due to the dissolution of Sr-bearing minerals. No natural fractionation of stable strontium isotopes has been observed during the natural processes. This property makes the strontium isotopic ratio a reliable tool for evaluating the mixing of groundwater and for tracing water-rock interactions (Clark and Fritz, 1997). The $^{87}\text{Sr}/^{86}\text{Sr}$ ratios are measured using thermal ionization mass spectrometry (TIMS).

Boron-11 (^{11}B) isotope

Geochemical indicators like Br/Cl, B/Cl, etc., along with ^{11}B isotopes can be used to identify the source of salinity. The content of B changes as a result of water-rock interactions, mixing with waters of different origin and by the addition of contaminants. Boron has two stable isotopes, ^{10}B and ^{11}B . ^{11}B is measured using the positive ion thermal ionization mass spectrometry (TIMS). Values are reported in δ notation with respect to NBS951 boric acid standard. The large isotopic variation (90‰) of B, which is recognized in natural reservoirs, makes $\delta^{11}\text{B}$ a potential tool for tracing the origin of dissolved salts in groundwater.

Chlorine-37 ($\delta^{37}\text{Cl}$) isotope

This radioisotope is used to identify the source of Cl^- in groundwater, the mixing of fluids, and the rock-water interaction in an aquifer. $\delta^{37}\text{Cl}$ has been proven to be a useful tool for identifying sources of solute and for distinguishing solute transport mechanisms (Godon et al., 2004). The separation of ^{35}Cl and ^{37}Cl takes place by diffusion processes in argillaceous formations due to higher mobility of ^{35}Cl than that of ^{37}Cl , and this causes a depletion of ^{37}Cl in groundwater with increasing distance from a Cl^- source. Therefore, the $\delta^{37}\text{Cl}$ content of modern seawater is expected to differ from those of paleo seawater, which has been modified by diffusion processes in argillaceous sediments. Therefore, the difference in $\delta^{37}\text{Cl}$ could allow distinction of salinity from different origins and provide more information on seawater intrusions in a confined aquifer system. The fractionation of chlorine isotopes occurs due to the physical processes such as salt precipitation, brine evaporation, ion filtration, ion exchange and diffusion.

Bromine-81 ($\delta^{81}\text{Br}$) isotope

The use of bromine-81 to assess the origin and evolution of groundwater is still in its early stage, and only a few researches presently exist on this topic, although considerable work has been done on its natural variation in other parts of the world. Shouakar-stash et al. (2007) measured the $\delta^{81}\text{Br}$ isotope of groundwater samples from the Siberian platform, Russia and assessed the sources of various saline waters, including halite dissolution and seawater evaporation. This work showed its potential use in investigation of source and evolution processes of groundwater. Hence, this isotope can be used to understand the geochemical evolution of groundwater.

Nitrogen-15 ($\delta^{15}\text{N}$) isotope

The probable sources of nitrate/ammonia in groundwater are from mineral or organic fertilizers used for agriculture, from sewage/manure systems, and from waste dumps. The environmental isotopes tracers can be used to identify the origin of nitrate pollutants and solutes. Stable isotopes of nitrate can be used to distinguish the potential source of fertilizer (organic and inorganic) or mineralization of a nitrogen bearing substance within soil and precipitation. In groundwater, the origin of nitrate and ammonia can be identified using the nitrate isotope ($\delta^{15}\text{N}$). However, the advancement in analytical methods now allows the determination of $\delta^{18}\text{O}$ -nitrate. Both, $\delta^{15}\text{N}$ -nitrate and $\delta^{18}\text{O}$ -nitrate can be used together to discriminate the source of atmospheric nitrate and nitrate from fertilizer. Typical ranges of different sources of nitrate are shown in Fig. 6.

Sulfur-34 ($\delta^{34}\text{S}$) isotopes

Sulfate is one of the dissolved constituents of the groundwater and surface water. Identification of its sources and biogeochemical evolution is important for water resource managers. Sulfate may cause hardness of water. Hence, the identification of the source of

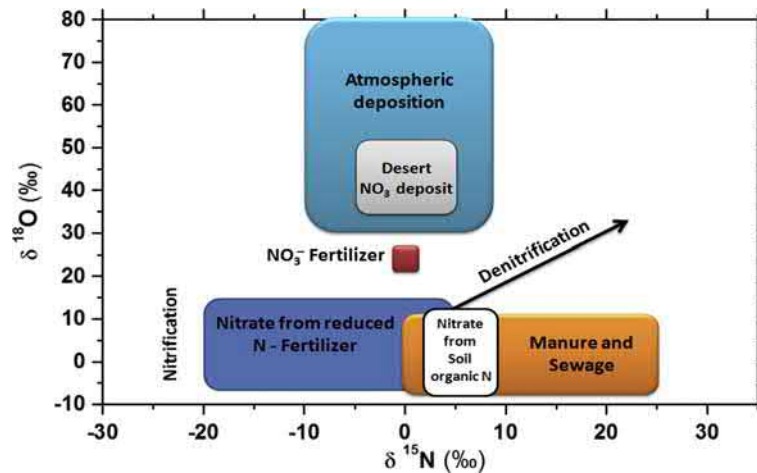


Fig. 6 Typical range of $\delta^{15}\text{N}$ and $\delta^{18}\text{O}$ of nitrate of different sources. Modified based on Kendall C and McDonnell JJ (1998) *Isotope Tracers in Catchment Hydrology*. Amsterdam: Elsevier, p. 839.; Mayer B (2005) Assessing sources and transformation of sulphate and nitrate in the hydrosphere using isotope techniques. In: Aggarwal PK, Gat JR, and Froehlich KFO (eds.) *Isotopes in the Water Cycle: Past, Present and Future of a Developing Science*, pp. 67–89. Springer.

sulfate is essential for water resource development. The stable isotopic composition of dissolved sulfate ($\delta^{34}\text{S}$) is often used to identify the source of sulfates. In addition, the spatial or temporal variability of sulfate concentration with respect to changing isotope ratios may divulge the biogeochemical processes controlling the occurrence of sulfate in the surface and sub-surface water bodies. Sulfur has four isotopes (^{32}S , ^{33}S , ^{34}S and ^{36}S), and due to their abundances, ^{34}S and ^{32}S are normally used for the measurement using the isotope ratio mass spectrometer (IRMS). The isotopic composition of sulfur is reported as $\delta^{34}\text{S}$, the ratio of $^{34}\text{S}/^{32}\text{S}$ in per mil (‰) relative to the V-CDT (Canon Diablo meteorite) standard. The $\delta^{34}\text{S}$ value ranges from +50‰ to –50‰ in the terrestrial environment. Geological formations (gypsum, evaporites, limestone, shales) that contain primary and secondary sulfide minerals, as well as chemically precipitated sulfate minerals, are the possible sources of sulfate in groundwater. The ranges of different sources of sulfate in a hydrological system are shown in Fig. 7.

Helium (He) isotopes

The stable isotopes of the noble gas helium (^3He , ^4He) offer a relatively new tool in isotope geochemistry. The solubility of atmospheric noble gases in water depends on the temperature (solubility increases with temperature) and their concentration in air. ^3He ,

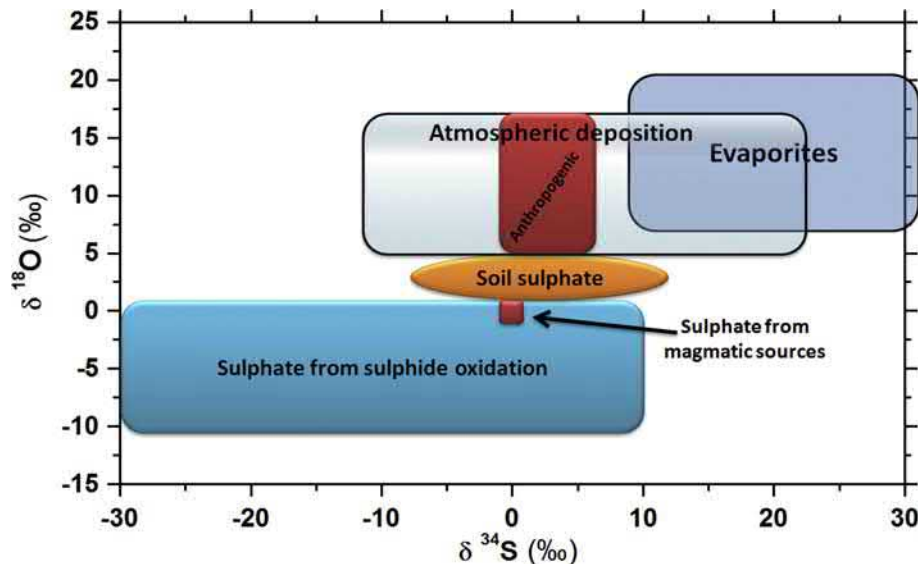


Fig. 7 Typical ranges of $\delta^{34}\text{S}$ and $\delta^{18}\text{O}$ values of sulfate. Modified based on Mayer B (2005) Assessing sources and transformation of sulphate and nitrate in the hydrosphere using isotope techniques. In: Aggarwal PK, Gat JR, and Froehlich KFO (eds.) *Isotopes in the Water Cycle: Past, Present and Future of a Developing Science*, pp. 67–89. Springer.

the daughter product of tritium decay, provides the age of young groundwater without knowing the tritium input function. ^4He in groundwater is used for dating old groundwater, as well as for understanding the possible mixing of waters of different ages ranging from 10^4 to 10^8 years (Clark and Fritz, 1997; Solomon and Cook, 2000).

Stable carbon-13/12 isotopes

Carbon, a ubiquitous element of the biosphere and hydrosphere, has two environmental stable isotopes (^{12}C , ^{13}C). Initially, carbon isotope ratios were used to reconstruct the palaeotemperature scale using marine carbonates. The chemistry of groundwater is strongly dependent on carbonate geochemistry (Clark and Fritz, 1997) and analyses of $\delta^{13}\text{C}$ of carbonate species is used to understand the carbonate geochemistry and geochemical evolution of groundwater. $\delta^{13}\text{C}$ is also essential for the quantification of the initial content of ^{14}C for dating purposes using ^{14}C . Carbon isotopes in geothermal systems have been used for gaining insights into reservoir conditions and for the geothermometry of such waters. They are also valuable in tracing the sources of contaminants to groundwater (Clark and Fritz, 1997; Bird et al., 2020). The ranges of $\delta^{13}\text{C}$ composition of the major sources of carbon are shown in Fig. 8.

Tritium (^3H)

Tritium is a radioactive isotope of hydrogen with a half-life of 4500 ± 8 days (Lucas and Unterweger, 2000), which decays into ^3He by beta (β) emission. Environmental tritium is a natural isotope that is formed in the upper atmosphere (stratosphere) by the interaction of cosmic rays with nitrogen (N) atoms, at an average rate of $2500 \text{ atoms m}^{-2} \text{ s}^{-1}$ (Masarik and Beer, 2009). In addition, it is also generated by various nuclear based anthropogenic activities like nuclear weapons explosions, bombardment of ^6Li in the reactors, etc. The main source of tritium in the atmosphere comes from atmospheric thermonuclear tests between the years 1953 and 1963. In 1963, tritium in precipitation reached the maximum level of several thousands of Tritium units (TU) (approximately 10,000 TU), which was 2–8 TU before nuclear tests (Michel, 2005). One TU is equivalent to one tritium atom per 10^{18} hydrogen atoms, corresponding to 0.120 Bq/L at standard temperature and pressure. Most of the environmental tritium (around 55%) is produced in the stratosphere and it has been estimated that $5\text{--}9 \times 10^5$ TU of atmospheric tritium resides in the vapor of the stratosphere (Fouere et al., 2006). Tritium is highly mobile and is easily exchanges with water molecules or hydrogen. Therefore, it is available everywhere in the hydrological systems. It is a useful environmental tracer for assessing the hydrological system, particularly in the dating of groundwater (Morgenstern and Daughney, 2012). It is an element of the water molecule (HTO) and has been used for

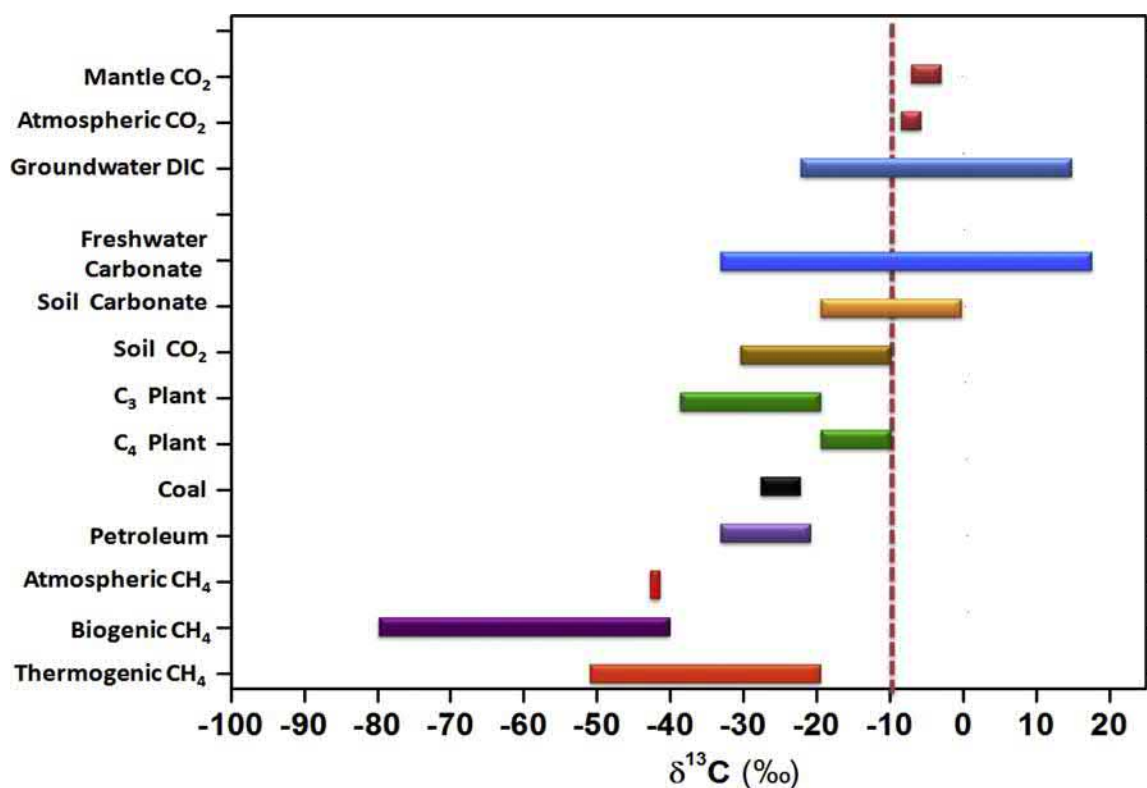


Fig. 8 The range of $\delta^{13}\text{C}$ composition of the major sources of carbon. Modified based on Bird MI, Haig J, Hadeen X, et al. (2020) Stable isotope proxy records in tropical terrestrial environments. *Palaeogeography, Palaeoclimatology, Palaeoecology* 538: 109445.

the past five decades as a tracer for the understanding of hydrological systems. The anthropogenic tritium that was released into the atmosphere during the nuclear tests is used to distinguish between modern groundwater recharge and pre-nuclear test recharge (Yangui et al., 2012). Current advanced models, particularly lumped parameter modeling, are used to quantify the mean transit time of groundwater, and are used to calibrate and validate other groundwater models such as the solute transport model, the 3D-GW flow model, etc. (Zuber et al., 2011). Tritium is also used to calculate recharge rates of groundwater and its renewability (Kumar et al., 2008), residence time as well as velocity and directions of groundwater flow (Sawdoni et al., 2000; Ansari et al., 2014, 2017).

In addition, tritium is extensively used in the study of the surface water system, such as lakes, rivers and streams (Michel, 2004). Time series tritium data of a river can be used to identify the river characteristic, whether it is the influent river or the effluent river.

The normal vertical distribution of tritium and deuterium in the atmosphere is shown in the Fig. 9. It is observed that $\delta^2\text{H}$ content gradually decreases up to the tropopause and then increases in the stratosphere up to 40 km (Johnson et al., 2001), while tritium increases gradually with increasing altitude in the atmosphere up to the tropopause (Zahn et al., 1998).

Age dating using tritium

Tritium decays to the noble gas ^3He . Since the absolute activity of ^3H has decreased over the past several decades and has almost reached the level of natural input, the combined use of $^3\text{H}/^3\text{He}$ helps to define groundwater ages based on the ratio of both isotopes. The age dating method using $^3\text{H}/^3\text{He}$ is based on the equation given below:

$$T = \frac{1}{\lambda} \ln \left(\frac{^3\text{He}_{\text{trit}}}{^3\text{H} + 1} \right) \quad (7)$$

where T is age of groundwater, λ is the tritium decay constant, and ^3He is the tritiogenic helium derived from the decay of tritium only.

The dating of groundwater using the $^3\text{He}/^3\text{H}$ ratio is less sensitive for older waters and requires precise determinations of excess air applicable for the close systems at the groundwater surface (Solomon and Cook, 2000). This method is not applicable for the groundwater nearby major fault zones, geothermal fields or geologically young igneous rocks since the concentration of geogenic sources may exceed the ^3He derived from tritium decay by an order of magnitude.

Radioactive carbon (^{14}C) isotope

Carbon-14 is a radioisotope of carbon having a long half-life of 5730 years. It is naturally produced in the upper atmosphere by the collision of cosmic ray produced neutrons with common nitrogen. These ^{14}C atoms are oxidized to carbon dioxide and enter the troposphere and participate in the biological and hydrological carbon cycle. By the biological action such as root respiration and decomposition of plant material, carbon dioxide enters into the soil. Rain dissolves small amounts of atmospheric carbon dioxide. Significantly, more carbon dioxide is added to water percolating through the soil layer, since soil air contains 100 times more carbon

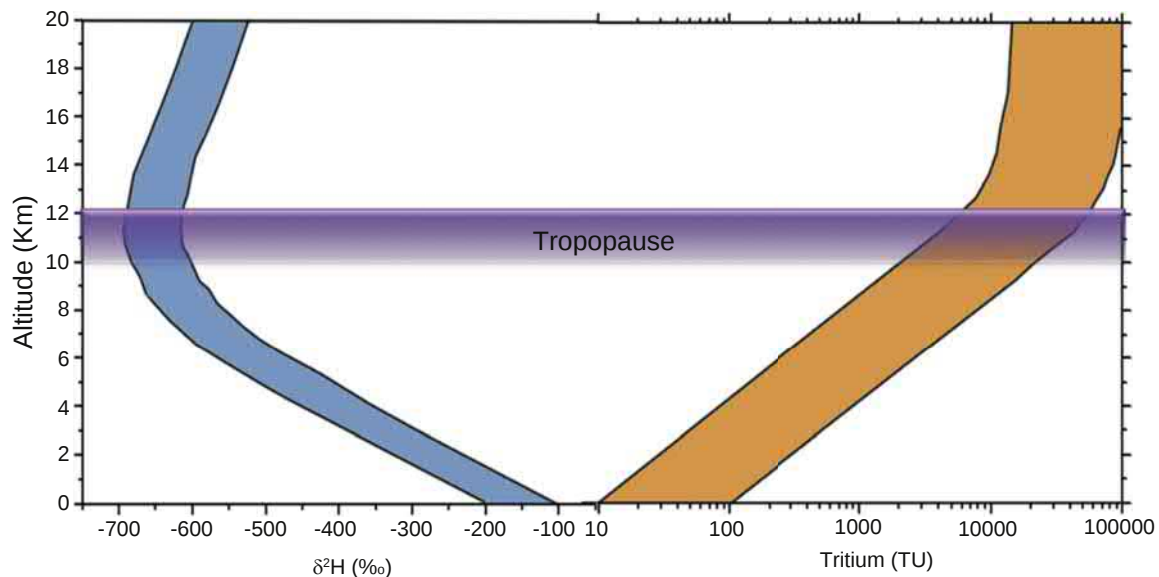


Fig. 9 Generalized vertical distribution of $\delta^2\text{H}$ and tritium (^3H) in the atmospheric moisture. Modified based on Rozanski K (2005). Isotopes in atmospheric moisture. In: Aggarwal PK, Gat JR, and Froehlich KFO (eds.). *Isotopes in the Water Cycle: Past, Present and Future of a Developing Science*, pp. 291–302. Springer.

dioxide than free air. Once water reaches the saturated zone, it is sealed from the atmosphere and its ^{14}C content decays, thus forming the basis for groundwater dating (Clark and Fritz, 1997). It decays according to:

$$a_t = ca_0 e^{-\lambda T} \quad (8)$$

$$T = -\frac{1}{\lambda} \ln \left(\frac{a_t}{ca_0} \right) \quad (9)$$

where, $\lambda = 1.2097 \times 10^{-4}$ is the decay constant, c is the dilution factor, a_t is the ^{14}C activity at time t , and a_0 is the initial ^{14}C activity (Geyh and Schleicher, 1990). The ^{14}C activities are expressed as pmC (percent modern carbon). The modern atmospheric carbon-14 (pre-nuclear test) content is 100 pmC corresponding to 13.56 dpm/gC in the year 1950 AD (Stuiver and Polach, 1977). The ^{14}C in the water samples is measured using a liquid scintillation counter or using an accelerator mass spectrometer. When the groundwater gets isolated from the atmosphere, the decay of ^{14}C in dissolved organic and inorganic carbon is used for dating the groundwater. The time at which groundwater became isolated from the atmosphere can be estimated using the ratio of initial and actual ^{14}C activities, a_0 and a_t as per the above equation. There are uncertainties associated with the estimation of initial activities, a_0 , because of potential contamination from post-nuclear detonations (post-1950), paleoclimatic recharge conditions, dilution, change in recharge rate and change from open to closed systems. Therefore, the estimation of a_0 should be done using geochemical adjustment models (Fig.10) such as the Vogel model (Vogel, 1967 and 1970), Tamers model (Tamers, 1967), Pearson model (Ingerson and Pearson, 1964), Eichinger model (Eichinger, 1983), IAEA model (Salem et al., 1980), Evans model (Evans et al., 1979), the Mook model (Mook, 1972, 1976, 1980) and the Fontes and Garnier model (Fontes and Garnier, 1979).

Chlorine-36 (^{36}Cl) isotope

Chlorine-36 is produced in the atmosphere predominantly by cosmic ray spallation of ^{40}Ar and to a lesser extent by the neutron activation of ^{36}Ar and ^{35}Cl . This chlorine-36 is then mixed with the atmospheric chloride (i.e. Cl derived from the ocean) and is deposited on the land surface during rains or snowfalls or as dry aerosols. When groundwater recharge takes place, it carries the ^{36}Cl into the subsurface where the radiometric “clock” is set. It is conservative within the subsurface. Therefore, it rarely retards by adsorption or any geochemical reactions in the subsurface. ^{36}Cl , which has a half-life of 301,000 years, is used for dating water in the age range 100,000 years to 1 million years. However, there are large uncertainties associated with the quantification of age. That is because of the variations in the Cl concentration in groundwater. The presence of evaporites in the catchment and evapotranspiration enrichment in the unsaturated zone, as well as variation in ^{36}Cl , fallout may cause complications.

This technique is also used to delineate the recharge rate of groundwater. The ^{36}Cl in water can be measured using an accelerator mass spectrometer (AMS) (Davis et al., 1983). It was first used to date old groundwater in the Great Artesian Basin, Australia and Milk River aquifer, Canada (Bentley et al., 1986; Phillips et al., 1986).

Process	Conventional $^{14}\text{C}_{\text{age}}$	Vogel	Tamers	Mass balance	Pearson	IAEA	Mook	Evans	Eichinger	Fontes and Garnier
Carbonate dissolution										
Soil gas CO_2 dissolution										
CO_2 gas-aqueous exchange										
Calcite- HCO_3 exchange										
Gypsum dissolution										
Ca/Na cation exchange										

Fig. 10 Geochemical process (reactions) and a_0 adjustment models.

Krypton-81 (^{81}Kr) isotope

This Krypton-81 isotope, with a long half-life of 229,000 years, is an ideal tracer because of (1) its inertness (non-reactive), (2) being derived only from an atmospheric source, and (3) the absence of any sub-surface source or sink except radioactive decay (Jiang et al., 2012; Matsumoto et al., 2018). The advancement of analytical techniques with high counting efficiency and shorter measurement time has made this tracer useful for dating old groundwater (Aggarwal et al., 2015). For ^{81}Kr analysis, a large volume of water samples of 2000–3000 L is pumped through a vacuum extraction chamber and the gas extracted is transferred to a container. Using preparative gas chromatography, Kr is then extracted from the bulk gas (Lu et al., 2014). The $^{81}\text{Kr}/\text{Kr}$ and $^{85}\text{Kr}/\text{Kr}$ ratios are determined using the Atom Trap Tracer Analysis (ATTA-3) instruments. As per the exponential decay law, the time elapsed since the water was in contact with the atmosphere can be estimated according to:

$$T = -\frac{1}{\lambda} \ln \left(\frac{R_{\text{sample}}}{R_{\text{air}}} \right) \quad (10)$$

where, $\lambda = 3.03 \pm 0.14 \times 10^{-6} \text{ year}^{-1}$ is the ^{81}Kr decay constant and R_{sample} and R_{air} are the $^{81}\text{Kr}/\text{Kr}$ ratios of the samples and modern air ($5.2 \pm 0.6 \times 10^{-13}$) respectively (Du et al., 2003).

Krypton-85 (^{85}Kr) isotope

^{85}Kr is another radioisotope of Kr that may also be used for dating of groundwater, due to its similar nature as that of ^{81}Kr . It has a short half-life ($t_{1/2} = 10.76$ years) and is used for dating groundwater less than 50 years old. It is the fission product of plutonium and uranium. The current specific activity of ^{85}Kr in the atmosphere is about $1.4 \text{ Bq/cm}^3_{\text{STP}}$ corresponding to a ratio of $^{85}\text{Kr}/\text{Kr}_{\text{total}}$ of 2.4×10^{-11} (Winger et al., 2005). The sampling and analytical procedure for the measurement of the $^{85}\text{Kr}/\text{Kr}$ ratio is same as that of $^{81}\text{Kr}/\text{Kr}$. ^{85}Kr is also used to monitor air contamination caused during sampling and gas extraction for the ^{81}Kr and ^{39}Ar dating process (described below) and to determine possible inter-aquifer leakage in the well.

Argon-39 (^{39}Ar) isotope

^{39}Ar is produced in the atmosphere by the interaction of cosmic rays with K and Ar nuclides. The equilibrium activity of Ar in the atmosphere is constant at $1.67 \times 10^{-2} \text{ Bq/m}^3$ of air, which corresponds to a $^{39}\text{Ar}/\text{Ar}$ ratio of $\sim 10^{-15}$ in the atmosphere. Argon-39 has half-life ($t_{1/2}$) of 269 years and is used for dating groundwater that falls between the modern water indicators ($^3\text{H}/^3\text{He}$, CFCs, etc.) and the ^{14}C radiocarbon methods. For ^{39}Ar analysis, a large volume of water samples of 2000–3000 L is pumped through a vacuum extraction chamber and the gas extracted is transferred to a container. Using preparative gas chromatography, Ar is extracted from the bulk gas (Lu et al., 2014). The ^{39}Ar activities are determined using low level gas proportional counters and the results are reported in %modern Ar (Loosli et al., 2000). ^{39}Ar can also be used to evaluate the presence of comparatively modern component (60–1000 years) water in the very old groundwater that cannot be identified using other isotopes, such as ^{85}Kr , ^3H , CFC analysis (Fig. 11).

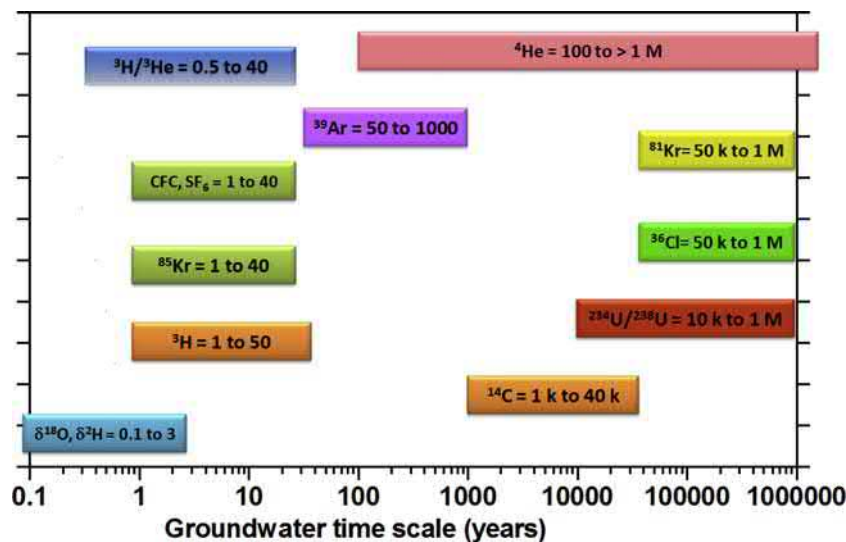


Fig. 11 Range of age dating using different radioisotopes and other environmental tracers.

Uranium-234/238 isotopes

The $^{234}\text{U}/^{238}\text{U}$ ratio in the groundwater samples can be used to constrain the groundwater dynamics and weathering processes (Clark and Fritz, 1997). A very low $^{234}\text{U}/^{238}\text{U}$ ratio indicates the lack of radioactive equilibrium in waters due to intense weathering as well as slow moving groundwater. A lower ratio is indicative of younger groundwater. In the higher transmissivity wells, there is a weak pattern of increasing activity ratio (AR) with increasing transmissivity.

Radon (^{222}Rn) and radium (^{226}Ra) isotopes

Radium has four natural isotopes, ^{223}Ra , ^{224}Ra , ^{226}Ra , ^{228}Ra and their half-lives are 11.4 days, 3.6 days, 1600 years, and 5.8 years, respectively. ^{226}Ra and ^{223}Ra are the decay products of ^{238}U and ^{235}U , while ^{224}Ra and ^{228}Ra are decay products of ^{232}Th . Radium is released into the groundwater by alpha recoil during the decay of the parent nuclide within the aquifer matrix, desorption from the surface of the rocks, and from dissolution. In surface water, radium isotopes are normally lesser than that of groundwater because of decay and adsorption (Wang et al., 2016). Therefore, this technique is used for evaluating the sub-surface groundwater discharge within the Sea (Liu et al., 2017). The distribution of radium in groundwater is mainly controlled by the chemistry of groundwater and the distribution of their parents' isotopes (U, Th) with the aquifer matrix (solid) (Sherif et al., 2018).

As for radon (^{222}Rn), it is a dissolved gas that is produced by the decay of the parent nuclide (^{226}Ra) present in the water and from alpha recoil during the decay of ^{226}Ra within the solid matrix. Radon sinks within the groundwater either by its radioactive decay or by the loss to the unsaturated zone due to degassing. ^{222}Rn is not absorbed by solids. Therefore, it is highly soluble in water, exactly the opposite of ^{226}Ra . In groundwater, radon variability is large because of the large scale variation in the fracture characteristics of the rock and groundwater flow dynamics (Noble et al., 2015). Owing to its short half-life (3.8 days), radon is indicative for young groundwater. Radon is also used to assess the surface water-groundwater interactions, to delineate the geological structure (faults, lineaments, etc.) (Ansari et al., 2016b) and to evaluate earthquakes.

Lead-210 (^{210}Pb) isotope

Lead (^{210}Pb) has little importance for hydrological applications, although, it is widely used for estimating the accumulation rate of sediments or ice. As ^{222}Rn gas escapes from the earth surface and decays by way of short-living daughters to ^{210}Pb in the atmosphere, after a short time in the air it settles down either by dry or wet precipitation along with snow (to form ice) or sediment in water. Since its half-life is 22.26 years, it can be used to estimate the age of ice and aqueous sediments deposit up to about 100 years.

Modeling of environmental water isotopes

Currently available stable isotope models have proved to be very valuable, both in local and global scales, in understanding and utilizing stable isotopes processes to widely varying degrees. Since the first use of Rayleigh distillation models, research has progressed towards GCMs which have enabled stable isotope processes to be coupled to the model's hydrological cycle.

There are two categories of stable isotope models; namely, Lagrangian and Eulerian, which are fundamentally distinct. The Lagrangian approach evaluates the fractionation of moisture in an air parcel during its transport in space and time, whereas the Eulerian model approaches evaluate isotopic fractionation based on first principles for the whole temporally and spatially discretized model domain. Both approaches are supported by parameterizations of sub-grid scale processes. Although increasingly complex models have been developed in both categories, several intermediate approaches have recently been introduced that combine the advantages of the two families of models, Lagrangian and Eulerian. Some of the important models available are the Lagrangian Rayleigh model, the 2D column model, the 2D advection model, the 3D GCMs and RCMs models, etc. All these available models have provided valuable insight on the climatological interpretation of the stable isotopic signal in many regions of the world.

Conclusions

A better understanding of hydrological systems is important for developing policies and practices for the sustainable management of freshwater resources. Isotope hydrology techniques have provided unmatched insights into the governing processes of the hydrological systems. Environmental isotopic tracers are now the most promising modern tools for providing critical hydrological information that sometimes cannot be obtained with other conventional techniques. The availability of advanced, inexpensive and easy-to-use instruments for measuring different isotopes has made progressive use of these tools in isotope hydrology. The applications of environmental isotopic tracers should be enormously multiplied and promoted worldwide in the coming years for ensuring sustainable management of freshwater resources.

See Also: Nuclear Driven Water Desalination.

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Environmental Protection: The Oceans—Formation and Global Climate Change

GiHoon Hong^a and Pavel P. Povinec^b, ^a GH Institute, Seoul, South Korea; and ^b Faculty of Mathematics, Physics and Informatics, Comenius University, Bratislava, Slovakia

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Glossary

Anthropogenic Relating to or resulting from the influence of human beings on nature.

Apohic zone The deep zone of an ocean or lake receiving too little light to permit photosynthesis.

Asthenosphere The ductile part of the earth just below the lithosphere, including the upper mantle.

Benthic nepheloid layer Relatively more turbid water layer near the bottom of the ocean.

Biomagnification Any concentration of chemicals in the tissues of tolerant organisms at successively higher levels in a food chain.

ϵ_{Nd} Epsilon notation where one epsilon unit represents a one part per 10,000 deviation from the Chondritic Uniform Reservoir (CHUR) isotopic composition of Nd.

Grand Ocean Conveyor Belt Thermohaline circulation that derives a global-scale system of currents.

HPGe detectors High Purity Germanium (HPGe) Radiation Detectors.

Lithosphere The outer solid part of the earth, including the crust and upper mantle.

Mass extinction Any substantial increase in the amount of extinction (lineage termination) suffered by more than one geographically wide-spread higher taxon during a relatively short interval of geologic time, resulting in at least a temporary decline in their standing diversity.

Nucleosynthesis The process of creating new atomic nuclei (chemical elements) from preexisting nucleons.

Primeval Of or resembling the earliest ages in the history of the world

Primordial Existing at or from the beginning of time.

Sv One Sverdrup (Sv) is equivalent to a volume transport of one million cubic meters per second. The volume flux unit *Sverdrup* (Sv) is widely used, after the Scandinavian pioneer meteorologist/oceanographer Harald Ulrik Sverdrup (1888–1957).

Tectonic plate A massive irregularly shaped slab of solid rock, generally composed of both continental and oceanic lithosphere. The plate size can vary greatly from a few hundred to thousands of kilometers across the globe.

Abbreviations

AMOC Atlantic meridional overturning circulation

AMS Accelerator mass spectrometry

CF Concentration factor

ITCZ Intertropical convergence zone

SPM Suspended particulate matter

THC Thermohaline circulation

TSPM Total suspended particulate matter concentration

Introduction

The ocean is vast, covering $3.6 \times 10^8 \text{ km}^2$, equivalent to approximately 72% of the earth's surface. It contains about $1.37 \times 10^{18} \text{ m}^3$ of water, and its mean depth is $3.8 \times 10^3 \text{ m}$ (Broecker and Peng, 1982). The world ocean provides the air we breathe by producing over half of the world's oxygen and absorbs 50 times more carbon dioxide than our atmosphere, regulating climate by transporting heat from the equator to the poles. The importance of seafood and other marine natural products in a balanced diet has been widely recognized by the full world human population.

"Climate refers to the average weather in terms of the mean and its variability over a certain period and a certain area. The classical period for averaging these variables is 30 years, as defined by the World Meteorological Organization." (IPCC, 2018). Thus, climate change refers to a change in the state of the climate due to natural or man-made forces, while the climate treaty of the United Nations Framework Convention on Climate Change 1992 (UNFCCC) defines climate change as "a change of climate which is attributed directly or indirectly to human activity that alters the composition of the global atmosphere and which is in addition to natural climate variability observed over comparable time periods" (Article 1.1). In this article, the former definition is used. The role of the oceans in climate may be best understood as a thermal lag on the climate variation due to its vast heat capacity. Oceans exchange with the atmosphere large quantities of heat, water, gases such as carbon dioxide and dimethyl sulphide, particles, and momentum. It is an important part of the global redistribution of heat from tropics to polar regions that keeps our planet habitable (e.g., Bigg et al., 2003). The grand thermohaline circulation of the global ocean, so called Grand Ocean Conveyor Belt, is the major mechanism by which the ocean contributes its share to the poleward heat transport required to counteract the Earth's surface radiation imbalance. This cyclic action repeats itself over a time period of about 1000 years. In the longer term, over a time scale of greater than 10,000 years, plate tectonics have shaped the distribution of land masses and ocean basins throughout the history of the Earth. Hence, the geologic time scales referred to eras of Precambrian, Paleozoic, Mesozoic, and Cenozoic and their individual periods, epochs and ages, are terms reflecting major climate and biological species compositional changes that have occurred in our planet.

About 4.6×10^9 years ago our solar system formed from a cloud of gas and dust that slowly contracted under the mutual gravity of all particles. The cloud was made largely of hydrogen (H) with some helium (He) and small amounts of the remaining naturally occurring chemical elements. The elements larger than He had to be produced in supernovae. Primordial radionuclides are residues from the Big Bang, from cosmogenic sources, and from ancient supernova explosions that occurred before the formation of the solar system. Bismuth (^{209}Bi , $t_{1/2} = 4.6 \times 10^{19} \text{ a}$), thorium (^{232}Th , $t_{1/2} = 14.05 \times 10^9 \text{ a}$), uranium (^{238}U , $t_{1/2} = 4.468 \times 10^9$; ^{235}U , $t_{1/2} = 704 \times 10^9$) and potassium (^{40}K , $t_{1/2} = 1.25 \times 10^9 \text{ a}$) are typical primordial radionuclides because they have half-lives long enough to still be found on the Earth. These radionuclides continue to undergo radioactive decay and produce heat in the core of the earth, turning iron into a convecting dynamo that maintains a magnetic field strong enough to shield the Earth from the solar wind and cosmic-ray particles. This molten iron leaks out into the mantle, causing convection in the rock that moves crustal plates and fuels volcanoes (Lee and Jeanloz, 2003; Sanders, 2003).

The origin of the primordial natural radionuclides of the earth is associated with the phenomenon of nucleosynthesis in stars (Garcia, 2021). The fact that the uranium, thorium, and actinium decay chains are found in the Earth is owing to the long half-lives of the parents of these chains. These primordial radionuclides have witnessed the evolution of the Earth system from its birth. Therefore, deciphering the nature of the presence of these radionuclides in various phases of the Earth is crucial to understand the past changes that occurred in the Earth over the billions of years ago. Also, the Earth since its birth is constantly bombarded by galactic cosmic rays, which primarily consist of protons and helium nuclei (Vermeesch, 2020). The primary cosmic rays managed to pass through the Earth's magnetic field, strongly interacting with the atmospheric molecules and producing secondary cosmic ray showers, mostly made of nucleons, muons and gamma-rays. Only a small fraction of the secondary cosmic rays reaches the surface of the Earth. The produced neutrons and protons then collide with the elements in the atmosphere and produce many cosmogenic radioactive and stable nuclides.

Determining the dates corresponding to climate and biota changes, as well as deciphering their underlying causes, are crucially important to investigate the underlying earth processes and also to predict the future climate changes. The natural radionuclides, along with anthropogenic radionuclides with known input functions, provide powerful means to investigate various earth processes—including the ocean. They allow the understanding of large-scale ocean circulation, input and formation of particulate matter in the ocean and its subsequent settling and decomposition in the water column, and accumulation and burial in the seafloor. These oceanic processes, with transfer of energy and material, play a crucial role in shaping the current climate of the Earth. The scientific understanding of the oceans' dynamics and their evolution over time is required to manage the wealth and sustainability of the ocean. In this regard, nuclear technology has been instrumental in tracing the movement of energy and material through the various oceans and through other compartments of the Earth, such as the atmosphere, cryosphere, and lithosphere both globally and locally. The rates and fluxes of material and energy occurring in the oceans examined by radionuclides are yielding many practical applications for sustainable management of the oceans.

Evolution of land and ocean through geologic history

The initial rotation or tumbling motion of our planetary system was accelerated as the nebula contracted, like a spinning skater who pulls in his arms to spin faster. Within the disc, the largest concentration of matter was in the center. This became the sun. Matter collected in smaller clumps out in the disc. These became the planets. The proto-sun and proto-planets grew by accretion. The materials that accreted into the early Earth were probably added piecemeal, without any particular order, by numerous collisions between small objects to form larger ones (Yu and Jacobsen, 2011). The early Earth was extremely hot from (1) gravitational compression, (2) impacts, and (3) radioactive decay. The early Earth was probably partially or largely molten. The denser metallic liquids sank to the center of the Earth and less dense silicate liquids rose to the top. In this way the Earth was very quickly differentiated into a metallic, mostly iron core and a rocky silicate mantle. Through igneous (volcanic and intrusive) activity, the crust of the Earth eventually formed.

At the same time, the atmosphere and the oceans accumulated gradually over billions of years with continual degassing of the Earth's interior. The ocean formed with the escape of water and other gases from the molten rocks of the Earth to the atmosphere surrounding the cooling planet. After the Earth's surface had cooled to a temperature below the boiling point of water, rain began to fall for centuries. As the water drained into the great hollows in the Earth's surface, the primeval ocean came into existence. This occurred as early as 4.4 billion years ago when the continent started to grow (Hadean Eon, Precambrian Era). This time period was determined by ^{40}Ar dating technology (Cawood et al., 2013; Guo and Korenaga, 2020). Argon is an inert gas that land masses emit into the atmosphere. It is a product of the radioactive decay of the primordial radionuclide, ^{40}K , that remained in the crust and mantle of continents. The ^{40}K , which decays to ^{40}Ar , was transferred to the continental crust from the mantle during partial melting, and part of it was recycled back into the mantle through subduction. About 5–6% of the Earth's mantle is made of recycled crust. The rates of crust formation were regarded similar to what they are today (Yang et al., 2020).

The forces of gravity prevented the water from leaving the planet. The Earth's solid outer crust, the lithosphere, is separated into plates that move over the asthenosphere, the molten upper portion of the mantle. Oceanic and continental plates come together, spread apart, and interact at boundaries all over the planet. Each type of plate boundary generates distinct geologic processes and landforms. At divergent boundaries, plates separate, forming a narrow rift valley. Here, geysers spurt super-heated water and magma, or molten rock, which rise from the mantle and solidify into basalt, forming a new crust. Thus, at divergent boundaries, oceanic crust is created. The mid-ocean ridge, the Earth's longest mountain range, is formed at the divergent boundary. It usually rises above about 2000 m from the sea floor but also rises above sea level (e.g., Iceland). Currently it is 5000 km long and 1500 km wide. The total length of the mid-ocean ridge has varied along the breakup of supercontinents, thus, impacting the atmospheric CO_2 concentration (Müller and Dutkiewicz, 2018).

As the seawater continues to percolate downward through fractured ocean crust along the volcanic ocean ridge system, it is first heated and then undergoes chemical modification through reaction with the host rock as it continues downward, reaching $>400^\circ\text{C}$. At these temperatures the fluids become buoyant and rise rapidly back to the seafloor where they are expelled into the overlying water column. Therefore, seafloor hydrothermal circulation plays a significant role in the cycling of energy and mass between the solid earth and the ocean to result in changes in seawater composition (e.g., von Damm, 2000) and it has changed seawater composition throughout the geological times (Antonelli et al., 2017).

At convergent boundaries, plates collide with one another. The collision buckles the edge of one or both plates, creating a mountain range or subducting one of the plates under the other, creating a deep seafloor trench. At convergent boundaries, continental crust is created, and oceanic crust is destroyed as it subducts, melts, and becomes magma. The highest mountain range above the sea level, the Himalayas, was formed 55 million years ago (Eocene Epoch of Tertiary Period, Cenozoic Era) when the Eurasian and Indo-Australian continental plates converged. Therefore, the ocean basins are to be recognized as transient features over geologic time, changing shape and depth due to the dynamic processes of plate tectonics that have occurred relentlessly.

Ocean basins

Tectonic plates have been known to form a supercontinent, then break away to drift for a while, and then aggregate again to form a supercontinent over the time scale of approximately 600 My. These recurring gatherings and dispersals of the continents have occurred throughout Earth's history (Nance et al., 2014). This assembly and breakup of continents have profoundly influenced the evolution of the geosphere, hydrosphere, atmosphere, and biosphere—including the climate changes of the Earth. Notable supercontinents having appeared so far include Ur (~ 3 Ga; Archean), Kenorland (~ 2.4 Ga; Neoproterozoic), Columbia/Nuna (~ 1.7 Ga; Paleoproterozoic), Rodinia (~ 1 Ga; Neoproterozoic), Pannotia (~ 600 Ma; Neoproterozoic ago), and Pangea (~ 250 Ma ago; Triassic). The abbreviation of Ga and Ma refer to giga years ago and million years ago, respectively.

The Pangea supercontinent has been breaking up to form the current continent and ocean. It is moving to form the next supercontinent (Amasia or Pangea Ultima, or Novopangea, or Aurica) over the next 200–300 My. The Arctic Ocean and the Caribbean Sea will disappear, and Asia will crash into the Americas by that time (Davies et al., 2018). It is clear that the distribution of continents and oceans was different from that of present day in the past and it will continue to be so in the future. For example, the huge landmasses Gondwana and Laurussia and other small ones were in the southern hemisphere in the early Devonian Period (ca. 410 My before present). In the next 200 My, most of the continental masses were gradually brought together to form a supercontinent called Pangea (Fig. 1; Domeier and Torsvik, 2014).

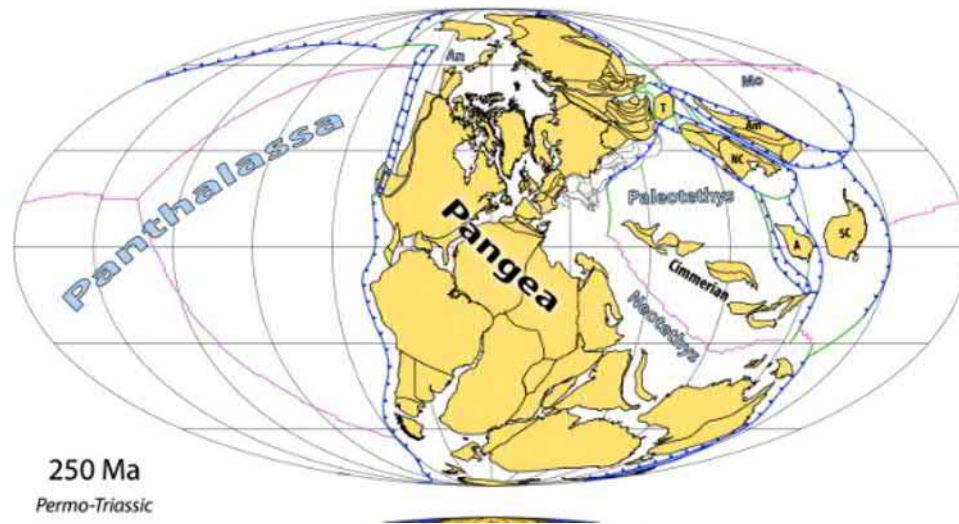


Fig. 1 250 Ma (Permo-Triassic) paleogeographic reconstruction showing simplified plate boundaries and labels of some major features. Abbreviated continental units: A, Annamia; Am, Amuria; NC, North China; SC, South China; T, Tarim; oceanic domains: An, Angayucham Ocean; Mo, Mongol-Okhotsk Ocean. From Domeier M and Torsvik T (2014) Plate tectonics in the late Paleozoic. *Geoscience Frontiers* 5: 303–350.

The fragmentation of Pangea led to the formation of an internal ocean (the present-day Atlantic Ocean) and partial consumption of the surrounding ocean (the present-day Pacific Ocean) around 180 million years ago. Currently, the spreading centers are mostly located in the oceanic ridges and their half spreading rates are fast as much as 100 mm per year (Fig. 2; Müller et al., 2008). This means that the age of the oceanic crust is young in the ridge crest, whereas the spreading ridges could be as much as 300 My old at the brink point of sinking back into the mantle in the subduction zone (e.g., Philippine Sea) (Fig. 3; Müller et al., 2008). Also, the oceanic depth is found to increase from the spreading ridge (~2500 m) to over 6000 m near the subduction zone (Hiller and Watts, 2005). It is generally thought that we are presently halfway through a supercontinent cycle.

Ocean waters

The tropical ocean, which is situated approximately between 30°N and 30°S, is generally defined as the sea surface with temperature above 20 °C. Its surface area constitutes more than a half of the global ocean. However, the thermal equator migrates to the north and south seasonally. The tropical ocean is, in fact, extended almost to the area between 40°N and 40°S, and comprises more than

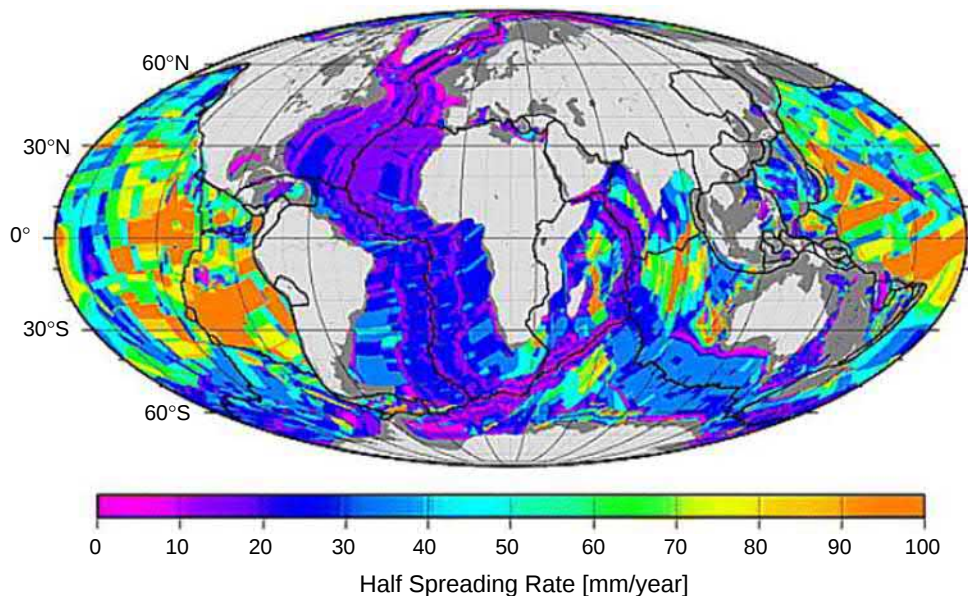


Fig. 2 Half-spreading rate of oceanic plates in the world ocean. From Müller RD, Sdrolias M, Gaina C and Roest WR (2008) Age, spreading rates, and spreading asymmetry of the world's ocean crust. *Geochemistry, Geophysics, Geosystems* 9. doi: 10.1029/2007GC001743.

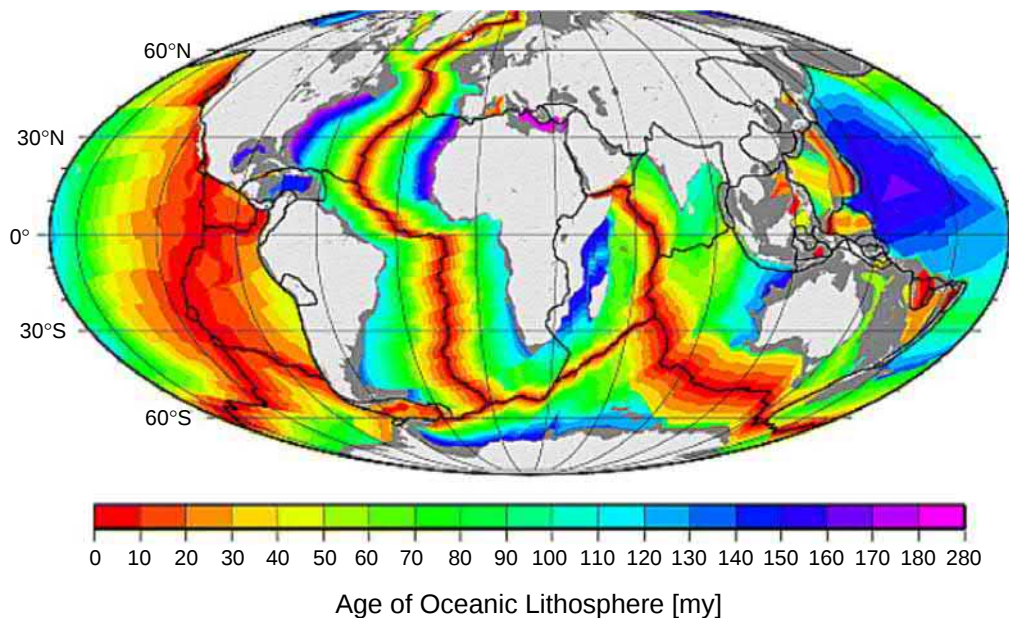


Fig. 3 Age of oceanic crust. From Müller RD, Sdrolias M, Gaina C and Roest WR (2008) Age, spreading rates, and spreading asymmetry of the world's ocean crust. *Geochemistry, Geophysics, Geosystems* 9. doi: 10.1029/2007GC001743.

70% of the global ocean (Peng, 1985; IAEA, 1988), as shown in Fig. 4. Warm tropical waters accumulate heat and energy harnessed from solar radiation and move westward by trade winds. They peel off in northerly and southerly directions (e.g., western boundary currents such as the Gulf Stream and the Kuroshio in the Northern Hemisphere, East Australia Current, Agulhas Current, Brazil Current in the southern hemisphere) in places where the landmass blocks its westward movement—resulting in water being pumped in bursts that progressively move toward the poles.

The intertropical convergence zone (ITCZ) is the region that circles the Earth near the equator, where the trade winds of the Northern and Southern Hemispheres come together. The intense solar radiation and warm water of the equator heat the air in the ITCZ, raising its humidity and making it buoyant. Aided by the convergence of the trade winds, the buoyant air rises. As the air rises, it expands and cools—releasing the accumulated moisture in an almost perpetual series of thunderstorms. Therefore, it is also called the global rain belt. It migrates latitudinally on a seasonal basis. Although it remains near the equator, the ITCZ moves farther north or south over land than over the oceans because it is drawn toward areas of the warmest surface temperatures. The location of the ITCZ can vary as much as 40°N in summer on land. The typhoons and hurricanes originating in the tropics carry a large amount of heat and rain to the poleward directions. The trade winds weaken every 2–7 years, and typically last for 9–12 months, changing the climate (e.g., heavy rainfall and droughts in countries around the Pacific from Peru to Indonesia and Australia). This phenomenon (called El Niño) is also extended to the Indian Ocean (DiNezio et al., 2020).

Because of its vast influence, tropical ocean water is of primary interest. Ocean water in the tropical region may be characterized as a warm surface water and cold deep water as shown in Fig. 5 (e.g., Northwestern Pacific Ocean, GEOSECS 214; 32°1'N 176°59'W). The whole water column, therefore, can be characterized as vertically segregated by density differences, that is, heavy cold water lies beneath the light warm water in the surface (Fig. 5; Broecker et al., 1982; Ostlund et al., 1987). The two layered ocean model has been useful to understand the vertical distribution of seawater properties as well as their horizontal segregation in the global ocean (Broecker and Peng, 1982).

In Fig. 5, the salinity profile at a depth just over 600 m reflects the presence of the oxygen depleted intermediate water that forms in the northern Pacific and sinks and flows laterally beneath the waters of the warm temperate ocean. Dissolved oxygen is utilized by animals and bacterial living in the deep sea to “burn” the organic matter falling from the surface. Thus, all the deep waters are deficient in oxygen, compared to the amount they equilibrated with the overlying atmosphere. The increase of the dissolved oxygen in the deep water reflects the intrusion of northward bottom water recharged with air in the Southern Ocean. As oxygen is consumed, silicate, phosphate and nitrate are regenerated. The life processes are one of the most salient features of the oceans. The silicon, nitrogen, and phosphorus are the essential constituents of the marine lives, along with carbon.

Three major types of particles formed in the sunlit surface ocean through photosynthesis are organic matter (e.g., Redfield molecule, $C_{106}H_{263}N_{16}P$), calcium carbonate ($CaCO_3$), and opaline silica (SiO_2). The plant and animals produce calcium carbonate tests or opaline silica frustules in the sunlit surface ocean and those particulate matters are decomposed or dissolved while they sink through the water column. A fraction of them arrive at the sea floor and are eventually buried there.

The ocean regulates the climate of the Earth (our human habitat) by the redistribution of heat from the tropics to the poles through its deep-water circulation (conveyor belt) and the exchange of the most important climate gas (carbon dioxide) through thermohaline ocean circulation and biological pumping. The biological pump refers to a particulate form of carbon transported

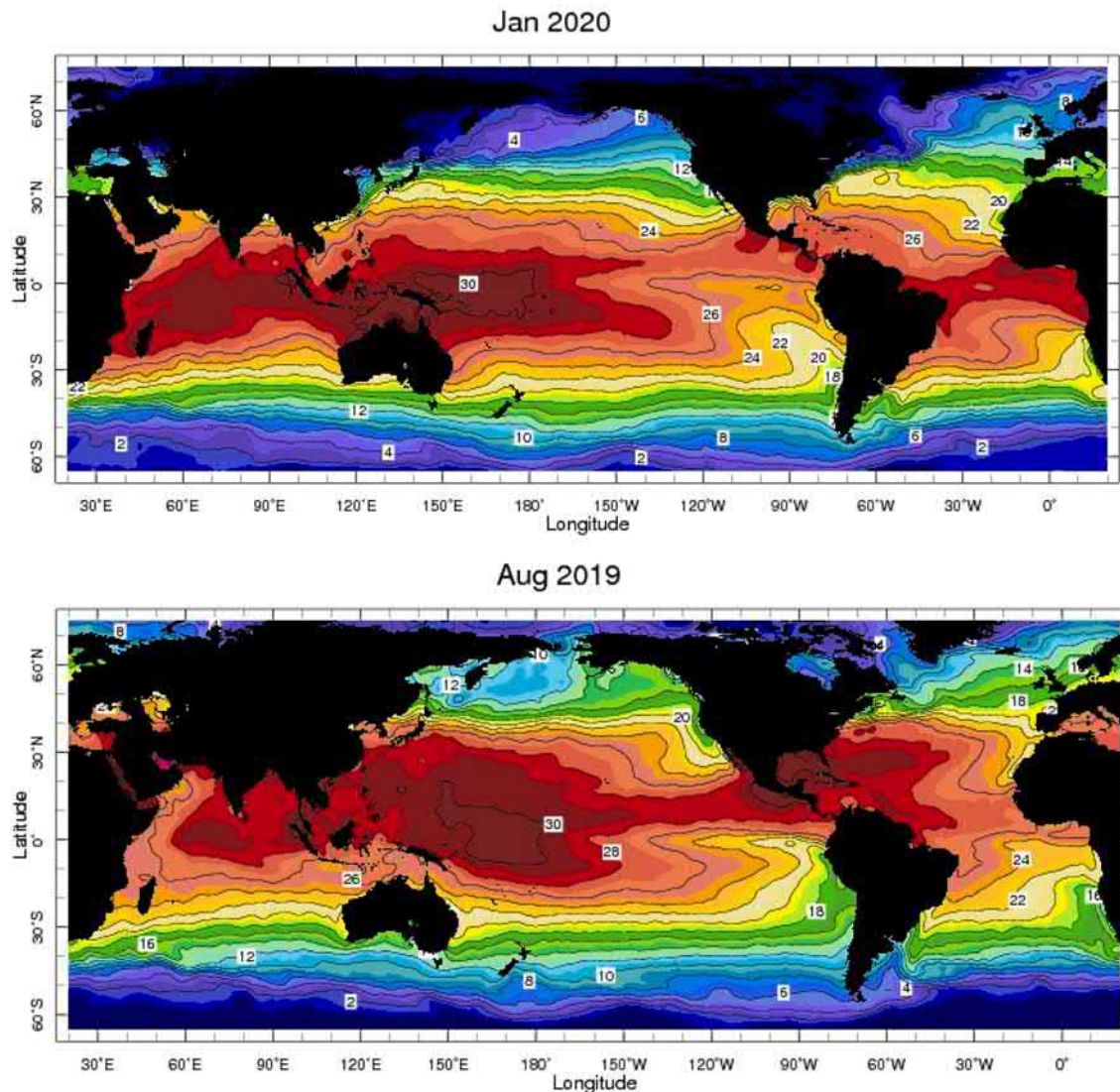


Fig. 4 Global sea surface temperature. Temperatures in squares are in degree Celsius (°C). Courtesy of Lamont Doherty Geological Observatory, 2020.

from the sunlit surface ocean to the dark deep sea and sea floor. In the real-world ocean, the concentrations of the biologically active constituents (e.g., dissolved SiO_2 (noted as $\text{Si}(\text{OH})_4$), NO_3^- , PO_4^{3-}) show considerable geographic and depth variability. As shown in Fig. 5, the concentrations of the biologically active constituents increase with depth in the North Pacific to a mid-depth maximum and then decrease somewhat toward the bottom. This type of profile characterizes the Pacific and Indian Oceans from the equator to about 40°N and 40°S. In the Atlantic, the concentrations of these constituents increase all the way to the bottom. In the deep water below 1000 m depth, the nutrient concentrations increase from the Atlantic sequentially to the Indian, South Pacific, and North Pacific oceans due to the thermohaline deep-water circulation, so called Great Ocean Conveyor Belt (Fig. 6; NOAA, 2020).

Marine biological organisms

It is important to recognize that oceanic water bodies contain many forms of life, largely supported by the organic matter produced by phytoplankton through photosynthesis converting carbon dioxide and nutrients in seawater using solar radiation in the upper euphotic 200 m water column. Chemosynthesis also occurs in some localized locations using reduced sulfur in oceanic ridges (e.g., 1 km long and several hundred meters wide, the Pacmanus hydrothermal vent field in the eastern Manus Basin, Papua New Guinea; Van Dover et al., 2018) or using hydrocarbons (usually methane) seeped from the organic matter rich continental margins (e.g., 1000 m² in the Peruvian margin; Tunnicliffe et al., 2003).

The intensity of light rapidly dissipates as depth increases, especially beyond a depth of 100 or 200 m. The zone between about 100–200 m and 1000 m is usually referred to as the “twilight” zone, which is usually substantially colder than the surface. About

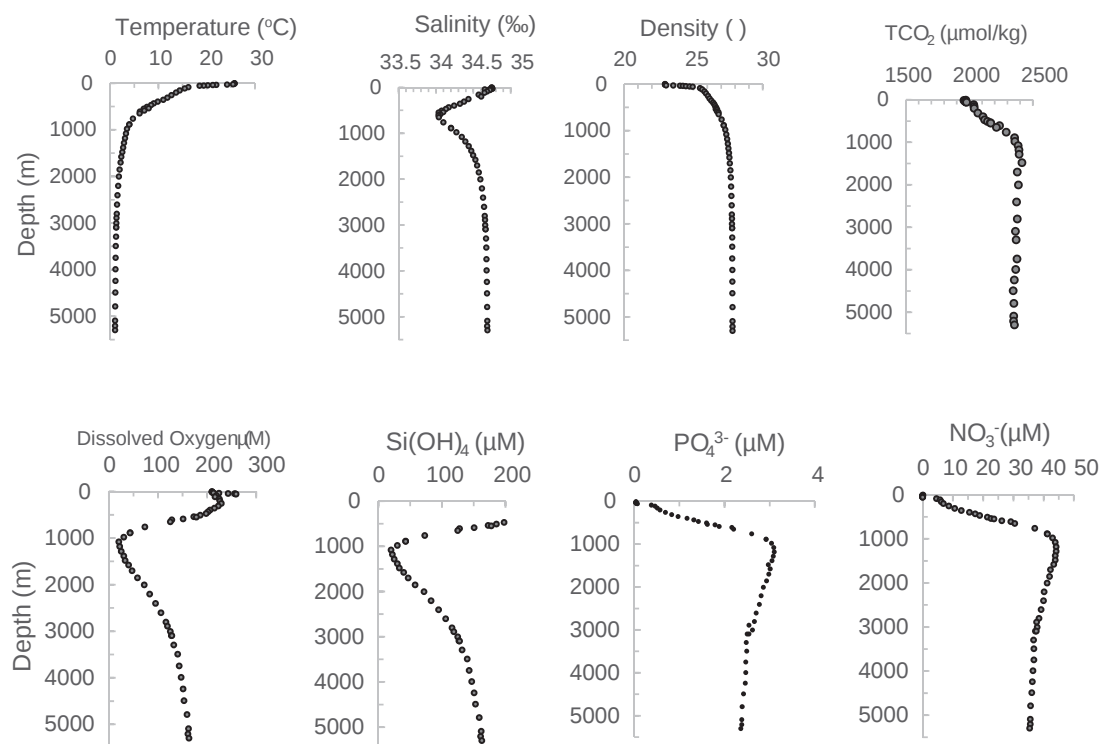


Fig. 5 Depth profiles of potential water temperature ($^{\circ}\text{C}$) corrected for the adiabatic pressure, salinity (‰ , by weight), potential density (σ_{θ} , the density calculated using in situ salinity and potential temperature minus 1000 kg m^{-3}), dissolved oxygen ($\mu\text{mol/kg}$), dissolved silicic acid (SiO_2 , $\mu\text{mol/kg}$), phosphate ion ($\mu\text{mol/kg}$), nitrate ion ($\mu\text{mol/kg}$) at GEOSECS station 214 in the North Pacific ($32^{\circ}1'N$ $176^{\circ}59'W$). Data from Broecker WS, Spencer DW and Craig H (1982) GEOSECS Pacific Expedition. In: *Hydrographic Data 1973–1974*, Vol. 3, 154 p. Washington, DC: International Decade of Ocean Exploration, National Science Foundation; and Ostlund HG, Craig H, Broecker WS and Spencer D (1987) *GEOSECS Atlantic, Pacific, and Indian Ocean Expedition. Shorebased Data and Graphics*, Vol. 7, 223 p. Washington, DC: National Science Foundation.



Fig. 6 The path of the global ocean conveyor belt. The blue arrows indicate the path of deep, cold, dense water currents. The red arrows indicate the path of warmer, less dense surface waters. It is estimated that it can take 1000 years for a parcel of water to complete the journey along the global conveyor belt. From NOAA (2020) The Global Conveyor Belt. National Oceanic and Atmospheric Administration. https://oceanservice.noaa.gov/education/tutorial_currents/05conveyor2.html.

90% of particulate organic matter sinking from the upper water column is decomposed in this twilight zone (Buesseler et al., 2020), thus playing a major role in removing carbon dioxide from the atmosphere and storing it for centuries or longer in the ocean. It is also privy to the largest migration on Earth. Huge numbers of fishes and zooplankton move hundreds of meters toward the surface to feed, before retreating down at dawn.

The twilight zone has been increasingly recognized for its importance in maintaining a healthy ocean. Phytoplankton growing in the sunlit layer fuel multiple food supply routes into the zone that sustain organisms from bacteria to giant squid. In the process of consuming this food, and each other, the twilight zone animals produce CO_2 , consume oxygen, and release nutrients back into the water as shown in the depth distributions of phosphate and nitrate in the North Pacific (Fig. 5). In the winter, cold and windy weather mixes the deep-water rich in the regenerated nutrients with water from the surface layer. In this way, the twilight zone has an important role in supporting phytoplankton growth in next spring. Although winter mixing can release carbon back into the atmosphere, a large fraction of carbon ends up in deeper waters where it can be locked away, typically for centuries. This downward transport of organic matter, mediated by life in the twilight zone, is called the biological carbon pump, and the twilight zone is central to its strength. This deeper flux of materials becomes food for the animals there. The small amount that eventually reaches the sea floor sustains everything from bacterial to sea cucumbers.

The aphotic, or “midnight,” zone exists in depths below 1000 m of the abyssal ocean. Sunlight does not penetrate the ocean deeper than 1000 m, beyond which it is shrouded in utter darkness. Biological organisms metabolize continuously in the abyssal ocean deeper than 4000 m, but their metabolic rates have been observed to be significantly lower than those in shallow water due to the increased hydrostatic pressure and/or diminished food and chemical energy available—rather than cold water temperature (Brown et al., 2018).

Behavior of radionuclides in the sea

The oceans and coastal waters are influenced by a complex variety of physical, geochemical and biological processes, which influence the behavior, transport and fate of radionuclides released into the marine environment. Among them, suspended particulate matter and biological organisms are of interest in terms of the transport and fate of radionuclides in the sea (Fig. 7; Benitez-Nelson et al., 2018). Although the suspended particulate matter is present at an order of a few tens of parts per billion ($\mu\text{g}/\text{kg}$) level, it scavenges metallic elements such as Th, Pu, Fe, Pb, and Cu from the ocean water column, thereby controlling their concentrations. The

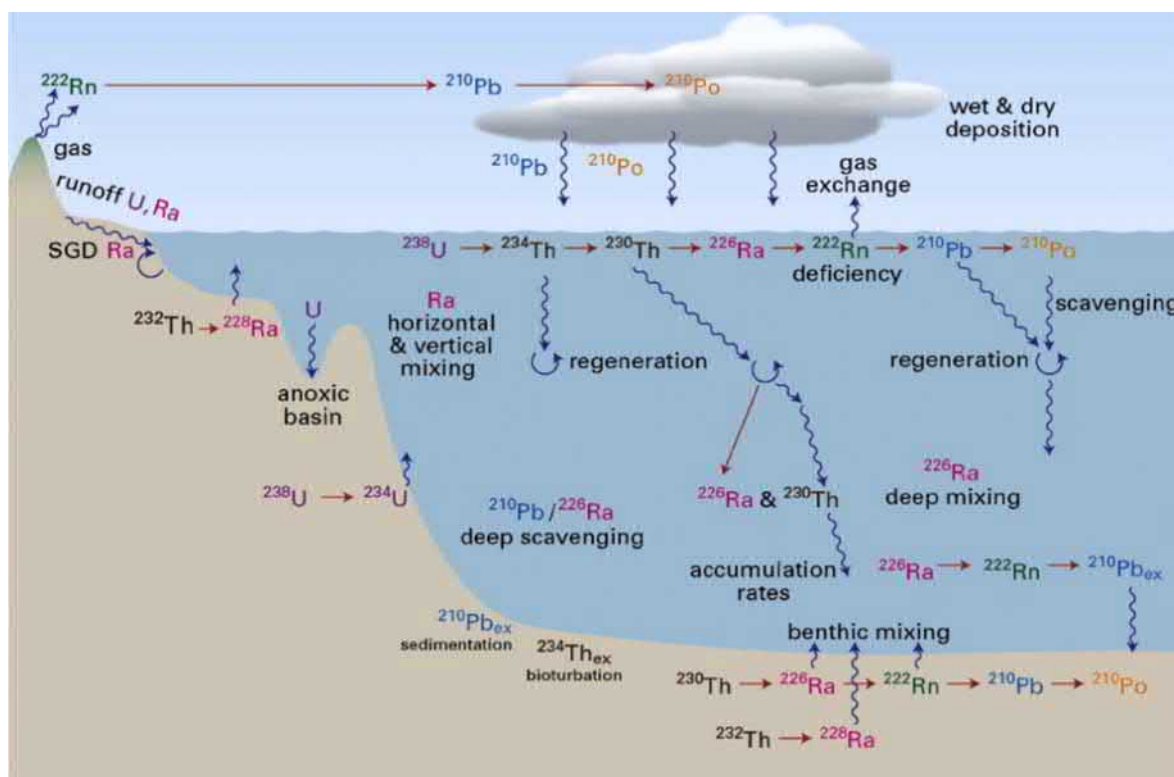


Fig. 7 Schematic diagram of the U-Th series radionuclides utilized to understand various processes occurring in the ocean using different chemical affinity to water borne particles between daughter and parent radionuclides. From Benitez-Nelson CR, Buesseler K, Dai M, Aoyama M, Casacuberta N, Charmasson S, Johnson A, Godody JM, Maderich V, Masqué P, Moore W, Morris PJ and Smith JN (2018) Radioactivity in the marine environment: Understanding the basics of radioactivity. *Limnology and Oceanography e-Lectures*. doi: 10.1002/loe2.10010. with permission.

suspended particulate matter is mostly made out of biogenic matter produced by phytoplankton in the upper water column, and it undergoes decomposition and dissolution during their decent. Therefore, a part of radionuclides once scavenged by suspended particulate matter is subject to be released in the deeper depth. This intricate nature of chemical processes is occurring relentlessly throughout the ocean water column and the benthic nepheloid layer adjacent to the sea floor (e.g., Lal, 1977).

Metallic radionuclides and suspended particulate matter

Many of the radionuclides introduced into the marine environment are likely to be taken up by organisms or adsorbed onto suspended and settling particulate matter in the water column and bottom sediments, e.g., Pu, Am. However, some metallic elements such as U, Pb, Cs, and Pu are largely present in the form of dissolved species in the ocean interior due to the low amount of suspended particulate matter (on the order of a few tens of $\mu\text{g/kg}$ or less). For example, the isotopes ^{239}Pu and ^{240}Pu are still largely present in the dissolved phase after their introduction more than seven decades ago (Hong and Povinec, 2021). Once introduced into the sea, they are ionized and dissolved in seawater and adsorbed onto the surface of particulate matter. Some of them are actively taken up by the plants and animals. In general, metal species present in seawater are in a form of free metal ions, inorganic pairs ($< 1\text{ nm}$), inorganic complexes, organic complexes, chelates, metal species bound to high molecular weight organic material ($< 10\text{ nm}$), metal species adsorbed on colloids ($< 10\text{ nm} \sim < 120\text{ nm}$) or aggregated colloids ($\sim < 0.2\text{ }\mu\text{m}$), and adsorbed on particulate matter ($\sim > 0.2\text{ }\mu\text{m}$) (Stumm and Morgan, 1981).

The combined amount of free metal ions, inorganic pairs, inorganic complexes, organic complexes, chelates, and metal species bound to high molecular organic material ($< 10\text{ nm}$) may be considered as the truly dissolved phase. For example, about 35%, 49%, 16% of ^{210}Pb are found in dissolved, colloidal, and particulate phases, respectively, while 21%, 34%, and 44% of ^{210}Po in the upper 200 m water column of a marginal sea of the Northwest Pacific are found in dissolved, colloidal, and particulate phases (Kim and Kim, 2012). However, it is often difficult to separate dissolved and colloidal substances at sea. The dissolved phase is operationally defined as the sum of truly dissolved and colloidal phases, and the cutoff diameter is usually taken as about $0.4\text{ }\mu\text{m}$. The metal fraction below the cutoff diameter is defined as the dissolved phase, where the metal fraction above the cutoff diameter is defined as the particulate phase. In order to understand the portion of particulate phase of metallic radionuclides, the distribution coefficient (K_d) is operationally defined as the ratio of the metallic contents between particulate and dissolved phases (Eq. 1).

$$K_d = \frac{C_p}{C_d \times \text{TSPM}} \quad (1)$$

where C_p (Bq/L of water) and C_d (Bq/kg) are activity concentrations of radionuclides in particulate and dissolved phases of seawater, respectively, and TSPM ($\mu\text{g/L}$ of water) is the total suspended particulate matter concentration. K_d values (L/kg) for metals or radionuclides are largely determined from seawater concentration and suspended particulate matter in a given parcel of water. For the modeling studies, IAEA (2004) have compiled K_d values with respect to pelagic clay and shale. Their values may be regarded as an upper limit, as most particulate matter in the ocean proper away from the coast is composed of many biologically produced particles of calcium carbonate, siliceous frustules, and organic matter that may be less adsorbed by metallic elements. Selected K_d values are listed in Table 1 (IAEA, 2004).

If TSPM is known, the relative proportion of metal concentrations in the particulate and dissolved phases may be estimated by the modification of Eq. (1).

$$\frac{C_p}{C_d} = K_d \cdot \text{TSPM} \quad (2)$$

The total concentration of a radionuclide (C_t) is the sum of $C_p + C_d$. Therefore, a fraction of particulate phase over total amount of metal or radionuclide (Eq. 2) may be estimated using Eq. (3).

$$\frac{C_p}{C_t} = \frac{C_p}{C_p + C_d} = \frac{1}{1 + \frac{1}{K_d \cdot \text{TSPM}}} \quad (3)$$

Table 1 Distribution coefficient of selected elements (recommended values taken from IAEA, 2004).

Element	K_d (L/kg)	Phytoplankton	Zooplankton	Crustacean muscle	Fish fillet (surface water)	Cetaceans muscle (whale, dolphins, porpoises)
Sr	2×10^2	1×10^0	2×10^0	5×10^0	3×10^0	
Tc	1×10^2	1×10^3	1×10^3	1×10^3	1×10^1	
I	2×10^2	8×10^2	3×10^3	3×10^0	9×10^0	
Cs	2×10^3	2×10^1	4×10^1	5×10^1	1×10^2	3×10^2
Pb	1×10^7	1×10^5	1×10^3	9×10^4	2×10^2	4×10^4
Po	2×10^7	7×10^4	3×10^4	2×10^4	2×10^3	
Th	5×10^6	4×10^5	4×10^3	1×10^3	6×10^2	
Pu	1×10^5	2×10^5	4×10^3	2×10^2	1×10^2	
Am	2×10^6	2×10^5		4×10^2	1×10^2	
U	5×10^2	2×10^1				

Since the K_d value of stable Pb in the ocean is about 1×10^7 and the concentration of dissolved Pb is 4.0×10^{-15} Pb g/L (IAEA, 2004), the C_p/C_d ratio should be equal at the TSPM of 100 $\mu\text{g/L}$. However, in the open ocean, it is still rare to witness simultaneous measurements of suspended particulate matter concentration and particulate radionuclide activity concentration. For example, without determining TSPM concentrations directly, particulate ^{210}Pb values are reported to account for less than 10% in most subsurface waters, and close to 40% in near bottom turbid waters in the global ocean. This is determined by measuring the ^{210}Pb activity concentration in the solid retained on the filter and in the filtrate (Rigaud et al., 2015; Nidermiller and Baskaran, 2019). Therefore, recommended values given by IAEA (2004) should be regarded as maximum estimates. It is also noted that the distribution coefficient as a quotient of activity concentration in ^{210}Pb in particulate and dissolved phases is inversely proportional to the concentration of TSM. In turbid coastal waters where TSPM concentration is greater than mg/L level (e.g., the Yellow Sea, Hong et al., 1999), a few tens of $\mu\text{g/L}$ level of TSPM are found in the deep ocean. The fraction of particulate phase of ^{210}Pb and ^{210}Po were found to reach almost 100% in seawater with concentrations greater than 4 mg/L (Hong et al., 1999).

Concentration factors

For a given substance in a liquid, the ratio of its concentration in the two different phases is constant at a given temperature, provided that the molecular state of the distributed solute is same in both phases according to Nernst's distribution or partition law. It is expressed as $CF = C_1/C_2$ where C_1 and C_2 are the concentrations of the solute in phase 1 and phase 2, respectively. C_2 is usually the concentration of a given substance in a liquid phase, operationally defined as the phase of less than 0.4 μm in this study. CF is called concentration factor in the case where C_1 is the concentration in biological organism phase (Eq. 4), assuming they are in equilibrium and the uptake rate is much shorter than the observation frequency.

$$CF (\text{L/kg}) = \frac{\text{Concentration per unit mass of organism} \left(\frac{\text{Bq}}{\text{kg wet weight}} \right)}{\text{Concentration per unit volume of seawater} \left(\frac{\text{Bq}}{\text{L}} \right)} \quad (4)$$

A strong positive correlation was observed for many metallic radionuclides, such as ^{137}Cs levels in the muscle of phytoplankton, zooplankton, crustacean, fish, cetaceans (Table 1). Bioavailability of a given substance may be strongly dependent upon the specification of the substance where the free ion form is commonly believed to be the bioavailable form for direct uptake from seawater. It should be noted that, except for algae, the term CF as used here does not imply that all the elements within the organism are concentrated by direct accumulation from the water. It is simply a value that relates the concentration in the organism, which may have been derived by uptake from sea water, particulate matter, and food, to that of the medium in which it lives (IAEA, 2004).

Biomagnification

Biomagnification, also known as bioamplification or biological magnification, is any concentration of a toxin, such as pesticides or metals, in the tissues of tolerant organisms at successively higher levels in a food chain. Concentration factors for selected radionuclide metals are compared among various trophic level organisms to elucidate whether they are biomagnified through the higher food web (Fig. 8; IAEA, 2004). Among the elements Cs, Pb, Po, Pu, Am, Cesium (Cs) appears to be biomagnified in a higher food web in the phytoplankton-zooplankton-fish food chain, and Pu in a higher food web in the macroalgae-molluscs, e.g., abalone (Kim et al., 2020). The ^{137}Cs bioaccumulation is driven by relatively rapid dietary uptake rates, moderate depuration rates in lower trophic level organisms and slow elimination rates in high trophic level organisms.

Uptake kinetics of radionuclides from water

The massive ^{137}Cs released into the sea from the Fukushima Dai-ichi Nuclear Power Plant (FDNPP) accident in 2011, and subsequent large-scale time-series measurements in both seawater and fish, provided a rare opportunity to observe the relationship between CF and the exposure time. The ^{137}Cs concentrations in seawater ranged from < 10 Bq/kg in 2016 (Hori et al., 2018) to about 1000 Bq/kg ($^{134}\text{Cs} + ^{137}\text{Cs}$) at the beginning of the accident in 2011 (Povinec et al., 2013a). Wang et al. (2018) collated those data during the initial 400 days and they found the time series variation of ^{137}Cs activity in fish flesh depended on the activity concentration in seawater and as a function of time. For the pelagic fish and for the surface seawater, ^{137}Cs (Bq/kg) = $305.51e^{(-0.012 \times \text{day})}$ ($R^2 = 0.448$) ($n = 102$) and ^{137}Cs (Bq/L) = $53.442e^{(-0.023 \times \text{day})}$ ($R^2 = 0.921$, $n = 2042$) were observed, respectively. R^2 and n refer to the square of sample correlation coefficient, and size of samples, respectively. If we calculate CF using the two equations, CF of ^{137}Cs in pelagic fish flesh is given as a function of time, $CF = 5.71e^{(0.011 \times \text{day})}$. This equation suggests that the exposure time is a limiting step for the accumulation of ^{137}Cs in the sea. The obtained CF value of 316 at the 365th day may be a little higher than the average values reported.

A similar feature was observed in a higher-trophic level animal than fish, such as marine mammals. The salmon-resident killer whale food web ^{137}Cs bioaccumulation model outcomes show that ^{137}Cs can be expected to gradually bioaccumulate in the food web over time as demonstrated by the increase of the apparent trophic magnification factor of ^{137}Cs , ranging from 0.76 after 1 month of exposure to 2.0 following 30 years of exposure (Alava and Gobas, 2016). In general, a substance including radionuclide

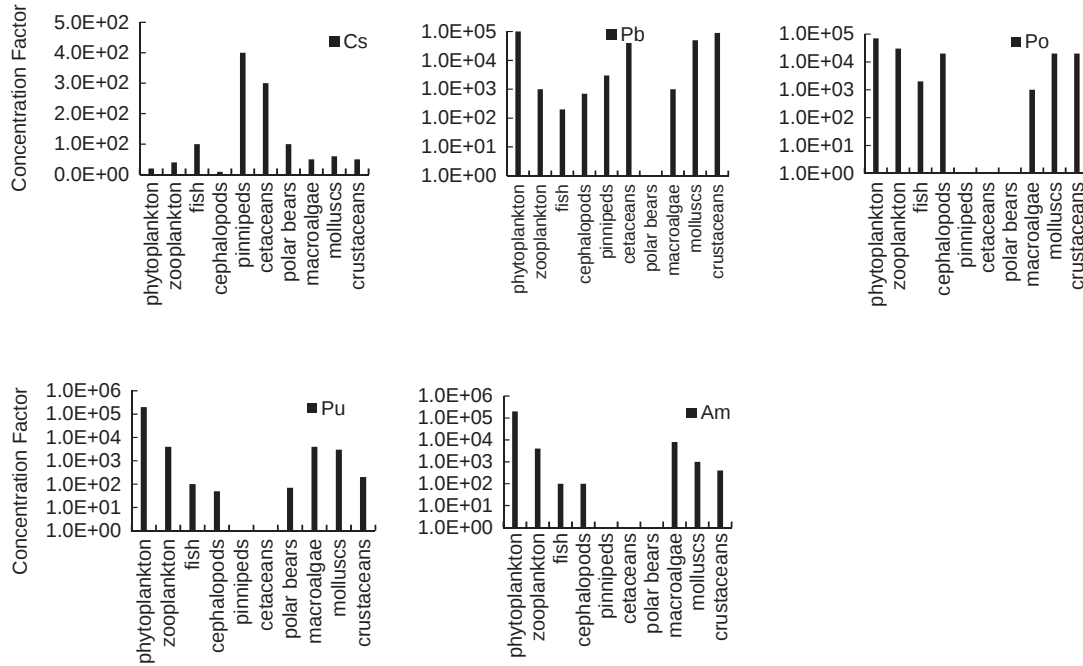


Fig. 8 Concentration factors of Cs, Pb, Po, Pu and Am. Data from IAEA (2004) Sediment distribution coefficients and concentration factors for biota in the marine environment, 103 p. Vienna: International Atomic Energy Agency, Technical Report Series no. 422.

accumulation in biological organisms in water has been described by the following first-order reaction in Eq. 5 with respect to the concentration of the substance in water and prey material (cited in Wang et al., 2000).

$$dC/dt = k_u \cdot C_w + AE \cdot IR \cdot C_f - k_e \cdot C \quad (5)$$

where C is the radionuclide concentration in the animal organs at time t ($\mu\text{g/g}$ dry or wet wt. or Bq/g), k_u is the radionuclide uptake rate constant from dissolved phase (L/g day or L/Bq day), C_w is the radionuclide concentration in seawater ($\mu\text{g/L}$ or Bq/L), AE is the radionuclide assimilation efficiency, IR is the animal's food particle ingestion rate ($\text{g/g of body wt}^{-1} \text{ day}^{-1}$), C_f is the radionuclide concentration in the ingested food particles ($\mu\text{g/g}$ or Bq/g), and k_e is the radionuclide efflux rate constant (day^{-1}). All weight units of tissues were in dry or wet weight. Eq. (6) assumes that the body concentration change with time is a function of the radionuclide concentration of seawater and concentration of prey organisms. This tacitly assumes the uptake via seawater and uptake by consumption of prey are both instantaneous reactions. The rate constant for uptake from the dissolved source (k_u) is metal and species specific and is an indicator of bioavailability of dissolved metals (Lee et al., 1998). The radionuclide concentration in tissues of animal at steady-state, C_{ss} , can be calculated as:

$$C_{ss} = (k_u \cdot C_w + AE \cdot IR \cdot C_f) / k_e \quad (6)$$

If animals are not fed, radionuclide uptake occurs only from seawater, and the ratio of rates of uptake over depuration (excretion) in Eq. (5) at steady state is the ratio of concentration in animal over concentration in water, $k_u/k_e = C_{ss}/C_w$, that is the concentration factor (CF) as defined in Eq. (4). In dilute solutions, as in the case of concentration in the order of $\text{fmol } ^{137}\text{Cs/L}$ (equivalent to 10 Bq/L of ^{137}Cs) in this study, n becomes 1, as described in the Freundlich isotherm ($C = \text{CF} \cdot C_w^{1/n}$) shown in Lee et al. (1998)'s metal uptake experiments on the bivalves. However, the empirical Freundlich isotherm does not predict C_{ss} or an adsorption maximum that we have often observed. The Langmuir isotherm has been used widely to describe empirically the amount of adsorbate per unit mass of adsorbent as a function of the adsorbate concentration remaining in solution to predict the adsorption maximum at equilibrium (Eq. 6). Smith (1999) briefly summarized the Langmuir isotherm as following Eq. (7)

$$q = \frac{kbC}{1 + kC} \quad (7)$$

where q is the amount of adsorbate per unit mass of adsorbent, C is the adsorbate concentration remaining in solution at equilibrium, b is the maximum adsorption capacity of the adsorbent (monolayer coverage), and k is a constant related to the binding strength. The Langmuir isotherm assumes (i) a finite number of surface sites, (ii) no electrostatic or chemical interactions, (iii) constant binding energy for all surface sites, (iv) assuming that the binding energy is independent of the adsorption density, (v) assuming that the maximum amount of adsorption is limited to a monolayer coverage of the surface sites, and (vi) there is a large excess of substrate over adsorbent.

When the above equation is rearranged, a plot of C/q against C results in a linear representation of adsorption data with a slope of $1/b$ and an intercept of $1/kb$. In many cases, the sorption in natural systems deviates from this linear representation. Rearranging variables in Eq. (7) gives the Michaelis-Menten equation of enzyme kinetics that is also widely used to describe kinetics of nutrient uptake by phytoplankton at sea, $V_M = \frac{V_{max,s} \cdot S}{K_s + S}$, where V_M is the uptake rate for a nutrient with the concentration S , $V_{max,s}$ is the maximum uptake rate for the nutrient, and K_s is the half-saturation constant. Alava and Gobas (2016) showed that ^{137}Cs activity (Bq/kg wet wt.) in fish body followed the hyperbolic function of seawater concentration (Bq/L).

The uptake of radionuclides from the dilute solution by animals may be similar to nutrient (or metal) uptake by phytoplankton from the dilute solution of seawater. That is, nutrient (or metal) concentration in sea water is rather given at the fixed concentration, but the incorporation of nutrient (metal) into the phytoplankton cells depends on the uptake (adsorption and absorption) characteristics of the phytoplankton of concern. In this case, uptake reaction is assumed to be completed instantaneous. However, the accumulation of extracellular compartments, characterized by an initial rapid accumulation, then follows a gradual slowing down over time. The result fits perfectly to a Michaelis-Menten model (Fernandez et al., 2006). Fowler et al. (2010) also observed a hyperbolic function like uptake of ^{207}Bi as volume concentration factor (VCF) with time in phytoplankton cells. The accumulation of radionuclide in the flesh of oysters and the digestive tract of flounders may be quite similar to the other metal uptake in seawater where the metal concentration in sea water is fixed.

The purposes of this study are to look at the response of oyster and flounder with time to uptake radionuclides in seawater of a fixed concentration of the radionuclides in relatively overwhelmingly large pool of radionuclide contaminated seawater—the entire north-west Pacific Ocean. Among the radionuclides tested, ^{137}Cs accumulation in the flesh of oysters and the digestive tract of flounders with exposure time appeared to follow the hyperbolic function (Fig. 9) identical to the Michaelis-Menten equation Eq. (8).

$$\frac{\text{Bq}}{\text{kg wet wt}} = \frac{m \cdot [t]}{k + [t]} \text{ or CF} = \frac{m \cdot [t]}{k + [t]} \quad (8)$$

where t is the duration of exposure.

The accumulation rate (expressed as CF) of a given radionuclide is proportional to the exposure time with two fixed values of m and k . The m and k are maximum or steady state concentrations, and the time t is taken to reach the half the value of m . For the CF, m is the maximum or steady state CF and k is the time taken to reach half of the value of maximum CF (Fernandez et al., 2006). They further defined steady state concentration in animal tissue as the tissue concentration from which the daily increase in concentrations is less than 1% of that of the previous day. The accumulation of radionuclide in the flesh of oyster faithfully appears to follow the hyperbolic function in the unfed oyster culture experiment, $[^{137}\text{Cs} \text{ Bq/kg wet wt.}] = 54.20 \times (\text{hour})/[24.87 + (\text{hour})]$, $R^2 = 0.89$ as shown in Fig. 9. The hyperbolic function predicts that ^{137}Cs activity will reach 51.6 Bq/kg wet wt. on the 21st day, 52.9 Bq/kg wet wt. on the 42nd day, and 54.1 Bq/kg wet wt. on the 420th day of exposure (Fig. 9).

Where the ocean is turbid (more SPM), the dissolved concentration is relatively less in the dissolved in seawater. The metals concentration in fish muscle is relatively higher in clear waters than in turbid waters because dissolved phase metal is only available for bioaccumulation. For example, ^{210}Po concentration is higher in squid living in the turbid Yellow Sea ($\sim 10 \text{ mg/L} < \text{TSPM} < \sim 110 \text{ mg/L}$) compared ones living in the clear East Sea/Sea of Japan where the TSPM is generally $< 0.55 \text{ mg/L}$ (Waska et al., 2013; Chen et al., 1996).

Climate change and the ocean

The level of atmospheric CO_2 concentration is measured as a dry air mole fraction, defined as the number of molecules of carbon dioxide divided by the number of molecules in air including CO_2 itself, after water vapor has been removed. The mole fraction is

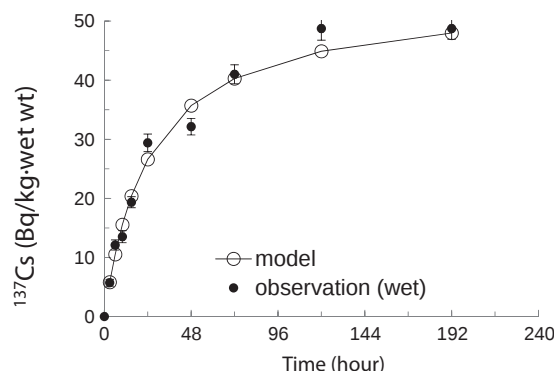


Fig. 9 ^{137}Cs accumulation in oyster flesh appears to follow the hyperbolic function, $y = m \cdot x/(k + x)$, $[^{137}\text{Cs} \text{ Bq/kg wet wt.}] = 54.20 \times (\text{hour})/[24.87 + (\text{hour})]$, $R^2 = 0.89$. The oysters were placed in each aquarium and kept unfed and aerated until being sacrificed. Data from Kim SH and Hong GH, unpublished.

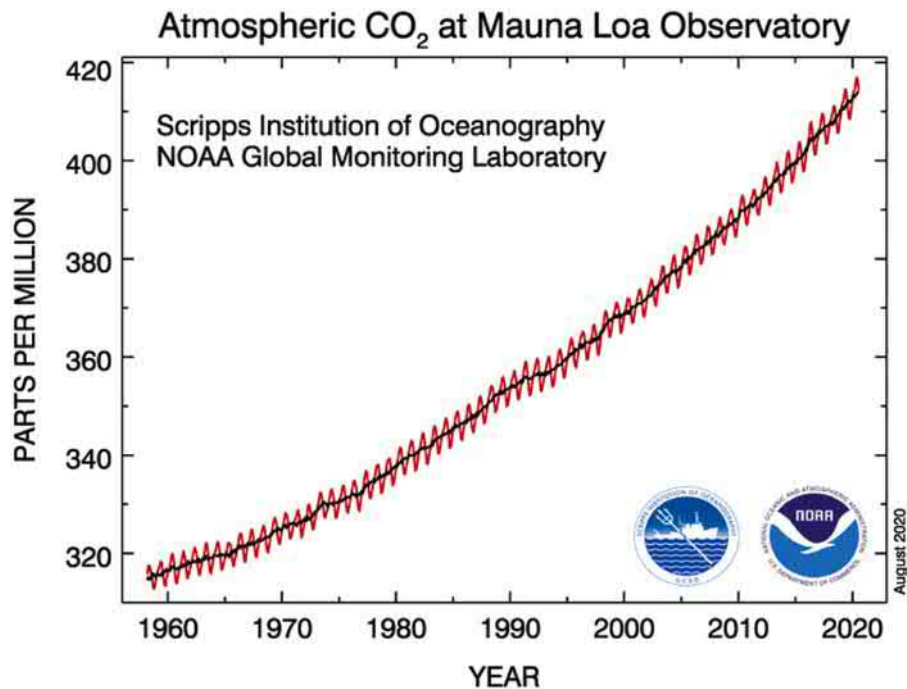


Fig. 10 Monthly mean carbon dioxide measured at Mauna Loa Observatory, Hawaii. The observatory is located on the north flank of Mauna Loa Volcano, on the Big Island of Hawaii, at an elevation of 3397 m and it has started in 1959 (316 ppm, annual average). The annual mean rate of growth of CO_2 has steadily increased from 0.94 ppm in 1959 to 2.46 ppm in 2019. Courtesy of NOAA, https://www.esrl.noaa.gov/gmd/webdata/ccgg/trends/co2_data_mlo.png.

expressed as parts per million (ppm). The current level of CO_2 in the air (June 2020) is about 416 ppm (Fig. 10; NOAA), much higher than that of preindustrial levels (~ 280 ppm) in 1750.

For the last decade (2009–18), combustion of fossil fuels including other industries (e.g., cement production) emits 9.5 ± 0.5 Gt C (giga ton carbon) annually into the atmosphere. A land use change emits 1.5 ± 0.7 Gt C annually to the atmosphere. The land use change here refers to clearing forest or grassland that emits the stored carbon dioxide, methane, and nitrous oxide in soil to the atmosphere. The emitted carbon dioxide accumulated in the atmosphere, ocean, and land is as much as 4.90 ± 0.02 Gt C, 2.5 ± 0.6 Gt C, and 3.2 ± 0.6 Gt C, respectively, with a budget imbalance of $0.4 \text{ Gt C year}^{-1}$ indicating over estimated emissions and/or underestimated sinks (Friedlingstein et al., 2019). About 829 Gt C of carbon is currently stored in the atmosphere, and 40,453 Gt C in the ocean (Fig. 11; IPCC, 2013).

Most carbon dioxide released into the atmosphere will eventually be absorbed by the ocean. Thus, it results in the acidification of surface ocean to decrease the pH of the oceanic surface waters from the preindustrial level of 8.2 to down to 7.8 (H^+ concentration rises by 150% (e.g., Gattuso et al., 2015, Caldeira and Wickett, 2003). At the same time the warmed atmosphere heats the surface ocean resulting in enhancing stratification, thereby decreasing nutrient availability to phytoplankton growth. Therefore, the global warming due to the increase in greenhouse gas concentrations, especially carbon dioxide, induces a significant change in the ocean chemistry (e.g., pH) and primary productivity (Gao et al., 2019).

We can see that the ocean is extremely important in the study of atmospheric carbon dioxide since it is such a large reservoir in intimate contact with the air. As soon as CO_2 migrates from the atmosphere into the water, where its concentration is lower than that of the overlying atmosphere, it reacts chemically with water molecules to form carbonic acid, which causes a shift in the concentrations of the hydrogen carbonate (HCO_3^-) and carbonate (CO_3^{2-}) ions that are derived from the carbonic acid. Because carbon dioxide is immediately processed in the sea, the CO_2 storing capacity of the oceans is 10 times higher than that of freshwater. Hence, the ocean can absorb large quantities of CO_2 . When a new carbon equilibrium between the atmosphere and the world ocean is re-established in the future, the oceanic reservoir will have assimilated around 80% of the anthropogenic CO_2 from the atmosphere, primarily due to the reaction with carbonate. The buffering effect of deep-sea calcium carbonate sediments is also important. These carbonates neutralize large amounts of CO_2 by reacting and dissolving it to some extent. Thanks to these processes, the oceans could ultimately absorb around 95% of the anthropogenic emissions. Because of the slow mixing of the ocean, however, it would take centuries before equilibrium is established. The very gradual buffering of CO_2 by the reaction with carbonate sediments might even take millennia.

For today's situation, this means that a marked carbon disequilibrium between the ocean and atmosphere will continue to exist for the decades and centuries to come. Therefore, the world ocean cannot absorb the greenhouse gas as rapidly as it is emitted into the atmosphere by humans. The absorptive capacity of the oceans through chemical processes in the water is directly dependent on

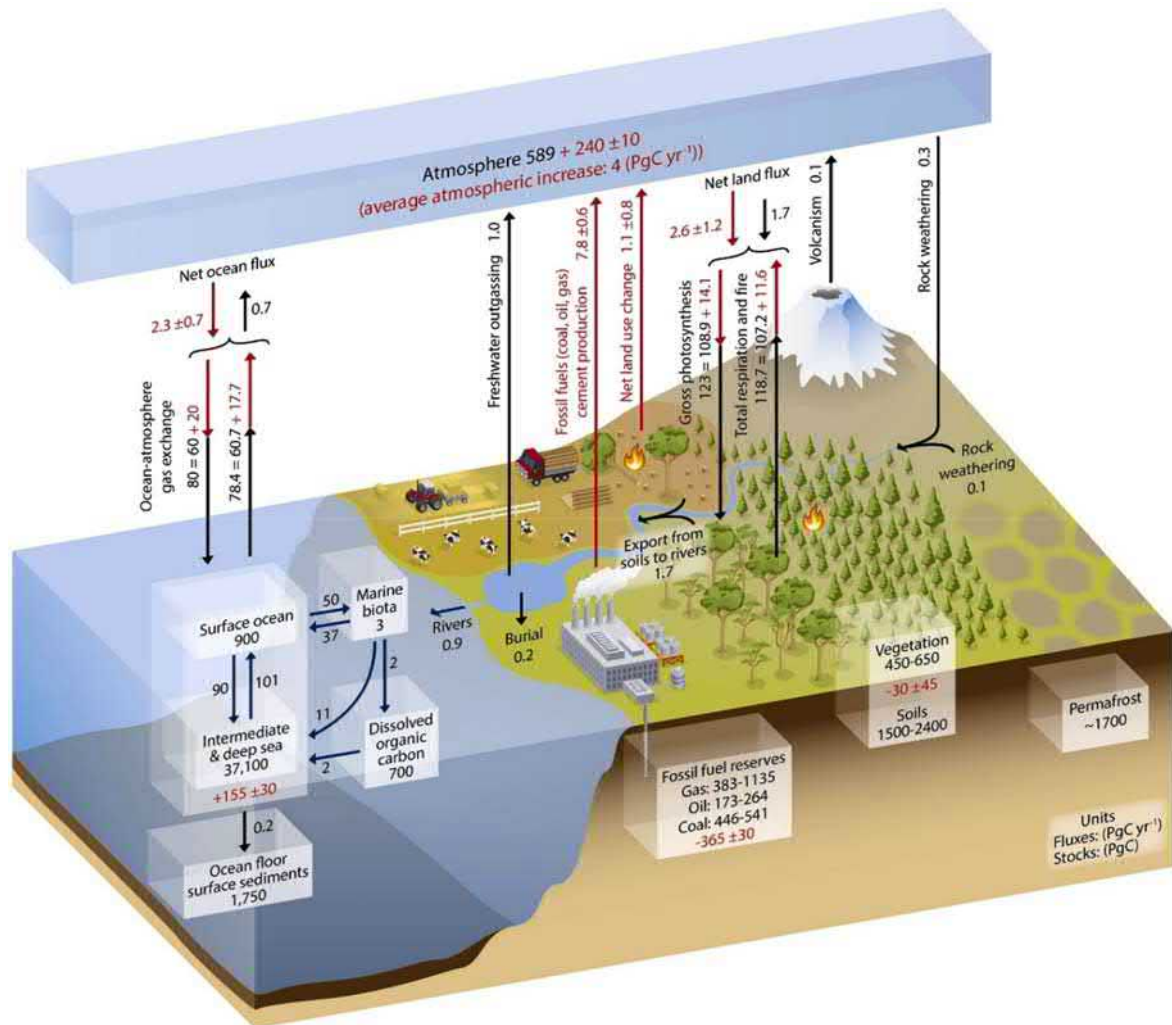


Fig. 11 Schematic of the global carbon cycle. From IPCC (2013) *Climate Change 2013: The Physical Science Basis*. Contribution of Working Group I to the Fifth Assessment Report of the Intergovernmental Panel on Climate Change. [Stocker TF, Qin D, Plattner G-K, Tignor M, Allen SK, Boschung J, Nauels A, Xia Y, Bex V, and Midgley PM (Eds.)]. Cambridge, United Kingdom and New York, NY, USA: Cambridge University Press, 1535 pp.

the rate of water mixing in the world ocean. Thus, the current oceanic uptake of CO_2 lags significantly behind its chemical capacity as the present-day CO_2 emissions occur much faster than they can be processed by the ocean (Bollmann et al., 2010).

If humans double atmospheric carbon dioxide from the pre-industrial levels (280 ppm to 560 ppm) within the next 40 years, and if the build-up of CO_2 continues at current rates (about 0.5% per year), based on the long-term record of measurements from the Mauna Loa Observatory (Fig. 10; Tans and Keeling, 2020), the likely warming range is estimated to be between 2.6°C and 3.9°C (Sherwood et al., 2020) and this will constitute a new basis for projections of sea-level rise, economic damage, and other impacts (Voosen, 2020). Heatwaves are becoming more frequent both in the land and the ocean. It is not unusual to observe dense surface blooms of toxic cyanobacteria in eutrophic lakes, leading mass mortalities of fish in lakes and rivers, and resulting in a serious health threat for cattle, pets, and humans in all over the world (Jöhnk et al., 2007). Similarly, an extensive fish kill event was found to be caused by the combination of a marine heatwave and a harmful algal bloom in South Australia (Roberts et al., 2019). A massive quantity of warm water, the Blob, suddenly appeared in the ocean area extending from Baja California in Mexico all the way to Alaska and the Bering Sea. During 2014 and 2019, it resulted in the death of many sea lions and salmon, including the production of toxic algae blooms that can poison mussels, crabs and other sea life (Amaya et al., 2020). The frequency and severity of the summertime marine heatwaves is expected to increase in various places in the ocean as the global climate warms. The impact of heat waves has been felt in almost all aspects of human activities, such as heat induced mortality increase, urban air quality deterioration, abrupt electricity demand, closing airports when the air temperature rises above 48°C , and the economic recession (Mazdiyasi et al., 2019).

Climate change is one of the important issues in our time. However, a statement like “climate change is occurring now” should be understood as “climate change is occurring now as it used to be, but at an accelerated rate.” Although human induced alternation

of the climate is certainly perturbing the natural climate system significantly, the Earth itself has undergone multiple changes over time since its formation. It will continue to do so for the next 5.4 By, until the sun will enter the red giant phase exhausting all hydrogen in the core, and the inert helium ash that will build up there becomes unstable and collapses under its own weight (Williams, 2016). In the context of the current attention to the climate change caused by human activities emitting carbon dioxide and other greenhouse gases to the atmosphere, it would be useful to look back on the climate changes that occurred in the geological past—including mass extinction of lives over time—while acknowledging the human caused changes that need to be corrected.

The effect of climate change on marine biota

Life as single-celled microbes probably began in the ocean at least 3.5 By ago (Paleo Archean Era of Archean Eon, Precambrian Era; Schopf et al., 2018). Oxygen continued to build up in the atmosphere and ocean to support complex life as photosynthesis began more than 2.5 By ago. More complex multicellular organisms evolved about 1.2 By ago (Meso-Proterozoic Era of Proterozoic Eon, Precambrian Era; Paleo Proterozoic Era of Proterozoic Eon, Precambrian Era) leading to the Great Oxidation Event. Since then, life forms have grown much diverse—though not continuously. Early life forms began to flourish during the Cambrian period (540 My). The record reveals bursts of evolution and expansion interrupted by massive extinctions that accompanied the big climate changes resulting from diverse causes. Mass extinctions—when at least half of all species died out in a relatively short time—have occurred several times over the course of our planet's history. The earliest one was Ordovician-Silurian extinction (440 My), which lasted for at least 1 My when a large portion of marine benthic animals died out due to glaciation (Wang et al., 2019). This was followed by the Devonian extinction that wiped out many tropical marine species, including reef building sponges and corals—all subject to extinction in the Devonian Period (365 My) (De Vleeschouwer et al., 2017), followed by the largest extinction event (elimination of over 95% of marine and 70% of terrestrial species in Permian-Triassic period (250 My) in the geological history of the Earth (Shen et al., 2019). More recently, the Cretaceous-tertiary extinction (K-T extinction) occurred in 66 My, when dinosaurs and the mosasaurs, plesiosaurs of land and ichthyosaurs, ammonites, and reef building bivalve rudists of the oceans disappeared. Recently, ancient specimens of rudist laying down daily growth rings revealed that Earth turned faster at the end of the dinosaurs era than it does today, rotating 372 times a year, compared to the current 365 in the late Cretaceous. (de Winter et al., 2020).

The cause of the mass extinctions in the past have been revealed with the aid of nuclear dating techniques. One of the recent examples is the 2nd largest extinction event; namely, the Ordovician-Silurian extinction. Widespread marine anoxia is hypothesized as the trigger for the second pulse of the Late Ordovician (Hirnantian) mass extinction, based on lithologic and geochemical proxies that recorded the redox level of local bottom waters or porewaters. In the ocean, $^{238}\text{U(IV)}$ is preferentially sequestered into anoxic marine sediments, leaving the residual U(VI) in seawater with a lower $^{238}\text{U}/^{235}\text{U}$ ratio, which is subsequently incorporated into carbonate sediments. The isotope ratio is often given as the relative ratio against that of standard material (Eq. 9).

$$\delta^{238}\text{U} = \left[\left(^{238}\text{U}/^{235}\text{U} \right)_{\text{sample}} / \left(^{238}\text{U}/^{235}\text{U} \right)_{\text{standard}} - 1 \right] \times 1000 \quad (9)$$

Therefore, $\delta^{238}\text{U}$ values can serve as a proxy for global-ocean redox conditions and provide a means of quantifying changes in the extent of anoxic sediment accumulation across a range of geologic time intervals. Bartlett et al. (2018) constructed a steady-state model under the assumptions (1) constant U isotopic composition of the riverine source (modern $\delta^{138}\text{U}$ river as -0.29‰), (2) constant fractionation factors for oxic/suboxic and anoxic/euxinic sinks (0.77‰), (3) constant U burial rates, and (4) constant diagenetic correction factor to account for ^{238}U enrichment during the formation of cement phases in O_2 -depleted porewaters (0.2‰). Bartlett et al.'s model shows that during the Hirnantian ocean anoxic event, there was a $\sim 15\%$ increase in anoxic seafloor area, and $\sim 80\%$ of seawater U was sequestered into anoxic sediments.

Great Ocean Conveyor Belt

Unlike the wind driven ocean currents that move with a maximum speed of an order of hundreds of cm/s in the upper few hundred meters of the ocean (e.g., Gulf Stream, Dong et al., 2019), the conveyor belt (Fig. 6) moves horizontally in the order of a few cm/s (e.g., Richardson, 2008) and vertically in the order of about 10^{-5} to 10^{-3} cm/s (Liang et al., 2017). Any direct currently available ocean current measurement methods, e.g., mechanical, electromagnetic, or acoustic meters, have measurement errors larger than a few cm/s (e.g., Guerra and Thomson, 2017). Hence, they are rarely applicable in the deep sea where the suspended particulate matter is extremely low (in the order of less than a few tens of $\mu\text{g/L}$). The apparent simple question, posed at a particular location of the sea, “what is the vertical mixing rate? and what is the age of deep water?” becomes much more complex to answer using direct measurement methods. Radionuclides present in the ocean, with known input function and half-lives along with modeling, fill this gap to measuring slow but important climate processes occurring in the ocean (Jenkins, 2003). Thanks to radionuclides quantitative information, such as rates and fluxes of energy and material, we can now better understand the ocean and are able to predict the future climate change.

The Great Ocean Conveyor Belt, termed by Broecker (1991, 2010), or the Atlantic meridional overturning circulation (AMOC), or sometimes, thermohaline circulation (THC), is the major mechanism by which the ocean redistributes heat over the globe. Significant changes in the Great Ocean Conveyor Belt will, therefore, affect the global radiation balance through altering the atmosphere's surface boundary conditions, and hence, lead to major climate change (Richardson, 2008). Broecker (1991) succinctly explained it based on the paleo air temperature inferred from the oxygen isotopes ($^{18}\text{O}/^{16}\text{O}$) record preserved in the Greenland

ice cap and radiocarbon ^{14}C -based estimate of the flux of the North Atlantic Deep Water among others, in plain language as following: “The ocean’s conveyor appears to be driven by the salt behind as the result of water-vapor transport through the atmosphere from the Atlantic to the Pacific basin. A byproduct of its operation is the heat that maintains the anomalously warm winter air temperatures enjoyed by northern Europe. A millennium of very cold conditions known as the Younger Dryas (around 12,900 to 11,700 years ago) appears to have been the result of temporary shutdown the conveyor.”

It is estimated that any given cubic meter of water takes about 600–1000 years to complete the journey along the great ocean conveyor. This movement consists of an immense volume of water, $17.0 \pm 4.4 \text{ Sv}$ ($1 \text{ Sv} = 1.0 \times 10^6 \text{ m}^3 \text{ s}^{-1}$), with seasonal and interannual variations (Frajka-Williams et al., 2019). This volume of water is about 15 times greater than all world river discharge combined (1.14 Sv , Suzuki et al., 2018). Therefore, any changes in the global ocean conveyor belt would bring drastic changes in our climate—as happened in the past. For example, if its slowdown since the middle of the last century continues, Europe will be much colder in winter and hotter in summer, along with increased iceberg streams in the North America as in the past 12.9 and 11.6 thousand years (Bromley et al., 2018).

On the other hand, in the glacial period, the ocean conveyor belt moved faster and stronger as evidenced from high resolution sedimentary records of $^{231}\text{Pa}/^{230}\text{Th}$ and $^{143}\text{Nd}/^{144}\text{Nd}$ in the deep western North Atlantic (Böhm et al., 2015). The isotopes ^{231}Pa ($t_{1/2} = 3.25 \times 10^4$ years) and ^{230}Th ($t_{1/2} = 7.52 \times 10^4$ years) are produced in the ocean at constant rates by radioactive decay of ^{235}U and ^{238}U , respectively. They have remained in the ocean for about 20 to 40 years and about 100–200 years (e.g., Yu et al., 1996), respectively, and both have exhibited high particle affinity. In the well-ventilated N Atlantic, the activity ratio of $^{231}\text{Pa}/^{230}\text{Th}$ in recently sedimented particles is low (0.093), as approximately half of ^{231}Pa produced in the North Atlantic is exported to the Southern Ocean where it is scavenged out in the high-opal (SiO_2) flux region. In the simplest terms, a slowdown of Atlantic meridional overturning (Great Ocean Conveyor Belt) should result in less ^{231}Pa exported to the Southern Ocean and, thus, higher Atlantic $^{231}\text{Pa}/^{230}\text{Th}$ ratios (Bradtmeier et al., 2014). ^{143}Nd ($t_{1/2} = 1.06 \times 10^{11}$ years) is produced from primordial ^{147}Sm ($t_{1/2} = 1.06 \times 10^{11}$ years), and ^{144}Nd ($t_{1/2} = 7 \times 10^5$ years) is produced from primordial ^{148}Sm ($t_{1/2} = 7 \times 10^{15}$ years). The Nd isotope ratio is expressed usually as $\epsilon_{\text{Nd}} = ((^{143}\text{Nd}/^{144}\text{Nd})/0.512638) - 1 \times 10^4$, where 0.512638 is a present day average Earth value ($^{143}\text{Nd}/^{144}\text{Nd}$) of the Chondritic Uniform Reservoir (CHUR) (Bizimis and Scher, 2016; Huang et al., 2012). The gradual ϵ_{Nd} increase has occurred for intermediate and deep-water masses from the northwest North Atlantic ($\epsilon_{\text{Nd}} < -13$), via the Southern and Indian oceans to the Pacific ($\epsilon_{\text{Nd}} < -3$), relative to the low ϵ_{Nd} in the North Atlantic being surrounded by old, highly unradiogenic crustal rocks, while the Pacific is surrounded by younger volcanic systems with contributions from the depleted mantle. These seawater ϵ_{Nd} values are recorded in fossil fish debris, deep-sea coral skeletons, and ferromanganese nodules, and coatings on bulk sediment particles, or inside foraminiferal calcite tests. These archived time series data reveal the strengthening or relaxation of Great Ocean Conveyor Belt.

Besides the above-mentioned application of radionuclides in investigating slow oceanic processes or oceanic processes too difficult to measure directly, Broecker and Peng (1982) illustrated succinctly that many natural and anthropogenic radionuclides have been used in the ocean studies, such as rate of gas exchange at sea-air interface (e.g., ^{14}C , ^{222}Ra), particle formation, sinking, and accumulation at the sea floor (e.g., $^{234}\text{Th}/^{238}\text{U}$, $^{228}\text{Th}/^{228}\text{Ra}$, $^{230}\text{Th}/^{234}\text{U}$, $^{231}\text{Pa}/^{235}\text{U}$, $^{210}\text{Pb}/^{226}\text{Ra}$, $^{210}\text{Po}/^{210}\text{Pb}$, ^{14}C), the rate of vertical mixing of ocean (^{14}C), dating of sediment core layers and coral growth bands (^{14}C , excess ^{230}Th , ^{10}Be , ^{40}K , ^{40}Ar), efflux of hydrothermal fluid from underwater volcano (^3He), photosynthesis and respiration ($^{13}\text{C}/^{12}\text{C}$), diapycnal and isopycnal water mixing and vertical diffusivity coefficient (^{222}Rn , ^{228}Ra), and downwelling (deep water formation or ventilation of deep ocean) (^{14}C , ^{39}Ar). Radium radionuclides of ^{223}Ra , ^{224}Ra , ^{226}Ra , and ^{228}Ra are also proven to be useful to trace and quantify submarine groundwater discharge in coastal waters (e.g., Moore, 2006; Povinec et al., 2008c). More recently, a coordinated international program to systemically study the marine biogeochemical cycles of trace elements and their isotopes to cover the global oceans is now underway. This study, under the name of the GEOTRACES program, should expand our ability to decipher climate and environmental changes occurring in our time (Anderson, 2020). It is scheduled for completion by the end of this decade.

Developments in radioanalytical technologies

Earlier analytical methods in marine radioactivity involved collection of relatively large volumes of water (often more than 100 L, e.g., Fonselius and Östlund, 1959; Bowen et al., 1980) followed by preconcentration, radiochemical processing and measurements following radioactive decays using alpha, beta and gamma ray spectrometers, or using mass spectrometry for long-lived radionuclides (Povinec et al., 2008a, 2020; Levy et al., 2011). New developments in instrumentation, and technology in sample collection, preconcentration, and analysis have reduced the sample size as well as the time involved in processing and measurement of samples for anthropogenic radionuclides (Baskaran et al., 2009). All the developments in marine radioactivity studies have been possible due to new radioanalytical technologies, which have considerably improved the sensitivity of radionuclide analyses. Many marine radioactivity studies were not possible to carry out before, either because of lack of sensitivity or requirements of large volume samples. There is no space available to go into details. Therefore, we mention only a few developments and their applications (Povinec, 2018a,b).

The first break-through in radiometric techniques for analysis of environmental radionuclides at very low-levels (down to $\mu\text{Bq/kg}$ scale) has been the development of large volume HPGe detectors and their operation in underground laboratories (e.g., Laubenstein et al., 2004; Povinec et al., 2004, 2005, 2008a, 2013b, 2020; Aoyama et al., 2011a,b; Levy et al., 2011). Semiconductor alpha-ray spectrometry and liquid scintillation spectrometry remain powerful techniques for the analysis of alpha- and beta-emitters because of the simplicity of measurement, reasonable resolution and high sensitivity (Aoyama and Hirose, 2004; La Rosa et al., 2005; Lee et al., 2001; Levy et al., 2011). It is expected that the radiometrics sector will dominate also in future in the analysis of short- and medium-lived radionuclides (Povinec et al., 2020).

Specific radiometric developments included in situ gamma-ray spectrometers to be applied in mapping radionuclide distributions in seawater and sediments. Underwater gamma-ray spectrometry has been widely applied for studies of both spatial and temporal variations of gamma-ray emitters in the marine environment, sometimes working with satellite data transmission from the field to the laboratory, specifically when large sea surfaces should be scanned, or continuous monitoring should be carried out (Povinec et al., 1997, 1999, 2008b; Osvath et al., 2001, 2005), including the investigation of submarine discharges to the sea (Povinec et al., 2008a,c,d, 2020).

The most exciting breakthrough in the analysis of long-lived radionuclides in the marine environment has been made by the development of accelerator mass spectrometry (AMS), which offers the best sensitivity, requires minimum sample size, and minimizes matrix and interference effects (Povinec et al., 2008a, 2009). A direct coupling of accelerators with chromatographs has enabled researchers to proceed from a bulk sample analysis to a compound specific analysis, even to single atom counting (e.g., Povinec et al., 2009, 2020).

We list here at least two interesting applications: (1) new radioanalytical technologies have helped develop the analysis of a small volume of water samples; for example ^3H , ^{14}C , and ^{137}Cs (the first data sets with analysis of 10–20 L water samples) deep water mapping of the Pacific Ocean (Aoyama et al., 2011a,b; Povinec et al., 2011a; Kumamoto et al., 2011), and (2) the transport of north Pacific waters through the Equator (Aoyama et al., 2011a,b), which have been an important prerequisite for later studies of the FDNPP accident impacts. In the Indian Ocean, deep water mapping indicated the accumulation of ^3H and ^{14}C within the Subtropical gyre, confirming that the south Indian Ocean has been acting on a time scale of several decades as a final reservoir of contaminants transported from the northern Indian and Pacific Oceans (Povinec et al., 2011a,b). The ^{14}C water profile data clearly showed a penetration of bomb fallout ^{14}C down to almost 5000 m, documenting that the south Indian Ocean is an important sink for anthropogenic carbon. If we accept that the world ocean generally controls the climate on the Earth, then the Indian Ocean in this respect is playing a crucial role.

Certified reference materials for radionuclides in seawater, sediment, and marine biota has proved to be a must in quality assurance and quality control of radionuclide analyses (e.g. Povinec and Pham, 2001; Povinec et al., 2002, 2007; Pham et al., 2006a,b, 2008, 2010a,b, 2014, 2016). Similarly, a great impact on the marine radioactivity studies has been due to the development of marine radioactivity databases, e.g. HAM database (Aoyama and Hirose, 2004), and the IAEA's MARIS database (<https://maris.iaea.org/home>) (Povinec et al., 2006, 2020).

Conclusions

The ecosystem contributions of the global oceans have been increasing continuously by the aid of advancements of scientific knowledge. Oceans are becoming recognized as indispensable habitats for the entire human race. Advancements in nuclear science and technology have greatly increased our knowledge in understanding the evolution of the earth and environmental changes that have occurred in the ocean from the beginning. Hence, climate change is now and will be better understood than ever. Many questions related to the climate and marine environmental change are being addressed using radionuclides as tracers by employing new analytical methods and technologies. Scientific understanding on the underlying climate change and measurement capabilities will enable us to manage our ocean more robustly in a sustainable manner.

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See Also: What Is the Universe Made of?

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Environmental Protection: The Oceans—Implications of Manmade Radiation

GiHoon Hong^a and Pavel P. Povinec^{b, a} GH Institute, Seoul, South Korea; and ^b Faculty of Mathematics, Physics and Informatics, Comenius University, Bratislava, Slovakia

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Glossary

CTBT Comprehensive nuclear test ban treaty.
ENMOD Convention on the prohibition of military or any other hostile use of environmental modification techniques.
FDNPP Fukushima Daiichi Nuclear Power Plant.
IAEA International Atomic Energy Agency.
LC London Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter 1972.
NPP Nuclear Power Plant.
PBq Petabecquerels (10^{15} Bq = 10^{15} disintegrations per second).
UNECE United Nations Economic Commission for Europe.
UNSCEAR United Nations Scientific Committee on the Effects of Atomic Radiation.

Introduction

The term “natural capital” is now widely accepted as it provides a way of thinking about nature as a stock that provides a flow of benefit to people and the economy. The stocks of natural assets refer to the living and non-living components of ecosystems. Multiple forms of natural capital interact to generate goods and services (Guerry et al., 2015). Nearly 2.4 billion people (about 40% of the world’s population) live within 100 km of the coast (UN, 2017), and many coastal inhabitants draw their livelihood directly from the coast and coastal ocean. Coastal oceans consisting of tidal marsh, mangroves, estuaries, seagrass, macroalgal beds, coral reefs, and continental shelf, covering 32×10^6 km² (equivalent to approximately 9% of the ocean surface), provided 28 trillion US dollars’ worth of services to the human society (at 2007 \$US value) in 2011. The ecosystem service of the global oceans amounted to about 50 trillion US dollars per year (at 2007 \$US value) and accounts for about 40% of the global total value of ecosystem services in 2011 (Costanza et al., 2014). Marine transportation handles about 11 billion tons of cargo and provides more than 80% of global trade by volume (UNCTAD, 2020).

The US Ocean alone produces \$282 billion in goods and services annually according to the various government sources (NOAA, 2020). Without the oceans, human survival and their civilization may not have been sustainable. The ocean natural capital has increased almost twofolds over between 1997 and 2011 (Costanza et al., 2014), and will continue to increase as more scientific investigation reveals its value and as more innovative engineering projects working with nature allow us to use more ocean space and previously untapped ocean resources. For example, the effect of the damping impact of storm surges and strong winds by coastal wetlands recently was estimated to be $\$1.8 \times 10^6$ /km² per year in the United States, and property damage due to recent wetland losses at the time of Hurricane Irma was found to incur as much as \$430 million dollars (Sun and Carson, 2020).

The meaning of marine pollution covers more broadly than mere chemical pollution that is usually referred in the terrestrial environments. The International Environmental Treaty defines marine pollution as “the introduction, directly or indirectly, by human activity, of wastes or other matter into the sea which results or is likely to result in such deleterious effects as harm to living resources and marine ecosystems, hazards to human health, hindrance to marine activities, including fishing and other legitimate uses of the sea, impairment of quality for use of sea water and reduction of amenities” (Article 1.10, 1996 Protocol to the Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter, 1972).

Radioactive substance poses two kinds of threat to the marine environment: radiation and the chemical toxicity. Due to the legacy of the enormous damage having arisen from nuclear weapon blasts, regardless of the amount of radiation doses stemming from the released radioactive substances into the marine environment, any accidental release of radioactive substance into the marine environment will significantly curtail many legitimate uses of the sea. These include fishing activities, marine ecosystem service, recreational activities, and tourism industry by forcing people to avoid visiting oceans and consuming fisheries products, as we observed during the initial stage of the aftermath of Fukushima accident (e.g., Fukumo, 2020).

Radionuclides have been released to the environment from various sources, both planned and accidental. They include global nuclear weapons fallout (largest by far), the Chernobyl accident, the Fukushima accident, controlled releases from Sellafield and Cap de la Hague reprocessing plants, sea dumping of radioactive wastes, river runoff of previously deposited radionuclides on land surface and from the inland nuclear facilities, and loss or accidents involving nuclear materials (weapons, nuclear submarines, satellite fuels, and thermoelectric generators for lighthouse or ocean buoys; IAEA, 1999, 2001; Livingston and Povinec, 2000; Hong et al., 2004). Due to the global fallout through the stratosphere, the entire surface of the earth has been contaminated with radioactive debris in the early 1960s.

The IAEA has compiled global ocean anthropogenic radionuclides activity concentrations since 1961. As part of their effort, a coordinated research project yielded a final report on the worldwide marine radioactivity studies in 2005, which is the most recent one (IAEA, 2005; Povinec et al., 2005). According to their report, the total ^{137}Cs input from global fallout was estimated to be 311 PBq for the Pacific Ocean, 201 PBq for the Atlantic Ocean, 84 PBq for the Indian Ocean, and 7.4 PBq for the Arctic Ocean. For comparison, about 40 PBq of ^{137}Cs was released to the marine environment from the Sellafield and Cap de la Hague reprocessing plants. The Chernobyl accident contributed about 16 PBq of ^{137}Cs to the sea, mainly the Baltic and Black Seas, where the present average concentrations of ^{137}Cs in surface water were estimated to be about 60 and 25 Bq/m³, respectively, while the worldwide average concentration due to global fallout is about 2 Bq/m³. Therefore, global fallout has been considered as the largest input to the ocean (IAEA, 2005).

Testing of nuclear weapons fallout in the air or at the surface

According to the Comprehensive Nuclear Test Ban Treaty Organization (CTBTO), the rapid, uncontrolled reactions involved in testing of weapons in the air (1945–1980 by the United States, USSR, United Kingdom, France, China) released radioactive fallout all over the surface (Fig. 1). Globally, there have been at least 423 nuclear weapons tests with a substantial amount of radioactive fallout between 1945 and 1980 (Hamilton et al., 1996) and over 2000 nuclear explosions were reported from 1945 to 1996. Since 1996, nuclear weapon tests were also conducted from India, Pakistan in 1996, and the Democratic People’s Republic of Korea in 2006–17 (CTBTO, 2020). The explosion of nuclear weapons in the stratosphere and troposphere resulted in the release of radioactive debris in the air and these airborne radioactive materials were gradually deposited on the surface of the earth, including oceans, within a few years (Comar, 1963).

Although the solubility of the debris is sensitive to the composition of the explosion environment, air explosions produce small particles with high water solubility (UNSCEAR, 1962). The explosion of nuclear weapons at the surface generated radioactive debris that is subject to be transported by the prevailing winds, often divided into separate radioactive cloud where there is a steep wind speed gradient with altitude. In the stratosphere, zonal winds distribute radio-active debris in a latitudinal band of the stratosphere circling the earth within a few days to 6 weeks after the injection (Machta et al., 1956; Bleeker, 1961; UNSCEAR, 1962). The fear of environmental contamination as the result of nuclear fallout convinced the international community to adopt the Treaty Banning Nuclear Weapons Tests in the Atmosphere, in Outer Space and Underwater in 1963, followed by moratoria by the United Kingdom, the United States and the USSR, although the Chinese tested nuclear weapons from 1964 to 1980 in the air. Later the Comprehensive Test Ban Treaty (CTBT) was signed in 1996 (Bergkvist and Ferm, 2000). Therefore, the possibility of the global fallout from the nuclear weapon testing appears to be largely eliminated for now, although undeclared nuclear tests have been carried out by some countries, but mostly underground.

Unlike terrestrial environments, although the source term is still important, the mixing and transport of waterborne radionuclides in the world’s oceans and seas over time became significantly redistributed over the vast ocean areas, as well as sinking to the bottom and becoming entrained in the settling particulate matter and biological organisms. Therefore, the marine environment labelled with radionuclides differs from one region to another (e.g., Povinec et al., 2005; Fig. 2), and their activity concentrations continue to evolve with time—especially for the long-lived radionuclides relative to the time of water renewal.

The largest ocean, the Pacific, has been a major repository for these releases for at least three reasons: (1) partly because of its sheer size, (2) due to its proximity to releases such as the fallout in Bikini and Eniwetok Atolls in Marshall Islands, and now from the FDNPP accident, and (3) due to the nature of the end of the deep ocean conveyor belt that has about 600–1000 years of turnaround time (Livingston and Povinec, 2002). Therefore, the deep waters of the North Pacific Ocean will continue to

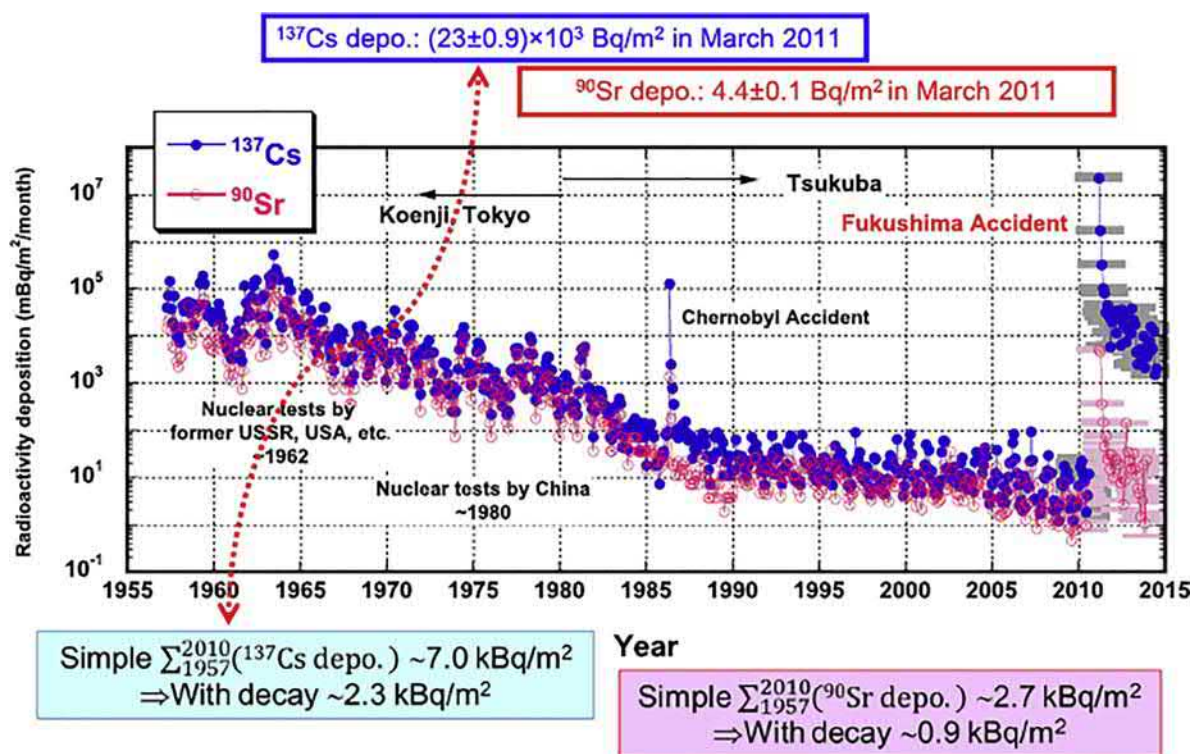


Fig. 1 ^{90}Sr and ^{137}Cs monthly deposition observed at the Meteorological Research Institute at Koenji Tokyo (35.7079° N, 139.6465° E, 1957–80) and at Tsukuba (36.0835° N, 140.0764° E, since 1981). Monthly deposition is expressed in millibecquerel per square meter on a logarithmic scale. The effects of atmospheric nuclear bomb tests have been recorded since 1957. Until the Partial Test Ban Treaty (PTBT) became effective in 1963, the United States, Soviet Union, and United Kingdom conducted atmospheric tests. France and China continued atmospheric testing until 1974 and 1980, respectively. The downward trend was interrupted by two excursion events, the Chernobyl accident in 1986 and Fukushima Daiichi Nuclear Power Plant (FDNPP) accident in 2011. The total amount of ^{90}Sr and ^{137}Cs deposition from 1957 to 2010 was compared to the FDNPP-derived deposition. Igarashi Y, Kajino M, Zaizen Y, Adachi K, and Mikami M (2015) Atmospheric radioactivity over Tsukuba, Japan: As summary of three years of observations after the FDNPP accident. *Progress in Earth and Planetary Science* 2: 44. <https://doi.org/10.1186/s40645-015-0066-1>.

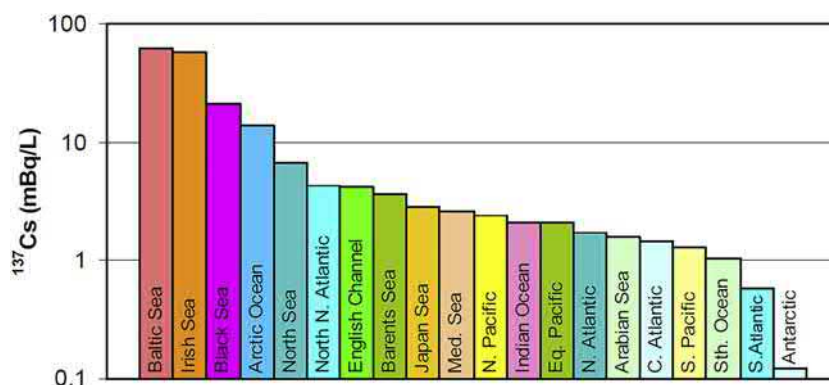


Fig. 2 Activity concentration of ^{137}Cs (adjusted to the year of 2000) in the world's oceans and seas. IAEA (2005) *Worldwide Marine Radioactivity Studies (WOMARS). Radionuclide Levels in Oceans and Seas*. IAEA-TECDOC-1429. Vienna: International Atomic Energy Agency, 197 p.

accumulate anthropogenic radionuclides over the centuries (Fig. 3). More than 80% of the total $^{239} + ^{240}\text{Pu}$ inventory originated from weapons testing are remains in the water column (Hamilton et al., 1996).

Nuclear power plants

Various forms of energy, such as coal, oil, natural gas, biomass, hydropower, wind, geothermal, and solar energies have been explored and exploited in the developmental course of human civilization. Nuclear energy was introduced for the first time as

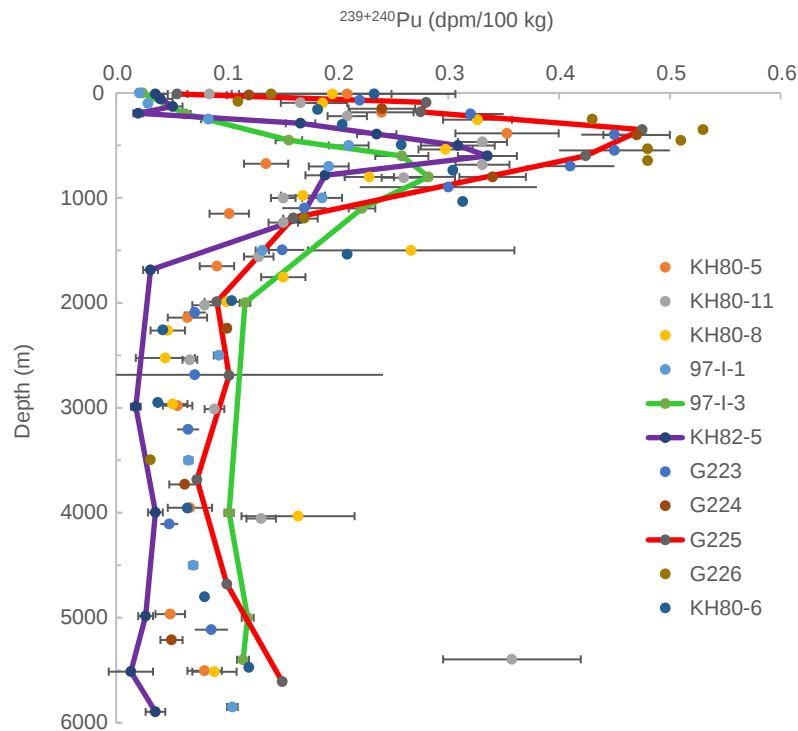


Fig. 3 A time series vertical profiles of the total $^{239}+^{240}\text{Pu}$ (dpm/100 kg) in the northwest Pacific Ocean from the year of 1973 to 1997. G223 (34.9667°N 151.8333°E), G224 (34.15°N, 141.9667°E), G225 (32.61667°N 161.9167°E), G226 (30.5667°N 170.6333°E) were sampled in 1973 (Ostlund et al., 1987), KH80-5 (40.0333°N 156.0333°E), KH80-6 (39.05°N 166°E), KH 80-8 (38.05°N 179.7167°E), KH80-11 (30.56667°N 170.6667°E) were sampled in 1980, KH 82-5 (25.0333°N 169.9833°E) was sampled in 1982 (Nagaya and Nakamura, 1984, 1984), and 97-1-1 (34.9983°N 145.9983°E) and 97-1-3 (30.5633°N 170.615°E) were sampled in 1997 (Povinec et al., 2003).

an option to generate electricity in 1948, when the X-10 Graphite Reactor was completed in Oak Ridge, Tennessee, the United States. A commercial NPP in Obninsk, Russia (formerly the Union of Soviet Socialist Republics (USSR)) began operation in 1954 to provide electricity from a nuclear source for the first time in human history. A total of 448 nuclear reactors were operating in 31 countries in 2016 according to International Atomic Energy Agency (IAEA, 2017). In addition, some 60 nuclear power reactors were under construction. The total net installed power in 2016 was 392,116 MWe according to IAEA. Nearly all the nuclear reactors are located near a large body of inland waters or the ocean, taking advantage of unlimited access to the heat sink of the reactor.

NPPs are mostly using uranium to produce electricity through the nuclear fission processes that produces thermal energy, which is then used to boil water and generate steam that in turn drives turbines connected to electromagnetic generators. Similar to the other thermal-electric generators, such as coal and oil burning power plants, less than 50% of the fission heat is converted to electricity, the extra heat needs to be disposed off in an ultimate heat sinks, such as a large body of water, e.g., rivers, lakes, and seas, or into the air. The sea is globally considered the most convenient ultimate heat sink to cool the condensate of the secondary cooling circuit in the coastal states. Therefore, large numbers of NPPs are located at the coast to take advantage of the sea as a heat sink. If not properly located or designed, those coastal reactors are subject to the risk of flooding, due to the increased frequency of severe weather or storm events and huge tsunamis caused by earthquakes that may cause accidents such as in FDNPP.

Contaminants emitted during the normal operation of nuclear power plants

Due to the scope of this encyclopedia, our discussion is largely confined to radioactivity. However, it should be noted that most industries and human activities serve the benefit of human civilization, and the unintended consequential damage to the human environment stems from those human activities that has been gradually curtailed over time largely thanks to both the scientific/technical engineering advances and adoption of innovative management practices. The biggest global pollution issue in our time is probably air pollution and human-caused climate change both due mostly to fossil fuel burning. Increase of greenhouse gas emissions, especially carbon dioxide, in recent decades is felt in the ocean in the form of acidification of the surface ocean where it contacts with the atmosphere (Hong and Povinec, 2021). The effects of nuclear power on human health (mortality) and the environment (climate) were recently quantified globally. Kharecha and Hansen (2013) used historical electricity production data and mortality and emission factors from the scientific literature to find that nuclear power prevented an average of over 1.8 million net deaths and an average of 64 Gt CO₂-equivalent net greenhouse gas emissions globally between 1971 and 2009. It is equivalent to

the past 35 years of CO₂ emissions from coal burning in the United States. Besides of CO₂, coal power plant emits particulate matter (PM₁₀ and PM_{2.5}), mercury, sulfur dioxide, nitrogen oxides, lead, carbon monoxide, volatile organics, arsenic, polonium and many other toxic chemicals to the air (e.g., Lin et al., 2019) as well as ²¹⁰Pb, ²¹⁰Po and other radionuclides, which may deliver locally higher radiation doses to the public than radionuclides released during normal operation of NPPs (e.g., Lauer et al., 2015; Mora et al., 2009).

Radioactive substances

Like other types of normally functioning industrial installations, including fossil fuel fired power plants, contaminants are released from NPPs during controlled nuclear fission operations. However, they are normally very small and do not represent a threat to humans and surrounding natural environment. Nonetheless, it should be acknowledged that NPPs do release some radioactive substances into air and sea. The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) reviews the sources and effects of ionizing radiation on a regular basis. UNSCEAR (2000) reported that radioactive noble gases, such as ¹³³Xe ($t_{1/2}$ = 5.3 days), and ¹³⁵Xe ($t_{1/2}$ = 9.2 h), ¹³¹I ($t_{1/2}$ = 8.02 days), tritium ($t_{1/2}$ = 12.3 years), ¹⁴C ($t_{1/2}$ = 5730 years), and other radionuclides are emitted to the atmosphere in small amounts. In addition, it is noted that the liquid effluents of the reactors during routine operation contain tritium, cobalt isotopes (⁵⁸Co, ⁶⁰Co), cesium isotopes (¹³⁴Cs, ¹³⁷Cs), ^{110m}Ag, ¹²⁴Sb, and other radionuclides (e.g., Kong et al., 2017).

Moreover, it is reported that many variations in types and quantities of radioactive materials emitted from different NPPs were detected, because of the great differences among the regulatory and operational measures of the different countries. The annual average effective dose to an individual due to the routine release of radiation from NPPs was estimated to be about 5 μ Sv/years, assuming that the total collective dose for each reactor type affects a single population group (UNSCEAR, 2000). Korean 24 NPPs were recently found to release about 0.4 PBq per year of total radioactivity into the environment in gaseous and liquid forms and their average effective dose to the members of public living near NPPs were estimated on the order of 1 μ Sv per year (Kong et al., 2017). This is orders of magnitude smaller than background radiation levels all human-beings are exposed to. For example, the total radioactivity of the naturally occurring gamma emitter ⁴⁰K in the global oceans of 15,000,000 PBq may be compared to that of 600 PBq, 16 PBq, and 15–18 PBq of ¹³⁷Cs from the atmospheric global nuclear weapons testing in 1950–1960s, Chernobyl in 1986, and Fukushima in 2011, respectively (Povinec et al., 2013a). ²³⁸U alone amounts to 37,000 PBq in the entire ocean (Buesseler, 2014). Therefore, the radiation impact from anthropogenic radionuclides in the world ocean on the human body is negligible because the maximum permissible annual limit for the public dose from exposure from external sources, including medical, educational, research and industrial facilities (IAEA, 2013) is usually set at most countries at 1 mSv/years, while the average natural global background dose is about 6.2 mSv/years (Health Physics Society, 2020). Fig. 4 shows the total quantities of radionuclides released from the East Asian NPPs.

Thermal effluent

Power plants powered by burning fossil fuel (coal and gas) or nuclear fission traditionally have required huge volumes of water to condense steam from the turbine exhaust (Pan et al., 2018). Generally, an NPP requires about 189 m³ of seawater cooling to produce 1 MWe (calculated using data provided by Korea Electric Power Corporation (<https://www.data.go.kr/data/15039187/fileData.do>), similar values are found in IAEA (2012)). Coal fired power plants require cooling water, but the volume required is slightly smaller (~10% less) because they usually have a higher thermal-to-electrical energy conversion efficiency. Advanced

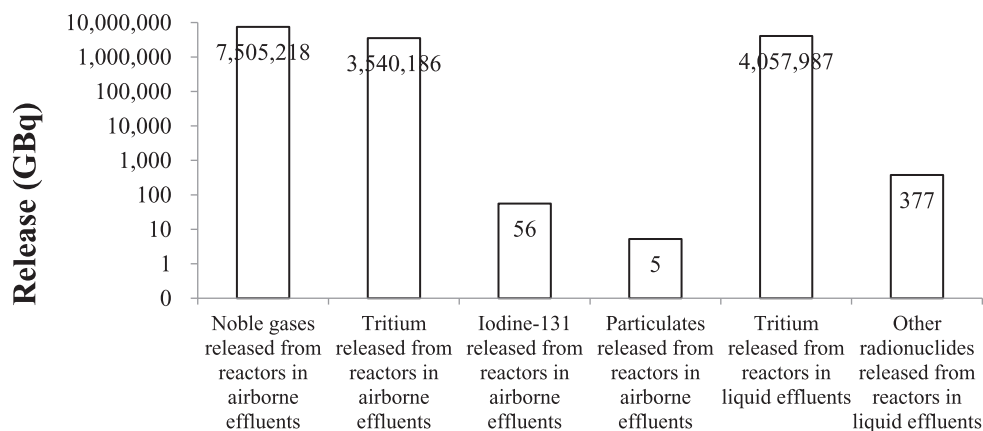


Fig. 4 Total amount of radionuclides released annually from operating NPPs in China, Japan, Republic of Korea, and Taiwan. Data from UNSCEAR (2000) *United Nations Scientific Committee on the Effects of Atomic Radiation: Sources, Effects and Risks of Ionizing Radiation UNSCEAR 2000 Report*. NY: United Nations, 659 p.

NPP are expected to have energy conversion efficiency comparable to, or larger than the efficiency of coal-fired power plants (Pan et al., 2018; Stanculescu, 2021).

The volume of seawater required varies for each reactor, because the ambient seawater temperature differs in each NPP location. For example, the Yeong Kwang NPP, which houses six reactors, requires about 7.5×10^9 m³/years, which is equivalent to the amount of water in the third largest river in the Republic of Korea (Kum River, 6×10^9 m³/years) (Hong et al., 2002). The condenser cooling water, which carries the residual heat, is discharged into the sea at a temperature higher than the ambient water (6–11 °C higher in case of a low temperature condenser and 14–17 °C higher in case of a high temperature condenser) and it affects ocean temperature in the immediate locale. The amount of seawater circulated through condensers of all the NPPs operating in China, Japan, Korea, and Taiwan is about 567×10^9 m³/years, which is equivalent to that of the Mekong River (448×10^9 m³/years), about 2/3 of the Changing River (896×10^9 m³/years), and about 20 times of the discharge of the Han River, the largest river in the Republic of Korea.

Therefore, seawater circulation through condensers may thermally modify the marine ecosystems significantly in the vicinity of the points of seawater intake and discharge. These thermal effluents have been affecting the surrounding marine ecosystems (e.g., Huang et al., 2019), cold water species in particular, because they raise the water temperature by more than about 0.5 °C above the ambient water temperature in an area of about 12–30 km² surrounding the discharge pipe outlet (Maeng et al., 2013). In addition, the abrupt halt of discharge of thermal effluent due to routine plant shutdown causes cold shocks to the thermally elevated ecosystem near the outfall and forces it to readjust to the new settings. For example, NPP operators in the Republic of Korea have paid about 21 million US\$ annually for the last 5 years (2015–19) to the local fisheries interest groups in response to the alleged thermal damages on their cold species culture ground adjacent to NPP (Today Energy, 2019). This is partly due to the peculiar legal settings surrounding some kinds of territoriality fishing rights of local residents and other legitimate users of the sea in the Republic of Korea.

In a broader context, NPPs, fossil fuel powered power plants, and desalination plants have discharged thermal effluent into the sea, which significantly modifies temperature and salinity in the immediate receiving body of water (e.g., Kim and Jeong, 2013).

Other chemicals

Like other seawater intake pipes in other industries, including fossil fuel powered thermoelectric generators, the inner wall of the condenser of NPPs is subject to scale formation, colonization of marine organisms, and corrosion. These phenomena compromise the designed cooling efficiency. Therefore, they must be strictly controlled. Usually, scales, corrosion, and fouling are prevented using acidic polyphosphate and chromate solutions, amine or organosilicon compounds, and sodium hypochlorite, respectively. Foam forms in the downstream of the outlet of the condenser effluent because air is entrained at the weir, and cascades build up at the outlet structure and become persistent around the outlets. As a result of the growing public concern about the buoyant foam, various chemical de-foaming agents, such as polydimethylsiloxane (Stevens et al., 2001; Craig and Caunter, 1990; GESAMP, 1986) or other mechanical options have been used (Oh et al., 2012). The residues of these chemicals are discharged into the sea along with the thermal effluents. However, their impact on the marine ecosystem and humans is usually negligible.

Impact of the cooling water intake operation on the ecosystem

For both fossil fuel and nuclear thermoelectric generators, drawing a large volume of water at a high-speed leads to the simultaneous intake of large amounts of debris from the sea. Therefore, bar screens, double through traveling screens, drum screens, and debris filters are placed at the mouth of the intake. However, fish and other animals collide in the wire screen at the mouth of the intake tube and eventually die. The total amount of fish killed at NPPs along the coast of the Republic of Korea ranges from 3.4 to 30 t/years (Yi and Lee, 2016). Similarly, 258 t of fish kill were reported in the 2400 MW Longannet Power Station in January 1999 to December 2000 in the United Kingdom (Greenwood, 2008). In addition, small fish, eggs, and larvae of other marine organisms may pass through the filter screens and enter the cooling system, which threatens their mortality rates. Zooplanktons, such as euphausiid and scyphozoan medusa, often approach the screens in large volumes, which results in the clogging of the intake screens forcing NPPs to temporarily halt operations. The maximum tolerable densities of jelly fish and euphausiid were estimated to be about 2 and 800 individuals/m³ seawater, respectively (Lee et al., 2006).

Contamination of oceans due to nuclear reactor accidents

Waste products from nuclear reactors arise in two ways; (1) from the slow controlled reactions in nuclear power plant operations, including the production of reactor fuels and the burning of fuels in power reactors (1954–present) on land and sea going vessel (submarines, 1958 to the present); (2) from accidental releases to the environment from failed nuclear reactors (e.g., Chernobyl in 1986 and FDNPP in 2011), and (3) from the emission of technologically enhanced naturally occurring radioactive material (TENORM) from the non-nuclear industries (e.g., coal fired thermal powered plants, Pandit et al., 2011).

In the hydrogen explosions during the FDNPP accident, little dissipation occurred during this time due to the nature of the rapid global air circulation system. The Fukushima radioactive plume contaminated the entire Northern Hemisphere for a relatively short period of time (<21 days) when the westerlies circulated the earth with an average zonal velocity of >60 km/h, as shown in Fig. 5

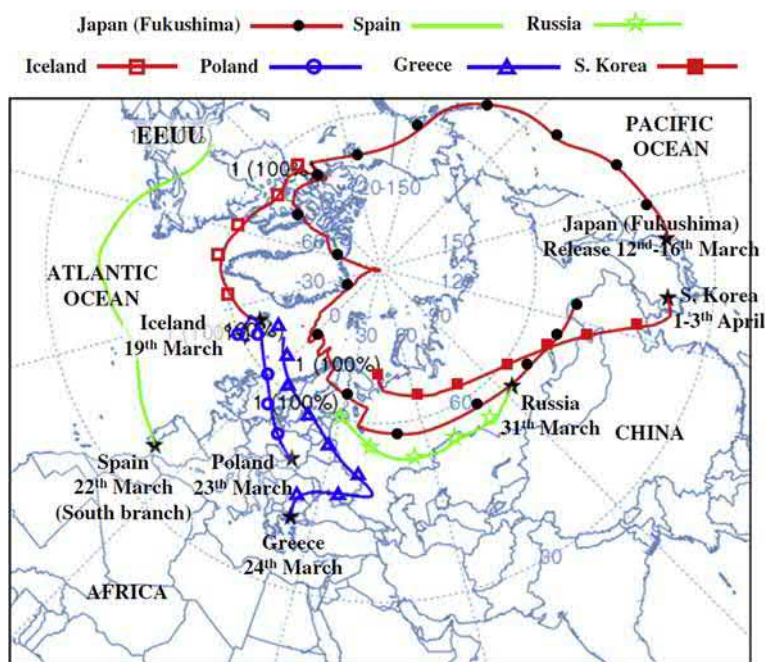


Fig. 5 The centroid of the 8 days backward trajectory analysis at different sites around the world indicating the transport of the Fukushima plume and the centroid of the 19 days forward trajectories from Fukushima from 12 to 16 March. The number 1 is the identification number of the centroid and the percentage (100%) indicates the number of hourly trajectories occurring in the cluster. Hernandez-Ceballos MA, Hong GH, Lozano RL, et al. (2012) Tracking the complete revolution of surface westerlines over Northern Hemisphere using radionuclides emitted from Fukushima. *Science of the Total Environment* 438: 80–85.

(Hernandez-Ceballos et al., 2012; Povinec et al., 2013a,b). Earlier in 1986, ^{137}Cs emitted from the Chernobyl accident also affected its levels in the Sea of Japan (East Sea) in the opposite side of the globe almost immediately (Fig. 6).

The radionuclides discharged into the sea or falling on the surface of the ocean near the FDNPP have been subject to spread elsewhere in the Pacific Ocean by the aid of prevailing water currents, entering e.g. the Bering Sea and the Arctic Ocean (Povinec et al., 2013a,c; Huang et al., 2020; Tsumune et al., 2020). It is interesting to note that the declining trend of ^{137}Cs activity concentration in the Sea of Japan (East Sea) was reversed in the time period from 2011 to 2016 (Fig. 6). Hirose and Povinec (2019) confirmed the intrusion of Fukushima contaminated seawater through Korea/Tsushima Straits from the East China Sea using decoupling of ^{137}Cs and ^{90}Sr in seawater after 2011 (Fig. 7). The activity ratio of $^{90}\text{Sr}/^{137}\text{Cs}$ is extremely low (as much as 0.01–0.03) (Casacuberta et al., 2013; Povinec et al., 2013b), but the water originating off Fukushima of Japan carries the Fukushima $^{90}\text{Sr}/^{137}\text{Cs}$ signal. The Subtropical Mode Water off Fukushima flows westward as a form of undercurrent and entrained in the Tsushima Current to enter the Sea of Japan (East Sea) via shallow strait (~130 m deep). A part of ^{137}Cs introduced in the surface water is incorporated into the living and non-living suspended particulate matter and sinks to the bottom water. Owing to the relatively long residence time of the deep water (~1000 years), ^{137}Cs will accumulate in the deep water of the Sea of Japan (East Sea) similar to that of Pu isotopes (Hong et al., 2011).

Nuclear reactor failures and international governance

Accidents can happen in NPPs like they happen in any other large industrial installations. An “accident” can be defined as an event that happens unexpectedly and unintentionally, and typically results in damage or injury. However, the aftermath of an accident that involves a radioactive substance is potentially long-lasting, widespread, and can cause severe effects, such as destruction, damage, or injury to another state party. The Convention on the Prohibition of Military or Any Other Hostile Use of Environmental Modification Techniques 1976 (ENMOD) defines the following expressions: “widespread” refers to an area encompassing several hundreds of square kilometers, “long-lasting” refers to a period of a few months, and “severe” refers to the state involving serious or significant disruption or harm to human life, natural and economic resources, or other assets.

Radionuclides can exist in gaseous, liquid, or solid forms. In addition, they may spread into the air or water by prevalent wind and rain cycles. More importantly, they are invisible in most cases. Moreover, the radioactive decays of some radionuclides can take place over several, hundreds, or even millions of years and may extend much beyond the human lifetime. Therefore, they may modify the surrounding environment, regardless of their immediate adverse effects, and, thus, create additional concern because of the lack of scientific knowledge on their fate and behavior in the environment and natural ecosystems. Furthermore, the duration of exposure to radioactivity is more seriously considered by regulatory organizations—compared to that of non-radioactive

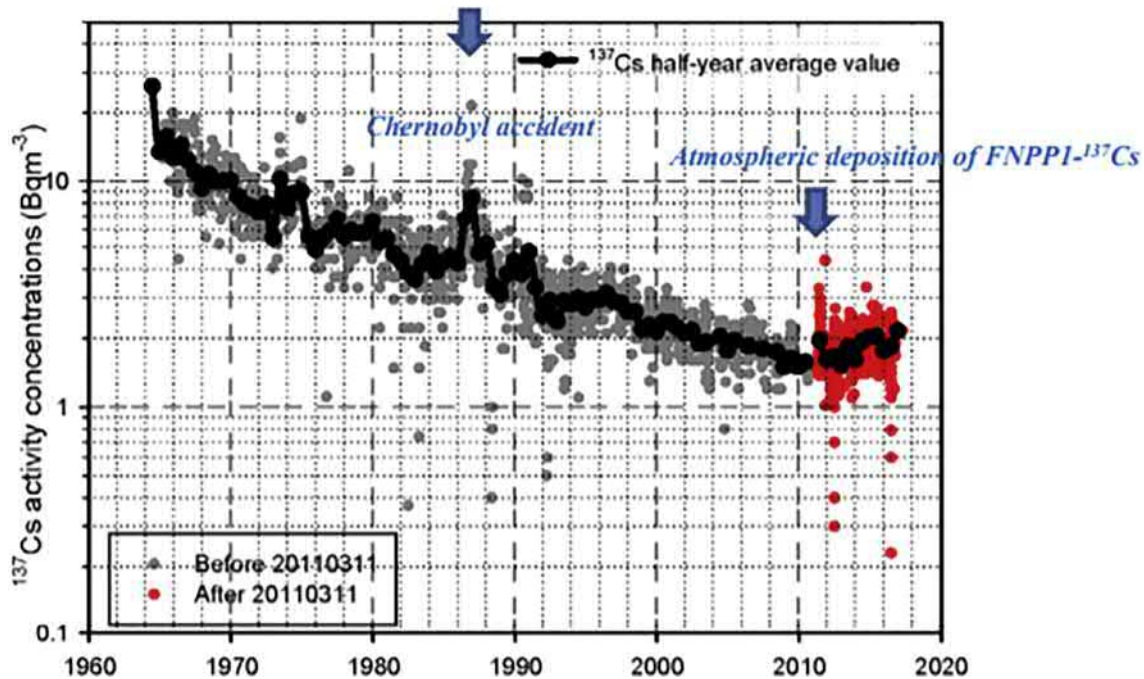


Fig. 6 Temporal variations in the ^{137}Cs activity concentrations in surface seawater in the Sea of Japan (East Sea) during the period from 1966 to 2016. Circles indicate measured values, and red circles indicate data measured after the Fukushima accident (March 11, 2011). Black circles indicate the 0.5-year average value. Standard deviations of data were removed to clearly show the temporal variation. Inomata Y, Aoyama M, Hamajima Y, and Yamada M (2018) Transport of FNPP1-derived radiocaesium from subtropical water in the western North Pacific Ocean to the Sea of Japan. *Ocean Science* 14: 813–826.

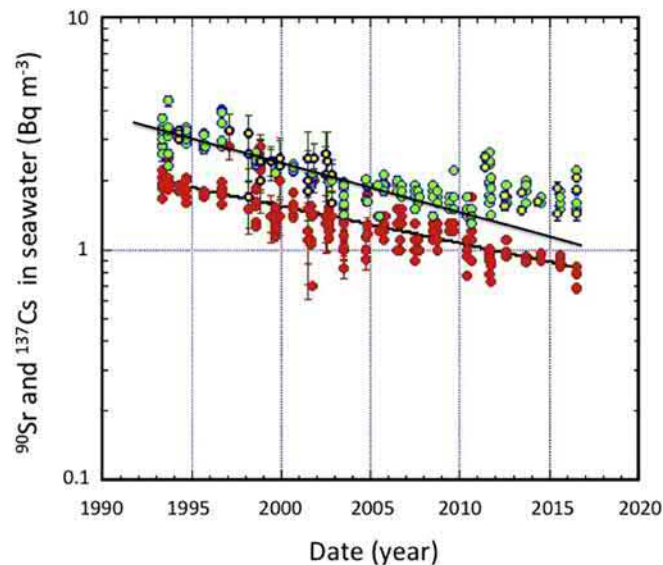


Fig. 7 Temporal variations of ^{90}Sr and ^{137}Cs in surface waters of the Sea of Japan (East Sea). Red dots show ^{90}Sr activity concentrations, and green dots show ^{137}Cs activity concentrations. The solid lines represent exponential regression curves. Hirose K, and Povinec PP (2019) ^{137}Cs and ^{90}Sr in surface waters of the Sea of Japan: Variations and the Fukushima Dai-ichi Nuclear Power Plant accident impact. *Marine Pollution Bulletin* 146: 645–652, with permission.

contaminants. The effluent standards of other contaminants, e.g. cadmium, is usually given in the form of concentration, although the health risk evaluation of these substances usually specifies the duration of exposure similar to that of the radioactive substances.

In 1956, 3 years after the first commercial nuclear reactor produced electricity, the cooling system of one of the waste storage tanks of the Mayak NPP and reprocessing facility, located in the Chelyabinsk region, failed, resulting in an explosion and contaminating a wide area (approximately 1000 km²). This accident was called the Kyshtym disaster and was ranked at level 6 on the

International Nuclear Events Scale (INES) (Thomas, 2012; Buttinger, 2017, 2017). In the year 1957, a fire broke out at the Windscale reprocessing plant in the United Kingdom with the emission of large quantities of ^3H , ^{131}I , ^{210}Po and other radionuclides into the air (e.g., Wakeford, 2007) and ^{137}Cs into the ocean along with other fission products (e.g., Livingston and Bowen, 1977; Povinec and Povinec, 2000).

In the same year (1957), the International Atomic Energy Agency was established. Since then, incidents and accidents involving nuclear material in nuclear power plants have occurred in more than 90 cases around the world (Rose and Sweeting, 2016). The nuclear reactor accidents that contaminated at least a half of the surface of the earth, including the ocean, were Chernobyl (Ukraine in 1986) and Fukushima (Japan in 2011). They occurred due to different causes, although human errors played a significant role in these accidents as well. However, a reactor of the Three Mile Island failed at level 5 on the INES in 1979 did not emit any noticeable amount of radioactive substance to the outside environment (Wheatley et al., 2017). In 1956 there was also a large-scale accident at the Windscale (UK) nuclear reprocessing facility.

Therefore, it is still difficult to generalize the cause of the accidents or the probability of nuclear reactor failures that discharge radioactive substances to the natural environment. But it is instructive to realize that the number of the major nuclear reactor accidents that contaminated marine environment is only two (Chernobyl and Fukushima) out of about 448 (0.45%) nuclear reactors in the world (see also Sich, 2021; and Mizokami and Rempe, 2021). That may be compared to that of the other accidents of manufacturing industries, for example, in S Korea (0.59% in 2012, Yi and Lee, 2016), though a different matrix was used. Also, in evaluating the total environmental impact of various power production sources, one must take into account the vast reduction in CO_2 pollution emissions provided by nuclear power in comparison to fossil fuel plants. Furthermore, it is important to note (i) the number of years the NPPs have operated or the total amount of energy they have generated should be factored in, and (ii) future advanced NPP are expected to be inherently safe, such that the probability that they will have accidents that contaminate the oceans will be much lower.

Unit 4 of the Chernobyl NPP was blown up at 01:23 on April 26, 1986 and caused the emission of a large amount of radioactive substances, which spread over the entire globe, especially in Northern Europe. However, Moscow did not announce the accident until April 28 at 21:02. The health, environmental, and socioeconomic impacts of the Chernobyl accident are still being assessed, and remedial measures are still being conducted in several countries. In addition, some affected areas in Ukraine are still abandoned (OECD, 2006; Beresford et al., 2016). The Chernobyl nuclear accident prompted various global countermeasures against nuclear accidents to clarify the duties of the accident states and create an international assistance mechanism.

The IAEA General Conference conducted the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency. Both were opened for signature on September 26, 1986 (5 months after the accident) and became effective on October 27, 1986 and February 16, 1987, respectively. Subsequently, Europe further developed collaboration mechanisms in a transboundary context, such as the Convention on Environmental Impact Assessment in a Transboundary Context 1991; the Convention on the Transboundary Effects of Industrial Accidents on March 17, 1992; the Convention on Access to Information, Public Participation in Decision-making and Access to Justice in Environmental Matters in 1998. Other activities were administered by the United Nations Economic Commission for Europe (UNECE) for industrial installations, e.g., NPPs and offshore oil and gas platforms that might have adverse environmental impacts across borders.

In 2011, the FDNPP accident (INES 7) occurred, resulting in the emission of a considerable amount of radioactive substances into the entire northern hemisphere. A magnitude 9.0 earthquake occurred at 14:46 on March 11, 2011 in the Pacific Ocean side of Northern Honshu, Japan, followed by a tsunami that struck the east coast of the Tohoku region. Three of the six BWRs operating at the FDNPP automatically shut down. However, a meltdown of the nuclear fuel reactor occurred shortly after the shutdown because the cooling system of the reactors failed due to the failure of the submerged diesel generators located in the flooded basement. Radionuclides were detected at the main gate of the plant at 12:00 on March 12, and a substantial number of radionuclides were released to the atmosphere after hydrogen accumulated within the reactor, resulting in the explosion of Unit 3 on March 14, 2011 at 11:00. The large magnitude of the atmospheric release continued until March 17 (Momoshima et al., 2012). In addition, about 24 days later, on April 4, nearly 10,000 m^3 of radioactive substances laden water was discharged to the Pacific Ocean. The Japanese government imposed a voluntary ban on fishing activities on March 15 (4 days after the nuclear disaster). Subsequently, China, Hong Kong, Taiwan, Singapore, the Russian Federation, the United States, and Australia have banned the imports of certain Japanese food items, including fishery products that originated in or near Fukushima.

The FDNPP accident revealed a number of important aspects of nuclear accidents in coastal areas:

- (i) Radioactive substances first released into the atmosphere had spread into the entire Northern Hemisphere within a few weeks (Hernandez-Ceballos et al., 2012; Povinec et al., 2013a) and had immediately arrived to the ocean surface (Buesseler et al., 2012; Povinec et al., 2013a,b).
- (ii) In the short period following the accident, all fisheries activities within a 20 km radius from the plant were suspended by the government.
- (iii) Seawater was used to cool the damaged reactors, which resulted in the discharge of water highly contaminated with radioactive materials into the sea.
- (iv) Sea bottom sediments adjacent to the plant were rapidly contaminated, which immediately led to the contamination of the overlying seawater.
- (v) Contaminated marine bottom sediments acted as a source of radioactive substances, to which the benthic fauna was exposed, i.e., longer retaining the radioactive substance in the bottom dwelling fish (Wang et al., 2018). Though none of the

- bottom-dwelling fish caught nearby exceed Japan's strict limit of radiocesium ($^{134}\text{Cs} + ^{137}\text{Cs}$) of 100 Bq kg^{-1} ww (wet weight) after 2015 (for surface-dwelling fish already after 2011), there has been public concern about the safety of the marine food.
- (vi) There were still more than 1000 massive water tanks on the premises containing waste water with many radiologically important radionuclides of relatively high concentrations (^3H , ^{14}C , ^{99}Tc , ^{125}Sb , ^{60}Co , ^{106}Ru , ^{137}Cs , ^{134}Cs , ^{90}Sr , ^{129}I) (Buessler, 2020), which were subjected to radio-purification, or eventually to be released into the sea (when containing only tritium). It is planned that about 1 Mt. of radioactive water containing tritium will be released to the sea for 30 years. A land-side impermeable wall (the so called "ice wall"), about 30 m deep and 1.5 km long, was constructed to prevent possible contamination of the groundwater. However, the stability of this wall has been recently questioned (Reuters, 2018).

The aftermath of the Fukushima accident was extraordinary, because all the details of the accident scene had been continuously broadcasted live from the start, and all the measures and the inter-governmental consultations were known to the general public before they were realized by the domestic and international authorities. The large uncertainties associated with the aftermath management, such as the amount and fate of the radioactive substances released from the accident beyond the country of origin (as well as the radioactivity records observed throughout the northern hemisphere) triggered many doubts and fear in ordinary people especially those living in neighboring countries. On the other hand, it should be stressed that there has not been reported any deaths which could be associated with radiation from the Fukushima accident.

During the last 25 years, the world has significantly changed, particularly after the Internet and World Wide Web became available to the public. Thereafter, the public gained access to an unprecedented amount of data and information previously owned by a few elite groups, such as state governments. The ocean and its resources are increasingly and simultaneously utilized by various international stakeholders. For example, the share of exports in fish production amounted to 37.6% in 2018 (FAO, 2020). Consequently, ordinary citizens who have access to up-to-date scientific data view marine space more like Transboundary Rivers. They regard it as shared transboundary waters similar to that of the Kiev Protocol on Civil Liability, the Compensation for Damage Caused by the Transboundary Effects of Industrial Accidents on Transboundary Waters, the 1992 Convention on the Protection and Use of Transboundary Watercourses and International Lakes, and the 1992 Convention on the Transboundary Effects of Industrial Accidents—all allowing individuals affected by the transboundary impact of industrial accidents on international watercourses (e.g., fishermen) to legally claim an adequate and prompt compensation. Pacific bluefin tuna flesh caught off California in August 2011, about 6 months after the March 2011 Fukushima accident, were found to have ^{134}Cs ($4.0 \pm 1.4 \text{ Bq kg}^{-1}$) and ^{137}Cs ($6.3 \pm 1.5 \text{ Bq kg}^{-1}$) above the pre-Fukushima accident ^{137}Cs ($\sim 1 \text{ Bq kg}^{-1}$) level (Madigan et al., 2012).

However, these increases are statistically insignificant increases above the global fallout background—which illustrates the misunderstanding of the public on radionuclide levels in the environment and specifically in food. The effective dose commitment estimated from consumption of such "radiocontaminated" tuna fish would be about $0.1 \mu\text{Sv/years}$, i.e. 10,000-times below the maximum permissible annual dose to the public from external sources (Povinec et al., 2013a; Povinec and Hirose, 2015).

The Japanese government imposed a strict limit of 100 Bq kg^{-1} ww (wet weight) on seafood (5-times lower than during the pre-Fukushima era, and 10-times lower than usually accepted international value). The total effective dose commitment estimated from the ingestion of ^{134}Cs and ^{137}Cs in fish, shellfish, and seaweed due to the FDNPP accident is 0.7 mSv/years (Povinec et al., 2013a). This value is still well below the maximum permissible annual dose to the public from external sources (1 mSv/years), and below the world average dose from natural radiation sources (2.4 mSv/years). The individual effective dose commitment from consumption of ^{134}Cs and ^{137}Cs in fish caught in the open north-western Pacific Ocean in 2012 was estimated to be about $2 \mu\text{Sv/years}$, i.e. 350-times lower than the dose from the consumption of natural ^{210}Po in fish and shellfish.

Ownership of the aftermath of marine environmental contamination in areas beyond the national jurisdiction

After the 2011 Great East Japan Earthquake, the media started to report real-time news of its development and its impact on the FDNPP accident and the subsequent release of radioactive substance into the air. Countries worldwide were notified of the radioactive substances released into the environment through the news media instead of the official diplomatic channels. Hence, they carried out extensive monitoring of the Fukushima originated nuclear substances according to their individual social needs, which were based on their individual economic and technological capabilities. Globally, the monitoring activities of the marine environment, such as sending ships to collect seawater, biota, and sediments—as well as carrying out subsequent laboratory analyses and a number of national and international meetings—had a significant cost. Therefore, recently mostly international cruises have been organized.

As a result of these efforts, radioactive substances originating from the Fukushima accident were found to have reached the west of Japan as well as passing the Pacific Ocean, America continent, Atlantic Ocean, Europe continent and going back to Japan (Povinec et al., 2013a). There have been several national and international sampling expeditions to collect samples of seawater and biota and subsequent analysis of key anthropogenic radionuclides in probably more than a few hundred laboratories around the globe. The sampling and further analyses are still being carried out and reported to the scientific communities and to the public. Handling and controlling the aftermath of the accident becomes an international concern, which may be regarded as a manifestation of the sense of the collective ownership of the aftermath, although each state carried out their monitoring activity for the protection of their state sovereignty. Therefore, there is good reason to establish a consultative committee of experts similar to that of the ENMOD to analyze the findings and provide expert views relevant to the problems encountered.

Precautionary principle versus risk-based approach

Contaminants in the environment are usually evaluated in terms of amounts, e.g., (i) a substance could be present in trace or negligible amounts; (ii) a tendency to be shortly converted to a harmless substance; (iii) safety factors; and (iv) safe or acceptable levels, which are often expressed in numeric figures. However, although these criteria appear to be scientific, they are almost impossible to be integrated in scientific methods, because uncertainty is a trait of science and scientific investigations. For example, the mean acute toxicity values (or LC_{50}) of marine organism species vary by five orders of magnitude, in which the highest concentration, for example of cadmium (135 mg Cd L^{-1}) is that of the oligochaete worm, and the lowest one ($41.3 \text{ } \mu\text{g Cd L}^{-1}$) is that of the mysid (Wang et al., 2010). Although these data do not permit the release of contaminants in concentrations higher than the LC_{50} value of a target species into the marine environment, some operators may use some of the higher limits as their operating criteria. These variations in values paralyze the decision on which value is the most suitable for the protection of the marine environment.

Body burden data is often established through toxicity testing, and then the relevant thresholds are decided based on the relation between them and their accompanying toxicity. Cd concentrations in the surface of relatively pristine seawater, which are commonly below 10 ng L^{-1} , are considered as a reference. In addition, the uncertainty increases if the health of organisms including humans is considered. Another example of the uncertainty is that of “black spot,” which is a necrotic shell disease of Crab (*Cancer pagurus*) that lives in British coastal waters. Many studies have investigated this disease, but they have generated very different explanations for its causes. For example, one scientific paper claimed that the cause of the disease is the high frequency of net injuries, while another paper in another area declared the cause to be aging, because of the low crab fishing activities in the area (Dethlefsen, 1986). Both of these arguments were used to advocate or reject sewage sludge dumping into the sea, respectively.

Therefore, in order to avoid the “paralysis of uncertainty” (McIntyre and Mosedale, 1997), the current environmental management practice is to adopt a no emission principle and waste prevention policy, i.e., the precautionary approach was chosen to be the administrative measure to avoid the inaccuracy of scientific predictions of the threats to the environment and human health and to seek less harmful alternative processes and available products (Santillo et al., 1998). Environmental regulations, such as wastewater discharge limits, continue to be based on numeric standards in various parts of the world, with the recognition that individual governments will adopt air, water, soil, or food quality standards to complement their individual end of pipeline discharge limit standards.

Unfortunately, the nuclear industry has yet to adopt the best practice to limit its impacts on the marine environment. The international law of nuclear energy originated as regulations for nuclear weapons, not as a part of environmental law for protection of the environment. Moreover, the International Atomic Energy Agency (IAEA) aims to promote atomic energy for peaceful use, and the protection of the environment is a byproduct of this effort. Therefore, it considers the subject primarily for an industrial activity rather than from an environmental point of view (e.g., article 2 of the IAEA Statute). In addition, the international community has been dealing with radioactive substances in a different manner, compared to the other substances. The Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter 1972 (LC) included radioactive wastes and other radioactive matter along with other contaminants, such as e.g. cadmium and cadmium compounds. However, it handed over the issues of the radioactive material to the IAEA because it is the competent authority to handle this subject among intergovernmental agencies (Sjoblom and Linsley, 1994; IAEA, 2015).

Before the LC in 1972, radioactive waste disposal at sea had been conducted by multiple countries ever since 1946 (IAEA, 1999) and the IAEA had already provided safety standards for the radioactive waste disposal into the sea in 1961 (IAEA, 1961). It gave numerical values for the concentration of the most hazardous radionuclides, e.g., ^{90}Sr in the surface layers of the sea (upper 1000 m) that should not exceed $10^{-9} \text{ } \mu\text{C/mL}$ ($37 \text{ Bq mL}^{-1} = 37 \text{ kBq L}^{-1}$ seawater). These earlier guidelines are no longer validated by the agency. Upon the request of contracting parties to the LC, IAEA submitted the provisional definition, and provided recommendations on dumping of radioactive wastes and other radioactive matters. It revised those definitions in 1978 and then again in 1986 (IMO, 1986). The value of high-level wastes has been lowered, and the condition of dumping sites has been modified to be more restrictive. Subsequently, the international consultancy requirement prior to the dumping was added in the 1986 version.

However, in 1994, the Contracting Parties to the LC suspended the use of the IAEA definitions, recommendations, and guidance, because of their decision to ban radioactive waste disposal at sea. In addition, the IAEA always stressed that, in the light of future technological developments and increased scientific knowledge, the guidance will continue to be revised when appropriate, which reflects the uncertain nature surrounding the use of numerical values as the sole decision criteria. For example, if the threshold of cadmium emission from a plant to the ambient air is 5 ppb, the operator may be immune from the prosecution by keeping the plant emission under 5 ppb. However, the plant operator should be held responsible for the harm caused to animals or plants living adjacent to the outfall as a result of the emission, even if the discharge standard is retained. This is a well-established legal norm in the domestic court, but it still has not been well founded in the general international law.

In summary, it is unfortunate that this continuing evolvement of often conflicting international approaches for prevention of potential ocean contamination remains cloudy. As such, the precautionary principle currently providing the overarching guidance suggests a goal of NO discharge of radioactive substances into the ocean. Whereas this sounds admirable, it may not be currently practical for all industries emitting contaminants into the sea. For example, municipal waste treatment plants discharge contaminants into the public waters according to the emission standards set by public authority, though some contaminants in treated wastewater remaining or not screened by the standards pollute the receiving water bodies (e.g., Rogowska et al., 2020; Freeman et al., 2020). It is fair to note that nuclear power plants operators, like any other industries, endeavor to evaluate feasible waste reduction and prevention techniques including process modifications to remove the hazardous constituents and on-site

closed-loop recycling. Common sense also recognizes the enormous degree of dilution provided by the incredibly vast amount of ocean water and naturally occurring radioisotopes are continuously getting into the oceans. Consequently, considerably more science and innovative engineering solutions (e.g., advanced fission reactors) needs to be included in future efforts to minimize/prevent pollution from all sources.

Application of the distribution of anthropogenic radionuclides present at sea for the environmental protection

Applications of anthropogenic radionuclides as tracers to investigate oceanic processes were extensively reviewed by Broecker and Peng (1982), Jenkins (2003), Povinec et al. (2011), and Hong et al. (2011). Since the ocean input history of anthropogenic radionuclides is relatively well known, many were utilized to trace their carrier phases in the sea as time markers. Several examples are listed here.

One of the most significant applications of anthropogenic radionuclides to climate studies up to the present is the utilization of the bomb tritium distribution in the North Atlantic. With its presence in intermediate and deep waters, it directly confirmed that the deep-water forms in the North Atlantic as was postulated by the box model of global ocean conveyor-belt circulation. The distribution of bomb-produced ^{14}C in the ocean has been instrumental in validating the global ocean carbon model, as well as providing a valuable tool to date marine samples for the recent past (~ 60 years) with high accuracy. ^{14}C has also been useful for tracing sources of organic matter in estuaries and the ocean.

^{54}Mn , ^{58}Co , ^{60}Co , ^{137}Cs can be used to study metal physiology, such as trophic transfer factors in biological organisms in the marine areas adjacent to the nuclear waste discharge facilities. ^{85}Kr , ^{90}Sr , ^{129}I , ^{137}Cs , ^{238}Pu , ^{239}Pu , ^{241}Am have great potential as tracers for ocean circulation, including deep water formation originated from regional climate changes (e.g., Pittauer et al., 2017). They likewise have the potential for tracing ice rafted sediment movements in the Polar seas (e.g., Landa et al., 1998).

Point-like sources of artificial radionuclides of ^{129}I and ^{236}U from nuclear reprocessing plants near La Hague and Sellafield were recently utilized to investigate spreading of the Atlantic Water pathways in the North Atlantic (Castrillejo et al., 2018, 2020). The arrival times and locations of radioactive fallouts over the globe from Mike and Bravo tests in the Pacific revealed vertical air structure over the Indo-Pacific Ocean (Machta et al., 1956). Similarly, the arrival times and locations of the Fukushima derived radionuclides at the North America, Europe and Asia were utilized to tracking the movement of westerlies in the Northern Hemisphere and its poleward shift in the current warming period (Hernandez-Ceballos et al., 2012).

Bomb radiocarbon assays have provided estimates of the age and growth of whale sharks—now shown to be living longer than 50 years. These sharks are the largest fish in the world and are highly valued in the eco-tourism industry (Ong et al., 2020). Given the rapid progress in marine environmental radioanalytical technology to enable small size samples and single atom counting, more scientific understanding utilizing temporal changes in the distribution of anthropogenic radionuclides in the ocean will be made in the subtle inner workings of the oceans (Povinec et al., 2020).

Although particle reactive short lived artificial radionuclides ^{134}Cs ($t_{1/2} = 2.06$ years) and ^{60}Co ($t_{1/2} = 5.27$ years) were introduced experimentally to trace fine sediment movement in a low energy environment such as pasture or to trace remobilization of recent overbank sediment on river floodplains during subsequent inundation events (e.g., Greenwood et al., 2014), fallout radionuclides of well-known input histories have been generally extensively used to trace the soil erosion in the terrestrial environment (e.g., FAO/IAEA, 2017) and subsequent transport and deposition in the aquatic environment such as lakes and coastal oceans (e.g., Froger et al., 2018; Fuller et al., 1999). Once the changes in provenance and accumulation rates of bulk sediment grains established over time since 1945 to the present using artificial or in combination with natural radionuclides, the concurrent particle bound contaminants or other matter input history have been reconstructed since the early 1970s (e.g., Hornberger et al., 1999; Kamula et al., 2017).

Conclusions

Global oceans have been contaminated due to large-scale atmospheric weapons testing from the 1940s to 1980s. Three large scale civilian nuclear industry installations (Windscale, Chernobyl, and Fukushima) have experienced serious accidents, triggered by various causes, resulting in the contamination of regional seas and oceans largely in the Northern Hemisphere in 1950s, 1980s, and 2010s, respectively. Radioactive contamination of oceans by human errors deteriorates the utility of the ocean in a near field coastal ocean area adjacent to the accident. However, the radionuclide contamination levels in a far-field coastal ocean areas have been insignificant with respect to human health. Considering that numerous NPPs are located at coastal areas and regional seas and that environmental protection treaties are not in place, there is an urgent need for developing a form of a legally binding agreement to ensure the safe operation of NPPs in a regional level (e.g., Convention for the protection of the Marine Environment of the North-East Atlantic (OSPAR Convention)) and to develop an international mechanism to deal with the radiological accidents in timely manner, even though the new GEN-IV of NPP are likely to be immune to operators' errors such as occurred in Chernobyl and FDNPP. On the other hand, the released radionuclides to the marine environment have been effectively used as tracers of marine and ocean-atmosphere processes for the protection of sustainable use of the oceans.

Despite the fact that the nuclear energy from primordial origin was largely responsible for the creation of planet Earth the way it came to be and for the evolution of life, the following warning from the 1957 publication (Revelle, 1957) may be still worth to repeat here. "As the use of atomic energy becomes more and more a part of our daily life it is essential that thoughtful attention in broad perspective be paid to the often subtle and perhaps profound effects of this new technology on man and his environment." The well-known input history of radioactive substances into the marine environment has been and will be extensively utilized to

deciphering ocean mixing and global circulation as well as to investigating sources, fates and transport of other chemicals and materials in the ocean environment, including estuaries and coastal environments.

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See Also: Natural and Man-Made Sources of Radiation.

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Environmental Protection: Reducing Environmental Pollution

Andrzej G. Chmielewski^a, Bumsoo Han^b, Sunil Sabharwal^{c,*}, and Maria Helena Sampa^d, ^a Institute of Nuclear Chemistry and Technology, Warsaw, Poland; ^b International Atomic Energy Agency, Vienna, Austria; ^c Radiation Technology Development Division, Bhabha Atomic Research Centre, Trombay, Mumbai, India; and ^d IPEN, Sao Paulo, Brazil

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Abbreviations

BWM	Ballast water management
cfu	Colony forming unit
D ₁₀	10-Fold reduction
DDIC	Daegu Dyeing Industrial Complex
DYETEC	Korea DYEing & Finishing TEchnology Institute
EB	Electron beam
EBFGT	Electron beam flue gas treatment
EB TECH	Accelerator manufacturing company, ROK
e/X	Conversion of electron beam in X-ray beam based on Bremsstrahlung
FGD	Flue gas desulfurization
Free Radical	In chemistry, very reactive molecule that contains at least one unpaired electron
HFO	Heavy fuel oil
IMO	International Maritime Organization
INCT	Institute of Nuclear Chemistry and Technology, Warsaw, Poland
KAERI	Korean Atomic Energy Research Institute
kGy	Radiation energy absorption in material unit [kJ/kg]
MCi	Isotopic source activity Mega (million) Curie [1 Ci = 3.7 × 10 ¹⁰ decays per second]
MW _{th}	Thermal power of the fossil fuel fired boiler
Nm ³	Volume of gas per cubic meter in normal conditions (20 °C, 1 atm)
NO _x	Nitrogen oxides
PM	Particulate matter
SCR	Selective catalytic reduction
SO _x	Sulfur oxides

Introduction

Radiation chemistry is a part of physical chemistry similar to photo-chemistry, plasma chemistry, ultrasonic-chemistry etc. Ionizing radiation produces abundant secondary electrons.

Following these primary events, the ions, secondary electrons and excited molecules undergo further transformations, exchanging charges and energy and reacting with surrounding molecules, thereby producing free radicals and other reactive species that finally evolve into new stable products. Therefore, radiation can modify physical, chemical and biological properties of materials.

Three main sources of radiation are applied for radiation processing. These are electron accelerators, gamma sources and X-ray units based on e/X conversion processes. Accelerators are capable of supplying electron beams in the energy range up to 10 MeV

*Retired

(Zimek and Chmielewski, 1993) and sources of the radionuclides Co-60 and Cs-137, which emit gamma rays of energy 1.17/1.33 and 0.662 MeV respectively (IAEA, 2005). The recent introduction of powerful new X-ray (Bremsstrahlung) radiation sources open new unexplored fields as well (Stichelbaut et al., 2004). Electron beams provide corpuscular radiation, characterized by limited penetration. The entire energy of high-energy electrons is deposited in relatively thin layers of material. In the case of X-rays and gamma rays, ionizing radiation is provided by photons that have no mass and are thus able to penetrate deeper into materials.

The cobalt-60 emitted gamma rays penetrate up to 300 mm of 1 g cm^{-3} density (in a material like water) on an equal intensity material entrance—exit basis. In contrast, the highest electron energy used in commercial applications, 10 MeV, penetrates only 38 mm. Dose rates of gamma and X-rays are 4–5 orders of magnitude lower as compared to electron beams. Therefore, throughput efficiencies of gamma and X-rays are significantly lower than those of e-beams. Electron beams are capable of delivering 100 kGy per second, whereas the typical dose rate for gamma rays is 2.8×10^{-3} kGy per second or ~ 10 kGy per h.

The radiation dose rate generated by an electron accelerator of 15 kW power is approximately equal to that generated by 1 MCi of the cobalt source. When an Electron beam passes through matter (or a metal target on its way), it slows down and a fraction of its energy is directly converted into X-rays. This phenomenon is called *Bremsstrahlung* (German word) from *bremsen* “to brake” and *Strahlung* “radiation”; i.e., “braking radiation.” The X-ray target conversion efficiency is defined to be the fraction of EB energy converted into X-rays. The actual fraction varies with the atomic number of the metal used, but it is typically no higher than 5–8%. In practice this means that in order for an X-ray to process products with the same speed as a 10 MeV, 50 kW e-beam, it will need to have 625 kW of power. Over the last few decades, extensive work has been carried out for utilizing radiation technology for environmental remediation like flue gases from the power industry based on fossil fuel combustion purification, industrial wastewater treatment and municipal wastewater sludge hygienization, hospital waste sanitization, etc. (Chmielewski, 2005; Chmielewski and Han, 2016).

Flue gas treatment

Fossil fuels, which include coal, natural gas, petroleum, shale oil and bitumen, are the primary global sources of heat and electrical energy production. They are responsible for emitting a large number of pollutants into the atmosphere with off-gases from industrial use, power stations, residential heating systems, and vehicles. All these fuels contain major constituents (carbon, hydrogen, oxygen) as well as other trace materials, such as metals, sulfur and nitrogen compounds. During the combustion process, different pollutants like fly ash, sulfur oxides (SO_2 and SO_3), nitrogen oxides ($\text{NO}_x = \text{NO}_2 + \text{NO}$) and volatile organic compounds are emitted. Wet and dry deposition of inorganic pollutants leads to acidification of the environment. These phenomena affect human health, increase corrosion, and destroy forests and agricultural crops.

Wet flue gas desulfurization (FGD) and selective catalytic reduction (SCR) can be applied for flue gas treatment and SO_2 and NO_x emission control. All these technologies are complex chemical processes; and waste, like wastewater, gypsum and used catalyst, are generated.

Electron beam flue gas treatment technology (EBFGT) is a dry-scrubbing process of simultaneous SO_2 and NO_x removal, where no waste except the usable by-products are generated. Studies show that irradiation of flue gases with an electron beam can bring about chemical changes that make removal of sulfur and nitrogen oxides easier. The main components of flue gases are N_2 , O_2 , H_2O , and CO_2 , with much lower concentrations of SO_x and NO_x . NH_3 is used as an additive to aid the removal of the sulfur and nitrogen oxides. Radiation energy is absorbed by gas components in proportion to their mass fraction in the mixture. The fast electrons slow down and secondary electrons are formed, which play an important role in overall energy transfer. After irradiation, fast electrons interact with gas, creating various ions and radicals. In the case of high-water vapor content in the treated off gases, the oxidizing free radicals concentration is high. These species take part in a variety of ion-molecule reactions, neutralization reactions, and dimerization. After humidification and lowering of the temperature, flue gases are guided to a reaction chamber, where irradiation by an electron beam takes place. Ammonia is injected upstream from the irradiation chamber. Nitrogen oxide is reduced to atmospheric nitrogen. The byproduct formed in the process is a mixture of ammonium sulfate and nitrate, which is a good fertilizer component.

Electron beam flue gas treatment industrial plants have been constructed in coal-fired plants in China and Poland (Fig. 1) (Chmielewski, 2011). The installation consists of four main separated systems: flue gas conditioning unit, ammonia storage and injection unit, process vessels, and by-product collecting and storage unit (Fig. 2).

The first industrial installation for the simultaneous removal of SO_2 and NO_x was located in the Electropower Station “Pomorzany” in Szczecin (Fig. 1), in the north of Poland. The installation purified flue gases from two Benson co-generation boilers, which provided 65 MW of electricity and 200 MW of thermal energy to help heat the town. The total maximum flue gases flow rate is $270,000 \text{ Nm}^3/\text{h}^1$ (Chmielewski, 2019).

Recently, there has been significant concern about the air pollution from marine sources that currently utilize low quality diesel fuels. As a result, research and development projects have focused heavily on creating a cost-effective technology that can clean off gases with a high level of efficiency.

Exhausts from marine engines contain nitrogen, oxygen, carbon dioxide and water vapor as well as nitrogen oxides, sulfur oxides, carbon monoxide, various hydrocarbons and complex particulate matter. The maritime transporters usually use heavy fuel oil (HFO) with a high content of sulfur, which naturally leads to the three main pollutants derived from shipping: nitrogen oxides (NO_x), sulfur oxides (SO_x) and particulate matter (PM). Around 15% of global NO_x and 5–8% of SO_x emissions are attributable to ocean-going ships. SO_2 emission as a smog component is a precursor to acid rain and it can have a negative influence on plant life as well as on wider ecosystems.



Fig. 1 View of EPS Pomorzany, Szczecin, Poland. Flue gas humidifier tower of EBFGT plant—front left.

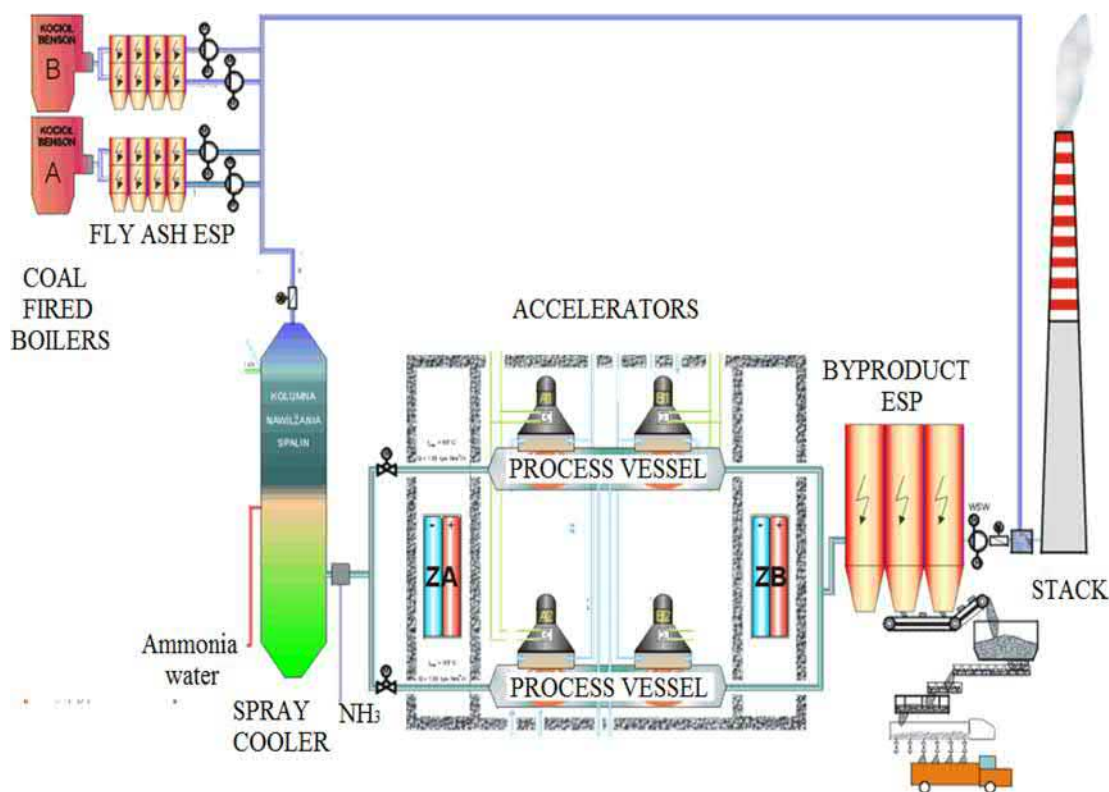


Fig. 2 Scheme of installation for radiation treatment of coal fired boiler flue gases.

A new, hybrid technology is based on the concept of combining two methods used to clean up the exhaust gases: Electron Beam (EB) and Wet Scrubbing. A new emerging hybrid technology that couples the Electron Beam with the reduced size wet scrubbing methods may provide an answer to reducing emissions from the marine shipping industry (King et al., 2019). There are two main stages involved: (1) SO_2 and NO_x (mainly nonsoluble and nonreactive NO) oxidation during irradiation by the Electron Beam from the accelerator and (2) the transformed by irradiation well soluble pollution products (SO_3 and NO_2) absorption into aqueous solution. The technology has been developed by INCT and tested in Warsaw, Poland in laboratory scale (King et al., 2019) and in pilot plant scale with application of mobile accelerator system developed by Fraunhofer FEP of Dresden (Germany) (CERN, 2019).

Wastewater treatment

Increasing urbanization in the last two centuries has been accompanied by an expansion of sewage collection systems without any or adequate treatment. Liquid waste loads have become so large that the self-purification capacity of receiving streams down-stream of large populations can no longer prevent adverse effects on water quality. These wastes now constitute significant sources of water pollution. The industrial effluents carry chemical contaminations such as heavy metals, organic pollutants, petrochemicals, pesticides and dyes, while the discharge of sewage and sludge gives rise to microbiological and chemical contamination of water bodies. The discharge of such materials into water bodies is responsible for risk of infection, health effects caused by contaminated drinking water and offensive odors. Therefore, all these industrial and municipal wastewater systems need adequate treatment.

Radiation processing of wastewater treatment is non-chemical, and uses fast formation of short-lived reactive species that can interact with a wide range of pollutants. High-energy irradiation produces instantaneous radiolytic transformations by energy transfer from high energy photons or accelerated electrons to orbital electrons of water molecules. Absorbed energy disturbs the electron system of the molecule and results in breakage of interatomic bonds. These become the most important products of the fragmentation and primary interactions (radiolytic products). High chemical reactivity is characteristic of water radiolysis products. A typical time for their reactions with the impurities in water is, as a rule, less than 1 μ s.

A large number of substances such as hard surfactants, lignins, and pesticides cannot be degraded by conventional biochemical methods. Thus, they escape from decomposition in biological treatment. Biodegradation quality of wastewater depends on the oxidation level and structure of pollutants. Preliminary oxidation and fragmentation of biologically resistant molecules contribute to the improvement of their biodegradability.

The above-mentioned mechanism of radiolytical oxidation provides the possibility for the required transformation of various pollutants (Sampa et al., 2007). Research and industrial treatments testify significant improvements of pollutant biodegradability after radiation-oxidation in aerated wastewater. Usually a dose of about 1–2 kGy is necessary for complete transformation of biologically resistant pollutants to a biodegradable state. EB TECH Co, DYETEC, and KAERI built an installation of a high-power accelerator (1 MeV 400 kW) and wastewater treatment system in the Daegu Dyeing Industrial Complex (DDIC), Korea, and started operation of the plant. This plant treats up to 10,000 m³ of textile dyeing wastewater (from the total of 80,000 m³) per day and has shown good removal of non-degradable organic impurities (Han et al., 2012). This is a breakthrough in the technology implementation, worldwide, and the biggest wastewater treatment plant ever built based on an electron accelerator unit (Fig. 3).

Other possible an accelerator application is ballast water treatment. The discharge of sea water used for ballasting ships poses a threat to coastal ecosystems. The introduction of alien species from ballast water discharge may lead to excessive development of these organisms in the new environment and cause a threat to native plant and animal communities. Low salinity is the basic natural factor shaping the biodiversity of the Baltic Sea and results in a small species diversity in comparison with other seas. Therefore, brackish seas, such as the Baltic Sea, are particularly sensitive to newly introduced species because they are poor in native species. In 2004, the International Maritime Organization (IMO) developed the Convention on the Control and Management of Ship's Ballast Water (BWM Convention), which provides a uniform legal regulation of the ballast water problem around the world.

In accordance with the legal requirements, ships, including docks, may only discharge ballast water if the number of microorganisms in the discharged water does not exceed:

- for *Vibrio cholerae* (contagious less than 1 colony forming unit (cfu) per 100 mL or less than 1 cfu per 1 g (wet weight) of zooplankton samples,
- *Escherichia coli* less than 250 cfu per 100 mL,
- *Enterococci* (intestinal) less than 100 cfu per 100 mL.

A ship, before entering a shipyard, has to replace the ballast water (with its antibiological treatment) in open waters. However, some waters still remain, together with solid state sediments containing biological contamination. When the ship is placed on the dry

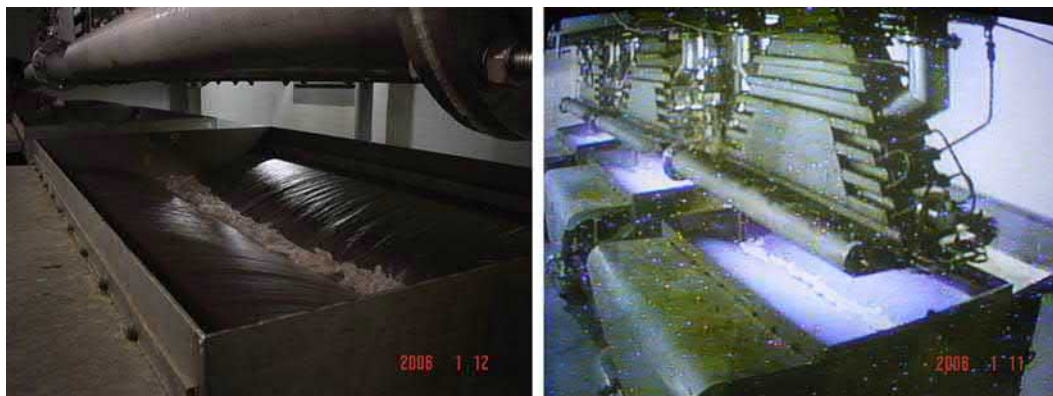


Fig. 3 Industrial EB plant (10,000 m³/day) for textile dyeing wastewater in Daegu, Korea.

dock for maintenance (repair, painting, etc.), this ballast water has to be discharged. For the controlled pathogens, the D_{10} dose [kGy] resulting in a 10-fold reduction in the population is as follows:

- *Vibrio cholerae*, *V. parahaemolyticus*, *V. vulnificus*, $D_{10} = 0.1$ kGy,
- *E. coli* $D_{10} = 0.5$ kGy,
- *Enterococci* $D_{10} = 0.6$ kGy.

To enhance a better implementation of technology from the laboratory scale to commercial scale, the continuous operation by pilot scale is essential. For this purpose, a mobile e-beam plant has been designed in several countries. This movable plant (Han et al., 2016) is for individual field applications with a self-shielded structure that fulfills necessary safety requirements (Fig. 4). It can be used for on-site test of waste up to 500m³ of liquid waste per day at 2 kGy or 2000 Nm³ of gases from oil fired boiler in Jeddah, Saudi Arabia at 15 kGy per h (Pawelec et al., 2016).

Sludge hygienization and biohazards

In addition to treating wastewater, research has shown that sewage sludge can be disinfected successfully by exposure to high-energy radiation. At a plant near Munich, doses of 2–3 kGy have destroyed more than 99.9% of the bacteria present in sewage sludge, and at a plant near Boston a slightly higher dose (4 kGy) was used. Higher doses (up to 10 kGy) were required to inactivate more radiation resistant organisms at plants in Albuquerque. Both gamma sources (Co-60, Cs-137) and electron accelerators can be used for the irradiation of sewage sludge (Chmielewski et al., 1995; Kim et al. 2005). Two industrial gamma irradiation plants are working in India and several pilot plants using both type of irradiators were tested all over the world. Gamma source plants were operated in Albuquerque USA and Geiselbullach, Germany along with electron beam-based plants at Deer Island, Boston (Lessel and Suess, 1984; Hossain et al., 2018).

Gamma sources have better penetration, allowing thicker layers of sludge to be irradiated, although they are less powerful and take a longer irradiation time than electron beam sources. A pilot plant using a gamma source is operating in India (Sabharwal et al., 2005). The irradiator system (Fig. 5) can be easily integrated with a conventional treatment plant with flexibility of operation (Fig. 6). Lower radiation dose treatments can be imparted to the sludge with an addition of sensitizing agents such as oxygen, air, ozone etc. About 3 kGy of absorbed dose in sewage sludge removes 99.99% of pathogenic bacteria consistently and reliably in a simple (from technical point of view) process.

The use of radiation treatments allows for the complete elimination of nematodes and parasitic eggs (*Ascaris* sp., *Trichuris* sp., *Toxocara* sp., ATT) and the practical elimination of pathogens, such as contagious bacterial contamination and *Coliform*



Fig. 4 Mobile e-beam plant on EBFGD in Saudi Arabia (upper) and Wastewater treatment in Brazil (bottom).

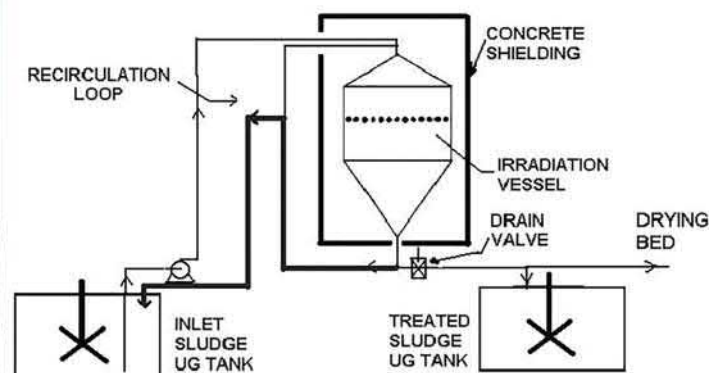


Fig. 5 Photo of the gamma irradiator in Baroda, India (left) and its scheme (right).

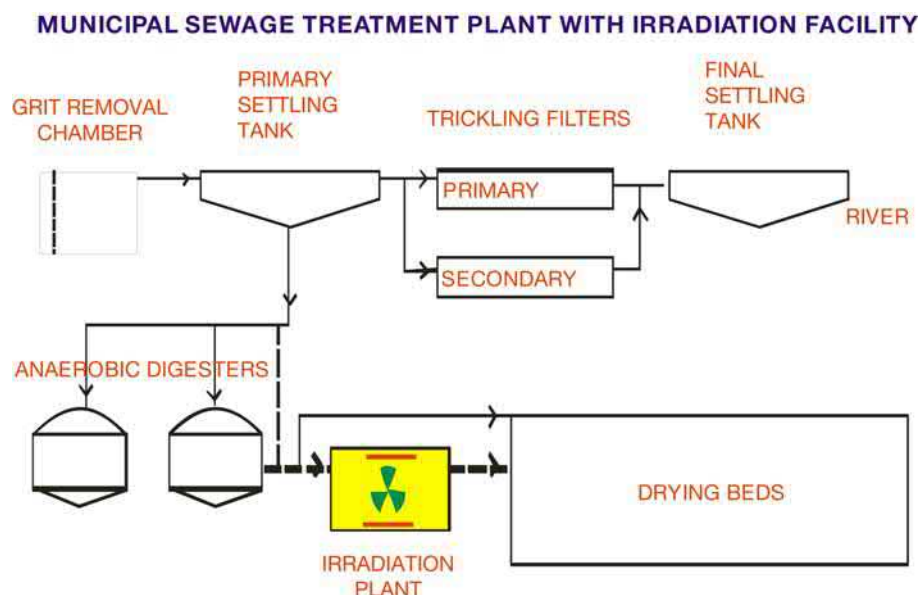


Fig. 6 Scheme of the installation—gamma irradiator treating excess biological sludge from conventional wastewater treatment plant.

titer (decreased by three orders of magnitude for dose 1 kGy), *Clostridium perfringens* titer (decreased by five orders of magnitude for a dose of 1 kGy). The irradiated sludge being pathogen free can then be beneficially used as manure in the agricultural fields as it is rich in nutrients required for the soil. Since the irradiated sludge is free from bacteria, this can also be used as a medium for growing useful bacteria like *rhizobium* and *azetobactor* to produce biofertilizers, which can be used to enhance the crop yields.

A new concept “zero energy” biogas plant is based on the construction of a complex system consisting of a biogas plant and an electron accelerator in the biofertilizer manufacturing line. Digested material (in which organic solids were decomposed into stable substances) originating from the methane fermentation of sewage sludge is irradiated to remove all pathogens using an electron beam from an accelerator powered by electric energy obtained from burning biogas in a co-generator. The product is a high-quality, biologically safe fertilizer (Chmielewski and Sudlitz, 2019).

Special applications have evolved regarding biohazards combat. For example, *Anthrax* that was sent in the US mail in October 2001 caused several deaths and large economical losses. Radiation proved to be very effective for mail decontamination. About 4000 tons of letter mail and 200 tons of parcels had been sanitized by the end of 2003 (Desrosiers, 2004). The mail is being irradiated at a Titan Facility in Lima, Ohio and an IBA facility in Bridgeport, New Jersey. Both of these facilities are operating as electron beam irradiation units (EPA, 2019).

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Public Safety: Fighting Crime and Terrorism; Border Security

Gregory W. Patton, Pacific Northwest National Laboratory, Richland, WA, United States

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Glossary

HEU Highly enriched uranium—uranium containing 20% or greater of the isotope U-235.

MORC Material out of regulatory control—nuclear or radioactive material that should be under regulatory control that is not currently, either from failure of the controls or because it was never under control.

NORM Naturally occurring radioactive material—radioactive material composed of only of materials that occur in nature.

PRD Personal radiation detector—small handheld gamma radiation detectors typically used to provide an early indication of radiation above background.

RDD Radiological dispersive device—illicit device designed to scatter radiation over a general area, for example a dirty bomb that combines an explosive with radioactive material.

RID Radioisotope identification device—radiation detector capable of identifying specific radioisotopes using gamma spectroscopy.

RPM Radiation portal monitor—passive radiation detectors used for screening pedestrians, vehicles, and cargo for the detection of illicit movement of gamma and neutron radioactivity, typically deployed at points of entry.

SPM Spectroscopic portal monitor—a radiation portal monitor that is able to identify specific radioisotopes using gamma spectroscopy.

Weapons-grade plutonium Plutonium greater than 90% in the isotope Pu-239.

Abbreviations

BMWG Border Monitoring Working Group

GICNT Global Initiative to Combat Nuclear Terrorism

IAEA International Atomic Energy Agency

INTERPOL International Criminal Police Organization

ITDB IAEA Incident and Trafficking Database

TRACE Tool for Radiation Alarm and Commodity Evaluation

Introduction

The movement of radioactive materials across international borders is vital to modern society for medicine, science, and commerce. However, when these valuable materials are no longer within regulatory control, either through accidental loss, inadvertent transport, or illicit activities, they represent a global concern requiring national and sometimes international cooperation. The loss of control of radiological and nuclear materials has the potential to effect human health, damage economies, degrade the environment, and impact society. The potential threats for using radioactive materials for terrorist acts include acquisition of a nuclear explosive device, nuclear material to build an improvised nuclear device, material to construct a radiological dispersive or exposure device (RDD), or the potential attack/sabotage of a facility using radioactive material.

The three major pillars of international nuclear security are prevention, detection, and response. Preventing the loss of regulatory control over nuclear and other radioactive material is the initial layer of defense and the strategy with the highest probability for success. Prevention is often summarized as “guns, gates, and guards,” but it involves more than just physical protection. Prevention includes material balance accounting, regulatory licensing and oversight, and establishing laws and regulations criminalizing the theft or diversion of materials. When material falls out of regulatory control, detection systems and measures focus on locating the material through information alerts and/or instrument alarms. Response is the third layer of defense where resources are used to control and mitigate circumstances resulting from a nuclear security incident, with the goal of controlling and stabilizing the situation and returning the material to regulatory control. This article focuses on international efforts to support border security and detection of radioactive material out of regulatory control (MORC) at traditional border crossing points.

Initial efforts for increasing border security for the detection of radioactive MORC were driven by incidents of orphan radioactive sources in shipments of scrap metals that resulted in radiation exposure to the public such as apartments built with cobalt-60 contaminated rebar (Chang et al., 1997) and contaminated materials from the Chernobyl accident. International efforts to improve border security for detecting radiological and nuclear materials increased dramatically following the breakup of the former Soviet Union and the September 2001 terrorist attacks on the United States.

Responsible organizations

Nuclear and radiological security, including nuclear security measures as they apply to border security, is a responsibility of individual countries; however, there is also tremendous international cooperation in this area. Major international organizations participating in nuclear security detection at borders include the International Atomic Energy Agency (IAEA), World Customs Organization, European Commission’s Directorate General for Taxation and Customs, International Criminal Police Organization (INTERPOL), European Union Agency for Law Enforcement Cooperation, United Nations Office of Drugs and Crime, United Nations Interregional Crime and Justice Institute, and the Global Initiative to Combat Nuclear Terrorism (GICNT). In addition, there are many countries that are involved in bilateral and multilateral cooperation. Rather than dealing with individual country efforts, this article summarizes joint efforts for international agencies that assist countries in developing capacity and sustaining a nuclear security regime and detection architecture. The Nuclear Security Summit series of international meetings held in Washington D.C., USA in 2010; Seoul, South Korea in 2012; the Hague, Netherlands in 2014; and Washington, DC, United States in 2016 provided a showcase of international efforts to improve the security of radiological and nuclear materials. In addition, there are informal mechanisms for international cooperation such as the Border Monitoring Working Group (BMWG).

The IAEA Nuclear Security Series documents are recommendations for IAEA Member States to establish and sustain systems and measures for nuclear security, and more broadly a national-level nuclear security detection architecture. These documents are organized into four categories: fundamentals, recommendations, implementation guides, and technical guidance. Fundamentals are the highest-level document and provide overarching objectives and principles, the basis for recommendation documents, and references to international agreements. Recommendations are the application of the fundamentals and provide general approaches, applications, concepts, and strategies. Implementing guides provide the ways and means on how the recommendations are to be applied. Technical guidance documents provide the details on implementing and operating nuclear security systems and measures, and can include reference manuals, training guides, and service manuals. The IAEA also produces non-consensus-based technical documents that can include training guides and service manuals.

The IAEA Incident and Trafficking Database (ITDB) was established in 1995 and provides a valuable resource for reviewing incidents of nuclear and other radioactive MORC. Reporting to the ITDB is voluntary with 136 Member States participating at the end of 2017. The database covers incidents and trafficking of nuclear material, naturally occurring radioactive material (NORM), radioactively contaminated material, and commercial radioactive material. In addition, reporting of incidents of fraud (e.g., scams or hoaxes) claiming nuclear or radioactive material is also encouraged. The ITDB is confidential for Member States and the IAEA; fortunately, the IAEA provides a public domain summary of the reported incidents through an annual fact sheet. The 2018 fact sheet shows a total of 3235 confirmed incidents have been reported from 1993 to 2017. Of these incidents, 278 are confirmed or likely cases that involved trafficking or malicious use. Of the confirmed trafficking or malicious use cases, 12 involved highly enriched uranium (HEU), two involved plutonium, and four involved plutonium/beryllium neutron sources. An additional 913 incidents were recorded without enough information to determine whether trafficking or malicious use took place. All other cases were determined to not, or likely to not, relate to trafficking or malicious use.

The World Customs Organization is an independent intergovernmental body whose Member States work to enhance the effectiveness of custom services relative to regulated trade, and public and environmental safety. Customs organizations have a unique role in border security through the ability to interdict MORC at national borders, either prior to entering or exiting a country's boundaries.

INTERPOL assists in border security for the detection of nuclear and radiological material through information sharing, capacity building, and support for operations, investigations, and prosecutions. INTERPOL supports border security by issuing cooperation notices at the request of each partner country's National Central Bureau. INTERPOL's color-coded notices are international requests for cooperation or alerts, ranging from wanted persons, missing persons, additional information, unidentified bodies, warnings and intelligence, imminent threats, modus operandi of criminals, and joint INTERPOL–UN Security Council special notices. INTERPOL maintains the Geiger database that compiles law enforcement information on potential illicit activity concerning radiological material. INTERPOL member countries can use the Geiger database for conducting risk/threat and trend analyses. Access to both the IAEA ITDB technical data and Geiger criminal data provides a tremendous resource for partner countries in investigating or prosecuting criminal activity and can assist with designing a nuclear detection architecture. INTERPOL also assists in building capacity for border security by facilitating training and workshops.

The BMWG was established in 2005 by the IAEA, European Union, and United States to promote cooperation between its Members and serve as a forum for discussion and exchange of information on plans and programs to be implemented by the Members in cooperation with recipient countries to combat the illicit trafficking of nuclear and other radioactive material (Abousahl et al., 2014). Specific areas of cooperation include radiation detection equipment deployment, training, and long-term sustainability. The BMWG provides an effective tool for avoiding duplication and maximizing resources. For example, joint assessment missions are conducted to ensure common approaches and practices for the assistance and support provided. Group members also implement joint projects to coordinate the deployment and sustainment of equipment through organizing joint workshops and exercises and providing joint training. In the field of training, representatives of the three organizations have worked on development of curricula for frontline officers on radiation detection techniques and for the development of officers as trainers. The BMWG cooperates on research and development aspects related to radiation detection technologies and is involved in the testing of the equipment. The group also encourages and supports targeted projects in areas where the scientific and technical work can bring measurable added value to end users, such as reducing the number of NORM alarms from scintillating plastic radiation portal monitor (RPM) detectors.

The GICNT is an international partnership of 88 nations and six international organizations that are committed to strengthening global capacity to prevent, detect, and respond to nuclear terrorism. The GICNT works toward this goal by conducting multilateral activities that strengthen the plans, policies, procedures, and interoperability of partner nations. All partner nations have committed to implementing the GICNT Statement of Principles, a set of broad nuclear security goals encompassing a range of deterrence, prevention, detection, and response objectives. The eight principles contained within the statement aim to develop partnership capacity to combat nuclear terrorism consistent with national legal authorities and obligations as well as relevant international legal frameworks such as the Convention for the Suppression of Acts of Nuclear Terrorism, the Convention on the Physical Protection of Nuclear Material, and United Nations Security Council Resolutions 1373 and 1540. The United States and Russia serve as co-chairs, while Morocco leads the Implementation and Assessment Group under guidance of the co-chairs. The GICNT conducts multilateral activities and senior-level plenary meetings, and is open to nations that share in its common goals and are committed to combating nuclear terrorism. The GICNT has a Nuclear Detection Working Group chaired by the United Kingdom, a Nuclear Forensics Working Group chaired by Canada, and a Response and Mitigation Working Group chaired by Argentina.

Radioisotopes of concern for border security

Major radioisotopes of concern for border security can be grouped into four major categories: nuclear material, radiological materials, NORM, and medical. The most significant radioactive threat materials are nuclear weapons or components, HEU, weapons-grade plutonium, and commercial radioisotopes with high activity and half-lives with reasonable persistence (e.g., cesium-137 and cobalt-60 irradiators and sterilization sources, iridium-192 radiography) that could be used for radiological disperse or exposure devices. Most of these isotopes of concern emit penetrating ionizing radiation with either a gamma or a gamma/neutron signature. Instrument-based detection equipment at border security locations can detect gamma and neutron radiation. Alpha and beta radioactivity is non-penetrating and is easily shielded by packaging, luggage, and vehicle panels and is therefore not useful for initial screening by radiation detectors at border crossing points. However, many of the alpha and beta emitting radioisotopes also emit gamma radiation and, in the case of beta radiation, there may be some detectable Bremsstrahlung X-ray radiation produced from interaction of the beta radiation with container walls.

Nuclear materials are fissile isotopes that can be used in commercial nuclear power reactors, naval vessels, and for nuclear weapons. The materials of most concern are HEU and weapons-grade plutonium. Although there are other materials that are technically nuclear, they are not practical or available in sufficient amounts for use in weapons (e.g., americium-241). HEU is defined as uranium containing 20% or more of the isotope uranium-235 (IAEA, 2001). HEU emits low-energy gamma radiation, no readily detectable neutrons, and is relatively easy to shield. Plutonium is primarily an alpha emitter, but it also emits both gamma and neutron radiation; neutron radiation is difficult to shield. A neutron-only alarm can be suspicious because it is likely that someone has tried to shield the material and may be an indication of plutonium.

NORM contains long-lived radioisotopes that are ubiquitous in our environment and can be found in many products, construction materials, and geological formations (e.g., truckloads of granite or fire bricks). NORM is the most frequent cause of gamma radiation alarms in customs and border security applications (Kouzes et al., 2002). In large quantities NORM can trigger a gamma alarm. The most commonly encountered NORM radioisotopes are potassium-40, uranium, and thorium and their associated decay products (e.g., radon daughters) (Eisenbud, 1987). Recently the IAEA has released a mobile application (the Tool for Radiation Alarm and Commodity Evaluation or TRACE) to assist frontline officers in determining if radiation alarms generated from cargo shipments are potentially from NORM or an illicit shipment.

Certain radioactive isotopes are used in a wide variety of industrial and medical applications and are commonly encountered in commercial shipping. For example, cesium-137 is used to measure flow in pipelines and as a transmission source in radiography devices, and cobalt-60 can be used to sterilize surgical instruments and in medical therapy. However, these isotopes are also likely source materials for making radiological dispersive devices (RDDs or dirty bombs) or radiological exposure devices. The properties suitable for constructing an RDD include availability in commercial or medical devices, high activity, and moderate to long radioactive half-life. This leads to a relatively short list of radioisotopes of concern that includes: americium-241, californium-252, cesium-137, cobalt-60, iridium-192, plutonium-238, polonium-210, radium-226, and strontium-90.

Medical pharmaceuticals (e.g., technetium-99m and iodine-131) are widely used throughout the world for diagnosis and treatment of disease. While these medical sources can have moderate to high activity, they generally have relatively short half-lives and are not considered a major threat for illicit activity in RDDs. Due to the short half-lives of most radiopharmaceuticals, frontline officers typically encounter these isotopes when monitoring airports and express shipping containers. Frontline officers more frequently encounter radiopharmaceuticals when patients under active treatment are detected by instruments at border crossing points. Some patients can have enough activity to set off radiation detection equipment in multiple vehicle monitoring lanes (Kouzes et al., 2002). Radiation alarms at border crossing points can be cleared through a medical document check and/or using a handheld radioisotope identification detector (RID).

Preventing the loss of regulatory control of radioactive materials is of paramount importance for nuclear security (IAEA, 2002a). Placing radioactive sources under strict physical security, eliminating the use of radioactive sources by applying alternative technologies, and applying regulatory oversight can reduce the chance of a source falling out of regulatory control. Once a source falls out of regulatory control, we must rely on the more limited detection and response pillars to resolve to the situation.

Detection

There are two approaches for detection of MORC: information alert and instrument alarms (IAEA, 2002b).

Detection by information alert

Information alerts can come from a wide range of sources but most are reports of regulatory noncompliance or loss of regulatory control, suspicious activity, public reporting, medical observations (e.g., emergency room indications of radiation exposure), international databases (e.g., IAEA and INTERPOL), bilateral or multilateral information and data sharing, and country-specific operational information such as informants and traditional intelligence. The observation and investigative skills of frontline officers have resulted in many seizures of radioactive MORC. The seizure of HEU without the aid of radiation detection equipment by an attentive and proactive frontline officer at the Bulgarian border is an excellent example (Niemeyer and Hutcheon, 2003; Keegan et al., 2016). A review of proliferation incidents involving fissile material trafficking reported that traditional investigative methods such as observations during unrelated investigations, undercover sting operations, reports of missing inventory material to security forces, anonymous tips, and observational skills during routine operations resulted in most of the incidents (Potter and Sokova, 2002). Detection by information alert is particularly valuable in cases where traffickers attempt to use masking or shielding to limit the ability of detection equipment.

Detection by instrument alarm

Alarms are generated when thresholds for detection equipment are exceeded, typically at an inspection point that is situated at a border crossing or other port of entry (e.g., airport, ferry terminal). Typical equipment at inspection points includes RPMs, personal radiation detectors (PRDs), handheld radiation detection and/or isotope identification equipment, metal detectors, and X-ray systems. RPMs are in wide use worldwide for border crossings or ports of entry. For example, the U.S. Department of Energy's National Nuclear Security Administration has partnered with more than 70 countries to jointly deploy nearly 800 radiation detection systems, primarily RPMs. RPMs are used to screen pedestrian, vehicle, truck traffic, and cargo containers for penetrating radiation (gamma and neutron). The IAEA Nuclear Security Series *Technical Guidance on Technical and Functional Specifications for Border Monitoring Equipment* (IAEA, 2006) provides detailed descriptions and specifications for permanently installed (fixed) pedestrian, vehicle, and rail monitors. In addition, the Technical Guide provides details and specifications for supplemental detection equipment such as PRDs, handheld radiation detectors, and neutron detectors. The combined operations of the fixed RPMs, which are typically used for primary detection, PRDs, and handheld detectors for secondary confirmation will be discussed later.

RPMs are designed to be automated and provide high throughput of pedestrians, vehicles, and cargo such that properly designed and maintained systems should not significantly impact traffic for non-alarming situations. The systems are typically installed at restriction points for the normal flow of traffic such as document inspection stations or at a vehicle drop bar. The systems are also designed to minimize additional labor for frontline officers. Officers can typically conduct their traditional activities while the RPM equipment runs relatively unattended. Once a radiation alarm is generated, officers will need to evaluate the alarm, determine if a confirmation inspection is required, and then follow their standard operation procedures to resolve the alarm.

Gamma detection at fixed RPMs

Most RPM locations have gross (total) counting gamma detection equipment using passive scintillating-plastic detectors (e.g., polyvinyl toluene). While the size of the detectors and lane width can vary based on the target object (e.g., pedestrian or vehicle) and the country's risk assessment analysis, most RPMs are based on an opposing two-pillar design, with gamma and neutron detectors on both sides. Vehicle and rail RPMs typically have two detector banks in each pillar to provide coverage for both low (cars) and high (tractor trailers) conveyances. The detectors in the RPMs are shielded from behind to reduce radioactive background and improve detection sensitivity. RPMs may have additional collimators added at locations with high background or where active equipment such as X-rays cause interference because gamma detectors also respond to X-rays.

Most RPMs operate by summing the counts from all detectors and then setting a background level that is automatically adjusted for minor changes in ambient radiation. The RPM detectors work in conjunction with an occupancy (presence) sensor that detects when a target object has entered the detection zone. Once an occupancy occurs, the system fixes the average background and an alarm threshold is then set above the average background (alarm threshold = average background + $n \sigma$, where σ = square root of the background and n is a positive number). The alarm threshold setting is optimized by adjusting the value of n to balance the need for sensitive detection of target materials versus the ability to process nuisance alarms (e.g., false alarms caused by variable background). Decreasing the n value lowers the alarm threshold but increases the probability of a false alarm. Increasing the n value decreases the probability of a false alarm but raises the alarm threshold. The threshold setting also considers the threat basis and site-specific operational limitations. When the alarm threshold is exceeded during movement of the conveyance through the RPM, a radiation alarm occurs and frontline officers are notified by visual, audible, or computer-based notification; this initiates the verification or secondary screening that will be discussed in the alarm response section. Most RPM systems are equipped with video recording equipment and alarms are adjudicated using computer-based software to process and store event details.

Spectroscopic gamma detection equipment

Considerable work has been conducted to design, evaluate, and deploy spectroscopic portal monitors (SPMs) for gamma detection that combine the sensitivity of the gross counting RPMs with a spectroscopic capability to identify radioisotopes ([National Research Council, 2009](#)). SPMs can both detect and identify radioisotopes during a pass through the portal with the potential for higher conveyance throughput and the simplification or possible elimination of a manpower-intensive secondary screening using handheld radiation detectors. SPMs have the potential to reduce the number of secondary inspections for alarms triggered by NORM or medical isotopes. SPMs have been designed using sodium-iodide and high-purity germanium detectors arranged in similar geometries as the gross-counting RPMs. Limitations of the SPMs include higher costs and greater complexity of the equipment, the need for additional training, and higher sustainability costs for maintenance. Currently SPMs are not widely deployed in comparison to the gross-counting RPMs for border security.

Neutron detection

Neutron alarms are especially important because, unlike gamma alarms, there are virtually no natural or background neutron sources. While it might be possible to have a neutron alarm due to a higher background or cosmic event, it is much more likely the alarm is real. The resolution of neutron alarms should be prioritized because they may indicate the presence of nuclear material, specifically plutonium, or the legal or inadvertent transit of a neutron source. While there are legitimate neutron sources such as californium-252 in well-logging equipment or americium/beryllium and plutonium/beryllium neutron sources in commerce, these should always be accompanied by a license and proper shipping documentation. The shipment of large containers of uranium hexafluoride, used to produce commercial power reactor nuclear fuel, provides one of the few examples where a neutron alarm may occur. With this material, neutrons are produced from the fluorine nucleus by interactions with alpha particles from the radioactive decay of uranium (an alpha-neutron reaction).

Helium-3 tubes are the most common neutron detectors and are considered the gold standard due to high detection efficiency, excellent gamma energy discrimination, and non-toxicity. Limitations for helium-3 include its limited supply and that the cost has been rapidly increasing ([Hanson, 2010](#)). Helium-3 tubes are gas-filled stainless steel tubes with a high-voltage wire strung down the middle. The tubes are housed inside a plastic enclosure to slow down (moderate) neutrons to improve detectability. The thermal (slow) neutrons are captured by the helium-3 atoms, which then split into ions that interact with the high-voltage wire and containing wall of the tube to generate an electrical pulse that is counted by the instrumentation. The moderating plastic surrounding helium-3 detectors increases detection sensitivity but effectively limits the ability to measure differing neutron energies. The detectors are not able to identify the specific neutron-emitting radioactive isotopes; however, this is not a major limitation because the

site-specific standard operating procedure should require resolution of any confirmed neutron alarm. In addition, most neutron-emitting isotopes have associated gamma rays that may allow for identification during a secondary inspection using a RID.

Alternative neutron detectors

With the shortage and high cost of helium-3, there have been considerable efforts to develop alternate neutron detectors. Boron-10 and lithium-6 are the leading alternatives. Boron trifluoride gas enriched in boron-10 was commonly used for gas-proportional detectors prior to the widespread use of helium-3 and is a potential low-cost alternative with a similar form factor for the detectors (Kouzes et al., 2009). However, boron trifluoride is a toxic gas and would require some additional mitigation to ensure safe use. Boron-lined proportional detectors have been developed using either a thin layer of boron-10 inside sealed gas-filled tubes or as boron-lined straw tubes (Kouzes et al., 2010a; Knoll, 2010). Detectors using lithium-6 scintillation systems are another promising option (Kouzes et al., 2010b; Knoll, 2010).

Operational factors affecting the sensitivity of radiation detection using RPMs

Many factors affect how well-fixed detection equipment works. Some are intrinsic such as the type and response of the detectors and the amount and nature of the radioisotope. Operationally, the three initial factors to consider for maximizing detection capability are controlling speed of the target past the detectors, optimizing the distance (d) between the detector and the target, and understanding the role of shielding between the target and the detectors. Controlling the speed of the vehicle or person passing through the detection zone increases the time spent near the detector which maximizes sensitivity to radioactive sources. Reducing the distance between detectors and source of radioactivity is particularly important because the intensity of the radiation decreases by the inverse square law ($1/d^2$) and is a major reason why RPMs use parallel detection pillars and data summing. Shielding provides a unique challenge because there is no effective way to reduce the amount of shielding between the detector and radioactive source. Frontline officers need to be aware that shielding may be present and if shielding is suspected, metal detectors, X-ray imaging equipment, or a detailed physical inspection could help confirm the situation. Masking is another situation to consider, where a benign radiation alarm from either NORM, a medical treatment, or a legally transported radioactive source would be used to conceal an illicit shipment by complicating both the radiation detection and inspection processes. Masking scenarios are attempts to make the radiation emitted from smuggled material appear to come from legitimate cargo. For both shielding and masking situations, detection may be possible from a careful and detailed inspection process combined with investigative skills or an information alert.

Handheld gamma radiation detection equipment

Personal radiation detectors

PRDs are small battery powered devices that are highly portable, have fast response, and are easy to operate. PRDs are primarily used as safety devices for frontline officers and are typically worn in a holster on a belt. The devices are set to provide an alarm indication at approximately two to three times background or at a site-specific dose rate; this allows the officers to maintain a safe distance from a radiation source while assessing the dose rate and then proceed according to their site-specific operational plans. A PRD can be used to set up an initial safety perimeter around a conveyance with an elevated dose rate. Most devices use scintillation crystals (e.g., cesium iodide) to detect gamma radiation. For ease of use some PRDs have a simple numeric scale (1–9) or color indicating lights (green, yellow, and red). There is a wide range of supplemental features available for PRDs including higher dose rate detectors (e.g., Geiger-Mueller) and neutron detection (helium-3 or lithium-6); however, due to the small size and limited interfaces, operating agencies should carefully evaluate if these features meet site-specific requirements for higher dose determinations and required neutron sensitivity. The dose rate information provided by PRDs and other handheld radiation detectors are estimated dose only (typically assuming cesium-137 energies) and are not certified dose rate meters.

Handheld radiation detectors and radioisotope identification devices

Handheld radiation detectors are typically larger, heavier, and more sensitive than the PRDs for detecting radioactivity. There is a wide range of handheld equipment used to screen pedestrians and conveyances and all have the general purpose of detecting and determining the location and distribution of radiation (point source or widely distributed). A localized point source usually indicates a discrete source and may be an indicator of illicit activity. A widely distributed source may indicate the presence of NORM; however, officers should be cautious that NORM cargo could be used to mask the signal from an illicit discrete source. Handheld detectors can also be used for officer safety. Radioisotope identification detectors (RID) are handheld devices that identify radioisotopes by analyzing the gamma energies present and comparing them to a library of spectra and most also provide a confidence level associated with the determination. Most RIDs classify the radioisotopes identified into general categories of NORM, industrial, medical, and special nuclear material.

RID detector types vary from thallium-doped sodium iodide [NaI(Tl)] or cerium-doped lanthanum bromide (LaBr₃:Ce) scintillation detectors to semiconductor detectors using high-purity germanium (HPGe). There are pros and cons for each type, with

scintillation detectors typically lower in cost and lighter weight but with only moderate energy resolution. NaI(Tl) detectors are the most widely used due to the relative ease of production and lower cost. The higher cost LaBr₃:Ce detectors have better energy resolution and better scintillation light output (Milbrath et al., 2007; The Royal Society, 2008), but also have an intrinsic background from radioactive lanthanum and actinium impurities. HPGe detectors have excellent gamma energy resolution but need to be cooled and maintained at cryogenic temperatures and are heavier and more expensive than scintillation-based detectors. As with PRDs there is a range of supplemental features available for RIDs that may or may not be appropriate for site-specific requirements.

Integrated alarm response at a border crossing point

Fig. 1 provides a generic depiction of an integrated alarm response for a border crossing point. The flowchart is not intended to be prescriptive but to illustrate some of the challenges and complexity involved during a detection event and is not intended as a replacement for a written site-specific concept of operations. Alarm response starts with the primary detection event, either an instrument alarm or an information alert. For example, a vehicle passing through an RPM at a border crossing point generates a gamma and/or neutron alarm, or for a location without RPMs the detection could occur from an information alert during a passport inspection (e.g., INTERPOL notice).

Either type of alarm would require verification by interviews, inspections with handheld equipment, and/or physical inspection. For simplicity this section will deal with the vehicle scenario. Gamma radiation alarms for small conveyances such as personal vehicles are usually not triggered by NORM because of the limited amount that can be carried. A small conveyance with point sources of gamma radiation could result from legal transport with associated documentation, medical treatment for the driver and/or passengers, or possible illicit transport. Any neutron alarm generated with a small conveyance is highly suspicious and should be confirmed with a second neutron inspection to rule out the possibility of a false alarm from a transient increase in cosmic neutron background. Larger vehicles with commercial cargo or containers have a greater potential to generate NORM alarms in addition to the other alarms discussed above. Radiation alarms can be verified at some locations with an additional pass through the RPM or a pass through an additional RPM dedicated as a verification lane, which can be a valuable tool for neutron verification since it can be difficult to verify small neutron sources with handheld equipment. If the alarm cannot be reproduced with a subsequent pass through an RPM or verified with handheld equipment and no other suspicious observations are present, then the event was likely a false alarm and the people and/or cargo are released.

If the alarm is verified, the officer needs to conduct a hazard assessment and, if safe to proceed, locate the source and determine if the radiation is from a point source or widely distributed. For this discussion an estimated dose rate of $100 \mu\text{Sv h}^{-1}$ at 1 m, a confirmed neutron source, or suspicion of contamination would be a significant hazard and would trigger additional safety

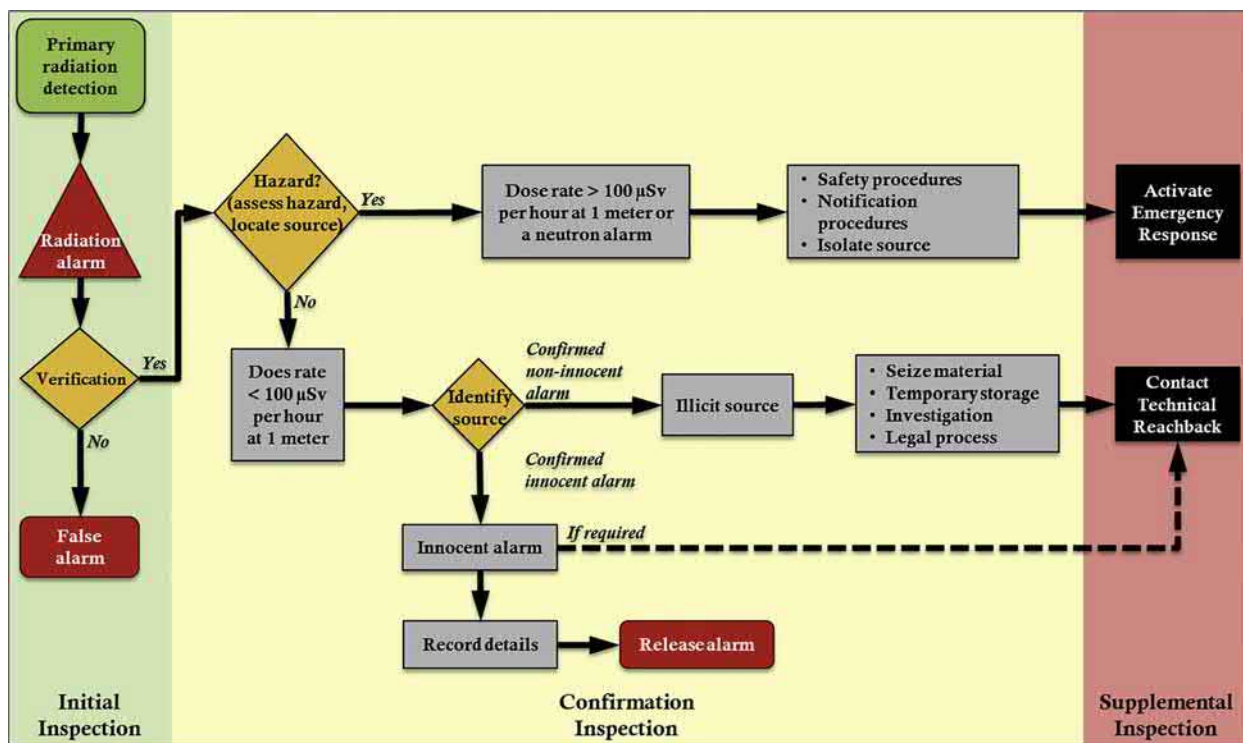


Fig. 1 Generic alarm response flowchart.

procedures, including notification of relevant authorities, initiate measures to isolate and control the source and potential suspects, and activate response procedures detailed in a site-specific plan.

For verified gamma-only alarms with estimated dose rates below $100 \mu\text{Sv h}^{-1}$ at 1 m, the officer can use a RID to identify the radioisotope and determine if the alarm was innocent (medical or NORM) or if an illicit non-innocent alarm has occurred. The officer can also make a gamma-emitting isotope identification without having to open vehicle compartments, cargo, or luggage, which reduces the risk from contamination and other hazards. However, the RID may not provide a high-confidence detection or may report the isotope as “unknown.” In these cases, the investigative skills and training of the officer will be needed to determine if the suspect vehicle and person(s) can be released. The officer may need to contact additional resources from technical experts prior to final release of the alarming conveyance or person. If confirmed non-innocent gamma sources are determined to be illicit sources, the process shifts from detection to response using site-specific procedures to seize the material and/or vehicle, apprehend suspects, provide secure and safe temporary storage of the source and/or vehicle, and initiate any follow-on investigations and legal processes. In most cases the agency that made the detection will need to contact and obtain additional resources from technical experts.

This discussion illustrates some of the complex interactions of officers resolving radiation alarms; however, discussions of high-activity sources, multiple sources, radioactive contamination control, and other hazards (e.g., explosives, physical violence, chemical hazards) are beyond the scope of this introduction.

Sustainability

The risks posed by MORC and efforts to design and deploy radiation detection equipment as part of a national nuclear security detection architecture are significant and require resources to sustain an effective capability. Sustainability planning should include: establishing national and site-level policy and procedures (e.g., concept of operations and response plans); conducting operations in accordance with policies and procedures; maintaining detection equipment and systems; conducting effective training and developing a training cadre; performing programmatic reviews, drills, and exercises to assess and promote continuous improvement of the detection efforts; participating in interagency or international training or evaluation activities; and producing a regular budget-allocated to the relevant agencies and organizations responsible for the sustainment of the nuclear security detection architecture.

Conclusion

The inadvertent or illicit movement of MORC, along with the potential use of radiological and nuclear materials by terrorists, is a real and continuing risk for the global community. Border security efforts to prevent, detect, and respond to these threats are complex and evolving. Radiation detection efforts at border crossing points help to provide an important layer of protection as part of an overall national nuclear security detection architecture and contribute to international nuclear security efforts.

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Public Safety and Security: Emerging Inspection Technologies—Active Interrogation for Nuclear Materials

Dan Strellis^a and Tsahi Gozani^b, ^a Rapiscan Systems, Andover, MA, United States; and ^b Gozani Consulting, Playa Vista, CA, United States

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Glossary

Active Interrogation A non-invasive interrogation method whereby a radiation source is used to induce nuclear reactions in a threat object and radiation detectors are used to detect radiation signatures emanating from the induced reaction.

CONOPS The Concept of Operations is the set of steps and procedures that a user follows to employ the detection technique to suit their application.

DDAA Differential Die-Away Analysis is a nuclear material detection method that uses a pulsed source of neutrons and the time dependent nature of neutron thermalization to measure fission prompt neutrons when the neutron source is not emitting neutrons.

DTOF Differential Time-of-Flight technique is similar to the DDAA technique while adding the impact of the thermal (or epithermal) neutron time-of-flight.

ENG Electronic Neutron Generator is a particle accelerator that typically produces either ~2.5 MeV or 14 MeV neutrons through the $d(D,n)He-3$ or the $d(T,n)\alpha$ reactions.

Fissile isotopes U-233, U-235, Pu-239, Pu-241. These isotopes fission by neutrons with any energy, in particular thermal neutrons.

Fissionable isotopes All fissile isotopes plus Th-232, U-238, Np-237, Pu-236, Pu-240. These isotopes can fission by neutrons with energies in the MeV range.

LEU Low Enriched Uranium has a concentration of U-235 of less than 20% (and the rest U-238).

LS A Liquid Scintillator detector is a radiation detector made of a liquid organic compound that creates scintillation light when exposed to radiation. Liquid scintillators can detect both neutrons or gamma-rays. It can distinguish between the two using PSD (see below).

NORM Naturally Occurring Radioactive Materials are objects and materials that contain radiation producing isotopes as a part of their natural composition. Such items include bananas, granite, and certain fertilizers containing potassium.

PBAR Photofission-Based Alarm Resolution system was a successful prototype detection system using a 9-MeV linear accelerator to induce photofission signatures for SNM detection in conjunction to regular high-resolution X-ray radiography.

PFA The Probability of False Alarm is a common metric for evaluating a detection system's performance. It is measured by taking the ratio between the number of times the system alarms on a non-threat object and the total number of times a non-threat object was shown to the system.

PFNA Pulsed Fast Neutron Analysis is a detection technique whereby nanosecond pulsed neutrons are used to induce gamma-ray emitting nuclear reactions in objects to measure their element composition through gamma-ray detectors and their location by considering the neutron time-of-flight and the time of radiation detection.

PNFP Prompt Neutron from Photofission is a detection technique whereby prompt neutrons are measured during the continuous X-ray exposure using detectors and the pulsed shape discrimination (PSD) technique to distinguish between source X-rays and the prompt neutron signature.

PSD Pulse Shape Discrimination is a detection technique whereby both photons and neutrons can be detected in the same scintillation detector but separated due to the fact that gamma-rays interact with atomic electrons inducing a longer decaying light pulse in the detector than the neutrons that interact by recoiling a proton and produce shorter light pulses.

RPM A Radiation Portal Monitor is a radiation detector system used to detect gamma-ray and neutron radiation that are passively emitted from the cargo to screen cargo in vehicles, trains, and other modes of transport.

Sensitivity A general metric used for a detection technique that refers to the strength of the signal and ultimately to the minimum amount or level of material a system is capable of detecting.

SNM Special Nuclear Material is fissile isotope, i.e. Pu-239, U-235 above 20% enrichment, and U-233, from which nuclear devices can be built.

Specificity A general metric used for characterization of a detection technique that signifies how distinctive the detected signal is to a fissile or fissionable isotope. It relates to the technique's ability to distinguish a threat item from a non-threat item.

TAD A Threshold Activation Detector is a neutron detector used for prompt neutron detection. The detector is comprised of a material that activates when exposed to neutrons above a threshold energy. The detector measures the level of its self-activation as a signal for prompt fission neutrons.

TOF Time-of-Flight refers to the time of flight of a particle from its origin to its point of detection.

Voxel A volume element within an inspected object's volume.

Introduction

In the previous chapter (Patton, 2021), current technologies for addressing border security and public safety in the international arena were presented. They included radiation portal monitors (RPMs), handheld dosimeters, and X-ray inspection. These technologies have been widely deployed throughout the world because they are relatively easy to use and they are relatively inexpensive to acquire and operate. They provide the end user with valuable tools for interdiction of threats in cargo conveyances. However, these tools do have their limitations as the complexity and density of the cargo and types of threat become greater.

For border security applications, there are three primary methods for screening cargo for threats of interest. The major threats are explosives, narcotics, radiological material, and Special Nuclear Material—SNM. The relative merits of the three methods of detection (Passive Detection, Radiography, and Active Interrogation) are dependent to a large extent on the cargo material being screened. As the cargo complexity, density, and atomic number (Z) increase, the effectiveness of one method is reduced and is better addressed by another one. A notional diagram outlining this relationship is shown in Fig. 1.

The simplest method, Passive Detection, was described in Patton (2021). Passive Detection may be sufficient for screening cargoes of low complexity and density. As these factors increase, the radiation signatures (either photons or neutrons) are more effectively attenuated by the cargo and, thus, cannot be detected. One should note that Passive Detection is only applicable to

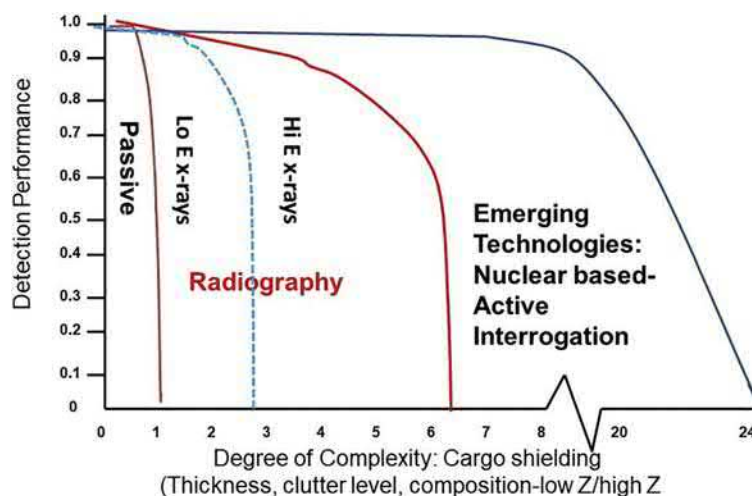


Fig. 1 Notional relationship between detection performance and cargo complexity for the three main technology areas (Passive Radiation Detection, low and high X-ray Radiography, and Active Interrogation).

detecting radiological material. This method has no relevance to detecting other threats such as narcotics and explosives. Moreover, it has only limited capability in detecting SNM.

The second method, X-ray radiography, provides the screener with a visual interpretation of the cargo by way of an X-ray image. Thus, the screener can make a better determination of the cargo contents, potential hiding locations, and a more accurate measurement of dense objects in the cargo. However, X-ray imaging does not provide specific identifiable evidence of radiological material, SNM or other threats. This method only provides indications of the presences of dense objects or objects with relatively high atomic numbers that are commonly used for shielding gamma-ray sources or SNM.

To overcome this deficiency, one can use the third method, active interrogation. This method uses an ionizing form of radiation to induce nuclear reactions in the threat material. The radiation signatures emanating from the reactions are then sensed by detectors. In some cases, active interrogation can be utilized in concert with passive radiation detection because the signatures can be collected with the same radiation detectors. Similarly, active interrogation can be deployed with high energy X-ray Radiography because detection signatures can be induced by the same high energy X-ray source deployed for radiography. Active Interrogation methods are advanced technologies offering capabilities that are lacking in existing technologies but are challenged by relatively high cost, large size, and implied complexities. However, the significant benefits in detection performance make it worthy of exploration in this chapter.

Fission signatures and basic physics of active interrogation

Active Interrogation is a category of screening techniques that utilize a radiation source to induce a unique detectable signature in the material of interest. In contrast to passive techniques that rely on the material alone to provide the detectable signature, active techniques rely on a nuclear reaction between the interrogating radiation source and the material to provide the radiation signature to detect the material. In general, active interrogation techniques are considered more difficult to field. However, they provide performance advantages not achievable with the passive techniques. The radiation sources used for active interrogation range from high-energy X-rays (6 MeV or higher) and neutrons (thermal to 14 MeV). Choosing the radiation source type for the application is a trade-off on complexity, size, performance, and cost.

X-rays interact with matter by way of the atomic electrons. X-rays are either absorbed, scattered or disintegrate after interacting with the atom. In general, the interaction rate with matter is a strong function of atomic number Z ($\sim Z^4$) in the “Photoelectric effect” domain at X-ray energies below 1 MeV. The interaction rate has very little to no Z dependence between X-ray energies in the “Compton” domain of 1–5 MeV. At X-ray energies above 5 MeV, the interaction rate is a strong function of Z ($\sim Z^2$) in the “Pair Production” domain. As a result, X-ray attenuation increases with increasing Z and atomic density. Thus, denser higher- Z materials like lead and tungsten attenuate X-rays to a larger degree than less dense lower- Z materials like plastic and water. It is for this reason that lead is typically used for X-ray radiation shielding.

A neutron is a subatomic particle with no charge and a rest mass energy slightly higher than a proton (939.6 vs. 938.3 MeV). To understand the physics on a neutron’s interaction with matter, it is helpful to do a thought experiment using billiard balls. Due to their similar mass, we can represent a neutron as one ball and a proton as another. If one ball is at rest and the other collides with it, the former imparts a percentage of its kinetic energy to the latter that is dependent on the angle of scattering. Because the masses are similar, the former will impart all of its kinetic energy to the latter with a head-on collision. This is analogous to a neutron interacting with a proton, or in a familiar case, with a hydrogen nucleus consisting of one proton and no neutrons. On average, a neutron loses half of its kinetic energy when colliding (and being scattered) by a hydrogen nucleus. The heavier the element (assumed at rest) the neutron collides with, the smaller will be the fraction of the neutron kinetic energy that is imparted to that element, thus reducing its own kinetic energy. This process is commonly referred to as the Slowing Down process in the field of nuclear physics. The last stage of this process is called Thermalization, in which the neutrons and target nuclei have a comparable kinetic energy. Neutrons will thermalize in smaller number of collisions and, hence thermalize faster in cargo comprised of lower- Z materials (plastics, water, and oil) compared to the higher- Z materials (machine parts, appliances, and cement). This difference in the way that X-rays and neutrons interact with matter is important when considering the effectiveness of active interrogation. X-rays and neutrons are used as sources for this technique. A significant number of the source particles needs to penetrate any cargo that is present before inducing a detectable signature in the material of interest.

The signatures of interest for the detection of SNM are initiated by the fission process. Fission is nuclear reaction whereby a nucleus splits into two lighter nuclei (Klay, 2020). Fission occurs as a result of the competition between the strong nuclear force (that binds the protons and neutrons of a nucleus together) with the Coulomb force (that repels like charged particles away from one another—in this case protons with positive charge). This reaction occurs when a photon or neutron of sufficient energy interacts with the nucleus of a heavy element (thorium and higher) to create a compound nucleus with excess energy that results in splitting into two unequal isotopic nuclei rather than remaining as one. A key parameter for fission based SNM techniques is the magnitude and energy of the interrogating radiation’s dependence of the fission cross section (see Table 1). It expresses the ability of the interrogation radiation to induce fission in the nuclear material. It varies greatly with energy and the type of the probing radiation. For example, U-235 fission cross section ranges from close to 600 b (barn = 10^{-24} cm²) for thermal neutrons, 300 b for the resonance neutrons (0.4–10 keV) down to about 1.2 b for fast neutrons. It is much lower for the photofission process, about 0.001 b at 6 MeV rising to about 0.1 b at 10 MeV photons.

Table 1 Fission cross-sections of long-lived fissile and fissionable isotopes (JAEA, 2020). Values without units in the table are in unit of b (barn).

Isotope	Thermal (0.0253 eV)	Resonance integral	Fission spectrum
^{232}Th	54 (μb)	375 (μb)	80 (mb)
^{233}U	531	775	1.91
^{235}U	585	274	1.22
^{238}U	17 (μb)	1.2	0.31
^{237}Np	20 (mb)	5.37	1.34
^{238}Pu	17.77	27.76	1.97
^{239}Pu	747.4	301	1.80
^{240}Pu	36 (mb)	8.13	1.30
^{241}Pu	1012	567.4	1.63

The fission cross section of Pu-239 has a similar behavior but it is higher by about a factor of 1.1–1.5 than that of U-235. Most other actinides isotopes (e.g., Th-232, U-238, Np-237, Pu-240) undergo fission only by higher energy neutrons (having thresholds above about 0.1 to 1 MeV) but not by thermal neutrons.

Signatures from fission can be grouped by their emission time relative to the fission event and are typically referred to as either prompt or delayed events. The signatures are prompt neutrons, prompt gamma-rays, delayed neutrons, and delayed gamma-rays. Each will be described here.

The prompt neutrons and gamma rays are emitted instantaneously—at the moment—of fission. They are among the strongest signatures available: about 2.4 and 4.5 n/f, corresponding to fission by thermal and 14-MeV neutrons, respectively. On average, about 3 n/f are generated in photofission with X-ray end point energy of 9 MeV (see Table 2).

For prompt signatures, the background radiation due to the interrogation particles is always very high and often overwhelms the detectable signal. In order to detect the prompt neutron signature, in the presence of such a high background, one attempts to discriminate between the two by suitable shielding and some sort of energy and/or temporal or multiplicity discrimination.

The prompt gamma ray signature emitted in the fission process is also a very strong signature. On average, close to seven photons are emitted in the fission process. Unfortunately, it is very difficult to use the prompt gamma ray signature because of the presence of the very strong gamma-ray background from the source neutron interaction with structural and cargo material (when neutron interrogation is used) or the X-ray field (when X-ray interrogation is used). All gamma detectors are generally blinded during the period the interrogation source is on and, hence, could not detect the prompt fission photons. Thus, no practical active interrogation technique is based on detecting prompt gamma-rays from fission. The exception is Pulsed Fast Neutron Analysis (PFNA) that is described in more detail later in this chapter. In this technique fissionable materials are detected via the prompt gamma ray signature employing the time-of-flight method afforded by its nanosecond neutron pulsing capabilities.

The delayed fission signatures (delayed neutrons and delayed gamma-rays) originate from the chain of beta decaying fission fragments (and their daughters). As such, they can be measured well *after* the fission events occurred when a much lower source related backgrounds exist. These signatures, in terms of both absolute intensity and their pattern of temporal decay, are unique to the fission process of the specific actinide isotope. It is common to group the fission fragments that emit delayed neutrons in six groups characterized by the radioactive decay half-life (Previtali, 2021). The absolute delayed neutron yields and their six time-groups for neutron fission are given in Table 3. The same quantities for photofission are given in Table 4. The yield increases slightly as the energy of the photon inducing fission increases. The delayed neutrons are easy to detect, but they suffer from low intensity and lower energy (average ≈ 450 keV), which make them susceptible to absorption, especially in hydrogenous cargo.

The absolute delayed gamma-ray yield and their five time-groups are given in Table 5. These quantities are, within the current experimental accuracy, applicable to both neutron fission and photofission. There are many tens of beta decaying chains of fission products with half-lives from milliseconds to thousands of seconds with multitude of excited states. There are a few weak discernable isolated gamma-ray lines that can be used as a detection signature. However, a much more useful signature is to measure the much stronger and readily detectable gross delayed gamma-ray spectrum. This spectrum (above a few tenths of an MeV) has an exponentially declining intensity vs. energy. The value of the negative energy exponent is roughly 1.1 ($\pm 10\%$). Most of the delayed gamma-rays are emitted within a few minutes, significantly slower than the delayed neutrons, most of which are emitted within 10–

Table 2 Average number of prompt neutrons emitted in fission by neutrons and photons (Caldwell and Dowdy, 1980; Gozani, 2009a, 2009b; JAEA, 2020; Verbeke et al., 2014).

Isotope	0–14 MeV neutrons in neutron fission (n, f) reaction	9 MeV Bremsstrahlung in photofission (γ, f) reaction	Number of higher energy fission neutrons with energy > 3 MeV for (n, f) or (γ, f) reactions
^{235}U	2.4–4.4	3	0.6–1.1
^{238}U	2.3–4.4	2.8	0.6–1.1
^{239}U	2.9–4.9	3.4	0.7–1.2

Table 3 Delayed neutron yield from neutron fission (thermal or fast) (Fisher and Engle, 1964; Tuttle, 1975; Gozani, 1981; Rudstam, 1982).

Group #	Group half-life (s)	Relative delayed neutron group yield		
		U-235	U-238	Pu-239
1	0.18	0.026	0.075	0.038
2	0.50	0.128	0.225	0.28
3	2	0.407	0.388	0.216
4	6	0.188	0.162	0.328
5	22	0.213	0.137	0.103
6	55	0.038	0.013	0.035
Absolute delayed neutron yield per fission (ν_d)		0.017	0.045	0.007

Table 4 Delayed neutron yield from photofission (Kull et al., 1970; Caldwell et al., 1973; Gozani, 1981, 2009a, 2009b; Padovani et al., 2007).

Group #	Group half-life (s)	Relative delayed neutron group yield	
		U-235	U-238
1	0.2	0.03	0.11
2	0.6	0.09	0.18
3	2	0.34	0.36
4	5	0.24	0.17
5	20	0.25	0.16
6	55	0.06	0.02
Absolute delayed neutron yield per fission (ν_d)		0.012	0.035

Table 5 Delayed gamma-ray yield from neutron fission and photofission (Walton et al., 1964; Fisher and Engle, 1964; Gozani, 1981, 2009a, 2009b).

Group #	Group half-life (s)	Relative delayed gamma-ray group yield	
		U-235	U-238
1	0.29	0.011	0.062
2	1.7	0.168	0.203
3	13	0.304	0.324
4	100	0.273	0.233
5	940	0.245	0.179
Absolute delayed gamma-ray yield per fission (γ_d)		6.3	8.7

20 s after the fission event. However, the intensity of delayed gamma-rays, even the high energy fraction above 3 MeV alone—about 2–3% of the total—is order of magnitude higher than the total delayed neutron yield and is much more penetrating in hydrogenous cargo.

The continuous non-discrete shape of the delayed gamma-ray spectrum allows the use of detectors with a mere medium or even poor energy resolutions (like plastic scintillators). This, in turn, allows lower cost large area detectors (Slaughter et al., 2007; Han, 2021) to be employed making the high energy fission delayed gamma-rays a very attractive and practical fission signature.

Signatures to detect nuclear material through active interrogation have been described in this section. The practical techniques that have been demonstrated to identify the presence of nuclear material, either through measuring of the signature itself or measuring a property of material that indicates its possible presence, will be covered in the next sections.

High energy X-ray inspection

X-rays were discovered by Wilhelm Conrad Roentgen in 1895 when he observed a glowing plantinobarium screen several feet away from an experimental study of glass bulb electrodes. One of his earliest images using these X-rays was of his wife's hand showing her wedding ring clearly in one of the most famous images (Fig. 2) in history (Riesz, 1995).



Fig. 2 One of Roentgen's first X-ray image.

Since that time, X-rays have been used in the fields of medicine, industrial sterilization, non-destructive testing and security (Waltar, 2021; Bjornstad, 2021a). Some of their uses may be well known to the reader; others may not. In hospitals and dental offices, an X-ray system is used to image bones and teeth of patients. These images are produced using an X-ray generator on one side of the patient and an X-ray detector (phosphorescent screen or digital x-ray detector) on the other side. The level of X-ray transmission through the body part of interest is dependent upon primarily its density and secondarily its composition, including, bones, fat, muscle, and other tissue. X-rays are either absorbed or scattered to a higher degree by objects with higher atomic numbers (Z) and higher density. For molecules, it is common practice to calculate the effective atomic number (Z_{eff}) based on the chemical formula and the atomic numbers of the elements in the molecule. Thus, when an X-ray interacts with a bone (comprised of mainly calcium) with $Z_{\text{eff}} = 13.8$, it is preferentially attenuated to a larger degree than an X-ray that interacts with muscle ($Z_{\text{eff}} = 4.5\text{--}6$). This differential attenuation of the X-rays results in a higher number of them detected in line with the muscle and tissue versus the bone. This provides a contrast in the image that shows the bone clearly as an X-ray "shadow" that shows the structure of the bone.

The use of X-rays in port and border security is a more specialized field. X-ray inspection is a common technique used at cargo seaports and land ports of entry. At these facilities, entire shipping containers are unloaded from ships in under a minute. In some customs applications, a select number of containers are set aside and screened by customs officials using X-rays to look for contraband materials or instances of duty violations. To screen an entire shipping container that typically contains a wide range of material (tires, engines, computers, tile, furniture, and clothing to name a few), the X-ray screening system typically uses an energy of 4.5 MeV or higher. Typical medical X-ray systems use X-ray energies between 20–150 keV. The energy level for border security is 2–3 orders of magnitude greater, but the general process is similar. The X-rays are produced from an X-ray source. In this case the source is usually a linear accelerator (linac) (Varex, 2020) or a betatron (JME, 2020; Garcia, 2021) capable of producing X-rays by accelerating electrons into a metal target with high atomic number (typically tungsten or a tungsten alloy). These X-rays are collimated into a beam shaped like a fan. X-ray detectors reside on the opposite side of the container to detect those X-rays that penetrate and transmit through the container and its contents and impinge on the detector. A diagram of the basic geometry and subsystem components is shown in Fig. 3.

Material contrast is generated by the structure and composition of the objects being screened. In the cargo screening application, the object being scanned is often a cargo container. The X-ray source is used to provide material contrast using a technique called dual-energy imaging. In this imaging modality, the X-ray source alternates between two different X-ray energies from pulse to pulse separated by 3–20 milliseconds. As a result, two X-ray images are created for the same container and its contents. Due to the Z -

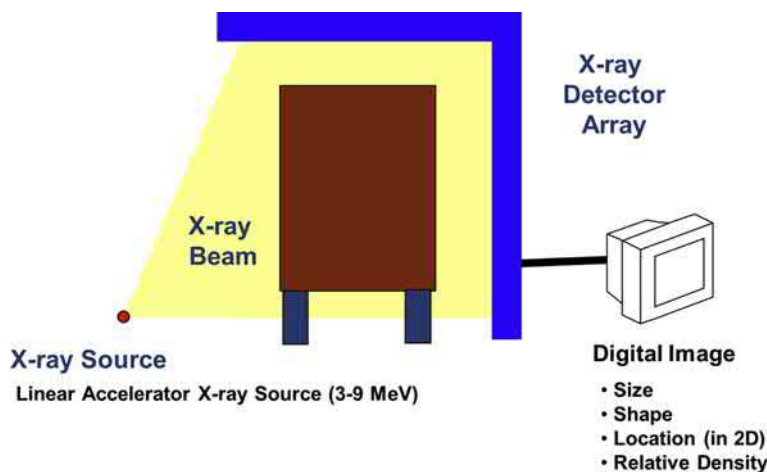


Fig. 3 Schematic of a standard X-ray imaging system used at land and seaports.

dependence on absorption and scattering, the two images provide two different energy dependent data points on the attenuation of objects in the image. These systems are calibrated in the factory with several different materials of known Z to determine the X-ray detector response to the two different X-ray source energies. In the field at a land border crossing or seaport, the system images containers where the contents and material composition are unknown. After the two X-ray images are produced, the Z -calibration from the known objects is used to roughly estimate the Z of the unknown objects.

Special nuclear material (SNM) of interest is comprised mainly of uranium or plutonium. Unlike other methods described later in this chapter, X-rays do not have the ability to specifically identify the presence of U or Pu through transmission imaging alone. Also, the X-ray detectors are not measuring the presence of radiation emanating passively from the SNM with detectors as described in (Patton, 2021). Instead, X-rays screening can provide some material contrast to a broad level. Due to their high Z (92) and high density (19 g/cc) for uranium, SNM, as well as possible shielding materials such as tungsten ($Z = 74$) or lead ($Z = 82$), are highly attenuating to X-rays of energies used in cargo screening techniques. As described above, these high- Z objects will show as a darker object on the image because very few X-rays penetrate through the object and are detected by the detector array. Vendors providing screening equipment to a Customs organization typically provide a color overlay on the image to guide the user's eye to different materials of interest. Organic materials with lower Z will have one color, while metals of higher Z will have another color. The presence of a dark object with a high- Z overlay color could then be possibly indicative of SNM, tungsten or lead shielding. However, a similar-looking image object could also be created by thicker material of lower- Z (such as steel) that is directly in line with the X-ray beam path between source and detector.

The ability to inspect the cargo depends on the ability of the X-ray beam to penetrate the cargo. The higher the X-ray source energy, the greater the penetration and the greater range of cargo can be inspected. Fig. 4 demonstrates the effect that beam energy has on cargo penetration. Notice that the entire cargo is revealed in the 9 MeV X-ray image, whereas areas of the cargo shown in red are not penetrated at lower energies. For cargo inspection, the highest energy currently allowed by the World Health Organization is 10 MeV.

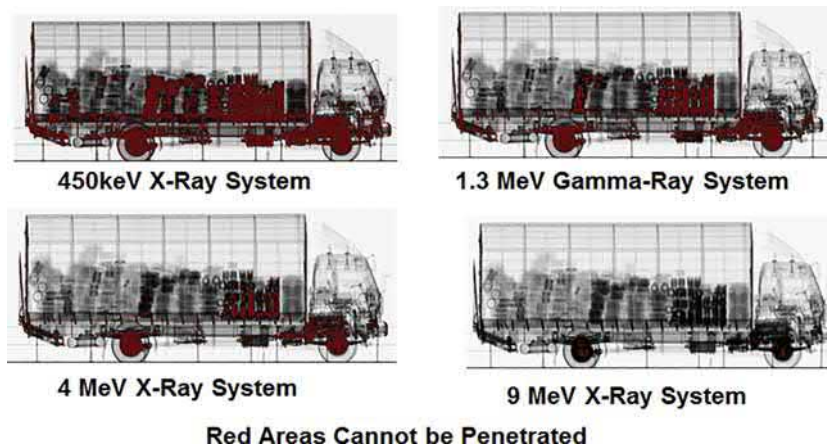


Fig. 4 Examples of cargo penetration vs. photon beam energy.

Transmission imaging alone cannot unequivocally determine the presence of SNM. Additional techniques are required and can in principle be used in concert to provide the information required to be more certain of the suspicious object's material.

Novel techniques and prototype technologies have been studied and developed to work with transmission X-ray screening by utilizing additional signatures emanating from the cargo material interactions with the high energy X-rays.

One technique measures the energy spectrum of the transmitted X-rays using energy sensitive detectors. Traditional methods measure the electrical current induced by the X-rays inside the detector. We already learned that X-ray attenuation is dependent on the type of material attenuating it and the X-ray energy. The linac sources emit X-rays with a broad Bremsstrahlung energy spectrum. The deviation in the shape of the X-ray energy spectrum can provide information about the material being interrogated (Gozani et al., 2014). The effect that a material (of known atomic number) has on the X-ray energy spectral shape is shown in Fig. 5. The energy spectrum of the transmitted X-ray beam through the higher Z material is far richer with lower energy X-rays than in the lower Z material case. A spectral ratio can be calculated by taking a ratio of the low energy spectral region to that of the high energy region. A strong Z dependence is revealed as shown in Fig. 6. Key issues with fielding this technique are the required level of collimation, the large background of multiple scattered Compton photons, and the need to have a stable very fast detector to handle a large dynamic range of count-rate.

Another high energy X-ray technique that provides high Z material detection is based on the high-energy Bremsstrahlung back-scattered radiation (Van Liew et al., 2017). The interrogating X-ray beam generates Compton electrons (and to a lesser extent electron-positron pairs) in its direction as part of its attenuation process by the cargo materials. Many of these electrons undergo multiple collisions with the cargo atoms and a few alter their direction of motion from forward (the X-ray beam direction) to backward direction. These electrons lose energy in part by the Bremsstrahlung process, emitting high energy X-rays that can be detected

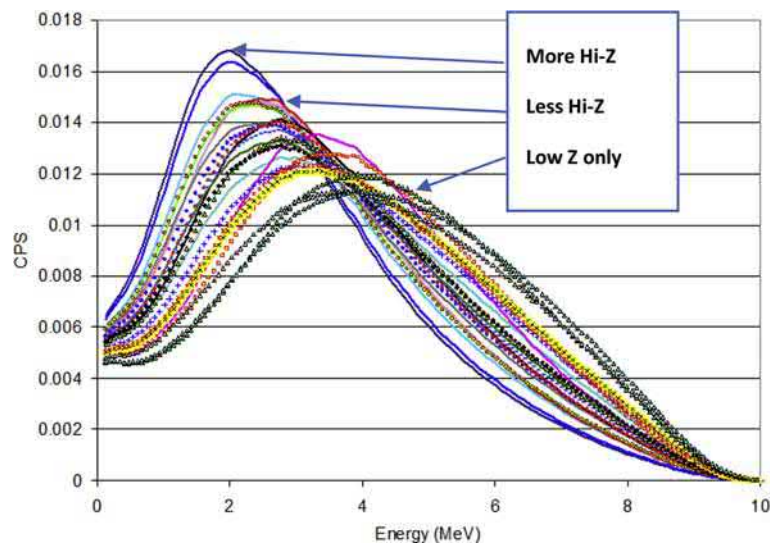


Fig. 5 Bremsstrahlung X-ray spectral shapes after cargo transmission vs Z at a fixed attenuation level.

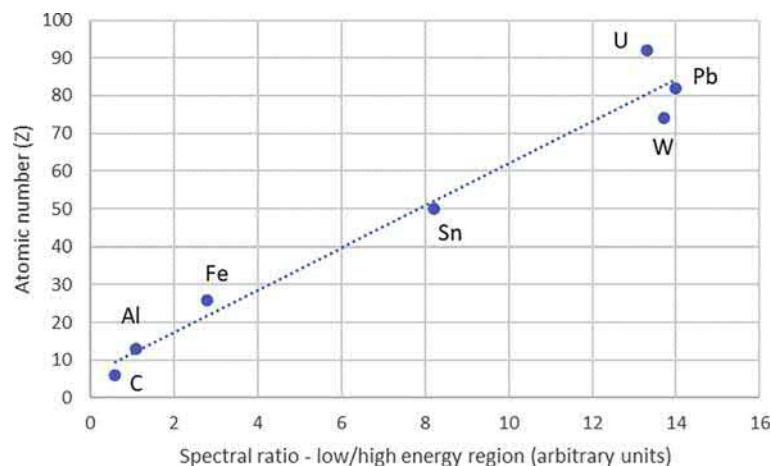


Fig. 6 Atomic number Z as a function of a spectral ratio of low to high energy regions in the transmission X-ray energy spectrum.

by energy sensitive detectors placed at the back angles ($>90^\circ$). The normal backscattering spectrum of Bremsstrahlung radiation is made up of low-energy photons with a maximum energy of 0.256–0.511 MeV (corresponding, respectively, to photon scattering by 180° -backscattering- and 90° scattering). These low energy backscattered X-rays are highly absorbed in cargo material limiting their usefulness for cargo examination to near the surface of the cargo container. There exists, however, a small but noticeable high-energy component (resulting from the Bremsstrahlung process mentioned above) in the X-ray backscattered spectrum that is significantly stronger for high-Z scatterers like tungsten and uranium. These processes are strongly Z dependent (about Z^2). These characteristics can provide a strong indication for high Z material presence. The intensity of the high-energy backscattering signal above 1 MeV for tungsten in air is estimated to be 10^{-12} to 10^{-11} photons/cm²/electron.

The strong Z dependence of the backscattered secondary electron Bremsstrahlung spectrum, expressed by the ratio of low energy part of the spectrum (<0.6 MeV) to that of the high energy measured by NaI detector for 6-MV linac, is shown in Fig. 7.

The Z information gleaned from this method is stronger at 9 MV, as the high energy flux of the Bremsstrahlung is far richer with high energy photons. The main strength of the technique is the strong dependence of the backscattered intensity on the Z of the scattering object. The main drawback of the method is the necessity to use a very strong X-ray source, a large array of efficient and highly collimated medium resolution detectors (such as NaI), and the need to employ a rastered narrow X-ray beam (Van Liew et al., 2017). The intersection area of the beam with the projection of detector collimation area defines the pixel of the scattering material. The motion of the cargo provides the third dimension, defining the voxel of the scattering volume.

High energy X-rays above 6 MeV (though practically >8 MeV) can interact directly and uniquely with SNM via photofission. The process generates unique signatures, like prompt and delayed neutrons and gamma-rays, which are measured with appropriate detectors. Neutron and photofission-based inspection techniques are discussed in the next sections.

Neutron-based inspection techniques

In the previous sections, technologies based mainly on the detection of the high atomic number (Z) and density of materials were described. These properties are characteristic of nuclear materials and their potential shields (such as tungsten or lead). In this section, active neutron interrogation technologies based on one or more fission specific signatures of nuclear materials are described. General papers on neutron interactions, sources and detectors and industrial applications can be found in Bjornstad (2021b), Theresa and Brisset (2021), and Han (2021). Overview publications on neutron and photon-based inspection can be found in Gozani (1981, 1985, 1996, 2010) and Runkle et al. (2012).

Neutron-based inspection techniques are considered to be the most versatile for probing the elemental composition of materials. Fast (>100 keV), epithermal (100 keV–0.4 eV) and thermal (<0.4 eV) neutrons interact with almost all isotopic elements existing in nature, generating characteristic gamma rays, indicating the presence of these isotopes, and by inference that of the corresponding elements. Thus, neutrons can and do detect the presence of explosives, narcotics as well as other substances and nuclear materials, specifically SNM. The latter can be detected very well also by linac-based photofission (see the following section).

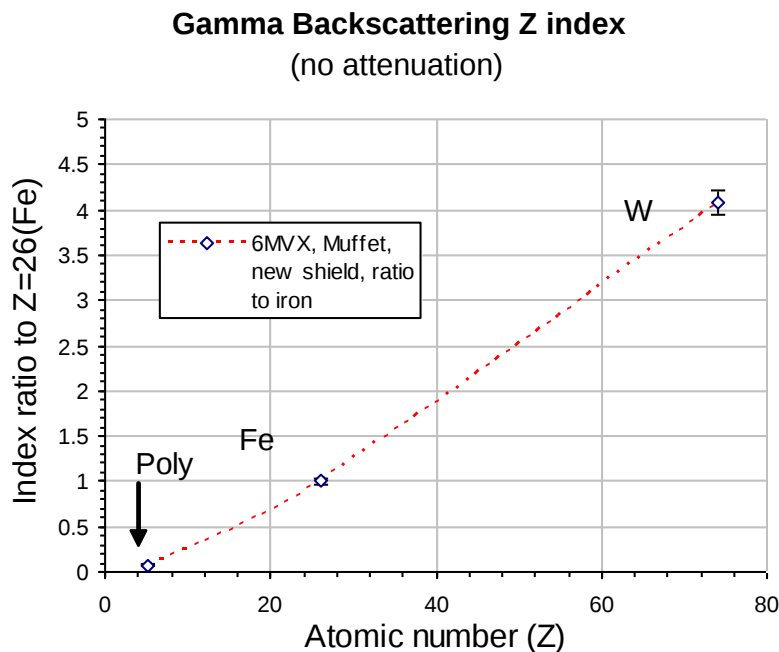


Fig. 7 High energy backscattering spectral ratio vs. Z.

Neutrons are generated as a result of charged particle interactions. The most common neutron producing reactions are the exothermic (d,D) and (d,T), where the accelerated projectile is a deuteron and the target is titanium (or similar metal capable of absorbing hydrogen isotopes like deuterium or tritium). The resulting neutron energy is 2.5 MeV and 14 MeV, respectively. The (d,T) reaction yields about 100 times more neutrons per deuteron than the (d,D) reaction at the low accelerating voltage of 100 keV, where most commercial neutron generators operate. The d-D generators, (depending on design, size, cost etc.) can yield neutron intensities of the order of 10^6 – 10^{10} n/s. The d-T generators yield 10^8 – 10^{13} n/s. These accelerators can provide continuous or pulsed beams with a pulse width as narrow as ≈ 5 μ s and with practically unlimited repetition rate (all the way to a continuous beam).

Neutrons of different energies interact differently with the target nuclei. Fast neutrons interact mainly via inelastic scattering process, where the colliding neutron loses part of its energy and the collided nucleus deexcites by emitting characteristic gamma rays related to its specific nuclear structure. Thermal and epithermal neutrons interact mainly via neutron capture process. Here the colliding neutron is captured by the nucleus that is converted into another nucleus. The two nuclei have the same number of protons but the product nucleus has one more neutron. The two nuclei pertain to different isotopes of the same element. The product isotope is created in an excited state, which then deexcites by emitting characteristic gamma rays. Generally, the capture gamma rays have higher energies (some exceeding 10 MeV) than those generated by the inelastic scattering process. Carbon, nitrogen, and oxygen are exceptions. They have very low thermal neutron capture cross sections but with a substantial fast neutron inelastic cross section resulting in relatively high inelastic gamma energies (2–7 MeV).

All neutrons from zero energy and above can induce fissions (with declining cross sections vs. energy) in fissile isotopes (i.e. SNM). All fissionable isotopes can undergo fission by fast neutrons with energy above roughly 1 MeV. Thus, any neutron-based technique can, in principle, offer detection of fissionable material as well as other threats such as explosive and narcotics. The very high fission cross section of thermal neutrons ($\approx > 600$ b), the strength and specificity of the fission signatures, result in a much higher probability of detection of SNM per gram than that for explosives.

Current technologies

The list of neutron-based technologies relevant to the Active Interrogation (AI) of nuclear materials available or being developed is provided below in Table 6.

Table 6 Main neutron based active interrogation (AI) techniques for SNM

#	Technique name	Probing radiation	Source	Signature	Detectors	Specificity/sensitivity (applicable to)
1	Differential Die Away (DDAA, DTOF)	Thermalized neutrons	Any pulsed ENG (most common (d,T), (d,D)), pulse width $\approx 100\mu$ s	(n_p) and secondarily (n_d)	(1) Moderated neutron detector (^3He & or similar) (2) LS + PSD (3) NaI	Very high/very high (fissile and other threats w/NaI)
2	Delayed gamma rays (γ_d & $\gamma_d(t)$)	All $< 10\text{MeV}$ neutrons inducing fission	$E_n < 10\text{ MeV}$ e.g., (d,D), time switchable (e.g., 5s on/5s off)	Gross delayed γ rays (amplitude & time dependence)	PS or LS, (CsI, LaBr ₃)	Medium high/high (fissionable)
3	Direct prompt fast neutrons (n_p)	Neutrons with $E_n < 3\text{MeV}$	(d,D)	n_p w/ $E_n > 3\text{MeV}$	TAD, LS + PSD	High/medium (fissionable)
4	Delayed neutrons (n_d & $n_d(t)$)	All neutrons inducing fission	Any switchable source (e.g. 5s on/5s off) such as shuffling ^{252}Cf , (d,T) or (d,D) ENG	Delayed neutrons $n_d(t)$ (amplitude & time dependence)	Moderated ^3He or similar detectors based on ^{10}B or ^6Li	Very high/very low (fissionable)
5	Time correlation techniques (FMD, Rossi α, Feynman, etc.)	Slowed down neutrons and/or neutrons with $E_n < 3\text{MeV}$	CW (d,D)	n_p , γ_p	PS, LS + PSD, Moderated neutron detectors (^3He & or similar det.)	Medium high/low (fissionable)
6	Pulsed Fast Neutron Analysis (PFNA)	Monoenergetic fast neutrons in the energy range of 7–9 MeV	ns pulsed (d,D) in frequency range of 1kHz to 1MHz rep rate	$(n, n'\gamma)$ & γ_p from (n,f)	PS or LS with very fast timing	High /high provides 3D localization for fissionable and other threats such as explosives
7	API (Associated Particle Imaging)	14 MeV-n in coincidence w/ the associated α particles	CW (d,T) + embedded α detector	$(n, n'\gamma)$ & γ_p from (n_{14}, f)	NaI (LaBr ₃)	Medium high/low provides limited 3D localization for fissionable and other threats as explosives

ENG, electronic neutron generator; CW, continuous wave (not pulsed); DF, duty factor; PS, plastic scintillator detector; LS, liquid scintillator detector; PSD, pulse shape discriminator; TAD, threshold activation detector; FMD, fission multiplicity detection; Rossi and Feynman are data analysis methods to derive the system's neutron multiplication.

The table lists the main techniques that are based on neutron interrogation. Key features of each technique are provided, including information on the neutron source, the main fission signature or signatures used, the means of detection employed, and two qualitative attributes of the technology: “Specificity” and “Sensitivity”. Specificity relates to the uniqueness of the measured signals to fissile or fissionable materials. For example, delayed neutrons have very high (possibly the highest) specificity, as no other materials in nature besides the fissionable ones (with a very few and rare exceptions) emit this type of delayed radiation. Sensitivity relates to the strength of the specific signature or signal and its other attributes (such as energy) that affects their penetrability through cargo. For example, the number of delayed neutrons emitted per fission dictates the magnitude of the measurable fission signature. This, in turn, determines the number and size of the detectors required to measure it within operational constraints.

An effective system is one that has high specificity and sensitivity. When quantified, the product of the two may define the system’s “effectiveness.” The table lists the technologies (#1–#7) in the rough order of their “effectiveness” from high to low. In actual AI systems, multiple complementary techniques are employed together to enhance the robustness of the overall interrogation system.

A brief synopsis of active Interrogation techniques available or being developed is provided below.

Differential die-away analysis (DDAA)

DDAA is a method to detect prompt neutrons emitted during the fission process induced by thermal neutrons only in fissile materials, such as U-235, Pu-239 (Caldwell et al., 1986).

Thermal neutrons are created by the slowing down and thermalization of a pulse of fast neutrons (i.e. from an electronic neutron generator). These processes occur rapidly within a few microseconds. Once the neutrons have thermalized, their population can remain for a long time period of a fraction of millisecond to a few milliseconds, depending on the moderating and absorption properties of the medium. These processes occur in the moderating material surrounding the source (by design) and/or inside the inspected object if it contains hydrogenous substances (e.g., cargo composed, among others, of organic materials). A fast neutron detector with a fast response time will see no signal after a short time of a few microseconds if no fissile material is present. A slower fast neutron detector, such as a He-3 detector covered with cadmium (to capture thermal neutrons), having intrinsic “die away” of 20–40 μ s, would not see any neutrons after about four die-away times (from 80 to 160 μ s). The thermal neutron population in the moderating material, however, can last for relatively long time and decays with time constants of the order of a few 100 μ s to a couple of milliseconds. If fissile material is present, fissions will be induced and fast prompt fission neutrons will appear and be detected by the DDAA neutron detector. The detection of this relatively slow decaying fast neutron population is an unequivocal indication to the presence of fissile material in the inspected object.

The DDAA technique is a very sensitive SNM detection technique because of the very high fission cross section of thermal neutrons in fissile materials and the high efficiency with which these neutrons can be detected.

There are variants of the DDAA technique (Jordan and Gozani, 2007) that have been adapted for specific inspection scenarios, such as the inspection of large bulky objects like 20/40-ft (6.1/12.2 m) shipping containers used in maritime and land transport. Another variant is the DTOF (Gozani et al., 2013) technique that is the manifestation of the DDAA technique when the inspection object is at a significant standoff distance (e.g., > 0.5 m) from the source. In this case, the thermal (and epithermal) neutron time-of-flight from the source to the object is to be considered in the detection process. Recently the DDAA technique was extended to demonstrate the feasibility to detect SNM with more penetrating epithermal and even sub-MeV neutron fraction of the 14 MeV slowing down spectrum (Gozani and King, 2016, 2017). However, these require an intense narrow-pulse (< 100 ns) dT neutron generators with high frequency pulsing, which are not available commercially.

Several different systems based on thermal neutron inspection have been built. An example is shown in Fig. 8. The system in the figure was designed to detect explosives by looking mainly for a high energy gamma-ray from the de-excitation of nitrogen (10.8 MeV). When cadmium covered banks of moderated He³ (or other thermal detectors) were added (see Fig. 9) highly sensitive DDAA capability was added to detect SNM. Furthermore, a large area thermal neutron detector array added above the inspected conveyance increases significantly the detection sensitivity of delayed neutrons. Typical raw signals from such a system are shown in Fig. 10.

Delayed gamma-ray based inspection system (Stevenson et al., 2011)

This technique detects the presence of fissionable material through the detection of the gross gamma rays emitted by the fission products from a fraction of a second to many seconds after the fission event.

The fission processes (induced by either neutrons or a high energy photons) in all fissionable isotopes generate a copious number of gamma rays. Though the majority have low energies, a useful fraction of a few percent is above 3 MeV. The decay times of the delayed gamma rays cover a wide range: from milliseconds to many seconds. The practical time range is from a few tenths to tens of seconds. The optimal irradiation/counting sequence is typically 5–10 s irradiation (i.e. source “ON”) and a similar time for counting (source “OFF”). In pulsed mode the delayed gamma rays appear conveniently as constant background between pulses following the complete decay of the capture gamma rays in a few milliseconds after the end of the neutron pulse.

Delayed gamma detection techniques can be employed with both neutron and high energy X-ray sources. The sources can be inherently pulsed (like linac or pulsed ENG) or continuous. To detect the delayed gamma-rays, the source (if not pulsed) is sequentially turn on for a few seconds and then off for a similar time. When the source is off, the detector acquisition is turned on to detect

Track mounted dual sided VEDS system inspecting a truck

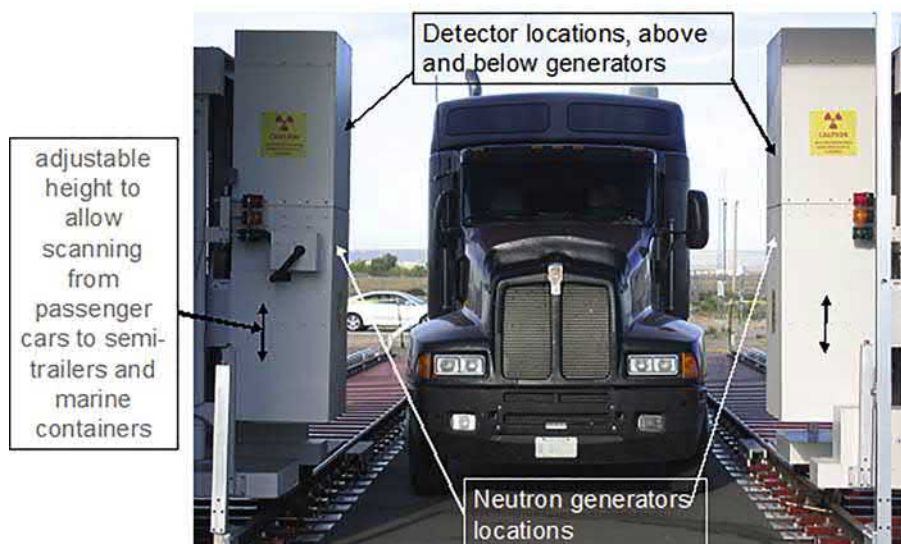


Fig. 8 A gantry active neutron interrogation system for inspecting trucks designed and built as VEDS (Vehicular Explosives Detection System) by Ancore Corporation.

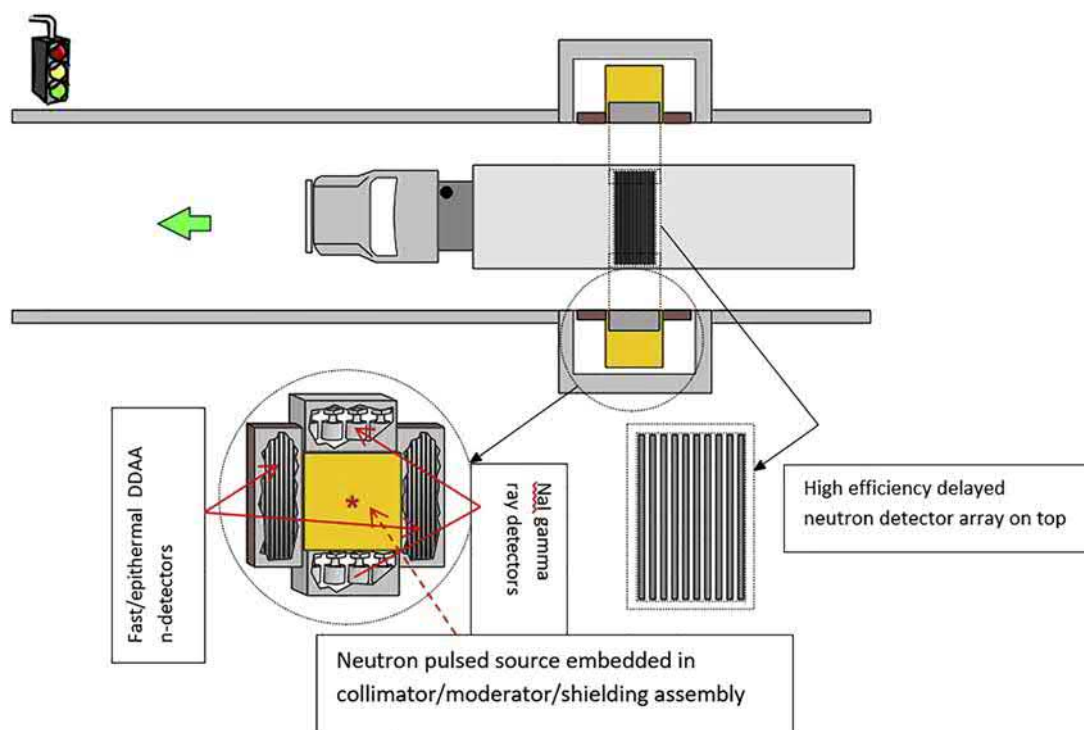


Fig. 9 Combined Explosive and Nuclear Detection system (Dual VEDS/SNMDS). A basic platform for all key neutron-based inspection techniques. The truck is being inspected simultaneously from both sides.

the delayed gamma rays. This process is repeated until sufficient statistical precision of the signal is achieved. The detector does not require having a particularly high energy resolution, as the gross delayed gamma spectrum is basically “featureless” that exponentially declines with energy; with an average energy of about 1 MeV (see Fig. 11).

The signal to background ratio for the high energy part of the spectrum, above most of the natural gamma-ray background (>2.6 MeV), is a strong indication for the presence of a fissionable material (hence, “good specificity”). The intensity of the signal is generally high when employing reasonably strong sources and large area efficient gamma ray detectors (hence, “high sensitivity”).

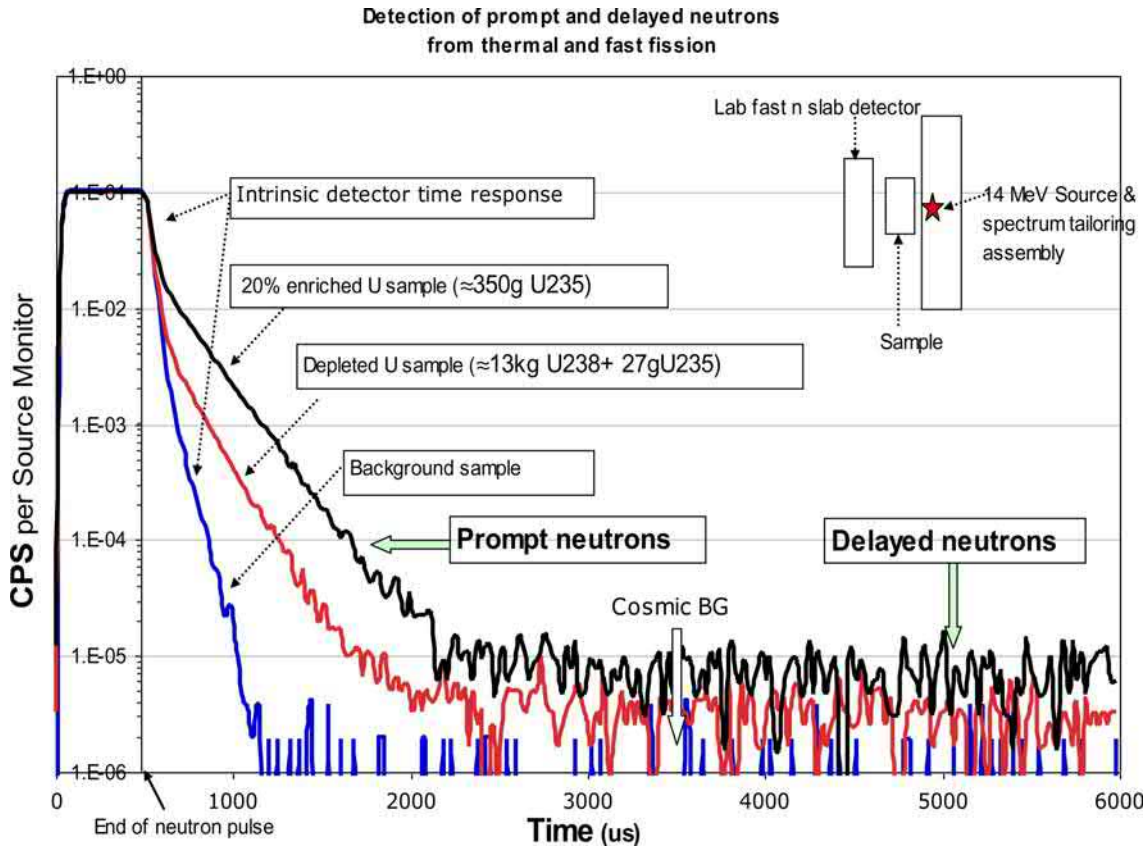


Fig. 10 Basic response of DDAA detector to fissile & fissionable material. Please note that the depleted uranium DDAA response is from the residual U-235. The delayed neutron response however is due to thermal fission of the residual U-235 and fast fission in U-238.

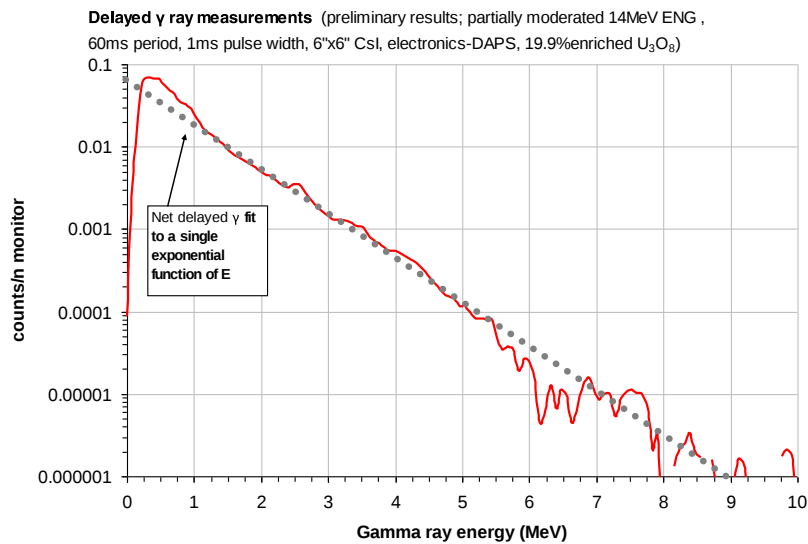


Fig. 11 Measured gross delayed gamma-ray spectrum.

The dominant background for the gamma ray signal is very different from that in the neutron detection modality and is much higher. It consists of thermal neutron-capture gamma rays (which, for example, enable explosive detection), which decays with the time constant of the thermal neutron flux (hundreds of μ s to ms) after the source pulse. The second background source is due to the possibility of neutron activation of any cargo present or of structural materials. This is the reason that it is difficult to use 14 MeV neutrons as an interrogation source if the delayed gamma rays are to be employed as a fission signature (Slaughter et al., 2007). The

14 MeV neutrons activate O-16 atoms via the (n, p) reaction with a threshold at 10.2 MeV. The reaction product, N-16, decays back to O-16 by emitting a β ray and a 6.1 MeV photon, with a half-life of 7.1 s. This activation can constitute a significant background, since oxygen-containing materials are very common in cargos. A third background is the background due to naturally occurring radioactive materials (NORM), which is significant at or below 2.6 MeV, emanating from the natural radioactivity of the soil and the environment in general. A fourth source of background, for this measurement, is passive radiation from the LEU sample itself. Thus, a commercial d-D ENG is the recommended option when a delayed gamma ray signature is crucial. No significant extraneous cargo activation occurs since the d-D neutrons are limited to energy of about 2.5 MeV. Unfortunately, the available neutron yield for d-D ENGs is still relatively low of the order of 10^8 n/s. More powerful generators for 14 MeV and lower energy (Slaughter et al., 2007; Phoenix, 2020) are larger and more expensive. Delayed fission gamma rays are arguably the most effective fission signatures in a photofission-based inspection system employing low neutron background electron linacs (see next section).

Direct prompt fast neutrons

This technique measures directly the high energy tail of the prompt neutron spectrum emitted during the fission process. Any penetrating particles (e.g., neutrons or high energy X-rays) that cause fission can be used as an interrogation source. Fast neutron detectors, which are able to discriminate between the numerous source particles and the higher energy fission neutrons and can be effectively shielded to cope with the inevitable severe “source flash,” are employed in this technique.

On average, about three fast neutrons/event are emitted during fissions induced by thermal neutrons. This number increases slightly as the energy of the neutrons increases. Thus, detection of fast fission neutrons during the fission process is a strong positive indication for the presence of a fissionable isotope (hence, it has “high specificity”). The discrimination between the two types of neutrons (i.e. between the fission neutron signal and the numerous background neutrons and X-ray photons), while retaining a good detection efficiency and ability to function during the “source flash,” are major challenges in this direct prompt fission neutron detection method. The energy of the source neutrons must be lower than at least that part of the high energy tail of the neutron fission spectrum one wishes to detect. The d-D ENG emitting neutrons with $E < 2.8$ MeV (highest energy at zero degree) is the highest source energy that is typically employed. The fission spectrum above 3 MeV still contains about 25% of all fission neutrons. X-rays with energy below 9 MeV produce in cargo mostly lower than 3 MeV background neutrons, via photoneutron interactions—mainly in the deuterium, which is always present in hydrogenous substances. These high energy X-rays are therefore very useful for interrogation for fissionable materials (see next section). The neutron detectors must be able to discriminate between the numerous < 3 MeV source neutrons from the relatively few > 3 MeV fission neutrons signature with a good detection efficiency. Hydrogenous liquid scintillators (“LS”) with good pulse shape discrimination (PSD) capabilities are used to directly detect the high energy fission neutrons by discriminating them from the lower energy (d,D) source neutrons or the low energy (γ ,n) neutrons generated by the X-ray source used to stimulate fission. The energy discrimination process though considerably reduces the effective detection efficiency of the LS.

An altogether different detector type, called Threshold Activation Detector (TAD) (Gozani et al., 2011b), is entirely insensitive to events during the neutron or X-ray source pulses (or “flashes”). Rather it is activated by the high energy fission neutrons rendering the scintillation material itself slightly radioactive, emitting beta (and some gamma) particles that are very efficiently detected milliseconds to seconds after the source pulse. This activation (with a few seconds life-time) is measured well after the fission event, where the most of the interfering background radiation does not exist anymore.

Delayed neutrons

This technique detects the presence of fissionable material through the detection of neutrons emitted by the fission products from a fraction of a second to several seconds after the fission event occurs.

This signature is very unique to the fission process (“high specificity”) and is relatively easy to detect, as the detection is done well after the fission process has occurred and the neutron or the X-ray source stimulating the fissions is shut off. Large area thermalized ^3He or other similar thermal neutron detectors are being used (see also Fig. 9). The delayed neutrons can be measured between pulses of a pulsed source (e.g., pulsed d-T ENG or linac) or after short irradiation time, when the source is off. The main drawback of the delayed neutrons is their low energy and especially their low intensity. Thus, it is characterized as a “low sensitivity” signal.

Time correlation/multiplicity techniques

Time correlation techniques include a family of techniques based on the high multiplicity of the prompt particles (0–10 prompt neutrons, with an average of 3 and/or 0–20 prompt gamma rays, with an average of 7) emitted during the fission process. The multiplicity is measured by detecting prompt and/or delayed coincidences (depending on type of detector employed) between detectors. The overall fission multiplicity detection efficiency is inherently low because it is the product of the efficiencies of the individual detectors contributing to the coincidence detection (e.g., the single detector efficiency to the power of the multiplicity level sought). These techniques have been used in applications for relatively small objects, which can be practically enveloped with efficient detectors (Verbeke et al., 2007; Gozani, 1976, 1981; Boldeman and Hines, 1985; Hausladen et al., 2009).

Pulsed fast neutron activation (PFNA)

Pulsed Fast Neutron Analysis (PFNA) (Brown and Gozani, 1995; Rynes et al., 1999; Hellborg, 2005; Gozani and Strellis, 2007) is a technique that uses a collimated pulsed beam of fast neutrons (typical to 8 MeV) to excite the nuclei of common elements in explosive and other threats (C, N, O, and others) (Stevenson, 2006). Direct imaging of the elemental contents of the material is accomplished by using time-of-flight analysis to identify the position of the interactions and gamma-ray spectroscopy to identify the elemental gamma-rays. From the ratios and absolute measurements of elemental abundances, the identification of the material can be deduced. Even a small volume of explosives can be detected as they are characterized by a narrow range of (C, N, O) elemental ratios.

While PFNA might be the ultimate and most versatile cargo threat inspection system, it is considered the most expensive and complex. It requires intense nanosecond pulses of deuterons at 4–6 MeV to impinge on a deuterium gas target producing pulses of monoenergetic neutrons. A unique high-speed data acquisition system digitizes and analyzes the time-energy data in real time. A wide range of threats can be detected and located within the inspected volume (be it marine shipping container, air cargo containers, etc.). Fissionable materials can be detected and located, like the other threats, by the TOF of the monoenergetic source neutrons (typical 8 MeV) to the point of interaction plus the very short flight time of the prompt fission gamma rays.

The PFNA requires a relatively large accelerator (e.g., Van de Graaff) capable of intense nanosecond wide pulses with a MHz repetition rate, the ability of beam line and target to raster-scan a container in the vertically direction, a large array of efficient gamma detectors (such as NaI) and all associated electronics, and very fast data processing.

Associated particle technique (API) (Beyerle et al., 1990; Pino et al., 2018)

The API technique is similar to other techniques utilizing a neutron generator neutron source; however, it adds an alpha particle detector to measure the associated particle emitted simultaneously and 180 degrees from the neutron in the neutron producing nuclear reaction. For the d-T source, this is an alpha particle. For the d-D source, it is a He-3 particle. The direction of the neutron is determined by measuring the location of the detected associated particle. TOF is used to get location of the nuclear reaction. The useable intensity of the neutron source with this technique is inherently limited by the alpha detectors time resolution and cross talk between them. The fission signal that can be detected is the fission multiplicity beset by the same low efficiency mentioned in the Time Correlation techniques above.

Cargo activation

It is a well understood and expected that neutron interactions can create isotopes that are radioactive and create ionizing radiation as a result. This phenomenon is called activation. Neutrons (thermal to 14 MeV in energy) can activate cargo material and potentially impact the utility of the inspected cargo.

Extensive cargo activation studies have been conducted over the years and have shown that even with a strong neutron source ($> 10^{11}$ n/s of 8 MeV neutrons employed in PFNA) and expected scanning speeds there is no significant activation that will prevent consuming inspected food (including medications) or create external radiation hazards of any significance.

High energy (< 10 MeV) X-ray based inspection systems induced very little activation products, these being mostly created from secondary (γ, n) neutrons.

Photofission-based techniques (Stevenson et al., 2011; King et al., 2011a; Danagoulian et al., 2010)

As described briefly earlier in this chapter, photofission is a nuclear reaction whereby a high-energy photon induces the fission process in an element. The probability that a photon initiates a fission event is defined by the cross section. The cross section for photofission is energy-dependent, ranging from about 0.001 b at 6 MeV to about 0.1 b at 10 MeV. To put this into perspective, at the higher photon energies the cross section for U-235 is about one order of magnitude smaller than for fission spectrum neutrons and 3–4 orders of magnitude smaller than for thermal neutrons. Despite this deficiency, for practical applications, photofission does have an advantage over neutron-based active interrogation techniques. For what it lacks in available cross section, it gains in radiation intensity. Due mainly to other industrial uses (e.g., medicine, non-destructive testing), X-ray sources are well designed and tested across a wide energy range from tens of kV to tens of MV. The most widely-used X-ray source for security applications capable of screening cargo is the electron linear accelerators (linac) with electron energies at or below 10 MeV (currently limited by regulations). Commercial 9-MV linac X-ray sources can produce a total X-ray intensity of 10^{15} X-rays/s/100 micro-amps. Compared to commercial 14 MeV neutron sources, this is 5–7 orders of magnitude higher. Not only do these devices provide excellent sources of strong X-ray beams for radiography that are highly penetrating into dense cargos, they provide a radiation source capable of initiating fission via photofission process. Systems utilizing a high-energy linac source and radiography detectors are commonly deployed today for cargo screening in trucks, sea containers, and rail cargo.

A relatively straightforward method to add nuclear material detection to a standard high-energy radiography system would be to add a subsystem capable of measuring the photofission signatures.

Several techniques can be used for nuclear material detection through photofission using commercial linac X-ray sources as outlined in **Table 7**. Taking the first row, one could employ large plastic or liquid scintillators (Eljen, 2020) (or inorganic scintillators like NaI (Saint-Gobain, 2020)) to detect the total number, energy, and detection time of photons emitted after passing through the linac X-ray beam. The majority of delayed gamma-rays emitted after photofission are emitted 10s to 100 s of seconds after fission. This characteristic allows the detector subsystem to be separated in time from the linac X-ray source so that the emitted delayed gamma-rays can be measured in a reduced radiation background field. On average, about six delayed gamma-rays are emitted per fission, so the signature is strong. This method is specific to fissionable and fissile materials.

Similarly, following the second row, one could employ large moderated thermal neutron detectors (e.g., He-3 detectors) to detect the delayed neutrons from the photofission process. To reduce the radiation background from neutrons and electronic noise caused by the X-ray flux, the detectors would need to be sufficiently well shielded from the linac. A majority of the delayed neutrons emitted after U-235 photofission are emitted on the seconds to tens of seconds times scale. On average, only 0.012 delayed neutrons are emitted per fission. The signature is considered very weak, but very specific to fissionable and fissile material.

Alternatively, following the third row, one could deploy a detector that can differentiate between the highly ubiquitous photons and the fission signature neutrons. One can use a technique called pulse shape discrimination (PSD) (Zaitseva et al., 2012) with liquid scintillator detectors and solid organic scintillators to separate photons from neutrons. PSD works on the principle that neutrons and photons interact with protons and electrons, respectively, in the detector material, causing different time decay behavior in the detector's scintillation light. Another detection method for prompt neutrons is using a threshold activation detector (TAD). A TAD detects neutrons above a specific energy by measuring the beta activation induced by the neutron in the detector material itself (Gozani et al., 2011b). These detectors also need to be well shielded from source and scattered photons and neutrons. This technique is highly specific and has a good sensitivity for fissionable material.

In addition to the more straightforward methods outlined above, a linac X-ray source can be converted to a dual species neutron/photon source through the photoneutron reaction (γ, n) (King et al., 2011b; Gozani et al., 2010, 2011a). Practically, this technique requires the installation of a neutron converter near the X-ray target using a material with a photoneutron reaction threshold energy below the X-ray source energy. A likely candidate is heavy water that contains deuterium (^2H) or beryllium. When both neutrons and photons are present for active interrogation, one can utilize both photofission and neutron fission to create fission signatures. The additional benefit of a dual species source is that they are complementary in terms of cargo attenuation. Where photons have difficulties in penetrating metals, neutrons do not. Where neutrons have challenges in hydrogenous materials, photons do not. Thus, most cargo types can be penetrated by at least one or the other.

One challenge with photofission-based systems is the relatively strong photoneutron production by the (γ, n) reactions (an energetic photon ejecting a neutron from the nucleus it hits) with material around the X-ray source (e.g., converter, shielding and collimators) and in the surrounding environment (e.g., cargo and the air). Material containing isotopes with lower (γ, n) thresholds than the end point energy of the X-ray source (such as deuterium, which is always present in natural hydrogen), cause undesirable radiation background for measuring the fission signatures. The most important (γ, n) thresholds relevant for a 9 MV linac X-ray source are listed in **Table 8** for structural materials and key cargo isotopes and in **Table 9** for the commonly-employed linac converter and shielding materials.

The photoneutron intensity rapidly increases, along with the neutron energy, as the end point X-ray energy increases beyond the 8–9 MeV range. These neutrons, especially those with higher energy, constitute a significant background for the prompt neutron detectors. For deploying this technique to the field, a significant reduction of this background is highly desirable by properly considering the electron/X-ray converter, structural material, collimation, shielding and its system design. Linacs capable of operating with wider pulse width (commercial linacs operate with pulse width of $< 1 \mu\text{s}$), and with a significantly lower (γ, n) neutron background, are being currently developed (Tantawi, 2015; Condrón et al., 2013).

Table 7 Methods and signature for detecting fissile and fissionable material through photofission.

Technique	X-ray source/electron energy (MeV)	Signature	Detector configuration	Specificity/sensitivity (applicable to)
Photofission Delayed Gamma Detection	Linac, CW (or high DF)/ ≤ 10 MeV	Measure gross delayed γ rays (amplitude and time)	NaI, organic scintillators (PS, LS)	Medium-high/high (fissionable/fissile)
Photofission Delayed Neutron Detection	Linac, CW (high DF)/ ≤ 15 MeV	Measure delayed neutrons (amplitude and time)	Moderated ^3He or similar detectors	Very high/very low (fissionable/fissile)
Photofission Prompt Fast Neutron Detection (PFNP)	Linac, CW/ ≤ 9 MeV	Measure prompt neutrons above an Energy threshold	Well shielded LS (+PSD), TAD	High/medium high (fissionable)
Differential Die Away	linac with (γ, n) converter/ ≥ 6 MeV	Measure prompt neutrons by time and delayed neutrons (secondarily)	Moderated n det. (^3He or similar) with $\tau < 50 \mu\text{s}$	very high/ medium-high (fissile only)
Nuclear Resonance Fluorescence (NRF)	Variable energy intense CW electron accelerator (Rhodotron)	Resonant isotopic gamma lines	Large array of well collimated Ge detectors	very high/very low (fissionable/fissile + other elements)

Table 8 Photoneutron production threshold in cargo materials, relevant for 9 MeV X-rays.

<i>Isotope</i>	<i>Natural abundance (%)</i>	<i>(γ, n) threshold (MeV)</i>	<i>Where present</i>
H2	0.015	2.225	Cargo
C13	1.11	4.947	Cargo
N14	99.634	10.559	Cargo
N15	0.366	10.838	Cargo
O16	99.762	15.672	Cargo
O17	0.038	4.144	Cargo
O18	0.2	8.046	Cargo
Fe54	5.845	13.38	Cargo and structural
Fe56	91.754	11.199	Cargo and structural
Fe57	2.119	7.647	Cargo and structural
Fe58	0.282	10.046	Cargo and structural

Table 9 Photoneutron production threshold in X-ray converter and nuclear materials, relevant for 9 MeV endpoint X-ray interrogation.

<i>Isotope</i>	<i>Natural abundance (%)</i>	<i>(γ, n) threshold (MeV)</i>	<i>Where present</i>
Ta181	100	7.577	X-ray converter
W180	0.12	8.412	X-ray converter
W182	26.5	8.065	X-ray converter
W183	14.31	6.19	X-ray converter
W184	30.64	7.411	X-ray converter
W186	28.43	7.191	X-ray converter
Au197	100	8.072	X-ray converter
Pb204	1.4	8.394	X-ray shielding
Pb206	24.1	8.086	X-ray shielding
Pb207	22.1	6.738	X-ray shielding
Pb208	52.4	7.367	X-ray shielding
U235	0.72	5.297	SNM
U238	99.275	6.154	Fissionable
Th232	100	6.44	Fissionable
Pu239	100	5.646	SNM

Photofission is a promising, albeit challenging, method for expanding the capability of currently deployed systems to detect nuclear material. The method does involve adding additional radiation detectors to measure the neutron or gamma-ray signatures from fission. Also, supporting tools like software and detection algorithms will require modification. However, the underlying methods for inspection and the X-ray source technology in many cases would not require modifications, except preferentially utilizing sources at the higher range of energy. For these reasons, photofission may be the most plausible method to deploy to real operational environments where nuclear material detection is considered a priority. A couple prototype systems utilizing photofission have been developed and have shown very good performance across a range of cargoes and nuclear materials employing the delayed gamma-ray signature. One system (Stevenson et al., 2011) utilizing a commercial 9-MV linac and the photofission delayed gamma-ray and prompt neutron detection methods is schematically shown in Fig. 12. This figure shows an X-ray radiography system mounted on a gantry that travels on rails to scan ISO 40-ft containers. Fission detectors (in blue) in the form of large plastic scintillators and TADs were mounted to the radiography detector array to measure the delayed gamma-rays and prompt neutrons from fission. Another successful manifestation of photofission-based inspection is the PNFP (Danagouliau et al., 2010), that is based on a powerful CW Rhodotron (JME, 2020) circular electron accelerator and multiple well shielded LS with PSD.

Nuclear resonance fluorescence (Bertozzi and Ledoux, 2005; Bertozzi et al., 2007)

NRF is not a photofission-based technique but it uses the ability of nuclei, (e.g. ^{14}N) to resonantly scatter/absorb gamma rays of specific and precisely defined energy (^{14}N : 9.17 MeV). For fissionable material detection, the resonantly scattered gamma rays are measured by collimated high resolution detectors (such as high purity germanium). The gamma rays originate from the different nuclear levels of fissionable isotope present in a voxel. The 3D voxels are defined by the volumetric intersection of the collimated X-ray beam and the subtended solid angle of collimated germanium gamma ray detectors. The gamma ray energies detected are very specific to each isotope. Thus, NRF is characterized as a technique with very high specificity. However, the technique has very low sensitivity because the number of photons that can be resonantly scattered is small even in an intense X-ray beam, and the overall germanium detector efficiency is very low due to the small solid angle the voxel subtends at the detector surface. Additionally, the specific gamma rays of the fissionable material are relatively low, 2–5 MeV making them more vulnerable to attenuation in the cargo material.

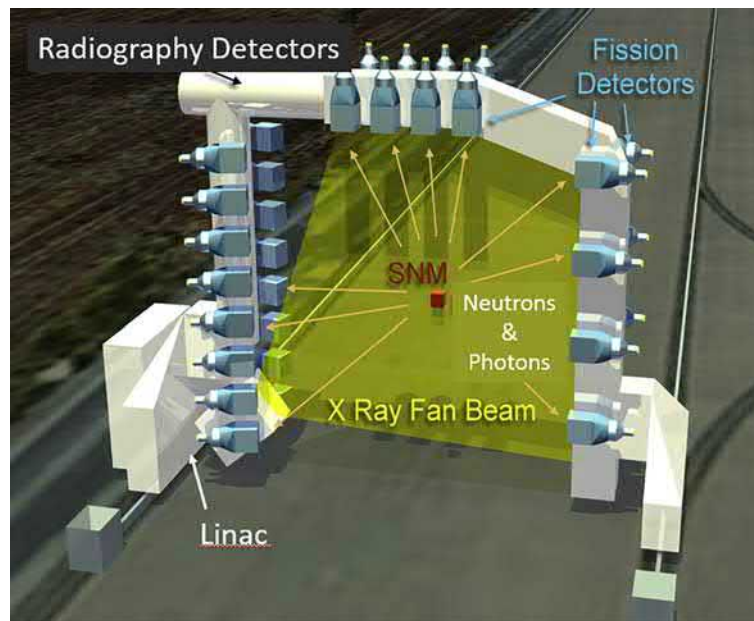


Fig. 12 A rendering of a photofission based active interrogation system that integrated a high-energy X-ray radiography system with fission signature detectors.

Conclusions

Active interrogation technologies employing, as appropriate, neutrons or high energy X-rays fill an important gap to detect nuclear and other threats. They can be combined with the existing technologies or stand alone. The research and development efforts over the last few decades clearly show that reliable active non-intrusive inspection of large objects (e.g., trucks, marine shipping containers, air cargo containers and smaller objects laden with commercial cargo) for nuclear material is an emerging technology. On-going research and development into electron linacs, detector materials, signal processing methods, and detection algorithms further increase the possibility of field deployment of these vital technologies as acquisition and operational costs are reduced.

See Also: Modern Industry: Application of Neutrons for Materials Testing and Inspection; Modern Industry: Gamma Methods for Materials Testing and Inspection; Modern Industry: Neutron Basic Interactions, Sources and Detectors for Materials Testing and Inspection.

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Public Safety: High Radiation Sources and Alternative Technologies

Valeriia N. Starovoitova, Radioisotope Products and Radiation Technology Section, Division of Physical and Chemical Sciences, Department of Nuclear Sciences and Applications, International Atomic Energy Agency, Vienna International Centre, Vienna, Austria

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Nomenclature

Bq Becquerel (SI unit of radioactivity, 1 Bq = 1 decay/s)
Ci Curie (unit of radioactivity, 1 Ci = 3.7×10^{10} Bq)
DC Direct current
Dose (or absorbed dose) Measure of energy deposited by ionizing radiation in matter
DUR Dose uniformity ratio
Gy Gray (derived SI unit of dose, 1 Gy = 1 J/kg)
IAEA International Atomic Energy Agency
ITDB IAEA Incident and Trafficking Database
NDT Nondestructive Testing
NNSA National Nuclear Security Administration
ORS Office of Radiological Security
RAM Radioactive Material
RDD Radiological Dispersal Device
RF Radiofrequency
RT Radiotherapy
SIT Sterile Insect Technique
TA-GVHD Transfusion-Associated Graft-Versus-Host Disease
UN United Nations
US DOE United States Department of Energy
Z Atomic number

Introduction: Uses of high activity sources and their benefits

Applications of radioisotopes have come a long way since their first practical use over a century ago. Today hundreds of radioisotopes, both natural and man-made, are routinely employed in medical, industrial, and research fields. The development and use of sealed radioactive sources have played a significant role in improving our quality of life as it touches almost every aspect of the modern society.

One of the most common applications of radioactive sources is radiotherapy (RT)—a widely used form of cancer treatment (Baskar et al., 2012). Radiation, used alone or in combination with surgery or chemotherapy, effectively destroys cancerous cells. External radiotherapy, also known as teletherapy, is provided by an external radiation source, usually cobalt-60, which emits 1.1 and 1.3 MeV gamma rays. Brachytherapy, or internal radiotherapy, places radioactive sources inside the patient. This technique

has the advantage of using a highly localized dose of radiation and significantly decreases the risk of radiation-induced second malignancies. Radioactive implants are inserted for hours, days, or, in some cases, even permanently. Currently, brachytherapy is predominantly used for localized tumours, such as cancers of the prostate, cervix, and endometrium. Mostly alpha- or beta-emitting radionuclides are used for brachytherapy due to their short penetration into biological tissues and high linear energy transfer. Among them are iodine-131, iridium-192, palladium-103, cesium-131, and ruthenium-106.

Another medical application of high activity sources is irradiation of blood before it is given to at-risk patients in blood transfusions. The irradiation is aimed at preventing transfusion associated graft-versus-host disease (TA-GVHD), a rare but nearly always fatal complication, in which transfused white blood cells attack the recipient's tissue (Moroff and Lunab, 1997). This process is commonly done using cesium-137 irradiators, and hundreds of them can be found worldwide. As will be discussed later, currently these sources are being replaced by non-radioisotopic irradiators in many countries.

Ionizing radiation is also used to sterilize healthcare products, including medical devices and pharmaceuticals (https://www-pub.iaea.org/MTCD/publications/PDF/Pub1313_web.pdf). Many single-use medical devices are supplied to the user as sterile products. Gamma irradiation, unlike other sterilization techniques, can penetrate products while sealed in their final packaging, which simplifies the manufacturing and distribution process. Implantable devices (orthopaedics, stents, etc.) as well as tissue-based devices coming from human donors, are also commonly sterilized with cobalt-60 to increase patient safety. Given the huge volume of devices which need to be sterilized, this processing is typically done in large commercial gamma facilities containing MCi (tens of PBq) quantities of cobalt-60.

High activity sources are also used for food safety and preservation and phytosanitary applications (<https://www-pub.iaea.org/MTCD/Publications/PDF/trs481web-98290059.pdf>). In the current global economy food is often consumed far from where it was produced. To extend its shelf-life during transport and storage, it needs to be preserved, which can be done by a variety of processes. In addition, transported produce needs to be free of pests that could disrupt the environment or agricultural regions where it is received. Ionizing radiation destroys pathogenic organisms such as parasites, bacteria, and viruses, and sterilizes insect pests. In many countries, food irradiation with cobalt-60 is routinely used to prevent sprouting or rotting, to delay ripening, and to reduce the risk of foodborne illness. Sterile Insect Technique (SIT) is another common application of gamma sources. *This insect* control method involves radiation sterilization of target *pests*, followed by the systematic area-wide release of the *sterile* males to mate with wild females without producing offspring. SIT has proved to be successful by eradicating several insect pests in various regions worldwide.

For almost a century radioactive sources have been used in industrial radiography to detect defects not visible from the outside (Committee on Radiation Source Use and Replacement, 2008). Cracks, flaws, or voids can be found by measuring the variation in the intensity of the radiation passing through an object. Examination of pipelines, boilers, parts and assemblies of vehicles and aircrafts, and other products has become widespread due to the convenience of compact and reliable sources of radiation. Radioactive sources are also used to monitor the thickness of items being formed, to monitor the density of material flowing through piping, or to measure the level of liquids in opaque containers (Monno, 2004). All of the above can be done by measuring the attenuation of radiation as it passes through matter.

Sealed radioactive sources are also routinely used in well-logging. Both gamma and neutron emitters are essential for evaluation of oil and gas formations. Formation density is typically measured by lowering Cs-137 source into a borehole and analyzing its scattered gamma rays. Porosity of formation is usually found with Am-Be neutron sources. As the formation is irradiated with neutrons, they scatter and thermalize. Low energy neutrons are easily absorbed by hydrogen, and therefore the neutron counting rate is inversely related to the amount of hydrogen (water and hydrocarbons) in the formation. The combination of neutron and gamma logs usually gives a rather accurate prediction of the feasibility of extracting oil or gas from a borehole (Pike and Duey, 2002).

Finally, high activity sources are widely used for R&D purposes. Thousands of research irradiators are installed worldwide and they are used intermittently for very diverse applications, including chemistry, biology, and material science. For example, small animals are irradiated to conduct preclinical studies in cancer research. Radiation-induced mutations help develop high yielding and disease resistant varieties of economically important plants and crops. New radiation resistant materials are tested in research irradiators.

From this brief review one can see the variety of applications of high activity sources in our daily life. The preferred radioisotopes for sterilization, blood irradiation, and food irradiation have historically been cobalt-60 and cesium-137. However, other isotopes are commonly used for brachytherapy and for non-destructive testing. Table 1 summarizes the properties of the most commonly used radioisotope sources.

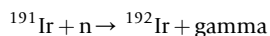
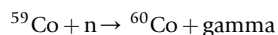
Radioisotope life-cycle: From cradle to grave

Numerous applications of radioactive sources, some of which are described above, require reliable supplies of radioisotopes. There are several approaches to isotope production, but the two main instruments required to make radioisotopes are particle accelerators and nuclear reactors (Starovoitova, 2017). While many radioisotopes are produced in accelerators, this method is mostly suitable for short lived (hours to days) isotopes, commonly used in radiopharmaceutical science and nuclear medicine. Longer lived radioisotopes, used for sealed sources, typically require long (months to years) irradiations. Therefore, most radioisotopes for radioactive sources are produced in nuclear reactors—both research and power. Neutron capture is the most common

Table 1 Most commonly used radioisotopes for radioactive sources, their properties and applications.

<i>Application</i>	<i>Radioisotope</i>	<i>Half-life</i>	
Industrial irradiators	Cobalt-60	5.3 years	Used in metallic form, which oxidizes in air, but is not soluble in water
	Cesium-137	30.1 years	Used typically in form of cesium salts (typically chloride), which are highly soluble in water. Sometimes used in ceramic form for weak sources
Industrial radiography	Cobalt-60	5.3 years	See above
	Iridium-192	74 days	Iridium is a noble metal, not oxidized in air, not dissolved in water
	Selenium-75	120 days	Used in form of thermally stable metal-selenide compound (typically with vanadium)
Gauges	Cobalt-60	5.3 years	See above
	Cesium-137	30.1 years	See above
	Californium-252	2.6 years	Californium is a highly reactive actinide and typically used in form of alloys or cermets
	Krypton-85	10.7 years	Gas
	Americium-241	432 years	Americium metal is not chemically stable, thus usually used as oxide powder in a capsule
Well-logging	Cesium-137	30.1 years	See above
	Americium-251/Beryllium	432 years	Used in form of mixed oxide powder or pellets
Research	Cobalt-60	5.3 years	See above
	Cesium-137	30.1 years	See above
	Americium-251/Beryllium	432 years	See above
Teletherapy	Cobalt-60	5.3 years	See above
Brachytherapy	Cesium-131	9.7 days	Cesium containing core in a thin aluminum jacket inside a Teflon capsule
	Cobalt-60	5.3 years	See above
	Iridium-192	74 days	See above
	Iodine-125	60 days	Often used as absorbed into silver matrices
	Palladium-103	17 days	Palladium doped polystyrene in titanium capsules
Blood irradiators	Cobalt-60	5.3 years	See above
	Cesium-137	30.1 years	See above

reactor-based production method due to high reaction cross section. Examples of radioisotopes produced by neutron capture include cobalt-60 and iridium-192:



Irradiated pellets are transferred to a manufacturing facility, where sealed sources are produced under strict quality control and quality assurance systems, such as ISO9001. Manufacturing the finished sources usually requires shielded hot cells with remote manipulators and specialized equipment. The finished products are normally double-encapsulated, welded stainless steel sources, identified with a unique serial number.

Radioactive sources (defined as Class 7 material in the UN Model Regulations on Dangerous Goods) need to be transported several times during their life-time. Multiple modes of transport are used including roads, railroads, air, and waterways for consignments. Often, country borders are crossed. Special packaging or casks are required to contain the radioactive content and control the external radiation levels. All of the above entails the stakeholders to adhere to various directives, laws, and regulations. The International Atomic Energy Agency (IAEA) has published international regulations for the transport of radioactive material since 1961, with the latest released in 2012 (https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1570_web.pdf). Many countries have adopted IAEA regulations into their national policies. Careful planning, effective communication and coordination is a must for safe and secure transport.

When the activity of the sources drops, they are no longer suitable for their intended purposes. Rarely, these sources still can be used for training or R&D at local universities and laboratories. However, most often, the sources are disposed of by being returned to the manufacturer or by storing them in specialized waste facilities. Unfortunately, many countries lack the appropriate disposal facilities. Therefore, management of radioactive wastes and the long-term safety and security of disused radioactive sources remains a concern at the international level.

Lost and stolen sources

Millions of radioactive sources have been produced worldwide, although the existing databases indicate a smaller number. For various reasons, a significant number of radioactive sources have become obsolete. At this stage, the sources are often not properly controlled and can be lost or stolen. Over the decades there have been a number of incidents and accidents. One of the most serious accidents, ranked as level 5 on the IAEA International Nuclear Event Scale (INES) (<https://www.iaea.org/publications/10508/ines-the-international-nuclear-and-radiological-event-scale-users-manual>), took place in 1983 in Goiania, Brazil.

A 51 TBq (1.4 kCi) cesium-137 source was stolen from an abandoned hospital building and sold to a local scrapyard. Shortly after, the capsule was broken and people were fascinated by the blue glow coming from cesium. Many family members and friends visited, and no one understood the nature nor the danger of the material even after people started feeling sick. Only days later radioactivity was detected, and a large-scale screening operation was initiated. By that time a large part of the city was contaminated as cesium was transferred by wind and rain. People were evacuated, houses were demolished, whole streets and squares decontaminated, and soil removed. Over 100,000 people were screened and more than 200 were found to be irradiated. Thousands got sick and four people died from radiation poisoning.

Another serious accident, INES level 4, took place in Myapuri, India, in 2010. Its story is similar to the one in Goiania: an abandoned cobalt-60 source from an old research irradiator was sold to a scrap yard. After immediate contact with the source, several people received a significant dose and one person died. Many other accidents and incidents involving abandoned radioactive sources, such as cobalt-60 and cesium-137, have transpired over the last several decades. Among them are events in Casablanca, Morocco; Mexico City, Mexico; Baku, Azerbaijan; Chiba, Japan; Los Angeles, USA; and many, many others (<http://www.irpa.net/irpa10/cdrom/00140.pdf>).

Often, during these accidents, people did not know what they were dealing with until they started feeling sick. However, there are several cases when radioactive material was stolen on purpose, with a clear malicious intent. For example, in 1999, six people attempted to steal cobalt-60 rods in the *Chechen Republic of Russian Federation*. After handling the rods, three of them died, and the other three were severely injured. It is possible that their intent was to use this radioactive material in their struggle against Russian federal troops, who at that time were re-establishing central government control over Chechnya. Moreover, many events have been reported when radioactive material was used in attempts, sometimes successful, to murder people (<http://www.johnstonsarchive.net/nuclear/radevents/LAUR073686.pdf>).

The IAEA Incident and Trafficking Database (ITDB), launched in 1993 and updated annually, contains information on voluntarily reported incidents involving radioactive materials (<https://www.iaea.org/sites/default/files/19/04/itdb-factsheet-2019.pdf>). While the database is confidential, the IAEA annually releases a summary without the details of individual accidents. Several hundred incidents are reported annually by participating countries and most of them involve industrial and medical radioactive sources. The majority of incidents are related to unauthorized disposal and shipment and the discovery of uncontrolled radioactive sources. Many cases involve detection of goods contaminated with radioactive material. The most common source of such contamination is feed metal, often obtained from the recycling industry. As the feed is melted down, it can become contaminated with undetected radioactive material such as cobalt-60. If the resulting contaminated metal is used for household goods, it could pose health problems to their consumers. Some incidents are related to missing or stolen material. Devices containing radioactive sources often look valuable to potential thieves. Most such devices use significant quantities of relatively long-lived isotopes such as cesium-137, cobalt-60, and iridium-192. Few incidents demonstrate intent to acquire or provide radioactive material for malicious use. Most of them are rather amateur or opportunistic in nature. However, several cases were reported to be better prepared and resourced.

Only some countries systematically report the incidents involving the loss of radioactive material to the IAEA ITDB. A more comprehensive database is annually prepared by the James Martin Center for Non-proliferation Studies (https://media.nti.org/documents/global_trafficking_2017.pdf) through exhaustive multi-language searches of media and government sources. According to their data, more than half of the incidents were at least partially caused by human error. Examples include material left unattended in the lab and lost, device not properly secured in a car, so it fell off, etc. About a quarter of incidents involve theft of radioactive material, and more than half of those took place when the material was in transit. Material was stolen both from the vehicles and with the vehicles. In most of the cases the vehicle was unattended at the time of theft and thieves were most likely unaware of what they had stolen. These results call for a stronger security culture in organizations that use, transport, and store radioactive materials.

Dirty bomb—Real danger

While the majority of incidents described above do not necessarily imply malicious use of radioactive sources, there have been attempts to buy and sell radioactive and nuclear material. While small quantities of cobalt or cesium cannot be used to make a nuclear bomb, they are capable of causing radiation poisoning and sickness. This makes high activity sources an attractive target for terrorists seeking to build a so-called “dirty” bomb. A *dirty bomb*, also known as a radiological dispersal device (RDD), combines radioactive material with a conventional explosive to spread it. A radiological weapon would not produce a nuclear explosion and would cause few (if any) immediate casualties. However, if detonated in a major city, an RDD could potentially contaminate a significant area and spread fear among population. Therefore, its *primary* destructive power would not be radiation, but *panic* and consequent financial losses. Analysis of the impact of a dirty bomb attack on the Port of Los Angeles and Long Beach predicts the extensive shutdown of the region’s operations and hundreds of billions of US dollars in losses (Rosoff and von Winterfeldt, 2007). Another study considers an RDD attack in the area of the New York Stock Exchange and estimates losses to reach \$1 trillion (Sudnik, 2011).

A dirty bomb has never been detonated. However this does not mean that it cannot or will not happen. Given the massive amount of existing radioactive sources, both used and disposed, many experts consider the eventual detonation of RDD to be highly likely. What can be done to reduce the risk and minimize the probability of such a frightening event? Obviously, more attention must be paid to the security of existing radioactive material. Numerous documents and guidelines exist to help. For example, IAEA’s

“Code of Conduct on the Safety and Security of Radioactive Sources” (<https://www.iaea.org/publications/7227/code-of-conduct-on-the-safety-and-security-of-radioactive-sources-guidance-on-the-import-and-export-of-radioactive-sources>) provides strategies for governments to establish the legal provisions necessary for the protection of high-activity sources. However, only two thirds of IAEA Member States endorsed the Code, and even fewer implemented all its recommendations. This means that many sources are still not safely and securely used, stored, transported, and disposed. In addition to taking proper care of the radioactive material already existing around us, can we minimize the number of sources? The answer is yes. Many benefits of radioactive sources without their risks can be provided through alternative technologies.

Alternative technologies

Many governments and international organizations, including the IAEA, are concerned about high activity source security and provide institutional support for the institutional support for the adoption and development of non-radioisotopic alternatives. The most common substitution for radioactive sources is X-ray tubes and electron accelerators, some of which can operate in e-beam mode and X-ray mode. The main advantages of these alternative technologies are their ability to be turned on and off for safety and their minimal security risks. These machines cannot produce residual radioactivity, and therefore do not result in radioactive waste. Historically, such accelerators were complex and rather expensive. However, lately, numerous models have been developed for various applications, and many of them are compact, versatile, and much more reliable.

One of the most widespread sources of X-rays is the X-ray tube—a simple tool, developed over a century ago, but still widely used in medicine, industry, and science. An X-ray tube consists of a cathode, which emits electrons. Accelerated by an electric field, the electrons bombard the metal anode. As the incident beam of electrons pass through the anode, they are primarily slowed down by the Coulomb field surrounding nuclei. The difference between the electron’s initial energy and its final energy is emitted in the form of photons, also known as bremsstrahlung radiation. The bremsstrahlung energy spectrum of photons is continuous, ranging from nearly zero up to the applied voltage. X-ray tubes are simple, compact, and robust. However, their energy output is limited to a few hundred keV, which is less than the energy of gamma rays emitted by cesium-137 and cobalt-60.

Electron accelerators also use electron guns (cathodes) to produce a collimated electron beam, which is accelerated with either DC voltage or RF cavities. Large machines can accelerate electrons to hundreds of GeV, however for industrial purposes, the energy of the electrons should not exceed 10 MeV and should be even less for some applications. This energy can be achieved with rather compact units. High energy electrons can be used “as is,” which is suitable for irradiation of low density or thin products. Alternatively, they can be converted into X-rays using a converter—a thin plate of high Z material, such as tungsten or tantalum. The resulting X-rays are highly penetrable, but the efficiency of the conversion process does not exceed more than a few percent. Fig. 1 depicts the bremsstrahlung radiation process (left) and the distribution of photon energies simulated for a 9-MeV electron beam impinging upon tungsten converter (right). It also shows lines corresponding to the energy of gammas emitted by cesium-137 (661 keV) and cobalt-60 (1.1 and 1.3 MeV) for comparison. The mean energy of bremsstrahlung photons can be roughly estimated as one third of the electron beam energy. Thus, a 9 MeV electron beam results in mean photon energy to be about 3, and a 4 MeV electron beam will be “gamma-energy-equivalent” to the cobalt-60 source.

Accelerators and X-ray tubes can potentially replace radioisotopic sources in many areas listed in Table 1. However, they have their own drawbacks, among them relative complexity of the technology, stable power supply requirements, and large unit size in some cases. As a result, non-radioisotopic technologies are more applicable to some uses of gamma sources than others. In the

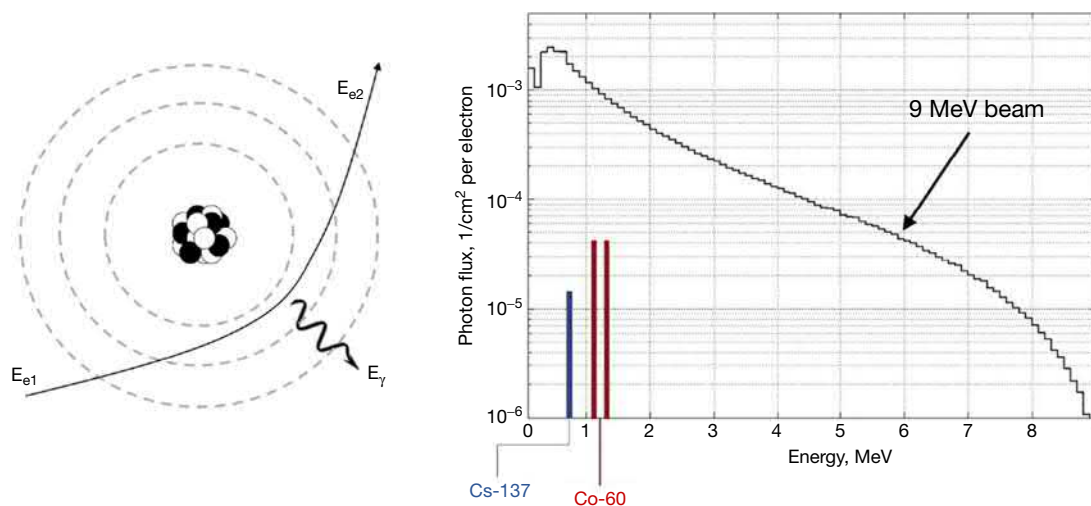


Fig. 1 (Left) Bremsstrahlung radiation process and (Right) bremsstrahlung spectra simulated with MCNPX for 9-MeV beams hitting a thick tungsten converter. Energies of gamma rays emitted by Cs-137 (blue) and Co-60 (red) are also shown for comparison.

following sections, several applications of sources and discuss advantages and disadvantages of their non-radioisotopic alternatives will be addressed.

Blood irradiators

To prevent TA-GVHD, a potentially fatal complication of blood transfusion—especially for immunocompromised patients—blood needs to receive 15–50 Gy of dose. Thousands of cesium blood irradiators were installed worldwide and have been used routinely for this purpose. However, the situation changed about a decade ago when several programs were initiated to motivate the users of cesium sources to switch to alternative technologies. Licensees who choose to replace their cesium irradiators with non-radioisotopic technologies are eligible to various incentives, including financial ones. As a result, in some countries, such as Norway, France, and Denmark, all cesium blood irradiators were replaced. In others, such as Japan or USA, the conversion is still ongoing, rather successfully. The US DOE NNSA ORS initiated the Cesium Irradiation Replacement Project and more than 100 cesium irradiators have already been phased out in several locations, including New York City and the University of California system. More replacements are planned for the next few years in the US.

Several low-energy self-shielded X-ray machines are currently available to replace cesium blood irradiators. These units are compact and self-shielded so they can easily be installed in a regular room in a hospital and are easy to use. While the energy of the X-rays they generate is significantly lower than 661 keV, their high current results in high dose rate, typically several Gy/min. This allows irradiating several blood units in just a few minutes. The main disadvantage of X-ray irradiators is their high maintenance cost.

The licensees interested in replacement consider different factors (see Table 2), but the main two are infrastructure requirements and costs, both capital and operational. Developing countries present a unique set of challenges due to their limited financial resources and lack of advanced infrastructure. In this situation cesium irradiators, which are easy to operate, might look very attractive over the short term. However, rather than merely accepting the “simpler,” but potentially threatening technology, the developing world should consider investments in personnel training and development of advanced infrastructure, and the costs should be mitigated through international support. The recent world-wide effort in replacing cesium irradiators with X-ray units is a positive step and a great example of successful introduction of alternative technologies. This work should continue to expand to effectively reduce the risk of terrorism.

R&D irradiators

Unlike blood irradiation, which is well defined in terms of required dose and irradiated product, R&D applications of gamma sources are extremely diverse. R&D gamma irradiators are used in biology, geology, material studies, and other areas of science and technology. Dose requirements differ greatly: from 2 to 10 Gy for small animal studies to 100 kGy for investigation of radiation-induced polymer degradation. Due to its long half-life, cesium-137 is widely used for R&D purposes, with hundreds of cesium research irradiators (1–100 kCi each) installed worldwide. Many research institutes, universities, and government laboratories consider and even already have begun the conversion from cesium to X-ray devices or electron accelerators. However, this process has a number of challenges. Existing low energy X-ray machines described above might be a good substitute for cesium blood irradiators; however, not all the models can provide the high penetration depth and good dose uniformity ratio (DUR) needed for some uses of research irradiators. To fully match the dose penetration of cesium research irradiators, an accelerator with at least 2.5 MeV electron beam energy is often required. A new generation of accelerators, for example 3–7 MeV machines, can potentially replace cesium sources for diverse R&D work.

Cobalt-60 has a significantly shorter half-life compared to cesium and, therefore, is less common for R&D applications. Nevertheless, about 500 self-shielded cobalt-60 installations (with up to 100 kCi of cobalt) are estimated to be present worldwide in research institutes, universities, and government laboratories. Uses of cobalt-60 research irradiators are as diverse as uses of cesium-137 research irradiators. Since cobalt has twice as high gamma ray energy, it is often used for studies that need even better penetration and DUR than provided by cesium units. Note that 3–7 MeV, 0.1–0.3 mA accelerators, which could replace cesium and

Table 2 Advantages and disadvantages of cesium blood irradiators and X-ray systems.

	<i>Cesium irradiators</i>	<i>X-ray systems</i>
Advantages	• Simple and reliable technology • No advanced infrastructure is required	• Long term cost saving • Safety and security are less of an issue • Can be turned on and off
Disadvantages	• Uncertainty in future cesium costs • Maintenance of expensive license, surveillance systems and security procedures • Costly disposal of cesium	• High initial cost to replace the unit • Requires stable power supply • Requires highly trained personnel • Lower reliability compared to cesium unit; need to have a backup system

cobalt research irradiators, will require extensive shielding and will most likely not fit into the original lab space. Lab expansion or construction of a new facility will increase the costs of the replacement project. On the other hand, accelerators with variable energy, dose rate, penetration depth, and field profile, will open new opportunities for scientists.

Larger (few hundred kCi) cobalt-60 research irradiators are even less common than self-shielded cobalt units. However, a few dozen such facilities are installed worldwide. A typical facility consists of a concrete bunker with a source pool (or source rack) in the middle. Products to irradiate are usually placed around the source, sometimes on turntables. This setup can accommodate larger volumes of samples compared to self-shielded units. It might be possible to retrofit such a facility by replacing the cobalt-60 source with a 3–10 MeV, 2–10 mA accelerator. In this case, water needs to be removed (in case of wet source), and the accelerator needs to be installed in the pit with the electron beam going upwards. Bending magnets can be used to bend the electron beam so that it's horizontal. Adding a bremsstrahlung converter and rastering magnets will result in a bremsstrahlung photon beam on the conveyor. In many cases it might not be possible to retrofit the existing facility, and a new facility needs to be built.

Replacement of research irradiators with alternative technologies has begun, but it is a challenging process. Accelerators to replace cesium and cobalt R&D irradiators already exist. However the current state of technology would not meet the expectations of the market in terms of price, complexity, reliability, and maintenance. Further development of the technology and transition should be supported by governmental and international organizations.

Teletherapy

The major modality of curative or palliative treatment of cancer, which continues to be one of the leading causes of mortality worldwide, is radiotherapy (RT). The first RT units were developed in the 1950s and contained cobalt-60 as a source of gamma radiation. In a typical design, the cobalt source is just stored in a shielded head where it can be moved pneumatically. During the treatment the source is moved next to an opening through which the gamma beam exits. After the treatment is over, the source is returned to the shielded position. Collimators, flatteners, jaws, and trimmer bars are used to adjust the radiation field as needed. A typical teletherapy machine contains a capsule with 5–10 kCi of cobalt and provides few Gy/min dose rate to a patient. The sources are usually replaced every 5–6 years when the dose rate becomes “too low.” However, financial considerations often result in longer usage.

Due to a relatively simple design and operating requirements, this invention provided a tremendous boost in cancer treatment and today more than 10,000 cobalt units are operational over the world (<https://dirac.iaea.org/>). However, at about the same time the first cobalt units appeared, development of medical linacs also began. Various types of linacs are currently available for clinical use. Some provide only X-rays, while others provide both X-rays and electrons at megavoltage energies. Modern machines offer dual photon energy and multiple electron energies anywhere in the 6–25 MeV range, photon beam intensity modulation with multi-leaf collimators, and full dynamic conformal dose delivery with intensity modulated beams.

The complexity of modern multimodal linacs makes it possible to perform conformal treatments of the tumor, which could not be done with cobalt units. It is no wonder that increasingly sophisticated generations of linacs replaced many cobalt machines and became the most commonly used tool in modern RT. Despite the clear technological advantages of linacs, cobalt units still play an important role in RT, especially in the developing world, which has higher populations and fewer financial resources. Replacement of cobalt machines in developing countries is difficult, due to their considerably lower capital, installation and maintenance costs, simplicity of design, and ease of operation. While linacs continue to become more reliable and affordable, cobalt units will likely play an important role in cancer therapy for the foreseeable future.

Industrial irradiators

Industrial cobalt-60 irradiators are primarily used for food or phytosanitary applications and medical device sterilization. They operate in either continuous or, less commonly, in a large batch mode, and products to be irradiated come in pallets or tote boxes. Food irradiation requires relatively low doses: less than 1 kGy to inhibit the growth of sprout, slow the ripening process, and kill insects and larvae that can be found in grains, fruits, and vegetables after harvesting; and up to 10 kGy to destroy pathogens in meat and fish, control molds and extend shelf life in fresh products. Therefore, food irradiation processing typically relies on small (~1 MCi) industrial irradiators, which can rapidly provide 1–10 kGy doses. Large industrial irradiators (greater than 1 MCi) are almost exclusively used for medical device sterilization as they can deliver high doses required for medical products (25–50 kGy) at high speed. If needed, they can also be used for food or phytosanitary applications with a split source rack.

Gamma irradiation is an established and simple technology with about 500 MCi of cobalt-60 installed at about 200 facilities (<http://iiaglobal.com/wp-content/uploads/2018/01/White-Paper-Comparison-Gamma-Eb-Xray-and-EO-for-Sterilisation.pdf>). Many producers of medical products have been using it for decades. Moreover, many new medical products are designed and developed with gamma sterilization modality in mind and it will be costly to change it. However, raising security concerns, recent perturbations in cobalt supply, and sharp increases in its cost have made many users consider alternative technologies. E-beam technology has gained popularity, but it is not suitable for many products. In addition, it requires much more complex machinery and maintenance, which can result in prolonged shutdowns. It also requires a reliable supply of electricity, which can be an issue in some countries. However, it is very cost efficient and provides very high product throughput. X-ray technology is similar to e-beam in requirements as it uses the same accelerators. However, during the electron-to-photon conversion a significant fraction of energy is lost and the economic viability is not as great as in the case of an e-beam. On the other hand, this method can be used for a much wider portfolio

Table 3 Gamma, e-beam, and X-ray irradiation and their advantages and disadvantages.

	<i>Gamma</i>	<i>E-beam</i>	<i>X-ray</i>
Technology	• Relatively simple and reliable • Good uptime	• Complex technology • Fair uptime • Requires more personnel	• Complex technology • Fair uptime • Requires more personnel
Products	• Good penetration, can treat “dense” products	• Poor penetration, limited to low density and thin materials	• Good penetration, can treat “dense” products
Security	• Good safety record • Cannot be turned off • Depleted sources are an issue	• Good safety record • Can be turned on and off • Does not produce radioactive material	• Good safety record • Can be turned on and off • Does not produce radioactive material
Maintenance	• Cobalt needs to be replaced regularly • Limited suppliers and uncertainty of future availability of cobalt • High price of cobalt	• Requires stable power supply and backup systems for critical utilities • Maintenance service contract usually required	• Requires stable power supply and backup systems for critical utilities • Maintenance service contract usually required
Costs	• High capital costs • High costs of cobalt	• High capital costs	• High capital costs
Public acceptance	• Low	• No concerns	• No concerns
User acceptance	• Good	• Good	• Low, as technology is relatively unknown

of products. Several facilities now equip their accelerators with removable X-ray converters so that they can be used in both modes, e-beam and X-ray, as needed. Nevertheless, this method remains relatively unknown to medical product producers. The three main irradiation modes described above, and their advantages and disadvantages are summarized in [Table 3](#).

Industrial irradiation is one of the most challenging domains for alternative technologies. To provide high dose rates required for high product throughput, one would need reliable accelerators delivering both high energy (as close to 10 MeV as local regulations allow) and high power (tens of kW) beams. Only a handful of companies in the world produce such machines, and they are still complex and expensive. To replace industrial cobalt irradiators, more than one accelerator should be installed in each facility—both to provide the throughput and to offer backup in case of maintenance or shutdown.

Conclusions

It is nearly impossible to imagine our life without radioactive sources and it is difficult to overestimate their importance. However, several hundred incidents involving sources are reported annually, and this might be just the tip of the iceberg. To make it worse, in some cases, it was obvious that the material was planned to be used maliciously. After the terrorist attacks on the United States on September 11, 2001, concerns about the safety and security of radioactive sources grew, including the fears of a “dirty bomb.” To minimize the threats of misuse of sources it is essential to improve the security culture, and to minimize the amount of radioactive material around us by introducing alternative technologies.

For many applications, alternative, non-radioisotopic technologies have been around for a while. However, due to their complexity and high costs, these methods were often overlooked. Raising safety and security concerns increased interest to these techniques. For some applications, such as blood irradiation, the transition to X-ray technology is very successful. For other applications, such as industrial sterilization of medical devices, the progress is not as great due to the lack of powerful, reliable, and inexpensive accelerators.

Radiation sources are essential to securing our health, safety, and prosperity, and they should be replaced with caution, to make sure that the alternative technologies are viable, practical, and preserve the benefits of the sources. Potential risks should be evaluated as well as potential benefits and all the trade-offs should be carefully considered. There is no doubt that alternative technologies will become more and more widespread. However, this process will take time and will require strong support from governmental organizations and international agencies. Market, regulatory, and licensing incentives should facilitate the introduction of alternative technologies and reduce the attractiveness of radioactive sources. This work should continue to expand worldwide to effectively reduce the threat of radiological terrorism.

Acknowledgments

It is a professional privilege for me to have been chosen by the Editor of the Volume, Dr. Alan Waltar, to handle the important task of discussing the risks of radiation sources and alternative technologies. I would like to acknowledge with much appreciation the excellent support received from my colleagues, especially Ms. Kristin Hirsh and Mr. Lance Garrison from the US DOE NNSA. It is envisaged that the present volume will be found valuable by not only scientists and engineers, but also by various stakeholders in countries across the world and in international agencies.

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- <https://dirac.iaea.org/> —IAEA's Directory of RAdiotherapy Centres.
- <https://www.energy.gov/nnsa> —US National Nuclear Security Administration.
- <https://www.nti.org/> —Nuclear Threat Initiative.
- <https://www.world-nuclear.org> —World Nuclear University.
- <https://iiaglobal.com/> —International Irradiation Association.

Public Safety and Security: Nuclear Forensics

Vitaly Fedchenko, Stockholm International Peace Research Institute (SIPRI), Stockholm, Sweden

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Glossary (IAEA, 2015a,b)

Categorization or identification (IAEA, 2014) A process performed to address the threat posed by a specific incident, with the goal of identifying the risk to the safety of first responders, law enforcement personnel and the public and determining if there is criminal activity or a threat to national security.

Chain of custody The procedures and documents that account for the integrity of physical evidence by tracking its handling and storage from its point of collection to its final disposition.

Characterization Determination of the nature of the radioactive material and associated evidence.

Improvised nuclear device A device incorporating fissile materials designed to produce a fission chain reaction. Such devices may be fabricated in a completely improvised manner or may be an improvised modification to a nuclear weapon.

National nuclear forensics library An administratively organized collection of information on nuclear and other radioactive material produced, used or stored within a State.

Nuclear forensic interpretation The process of correlating sample characteristics with existing information on types of material, origins and methods of production of nuclear and other radioactive material, or with previous cases involving similar material.

Nuclear forensics Nuclear forensics is the examination of nuclear or other radioactive material, or of evidence that is contaminated with radionuclides, in the context of legal proceedings under international or national law related to nuclear security.

Nuclear security event An event that has potential or actual implications for nuclear security that must be addressed.

Nuclear security The prevention of, detection of, and response to, criminal or intentional unauthorized acts involving or directed at nuclear material, other radioactive material, associated facilities, or associated activities.

Radiation exposure device A device with radioactive material designed to intentionally expose members of the public to radiation.

Radiological crime scene A crime scene at which a criminal act or intentional unauthorized act involving nuclear or other radioactive material has taken place or is suspected.

Radiological dispersal device A device to spread radioactive material using conventional explosives or other means.

System for nuclear material accounting and control An integrated set of measures designed to provide information on, control of and assurance of the presence of nuclear material, including those systems necessary to establish and track nuclear material inventories, control access to and detect loss or diversion of nuclear material, and ensure the integrity of those systems and measures.

Abbreviations

CMX Collaborative material exercises (exercises developed and conducted by the ITWG)

CRP Coordinated research project (research coordination tool used by the IAEA)

GICNT Global Initiative to Combat Nuclear Terrorism

HEU Highly enriched uranium

IAEA International Atomic Energy Agency

ICP-MS Inductively coupled plasma mass spectrometry
 ICSANT International Convention for the Suppression of Acts of Nuclear Terrorism
 IND Improvised nuclear explosive device/Improvised nuclear device
 INSSP Integrated Nuclear Security Support Plan
 ITBD Incident and Trafficking Database
 ITU Institute for Transuranium Elements (Karlsruhe)
 ITWG Nuclear Forensics International Technical Working Group
 JRC Joint Research Centre (Karlsruhe)
 MORC Material out of regulatory control
 NFWG Nuclear Forensics Working Group
 NNFL Natural Nuclear Forensics Library
 SEM Scanning Electron Microscope
 SIMS Secondary ion mass spectrometry
 TIMS Thermal ionization mass spectrometry
 XRD X-ray diffraction analysis
 XRF X-ray fluorescence analysis

The role and origins of nuclear forensics

The role of nuclear forensics in combating crime and terrorism

In 2005 the International Atomic Energy Agency (IAEA) officially defined the new discipline and the major part of its work, nuclear security, as the complex of laws, regulations, institutions, systems and measures aimed at the “prevention of, detection of, and response to, criminal or intentional unauthorized acts involving or directed at nuclear material, other radioactive material, associated facilities, or associated activities” (IAEA, 2005). Even though the official definition of nuclear security does not itself mention nuclear terrorism, the document that introduced it did connect nuclear security with “measures to protect against nuclear terrorism” (IAEA, 2005). A nuclear security event is an “event that has potential or actual implications for nuclear security that must be addressed” (IAEA, 2015b).

Nuclear terrorism was officially defined the same year with adoption of the 2005 International Convention for the Suppression of Acts of Nuclear Terrorism (ICSANT). This convention provides a strict legal definition of nuclear terrorism as a set of specific offences (listed in Article 2) that can be performed by non-state actors, and provides a framework for legal action. ICSANT does not cover actions of states, nor offences that were completely confined to the territory of one state and did not include anyone other than the nationals of that same state. Thus, nuclear security is understood as defence against nuclear terrorism within the framework of the international, UN-linked legal mechanisms and organizations. Nuclear security also works at the level of individual states to deal with nuclear security events, but those, even if the same in nature, would not qualify as acts of nuclear terrorism under ICSANT.

While the exact legal definition of nuclear terrorism is given in ICSANT, in practical terms it is understood as one (or more) of the following four actions by a non-state actor: (a) theft and/or use of a nuclear warhead obtained from a state; (b) theft of a nuclear material and making of an ‘improvised’ nuclear explosive device; (c) obtaining radioactive material and using it directly or making a “dirty bomb” or any other device designed to disperse radioactivity; (d) attacking a nuclear installation (Ferguson et al., 2005).

Nuclear forensics has been established well before nuclear security was defined as a field. (See section **Nuclear forensics and its origins** below.) The newest IAEA definition of nuclear forensics states: “Nuclear forensics is the examination of nuclear or other radioactive material, or of evidence that is contaminated with radionuclides, in the context of legal proceedings under international or national law related to nuclear security” (IAEA, 2015a,b). Nuclear forensic science (also referred to as nuclear forensics) is seen as a subdiscipline of a general forensic science dealing with nuclear or other radioactive materials within the framework of legal mechanisms pertinent to nuclear security. It is aimed at strengthening the nuclear security regime through two out of three mechanisms mentioned in the definition of nuclear security: (1) response (including an investigation of a nuclear security event and possibly determination of the material’s source, point of origin and route of transfer), (2) prevention of future cases by identification of suspect material origin or point of loss of control, and improvement of material’s control, accounting and physical protection there (IAEA, 2015a,b).

Nuclear forensics can be useful in at least three out of four nuclear terrorism scenarios described above: (a) theft and/or use of a nuclear warhead, (b) theft of nuclear material for the purpose of creating a nuclear explosive device, and (c) theft and/or other illicit use of radioactive materials or sources. Each of these scenarios has two subscenarios: before and after the actual terrorist act that would result in an explosion of some kind or possibly a dispersion of radioactive material by other means, including poisoning (McFee and Leikin, 2009). Table 1 mentions indicative or typical tasks of the nuclear forensic science in those scenarios.

Although there is some discussion concerning this general topic in the scientific literature, nuclear terrorism scenarios that involve an actual stolen nuclear weapon are considered less likely than those that involve an improvised nuclear explosive device. Those that involve dispersal of radioactive materials are considered the most likely of the three (Volders and Sauer, 2016).

Table 1 Potential tasks for nuclear forensic science aimed at prevention of or response to nuclear security events, including nuclear terrorism acts.

	<i>Nuclear security events, involving</i>		
	<i>A nuclear warhead</i>	<i>An improvised nuclear explosive device (IND) from illegally obtained nuclear material</i>	<i>Dispersal of radioactive material (including poisoning)</i>
Before the explosion or dispersion	Identification of origin and/or production date of fissile material in a warhead	Identification of origin and/or production date of fissile or other nuclear material found outside of regulatory control	Analysis of radioactive material to assist investigation (e.g. by determining the material's origin or age); or collection of "traditional" evidence from surfaces contaminated with radionuclides.
Post-explosion or dispersion	Analysis of particles and gases (from weapon debris) for understanding the design features of the exploded device and assisting attribution	Analysis of particles and gases (collected after the explosion) for understanding the design features of the exploded device and assisting attribution	Analysis of particles (from a radiological dispersal device) and/or human tissue, clothing and excretions; or collection of "traditional" evidence from surfaces contaminated with radionuclides.

Highlighted table cells represent areas where real incidents happened, and they are also the ones that are most often discussed in open literature on nuclear forensics and nuclear security.

Luckily, successful dispersals of radioactive materials with malicious purposes are rare and there has not as yet been a terrorist act that would involve a real nuclear explosive device. In order to prevent such instances from happening, a large amount of work has been done internationally to address a potential precursor to such events; namely, the existence of nuclear or other radioactive materials outside of regulatory control (MORC), and in particular, an illicit trafficking of such materials across national boundaries—a phenomenon known colloquially as nuclear smuggling. In its broadest sense, illicit trafficking refers to criminal or unauthorized acts involving MORC, such as their "import, export, possession, sale, delivery, movement, use, storage, disposal or transfer" (IAEA, 2007).

To summarize, nuclear forensics is most often seen as a nuclear security tool intended to aid criminal investigation of illicit trafficking or nuclear terrorism cases. One should keep in mind, however, that nuclear smuggling was happening (and nuclear forensics was used to investigate it) before nuclear security was formally defined as a discipline, and nuclear forensics techniques have also been used to investigate cases not pertinent to nuclear security.

Nuclear forensics and its origins

Before the 1990s, the issue of nuclear and other radioactive materials outside of regulatory control was not completely unknown. But the threat was low level, isolated and more relevant to the field of nuclear non-proliferation than nuclear security (Fedchenko, 2017). This changed in the early 1990s, following the disintegration of the Soviet Union.

By the time of its breaking up into independent states, the Soviet Union had by far the largest arsenal of nuclear weapons in the world (Norris and Kristensen, 2010). It is also estimated to have had the largest stockpile of highly-enriched uranium (HEU) and plutonium (Glaser and Mian, 2008). It is clear that vast quantities of nuclear and especially of other radioactive materials and radioactive sources, as well as nuclear facilities, have been widely scattered across and extensively used in the USSR for many decades. The controlled nature of the Soviet society allowed for secure operation of the Soviet nuclear complex. However, as soon as the USSR began to disintegrate and pervasive controls have disappeared, security of the Soviet nuclear complex materials became lax and even the knowledge concerning location of some of them became fragmented. This happened against the background of the sharp economic decline and "wrenching societal change" in the states of the former Soviet Union, as well as the lack of public awareness of market values and dangers of various materials. As a result, people that had access to some kind of radioactive materials had a strong incentive to attempt to profit from it, and that led to theft and illicit trafficking.

The earliest known investigated cases of nuclear smuggling reportedly occurred in Switzerland and Italy in 1991, although it is not known whether a connection between the material in those cases and the Soviet Union was established (Mayer et al., 2007). These incidents were quickly followed by others in Germany, the Czech Republic, Hungary and several other Central European countries. The illicit trafficking phenomenon became so widespread that the IAEA established, in 1995, the Illicit Trafficking Database (ITDB, renamed the Incident and Trafficking Database in 2012) to record and analyze its member states' reports of illicit trafficking and other unauthorized activities involving MORC. The ITDB has indeed registered about 40 cases related to trafficking or malicious use, including scams, between 1993 (earliest year when data was collected) and 1995 (IAEA, 2020). Many of these cases occurred in Europe or its immediate neighborhood, and some involved HEU and plutonium in various quantities (Zaitseva and Steinhausler, 2014).

Investigations were commenced in response, and the authorities often had questions about the nature of the intercepted material, its intended use, origin and potential trafficking route. In cases that do not involve radioactive material, answering these questions is assisted by forensic science laboratories, but it is not normally possible for them to handle radioactive material. It has become clear that in these new cases forensic analysis would have to be done by nuclear measurement laboratories and research institutes.

The first known official case of nuclear forensic analysis for the purposes of a criminal investigation was triggered by the seizure of 72 uranium pellets in Augsburg, Germany, in March 1992. The pellets were analyzed at the EU Joint Research Centre's Institute of Transuranium Elements (ITU) in Karlsruhe, Germany (currently called JRC Karlsruhe). The laboratory was chosen because of its expertise in analytical methods pertinent to IAEA nuclear safeguards (which provided correct and reliable measurements in this case), as well as available nuclear fuel cycle expertise (which allowed for correct interpretation of results of those measurements). The ITU analysis concluded that the seized pellets were nuclear fuel pellets for a Russian RBMK-type graphite-moderated nuclear power reactor, and identified two possible production plants in Russia and Kazakhstan (Mayer et al., 2007).

Thus, the discipline of nuclear forensics was born. But it was not yet defined, and its processes, protocols, purpose and context were not yet described. The Augsburg case was quickly followed by others, and the need to assist criminal investigations arose not just in Europe, but also in other countries, including the United States and Russia. The instances of nuclear smuggling of material were often international. Perhaps the most high-profile case of the early 1990s involved, first, a seizure of a small sample of PuO₂-UO₂ mixture on July 25, 1994, and then a second, much larger seizure of 363.4 g of plutonium in a similar mixture at the Munich Airport on August 10, 1994. Nuclear material was brought to Munich by a Colombian physician on a Lufthansa flight from Moscow. He was found carrying a suitcase containing 560 g of mixed plutonium and uranium oxide powder (containing 363.4 g of plutonium) and 210 g of enriched lithium metal. The analysis conducted by the ITU revealed that the plutonium was probably produced in a graphite-moderated nuclear reactor at the end of 1979, and that the lithium metal was highly enriched in lithium-6, the isotope required for the deuterium-tritium nuclear fusion utilized in thermonuclear weapons (Edwards, 1994; Mayer et al., 2019).

It became clear that nuclear forensics needed an international dimension.

International mechanisms of nuclear forensics

While materials outside of regulatory control certainly exist and can be moved within territories of individual countries, their smuggling across borders is an especially worrisome phenomenon, because it likely demonstrates not just an intent but also a certain capability to conduct criminal acts. It became obvious that combatting the cases of cross-border nuclear smuggling would require international cooperation. This task was eventually undertaken by three international mechanisms: Nuclear Forensics International Technical Working Group (ITWG), the IAEA, and the Nuclear Forensics Working Group of the Global Initiative to Combat Nuclear Terrorism (NFWG).

Nuclear Forensics International Technical Working Group (ITWG)

In May 1995, the US Department of Energy (DOE) proposed holding an international conference on the role of nuclear forensics in addressing nuclear smuggling. The "International Conference on Nuclear Smuggling Forensic Analysis" met in November of 1995 at the Lawrence Livermore National Laboratory in the United States, with more than 70 scientists, law enforcement officials, intelligence and policy-making professionals from 14 countries and organizations in attendance. That meeting came up with a proposal to establish an ongoing forum for international technical cooperation in nuclear forensics by forming a Nuclear Smuggling International Technical Working Group, subsequently renamed the Nuclear Forensics International Technical Working Group (ITWG) (Niemayer, 2019). Two of the founding co-chairs of the ITWG, Lothar Koch and Sidney Niemeyer, explained in 2002 their vision for the group's goal: to "develop a preferred approach to nuclear forensic investigations that is widely understood and accepted as credible" (Fedchenko, 2017). They also formulated the elements of the group's work program aimed at achieving that goal: the development of procedures and protocols for the collection and preservation of evidence, and for laboratory investigation; the development of techniques for material characterization; the development of methods and tools, such as nuclear forensics libraries, to assist with the interpretation of the characterization's results; executing inter-laboratory exercises; and facilitating technical assistance to countries on request (Fedchenko, 2017). This vision of the purpose and work of the ITWG remains unchanged to this day, and the structure of the ITWG task groups, described below, reflects it.

As of 2020, the ITWG has developed into a worldwide "informal and unaffiliated association of nuclear forensics practitioners who are dedicated to advancing the scientific discipline of nuclear forensics in order to prevent and respond to incidents involving nuclear and other radioactive material out of regulatory control" (Smith, 2016). The ITWG preserves its informal status, and is able to remain open for participation of all states, as well as organizations and individuals with a recognized role or expertise in nuclear forensics. It is successful in fostering active cooperation and collaboration between members. At the same time, the ITWG has established and preserved its international credibility through ensuring active participation of the world's leading nuclear forensics laboratories and top experts. It has become an international resource for governments and international organizations, assisting them in fulfilling their nuclear security obligations. ITWG's current mission is to identify, socialize and promote best practices in the field of nuclear forensics.

The ITWG's work throughout the year is steered by two co-chairpersons (traditionally, one representing the US government and another the European Commission), and a small Executive Committee. During the year the main work of the ITWG is done in the five Task Groups (TG): the Evidence Collection Task Group, the Exercise Task Group, the Guidelines Task Group, the National Nuclear Forensics Library Task Group, and the Outreach and Training Task Group (Smith, 2016).

The Evidence Collection TG focuses on procedures involved in the safe collection of evidence at a nuclear security event or a crime scene. This TG's output includes guidelines on topics such as "Evidence Collection in a Radiological or Nuclear Contaminated Crime Scene" (ITWG, 2012).

Formulating and executing inter-laboratory exercises has been one of a few primary focus areas of the ITWG work since its inception. The Exercise Task Group's organization of collaborative material exercises (CMXs, called "Round Robin" exercises in the earlier years), in which the nuclear forensics laboratories that elect to participate, analyze the same material circulated by organizers. The results are later compared to identify "lessons learned", but without attribution of the analytical results to a specific lab. The first CMX involved plutonium oxide (1996); the second investigated highly enriched uranium oxide (2001); the third comprised highly enriched uranium metal (2010); the fourth covered low-enriched uranium (LEU) nuclear fuel pellets (2015), the fifth involved two LEU oxide pellets and one "virtual sample" for comparison (2017), and the sixth utilized depleted U metal ingot and one stable Ce metal ingot, each contaminated with a trace amount of weapons grade Pu fluoride (2019). The second and the sixth CMX also included specially planted "traditional" evidence - fingerprints and toolmarks (Schwantes and Marsden, 2020).

The Guidelines Task Group has so far published 13 guideline documents describing "best practices" of using destructive and nondestructive measurement methods for the purposes of nuclear forensics. These guidelines are designed to be widely available and applicable to nuclear forensics laboratories across the world.

The National Nuclear Forensics Library Task Group's purpose is to organize a series of virtual table top exercises that advance and test the concept of nuclear forensic libraries by facilitating comparisons of samples of interest and available data on known materials used, produced, or stored as part of the nuclear fuel cycle. The "Galaxy Serpent" exercises challenge participants to construct a comparative database and subsequently determine the consistency of exercise data with the holdings in the database. As of 2020 the Task Group has conducted four "Galaxy Serpent" exercises.

Finally, the Outreach and Training Task Group works to distribute the ITWG's technical output to individuals, laboratories, organizations and states with an interest in nuclear forensics. The ITWG has an open website that publishes the Guidelines mentioned above, as well as a tool to promote cohesiveness of the nuclear forensics' community across the world—the "ITWG Nuclear Forensics Update" newsletter.

The ITWG's main events are the Annual meetings. They gather a significant number of professionals every year and provide space for participants to exchange information on topics such as recent cases where nuclear forensic analysis was used, national achievements in nuclear forensic practice, new analytical techniques for measurement of radioactive materials, best practices in presentation of analytical methods or collected evidence to non-scientist audiences (e.g. in court), and outcomes of exercises.

The ITWG has received political support and international recognition on a number of levels. First, nuclear forensics was officially recognized as an element of response to nuclear smuggling events as early as at the Group of Seven (G7) Moscow Nuclear Safety and Security Summit in 1996. The ITWG was later also endorsed by the Group of Eight (G8) Non-Proliferation Experts Group (NPEG) in May 1999, and by the G8/G7 Nuclear Safety and Security Group in 2005.

The International Atomic Energy Agency (IAEA)

The IAEA is an international organization within the UN system that was established in 1957. It works to promote the peaceful use of nuclear energy, and to inhibit its use for harmful purposes, including nuclear weapons and, more recently, nuclear terrorism through promoting nuclear security worldwide. However, implementation of a nuclear security regime remains a responsibility of the IAEA Member States. Upon request of its Member States, the IAEA can support them in establishing and maintaining their nuclear security regimes and infrastructures, including nuclear forensic capabilities. One or more of the following tools can be used to achieve this.

First, the IAEA issues international guidance documents on nuclear security, including in its Nuclear Security Series. This includes the general guidance such as "Nuclear Forensics in Support of Investigations" and "Combating illicit trafficking in nuclear and other radioactive material", as well as more specific technical advice, e.g. "Development of a National Nuclear Forensics Library: A System for the Identification of Nuclear or Other Radioactive Material out of Regulatory Control."

Second, the IAEA has specific advice mechanisms able to assist states, upon their request, in applying that guidance or assessing their nuclear security regime in general and their nuclear forensic capabilities in particular. For instance, the IAEA may develop for the states the Integrated Nuclear Security Support Plan (INSSP), reviewing their nuclear security regimes and identifying areas where those regimes need to be strengthened.

Third, the IAEA, by itself or in cooperation with the European Union, the United States, and other countries, can provide training on issues including nuclear forensic awareness, radiological crime scene management and nuclear forensic methodologies.

Finally, the IAEA promotes the quality of nuclear forensics research and cohesion of research teams through a mechanism of coordinated research projects (CRPs) (Smith, 2019).

The Global Initiative to Combat Nuclear Terrorism (GICNT)

The GICNT, launched in 2006 by the United States and Russia, is a voluntary international initiative of 89 states and six international organizations aiming at strengthening global efforts to prevent, detect and respond to the threat of nuclear terrorism. The GICNT works at a decision-making level, conducting multilateral activities that strengthen the plans, policies, high-level procedures, and interoperability of partner nations. The GICNT has the Nuclear Forensics Working Group that is designed to assist the political leadership in individual countries in their build-up of domestic nuclear forensic capabilities. The group's activities are aimed at raising awareness, developing intergovernmental relationships, conducting joint policy-level exercises and promoting best practices.

The model action plan and the nuclear forensics process

The ITWG recognized very early on that nuclear forensics is a new and emerging discipline that has to apply methods of natural sciences such as physics and chemistry to judicial and other investigative environments. Therefore, nuclear forensics had to be built up conceptually, starting with definitions. The ITWG was exactly the right international body for this job. The central part of the ITWG's vision to "develop a preferred approach to nuclear forensic investigations that is widely understood and accepted as credible," began to be realized in 1997 when the ITWG started to discuss an action plan (a step-by-step description of a general process) that would be applicable worldwide and would allow professionals to "identify and trace back seized nuclear material from illicit trafficking" (Fedchenko, 2017). In June 2000, the ITWG adopted an early version of such a plan at its meeting in Vienna, and named it the Model Action Plan. It contained both the description of the process and the early definitions underlying the discipline of nuclear forensics. It was published initially as a US government document in 2004, and then, with some updates, became an integral part of the IAEA guidance documents on nuclear forensics in 2006 and 2015 (Dudder et al., 2004; IAEA, 2006). The contemporary version of the Model Action Plan is given in Fig. 1.

The nuclear forensics process begins at the site of a nuclear security event, in this case referred to as a "radiological crime scene." At this stage, forensic analysis is a part of a large set of necessary activities included into the radiological crime scene conduct of operations. While it is important that all responders are aware of forensic examination needs, such as the integrity of evidence and preservation of its later admissibility in a courtroom setting, those needs are secondary to safety and security of all involved personnel and the general public. At this stage, the nuclear or other radioactive material found on the scene should go through the *first* stage of the nuclear forensic analysis—categorization or identification. *Categorization (identification)* is the quick assignment of the material of interest for further investigation that determines its further handling, for example, modalities of transportation or an ability of a certain lab to accept it. It is conducted to identify potential nuclear security implications and risks to personnel and the public. For example, a large cache of HEU intercepted in a sting operation would have different implications than an abandoned radioactive source. Transportation of materials to laboratories usually happens after this stage.

Results of material categorization serve as input to the forensic examination plan that is then drafted by investigative authorities—together with "traditional" and nuclear forensic laboratories. The forensic examination plan describes the sequence and requirements to the examinations that need be conducted in order to support a potential criminal prosecution or other legal action. The plan is based on specific questions normally defined by investigative authorities (IAEA, 2015a).

A nuclear forensic analytical plan is developed on the basis of the forensic examination plan to define the specific types and sequence of measurements to be performed over a nuclear or other radioactive material or over evidence contaminated with radionuclides. A nuclear forensic analytical plan then guides the *second* stage of nuclear forensic analysis—characterization.

Characterization is the determination (i.e. measurement) of a sample's characteristics. It may involve (depending on the plan) measurements of physical, chemical, elemental and/or isotopic characteristics of the material in the sample.

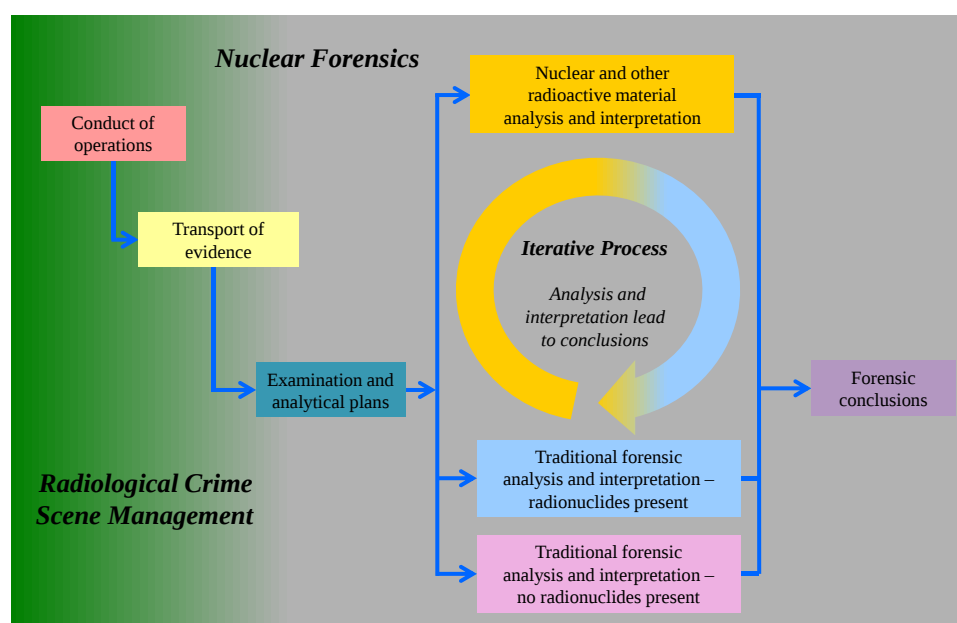


Fig. 1 Nuclear forensics model action plan: A process which supports an investigation of a nuclear security event. *Note:* Background shading indicates the transition from radiological crime scene management to nuclear forensics. IAEA (2015) Nuclear forensics in support of investigations. IAEA Nuclear Security Series, p. 7.

Table 2 Categories of characteristics of materials or items subject to measurement (Moody et al., 2014; IAEA, 2015a; L'Annunziata, 2020; Fedchenko, 2015).

Category of characteristic	Characteristics	Commonly used characterization techniques
Physical	Sizes, shapes and textures of solid objects Characteristics of powders: particle size distribution; morphology of individual particles Characteristics of liquids	Visual inspection, photography, dimensional and weight measurements, color, optical microscopy, Scanning Electron Microscopy (SEM), radiography
Chemical (molecular)	Chemical form of nuclear or radioactive material Non-radioactive chemicals used in the nuclear fuel cycle	X-ray spectrometry, XRD, Fourier transform infrared spectroscopy
Elemental	Major elements constituting the sample (if not clear from determination of chemical composition) Minor elements ^a Trace elements ^b	XRF, chemical titration, controlled potential coulometry
Isotopic	Isotopic composition of nuclear materials Isotopic composition of non-nuclear materials	Gamma spectrometry, alpha spectrometry, multiple mass spectrometry methods (e.g. inductively coupled plasma mass spectrometry (ICP-MS), thermal ionization mass spectrometry (TIMS), secondary ion mass spectrometry (SIMS))

^aMinor elements are usually those intentionally added to the material in small quantities to improve or change its physical or chemical characteristics.

^bTrace elements are usually added unintentionally during material production or manufacturing; e.g. machining of uranium metal with steel instruments will leave traces of iron and chromium behind.

For example, characterization of a hypothetical intercepted fuel pellet will likely include at least measurements of its size, shape and surface roughness, chemical composition (e.g. uranium oxide or metal), and isotopic composition (e.g. natural or enriched uranium) (Fedchenko, 2015; IAEA, 2015a). Most common techniques used for characterizing materials in this context as well as their purpose is discussed in Table 2. The ITWG has published on its open website specific guidelines on the use of some of these techniques in the nuclear forensic context.

This stage of characterization produces results of measurements (“analytical results”) in the form of raw data, e.g. numbers or photographs. Such data is rarely useful by itself for answering questions put forward by investigative authorities. In order to make use of it, the data has to be transformed into meaningful information. This is done at a *third* stage of nuclear forensics—*interpretation*.

Nuclear forensic interpretation is defined as “the process of comparing and associating sample characteristics with existing information pertaining to types of material, and origins and methods of production of nuclear or other radioactive material, or with previous cases involving similar material” (IAEA, 2015a,b).

This process utilizes two concepts crucial for nuclear forensics—nuclear forensic signatures, and national nuclear forensic libraries. Nuclear forensic signatures are, in essence, meaningful combinations of material characteristics. The IAEA defines them as “a set or sets of data characteristics of a given sample of nuclear or other radioactive material that may enable the sample to be identified as being consistent with, or as not being consistent with, particular nuclear or other radioactive material used, produced or stored in a State by way of either exclusion or inclusion” (IAEA, 2018).

The signatures are useful because the whole nuclear fuel cycle can be imagined as a flow of nuclear materials through nuclear facilities, each of which by definition transforms the material in some way and, therefore, leaves in the material some information about its history and possibly intended future. Even if nuclear material is simply stored, the process of radioactive decay ensures that the “parent” radioactive isotopes decay into “daughter” isotopes and their ratio may in principle be used to determine the “age” of the material and, possibly, the length of the storage time. In other words, with nuclear material moving through the nuclear fuel cycle, nuclear forensics signatures are being constantly introduced to and erased from it. This fundamental consideration is a nuclear forensic version of the famous Locard’s exchange principle used in “traditional forensics”: “whenever two objects come into contact with each other, they transfer material from one to the other” (Shaler, 2011).

A single nuclear forensic signature is not normally enough to identify a specific sample uniquely from known classes of similar materials. Nuclear forensics laboratories would, therefore, use multiple signatures to increase their confidence in the results of the nuclear forensic interpretation process. In addition, as indicated in Fig. 1, forensic analysis of any kind may be iterative, in the sense that the results of interpretation stage may reveal the need to get back to the characterization stage and conduct additional measurements in order to answer questions put forward by investigative authorities.

To facilitate the use of nuclear forensic signatures in individual states, the ITWG and the IAEA have developed a concept called the National Nuclear Forensics Libraries (NNFL). A NNFL is a national system for the identification of nuclear and other radioactive materials found out of regulatory control (MORC). It should contain, in an available form, reference data on nuclear and other radioactive materials produced, used, or stored within a State, as well as be connected to appropriate experts able to interpret that data. An NNFL enables the use of nuclear forensic signatures to compare existing data on known materials and data obtained from analytical measurements of samples of MORC or, possibly, data provided by other states (IAEA, 2015a). Implementation of this concept is much more straightforward in countries with a smaller nuclear fuel cycle. In some cases, an NNFL would look, for instance, at a list of radioactive sources existing in the state, if that is the extent of that country’s use of radioactive materials.

Thus, at the stage of nuclear forensic interpretation, the data from the characterization stage is interpreted and converted to usable information with the help of nuclear forensic signatures, nuclear forensic libraries, calculations, and subject matter expertise. The product of this fourth stage is nuclear forensic findings, i.e., typically, determinations on material origin and history. Because nuclear forensic findings may be used in legal proceedings or to identify nuclear security vulnerabilities, they are expected to be defensible (e.g. in court), and, therefore, have to be arrived at using arrangements that would allow their use in a specific country. Such arrangements often include the use of validated measurement methods and certified reference materials, reliance on demonstrated competencies, adherence to chain of custody and implementation of quality assurance and quality control procedures (IAEA, 2015a,b).

The final, fifth, stage of the process involves combining nuclear forensic findings with other aspects of the investigation (see Fig. 1), including traditional forensic findings, allowing conclusions to be drawn about the associations between the material and people, places, events and production processes.

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Relevant Websites

<http://www.nf-itwg.org>—Nuclear Forensics International Technical Working group.

<http://www.gicnt.org/working-groups.html>—Nuclear Forensics Working Group.

<https://www.iaea.org/topics/nuclear-forensics>—IAEA, Nuclear Forensics.

<https://www.nuclearforensics.eu>—Forensics in Nuclear Security: Sharing Knowledge.

<http://nuclear-forensics.org>—Nuclear Forensic Science.

Cultural Heritage Preservation

P.A.S. Vasquez and M.L.E. Nagai, Nuclear and Energy Research Institute—IPEN, São Paulo, Brazil

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Abbreviations

AMS accelerator mass spectrometry
BCE Before Common Era
C14 carbon-14
CE Common Era
CH cultural heritage
 $D_{\max}$ maximum acceptable dose
 $D_{\min}$ minimum acceptable dose
EB electron beam
EPMA electron beam microanalysis
GC-MS gas chromatography mass spectrometer
Gy gray (unit of radiation absorbed dose; 1 Gy = 1 J/Kg)
HPGe High Purity Germanium (radiation detector)
IBA ion beam analysis
ICP-MS inductively coupled plasma mass spectrometry
IR infrared
ISO International Standards Organization
NAA neutron activation analysis
NDE non-destructive examination
NDT non-destructive testing
PGAA prompt-gamma activation analysis
PIGE particle induced gamma emission
PIXE particle induced X-ray emission
RBS Rutherford back scattering
SEM-EDX scanning electron microscopy with X-ray microanalysis
SIMS secondary ion mass spectroscopy
TL thermoluminescence dating

TOF-ND time of flight neutron diffraction
 TXRF total reflection X-ray fluorescence analysis
 μ -FTIR micro-Fourier transform-infrared spectroscopy
 UV ultra violet
 XRD X-ray powder diffraction
 XRF X-ray fluorescence analysis

Introduction

Cultural heritage is the legacy of physical artifacts and intangible attributes of a group or society that is inherited from past generations and maintained for the benefit of future generations. Protecting the world cultural heritage by maintaining the physical integrity of cultural assets has become a consensus for the entire technical and scientific community. Many analytical techniques used for examination, characterization, analysis, preservation, consolidation and conservation of cultural heritage artifacts, are grounded on ionizing radiations. Among these can be cited radiography using X-rays, gamma photons, neutrons or beta particles, X-ray fluorescence analysis, neutron activation and neutron prompt gamma analysis, ion beam analysis as well PIXE (particle induced X-ray emission), and synchrotron radiation-based characterization or analysis techniques.

The knowledge and understanding of the composition of materials and the processes of manufacturing of ancient or modern artifacts has been significantly improved by the application of scientific analytical techniques, which provides a scientific basis for conservation and restoration work and benefits preservation and conservation institutions, universities, museums, libraries, etc. being well recognized by a large public.

Treatments of cultural artifact collections for disinfestation and disinfection by radiation processing, or simply irradiation, which use ionizing radiation (usually gamma rays from cobalt-60 sealed sources) have been successful mainly in the tangible materials of organic raw material. Tangible cultural heritage materials include works of art, artifacts in museum collections, books, manuscripts, drawings, archive documents, musical instruments, ethnographic objects, archaeological findings, natural history collections, historical buildings and historical places, monuments and industrial heritage objects. Furthermore, the consolidation technique based on the impregnation by an appropriate resin inside of deteriorated objects made of dry or waterlogged wood, porous stones or other fragile materials, also uses gamma irradiation to promote a polymerization process. This technique uses organic polymers with high molecular weight such as polyethylene-glycol enabling to consolidate and to save deteriorated precious artifacts that can be presented to the public in cultural institutions. Commonly, before the consolidation process, the treated artifacts become more resistant to mold or insect attacks.

To this end, the IAEA has been collaborating with the community engaged in the preservation of cultural heritage through research programs, training of scientists, and especially with publications addressed to the development and application of analytical techniques and techniques based on ionizing radiation (radiation processing) for conservation of historical and cultural artifacts. Recently, the dissemination of this knowledge to curators, conservators/restorers, conservation scientists, and students has aimed to improve the global efficiency of the safeguard of our common cultural heritage (Fig. 1).

Ionizing radiation treatment of cultural objects and archived materials

Biodeterioration of tangible cultural heritage objects

Tangible cultural heritage objects are mainly made of wood, leather, textile or paper. Since these materials are basically natural polymers, they are vulnerable to environmental aggressors. Among the physical and chemical agents of deterioration found in the environments are water, oxygen, polluting gases from the atmosphere, acids, temperature, and light. However, the deterioration of artifacts is mostly due to the biological aggressors as well microorganisms and insects (Fig. 2).

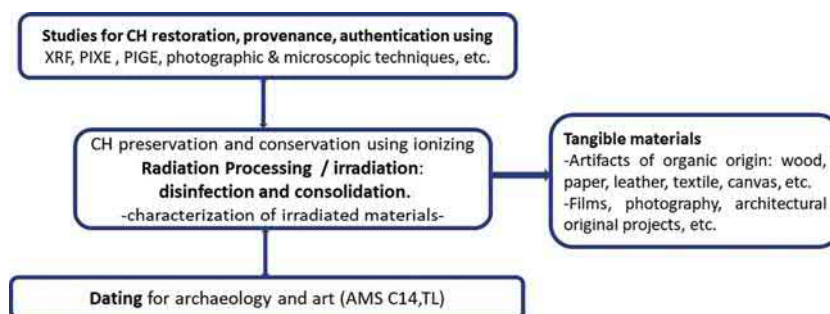


Fig. 1 Overview of nuclear techniques to characterization and preservation of cultural heritage objects and archived materials.

Natural disasters



Insects & Fungi attack



Fig. 2 Natural disasters promoting biological attack in tangible materials.

Degradation due to physical-chemical factors evolves slowly and seems to be dependent on noxious exposure intensity, while during an active biological attack the artifact degradation evolves with high speed. In the last case, the conservation prime target must be to stop the degradation. It is important to notice the synergism between improper physical—chemical conditions and the development of biological aggressors.

A preventive conservation plan, including the regulation of microclimate, can be efficient against physical and chemical degradation factors. Usually, controlling the environment can be sufficient to obtain control of biological factors, but only if their attack was not started.

Once the biological attack is underway, each object has to be treated, since the aggressors are present inside the artifact in the active biological form and take it as nutrient. Then, interventive conservation actions are necessary for disinfection because fungi, bacteria and insects have a complex life cycle, including dormant stages resistant at various biocides and unfavorable living conditions (IAEA, 2017).

Biodegradation takes place when insects are present. An example is provided by the ecological food chain involving termites. These insects are xylophages for excellence. They destroy the wood, but they do not have the intrinsic ability to break down cellulose. To overcome this lack of ability, termites carry microorganisms in their digestive tract that produce cellulolytic enzymes. The insects do not feed on the cellulose but on sugars resulting from the microorganisms' digestion of cellulose. In this way the termites profit from the metabolic by-products of the microorganisms.

Molds are microorganisms that are part of the kingdom of fungi and differ from members of the common plant kingdom by the absence of chlorophyll and the lack of ability to create energy from sunlight. Variation in the equilibrium of the moisture content of the material can initiate the development of mold in the substrate. The threshold of water content in the substrate for mold growth seems to be non-uniformly defined, and therefore different threshold levels of relative humidity can be found in literature: 50–60%, 65–70%, 70 and 75% (Thomson, 1986). They are incapable of synthesizing organic components from inorganic ones and, therefore, need carbon as a main source to grow. Most of the mold families are saprophytes (i.e. they are capable of deteriorating plant and animal substrates). The color of the mold is based on the substrate on which it grows. If the level of moisture in the environment drops, mold species are capable of entering a dormant stage in which they are inactive and are therefore able to survive (Great-house and Wessel, 1954).

Mold not only affects archival and library materials, but also affects occupational health through infections (mycoses), allergic reactions and toxic effects (mycotoxicosis). Some families are extremely poisonous and even carcinogenic. *Aspergillus flavus* is one such mold. In 1990 a deadly mold species was found in the cellars of the New Museum of Contemporary Art in New York. Once killed, mold can still be dangerous because of substances remaining in the substrate (Libert, 2013; Nyberg, 1987).

Considering biodegradation, mold, for example, causes an irreversible change to cellulose based substrates. The nutrient matrix for mold in cellulose materials is mainly the amorphous region that also contains polysaccharides such as hemicellulose. Owing to the presence of excess moisture, the fibers swell and become more attractive for mold (Fellers et al., 1989). The enzymes created by the mold deteriorate the cellulose. Mold may also produce radicals which, in the presence of transition metals, will form hydroperoxides (Ritschkoff and Mahlberg, 2001). Based on this progression, two treatments are demanded. First the mold has to be made inactive and then the material needs to receive a preventive treatment to stop chemical deterioration caused by the waste products remaining from the mold, for example removing the residue of disinfection process (e.g. irradiated mold).

Radiation sterilization

The biocidal effect of irradiation was first noticed at the beginning of the 20th century, immediately after the discovery of natural radioactivity. But irradiation treatment, especially for sterilization of medical devices, has only been used in industry for the last several decades and its use is growing. Radiation sterilization, an alternative recommended by the European Pharmacopoeia

(Council of Europe, 2014), has been used more frequently as a final sterilization method. Over 260 million cubic meters of products are sterilized each year using irradiation (GIPA, 2013). It is worth emphasizing that when irradiated under typical conditions (using gamma rays or electron beams (EB) with energies of less than 10 MeV), no radioactivity is induced in the products. Radiation sterilization of medical devices is now a well-defined industrial process, and a large body of knowledge exists on the treatment itself (Fig. 3).

Research conducted to study the influence of irradiation on living organisms led to establishment of a scientific specialty called radiobiology. Other studies were devoted to qualification and testing of materials exposed to radiation, especially plastics and natural polymers (cotton), as well as coatings and adhesives. Engineering studies allowing different various facility designs have been implemented with the objective of optimizing the cost–benefit ratio and improving the production yield, radiation safety or reliability. In addition, guides and standards have been developed covering safety design, installation and exploitation, quality assurance and quality control (IAEA, 2017).

Disinfection of cultural artifacts using irradiation

Irradiation is a transfer of energy from ionizing radiation to the irradiated material. Ionizing radiation has the potential to cause damage in living organisms if they are exposed to excessive levels. Experiments by Bletchly in the late 1950s involving gamma irradiation of xylophagous insects suggested that the biocidal effect of ionizing radiation can be used to stop biodeterioration of cultural heritage artifacts (Bletchly, 1961, 1962). It has been scientifically established that biological response is proportional to dose. The materials remain the same as they were before the irradiation treatment, only decontaminated. Treated materials do not retain radiation and, therefore, cannot irradiate people handling them. In addition to being used to disinfect cultural heritage artifacts, ionizing radiation (e.g. X-rays and gamma rays) is used in radiotherapy (medical) and for non-destructive testing of objects such as cultural artifacts (gamma radiography). The energy transferred modifies the material structure only at the molecular level. The irradiation effects depend on the chemical bonds existing in the irradiated material. In metals and inorganic materials, where there are either metallic or electrostatic bonds, the primary effect of irradiation is warming; in certain situations, color changes may occur. This effect is comparable to heating food in a microwave oven. In organic substances, characterized by covalent bonds, irradiation leads mainly to the breaking of the chemical bonds. The more complex the molecule, the more easily the bonds can break. Therefore, for example, bonds in DNA may be broken more easily than those in cellulose. Organic substances are present in both artifacts and their living contaminants; therefore, degradation is expected in both of them. To avoid degradation induced by irradiation process, the dose should be limited to a maximum of 10 kGy to destroy mold. For many materials, irradiation contributes less to deterioration than mold. The radiation dose (D) quantifies the transfer of energy during the irradiation. It is measured in units called gray (Gy). The multiple kGy (kilogray) is often used. The intensity of all disinfection effects and side effects, is dose dependent. The irradiation disinfection of cultural heritage is an acceptable intervention method, because doses for disinfection are lower than doses producing significant side effects.

Radiation processing has become a mature industrial branch involved in vital economic areas (medical, electrical, food industry, etc.). This increased confidence in the use of radiation treatment for decontamination of cultural heritage artifacts (Fig. 4).

Recommended treatment doses

There are two different of dose range levels that can be recommended for irradiation treatment related to disinfestation and disinfection. The dose for disinfestation of insects is lower than that for disinfection of fungi (e.g. mold). This is based on the complexity of the DNA structure of the species; complex DNA is more easily degraded than simpler DNA (IAEA, 2017).



Fig. 3 A typical panoramic, Cobalt-60 wet storage Multipurpose Gamma Irradiation Facility (IPEN, Brazil).



Fig. 4 Painting and ethnographic sculptures disinfected by gamma radiation at the Multipurpose Gamma Irradiation Facility (IPEN, Brazil).

Insects

In the case of an insect attack, the recommended treatment dose is around 0.5 kGy, but can be up between 2 and 3 kGy depending of the attacked material. This dose is also includes the eradication of insect eggs (Fig. 5). Treatment with 2–3 kGy can be recommended for furniture (wood) that has been stored in inadequate conditions of temperature and humidity for a not too long period. In uncontrolled environmental conditions, a fungal attack can occur simultaneously with an insect attack. Therefore, the range of dose between 2 and 3 kGy is too low for disinfection of all kinds of fungi that can remain and even form a basis for a new infestation by insects. Thus, the use of 2–3 kGy is not recommended if the artifacts are reintroduced after treatment into a non-controlled environment. There is a serious risk that a 2–3 kGy treatment may be ineffective in warm and wet climatic conditions. This is because the termites feeding system is based on a symbiotic relationship with certain microorganisms (IAEA, 2017).

Mold and overall treatment

A treatment with a maximum dose of 10 kGy can be considered as a reference dose for overall disinfection of cultural heritage artifacts. At this level of dose, eradication of fungi occurs. The dose applied to a batch of materials should be considered as an average dose. The final dose depends on the homogeneity of the batch density. Therefore, it is preferable that high density materials are not combined in the same batch with low density materials. Thus, if possible, materials should be classified into batches by density.

The larger the object is, the larger the difference between $D_{\min}$ and $D_{\max}$. Effective distinction should be ensured by $D_{\min}$, and $D_{\max}$ must not exceed the dose value at which side effects become unacceptable. Dose can be measured by means of dosimetry. Dosimetry can estimate the geometric parts of the artifact that get $D_{\min}$ and $D_{\max}$, respectively. The values $D_{\min}$ and $D_{\max}$ are dependent on each other for any given radiation geometry. The irradiation time is chosen in order to obtain the $D_{\min}$ (disinfection effectiveness).

Based on research, an average dose of 8 ± 2 kGy can be set for effective disinfection. This dose is sufficient to eradicate most fungi and will have minor side effects on the irradiated materials. Irrespective of these recommended doses, the conservator and the staff of the irradiation facility may decide to use another average dose. This dose can be lower or higher than the recommended dose (IAEA, 2017).

In the case of artifacts subjected to flooding, microorganisms—and fungi in particular—develop explosively in conditions of excessive moisture. This happens frequently in the case of artifacts that have been subjected to flooding and have remained after the flooding in an environment with a high moisture content. Also, the conditions of storage between the time of the flood and



Fig. 5 Large scale disinfestation of books by gamma radiation at the Multipurpose Gamma Irradiation Facility (IPEN, Brazil).

treatment may encourage the growth of mold. Therefore, a dose of 10 kGy has to be considered, and a dose higher than 10 kGy could be taken into consideration in special situations. Up to the present time, irradiation disinfection has been successfully applied to several cases to preserve materials that had contact with floods, war and poor storage conditions, and for very complex artifacts such as mummies (Fig. 6).

Consolidation of organic materials using radiation technology

The technique for consolidation of porous materials, such as wood or concrete, was introduced and carried out in the 1960s world-wide (the USA, Japan and Europe) by using the process of impregnating these materials by acrylic, vinylic monomers under pressure, and then their in-situ polymerizing or solidifying by gamma irradiation. It was in this period that wood plastic composites were developed, whose main industrial application was flooring in public areas. Very hard surfaces were obtained because the resin fills completely the wooden vessels, giving a densified wood that is much less sensitive to relative humidity.

Application in the field of cultural heritage was initiated in the 1970s in Europe. In 1970, the Grenoble laboratory took up the challenge to consolidate the 19th century mosaic parquet from the ancient Grenoble city hall by dismantling the wooden panels, which were impregnated by the monomer methyl-methacrylate (Fig. 7). During the late 1970s, the consolidation of many degraded wooden artifacts was carried out by using more appropriated radiation-curing resin based on unsaturated polyester and styrene. This latter approach was also implemented in conservation of waterlogged archaeological artifacts that require additional liquid phase exchange steps with the solvent acetone (IAEA, 2017).

Impregnation of wooden artifacts and irradiation

The resin impregnation in degraded wooden artifacts in dry condition is carried out in steel tanks suitable for vacuum and pressure applications. Inside the adapted tank, the artifact must be wedged on its support to avoid floating in the resin bath, and then a low vacuum is set up (around 1 mmHg) during several hours to extract the air from the wood pores. Liquid resin then fills the tank by



Fig. 6 The mummy of Ramses II disinfected by gamma radiation. IAEA (2017) *Uses of Ionizing Radiation for Tangible Cultural Heritage Conservation*. IAEA Radiation Technology Series no. 6. Vienna: IAEA.



Fig. 7 Left: parquet panel before consolidation treatment; right: impregnation of a series of parquet panels. IAEA (2017) *Uses of Ionizing Radiation for Tangible Cultural Heritage Conservation*. IAEA Radiation Technology Series no. 6. Vienna: IAEA.

vacuum suction until the complete immersion of the artifact in the resin bath. In order to ensure the diffusion of the resin in the core of the artifact, nitrogen pressure is then applied in the tank, in the range of 1–3 bars following the decay state of the wood. The duration is a few hours for thin artifacts to more than 24 h for large sizes. At the end of impregnation, the excess resin is flowed back to the storage tank for further use. This feature is one of the main advantages of the irradiation process: the resin, without any catalyst as mentioned previously, can be reused and stored for a long period at room temperature. Back to atmospheric pressure, the artifact is left to drain inside the tank until no more flowing of resin from the object. Out of the tank, the object is cleaned with textile to absorb any resin residue on the surface and then is wrapped entirely with textile and plastic film prior to irradiation. The artifact can be placed at 10 cm from the panel cobalt-60 source to start the in-situ resin polymerization after putting thin thermocouples inside the object to monitor the temperature, which must not exceed 50–60 °C.

The other advantage of radiation-curing is the temperature control by varying the dose rate. For instance, increasing the distance between the artifact and the panel source will result in lowering the wood temperature. This parameter is important for the behavior of the artifact in terms of potential dimensional variations of the surface area or its internal structure. Thanks to the penetrating power of the gamma rays, the polymerization is performed at each point on and inside the object, resulting in a homogeneous and complete reaction. During the first 48 h of irradiation, cleaning of the surface with a textile and the styrene monomer, plus putting new wrapping textile, are both crucial for the wood aspect without the minimum of residual resin on it or glossy surfaces.

The two faces of the artifact are exposed to the sources for the homogeneity of the absorbed irradiation dose, which is around 30–40 kGy after many days of treatment. This dose range is not harmful for the wood structure. Finally, the consolidated artifact is placed in a ventilated chamber during many weeks to eliminate any residual styrene monomer trapped in the object.

The quantity of resin absorbed by the wood increases its weight correspondingly and gives it a composite wood/polymer structure. The material obtained is hard throughout. Hence, its bulk and its mechanical strength are considerably increased. This gives greater resistance to abrasion and friction at the surface and improves solidity and resistance to shocks.

The densified wood is also unaffected by temperature variations and hardly sensitive to changes in climatic conditions when it is displayed or manipulated indoors. Impregnation darkens slightly the color of wood, depending on its species (broad-leaved species darken more than conifers).

Regarding polychromic sculptures, much care must be taken in testing to be sure of minimizing the interaction of pigment layers with the monomer or resin. The method is obviously avoided if any dissolution of pigment by the resin is detected. In some cases, protection of the pigment layer could be realized by wax application prior to impregnation.

Undoubtedly, this process, using insoluble cross-linked polymer and its maximum resin content, is at the opposite end of the conventional application of diluted solutions of polymers, which are theoretically reversible and film forming in the wood structure. Nevertheless, it could be considered as a useful method for the preservation of many degraded artifacts, the “last chance method,” because of the irreversible characteristic method (IAEA, 2017).

Advantages of irradiation techniques

The use of radiation techniques in the preservation of cultural heritage has specific and indisputable advantages over classical procedures. The first advantage is *harmlessness*. This should be highlighted and explained extensively to counterbalance public resistance to nuclear related areas in general. The nuclear domain is vast, and it is a great mistake, due to the misunderstanding of the application of nuclear energy, to judge radiography (which has undoubtedly improved medical diagnosis), Co-60 radiotherapy (one of the few ways to fight cancer), or sterilization by irradiation (a method that brought considerably cheaper medical devices).

Besides, there are no notable differences between sterilization, food treatment or decontamination of artifacts using ionizing radiation. In all cases, the same technology and the same irradiation equipment can be used. Radiation decontamination is performed in a confined, protected and well surveyed area. By design, such a facility can only be used under strict safety conditions. This technology does not leave any residue in the treated artifact or cause any damage to the environment. The artifacts do not become radioactive. Therefore, there is no risk for conservators/restorers, museum curators and registrars, or irradiation facility operators, and there is no risk to the environment. Side-effects in irradiated materials are avoided if recommended dose limits are set in the process.

Another important advantage is *effectiveness*, because gamma radiation penetrates any material and is effective up to its penetration depth, which depends on the density of the material and the quantity of the load, and the biocidal effect is controlled by a single processing parameter, the absorbed dose, commonly called dose. It can be confidently calculated, delivered at a known level, measured and certified. Radiation decontamination can be described as a process that is inherently effective. It does not depend on the material treated. In all other decontamination techniques, effectiveness is conditioned by the diffusion of a gas or the temperature, and these depend on the type and structure of the treated material. There is another practical consequence of radiation penetration: the artifacts can be irradiated without being removed from the package or container used for their transportation.

A further advantage is the *reliability* of the treatment. This is based on the fact that decontamination effectiveness depends only on the irradiation dose. There are international standards, developed by the International Organization for Standardization (ISO), for all dosimetry systems. Additionally, there is an obligation in industrial irradiation facilities to assure the traceability of dosimetry systems to an international reference laboratory. This is a key part of the certification of the quality management system. The composition and structure of the treated artifact do not influence reliability. It is the same for wood, paper, leather, parchment, textile and others, regardless of their degradation stage. The dose can be accurately calculated at different points in large objects. This makes it possible to irradiate oversized objects as effectively as smaller objects. As a method of remedial rather than preventive conservation, irradiation decontamination is applicable in emergency situations. Industrial radiation processing facilities (designed for

sterilization or food treatment) are best suited in such cases. Disinfection using ionizing radiation takes significantly less time than the classical methods.

Provenance and authentication of cultural heritage objects by radiation analytical techniques

Nuclear techniques have long been considered to be one of the most important research techniques for identifying works of art, due to their great sensitivity and the possibility of discovering features that are invisible to the naked eye (IAEA, 2011).

Provenance

The application of analytical methods to objects of cultural heritage must extract the minimum sample, in the case of destructive techniques, or no sample consumption, in the case of non-destructive techniques. In this case, optical methods, such as ultraviolet (UV) or infrared (IR) spectroscopy, and nuclear techniques became viable for application in research with cultural assets after the dawn of nuclear research reactors. Since the 1970s, neutron activation analysis (NAA) has been recognized as the method for investigating materials of archaeological provenance. This enabled the development of databases of archaeological artifacts, and scientists at the Brookhaven National Laboratory were pioneers in adopting multivariate statistical procedures for the study of ceramics and other archaeological materials (Bieber et al., 1976). The provenance presupposes that differences in the chemical composition between different natural sources overcome the differences within a given source. In this way it is possible to trace the artifacts by their source or to group artifacts from unknown sources (Weigand et al., 1977). In the example of sources being localized and easy to identify, such as obsidian, ceramic or clay, raw materials of known sources are usually characterized and, therefore, artifacts of unknown origin can be compared with the variation of the source group known. The tracing of materials to the place of origin, such as ceramics and clay paste, precious stones, basalt, marble, antique glass, copper and other precious metals, were determined using the trace element digital printing method through the application of a set of nuclear analytical techniques.

The prompt-gamma activation analysis (PGAA) is based on the interaction of a given material with neutrons, followed by the measurement of induced radioactivity. In general, irradiation is performed with thermal neutrons generated in a nuclear reactor and the radioactivity of each radionuclide generated is measured by gamma ray spectrometry. It is a non-destructive method with good precision and accuracy, useful for qualitative and quantitative multi-element analysis of the main elements, minor elements and trace elements, in archaeological materials. PGAA is particularly sensitive to light (low *Z*) isotopes, such as hydrogen (H) and boron (B), in addition to cadmium (Cd), mercury (Hg) and rare earth elements and also some bulk elements, such as carbon (C), nitrogen (N) and sulfur (S) can be quantified (Sajó-Bohus et al., 2006; Zoeldfoeldi et al., 2006).

Provenance investigations may rely on the complementary time-of-flight neutron diffraction (TOF-ND), which provides a direct and comprehensive examination of the structural properties of materials, such as ceramics, pigments, stones, marbles and metals, with information about the composition of phases, crystalline and magnetic structure, microspheres and texture, that is, whether or not the grains have a random or preferred orientation. The microstructure of the material is related to the properties of the material and the manufacturing process. Quantitative Rietveld analysis is used for data processing for phase identification, determination of the crystalline lattice, quantitative evaluation of mineral composition and texture analysis. In this way it is possible to understand the representation of the orientation of grains that have been changed in a sample body due to, for example, casting, rolling, hammering or heating of the production process. Therefore, if historical manufacturing techniques are known, texture maps can also help to distinguish genuine from fake artifacts (Grazzi et al., 2009; IAEA, 2011; Kockelmann et al., 2006).

Authentication

One of the problems that affects particularly the artifact trade and has a considerable economic impact is the forgery of cultural heritage objects. In this case, scientific investigations to study composition or age are the only definitive tools to clarify the authenticity of materials in a reliable way, since often the forgery perfectly resembles the original object. The methods already described so far have the potential to also help with authenticity investigations. Other methods, such as ion beam analysis (IBA) techniques, include particle induced X-ray emission (PIXE), particle induced gamma ray emission (PIGE) and Rutherford backscattering (RBS), allow obtaining data on the elemental composition of large objects without affecting the integrity of the materials. Although the analysis is from the surface, adjusting the energy of the proton beam allows information to be obtained in deeper layers of the material. National museums and research institutes, such as the Accélérateur Grand Louvre d'analyse élémentaire (AGLAE) facility at the Louvre Museum in Paris, France, the Laboratory of Nuclear techniques for the Cultural Heritage (LABEC) facility at National Institute for Nuclear Physics (INFN), Florence, Italy (Mandó, 2009) or the IBA facility at the Forschungszentrum Dresden, Germany, have exclusive accelerators for the investigation of cultural heritage and art objects. A detailed survey of available accelerator based analytical techniques was compiled by the IAEA in 2005 (Calligaro et al., 2004; IAEA, 2007, 2011; Jembrih-Simbürger et al., 2004; Mandó, 2005).

The need to prove the authenticity of cultural heritage objects arose from the spread of forgery of cultural heritage artifacts in museums and private collections. The demand for scientific techniques to identify fraud is related not only to the economic value of the cultural asset, but also to the historical, cultural, aesthetic and scientific meanings. In this sense, the techniques based on

nuclear energy certainly contribute to prove the authenticity of cultural heritage assets and are based on principles of forensic science (IAEA, 2011; Stanish, 2009).

Dating

A particular method for proving authenticity is carbon-14 dating. Radiocarbon dating depends on the continuous production of a radioactive carbon-14, or radiocarbon, by cosmic rays in the upper atmosphere. The isotope mixes with oxygen to form carbon dioxide, which is filtered to the biosphere and absorbed by plants, which are then consumed by animals. Carbon-14 (half-life of 5715 years) is lost by radioactive decay, but is balanced by constant production by cosmic rays. All living things, plants and animals, have the same concentration of carbon-14. However, when a plant or animal dies, its replacement of carbon-14 by the atmosphere is interrupted. The content of this isotope in the remains of fossils gradually decreases to the point of being undetectable. Still this technology can be used to date materials as old as approximately 50,000 years. Radiocarbon dating procedures measure accurately the carbon-14 content in various materials, and the results of carbon dating can be used to calculate the age of the object of study. The dating system is an indispensable tool in archeology, geology and other earth sciences. Currently, two procedures are used: a radiometric treatment and a treatment with accelerated mass spectrometry (AMS). Presently, some tens of Nuclear Physics laboratories worldwide work on radiocarbon dating, some of them being almost completely dedicated (Mandò, 2005).

For the AMS technique, an insignificant amount of the material sample is required to obtain the dating. This is important when thinking about dating analyses like The Dead Sea Scrolls and the Shroud of Turin. The Dead Sea Scrolls refer to about 1200 manuscripts first found in 1947 in caves on the hills of Qumran on the west coast of the Dead Sea. They range from small fragments to complete Old Testament books. In only a few cases, direct information about the date of writing was found on the scrolls. In all other cases, dating is based on indirect archaeological and paleographic evidence. To verify this evidence, analyses of AMS contributed to determine the radiocarbon ages of selected scrolls and proved the authenticity of the specimens as belonging to the period from 408 BCE to 318 CE (Bonani et al., 1992). The case of the Shroud of Turin was popularly attributed to the mantle that would have covered the body of Jesus Christ after his crucifixion. The first display of the mantle occurred in France in 1353. In 1578, the shroud was brought to Turin and in 1694 was placed in the Royal Chapel of Turin Cathedral. Scientific analysis using AMS attributed that the mantle was probably from medieval times (Damon et al., 1989).

Scientific studies of paintings

The starting point of a scientific study of a painting is to carry out analyses to identify the pigments. For this purpose, X-ray, IR and UV techniques are applied to the stratigraphic layers of the paints. The results obtained in sets with information on the painting technique support the appropriate decisions for conservation treatments.

X-ray radiography in easel paintings reveals the layers of paint, including elements of high atomic number, and allows visualization of the artist technique and allows the identification of the pigments used in the painting (Fig. 8). However, new research possibilities have been discovered with the use of neutron induced autoradiography.

Therefore, with the detailed study of easel paintings, it is possible to identify the materials and techniques used during the artist's creation process and subsequent restoration interventions. In addition, the step of removing the protective varnish, touch-ups and interventions, allows more accurate identification of the original painting (van den Berg, 2002; IAEA, 2011; Keune, 2005; Stanish, 2009; Van Der Weerd, 2002; Wyplosz, 2003).

An analysis of the pigments used in paintings is of the most importance in order to develop appropriate methods for their conservation, as it provides the detailed characteristics of the materials and enables their origins to be determined.

Studies on pigments used in easel paintings have been carried out for many years in various research centers around the world. These surveys contribute to the creation of databases and publication of manuals. (Eastaugh et al., 2004; FitzHugh and Harley, 2002; Genestar, 2002; Hochleitner et al., 2003). Examples of analytical methods used:

- a. X-ray powder diffraction (XRD), NAA, XRF, total reflection X-ray fluorescence analysis (TXRF), scanning electron microscopy with X-ray microanalysis (SEM-EDX), electron probe microanalysis (EPMA), secondary ion mass spectrometry (SIMS), micro-Fourier transform interferometry (μ -FTIR), thermal microscopy, PIXE, PIGE, Mössbauer spectroscopy, Fourier transform spectroscopy, inductively coupled plasma mass spectrometry (ICP-MS) and gas chromatography mass spectrometry (GC-MS)—for identification of organic substances (varnishes, mediums and organic pigments);
- b. Neutron autoradiography to determine the distribution of pigments and fillers used, and microchemical methods and microscopic methods (microcrystallography, microtexture analysis, petrographic analysis and mineral analysis) (Adriaens and Dowsett, 2006; Arbizzani et al., 2004; Demortier and Adriaens, 2000; Denker et al., 2017; Lill et al., 2002; Mandò, 2005; Respaldiza and Gomez-Camacho, 1997).

Currently, new methods of imaging are being developed that use different types of ionizing radiation, such as confocal X-ray microscopy, multispectral illumination methods, thermographic techniques, scanning acoustic imaging, air coupled ultrasound, Raman spectroscopy and imaging, non-destructive synchrotron X-ray diffraction, synchrotron based K edge imaging and others. The neutron autoradiography method continues to reveal a huge potential for studies of paintings. The basic drawback is the high cost of research and the need to have a nuclear reactor for research purposes (Fig. 9). (Calligaro et al., 2003; Denker et al., 2017; Dooryhée et al., 2005; Kanngießer et al., 2003, 2005; Krug et al., 2006; Maev et al., 2008; Mass et al., 2007; Woll et al., 2004).

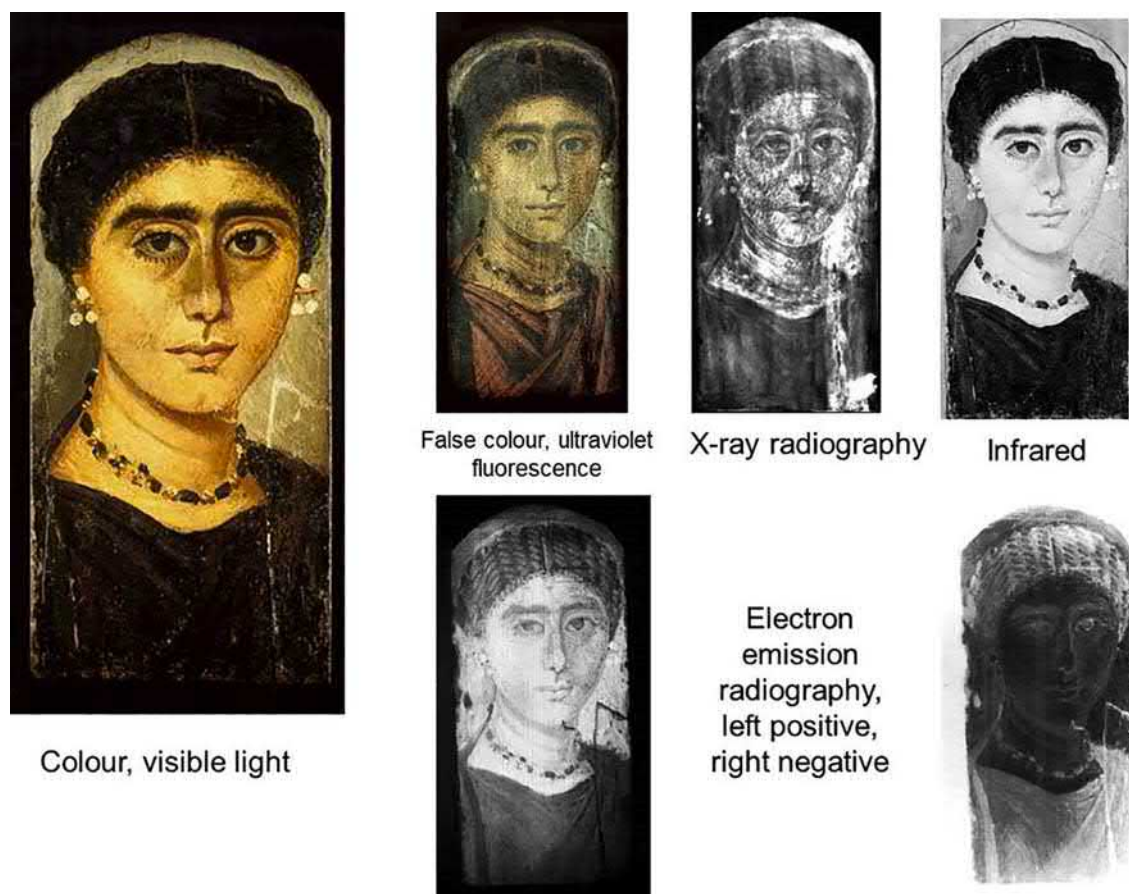


Fig. 8 Complete set of examination of a wax painting—Ptolemaic Egypt—Louvre Museum C2RMF. Boutaine JL (2017) A survey of the possibilities of various radiographic techniques for the non-destructive examination of cultural heritage artefacts. In: *International Conference on Applications of Radiation Science and Technology (ICARST-2017)*, Vienna, p. session B14.

Provenance of artistic materials

In the case of investigations related to the history of art, it is important to know the origin of the materials incorporated in the works of art, specifically in works of unknown authorship. This can be identified by measuring the concentration of trace elements and the isotopic proportions in the artistic material and in the respective supposed source material. The analyses have revealed that the isotopic relationships of lead from lead white used by artists before 1800 and, specifically, those found in paintings by Dutch, Flemish and Belgian masters, are limited (Fortunato et al., 2005; Keisch, 1971; Keisch and Callahan, 1976). Based on quantitative analyses to date of trace elements in lead white collected from paintings dating from the fifteenth to the nineteenth centuries, it can be concluded that lead white originating from regions south of the Alps, found inter alia in Venetian paintings, shows a higher content of copper and manganese and a lower content of silver and antimony compared with lead white from regions north of the Alps, mainly applied in Northern Europe (Fortunato et al., 2005; Keisch, 1971; Keisch and Callahan, 1976; Panczyk et al., 1992, 2000).

Nondestructive testing of cultural heritage objects

For cultural heritage artifacts or art objects and their component materials, the conservation scientist needs several non-destructive examination, characterization, and analysis techniques to improve understanding of their manufacture, their evolution and degradation during time. This understanding is necessary to give a rational basis for the restoration and conservation of objects.

Non-destructive testing (NDT) (or non-destructive examination—NDE), includes the scope of methods such as photography, endoscopy, radiography, holography, liquid penetrant testing, magnetic testing, Eddy current, infrared and thermal testing (thermography), acoustic emission, ultrasonic testing, leak testing, etc. These methods are widely used in industrial control and generally as complementary ones (Fig. 10).

Special attention should be paid on the different techniques that covers the general term radiography and their advantages to the study of cultural heritage artifacts.

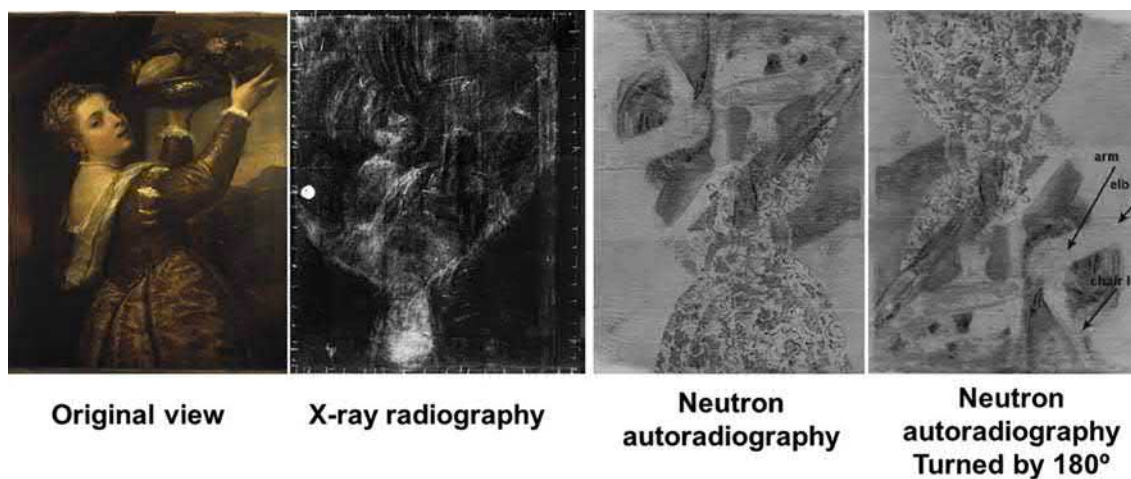


Fig. 9 Autoradiographs from « Girl with a platter of fruits » (c. 1555) by Tiziano Vecellio, called Tizian—Gemälde Galerie—Berlin C. Laurenze-Landsberg, C. Schmidt, C.O. Fischer, B. Schröder-Smeibidl—BENSC Report—Berlin (2001). Denker A, Laurenze-Landsberg C, Kleinert K, and Schröder-Smeibidl B (2017) Paintings reveal their secrets: Neutron autoradiography allows the visualization of hidden layers. In: *Neutron Methods for Archaeology and Cultural Heritage*, pp. 41–52. Cham, Switzerland: Springer.



Fig. 10 Complementary NDT methods for the study of the bronze casting techniques: direct visual examination, endoscopy, X and gamma ray radiography, ultrasound, computed tomography.

Radiographic examination for the study of a cultural heritage artifacts

From pertinent radiographs of a cultural heritage artifact (often taken from different incidences), it is possible to deduce various information related to:

- Determination of the internal structure of the object: hollow / fill, thickness and steadiness of the walls, homogeneity and heterogeneity of the constitutive materials, presence of structural defects (porosities, shrink holes, inclusions, cracks, slag, bubbles, voids, delamination, corroded or altered zones, etc.)
- Assembling features, nails, screws, welding, brazing, armatures, core pins.
- Repairs: inserts, secondary casting, screws, etc.
- Presence of eventual previous restorations.

These results may also provide pertinent information regarding the conservation state and arguments for eventual consolidation and restoration procedures (Fig. 11).

Neutron radiography

Neutron radiography is seldom used for the examination of cultural heritage artifacts, but it could be a complementary technique if one wants to look for organic materials inside a closed metal or stone object (Fig. 12). The difference between neutron and X-ray interaction mechanisms produce significantly different and often complementary information. While X-ray attenuation is directly dependent on atomic number, neutrons are efficiently attenuated by only a few specific elements like H, B, rare earths, Co, Li, In, Hg, Au, Ag. For example, organic materials or water are clearly visible on neutron radiographs because of their high hydrogen content, while many structural materials such as silica, chalk, aluminum, steel or lead are nearly transparent. The examination must be done in a thermal neutron beam extracted from a research nuclear reactor.



Fig. 11 Gamma radiography (iridium 192—Kodak Définix film) of the Gold mask of Toutankhamon, at the occasion of the exhibition «Toutankhamon et son temps»—Petit Palais Paris (1967)—© SGS—Qualitest. Boutaine JL (2017) A survey of the possibilities of various radiographic techniques for the non-destructive examination of cultural heritage artefacts. In: *International Conference on Applications of Radiation Science and Technology (ICARST-2017)*, Vienna, p. session B14.

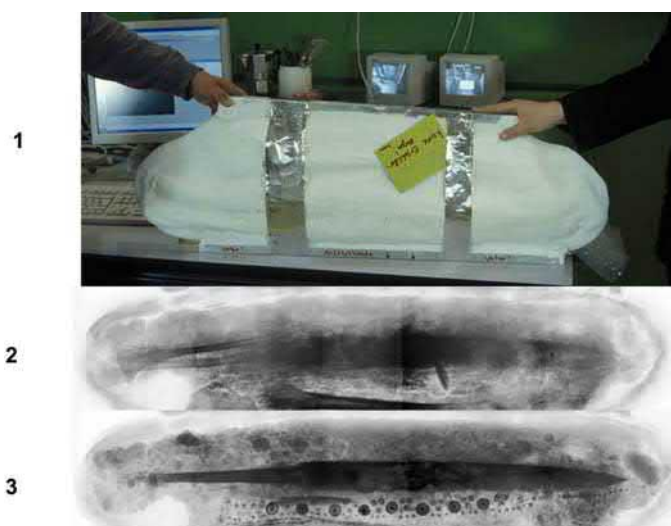


Fig. 12 Block excavation done in Kanton Zug prepared for radiographic examination. 2—Neutron radiograph showing bones & leather (PSI—Villigen). 3—X-ray radiograph showing sword, knife and metallic decoration. Boutaine JL (2017) A survey of the possibilities of various radiographic techniques for the non-destructive examination of cultural heritage artefacts. In: *International Conference on Applications of Radiation Science and Technology (ICARST-2017)*, Vienna, p. session B14.

Computer assisted tomography (densitometry)

Computed tomography is a powerful radiographic technique for producing 2D cross-sectional images (and eventually 3D images) of an object from flat X-ray images. The object to be examined is placed on a turntable stage that is between the radiation source (generally an X-ray tube) and an imaging system (an array of radiation detectors or a 2D pixelized detector). The turntable and the imaging system are connected to a computer. The instrument produces primarily a 2D shadowgraph image of the object like a film radiograph. Dedicated software makes it possible to produce cross-sectional images of the object as if it was being sliced and also reconstruction of its internal structure in 3D. This permits to characterization of the internal structure of an object such as dimensions, shape, internal defects, local density (Fig. 13).

In the cultural heritage area, computed tomography is used for the non-destructive examination and study of mummies, metal statues, archaeological objects, musical instruments, among others. Medical tomographs (scanners), industrial tomographs or dedicated instruments can be used for such purposes.

Microradiography and microtomography

This technique was based on the use of the plate of a scanning electron microscope. The very small focal spot (some mm in diameter) of the focused electron beam (up to 100–150 kV) striking an appropriate metallic anti-cathode (Cu, Ag, Rh, W, etc.) permits obtaining a sharp conical beam of nearly monochromatic X-rays. This beam permits to obtain, by direct projective magnification without loss of image sharpness, enlarged radiographic images of the absorption pattern of thin metallographic samples, and thus obtain precise information about the micro-structure of the studied alloy. The first application was dedicated to metallographic studies of refractory alloys for turbojet engine blades. Later on, companies like Pantak (UK), Feinfocus (DE), Kevex (US), Hamamatsu (JP) marketed dedicated microfocus equipment derived from this development. More recently, beams of mono-chromatic high intensity low energy X-rays have been instrumented at synchrotron machines, which permit similar examinations, but also microtomography to study 3D fine structures of samples of metals, stones, ceramics, wood, bone, fossil (Fig. 14).

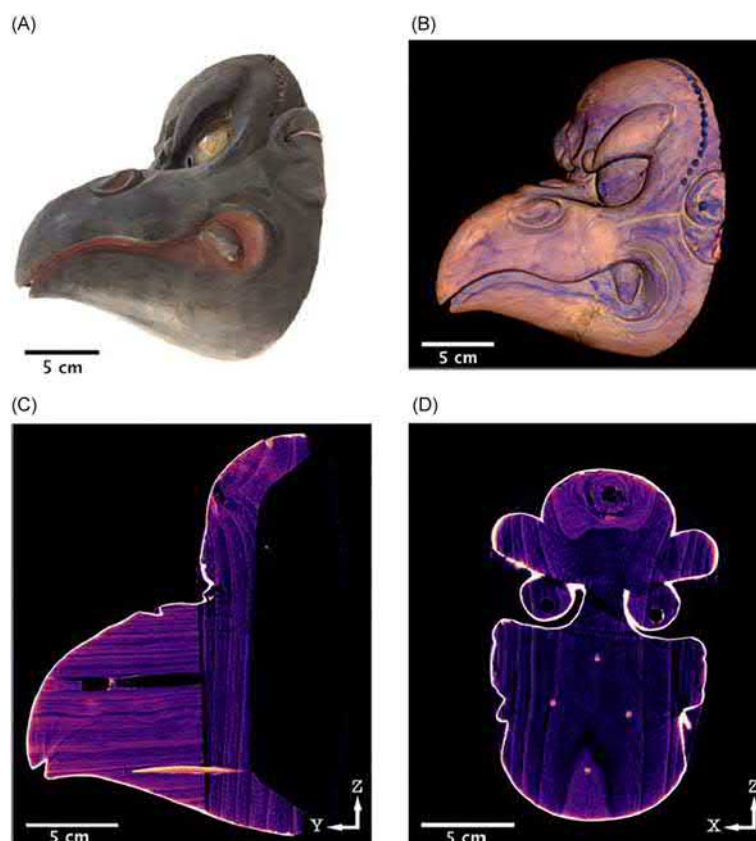


Fig. 13 Computed tomography of a Karasu Tengu mask—Kagura rituals—XIX century. (A) photograph of the mask, (B) computed tomography rendering (in false colors) of the entire volume. Tomographic slices (in false colors) showing the junction between the nose and the visage and the mallow removal (C), and the wood knot and the forehead (D). Albertin F, Bettuzzi M, Brancaccio R, Morigi MP, and Casali F (2019) X-ray computed tomography in situ: An opportunity for museums and restoration laboratories. *Heritage 2*: 2028–2038.

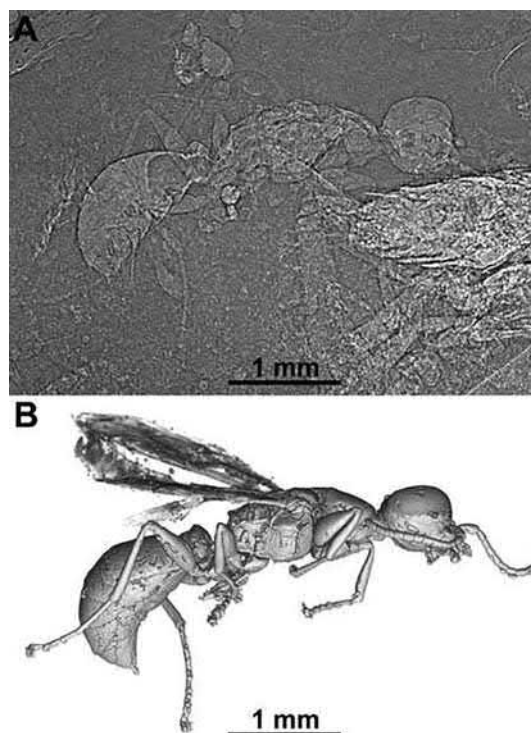


Fig. 14 (A) Phase contrast microradiography of a fossil Hymenoptera obtained with a 5.03-mm pixel size. (B) Result of 3D segmentation on a phase contrast microtomographic data set with an isotropic voxel size of 5.03 mm. Wings that were poorly visible on the radiograph were successfully extracted from 3D data due to the moderate phase contrast used for tomography. Lak M, Néraudeau D, Nel A, Cloetens P, Perrichot V, and Tafforeau P. (2008) Phase contrast X-ray synchrotron imaging: Opening access to fossil inclusions in opaque amber. *Microscopy and Microanalysis* 14: 251–259.

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Global Market for Radiation Applications

Roger H. Bezdek, Management Information Services Inc., Virginia, United States

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Glossary

CAGR Compound Annual Growth Rate
CT Computerized Tomography
FAO Food and Agriculture Organization
FDA Food and Drug Administration (USA)
IAEA International Atomic Energy Agency
IR Ionizing Radiation
MRI Magnetic Resonance Imaging
PET Positron Emission Tomography
SIT Sterile Insect Technique
SPECT Single Photon Emission Computed Tomography
TAT Targeted Alpha Therapy
THz Terahertz Radiation
WHO World Health Organization

Introduction

The use of nuclear technologies in modern society is widespread and pervasive, the economic benefits of radiation applications are enormous, and the global market for radiation applications is huge and is increasing rapidly (RadiationAnswers.org, 2020; [World Nuclear Association](#), May, 2017; [Lowenthal and Airey](#), 2001; “The Benefits of Nuclear Technology,” November–December, 1992). However, the benefits of these non-energy nuclear technologies – which include medical and other radioisotopes and radioactive materials – are not widely known or sufficiently appreciated. The standard of living for most people in developed countries would not be possible without nuclear technologies. They are ubiquitous and extraordinarily important in many ways, both large and small, throughout all aspects of life ([Waltar](#), 2004). In fact, Glenn Seaborg, one of the most prominent scientists of the 20th century, stated that even if not 1 W of electrical power had ever been generated from nuclear energy, all of the money invested in the nuclear energy field from the Manhattan Project forward would have been justified many times over by radioisotopes and radioactive materials ([Bezdek and Wendling](#), 1997).

Aside from the generation of electricity, nuclear technologies play a critical role in:

- Maintaining the safety and structural integrity of buildings, airplanes, and roads and bridges.
- Maintaining the health and quality of the foods and liquids people consume.
- Testing and improving motor vehicles.
- Improving health and saving lives, while at the same time reducing the costs of health care.
- Increasing crop yields and improving the health and productivity of farm animals.
- Reducing the costs of energy exploration and production, and increasing energy efficiency.
- Reducing the threat of terrorism in air travel, in public places, and in other venues.
- Controlling insect pests in an environmentally benign manner.
- Controlling and abating air, water, chemical, and solid waste pollution.
- Facilitating R&D breakthroughs in all fields of science, industry, and technology.

It is generally recognized that nuclear power plants produce electricity and that radioisotopes are used in research. However, it is not widely known that:

- One-third of the patients hospitalized every year in many nations are treated with nuclear medicine techniques.
- Many new drugs must be tested with radioactive materials before they can be approved by regulatory authorities and sold to the public.
- Worldwide, industry depends on radioisotopes and radioactive materials for measurement and automation, process development, quality control and testing, and cost reduction. In many cases, there are no feasible substitutes for these materials.
- Many common, widely used consumer products, such as smoke detectors, non-stick pans, and radial tires, require radioisotope methods for their development, production, or operation. For example, smoke detectors rely on a tiny radioactive source to function, and non-stick pans are treated with radiation to ensure that the plastic coating adheres.
- Radioactive materials are used in thousands of applications ranging from bridge and building construction to police work and anti-terrorism, from dating of archeological artifacts to development of agricultural crops.
- The use of nuclear technologies in chemistry has had as profound an impact in that field of science as use of the electron microscope has had in physics.
- Businesses in all fields of industry and commerce rely heavily on nuclear technologies to maintain and enhance their competitiveness in an increasingly competitive world.
- Nuclear technologies have been used to resolve two of the most controversial archeological disputes in recent decades: The Dead Sea Scrolls and the Shroud of Turin.
- Nuclear technologies have been used to estimate the age of the earth and to support the theory that a cataclysmic impact of a huge asteroid was responsible for extinction of the dinosaurs and the evolution of current species, including homo-sapiens.
- Nuclear medicine techniques permit the detection, treatment, and cure of breast cancer and prostate cancer and many other diseases without surgery.
- One-third to one-half of the food produced in the world is lost due to spoilage and infestation between production and consumption, and nuclear technologies can prevent most of this loss.
- Nuclear technologies permit the exploration of space, which would be impossible without small, radioisotope-powered generators.
- Use of nuclear technologies has reduced the number of patients in the USA treated annually using surgery for hyperthyroidism from 3000 to 50.
- Radioactive iodine is the most reliable treatment available for hyperthyroidism.
- Radiography is used to check the welds on virtually all new oil and gas pipelines and to examine the structural integrity of bridges.
- Radionuclides were key in determining the structure of DNA and in unlocking the genetic code.
- Radioactive materials are used to measure pollution in reservoirs and coastal aquifers.
- Pipeline leaks can be detected using nuclear technologies in a matter of days or weeks at a cost of \$40,000–\$60,000; alternative methods can take six months to a year and cost \$1 million, plus many millions of dollars in pipeline downtime costs.
- In recent decades the application of a single nuclear technique – tracers – in one industry – machine tools – has saved the USA economy nearly \$100 billion.

- Nuclear electron beam processing can eliminate the air pollutants that can cause acid rain and global warming, without producing harmful by-products.
- Solid wastes and sewage can be treated with nuclear technologies without using toxic chemicals.

Perhaps most important, in many cases there are no adequate substitutes for nuclear technologies at virtually any price.

The myriad uses of nuclear technologies

Radioisotopes, nuclear power process, ionizing radiation heat and non-stationary power reactors, and other nuclear technologies have numerous essential uses across multiple sectors, including consumer products, food and agriculture, industry, medicine, scientific research, transportation, water resources, and the environment – Fig. 1. These uses are discussed below.

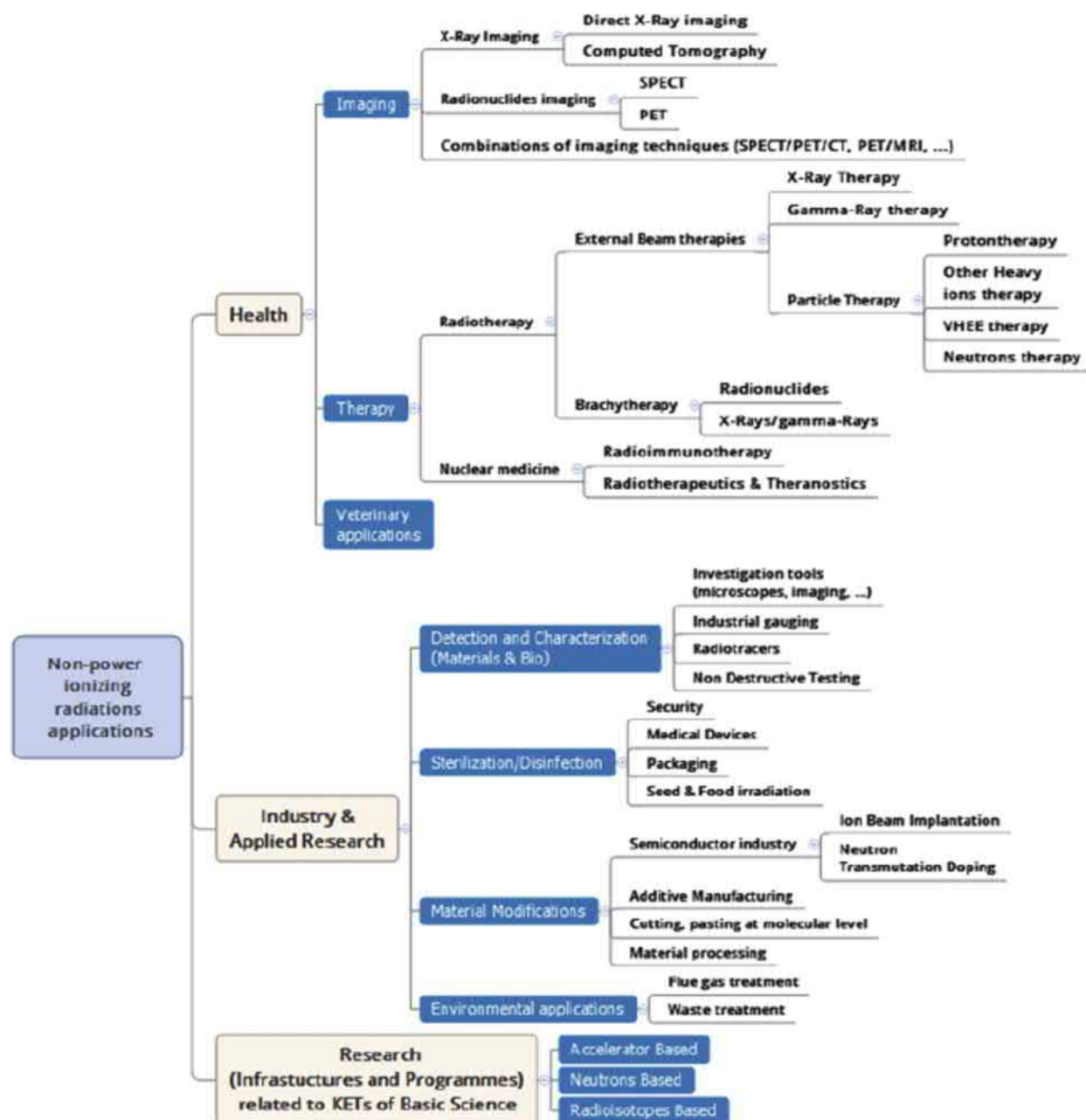


Fig. 1 Global summary of non-energy applications of ionizing radiation. From Kolmayer A, Mario M, Ligtvoet A, Van Barnevald J, Scholten C, and Davé A (2019) *European Study on Medical, Industrial and Research Applications of Nuclear and Radiation Technology*, Contract ENER/17/NUCL/SI2.755660, Luxembourg: Publications Office of the European Union.

Carbon dating

Analyzing the relative abundance of particular naturally-occurring radioisotopes is of vital importance in determining the age of rocks and other materials that are of interest to geologists, anthropologists, hydrologists, and archeologists, among others (American Chemical Society, 2019). Nuclear technologies play a critical role in these analyses.

Consumer products

The function of many common consumer products is dependent on the use of small amounts of radioactive material (World Nuclear Association, January, 2018). For example, smoke detectors, watches and clocks, radial tires, and non-stick materials, among others, all utilize the natural properties of radioisotopes in their design. One of the most common current uses of radioisotopes is in household smoke detectors. These contain a small amount of americium-241 which is a decay product of plutonium-241 originating in nuclear reactors. The Am-241 emits alpha particles which ionize the air and allow a current between two electrodes. If smoke enters the detector it absorbs the alpha particles and interrupts the current, setting off the alarm.

Environmental tracers

Radioisotopes play an important role in detecting and analyzing pollutants. Nuclear techniques have been applied to a wide range of pollution problems including smog formation, Sulfur dioxide contamination of the atmosphere, sewage dispersal from ocean outfalls, oil spills, and other pollution problems (Elliot, 2014).

Fertilizers

Fertilizers are expensive and if not properly used can damage the environment. It is important that as much used fertilizer as possible is “fixed” in the plant matter and that a minimum is lost to the environment. “Labeling” fertilizers with a particular isotope (e.g. nitrogen-15) provides a means of ascertaining how much has been taken up by the plants, allowing for optimal management of fertilizer use (Glibert et al., 2018).

Food irradiation

Approximately 25–30% of food harvested is lost as a result of spoilage before it can be consumed. This problem is particularly prevalent in hot, humid countries. Food irradiation is the process of exposing foodstuffs to gamma rays to kill bacteria that can cause food-borne disease, and to increase shelf life (Singh et al., 2013). In all parts of the world there is growing use of irradiation technology to preserve food. More than 60 countries worldwide have introduced regulations allowing the use of irradiation for food products.

In addition to inhibiting spoilage, irradiation can delay ripening of fruits and vegetables to give them longer shelf life, and it also helps to control pests. Its ability to control pests and reduce required quarantine periods has been the principal factors behind many countries adopting food irradiation practices.

Industrial tracers, inspection, and instrumentation

Radioisotopes are used by manufacturers as tracers to monitor fluid flow and filtration, detect leaks, and gauge engine wear and corrosion of process equipment. Small concentrations of short-lived isotopes can be detected, and no residues remain in the environment (Pandey, 2015). By adding small amounts of radioactive substances to materials used in various processes it is possible to study the mixing and flow rates of a wide range of materials, including liquids, powders and gases, and to locate leaks.

Radioactive materials are used to inspect metal parts and the integrity of welds across a range of industries. For example, new oil and gas pipeline systems are checked and tested by placing the radioactive source inside the pipe and the film outside the welds. Gauges containing radioactive (usually gamma) sources are in wide use in all industries where levels of gases, liquids, and solids must be assessed. They measure the amount of radiation from a source that has been absorbed in materials. These gauges are most useful where heat, pressure, or corrosive substances, such as molten glass or molten metal, make it impossible or difficult to use direct contact gauges.

The ability to use radioisotopes to accurately measure thickness is widely utilized in the production of sheet materials, including metal, textiles, paper, plastics, and others. Density gauges are used where automatic control of a liquid, powder, or solid is important, for example, in detergent manufacture.

Insect control

Crop losses to insects are significant. Despite the widespread use of insecticides, these losses total in excess of 10% globally, and are often notably higher in developing countries. One approach to reducing insect depredation in agriculture is to use genetically-modified crops, so that much less insecticide is needed. Another approach is to disable the insects.

Radiation is used to control insect populations via the Sterile Insect Technique (SIT) (Rossler, 2015). SIT involves rearing large populations of insects that are sterilized through irradiation (gamma or X-rays), and introducing them into natural populations. The sterile insects remain sexually competitive, but cannot produce offspring. The SIT technique is environmentally-friendly, and has proved an effective means of pest management and control even where mass application of pesticides had failed. The International Plant Protection Convention recognizes the benefits of SIT, and categorizes the insects as beneficial organisms (United Nations Food and Agriculture Organization, 2019).

SIT was first developed in the USA and has been used successfully for more than 60 years. At present, SIT is applied across six continents. Since its introduction, SIT has successfully controlled the populations of a number of high profile insects, including mosquitoes, moths, screwworms, tsetse flies, and various fruit flies (Mediterranean fruit fly, Mexican fruit fly, oriental fruit fly, and melon fly).

The most recent high-profile application of SIT has been in the fight against the deadly Zika virus in Brazil and the broader Latin America and Caribbean region (United Nations, April 19, 2018). Following its outbreak, impacted countries requested urgent support from the International Atomic Energy Agency (IAEA) to help develop the established technique to suppress populations of disease-carrying mosquitoes. The IAEA responded by providing expert guidance, extensive training, and by facilitating the transfer of gamma cell irradiators to Brazil. Three UN organizations – the IAEA, the Food and Agriculture Organization (FAO), and the World Health Organization (WHO) – with the governments concerned, are promoting new SIT programs in many countries.

Medicine

There is widespread use of radiation and radioisotopes in medicine, particularly for diagnosis (identification) and therapy (treatment) of various medical conditions (American Nuclear Society, 2014; Patton, 1993). In developed countries about one person in 50 uses diagnostic nuclear medicine each year, and the frequency of therapy with radioisotopes is about one-tenth of this.

Diagnosis

Diagnostic techniques in nuclear medicine use radiopharmaceuticals (or radiotracers) which emit gamma rays from within the body. These tracers are generally short-lived isotopes linked to chemical compounds which permit specific physiological processes to be scrutinized (Center for Disease Control and Prevention, 2014).

Depending on the type of examination, radiotracers are either injected into the body, swallowed, or inhaled in gaseous form. Emissions from the radiotracers are detected by the imaging device, which provides pictures and molecular information. The superimposition of nuclear medicine images with computed tomography (CT) or magnetic resonance imaging (MRI) scans can provide comprehensive views to physicians to aid diagnosis. An advantage of nuclear over X-ray techniques is that both bone and soft tissue can be imaged very successfully.

The most widely used diagnostic radioisotope is technetium-99 m, with a half-life of 6 h, and which gives the patient a very low radiation dose. Such isotopes are ideal for tracing many bodily processes with the minimum of discomfort for the patient. They are widely used to indicate tumors and to study the heart, lungs, liver, kidneys, blood circulation and volume, and bone structure.

Sterilization

Hospitals use gamma radiation to sterilize medical products and supplies such as syringes, gloves, clothing, and instruments that would otherwise be damaged by heat sterilization. Many medical products today are sterilized by gamma rays from a cobalt-60 source, a technique which generally is much cheaper and more effective than steam heat sterilization. The disposable syringe is an example of a product sterilized by gamma rays. Because it is a “cold” process, radiation can be used to sterilize a range of heat-sensitive items such as powders, ointments, and solutions, as well as biological preparations such as bone, nerve, skin, etc., used in tissue grafts (Finkiel, 2016).

There are significant benefits to sterilization by radiation. It is safer and cheaper because it can be done after the item is packaged. The sterile shelf life of the item is then practically indefinite provided the package is not broken open. Apart from syringes, medical products sterilized by radiation include cotton wool, burn dressings, surgical gloves, heart valves, bandages, plastic and rubber sheets, and surgical instruments.

Therapy

Nuclear medicine is also used for therapeutic purposes. Most commonly, radioactive iodine (I-131) is used in small amounts to treat cancer and other conditions affecting the thyroid gland. Cancerous growths are sensitive to damage by radiation, which may be external (using a gamma beam from a cobalt-60 source), or internal (using a small gamma or beta radiation source). Short-range radiotherapy is known as brachytherapy, and this is becoming the main means of treatment. Many therapeutic procedures are palliative, usually to relieve pain (Florida Medical Clinic, June 28, 2016).

A new field is targeted alpha therapy (TAT), especially for the control of dispersed cancers (Kozempel et al., 2018). The short range of very energetic alpha emissions in tissue means that a large fraction of that radiative energy goes into the targeted cancer cells once a carrier, such as a monoclonal antibody, has taken the alpha-emitting radionuclide to exactly the right places.

Plant mutation breeding

Plant mutation breeding is the process of exposing the seeds or cuttings of a given plant to radiation, such as gamma rays, to cause mutations (Beyaz, 2017). The irradiated material is then cultivated to generate a plantlet. Plantlets are selected and multiplied if they show desired traits. A process of marker-assisted selection (or molecular-marker assisted breeding) is used to identify desirable traits based on genes. The use of radiation essentially enhances the natural process of spontaneous genetic mutation, significantly shortening the time it takes.

Countries that have utilized plant mutation breeding have realized large socio-economic benefits. For example, in Bangladesh, new varieties of rice produced through mutation breeding have increased crops three-fold in the last several decades. During a period of rapid population growth, the use of nuclear techniques has enabled Bangladesh, and large parts of Asia in general, to achieve food security and improved nutrition.

Water resources

Adequate potable water is essential for life. However, in many parts of the world fresh water has always been scarce and in others it is becoming so. Isotope hydrology techniques enable accurate tracing and measurement of the extent of underground water resources (Barbieri, 2019). Such techniques provide important analytical tools in the management and conservation of existing supplies of water and in the identification of new sources. They provide answers to questions about origin, age, and distribution of groundwater, as well as the interconnections between ground and surface water, and aquifer recharge systems. The results permit planning and sustainable management of these water resources. For surface waters they can provide information about leakages through dams and irrigation channels, the dynamics of lakes and reservoirs, flow rates, river discharges, and sedimentation rates. Neutron probes can measure soil moisture very accurately, enabling better management of land affected by salinity, particularly in respect to irrigation.

Estimates of the sizes of the global markets for radiation applications

As discussed above, non-energy uses of nuclear technologies are widely used in health, industry, research, and many other areas. The markets for these technologies extends from the nuclear technologies and equipment themselves, to equipment servicing (maintenance, upgrades, training, etc.) and to health, industrial and research products, and services in which they are embedded, and where their specific added value is most often difficult to isolate. This makes evaluating and estimating the market particularly challenging.

Ionizing radiation (IR) technologies rely on charged particles beams (accelerators), X-Rays, or α , β , or γ -rays, and neutrons. It has been estimated that the Ionizing radiation (IR) technologies alone underpin nearly half a trillion dollars' worth of global commerce a year (Hamm and Kephart, 2017). More generally, the non-energy economic impact of nuclear technologies and applications is enormous. For example, the 2019 economic impact of these technologies is estimated to be \$532 billion in the USA and \$233 billion in Japan (Bezdek and Wendling, 1997; Bezdek et al., 2008; Waltar, 2004) (Economic impact estimates converted to USA 2019 dollars by the author).

The evaluation of the IR equipment market is somewhat more straightforward. The global market value of IR equipment is estimated at more than USA \$40 billion per year, with health applications being the most important non-energy sector – see Fig. 1. This global equipment market is attractive, with a high 3–6% annual growth rate and favorable export prospects (Kolmayer et al., 2019). The equipment market is also competitive, driven by constant innovation, which requires substantial investments.

Assessing the global economic impact of such diverse applications is difficult because they are generally embedded within products and services and manufacturing processes and research where their specific added value is difficult to estimate. Nevertheless, as noted, it has been estimated that ionizing-radiation applications of accelerators alone underpin nearly half a trillion dollars-worth of commerce a year, without taking into account their invaluable health benefits (Hamm and Kephart, 2017).

Estimates of the current size of the global markets for radiation applications are presented below.

Food irradiation market

The global food irradiation market currently totals USA \$250 million and is growing at a compound annual growth rate (CAGR) of 5% (McHugh, 2019; Hallman et al., 2016; Lacroix and Follett, 2015). This growth rate is due to increased consumer acceptance since the U.S. Food and Drug Administration (FDA) approved phytosanitary treatment of fresh fruits and vegetables by irradiation. Phytosanitary treatments disinfect traded commodities of potential quarantine pests, and phytosanitary irradiation treatments use ionizing radiation to accomplish this. The food irradiation market in Asia is also growing very rapidly owing to approval of government agencies in India and other countries. At present, over 40 countries have approved applications to irradiate over 40 different foods. More than half a million tons of food is irradiated worldwide annually. For, example about a third of the spices and seasonings used in the USA are irradiated.

Ionizing-radiation equipment market

Global market data are available for the ionizing-radiation equipment market (Kolmayer et al., 2019). The global annual market currently totals about USA \$40 billion and is growing at a CAGR of about 5%. The market consist of the following major segments – data presented in USA \$:

- Health – USA \$32 billion
- Industry (excluding health) – USA \$6 billion, excluding the consumer-products segment, which is likely to be significant
- Research – USA \$2 billion

Healthcare applications are probably the most important non-energy field to use ionizing-radiation tools. The market is driven by Asian markets in particular. Competition in these markets is fierce, with an increasing presence of USA and Asian companies relying on strong domestic markets. Ionizing radiation technologies are used daily in Europe in a number of fields, including health, industry, and research, with significant impact on the health of European citizens, the European economy, and Europe's international influence.

Magnetic resonance imaging (MRI) market

The Magnetic Resonance Imaging (MRI)-guided radiation therapy systems market currently totals nearly USA \$300 million and is increasing at a CAGR in excess of 20% (Eminent Research and Advisory Services, May, 2019). The MRI-guided radiation therapy systems industry has been underpinned by the emergence of advanced technologies in the healthcare and medical devices industries. Increasing adoption of radiotherapy in cancer treatment is fueling demand for MRI-guided radiation therapy systems.

Medical X-ray radiation protection glass market

The worldwide market for medical X-ray radiation protection glass currently totals about USA \$350 million and is forecast to increase at a CAGR of nearly 2% (360 Research Reports, 2019).

Medicine and radiopharmaceuticals market

The medicine and radiopharmaceuticals market currently totals about USA \$6 billion and, driven by the increasing number of successful clinical trials, is increasing at a CAGR of 9% (Fortune Business Insights, December 24, 2019). The market includes PET radiopharmaceuticals and SPECT radiopharmaceuticals, applications in neurology, cardiology, oncology, and other fields, and end users including hospitals, clinics, and diagnostic centers. The major growth factors driving this market include:

- Radiopharmaceuticals are substances that are used to diagnose specific medical problems or diseases, and increasing imaging capabilities and efficiencies have led to a wide product adoption across the world. Increasing numbers of successful clinical trials associated with radiopharmaceuticals are increasing demand for the product.
- Recent drug application area discoveries have showcased promise for the companies operating in the market, and technological advancements in nuclear imaging and their applications in diagnosis of cancer and other serious diseases have opened up a huge potential for growth.
- Growing awareness regarding the adverse effects of chronic diseases, and the need for early diagnosis, are facilitating market growth. The advancements in imaging systems have played a major role in the growth of the market.

The need for early and accurate diagnostic methods, coupled with increasing need for better therapies, is driving the market (Grand View Research, Inc. August, 2018). Increasing incidence of cancer and cardiovascular disorders is also contributing toward market growth, since nuclear medicine has proven to be critical in the diagnosis and treatment of such conditions (International Atomic Energy Agency, 2015). Medical exposures have grown considerably in the past several decades, mostly due to the marked increase in the use of CT scanning and other advanced imaging (United Nations Scientific Committee on the Effects of Atomic Radiation, 2008).

According to the World Health Organization (WHO), the number of new cases of cancer is expected to increase by around 70% over the next two decades (World Health Organization, September, 2018). Globally, about 10 million people die annually from cancer, and cancer accounts for one out of every six deaths worldwide.

Increasing prevalence of cardiovascular diseases is fueling market growth. These diseases account for nearly 20 million deaths annually worldwide. The number is anticipated to exceed 23 million by 2030. An estimated 31% of global deaths can be attributed to cardiovascular diseases (World Health Organization, September, 2017).

At present, radiopharmaceuticals are widely used in the field of oncology and cardiovascular diseases. Ongoing research and studies demonstrating positive results is widening the scope of radioisotope applications for diagnosis and treatment of bone diseases, respiratory diseases, thyroid-related diseases, and conditions of the digestive tract. In addition to these applications, radioisotopes are widely used in radiopharmacology to study drug movement in lab subjects.

Radiation cured products market

The radiation cured products market currently totals about USA \$8.5 billion and is increasing at a CAGR of nearly 7% ([Global Industry Analysts, Inc. April, 2018](#).) The global market for radiation cured products, including adhesives, coatings, inks, and electronics, currently exceeds 600,000 thousand metric tons and is driven by growing awareness of the numerous environmental and performance benefits offered by radiation curing. Also driving growth in the market are innovations in radiation curable chemistries and stringent environment safety norms that encourage environmentally-responsible curing of resins, inks, coatings, and adhesives. The adoption of radcure technology in a broad range of applications is attributed to several advantages, such as more rapid curing and processing time, lower volatile organic compounds emissions, improved physical characteristics, and cost-effectiveness. Construction, furniture, medical, and automotive industries represent attractive end-use markets for radcure technology. Europe represents the largest market worldwide. However, Asia-Pacific ranks as the fastest growing market led by strong Chinese demand for ultraviolet and electron beam curing technologies, which are rapidly replacing traditional methods in several applications such as wood coatings, industrial coatings, and electronics coatings.

Radiation detection, monitoring, and safety market

The global radiation detection, monitoring, and safety market currently totals about USA \$30 billion and is increasing at a CAGR greater than 4% ([Global Radiation Detectors Market by Manufactures, Types, Applications, and Forecast 2019-2024, 2019](#); [Radiation Detection, Monitoring, and Safety Market by Product, Composition, and Application, 2019](#)). The market includes products, such as detection & monitoring and safety equipment, gas-filled detectors, scintillators, solid-state detectors, and various applications in healthcare, homeland security, defense, and industry. The key factors driving the growth of this market include increasing security threats, growing prevalence of cancer worldwide, increasing safety awareness among people working in radiation-prone environments, growing safety concerns following the Fukushima disaster, growing security budgets of global sporting and entertainment events, growth in the number of PET/CT scans, increasing usage of nuclear medicine and radiation therapy for diagnosis and treatment, and use of drones for radiation monitoring.

Radiation dose management solution market

Radiation dose monitoring, dose management, dose recording, and radiation safety technologies include systems to reduce or block dose via barriers or computed tomography (CT) dose reduction technologies such as iterative reconstruction. The global radiation dose management solution market currently totals USA \$250 million and is increasing at a CAGR of over 14%. The market is driven by the increasing adoption of medical imaging equipment. In addition, the demand for cloud-based systems is anticipated to further increase the growth of the radiation dose management solution market.

Factors such as increasing applications of diagnostic imaging modalities and rising prevalence of various diseases are driving the need to conduct medical imaging examinations. In addition, the increasing focus on prevention and early diagnosis of diseases along with the availability of reimbursement in countries such as the UK and the USA will further drive the demand for medical imaging equipment such as picture archiving, communication systems, and radiology information systems. This in turn, will drive the adoption of radiation dose management solution, which captures, tracks, and reports the radiation dose directly from these imaging devices. Thus, the increasing adoption of medical imaging equipment is expected to drive market growth during the forecast period.

Market growth has been primarily attributed to the major drivers such as increasing prevalence of chronic and lifestyle associated diseases, increasing installed bases of radiology equipment, growing geriatric population, increasing regulatory requirements for diagnostic devices, increasing concerns related to radiation overexposure, and growing awareness and initiatives for radiation dose management. The market is expected to grow at a significant growth rate due to the opportunities that lie within its domain, which include technological advancement in medical imaging, impact of cloud-based solutions, increasing adoption of radiation dose management technologies in emerging technologies, and radiation dose management for pediatric procedure. However, there are significant challenges which are restraining the market growth. These challenges include lack of trained and skilled professionals, and lack of benchmarking for dose optimization ([Technavio, 2019a](#); [BIS Research, 2019](#)).

Radiation-hardened electronics market

Radiation-hardened electronics are electronic components, packages and products that have been designed and tested to provide protection against penetration radiation that may, if unimpeded, cause malfunction, damage circuitry, or cause the electronic device to shut down. The global radiation-hardened electronics market currently exceeds USA \$700 million and is increasing at a CAGR greater than 4% ([Global Industry Analysts, 2019](#)).

Radiation safety and protection and shielding applications market

The global market for radiation safety and protection and shielding applications currently totals about USA \$60 million and is increasing at a CAGR greater than 4% ([Technavio, 2019b](#); [BCC Research, June, 2019](#)). The market is driven by a number of factors,

including the increasing number of installations of radiology equipment. In addition, advances in radiation shielding equipment are anticipated to further increase the growth of the medical radiation shielding market.

The adoption of diagnostic radiology equipment is increasing with the growing number of imaging examinations resulting from the high prevalence of chronic diseases. Other factors such as the increasing number of radiologists and high demand for technologically advanced diagnostic radiology equipment are further driving the installation of the radiation equipment. This is increasing the need for medical radiation shielding to minimize the exposure to excess radiation and protect patients from acute health effects. Thus, the increasing number of installations of radiology equipment is a major factor driving market growth.

Radiopharmaceutical and therapeutics market

The global radiopharmaceutical and therapeutics market current totals about USA.

\$17 billion and is increasing at a CAGR in excess of 9% (Laxmi, 2019).

Radiation therapy market

The current global market for radiation therapy, also known as radiotherapy, exceeds USA \$7 billion and is growing at a CAGR of nearly 7% (MarketsandMarkets Inc. July, 2019; Radiotherapy Market by Type, Product, Application and End User, 2019). Radiation therapy uses ionizing radiation, generally as part of cancer treatment to control or kill malignant cells, is normally delivered by a linear accelerator and includes *Stereotactic, Brachytherapy, CyberKnife, Gamma Knife, Tomotherapy, Particle Therapy, and related applications*. Radiation therapy may be curative in a number of types of cancer, such as breast cancer and prostate cancer, if they are localized to one area of the body. Market growth is driven largely by factors such as technological advancements, the need for cancer treatments, the increasing number of conferences and symposia focusing on spreading awareness about the benefits of radiotherapy, and the growing use of particle therapy for cancer treatment. In addition, the increasing adoption of radiotherapy procedures for cancer treatment, increasing use of particle therapy for cancer treatment, and the rising number of conferences and symposia focusing on the advancements in radiotherapy are some of the other major factors driving the growth of this market. The emerging markets, growing government and private investments to meet the increasing demand for cancer treatment, and the improving reimbursement scenario are expected to present a wide range of growth opportunities for market players.

Terahertz radiation devices and systems market

Terahertz (THz) radiation can penetrate fabrics and plastics and is used in surveillance, such as security screening, to remotely uncover concealed weapons and explosives. The global market for terahertz radiation devices and systems currently totals USA \$170 million and is increasing at a CAGR in excess of 31% (BCC Research, August, 2019).

Summary

It is clear that modern society cannot function without nuclear technologies and that the benefits of these technologies are enormous and widespread. International markets for these technologies exceed USA \$500 billion annually and are growing rapidly. For example the global market for:

- Food irradiation is growing at a CAGR of 5%.
- Ionizing-radiation equipment is growing at a CAGR of 5%.
- Magnetic resonance imaging (MRI)-guided radiation therapy systems are growing at a CAGR in excess of 20%.
- Medicine and radiopharmaceuticals is increasing at a CAGR of 9%.
- Radiation cured products is growing at a CAGR rate of nearly 7%.
- Radiation detection, monitoring, and safety is increasing at a CAGR greater than 4%.
- Radiation Dose Management is growing at a CAGR rate of over 14%.
- Radiation-hardened electronics is increasing at a CAGR greater than 4%.
- Radiation safety and protection and shielding applications is increasing at a CAGR greater than 4%.
- Radiopharmaceutical and therapeutics is increasing at a CAGR in excess of 9%.
- Radiation therapy is growing at a CAGR of nearly 7%.
- Terahertz radiation devices and systems is increasing at a CAGR in excess of 31%.

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Introduction to the Social Issues of Nuclear Energy

Andrew C. Kadak, Kadak Associates, Inc., Port St. Lucie, FL, United States

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This section of Elsevier's Encyclopedia of Nuclear Energy is devoted to the "Social Issues" surrounding the technology. Perhaps no modern technology has been so misunderstood by the public. Initially conceived to address mankind's need for an abundant source of clean energy, its birth with the atomic bomb has for many affected people's attitude toward the technology. When President Dwight D. Eisenhower initiated the "Atoms for Peace" program at the United Nations in 1954, the hopes for the technology were high and aspirational. No one could have imagined how much the fear of radiation affected people's perceptions of the technology. This fear can be understood because of the low level of knowledge about radiation effects by the public and little connection with the contribution nuclear energy can make to de-carbonization of the climate.

Experience to date indicates that nuclear energy, if managed carefully, can be very beneficial and safe in comparison to other energy technologies. Nuclear energy has many attributes that render it a potentially important complement to other energy technologies that do not generate greenhouse gases. If nuclear energy is not considered in the portfolio of other clean energy sources, mankind will suffer great needless harm because of these unfounded but widely held perceptions (Golay, 2021).

This section will explore and hopefully shed light on some of the key social issues affecting the application of nuclear energy for peaceful purposes. Authors of the various sections of this chapter are experts in the areas they cover. The topics that are addressed are:

1. Factors Affecting Public Opinion of Nuclear Energy in the US

A 36-year program of research shows that the majority of the US public favors nuclear energy and this support is even stronger among nuclear power plant neighbors. This chapter draws primarily from that extensive and in-depth program of research on public attitudes toward nuclear energy by Bisconti Research. The long-term trends show that support for nuclear energy is much greater today than it was when the survey program began in 1983. Gender, education, and political affiliation continued to be associated with attitudes. Women are more uncertain about nuclear energy, not more opposed. A stigma attached to nuclear energy may be diminishing, as recent surveys show that the public's view of "public opinion" improved recently.

2. How Innovative Reactors Could Improve Public Acceptance of Nuclear Energy

Society's growing concern about climate change, desire to move more people out of energy poverty, and aim to bring equity to technology deployment have created an opportunity for new, innovative reactor designs, called advanced reactors, to contribute to the global clean energy transition. To realize this opportunity, however, there is a strong need for innovation across the technology space, within social frames, in organizational structures, and in cultivating leadership. There are many reactor designs to choose from, with different types using different technical approaches to resolve the challenges the designers think are most important—including political and social challenges. Each community and country must choose for themselves what they value since the answer is unlikely to be the same everywhere. This chapter examines the case for advanced reactors, summarizes the current state of development, and assesses how technical and social systems must be considered collectively for the best outcomes.

3. Relative Nuclear Energy Health Hazards and Popular Acceptance

Experience to date indicates that nuclear energy if managed carefully can be very beneficial and safe in comparison to other energy technologies. Yet it is perceived by many as being unnatural, highly threatening, and too dangerous to be used. In wealthy democracies this perception has become common over the past 40 years, and experience to the contrary seems not to be able to budge it. This can be understood in terms of the weak state of knowledge of existence of natural radiation on Earth and of low dose radiation health effects and the public perception of high hazard combined with absence of important benefits provided by nuclear technology. Whether these views will cause nuclear energy to be omitted from the technological portfolio to be used in response to impending climate change may be very important in determining how harmful global warming becomes. Nuclear energy has many attributes that render it a potentially important complement to other energy technologies that do not generate greenhouse gases. It is plausible that mankind will suffer great needless harm because of these unfounded but widely held perceptions.

4. Securing Nuclear Power's Contribution to Decarbonization

The results of the MIT study on *The Future of Nuclear Energy in a Carbon-Constrained World* are summarized in this chapter. These results of an electricity generation capacity and dispatch optimization model under carbon constraints demonstrate the opportunity for nuclear across various regions with different renewable resource endowments and costs of nuclear. Nuclear power contributes a firm, dispatchable element to a portfolio of low carbon technologies, which lowers the cost of achieving decarbonization targets. However, high construction costs of nuclear plants in western Europe and the U.S. have undercut this

case. Pinpointed are the key contributors to high cost and identify opportunities for reducing costs. Finally, there is a review of the role for government policy in enabling nuclear energy to contribute to decarbonization.

5. Economics of Nuclear Power vs Other Energy Sources

Two types of costs, total cost and levelized cost of electricity provide much of the economic information necessary to compare competing electricity generating alternatives. Levelized cost is important because it suggests whether the nuclear power plants can compete in the long run against other generating technologies. These other technologies include combined cycle natural gas, coal, hydro, and renewables. This chapter calculates and compares the levelized costs of these electricity generating sources. Also, electric utilities, when making investment decisions, must consider how a specific generating unit interacts with its portfolio of generating assets.

6. Impact of Small Modular Reactors on the Acceptance of Nuclear Power by the Public, Investors, and Owners

The acceptability of nuclear power has varied widely throughout its 60-year history influenced both by externalities, such as regional economic strength, availability of energy alternatives, and political biases, and by the successes and failures of the nuclear industry itself. The nuclear industry is on the verge of introducing a new paradigm in the delivery of nuclear power: small modular reactors (SMRs) that promise to improve on all aspects of design, engineering, construction and operation of commercial nuclear power plants. If fulfilled, these promising benefits can have a significant impact on the acceptance of nuclear power to all stakeholder groups, including the general public, financial investors, and plant owners.

7. The Environmentalist's Dilemma

The environmental movement is facing difficult decisions in the years ahead if they want to maintain their credibility relative to the dangers of global climate change. On the one hand, they aggressively proclaim the dangers of global warming due to manmade increases in CO₂ levels and on the other, reject one of the largest non-CO₂ electricity generating sources – nuclear energy. As a proven performer for over 50 years, nuclear generated electricity is produced by over 440 nuclear plants world wide. Nuclear energy combined with solar and wind can be a winning combination in the battle against global warming. This chapter highlights reasons why major environmental organizations oppose nuclear energy; identifies groups and people who have switched their positions in support of nuclear energy; highlights positions of environmental organizations who support nuclear energy and finally provides environmental and public health reasons for these groups to support nuclear energy. It is hoped that this article will help those who feel uncertain about supporting nuclear energy reasons for actively supporting it as a resolution to the “environmentalist's dilemma.” We conclude with a sentence from top climate scientists in their 2013 letter to the environmental community: said “continued opposition to nuclear power threatens humanity's ability to avoid dangerous climate change.”

8. Nuclear Waste Can Be Disposed of Safely

The disposal of nuclear waste is one of the key issues facing the use of nuclear energy. Many question the feasibility and safety of disposal for tens of thousands of years. Scientists and engineers have recommended deep geological disposal as a final solution. The goal is to keep the radioactive materials safely in geological structures and away from our surface environment. Nature provides us with evidence that geological disposal is possible based on natural reactors that operated several billion years ago. Thus far, there is one geological repository for nuclear waste operating in New Mexico since 1999 at the Waste Isolation Pilot Plant. The United States proposed repository in Yucca Mountain, Nevada was politically cancelled after 20 years of study and the submission of a license application to the Nuclear Regulatory Commission in 2008. The results of the studies and NRC review provide the data and analysis that shows the repository can be safely operated for at least 10,000 years. International efforts in Europe also are making progress with Finland scheduled to open their geological repository which is now under construction in 2023 after a vigorous technical safety review. Thus, to answer the question can nuclear waste be safely disposed, the answer is yes. The big challenge is overcoming political opposition and public concerns.

9. The Sustainability of Nuclear Energy

One can have confidence that uranium and thorium will not be a limitation on the sustainability of nuclear energy. We do not know today the exact locations of the ore bodies containing future sources of uranium and thorium, but the global measurements of supernova nucleosynthesis, geoneutrinos and thermal gradients discussed in this article indicate that those sources are very likely to exist. Furthermore, it is not necessary that we know those precise locations today since the fuel itself will not be needed for several centuries. The geoneutrino measurements indicate that at least 25% of global uranium and thorium are located in the continental crust and the thermal gradient data indicate that these elements are within a few kilometers of the surface. The discoveries of unexpected IOCG deposits indicate that multiple modes for the concentration of uranium exist. The development of chelating ligands for the extraction of uranium from seawater is not yet economical compared to existing mines and *in situ* extraction on land, but the technology places a ceiling on future uranium prices and assures a widely accessible source. Although the quantity of uranium in the oceans is immense, hundreds of times presently identified resources, the oceans are being constantly replenished by rivers which carry dissolved uranium from the continental crust.

10. Non-Proliferation Aspects of Nuclear Energy

In December 1942, the Chicago Pile-1 (CP-1) demonstrated the feasibility of self-sustaining nuclear chain-reaction. However, the public realization of the enormous power of nuclear energy was not until the detonation of a nuclear weapon in August 1945. Ever since the dawn of the nuclear age, the potential for peaceful uses of nuclear energy often raised concerns that the potential spread of nuclear weapons would be catalyzed by a civilian nuclear power program. There is also the possibility of theft of fissile material or nuclear technology by non-state entities or terrorists. Though nuclear power reactors and nuclear

weapons operate under different conditions, they rely on a similar scientific basis and use similar materials and technologies. This article discusses some of the activities and characteristics related to curtailing and deterring proliferation associated with worldwide deployment of nuclear energy; that is, the non-proliferation aspects of nuclear energy from the perspective of policy, regulation and design.

11. Public Images of Nuclear Energy

Radioactivity and nuclear energy were entangled from the outset in an ancient web of associations: cosmic secrets of transmutation and life-force, mad scientists with death rays and monsters, apocalyptic destruction, a future golden age. Fear of nuclear war reinforced the links to psychologically profound symbols. The anti-nuclear-bomb movement taught that any trace of a radioactive substance is horrible contamination and authorities are untrustworthy, convictions that carried over into the anti-nuclear-reactor movement. These associations have lingered, less consciously, into the present. Ameliorating excessive public fear of nuclear radiation requires sophisticated communication techniques.

12. Societal acceptability of geological disposal: a crucial issue—See Chapter 00622

13. Nuclear Waste Can Be Reduced by Recycling and Transmutation

This article first reviews the problem of radioactive waste (RW) in particular the problem of long-lived high-level radioactive elements contained in spent nuclear fuels from nuclear reactors. In this respect, it discusses the various possible options for their management and analyses more particularly the option of reprocessing these spent fuels that allows the recycling of the nuclear materials recovered after this reprocessing operation (uranium and plutonium). Main conclusions of this paper are reprocessing makes sense from the standpoint of recycling because it allows a much more efficient use of natural uranium to produce energy. Reprocessing is important to high level nuclear waste disposal since it greatly reduces their volume and puts them in a more stable configuration making it more difficult to release into the environment. Additional steps can be taken to make risk of disposal even lower by partitioning and transmutation of minor actinides and certain fission products. Regardless of all the above, a repository for high level nuclear waste is still needed whether for once through or reprocessed waste.

14. Floating Nuclear Plants to Improve Economics, Safety, Siting and Proliferation Resistance

Floating nuclear plants have been around for decades, first powering nuclear submarines and aircraft carriers. The first “commercial” application was in 1967 when a reactor was placed in a converted liberty ship in 1967 to provide power to the Panama Canal. Today many countries are developing commercial nuclear plants to provide power to remote areas as well as grid power for nations that may not have the technical infrastructure to support nuclear energy. Many designs are being developed from subsurface submarines without propulsion to floating barges to offshore floating platforms. These concepts attempt to address many issues some of which are social, such as public acceptance while others are more economic, in that it is expected that factory fabrication in a shipyard using modularity principles can lower overall cost and improve quality. Concerns about safety, security and non-proliferation aspects of floating nuclear plants are addressed showing numerous improvements over land based nuclear plants.

15. The Case for International Fuel Cycle Centers

Expansion of nuclear energy is a part of the world’s battle against climate change. The International Panel on Climate Change recognizes the importance of nuclear energy as a contributor to energy supply. A key element in the production of nuclear energy is the “fuel cycle” which is what is required to make nuclear fuel and treat used nuclear fuel. The fuel cycle includes mining, milling, enrichment and fuel fabrication in the “front-end” and used fuel management in the “back-end.” In future nuclear energy system the “back-end” involves recycling the used nuclear fuel to extract the remaining useful materials for reintroduction into the reactors, and to process and dispose of the nuclear waste. In the recycling process, the used fuel needs to be chemically separated to extract the uranium, plutonium (and possibly minor-actinides) for reuse. This extraction process and enrichment is a proliferation risk since these materials can be used to make nuclear weapons. The goal of the non-proliferation regime is to prevent such diversions while not denying countries the benefit of using nuclear energy. International fuel cycle centers have been proposed to address this problem. These centers would be centrally located around the world to enrich natural uranium, receive used fuel, reprocess it and recycle it into new fuel and embed the left-over nuclear waste in a stable glass form for geological disposal. Such a center would avoid the need for each country to develop enrichment plants which are a direct line to a nuclear weapon. It would also avoid the need for building reprocessing plants and help address nuclear waste disposal by creating a greatly reduced and stable waste form. The concept calls for the host nation to have the necessary skill set and infrastructure to prevent diversion. The developing nation seeking the benefits of nuclear energy would purchase fuel cycle services from the center, ship the used fuel to the center for recycling and have the nuclear waste returned to them for disposal or to a centralized disposal facility as discussed in chapter “The Case for International Fuel Cycle Centers.” If such a scheme was to be implemented, the proliferation risk would be greatly reduced. France has established a small version of such a center with its complex at LeHague which is reprocessing used fuel from France and Japan for reuse. The success of the international fuel cycle centers depends on security of supply not subject to political disruptions.

16. The Case for International Waste Disposal Centers

There have been several proposals for regional and international repositories for disposal of high-level nuclear wastes, and in 2003 the concept received strong endorsement from the head of IAEA. The European Commission is funding studies to assess the feasibility of European regional waste repositories. Arising from these studies, 14 EU countries resolved to set up a European Repository Development Organisation (ERDO) to collaborate on nuclear waste disposal. A similar initiative is under way for the Middle East and North Africa. In May 2016 a high-level commission in South Australia recommended establishment of an international repository there. Since 2009 the International Framework for Nuclear Energy Cooperation (IFNEC) has focused

on possible multinational repositories for their safety, security and environmental benefits, especially for the approximately 20 countries with less than 3000 tonnes of used fuel in storage where they would be most appropriate. However, in 2019 IFNEC suggested that it was not timely to develop a multinational repository until national repository projects were more advanced.

17. Mankind Benefits of Nuclear Energy and Radiation

There are multiple uses of nuclear energy and radiation. They may be broadly classified into those using heat, and those using neutrons directly. The main heat application is electricity, but industrial process heat has enormous potential and desalination is already operational. Marine propulsion is an important application of heat from nuclear fission, and space exploration is a small niche application. Neutrons are supplied by research reactors for a variety of uses, most well-known being making radioisotopes for medicine and industry. Radiation from research reactors, synchrotrons and cyclotrons may be used directly for diverse purposes, and that from radioisotopes made in them has immense application in nuclear medicine both for diagnosis and therapy, and in industry. As the hydrogen economy takes shape, nuclear power will have a role in producing it without CO₂ emissions.

18. Micro Reactors for Decentralized Power Sources

Dozens of developers are working on very small modular reactors, referred to here as microreactors, with sizes ranging from 0.5 MWe–50 MWth. At one-20th to one-2000th of the size of a typical light-water reactor under construction globally, these new microreactor concepts offer novel benefits and challenges. Near-term applications include off-grid and islanded micro-grids, such as defense installations, industrial sites, and arctic communities. While these microreactors may appear expensive, they could provide affordable power to off-grid diesel-dependent sites. Unique risks that need to be further studied include how to transport fueled reactor cores, and the security implications of autonomous or minimally staffed nuclear reactors.

As can be seen by the wide range of topics covered, the expectation is that the reader will gain a better appreciation of the contributions that are being made by nuclear energy and the key topics of public concern including nuclear waste disposal, proliferation, sustainability, climate change and the prospects for future nuclear innovations.

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Public Images of Nuclear Energy

Spencer Weart, American Institute of Physics, College Park, MD, United States

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Introduction: Images and memory

Civilizations have long memories. The ideas that circulate today, the stories, the pictures, the very words we use have been shaped over generations extending back into prehistory. Much of this is embodied in images, which can be represented in pictures or in words or in primitive concepts that barely reach consciousness. This imagery has a profound effect on our thinking, outweighing rational deliberation more often than we would like to think. Old images weigh with particular force on how people think, and feel, about nuclear energy (Weart, 1988, 2012).

Images associated with nuclear energy do not align strictly with rationality and facts. The familiar Pavlovian association can connect any two things, for example the ringing of a bell with food. The link is often forged by simple repetition. Our brains can also form a durable association from even a single incident when a strong basic emotion such as terror is involved, as may sometimes be seen in post-traumatic stress disorder. This is especially common for disgust, a very ancient mechanism. You may have a lifetime antipathy against something that revolted you on a single memorable occasion. Of course associations are often based on genuine facts. It is sensible for a cat to salivate when it hears the can opener, and if a bird is nauseated after eating a single monarch butterfly it is quite right to avoid similar butterflies in future. But as with Pavlov's dogs, an association can be created by any things that happen to occur together, independent of a meaningful relationship. And once an arbitrary association is entrapped in the sticky web of culture it can become permanent. Thus Christians celebrate the birth of Jesus by gathering around an ever-green tree on December 25th, obedient to associations of new life with greenery and the solstice that are much older than the Bible.

Nuclear energy before Hiroshima

In 1898 Marie and Pierre Curie announced the discovery of radium. The new element emitted radiation that manifested a new form of energy: atomic energy, later called nuclear energy. Other scientists soon declared that atomic energy was not only new but of cosmic import. These atoms had been emitting rays inexhaustibly since the creation of the universe! The press reported the scientists' statements and exaggerated them, yet nothing could seem more sensational than the facts. Then in 1903 Ernest Rutherford and Frederick Soddy announced that the emission of rays by radioactive elements signaled a transformation of one element into another. "Rutherford," Soddy exclaimed at the moment of discovery, "this is transmutation!" Soon everyone within reach of a newspaper was reminded of the notorious goal of alchemists far into the past. Thus radioactivity became bound into an ancient web of images—in fact one of the most ancient and profoundly meaningful webs of associations found in human culture.

Many of the old alchemists had sought to change one chemical element into another, usually lead into gold, but "transmutation" meant more than that. The central goal of alchemy was to create the Philosophers' Stone, which had powers far beyond the transformation of metals. The Stone embodied the deepest secrets of creation. Nothing could be easier than to associate this with radioactivity and atomic energy, for physicists hastened to tell a fascinated public that the new energy manifested in their laboratories was exactly a profound secret of nature. The potential seemed limitless.

When people learn of a new physics discovery the first practical application they look for is usually in medicine, and radioactivity followed the rule. Pierre Curie was the first to report that radium rays (like another recent discovery, X-rays) could cause serious damage to living flesh. This was promptly turned to advantage in attacking some forms of cancer. But journalists, and even some speculative scientists, would not stop there. Might atomic energy, so potent and mysterious, represent a fundamental life-force, the energy of creation itself? The idea had long since been incorporated in ideas about radiation, for since the spread of agriculture if not earlier the brilliant rays of the Sun had been recognized as a basic source of vitality.

This fitted with curious ease into the web of transmutation imagery. The Philosophers' Stone was also the Elixir of Life, the arcane force that could transmute inert dead matter into living things. Turning lead into gold in their crucibles was only a first symbolic step for some of the premodern alchemists. If they could break down matter into its primeval formlessness and recreate it in a perfected condition, so might they renew living flesh. The goal was rebirth and onward to immortality.

In the same tradition, enthusiastic writers of the early 20th century claimed that the magic of radium could not just cure illness but restore youthful vitality. The association of radioactivity with life-force became so widespread that hundreds of entrepreneurs raked in cash by selling patent medicines containing radium or other radioactive materials. You could eat radioactive chocolate and brush your teeth with radioactive toothpaste.

It was not until the 1920s that sensational news accounts reported problems. It turned out that an aging American playboy had swallowed so much radium tonic, in hopes of reviving his flagging sexual powers, that it killed him. Worse, women painting luminous watch dials who pointed the brushes on their lips had fallen prey to “jaw rot,” in fact cancer. Most people continued to view radioactivity positively. Like many things in medicine, it could be dangerous if misused but beneficial in the right hands. Any potent force is a double-edged blade.

If the first application most people look for when they learn of a new physics discovery is medical, the second one is weapons. Science fiction tales that featured “death rays” were already a cliché when Pierre Curie in his 1905 Nobel Prize address warned that “radium could become very dangerous in criminal hands.” Another Nobelist, Soddy, told the public that every grain of matter is a storehouse of stupendously perilous energies. Speculations about atomic explosions followed instantly, although it was not until 1913 that the specific term “atomic bomb” appeared. That was in H.G. Wells’s novel *The World Set Free*, featuring a world war that was—to use a term rapidly becoming associated with atomic energy—apocalyptic.

The Biblical apocalypse, the descent into darkness and chaos followed by a glorious Millenium, is another ancient image associated with the Great Work of the alchemists. Their mystery of transmutating corrupt inert matter into glorious eternal life could also represent a social change. Some mystics dreamed that their secret knowledge could not merely turn lead into gold but bring a Golden Age. In the same vein, Wells’s novel foresaw that after atomic warfare destroyed the world’s cities, a glorious new civilization would rise from the ashes with atomic flying cars and free love. Or could we bypass the danger? Countless newspaper articles, echoing claims by various scientists, exclaimed that science would apply atomic energy to bring global prosperity. The “new alchemists” would create miraculous new materials and supply limitless power.

On the other hand, Soddy warned of the possibility that some careless experimenter might set off an atomic chain reaction that would spread uncontrollably. The horrifying image of an exploded planet joined the associations clustering around radioactivity. There was nothing new in contemplating the end of the world, but the association with atomic energy gave the ultimate apocalypse a worrisome scientific plausibility.

Apocalyptic imagery connected with the deepest roots of the ancient transmutation enterprise. Whether they understood it or not (only some did), the alchemists’ labors in their laboratories involved hopes of personal transformation, call it psychological or spiritual as you will. The Great Work could be a plunge into your darkest places, to embrace sinful urges, overcome them, and emerge reborn. This is also the hero’s journey which, as Joseph Campbell explained, underlies many of the fairy tales and myths and religious writings of every human society. It is a perilous journey. There is a terrifying risk of getting stuck in the midst of the work, descending into chaos and never reborn.

There is much that could be said here about basic emotions and universal childhood experiences and so forth, the sort of thing Freud explored, but we can leave all that aside. However, all that was not left aside by the screenwriters of Hollywood horror movies of the 1930s, some of whom manifestly knew their Freud. A fictional scientist investigating forces could remind the audience (usually unconsciously) of a child who probed forbidden secrets, by implication partly sexual—of course the life-force in penetrating radiation always carried sexual weight. The movie scientist irradiating a pretty actress with his weird ray device could represent not only an overly inquisitive child but at the same time a dangerous and overweening father. Such a dual nature is common; as the anthropologist Claude Levi-Strauss explained, it is the very essence of myths to embrace dualities in a quest to reconcile them. Not by accident did people think “Frankenstein” was the name of the monster as well as the scientist who created it, the child and its father together. Most explicit was *The Invisible Ray* (1936), with Boris Karloff portraying a scientist whose impious experiments with uncanny rays turned the scientist himself into a murderous creature. (He glowed in the dark like a radium watch, and killed people with a touch of his hand.) This was the hero’s rebirth aborted, the seeker of transformation stuck in the darkness, his own wretched victim.

Nothing new here. Standing behind the Hollywood mad scientist was the prideful and damned Dr. Faust. And standing behind the Faust legend (which referenced an actual self-styled magician and alchemist of the 16th century) was a long line of alchemists and wizards tracing their ancestry back to the mages of antiquity. Going still farther back we meet the village witches and tribal shamans found in virtually every human culture, wielding secret spells to heal or harm. And in the shadows behind all of these moves someone very familiar: the parent who dominated our childhood, an overwhelming authority wielding mysterious powers that could be as frightening as they were attractive.

By the late 1930s all this, each and every one of the primitive and potent images described above, could easily come to mind when any member of the public thought for a minute about radioactivity and atomic energy. But the negative aspects of the imagery were peripheral, confined to the adolescent ghettos of pulp science fiction and horror movies. The sober newspapers and magazines aimed at attentive citizens (a small minority, but the makers of opinion and policy) featured mostly optimistic stories. A rising profession of science journalists echoed the scientists they interviewed, describing radioactivity and atomic energy in glowing terms. Medicine was as always the heavyweight in science writing, and new radioactive isotopes and “atom smasher” ray machines were in fact making significant advances. Atomic power plants, not to mention the flying cars Wells had predicted decades earlier, seemed a more distant promise. But when uranium fission was discovered, offering a practical way to release atomic energy, science writers redoubled their claims that a millennial age of plenty was in prospect.

This was 1939. The scientists themselves were thinking about atomic bombs.

Hiroshima and cold war

When the United States announced it had dropped an “atomic bomb” on Hiroshima, everyone knew what that meant. Even before any actual report arrived from the destroyed city, radio announcers spoke portentously of cosmic secrets, apocalyptic perils, golden future prospects—in short, all the things long associated with atomic energy, not excepting Frankenstein’s monster (for the postwar era see [Boyer, 1994](#); [Piehler, 2009](#)).

The first photographs from Hiroshima, censored by American authorities, showed a landscape of endless rubble with no living thing in sight. Atomic reality once again resonated with old imagery. Pictures of abandoned ruins, the desolate remnants of a fallen civilization (Babylon, Rome, Angkor Wat...), had been popular from the 19th century forward. This Empty City was yet another image with transmutational associations, for it could well represent the psychological state of utter loneliness and emptiness at the nadir of the attempt at rebirth.

In the next few years more complete reports and photographs from Hiroshima came to light, and these prodigiously reinforced some particular associations. Shocking stories and gruesome photographs of suffering Japanese drove straight into the regions of the brain where emotions reside. The images evoked horror—that is, a combination of the basic emotions of fear and disgust. The Hiroshima images could not be unseen, permanently associating atomic energy and radiation with pure abhorrence.

The assault on public emotions redoubled in the 1950s when movie-makers discovered a rich vein of public interest in atomic monsters. The pathbreaker was *Godzilla* (1954), a mad scientist’s monster on steroids, awakened by an atomic bomb test. The enormous lizard with radioactive death-ray breath stomping on Tokyo was immediately recognized as a symbol for the atomic bomb. Hollywood responded with *Them!* (1954) featuring ants the size of busses engendered by a bomb test. The movie was successful enough to spawn countless other gigantic monsters, mostly invertebrate. A movie-goer shocked by the sudden onset of a huge, hairy insect would experience pure horror, another blow of the hammer nailing primal emotions onto atomic energy.

Here again atomic science had a hook to attach bizarre associations. Since the 1930s biologists had recognized that radiation can cause genetic mutations. Such tampering with living creatures was immediately felt to carry the deep dangers associated with any transmutation. Mutations mean birth defects, and it is telling that the oldest meaning of “monster” was a malformed birth such as a two-headed calf. The facts of mutation associated radioactivity with yet another primitive image—the defective baby—almost too dreadful to bring to mind.

Yet nothing was as dreadful as the simple fact that on any given day your city could be blasted to smoking rubble. Reports from Hiroshima were quickly outpaced by sober magazine articles portraying a future atomic war, not science-fiction but government-endorsed statements of facts. However awful those might be, fiction went further. The public grew familiar with fantasies of a few lonely survivors roaming among the ruins, an Empty City conveniently stocked with canned food. The survivors might venture into a radioactive Forbidden Zone, the hero’s journey with mutant monsters. Some stories promised the eventual rise of a new and better civilization, the classic apocalyptic transmutation. In other stories apocalypse failed to engender rebirth. An actual end of the world, not as a miracle of divine wrath but from plain human misuse of physics, seemed dreadfully plausible. Hundreds of millions felt tears come to their eyes as they viewed the final scenes of *On the Beach* (1959), the streets of the world’s cities rendered radioactive and emptied of all life.

An actual world war would produce none of these outcomes, but the real possibilities were terrifying enough. Government officials and media editors worked to make everyone aware of the peril, aiming to promote civil defense. With Hiroshima-sized bombs it seemed sensible to adopt the measures that Europeans had recently used as their cities were pulverized. Authorities took care to present images that would prompt effective preparation, images accurate enough to “inoculate” the public so they would not panic in an attack, but not images so graphic that citizens would recoil in horror, outrage, or despair. Yet no amount of care could make palatable the dreadful prospect: hundreds of ruined cities populated by millions of corpses. (Hardly anyone dared to recognize that tens of millions more would be refugees, diseased and starving in squalid camps or wandering on their own.) The most emotionally vulnerable citizens, of course, were children. “Duck and cover” school exercises and instructions on how to escape radioactive poisons gave an entire generation nightmares—and a growing skepticism that authorities could be trusted to protect them.

Adult citizens too were beginning to feel misgivings about the authorities who controlled the fearsome new power, as represented primarily by the US Atomic Energy Commission (AEC). The first questions arose following widely publicized bomb tests in the Nevada desert. The radioactive dust from the mushroom clouds was easily detected when it fell to earth hundreds or even thousands of miles away. The obvious association was the “black rain” following the Hiroshima and Nagasaki bombs, fallout that many survivors believed irretrievably contaminated them and perhaps their progeny as well. AEC spokesmen insisted that everything was fine, and most Americans patriotically accepted that the tests were necessary to prevail in the Cold War. But doubts lingered, well-founded doubts as later investigations revealed, that the officials had been over-confident about the risks if not deliberately disingenuous.

To excuse their bland and elliptical reassurances the AEC pled the necessities of Cold War secrecy. Almost anything related to atomic energy was treated as a mysterious magic, more potent and perilous than any other secret in the public mind—and even in American law, which enacted a death penalty specifically for atomic espionage. The public of the 1950s was obsessed with atomic secrets, for here too facts aligned all too well with myths. Communist spies both real and fictional took up the mantle of the 1930s mad scientists who might blow up the world. By ancient tradition and deep in the psychology of childhood, anyone with powerful secrets seemed threatening. Critics remarked that the AEC with its secrets had adopted a “father knows best” attitude that was not always reassuring.

These feelings broke into the open in 1954 when a test of a hydrogen fusion device in mid-Pacific was three times more powerful than calculated, projecting clouds of intensely radioactive dust hundreds of miles across the ocean. Distant islanders were struck with radiation illness, and the crew of a Japanese fishing vessel, the misnamed *Lucky Dragon*, required urgent medical care. When the AEC claimed that everything was safe (details withheld for national security), nothing to see here folks, many people recognized the statements were misleading and some saw outright dishonesty.

By 1960 the number of hydrogen bombs was rapidly climbing, each three orders of magnitude more powerful than a fission bomb. Cowering under a desk would not save you if you were *inside* the fireball. If you were not directly incinerated you could face highly radioactive residues scattered across the countryside; your only recourse would be to hide underground for weeks. A rage for building fallout shelters swept the United States in the early 1960s. The media featured anguished debates about whether you should shoot neighbors who tried to get in. And if you did survive, what would you find when you crept out? The fallout-shelter movement collapsed in futility as the number of hydrogen bomb warheads climbed into the thousands, but it had powerfully reinforced associations between atomic energy and images of post-apocalyptic savagery and desolation.

To many citizens it seemed plain insanity that national authorities were building enough ballistic warheads to truly shatter civilization. The mad scientist (exemplified by the physicist “father” of the hydrogen bomb, Edward Teller) became allied in public imagery with another trope of long standing, the bloody-minded military leader (exemplified by the actual commander of the American strategic forces, General Curtis LeMay, who had incinerated a million residents of Tokyo in a single night and planned to repeat the performance on Moscow). The mad scientist/general was connected in turn with another traditional villain, the warmongering government leader (exemplified by your enemy of choice, be it a Communist tyrant or the fervently anti-Communist 1964 Republican Presidential candidate, Barry Goldwater). Here was another dense tangle of imagery, immortalized in the characters of the 1964 movie *Dr. Strangelove*. Apparently such men had to be stopped before they stumbled into World War III and destroyed everything.

A public movement against nuclear weapons had begun in Japan in the wake of the *Lucky Dragon* affair, reinforced by reports that some tuna caught in the Pacific Ocean were radioactive. The Japanese reaction was similar to what Americans would feel if they heard their hamburgers were polluted. Opposition to the arms race rapidly spread around the world, bringing millions out in mass demonstrations. Originating in a reaction to bomb test fallout, the movement gave most of its attention to this “death dust.” Fallout was potent as a “backyard” issue, for sensitive geiger counters could detect traces of radioactivity thousands of miles downwind from a test. Every mother could worry that her child was eating contaminated food.

The movement’s ultimate goal was to ban the bombs altogether, but leaders understood that taking on the entire Cold War was impracticable. They concentrated on demanding a ban on weapon tests. Halting the development of even worse devices was sensible in itself, and could be a first step toward international détente.

The history of the vast “ban-the-bomb” movement has filled entire books (Wittner, 1993-2003) and this essay will restrict itself to one point. Through every medium from songs to mass demonstrations the movement pounded into the public mind one message: radioactivity, even the minuscule amount that might taint a snowfall 10,000 miles distant from a bomb test, was horrible above all horrors. Nobel-prize scientists exclaimed that exposing billions of people to billionths of a dose would strike down millions with cancer, while millions of babies would suffer mutational defects.

These were unprovable claims, since even a million cancers or birth defects would be an undetectably small addition to these great burdens on humanity. Other scientists insisted that our bodies are well adapted to radiation at the level found in nature from cosmic rays and rocks, a level which global fallout increased by only a tiny fraction. Some went on to argue, like the believers in healthful rays who were still patronizing spas that advertised radium water, that a bit of radiation stimulates the body’s repair mechanisms. Perhaps so; the science remains uncertain and contentious to the present day. For many experts and the bulk of the public, the argument that prevailed was that a little additional radiation might possibly bring mass death, best to take no chances. Doctors began to observe that if most people were grateful for therapies with X-rays and radioactive materials (which every year preserved millions of lives), a significant minority recoiled from anything that involved the slightest trace of radiation. Nobody asserted that fallout could actually bring the end of the world... and then *On the Beach* did assert it. The arguments continued for years in every public venue until many people came to see all radioactivity—a basic feature of the physical world—as an unrelieved evil.

Underlying these ideas was a threat that even the most ancient and primitive organisms know and shun: pollution. The response to a poison in the environment is disgust, fear, and eventually horror. A tight connection between radioactivity and pollution had emerged in reports about the Hiroshima survivors who believed that simply being in the vicinity of the blast had contaminated them for life. But it was the movement against nuclear weapons that solidified the association as a basic component of modern culture.

Disgust is not only a physical but a social phenomenon. Brain scans find that regions activated by a mention of, say, excrement, are also activated by “disgusting” behavior, for example a description of someone who molested little boys. It appears that human evolution hijacked an ancient physiological defense for use in social situations. When we say, “that was a shitty thing to do” we are invoking something very deep. Thus there was a fundamental connection between the opposition to bomb test fallout and distrust of the authorities responsible for it. Such distrust was indeed central in the movement’s politics.

The pressure on governments became too strong to withstand. After the 1963 Cuban Missile Crisis brought a spike of unbearable terror, the Cold War protagonists undertook negotiations. The result was an agreement to restrict bomb tests to underground explosions. That had no practical effect on the arms race, but it meant there would be no more fallout drifting into people’s backyards. The agreement robbed the anti-bomb movement of its central argument, and also reassured the public as a small step away from obdurate hostility. The movement collapsed. Not only mass demonstrations but newspaper articles, books, movies, cartoons, every

sort of production about nuclear war, which had all proliferated after 1945 and became even more numerous in the early 1960s, abruptly disappeared from the public stage. People who were asked in the early 1960s to list their main concerns usually put nuclear war at or near the top; asked in the 1970s, few brought it up at all. Since nations were as prepared as ever to launch their missiles at a moment's notice, this looks like a case of classic denial. In the psychology of an individual, denial happens when something is so disturbing that it never rises to conscious thought. After 1964 this appears to have become a mass phenomenon.

Nuclear reactors

For all the bomb terrors, through the 1950s and 1960s the public was enthusiastic about employing "Our Friend the Atom" (the title of a widely seen Disney television episode and educational film). Scientists since the Curies had staked their hopes and careers on using the new energy for the benefit of humanity, and the onus of building weapons only redoubled their efforts. The scientists' well-publicized enthusiasm was seconded by government action. To counter the public distrust aroused by the threat of nuclear war, and to score Cold War points by promoting peace and prosperity, the US Atomic Energy Commission and its counterparts abroad launched extensive education and propaganda campaigns. "Atoms for Peace" was both a rallying cry and a justification for serious research and development work.

Experts, journalists, politicians, corporate and government officials, opinion leaders of every stripe repeated the ambitious claims made ever since the discovery of radioactivity. The magic of the atom would bring astounding new materials, incredible benefits to medicine and agriculture, and not least, a limitless source of cheap power. If personal flying atomic cars posed difficulties, engineers set to work designing atomic aircraft, and atomic ships were actually built (starting with submarines on the military budget, but peaceful ones too). Most significant of all was the development and deployment of full-scale atomic power plants, nuclear reactors generating electricity. Government support and the enthusiasm of a handful of corporate executives culminated in 1957 when, to universal acclaim, the first large and entirely "peaceful" reactor started feeding power into the electrical grid.

It might be a business project but it still seemed like magic. Physicists themselves continued to tout atomic energy as the cosmic force that powered the universe, a mystery just short of the Philosophers' Stone. Few admitted that the forces driving radioactivity are no more, and no less, wondrous than familiar electrochemical forces; operating a reactor is metaphysically equivalent to burning wood. But nothing could loosen the association of atomic energy in general and radioactivity in particular with uncanny secrets and powers. Anything so potent could be terrifying if mishandled. Were the authorities trustworthy?

The "anti-nuclear movement" of the early 1960s, which is to say the anti-nuclear-bomb movement, had not entirely dissolved. In the late 1960s many of the leaders and organizations that had campaigned against bomb tests began to incubate a new "anti-nuclear movement" that opposed building power plants. In the 1970s crowds again chanted "No nukes!" but now referring to reactors rather than bombs. Not only the language but the technical arguments had scarcely changed. It was again a backyard issue, as communities in the vicinity of a planned reactor recoiled at the thought that an accident or even routine emissions could contaminate them with radioactive materials. You didn't have to live downwind from a reactor to worry, as nuclear materials and particularly radioactive waste traveled the rails and the highways. Once again experts debated fruitlessly whether exposing a large population to minuscule amounts of radiation was virtually harmless or virtually genocidal. Of course the actual genocidal threat was the proliferation of hydrogen bombs, mounting relentlessly into the tens of thousands, but that was rarely mentioned. Denial persisted through the 1970s.

Denial is often accompanied by displacement. Displacement can be entirely conscious, as when a monk turns from lascivious thoughts to contemplating the Virgin. Often it is unconscious. The classic example is a monkey who, attacked and defeated by a monkey of higher status, turns and attacks one below him (much the same has been observed in schoolyards and offices). The deepest motivation of the anti-reactor movement may well have included displacement of the dread of bombs onto something quite different.

The movement also redeployed the growing distrust of authorities. Originally provoked by bomb test fallout and the spread of apocalyptic weapons, distrust escalated with the coming of age of the postwar "baby boom" generation. This was a generation mobilized not only by their experience of civil defense delusions and the perilous Cold War standoff but equally by fierce struggles for civil rights in the American South, the Vietnam War, and many other issues (European students were particularly outraged by the persistent residues of fascism and their own hidebound university authorities).

The scientists who devised bombs and the generals and political leaders who threatened to use them now shared the stigma of villainy with executives of the rising nuclear industry, and onward to capitalists in general. It was no big stretch, for corporate bosses had been attacked for a century in everything from muckraking books to cartoons. The movement used its tactics of mass demonstrations and civil disobedience and media messaging, developed to oppose weapons, not only to block new reactors but to assail the entire military-scientific-government-corporate "establishment."

The movement opposing reactors, like the atom-bomb movement a decade earlier, has a history spanning countless significant actions of many kinds in many countries; this essay will restrict itself to one point. Through every medium from lawsuits to songs to invasions of reactor sites, the movement pounded into the public mind the message that reactors were as horrible as bombs.

After all, couldn't an atomic power plant explode like the Hiroshima bomb? The public watched closely as the industry struggled with the threat of accidents. Reactor experts were at the forefront in demanding strict safety measures—and in dreaming up the worst possible disaster scenarios, however unlikely, in order to figure out how to prevent them. To be sure, it is physically impossible for a reactor to blow up like an atomic bomb, since a runaway reaction would simply melt the fuel. The experts calculated, however,

that during a meltdown a simple steam explosion could release a cloud of radioactive material. As usual, the facts resonated with apocalyptic fears. Opponents of reactors, drawing on the familiar scenarios of hydrogen-bomb warfare, exclaimed that fallout from a reactor accident could kill countless citizens and render entire states uninhabitable.

Some of the most effective arguments looked at radioactive waste. The very word “waste,” described as coming from the “back end” of the fuel cycle, evoked disgusting pollution, something so awful it had to be hidden away. The experts themselves took it for granted that they must take extraordinary measures to isolate radioactive materials. Most passionately reviled was plutonium, the very stuff of bombs; with its radioactive half-life of 24,000 years it seemed to imperil all posterity. Few remarked that if even the tiniest trace of a carcinogen like plutonium really was dangerous, then over future generations the waste from coal-burning plants posed a far greater danger. The carcinogen and mutagen arsenic, for example, concentrates in coal ash, and as an element its lifetime is infinite. While the thousands of tons of reactor waste were locked up in costly containers, the millions of tons of coal waste were piled up in open-air sludge dumps. (Nor are solar panels innocent, junked after a few decades with their burden of the carcinogenic element cadmium.) The public knew nothing of this. It was radioactive materials that inspired awe and horror.

The growing nuclear industry and its government supporters fought their opponents with information and public relations exercises. For example, it was unfortunate that the first thing people thought of when they heard the word “atom” was “bomb.” So in the 1960s experts deliberately changed their language to “nuclear,” e.g., replacing “atomic power” with “nuclear power.” After all, the energy arises from changes in the nucleus rather than in the atom as a whole. But it took only a few years for all the negative associations to glue themselves firmly onto anything “nuclear,” beginning with “nuclear weapons.” Doctors who prescribed Nuclear Magnetic Resonance Imaging began to omit the first word to avoid repelling patients.

The advocates for nuclear energy mostly had roots in engineering, science, and corporate or government bureaucracy; their native language was facts and rational argument. Many held what was later called the “deficit model” of public opinion: people were ignorant, just give them the facts and all would be well. The films, advertisements, museum exhibits, and so forth that industry and government deployed were reasonable and somewhat effective, but many of the efforts were factual and dry, devolving into exhaustive technical reports.

Most of the opponents of reactors came from the liberal arts. They made expert use of every medium from clever posters to full-scale Hollywood movies, rarely paying scrupulous attention to facts. In a battle of vivid imagery against statistics the advantage lies with imagery.

Social scientists were barely beginning the studies that would demonstrate that facts play a minor role in public opinion. Experiments eventually showed that once people have adopted an opinion they hold to it tightly. Confronted with contrary facts, people will invent counter-arguments, or mis-hear the uncomfortable facts, or simply forget them. To be sure, a curious and fair-minded citizen can be swayed by a sound argument. But many other things are no less potent. Your identification with a social group—your neighbors, your favorite media, the political leaders and other celebrities you admire—this tends to align your opinions. In particular an idea is most easily accepted if it fits with your political ideology. It should also fit with your gut feelings about such matters as the trustworthiness of officials, how important it is to avoid anything risky, whether natural resources ought to be used or protected.

Underlying all attitudes are features in the way evolution shaped human brains (Kahneman et al., 1982; Slovic, 2010). Mechanisms that efficiently warn us of a lion hiding in the grass are less useful in an industrial civilization. Instinctively we feel that something invisible (like nuclear radiation) is more dreadful than something visible (like sunlight, the cause of countless skin cancers). A novel risk (nuclear waste) generates more murky anxiety than a familiar risk however deadly (tobacco smoke). An unusual and spectacular risk (airplane crashes) gets far more attention from the media and the public than an everyday risk with far worse mortality statistics (car crashes). A risk that seems to be imposed by an outside authority (nuclear waste again) provokes greater dismay than a risk people imagine they undertake voluntarily (tobacco again). And of course a risk that for decades has been intimately associated with uncanny secrets, and insidious pollution, and horrifying monsters, and their overweening masters, not to mention blasted cities and the end of the world...

As usual some realities of nuclear energy resonated with the worries. The infant industry naturally made mistakes. If opponents exaggerated every problem, some officials reinforced distrust by covering up difficulties and imperiously scoffing at criticism as ignorant and regressive. Wastes in particular were a magnet for actual problems as well as visceral anxieties. To give only one example, in 1971 the AEC announced that it would solve the problem by burying wastes in a Kansas salt dome, since salt deposits are naturally self-healing. Shortly it appeared that the dome had been drilled extensively for brine. The AEC had picked a remarkable rarity, a wet and leaky salt dome. The agency retreated amid great embarrassment, setting the tone for an endless dispiriting saga of geological and political feuding over waste disposal.

The AEC officials' unbridled enthusiasm for nuclear energy had long raised doubts that they could be trusted to regulate their own creations. In 1975 Congress responded by splitting the agency. Development and promotion of reactors were entrusted to a new Energy Research and Development Administration (later the Department of Energy), while a new Nuclear Regulatory Commission would serve as a strict independent watchdog. Meanwhile the nuclear industry, learning from its mistakes, devoted resources to studying public opinion and developing sound communication practices. Executives and government officials gradually weaned themselves from the authoritative “father knows best” attitude of the early AEC. Their efforts were especially effective in the neighborhood of reactors, where citizens could have personal interactions with their engineer neighbors and could inspect the plants themselves. Eventually almost the only places where a new reactor could be planned without provoking vehement opposition were sites where reactors were already running (Bisconti, 2021). But for the public at large the damage had already been done. Opinions and associations formed over decades and sealed with powerful emotions at the height of the Cold War were not to be lightly overturned.

The public's distrust redoubled in 1979 when a partial meltdown wrecked a reactor at Three Mile Island, Pennsylvania. Weeks of sensational news reports panicked people for thousands of miles around before authorities could confirm that only inconsequential traces of radioactive material had escaped. Faith in the nuclear industry had scarcely recovered by 1986 when a reactor exploded at Chernobyl in Ukraine. That did contaminate a wide region, creating another resonance of fact with myth: an actual empty city, an actual forbidden radioactive death zone. The accident directly killed a few dozen workers and probably caused a barely visible increase in cancer rates among the nearby populace. But the main cause of morbidity and mortality lay elsewhere: in dread of nuclear radiation. A mass evacuation swept up more than a hundred thousand people. Displaced from their communities and their livelihoods, tormented by a suspicion that they were irrevocably contaminated by radioactivity, tens of thousands fell prey to poverty, chronic depression, alcoholism, and even suicide (not to mention uncounted abortions by women who thought their infants might be deformed).

The lesson was overlooked in 2011 when a great tsunami caused massive reactor failures at Fukushima, Japan. The radioactive materials that escaped caused no direct fatalities but might bring a few hundred future cancer deaths. Far more important was a mandatory evacuation that displaced more than a hundred thousand people from mildly contaminated regions, in the process killing more than 2000 of them (for example, hospital patients). The refugees experienced much the same miseries as their Ukrainian counterparts; most would suffer far more over their lifetimes than if they had remained in their homes, even if there was a slight increase in the cancer risk (Oughton, 2016). Yet the evacuation was not the largest cause of illness and death. Distrust of the nuclear industry was now so powerful that Germany and Japan quickly shut down many of their reactors, some of them never to restart. Over the following decade, air pollution from the fossil fuel plants that took the place of the reactors would inflict an estimated 30,000 premature deaths (Kharecha and Sato, 2019). Nuclear fear can be deadly.

Nuclear imagery in the 21st century

Civilizations have long memories, but images fade unless they are renewed. Hardly anyone born after the Chernobyl disaster experienced the shock, disgust, and dread raised by the events and images familiar to their elders. To be sure, all the old atomic war and monster movies were available. Many youngsters watched them on the television show *Mystery Science Theater 3000* ("MST3k," 1988–99, now available on Netflix), where wisecracks by a fictional audience subverted the black-and-white action. This was the realm of the postmodern. The postmodern attitude takes up images and ideas from the past without taking them seriously, it throws about references with a wink, it sells a world-weary irony. These lurching monsters might have terrified your parents, but their power was spent (Weart, 2012).

When the television comedy *The Simpsons* (1989–) based its opening sequence on a dangerously inept nuclear reactor operator, and played episodes featuring the greedy owner of the reactors or perhaps a three-eyed mutant fish, it took for granted that the audience would understand its references—and be amused. Referencing was equally plain in another production enormously popular among young people, the video game series *Fallout*. The player wanders in a post-apocalyptic city reduced to radioactive rubble and combats mutant monsters, but unlike the monsters of the 1950s productions, these are not imagined to represent anything more real than the dragons and witches of other video games. With postmodern detachment you might get less of a visceral shock than the 1950s productions bestowed. But there is also less actual thinking involved: the sly references do not ask to be questioned. The power of the entire tangle of transmutational imagery did not go away when the mad scientist became a cartoon figure of fun. The emotional associations linger underground in the unconscious realms where they had dwelt for generations.

That was not all that remains out of sight and out of mind. Nations still deploy more than 3000 nuclear warheads that can be launched on a few minutes' notice. The President of the United States and the President of Russia each holds the unchallenged power to launch their share and destroy civilization: nobody is in a position to stop them. These facts are hardly ever discussed, a striking persistence of denial through half a century. (We can leave aside a brief but intense revival of war fears in 1980–83 when the bellicose President Ronald Reagan confronted a hidebound and decaying Soviet regime.) In classic denial, what is repressed is never really gone. It is reasonable to suppose that anything having to do with nuclear energy has strings that tie it to deeply buried terrors, transmitting an influence all the more powerful for being unacknowledged.

In the absence of war, historical events pulled up other old associations. Terrorist attacks on November 9, 2001 froze Americans in front of their televisions for a week, obsessively watching the World Trade Center collapse again and again. The event marked a generation much as Hiroshima had marked their elders, traumatic less for what had happened than for what it threatened about the future. The cloud that rose over New York City was widely described as a mushroom cloud, a phrase that had long been a metonym for an atomic blast. Without conscious decision the site came to be called "ground zero," a phrase invented in 1945 by scientists describing the effects of an atomic bomb explosion. A thought came simultaneously to many: what if terrorists set off a nuclear device?

The idea had a history reaching back through the Communist spies and infiltrators of the 1950s to the mad atomic scientists of the 1930s and beyond. Terrorists had already set off portable atomic bombs in a 1908 novel (Anatole France's *Penguin Island*), following still earlier stories of evil chemists devising novel explosives. In a popular variation the evildoer was the ruler of a powerful corporation, well-funded conspiracy, gang, or entire nation, a trope featured in so many James Bond movies (*Dr. No*, 1962, *et seq.*) that another successful movie series was based on a parody. "Let's just do what we always do," declared Dr. Evil, "hi-jack some nuclear weapons and hold the world hostage" (*Austin Powers: International Man of Mystery*, 1997). You couldn't get more post-modern than that, an ironic reference to the James Bond villain who was himself an ironic reference to earlier villainous rulers

wielding terrible secret weapons. Returning to the real world, any dictator who might get hold of a nuclear weapon was feared not only because he might use it himself, but because he might let it fall into the hands of terrorists.

These venerable fears were prominent in the actual history of early 21st century international relations. The American invasion of Iraq in 2003, the repeated war scares involving Iran or North Korea—each of these conflicts revolved around potential possession of nuclear weapons. The public feared and despised the rulers of these states as if they were traditional mad-science villains.

As for the nuclear power industry, when experts discussed reactors their greatest worry was proliferation of weapons. Reactors cannot produce power without manufacturing plutonium, the stuff of bombs, and the isotope uranium-235 is excellent for both reactors and weapons. Was it wise to encourage almost any nation to build reactors? Below the national level, while no terrorist organization seemed capable of crafting a Hiroshima-style bomb, if they simply stole some radioactive material and spread it around a city they would cause devastating panic. To be sure, there are many more noxious materials that are far easier to obtain, but only radioactive stuff would rouse apocalyptic dread.

To the public, unlike the experts, terrorism was not the most worrisome thing about reactors. Now as for half a century the public's first concern was bomb-like explosions and fallout as at Chernobyl and Fukushima. To address reactor safety the nuclear industry in the United States had developed a new system after the shock of the Three Mile Island accident. Executives had believed themselves free from responsibility so long as they satisfied the mountains of paperwork rules set by the Nuclear Regulatory Commission, but that had proved inadequate. From the 1980s onward the industry developed an independent program of self-criticism and rigorous inspections. If the industry's own engineers determined that a company's procedures were lax, the people responsible could be fired. Comparable systems were urged on the Russians, Japanese, and others (and might have prevented the Chernobyl and Fukushima disasters), but adoption was slow.

The public's second concern was radioactive waste. By now the technical problems of sequestering materials were essentially solved; the real problems were political. The anti-nuclear movement had faded away in the 1980s as nations stopped building reactors in any location where they could meet serious public opposition (thus no new reactors in the United States, but many in China). Time and the passing of Cold War stresses had dimmed nuclear fear among the public at large; only a tenth clung to powerful anti-nuclear convictions (Bisconti, 2021). But in political systems like the United States that offered many points where opposition could be raised, a passionate minority could block a political solution. The anti-nuclear impetus was only dormant and any appropriate stimulus could revive it. Nothing roused the opponents so much as anything involving radioactive materials, such as proposals for the disposal of wastes.

The 21st century public was highly sensitive to pollution in any form. Tobacco smoke, lead in paint or gasoline, DDT and other pesticides, oil spills in the oceans, and much else had turned out to be far more harmful than corporations and their tame scientists had claimed. Purity was the new watchword. Movements even arose to oppose substances that scientists fully endorsed, from genetically modified organisms to measles vaccine. Mr. Burns, the evil proprietor of the nuclear reactors in *The Simpsons*, could have been a representative of any of a dozen distrusted industries—although the creators of the series had chosen the nuclear industry as iconic.

The future: Debating climate and nuclear power

In the 21st century the public became increasingly concerned about another type of pollution: greenhouse gases. Global warming, like nuclear war, posed a novel and existential threat to world civilization.

Global warming had no resonances with ancient psychological and social themes. The mascot for the issue was the endangered polar bear, and not everyone cared much about this ferocious man-eater. Imagery of collapsing glaciers and ruined coral reefs were likewise remote from the back yard. Climate change looked far away not only in space but in time. Economic and political plans that could barely see 5 years ahead refused to worry about the next generation (Weart, 2008, 2019).

Public concern awakened in the 2010s in reaction to a manifest increase in the worst weather disasters—heat waves, droughts, wildfires, floods, hurricanes—and unequivocal statements by all reputable experts that our emissions were to blame. As early as the 1970s some advocates of nuclear power had mentioned the prospect of global warming as one among many arguments for promoting reactors. Now some citizens, including respected environmentalists who had once opposed nuclear power, recognized it as a lesser danger. Evidence mounted that climate change is a threat so dire that taking every conceivable measure to replace fossil fuels might barely be enough.

When climate scientists had first addressed the public they followed the deficit model of public opinion, like nuclear engineers and their allies during the first decades of reactor development. The scientists devoted countless hours to crafting reports by national academies and scientific societies, followed by entire years compiling evidence for the Intergovernmental Panel on Climate Change. Their success was modest. Explanations and facts are insufficient to sell an idea that conflicts with cherished ideologies, to say nothing of the colossal fossil fuel industries. Coal and oil interests had lobbied quietly against nuclear power ever since the 1950s, and starting in the late 1980s they spent billions of dollars on climate lobbying, misinformation, and obfuscation.

Advocates of action against climate change gradually discovered the immense existing literature on communication, the studies of everything from commercial advertising to public health campaigns. Scholars pressed on to develop a focused academic “climate communication science” that went well beyond the sort of opinion and attitude studies that the nuclear industry had conducted over the years. The advocates even wrote handbooks and maintained websites for climate debaters and schoolteachers (Boykoff, 2019; Cook, 2019; Cook and Lewandowsky, 2011; Moser, 2009, 2016; Shepardson et al., 2017). Although the studies were subject to the uncertainties of all social science and psychology, researchers offered some reasonably reliable guidelines.

Nothing is more important for an advocate, they explained, than to come to the audience without preconceptions, as eager to listen as to talk. “Believe me because I am an expert” had already in the 1960s turned many against the nuclear establishment. The messenger needs to seek out common ground with the audience’s goals, meeting them where they can feel comfortable. There are indications, for example, that liberals can be swayed by appeals to their empathy with the global poor, whereas conservatives may respond better to arguments involving jobs and the national interest. An issue should thus be “framed” in terms appropriate to the audience. Framing the next generation of reactors as marvels of engineering prowess could work well for some audiences, but boasts of mastering the forces of nature could also call up masculine associations reaching back to the mad scientist. That could alienate, for example, women who would respond better to reactors framed as a technology valuable for preserving the global environment.

Framing begins with the choice of the messenger, preferably someone clearly sympathetic to the listeners’ ideological orientation. A Republican member of an evangelical church will get much farther with that particular audience than an agnostic Democrat, regardless of what arguments they deploy. As the nuclear industry learned long ago, nothing is so effective as neighbor talking to neighbor. Also useful, when applicable, is explaining that a belief is more popular than the audience imagines, for humans want to belong. Most Americans, for example, are surprised to learn that nearly all climate scientists agree that emissions are dangerously warming the planet (by the 2010s the few dissenters were no longer publishing in peer-reviewed journals, but they continued to attract attention thanks to ample funding and political connections). Similarly, barely half of Americans know that a solid majority of their fellow citizens accept nuclear reactors as part of the electrical power system. Experiments have found that giving people such information about their peers can change their attitude.

As noted earlier, factual information about the issue itself gives surprisingly little help in winning people to a viewpoint (and technical or scientific information least of all). Facts and logic are certainly important and it is essential to lay them out. But they need to be embedded in a humanly appealing story.

A messenger may need to confront specific objections, based as often on misinformation as on facts. Arguing is risky: when people hear, “vaccination does not cause autism,” a week later what might come to their mind is, “there is some kind of connection between vaccination and autism.” And as anyone who has argued politics with a relative knows, confrontation often just makes your interlocutor dig in with redoubled conviction. A wise messenger will begin with a positive statement that addresses concerns you share with your audience (“vaccination can save your child’s life”) to frame the issue before proceeding to attack opposing arguments.

Framing should include a choice of rhetorical style appropriate to the occasion (dramatic? informational? inspirational?). Every technical field has its own specialized language which can mean something quite different to the lay public. Climate scientists, for example, must beware that when they speak of positive feedback (such as a warmer tundra emitting more greenhouse gases that cause more warming), a listener may think “positive” means “good.” Radiation “exposure” even at the most innocuous level sounds frightening; “contamination” of water by a single atom may be perceived as disgusting. It is even more important to think about what a pictorial image may suggest. The countless published photographs of a cavernous space with a vast pool of water full of spent reactor fuel, glowing with an eerie blue light, are not necessarily reassuring. Logic itself can have an unexpected meaning. Nuclear experts often contrast the tiny amount of radiation that might leak from a reactor to the much larger amounts of natural radiation from cosmic rays or even from bananas. But for some people anything “natural” is acceptable while anything artificially imposed is suspect (Bisconti Research Inc., 2019; Bisconti, 2021).

With distrust of authorities central to the psychology of opposition, it helps to empower the public as far as possible. Some of this can be channeled through standard democratic government mechanisms; the law courts and the Nuclear Regulatory Commission have largely retained the public’s trust to represent their concerns. More directly empowering are offering frank dialogs between industrial and public representatives, and giving the local community a meaningful role in radiation monitoring, evacuation planning, and the like.

None of these methods can prevail in the long run unless your argument is factually valid. The nuclear industry will not be fully accepted until it can provide convincing evidence that it will handle reactor safety and wastes more reliably than in the past. And of course the industry must show that it can compete economically with power provided by renewable sources plus energy storage. People can be remarkably flexible when they expect to save money. Nothing could improve public acceptance so much as the advances in reactor design described elsewhere in this volume, which demonstrably make disasters all but impossible (Rothwell, 2021a; Ingersoll, 2021; Greneche, 2021), and the economic efficiencies of scale resulting from mass deployment such as some countries are undertaking today (Rothwell, 2021b).

One thing is certain: the facts about climate change will become increasingly obvious and increasingly grim. Year by year disasters are happening more often and with greater damage, just as scientists predicted for decades. The pain is only beginning with much worse to come. It becomes obvious that every available technology must be pressed into service. However great the public’s fears of nuclear energy and radiation, the catastrophic costs and personal distress from climate disruption must eventually push them aside.

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Factors Affecting Public Opinion of Nuclear Energy in the United States

Ann Stouffer Bisconti, Bisconti Research, Inc., Chevy Chase, MD, United States

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Public opinion in action

Between 1976 and 1992, antinuclear activists sponsored 21 ballot initiatives in 11 US states aimed at banning the construction of nuclear power plants or shutting down existing nuclear power plants. The votes on these initiatives provided an unequivocal test of public opinion. In all but one case, the antinuclear effort failed and the public voted in favor of the plants. Why? With all the dramatic imagery linked to nuclear power and the fear this imagery can evoke (Weart 2021), the results of these ballot initiatives must have come as a surprise.

Also surprising must have been the first time a nuclear power plant received regulatory approval to extend its operating license by 20 years (subject to continuing regulatory review)—with little, if any, opposition. License renewals for another 90 nuclear power plants followed smoothly.

In fact, a 36-year program of research shows that the majority of the US public favors nuclear energy, and this support is even stronger among nuclear power plant neighbors. This article draws primarily from that extensive and in-depth program of research on public attitudes toward nuclear energy by Bisconti Research.¹

Favorability trends

Bisconti Research has tracked public opinion on some questions at least annually since 1983. The surveys used nationally representative samples of at least 1,000, with a margin of error of plus or minus three percentage points.

In 2019, 61% of the US public said they favored “the use of nuclear energy as one of the ways to provide electricity in the United States.” The national survey findings for 2018 were practically identical (see Findings 1).

The long-term trends show that support for nuclear energy is much greater today than it was when the survey program began in 1983 (Findings 2).

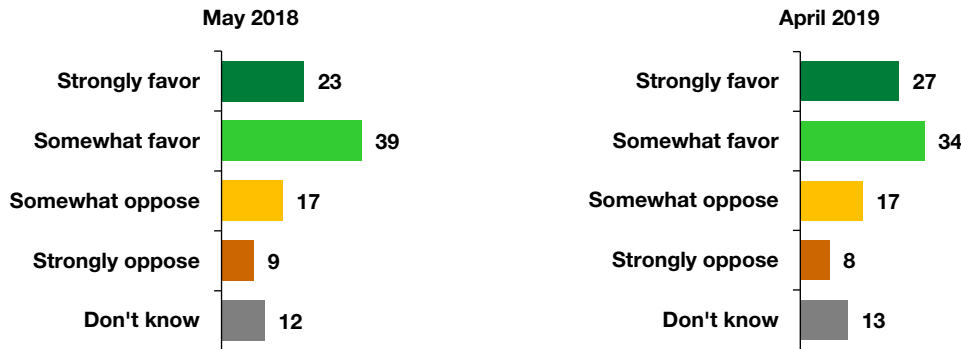
In 1983, the public was equally divided between those in favor of nuclear energy and those opposed. In 2019, those who favored nuclear energy outnumbered those who opposed by 61% to 26%; 13% had no opinion. There have been some downturns—most notably after the accident at Chernobyl in 1986 and the tsunami at Fukushima in 2011. Upturns coincided with periods of perceived need for energy. Those in favor peaked in 2010, as the industry geared up for a nuclear renaissance and the Democrats in power (historically unlikely allies) began to see the environmental benefits of this clean-air, carbon-free energy. The Fukushima accident the next year put a damper on this new enthusiasm.

Over the years, open-ended questions asking reasons for respondents’ opinions of nuclear energy have found that accidents are the main concern people have, and, therefore, real accidents like Chernobyl and Fukushima reinforce this concern. That is especially true when the media coverage shows images of devastation or human tragedy. In the case of Fukushima, images of devastation from the earthquake and tsunami could easily have been associated with the accident at the nuclear power plant that was caused by the tsunami. As of 2015, top reasons mentioned by those who *favored* nuclear energy were:

- Need the energy (13%)
- Environment, clean, less air pollution (10%)

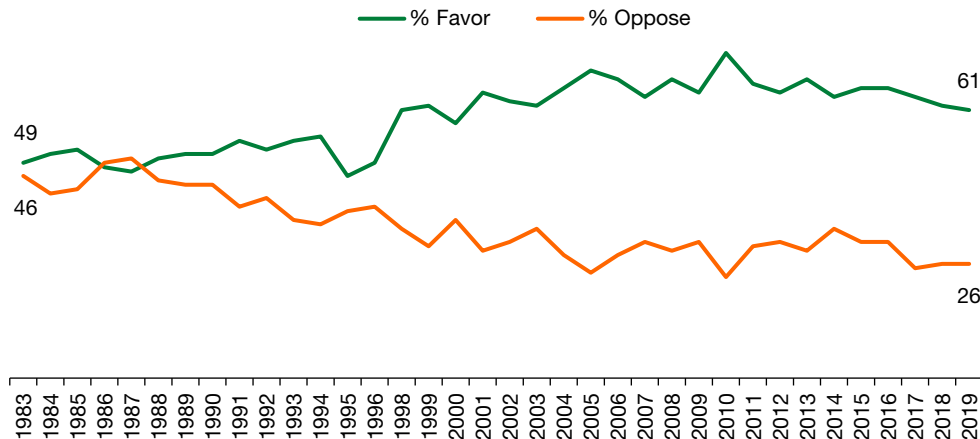
¹Dr. Ann Stouffer Bisconti initiated the research program in 1983 at the Nuclear Energy Institute (NEI) and its predecessor, the US Council for Energy Awareness (USCEA). Bisconti Research, Inc. continues the program with various sponsors, through the present day. For the past 15 years, the fieldwork of all surveys has been by Quest Global Research.

Overall, do you *strongly favor*, *somewhat favor*, *somewhat oppose*, or *strongly oppose* the use of nuclear energy as one of the ways to provide electricity in the United States? (%)



Findings 1 Favorability to nuclear energy steady (General US Public, 2018 and 2019).

Overall, do you *strongly favor*, *somewhat favor*, *somewhat oppose*, or *strongly oppose* the use of nuclear energy as one of the ways to provide electricity in the United States? (%)



Findings 2 Favorability to nuclear energy increased over long term (General US Public, Annual Averages).

- Low cost (8%)
- Others: safe, reliable, efficient, good alternative

The top reasons that came to mind by those *opposed* to nuclear energy were:

- Danger, accidents (49%)
- Waste (10%)
- Others: prefer other sources, not needed, scary

Two points jump out from these numbers:

- First, favorable attitudes are less focused than unfavorable attitudes. There is no single favorable attribute such as clean air energy that stands out. The attributes are diffuse and fuzzy. That indicates a need to focus more clearly on how nuclear energy benefits humanity.
- Second, among reasons to oppose nuclear energy, accidents are the main concern, far more than waste. However, waste does become a concern once it is raised, and some policymakers view waste as an obstacle.

One of the difficulties in communicating about nuclear waste is that people hold an image of spent fuel as a liquid that is uncontrolled. Words alone are not sufficient to convey that spent nuclear fuel is solid, compact, and contained. For this reason, advocates find that actual models, such as a simulated fuel pellet accompanied by photos of the many layers of containment are helpful. Some advocates use props such as a Coke can to convey the amount of spent fuel that would be created by a family of four over a lifetime of using only nuclear energy for electricity.

Gender continues to be associated with attitudes toward nuclear energy (**Findings 3**). Among the general public surveyed in 2019, more men (69%) than women (53%) favored nuclear. But women were not more opposed; more women (20%) than men (7%) had no opinion. Education also continues to be a factor, as 68% of college graduates, compared with 56% of those

Findings 3 Favorability to nuclear energy, demographic groups (General US Public, April 2019).

Overall, do you strongly favor, somewhat favor, somewhat oppose, or strongly oppose the use of nuclear energy as one of the ways to provide electricity in the United States? (%)

	<i>Favor</i>	<i>Oppose</i>	<i>No opinion</i>
Men	69	24	7
Women	53	27	20
Age: 18–34	57	26	17
Age: 35–49	61	22	16
Age: 50+	64	27	8
College graduates	68	25	7
Not college graduates	56	27	17
Democrat	57	33	11
Republican	73	18	9
Independent	59	26	15
Conservatives	79	15	6
Moderates	58	29	14
Liberals	49	42	9

with less than a bachelor's degree, were in favor in 2019, again due to more college graduates having an opinion. Age differences were minimal, except that more of the public over age 50 had an opinion. Majorities of all political parties favored nuclear energy: 73% of Republicans, 57% of Democrats, and 59% of Independents. Political philosophy shows a wider divide; those favoring nuclear energy included 79% of conservatives, 58% of moderates, and 49% of liberals.

Public majorities have supported specific uses of nuclear energy, including the following propositions, for decades. In 2019, majorities agreed we should:

- Renew the license of nuclear power plants that continue to meet federal safety standards—75% agreed; 16% disagreed.
- Prepare now so that new nuclear power plants can be built if needed in the next decade—67% agreed; 21% disagreed.
- Definitely build more nuclear power plants in the future—55% agreed; 31% disagreed.

Perception gap

The origins of nuclear fear are well described in Spencer Weart's chapter in this encyclopedia, "Public Images of Nuclear Energy." The negative imagery he describes persists today. Because of the negative imagery, it is often assumed that the public opposes nuclear energy, when this is clearly not the case. Over many years, the Bisconti Research survey program asked:

(1) respondents' own opinion of nuclear energy and (2) their perceptions of the opinions of others. Over the years, the data have shown a very large perception gap—that is, the US public perceives public opinion toward nuclear energy as less favorable than their own opinion.

This perception gap can lead persons who favor nuclear energy to fear speaking out in support of nuclear energy. That silence, in turn, could reinforce erroneous perceptions of public opinion. In her book, *The Spiral of Silence*, political scientist [Noelle-Neumann \(1984\)](#) proposed the theory that people fall silent when they perceive that their position on a topic is in the minority, and this silence adds to the perception that the position is unpopular. More silence accentuates the perception and further reduces vocal support in a spiral until the silent side loses.

The 2019 survey found a remarkable change in the perception gap. In 2019, 61% personally favored nuclear energy, and 53% thought that others shared this view. The 2019 finding is especially remarkable because, in three decades of asking the question in slightly different ways, this was the first survey in which a majority of the public (53%) described public opinion as favorable.

This change indicates the possibility of greater attention in the public discourse to nuclear energy as a solution to future energy problems. Are more supporters speaking out? Observation indicates they are doing so ([Findings 4](#)).

Another remarkable finding is that slim majorities across demographic groups described public opinion as favorable. Notably, 58% of the young generation of adults (Millennials and Gen Xers) perceived public opinion toward nuclear energy to be favorable ([Findings 5](#)).

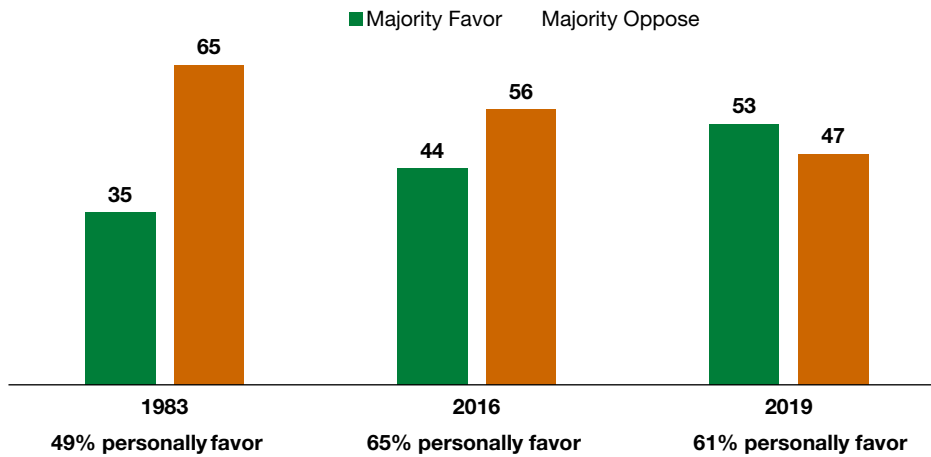
Mushiness

In the 1980s, [Yankelovich \(1991\)](#), in conjunction with *Time Magazine*, developed an index to distinguish issues about which public views were "firm and stable" from those that were "unstable and mushy." Factors associated with greater mushiness include low congruence between one's own perceptions and those of others, low personal engagement, and a low sense of feeling knowledgeable about the topic.

(Includes only respondents who said others favor or oppose)

1983 Question: To the best of your knowledge, are most people in your community in favor of or opposed to nuclear energy? (%)

2016/2019 Question: Do you think that the majority of people in your community favor or oppose the use of nuclear energy? (%)



Findings 4 For the first time in 36 years, more say the majority favors nuclear energy (General US Public). Bisconti Research, Inc. (2019) *Public Opinion on Nuclear Energy: Turning a Corner?* American Nuclear Society.

Findings 5 Perceived favorability to nuclear energy, demographic groups (General US Public, April 2019).

	Do you think that the majority of people in your community favor or oppose the use of nuclear energy? (%)	
	Favor	Oppose
Men	54	46
Women	53	47
Age: 18–34	58	42
Age: 35–49	50	50
Age: 50+	52	48
College graduates	55	46
Not college graduates	52	48
Democrat	50	50
Republican	60	40
Independent	53	47
Conservatives	65	35
Moderates	52	48
Liberals	40	60

Public opinion toward nuclear energy tends to be mushy, Bisconti Research surveys find. It is easily influenced by new information and even by slight changes in the context in which public opinion questions are asked. For this reason, the ability to track the same identical question in the same context is valuable (Bisconti, 2016). A context of discussing environmental hazards and alternative solutions can trigger negative thoughts and images about nuclear energy such as Weart (2021) described in his chapter on “Public Images of Nuclear Energy.” Recently, Gallup has asked the exact same question that the Bisconti Research polls have tracked for 36 years but in the context of questions about environmental hazards. The answers are always less favorable to nuclear energy in that context; in 2019, Gallup found the public divided 49% in favor to 49% opposed; Bisconti Research found 61% in favor to 26% opposed; 13% had no opinion.

What remains in common, even with the influence of a negative context, is that the majority of the public in both surveys sits in the middle—*somewhat* favor or *somewhat* oppose or don’t know. That is another sign of mushiness.

There is a common misperception that a large portion of the public is firmly opposed to nuclear energy because opponents have been outspoken. Those saying they were strongly opposed in 2019 included 21% of those surveyed in the Gallup question context and 8% in the Bisconti question context. Obviously, being strongly opposed does not equal being firmly opposed, as opinions of many potentially strongly opposed respondents change depending on the question context.

Familiarity builds favorability

Surveys consistently have found that few members of the US public feel very well informed about nuclear energy. In 2019, only 19% said they felt very well informed about the topic. These include more men, college graduates, and Millennials and Gen Xers (**Findings 6**).

There is a close correlation between the level of feeling informed on nuclear energy and attitudes toward this energy source. Among those who felt very well informed about nuclear energy, 68% *strongly* favored nuclear, and 7% were *strongly* opposed. Those who felt less well informed tended to express opinions in the middle or no opinion at all (**Findings 7**).

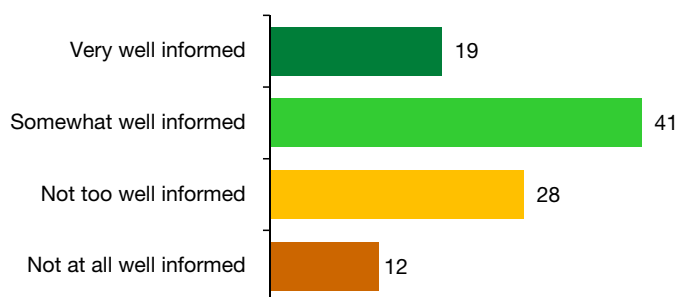
Plant neighbors more favorable

Familiarity is shown to make a difference in surveys of nuclear power plant neighbors. Persons who live near nuclear power plants are overwhelmingly favorable to nuclear energy. Their opinions are the opposite of NIMBY (Not In My Backyard).

Bisconti Research conducted seven biennial surveys of nuclear power plant neighbors from 2005 through 2017. Each survey included at least 1,000 persons selected within the 10-mile radius of the US sites with operating nuclear power plants—59 sites in 2017. The surveys excluded households with people who work at a nuclear power plant and might have a vested interest.

Over the seven surveys, public support has continued to be broad and deep. In 2017, 81% favored the use of nuclear energy, and half were strongly favorable. Those opinions were more favorable than those of the general public who were surveyed at the same time (**Findings 8**).

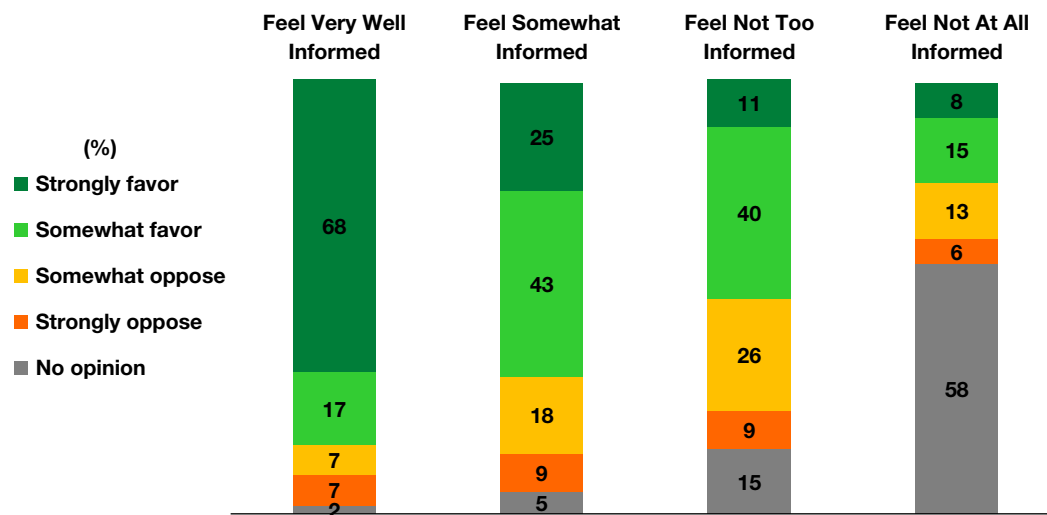
How well informed do you feel about nuclear energy used to produce electricity? (%)



Findings 6 Level of feeling informed about nuclear energy (General US Public, April 2019).

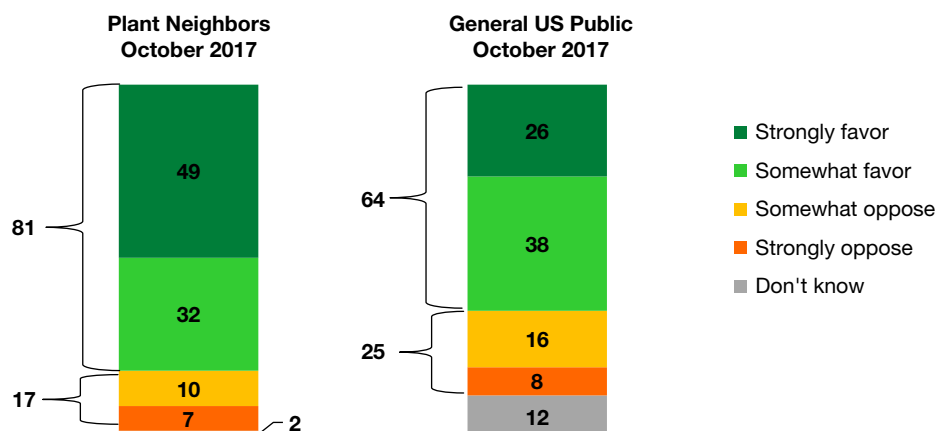
Overall, do you strongly favor, somewhat favor, somewhat oppose, or strongly oppose the use of nuclear energy as one of the ways to provide electricity in the United States? (%)

How well informed do you feel about nuclear energy used to produce electricity?



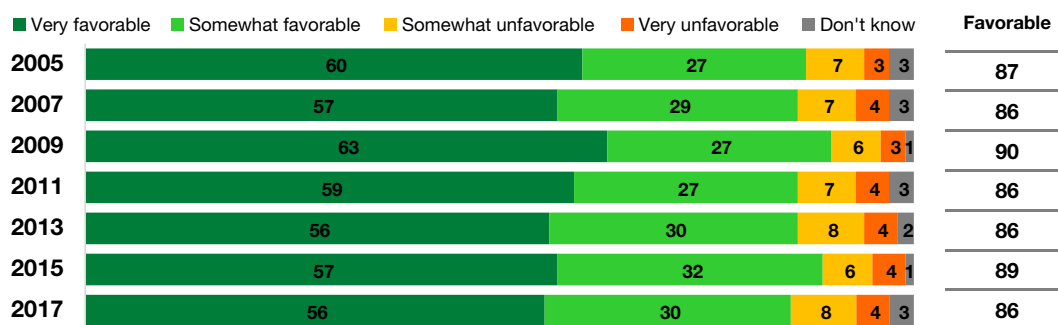
Findings 7 Informed about nuclear energy, by favorability to nuclear energy (General US Public, April 2019). Bisconti Research, Inc. (2019) *Public Opinion on Nuclear Energy: Turning a Corner?* American Nuclear Society.

Overall, do you *strongly favor*, *somewhat favor*, *somewhat oppose*, or *strongly oppose* the use of nuclear energy as one of the ways to provide electricity in the United States? (%)



Findings 8 Favorability to nuclear energy (Plant Neighbors and the General US Public, October 2017).

Thinking of the nuclear power plant closest to where you live, would you describe your general impression of this plant and the way it has operated recently as very favorable, somewhat favorable, somewhat unfavorable, or very unfavorable? (%)



Findings 9 Impression of nearby nuclear power plant (Plant Neighbors Surveys).

Those favorable attitudes are grounded in excellent performance, according to plant neighbors. Almost everyone (86% in 2017) reported a favorable impression of the nearby plant and how it had operated recently; 56% were very favorable (**Findings 9**).

When asked about the community's impression of the nearby plant, there is virtually no perception gap appearing in any of the seven surveys. Residents continued to be aware that most people in the community have a favorable impression of the plant: most recently, in 2017, 75% thought most people in their community have a favorable impression, 12% thought they were unfavorable, and 13% were unsure.

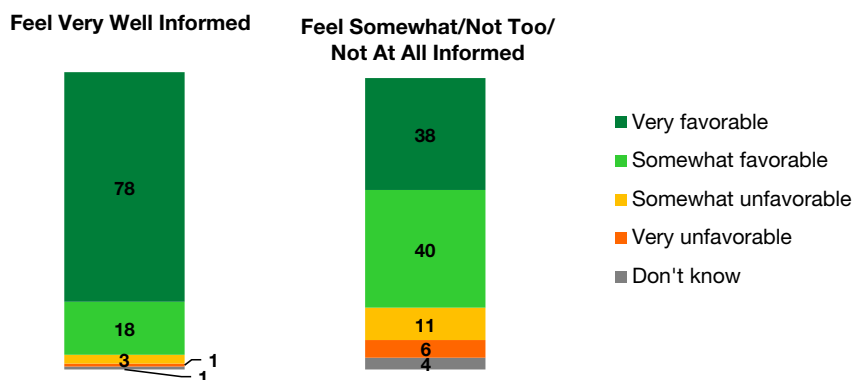
Plant neighbors feel better informed about nuclear energy than does the general public; in 2017, 45% said they felt very informed. Of those who said they felt very well informed, 78% were very favorable to the nearby plant, and 1% were very unfavorable (**Findings 10**). Because activism is usually dependent on feeling very well informed without a topic, the chances of nuclear plant neighbors becoming activists in favor of a nearby plant versus opposed to a plant are on average 78 to 1.

Local support is so strong that, in 2017, 68% said they would find it acceptable to add a new reactor at the nearby plant. Just 27% would find it not acceptable, and 5% were unsure. Acceptability slightly declined after Fukushima—from 76% in 2009 to 67% in 2011 and then plateaued at 68% in 2013, 2015, and 2017. This figure is for all US nuclear power plants and is lower in some areas and higher in others. It is clear, however, that there are numerous sites where a new reactor would be readily acceptable. In a time of BANANA (Build Absolutely Nothing Anywhere Near Anyone), nuclear energy has the advantage of extraordinary local support (**Findings 11**).

A 2019 survey of Canadians by Abacus Data also found a similar pattern of greater support for nuclear energy in provinces that are most familiar with the technology. After being told that "nuclear energy produces very little carbon emissions, similar to solar, wind, and hydroelectricity," respondents were asked their opinion on "using nuclear energy technologies in situations where they could replace the use of higher emitting fuels." Nationally, 49% said they would support using nuclear energy in those situations, 35% would be open to it, and 17% would oppose. In Ontario, where much of Canada's nuclear energy capacity is located, most would support (56%) or be open to (34%) substituting nuclear for higher emitting sources; only 10% would oppose. In Quebec province, which does not produce any nuclear energy, 30% would oppose.

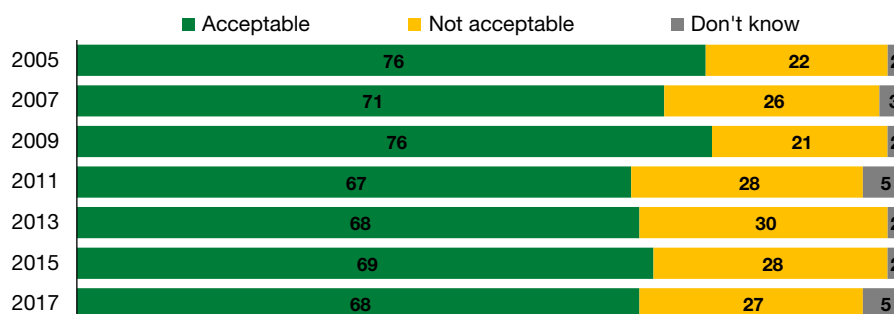
Thinking of the nuclear power plant closest to where you live, would you describe your general impression of this plant and the way it has operated recently as very favorable, somewhat favorable, somewhat unfavorable, or very unfavorable? (%)

Do you feel very informed, somewhat well informed, not too well informed, or not well informed at all about nuclear energy used to produce electricity? (%)



Findings 10 Impression of nearby plant, by how informed residents feel about nuclear energy (Plant Neighbors Survey, October 2017).

If a new power plant were needed to supply electricity, would it be acceptable to you or not acceptable to you to add a new nuclear reactor at the site of the nearest nuclear power plant? (%)



Findings 11 New reactor acceptability (Plant Neighbors Surveys).

Foundations of local public support for nuclear power plants

Local support for the plants is grounded in familiarity, perceptions of safe plant operations, and favorable views of the company regarding safety, the economy, jobs, the environment, and community outreach. Nuclear power plant personnel are actively engaged in community outreach and projects such as support for schools and the local environment (**Findings 12**).

Local voluntary environmental programs around US nuclear power plants are especially extensive and were initiated long before such programs became fashionable among corporations. Because nuclear power plants are often located next to lakes, rivers, or ocean water, they are surrounded by diverse species of wildlife and plants. Several nuclear utilities sponsor parkland near their plants for use by the local community. These programs are on top of clean and safe performance and environmentally safe waste management, both of which are regulated.

A 1995 survey by [Landsberger and Creatchman \(1995\)](#) produced an inventory of US nuclear utility voluntary environmental programs to protect the land, water, and wildlife around their plants. The programs, published in the report, *Nuclear Power Plants and the Environment: 254 Things you Never Knew*, include many activities co-sponsored by environmental organizations such as the Nature Conservancy, Audubon Society, and Boy and Girl Scouts—as well as local government agencies, universities, and community groups. The report cites a manager at an electric utility that owns a nuclear power plant, as follows: “We’re not a manufacturing plant that can just up and move. This is our service territory. We’ve been here, serving this region, for nearly 100 years—and we’re not going anywhere. That gives us a major stake in preserving the beauty of the area.”

In addition to programs such as those to protect the environment, the nuclear utilities put considerable effort into energy education and outreach. Energy education centers were once the focal point for their energy education. These centers, like mini-museums, included exhibits and personnel to explain energy, nuclear energy, radiation, waste management, and other topics of interest. These centers were open to passersby from anywhere, as well as to the local community. They also served as places for community groups to hold meetings. After the 9/11 attacks and the increased imposition of security everywhere,

Findings 12 Percent agree with statements about company that operates the nearby nuclear plant (Plant Neighbors Survey).

Now, I'd like to ask you about the company that operates the nuclear power plant nearest to you. Please tell me if you strongly agree, somewhat agree, somewhat disagree, or strongly disagree with the following statements about this company. (%)

	2005	2007	2009	2011	2013	2015	2017
I am confident in this company's ability to operate a nuclear power plant safely.	88	87	91	87	89	90	88
I am confident that the company has prepared the plant to withstand the most severe natural events that may occur in this region.	n/a	n/a	n/a	79	83	82	81
This company is doing a good job of protecting the environment.	84	81	86	84	82	83	79
The plant helps the local economy.	n/a	n/a	90	87	91	89	90
There are good jobs for local people at the plant and in local businesses that provide services to the plant.	n/a	n/a	89	87	87	89	87
This company is involved in the community. (<i>"...as a good citizen."</i> 2005)	83	77	83	76	80	81	78

centers located within the boundaries of the plants had to be closed. Plant tours, another important education vehicle, also were largely curtailed.

Personnel from nuclear power plants still actively engage with local community groups and schools. In some cases, the companies organize outreach formally. A model is the Fan Team approach used by the nuclear utility Baltimore Gas & Electric for two-way communication with the community before seeking license renewal of the Calvert Cliffs plant—the first plant to seek license renewal in the United States. Designated personnel, like spokes on a fan, took information to their peer groups and brought back questions and opinions from these groups to a central point. The result was that the community gained a good understanding of license renewal, and the company had a good understanding of public opinion.

The local newspaper reported the results:

We are sure some of the opponents to relicensing Calvert Cliffs Nuclear Power Plant were left confused following a hearing last week. Critics of the utility and the Nuclear Regulatory Commission denounced the relicensing process as well as nuclear energy. Then what happened next was a little startling. Calvert Cliffs residents stood up and told them to leave their plant alone. That Baltimore Gas & Electric had been a good neighbor. (*The Calvert Recorder*, 1999)

Some nuclear power plants report that establishing Community Advisory Groups has been helpful in engaging the public and creating more ways for two-way communication. They have been used, for instance, for advising the utility on community interests when storage facilities for spent nuclear fuel are built on the plant site or when a plant is being decommissioned.

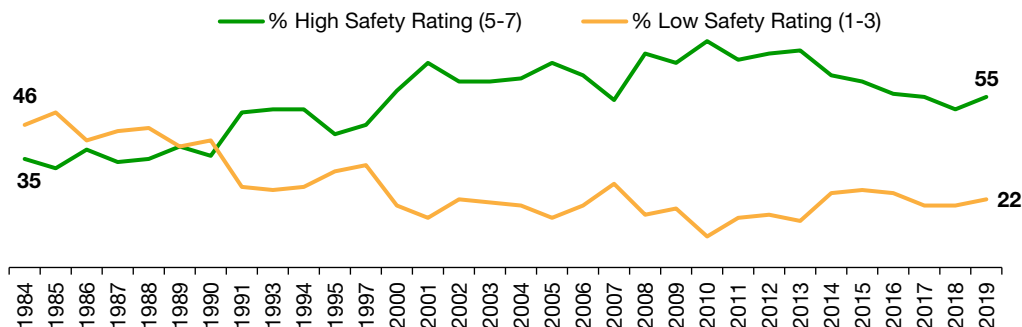
Building support beyond local communities

Expanding familiarity to a population of millions is obviously challenging. Beyond local communities, the general public can best be reached when there is something related to nuclear energy that is on their minds. For example, when the Chernobyl accident occurred in 1986, there was vast interest in the safety of nuclear power plants. Quick research showed that the public was greatly reassured to learn about the main differences between the Chernobyl plant and those in the United States. The industry launched an advertising campaign that explained, "Why What Happened at Chernobyl Did Not Happen at Three Mile Island." The advertising showed the containment structure that surrounds US reactors and did not surround the Chernobyl plant. It also, for the first time, communicated the reality of the Three Mile Island accident—that no one was harmed. It helped that the news media, politicians, and even some nuclear critics reiterated these same themes.

The result was remarkable. Although Chernobyl caused a slight drop in favorability to nuclear energy, the ratings of the safety of nuclear power plants became slightly *more* favorable than before. The ratings of nuclear power plant safety then began an upward trend that continued with a couple of aberrations until a descent following the Fukushima accident in 2013 (**Findings 13**).

Today, nuclear power plant safety is not part of the public discourse. The tremendous influence of the Institute of Nuclear Power Operations on safety, including changes to all US plant operations based on lessons learned from Three Mile Island and Fukushima, is reassuring but is not a current topic for the news media. The ongoing daily oversight of all US plants by the Nuclear Regulatory Commission (NRC) and the fact that, if a nuclear power plant were not operating safely, the regulator would shut it down, is also very reassuring but not a current news topic.

Thinking about the nuclear power plants that are operating now, how safe do you regard these plants? Please select one number on a scale of “1” to “7” where “1” means very unsafe and “7” means very safe. The safer you think they are, the higher the number you would give.



Findings 13 Perceptions of nuclear plant safety more favorable over long term (General US Public).

Today, the energy related topic that is on the public mind and of interest to the news media is environmental, including clean air, carbon-free energy solutions, and transitioning away from fossil fuels. Within that context, there is a great deal of innovation taking place at the national laboratories and in private industries that can convey to the public that the nuclear energy enterprise is thriving. A thriving industry seems safer than one that is on its last legs. Some of that innovation is exciting and includes different sizes of reactors to meet different location needs. Innovative nuclear reactor designs can make real the promise that nuclear waste does not have to be wasted: some of them use the waste to create electricity and others can minimize waste.

These innovations are consistent with the international Clean Energy Ministerial that is taking a fresh visionary look at energy systems of the future in which nuclear energy can supply electricity, as well as clean drinking water, heat, hydrogen, and support for variable renewables.

The industry should not fear that talking about the technological advancements might shed a bad light on the current fleet, as the public expects a thriving industry to evolve and improve its products. Negative impact will happen only if those promoting new reactors disparage the current fleet by calling existing reactors unsafe, outdated, or old. None of those claims is true, as is evidenced by the fact that if a plant were unsafe the Nuclear Regulatory Commission would shut it down. All the existing plants are continually updated with new technology.

The interest in technological discussions extends to preferred spokespersons. Americans would prefer, by a wide margin, information from nuclear scientists and engineers, as well as from a safety engineer at a nuclear power plant in their area. Research finds consistently that plant neighbors and the general public both give greatest credibility to persons with hands-on experience with the technology (**Findings 14**).

Given the option of using solar or wind, there needs to be a compelling reason to use nuclear energy. Public discussions about climate change solutions often fail to mention nuclear energy even though nuclear energy produces more than half the US carbon-free energy. Even more important, nuclear energy offers *both* clean air and reliable electricity. It is the only carbon-free source of electricity that offers both in large quantities day in and day out. It keeps on generating electricity when the wind doesn't blow and the sun doesn't shine. Those are compelling reasons.

In fact, the public agrees, as seen in answers to questions designed to test messaging *after* completing the trend questions about public opinion. Most recently, in the 2019 survey, 75% agreed with the following statement: "We should take advantage of all carbon-free energy sources including nuclear, hydro, and renewable energy to produce the electricity we need while limiting greenhouse gas emissions" (**Findings 15**).

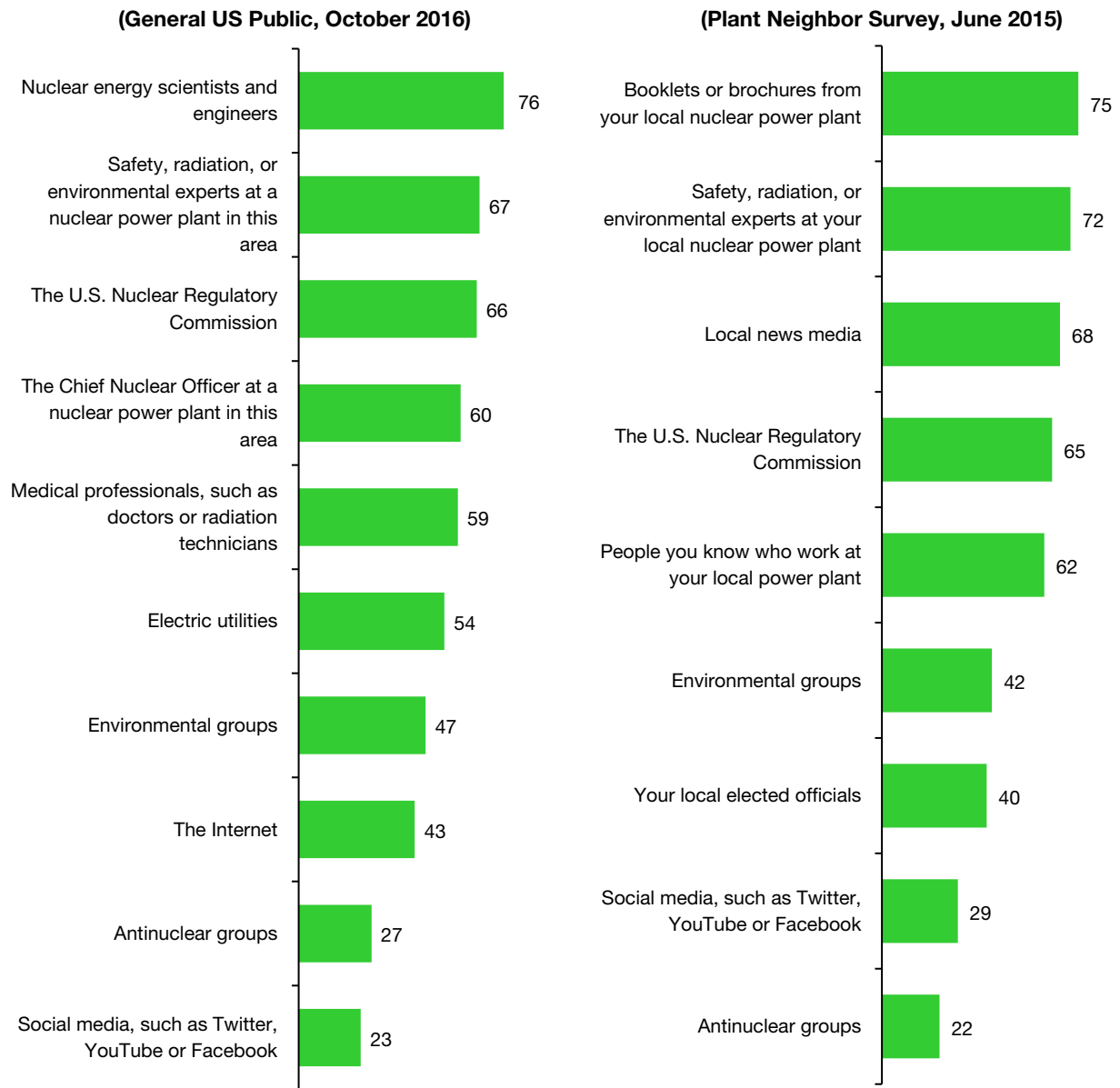
The survey continued: "Nuclear energy produces more than half of all the carbon-free electricity in the United States. And given that nuclear energy is the only electricity source that provides both clean air and continuous 24-7 electricity, do you think nuclear energy should be very important, somewhat important, not too important, or not important at all in the future?" Given these points, three-fourths said nuclear energy should be important (**Findings 16**).

In the 2019 national survey, respondents were asked to rate the importance of considerations for the way electricity is produced. All of them are attributes of nuclear energy. The considerations given highest importance were reliable electricity, affordable electricity, and clean air. More than 60% rated each of them "top" in importance. Other attributes given top or medium importance by almost everyone were: efficiency, energy security, economic growth, job creation, climate change solution, and resilience.

The concept of resilience, the ability to resist natural disasters, has been on the lips of policymakers recently due to dramatic disasters that many observers link to climate change. Nuclear energy plants can claim resilience, as their plants continue to provide electricity when many other types of plants are forced to shut down. That attribute trailed the others in the importance given, largely because the concept could have many meanings other than the ability to resist natural disasters (**Findings 17**).

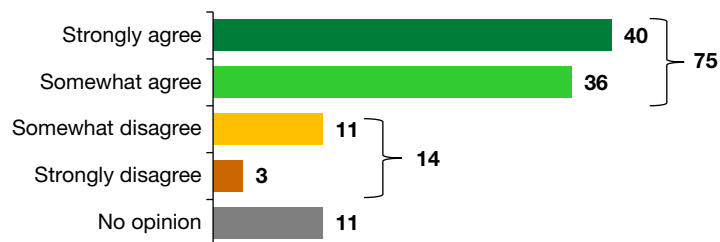
In multiple surveys, clean air scores as a top consideration in electricity production, but climate change does not. There is strong agreement across political parties on the importance of clean air but a partisan divide on climate change: fewer Republicans assign

Please tell me if you think each of the following would be an excellent, good, fair, or poor source of accurate and reliable information about nuclear energy. (%)



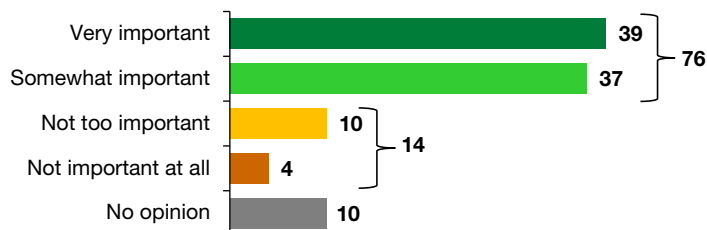
Findings 14 Percent rating information sources excellent or good.

Please indicate if you strongly agree, somewhat agree, somewhat disagree, or strongly disagree with the following statement. We should take advantage of all carbon-free energy sources including nuclear, hydro, and renewable energy to produce the electricity we need while limiting greenhouse gas emissions. (%)



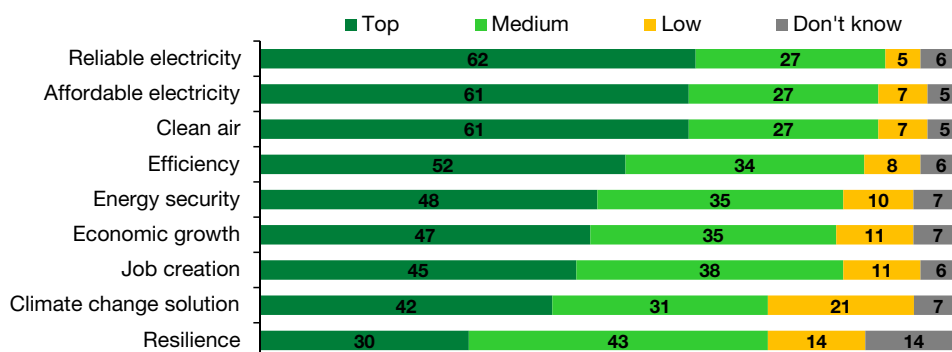
Findings 15 Agreement with using carbon-free energy mix, including nuclear, hydro, renewables (General US Public, April 2019).

Nuclear energy produces more than half of all the carbon-free electricity in the United States. And given that nuclear energy is the only electricity source that provides both clean air and continuous 24-7 electricity, do you think nuclear energy should be very important, somewhat important, not too important, or not important at all in the future? (%)



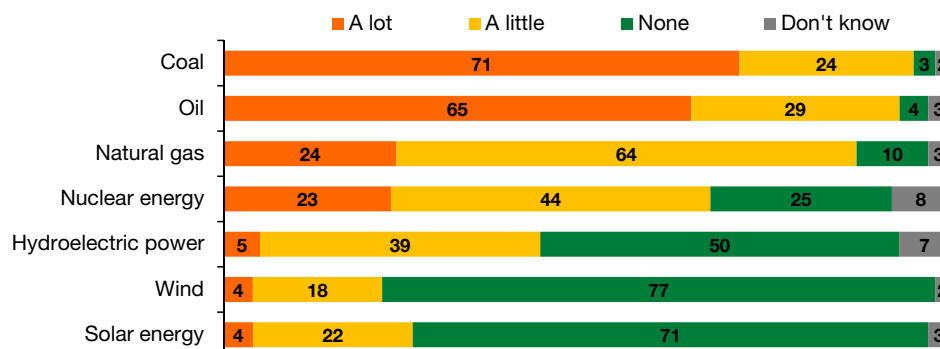
Findings 16 Importance of nuclear energy as a carbon-free electricity source (General US Public, April 2019).

Following are some considerations for the way electricity is produced. For each one, indicate if that consideration for the way electricity is produced is of top importance to you, medium importance to you, or low importance to you.



Findings 17 Top considerations for electricity production (General US Public, April 2019).

Greenhouse gases, such as carbon, may cause climate change. For each source of electricity that I'll read, please tell me if you think it releases a lot, a little, or no greenhouse gases, such as carbon, into the air. (%)



Findings 18 Perceived sources of greenhouse gas emissions (General US Public, March 2014).

top importance to climate change. Nevertheless, in 2019, 60% of Republicans assigned at least medium importance to climate change, indicating at least openness to climate change messages.

The challenge in gaining entry into climate change discussions is that few members of the public recognize nuclear energy's large clean air role. In a Bisconti Research survey in 2014, only 25% answered that nuclear energy releases no greenhouse gases, compared with more than 70% for wind and solar energy (**Findings 18**).

In 2019, Abacus Data asked a sample of 2500 Canadians about the carbon impact of six sources of electricity: "Does using these energy sources create more carbon pollution than oil, about the same or less?" Only 38% of the public knew that nuclear energy creates less carbon pollution than oil. That compares with 82% for solar, 80% for wind, and 63% for hydro.

Implications

Despite the majority public support, there is no doubt that a stigma still attaches to nuclear energy and all things associated with radiation. The origins of this stigma are well chronicled by Weart (2021). Education about radiation with attentive audiences such as students and Scout troops is helpful. Some communities encourage their residents to measure radiation by handing out Geiger counters; they are, in fact, empowering the public with hands-on tools. For the broad inattentive public, just hearing information about radiation could trigger the old ingrained phantasmagoric images. Instead, the best way to reduce the stigma is to make the beneficial uses of radiation normal and well known—familiar (Bisconti 2011). In the case of nuclear energy, the more people see it to be needed, the less they see danger. Useful information on nuclear energy's benefits is provided by Hor-Lacy (2021), Waltar (2021), and by Bezdek (2021).

Today, the public senses that nuclear energy is needed, but they do not articulate a focused reason why it is needed. Nuclear energy has an essential role as the one energy source that delivers a large amount of carbon-free energy reliably all day every day. Learning about nuclear energy's unique role for meeting climate change and clean air goals greatly increases support.

A focused message about nuclear energy's contribution needs to be heard widely in order to build support. In the words of World Nuclear Association Director General Rising (2019): "The 445 nuclear reactors in 30 countries are the low-carbon backbone of electricity systems, operating in the background, day in day out, often out of sight and out of mind, capable of generating an immense amount of clean power. They are the silent giants upon which we rely today." Silence is the enemy of winning public hearts and minds.

Time and effort by those who recognize nuclear energy's important clean air value will be necessary. They must focus communications on vision for the future and a world with clean and abundant energy. That may include the innovations that support this future vision. The vision of nuclear energy as a part of the solution to major societal concerns gains substance by telling the interesting story of nuclear energy innovation, including the ongoing work of the national labs.

The fact that the perception gap may be narrowing—the public perception of public opinion was more favorable in 2019—suggests that more positive messages about nuclear energy's contribution are reaching the public. Credit is due to the many nuclear professionals and environmental advocates who are giving valuable time to outreach.

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Relative Nuclear Energy Health Hazards and Popular Acceptance

Michael W Golay, Massachusetts Institute of Technology, Cambridge, MA, United States

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Abbreviation

GWe-Yr Gigawatt—Year

Introduction

Concerning any energy technology important questions are those of the human health and environmental hazards that it might pose and the cost—benefit balancing that could affect the acceptability of these hazards. For civilian nuclear energy experience and all except politically partisan analyses have indicated that the health and environmental risks of the technology are small compared to those of other energy technologies. In its current form the technology provides mainly electricity, a commodity available from many other technologies as well. Consequently, the benefits of using nuclear technology are not valued highly, at least in wealthy countries, and the risks, while small, are also sufficiently uncertain for many people that the balancing does not lead to nuclear acceptance. This balancing is sometimes different in rapidly developing societies where the energy supply diversity offered by nuclear is more appreciated, and its risks are seen as being relatively smaller.

Hazards overview

Table 1 compares the types of environmental and health hazards of many of the currently competing energy technologies. It is organized according to different facility life cycle stages, and reflecting accidental hazards. It is seen that technologies using fossil fuels have more entries in different categories, with coal being the most dangerous fuel, followed by petroleum and then by natural gas. Among the remaining technologies nuclear has fewer entries than those using fossil fuels, but more than some of the renewable energy technologies (e.g., wind and solar PV). Notably hydroelectric is a renewable energy technology that has been responsible for many fatalities killing scores of persons every few decades yet it is not viewed as being more hazardous than nuclear energy. Some technologies have chronic health consequences that show up as actuarial evidence, as with excess mortalities and morbidities from environmental pollution (e.g., coal and petroleum—to a lesser degree). Other technologies evidence accidental hazards, especially any involving underground mining (i.e., coal and uranium-in the past). Corresponding quantitative results from several studies are provided by [Haddad and Dones \(1991\)](#) and excerpted in **Table 2**.

Radioactive materials

Nuclear energy is special as it involves radioactive materials. The doses to workers are greater than to the general public, and for both groups are at levels where there is considerable expert disagreement from whether any effects exist to the position that the actual effects upon which public and occupational protection policies are based are in reality underestimated. The problem here is that the scientific foundation for reaching conclusions in this exposure range is still substantially nonexistent. The arguments that no effects should be expected range from expectations that natural repair of radiation effects will occur-given that the Earth's organisms have evolved in previously more intense radioactive environments than obtain today. The arguments that actual effects are being underestimated are based upon suspicions that all or most radiation effects are harmful. In democracies the actual experience with radiation has been that no radiation related fatalities or serious injuries have been observed in connection with the four reactors

Table 1 Matrix of environmental effects of various energy technologies.

<i>Environmental effects matrix</i>								
<i>Fuel/Phase</i>	<i>Coal</i>	<i>Petroleum</i>	<i>Natural gas</i>	<i>Nuclear</i>	<i>Hydro</i>	<i>Solar Power Tower</i>	<i>Fusion</i>	<i>Geothermal</i>
Extraction	Mining Accidents Lung Damage Aquatic Ecosystems Damage	Drilling-Spills (off-shore)	Drilling Fracking Effluents	Mining Accidents Lung Damage	Construction	—	H ² , Li Production	—
Refining	Refuse Piles	Water Pollution	—	Milling Tails	—	—	—	—
Transportation	Collision	Spills	Pipeline Explosion	—	—	—	—	—
On-Site								
Thermal	High Efficiency	High Efficiency	High Efficiency	Low Efficiency	—	Ecosystem Change	—	—
Air	Particulates-SO ₂ , NO _x	SO ₂ , NO _x	NO _x	BWR Radiation Releases	—	—	—	H ₂ S
Water	Water Treatment Chemicals	Water Treatment Chemicals	Water Treatment Chemicals	Water Treatment Chemicals	Destroys Prior Ecosystems, Starved Sediment Transport	Water Treatment Chemicals	Tritium in Cooling Water	Brine in Steams
Aesthetic	Large Plant Transmission Lines	Large Plant Transmission Lines	Large Plant Transmission Lines	Small Plant Transmission Lines	Small Plant Transmission Lines	Poor Large Area	Large Large Area	Poor Large Area
Wastes	Ash, Slag	Ash	—	Spent Fuel Transportation Reprocessing Waste Storage	—	—	Irradiated Structural Material	Cool Brine
Special Problems	—	—	—	—	—	—	Occupational Radiation Doses	—

Table 2 Summary of comparative assessments of energy technology risks.

<i>Technology</i>	<i>Public accident mortalities^a</i>		<i>Public disease mortalities^a</i>	
	<i>Low</i>	<i>High</i>	<i>Low</i>	<i>High</i>
Coal	< 0.001	1	< 0.001	8
Petroleum	< 0.001	0.1	< 0.001	8
Natural gas	< 0.001	0.2	< 0.001	0.2
Solar photovoltaic	< 0.001	2	< 0.001	2
Wind	< 0.001	0.7	< 0.001	0.1
Hydroelectric	< 0.001	> 0.01	< 0.001	> 0.01
Nuclear—LWR	< 0.001	> 0.01	< 0.001	> 0.01

^aEvents/GWe—year.Haddad, S and Dones R (1991) Comparative health and environmental risks for various energy sources. *IAEA Bulletin* 33: 14–19.

destroyed in the Three Mile Island and Fukushima Daiichi accidents. Also, in those cases the offsite radiation levels typically stayed below occupational exposure limits. However, these events were very scary. Democracies have shown that they are unwilling to tolerate such stressful experiences, regardless of how small their actual consequences turn out to be.

The only other core damage event involving civilian nuclear power was that in 1986 at Chernobyl, Ukraine, where 32 firefighters died due to acute radiation exposures. Many others in surrounding regions may have been exposed to lower doses. The reality will likely never be known as data on both exposures and effects were not well developed following the event. As a consequence, a broad

range of estimates of both have appeared. This event has largely been viewed as the consequence of operational practices and faulty reactor design reflecting causes associated with the societies of the Soviet Union and allied states, and not with nuclear energy operations encountered elsewhere (Sich, 2021). The Chernobyl experience carries a lesson of the need for care in nuclear power operations to ensure that operational practices are well-designed and adhered to. This lesson is probably most important in developing countries, especially those new to using nuclear technology, where the safety practices may not be as demanding and strictly adhered to as in the industrialized democracies. The overall lesson is that nuclear power has radiation hazards that can be managed successfully, but also that can be harmful if not managed carefully.

Most assessments of nuclear energy hazards reflect broad uncertainties, as the theoretical and data base needed for high accuracy have not yet been developed—reflecting a lack of clearly attributable harmful effects of radiation. A good example of this is reflected in Table 2, summarizing prior estimates of electricity related health risks.

Important causes of mortalities include explosions of natural gas pipelines, collisions with freight trains carrying coal, flooding following hydroelectric dam failures and facility construction accidents. Radiation related fatalities due to accidental releases from nuclear facilities have been widely feared but not realized, except in the case of the 1985 Chernobyl reactor explosion, which was more of a Soviet Union cultural failure than a technological accident (Sich, 2021). Public disease mortalities have arisen mainly from fossil fuel-related air pollution. In cases where the theoretical basis is substantially uncertain high values of estimated mortalities are associated with pessimistic assumptions, but without clear evidence, about the extremes of causality, as with radiation, water pollution, electromagnetic field and air pollution effects.

Radiation dose sources and public ignorance

Radiation exposures to the general public are dominated by those for medical (i.e., cancer) therapy and imaging, with natural sources of radiation (cosmic radiation and those from naturally occurring radioactive decay of elements that are part of the Earth crust), miscellaneous sources such as cosmic radiation encountered in high altitude flying. Cancer treatment doses are of the order of 1000 Rad or 10 Gy; naturally occurring and occupational doses are typically four orders of magnitude smaller. That is, industrial exposures, including those due to nuclear energy, are negligible (Nichols and Byrne, 2021). Despite this reality radiation associated with nuclear energy is and has long been viewed by many people as a serious hazard. This belief has been persistent over several decades, and appears to be fundamental to popular attitudes concerning the acceptability of the nuclear technology.

The combination of hypotheses has been offered to explain this persistence. The main elements identified include both scientific and popular ignorance, a literature reflecting scary themes dating back roughly a century (Weart, 2012, 2021), introduction of powerful nuclear technologies via use of nuclear weapons in 1945—followed about a decade later by the United States' leadership in establishing the UN's Atoms for Peace program to develop civilian nuclear energy internationally—all in the context of the cold war and nuclear competition with the Soviet Union until 1991.

The universe has been created and powered by nuclear reactions from the beginning of time (Previtali, 2021). Without them there would be no universe. Life on earth has evolved in a persistent and decreasingly intense environment of ionizing radiation. The biological effects of such radiation at high doses and rates has long been recognized as potentially harmful. The corresponding state of knowledge for low-dose and low-dose rate exposures still remains to be developed. Extrapolation from high dose experiences to low-dose ones has been made in order to permit occupational and public exposure standards to be formulated. However, it remains recognized that the basis for such extrapolation has not been well established. Rather, the linear, no threshold model of radiation dose-health effects has been adopted as a social expedient. The corresponding uncertainty has allowed various social interpretations concerning radiation and the risks of exposures to take hold. Some of these have led to strong opposition to use of alternative models ranging from ones of beneficial effects to those of more severe adverse effects. These disagreements are likely to persist, perhaps far into the future, until the state of biophysical understanding becomes much more advanced (for more on this see Nichols and Byrne, 2021).

Nuclear fission and societies

Despite their foundational roles in creating our reality nuclear reactions entered human consciousness only very recently, via discovery of radioactivity by Antoine Henri Becquerel at the end of the 19th century. Thus, it is not surprising that many people view nuclear phenomena as a recent alien intrusion; and a mysterious, dangerous threat.

Nuclear reactions were of little practical importance until the 1930s when the fission reaction was discovered by O. Hahn and F. Strassmann in collaboration with L. Meiter and O. Frisch. It is an essential phenomenon in both nuclear weapons and in civilian nuclear power. It occurs by exciting a heavy nucleus via neutron absorption. The excited nucleus splits typically into two fission fragments, and associated nuclear particles. In doing this it releases approximately a 100 million times the amount of energy obtained from a single chemical reaction. Thus, a small mass of nuclear fuel can store a large amount of potential energy, that when released can be very destructive (as with nuclear weapons) or can be a long duration energy source (as with nuclear powered ships). The fission reaction products also constitute nuclear wastes and create the radiation environment of nuclear energy which renders nuclear systems more difficult to manage, and their failures more difficult to recover from.

Initially support for both civilian and military nuclear applications was strong, reflecting their contributions to success of the allies in World War II and then in competition with the Soviet Union. This included the Atoms for Peace program, especially within the United States, where it originated, and where support was quite strong. A broad consensus existed that nuclear technologies, especially for naval propulsion, energy and medicine, were beneficial and that their use should be promoted internationally. Subsequent literature, (Slovic and Peters, 2006) has developed seeking to explain later popular rejection of technologies, including nuclear, seen to be hazardous. Causal factors cited include individual unfamiliarity with the technologies, remoteness of the facilities and their operators from ordinary experience, unknowability of the harmful agents (e.g., invisible, odorless, mysterious, as with radiation) among others. The validity of this literature is challenged by the previously broad acceptance of nuclear technologies. It is credible that other factors, such as a loss of trust in authority figures maybe more important.

Nuclear power in the United States

Such support persisted until 1963 when the US became involved in the Vietnam War, which lasted until 1975. That involvement coincided with the advent of the US civil rights movement, the environmental protection movement and growing opposition to civilian nuclear power—particularly by challenging its technological soundness, especially concerning safety. From the 1960s onward, a persistent criticism was that nuclear power was too dangerous to be used, and that the American policy elite had deceived society via its promotion. This charge can be understood as a means of opposing the war, where the means of opposition were opposition to the war itself, and challenging the legitimacy of national policies generally. As the civilian nuclear power program was a flagship US national enterprise attacking the program was an attractive way to oppose the war policy.

This view was reinforced later by the 1979 Three Mile Island accident, which caused no radiation related injuries, but was attributed to have been an important cause of mental health problems. More importantly, that event cause a loss of confidence in the leadership of the worldwide nuclear power enterprise. These changes overthrew the previously uncritical pro-nuclear consensus. Implicitly support for all of the popular movements mentioned here reflected broad dissatisfaction with national policies and constituted demands for change.

These factors have been influential to the point that a little less than half of the approximately 180 nuclear power reactors ordered in the United States ultimately did not go into operation, and only two reactors ordered since 1974 appear today to be approaching operation. Other factors contributing to this result have included the disappointing financial and regulatory experiences of many of the units initially ordered during the 1970s.

It is important to note that anxieties about radiation risks have not been substantiated, but still have powered an effective prohibition since the 1970s on the growth of the civilian nuclear power enterprise. This comes about not through a formal policy, but rather from creation of a highly unstable project management environment that renders the probability of successful completion of a new power reactor facility low enough to deter investors. The safety regulatory system is especially important in this, as it has ended up being unpredictable, slow and expensive to deal with. In the rest of the world growth of the nuclear power economies has been steadier, partially reflecting the more typical situations that nuclear safety regulation excludes citizen participation in licensing of individual units—a major source of construction project instability that is unique to the United States.

This system has also hindered innovation, such that the dominant nuclear power technology of today uses light water-cooled reactors, derived from submarine propulsion systems, that was introduced at the start of the Atoms for Peace Program for achieving a quick, inexpensive, politically-inspired success. It then became ensconced worldwide via the technology lock-in process. This technology is good enough for utility level electricity production, but not for important innovative roles such as aiding decarbonization of the global energy economy. Other reactor concepts, capable of much higher temperatures and large power capacity, consistent with safety and environmental compatibility, such as those cooled by gas, molten salts or liquid metals appear much better suited for such role (for more on this see Stanculescu, 2021). Some of these have been taken through the early stages of development and commercial introduction, but have not yet achieved large scale marketplace success.

The nuclear power enterprise has been engaged for more than six decades, resulting in the approximately 450 nuclear power reactors operating worldwide. In three countries it is the most important electricity source, and most industrialized countries have a nuclear power industry. It has an excellent occupational safety record, being one of the safest industries for workers in the world (although the quality of the results observed reflect differences in the different societies, all using essentially the same technology).

Weighing of risks and benefits

In the face of this positive record the question arises as to why nuclear power has not been embraced popularly since the end of the Vietnam War in 1975, at which time the anti-war genesis of nuclear opposition became moot. A plausible explanation is that this technology is viewed as being possibly hazardous and safe enough when managed well, but without strong redeeming benefits. For many people this combination can provide sufficient justification for being wary and reluctant to invest in efforts to understand matters more deeply. Civilian nuclear energy is simply one means of obtaining electricity, which is available at low costs from many sources. Consequently, this weak benefit-cost comparison can be a sufficient explanation for lack of enthusiasm for using this technology. The contrasting example of commercial flying, where hazards are evident on an actuarial basis yet the demand

for flying remain strong illustrates the effect of cost-benefit balancing upon popular technological acceptance. In the case of flying the convenience that it provides is valued sufficiently that people are willing to expose themselves to potential, and clearly evident, destruction in order to capture it. Should nuclear energy come to be seen as providing substantially higher benefits—as in saving the planet from global warming—conceivably popular preferences could change strongly and quickly.

Nuclear energy and climate change

Should nuclear energy come to be used more broadly throughout the world these culturally dependent outcomes may prove to be more important than those arising from technological factors. This dependence of nuclear success upon human decisions and practices is a little recognized, but very important aspect of factors affecting civilian nuclear power performance, and may be central to the future acceptability of the nuclear enterprise.

A particularly important reason for a much larger future role for nuclear energy is its ability to contribute to mitigating climate change. It is currently not exploited near to its potential in reducing use of fossil fuels, an essential condition for ending global warming. In the technological portfolio for such a response a combination of the renewable energy technologies, nuclear and efficiency improvements combined with an energy storage capability has become recognized as being economically more attractive than a sparser toolkit. This outcome results from the greater benefits of the complementary strengths of the diverse elements of the portfolio, and less need to utilize any particular technological options in its domain of decreasing benefits. For example, the intermittency of many of the renewable energy technologies causes the effective marginal costs of an expanded role to increase as its role grows. This is due to need for greater investments in back up energy sources such as batteries, and in the grid capable of dealing with a broader range of operations. This outcome can be avoided when the renewables are used in combination with steady generation sources such as nuclear energy. Recent studies (Sepulveda et al., 2018) indicate that costs of mitigation decrease and decarbonization success probabilities increase substantially by using a more diverse portfolio.

Currently persistent suspicion of the hazards of nuclear energy have discouraged its inclusion in the portfolio. This may be a dangerous choice for humanity, as current nuclear technologies can be useful, but are not designed for climate change mitigation. It takes time and investments to create better technological choices. If the decision to pursue them is delayed too long the needed results cannot be obtained in time for an effective implementation of new facilities. Thus, the stakes are high and decisions based upon perceived risks may deserve reexamination.

See Also: Estimation of Health Risks From Exposure to Ionizing Radiation.

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Securing Nuclear Power's Contribution to Decarbonization

John E. Parsons, Sloan School of Management, Massachusetts Institute of Technology, Cambridge, MA, United States

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Glossary

Baseload generation A generation plant that operates at roughly full capacity throughout the day and week.

Capacity expansion and dispatch model A mathematical program that optimizes the capacities across a portfolio of alternative generating technologies, and the dispatch of those capacities to generate electricity. A typical optimization criterion is lowest total cost, including investment cost and operating cost. A primary constraint is that electricity generation in each period exactly equal load in the period. Another constraint is that aggregate emissions remain below some cap.

Capacity factor The actual generation of a power plant or set of power plants over a window of time divided by the potential generation.

Carbon intensity Carbon emissions per unit of output. For the electricity sector one commonly used measure is grams CO₂ per kilowatt hour of generation (g CO₂/kWh).

Cross-cutting technologies Technologies that are relevant across different reactor concepts. For example, ultra-high performance concrete can be utilized for light-water reactors as well as for molten salt reactors.

Deep decarbonization A reduction in carbon emissions of between 80% and 100% from 1990 levels.

Firm, low-carbon resources Generation technologies that can reliably operate as needed, especially in hours when renewable resources such as wind or solar are unavailable.

Fuel-cycle services Manufacture and handling of nuclear fuel. Handling of used fuel and waste, including storage, processing and waste disposal.

Generation-III reactors It is common to classify reactor designs according to "generation." Generations I includes the early designs deployed by industry in roughly the 1950s and 1960s. Generation II designs begin in the late 1960s and include a large fraction of plants in operation today. Generation III are the most modern designs which began deployment in the 1990s. The overwhelming majority of Gen III reactors are Light-water reactors (LWRs).

Generation-IV reactors The next generation of reactor concepts, only now coming into actual construction and operation. The term is also often used to distinguish new, alternative reactor concepts from the light-water reactor (LWR) concept since most Gen IV concepts are not LWRs. For example, Gen IV concepts include high-temperature gas-cooled reactors, molten-salt reactors and liquid metal cooled fast reactors (Stanculescu, 2021).

Levelized cost of electricity (LCOE) The average cost of electricity generated by a given technology, including the investment cost of the installed capacity and the operating cost. The LCOE varies with utilization as measured by the capacity factor. A higher capacity factor lowers the LCOE since the fixed investment cost is amortized across a larger number of units of generation (Rothwell, 2021).

Light-water-reactor (LWR) This is the overwhelming majority of reactors in use today for generating electricity. Water serves as both the coolant and the neutron moderator. Light-water is the common form of water and stands in contrast to heavy water.

Load curve A representation of the temporal variation of aggregate load throughout the year. Usually reported at hourly granularity, and in the form of a duration curve with the hours ordered not in temporal sequence but by the size of the load.

Load-following generation A generation plant that cycles its output level with fluctuations in load, usually through the diurnal and weekly pattern.

LWR See light-water-reactor.

Peaker generation A generation plant that is utilized infrequently, to meet the peak hours of load.

Premature closure Generally refers to closure prior to the expiration of a plant's license or life as a result of economic pressures.

Production credits A monetary subsidy per unit of electricity sold. This is paid on top of the wholesale market price received from sales. Similar to production tax credits, except that it is a direct payment independent of the company's tax status.

Prototype reactor An early version of a reactor design, usually small scale and built to experiment with and demonstrate operation of the reactor. In general usage, there is significant overlap with the terms demonstration reactor and first-of-a-kind

reactor. In US Nuclear Regulatory Commission regulations the term prototype reactor has specific meaning with respect to licensing requirements.

Reactor concepts Reactor designs that differ in fundamental ways. For example, a light-water-reactor design is a distinct reactor concept from a gas-cooled reactor.

Renewable resource The determinant of output for renewable generation technologies, such as the specific power of the wind (watts per meter squared, W/m^2) or the level of solar irradiance (watts per meter squared, W/m^2).

Small modular reactors (SMRs) A new variant of light-water reactor (LWR) which is much smaller than the traditional LWR design. Multiple reactors, or modules, can be deployed within a single power plant to create flexibility in scaling the plant. Other reactor concepts beside LWRs can also be designed as SMRs (Ingersoll, 2021).

System cost The aggregate cost of serving load. Different technologies may serve different portions of the load curve, and the cost of serving these different portions vary. The system cost is the cost of all the technologies used to meet the full load curve. System cost is contrasted with LCOE since the LCOE is typically reported for a single technology serving a portion of the load curve.

Introduction

The 21st century confronts the world with the twin challenges of expanding energy access to billions of people while simultaneously slashing greenhouse gas (GHG) emissions. Electricity has been the world's fastest-growing form of energy for many decades and it will likely continue to be so for decades to come. In the US Energy Information Administration's (EIA) Reference Case scenario net electricity generation worldwide is growing by almost 84% to mid-century, from 24.1 trillion kilowatt hours (kWh) in 2017 to 44.2 trillion kWh in 2050 (U.S. Energy Information Administration, 2019). Non-OECD countries record the strongest growth at 2.3% per year, while in the OECD countries the average projected growth rate is 1% per year.

This expanded access to electricity will benefit the global population enormously, but the large GHG emissions it could entail represent a clear and present danger to humankind. Already, "[h]uman activities are estimated to have caused approximately 1.0 °C of global warming above pre-industrial levels..." according to the Intergovernmental Panel on Climate Change (IPCC, 2018). In order to limit the global temperature increase to around 1.5 °C or 2.0 °C, total CO₂ emissions need to be reduced to net zero by around 2050 or 2070, and be further reduced to net negative thereafter. In contrast, the EIA's Reference Case shows CO₂ emissions growing by almost 23% to mid-century.

Deep decarbonization of the electricity sector is key to simultaneously meeting the need for energy access while drastically reducing emissions. This is both because the costs of significant emission reductions in this sector are believed to be lower than in other major energy sectors, and because electrification is seen as a tool for decarbonizing other sectors such as transportation and building conditioning. Table 1 benchmarks the task. It contrasts the recent historical level of carbon intensity of the electricity sector with the 2050 goal established for a scenario developed by the International Energy Agency (2017) to be consistent with a 2 °C target for global warming. For example, the 2014 level in the United States was 486 gCO₂/kWh, while the 2050 goal is 11. For China in 2014 it was 698 g CO₂/kWh, while the 2050 goal is 24. Much more stringent goals are also widely advocated by various organizations and experts.

Nuclear power is a dispatchable, low-carbon technology. It is the largest source of low-carbon energy in the United States and Europe, and the second largest source worldwide after hydropower. However, the growth in low-carbon generation in recent years has been dominated by wind and solar technologies. Most outlooks for the future out to 2050 see this dominance continuing due to the falling cost of these technologies. While the increasingly urgent need for more low carbon electricity is an opportunity for the nuclear industry, the prospects for the expansion of the industry remain measured overall, and decidedly dim in many parts of the world. For example, the EIA's Reference case reports an average annual growth rate for nuclear of approximately 1% over the years 2018–50, while the average rates of growth reported for wind and solar technologies over this long period are 4.9% and 7.2%, respectively.

Table 1 Carbon intensity of the electricity sector (g CO₂/kWh), 2014 actual versus 2050 scenario goal.

	2014	2050
World	540	35
United States	486	11
China	698	24
Europe	350	11

Source: International Energy Agency (2017) *Energy Technology Perspectives, 2° C Scenario*.

The outlook for nuclear varies greatly by country and region. Anticipated growth is concentrated in non-OECD countries. The EIA's Reference Case reports an average annual growth rate for both China and India above 4% over the years 2018–50. Smaller non-OECD countries also have noteworthy plans to develop new nuclear capacity. For example, in the United Arab Emirates, four Korean-designed nuclear power reactors with combined capacity of 5.6 GWe are being installed at the Barakah site, with the first unit complete and scheduled to go on-line in 2020. In contrast, overall nuclear capacity is expected to decline in the United States and in the OECD portion of Europe. In Japan, capacity is expected to remain below one-half of the pre-Fukushima level through 2050.

Many factors are at play in these outlooks, but one fundamental problem is cost. As other low-carbon generation technologies have become cheaper in recent decades, new nuclear plants have only become costlier (Rothwell, 2021). This disturbing trend undermines nuclear energy's potential contribution and increases the cost of achieving deep decarbonization. A recent MIT study on *The Future of Nuclear Energy in a Carbon-Constrained World*, analyzed the opportunity as well as the steps needed to arrest and reverse that trend (Buongiorno et al., 2018). This paper summarizes some of those results.

The opportunity for nuclear energy in a decarbonized electricity system

Looking out to 2050, we asked the question, what are least-cost mixes of generation technologies to serve loads in diverse regions while achieving targeted reductions in carbon intensity, and where does nuclear power fit into those mixes? We paid particular attention to how the accelerated growth of variable renewable technologies, such as wind and solar Photo-Voltaic (PV), alter the optimal portfolio mixes and nuclear's place in these new mixes. To address the question, we applied a capacity expansion and dispatch model to the conditions in a variety of regions with different load and renewable resource patterns. We chose six regions in total, two in the United States—New England and Texas—two in China—the Tianjin, Beijing and Tangshan region and the Zhejiang region—and two in Europe—France and the United Kingdom.

We used the GenX model, which is a constrained optimization model that determines the least-cost mix of investments required to serve electricity demand in a future planning year (Jenkins and Sepulveda, 2017). The optimization criterion is total system cost, which includes the capital expenditures required to install the capacity, as well as the operating expenditures incurred running the capacities installed. The model assumes that once capacity has been installed, it is dispatched and operated to minimize the total system cost, subject to constraints. The constraints include the requirement that net generation equal load in each of the 8760 h of a representative year, taking into account the availability of storage and demand response. Hourly renewable generation is constrained by installed capacity and by the hourly availability of the renewable resource.

Generation units must operate within their technical constraints, such as minimum load, maximum ramping capacity and so on. For example, we assume that nuclear plants have a minimum generation level of 50% and can ramp at a rate of 25% per hour, while gas turbines have a minimum generation level of 24% and can ramp 100% per hour. Finally, aggregate CO₂ emissions must be within a specified constraint. GenX can be configured for different levels of detail. For example, it can incorporate opportunities for demand response, in which customers can displace consumption away from peak load hours, although the study discussed here did not include them. It has the capacity to incorporate certain defined transmission constraints, although the study discussed here treated each region as if there were no transmission constraints within the region and no trade outside the region. Trade across larger regions can serve as a partial substitute for firm, dispatchable generation, but we believe our chosen regions are large enough to extract useful lessons. The model can be parameterized to include existing capacity and to choose new investments, although the study discussed here focused on a greenfield mix for 2050—i.e., with no inherited capacity. The exception is hydro facilities which were fixed at the existing level.

The model requires inputs on the set of available technologies, their operating constraints, and the capital and operating costs. For example, the technologies included in the study discussed here were: utility scale solar PV, on-shore wind, LWR nuclear, natural gas with carbon capture and sequestration (CCS), coal with CCS, open cycle gas turbine (OCGT), combined-cycle gas turbine (CCGT), coal (IGCC), pumped-hydro storage, and battery storage.

This is obviously a limited set of technology options. However, the range of options is broad in terms of key characteristics. Moreover, technologies compete among each other in important ways. Adding additional technologies would provide granularity to the results, but would not affect the key conclusions.

A portfolio optimization that looks at total system cost is essential, and far superior to casual comparisons of levelized cost of electricity (LCOE) numbers for competing technologies. LCOE comparisons assume that technologies are competing head-to-head to serve similar loads. In reality, technologies are often best suited to serve certain portions of the load, and ill-suited to serve others, so that what may seem to be competing technologies are actually complementary technologies on a system.

Historically, because dispatchable thermal technologies have dominated most systems, portfolio optimization sorted technologies into categories such as baseload, load-following, and peaker. What primarily distinguished technologies was the tradeoff between fixed costs (mostly capital investment, but for nuclear also fixed operating costs) and variable costs (primarily fuel). For the baseload portion of the load curve which can be served with a high capacity factor, high fixed costs technologies can yield a low average cost of generation. For the peak load portion of the load curve, which can only be served with a low capacity factor, low fixed cost technologies yield a lower average cost of generation. The optimal mix of different types of technologies is determined by the shape of the load curve and the set of fixed and variable costs across technologies.

The introduction of renewables into the set of available technologies brings in a new factor: the annual profile of renewable resource availability. It is no longer enough to focus on the load curve itself. It is now necessary to appreciate the interaction between the load curve and resource availability. Expanding investments in renewables adds a high volume of generation to some hours of the day and year, but not others. While the dramatic drop in costs have made wind and solar PV the economic choice for incremental investment, as penetration expands the value of additional investment falls because it is not serving load in the most deficient hours. Other technologies are required to serve these hours, or to store renewable generation and deliver it to these hours. The growth in intermittent renewables forces us to move from the old taxonomy of baseload/load following/peaker, to a new portfolio paradigm proposed by Sepulveda et al. (2018) in which resources like nuclear compete as “firm” low-carbon resources complementing intermittent renewables. Jenkins et al. (2018) survey a variety of studies on deep decarbonization pathways and identify this role as a critical one. Our implementation of GenX addresses this issue and brings out the opportunity for nuclear and the challenge of cost.

Full details on the cost assumptions are available in the MIT study report, however, a couple of key figures are useful to call out here. The nominal or base case assumption for the overnight capital cost of nuclear in the two US regions for 2050 is set at \$5500/kW in 2014 dollars. A widely used metric for benchmarking the costs of alternative generation technologies is the Levelized Cost of Electricity (LCOE), so we can concisely describe our assumptions by translating them into an LCOE. Assuming the nuclear plant operates at a 90% capacity factor, our assumptions on the cost of nuclear translate into an LCOE of just over \$100/MWh. In contrast, assuming capacity factors for wind and solar of 34% and 25%, which are in the neighborhood of the available resource factors across our US regions, our assumptions on the cost of wind and solar PV translate into LCOEs of \$72/MWh and \$52/MWh. So nuclear is an expensive alternative in our US regions from the viewpoint of an average unit of generation. Of course, as mentioned above and as will be apparent in the results below, it may be a valuable part of a portfolio because of its capability to generate in hours when the renewable technologies are less available.

The results reported below also consider a case in which the capital cost for nuclear is decreased by 25% to \$4100/kW in 2014 dollars. Where the base case US capital cost assumptions for nuclear are high relative to the cost of other technologies, our assumption for China is markedly lower, at \$2800/kW in 2014 dollars, based on a comparison of US and China costs reported by International Energy Agency (2015). The study contains a number of sensitivity analyses on various other parameters such as the cost of storage.

Fig. 1, below, shows the optimal builds of new nuclear capacity in the New England region depending upon the level of decarbonization targeted and upon the assumed cost of a nuclear plant. The blue bars show the results for the nominal case assumptions. Given the high cost assumed for nuclear and the quality of renewable resources in New England, significant emission reductions are accomplished without nuclear: both the 100 and the 50 g-CO₂/kWh carbon intensity targets are achieved at lowest cost without any nuclear capacity. However, pursuing deep decarbonization brings nuclear into the optimal mix. Given a peak demand of approximately 34 GW, at the deep decarbonization levels the optimal capacity jumps to the neighborhood of 17 GW, which would be a dramatic increase for this one region of the United States.

Fig. 1 shows that for deep decarbonization (10–1 g CO₂/kWh) nuclear is essential to achieve that goal. Fig. 1 also shows that lowering the cost of nuclear makes a dramatic difference to the scale of nuclear. It is a potentially valuable player in Low C, but cost becomes a determining factor to scale.

Fig. 2, below, reports the average generation cost at each decarbonization target and for different assumptions about nuclear. In all cases, costs increase as the target for carbon intensity becomes tighter. The red bars show the impact on costs of excluding nuclear from the mix.

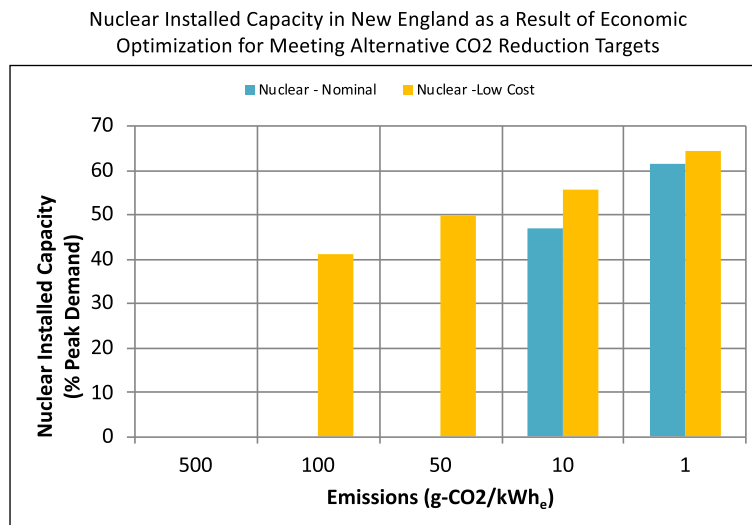


Fig. 1 Nuclear installed capacity in New England as a result of economic optimization for meeting alternative CO₂ reduction targets.

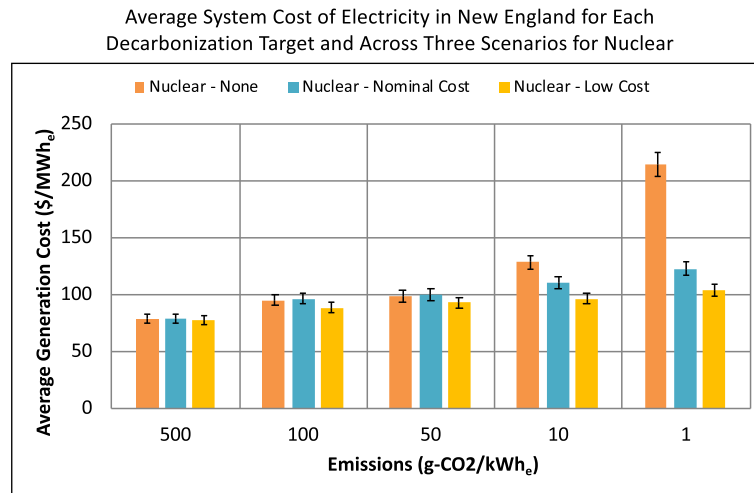


Fig. 2 Average system cost of electricity in New England for each decarbonization target and across three scenarios for nuclear.

Figs. 3 and 4 show these results for the Tianjin, Beijing and Tangshan (T-B-T) region of China where the renewable resources are poorer and the relative cost of nuclear is smaller. Consequently, nuclear is a part of the optimal mix even when targeting modest emission reductions.

Fig. 4 shows that excluding nuclear is very expensive in terms of climate change mitigation. What the GenX analysis shows is that the use of nuclear energy in regions and nations is regionally dependent which one would assume given the variability of solar and wind as well as the costs to deploy the technologies.

Industrial strategy to reduce the cost of nuclear power

In western Europe and the United States, a string of new nuclear plant construction projects have suffered severe construction delays and cost escalations. These include the projects at Olkiluoto, Finland, Flamanville, France, Hinkley Point C, United Kingdom, and the Vogtle and Summer projects in the United States. The cost overruns at Vogtle caused the reactor supplier Westinghouse to declare bankruptcy, and at Summer they eventually led to the project being abandoned midway. In all of these cases, the ultimate cost proved to be far above the nominal cost used in the analysis above, not to speak of the low-cost case. It must be said, however, that many nuclear plant construction projects in South Korea, China, Russia, and the United Arab Emirates have been completed more or less on time, and likely at significantly lower cost than comparable projects in the West, although it is often challenging to

Nuclear Installed Capacity in the Tianjin, Beijing and Tangshan (T-B-T) Region of China as a Result of Economic Optimization for Meeting Alternative CO₂ Reduction Targets

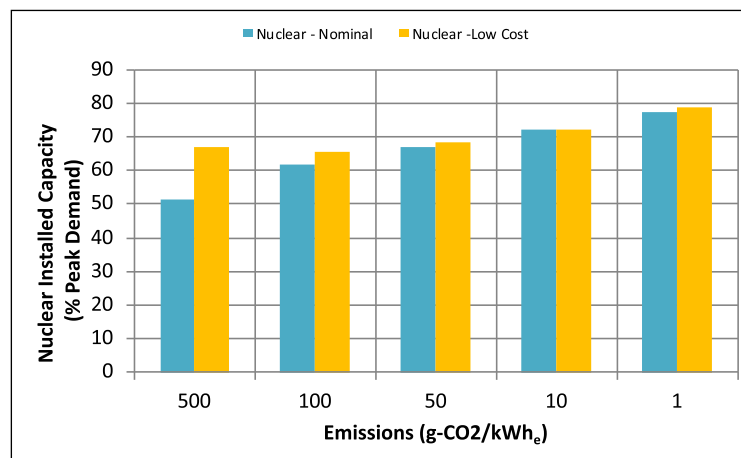


Fig. 3 Nuclear installed capacity in the Tianjin, Beijing and Tangshan (T-B-T) Region of China as a result of economic optimization for meeting alternative CO₂ reduction targets.

Average System Cost of Electricity in the Tianjin, Beijing and Tangshan (T-B-T) Region of China for Each Decarbonization Target and Across Three Scenarios for Nuclear

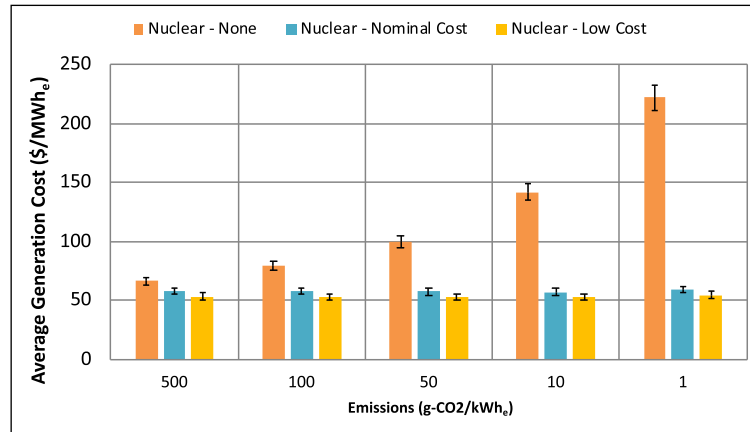


Fig. 4 Average system cost of electricity in the Tianjin, Beijing and Tangshan (T-B-T) Region of China for each decarbonization target and across three scenarios for nuclear.

independently validate cost figures published for those projects. Since reducing the capital cost of new nuclear power plants is an existential issue for the nuclear industry and for the pathways to decarbonization, our study made a detailed investigation into the drivers of cost and developed detailed recommendations about how to reduce cost.

The equipment for the reactor and the nuclear steam supply system (NSSS) represents a small share of the total capital cost of a new reactor, as does the equipment for the power island—typically less than 20% in total. The majority of plant costs are in the site preparation, civil works, equipment installation, and project engineering. It is also in these categories where many projects experience large delays and cost overruns. The duration of construction also adds an enormous financing cost. Therefore, cost reduction efforts need to be focused not on the NSSS design or the specific reactor technology but on (a) improvements in how the overall plant is constructed (or delivered to the site), and (b) ways to accelerate the construction process.

Opportunities exist to significantly reduce the capital cost and shorten the construction schedule for new nuclear power plants. First, the deployment of multiple, standardized units, especially at a single site, affords considerable learning from the construction of each unit. In the United States and Europe, where productivity at construction sites has been low, we also recommend expanded use of factory production to take advantage of the manufacturing sector's higher productivity when it comes to turning out complex systems, structures, and components. The use of an array of cross-cutting technologies, including modular construction in factories and shipyards, advanced concrete solutions (e.g., steel-plate composites, high-strength reinforcement steel, ultra-high performance concrete), seismic isolation technology, and advanced plant layouts (e.g., embedment, offshore siting), could have positive impacts on the cost and schedule of new nuclear power plant construction. For less complex systems, structures, and components, or at sites where construction productivity is high (as in Asia), conventional approaches may be the lowest cost option.

It is important to emphasize the broad applicability of these recommendations across all reactor concepts and designs. Cost-cutting opportunities are pertinent to evolutionary Generation-III LWRs, small modular reactors (SMRs), and Generation-IV reactors. The converse is true as well: Without design standardization and innovations in construction approaches, we do not believe the inherent technological features of any of the advanced reactors will produce the level of cost reductions needed to make nuclear electricity competitive with other generation options. The challenge for any proposed plant design is to achieve the radical cost reductions needed to make a new nuclear plant competitive in the on-grid electricity market.

In addition, a continued shift toward reactor designs that incorporate inherent and passive safety features can further reduce the probability a severe accident occurs while mitigating the offsite consequences in the event an accident does occur, and can ease the licensing of new plants and accelerate their deployment in developed and developing countries. These include core materials that have high chemical and physical stability, high heat capacity, negative reactivity feedbacks, and high retention of fission products, together with engineered safety systems that require limited or no emergency AC power and minimal external intervention, will likely make operations simpler and more tolerable to human errors. Such design evolution has already occurred in some Generation-III LWRs and is exhibited in new plants built in China, Russia, and the United States. Advanced reactors like LWR-based SMRs (e.g., NuScale) and mature Generation-IV reactor concepts (e.g., high-temperature gas reactors and sodium-cooled fast reactors) also possess such features and are now ready for commercial deployment. The first commercial high temperature pebble bed gas reactor is scheduled of operation in China in 2020.

Government policy

Government policy can facilitate nuclear power making a valuable contribution to decarbonization. First, decarbonization policies need to create a level playing field that allows all low-carbon generation technologies to compete on their merits. Existing nuclear

power plants are usually a cost-efficient source of low carbon electricity, even when building a new plant might not be. Nevertheless, in a number of markets, supports for low-carbon generation exclude nuclear and alter market prices so that these cost-efficient plants are made unprofitable. Several have closed already and others may close soon. Elsewhere, the policies more directly target closures. Premature closures of existing plants undermine efforts to reduce carbon dioxide and other power sector emissions and increase the cost of achieving emission reduction targets. Fixing these policies is not just relevant for existing plants. It is also necessary to properly incentivize innovations in nuclear energy technology and investments in future new builds of nuclear plants where appropriate. Investors pursuing new, truly low-cost nuclear designs must see the possibility of earning a fair profit based on selling their products for their full value, including the social value of providing low-carbon generation. Current policies that disregard the social value of nuclear energy's contribution to climate change mitigation reduce the prospect for profit and discourage investments in nuclear innovation.

Second, there are direct ways for governments to enable and support development of new, cost-efficient reactor designs. In our report, we recommend that governments should establish reactor sites where companies can deploy prototype reactors for testing and operation oriented to regulatory licensing. Such sites should be open to diverse reactor concepts chosen by the companies that are interested in testing prototypes. The government should provide appropriate supervision and support—including safety protocols, infrastructure, environmental approvals, and fuel-cycle services—and should also be directly involved with all testing. We also recommend that governments should establish funding programs around prototype testing and commercial deployment of advanced reactor designs using four levers: (a) funding to share regulatory licensing costs, (b) funding to share research and development costs, (c) funding for the achievement of specific technical milestones, and (d) funding for production credits to reward successful demonstration of new designs.

Conclusion

Nuclear power has a vital role to play in simultaneously expanding energy access while decarbonizing generation. Its ability to provide firm, low-carbon electricity is a complement to variable renewable generation like wind and solar PV. However, in the United States and Western Europe, the high cost of constructing new plants is a major obstacle to an expanded role for nuclear power. This makes decarbonization more expensive and limits energy access. There are important advances that have been made in a variety of non-nuclear technologies that can be employed in the construction of nuclear power plants to significantly reduce the cost of construction. These cross-cutting technologies can be employed across various reactor concepts and designs, and it is the effective application of these technologies that is likely to be the determining factor in cost reduction rather than the choice of reactor concept. Government policy has a role to play in encouraging and facilitating these innovations and cost reductions. Carbon policy needs to recognize the low-carbon value of nuclear generation alongside other low-carbon technologies in order that nuclear investments can be profitable. Governments can also support innovation by creating facilities where prototypes of new designs can be built and their operation demonstrated.

See Also: Introduction to Nonelectric Applications of Nuclear Energy.

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The Environmentalist's Dilemma

Andrew C. Kadak, Kadak Associates, Inc., Port St. Lucie, FL, United States

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Introduction

The issue of climate change has faced the environmental community with a difficult dilemma as it relates to the use of nuclear energy. On the one hand, nuclear energy is demonstrably reducing CO₂ emissions. On the other hand, most environmental organizations have opposed the use of nuclear energy for safety reasons and concerns about nuclear waste disposal (Daily Caller, 2015). At present, worldwide there are over 400 operating nuclear plants producing 10% of world's electricity without emission of greenhouse gases. In some countries the proportion of nuclear generated electricity is much larger. For example France produces 72% of its electricity from nuclear energy. Other surprising countries and their nuclear contributions are: Switzerland (38%), Finland (32%), Sweden (40%), United States (19%), Ukraine (56%). In all 440 nuclear plants are deployed in the world with 55 under construction.

It has been estimated that in 2018 these nuclear plants have avoided the release of 60,000,000,000 tonnes of CO₂. Nuclear is the largest base load source of clean energy in the world. While the environmental community recognizes this fact, they are still unwilling to lend their support to their continued operation and further expansion to help reduce CO₂ emissions. Their hope is that through conservation, use of solar and wind energy, taking the “nuclear risk” is not needed (Follett, 2015). These hopes do not reflect the fact that the use of variable energy sources such as wind and solar cannot reliably supply clean electricity on a 24 h per day, 365 days per year operation without massive investment in either energy storage and/or nuclear power reactors. Nuclear energy can make clean electricity on a scale needed to make a difference in combating climate change with wind and solar energy. What would be a desirable outcome is for the environmental community to actively support the expansion of safe nuclear plants in addition to solar, wind, and other renewable sources of energy.

The objective of this article is to highlight reasons for why the environmental community is opposed to nuclear energy; summarize the positions of key national and international environmental organizations; review reasons why some key environmental opinion leaders have changed their positions to become more supportive of nuclear; summarize why some environmental organization support nuclear and finally make the case for why all people, especially those in the environmental community should

support nuclear energy. We conclude that evidence shows that without nuclear energy, in combination with renewables, worldwide success in dealing with climate change is in jeopardy and will be much more costly.

Who and What is the “Environmental Community”

The “Environmental Community” is a self identified group of like minded people who believe their mission is to protect the environment as they see it from human activities. Their mission is narrowly focused on environmental issues typically without regard to the benefits that the human activity may provide. They rarely consider the likelihood of the risks posed but are rather absolutist in their positions. These groups raise funds from people who believe in their causes and use events that may arise in their fund raising campaigns many times exaggerating claims of damage to encourage more funding. Huge contributors to these organizations are green-leaning philanthropies. Grants from 2011 to 2015 totaled close to \$ 500 million none of which supported the development of nuclear energy. \$175,000 was actually dedicated to opposing nuclear energy. (Bryce, 2018). Many of the supporters of these groups do not have a full understanding of the issues but support the groups based on a perceived environmental good. Some groups get government grants to support some of their activities. While their goals are laudable, they, at times, do not consider the totality of the impact of their efforts and may cause negative reaction by the public to overly aggressive actions or dramatic claims.

The leadership of these organizations are typically well versed in the sciences and technology supported by a board of directors with highly regarded reputations. The challenge for the leadership is that should they take a contrary but correct position on an environmental issue that their supporters who are not as well informed but are committed to a position, will stop funding the organization. This is a main impediment to changing views of environmental groups to issues such as support of nuclear energy. In an era of dramatic climate change where constructive action is needed to make a difference according to their own positions, it is imperative that the leadership of these groups begin an education campaign on all energy options available including nuclear to enable the outcomes they seek.

Positions of key environmental groups

World Wildlife Fund

The best summary of why environmental groups oppose nuclear energy is a position statement issued by the World Wildlife Fund in May 2003 (World Wildlife Fund, 2003). Their position summarized below:

While the WWF's priority is to “accelerate implementation of solutions to combat global warming,” they oppose the use of one of the largest clean air electricity sources used today, namely nuclear energy. They do so because they incorrectly assume it is “not sustainable.” The reasons for this belief in their words are:

- “The entire commercial chain of the processing of nuclear raw materials from nuclear mining; operating nuclear power stations; handling nuclear waste and finally re-processing, is full of leaks contamination and produces a highly toxic legacy for thousands of years to come.
- The creation and handling of highly toxic nuclear products and the unsolved issue of safe storage of waste demonstrates the unsustainability of the technology.”

Kadak (2021) and Greneche (2021) adequately address the fallacy of their position however they cite other reasons such as the technology is particularly dangerous and hard to control as the accidents at Three Mile Island (1979), Chernobyl (1986) and Fukushima (2011) demonstrate. These accidents are discussed on other sections of this encyclopedia. Two of these nuclear accidents occurred from 25 to 40 years ago with the more recent Fukushima accident caused by a tsunami not any technical fault of the reactors. They also argue that investments in nuclear energy take funding away from other renewable energy sources which are lower cost. Parsons (2020) explains the deep decarbonization is only possible with a strong mix of nuclear and other renewable energy sources and would be much more expensive to implement even with the higher cost of nuclear plant construction. They also ignore the value of nuclear heat in process heat applications such as desalination of water.

What is unfortunate, nuclear energy plants are a quantifiable source of non-CO₂ electricity saving on a world wide basis 60,000,000,000 tonnes of CO₂ emissions per year which are threatened by the lack of the environmental movement's active support. The electricity generated by nuclear energy plants will be very difficult to replace with variable solar and wind energy plants.

Despite these obvious facts, their anti-nuclear views are shared by many of the other environmental organizations as summarized below.

Sierra Club

The Sierra Club maintains its historical anti-nuclear position based on the three nuclear accidents they refer to as “disasters.” They point out that the Fukushima tsunami which caused the death of more than 20,000 people none of which were caused by the nuclear plant as evidence that the fundamental problems of nuclear power have not been addressed (Sierra Club, 2020) (<https://www.sierraclub.org/nuclear-free>).

Natural Resources Defense Council (NRDC)

Although the NRDC recognizes the “magnitude and urgency” of climate change and the US must consider all forms of energy, they reject nuclear power since it suffers from “too many security, safety and environmental problems.” They also point out that the costs are high while ignoring the high cost of solar and wind alternatives (Cochrane et al., 2005) (<https://www.nrdc.org/sites/default/files/power.pdf>).

Greenpeace (US and Europe)

The Greenpeace organization claims that nuclear energy is expensive and dangerous and poses a risk of a nuclear meltdown. They reject the evidence of nuclear's clean energy contribution today while emphasizing the risks and ignoring the benefits. It is interesting to note that they, as have others in the anti-nuclear community are now using nuclear energy's relatively high cost compared to natural gas as a reason for excluding nuclear. They are, however, willing to pay the high price of solar and wind energy (Greenpeace, 2020) (<https://www.greenpeace.org/usa/global-warming/issues/nuclear/>).

Union of Concerned Scientists

What is most interesting is that of all the historically anti-nuclear groups, the Union of Concerned Scientists has recognized the importance of nuclear energy to combat the global impacts of climate change. In their most recent report on the subject “The Nuclear Power Dilemma: Declining Profits, Plant Closures, and the Threat of Rising Carbon Emissions,” they support a national carbon price or low carbon electricity standard that would help avoid premature shutdown of existing nuclear plants. They suggest that such a carbon price policy would help the USA meet its commitment under the Paris climate agreement (Union of Concerned Scientists, 2015; World Nuclear Association, 2018) (<https://www.ucsusa.org/resources/nuclear-power-global-warming>, <https://www.world-nuclear-news.org/Articles/UCS-calls-for-policy-to-preserve-nuclear>).

Environmental Defense Fund

The position of EDF on nuclear has changed from hostile to neutral finally recognizing that it is indeed a clean energy source. They have recognized that prematurely shutting down existing nuclear plants will result in more CO₂ emissions since they will be replaced with natural gas plants since they are cheaper than solar or wind. They grudgingly support subsidies to keep existing plants open until renewable are cheaper. They do not appear to support the construction of new nuclear plants but they do not oppose them either (Environmental Defense Fund, 2017; Finnigan, 2017) (https://www.edf.org/sites/default/files/content/Clean_energy_fact_sheet_Nov_2017.pdf, <http://blogs.edf.org/energyexchange/2017/04/17/why-we-still-need-americas-nuclear-power-plants-at-least-for-now/>).

Example of change agents in the environmental community

Perhaps the most significant change agents were climate scientists Dr. Ken Caldeira, Senior Scientist, Department of Global Ecology, Carnegie Institution, Dr. Kerry Emanuel, Atmospheric Scientist, Massachusetts Institute of Technology, Dr. James Hansen, Climate Scientist, Columbia University Earth Institute, Dr. Tom Wigley, Climate Scientist, University of Adelaide and the National Center for Atmospheric Research. In 2013, they published a letter to the environmental community which is included in the **Appendix A** to this article. This letter called for those in the environmental community who were opposed to nuclear power to “support the development and deployment of safer nuclear power systems as a practical means of addressing the climate change problem.” They fear “that continued opposition to nuclear power threatens humanity's ability to avoid dangerous climate change.”

Their letter recognized some inherent risks of nuclear energy and called for more advanced designs to reduce the risk but they concluded that the risks of nuclear energy are orders of magnitude smaller than the risk of fossil fuels. They conclude that “there is no credible path to climate stabilization that does not include a substantial role for nuclear power” and urges the environmental community to “support the development and deployment of advanced nuclear energy.”

It is not clear whether their letter triggered other prominent people in the environmental community to change their views but a brief list of people who now see nuclear energy as vital to our efforts to combat climate change is listed below: (Brook, 2011).

Anti- to Pro:

George Monbiot (environmental journalist, went anti- to neutral in 2009, pro- in 2011);
Mark Lynas (author of “Six Degrees”, 2008);
Stewart Brand (author of “Whole Earth Catalog”, 2005);
Dick Smith (Australian entrepreneur, 2011);
Ben Heard (eco-consultant, 2010);
Patrick Moore (co-founder of Greenpeace, 2003);
Stephen Tindale (former director of Greenpeace, 2009);
Chris Smith (chairman of the Environment Agency, 2009);
Chris Goodall (Green Party activist and prospective parliamentary candidate, 2009);
Gwyneth Cravens (author, 2007);
Hugh Montefiore (a founder of Friends of the Earth, 2004);
Chicco Testa (Italian politician, 2008);

Ian McEwan (author, 2010);
 Steve Kirsch (US entrepreneur, 2008); [Tony Blair](#) (former UK PM, 2006);
 Al Franken (Democrat Senator and media host, 2011);
 Michael Shellenberger (Activist, Founder of Breakthrough Institute).

Neutral to pro:

James Lovelock (Gaia theory, 2004);
 Bob Hawke (former Australian Prime Minister, 2008);
 Bob Carr (former NSW Premier, 2008);
 Jared Diamond (scientist and author, 2005);
 Jesse Jenkins, (Breakthrough Institute, 2008–2010).

“Can do with 100% renewables” to “nuclear required”:

Jim Hansen (climatologist, 2008);
 Barry Brook (biologist and risk modeller, 2009);
 David Mackay (physicist, maybe, 2008);
 Jeffrey Sachs (economist, 2009).

Anti- to neutral:

Chris Huhne (UK Lib Dem Energy Minister, 2010);
 Paul Ehrlich (conservation biologist, 2010).

There are likely many more as evidence mounts that without nuclear energy, climate change goals will be extremely difficult if not impossible to attain (<https://bravenewclimate.com/2011/04/15/who-has-changed-their-mind/>).

Described in more detail below are people and organization who believe that nuclear energy is vital to dealing with climate change with some reasons these individuals or groups reached this position.

Michael Shellenberger

A key environmental leader who changed his position from anti-nuclear activist to being an advocate is Michael Shellenberger who co-founded the Breakthrough Institute and is a founder of Environmental Progress. In 2008, he was named by Time magazine as one of the “Heroes of the Environment.” His conversion from anti to being pro nuclear was based on several key points:

- Solar and wind are intermittent sources of energy which require back up power.
- The cost of renewable energy is higher than alternatives.
- Life cycle carbon emissions from nuclear are comparable to solar and wind and much lower than natural gas or coal.
- The safety record of nuclear plants is far better than other sources of energy which they are replacing such as coal, oil and natural gas and equivalent to solar and wind including the major accidents experienced over 50 years in nuclear plants ([OECD, 2010](#); [Mackay, 2015](#)).
- British Medical Journal Lancet finds that nuclear power is already the safest way to make electricity compared to 7 million deaths per year from air pollution from fossil plants ([Markandya and Wilkinson, 2007](#)).
- The lack of scientific evidence of health effects from the Chernobyl accident outside of the Soviet Union. He was particularly struck by the finding that there was no increase in cancer even from the cohorts of people who were fighting the fire at the plant and those that cleaned up afterwards.
- Regarding Fukushima, he notes that no one died from radiation exposure but rather 1500 people who did die were the result of the evacuation and stress caused by the accident. This compared to the 20,000 or so who died as a result of the tsunami that engulfed the mainland of Japan ([World Health Organization, 2020](#); [UNSCEAR United Nations, 2014](#)).
- Regarding the issue of nuclear waste, he notes that nuclear waste is the only waste from the production of electricity that is “contained.” He points out that the toxic heavy metals waste from solar plants does not have a plan for their disposal which is 300 times that produced by nuclear energy which will be safely disposed of in a geological repository.
- In terms of weapons proliferation, his review concluded that no nation that started with a commercial nuclear program ever developed nuclear weapons. He contrasts that with North Korea that does not have a nuclear generating plants but does have nuclear weapons.

In his final summary, he concludes that “If Germany hadn’t closed its nuclear plants, its emissions would be 43% lower than it is today. And I think if you care about climate change, that’s at least something that you have to wrestle with, especially in light of some of these facts about the harms and benefits of different energy sources” ([Shellenberger, 2017, 2018, 2019](#)) (<https://www.youtube.com/watch?v=ciStd9Y2ak>, <https://www.forbes.com/sites/michaelshellenberger/2018/05/08/we-dont-need-solar-and-wind-to-save-the-climate-and-its-a-good-thing-too/#2e2846bde4de>, <https://quilllette.com/2019/02/27/why-renewables-cant-save-the-planet/>, https://www.who.int/ionizing_radiation/a_e/fukushima/faqs-fukushima/en/, <https://www.thelancet.com/action/showPdf?pii=S0140-6736%2807%2961253-7>, http://www.unscear.org/docs/publications/2013/UNSCEAR_2013_Report_Vol.I.pdf, <https://www.oecd-nea.org/ndd/reports/2010/nea6861-comparing-risks.pdf>, http://www.inference.org.uk/withouthotair/c24/page_168.shtml).

Breakthrough Institute

The Breakthrough Institute was founded in 2008 by Michael Shellenberger and Ted Norhaus whose visions for the environmental movement were different than those of the past. They believed that to truly deal with climate change a new approach was needed. In 2004, they published an essay entitled "The Death of Environmentalism" which pointed out that "20th century environmentalism cannot address complex global environmental challenges like climate change." Their focus was on making clean energy "cheap" that would not affect global economic development at the expense of addressing climate issues which is needed to address the needs of billions of people trying to live "modern" lives. They call for substantial innovation in energy production to meet these aggressive goals (Breakthrough Institute, 2020).

As an organization, the Breakthrough institute believes that nuclear energy is essential to the clean air revolution. They call for nuclear innovation, (Lovering et al., 2017), identify opportunities to make nuclear energy cheap, (Lovering et al., 2013) and how to "plant the seeds for a nuclear revolution," (Lovering et al., 2018). They conclude that despite impressive shifts in energy production in Germany, California, Denmark and other nations, no large economy has been able to power all of its electric grid with only renewables. They bemoan the fact that despite nuclear energy's clean power, environmental organizations continue to push for their closure making it an illogical argument given their view that climate change is an "untenable threat to human societies." They urge their peers to change their positions based on facts not politics. They recognize that there is a political battle to be won to deploy more nuclear plants but the battle is worth waging for all including environmental groups historically opposed (Nordhaus, 2018; McBride, 2018) (<https://thebreakthrough.org/articles/how-to-make-nuclear-innovative>, <https://thebreakthrough.org/articles/how-to-make-nuclear-cheap>, <https://thebreakthrough.org/articles/planting-the-seeds-of-a-distributed-nuclear-revolution>, <https://thebreakthrough.org/issues/energy/a-new-day-for-nuclear-advocacy-and-environmentalism>, <https://thebreakthrough.org/issues/energy/nuclear-for-1-5-c>).

Conservation Law Foundation

The Conservation Law Foundation was one of the lead organizations seeking to block the construction and operation of the Seabrook nuclear plants in New Hampshire in the 1970s and proudly claims that it helped cancel the second unit at Seabrook. Today, it is concerned that with the tendency to shutdown existing nuclear plants in New England, Yankee Atomic, Vermont Yankee, Maine Yankee, Vermont Yankee and Pilgrim that have already been shutdown. This generation is being replaced with CO₂ and greenhouse gas emitting natural gas plants. As an example, in the 1980s nuclear energy generated about 40% of the electric energy mix. Today 40% is natural gas fired with nuclear's contribution reduced to less than 20%. The Conservation Law Foundation does not support new nuclear construction believing that solar and wind can do the job. They are also actively opposing new natural gas plants and associated pipelines in New England believing in the totally renewable electric grid. They have not actively supported keeping existing nuclear plants operating despite their concern about the loss of this clean energy source.

The pillars of the Conservation Law Foundation are based on three principles which include pushing policies that boost low-cost locally made clean energy; advocating against natural gas expansion in New England; and shutting down the rest of the coal plants. They do call innovative ideas to lessen carbon emissions. They do not identify what those "low-cost locally made clean energy sources are." The policies supported by CLF include increasing the amount of electricity that must be purchased by utilities in renewable portfolio standards, increasing transmission line capacity to handle the distributed solar and wind installations and lifting caps on net metering. They also do not address the major issue of how they will supply power at night or during periods of no wind. Many of these suggestions are disruptive to electric grid operation and raise the cost of electricity. Noticeably absent is any support for existing or new nuclear plants (Conservation Law Foundation, 2020) (<https://www.clf.org/our-focus/climate-change/>).

UN Intergovernmental Panel on Climate Change (https://www.ipcc.ch/site/assets/uploads/sites/2/2019/05/SR15_SPM_version_report_LR.pdf, <https://www.ipcc.ch/sr15/chapter/spm/>)

The October 2018 report of the Intergovernmental Panel on Climate Change identified pathways to reduce climate change temperature increase to 1.5C. The IPCC report outlined several scenarios in which this is possible. All of the scenarios include a major increase in the use of nuclear energy. Of the four major scenarios reviewed shown on Fig. 1 below P1 represents the 1.5° pathway with the least economic and energy growth; P4 the most. According to their estimates which was summarized in a results oriented chart by the Breakthrough Institute, nuclear's share of primary energy doubles by 2030 and quintuples by 2050 in the scenarios with the highest economic growth and energy demand to meet the 1.5C increase limit (International Panel on Climate Change, 2019; International Panel on Climate Change, 2018).

The IPCC report compared nuclear with other low carbon technologies such as wind, solar, biomass and carbon capture and sequestration each of which have significant challenges as reported by the IPCC. They include huge land use commitments, supply of raw materials needed for such distributed low energy density sources compared to nuclear and carbon capture and sequestration. They point out that nuclear energy faces major obstacles in regulation and social acceptance to achieve this large but needed expansion.

International Energy Agency (<https://www.iea.org/reports/nuclear-power-in-a-clean-energy-system>)

The International Energy Agency (IEA) is an intergovernmental organization which is part of the Organization for Economic Cooperation and Development. The IEA was formed to assist nations in dealing with oil and energy issues. It is largely a data collection agency that publishes periodic reports on energy issues and serves as a policy advisor to its member states. They have adopted a three

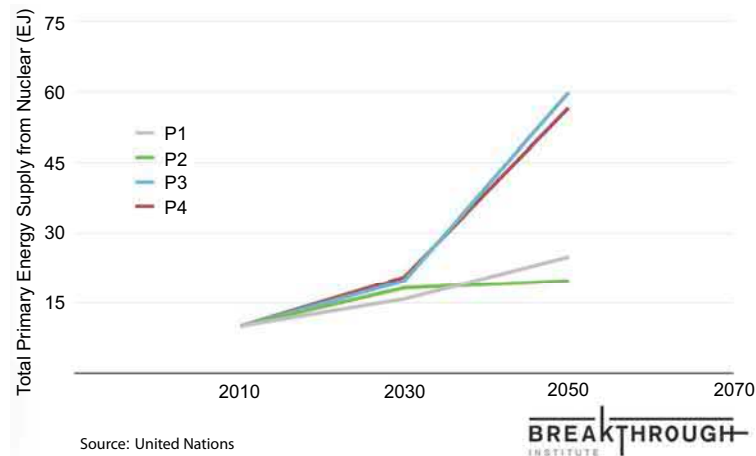


Fig. 1 IPCC estimates of nuclear in 2030 and 2050.

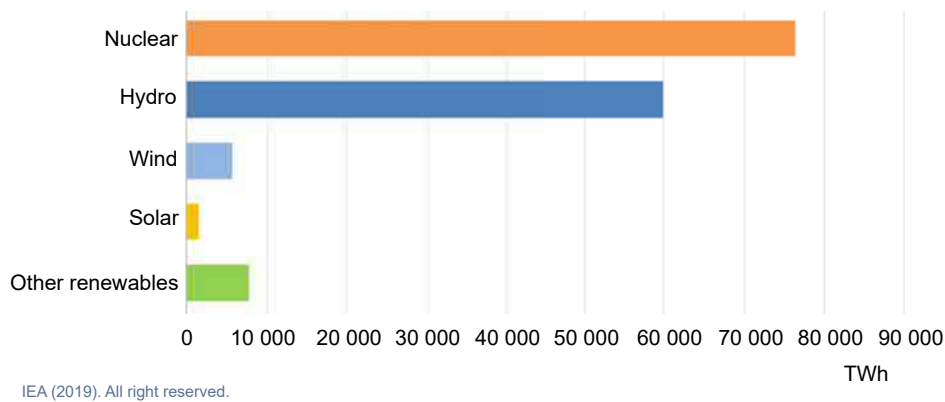


Fig. 2 Low-carbon electricity generation in advanced economies by source, 1971–2018.

prong area of energy policy for focus: Energy Security, Economic Development, and Environmental Protection. They have a broad role in promoting alternate energy sources, rational energy policies and energy technology cooperation among nations.

The International Energy Agency's recently released report "Nuclear Energy in a Clean Energy System" (International Atomic Energy Agency, 2019) summarizes why nuclear energy must play a role in significant decarbonization. Fig. 2 points out the existing role nuclear energy plays in the clean energy sector from 1971 to 2018. This figure shows that in advanced economies nuclear energy is by far the most abundant source of clean power.

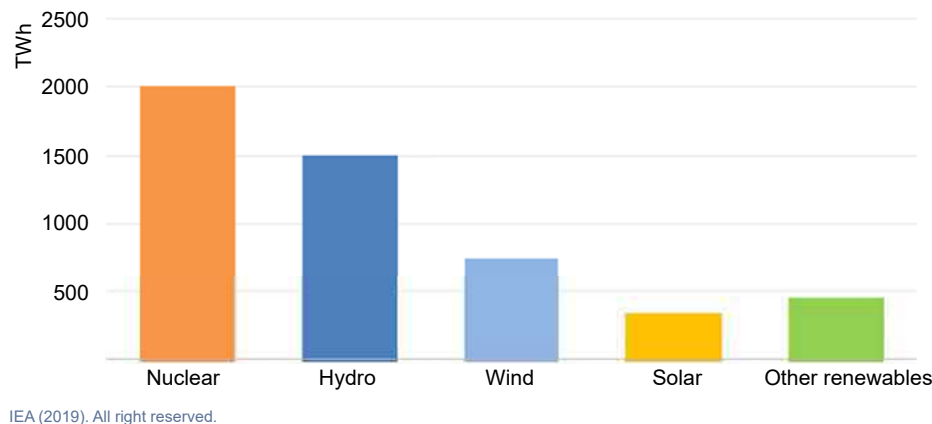
This trend continues in 2018 as shown in Fig. 3 below. Most people may not be aware of this enormous contribution to clean electricity supply.

Shown on Fig. 4 are the country by country savings in carbon dioxide emission avoided. Globally emissions would have been 20% higher without nuclear energy. In absolute terms over 60 Gigatonnes (60,000,000,000 tonnes) would have been emitted in 2018 without nuclear energy. As these figures adequately demonstrate nuclear energy is a proven clean energy source.

This report highlights what is necessary for nuclear to continue to play a productive role in the transition to renewable energy sources. One of the concerns expressed is that with the premature shutdown of existing nuclear plants worldwide, the transition to clean energy has slowed. Additionally, they point out, consistent with IPCC recommendations that to achieve 85% of global electricity coming from clean energy sources such as wind, nuclear and solar it will require an expansion three times the rate currently being deployed by 2040. This trajectory would need an 80% increase in global nuclear power production by 2040. To keep from further retardation of progress, they recommend extending the licenses of operating nuclear plants from 40 to 60 and perhaps even 80 years (as is being done in the US).

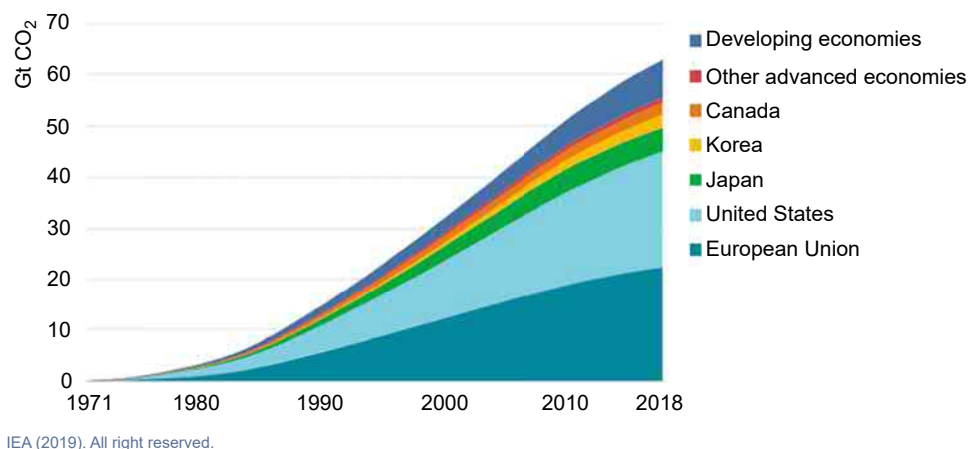
As the share of variable renewables increases, nuclear energy will be even more needed to provide grid stability increasing energy security and improving reliability. IEA also points out that often times, markets and regulatory systems do not value nuclear energy's contribution to clean energy and electricity security causing early closures of sound operating plants.

They conclude that without nuclear energy plants, achieving a sustainable energy system will be much harder to achieve and more costly. They call for intrusive policy making to enable existing nuclear plants to continue to operate and the needed new plants



Nuclear power is the leading low-carbon source of electricity in advanced economies today.

Fig. 3 Low-Carbon electricity generation in advanced economies by source, 2018.



Without nuclear power, global CO₂ emissions from electricity generation would have been almost 20% higher over the last half-century.

Fig. 4 Cumulative CO₂ emissions avoided by global nuclear power to date.

to be built. Their analysis suggests that at least \$ 1.6 Trillion of new investment would be required in the electricity sector from 2018 to 2040. The costs will be even higher if existing nuclear plants' lives were not extended.

MIT Report on decarbonization

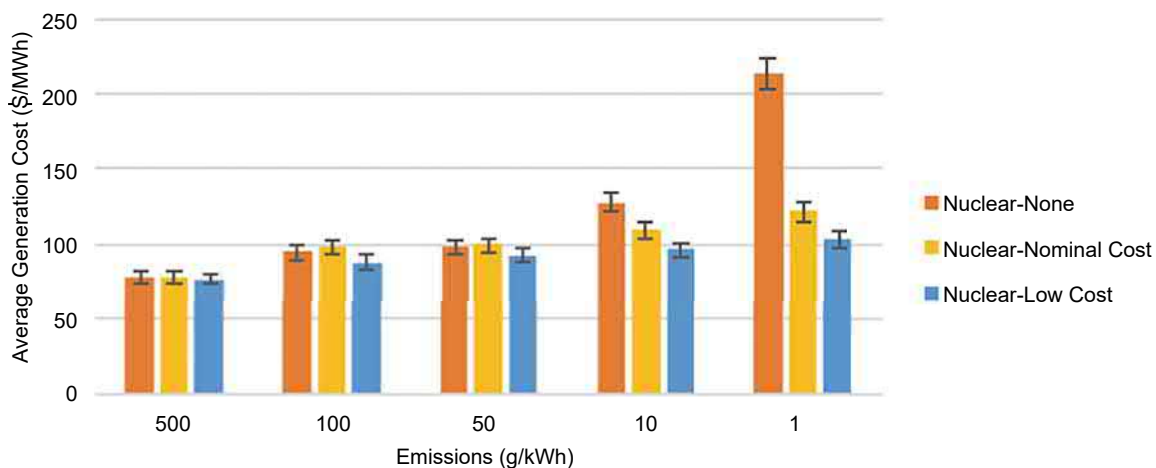
In 2018, MIT's Energy Initiative project published "The Future of Nuclear Energy in a Carbon Constrained World" (MIT Energy Initiative, 2018). This report examined many electricity generating options with the goal of maximizing the elimination of carbon and minimizing costs. Their exhaustive report reviewed alternatives including nuclear, solar, wind, natural gas, coal, hydro, battery storage. The options considered are listed below (Table 1).

MIT employed a GenX model which optimizes the cost of electricity for a given carbon limit target. For a given power market the required inputs include hourly electricity demand, hourly weather patterns, economic costs (capital, operations, and fuel) for all power plants (nuclear, wind and solar with battery storage, fossil with and without carbon capture and storage), and their ramp-up rates. The GenX simulations were used to identify the electrical system generation mix that minimizes average system electricity costs in each of these markets to identified CO₂ targets.

The goal is to find out the optimal cost effective mix that achieves the carbon targets. These targets are quantified as gm-CO₂ emissions per kilowatt-hours (kWh) of power produced for the optimal economic mix. The targets chosen were 500 (the current carbon density), 100, 50, 10, and 1 g-CO₂/kWh. For nuclear energy they modeled five scenarios for each target to see what the economic model would predict—no nuclear, high cost, nominal cost, low cost and extremely low cost. For the low and extremely low cost, the assumption was that they were 25% and 50% respectively lower than the nominal in the region. The high cost was assumed to be 25% higher than nominal.

Table 1 Technology options for each pathway.

<i>Nuclear Energy Is An Allowed Option</i>	<i>Nuclear Energy Is NOT An Allowed Option</i>
<i>Carbon-Free Options</i>	
Photovoltaic (PV) Solar	PV Solar
On-shore Wind	On-shore Wind
Light-water Reactor (LWR) Nuclear	Coal with CCS (IGCT-CCS)
Coal with Carbon Capture and Storage (CCS)	Natural Gas with CCS
Natural Gas with CCS	
<i>Carbon options</i>	
Open Cycle Gas Turbine (OCGT)	OCGT
Combined Cycle Gas Turbine (CCGT)	CCGT
Coal (IGCC)	Coal (IGCC)
<i>Storage Options</i>	
Battery Storage	Battery Storage
Hydro-electric Storage (Fixed)	Hydro-electric Storage (Fixed)

**Fig. 5** New England cost of electricity generation.

They compared several regions. For the US, they chose New England and Texas. For international markets they chose France, the United Kingdom and China (the Tianjin-Beijing-Tangshan (T-B-T) area). The nominal overnight capital cost for nuclear (Rothwell, 2021) in the New England area (US) chosen was \$ 5500/kWe installed in 2050. In China, they are able to build the nuclear plants considerably cheaper and a \$ 2800/kW was assumed. To test the sensitivity of their model to fluctuating prices for alternative energy sources similar variability studies were included which are thoroughly documented in their report.

The results of their optimization model are revealing. Clearly the cost of nuclear energy reflects heavily on the contribution to various emission targets. Figs. 5 and 6 shown below are the results for New England and China.

As can be seen, without nuclear energy, the cost of achieving clean energy targets is much higher for deep decarbonization. For France similar results are shown on Figs. 7 and 8.

It should be pointed out that according to climate change stabilization scenarios developed by the International Energy Agency in 2017, the power-sector carbon intensity targets to limit global average warming to 2 °C range from 10 to 25 g/MWh of CO₂ by 2050 and less than 2 g/MWh by 2060.

The conclusion of the MIT study is that without nuclear energy deep decarbonization, the cost will be higher and targets for limiting CO₂ emissions will be difficult to meet. This finding supports the 2014 UN Intergovernmental Panel on Climate Change which concluded that deep cuts in greenhouse gas emission “will require more intensive use of ... technologies such as renewables [and] nuclear energy.” Additionally in 2015, the International Energy Agency called nuclear power “a critical element in limiting greenhouse gas emissions.” At that time it said that “global nuclear-generation capacity must more than double by 2050 (to about 750 GW) for there to be any hope of limiting temperature increases to the 2-degree scenario that is widely agreed as the acceptable limit.”

Unfortunately, with the exception of China, no other nation is pursuing nuclear energy to the extent needed. To achieve these aggressive targets the MIT report made several recommendations:

1. The need to reduce nuclear energy's cost should be done by improved management, advanced construction and manufacturing including standardization and modularization.

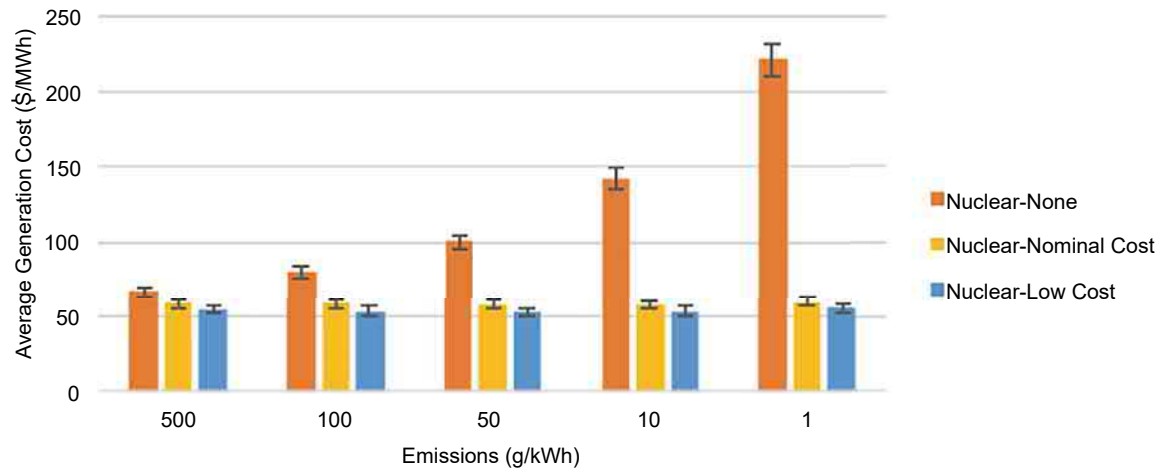


Fig. 6 T-B-T cost of electricity generation.

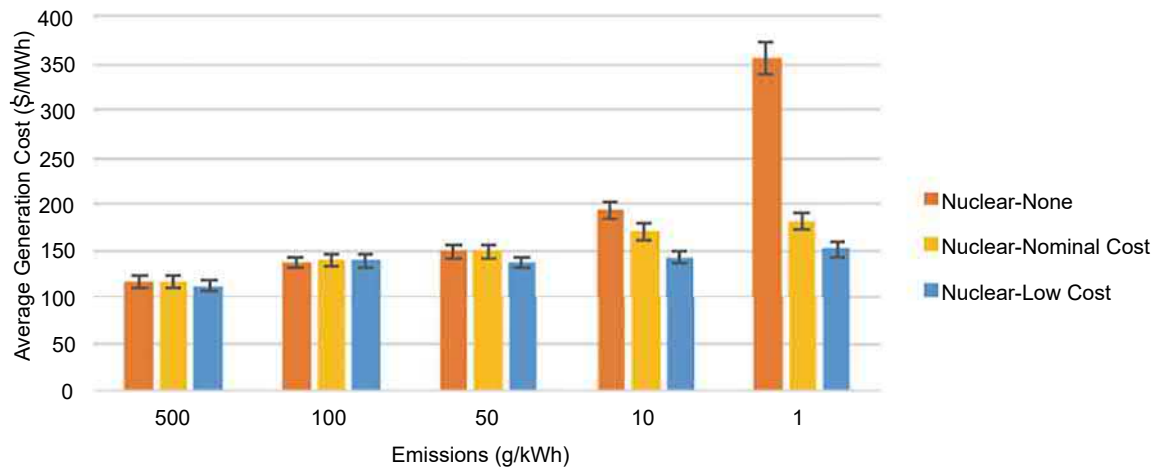


Fig. 7 United Kingdom cost of electricity generation.

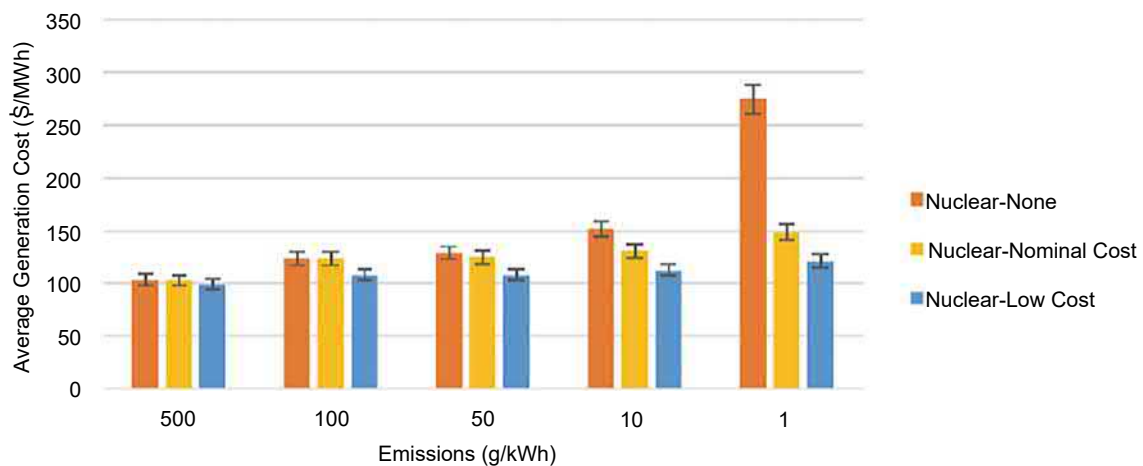


Fig. 8 France cost of electricity generation.

2. Research, design and development to introduce innovative new reactor concepts aimed at reducing capital and operating costs.
3. A more innovative approach to deployment is needed for these advanced concepts to allow for faster demonstration and deployment.
4. There should be no public policies that discriminate against nuclear energy but rather provide a level playing field for all clean energy sources.
5. The government should provide funding to test prototype nuclear plants.
6. The Nuclear Regulatory Commission should move more rapidly to employ risk informed performance based regulatory regime and facilitate the licensing of prototype reactors and funding for their review.

These are just a few of the recommendations found in the MIT report. Their report is rich with findings and other numerous recommendations to enable an important and vital role for nuclear energy in the quest for a real climate change solution and deep decarbonization (<https://energy.mit.edu/wp-content/uploads/2018/09/The-Future-of-Nuclear-Energy-in-a-Carbon-Constrained-World.pdf>).

The environmental case for nuclear energy

As can be seen in the previous sections, the role of nuclear energy in dealing with climate change is very important not only from a reduction in CO₂ but in terms of the affordability of the transition from a fossil based economy to a clean air renewable energy economy. A mix of electricity sources will be required. Nuclear's role will likely be a base load source of power with some ability to follow the load during the day and night. Studies are now being conducted to use nuclear energy when too much solar or wind are put into the grid. These applications include producing water, alternative energy storage systems such as liquid metals, process heat applications for industry and heating or making hydrogen to replace fossil fuels for transportation. These innovations can make nuclear energy even more economical since they can then be run essentially base loaded reducing the cost per kilowatt-hour and these other uses provide income to offset nuclear's relatively higher capital cost.

As an example of how an integrated energy system using nuclear generated heat can be applied with renewable energy systems is shown on Fig. 9 (Hezlar, 2021). The nuclear plant shown on the left can be combined with industrial applications shown on the right through decoupled energy storage systems (blue and brown storage tanks). This configuration when also coupled with supercritical CO₂ power conversion system enables rapid load following as well. High temperature molten salts can be used as the heat storage and transport system.

What is key to understand is that nuclear energy is one of the cleanest energy sources comparable to that of wind and solar. Nuclear energy is a highly concentrated energy source requiring minimal land or water use compared to wind or solar. It can be integrated with solar and wind if properly managed. Nuclear energy's safety record is very good. The accidents that have occurred were due to bad design (Soviet Union), a poor siting decision (Japan) and poor operator response and design (United States). The 440 reactors operating today in 31 countries are doing so safely and have performed well for over 50 years.

The issue of nuclear waste disposal needs a political consensus. Technically all nations agree that geological disposal is the safest alternative. In the United States, Yucca Mountain had completed its technical review finding that the facility could be safely build, operated and then sealed for hundreds of thousands of years but the US government chose to stop the project based on political, not technical reasons. Other countries such as Finland and Sweden are making great progress in actually building a repository. France, United Kingdom, Germany, China, and Japan are investigating suitable geological sites. Some nations are pursuing recycling of used nuclear fuel. France and the United Kingdom are reprocessing the used fuel to reduce the volume of waste needing disposal (which is very small in any case). France is making new fuel from the recycled product and making additional electricity. Advanced reactor designs using waste from enrichment plants can be used to make more fuel in sodium cooled fast breeder reactors essentially extending our uranium supplies for thousands of years without additional mining.

The opportunities nuclear energy offers to the world are enormous. Nuclear energy can be safely operated, advanced reactors with inherently safe design features, small modular reactors allowing for distributed electricity sources, clean water, process heat



Fig. 9 An example of an integrated energy system with nuclear power.

applications as well as a clean way to make hydrogen for the next generation for transportation fuels are but a few examples. Should we move to electric vehicles, more electricity will be required for the world in its continuing quest for a clean environment.

While solar and wind are often referred to as “renewable” sources of energy, nuclear power can also be considered as renewable in the sense that the fuel, uranium, currently already mined and in seawater is almost an infinite supply. In the uranium enrichment process, U-238 contained in residual mill tailings when used in fast breeder reactors can provide thousands of years of electric power. Additionally, even though the amount of natural uranium dissolved in seawater is very low (parts per billion), new techniques for extracting uranium are becoming available. This could essentially be an infinite supply should it be needed.

Unfortunately nuclear energy's potential is limited by the social and political aspects of the technology as pointed out by the IPCC. The lack of active support from the environmental community believing that solar and wind can do it all affects the social and political support needed for nuclear energy to continue to contribute. Studies already described earlier disprove that unfortunate assumption that solar and wind can do it all. Many opinion leaders have already lead the way for others to follow. The mistakes of Germany, in their shutting down their nuclear plants, in the hopes of reducing their carbon emissions has not worked (Energiewende program) and the costs to consumers has gone up significantly. The social cost of the phase out has been over \$ 12 billion in terms of increased health costs due to air pollution due to increased coal and gas usage to replace the nuclear capacity. Additionally in 2107, Germany spent \$ 26 billion Euros in transition costs alone—all borne by industry, taxpayers and consumers. In all, Germany has spent over \$ 400 billion on renewables with not much to show for it in terms of CO₂ reductions (Jarvis et al., 2020) (<https://haas.berkeley.edu/wp-content/uploads/WP304.pdf>).

A example of where a reduction in carbon emission strategies have worked can be found in France and Sweden. While this chart is relatively old it does show the effect of a strong nuclear energy program on the lowering of CO₂ emissions compared to Denmark which is an anti-nuclear country and China which at that time was heavily dependent on coal. Since 1990 China has embarked on an aggressive nuclear power expansion program with 46 nuclear plants in operation and 11 under construction.

Fig. 10, taken from Goldstein and Qvist (Goldstein and Qvist, 2019) shows future global emission intensity if the world transitions to clean energy like Germany did in its Energiewende program versus if the world builds nuclear power plants at a rate comparable to that done by France and Sweden during the global oil crisis in the early 1970s. It confirms studies of the IPCC and others that nuclear energy can play a most significant role in decarbonization of the planet.

A summary of the key reasons to support nuclear energy are provided below.

Intermittency of renewables

The technical reality of solar is that it is an intermittent source of electricity. The best solar farms produce electricity about 25% of the time which means that in a truly renewable economy, batteries or alternative energy sources like diesel-generators will be needed for 75% of the power. Unfortunately such battery technologies are not available so back up sources such as nuclear will be needed.

Wind can do a better job, depending on location. Denmark is a great example. There were days in 2018 when wind supplied all of Denmark's power, but as we all know, wind does not blow all the time. In fact for the full year, wind produced only 50% of Denmark's electricity demand. Denmark was lucky to have been able to fill the gap from imported power from other countries, most likely France's nuclear fleet or fossil plants elsewhere on the grid. The bottom line for even Denmark, they will need back up power for over 50% of the time. This is certainly an untenable position if all Europe gets rid of its nuclear or fossil plants.

As an example, Michael Shellenberger uses the example of how an activist believed that solar could replace the Palo Verde nuclear plants in Arizona. These nuclear plants generate 4000 MW of power on a 24/7 basis. It supplies over 1/3 of Arizona's electricity needs. He calculates that for a week of cloudy weather which sunny Arizona gets from time to time, it would take 670 GW of battery capacity. He found that the total battery storage available in the United States at the end of 2019 is about 3 GW. This

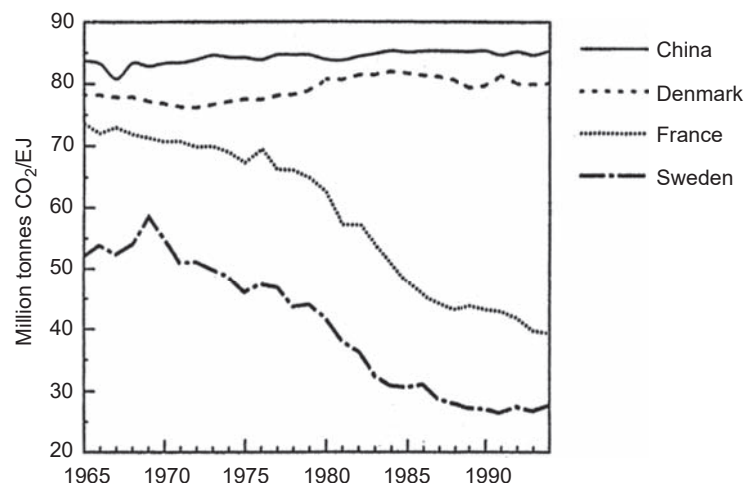


Fig. 10 CO₂ emissions from nuclear and non-nuclear countries.

includes all the batteries found in utilities, homes and cars. What these numbers mean is that to replace this one plant, one would need 200 times the available battery capacity in the US. While technology development is ongoing, the scale needed is enormous. The largest current battery in the world is a 126 Mw Tesla battery being used in Australia which cost \$ 66 million. To replace the Palo Verde plant, one would need 5000 of them having a total cost of \$330,000,000,000! The limitation of solar and wind requiring back up power should not be dismissed since at scale, it cannot be a solution.

Life cycle emissions of energy sources (http://www.world-nuclear.org/uploadedFiles/org/WNA/Publications/Working_Group_Reports/comparison_of_lifecycle.pdf)

The World Nuclear Association compiled research papers and reports from numerous sources reviewing the life cycle emissions from all sources of electric energy production. Reports from 21 sources were compiled and analyzed (World Nuclear Association, 2011). Shown on Fig. 11 are the sources used in this analysis.

Their summary report "Comparison of Lifecycle Greenhouse Gas Emissions of Various Electricity Generation Sources" provides the following results as shown on Table 2.

Graphically the results can be presented as shown on Fig. 12 with uncertainty ranges indicated in the estimates by the research reports.

These results show unquestionably that nuclear energy is a very low source of greenhouse gas emissions comparable to wind and solar and other so called "renewable" electric generating sources. The highly reputable Lancet journal seems to agree according their quote "Nuclear power has one of the lowest levels of greenhouse-gas emissions per unit power production and one of the smallest levels of direct health effects "and they add" it would add a substantial further barrier to the achievement of urgent reductions in greenhouse gases if the current 17 percent of world electricity generation from nuclear power were allowed to decline" (Markandya and Wilkinson, 2007) ([https://www.thelancet.com/article/S0140-6736\(07\)61253-7/fulltext](https://www.thelancet.com/article/S0140-6736(07)61253-7/fulltext)).

Land use

In 2017, a comparative study of land use of electricity production sources was performed by Strata, a non-government research organization that conducts scholarly research and analysis on a variety of public policy issues. Their papers are designed to influence and inform policy makers, citizens and civic leaders. One of their reports focuses on land use of alternative energy systems entitled "The Footprint of Energy: Land use of U.S. Electricity Production" (Stevens et al., 2017).

Their analysis includes direct and indirect land requirements for coal, natural gas, solar, nuclear, hydro and wind generating systems. Included were land required for mining, transport, transmission and waste storage. See Fig. 13. Due to the high energy density of coal, natural gas and nuclear energy, they have the smallest physical footprint of about 12 acres/MW produced. The highest land use commitment was hydro at 315.2 acres/MWe. Solar was found to be 43.5 and wind at 70.6 acres/MWe. According to a study done by Entergy Arkansas to replace the energy produced by their Arkansas One Nuclear station which generates 1800 MWe takes up 1100 acres (1.7 mile²). To replace that amount of power would require 108,000 acres (169 mile²) for wind and 13,200 acres (21 mile²) for solar (Fig. 14).

Thus the land commitment for renewables is huge for renewables which also creates other environmental impacts. For the CO₂ reduction benefit nuclear energy is a clear winner in terms of land use (<https://www.strata.org/pdf/2017/footprints-full.pdf>).

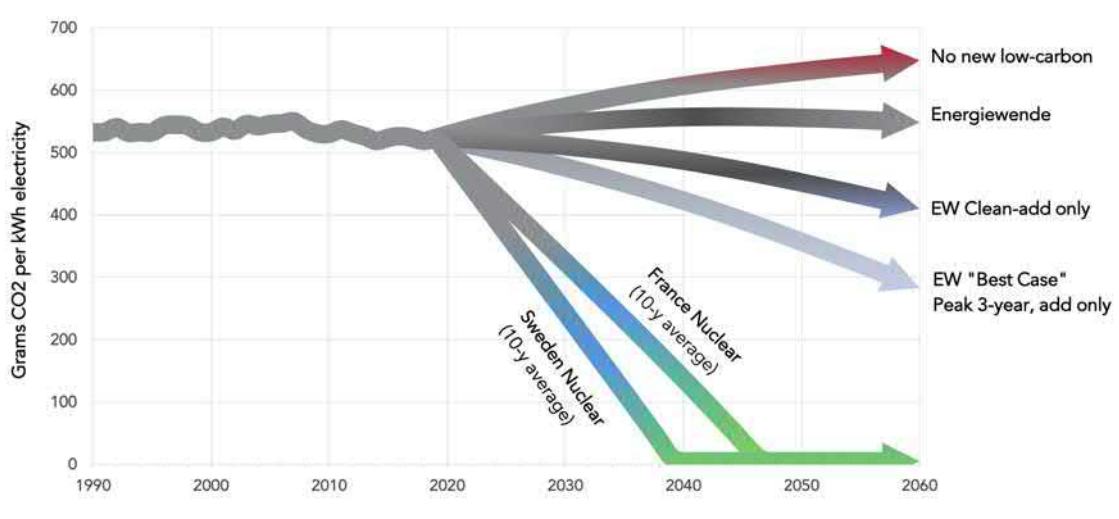
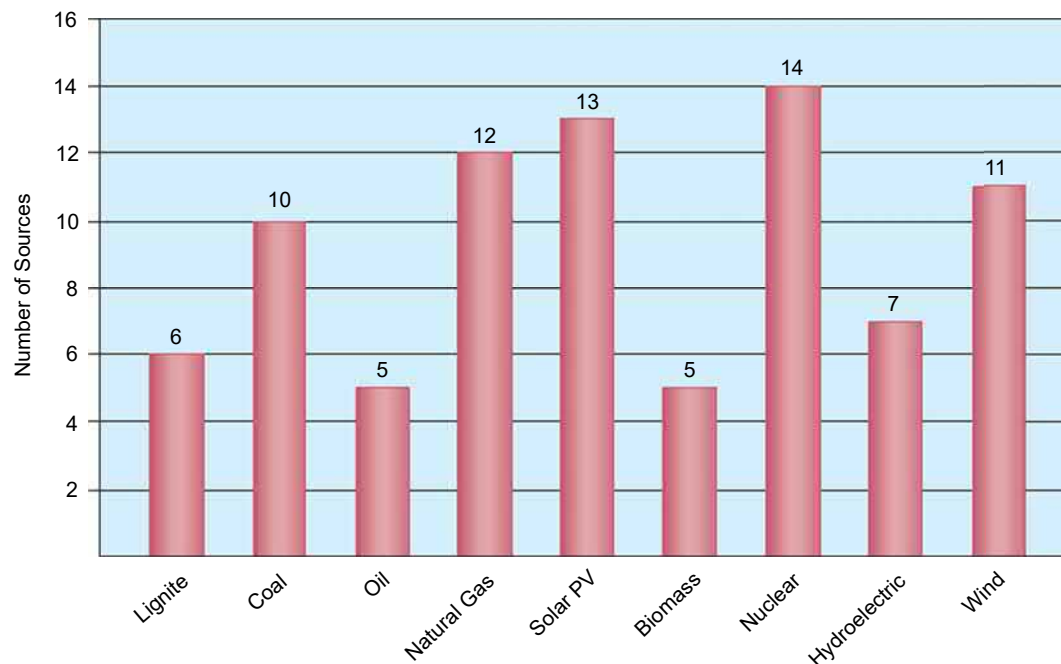


Fig. 11 Comparison of Future World Carbon Intensity under German Energy Strategies.

Table 2 Summary of GHG Lifecycle Emission Intensity From Various Sources.

Technology	Mean	Low	High
	tonnes CO ₂ e/GWh		
Lignite	1054	790	1372
Coal	888	756	1310
Oil	733	547	935
Natural Gas	499	362	891
Solar PV	85	13	731
Biomass	45	10	101
Nuclear	29	2	130
Hydroelectric	26	2	237
Wind	26	6	124

**Fig. 12** Summary of the number of literature sources cited for each generation method.

Cogeneration

To improve the competitiveness of nuclear and to better utilize nuclear energy in an era of increasing solar and wind systems on the electric grid, Dr. Charles Forsberg of the Massachusetts Institute of Technology organized a workshop at MIT to discuss possible options in June 2017 (Forsberg, 2017). The results of the seminar was summarized in a Report entitled "Light Water Reactor Heat Storage for Peak Power and Increased Revenue: Focused Workshop on Near-Term Options." The workshop discussed examples of how nuclear electric plants could be integrated with these renewable sources.

One of the problems has been that during the day, solar plants can contribute significant amounts of power during the day requiring reducing the power of nuclear plants due to grid overload and reducing prices to almost zero and at times negative. Later during the day as the sun sets, the electricity is needed and the price of power increases requiring back up power. At night, these plants may be needed but due to the negative or low prices during the day, they are uneconomic to run since they cannot offset their high capital costs. This mismatch is causing some utilities to prematurely shutdown their plants. To make up the difference, utilities are installing greenhouse gas emitting natural gas turbine plants to provide the needed power which is counter to reducing CO₂ emissions.

In order to maintain a base load capability for this plants, i.e., having the nuclear plant operate 24 h a day at full power, options have been proposed to utilize the steam produced instead of electricity generated. The goal is to bypass the electric generator when prices are low and when prices are high during the day, reconnect the generator to produce electricity. When the steam turbine is bypassed, the steam would be sent to other industrial operations that need the steam, steam storage systems to include steam accumulators (tanks), sensible heat storage, cryogenic air storage, packed pebble beds heat storage, hot rock storage, and geothermal storage.

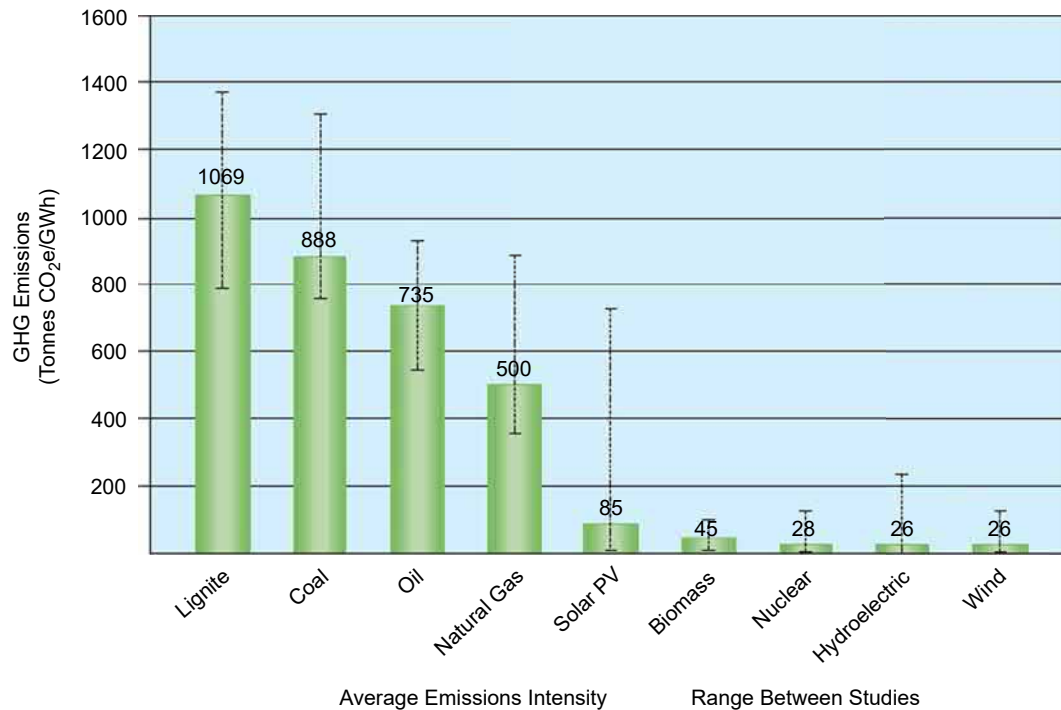


Fig. 13 Life cycle GHG emission intensity of electricity generation methods.

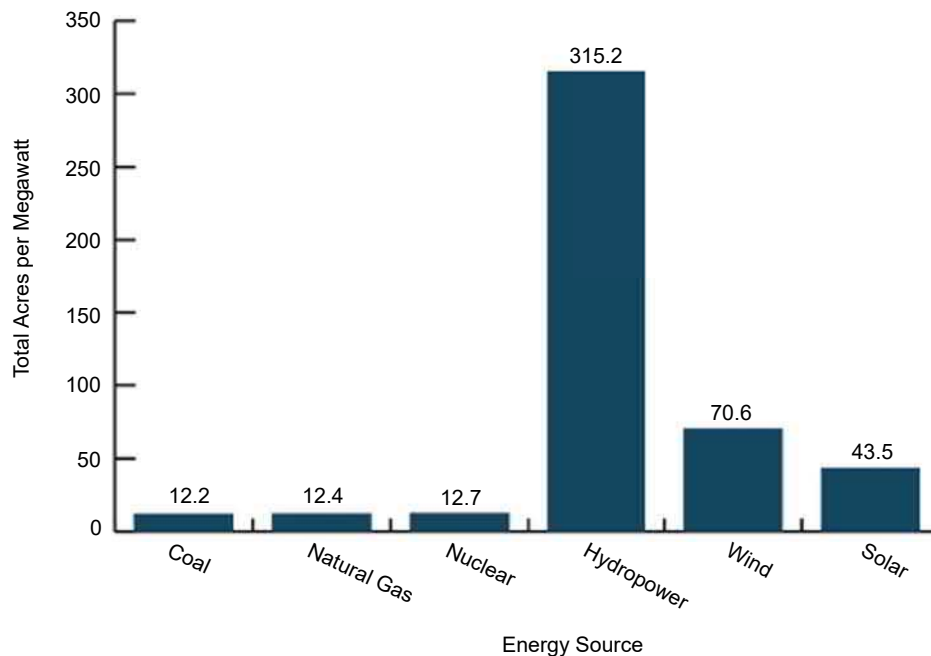


Fig. 14 Comparative land use of energy options.

The value of such systems are dependent on the market economics that would justify whatever investments might be required to offset the energy savings. Some of these technologies have been deployed on a limited basis but others require more development. Regulatory approvals might be required to utilize nuclear heat in these applications but one US utility has shown the selling steam to a local industry would be acceptable. Other countries use low grade heat coming from nuclear power stations to provide district heating (Russia) and Germany has used similar heat to warm soils to allow for longer growing seasons.

Another application that would be welcome is that of making fresh water from salt water in countries where water resources are scarce. This can be done by reverse osmosis (using electricity) or desalinization using the evaporative process. Some have also

suggested that the electricity could be used to make hydrogen for fuel cells or using it to convert CO₂ in the air to make hydro-carbon fuels for transportation. Hydrogen can also be made using electricity in a water separation process.

Thus, there are many compatible applications to use nuclear's excess electricity and unused heat to integrate nuclear energy in a renewable economy (<https://energy.mit.edu/wp-content/uploads/2017/12/Light-Water-Reactor-Heat-Storage-for-Peak-Power-and-Increased-Revenue.pdf>).

Advanced reactors

Today, there are more than 55 advanced reactor concepts under development world wide. These include micro-reactors of several Kilowatts to megawatt sized plants for distributed applications. Many small modular reactors (SMR) that are under development to address the concern about the total cost of large 1000 MW sized reactors; they offer significantly lower financing risk and inherent safety in which basic laws of physics protect against accidents involving emission of radioactive effluents without human intervention even in case of loss of power (Ingersoll, 2021). The SMR market targets distributed electricity avoiding the need to build large transmission system to support the larger plants. Other new nuclear technologies under development include nuclear plants using liquid metal coolants, helium gas, molten salts and heat pipe designs. All have technological features that greatly reduce the risk of accidents and have inherently safe designs that all but eliminate the possibility of a reactor meltdown scenario so feared by many; significantly higher efficiency for converting fission energy to electricity; up to two order of magnitude improvement in uranium utilization and significant reduction in the volume and toxicity of the nuclear waste. A complete discussion of advanced reactor concepts is found in Section "Example of change agents in the environmental community" of this article.

The summary of the case for nuclear energy

- Solar and wind energy are dilute source of energy. To produce the 4.4 Terrwatts of power the US will require in 2050 will mean covering 300,000 km² with solar panels and 1.5 million km² with wind farms for balanced wind production. This is about 20% of the total land area of the US.
- Solar and wind consume a great deal of material for manufacture of the panels and turbines some of which are toxic. To build an all renewable grid by 2050 which is a goal of some, will require about 200 million tonnes of glass, aluminum and steel, all of which are energy intensive.
- Solar will require extensive amounts of cadmium and tellurium both of which are toxic and relatively rare. Present known reserves of tellurium are about 20,000 tonnes. These reserves will have to increase by 1500 times to supply the tellurium needed for a 2050 transition to an all solar and wind economy. This might require deep sea mining with its ecological impacts which will increase costs for solar.
- For massive solar and wind installations, they will likely be centralized in massive solar and wind farms located in relatively rural areas. This will necessitate considerable grid expansion since existing nuclear and fossil plants are not located in such areas. The reliability of such long distance grids is questionable as are the losses in transmission of power over such long distances.
- There is currently no feasible means to provide the back up power needed in during periods of no sun or wind. Natural gas is the solution being deployed in large amounts which contributes to the release of CO₂ and other greenhouse gases and penalizes the economics of the solar/wind electricity generation.
- Nuclear energy is a proven performer providing clean CO₂ free electricity on a massive scale which is scalable to meet world energy needs.
- The current challenge for nuclear energy is competing with subsidized renewable energy and cheap natural gas which is a major CO₂ emitter.
- Advanced nuclear plants will be built with more enhanced safety features than already highly safe operating reactors that supply more than 10% of the world's electricity.
- To meet aggressive climate change goals, nuclear and other clean energy sources will have to massively be expanded in addition to keeping existing nuclear plants operating.
- The most significant public objection to nuclear energy is waste disposal. Scientists all agree that geological disposal is the proper solution but it is being hampered by political opposition. With political leadership such as that found in Finland and Sweden, the solutions are being implemented.

(LLP Fusion, 2020; National Renewal Energy Laboratory, 2013; US Energy Information Agency Administration, 2019; Energy Efficiency and Renewable Energy, 2019) (<https://lppfusion.com/environment/why-solar-and-wind-cant-replace-fossil-fuels-by-themselves>, <https://www.nrel.gov/news/press/2013/2269.html>, https://www.eia.gov/energyexplained/?page=us_energy_home, <http://css.umich.edu/factsheets/us-cities-factsheet>, <https://www.energy.gov/eere/solar/articles/solar-plus-storage-101>).

Will the real environmentalists please stand up

Given the global challenge that these groups have identified and call for "dramatic action," it is time for them to address their dilemma. They need to confront their objections and rationally conclude that nuclear energy is a part of the solution. They need not only to accept the science of climate change but also the science of practical solutions overcoming their historical opposition to nuclear energy. As existing nuclear plants are being prematurely shutdown due to economic, not safety reasons, it is time to act before we lose more clean energy plants that provide a huge source of electricity. These plants are shutdown due largely to cheap natural gas and subsidies given to solar and wind, it is time to stand up and politically support not only continued operation and new plant construction with financial support similar to the given to other "clean energy" technologies.

It is time for the leadership of these organizations to embrace nuclear energy as part of the solution to climate change in an integrated electrical system that harnesses the value the nuclear energy offers in reducing CO₂ emissions. This will take bold action by the leaders of environmental organizations and a call to action by helping educate their members about the need and value of nuclear energy. This task will be difficult but important if we are to deal realistically and positively with the global challenge if they really believe that it is an existential threat.

As the climate scientists' letter of 2013 said "continued opposition to nuclear power threatens humanity's ability to avoid dangerous climate change."

Appendix A

MIT Climate Scientists Letter

Shared publicly—Nov 3, 2013

Kerry Emanuel originally shared:

To those influencing environmental policy but opposed to nuclear power:

As climate and energy scientists concerned with global climate change, we are writing to urge you to advocate the development and deployment of safer nuclear energy systems. We appreciate your organization's concern about global warming, and your advocacy of renewable energy. But continued opposition to nuclear power threatens humanity's ability to avoid dangerous climate change.

We call on your organization to support the development and deployment of safer nuclear power systems as a practical means of addressing the climate change problem. Global demand for energy is growing rapidly and must continue to grow to provide the needs of developing economies. At the same time, the need to sharply reduce greenhouse gas emissions is becoming ever clearer. We can only increase energy supply while simultaneously reducing greenhouse gas emissions if new power plants turn away from using the atmosphere as a waste dump.

Renewables like wind and solar and biomass will certainly play roles in a future energy economy, but those energy sources cannot scale up fast enough to deliver cheap and reliable power at the scale the global economy requires. While it may be theoretically possible to stabilize the climate without nuclear power, in the real world there is no credible path to climate stabilization that does not include a substantial role for nuclear power.

We understand that today's nuclear plants are far from perfect. Fortunately, passive safety systems and other advances can make new plants much safer. And modern nuclear technology can reduce proliferation risks and solve the waste disposal problem by burning current waste and using fuel more efficiently. Innovation and economies of scale can make new power plants even cheaper than existing plants. Regardless of these advantages, nuclear needs to be encouraged based on its societal benefits.

Quantitative analyses show that the risks associated with the expanded use of nuclear energy are orders of magnitude smaller than the risks associated with fossil fuels. No energy system is without downsides. We ask only that energy system decisions be based on facts, and not on emotions and biases that do not apply to 21st century nuclear technology.

While there will be no single technological silver bullet, the time has come for those who take the threat of global warming seriously to embrace the development and deployment of safer nuclear power systems as one among several technologies that will be essential to any credible effort to develop an energy system that does not rely on using the atmosphere as a waste dump.

With the planet warming and carbon dioxide emissions rising faster than ever, we cannot afford to turn away from any technology that has the potential to displace a large fraction of our carbon emissions. Much has changed since the 1970s. The time has come for a fresh approach to nuclear power in the 21st century.

We ask you and your organization to demonstrate its real concern about risks from climate damage by calling for the development and deployment of advanced nuclear energy.

Sincerely,

Dr. Ken Caldeira, Senior Scientist, Department of Global Ecology, Carnegie Institution.

Dr. Kerry Emanuel, Atmospheric Scientist, Massachusetts Institute of Technology.

Dr. James Hansen, Climate Scientist, Columbia University Earth Institute.

Dr. Tom Wigley, Climate Scientist, University of Adelaide and the National Center for Atmospheric Research.

See Also: Introduction to the Social Issues of Nuclear Energy; Relative Nuclear Energy Health Hazards and Popular Acceptance; Securing Nuclear Power's Contribution to Decarbonization.

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Mankind Benefits From Nuclear Energy and Radiation

Ian Hore-Lacy, World Nuclear Association, Blackburn North, VIC, Australia

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Introduction

It is time to mainstream the use of peaceful nuclear technology at the highest level. That means raising public awareness about nuclear technology, incorporating it explicitly into national development plans, and stressing its importance to aid agencies and donors. (Yukiya Amano, IAEA)

Nuclear energy today is derived from the fission of uranium and plutonium in nuclear reactors either to produce heat internally within the reactor or to supply neutrons externally. The internal heat is typically used to make steam which drives turbine generators for electricity, but other applications are possible. The most energy-dense source of heat anywhere is nuclear, and high temperatures are readily achieved. But in contrasting application, the neutrons produced by research reactors are for a variety of applications, most obviously the production of radioisotopes for industry and medicine. Any heat is incidental.¹

Radiation, in the context of this chapter, is ionizing radiation employed for human benefit in a wide range of applications, including industry, agriculture and medicine. Ionizing radiation includes X-rays, which are of atomic rather than nuclear origin.

The International Atomic Energy Agency (IAEA) is the United Nations body overseeing world nuclear energy developments to benefit mankind. Its late Director General, Yukiya Amano, said at the opening of the *IAEA Ministerial Conference on Nuclear Science and Technology: Addressing Current and Emerging Development Challenges* in Vienna in November 2018, “Developing countries are especially interested in how the Agency can help them to grow more food, treat cancer, manage water supplies, protect the oceans and monitor climate change. However, I find that such awareness is often limited to the nuclear community—the scientists, engineers and doctors working in the field. At national level, there is often a lack of awareness of the major contribution nuclear science and technology make to development. As a result, the full potential of peaceful nuclear science and technology is not being realised.”

¹Some radioisotopes are also produced in certain power reactors, incidentally.

Electricity

By far the major application of nuclear energy is for generating electricity on a large scale without any CO₂ or other greenhouse gas emissions. Nuclear power avoids the emission of about 2 billion tonnes of CO₂ per year, relative to coal, and it could do much more.

Over 10% of the world's electricity is from nuclear power, and in France it supplies three quarters of the country's electricity. About 440 power reactors are operational, total almost 400 GWe, and over 18,000 reactor years of experience have been accumulated. In 2018 they supplied 2563 TWh in 30 countries. About 55 more reactors are under construction, equivalent to about 15% of existing capacity, while an additional 110 are planned, equivalent to nearly one-third of existing capacity.

The significance of this role of nuclear energy is that it provides a reliable source of dispatchable power to meet base-load demand, continuous 24/7, with some scope for varying output for load-following. It does so generally at competitive cost and with no carbon dioxide emissions from the plants. It is extremely safe and its wastes are of relatively small volume and easily managed (Greeneche, 2021). The cost of waste management and decommissioning is normally internalized.

In its 60-year history nuclear power has mostly competed with coal and gas-fired generation, both of which are also dispatchable to meet demand, and hence the focus has been on cost, availability of fuel, and wastes—most recently CO₂—for each. This century the comparison has been increasingly with wind and solar power, and the contrast has been scale and reliability, with wind and grid-scale solar photovoltaic (PV) having significant system costs in addition to generation costs to consider. These escalate dramatically with increased share of intermittent renewables in the grid supply.² In addition, the proliferation of wind turbines across landscapes is in marked contrast to the limited space and profile of nuclear power plants.

Electricity has many uses in homes, commerce and industry. In addition there is both immediate and long-term potential for nuclear energy to provide a major contribution to world transport needs through expansion of its present role. Most obviously, there is the current provision of nuclear electricity to run major railway networks and the future expansion of this as electrification is extended. Here it replaces fossil fuels as a source of the electricity, making those networks a genuinely clean form of transport.

Nuclear power is immediately relevant to road transport and motor vehicles by replacing fossil fuels as a source of the electricity used for charging electric vehicles. This would ensure that these vehicles are properly zero-emission transport from the source of the electricity through to the actual on-road mobility. Plug-in hybrid and pure electric vehicles use electricity, and especially off-peak base-load power from the grid, for recharging. Their increased use is changing the pattern of daily demand, and this will become a major factor in electricity supply as car manufacturers increase the electric vehicle proportion of their output.

In addition, nuclear heat can be used for production of liquid hydrocarbon fuels from coal (see **Process heat** below) and also for biofuels. Further out, hydrogen for oil refining and for future fuel cell vehicles may be made electrolytically and, eventually, thermochemically using high-temperature reactors.

Widespread use of plug-in electric vehicles (PHEVs) and full-electric vehicles (EVs)—which get much of their energy from the electricity grid overnight at off-peak rates—will increase electricity demand by up to about 15% in developed countries. But more significantly it will mean that a greater proportion of the electricity can be generated by large-scale plant running continuously, and hence at low cost. This is likely to cause a 30–40% increased requirement for capacity to meet base-load demand, and a corresponding decrease in intermediate- and peak-load plant. This has the potential to lower all electricity costs significantly (large-scale continuous power supply being lower-cost per kWh), and where those base-load plants are nuclear, the power will also be emissions-free.

Industrial heat

Nuclear energy can produce heat for a variety of industrial applications, most obviously desalination to augment surface or artesian potable water supplies. Industrial process heat, particularly at high temperatures using gas-cooled reactors is a further application. Nautical propulsion, especially naval, is well established.

Desalination

World Energy Outlook 2016 (IEA, 2016) reported that in 2015, there were about 19,000 desalination plants worldwide, to provide water to both municipal and industrial users. Total world capacity in 2016 was 88.6 million m³/day (32,300 GL/year) of potable water. Almost half of global installed desalination capacity was in the Middle East, followed by the European Union with 13%, the USA with 9%, and North Africa with 8%. Globally, seawater is the most common feed water type, supplying about 60% of installed capacity, followed by brackish water at over 20%.

Desalination is very energy-intensive. The energy consumption for desalination is considerable: in 2016 the UAE used 556 TJ/year, followed by Saudi Arabia 168 TJ/year, Qatar 118 TJ/year and Kuwait 76 TJ/year, virtually all from fossil fuels.

²System costs are those costs between generation costs and providing a steady reliable supply to meet demand. They escalate dramatically as the proportion of intermittent wind and solar input to a system increases. *The Costs of Decarbonisation: System costs with high shares of nuclear and renewables* (NEA, 2019).

Reverse osmosis (RO) needs up to 6 kWh of electricity per cubic meter of seawater, hence 1 MWe will produce about 160 m³/h. For brackish water and reclamation of municipal wastewater, RO requires only about 1 kWh/m³. The choice of process generally depends on the relative economic values of fresh water and particular fuels, and whether cogeneration is a possibility. A variety of low-temperature heat sources may be used for thermal processes, including solar energy.

Small and medium sized nuclear reactors are suitable for desalination, often with cogeneration of electricity using low-pressure steam from the turbine and hot seawater feed from the final cooling system. The main opportunities for nuclear plants have been identified as the 80–100,000 m³/day and 200–500,000 m³/day ranges. US Navy nuclear powered aircraft carriers reportedly desalinate 1500 m³/day each for use onboard. Desalination can provide a way to vary substantially the amount of electricity supplied to the grid by a plant operating continuously at full power, in response to varying grid demand. Water is easily stored on any scale.

The feasibility of integrated nuclear desalination plants has been proven with over 150 reactor-years of experience, chiefly in Kazakhstan, India and Japan. Large-scale deployment of nuclear desalination on a commercial basis will depend primarily on economic factors. Egypt's planned El Dabaa nuclear power plant will use about 12% of its power for desalination, each reactor taking 432 MWt from the secondary circuit to produce 170,000 m³/day at a cost of less than \$1/m³.

Process heat

Apart from desalination, nuclear energy is an excellent source of process heat which can potentially be applied for various industrial applications including synthetic and unconventional oil production, oil refining, biomass-based ethanol production, and in the future: hydrogen production.

Coal can be used to make synthetic oil. The Fischer-Tropsch process was originally developed in Germany in the 1920s, and provided much of the fuel for Germany during the Second World War. It then became the basis for oil production in South Africa by Sasol, which now supplies about 30% of that country's gasoline and diesel fuel. The process consumes a significant quantity of hydrogen, using carbon monoxide in a catalytic reaction. The hydrogen is now produced with the carbon monoxide by coal gasification. The carbon monoxide then undergoes the water shift reaction to produce more hydrogen. A nuclear source of hydrogen coupled with nuclear process heat would triple the amount of liquid hydrocarbons from the coal and eliminate most CO₂ emissions from the process. There is considerable potential to extend world oil supplies, with negligible carbon footprint in their production, by this means.

The potential application of nuclear heat depends mainly on the temperature required. High temperature heat transfer from reactor to process can be by helium, salt or liquid metals. With future reactor output temperatures of up to 700 °C there is a wide range of possible applications, at 900 °C there are further possibilities, and at 950 °C an important application to hydrogen production opens up. About 20% of US energy consumption goes into process heat applications, compared with 35–40% into electricity. In this 20%, replacing fossil fuels with nuclear heat promises much in energy security, price stability and reduced regulatory risks, and it is the only option if carbon dioxide emissions are avoided.

In the European Union, the Nuclear Cogeneration Industrial Initiative (NC2I) (<http://www.snetp.eu/nc2i/>), part of the Sustainable Nuclear Energy Technology Platform (SNETP) is focused on high-temperature gas-cooled reactors (HTRs) producing 550 °C steam potentially for a variety of industrial applications.

The nuclear—hydrogen connection

Hydrogen is already a significant chemical product, about half of annual production being used in making nitrogen fertilizers via the Haber process and about half to convert low-grade crude oils (including those from tar sands) into transport fuels. Both uses of pure hydrogen are increasing significantly. Some is used for other chemical processes. World demand is about 70 million tonnes of pure hydrogen plus 45 million tonnes mixed with other gases and used in industry such as steel and methanol production, both kinds of demand growing steadily (IEA, 2019).

About 76% of pure hydrogen is made from natural gas and 23% from coal. This gives rise to 830 million tonnes of carbon dioxide emissions annually—each tonne produced gives rise to at least 11 tonnes of carbon dioxide. Electrolysis accounts for only 2%.

Hydrogen for transport

Like electricity, hydrogen is an energy carrier (but not a primary energy source). As oil becomes more expensive, hydrogen may replace it as a transport fuel and in other applications. This development becomes more likely as fuel cells are developed, with hydrogen as the preferred fuel. If natural gas also becomes expensive, or constraints are put on carbon dioxide emissions, non-fossil sources of hydrogen will become necessary.

As with electricity, hydrogen for transport use will tend to be produced near where it is to be used. This will have major geopolitical implications as industrialized countries become less dependent on oil and gas from distant parts of the world.

Hydrogen can be produced economically by electrolysis of water in off-peak periods, enabling much greater utilization of base-load plants, and in particular nuclear plants. In future, the heat from high-temperature reactors could be used directly for thermochemical production of hydrogen.

All this points to the fact that while a growing hydrogen economy already exists, linked to the worldwide chemical and refining industry, a much greater one is possible (IEA, 2019). With new uses for hydrogen as a fuel, the primary energy demand for its production may approach that for electricity production. Nuclear power already produces electricity as a major energy carrier. It is well placed to produce hydrogen if this becomes a major energy carrier also.

An electricity system based on nuclear power could produce hydrogen at a steady rate or simply off-peak and store it underground so that it was used in large banks of fuel cells (e.g. 1000 MWe) at peak demand periods each day. Efficiency would be enhanced if by-product oxygen instead of air were used in the fuel cells.

In motor vehicles, hydrogen can be burned in a normal internal combustion engine, and some test cars are thus equipped. However, its main use is likely to be in fuel cells. A fuel cell is conceptually a refuelable battery, making electricity as a direct product of a chemical reaction. But where normal batteries have all the active ingredients built in at the factory, fuel cells are supplied with fuel from an external source. Fuel cells catalyze the oxidation of hydrogen directly to water at relatively low temperatures while producing electricity.

Hydrogen can also be used for stand-alone small-scale stationary generating plants using fuel cells—where higher temperature operation (e.g. of solid oxide fuel cells) and hydrogen storage may be less of a problem, or where hydrogen is reticulated like natural gas.

Considering the storage and portability challenges of hydrogen itself, as well as the radical change to fuel cell cars, attention has turned to methanol (CH_3OH), which can be made from CO_2 and hydrogen, and used in conventional internal combustion engines. It has about half the energy density of petrol/gasoline. For diesel engines, dimethyl ether (DME, $\text{CH}_3\text{—O—CH}_3$) is better than methanol, and DME is made by dehydrating methanol molecules. World production capacity of DME is over 10 million tonnes per year.

Hydrogen for agricultural fertilizers

According to Norman Borlaug, 1970 Nobel laureate and “grandfather of the green revolution,” organic nitrogen compounds present in the world’s soils are only sufficient to feed one-third of today’s population. The rest must come from inorganic additions. Most of the world’s nitrogen fertilizers are made using the Haber³ process, combining abundant atmospheric nitrogen with hydrogen. The resulting ammonia is then oxidized to nitrates. But the hydrogen has to be made from fossil fuels, principally methane, i.e. natural gas. This is costly and it gives rise to substantial carbon dioxide emissions.

If nuclear energy is used to make the hydrogen from water, the carbon dioxide is avoided and a valuable organic chemistry feedstock is conserved. An abundant supply of low-cost hydrogen would greatly boost world agricultural productivity through increased availability of nitrogen fertilizers.

Hydrogen for oil refining

A major use of hydrogen today is hydrogenation of heavy crude oil, a process that breaks down the long-chain hydrocarbons to yield synthetic crude oil (about 5 kg of hydrogen is used per barrel). Steam reforming of natural gas is used to produce the hydrogen feedstock. Nuclear power could make this steam and electricity, and use some of the electricity in high-temperature electrolysis for hydrogen production.

Nuclear propulsion

Marine propulsion

Work on nuclear marine propulsion started in the 1940s, and the first test reactor started up in the United States in 1953. The first nuclear-powered submarine, *USS Nautilus*, put to sea in 1955. This led to the development of reactors and propulsion systems mostly for naval vessels in several countries. Today over 140 vessels are powered by about 180 small nuclear reactors and more than 12,000 reactor-years of marine operation has been accumulated.

Nuclear propulsion has proven technically and economically essential in the Russian Arctic where operating conditions are beyond the capability of conventional icebreakers. The power levels required for breaking ice up to 3 m thick, coupled with refueling difficulties for other types of vessels, are significant factors. Russia’s nuclear fleet, with six nuclear icebreakers and a nuclear freighter, has increased Arctic navigation from 2 to 10 months per year, and in the Western Arctic, to year-round.

The Russian LK-60 icebreakers now being commissioned are “universal” dual-draught (10.5 m with full ballast tanks, minimum 8.55 m), displacing up to 33,540 t (25,450 t without ballast), for use in the Western Arctic year-round and in the eastern Arctic in summer and autumn. They are 173 m long, 34 m wide, and designed to break through 2.8 m thick ice at up to 2 knots. The wide 33 m beam at waterline is to match the 70,000 t ships they are designed to clear a path for, though a few ships with reinforced hulls

³German scientist Fritz Haber invented the process in 1909 and received the Nobel Prize for chemistry in 1918 for creating “an exceedingly important means of improving the standards of agriculture and the well-being of mankind,” which seems a considerable understatement. $\text{N}_2 + 3\text{H}_2 \rightarrow 2\text{NH}_3$ (at 15–25 MPa and 400–500°C). The Haber process produces about 500 million tonnes of nitrogen fertilizer per year and consumes about half of the hydrogen produced annually.

are already using the Northern Sea Route. The LK-60 is powered by two RITM-200 reactors of 175 MWt each which together deliver 60 MW at the three propellers via twin turbine-generators and three electric motors.

In 1988, the *NS Sevmorput* was commissioned in Russia, mainly to serve northern Siberian ports but more recently as a freighter for fresh food from east coast across the north of Siberia to the west. It is a 61,900 t LASH carrier (taking lighters to ports with shallow water) and container ship with ice-breaking bow. It is powered by a similar KLT-40 reactor to that used in the ice-breakers, delivering 32.5 propeller MW from the 135 MWt reactor, and it needs refuelling only every 15 years.

Nuclear power gives submarines and large naval ships unmatched performance. The Russian, US and British navies as well as Russian icebreakers until the latest models rely on steam turbine propulsion, the French and Chinese in submarines use the turbine to generate electricity for propulsion. All power plants are pressurized water reactors (PWR), except the ill-fated Russian *Alfa* class.

With increasing attention being given to greenhouse gas emissions arising from burning fossil fuels for international air and marine transport, particularly dirty bunker fuel for the latter, and the excellent safety record of nuclear-powered ships, it is likely that there will be renewed interest in marine nuclear propulsion. Large bulk carriers that go back and forth constantly on few routes between dedicated ports could be powered by a reactor delivering 100 MW thrust. The world's merchant shipping is reported to have a total power capacity of 410 GWt, about one third that of world nuclear power plants. So far, exaggerated fears about safety have caused political restriction on port access.

Space propulsion

Nuclear fission reactors for space programs have mainly been Russian. But for higher power requirements than RTGs can provide, fission power systems (FPS) have a distinct cost advantage over Radioisotope Thermoelectric Generators (RTGs). As currently conceived, FPS would be launched cold, with essentially no radioactive hazards. Reactor start-up is after the device is in orbit. Then the reactor automatically responds to thermal load changes and maintains safe operating temperatures based on negative temperature reactivity feedback, giving it load-following capability.

Between 1967 and 1988 the former Soviet Union launched 31 low-powered fission reactors in Radar Ocean Reconnaissance Satellites (RORSATs) on Cosmos missions. They utilized thermoelectric converters to produce electricity. Russia is now working on reactors in the megawatt power range, for long-haul interplanetary missions.

In USA, the National Aeronautics and Space Administration (NASA) and the Department of Energy (DOE) are developing fission power technology to serve a wide range of future space uses. More-conventional fission power systems employing liquid-metal-cooled pin-fuel reactors and dynamic power conversion are well suited for lunar and Mars surface power or nuclear electric propulsion (NEP) applications.

Heatpipe Power System (HPS) reactors are compact fast reactors producing up to 100 kWe for about 10 years to power a spacecraft or planetary surface vehicle. They have been developed since 1994 at the Los Alamos National Laboratory as a robust and low technical risk system with an emphasis on high reliability and safety. NASA is currently focused on its KiloPower space reactors. They may include a variety of designs of comparable power and mass to RTGs. They use liquid metal heat pipes to transfer fission heat to either thermoelectric or Stirling power conversion. Beyond that it plans MegaPower reactors producing 2 MWe.

Neutron applications and treatments

Research reactors, much smaller than power reactors designed to produce and exploit heat, are well established around the world to supply neutrons for research, to treat or test materials for specialist applications, and to produce radioisotopes for medicine and industry. Research reactors are essentially neutron factories, and about 220 operate in 53 countries (86 in developing countries). Their neutron beam output can have different characteristic depending on designed use. Test reactors are more powerful than most.

Some research reactors operate with high-enriched uranium fuel and international efforts have substituted low-enriched fuel (below 20% U-235) in most of these. Some radioisotope production also uses high-enriched uranium as target material for neutrons and this is being phased out in favor of low-enriched uranium. Research reactors are simpler than power reactors, and operate at lower temperatures. They need far less fuel, and far less fission products build up as the fuel is used.

Many research reactors are used with international collaboration, and the products of those used for isotope production are certainly traded internationally. The IAEA has designated two international research hubs based on research reactors, giving them ICERR⁴ status valid for 5 years. The first is in France, based on CEA's Saclay and Cadarache facilities, the second is in Russia, the Research Institute of Atomic Reactors (RIAR) at Dimitrovgrad, with six research reactors available to IAEA member states.

Neutron beams are uniquely suited to studying the structure and dynamics of materials at the atomic level. Neutron scattering is used to examine samples under different conditions such as variations in vacuum pressure, high temperature, low temperature and magnetic field, essentially under real-world conditions.

A fuller treatment of the wide range of radiation applications is in the chapter by Bezdek (2021) in this encyclopedia. He makes it clear that modern society cannot function without nuclear technologies and that the benefits of these technologies are enormous and widespread.

⁴International Centre based on Research Reactors <https://www.iaea.org/about/partnerships/international-centres-based-on-research-reactors-icerrs>.

Neutron activation

This process is used to produce the radioisotopes, widely used in industry and medicine. It involves bombarding particular elements with neutrons which are captured by atomic nuclei so that each target nucleus has a neutron added and becomes radioactive, or possibly fissions.

For instance, cobalt-59 can capture a neutron to become cobalt-60, a widely-used source of gamma radiation as it decays to nickel-60. But if lithium-7 captures a neutron it fissions into helium and tritium. The most widely used isotope in nuclear medicine is technetium-99 m, a decay product of molybdenum-99. This is usually produced by irradiating a target of uranium foil with neutrons (for a week or so) and then separating the molybdenum-99 from the other fission products of U-235 in a hot cell—the Mo-99 being about 6% of the fission products.

Neutron activation analysis is a sensitive and accurate means of trace element analysis, where emissions from the radioactive product can be measured, regardless of its chemical form.

Neutron transmutation doping

NTD changes the properties of silicon, making it highly conductive of electricity. Large, single crystals of silicon shaped into ingots, are irradiated inside a reactor reflector vessel. Here the neutrons change one atom of silicon in every billion to phosphorus. The irradiated silicon is sliced into chips and used for a wide variety of advanced computer applications. NTD increases the efficiency of the silicon in conducting electricity, an essential characteristic for the electronics industry.

Materials testing

In materials testing reactors, materials are also subject to intense neutron irradiation to study changes. For instance, some steels become brittle under such conditions, and alloys which resist embrittlement must be used in nuclear reactors.

Radioisotopes

The attributes of naturally decaying atoms, known as radioisotopes, give rise to their multiple applications across many aspects of life. There are about 50 million nuclear medicine procedures each year, saving many lives.

There are hundreds of natural radioisotopes representing stages in the decay of those which were formed primordially. But most of those used in the applications described below are made in research reactors, and are relatively short-lived. Their decay sequence is according to their physical characteristics, just the same as for those occurring naturally.

Nuclear medicine

The use of radiation and radioisotopes in medicine, particularly for diagnosis (identification) and therapy (treatment) of various medical conditions is widespread and well established. In developed countries (a quarter of the world population) about one person in 50 uses diagnostic nuclear medicine each year, and the frequency of therapy with radioisotopes is about one-tenth of this.

Nuclear medicine uses radiation to provide information about the functioning of a person's specific organs or to treat disease. In most cases, the information is used by physicians to make a quick, accurate diagnosis of the patient's illness. The thyroid, bones, heart, liver and many other organs can be easily imaged, and disorders in their function revealed. In some cases radiation can be used to treat diseased organs, or tumors. Five Nobel Laureates have been intimately involved with the use of radioactive tracers in medicine.

Over 10,000 hospitals worldwide use radioisotopes in medicine, and about 90% of the procedures are for diagnosis. The most common radioisotope used in diagnosis is technetium-99 (Tc-99), with some 40 million procedures per year, accounting for about 80% of all nuclear medicine procedures worldwide. In the United States there are over 20 million nuclear medicine procedures per year among 310 million people, and in Europe about 10 million among 500 million people.

Nuclear medicine was developed in the 1950s by physicians with an endocrine emphasis, initially using iodine-131 to diagnose and then treat thyroid disease. In recent years specialists have also come from radiology, as dual computerized tomography with positron emission tomography (CT/PET) procedures have become established, increasing the role of accelerators in radioisotope production. However, the main radioisotopes such as technetium cannot effectively be produced in useful quantities and purity without reactors.

Sterilization is an important medical-related use of radioisotopes having strong gamma emission, such as cobalt-60, or X-rays or electron beams. Many medical supplies are sterilized without heat, and after being packaged. More than 160 gamma irradiation plants worldwide are used to sterilize medical equipment and supplies. The coronavirus pandemic has highlighted the importance of sterile personal protective equipment, though face masks are an exception due to damage from the high levels of radiation involved.

Radioisotopes in industry

Radioisotopes are used by manufacturers as tracers to monitor fluid flow and filtration, detect leaks, and gauge engine wear and corrosion of process equipment. Small concentrations of short-lived isotopes can be detected while no residues remain in the environment. By adding small amounts of radioactive substances to materials used in various processes it is possible to study the mixing and flow rates of a wide range of materials, including liquids, powders and gases and to locate leaks. Radiotracers are used widely in industry to investigate processes and highlight the causes of inefficiency. They are particularly useful where process optimization can bring material benefits, such as in the transport of sediments. Radiotracers are also used in the oil and gas industry to help determine the extent of oil fields.

Radioactive materials are used to inspect metal parts and the integrity of welds across a range of industries. Industrial gamma radiography utilizes the ability of various types of radiation to penetrate materials to different extents. Gamma radiography works in much the same way as X-rays screen luggage at airports. Instead of the bulky machine needed to produce X-rays, all that is needed to produce effective gamma rays is a small pellet of radioactive material in a sealed titanium capsule. The capsule is placed on one side of the object being screened, and some photographic film is placed on the other side. The gamma rays, like X-rays, pass through the object and create an image on the film. Just as X-rays show a break in a bone, gamma rays show flaws in metal castings or welded joints. The technique allows critical components to be inspected for internal defects without damage.

The process of gamma radiography, a type of non-destructive testing, is used to validate the integrity of poured concrete and welds on fluid vessels, pipelines or critical structural elements. The unique characteristics of gamma radiography have resulted in the technique becoming a crucial tool throughout multiple industries.

Gauges containing radioactive (usually gamma) sources are in wide use in all industries where levels of gases, liquids and solids must be checked. The IAEA estimates that several hundred thousand such gauges are operating in industry worldwide. They measure the amount of radiation from a source which has been absorbed in materials. These gauges are most useful where heat, pressure or corrosive substances, such as molten glass or molten metal, make it impossible or difficult to use direct contact gauges.

Radioisotopes in environmental management and water resources

Radioactive tracers (or radiotracers) play an important role in detecting and analyzing pollutants since even very small amounts of a given radioisotope can easily be detected, and the decay of short-lived isotopes means that no radioactive residues remain in the environment. Hydrologists use radiotracers to determine the passage and pace of pollutants moving through groundwater to assess the level of vulnerability.

Radioisotope hydrology techniques enable accurate tracing and measurement of the extent of underground water resources. Such techniques provide important analytical tools in the management and conservation of existing supplies of water and in the identification of new, renewable sources. They provide answers to questions about origin, age and distribution of groundwater, as well as the interconnections between ground and surface water and aquifer recharge systems. The results permit planning and sustainable management of these water resources. For surface waters they can give information about leakages through dams and irrigation channels, the dynamics of lakes and reservoirs, flow rates, river discharges and sedimentation rates. There are some 60 countries, developed and developing, that have used isotope techniques to investigate their water resources in collaboration with IAEA. The isotopes measured may be naturally-occurring or added.

An allied technology, neutron probes, can measure soil moisture very accurately, enabling better management of land affected by salinity, particularly in respect to irrigation. A neutron probe suspended on a cable is lowered down a hole. A typical one contains a pellet of americium-241 and beryllium. The alpha particles emitted by the decay of the americium produce neutrons when they collide with the light beryllium nuclei. The probe also contains a measuring device which detects what happens to the neutrons after they encounter hydrogen nuclei and gives an indication of soil moisture.

Radioisotopes in the home

The main application of radioisotopes in the home is in smoke detectors. A common type is ionization smoke detector which contains a small amount of americium-241 as an oxide. Americium-241 emits alpha particles and low energy gamma rays. The alpha particles emitted by the Am-241 collide with the oxygen and nitrogen in air within the detector's ionization chamber to produce charged particles (ions). A low-level electric voltage applied across the chamber is used to collect these ions, causing a steady, small electric current to flow between two electrodes. When smoke enters the space between the electrodes, the smoke particles attach to the charged ions, neutralizing them. This causes the number of ions present—and therefore the electric current—to fall, which sets off an alarm.

Tritium and promethium-147 release beta particles during decay, triggering a chemical reaction upon contact with other materials, which creates a "glow" as perceived by the human eye. These properties—known as radioluminescence—make the isotopes useful for providing "light" in the absence of electricity. These materials are used in clocks, watches and gun sights so that they can be used at night, as well as in "exit" signs in commercial buildings and aircraft/ships. Earlier watches and the dials of other instruments utilized radium for the same purpose.

Radioisotopes in space

Radioisotope power sources have been an important source of energy in space since 1961. Radioisotope Thermoelectric Generators (RTGs) have been the main electricity source for space work.

The high decay heat of plutonium-238 (0.56 W/g) enables its use as an electricity source in the RTGs of spacecraft, satellites and navigation beacons. Its intense alpha decay process with negligible gamma radiation calls for minimal shielding. Americium-241, with 0.15 W/g, is another source of energy, favored by the European Space Agency, though it has some relatively low-energy gamma radiation.

In an RTG, heat from the oxide fuel is converted to electricity through static thermoelectric elements (solid-state thermocouples), with no moving parts. RTGs are safe, reliable and maintenance-free and can provide heat or electricity for decades under very harsh conditions, particularly where solar power is not feasible. So far over 45 RTGs have powered over 25 US space vehicles including Apollo, Pioneer, Viking, Voyager, Galileo, Cassini, Ulysses and New Horizons space missions as well as many civil and military satellites. RTGs range up to nearly 1 kW.

As well as RTGs, Radioactive Heater Units (RHUs) are used on satellites and spacecraft to keep instruments warm enough to function efficiently. Their output is only about 1 W and they mostly use Pu-238.

Summary

The enormous range of uses for nuclear energy itself, the neutrons it produces, and their derivative products in hundreds of different radioisotopes means that the suite of nuclear technologies is a vital, if inconspicuous, part of today's world. Modern science, medicine and industry could not function without them. Bezdek (2021) estimates that international markets for these technologies exceed US\$500 billion annually and are growing rapidly.

See Also: Global Market for Radiation Applications; Relative Nuclear Energy Health Hazards and Popular Acceptance; Securing Nuclear Power's Contribution to Decarbonization.

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Origins and Resources of Uranium and Thorium

J. Stephen Herring, Center for Space Nuclear Research, Universities Space Research Association, LLC, Idaho Falls, ID, United States

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Glossary

Cross-section Probability of neutron interaction with a nucleus, expressed in terms of area, in units of barns (b). One barn equals $1.0 \times 10^{-24} \text{ cm}^2$.

Enrichment Tails (also Depleted Uranium) The uranium remaining after the enrichment of natural uranium into fuel, today about 0.3% ^{235}U , earlier 0.2–0.25% ^{235}U .

Enrichment The fraction of an isotope, usually fissile ^{235}U , in a mass of uranium. Enrichment is commonly quoted as the weight percent of the particular isotope. Natural uranium has an enrichment of 0.711 wt%, commercial reactor fuel is 3–5% enriched and depleted uranium is 0.2–0.3 wt% ^{235}U .

Fractionation Crystallization from a magma in which the initial crystals are prevented from equilibrating from the parent liquid, resulting in a series of residual liquids of more extreme composition than would have resulted from continuous reaction.

Gangue Material that has no commercial value surrounding or mixed with the commercially valuable material in an ore deposit.

Geoneutrino An electron anti-neutrino emitted within the earth in the β -decay of ^{232}Th , ^{238}U or ^{40}K and their decay product nuclei.

High Activity Low Enriched Uranium (HALEU) Uranium containing between 5 wt% and 20 wt% ^{235}U .

Highly Enriched Uranium (HEU) Uranium containing more than 20 wt% ^{235}U .

J_{th} Joule (i.e. Watt-second) thermal. One British Thermal Unit (BTU) equals $1055 J_{\text{th}}$.

Log-Normal Distribution A distribution of the form $f(x) = e^{-(\ln x)^2}$. In the present usage the tonnage of an element available at concentration c , $T(c)$, is given by $T(c) = C_1 e^{-(\ln c_0 - \ln c)^2}$, where c_0 is the average crustal abundance and C_1 is a constant.

Low Enriched Uranium (LEU) Uranium containing less than 5 wt% ^{235}U .

Mafic Composed chiefly of dark ferromagnesian minerals.

MOX Mixed OXide fuel, usually consisting of a ceramic mixture of uranium dioxide and plutonium dioxide.

MSWU Mega-Separative Work Unit, a million separative work units. A Separative Work Unit is the separative work that must be done to one kilogram of a mixture of isotopes to change its separation potential by one unit. The separation potential, a dimensionless function, is defined by $\phi x_k = 2x_k - 1 \ln \frac{x_k}{1-x_k}$, where x_k is the atomic fraction of the isotope, k . See [Benedict \(1981, p. 667\)](#) for a more complete definition.

Nebula Clouds of gas and (and later dust particles) surrounding the site of a core-collapse supernova explosion. Depending on the density and energy of the nebula, the clouds gradually accrete into new stars and planetary systems.

Pegmatite An exceptionally coarse-grained igneous rock, with interlocking crystals, often found at the margins of batholiths

Placer A mineral deposit at the surface formed by sedimentary concentration of heavy mineral particles from weathered debris.

Quad Quadrillion (i.e. 1×10^{15} , also written 1E15) British Thermal Units. One quad = $1.055 \times 10^{18} J_{\text{th}}$.

t Metric ton, also used in Mt, million metric tons, and Tt, trillion metric tons (teratons).

Unconformity A break or gap in the geologic record, such as an interruption in the normal sequence of deposition of sedimentary rocks, or a break between eroded metamorphic rocks and younger sedimentary strata ([Bates and Jackson, 1984](#)).

wppb Weight parts per billion, also: wppm weight parts per million and wppt weight parts per trillion.

Yellowcake A concentrate of uranium ore, containing 80–90% U_3O_8 . Yellowcake ranges from yellow to black, depending on impurities, processing temperature and degree of hydration (NRC, 2020). Although uranium prices are sometimes colloquially cited as “dollars per pound of yellowcake,” the actual prices are \$ per lb of U_3O_8 , where all of the uranium is assumed to be present in the yellowcake as that oxide.

Introduction

Beginning with the discovery of nuclear fission in 1938, uranium has been seen as a valuable but scarce resource. Natural uranium is predominately ^{238}U (99.27 atom %) and only about 0.72 atom % is ^{235}U , the only naturally occurring isotope that can be made to fission with thermal neutrons. Consequently, the resources of uranium have been believed to inherently limit the sustainability of nuclear energy. Both ^{238}U and thorium (entirely ^{232}Th) are fertile, meaning that they can be converted to the fissile isotopes ^{239}Pu and ^{233}U respectively, through neutron irradiation. Therefore a consideration of nuclear fuel resources for the coming centuries must include both uranium and thorium.

Origins of fissile materials

Before discussing the resources and distribution of uranium and thorium on earth, it is important to first have a cursory understanding of the original sources of these elements, and all elements heavier than iron. This process of nucleosynthesis occurs in core collapse supernovae and in neutron star mergers. A supernova occurs when a massive star, generally more than ten solar masses, has consumed so much of the hydrogen and helium in its core that it can no longer support the gravitational pressure of outer layers of gas. The core temperature and pressures reach a point where the nuclei dissociate and the overlying layers collapse into the center. The dissociated neutrons collide with nuclei in the infalling layers and form heavier nuclei, including uranium and thorium. A binary neutron star merger occurs when two neutron stars spiral together to form a larger neutron star or a black hole. Neutron star mergers are now being detected by the Laser Interferometer Gravitational-Wave Observatory (LIGO) facilities in Washington state and Louisiana and a similar observatory, Virgo, near Pisa, Italy (Abbott et al., 2017). Gravitational wave signals from the neutron star merger GW170817 in 2017, indicated that during the 100 seconds before the merger, the two neutron stars increased their spiraling frequency from 24 revolutions per second to several hundred revolutions per second. The total mass of the pair was 2.8 solar masses.

The energy available to stars is governed, to first order, by the Curve of Binding Energy, shown in Fig. 1. Binding energy, as the name implies, is the energy required to separate a proton or neutron from a nucleus. Generally, binding energy is expressed as MeV per nucleon. Larger binding energies indicate that a particular nucleus is more tightly bound together. Thus the light nuclei, such as

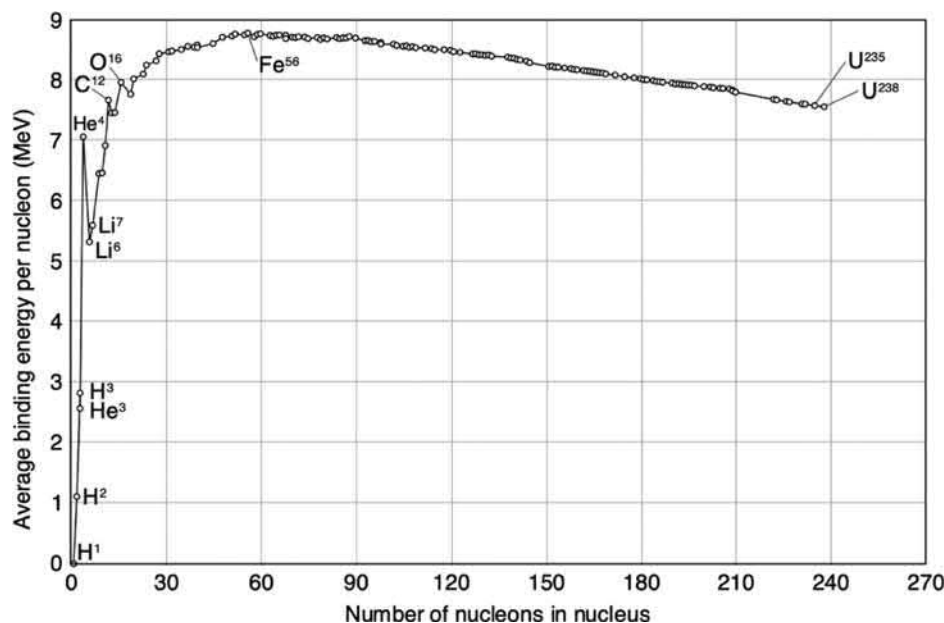


Fig. 1 Curve of binding energy.

^2H (“deuterium”), ^3H (“tritium”) and ^3He have small binding energies, while ^4He has a much higher binding energy. The reactions where two nuclei collide and form a larger nucleus, fusion reactions, such as $^2\text{H} + ^2\text{H} \rightarrow ^3\text{H} + \text{p}$ or $^2\text{H} + ^3\text{H} \rightarrow ^4\text{He} + \text{n}$, release energy because their product nuclei are more tightly bound. The energy is primarily kinetic energy of the resulting nuclei. Thus a fusion reaction between two nuclei heavier than ^{56}Fe requires the input of energy, usually as kinetic energy of the two colliding nuclei. If a ^{235}U nucleus is struck by a neutron and fissions into two fission fragments containing 90–140 nucleons, the reaction releases about 0.85 MeV/nucleon or about 200 MeV for the total reaction, again mainly as kinetic energy of the fission fragments.

Stars derive their energy through the fusion of hydrogen into helium and then further fusion reactions combine nuclei into carbon and all of the elements up to iron. As this process proceeds over millions of years the hydrogen and helium are consumed, especially at the center of the stars. The fusion reactions among nuclei from carbon up to iron yield less and less energy, while the outer layers of the stars continue exerting tremendous gravitational pressures on the core. A star the size of our sun is burning slowly and will expand as a red giant, engulfing the earth and collapse into a white dwarf in about 7 billion years. Our sun can produce the elements up to iron but doesn’t have the mass to drive the endothermic reactions producing substantial amounts of the elements heavier than iron.

Thus the question puzzling most astrophysicists in the mid-20th century was: Why is there tungsten or molybdenum or rhenium or uranium or any of the other elements heavier than iron in the earth?

Newly-built particle accelerators and accumulating data combined with the understanding of transient processes coalesced in a paper by Burbidge, Burbidge, Fowler and Hoyle, known as B²FH (Margaret Burbidge et al., 1957). (Margaret Burbidge, the first author, died on April 5, 2020 at the age of 100. She was an inspiration to generations of astrophysicists.) That paper explains the difference between the slow process (“s-process”) and a rapid one (“r-process”). In our sun the s-process is dominant and progressively heavier isotopes accumulate on a time scale long compared with the radioactive decay of the resulting nuclei (e.g. 10–100 years). Thus nuclei decay without absorbing additional neutrons. The r-process occurs in cataclysmic supernovae and neutron star mergers, where the average nucleus undergoes about 100 neutron collisions per second, much faster than the decay rate.

The first cataclysmic event suggested for the r-process was a supernova in which the depletion of light nuclei at the center of large stars (> 10 solar masses) reduces energy production in the core and there is a sudden collapse of the outer layers into the core. The compression of the core causes a degenerate state within the core in which the nuclei separate into the constituent protons and neutrons over the course of a few minutes. The resulting flash of neutrons and neutrinos meet the infalling outer layers, flowing at a rate of about one solar mass per second, and are sufficient to stop that inflow and to blow off the outer layers of the star. The collisions between the infalling heavy nuclei and outflowing neutrons produce heavier nuclei up to and beyond uranium with neutron fractions out to the “neutron drip edge.”

Wanajo et al. (2003) modeled the distribution of isotopes during 2.5 s of nucleosynthesis following the flash of neutrons and neutrinos in a supernova. Three charts of the nuclides at 0.48, 0.58 and 2.13 s after the neutrino flash are shown in Fig. 2

At $t = 0.48$ s in Fig. 2, nuclei are repeatedly absorbing neutrons as they progress up the chart near the neutron “drip edge,” where no more neutrons can be absorbed. By $t = 0.58$ s, short-lived beta decays have taken place to move the abundances up and to the left. Notice that higher abundances appear with values of N just below the quantum “magic numbers” of 50, 82 and 126.

Our understanding of the astrophysical nucleosynthetic processes occurring during the collapse of a supernova has progressed in the last 25 years both through spectroscopic observations of the interstellar dust that is eventually accreted into planets and through simulations of the nuclear processes during the collapse. Since these supernovae and neutron star mergers are the source of all elements heavier than iron, these observations and simulations can provide estimates of the constituents that formed the earth.

Core-collapse supernova implosions are estimated to occur, somewhere in the universe, at the rate of one per second. Obviously, most such implosions are too distant or masked by dust clouds and are not detected from the earth. Since the beginning of the universe, some interstellar material has gone through multiple cycles of collapse, explosion, dispersal, and accretion into new stars and planets.

The hydrodynamic instabilities of the implosion result in a wide variation in the shapes of the resulting nebulae. Nevertheless, neutron transport and reaction codes have been developed to estimate the distribution of isotopes resulting from nucleosynthetic events. Further simulations of planetary formation have produced models of the Bulk Silicate Earth based on the abundance of various elements and their chemical characteristics (McDonough and Sun, 1995). These BSE models assume that the earth has a uniform composition. As shown in Fig. 3, the uranium abundance in mass should be about seven orders of magnitude less than that of silicon. Since the chemical and planetary accretion characteristics of silicon, uranium and thorium are similar, and since the earth is about 10% silicon, one would expect that the overall concentration of uranium in the earth is about 10 weight parts per billion (wppb) and that thorium should be about 40 wppb.

Stated in other terms, the global inventory is approximately 60 Tt (60×10^{12} t) of uranium and approximately 240 Tt of thorium. Although this inventory is a vast amount of both elements, if uranium and thorium had a uniform distribution throughout the earth, as assumed in the cold accretion model, concentrations of uranium and thorium would be far too small to be economically extracted.

Geoneutrino estimates of uranium and thorium

In the last twenty years, however, a decay product of the 4.5 billion year-half-life of ^{238}U and the 14.2 billion year-half-life of ^{232}Th , has been used to estimate the total global inventory of uranium and thorium. These particles, called neutrinos, are extremely

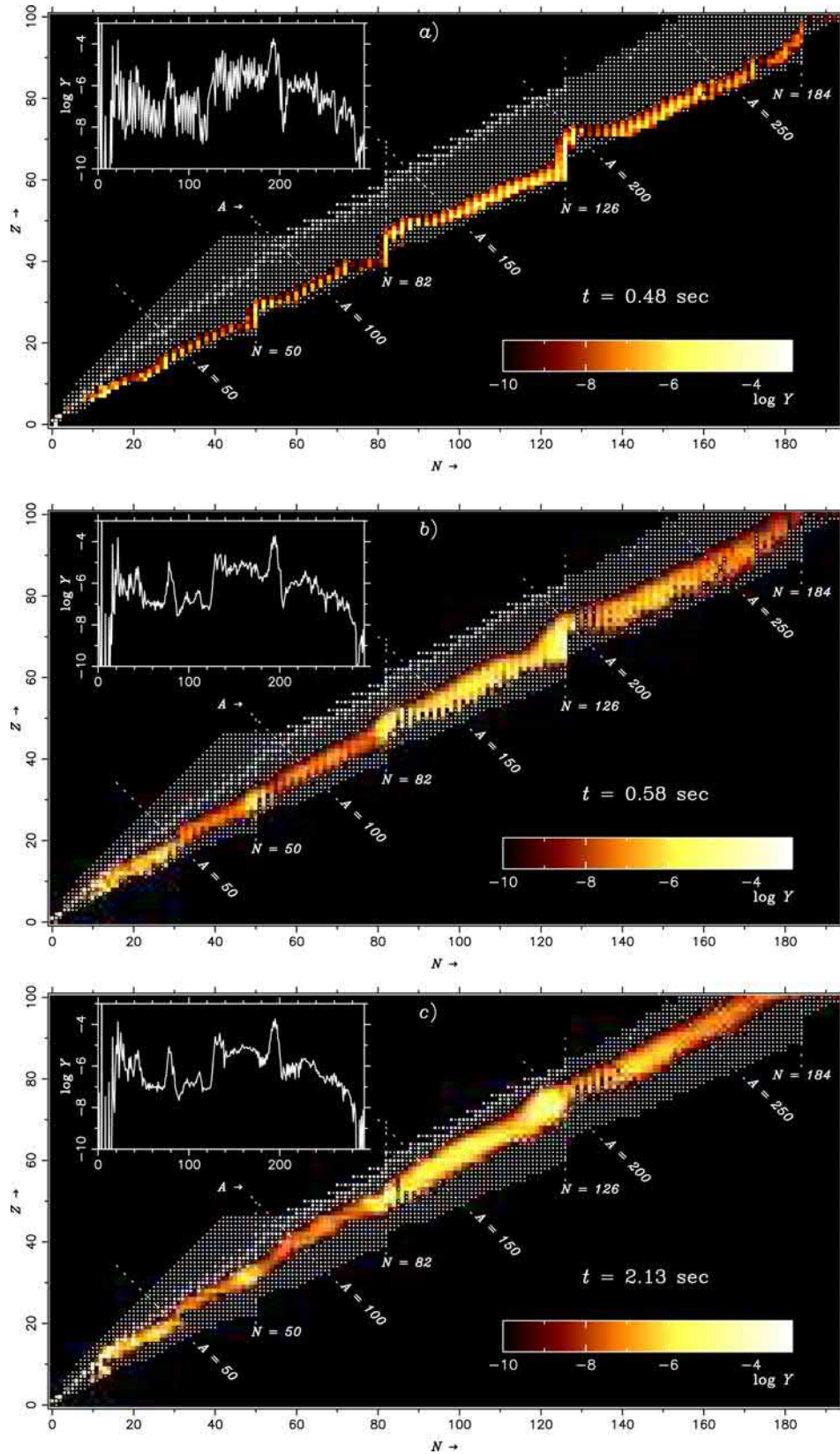


Fig. 2 Isotopic distribution following a core-collapse supernova (Wanajo et al., 2003). Y is the abundance of each isotope normalized to the initial proton abundance. The color scale for the abundances range from $1\text{E-}10$ (dark red) to $1\text{E-}02$ (white) times the initial proton abundance. A is the mass number, Z is the proton number and N is the neutron number for the nuclei. The larger white dots indicate stable or long-lived isotopes. Note that ^{232}Th , ^{235}U and ^{238}U are the uppermost stable or long-lived isotopes. Short-lived neutron-poor isotopes are not shown.

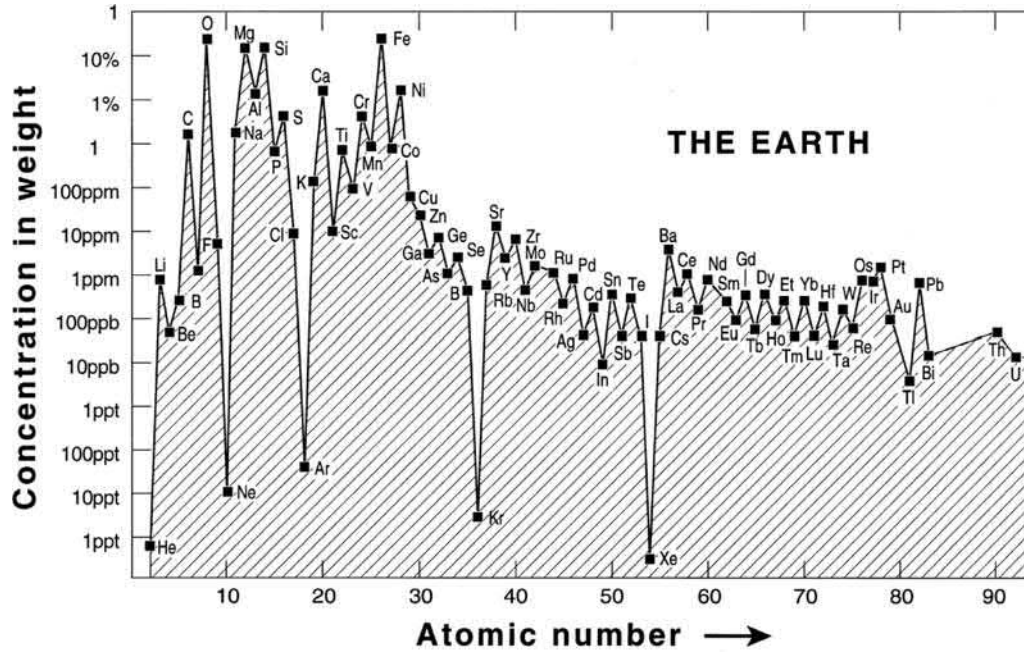


Fig. 3 Composition of the bulk silicate Earth.

difficult to detect and most neutrinos pass completely through the earth without interacting. Thus neutrino detectors are usually a thousand tons in mass and must be located deep underground to avoid unwanted signals caused by cosmic rays.

Neutrinos occur in three types: electron, muon and tau. Each of the three types has a corresponding anti-neutrino. Neutrinos originating within the earth, termed *geoneutrinos*, are actually electron anti-neutrinos primarily resulting from the decay of ^{40}K , ^{238}U and ^{232}Th . Geoneutrinos provide a means for estimating the total uranium and thorium content of the earth and also may provide limited information on the location of those resources. These elementary particles have been measured over the past decade by massive detectors in Japan, Canada and Europe in an effort to differentiate the radiogenic and gravitational components of the total geothermal energy flux through the earth's surface (Davies and Davies, 2010; Ludhova and Zavatarelli, 2013). Neutrino and antineutrino fluxes have also been measured to understand neutrino oscillations, to investigate solar fusion processes and as a first signal of supernova events. Neutrinos (and anti-neutrinos) travel close to the speed of light, have a small mass (< 2 eV) and lack an electric charge. When an electron anti-neutrino collides with a proton, the result is a neutron and a positron (i.e. an anti-electron). This reaction, known as the neutron inverse β decay, was used in the first detection of the neutrino in the Cowan-Reines experiment of 1956. Following the neutron inverse β decay, the positron reacts with a nearby electron to produce two 511 keV gamma rays. The neutron is absorbed by a hydrogen nucleus, releasing a characteristic 2.2 MeV gamma with a mean delay of ~ 200 μs . Circuitry in the detector registers a neutrino event through the delayed emission of a 2.2 MeV gamma following two 511 keV gammas.

As shown in Table 1, the decay chain of ^{238}U into ^{206}Pb results in 6 anti-neutrinos, one anti-neutrino for each beta decay. Similarly, the decay of ^{232}Th in ^{208}Pb results in 4 anti-neutrinos. (Fiorentini et al., 2004). Because of the neutron inverse β decay requires an electron anti-neutrino threshold energy of 1.80 MeV, KamLAND cannot detect ^{40}K anti-neutrinos, but anti-neutrinos from both ^{238}U and ^{232}Th are within the range of this instrument.

The KamLAND (the Kamioka Large Anti-Neutrino Detector), in central Japan, consists of a 18 m diameter spherical vessel which in turn contains a 13 m diameter nylon balloon (Araki et al., 2005). The balloon contains approximately 1000 metric tons of a liquid scintillator (mineral oil, benzene and fluorescent compounds). The volume between the balloon and the spherical vessel contains highly purified oil which shields the balloon from external radiation and provides buoyancy to support the liquid scintillator. About 1900 photomultiplier tubes are mounted on the inner surface of the spherical vessel. Surrounding the spherical vessel is a water Cherenkov detector which provides additional shielding and acts as a muon veto counter.

Table 1 The main properties of geoneutrinos.

Decay	Q (MeV)	$\tau_{1/2}$ (10^9 year)	E_{max} (MeV)	ϵ_H (W/kg)	$\epsilon_{\bar{\nu}_e}$ ($\text{kg}^{-1} \text{s}^{-1}$)
$^{238}\text{U} \rightarrow ^{206}\text{Pb} + 8\ ^4\text{He} + 6\ e + 6\ \bar{\nu}_e$	51.7	4.47	3.26	0.95×10^{-4}	7.41×10^7
$^{232}\text{Th} \rightarrow ^{208}\text{Pb} + 6\ ^4\text{He} + 4\ e + 4\ \bar{\nu}_e$	42.7	14.0	2.25	0.27×10^{-4}	1.63×10^7
$^{40}\text{K} \rightarrow ^{40}\text{Ca} + e + \bar{\nu}_e$	1.32	1.28	1.31	0.36×10^{-8}	2.69×10^4

$\bar{\nu}_e$ denotes electron anti-neutrinos. Q is the energy release for the overall decay chain; $\tau_{1/2}$ is the half-life of the parent isotope; E_{max} is the maximum antineutrino energy in the decay chain; ϵ_H is the heating rate, per kg of the parent isotope; $\epsilon_{\bar{\nu}_e}$ is the electron antineutrino source rate, per kg of the parent isotope.

Table 2 U and Th masses in Reservoirs and in the Bulk Silicate Earth (BSE) model as described in the reference model based on geoneutrino signals from 16 locations (Huang et al., 2013).

	Thickness (km)	U mass (10^{15} kg)	Th mass (10^{15} kg)
Continental crust			
Sediment	1.5	1.2	5.8
Upper crust	11.6	18.2	70.7
Middle crust	11.4	6.6	33.3
Lower crust	10	1.0	6.0
Lithospheric mantle	140	2.9	14.5
Oceanic crust			
Sediment	0.6	0.6	2.8
Crust	7.4	0.4	1.3
Depleted mantle	2090	25.7	70.6
Enriched mantle	710	24.0	113.7
Bulk silicate earth	2891	80.7	318.8

Huang et al. (2013) developed a Reference model of the heat-producing elements in the reservoirs shown in Table 2. The overall results of the KamLAND and Borexino geoneutrino detectors were used to estimate the distribution of Heat Producing Elements (i.e. U, Th and K) in the various reservoirs and those trial distributions were used in a Monte Carlo study to predict the geoneutrino flux at 16 detectors worldwide. The total masses of U and Th in the reservoirs show good agreement with the Bulk Silicate Earth model and also agree with the predictions of (U+Th)/Si ratios from supernova and neutron star merger studies.

The geoneutrino signal and the reference model also indicate that the masses of uranium and thorium in the upper continental crust (UCC) are about 18×10^{15} kg and 70×10^{15} kg, respectively.

Concentration mechanisms for uranium in the Earth's crust

Unlike other energy resources such as coal or petroleum, the resources of uranium are not fundamentally changed by geological processes. Whereas petroleum might be lost through evaporation or combustion or a natural gas reservoir may vent into the atmosphere, uranium is lost only through radioactive decay or through the relatively rare formation of a natural reactor. Therefore the primordial inventory of uranium, reduced by radioactive decay, remains present somewhere in the earth. The crucial question is "where?".

The natural distribution of elements in the earth's crust is controlled by two major factors. The first is the set of ambient geological fractionating processes that leads to regions of depletion and concentration of the element. The second factor includes the overall geochemical characteristics of the element. Elements that are concentrated by a small number of fractionation processes can be expected to have a multi-modal distribution, with a peak in the tonnage versus grade curve for each of the modes of geochemical concentration. For elements having a large number of applicable concentration processes, the peaks overlap and the resulting tonnage versus grade curve takes on a log-normal characteristic. For example, the element chromium, whose distribution at high concentrations is solely governed by fractional crystallization in mafic magmas (i.e. high in magnesium and iron), one would expect a bimodal distribution of concentrations, with one peak at the average crustal abundance and the high concentration peak at the mafic fractionation concentration. On the other hand, most elements, uranium included, can undergo a wide variety of fractionating processes and deposits would be expected over a wide range of concentrations. In this latter case, the tonnage versus grade distribution would be expected to be log-normal. Bear in mind that geological conditions change over time and therefore the distribution patterns have varied with time.

Thermal models of the earth point to inevitable melting of the earth soon after its accretion due to gravitation energy and due to radioactive decay of uranium, thorium and potassium. Because of its large ionic size and heating due to radioactive decay, uranium is transferred into low melting temperature fractions and out of the earth's core and mantle into the crust. These geochemical and geophysical models predict that two thirds of initial uranium present in the earth are now concentrated in the crust, which constitutes only 0.4 % of the earth's total mass. The low uranium and high iron concentrations predicted for the earth's mantle and core have been supported by concentrations in iron meteorites and in mantle issuing from oceanic spreading zones (0.1 ppm U), compared with U concentrations in magma and crust in subduction zones (2 ppm U).

In considering uranium in particular, it is important to examine the tectonic and igneous processes that have redistributed the uranium within the crust. In the past four billion years, the most important processes are continental accretion and plate tectonics. In the accretion process, crust formed into masses of continental dimensions. In the second, continuing, process, the continental crust and the oceanic crust have taken on quite different characteristics in terms of uranium concentration.

Igneous processes

Igneous processes begin with the melting of mantle rocks at depths of 60–200 km, followed by the migration of less dense liquids to the surface. The migration of these less dense minerals to the surface is a predominant process in the formation of the continental

crust. The extruded liquid forms crust in two general locations, at mid-oceanic ridges, where the upwelling material forms new oceanic crust and in subduction zones, where the oceanic crust plunges back into the mantle, usually passing under the edge of a continent.

The behavior of uranium in igneous processes is dominated by two characteristics of the element. In the +4 oxidation state, the condition expected in the earth's mantle, the U^{+4} ion has an ionic radius of 97×10^{-12} m (picometers, pm) about the same as Na^{+1} ion (97 pm). Other ions common in the core and mantle are significantly smaller in radius: Fe^{+2} , 74 pm; Ni^{+2} , 69 pm; Mg^{+2} , 66 pm; and Al^{+3} , 51 pm. Thus, like sodium and the other large ions, uranium ions selectively enter partial melts within the mantle and are transported to the surface.

The second characteristic of uranium is its radioactivity, serving as a source of heat for melting the mantle and core. Like Th^{+4} (ionic radius 102 pm) and K^{+1} (133 pm), these heat-producing elements are readily fractionated out of the mantle and toward the surface. Deffeyes and MacGregor (1978, 1980) note that the earth would be a radically different place if the heat-producing elements had small radii, since the geothermal energy source would then be located deep within the core and the convection currents driving plate tectonics would be much stronger.

The rocks forming the oceanic crust at mid-oceanic ridges are characterized by a uniform uranium concentration of about 0.1 wppm. Conversely, the crust formed above subduction zones is characterized by uranium concentrations of about 2 wppm. The wide difference in concentration is due to the differences in the source materials and to the different chemistry. The upwelling mantle at the oceanic ridge has a uranium concentration of about 0.005 wppm, while the subduction zones have as their source material oceanic crust and bits of continental crust, with an average uranium concentration of about 0.1 wppm. The continuous upwelling at the oceanic ridges serves as a mechanism for depleting the core and mantle of uranium and incorporating that uranium in the oceanic crust. The relatively low concentration of uranium in the oceanic crust is augmented with uranium from continental runoff, which subsequently precipitates in the ocean basins. At the subduction zones, the oceanic crust is again subjected to partial melting and the uranium is again fractionated in the melt and transported to the surface.

Average vertical distribution of uranium and thorium

Thermal gradients in the continental crust are dependent on both heat emanating from the mantle and core of the earth and heat released by heat-producing isotopes within the crust. Temperature measurements as a function of depth in conjunction with measurements of the thermal conductivity of local minerals have indicated that the average continental crustal abundance of uranium decreases exponentially with an e-folding distance ("depth parameter") of 7100–10,000 m.

The anticipated variation of uranium concentration with depth is given by the equation, $U(z) = U(z=0)e^{(-z/h_r)}$ where z is the depth in m, h_r is the depth parameter (discussed below) and $U(z)$ is the concentration at depth z , in wppm. $U(z=0)$ is the average continental crustal abundance of uranium at the surface, 2.76 wppm.

This approximation is based on the presence of heat producing elements, U-238, Th-232 and K-40, in the continental crust, measurements of the thermal conductivity of the crustal materials and the linear temperature distribution with depth measured at many locations. The heat produced in the crust is divided about evenly between U-238 and Th-232, since the crustal abundance mass ratio between Th and U is 3.9 (Table 3). K-40 is about four orders of magnitudes lower, although potassium has a crustal abundance of 2.1%, since K-40 is only 117 ppm of natural potassium and the thermal energy output of K-40 is about four orders of magnitude below U-238 and Th-232, as shown by Lachenbruch (1968, 1970).

Obviously, this method assumes one-dimensional heat transport and a fairly uniform thermal conductivity, without a significant contribution from flowing fluids. A more recent review by Brady et al. (2006) provides more details on the technique.

Several measured values of the depth parameter h_r are listed in Table 4.

If we assume a depth parameter of 8500 m, based on the above data, then 11% of the crustal uranium inventory would be expected to be within 1000 m of the surface and 21% within 2000 m.

Important trends and discoveries

Concerns for the long-term sufficiency of uranium resources have been steadily alleviated over the last fifty years. The uranium mined to supply fuel for the first 40 years following the discovery of fission came primarily from deposits caused by changes in the pH and oxygen content of groundwater and the precipitation of U_3O_8 . In 1975 however, the Western Mining Corporation

Table 3 Sources of heat in the upper continental crust.

Isotope	Thermal output
U-238	0.095 mW/kg
Th-232	0.027 mW/kg
K-40	3.6 nW/kg

Table 4 Temperature distribution depth parameter.

<i>Location</i>	<i>h_r (m)</i>
Sierra Nevada	10,000
Eastern US	7,500
Norway and Sweden	7,200 ± 700
Eastern Canadian Shield	7,100 ± 1700
Canadian Appalachians	10,000 ± 2000
US Appalachians	8,100 ± 1300

Note: In the other references the depth parameter is denoted D, rather than h_r.

discovered the Olympic Dam deposit in South Australia (Schubert, 2001). Western Mining was not exploring for uranium but rather for copper. The deposit that they found has since been termed an iron oxide copper gold (IOCG) deposit, though these ore bodies contain substantial amounts of silver, uranium and the light rare earth elements (Williams et al., 2005). Today, Olympic Dam, with about two million tons of uranium reserves, represents about a third of the known uranium reserves worldwide. In addition, because the uranium is a co-product with copper, gold, silver and iron, the supply is more stable than would be the case for a uranium-only mine. Though the term “IOCG” is still only loosely defined, these deposits represent a new geochemistry in which iron oxide, either as magnetite or hematite, is the dominant gangue mineral. Other IOCG deposits have been identified in Australia, Peru, Brazil, Chile, Mongolia and Sweden.

Extraction of uranium from seawater

Uranium is present in the seawater at a fairly uniform concentration of 3 wppb. Simple calculations can indicate that this concentration is insufficient to warrant pumping seawater through an extraction plant. Various groups have, however, developed surface ligands for polyethylene fibers that effectively chelate uranium ions from seawater. The buoyant fibers, gathered together into loose bundles, are about 40 m long and anchored at a depth of 100 m, allowing 60 m for the passage of ships. The bundles of fibers are immersed for ~90 days, at which time the dry-weight of the fibers is ~0.1% uranium. The bundles are then winched aboard a service ship where the uranium is eluted with an acid solution, after which the bundles are returned to the sea. The estimated lifecycle cost of the uranium recovered in this scenario is about \$640/kg_U (Conca, 2016; Schneider et al., 2014), though more pessimistic assumptions concerning fiber/ligand lifetime, handling costs and extraction rates indicate costs of order \$1000/kg_U.

While \$640 or \$1000/kg_U is well above current uranium prices, the cost of uranium from seawater serves to place an upper limit on future prices for nuclear fuel, first because the inventory in the oceans is immense (about 4 Gt_U) and secondly because only 3% of the global population lives in landlocked countries. Thus no country or international cartel can monopolize the uranium market, as has been the case with petroleum.

To put the increased cost of uranium extracted from seawater in perspective, an increase in the price of natural uranium from the present \$56 per kg_U (i.e. \$30 per lb of U₃O₈) to \$640 per kg_U increases the cost of electricity by \$20 per MWe-hr. An increase in the cost of natural gas from the present \$2.50 per million BTU to \$5.00 per million BTU increases the cost of electricity by \$21 per MWe-hr. Reducing the tails assay after enrichment would decrease the impact of seawater extraction, even accounting for the added additional enrichment costs.

Conclusions

Thus, considering these trends and discoveries, one can have confidence that uranium and thorium will not be a limitation on the sustainability of nuclear energy. We do not know today the exact locations of the ore bodies containing future sources of uranium and thorium, but the global measurements of supernova nucleosynthesis, geoneutrinos and thermal gradients discussed in this article indicate that those sources are very likely to exist. Furthermore, it is not necessary that we know those precise locations today since the fuel itself will not be needed for several centuries.

The geoneutrino measurements indicate that at least 25% of global uranium and thorium are located in the continental crust and the thermal gradient data indicate that these elements are within a few kilometers of the surface. The discoveries of unexpected IOCG deposits indicate that multiple modes for the concentration of uranium exist.

The development of chelating ligands for the extraction of uranium from seawater is not yet economical compared to existing mines and *in situ* extraction on land, but the technology places a ceiling on future uranium prices and assures a widely accessible source. Although the quantity of uranium in the oceans is immense, hundreds of times presently identified resources, the oceans are being constantly replenished by rivers which carry dissolved uranium from the continental crust.

See Also: Unconventional Uranium Resources From Phosphates; Uranium Extraction From Seawater Makes Nuclear Power Virtually Renewable; Uranium Resources.

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Economics of Nuclear Power versus Other Energy Sources

Geoffrey S. Rothwell*, Department of Economics, Stanford University, Stanford, CA, United States

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Glossary

BTU *British Thermal Unit* ($= 1055 \times \text{joule}$), MMBTU is 1 million BTUs.

CRF *Capital recovery factor* is the (annual) rate at which capital expenditures financed with debt and equity capital are paid to banks and investors.

D&D *Decontamination and demolition (decommissioning)* is the process by which an operating facility site is returned to its original state before operations. This involves the decontamination of contaminated structures and equipment, the removal of equipment, and the demolition of structures. Under different regulatory regimes this leads to the end (“decommissioning”) of the operating license.

Debt Two forms of debt are (1) a loan from a financial institution and (2) sales of bonds to investors. Both involve repayment of principle and *interest*. Both forms are repaid before payments to equity providers in the case of bankruptcy (which involves the restructuring of assets and liabilities).

Equity At least two forms of equity are (1) funds provided by the owner/operator out of earnings and (2) funds provided by investors hoping for a *rate of return* from profits.

FOAK *First-of-a-kind* technology refers to the set of early (“first”) units over which design and licensing costs can be recovered, justifying the construction of dedicated manufacturing facilities.

GIF *Generation IV International Forum* is a group of countries working together to develop advanced reactor technologies, known as “fourth generation” technologies.

IDC *Interest during construction* is the amount of financing charges (interest and returns to equity) that accumulate during the construction of a facility.

Interest rate Is the rate of return required by banks or on bonds, see *debt*.

LC *Levelized cost* is the value that equates (1) the discounted costs of a facility over its lifetime, including expenditures during construction and decommissioning with (2) the discounted revenues of a facility’s lifetime.

LCOE *Levelized cost of electricity* is the lifetime levelized cost of a generated megawatt-hour.

Nominal Is the cost or price of a good or service in the year of currency when expended.

PPI *Producer Price Index* is a measure of inflation relevant to producers, similar to the CPI, the Consumer Price Index, which is a measure of inflation facing consumers.

Real Is the nominal cost or price of a good or service inflated or deflated, e.g., by the PPI, to a currency in a particular year, e.g., 2018.

SMR *Small modular reactors* refers to nuclear power reactors (not plants) less than 300 MW that can be manufactured in a dedicated facility and shipped to a licensed power plant site.

Introduction

One of the issues challenging nuclear energy is its competitiveness with other forms of electricity production, such as coal, natural gas, and renewables. Because climate change requires moving away from fossil fuels to clean energy alternatives, such as wind and solar, nuclear energy (the largest source of carbon-free electricity) faces competitive challenges from low-priced natural gas,

*Retired.

subsidies for renewables, state policies that mandate new renewable plants despite their cost, and liberalized market policies regarding the dispatch of power based on variable cost.

This article reviews the current economics of nuclear versus other electricity generating alternatives. The future of nuclear energy depends on whether markets value the long-term investment that nuclear energy represents and on whether regulators allow this value to be represented in the market-based prices that utilities can charge. The nature of nuclear energy is that it is a “base load” supplier of power on a 24-h/365-day basis. This is in contrast to the hourly variability of solar and wind, and the need for backup power when the wind does not blow, or the sun does not shine. Pricing of these alternatives and economic comparisons do not consider the cost of backup power under such circumstances. This article focuses on the basic cost of power (the cost of delivering power to the grid, or “bus bar”) coming from each of the sources without considering system effects or grid reliability. For these, see Rothwell (2021b).

Levelized cost of electricity (LCOE) is an important measure for comparing nuclear energy with other forms of electricity generation. See Rothwell (2021a) for more details regarding the economics of nuclear energy. The LCOE is a measure of discounted life-time costs averaged over discounted electricity revenues. *LCOE* (note: variables are in italic and parameters are in bold) can be defined as in NEA/IEA (2015, p. 28):

$$LCOE = \sum_t [(I_t + O\&M_t + F_t)/(1+r)^t] / \sum_t [E_t/(1+r)^t] \quad (t = -LT, \dots, T), \quad (1)$$

where

I_t are investment expenditures (including decommissioning costs) in year t ,

$O\&M_t$ are Operations and Maintenance expenditures (including refurbishments) in year t ,

F_t is the product of fuel expenditures, F_{MWh} , and E_t in year t , including a fee for spent nuclear fuel, SNF, management and disposal, r is the discount rate (a parameter) and $(1+r)$ is the discount factor, e.g., $(1+5\%)$,

E_t are the MWh of electricity generated in year t , and

LT is the construction lead time.

The summations in Eq. (1) run from (1) the start of large construction expenditures when the foundation is poured ($t = -LT$, the construction lead time) to the start of commercial operation ($t = 0$) and (2) from the start of commercial operation to the end of commercial operation, T , the life of the plant. (There is a third period from the end of the commercial life to the end of decommissioning; the cost of which is treated as a capital cost or as a contribution to a sinking fund accounted for in $O\&M$.) The discount rate is used to estimate how much the project’s future (or past) cash flows (expenditures or revenues) would be worth at a specific time, such as, $t = 0$, the present. At a discount rate of 5%, an $O\&M$ expenditure of \$1,000,000 in $t = 2$ would be $\$1,000,000/(1.05)^2 = \$907,029$. See Rothwell (2021b).

In this discussion, a distinction must be made between nominal and real values. Nominal prices and costs are those in the currency of the year of the expenditure. Real prices and costs are those in the currency of a specific year (here, prices are in 2018 USD). (Prices are determined in markets; costs are the resources required to produce commodities and services traded in markets.) Here, nominal prices are deflated (inflated) to real prices with the Producer Price Index (PPI), see Federal Reserve Bank of St. Louis (2019). It is often difficult to tell in the literature whether prices and costs are in nominal currency or in real currency, if not explicitly stated. Because taxes are assessed on, and budget outlays are stated in, nominal values, most cost estimates associated with governmental or regulatory oversight are in nominal currency. This article uses real prices and costs. However, the PPI increased an average of 5.7% per year from 2000 to 2008, then experienced an average of 0% from 2008 to 2018 (the average from 2000 to 2018 was 2.6%). This means that cost estimates made before July 2008 should be discounted by the PPI, but cost estimates after July 2008 are *similar* to what they would be in 2020 (ignoring real escalation and real changes in the costs of construction, $O\&M$, and fuel).

Examples of electricity levelized costs can be found in NEA/IEA (2015). Table 1 presents levelized costs for nuclear power plants (NPPs). In each case the investment cost (construction cost plus financing cost) is the largest component of NPP LCOE. Investment costs are lowest in East Asia. The $O\&M$ cost (which is dominated by the cost of labor, see NEA/IAEA 2018) is less than the fuel cost for Chinese and Korean NPPs and greater than the fuel cost for European and US NPPs. LCOE is lowest in China and Korea, about the same in the United States, Continental Europe, and Japan, and highest in the United Kingdom.

To provide a historical context consider the types of nuclear power plants constructed since 1951. Nuclear power plants can be classified by means of a few characteristics, as in Fig. 1. First, a reactor has either a “fast” or “slow” chain reaction. There have been 12 fast power reactors built or under construction. Second, thermal reactor technologies (with “slow” chain reactions) can be characterized along two dimensions: (a) the choice of nuclear reaction moderator, to capture nuclear particles; and (b) the choice of nuclear reactor coolant, to remove heat. To moderate and control the chain reaction, light water (H_2O), heavy water (D_2O , in which deuterium, D , has twice the mass of hydrogen), or graphite (carbon) are used. Coolants are either water (light or heavy) or gas (air, carbon dioxide, or helium). (For compatibility with IAEA, 2019, 2020, two demonstration reactors are included in Fig. 1: the organic [hydrocarbon] cooled and moderated reactor, OCMR, and the liquid-metal cooled and graphite-moderated reactor, LMGMR.) Table 2 looks more specifically at light and heavy water nuclear power plants. Most under construction are PWRs (pressurized water reactors) and most of these are Russian (VVERs, Rosatom, 2018), Chinese (many models), and Korean (APR1400), some of which have been under construction for 10 years or more. Many American and French nuclear power units have been operating for more than 33 years (the median age of all units). On the profitability of NPPs in the United States, see UCS (2018).

Table 1 LCOE and cost components for nuclear power plants at a 5% discount rate (2018 USD).

Country		China	Korea	China	Belgium	United States	Hungary	Japan	United Kingdom
Type		CPR 1000	APR 1400	AP 1000	Gen III	PWR Gen III	VVER 1200	ALWR	EPR
Size	MWe	1080	1343	1250	1000	1144	1180	1152	1650
Overnight cost	\$/kWe	1825	2041	2641	5132	4920	6277	3922	7092
Investment cost	\$/kWe	2175	2310	3147	6117	5858	7208	4673	8322
Investment + D&D	\$/MWh	\$15.13	\$16.08	\$21.90	\$42.50	\$43.70	\$50.59	\$32.54	\$57.87
O&M cost	\$/MWh	\$6.57	\$9.75	\$7.39	\$13.69	\$19.06	\$10.50	\$27.70	\$23.68
Fuel + waste cost	\$/MWh	\$9.42	\$8.67	\$9.42	\$10.56	\$7.40	\$9.70	\$14.29	\$11.42
Variable cost	\$/MWh	\$15.99	\$18.41	\$16.73	\$24.25	\$26.46	\$20.20	\$42.00	\$35.11
Levelised cost	\$/MWh	\$31.00	\$34.00	\$39.00	\$67.00	\$70.00	\$71.00	\$75.00	\$93.00

Based on data from NEA (Nuclear Energy Agency) (2018a) *Sustainable Development and the Application of Discounting to the Calculation of the Levelised Costs of Electricity*. Nuclear Technology Development and Economics, NEA/NDC/R(2018)1 (June). [http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDC/R\(2018\)1&docLanguage=En](http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/NDC/R(2018)1&docLanguage=En); NEA (Nuclear Energy Agency) (2018b) *The Full Costs of Electricity Provision*. Paris: OECD. https://inis.iaea.org/collection/NCLCollectionStore/_Public/49/047/49047241.pdf?r=1&r=1, updated to 2018\$.

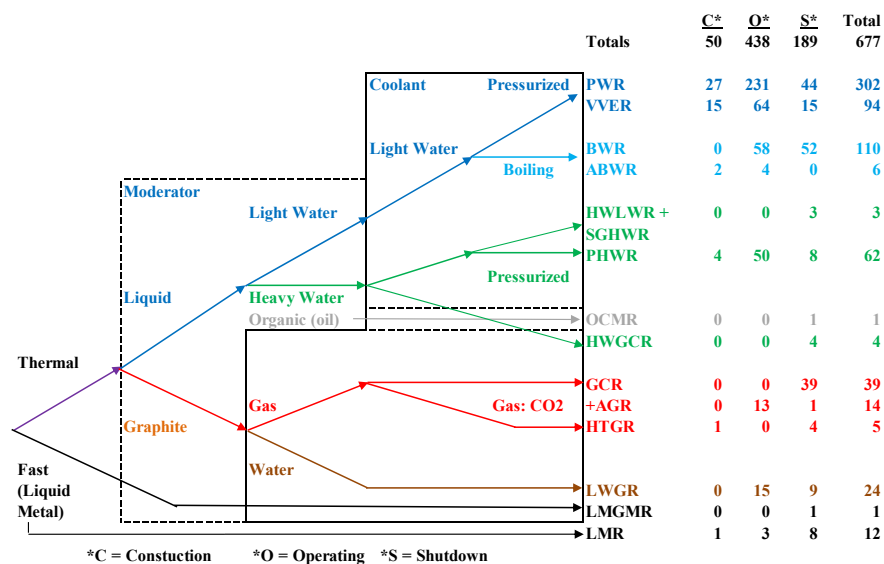


Fig. 1 Commercial nuclear power reactor technologies, 2020. Based on Rothwell GS (2016) *Economics of Nuclear Power*. Routledge Publishers, Francis and Taylor Group. https://www.academia.edu/37151189/The_Economics_of_Future_Nuclear_Power (Figure 1.2), updated with data from with IAEA (2019) *Nuclear Power Reactors in the World*. Reference Data Series No. 2, 2019 Edition. Vienna, Austria: International Atomic Energy Agency. <https://www.iaea.org/publications/13552/nuclear-power-reactors-in-the-world> and IAEA (2020) *Power Reactor Information System (PRIS)*. Vienna, Austria: International Atomic Energy Agency. <https://www.iaea.org/resources/databases/power-reactor-information-system-pris>.

Section “Components of the levelized cost of electricity” of this article discusses the components of levelized cost. Section “Comparisons of nuclear generation costs with other power generating technologies” compares the levelized costs of nuclear power with Combined Cycle Gas Turbines (CCGT) burning natural gas, coal, geothermal, photovoltaic solar, concentrating solar, onshore and offshore wind, and hydropower. Section “Observations on the past and future of the nuclear power industry” offers observations on the future of commercial nuclear power.

Components of the levelized cost of electricity

Fixed costs: Generating plant construction and its financing costs

Levelized capital cost, LKC, is a function of total capital investment cost and the cost of capital:

$$LKC = KC\{OC(MW) \cdot [1 + IDC(r, LT)]\} \cdot CRF(r, T)/E, \quad (2)$$

Table 2 Commercial light and heavy water reactors in 2020.

	C^*	$C-GW$	$C > 9$	O^*	$O > 33$	S^*	Total
Total	50	51.627	11	438	223	189	677
PWR	42	45.784	8	295	139	59	396
VVER	15	14.602	5	64	23	15	94
Chinese	12	11.693	0	36	0	0	48
Korean	7	9.400	0	14	0	0	21
French	4	6.490	2	70	41	2	76
American	2	2.234	0	87	64	27	116
Other ($\approx$ DE + JP)	2	1.365	1	24	11	15	41
GE-BWR	2	2.653	2	50	32	24	76
Non-GE-BWR	0	0.000	0	12	10	28	40
CANDU	0	0.000	0	30	18	5	35
Non-CANDU	4	2.520	0	20	5	3	27
Other	2	0.670	1	31	19	70	103

Notes: C^* , under construction; O^* , in operation; S^* , shutdown.

Based on Rothwell GS (2016) *Economics of Nuclear Power*. Routledge Publishers, Francis and Taylor Group. https://www.academia.edu/37151189/The_Economics_of_Future_Nuclear_Power (Figure 1.2), updated with data from IAEA (2019) *Nuclear Power Reactors in the World*. Reference Data Series No. 2, 2019 Edition. Vienna, Austria: International Atomic Energy Agency. <https://www.iaea.org/publications/13552/nuclear-power-reactors-in-the-world> and IAEA (2020) *Power Reactor Information System (PRIS)*. Vienna, Austria: International Atomic Energy Agency. <https://www.iaea.org/resources/databases/power-reactor-information-system-pris>.

where

KC is the total capital investment cost, a function of OC (see Rothwell, 2021a),

OC is overnight cost, including contingency, a function of size, MW,

MW is the size of the plant in net megawatts-electric,

IDC is the rate of Interest During Construction to finance KC , a function of r and LT ,

r is the appropriate real cost of capital,

LT is the construction lead time,

T is the economic life of the facility,

CRF is the capital recovery factor, a function of r and T , see Rothwell (2021b), and

E is the annual electricity output expressed in megawatt-hours (MWh).

Construction cost, KC , is the total amount spent on construction and testing before any electricity or revenues can be generated (as defined in GIF/EMWG, 2007, p. 79). KC is equal to the overnight construction cost with contingency plus financing costs, as measured by IDC . To measure these consistently, a set of standard definitions of construction accounts, structures, equipment, and personnel is required. The Code of Accounts (COA) used here is derived from the *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (GIF/EMWG, 2007), developed by the Economic Modelling Working Group (EMWG) of the Generation IV International Forum (GIF). While applied to the cost of nuclear power, with the GIF COA any electricity generating technology can be characterized by this COA. The total capital investment cost, KC , includes:

- Account 20, Direct construction costs plus Account 10, which includes pre-construction costs, i.e., those costs before the first concrete is poured for the foundation.
- Account 30, Indirect costs include engineering and administrative costs that cannot be associated with a specific cost category in Account 20.
- Account 40, Owners' costs include expenses incurred by the owner(s) associated with the plant and plant site, but not including off-site transmission system costs; the owners' costs associated with the development of the site, e.g., US Environmental Protection Agency (EPA) licensing fees.

A technology appropriate contingency is added to direct, indirect, owners' plus any supplemental costs. And to these accounts is added "Interest During Construction," IDC , expressed as a percentage mark-up on overnight costs, known as the "IDC factor."

The sum of Accounts 10–50 is the base overnight construction cost, $BASE$. Table 3 portrays the estimated costs of a System 80+, which was based on the System 80 design used at Palo Verde Units 1, 2 and 3, the largest NPP in the United States. The term "overnight" is used to describe what the construction cost would be if money had no time value. $BASE$ plus contingency, CN , equals total overnight cost, OC . Some references define OC without contingency and some references define overnight cost with contingency, as does the US Energy Information Administration (US EIA, 2019), see below. To make this distinction, $BASE$ excludes contingency and OC includes contingency. OC plus Account 60, IDC , equals total capital investment cost, KC . To understand the relationship between lead time and compounding IDC , see Rothwell (2021b). Although usually not a part of KC , "decontamination and dismantling" (D&D) costs must be set aside before or during operation for plant D&D after operation. In the latter case, D&D funding is treated as a sinking fund (and is assigned to O&M costs, see next section). (D&D cost might not include decontamination if hazardous materials have not been used.)

Table 3 Construction cost estimates for a single unit of a twin PWR in the United States.

COA	System 80+ General description	Cost USD-2018 (in millions)	Cost USD-2018 (per kWe)
20	Capitalized direct costs	\$2626	\$2296
30	Capitalized indirect costs	\$1556	\$1360
40	Capitalized owner's cost	\$486	\$425
	Base construction costs (<i>BASE</i>)	\$4668	\$4081
	Contingency (20% of <i>BASE</i>)	\$904	\$791
	Overnight construction cost	\$5573	\$4871
60	Capitalized financing costs (<i>IDC</i>)	\$1062	\$929
	Total capital investment cost (<i>KC</i>)	\$6635	\$5800
	Annual payment to capital (5% for 60 years)	\$351	
	Annual generation with 85% capacity factor (GWh)	8524	
	Levelized capital cost (<i>LKC</i>)	\$41.12	

Note: Cost have been increased from ORNL (2011, p. 81) with a 2% real annual escalation rate for NPPs not yet under construction.

Based on ORNL (2011) *Advanced High Temperature Reactor Systems and Economic Analysis*. ORNL/TM-2011/364, prepared by Oak Ridge National Laboratory under the direction of UT-Battelle, Oak Ridge. <http://info.ornl.gov/sites/publications/files/Pub32466.pdf>.

Variable costs: Operating and maintenance and fuel

This section discusses operations and maintenance (O&M) and fuel costs; for more details see Rothwell (2021a). While O&M costs can be treated as variable costs, for many generating technologies some O&M costs are fixed, and some vary with output. For this reason, US EIA (2019) gives both "Variable O&M" in \$/MWh and "Fixed O&M" in \$/kW per year. Given the LCOE is in dollars per MWh, Fixed O&M must be converted into \$/MWh as a function of the capacity factor and generator capacity:

$$O\&M_{MWh} = \text{Variable O\&M} + (\text{Fixed O\&M} \times 1,000)/(CF \times 8766), \quad (3)$$

where

$CF \equiv E/(MW \times 8766)$. In Table 6 (below), the assumed Variable O&M for an NPP is \$2.37/MWh, the assumed Fixed O&M is \$103.31/kw per year, and the assumed capacity factor is 85% (Table 5). $O\&M_{MWh}$ is equal to $\$2.37 + (\$103.31 \times 1000)/(0.85 \times 8766) = \$2.37 + \$13.87 = \$16.24/\text{MWh}$ (Table 7). On the other hand, Table 4 shows annual capital and O&M cost estimates from ORNL (2011, p. 81). In this estimate annual O&M costs are approximately \$19/MWh. Other discussions of the relationship between staff size (the largest component of O&M) and O&M can be found in Rothwell (2016, Section 4.3) and NEA/IAEA (2018, Chapter 2). Annual fuel and irradiated fuel management costs are added to Table 4 next.

Second, consider levelized fuel costs over the life of the NPP and fuel expenditures in each year. The cost per MWh of LEU fuel, F , is

$$F_{MWh} = [FC/(\epsilon \cdot \text{BURN} \cdot 24)] + \text{SNF}, \quad (4)$$

where

FC is the cost of nuclear fuel in USD per kilogram of uranium (USD/kgU),

ϵ is the thermal efficiency of converting MW-thermal into MW-electric,

BURN is the burnup rate measured in thermal megawatt-days per kgU,

24 is the number of thermal MWh in a thermal megawatt-day,

E are the MWh of electricity generated in year t , and

SNF is the interim storage cost of Spent Nuclear Fuel per MWh (assumed to be \$1.33/MWh real) plus geologic disposal costs (assumed to be \$1/MWh real).

Table 4 Levelized cost of electricity estimates for a generic PWR unit in the United States.

System 80+ PWR 4-loop	2018\$
Annual capital costs per MWh	\$41.12
Annual D&D contributions per MWh	+\$2.15
Annual O&M costs per MWh	+\$18.87
Annual fuel and SNF management costs per MWh	+\$7.33
Levelized costs of electricity per MWh	=\$69.47

See text.

For example, if the burnup rate is 50 MW-days per kgU and the efficiency is 33.3%, then the cost of fuel per MWh is (1) the cost of fuel per kgU divided by $[50 \text{ MWth-days} \times 24 \text{ h per day} \times 0.33 \text{ MWh per MWe}] = 400 \text{ MWh}$ plus (2) the cost of irradiated fuel management and disposal, assumed here to be \$2.33 from NEA/IEA (2015, pp. 32–33). If the cost per kgU is \$2000, then F_{MWh} equals \$5/MWh plus \$2.33/MWh, or about \$7.33/MWh. (Fuel prices in NEA/IEA, 2015, were estimated in 2014 when both the prices of natural uranium and SWU were higher, i.e., the implicit assumption in NEA/IEA, 2015, is that the cost of nuclear fuel per kg is \$2800 for fuel with a burnup of 50 MW-days per kgU.) Table 4 adds NPP fuel costs to capital and O&M costs.

While construction and O&M costs follow the same structure for electricity generators other than nuclear power, fossil fuel costs are calculated with a different formula. Here, the cost of fuel for a MWh generated with natural gas in a CCGT is given (this is applicable to all fossil fuels):

$$F_{\text{CCGT/MWh}} = [(P_{\text{NGAS}} \cdot \text{HR}_{\text{CCGT}})/1000] + (P_{\text{CO}_2} \cdot \text{CO}_{2\text{CCGT}}), \quad (5)$$

where

$F_{\text{CCGT/MWh}}$ is the cost of natural gas per MWh,

P_{NGAS} is the price of natural gas in \$/MMBTU,

HR_{CCGT} is the heat rate (in BTU/kWh) for CCGTs,

P_{CO_2} is the fee for emitting a tonne of CO_2 , and

$\text{CO}_{2\text{CCGT}}$ is the tonnage of CO_2 produced by the CCGT unit per MWh, known as the “carbon intensity factor.”

For example, if (1) the price of natural gas is \$5.50/MMBTU, (2) the heat rate is 6200 BTU/kWh, (3) the carbon intensity factor is 0.330, and (4) the price of a tonne of CO_2 is assumed to be \$30.00, then the cost of natural gas per MWh is \$34.10 without a carbon fee and is \$43.99/MWh with a carbon fee. These costs are explored for other electricity generators in the next section and a sensitivity analysis is conducted on the volatile price of natural gas.

Comparisons of nuclear generation costs with other power generating technologies

This section compares nuclear generation costs with other electricity generating technologies following the assumptions in US EIA (2019) and in NEA/IEA (2015), in anticipation of the publication of the next edition of this report: *Projected Costs of Generating Electricity, 2020 Update*. The 2020 edition will survey NEA and IEA member country experts regarding their expectations of the costs of generating electricity in 2025 (not all member countries submit fully completed surveys). One of the primary differences in the 2010 report (NEA/IEA, 2010) from the 2015 report was the use of 5% and 10% as discount rates in the 2010 report and 3%, 7%, and 10% in the 2015 report, making direct comparisons between the two reports at 3%, 5%, and 7% difficult. To simplify the discussion here a 5% discount rate is used. (On the sensitivity of the competitiveness of nuclear power to the assumed cost of capital, see Rothwell 2021b.)

To compare the projected costs in Table 1 with those of other electricity generating technologies, Table 5 gives default values for construction lead times (LT) and lifetimes (T) from NEA/IEA (2015). Where country survey data was provided for contingency percentage rates, the country data was used. As stated in NEA/IEA (2015, p. 31), “the following conventions have been adopted [for contingency rates] if national data were not available: (1) nuclear power: 15% of overnight costs, and (2) all other technologies: 5% of overnight costs.” These assumptions did not follow published guidelines (as discussed in Rothwell 2021a) and were determined through debate among the member country experts.

Also, there is little guidance regarding appropriate end-of-life costs (NEA/IEA, 2015, p. 33):

Table 5 Default assumptions for cost projections in NEA/IEA (2015).

Electricity generating technologies	Lead time (years)	Life-time (years)	Contingency (%)	Decommissioning (% of OC)	Capacity factor (%)
Nuclear	7	60	15%	15%	85%
Combined cycle natural gas turbines (CCGTs)	2	30	5%	5%	85%
Coal	4	40	5%	5%	85%
Geothermal	1	40	5%	5%	80% ^a
Solar (photovoltaic, PV)	1	25	5%	5%	14% ^a
Solar (concentrating solar power, CSP)	1	25	5%	5%	25% ^a
Wind (onshore)	1	25	5%	5%	28% ^a
Wind (offshore)	1	25	5%	5%	40% ^a
Hydropower	Survey	80	5%	5%	68% ^a

^aData from survey of United States in NEA/IEA (2015, Chap. 3).

Based on NEA (Nuclear Energy Agency) and IEA (International Energy Agency). (2015) *Projected Costs of Generating Electricity, 2015 Update*. Paris: OECD. <https://www.oecd-nea.org/ndd/pubs/2015/7057-proj-costs-electricity-2015.pdf>, pp. 30–34.

"At the end of a plant's lifetime, decommissioning and waste management costs are spread over a period of ten years for all technologies. In case of any positive 'residual value' after the operating lifetime of a plant (iron scrap value, left-over carbon permits, etc.), there was a possibility to also report it. For fossil fuel plants, the residual value of equipment and materials is assumed to be equal to the cost of dismantling and site restoration, resulting in a zero-net cost of decommissioning. For wind turbines and solar panels, rather than decommissioning, what takes place in practice at the end of their operating lifetime is a replacement of equipment, and the scrap value of the renewable installation is estimated to amount to 20% of the original capital investment. However, no country reported such residual values."

Because no country surveyed reported such residual values, an assumption was made that the end-of-life costs would be equal to those in [Table 5](#). Because these costs were discounted at the same rate as the cost of capital (3%, 7%, and 10%), annual contributions to end-of-life funds were mute (usually much less than \$0.25/MWh at 7% and 10% discount rates). See discussion of the appropriate discount rate in [NEA \(2018a\)](#).

Regarding capacity factors, [NEA/IEA \(2015, p. 31\)](#) states, "a standard capacity factor of 85% was used for all CCGTs, coal, and nuclear plants under the assumption that they operate in baseload. While it is clearly understood that many CCGTs are frequently used in mid-load or even peak load rather than in baseload, since the overarching concern here is with baseload capacity, the 85% assumption is also used as a generic assumption for CCGTs." Capacity factors for other technologies were taken from country surveys. The capacity factors for renewables varied enormously across countries. Therefore, [Table 5](#) gives capacity factors for various renewables (including hydro) for the United States, corresponding to expected costs in [Table 6](#). Generally, no storage costs are added to the costs of generating electricity if the capacity factor is less than what might be expected for a baseload technology given that each technology is dispatched in a electricity system as a function of its fixed and variable costs; see discussion in [Rothwell \(2021b\)](#).

Default fuel prices are given in [NEA/IEA \(2015, pp. 32–33\)](#). For the United States, hard coal is assumed to have a price of \$101/t (USD 2013 = USD 2018), and natural gas is assumed to have a price of \$5.50/MMBTU (USD 2013 = USD 2018), both are assumed to have a carbon tax of \$30/t of CO₂. Nuclear power is assumed to have the following once-through fuel cycle costs across the world (unless given in a country's survey):

"For uranium prices, an indicative value that does not directly enter the calculation of the front end of the nuclear fuel cycle is USD 100/kg of U₃O₈, which has been the average price during the last half century (1) Front end of nuclear fuel cycle (mining, enrichment, conditioning): USD 7.00/MWh and (2) Back end of nuclear fuel cycle (spent fuel removal, disposal and storage): USD 2.33/MWh".

For the United States, assumptions in [NEA/IEA \(2015\)](#) are updated using the assumptions in [US EIA \(2019\)](#). [Table 7](#) compares levelized cost of ("Advanced") CCGT, ("Advanced") Nuclear, ("On-shore") Wind, and (fixed-tilt) Solar PV as defined in US EIA (2019), supplemented with assumptions from US EIA (2016), [NEA/IEA \(2015\)](#), and calculations as in [Rothwell \(2016\)](#). (See [NEA, 2018b](#), comparing the costs of pollutants from each of these generating alternatives.) At \$5.50 per MMBTU, CCGT is the cheapest alternative. The levelized cost of nuclear and onshore wind are approximately the same. Solar PV is significantly higher due to its low capacity factor. These levelized costs are based on published assumptions and depend on the price of natural gas.

Because many renewables are dispatched when they generate electricity, their non-economic introduction into liberalized markets can create unanticipated load and profitability losses. As explained in [NEA \(2018b, p. 82\)](#), based on information from [NEA, 2012](#)):

"it should be recognized that the recent rapid deployment of subsidized [Variable Renewable Energy] in many OECD countries has had a profound impact on the electricity market, in particular contributing to a sharp decrease of wholesale electricity prices and to an increase of their volatility. This, in

Table 6 US Energy Information Administration costs of new generating stations.

Electricity generating technology	Size (MW)	Lead time (years)	Contingency	Over night (\$2018/kW)	Variable O&M (\$/MWh)	Fixed O&M (\$/kW/year)
Nuclear	2234	6	15.5%	\$6034	\$2.37	\$103.31
CCGT	1100	3	8%	\$794	\$2.06	\$10.30
Coal (with 30% CCS) ^a	650	4	10.2%	\$5169	\$7.31	\$72.12
Geothermal	50	4	5%	\$2787	\$0.00	\$122.28
Solar (photovoltaic)	150	2	5%	\$1969	\$0.00	\$22.46
Solar thermal (CSP)	100	3	7%	\$4291	\$0.00	\$72.84
Wind (onshore)	100	3	7%	\$1624	\$0.00	\$48.42
Wind (offshore)	400	4	37.5%	\$6542	\$0.00	\$80.14
Hydropower	500	4	10%	\$2948	\$1.36	\$40.85

^aCCS: carbon capture and storage.

Based on US EIA (Energy Information Administration) (2019) *Cost and Performance Characteristics of New Generating Technologies. Annual Energy Outlook 2019*. US EIA DOE/EIA – 0383(2019). https://www.eia.gov/outlooks/aeo/assumptions/pdf/table_8.2.pdf.

Table 7 Levelized cost for four electricity generation technologies, 2018 US\$, $r = 5\%$.

Input		CCGT	Nuclear	Wind	Solar-PV	Source
Net electrical capacity	MWe	1100	2234	100	150	US EIA (2019)
Average capacity factor	%	85%	85%	28%	14%	NEA/IEA (2015)
Plant economic life	Years	30	60	25	40	NEA/IEA (2015)
Construction lead time	Years	3	6	3	2	US EIA (2019)
Base overnight cost	\$/kw	\$736	\$5224	\$1518	\$1698	US EIA (2019)
Contingency	%	8%	15.5%	7%	5%	US EIA (2019)
Total overnight cost	\$/kw	\$794	\$6033	\$1623	\$1782	+ Contingency
IDC factor	%	7.5%	15.8%	7.5%	4.8%	Rothwell (2021b)
KC per kW with IDC	\$/kw	\$853	\$6988	\$1744	\$1868	+ IDC
Decommissioning cost/kW	\$/kw	\$43	\$1048	\$87	\$93	NEA/IEA (2015)
Decommissioning cost/MWh	\$/MWh	\$0.01	\$0.02	\$0.05	\$0.04	NEA/IEA (2015)
Fuel price + SNF fee (nuclear)	\$/MMBTU	\$0.00	\$9.33	\$0.00	\$0.00	NEA/IEA (2015)
Fuel price (fossil)	\$/MMBTU	\$5.50	\$0.00	\$0.00	\$0.00	NEA/IEA (2015)
CO ₂ price (\$/tonne)	\$/tonne	\$30	\$30	\$30	\$30	NEA/IEA (2015)
CO ₂ /MWh (CO ₂ intensity)	t/MWh	0.330	0.000	0.000	0.000	US EIA (2016)
Heat rate (from US EIA, 2019)	BTU/kWh	6200	10,461	NA	NA	US EIA (2019)
Variable O&M	\$/MWh	\$2.06	\$2.37	\$0.00	\$0.00	US EIA (2019)
Fixed O&M	\$/kW	\$10.30	\$103.31	\$48.42	\$22.46	US EIA (2019)
Levelized capital cost + D&D	\$/MWh	\$7.45	\$49.57	\$50.48	\$88.74	
Levelized O&M cost	\$/MWh	\$3.44	\$16.24	\$19.73	\$18.30	
Levelized fuel	\$/MWh	\$34.10	\$9.33	\$0.00	\$0.00	
Levelized fuel CO ₂ fee	\$/MWh	\$9.89	\$0.00	\$0.00	\$0.00	
Levelized cost without waste fee	\$/MWh	\$45.00	\$72.80	\$70.20	\$107.04	
Levelized cost with waste fee	\$/MWh	\$54.89	\$75.13	\$70.20	\$107.04	

Based on US EIA (Energy Information Administration) (2019) *Cost and Performance Characteristics of New Generating Technologies. Annual Energy Outlook 2019*. US EIA DOE/EIA – 0383(2019). https://www.eia.gov/outlooks/aeo/assumptions/pdf/table_8.2.pdf, p. 5, US EIA (Energy Information Administration) (2016) *Carbon Dioxide Emissions Coefficients*. US EIA. https://www.eia.gov/environment/emissions/co2_vol_mass.php, NEA (Nuclear Energy Agency) and IEA (International Energy Agency) (2015) *Projected Costs of Generating Electricity, 2015 Update*. Paris: OECD. <https://www.oecd-neo.org/ndd/pubs/2015/7057-proj-costs-electricity-2015.pdf>, pp. 27–35, and Rothwell GS (2021b) *Economic challenges of nuclear power*. Chapter 00225, this Encyclopedia.

Table 8 Short-term electrical load and profitability losses with renewables penetration rates.

Penetration		Wind 10%	Solar 10%	Wind 30%	Solar 30%
Load	CCGT	–34%	–26%	–71%	–43%
Losses	Coal	–27%	–28%	–62%	–44%
	Nuclear	–4%	–5%	–20%	–23%
Profitability	CCGT	–42%	–31%	–79%	–46%
Losses	Coal	–35%	–30%	–69%	–46%
	Nuclear	–24%	–23%	–55%	–39%

Based on data from NEA (Nuclear Energy Agency) (2012) *Nuclear Energy and Renewables: System Effects in Low-carbon Electricity Systems*. Paris: OECD. <http://www.oecd-neo.org/ndd/pubs/2012/7056-system-effects.pdf>, p. 136.

turn, has severely affected the revenue stream and the profitability of incumbent utilities, thus jeopardizing their financial stability and their ability to take on new investments. The short-term impact on market prices has been, by far, the most acutely felt by the industry... The combination of a reduced capacity factor and reduced electricity prices may have a severe impact on the revenues of existing generating plant, ... as shown in [Table 8].”

To understand the influence of natural gas prices, Fig. 2 (based on US EIA, 2014, US EIA, 2015, and updated with information from Federal Reserve Bank of St. Louis, 2020) presents three US natural gas price series: (1) interpolated monthly Texas natural gas prices (from annual data) for electric utilities from 1970 to 2012 from the US EIA’s SEDS for Texas; (2) the monthly US natural gas wellhead price from 1977 to 2014; and (3) monthly Henry-Hub spot-market prices in Louisiana from 1994 to 2020. Although most CCGT generators have long-term fuel supply contracts, the opportunity cost of using their contracted natural gas for electricity generation is the spot price. For this reason, this price is used to convert the price of natural gas into the price of electricity.

Comparing these results with the calculations in Tables 1 and 7, new nuclear was competitive with natural gas CCGT between 2000 and 2008. However, new nuclear cannot compete with natural gas prices below \$4/MMBTU (Table 9) in deregulated markets

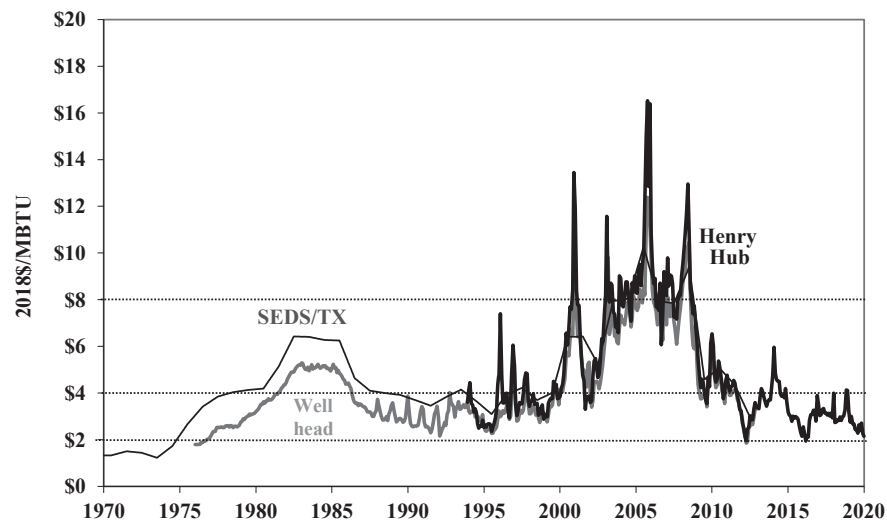


Fig. 2 Natural gas prices, 2018 US\$/MMBTU, 1970–2020. Data from US EIA (Energy Information Administration) (2014) *State Energy Data System (SEDS): 2013 Updates by Energy Source*. US EIA. <http://www.eia.gov/state/seds/seds-data-fuel.cfm?sid=US>; US EIA (Energy Information Administration) (2015) *Monthly U.S. Natural Gas Wellhead Price*. US EIA. <http://www.eia.gov/dnav/ng/hist/n9190us3m.htm>; Federal Reserve Bank of St. Louis (2020) *Henry Hub Natural Gas Spot Price*. <https://fred.stlouisfed.org/series/DHHNGSP>.

Table 9 Levelized cost for new natural gas generation, 2018 US\$, $r = 5\%$.

Combined cycle gas turbine (CCGT)		CCGT	CCGT	CCGT	CCGT
Levelized cost (2018 dollars)	$r =$	5.0%	5.0%	5.0%	5.0%
Fuel price	\$/MMBTU	\$2.00	\$4.00	\$6.00	\$8.00
Levelized capital cost (fixed)	\$/MWh	\$7.45	\$7.45	\$7.45	\$7.45
Levelized O&M cost (variable)	\$/MWh	\$3.44	\$3.44	\$3.44	\$3.44
Levelized fuel (variable)	\$/MWh	\$12.40	\$24.80	\$37.20	\$49.60
Levelized fuel CO ₂ fee (variable)	\$/MWh	\$9.89	\$9.89	\$9.89	\$9.89
Levelized variable cost	\$/MWh	\$15.84	\$28.24	\$40.64	\$53.04
Levelized cost without waste fee	\$/MWh	\$23.29	\$35.69	\$48.09	\$60.49
Levelized cost with waste fee	\$/MWh	\$33.18	\$45.58	\$57.98	\$70.38

Based on calculations in Table 7.

in the United States: it is unlikely that revenues would cover the fixed costs of NPP construction. Even if there were carbon taxes of \$30/tCO₂, new nuclear would not be competitive outside of China and Korea. If existing nuclear O&M and fuel costs are similar to new nuclear O&M and fuel costs (i.e., construction costs have been paid off for plants older than 30 years), existing NPPs could be competitive with CCGT with natural gas prices above \$4/MMBTU. However, with natural gas prices bouncing between \$2 and \$4/MMBTU, it would be difficult to justify investment in upgrades, uprates, or life extensions without the certainty that these investments could be recovered. On the other hand, natural gas prices have generally been twice as high in Europe and three times as high in Japan, see Knoema (2019). Therefore, it is more likely that incremental investments and new nuclear power could be justified in Continental Europe assuming rising natural gas prices and carbon taxes.

Finally, the various prices in Tables 1, 7, and 9 can be compared to wholesale electricity prices in the United States. Fig. 3 (based on information from US EIA, 2020) gives an approximation of daily electricity prices across the United States. (Day-ahead prices are the prices bid into the electricity market for the next day; see Section 3, “Competition on the Transmission Grid: Regulated Tariffs Versus Market Pricing” in Rothwell 2021b.) Because these are monthly averages, the daily highs and the lows have been averaged out. Only part of the time will electricity generated by new nuclear, new wind, and new solar-PV be adequately compensated. As explained in US EIA (2020):

“Wholesale electricity prices at several major hubs were generally lower in 2019 than in 2018, except in Texas. Record-high electricity demand in the summer led to much higher 2019 wholesale electricity prices in the Electric Reliability Council of Texas (ERCOT) electricity market. Day-ahead, around-the-clock wholesale electricity prices averaged \$38 per MWh in ERCOT in 2019, up 13% from their 2018 average. At other representative hubs—such as those in the Independent System Operator of New England (ISO-NE), New York Independent System Operator (NYISO), and PJM [Pennsylvania-New Jersey-Maryland] Interconnection—annual average wholesale electricity prices were generally 15% to 30% lower than in 2018. Much of this decline in wholesale electricity prices was the result of lower natural gas prices in 2019.”

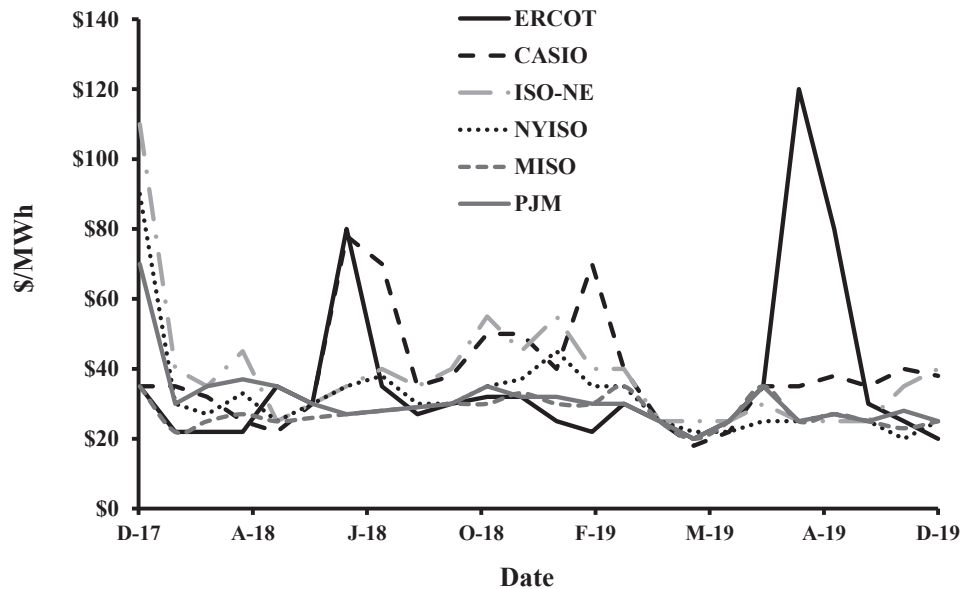


Fig. 3 Monthly average day-ahead prices in selected US electricity markets, 2018–2019. Data from US EIA (Energy Information Administration) (2020) *Wholesale Electricity Prices Were Generally Lower in 2019, Except in Texas*. (January 10). <https://www.eia.gov/todayinenergy/detail.php?id=42456#>.

Therefore, only natural gas with prices less than \$4/MMBTU without carbon taxes can be effectively compensated, as shown in Table 9. (Of course, this is because electricity generated by natural gas sets the price in most electricity markets in the United States.)

Observations on the past and the future of the nuclear power industry

As utilities shut down coal burning plants in response to political pressure, they tend to replace them with natural gas because in deregulated systems the market price is generally determined by the price of natural gas, so the lowest risk alternative to select is CCGT. Thus, competitiveness of nuclear power in the United States (and in some regions of the world) is a function of the price of natural gas. In areas of the world that must import much of their natural gas as liquefied natural gas and where NPP construction costs are relatively low (in China and in Korea), new nuclear power is competitive with alternatives, particularly if CO₂ fees are added. In Continental European countries and Japan that must import their natural gas, new nuclear power can be competitive, particularly if financing is available. In North America, the “fracking” of natural gas has led to an oversupply, a decline in electricity prices, and a difficult economic environment for both existing and new NPPs. The United Kingdom imports one-half of its natural gas as the North Sea reserves are being depleted. In response to this depletion and the aging of its NPP fleet, the United Kingdom seems to be willing to pay the highest prices for new NPPs.

While NPPs are closing in Europe and the United States, China and Russia are building much of the new NPP capacity. Hence, the center of the NPP industry is moving eastward from Europe and westward from North America and Japan. Russia is encouraging orders of its various models of VVERs with financing and the provision of both front and back end nuclear fuel services, e.g., the take-back of used nuclear fuel for reprocessing in Russia. The Chinese have been slower to export their nuclear units. However, China hopes to export its Hualong One (HPR1000) NPP, now being constructed in China. The Chinese are able to build their NPPs on time: (1) they started building their French EPRs in Nov 2009 and April 2010, and finished them in June 2018 and June 2019, surpassing the French in Finland (started in August 2005) and France (started in December 2007), and (2) they started building their Westinghouse AP1000s in March and December 2009 and finished them in June and August 2018, well ahead of the AP1000s at Vogtle (as yet incomplete) and Summer (since suspended). In both cases they passed ahead of the construction of the “First-of-a-Kind” (FOAK) units in home countries, showing the path to completion for Western Europe and the United States. Areva announced its reorganization in June 2016 and Westinghouse filed for bankruptcy in March 2017.

The US Government (USG) through the US Department of Energy (US DOE) tried to encourage the construction of NPPs in the United States in the early 2000s. US DOE engaged in a series of programs to provide incentives for new nuclear investments. It was envisaged that these programs would result in new NPPs operating in the United States by 2010 (Johnson, 2002, p. 2): “Goals: [1] Orders for one or more new nuclear plants by 2005 [and 2] Operation of new nuclear power plants by 2010.” However, electric utilities did not order new NPPs until after the passage of the US Energy Policy Act of 2005.

Three provisions in the 2005 Act promoted new nuclear capacity construction: production tax credits (PTCs), a standby support program (SSP), and loan guarantees (see Rothwell, 2016, Table 2.5). The subsidies were disproportionately granted to the first half-

dozen plants built and only to those that applied for Construction and Operating Licenses (COLs) before 31 December 2008. Within this group, there was a set of four AP1000s: Vogtle Units 3 and 4 (started in 2011), and Summer Units 2 and 3 (started in 2012). The FOAK units do not necessarily refer to the very “first” unit(s), but the set of first units over which design and licensing costs can be recovered, which justifies the construction of dedicated manufacturing facilities. While the PTC program was intended to support new nuclear operation (by means of tax reductions, if the units were in commercial operation by January 2021) and the SSP was intended to stabilize the new nuclear licensing process (by means of financial support during unforeseen regulatory delays), the US DOE loan guarantee program was intended to promote carbon-free emissions in electricity production from low-carbon energy technologies, including new nuclear capacity. Under this program, the total loan guarantee amount was limited to congressional appropriations. The borrower must pay fees for these loan guarantees that are similar to those that international financial markets would charge for similar loan guarantees.

The Energy Policy Act of 2005 plus various appropriations bills set the total amount of the loan guarantee program for advanced nuclear plants at \$18.5B. The owner-investors in Vogtle Units 3 and 4 were provisionally offered a loan guarantee of \$8.3B. After 4 years of negotiations, in February 2014, the US DOE awarded Vogtle co-owners Georgia Power and Oglethorpe Power with a \$6.5B loan guarantee and with a loan guarantee of \$1.8B to the remaining co-owner, the Municipal Electric Authority of Georgia. (US DOE provided an additional \$3.7B loan guarantee for Vogtle in March 2019, for a total of \$12B.) To follow the progress at Vogtle, see ANS (2020).

The US DOE gave three other plants positive initial reviews: Summer Units 2 and 3 (AP1000s) with South Carolina Electric & Gas (SCE&G), Calvert Cliffs Unit 3 (a single EPR) with Constellation in Maryland, and South Texas Project Units 3 and 4 (ABWRs) with NRG Energy in Texas. Even with loan guarantees, the builders of new nuclear would be required to convince Wall Street that investing in nuclear generation did not involve more risk than similar investments. In 2011 Moody's (2011) wrote (referring to SCE&G, and updating its 2007, 2008, and 2009 reports), “Today’s downgrade of SCE&G’s senior unsecured and [SCE&G] to Baa2 ... consider[ing] the heightened risk associated with a large nuclear construction program extending through 2019 that is expected to be about 50% debt financed and will pressure future financial metrics.” None of these projects received a US DOE loan guarantee. Until Vogtle 3&4 are finished, it is unlikely that the USG will subsidize large NPPs. US DOE has turned its attention to supporting Small Modular Reactors (SMRs); see Rothwell (2016, pp. 91–100), NEA (2016), Energy Strategies (2019), and Ingersoll (2021).

One of the leading SMR contenders is NuScale Power with a 12-module SMR where each module has a 60-MW (gross, 57 MW net) integrated PWR with an off-the-shelf turbine-generator and condenser. Table 10 (based on NuScale Power, 2015) reports the list price of the first two modules plus the balance of plant (BOP). Note, “The amounts shown for the individual module costs for modules 3 through 12 do not include any attributable installation labor ...” and “The lead time for ordering modules is approximately 5 years.” Unless most of the modules are installed, the NuScale SMR (as estimated in Table 10) might not be competitive with larger NPPs. Further, before seeing them in operation, it is difficult to predict their O&M costs. Therefore, it might be too early to understand the economics of SMR construction and operation.

Depending on the country and its policies regarding clean energy, nuclear power can be competitive. The low price of natural gas in the United States greatly limits the potential for new, large baseload plants due to the high initial capital investment, long construction times, and high financing costs. Although nuclear plants are a long-term investment, 60 years and possibly longer, levelized cost analysis using a single discount rate does not give credit to such investments. Some countries and states provide clean air credits for renewable energy projects, but few provide similar credits to nuclear plants. At present, China and Russia are expanding their nuclear fleets, but elsewhere, as in Europe and the United States, nuclear plants are being retired and few new plants are being built. For this situation to be reversed, new policies, designs, and construction techniques must be developed, such as inexpensive SMRs, to improve the competitiveness of nuclear with other sources of electricity.

Table 10 NuScale Power Small Modular Reactor Cost by Reactor Module.

<i>Systems installed</i>	<i>Cost</i>	<i>Net plant output (MWe)</i>	<i>Gross plant cost (kWe)</i>
All BOP plus Modules 1&2	\$1,800,000,000	95	\$18,947
Module 3	\$110,000,000	143	\$13,404
Module 4	\$110,000,000	190	\$10,632
Module 5	\$110,000,000	238	\$8968
Module 6	\$110,000,000	285	\$7860
Module 7	\$110,000,000	333	\$7068
Module 8	\$110,000,000	380	\$6474
Module 9	\$110,000,000	428	\$6012
Module 10	\$110,000,000	475	\$5642
Module 11	\$110,000,000	523	\$5340
Module 12	\$110,000,000	570	\$5088
Total	\$2,900,000,000	\$5088	

Based on NuScale Power (2015) Attachment 1: Indicative, Incremental Costs for Discontinuous Installation of NuScale Power Modules. LO-0715-16002, submitted to the US NRC (July 14). <https://www.nrc.gov/docs/ML1520/ML15203B306.pdf>.

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See Also: Pros and Cons of Commercial Reactor Designs: Part 2—Current Status and Future Trends in the World Nuclear-Power Industry and Technical Considerations of Nuclear-Power Reactors.

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How Innovative New Reactors Could Improve Public Acceptance

Rachel Slaybaugh^a, ^a Department of Nuclear Engineering, University of California, Berkeley, Berkeley, CA, United States

Suzanne Baker^b, ^b Nuclear Engineering and Radiological Sciences Department, University of Michigan, Ann Arbor, MI, United States

Jessica Lovering^c, ^c Good Energy Collective, Brentwood, MD, United States

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Abbreviations

AR	Advanced reactor
CCS	Carbon capture and sequestration
DOE	US Department of Energy
EIA	US Energy Information Administration
EPZ	Emergency planning zone
GCR	Gas-cooled reactor
HALEU	High-assay low-enriched uranium
HTGR	High-temperature gas reactor
LEU	Low-enriched uranium
LFR	Lead-cooled fast reactor
LULU	Locally unwanted land use
LWR	Light water reactor
MSR	Molten salt reactor
NEIMA	Nuclear Energy Innovation and Modernization Act
NRC	Nuclear Regulatory Commission
O&M	Operations and maintenance
PWR	Pressurized water reactor
SFR	Sodium fast reactor
TRISO	Tri-structural isotropic fuel
UAMPS	Utah Associated Municipal Power Systems

Introduction

Society's growing concern about climate change, desire to move more people out of energy poverty, and aim to bring equity to technology deployment have created an opportunity for advanced reactors to contribute to the global clean energy transition. New,

innovative reactor designs have the potential to address problems across a multitude of facets: enabling faster and more resource efficient decarbonization by balancing out intermittent resources and/or providing heat for industrial processes (Jenkins et al., 2018); improving the economic case for deep decarbonization; offering versatile products for more markets, which can help bring more people out of energy poverty; and improving social license by integrating social justice considerations, mitigating concerns about used fuel, and simplifying safety systems.

The reactor designs that are being developed in a serious way in the early 2020s are not entirely new ideas as the basic concepts generally originated in the 1950s, but the potential for these designs to be broadly commercially deployed is new. The combination of societal need and technology development has created a fresh opportunity for the nuclear industry to help solve some of society's toughest challenges. This chapter examines the case for advanced reactors (ARs), summarizes the current state of development, and assesses how technical and social systems must be considered collectively for the best outcomes.

To realize the potential of these new reactors, however, there is a strong need for innovation across the technology space, within social frames, in organizational structures, and in cultivating leadership. The current state of the nuclear industry in the United States and Western Europe is mixed, with new builds mired in cost overruns and schedule delays (McMahon, 2018); and light water reactors (LWRs) are shutting down early in tough-competition markets, taking valuable clean energy off of the grid (<http://www.beyondnuclear.org/reactors-are-closing/>). It is also clear that the nuclear industry needs to think and act differently regarding equity of access and participation across boundaries of race, gender, and more (Turner et al., 2020). If the nuclear industry is serious about getting new outcomes, they must do things differently than they have in the past.

Fortunately, the industry can thoughtfully engage in developing new approaches and structures for nuclear energy deployment as the next decade provides a unique opportunity to execute. New companies have formed and are growing, new leaders are emerging, the regulatory environment is being defined for non-LWRs, the US government is providing funding in new and creative ways for ideas to move from the laboratory up the commercialization ladder, conversations around international engagement are changing, and new supply chains are being envisioned and the groundwork for them is being laid. All of this is of course a snapshot in time and some of it will change by the time you read this chapter.

There are many reactor designs to choose from, with different types using different technical approaches to resolve the challenges the designers think are most important. The answer to which design is best depends very much on the value judgment of what is most important, or which combination of things are most needed. Is it the best economics? Is it balancing renewables on the grid? Is it limiting and eliminating nuclear waste? Is it providing high-temperature heat? Each community and country must choose for themselves what they value since the answer is unlikely to be the same everywhere. This chapter endeavors to provide some framing for how these choices might be made.

The case for innovative new reactors—What problems could we solve?

Society is facing climate change, economic and racial disparity in access to energy and quality of life, and health impacts from air pollution. Nuclear energy is already helping to mitigate some of these problems as it is a major source of clean electricity globally and has the additional benefit of being a firm resource (Jenkins et al., 2018) ([https://www.iea.org/data-and-statistics?country=WORLD&fuel=Energy supply&indicator=Electricity generation by source](https://www.iea.org/data-and-statistics?country=WORLD&fuel=Energy%20supply&indicator=Electricity%20generation%20by%20source)). New reactor designs that are available today, termed Generation III+, have enhanced safety features compared to the existing Generation II and III reactors, but their construction costs and timelines have varied globally and not all electricity grids and markets need the 1 GWe-sized, baseload power plants that are Gen III+ (CleanTech Catalyst Ltd., 2018; Sah et al., 2018).

The combined need for clean, firm energy resources and the limitations of Gen II to III+ LWRs create the need for nuclear energy systems that can fill the gaps and improve upon specific challenges—though there is certainly still a strong need for the existing reactors everywhere and new Gen III+ plants in many parts of the world. Advanced reactors are poised to do just this, and the remainder of this section describes the existing barriers and how ARs may overcome them.

Market fit

Advanced reactors have the potential to improve the “buy-ability” of nuclear systems. The often-determining consideration for what electricity sources are constructed is economic, where the economics are set by overnight capital cost, construction time and timeline certainty, and cost of operations and maintenance (Rothwell, 2021). A second prerequisite for market fit is generating the amount of electricity needed for a particular market. In the United States, due to the fragmented electricity market spurred by deregulation, none but the largest utilities can currently afford to build nuclear energy projects. This is also true in many countries. Smaller reactors with factory fabrication and certain delivery dates could open up entirely new markets (Ingersoll, 2021). An important area for innovation is in reducing the cost and risk to build and operate new nuclear plants.

Overnight capital cost

Generation III+ plants are being built in several parts of the world at widely varying cost. In Asia and the Middle East, the costs are fairly reasonable and in some places are even competitive with other generation sources. In the United States and Western Europe, however, the costs are shockingly high, multiple times higher than the predicted project prices, and much higher than other generation choices. It is unlikely that new large LWRs will be built in many places because the costs are so high and so unpredictable.

An item that can exacerbate high and uncertain capital cost is a long and uncertain project timeline. Most nuclear projects are financed through loans. Riskier projects garner higher interest rates. Further, the longer the project takes, the more time before revenue is generated and the more interest is paid on the loan. Thus, when projects have high variability of cost compounded by a longer than planned schedule, finance costs can become significant. Increasing predictability of cost and timeline would reduce finance cost, and reducing time until revenue generation would have an additional benefit.

Advanced reactors are projected to cost slightly less to significantly less than Gen III+ reactors, depending on reactor type. The Energy Options Network (EON) worked with eight AR companies to do a rigorous cost analysis. EON used the plant cost accounting framework developed by the Generation IV International Forum (Stanculescu, 2021) as the basis for a cost model for n-th of a kind nuclear plants (Energy Options Network, 2017). This work found ARs could cost as much as \$5885/kW to as little as \$3782/kW (in 2016 \$). This range is significantly lower than what is being constructed currently in the United States and Western Europe and lower than what is projected by the US Energy Information Administration (EIA) for Gen III+ plants. It is noteworthy, however, that the projected lowest-cost reactors are still more expensive than projections for many other new generation forms and more expensive than the Gen III+ reactors being built in parts of Asia and the Middle East.

ARs aim to cut overnight costs using a few different approaches. Many are taking the approach of factory fabrication and design standardization, avoiding the redesign and bespoke work that drives up costs of projects now and increasing predictability of schedule and cost. Factory fabrication will allow for part standardization and to take advantage of new techniques like additive manufacturing for complex parts. Some are operating at lower or no coolant pressure, which means components can be thinner, lighter, and less expensive. Inherent safety features can lessen the complexity of design and number of components, and reduce the size and weight of the concrete installations needed. Smaller reactors simplify things. Overall, these designs are working very hard to reduce cost and construction schedule, and increase predictability.

Operations and maintenance costs

Once a reactor is constructed and operating, there are costs to keep that reactor running. Those costs include fuel, staff, supplies, periodic improvements, etc. We typically break the cost of electricity into levelized capital costs, which recoups the cost of construction and any capital improvements over time; variable operations and maintenance (O&M), which includes variable costs like fuel; and fixed O&M, which includes costs that don't change with electricity output, like staff salaries. The fixed costs are important as they must be paid whether electricity is being sold or not—where no sale means no income.

Existing nuclear reactors have very high fixed O&M compared to other generation sources, and this has become a major challenge in the United States. Ten nuclear reactors closed early between 2013 and April 2020 because the costs to operate them are too high (<http://www.beyondnuclear.org/reactors-are-closing/>). O&M costs for nuclear are more than 50% higher than the costs for hydro and fossil steam, and about three times the costs for natural gas (https://www.eia.gov/electricity/annual/html/epa_08_04.html). The EIA's projected O&M costs for new assets coming online after 2025 slates Gen III+ plants with the highest fixed O&M costs, with only geothermal and biomass having fixed costs that are nearly as high. This gives the concern that the financial challenges facing the current fleet will still be a concern for Gen III+ reactors.

What about the next generation of reactors? Since we have little or no operating experience with ARs, all O&M estimates are based on best-guess values largely based on what we know about operating large LWRs (Buongiorno et al., 2018; Energy Options Network, 2017). Some reactors have the potential to have competitive O&M costs, and others are currently projected to face the same challenges as existing reactors.

This is a key area where new technology could make a difference. Many industries are dramatically reducing O&M costs by using predictive maintenance, advanced control systems, and asset performance management. These ideas use modeling and simulation, real time data collected from the assets, and artificial intelligence to accurately determine the operational state of individual components, a whole plant, or a fleet of plants. This information can automate operations, optimize maintenance, and reduce staff—all of which can dramatically reduce cost and make reliability easier to attain. These types of technologies will be especially important for new reactors since we do not have decades of operating experience. Simpler plant designs and improvements in how we can design plants to be easy to defend may allow for significant reduction in security forces. With all of these ideas, some reactors designs may be able to have very low staffing numbers, and microreactors may be able to be fully autonomous for long periods of time (Slaybaugh et al., 2019).

Multiple products for multiple markets

The reactors available to purchase today generate approximately 1 GWe (1 Giga-watts of electricity), require a substantial water resource for cooling, and a specifically-trained workforce for construction and operation. It may seem obvious, but not all locations that need electricity need 1 GW of electricity in one large installation, have a lot of water available for cooling, or have the desire to build the workforce and infrastructure needed. One of the major benefits of ARs is that a variety of reactor sizes and types will be available and able to meet the needs of more locations.

Some places may need less than 1 GWe because of limitations of the grid for transmission of electricity or overall electrical needs. Natural gas plants average about 500 MWe, and the capacities of coal plants that might be retired range from ~50 to ~800 MWe. With these plant sizes, it may not be possible to replace fossil plants with nuclear plants if the nuclear plant is much bigger than the system it would be replacing. Another barrier is that some countries or some utilities cannot finance a large reactor. Being able to buy a nuclear power plant made of a number of small reactors sequentially installed would decrease the cost per unit installation such that a new system could be affordable.

The smallest AR may be as small as 1 MWe, with many intermediate increments: 10, 50, 300 MWe, etc. and may have other benefits in addition to size choice. Some new designs don't require water for cooling, which dramatically expands which locations are appropriate for installation. Many new reactors don't require refueling, a complex operation requiring a surge in highly-trained personnel, for 5 to 30 years. Others need very few operational staff and can be buried underground to make physical protection much simpler. The range and versatility of products allows communities to find an AR that works for them, should they choose to benefit from nuclear energy.

Nuclear energy is a firm resource that does not emit carbon, which will be essential for a decarbonized grid. The considerations about construction cost, operating cost, water use, operational complexity and workforce needs, and generation size are important ones. However, this remains to be proven and the onus is on the nuclear industry to ensure they execute the vision they created.

Grid integration and process heat

Electricity generation sources are not islands; they integrate with all the other sources on the electricity grid and interface with the grid itself because, in the end, the supply for electricity must meet the demand. Thus, choices about generation sources need to consider all of those factors collectively. Advanced reactors may make it easier and cheaper for us to meet demand reliably in a very low carbon economy.

To be able to meet the demand, it is helpful for supply to be predictable and controllable. However, society also has very strong goals around not emitting air-pollution. The intersection of these goals can be difficult because some low-carbon resources are intermittent, such as wind, solar, and hydropower without large reservoirs. That is, we can't control when the wind blows, the sun shines, or the rain comes down. These sources therefore vary over timescales of seconds to seasons.

Historically, balancing intermittent resources has not been too concerning because there were not many of them on the grid, but that is changing. To reach our emissions goals, however, we need more renewable, intermittent resources, and many states in the United States and countries around the world have been adding large quantities of intermittent resources to meet their carbon and air pollution goals. We expect this trend to continue and accelerate, so we will need good mechanisms to bring on more and more intermittent resources while reliably meeting demand.

We can balance some of this variability with good forecasting and controls combined with balancing or "fast-burst" resources such as batteries, simple cycle gas turbines (with carbon capture and sequestration (CCS)), demand response, or flexible demand (Ela et al., 2013). These balancing resources have a response time that matches the shortest variation time of intermittent renewable resources. Coal, combined cycle natural gas, and nuclear do not vary their output too quickly, so are not well poised to balance the quickest changes in variable resource output (Gonzalez-Salazar et al., 2018).

For the multi-hour to seasonal scale of variation, we need "firm" low-carbon resources that can be counted on. This is where nuclear, coal and natural gas with CCS, large hydro, geothermal, and biomass come in. We need these technologies to operate flexibly enough to balance out naturally-varying resources over longer time scales. To date, combined cycle natural gas has largely been used as the variable firm operational source to meet varying demand. Nuclear and coal are operated in a "baseload" frame: they run at constant power unless they need to be shut down for maintenance or refueling. If we are serious about our decarbonization goals, this will need to change because these baseload resources will need to become variable in a controlled fashion.

Existing LWRs can vary their electrical output to some extent. They can directly vary their electrical output using reactor power changes. There are physics limitations on how fast, how much, and how often this strategy can be employed though, even for plants that were built with the intention to "load follow" or vary their output (OECD, NEA, 2011). Larger, more rapid power changes can be achieved by temporarily directing the steam away from the generator, but this is not a desirable solution as it is wasting energy (Jones, 2011).

Advanced nuclear reactors offer a few pathways to higher levels of variability that may enable really large deployments of intermittent resources. This is an essential feature of advanced reactors. Some reactor designs are intended to directly vary their electrical output. Other designs are intended to run constantly at full power and have a thermal storage system attached that can vary electrical output. When demand is low, heat will be directed to the storage system rather than the electrical generator. When demand is high, the stored heat can be used in conjunction with the reactor to produce electricity. Finally, many ARs are designed to operate at higher temperatures. They can divert the produced heat directly for industrial processes, or to generate co-products like hydrogen or synthetic fuels in a manner similar to the thermal storage approach. Which approach will be best will depend on the specific market needs, the reactor technology, and many implementation details that will dictate cost tradeoffs (Lucid Catalyst, 2020; Clean Energy Ministerial, 2020). The main point is that there are many pathways to flexible nuclear reactors that can match the needs of a very low carbon grid.

To fully decarbonize the global economy, we need to focus on more than electricity alone since greenhouse gas emissions come from many sectors that cannot be electrified (<https://www.epa.gov/ghgemissions/global-greenhouse-gas-emissions-data>). The EPA notes that the 21% of global emissions associated with industry "primarily involve fossil fuels burned on site at facilities for energy." The heat needed for industrial processes needs to be replaced by non-emitting sources.

The temperatures needed for industrial process heat range from 50 °C to 2200 °C. A combination of nuclear with solar thermal and geothermal energy in the locations where those options are viable has a strong potential to fill some of these needs, with technology match depending on exact temperature needed, heat transfer modes, quality of heat, etc. Light water reactors produce steam a little under 300 °C; concentrated solar plants typically operate around 400 °C, though this range is expanding; and geothermal plants typically operate at lower temperatures but vary by installation detail (<https://www.eia.gov/energyexplained/solar/solar-thermal-power-plants.php>). As detailed below, advanced nuclear plants are slated to reach up to 1000 °C, so they could provide heat to a large number of applications (McMillan et al., 2016).

Beyond industrial heat, nuclear reactors are particularly well suited for other industrial applications like hydrogen production, desalination, and district heating. The Department of Energy (DOE) has a program that is researching flexible use of small modular light water reactor modules, whether for hydrogen generation, process heat, or load following. Such a program will help prove out the details of some of these ideas so industry and communities will be able to have the details they need for decision making.

In many countries, intermittent electricity resources are becoming more prevalent and to meet our decarbonization goals this trend needs to increase over time. Further, sectors like heavy industry need to be decarbonized. Many studies show that the availability of flexible and high-temperature reactors can help decarbonize the global economy faster and at lower cost because they can balance renewables and provide heat (Jenkins et al., 2018; Lucid Catalyst, 2020; <https://inl.gov/article/advanced-reactors/>).

Social license to operate

While existing nuclear power plants typically have strong public support, there has long been ardent opposition to nuclear facilities across all aspects of the technology, particularly mining and milling of uranium as well as waste disposal sites. Certain characteristics of advanced reactor concepts could improve social license to operate—the ongoing acceptance or approval of a project from the relevant stakeholders—as detailed below, but other challenges cannot be mitigated with technology alone and will need commensurate public policy to ensure positive stakeholder experiences with technology adoption. Section “What’s needed in tandem with technological innovation to improve outcomes?” details these ongoing challenges.

Improving the social license for nuclear energy might start with expanding the market to new customers like smaller municipal utilities, rural electric cooperatives, off-grid communities, and a larger range of industrial users. This would allow for gaining social acceptance by providing local benefits such as zero-emissions electricity generation and high-paying jobs. The ability to have local control over a power plant may facilitate stronger social license.

Some of the greatest environmental justice issues related to nuclear energy concern the fuel cycle. Significant work still needs to be done to address legacy pollution issues from nuclear weapons production and testing as well as uranium mining and to make progress on nuclear waste disposal. Some AR designs are looking to address fuel cycle concerns by recycling used fuel, which will reduce mining needs and simplify disposal, resolving some of the largest objections to the nuclear industry. While fuels vary widely across reactor technologies, some have the potential to use fuel more efficiently and even burn nuclear waste as fuel. These characteristics could reduce the need for fresh fuel mining, reduce the volume of spent fuel produced, and extend the time between reactor refueling (meaning less transport and handling of fuel in local communities).

The simple, “walkaway” safe nature of many advanced reactors also has the potential to strengthen social license. Power plants that rely on inherent and passive safety features may have an easier time gaining the trust of potential host communities than those designs with complex engineered safety systems. The promise of no off-site consequences in the event of an accident, along with a reduced need for back-up power and emergency cooling reserves, may also reduce dread. Yet these safety attributes will need to be proven to host communities before they provide meaningful reassurance.

Although not intrinsic to any advanced reactor concept, the makeup and structure of the developers has also changed and may improve social license. Originally, nuclear technologies came out of the military and so did much of the workforce. While many advanced reactor concepts were also born from government programs, their modern commercial development is coming from new companies that look very different: younger, more diverse, more focused on equity. Many of these reactor companies were spun out of university programs, raising a mix of private and government investment, and their workers are driven by concern for climate change and energy poverty. Such changes in the fabric of technology developers may not change the public’s minds about nuclear technology on its own, but these new companies may also come with different models for how community engagement, government interaction, and how they do business more generally.

Advanced reactors are quite different from existing reactor designs. These technical differences mean they may be more economic, fit into more markets, help us achieve deep decarbonization faster and in hard sectors, and improve social license. Changes to who is building and designing these reactors may also improve social license. In all cases, we must execute the vision thoughtfully and diligently for these possibilities to be realized.

Overview of new innovative reactors under development

There are many AR designs currently in development, each with a different combination of materials, sizes, and operating regimes, that may be better suited to different purposes and markets. Commensurately, there are many companies, new and old, working to develop what they think is the best reactor. No two designs are the same, though there are some general categories. The major variables that people select from to comprise their reactor design include the following, noting that this list is likely not exhaustive (Milko et al., 2018; Freed et al., 2015):

- *Electrical output:* microreactors may be 1–10 MWe, small modular reactors are often 50–300 MWe, and large reactors are on the order of 1 GWe.
- *Power conversion system:* most ARs plan to use a steam Rankine cycle, but some are planning for CO₂ Brayton cycles.
- *Heat transfer medium:* light (regular) water, sodium, gas (usually He), lead (or lead-bismuth alloy), molten salt (fluorides, chlorides, or nitrides), heat pipes, supercritical water, organics, and supercritical CO₂ are all options.

- *Fission operating regime*: fission reactors use neutrons to split unstable isotopes. The energy of the neutron causing fission can be near the energy they are born, “fast” (on the order of MeV), or in thermal equilibrium with the system, “thermal” (on the order of meV). Whether a neutron is fast or thermal determines which isotopes it can fission.
- *Fuel form*: oxides, nitrides, metal, tri-structural isotropic (TRISO), molten salt, silicon carbide clad, metal clad, are all options.
- *Fuel enrichment*: some reactors will operate with the enrichment level currently used, 3–5 wt% U-235, while others will use nearly 20 wt%, and many somewhere in between.
- *Operating Temperature*: LWRs have a core outlet temperature of $\sim 285\text{--}330^\circ\text{C}$, while many non-LWRs will operate around 650°C or even 1000°C .
- *Operating Pressure*: some reactors operate at atmospheric pressure, many operate in the range of 7–17 MPa, while the highest pressure systems are 25 MPa.

These different options are combined in different ways. **Table 1** provides a non-exhaustive list of some key features of different reactor categories. The World Nuclear Association provides a good overview of many different advanced reactor designs, details, projects, and collaborations (<https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/advanced-nuclear-power-reactors.aspx>).

With all of these options, how does a community think about what to choose? Each reactor type fills a particular market purpose. However, each also has drawbacks or challenges. Which reactor is the best choice will depend on the needs of a particular community, market, or country. Some brief points on several reactor types being pursued are given below, followed by a more in-depth look at two particular reactor designs. Note that the following discussion does not cover all reactor designs or design variations.

Small modular LWRs use the same basic technology as large LWRs, but their smaller size could give them smaller financial risks and simplify safety systems. It is possible to manufacture modules of the reactor in a factory and ship them to site, which should provide an economic benefit via standardization and economy of mass production (Ingersoll, 2021). Each individual module would be lower total cost (even if the cost per MW is higher), which should make it easier to finance each module. The smaller size makes it easier to match load if there is a collection of modules as some modules can be turned off or on while others are ramped a little bit. Finally, the smaller core size gives safety benefits just because each reactor has less material, so the maximum accident is less impactful. The main drawback compared to large LWRs is that the smaller size reduces the economy of scale, which may not compensate for the economy of numbers, ease of financing, and smaller maximum accident impact.

The main benefits of gas-cooled reactors are high temperatures that can be used for industrial processes and extreme safety characteristics (the VHTR and GCR have similar characteristics). Because they use a gas coolant and TRISO fuel (Helmreich, 2021; Gerczak, 2021) the maximum accident the plant could experience is limited, with no scenarios causing fuel melt. GCRs have been built and operated commercially, so we have more operating experience and data with them than many with other AR types. The graphite moderator used in this reactor core causes the power density to be low, so GCRs tend to have cost challenges because they are large compared to how much power they generate. A second drawback is that the TRISO fuel form cannot be recycled and GCRs are not set up to consume used fuel.

The main benefits of SFRs (Heidet, 2021) are that the fast spectrum can be used to increase uranium utilization by two-orders of magnitude as well as to consume nuclear waste generated in thermal reactors. These reactors have moderately-high temperatures that could be used for industrial purposes. SFRs have very strong neutronic safety features and benefit from operating at atmospheric pressure, but that must be balanced against the fact that sodium is highly pyrophoric which greatly complicates operations. An additional challenge is a relatively high radiation damage rate to structural materials. There is some experience in operating SFRs primarily in Russia and France. The United States has operated a number of experimental SFRs.

The LFR has similar benefits to the SFR (Alembreti, 2021). The temperature and pressure regimes are similar, and LFRs can also be used to efficiently utilize uranium and to consume waste. LFRs also have strong safety features. A main reason people are

Table 1 Physical properties of several advanced reactor categories.

Reactor	Neutron spectrum	Coolant	Core outlet temperature ($^\circ\text{C}$)	Pressure (MPa)	Size (MWe)
Current LWRs	Thermal	Water	285–330	15.5 (PWR) 7 (BWR)	1000–1200
Very-high-temperature reactor (VHTR)	Thermal	Helium	900–1000	7–17	250–300
Sodium-cooled fast reactor (SFR)	Fast	Sodium	500–550	~ 0.1 (1 atm)	50–150 300–1500 600–1500
Supercritical-water-cooled reactor (SCWR)	Thermal/fast	Water	510–625	25	300–700 1000–1500
Gas-cooled reactor (GCR)	Thermal	Helium	850	9	1200
Lead-cooled fast reactor (LFR)	Fast	Lead	480–570	~ 0.1 (1 atm)	20–180 300–1200 600–1000
Molten salt reactor (MSR)	Thermal/fast	Salts (F, Cl, or N based)	700–800	< 5	1000
Microreactor	Thermal/fast	Heat pipes/helium			1–10

interested in lead is that it removes the pyrophoric consideration when compared with SFRs. However, the lead introduces major corrosion challenges that have not yet been reliably resolved, can introduce issues of freezing, and the production of significant amounts of polonium introduces a radiological hazard for operators. Russia has experience operating LFRs.

MSRs have more variations than the other reactor types (Ignatiev, 2021). The main benefits across all MSRs are potential to provide process heat and to load follow either directly or with secondary power conversion systems—the secondary salt coolants are good candidates for using in thermal storage; and very strong safety features—they cannot have melt-down accidents. Some designs have liquid fuel and some have solid fuel. Some are fast spectrum and could therefore burn waste. Different designs use different types of salt. For many MSRs there are challenges with corrosion and large operational radiation environments such that all maintenance will need to be conducted by robots. We have built one experimental MSR.

Microreactors have a whole chapter devoted to them (Lovering, 2021). Briefly, their main benefits are ability to fit into totally different markets, load following, and extreme safety. The different microreactor designs are quite different from one another, but their key characteristics are similar. For many microreactors the designs focus on simplicity of design and construction, which should ease deployment. The major question is whether the cost/MWe can be low enough and whether they will be able to achieve the very low staffing numbers they are targeting.

In this article we will describe two advanced reactor concepts that have engaged with the Nuclear Regulatory Commission (NRC) for review and approval in different capacities. These are indicative of the types of plants being considered in the future employing some of the attributes mentioned earlier.

NuScale: Small modular PWR

NuScale has designed a small modular Pressurized Water Reactor (PWR), and this technology may be one of the first new reactors deployed. The NuScale design uses the basic technology of PWRs: regular water coolant, low-enriched uranium (LEU) dioxide fuel, zirconium-alloy cladding, and operates at similar temperatures and pressures. However, the first NuScale reactor has many design innovations (<https://www.nuscalepower.com/technology>). For example, it uses the so-called integrated design in which all the primary reactor coolant components are within the pressure vessel and the primary coolant removes the fission energy by natural circulation. This design approach eliminates the primary pumps, valves, and external piping and vessels, thus increasing the plant's robustness and also reducing construction and maintenance costs. The reactor vessel is located below grade in a pool of water. The water serves multiple purposes, including passive containment cooling and decay heat removal. That is, the pool provides a heat sink with a capacity to absorb all the decay heat produced by a fully mature core for greater than 30 days.

Each unit is 60 MWe instead of the 1000 MWe of present day commercial reactors. The relatively small power per unit has important safety advantages (Ingersoll, 2021) and enables modular construction. Each NuScale reactor vessel is 2.7 m in diameter and 20 m tall, weighing 590 metric tons. The reactor module would be pre-fabricated, delivered by railcar, barge, or special trucks, and assembled on-site. The NuScale power plant construction time is expected to be much smaller than that of conventional LWRs.

NuScale is betting that the similarities with existing technology means they will be able to leverage a lot of existing supply chains and they will not need to get materials or fuel newly qualified, which can take a lot of time and money. Similarly, the knowledge base to keep nuclear materials safe from proliferation should apply to NuScale's design.

Historically, nuclear power plant construction has been site-customized, with everything built on site, and many items redesigned and customized for a specific plant installation. In the United States and Western Europe, this approach has been demonstrated to be very slow and expensive. NuScale's design will avoid this pitfall by having all the plant components manufactured at a factory and shipped to site for assembly—though the civil works will still be site-specific, which may prove to be a time and cost burden (<https://www.nuscalepower.com/technology/fabrication-and-assembly>). With a large number of plant orders, this factory approach should lower costs and improve learning and quality control.

As discussed, load following is likely to become increasingly important in a low-carbon electricity grid. The plan to install multiple units at one site gives granularity to power output. A "full pack" is twelve 60 MWe units, creating a 720 MWe plant. The individual modules can come on or offline in a ~1 day timeframe. For faster power changes, individual units can be ramped up or down and for very fast changes they can execute a steam bypass strategy. It's unclear whether these approaches will be economic—that will ultimately depend on the market and the operational costs. Instead of power change, heat could be diverted for industrial processes. How well this works and how financially viable this approach might be will be studied by the DOE.

Because the modules are independent they can be purchased individually and in series, which could make financing easier. The cost of an individual module will be much less than 12 at one time, so the total financial risk is lower which could enable better loan rates. The combination of lower capital outlays and better loan terms means smaller entities could more easily buy a reactor. Additional modules can be added after the first one or several are already operating and generating revenue, again reducing financial risk and making these easier to buy.

NuScale is betting that the similarities with existing LWR technology means their design will be easier to license and regulate because the NRC is familiar with LWRs. There were still major differences to be addressed, like using one control room for up to 12 modules and shrinking the emergency planning zone (EPZ) to the site boundary rather than 10 miles. Having one operating room will improve operational efficiency and reduce fixed O&M. The small unit size means natural convection can be used in emergencies to remove decay heat and the plants are very robust, so emergencies would never go beyond the site boundary. NuScale submitted their design certification application in January 2017 and the Final Safety Evaluation Report was completed and approved in September of 2020 (<https://www.nrc.gov/reactors/new-reactors/smr/nuscale/review-schedule.html>).

Not only is NuScale's the first new type of reactor to get a design certification, in principle it already has a customer. The Utah Associated Municipal Power Systems (UAMPS) is planning to site 12 modules at Idaho National Lab. DOE is providing matching funds for site preparation and the combined operating license application. At first the plan was for UAMPS to vote on the final notice to proceed for construction in 2023, with plans for the first NuScale plant to achieve operation by 2026, but by August of 2020 that plan was delayed to a 2030 operation date and made contingent upon DOE providing \$1.4 B to offset increasing costs (Cho, 2020; <https://www.nuscalepower.com/newsletter/nucleus-spring-2019/powering-the-next-generation-of-nuclear>). The delay is based on the projected power needs of the UAMPS communities, not because of changes in NuScale's trajectory. NuScale is doing customer engagement and communications in a more modern and forward thinking way, which may help them work through the timeline and cost issues with UAMPS.

X-energy: High-temperature gas reactor

X-energy is designing the Xe-100, a small, modular, helium-cooled, pebble bed reactor that is to generate 80 MWe, intended to be sited with up to 4 units for 320 MWe installations (<https://x-energy.com/reactors/xs-100>). They will use TRISO fuel that is enriched above 5% but below 20% U-235, called high-assay low-enriched uranium (HALEU), with a graphite moderator. The Xe-100 uses online refueling, has a 750 °C He outlet temperature, uses a steam power conversion cycle, and focuses on using approved materials as much as possible. They, too, will have an EPZ that does not extend beyond their site boundary. X-energy is counting on safety, modularity, high-temperatures, and fuels and materials that the industry has some experience with for their technology to be adopted. They are focused on designing a plant with strong focus on economics and deployment—something that can be factory manufactured and use existing knowledge, supply chains, and infrastructure as much as possible. They are also working on sophisticated operations and maintenance strategies for fast licensing and low-cost operations (https://arpa-e.energy.gov/sites/default/files/documents/files/GEMINA_Project_Descriptions_FINAL.pdf).

One of the potential benefits of gas reactors for a future grid is the high temperature heat. The heat in the form of He or steam (565 °C) enables generating electricity at a higher efficiency than in LWRs and could be directly used for industrial purposes. There is the option to add a secondary power conversion system that could be used for load following, which might be a good choice for this plant since the higher temperatures improve the efficiency compared to lower-temperature systems. The Xe-100 can refuel without shutting down, which eliminates downtime for refueling and should be an economic advantage.

While a more efficient power conversion could be attained if X-energy used the helium directly to drive a gas-turbine, by choosing a steam cycle they can use existing supply chains and rely on technologies the industry knows and trusts. X-energy has selected an ASME standards-compliant core barrel, steam generator, and reactor vessel, making it easier to use existing supply chains and fit into existing regulatory frameworks. This should reduce risk and time to deployment. X-energy has engaged in pre-licensing activities with the NRC, but has not submitted a license application (<https://www.nrc.gov/reactors/new-reactors/advanced/xs-100.html>). The Xe-100's will also keep their design small and modular so they can take advantage of the benefits of factory fabrication.

There is not a supply chain for HALEU, but the DOE is working to change that. X-energy has TRISO fuel manufacturing capability, so once a HALEU supply is secured they will have a secure fuel supply chain. There are several key benefits from using TRISO. There is significant experience with this fuel form, which means it is likely easier to qualify than many other new fuel forms. It is extremely safe, which creates the foundation of the very strong safety of gas reactors. However, the TRISO fuel form is comparatively bulky, and it is nearly impossible to reprocess it. This may be beneficial from a non-proliferation perspective and detrimental from the perspective of closing the fuel cycle. It remains to be seen how cost effective HALEU TRISO is since the low fuel density is the main driver of the low power density that challenges gas reactors economically.

Like NuScale, the Xe-100 will not need an EPZ. Gas reactors have low power density, high thermal inertia, and very robust fuel. These very strong safety features mean the reactor cannot melt down. Having a low EPZ means the reactor can be sited in more locations more easily, reduces emergency planning burden, and reduces staff required to manage accident response. Mostly, though, not ever needing to evacuate a community is strongly beneficial to gaining social license to operate.

Overall, X-energy is relying on the combination of these features to give them a competitive reactor: modular manufacturing, small size, high-temperature heat source, very long up times (high capacity factor), many proven components, and extreme safety. The design has yet to be certified, and a gas reactor is quite different from a light water reactor. There are questions about fuel supply and cost, and the overall economics remain to be proven.

A historic challenge that must be avoided for whatever reactor design moves forward is that the nuclear sector needs to be accurate in its time and cost estimates. That is, not only does the sector need to develop technologies that can be economically viable and adhere to reasonable timelines, they also need to be better at the projections they make. For example, In 2008 NuScale thought perhaps they could have a reactor online by 2015. They did not even submit a license application until 2017. While there were extenuating circumstances that caused this delay, the pattern has remained consistent across the sector in the United States: wildly inaccurate timelines and costs. These issues do not demonstrate that the industry is investible or a good partner. For nuclear energy to have a future, it has to be able to keep the promises that it makes.

What's needed in tandem with technological innovation to improve outcomes?

It is clear that advanced reactors offer many unique attributes that have the potential to alleviate many of the challenges with existing nuclear energy, including public opposition. Yet the success of these designs is not guaranteed based upon improved engineering. There is a host of complementary policies that need to be developed to facilitate equitable adoption of these novel technologies.

Along with technological innovation, there is an opportunity now to develop novel organizational structures, modern infrastructure and regulatory regimes, and new business models that will facilitate equitable adoption of advanced nuclear designs. Below is a summary of the areas that still require parallel innovations.

Fuel cycle

The United States has struggled to establish a holistic, sustainable, and closed fuel cycle that accounts for mining, uranium supply chains, fuel fabrication, waste management, and legacy nuclear materials—which includes clean up from the nuclear weapons program still ongoing at more than a dozen sites across the country (<https://www.energy.gov/em/mission/cleanup-sites>)—and the respective impact of these activities on communities. Environmental contamination issues across the fuel cycle have resulted in legitimate concerns about the viability and safety of nuclear energy. As a result, many states have explicit policies that ban the construction of new nuclear facilities until a nuclear waste solution is available (<https://www.ncsl.org/research/environment-and-natural-resources/states-restrictions-on-new-nuclear-power-facility.aspx>). One way forward could be to develop a national strategy aimed at addressing and expediting ongoing clean-up efforts and redesigning the US fuel cycle for a new generation of advanced reactors. The introduction of new fuel forms invites the possibility of creating a new, more sustainable fuel cycle that considers where and how materials are sourced and manufactured, and where those materials ultimately go after their use.

Workforce

The new suite of advanced reactor designs will look quite different from the LWRs we have now, and those differences offer an opportunity to rethink how the industry approaches problems as well as what problems need to be addressed. The industry needs to foster a new generation of workers that is creative and adaptable, ready to meet the complete set of complex challenges associated with nuclear energy, not just the technical subset. The workforce should reflect the diversity of the communities that will be hosting these advanced reactor projects. Going beyond engineering, this new workforce will need to be broader and more interdisciplinary; reactor designers, management, and anyone public-facing will need to understand the complex challenges of social license, environmental justice, and an evolving regulatory process. For example, employees at vendor companies who are working in community engagement and siting should have a broader background that could include but is not limited to: anthropology, geography, history, politics, communications, psychology, sociology, and science and technology studies. Management should work to integrate teams with both technical and non-technical backgrounds to address emerging challenges.

Regulation

Recent legislation signed into law in the United States, like the Nuclear Energy Innovation and Modernization Act (NEIMA), have made incremental progress on improving the regulatory process for advanced reactors. Specifically, NEIMA instituted a more equitable and transparent fee structure for applicants and directed the NRC to develop a technology-inclusive licensing framework (<https://www.nei.org/news/2019/president-signs-bill-to-modernize-nuclear>). Yet much more work needs to be done to streamline the licensing process for future owners and operators of advanced reactor projects, especially for smaller utilities and municipal customers. Some modular reactor developers are envisioning a deployment model that is closer to “plug-and-play” than in existing nuclear power plants, with much less on-site construction. To facilitate this adoption model, there will need to be parallel innovations to create licensing templates for potential customers to standardize and streamline the process. Without such improvements, the challenges of the licensing process will continue to privilege the largest of electric utilities.

Siting

Traditionally, the siting of nuclear power plants has followed a simple model: announce, defend, build. Considered a Locally Unwanted Land Use (LULU), strategies to site new large LWRs in the United States typically involve the expansion of existing nuclear facilities in rural and suburban areas rather than looking to build projects in totally new locations (<https://onlinelibrary.wiley.com/doi/abs/10.1111/j.1539-6924.2009.01262.x>). However, advanced reactors have the potential to dramatically expand feasible siting areas, including: to both geographically isolated and urban domains; as direct replacements for coal and natural gas facilities; and as co-located with industry to provide high temperature steam, to be used for hydrogen production, desalination, or other co-products. All of these new adoption scenarios point to the limits of new nuclear plant siting at existing facilities and raise critical questions about the viability of traditional siting approaches going forward. New methods for identifying and working with host communities will likely be warranted and should take a bottom-up rather than top-down approach. Consideration must be given to how nuclear energy and nuclear fuel cycle facilities impact communities, and consent-based approaches should be developed to ensure those who are exposed to risks accrue commensurate benefits and that the benefits match with community needs.

Financing

New nuclear technologies may be smaller and cheaper to build overall, but financing such projects could still put them out of reach for all but the largest utilities. There are over 3000 electric utilities in the United States, and most are too small to carry the large debt

required for even a small nuclear power plant. Therefore, to make nuclear energy a viable option for these utilities, there will need to be new financing models appropriate for the innovative reactor designs. Public-private partnerships will be critical for the first handful of demonstration projects, but beyond that there will be a need for continued access to financing to allow a more diverse set of utilities to access these technologies. Loan guarantees for small utilities could help de-risk investment, along with continued production tax credits. Further support could come through agencies like the Rural Electrification Administration or state governments, especially those that have aggressive decarbonization targets.

Summary

Public perception about nuclear energy is mixed, with some people being very strongly against it and others being open to it or in favor of using it. The large challenges society is facing has caused many people to reconsider nuclear energy because of the technology's benefits. Advanced reactors provide an opportunity to improve public acceptance even more. Innovation in how nuclear plants are designed, built, and operated is needed. Innovations in design and construction are needed to reduce the cost of building and operating the plants to make them economically desirable. New markets beyond simply electricity generation are needed with designs that enable multiple applications such as process heat, desalinization, district heating, hydrogen production, and other industrial applications. To enable these applications, innovations are needed in safety improvements allowing for such collocation. Since many countries and companies cannot afford to build large power stations, small modular and right-sized reactors are needed to make them affordable. In addition to the technological innovations that are needed, innovations in licensing, financing, and community engagement from the local level to the federal level are needed.

Advanced reactors offer the possibility of making nuclear energy easier to use as a tool to fight climate change, reduce air pollution, and help electrify developing economies. These technologies offer real technical benefits to complement what large LWRs can offer. Moreover, ARs offer an opportunity to do things differently. Because the industry is inventing something new, there is an opportunity to implement new public engagement practices, new finance models, new business plans, etc. At present, there are over 50 innovative new designs at various stages of design and development all over the world. That gives us many chances to do it right.

See Also: Worldwide Status of Advanced Reactors (GEN IV) Research and Technology Development.

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Impact of Small Modular Reactors on the Acceptance of Nuclear Power by the Public, Investors, and Owners

Daniel T. Ingersoll^a, Oak Ridge National Laboratory, Oak Ridge, TN, United States; and NuScale Power, LLC, Corvallis, OR, United States

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Introduction

The acceptability of nuclear power from perspectives of the public, investors, and plant owners is a complex, multi-faceted issue that has seen repeated cycles throughout its 60-year history. These cycles have been influenced by many externalities such as regional or global economic strength, availability of energy alternatives, and political biases. However, the nuclear industry also contributes to the social, political, and economic swings of acceptance by virtue of its successes and failures of technology choices, business models, operational performance, and management acumen through all phases of design, engineering, construction, and operations.

The nuclear industry is on the verge of introducing a new paradigm in the delivery of nuclear power: small modular reactors (SMRs) (Ingersoll, 2016). The promise of SMRs is to improve on all aspects of design, engineering, construction and operation of commercial nuclear power plants in terms of enhanced safety and resilience, improved affordability and expanded utility. If fulfilled, these promises will have a significant impact on the acceptance of nuclear power to all stakeholder groups, including the general public, financial investors, and plant owners. Before reviewing these potential impacts in detail, a brief discussion of the nature of and motivations for SMRs is provided.

Background

The commercial use of nuclear power began in 1957 with the 60 MWe Shippingport reactor. Less than 15 years later, most new commercial plants built in the U.S. had capacities nearly 20 times larger, a trend that continues today world-wide. Fig. 1 demonstrates this rapid expansion of nuclear plant unit size. Currently, most new plant designs available on the world market range from 1100 to 1600 MWe, including several designs from France, Japan, the Republic of Korea, Russia and the United States. During the past few years, a number of reactor designers have turned to developing a new generation of commercial nuclear power plants with much smaller unit capacities. Commonly designated as “small modular reactors (SMRs),” they are generally characterized by having an electrical output of less than 300 MWe, are substantially fabricated in a factory and transported to a site for installation into a fixed building structure, and operated in combination with one or more identical reactor modules. Currently, there are over

^aRetired.

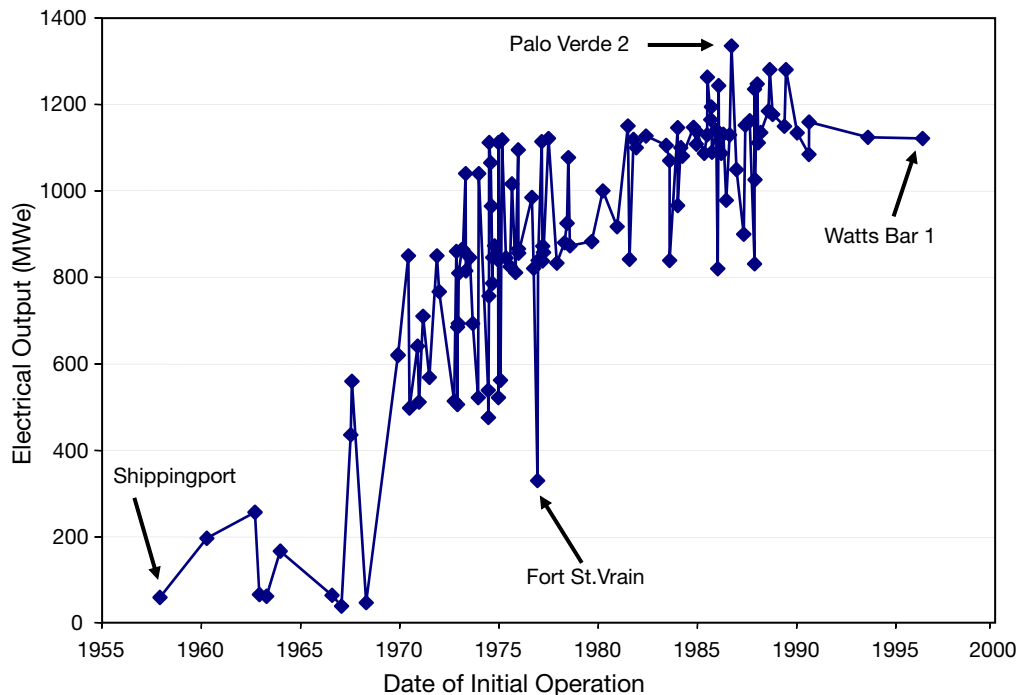


Fig. 1 Electrical output of commercial nuclear power plants build in the U.S. as a function of their initial date of operation (EIA, 2008).

50 SMR designs being commercially developed world-wide motivated by three key drivers: enhanced safety, reduced financial risk, and increased flexibility in deployment and use.

It is important to note that an SMR does not necessarily represent a new reactor technology, although it may. The reactor technology used in a specific design is largely driven by the intended use of the nuclear plant, such as electricity generation, process heat production, or fuel and waste management. Reactor technology is typically categorized by the type of coolant used to remove heat from the reactor core, and includes: water (H_2O or D_2O), gas (helium or carbon dioxide), liquid metal (sodium, lead, or lead-bismuth), and molten salt (fluoride or chloride). Of the nominally 446 commercial nuclear plants operating world-wide, 430 are water-cooled, 14 are gas-cooled, and 2 are sodium-cooled (Nuclear News, 2018).

Based on power level alone, an SMR may be a down-sized replica of a large plant design. For example, 29 of the 446 operating nuclear plants are small versions of larger plants, i.e. they use the same design and technology as large plants but have a power output less than 300 MWe. However, many new SMR designs use either dramatically different engineering principles to simplify the design or advanced technology, or both. They differ from large plants in the physical scale of the components, the relative placement of the components, and the multiplicity of the nuclear systems. In most cases, these SMRs are “deliberately small,” i.e. they cannot be reasonably scaled to larger sizes (Ingersoll, 2009).

The essence of the new SMR paradigm is to significantly change the fundamental philosophy of nuclear plant design and the approach to deploying nuclear power. The intent is to bring to the nuclear industry the kind of modularity that has already found success in most other energy generating technologies (coal, natural gas, wind, and solar) and to meet the needs of an expanded set of potential customers. With modularity comes the opportunity for standardization, scalability, and adaptability due to smaller component sizes and multiplicity of modules. Additionally, the reduced size of the individual nuclear units can provide enhanced safety (robustness) to the design, thus reducing hazard risk to the public and reducing financial risk to the investors and owners. These motivations and justifications are detailed in the sections below.

Motivations for SMRs

Most succinctly, the current global interest to design, license and deploy new SMRs is motivated by their promise of enhanced safety (resilience), affordability, and flexibility. The potential for smaller reactors to enhance plant safety/resilience beyond existing large plants comes from the application of different design principles that take advantage of system scale and natural phenomena. As an example, some SMR designs use gravity-driven natural circulation of the reactor coolant during normal operation, thus eliminating the possibility of a pump failure event and reducing or eliminating the need for emergency power during a station blackout event. Removal of the residual decay heat in a reactor core is made easier since the smaller reactor core produces less heat than a large core. Similarly, the inventory of radionuclides produced by the fission process in the fuel is less for small cores since the total quantity of radionuclides is roughly proportional to the power level. Some SMRs have module power levels less than 5% of a large reactor and

therefore will have less than $1/20^{\text{th}}$ of the potential for radionuclide release during a severe accident, even accounting for common-cause failures.

Affordability of SMRs comes mainly from having a lower purchase price. Values of \$6-10B have been cited for recent purchases of large nuclear plants. This huge price tag, and the financing challenge it creates, presents an insurmountable barrier for all but the largest utilities. SMRs offer the potential for improved affordability due to: (1) a lower total plant cost, and (2) the ability to add new generating capacity incrementally. Estimates of initial plant cost range from \$1-3B for SMR plants with total output of 200–500 MWe. Also, cash flow or capital-at-risk is much different for a multi-unit SMR plant than for a single unit large plant. The owner has the flexibility to stagger the construction or installation of the modular units, allowing the first unit to generate revenue earlier. The lower financing cost resulting from the reduced debt profile and the ability of the owner to better match their demand growth profile further reduces the total life-cycle cost of the plant.

Regarding operational flexibility, SMRs allow the owner to install incremental capacity at a rate that more closely matches demand growth. Additionally, the smaller generating capacity provides for better grid stability, especially in small-grid regions, since the unexpected loss of the power source creates a smaller perturbation on the grid. Finally, several plant features allow greater flexibility in plant siting, such as reduced water usage due to the smaller amount of rejected heat and simplified emergency preparedness due to reduced risk of radiation release. In the following sections, these motivations are explored in more depth and their potential impacts on acceptance by the public, investors and plant owners are reviewed.

Safety and public acceptance

The three most significant concerns commonly cited by the general public regarding nuclear power are: safety, nuclear waste, and nuclear proliferation. As mentioned above, SMRs are simply a different design approach than used for large reactors and they span the full gamut of reactor technologies. As such, they do not offer any particular advantage or disadvantage regarding nuclear waste and nuclear proliferation, which is primarily impacted by the choice of reactor technology. Therefore, these issues and their impact on public acceptance of nuclear power are deferred to other sections of this encyclopedia, specifically (Kadak, 2021) for waste and (Cheng, 2020) for proliferation. This leaves only the concern regarding nuclear plant safety to be addressed in this chapter, for which SMRs can have a dramatic impact.

The safety of nuclear plants in the U.S. initially crept into the national dialog in the early 1970s, due in part to an industry-led study to evaluate the statistical probability of reactor accidents. Anti-nuclear groups were quick to interpret the study as casting doubt on the safety of nuclear power and have been driving the public dialog in that direction ever since. The nuclear industry perpetuates this perception by being defensive about nuclear plant safety despite all evidence to the contrary. In reality, nuclear plants are incredibly safe. By any metric, the nuclear industry has one of the best records among all industries for protection of the workforce and the general public.

In a 2007 book, *The Geopolitics of Energy*, several common myths of nuclear power were discussed, including safety (Wright and Conca, 2007). The data in Table 1 are derived from the assessment of US fatalities in the period 2001–2006 and show the average annual number of deaths in the U.S. resulting from the listed activities. The leading cause of death was “iatrogenic causes,” which result from unexpected and unexplained outcomes of medical procedures. It was not surprising that smoking, alcohol and automobile accidents are high on the list—most people would expect that ranking. The fact that nuclear power is at the bottom of the list is likely a big surprise to many people. Operational injuries follow a similar pattern with the nuclear industry showing the lowest rate of injuries—even less than banking.

Despite the exceptional safety record of the nuclear industry, which has been consistently demonstrated over multiple decades of world-wide experience, many people, including industry advocates, persist with casting doubt on the safety of nuclear power. A key to improving public acceptance of nuclear power may rest with a clarification of terminology, specifically, distinguishing between *safety* and *resilience*.

Table 1 Average annual US fatalities for the period 2001–2006.

<i>Activity</i>	<i>Number of deaths</i>
iatrogenic causes	190,000
Smoking	152,000
Alcohol	100,000
Automobile accidents	50,000
Firearms	31,000
Coal use	6,000
Construction	1,000
Hunting	800
Police work	160
Nuclear power use	0

Wright J and Conca J (2007) *The GeoPolitics of Energy: Achieving a Just and Sustainable Energy Distribution by 2040*. North Charleston, SC: BookSurge Publishing.

Safety and resilience

Plant *safety* is not at the discretion of the designer; it is an expectation set by the regulator. The designer is obliged to develop a design that meets those expectations in a convincing and substantiated manner. Safety standards established by the regulator are focused on reasonable assurance of adequate protection of the workforce and the public. The often cited concern for nuclear safety is not as much about its actual safety but rather a perception of its risk, which has both a technical basis and an element of emotion. For example, a person may be too afraid to board a commercial airliner but would not hesitate to jump into the family car—a much riskier proposition.

The US Nuclear Regulatory Commission (NRC) sets an extremely high bar on its established safety standards and is frequently referred to as the “gold standard” for nuclear regulation world-wide (Rempe, 2021). All reactor designs that have been certified by the NRC meet these exacting standards. The trend in new plant designs, large and small, is to meet the NRC safety standards in more assured ways, principally by making use of fundamental laws of nature whenever possible rather than relying on engineered systems that require operator action and a sustained source of electrical power to function. In doing so, the plant naturally becomes more resilient against unanticipated upsets.

All existing plants have been properly licensed and therefore are considered safe according to NRC standards. To further improve the designs, it may be instructive to think of designing beyond safety. This requires introducing new terminology that is a superset of *safety*, which is defined in the Webster New World Dictionary (Guralnik and Friend, 1962) as “freedom from danger, injury, or damage.” A suitable new term for a more expansive quality may be *robustness* or the more recently popular term *resilience*, which Webster defines as “ability to bounce or spring back into shape...after being pressed or stretched” (Guralnik and Friend, 1962). While safety relates to protection of the workforce and public from hazards within the plant, resilience relates to protecting the plant itself—ensuring that the financial investment and the generating capacity are not lost after a major event. As an example, the accident at the Three Mile Island nuclear plant in Pennsylvania in 1979 (Skillman and Rempe, 2021) resulted in no direct impact on the workers or the public (other than extreme anxiety); however, the damage to the reactor core was a financial disaster for the plant’s owner. Said differently, if a plant can survive a major upset intact, protection of the workforce and public will also be ensured. Realizing this distinction, many SMR designers seek to design for plant resilience.

Designing for resilience

Design choices impact its safety characteristics and also its resilience. Compare a delicate wine chalice and a beer stein: both are fundamentally used for serving an alcoholic beverage but the stein is considerably more robust. Similarly, engineered systems can be designed with robustness in mind or not. Many SMR designers understand this principle and design to a high level goal of significantly enhancing the resilience of their design, even if they articulate it as enhanced safety. Mario Carelli did this with his “safety by design” approach for the IRIS design (Carelli et al., 2004) and Jose Reyes is doing this for the NuScale design (Reyes, 2012).

Nuclear plant resilience fundamentally comes down to one requirement: keep the reactor fuel below its failure temperature at all times. If the heat produced in the reactor is successfully removed, the nuclear fuel remains intact and there are no bad consequences. Some reactor designers address this requirement by using a reactor fuel with very high temperature tolerance. Other designers focus on assured removal of the heat. Smaller sized reactors offer more options to address the heat removal requirement by the simple fact that there is less decay heat to remove and more options for removing it. Different SMR vendors have chosen a variety of ways to capitalize on this advantage.

Despite the diversity of SMR designs, many of them share a common set of design principles to enhance plant safety and resilience. The top-level principles include: (1) eliminate as many features as possible that have the potential to initiate a serious accident, (2) for those features that cannot be eliminated, reduce the probability that an accident will be initiated, and (3) design the system to substantially mitigate the consequences of remaining potential accidents.

An example of principle (1) is the integral primary system design approach in which all or nearly all of the primary system components are contained in a single pressure vessel. Fig. 2 shows a simplified comparison of a loop-type pressurized water reactor (PWR) design typical of most large operating reactors (Brown, 2021) and an integral primary system design, sometimes referred to as an iPWR. An integral configuration greatly simplifies the design in two ways: (1) it eliminates the external piping and vessels, and (2) it eliminates the need for active backup safety systems required to mitigate the extreme consequences resulting from a major break in any of the external pipes or vessels. These design changes not only increase the plant’s robustness but also reduces construction costs and operational costs associated with the eliminated systems.

A comprehensive review of many design features for small and medium sized reactors spanning all reactor technologies is given in a 2009 report issued by the International Atomic Energy Agency (IAEA) (IAEA, 2009). Some of the specific features that help to implement these principles and achieve a higher level of plant resilience include: the use of passive systems (rely on natural forces such as gravity, conduction and convection), arrangement of the primary system components to eliminate accident initiators, provide assured removal of decay heat produced within the reactor core.

An important consideration emphasized by the accident at the Fukushima Dai-ichi plant in 2011 (Mizokami and Rempe, 2021) is associated with the importance of having assured backup electricity in the case of a reactor upset. This vulnerability is well known and designers provide redundant layers of backup power options, including emergency diesel generators and safety-grade batteries. The power is needed to run pumps that circulate the cooling water and to operate valves that direct the various sources of cooling

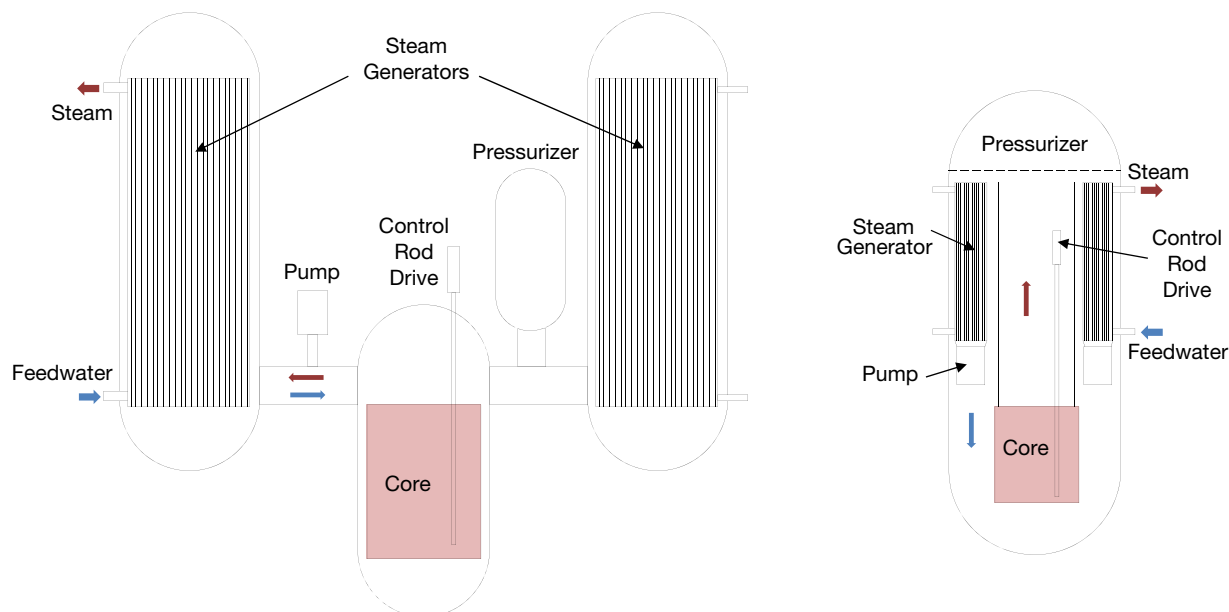


Fig. 2 Comparison of traditional loop reactor design (left) with integral reactor design (right).

water to the required locations. A more elegant solution to this plant vulnerability is to eliminate the need for backup power, at least for maintaining the reactor core cooling function. Because of the smaller amount of decay heat within an SMR, it is practical to design the reactor system to not require backup power for extended and potentially unlimited time. In the NuScale design, this is accomplished by eliminating pumps entirely from the primary reactor system and immersing the module directly into a large pool of water to provide assured flow of heat from the reactor core to the ultimate heat removal system. In 2018 as part of the on-going review of NuScale's design certification application, the NRC formally agreed with NuScale that no safety-grade power source is needed to achieve safe shutdown of its modular reactors—a landmark accomplishment in nuclear plant resilience.

Another consideration, which is related to the previous issue, is the broader concept of "grace period." The grace period is the length of time after an accident during which no external power, make-up water, or operator action is required to maintain the plant in a safe condition. Initially, 24 h was used as the acceptable grace period, although newer plants are expected to provide a 72 h grace period. The accident at Fukushima demonstrated that more than 72 h may be needed in severe accident situations. The devastating tsunami made access to the plant by emergency workers very difficult. The few workers that were available were heavily burdened by the many actions needed to deal with the accident progression in four large reactor units. In recognition of this challenge, and due to the smaller heat loads and radionuclide inventories in SMRs, several designs appear to have achieved severe accident grace periods of several weeks. At least one water-cooled design (NuScale) offers an unlimited grace period, owing largely to its small module size of only 60 MWe and the fail-safe nature of its simplified emergency core cooling system. The modular high-temperature gas-cooled reactor (MHTRG) also promises unlimited grace period by virtue of its highly robust fuel form and assured removal of the decay heat (Shenoy et al., 2012).

The enhanced resilience that many SMR designs offer is perhaps the most significant benefit that SMRs can bring to the nuclear industry and has the potential to catalyze a major redirection of the public's perception about nuclear power. Moving the dialog from *safety* to *resilience* will be an important step in developing the new methods and metrics that will allow the industry to confidently advance the resilience of new plant designs against a broader range of unexpected threats and allow the public to recognize its benefits. Most likely, the first discernable impact of this enhanced resilience will be seen by the general public as a simplified emergency planning associated with new SMR builds—a crucial consequence of design resilience. The relationship between design resilience and emergency planning is discussed more thoroughly in Section "Financial risk."

While many studies have concluded that SMRs can provide exceptional safety and robustness, the consistent concern in these studies has been that the enhanced resilience comes at a price and the price is economic viability. The next section addresses this concern and offers justification for why improved economics will in fact be another significant benefit that SMRs bring to the nuclear industry.

Affordability and investor acceptance

Given the admirable safety record demonstrated by the nuclear industry over the past few decades, economics has now replaced safety as the greatest challenge and uncertainty for expanding the use of nuclear power. This concern is further exacerbated for SMRs due to a lack of direct experience with their construction and operation, and a prevailing mindset that "bigger is better"

(or at least cheaper). Consequently there have been several studies in recent years generating a plethora of papers and reports dedicated to proving or disproving the economic viability of SMRs. Unfortunately, the results and conclusions from the many studies are quite diverse, sometimes contradictory and frequently lack a clear understanding of SMR fundamentals.

Alvin Weinberg, a pioneer in the development of nuclear power and a staunch promoter of new technologies, also recognized the importance of economics. In his book *The Second Nuclear Era*, he wrote: “Unless a forgiving reactor is affordable, no one will buy it” (Weinberg et al., 1985). He was speaking specifically about the new smaller sized, highly robust reactor designs that were emerging during the late 1970s and early 1980s. Many subsequent studies of smaller sized reactor designs by the IAEA and Nuclear Energy Agency (NEA) also identified potentially unfavorable, or at least uncertain, economics as the key concern for smaller sized reactors.

With the rapid reemergence of interest in SMRs in the past ten years, both the IAEA and the NEA have taken a fresh and more refined look at the anticipated economics of SMRs. Additionally, several research organizations have conducted studies related to SMR economics with varied emphasis on market potential, cost projections, economic competitiveness, and economic impacts. An excellent overview of SMR economic considerations is given in a recent paper entitled “Small modular reactors: A comprehensive overview of their economics and strategic aspects” by Giorgio Locatelli and colleagues (Locatelli et al., 2014). Sara Boarin and colleagues also provide a good foundation of SMR economic principles in Chapter 10 of the *Handbook of Small Modular Nuclear Reactors* (Carelli and Ingersoll, 2014).

From the nuclear power industry’s infancy, the dominant metric for evaluating the economic viability has been “levelized cost of electricity” (LCOE). Roughly speaking, this is the total life-cycle cost of the plant normalized by the total energy production of the plant and is usually expressed as cents per kilowatt-hour (or dollars per megawatt-hour) (Rothwell, 2021). This metric continues to be appropriate for large capacity plants, which are typically located on extensive inter-connected grids with a robust wholesale electricity market. In this environment, the unit cost of electricity from each plant must be competitive with electricity generated by other plants—nuclear or otherwise—in the same wholesale market. While the LCOE metric makes sense for large plants, it is not obvious that it is equally important for SMRs, especially for those that are located on smaller regional grids or used for off-grid applications. Even for large-grid applications, it is necessary to expand the set of economic metrics in order to capture the new features that SMRs bring to the market, especially those that reduce investor risk such as incremental capacity growth.

A few recent studies have acknowledged the need for better economic metrics for SMRs (Rosner and Goldberg, 2011; NEA, 2011a). The collective studies highlight several financial metrics that are important to a fair assessment of the economic viability of SMRs, including:

- *Upfront capital cost.* This metric is basically the purchase price of the nuclear plant. It is central to the affordability of SMRs and is perhaps the leading discriminator between customers today who can consider building a nuclear plant and customers for whom this is not an option.
- *Interest during construction.* This metric is a subset of the total project cost and represents the cost of borrowed money. It includes several considerations such as availability of money, credit rating of the owner, length of construction, total debt, and overall project risk.
- *Maximum cash outlay.* This metric, similar to capital-at-risk, is the largest amount of debt that has to be financed at any time during the project and influences the interest rate of borrowed money.
- *Time to first revenue.* This metric helps to capture investor risk concerns, especially return-on-investment and financing costs.
- *Sensitivity to market changes.* This metric also captures investor risk concerns and impacts the uncertainty of the potential return on investment once the plant is operating.

Specific examples of these metrics, as well as the traditional LCOE metric, are discussed in the following sections to better characterize the potential economic viability and competitiveness of SMRs.

Affordability

Purchase price matters. Utilities have limits on what they can responsibly finance and what their investors or public service commissions will allow. The high cost of current large nuclear plants and the fragmented utility structure in the U.S. are driving a situation in which only the few most affluent utilities can consider building new nuclear plants. This effect is portrayed in Fig. 3, which compares the capital cost of four different energy sources: a multi-module SMR, a large dual-unit nuclear plant, a natural gas plant and a coal plant. Cost data for the latter three plants were taken from an Energy Information Administration report. (EIA, 2013) The cost range shown in the figure for the SMR was gleaned from publicly available data for some near-term SMRs that are under development in the U.S. Fig. 3 also shows the average annual revenue of all US investor-owned nuclear utilities (Rosner and Goldberg, 2011). The reality is that the purchase of a dual-unit large nuclear plant, similar to the plants being built in Georgia and recently abandoned in South Carolina, is a “bet the company” proposition for the larger utilities and is beyond consideration for smaller utilities. Although producing significantly less power than the large nuclear plant, an SMR offers a purchase price that is within reach of a smaller utility and may better match their capacity needs.

Another aspect of affordability is the flexibility to purchase an item incrementally. A group at the Politecnico di Milano (POLIMI) has been very active in the economic assessment of SMRs and developed the “integrated model for the competitiveness assessment of SMRs—the INCAS code.” Fig. 4, which is derived from their analyses (Boarin, 2015), shows an example of the impact of a staggered deployment of an SMR using INCAS-generated data provided by POLIMI. Three cases are presented: (1) a compressed

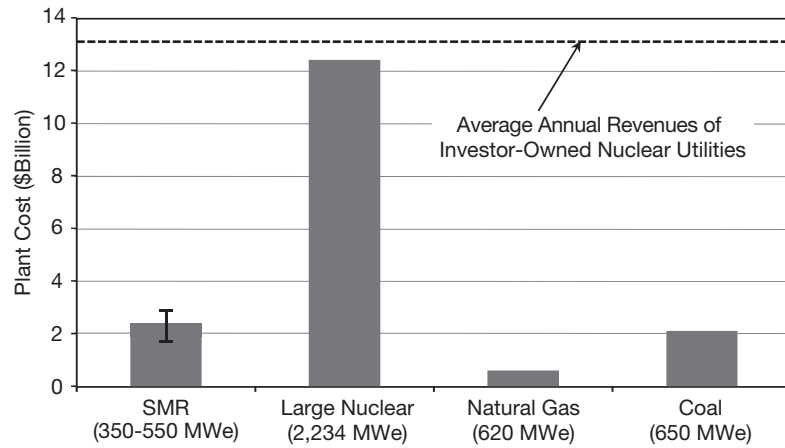


Fig. 3 Comparison of power plant investment costs for different energy sources.

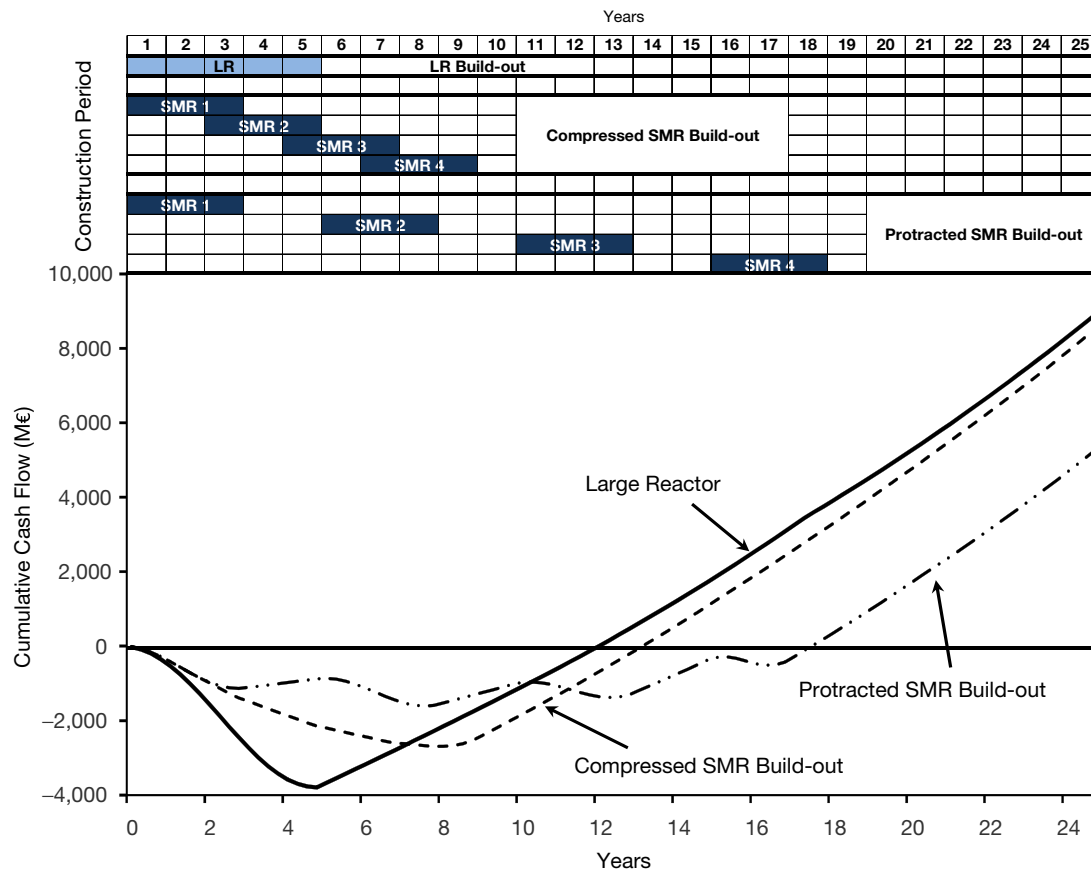


Fig. 4 Impact of staggered build-out on maximum cash outlay (Boarin, 2015).

build-out of four 300 MWe single-unit SMR plants with a one-year overlap of each consecutive three-year SMR construction, (2) a protracted build-out of the same four SMRs with a two-year gap between consecutive plant constructions, and (3) a single 1,200 MWe plant with a 5-year construction duration. Both the SMR and large reactor were assumed to be PWRs. The construction schedules are depicted in the upper portion of Fig. 4. In the lower portion of the figure, the cumulative cash outlay profiles for the two SMR cases are compared to the large reactor cash outlay. The maximum cash outlay for the large reactor occurs in year 5 and is approximately 3,800 M€ (million euros), which is the full amount of the plant's construction cost. For the compressed SMR build-out case, the maximum cash outlay is reduced by roughly 30% to 2,700 M€, which occurs in year 8. The reduction in peak cash outlay is a consequence of the fact that revenue is being generated by the earlier SMRs while the latter units are being constructed, thus increasing the amount of self-financing. This effect is even more apparent in the protracted build-out case, which results in

a much smaller and flatter cash outlay profile. In this case, the maximum outlay is only 1,600 M€—nearly 60% lower than the large reactor case.

It is important to note that the three cases in this example result in different rates of new capacity addition to the grid. In the large reactor case, the full 1,200 MWe of generation is available at the beginning of year 6, while this does not occur with the SMR cases until year 10 for the compressed schedule and year 19 for the protracted schedule. However, if the reduced rate of cumulative capacity is acceptable to the plant owner based on demand growth, then the added flexibility of staggering the construction of the individual SMR units can have a significant impact on the owner's maximum debt. This, in turn, can help to reduce overall project cost by reducing the cost of financing the plant.

Economic competitiveness

Economic competitiveness of SMRs continues to be their most hotly contested feature. The controversy is probably a result of many factors, foremost of which is an ingrained mindset in the nuclear industry favoring economies of scale. As discussed earlier, LCOE quickly became the dominant metric in the early phases of the nuclear industry. It was also widely assumed that LCOE is lowest for large plants due to economies of scale. However, the fixation on economies-of-scale principle actually drove the industry in the opposite direction, i.e. it generated a succession of larger and more complex plants that ultimately resulted in a higher unit cost of electricity.

Economies of scale have been demonstrated repeatedly in many industries. Every system designer must contend with this reality. However, a closer look at the economics of SMRs suggests that the economies-of-scale factor can be substantially mitigated by a combination of other economic considerations. One of the earliest studies of the “economies of small” was conducted by Hayns and Shepherd of AEA Technology during the time of the development of the Safe Integral Reactor design (Hayns and Shepherd, 1991). They provided a qualitative discussion of a dozen factors that can offset economies of scale in SMRs, which are paraphrased in Table 2.

In another study, Ning Li recognized the importance of design simplification (Li, 2009). He cites the example of emergency core cooling systems, which are added to a nuclear plant to protect against the severe consequences of a loss-of-coolant accident. He argues that the lower power rating of an SMR results in design solutions that allow simple passive systems to suffice. In contrast, large (high power) plants require sophisticated active emergency core cooling systems.

In the case of the NuScale design, the unique integrated module configuration and small unit power—just 60 MWe—allows 15 major systems or components found in traditional PWR plants to be eliminated while introducing only one new system. A recent study by the Energy Policy Institute at Boise State University compared detailed cost information for a NuScale plant against a standard 4-loop PWR (Black et al., 2019). As expected, the NuScale plant was significantly less expensive than the PWR on an absolute basis—less than half the total capital cost. Surprisingly, it was also found to be less expensive on a per-unit-output basis. The normalized direct capital cost (structures, equipment, etc.) was similar for the NuScale and PWR plants, while the normalized indirect capital cost (design services, construction supervision, etc.) was 68% lower for NuScale, yielding a total capital cost that was 38% lower than the large PWR cost. They attribute the cost savings to the “modular design and off-site manufacturing that dramatically reduces the on-site construction activities.”

Another consideration regarding the competitiveness of SMRs is operating cost. Given the complete lack of experience in the operation of any contemporary SMR designs, anticipated operating costs are highly speculative and generally lacking in the

Table 2 Factors that can mitigate economies-of-scale in an SMR.

<i>Factor</i>	<i>Description</i>
Increased factory fabrication	Smaller unit size permits a greater degree of factory fabrication and less on-site construction to reduce labor cost
Skilled work force retention	Increased factory fabrication will help to attract and maintain skilled workers without drawbacks of frequent relocations
More replication	Production of a larger number of small standardized units increases efficiency of fabrication/construction
Multiple units at a single site	Shared infrastructure at a common site can reduce capital cost per installed unit
Improved availability	Multi-module plant using smaller reactor modules improves the possibilities for partial operation of the plant during maintenance of a single module
Learning curve	A larger number of small identical units expedites achieving fabrication/construction efficiencies through learning process
Bulk ordering	A larger number of small units can reduce cost of bulk orders for individual components
Better match to demand	Small unit size allows better match to demand, thus reducing costs associated with purchased power or stranded assets
Smaller front-end investment	Smaller total project cost reduces construction financing cost
Reduced construction time	Shorter construction time reduces construction financing cost and introduces revenue sooner
Increased plant lifetime	Small modules may lead to easier refurbishment/replacement costs and extend lifetime of the facility
Reduction in planning margin	Smaller unit capacity shortens planning horizon and reduces investment sensitivities to market changes
Design appropriate to the size	Small unit size can enable simplification of systems and reduce costs of materials, fabrication, maintenance and disposal

open literature. There are reasonable bases for speculating that SMR operating costs will be higher due to a lack of scaling of labor force requirements; however, there are also valid bases to speculate that innovations and efficiencies in operations will more than compensate. Given that even the most mature contemporary SMR designs are still in the licensing or construction phases, it may be more than a decade before substantiated data on operating costs are obtained.

Financial risk

Total project cost is clearly the dominant factor in the purchase of a nuclear power plant. However, financing risk is also a major consideration, especially in merchant markets. The bigger the risk, the higher is investor concern and the more they charge for using their money. Also, as risk increases, some financing options disappear such as venture capital money, which are typically smaller investments (less than \$100M) with an expectation of rapid pay-back (less than 5 years). In contrast, bringing a new nuclear project to reality generally costs billions of dollars and may require decades before the first dollar of revenue is generated.

Undeniably, nuclear power conjures up a higher perception of risk to the investor community than most other industrial endeavors. This perception was articulated at a 2009 workshop on financing of nuclear projects as part of the Global Nuclear Energy Partnership (GNEP) program. Several individuals were invited from the financing community, including Deutsche Bank, Fitch Rating Services, Towers Perrin, Nippon Export and Investment Insurance, and the U.S. Export-Import Bank. These representatives shared a consistent message: nuclear projects are among the most difficult for investors to embrace. Some of the more common justifications included the capital-intensive nature of nuclear projects with very long payback times, impact of external factors such as regulatory uncertainties and public response, construction delays and cost overruns, and the need for reliable long-term market forecasts. In Grubler's paper on the French nuclear program, he writes: "Perhaps the nuclear 'valley of death' is its inherently high investment costs and their tendency to rise beyond economically viable levels" (Grubler, 2010).

The POLIMI group also assessed financing risks associated with nuclear power projects (Carelli and Ingersoll, 2014). They cite many of the same issues that were voiced by the bankers at our GNEP meeting, which they categorize into those that are associated with capital-intensive projects and those that are unique to nuclear projects. Their major risk factors are delineated in Table 3.

The POLIMI group further asserts that construction cost and schedule overrun is the leading factor that contributes to nuclear project risk. As mentioned earlier, total project cost is highly sensitive to construction duration. The most significant component of this sensitivity is financing cost, but other cost factors can contribute such as retention of the labor force and protracted rental fees for major construction equipment. The inherent features of SMRs mitigate many of the risks. These include shorter construction time, faster time to first revenue, and higher reliability in fabrication costs and schedule. The shorter construction time and sequential construction features also result in lower sensitivity to several market-related factors such as interest rates, electricity prices, and changes in electricity demand due to the owner's ability to respond faster to changing conditions. Of special note was the fact that the combined effect of shorter construction time and reduced capacity of an SMR helps to mitigate the impact of miscalculating the long-term forecast of electricity demand, which can result in the purchase of expensive replacement power or result in a stranded asset—either of which hurts the plant's profitability.

While SMRs appear to have a clear advantage over single large plants in terms of reducing investment risks listed in the left-hand column of Table 3, it is less obvious that they will have any impact on the nuclear-specific risks. Their ability to impact these factors will depend substantially on the successful licensing and deployment of the first SMRs in terms of demonstrating the first two major strengths of SMRs: their enhanced safety/resilience and their affordability. If this can be done for traditional electricity applications, it will enable realization of the third major benefit of SMRs: their suitability for expanding the use of nuclear power to non-traditional markets and non-electrical applications—the topic of the next section.

Flexibility and owner acceptance

Small modular reactors are not generally intended to compete with large plants. Many SMR designs are motivated by the opportunity to expand the use of nuclear power to more diverse customers and applications. In some cases, the SMR features that enhance

Table 3 Major risk factors impacting financing of nuclear power plant projects.

<i>Common to capital-intensive projects</i>	<i>Unique to nuclear power projects</i>
• High up-front capital costs • Cost uncertainty • Construction supply chain risks • Long lead times and long payback periods • High sensitivity to interest rates • Plant reliability and capacity factor • Market price of product (electricity)	• Unstable public support • Negative public acceptance • Regulatory and policy changes • Decommissioning/waste disposal costs and liabilities

Carelli MD and Ingersoll DT (2014) *Handbook of Small Modular Nuclear Reactors*, Cambridge, UK: Woodhead Publishing.

their adaptability to non-traditional plant owners also enhance the usability of the plant for even the traditional users. The first two features of SMRs that help to provide their expanded flexibility are captured in their name: small and modular. Two additional features that are derived from their smallness and modularity are worth highlighting: their benefits for more flexible plant siting and their adaptability to non-electrical applications. All four factors, which are discussed in more detail in the following sections, contribute to the promising potential that SMRs offer to better accommodate the needs of both traditional and non-traditional owners.

Size considerations

There are some applications for which an SMR (with output typically greater than 50 MWe per module) is actually too big for the energy demand. These include customers such as small isolated communities, single industrial facilities, mining operations, and defense installations. For these applications, there is an emerging class of reactors referred to as micro-reactors with power outputs generally below 10 MWe (Wallenius, 2021). Although SMRs may not be a solution for the very small consumer, there are many cities and industrial complexes that have significant electrical demand, but for which large nuclear plants or other energy alternatives may not be practical. Some of these communities may have access to adequate electrical grids but many more have outgrown their local grid infrastructure or have only marginal grid interconnection. For these customers, SMRs offer a very promising solution, suggesting the value in a more distributed power generation model than conventional nuclear allows.

Installing new overhead high-voltage lines can cost \$1–2 million per mile and the cost of underground lines can be four to five times higher. In some locations such as the western U.S., it is especially challenging to add new transmission lines. Whether because of the vast amount of protected lands, the harsh topology, or safeguarding of protected species, the acquisition of new transmission right-of-ways and the construction of the lines are both difficult and expensive. Extended transmission lines also create grid vulnerabilities since they can be disrupted by severe weather or sabotage. Recently in California, downed power lines have contributed to devastating fires. This reality highlights a significant benefit of an SMR's smallness: its small unit output allows for a more distributed model of nuclear power generation than currently exists. In other words, SMRs are about offering more choices to the utility executives, not only choices among power generation options but also the choice between distributed generating assets versus new transmission and distribution networks.

Having smaller sized generators also helps to provide grid stability, especially in regions with limited grid interconnection. An often-cited rule-of-thumb for stable grid operation is that the output of a single generating unit should not exceed 10% of the total generating capacity. Even in large grid regions, utilities are accustomed to operating smaller sized generating units. Fig. 5 shows the size distribution of all types of electrical generation plants worldwide, which indicates that roughly 93% of the generating plants have capacities below 500 MWe (Research and Markets, 2006). Even in the U.S., only six of the nearly 1,400 new generating units added between 2000 and 2005 had power ratings of greater than 350 MWe. Consequently, most utilities are very familiar with the operation and grid management strategies for smaller generating units.

Related to this is the requirement for “spinning reserve,” which is the excess capacity that must be available to the utility to accommodate the loss of the largest generator on the system without major disruption. Large, single-unit generating plants dictate

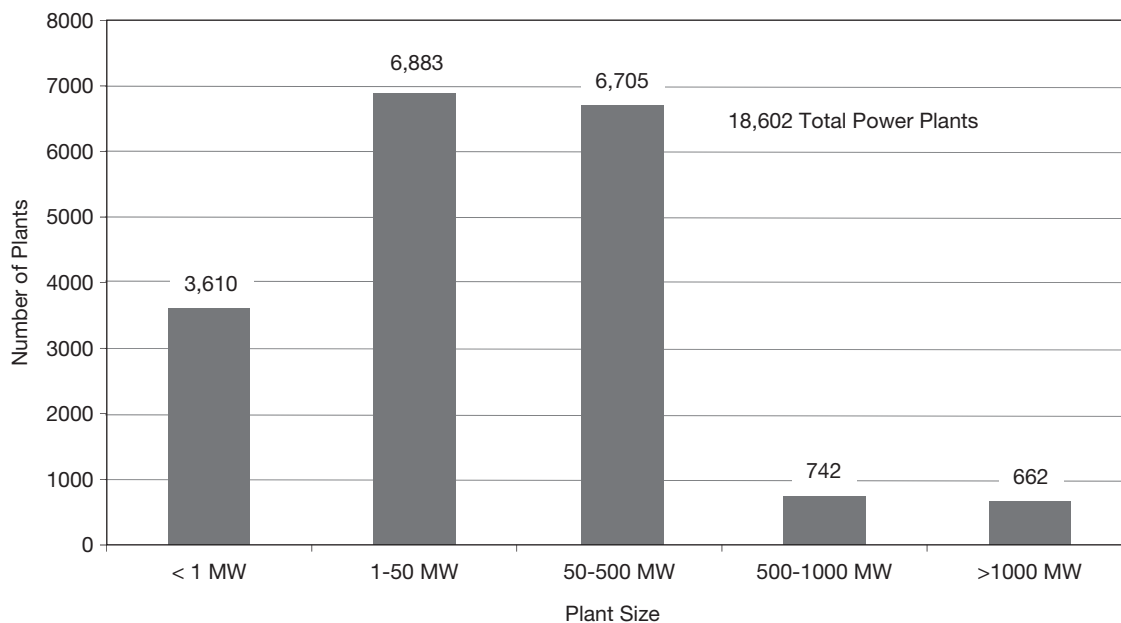


Fig. 5 Size distribution of electricity generation plants globally (Research and Markets, 2006).

the need for a large amount of spinning reserve. This can be accomplished by running many of the existing stations at slightly less than full capacity, or by operating dedicated units on continuous standby. Either way, energy is being wasted. The smaller output of an SMR provides a more manageable power distribution and reduces wasted resources associated with the spinning reserve capacity.

Also related to the integrity of the grid infrastructure is the issue of load-following, i.e. adjusting generator output to match demand fluctuations. Conventional wisdom suggests that nuclear power plants should be operated continuously at full capacity and that natural gas plants are best suited to provide “peaking” capability to meet excess demand. This historical strategy has been driven largely by economic considerations. The logic is that because a nuclear plant has a relatively high capital cost and relatively low fuel cost compared to a natural gas plant, it makes sense to keep the more expensive hardware running continuously. Because of its low fuel cost, running the nuclear plant at 50% power has minimal impact on its operational costs but cuts revenue in half. Increased maintenance costs are also normally higher with load-following operation due to fatigue issues introduced by the thermal cycling of the reactor and balance-of-plant systems.

Some coal plants, which also operate best at full power, are now being cycled for load following in the U.S. and nuclear plants are expected to follow. Worldwide, France’s pressurized water reactors load-follow on a regular basis due to the high percentage of nuclear-generated electricity on their grid (nominally 75%). Canadian reactor units are also required to load-follow due to the relatively high percentage of nuclear power there and German reactors load-follow primarily because of a relatively high contribution of intermittent wind generation on their grid (NEA, 2011b).

There are now two reasons to consider load-following with SMRs. The first is for small grid applications enabled by SMRs. In these cases, similar to what is experienced in France and Canada, the relatively high percentage of nuclear generation on the regional grid will likely force the SMRs to follow the changing demand. The second reason to consider load-following with nuclear plants is in response to an increasing penetration of renewables on the grid, similar to the situation in Germany. The increasing penetration of renewable sources, especially wind, has altered the economic argument since wind turbines are also capital-intensive (more expensive than nuclear plants per unit of power produced) and their fuel cost is very low—zero, in fact. Also, some regional policies require grid dispatchers to preferentially use renewable energy first, thus requiring baseload plants to accommodate demand variations. Consequently plant owners are already beginning to perform power maneuvering with coal plants and are being pressured to extend this to nuclear plants.

Fortunately, the smallness of a typical SMR allows it to better accommodate load-following operations and several vendors advertise this capability. In general, load-following impacts are reduced relative to large plants because of the greater agility of the plant to respond to thermal cycling. The greater agility is provided by the smaller reactor system components, smaller turbine/generator equipment, and system simplifications that are enabled by the smaller reactor size. Fast spectrum reactors such as sodium-cooled SMRs (Heide, 2021) have a further advantage that the core reactivity is much less sensitive to xenon buildup and depletion in the reactor core, a phenomenon that complicates core power maneuvers in thermal spectrum reactors such as water-cooled reactor systems.

Modularity (scalability)

Related to small size is the equally important factor of modularity. They work together to provide a scalable solution to meet a wide range of potential applications. Successful examples of modularity exist throughout the energy industry. The world’s largest solar farm, which was recently completed in California, produces over 500 MWe from nine million individual solar panels. Multi-gigawatt wind farms are built from thousands of 1–2 MWe wind turbines. Also, many coal plants are comprised of multiple small boilers such as the 1,400 MWe Kingston Steam Plant in Tennessee that uses nine 175–200 MWe boilers. In fact, nuclear appears to be the last holdout in the energy industry for the sole use of large single-unit power generating stations.

Multi-module plants allow the owner to build new capacity at a rate that more closely matches demand growth and allows the local grid capacity to be built up in modest increments. This provides the owner with operational flexibility and favorably impacts the economics of the plant, as discussed in the previous section. Besides the economic advantages of reduced cash outlay and better demand matching, modularization of nuclear plants provides an opportunity to take standardization to a new level. The U.S. did a dismal job at standardizing nuclear plants in our original build-out during the 1960s and 1970s. Nearly every plant was a unique design. Subsequent nuclear build-outs in countries such as France and the Republic of Korea have done a better job at maintaining standardization in their fleet of commercial reactors with an observable gain in improved efficiencies and reduced costs. The new plants being built in the U.S. also anticipate improved standardization, but it is too early to tell if the vendors and new customers will adhere to this goal. Modular plants facilitate the possibility of a highly standardized reactor system by virtue of their intrinsic multiplicity within a plant, smaller component sizes, and a higher fraction of factory-fabricated components.

The higher degree of reactor system standardization that SMRs can provide should also accelerate the implementation of fleet-wide system management, similar to what is done in other industries such as jet engines in the aircraft industry. Rolls Royce is able to monitor a large array of operating parameters on thousands of in-service jet engines using sensors and live satellite feeds to predict potential problems before they occur (Rolls Royce, 2015). The ability to monitor the operating condition of each module in real time could further enhance plant safety through early identification of component aberrant behavior and allow for timely repair or replacement. A further simplification and economic advantage of operating a fleet of standardized modules is the efficiencies gained in maintaining a reduced inventory of replacement parts and qualification of maintenance personnel.

Siting flexibilities

A significant siting flexibility of SMRs, if it can be realized, is the opportunity to simplify the plant's emergency management plan, including potential reduction in the plant's emergency planning zone (EPZ). The NRC currently sets the EPZ for existing and new nuclear plants at 10 miles for plume exposure and 50 miles for ingestion pathways. However, they allow a license applicant to request a reduced EPZ and have licensed several small reactors with an EPZ of 5 miles. These include the Fort St. Vrain (842 MWt), Big Rock Point (240 MWt) and La Crosse (165 MWt) plants, although none of these plants are still in operation (INL, 2014). As discussed in a previous section, new SMR designs are expected to have significantly enhanced robustness against accidents, which results from innovative engineering and extensive use of passive safety systems. They also will have reduced accident source terms by virtue of a smaller reactor core size and may include additional barriers for fission product release if fuel damage does occur. Hence, it should be possible for an SMR to justify a reduced EPZ size while not increasing risk to the general public.

In recognition of this opportunity and its value, the SMR community has been working through the Nuclear Energy Institute (NEI) to develop and pursue a methodology for deriving the appropriately "optimized" emergency preparedness of an SMR that is commensurate with the plant's relative risk. The NEI submitted a whitepaper on the topic to the NRC in December 2013 describing a methodology for establishing a "technology-neutral, dose-based, consequence-oriented emergency preparedness framework for small modular reactor sites" (NEI, 2013). The methodology requires that a license applicant provide a substantiated technical basis for the proposed emergency planning actions, including the extent of the EPZ.

The first test case for the new methodology was an early site permit application submitted by the Tennessee Valley Authority (TVA) in 2016. As part of their application, which enveloped four SMR designs, TVA proposed a scalable EPZ and supported their proposal with results from a NuScale analysis using the new methodology. In another landmark accomplishment for SMRs, the NRC staff concurred with a site boundary EPZ for a NuScale design.

Optimization of the EPZ for SMRs is valuable for a number of reasons. First, a recent analysis at the Idaho National Laboratory determined that the average cost of establishing a 10-mile EPZ averages about \$10M per plant and adds greater than \$2M to the annual operating cost (INL, 2014). A substantial portion of these costs would be saved if a reduced EPZ can be justified. More importantly, a reduced EPZ facilitates siting of the plant closer to population centers without the costly and socially alarming aspects of emergency planning requirements. This is especially important if SMRs are to be successful as an option for distributed power generation for smaller, more remote communities. A reduced EPZ will also help to facilitate broader missions for SMRs, especially process heat missions. In this case, co-location of the reactor with the industrial user is very important to minimize heat losses due to lengthy steam distribution lines. This brings us to the fourth major flexibility that SMRs have to offer its owner: suitability for non-electrical applications.

Non-electrical applications

A significant attractiveness of small reactors is their flexibility to enter traditionally non-nuclear energy markets. Several energy-intensive applications could significantly benefit from using nuclear-generated heat to replace the combustion of fossil fuels to produce heat. This would not only reduce emission of greenhouse gases, but would also allow the fossil resources to be used as feedstock for producing higher value products such as petrochemicals and plastics. In the U.S., electricity generation contributes only 40% to our total carbon emissions, so a concerted effort to reduce our total emissions will necessarily require replacing fossil fuels with non-emitting fuels in the industrial sector as well as the electricity generation sector.

The suitability of SMRs for this mission is largely a result of their smallness and modularity and to a lesser extent their siting flexibilities. Primary applications for nuclear heat include:

- District heating and cooling
- Water desalination and purification
- Advanced oil recovery processes and oil refining
- Hydrogen production for the enrichment of liquid fuels and eventually fuel cell applications
- Advanced energy conversion processes such as coal-to-liquids and petrochemical production
- General process heat for chemical or manufacturing processes

In all cases, smaller sized, more robust reactors are better suited for integration with these applications than are large plants because of the many considerations discussed earlier.

The most logical integration of a nuclear plant with non-electrical applications is to co-locate the two facilities and dedicate the nuclear plant to the industrial facility. The energy coupling may be heat only or a combination of heat and electricity. The latter arrangement is typically referred to as either "co-generation" or as "combined heat and power" (CHP). The aspect of co-location is what gives SMRs a distinct advantage over large power plants because their smaller output is typically a better match for a single industrial plant. Co-location is further facilitated by the reduced emergency planning zone for some SMRs, as discussed in the previous section. The International Atomic Energy Agency (IAEA) has conducted a number of studies of non-electrical applications for nuclear plants, including for traditional water-cooled reactors and advanced non-water-cooled reactors. An especially thorough review on the subject for water-cooled reactors is reported in *Advanced Applications of Water-Cooled Nuclear Power Plants* issued in 2007 (IAEA, 2007). Non-water-cooled SMRs also are attractive for non-electrical applications, especially due to the higher temperature output of gas-, metal- and salt-cooled reactors.

Another operational flexibility with smaller nuclear plants is the ability to better match the plant output to the user requirements, which is especially valuable for process heat applications. Many process heat users actually require a relatively modest

amount of heat—far less than the 3,000–4,000 MWt that a large nuclear plant produces. For example, a 2008 study by Greene concluded that a nuclear plant producing approximately 100 MWt is sufficient to provide all the process heat needed for the production of liquid fuels in a state-of-the-art biorefinery (Greene et al., 2008). This modest heat demand is a result of the fact that the biorefinery size is dictated by the transportation cost associated with transport of the bulk biomass, which limits the reach of the feedstock supply to about 50 miles. Similar results were reported in a NuScale study, which found that a large commercial-scale water desalination plant producing 190,000 m³/d (50 million gallons per day) of clean water consumes between 150 and 550 MWt, depending on the choice of desalination technology (Ingersoll et al., 2014). Finally, a paper published by the European Commission's Joint Research Institute indicates that in the European industrial market, the average thermal demand per site of several industrial processes ranged between 100 and 400 MWt. This included factories for the production of various products such as chemicals, cement, iron and steel, soda ash, and alumina (Carlsson et al., 2012).

Bringing promises to reality

Despite the several benefits of smaller sized reactors and their potential to positively impact the broad acceptance of nuclear power, many technical and institutional challenges must be addressed and overcome. From a technical perspective, design innovations, such as integral primary systems and passive safety systems, will require comprehensive testing and demonstration through separate effects tests, scaled simulators, or perhaps even a prototype unit. Examples of some non-traditional components include helical coil steam generators, internal control rod drive mechanisms and internal coolant pumps. Much of the required testing has been done for the NuScale SMR but will need to be similarly demonstrated for other new water-cooled designs. In the case of the non-water-cooled designs, significant material and fuel qualification efforts may be needed in addition to new component and systems testing. Material and fuel qualification will be especially important for SMR designs that are intended to have very long fuel cycle lengths—some designs claim cycle lengths as much as 10 to 15 times longer than current LWR fuels.

A number of institutional challenges also exist for the deployment of new SMR designs. The most significant challenge will be the regulatory approval of the new designs. As mentioned in the previous section, the NRC is already working to adapt the existing licensing framework to SMRs, including those that use advanced technologies; however, there are many explicit and implicit biases in the process in favor of large light-water reactor plants. The NRC is currently working to expand the framework to be both size and technology neutral (Rempe and Williams, 2021).

Summary

In the past decade, a number of new SMR designs have emerged that promise notable improvements in safety and resilience, construction, operations and affordability, and also are well suited for a variety of energy-intensive applications. All of these benefits have the potential to significantly impact the acceptability of nuclear power. Improved safety/resilience is accomplished by lower fuel inventory, greater use of passive safety features, and the elimination of design features and components that are vulnerable to potential accidents. Lower total capital cost, extensive factory fabrication, and shorter construction time contribute to reduced investment risk and cash outlay profiles, thus addressing affordability. Finally, the smaller physical size and modularity contribute to added flexibilities in fabrication, construction and plant siting. If realized, these improvements should produce a discernable improvement on the acceptance of nuclear power from all stakeholder groups, including the general public, investors, and plant owners. In short, successful demonstration of the SMR promise provides an opportunity to significantly change the nuclear dialog.

See Also: Worldwide Status of Small Reactors Research and Technology Development.

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Floating Nuclear Plants to Improve Economics, Safety, Siting, and Proliferation Resistance

Michael Golay^a and Andrew C. Kadak^b, ^a Department of Nuclear Science and Engineering, Massachusetts Institute of Technology, Cambridge, MA, United States; and ^b Kadak Associates, Inc., Port St Lucie, FL, United States

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Introduction

Floating nuclear plants have been used since the 1950s. Nuclear submarines, aircraft carriers, and ice breakers have been powered by nuclear plants for over 60 years. So it is not unreasonable to consider such plants for commercial nuclear electricity power production. This is becoming especially important as sites for new electricity generation are becoming limited forcing companies to look at offshore locations for electric plants. Off shore wind generators have led this trend which provides a solution to visual pollution and disruption of television and other electronic signally devices which typically are reasons for siting objections to wind turbines. In the case of nuclear energy, objections by people who may not want a nuclear facility in their neighborhood is a driving force for moving these plants offshore—10–20 miles away. This distance also reduces the need for emergency planning of civil populations in the unlikely event of an accident. Offshore locations also enhance safety by providing a vast reservoir of cooling water and has security advantages due to its isolation. Additionally, factory manufacture can lower the capital cost of new nuclear construction much as it has lowered the cost of modern nuclear submarines using modularization techniques. There are also non-proliferation advantages for floating nuclear plants in that nations do not need to build enrichment or reprocessing plants since these services could be provided by the vendor or operator.

At present, Russia is the leader in developing an operable floating nuclear plant. With the launch in 2018 of the Akademik Lomonosov, a 60 MWe floating nuclear plant now operating in Pevek, a rural Siberian community, they have shown the feasibility of floating nuclear plants ([World Nuclear News, 2019](#)). This paper reviews current thinking about such plants discussing advantages and disadvantages leading to the conclusion that such plants offer many countries a flexible option to use clean nuclear energy safely avoiding many public objections ([Boyd, 2019](#)). Russia claims to have 70 orders for their Lomonosov type plants and China intends on building 20 floating nuclear plants. Presently countries developing floating nuclear plants, in addition to Russia and China, are the United States, Korea, and recently, Denmark.

History of floating nuclear plants

One of the first applications of floating nuclear plants was in 1967 with the installation of a nuclear reactor on the Sturgis (MH-1),¹ a converted Liberty ship, to supply electricity to the Panama Canal Zone ([Horizon Environmental Services, 2019](#)). This project started in 1961 with the installation and testing of a 10 MWe pressurized water reactor at Ft. Belvoir. This project was one of several

¹Mobile High Power.

conducted by the US Army Nuclear Power Program to employ nuclear energy at remote locations to supply power and heat for their outposts (US Core of Engineers, 2019). The ship was towed to the Panama Canal Zone to supply power to the Gatun Lake pumps to control water flow. The ship provided power from 1968 to 1978 during which it was refueled five times with low enriched uranium fuel. The Sturgis has since been decommissioned (Wikipedia, 2020d; Horizon Environmental Services, 2019) (Fig. 1).

The second most serious effort at designing and building floating offshore nuclear plants occurred in the United States in 1969 with the development of the Atlantic Generating Station whose concept was to use a factory much like a ship yard to build a full size 1150 MWe Pressurized Water Reactor on a 400 ft square barge drawing 30 ft which would be floated out to a location 2.8 miles off the New Jersey shore surrounded by a break water (Wikipedia, 2020e). An artist rendering of the concept showing twin barges is shown below.

Public Service Electric and Gas teamed up with Westinghouse and Tenneco to build this plant in a factory in Jacksonville Florida (Touran, 2020). The company formed was called Offshore Power Systems which filed an application with the Nuclear Regulatory Commission to manufacture these floating plants. A significant accomplishment was the issuance of a “manufacturing license” ML-1 for the factory to produce as many standardized floating plants as the market would order (Nuclear Regulatory Commission, 1982). This was revolutionary for an industry that was characterized by uniquely building each plant as “stick built” requiring extensive regulatory review. The goal of Offshore Power Systems was to take advantage of standardized mass production in a factory setting to offset on site construction costs and avoid siting disputes on the limited and highly populated east coast (Fig. 2).

As can be seen in the figure, the floating nuclear plant barges are moored behind massive breakwaters in 30 m of water. These breakwaters were designed for wave heights of 43 ft, 156 mph sustained hurricane winds and 300 mile/h tornados. Electricity would be cabled under water to the shore using AC power to a base that would be used to support plant operation several miles out to sea.

The design proposed also had significant safety advantages since the location in the ocean provided what is referred to as an “ultimate heat sink” for cooling water in the event of an accident. Its location offshore also provided additional margin of safety without requiring public evacuation in the unlikely event of a core damage accident. The Westinghouse designed plant was also supplied with a “core catcher” as an additional level of protection in the unlikely event of a meltdown preventing the release of radioactive materials.

The plants were first announced in 1971 with an initial order of two followed by another two in 1973. Unfortunately during this period the US was confronted with an Arab oil embargo and declining energy demand. In 1978 PSEG cancelled all 4 units despite the issuance of a manufacturing license in December of 1978. This license has since expired in 1999. The breakthrough of using a manufacturing license is important for future small modular reactors being considered in the United States.

The interest in floating nuclear plants has seen a recent resurgence as shown on Table 1. Many of these plants are still in the conceptual design stage. As can be seen, Russia is the leader using its nuclear ice breaker technology.

What problems could floating nuclear plants solve

Offshore floating nuclear plants are aimed at providing a flexible power source to remote communities, and nations, especially those that might not have the regulatory or technical infrastructure. In addition, they can avoid land siting problems, improve economics, provide additional safety margin and avoid national proliferation problems. They can provide a safe clean platform for varying size reactors to power cities without needing land resources. They have an overall increase in the level of safety, and hopefully public acceptance. They offer the opportunity for increased quality of construction and fabrication in factory environment. As land resources for power generation become scarce, developed nations might find such floating plants as suitable alternatives for bulk power production near load centers but offshore.



Fig. 1 Sturgis first floating nuclear plant. Wikipedia (2020d) MH-1A. Retrieved from Wikipedia: <https://en.wikipedia.org/wiki/MH-1A>.



Fig. 2 Proposed Atlantic Nuclear Generation Station. Artist Rendering of PSEGC Design. Timeline.

From the generators point of view, the expectation is reduced cost of construction as a result of the shipyard environment with economies of production offsetting economies of scale. In addition, this onshore factory would provide the essential services needed for each nuclear plant in a central location, further reducing costs. Much of the cost of nuclear construction goes to civil and structural work which would be eliminated saving hundreds of millions of dollars for each plant. The advantages of standardization can also be fully realized in such a setting. These plants can be located where needed and when needed adding additional flexibility. They can be a significant part of the climate change solution for developing nations as well since the plants can be operated by the company selling the plants who would also be responsible for training, maintenance and refueling with the host nation only paying for the power. This could be very advantageous to developing nations as they seek to improve the quality of life for their people and avoiding proliferation concerns.

The bottom line goal of floating nuclear plants is to avoid local land use siting objections and challenges, gain public acceptance due to improved safety and reduce costs by building in central shipyards for world wide deployment.

Concept of floating nuclear plants

Floating nuclear plants are generally considered to be in the small modular reactor class of new nuclear plants. A proposal by researchers at the Massachusetts Institute of Technology is focused on designing an offshore platform that can house a full size 1000 MWe reactor plant among other sizes to address large load center needs. These plants depend on advanced manufacturing and assembly in factories instead of massive on site construction programs. Through such mass production, costs should be lowered allowing smaller reactors to be in a better position to be competitive with larger plants. In addition, these floating plants are typically barge mounted not requiring large civil structures which are typically found in land based plants. Most of the floating plants can be floated back to central maintenance or refueling facilities for periodic maintenance further lowering the operating cost by using common facilities. The offshore or undersea locations also provide additional safety due to the ocean being a source of cooling water should the plant suffer a loss of coolant accident. The greatest attribute seems to be one of siting since most opposition to nuclear plants comes when a local site is selected. Much as offshore wind turbines have reduced opposition to their siting, floating nuclear plants offshore might gain similar acceptance.

One of the major impediments to new nuclear deployment is the high cost of construction to meet detailed technical safety requirements. Small modular reactors, on a per unit basis, are more costly than larger plants due to the larger plant's economy of scale—more power for the footprint. To offset this disadvantage floating nuclear plants, like small modular reactors, rely on the economies of mass production to reduce the cost in addition to manufacturing in a central factory. The avoidance of many geological studies, site licensing processes, and civil structural work offer huge savings in time and money. It is estimated that at least 20% of the capital cost of a nuclear plant is associated with site preparation and civil works ([World Nuclear Association, 2020](#)). Also using the shipyards as central service centers for major maintenance is a savings in on site facility and operating costs. The barge only needs to be floated to the central location and returned when ready.

Review of existing designs

Countries currently active in the development of floating nuclear plants of many different types include the Russia, France, China, Korea and the United States and recently Denmark. These will be briefly described in this section.

Table 1 Summary of civilian offshore nuclear plants built or designed.

<i>Name</i>	<i>Reactor type and power rating</i>	<i>Operating conditions</i>	<i>Notes and references</i>
ABV—6M (Russia)	VVER Thermal power—38 MW Electric power—8.6 MW	Design pressure, — 15.7 MPa Design temperature — 330 °C Steam generation capacity—248 t/h Fuel material and enrichment—UO ₂ , 19.7% Superheated steam pressure /temp.— 3.14 MPa/290 °C	Integral reactors with 100% natural circulation in the primary circuit; modular design for both land-based and floating nuclear power plants
KLT—40S (Russia)	VVER Thermal power — 150 MW Electric power — 35 MW	Design pressure — 12.7 MPa Design temperature — 316 °C Core coolant flow rate — 2600 t/h Coolant temperature at core inlet/outlet — 280/317 °C Fuel material and enrichment — UO ₂ , <20% Superheated steam pressure/temp.— 3.8 MPa/290 °C Steam generation capacity — 2 × 240 t/h	Serial modular reactors for nuclear icebreakers and sea vessels; floating nuclear power plant
RITM—200 (Russia)	VVER Thermal power — 175 MW Electric power — 50 MW	Design pressure, — 15.7 MPa Steam generation capacity — 248 t/h Fuel material and enrichment — UO ₂ , <20% Superheated steam pressure/temp.— 3.82 MPa/295 °C	Integral reactor with forced circulation in the primary circuit for nuclear icebreakers
VK—300 (Russia)	BWR Thermal power — 750 MW Electric power — 250 MW	System pressure — 6.9 MPa System temperature — 285 °C Fuel material and enrichment — UO ₂ , 4.0%	Natural circulation in primary circuit
VBER—300 (Russia)	VVER Thermal power — 917 MW Electric power — 325 MW	System pressure/temp.— 12.7 MPa/316 °C Fuel material and enrichment — UO ₂ , 4.95%	Modular reactor based on marine propulsion reactor technologies for land-based and floating nuclear power plants
Atlantic generating station—Offshore nuclear power plant (United States)	PWR Electric power — 1150 MW Thermal output — 3425 MW	Operating pressure — 2250 psia Containment design Pressure — 15 psi Reactor inlet and outlet temp. — 557.3 °F/616.9 °F Fuel material and enrichment — UO ₂ , 3.2% No of coolant pumps—4 Total reactor flow — 142.1 × 10 ⁶ lb./h	Ice condenser containment; large breakwater; twin units
FLexBlue underwater power plant (France)	PWR Electric power—50–250 MW Thermal power—150–750 MW	System pressure — 15.5 MPa System temperature — 310 °C Fuel material and enrichment—UO ₂ , 3.1%	At conceptual design stage: autonomous operation, seabed, torpedo resistant
Offshore nuclear power plant (South Korea)	PWR Electric power—1400 MW × 2 units	Standard APR1400 PWR nuclear island with passive seawater-based containment and core cooling systems	Plant housed in concrete/steel structure resting on seabed, suitable only for shallow waters

Russia Akademik Lomonosov

As shown on Table 1, Russia has a number of such marine nuclear applications in addition to the ice breaker KLT 40S. This reactor design was used for their floating reactor barge plant—The Akademik Lomonosov which was launched in 2018 and placed in operation in 2020 at Pevek, Russia, a remote town in northern Siberia. An artist rendering of the plant is shown below. The actual photograph of the plant is shown as well (Power Technology, 2020) (Figs. 3 and 4).

This barge actually looks like a ship but needs to be towed to its desired location typically in a port. It is 140 m long with a 30 m beam and draft of 5.56 m. It displaces 21,500 t with a crew size of 70 people for operation and maintenance. The barge has two 35 MWe KLT reactors sufficient to power a city of about 200,000 people for 40 years. Additionally the ship can supply 300 MWth of heat to the community and can be converted to supply 240,000 L of fresh water each day.

France—FLEXBlue

The FLEXBlue plant has been under development in France for many years. This plant is being designed by DCNS² for undersea deployment to support on shore facilities (Gourmel et al., 2016). It is basically a submarine without engines carried or towed to



Fig. 3 Artist rendering of Akademik Lomonosov. <https://www.crediblecarbon.com/news-and-info/news/marine-nuclear-plant-leads-the-way-to-the-future/>.



Fig. 4 Actual photo of Akademik Lomonosov. Lev Fedoseyev—Lev Fedoseyev/TASS.

an undersea location where they are anchored to supply needed power. It is designed for autonomous operation with minimal

²Direction des Constructions Navales (DCNS) is predominately a French state owned defense contractor that designs, builds, maintains and decommissions French Navy ships. Their specialty is design of nuclear submarines and aircraft carriers. Over the last 50 years they have made more than 17 such vessels.



Fig. 5 FLEXBlue in transit and on location.

maintenance crew when needed. Each module is capable of producing from 50 to 250 MWe using light water reactor technology. An artist rendering of the plant is shown below (Fig. 5).

These modules are roughly 150 m long with a 14 m diameter. The reference module is 160 MWe Pressurized Water Reactor with two steam generators and four canned rotor main coolant pumps. These plants can be linked to provide more power to onshore facilities. They can be anchored in 50–100 m of water a few kilometers from shore. A preliminary safety analysis was performed on this design by DCNS (Haratyk and Gourmel, 2015) which demonstrates that all safety functions can be passively operated without operator actions or electrical power.

There are several goals of this design, the most significant of which is to avoid public objections from placing these plants on land near people while also having the ability to site plants in areas that have high energy demand and where land is scarce or unsuitable for nuclear construction due to the presence of natural or manmade hazards such as other industrial facilities. The subsea location also provides additional safety benefit for having essentially an infinite source of cooling water in the event of an accident. It would be less prone to terrorist attacks. As in other floating reactor concepts, the advantage of factory manufacture, mass production and no major civil engineering structures or massive containments should lower capital costs.

This small modular reactor is designed for a 40 months operating cycle after which it is removed and transported back to a central base for refueling. In its place, another previously refueled module is transported to the site to maintain power capability. It is expected that every 10 years the modules are shipped back for major maintenance. What is interesting in this design is that the reactors are remotely operated.

Korea—BANDI

South Korea has recently signed a memorandum of understanding with South Korea's KEPCO Engineering & Construction Company and Daewoo Shipbuilding & Marine Engineering to develop floating nuclear plants (World Nuclear News, 2020) Korea is a likely candidate for such plants due to their unavailability of suitable shore based sites for new construction. Shown below is an artist rendering of their proposed plant (Fig. 6).

These companies hope to leverage their experience in nuclear reactor design and ship building to construct a floating small modular reactor. Kepco has been working on BANDI-60, a small modular reactor SMR since 2016. This reactor is a 60 MWe reactor which they call a "block" type pressurized water reactor since they have eliminated main coolant pipes in an effort to avoid large break loss of cooled accidents. (Kim et al., 2019) Another unique innovation is to eliminate the use of soluble boron for reactor control replacing it with an in-vessel control rod drive mechanism and top mounted core instrumentation.

The target market for the BANDI 60 is to provide power and heat to remote communities and water desalinization with the additional goal of supporting the variability of renewables on the grid.

China

China has two competing floating nuclear plant designs underway to provide power in the South China Sea to support offshore oil and gas operations. They also have recently been building islands which will require power (Lempriere, 2018). Two companies are developing small pressurized water reactors based on their land based designs. The Nuclear Power Institute of China (NPIC) which is a subsidiary of the China National Nuclear Corporation (CNNC) is developing the ACP100 S which is to be a marine version of the ACP-100 (CNNC to Develop Floating Reactor, 2016). While the details are not known due to secrecy concerns, it appears that the reactors are capable of producing 310 MWt of heat and 100 MWe of power. It has 57 fuel assemblies 2.15 m tall using integral steam generators much like US versions of similar integral nuclear plants.

An artist rendering of this plant is shown below. It is not clear whether this is simply a barge or a motorized ship which contains the reactor (Fig. 7).



Fig. 6 Korean BANDI floating nuclear plant.



Fig. 7 Nuclear Power Institute of China's Floating Nuclear Plant. <https://www.world-nuclear-news.org/Articles/CNNC-to-construct-prototype-floating-plant>.

The second floating nuclear plant is being developed by China General Nuclear Power Group (CGN) is a 200 MWt/60 MWe pressurized light water reactor (CGN to Build Floating Reactor, 2016). It's based on the ACPR50S design employing 37 fuel assemblies providing steam using two primary coolant loops feeding 4 steam generators. CGN has signed an agreement with China National Offshore Oil Corporation.

An artist rendering of the CGN floating plant is shown below (Fig. 8):

The present status of these projects is unknown.

Offshore floating nuclear plant (MIT)

Another interesting concept being developed by researchers at MIT is the Offshore Floating Nuclear Plant (OFNP) which builds on successful offshore oil rig platforms. The MIT team designing the OFNP is made up of seasoned marine industry companies familiar with ship construction and undersea operations. The MIT team has consulted with Newport News Shipbuilding, a US builder of nuclear submarines and air craft carriers, and Sevan Marine, a Norwegian designer of offshore floating oil storage barges. The fundamental idea is similar to other floating designs but on a much larger scale in that a full size 1100 MWe pressurized water reactor could be installed in one of the modules rather than the smaller 60 MWe plants currently being proposed by others (Buongiorno et al., 2016). Other options do include smaller reactors in the 300 MWe such as those small modular reactors being considered by some utilities and countries to meet their specific needs. What is unique in their concept is that the floating platform can be designed to hold any nuclear plant design. Shown on the figure below is a typical rendering of their plant (Fig. 9).

The proposal is to install the reactor in a floating platform similar to that designed by the Sevan Corporation (Sevan Floating Production Storage and Offloading, 2020) for their Floating Production Storage and Offloading platforms (Wikipedia, 2020,c).



Fig. 8 China General Nuclear (CGN) Floating Nuclear Plant. <https://www.world-nuclear-news.org/Articles/CGN-to-build-floating-reactor>.



Fig. 9 MIT Offshore Nuclear Plant. Buongiorno J, Jurewicz J, Golay M and Todreas N (2016) The offshore floating nuclear plant concept. *Nuclear Technology* 194: 1–14.

These platforms are built in shipyards and are routinely used in the offshore oil industry. One of the unique advantages of this system is that they are extremely stable platforms for nuclear plants able to endure storms, offshore tsunamis and earthquakes (Fig. 10).

The MIT design calls for building these platforms in a shipyard and installing proven licensed pressured water reactors or other reactor types into the vessel in a controlled environment. The platform would then be shipped or towed to a location 20 km offshore and anchored at the location from which 345 KV AC cables would be used to transport the power to a land location for distribution (Fig. 11).

The uniqueness of this design is that it can be moored virtually anywhere in relatively deep water (100 m) which would address earthquake, hurricane or tsunami risks. The plant would be self-sufficient in that refueling would occur at sea and when its useful life of 60 or so years is completed, it could be floated back to the shipyard for decommissioning, much like the US fleet of nuclear ships and submarines. The figure below shows what such a nuclear plant might look like being transported to an offshore site (Fig. 12).

The design team has developed the concept to allow for a visual of the layout which is shown below.

The design calls for placing all safety critical equipment in water tight compartments. As in oil platforms, the operating staff will have living quarters with monthly shift changes. Minor maintenance would be conducted at sea with a major outage every 10 years at which time the platform would be towed back to a central maintenance facility. Refueling would be conducted at sea every 12–48 months since an on-site spent fuel pool would be located on the platform capable of holding all the spent fuel for the 40 year expected life of the facility. To supply freshwater, the plant would also have a desalinization plant (Fig. 13).



Fig. 10 Sevan offshore oil storage platform being shipped. <https://sevanssp.com/sevan-fps0/#>.

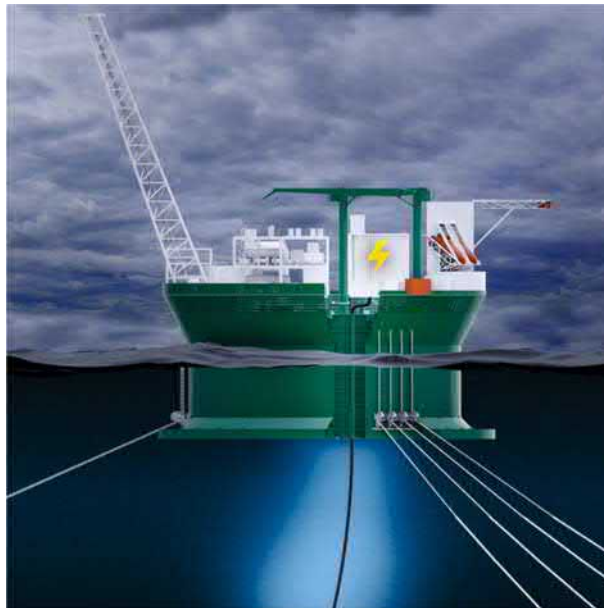


Fig. 11 Rendering of anchorage system of offshore floating nuclear plant. <https://sevanssp.com/sevan-electrified-fps0/>.

The feasibility of this concept is best demonstrated what is already being deployed worldwide using the Sevan design. Their floating platforms for oil storage and operation are 110 m tall, 90 m in diameter with a draft of 30.5 m. The expectation is that existing reactor designs such as the AP-1000 can be inserted into these platforms in a shipyard (Fig. 14).

As with other floating nuclear plants, the safety is ensured by the use of the ocean as the ultimate heat sink to remove decay heat in the event of an accident. Numerous safety studies have been performed on this design showing not only the feasibility of the anchoring system during storms but also using conventional safety analysis (Zhang et al., 2018), (Strother, 2015). The defense in depth principles are shown in the figure below. What is noticeably different from land based plants is the lack of a 3–4 ft thick reinforced concrete structure needed for loss of coolant accident pressure control and security against aircraft impact. This is a significant savings in capital cost (Fig. 15).

Seaborg floating nuclear plant—Denmark

A surprising new entrant into the floating nuclear plant market comes from Denmark, a historically anti-nuclear country. What is unique about their design is that instead of using light water reactors, they have chosen a molten salt fueled reactor of 250 MWth/100 MWe with barge capacities ranging from 100 to 600 MWe. (Jaspen, 2020) Molten salt reactors (Ignatiev, 2021) use a fluoride salt to dissolve thorium fuel so that the core is essentially circulating liquid fuel. Seaborg's design is unique in that the core consists



Fig. 12 Artist rendering of OFNP being transported. Buongiorno J, Jurewicz J, Golay M and Todreas N (2016) The offshore floating nuclear plant concept. *Nuclear Technology* 194: 1–14.

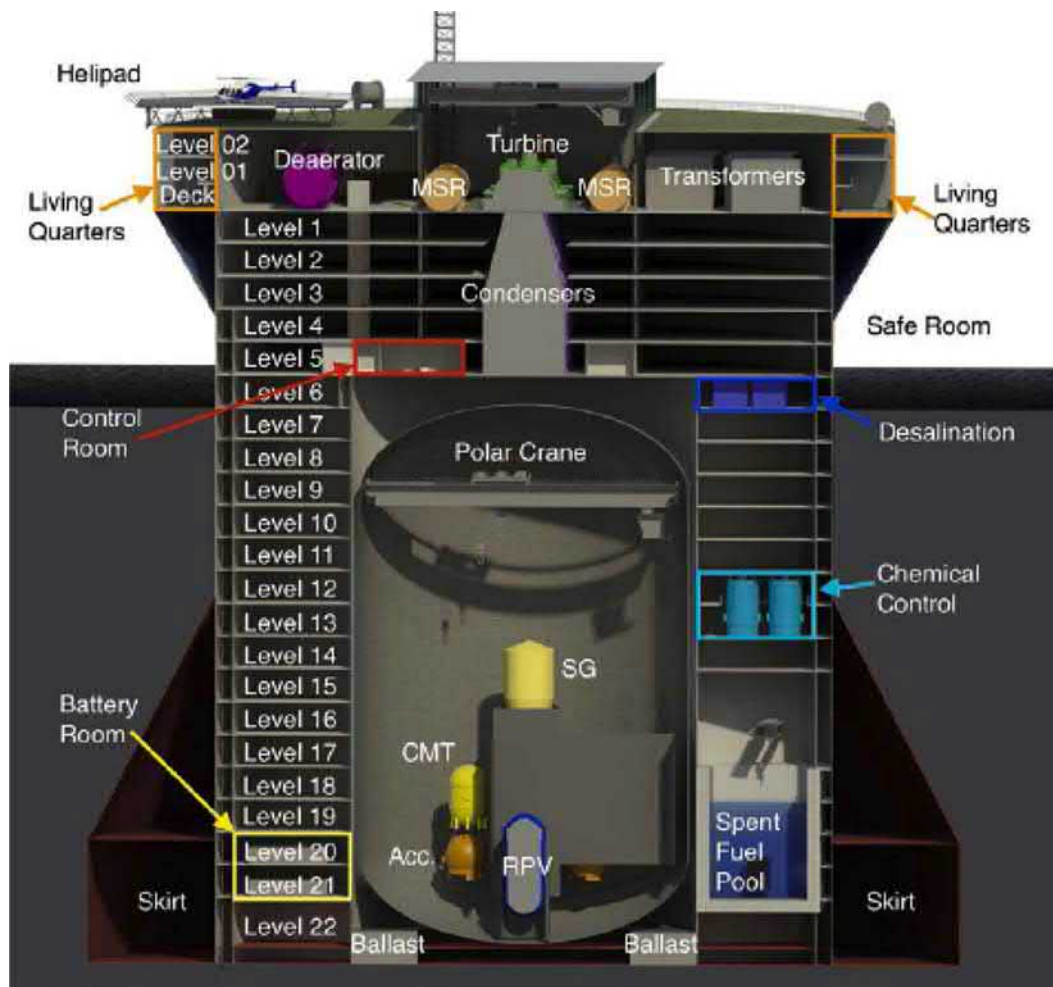


Fig. 13 Design platform for OFNP. Buongiorno J, Jurewicz J, Golay M and Todreas N (2016) The offshore floating nuclear plant concept. *Nuclear Technology* 194: 1–14.

of a collection of tubes through which the fuel flows surrounded by a molten salt moderator likely sodium hydroxide (Lovschall-Jensen, 2019) (Fig. 16).

These are higher temperature reactors that operate at 600 °C but at atmospheric pressure which is an advantage. Few such reactors (especially thorium) have operated especially on a commercial scale which suggests a great deal of



Fig. 14 Shipment of offshore oil storage platform. Sevan Floating Production Storage and Offloading (2020) Retrieved from Sevan SSP: <https://sevanssp.com/sevan-fps0/#>.

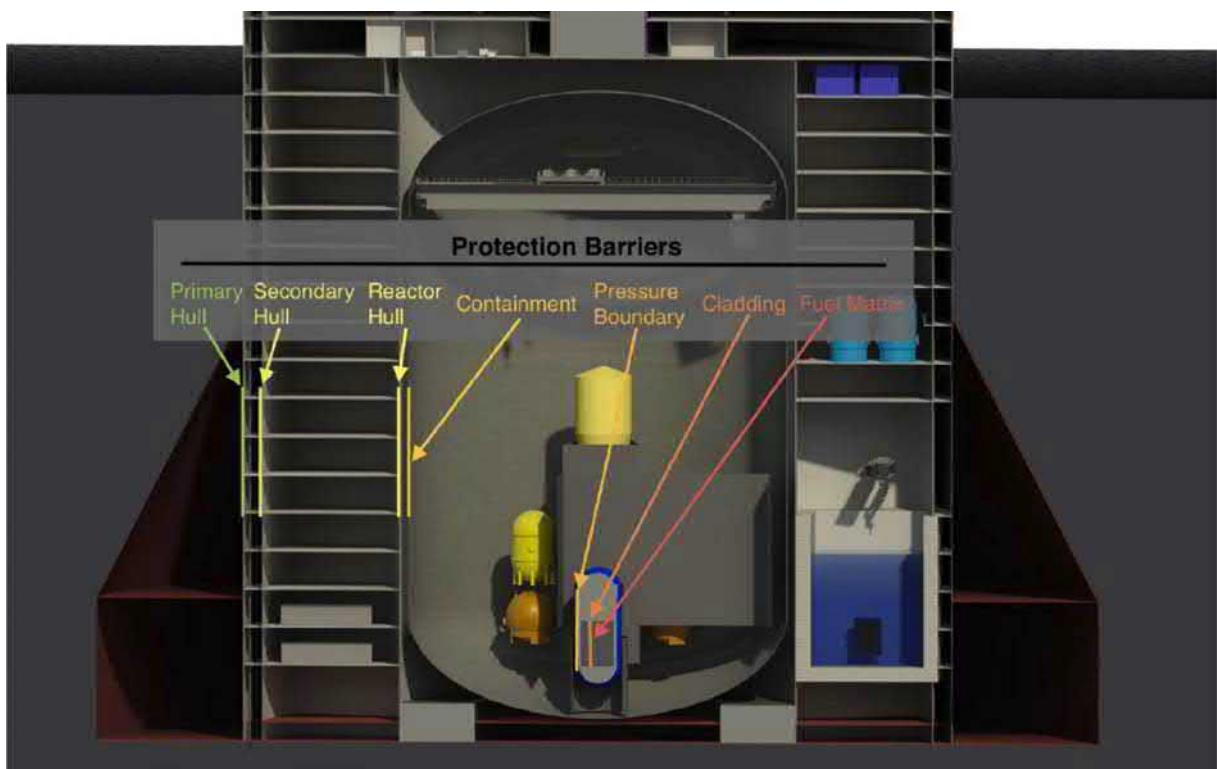


Fig. 15 Defense in depth barriers. Buongiorno J, Jurewicz J, Golay M and Todreas N (2016) The offshore floating nuclear plant concept. *Nuclear Technology* 194: 1–14.

development will be required. They might start with a uranium core to allow for earlier deployment. The challenge is dealing with the corrosion due to the molten salts on metals. The expectation that the Seaborg will have an operating cycle of 12 years before refueling.

Shown on Fig. 17 below is an artist rendering of the Seaborg CMSR in an arctic location providing heat and power to such a remote location. This concept is very early in development funded by private investors.



Fig. 16 Core design for Seaborg MSR. Lovschall-Jensen AE (2019) Making Nuclear Sustainable. <https://www.dualports.eu/wp-content/uploads/2019/06/Seaborg-making-nuclear-sustainable.pdf>.



Fig. 17 Seaborg CMSR floating nuclear plant concept. Lovshell-Jensen AE (2019) Making Nuclear Sustainable. Sweden. Retrieved from <file:///C:/Users/kadak/Desktop/Seaborg-making-nuclear-sustainable.pdf>.

Other concepts

There are other floating nuclear plant concepts that are being investigated mostly by Russia based on their experience with marine propulsion systems (International Atomic Energy Agency, 2013). The most notable is a lead/lead bismuth cooled fast reactor which was used in early Russian Alpha class submarines however not very successfully. All have been retired. The SVBR 100 was a joint venture with AKME Engineering. The 100 MWe plant was to have an operating cycle of 7–8 years before refueling with uranium dioxide, mixed oxide or nitride fuel. In all, the Russians have about 80 reactor years of experience with these reactors (Fig. 18).

Another Russian concept is the ABV-6M which has a power rating of 8 MWe (Bakhmetyev et al., 2015) This reactor is being designed under the leadership of OKBM with the goal of autonomous operation. It is an integral pressurized water reactor with steam generators in the reactor vessel designed for a fuel cycle of 10–12 years without refueling. It is to be barge mounted with twin units (Fig. 19).

Proliferation and security concerns

A concern about deployment of many such plants around the world relate to the risk of nuclear proliferation and security. Depending on the operational model, these floating plants could be operated as a fleet by responsible companies or nations ensuring not only the safety of operation but the ability to keep the spent nuclear fuel in a secure state. Commercial nuclear plants use low enriched uranium, 4–5%, which make it impossible for a nuclear weapon which needs over 93%. The spent nuclear fuel, once its useful life for power production is over, does contain a mix of radioactive materials including plutonium 239 which is fissile.

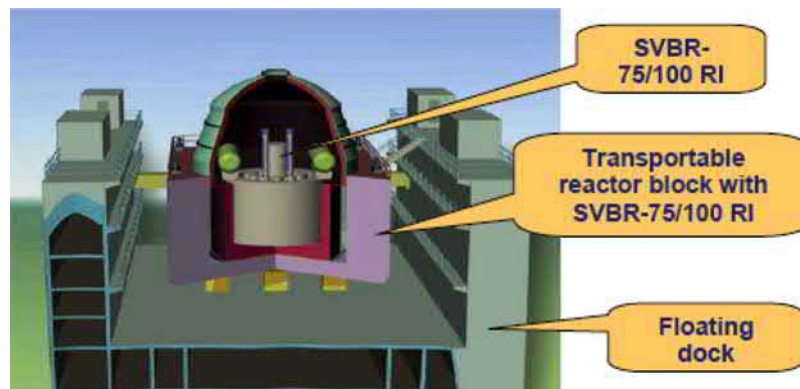


Fig. 18 Russian SVBR floating reactor barge concept. International Atomic Energy Agency (2013) *Legal and Institutional Issues of Transportable Nuclear Power Plants: A Preliminary Study*. Vienna: IAEA.

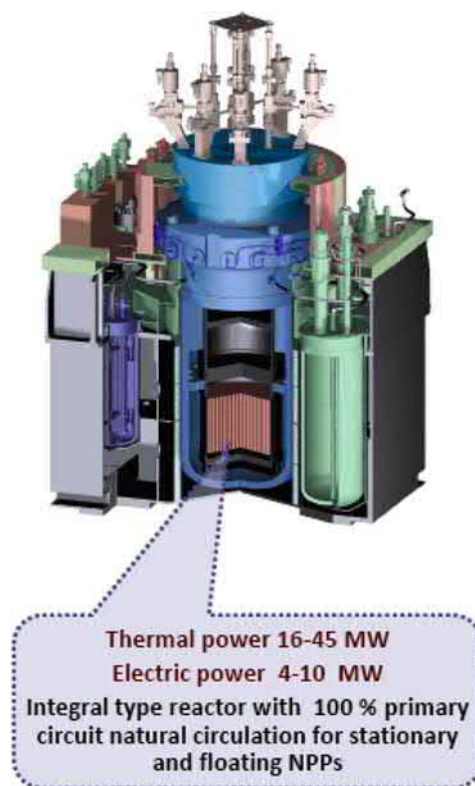


Fig. 19 ABV-6M Russian design. (Bakhmetyev A, Bylov I, Lepekhin A and Fadeev Y (2015) *Small Modular Reactors Engineering Solutions for Safety Provisions*. OECD/NEA Workshop on Innovations in Water Cooled Technologies: Issy-les-Moulineaux.)

However, in order to obtain sufficient quantities for a weapon, a complex chemical reprocessing process is required which would be easily detected should some nation decide to pursue a nuclear weapon via plutonium extraction from irradiated fuel. In addition, the International Atomic Energy Agency which inspects international nuclear facilities for proliferation concerns would also apply their protocols to these floating plants. Thus, from a proliferation point of view, commercial nuclear plants themselves are quite proliferation resistant.

What makes the floating nuclear plants attractive from a non-proliferation perspective is that nations do not need to have uranium enrichment plants that can be a direct path to making high enriched uranium nor would they need reprocessing plants, and still have the benefits of clean nuclear energy. The best model for use of floating nuclear plants from a non-proliferation perspective is that the supplier of the plants would provide all services from operation, spent fuel storage and disposal at their (Ignatiev, 2020) central facility (Mckinzie, 2021). Some of the floating nuclear plants have life time storage of spent fuel capability while others require the transportation of the barge back to the central facility after a number of years of operation. During transportation of the barges, there is some risk of interception but given the security protocols that would be in place, the likelihood of a successful

theft of the spent fuel is very low. At present, Sweden, France, England, Japan, and the United States have shipped spent fuel and fresh fuel safely by sea for over 40 years.

It is expected that some nation or international consortium could possibly provide all such fuel cycle services and waste disposal solutions (Hore-Lacy, 2020). Thus, under such a model, these plants provide a non-proliferation alternative for the use of nuclear energy which would avoid the need for enrichment and reprocessing plants.

From a security perspective, clearly offshore locations make access to the plant much more difficult. As an example, security has been considered in the MIT design as shown the by figure below which establishes several security zones for access and monitoring. As in all nuclear plants, a security team working from defensive positions would be stationed on the platform to monitor and repel any intrusion attempt (Fig. 20).

Objections and disadvantages

The political pressure group, Greenpeace, has objected to the use of the Akademik Lomonosov in Siberia for several reasons beyond their general dislike of nuclear energy despite the reluctant admission that it will help reduce carbon emissions in these remote communities. They argue that due to its remote location, it will be hard to get qualified workers to staff the facility and respond to emergencies should it be needed (Cholteeva, 2020). They are concerned about the barge sinking and that spent fuel will remain on the barge for years before being removed causing a potential pollution problem. They are also concerned about the transportation of the barges to and from their moored location. The designs, such as the MIT design, have double hulls to avoid such problems and others would be shipped on other vessels such as the FLEXBlue design. While legitimate concerns, they are recognized in design and operation.

While the advantages are many, transporting such large structures over long distances (shipyards in the Southeast Asia) can be a challenge especially with changing weather conditions for such long journeys. Transporting unfueled reactors is not the concern but once returned to the factory shipyard for refueling, the risks do increase.

Licensing and regulatory challenges

As in any nuclear enterprise, regulatory compliance can be a challenge especially in the case of a floating nuclear plant traveling through international and national waters. In the case of the proposed Atlantic Nuclear Generating Station, the Nuclear Regulatory Commission not only approved the manufacturing license, it also approved the siting and shipment of the barge to a location off the New Jersey coast. This approval and process was a good precedent for a regulatory approach to a national installation. Russian regulatory authorities approved the design, construction and siting of the Akademik Lomonosov in Pevek.

Licensing barges or floating offshore platforms containing nuclear reactors would not only have to comply with nuclear safety regulations by the host country but also international rules for shipping nuclear materials. While the nuclear navies of the world are not internationally regulated for transport, at present there are no legally binding international conventions that address the floating nuclear plant options being considered. One example is how should a fueled reactor which would be tested prior to transport is to be treated under international conventions such as that expected for FLEXBlue. There have been four commercial nuclear powered cargo /passenger ships built to date, only one of which is still operating—Sevmorput (Russia) (Wikipedia, 2020a,b). Perhaps the most famous commercial ship was the NS Savannah built in 1959 by President Eisenhower under the Atoms for Peace Program. This ship visited 45 countries, 32 domestic US ports traveling over 450,000 miles. She was placed in active passenger and cargo service from 1962 until 1971 when she was retired and now is

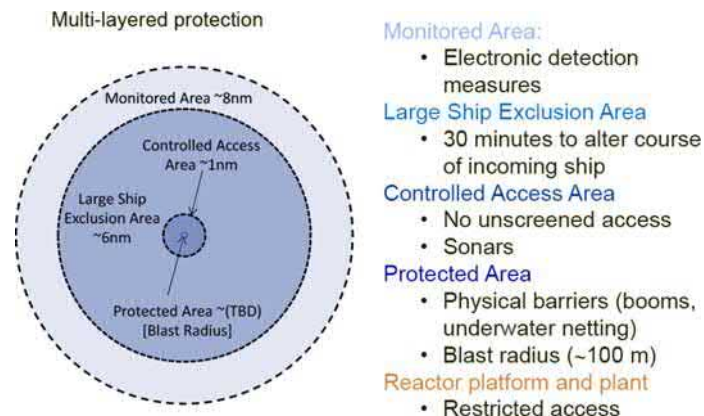


Fig. 20 Security Considerations (nm stand for nautical miles). Buongiorno J, Jurewicz J, Golay M and Todreas N (2016) The offshore floating nuclear plant concept. *Nuclear Technology* 194; 1–14.



Fig. 21 NS Savannah Nuclear Powered Commercial Ship. Wikipedia (2020a) NS Savannah. Retrieved From NS Savannah: https://en.wikipedia.org/wiki/NS_Savannah.

a National Historic site museum in Maryland. During her visits to ports around the world she hosted over 1.4 million people for tours. The Savannah experience shows that shipping commercial nuclear power plant is possible under international regulations and conventions (Fig. 21).

Shipping an unfueled reactor is much simpler since shipments of all parts of the reactor are routinely done as are shipments of fresh unirradiated fuel such as was the case with the Russian Akademik Lomonosov. However once the fuel is irradiated after operation, all would be subject to national and international regulations, some of which have not yet been agreed to. While shipments of irradiated fuel are also routinely done by ship, it should not be a more difficult challenge to gain such approvals for transit through waters of other countries. The International Atomic Energy Agency, among others, have studied this issue reviewing a number of options (International Atomic Energy Agency, 2013).

Countries accepting such floating nuclear plants must either accept the regulatory review and approval of the supplier nation or develop their own for floating plants within their territorial limits. For developing nations, this is seen as a positive feature since while they need to have a regulatory oversight function, they do not need a formal regulatory system needed to license such a facility.



Fig. 22 Summary of floating nuclear plant designs.

It is unclear how floating nuclear plants would be treated if moored outside a nation's territorial limits in international waters. International treaties covering this option would need to be negotiated.

International liability laws also need to be developed such that insurance can be provided to the developer and host country in the event of an accident during operation or during transport. These agreements need to be in place prior to any international commerce of such technologies. Land based reactors sold to countries have forms of liability protection which seem to be possible for floating plants but not yet agreed to.

Summary

Fig. 18 below summarizes the various offshore power systems being considered today. As can be seen there are many choices each with its unique advantages given the location, electricity needs and siting considerations. Other applications include moored barges protected by breakwater such as the Atlantic Offshore plant or barges resting on the seabed. The goal of all these concepts is to avoid local siting problems and challenges, standardize construction and increase efficiency and reduce costs by building in central shipyards for world wide deployment. The additional safety advantage of the sea provides an assured supply of cooling water minimizing accidental risks. Regulatory, legal and institutional challenges remain for international use of floating nuclear plants but are not seen as insurmountable (Fig. 22).

Floating nuclear plants offer the international community a safe and potentially economical source of clean energy using existing technology already deployed in both the nuclear and marine industries. Additionally, there are significant non-proliferation advantages to floating nuclear plants for many nations since they would not need a nuclear fuel cycle infrastructure. More detailed engineering is needed to implement some of the concepts but the challenges are manageable. The Russians with their first barge mounted plant and many nuclear navies around the world have shown the feasibility of offshore floating nuclear plants.

See Also: Floating Nuclear Power Plants With Small Modular Reactors; Marine Propulsion.

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Microreactors for Decentralized Power Sources

Caroline Cochran^a and Jessica R. Lovering^b, ^a Oklo, Inc., Sunnyvale, CA, United States; and ^b Carnegie Mellon University, Pittsburgh, PA, United States

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Abbreviations

HALEU High-assay low-enriched uranium
HPR Heat pipe reactor
HTGR High-temperature gas-cooled reactor
LEU Low-enriched uranium
LFR Lead-cooled fast reactor
LWR Light-water reactor
MW Megawatt
SMR Small modular reactor

Definition of microreactors and what role they have

Microreactors are generally understood to be the smallest fission reactor category, although specific definitions vary significantly. The term tends to be used more in certain parts of the world, especially the United States of America, while some parts of the world tend to consider microreactors as a subcategory of small modular reactors, occasionally termed “very small modular reactors (vSMRs)” (<https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/small-nuclear-power-reactors.aspx>). The International Atomic Energy Agency (IAEA), for instance, does not define a separate subcategory of SMRs (<https://www.iaea.org/topics/small-modular-reactors>). The interest in microreactors has grown globally with the concurrent interest in low greenhouse gas emission energy, growing global energy use, the expansion of microgrids, and the need for reliable energy supplies for critical facilities like military bases and hospitals.

As the smallest reactor class, the definition of a microreactor usually involves some kind of upper limit to the size or capacity of a reactor in order to be defined as a microreactor. The definition of microreactor has ranged from below 10 MW-electric (MWe), to below 30 MWe/100 MW-thermal (MWth) to below 50 MW of unspecified output as shown in **Table 1**. A possible reason for this variety of definitions is that microreactors will be serving different markets than large-scale nuclear reactors.

Table 1 Definitions of microreactor power ranges.

Entity	Thermal power	Electrical power	Notes
US Department of Energy (https://www.energy.gov/ne/articles/what-nuclear-microreactor)	< 20 MW		Also has described characteristics not related to power level
US Department of Defense (https://www.nei.org/CorporateSite/media/filefolder/resources/reports-and-briefs/Road-map-micro-reactors-department-defense-201810.pdf)		< 10 MW	
US Congress in the National Defense Authorization Act (https://www.nei.org/news/2018/congress-directs-doe-study-of-micro-reactor-deploy)			< 50 MW of “power production”
UK Department of Energy and Climate Change (https://assets.publishing.service.gov.uk/government/uploads/system/uploads/attachment_data/file/787411/Market_and_Technical_Assessment_of_Micro_Nuclear_Reactors.pdf)	< 100 MW	< 30 MW	

Historically, the primary output of commercial nuclear reactors has been electricity, and reactors were commonly rated by the electrical output. However, end users for microreactors may be more interested in direct heating, cogeneration of electricity and steam, or use of industrial process heat. This means that a discussion of MWth is involved more frequently in the definition of microreactors as opposed to more traditionally-sized larger plants. In general, microreactors are defined by their output, although some definitions have attempted to include other attributes, such as operational characteristics. Because the market demand and supply are still evolving, it is to be expected that there is no single clear definition which has emerged. Ultimately, at the time of writing, the most typical definition of microreactors is based on producing on the order of single-digit megawatt electrical output, i.e., less than 10 MWe although up to 50 MWe may be considered.

On occasion, radioisotope thermoelectric generators (RTGs) may be considered a subtype of microreactors. These would not be a “reactor” in a literal sense because they do not sustain a fission reaction in order to produce heat and thereby electrical power. RTGs utilize heat from isotope decay to generate power (Bennett, 2021a,b) and usually produce tens to hundreds of watts of electrical power. However, this article will focus on fission microreactors.

Near term applications

Microreactors, because of their size and technology flexibility, have a range of potential near term applications. Many research and test reactors historically have had power capacities within the range of microreactors, and there are currently about 31 such reactors in operation in the United States alone (<https://www.nrc.gov/reactors/operating/map-nonpower-reactors.html>). Microreactors also have a history of use for power and cogeneration around the world, and there are a few currently operating and currently under construction for various applications as shown in Table 2. In fact, the first privately owned and operated reactor to commercially provide power on the US grid was a 5 MWe microreactor, the Vallecitos BWR which operated in California between 1957 and 1963

Table 2 Current and near-term applications of microreactors (reactors under 50 MWe, not including research-only reactors which do not produce electrical power).

Reactor	Status	Application	Location	Type	Size
EGP-6 #2 (http://rosenergoatom.ru/stations_projects/sayt-bilibinskoy-aes/)	Currently operating	Cogeneration power	Bilibino, Siberia	Graphite moderated boiling water reactor (BWR)	12 MWe
EGP-6 #3 (http://rosenergoatom.ru/stations_projects/sayt-bilibinskoy-aes/)	Currently operating	Cogeneration power	Bilibino, Siberia	Graphite moderated BWR	12 MWe
EGP-6 #4 (http://rosenergoatom.ru/stations_projects/sayt-bilibinskoy-aes/)	Currently operating	Cogeneration power	Bilibino, Siberia	Graphite moderated BWR	12 MWe
China Experimental Fast Reactor (CEFR)	Currently operating	Cogeneration power	Tuoli, China	Sodium-cooled fast reactor	20 MWe
CAREM-25 (https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/small-nuclear-power-reactors.aspx)	Under construction	Electrical power	Argentina	Integral PWR	27 MWe
KLT-40S (https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/small-nuclear-power-reactors.aspx)	Under construction	Electrical power	Russia	Pressurized water reactor (PWR)	35 MWe
RITM-200 (https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/small-nuclear-power-reactors.aspx)	Under construction	Electrical power	Russia	Integral PWR	50 MWe

(<https://www.asme.org/wwwasmeorg/media/resourcefiles/aboutasme/who%20we%20are/engineering%20history/landmarks/128-vallecitos-boiling-water-reactor-1957.pdf>).

Because there is increased interest worldwide, there are also a number of commercial microreactor designs in development. For these newer commercial offerings, it is expected that early units will have higher costs than some mainline markets within the United States, and so the near term applications are likely to be locations where either the cost for current electricity supply is high, the logistics for electricity supply is precarious, or the need for reliable power is significant; or a combination of all three. Longer term, microreactors have the potential to achieve cost competitiveness with mainline grid prices throughout the world.

Microgrids are a natural application for microreactors. This may include artificially created microgrids, i.e., where an end user may be connected to a large grid but prefers its own supply, or where an end user purchases a certain level of service from a large provider that exceeds the reliability or guarantee of the larger grid as a whole. Examples of microgrids which represent near term applications for microreactors include remote residential, commercial, or industrial sites, resorts, campuses, or military bases.

Specific types of residential, commercial, industrial and military sites that may find particular benefit by the low emissions, reliability, and resilience of this power source include emergency power supply to large nuclear power plants, some data centers, large hospitals, mining and other resource production sites, military bases, island communities, and communities in far northern or southern areas of the world with few energy options.

Emission-free power in the far north

Communities in far northern areas around the world do not have ready access to energy. In many areas, transportation of fuel may only be possible in certain times of the year. Due to the northern geography, solar power may be ineffective for significant portions of the year. Similarly, hydropower resources may not be available in times of the year. There is a history of use of microreactors in northern areas in Siberia in Russia, where four reactors producing approximately 12 MWe have been operating since the 1970s. These reactors also cogenerated process heat. Other northern areas, such as the northern territories of Canada as well as the US state of Alaska, have explored the use of microreactors. Current remote microgrid fossil fuel capacity installed in Canada is about 750 MWe (<https://atlas.gc.ca/rced-bdece/en/index.html>), and in Alaska there is approximately 150 MWe of installed capacity (<https://www.eia.gov/state/?sid=AK>).

Emergency power backup to existing plants

There has been interest to utilize microreactors as emergency power backup for existing large nuclear plants. Licensing a very small additional reactor on a site already prepared for a large reactor should present a relatively manageable challenge, while the benefit would allow for the site to have a robust source of power available to the larger reactor should it be needed. There is interest in improving the startup time and reliability for emergency diesel generators for large plants, but if a power source is already onsite and already powered up, this would be of use for the plant safety and reliability as a whole.

Resilient power for defense applications

Another particular application which has received attention particularly in the United States is the use of microreactors to improve energy independence and resilience of military bases and other critical assets. Both the United States and Russia have had microreactors for defense purposes built and tested or operated in the mid-20th century, although the interest has been renewed in recent years with modern technologies. The US Department of Defense spends billions of dollars a year on energy. Ninety percent of US military installations have an average annual energy use that can be met by an installed capacity of nuclear power of 40 MWe or less (<https://www.nei.org/CorporateSite/media/filefolder/resources/reports-and-briefs/Road-map-micro-reactors-department-defense-201810.pdf>).

Global developing microgrids and new markets

Most broadly, microreactors could present a valuable supply of emission-free and reliable power to any microgrid. Worldwide, there is more than 22,500 MW of installed remote power sources utilizing fossil fuels, and more than two-thirds of these are located in developing and emerging countries (<https://reiner-lemoine-institut.de/en/global-analysis-of-the-utilization-of-diesel-generators-for-remote-power-supply/>). Another estimate puts the global market demand at 1000 GW (i.e., 1,000,000 MWe) (<https://www.esi-africa.com/news/hybrid-micro-grids-beginning-of-the-end-for-diesel-gensets/>). This global market size, as well as the market need, is only expected to grow. The specific fraction of this need which could be met by microreactors could depend on a number of factors. According to the UK Department of Energy & Climate Change, there will be a potential global accessible market for microreactors of up to 2850 MWe by 2030 (https://assets.publishing.service.gov.uk/government/uploads/system/uploads/attachment_data/file/787411/Market_and_Technical_Assessment_of_Micro_Nuclear_Reactors.pdf). That said, there are areas of the world where markets have not even been allowed to develop because neither fossils nor renewables may be practical. It is in these emerging and niche markets that microreactors may create wholly new means to supply energy to markets that are not yet in existence.

Unique benefits and risks of microreactors

While microreactors offer many of the same benefits and risks as traditional large-scale nuclear reactors, their tiny size and simplified engineering also provide unique benefits as well as risks. Many of the benefits will depend on the specific reactor technology and fuel cycle, although some are universal to such small designs. Similarly, inherent risks are dependent on the deployment model and safety features of each design.

Benefits

The main benefit of microreactors is lower overall cost. While the cost to generate electricity may be higher than a large light-water reactor per kWh, the total cost is low enough to open up new markets. For example, traditional nuclear power plants could only be owned and operated by the largest utilities in the United States, and typically operated by state-owned or state-supported utilities outside of the United States. If total project costs for a microreactor plant are in the tens of millions range, they could be purchased by small utilities, rural electric coops, and municipal utilities. In addition, they could be financed by private investment instead of relying on federal loan guarantees, investment tax credits, or other state support. Similarly, shorter and predictable construction duration is another expected benefit of microreactors over large-scale nuclear power plants. After the first demonstration projects, customers should be able to order a microreactor with confidence that it will be delivered on a specified time range, measured in tens of months not tens of years. Beyond more predictable construction schedules, factory-fabricated reactors should also come with fixed, predictable costs for customers.

Most microreactors are non-LWRs that come with a range of unique safety benefits. Yet, the very small size of microreactors allows further inherent safety due to simplified engineering and a reliance on physics rather than engineered safety systems. Depending on the design, microreactors are able to eliminate many of the redundant and active safety systems, relying more on natural physical properties to maintain reactor control. This not only lowers costs, but also results in reactors with reduced risk of accidents, and minimized need for operator intervention in the case of an accident. Additionally, the reduced risk allows for a much smaller Emergency Planning Zones—the area around a nuclear power plant where owners must prepare emergency evacuation plans in the rare event of an accident—meaning that microreactors could be located closer to populations centers where the electricity is needed.

Both the small size and non-LWR designs also allow microreactors to significantly reduce or even eliminate their water consumption especially those that use direct cycle gas turbines. This has economic, safety, and environmental benefits. On the economic side, customers do not have to worry about securing access to a reliable water supply for a potential power plant, which also opens up the market to inland and arid regions. On the safety side, operators do not have to worry about supplying consistent water to cool the reactor in the event of an accident. And on the environmental side, these designs reduce the consumption of fresh water from ecosystems, as well as reduce thermal pollution from warm water returning to aquatic ecosystems.

While the first commercial microreactor projects will likely be off-grid locations, many currently on-grid communities are looking to take ownership of their power generation. This movement towards distributed and decentralized power generation is normally associated with renewable energy, but declining costs of microreactors in the longer term could open up this potential market. Microreactors would offer these communities reliable, low-carbon electricity at a long-term fixed rate.

Lastly, large-scale nuclear power is valued on the power grid for its reliability. Microreactors would open up this valuable attribute to smaller off-grid and microgrid markets. For example, many US defense installations require back-up diesel generators to maintain critical operation in the event of a failure of the central grid. Microreactors could provide consistent power for mission critical operations or the entire installation, depending on size. Additionally, nuclear can provide islanding and black start capabilities, allowing the grid to be isolated during an emergency event or turned back on after a power outage. Currently, these services are only available from fossil fueled generators.

Risks

The specific risks from nuclear power are well-understood, and microreactors do not seem to come with many unique risks. However, most microreactor vendors expect lower staffing levels than required for regulated facilities which are hundreds of times larger. And many developers tout use of automatic controls for reactor operation. Some have proposed use of remote operation. Such capabilities provide many economic benefits, but some capabilities pose new risks. For instance, remote operation will create a cyber-security risk which is not present in existing LWRs which utilize analog controls and are not connected to the internet or a network offsite. Since many of the regulations governing existing nuclear power plant safety and physical security are based on requiring safety systems to actively function to ensure safety, and for well-trained staff to initiate those active systems, designs will need to show that passive or inherent safety or active controls have lower levels of error than human error, if the controls affect safety or security of the plant. This challenge may be straightforward, but may make some potential customers and regulators wary.

Some proposed uses of microreactors include requirements for transportability and minimizing weight which allows for minimal amounts of shielding; for instance, the recent US DoD solicitation of information targeted designs weighing less than 40 tons, able to operate for at least 3 years, in the range of 1–10 MW ([FedConnect, n.d.](#)). This use capability would have new risks or challenges regarding ensuring dose levels remain below safe limits for those in forward-operating areas.

Current development efforts of microreactors

While the concept of microreactors is not new, there is renewed interest in commercializing such designs that has led to a relative boom in concepts. In addition, new US federal programs aimed at advanced nuclear technologies may be accelerating commercialization of new nuclear technologies. Among the list of microreactors put forward, many are scaled-down versions of larger reactors the vendor ultimately hopes to build, meaning the microreactor could be a commercial demonstration of a new technology that is the first step toward licensing of a larger advanced reactor design. Yet other vendors are specifically developing a microreactor as their final product. While Russia and China are also developing microreactor concepts, their focus for advanced reactors appears to be at larger scales and thus are excluded from the following descriptions. **Table 3** provides a summary of the most mature microreactor concepts for US and Canadian markets.

All vendors are US-based unless otherwise noted. Reactor technology acronyms: *HTGR*, high-temperature gas-cooled reactor; *LFR*, lead-cooled fast reactor; *HPR*, heat pipe reactor; *LWR*, light-water reactor. Fuel terminology: Low-enriched uranium (LEU) less than 5% U235, high-assay low-enriched uranium (HALEU) is enriched to 5%–20%.

Looking at the microreactors under development in **Table 3**, all of the designs are non-LWR with the exception of NuScale. All developers anticipate using HALEU, with the exception of NuScale and General Atomics. Gas-cooled reactors may be the most common in part because the US Department of Defense laid out a requirement for a transportable reactor to utilize TRISO fuel, and many developers are in competition for this solicitation. There are also a few molten salt designs (either in the primary or secondary cooling loop). There are two sodium-cooled designs and one lead-cooled design, which are likely taking advantage of experience with sodium-cooled designs globally. Many developers are planning to use TRISO fuel, which has some history of commercial-scale fabrication. Several vendors claim load-following capabilities and autonomous operation.

DOE's Gateway for Accelerated Innovation in Nuclear (GAIN) program began offering vouchers in 2016 which allow for payments directly from DOE to the national laboratories, and enable advanced nuclear developers to direct research and access experimental facilities and expertise at national laboratories (<https://gain.inl.gov/SitePages/Nuclear%20Energy%20Vouchers.aspx>). Several microreactor developers have been awarded these vouchers, detailed below. DOE has also awarded small grants for specific research and development projects to several advanced reactor developers.

On the demonstration side, there are several programs in the United States and Canada helping to accelerate first-of-a-kind (FOAK) demonstrations. In the United States, there have been several studies and a recent request for information from DOE for a pilot program to deploy microreactors at federal sites, such as DoD and national laboratories. In 2016, Canadian Nuclear Laboratories put out an invitation for SMR developers to submit proposals to build a demonstration reactor at their Chalk River site, with the goal of commissioning before 2026. This has spurred several companies to initiate the pre-licensing, four-stage vendor design review with CNL (CNL, n.d.).

Details on specific microreactor concepts in development

What follows are more details on where each vendor stands on commercialization of their microreactor concept, including specific funding and licensing progress. Companies are in alphabetical order.

General atomics

General Atomics has a long history with building very small reactors as one of the leading vendors for research reactors globally. Their energy multiplier module is a 4–10 MWe high-temperature gas-cooled reactor that has a unique breed and burn configuration (Qvist, 2021), allowing it to run on low-enriched uranium (LEU) while also having a long core lifetime (30 years). They claim the reactor can load-follow, operate mostly autonomously, and be transported in a standard shipping container (<http://www.ga.com/advanced-reactors>).

As GA's design relies on silica carbide fuel, much of their development work is focused on qualifying this novel fuel. In 2018, they received \$2.2 million to test a new method of qualifying fuels that could reduce the cost and schedule. They received an

Table 3 A summary of microreactor concepts farthest along in development.

Vendor	Technology	Capacity	Core lifetime (years)	Fuel type and enrichment
General Atomics	HTGR	4–10 MWe	30	1% U235, silica carbide fuel rods
HolosGen	HTGR	13 MWe	12–20	Up to 15% HALEU
LeadCold (Sweden)	LFR	3 MWe	10–30	19.9% UO ₂
NuScale	HPR	1–10 MWe	> 10	HALEU
Oklo	HPR	1.5 MWe	< 20	Low-enriched metal fuel
Ultra Safe	HTGR	5 MWe	20	9%–12% TRISO
Urenco (UK-Dutch-German)	HTGR	10 MWth	5–10	TRISO (mix of UO ₂ and ThO ₂)
Westinghouse Evinci	HPR	0.2–15 MWe	> 10	HALEU oxide
X-energy	HTGR pebble bed	10 MWth	~ 10	HALEU TRISO

additional \$0.4 million to engage with the NRC on a pre-application license review of their carbide fuel (<https://www.energy.gov/articles/secretary-energy-rick-perry-announces-60-million-us-industry-awards-support-advanced>) (Fig. 1).

HolosGen

Based on technology originally developed for airborne reactors at Los Alamos, HolosGen is developing a gas-cooled fully mobile microreactor. The design is scalable from 3 to 13 MWe, cooled by Helium or Carbon Dioxide, with fuel enriched up to 15% U235. The design has an integrated balance-of-plant, making the entire power plant fit inside a standard shipping container (<http://www.holosgen.com/>) (Fig. 2).

LeadCold nuclear

Sweden-based LeadCold is designing a 3–10 MWe lead-cooled fast reactor for remote arctic applications. The fuel is uranium oxide enriched to 19.9%, allowing a core lifetime of 10–30 years (<https://www.leadcold.com/sealer.html>). LeadCold has submitted a 55 MWe reactor of similar design for review to the UK's Department for Business, Energy & Industrial Strategy. They have also submitted an Expression of Interest to Canadian Nuclear Laboratories' invitation for vendors to build and operate an SMR at their site. They began Phase 1 of pre-licensing vendor design review in January 2017 with the Canadian Nuclear Safety Commission, but their review is currently on hold at their request (<http://nuclearsafety.gc.ca/eng/reactors/power-plants/pre-licensing-vendor-design-review/index.cfm>) (Fig. 3).

NuScale power

NuScale, the clear front-runner for SMR light-water reactors in the United States for their 50 MWe SMR design, is also developing a heat pipe reactor under 10 MWe that will use TRISO fuel. This new design will have a smaller footprint and longer core lifetime than their 50 MW LWR reactor, but further details are not publicly available (Fig. 4).

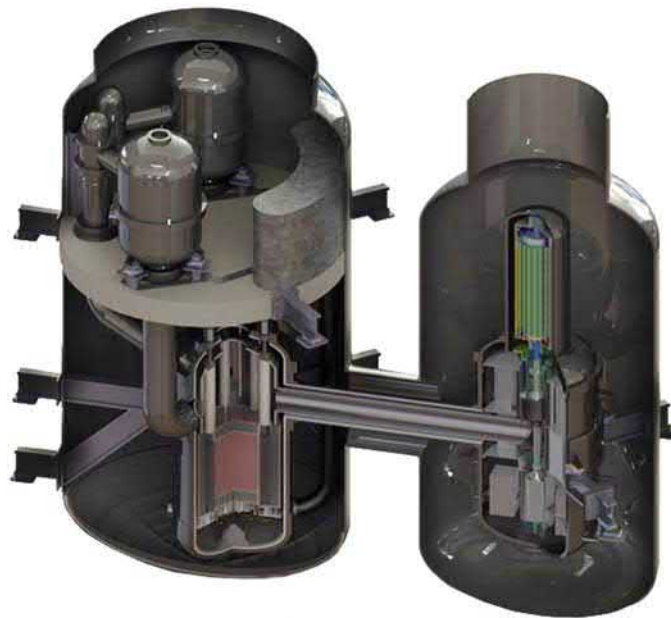


Fig. 1 Artist rendering of General Atomics' Energy Multiplier Module (EM²). On the left is the reactor module and on the right is the energy conversion module. Photo Credit: General Atomics, <http://www.ga.com/advanced-reactors>.



Fig. 2 Artist rendering of the Holos reactor in transport. Photo Credit: HolosGen, <http://www.holosgen.com/generators/>.



Fig. 3 Mockup of LeadCold's SEALER-Arctic reactor. Photo Credit: LeadCold, <https://www.leadcold.com/sealer.html>.

Oklo

Oklo is a company based in California, and they have long been a front-runner in micro-scale advanced fission development. In November of 2019, Oklo launched their first product, the Aurora powerhouse. In December of 2019, Oklo received a site use permit to from the US Department of Energy to build their microreactor at the Idaho National Laboratory (INL) ([Business Wire, Oklo Inc., 2019](#)). In February of 2020, Oklo announced that INL selected Oklo to demonstrate fuel reuse in the Aurora powerhouse. In March of 2020, Oklo submitted a Combined License Application to the US Nuclear Regulatory Commission, making them the first non-LWR design to submit such an application ([Bandyk, 2020](#)). The Aurora powerhouse utilizes solar panels and a 1.5 MWe advanced fission battery that relies on heat pipes to minimize moving parts. They are looking to deploy the Aurora powerhouse in off-grid communities that currently rely on diesel, particularly in Alaska ([Fig. 5](#)).

Ultra safe nuclear corporation

Ultra Safe Nuclear is the front-runner in Canada's pre-licensing review process. Their design is gas-cooled in the primary loop, but the secondary cooling loop is molten salt, allowing more flexibility in thermal storage (<https://usnc.com/MMR.html>). Ultra Safe began pre-licensing vendor design review with the Canadian Nuclear Safety Commission in December 2016 and was the first—and so far only—SMR vendor to begin Phase 3 of the review (<http://nuclearsafety.gc.ca/eng/reactors/power-plants/pre-licensing-vendor-design-review/index.cfm>). Ultra Safe has also received \$400–\$800 thousand annually since 2016 from the Small Business Innovation Research program, mostly for research on space-based nuclear thermal propulsion systems (<https://www.sbir.gov/sbc/ultra-safe-nuclear-corporation>) ([Fig. 6](#)).

Urenco

Urenco is one of the largest uranium enrichment companies, supplying about 30% of the world's enrichment services. In 2008 they announced a challenge to develop a microreactor that functioned like a battery for use at remote mining sites. The University of Manchester (UK) and Delft University of Technology (NL) collaborated to develop the winning project which is now called U-battery (<https://www.u-battery.com/>). The U-battery is a 10 MWth high-temperature gas-cooled reactor with a lifetime of 5–10 years. They plan to use TRISO fuel that is a mix of UO_2 and ThO_2 . Urenco has partnered with several large industrial firms to complete and commercialize the design. Urenco submitted their application for Phase 1 of vendor design review to the Canadian Nuclear Safety Commission in February 2017 ([Fig. 7](#)).

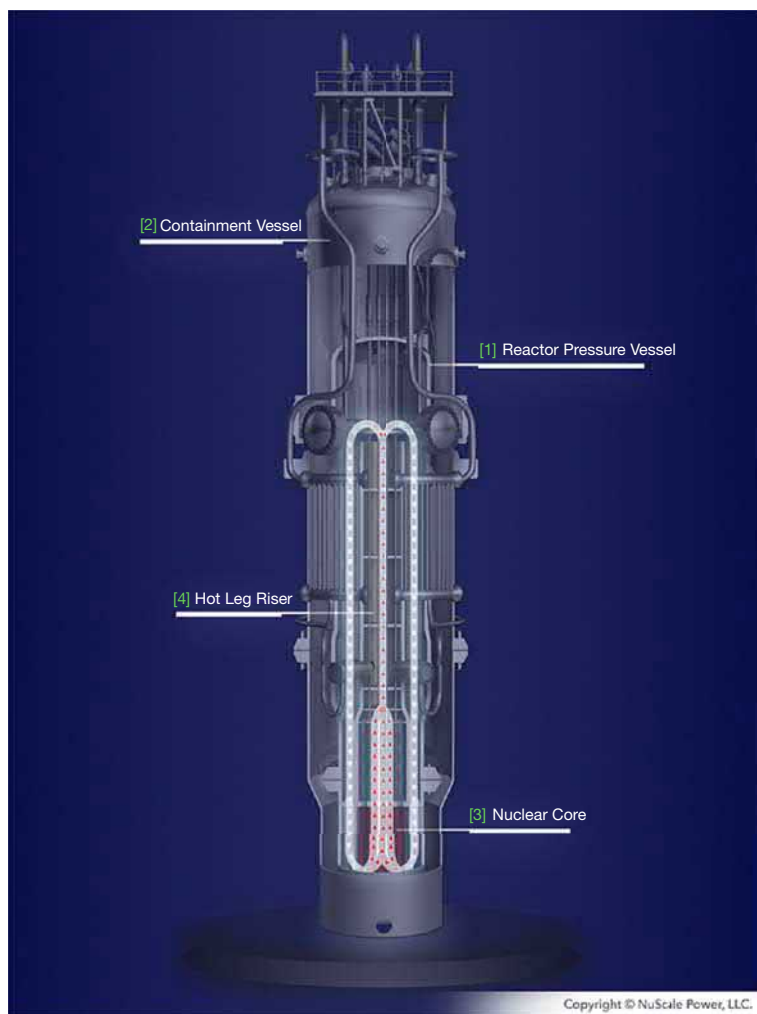


Fig. 4 Schematic of NuScale Power's reactor concept. Photo credit: <https://www.nuscalepower.com/technology/design-innovations>.

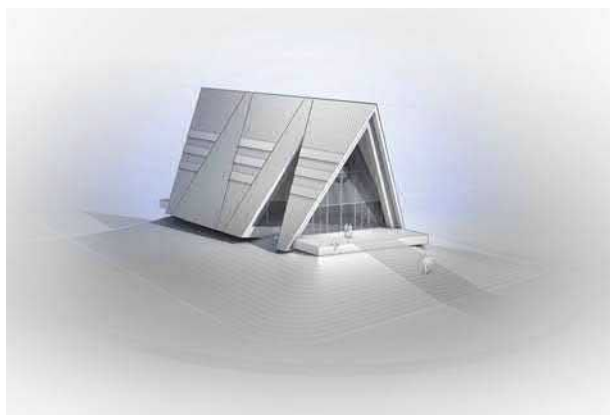


Fig. 5 Site rendering of Oklo's 1.5 MWe Aurora powerhouse. Credit: Oklo Inc. and Gensler.

Westinghouse

While Westinghouse has many reactor concepts in development, their microreactor is based on the megapower reactor developed at Los Alamos National Labs, which was a scaled-up version of a space-based reactor concept. The design, called eVinci, is radically simplified, relying on heat pipes to minimize moving parts and allow mostly autonomous operation (<http://www>.

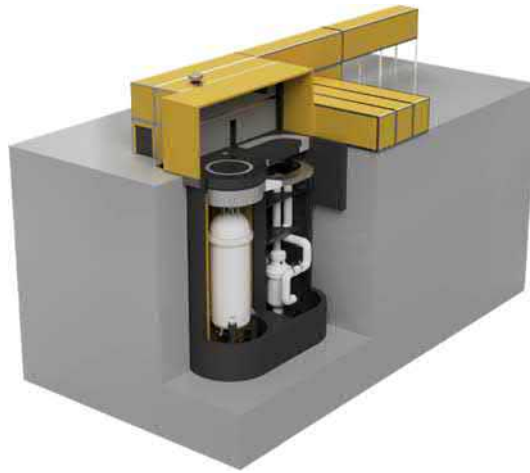


Fig. 6 Ultra Safe Nuclear Corporation's reactor concept. <https://usnc.com/MMR.html>.



Fig. 7 Urenco's U-battery design. Photo credit: <https://www.u-battery.com/>.

westinghousenuclear.com/new-plants/evinci-micro-reactor). The design is scalable from 200 kW to 15 MWe with a core lifetime of over 10 years. They are planning to use a HALEU oxide fuel. In 2019, Westinghouse received \$13 million from DOE to prepare the eVinci design for demonstration by 2022 (<https://www.energy.gov/ne/articles/us-department-energy-further-advances-nuclear-energy-technology-through-industry-awards>). Westinghouse contributed another \$15 million toward this effort. In February 2018, Westinghouse submitted their application for Phase 2 of vendor design review to the Canadian Nuclear Safety Commission (<http://nuclearsafety.gc.ca/eng/reactors/power-plants/pre-licensing-vendor-design-review/index.cfm>) (Fig. 8).

X-energy

X-energy is developing a 10 MWth high-temperature, gas-cooled pebble bed reactor. The design will be 10 MWth with a core lifetime of about 10 years. The company has been very focused on scaling up the commercial supply chain of UCO TRISO fuel pebbles (<https://x-energy.com/reactors/>). X-energy began pre-application licensing activities with the NRC in September of 2018 (<https://www.nrc.gov/reactors/new-reactors/advanced/xe-100.html>). DOE granted X-energy \$4.5 million in 2018 to design and license a TRISO fuel fabrication facility. X-energy contributed \$4.5 million of their own to this project (Fig. 9).

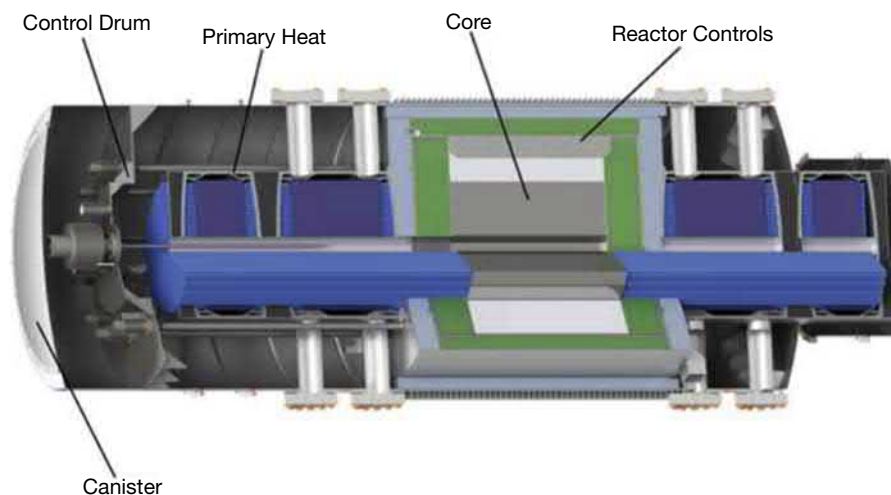


Fig. 8 Schematic for Westinghouse's eVinci reactor. Photo credit: "eVinci™ Micro Reactor" by Yasir Arafat, and Jurie Van Wyk, Westinghouse Electric Company LLC. <http://www.westinghousenuclear.com/Portals/0/new%20plants/evincitm/eVinci%20Micro%20Reactor%20NPJ%20M-A%202019.pdf?ver=2019-04-30-211410-367>.



Fig. 9 X-energy's reactor concept; Reactor module on left and energy conversion module on right. Photo credit: <https://www.x-energy.com/nuclear-reactors/>.

Summary

Microreactors have the potential to be a disruptive nuclear technology due to their unique attributes and niche market opportunities. These designs are moving faster through the commercialization process than many anticipated just 5 years ago, and policy may need to play catch up with where the technology is moving. The very small size of microreactors not only reduces the total cost, but

allows for real factory fabrication along with consistent production schedules. The improved safety profiles of such small, non-LWR technologies can also simplify operations and reduce staffing levels on-site.

The lower absolute cost combined with the reliability of nuclear energy could open up new markets for microreactors. These very small reactors could soon be powering off-grid communities at northern latitudes or island cities. Customers that require high-reliability from the power sources—such as military bases, large hospitals, or off-grid mining sites—could also benefit from commercial microreactor technology.

Policy-makers should begin the process of developing supportive policies to accelerate commercialization of these designs, but also level the playing field in the power sector to allow microreactors to compete with fossil fuels and cooperate with renewable energy to expand low-carbon generation.

See Also: Micro-Reactors.

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Non-Proliferation Aspects of Nuclear Energy

L.-Y. Cheng and R.A. Bari, Brookhaven National Laboratory, Upton, NY, United States

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In December 1942, the Chicago Pile-1 (CP-1) demonstrated the feasibility of self-sustaining nuclear chain-reaction. However, the public realization of the enormous power of nuclear energy was not until the detonation of a nuclear weapon in August 1945. Ever since the dawn of the nuclear age, the potential for peaceful uses of nuclear energy often raised concerns that the potential spread of nuclear weapons would be catalyzed by a civilian nuclear power program. There is also the possibility of theft of fissile material or nuclear technology by non-state entities or terrorists. Though nuclear power reactors and nuclear weapons operate under different conditions, they rely on a similar scientific basis and use similar materials and technologies. This article discusses some of the activities and characteristics related to curtailing and deterring proliferation associated with worldwide deployment of nuclear energy; that is, the non-proliferation aspects of nuclear energy from the perspective of policy, regulation and design.

The early years

Leo Szilard, a Hungarian-German-American physicist and inventor, filed for a patent in England on the concept of neutron-induced nuclear chain reaction in 1933 (granted in 1936) while the mechanism of nuclear fission was not discovered until 1938. In December 1939, Szilard drafted a letter signed by Albert Einstein to President Franklin D. Roosevelt, alerting the President to the potential for a vast amount of power that could be released from a nuclear chain reaction in uranium. The Szilard-Einstein letter prompted the birth of the Manhattan Project that ushered in the nuclear conundrum – the potential for harnessing the power of nuclear energy as a weapon and also as an alternate source of energy. For more information on this topic see [Reed \(2021\)](#).

An early attempt to manage nuclear energy after World War II was suggested in the Report on the International Control of Atomic Energy, written by a committee chaired by Dean Acheson and David Lilienthal in 1946 (aka the Acheson-Lilienthal Report). It proposed methods for the international control of nuclear weapons and the avoidance of future nuclear war. The report also recommended that all fissile material be controlled by an international agency and small amounts would be released to individual nationals for peaceful uses of nuclear energy.

In the United States, the first domestic legislation to control nuclear energy was enacted in the Atomic Energy Act of 1946. The Act established the United States Atomic Energy Commission for the purpose of transferring nuclear weapons development and nuclear power management from military to civilian control. The Act established a US policy to prevent the spread of nuclear technology by secrecy and denial. Private development of nuclear energy did not start until the Act was amended under the Eisenhower administration's Atoms for Peace Program in 1954. The Program actively promoted peaceful applications of nuclear technology, establishing new scientific curricula and sending equipment to schools, hospitals and research institutions within the US and throughout the world. The first nuclear reactors in several countries were built under the Program. The enactment of the Atomic Energy Act of 1954 was a watershed moment in the civilian nuclear industry. It amended and refined the Atomic Energy Act of 1946 by enabling private companies access to technical information about production of nuclear energy and fissile materials and allowing for greater exchange of information with foreign nations as part of the Atoms for Peace Program. The 1954 Atomic Energy Act adopted the new goals of international cooperation and the development and utilization of atomic energy for peaceful purposes as a vehicle to support non-proliferation by averting further activities in the development of independent nuclear programs. Section 123 of the US Atomic Energy Act of 1954 authorized the conclusion of Agreements for Cooperation with other countries. An Agreement for Cooperation, commonly known as "123 agreement" established conditions for non-proliferation commitments and safeguards for nuclear technology supplied by the United States, including nuclear reactors, enrichment and reprocessing facilities and special nuclear material. The Atomic Energy Act of 1954, as originally drafted, did not prohibit

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cooperating partner countries, through indigenous effort, develop nuclear weapons or other nuclear-related military applications. Strengthening of non-proliferation requirements and compliance progressed with the formation of the International Atomic Energy Agency (IAEA) in 1957 and the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) that entered into force in 1970. The NPT represents a new international regime to control and verify that the sharing of nuclear technology does not contribute to nuclear proliferation either directly or indirectly. An online monograph (Rosenthal et al., 2019) provides a more comprehensive introduction to the safeguards systems and the history of the International Atomic Energy Agency. For more on various subjects related to IAEA safeguards see the “IAEA Safeguards Glossary” (IAEA, 2001).

Special nuclear material

A crucial step in the making of a fissile weapon is to acquire fissile material necessary to reach a critical mass (the minimum amount of fissile material needed to sustain a nuclear chain reaction). All currently deployed nuclear weapons are based on uranium enriched in the isotope U-235 or plutonium with high isotopic content of Pu-239. Both U-235 and Pu-239 are fissile nuclides in nuclear fuel for power reactors. Fresh (unirradiated) light water reactor (LWR) fuel typically has a U-235 enrichment of 3–5%. Neutron capture in U-238 (the dominant isotope in LWR fuel) breeds Pu-239. U-233 is a fissile isotope bred from thorium-232 by neutron capture in nuclear fuel based on thorium. While it may be possible to weaponize U-233 there is no public information on its actual implementation. For more information on fissile materials see Shusterman (2021).

The usefulness of nuclear material for use in a nuclear weapons program is commonly referred to as material attractiveness. From a safeguards perspective not all nuclear material is alike. Material attractiveness generally refers to key attributes of the material, such as isotopic content, chemical form, etc. Examples of categorization schemes for use in coarse comparison of material attractiveness include the US DOE material attractiveness categories I–IV for nuclear materials in DOE nuclear facilities (US DOE Order 474.2) (USDOE, 2015) and the IAEA safeguards categories described in the Safeguards Glossary (IAEA, 2001). Other schemes, such as the figures of merit (FOM), have been advanced to delineate the attractiveness or preferences for a range of nuclear materials in advanced nuclear fuel cycle (Bathke et al., 2012).

Technical difficulties and time required to build a nuclear weapon from using certain nuclear material are two general attributes for characterizing the material attractiveness. Relevant material properties for consideration are the type, quantity and quality of material. Two intrinsic features of nuclear material that are key in the IAEA safeguards categories are conversion time and significant quantity. Conversion time (IAEA, 2001) is the estimated time required to convert different forms of nuclear material to the metallic components of a nuclear explosive device. Significant quantity (IAEA, 2001) is the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded and it includes unavoidable losses from conversion and manufacturing processes. As a general rule-of-thumb the attractiveness of nuclear material increases with shorter conversion time and smaller significant quantity. In terms of quality the IAEA distinguishes direct versus indirect use material. Direct use material is nuclear material that can be used for the manufacture of nuclear explosive devices without transmutation (the conversion of an isotope of one element into an isotope of another element through one or more nuclear reactions) or further enrichment. Indirect use material is all nuclear material which must be further processed (e.g. transmutation or enrichment) to produce direct use nuclear material. Table 1 based on Tables I and II in Section 3.13 of the IAEA safeguards glossary (IAEA, 2001) summarizes attributes of direct and indirect use nuclear materials.

Table 1 Direct and indirect use nuclear materials.

<i>Beginning material form</i>	<i>Conversion time</i>	<i>Significant quantity (SQ)</i>
<i>Direct use nuclear materials</i>		
Metal: Pu, HEU ($^{235}\text{U} \geq 20\%$), or ^{233}U	Order of days (7–10)	Pu ^a : 8 kg Pu
PuO ₂ , Pu(NO ₃) ₄ or other pure Pu compounds;	Order of weeks (1–3)	^{233}U : 8 kg ^{233}U
HEU or ^{233}U oxide or other pure U compounds		HEU ($^{235}\text{U} \geq 20\%$): 25 kg ^{235}U
MOX or other non-irradiated pure mixtures containing Pu, and U ($^{233}\text{U} + ^{235}\text{U} \geq 20\%$)	Order of weeks (1–3)	
Pu, HEU and/or ^{233}U in scrap or other miscellaneous impure compounds	Order of weeks (1–3)	
Pu, HEU or ^{233}U in irradiated fuel	Order of months (1–3)	
<i>Indirect use nuclear materials</i>		
<i>Beginning material form</i>	<i>Conversion time</i>	<i>Significant quantity (SQ)</i>
U ^b containing $< 20\%$ ^{235}U and ^{233}U	Order of months (3–12)	75 kg ^{235}U (or 10 t natural U or 20 t depleted U)
Th	Order of months (3–12)	20 t Th

^aFor Pu containing less than 80% ^{238}Pu .

^bIncluding low enriched, natural and depleted uranium.

The above table categories Pu and ^{233}U in metal form to have the highest attractiveness of nuclear materials, as they have the smallest SQ and shortest conversion time, followed by high enriched uranium (HEU, i.e. $^{235}\text{U} \geq 20\%$ by weight) metal that has a similar conversion time but a larger SQ. Direct use nuclear materials also include mixed oxide (MOX is a nuclear fuel that contains more than one oxide of fissile material, such as Pu recovered from used nuclear fuel mixed with depleted uranium) and non-irradiated pure mixtures containing Pu and $^{233}\text{U} + ^{235}\text{U}$ enriched to above 20%. Other parameters of nuclear materials that characterize their attractiveness or non-attractiveness for weaponization include critical mass, internal heat generation, dose rate and spontaneous fission (Bathke et al., 2012).

The nuclear fuel cycle

The nuclear fuel cycle of a reactor represents the progression of nuclear fuel from creation to disposal. It involves front-end steps to manufacture the fuel for use in nuclear reactors and back-end steps to manage and handle used or spent fuel. The following overview examines the non-proliferation aspects of the key elements of the nuclear fuel cycle. The discussion uses the uranium fuel cycle, Fig. 1, for illustration (see also Crawford (2021)).

At the front-end of the nuclear fuel cycle the enrichment stage is of primary proliferation concern. The bare critical mass of uranium metal has a direct correlation with its level of enrichment in ^{235}U (natural U contains about 0.7% ^{235}U), as depicted in Fig. 2 (Bari et al., 1996). The calculated critical mass for bare uranium spheres of various enrichments using the MCNP Monte Carlo code and its associated cross-section library show that there is a significant increase in critical mass for enrichments below 20%, a value consistent with the threshold for HEU. Appendix A in Bathke et al. (2012) presents the calculated bare critical mass for 40 actinide isotopes.

Nuclear weapons typically contain uranium enriched to 80% ^{235}U or more, which is known as weapons-grade uranium. It is a matter of more SWU (separative work unit, a quantitative measure of the work that need be invested in an enrichment process)

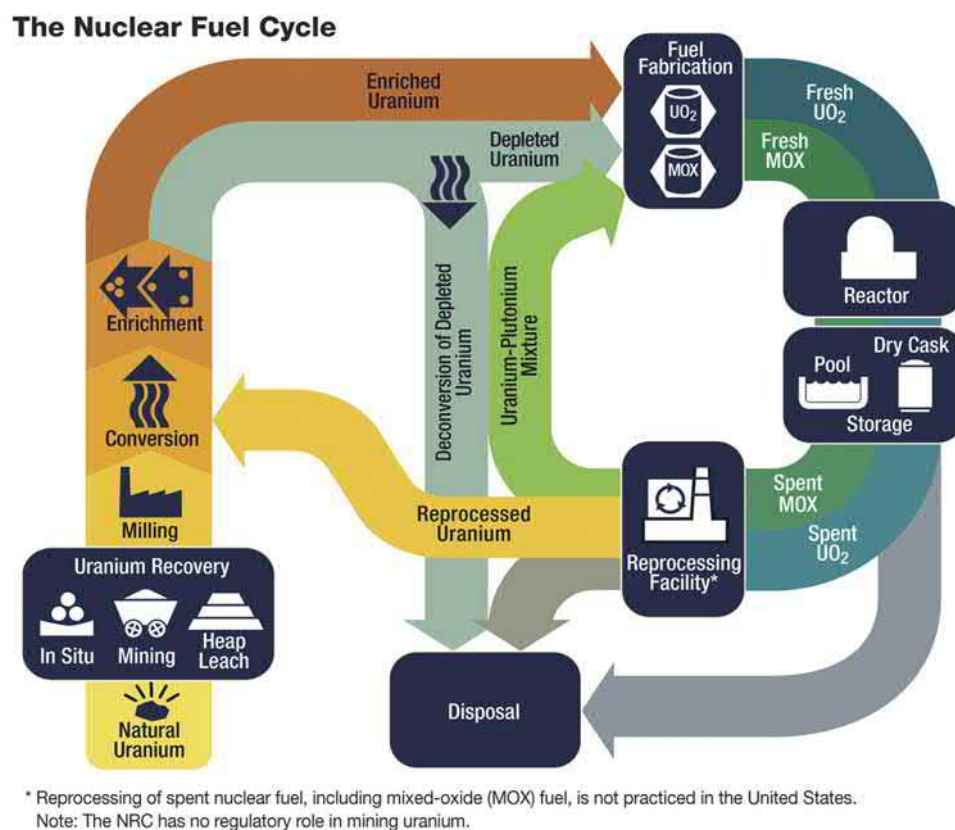


Fig. 1 Nuclear fuel cycle (USNRC graphics).

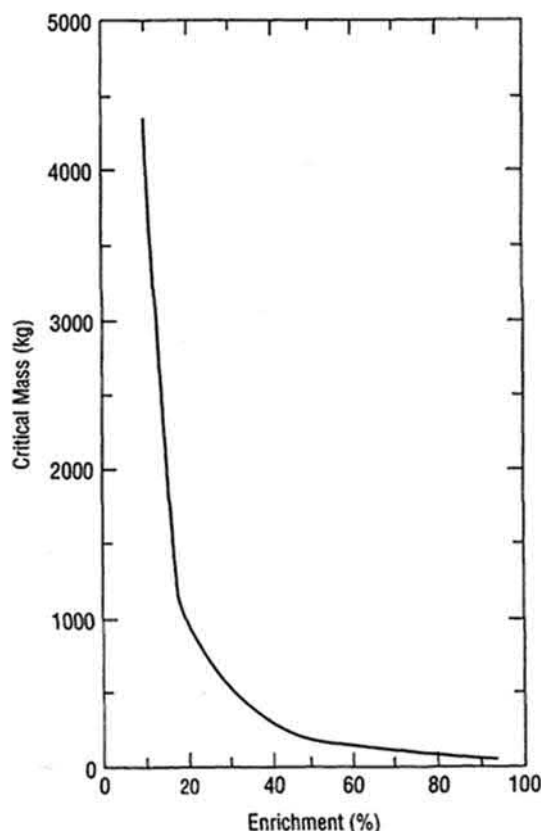


Fig. 2 Enrichment dependence of the critical mass of bare metallic uranium sphere.

to reach higher enrichment in an enrichment facility. The required SWU for a given throughput and final enrichment target is dependent on the enrichment of the feed and the tailings (depleted uranium by-product of enrichment). For a given SWU the energy requirement and the size of an enrichment facility are dependent on the sophistication of the enrichment technology. The time and energy required to enrich uranium is reduced generally by starting with a higher enrichment feed. In one estimate (NRC, 2009), the time for obtaining 1 kg of 90% ^{235}U would be reduced by a factor of about 4 if the facility had used 5% enriched uranium instead of the usual natural uranium as the feed. Also, the SWU requirements for producing HEU drop even further if the process leaves higher ^{235}U concentration than the usual 0.3% in the tailings.

A monograph (Krass et al., 1983) reviewed the impact of enrichment technology on proliferation. The clandestine network that transferred gas centrifuge technology from the Netherlands to Pakistan and on to Libya, Iran, and North Korea has brought enrichment to the forefront of proliferation. Potential for clandestine enrichment facilities and the breakout scenario, where a proliferating state acquires a nuclear weapon disregarding international safeguards and sanctions, present challenges to deter proliferation. Suggestions to enhance the safeguards of enrichment facilities include to only store enriched uranium offsite and in a chemical form not suitable for direct re-enrichment, such as uranium oxide (Wood et al., 2008). It is interesting to note that some of the requirements and actions mandated by the Joint Comprehensive Plan of Action (JCPOA), aka the Iran nuclear deal (IAEA, 2015) set limits on the enrichment capacity and cap the level of enrichment to 3.67% ^{235}U . The JCPOA is based on a framework agreement consisting of a short-term freeze of portions of Iran's nuclear program and an increased intensity of international inspections in exchange for decreased economic sanctions on Iran, as the countries work towards a long-term agreement. By curtailing Iran's nuclear capabilities and obstructing its path to a nuclear weapon in the near term, the JCPOA is designed to prevent Iran move away from the rule-based non-proliferation regime created by the NPT and including especially the IAEA safeguards systems. (Current status: In May 2018 the United States announced that it will no longer be a party to the JCPOA (US, 2018). We also note that the new administration which will be in place on January 20, 2021 has indicated the intention of returning to consideration of the JCPOA (NYT, 2020).)

Fuel fabrication plants for light water reactors (LWR) typically receive uranium from enrichment plants in the form of uranium hexafluoride (UF_6) and then a chemical process converts the UF_6 to uranium dioxide (UO_2) powder. A sintering process fuses the UO_2 powder to form fuel pellets. For mixed oxide fuel (MOX) the powder used to form fuel pellets consists of both UO_2 and plutonium dioxide (PuO_2). LEU and Pu are targets of proliferation at the fuel fabrication stage of the uranium fuel cycle. It is noted that the uranium enrichment level does not change at fuel fabrication plants.

Plutonium is one of the few elements that occur in nature only in trace quantities. In uranium fueled reactors Pu is a product of neutron capture and beta decay. A neutron absorption by ^{238}U forms ^{239}U which beta decays first into ^{239}Np then ^{239}Pu . The Pu

isotopic composition varies according to the level of fuel utilization or burnup. The buildup of ^{238}Pu and ^{240}Pu in reactor fuel over time (Shusterman, 2021) presents challenge to using power reactors to produce weapons-grade plutonium. Weapons-grade plutonium (USDOE, 1997) is typically about 93% ^{239}Pu and is obtained from low burnup uranium fuel. In the US plutonium containing between 80 and 93% ^{239}Pu is referred to as “fuel-grade” plutonium, while plutonium with less than 80% ^{239}Pu , typical of Pu isotopic in LWR and CANDU reactors at normal burnup, is referred to as “reactor-grade” plutonium. Decay heat in ^{238}Pu makes handling of the material in a weapon problematic. The high rate of spontaneous fission in ^{240}Pu can cause a nuclear weapon to pre-initiate, thereby jeopardizing its yield. Employing heavy water as the neutron moderator, such as CANDU reactor, affords much greater flexibility in the types of fuel that can be used, including natural uranium. CANDU reactors with natural uranium fuel and their operation as on-load or online refueling (Nichita, 2021) present special safeguards attention. IAEA has developed robust approaches for CANDU reactors to track core and spent fuel items. For example, special unattended bundle counters are used to monitor movement of irradiated fuel bundles from the reactor core to the spent fuel pond. It is noted that one of the stipulations in the JCPOA requires Iran to convert its heavy water research reactors from natural uranium to LEU to reduce weapons-grade plutonium output.

Extending the period between refueling enhances the proliferation resistance of the fuel cycle for several reasons. A longer cycle means higher burnup and less frequent fuel maneuver into and out of the reactor. A higher burnup increases the fraction of the even isotopes in the plutonium thereby reduces the attractiveness of the fuel for weapons utilization. Less fuel maneuver reduces the opportunity to introduce fertile material into the reactor for undeclared production of nuclear material. Unplanned shutdown in the pretext of retrieving leaky fuel is a diversion scenario to obtain low burnup fuel. Pin diversion is possible under a scenario of reconstituting leaky fuel rods instead of replacing the whole fuel assembly (for more on fuel design see Young (2021)). Reactor with a lifetime core is a concept that precludes the need of onsite refueling (IAEA, 2007). These reactors are being designed to rely on outsourced front-end and back-end fuel cycle services thus alleviates the operator to have enrichment or reprocessing capabilities. An example of a State forgoing the right to develop enrichment and reprocessing capabilities is the commitment made by the United Arab Emirates in the US-UAE Agreement for Cooperation (Blanchard and Kerr, 2010).

At the back-end of the nuclear fuel cycle the proliferation concern centers around access to plutonium either from the spent/used nuclear fuel or the process streams from fuel reprocessing facilities. The assessment in DOE/NN-0007 (USDOE, 1997) concludes even reactor-grade plutonium can be used to make nuclear weapons. Further, the assessment indicates the only plutonium isotopic that cannot realistically be used for nuclear weapons is nearly pure ^{238}Pu which generates so much heat that would render the weapon unstable.

The amount of ^{238}Pu in the plutonium isotopic vector of spent fuel is typically a few percent and depending on burnup can reach up to 8%. Denaturing the plutonium isotopic in nuclear fuel by increasing the ^{238}Pu content can reduce the attractiveness of the plutonium for use in nuclear weapons (Kessler, 2008; Sagara et al., 2005; Lloyd and Goddard, 2018). Using a set of figures of merit (FOM) for attractiveness, Bathke et al. (2012) estimated that only 8% ^{238}Pu is required to render the plutonium isotopic unattractive for an unadvanced proliferant state that requires reliably high-yield nuclear devices. An evaluation by Kessler (2008) confirmed that plutonium with up to 9% ^{238}Pu is weapons usable only if high technology is used. Non-nuclear weapon states (NNWS) are not expected to have access to high technology and if any nuclear yield is acceptable then the unattractiveness threshold for ^{238}Pu should be lowered accordingly (Kessler, 2008; Bathke et al., 2012).

The principal motivation to reprocess spent/used fuel from power reactors is to recycle fissile materials. The current standard method of reprocessing nuclear fuel is the PUREX (plutonium uranium redox extraction) process (IAEA, 2010; Long, 1978) (for more on this topic see Senentz (2021)), an aqueous method to recover uranium and plutonium from spent/used nuclear fuel for use in nuclear power reactors or nuclear weapons. The process extracts uranium and plutonium as separate product streams. Other reprocessing technologies have been under development to avoid separated plutonium. They include COEX, UREX and pyroprocessing, etc. COEX (Coextraction) (IAEA, 2010) a proprietary Areva process in which plutonium and uranium are extracted together, thereby making it more difficult to isolate plutonium than if PUREX were used. The two major products from the COEX process are pure uranium and a mixture of plutonium and uranium. The UREX+ suite of technologies (Laidler, 2008; Kumari et al., 2020) is developed by the US DOE for the reprocessing of LWR spent fuel and for meeting waste management goals by recycling actinides (a group of 15 metallic elements with atomic numbers from 89 (actinium) to 103 (lawrencium)) recovered from spent fuel in fast reactors. A common feature of the UREX+ aqueous processes is that there is no separation of pure plutonium in any stage of the reprocessing. The co-extraction of uranium with technetium (Tc) is the initial process stage for UREX+. Alternatively, U may be co-extracted with Pu and Np, as well as Tc. Further processing yields separated streams of U, Tc, Pu/Np, the transuranic elements (TRU) (mainly Np, Pu, Am and Cm), or a mixed U/Pu/Np stream. Pyroprocessing is also referred to as electrochemical processing (Laidler, 2008) (for more on this topic see Bourg (2021)). It is based on electrochemical separation performed in molten salts at elevated temperature. Separation of individual actinide elements depends on the relative free energy of formation of the actinide-chloride compounds. As a result, it is usually simple to collect a uranium product with low TRU contamination, while it is more difficult to separate any of the transuranic from one another. Consequently, pyroprocessing is designed to recover the TRUs together with uranium, the amount of uranium being set by processing operations. The stored product will be a U/TRU metallic ingot.

In 1998 the IAEA Director General provided a report to the IAEA Board of Governors (IAEA, 1998) with information that, if available in enough quantities in separated form, Np, and with considerably greater difficulty, Am could be used to manufacture nuclear explosive devices. The proliferation risk could certainly grow with fuel cycle choices that incorporate reprocessing technologies, such as the UREX+ suite, producing separated Am as one of the product streams. Short of putting Np and Am under

safeguards, the IAEA eventually developed and implemented a flow sheet verification technique at reprocessing plants subject to IAEA safeguards (Albright and Kramer, 2005; Rosenthal et al., 2019). Absent such monitoring, it is possible that States would be able to remove significant quantities of Am or Np from the plant clandestinely in order to manufacture nuclear weapons, either taking product streams of these materials or separating it from mixed product streams. Proliferation resistance studies have been performed on a comparative basis to evaluate reprocessing alternatives to PUREX (Bari et al., 2009). Using the figures of merit approach, Bathke et al. evaluated the material attractiveness of product streams from several reprocessing schemes for the LWR uranium fuel cycle, including UREX, COEX and PYROX (a pyroprocessing scheme) (Bathke et al., 2012). Their general conclusion is that co-extracting Pu with minor actinides (these are actinide elements in used nuclear fuel (spent fuel) other than uranium and plutonium, which are termed major actinides) does not reduce its attractiveness for use in nuclear weapons. However, extracting Am with Cm produces a product that is unattractive for use in a nuclear weapon. In the COEX process, where Pu is co-extracted with U, the U content determines the attractiveness level with higher proportions of U trending towards lower level of attractiveness.

Thorium fuel cycle

The first core of Indian Point Unit 1, a Babcock and Wilcox (B&W) Pressurized Water Reactor (PWR) in Buchanan, New York, used thorium-based fuel with stainless steel cladding. A thorium fuel cycle starts with the conversion of ^{232}Th to ^{233}U by neutron capture and beta decay (Shusterman, 2021). The only naturally occurring isotope of thorium is ^{232}Th . The absence of fissile isotope in thorium requires the thorium fuel cycle to rely on fissile fuel, like ^{235}U or ^{239}Pu , to establish and sustain a fission chain reaction that can provide excess neutrons to start the ^{233}U production. ^{232}Th , the fertile component in the thorium fuel cycle, is thus analogous to ^{238}U in the U–Pu fuel cycle. There is no plutonium in a thorium fuel cycle. However, ^{233}U is under IAEA safeguards because in many respects it is well suited for nuclear weapons use, such as a low critical mass, low spontaneous fission neutron source and low heat output. The proliferation concern for a thorium fuel cycle are the diversion of ^{233}U and the misuse of reactor to produce ^{233}U clandestinely (Kang and von Hippel, 2001; Uribe, 2018). The ^{232}U co-produced with ^{233}U in a thorium fuel has ^{208}Tl daughter products that emit highly energetic (2.6 MeV) gamma rays with high absolute emission probability. Gamma radiation from ^{208}Tl presents challenges not only to operation and monitoring but also to the potential proliferator. Some people argue the radiation exposure may not be a sufficient deterrent to a technically advanced proliferator. In some proliferation scenarios, ^{233}Pa a precursor to ^{233}U , may be removed from a reactor, such as from a molten salt reactor (MSR), allowing formation of ^{233}U without the undesirable ^{232}U (Uribe, 2018). Considering the spectrum of MSR designs and material flowsheets (for more on MSR see Ignatiev (2021)) several safeguards challenges were noted (Kovacic et al., 2018) including nuclear material measurement methods and technologies need to be developed (destructive/non-destructive analysis, etc.) given that the nuclear signatures are likely to be notably different from conventional LWR fuel cycles.

Research reactors

Research reactors primarily serve as a neutron source for research and isotope production. Early designs often use HEU for fuel to increase the power density in a more compact configuration and reduce the frequency of refueling. There has been a movement to convert existing research reactors from HEU to LEU (Glaser, 2002). A mission to develop new fuel for research reactors to increase their proliferation resistance was initiated by the IAEA International Nuclear Fuel Cycle Evaluation (INFCE) conference (IAEA, 1980) in the late 1970s. In 1978 the US Department of Energy initiated the Reduced Enrichment for Research and Test Reactors (RERTR) Program (<https://www.rertr.anl.gov/>) to develop the technical capabilities and resources to convert research reactors and isotope production processes from using HEU to LEU through the development of new LEU fuels and targets. For more on research reactors see O’Kelly (2021).

A global fuel cycle

A variety of programs have looked at developing an international fuel cycle or a reliable supply of fresh fuel and spent fuel take-back arrangements to encourage aspiring nuclear energy states to forgo the development of an indigenous sensitive nuclear infrastructure, thereby minimizing the proliferation potential for these aspiring states. Mohamed ElBaradei, former director general of the IAEA, had called for the establishment of a multinational nuclear fuel bank, envisioning that such a fuel bank would remove the incentive for countries to develop their own enrichment and reprocessing facilities, and consequently would minimize the proliferation risk posed by countries seeking nuclear energy (IAEA, 2005). In such a system, however, incentives for participation must be coherent and compelling and must alleviate the anxieties of states vying for energy independence by providing an economic and assured fuel supply. Programs must address the issues of where multinational facilities will be located and who will provide the fuel, without making states feel that they are being held hostage or abdicating their right to peaceful nuclear technologies as granted by Article IV of the NPT. Many issues related to the multinational approaches to the nuclear fuel cycle were examined in a series of articles published by the American Academy of Arts and Sciences (AAAS, 2009, 2010a,b). A joint committee of the US National Academy of Sciences (NAS) and the Russian Academy of Sciences (RAS) produced a report (NRC, 2009) analyzing the proposals and options

for future international nuclear fuel cycles, including the incentives that might be required for countries to accept fuel assurance guarantees and not develop enrichment or reprocessing facilities, as well as technical fuel-cycle issues. One of the joint committee's recommendations is that developers of nuclear fuel cycle technologies should assess the technologies' proliferation risks and projected economic costs and benefits as critical elements of design. For more information on global fuel cycle see also [McKinzie \(2021\)](#) and [Hore-Lacy \(2021\)](#).

Evaluating proliferation risk

In order to reduce proliferation risk of deploying nuclear energy more globally, risk assessments of the threats and consequences of a potential proliferation event are necessary. Methods addressing the technical aspects of proliferation resistance have been developed, most notably the Proliferation Resistance and Physical Protection methodology ([GIF, 2011b](#)) of the GEN IV International Forum (GIF). Such methodologies, however, have concentrated on the technological factors of proliferation resistance, but have not addressed the equally important political factors. Many of the nonproliferation efforts in the geo-political arena are discussed in a Special Section on Nonproliferation in The November 2010 issue of the Nuclear News, a publication of the American Nuclear Society. It is recognized that there are large uncertainties in evaluating the impact of national nuclear policy on non-proliferation. After conducting a review of worldwide research activities on nuclear non-proliferation a proliferation evaluation framework was proposed ([Wu et al., 2020](#)). The new assessment framework utilizes the analytic hierarchy process (AHP) to quantify the weight of national policy, among other extrinsic and intrinsic barriers, in influencing the outcome of nuclear non-proliferation. AHP had been applied previously in a comparative study of nuclear fuel cycles and the three cycle options (once-through, thermal recycling using MOX fuel and fast reactor recycling employing pyro-processing) were ranked against five attributes: sustainability, environmental friendliness, economics, proliferation resistance and technical feasibility ([Yoon et al., 2016](#)).

Several steps are involved in preparing a good evaluation of the proliferation risk or resistance of a nuclear energy system or facility. First, one must determine how to characterize and measure proliferation resistance or risk, and, second, one must evaluate the risk or resistance. Most research on proliferation resistance and risk has focused on these steps. Nuclear proliferators may be states, state-sponsored terrorists, or in dependent non-state actors. Each of these proliferators possesses differing motives, tactics, and means that dictate how the system should be optimally safeguarded and protected. In addition, there are intrinsic features of the system that could delay or present technical difficulties to the proliferator.

The Generation IV Roadmap ([GIF, 2002](#)) defined the following Proliferation Resistance and Physical Protection (PR&PP) goal for future nuclear energy systems:

Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

The Proliferation Resistance and Physical Protection Working Group (PRPPWG) was created to develop, implement and foster use of an evaluation methodology ([GIF, 2011b](#)) to assess Generation IV nuclear energy systems ([Stanculescu, 2021](#)) with respect to the GIF PR&PP goal. The following is a brief discussion of the PR&PP methodology for evaluating proliferation resistance.

The overall PR&PP framework involves three steps: challenges, system responses, and outcomes, shown in [Fig. 3](#). For a proliferation risk scenario occurring at a nuclear facility, "challenges" are threats to the nuclear facility, such as diversion, misuse, breakout, or the establishment of a clandestine facility. System responses to the challenge are then evaluated, for example, whether there are physical and technical design features that would combat or slow this particular attempt or safeguards in place that would alert the international community. Finally, the possible outcomes resulting from the challenge and the system response are evaluated. These steps are repeated for many potential challenges and system response variations.

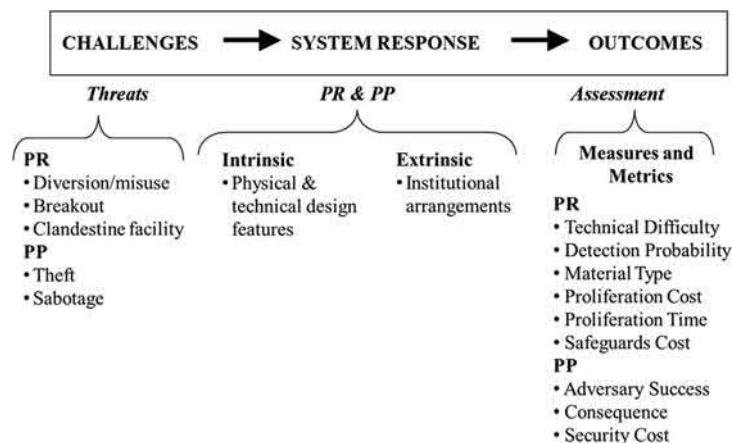


Fig. 3 The PR&PP methodology framework.

The PR&PP analysis of the system response occurs in three stages. First, the nuclear system is decomposed into system elements to permit a pathway analysis. This involves identifying elements such as the materials, facilities, processes, and transportation links that an adversary could exploit to accomplish his or her goals. Second, the location of operations and materials, their accessibility, and characteristics are identified. In addition, any extrinsic measures and the locations where they are applied are noted, such as material balance areas and locations of safeguards and physical protection systems. Finally, interfaces with other systems that are not part of the analysis (i.e., links to clandestine facilities) are evaluated to identify any additional potential vulnerabilities.

In assessing the degree of proliferation resistance of a nuclear system both intrinsic and extrinsic features/barriers are considered. Intrinsic features relate to physical design features of a nuclear energy system; hence, intrinsic barrier: physical design features that impede proliferation, sabotage or theft. Extrinsic or institutional features relate to States' commitments, obligations, and policies regarding nuclear non-proliferation. Hence, extrinsic barriers include treaties, agreements, the application of IAEA safeguards, and security forces and equipment to impede proliferation, sabotage and theft. Examples of intrinsic and extrinsic features are presented in [IAEA \(2002\)](#), [USDOE \(2000\)](#), [USDOE \(2001\)](#) and [IAEA \(2008\)](#).

Intrinsic features consist of technical features that:

- (a) Reduce material attractiveness for use in nuclear weapons programs;
- (b) Prevent or inhibit diversion of nuclear material;
- (c) Prevent or inhibit undeclared production of direct-use material, including reactors designed to prevent undeclared target materials from being irradiated in or near the core; reactor cores with small reactivity margins to prevent operation of the reactor with undeclared targets; and fuel cycle facilities and processes that are difficult to modify
- (d) Facilitate nuclear material accounting and verification, including continuity of knowledge.

Extrinsic features consist of:

- (a) Commitments, obligations and policies of States, such as the Treaty on the Non-Proliferation of Nuclear Weapons and the IAEA safeguards agreements and protocols additional to such agreements;
- (b) Agreements between exporting and importing states on exclusive use of nuclear energy systems for agreed purposes;
- (c) Commercial, legal or institutional arrangements that control access to nuclear material and technology;
- (d) Verification measures by the IAEA or by regional, bilateral and national measures; and
- (e) Legal and institutional measures to address violations of measures defined above.

For comparing the relative risk of different proliferation scenarios, separate measures for proliferation resistance (PR), which is relevant to the Host State threat, and physical protection (PP) robustness, which is relevant to non-state threats, have been developed under the PR&PP approach. For proliferation resistance (PR) the measures are:

- *Proliferation Technical Difficulty (TD)*—The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the barriers to proliferation.
- *Proliferation Cost (PC)*—The economic and staffing investment required to overcome the technical barriers to proliferation including the use of existing or new facilities.
- *Proliferation Time (PT)*—The minimum time required to overcome the barriers to proliferation (i.e., the total time planned by the Host State for the project).
- *Fissile Material Type (MT)*—A categorization of material based on the degree to which its characteristics affect its utility for use in nuclear explosives (e.g., bare sphere critical mass, neutron emissions, decay heat).
- *Detection Probability (DP)*—The cumulative probability of detecting a proliferation segment or pathway.
- *Detection Resource Efficiency (DE)*—The efficiency in the use of staffing, equipment, and funding to apply international safeguards to the nuclear energy system.

For physical protection (PP), the measures are:

- *Probability of Adversary Success (Ps)*—The probability that an adversary will successfully complete the actions described by a pathway and generate a consequence.
- *Consequences (C)*—The effects resulting from the successful completion of the adversary's action described by a pathway.
- *Physical Protection Resources (PPR)*—the staffing, capabilities, and costs required to provide PP, such as background screening, detection, interruption, and neutralization, and the sensitivity of these resources to changes in the threat sophistication and capability.

Example applications of the PR&PP methodology include a case study of an example sodium fast reactor ([GIF, 2009](#)) and a compendium report that examined the PR&PP characteristics of each of the six GIF nuclear energy systems ([GIF, 2011a](#)). The report ([GIF, 2011a](#)) captures the current salient features of the GIF system design concepts that impact their PR&PP performance. Appendix A to the report provides an extended table which lists, in summary form, each design concept and its robustness characteristics in response to potential PR&PP threats to the design.

An early application of the GIF PR&PP methodology was its adaptation in the assessment framework ([Bari, 2008](#)) for the draft nonproliferation impact assessment (NPIA) ([NNSA, 2008](#)) of the Global Nuclear Energy Partnership (GNEP). GNEP was part of President Bush's Advanced Energy Initiative (announced in 2006 and domestic components terminated in 2009), aiming to expand nuclear energy while reducing nuclear waste impacts and risks of nuclear proliferation. The draft NPIA evaluated GNEP programmatic alternatives by using technical factors and policy factors. The assessment framework ([Bari, 2008](#)) developed the linkage between the technical factors and the PR&PP methodology, aligning the technical factors with associated PR&PP measures and their

underlying metrics. **Table 2** is a crosswalk between the three technical factors and the associated PR&PP measures. The metric for each PR&PP measure provided the foundation of a grading structure for the technical factors, giving qualitative notion of time and scale of possible consequence.

The draft NPIA (NNSA, 2008) conducted comparative assessments of the nonproliferation features of the GNEP alternatives. The study considered three fuel cycle alternatives, once-through, full actinide recycle and partial actinide recycle, and included both fast and thermal reactors. For thermal reactors, in addition to the LWR, the other alternatives were heavy water moderated reactor (HWR) and high temperature gas-cooled reactor (HTGR). The draft NPIA compared the nonproliferation impacts of alternatives by assessing several PR&PP attributes of each alternative, including demand for enrichment services (e.g. HTGR was higher than HWR), risk of misuse (e.g. HWR was higher than LWR), risk of separating weapons-usable material (e.g. recycle was higher than once-through), material attractiveness (e.g. recycled fuel was higher than once-through fuel) and safeguards cost (e.g. HWR was higher than LWR).

Concurrently, a multi-laboratory team (Zentner et al., 2010), using the GIF PR&PP methodology, evaluated the proliferation resistance technical risk characteristics of a number of generic nuclear reactors types in a variety of fuel cycle implementations. They included a sodium fast reactor, a high temperature gas reactor, a heavy water reactor, and a light water reactor. Three general types of material acquisition scenarios were evaluated for each reactor type: (1) concealed diversion of material; (2) concealed misuse of the reactor to produce materials; and (3) breakout. The evaluations considered both the intrinsic and extrinsic PR&PP characteristics of each reactor. This study showed that each reactor type has particular features affecting its respective PR&PP characteristics and that no type was clearly superior or inferior for the material acquisition scenarios considered. Areas were identified where safeguards approaches and technology could be improved to enhance the safeguardability of the designs. For example, minimizing the number of entry and exit points for fuel transfer between system elements will enhance material containment, protection, and accountability (MCP&A), thus partially compensating for any lack of knowledge continuity by visual inspection during a fuel transfer. Another design feature with potential to reduce the risk of misuse is the modern approach of using burnable poisons in LWR fuel assemblies. Its implementation results in fewer guide tubes in PWR fuel assemblies, complicating misuse by insertion of target rods in the guide tubes. Many advanced reactor designs under research and development incorporate intrinsic proliferation resistance features, in particular in the fuel cycle architecture. Some micro (typically up to 10 MWe) reactor designs that feature lifetime cores (IAEA, 2020) can operate up to 20–30 years without refueling. These so-called ‘nuclear batteries’ replace their cores as a whole in a dedicated fuel facility, simplifying the fuel cycle infrastructure and its related safeguards requirements. Some fast reactor designs practice breed-and-burn that sustains fission power generation with minimal need for enrichment. The Traveling Wave Reactor-Prototype (TWR-P) (Hejzlar, 2021), for example, only needs enriched uranium for its first fuel loading (to establish criticality), and thereafter it is fed only with natural or depleted uranium and operated on the once-through fuel cycle for very long core life (~60 years) with no need for reprocessing and enrichment. There is also a scenario of deploying floating nuclear power plants (Golay and Kodak, 2021) whereby the supplier provides all fuel services from cradle-to-grave such that the host nations do not need to have their own infrastructure for enrichment, fuel fabrication and spent fuel storage and disposal, thus minimizes the risk of proliferation.

Prospects of mitigating proliferation risk

For nuclear energy, proliferation resistance does not mean proliferation proof; there is no panacea for the ever-dynamic evolution of the nature of proliferation threats. Ultimately, no fuel cycle, technology, or international policy can entirely eliminate the possibility of nefarious proliferation of nuclear expertise, technology or materials. The risk of nuclear proliferation, however, can be mitigated with a balanced approach that includes intrinsic proliferation resistance inherent to technology design and material management, in conjunction with extrinsic efforts through safeguards, international collaboration, and verifiable treaties. The incorporation of each of these approaches, both technical and institutional, should work in tandem to produce a risk-informed proliferation-resistance strategy.

Proliferation resistance can raise hurdles, slow progress, and provide alarm signals of proliferation activity. Risk assessments of the threats and consequences of a potential proliferation event are thus necessary to reduce proliferation risk. There are many

Table 2 Correspondence between NPIA technical factors and PR&PP measures.

<i>Technical factors</i>	<i>Associated PR&PP measures</i>
Avoiding proliferator success (T1)	• Technical Difficulty (PR) • Proliferation Cost (PR) • Proliferation Time (PR) • Detection Probability (PR) • Probability of Adversary Success (PP) • Consequences (PP)
Facilitating cost effective international monitoring (T2)	• Safeguards Cost (PR) • Detection Probability (PR) • Security Cost (PP)
Resulting in less attractive material types and forms (T3)	• Material Type (PR) • Consequences (PP)

potential pathways to proliferation, and adversary success is highly dependent on pathway choice and creativity. In addition, the results and applicability of an analysis are dependent on a number of assumptions about adversary capabilities and objectives. These uncertainties make it difficult to effectively collapse the proliferation resistance of a facility into a single value denoting overall proliferation resistance. However, one key conclusion has emerged from the work performed to date for PR&PP: Safeguardability is a very important consideration. The PR&PP methodology provides designers and policy makers a generic and formal comprehensive approach to evaluate, through measures and metrics, the Proliferation Resistance (PR) and Physical Protection (PP) characteristics of advanced nuclear systems. As such, the application of the evaluation methodology offers opportunities to improve the PR and PP robustness of system concepts throughout their development cycle.

Safeguards-by-design (SBD) is generally recognized as an approach wherein international safeguards are fully integrated into the design process of a nuclear facility—from initial planning through design, construction, operation, and decommissioning. In a workshop (IAEA, 2009), safeguards and facility design experts convened to discuss basic principles to design and implement an SBD process and to identify fundamental design features deemed critical to the effective and efficient implementation of IAEA safeguards. The PR&PP evaluation could be an integral part of the SBD process, to elucidate the interactions between the intrinsic and the extrinsic features, study their interplay, and then guide the path towards an optimized design. The SBD process was applied to an advanced CANDU reactor design (ACR-1000) (Whitlock, 2010) using the PR&PP methodology. The pathway comparison consisted of a simplified qualitative ranking based on expert elicitation. Results of this exercise were used to stimulate discussions involving the design team, the state regulator, and the IAEA, and propose steps that accommodate safeguards without negatively impacting other design requirements. The GIF PR&PP Working Group is implementing an SBD process to evaluate the GEN IV system designs (Cojazzi et al., 2018; Cheng et al., 2019).

There are now many new and diverse developments and applications of peaceful nuclear energy in many countries. It is of paramount importance that the development of new designs of reactors and fuel cycles incorporate enhanced intrinsic proliferation resistance especially during the early stages of design. It is the aspiration of Generation IV nuclear energy systems to increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism. Adherence to stringent and powerful international safeguards regime throughout the entire lifecycle of nuclear energy systems also provides confidence that proliferation risk can be mitigated and reduced to a level acceptable by society.

See Also: International Nuclear Waste Disposal Centers; Research Reactor Conversion to Low Enriched Uranium (LEU) Fuel; Social Acceptability of Geologic Disposal; The Case for International Fuel Cycle Centers to Reduce Proliferation Risk.

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The Case for International Fuel Cycle Centers to Reduce Proliferation Risk

Andrew C. Kadak, Kadak Associates, Inc., Port St. Lucie, FL, United States; and Massachusetts Institute of Technology, Cambridge, MA, United States

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The fuel cycle

The nuclear fuel cycle considered in this chapter assumes nuclear power reactors cooled and moderated by regular water (Light Water Reactors or LWR). The international fuel cycle center concept applies to all types of nuclear reactors, including advanced future generation reactors ([Krepel, 2021](#)), ([Stanculescu, 2021](#)).

The fuel cycle starts with natural uranium that is mined which needs to be processed in the milling step ([Crawford, 2021](#)) and converted to uranium hexafluoride which then needs to be enriched in Uranium 235 from 0.7% to between 3% and 5% ([World Nuclear Association, 2020b](#)). Once enriched, the U235/U238 powder is converted into pellets and they are inserted into fuel pins and arranged into fuel assemblies to be loaded into the nuclear reactor for 3–4 years of operation. These fuel assemblies constitute the “core” of the reactor. Once removed from the reactor, the used fuel can either be directly disposed after about 5 years of on-site storage in spent fuel pools of water, stored on an interim basis in dry cask storage casks or reprocessed and then recycled. The residual waste, either as used fuel or reprocessed high level waste (HLW), needs to be disposed in a geological repository. [Fig. 1](#) highlights the key steps in the fuel cycle.

The problem

The problem with this fuel cycle is that at various points it is susceptible to diversion of special nuclear materials (primarily Uranium 235 and Plutonium 239) for the purpose of making nuclear weapons. The major points of vulnerability are during the enrichment and reprocessing step. While enrichment of uranium is required to produce nuclear energy, reprocessing is not. Reprocessing is an added step to recycle unused uranium 235 and plutonium 239 which is produced during operation ([World Nuclear Association, 2020a](#)). This recycling has two objectives: (1) to reduce demand for mining of uranium to extend the supply of uranium; and (2) to reduce the volume of high level nuclear waste that needs to be geologically disposed by 90% compared to direct used fuel disposal. It is interesting and important to note that there has not been a case where reactor grade plutonium (without complicated reprocessing technology) has been used to make a nuclear weapon. The feasibility of using “reactor grade” plutonium for weapons is in doubt but it is assumed to be a proliferation risk should use fuel be stolen.

Another nuclear technology that requires reprocessing are “breeder” or “burner” reactors. One type of this class of reactors is called “fast”¹ reactors which use liquid sodium as a coolant instead of water to allow for “breeding” Plutonium 239 from natural uranium (mostly U-238). These reactors will avoid the environmental impact of uranium mining and the enrichment since Pu-239 is fissionable and can power the nuclear reactor. Breeder reactors can make at least as much Pu-239 (by neutron capture in U-238) as they consume. There is enough U-238 in storage left over from uranium enrichment plants to power such breeding nuclear energy plants for thousands of years making nuclear energy “sustainable.” This technology is unique in that it takes a non-fissionable “waste” from enrichment (U-238) and converts it in the reactor to make Plutonium 239 as a result of the neutronic reaction during operation ([Boullis, 2021](#)). Depending on the design of the core, more plutonium can be produced than is consumed during operation which is why it is called a “breeder.”

The so called “burner” reactors can use the same fast reactor sodium cooled core but are designed to only consume the Plutonium generated in contemporary light water reactors and not breed it. The first operating reactor in the United States to produce electricity was a sodium cooled fast reactor built in Idaho at the Idaho National Laboratory. Breeder reactors are operating in Russia and once operated in France and the United States. The United States breeder reactor program was politically shutdown due to concerns about proliferation.

¹The term fast is used since the reactor uses higher speed neutrons instead of slow thermal energy neutrons needed for Light Water Reactors.

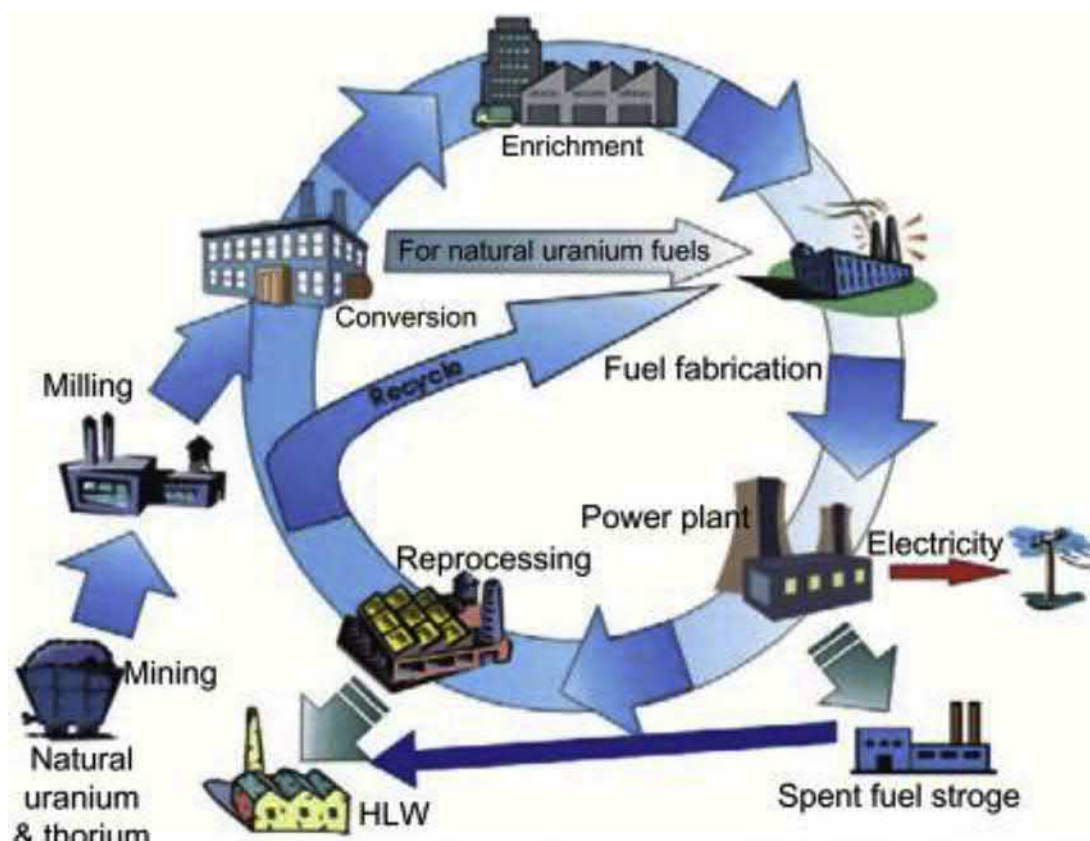


Fig. 1 Nuclear fuel cycle. <https://www.sciencedirect.com/topics/materials-science/nuclear-fuel-cycle>

Most nations with developed nuclear energy programs have the capability of natural uranium enrichment which is a requirement for self-sufficiency in nuclear energy production. Most nuclear nations also have reprocessing² capabilities including China, Russia, France, United Kingdom and the United States (for military uses) (World Nuclear Association, 2020b). The US has decided not to reprocess commercial used fuel to set an example to other nations to forego this technology with little success.

With many new emerging nations seeking to use nuclear energy for power production, concerns exist for proliferation of the special nuclear materials. For this reason, international fuel cycle centers have been proposed to avoid the need for each nation that seeks to use nuclear energy to develop their own proliferation risk technologies. Those nations that have reprocessing capability have high security proliferation resistant systems in place with checks by the International Atomic Energy Agency to monitor the location of these special nuclear materials.

The solution

The concept of an international fuel cycle center would be to have a nation, likely a nation that already has the infrastructure and facilities to perform not only enrichment, and fuel fabrication, but also reprocessing and recycling at the same secure facility, host such a facility under IAEA supervision (Richter, 2008). While combining all these functions at one facility is not necessary, it would limit the security requirements associated with separate facilities. The challenge is to get agreement from the other nations to assure them that they can have a reliable supply of fuel for operations. The hope is to avoid the national desire to “become a nuclear weapons nation” for the status that it provides. We have such examples in North Korea, Indian, Pakistan, and Iran.

The backbone of the effort to limit the spread of nuclear weapons and technology is the Non-Proliferation Treaty of International Atomic Energy Agency (1970). The goal was to limit the spread of nuclear weapons while affording nations the opportunity for development of peaceful applications of nuclear science and technology including energy production. The International Atomic Energy Agency is charged with overseeing compliance by hosting inspections of nations with nuclear programs. At this point 191 nations have signed the treaty.

²Reprocessing is currently used to recycle unused uranium from used fuel and recycled in LWRs. Plutonium is also recycled in LWRs by mixing with uranium creating Mixed Oxide (MOX) fuel which is being done in France. This can save about 25–30% of uranium resources by extracting this unused energy in uranium.

While this effort has been largely successful, India, Pakistan and Israel have not signed the Treaty and have nuclear weapons. North Korea signed and then resigned from the Treaty which has led to their development of an atomic bomb. Although Israel has not declared itself as a nuclear weapons state, it is widely believed it have them. As a counter, Iran is suspected of developing nuclear weapons despite assurances not to do so. Iran has its own independent enrichment plant capable of enriching uranium to levels needed for nuclear weapons with only one operating nuclear power plant.

In an effort to deal with these evolving threats and nations that desire to have access to nuclear technology, the then Director General of the Atomic Energy Agency in 2003, Dr. Mohamed ElBaradei, published a paper entitled, "Towards a Safer World" (ElBaradei, 1970). In this article, he recommended, among many other things dealing with weapons proliferation, to limit reprocessing and enrichment to facilities under multinational control with assurances to user nations that there would be a reliable, non-political source of supply. The obvious difficulty of his proposal was dealing with national security and economic interests of the nations. His proposal was complicated by the globalization of countries' own existing fuel production facilities in terms of placing them under international control. The barriers to obtaining agreement on surrendering their rights to enrich and reprocess dwell on trust of the multilateral agency to be non-discriminatory and transparent in its dealings.

As a result of the ElBaradei initiative, many proposals were considered including the creation of IAEA's international emergency fuel supply to deal with possible disruptions for enrichment (Carrell-Billiard and Wing, 2010). In 2007, ElBaradei presented a report to the IAEA Board of Governors (all member states) entitled "Possible New Framework for the Utilization of Nuclear Energy" (ElBaradei, 2008). In this report he calls for the creation of an international fuel bank to avoid the need for country's to build enrichment plants funded by member states and individual contributions. As a result several nations, the United States, Germany, Austria, Russia and the World Nuclear Association presented proposals to forward the concept of international fuel cycle facilities (World Nuclear Association, 2011), (Simpson, 2009), (National Research Council, 2009).

The challenges

The challenge is that there are only four places to enrich natural uranium—the United States (URENCO-USA foreign owned and operated), Eurodif (which is internationally owned but operated by the French), URENCO (internationally owned and operated) and Russia (state owned and operated). China has been enrichment and reprocessing capabilities but so far has them for their domestic market. The concern is that since the number of options are few, assurance of supply and putting the nation's electricity supply at the political mercy of these countries is not necessarily viewed as a sound policy. What ElBaradei and others sought to achieve is to provide an IAEA independently managed stockpile of enriched uranium to those countries who forgo enrichment. With this stockpile, nations would then be able to purchase fuel fabrication services from at least 16 countries worldwide eliminating concerns about diversity of supply.

The most ambitious new proposal was the Global Nuclear Energy Partnership (GNEP) proposed by President G.W. Bush of the United States in Department of Energy (2007). GNEP called for a consortium of nations with advanced nuclear programs to enter into supply arrangements for nuclear fuel and to take back the used fuel from nations who agree to forgo enrichment and reprocessing. GNEP also called for development of reprocessing technologies that would be less proliferation prone. The big incentive for the GNEP program was the return of the used fuel which makes dealing with their nuclear waste less of a problem. Ultimately, the high level nuclear waste from reprocessing might be have to be returned to the host country for storage or disposal. There are companion proposals to host international disposal centers (Hore-Lacy, 2021) to avoid the need for each nation building its own geological repository for their nuclear power program.

Due to the difficulties in obtaining agreement with participant nations to forego enrichment and reprocessing as a pre-condition, the United States modified the proposal to sign separate memoranda of understandings for a guarantee of supply. Despite these heroic efforts in 2009, the United States changed administrations to democrats who bizarrely defunded the entire effort due to the concerns about reprocessing as "undermining the Nation's non-proliferation policy."

International Uranium Enrichment Centers

Because of the difficulties in obtaining a more complete agreement on the nuclear fuel cycle, ElBaradei focused on the more immediate and important question of enrichment which is the easiest path to nuclear weapons while also an important first step in using nuclear power. Establishing an enrichment program is costly and complicated which is why nations might support an international enrichment supply system. Reprocessing is even more complicated and costly and likely unneeded at their early stage of a nuclear energy program. The high cost of developing either or both of these capabilities is a big disincentive for new nuclear nations.

One successful program that did move forward was the Russian establishment of a low enrichment uranium reserve with agreement with the IAEA. Russia established an "International Uranium Enrichment Center" (IUEC) in Angarsk, Siberia in 2007 with Kazakhstan, Ukraine and Armenia as initial shareholders and members (IAEA, 2007). In 2010, this facility won acceptance by the IAEA as an emergency storehouse of low enriched uranium (4.95%) under IAEA control (International Uranium Enrichment Center, 2010). For nations seeking to withdraw LEU from the emergency supply it must meet "established non-proliferation criteria."

The IUEC has already placed 120 t of low enriched uranium in storage for use of all IAEA member countries as an emergency reserve due to political constraints from purchasing on the commercial market. This facility is under IAEA safeguards as a monitored facility. The other interesting feature of the IUEC is that the technology is "black boxed" which means the owners will not have

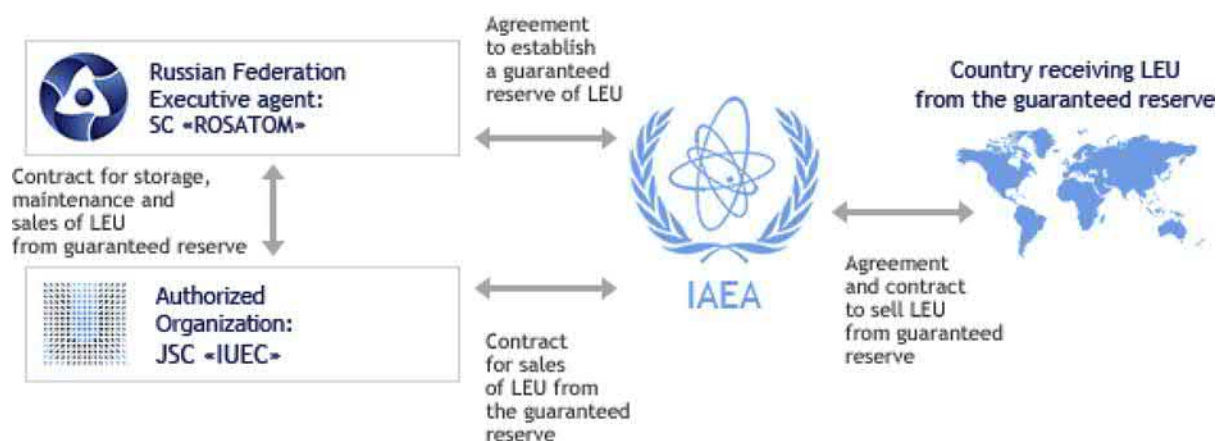


Fig. 2 International uranium enrichment center fuel bank. http://eng.iuec.ru/activities/fuel_bank/

Table 1 Fuel bank operating criteria.

Amount of reserve	More than 120 tU LEU as uranium hexafluoride (UF ₆)
Enrichment	From 2.00% up to 4.95%; not less than one third of the reserve (40 tU) with an enrichment of 4.95%
Location	JSC IUEC storage sites within the Angarsk Electrolysis Chemical Complex
Financing	The guaranteed reserve is created and maintained at the expense of the Russian Federation
Selling price	Based on spot market prices published by internationally acknowledged companies
JSC IUEC functions	Storage, maintenance and sales of LEU from the guaranteed reserve

access to the technology, only the service of enrichment and materials. This is the first such agreement on fuel cycle technology that could serve as model for others including reprocessing services. Fig. 2 shows how functionally this system works (International Uranium Enrichment Center, 2020). The Russian concept appears to meet ElBaradei's goals for at least one step to a "safer world."

The basic parameters under which this center operates are shown on Table 1.

Concluding remarks

No nuclear fuel cycle can be made completely risk free. Although reprocessing and enrichment technologies can be made more proliferation resistant and secure, reducing proliferation risk is a political not technical decision (Chang and Bari, 2021). It is clear however, that as more countries seek to obtain the benefits of nuclear energy increasing the number of enrichment and reprocessing facilities, the larger the risk even under the most secure conditions. An individual country can chose to "break out" and build a nuclear weapon with these technologies unless politically and economically incentivized not to do so. It is important to establish international fuel cycle centers to discourage individual countries from developing their own enrichment and reprocessing capabilities. However, guarantees and incentives are needed to enable such agreements among nations so that may be assured of nuclear fuel for their operating nuclear power stations.

The current commercial nuclear fuel cycle market can handle the current demands for uranium fuel and reprocessing needs. The IAEA fuel cycle bank addresses a major step forward avoiding the need for nations to build their own enrichment plants if there is trust in the system and the commercial market accommodates the need. The reprocessing facility in France already serves the international market with the production of Mixed Oxide Fuel from recycled Plutonium being used in Japan, France and other countries. The so called "closing" the fuel cycle with the use of breeder or burner reactors and a centralized nuclear waste repository awaits leadership similar to that shown by the Russian government in their creation of the International Uranium Enrichment Center.

See Also: Non-Proliferation Aspects of Nuclear Energy.

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Nuclear Waste Can Be Reduced by Recycling and Transmutation

Dominique Greneche, Nuclear Consulting Company, Marcoussis, France

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What is a “nuclear waste”?

Nuclear waste (NW) in its broadest sense are radioactive materials produced by nuclear industry (including nuclear research facilities) or nuclear defense activities for which no further use is foreseen in a given country in accordance with its national legislative and regulatory framework. NW may arise in a wide range of concentrations of radionuclides and in a variety of physical and chemical forms. These differences result in an equally wide variety of options for their management and processing the more so that practices and terminologies as well as regulations differ from different countries. Classifying NW is therefore needed in order to implement and optimize appropriate management systems adapted to their physical/chemical characteristics.

To this regard, the IAEA (International Atomic Energy Agency) has published a reference document ([IAEA SAFETY STANDARDS, 2009](#)) which sets out a general scheme for classifying radioactive waste (that include NW) which is based primarily on considerations of long-term safety, and thus, by implication, disposal of the waste. This “Safety Guide” identifies the conceptual boundaries between different classes of waste and provides guidance on their definition on the basis of long-term safety considerations. This classification is mainly based on two parameters that characterize a radioactive waste: its “activity content” (a generic term that covers activity concentration, specific activity and total activity) and half-life of the radionuclides contained in the waste (time after which its radioactivity has been halved).

As a practical matter, nearly all the nuclear waste of concern to the public is made of radioactive products contained in the spent fuel which is unloaded from nuclear reactors when it has been fully used in the reactor core to produce its energy (hence the term “used fuel” or “spent fuel”). These radioactive products consist of two main categories. The first one comprises all fission products (FPs) which are directly created from fissions of atomic nuclei that are broken by neutrons into two pieces in the reactor core ([Klay, 2021](#)). The second is made up of actinides (or transuranics), which originate from various nuclear reactions (mainly neutron captures) on heavy nuclei (mainly uranium) during irradiation in the reactor core. The primary actinide is plutonium (Pu) and the other more common ones referred to as “minor actinides” (MAs) are neptunium (Np), americium (Am), and curium (Cm).

What to do with spent nuclear fuels (SNFs)?

When SNFs are unloaded from the reactor they are temporarily stored at the reactor site in 40-ft-deep water pools ([Glatz, 2021](#)). Then, the main question that arises is what to do with these highly radioactive SNFs? There are two options. The first one is to dispose of SNFs permanently (that is without further use) in an appropriate deep geological media once having encased them in leak proof shielded casks. Until final disposal, spent fuel can be stored in above ground shielded casks at interim storage facilities either at the reactor site or in a centralized storage facility ([Kessler, 2021](#)). This option is called “open” (or “once-through”) fuel cycle.

The second option is to “reprocess” SNFs in order to separate its constituents and then specifically manage each of them. In effect, it is a unique characteristic of nuclear energy is that used fuel may be reprocessed to recover fissile and fertile materials in order to recycle them in nuclear reactors. For a light water reactors either Pressurized or Boiling the main constituents in the used fuel are: uranium (which still contains about 0.8–0.9% of U235 fissile isotope and which makes up about 94% of the used fuel; plutonium

(which contains on average 60% to 70% of fissile isotopes (Pu239 and Pu241), and which makes up a little more than 1% of the used fuel; and lastly Fission Products (FPs) and Minor Actinides (MAs) which makes up the remaining 5% of the used fuel. In the PUREX process as it is implemented today at the La Hague reprocessing plant in France, FPs and MAS are considered as ultimate waste that are vitrified together. This reprocessing option is called “closed fuel cycle” because uranium and plutonium can be recycled to produce energy.

The choice between these two options remains open in several of the 31 countries (including Taiwan) in which at least one nuclear power reactor operates (at the end of the year 2018). These countries have taken the “wait and see” solution for the management of spent nuclear fuel. This unfortunately leads to the accumulation of SNFs at reactor sites or in a central interim storage facility. Some countries like Sweden, Finland, the United States and Canada (which has only CANDU-type nuclear reactors in operation) have decided to consider SNFs as a nuclear waste and thus to dispose of it permanently in a deep geological media (open cycle). Other large nuclear countries like France, Japan, Russia, and China have instead decided to reprocess their SNFs (closed cycle). **Table 1** below provides a summary of spent fuel management strategies in some OECD countries (NEA CSNI/R, 2015) (Russia and China are not OECD members so are not listed in this table).

France is by far the most advanced and experienced country in the world for reprocessing since it has chosen to reprocess all its spent fuel from the outset of its large nuclear power program in early 1970s (in 2019, there are 58 PWRs in operation in France, totaling 63 GWe of installed power). In total more than 36,000 metric tons (MT) of SNFs (content in heavy metal: MTHM) that have been reprocessed at the La Hague Reprocessing Plant (including SNF from foreign customers) which is equivalent to 3 years of SNF discharged from all the world’s reactors.

All countries which have to manage NW or SNF are signatories of the IAEA “Joint Convention on Spent Fuel and Radioactive Waste Safety” which entered into force in 2001 (IAEA INFCIRC/546, 1997). This Convention is the only legally binding international instrument to address on a global scale the safety of spent fuel and radioactive waste (RW) management. The goal is to achieve and maintain a high level of safety worldwide in SNFs and RW management through ensuring the availability of effective defenses against potential hazards, preventing radiological accidents and mitigating their consequences should they occur.

Reprocessing and recycling SNFs save natural resources

The most common industrial route used today to reprocess SNFs is the so-called “Purex” process. Schematically it consists in the shearing of fuel assemblies to allow for dissolution of the fuel matrix in nitric acid (no dissolution of metallic structures of fuel assemblies). This liquid mixture obtained after dissolution is treated with selective organic solvents to separate the Fission Products and Minor Actinides (NEA, 2012). The U and Pu are then chemically separated and undergo a specific treatment dependent on their future use. The fission products are processed to create a solid glass log for interim storage and the separated Uranium and Plutonium are stored for future fabrication to be recycled into fresh nuclear fuel.

Single recycling of Pu in nuclear reactors (sometimes called mono-recycling) is now commonly practiced in France for more than 30 years using what is called “Mox” fuels that are a mixture of plutonium and depleted uranium in a standard fuel assembly. This leads to an increase of more than 10% of global electricity production from the same amount of natural uranium extracted from the ground. In total near 300 metric tonnes (MT) of Pu have been recycled in the world using Mox fuels in LWRs. This corresponds to a production of electricity of about half of world’s nuclear power for 1 year (for the justification of this figure, see note 1 at the end of the text).

Even more energy can be achieved with the multi-recycling of Pu in LWRs (PWRs or BWRs). This has been demonstrated as feasible on an industrial scale and it will be implemented in France in the future (using a mixture of Pu and slightly enriched uranium, called Mix fuel). This should allow an additional increase of about 5% in global electricity production from a given

Table 1 Present spent fuel management strategies in some OECD countries.

Country	On reactor site interim storage	Centralized interim storage	Reprocessing	Direct disposal
Belgium	Yes	Yes (wet and dry)	Not any longer	Long term policy still to be developed
Canada	Yes	No	No	Future plans
Czech Republic	Yes	No	No	No
Finland	Yes	No	No	Yes
France	No	At reprocessing plant	Yes	No
Germany	Yes	Yes for some plants	Not any longer	Currently not but foreseen
Hungary	Yes	Yes (Dry)	No	Planned, site investigation in progress
Japan	Yes	Yes	Yes	No
Korea (Republic of)	No	Yes (Dry)	Under plan	Under review
Spain	Yes for some plants	Planned, dry	—	—
Sweden	No	Yes	No	Yes
USA	Yes	Under study in Texas and New Mexico)	No	Under study

quantity of natural uranium and should avoid the accumulation of spent Mox fuels. Both these options reduce the need to mine more uranium.

If plutonium could be recycled in fast neutron reactors (FNRs), operated in a breeding mode (which is a proven technology thanks to the successful operation of several power FNRs in the past), the increase in energy production could be enormous. Indeed, in that case the energy potential of natural uranium could be multiplied by a factor 100 making nuclear energy a sustainable source of energy because it is nearly independent of uranium natural resources.

Additionally, if full recycling of reprocessed uranium, after its re-enrichment, allows supplementary energy gain of 10% from existing uranium resources now in the fuel cycle.

In summary the recycling of U and Pu recovered from SNF in LWRs can reduce up to 30% natural uranium requirements to operate a nuclear reactor that produces the same amount of energy.

Reprocessing SNFs makes waste management easier and safer

Beyond the very significant natural uranium savings, reprocessing–recycling allows a substantial improvement regarding safety and management of NW. Indeed, as mentioned before, SNFs contain only 5% (or so) of highly radioactive ultimate waste (FPs + MAs), mixed with 95% of recyclable materials U and Pu, which are relatively very low in radioactivity. Therefore, it is much more efficient to deal with the two groups of nuclear materials separately as we now discuss.

A safer way to dispose of NW

In the Purex process, FPs and MAs are mixed together to form what is called “High Level Long-lived Waste” (HLLW) which are conditioned (at the atomic level) in a very robust and durable matrix which can last for hundreds of thousands years: The material created after mixing with Silica (sand) is a GLASS. This process is called vitrification (for details, see chapter 00611 of this encyclopedia). Many “natural analogs” demonstrate that this glass matrix is particularly resistant to water leaching for hundreds of thousands of years. It has been studied for more than 40 years during which it has been tested, qualified and manufactured for the sole purpose of efficiently confining HLLW.

This is not the case of a nuclear fuel assembly, which has been designed, tested, qualified and manufactured ONLY to produce energy in a reactor. It was not designed to confine HLW for hundreds of thousands of years. Therefore, if SNF are directly disposed of in a deep geological repository, they must be enclosed in a wear-resistant container which absolute tightness must be fully guaranteed during this very long period of time. This is not easy to demonstrate to safety authorities and it is not easy to convince the public that such human made structure is able to ensure total confinement of radiotoxic materials beyond 10,000 years. See Chapter 13.8 for the safety of disposal of HLW.

That said it is important to underline that the amount of HLLW generated by nuclear power is extremely small in comparison to the huge amount of energy that leads to the production of a small amount of HLLW. To give an example, for FP it is possible to make a simple calculation easily understood. Consider the energy released by the fission of an atomic nucleus which is converted to electrical energy in a nuclear power plant with an efficiency of about 34%. This fission gives rise to two FP which total mass is very close to that of the initial atomic nucleus. This allows a given mass of FP to be associated with a given amount of electricity in a rigorous and irrefutable way. For example, by taking an annual electricity production of a 1000 MWe nuclear reactor operating at full power for 1 year (that is 8.760 TWhe) we produce 1.095 tonnes of FPs per year. Furthermore, only 15% of these FPs, hence 164 kg, remain radioactive at the time of vitrification. They consist of 7 long-lived but low-level radioactive radionuclides (by definition!) and 2 radionuclides that have 30 years half-life, Cs137 and Sr90, which activity is reduced by more than a factor of a thousand after 300 years (divide by 2 ten times giving 1024). As for MA, which are all radioactive, the annual production of the reactor considered here is about 35 kg (this mass cannot be calculated “by hand” as for FPs, but of course data are available on this point). In total the amount of radioactive HLLW is less than 200 kg per year for this reactor running at full power all the time. This is an extremely small amount given the energy produced. Compared to this, a 1000 MWe coal-fired power plant releases nearly 300,000 tonnes of ash every year, including 400 tonnes of toxic heavy metals, arsenic, mercury, cadmium, including 5 tonnes of uranium 238 and 13 tonnes of thorium, 350 tonnes of soot and fine particles that escape into the atmosphere, and also 7.8 million tons of CO₂ and 10,000 tons of nitrogen oxide (NOX).

A drastic reduction of potential harmfulness of NW once disposed of in a final repository

Radiotoxicity of a radioactive element decreases over time due to its radioactive decay characterized by the radioactive half-life. A good indicator of the potential harmfulness of Radioactive Waste (RW) is the notion of “potential radiotoxicity” (PR) calculated at any given time. For a given mass of a radionuclide this indicator is obtained by multiplying two factors:

1. its effective dose coefficient by ingestion: The effective dose due to one Becquerel of activity of the radionuclide, that is of one nuclear radioactive decay per second (i.e. one disintegration or nuclear transformation per second). Effective dose is a measure of the hazard of the radionuclide provided by its radiotoxicity which accounts for type of radiation and tissue weighting factors as well as metabolic and biokinetic information. The unit of effective dose is the Sievert.
2. its specific activity at a given moment expressed in Becquerel per unit of mass.

Therefore, for a given mass of a radionuclide, PR is expressed in Sieverts (symbol Sv).

One would add the PR of all radionuclides contained in a RW (with the individual masses of each radionuclide contained in the waste) to have the overall PR of this RW being sure to include in PR calculations of the possible radioactive decay products (daughter) of each radionuclides.

If one takes the total amount of RWs contained in a repository one obtains what is called the “radiotoxic inventory” of the repository. In fact, the values obtained do not have much meaning in the absolute except perhaps for very small masses since the harmfulness thus calculated assumes that a person directly ingests this mass of waste. On the other hand, this PR can be compared with that of the mass of natural uranium (Unat) extracted from the subsoil that was used to generate a fixed amount of energy and thus a given mass of waste (FPs + MAs).

This mass of Unat depends mainly on the initial enrichment of the fuel which is determined by two parameters: (1) the “burnup” (or consumption) of the fuel (that is the total amount of energy extracted from one ton of uranium loaded into the reactor) and (2) the reload fraction of the reactor core (that is the fraction of fuel assemblies that is replaced at each reload, which generally is one third but can be also on half or one forth). For example, for an initial average uranium enrichment of 4.2% (average burnup of about 50 GWd/MTHM, load fraction one third) this mass of Unat which is used to make a ton of fuel (MTHM) is 7.84 t (for a tail enrichment of 0.2%). This comparison provides a valid benchmark for estimating the adverse biological effects on humans of waste since it is compared to the radioactivity of the Unat resource. For vitrified HLLW from reprocessing of SNFs these calculations show that the RP becomes lower than that of natural uranium after about 10,000 years.

Now, if SNF is considered as a nuclear waste and directly disposed of in a repository this achievement is entirely changed because of the presence of plutonium in SNFs. Indeed, the PR of Pu is by far the chief contributor to total PR of a SNF between few hundred years and 100,000 years or so (around 80% to 90%). This is illustrated clearly by Fig. 1 which shows the relative PR of SNF versus time (PR is normalized to one at each time).

Another important point to mention is that PR of a SNF becomes lower than that of natural uranium after 200,000 years or so which is 20 times longer than that of a vitrified waste.

In short, at a time when *recycling is key to human activities*, it is much more appropriate to *not dispose of plutonium* and use it to produce energy.

A significant reduction of volume of conditioned waste

The ratio of gross volumes of an SNF assembly to the volume of glass matrix enclosing FPs and MAs contained initially in this SNF is 3.85 (see note 2 at the end of the text for the calculation). This significant reduction in volume has two consequences.

1. During the interim storage period, SNF are stored either in large pools or air-cooled dry casks, either at-reactor or away-from-reactor in centralized facilities for long-term storage. The vitrified wastes from reprocessing are simply stored at the reprocessing plant site in air cooled steel tubes (shafts) enclosed in a concrete structure (building designated by the generic acronym ‘EV’ at the La Hague reprocessing plant (see Fig. 2). To this regard we can compare the volume of water in cooling pools which is around 2.5 m^3 – 2.8 m^3 /MTHM on average (see Table A3 of Reference NEA CSNI/R (2015)) whereas the volume of the EV concrete structure in which shafts are enclosed (see Fig. 2) is only 0.4 m^3 /MTHM.

It must be noted on this point that adequate cooling of SNFs stored in cooling pools dedicated to long-term storage of SNF may be lost either by malfunction of the pool cooling system (loss of cooling accident) or by loss of the pool water inventory (loss of

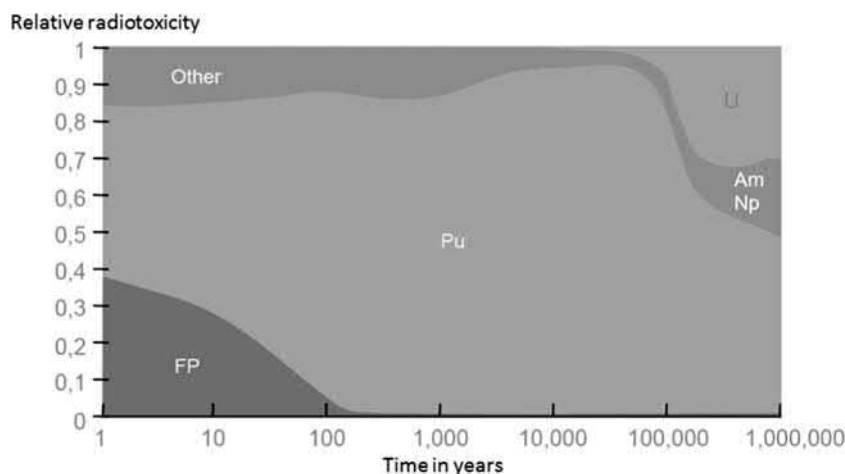


Fig. 1 Relative radiotoxicity of a spent fuel versus time – “Other” are mainly MAs.

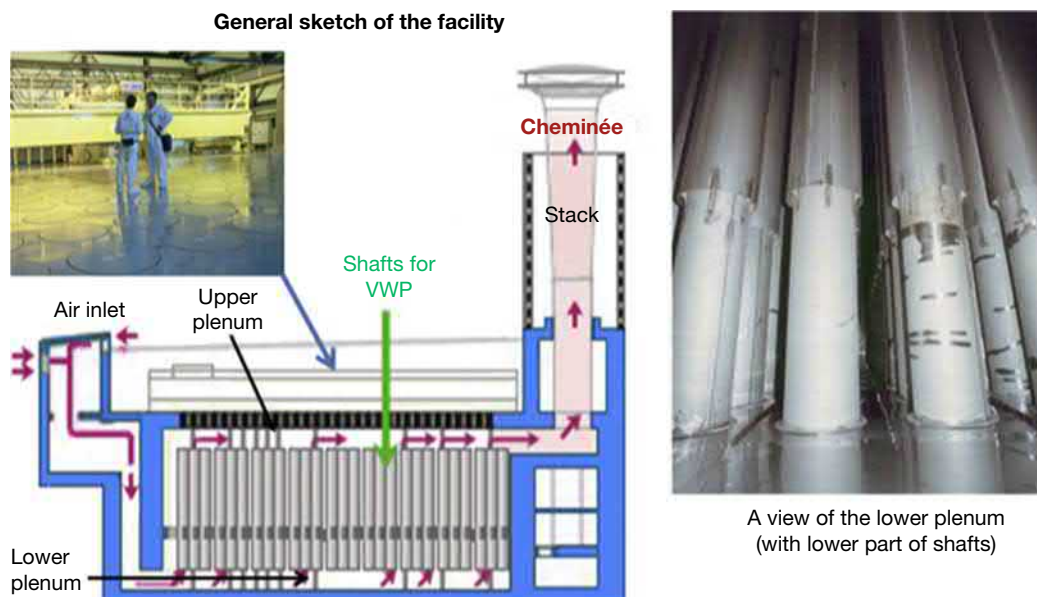


Fig. 2 Storage facilities ('EV') for vitrified waste packages (VWP) - La Hague reprocessing facility – France.

coolant accident). This safety issue must be assessed very carefully: see Reference. (NEA CSNI/R, 2015). This issue is not as acute for the long-term storage of vitrified waste since simple air-cooling systems are enough to remove the heat that they generate. After about 6 to 7 years the system can operate using only natural convection.

2. Prior to geological disposal the SNF must be enclosed in a rugged casks designed to withstand a multitude of possible degradation mechanisms (See Chapter 13.8) The glass matrix is simply enclosed in a light protective steel case in which the mixture of melted glass with FPs and MAs is poured. To make a meaningful comparison in term of volume (and weight) between the two options (disposal of SNFs versus disposal of corresponding vitrified waste), we can take the example of the so-called "KBS3" Swedish geological repository which is one of the most advanced project in the world for final disposal of SNF, located in Forsmark site (140 km north of Stockholm). In this project, the SNF container is a large copper cylindrical canister which accommodates 4 PWR SNF assemblies which contains 1.840 MTHM. Its overall dimensions are 1050 mm of external diameter and 4835 mm of external height (thickness of copper walls is 50 mm); hence a volume of 1.33 m³. Its total weight with the 4 PWR SNF assemblies inside is 26,800 kg (SKB report TR-10-14, 2013). The corresponding equivalent volume of vitrified waste package would be 0.23 m³ with a weight 631 kg. The volume ratio between the two options is thus 5.8 and the weight ratio is 42. This is again a large volume reduction (and a very large weight reduction). It must be mentioned that the real benefit of such reduction in volume for the final disposal depends on many other parameters such type of geological media, architecture of the disposal facility, cooling time of waste before its disposal in the repository and design of SNF canister.

Partitioning and transmutation (P&T): a further step towards reduction of ultimate nuclear waste (NW) burden

Long term radiological impact of NW disposal depends on the amount of long-lived radioactive elements disposed. These elements are mainly the 7 long-lived Fission Products (LLFP), Tc99, I129, Se79, Zr93, Pd107, Cs135, Sn126 and the main isotopes of the 3 Minor Actinides, Am, Cm and Np. To this end, the starting point is to separate (partition) them and, once they are separated to "destroy" them individually via appropriate nuclear reactions. To achieve this, the only practical way known today is to get them to absorb a neutron which either transforms them into another element with a shorter half-life (or even a stable element) or to "break" them through the fission process. This process is called "Partitioning and transmutation", in short P&T.

For a better understanding of this section, it helps first to recall some important characteristics of actinides, which are shown on Table 2.

As stated before, one can see that plutonium is the "major" actinide" in a SNF in terms of mass with a total of 11.2 kg Pu per ton of heavy metal (THM) (> 1%). Dominant minor actinides are Np237 and Am241 (and, in a lesser extent, Am243).

The long term impact of geological repositories for NW

Let's first recall that deep geological repository in an appropriate media is unanimously recognized as the best solution for final disposal of HLLW (Kadak, 2021). Thank to numerous safety studies carried out in the world this mode of isolation of

Table 2 Some important characteristics of actinides in a spent fuel.

Element	Isotope	Half-live (years) ^a	Decay mode	Radioactive daughter	Specific activity (Bq/g) ^b	Specific power (w/g)	Neutron emission from spontaneous fissions (n/g.s)	Ingestion dose coefficient E-7 Sv/Bq ^c	Mass (kg) per THM after 3 years of cooling time ^d
Plutonium	238	87.74	Alpha	U234	6,30E+11	0.57			0.275
	239	24,110	Alpha	U235	2,30E+09	0.0019			6.220
	240	6563	Alpha	U236	8,40E+09	0.0071	900	2.5	2.696
	241	14.35	Beta(–)	Am241	3,80E+12	0.0033		0.047	1.353
	242	375,000	Alpha	U238	1,50E+08	0.00012	1700	2.4	0.720
Neptunium	237	2,144,000	Alpha	U233 (via Pa233)	2,60E+07	0.000021		1.1	0.633
Americium	241	432.2	Alpha	Np237	1,30E+11	0.11		2	0.412
	242 m	141	Internal	Am242	3,90E+11	0.0045		1.9	0.00087
	242	0.00183	83% Beta (–)	Cm242	3,00E+16	950		0.003	0
	243	7370	Alpha	Pu239 (via Np239)	7,40E+09	0.0064		2.0	0.165
Curium	242	0.446	Alpha	Pu238	1,20E+14	120	20 millions	0.12	0.000012
	243	29.1	Alpha	Pu239	1,80E+12	1.8		1.5	0.00054
	244	18.1	Alpha	Pu240	3,00E+12	2.8	12 millions	1.2	0.0553
	245	8500	Alpha	Pu241	6,40E+09	0.0057		2.1	0.00456
	246	4730	Alpha	Pu242	1,15E+10	0.01	9 millions	2.1	0.00050

^aData from "Chart of nuclides" 7th edition 2006 - Karlsruhe.^bFigures in this column are noted with a comma (European notation) instead of a period (US notation) due to Excel format constraints.^cData from ICRP 2012 publication 119 (based on ICRP publication 60).^dUO₂ fuel - Initial enrichment 4.2% - Burnup 52 GWd/THM – 1/3 of core reloaded at every cycle (16 months) - (5 years of cooling after unloading).

long-lived radioactive waste has proved to be highly secure and environmentally sound subject to the selection of an adequate geologic media. In particular this media must stable for at least several hundred thousands years and away from any significant circulation of free water in order to ensure radionuclide radiotoxicity has become sufficiently low (due to natural radioactive decay) before their possible migration to the surface could constitute a danger to living species and the environment. Another criterion generally taken into account in choosing a storage site is that of minimizing the risk of human intrusion once the site becomes unmarked (i.e., the site is released for public use after the monitoring period). To achieve this, waste must be disposed of in a repository as deep as possible (typically at least 200 or 300 m below the surface) and as far away from any natural resource as possible.

Almost all host formations (granite, clay, salt) feature excellent confinement properties in the long term for HLW for the normal evolution of geologic repositories, that is for the future evolution of the disposal system under the predicted (expected) thermal, hydrological, chemical and mechanical conditions based on the knowledge of the past evolution of these geological conditions. Hence, calculated dose levels at the surface are significantly lower than the most stringent regulatory limits and natural radiological background. This very small long term radiological impact is mainly due to the soluble long-lived FPs (LLFPs) such as Tc99 which half-life is 211,000 years (although sparingly water-soluble as metal or oxide in reducing conditions of deep aquifers), and I129, which half-life is 15.7 million years for SNF disposal. One may note in passing that there are only tiny amounts of I129 in glass because iodine is highly volatile at the temperature of vitrification (1100 °C) and thus cannot be incorporated in the glass with other FP. It is therefore released into the environment at the La Hague plant (under stringent regulatory limits and strict control) mainly in the NaI liquid form (> 96%) in sea water where it is diluted with natural iodine. Because of this dilution and from the fact of its very low specific activity (due to its very long half-life) the radiological impact of this dispersion in marine environment is practically nil.

Unlike long-lived fission products (LLFPs), MAs are almost insoluble in underground waters and they migrate extremely slowly in reducing conditions prevailing in deep geological media. Consequently, MAs have practically no radiological impact even in the very long term under normal evolution of a NW repository. Nevertheless, the presence of minor actinides in the waste is the main source of long-term potential radiotoxicity (PR): after three centuries, nearly 99% of the PR of the vitrified waste forms currently produced will be due to the presence of americium, curium, and their decay products. The share of the different MAs in this PR is illustrated by the curves of the Fig. 3A drawn from (Magill et al., 2003) and Fig. 3B drawn from (CEA, 2012a) (these figures have been reformatted for the purpose of this article).

These curves show that americium is the dominant contributor to the PR of MAs from 100 years and up to 100,000 years, mainly because of its isotope Am241 whose half-life is 433 years (its radioactive daughter being Np237) and which proportion by mass is 2/3 of that of the total of all isotopes of americium and curium combined. The intrinsic PR of curium, which proportion by mass is less than 10% of americium plus curium, is always much lower than that of americium. The PR of curium falls below that of natural uranium after few hundred years. As a matter of fact, the major isotopes are Cm243, whose half-life is 30 years (its radioactive daughter being Pu239) and Cm244, whose half-life is 18 years (its radioactive daughter being Pu240). As for neptunium (which has only one isotope Np237), it is clear that its contribution is much lower than that of other MAs up to about 100,000 years,

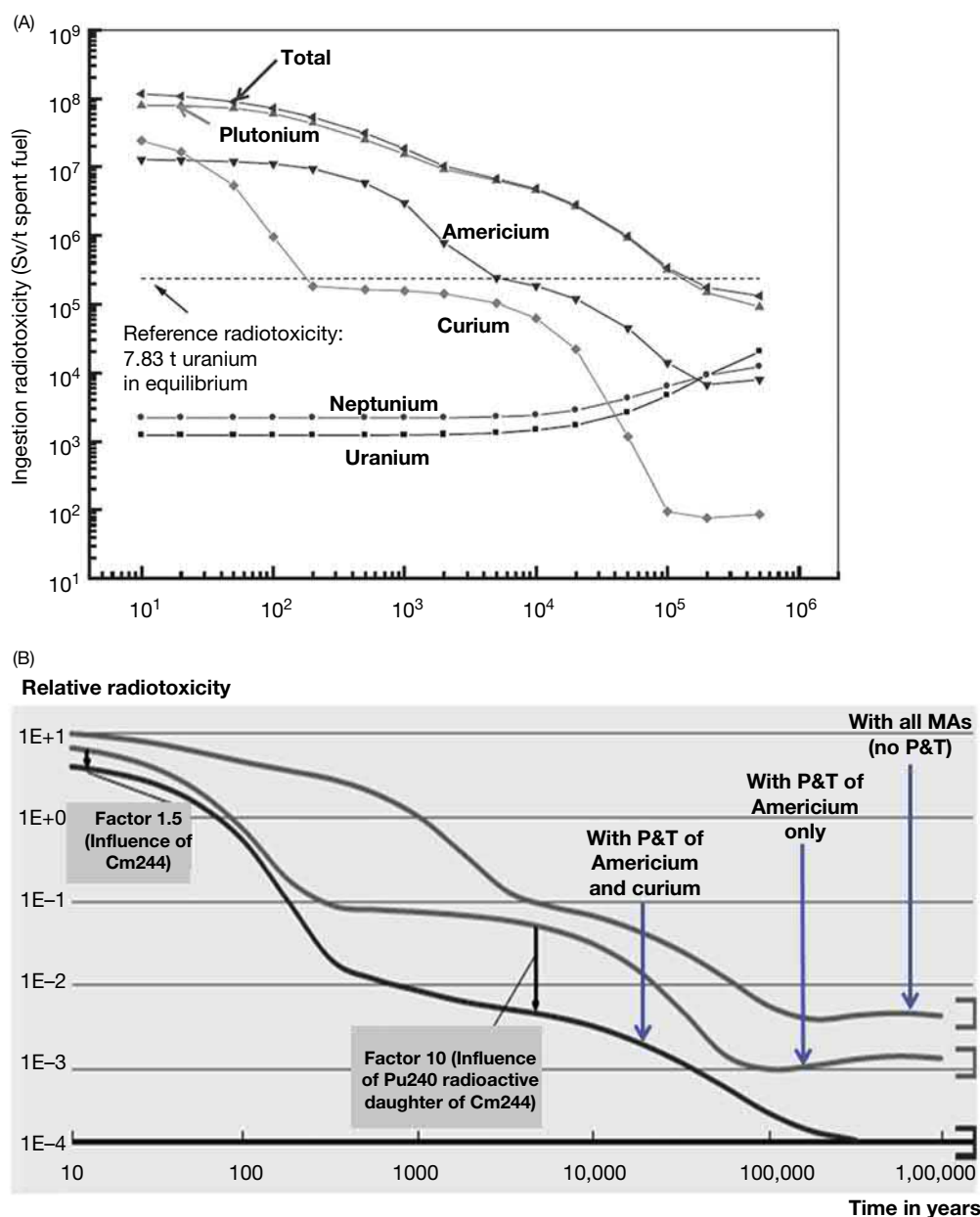


Fig. 3 (A) Actinide radiotoxicity in spent PWR fuel vs. time. Results are grouped by element present after 6 years of cooling and the radiotoxicity of each isotope of an element includes that of their radioactive daughters. (B) Relative radiotoxicity as function of time normalized to 10 at 10 years: effect of P&T on americium only and on americium + curium.

a date after which total PR remains lower than the reference radiotoxicity of natural uranium in equilibrium. This is due to its very long half-life (2.144 million years) and to the low specific radiotoxicity of its radioactive daughter, U233 (half-life 160,000 years).

Performance assessment studies

To obtain a quantitative evaluation of potential releases of radioactivity from a disposal facility into the environment and to assess the resultant radiological doses, repository “performance assessment” (PA) studies are carried out. PA can refer to the process, model, or collection of models used to estimate future doses to human receptors. Typically, a PA is conducted to demonstrate whether a disposal facility has met its performance objectives.

Let us just note here that these studies allow not only to estimate the long-term radiological impact of NW disposal but also to assess the potential benefit of the reduction of amounts of long-lived radioactive elements on this impact. Before examining the technical feasibility of P&T, let’s discuss its potential benefits on geological repositories.

Effect of P&T of NW on sizing of geological repositories and on its long-term impact

We have seen above that the radiological impact of a NW repository in the very long term (> hundreds of thousands years), if any, would come only from Long-Lived Fission Products (LLFPs), that cannot be easily transmuted at industrial scale (and even cannot be transmuted at all for several of them). Fortunately, this impact is so low under all credible scenarios that this very partial transmutation would have practically no effect on this radiological impact.

The situation is quite different for MAs. As a matter of fact, lessons learned from numerous PA studies carried out in the past have shown that advanced fuel cycles with P&T of MAs can offer potential benefits to NW disposal in geological repositories. More specifically, two main advantages can be anticipated from P&T of MAs:

- Reduction the potential harmfulness arising from ingestion of the radioactivity of MAs present in long-lived high-level waste in the so-called “human intrusion” scenario (though very unlikely scenario) or in scenarios considering an unforeseen change of the geochemical conditions favoring actinide migration, e.g. establishment of an oxidizing environment in the repository
- Reduction of the heat load (“thermal load”): larger amount of wastes can be stored in the same repository (depending on the host formations). Studies carried out in the frame of the European “Red Impact” project ([RED IMPACT, 2006](#)) have shown for example that removing a significant part of MAs from waste allow a reduction of the emplacement galleries length up to a factor 3–6 after interim storage cooling time of at least 50 years for deep geological repositories in clay and hard rock formations.

As for the first point (potential radiotoxicity PR) we have already indicated that Am241 is the major contributor (see [Fig. 3A](#) and [B](#)). We have also observed on these curves that P&T of both americium and curium would diminish the PR by a factor of 100 after a few centuries up to few thousand years and a factor 15 to 20 beyond (up to 100, 000 years or so). If full P&T was implemented for all MAs (Am, Cm and Np), i.e. assuming no loss of material during reprocessing, the total PR of vitrified waste would fall to a level equivalent to that of all the uranium mined for fuel fabrication today in less than 500 years (because in that case the remaining PR would be only that of LLFPs).

As for the second point (heat load) the contribution of FPs and MAs are plotted on [Fig. 4A](#) drawn from ([Wigeland, 2006](#)) for spent fuels and [Fig. 4B](#) drawn from ([CEA, 2012a](#)) for vitrified waste.

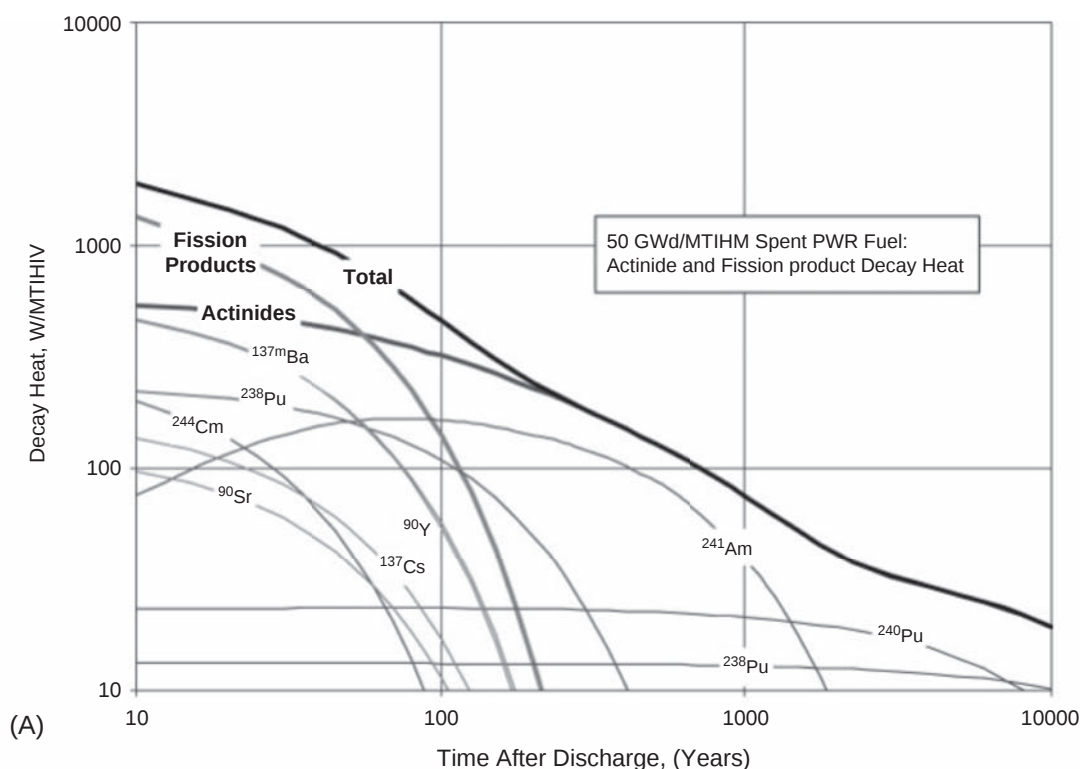
For spent fuels ([Fig. 4A](#)), the decay heat is mostly from actinide elements after 60 years, with the important actinide elements being plutonium and Am241. Beyond 200 years, the decay heat is caused entirely by isotopes of plutonium and americium, out to at least 10, 000 years. For vitrified waste ([Fig. 4B](#)), Am241 becomes dominant (> 50%) after about 80 years and then it becomes more and more the dominant contributor to the total decay heat: 70% at about 120 years and almost 100% after 300 years. Yet, the heat released from vitrified waste packages determines to a large extent the design of repository disposal cells: the lower the heat load in the waste, the higher the disposal density of the repository. Removing MAs from the final waste and providing a prior interim storage period long enough to allow significant radioactive decay of the short-lived fission products could thus significantly reduce the size of the repository. To this regard, the French agency for radioactive waste management (ANDRA) estimated a gain by an order of magnitude on the footprint of the disposal cells for vitrified waste packages in the clay disposal concept currently envisaged in France (CIGEO project) for a 120 years interim storage period before their emplacement in the repository. This would represent an overall footprint reduction for the entire repository, including intermediate level waste storage and access infrastructures by about a factor of 3 (see Reference [CEA \(2012b\)](#)).

Feasibility of P&T

Numerous studies have been carried out on P&T issues in most countries having a significant nuclear program, in particular in France and some other European countries but also in the USA and Japan (see for example Reference [OECD \(2011\)](#) pages 39 to 59 for an overview of these studies).

These studies (theoretical and experimental) have shown that transmutation of LLFPs by neutron irradiation is ineffective for the majority of them. Nevertheless, there has been some examination of the potential for treating Tc99 and I129, which appear to be the only LLFPs that can be transmuted in reactor systems. Transmutation of Tc99 is possible if it is present as a metallic target, since the transmutation product is inactive Ru100. However, due to the very small thermal cross-section of 20 barns, a high thermal neutron flux is needed as is a high loading in the reactor with an optimized moderator to target radius. Furthermore, multi-recycling is needed to achieve a high rate of transmutation of Tc99. Another fact which makes the transmutation of Tc99 less interesting is that PA studies have shown that the chemical form of Tc in deep geological conditions makes it much less soluble in water, thus not too mobile (Reference [CEA, 2012a](#)).

Transmutation of I 129 is a very difficult task because it has a moderate to small thermal cross-section (27 barns) and no thermally stable iodine matrix has been found. In addition, two conflicting safety requirements must be fulfilled: on the one hand, the confinement of the iodine compound in the target capsule during irradiation and, on the other hand, the discharge of the produced xenon. A vented capsule with an iodine filter needs to be investigated. The upscaling of such complex irradiation procedures to industrial quantities is not obvious. More detailed technical information on P&T of LLFPs can be found in Reference [OECD \(2006\)](#) pages 88 to 100 and [IAEA \(2004\)](#) pages 97 to 100.



Vitrified waste package from co-reprocessing of UOX spent fuel (85%) and MOX spent fuel (15%)

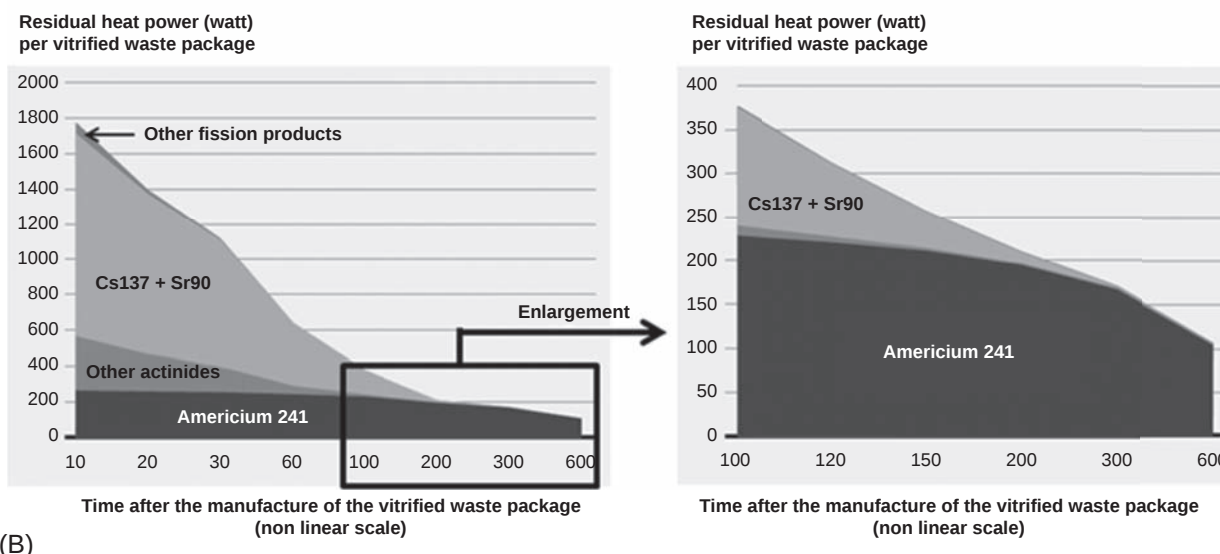


Fig. 4 (A) Key contributors to the decay heat (residual heat) in spent PWR fuel irradiated to 50 GWd/MTIHM. Note that plutonium isotopes are not present in vitrified waste except Pu losses which are very small ($< 0.1\%$) but more importantly as decay products of curium isotopes which are produced in a very low amounts and therefore which does not contribute significantly to the total decay heat of vitrified waste: see (B) Dominant decay heat (residual heat) contributors in vitrified waste. (A) Wigeland A (2006) Separation and transmutation criteria to improve utilization of geologic repository. *Nuclear Technology*. 154. (B) CEA, Nuclear Energy Division (2012) *Séparation et transmutation des éléments radioactifs à vie longue*, Tome 2.

Studies of the partitioning of MAs have been carried out (and are being continued) in France mainly by the CEA (in cooperation with EDF and other industrial actors) within the framework of the laws governing research and management on NW. This important program includes:

- Experimental test at laboratory scale on actual spent fuel samples (several kg) of the processes that have been developed by the CEA to recover minor actinides using new extractants that are both selective and resistant. Several options were considered,

corresponding to various possible recycling routes: group separation of all the actinides for homogeneous recycling, sequential separation for minor actinide recycling in core blankets, separation of americium alone, etc. The observed performance was excellent (recovery yields above 99%) and additional tests were carried out to approach the operating conditions at industrial scale. The results are very encouraging (especially with regard to the endurance of the molecules involved) and are very promising for possible commercial application of the concepts investigated.

- As for transmutation, results acquired since 2005 by the CEA have shown that the minor actinides could be efficiently transmuted only in a fast neutron spectrum (whether in power reactors or in dedicated devices). This was confirmed later by analyzing irradiation in the Phenix fast neutron reactor in homogeneous mode ("Superfact" experiment) and heterogeneous mode (irradiation of pin segments containing americium-bearing pellets on various media).

In the frame of this article we can only provide a brief summary of the main outcomes of this program.

Considering the previous discussion on radiotoxicity and decay heat, these studies confirm that the first target for a transmutation strategy could be americium since it has the most limited impact on existing recycling operations. On the other hand, neptunium is seen as a secondary candidate for transmutation due to its very long half-life and low activity. Curium transmutation is virtually ruled out due to the high activity of curium isotopes. As a matter of fact, its transmutation in heterogeneous mode (target fuel assemblies) is almost impossible even if it is mixed with americium especially because of its intense specific heat. Its recycling in homogeneous mode is subjected to severe constraints (fabrication and transport of fuels) which are an order of magnitude greater than for recycling americium, because of its very high radiation level. In particular, curium is an intense source of neutrons (from spontaneous fissions of even isotopes): see [Table 2](#). Furthermore, transmutation of curium is a slow and complex process that generates higher isotopes of which Cf252 that can be extremely troublesome due to its very high specific neutron emission rate. In one word, it seems that disadvantages of the P&T of curium far outweigh its potential benefits.

Another result of these studies is that homogeneous recycling of MAs in fast neutron reactors may affect some core safety parameters. Recycling them in core blankets (heterogeneous mode) leads to difficulties in fabrication and handling of assemblies, before and after irradiation in the reactor. P&T of MAs at industrial scale could also increase occupational doses of workers (this point has been particularly addressed in Reference [Grenèche et al. \(2001\)](#)). More generally, it could probably increase the risk associated with the industrial implementation of the entire P&T process (for example consequence of a failure somewhere in one of the facilities or equipment dedicated to P&T). It could also increase the overall cost of the fuel cycle. To this regard, the CEA assessment is that P&T would increase the levelized cost of electricity from 5% to 10% depending on various options (homogeneous or blankets, minor actinides or americium alone). This extra cost is nonnegligible and must be carefully considered against expected benefits of P&T, including a greater social acceptance of nuclear waste disposal (this point is discussed in more details in Reference [Grenèche et al. \(2006\)](#)).

Conclusion

Recycling of Plutonium and Uranium in light water reactors can increase energy production from natural uranium ore by 30%. If fast breeder reactors are used, nuclear energy can be a sustainable long-term energy source. Indeed, fast breeder reactors are able to use the energy potential of the totality of natural uranium (100%) by transforming most of the U238 into plutonium, whereas current reactors use barely 0.5% of this natural uranium to produce nuclear energy.

Recycling of used nuclear fuel has already demonstrated a significant reduction of the volumes and long-term radiotoxicity of nuclear high-level waste through the reprocessing of 37,000 tonnes of spent LWR fuel to recycle the plutonium and uranium. These closed fuel cycles provide the opportunity to partition classes of nuclear waste and to manage each class in a separate waste form according to its individual characteristics. Advanced waste management strategies include transmuting of selected nuclides, cost effective decay heat management, flexible interim storage, and customized waste forms for specific geologic repository environments. These strategies hold the promise to significantly reduce the long-lived radiotoxicity of waste destined for geological repositories by at least an order of magnitude. Such reductions, and the ability to optimally condition the residual wastes and manage heat loads, will permit more efficient use of repository capacity and further enhance overall safety of the final disposal of radioactive wastes.

Nevertheless, whatever procedure is implemented to manage waste streams from different back-end fuel cycle options including advanced processes for P&T of long-lived radionuclides, a deep geological repository to host HLLW is unavoidable and it is a safe way to dispose of them.

Notes

1. This equivalence is obtained using the French case for which about 10 MT of Pu is recycled every year in MOX fuels (in 24 PWRs of 900 MWe power each) thereby providing 10% of its 400 TWhe nuclear electricity (rounded up), that is 40 TWhe. Since nuclear power in the world produces about 2500 TWhe per year (2563 TWhe exactly in 2018), half of this production (1250 TWhe) can be produced with 300 MT of Pu recycled in MOX fuels (result obtained with a single rule of three). Another way to obtain the same result is to consider directly the energy released by one fission of a Pu239 atomic nucleus (210 MeV) and suppose that only half of the 300 MT of Pu can fission (because of isotopic composition of Pu and because incomplete fuel

burnup). The exact calculation of plutonium mass to produce 1250 TWhe gives 156 MT, which is coherent with the previous result (about half of the 300 MT).

2. We take the example of a standard fuel assembly of a typical 900 MWe PWR which fuel pins (tight zircaloy tubes) containing enriched uranium in the form of oxide are arranged in a 17×17 tubes square array. The square cross-section is 462 cm^2 and height is 4 m, hence a volume of 0.185 m^3 . We also take a standard average burn-up of 45 gigawatt-days per metric tonnes of heavy metal (HM) loaded (GWd/MTM) which leads to a content of 21.5 kg of FPs and 0.6 kg of MAs in the corresponding spent fuel assembly. The initial mass of HM in this fuel is 460 kg of 3.7% enriched uranium. If this SNF is reprocessed using the Purex process (as implemented in France at the La Hague plant), the glass matrix obtained from vitrification contains 16.5% of FPs and 0.5% of MAs *w/w* and has a density of 2.67. Therefore, the vitrification of $(21.5 + 0.6) = 22.1 \text{ kg}$ of (FPs + MAs) from one SNF assembly leads to 130 kg of glass, that is a volume of 0.048 m^3 of glass. Hence the *ratio* of 3.85 between the gross volume of an SNF assembly and the corresponding glass matrix. If we take into account the protective steel case and gripping device of the standard vitrified waste package containing 400 kg of glass (that is a volume of 0.15 m^3) the volume of the package is in fact 0.18 m^3 and this ratio is somewhat less: 3.2. We do not consider here the other category of nuclear waste from reprocessing that is “compacted waste.” This waste result from the compaction of metallic structures of the fuel assembly (cladding hulls, end pieces (top and bottom nozzles), grid spacers and some other metallic components like springs, etc.) produced by the shearing of the SNF at the head end of the process (before dissolution of the fuel matrix). This is because they are *not* high level waste and almost *not irradiating* (their radioactivity is at least 100 times less than that of vitrified waste and is mainly alpha type of radiation) which does not generate heat (less than 10 watts per package). Consequently, they can be managed in much more simple way than vitrified waste whatsoever for the duration of interim storage or for final disposal.

See Also: Fuel Rod, Fuel Assembly, and Reactivity Control Assembly Design; Routes for Pu Multi-recycling; Self-Sustaining Breeding in Advanced Reactors: Comparison of Fuel Cycle Performance; Synthetic Hydrocarbon Fuels From H_2 and Captured CO_2 ; The Current Industrial Mono-Recycling of U and Pu: Recycling Step.

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Nuclear Waste Can Be Disposed of Safely

Andrew C. Kadak, Kadak Associates, Inc., Port St. Lucie, FL, United States

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Introduction

One of the key social issues affecting public opinion on the use of nuclear energy is that associated with nuclear waste disposal. The public's primary concern is reactor safety but very quickly comes the question: "What are you going to do with the waste"? In the United States, this issue has significant technical and political ramifications. In a 1957 report by the National Academy of Sciences entitled "The Disposal of Radioactive Waste on Land" (Hess, 1957), Committee of Waste Disposal of the Division of Earth Sciences recommended that high level nuclear waste (primarily used fuel from nuclear reactors) could be safely disposed of in stable geological formations. In 1970, the US began searching for possible disposal sites based on this recommendation. This same recommendation was followed by other nations of the world. Table 1 summarizes the national waste disposal efforts of many nations some of which are still in the research phase and others are in the development phase. What is interesting in this table is the different geologic media being used by a country depending on their particular geology (Wikipedia, 2019; International Atomic Energy Agency, 2018).

What follows is a discussion of the evidence of the feasibility of safe geological disposal of radioactive materials using as examples natural nuclear reactors and an existing United States nuclear waste geological disposal site in New Mexico. A summary of the US licensing studies justifying the safety of the proposed Yucca Mountain High Level Waste (HLW) site and the Finland's HLW disposal project which is now under construction is provided.

Reactors in nature

The real question is can nuclear waste be safely disposed? Perhaps the best answer to this question is found in nature. More than 2 billion years ago, natural uranium was of sufficient concentration in U-235 and under the right geological conditions was able to sustain a natural fission chain reaction, i.e. a natural nuclear reactor (Stuchbery, 2021). The concentration (enrichment) of U-235 in natural uranium 2 billion years ago was about 3% which is about what is necessary for a fission reaction to occur when in the presence of water. This reaction produces byproducts which are referred to as fission-products or nuclear waste. The half-life of U-235 is about 700 million years during which time about half of the U-235 will have decayed away to other non-fissile isotopes. Thus, over 2 billion years, the concentration of uranium-235 was reduced from about 3% to today's concentration of 0.72%. Many of the fission products are also radioactive and their natural decay is an important feature of nuclear waste since its radioactivity or hazard decreases with time. After about 10,000 years the radioactivity level from this waste is about the same level as natural uranium.

To date, 15 such natural reactors have been identified in a uranium mine in Oklo, Gabon, South Africa (Hess, 1957; Smellie, 1995; Cowen, 1976). This site has been studied for many years to determine how such a natural reactor might have been possible. Good summaries of how the natural reactor worked are described by Bressen (2018) and Meshik (2009). In essence, given that the concentration of U-235 was about 3% and sufficient water was present with no strong neutron absorbers that might capture the natural neutron emission from uranium, the "reactor" had the needed moderation to slow down the neutrons emitted from uranium fission allowing for fission-chain-reaction to take place. What happened when the "reactor" went critical, namely had a self-sustaining nuclear chain reaction, was the water was heated by the energy released as a result of the fission process and boiled off. This essentially shut the reactor down. Once cooled down and water returned, the reactor was restarted repeating the boil off cycle. This on/off cycle continued for about a million years until there was not a sufficient concentration of uranium-235 left since it

Table 1 Status of nation's high level waste repository programs.

Country	Location	Waste	Geology	Status
Argentina	del Gastre	VHLW	Granite	Under discussion
Belgium		VHLW	Clay	Under discussion
Canada	Ontario	SF		Siting
China				Under discussion
Finland	Eurajoki	SF	Granite	Under construction
France	Burre	VHLW	Mud stone	Siting
Germany	Gorleben	VHLW	Salt dome	Proposed on hold
Japan		VHLW		Under discussion
Sweden	Forsmark	SF	Granite	License application
Switzerland		VHLW	Clay	Siting
United Kingdom		VHLW		Under discussion
United States	WIPP	Trans U	Salt bed	In operation 1999
United States	Yucca Mtn	SP	Ignimbrite	Canceled 2010

VHLW, Vitrified high level waste; SP, Spent fuel; Trans U, Transuranic waste.

was consumed by fissioning or reduced by natural radioactive decay. The accumulation of fission products also interferes with sustenance of a chain reaction.

The Gabon site was used to study what happened to the fission products over time. This is a critical issue for safe nuclear waste disposal since the goal is to understand how these byproducts may have migrated from this underground mine to the environment. The good news is that the studies show minimal migration from the original site of the fission reaction 2 billion years ago. In fact, one of the key waste isotopes of concern is plutonium produced by neutron capture in U-238 (which makes up over 99% of the uranium in the earth today, and ~97% 2 billion years ago). Analysis has shown that the plutonium moved less than 10 ft away from its original creation some 2 billion years ago (Kaushuk, 1976). This natural analog provides scientists great confidence in geological disposal options under consideration. Fig. 1 is a photograph of one of the Oklo natural reactors in the mine.

What is nuclear waste?

High level nuclear waste comes from the operation of reactors, nuclear reprocessing (recycling) plants, nuclear weapons production facilities, nuclear ships and submarines operated by naval forces and wastes that come from nuclear accidents such as Fukushima, Chernobyl and Three Mile Island. It is called high level because of its intense radioactivity that requires special shielding to prevent human exposure.

What people commonly refer to as nuclear waste as it affects the social issue of the use of nuclear energy is that associated with the used fuel that comes from the nuclear reactor after about 4 years of operation. This ceramic uranium fuel pellet is contained in tubes of zirconium metal (fuel rods) about 12 ft tall in box like assemblies as shown in Fig. 2 below. The composition of the used



Fig. 1 A natural nuclear reactor in Oklo, Gabon. Uranium oxide remains are visible as the yellowish rock. Photo credit: Robert D. Loss/APOD.

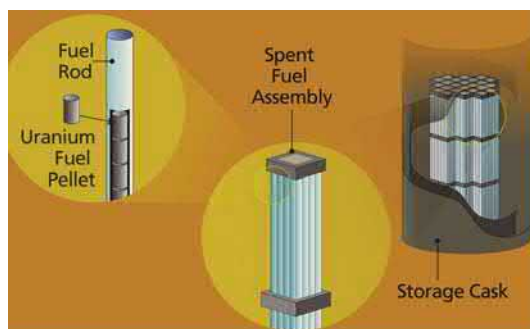


Fig. 2 Schematic of a nuclear fuel assembly. Source: <https://public-blog.nrc-gateway.gov/2015/03/12/dry-cask-storage-the-basics/>

fuel, its radioactivity and its heat generation as a function of time following removal from the reactor core are described in Glatz (2021).

At the reactor sites, this used fuel is stored in 40 ft deep water pools as shown in Fig. 3, which remove the heat generated in this fuel due to the radioactive decay of its fission products and provide shielding from any radiation emitted from the used fuel. Once the used fuel has cooled down sufficiently not requiring anything but natural air cooling, the used fuel is transferred into dry cask storage facilities, as shown in Fig. 4, that provide adequate air cooling, shielding against radiation and physical protection (Nuclear Regulatory Commission, 2016).

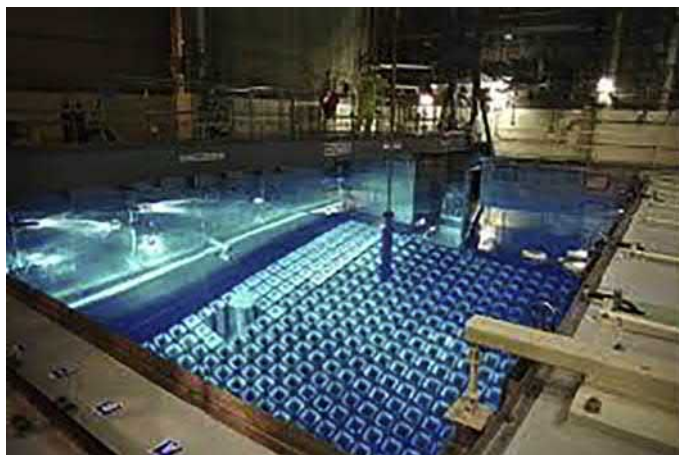


Fig. 3 Fuel assembly in a spent fuel storage pool at reactor site. Source: ansnuclearcafe.org.



Fig. 4 Dry cask storage of used fuel (naturally cooled by vented air in bottom). Source: <https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/dry-cask-storage.html>

The used fuel is very radioactive and needs to be shielded by concrete or metal casks. The good news is that the quantity of this used fuel is very small. As an example, all of the spent fuel produced by all the US operating reactors since 1950 can be placed on a football field 10 ft deep. This quantity of material can easily be stored in a single geologically mined cavity in the earth.

At present, many of the spent fuel storage pools at operating reactors have reached their maximum capacity and utilities are building dry cask storage facilities at their sites pending the Department of Energy's readiness to start taking the spent fuel from the sites either to an underground repository or a centralized interim storage facility which also has not been built. By contract and by law, the United States government is obligated to remove and dispose of used nuclear fuel from nuclear plant sites.

Waste isolation pilot plant—The US first geological disposal site

In the United States, nuclear waste disposal efforts initially focused on underground salt domes that have been stable for millions of years. Early efforts in siting a high level waste repository focused on Lyons, Kansas which was ultimately found unsuitable due to water in-leakage. Despite this early failure, the United States was able to site an underground geological disposal site in bedded salt for plutonium contaminated nuclear waste from US nuclear weapons facilities.

The United States has an operational geological underground repository for nuclear waste in New Mexico. The Waste Isolation Pilot Plant (WIPP) has been in operation since 1999 for transuranic¹ (TRU) (primarily plutonium) contaminated wastes in an underground salt deposit (Energy, 2019; Environmental Protection Agency, 2020). The nuclear wastes disposed in the WIPP consist of wastes which are byproducts of the nation's nuclear weapons and defense programs. These wastes include plutonium contaminated tools, rags, sludge, and other materials that have been contaminated with radioactive materials.

In 1974 the Atomic Energy Commission selected a bedded salt deposit near Carlsbad, New Mexico for disposal of TRU wastes. The bedded salt was created 250 million years ago from an ancient Permian sea bed in an underground 2000 ft thick salt deposit. After years of study, the US Environmental Protection Agency licensed WIPP for safe long term disposal of TRU wastes. The state of New Mexico and the US Department of Energy reached an agreement to allow for the construction, operation and oversight of WIPP for its operating lifetime.

A picture of the WIPP from the surface is shown in Fig. 5 with Fig. 6 showing a mined salt cavity and barrels of plutonium and other TRU contaminated wastes. As can be seen in the figures, the TRU waste is contained in barrels which are contact-handled meaning that the radioactivity is very low. WIPP also has remote-handled TRU waste which requires special handling. As of January 2017, 170,000 containers plus other boxes and overpacks amounting an equivalent of 450,000 55 gal drums have been disposed of in the WIPP facility.

The value of salt as a disposing medium is that it has been stable for millions of years, contains no flowing water and is self-sealing in that over time the salt encapsulates the waste packages. One of the challenges is that periodically, the operating shafts have to be re-mined to keep the shafts open due to the creeping of the salt.



Fig. 5 Waste isolation pilot plant. Source: <https://www.mat.ucsb.edu/~g.legardy/academic/courses/09f130/10000yrs/WEB/WIPP2.htm>

¹Transuranic elements are those whose atomic number is greater than 92 (uranium).

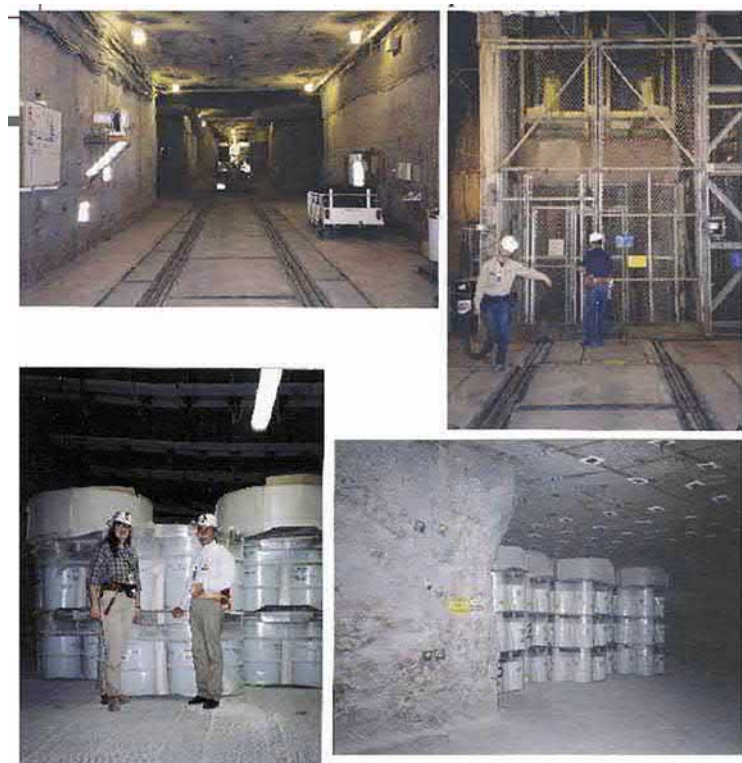


Fig. 6 Underground salt vaults. Source: Kadak Photo.

The problem of disposing high level nuclear waste

The political challenge of nuclear waste disposal is dominated by opposition to siting a nuclear “waste dump” enabling politicians to use the waste issue to help their election prospects especially in the United States. While this may not be universally applicable in other countries, public fear and apprehension of nuclear waste disposal drives the political opposition to siting. In the United States after years of study for possible sites by the Department of Energy, the US Congress passed the Nuclear Waste Policy Act of 1982 and amended in 1987 ([United States Law, 1982](#); [Department of Energy, 2004](#); [O'Donnell, 2019](#); [US Congress, 1987](#)) which ultimately selected the Yucca Mountain site in Nevada as the primary location for disposal of used fuel subject to successful licensing. This law is still the law of the land. Yucca Mountain is approximately 90 miles northwest of Las Vegas on the former nuclear weapons testing site. It is a relatively desolate area with very little rainfall. A picture of the Yucca Mountain site is shown in [Fig. 7](#). The potential repository would be about 1000 ft below the mountain in a series of mined tunnels.

Part of the law mandates the collection of funds from nuclear generating stations based on the amount of electricity generated by the plants. This fee is \$0.001 per kilowatt-hour of electricity which is to be used to fund the cost of building and operating a high level waste disposal site. Up to 2018, about \$41 billion has been collected for nuclear waste disposal ([General Accounting Office, 2011](#); [DOE Office of Environmental Management, 2019](#)). By 2011, the Department of Energy and many of the nation's national laboratories spent \$15 billion to study the Yucca Mountain site and other possible sites.

Following the determination of site suitability in 1998, the DOE proceeded to prepare a license application to the Nuclear Regulatory Commission (NRC) for construction and operation of the waste disposal site. In 2008, the DOE submitted its application to the NRC for a detailed review. Unfortunately, once the review was nearly complete, Senator Reid of Nevada, who was the majority leader of the Senate at the time, withheld funding from the NRC to conduct hearings on the safety of the project. The NRC did issue a Safety Evaluation Report ([Nuclear Regulatory Commission, 2010](#)) for the entire project which could have led to a construction permit and eventually an operating license. The NRC's conclusion of the Safety Evaluation Report was:

On the basis of the information provided in the license application and the commitments specified in the Safety Evaluation Report (SER), Volume 1, Chapters 1–5 and Appendix, the NRC staff concludes that the Yucca Mountain repository meets the following requirements of 10 CFR Part 63² with respect to a construction authorization.

²10 CFR 63 outlines the requirements that the repository must meet to be licensed for construction and operation.



Fig. 7 Aerial view of Yucca Mountain looking south. Showing the desert environment and remote location. Source: <https://www.energy.gov/sites/prod/files/edg/media/SER.PDF>

The Obama administration subsequently ordered the Department of Energy to withdraw the application essentially canceling the project (Northey, 2011). However, the law of the land still requires that the repository be built at Yucca Mountain. At the present time, no work is being done at Yucca Mountain.

United States Yucca Mountain high level waste repository

What follows in this section is a short discussion of the technical viability of the Yucca Mountain site based on the license submittal to the NRC showing that safe geological disposal is possible. Prior to its suspension, the Yucca Mountain Repository site had been studied for over 20 years. We use the Yucca Mountain NRC licensing review of the safety of deep geological disposal and an example of why nuclear waste can be disposed of safely. While the licensing was not complete, much of the technical work was justifying the safety of the disposal option.

One of the major accomplishments of the project was the completion of the Exploratory Studies Facility which started in late 1992 and was completed in 1997. The large tunnel and side drifts provided scientists and engineers to conduct studies of the hydrology and geology of the mountain to support the design and licensing application for construction of the repository (see Fig. 8).



Fig. 8 Photograph of the tunnel boring machine in Yucca Mountain. Source: <https://www.energy.gov/sites/prod/files/edg/media/SER.PDF>

The studies leading up to the license application for the Yucca Mountain High Level Waste Repository and the review by the Nuclear Regulatory Commission of this application form the basis for stating that high level nuclear waste can be safely disposed. The basic design of the Yucca Mountain repository is based the National Academy of Science's 1957 finding that the best means for long term disposal of HLW is in geological media deep underground. Based on this fundamental finding, Yucca Mountain was chosen as the primary site in igneous rock 2000 ft underground in an engineered series of tunnels designed to deal with events including water ingress, seismic, climate changes, volcanic disruptions and human intrusion events that may occur for the licensed period of 10,000 years. Analyses were performed even out to 1 million years to check for any outlier events or processes that might compromise the geological isolation of the nuclear waste. The design calls for manufacturing spent fuel canisters in specially designed casks. Although the basic concept relies on the fundamental geology to contain the radioactive materials, the Department of Energy chose to add additional engineered barriers to prevent water from attacking the waste packages and ultimately reaching the used fuel pellets. This system was called the Engineered Barrier System (EBS) to provide for a defense in depth approach to system safety. The EBS mitigates any possible corrosion of the waste packages which contain the used fuel under a multitude of possible events that could disrupt the repository.

The basic layout of the Yucca Mountain repository is shown on Fig. 9 (Department of Energy, 1998). As can be seen, the design calls for a series of 100 placement tunnels about 73 miles in total length 2000 ft underground. Each tunnel would hold about 100 waste packages containing the used fuel in specially fabricated casks designed to provide radiation shielding and corrosion protection. One such series of tunnels would provide disposal of 70,000³ tons of used fuel from operating reactors. This value could be expanded since there is adequate additional space available to store all of the nation's used fuel for the life of all operating nuclear plants.

The studies by the Sandia National Laboratory, the lead national laboratory for the Department of Energy, and other national laboratories including the US Geological Survey culminated in the submission of a license application for the construction of the repository in 2008. The NRC reviewed the license application containing the results of numerous studies and analyses showing that the repository could be safely built and operated. Opponents of the repository filed numerous contentions requiring resolution by the Atomic and Licensing Board after public hearings. Unfortunately, due to the action of Senator Reid of Nevada, these hearings

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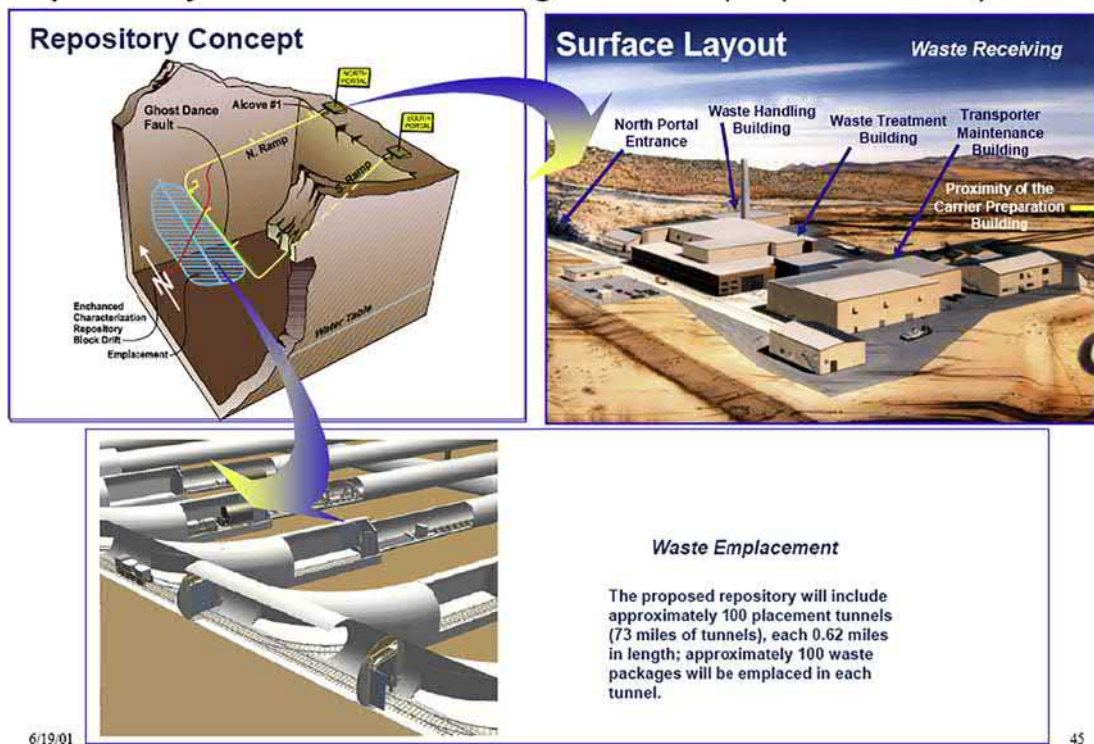


Fig. 9 Conceptual design of the Yucca Mountain repository. Source: https://www.energy.gov/sites/prod/files/Yucca_Viability_01_Overview.pdf

³The 70,000 tons is a law limit at present.

were never held since under his leadership Congress withheld monies for this review which have yet to be restored continuing the political stalemate on siting a repository.

The safety case for the Yucca Mountain high level waste repository

A summary of the findings of the NRC review is presented below showing that the repository could be safely operated for the 10,000 year below the post closure individual protection standard as defined in 10 CFR 63.311 (Nuclear Regulatory Commission, 2009) according to the analysis. These results are documented by the “Total System Performance Assessment Model and Analysis for the License Application” (Sandia National Laboratory, 2008; Department of Energy, 2002). The key challenge for the design of an underground nuclear waste repository is demonstrating that over the period of 10,000 years that the radioactive materials contained in the fuel, the canisters, and metal overpack casks will not reach the habitable environment over that period. The analysis focuses on possible mechanisms that can deteriorate the protective canisters and eventually degrade the metallic fuel pins allowing the fission products (so called waste) to somehow migrate to the surface through water pathways that are deep underground. Thus, site selection is very important in that sites should be selected that have minimal rainfall such as the Nevada desert.⁴ Additionally, sites should also have minimal seismic or volcanic activity that could create direct pathways to the environment. These are likely reasons that Congress selected the Yucca Mountain site among numerous alternatives.

One of the first major studies completed prior to continuing with a formal license application to construct the repository at Yucca Mountain in the Nevada desert was a viability assessment prepared by the nation’s national laboratories. This viability assessment was the first study that preliminarily demonstrated that subject to more detailed studies, Yucca Mountain was acceptable for further study and development.

This viability assessment modeled all possible mechanisms that could lead to the migration of the radionuclides into the environment as shown on the figure below. Since water is a very important parameter affecting possible degradation of the waste packages and migration into habitable environment, modeling climate, infiltration, water transport, package degradation, and movement of any dissolved waste products into the human biosphere as illustrated in Fig. 10.

As can be seen, the analysis attempts to consider all possible mechanisms for container failure due water pathways which may cause corrosion and eventual package failure. Once the package fails, analyses are also conducted considering the chemical makeup of the used fuel by radioisotope to understand how those isotopes might migrate through water pathways to the habitable environment post repository closure thousands of years in the future. While the geological barrier is the primary barrier to radioactive release to the habitable environment, as mentioned earlier the “Engineered Barrier System” (EBS) provides multiple barriers as the first line of defense for the natural geological barrier.

Yucca Mountain is designed not only to store commercial used nuclear fuel from many different types of reactors, but also used fuel and nuclear waste from the US nuclear weapons production facilities. Shown on Fig. 11 is an example of the types of waste packages being proposed for disposal.

Modeling these mechanisms for such long durations required developing sophisticated analyses of possible scenarios and likelihood of water migration events (Meecham et al., 2011). This led to the development of probabilistic risk analysis over time periods that span a million years. Due to the uncertainty of such predictions the probabilistic methods are best suited for such analysis. These analyses model degradation mechanisms assuming many different climatic scenarios which can cause possible water intrusion events for thousands of years including uncertainties in data and modeling assumptions.

The models assume that overtime, water will enter the drift, fall on the drip shield eventually failing it thousands of years later. Eventually the water will attack the actual waste canister made of a special alloy material as shown on Fig. 11 after which it will attack the spent fuel pins shown earlier. This process, should it happen as modeled, takes many thousands of years. The EBS is designed to prolong the degradation process. Since water is the primary means of degradation and the pathway to the environment, knowledge of the chemistry of the materials and reactions with water and the isotopic mobility need to be understood and modeled. This understanding, including the underground hydrology, is necessary to predict the pathway to the habitable environment which is the primary concern. Recall that the geologic barrier is the prime barrier to the prevention of radiological releases to the environment.

The Total System Performance Analysis (Sandia National Laboratory, 2008; Department of Energy, 2008) documents the results of thousands of calculations to predict possible releases from the repository over time after repository closure. Each line shows a summary of the collective probabilistic failure analyses under many scenarios postulated to occur including uncertainties of the assumptions used in the analyses. The results are presented as the likelihood of a public radiation exposure over the time period analyzed. Shown on Fig. 12 are the results of the analyses for the 10,000 year licensed period. These results show that expected annual dose to the nearest person is less than 1 mrem per year given all the possible failure modes and uncertainties in modeling. For reference, natural background radiation averages over 250 mrem per year.

Sandia National Laboratory also carried out the analyses to 1 million years to see if there are any radiological outliers. This analysis is summarized in Fig. 13. These results show that the annual expected dose to the nearest individual is less than 10 mrem per year out to 1 million years. Carrying out the analysis to 1 million years, while not required for licensing, shows that even out that long term horizon, the repository and still meet the individual protection standard under many challenging scenarios.

⁴Yucca Mountain receives about 7.5 in. of rainfall per year of which 95% is either evaporated or is taken in by vegetation.

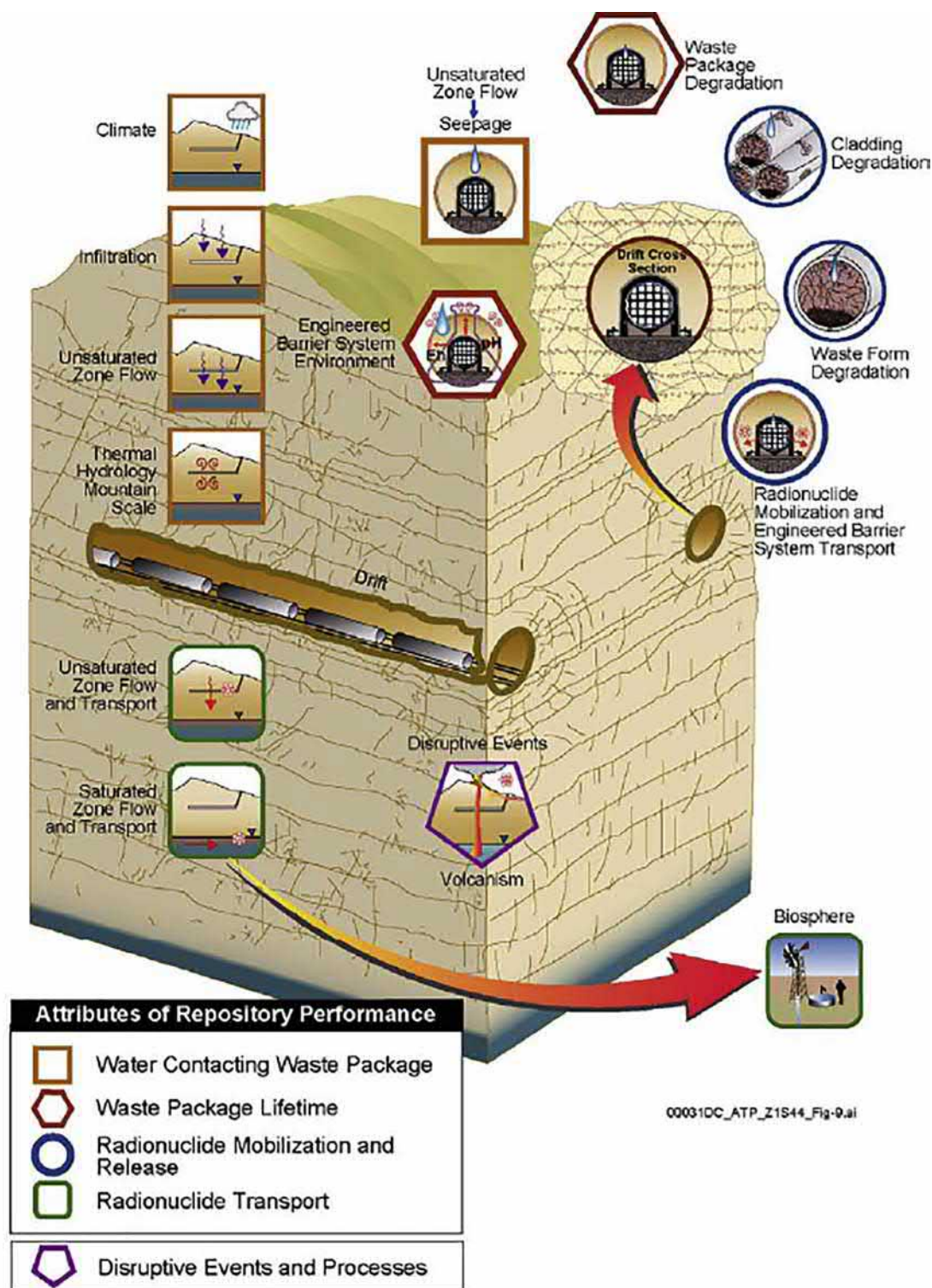


Fig. 10 Considerations Incorporated in the total system performance model. Source: <https://prod-ng.sandia.gov/techlib-noauth/access-control.cgi/2011/118270.pdf>

While analyses over such long times are suspect, the detailed treatment of models, assumptions, input data and uncertainties associated with them provide confidence that the repository will be safe for hundreds of thousands of years. Shown on this plot is the Individual Protection Standard as defined in (Nuclear Regulatory Commission, 2009) which is the NRC requirement for the repository for this time period.

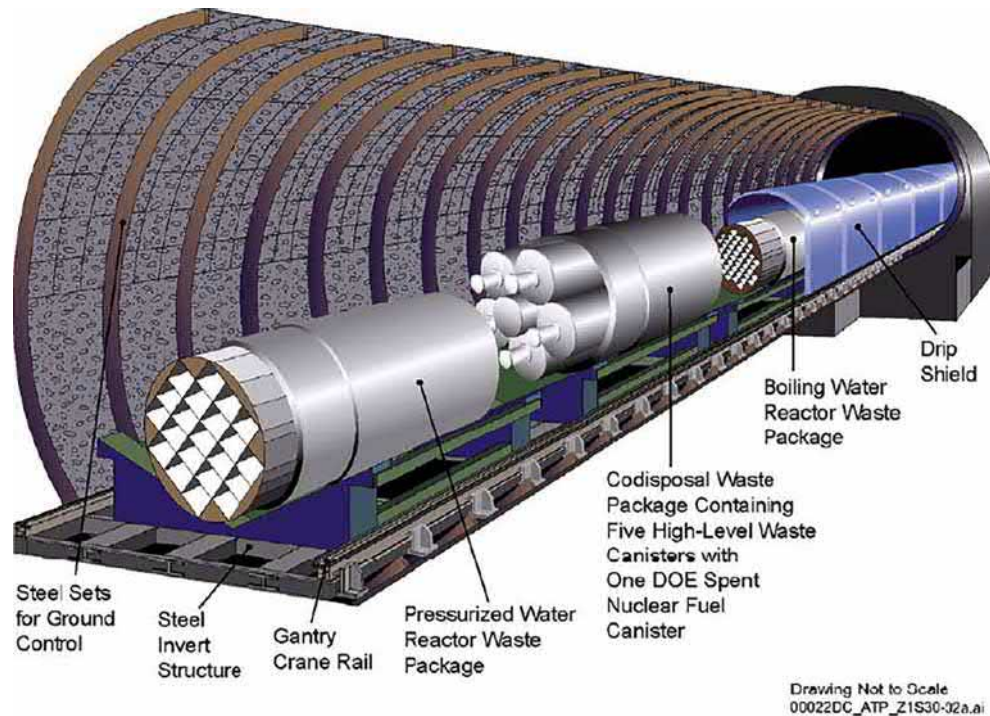


Fig. 11 Schematic illustration of the emplacement drift with cutaway views of different waste packages. Source: <https://prod-ng.sandia.gov/techlib-noauth/access-control.cgi/2011/118270.pdf>

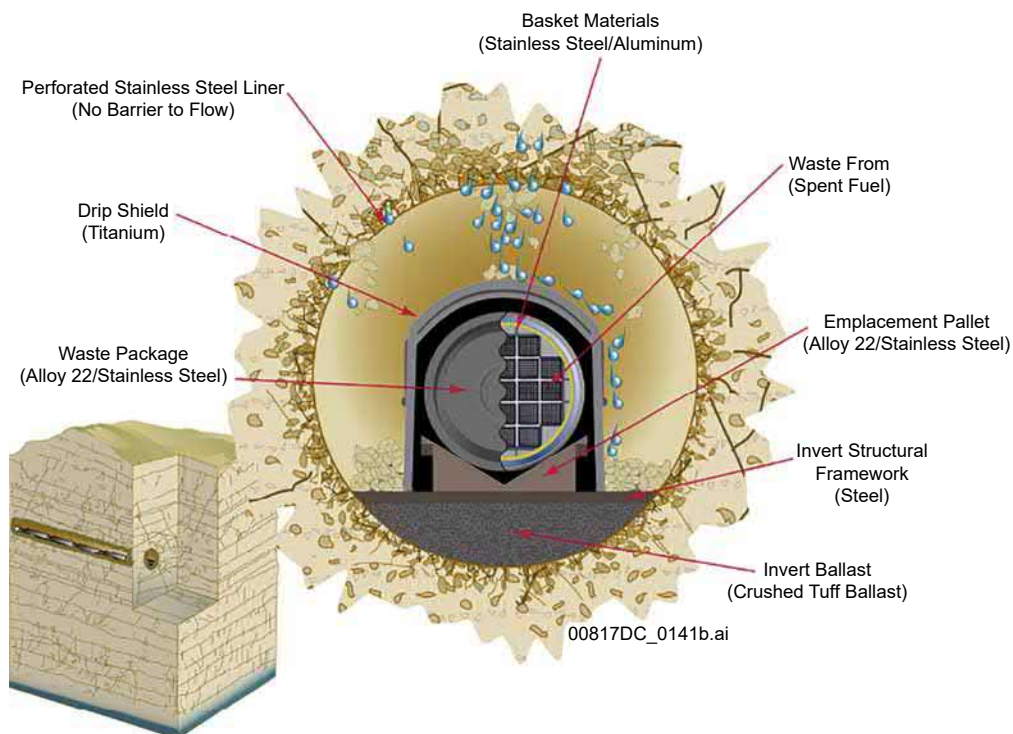


Fig. 12 Illustrative cross section of a typical emplacement drift. Source: MD-WIS-PA-000005 Rev 0,AD 01.

It should be noted that during the loading process and subsequent monitoring period prior to closure, the nuclear waste casks are meant to be retrievable should monitoring results not support expected behavior (Fig. 14) (Sandia National Laboratories, 2008).

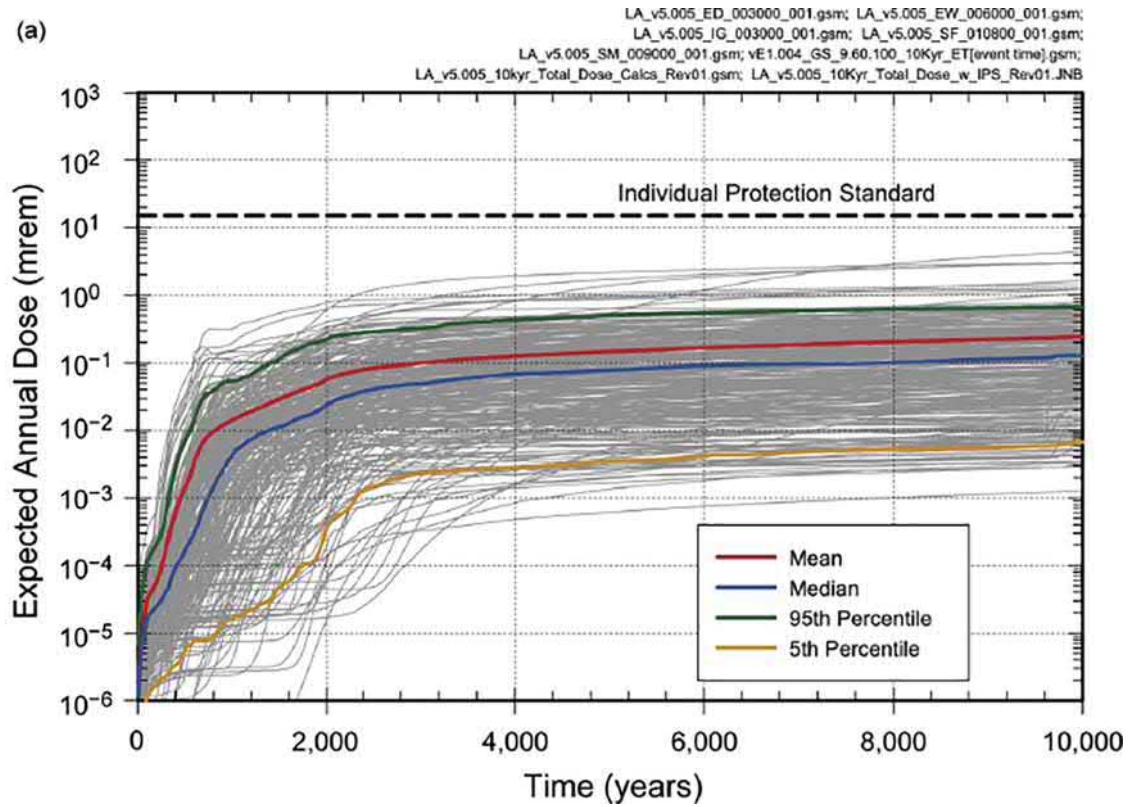


Fig. 13 Expected annual dose to the nearest person for 10,000 years. Source: <https://www.nrc.gov/docs/ML0907/ML090770579.pdf>

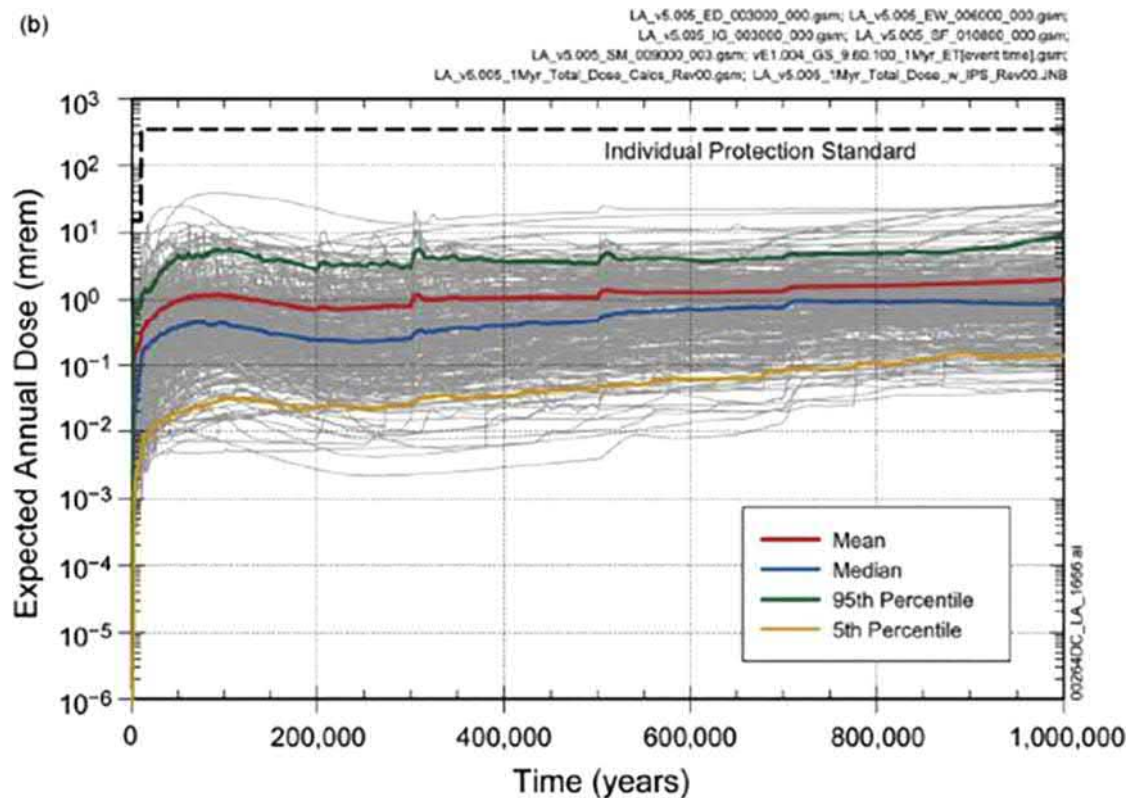


Fig. 14 TSPA-LA results for total expected dose over 1 million years. Source: <https://www.nrc.gov/docs/ML0907/ML090770579.pdf>

A review of the technical findings of the DOE license application by the NRC indicated that they believed that the Yucca Mountain disposal site was acceptable for construction. Anti-nuclear organizations and the state of Nevada filed contentions to be addressed by the NRC Atomic and Safety Licensing Board prior for permission to be granted for construction. Due to the withdrawal of funding by Congress, these hearings were not held leaving the entire question of waste disposal unresolved despite the federal law still requires that Yucca Mountain be the site unless it is found unacceptable for safety reasons.

Gridlock on a solution

Today, there continues to be gridlock on siting repository upon Obama's cancellation of the Yucca Mountain project. The DOE has chosen a "volunteer consensus" siting approach to try to restart the entire process without much success. This decision was based on the recommendation of a "Blue Ribbon" commission. Their report, "Blue Ribbon Commission on America's Nuclear Future, Report to the Secretary of Energy" ([Blue Ribbon Commission, 2012](#)) outlined several actions that need to be taken one of which was volunteer siting. It should be noted that consideration of the Yucca Mountain site was not permitted in this study. To date, no volunteers have come forward for obvious political reason resulting in a situation where the used fuel is currently being stored at existing nuclear plant sites in dry storage or in existing used fuel storage pools under water. To alleviate the problem of removing used fuel from decommissioned nuclear sites and avoid building more dry cask storage facilities at existing nuclear power plant sites, private efforts are now underway in New Mexico and Texas to site consolidated above ground dry cask interim storage facilities which are now going through the licensing process ([Nuclear Regulatory Commission, 2018](#)).

This situation is particularly embarrassing and costly to the taxpayer since taxpayers are paying utilities for storage of the used fuel at the reactor site since under contract, the DOE was obligated to begin taking used fuel by 1998. Thus far, taxpayers have paid US utilities over \$7 Billion which is expected to grow to \$27 Billion assuming that the government takes the spent fuel by 2021 (an unlikely event given progress to date on locating even a temporary storage location for the United States). Industry estimates for the government liability double that to over \$ 50 Billion taxpayer dollars ([Korsnick, 2019](#)).

International status

[Metlay \(2020\)](#) discusses in some detail progress that is being made in other countries in their efforts to dispose of their nuclear waste from their operating plants. The two leading nations in this effort are Finland and Sweden while progress is being made in France, Germany and England. Finland and Sweden have made significant progress. The sites selected were near existing nuclear plants in both countries with public support from local residents and the national government. Each country recognized that they bore the responsibility for disposal of the nuclear waste and assigned private companies (not the government) for managing and disposing the used fuel. The major differences in terms of success between the Finnish and Swedish programs and the United States are that the sites chosen were near nuclear plants so that the residents were familiar with nuclear energy and that the federal government was not responsible for disposal. This decision left national politics out of the picture which has stymied US nuclear waste disposal efforts. Both Sweden and Finland use the same technology and plan to dispose of the spent fuel in a granite geological formation. Since Finland is preparing to open a repository in 2021, it will be briefly described below.

Finland's high level waste repository

Finland produces about 30% of its electricity from nuclear power using 4 reactors. Another nuclear plant is under construction that will increase Finland's nuclear share of electricity production to 60%. The two utilities (Fortum and TVO) operating reactors formed a special company, Posiva, to design, construct and operate the proposed repository ([Posiva, 2019](#)). In 1999 Posiva proposed to the government that the Olkiluoto site be used for what they refer to as a "(very) long-term underground storage facility." The local municipality, Eurajoki, approved the selection which was approved by the national government in 2001. The official name for the repository is "Onkalo." An excavation permit was issued by the municipality after years of site specific study and engineering analysis in 2004. A construction license approved by 2015 with first loading of the nuclear waste is expected by 2023.

The Finnish and Swedish designs call for encapsulation of the used fuel in copper disposal packages to be emplaced in underground granite deposits that have been stable for 9 billion years in a series of tunnels backfilled with an impermeable barrier of bentonite clay. Shown in [Fig. 15](#) are the copper canisters into which the spent fuel will be placed.

The basic repository design calls for a series of underground tunnels cut in granite bedrock as shown in [Fig. 16](#).

The plan is to load the 12 used fuel assemblies in boron steel canister baskets and sealing it in the copper cask. Each cask will be disposed of in its own hole after which it is overpacked with bentonite clay which is designed to keep water from the copper cask. An actual picture of the below ground repository is shown in [Fig. 17](#).

Posiva's license application and safety case was approved by the Finnish regulator STUK in 2015 ([World Nuclear News, 2015](#)). The regulator concluded after the safety review that the repository "can be built to be safe." As opposed to the United States approach where everything needs to be conclusively addressed in the application, the Finnish government has recognized that the process requires a phased approach to licensing in that as more information is learned by the excavation and placement of the waste packages, requirements will be modified as necessary. The safety standards applied by STUK are similar to those applied to nuclear power plants in terms of public health impact. The safety analysis of the repository identifies possible means whereby any of the radioactive materials contained in the waste package should not result in a public exposure no higher than 0.1 millisievert (10 millirem) per year.



Fig. 15 Finnish and Swedish canister design. Source: Posiva.

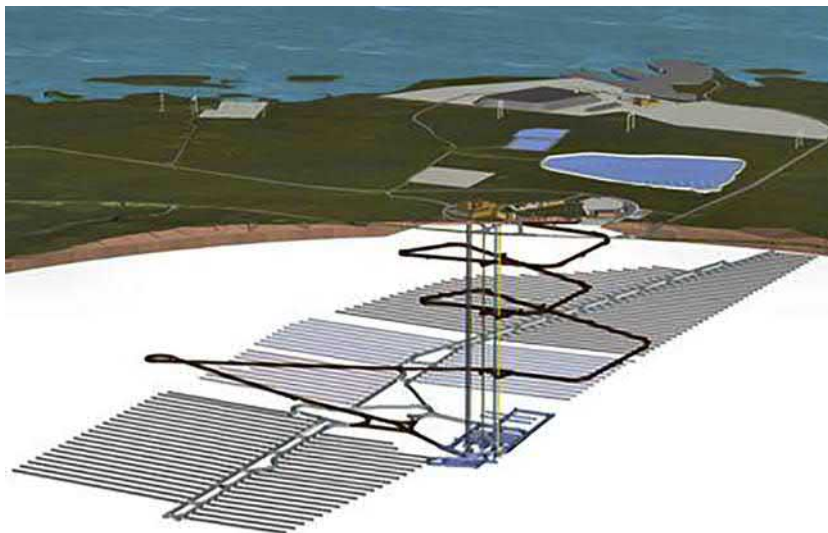


Fig. 16 Conceptual design of Finland's nuclear waste repository. Source: http://www.posiva.fi/en/media/image_gallery?gfid_2061=92#gallery_2061.



Fig. 17 Onkolo repository shaft for disposal of copper casks. Source: http://www.posiva.fi/en/media/image_gallery?gfid_2061=92#gallery_2061.

For reference, the average Finnish person (Posiva Safety Case, 2012) is naturally exposed to about 320 millirem per year. Posiva's analysis shows the projected releases to be 10,000 times lower (Posiva Safety Case, 2012).

As in the US case, Finland's and Sweden rely on the same conceptual engineered barrier approach to geological disposal with the geology being the primary barrier to release. The reference time period for the analysis is 250,000 years which should encompass the next ice age (STUX Finnish Regulator, 2016). Although the Finnish approach is considered more deterministic, the analysis does include a risk assessment based on the uncertainties and probability considerations. The expectation is that after 250,000 years, the level of radioactivity remaining is about that of a large uranium deposit.

The regulator and Posiva accept the fact that it is not possible to consider every event that may occur for the 250,000 years but there is high confidence that the multi-barrier and phased approach will be safe for people and the environment. The Onkalo repository is expected to operate for about 100 years after which the access tunnel will be backfilled and sealed.

The estimated cost of the entire Finland's nuclear waste disposal program which is funded by the utilities is about \$ 910 million. The State Nuclear Waste Management Fund which collects the money from the utilities for the nuclear energy generated has about \$1.6 Billion. This is in sharp contrast to the US where the DOE has spent more than \$ 17 billion with not much to show for it.

Public acceptance

Metlay (2020) summarizes the difficulty in public acceptance in many nations including the United States. His conclusions are that there are five reasons for difficulty in siting. They include contested technical claims, public perception of risk and stigma associated with nuclear energy, mistrust, inequitable processes and decision making, and lastly opposition to nuclear power. In the United States, as mentioned previously, politics plays a major role especially in presidential election years.

Summary and conclusions

Nuclear waste can be disposed of safely. While natural analogs provide the best examples of billions of years of confinement in geological media, the best examples found in the United States are the Waste Isolation Pilot Plant in New Mexico, the safety and licensing submittals to the NRC for the Yucca Mountain site, and the Swedish and Finnish licensing applications for an underground repositories. While the US Yucca Mountain Project licensing process was canceled for political reasons, a technical basis exists demonstrating that for 1 million years, far beyond the 10,000 year requirement, that used nuclear fuel can be safely disposed of in an engineered geologic repository. Construction is underway in Finland for a geological repository after safety reviews and the Swedish application is currently undergoing licensing reviews. Siting discussions are underway in France, Switzerland, and Germany.

The extensive US application features a total system performance assessment using probabilistic risk assessment techniques to address potential modeling and input uncertainties for geologic time frames demonstrating that the potential dose to the nearest member of the public over these long times show minimal possible exposures of less than 1–10 millirem per year for the nearest neighbor compared to an annual natural radiation exposure from all sources of 245 millirem for each person for at least 10,000 years. Thus, the safety of the repository can be safely assumed to be acceptable given these results.

See Also: Social Acceptability of Geologic Disposal.

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International Nuclear Waste Disposal Centers

Ian Hore-Lacy, World Nuclear Association, Blackburn North, VIC, Australia

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Legal and ethical context

At present there is clear and unequivocal understanding that each country is ethically and legally responsible for its own wastes, especially nuclear wastes, therefore the default position is that all wastes will be disposed of in each of the 50 or so countries concerned. McCombie and Chapman (2002) discuss concerns that discussions on shared facilities should not derail national programs then underway. They caution that it is equally important that future development of regional and international projects is not unnecessarily blocked by short-term national considerations. The International Atomic Energy Agency soon published a technical document on the subject (IAEA, 2004).

The main volume of high-level nuclear waste is created in the nuclear power reactors which make electricity in over 30 countries. The elements within the used fuel which make it “waste” are essentially not left-overs from imported uranium—they are created in the operating reactors (by fission and neutron capture). There is thus no moral obligation on uranium suppliers in respect to the wastes, other than that involved in safeguards procedures.

Those International Atomic Energy Agency (IAEA) and bilateral national safeguards procedures track used fuel and its constituents, so any international waste repository has implications under the Nuclear Non-Proliferation Treaty (NPT). The trustworthiness and standing of the host country is fundamental to the project’s acceptability to NPT states, which comprise virtually every country but India, Pakistan, Israel and North Korea.

Also, the international treaty produced by IAEA and signed by most nations of the world in 1997 covering the management and disposal of used fuel and high-level wastes requires that the host facility or system meets the highest national and international standards.

There are over 20 countries with no nuclear power that have need for secure disposal of long-lived radioactive wastes from their research reactors, or from nuclear medicine or other technology.

International repository concepts

In November 2003, Dr. Mohamed ElBaradei, Director-General of the UN’s International Atomic Energy Agency (IAEA), said to the UN General Assembly: “We should ... consider multinational approaches to the management and disposal of spent fuel and radioactive waste. Over 50 countries currently have spent fuel stored in temporary locations, awaiting reprocessing or disposal. Not all countries have the appropriate geological conditions for such disposal—and, for many countries with small nuclear programs, the financial and human resources required for the construction and operation of a geological disposal facility are daunting.”

In a later article he specifically included research reactors in the scope of this suggestion and concluded that “considerable advantages—in cost, safety, security and non-proliferation—would be gained from international co-operation in these stages of the nuclear fuel cycle.” These principles are expressed in IAEA (2004) and an expert group report to him followed up on this (IAEA, 2005).

Nearly 25 years earlier, in 1980, the IAEA-sponsored International Nuclear Fuel Cycle Evaluation (INFCE) waste management and disposal report had firmly recommended that proposals “for establishing multinational and international repositories should be elaborated” due to their non-proliferation advantages. “Centralised facilities for disposal of spent fuel and/or vitrified high-level wastes ... would reduce the diversion risk” and be more economical.

Individual waste repositories for spent nuclear fuel and other high-level wastes need to be reliably secure. Achieving high security means:

- They can make a vital contribution to global environmental *safety* by ensuring that radioactive substances are permanently removed from the human environment,
- They can greatly enhance global *security* in the broader sense by preventing malicious use of fissile and radiological materials.

Insofar as these functions are less than fully assured in any of the 50 countries concerned with radioactive wastes, there is justification for some kind of international collaboration and facilities, possibly on a regional basis. In particular, the second point is arguably best achieved by international collaboration under IAEA auspices. McCombie and Chapman (2004) present arguments that the international community should take concrete steps to expedite credible proposals for shared repositories.

In 2004 the IAEA (IAEA, 2004) put forward three concepts for multinational repositories:

- Incremental addition to a large national program
- A supra-national facility with international management and control
- Collaborative partnering among countries for a multinational repository.

While most countries should be able to find suitably safe sites in stable geological formations, demonstrating this safety so as to create public confidence is best achieved where there is simple geology.

Certainly, there is wide consensus that geological disposal is the only way of ensuring adequate safety and security in the long-term management of used fuel which is treated as waste, and also of separated high-level radioactive wastes which, if recycling of actinides becomes established, eventually will be mainly fission products (Greeneche, 2021; Wigeland and Taiwo, 2021).

While acknowledging each country's responsibility for its own wastes, the limits to the logic on indigenous disposal can be seen from the changing national borders within Europe over the last century. For Slovenia for instance (which has one nuclear power reactor), its capital city Ljubljana has politically lain within seven different states in the last hundred years or so.

The South Australian Royal Commission into the Nuclear Fuel Cycle (2016) report notes that:

There are international models that address the transfer of waste between countries. The Basel Convention, which applies to hazardous wastes other than radioactive waste, imposes requirements upon the transfer of hazardous wastes between countries; namely the transfer shall only take place where prior informed consent has been received and only if the transfer represents an environmentally sound solution. Hazardous wastes are commercially transferred under this regime. While the Joint Convention applies equivalent requirements to transfers of radioactive waste between countries, there are no operating models for the commercial transfer of used fuel for disposal.

Various organisations have looked into potential concepts. There are, however, commercial models for the transfer of used fuel between countries for reprocessing, as well as the take-back of fuel from reactors built by Rosatom, the Russian state nuclear corporation. The United Kingdom has reprocessed used fuel for many countries but does not accept the waste products for disposal. In all cases, transfers can only take place if the recipient country has the capacity to manage the waste safely and where such transfer has been agreed between the countries concerned.

In October 2016 the *reliable nuclear fuel services* working group of the International Framework for Nuclear Energy Cooperation (IFNEC, 2016) released a position paper: *Overview of the Dual Track Approach for National Back-end Programs*, focused on the needs of countries with small nuclear programs. It was based on previous work initiated by Arius as well as studies done by IAEA and NEA. It notes that under the IAEA Joint Convention on wastes and spent fuel, contracting parties have agreed that the country that discharges the spent fuel and receives the benefits of the power generated bears the responsibility for its management, including disposal. Since disposal is a relatively major burden with small quantities, due to high fixed costs, some international collaboration is suggested. This is the dual track beyond simply national approaches, allowing a country to offer repository facilities as a service provider.

This was fleshed out in an IFNEC (2019) presentation to the IAEA in mid-2019: *Development of Multinational Repository Concept: Exploring Alternative Approaches to Financing Multinational Repository*. This reported on an IFNEC workshop held in Paris in 2018 which considered the spending profile on a repository from siting and licensing, through construction (35% of total), operation (45% of total) and closure. It concluded however that multinational repository development was not yet timely, though the concept had potential and should continue to be studied. Countries with small nuclear programs should set aside funds in anticipation. IFNEC should continue to work on linking any potential service provider with potential customer countries.

South Australian proposal

In May 2016 the South Australian Royal Commission into the Nuclear Fuel Cycle reported. A major recommendation was that a facility for the disposal of international used nuclear fuel and intermediate-level waste should be established in South Australia. It found that the state "has the necessary attributes and capabilities to develop a world-class waste disposal facility, and to do so

safely” as a service provider. Based on a “cautious and conservative approach,” from assessments of used fuel inventories and potential global interest the commission determined that such a facility could generate more than A\$ 100 billion in income in excess of expenditure (including a reserve fund of A\$ 32 billion for facility closure and ongoing monitoring) over the 120-year life of the project.

The World Nuclear Association said that the report had “fundamentally changed the nature of the global nuclear waste discourse,” and a multi-national waste facility based in South Australia would provide a welcome option for many countries operating nuclear facilities today. It would be a “viable alternative” to national projects. Such a large multi-national waste storage facility and repository would be a world-first and should offer advantages in terms of siting and economics when compared with smaller national approaches.

The [South Australian Royal Commission into the Nuclear Fuel Cycle \(2016\)](#) report continues:

Under the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management, any proposal to store and dispose of used fuel in South Australia would require agreement between the countries concerned. In Australia, treaty level agreements would need to be developed between the federal government and the relevant overseas government. An agreement would also need to specify arrangements between the Australian Government and the Government of South Australia, to ensure these commitments were fulfilled. Further agreements may be required with third party countries: for example, if they have supplied uranium to the country wishing to store and dispose of used fuel in South Australia.

As Australia is a net exporter of energy, it has a significant role to play in assisting other countries to lower their carbon emissions. This includes countries with less opportunity for large scale renewable energy deployment than Australia, for whom nuclear power makes a substantial contribution to their production of low-carbon energy. For new nuclear entrants or countries with little prospect of siting their own used fuel disposal facilities, an international solution would remove a significant impediment to the new or ongoing use of nuclear power as a low carbon technology. As a result, Australia would derive a reputational and financial benefit by hosting a facility for the disposal of international used fuel.

Excluding countries which are committed to developing national solutions or already have structured programs leading to a domestic facility, and those with laws prohibiting export of wastes, the report identifies 90,000 tonnes of used fuel and almost 270,000 cubic metres of intermediate-levels wastes as a potentially accessible market and available for an Australian repository today. These amounts, comprising about one quarter of actual world used fuel and ILW, are increasing annually. By 2090 the amounts are 276,500 tonnes and 782,000 m³ respectively. The viability analysis is based on half these amounts.

Based on detailed analysis, the Royal Commission considers that a reasonable baseline price for the purpose of assessing viability would be A\$ 1.75 million/tHM for used fuel. This is based on a reasonable baseline “willingness to pay” estimate of A\$ 1.95 million/tHM, less A\$ 0.2 million/tHM to account for costs incurred by customers in preparing and delivering the waste to South Australia.

Regarding intermediate-level wastes, the [Royal Commission into the Nuclear Fuel Cycle \(2016\)](#) report says:

The management and disposal of intermediate level waste commands a far lower willingness to pay than for used fuel since it is more readily managed and stored.

However, a 2011 report from the UK Department of Energy and Climate Change has suggested that £25 900 per m³ (in current terms, A\$66 000 per m³) represents a levy that ought be imposed on nuclear power plant operators to reflect current costs and the potential for future increases. In the interests of conservatism, and to address the costs of packaging and transport (which are not as well defined as for used fuel) a price to charge of A\$40 000 per m³ is considered appropriate for the purposes of a viability analysis.

Interim above-ground storage for used fuel inside heavily engineered, purpose-built casks would be followed by deep geological disposal, the repository design based on that under construction at Olkiluoto in Finland at 400–450 metres depth. In the analysis, the geological disposal facility for used fuel is notionally co-located with an intermediate level waste facility, where those packages are placed in medium-depth vaults of 50–250 metres. The report said that integrated facilities with capacity to store and dispose of used fuel would be viable.

On a number of realistic scenarios, such a facility would be highly profitable. The timeline for establishing an interim storage facility and associated transport infrastructure, including harbor, port and railway would be 11 years after project commencement. Transferring used fuel and ILW from surface storage to underground repository would begin at 28 years. Revenue would thus commence at year 11, based on 3000 t/year used fuel, so \$5.25 billion/year over 30 years from then simply for the fuel. Total capital cost would be \$41 billion over some years including 25% contingency, and operating expenses from year 11 about A\$ 0.878 million (\$ 2015). Overall total revenue (in undiscounted terms) would be more than A\$ 257 billion, with total costs of A\$ 145 billion. The project would add 4.7% to gross state product and 1.9% to employment by 2030.

ARIUS and Europe—ERDO

Early in 2002 a new, non-commercial body to promote the concept of regional and international facilities for storage and disposal of all types of long-lived nuclear wastes was set up. This is ARIUS—the Association for Regional and International Underground Storage. A key objective is to explore ways of providing shared radioactive waste management approaches and facilities, in particular storage and disposal facilities for smaller users. Membership is open and comprises countries with small nuclear programs as well as industrial organisations with relevant interests. Arius is a successor to the Pangea proposal (see below).

Arius focused initially on Europe, and the feasibility of regional repositories there. In 2002 a European Commission Directive said that geological disposal of radioactive wastes was preferred and that “A regional approach, involving two or more countries, could also offer advantages especially to countries that have no or limited nuclear programs, insofar as it would provide a safe and less costly solution for all parties.”

In mid-2003 Arius initiated the SAPIERR Pilot Project for European Regional Repositories, which obtained European Commission approval. This was undertaken over 2 years to 2005 to help the EC grapple with the regional repository issue as flagged earlier in the EC Radioactive Waste Directive. It allowed potential options for regional collaboration and for regional repositories to be identified, though it did not extend to site identification. Slovakia provided the project coordination.

Following this pilot study, in September 2006 a new EC-funded SAPIERR (Strategic Action Plan for Implementation of European Regional Repositories) project to assess the feasibility of European regional waste repositories was commenced, indicating a recognition in the EU that implementing 25 national repositories is not optimal economically or for safety and security. The project was in line with proposals from the International Atomic Energy Agency (IAEA), Russia and the United States (with GNEP, now IFNEC) for multilateral cooperation in the fuel cycle in order to enhance global security. Shared repositories for high-level nuclear wastes are an important element of this.

The SAPIERR project held its final symposium January 2009 in Brussels. The results of studies on the viability of shared, regional European geological repositories were presented to 50 participants from 21 countries. The aspects considered included organisational and legal issues, economic impacts, safety and security considerations, and public and political attitudes to multinational repositories (McCombie, 2009).

The main outcome of this was that 14 countries¹ resolved to move towards setting up a European Repository Development Organisation (ERDO). The first step in the strategy was the establishment of a self-financing Working Group (ERDO-WG) of interested countries in 2009 to prepare a consensus model to be agreed for ERDO, using the SAPIERR findings as a starting point (see www.erdo-wg.eu). It proposes a sister organisation to waste agencies from European countries that have opted for a purely national repository program. This model was presented to potentially interested countries, and internal discussions are still in progress in several of these on adopting the multinational solution as one option when responding to the requirements placed on EU Member States by the EC Waste Directive of 2011.² Some member states have already decided to follow this course. The ERDO-WG secretariat is provided by ARIUS, Switzerland, and the administration by the Netherlands waste agency, COVRA (<https://covra.nl/en/disposal/international>).

ARIUS: The Persian Gulf and SE Asia

Alongside ERDO, Arius is evaluating whether similar, regional shared solutions would be appropriate for and of interest to emerging nuclear power programs in the Gulf, Middle East and North Africa (MENA) region and also SE Asia. The overall aim is to assess the interest within each region of working towards Regional Repository Development Organizations (RDOs) similar to the Europe’s ERDO. A scoping study was completed in 2011 and funding for a further two-year project was awarded in mid-2011 by two US Foundations. In 2011 the UAE’s permanent representative to the IAEA raised the possibility of a regional repository and said that “Among GCC countries there is potential for a lot of co-operation in this area, including waste management and waste disposal.” A final project report from the scoping study was in February 2016.

¹Austria, Bulgaria, Czech Republic, Denmark, Estonia, Ireland, Italy, Latvia, Lithuania, Netherlands, Poland, Romania, Slovakia, and Slovenia, though some subsequently did not continue in the ongoing ERDO-WG. In April 2020 the WG active membership comprised Austria, Croatia, Denmark, Italy, Netherlands, Norway, Poland and Slovenia.

²At the end of 2011, the ERDO Working Group, led by representatives of the Dutch and Slovakian national programs, prepared a set of documents outlining the possible structure, operating methods and financing options for a formal, multinational, European waste management agency (the ERDO). These documents were submitted to all EU governments that might be interested in actively participating in the preparation of this next step, and also provided as information to those European countries, such as Sweden, Finland and France, that have decided on purely national approaches to managing their radioactive wastes. Discussions with potential participants have been underway since 2012, and a number of positive responses received. In December 2013 the EC, the European Nuclear Energy Forum (ENEF) Waste Group and the ERDO-WG organised a meeting in Luxembourg to consider how to help all EU member states meet the requirements set by the EC Waste Directive of 2011. The meeting addressed the fact that some EU member states are not well equipped to meet Waste Directive targets, but could develop a common program for addressing many of its requirements. This led in June 2014 to a presentation to the European Parliament Working Party on Atomic Questions on the COMS-WD (Cooperation between Member States Responding to the Waste Directive) proposal. http://www.erdo-wg.eu/Documents_files/COMS-WD.pdf. In September 2014 Arius together with European partners submitted the COMS-WD proposal to the €80 billion Horizon 2020 EU research and innovation program. Nine EU countries (Austria, Croatia, Denmark, Greece, Italy, Netherlands, Poland, Portugal, and Slovenia) and the Joint Research Centre Institute for Transuranium Elements (JRC-ITU) were involved with Arius in this. JRC-ITU involvement is to ensure continued cooperation with other EU projects. However, the proposal was rejected as being “out of scope” for Horizon 2020. Nevertheless, the multinational or dual-track option is being included by some countries in their strategies responding to the Waste Directive—Netherlands and Slovenia so far.

In April 2012, an initial meeting took place in Abu Dhabi, hosted by the Federal Authority for Nuclear Regulation (FANR) of the United Arab Emirates (UAE) and supported by the IAEA. Around 35 participants attended, primarily from UAE waste management and planning organisations. UAE was an obvious host for a first meeting since it has the most advanced nuclear power program in the region and has also formally committed to a “dual track” radioactive waste management strategy that involves developing a national storage and disposal program in parallel with exploring regional cooperation options. A follow up to the Abu Dhabi meeting was in November 2012 in Tunisia, hosted by the Arab Atomic Energy Agency (AAEA), with a widened group of participating MENA countries. In the six countries comprising the Gulf Cooperation Council region, which includes two nations with expanding nuclear power programs (the UAE and Saudi Arabia), consideration is also being given to launching a joint project on the feasibility of shared storage and/or disposal facilities. Further discussion has followed, including at a January 2015 workshop in Abu Dhabi.

Arius estimated that the cost of such a shared regional repository in the Middle East might be \$4 billion, but would offer a large payback in the form of regional security. It would not be needed before about 2080.

The European and MENA initiatives will act as role models for a possible SE Asian initiative, where interest has been shown by representatives of potential nuclear power newcomer nations such as Malaysia and Vietnam and where involvement with the mature nuclear power programs of the region (e.g. in the Republic of Korea and Taiwan) would add an extra dimension to the issue. Following a January 2014 meeting in Indonesia, further meetings are envisaged.

An EU group, the Implementing of Geological Disposal of Radwaste Technology Platform (IGD-TP) was established in 2009 by several EU organisations to support research related to its vision and assist the various EU programs. It does not have government-level representation but is partly funded by the EU.

Fuel leasing

There have been a number of proposals for fuel leasing. At present the normal process is that a utility buys uranium from a mining company, then gets it converted, enriched and fabricated, before using it and discharging the used fuel. That used fuel becomes the responsibility of the country in which it has been used, as outlined in the first paragraphs of this article.

Fuel leasing is an alternative concept whereby the utility leases its fabricated fuel from a supplier, probably in another country, and after it has been used that supplier takes it back. This concept is not yet in use except for some very limited applications, mainly for Russian-built nuclear power plants in NPT non-weapons states (e.g. Bangladesh, Turkey, Iran). The supplier would then add the leased used fuel to its own larger stocks to be stored for later disposal or reprocessing and recycling, in which case the valuable components would belong to the fuel supplier/leaser.

The concept was part of the Global Nuclear Energy Partnership (now IFNEC) launched in 2006, in order to restrict the spread of sensitive technologies such as enrichment, but it did not come to fruition under that program. Of all the nuclear suppliers, Russia has expressed the most support for fuel leasing and take-back. Russia's fuel supply contract with Iran involves leasing, and Iran is required to send the used fuel back to Russia. Russian law allows import of foreign used fuel for reprocessing and the return of wastes (as with reprocessing other countries' used fuel in France and Britain), and Russia's default position in supplying reactors to non-nuclear weapons states is to take back Russian-origin fuel without requiring return of the wastes to those countries, as with Iran.

The Pangea proposal

In the 1990s Pangea Resources, a UK-based company, identified Australia, southern Africa, Argentina and western China as having the most appropriate geological credentials for a deep geologic repository, with Australia being favored on economic and political grounds. It would be located where the geology has been stable for several hundred million years, so that there need not be total reliance on a robust engineered barrier system to keep the waste securely isolated for thousands of years.

It would be a commercial undertaking and would have dedicated port and rail infrastructure. It would take spent fuel and other wastes from commercial reactors, and possibly also material from weapons disposal programs.

Pangea summed up the situation thus:

By taking a fresh look at the reasons for the difficulties which have faced most national repository programs, and discarding the preconception that each country must develop its own disposal facilities, it is possible to define a class of simple, superior high-isolation sites which may provide a multinational basis for solving the nuclear waste disposal problem.

The relatively small volumes of high-level wastes or spent fuel which arise from nuclear power production make shared repositories a feasible proposition. For small countries, the economies of scale which can be achieved make the concept attractive. For all countries, objective consideration of the relative merits of national and multi-national solutions is a prudent part of planning the management of long-lived radioactive wastes.

Early in 1999 Pangea Resources released its project proposal to the Australian public, expecting this “to initiate discussions which will enable us to more fully assess the feasibility and strategy of our proposal ... on (its) merits.”

Pangea’s objectives were to site a deep geologic disposal facility in a region where the geology and biosphere conditions meet the test of simplicity coupled with robustness in order to demonstrate that the performance of the facility from a safety standpoint will meet the highest international standards and international safeguard requirements. In addition to its ideal geological characteristics, the host country should preferably be a first-world, stable democracy, familiar with high-technology enterprises.

Pangea’s strategy implied that the geological barrier can be the primary safety barrier, in contrast to some other potential repository concepts where the waste form and the engineered barriers are required to be more dominant. There is a side benefit in that less complex and less expensive engineered barriers may be sufficient. Pangea also saw a potential public acceptance benefit, in that reliance on a simple geological barrier might be more readily understood and accepted.

The Pangea concept envisaged a dedicated port and rail link to an inland repository site covering perhaps 5 km² on the surface and 20 km² underground (500 m down). There would be a fleet of 35 dedicated and purpose-built ships at any one time.

Pangea’s business plan was based on taking 75,000 t of spent fuel and high-level waste from reprocessing spent fuel, plus some intermediate-level wastes from decommissioning nuclear facilities, over some 40 years. Spent fuel would be shipped to the facility at a rate of about 2000–3000 t per year once it was fully operational. This rate is about 20% of the spent fuel generated annually by commercial reactors around the world, or to put it another way, the repository is designed to take 25% of the world’s civil waste inventory at the time it opens. The projected size of the repository is thus similar to that proposed at Yucca Mountain, Nevada.

The capital cost was estimated at US\$ 6 billion, with some US\$ 400 million per year operating cost. An Access Economics study projected total export revenues over 40 years of about US\$ 100 billion, with payments to governments of about \$50 billion (1998 dollars) before considering multiplier effects. This would have added about 1% to the Australian GNP. The project envisaged establishment of a shipyard and foundry for the manufacture of 70 specialized ships and some 3000 large stainless steel transport casks as well as port and fleet maintenance facilities. Direct employment would be about 2000, indirect about 6000 people.

While the concept aims at nuclear waste generated by countries other than the United States, that country would need to be closely involved because through Non-Proliferation Treaty provisions, it controls some 60% of the nuclear fuel worldwide and would have to authorize any international movement of it.

The commercial scheme would also provide the repository which could be used to dispose of unwanted nuclear materials from disarmament, which was seen as a major spin-off. These would include excess plutonium from nuclear weapons either in spent mixed oxide fuel or in the Synroc-based ceramic developed by the Australian Nuclear Science & Technology Organisation (ANSTO) and chosen in 1998 by the US Department of Energy as the wasteform to immobilise some of its excess weapons plutonium (see link at end).

The initial response from the federal government however was to reiterate Australia’s long-standing and bipartisan policy of not importing nuclear wastes and saying that there was no immediate intention of considering such a proposal. Then, after only cursory consideration, the Western Australian parliament passed a Bill to make it illegal to dispose of foreign high-level waste in the state without specific parliamentary approval. Pangea continued its geological investigations in that state while extending its feasibility study to other potential host regions.

The Pangea concept can be traced to the Synroc Study Group, which began its activities in 1988, set up by the Australian government to study the commercial potential for Synroc in a global context. It was conducted by four leading Australian resource companies, assisted by ANSTO and the Research School of Earth Sciences at the Australian National University.

Russian plans 2001–06

In 2001 the Russian parliament (Duma) passed legislation to allow the import of spent nuclear fuel. The President signed this into law and set up a special commission to approve and oversee such imports. The commission would have 20 members, five each from the Duma, the Council (upper house), the government and presidential nominees. It would be chaired by Dr. Zhores Alferoy, a parliamentarian who is also Vice President of the Russian Academy of Sciences and a Nobel Prize physicist.

This scheme was progressed in 2005 when the Duma ratified the Vienna Convention on civil liability for nuclear damage. However in July 2006 Rosatom announced it would not proceed with taking any foreign-origin used fuel.

In 2003 Krasnokamensk had been suggested as the site for a major spent fuel repository—it is a city 7000 km east of Moscow in the Chita region and is a center for uranium mining and milling. [McCombie and Chapman \(2005\)](#) discuss Krasnoyarsk’s potential. Then in 2008 the Nizhnekansky Granite Massif at Zheleznogorsk in Krasnoyarsk Territory was put forward for a national deep geological repository. In August 2016 the Territorial Planning Scheme to 2030 confirmed the site and approved construction of repository facilities here for 4500 m³ net of class 1 wastes and 155,000 m³ net of class 2 wastes.

(Russia continues to take back for reprocessing all fuel that it supplies in connection with reactors that it builds in foreign countries other than India.)

Summary

International waste repositories are a logical development for countries with a relatively small amount of high-level wastes. However, working through the political implications for potential host countries is not straightforward.

See Also: Assessment of the Relative Environmental Footprint of Nuclear Energy and Its Fuel Cycle; Main Repository Projects Worldwide; Social Acceptability of Geologic Disposal; The Concept of Geological Disposal of Highly Radioactive Nuclear Waste.

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Notes

Cross-reference terms in *italics> are general cross-references, or refer to subentry terms within the main entry (the main entry is not repeated to save space). Readers are also advised to refer to each article for additional cross-references – not all of these cross-references have been included in the index cross-references.*

The index is arranged in set-out style with a maximum of three levels of subheading. Major discussion of a subject is indicated by bold page numbers. Page numbers suffixed by t, f, and b refer to Tables, Figures, and Boxes respectively. *vs.* indicates a comparison.

The index entries are presented in word-by-word alphabetical sequence in which a group of letters followed by a space is filed before the same group of letters followed by a letter. For example, entries beginning 'air density' are alphabetized before 'aircraft.' Prefixes and terms in parentheses are excluded from the initial alphabetization.

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